March 26, 1976

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

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Gentlemen:

The Commission has issued the enclosed Amendments No. 20, 20, and 17 for Licenses No. DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Technical Specifications and are in response to your request dated December 1, 1975, as supplemented February 24 and 27, 1976.

These amendments (1) revise the Technical Specifications to establish operating limits for Unit 1 cycle 3 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR 50.46, (2) terminate the operating restrictions imposed on Unit 1 by the Commission's December 27, 1974 Order for Modification of License and (3) revise the Technical Specifications to specify quadrant power tilt limits for Units 1, 2 and 3 independent of the measurement system used.

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

Sincerely,

/S/ Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 20 to DPR-38
2. Amendment No. 20 to DPR-47
3. Amendment No. 17 to DPR-55
4. Safety Evaluation
5. Federal Register Notice

SS:RSB	DOR:RSB	SS:AB	SS:CPB
TNovak	BBaer	ZRostoczy	PCheck
3/ /76	3/ /76	3/ /76	3/ /76

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SURNAME	SSheppard	GZech:esp		RAPurple	KRGoller	
DATE	3/ /76	3/ /76	3/ /76	3/ /76	3/ /76	

cc w/enclosures:
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Walhalla, South Carolina 29621

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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. 20
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 December 1, 1975, as supplemented February 24, and 27, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change 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: March 25, 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. 20
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 December 1, 1975, as supplemented February 24, and 27, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change 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

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 25, 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. 17
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 December 1, 1975, as supplemented February 24, and 27, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change 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

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 25, 1976

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 20 TO FACILITY LICENSE NO. DPR-38

AMENDMENT NO. 20 TO FACILITY LICENSE NO. DPR-47

AMENDMENT NO. 17 TO FACILITY LICENSE NO. DPR-55

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

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-4	2.1-4
2.1-7	2.1-7
2.1-10	2.1-10
2.3-1	2.3-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-4	2.3-4
2.3-5	2.3-5
2.3-8	2.3-8
2.3-11	2.3-11
3.1-1	3.1-1
3.1-2	3.1-2
3.1-19	3.1-19
3.1-20	3.1-20
3.3-2	3.3-2
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
3.5-12	3.5-12
3.5-13	3.5-13
3.5-18	3.5-18
--	3.5-18a
3.5-21	3.5-21
--	3.5-21a
3.5-24	3.5-24
4.1-9	4.1-9

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A-Unit 1. If the actual pressure/temperature point is below
2.1-1B-Unit 2
2.1-1C-Unit 3
and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A-Unit 1. If the actual reactor-thermal-power/power
2.1-2B-Unit 2
2.1-2C-Unit 3
imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases - Unit 1

The safety limits presented for Oconee Unit 1 have been generated using BAW-2 critical heat flux (CHF) correlation(1) and the actual measured flow rate at Oconee Unit 1 (2). This development is discussed in the Oconee 1, Cycle 3 Reload Report, reference (2). The flow rate utilized is 107.6 percent of the design flow (131.32×10^6 lbs/hr) based on four-pump operation.(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 setponts to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A 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 107.6 percent of 131.3 x 10^6 lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

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

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

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 2.1-2A 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 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

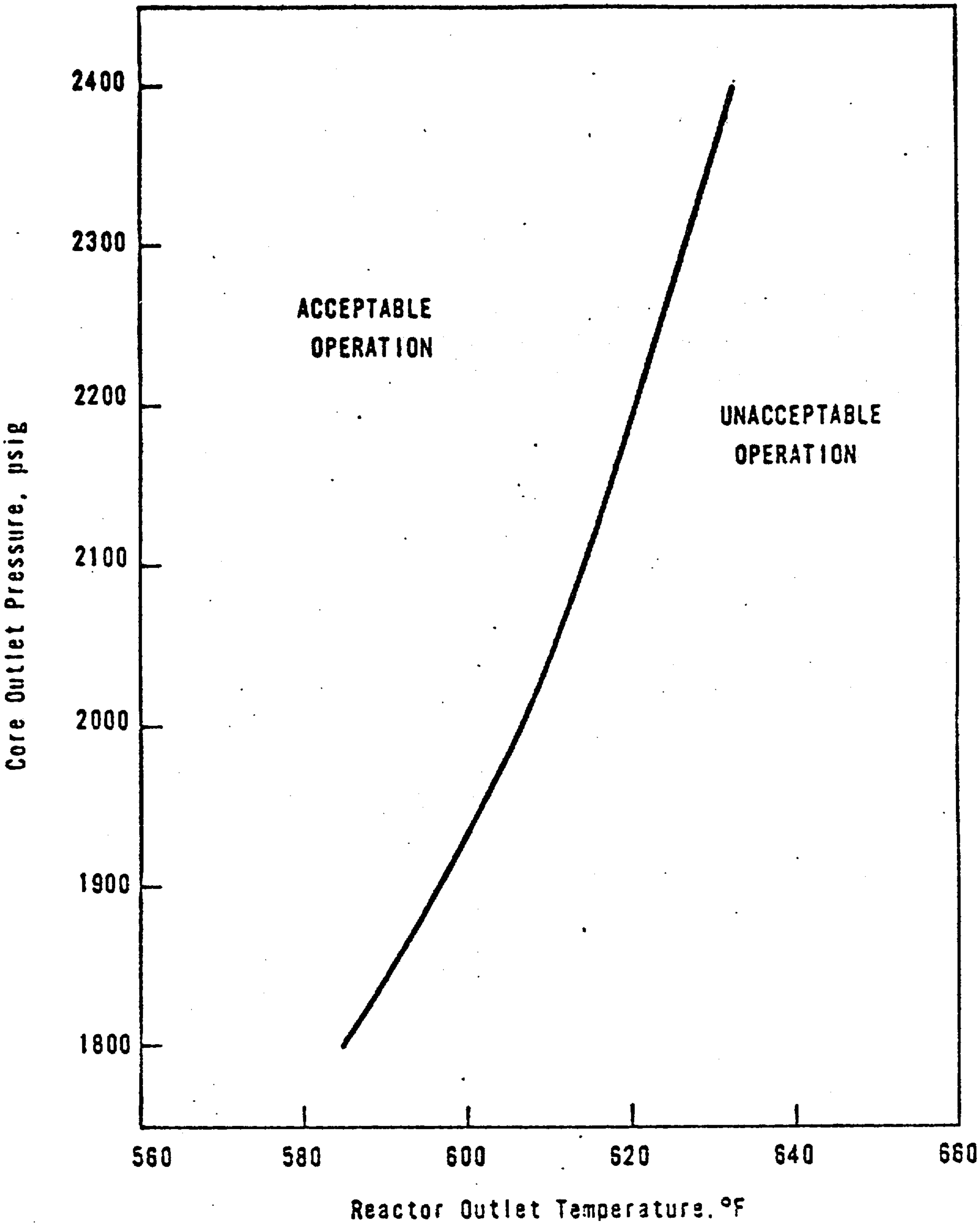
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055 = 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a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

For Figure 2.1-3A, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30. 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>

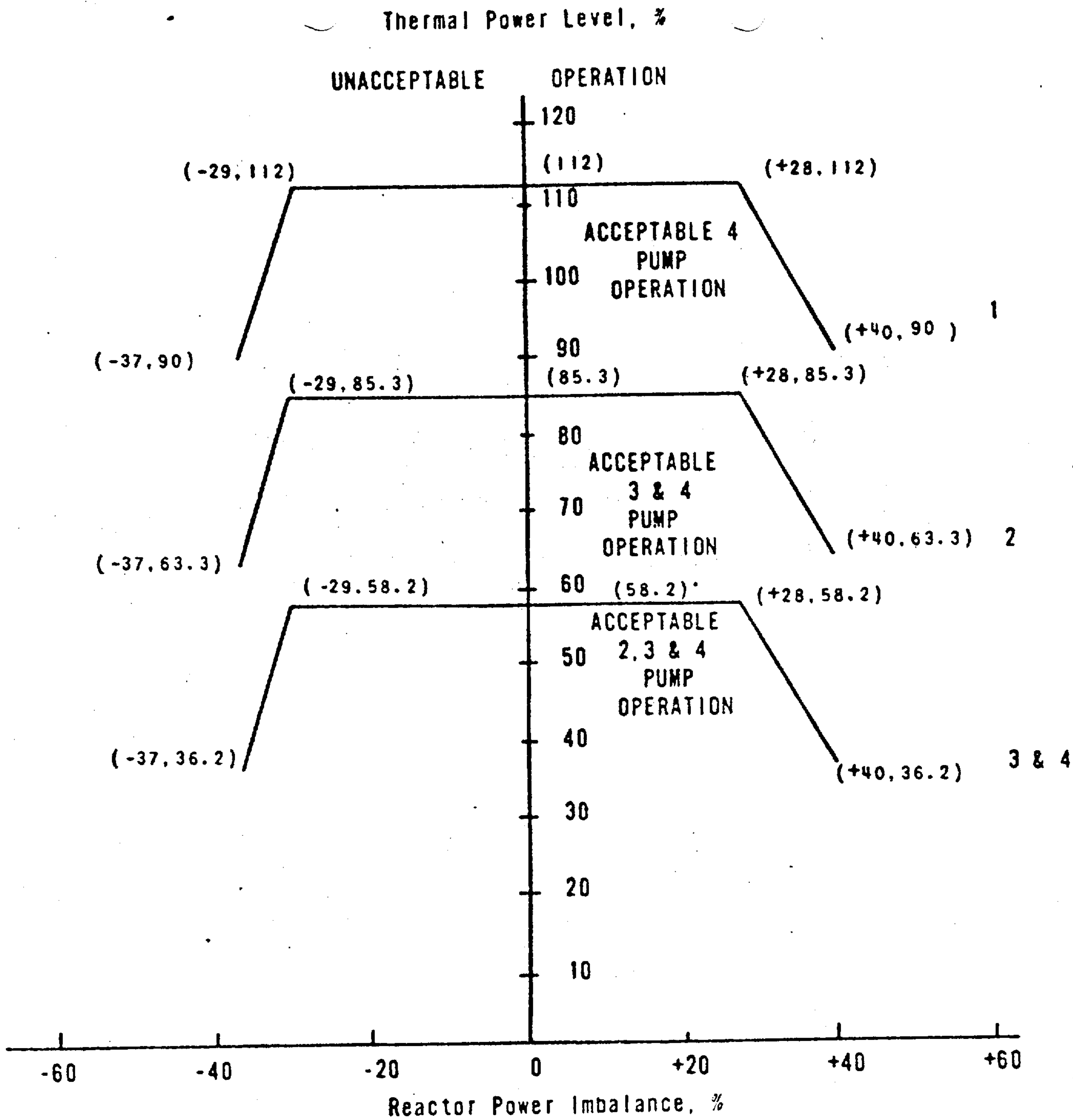
(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March, 1970.

