October 4, 1977

DISTRIBUTION
Dockets
NRC PDRs
LOCAL PDR
ORB#1 Reading
VStello
KRGoller
SMSheppard
DNeighbors
ASchwencer
OELD
OI&E(5)
BJones(12)
BScharf(15)
ACRS(16)
JMcGough
JSaltzman
BHarless
DEisenhut
CMiles, OPA
DRoss
TBABernathy
JRBuchanan

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

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 Amendment Nos. 47 , 47 and 44 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Unit Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated March 30, 1977, as supplemented June 21, August 23, September 8, 14 and 24, 1977.

These amendments revise the Technical Specifications to establish operating limits for Unit 1 cycle 4 operation and tighten leakage limits through the Steam Generator tubes.

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

Sincerely,

/s/

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

Enclosures:
1. Amendment No. 47 to DPR-38
2. Amendment No. 47 to DPR-47
3. Amendment No. 44 to DPR-55
4. Safety Evaluation Supporting Amendment Nos. 47 , 47 and 44
5. Safety Evaluation of Steam Generator Tube Degradation Phenomenon
6. Notice of Issuance

cc w/encl: See next page

DOR:ORB#1	DOR:ORB#1
DNeighbors:lb	ASchwencer
10/04/77	10/ /77

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

October 4, 1977

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

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 Amendment Nos. 47 , 47 and 44 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Unit Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated March 30, 1977, as supplemented June 21, August 23, September 8, 14 and 24, 1977.

These amendments revise the Technical Specifications to establish operating limits for Unit 1 cycle 4 operation and tighten leakage limits through the Steam Generator tubes.

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

Sincerely,

A. Schwencer

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

Enclosures:
1. Amendment No. 47 to DPR-38
2. Amendment No. 47 to DPR-47
3. Amendment No. 44 to DPR-55
4. Safety Evaluation Supporting Amendment Nos. 47 , 47 and 44
5. Safety Evaluation of Steam Generator Tube Degradation Phenomenon
6. Notice of Issuance

cc w/encl: See next page

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

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

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

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

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

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

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

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated March 30, 1977, as supplemented June 21, August 23, September 8, and 14, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated March 30, 1977, as supplemented June 21, August 23, September 8, and 14, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 4, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated March 30, 1977, as supplemented June 21, August 23, September 8, and 14, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 47 TO DPR-38

AMENDMENT NO. 47 TO DPR-47

AMENDMENT NO. 44 TO DPR-55

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

Revise Appendix A as follows:

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

2.1-1	3.5-8	5.3-1
2.1-2	3.5-9	
2.1-3	3.5-10	
2.1-4	3.5-11	
2.1-7	3.5-12	
2.1-10	3.5-13	
2.3-1	3.5-18	
2.3-2	3.5-18a	
2.3-3	3.5-21	
2.3-4	3.5-21a	
2.3-11	3.5-24	
2.3-12	4.1-9	
3.1-14	3.1-15	

2. Add pages:

3.5-13a
3.5-18b
3.5-21b
3.5-23c
3.5-23d
3.5-23e

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]. The reactor coolant system flow rate utilized is 106.5 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 106.5 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.

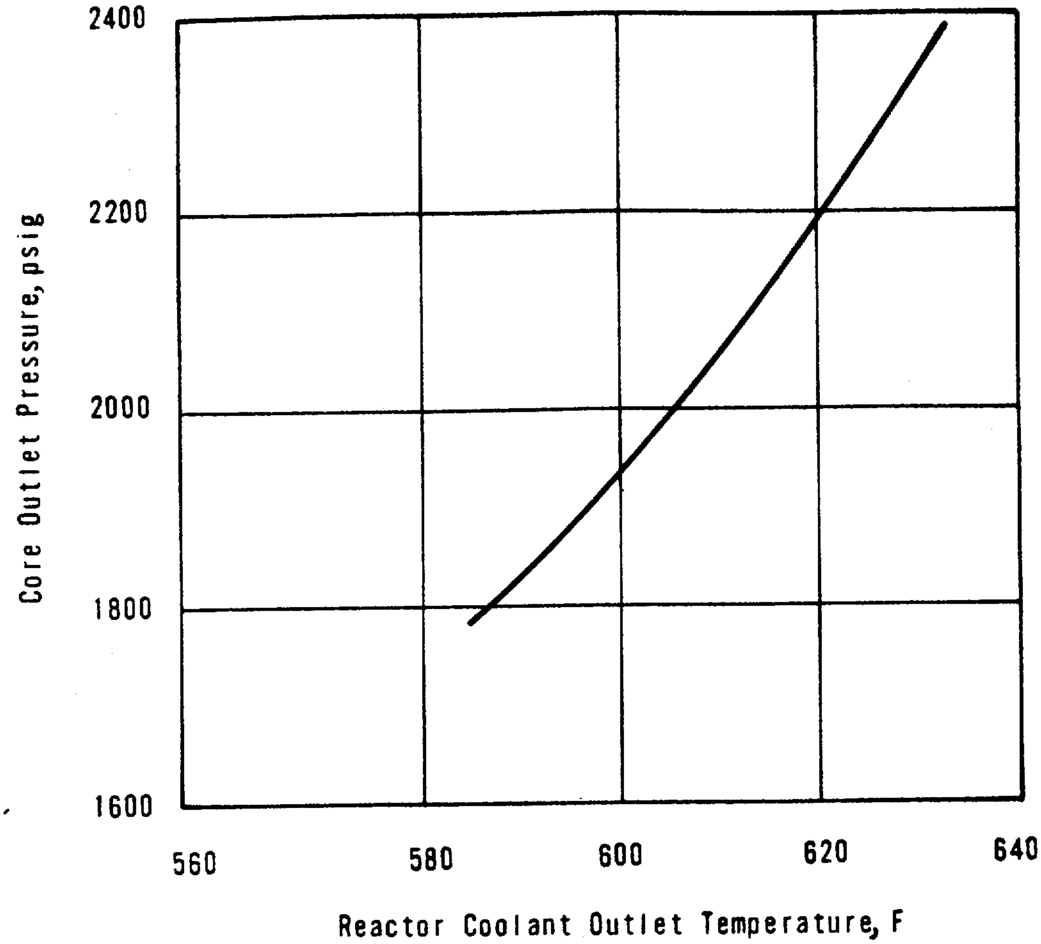
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055 = 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a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

References

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

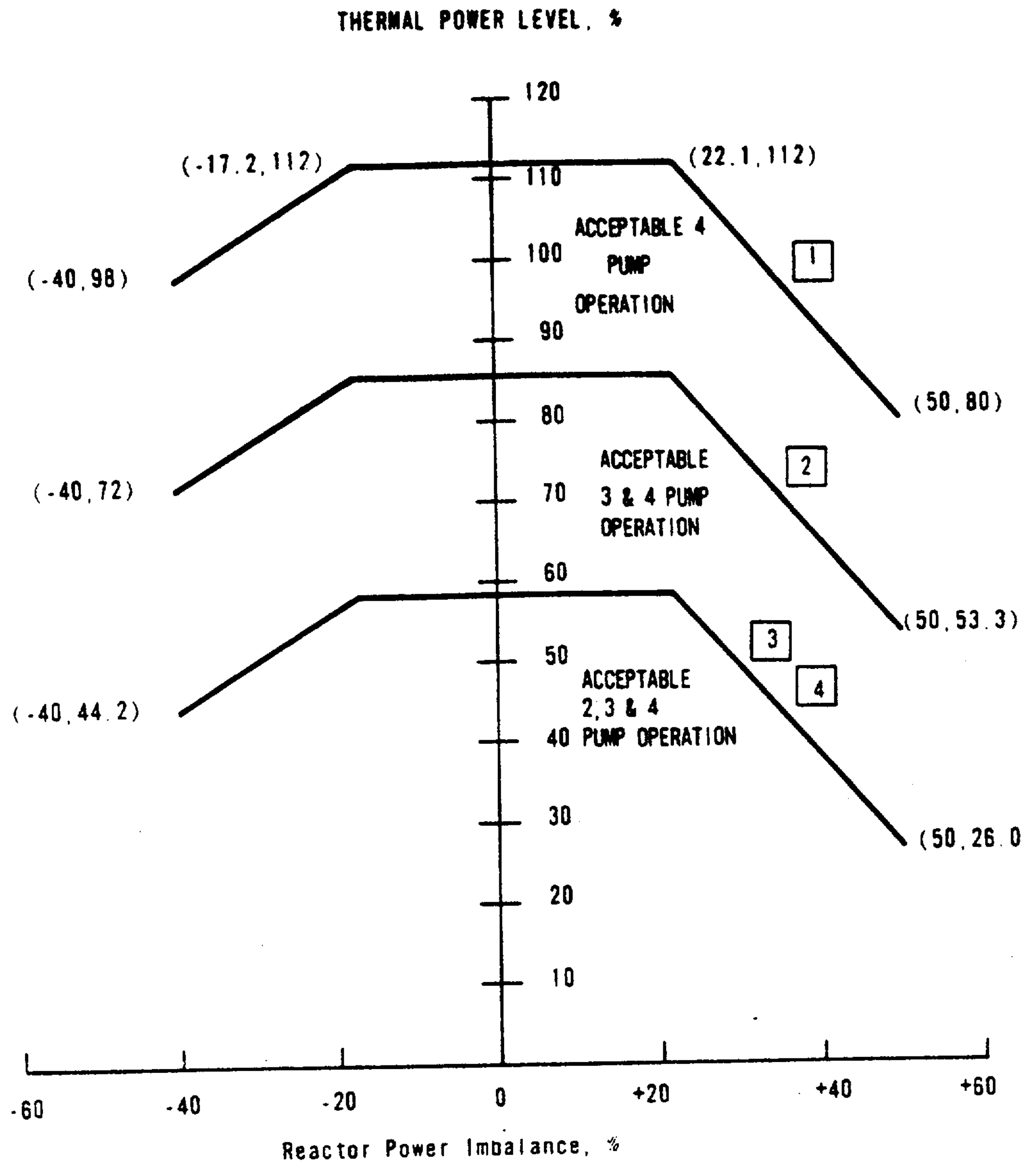
(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March, 1977.



CORE PROTECTION SAFETY LIMITS, UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-1A

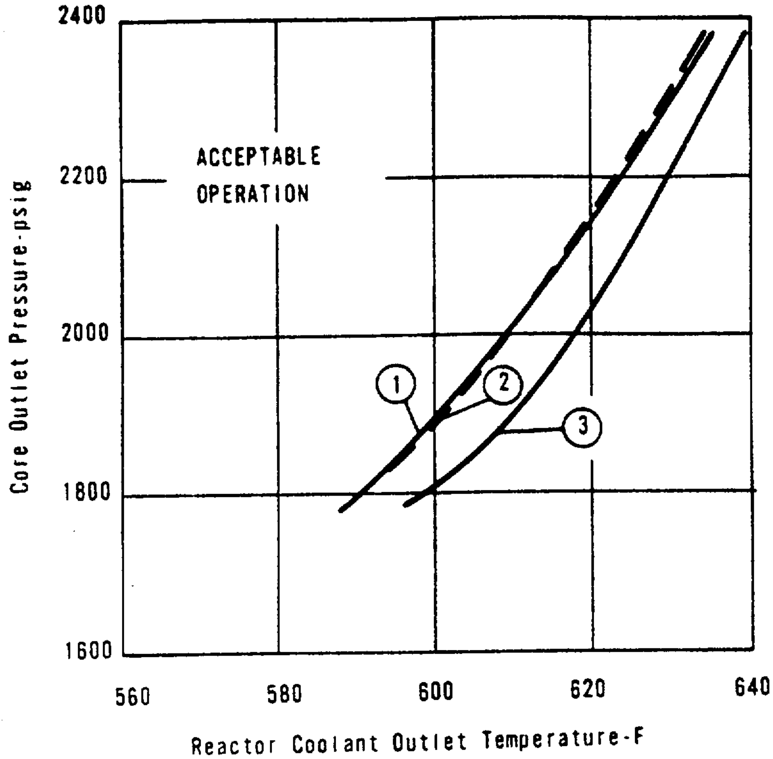


CURVE	REACTOR COOLANT FLOW (GPM)
1	374880
2	280035
3	183690
4	204310

CORE PROTECTION SAFETY LIMITS, UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-2A



CURVE	REACTOR COOLANT FLOW (GPM)	POWER	PUMPS OPERATING	(TYPE OF LIMIT)
1	374880 (100%)*	112%	4	(DNBR)
2	280035 (74.7%)	86.7%	3	(DNBR)
3	183690 (49.0%)	59.0%	2	(QUALITY)

* 106.5% OF FIRST CORE DESIGN FLOW

CORE PROTECTION SAFETY LIMITS, 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

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

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

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

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

Bases

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

The trip setting limits for protective system instrumentation are listed in Table 2.3-1A - Unit 1. The safety analysis has been based upon these protective
2.3-1B - Unit 2
2.3-1C - Unit 3
system instrumentation trip 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 account for the maximum calibration and instrument 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.
1.055%-Unit 2
1.07% -Unit 3
For Units 1 and 2 the power-to-flow reduction ratio is 0.949, and for Unit 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 ($11.14\ T_{out}-4706$)
(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 ($11.14\ T_{out}-4746$)
($11.14\ T_{out}-4746$)
($10.79\ T_{out}-4579$)

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.

