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. 62
License No. DPR-55

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

   A. The applications for amendment by Duke Power Company (the licensee) dated April 20, 1978, and June 26, 1978, as supplemented April 27, August 21, 28, September 6, 22 and 26, 1978, 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 applications, 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.

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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 Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 23, 1978

<u>ATTACHMENTS TO LICENSE AMENDMENTS</u>

<u>AMENDMENT NO. 65 TO DPR-38</u>

<u>AMENDMENT NO. 65 TO DPR-47</u>

<u>AMENDMENT NO. 62 TO DPR-55</u>

<u>DOCKETS NOS. 50-269, 50-270 AND 50-287</u>

Revise Appendix A as follows:

Remove the following pages and insert the revised identically numbered pages.

2.1-2

2.1-7 (Figure 2.1-2A)

2.1-10 (Figure 2.1-3A)

2.3-8 (Figure 2.3-2A)

3.2-2

3.3-3 & 3.3-4

3.5-9

3.5-10

3.5-11

3.5-11a

3.5-11b* (Table 3.5-1)

3.5-12 (Figure 3.5.2-1A1)

3.5-13 (Figure 3.5.2-1A2)

3.5-18 (Figure 3.5.2-2A1)

3.5-18a (Figure 3.5.2-2A2)

3.5-21 (Figure 3.5.2-3A1)

3.5-21a* (Figure 3.5.2-3A2)

3.5-23c (Figure 3.5.2-4A1)

3.5-23d (Figure 3.5.2-4A2)

4.1-1

6.4-1

Changes on the revised pages are identified by marginal lines.

*New Page

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 2.1-1A 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.

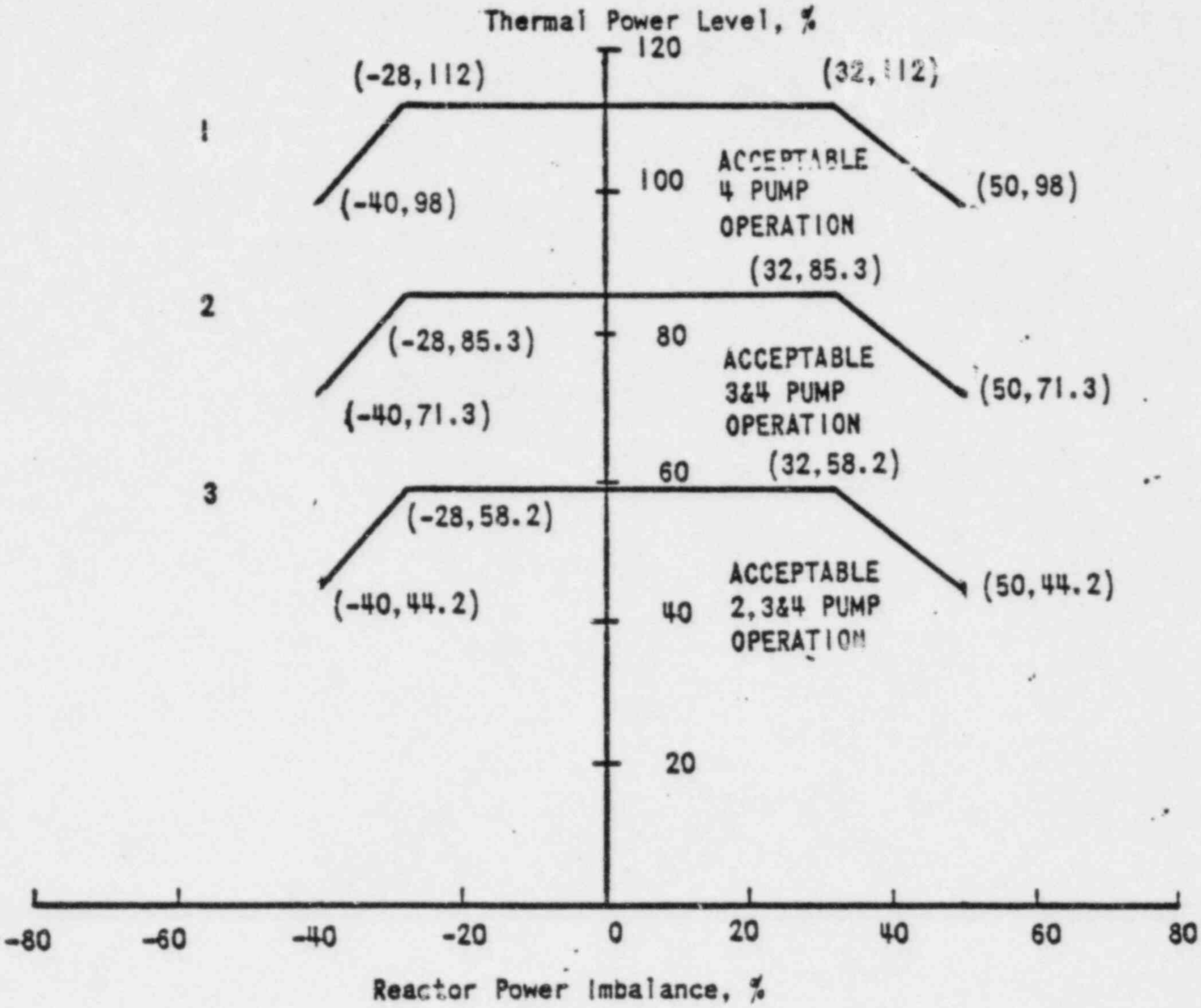
2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

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, and 3 of Figure 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1A 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.

