March 17, 1978

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Docket No. 50-313

Arkansas Power & Light Company
ATTN: Mr. William Cavanaugh, III
Executive Director, Generation and Construction
Post Office Box 551
Little Rock, Arkansas 72203

Gentlemen:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications in response to your application for amendment submitted by letter dated December 28, 1977, as supplemented by letters dated January 17, and 30, 1978, and March 3, 1978.

The amendment authorizes operation and modifies the Technical Specifications for Cycle 3.

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. 31
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Arkansas Power & Light Company

cc w/enclosures:
Phillip K. Lyon, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Mr. Daniel H. Williams
Manager, Licensing
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72203

Mr. John W. Anderson, Jr.
Plant Superintendent
Arkansas Nuclear One
Post Office Box 608
Russellville, Arkansas 72801

Arkansas Polytechnic College
Russellville, Arkansas 72801

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

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

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

w/cy of AP&L filings dtd:
12/28/77, 1/17/77, 1/30/78,
3/3/78:
Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. DPR-51

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

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated December 28, 1977, as supplemented by letters dated January 17 and 30, 1978, and March 3, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 17, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Revise the Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
9	9
9b	9b
12	12
14b	14b
30 & 30a	30
47 & 48	47 & 48
48b	48b
48bb	48bb
48bbb	48bbb
48c	48c
48cc	48cc
48ccc	48ccc
48d	48d
48dd	48dd
48ddd	48ddd
-	48f - 48h

New pages and changes in the revised pages are identified by marginal lines.

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.

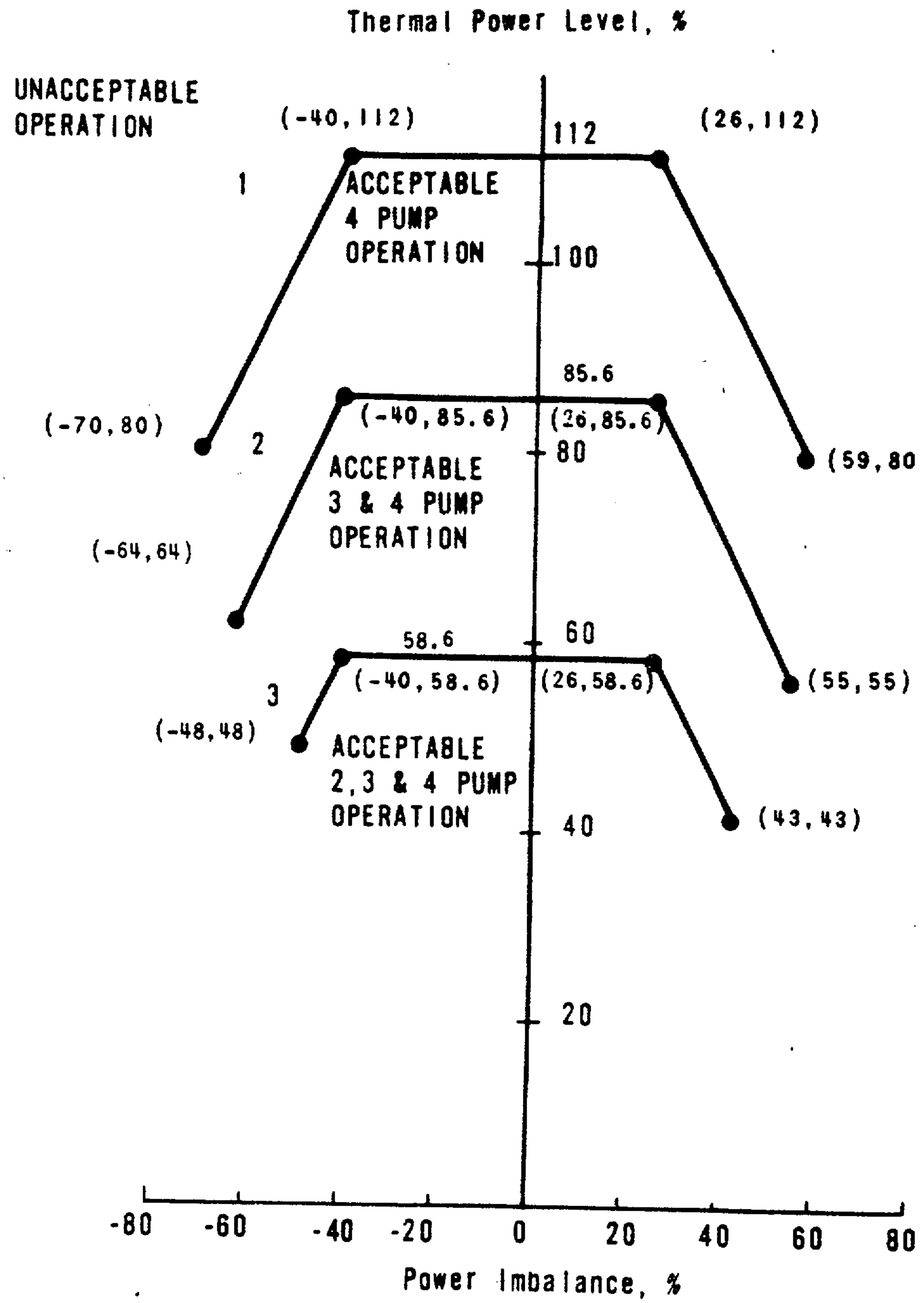
The maximum thermal power for three pump operation is 85.6 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.060 79.1 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced 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

ARKANSAS POWER AND LIGHT
CORE PROTECTION SAFETY LIMITS

Figure 2.1-2

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

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 106.0 percent and reactor flow rate is 100 percent or flow rate is 94.3 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.1 percent and reactor flow rate is 74.7 percent or flow rate is 70.7 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 52.3 percent and reactor flow rate is 49.2 percent or flow rate is 46.2 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.

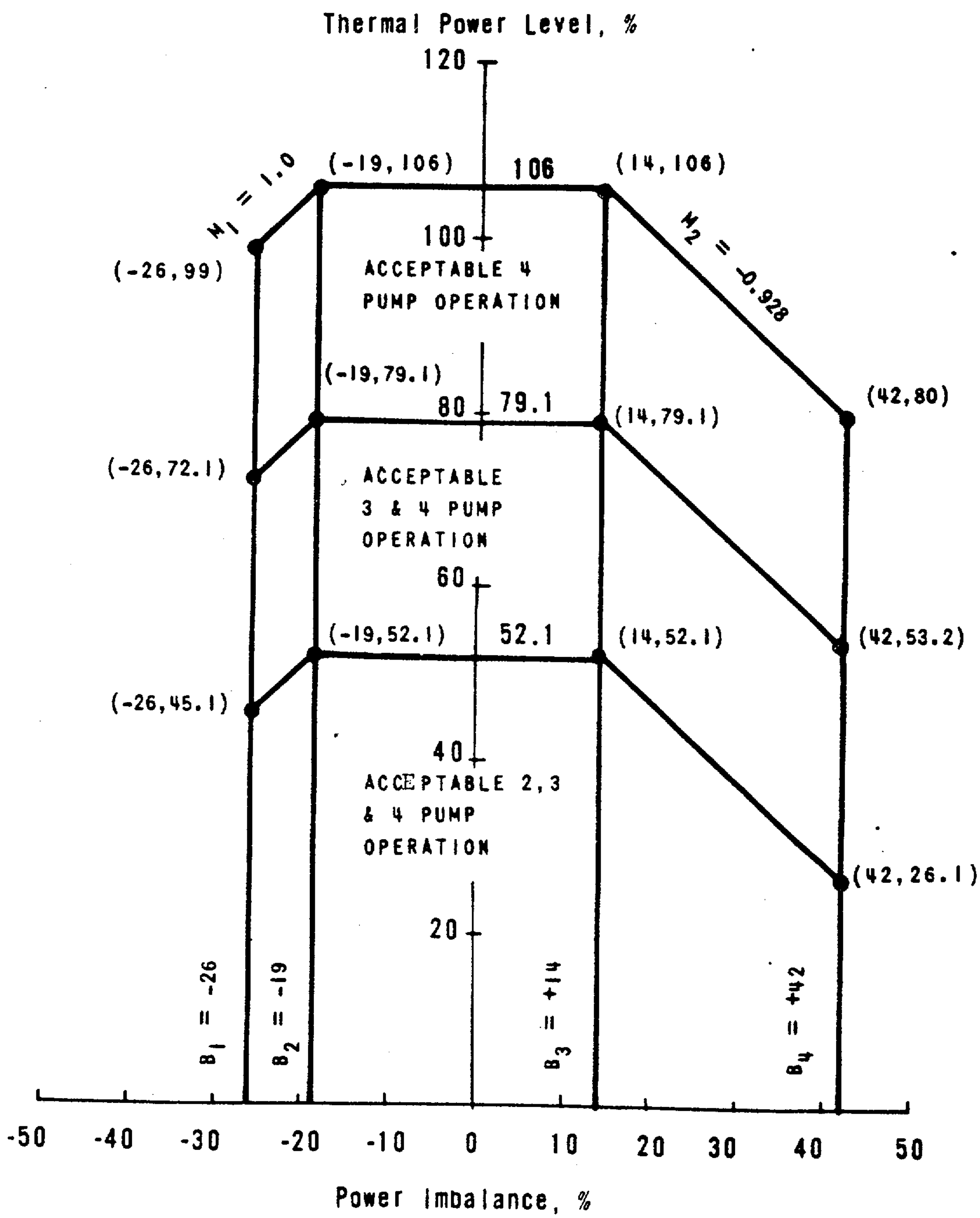
The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level 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 1.060 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