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

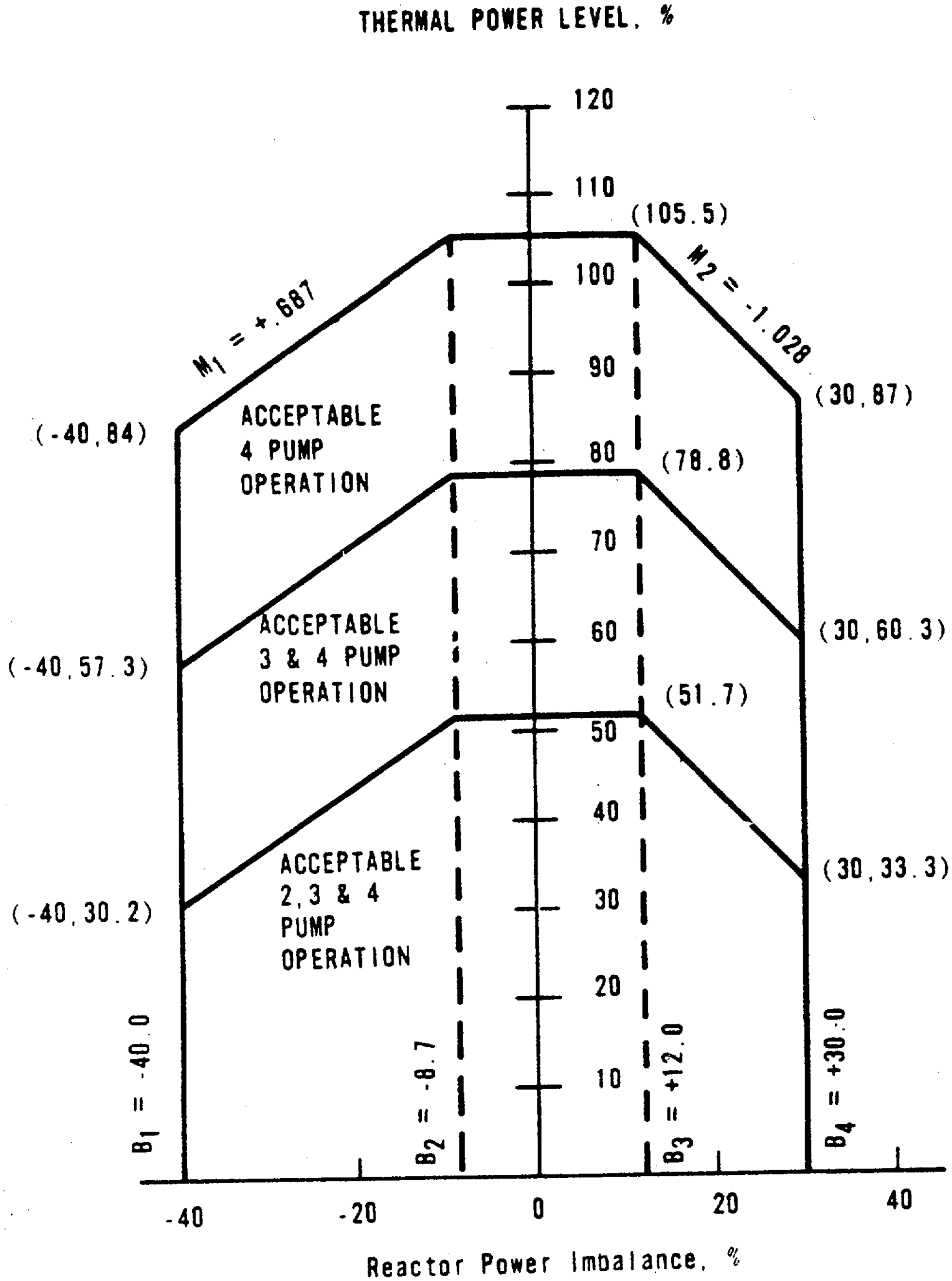
Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

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)
0.949 (Unit 2)
0.961 (Unit 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



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.3-2A

Amendments 47, 47 & 44

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	80%	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 (°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.

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating -49% Rated)	Shutdown Bypass
1. Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0 (3)
2. Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.055 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	80%	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.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	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.

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If at any time, the leakage through the Unit 1 steam generator tubes equals or exceeds 0.3 gpm, a reactor shutdown shall be initiated within 4 hours and and the reactor shall be in a cold condition within the next 36 hours. If the leakage is less than 0.3 gpm, as assessment shall be made whether operations may be continued safely or the plant should be shutdown. In either case, the NRC shall be notified in accordance with Section 6.6.2 i.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7 except that such losses when added to leakage shall not exceed 30 gpm.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage including evaporative losses (which may be on the order of .5 gpm) to the lowest possible rate and at least below 1 gpm in order to prevent a large leak from masking the presence of a smaller leak. Water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small breaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one high pressure injection pump and makeup would be available even under the loss of off-site power condition.

If leakage is to the reactor building it may be identified by one or more of the following methods:

a. The reactor building air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are .10 gpm to greater than 30 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in coolant leakage of 1 gpm is detectable within 10 minutes after it occurs.

b. The iodine monitor, gaseous monitor and area monitor are not as sensitive to corrosion product activity.(1) It is calculated that the iodine monitor is sensitive to an 8 gpm leak and the gaseous monitor is sensitive to a 230 gpm leak based on the presence of tramp uranium (no fission products from tramp uranium are assumed to be present). However, any fission products in the coolant will make these monitors more sensitive to coolant leakage.

c. In addition to the radiation monitors, leakage is also monitored by a level indicator in the reactor building normal sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as reactor coolant system, low pressure service water system, component cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The sump capacity is 15 gallons per inch of height and each graduation on the level indicates 1/2 inch of sump height. This indicator is capable of detecting changes on the order of 7.5 gallons of leakage into the sump. A 1 gpm leak would therefore be detectable within less than 10 minutes.

(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. **Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours, or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.**

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1, 3.5.2-1A2 and 3.5.2-1A3 (Unit 1); 3.5.2-1B1, 3.5.2-1B2 and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1, 3.5.2-2A2 and 3.5.2-2A3 (Unit 1); 3.5.2-2B1, 3.5.2-2B2 and 3.5.2-2B3 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, 3.5.2-4A2 and 3.5.2-4A3 (Unit 1) and on figures 3.5.2-4B1, 3.5.2-4B2, and 3.5.2.4B3 (Unit 2). If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

d. Except for physics tests, power shall not be increased above the power level cutoff as shown on Figures 3.5.2-1A1, 3.5.2-1A2,3.5.2-1A3, (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, 3.5.2-3A3, 3.5.3-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

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

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (See Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification effects
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing effects

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

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

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

**Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument 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.
5.10% for Unit 2
5.10% for Unit 3
The limits shown in Specification 3.5.2.4 are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

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

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

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

REFERENCES

[1] FSAR, Section 3.2.2.1.2

[2] FSAR, Section 14.2.2.2

[3] FSAR, SUPPLEMENT 9

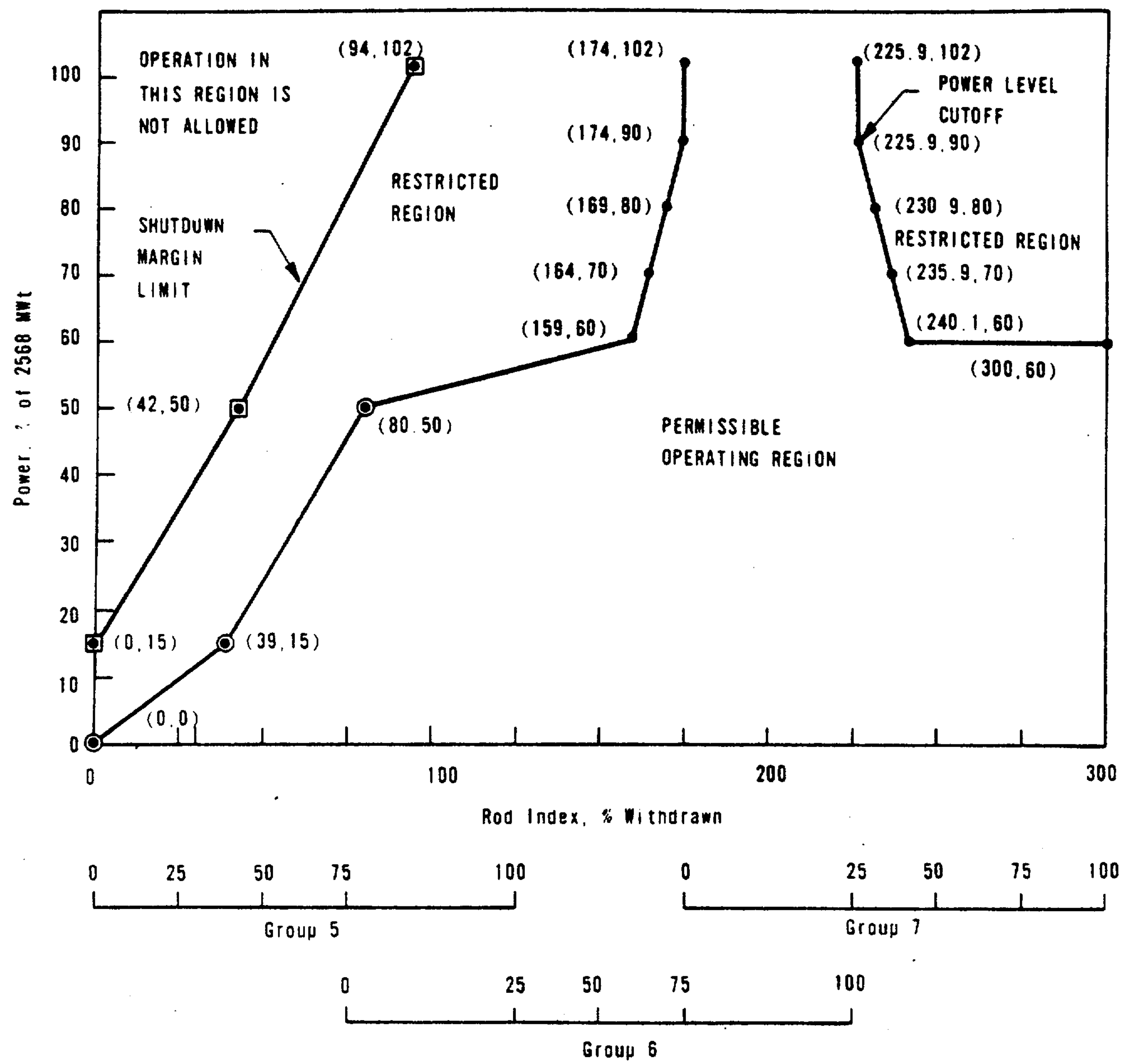
[4] B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

BAW-1400 (UNIT 3)

[5] Oconee 1, Cycle 4 - Reload Report - BAW-1447, March 1977.



