May 23, 1979

Docket No.: 50-313

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Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your applications dated November 9, 1978, as supplemented, and February 23, 1979, as supplemented.

This amendment revises the Technical Specifications to reflect plant operating limits for the fuel loading to be used during Cycle 4. The amendment also adds additional smoke/heat detectors and fire suppression systems in appropriate tables of the Technical Specifications. To support the modification in the area of fire protection, Supplement No. 1 to the Fire Protection Safety Evaluation Report for this facility has been prepared.

In accordance with our discussions, you have agreed to provide the deviations between the actual power distributions and the predicted power distributions in the monthly operating report.

7906290292 P

On May 17, 1979, the Commission issued an Order confirming your commitment to keep the facility shutdown until certain modifications and changes to procedures are implemented to decrease the likelihood of occurrence of an event similar to that which recently occurred at Three Mile Island, Unit No. 2 (TMI-2). The Order requires that ANO-1 be maintained in a shutdown condition until satisfactory completion of the items in the Order have been confirmed by the Director, Office of Nuclear Reactor Regulation. Therefore, upon issuance of this amendment related to the fuel loading Cycle 4 and to fire protection, operation of the facility can be commenced only after confirmation of completion of the items in the Order.

CP 1

GD

On March 28, 1979, TMI-2 experienced core damage which resulted from a series of events which were initiated by a Loss of Feedwater Event and apparently compounded by operational errors. We believe that several aspects of this accident have generic applicability to all light water power reactor facilities such as ANO-1. To identify corrective actions to be taken by all licensees, I&E Bulletins have been issued since the TMI-2 accident. The particular bulletins that apply to the B&W facilities are Bulletins Nos. 79-05A and 79-05B.

AP&L provided their response to Bulletin No. 79-05A by letters dated April 11 and 14, 1979, and to Bulletin No. 79-05B by letter dated May 4, 1979. Our evaluation of your response indicates that you have correctly interpreted I&E Bulletins Nos. 79-05A and 79-05B, the actions you have taken demonstrate understanding of the salient concerns arising from the TMI-2 incident in reviewing their implications on ANO-1 operations, and provide added assurance for the protection of the public health and safety during plant operations. A separate Safety Evaluation will be issued documenting our review of the AP&L response to I&E Bulletins Nos. 79-05A and 79-05B and identifying certain areas where additional information or action is needed.

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

Sincerely,

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

Enclosures:
1. Amendment No. 43
2. Safety Evaluation
3. Supplement No. 1 to the Fire Protection Safety Evaluation for ANO-1
4. Notice

cc w/enclosures: See next page

C-RSB:DOR
PCheck*
5/9/79

OELD
CWoodhead*
5/11/79

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB#4:DOR	ORB#4:DOR	ORB#2:DOR	C-PSB:DOR	AD-E&P:DOR	C-ORB#4DOR
SURNAME	RIngram*	GVissing:rf	TWambach*	GLainas*	BGrimes	RReid
DATE	5/7/79	5/21/79	5/8/79	5/8/79	5/18/79	5/22/79

Docket No.: 50-313

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In accordance with our discussions, you have agreed to provide the deviations between the actual power distributions and the predicted power distributions in the monthly operating report.

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

Sincerely,

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

Enclosures and cc: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	ORB#2:DOR	C-PSB:DOR	AD-E&PDOR	C-ORB#4:DOR	C-PSB:DOR	OELD
SURNAME	RIngram	GVissing:rf	TWambach	GLainas	BGrimes	RReid	PCheck	
DATE	5/7/79	5/7/79	5/8/79	5/8/79	5/ /79	5/ /79	5/9/79	5/11/79

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Acting County Judge of Pope County
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cc w/enclosure(s) and incoming
dtd.: 11/9/78, 2/27, 4/28,
2/23, 3/19, 3/20, & 3/30/79
Director, Bureau of Environmental
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Little Rock, Arkansas 72201

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. DPR-51

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

   A. The applications for amendment by Arkansas Power and Light Company (the licensee) dated November 9, 1978, as supplemented February 27 and April 26, 1979, and February 23, 1979, as supplemented March 19, 20 and 30, 1979, 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 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   2.c.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   The license is further amended by revising paragraph 2.c.(3) to read as follows:

   2.c.(3) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1 through 3.19 of the NRC's Fire Protection Safety Evaluation (SE) on the facility dated August 22, 1978 and supplements thereto. These modifications shall be completed as specified in Table 3.1 of the Safety Evaluation Report or supplements thereto. In addition, the licensee may proceed with and is required to complete the modifications identified in Supplement 1 to the Fire Protection Safety Evaluation Report, and any future supplements. These modifications shall be completed by the dates identified in the supplement.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: May 23, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

Remove	Insert
9	9
9b	9b
11	11
12	12
14b	14b
15	15
35	35
35a	35a
47	47
48	48
48a	48a
48b	48b
48bb	48bb
48bbb	48bbb
48c	48c
48cc	48cc
48ccc	48ccc
48d	48d
48dd	48dd
48ddd	48ddd
48f	48f
48g	48g
48h	48h
53d	53d
-	53e
66n	66n

Using a local quality limit of 22 percent at the point of minimum DNBR as a basis for curve 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

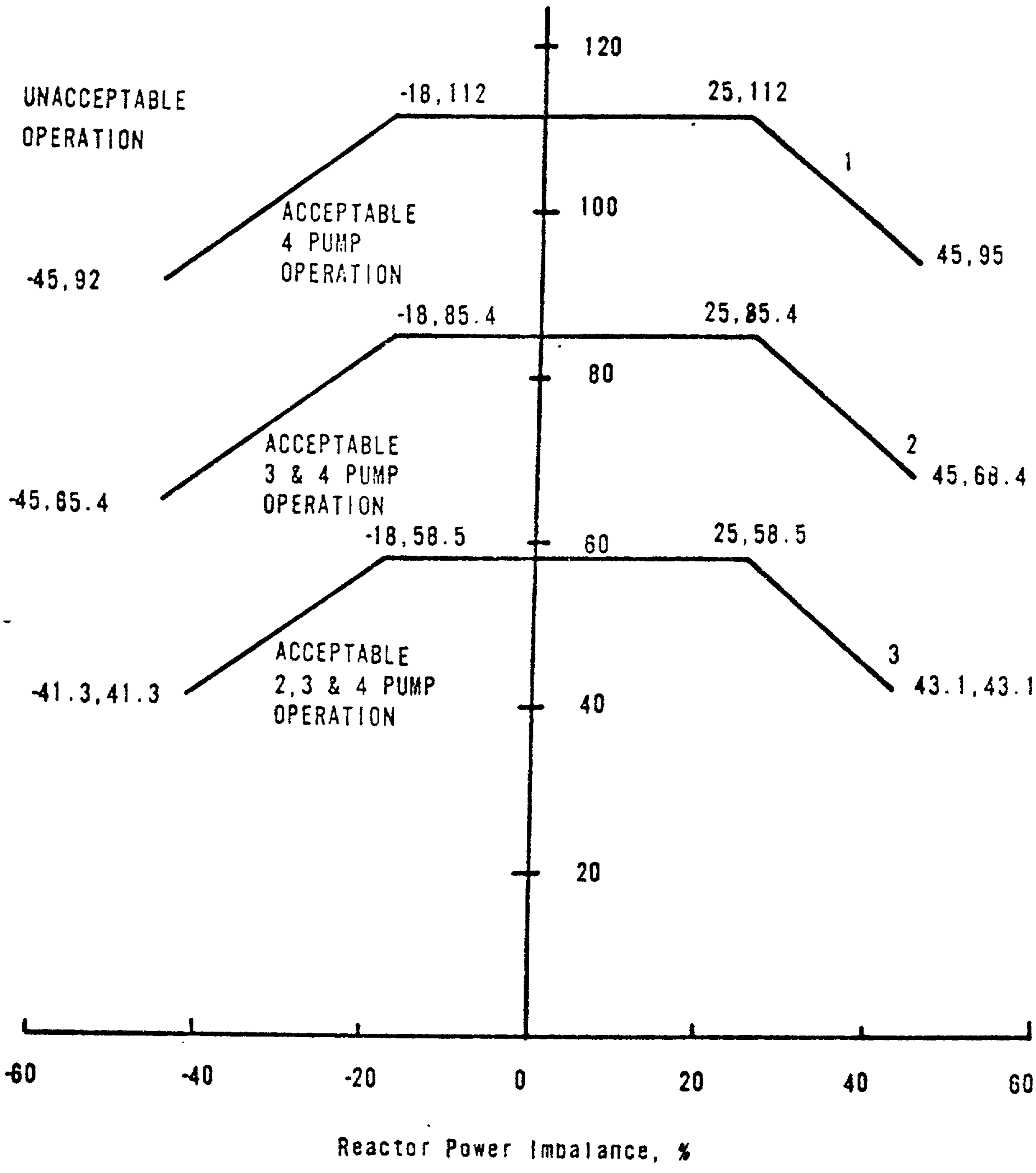
The maximum thermal power for three pump operation is 85.4 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.057 = 78.9 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curves 1 & 2 of Figure 2.1-3 are the most restrictive because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curve.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) FSAR, Section 3.2.3.1.1.c



CURVE	REACTOR COOLANT FLOW (GPM)
1	374,880
2	280,035
3	184,441

CORE PROTECTION SAFETY LIMITS

Figure 2.1-2

Amendment No. ~~5~~, ~~21~~, ~~31~~, 43

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

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

Objective

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

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

Bases

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

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

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

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5 percent of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower trip based on flow and imbalance

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is 105.7 percent and reactor flow rate is 100 percent or flow rate is 94.6 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 78.9 percent and reactor flow rate is 74.7 percent or flow rate is 70.9 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.0 percent and reactor flow rate is 49.2 percent or flow rate is 46.3 percent and the power level is 49.0 percent.

The flux/flow ratios account for the maximum calibration and instrumentation 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.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

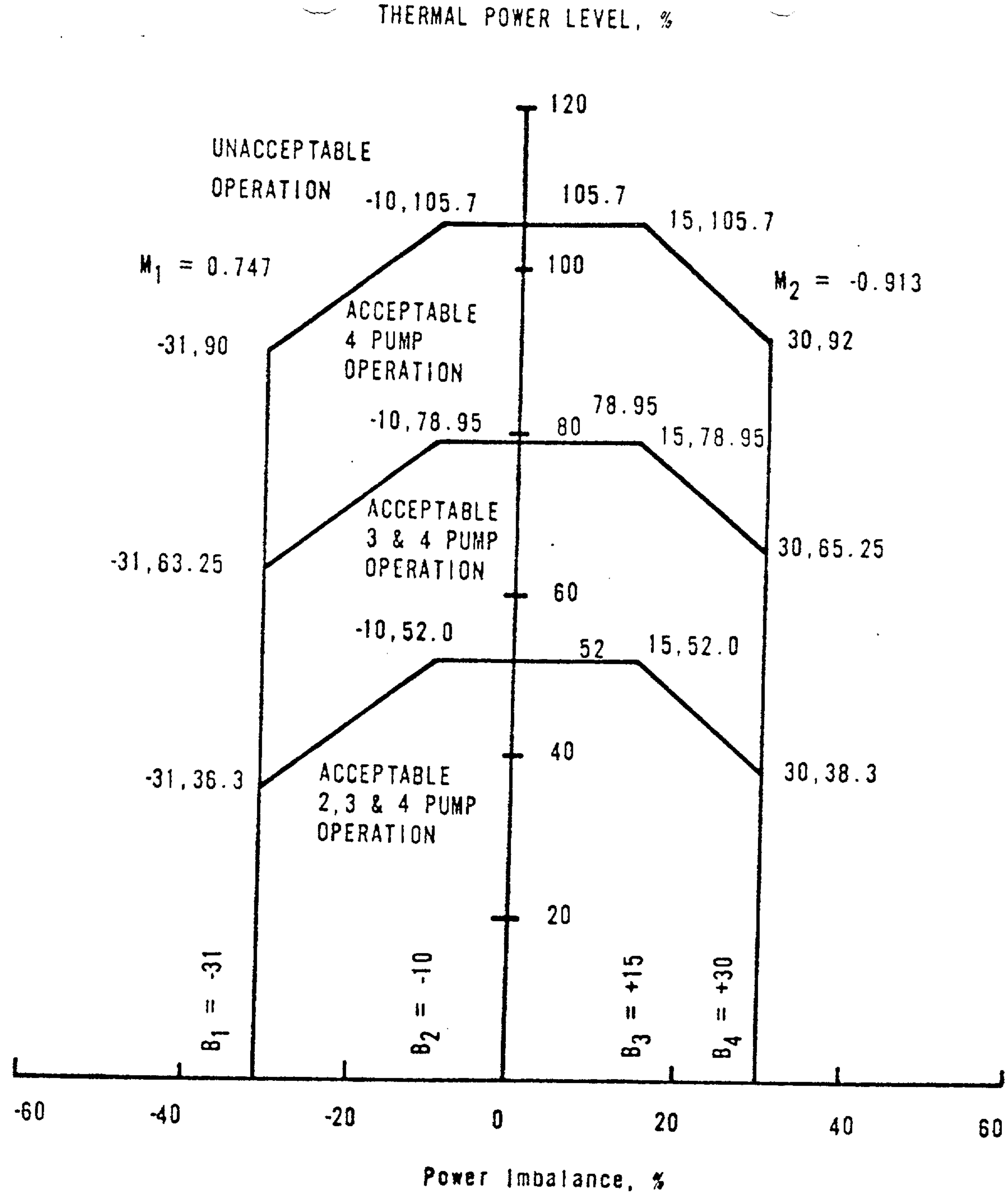
The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by 105.7 percent for a 1 percent flow reduction.

