Docket Nos. 50-269/270/287

JUN 3 0 1976

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 Amendments Nos. 27, [illegible]7, and [illegible] for 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 Technical Specifications and are in response to your requests dated February 25, 1975, as revised May 7, 1976, and dated June 11, 1976.

These amendments (1) revise the Technical Specifications to establish operating limits for Unit 2 Cycle 2 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR Section 50.46 and (2) terminate the operating restrictions imposed on Unit 2 by the Commission's December 27, 1974 Order for Modification of License.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed By

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

Enclosures:
See next page

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Enclosures:
1. Amendment No. to DPR-38
2. Amendment No. to DPR-47
3. Amendment No. to DPR-55
4. Safety Evaluation
5. Federal Register Notice

cc w/enclosures:
See next page

bcc: JRBuchanan
TBAbernathy

<u>DISTRIBUTION</u>
Docket Files (3)
NRC PDRs(3)
Local PDR
ORB#1 Reading
KRGoller/TJCarter
ASchwencer
GZech
SMSheppard
JMcGough
ELD
OI&E(3)
ACRS(16)
BJones(12)
BScharf(15)
CMiles,OPA
VStello

OFFICE ➤	ORB#1	ORB#1	OELD	OR/AD		
SURNAME ➤	GZech:dc	ASchwencer		KRGoller		
DATE ➤	6/22/76	6/ /76	6/ /76	6/ /76		

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

Mr. Troy B. Conner
Conner & Knotts
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

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

Honorable Reese A. Hubbard
County Supervisor of Oconee County
Walhalla, South Carolina 29621

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

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Duke Power Company (the licensee) dated February 25, 1975, as revised May 7, 1976, and dated June 11, 1976 comply 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1976

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Duke Power Company (the licensee) dated February 25, 1975, as revised May 7, 1976, and dated June 11, 1976 comply 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1976

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Duke Power Company (the licensee) dated February 25, 1975, as revised May 7, 1976, and dated June 11, 1976 comply 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1976

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 27 TO DPR-38

AMENDMENT NO. 27 TO DPR-47

AMENDMENT NO. 23 TO DPR-55

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

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-3a	2.1-3a
2.1-3b	2.1-3b
2.1-3c	2.1-3c
- -	2.1-3d
- -	2.1-3e
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
2.3-6	2.3-6
2.3-9	2.3-9
2.3-12	2.3-12
3.5-7	3.5-7
3.5-8	3.5-8
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-14	3.5-14
3.5-14a	3.5-14a
3.5-15	3.5-15
3.5-19	3.5-19
- -	3.5-19a
- -	3.5-19b
3.5-22	3.5-22
- -	3.5-22a
- -	3.5-22b
3.5-24	3.5-24
3.11-1	3.11-1
4.1-9	4.1-9
4.2-3	4.2-3
4.6-1	4.6-1
4.6-2	4.6-2
5.3-1	5.3-1

The safety limits presented for Oconee Unit 2 have been generated using BAW-2 critical heat flux correlation [1] and the Reactor Coolant System flow rate of 107.6 percent of the design flow (131.21×10^6 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 2.1-1B 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 141.3×10^6 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 2.1-2B 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 19.8 kw/ft for Unit 2.

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 2.1-2B 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 maximum thermal power for three-pump operation is 86.4 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.9 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 2.1-3B, 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 1.30 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

References

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

(2) Oconee 2, Cycle 2 - Reload Report - BAW-1425 (Rev. 1), April 1976.

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 W-3 correlation.(1) The W-3 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 1.3. A DNBR of 1.3 corresponds to a 94.3 percent probability at a 99 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 1.3 is predicted for the maximum possible thermal power (112%) when four reactor coolant pumps are operating (minimum reactor coolant flow is 131.3×10^6 lbs/hr). This curve is based on the following nuclear power peaking factors(2) with potential fuel densification 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 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:

1. The 1.3 DNBR limit produced by a nuclear power 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 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.8 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, 3, and 4 of Figure 2.1-2C correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

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.

The curves of Figure 2.1-3C represent the conditions at which a minimum DNBR of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%,(3) whichever condition is more restrictive.

Using a local quality limit of 15 percent at the point of minimum DNBR as a basis for Curves 2 and 4 of Figure 2.1-3C is a conservative criterion even though the quality of the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from point of minimum DNBR, so that the exit DNBR is 1.7 or higher, depending on the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15 percent is justified on the basis of experimental data.(4)

The maximum thermal power for three pump operation is 86.4% - Unit 3 due to a power level trip produced by the flux-flow ratio 75% flow x 1.07 = 80% power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner. A flux-flow ratio of 0.961 is used for single loop conditions.
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 1.3 or a local quality at the point of minimum DNBR less than 15 percent for that particular reactor coolant pump situation. The 1.3 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

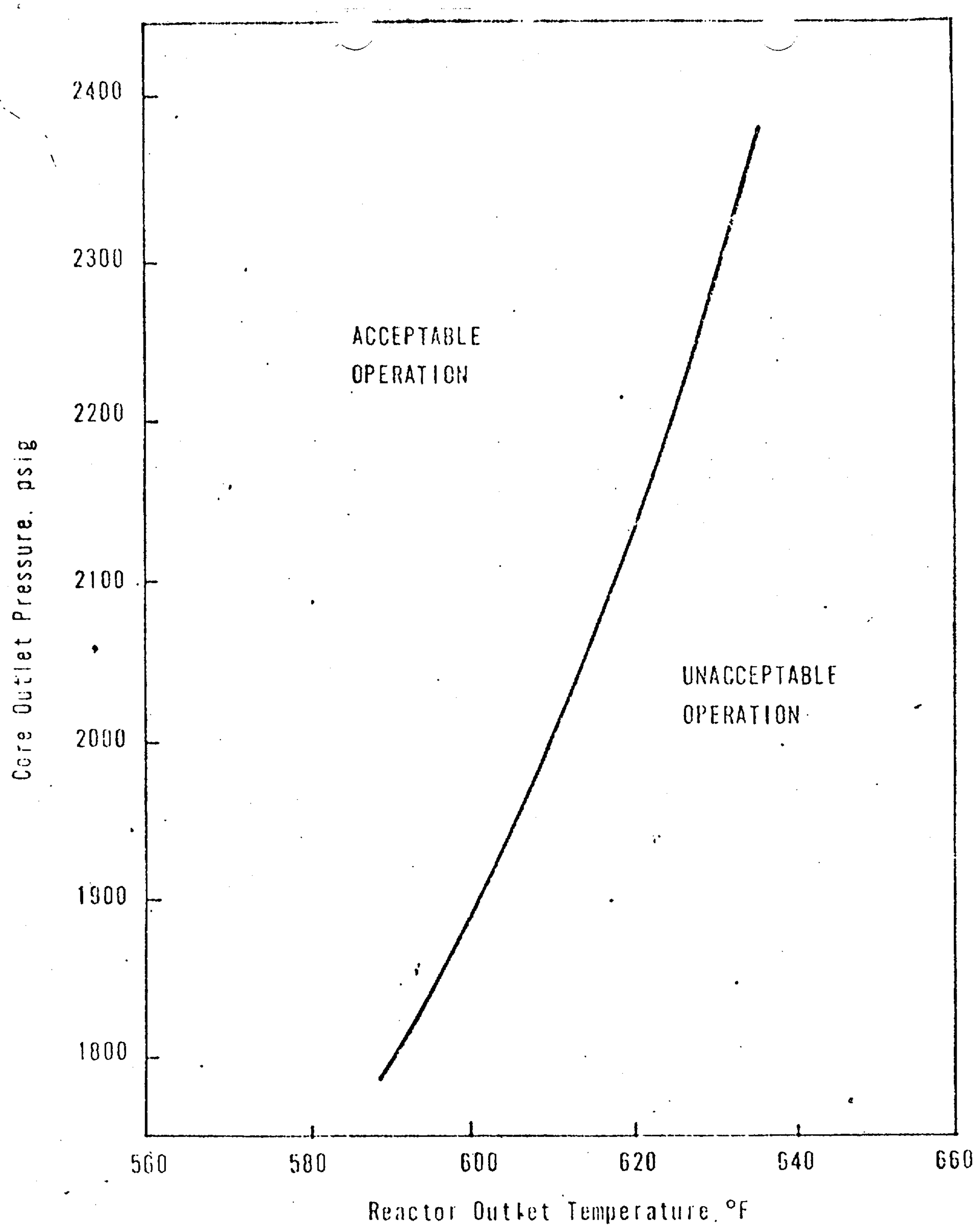
REFERENCES

(1) FSAR, Section 3.2.3.1.1
(2) FSAR, Section 3.2.3.1.1.c
(3) FSAR, Section 3.2.3.1.1.k

(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the "Two-phase Flow and Heat Transfer in Rod Bundles Symposium:"

(a) Wilson, et al.
"Critical Heat Flux in Non-Uniform Heater Rod Bundles"

(b) Gellerstedt, et al.
"Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water"

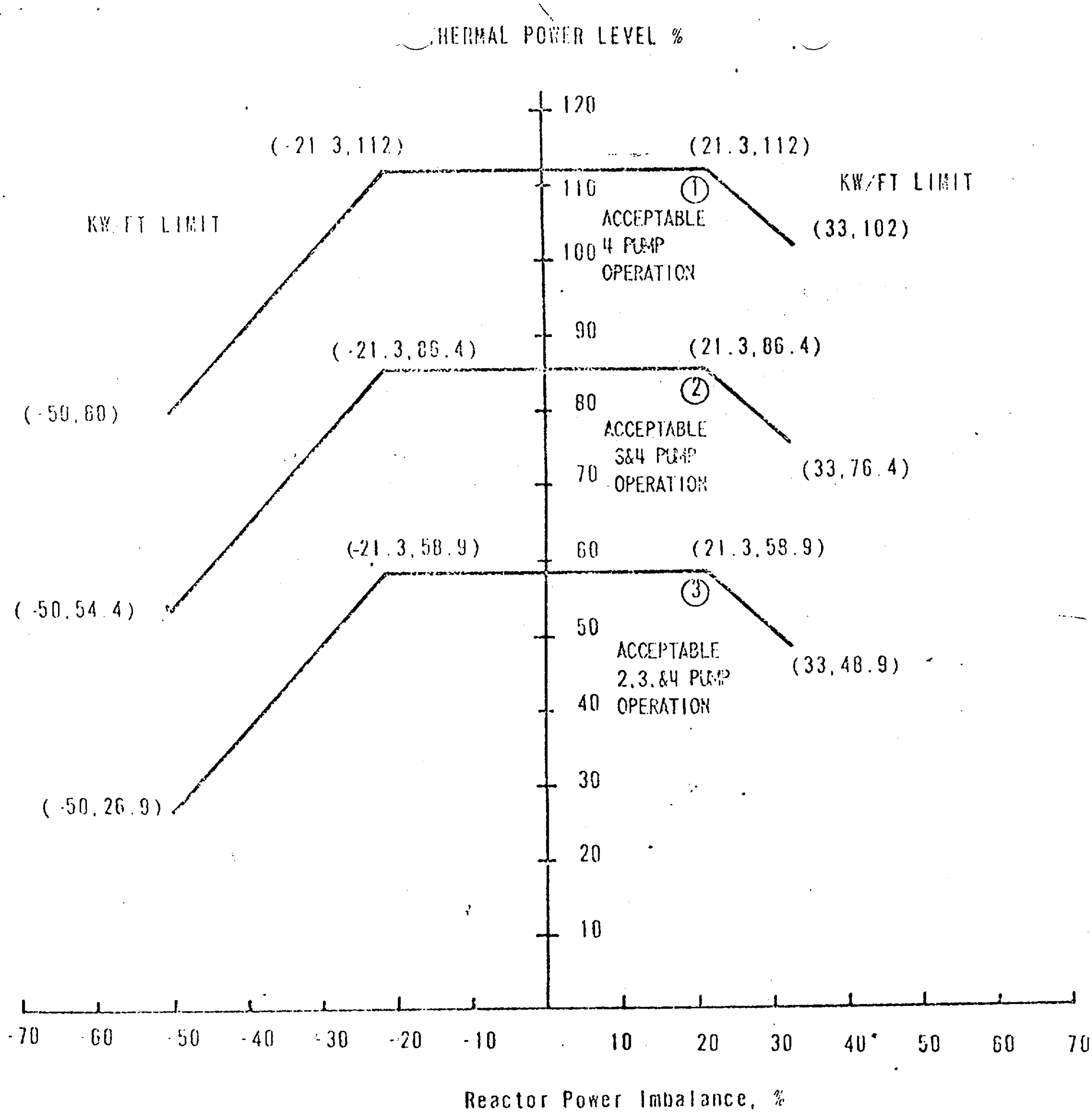


CORE PROTECTION SAFETY LIMITS

UNIT 2

OCONEE NUCLEAR STATION

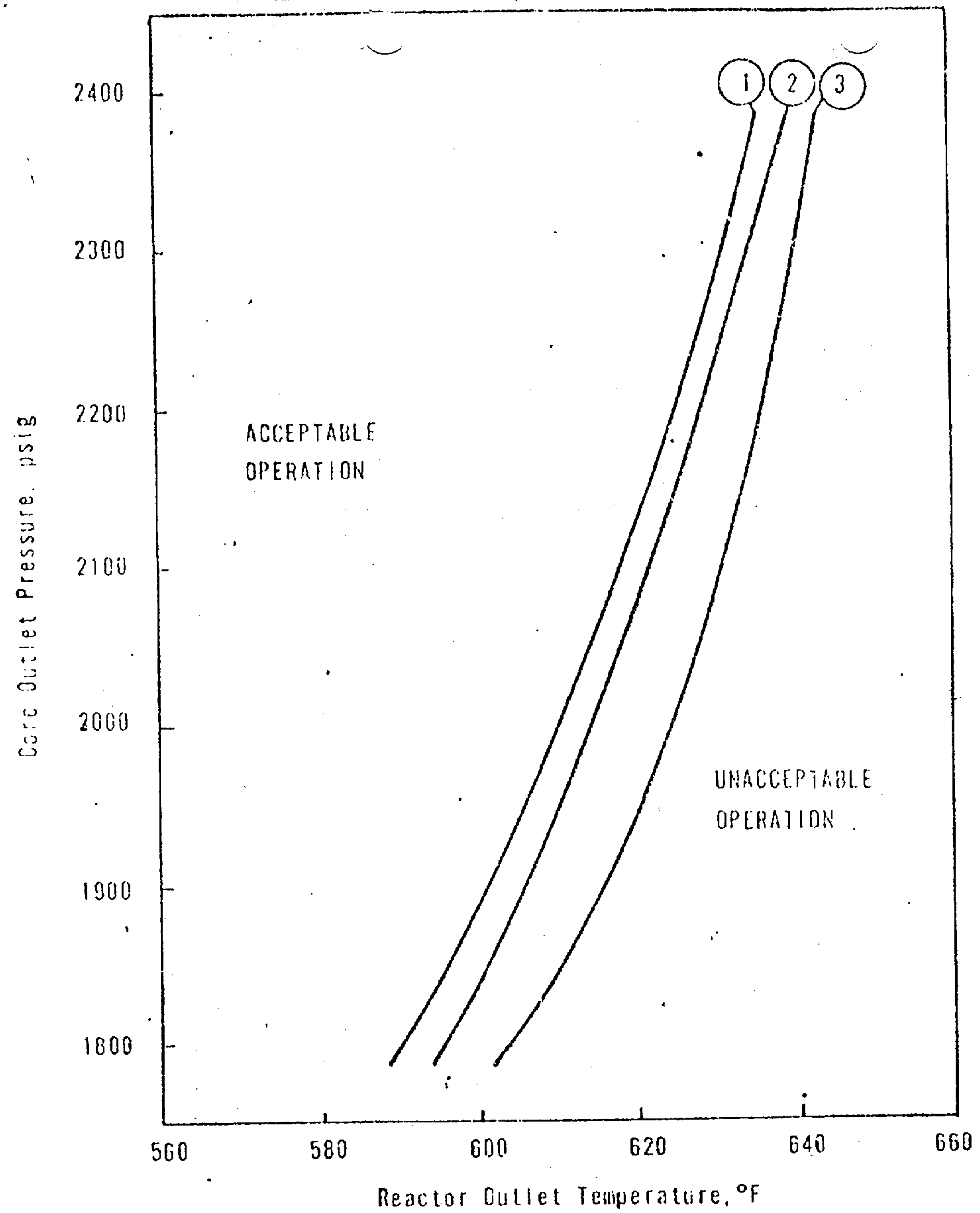
FIGURE 2.1-1B



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6

CORE PROTECTION SAFETY LIMITS
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 2.1-2B