(2) Oconee 1, Cycle 3 - Reload Report - BAW-1427, December 1975.



UNIT 1
CORE PROTECTION SAFETY LIMITS
OCONEE NUCLEAR STATION
FIGURE 2.1-1A

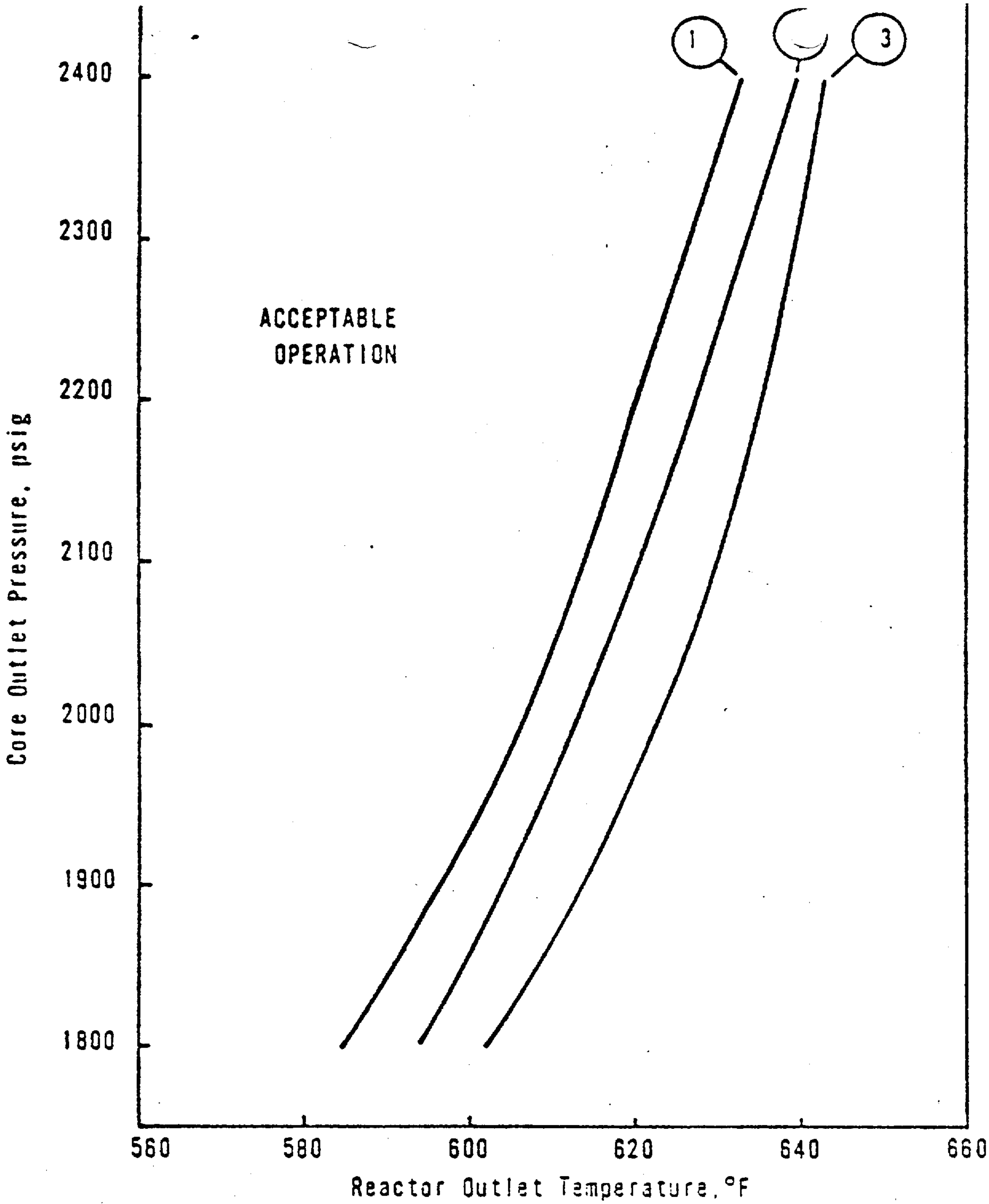
Amendment No. 20, 20 & 17
March 25, 1976



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	141.3×10^6
2	105.6×10^6
3	69.3×10^6
4	64.7×10^6

*THE FLUX/FLOW SETPOINT FOR 2/0
PUMP OPERATION MUST BE SET AT 0.949

UNIT 1
CORE PROTECTION SAFETY LIMIT
OCONEE NUCLEAR STATION
FIGURE 2.1-2A



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	141.3×10^6 DNBR Limit
2	105.6×10^6 DNBR Limit
3	69.3×10^6 Quality Limit

UNIT 1
OCONEE NUCLEAR STATION
FIGURE 2.1-3A

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

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

Objective

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

Specification

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

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

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

a. Loss of two pumps and reactor power level is greater than 55% of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power. (Power/RC pump trip setpoint is reset to 55% for all modes of 2 pump operation.)

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

Bases

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

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

Nuclear Overpower

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

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 Unit 1 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 ($16.25\ T_{out}-7756$)
(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 ($11.14\ T_{out}-4746$)
($16.25\ T_{out}-7796$)
($16.25\ T_{out}-7796$)

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°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.

Amendment No. 20, 20 & 17
March 25, 1976

Shutdown Bypass

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

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

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

Two Pump Operation

A. Two Loop Operation

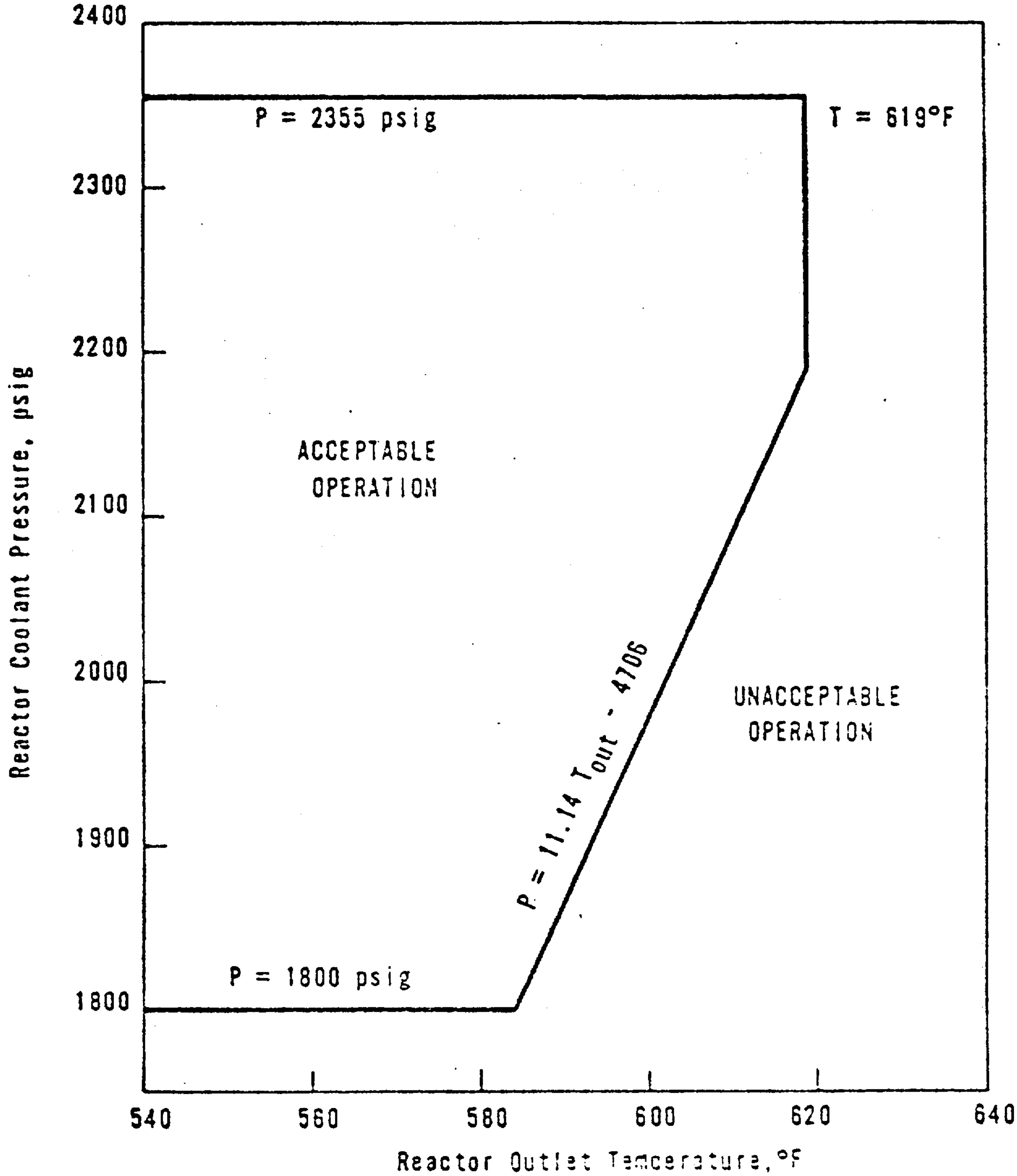
Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