B. Pump monitors

In conjunction with the power imbalance/flow trip, 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 pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS

Figure 2.3-2

Amendment No. ~~8~~, ~~21~~, ~~31~~, 43

Table 2.3-1
Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
Nuclear power, % of rated, max	105.5	105.5	105.5	5.0(3)
Nuclear power based on flow(2) and imbalance, % of rated, max	1.057 times flow minus reduction due to imbalance(s)	1.057 times flow minus reduction due to imbalance(s)	1.057 times flow minus reduction due to imbalance(s)	Bypassed
Nuclear power based on pump monitors, % of rated, max (4)	NA	NA	55%	Bypassed
High reactor coolant system pressure, psig, max	2355	2355	2355	1720(3)
Low reactor coolant system pressure, psig, min	1800	1800	1800	Bypassed
Variable low reactor coolant system pressure, psig, min	$(11.75T_{out}-5103)$ (1)	$(11.75T_{out}-5103)$ (1)	$(11.75T_{out}-5103)$ (1)	Bypassed
Reactor coolant temp, F, max	619	619	619	619
High reactor building pressure, psig, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

(1) T_{out} is in degrees Fahrenheit (F).
(2) Reactor coolant system flow, %.
(3) Automatically set when other segments of the RPS (as specified) are bypassed
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.

Minimum volumes (including a 10% safety factor) as specified by Figure 3.2 1 for the boric acid addition tank or 35,659 gallons of 2270 ppm boron as boric acid solution in the borated water storage tank (3) will each satisfy this requirement. The specification assures that adequate supplies are available whenever the reactor is heated above 200°F so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The principal method of adding boron to the primary system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using the 25 gpm boric acid pumps.

The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps.

Concentration of boron in the boric acid addition 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 this tank and its associated piping will be kept 10°F above the crystallization temperature for the concentration present. Once in the makeup system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

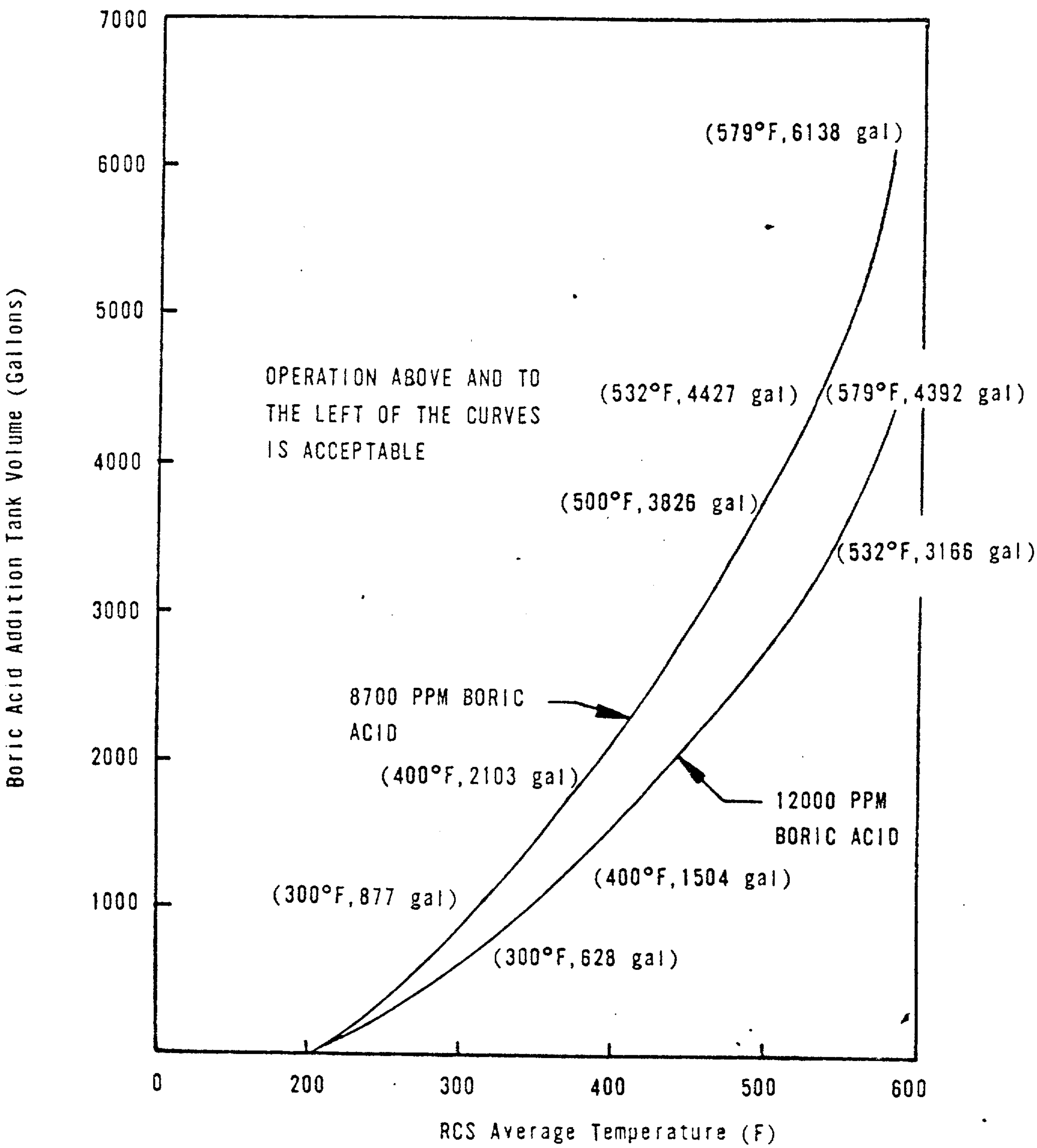
REFERENCES

(1) FSAR, Section 9.1; 9.2

(2) FSAR, Figure 6-2

(3) FSAR, Section 3.3

Figure 3.2-1 BORIC ACID ADDITION TANK VOLUME AND CONCENTRATION REQUIREMENTS VS RCS AVERAGE TEMPERATURE



Amendment No. ~~33~~, 43

6. If a control rod in the regulating or axial power-shaping groups is declared inoperable per Specification 4.7.1.2. operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant tilt:

1. Except for physics tests, if quadrant tilt exceeds 4.92% power shall be reduced immediately to below the power level cutoff (see Figures 3.5.2-1A and 3.5.2-1B). Moreover, the power level cutoff value shall be reduced 2% for each 1% tilt in excess of 4.92% tilt. For less than 4 pump operation, thermal power shall be reduced 2% of the thermal power allowable for the reactor coolant pump combination for each 1% tilt in excess of 4.92%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4.92% except for physics tests, or the following adjustments in setpoints and limits shall be made:

   a. The protection system maximum allowable setpoints (Figure 2.3-2) shall be reduced 2% in power for each 1% tilt.

   b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of 4.92%.

   c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 4.92%.

3. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.

4. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

1. Technical Specification 3.1.3.5 (safety rod withdrawal) 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.

2. Operating rod group overlap shall be 20% ± 5 between two sequential groups, except for physics tests.

3. Except for physics tests or exercising control rods, a) the control rod withdrawal limits are specified on Figures 3.5.2-1A, 3.5.2-1B and 3.5.2-1C for four pump operation and on Figures 3.5.2-2A, 3.5.2-2B and 3.5.2-2C for three or two pump operation and b) the axial power shaping control rod withdrawal limits are specified on Figures 3.5.2-4A, 3.5.2-4B and 3.5.2-4C. If any of these control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within four hours.

4. Except for physics tests, power shall not be increased above the power level cutoff (see Figures 3.5.2-1) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelopes defined by Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C. If the imbalance is not within the envelopes defined by Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

Bases

The power-imbalance envelopes defined in Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C are based on 1) LOCA analyses which have defined the maximum linear heat rate (See Fig. 3.5.2-4) such that the maximum clad temperature will not exceed the final Acceptance Criteria and 2) the Protective System Maximum Allowable Setpoints (Figure 2.3-2). Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the final acceptance criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

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

The 20 ±5 percent 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:

*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

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

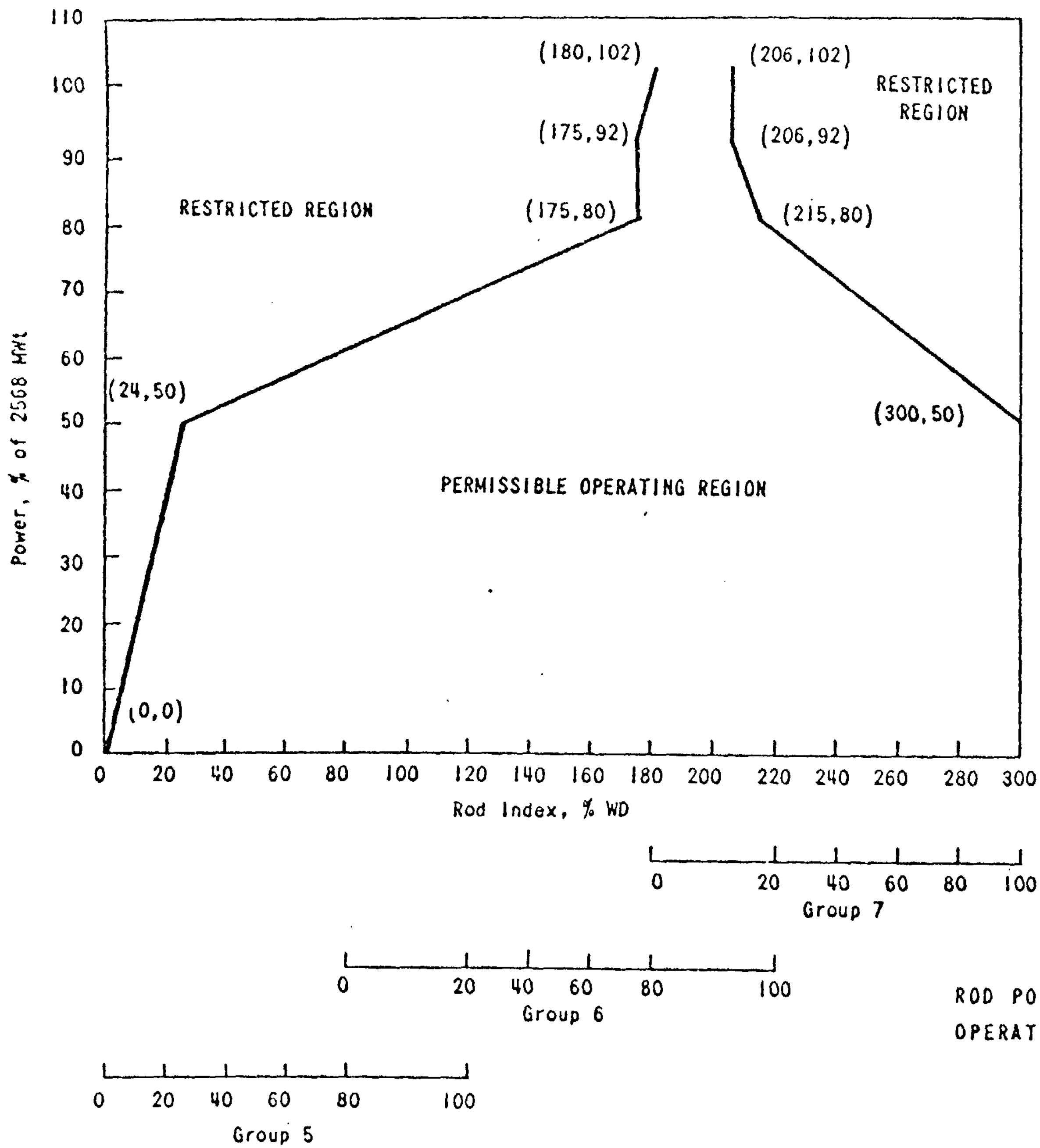
The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent 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) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod positions limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6, and 7 are overlapped 20%. 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 within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5.3 ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