Amendments Nos. 27, 27, & 23



CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	141.3×10^6 (100%)*	112%	FOUR PUMP (DNBR LIMITED)
2	105.6×10^6 (74.7%)	86.4%	THREE PUMP (DNBR LIMITED)
3	69.3×10^6 (49%)	58.9%	ONE PUMP IN EACH LOOP (QUALITY LIMITED)

* 107.6% OF CYCLE 1 DESIGN FLOW

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 set point 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 set point 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 situtations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 105.5% and reactor flow rate is 100%, or flow rate is 94.8% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.8% and reactor flow rate is 74.7% or flow rate is 71.1% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 51.7% and the operating loop flow rate is 54.5% or flow rate is 48.5% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 51.7% and reactor flow rate is 49.0% or flow rate is 46.4% and the power level is 49%.

The flux-to-flow ratios for Units 1 and 2 account for the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

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 1.055%-Unit 1 for a 1% flow reduction.

1.07% - Unit 2

1.07% - Unit 3

For Unit 1, the power-to-flow reduction ratio is 0.949, and for Units 2 and 3, the power-to-flow reduction factor is 0.961 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 overpower 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 ($10.79\ T_{out}-4539$)

(1800) psig ($16.25\ T_{out}-7756$)

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 ($11.14\ T_{out}-4746$)

($10.79\ T_{out}-4579$)

($16.25\ T_{out}-7796$)

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

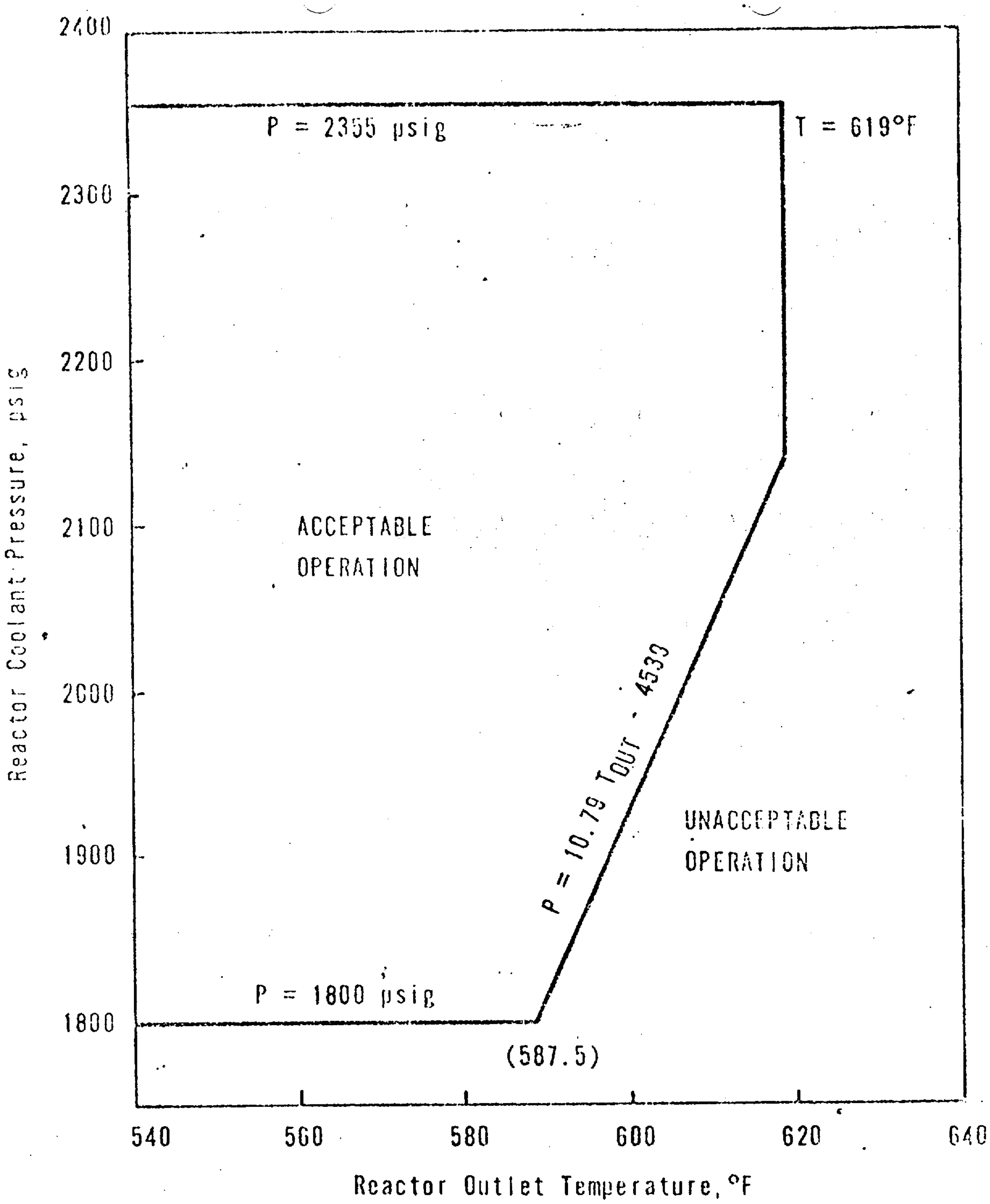
2.3-1B

2.3-1C

temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620^{o}F.

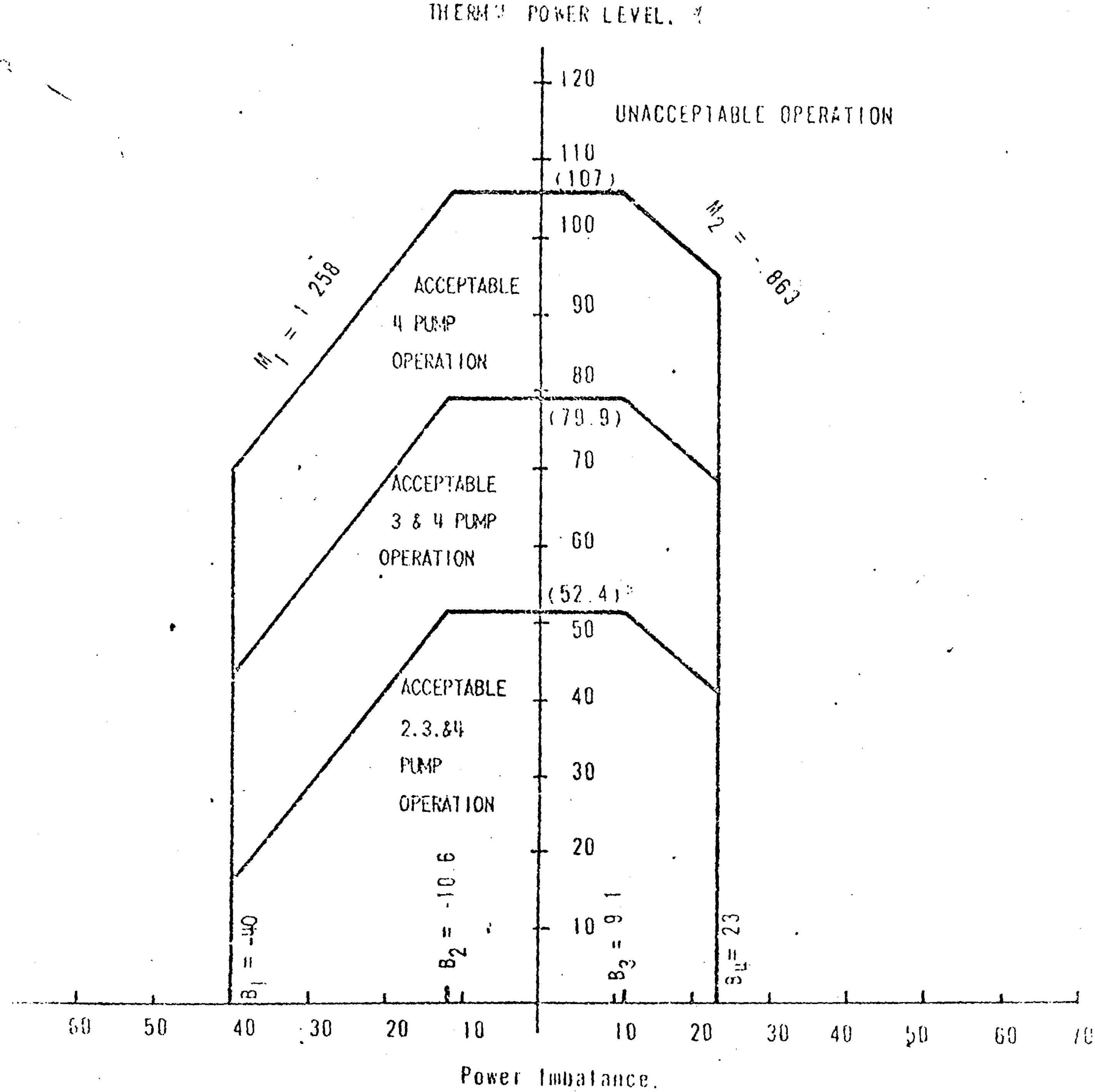
Reactor Building Pressure

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 2.3-1B



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 2.3-2B

Amendments Nos. 27, 27, & 23

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)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% 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	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	0.961 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% (5) (6)	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720[4]
5. Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6. Variable Low Reactor Coolant System Pressure psig, Min.	$(10.79\ T_{out}-4539)^{(1)}$	$(10.79\ T_{out}-4539)^{(1)}$	$(10.79\ T_{out}-4539)^{(1)}$	$(10.79\ T_{out}-4539)^{(1)}$	Bypassed
7. Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8. High Reactor Building Pressure, psig, Max.	4	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.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

g. If within one (1) hour of determination of an inoperable rod, it is not determined that a 1%Δk/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

h. Following the determination of an inoperable rod, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

i. If a control rod in the regulating or safety rod groups is declared inoperable, power shall be reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.

j. If a control rod in the regulating or axial power shaping groups is declared inoperable, operation above 60 percent of rated power may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 3.5.2.2.a and the withdrawal limits of Specification 3.5.2.5.c.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, if the maximum positive quadrant power tilt exceeds +3.41% Unit 1, either the quadrant power tilt shall
3.41% Unit 2
4.92% Unit 3
be reduced to less than +3.41% Unit 1 within two hours or the
3.41% Unit 2
4.92% Unit 3
following actions shall be taken:

(1) If four reactor coolant pumps are in operation, the allowable thermal power shall be reduced below the power level cutoff (as identified in specification 3.5.2.5) and further reduced by 2% of full power for each 1% tilt in excess of 3.41% Unit 1.
3.41% Unit 2
4.92% Unit 3

(2) If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% of full power for each 1% tilt.

(3) Except as provided in specification 3.5.2.4.a, the reactor shall be brought to the hot shutdown condition within four hours if the quadrant power tilt is not reduced to less than 3.41% Unit 1 within 24 hours.
3.41% Unit 2
4.92% Unit 3

b. If the quadrant tilt exceeds +3.41% Unit 1 and there is simultaneous
3.41% Unit 2
4.92% Unit 3
indication of a misaligned control rod per Specification 3.5.2.2, reactor operation may continue provided power is reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

c. Except for physics test, if quadrant tilt exceeds 9.44% Unit 1,
9.44% Unit 2
11.07% Unit 3
a controlled shutdown shall be initiated immediately, and the reactor shall be brought to the hot shutdown condition within four hours.

d. Whenever the reactor is brought to hot shutdown pursuant to 3.5.2.4.a(3) or 3.5.2.4.c above, subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the power range high flux setpoint allowable for the reactor coolant pump combination are restricted by a reduction of 2 percent of full power for each 1 percent tilt for the maximum tilt observed prior to shutdown.

e. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated 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. Operating rod group overlap shall be 25% $\pm$ 5% between two sequential groups, except for physics tests.

c. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1A1 and 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, and 3.5.2-1C3 (Unit 3) for four pump operation and on Figures 3.5.2-2A1, 3.5.2-2A2 (Unit 1), 3.5.2-2B1, 3.5.2-2B2, 3.5.2-2B3 (Unit 2), and 3.5.2-2C (Unit 3) for three or two pump operation. If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. 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 shall be asymptotically approaching the 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, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, and 3.5.2-3C. If the imbalance is not within the envelope defined by Figure 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, and 3.5.2-3C, 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.

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, and 3.5.2-3C is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2-4) 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 effects
d. Hot rod manufacturing tolerance 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 0.5% Δk/k (Unit 1) or 0.65% Δk/k (Units 2 and 3) at rated power. These values have been shown to be safe by the safety analysis (2,3,4) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod positions limits at hot zero power. A single inserted control rod worth of 1.0% Δk/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 0.5% Δk/k (Unit 1) or 0.65% Δk/k (Units 2 and 3) ejected rod worth at rated power.

**Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and 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 with consideration of potential effects of rod bowing (Units 1 and 2 only) and fuel densification 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
7.36% 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 withour 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 the "undershoot" region and asymptotically 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

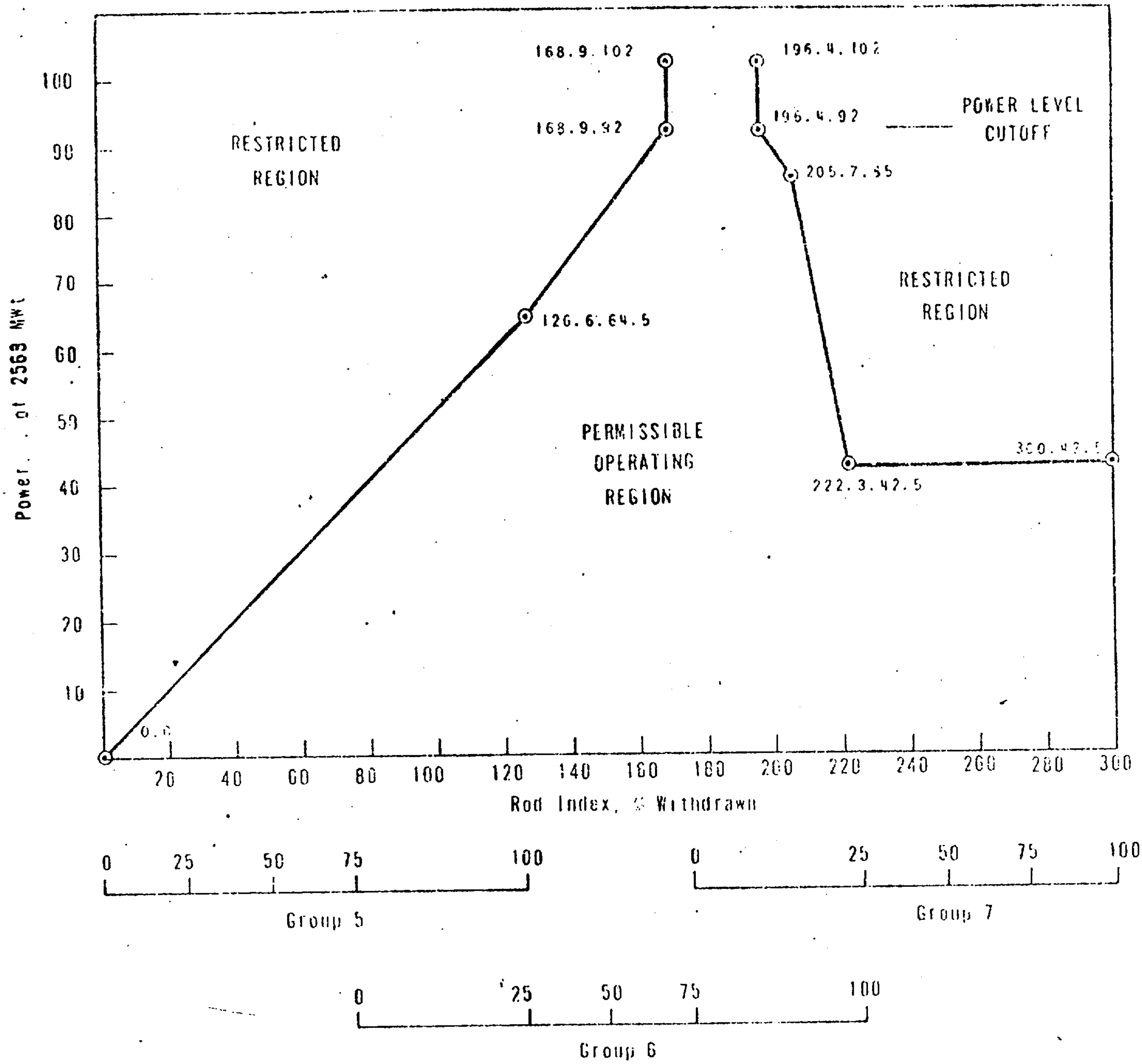
[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

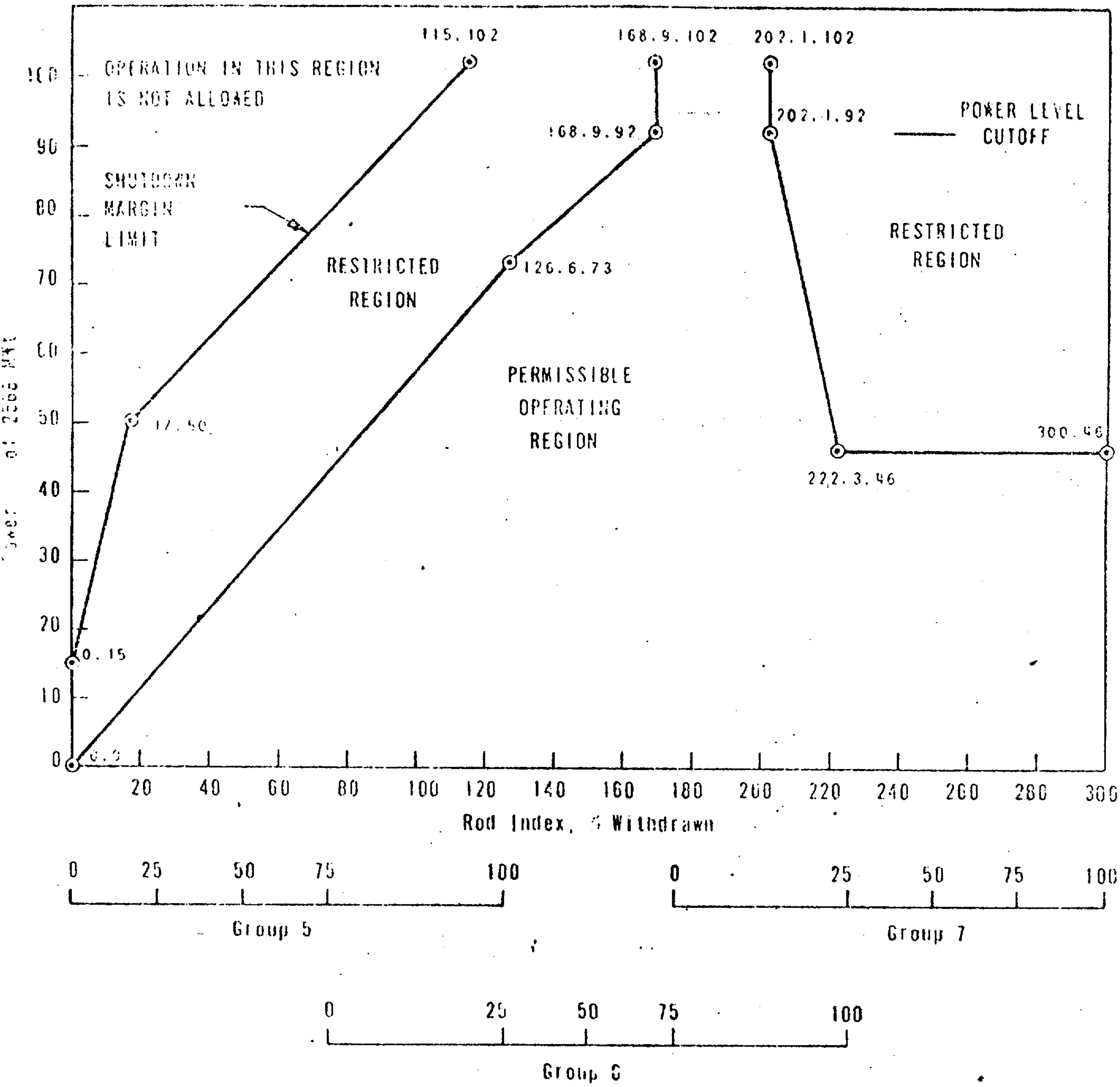
BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

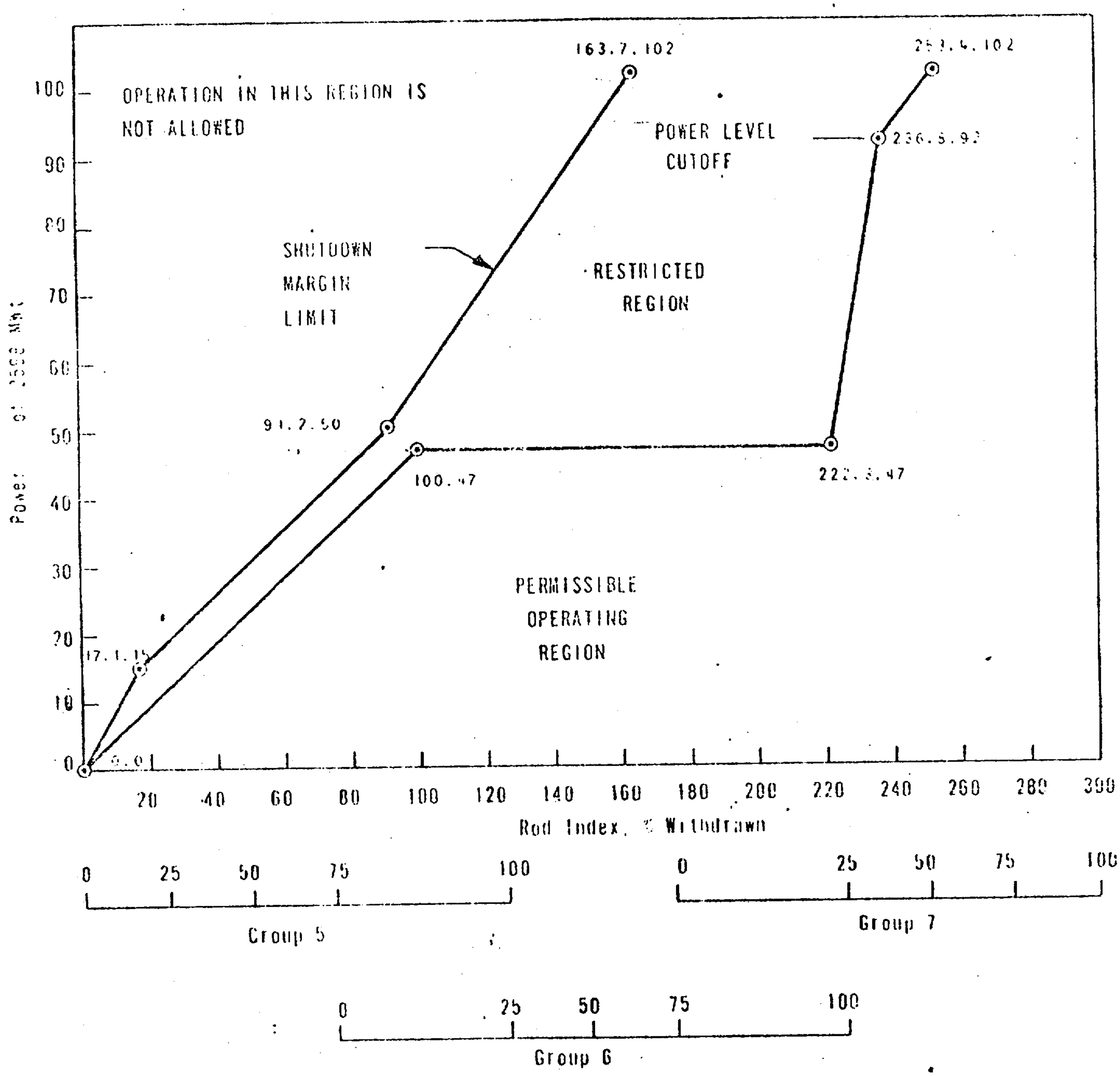
BAW-1400 (UNIT 3)



ROD POSITION LIMITS
FOR 4 PUMP OPERATION
FROM 0 TO 150±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1B1

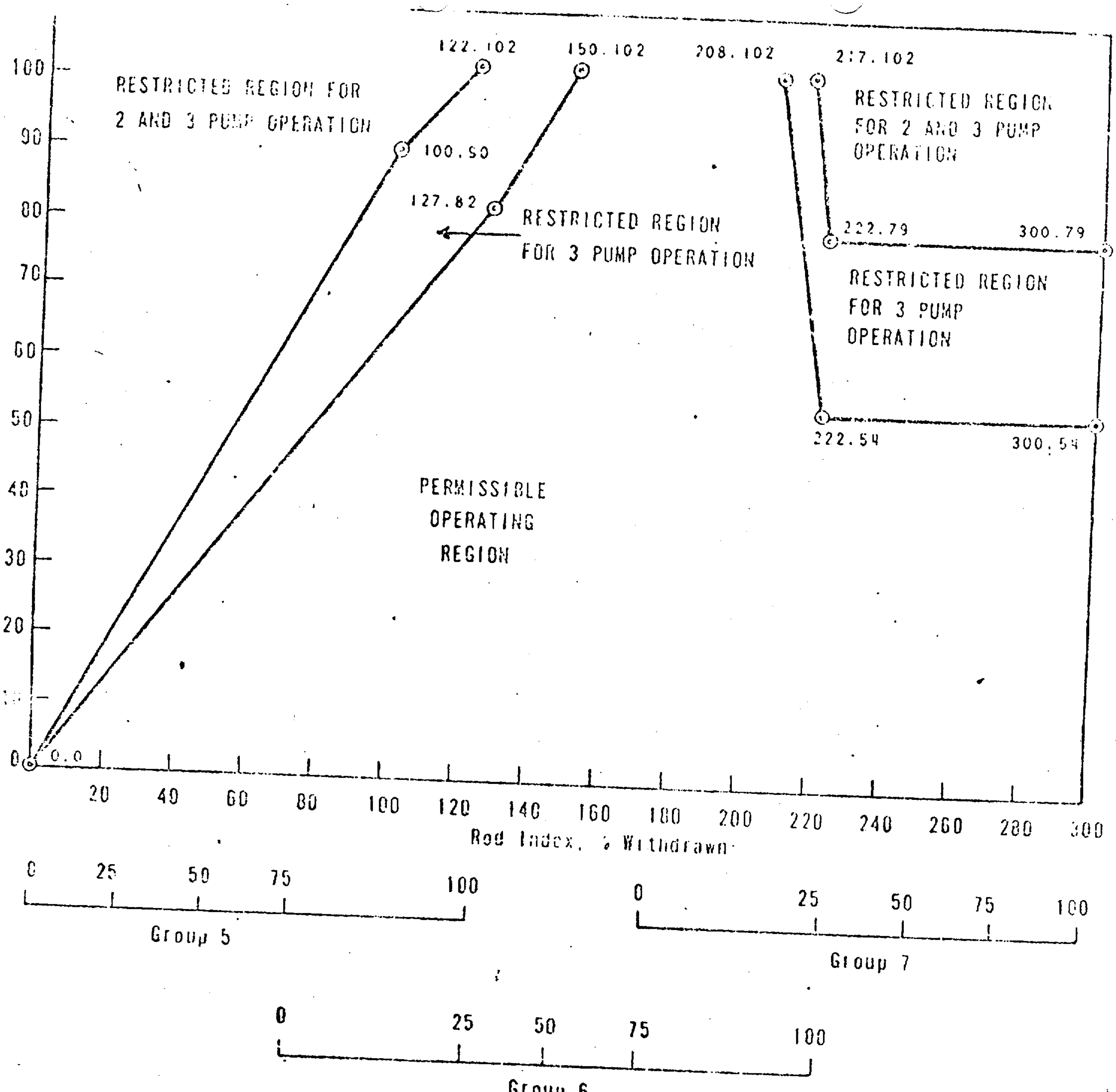


ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
150±10 EFPD TO 267±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1B2