ARKANSAS POWER AND LIGHT COMPANY-UNIT 1
PROTECTIVE SYSTEM MAXIMUM ALLOWABLE
SETPOINTS

Figure 2.3-2

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is $\geq$95% of rated thermal power and shall be less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever thermal power is <95% of rated thermal power and the reactor is not shutdown.

3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.

Bases

A non-positive moderator coefficeint at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ corrected to 95% of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta k/k/°F$.

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 25% ± 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

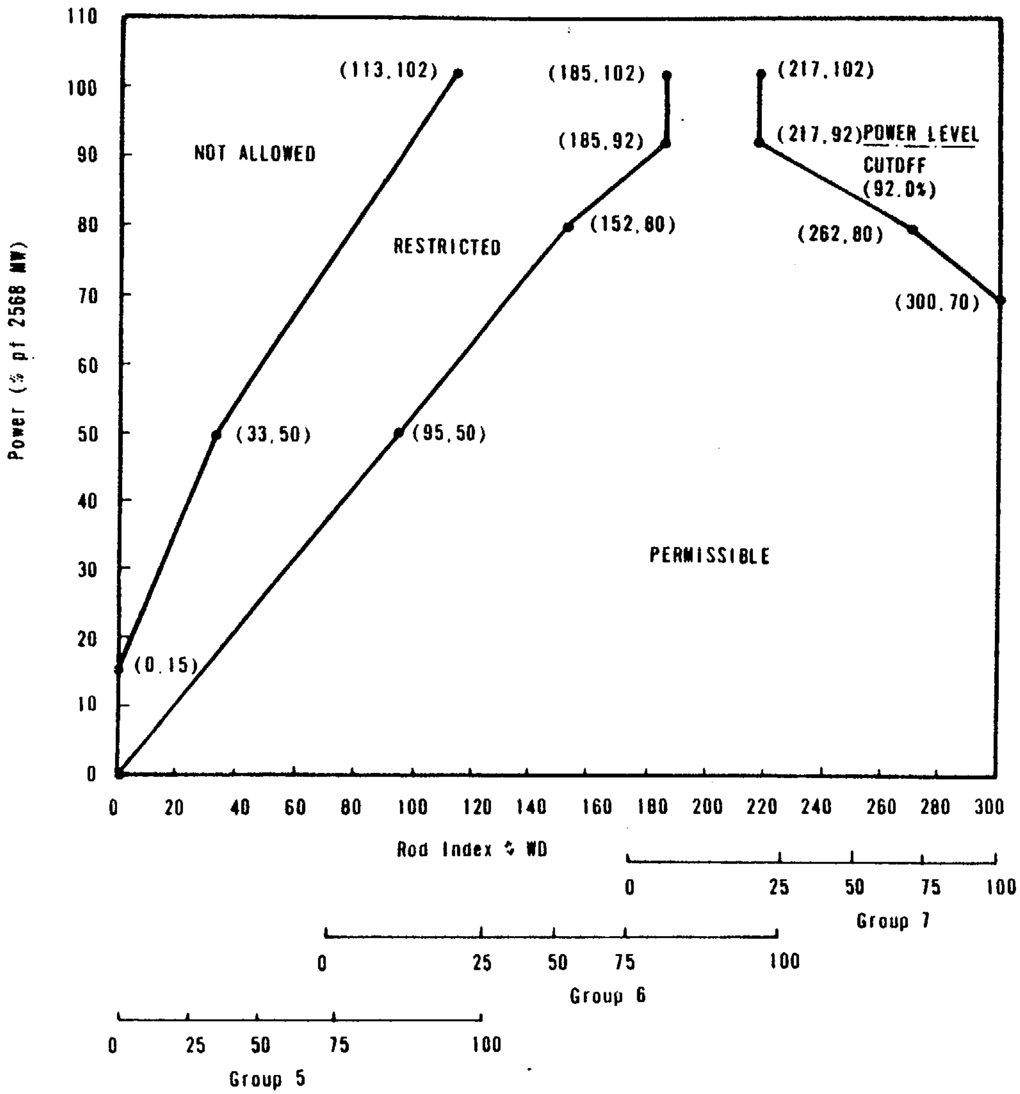
The power-imbalance envelopes defined in Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C are based on 1) LOCA analyses which have defined the maximum linear heat rate (See Fig. 3.5.2-4) such that the maximum clad temperature will not exceed the final Acceptance Criteria and 2) the Protective System Maximum Allowable 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 25 ±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:

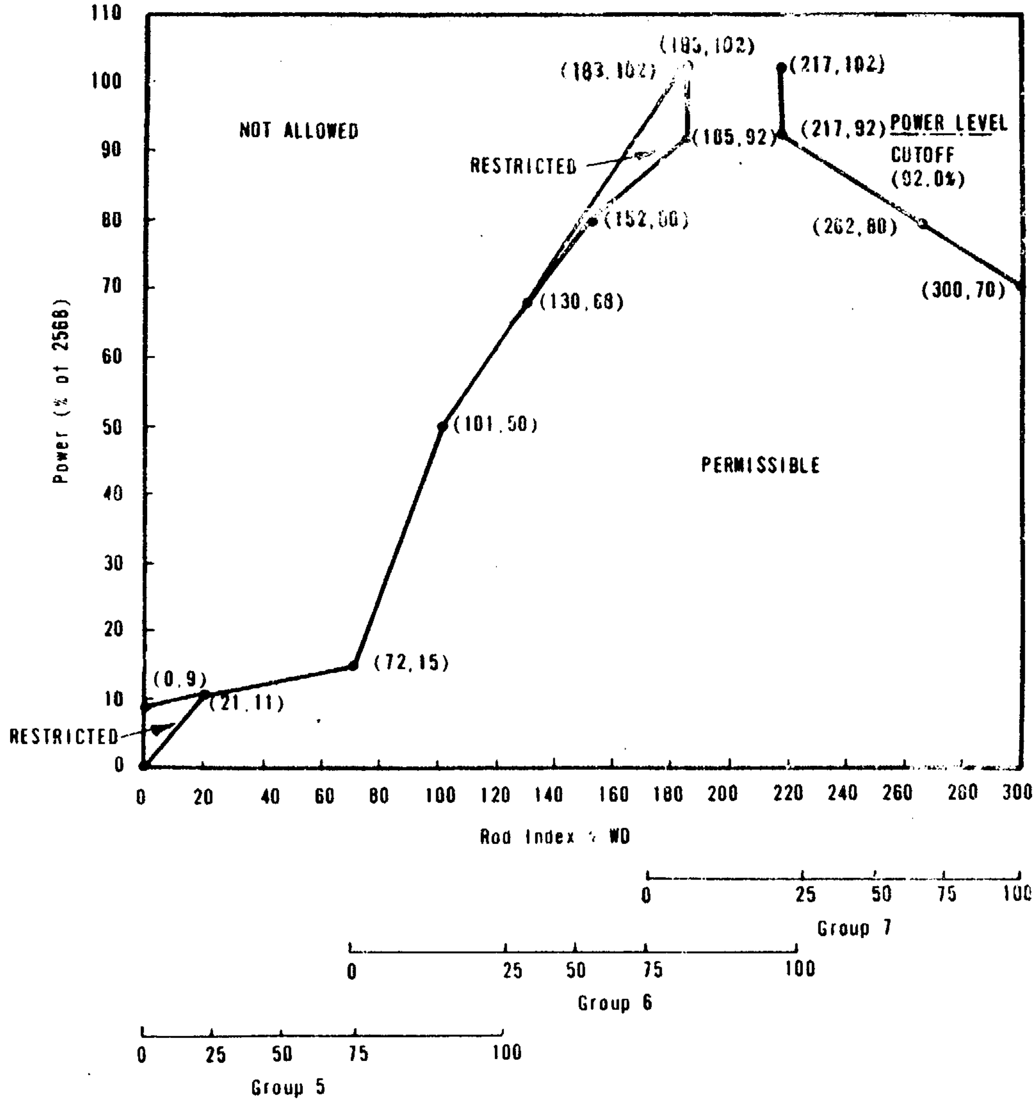
*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