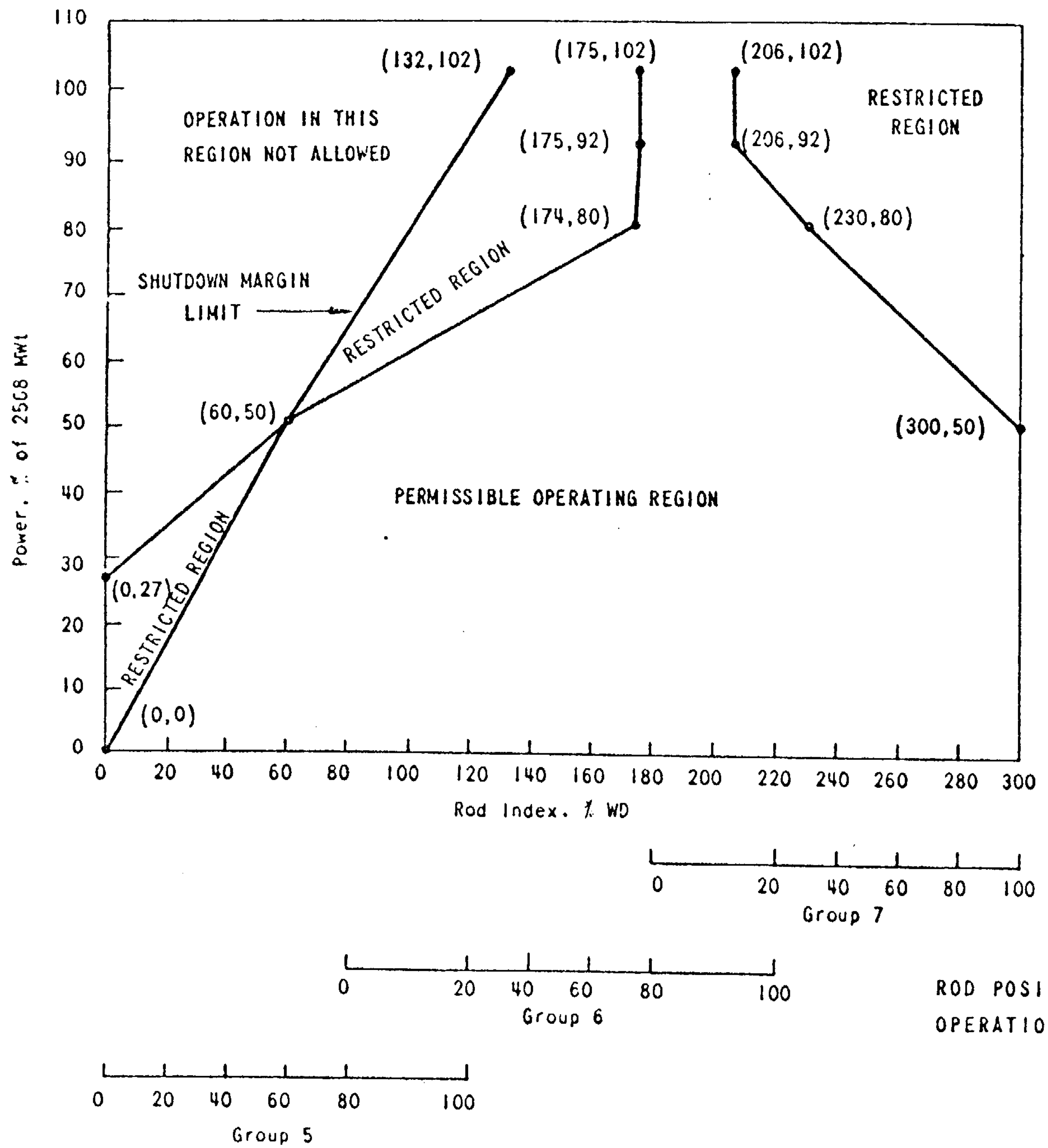
The quadrant tilt and axial imbalance monitoring in Specifications 3.5.2.4.4 and 3.5.2.6, respectively, will normally be performed in the plant computer. The two hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided:



ROD POSITION LIMITS FOR FOUR-PUMP OPERATION FROM 0 TO 100 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-1A



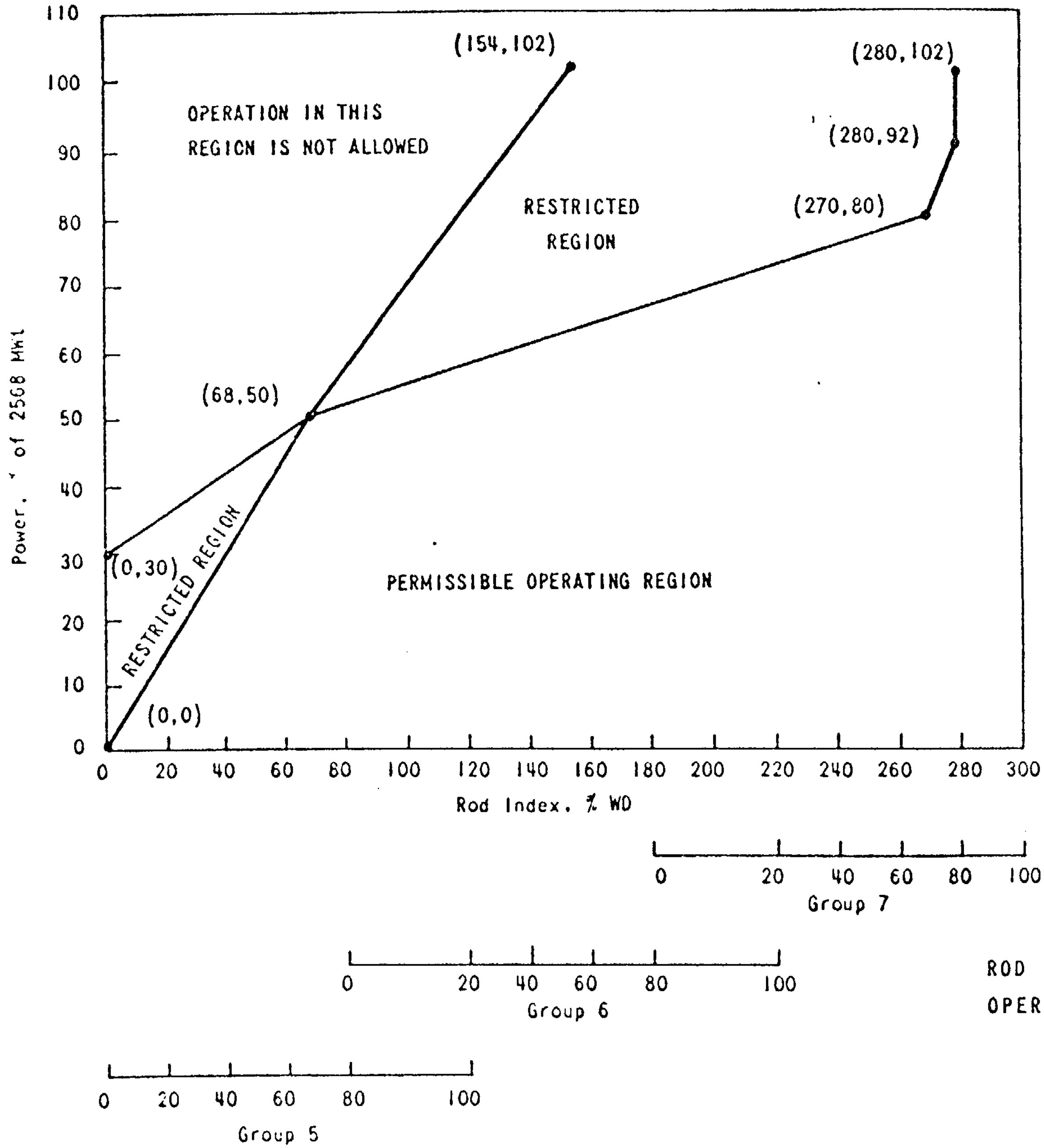
ROD POSITION LIMITS FOR FOUR PUMP OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-1B