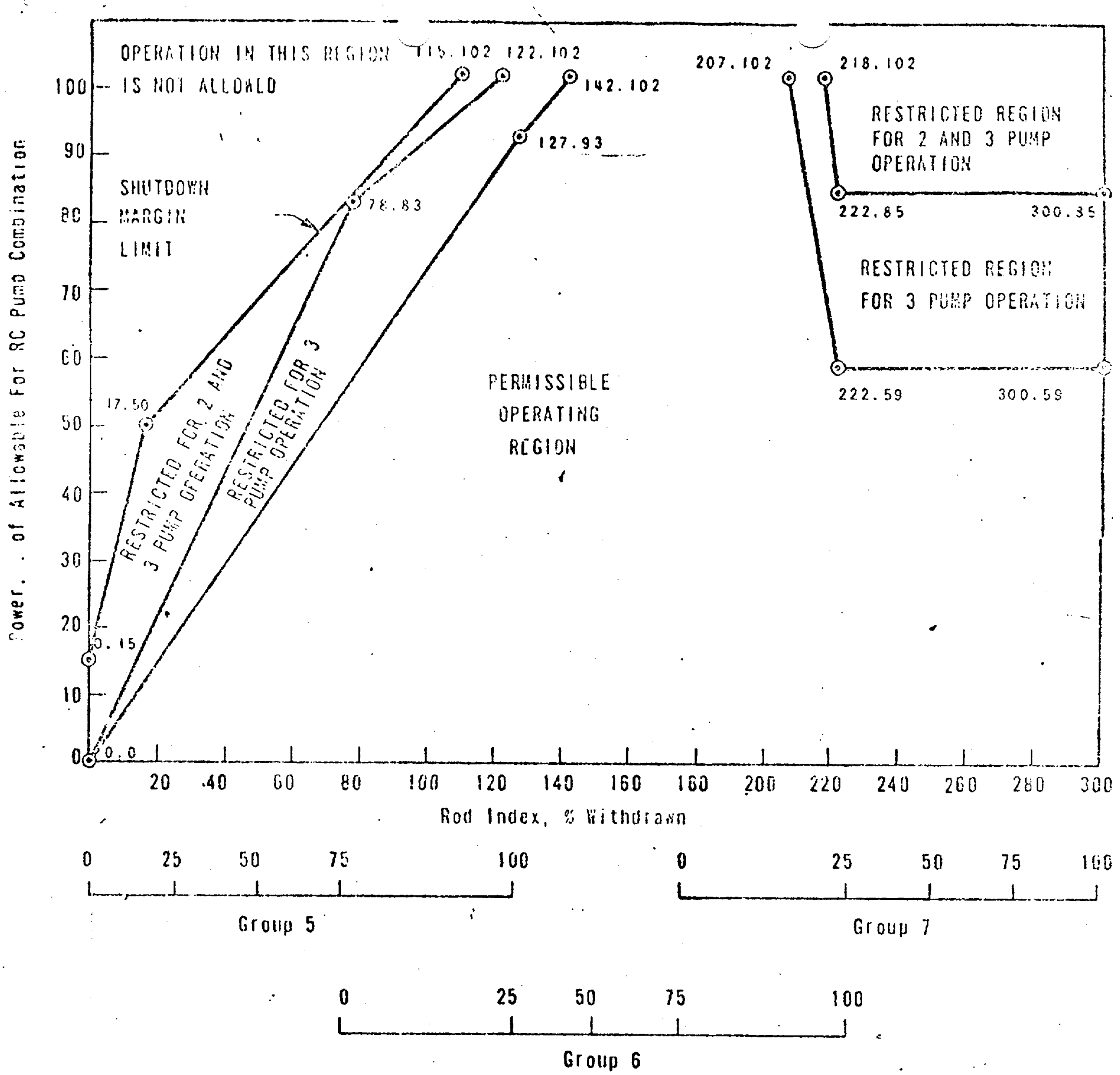
ROD POSITION LIMITS
FOR 4 PUMP OPERATION
AFTER 267±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION

FIGURE 3.5.2-1B3



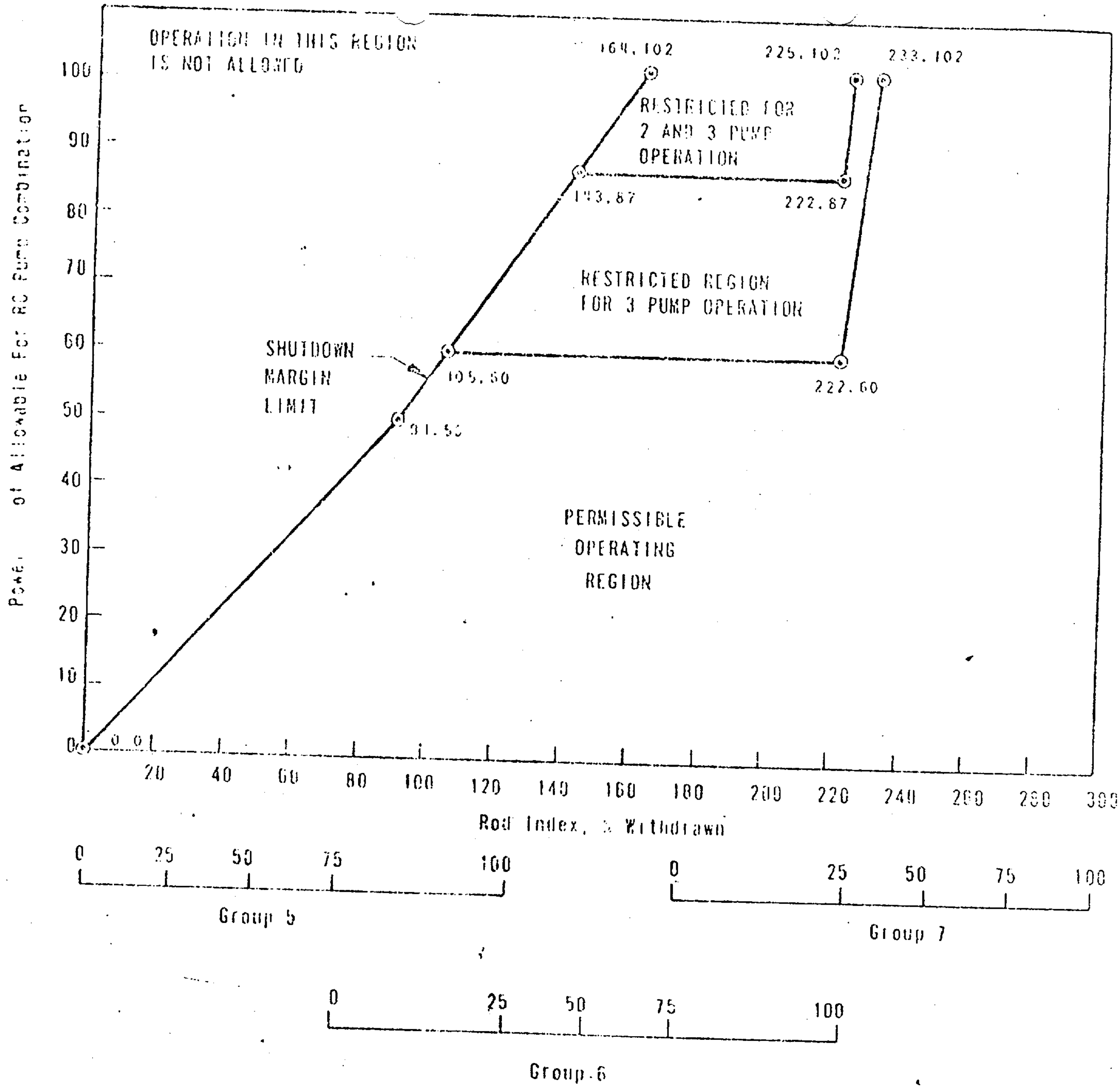
ROD POSITION LIMITS FOR
2 AND 3 PUMP OPERATION
FROM 0 TO 150±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION

FIGURE 3.5.2-2B1

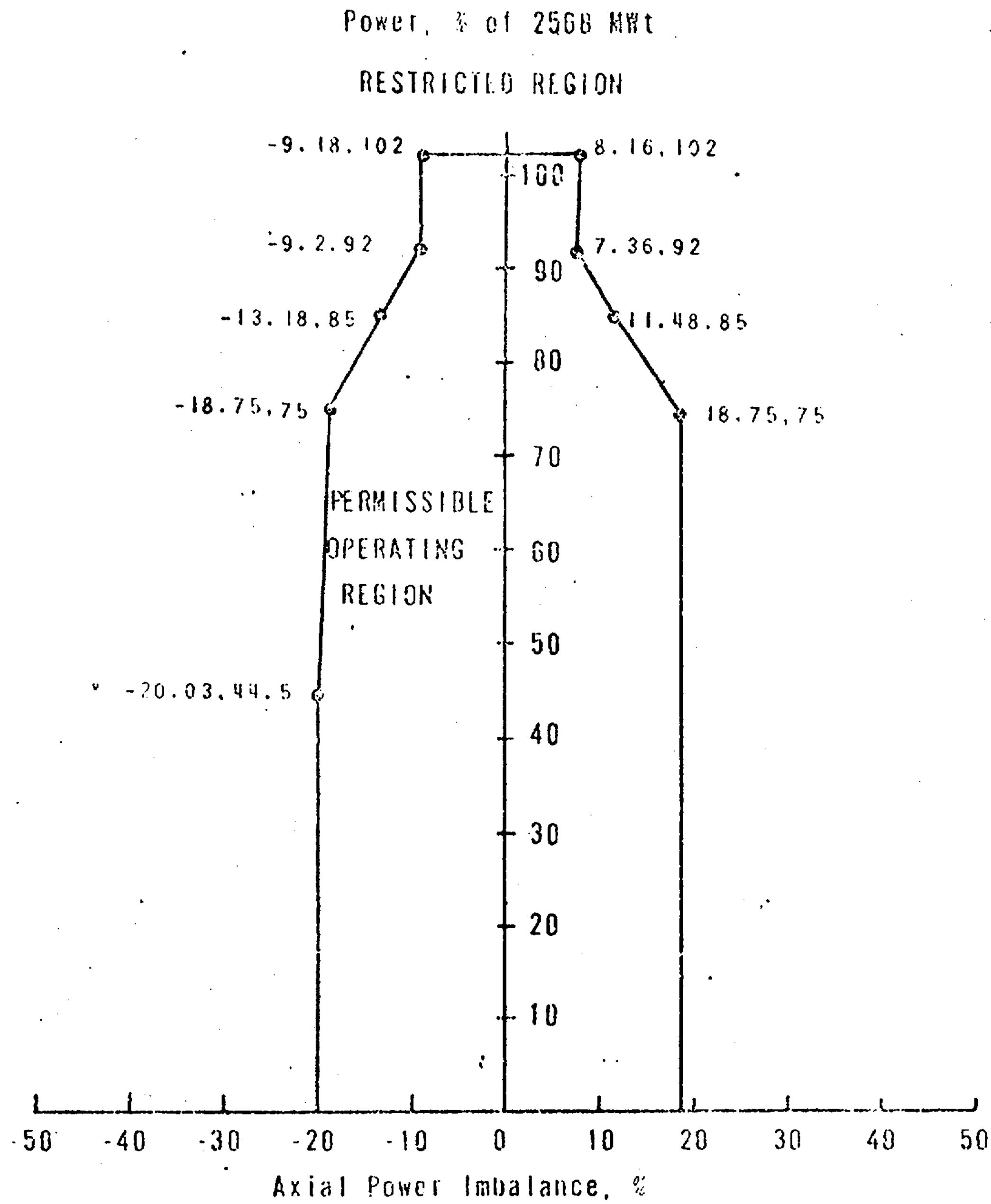


ROD POSITION LIMITS FOR
2 AND 3 PUMP OPERATION FROM
150±10 EFPD TO 267±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B2