Amendment No. ~~6~~, ~~21~~, 31



ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 0 TO 100 ± 10 EFPD ARKANSAS CYCLE 3

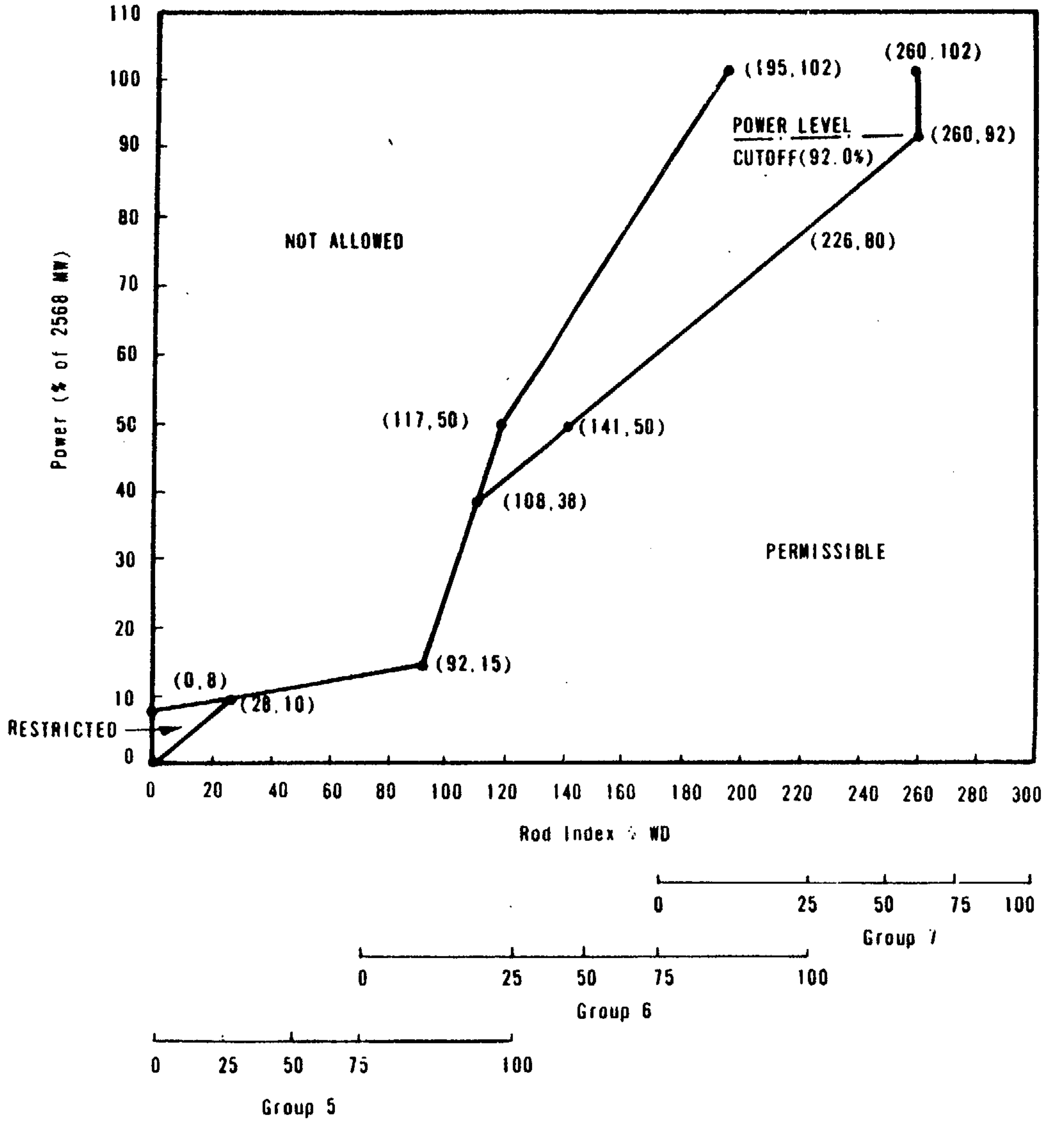
Figure 3.5.2-1A



ROD POSITION LIMITS FOR 4 PUMP OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD ARKANSAS CYCLE 3

Figure 3.5.2-18

Amendment No. ~~21~~, 31