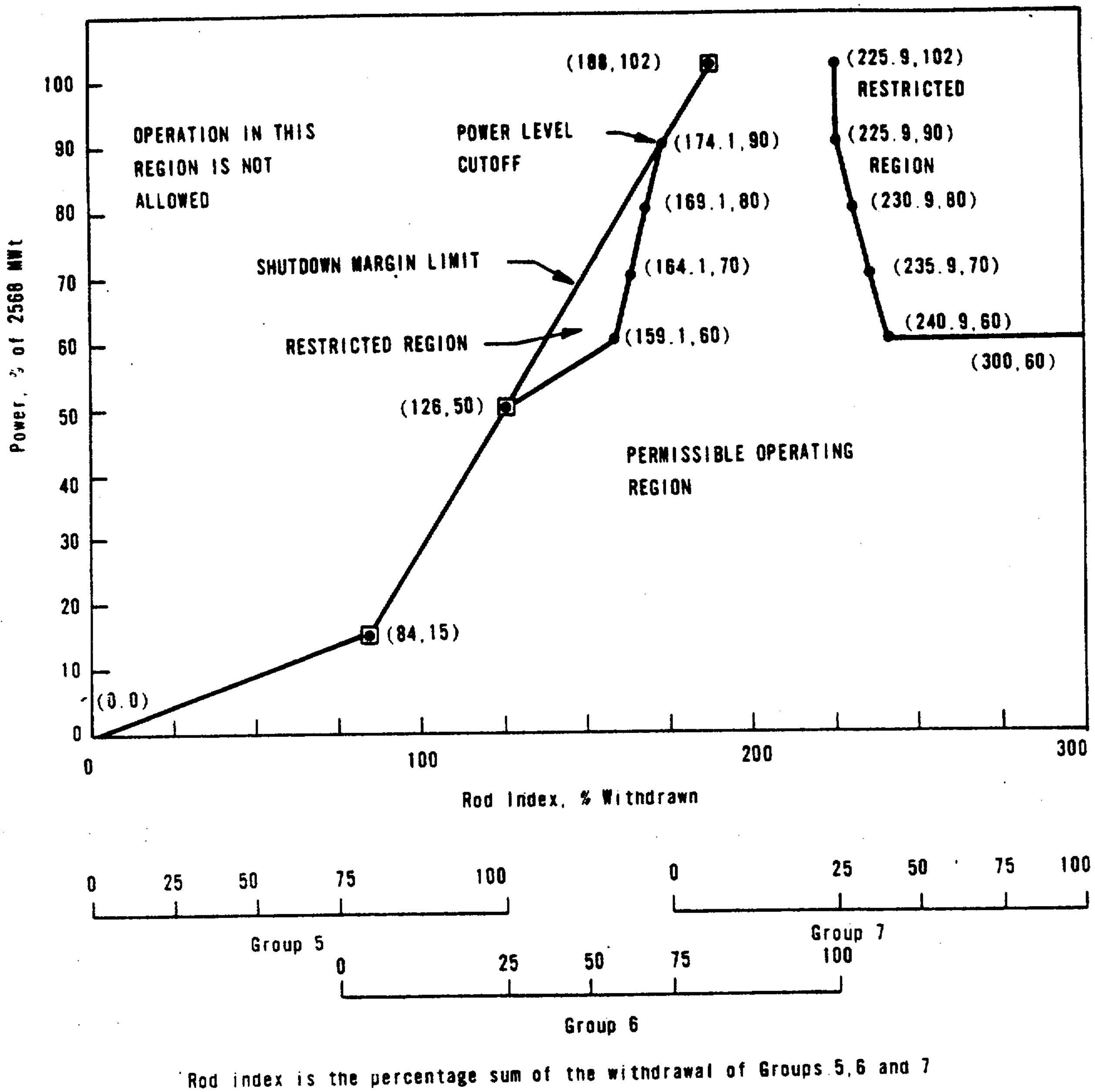
Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

ROD POSITION LIMITS FOR FOUR-PUMP OPERATION FROM 0 TO 100 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1A1

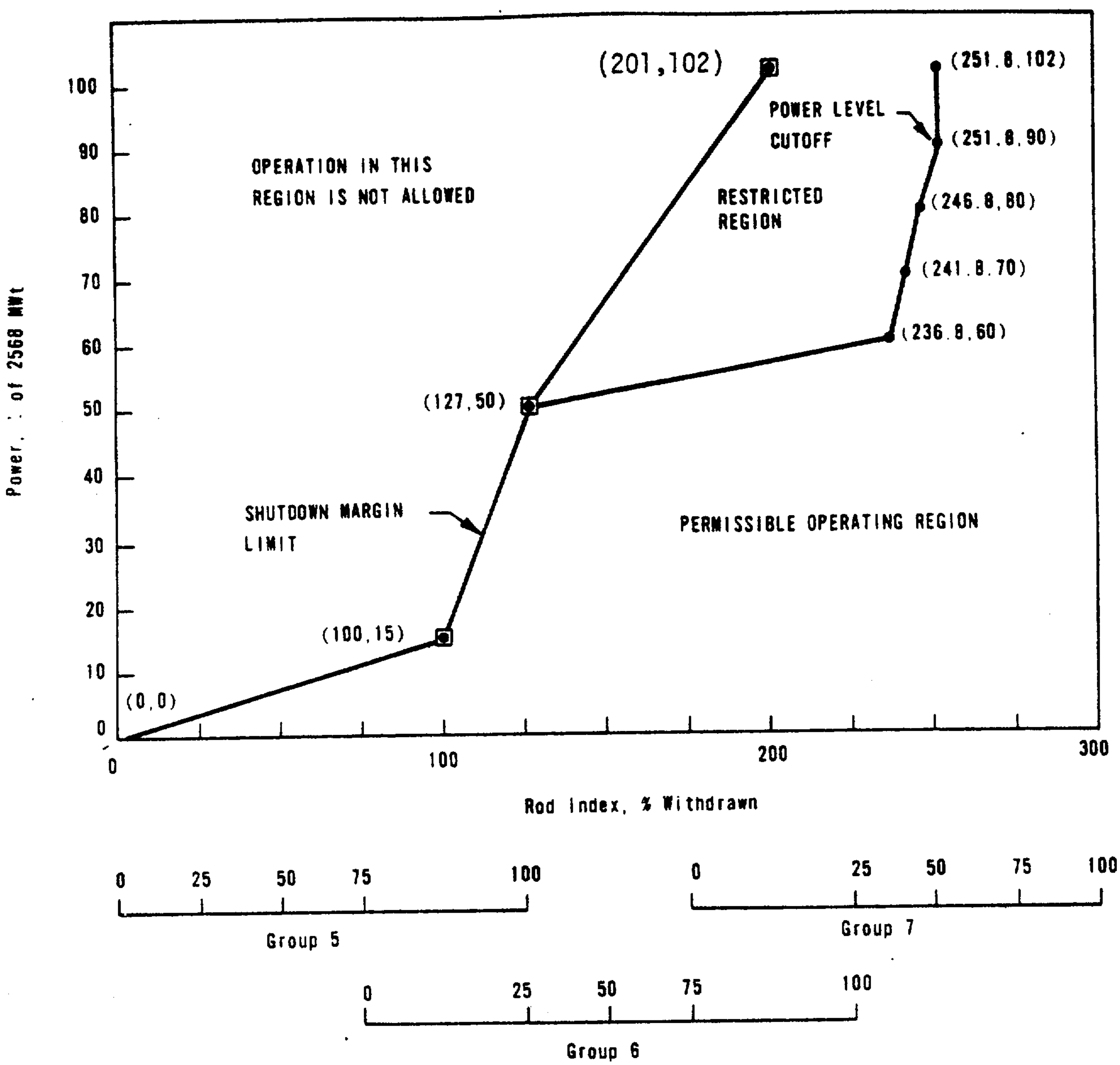
Amendments 47, 47 & 44



ROD POSITION FOR FOUR-PUMP OPERATION FROM 100 (±10) TO 235 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1A2



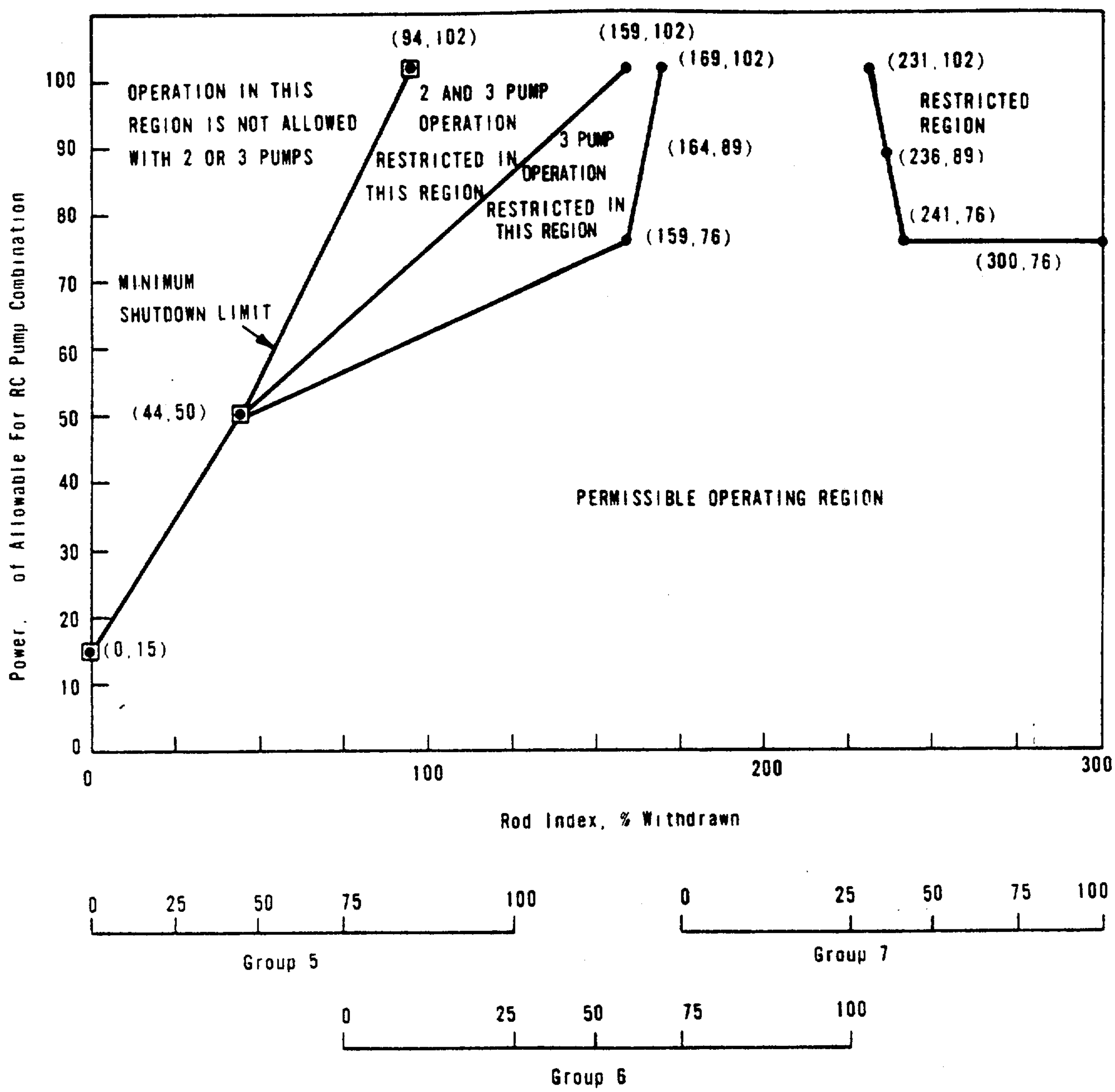
Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