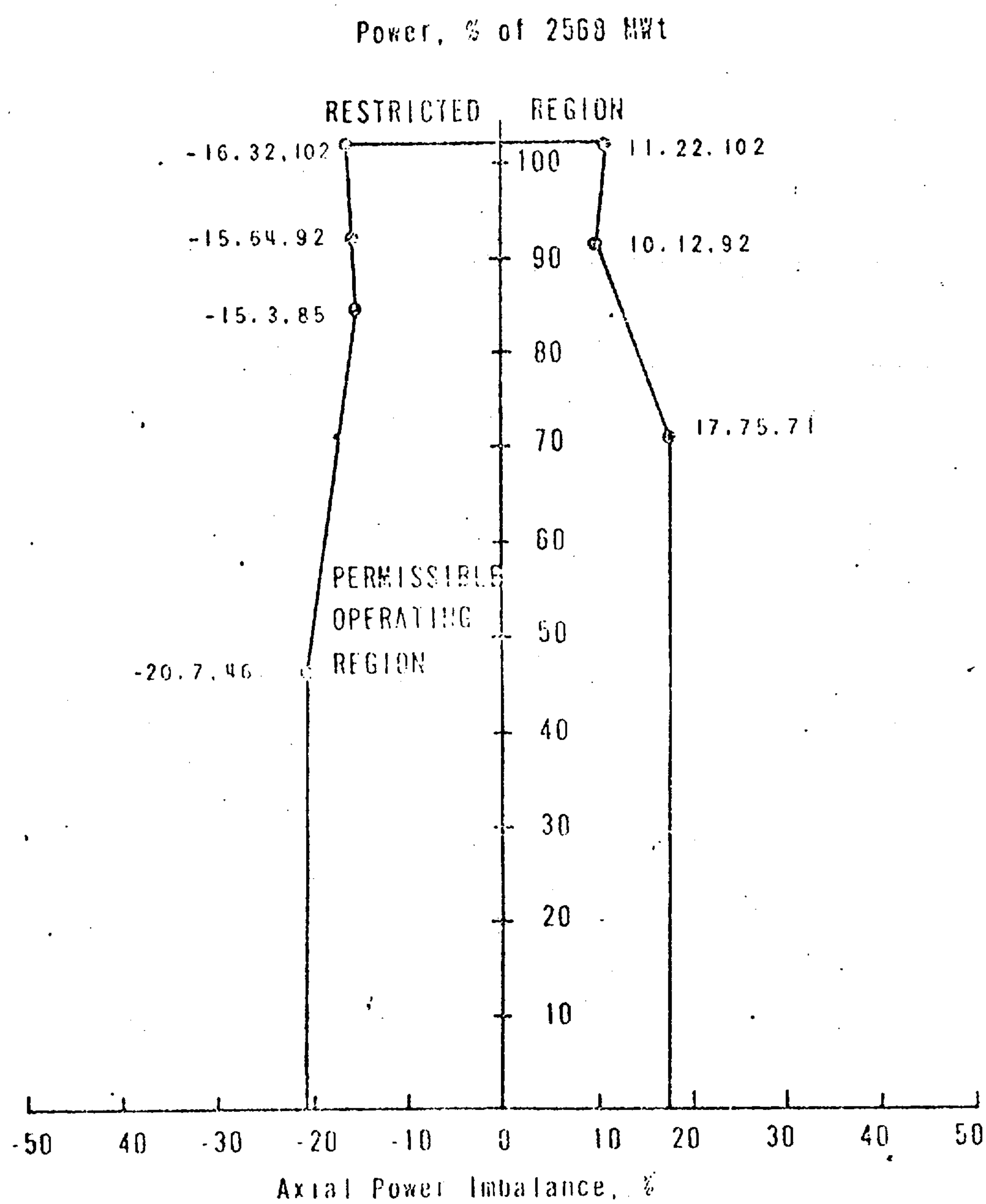
 Amendments Nos. 27, 27, & 23



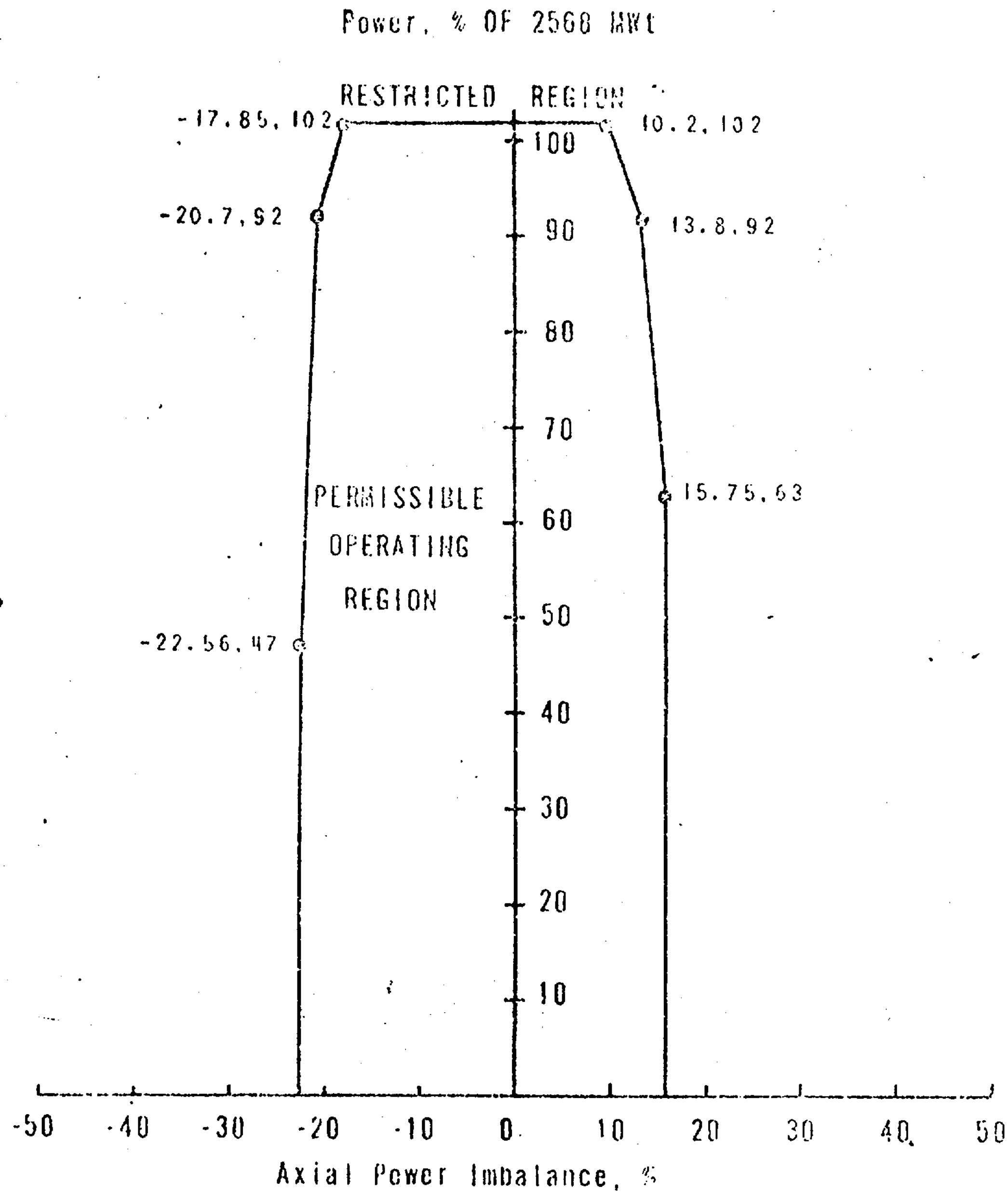
ROD POSITION LIMITS
FOR 2 AND 3 PUMP OPERATION
AFTER 267±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2B3



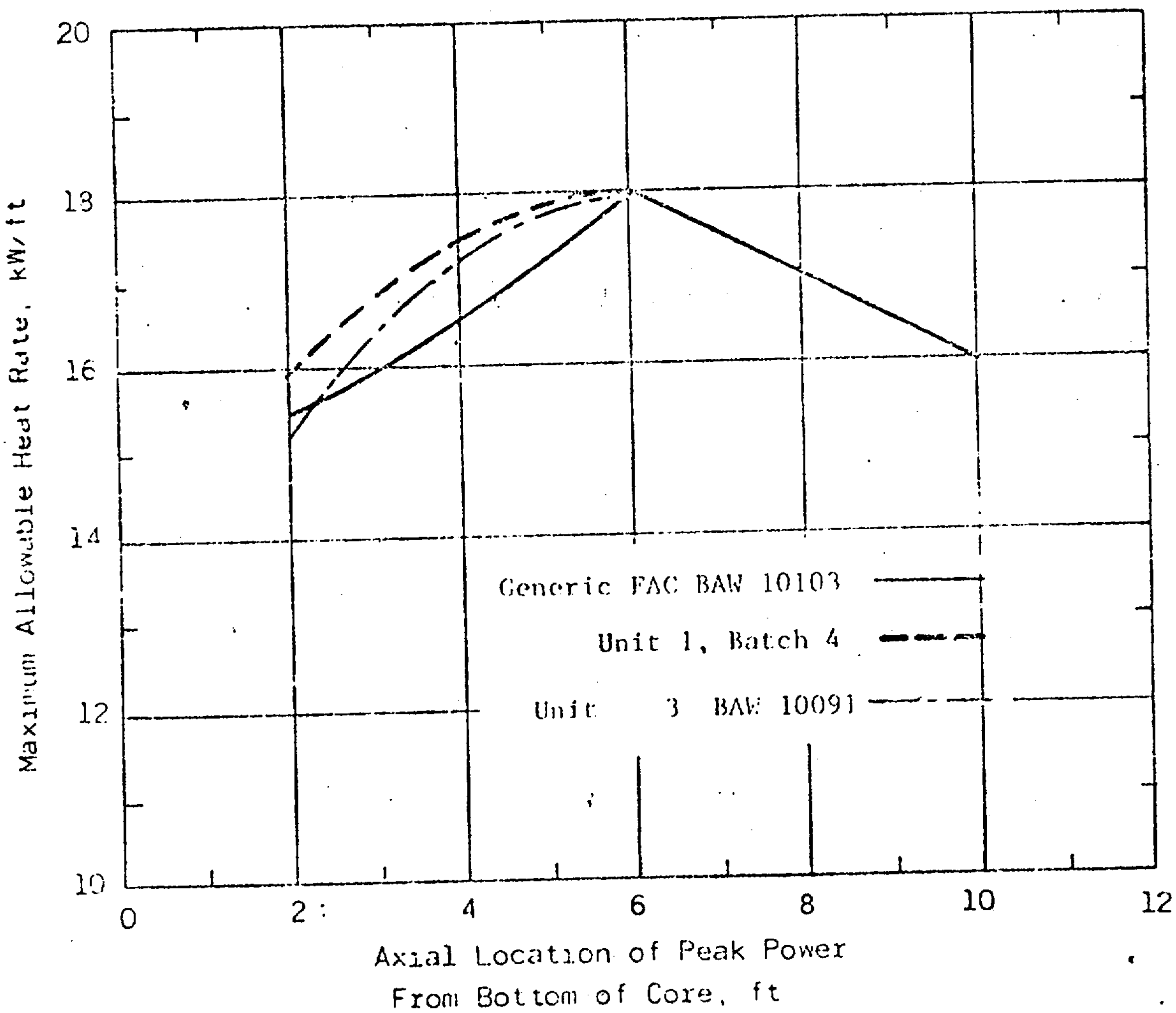
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 150±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3B1



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
150±10 EFPD TO 207±10 EFPD
Unit 2
OCONEE NUCLEAR STATION
Figure 3.5.-[illegible]2



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION
AFTER 267+10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3B3



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
OCONEE NUCLEAR STATION
FIGURE 3.5.2-4

3.11 MAXIMUM POWER RESTRICTION

Applicability

Applies to the nuclear steam supply system of Unit 3 reactor.

Objective

To maintain core life margin in reserve until the system has performed under operating conditions and design objectives for a significant period of time.

Specification

The first reactor core in Unit 3 may not be operated beyond 10,944 effective full power hours until supporting analysis and data pertinent to fuel clad collapse under fuel densification conditions have been approved by the Directorate of Licensing.

Bases

The licensing staff has reviewed the effects of fuel densification for the first core in Oconee Unit 3 and concluded that clad collapse will not take place within the first fuel cycle (10,944 effective full power hours). Detailed clad creep collapse analyses are yet to be performed to demonstrate that clad collapse will not occur during operation beyond the first fuel cycle.

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.	Hydraulic Snubbers on Safety-Related Systems	Visual Inspection	Annually
11.	High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing
12.	Reactor Coolant System Flow	Validate Flow to be at least: Unit 1 141.30×10^6 lb/hr Unit 2 141.30×10^6 lb/hr Unit 3 131.32×10^6 lb/hr	Once Per Fuel Cycle

(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.

4.2.10 For Unit 1, Cycle 3 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 4 operation. For Unit 2, Cycle 2 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.9 will be revised prior to Cycle 3 operation. For Unit 3, Cycle 1 operation, the surveillance capsules will be removed from the reactor vessel for a portion of the cycle and the provisions of Specification 4.2.9 will be revised prior to Cycle 2 operation.

4.2.11 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

4.2.12 To assure that reactor internals vent valves are not opening during operation, all vent valves will be inspected during each refueling outage to confirm that no vent valve is stuck open and that each valve operates freely.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel specimen surveillance program for Unit 1 and Unit 2 is based on equivalent exposure times of 1.8, 19.8, 30.6 and 39.6 years. The contents of the different type of capsules are defined below.

A Type	B Type
Weld Material	HAZ Material
HAZ Material	Baseline Material
Baseline Material	

For Unit 3, the Reactor Vessel Surveillance Program is based on equivalent exposure times of 1.8, 13.3, 26.7, and 30.0 years. The specimens have been selected and fabricated as specified in ASTM-E-185-72.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.

4.6 EMERGENCY POWER PERIODIC TESTING

Applicability

Applies to the periodic testing and surveillance of the emergency power sources.

Objective

To verify that the emergency power sources and equipment will respond promptly and properly when required.

Specification

4.6.1 Monthly, a test of the Keowee Hydro units shall be performed to verify proper operation of these emergency power sources and associated equipment. This test shall assure that:

a. Each hydro unit can be automatically started from the Unit 1 and 2 control room.

b. Each hydro unit can be synchronized through the 230 kV overhead circuit to the startup transformers.

c. Each hydro unit can energize the 13.8 kV underground feeder.

4.6.2 Annually, the Keowee Hydro units will be started using the emergency start circuits in each control room to verify that each hydro unit and associated equipment is available to carry load within 25 seconds of a simulated requirement for engineered safety features.

4.6.3 Monthly, the Keowee Underground Feeder Breaker Interlock shall be verified to be operable.

4.6.4 During each refueling outage, for the affected unit, a simulated emergency transfer from the 4160 volt main feeder buses to the startup transformer (i.e., CT1, CT2 or CT3) and to the 4160 volt standby buses shall be made to verify proper operation.

4.6.5 Quarterly, the External Grid Trouble Protection System logic shall be tested to demonstrate its ability to provide an isolated power path between Keowee and Oconee.

4.6.6 Annually, it shall be demonstrated that a Lee Station combustion turbine can be started and connected to the 100 kV line. It shall be demonstrated that the 100 kV line can be separated from the rest of the system and supply power to the 4160 volt main feeder buses.

4.6.7 Batteries in the 125 VDC systems shall be tested as follows:

a. The voltage and temperature of a pilot cell in each bank shall be measured and recorded five times per week for the Instrument and Control, Keowee Hydro, and Switching Station batteries.

b. The specific gravity and voltage of each cell shall be measured and recorded monthly for the Instrument and Control, Keowee Hydro, and Switching Station batteries.

c. During each refueling outage, for the affected unit, a one-hour discharge test at the required maximum safeguards load shall be made on the Instrument and Control batteries.

d. Before initial operation and annually thereafter, a one-hour discharge test shall be made on the Keowee Hydro and Switching Station batteries.

4.6.8 The operability of the individual diode monitors in the Instrument and Control and Keowee Station 125 VDC systems shall be verified monthly by imposing a simulated diode failure signal on the monitor.

4.6.9 The peak inverse voltage capability of each auctioneering diode in the Instrument and Control, Switchyard and Keowee Hydro 125 VDC systems shall be measured and recorded semiannually.

4.6.10 The tests specified in 4.6.7, 4.7.8 and 4.6.9 will be considered satisfactory if control room indication and/or visual examination demonstrate that all components have operated properly.

Bases

The Keowee Hydro units, in addition to serving as the emergency power sources for the Oconee Nuclear Station, are power generating sources for the Duke system requirements. As power generating units, they are operated frequently, normally on a daily basis at loads equal to or greater than required by Table 8.5 of the FSAR for ESF bus loads. Normal as well as emergency startup and operation of these units will be from the Oconee Unit 1 and 2 Control Room. The frequent starting and loading of these units to meet Duke system power requirements assures the continuous availability for emergency power for the Oconee auxiliaries and engineered safety features equipment. It will be verified that these units are available to carry load within 25 seconds, including instrumentation lag, after a simulated requirement for engineered safety features. To further assure the reliability of these units as emergency power sources, they will be, as specified, tested for automatic start on a monthly basis from the Oconee control room. These tests will include verification that each unit can be synchronized to the 230 kV bus and that each unit can energize the 13.8 kV underground feeder.