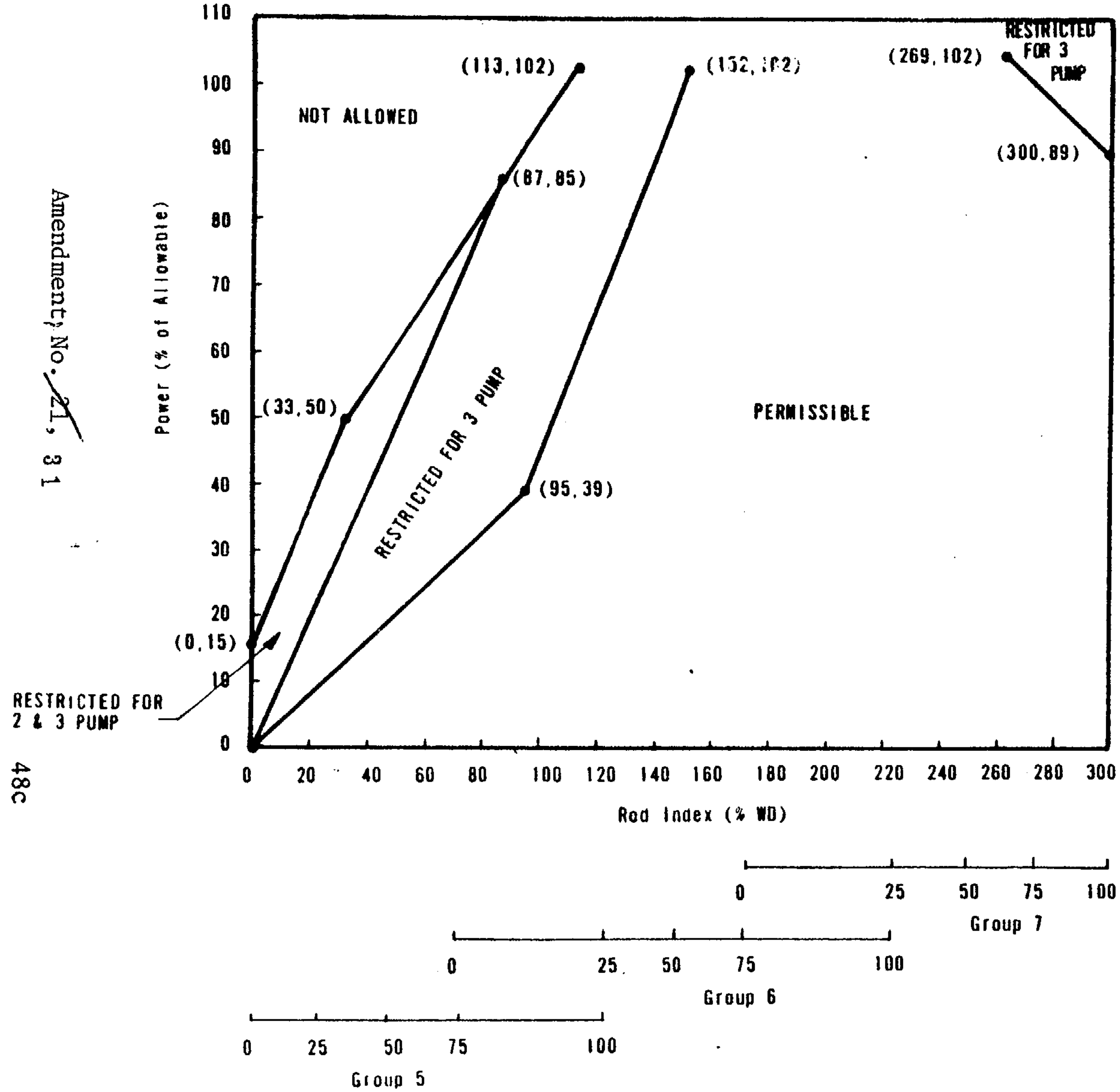


ROD POSITION LIMITS FOR 4 PUMP OPERATION AFTER 250 ± 10 EFPD ARKANSAS CYCLE 3

Figure 3.5.2-1C

48bbb

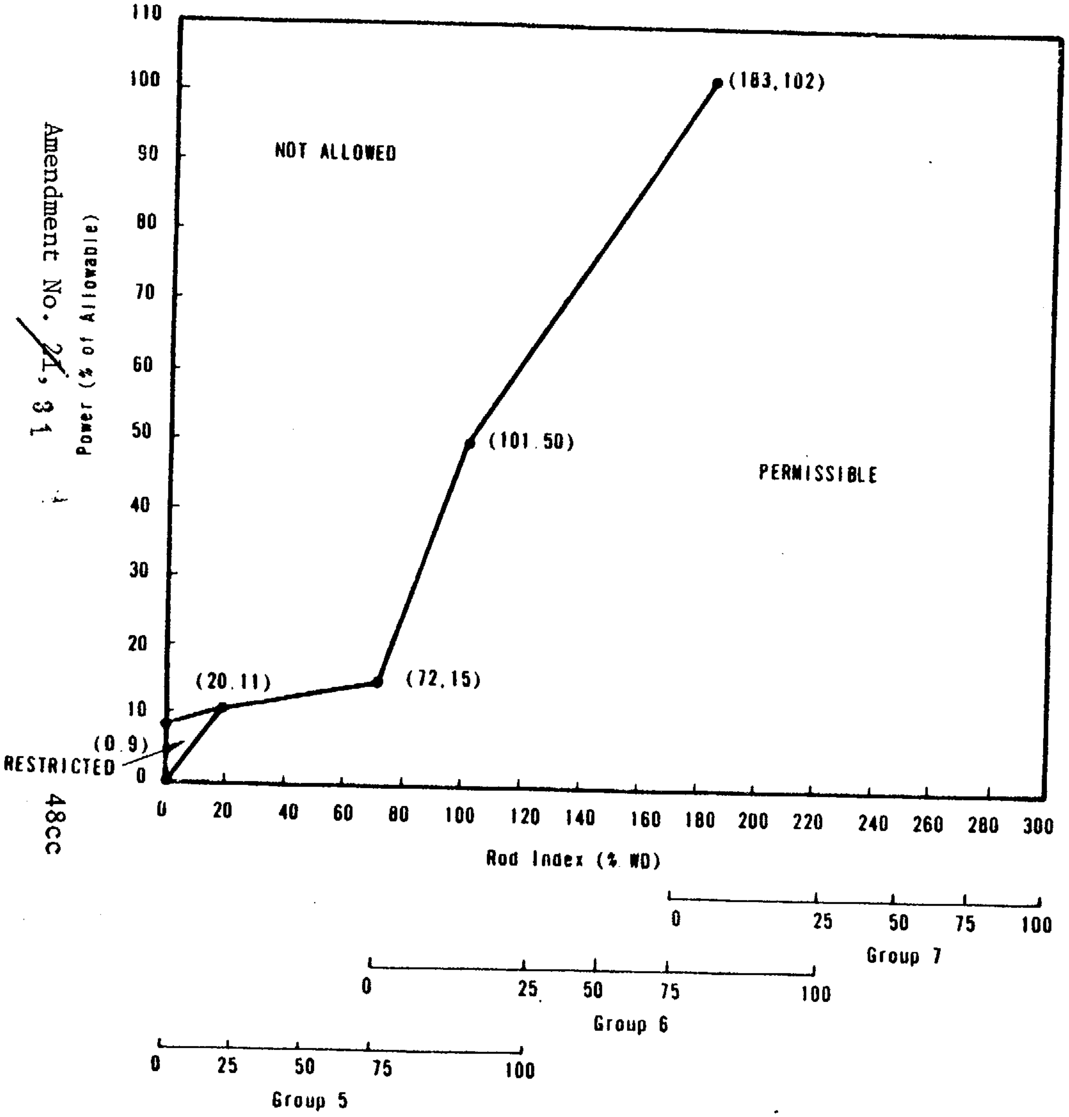
Amendment No. ~~21~~, 31



ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION
FROM 0 TO 100 ± 10 EFPD ARKANSAS CYCLE 3