ROD POSITION LIMITS FOR FOUR-PUMP OPERATION AFTER 235 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1A3

Amendments 47, 47 & 44

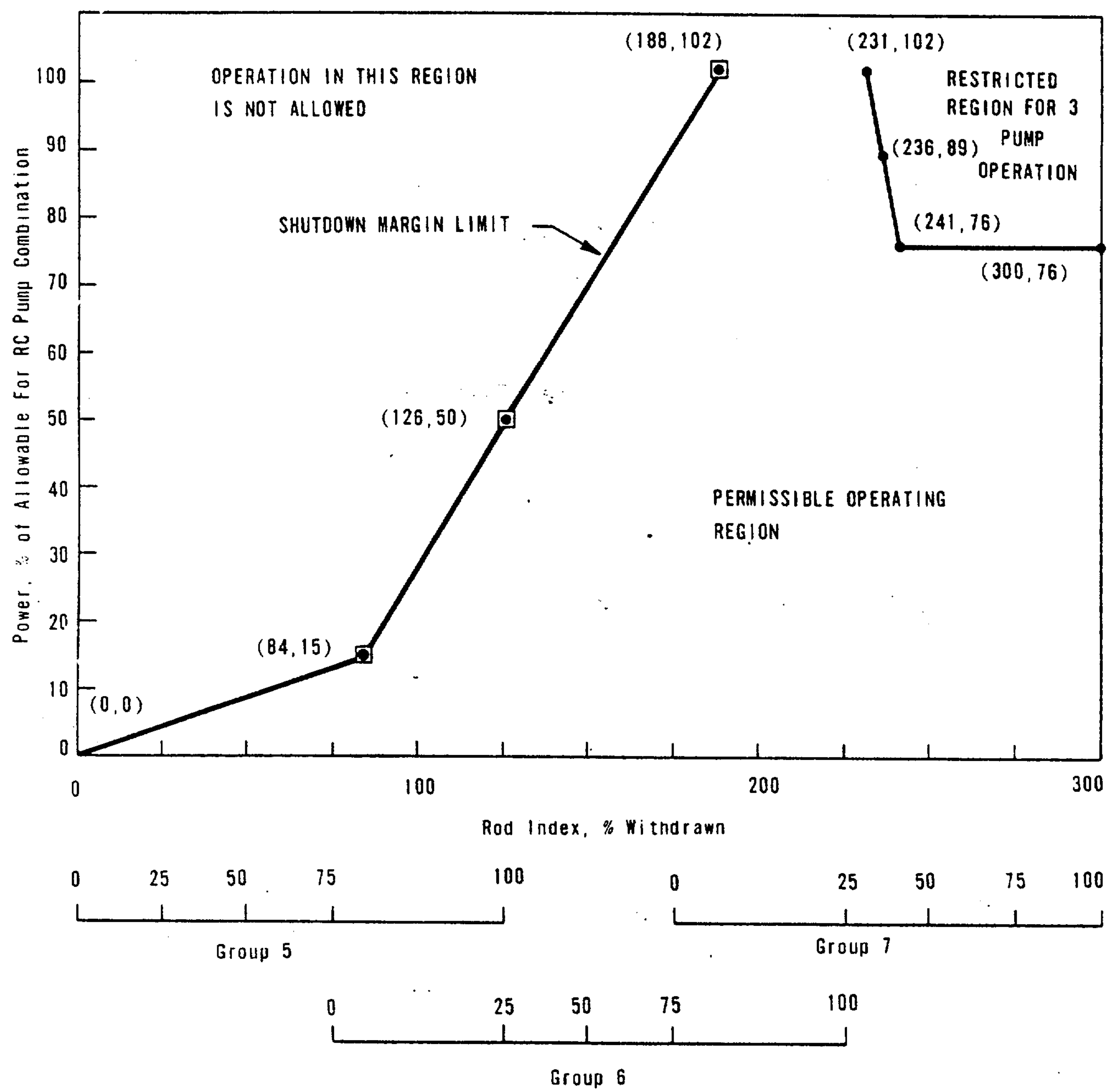


Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION FROM
0 TO 100 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

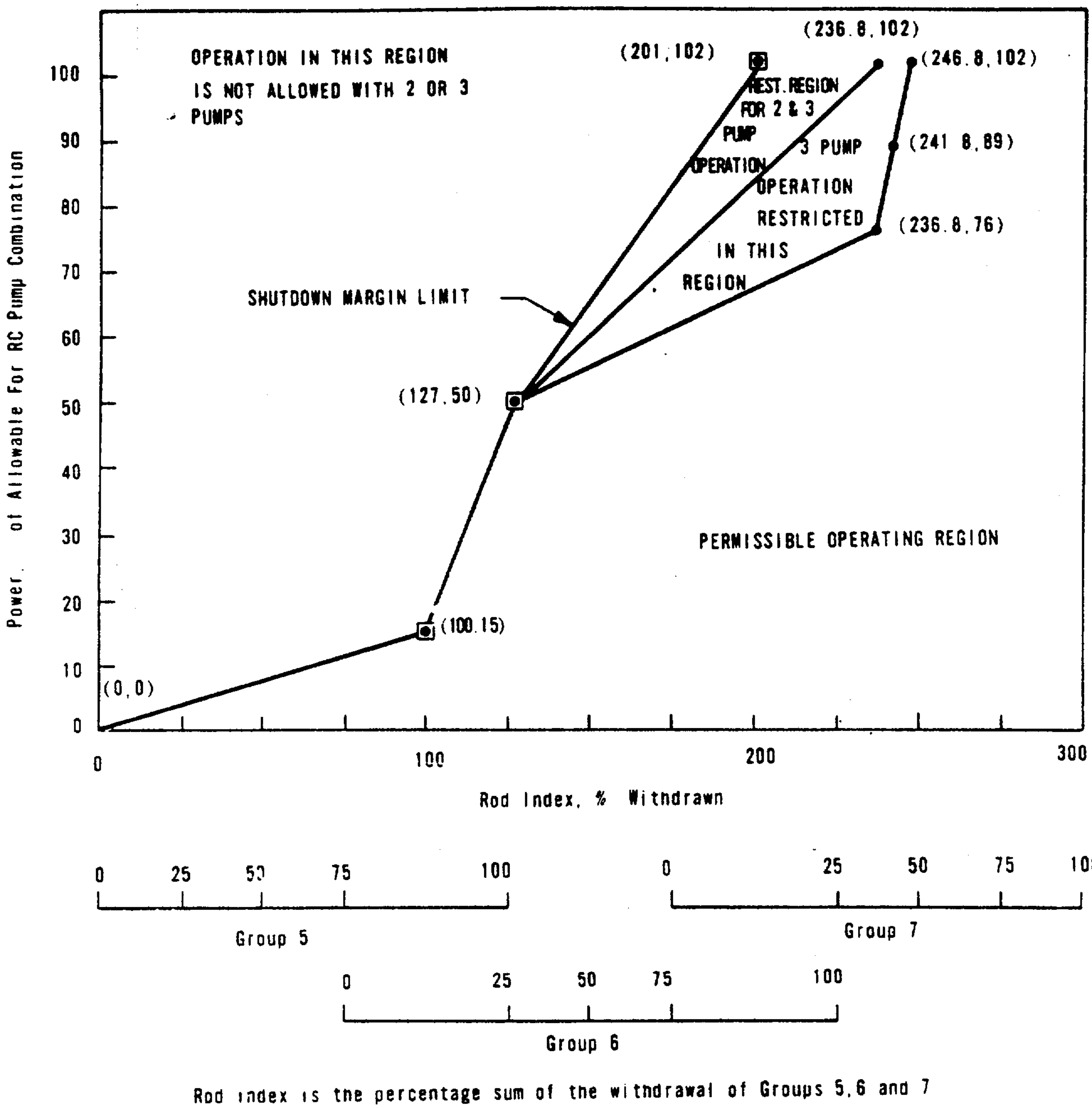
Figure 3.5.2-2A1



ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION FROM
100 (±10) TO 235 (±10), EFPD
UNIT 1