The interval specified for testing of transfer to emergency power sources is based on maintaining maximum availability of redundant power sources.

Starting a Lee Station gas turbine, separation of the 100 kV line from the remainder of the system, and charging of the 4160 volt main feeder buses are specified to assure the continuity and operability of this equipment.

REFERENCE

FSAR Section 8

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies, all of which are prepressurized with Helium.

5.3.1.2 The fuel assemblies shall form an essentially cylindrical lattice with an active height of 144 in. and an equivalent diameter of 128.9 in. (2)

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 3-46. The full-length CRA contain a 134 inch length of silver-indium-cadmium alloy clad with stainless steel. The APSR contain a 36 inch length of silver-indium-cadmium alloy. (3)

5.3.1.4 Initial core and reload fuel assemblies and rods shall conform to design and evaluation described in the FSAR or Reload Report and shall not exceed an enrichment of 3.5 percent of U-235.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements. (4)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F. (5)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 ft^3.

REFERENCES

(1) FSAR Section 3.2.1

(2) FSAR Section 3.2.2

(3) FSAR Section 3.2.4

(4) FSAR Section 4.1.3

(5) FSAR Section 4.1.2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

AMENDMENT NO. TO FACILITY LICENSE NO. DPR-47

AMENDMENT NO. TO FACILITY 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

Introduction

By letter dated February 25, 1975 and as amended May 7, 1976, Duke Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2, and 3. The proposed changes would permit operation of Unit No. 2 as reloaded for cycle 2 operation. Included in the bases of the analyses performed are the Final Acceptance Criteria (FAC) for Emergency Core Cooling Systems, as required by the Commission's Order for Modification of License dated December 27, 1974.

Discussion

The Oconee Unit No. 2 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The cycle 2 reload will involve the removal of all of the Batch 1 fuel (56 assemblies) and the relocation of the Batch 2 and Batch 3 fuel. The fresh Batch 4 fuel (56 assemblies) will occupy primarily the periphery of the core and eight locations in its interior. Two of the new Batch 4 fuel assemblies are demonstration Mark C assemblies, each of which consists of a 17x17 array of fuel rods. A description of the program to irradiate the two Mark C assemblies in the cycle 2 core was provided by letter dated January 28, 1976. In addition, Babcock & Wilcox (B&W) Report BAW-1424, "Irradiation of Two 17x17 Demonstration Assemblies in Oconee 2, Cycle 2," January 1976, was provided which describes the mechanical, nuclear, and thermal-hydraulic characteristics of the two demonstration assemblies. Table 1 summarizes the reload core fuel assembly parameters.

TABLE 1

	Residual Fuel Assemblies Batch 2	Residual Fuel Assemblies Batch 3	New Fuel Assemblies Batch 4	
Fuel assembly type	Mark B-3	Mark B-3	Mark B-4*	Mark C
Fuel rod array	15x15	15x15	15x15	17x17
No. of assemblies in core	61	60	54	2
Initial fuel enrich., wt/% U235	2.75	3.05	2.64	2.64
Initial fuel density, % TD	92.5	92.5	93.5	94
Batch burnup, BOC, MWd/mtU	16,135	10,318	0	0
Fuel rod OD, in.	0.430	0.430	0.430	0.379
Fuel rod ID, in.	0.377	0.377	0.377	0.332
Fuel pellet OD, in.	0.370	0.370	0.370	0.324
Fuel pellet length in.	0.700	0.700	0.700	0.600, 0.375**
Undensified active fuel length, in.	144.0	144.0	142.6	143.0
Type of flexible spacer	Corrugated	Corrugated	Spring	Spring
Solid spacer material	ZrO_2	ZrO_2	Zr-4	Zr-4

*Two fuel assemblies have fuel rods raised 0.6 inch above bottom grillage.

**One assembly with 0.375-inch pellet only. One assembly with 11 fuel rods at 0.375-inch pellet length while the remaining rods have 0.600-inch pellets. The 0.600-inch length is of similar L/D as the Mark B assemblies. The smaller L/D is to investigate fabrication and loading techniques.

The licensee's reload analyses and Technical Specification changes submitted by letter dated February 25, 1976 were based on an originally planned 460 equivalent full power days (EFPD) of Unit No. 2 cycle 1 operation. The licensee, however, advised us by letter dated May 7, 1976 that cycle 1 operation was terminated early at 440 EFPD and, as a result, the burnup distribution in the Batch 2 and 3 fuel assemblies, which are to remain in the core for cycle 2 operation, will be different from that assumed in the original reload analysis. Based on a reanalysis of the new burnup distribution of the Batch 2 and 3 fuel assemblies, the licensee submitted by letter dated May 7, 1976 revisions to certain core physics parameters and those Technical Specifications which were affected. Also included in the May 7, 1976 submittal are the results of an analysis performed to determine the effects of fuel rod bow on Unit No. 2 cycle 2 operation.

Evaluation

1. Fuel Mechanical Design

The outside dimensions and configurations of the new Mark B-4 (Batch 4) fuel assemblies and the once-burned Mark B-3 fuel assemblies are identical except that the Mark B-4 have a spring-type flexible spacer and the Mark B-3 have a corrugated-type flexible spacer. This new fuel rod spacer design has been reviewed and found acceptable by us and is currently operating in the Oconee Unit No. 3 plant. The new Mark B-4 fuel assemblies therefore do not represent any unreviewed change in mechanical design from the reference cycle.

There are four demonstration fuel assemblies proposed for operation in Oconee Unit No. 2 cycle 2. Two of the demonstration assemblies are a raised fuel rod design. These assemblies are identical to the Mark B-4 assemblies, except that the fuel rods are raised 0.6 inches above the bottom grillage. These assemblies are being introduced in the cycle 2 core to investigate the raised fuel rod effect on rod bow.

Two Mark C fuel assemblies are to be placed in the cycle 2 core. These assemblies have a 17x17 fuel rod configuration. As described in Table 1, there are two different length fuel pellets used in these 17x17 assemblies. Also the fuel rod outside and inside diameters have been decreased in the Mark C demonstration assemblies. The Mark C demonstration assemblies are mechanically compatible and interchangeable with Mark B assemblies with the exception of the control rod component interface.

These mechanical design changes have been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that the fuel assembly mechanical design differences in the Oconee Unit No. 2 cycle 2 core are of negligible effect and that the once burned fuel is generally limiting.

Fuel rod cladding creep collapse analyses were performed for the three fuel batches which will be present in the Unit No. 2 cycle 2 core. The calculational methods, assumptions, and data have been previously reviewed and approved by the staff. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse. The most restrictive power profiles the new fuel assemblies may be exposed to were used in the Batch 4 analysis: The actual reactor operating history along with the most restrictive power histories were used in the analyses of the Batch 2 and Batch 3 fuel. The fuel cladding material properties are the same as those used in the CROV code. The analysis performed assumed a 2000-hour densification time (maximum creep), no fission gas production (maximum differential pressure), lower tolerance limit on cladding thickness, and upper tolerance limit on cladding ovality. Based on the analyses performed, the fuel rod design has been shown to meet the required design life limits for fuel cladding creep collapse and is therefore acceptable.

From the viewpoint of cladding stress (creep stress due to differential pressure, thermal stress due to temperature gradient and bending stress), neither the yield stress or ultimate strength of the cladding material will be exceeded in the cycle 2 core. The cladding stress estimated in the Unit No. 2 cycle 1 core will be limiting in the cycle 2 core, because of the lower prepressurization and lower fuel pellet density.

The Batch 4 fuel assemblies are not new in concept and do not utilize different component materials. In addition, the introduction of the four demonstration assemblies into the cycle 2 core has been shown to have an insignificant effect on the cycle 2 operation. Therefore, on the bases of the analysis presented we conclude that the fuel mechanical design for cycle 2 operation is acceptable.

2. Fuel Thermal Design

The fuel thermal design analysis was conducted using the TAFY-3 computer code, as described in "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," BAW-10044, May 1972, to establish heat flux limits to centerline melt. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in "Fuel Densification

Report," BAW-10055, Revision 1, June 1973. Modifications to BAW-10055 consisting of changes to the void probability, Fg, and size distribution Fk, have been previously reviewed and approved by us for use in the densification model.

As part of our interim evaluation of the TAFY code, the following modifications to the code were approved for use in "Technical Report on Densification of Babcock & Wilcox Reactor Fuels", July 6, 1973.

1) The code option for no restructuring of fuel has been used in this analysis in accordance with our interim evaluation of TAFY.

2) The calculated gap conductance was reduced by 25% in accordance with our interim evaluation of TAFY.

During cycle 2 operation the highest relative assembly power levels occur in Batch 3 fuel. The fuel temperature analysis for this fuel documented in the Oconee Unit No. 2 Fuel Densification Report is applicable for cycle 2 and is based on limiting beginning-of-cycle (BOC) conditions (zero burnup). Although Batch 4 fuel has a reduced active fuel length and a correspondingly higher average linear heat rate, the maximum predicted centerline temperature of this fuel is lower than that of Batch 3 fuel, even with the same peaking factors applied. This is due to the higher initial density of the Batch 4 fuel.

Based on the above, we conclude that the fuel thermal design for Oconee Unit No. 2 cycle 2 core is acceptable.

3. Nuclear Analysis

The reactor core physics parameters for Unit No. 2 cycle 2 operation were calculated using the PDQ07 computer code which has been previously approved by us for use. Since the Unit No. 2 core has not yet reached an equilibrium cycle, the minor differences in the physics parameters which exist between the cycle 1 and cycle 2 core are to be expected and are not significant.

The effects of the four demonstration fuel assemblies in the Batch 4 fuel on the cycle 2 nuclear design have been reviewed and shown to be negligible.

In view of the above and the fact that startup tests (to be conducted prior to power operation) will verify that the critical aspects of the core performance are within the assumptions of the safety analysis, we find the licensee's nuclear analysis for cycle 2 to be acceptable.

4. Rod Bow Penalty

The effect of fuel rod bow was evaluated with consideration given to the hot channel power spike and the effect of flow area reduction on the Departure from Nucleate Boiling Ratio (DNBR). These phenomena were evaluated separately since they are mutually exclusive and one cannot exist when the other is present. In a letter of May 7, 1976, the licensee summarized the results of the rod bow analysis in which the methods described in its letter of February 27, 1976 were used. The results of this analysis indicate the following:

Effect of Rod Bow on DNBR

1) The rod bow effect on the flow area of the hot channel is adequately compensated for by the flow area reduction factor employed in the hot channel analysis, and

2) The power spike caused by the rod bow effect away from the hot channel when added to the hot rod in the area of the minimum DNBR, shows that the Unit No. 2 cycle 2 DNBR limit (1.30) conservatively accounts for the effects of rod bowing.

Local Power Peaking Effects of Fuel Rod Bow

1) A power spike of 1.6% may occur as a result of rod bowing during cycle 2 operation.

The effects of the rod bow power spike of 1.6% on the limiting heat rate criteria (central fuel melt -kW/ft limit and LOCA - kW/ft limit) have been evaluated and compensated for by reducing the quadrant power tilt limit for Oconee Unit No. 2 from 4.92% to 3.41%. We have reviewed the licensee's analysis on the effects of rod bow and have found the results to be acceptable.

5. Thermal-Hydraulic Analysis

The major acceptance criteria for the thermal-hydraulic design are specified in Standard Review Plan (SRP) 4.4. These criteria establish the acceptable limits on DNBR and on the Critical Power Ratio (CPR). The thermal-hydraulic analysis for the Unit No. 2 cycle 2 reload were made using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the cycle 2 core and are discussed below.

Reactor Coolant System Flow Rate

The reactor coolant flow rate was accurately measured during cycle 1 operation and determined to be 111.5% of the system design flow. The licensee has proposed to take credit in the cycle 2 thermal-hydraulic analysis for the fact that actual system flow is higher than design flow, and has also included conservatisms representing uncertainties in the measurement of the flow. Considering these conservatisms and, to be consistent with the flow rate used in the Unit No. 1 cycle 3 thermal-hydraulic analysis, the licensee has utilized a flow rate of 107.6% in the Unit No. 2 cycle 2 analysis.

In the past, a 4.6% reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for the Oconee units. This penalty was assessed to allow for the potential of a core vent valve being stuck open during normal operation. The core vent valves are incorporated into the design of the reactor internals to preclude the possibility of a vapor lock developing in the core following a postulated cold-leg break. By letter dated January 30, 1976, we advised the licensee that we had concluded that sufficient evidence had been provided by B&W to assure that the core vent valves would remain closed during normal operation and that it could, therefore, submit an application for a license amendment to eliminate the vent valve flow penalty. In addition, the submittal should include appropriate surveillance requirements to demonstrate, each refueling outage, that the vent valves are not stuck open and that they operate freely. By letter dated June 11, 1976, the licensee proposed the surveillance requirements referred to by us in our January 30, 1976 letter.

By letter dated June 15, 1976, the licensee advised us that an error had been identified in the Oconee Unit No. 2 cycle 2 DNBR fuel densification penalty calculations. This error resulted from the use of inconsistent heat flux (flux shape) and enthalpy rise calculations in evaluating the DNBR densification peanlty. The revised calculations indicate that the reduction in the DNBR margin due to fuel densification effects and the reduction in power peaking margin should be greater than those values previously identified. In the analysis incorporating the revised DNBR densification penalty, the licensee took credit for removal of the flow penalty previously assessed for a stuck open core vent valve, as discussed above. The four-pump Pressure-Temperature (P-T) limit curve based on this new analysis is less restrictive than the P-T limit curve as included in the licensee's May 7, 1976 submittal. The licensee indicates that since the variable low pressure trip setpoint is based on the four-pump P-T limit curve, the variable low pressure trip setpoint included in the May 7, 1976 is conservative. In addition, with regard to the flux/flow trip setpoint, which is based on a two-pump coastdown analysis, the licensee indicates that in the analysis incorporating the revised DNBR densification penalty and removal of the core vent valve flow penalty, a flux/flow trip setpoint of 1.08 can be justified. This setpoint includes a 1.2% flow error to account for the precision of the various components in the RPS flow instrument string. The flux/flow trip setpoint of 1.07 as proposed in the licensee's May 7, 1976 submittal for Unit 2 cycle 2 is therefore conservative in comparison to the 1.08 value identified by the licensee in the new analysis.

The Oconee Technical Specifications include monthly and annual surveillance requirements for the flux/flow comparator instrumentation channels. The monthly calibration check verifies the trip setpoint using known test signals and the annual requirement includes the calibration of the entire reactor coolant flow instrumentation string using an actual differential pressure as input to the system d/p cells. In addition, a surveillance requirement exists which requires that the reactor coolant system flow be verified to be at least 141.3 x 10^6 lbs/hr (107.6% design flow) at least once each fuel cycle.

