Docket File

OCT 22 1976

Dockets Nos. 50-269/270/287

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. 34, 34 and 31 for Licenses Nos. DPR-38, DPR-47 DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the station Technical Speicifications and are in response to your request dated July 21, 1976, as supplemented August 20, October 7, October 19, October 20, and October 20, 1976.

These amendments (1) revise the Technical Specifications to establish operating limits for Unit 3 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 3 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

W

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Duke Power Company

OCT 22 1976

Enclosures:
1. Amendment No. 34 to DPR-38
2. Amendment No. 34 to DPR-47
3. Amendment No. 31 to DPR-55
4. Safety Evaluation
5. Federal Register Notice

cc w/encl:
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
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Walhalla, South Carolina 29691

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

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

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DATE	10/20/76	10/ /76	10/ /76	10/ /76	10/ /76	10/ /76

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated July 21, 1976, as supplemented August 20, October 7, October 19, October 20, and October 20, 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.

OFFICE➤						
SURNAME➤						
DATE➤						

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

Original signed by

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 22 1976

OFFICE						
SURNAME						
DATE						

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated July 21, 1976, as supplemented August 20, October 7, October 19 October 20, and October 20, 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.

OFFICE➤						
SURNAME➤						
DATE➤						

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

Original signed by

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 22 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated July 21, 1976, as supplemented August 20, October 7, October 19, October 20, and October 20, 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.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

Original signed by

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: OCT 22 1976

OFFICE ➤						
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ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 34 TO DPR-38

AMENDMENT NO. 34 TO DPR-47

AMENDMENT NO. 31 TO DPR-55

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

Revise Appendix A as follows:

Remove the following pages:

2.1-3c	3.5-8	3.1-17
2.1-3d	3.5-9	
2.1-6	3.5-10	
2.1-9	3.5-11	
2.1-12	3.5-16	
2.3-2	3.5-16a	
2.3-3	3.5-17	
2.3-7	3.5-20	
2.3-10	3.5-23	
2.3-13	3.5-24	
3.5-7	4.1-9	

Insert identically numbered pages, as above.

Add pages:

3.5-20a
3.5-20b
3.5-23a
3.5-23b

Delete pages:

3.17-1
3.17-2

Bases - Unit 3

The safety limits presented for Oconee Unit 3 have been generated using BAW-2 critical heat flux correlation[1] and the Reactor Coolant System flow rate of 107.6 percent of the design flow (131.32×10^6 lbs/hr for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate.[2]

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

The curve presented in Figure 2.1-1C represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 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 2.1-2C are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing.

1. The 1.30 DNBR limit produced by a nuclear peaking factor of $F_q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 3.

Power peaking is not a directly observable quantity, and, therefore, limits have been established on the bases of the reactor power imbalance produced by the power peaking.

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

The 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.

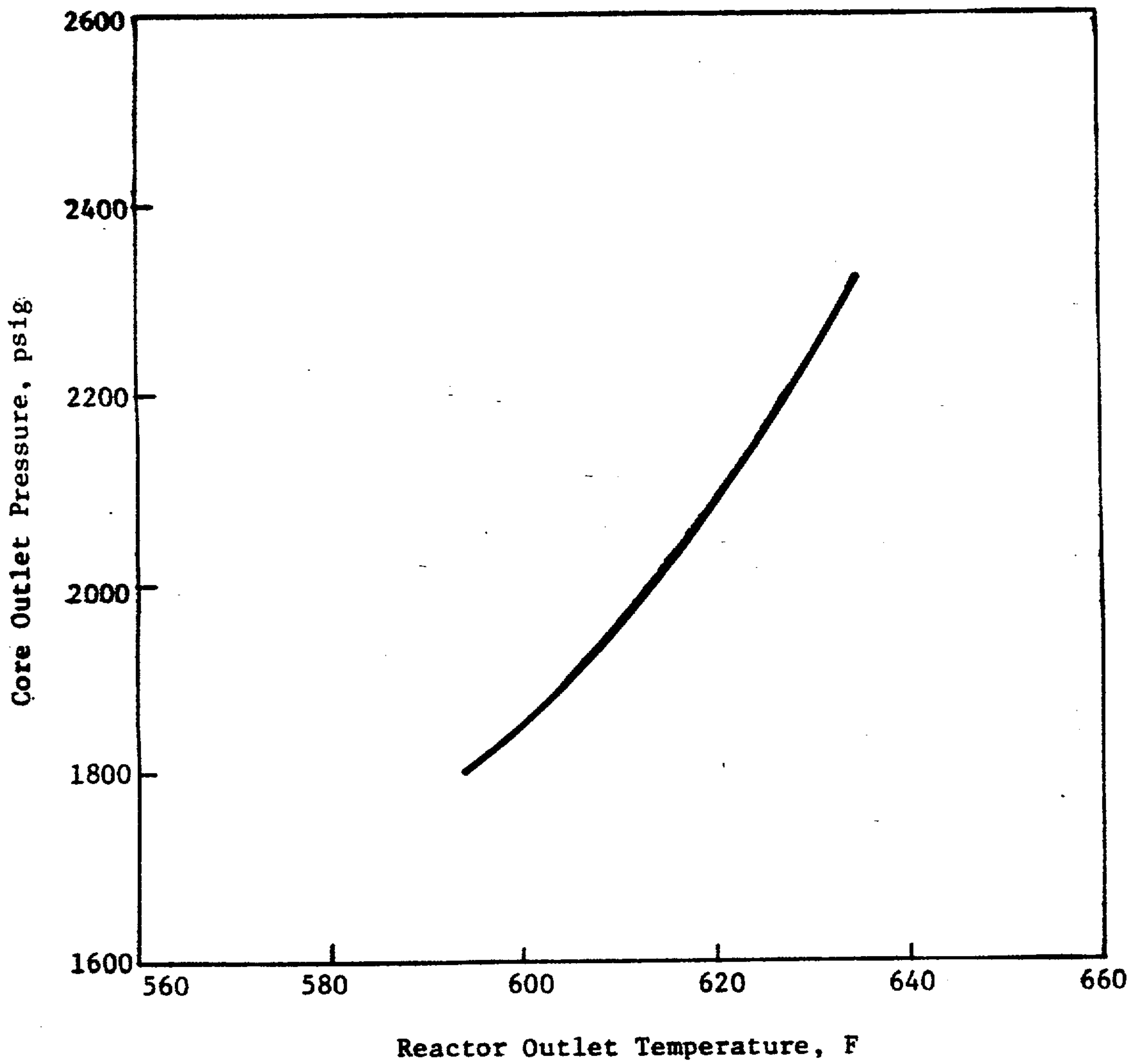
The maximum thermal power for three-pump operation is 86.4 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.9 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3C a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 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.

<u>References</u>

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

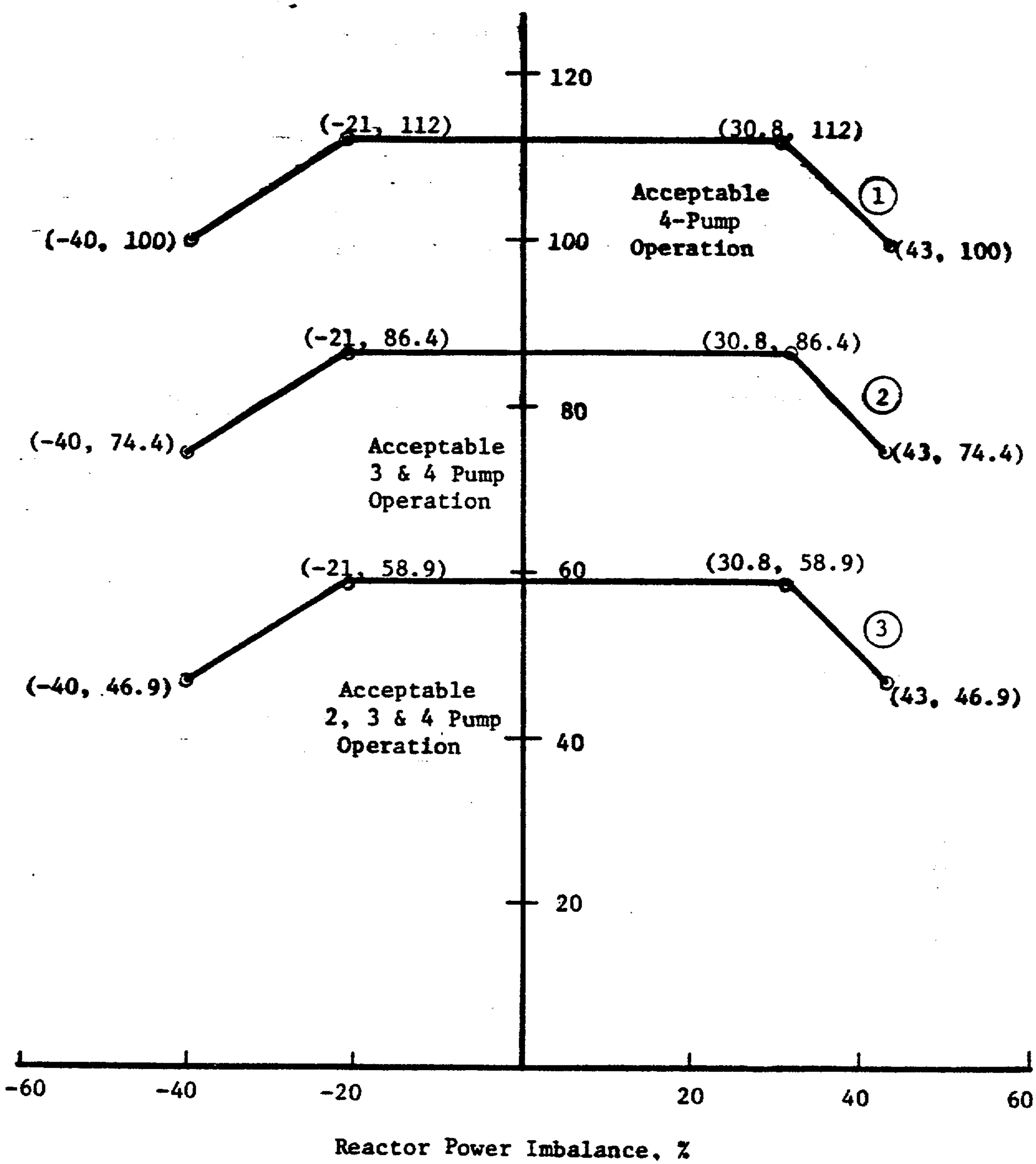
(2) Oconee 3, Cycle 2 - Reload Report - BAW-1432, June 1976.



CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

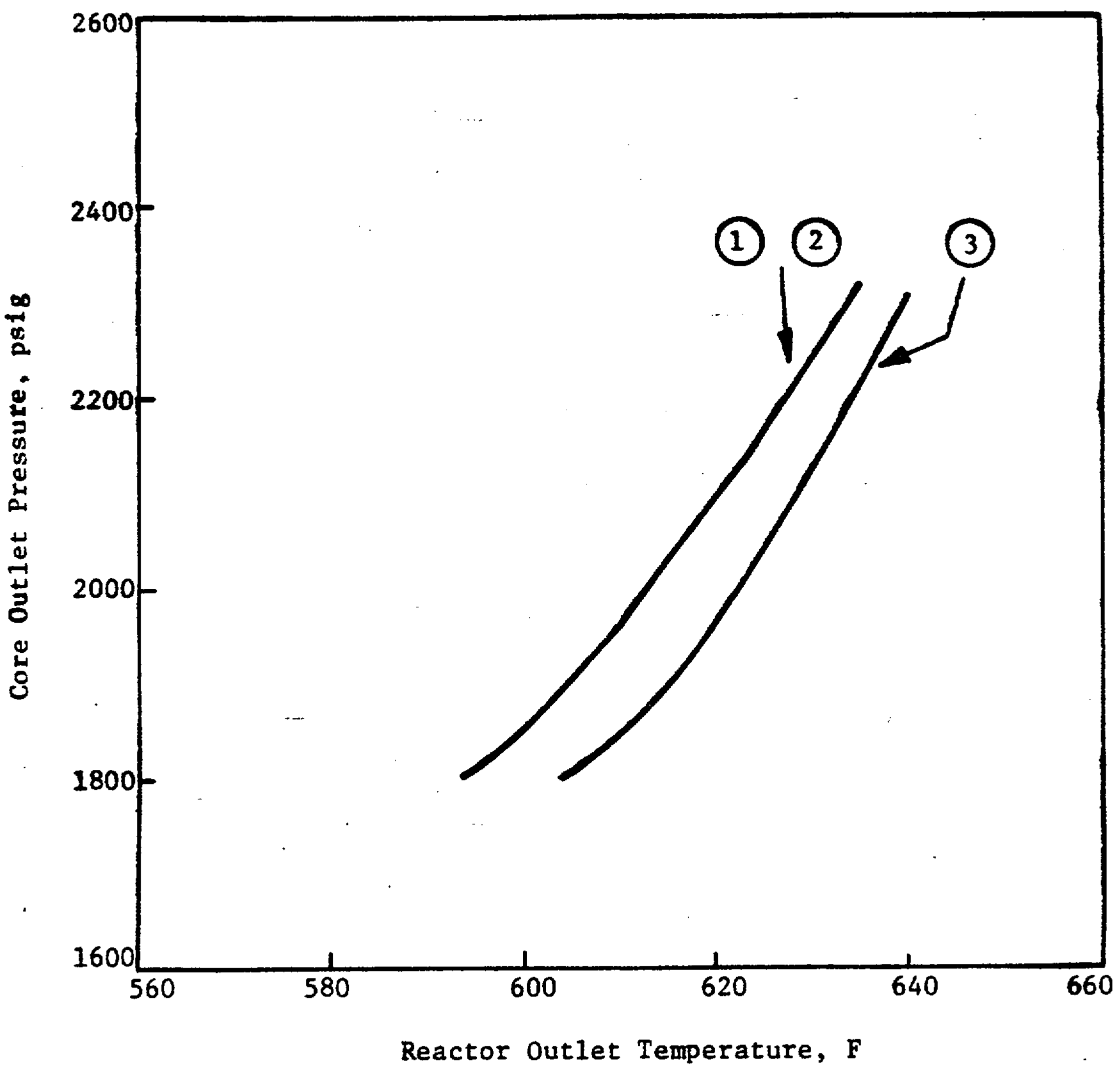
Figure 2.1-1C



Curve	Reactor Coolant Flow (lb/h)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION
Figure 2.1-2C



Curve	Reactor coolant flow (lbs/h)	Power	Pumps operating (type of limit)
1	141.3×10^6 (100%)	112%	Four pumps (DNBR limit)
2	105.6×10^6 (74.7%)	86.4%	Three pumps (DNBR limit)
3	69.3×10^6 (49.0%)	58.9%	One pump in each loop (quality limit)

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION
Figure 2.1-3C

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 account for the maximum calibration and instrumentation errors and 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 over-power trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}-4706$) trip
(1800) psig ($10.79\ T_{out}-4539$)
(1800) psig ($10.79\ T_{out}-4539$)
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)

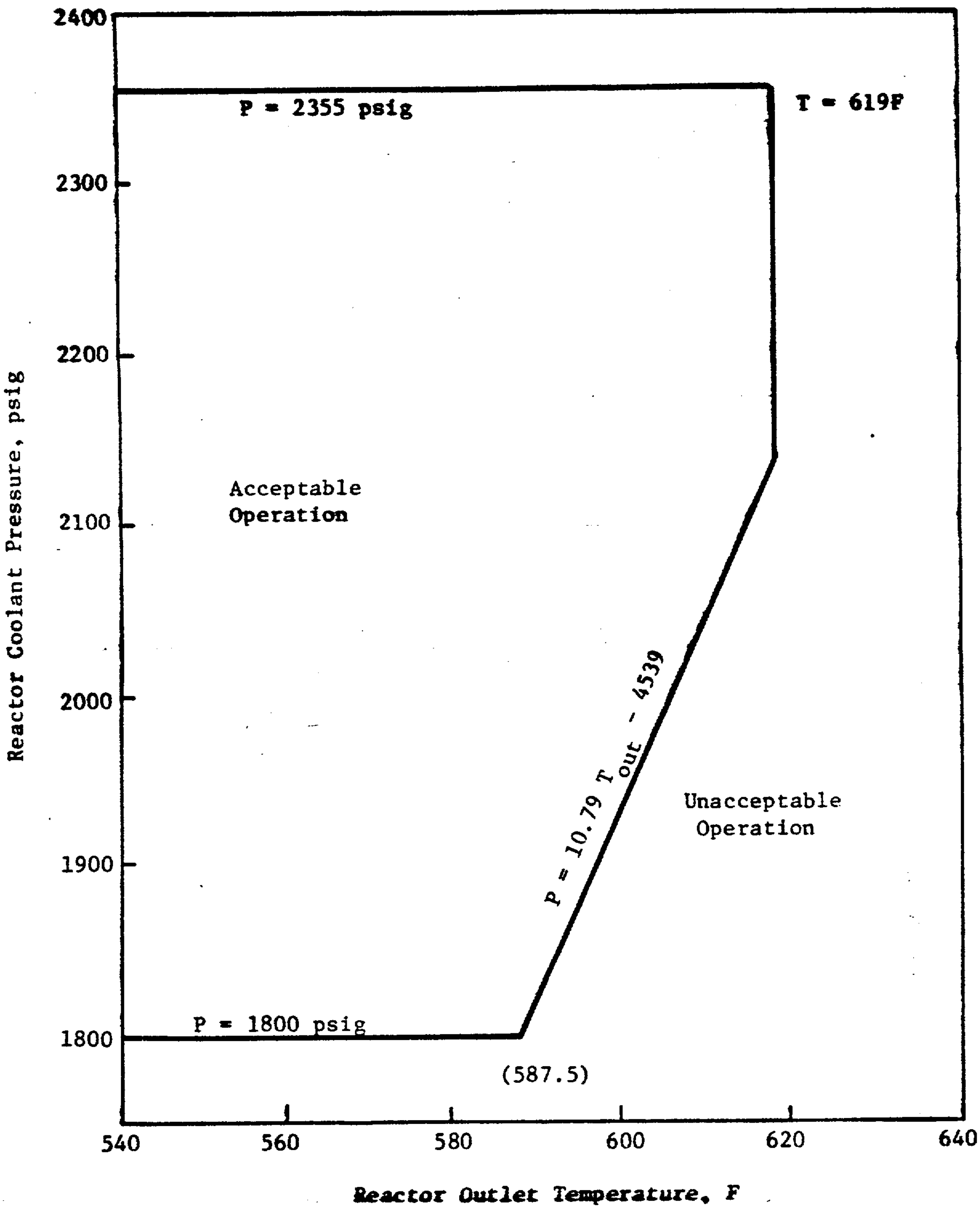
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}\ -4746$)
($10.79\ T_{out}\ -4579$)
($10.79\ T_{out}\ -4579$)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620°F.

