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July 29, 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 Amendments Nos. 45, 45, and 42 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated May 6, 1977, as supplemented June 21, and July 11, 1977, and your request dated March 1, 1977, as supplemented May 5, 1977.

These amendments revise the Technical Specifications (1) to establish operating limits for Unit 2 Cycle 3 operation and (2) to establish requirements for testing reactor core internal vent valves.

Copies of the Safety Evaluation 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. 45 to
License No. DPR-38
2. Amendment No. 45 to
License No. DPR-47
3. Amendment No. 42 to
License No. DPR-55
4. Safety Evaluation
5. Notice of Issuance

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SURNAME ➤	SSheppard	DNeighbors		ASchwencer		
DATE ➤	7/ /77	7/ /77	7/ /77	7/ /77		

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

cc: Mr. William L. Porter
Duke Power Company
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Oconee Public Library
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Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
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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.45
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 May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, 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: July 29, 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. 45
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 May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment 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. 45 , 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: July 29, 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. 42
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 May 6, 1977, as supplemented June 21 and July 11, 1977, and application dated March 1, 1977, as supplemented May 5, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 42, 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: July 29, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 45 TO DPR-38

AMENDMENT NO. 45 TO DPR-47

AMENDMENT NO. 42 TO DPR-55

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

Revise Appendix A as follows:

Remove the following pages and insert revised identically numbered pages.

2.1-3a
2.1-3b
2.1-5
2.1-8
2.1-11
2.3-3
2.3-4
2.3-6
2.3-9
2.3-12
3.5-8
3.5-9
3.5-10
3.5-11
3.5-14
3.5-14a
3.5-15
3.5-19
3.5-19a
3.5-19b
3.5-22
3.5-22a
3.5-22b
3.5-23f
3.5-23g
3.5-23h
4.1-9
4.2-3
4.20-1

Bases - Unit 2

The safety limits presented for Oconee Unit 2 have been generated using BAW-2 critical heat flux correlation [1] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (design flow is 352,000 gpm for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[2].

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

The curve presented in Figure 2.1-1B represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 374,880 gpm). This curve is based on the following nuclear power peaking factors with potential fuel densitication and fuel rod bowing effects: $F_q^N = 2.67$; $F_{\Delta H}^N = 1.78$; $F_z^N = 1.50$. The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

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

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.8 kw/ft for Unit 2.

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

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

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

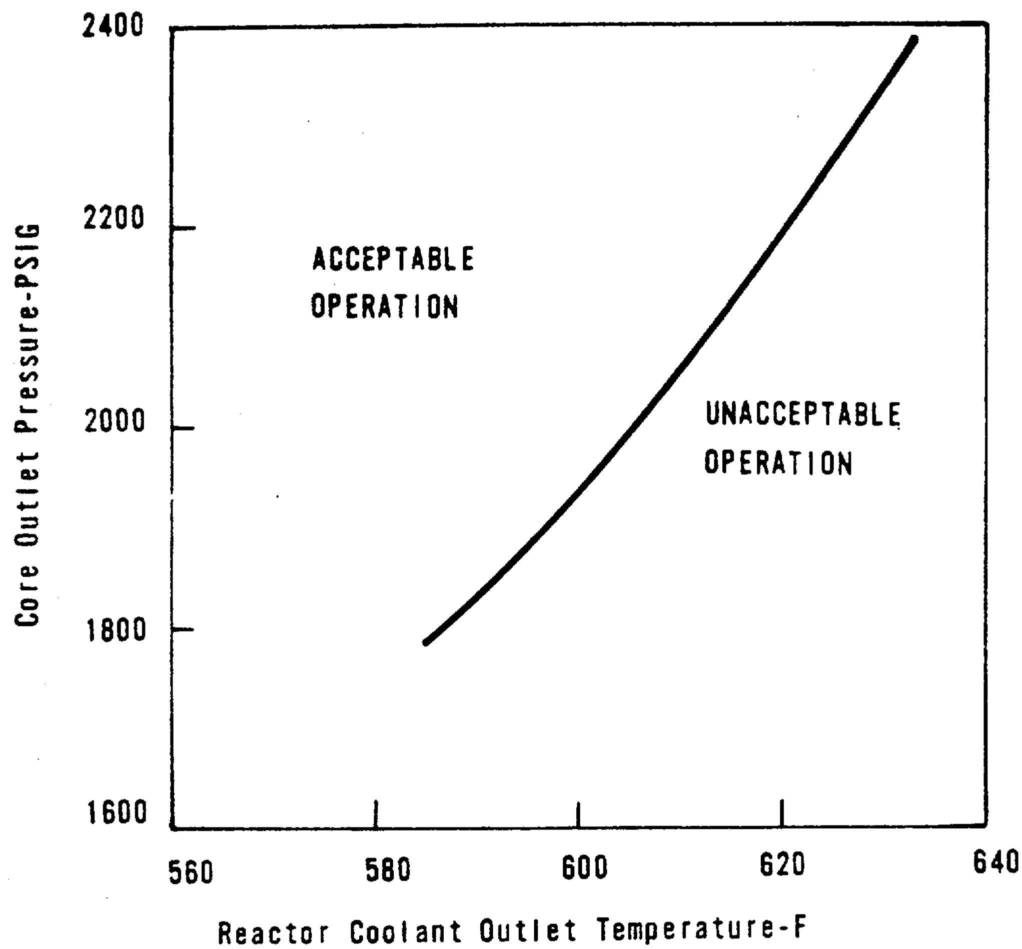
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055= 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar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

For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.30 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

References

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

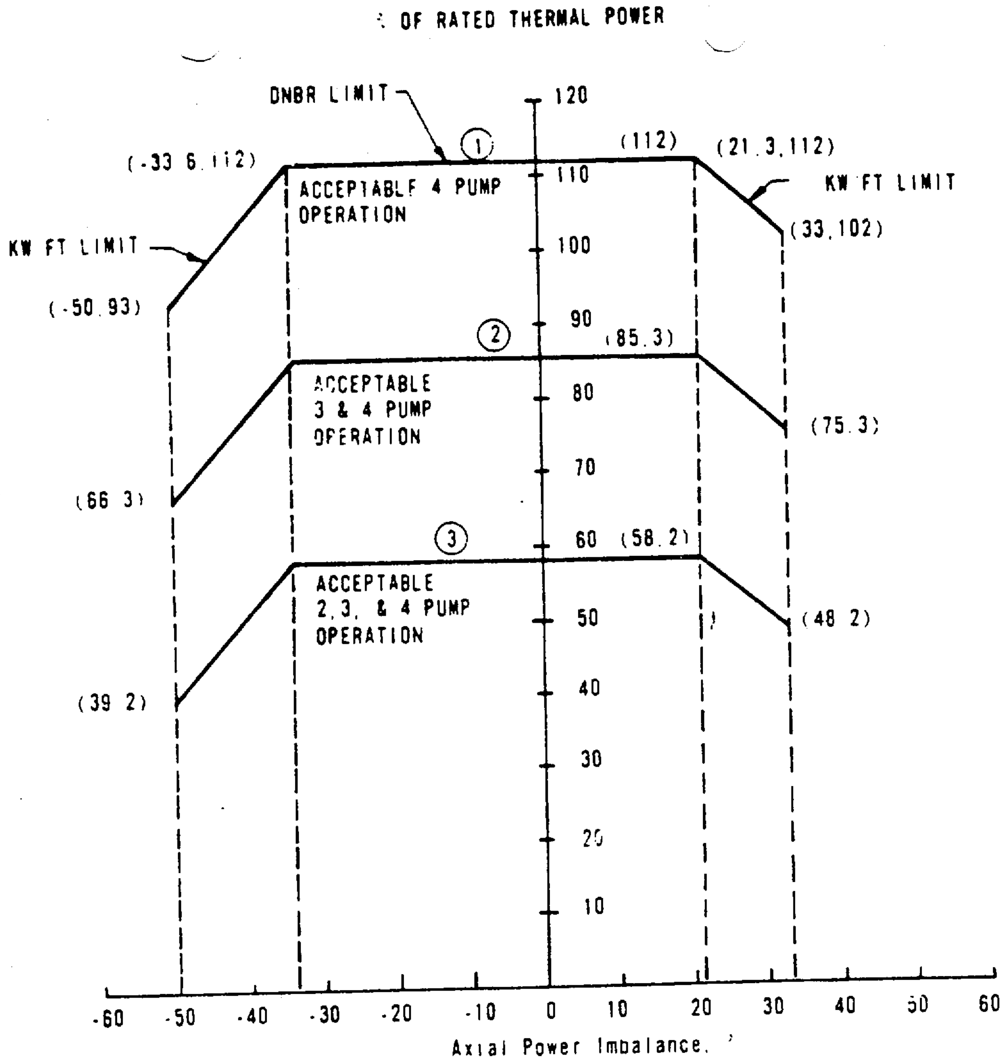
(2) Oconee 2, Cycle - Reload Report - BAW-1452, April, 1977.



CORE PROTECTION SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1-1B

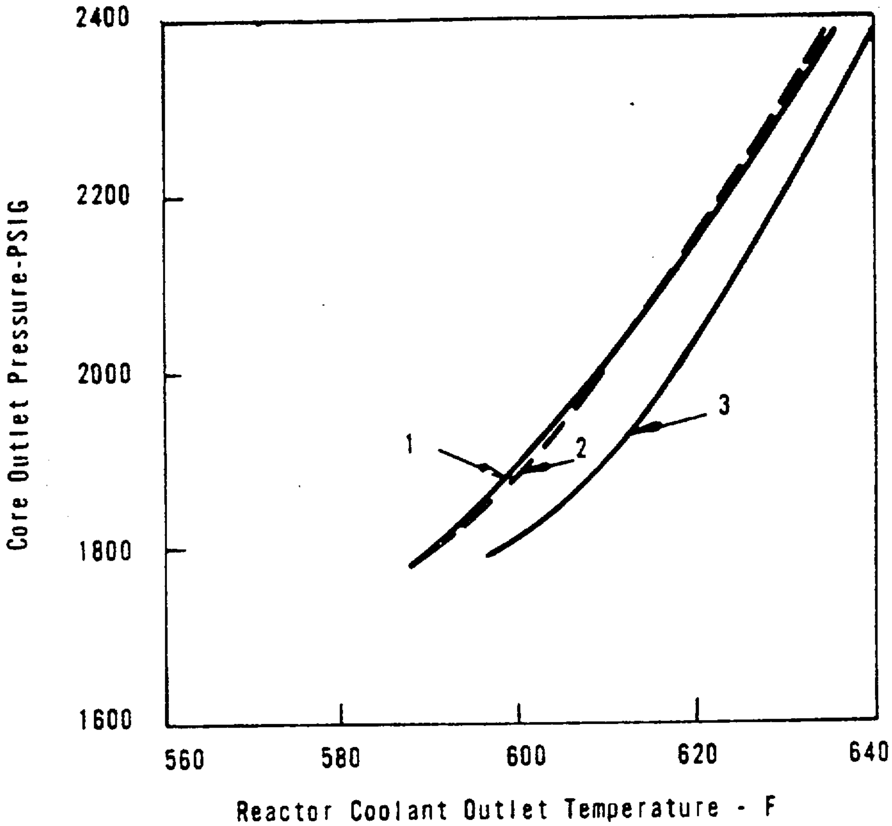


CURVE	REACTOR COOLANT FLOW (GPM)
1	374,880
2	280,035
3	183,690

UNIT 2
CORE PROTECTION SAFETY LIMITS

OCONEE NUCLEAR STATION
Figure 2.1-2B

Amendments 45, 45 & 42



CURVE	REACTOR COOLANT FLOW (GPM)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374880 (100%)	112%	Four Pump (DNBR Limited)
2	280035 (74.7%)	86.7%	Three Pump (DNBR Limited)
3	183690 (49%)	59.0%	One Pump In Each Loop (Quality Limited)

CORE PROTECTION SAFETY LIMI
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1-3B

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% -Unit 1 for a 1% flow reduction.
1.055%-Unit 2
1.07%-Unit 3
For Unit 2, the power-to-flow reduction ratio is 0.949, and for Units 1 and 3 the power-to-flow reduction factor is 0.961 during single loop operation.

