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

November 30, 1981

Dockets Nos. 50-269, 50-270
and 50-287

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Amdt. 105
to DPR-47

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 105 , 105 , and 102 to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your applications dated May 29 and August 25, 1981, as supplemented by letter dated October 16, 1981.

These amendments revise the TSs to allow full power operation of Unit 1 for fuel Cycle 7, reflect completed modifications to the high pressure injection system and revisions to the boron concentration requirements.

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

Sincerely,

Philip C. Wagner

Philip C. Wagner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 105 to DPR-38
2. Amendment No. 105 to DPR-47
3. Amendment No. 102 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

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

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 (Unit 1); 3.5.2-1B1, and 3.5.2-1B2 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1, 3.5.2-2A2, and 3.5.2-2A3 for three pump operation and 3.5.2-2A4, 3.5.2-2A5, and 3.5.2-2A6 for two pump operation (Unit 1); 3.5.2-2B1, and 3.5.2-2B2 (Unit 2); and 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, and 3.5.2-4B2, (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.

Duke Power Company

cc w/enclosure(s):

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5/29/81, 8/25/81, 10/16/81

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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. 105
License No. DPR-47

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 29 and August 25, 1981, as supplemented October 16, 1981, 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.

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-47, 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. 105 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 30, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 105TO DPR-38

AMENDMENT NO. 105TO DPR-47

AMENDMENT NO. 102TO DPR-55

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

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
2.1-2	2.1-2
2.1-3	2.1-3
2.1-7	2.1-7
2.1-10	2.1-10
2.3-2	2.3-2
2.3-3	2.3-3
2.3-8	2.3-8
3.2-1	3.2-1
3.2-2	3.2-2
3.3-1	3.3-1
3.3-2	3.3-2
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	——

REMOVE PAGES	INSERT PAGES
3.5-9	3.5-9
3.5-10	3.5-10
3.5-15	3.5-15
3.5-15a	3.5-15a
—	3.5-15b
3.5-18	3.5-18
3.5-18a	3.5-18a
—	3.5-18b
—	3.5-18c
—	3.5-18d
—	3.5-18e
3.5-21	3.5-21
3.5-21a	3.5-21a
—	3.5-21b
3.5-24	3.5-24
3.5-24a	3.5-24a
—	3.5-24b
3.8-3	3.8-3
4.4-12	4.4-12

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 20.05 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 magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis (3).

All plant operating limits are presently based on an original method of calculating rod bow penalties that are more conservative than those that would be obtained with new approved procedures (3). For Cycle 7 operation, this subrogation results in a 10% DNBR margin, which is partially used to offset the reduction in DNBR due to fuel rod bowing.

The maximum thermal power for three-pump operation is 89.899 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.929 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is produced in a similar manner.

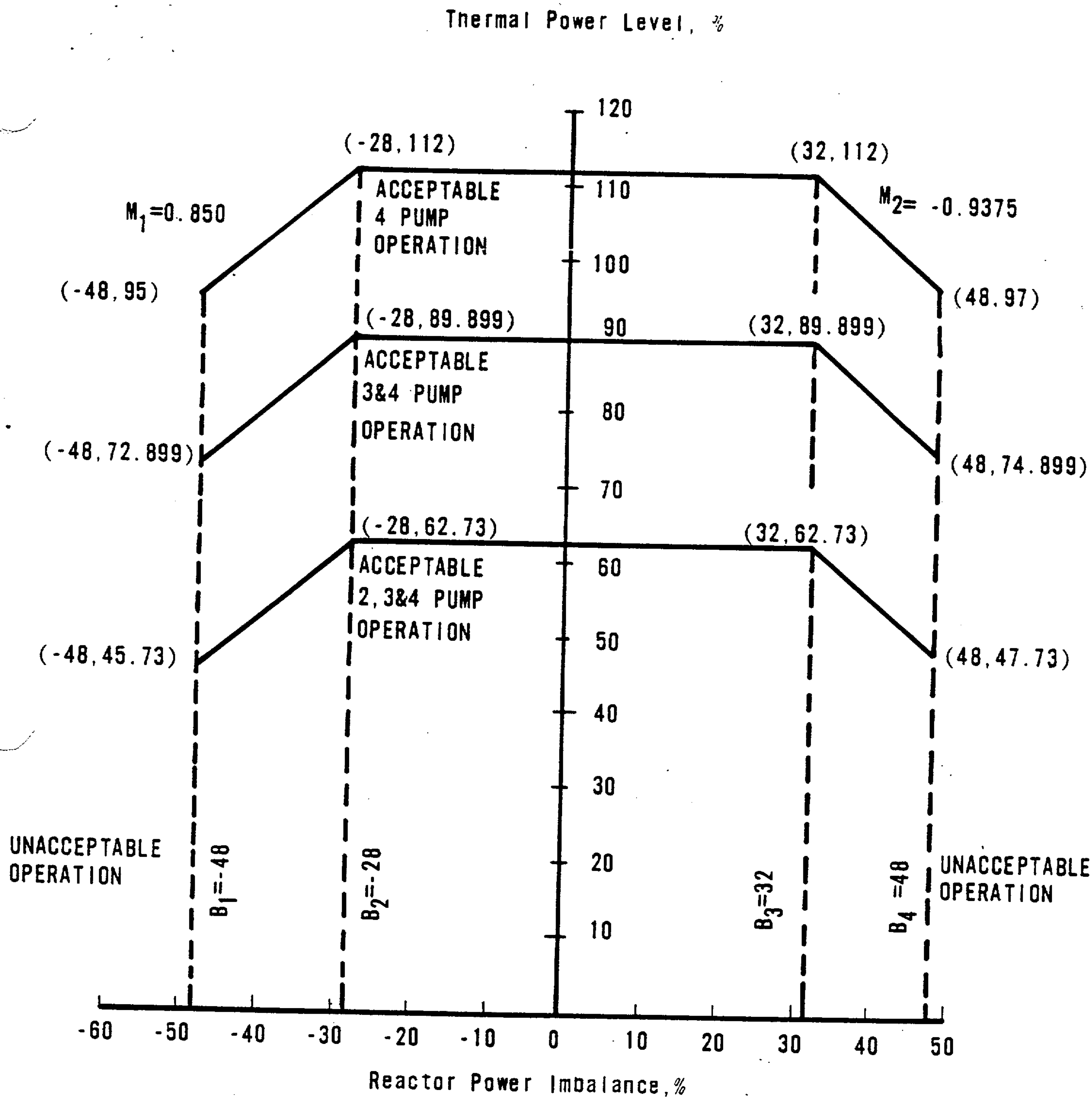
For Figure 2.1-3A, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30.

References

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

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

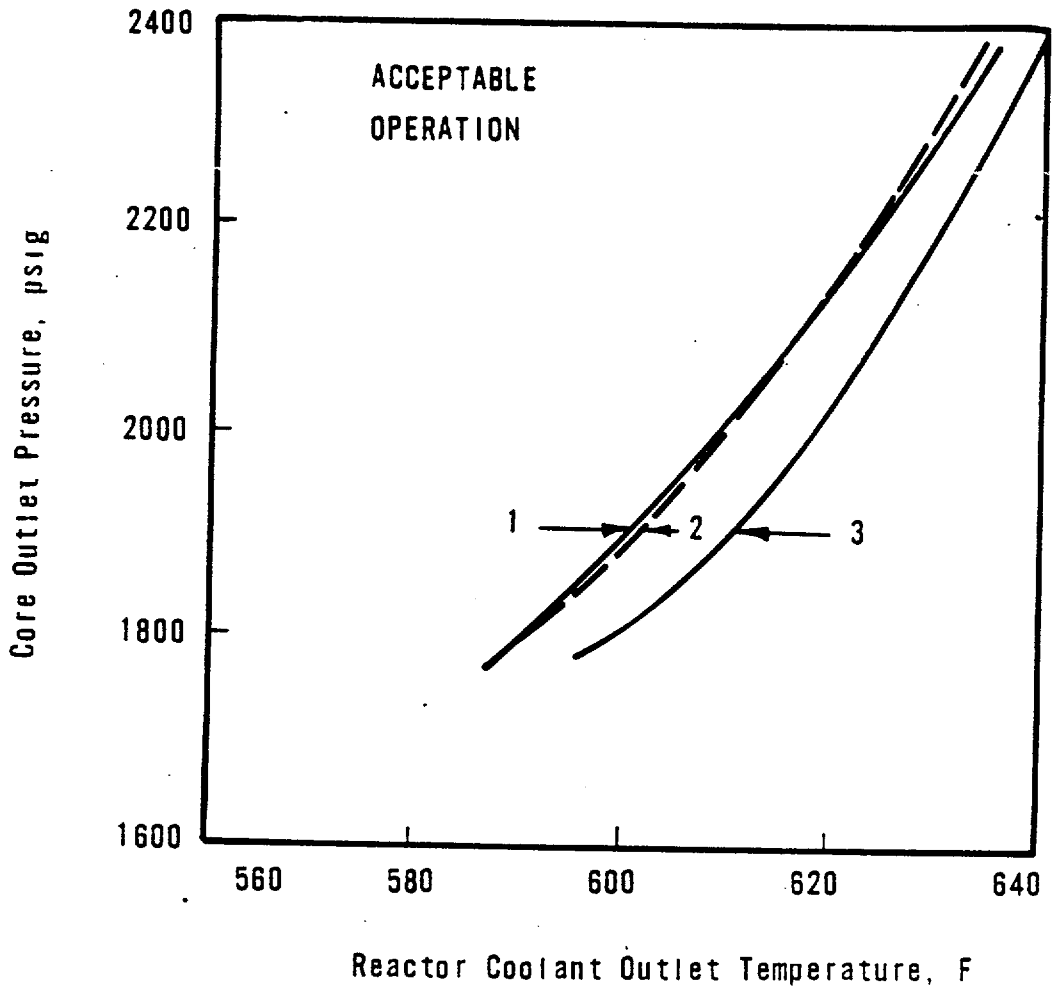
(3) Oconee 1, Cycle 7 - Reload Report - BAW-1660, March, 1981.



CORE PROTECTION
SAFETY LIMITS
UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-2A



CURVE	COOLANT FLOW, GPM	POWER, %	PUMPS OPERATING	TYPE OF LIMIT
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	89.899	3	DNBR
3	183,690 (49.0%)	62.73	2	QUALITY

*106.5% OF FIRST CORE DESIGN FLOW

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

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

The power level trip setpoint produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.92% and reactor flow rate is 74.7% or flow rate is 70.09% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.43% and reactor flow rate is 49.0% or flow rate is 45.79% and the power level is 49%.