Reactor Building Pressure

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

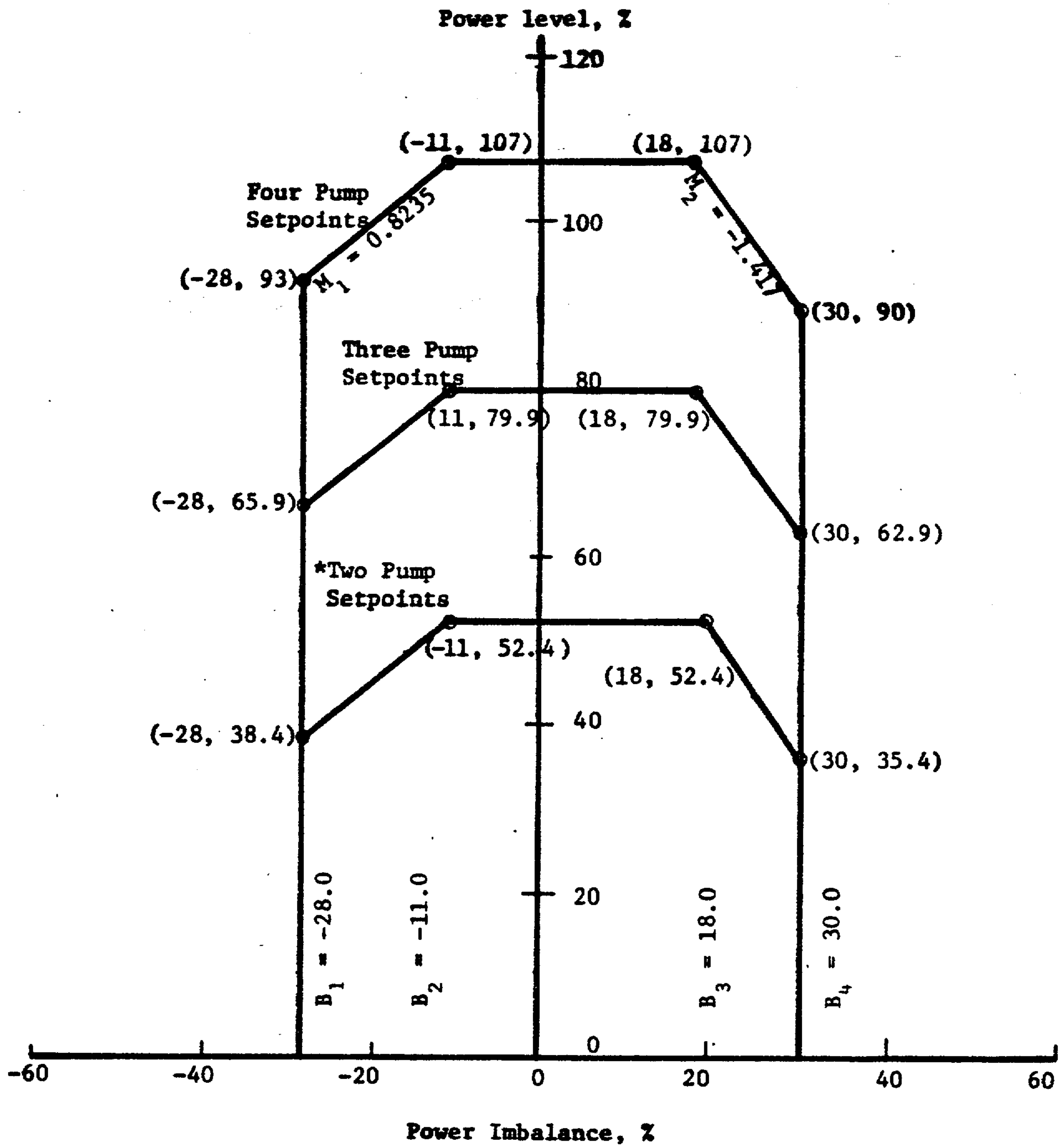


PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-1C

Amendments Nos. 34, 34 & 31



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-2C

Table 2.3-1C

Unit 3

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	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 (^{o}F).

(2) Reactor Coolant System Flow, %.

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

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

(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.

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95 percent of rated power.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/^{o}F$ corrected to 95% rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k/^{o}F$. The moderator coefficient is expected to be zero or negative prior to completion of startup tests.

When the hot zero power value is corrected to obtain the hot full power value, the following corrections will be applied.

A. Uncertainty in isothermal measurement

The measured moderator temperature coefficient will contain uncertainty on the account of the following:

1. $\pm 0.2°F$ in the ΔT of the base and perturbed conditions.

2. Uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4}$ $\Delta k/k$.

Proper corrections will be added for the above conditions to result in a conservative moderator coefficient.

B. Doppler coefficient at hot zero power

During the isothermal moderator coefficient measurement at hot zero power, the fuel temperature will increase by the same amount as the moderator. The measured temperature coefficient must be increased by $0.16 \times 10^{-4} (\Delta k/k)/°F$ to obtain a pure moderator temperature coefficient.

. Moderator temperature change

The hot zero power measurement must be reduced by $.09 \times 10^{-4}$ $(\Delta k/k)/°F$. This corrects for the difference in water temperature at zero power (532°F) and 15% power (580°F) and for the increased fuel temperature effects at 15% power. Above this power, the average moderator temperature remains 580°F. However, the coefficient, α_m, must also be adjusted for the interaction of an average moderator temperature with increased fuel temperatures. This correction is $-.001 \times 10^{-4}$ $\Delta\alpha_m/\Delta\%$ power. It adjusts the 15% power α_m to the moderator coefficient at any power level above 15% power. For example, to correct to 100% power, α_m is adjusted by $(-.001 \times 10^{-4})$ (85%), which is $-.085 \times 10^{-4} \Delta\alpha_m$.

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
3.41% Unit 3
be reduced to less than +3.41% Unit 1 within two hours or the
3.41% Unit 2
3.41% 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
3.41% 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.b, 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
3.41% Unit 3

b. If the quadrant tilt exceeds +3.41% Unit 1 and there is simultaneous
3.41% Unit 2
3.41% 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
9.44% 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% ± 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 and 3.5.2-2B3 (Unit 2), and 3.5.2-2C1, 3.5.2-2C2, and 3.5.2-2C3 (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, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager.

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, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (See Figure 3.5.2-4) 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% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (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% $\Delta k/k$ is allowed by the rod positions limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.5% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (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 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
5.10% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

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

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

Operating restrictions are included in Technical Specification 3.5.2.5d to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond the "undershoot" region and asymptotically approaching its equilibrium value at the power level cutoff.

<u>REFERENCES</u>

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

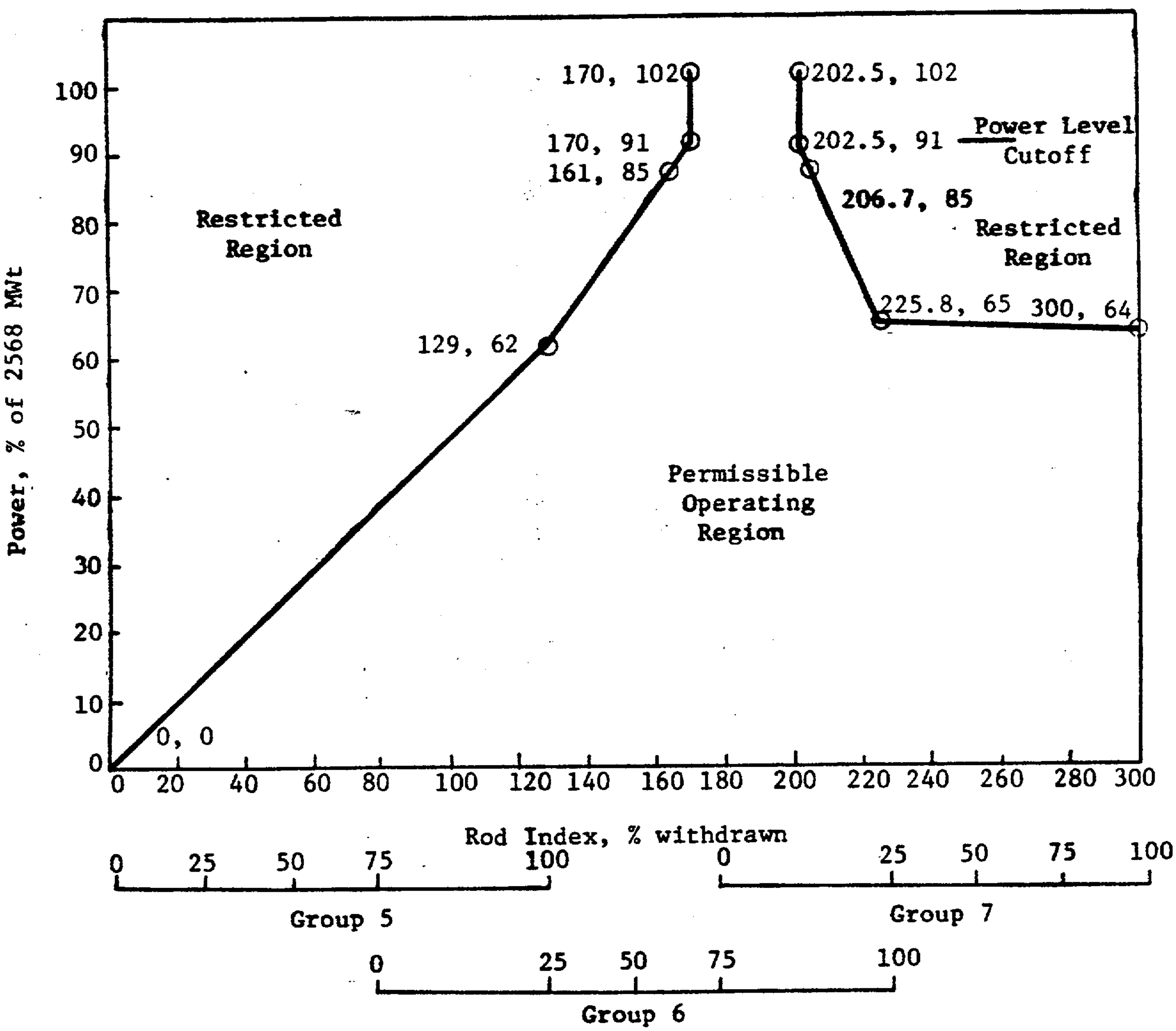
[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

BAW-1400 (UNIT 3)