Amendment No. 21, 31, 43

Amendment No. ~~21~~, ~~31~~, 43

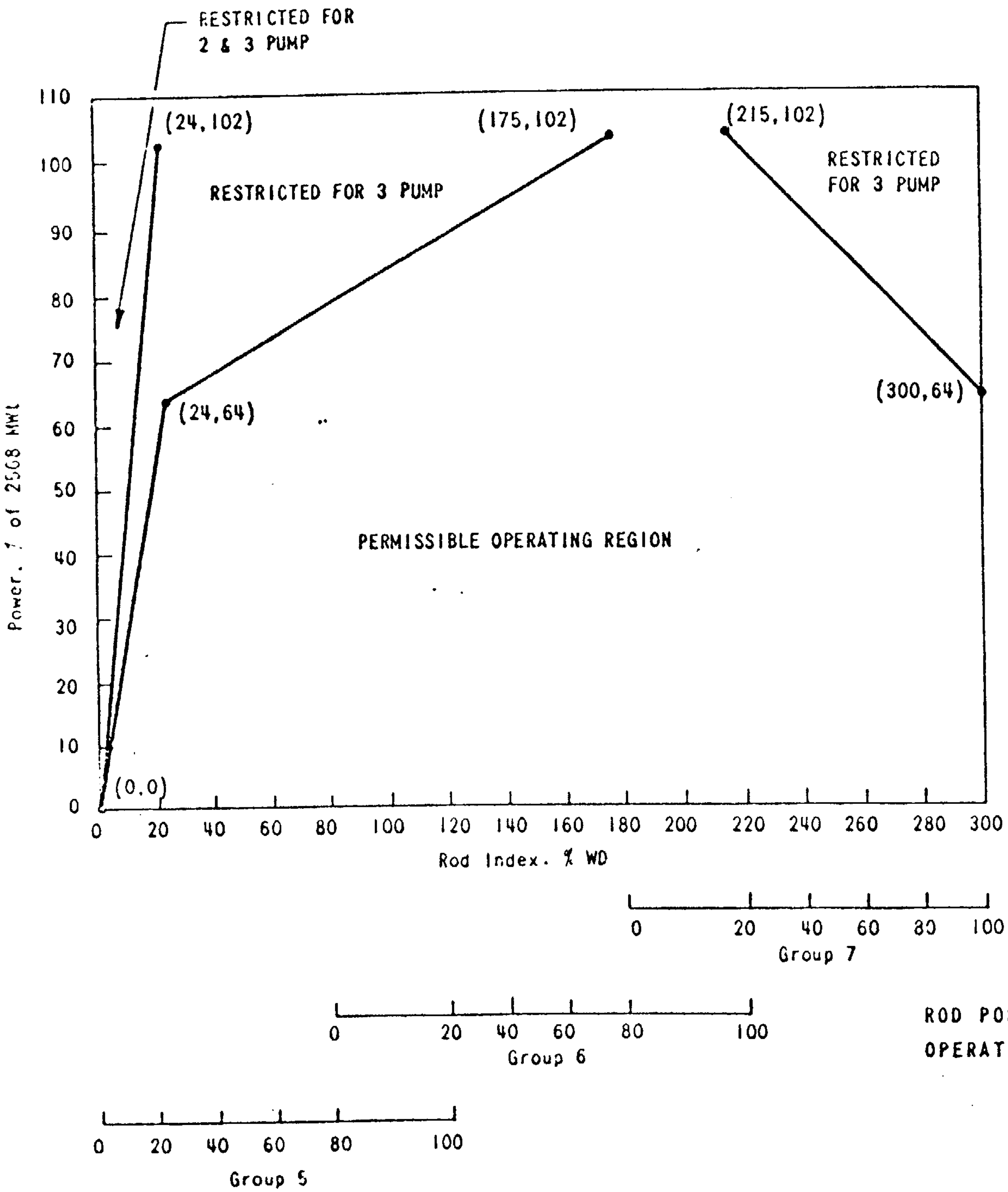
48bbb



ROD POSITION LIMITS FOR FOUR PUMP OPERATION AFTER 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-1C

Amendment No. ~~21~~, ~~31~~, 43



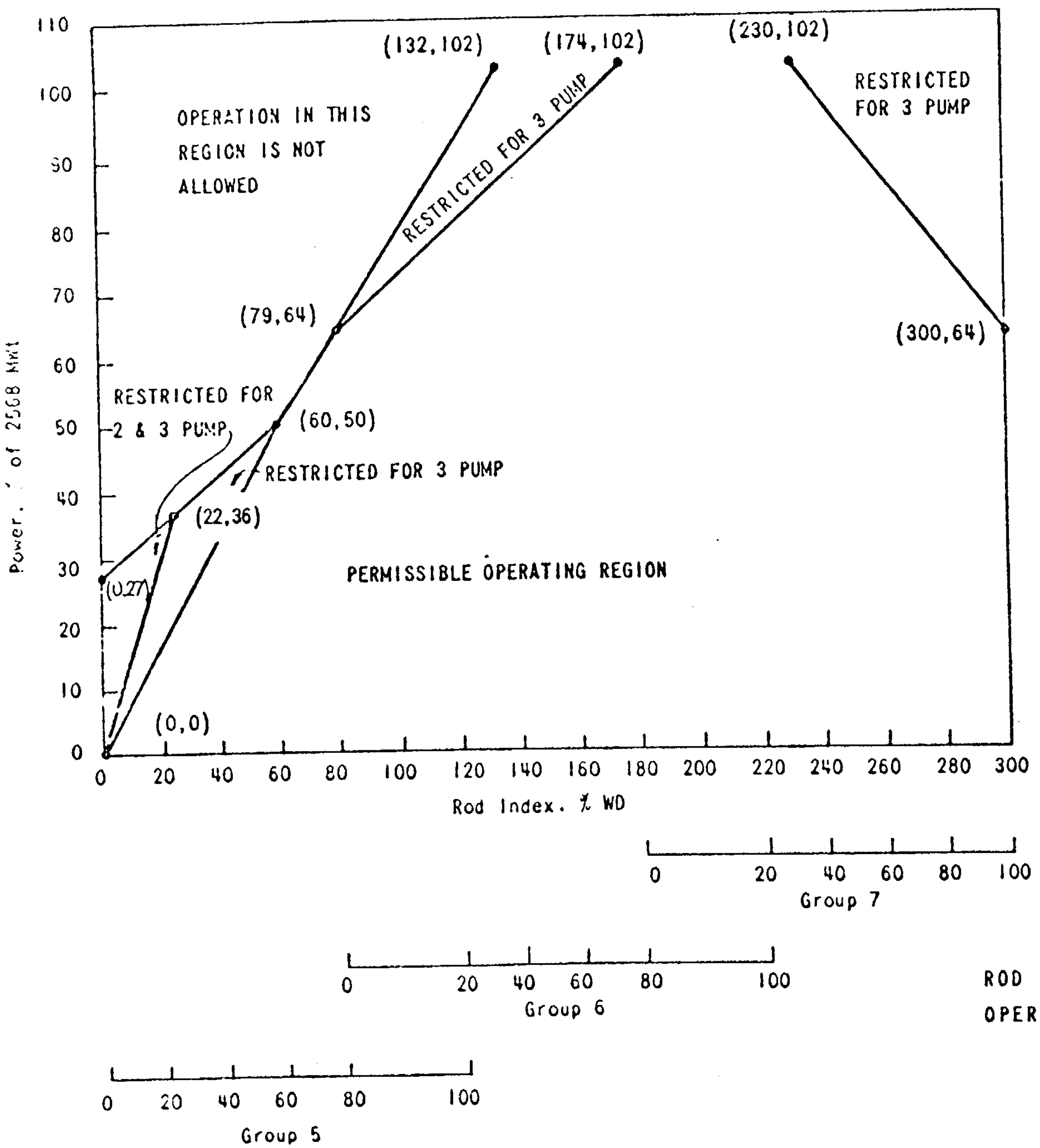
ROD POSITION LIMITS FOR TWO AND THREE PUMP OPERATION FROM 0 TO 100 ± 10 EFPD

ANO-1, CYCLE 4

Figure 3.5.2-2A

Amendment No. ~~21~~, ~~31~~, 43

48cc

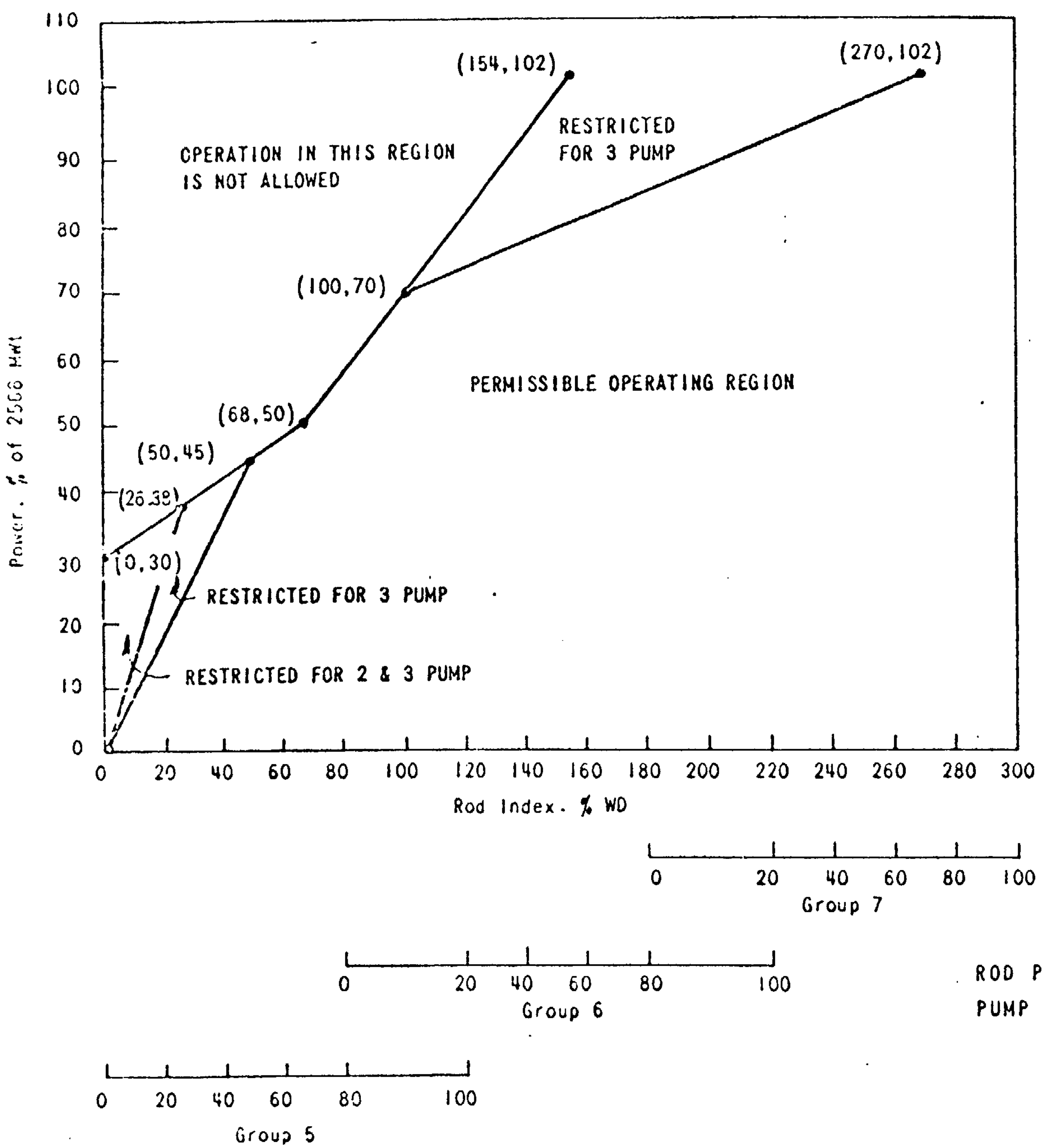


ROD POSITION LIMITS FOR TWO AND THREE PUMP
OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-2B

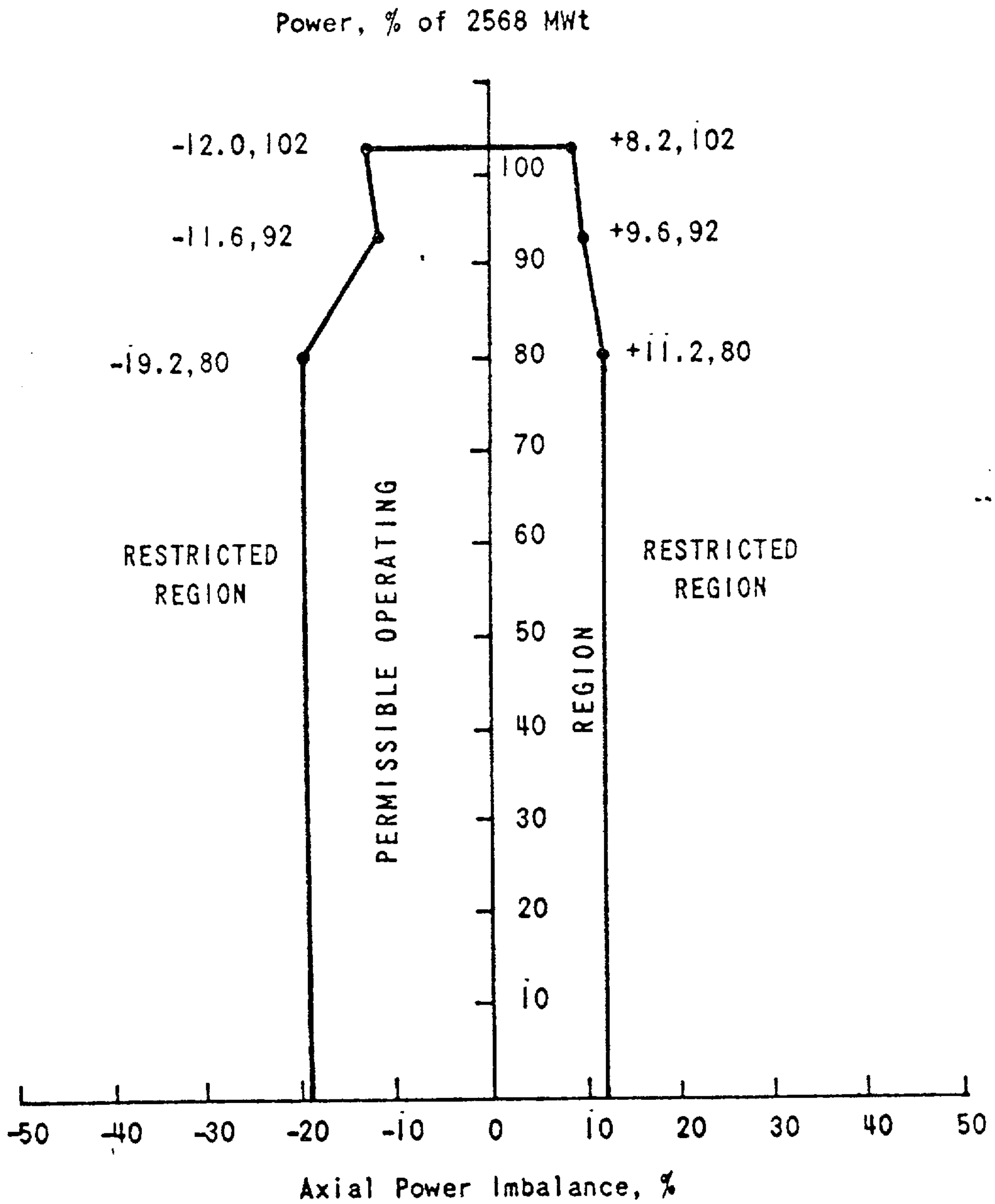
Amendment No. ~~21~~, ~~31~~, 43

48ccc



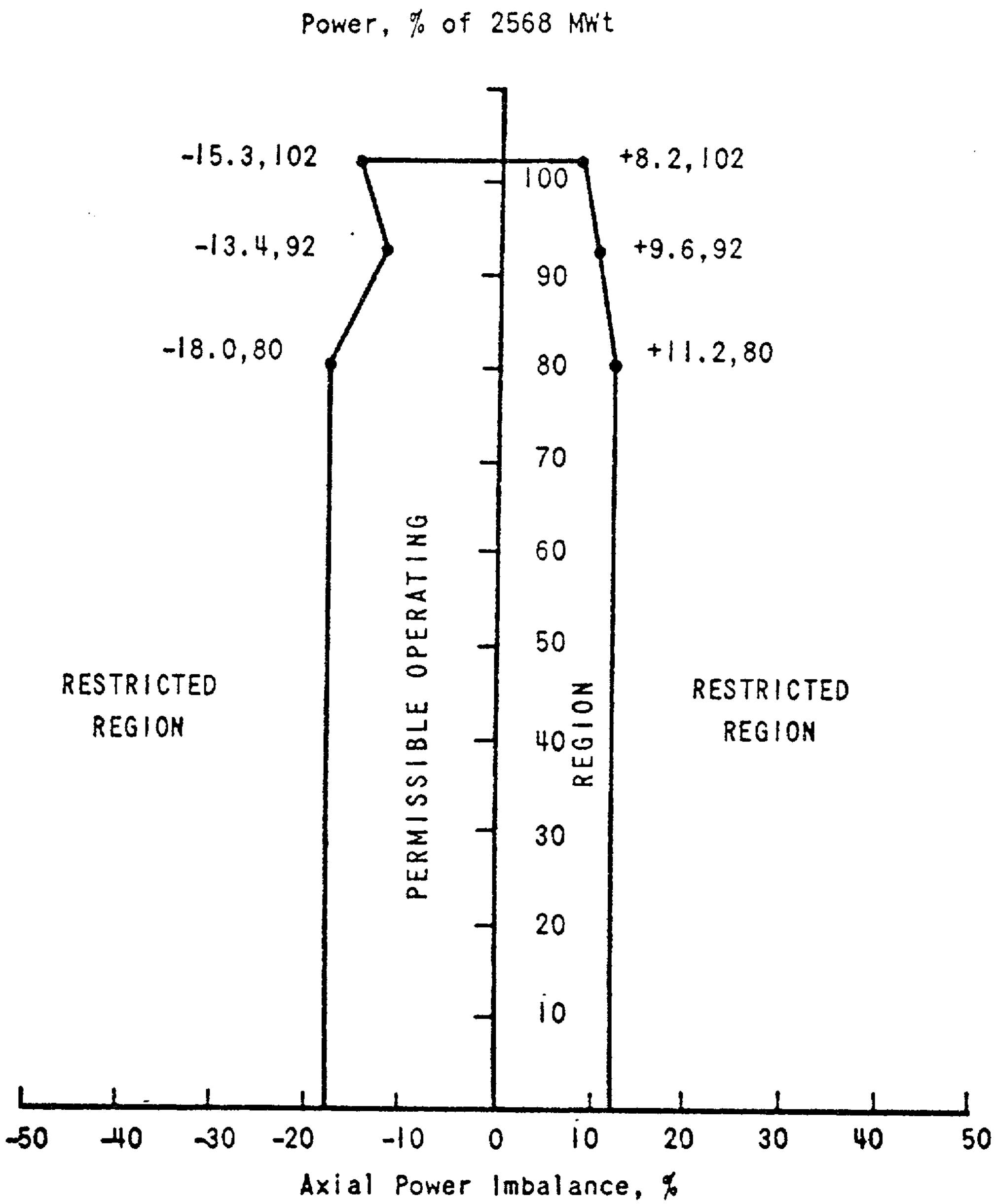
ROD POSITION LIMITS FOR TWO AND THREE PUMP OPERATION AFTER 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-2C



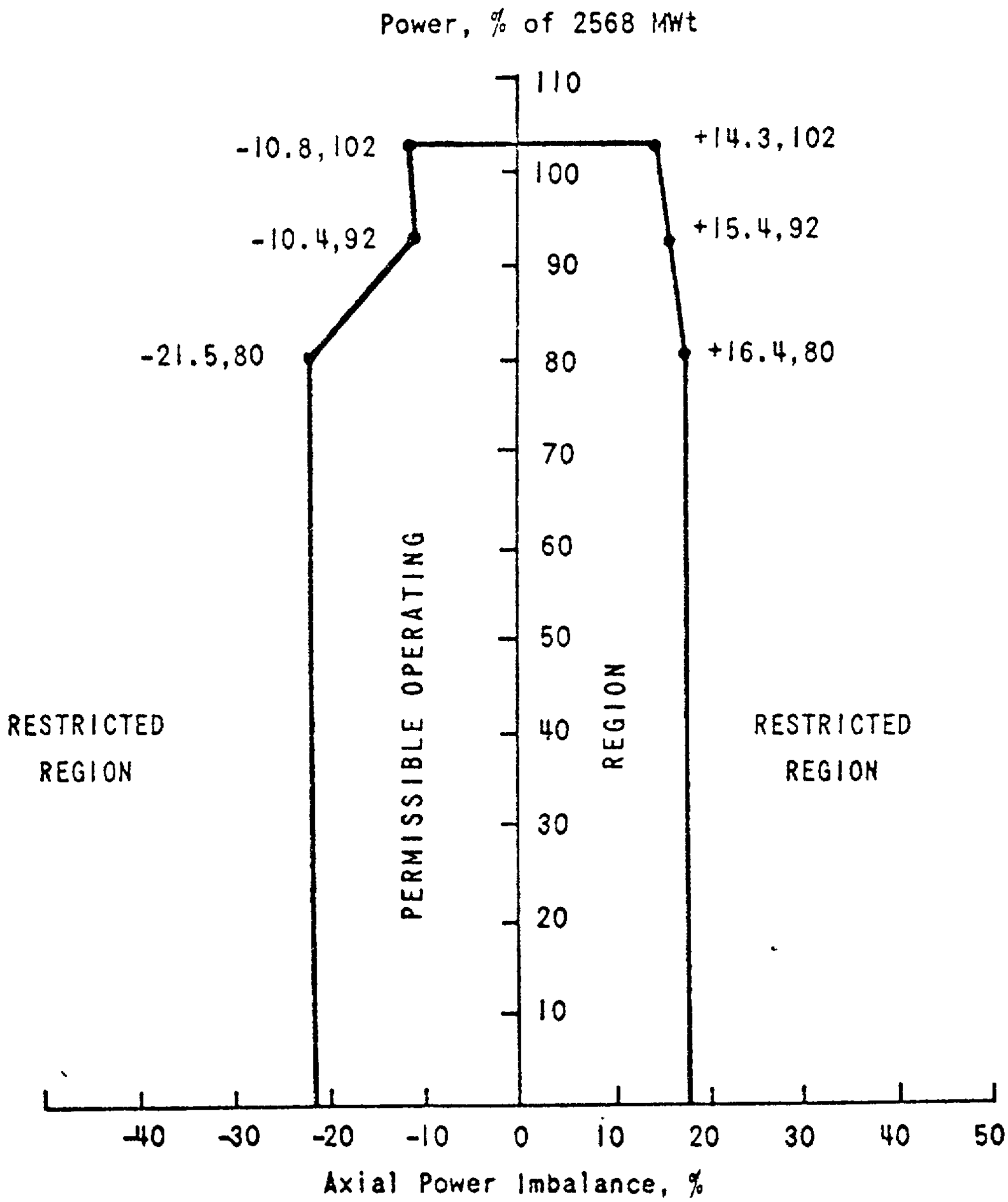
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 100 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-3A



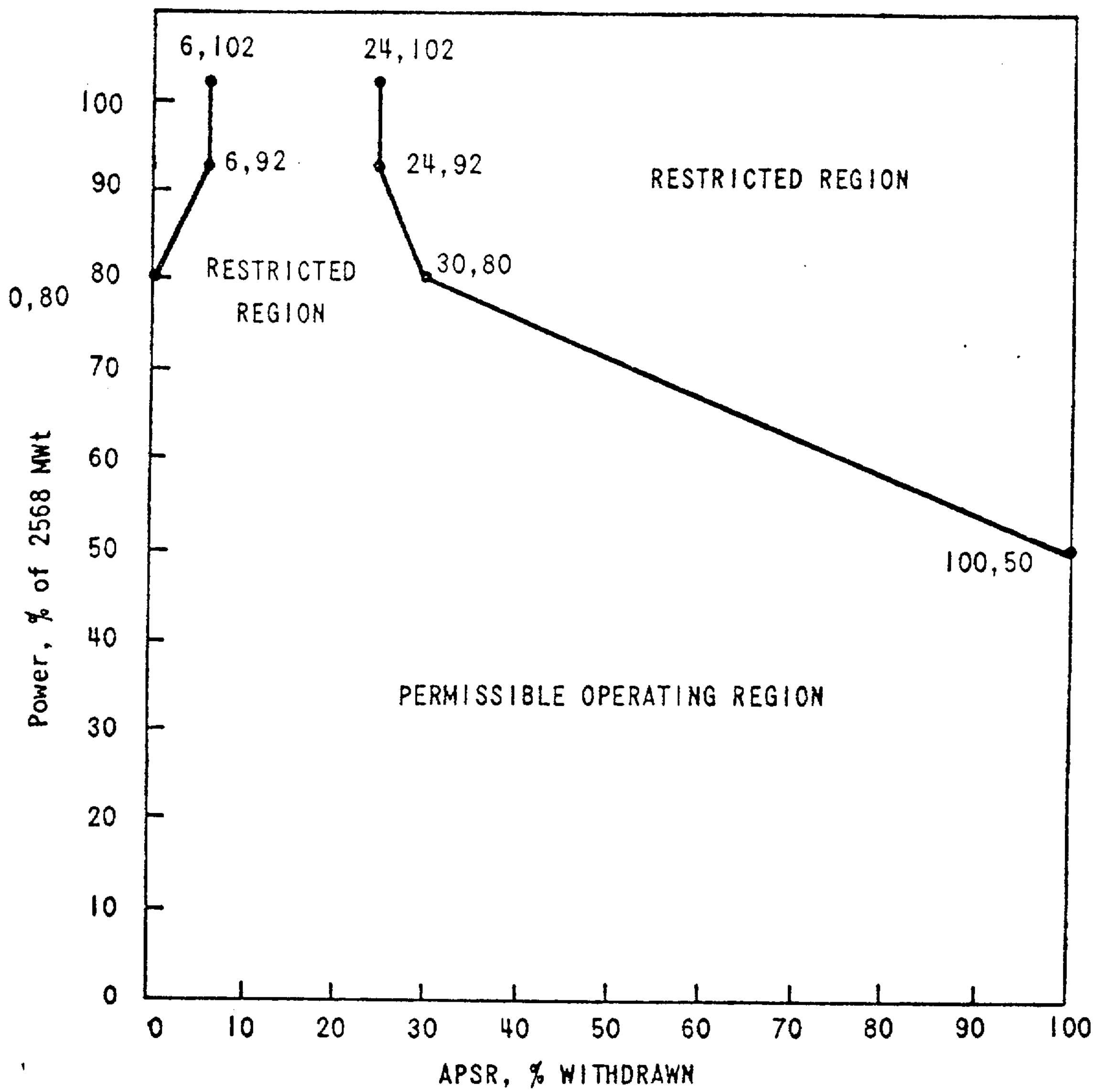
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD ANO-1, CYCLE 1