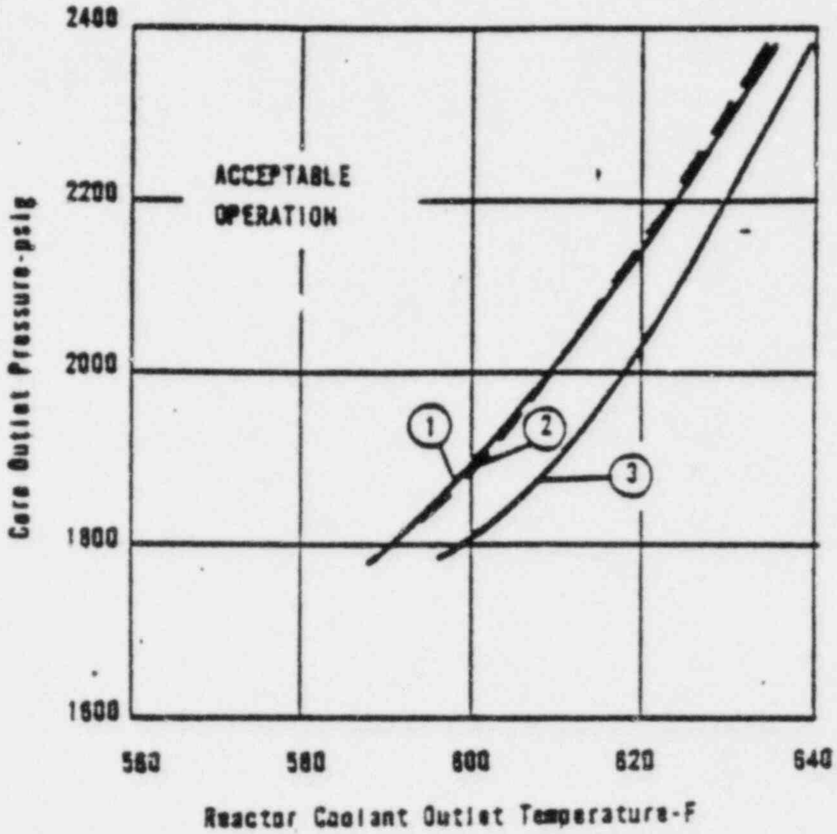


CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

CORE PROTECTION
SAFETY LIMITS
UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-2A

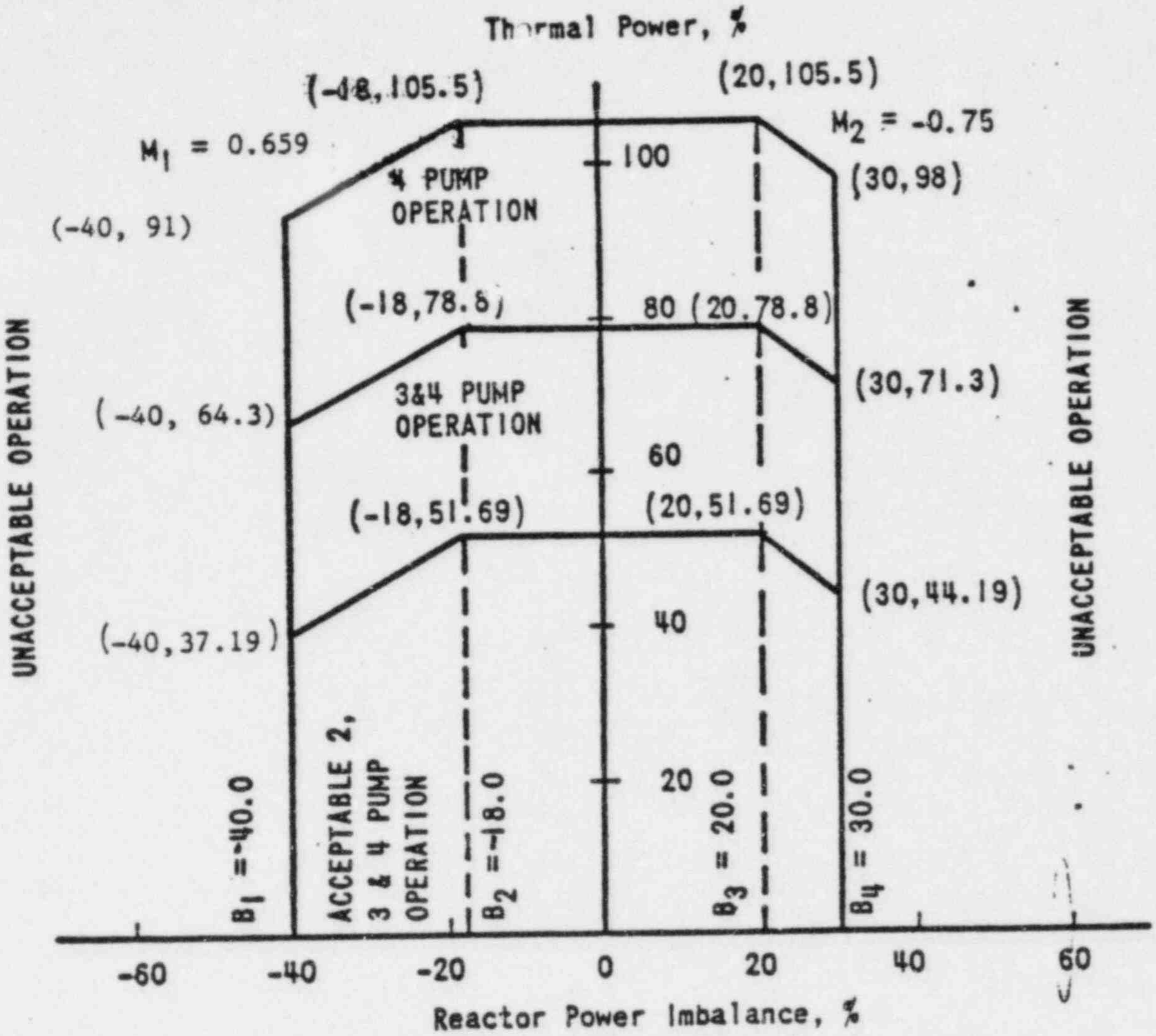


CURVE	REACTOR COOLANT FLOW (GPM)	POWER	PUMPS OPERATING	TYPE OF LIMIT
1	374880 (100%)*	112%	4	(DNBR)
2	280035 (74.7%)	85.3%	3	(DNBR)
3	183690 (49.0%)	58.2%	2	(QUALITY)

*106.5% of first core design flow

CORE PROTECTION
SAFETY LIMITS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1-3A

Amendments Nos. 65, 65 & 62



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 1

OCONEE NUCLEAR STATION

Figure 2.3-2A

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration. (1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank. (2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1%Δk/k subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit, Oconee 1 Cycle 5, Oconee 2, Cycle 3, and Oconee 3, Cycle 4 were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 980 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1800 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and in addition allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.25 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Section 9.1; 9.2
(2) FSAR, Figure 6.2
(3) Technical Specification 3.3

3.3.6 Exceptions to 3.3.5 shall be as follows:

(a) Both core flooding tanks shall be operational above 800 psig.