There are differences in the flow resistance between the current Mark B-3 fuel assemblies and the new fuel assemblies. The flow resistance for the Mark B-4 fuel assemblies, which includes the two raised fuel rod assemblies, is less than that measured for the Mark B-3 assemblies. Also, the Mark C assemblies have a greater flow resistance than either of the other two fuel assembly types. These differences have been analyzed and from this analysis it was concluded that the Mark B-3 assemblies are limiting for the Oconee Unit No. 2 cycle 2 operation. This analysis considered the possible introduction of core cross flow due to the different flow resistances and this phenomenon was shown to be a negligible effect.

In summary, the licensee has proposed that a reactor coolant flow rate based on actual measured flow rather than design flow be used in the Unit No. 2 cycle 2 thermal hydraulic analysis. The licensee has also applied for elimination of a 4.6% vent valve flow penalty. This application includes revisions in the cycle 2 DNBR fuel densification penalty. Based on our review, we have concluded that the licensee has included appropriate conservatisms in its analysis and that existing Technical Specifications provide added assurance that the reactor coolant flow is properly monitored. Based on the above we find the use of measured flow in the thermal-hydraulic analysis to be acceptable and that the Technical Specifications related to the cycle 2 thermal-hydraulic analysis, as proposed in the May 7, 1976 submittal, are also acceptable.

Critical Heat Flux Correlation (CHF)

The W-3 CHF correlation was used in the reference cycle. For the Unit No. 2 cycle 2 thermal-hydraulic analysis the BAW-2 CHF correlation was used. The BAW-2 correlation was approved for the Oconee Unit 1 cycle 2 and 3 cores. Two modifications to the BAW-2 correlation were introduced for its application in the Unit 1 cycle 3 core and are also used in the Unit 2 cycle 2 thermal hydraulic analysis. These modifications are:

1. An extension downward from 2000 psia to 1750 psia of the pressure range applicable to the correlation, and

2. A reduction in the DNBR from 1.32, (representing a 99% confidence level that 95% of the hot rods will not experience DNB) to 1.30 (representing a 95% confidence level that 95% of the hot rods will not experience DNB).

Item 1. above, was based on a review of rod bundle CHF data taken at pressures below 2000 psia which indicate that the BAW-2 correlation conservatively predicts the data in this range. Item 2. above is consistent with the standard review plan and industry practice.

We have previously reviewed the modifications identified above to the BAW-2 correlation and have concluded that they are acceptable for use in the Unit No. 2 analysis. In addition, we recently completed a re-evaluation of the BAW-2 CHF correlation to verify its continued suitability in relation to available rod bundle data. We determined that the BAW-2 correlation continues to be an acceptable correlation over the pressure, quality, mass flux, rod diameter and rod spacing range of its original data base.

6. Accident and Transient Analysis

The accident and transient analysis provided by the licensee demonstrates that the Oconee FSAR analyses conservatively bounds the predicted conditions of the Unit 2 cycle 2 core and is therefore acceptable.

7. Startup Program

The startup program tests verify that the core performance is within the assumption of the safety analysis and provide the necessary data for continued plant operation. The licensee has agreed to provide certain confirmatory information from the startup program. Specifically, a measurement of the temperature reactivity coefficient will be provided for at least two control rod configurations, i.e., all-rods-out and a normal rod configuration. In addition, the licensee has agreed to provide the measurement of at least two control rod group worths.

8. ECCS Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR 50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, the licensee, by letter dated July 9, 1975 and as supplemented August 1, 1975, submitted an ECCS reevaluation and related Technical Specifications. Included in the reload application of February 25, 1976 and as revised May 7, 1976, the licensee has submitted the related Technical Specifications for Unit 2, cycle 2. The reevaluation and Technical Specifications were submitted using the B&W ECCS evaluation model as described in BAW-10104 of May 1975.

The background of the staff review of the B&W ECCS evaluation model and its application to Oconee is described in the staff SER for this facility dated December 27, 1974, issued in connection with the Order for Modification of License. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 and the Supplement to the Status Report of November 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier version of the B&W model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Oconee 2 ECCS evaluation which is covered by this safety evaluation report properly conforms to the accepted model. The licensee's July 9, 1975 submittal contains documentation by reference to B&W Topical Reports of the revised ECCS model (with the modifications described in our December 27, 1974 SER) and a generic break spectrum appropriate to Oconee 2; BAW-10104, May 1975 and BAW-10103, June 1975, respectively.

The generic analysis in BAW-10103 identified the worst break size as the 8.55 ft^2 double-ended cold leg break at the pump discharge with a $C_D = 1.0$. The table below summarizes the results of the LOCA limit analyses which determine the allowable linear heat rate limits as a function of elevation in the core for Oconee Unit 2:

Elevation (ft)	LOCA Limit (kw/ft)	Peak Cladding Temperature (°F)		Max. Local Oxidation (%)	Time of Rupture (sec)
		Ruptured Node	Unruptured Node		
Oconee 2					
2	15.5	2002	1978	3.92	12.25
4	16.6	2136	2072	4.59	13.01
6	18.0	2066	2146	5.16	15.55
8	17.0	1743	2110	5.19	15.01
10*	16.0	1642	1931	2.93	39.20

*See discussion below.

The maximum core-wide metal-water reaction for Oconee 2 was calculated to be 0.557 percent, a value which is below the allowable limit of 1 percent.

As shown in the tabulation, the calculated values for the peak clad temperature and local metal-water reaction were below the allowable limits specified in 10 CFR 50.46 of 2200°F and 17 percent, respectively. BAW-10103 has also shown that the core geometry remains amenable to cooling and that long-term core cooling can be established.

The staff noted during its review of BAW-10103 that the LOCA limit calculation at the 10-foot elevation in the core showed reflood rates below 1 inch/second at 251 seconds into the accident (Section 7.2.5). Appendix K to 10 CFR 50.46 requires that when reflood rates are less than 1 inch/second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer. As indicated by the staff in the Status Report of October 1974 and supplement of November 1974, a steam cooling model for reflood rates less than 1 inch/second was not submitted by B&W for staff review. The steam cooling model submitted by B&W in BAW-10103 is therefore considered to be a proposed model change requiring further staff review and ACRS consideration. Accordingly, B&W was informed that until the proposed steam cooling model is reviewed, the heat transfer calculation at the 10-foot elevation during the period of steam cooling specified in BAW-10103 must be further justified. In lieu of using their proposed steam cooling model, B&W has submitted the results of calculations at the 10-foot elevation using adiabatic heatup during the steam cooling period, where this period is defined by B&W as the time when the reflood rate first goes below 1 inch/second to the time that REFLOOD predicts the 10-foot elevation is covered by solid water. The new calculated peak cladding temperature, local metal-water reaction and core-wide metal-water reaction at the 10-foot elevation are 1946°F, 3.02%, and .647% respectively. These values remain below the allowable limits of 10 CFR 50.46 and are acceptable to the staff. Until a steam cooling model has been accepted by the staff, these values will serve as the LOCA results for Oconee 2 at the 10-foot elevation.

We have reviewed the Technical Specifications proposed by the licensee in the July 9, 1975 submittal, to assure that operation of Oconee Unit 2 will be within the limits imposed by the Final Acceptance Criteria (FAC) for ECCS system performance. These criteria permit an increase in the allowable heat generation rate from 15 to 16 Kw/ft at the 10 foot elevation, as compared to the Interim Acceptance Criteria (IAC). For Unit 2, the LOCA-related heat generation limits are bounded by the generic limit of 18.0 Kw/ft as contained in BAW-10103. We have concluded that the proposed Technical Specifications, as submitted for Unit 2 cycle 1 operation meet the necessary FAC and are acceptable. Since Oconee Unit 2 is currently undergoing refueling for cycle 2 operation, we have also reviewed the proposed Technical Specifications for cycle 2 operation to assure that they also meet the FAC. We have determined that the LOCA related heat generation limits used in the BAW-10103 LOCA limits analysis are conservative compared to those calculated for this reload. Based on the above, we find that the proposed Technical Specifications for cycle 2 operation also meet the FAC of ECCS performance and are therefore acceptable.

Our review of other plant-specific assumptions discussed in the following paragraphs regarding Oconee 2 analyses addressed the areas of single failure criteria, long-term boron concentration, potential submerged equipment, partial loop operation, ECCS valve interlocks and the containment pressure calculation.

Single Failure Criterion

Appendix K to 10 CFR 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. The licensee has assumed all containment cooling systems are operating to minimize containment pressure and has separately assumed the loss of a 4160 Volt Feeder Bus resulting in the operation of only one LPI and one HPI pump to minimize ECCS cooling.

A review of Unit No. 2 piping and instrumentation diagrams indicated that spurious actuation of certain motor-operated valves could affect the appropriate single failure assumptions. A spurious actuation of core flooding tank (CFT) vent valves CF-5 or CF-6 would result in a decrease in CFT pressure. Since it is clear that CFT pressure is important to mitigating the consequences of a LOCA, Technical Specifications require that the normally closed motor-operated valves CF-5 and CF-6 have their breakers locked open and tagged except when adjusting CFT pressure.

To further minimize the potential for a water hammer due to the discharge of ECC water into a dry line, valves LP-21 and LP-22 will be left in the open position during normal operation. This maintains the LPI lines filled with a continual supply of water from the BWST due to the available static head built into the system. The normal value lineup in the HPI system provides a similar supply of water to the HPI pumps. In addition, Technical Specifications require the monthly venting of ECCS (HPI and LPI) pump casings to ensure that no air pockets have formed. Such venting will also be performed prior to any ECCS flow tests.

The Engineered Safeguards Protection System (ESPS) monitors parameters to detect the failure of the reactor coolant system and initiates operation of the high and low pressure safety injection systems, building isolation, and reactor building (containment) cooling and spray systems. The ESPS consists of eight two-out-of-three coincidence logic networks which actuate equipment in four safeguards systems. Therefore, each system is actuated at least by two redundant two-out-of-three logic trains.

Typically, one ESPS train actuates one piece of equipment in one safeguards system while the opposite train actuates a redundant component in the same safeguards system. However, whenever any system has a third piece of equipment, the licensee's design uses both ESPS trains to actuate this third component. We requested that the licensee determine if any single failure could compromise redundant trains. The licensee provided a control circuit schematic typical of that which would be used for all safeguards equipment actuated by redundant trains. Since two relay failures in redundant safeguards cabinets would be required to compromise redundant trains, this design provides adequate isolation between trains. This configuration is similar to that used in other nuclear power plants whose designs have been found acceptable. Therefore, this portion of the actuation system is in conformance with the fundamental single failure criterion at the electric component level.

The licensee has provided information identifying all types of equipment located inside the Reactor Building which are required to be operable during and after a LOCA. Included in this list are valve motors, fan cooler motors, penetrations, cables, and all required instrumentation. Qualification parameters include containment pressure, temperature, radiation, humidity, and chemistry. The licensee has provided sufficient information to give adequate assurance that type tests representing conditions that will be encountered in the LOCA environment were performed on samples of equipment required to function during and after a LOCA. Since this information was originally evaluated during the licensing phase of the FSAR application and an operating license was subsequently issued, we find that there is sufficient assurance that all equipment required to function in the LOCA environment is qualified.

Emergency Electric Power

1. Introduction

The design of the power distribution system for the Oconee Nuclear Station consists of two 87.5 MVA hydroelectric power generators at Keowee Dam that serve as onsite emergency power sources. One of these hydroelectric units is capable of supplying all the essential loads of all the Oconee Units. There are two diverse methods of feeding emergency power to each of the three Oconee Units. These are (1) an overhead line from the Keowee Dam through the 230KV site switchyard and respective unit startup transformers whenever offsite power is unavailable, and (2) a 13.8KV underground feeder cable feeding each unit's safeguard buses through a single stepdown transformer, redundant feeder breakers (SK1 and SK2) and 4160V standby buses.
In addition to the two Keowee hydro units, backup power is available from one of three gas turbine generators located 30 miles away at the Lee Steam Station via an independent overhead 100KV transmission system.

We reviewed the design of this system on the following basis: The design of the entire emergency electric power system, including generating sources, distribution system and controls, is such that a single failure of any single electric component will not preclude the Emergency Core Cooling System of either Units 2 or 3 from performing its function.

2. Standby bus breakers (SK1 and SK2) from the Keowee underground feeder

Breakers SK1 and SK2 are provided to connect the Keowee underground feeder cable to the Oconee redundant 4160 volt standby buses. These buses serve as standby power sources to all three Oconee Units. In this way the Engineered Safeguards Protection Systems of all three Oconee Units interface with the SK1 and SK2 breaker controls.

Each breaker can be actuated by an engineered safeguards (ES) signal from any of the three Oconee Units. Each Oconee Unit provides one ES input to breaker SK1 and a separate ES input from the redundant ES channel to breaker SK2. The signals, derived from dry contact outputs, interface directly into the breaker control logic. It has been determined that these contacts provide adequate isolation protection.

Each SK breaker requires a primary control source to operate its close-trip circuits and a secondary source to operate a redundant trip circuit. Power for these DC control circuits is supplied from Unit No. 1 control power panelboards. Panelboard 1DIC provides the primary control source to breaker SK1 and a secondary source to breaker SK2. Similarly, Panelboard 1DID provides the primary control source to breaker SK2 and the secondary source to breaker SK1. Each of these control power panelboards is supplied from Unit No. 1 and/or Unit No. 2 control batteries through isolating diodes.

Since breakers SK1 and SK2 are provided with individual redundant controls for the Keowee underground feeder and since there is a redundant overhead emergency feed from the Keowee hydros, no credible single electrical component failure can result in the loss of emergency power to the Engineered Safeguards buses of Units Nos. 2 or 3.

3. Electrical Interlocks

The 13.8KV underground feeder from Keowee to Oconee is fed by either one of Keowee's two hydros. One Keowee unit is always dedicated to the underground feeder through its respective breaker (Keowee Unit 1 - ACB 3 and Keowee Unit 2 - ACB 4). These Keowee breakers have an electrical interlock that prevents simultaneous closure of both breakers.

The licensee has stated the failure of this electrical interlock alone cannot prevent the Keowee units from providing emergency power to Oconee. All of the following conditions would have to exist to compromise the ability of the Keowee units to provide emergency power to Oconee:

a. Failure of the electrical interlock for ACB 3 and ACB 4.

b. Failure of the operator to follow established procedures. ACB 3 and ACB 4 are controlled manually from either the Oconee Units Nos. 1 and 2 control room or from the Keowee control board. The operator's procedures require that a closed underground feeder breaker must be in the open position before closing the other breaker; therefore, it would require an operator error to parallel the keowee units.

c. The Keowee units would have to be in a condition that would result in an electrical failure. Those conditions are:

(1) The two units must be running without being synchronized together. This is unlikely since the unsynchronized condition exists only temporarily when the units are being placed on line for peaking.