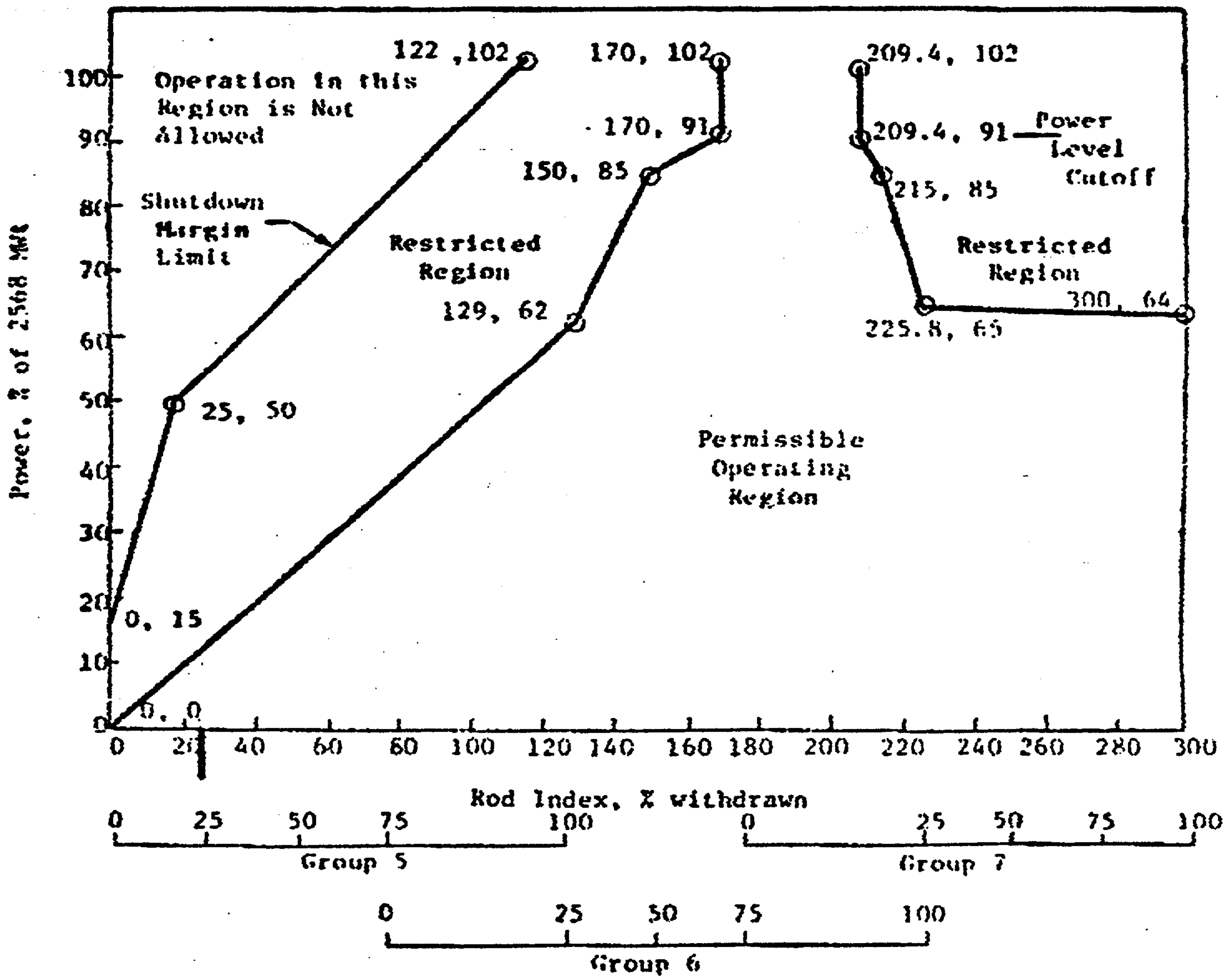


ROD POSITION LIMITS FOR FOUR PUMP OPERATION FROM 0 TO 115 (± 10) EFPD UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-1C1

Amendments Nos. 34, 34 & 31

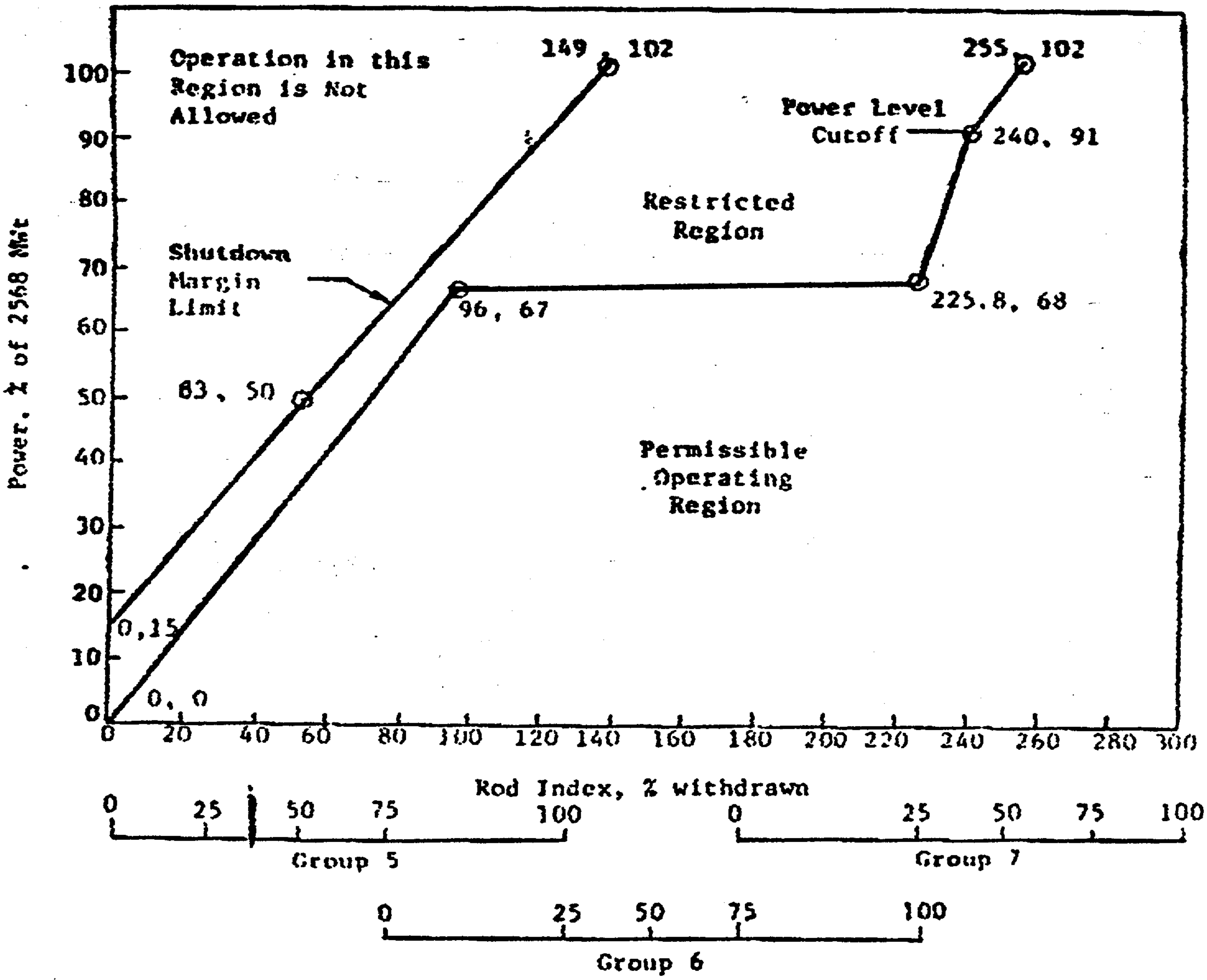


ROD POSITION LIMITS FOR FOUR PUMP OPERATION FROM 115 (± 10) EFPD TO 226 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-1C2