(b) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(c) One pressure instrument channel and one level instrument channel per core flood tank shall be operable above 800 psig.

(d) One reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days provided both reactor building spray pumps and associated spray nozzle headers are in service at the same time.

(e) If the requirements of Specification 3.3.1(f) are not met, the borated water storage tank shall be considered unavailable and action shall be initiated in accordance with Specification 3.2.

3.3.7 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

3.3.8 (a) Reactor power shall not be increased above 60% FP unless three HPI pumps and two HPI flow paths are operable.

(b) During power operation above 60% FP, tests or maintenance shall be allowed on any one HPI pump, provided two trains of the HPI system are operable. If the inoperable HPI pump is not restored to operable status within 72 hours, reactor power shall be reduced below 60% FP within an additional 12 hours.

(c) If during power operation above 60% FP a high pressure injection flow path becomes inoperable, reactor power shall be reduced below 60% FP within 12 hours.

Bases

The requirements of Specification 3.3 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two high pressure injection pumps and two low pressure injection pumps are required (except as specified in Specification 3.3.8 and as discussed further on in these bases.) However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core. (1)

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation. (2)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70^0F without any control rods in the core. This concentration is 1,338 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5 and 3.3.6 provided requirements in Specification 3.3.7 are met which assure operability of the duplicate components. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal. The basis of acceptability is a likelihood of failure within 24 hours following such demonstration.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the reactor building design pressure will not be exceeded with one spray and two coolers operable. Therefore, a maintenance period of seven days is acceptable for one reactor building cooling fan and its associated cooling unit. (3)

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop serverence, limit the peak clad temperature to less than $2,200^0$F and the metal-water reaction to that representing less than 1 percent of the clad.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Units 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

The requirement to have three HPI pumps and two HPI flow paths operable during power operation above 60% FP (Specification 3.3.8) is based on considerations of a 0.04 square foot break at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling. The analysis of this break indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

REFERENCES

(1) FSAR, Section 14.2.2.3
(2) FSAR, Section 9.5.2
(3) FSAR, Supplement 13

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 test, 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); 3.5.3-2B1, 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, and 3.5.2-4A2 (Unit 1); 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

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.

3.5.2.6 Xenon Reactivity

a. Except for physics tests, reactor power in Unit 1 shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, and 3.5.2-1A2 unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours in the soluble poison control mode.

b. Except for physics tests, reactor power in Units 2 and 3 shall not be increased above the power level cutoff shown in Figures 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2); 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.

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.8 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 power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

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

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

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

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument 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 to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding
7.50% for Unit 1. The limits shown in Specification 3.5.2.4
5.10% for Unit 2
7.50% 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 3.5.2.7, 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 3.5.2.6 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)

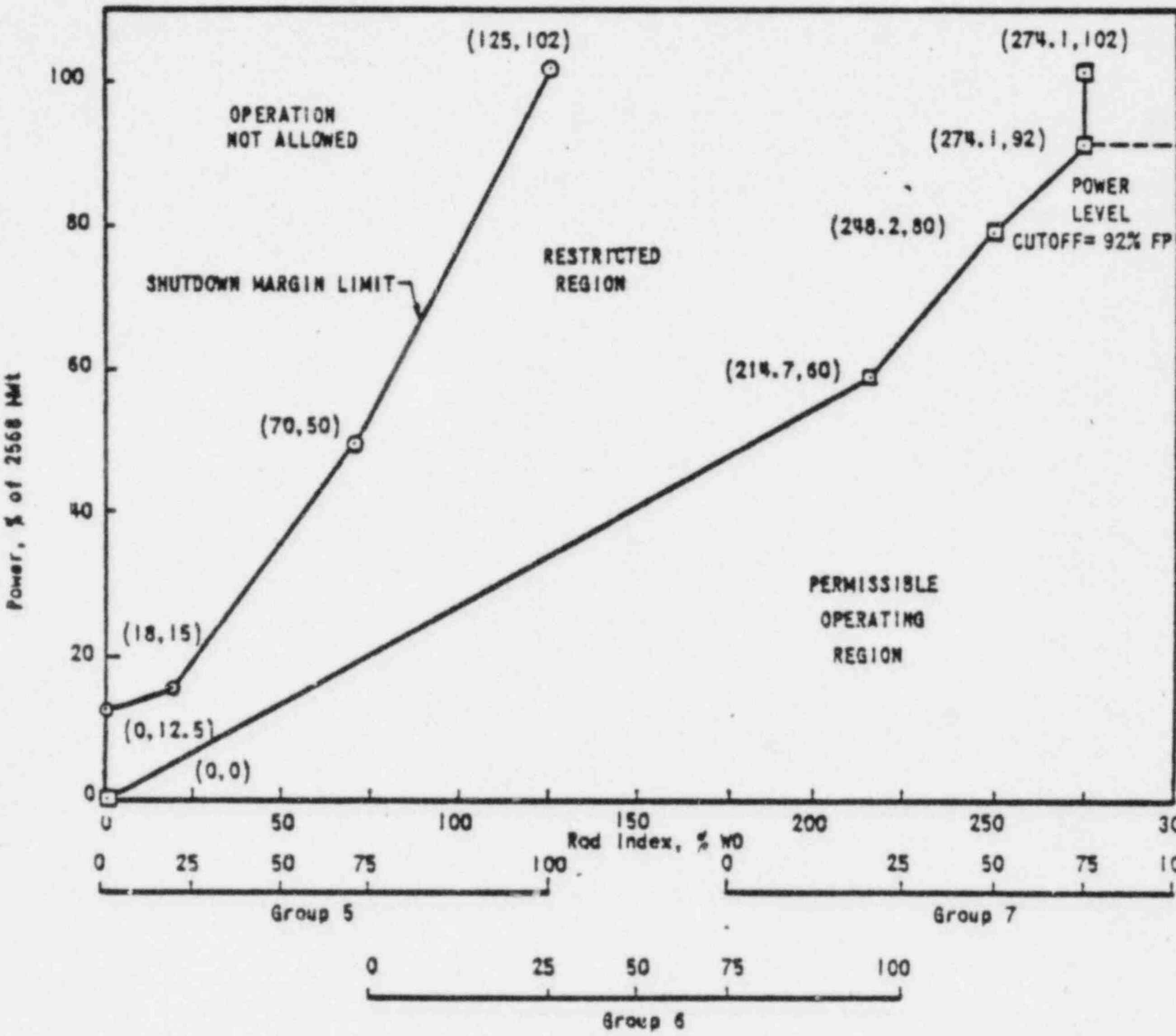
BAW-1400 (UNIT 3)

[5]Oconee 1, Cycle 4 - Reload Report - BAW-1447, March 1977, Section 7.11.

TABLE 3.5-1

Quadrant Power Tilt Limits

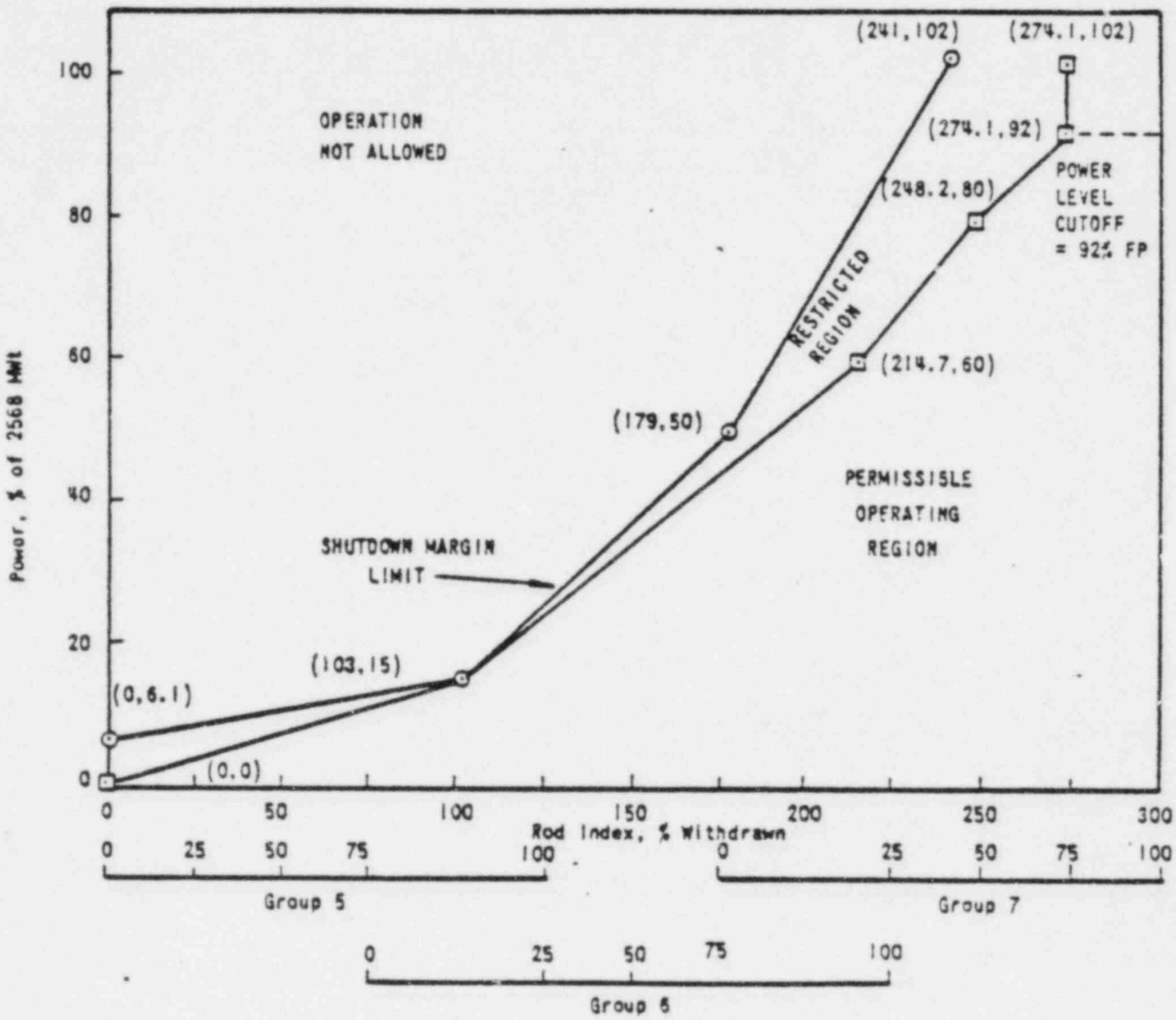
	Steady State Limit	Transient Limit	Maximum Limit
Unit 1	5.00	9.44	20.0
Unit 2	3.41	9.44	20.0
Unit 3	5.00	9.44	20.0



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 100 $\pm$ 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

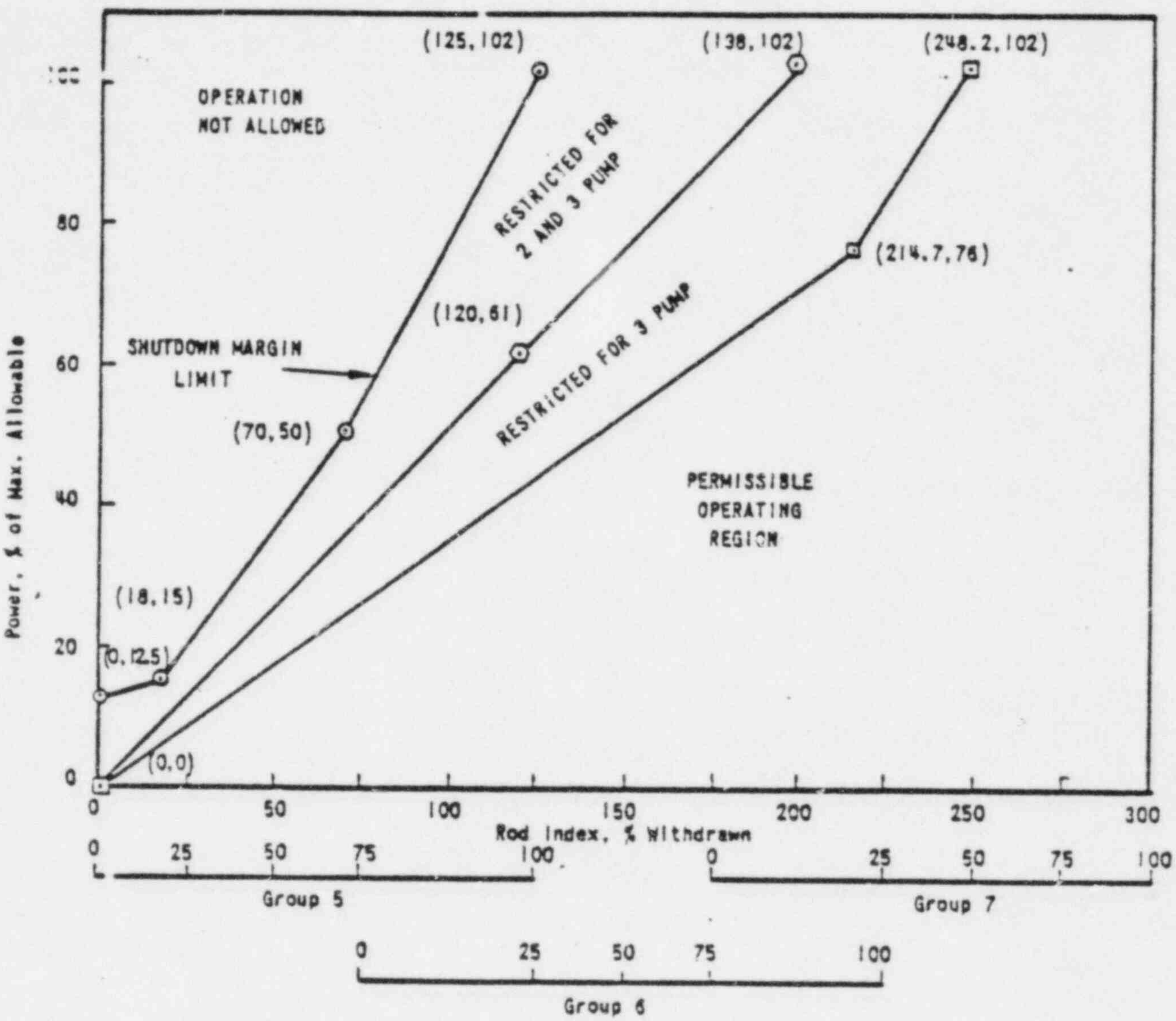
Figure 3.5.2-1A1



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

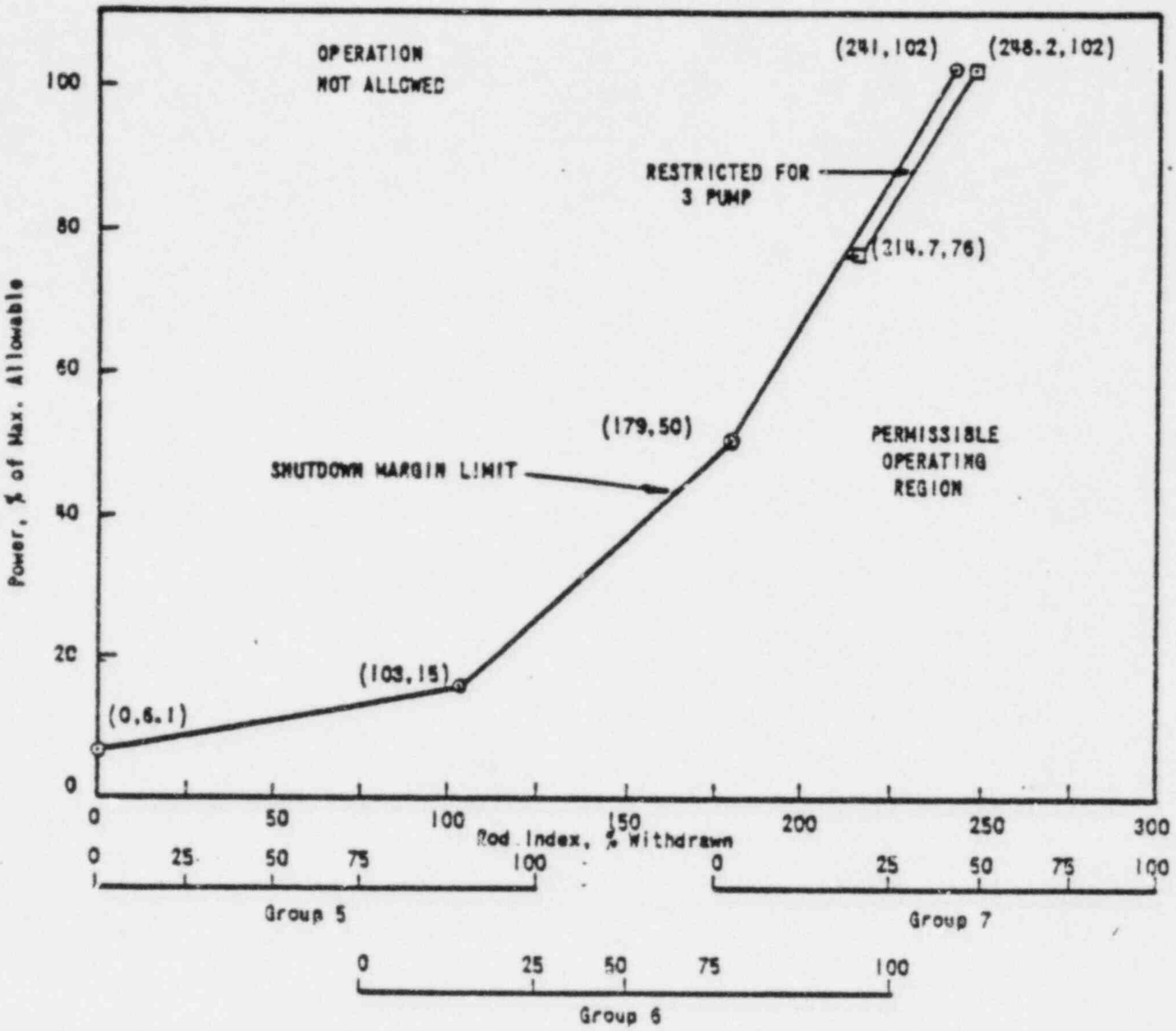
Figure 3.5.2-1A2