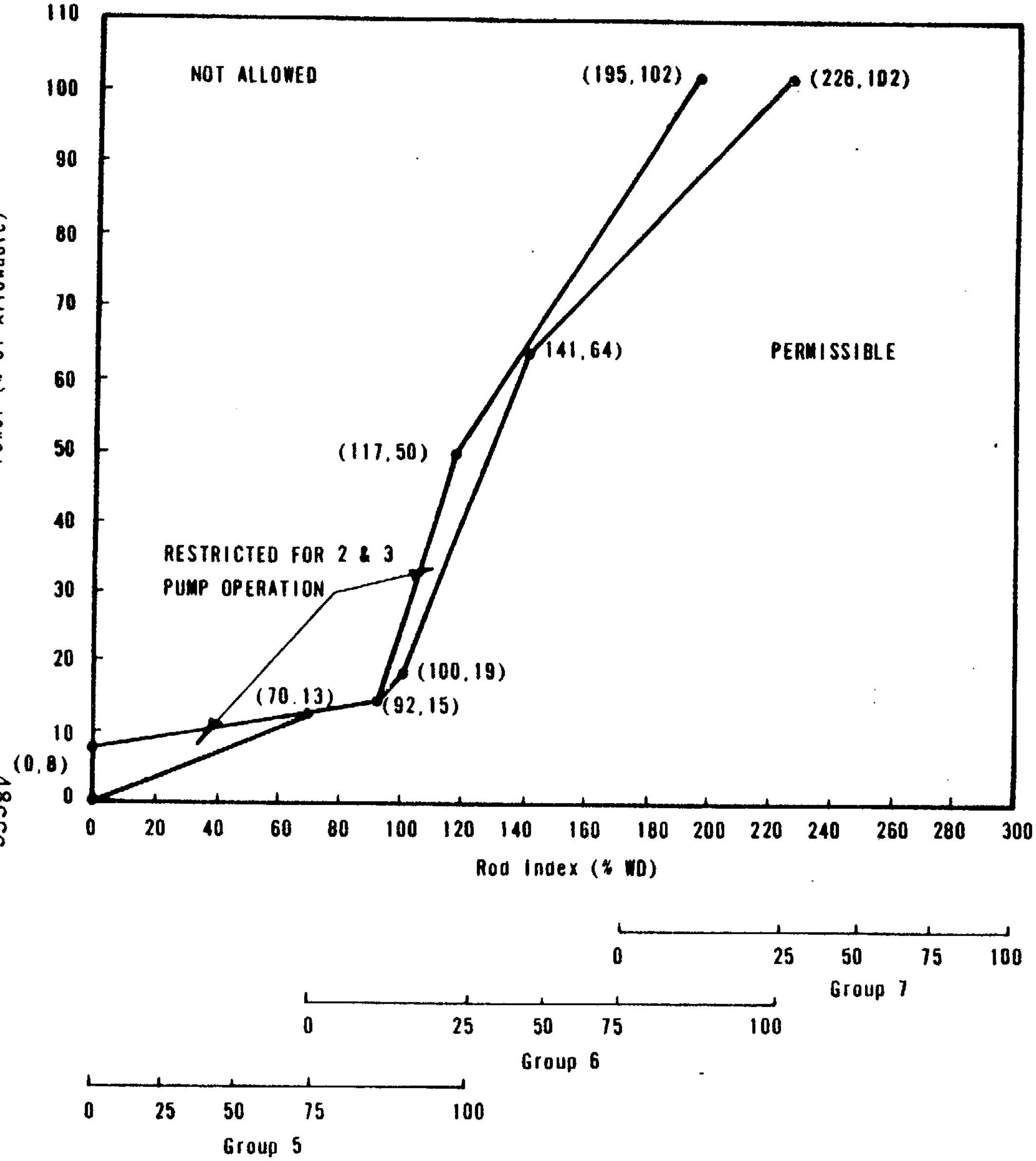
Figure 3.5.2-2A

Amendment No. ~~21~~, 31



ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD ARKANSAS CYCLE 3

Figure 3.5.2-2B

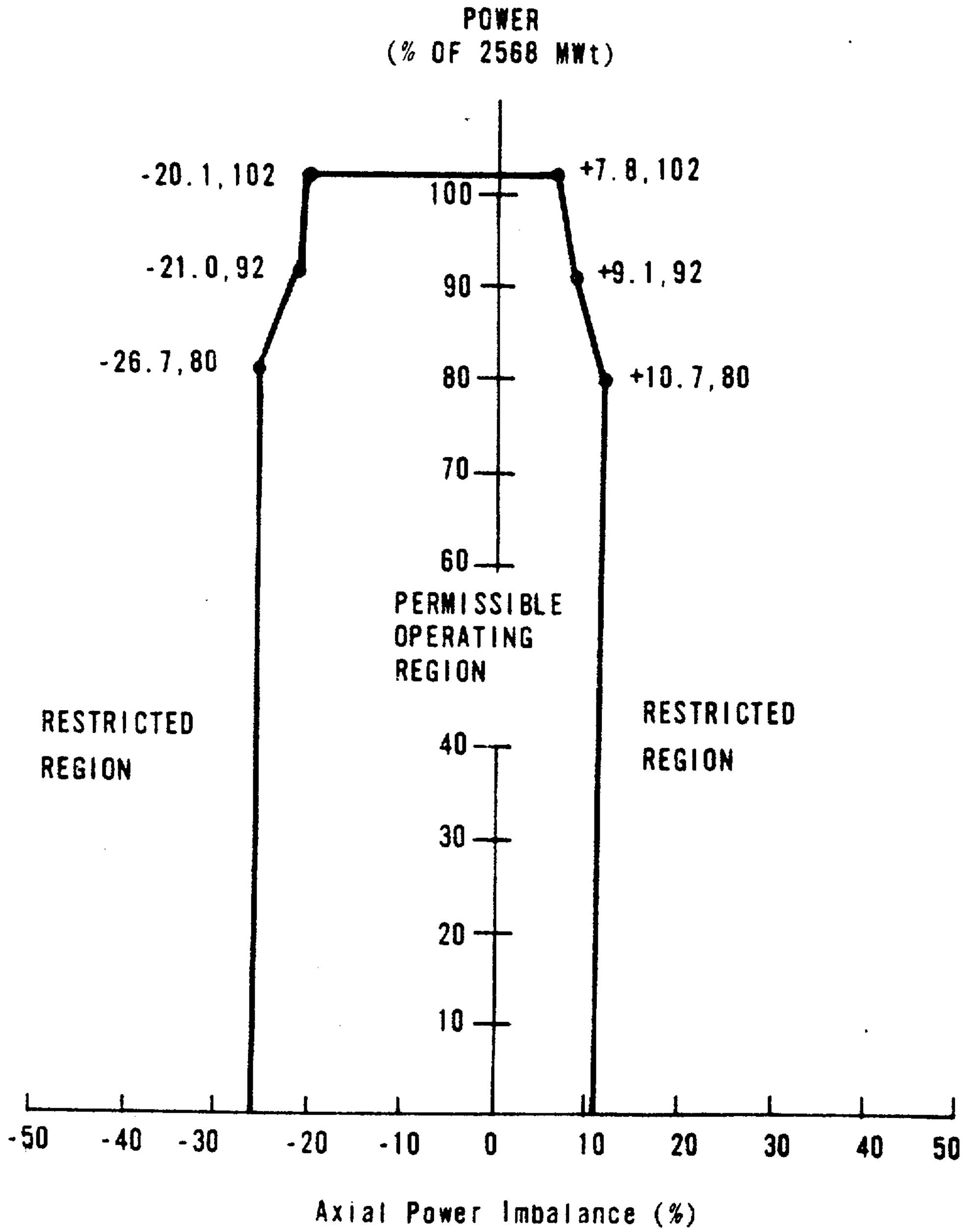


Amendment No. ~~21~~, 31

48ccc

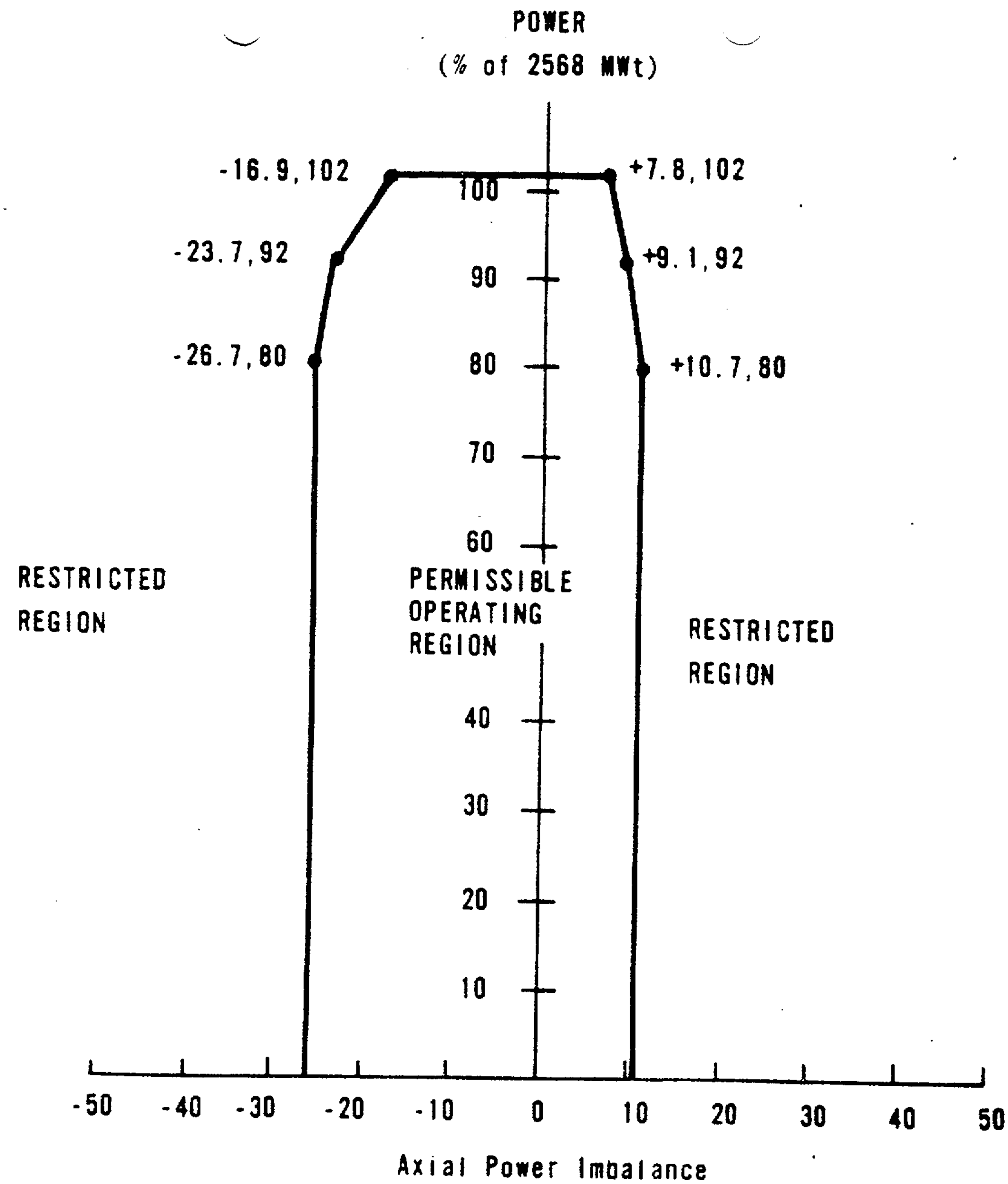
ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION
AFTER 250 ± 10 EFPD ARKANSAS CYCLE 3