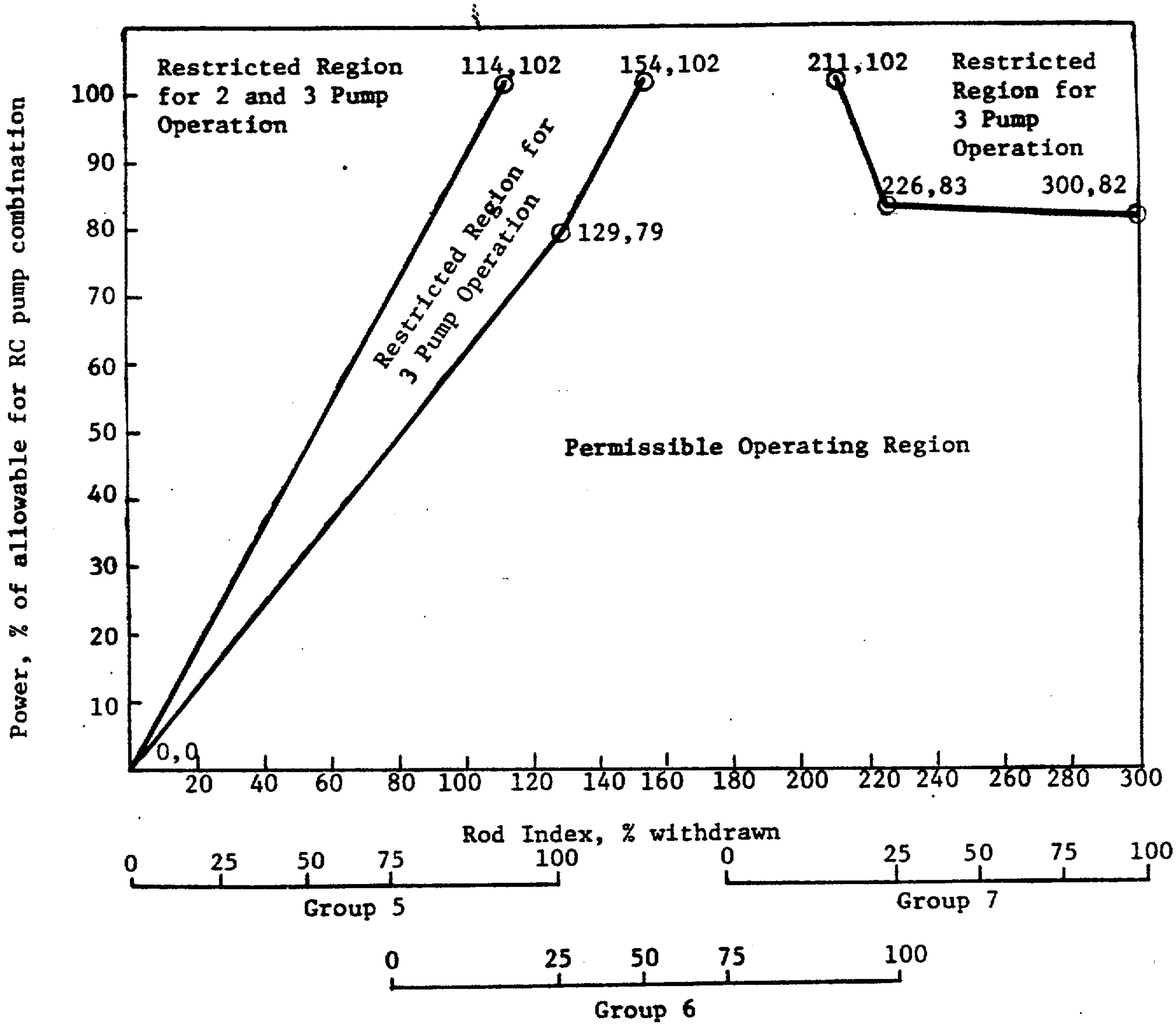
Amendments Nos. 34, 34 & 31



ROD POSITION LIMITS FOR FOUR PUMP OPERATION AFTER 226 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-1C3

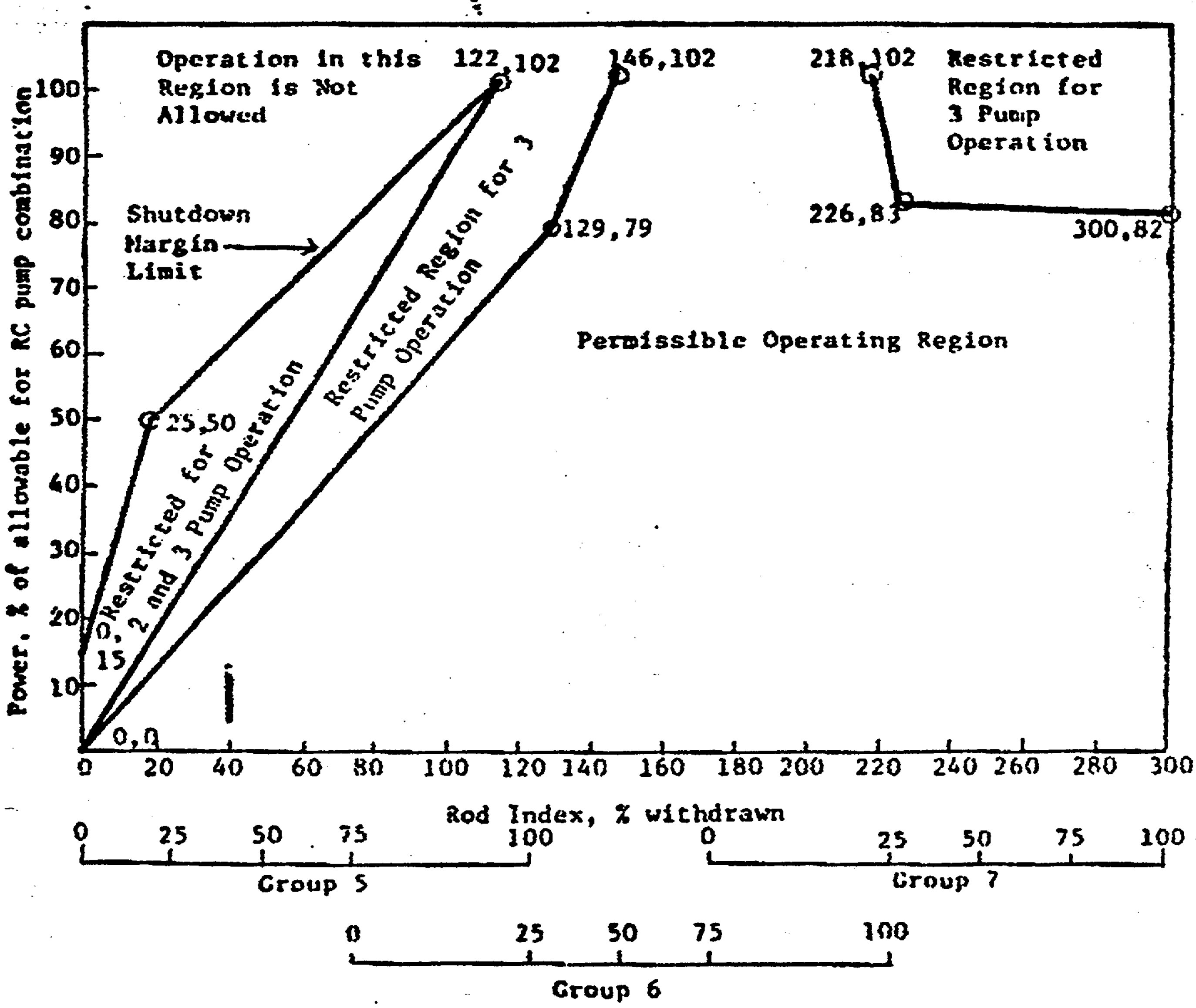
Amendments Nos. 34, 34 & 31



ROD POSITION LIMITS FOR TWO- AND THREE-PUMP OPERATION FROM 0 TO 115 (± 10) EFPD UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-2C1

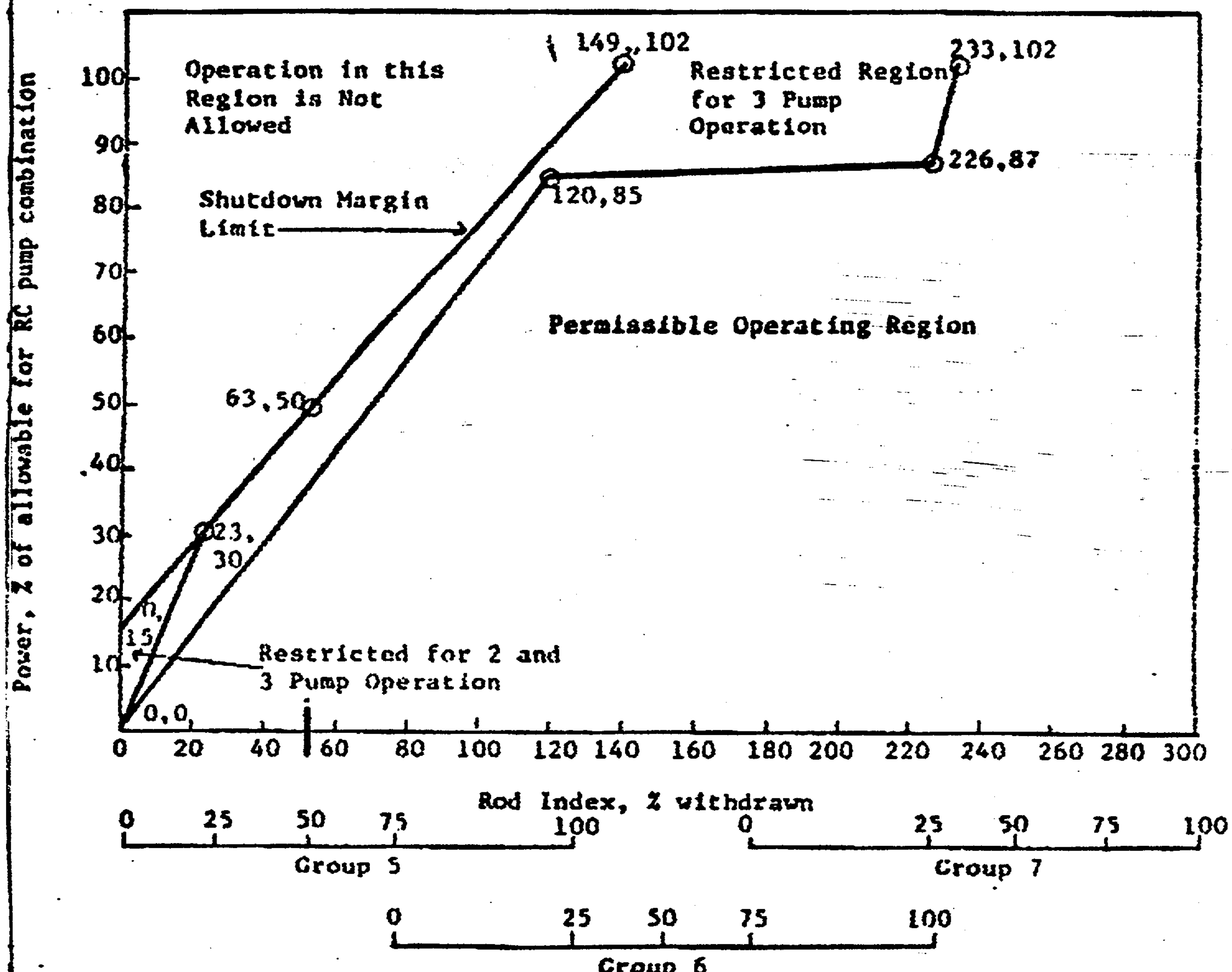


ROD POSITION LIMITS FOR TWO- AND THREE-PUMP OPERATION FR 115 (+ 10) TO 226 (+ 10) EF UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-2C2