(2) One unit is operating and the other is shut down.

Nevertheless, we identified a situation where an undetected failure of the interlock coupled with an operator error could compromise redundant power sources. To preclude the likelihood of an undetected failure, Technical Specifications will be required to include a monthly surveillance of this interlock. By including periodic testing of this interlock, we are satisfied that the same level of safety has been achieved for this interlock as exists for all other safeguards equipment that are tested monthly.

In addition, the licensee has stated that there are no electrical interlocks between redundant portions of the ECCS and supporting subsystems.

With this commitment and the above Technical Specification change there is sufficient assurance that no single failure of an interlock will compromise Emergency Core Cooling capability.

The Low Pressure Service Water System (LPSW) is the only shared safeguards or safeguards support system at Oconee Nuclear Station. This system is shared between Oconee Units Nos. 1 and 2 and contains three redundant LPSW pumps, any one of which is capable of supplying the system requirements.

One LPSW pump derives its power from Unit No. 1 switchgear group 1TC. The second LPSW pump is powered from Unit No. 2 switchgear group 2TC. The third pump is capable of receiving power from either Unit No. 1 switchgear group 1TD or Unit No. 2 switchgear group 2TD.

A manual transfer switch is provided to select the power source for the third LPSW pump. This switch is Kirk Key interlocked with both the Unit No. 1 and Unit No. 2 4160 volt feeder breakers to preclude the possibility of crossconnecting the two units' switchgear buses together. Manual operation of the transfer switch not only transfers pump power, but also transfers pump control circuits to the appropriate configuration.

Because of the redundancies provided in the LPSW system, no single failure can result in a loss of Low Pressure Service Water to the plants.

4. Availability of Keowee Units

The licensee has stated that based on previous hydroelectric experience, the cumulative need to dewater the penstock and hence remove both Keowee units from service can be expected to be limited to about one day a year plus perhaps four days every tenth year.

We requested that the licensee provide information on the outage of both Keowee units in order to substantiate the previous bases for their acceptance. Outages of both units are as follows:

July 24, 1973	10:11 - 10:14	Emergency start test
January 16, 1974	1345 - 1500	Keowee minimum flow test
August 17, 1974	0730 - Aug 18 0130	Keowee inspection
February 7, 1975	0910 - 1030	Keowee minimum flow test
May 26, 1976	0910 - 1030	Keowee minimum flow test

In all cases, the Lee combustion turbine was in operation through the isolated 100KV transmission line prior to Keowee removal from service.

As can be seen, the total outage time of both Keowee units has been less than 24 hours since July 24, 1973. This trend is well within the 24 hours a year predicted outage.

We find that the basis for the availability of both Keowee hydroelectric generators has been substantiated by the licensee's record of outage times to date.

5. Seismic Qualification of the Keowee Overhead Emergency Electric Power Source

As discussed above, one of the two redundant methods of supplying emergency power from the Keowee Hydro Units is via an overhead line through the 230 KV site switchyard. In order to take credit for this source of emergency electrical power following a LOCA, that portion of the 230KV switchyard used must be designed to the same seismic criteria as the Oconee Units were. Since the Oconee FSAR does not specifically address the seismic qualification of this part of the emergency power system, we requested that the licensee provide confirmatory information that the overhead emergency power path was properly qualified.

In response to our request, the licensee advised us that the emergency power path through the 230KV switchyard had been seismically designed to withstand the .15g earthquake referred to in the Oconee FSAR for Class I structures. We have requested and the licensee has agreed to furnish additional supporting information identifying the details of the seismic design of this portion of the emergency power system. In view of the fact that the licensee has verified that this portion of the system has been seismically designed and considering the fact that confirmatory details of the design criteria and analysis are forthcoming, we conclude that it is acceptable for the Oconee Station to operate pending our review of this confirmatory information. Also considered in our conclusion was the extremely low probability of a seismic event at the Oconee Station.

The licensee has committed to provide the confirmatory information requested in sufficient time for us to complete our review prior to the restart of Oconee Unit 3 following refueling in the Fall of 1976.

Submerged Electrical Equipment

The licensee has identified the following electrical equipment that may become submerged as a result of a LOCA.

Letdown Cooler 1A Inlet Valve HP-1
Letdown Cooler 1A Isolation Valve HP-3
Letdown Cooler 1B Inlet Valve HP-2
Letdown Cooler 1B Isolation Valve HP-4
Letdown Cooling Inlet Valve CC-1
Letdown Cooling Inlet Valve CC-2
Quench Tank Suction Valve CS-5
Core Flood Tank 1A Outlet Valve CP-1 Controller
Steam Generator 1A Level Detector (5)
Steam Generator 1B Level Detector (5)
Reactor Coolant Pump Oil Tank Level Detector (4)
Reactor Coolant Pump Standpipe Level Detector (4)
Letdown Cooling Component Cooling Outlet Temperature Detector (2)
Quench Tank Level Detector
Quench Tank Press Detector
Quench Tank Heat Exchanger Discharge Temperature Detector
Quench Tank Temperature Detector
Quench Tank Heat Exchanger Inlet Valve CC-49 Position Indication
Quench Tank Heat Exchanger Outlet Valve CC-53 Position Indication
Quench Tank Cooler Inlet Valve CS-13 Position Indication
Quench Tank Cooler Outlet Valve CS-14 Position Indication
Quench Tank Outlet Valve CS-3 Position Indication
Core Flood Tank 1A Level Detector (2)
Core Flood Tank 1B Press Detector
Reactor Building Normal Sump Temperature Detector
Reactor Building Normal Sump Level Detector
Reactor Building Emergency Sump Level Detector
Lighting Panels EL1 and WL1
Reactor Vessel Water Level Detector
Telephones
PA Speakers
PA Amplifier
PA Power Supply

The first eight items above are safety related equipment which are required to mitigate the consequences of an accident. However, submergence of Core Flood Tank outlet valve motor controller, CP-1, will not affect ECCS capability because the valve is locked open with electric power disconnected outside the reactor building during normal plant operation. In addition, the valve is not required to operate subsequent to a LOCA. The other seven valves are automatically actuated by an engineered safeguards actuation signal and will close before becoming submerged. After submergence these valves will remain in the closed position and will not reopen as a result of flooding.

The remaining items listed above are not considered necessary to place the reactor in a shutdown condition nor to mitigate the consequences of a LOCA. Therefore, the failure of the equipment to function has no safety significance and there is no impact on safety due to submergence of electrical equipment.

The electrical power for the aforementioned equipment is fed from Non-Class 1E power sources except for the following:

1. Reactor Coolant Pump Oil Tank Level Detectors (4)
2. Letdown Cooler 1A Isolation Valve HP-3
3. Letdown Cooler 1B Isolation Valve HP-4
4. Quench Tank Suction Valve CS-5

The licensee has reviewed the circuit breaker and fuse coordination scheme for these circuits and has determined that there is adequate protection so that the safety function of other Class 1E equipment is not rendered inoperative. However, the licensee has identified a situation in which the flooding of limit switches on the three valves (items 2, 3 and 4 above) could possibly result in the loss of the normal control power to an engineered safeguard cabinet. To preclude this possible failure, the licensee has agreed to install fuses in the circuits from the valve limit switches to the safeguard cabinets prior to September 1, 1976. These fuses will be properly coordinated with the circuit breakers to ensure that normal control power to that cabinet is not lost because of submergence. We find this to be acceptable.

Single Failure Conclusion - On the basis of our review, including the above indicated changes to Technical Specifications and commitments by the licensee, we find that there is sufficient assurance that the ECCS will remain functional after the worst damaging single failure of ECCS equipment at the component level has occurred.

Containment Pressure

The ECCS containment pressure calculations for Oconee Class plants were performed generically by B&W for reactors of this type as described in BAW-10103 of June 1975. Our review of B&W's evaluation model was published in the Status Report of October 1974 and supplement of November 1974.

We concluded that B&W's containment pressure model was acceptable for ECCS evaluations. We required that justification of the plant-dependent input parameters used in the containment analyses be submitted for our review of each plant. A containment pressure calculation specific to Oconee 2 was submitted in the licensee's submittal of July 9, 1975.

Justification for the containment input data was submitted for Oconee Unit 2 by letter dated October 10, 1975. This justification allows comparison of the actual containment parametersffor Unit 2 with those assumed in the July 9, 1975 submittal and BAW 10103 of June 1975. The licensee has evaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat-removal systems with regard to the conservatism for the ECCS analysis. This evaluation was based on as-built design information. Since the minimum containment pressure following a LOCA is more limiting, the containment heat removal systems were assumed to operate at their maximum capacities, and minimum operation values for the spray water and service water temperatures were assumed. The containment pressure analysis was demonstrated to be conservative for Unit No. 2.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Oconee 2 is conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR 50 of the Commission's regulations.

Long-Term Boron Concentration

We have reviewed the proposed procedures and the system designed for preventing excessive boric acid buildups in the reactor vessel during the long-term cooling period after a LOCA. By letter dated December 18, 1975, the licensee committed to the implementation of procedures for Unit 2 which would allow adequate boron dilution during the long-term and which will comply with the single failure criterion. These procedures will employ a hot leg drain network similar to the concept described in BAW-10103. To employ a single failure proof mode, the licensee recently completed modifications during the current cycle 2 refueling outage. The modification consists of the addition of one drain line from the decay heat drop line to the reactor building sump. The line (installed upstream of the DHR isolation valves LP-1 and LP-2) includes two qualified motor-operated valves. The existing flowpath through valves LP-1, LP-2, LP-3 and LP-4 to the "A" LPI pump suction or to the reactor building sump through valve LP-19 provides the alternate flowpath to meet the single failure criteria. By letter dated February 24, 1976, the licensee indicated its intention to test the design and installation of the drain lines by conducting a preoperational test prior to reactor startup. In addition, by letter dated March 4, 1976, the licensee indicated its intent to install flow equipment to provide positive indication of flow in the drain lines. This equipment will not be installed for cycle 2 operation, however, this is acceptable to us because the drain line modification will be tested prior to cycle 2 startup and we will have the opportunity to review this design prior to cycle 3 operation. We have concluded that the licensee's proposal to prevent long-term boron concentration is acceptable and that the preoperational test to confirm proper installation and functioning will provide adequate assurance during cycle 2 operation with the system will function under post-LOCA conditions.

Partial Loop Analyses

To allow an operating configuration with less than four reactor coolant pumps on the line (partial loop), the staff requires an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operating mode(s). By letter dated August 1, 1975, the licensee submitted an analysis for partial loop operation with one idle reactor coolant pump (three pumps operating). Using a reduced power level of 77% of rated power, B&W performed this analysis assuming the worst-case break (8.55 ft^2 DE, $C_D = 1$) and maximum Linear Heat Generation Rate (LHGR) (18.0 kw/ft) from the 4-pump analysis discussed above. The worst break selected was located in the active leg of the partially idle loop. Placing the break at the discharge of the pump in an active cold leg of the partially idle loop (instead of at the discharge of the pump in an active cold leg of the fully active loop) yields the most degraded positive flow through the core during the first half of the blow-down and results in higher cladding temperatures. The maximum cladding temperature for the one-idle-pump mode of operation was 1766°F. A staff review of all input assumptions and conclusions resulted in a set of inquiries which were answered by the licensee's letter of October 31, 1975 and B&W's letter of October 10, 1975. The results of a new analysis were submitted to reflect a more appropriate value of initial pin pressure. The original partial loop analysis contained in the licensee's letter of August 1, 1975, used an initial pin pressure of 1600 psi. As was demonstrated in the time-in-life sensitivity study, submitted by letter dated August 1, 1975, the worst pin pressure for this analysis should have been 760 psi. The maximum cladding temperature for the re-analysis is 1784°F, a value which is within the criterion of 10 CFR 50.46. Therefore, this analysis may be used to support Duke Power Company's proposed operation with one idle reactor coolant pump.

Since an analysis of ECCS cooling performance with one idle reactor coolant pump in each loop has not been submitted, power operation in this configuration is limited by Technical Specifications to 24 hours.

We consider the probability of a LOCA occurring within a 24 hour period to be extremely remote and, based on the operating history of the Oconee Units, it is anticipated that this pump configuration will occur very infrequently.

Single loop operation (i.e., operation with two idle pumps in one loop) is prohibited, by Technical Specifications, without notifying the Commission.

We have completed the review of the Oconee 2 ECCS performance re-analysis and have concluded:

(a) The proposed Technical Specifications are based on a LOCA analysis performed in accordance with Appendix K to 10 CFR 50.

(b) The ECCS minimum containment pressure calculations were performed in accordance with Appendix K to 10 CFR 50.

(c) The single failure criterion will be satisfied.

(d) The proposed procedures for long-term cooling after a LOCA are acceptable. The implementation of these procedures during the cycle 2 refueling outage is required to provide assurance that the ECCS can be operated in a manner which would prevent excessive boric acid concentration from occurring. A commitment by the licensee to install the positive indication to show that the hot leg drain network is working during post-LOCA conditions is required and has been received by letter dated March 4, 1976.

(e) The proposed mode of reactor operation with one idle reactor coolant pump is supported by a LOCA analysis performed in accordance with Appendix K to 10 CFR 50. Operation with one idle pump in each loop is restricted to 24 hours. Requests for single loop operation will be reviewed on a case-by-case basis.

We have completed our evaluation of the licensee's Unit 2 cycle 2 reload application and conclude that the licensee has performed the required analyses and has shown that operation of the cycle 2 core will be within applicable fuel design and performance criteria. In addition, we conclude that the licensee's proposed Technical Specification changes meet the Final Acceptance Criteria based on an acceptable ECCS model conforming to the requirements of 10 CFR 50.46 and that the restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License should be terminated and replaced by the limitations established in accordance with 10 CFR 50.46.

We have determined that the amendment does not authorize a change in e 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 amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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: June 30, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS 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 Amendments Nos. 27, 27, and 23to 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 Sation Units Nos. 1, 2, and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments (1) revise the Technical Specifications to establish operating limits for Unit 2 cycle 2 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR Section 50.46 and (2) terminate the operating restrictions imposed on Unit 2 by the Commission's December 27, 1974 Order for Modification of License.

The applications for these amendments comply 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. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with item (1) above was published in the FEDERAL

REGISTER on April 12, 1976 (41 FR 15370) and in connection with item (2) above was published August 13, 1975 (40 FR 34028). No request for a hearing or petition for leave to intervene was filed following notice of the proposed actions.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental 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 applications for amendments dated February 25, 1975, as revised May 7, 1976, and dated June 11, 1976, (2) Amendments Nos. 27 , 27 , and 23 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, NW., Washington, D.C. 20555 and at the Oconee County Library, 201 South Spring Street, 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 30th day of June, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

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