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear over-power trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure (11.14 T_{out}-4706) trip
(1800) psig (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%.

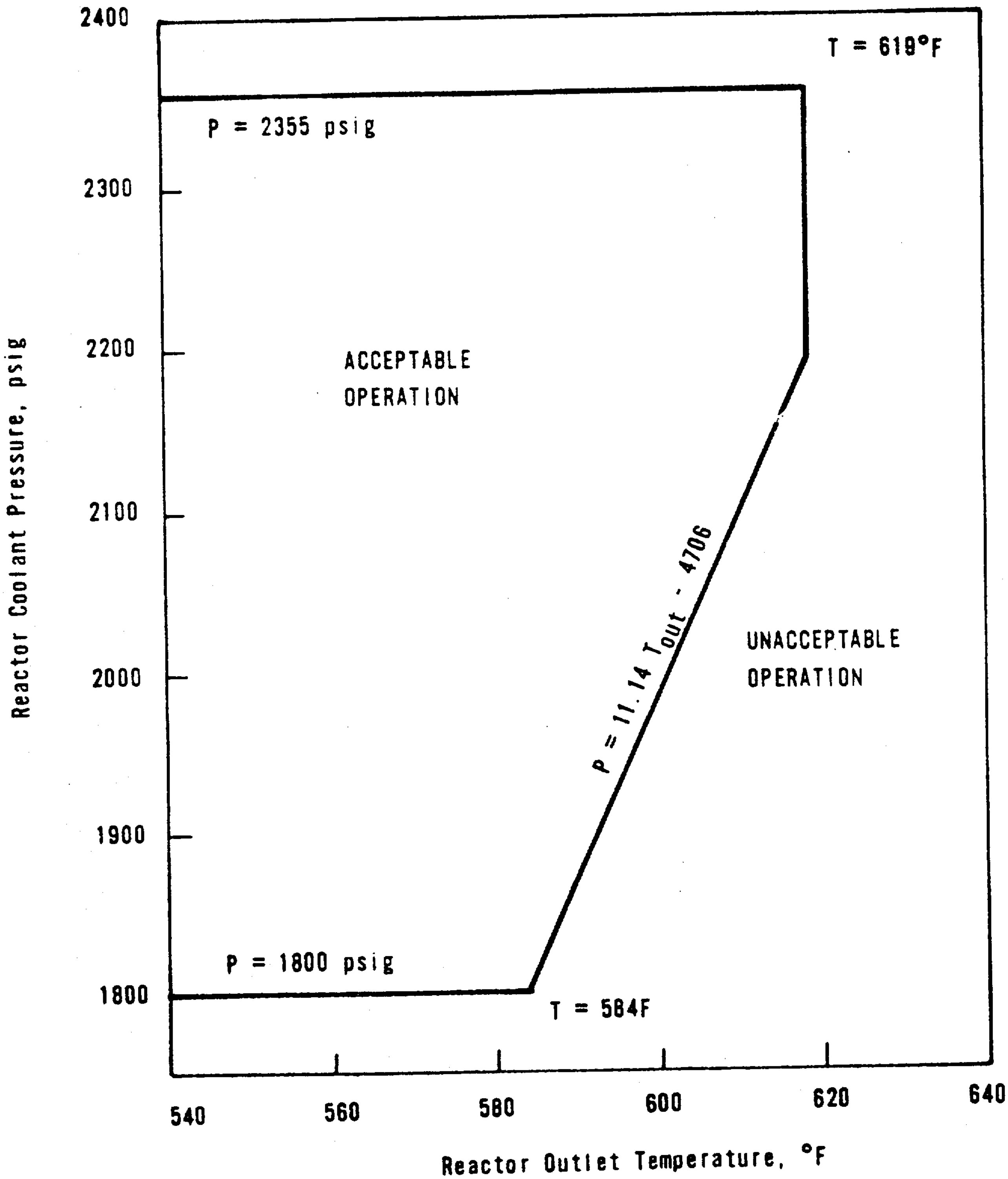
B. **Single Loop Operation**

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 0.961 (Unit 1)**
   0.949 (Unit 2)
   0.961 (Unit 3)

REFERENCES

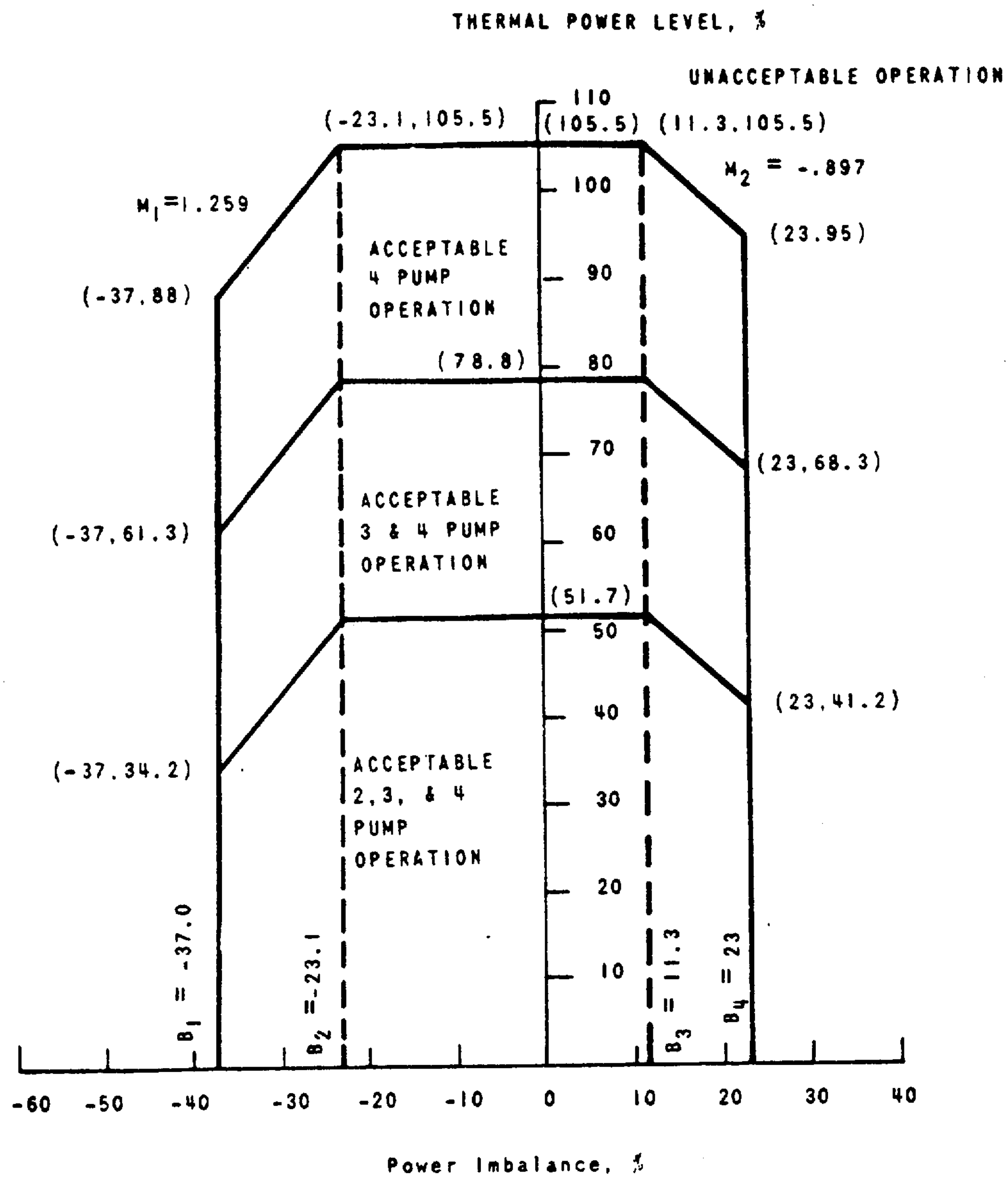
(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS, UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3-1B



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION
Figure 2.3-2B

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	NA	55% (5) (6)	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5. Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6. Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out}-4706)$(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) 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 and 3.5.2-1A2 (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 and 3.5.2-2A2 (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 3.5.2-4B1, 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 (Unit 1), 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2), and 3.5.2-1C1, 3.5.2-1C2, 3.5.2-1C3 (Unit 3), unless the following requirements are met.

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

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

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.3-3B1, 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, 3.5.2-3B1, 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 0.5% $\Delta k/k$ (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, 5) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.5% $\Delta k/k$ (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 calibration errors. The method used to define the operating limits is defined in plant operating procedures.