OCONEE NUCLEAR STATION

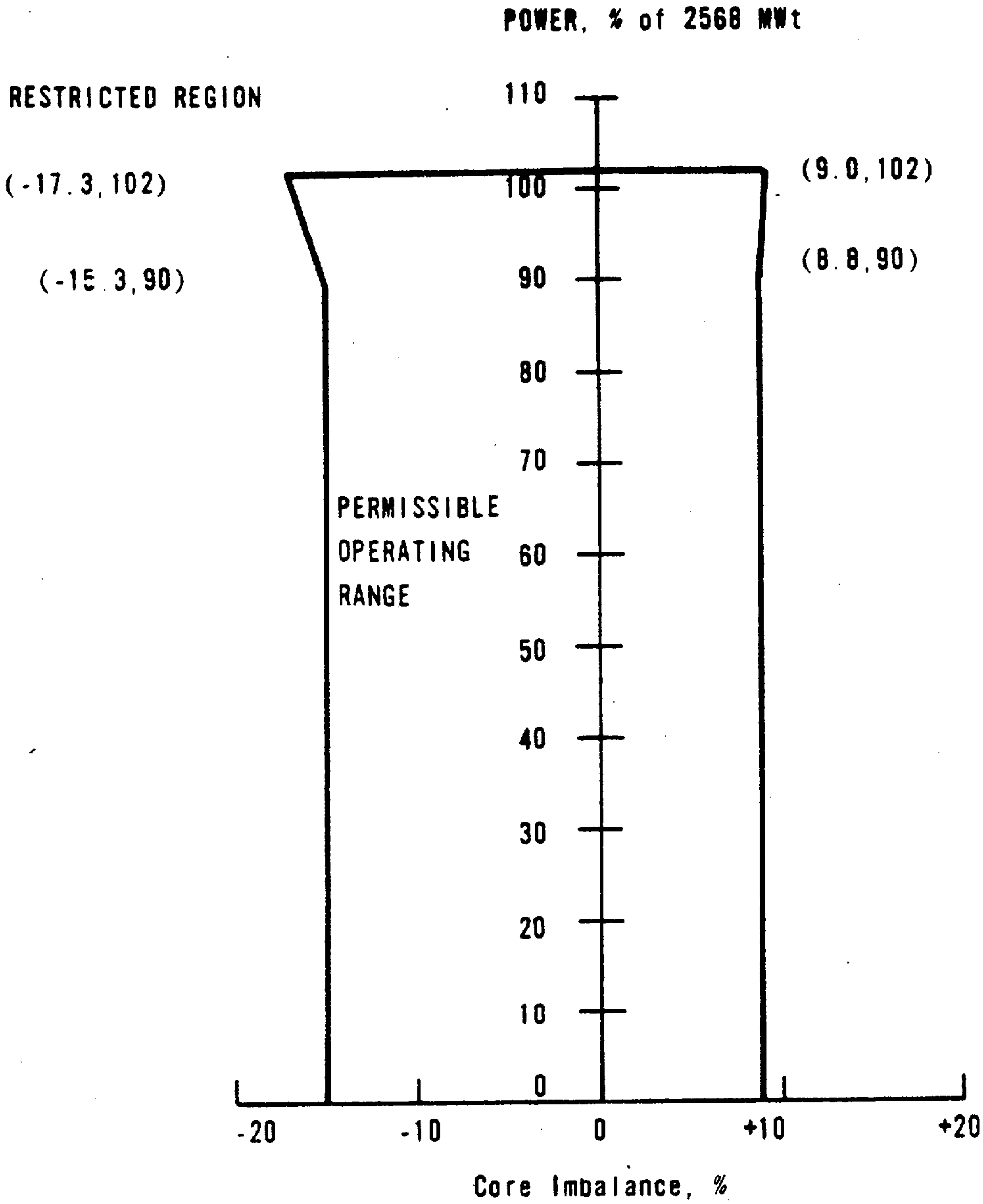
Figure 3.5.2-2A2



ROD POSITION LIMITS FOR TWO- AND THREE-PUMP OPERATION AFTER 235 (±10) EFPD, UNIT 1

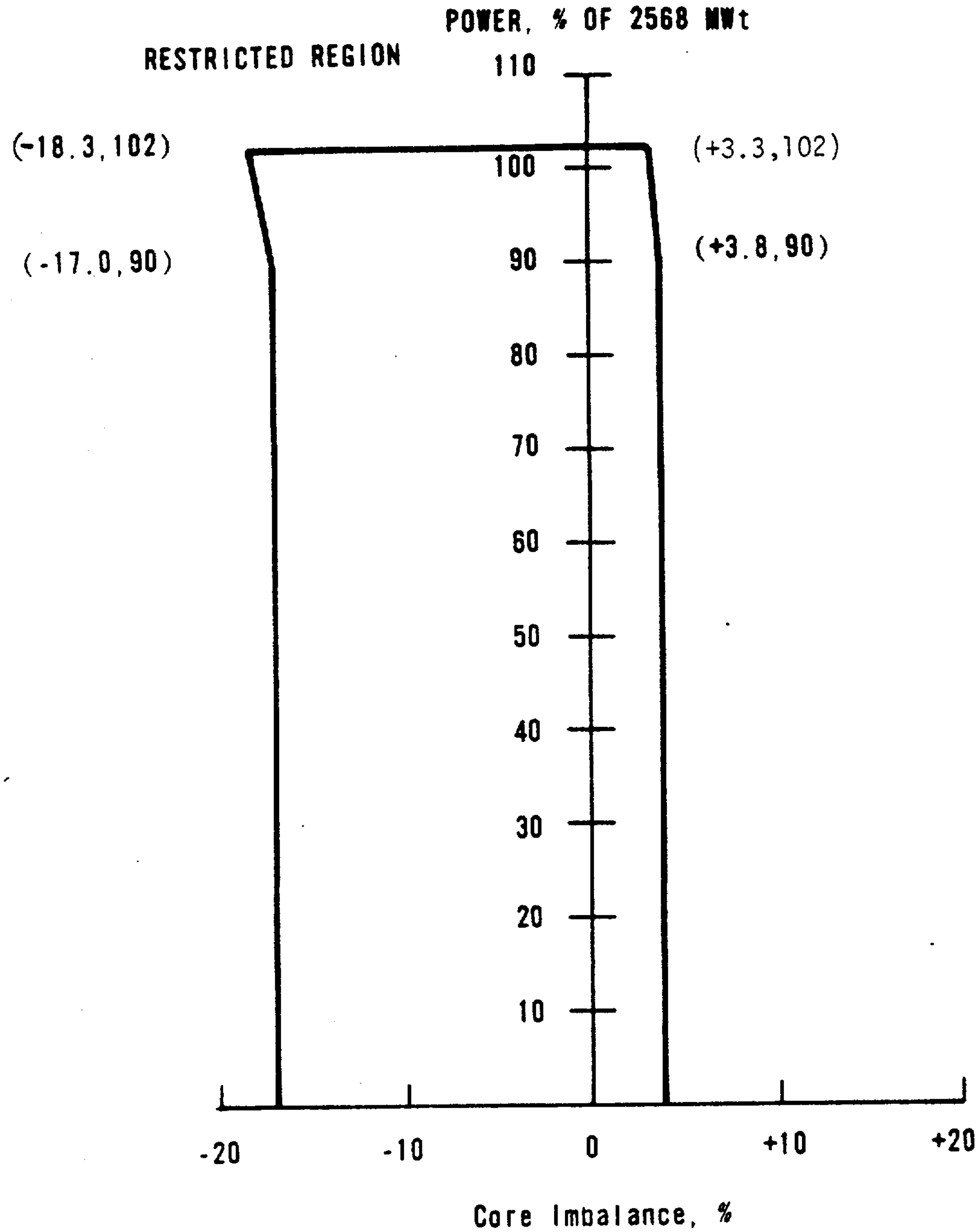
OCONEE NUCLEAR STATION

Figure 3.5.2-2A3



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 100 (±10) EFPD, UNIT 1

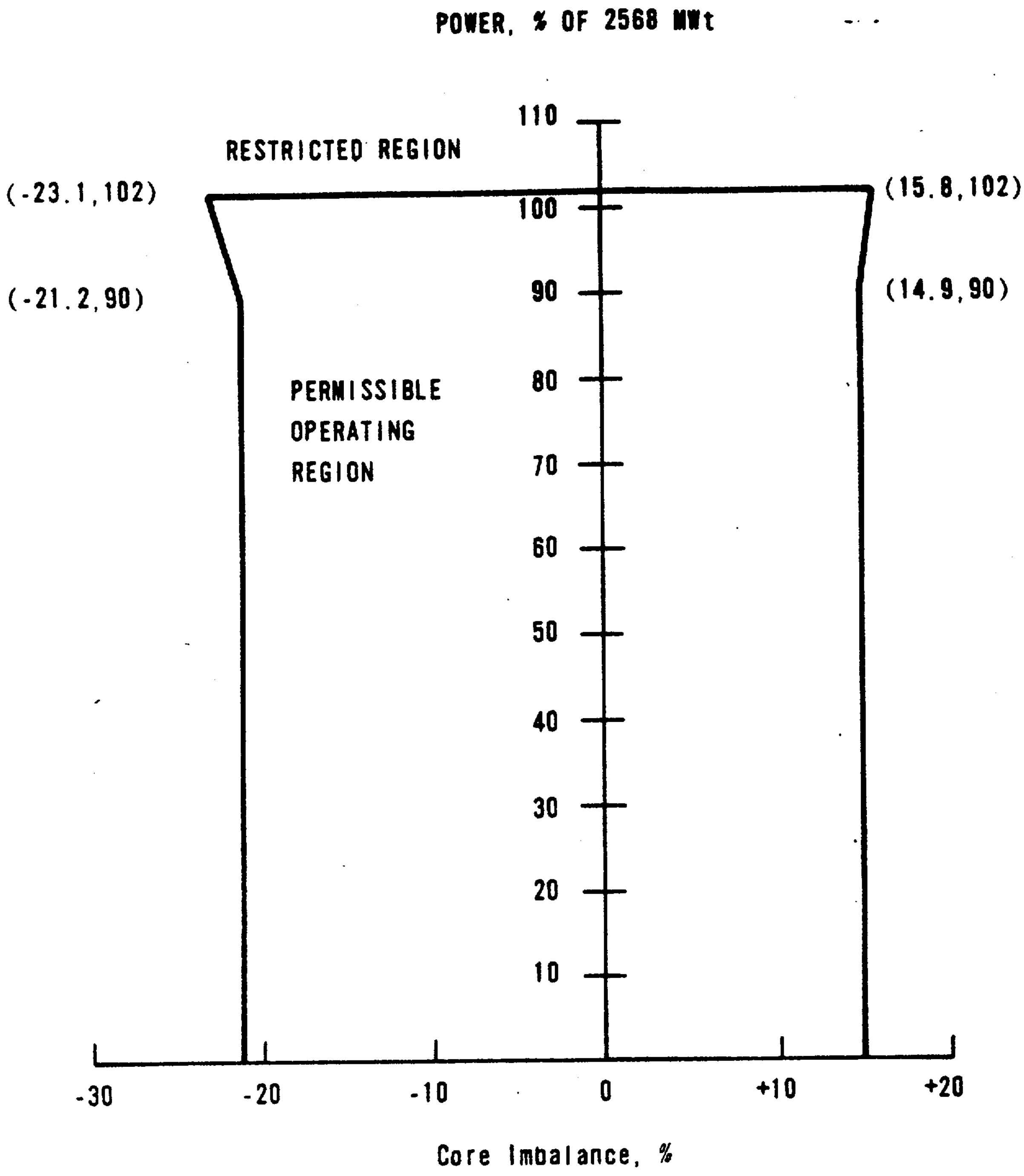
OCONEE NUCLEAR STATION
Figure 3.5.2-3A1



OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 100 (±10) TO 235 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

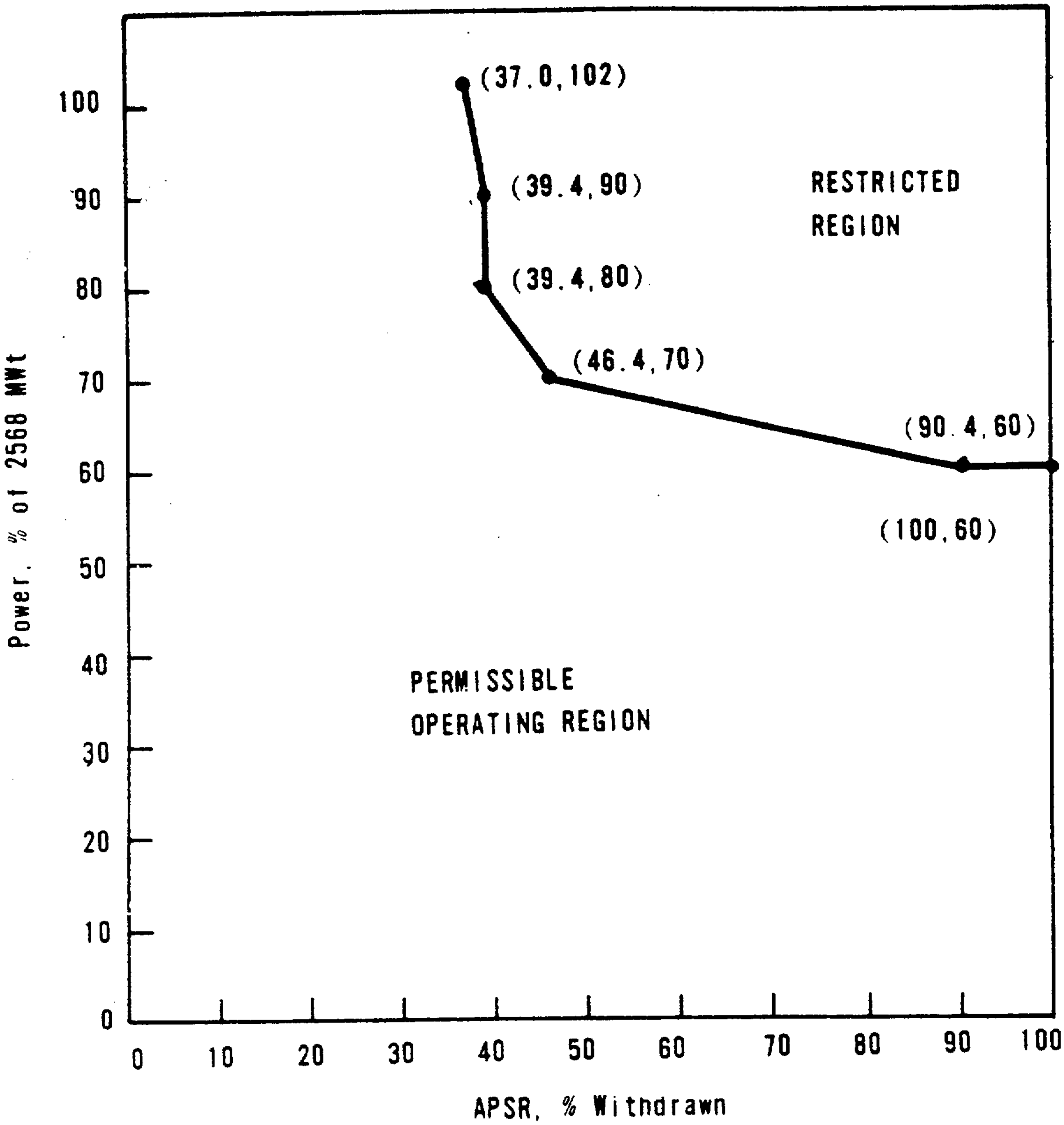
Figure 3.5.2-3A2



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION AFTER
235 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