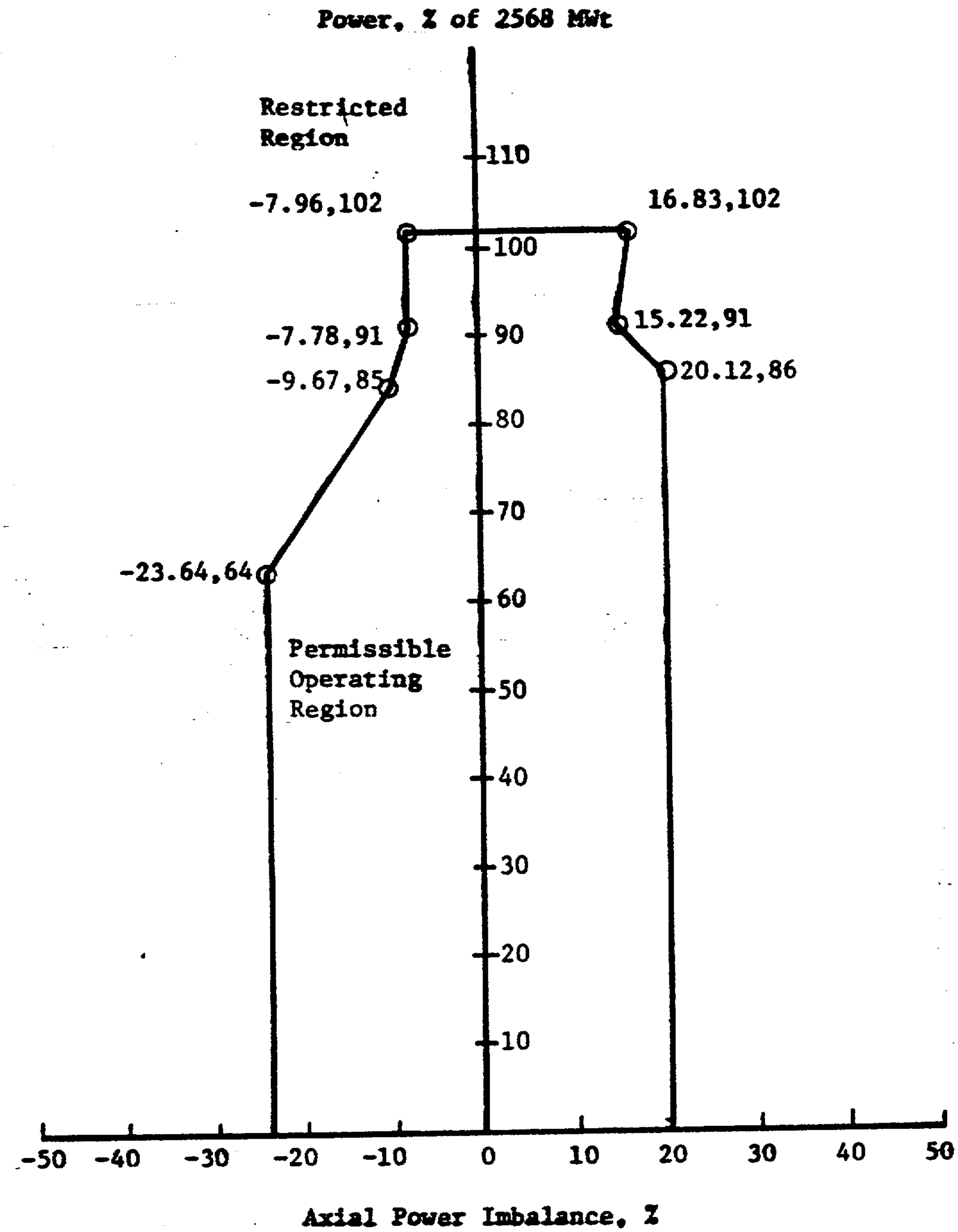
Amendments Nos. 34, 34 & 31



ROD POSITION LIMITS FO
AND THREE-PUMP OPERATI
AFTER 226 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATI

Figure 3.5.2-2C3

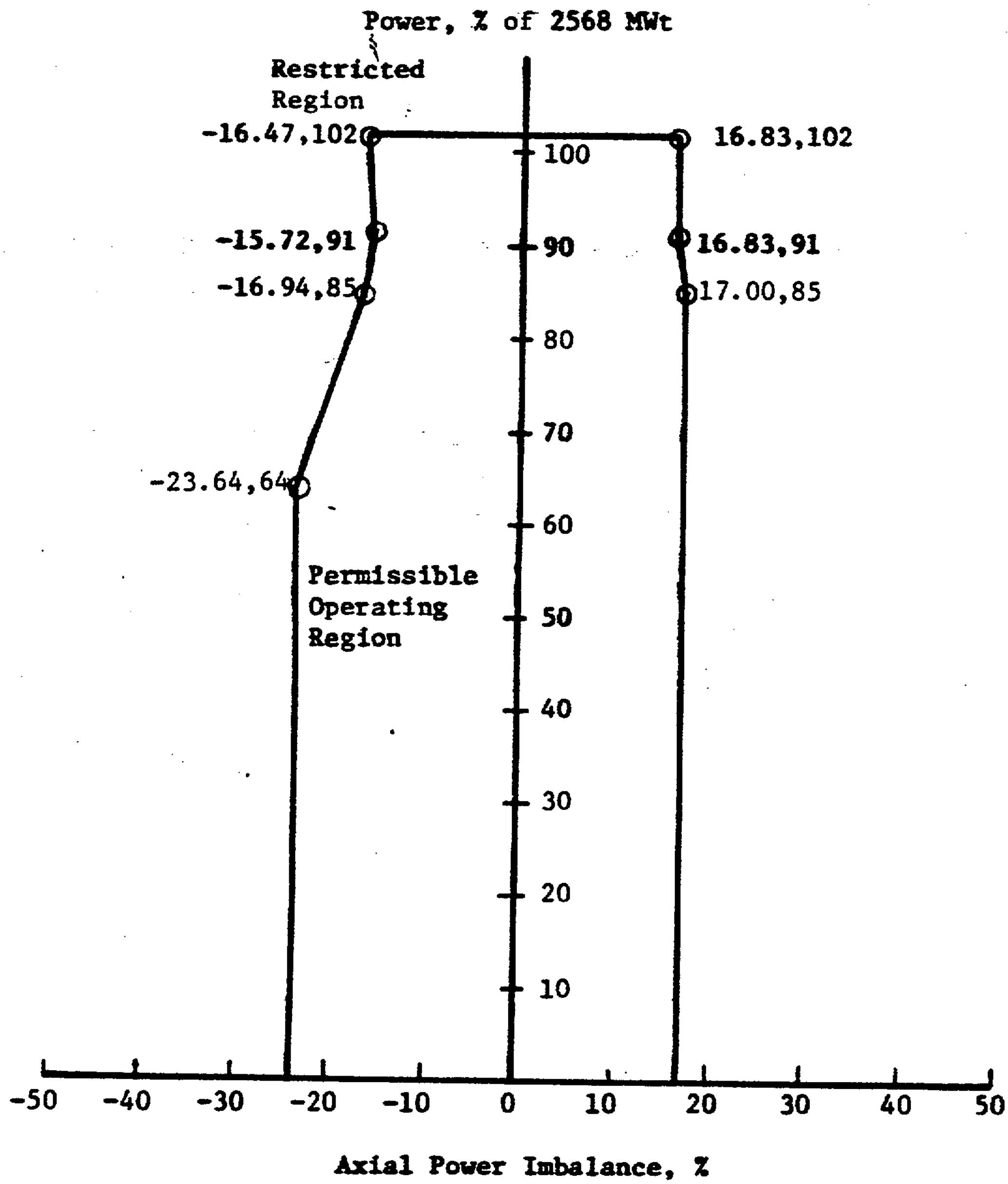


OPERATIONAL POWER IMBALANC
ENVELOPE FOR OPERATION FRC
0 TO 115 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-3C1

Amendments Nos. 34, 34 & 31

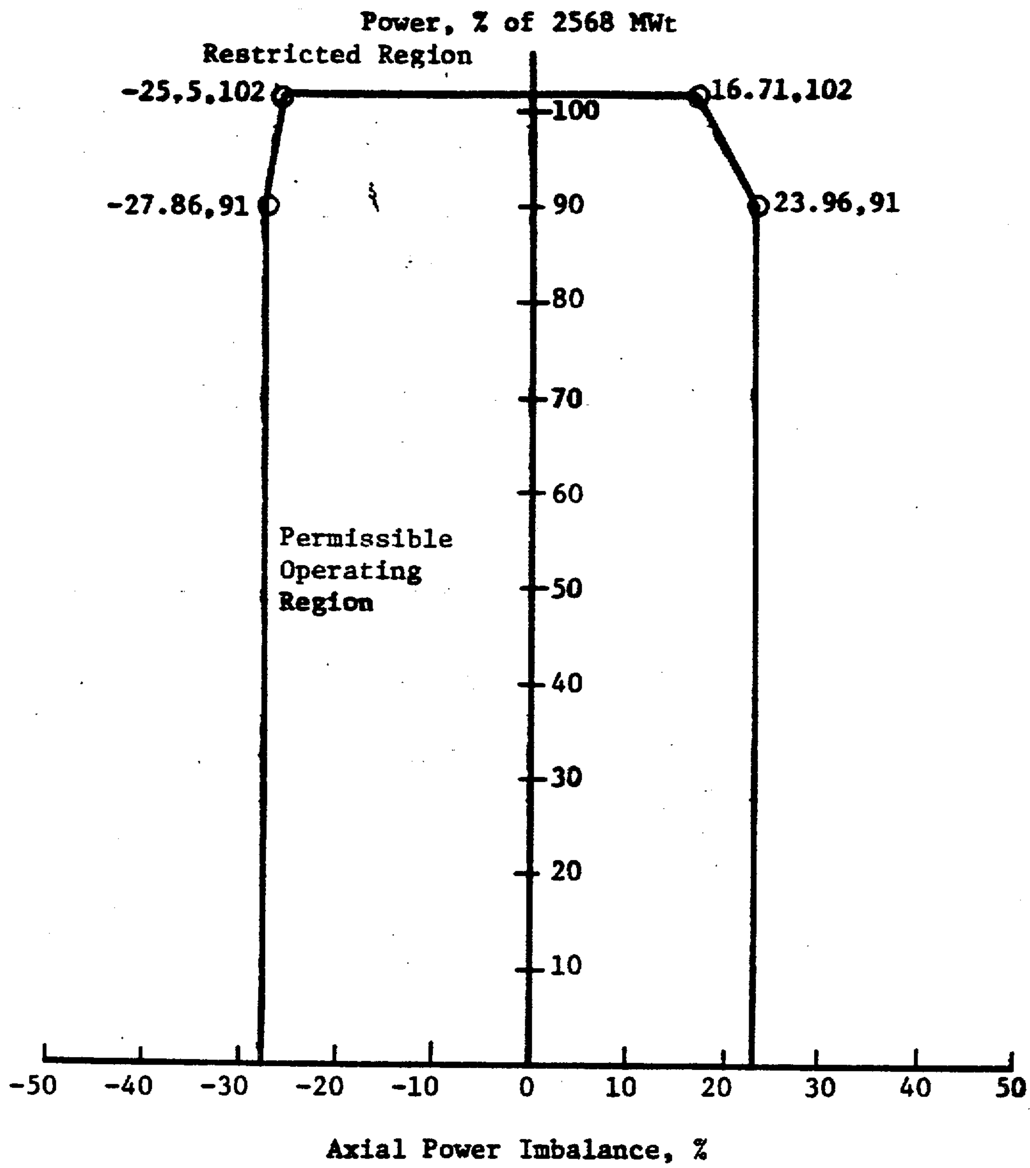


OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
115 (+) EFPD to 226 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

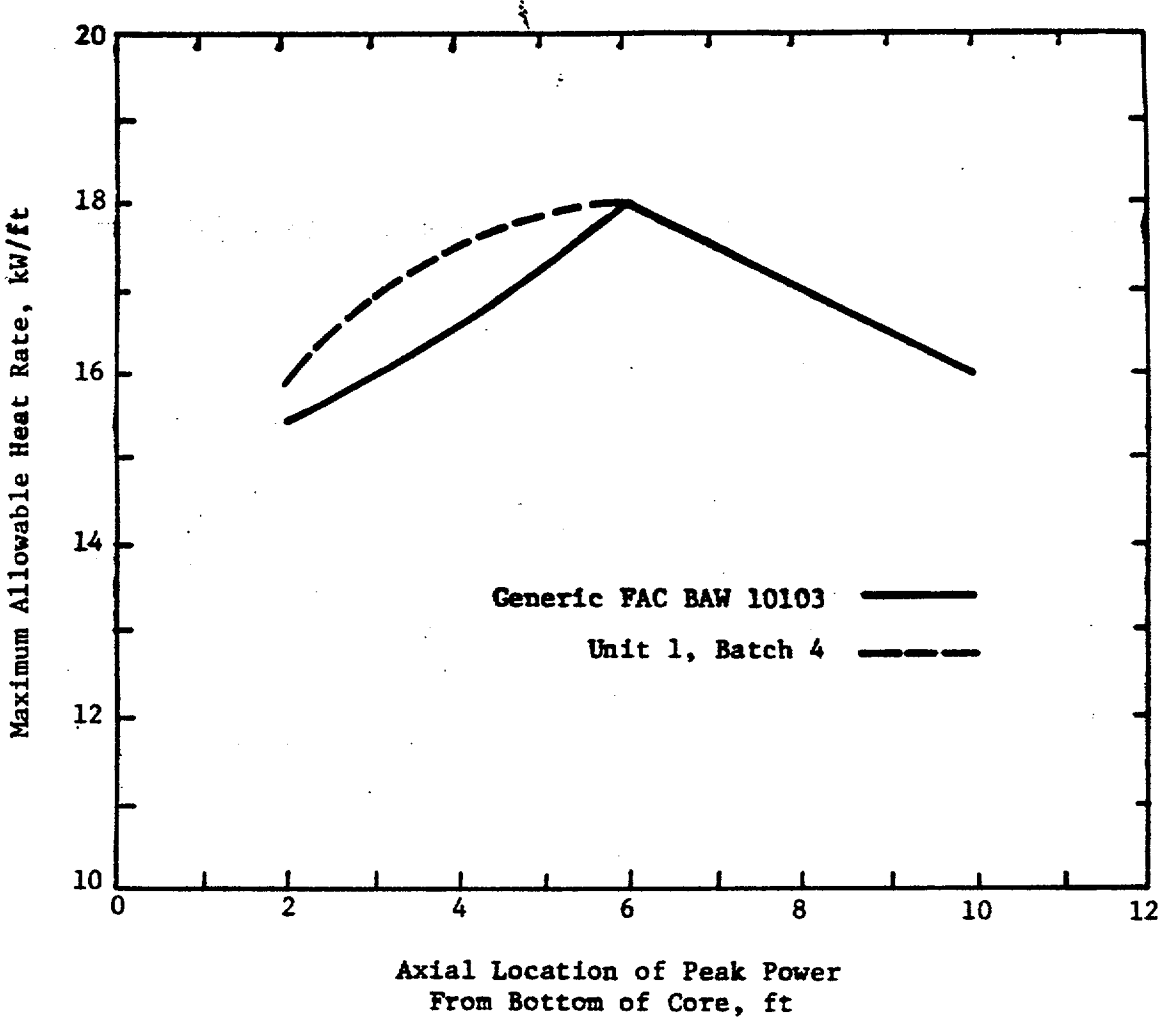
Figure 3.5.2-3C2

Amendments Nos. 34, 34 & 31



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION
AFTER 226 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-3C3



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-4

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 3 141.30×10^6 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 TO FACILITY LICENSE NO. DPR-38

AMENDMENT NO. 34 TO FACILITY LICENSE NO. DPR-47

AMENDMENT NO. 31 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 July 21, 1976, as supplemented August 20, October 7, October 19, October 20, and October 20, 1976, Duke Power Company (the licensee) requested changes to the Oconee Nuclear Station Technical Specifications appended to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for Units Nos. 1, 2, and 3. The proposed changes, which apply only to Unit 3, would permit operation of Unit No. 3 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. Our review of the Unit 3 ECCS single failure criterion was done concurrently with the review of the Unit 2 single failure criterion. Since the two plants are identical in regard to single failure, the evaluation we made for Unit 2 dated June 30, 1976, equally applies to Unit 3. The licensee will adopt the changes in plant Technical Specifications and design hardware identified in the June 30 evaluation for Unit 2 for Unit 3 also.

The Oconee Unit No. 3 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 will occupy primarily the periphery of the core and eight locations in its interior.

The licensee's reload submittal justifies the operation of the second cycle of Oconee Unit 3 at the rated core power of 2568 MWt. The analyses performed take into account the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems. We have concluded that Oconee Unit 3 can be operated safely during Cycle 2 at the rated power level of 2568 MWt. Details of our review are presented in this safety evaluation.

Evaluation

1. Fuel Mechanical Design