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

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

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

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

Operating restrictions are included in Technical Specification 3.5.2.5d to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond 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

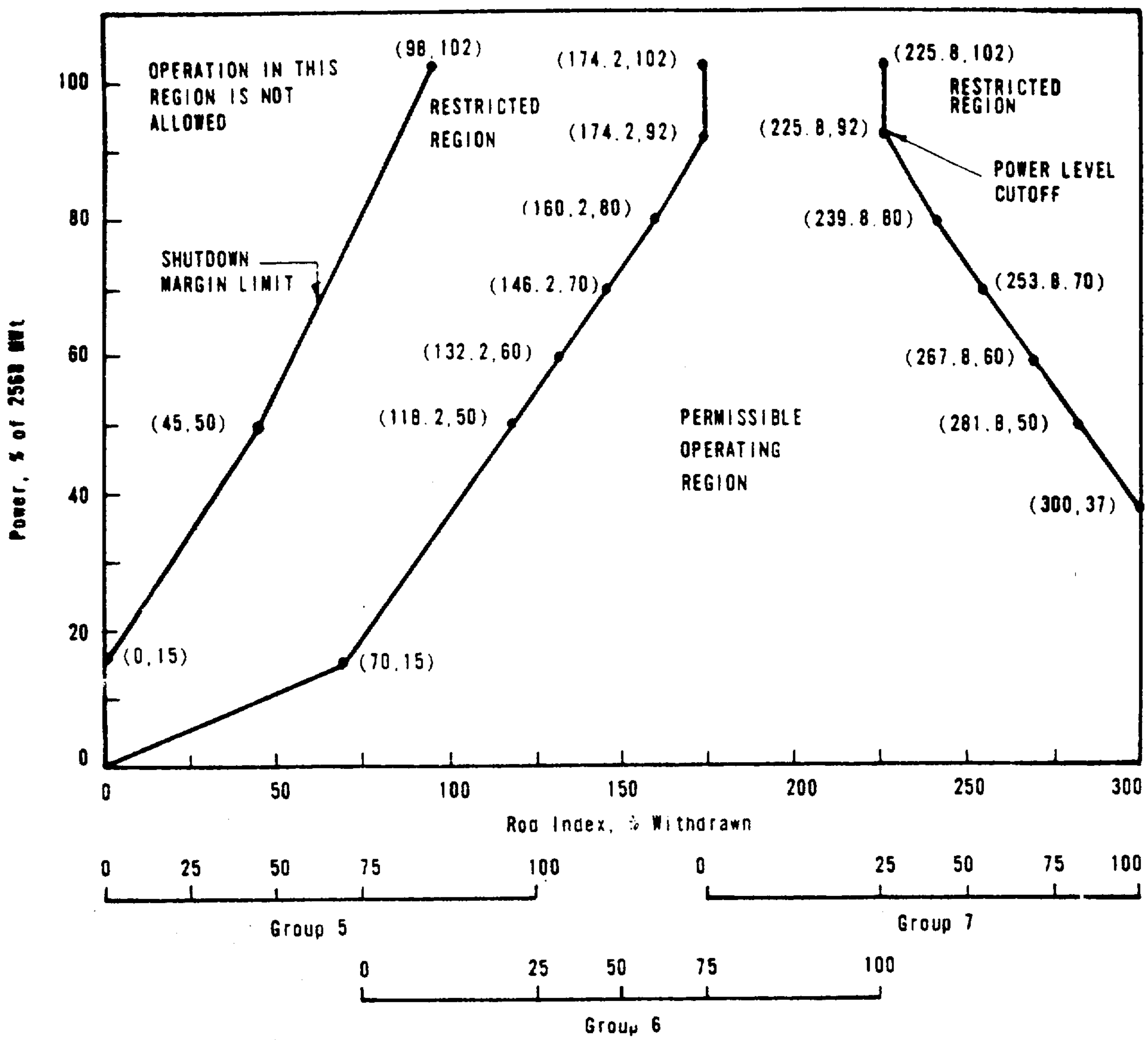
[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

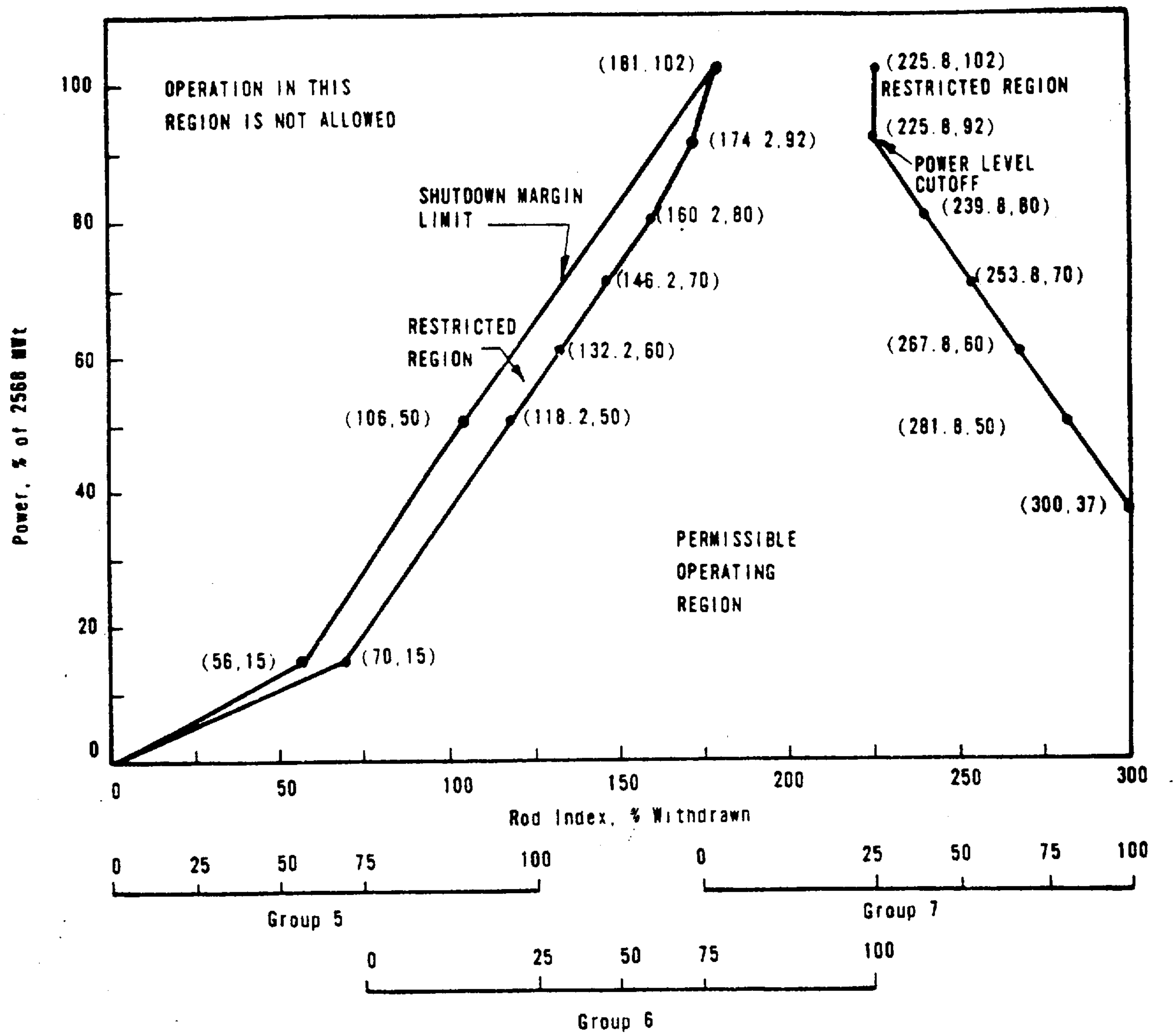
BAW-1400 (UNIT 3)



ROD POSITION LIMITS FOR
4 PUMP OPERATION FROM
0-100 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

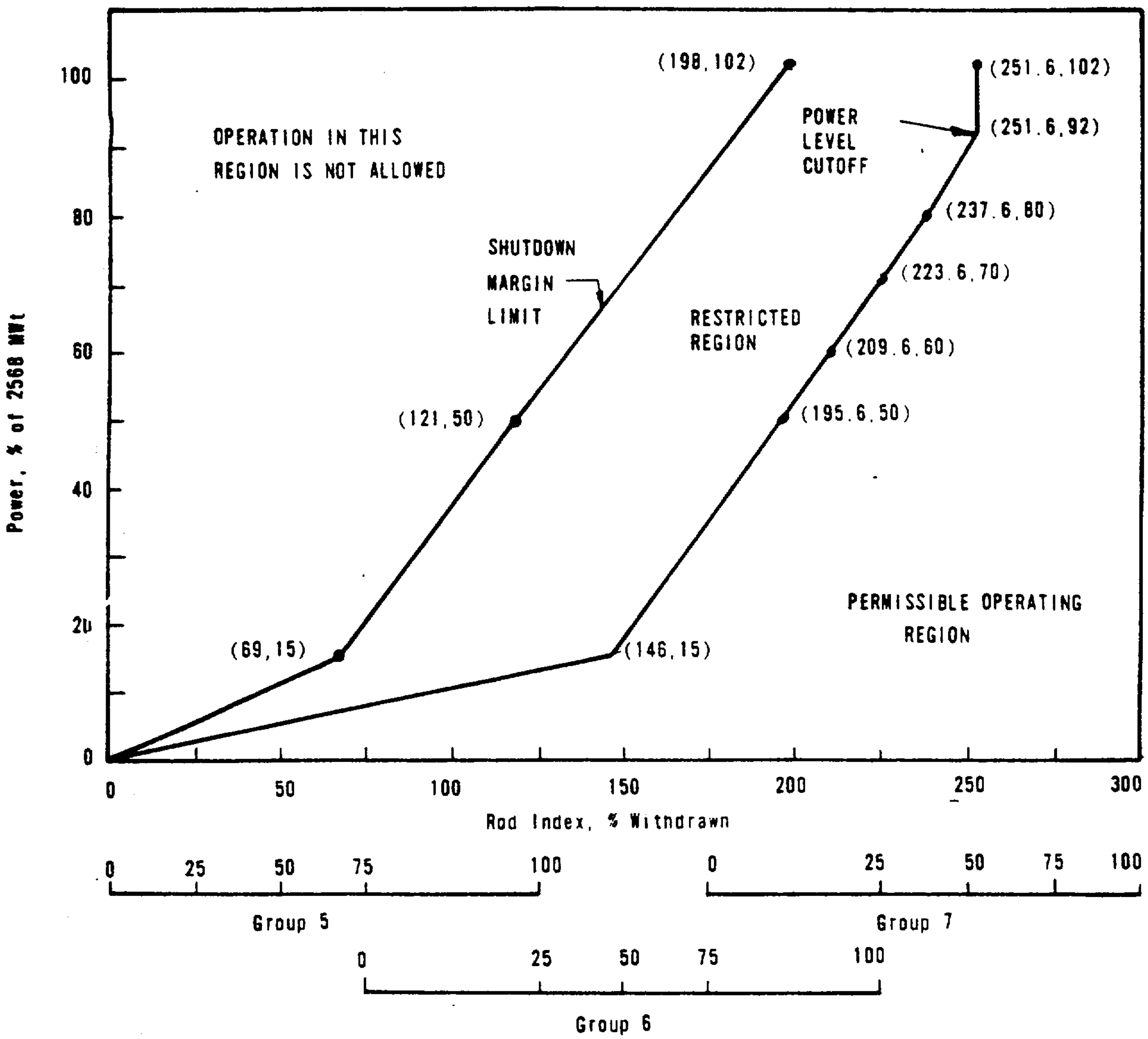
Figure 3.5.2-1B1



ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD UNIT 2

OCONEE NUCLEAR STATION

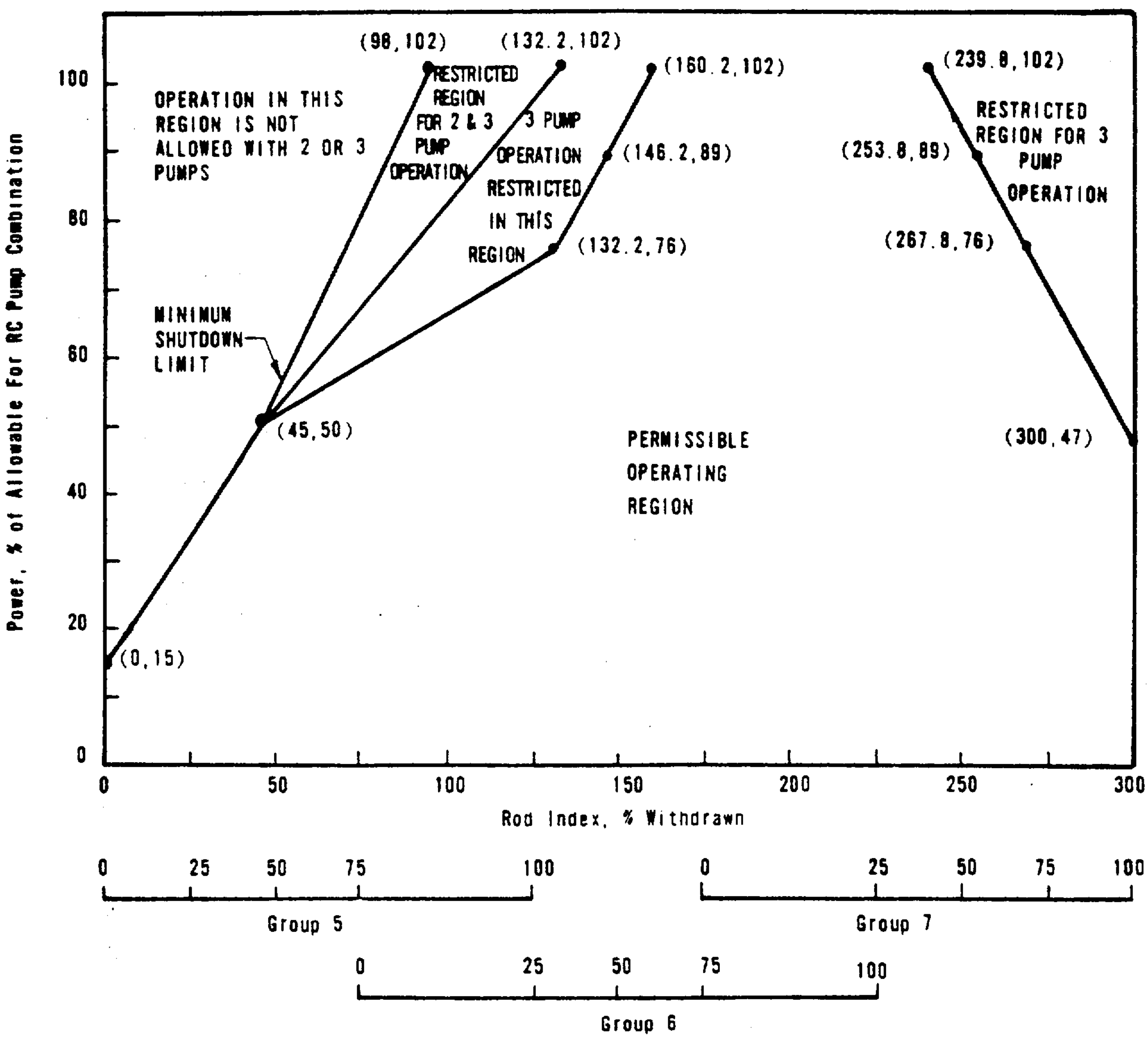
Figure 3.5.2-1B2



ROD POSITION LIMITS
FOR 4 PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 2

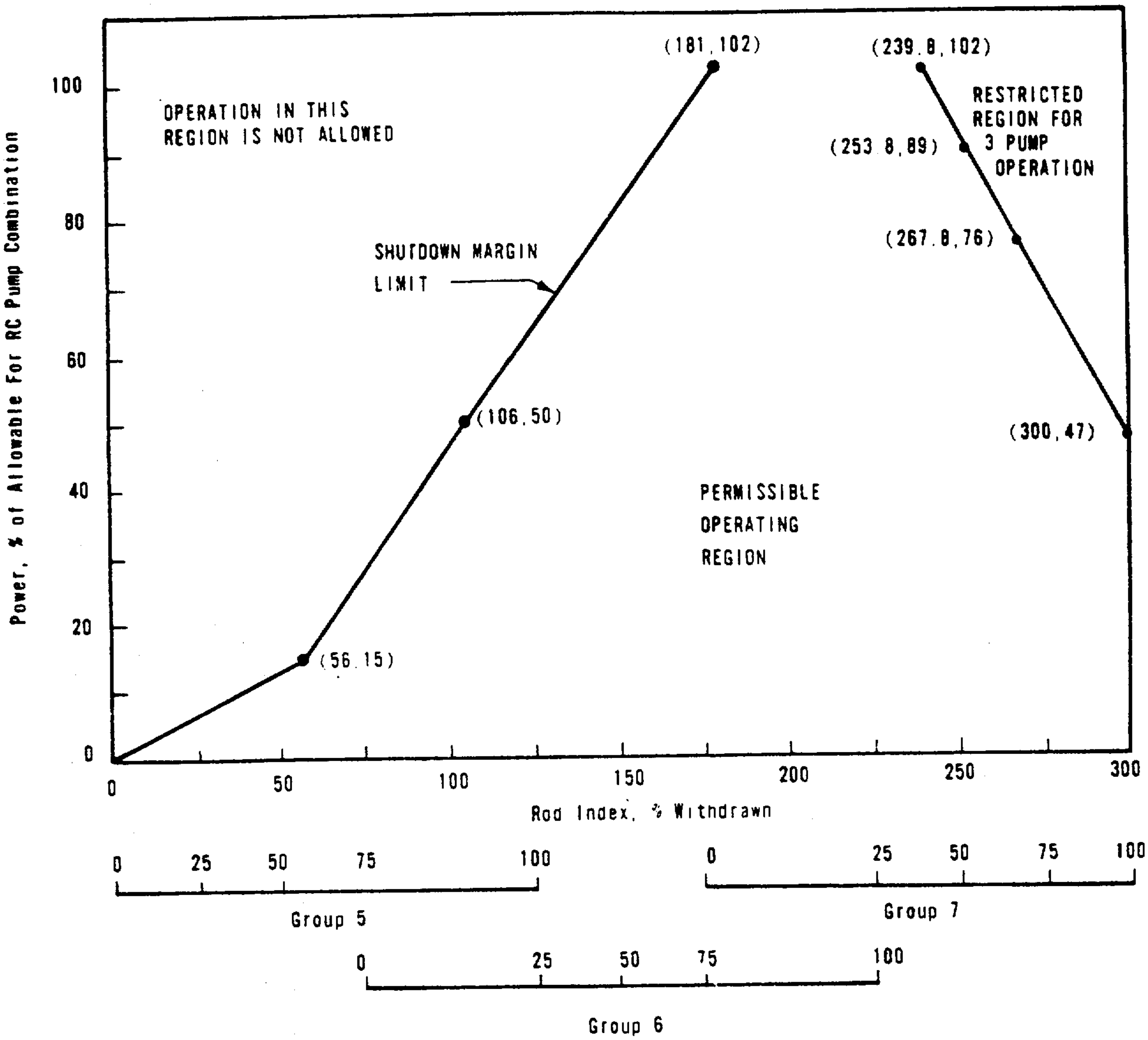
OCONEE NUCLEAR STATION

Figure 3.5.2-1B3



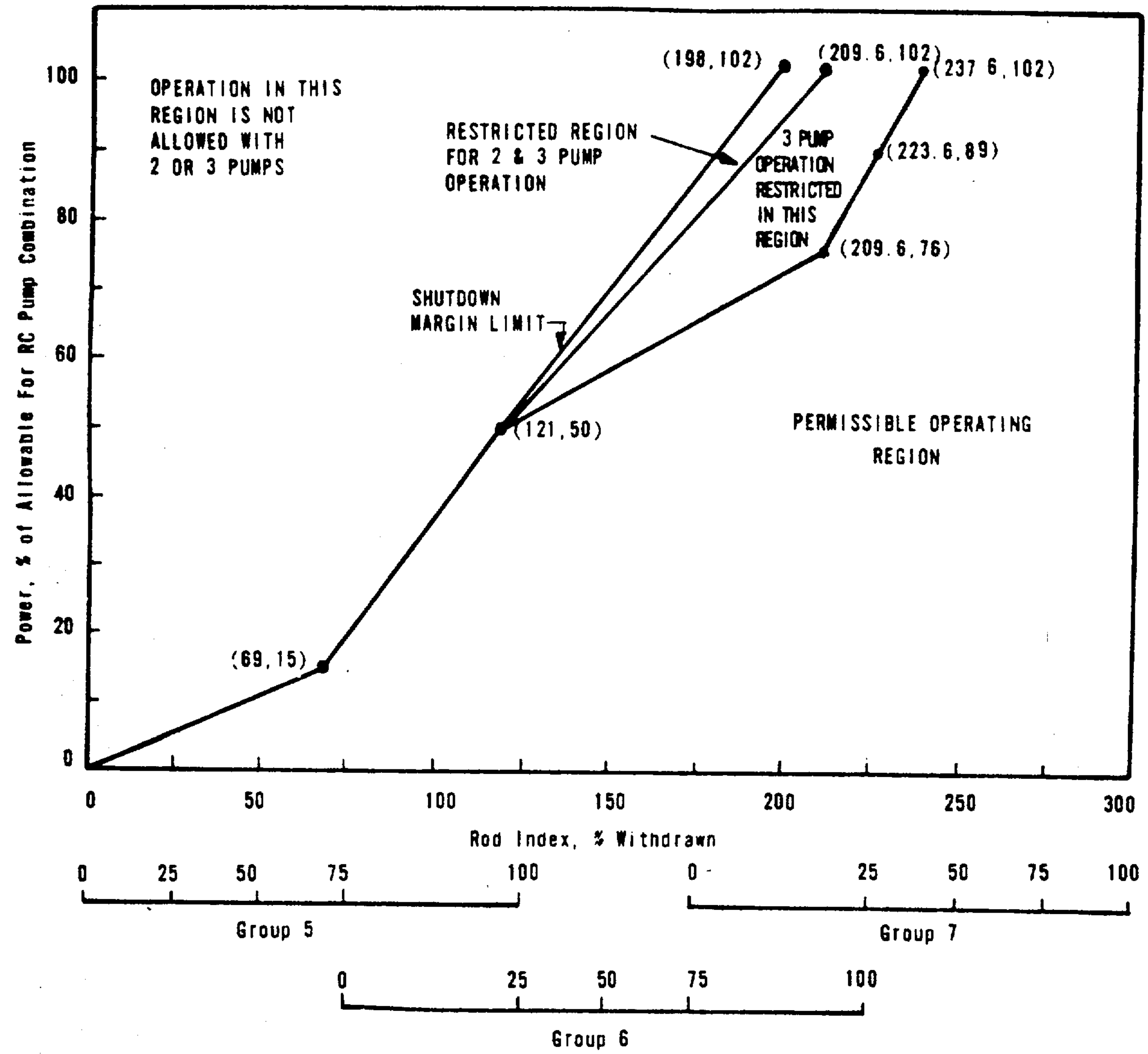
ROD POSITION LIMITS
FOR TWO AND THREE PUMP
OPERATION FROM 0 TO 100 EFPD
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



ROD POSITION LIMITS FOR TWO
AND THREE PUMP OPERATION
FROM 100 ± 10 TO 250 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-2B2

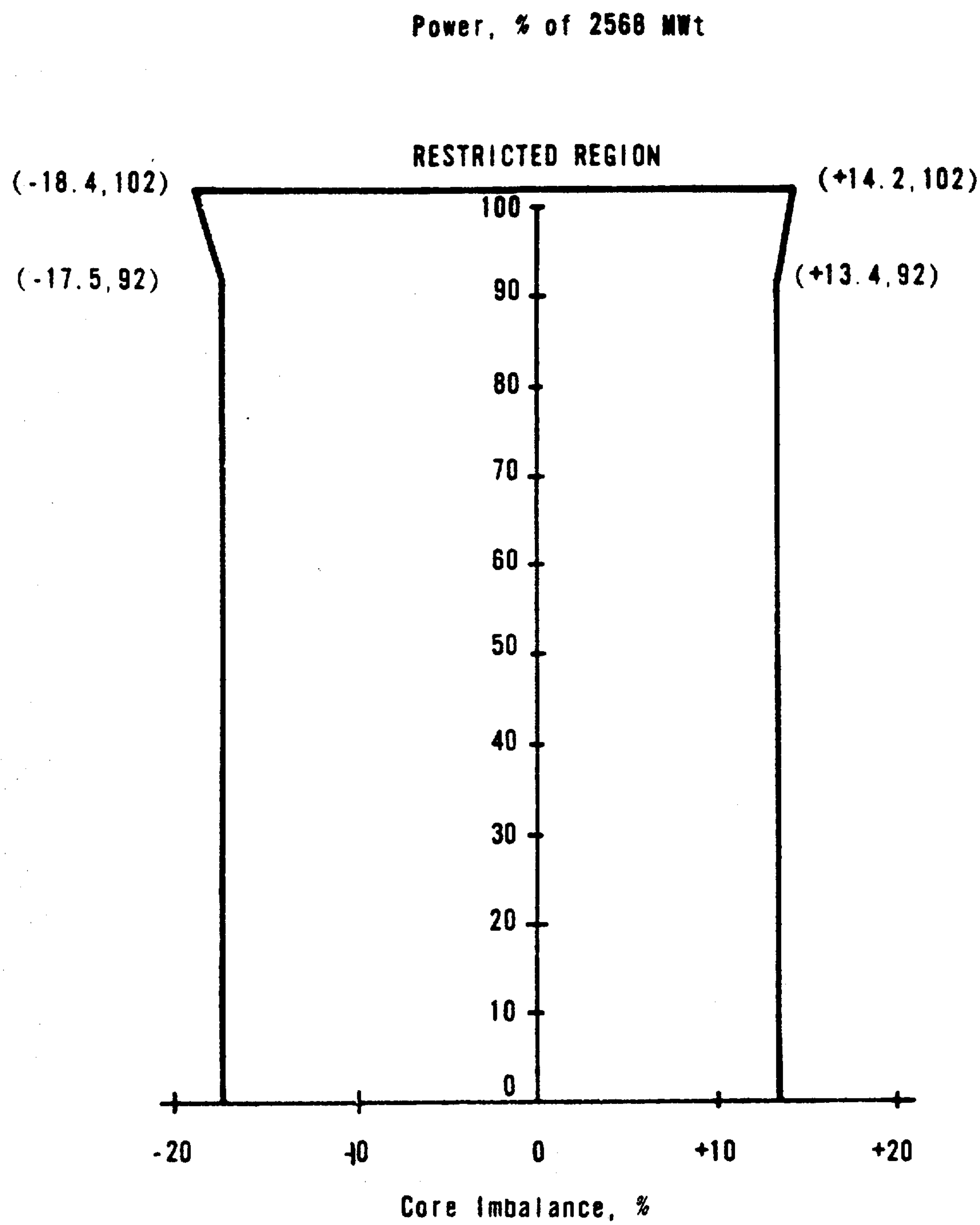


ROD POSITION LIMITS FOR TWO AND THREE PUMP OPERATION AFTER 250 ± 10 EFPD
UNIT 2

CITIZENSHIP BY DUKE POWER IN SERVICE

OCONEE NUCLEAR STATION

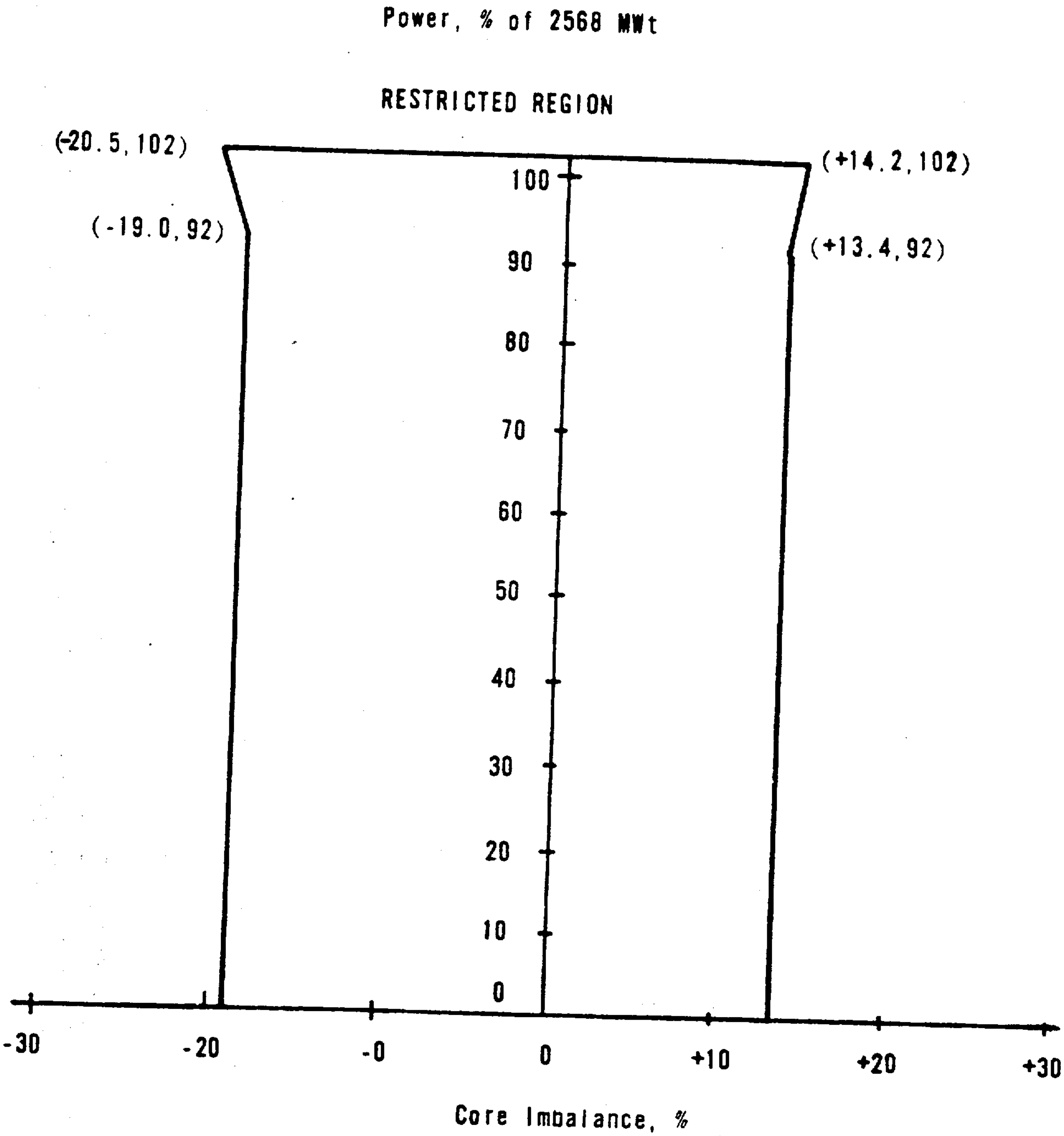
Figure 3.5.2-2B3



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 100 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

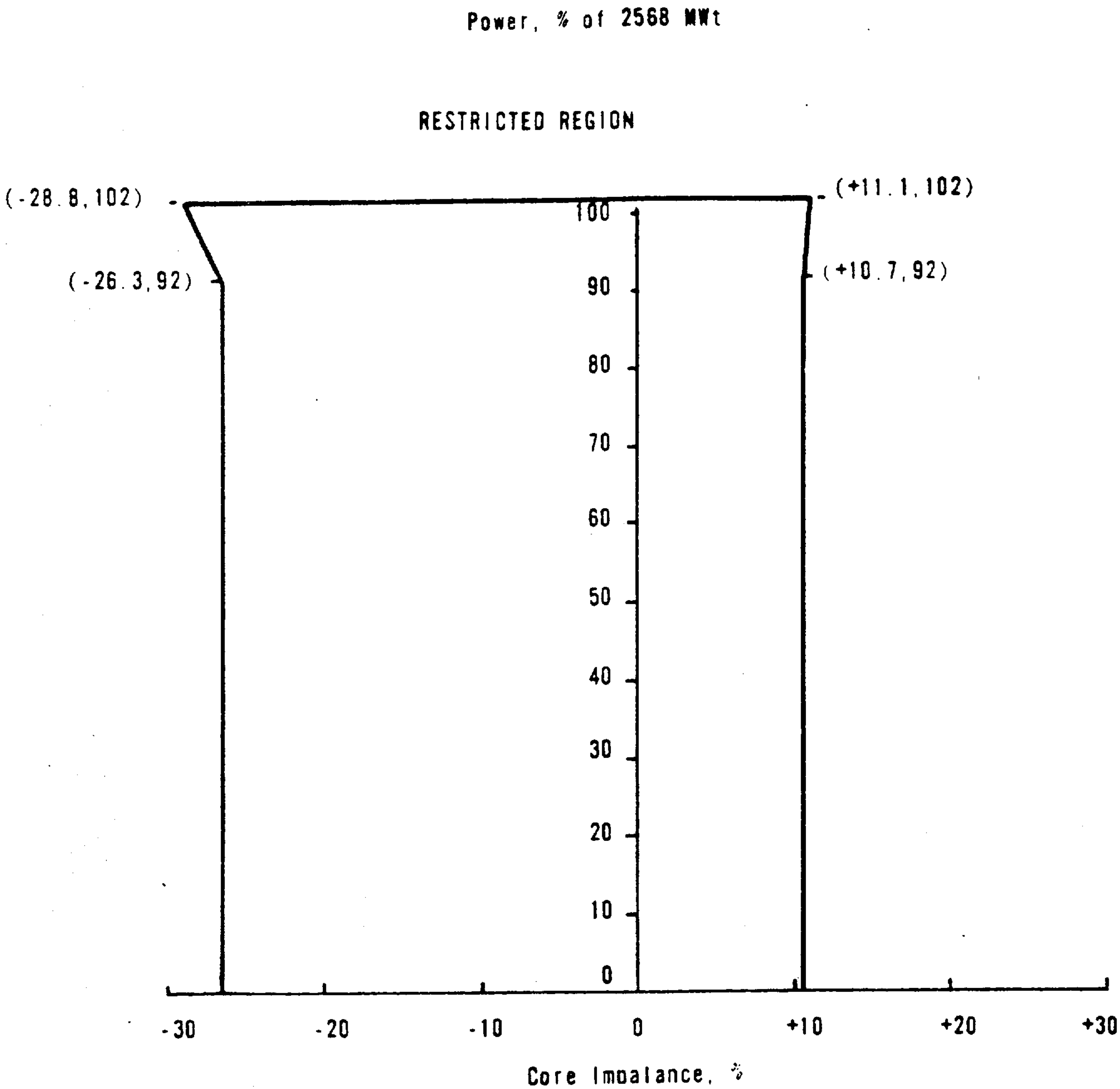
Figure 3.5.2-3B1



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
100 ± 10 TO 250 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-3B2

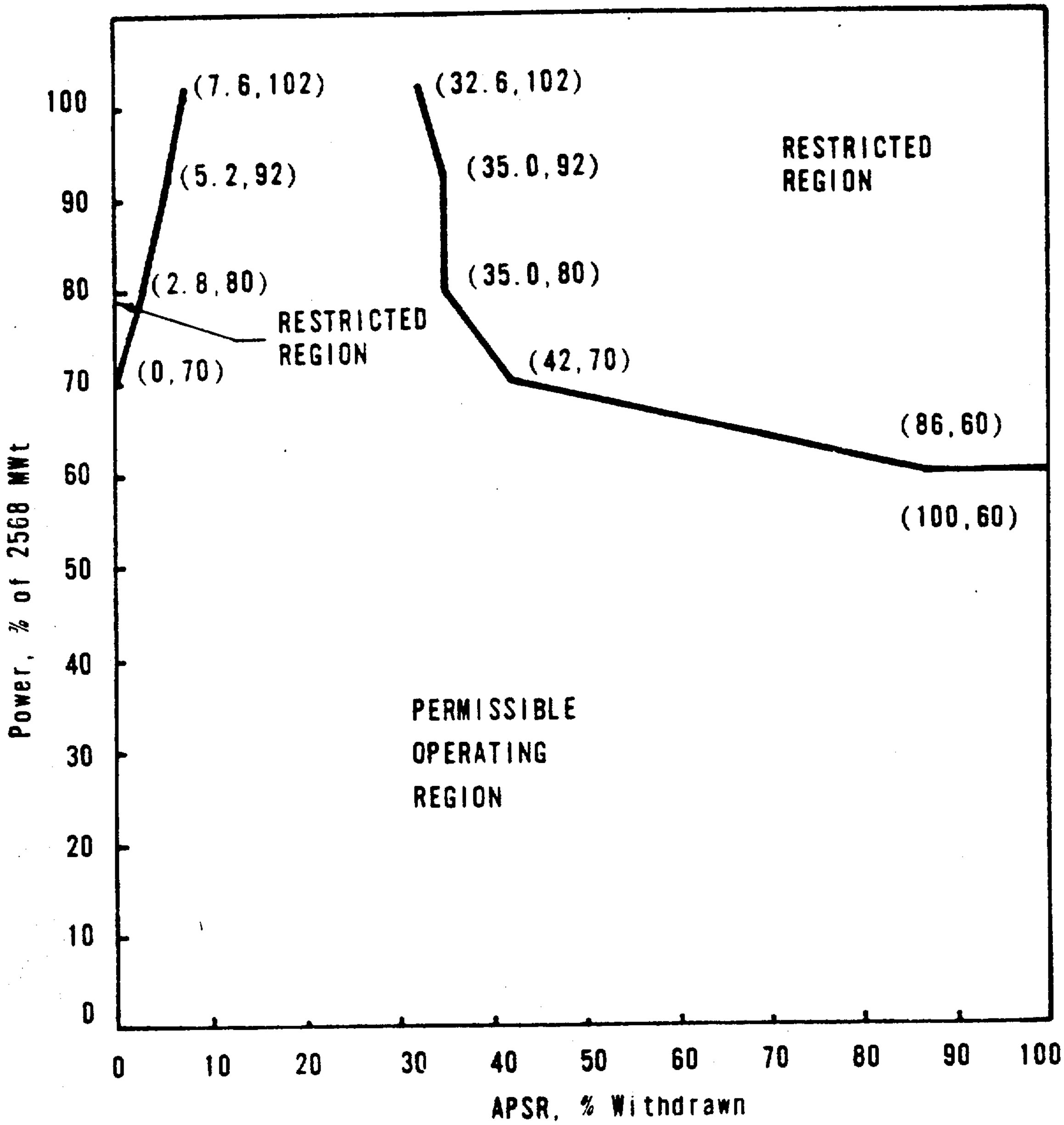


OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 250 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-3B3

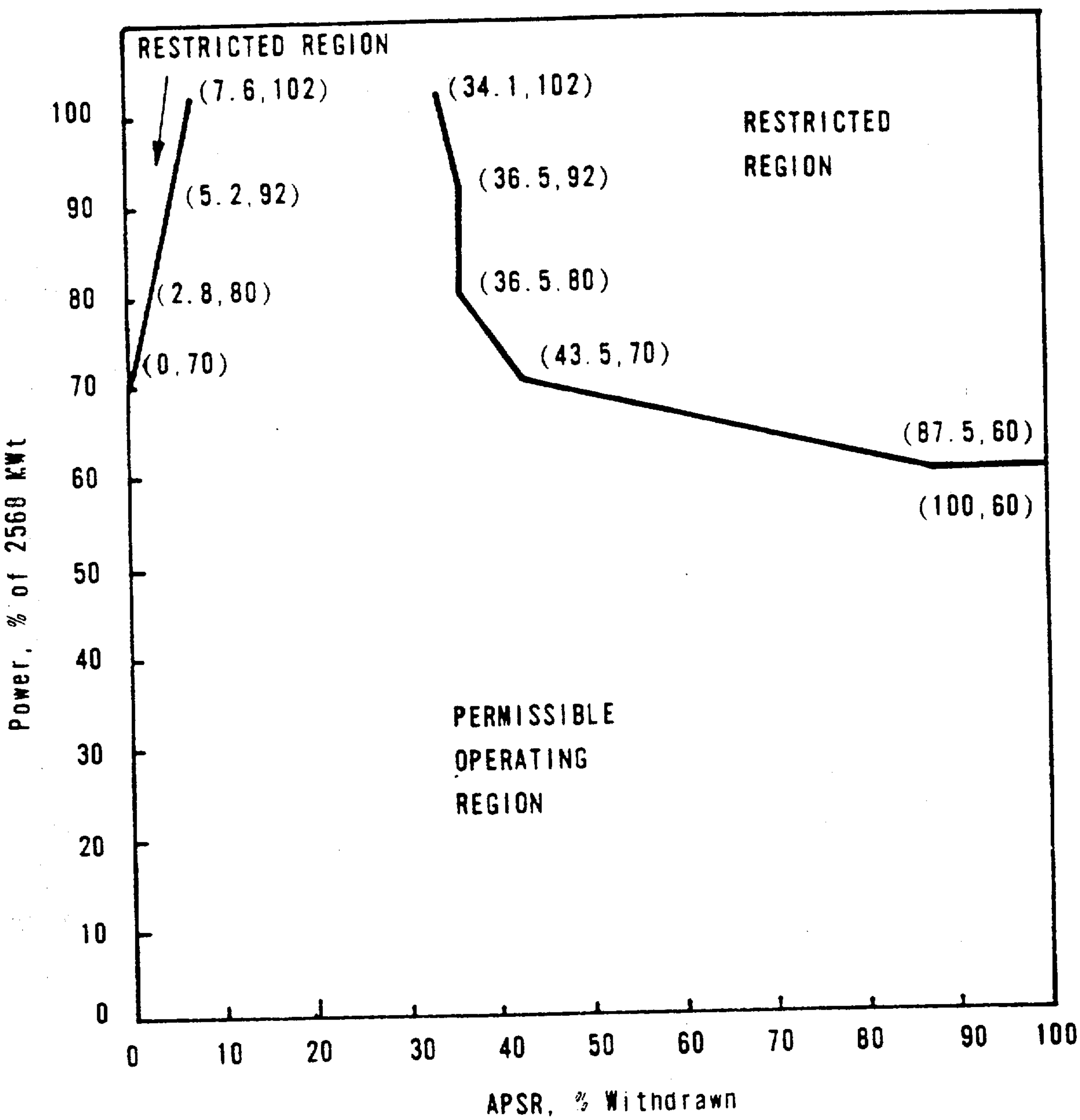
Amendments 45, 45 & 42



APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

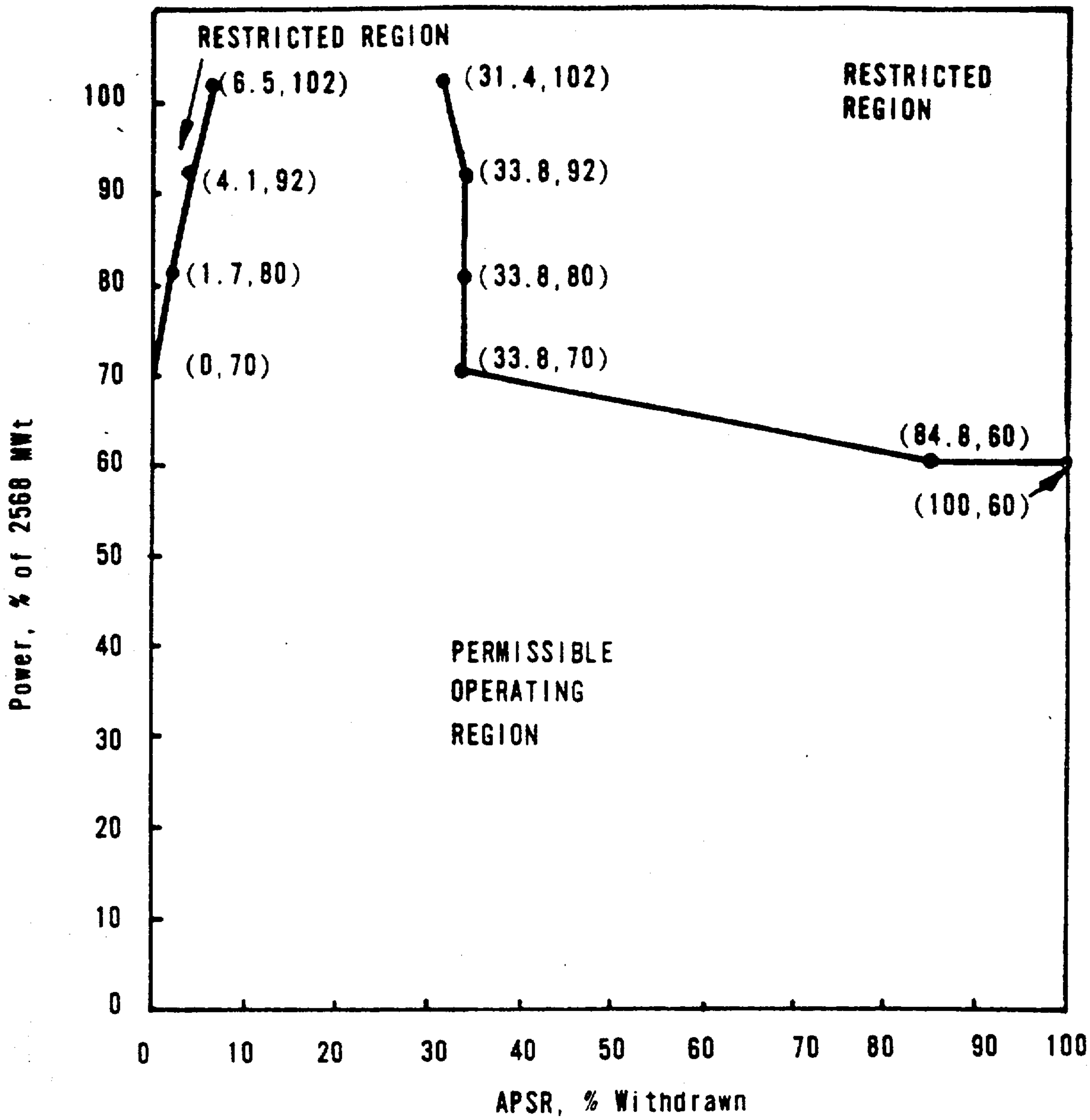
Figure 3.5.2-4B1



APSR POSITION LIMITS FOR
OPERATION FROM 100 ± 10
TO 250 ± 10 EFPD, UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B2



APSR POSITION LIMITS FOR
OPERATION AFTER 250 $\pm$ 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B3

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 2 143.8 x 10^6 lb/hr Unit 3 141.30 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.

4.2.10 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of the following: After March 13, 1978, any time that Crystal River Unit No. 3 fails to maintain a cumulative reactor utilization factor of greater than 45%.

The report shall provide justification for continued operation of Oconee Nuclear Station Units 1, 2 and 3 with the reactor vessel surveillance program conducted at Crystal River Unit No. 3 or the application for license amendment shall propose an alternative program for conduct of the reactor vessel surveillance program.

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

Bases

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

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

For the purpose of Technical Specification 4.2.10. Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commerical operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)]. The definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commerical operation".

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

4.20 REACTOR VESSEL INTERNALS VENT VALVES

Applicability

Applies to reactor vessel internals vent valves used to prevent vapor lock in the reactor vessel following a postulated reactor coolant inlet pipe rupture.

Objective

To verify that the reactor vessel internals vent valves operate as required.

Specification

At least once each refueling cycle, each reactor vessel internals vent valve shall be demonstrated operable by:

a. Conducting a remote visual inspection of visually accessible surfaces of the value body and disc sealing faces and evaluating any observed surface irregularities.

b. Verifying that the value is not stuck in an open position, and

c. Verifying that the valve can be fully opened with force equivalent to or less than 1.00 psid.

Bases

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves (1) assures operability, (2) assures that the valves are not open during normal operation, and (3) demonstrates that the valves are fully open at the forces equivalent to the differential pressures justifiable by the ECCS analysis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 TO LICENSE NO. DPR-38

AMENDMENT NO. 45 TO LICENSE NO. DPR-47

AND AMENDMENT NO. 42 TO LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

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

Introduction

By letter dated May 6, 1977(1) and as supplemented June 21, 1977(2) and July 11, 1977, Duke Power Company (the licensee) requested changes to the Technical Specifications appended to the Facility Operating License for the Oconee Nuclear Station, Unit 2 (Oconee 2). The proposed changes would permit Oconee 2 operation as reloaded for Cycle 3. By letter dated March 1, 1977(12) as supplemented May 5, 1977(10), the licensee also requested a change to Technical Specifications appended to the Facility Operating Licenses which would revise the reactor internal vent valves testing program for all three Oconee Units.

Evaluation

The Oconee 2 reactor core consists of 177 fuel assemblies. The reload for Cycle 3 will involve the removal of all 61 Batch 2 fuel assemblies and 12 of the Batch 3 fuel assemblies, and the relocation of the residual Batch 3 and Batch 4 fuel assemblies. The removed fuel will be replaced by 5 Batch 1 fuel assemblies from Oconee 2, 12 Batch 1 fuel assemblies from Oconee 3, and 56 new Batch 5 fuel assemblies. These assemblies will occupy primarily the periphery of the core and four locations in the interior regions of the core. All but four of the Cycle 3 core fuel assemblies have a 15x15 array of fuel rods. Of these four, two of the Batch 4 and two of the new Batch 5 fuel assemblies are demonstration Mark C & CR assemblies, respectively. Each of these demonstration assemblies consist of a 17x17 array of fuel rods. A description of the program to irradiate the assemblies was provided by letter dated January 28, 1976(3). In addition, a Babcock & Wilcox (B&W) report on the irradiation of 17x17 demonstration assemblies in Oconee 2 of January 1976(4), was provided which describes the mechanical, nuclear and thermal-hydraulic characteristics of these demonstration assemblies.

The licensee's reload analyses and Technical Specification changes submitted by letter dated May 6, 1977 were based on an originally planned 296 effective full power days (EFPD) of Oconee 2 Cycle 2 operation. The licensee, however, advised us by letter dated June 21, 1977[2] that Cycle 2 operation was being terminated early at 277 EFPD. As a result, the burnup distribution in the Batch 3 and 4 fuel assemblies, which are to remain in the core for Cycle 3 operation, will be different from that assumed in the original reload analysis. This June 21, 1977 submittal also included changes to the reload configuration (Removal of the 12 Batch 3 fuel assemblies and insertion of the 12 Batch 1 fuel assemblies from Oconee 3). Based on a reanalysis of the new burnup distribution of the Batch 3 and 4 fuel assemblies, and the change in the fuel reload configuration, the licensee submitted revisions to the reload report and Technical Specifications.

Fuel Mechanical Design

Table 4-1 of Reference 5 summarizes the reload core fuel assembly parameters. The Batch 5, 15x15, (Mark B-4) fuel assembly design has been reviewed and accepted by us for use in Oconee 2.[6] This type of assembly is currently operating in Oconee 2. The 56 new Batch 5 fuel assemblies, therefore do not represent any unreviewed change in mechanical design from the reference cycle.

Five of the new Cycle 3 replacement fuel assemblies are once-burned Batch 1 fuel assemblies. These assemblies were removed from the Oconee 2 reactor following Cycle 1 operation and have been stored in the spent fuel pool during Cycle 2. These assemblies are of the Mark B-2 type and have been previously reviewed and approved by us for operation in Oconee 2. They do not significantly change the mechanical design of the Cycle 3 core.

Twelve of the Cycle 3 replacement fuel assemblies are once-burned Batch 1X fuel assemblies removed from Oconee 3 (Batch 1X). These assemblies are of the Mark B-3 design similar to the twice-burned, Batch 3, Oconee 2 assemblies. This mechanical design which has been previously reviewed and approved for use in Oconee 2, does not significantly change the mechanical design of the Cycle 3 core.

As stated earlier, there are four demonstration fuel assemblies proposed for Cycle 3 operation in Oconee 2. Two of these are Mark C fuel assemblies and two are Mark CR fuel assemblies. The Mark CR assemblies were placed in the Cycle 2 core, and will continue irradiation in the Cycle 3 core. These assemblies have a 17x17 fuel rod configuration. There are two different length fuel pellets used in these 17x17 assemblies. The fuel rod outside and inside diameters have been decreased in the Mark C assemblies. The Mark C assemblies are mechanically compatible and interchangeable with Mark B type assemblies with the exception of the control rod component interface. These assemblies have been previously reviewed and found acceptable for operation in Oconee 2.[6]

The two demonstration Mark CR fuel assemblies[16] and the Mark C fuel assemblies are identical except that the Mark CR assemblies have reconstitutable lower end fittings. The reconstitutable feature is provided by positioning the lower end fitting to the lower grid by flange sleeves on the guide tubes rather than by welding the lower grid to the lower end fitting. Also, the lower end fitting is fastened to the guide tubes by torque nuts as in the Mark C demonstration assemblies; however, the nuts are prevented from rotating by swaged locking cups rather than by welding. The cups are based on a retainer plate that is restrained flush against the lower end fitting by a guide tube nut. The retainer plate cup brazement captures all 24 nuts by means of deformed metal tangs, so all nuts have to be untorqued before the brazement, including nuts, can be removed.

The Mark CR design has been subjected to a 300 hour test at simulated reactor full power conditions with no deterioration or wear of any of the parts of the retainer system (brazement nuts, lower end fittings, flange sleeves). Cold water tests were performed and the Mark CR design was found to have the same resonant frequency and amplitude as the Mark C design. Bench tests have been performed on the retainer system to assure the locking cups are securely brazed to the nut plate and the swaged cups will prevent loosening of the nuts which are captured within the nut plate brazement. The Mark CR demonstration assemblies have been designed to maintain their structural integrity through three cycles of operation and to successfully withstand seismic and loss-of-coolant loads. This reconstitutable mechanical locking configuration has been used in a similar function (new B&W tube specimen holder design); tested under simulated conditions, and completely analyzed and is being used in other B&W reactors. Therefore, the use of the reconstitutable lower end fitting is acceptable for the demonstration Mark CR assemblies.

These mechanical design variations have been taken into account in the various mechanical analyses. The Batch 3 fuel is generally limiting, because of it's low initial fuel pellet density, low initial pre-pressure, and previous incore exposure. The results of these analyses have shown that the mechanical design differences in the Oconee 2 Cycle 3 fuel assemblies are of negligible affect and are acceptable.

Fuel rod cladding creep collapse analyses were performed for the fuel batches which will be present in the Cycle 3 core. The calculational methods, assumptions, and data have been previously reviewed and approved by us.(13) 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 4 and Batch 5 analysis. The actual reactor operating history along with the most restrictive power histories for the forthcoming cycle were used in the analyses of the Batch 3 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 3 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 3 fuel is most limiting with respect to stress, because of its low prepressurization and density. The cladding strain for Cycle 3 operation is less than the generally used 1% plastic strain acceptance criteria. The strain analysis assumed maximum specification value for the fuel pellet diameter, density, and burnup, and minimum specification tolerance on fuel cladding inside diameter. These assumptions conservatively represent the cladding strain. The Batch 3 fuel will again be limiting in the Cycle 3 core based on the cladding strain. Again this is because of its irradiation history, lower prepressurization, and lower fuel pellet density.

The Batch 5 fuel assemblies are not new in concept and do not use different component materials. The fuel assemblies for Cycle 3 operation will not exceed their design life limits. In addition, it has been shown that the presence of the demonstration assemblies in the Cycle 3 core will have an insignificant effect on operation. We conclude therefore that the fuel mechanical design for Cycle 3 operation is acceptable.

Fuel Thermal Design

The fuel thermal design analysis was conducted using the TAFY-3 computer code.(7). This analysis established heat flux limits to fuel centerline melt. The analysis considered the effect of a power spike from fuel pellet densification.(8) Modifications consisting of change to the void probability, F_g, and size distribution, F_k, have been previously reviewed and approved by us for Oconee 2 fuel thermal design analysis.(6) This analysis is based on the lower tolerance limit on fuel density and assumes isotropic diametral densification shrinkage and anisotropic axial shrinkage densification. The calculated gap conductance was reduced by 25% in accordance with our interim evaluation of TAFY. These assumptions have been approved by us.(9)

During Cycle 3 operation, the highest relative assembly power levels occur in Batch 1 and 3 fuel assemblies. The fuel temperature analysis for Cycle 3 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.

Although Batch 4 and 5 fuel assemblies have a reduced active fuel length and greater linear heat generation rate, the maximum predicted centerline temperature of this fuel is lower than that of Batch 1 and 3 fuel assemblies. The maximum predicted centerline temperature for the Batch 1X fuel assemblies was also lower than that predicted for the Batch 1 and 3 fuel assemblies. This is due to the higher initial density of the Batches 1X, 4 and 5 fuel assemblies. Therefore, the thermal design analysis for the Batch 1 and 3 fuel assemblies thermal design analysis is bounding, and we conclude that the fuel thermal design for Oconee 2 Cycle 3 core is acceptable.

<u>Nuclear Analysis</u>

The reactor core physics parameters for Oconee 2 Cycle 3 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 2 and Cycle 3 cores. For example, EOC Doppler and moderator coefficient changes by less than 1% from Cycle 2 to Cycle 3. These changes are to be expected and are not significant.

The effects of the four demonstration fuel assemblies on the Cycle 3 nuclear design have been reviewed and found to be negligible.

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

<u>Thermal-Hydraulic Analysis</u>

The major acceptance criteria which we 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. The thermal-hydraulic analysis for Oconee 2 Cycle 3 reload were made using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the Cycle 3 core and are discussed below.

<u>Reactor Coolant System Flow Rate</u>

The reactor coolant flow rate was accurately measured during Cycle 1 operation and determined to be 111.5% of the system design flow. The licensee has proposed to take credit in the Cycle thermal-hydraulic analysis (as was done in Cycle 2) for this higher flow. The licensee will by measurement verify the reactor coolant system flow rate for Oconee 2 Cycle 3. The licensee will include conservatisms for uncertainties in the measurement of the flow in the thermal-hydraulic analysis. The licensee has used a flow-rate of 106.5% in the Oconee 2 Cycle 3 analysis with these conservatisms to be consistent with the flow rate used in Oconee 1 thermal-hydraulic analysis.

There are differences in the flow resistance between the Mark B-3 fuel assemblies and the Mark B-4 fuel assemblies. The flow resistance for the Mark B-4 fuel assemblies is less than that measured for the Mark B-3 assemblies. Also, the Mark C and CR assemblies have a greater flow resistance than either of the other two fuel assembly types. These differences have been analyzed. The Cycle 3 core has flow resistance characteristics that are similar to the Cycle 2 core. The possible introduction of core cross flow due to the different flow resistances has been considered. This phenomenon was shown to be of negligible effect from analyses for the previous cycle as discussed in reference 6.

Fuel Rod Bow

In the submittal dated May 6, 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 analyses with the NRC approved rod bow model or to show sufficient compensatory margin.

The licensee chose to show sufficient core flow 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 approximately a 3% margin in reactor coolant system flow rate, which is flow margin above the design valve of 106.5%. The required flow rate is specified in the Technical Specifications. The licensee will verify by measurement the reactor coolant system flow rate. From previous experience, we are assured that reactor coolant flow will remain essentially constant throughout the cycle. This approach will assure adequate thermal hydraulic design margin.

Mark C and CR Demonstration Assemblies

The thermal hydraulic design analysis has been based on a core configuration consisting of 177 Mark B (15x15) fuel assemblies. Comparative analyses have been performed to show that the insertion of the Mark C and CR (17x17) demonstration assemblies would have decreased the core thermal hydraulic margin if they were located in the high power region of the core. Therefore, the demonstration assemblies have been placed in low power producing core locations to ensure that these assemblies will not be limiting and to provide minimum impact on the hot assembly performance. Thus, the two Mark C and the two Mark CR demonstration assemblies are not limiting and their presence in Cycle 3 will not significantly affect the thermal-hydraulic characteristics of the reactor.

In summary, the licensee has proposed that a reactor coolant flow rate based on actual measured flow with uncertainties taken into account, be used in the Oconee 2 Cycle 3 thermal hydraulic analysis. The licensee has also assured us that there will be sufficient RCS flow to compensate for the difference between the approved and the unapproved rod bow models. The licensee has considered the impact of the Mark C and CR demonstration assemblies and has ensured the impact to be negligible. 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 analysis as previously approved and discussed are acceptable.

Accident and Transient Analysis

The accident and transient analyses provided by the licensee demonstrates that the Oconee FSAR analyses conservatively bounds the predicted conditions of the Oconee 2 Cycle 3 core and are therefore acceptable. Each FSAR accident analysis has been examined, with respect to changes in Cycle 3 parameters, to determine the effects of the Cycle 3 reload and to ensure that thermal performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analyses were design operating values based on calculated values plus uncertainties. Cycle 1 values (FSAR values) of core thermal parameters were compared with those used in the Cycle 3 analysis. For each accident of the FSAR, a discussion and the key parameters were provided. A comparison of the key parameters from the FSAR and Cycle 3 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 2 fuel densification report. Since Cycle 3 reload fuel assemblies contain fuel rods with theoretical density higher than those considered in this report, the conclusions derived in that report are still valid. Calculational techniques and methods for Cycle 3 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for the reload report. The dose considerations in the FSAR were based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

Startup Program

A startup program will be conducted to verify that the core performance is within the assumptions of the safety analyses and provide the necessary data for continued plant operation. The startup test program is similar to that previously approved for Cycle 2 operation.[6] Additionally, the program was discussed with the licensee for clarafication of control rod worth and power distribution measurements and comparison to predicted values. These measurements and comparisons will be performed by the licensee. The licensee also will provide a summary within 90 days following completion of physic testing. This startup test program is acceptable.

ECCS-U-Baffle Pressure Drop

By letter dated June 11, 1977, the licensee referenced a reanalysis of the ECCS performance with revised reactor coolant system pressure drop characteristics(14) using the same ECCS model previously approved for Oconee 2. This reanalysis was performed because of an identified error in the input value to the reactor vessel inlet nozzle U-baffle pressure loss characteristics. The reanalysis shows that lower peak cladding temperatures (PCT) would be obtained for the worst break analysis during a postulated LOCA. The trends of the break spectrum, sensitivity, and LOCA limits studies for the previously approved analysis for Oconee 2 remain valid. Therefore, only the limiting size break needed reanalysis. The licensee has confirmed that the reanalysis is appropriate to all three Oconee plants.

The reduction in PCT as compared to that for the generically approved ECCS analysis (B&W-10103) was due to the enhanced core flow during blowdown (more cooling), lower metal-water reaction rates (because of lower temperatures, less heat generation due to exothermic reaction), and improved reflooding of the core (cooling attained sooner). These benefits are based on an improved system pressure distribution, i.e., the reanalyzed RCS pressure drops are less than that assumed from B&W-10103.

The revision to the RCS pressure drops is based on both experimental and analytical verification techniques. Pressure drop measurements were made during the Oconee 1 hot functional testing. The other two Oconee plants are identical to Oconee 1, so that from this data the Once Through Steam Generator (OTSG) and reactor vessel pressure drops were established for all the Oconee plants. The pressure drop characteristics within the reactor vessel and the OTSG were then analytically established to match this data. Additionally, there were vessel model flow tests(15) which further substantiate the decrease in pressure drop observed in the hot functional test data and established by analysis. The reactor vessel inlet nozzle was originally assumed to be a long leg U-baffle, however, it is not. As shown by tests, the change in pressure drop for this component between originally assumed and-as-built conditions is substantial. All the changes to RCS pressure drops have been verified experimentally and analytically.

We have reviewed the RCS pressure drops and their impact on the ECCS performance analysis. We agree with the licensee that the ECCS calculations for the current Oconee 1, 2 & 3 fuel loading and those submitted by Units 1 and 2 are in compliance with the criteria of 10 CFR 50 Section 50.46 and Appendix K. Although the reanalysis has lower PCT than those of B&W-10103, the allowable Linear Heat Generation Rate (LHGR) limits for all Oconee plants will be maintained at the same values as previously approved. We find this analysis acceptable.

Surveillance Testing of Reactor Internal Vent Valves

The licensee proposed a change to the subject testing, such that, the required opening differential pressure for the reactor internals vent valves would be equivalent to 1.0 psid. The licensee has shown that this change has no significant effect on the peak cladding temperature (PCT) during the limiting LOCA, i.e., <3°F.(10) This is not a significant increase and does not cause the limiting LOCA PCT to exceed any of the 10 CFR 50.46 criteria, nor does this change affect which LOCA break is limiting. The licensee has supplied this information in reference 10. Based on this information and the continued surveillance requirements on the reactor internals vent valves we find this change to be acceptable.

The previously discussed analyses, which were presented as justification for operation, were conducted in compliance with NRC's regulations and approved methods and, furthermore, are conservative relative to NRC regulations. The proposed changes to the Technical Specifications are acceptable on the bases that the health and safety of the public will not be endangered by operation in the proposed manner.

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: July 29, 1977

References

1. Letter from W. O. Parker, Jr., (Duke Power Company) to Director NRR (NRC) dated May 6, 1977.

2. Letter from W. O. Parker, Jr., (Duke Power Company) to Director NRR, dated June 21, 1977.

3. Letter from W. O. Parker, Jr., (Duke Power Company) to B. C. Rusche, dated January 28, 1976.

4. "Irradiation of Two 17x17 Assemblies in Oconee 2, Cycle 2," BAW-1424, January 1976.

5. "Oconee Unit 2, Cycle 3-Reload Report", BAW-1452, April, 1977.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendments Nos. 27, 27 and 23 to Facility License Nos. DPR-38, DPR-41, and DPR-55 Duke Power Company, Oconee Nuclear Station 2, June 30, 1976.

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

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

9. "Technical Report on Densification of Babcock and Wilcox Reactor Fuels", ONRR, July 6, 1973.

10. Letter from W. O. Parker, Jr., (Duke Power Company) to Director NRR, dated May 5, 1977.

11. Letter from W. O. Parker, Jr., (Duke Power Company) to N. C. Moseley (NRC) dated March 10, 1977.

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

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

14. J. H. Taylor (Babcock & Wilcox) letter to R. L. Baer (NRC) dated July 8, 1977

15. Reactor Vessel Model Flow Tests, B&W-10012, October 1969

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 45, 45, and 42 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company (the licensee), which revised Technical Specifications for operation of the Oconee Nuclear Station Units Nos. 1, 2 and 3, (the facilities) located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications (1) to establish operating limits for Unit 2 Cycle 3 operation and (2) to establish requirements for testing reactor core internal vent valves.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. 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 50.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendments dated May 6, 1977, as supplemented June 21 and July 11, 1977, (2) application for amendments dated March 1, 1977, as supplemented May 5, 1977, (3) Amendments Nos.45 ,45 , and 42 to Licenses Nos. DPR-38, DPR-47 and DPR-55, and (4) 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), (3) and (4) 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 29th day of July 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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