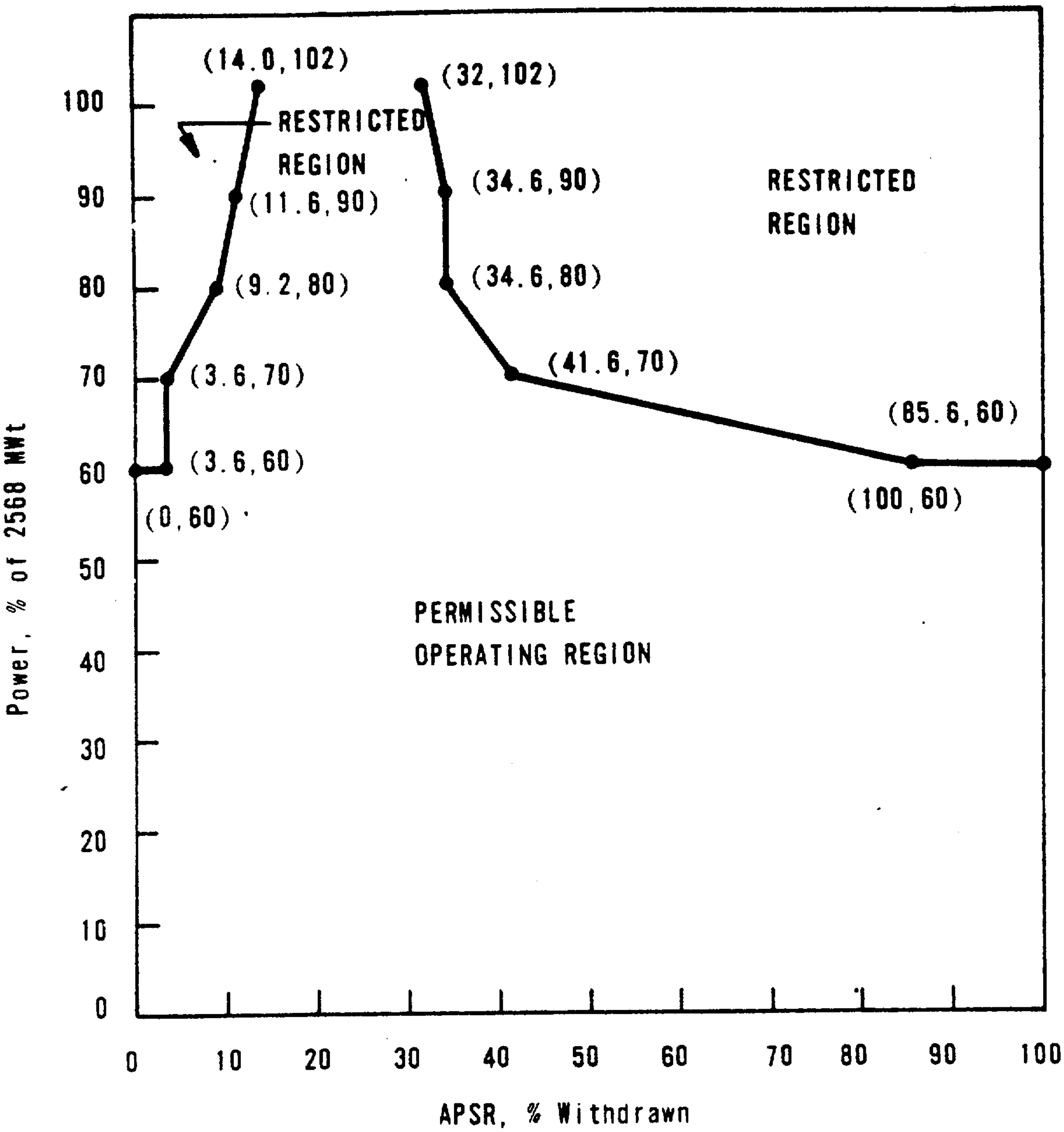
Figure 3.5.2-3A3



APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 (±10) EFPD, UNIT 1

OCONEE NUCLEAR STATION

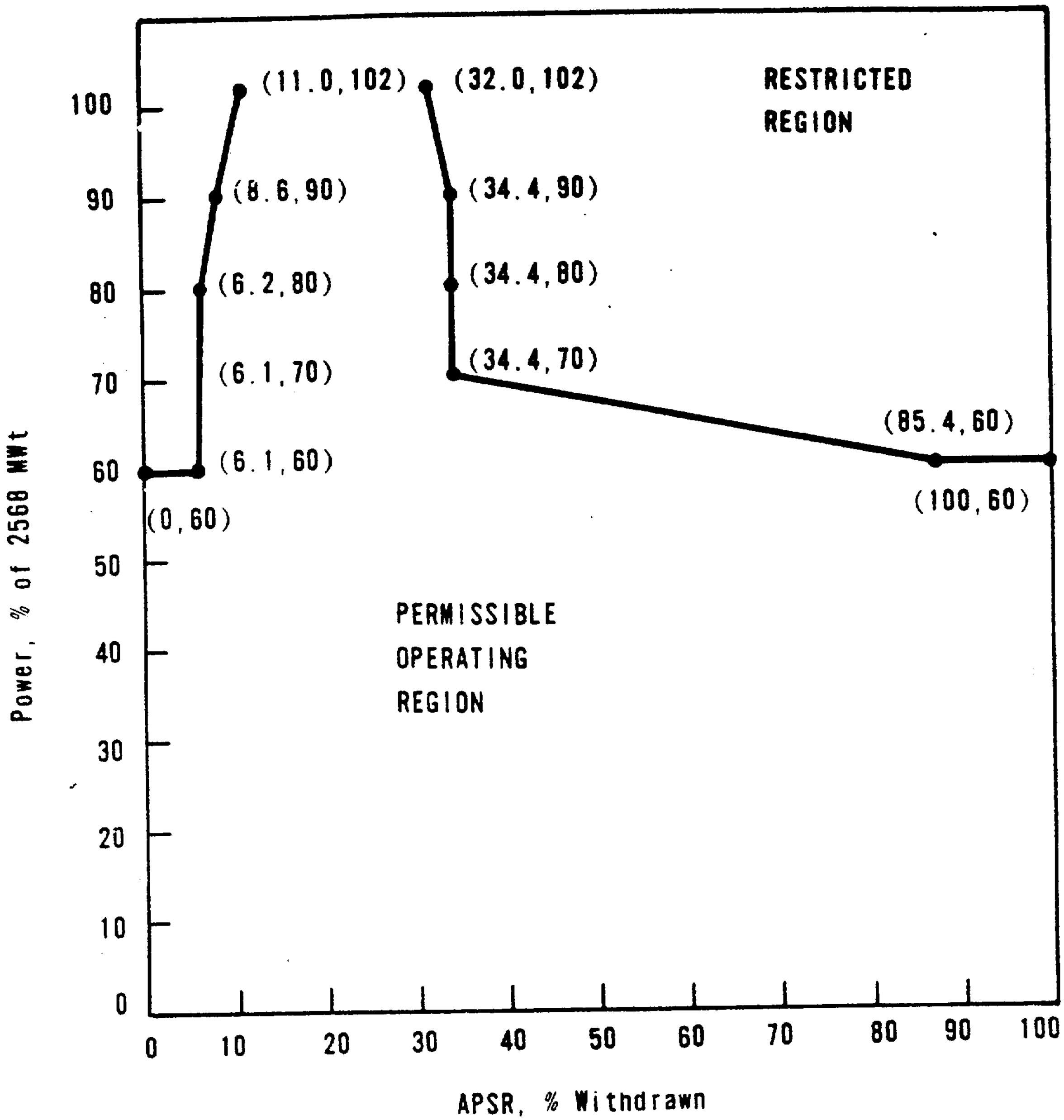
Figure 3.5.2-4A1



APSR POSITION LIMITS FOR
OPERATION FROM 100 (±10)
TO 235 (±10) EFPD, UNIT 1

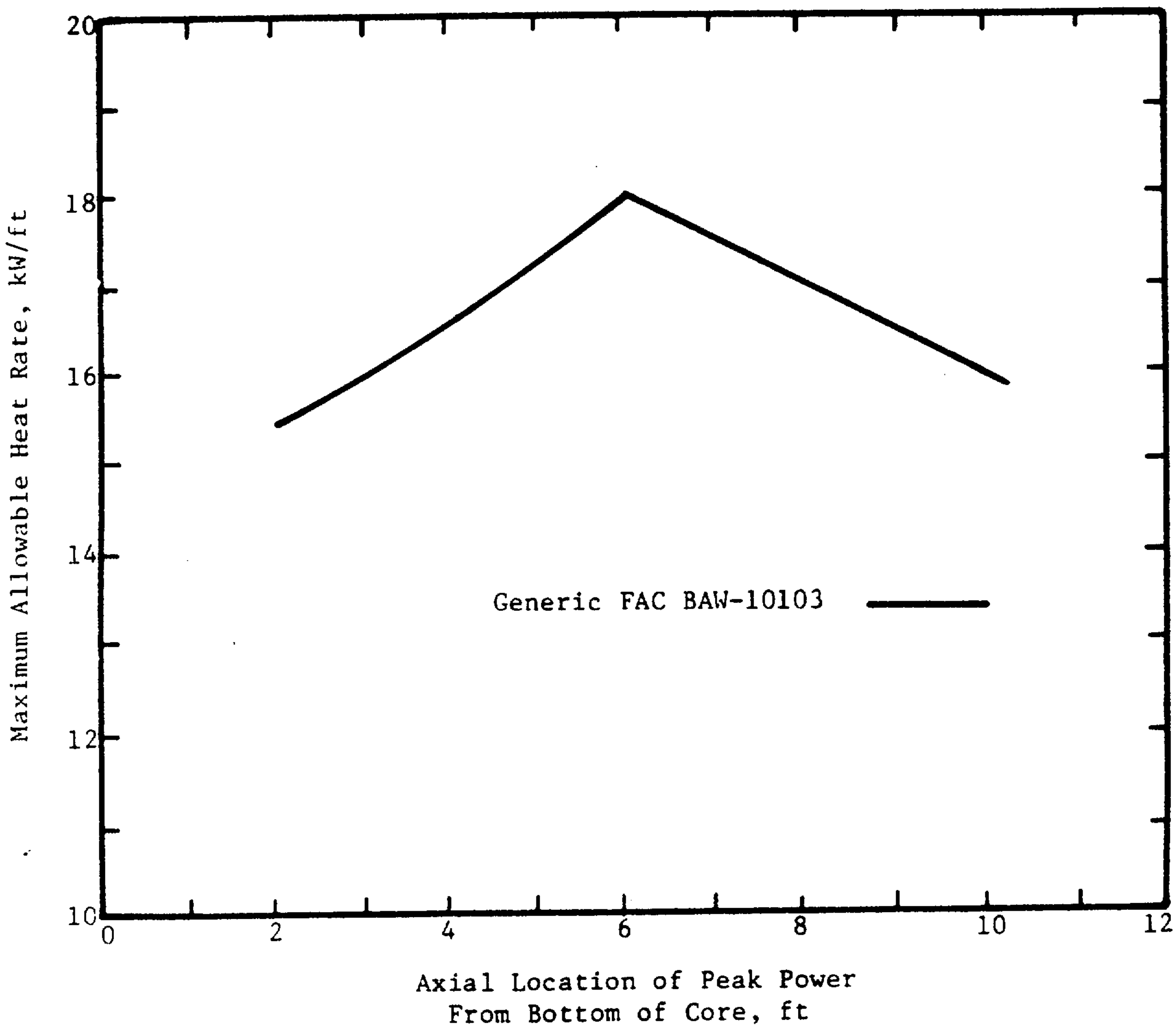
OCONEE NUCLEAR STATION

Figure 3.5.2-4A2



APSR POSITION LIMITS FOR
OPERATION AFTER 235 (±10)
EFPD, UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-4A3



LOCA-LIMITED MAXIMUM ALLOWABL
LINEAR HEAT

OCONEE NUCLEAR STATION

Figure 3.5.2-5

Amendments 47, 47 & 44

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

(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.