B. Single Loop Operation

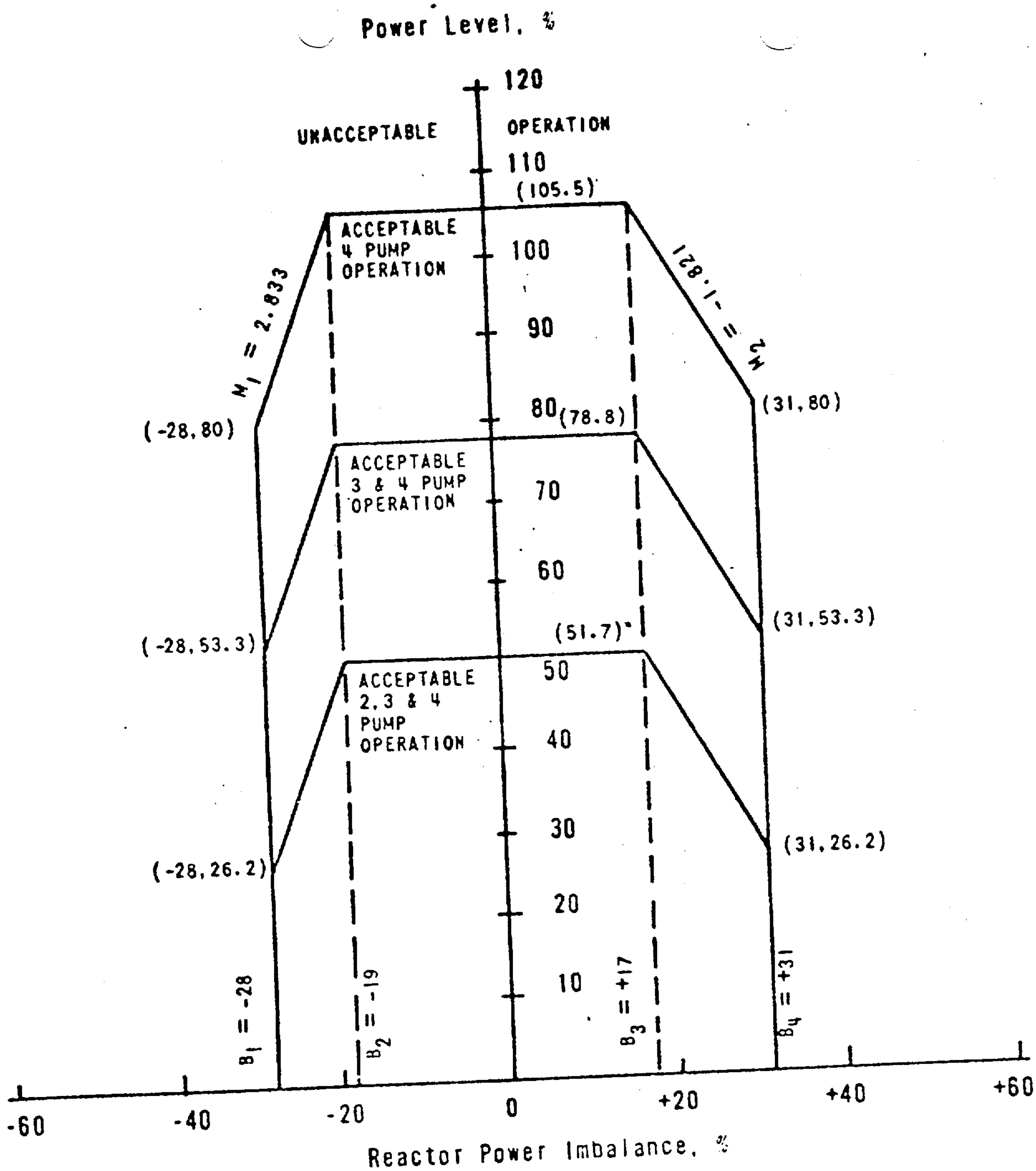
Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:
1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. Reset flux-flow setpoint to 0.949 (Unit 1).
0.961 (Units 2,3)

REFERENCES

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



UNIT 1
PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS
OCONEE NUCLEAR STATION
FIGURE 2.3-1A



*THE FLUX/FLOW SETPOINT
FOR 2/0 PUMP OPERATION
MUST BE SET AT 0.949

UNIT 1
PROTECTION SYSTEM MAXI
ALLOWABLE SETPOINTS
OCONEE NUCLEAR STATI
FIGURE 2.3-2A

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1. Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0(3)
2. Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.055 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	0.949 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55% (5)	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.	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	Bypassed
7. Reactor Coolant Temp. F., Max.	619	619	619 (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 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, single pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

2. Except for test purposes and limited by Specification 2.3, power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24 hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

3. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

c. Pressurizer Safety Valves

1. All pressurizer code safety valves shall be operable whenever the reactor is critical.

2. At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24 hour period is considered very remote.

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ± 1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR Tables 9-11 and 4-3 through 4-7.

(2) FSAR Sections 4.2.5.1 and 9.5.2.3.

(3) FSAR Section 4.2.5.4.

(4) FSAR Sections 4.3.10.4 and 4.2.4.

(5) FSAR Sections 4.3.7 and 14.1.2.2.3.

3.1.8 Single Loop Restrictions

Specification

The following special limitations are placed on single loop operation in addition to the limitations set forth in Specification 2.3.

3.1.8.1 Single loop operation is authorized for test purposes only and requires prior Commission approval.

3.1.8.2 At least 23 incore detectors meeting the requirements of Technical Specification 3.5.4.1 and 3.5.4.2 shall be available throughout this test to check gross core power distribution.

3.1.8.3 The pump monitor trip setpoint shall be set at no greater than 50 percent of rated power.

3.1.8.4 The outlet reactor coolant temperature trip setpoint shall be set at no greater than 610^{o}F.

3.1.8.5 At 15 percent of rated power and every 10 percent of rated power above 15 percent, measurements shall be taken of each operable incore neutron detector and each operable incore thermocouple, reactor coolant loop flow rates and vessel inlet and outlet temperature, and evaluation of this data determined to be acceptable before proceeding to higher power levels.

3.1.8.6 A report covering single loop operation, permitted by Specification 3.1.8, shall be submitted within 90 days after completion of testing. This report shall include the data obtained together with analyses and interpretations of these data which demonstrate:

(1) Coolant flows in the idle loop and operating loop are as predicted.

(2) Relative incore flux and temperature profiles remain essentially the same as for four pump operation at each power level taking into account the reduced flow in single loop operation. 26/21

(3) Operating loop temperatures and flows are obtained which justify the revised safety system setting prescribed for the temperature and flow instruments located in the operating loop (which must sense the combined core flow plus the cooler bypass flow of the idle loop).

Bases

The purpose of single loop testing is to (1) supplement the 1/6 scale model test information, (2) verify predicted flow through the idle loop, (3) verify that changes in power level do not affect flow distribution or core power

3.1.9 Low Power Physics Testing Restrictions

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protective System Requirements

a. Below 1720 psig shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1A - Unit 1.
2.3-1B - Unit 2.
2.3-1C - Unit 3.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1A - Unit 1.
2.3-1B - Unit 2.
2.3-1C - Unit 3.

3.1.9.2 Startup rate rod withdrawal hold shall be in effect at all times. This applies to both the source and intermediate ranges.

3.1.9.3 Shutdown margin may not be reduced below 1.0% $\Delta k/k$ as required by Specification 3.5.2.1 with the exception that the stuck rod worth criterion does not apply during rod worth measurements.

Bases

Technical Specification 3.1.9.2 will apply to both the source and intermediate ranges.

The above specification provides additional safety margins during low power physics testing.

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350°F and irradiated fuel is in the core:

(a) Two high pressure injection pumps shall be maintained operable to provide redundant and independent flow paths.

(b) Engineered Safety Feature valves and interlocks associated with 3.3.2a above shall be operable.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:

(a) The two core flooding tanks shall each contain a minimum of 13 $\pm$ .44 ft. (1040 $\pm$ 30 ft^3) of borated water to 600 $\pm$ 25 psig.