Figure 3.5.2-3B



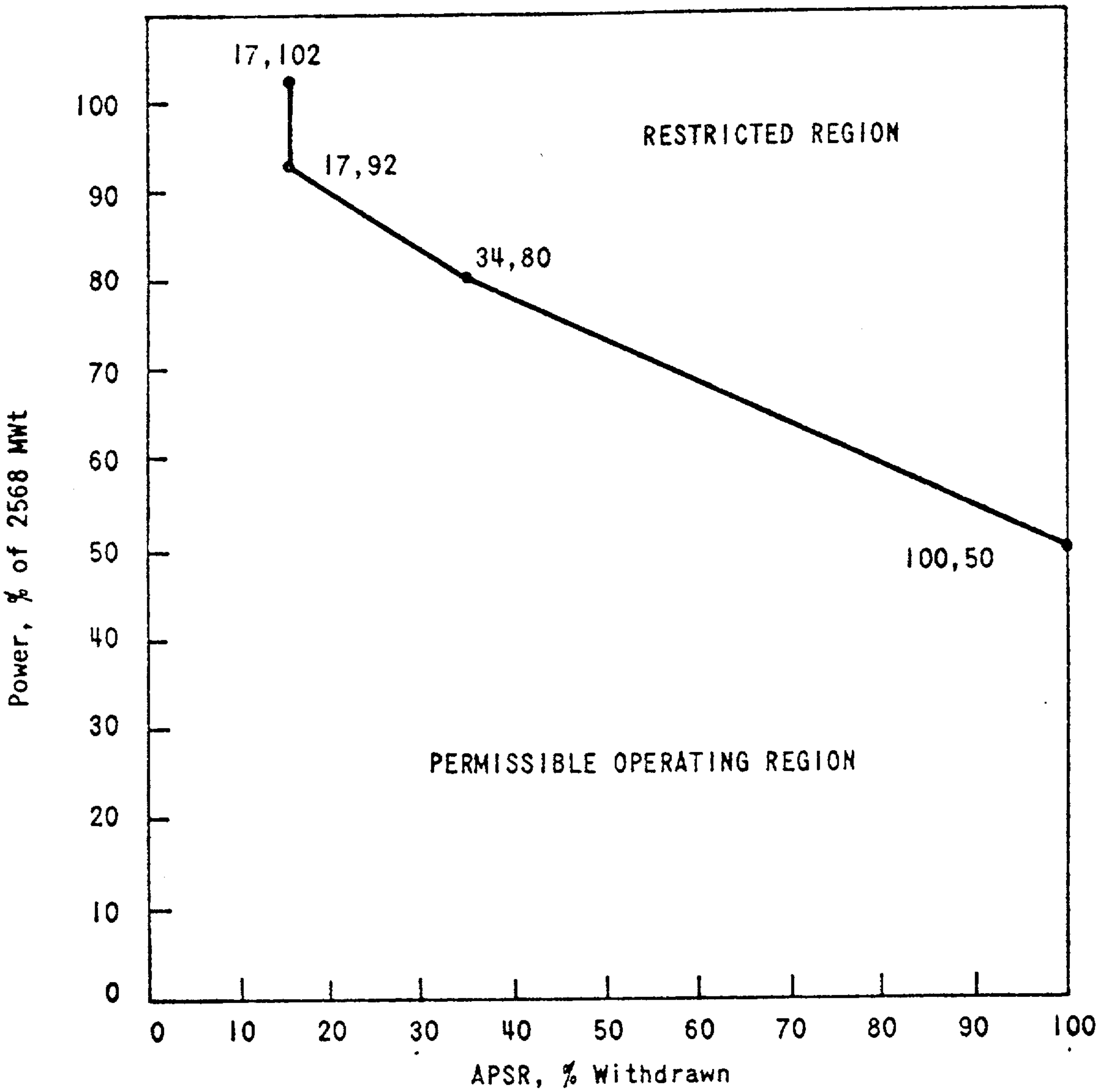
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-3C



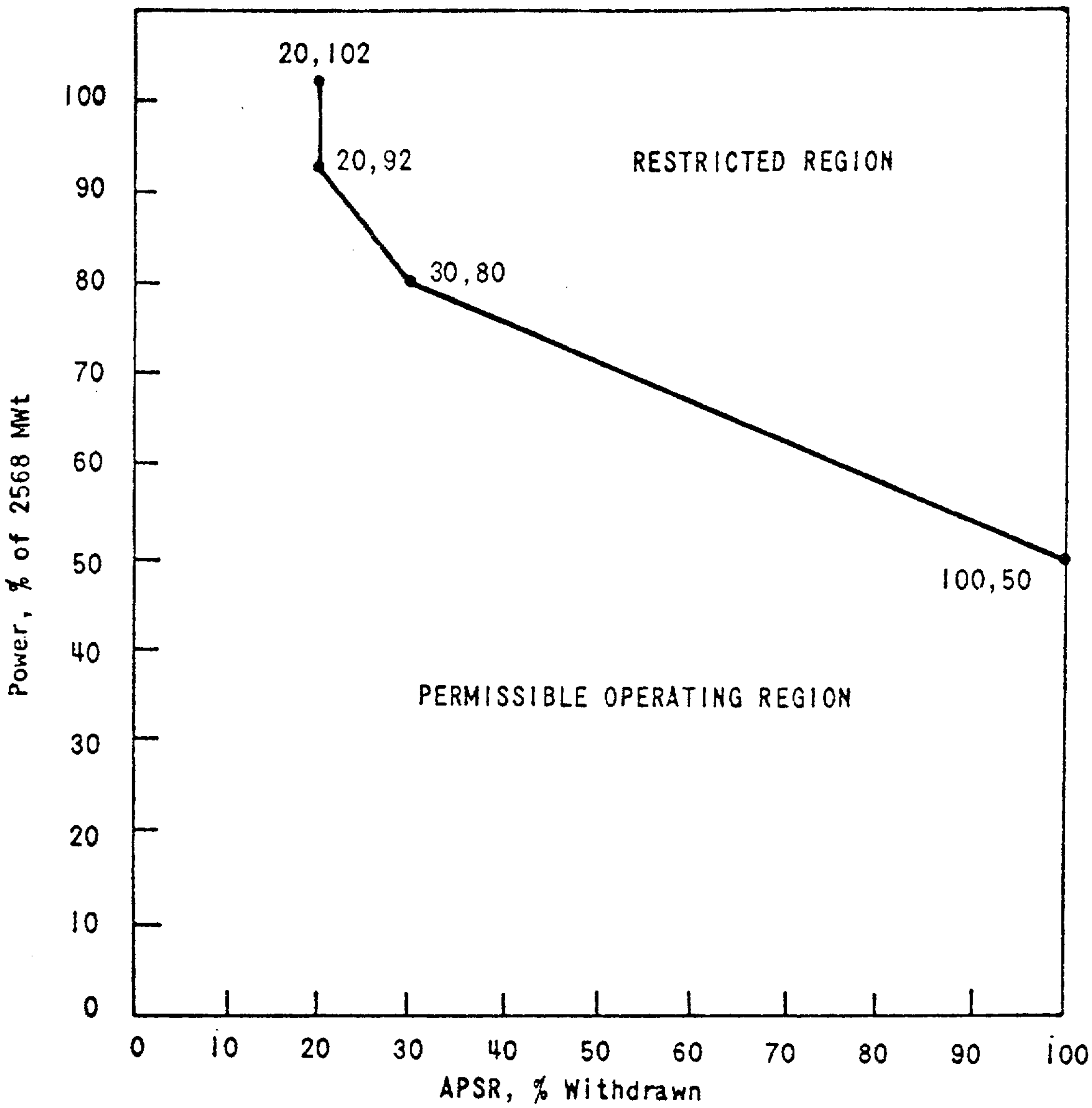
APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 ± 10 EFPD ANO-1, CYCLE 4

Figure 3.5.2-4A



APSR POSITION LIMITS FOR OPERATION
FROM 100 ± 10 TO 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-4B



APSR POSITION LIMITS FOR OPERATION
AFTER 250 ± 10 EFPD
ANO-1, CYCLE 4

Figure 3.5.2-4C

3.5.5 Fire Detection Instrumentation

Applicability

This specification applies to fire detection instrumentation utilized within fire areas containing safety related equipment or circuitry, for the purposes of protecting that safety related equipment or circuitry.

Objective

To provide immediate notification of fires in areas where there exists a potential for a fire to disable safety related systems.

Specification

3.5.5.1 A minimum of 50% of the heat/smoke detectors in the locations specified in Table 3.5-5 shall be operable.

3.5.5.2 If less than 50% of the fire detectors in any of the locations designated in Table 3.5-5 are operable, within one hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour and restore the equipment to operable status within 14 days or prepare and submit a report to the Commission pursuant to Specification 6.12.3.2(b) within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to operable status.

Bases

The various detectors provide alarms that notify the operators of the existence of a fire in its early stages thus providing early initiation of fire protection. The detectors in the main and auxiliary control rooms also provide automatic fire protection initiation.

The detectors required to be operable in the various areas represent 1/2 of those installed.

Operability of the fire detection instrumentation ensures that operable warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas(s) is required to provide detection capability until the inoperable instrumentation is restored to operability.

Table 3.5-5

Safety-Related Areas Protected By Heat/Smoke Detectors

1. Each of the four reactor building cable penetration areas.
2. Each of the four cable penetration rooms.
   (Zones 149-E, 144-D, 112-I, 105-T)*
3. Each of the two emergency diesel generator rooms.
   (Zones 86-G, 87-H)
4. North switchgear room.
   (Zone 99-m)
5. South switchgear room.
   (Zone 100-N)
6. Main control room.
   (Zone 129-F)
7. Auxiliary control room ceiling.
8. Auxiliary control room floor.
9. Each diesel generator fuel vault.
10. Cable spreading room.
    (Zone 97-R)
11. Battery charger and inverter rooms and hallway.
    (Zone 98-J)
12. Spent fuel area.
    (Zone 159-B)
13. Computer transformer room.
    (Zone 167-B)
14. Controlled access area.
    (Zone 128-E)
15. Tank room.
    (Zone 68-P)
16. Electrical equipment room.
    (Zone 104-S)
17. North upper piping penetration room.
    (Zone 79-U)
18. South upper piping penetration room.
    (Zone 77-V)
19. Condensate demineralizer area.
    (Zone 73-W)
20. Compressor room.
    (Zone 76-W)
21. Radwaste processing area.
    (Zone 20-Y)
22. Storage and pipe area.
    (Zone 34-Y)
23. Pipe area.
    (Zone 40-Y)
24. South lower piping penetration room.
    (Zone 46-Y)
25. Penetration Ventilation room.
    (Zone 47-Y)
26. North lower piping penetration room.
    (Zone 53-Y)
27. East decay heat removal pump room.
    (Zone 10-EE)
28. West decay heat removal pump room.
    (Zone 14-EE)
29. Intake structure.

* Zone numbers reflect nomenclature in the Fire Hazards Analysis and are listed for clarification only.