Figure 3.5.2-2C



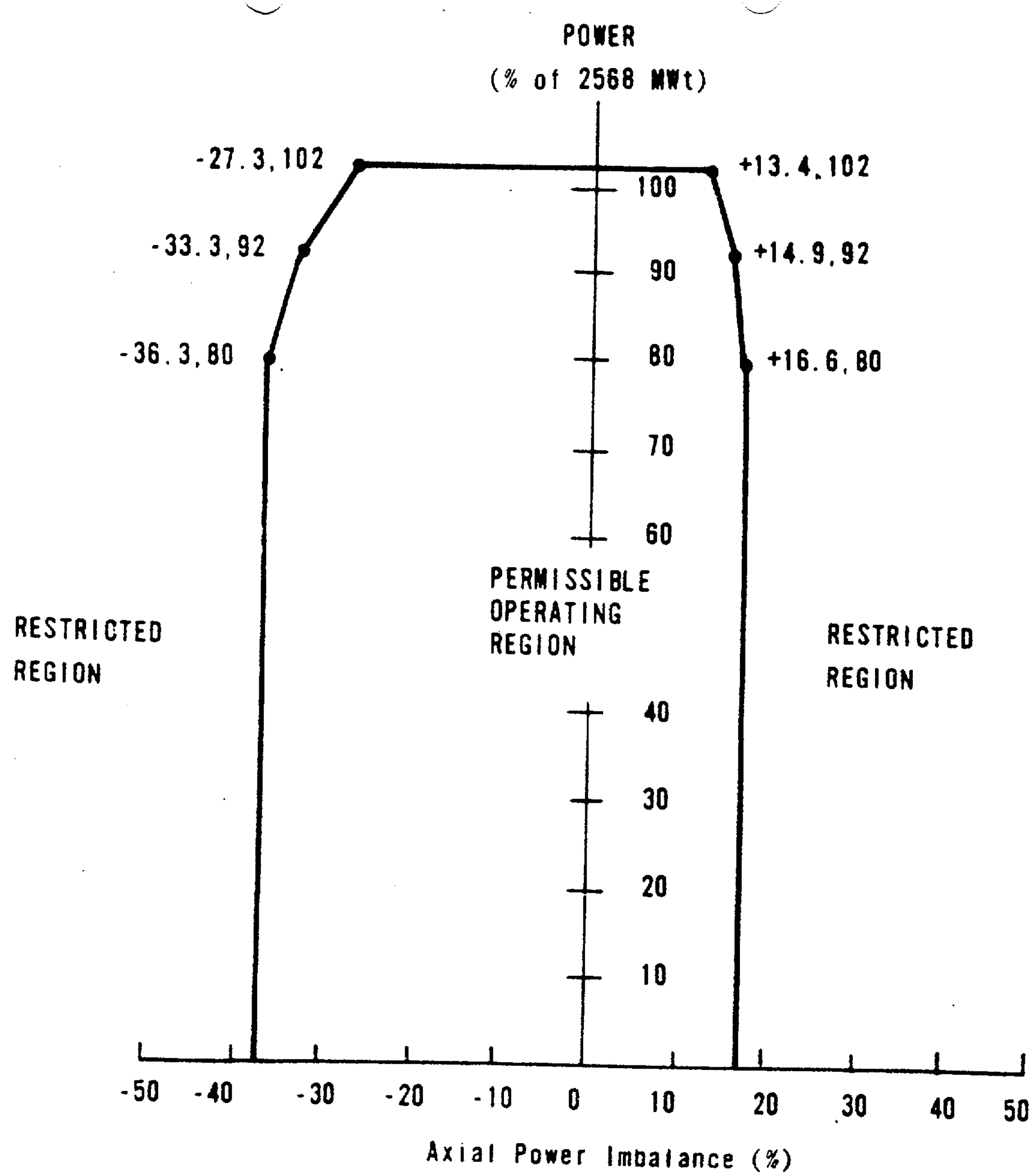
OPERATION POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 100 ± 10 EFPD
ARKANSAS CYCLE 3

Figure 3.5.2-3A



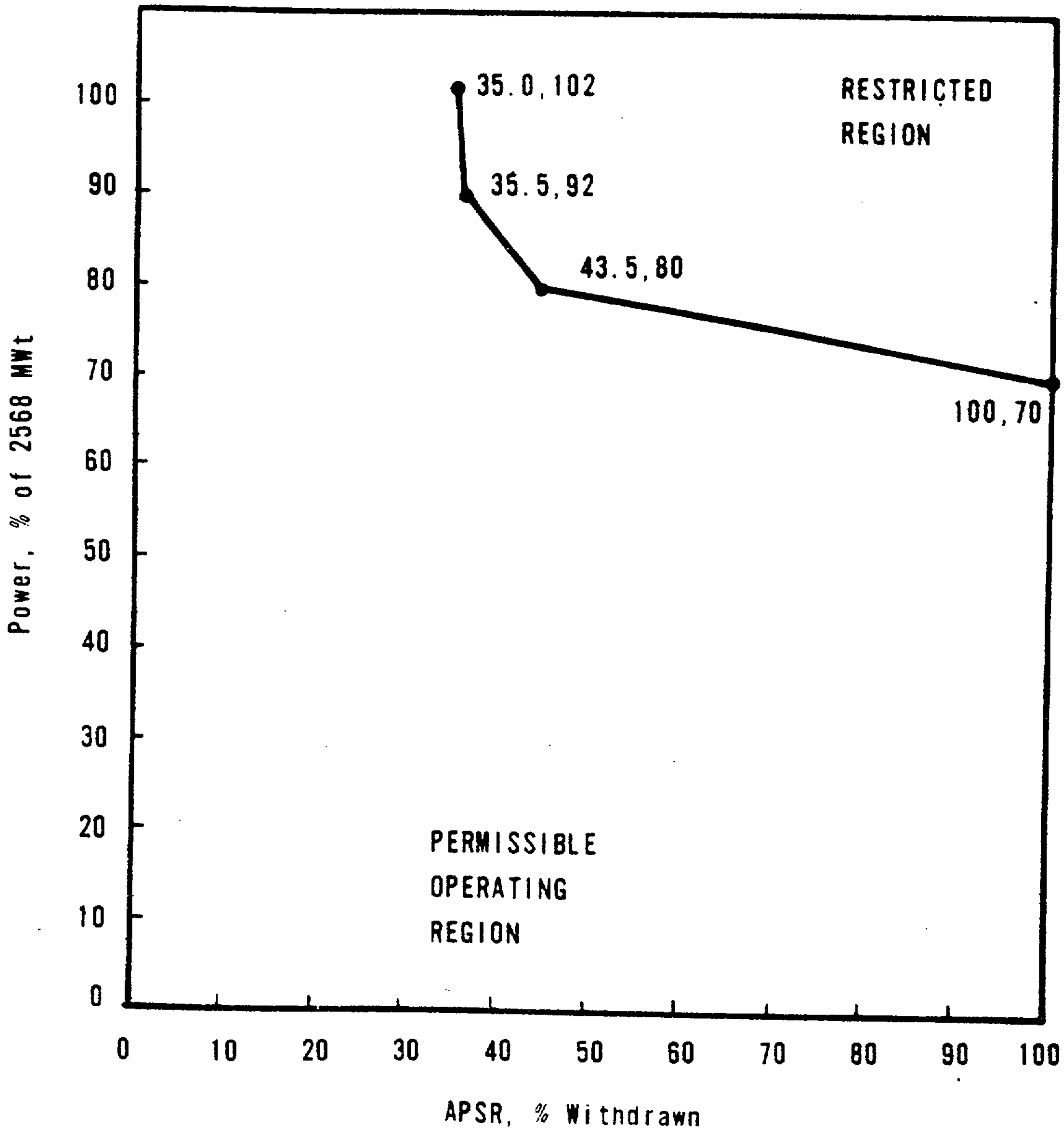
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 100 ± 10 TO 250 ± 10 EFPD
ARKANSAS CYCLE 3