All of the Cycle 2 fuel assemblies are identical in concept and are mechanically interchangeable. The assemblies are described in the licensee's reload submittal of July 21, 1976 as supplemented October 20, 1976. The fresh fuel does have minor modifications to the end fittings to reduce assembly pressure drop and increase the holddown margin. The only effect of these modifications is a slight redistribution of core flow which is discussed under thermal-hydraulic design in Paragraph 4 below. Also, four of the assemblies have a slightly higher enrichment and pellet stack length. These four assemblies were substituted for four of the original assemblies after two of the original assemblies were damaged during handling. These four assemblies are described in the licensee's October 20, 1976 letter.

Fuel rod cladding creep collapse analyses were performed for the three fuel batches for the Cycle 2 core. The calculational methods, assumptions, and data have been previously reviewed and approved by the staff. The CROV computer code (BAW-10084 PA) was used to calculate the time to fuel rod cladding collapse. The most restrictive power profiles the new fuel assemblies may be exposed to were used in the analyses. Conservative values were used for the cladding thickness and ovality and no credit was taken for fission gas release which yields conservative net differential pressures. Also, batches 2 and 3 cladding temperatures were calculated using outlet temperature which is also conservative. 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, Batches 2, 3, and 4 are identical.

The Batch 4 fuel assemblies are not new in concept and previously approved methods of analysis were used to analyze the mechanical performance of the fuel. Also, this design was used in Oconee 2, Cycle 2, which we approved on June 30, 1976. Based on our review, we conclude that the fuel design is acceptable.

2. Thermal Design

The fuel thermal design analysis was performed using the TAFY-3 computer code, as described in "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," BAW-10044, May 1972.

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) a code option for no restructuring of the fuel.

(2) calculated gap conductance was reduced by 25%.

Using the TAFY code, the damage threshold of the fuel has been shown to be 20.15 kw/ft for the 56 fuel assemblies, which is substantially above any value expected during normal operation, anticipated operating transients, or a LOCA.

Based on our review, we conclude that the fuel thermal design for Cycle 2 is acceptable.

3. Nuclear Design analysis

The reactor core physics parameters for Cycle 2 operation were calculated using the PDQ07 computer code which has been previously approved by us for use. Since the core has not yet reached an equilibrium cycle, the minor differences in the physics parameters which exist between the Cycle 1 and Cycle 2 cores are to be expected and are not significant.

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 design analysis for Cycle 2 to be acceptable.