(b) Core flooding tank boron concentration shall not be less than 1,800 ppm boron.

(c) The electrically-operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(d) The electrically-operated core flood tank vent valves CF-5 and CF-6 shall be closed and the breakers locked open and tagged except when adjusting core flood tank pressure.

(e) One pressure instrument channel and one level instrument channel per core flood tank shall be operable.

3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 is operable.

(a) The other reactor building spray pump and its associated spray nozzle header.

(b) The remaining reactor building cooling fan and associated cooling unit.

(c) Engineered Safety Feature valves and interlocks associated with 3.3.4a and 3.3.4b shall be operable.

3.3.5 Except as noted in 3.3.6 below, tests or maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, low pressure service water, reactor building spray, reactor building cooling or penetration room ventilation systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1, 3.3.1, 3.3.3, or 3.3.4 within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 are not met within an additional 48 hours, the reactor shall be placed in a condition below that reactor coolant system condition required in Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 for the component degraded.

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
4.92% Unit 2
4.92% Unit 3
be reduced to less than +3.41% Unit 1 within two hours or the
4.92% 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.
4.92% 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.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.
4.92% Unit 2
4.92% Unit 3

b. If the quadrant tilt exceeds +3.41% Unit 1 and there is simultaneous
4.92% 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,
11.07% 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), 3.5.2-2B (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 amintained 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, 3.5.2-3B, and 3.5.2-3C. If the imbalance is not within the envelope defined by Figure 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B, 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.

Amendment No. 20, 20 & 17
March 25, 1976

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B, 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% $\Delta k/k$ (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.

Amendment No. 20, 20 & 17 March 25, 1976

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 (Unit 1 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
7.36% for Units 2 & 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

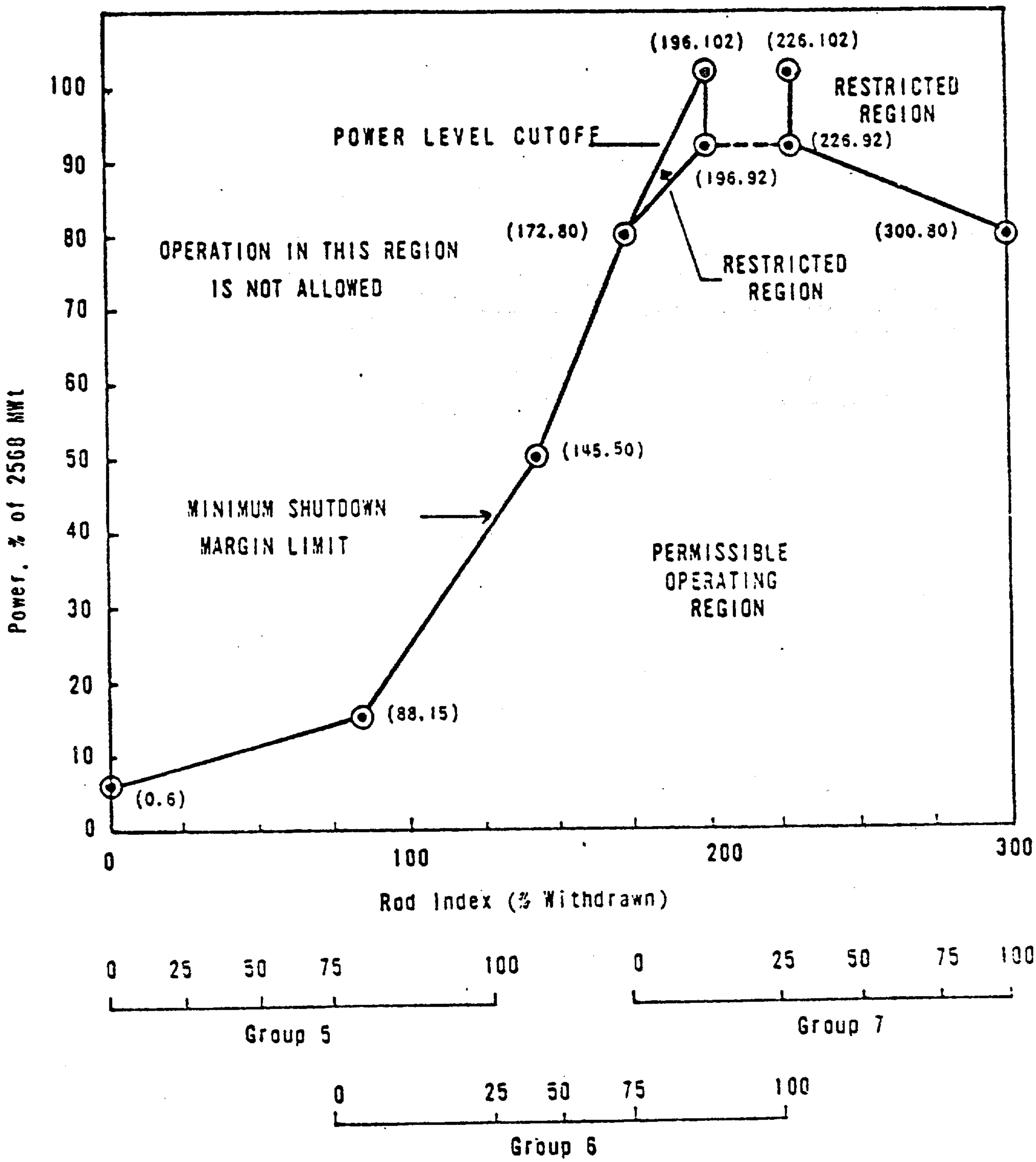
[4] B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

BAW-1400 (UNIT 3)

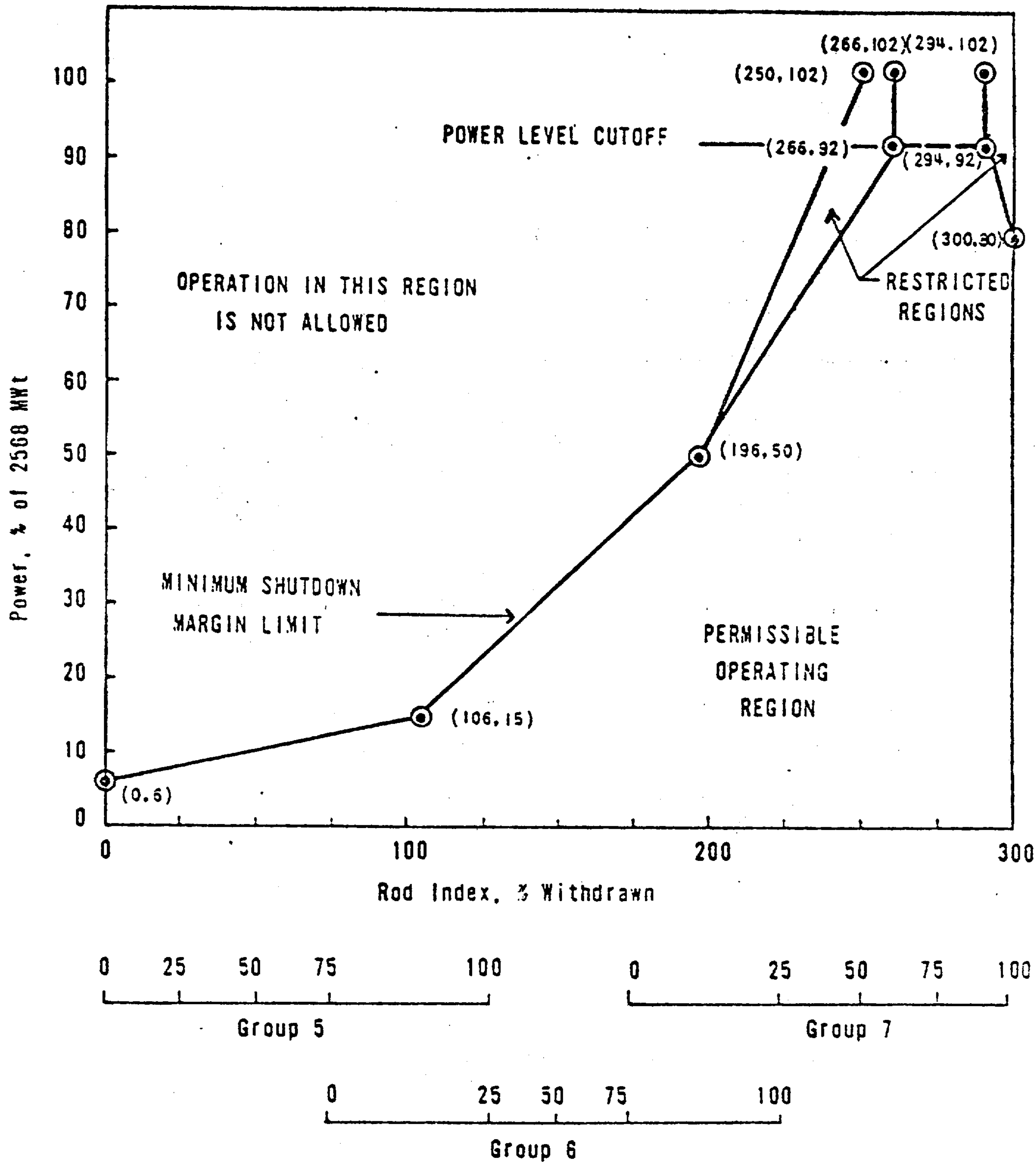
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 0 TO 230 ± 5 EFPD



Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7.