Figure 3.5.2-3B



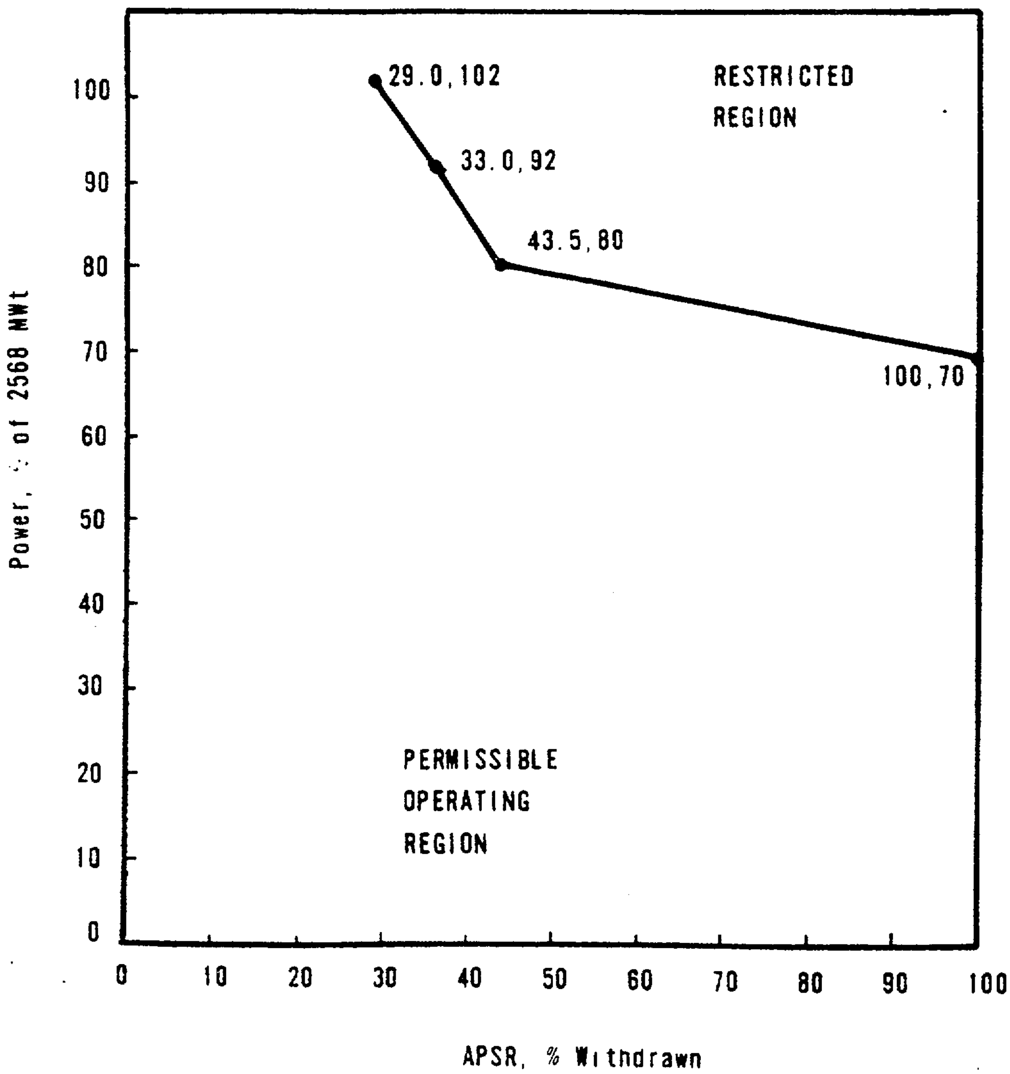
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 250 ± 10 EFPD
ARKANSAS, CYCLE 3

Figure 3.5.2-3C



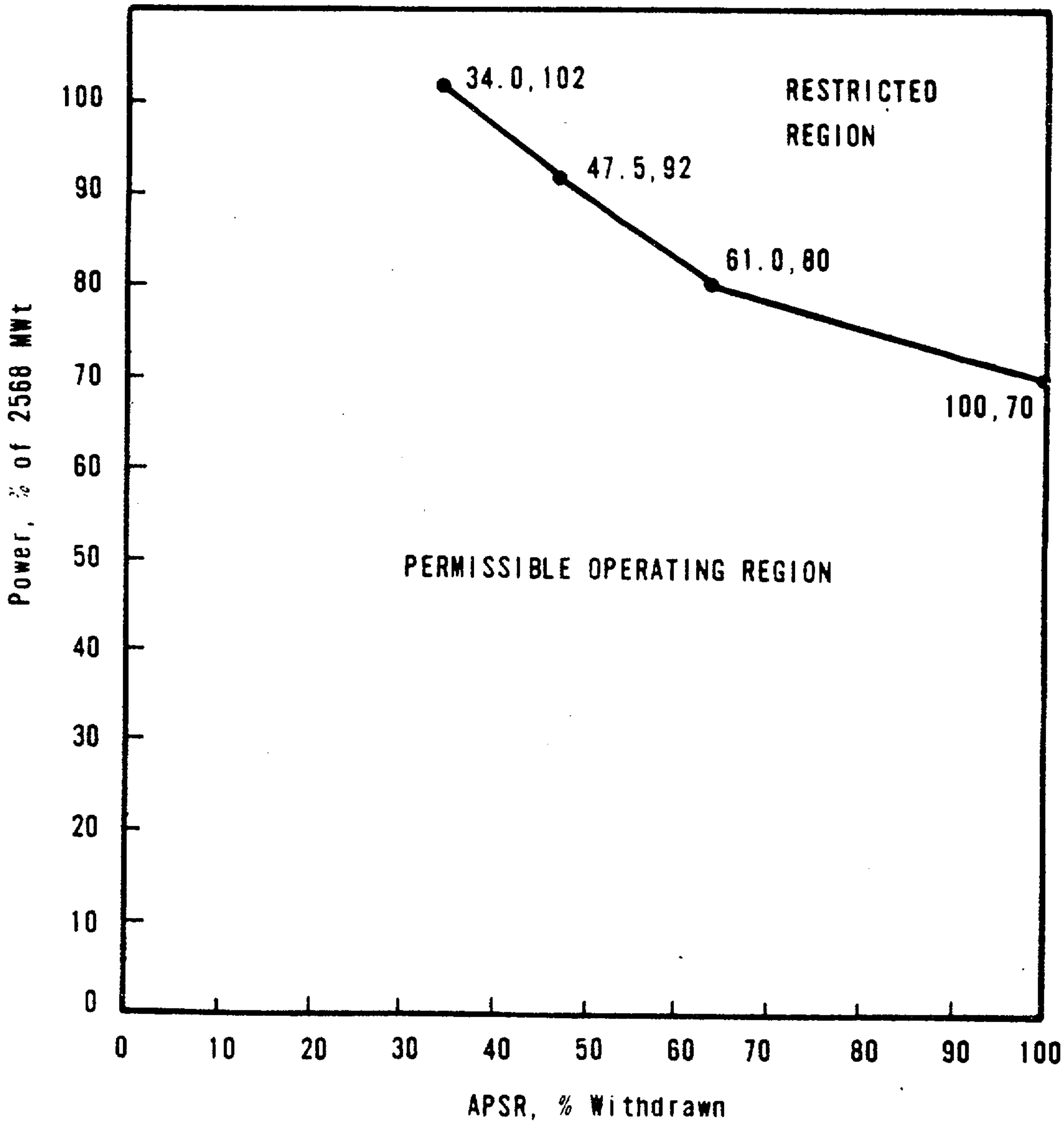
APSR POSITION LIMITS FOR OPERATION FROM
0 TO 100 ± 10 EFPD AND, CYCLE 3

Figure 3.5.2-4A



APSR POSITION LIMITS FOR OPERATION
FROM 100 ± 10 TO 250 ± 10 EFPD AND, CYCLE 3

Figure 3.5.2-4B



APSR POSITION LIMITS FOR OPERATION
AFTER 250 ± 10 EFPD - ANO, CYCLE 3

Figure 3.5.2-4C

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT NO. 1

DOCKET NO. 50-313

1.0 Introduction

By letter dated December 28, 1978 (Reference 1), as supplemented by letters dated January 17, and 30, 1978, and March 3, 1978 (References 2, 3 and 4, respectively), the Arkansas Power and Light Company (AP&L or the licensee) requested an amendment to Facility Operating License No. DPR-51. The amendment would modify the Technical Specifications for Arkansas Nuclear One, Unit No. 1 (ANO-1) for Cycle 3 operation.

2.0 Evaluation

The ANO-1 reactor core consists of 177 fueled assemblies, each containing a 15x15 array of fuel rods. Each 15x15 array contains 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube.

For Cycle 3 operations all Batch 2 assemblies will be discharged from the core. Five once-burned Batch 1 fuel assemblies will be reloaded into the center of the core. Sixty (60) Batch 3 assemblies and 56 Batch 4 assemblies will be shuffled into new locations. Fifty-six (56) Batch 5 fresh assemblies will occupy the core periphery and eight interior locations. Tables 4-1 and 4-2 of Reference 1 contain summaries of fuel design parameters, dimensions and thermal analysis parameters for the fuel batches which will be burned in Cycle 3.

Reactivity control will be supplied by 61 full length Az-In-Cd control rods and soluble boron shim. In addition, eight (8) partial length axial power shaping rods (APSRs) are provided for control of the axial power distributions. Control rod interchanges or burnable poison rods are unnecessary for Cycle 3 operation.

2.1 Fuel Mechanical Design

The Batch 5 fresh fuel uses the Mark B-4 fuel assembly design which was initially used in Batch 4 during Cycle 2. The reload fuel assemblies incorporate minor changes in the spacer grid corner cells which reduce spacer grid interaction during handling. Additionally, dynamic impact testing has shown that the spacer grids have a higher seismic capability and consequently an increased safety margin over the values reported in Reference 5. The dynamic impact testing techniques are described in Reference 6.

Creep collapse time was calculated to be in excess of 30,000 effective full power hours (EFPH) which is longer than the projected three cycle exposure of 25,584 EFPH. The calculation of creep collapse time was performed using the power history of the limiting fuel assembly. As was done in Cycle 2, the CROV computer code was used to predict the collapse time (Reference 7). The licensee stated (Reference 8) that the CROV code demonstrated its ability to conservatively predict cladding collapse.

Additional conservatisms used in the CROV calculations were that no credit was taken for fission gas release; the cladding thickness used in CROV was the lower tolerance limit (LTL) of the as-built measurements; and the lowest as-fabricated pellet densities were assumed to be located in the worst case power region of the core.

The fuel clad strain analysis was performed using a number of conservative assumptions: maximum allowable fuel pellet diameter and density, lowest permitted tolerance for the cladding inner diameter, conservatively high local pellet burnup, and conservatively high heat generation rate. This insures that the 1.0% limit on cladding plastic circumferential strain is not violated.

The Batch 5 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previous successful operations with equivalent fuel, we conclude that the fuel mechanical design for Cycle 3 operations is acceptable and does not decrease the safety margin.

2.2 Fuel Thermal Design

The Batch 5 fuel produces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. As was done in the Cycle 2 reload calculations, the linear heat rate (LHR) capability of ANO-1 was calculated using the TAFY computer code (Reference 9). The nominal LHR for Cycle 3 varies from a value of 5.77 for the Batch 1 fuel to 5.80 for the Batch 5 fuel. The LHR capability varies from 19.40 for Batch 3 to 20.15 for Batches 4 and 5.

The densification power spike model for Cycle 3 used the conservative combination of initial density and enrichment to calculate the spike factor. The power spike model is the same as that presented in Reference 10 with modifications to F_g and F_k. These changes reflect additional data from operating reactors which support a different approach and yield less severe penalties due to power spikes. Based on the analyses presented in Reference 1 and comparison with the allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations (Reference 11), the fuel thermal design for the ANO-1 Cycle 3 core is acceptable and does not decrease the safety margin.

2.3 Fuel Material Design

Cycle 3 fuel for ANO-1 will not have any significant material changes from previous cycles. Batch 4 started the use of a Zircaloy-4 (Zy-4) spacer material rather than Zirconium dioxide (ZrO_2) material. The use of Zy-4 spacer material is continued in Batch 5 assemblies. It was concluded in Reference 12 that the change from ZrO_2 to Zy-4 does not affect the primary coolant system chemistry. Therefore, the fuel material design for ANO-1 Cycle 3 operations is acceptable.

2.4 Nuclear Analysis

Physics parameters were calculated for the ANO-1 Cycle 3 core. There are minor differences between Cycle 3 and the Cycle 2 reference cycle physics parameters since Cycle 3 is not yet an equilibrium cycle. However, the differences in these parameters are minor.

The licensee requested a change in the ANO-1 Technical Specification regarding the correction of the hot zero power (HZP) measured moderator temperature coefficient (MTC) to compare with the 95% power Technical Specification limit (Reference 2). The proposed change would allow the use of cycle dependent parameters measured in the physics startup testing to project or extrapolate the 95% power value. The current Technical Specification requires a Technical Specification change each cycle because the cycle dependent corrections to the MTC at HZP are explicitly stated in the Technical Specification. We find that this approach will eliminate an unnecessary administrative step and is therefore acceptable.

The licensee also proposed a change in the plant Technical Specifications increasing the allowable quadrant tilt from 3.4% to 4.92%. The additional peaking allowed is a result of the statistical combination of the nuclear uncertainty factor, the hot channel factor, and the rod bow peaking penalty. We find that this Technical Specification is acceptable and does not decrease the safety margin.

The only significant proposed operational procedure change is the proposed Technical Specification change of the axial power shaping rod (APSR) position limits. The APSR position limits would provide added control of power peaking to insure that peak power limits for Loss of Coolant Accident (LOCA) conditions would not be violated.

We find that, based on the AP&L's nuclear analysis techniques and their commitment to perform acceptable physics startup testing, the ANO-1 nuclear analysis is acceptable. We also find the proposed Technical Specifications of APSR position limits and the usual regulating control rod and imbalance limits, which assure that the loss of coolant accident (LOCA) LHGR limits are not exceeded, are acceptable.

2.5 Thermal-Hydraulic Analyses

The thermal-hydraulic analyses for ANO-1 Cycle 3 were performed using previously approved methods and models per the ANO-1 Final Safety Analysis Report (FSAR). The only change in the thermal-hydraulic analysis for Cycle 3 is the removal of the densification power spike from Departure from Nucleate Boiling Ratio (DNBR) calculation, resulting in an increase in the minimum calculated steady-state DNBR from 1.84 for Cycle 2 to 1.90 for Cycle 3.

The maximum fuel rod bow, calculated using the interim NRC fuel rod bow model, is 11.2% and occurs at the end of Cycle 3. The licensee provides the requisite margin by the flux/flow trip setpoint of 1.060 and the variable low-pressure trip. We find that the thermal-hydraulic analysis for ANO-1 Cycle 3 operations is acceptable.

2.6 Accident and Transient Analysis

The generic Babcock and Wilcox (B&W) Loss of Coolant Accident (LOCA) analysis is contained in BAW-10103 (Reference 13). The analysis in BAW-10103 is generic since the limiting values of key parameters for all plants in the category I (177 FA-lowered loop) Nuclear Steam Supply System (NSSS) are used. The combination of average fuel temperature and pin pressure data, for the lifetime of the fuel, as used in the BAW-10103 LOCA limits analysis is conservative compared to those used in the Cycle 3 reload analysis. In Reference 14, B&W submitted a change to the BAW-10103 LOCA analysis because of an incorrect pressure drop assumed for the inlet nozzle region. The correction incorporates a revised reactor coolant system pressure distribution. The result is that the peak clad temperature in the revised calculation is 2060°F for the unruptured node and 1826°F for the ruptured node. This is a reduction of 86°F and 240°F, respectively, relative to the BAW-10103 results. Therefore, tne analysis presented in BAW-10103 is valid for the reload cycle.

Relative to plant transients, the Cycle 3 evaluation is bounded by the FSAR, the fuel densification report (Reference 15) and previous cycle analyses.

We conclude that the LOCA analyses performed for ANO-1 meet 10 CFR 50.46 criteria and insure that the plant can be operated without undue risk to the public safety.

2.7 Physics Startup Tests

The proposed physics startup program is discussed in Reference 4. The licensee has committed to conduct physics startup tests to insure that the significant aspects of the ANO-1 Cycle 3 core would be within the acceptable criteria. These include control rod functional tests, scram times, control rod worth tests, temperature reactivity coefficient tests, and power distribution tests. The licensee has also committed to provide a report on these tests within 45 days after completion of the test program. The program has been reviewed and found acceptable.

3.0 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.

4.0 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: March 17, 1978

References

1. Letter, Rueter (AP&L) to Davis (NRC), Docket No. 50-313, December 28, 1977, forwarding the Arkansas Nuclear One, Unit No. 1, Cycle 3 Reload Report, BAW-1471.

2. Letter, Cavanaugh (AP&L) to Davis (NRC), Docket No. 50-313, January 17, 1978.

3. Letter, Cavanaugh (AP&L) to Davis (NRC), Docket No. 50-313, January 30, 1978.

4. Letter, Williams (AP&L) to Reid (NRC), Docket No. 50-313, March 3, 1978.

5. BAW-10035, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation," June, 1970.

6. BAW-10133, "Mark-C Fuel Assembly Topical Report on LOCA - Seismic Analyses," October 1977.

7. BAW-1433, "Arkansas Nuclear One, Unit No. 1 - Cycle 2 Reload Report," November, 1976.

8. BAW-10084P-A, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse," January, 1975.

9. BAW-1004, "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," May, 1972.

10. BAW-10055, Rev. 1, "Fuel Densification Report, June, 1973.

11. Standard Review Plan, Section 4.4, pp. 4.4-2 and 4.4-3.

12. Letter, Davis (NRC) to Phillips (AP&L), Safety Evaluation for Amendment No. 21 to Facility Operating License No. DPR-51, March 31, 1977.

13. BAW-10103, Rev. 1, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," September, 1975.

14. Letter, Taylor (B&W) to Baer (NRC), July 8, 1977.

15. BAW-1391, "Arkansas Nuclear One, Unit No. 1, Fuel Densification Report," June, 1973.

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. 31 to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 1 (ANO-1) (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment authorizes operation and modifies the Technical Specifications for Cycle 3.

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 application for amendment dated December 28, 1977, as supplemented January 17, and 30, 1978, and March 3, 1978, (2) Amendment No. 31 to License No. DPR-51, 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 Arkansas Polytechnic College, Russellville, Arkansas 72801. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 17th day of March 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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