3.18 Fire Suppression Sprinkler Systems

Applicability

This specification applies to the following fire suppression sprinkler systems protecting safety-related areas:

a. Each of the four reactor building cable penetration areas.

b. Each of the four cable penetration rooms.

c. Each of the two emergency diesel generator rooms.

d. Cable spreading room.

e. Each of the two diesel generator fuel vaults.

f. Hallway-El 372. (Zone 98-J)

*g. Condensate demineralizer area.

Objective

To assure that fire suppression is available to safety-related equipment located in the above-listed areas.

Specification

3.18.1 The above-listed sprinkler systems shall be operable at all times.

3.18.2 With one or more of the above-listed sprinkler systems inoperable, establish a continuous fire watch (or operable smoke and/or heat detection equipment with control room alarm) with backup fire suppression equipment for the applicable area(s) within one hour. Restore the system(s) to operable status within 14 days or prepare and submit a Report to the Commission pursuant to Specification 6.12.3.2(b) within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system(s) to operable status.

Bases

Safety related equipment located in various areas is protected by sprinkler systems. The operability of these systems ensures that adequate fire suppression capability is available to confine and extinguish a fire occurring in the applicable areas. In the event a system is inoperable, alternate backup fire fighting equipment or operable detection equipment is required to be made available until the inoperable equipment is restored to service.

*To be implemented no later than July 30, 1979.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 43 TO LICENSE NO. DPR-51
ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE, UNIT NO. 1
DOCKET NO. 50-313

1.0 Introduction

By letters dated November 9, 1978, as supplemented February 27 and April 26, 1979 (References 1 and 2, respectively), Arkansas Power and Light Company (AP&L) requested amendment of Appendix A to Facility Operating License No. DPR-51 for Arkansas Nuclear One - Unit No. 1 (ANO-1). Section 5 summarizes the proposed changes of this amendment to the Technical Specifications (TS).

The AP&L submittal of November 9, 1978 was presented to support operation of Cycle 4 following the refueling performed at the end of Cycle 3. As such, the analysis presented in the submittal was based on the intended exposure for Cycle 4 of 387 effective full power days (EFPD). Information submitted describes the fuel system design, nuclear design, thermal-hydraulic design, accident analyses, and startup test program.

The refueling of ANO-1 for Cycle 4 will result in a core loading consisting of 64 fresh Mark B-4 assemblies, 57 once burned assemblies, and 56 twice burned assemblies. The fuel management has been changed from a conventional three fuel batch out-in scheme to a three fuel batch in-out-in scheme. The key feature of this scheme is the extensive use of fixed burnable poison in fresh reload fuel which will be loaded in the core interior rather than on the core periphery. The maximum fuel batch exposure at the end of Cycle 4 is predicted to be 28,300 MWD/MTU and hence is considerably less than the value of 33,000 MWD/MTU used in staff environmental considerations. This report addresses Cycle 4 operation only. Operation of successive nominal 18 month fuel cycles will result in fuel batch exposures in excess of 33,000 MWD/MTU during Cycle 6. The environmental impact of extended fuel burnup is to be addressed prior to Cycle 6 operation.

2.0 Evaluation of Modifications to Core Design

2.1 Fuel System Design

The 64 fresh Mark B-4 fuel assemblies which are to be loaded for Cycle 4 are mechanically identical to previously approved and utilized fuel assemblies at ANO-1 and other Babcock and Wilcox (B&W) supplied nuclear steam supply systems, NSSS. The mechanical design of the fresh fuel was not re-evaluated by the staff for Cycle 4.

Modified burnable poison rod assembly retainers (Ref. 3) are to be used in Cycle 4 to insure positive retention of the burnable poison rod assemblies (BPRA's). These retainers have been previously approved for retention of Orifice Rod Assemblies (ORA's). Mechanical and thermal-hydraulic compatability of the BPRA retainers has been previously reviewed and accepted. The BPRA's weigh 49 lbs. Coolant flow past the BPRA's has been predicted by the licensee to provide a net lift force of 53 lbs. Therefore, there is a net upward force of 4 lbs. The retainers provide a downward force of 47 lbs. and hence a minimum positive holddown force in excess of 30 lbs. Use of the modified BPRA retainers will therefore insure positive retention of the BPRA's. These retainers have been designed for one operating cycle and are to be replaced if BPRA's or ORA's are utilized in future cycles.

2.1.1 Cladding Creep Collapse

Fuel rod cladding creep collapse analyses have been performed for the most limiting (i.e., twice burnt exposed batch 4 fuel assemblies) fuel assembly to be used in Cycle 4. The analyses were performed according to the methods and assumptions described in References 4 and 5. These analyses predict that the time to rod cladding collapse will be in excess of 30,000 effective full power hours. The maximum batch 4 assembly burnup during Cycle 4 is predicted to be 23,112 EFPH (Table 4.1, Ref. 1). We conclude that cladding creep collapse has been suitably considered.

2.1.2 Cladding Stress and Strain

Stress calculations have been performed for a generic fuel rod model and strain calculations for a generic pellet model. These models and calculations have been approved for prior ANO-1 reloads. The licensee has asserted that Cycle 4 parameters are enveloped by these generic models. The licensee's calculations show that in no case does the stress exceed the yield.

2.1.3 Fuel Thermal Design

The introduction of the batch 6 fuel does not introduce significant differences in fuel thermal performance relative to the other fuel remaining in the core. The predicted linear heat rate to centerline melt (20.15kw/ft) is the same for batches 4, 5 and 6. At the core average linear heat rate

(5.8kw/ft), the licensee predicted nominal fuel temperatures of batches 4, 5 and 6 would be approximately 1300 F. These values are typical of all PWR's. Licensee calculations were performed using the approved computer code TAFY-3 (Ref. 6). It is noted that the code TACO (Ref. 7) has also been approved for fuel temperature calculations and is the staff preferred code. Based on the Cycle 4 predicted values and current approval of the analytic techniques used to make these predictions, the staff considers the fuel thermal design acceptable and provides for no reduction in the margin of safety.

2.2 Nuclear Design

Figure 3-1 of Reference 1 indicates the core loading arrangement for ANO-1 Cycle 4; the initial enrichments and burnup distributions are given in Figure 3-2. An unconventional fuel management scheme has been utilized.

An in-out-in fuel management scheme has been adopted. Fresh 3.19 w/o U235 fuel will be loaded in the core interior in a checkerboard pattern. Next cycle this fuel will be loaded on the core periphery. In its third resident cycle the fuel will once again be loaded in the interior of the core in a checkerboard fashion, hence the term "in-out-in". The fresh fuel will contain BPRA's to hold down local reactivity. Three concentrations of boron carbide (in an alumina matrix) will be employed to tailor the radial power distribution. By loading fresh fuel in the core interior, rather than on the periphery, neutron leakage is reduced. In turn for a fixed core enrichment the cycle length will be increased. Alternately the designer may increase the cycle length by increasing the average core enrichment. Both techniques have been used for Cycle 4.

In-out-in fuel management is believed to tax the nuclear designer's analytic capability to a greater extent than conventional three batch fuel management. To insure that achieved power distributions in the core are within the bounds assumed in the safety and setpoint analyses, monthly incore power maps are to be taken. This is a current TS requirement. Power distributions are to be compared with predicted distributions and the licensee has committed to report the deviations as part of the plant's monthly operating report.

Reactivity control and power distribution control will be maintained by control rods, axial power shaping rods (APSR) and soluble boron concentration control. The rod locations are given in Figure 3-3 of Reference 1. The core will be operated with control rods inserted at power to 250 EFPH and the APSR's deeply inserted.

The projected Cycle 5 length is 387 EFPD with a predicted cycle burnup of 12,111 MWD/MTU.

Cycle 5 nuclear parameters including critical boron concentrations, control rod worths, Doppler coefficients, moderator coefficients, xenon worth and effective delayed neutron fractions have been calculated using the approved PDQ07 code (Reference 8). These are presented in Table 5-1 of Reference 1 and compared to the Cycle 3 values. Relative to Cycle 3, predicted critical boron concentrations have increased due to the greater excess reactivity at beginning of life which is required to achieve the 18 month fuel cycle. The increased soluble boron concentration and use of BPRA;s will make the core "blacker" to thermal neutrons at beginning of Cycle 4 relative to Cycle 3. The extended cycle will result in more fission products and hence a "blacker" core at end of Cycle 4 relative to Cycle 3. Small changes in the power defect, Doppler coefficient, moderator temperature coefficient and inverse boron worth are consistent with increased core blackness.

Shutdown margins have been calculated for BOC and EOC (Table 5-2 of Reference 1). The calculated minimum shutdown margin during Cycle 4 is 1.77% $\Delta K/K$ which is larger than the value of 1% $\Delta K/K$ assumed in cooldown accident analyses by an adequate margin.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic design conditions for ANO-1 Cycle 4 are included in Table 6-1 of Reference 1. Only the minimum departure from nucleate boiling ratio at steady state differs from the Cycle 3 value.

The small difference, a 0.02 reduction in predicted DNBR at steady state 112% overpower, is attributable to the assumption in the thermal-hydraulic analyses that the hot assembly contained a BPRA. This assumption is consistent with beginning of Cycle 4 power distribution calculations (Fig. 5-1, Reference 1) which predict that the hot assembly will in fact contain a BPRA.

2.3.1 Removal of Orifice Rod Assemblies

Orifice rod assemblies were removed, bypass flow reanalyzed, and required setpoints adjusted as part of the Cycle 3 evaluation. Relative to Cycle 3 an additional six ORA's will be removed (44 to 50 ORA's) for Cycle 4. The core bypass flow will be negligibly effected (approximately 0.1% decrease in core flow) by this change. The flux/flow trip setpoint will be reduced to accommodate this change. This setpoint is based on an assumed two-pump coastdown from 102% indicated power (108% core power assumed in analyses).

2.3.2 Effect of Rod Bow on Thermal Design

The potential effects of fuel rod bow has been reviewed generically in Reference 9. Based on the rod bow model approved by the staff, ANO-1 has applied a DNBR penalty of 11.2% for fuel rod bow which has been incorporated in the variable low-pressure trip function and flux/flow setpoint.

3.0 Evaluation of Accidents and Transients

General:

The licensee has stated that each accident analyzed in the FSAR has been examined and has found, with the exception of the moderator dilution postulated event, to be bounded by the FSAR and/or the Fuel Densification Report and/or subsequent cycle analyses.

The staff has concluded that the consequences of hypothesized events are no worse than those stated in the FSAR or previous submittals, that is, part 20 and part 100 dose rate limits will not be exceeded in the event of an anticipated operating occurrence or accident respectively.

The removal of ORA's in Cycles 3 and 4 has resulted in increased bypass flow and corresponding decrease in core flow (approximately 1% decrease). This effect has been considered in calculating the steady state DNBR conditions. The licensee has assumed that the incremental DNBR degradation during an anticipated operating occurrence (AOO) or accident has not been substantially altered by these changes. Hence, the FSAR analyses are bounding. This approximation is considered acceptable.

Specific Analysis:

The licensee has stated (Reference 1) that the generic B&W ECCS analysis (Reference 10) is applicable to ANO-1, Cycle 4. Based on the minimal core changes for Cycle 4 the staff accepts this assertion.

The conclusion presented in the FSAR is that, in the event of a steam line break (SLB) accident, a small fraction of the 10 CFR 100 dose rate would be reached. The supporting analysis assumed a 1% $\Delta\delta$ safeguards allowance (shutdown margin). The predicted minimum shutdown margin during Cycle 4 is 1.77% $\Delta\delta$. On these bases the consequences of a hypothesized SLB are considered acceptable for Cycle 4 operation.

The larger initial soluble boron concentration at beginning of Cycle 4, relative to the reference analyses, will result in a slightly larger reactivity insertion rate for the postulated moderator dilution accident. For an assumed 500 gpm dilution rate the reactivity insertion is predicted to be 1.235×10^{-5} $\Delta K/K/sec$. A value of 1.227×10^{-5} $\Delta K/K/sec$ was assumed in the Cycle 4 analysis. The higher insertion rate will result in a faster reactor trip on high reactor coolant system pressure. The licensee has predicted that the reactor coolant pressure will increase by less than 10 psi relative to the FSAR analyses leaving a margin of approximately 300 psi to the safety limit. This change is insignificant.