UNIT 1
ROD POSITION LIMITS
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1A1

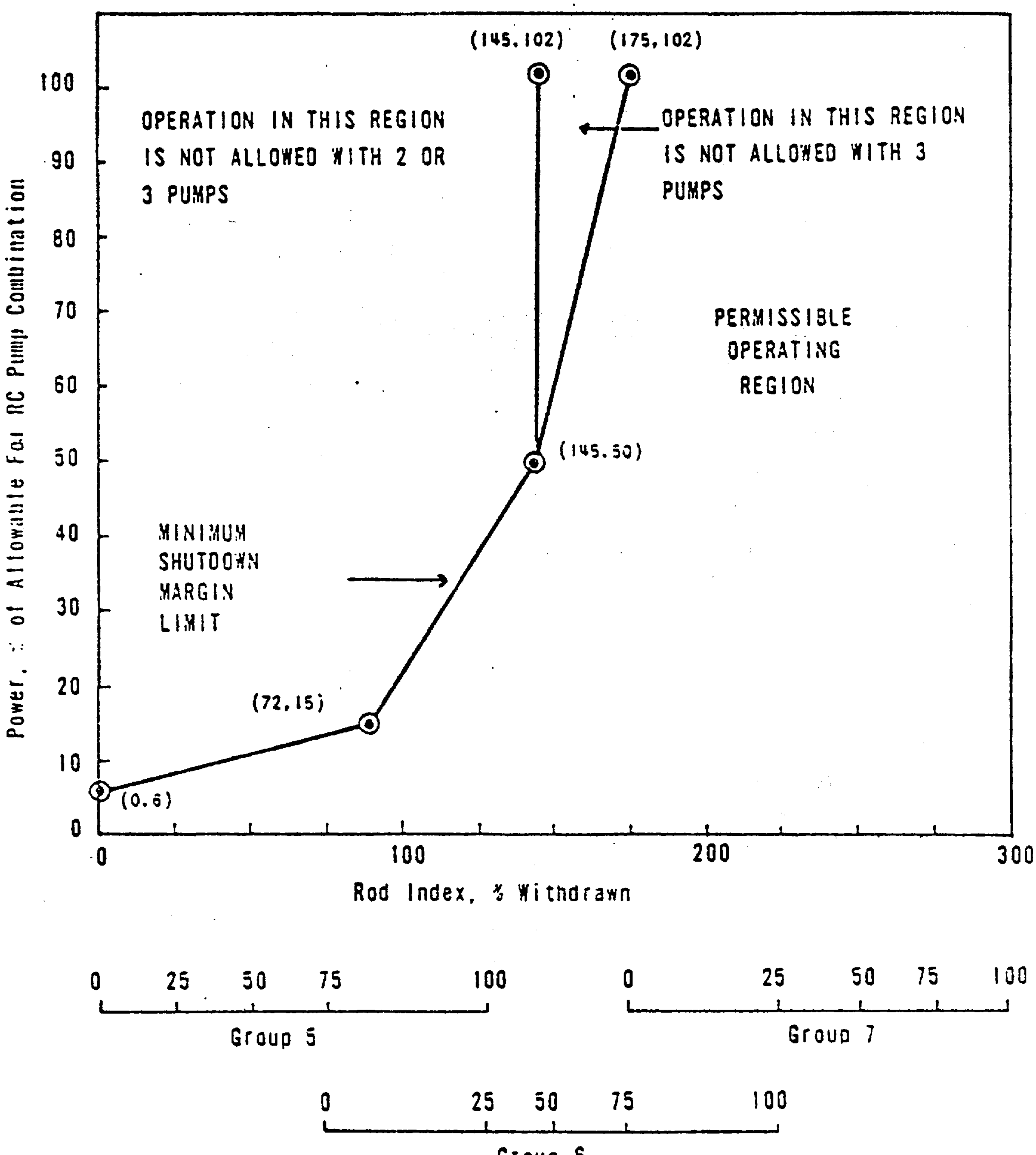
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD AFTER 230 ± 5 EFPD



Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

UNIT 1
ROD POSITION LIMIT
OCONEE NUCLEAR STATION
FIGURE 3.5.2-1A2

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 0 TO 230 ± 5 EFPD

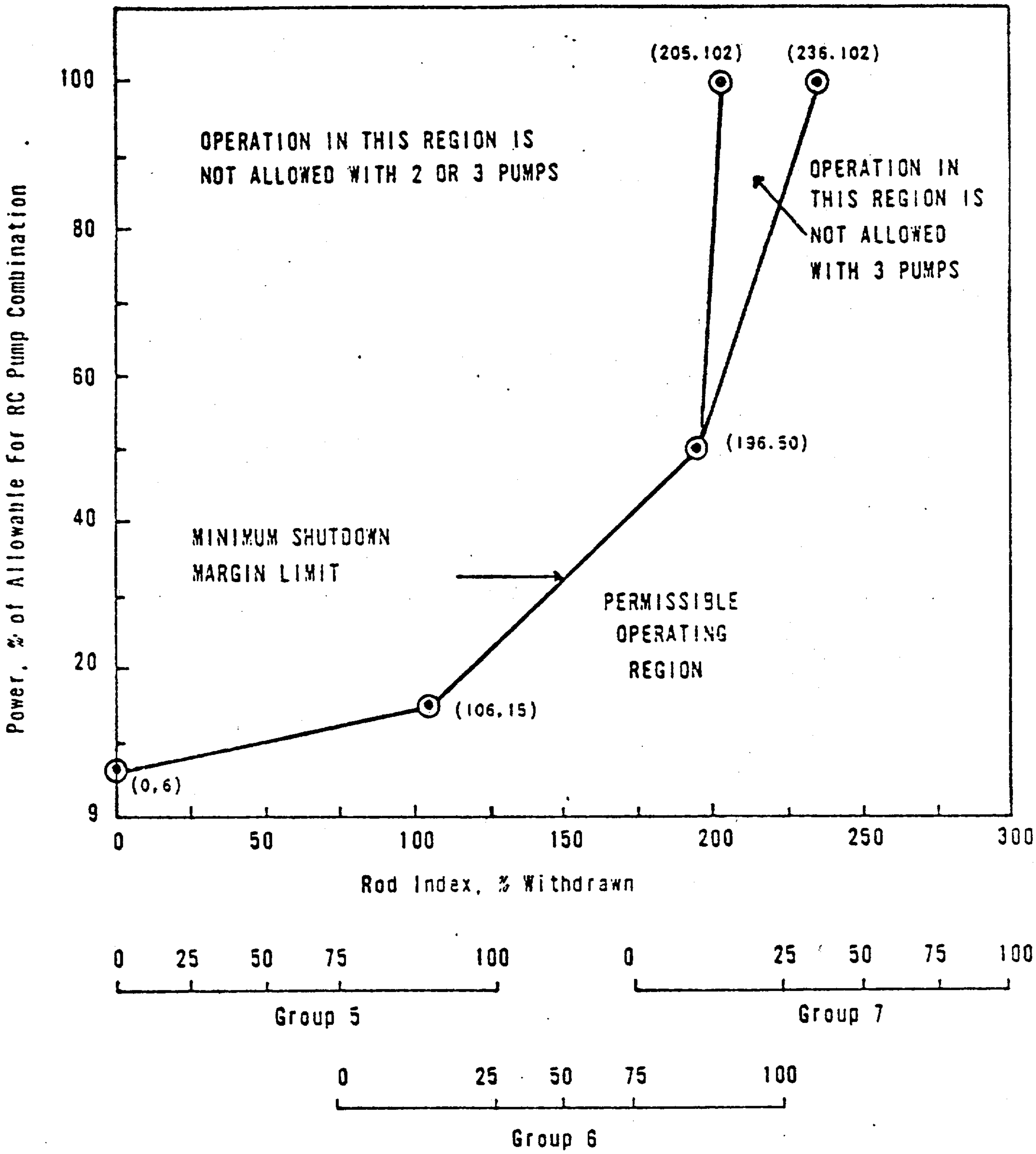


Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

UNIT 1
ROD POSITION LIMITS
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2A1