5.3 REACTOR

Specification

5.3.1 Reactor Core

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

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

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 3-46. The full-length CRA and the APSR shall conform to the design described in the FSAR or reload report.

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

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements.(3)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F.(4)

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

REFERENCES

(1) FSAR Section 3.2.1

(2) FSAR Section 3.2.2

(3) FSAR Section 4.1.3

(4) FSAR Section 4.1.2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO LICENSE NO. DPR-38

AMENDMENT NO. 47 TO LICENSE NO. DPR-47

AMENDMENT NO. 44 TO LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

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

Introduction

By letter dated March 30, 1977(1) and as supplemented June 21,(2) August 23,(3) September 8 (13) and 14,(14) 1977, Duke Power Company (the licensee) requested changes to the Technical Specifications appended to the Oconee Unit 1 Operating License for operation as reloaded for Cycle 4.

Evaluation

The Oconee Unit 1 reactor core consists of 177 fuel assemblies. The reload for Cycle 4 will involve the removal of all 56 Batch 3 fuel assemblies and 4 of the Batch 4 fuel assemblies, and relocation of the residual Batch 4 and Batch 5 fuel assemblies. The removed fuel will be replaced by 56 new Batch 6 fuel assemblies and 4 Batch 2 fuel assemblies. The new assemblies will occupy the periphery of the core.

The licensee's reload analyses and Technical Specification changes submitted by letter dated March 30, 1977, were based on an originally planned 292 effective full power days (EFPD) of Oconee Unit 1 Cycle 3 operation. The licensee, however, advised us by letter dated July 27, 1977,(4) that Cycle 3 operation was being extended to 312 EFPD. As a result, the burnup distribution in the Batch 4 and 5 fuel assemblies, which are to remain in the core for Cycle 4 operation, will be different from that assumed in the original reload analysis. Based on a reanalysis of the new burnup distribution the licensee submitted revisions to the reload report and Technical Specifications.(3)

Fuel Mechanical Design

Tables 4-1 and 4-2 of Reference 5 summarize the reload core fuel assembly parameters. The Batch 6, 15 x 15 (Mark B-4), fuel assembly design and the Batch 2, 15 x 15 (Mark B-2), fuel assembly design have been previously reviewed and accepted by us for use in Oconee Unit 1.(6) Also, these types of assemblies are currently operating in Oconee Unit 1. The reload assemblies, therefore, do not represent any unreviewed change in mechanical design from the reference cycle.

The reload fuel assemblies (Batch 6) are the same as the residual fuel assemblies except for minor design modifications to the spacer grid corner cells, which reduce spacergrid interation during handling. Dynamic impact testing has shown this design to have a higher seismic capability than the previous design.(1) The current design and the reload design meet all requirements of the fuel assembly design and are acceptable.

These mechanical design variations have been taken into account in the various mechanical analyses. The Batch 4 fuel is generally limiting, because of its relatively low initial fuel pellet density, and previous incore exposure. The results of these analyses have shown that the mechanical design differences between Cycle 3 and Cycle 4 are of negligible effect and are acceptable.

Fuel rod cladding creep collapse analyses were performed for the fuel batches which will be present in the Cycle 4 core. The calculational methods, assumptions, and data have been previously reviewed and approved by us.(7) The CROV computer code was used to calculate the time to fuel rod cladding creep collapse. The most restrictive power profiles, to which the once-burned and new fuel assemblies may be exposed, were used in the Batch 5, Batch 2, and Batch 6 analyses. The actual reactor operating history along with the most restrictive power histories for the forthcoming cycle were used in the analysis of the Batch 4 fuel. The fuel cladding material properties are the same as those used in the CROV code. The analysis assumed no fission gas production (maximum differential pressure), lower tolerance limit on cladding thickness, and upper tolerance limit on cladding ovality. Based on the analyses performed, the fuel rod design has been shown to meet the required design life limits for fuel cladding creep collapse and is, therefore, acceptable.

From the viewpoint of cladding stress and strain, Cycle 4 operation is acceptable. The cladding stress (creep stress due to differential pressure, thermal stress due to temperature gradient and bending stress due to axial loads and restraints) will not exceed the yield stress or ultimate strength of the cladding material. The Batch 4 fuel is most limiting with respect to stress, because of its irradiation history and lower fuel pellet density. The cladding strain for Cycle 4 operation is less than the generally used 1% plastic strain acceptance criteria. The strain analysis assumed maximum specification values for fuel pellet diameter, density, and burnup, and minimum specification tolerance on fuel cladding inside diameter. These assumptions conservatively represent the cladding strain. The Batch 4 fuel will again be limiting in the Cycle 4 core based on the cladding strain. Again this is because of its irradiation history and lower fuel pellet density.

The Batch 6 and Batch 2 fuel assemblies are not new in concept and do not use different component materials. The fuel assemblies for Cycle 4 operation will not exceed any design life limits. We conclude, therefore, that the fuel mechanical design for Cycle 4 operation is acceptable.

Fuel Thermal Design

The fuel thermal design analysis was conducted using the TAFY-3 computer code.(8) This analysis established heat flux limits to fuel centerline melt. The analysis considered the effect of a power spike from fuel pellet densification.(9) Modifications to the void probability, Fg, and size distribution, F_k, have been previously reviewed and approved by us for Oconee Unit 1 fuel thermal design analysis.(15) This analysis is based on the lower tolerance limit on fuel density and assumes isotropic diametral densification shrinkage and anisotropic axial shrinkage densification. These assumptions have been approved by us.(10)

During Cycle 4 operation, the highest relative assembly power levels occur in Batch 5 fuel assemblies. The fuel temperature analysis for Cycle 4 is based on limiting beginning-of-cycle (BOC) conditions (zero burnup) and conservative peaking factors. The analysis is performed to establish linear heat generation rates to preclude central fuel melting and stored energy limits for LOCA analyses. The thermal design analysis for the Batch 5 fuel assemblies thermal design analysis is bounding, and we conclude that the fuel thermal design for Oconee Unit 1 Cycle 4 is acceptable.

Nuclear Analysis

The reactor core physics parameters for Oconee Unit 1 Cycle 4 operation were calculated using a PDQ07 computer code. Since the core has not yet reached an equilibrium cycle, there were minor differences in the physics parameters between the Cycle 3 and Cycle 4 cores. For example, EOC Doppler and moderator coefficients change by less than 2% from Cycle 3 to Cycle 4. These changes are to be expected and are not significant.

Axial Power Shaping Rod (APSR) Testing

By letter dated June 21, 1977, a program was prepared to remove one of the axial power shaping rod (APSR) assemblies for destructive examination to obtain more information on the effects of irradiation on the material properties. Since this assembly has been exposed to three cycles of irradiation, it will be replaced with an APSR assembly with an equivalent poison worth. An evaluation of the nuclear, mechanical and thermal hydraulic considerations of this program have been conducted and it is concluded that safe operation of the reactor will not be adversely affected by this program .

In view of the above and the fact that startup tests (to be conducted prior to power operation) will verify that the significant aspects of the core performance are within the assumptions of the safety analysis, we find the licensee's nuclear analysis for Cycle 4 to be acceptable.

Thermal-Hydraulic Analysis

The major acceptance criteria which are used for the thermal-hydraulic design are specified in Standard Review Plan (SRP) 4.4. These criteria establish acceptable limits on departure from nucleate boiling (DNB). The thermal-hydraulic analysis for Oconee Unit 1 Cycle 4 reload was made using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the Cycle 4 core and are discussed below.

Reactor Coolant System Flow Rate

The reactor coolant flow rate was accurately measured during Cycle 1 operation at 108.6% of the system design flow. The licensee has proposed to take credit in the thermal-hydraulic analysis for this higher flow (as was done in the previous cycle).

The core configuration for Cycle 4 differs slightly from that of Cycle 3 in that the depleted batch 3 fuel removed at the end of Cycle 3 is the Mark B-2 fuel assembly design. Mark B-4 fuel assemblies exhibit a slightly lower resistance to flow than do the Mark B-2 assemblies, which have a revised end fitting design. This change has been considered in the Cycle 4 core flow distribution analysis. No credit has been taken for the increase in system flow that will result from the reduction in total core pressure drop.

Fuel Rod Bow

In the submittal dated March 30, 1977, the licensee summarized the method and results of the rod bow analysis. This rod bow analysis was performed with an as yet unapproved model. Therefore, the licensee was requested to provide an analysis with the NRC approved rod bow model or to show sufficient compensatory margin.

The licensee chose to show sufficient margin in order to offset the difference between models. The approved rod bow model requires a DNBR penalty of approximately 12% as compared to the unapproved rod bow model which has about a 6% DNBR penalty. The 6% difference in DNBR penalty will be accommodated by a change in the protective pump monitor trip function by tripping the reactor upon loss of one pump during four pump operation if the indicated reactor power is greater than 80% of full power.

The existing pump monitor trip function is set to trip the reactor upon loss of two pumps, and as such the analytic basis for the existing flux/flow trip setpoint is a two-pump coastdown. By adding the pump monitor setpoint trip on the indicated loss of one pump, the flux/flow trip setpoint analysis need only consider a one-pump coastdown while still providing the same protection for loss of two pumps at lower power. The proposed change provides DNBR margins of 30% for Oconee Unit 1, 34% for Oconee Unit 2, and 32% for Oconee Unit 3.

In the case of Oconee Unit 2 Cycle 3 and Oconee Unit 3 Cycle 3, the necessary DNBR margins for the flux/flow trip setpoints were demonstrated by taking credit for additional RC flow available over the thermal-hydraulic design flow. Because of the pump monitor trip on loss of one pump, these flow credits are not required.

In summary, a reactor coolant flow rate based on actual measured flow with uncertainties was used in the Oconee Unit 1 Cycle 4 thermal hydraulic analysis. The licensee has also assured us that there will be sufficient margin in the reactor protective trip function to compensate for the difference between the approved and the unapproved rod bow models. Based on our review, we find that the licensee has included appropriate conservatisms in its analysis and that the proposed Technical Specifications provide assurance that the criteria of SRP 4.4 will be met. Therefore, we conclude that the thermal hydraulic analyses as previously approved and discussed are acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by the licensee demonstrate that the Oconee FSAR analyses conservatively bound the predicted conditions of the Oconee Unit 1 Cycle 4 core and are, therefore, acceptable. Each FSAR accident analysis has been examined, with respect to changes in Cycle 4 parameters, to determine the effects of the reload and to ensure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those used in the Cycle 4 analysis. For each accident of the FSAR, a discussion and the key parameters from the FSAR and Cycle 4 was provided with the accident discussion to show that the initial conditions of the transient are bounded by the FSAR analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the Oconee Unit 2 fuel densification report[11] Since Cycle 4 reload fuel assemblies contain fuel rods with theoretical density higher than those considered there, the conclusions derived in that report are valid for Oconee Unit 1 Cycle 4. Calculational techniques and methods for Cycle 4 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for the reload report. The dose considerations in the FSAR are based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

A review of the ECCS U-baffle pressure drop error has been performed and documented in reference 12. The review considered a reanalysis of the reactor coolant system pressure loss characteristics and the effects and ECCS performance. The review found the current ECCS performance analysis acceptable for all three Oconee units. Reference 12 also found that a new surveillance testing program of the reactor internals vent valves is acceptable for all three Oconee units. The review considered the impact of these changes on ECCS performance and the adequacy of the surveillance techniques.

Startup Tests

A startup program will be conducted to verify that the core performance is within the assumptions of the safety analyses and provide the necessary data for continued plant operation. The startup test program is similar to that previously approved for Cycle 3 operation.[6] Additionally, the program was discussed with the licensee for clarification of control rod worth and power distribution measurements and comparison to predicted values. These measurements and comparisons will be performed by the licensee. Within 90 days following completion of physics testing the licensee also will provide a summary of the test program results. This startup test program is acceptable.

Environmental Consideration

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 4, 1977

REFERENCES

1. Letter from W. O. Parker, Jr., (Duke Power Company) to B. C. Rusche (NRC) dated March 30, 1977.

2. Letter from W. O. Parker,Jr., (Duke Power Company) to E. G. Case (NRC) dated June 21, 1977, Re: Oconee Unit 1 Docket No. 50-269

3. Letter from W. O. Parker, Jr., (Duke Power Company) to B. C. Rusche, dated August 23, 1977.

4. Letter from W. O. Parker, Jr., (Duke Power Company) to E. G. Case (NRC) dated July 27, 1977.

5. "Oconee Unit 1, Cycle 4-Reload Report", BAW-1447, March 1977.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 20, 20 and 17 to Facility License Nos. DPR-38, DPR-47 and DPR-55, Duke Power Company, Oconee Nuclear Station, Unit Nos. 1, 2 and 3, March 25, 1976.

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

8. TAFY-Fuel Pin Temperature and Gas Analysis," BAW-10044, May 1972.

9. "Fuel Densification Report", BAW-10055, Revision 1, June 1973.

10. "Technical Report on Densification of Babcock & Wilcox Reactor Fuels", NRR, July 6, 1973.

11. Oconee 2 Fuel Densification Report, BAW-1395, Babcock & Wilcox, June 1973.

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

13. Letter from W. O. Parker, Jr., (Duke Power Company) to E. G. Case (NRC) dated September 8, 1977.

14. Letter from W. O. Parker, Jr., (Duke Power Company) to E.G. Case (NRC) dated September 14, 1977.

15. Letter from S. A. Varga (NRC) to J. H. Taylor (B&W), Subject: Evaluation of BAW-10083P, Revision 1, dated May 16, 1977.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 & 3

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

Introduction

By letter dated September 24, 1977, the licensee informed the NRC of the discovery of an apparently different steam generator tube degradation phenomenon at Oconee Nuclear Station Unit 1. This phenomenon was discovered during the steam generator tube inservice inspection completed in September 1977. Results of this inspection were not included in Duke Power's Safety Assessment Report, submitted in August 1977, which is currently under review. The new degradation phenomenon is described as a localized erosion or cavitation mechanism resulting in a tube wall thinning. This thinning has been detected by eddy current measurements and a total of eighty-nine tubes in Oconee Unit 1 steam generators "1A" and "1B" have been affected by this phenomenon.

Discussion

Inspection Results

The eddy current inspection initially performed during this outage in each steam generator included all of the tubes adjacent to the open tube lane, a 3.0 percent sample randomly selected throughout the steam generator and a 2.5 percent sample randomly selected within the peripheral region of the steam generator. As a result of this inspection five defective tubes (tubes with eddy current indications greater than 40% wall thinning) were identified in the "1B" steam generator and three defective tubes were identified in the "1A" steam generator. These tubes were located in the peripheral region away from the open tube lane and were predominately at the 14th support plate elevation. Based on these results a second tube sample consisting of 3 percent of the tubes in each steam generator was inspected. This second sample was concentrated in the areas around the defects and in the peripheral region. The results of this sample inspection revealed one additional defective tube in the "1A" steam generator and 10 additional defective tubes in the "1B" steam generator. For the "1B" steam generator the majority of the additional defects were located in the half of the steam generator opposite the open tube lane, consisting of quadrants WX and XY as identified in the licensee's submittal.

The steam outlet lines leave the steam generator shell from these quadrants. A third 6 percent sample was examined in that region of the "1B" steam generator which detected 10 more defective tubes.

In an effort to validate that the problem regions of the steam generator had been identified, a third 6 percent sample was inspected in the periphery of the WX-XY quadrants in steam generator "1A" and a fourth 6 percent sample was examined in the periphery on the open tube lane side of steam generator "1B". These samples revealed one defective tube in steam generator "1A" and two additional defective tubes in steam generator "1B".

Due to the large number of defective eddy current indications, some of which were interpreted as 90% - 100% wall thinning, the licensee considered it essential to obtain tube samples for direct examination to help identify the degradation phenomenon. Therefore, the licensee removed two peripheral tubes from steam generator "1B" for visual and laboratory examination. The first tube removed had an eddy current indication just above the 14th support plate which interpreted as 45% - 50% wall thinning. Visual inspection of this tube revealed an eroded slot area about 1/8 inch long and 1/16 inch wide and approximately .020 inch deep. The second tube removed had an eddy current indication interpreted to be 90% wall thinning just above the 14th support plate. Visual inspection of this tube revealed a shallower erosion wear spot covering substantially more tube surface area. This spot was about 1 inch long and 0.3 inches wide. As a result of these observations, the licensee became aware of a form of tube degradation that is of a different nature than that previously observed in Oconee steam generators.

In view of this, the licensee conducted eddy current testing on an additional 11% of the tubes in steam generator "1B". This 11% sample included all of the tubes in the periphery of quadrants WX and XY and one-third of the periphery tubes in the open tube lane side of the steam generator. This inspection included examination of four tubes with 14th tube support plate indications which had been eddy current tested four months previously. Three of these four tubes showed no change in degradation size while for the fourth tube the eddy current test indicated that tube thinning had gone from less than 20 percent to 35 percent of wall thickness.

In total, the licensee has eddy current inspected 33% of the tubes in steam generator "1B" and 16% of the tubes in steam generator "1A". Of the approximately 7,350 tubes thus inspected in both steam generators, 32 were classified as defective tubes and 44 degraded (greater than 20% and less than 40% wall thinning) tubes were found in steam generator "1B"; in steam generator "1A" the totals were 5 defective tubes and 8 degraded. All of the 37 tubes identified as defective were plugged.

Burst Test Data

The licensee's submittal also included the results regarding B&W tube rupture test. B&W has demonstrated that a tube with a flat defect 70 percent through the wall will not fail under 5,000 PSI internal pressure. This is greater than twice the pressure which would occur during a postulated main steam line break accident.

Evaluation

The earlier problem of fatigue related circumferential cracking in tubes along the missing tube lane at the upper support plate locations has been carefully followed by us for some time. The leaks experienced so far have been quite small and have entailed orderly planned shutdown to investigate the leak and to remove the leaking tube from service by plugging. The licensee has recently, at our request, provided an extensive safety assessment of the effect of such leaks on reactor functions as part of a proposed revision of technical specifications relating to steam generator inspection and integrity protection. This is currently under review by us.

We have reviewed the information submitted by the licensee on September 24 regarding the newly identified steam generator tube degradation phenomenon. The licensee has completed a comprehensive eddy current examination of both Oconee Unit 1 steam generators. This eddy current inspection program began with a broad sampling plan which was expanded when the additional degradation phenomenon was identified and continued until the problem areas in the steam generators were identified and thoroughly inspected. Based on our evaluation we conclude that the eddy current testing program conducted has been sufficiently extensive to identify areas of the steam generator where there is a high probability of tubes being affected by this cavitation or erosion mechanism. Furthermore , the 100% eddy current examination performed in the high probability areas (periphery of quadrants WX-XY) in steam generator "1B" and additional eddy current sampling in lower probability areas in steam generator "1B" and steam generator "1A" provide sufficient confidence that the defective tubes have been identified.

The NRC has also reviewed photographic and measurements results of the visual examinations of the two tubes removed from steam generator "1B". The photographs of the tube defect locations give the appearance of a cavitation and erosion phenomenon consistent with the mechanism suggested by Babcock and Wilcox. Due to the nature of this phenomenon and experience in non-nuclear cases of this type of tube erosion a high degradation rate is not expected. This view is reinforced by examining the data collected on the four tubes that previously had eddy current indications when inspected in May and were reinspected during this latest inspection. Three of these four tubes indicated no further degradation rate while the fourth showed an approximate 15% degradation increase in four months. The effect of this type of degradation on steam generator tubes will be very similar to the wastage type of degradation previously observed in recirculating type of steam generators. Experience has shown that this type of degradation has not lead to catastrophic tube failure but rather has resulted in a leak before break situation. This type of gradual degradation is predictable and results, at worst, in an orderly plant shutdown with no danger to the public health and safety.

Moreover, from experience with wastage corrosion we would expect that a detectable leak will penetrate the tube wall before general tube wall thinning reaches a level at which the tube would be incapable of withstanding loads imposed by the full range of normal operating and accident conditions. As a result we believe that it is quite unlikely that a significant number of tubes (5 to 10) could reach a level of thinning at which they would fail in the event of an MSLB or LOCA, without prior detection by leakage in at least one tube. Because of the importance of leakage detection in assuring steam generator integrity, particularly since we appear to be dealing with a new phenomenon for which rate data is not well developed, it is very important for the licensee to continue its program of rapid repair of any detectable leakage and consultation with the NRC staff in the event such leakage is detected. This enables the staff, on the basis of the information available , to determine whether any additional investigation is required.

Based on our previous experience with similar forms of tube degradation, B&W's burst test data, and on the continuation of licensee's program of leakage detection and staff consultation, we believe that there is reasonable assurance that there will be no significant reduction in overall steam generator integrity resulting from the new erosion phenomenon in the period until the next inspection. Nevertheless, careful investigation of this dedradation phenomenon should continue in order to expand the preliminary data reported above. In this respect the NRC has requested and the licensee has committed to the following:

1. Information will be provided in a subsequent status report on the metallurgical examination conducted on removed tubes 43/108 and 83/117. This information is expected to be available by December 15, 1977.

2. Evaluations will be performed to evaluate a plugging limit criteria for defective tubes.

3. An attempt will be made to develop an inservice inspection calibration standard which will permit a more realistic, less conservative evaluation of large-area, shallow defects.

4. An attempt will be made to determine the rate of growth, if any, of indications at the 14th support plate at future Oconee 1 outages.

5. At the next Oconee 1 outage, additional peripheral tubes will be examined consistent with critical path scheduling.

6. Technical Specifications concerning inservice inspection of steam generator tubing will be reevaluated, and resubmitted if necessary, to incorporate the most recent experience.

7. Information will be provided in the near future concerning the visual examination of previously leaking, stabilized tube 114/109.

We conclude that the efforts represented by these commitments constitute a responsible and necessary approach to further understanding this steam generator tube degradation phenomenon and assessing its long term significance for the safe operation of these steam generators. Until the licensee completes the above investigation, we are adding the following Technical Specification to Unit 1

> If at any time, the leakage through the Oconee Unit 1 steam generator tubes equals or exceeds 0.3 gpm, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours. If the leakage is less than 0.3 gpm, an assessment shall be made whether operations may be continued safely or the plant should be shutdown. In either case, the NRC shall be notified in accordance with Section 6.6.2.1.

The 0.3 gpm is consistent with the limits imposed on other facilities with tube degradation problems.

Based on the above evaluation and commitments we conclude that Oconee Nuclear Station, Unit 1 is safe for continued operations. Progress in the licensee's investigation of this tube degradation phenomenon as well as performance of the Oconee Nuclear Station steam generators will continue to be under close observation by the NRC staff, and appropriate actions will be taken in the event of any unexpected developments not bounded by the above evaluation.

For this reason we have concluded that, with respect to newly identified tube erosion phenomenon, continued operation of the facility can be authorized under the conditions discussed above without significant reduction in overall steam generator safety margin.

Environmental Consideration

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 4, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 47 , 47 and 44 to Facility Operating License Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company which revised the Technical Specifications for operation of the Oconee Nuclear Station, Unit Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to establish operating limits for Unit 1 cycle 4 operation and tighten leakage limits through the Steam Generator tubes.

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. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement 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 March 30, 1977, as supplemented June 21, August 23, September 8, 14 and 24, 1977, (2) Amendment Nos. 47, 47 and 44 to License 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 4th day of October 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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