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

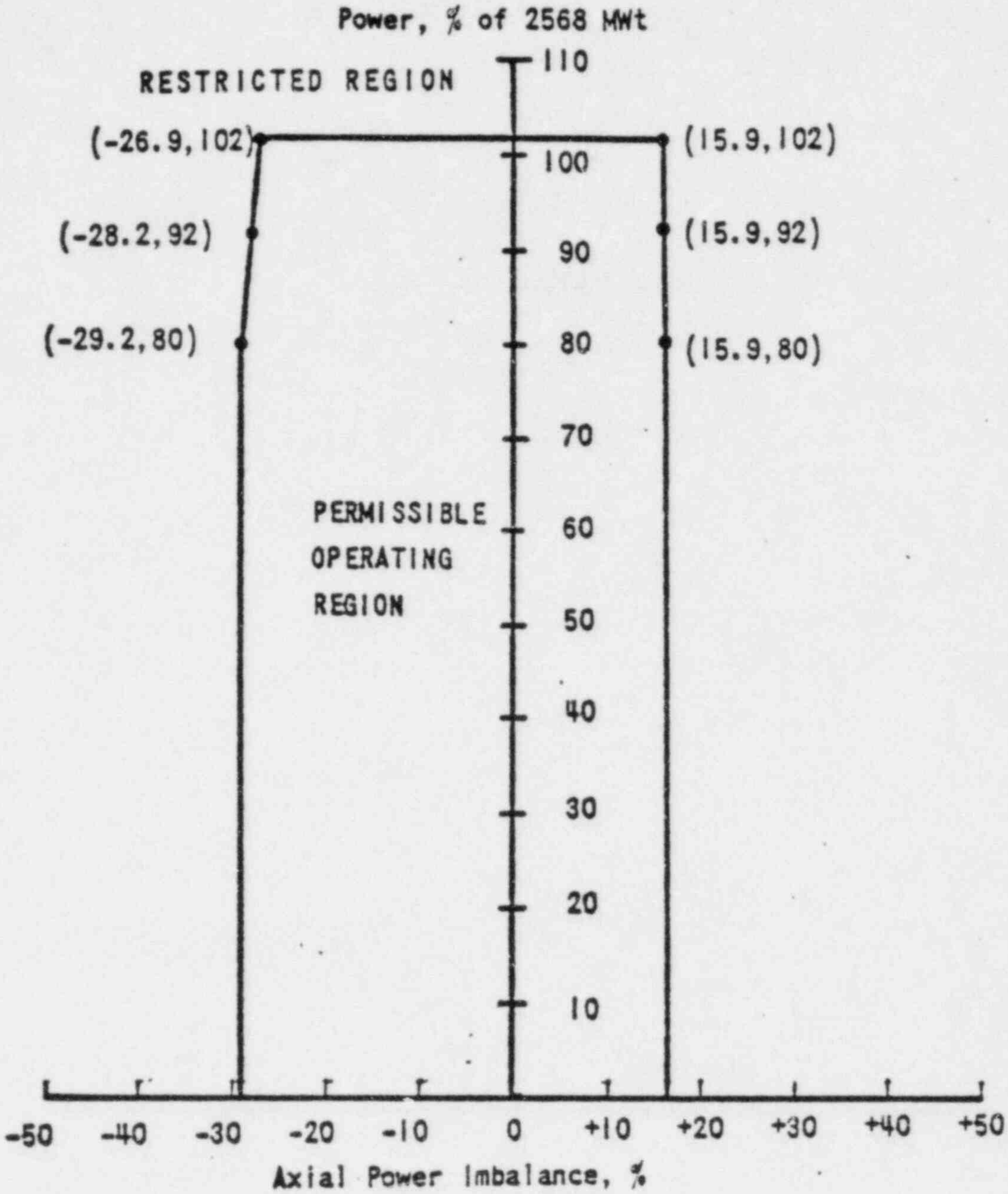


ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
AFTER 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-2A2

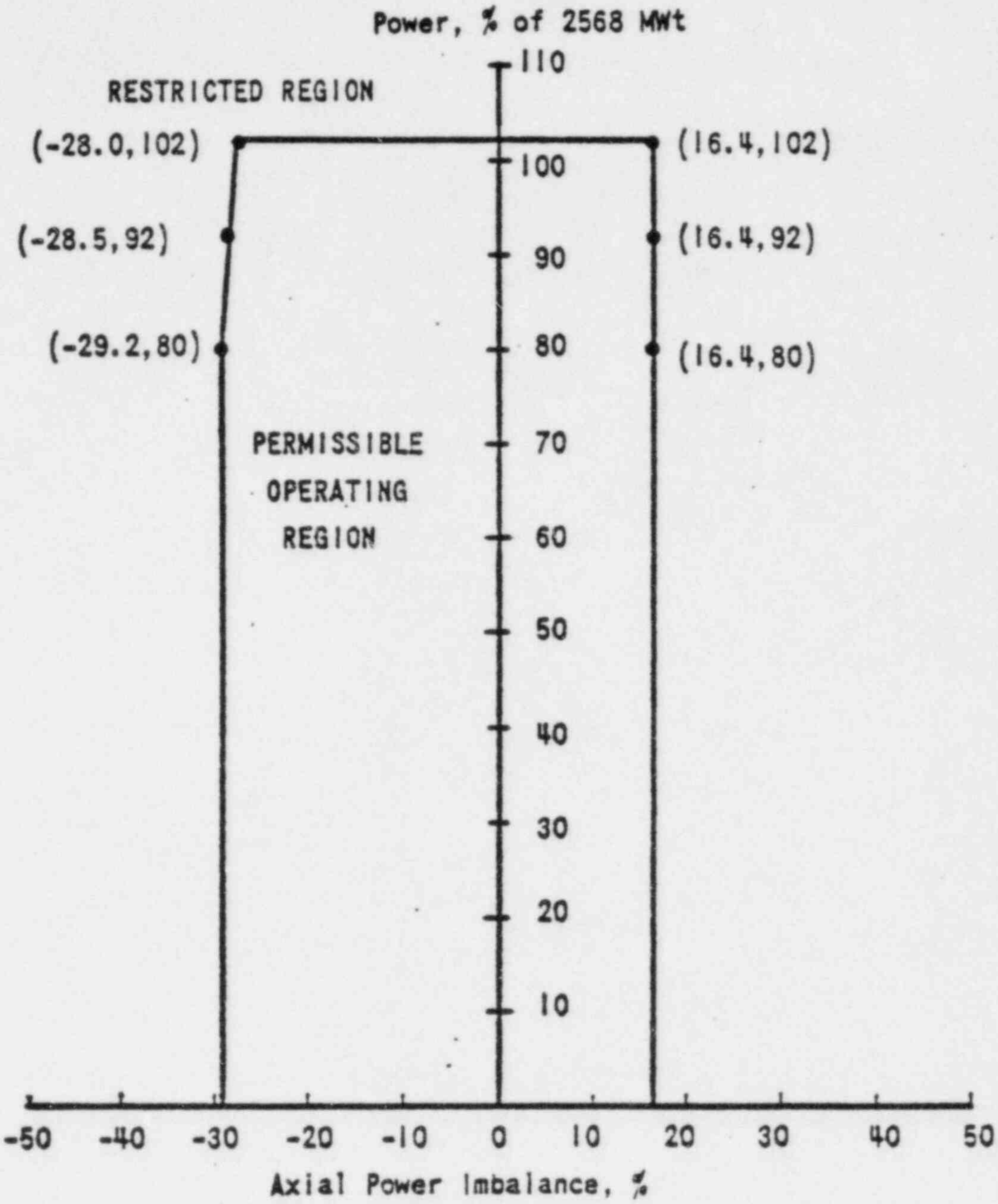
Amendments Nos. 65, 65 & 62



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 to 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

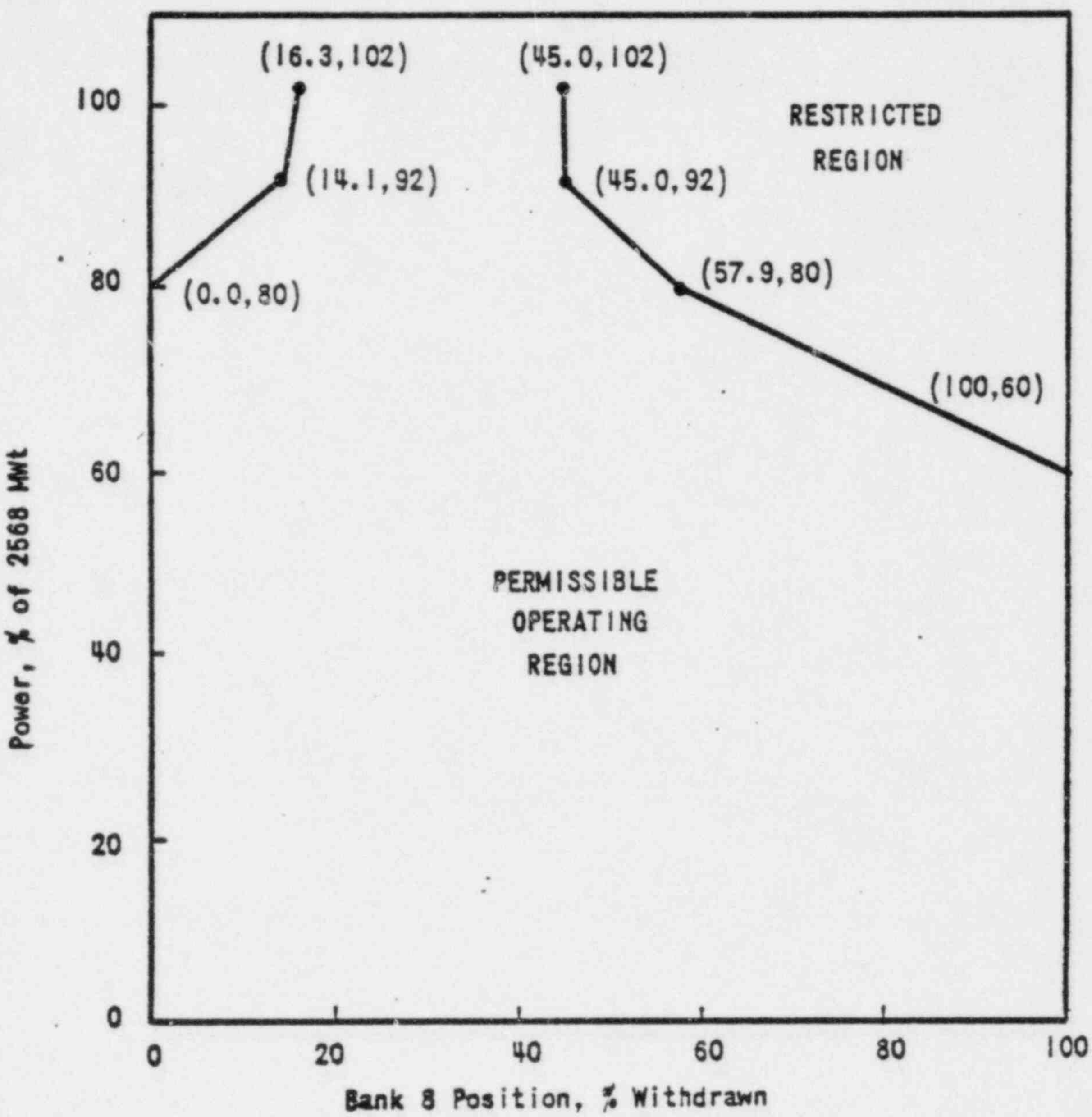
Figure 3.5.2-3A1



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 100 $\pm$ 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

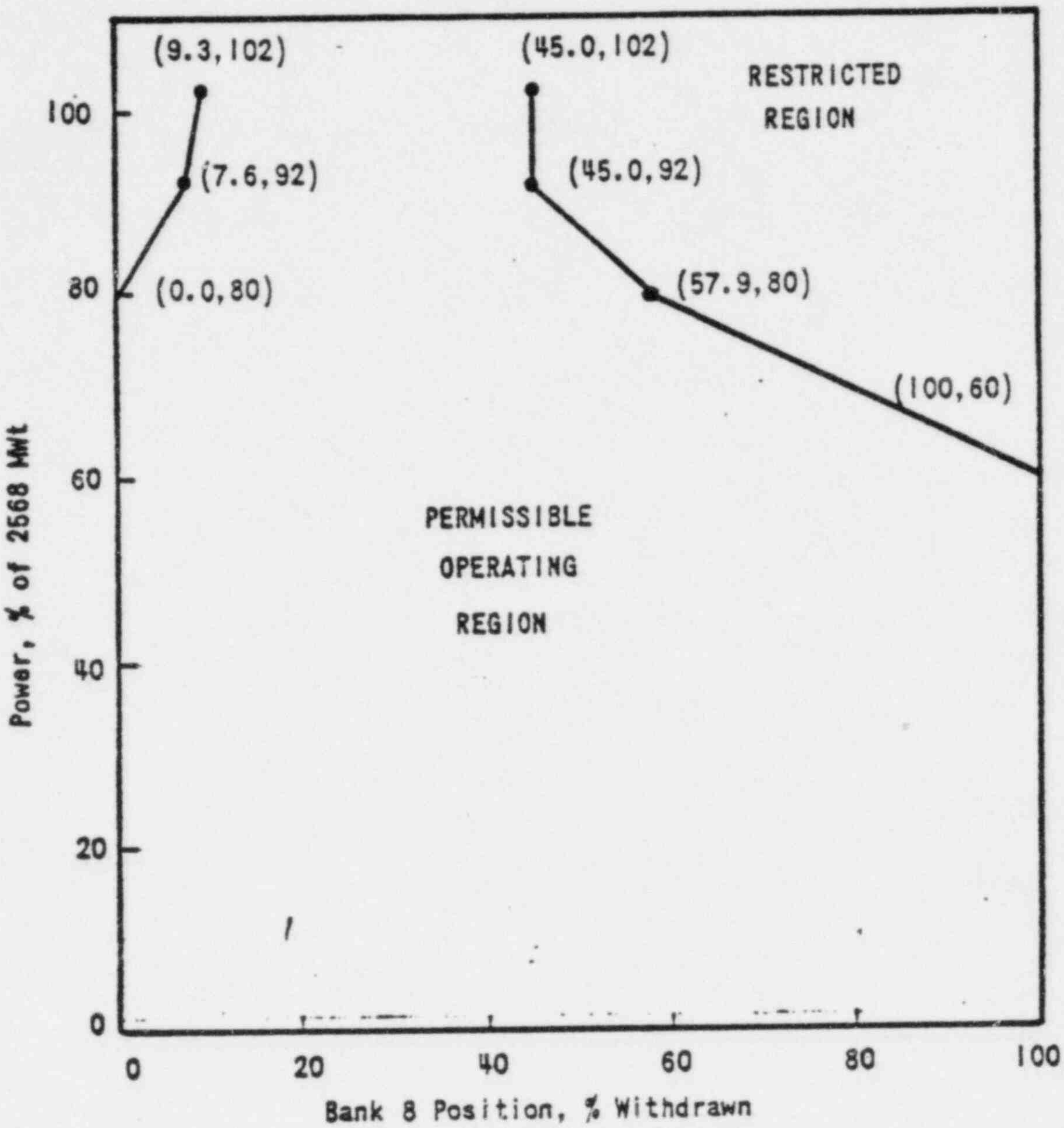
Figure 3.5.2-3A2



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 AFTER 100 ± 10 EFPD
Unit 1

OCONEE NUCLEAR STATION

Figure 3.5.2-4A2

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

6.4 STATION OPERATING PROCEDURES

Specification

6.4.1 The station shall be operated and maintained in accordance with approved procedures. Written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel.

f. Station survey following an earthquake.

g. Radiation control procedures.

h. Operation of radioactive waste management systems.

i. Control of pH in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

j. Nuclear safety-related periodic test procedures.

k. Long-term emergency core cooling systems. Procedures shall include provision for remote or local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break.

l. Fire Protection Program implementation.

6.4.2 Quarterly selected drills shall be conducted on site emergency procedures including assembly preparatory to evacuation off site and a check of the adequacy of communications with off-site support groups.

6.4.3 A respiratory protective program approved by the Commission shall be in force.