Amendment No. 20, 20 & 17
March 25, 1976

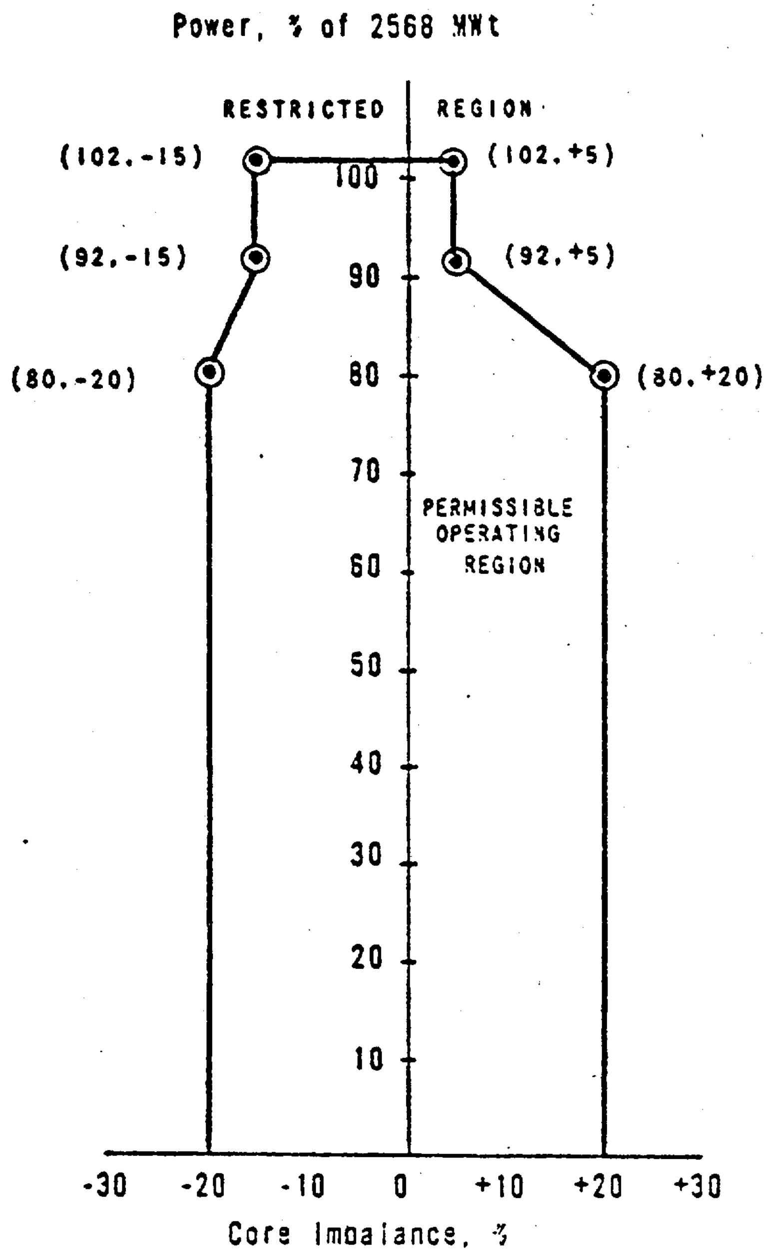
ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE TO THE PERIOD AFTER 230 ± 5 EFPD



Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7.

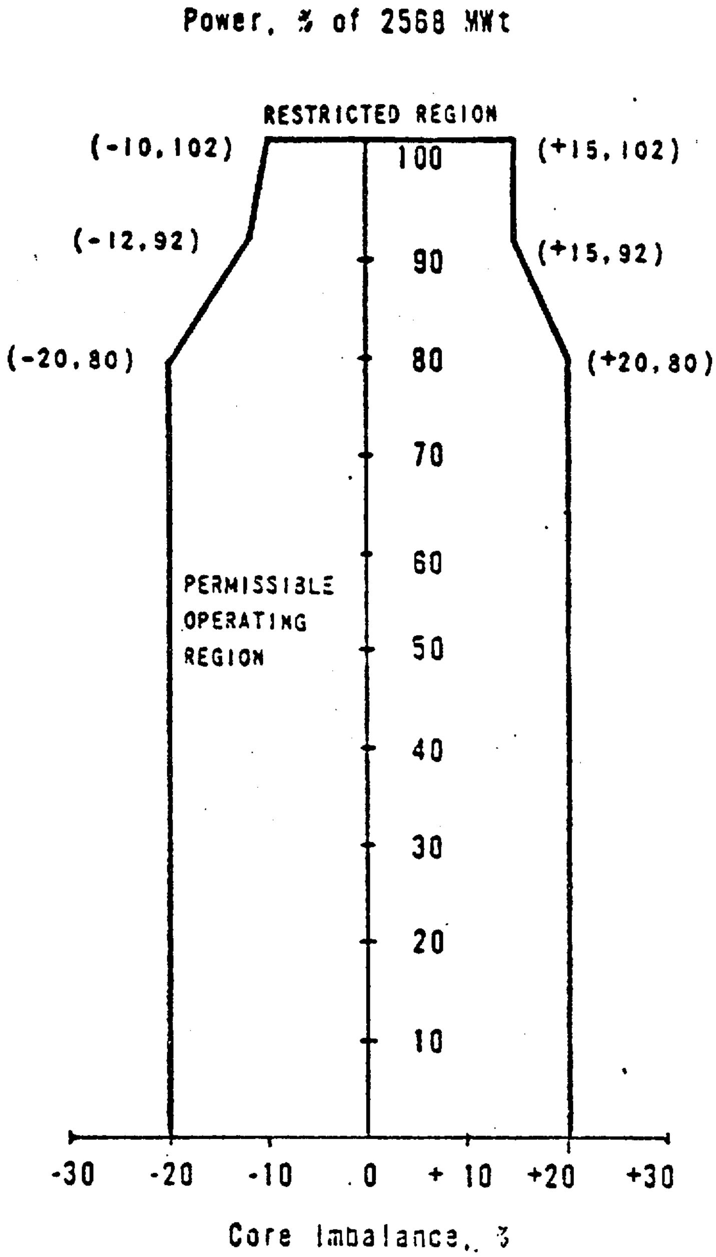
UNIT 1
ROD POSITION LIMITS
OCONEE NUCLEAR STATION
FIGURE 3.5.2-2A2

Amendment No. 20, 20 & 17
March 25, 1976



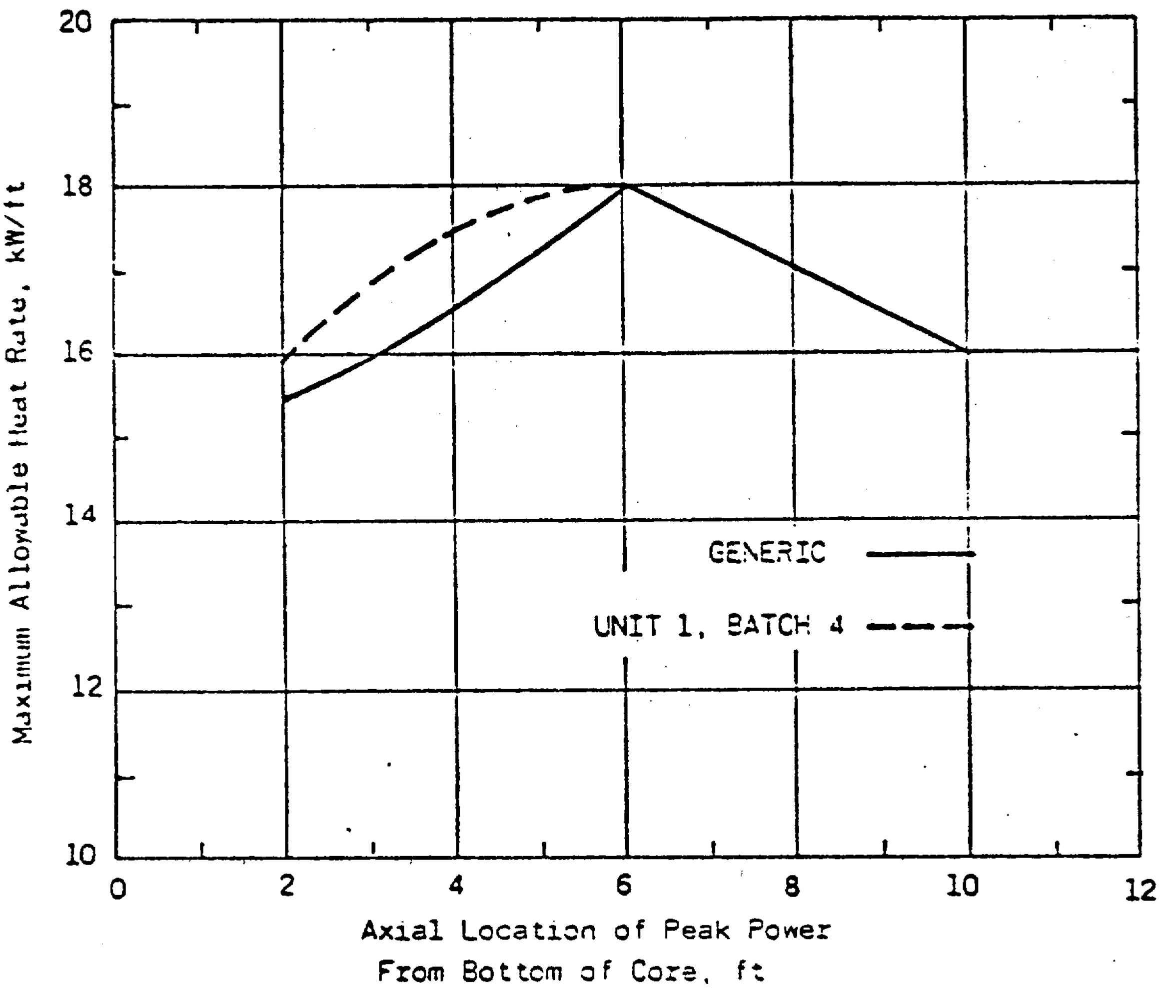
UNIT 1
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 230 ± 5 EFPD
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3A1

Amendment No. 20, 20 & 17
March 25, 1976



UNIT 1
OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION AFTER
230 ± 5 EFPD
OCONEE NUCLEAR STATION
FIGURE 3.5.2-3A2

Amendment No. 20, 20 & 17
March 25, 1976



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

Amendment No. 20, 20 & 17
March 25, 1976

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 x 10^6 lb/hr Unit 2 131.32 x 10^6 lb/hr Unit 3 131.32 x 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. 20 TO FACILITY LICENSE NO. DPR-38

SUPPORTING AMENDMENT NO. 20 TO FACILITY LICENSE NO. DPR-47

SUPPORTING AMENDMENT NO. 20 TO FACILITY LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

Introduction

By letter dated December 1, 1975, Duke Power Company (the licensee) requested a change in the Technical Specifications of License No. DPR-38 for the Oconee Nuclear Station, Unit 1. The proposed amendment is to permit operation of Unit 1 as reloaded for Cycle 3 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 1 reactor core consists of 177 fuel assemblies, each with a 15 x 15 array of fuel rods. The cycle 3 reload will involve the removal of all of the batch 2 fuel (36 assemblies) and 24 of the batch 3 assemblies. The remainder of the batch 3 assemblies and the batch 4 assemblies will be reassigned to new locations for cycle 3 operation. The fresh batch 5 assemblies will occupy primarily the periphery of the core and 4 major axes positions slightly interior to the core. The fuel to be added to the core is not significantly different in design or in operating characteristics from the original fuel it replaces. The rearrangement of fuel assemblies in the reloaded core will affect core physics and thermal hydraulic calculations, and as a result, appropriate changes to the Technical Specifications have been submitted.