   The flux-to-flow ratios account for the maximum calibration and instrument errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% - Unit 1 for 1% flow reduction.
1.08% - Unit 2
1.08% - Unit 3

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 setpoint is reached before the nuclear overpower trip setpoint. 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 (2300 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 ($11.14\ T_{out}$-4706)
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)

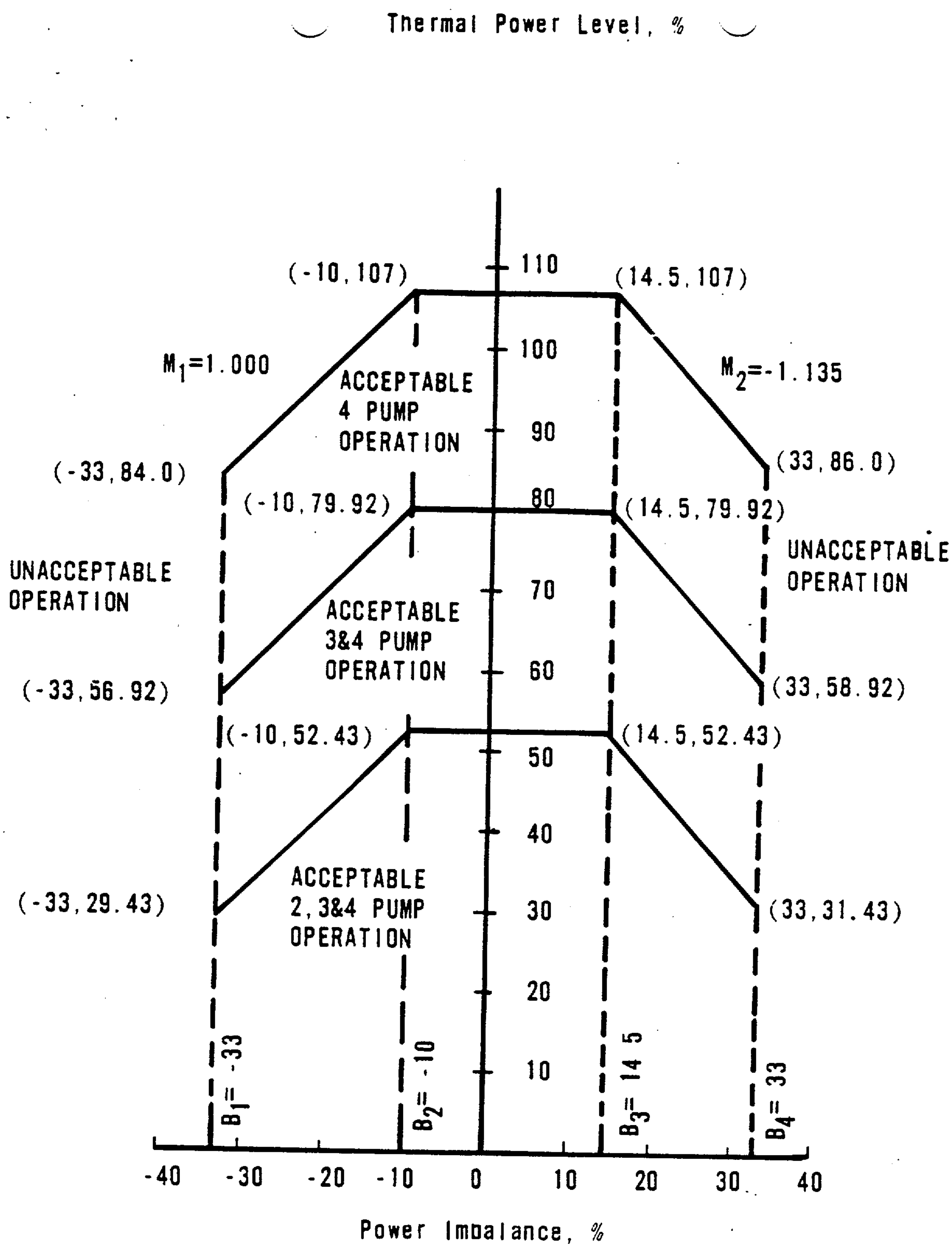
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperature in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

Reactor Building Pressure

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



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

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable.

This source will be the concentrated boric acid storage tank containing at least the equivalent of 1020 ft^3 of 8700 ppm boron as boric acid solution with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition and be borated to a shutdown margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

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% $\Delta k/k$ 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 7, Oconee 2, Cycle 5, and Oconee 3, Cycle 6 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 1020 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835 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 12.7 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 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY, AND LOW PRESSURE SERVICE WATER SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building cooling, reactor building spray, and low pressure service water systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling, reactor building spray and low pressure service water systems.

Specification

3.3.1 High Pressure Injection (HPI) System

a. Prior to initiating maintenance on any component of the HPI system, the redundant component shall be tested to assure operability.

b. When the reactor coolant system (RCS), with fuel in the core, is in a condition with temperature above 350°F and reactor power less than 60% FP:

(1) Two independent trains, each comprised of an HPI pump and a flow path capable of taking suction from the borated water storage tank and discharging into the reactor coolant system automatically upon Engineered Safeguards Protective System (ESPS) actuation (HPI segment) shall be operable.

(2) Test or maintenance shall be allowed on any component of the HPI system provided one train of the HPI system is operable. If the HPI system is not restored to meet the requirements of Specification 3.3.1.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS temperature below 350°F within an additional 24 hours.

c. For all Units, when reactor power is greater than 60% FP:

(1) In addition to the requirements of Specification 3.3.1.b(1) above, the remaining HPI pump and valves 3HP-409 and 3HP-410 shall be operable and valves HP-99 and HP-100 shall be open.

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

3.3.2 Low Pressure Injection (LPI) System

a. Prior to initiating maintenance on any component of the LPI system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two independent LPI trains, each comprised of an LPI pump and a flowpath capable of taking suction from the borated water storage tank and discharging into the RCS automatically upon ESPS actuation (LPI segment), together with two LPI coolers and two reactor building emergency sump isolation valves (manual or remote-manual) shall be operable.

(2) Tests or maintenance shall be allowed on any component of the LPI system provided the redundant train of the LPI system is operable. If the LPI system is not restored to meet the requirements of Specification 3.3.2.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.2.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.3 Core Flood Tank (CFT) System

When the RCS is in a condition with pressure above 800 psig both CFT's shall be operable with the electrically operated discharge valves open and breakers locked open and tagged; a minimum level of 13 ± .44 feet (1040 ± 30 $ft.^3$) and one level instrument channel per CFT; a minimum concentration of borated water in each CFT of 1835 ppm boron; and pressure at 600 ± 25 psig with one pressure instrument channel per CFT.

3.3.4 Borated Water Storage Tank (BWST)

When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

a. The BWST shall have operable two level instrument channels.

(1) Tests or maintenance shall be allowed on one channel of BWST level instrumentation provided the other channel is operable.

(2) If the BWST level instrumentation is not restored to meet the requirements of Specification 3.3.4.a above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.4.a are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of 1835 ppm boron at a minimum temperature of 50°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

3.3.5 Reactor Building Cooling (RBC) System

a. Prior to initiating maintenance on any component of the RBC system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on any component of the RBC system provided one train of the RBC and one train of the RBS are operable. If the RBC system is not restored to meet the requirements of Specification 3.3.5.b(1) above within 24 hours, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

c. When the reactor is critical:

(1) In addition to the requirements of Specifications 3.3.5.b(1) above, the remaining RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on one RBC train under either of the following conditions:

(a) One RBC train may be out of service for 24 hours.

(b) One RBC train may be out of service for 7 days provided both RBS trains are operable.

(c) If the inoperable RBC train is not restored to meet the requirements of Specification 3.3.5.c(1) within the time permitted by Specification 3.3.5.c(2)(a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.5.c(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.6 Reactor Building Spray (RBS) System

a. Prior to initiating maintenance on any component of the RBS system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) One RBS train, comprised of an RBS pump and a flowpath capable of taking suction from the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation (RBS segment) shall be operable.

(2) Tests or maintenance shall be allowed on any component of the RBS system under the following conditions:

(a) One RBS train may be out of service for 24 hours provided two RBC train are operable.

(b) If the inoperable RBS train is not restored to meet the requirements of Specification 3.3.6.b(1) within 24 hours, the reactor shall be placed in a condition with the RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

c. When the reactor is critical:

(1) In addition to the requirements of Specifications 3.3.6.b(1) above, the other RBS train comprised of an RBS pump and a flowpath capable of taking suction of the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation (RBS segment) shall be operable.

(2) Tests or maintenance shall be allowed on one RBS train under either of the following conditions:

(a) One RBS train may be out of service for 24 hours.

(b) One RBS train may be out of service for 7 days provided all three RBC trains are operable.

(c) If the inoperable RBS train is not restored to meet the requirements of Specification 3.3.6.c(1) above within the time permitted by Specification 3.3.5.c(2)(a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.6.c(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.7 Low Pressure Service Water (LPSW)

a. Prior to initiating maintenance on any component of the LPSW system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two LPSW pumps for the shared Unit 1, 2 LPSW system and two LPSW pumps for the Unit 3 LPSW system shall be operable with valves LPSW-108, 2LPSW-108, and 3LPSW-108 locked open.

(2) Tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. If the LPSW system is not restored to meet the requirements of Specification 3.3.7.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.7.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250° within an additional 24 hours.

<u>Bases</u>

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad. (1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks 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.(2) The analysis of these breaks 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.

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.(3)

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 above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. The minimum value specified in the tanks is 1835 ppm boron.

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. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

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.

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

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.5.2

(4) FSAR, Supplement 13

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 for Unit 1; Figures 3.5.2-1B1, and 3.5.2-1B2, for Unit 2; and Figures 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 for Unit 3 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.

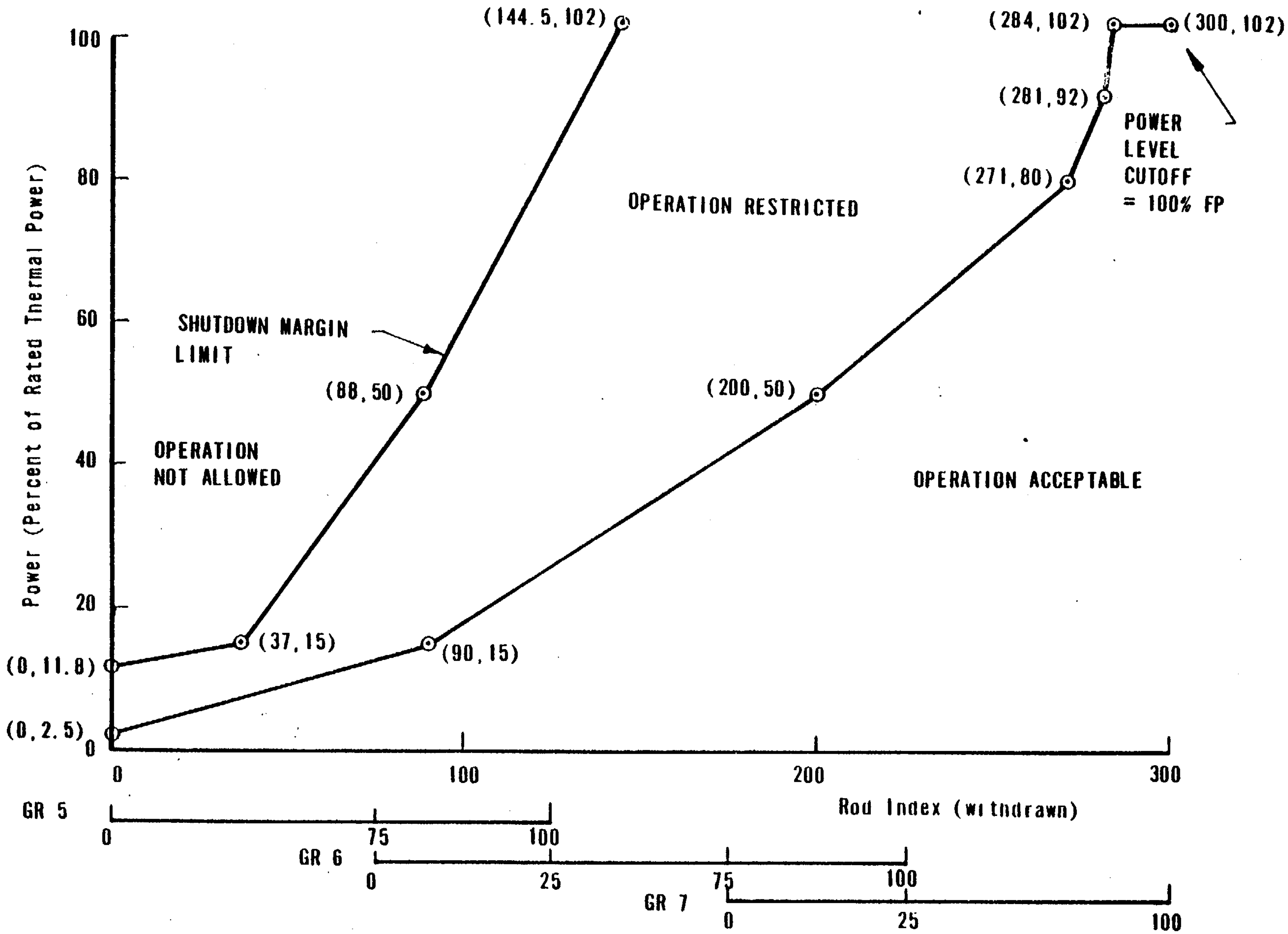
3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. 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.