The dropped rod accident analysis reported in the FSAR is based on an assumed dropped rod worth of 0.65% $\Delta K/K$, and a peak post dropped enthalpy rise, $F\Delta H$, equal to the design value, 1.78. Turbine runback to 60% of rated power was assumed not to function. The licensee has predicted that the maximum dropped rodworth is 0.2% $\Delta K/K$. Post drop values of $F\Delta H$ have not been provided by the licensee. The peak enthalpy rise would increase by less than 20% following the drop of a control rodworth only 0.2% $\Delta K/K$. Following the rod drop and assuming no turbine runback, the core will return to rated power. Since the core is typically operated with an initial enthalpy rise approximately 15% (or greater) less than the design peak, and even if the core was initially at the design peak and the peak **were** to increase by 20% there would still exist margin to DNBR limits (at 100% power), the dropped rod analysis is considered adequate for Cycle 4.

The most limiting transient considered as a part of the original licensing process was the postulated loss of AC power. The loss of all AC power would result in loss of reactor coolant pumps and leave forces flow as well as loss of normal feedwater. The postulated loss of feedwater is considered less limiting than the loss of AC power assuming no other single or multiple failures. Therefore, loss of feedwater has not been reviewed as a part of the proposed license amendment.

The maximum ejected rod at hot full power, Cycle 4, is predicted to be 0.55% $\Delta\delta$ or 0.89\$. The FSAR analysis assumed a rod worth of 0.65% $\Delta\delta$ or 0.92\$. The predicted Doppler coefficients during Cycle 4 are substantially less than the values used in the FSAR analyses. These are conservative changes relative to the FSAR analyses. The delayed neutron fraction (βeff), is predicted to be smaller than assumed in the FSAR. The effect of the smaller value of βeff is a slower decay of the neutron flux once the peak value is reached. This is a non-conservative change. The above cited conservatisms are substantially larger than this non-conservatism. FSAR calculations were run using a point kinetics design model assuming the design three dimension peak and compared to two dimensional space-time kinetics calculations. The design model was shown to be conservative. Post ejected rod peaking factors have not been presented for Cycle 4 nor for the FSAR analyses. Hence, a direct comparison cannot be made of these values. The conclusion of this analysis presented in the FSAR is that there exist substantive margins for this accident to limiting enthalpy deposition values. On this basis the applicability of the FSAR analysis to Cycle 4 is accepted.

4.0 Startup Tests

Startup tests are described in Reference 1. These tests are consistent with the startup tests performed in association with other recent B&W reloads. We have reviewed the tests in terms of their intended purpose and consider them acceptable. AP&L has agreed to provide a startup test report (Reference 2).

5.0 Evaluation of TS Changes

Proposed modifications to the ANO-1 TS are described below:

(1) TS Fig. 3.5.2-1A, B, C, Fig. 3.5.2-2A, B, D

Rod position limits insure that values of shutdown margin, ejected rod worth, peak linear heat rate, and peak enthalpy rise, realized during the operating cycle are less than or equal to the values used in the safety analyses. They have been modified, relative to Cycle 3, to accommodate changes in predicted peaking factors with rods inserted in the core which in turn have been altered by the revised fuel management. The core is to be run at full power with control group 7 inserted and group 6 inserted as a bite bank till 250 EFPD and thereafter essentially unrodded with group 7 used as the bite bank. The Cycle 4 limits at full power are somewhat more restrictive than the Cycle 3 limits.

(2) TS Fig. 3.5.2-4A, B, D

The axial power shaping rods (APSR) are to be inserted near the bottom of the core throughout the cycle. APSR limits for Cycle 4 are more restrictive than the Cycle 3 limits.

(3) TS Fig. 3.5.2-3A, B, C

Operational power imbalance limits in conjunction with rod position limits insure that the peak linear heat rate as a function of core height limits are not exceeded. Relative to Cycle 3, limiting bottom peaked axial power shapes are to be excluded for Cycle 4. Hence, these Cycle 4 limits are, as the rod position limits, more restrictive than the Cycle 3 values.

(4) TS 3.5.2.5

Control group overlap is to be reduced from 25% to 20% overlap. The change reduces the extent over which an overlap region must be considered at any given power. Overlap regions exhibit higher planar peaking than non-overlap regions.

Items (1), (2), (3) and (4) are considered together and may be traded against each other; it is the convolution of these limits which determines the peak linear heat rate and enthalpy rise. We find that these proposed changes are acceptable and do not decrease the margin of safety.

(5) TS 3.2 and Figure 3.2-1

The increased boron acid addition tank volume is required to insure shutdown of the reactor to cold conditions (200 F) using soluble boron. The Cycle 4 core at beginning of cycle is simply more reactive (higher core average enrichment, lower core average burnup) than the Cycle 3 core.

(6) TS 2.3.1, Table 2.3-1, Figure 2.1.2 and 2.3.2

These small changes reflect revision of the flux/flow setpoint from 1.06 to 1.057. This proposed revision accommodates the addition of BPRA retainers and removal of an additional six ORA's.

Items (5) and (6) are considered separately. We find the proposed changes are acceptable and do not decrease the margin of safety.

6.0 High Pressure Injection (HPI) System Modifications

On April 28, 1978, the Commission issued an Order modifying License No. DPR-51 to require that certain operating procedures be implemented until facility modifications could be implemented to alleviate the ECCS small break problem. By letter dated October 27, 1978, supplemented by letter dated January 3, 1979, the licensee proposed certain facility modifications at ANO-1 to mitigate a small break LOCA without requiring operator action. By letter dated March 1, 1979, we accepted the licensee's proposed modifications. The proposed modifications would ensure that the proper flow uplift in the HPI lines and assure the minimum ECCS flow to the reactor coolant system. By letter dated May 14, 1979, the licensee verified that the proposed modifications were implemented and tested to assure the proper flow split to the redundant legs of the HPI system. Thus the actions required by the Commission's Order dated April 28, 1978 have been completed.

7.0 Environmental Consideration

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

8.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 23, 1979

References:

(1) W. Cavanaugh, III, Proposed TS, November 9, 1978, AP&L ltr (File 1511.1).

(2) D. A. Rueter, Cycle 4 Reload Report Questions, February 27, 1978, AP&L letter (File 024.6, 1511.1), and April 26, 1979 (File 0242.6, 1511.1).

(3) BPRA Retainer Design Report, BAW-1496, B&W, May 1978.

(4) Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 1, B&W, November 1976.

(5) ANO-1, Fuel Densification Report, BAW-1391, B&W, June 1973.

(6) C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-1004, B&W, May 1972.

(7) R. H. Stoudt, et al., TACO - Fuel Performance Analysis, BAW-10087, B&W, June 1976.

(8) H. A. Hassan, et al., B&W's Version of PDQ07 - User's Manual, BAW-10117, B&W, June 1976.

(9) Memo to D. B. Vassallo (NRC) from D. F. Ross (NRC), Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Therma. Margin Calculations for Light Water Reactors, February 16, 1977.

(10) R. C. Jones, et al., ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103A, Rev. 1, B&W, July 1977.

FIRE PROTECTION

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

SUPPLEMENT NO. 1

Dated: May 23, 1979

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Introduction

By letter dated February 23, 1979, the Arkansas Power and Light Company (the licensee) proposed amendment to Operating License No. DPR-51 which would change the Technical Specifications for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment would add additional smoke/heat detectors and fire suppression systems in appropriate tables of the Technical Specifications. The amendment was submitted in response to License Condition 2.c.(3) which requires certain modification related to fire protection completed by the end of the current refueling outage.

By letters dated January 18 and 31, 1979, and March 9 and 30, 1979, and supplemented by letters dated March 19 and 20, 1979, the licensee submitted design details, test plans and test results on facility modifications which are required to be implemented no later than the end of the third refueling outage in accordance with License Condition 2.c.(3).

By letter dated April 24, 1979, the licensee committed to have all items, which were scheduled for completion by the end of the current refueling outage completed as scheduled.

By letter dated March 13, 1979, the licensee requested relief as to the implementation date of installing all three hour related dampers in the ventilation ducts penetrating fire barriers.

Background and Discussion

On August 22, 1978, the Commission issued Amendment No. 35 to the ANO-1 operating license. This amendment added a condition to the license (2.c.(3)) which requires completion of the modifications identified in paragraphs 3.1 through 3.19 of the NRC's Fire Protection Safety Evaluation (FPSE) for ANO-1 dated August 19, 1978. This amendment also added a license condition which requires completion of these modifications in accordance with the schedule given in Table 3.1 of the FPSE. Of these modifications, the schedule calls for completion of eight items by the end of the third refueling outage which is currently underway and nearing conclusion.

By letter dated July 25, 1978, the licensee committed to install three-hour fire rated fire dampers in all ventilation ducts penetrating fire barriers.

Since that time the licensee identified many more locations where three-hour dampers are needed than what was expected. The licensee has proposed to install three-hour dampers in safety related areas where no dampers presently exist, but are needed. The licensee has proposed to test the dampers in locations which presently have one and one-half hour dampers. The results of these tests would not be available until September 1977. If any damper would not pass a three-hour rating test the licensee has proposed to replace it with a three-hour damper by the end of the fourth refueling outage. Also, the licensee has proposed to install three-hour dampers in as many nonsafety related areas as possible during the current refueling outage and complete the installation of three-hour dampers in the remaining nonsafety areas by the end of the fourth refueling outage.

Evaluation

We have reviewed the licensee's proposed Technical Specifications and found minor modifications were necessary. We have discussed the modifications with the licensee's staff and they have agreed to the modifications. We find the proposed Technical Specification changes as modified meet the requirements of License Condition 2.c.(3) and therefore are acceptable.

The licensee has indicated by letter dated April 24, 1979 that all modifications which were scheduled for completion by the end of the third refueling outage will be completed and tested as scheduled. We find it acceptable to implement these modifications by the end of the current refueling outage. We find that these modifications would improve the fire protection at ANO-1. We have reviewed the design details of Items 3.4, 3.8 and 3.18 and find them acceptable. However, our findings on items 3.3, 3.7 and 3.11 are pending completion of our review as to acceptability which is continuing. Items 3.6, 3.10 and 3.11 were found acceptable by the FPSE.

We have reviewed the licensee's request with respect to the installation of the three-hour rated fire dampers in ventilation ducts penetrating fire barriers. We find that the licensee will greatly increase the fire protection of the facility in the installation of additional three-hour dampers in ventilation ducts by the end of the current refueling outage.

We also find that the licensee would not significantly increase the fire protection capability of the facility by replacing those one-half hour dampers, which would not pass a three-hour test, with three-hour dampers and by completion of all installation of three-hour dampers in nonsafety related areas. Therefore, we find it acceptable to delay completion of installation of all three-hour dampers as requested until the end of the fourth refueling outage.

We have reviewed the test plan submitted by the licensee on March 9, 1979 for the testing of cable penetration fire stops installed in metal lath and plaster walls. We find the proposed test acceptably follows the recommendations and acceptance criteria of the staff.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Facility Operating License No. DPR-51, issued to Arkansas Power and Light Company, which revised Technical Specifications for operation of the Arkansas Nuclear One, Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to reflect plant operating limits for the fuel loading to be used during Cycle 4 and adds additional smoke/heat detectors and fire suppression systems in appropriate tables. The amendment does not authorize operation of the facility. Operation of the facility will be authorized only after the Director of Nuclear Reactor Regulation has confirmed satisfactory completion by the licensee of certain actions set forth in the Commission's Order of May 17, 1979 (44 F.R. 29997, May 23, 1979).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendments dated November 9, 1978 (as supplemented February 27 and April 26, 1979), and February 23, 1979 (as supplemented March 19, 20 and 30, 1979), (2) Amendment No. 43 to License No. DPR-51, (3) the Commission's related Safety Evaluation, and (4) Supplement No. 1 to the Fire Protection Safety Evaluation for ANO-1. 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 Arkansas Polytechnic College, Russellville, Arkansas. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 23rd day of May 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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