The licensee has provided technical information which includes a general description of the reload core, detailed mechanical design data on the reload fuel, nuclear and thermal-hydraulic design data, accident and transient analyses, fuel rod bow analyses and the loss of coolant accident (LOCA) analysis in support of the reload.

Evaluation

1. Fuel and Mechanical Design

Creep collapse calculations were performed by the licensee for three-cycle assembly power histories for Oconee Unit 1 using the Babcock & Wilcox (B&W) computer code, CROV, which we approved in our Generic Review of the B&W Cladding Creep Collapse Analysis Topical Report, BAW-10084, issued on August 9, 1974. The calculations included conservative treatment of effects of fission gas (no credit taken), cladding thickness (lower tolerance limit), initial cladding ovality (upper tolerance limit), and cladding temperature (assembly outlet temperature) on collapse time. The most limiting assembly was found to have a collapse time of more than 26,000 hours which is greater than the maximum projected cycle 3 life of 21,500 hours and is therefore acceptable.

Fuel thermal analysis calculations that account for the effects of fuel densification were performed with our approved version of the B&W analytical model TAFY as described in B&W Topical Report BAW-10044 of May 1972. Fuel densification results in increases in stored energy, linear thermal output and the probability of local power spikes from axial gaps. During cycle 3 operation, the highest relative assembly power levels will occur in batches 4 and 5 fuel. Fuel temperature analysis for batches 1, 2 and 3 fuel is documented in the Oconee 1 Fuel Densification Report, BAW-1388, Revision 1 of July 1973. We agree that this analysis is also applicable to batches 4 and 5 fuel because they have the same linear heat rate capabilities to centerline melt as batches 1, 2 and 3 (20.15 kw/ft). In view of the above, we find the licensee's fuel thermal analysis acceptable.

The batch 5 fuel assemblies are not new in concept and they do not utilize different component materials. Therefore, on the bases of the analysis presented in the reports referenced, we conclude for Oconee Unit 1 cycle 3 that:

(a) The fuel rod mechanical design provides acceptable safety margins for normal operation, and

(b) The effects of fuel densification have been adequately accounted for in the fuel design.

2. Thermal-Hydraulic Analysis

The thermal-hydraulic calculations for the Unit 1 cycle 3 reload core were made using previously approved models and methods. There were no changes due to mechanical differences since the new fuel assemblies are mechanically similar and flow resistances are identical to the previously analyzed cycle 2 core.

As reported in the licensee's letter of August 23, 1973, precision measurement tests of reactor coolant flow were conducted at Oconee Unit 1. As in the cycle 2 reload, a measured flow value based on the coolant flow measurements, instead of the system design flow, is used for the thermal hydraulic analysis for cycle 3.

The coolant flow measurement test results referred to above showed a measured flow rate of 107.8±.82% of design flow. As discussed in the licensee's Startup Report for Unit 1 dated November 16, 1973, corrections to the test data increased this value to 108.6% of design flow. The value of system flow selected for the cycle 3 (and cycle 2) thermal hydraulic analysis, 107.6%, is conservative with respect to the test results referenced above.

The flux/flow trip setpoint for a two-pump coastdown previously determined for cycle 1 (supplement 17 to the Oconee FSAR) has been reevaluated for the cycle 3 core. The procedure was revised to use the measured flow, 107.6% of design flow, instead of the previously used design flow rate. Because of higher system flow rates, most of the orifice plugs have been removed from peripheral fuel assemblies. This increased the predicted core bypass flow by 2.3% (from 6.04% to 8.3%) and has resulted in a 5.3% increase in core flow from the measured 7.6% excess in system flow rate. The core bypass flow was taken into account in the analyses based on the increased system flow rate. In addition, a 4.6% flow penalty for an assumed stuck open core vent valve was used in the analysis.

Based on the licensee's reevaluation, a flux/flow ratio of 1.07 was determined to give a satisfactory minimum Departure from Nucleate Boiling Ratio (DNBR) of 1.31 under two-pump coastdown conditions, starting from 108% power. In the reevaluation, the licensee considered the maximum variation from the average value of the reactor coolant flow signal to provide a conservative indication of flow to the Reactor Protective System. Consequently, the flux/flow trip set point, as proposed for cycle 3 operation, is more conservatively established as 1.055.

In addition to consideration of the variations in the reactor coolant flow signal, as discussed above, the licensee has also included an allowance for the accuracy of the RPS instrumentation string. This error was accounted for in the flux value used to establish the flux/flow trip setpoint.

The present 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 primary flow instrumentation string using an actual differential pressure as input to the system d/p cells. The accuracy of these checks are on the order of $\pm 1\%$.

To assure continual confidence in the calibration discussed above, a Technical Specification has been included which will require that the reactor coolant system flow be verified to be at least 141.3×10^6 lbs/hr (107.6% design flow) at least once each fuel cycle.

In summary, the licensee has proposed, as in cycle 2, that a reactor coolant flow rate based on measured flow be used in place of design flow in the analyses involving reactor coolant flow. In conjunction with this, the flux/flow trip setpoint has been reevaluated to meet the revised limiting DNBR of 1.3. In our review of these items, we considered the difference between the value of reactor coolant flow used in the calculations (107.6% design flow) and the actual measured flow (108.6% design flow), the accuracy of the calibrations performed and the conservative allowances taken by the licensee in the analyses. In addition, the 4.6% reactor coolant flow penalty imposed for an assumed stuck open core vent value has been determined to no longer be necessary. This has the effect of adding additional conservatism to the analyses performed for the cycle 3 core. In view of the above, we conclude that the use of measured rather than design flow is acceptable.

Thermal hydraulic design calculations for cycle 3 operation utilized the same analytical methods previously documented in the Unit 1 FSAR and the Unit 1 Cycle 2 reload submittal. Adjustments to the calculations were made to account for modifications in the use of the BAW-2 Critical Heat Flux (CHF) correlation which was used for the cycle 2 reload. Two modifications to the BAW-2 CHF correlation have been introduced for its application to the cycle 3 core. These are:

(a) An extension downward from 2000 psia to 1750 psia of the pressure range applicable to the correlation, and

(b) 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 rods will not experience DNB.

We recently completed a re-evaluation of the BAW-2 CHF correlation to verify its continued suitability in relation to available rod bundle DNB 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.

In conjunction with our reevaluation of the BAW-2 CHF correlation we also reviewed the licensee's proposed modifications to the correlation for the cycle 3 core. The original data base for the correlation covered the pressure range 2000-2450 psia and resulted in a 1.32 minimum allowable DNB ratio to ensure with 99% confidence that 95% of the hot rods did not experience DNB. As an attachment to their letter of February 3, 1976, B&W provided information which compared the BAW-2 CHF correlation with data in the low pressure range from five different test bundles. The mean measured-to-predicted ratio for all data was 1.05 and the minimum allowable DNBR was 1.29 for a 95% confidence that 95% of the hot rods at the DNBR would not experience DNB.

The 1.32 minimum DNB ratio used by B&W is based upon 95% of the hot rods at that DNBR not experiencing DNB, with a 99% confidence. If the confidence level is changed to 95%, which is consistent with the standard review plan and industry practice, the minimum allowable DNBR becomes 1.30.

Based on the above, we find both the extension of the BAW-2 CHF correlation to pressures down to 1750 psia and the change to a minimum DNBR of 1.30 to be acceptable. The BAW-2 CHF correlation has been shown to be conservative in the low pressure region and the change to a 1.30 minimum DNBR is consistent with the requirements of Standard Review Plan 4.4. In addition, the proposed reduction in the reactor coolant low pressure trip (1800 psig from 1985 psig) is consistent with the extension of BAW-2 CHF correlation downward to 1750 psig and is therefore also acceptable.