3.5.2.9 The operational limit curves of Technical Specifications 3.5.2.5.c and 3.5.2.7 are valid for a nominal design cycle length, as defined in the Safety Evaluation Report for the appropriate unit and cycle. Operation beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are <u>not</u> valid for the extended period of the operation, appropriate limits will be established and the Technical Specification curves will be modified as required.

Amendments Nos. 105 , 105 & 102

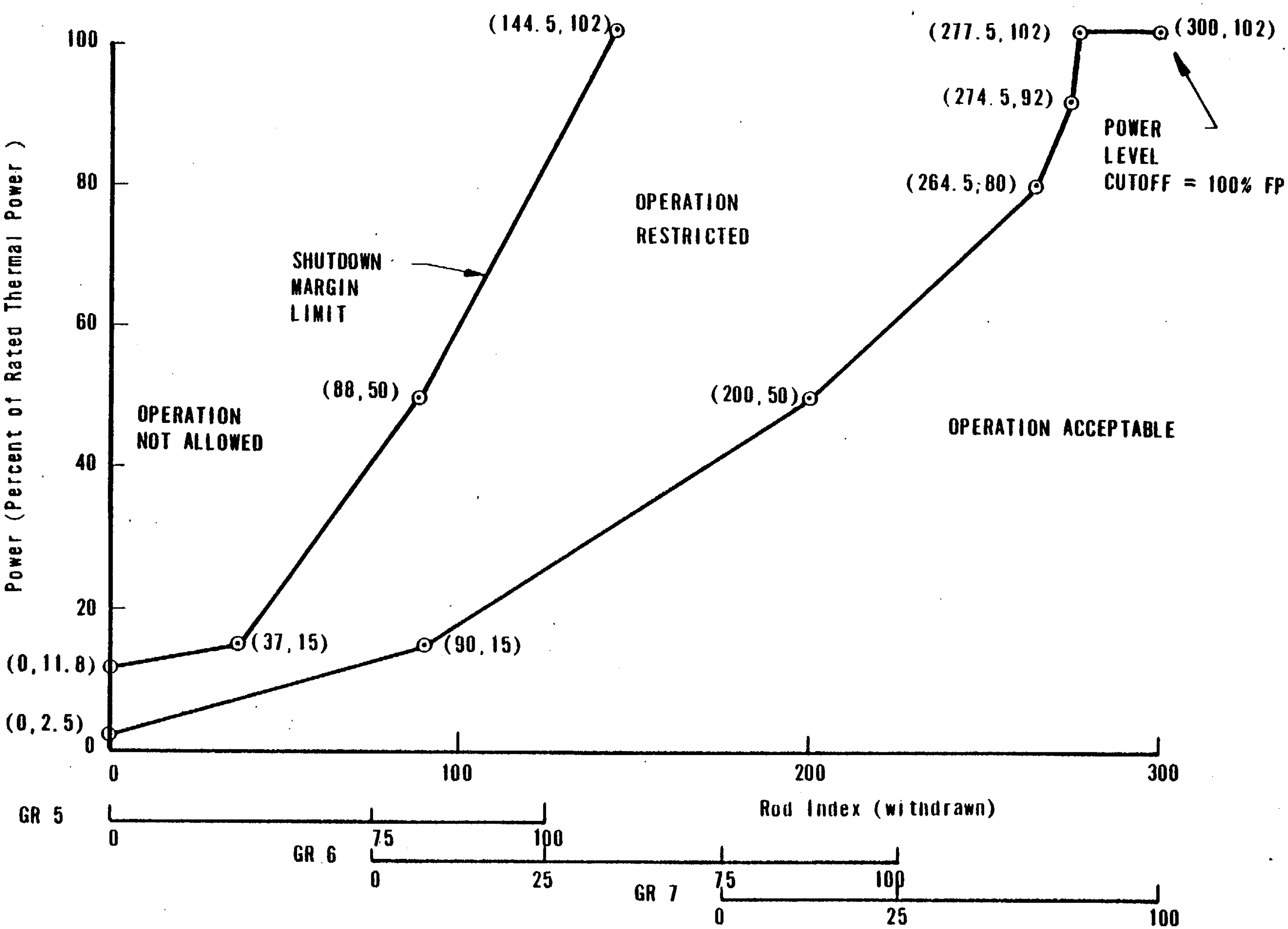
3.5-15



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 1

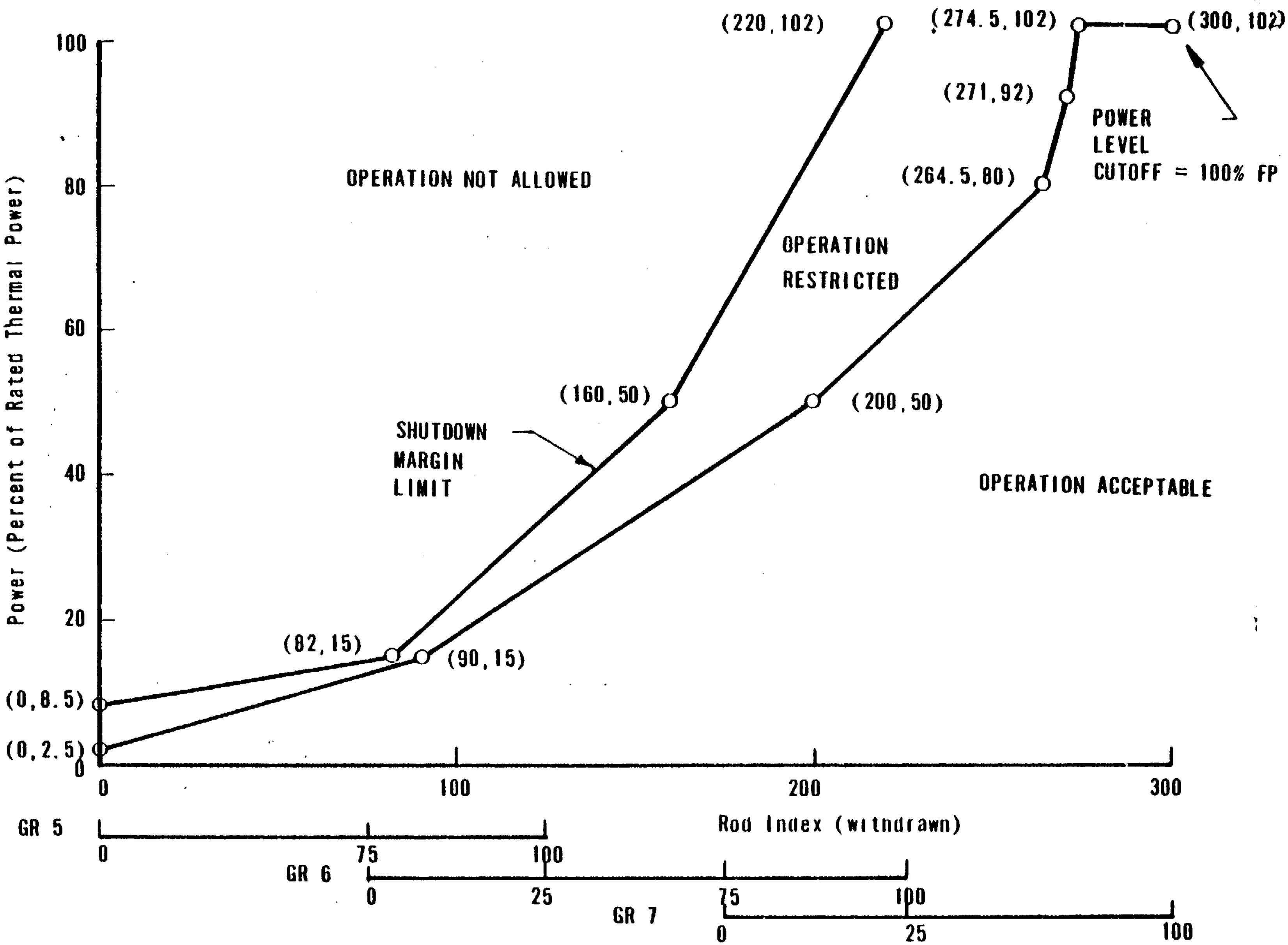
OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 1A1



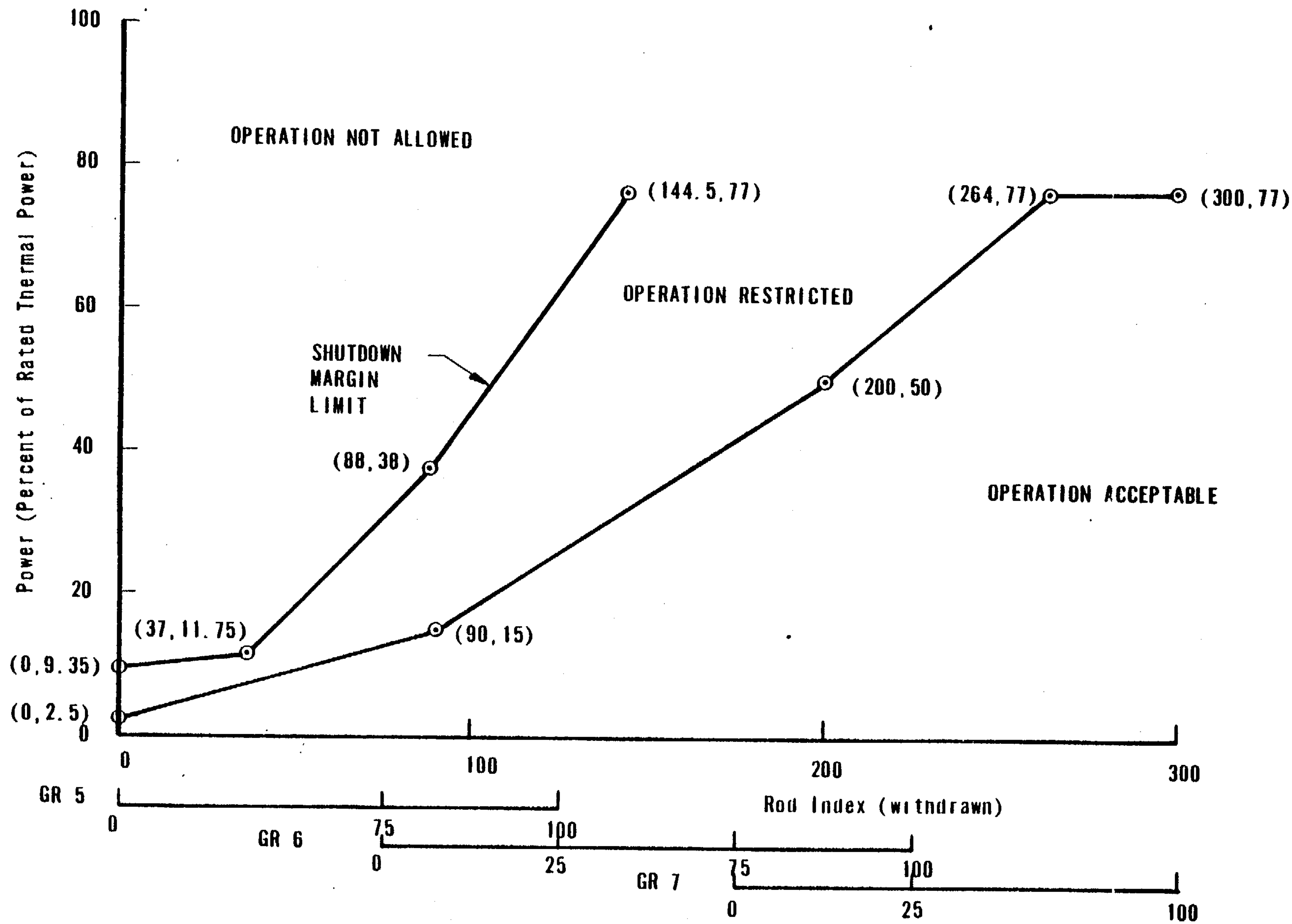
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 50 (+10, -0) to 200 $\pm$ 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 1

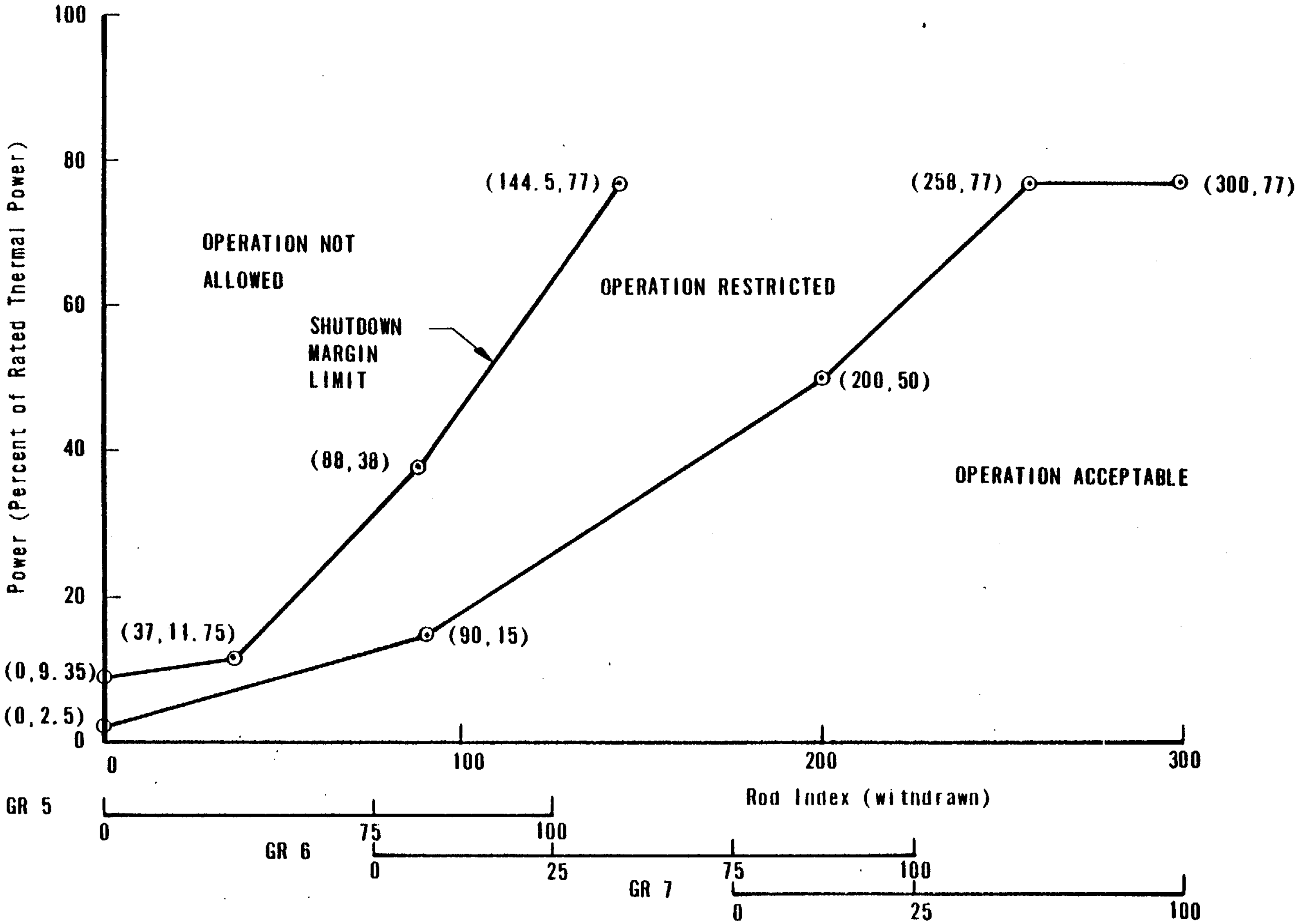
OCONEE NUCLEAR STATION



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 1

OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 2A1



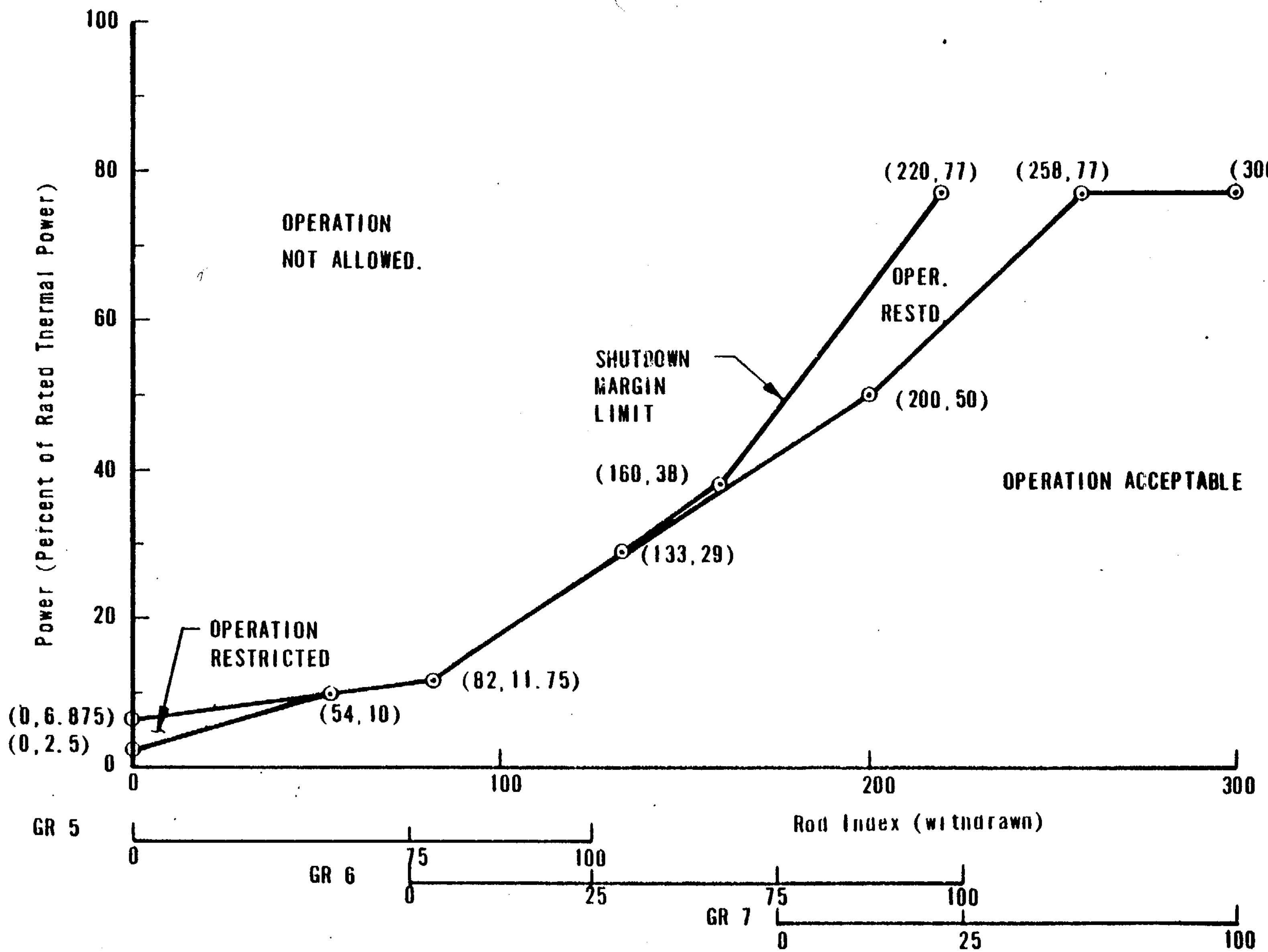
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 50 (+10, -0) to 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 2A2

Amendments Nos. 105 , 105 & 102

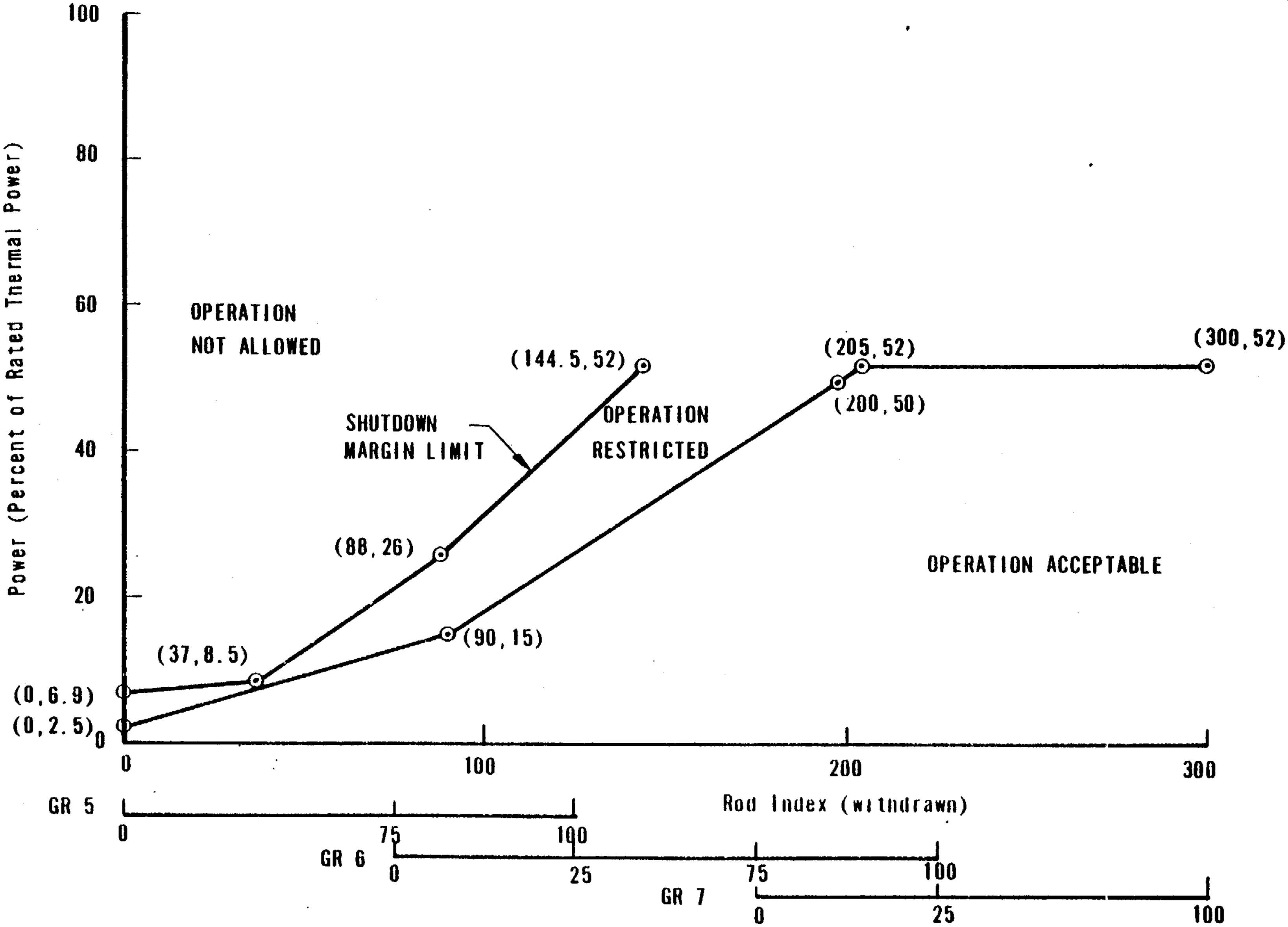
3.5-18a



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

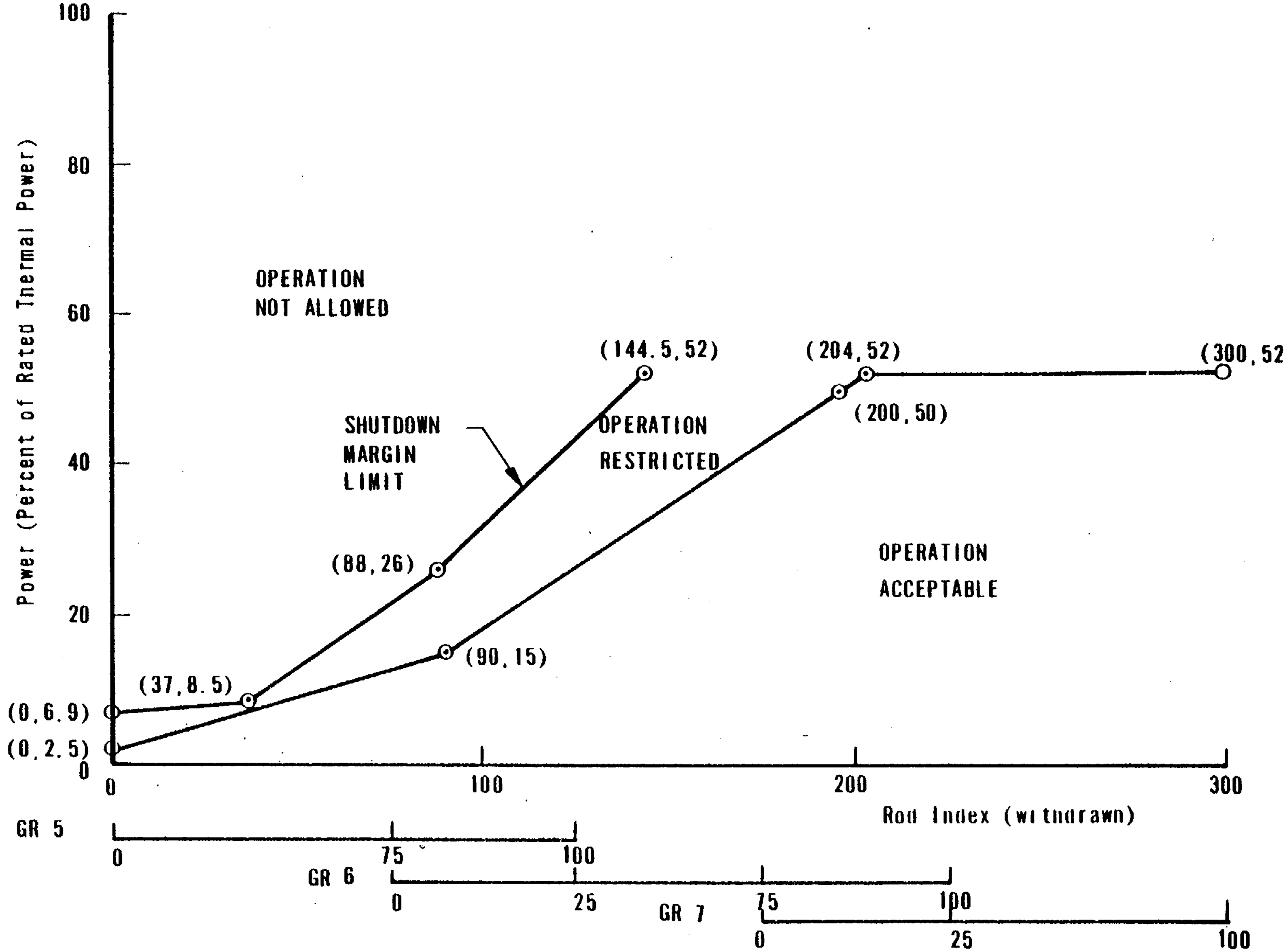
FIGURE 3.5.2 - 2A3



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 1

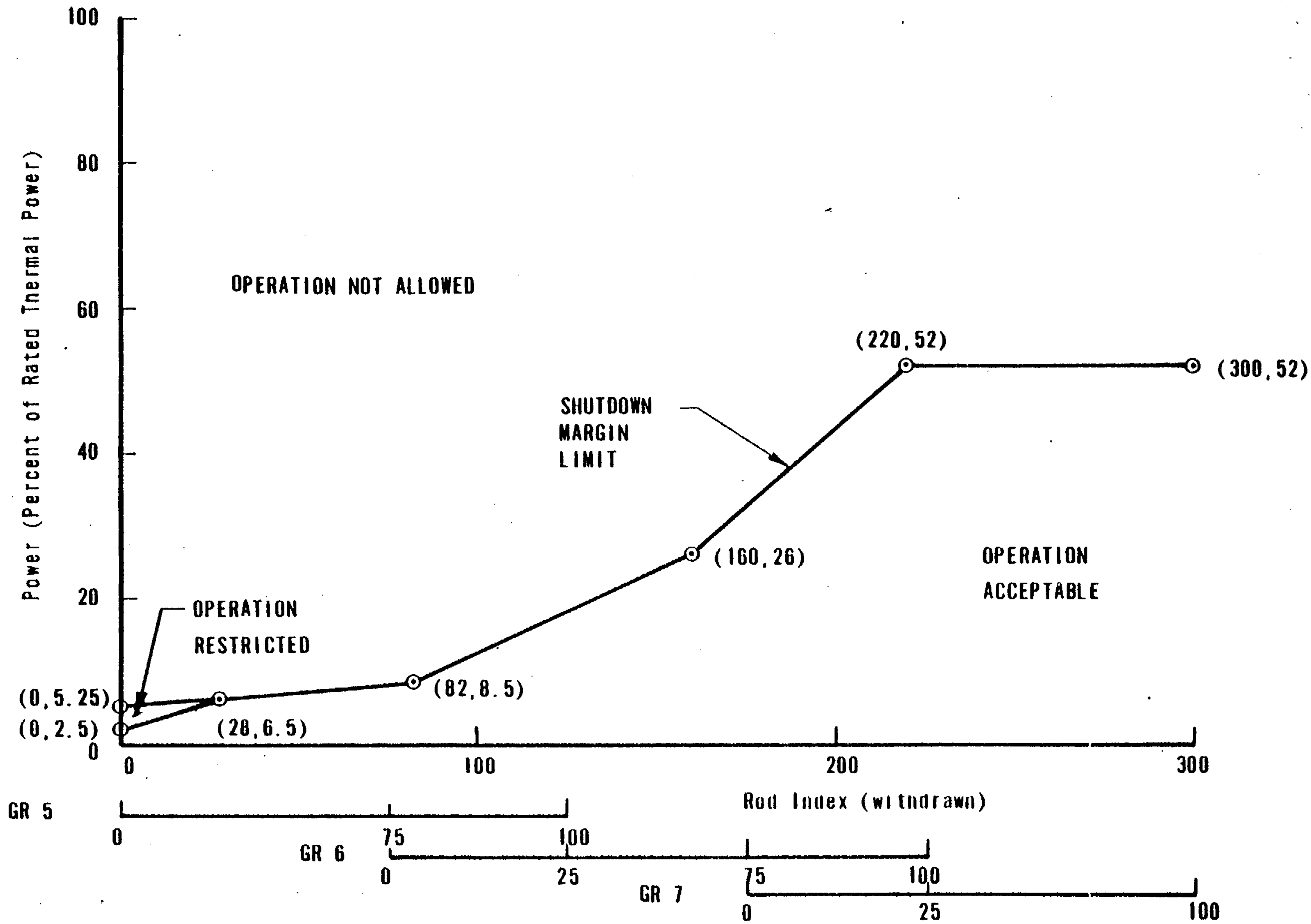
OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 2A4



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 50 (+10, -0) to 200 ± 10 EFPD
UNIT 1

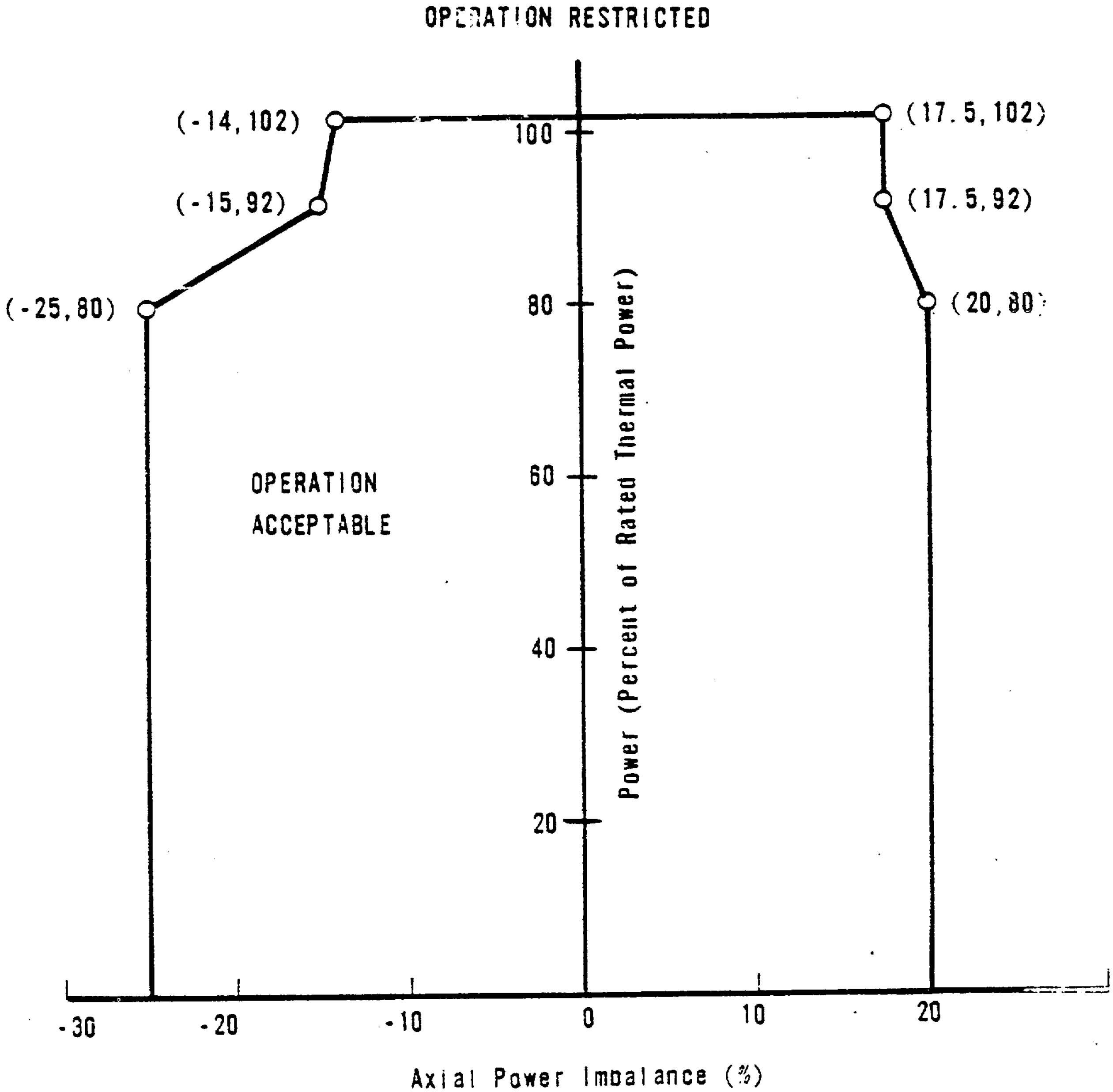
OCONEE NUCLEAR STATION



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

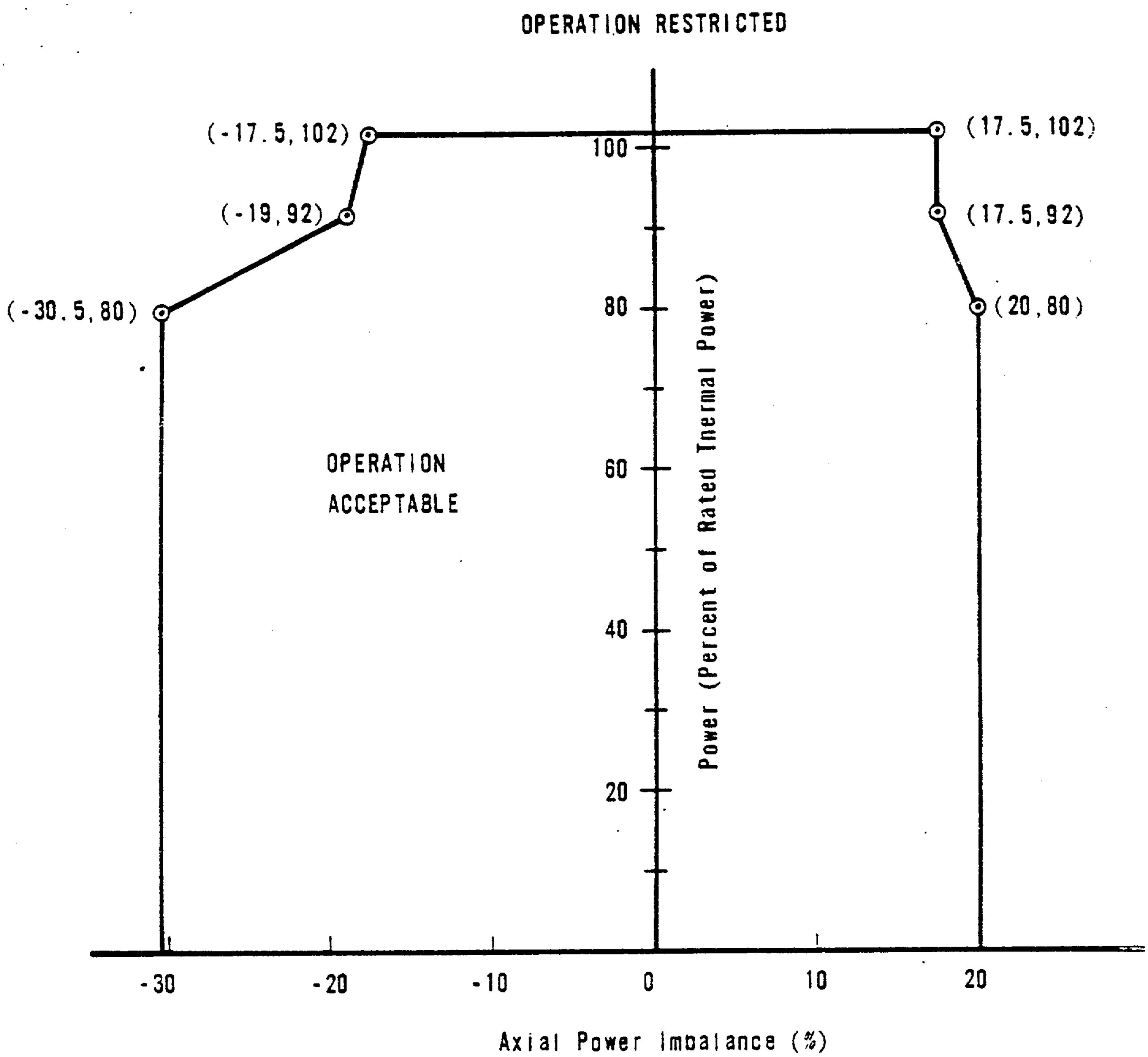
FIGURE 3.5.2-2A6



POWER IMBALANCE LIMITS
FOR OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 1

OCONEE NUCLEAR STATION

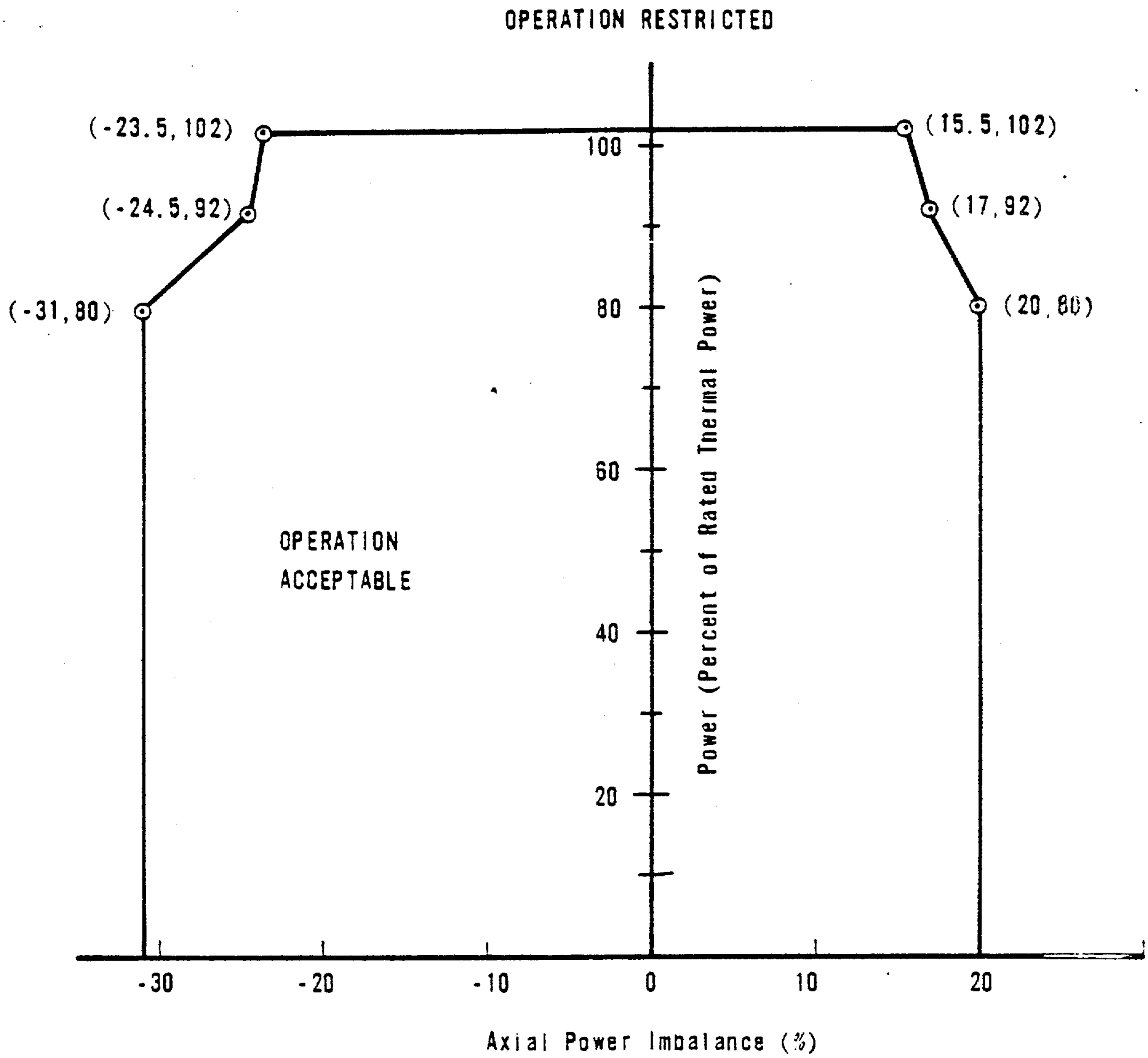
FIGURE 3.5.2 - 3A1



POWER IMBALANCE LIMITS
FOR OPERATION
FROM 50 (+10, -0) to 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

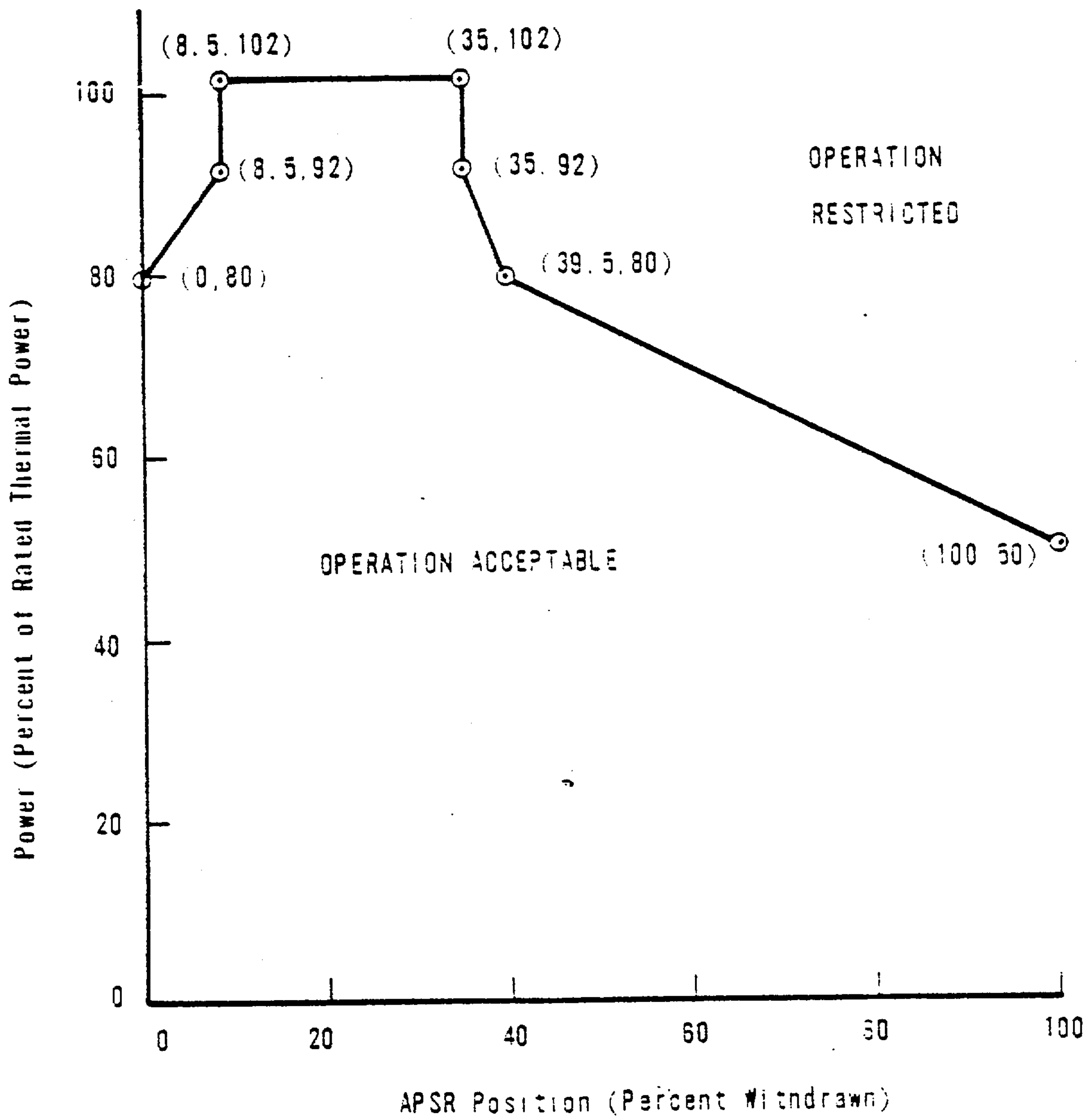
FIGURE 3.5.2 - 3A2



POWER IMBALANCE LIMITS
FOR OPERATION
AFTER 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

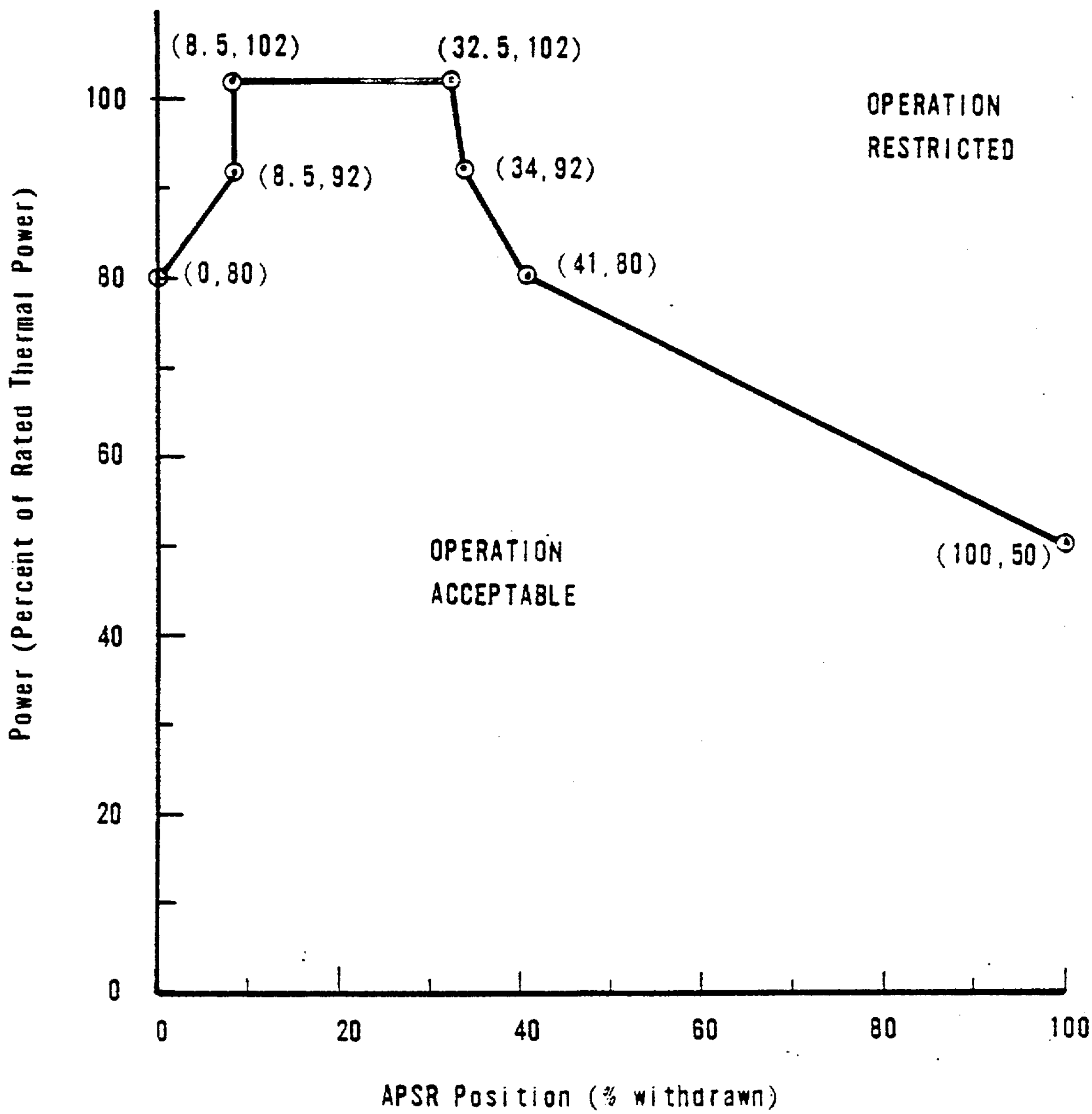
FIGURE 3.5.2 - 3A3



APSR POSITION LIMITS
FOR OPERATION
FROM 0 to 50 (+10, -0) EFPD
UNIT 1

OCONEE NUCLEAR STATION

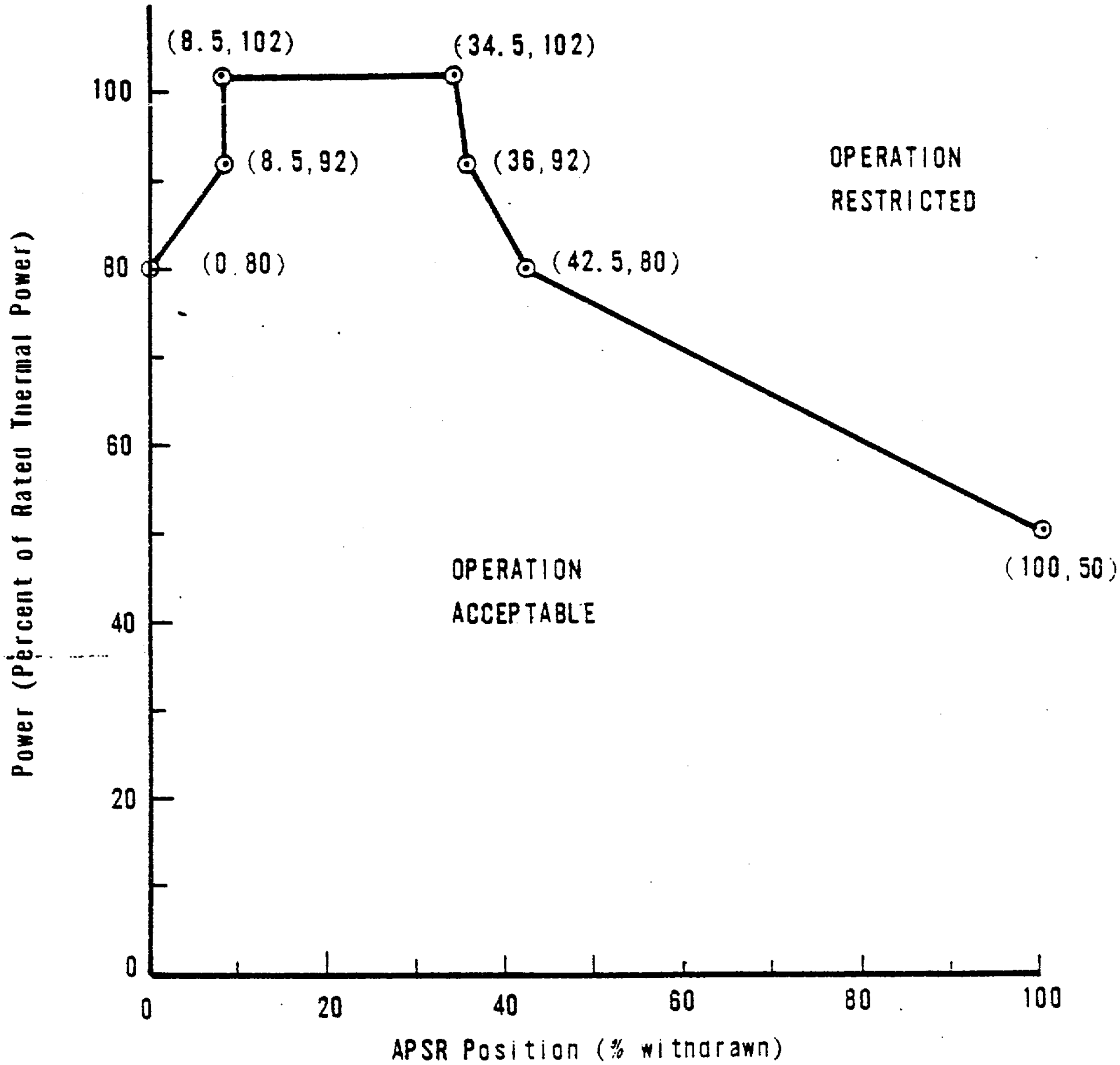
FIGURE 3.5.2 - 4A1



APSR POSITION LIMITS
FOR OPERATION
FROM 50 (+10, -0) to 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 4A2



APSR POSITION LIMITS
FOR OPERATION
AFTER 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

FIGURE 3.5.2 - 4A3

Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1835 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

The off-site doses for the fuel handling accident are within the guidelines of 10CFR100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

(1) FSAR, Section 9.7
(2) FSAR, Section 14.2.2.1
(3) FSAR, Section 14.2.2.1.2

TABLE 4.4-1
(NOTES)

NOTE 1 All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment atmosphere and to assure they will be subjected to the test differential pressure.

NOTE 2 Fluid system that is part of the reactor coolant pressure boundary and open directly to the containment atmosphere under post-accident conditions (vented to containment atmosphere during Type A test).

NOTE 3 Closed system inside containment that penetrates containment and postulated to rupture as a result of a loss of coolant accident (vented to containment atmosphere during Type A test).

NOTE 4 System required to maintain the plant in a safe condition during the test (need not be vented).

NOTE 5 System normally filled with water and operating under post-accident condition (need not be vented).

NOTE 6
a. Containment penetration whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetration filled with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

b. Air lock door seals including door operating mechanisms which are part of the containment pressure boundary.

c. Doors with resilient seals or gaskets except for seal welded doors.

d. Components other than those above which must meet the acceptance criteria of Type B tests.

NOTE 7
a. Isolation valves provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves.

b. Isolation valves are required to close automatically upon receipt of a containment isolation signal in response to controls intended to affect containment isolation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 102 TO FACILITY OPERATING 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

1.0 Introduction

By letter dated May 29, 1981, Duke Power Company (Duke or licensee) submitted an application to change the common Oconee Nuclear Station (ONS) Technical Specifications (TSs) to support the full power operation of Unit 1 during Cycle 7 operation. Included in this application was a change to the high pressure injection system TSs to reflect the present design of the ONS systems. Another application was submitted by Duke on August 25, 1981, requesting that the minimum temperature of the Borated Water Storage Tank be increased from 40 to 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. Additional changes to the TSs were determined to be needed as a result of reevaluation of the Reactor Protective System accuracies. These changes were submitted by a supplemental application dated October 16, 1981.

2.0 Background

The ONS Unit 1 core contains 177 fuel assemblies. The fuel assemblies for Cycle 7 operation include 68 new assemblies designated as Batch 9 and previously loaded assemblies designated as Batch 7B, 8A, 8B and 4E. The Batch 9, 7B, 8A and 8B assemblies are mechanically interchangeable, Babcock and Wilcox (B&W) Mark B4 designs, while the single Batch 4E assembly is an older Mark B3 design. Reactivity control will be accomplished through the use of 69 full length Ag-In-Cd control rods, 60 burnable poison rod assemblies (BPRAs) and soluble boron shim. The BPRAs have the redesigned holddown latching mechanism which was previously approved by the NRC staff.

3.0 Evaluation

3.1 Fuel Assembly Design

Although all fuel assemblies for Cycle 7 operation are Mark B design, the Batch 9 and some Batch 8 assemblies have a slightly higher initial fuel density (94 to 95 percent theoretical density) as a consequence of a modified fuel fabrication process. In addition, four Batch 9 assemblies have Zircaloy rather than Inconel intermediate spacer grids and are designated Mark BZ design, Batch 9B. The Mark BZ design demonstration fuel assemblies were described in B&W's Report, BAW-1661P, submitted by letter dated April 10, 1981. We have reviewed these changes in the fuel assembly design and find them to be relatively minor and not limiting for Cycle 7 operation.

The fuel assemblies were analyzed by the licensee for cladding collapse, stress and strain using methods and limits previously reviewed and approved by the NRC and were found to be bounded by either previously analyzed, or specifically analyzed for Cycle 7 conditions.

Fuel rod internal pressure was evaluated in accordance with approved methods and found to remain below normal system pressure for all assemblies except the Batch 4E assembly. We find this acceptable, however, because (1) the consequences of underestimating fuel rod internal pressure would be limited to the single assembly, and (2) the relative power density of this assembly (due to the higher burnup) is significantly lower than the average for the core and is not limiting for postulated transients and accidents. The differences in the computer codes used to calculate internal pressures may result in values that are too low at the beginning of core life and since these values are used to determine swelling and rupture behavior during a Loss of Coolant Accident (LOCA), reduced KW/ft limits at low core elevations during the first 50 effective full power days have been included in the TSs.

We have reviewed the factors related to fuel assembly design and find that they have been acceptably considered for Cycle 7 operation.

3.2 Core Physics

The licensee described the core loading to be used in Cycle 7. Sixty-eight fresh assemblies having an initial enrichment of 3.28 weight percent U-235 will be loaded. A single high burnup assembly will be located at the core center for its fifth cycle. Cycle 7 is to have an extended length of 427 effective full power days. For this reason burnable poison assemblies are used to limit the required beginning of cycle soluble boron concentration.

The nuclear characteristics of the core have been computed by methods previously used and approved for B&W reactors. Comparisons were made between the physics parameters for Cycles 6 and 7. The differences that exist between the parameters are due to the increased cycle length which tends to increase values of critical boron concentrations, stuck and ejected rod worths and moderator coefficients. All safety criteria are still met. Shutdown margin values at beginning and end of cycle are 3.89 and 2.40 percent $\Delta k/k$, respectively, compared to the required 1.0 percent. Beginning of cycle radial power distributions show acceptable margins to limits. Based on our review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Cycle 7 is acceptable.

The key kinetics parameters for Cycle 7 have been compared to the values used in the Final Safety Analysis Report (FSAR) and densification report. It is shown that in all cases Cycle 7 values are bounded by those previously used. We conclude that the FSAR transient and accident analyses are valid.

We have reviewed the proposed TSs for Cycle 7. The limiting safety systems settings and the limiting conditions for operation have been established by previously used and approved methods. The rod withdrawal limits for the various pump combinations and times in life are presented. On the basis that previously approved methods were used to obtain the limits, we find them acceptable.

The effects of the recently discovered under-estimate of the errors in certain modules of the reactor protection system have been included. By letter of September 10, 1981, the nuclear overpower trip setpoint was reduced from 105.5 to 104.9 percent full power. This is the same change that has been made on other B&W plants and is acceptable. Likewise the high reactor coolant temperature trip has been reduced from 619 to 618 degrees Fahrenheit. By letter dated October 16, 1981, the flux-flow-imbalance safety system setpoints were revised to restrict operation to narrower imbalance limits. On the basis that these setpoints were established by previously accepted methods, we conclude that the revised limits are acceptable.

3.3 Core Thermal-Hydraulics

The thermal-hydraulics design conditions for Cycle 7 operation were compared to the Cycle 6 values in Duke's May 29, 1981, application (Table 6.1) and were shown to be identical. The major differences of thermal-hydraulic concern between Cycle 6 and Cycle 7 are related to the Mark BZ demonstration assemblies which are discussed above and the rod bow Departure from Nucleate Boiling Ratio (DNBR) compensation.

The rod bow DNBR compensation for Cycle 7 operation was calculated using approved interim evaluation procedures which demonstrated that the Batch 9 fuel assemblies are the most limiting. The burnup used to calculate the rod bow penalty was the highest assembly burnup in Batch 9 (17,669 MWd/MTu) which contains the limiting (maximum radial peaking factor) fuel assembly. The resultant net rod bow penalty, after inclusion of the 1% flow area reduction factor credit, is a 0.2% reduction in DNBR.

We have reviewed the details of these calculations provided by letter dated October 16, 1981, and have concluded that the thermal-hydraulics design for Cycle 7 includes a margin of greater than 0.2% above the minimum acceptable DNBR and is therefore acceptable.

3.4 Startup Physics Testing

Included in the May 29, 1981 application, was a revision to the "Oconee Nuclear Station Startup Physics Test Program" which had been approved by the NRC letter dated March 23, 1981. The proposed test would use the incore detector outputs to calculate the quadrant power tilt at 17% full power. Data are presented which show that the quadrant power tilts are consistent with those obtained from the previously used test.

The proposed test is a better measure of core symmetry and it provides information to help determine and correct the cause of a possible core asymmetry. The only disadvantage is that the proposed test cannot be performed until core power is at about 17% of full power. Because the proposed test would provide more usable data and because no core limits would be approached at 17% full power even for a large core asymmetry, we find the proposed test an acceptable substitute for the zero power rod swap asymmetry test for this and subsequent startup testing on all Oconee Units.

3.5 Technical Specifications

Included in the May 29, 1981 application, were proposed revisions to the high pressure injection (HPI) cross connect system, the required volume of the boric acid storage tank (BAST) and the concentration of the boric acid solution in the borated water storage tank (BWST). The changes to the HPI system specification reflect completion of modifications which were approved by NRC letter dated December 13, 1978. The proposed specification had been approved, by Amendments 81, 81 and 78 issued on February 22, 1980, for Unit 3 and is therefore acceptable for the other Units. The changes to BAST volume (from 995 to 1020 cubic feet of 8700 ppm boron solution) and BWST concentration (from 1800 to 1835 ppm boron solution at a minimum volume of 46 ft.) were required to increase the minimum available boric acid solution to provide assurance that the reactor can be borated to an adequate subcritical margin during Cycle 7. Since these changes maintain previously approved design bases, we find them to be acceptable.

By letter dated August 25, 1981, Duke proposed increasing the minimum temperature of the BWST from 40 to 50°F. This change would ensure that a higher temperature fluid would be injected, should the HPI system be operated. An increased injection temperature would lessen the thermal shock to the reactor pressure vessel. We have reviewed this change and find it acceptable since it will provide added protection against thermal shock.

The TS changes related to full power operation of Unit 1 for Cycle 7 were reviewed as discussed in Section 3 of this evaluation and were found acceptable.

One additional TS page was revised to remove a source of confusion. The last sentence of Note 5 of Table 4.4-1, on page 4.4-12, is confusing since it could be interpreted to conflict with the requirements contained in Table 4.4-1. To remove this confusion, this sentence was removed. Since no technical content of the requirements was changed by this action, we consider this to be an administrative action and find it to be acceptable.

4.0 Environmental Consideration

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

5.0 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.

Dated: November 30, 1981

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

These amendments revise the TSs to allow full power operation of Unit 1 for fuel Cycle 7, reflect completed modifications to the high pressure injection system and revise the boron concentration requirements.

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 §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated May 29 and August 25, 1981, as supplemented by letter dated October 16, 1981, (2) Amendments Nos. 105 , 105 , and 102 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 30th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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