4. Thermal-Hydraulic Analysis

The Mark B4 (Batch 4) assembly differs from the Mark B3 (Batch 3) assembly primarily in the design of the end fitting. This produces a slightly smaller flow resistance for the B4 assemblies. Introducing B4 assemblies into the core causes a slight change in the core flow distribution, which we conclude to be a negligible effect. To obtain the Cycle 2 core flow distribution, the thermal-hydraulic model utilized the actual 56 B4, 121 B3 configuration with B3 assemblies in the hottest core locations.

Reactor coolant flow was measured during Cycle 1 operation. The measured flow was 110% of the design flow. For the Cycle 2 thermal-hydraulic design analysis, system flow was assumed to be 107.6% of design which is consistent with Units 1 and 2. This value is acceptable as it includes adequate conservatisms representing uncertainties in the measurement of flow. Incorporation of this increased flow in the thermal-hydraulic calculations was accompanied by a corresponding increase in the core inlet temperature from 554 to 555.9F. The increases in RC flow and inlet temperature are changes in calculational parameters only and do not represent changes in operation of the plant. The Cycle 2 analysis indicates that the margin to DNB is greater for Cycle 2 than had been predicted for Cycle 1 operation.

The DNBR analysis for Cycle 2 operation considered maximum design conditions, as-built fuel assembly geometry, and hot operating conditions. This analysis resulted in the hot channel (Batch 3 fuel) minimum DNBR of 1.98 of 112% power for undensified fuel. The DNBR calculations for undensified fuel are based on a 144-inch active length.

The shortened stack length used in a second analysis for densified fuel was 141.12 inches. Although this is longer than the densified stack length of the Batch 3 fuel (140.30 inches) the gap size and power spike magnitude were large enough to give conservative results. The densification effect results in a 5.93% reduction in the minimum DNBR. The minimum DNBR for Cycle 2, considering this effect, is still greater than for Cycle 1.

Rod Bow

An analysis was performed with the COBRA III-C code to determine the effect of a fuel rod bowing into the hot channel and reducing its flow area. The results indicate that rod bow of the magnitude predicted is adequately compensated for by the flow area reduction factor. Rod bow away from the hot channel was also analyzed. In this analysis the effect of a power spike was added to the hot rod in the area of the minimum DNBR. This analysis indicates that Cycle 2 DNBR results account for the effects of fuel rod bowing.

Core Vent Valve

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 surveillance requirements.

Our letter dated June 30, 1976, issued the license amendments applying these surveillance requirements to all units. By letter dated August 20, 1976, the licensee requested that the requirement for a flow penalty be removed for Unit 3. Since the June 30, 1976 amendments provided for the necessary surveillance, we find the licensee's request to remove this flow penalty to be acceptable.

<u>Critical Heat Flux Correlation</u> (CHF)

The W-3 CHF correlation was used for the Unit 3 Cycle 1 core. The BAW-2 correlation has been reviewed and approved for use with the Mark B fuel assembly design. In the application to the Oconee 3, Cycle 2 core, two modifications, which have also been applied to the Oconee 1, Cycle 3, and Oconee 2, Cycle 2 cores, have been instituted.

1. The pressure range applicable to the correlation has been extended downward from 2000 to 1750 psia.

2. The limiting design DNBR of 1.30 was used. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur.

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. 3 analysis. In addition, we recently completed a reevaluation 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, massflux, rod diameter and rod spacing range of its original data base.

In summary the licensee has proposed a reactor coolant flow rate consistent with Units 1 and 2 for the Unit 3, Cycle 2 thermal-hydraulic analysis. The licensee has also requested elimination of a 4.6% vent valve flow 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 that the thermal-hydraulic analysis is acceptable and that the Technical Specifications related to the Cycle 2 thermal-hydraulic analysis, as proposed in the July 21, 1976 submittal, are also acceptable.

5. Accident and Transient Analysis

Each FSAR accident and transient analysis was reviewed. In all cases the important parameters are bounded by FSAR assumed parameters or the results are conservative with respect to the FSAR and reference cycle analyses. Therefore, we conclude that the accident and transient analyses are adequate.

6. Startup Program

The startup program tests will verify that the core performance is within the assumption of the safety analysis and will provide the necessary data for continued plant operation. The licensee has agreed by letter dated October 20, 1976, to provide certain confirmatory information from the startup program. We find this to be acceptable.

7. ECCS

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 of other license amendments as may be necessary to implement the evaluation results. As required by the Order, the licensee, by letter dated July 9, 1975 as supplemented August 1, 1975, submitted an ECCS reevaluation and related Technical Specifications. In the reload application of July 21, 1976, the licensee has submitted the related Technical Specifications using the B&W ECCS evaluation model as described in BAW-10104 of May 1975.

The background of our review of the B&W ECCS evaluation model and its application to Oconee is described in our Safety Evaluation Report 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 our Status Report of October 1974 and the Supplement to the Status Report of November 1974 which are referenced in the December 27, 1974 SER. That SER describes the various changes required in the earlier version of the B&W model. Together, that SER, the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for our acceptance of the model. The Oconee 3 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 3; BAW-10104, May 1975 and BAW-10103, June 1975 (Revised April 1976), 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 3:

Elevation (ft)	LOCA Limit (kw/ft)	Peak Cladding Temperature (°F)		Max. Local Oxidation (%)	Time of Rupture (sec)
		Ruptured Node	Unruptured Node		
Oconee 3					
2	15.5	2002	1978	3.92	12.25
4	16.6	2136	2072	4.59	13.01
6	18.0	2066	2146	5.46	14.55
8	17.0	1742	2110	5.19	14.01
10*	16.0	1642	1931	2.93	39.20

*See discussion below.

The maximum core-wide metal-water reaction for Oconee 3 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.

We noted during our review of BAW-10103 that the LOCA limit calculation at the 10-foot elevation in the core showed reflood rates below 1 inch/second, 251 seconds into the accident (Section 7.3.5). 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 us 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 our review. The steam cooling model submitted by B&W in BAW-10103 is therefore considered to be a proposed model change requiring our further 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 us. Until a steam cooling model has been accepted by us, 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 3 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 3, 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 3, Cycle 1 operation meet the necessary FAC and are acceptable. Since Oconee Unit 3 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 3 analyses addressed the areas of single failure criterion long-term boron concentration, potential submerged equipment, partial loop operation, emergency electrical power 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.

Our review of the Unit 3 ECCS single failure criterion was done concurrently with the review of the Unit 2 single failure criterion. Since the two plants are identical in regard to single failure, the evaluation we made for Unit 2, dated June 30, 1976, equally applies to Unit 3.

One of our requirements in the Unit 2 safety evaluation was that valves LP-21 and LP-22 would be left in the open position during normal operation to minimize the potential for a water hammer due to the discharge of ECC water into a dry line. By letter dated August 20, 1976, the licensee committed to this procedure for Unit 3 also.

Based on our review of the single failure criterion, we conclude that the criterion has been met and is therefore acceptable.

Emergency Electric Power

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 step-down 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.

Our evaluation of the Unit 2 emergency electric power system dated June 30, 1976, applies to the Unit 3 as well. We have concluded that the design of the electric power system is such that a single failure of any single electric component would not preclude the ECCS of either Units 2 or 3 from performing its function. Our conclusion was based in part, on the seismic qualification of the Keowee Overhead Electric Power Source, which the licensee had advised us was seismically designed to withstand the .15g earthquake referred to in the Oconee FSAR. The licensee had committed to provide us with confirmatory information prior to the startup of Unit 3.

The licensee, by letter dated October 7, 1976, stated that although the analyses are being completed as expeditiously as possible, the complexity, diversity, and vintage of the equipment has precluded completion of the tasks in the short period of time which has transpired. The licensee has provided a schedule which shows completion of the tasks involved by March 1, 1977.

We conclude that since the confirmatory information is forthcoming on a reasonable schedule and a seismic event at Oconee is an extremely low probability, that it is acceptable for Unit 3 to operate pending our review of this confirmatory information.

Submerged Electrical Equipment

The Unit 3 review and evaluation are identical to that performed for Unit 2. Our Safety Evaluation issued on June 30, 1976, applies to Unit 3, also, and is 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

Our Safety Evaluation dated June 30, 1976, is applicable to Unit 3 also. 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 supplemented of November 1974.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Oconee 3 is conservative and, therefore, the calculated containment pressure 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 3 which would allow adequate boron dilution during the long-term and which will comply with the single failure criterion.

As indicated in our June 30, 1976 Safety Evaluation and our letter dated October 4, 1976, we concluded that the proposed procedures and modifications are acceptable for preventing long-term boron concentration provided that some type of flow indication is provided on the hot leg drain lines. We indicated that the next refueling cycle would be acceptable for installation on Unit 3 since we required testing of the hot leg drain system prior to cycle 2 startup. The licensee has committed to this by letter dated October 19, 1976. We find this to be acceptable.

Partial Loop Analysis

Our Safety Evaluation dated June 30, 1976, evaluated the operating mode of one idle reactor coolant pump and showed that this mode is supported by a LOCA analysis performed in accordance with Appendix K of 10 CFR 50.

An analysis of ECCS cooling performance with one idle reactor coolant pump in each loop was not submitted and power operation in this configuration was limited by Technical Specifications to 24 hours.

The June 30, 1976 evaluation is applicable to Unit 3 and we conclude that this mode of operating is acceptable as indicated above.

We have completed the review of the Oconee 3 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 3 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 October 19, 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 3 Cycle 2 reload application and conclude that the licensee has performed the required analyses and has shown that operation of the Cycle 3 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 amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

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.

Date: October 22, 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. 34, 34 and 31 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company which revised the licenses for operation of the Oconee Nuclear Station 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 3 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 3 by the Commission's December 27, 1974 Order for Modification of License.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendment to Facility Operating License No. DPR-55 in connection with this action was published in the FEDERAL REGISTER on September 16, 1976 (41 FR 39848). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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

For further details with respect to this action, see (1) the application for amendments dated July 21, 1976, as supplemented August 20, October 7, October 19, October 20, and October 20, 1976, (2) Amendments Nos. 34,34 and 31 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. 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 22nd day of October 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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