3. Nuclear Analysis

The licensee has provided values for core physics parameters for the Unit 1 cycle 3 core which reflect minor differences when compared to those for cycle 2. These differences are attributable to the fact that the core has not yet reached an equilibrium cycle and such differences are to be expected. We have concluded that no significant changes exist in the core design between cycles 2 and 3. In addition, the same calculational methods and design information were used to obtain the important nuclear design parameters. Based on the above and the fact that startup tests (to be conducted prior to power operation) will verify that the critical aspects of core performance are within the assumptions of the safety analysis, we find the licensee's nuclear analysis for cycle 3 to be acceptable.

4. Transient and Accident Analysis

The licensee has provided the results of examinations conducted of each FSAR accident analysis with respect to changes in cycle 3 parameters to determine the effects of the reload and to ensure that thermal performance during hypothetical transients is not degraded. We have reviewed the licensee's submittal and agree that in most cases the consequences of transients are less severe and in no case are they more severe.

5. Rod Bow Penalty

By letter dated February 27, 1976, the licensee provided information to supplement its December 1, 1975 cycle 3 reload submittal which would revise the Technical Specifications to account for the effect of rod bow on core parameters. In conjunction with these revisions, the licensee is also proposing changes to quadrant tilt specifications, applicable to all three Oconee units, which would specify the limit on actual quadrant power tilt, using as a frame of reference the real core power ratio instead of the power ratio measured by just the out-of-core detector system, as is presently done.

In the analysis supporting the proposed Technical Specification changes for Unit 1 the licensee indicated that:

(a) The rod bow effect on the flow area of the hot channel is adequately compensated for by the flow area reduction factor,

(b) 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 1 cycle 3 DNBR limit (1.30) conservatively accounts for the effects of rod bowing, and

(c) The power spike due to rod bow, when added to the other factors affecting the power imbalance limit for the Reactor Protection System (RPS), necessitates a reduction in the core safety and RPS imbalance limits. These limits exist to preclude exceeding the central fuel melt criteria which is more limiting than DNBR for cycle 3.

In view of the considerations identified in (c) above, the licensee is proposing that a rod bow spike penalty of 2.15% be absorbed by reducing the quadrant tilt limit for Unit 1, from 4% to 2.77%. These values would be the limit when the out-of-core detectors are used for quadrant tilt measurements. To improve clarity and provide a quadrant tilt limit which would be independent of the measurement system used (out-of-core or in-core detector system) the licensee is proposing to also revise the quadrant tilt specifications to refer to actual quadrant tilt and to use this method in the operation of all three Oconee Units. The equivalent peaking increase for unit 1 would then be revised from 7.36% to 5.10%, to account for rod bow effects.

In addition to the power spike penalty associated with the rod bowing phenomenon there has been determined to be a DNB penalty resulting from displaced coolant flow. This penalty, however, is essentially compensated for by allowances made in the design. B&W has not yet formally submitted a rod bow model for our review. The model we have utilized is appropriately conservative, however, due to the uncertainties involved and the lack of sufficient supportive data, we have imposed an additional 2% DNB penalty.

The licensee's proposed reduction in the quadrant tilt limit to accommodate the rod bow spike penalty is more limiting than the 2% DNB penalty we have imposed and is therefore more conservative. Based on the above, we find the proposed Technical Specifications for Units 1, 2 and 3 to be acceptable.

6. 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 December 1, 1975, the licensee has submitted the related Technical Specifications for Unit 1, cycle 3. 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 1 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 1; BAW-10104, May 1975 and BAW-10103, June 1975, respectively. In addition, Duke Power Company included in this July 9th submittal a separate analysis of the worst break for Oconee Unit 1, using the following plant-specific parameters:

(a) Power level = 1.02 x 2568 Mwt. The generic analyses in BAW-10103 used 1.02 x 2772 Mwt.

(b) Initial average fuel temperature assumed reflects the reload core (T = 3030°F for 18 kw/ft with 580°F sink temperature). The generic analyses used T = 3050°F.

(c) Different pin dimensions were employed to reflect fuel changes.

(d) Core flood tank line resistance was changed to reflect the as-built value for Oconee Unit 1 (6.5 versus 7.75 in generic analyses).

(e) System enthalpies and steam generator heat loads were changed to reflect the lower power level of 2568 Mwt.

(f) Initial pin pressures and oxide layer thicknesses were changed to reflect the different fuel in Oconee 1.

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 1:

Elevation (ft)	LOCA Limit (kw/ft)	Peak Cladding Temperature (°F)		Max. Local Oxidation (%)	Time of Rupture (sec)
		Ruptured Node	Unruptured Node		
Oconee 1					
2	16.0	1882	1930	3.40	10.90
4	17.5	1975	1978	3.17	12.39
6	18.0	2066	2146	5.46	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 reactor for Oconee 1 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 reactor 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 1 at the 10-foot elevation.

As indicated above, Duke Power Company elected to provide a plant-specific calculation for Oconee Unit 1 utilizing selected as-built data. We have reviewed the input changes used (relative to BAW-10103) and believe them appropriate for Oconee Unit 1.

We have reviewed the Technical Specifications proposed by the licensee in the July 9, 1975 submittal, and as revised October 31, 1975, to assure that operation of Oconee Unit 1 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. For Unit 1, the LOCA-related heat generation limits (maximum of 18.0 Kw/ft) occur in the Cycle 2 reload fuel (batch 4). We have concluded that the proposed Technical Specifications, as submitted for Unit 1 cycle 2 operation, meet the necessary criteria and are acceptable. Since Oconee Unit 1 is currently undergoing refueling for Cycle 3 operation we have also reviewed the proposed Technical Specifications for Cycle 3 operation to assure that they also meet the FAC. We have determined that the LOCA related heat generation limits, as for cycle 2, occur in the batch 4 fuel. The maximum LOCA related heat generation rate is therefore unchanged at 18.0 Kw/ft. Based on the above, we find that the proposed Technical Specifications for cycle 3 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 the Oconee 1 analyses addressed the areas of single failure criterion, 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 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 Oconee 1 piping and instrumentation diagrams indicated that the 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. The rate at which this decrease occurs is controlled by a preset needle throttling valve (CF-16 or CF-18) downstream of the electrically-operated valve. The predetermined position of the needle valve is provided by manually turning the local handwheel such an amount as to limit the rate at which a depressurization of the CFT could take place. A recent test at Oconee indicated that the tested valve setting allowed 17 minutes for the CFT pressure to decay from 625 psi to the low pressure alarm, 580 psi, when the electrically-operated valves were opened. Since it is clear that CFT pressure is important to mitigating the consequences of a LOCA, a Technical Specification is included which will require that the normally closed motor-operated valves CF-5 and CF-6 have their breakers locked open and tagged except when adjusting core flood tank pressure.

A review was also conducted of the electrical schematics for ECCS motor-operated valves. It was determined that a single failure of valve interlocks could not affect the appropriate single failure assumptions.

To further minimize the potential for a water hammer due to the discharge of ECC water into a dry line, we will require that valves LP-21 and LP-22 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. Such a configuration will also eliminate the need for one automatic safety action in the event of a LOCA; that is, the automatic opening of these valves to provide water to the LPI pumps.
The normal value lineup in HPI system provides a similar supply of water to the HPI pumps. In addition, a Technical Specification is included to 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.

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 1 was submitted in the licensee's submittal of July 9, 1975.

Justification for the containment input data was submitted for Oconee Unit 1 by letter dated October 10, 1975. This justification allows comparison of the actual containment parameters for Unit 1 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. 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 Oconee Unit 1.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Oconee 1 is reasonably 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 systems designed for preventing excessive boric acid buildups in the reactor vessel during the long-term cooling period after a LOCA. Duke Power Company has agreed to implement procedures for Unit 1 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, Duke Power Company will make modifications to the existing Decay Heat Removal (DHR) design during the cycle 3 refueling outage. The proposal consists of the addition of two drain lines from the decay heat drop line to the sump. One line (installed upstream of the DHR isolation valves) will include two qualified motor-operated valves. The other line (installed downstream of the DHR isolation valves) will include one qualified motor-operated valve. 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 committed to the installation, prior to cycle 4 operation, of equipment to provide positive indication of flow in the drain lines.

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 3 operation that the system will function under post-LOCA conditions.

Submerged Valves

The applicant has conducted a review of equipment arrangement to determine if any valve motors inside the containment will become submerged following a LOCA. Based on this review, no valves were identified which would be flooded and which would affect short-term or long-term ECCS functions or containment isolation.

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 blowdown 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 will be limited by Technical Specifications to 24 hours.

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

We have completed the review of the Oconee 1 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 provided that the modifications as specified above are implemented.

(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 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 1 cycle 3 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 amendment does 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 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: March 25, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 20, 20 and 17 to Facility Operating Licenses No. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company which revised Technical Specifications for operation of the Oconee Nuclear Station, Units 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of their date of issuance.

These amendments (1) revise the Technical Specifications to establish operating limits for Unit 1 cycle 3 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR 50.46, (2) terminate the operating restrictions imposed on Unit 1 by the Commission's December 27, 1974 Order for Modification of License and (3) revise the Technical Specifications to specify quadrant power tilt limits for Units 1, 2 and 3 independent of the measurement system used.

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-38 in connection with Unit 1 Cycle 3 reload was published in the FEDERAL REGISTER on February 5, 1976 (41 F.R. 5354). 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 statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendment dated December 1, 1975, as supplemented February 24 and 27, 1976, (2) Amendments No. 20 , 20, and 17 to Licenses No. DPR-38, DPR-47, and DPR-55, (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 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 25th day of March 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors