DMB016

December 20, 1984

Docket No. 50-313

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Mr. John M. Griffin
Senior Vice President
of Energy Supply
Arkansas Power and Light Company
P. O. Box 551

Dear Mr. Griffin:

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 26, 1984, supplemented by letter dated October 31, 1984.

The amendment revises the TSs to support the operation of ANO-1 at full rated power during the Cycle 7.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

"ORIGINAL SIGNED BY:"

Guy S. Vissing, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 92 to DPR-51
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	GVissing;cf	JFStolz		GLainas
12/11/84	12 / /84	12/ /84	12/ /84	12/10/84

McGuire Noted objection

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PDR ADOCK 05000313
P PDR

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated September 26, 1984, as supplemented October 31, 1984, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 92 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 20, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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
iv	iv
v	v
vi	vi
8	8
9	9
9b	9b
9c	9c
12	12
14b	14b
15	15
35a	35a
47	47
--	47a
48	48
48a1	48a1
48b thru 48b3	48b
48c thru 48c7	48c
48d thru 48d3	48d
48e	48e
48f	48f
48g	48g
48h	48h
48i	--

LIST OF FIGURES

Amendment No. ~~52~~, ~~80~~, ~~86~~, 92

DNBR of 1.3 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure was actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR greater than 1.3 is predicted. The curve is the most restrictive combination of 3 and 4 pump curves, and is based upon the maximum possible thermal power at 106.5% design flow per applicable pump status. This curve is based on the following nuclear power peaking factors (2) with potential fuel densification effects:

$$F_q^N = 2.83; \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.65.$$

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

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.83$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than 1.3 DNBR.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 20.5 kW/ft.

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

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

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which a minimum DNBR greater than 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. The local quality at the point of minimum DNBR is less than 22 percent (1).

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, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

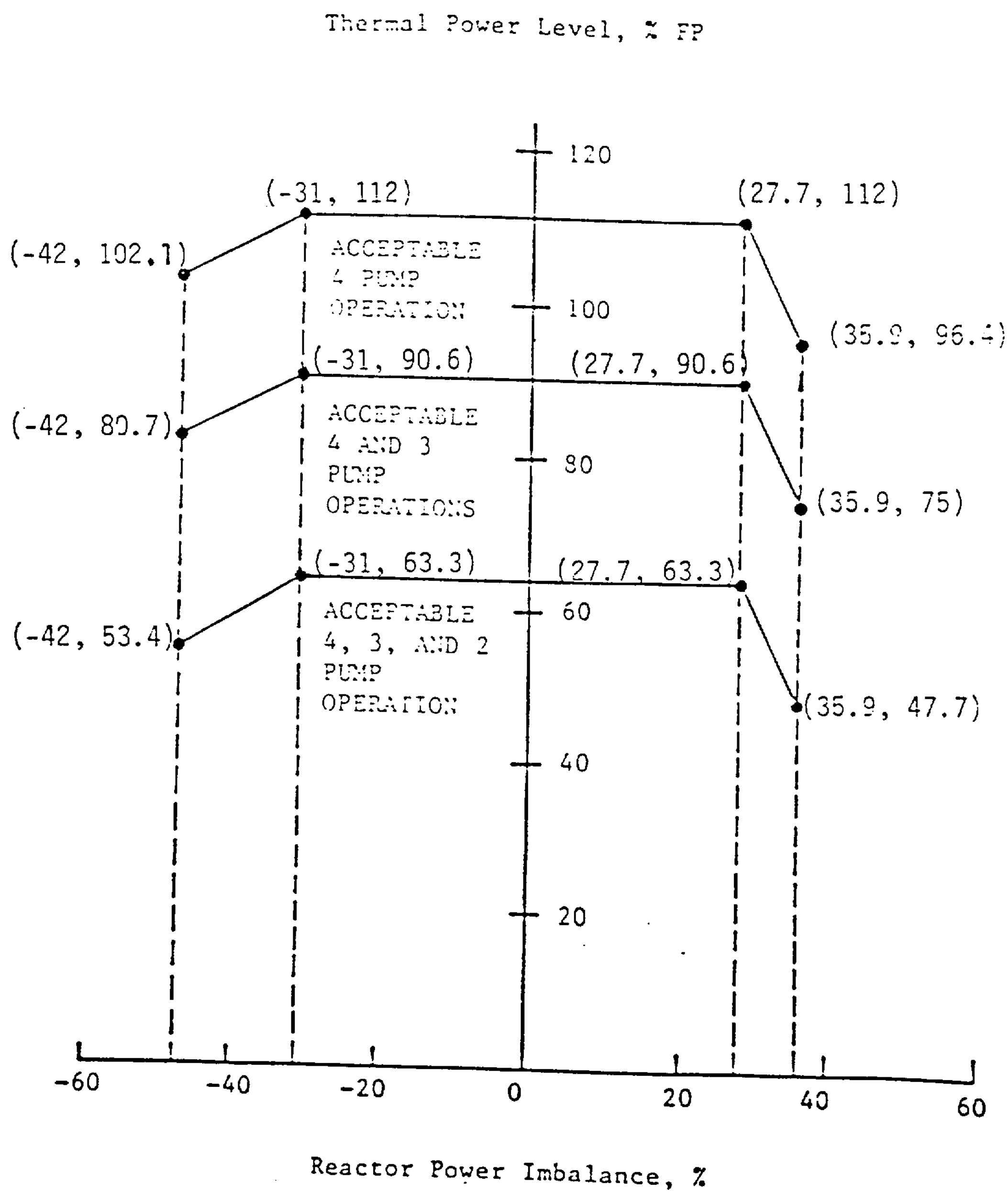
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. Curves 1 and 2 of Figure 2.1-3 are the most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

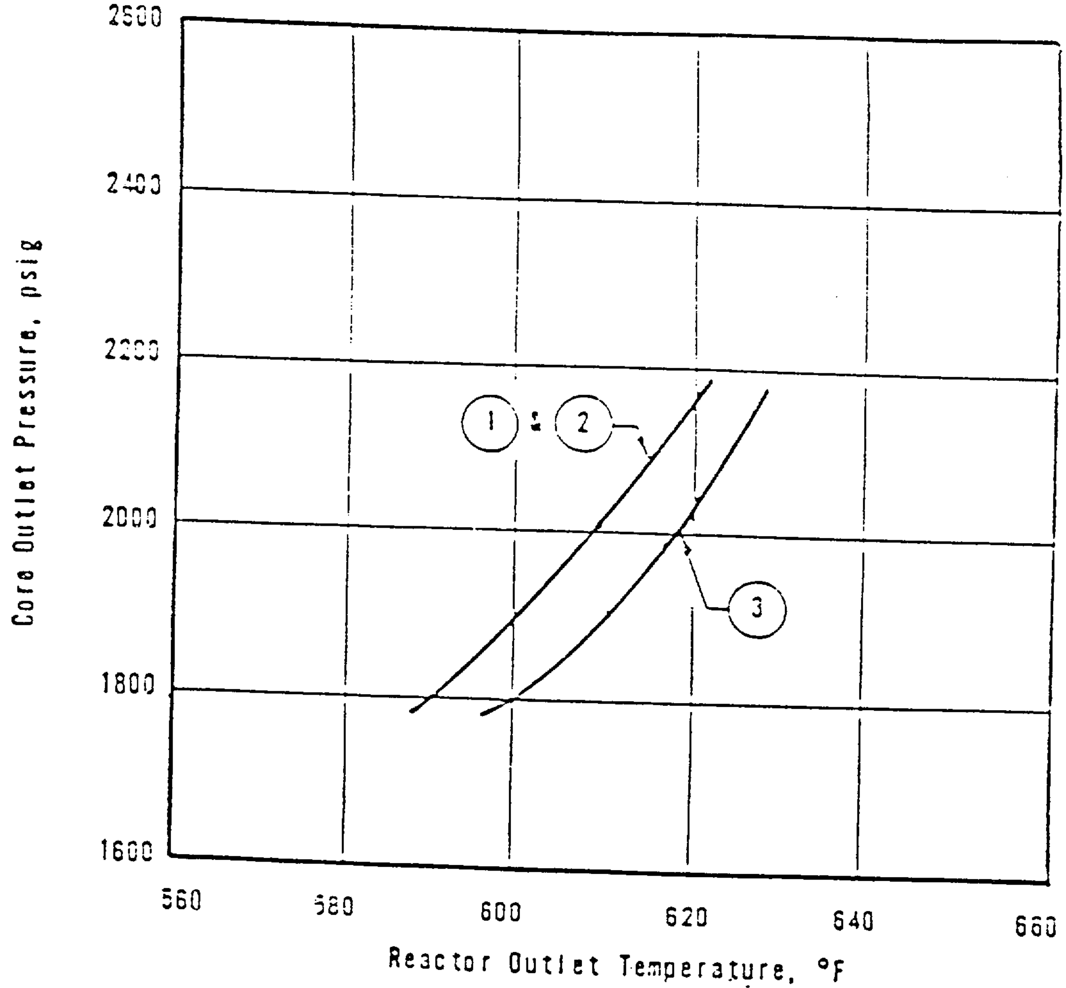
(2) FSAR, Section 3.2.3.1.1.c

Core Protection Safety Limits - ANO-1
Figure 2.1-2



Amendment No. ~~5~~, ~~21~~, ~~31~~, ~~43~~, ~~52~~. ~~67~~, ~~71~~,
92

Core Protection Safety Limits - ANO-1
Figure 2.1-3



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%)*	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	90.6%	THREE PUMPS (DNBR LIMIT)
3	184,441 (49.2%)	64.1%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

*106.5% OF DESIGN FLOW

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 107 percent and reactor flow rate is 100 percent or flow rate is 93.5 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80 percent and reactor flow rate is 74.7 percent or flow rate is 70 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 percent and reactor flow is 49.2 percent or flow rate is 45.8 percent and the power level is 49 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 with reactor power-to-reactor power imbalance boundaries by 1.07 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

Protective System Maximum Allowable Setpoints
ANO-1, Figure 2.3-2

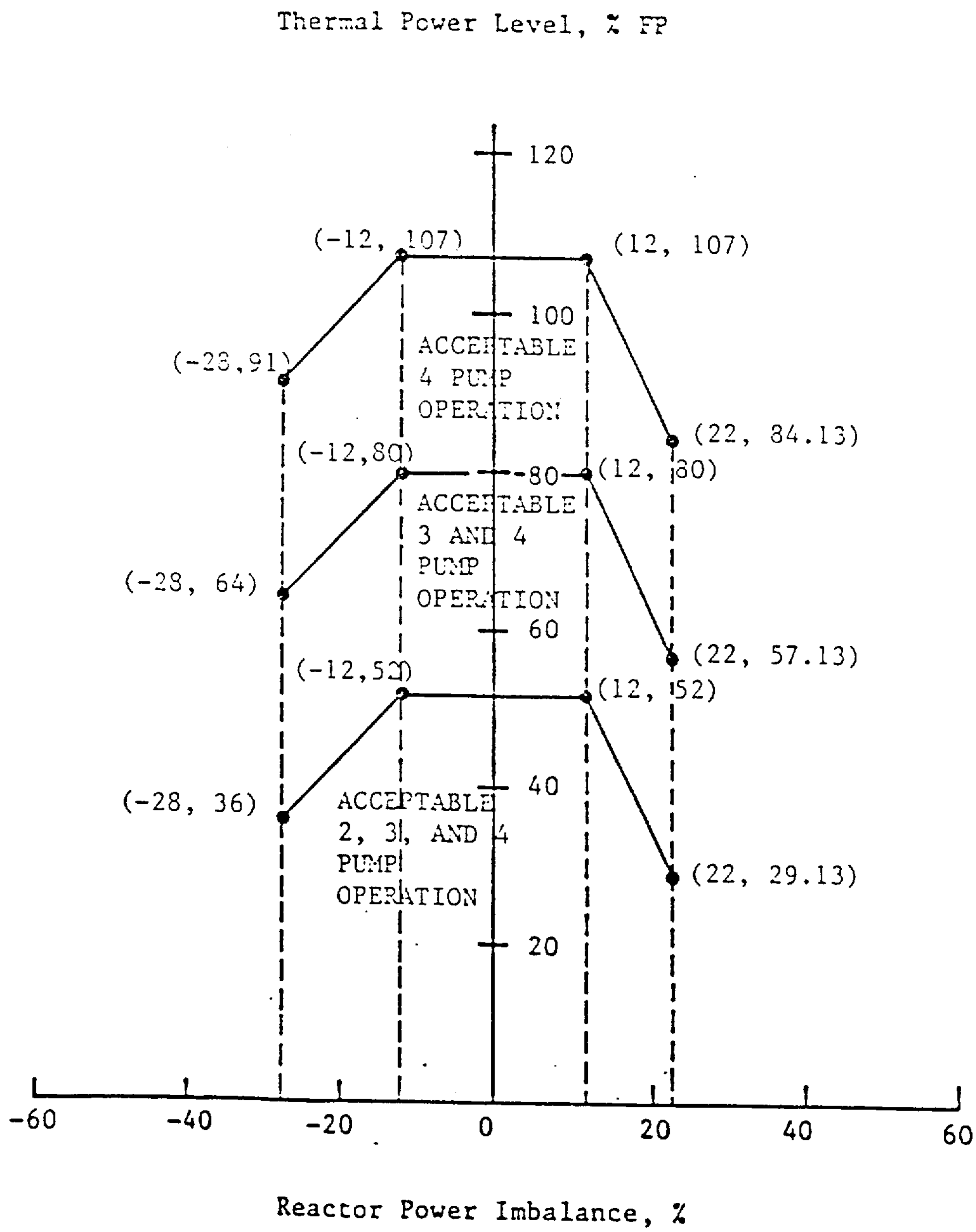


Table 2.3-1
Reactor Protection System Trip Setting Limits

	Four RC Pumps Operating (Nominal Operating Power - 100%)	Three RC Pumps Operating (Nominal Operating Power - 75%)	One RC Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
Nuclear Power based on pump monitors[c], % of rated, max	NA	NA	55	Bypassed
High RC system pressure, psig, max	2300	2300	2300	1720[a]
Low RC system pressure, psig,min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	11.75 T_{out}-5103[d]	11.75 T_{out}-5103[d]	11.75 T_{out}-5103[d]	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a]Automatically set when other segments of the RPS (as specified) are bypassed.

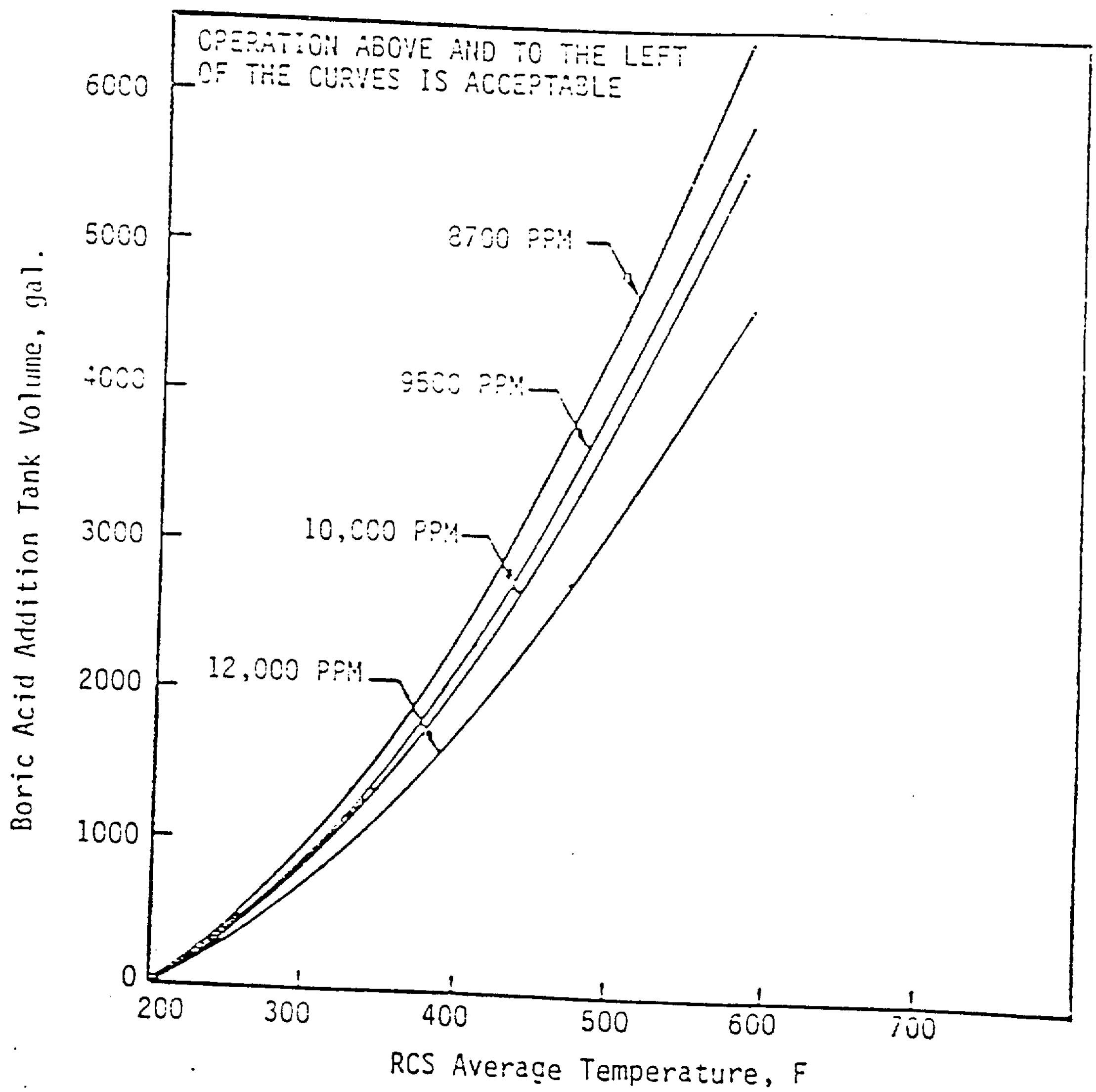
[b]Reactor coolant system flow.

[c]The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d]T_{out} is given in degrees Fahrenheit (F).

Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature - ANO-1

Figure 3.2-1



TEMP. F	REQUIRED VOLUME, GAL.			
	8700PPM	9500PPM	10,000PPM	12,000PPM
579	6436	5863	5554	4589
532	5289	4817	4564	3769
500	4488	4087	3872	3199
400	2434	2218	2101	1737
300	986	898	851	705
200	0	0	0	0

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 3.1% power shall be reduced immediately to below the power level cutoff (92% FP). Moreover, the power level cutoff value shall be reduced 2% for each 1% tilt in excess of 3.1%. 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 3.1%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 3.1% 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 3.1%.

   c. The operational imbalance limits shall be reduced 2% in power for each 1% tilt in excess of 3.1%.

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 3.5.2-1, 3.5.2-2A and 3.5.2-2B for 4, 3 and 2 pump operation respectively; and (b) the axial power shaping control rod withdrawal limits are specified on Figures 3.5.2-4A and 3.5.2-4B. 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 4 hours.

4. Except for physics tests, power shall not be increased above the power level cut-off of 92% of the maximum allowable power level unless one of the following conditions is satisfied:

   a. Xenon reactivity is within 10% of the equilibrium value for operation at the maximum allowable power level and asymptotically approaching stability.

   b. Except for xenon free startup, when 3.5.2.5.4a applies, the reactor has operated within a range of 87 to 92% of the maximum allowable power for a period exceeding 2 hours.

3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within 4 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

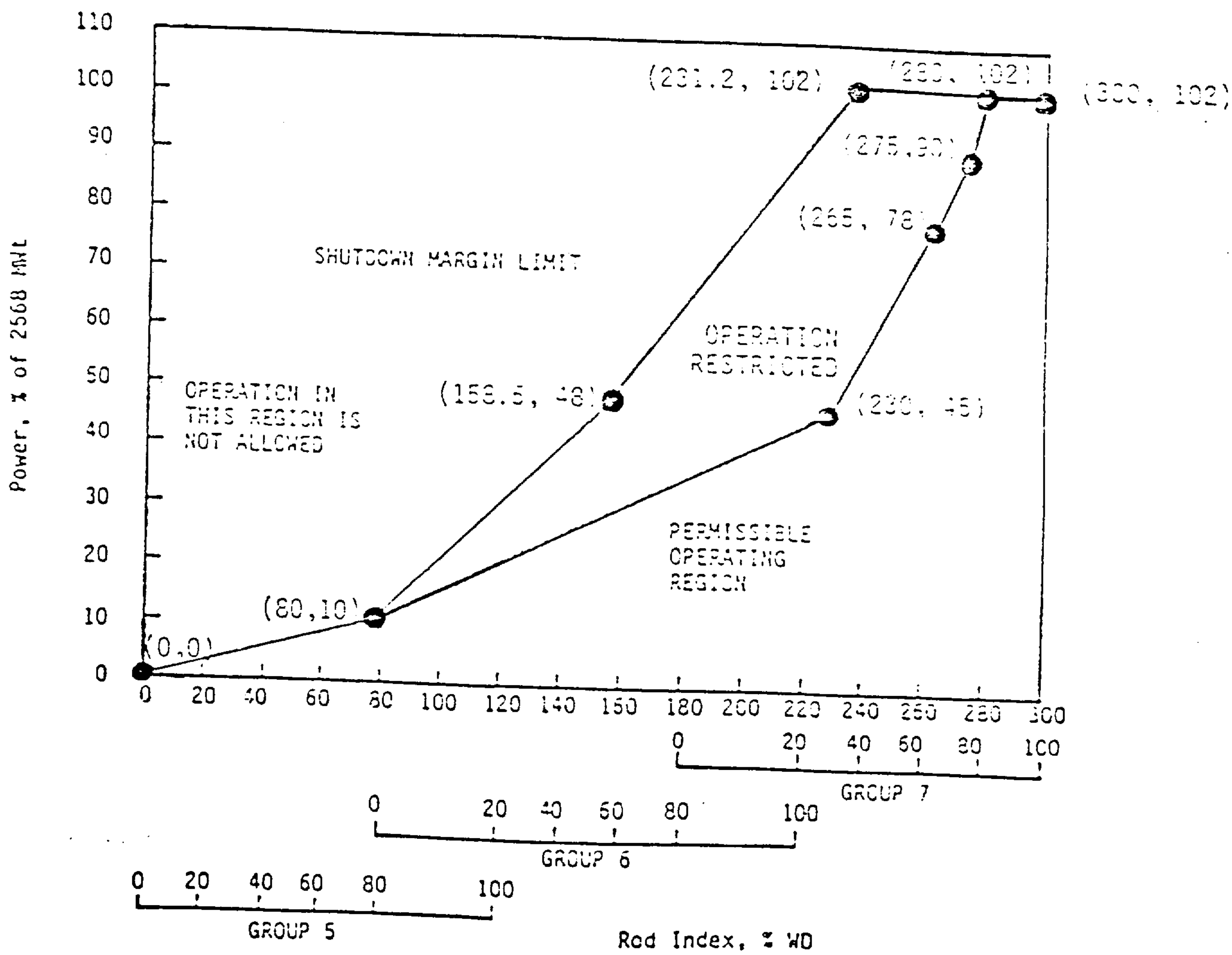
The power-imbalance envelope defined in Figure 3.5.2-3 is based on (1) LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria and (2) the Protective System Maximum Allowable Setpoints (Figure 2.3-2). Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundaries. 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

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.

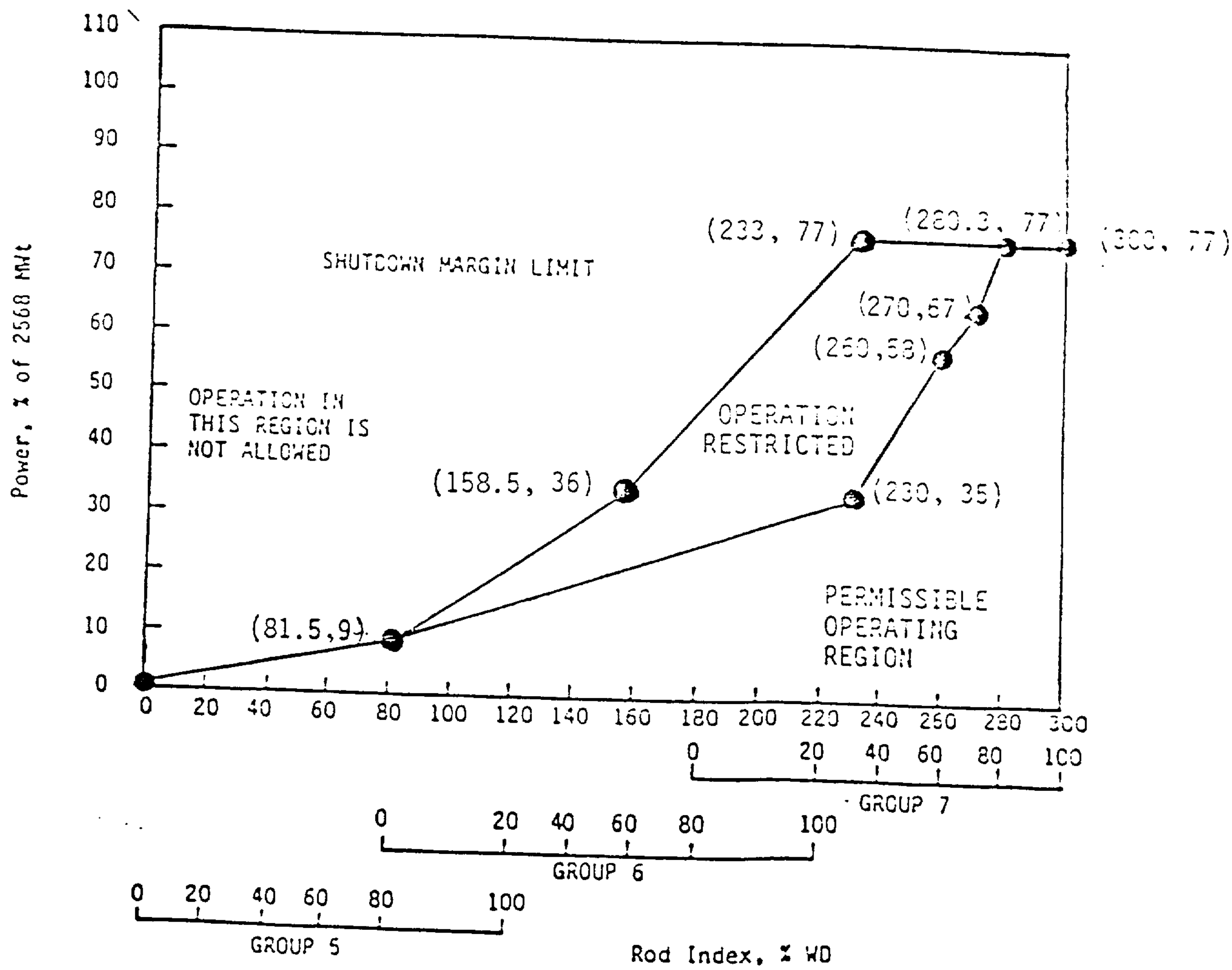
The quadrant tilt and axial imbalance limits in Specifications 3.5.2.4 and 3.5.2.6, respectively, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

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

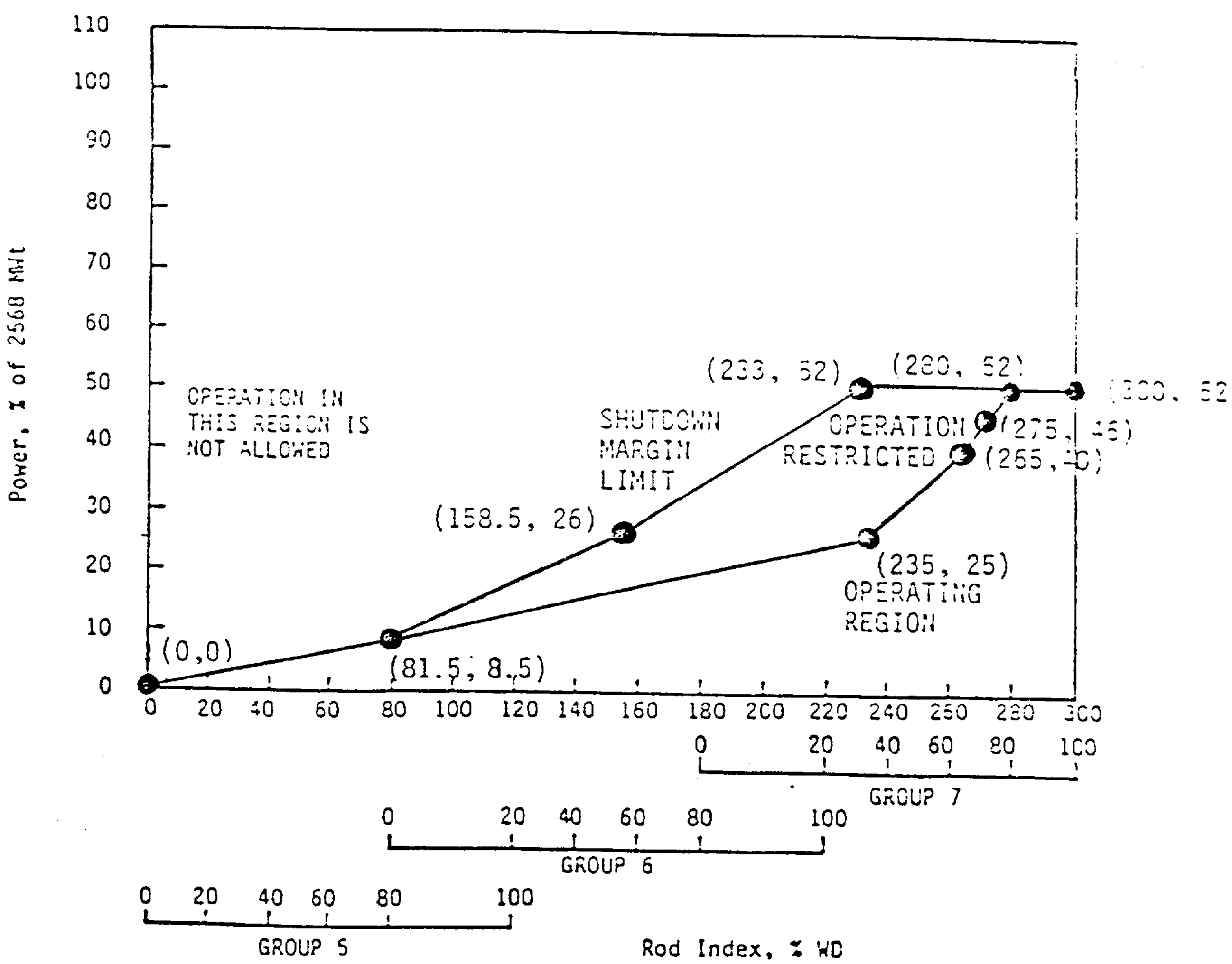
Rod Position Limits for 4-Pump Operation
From 0 EFPD TO EOC ---- ANO-1
Figure 3.5.2-1



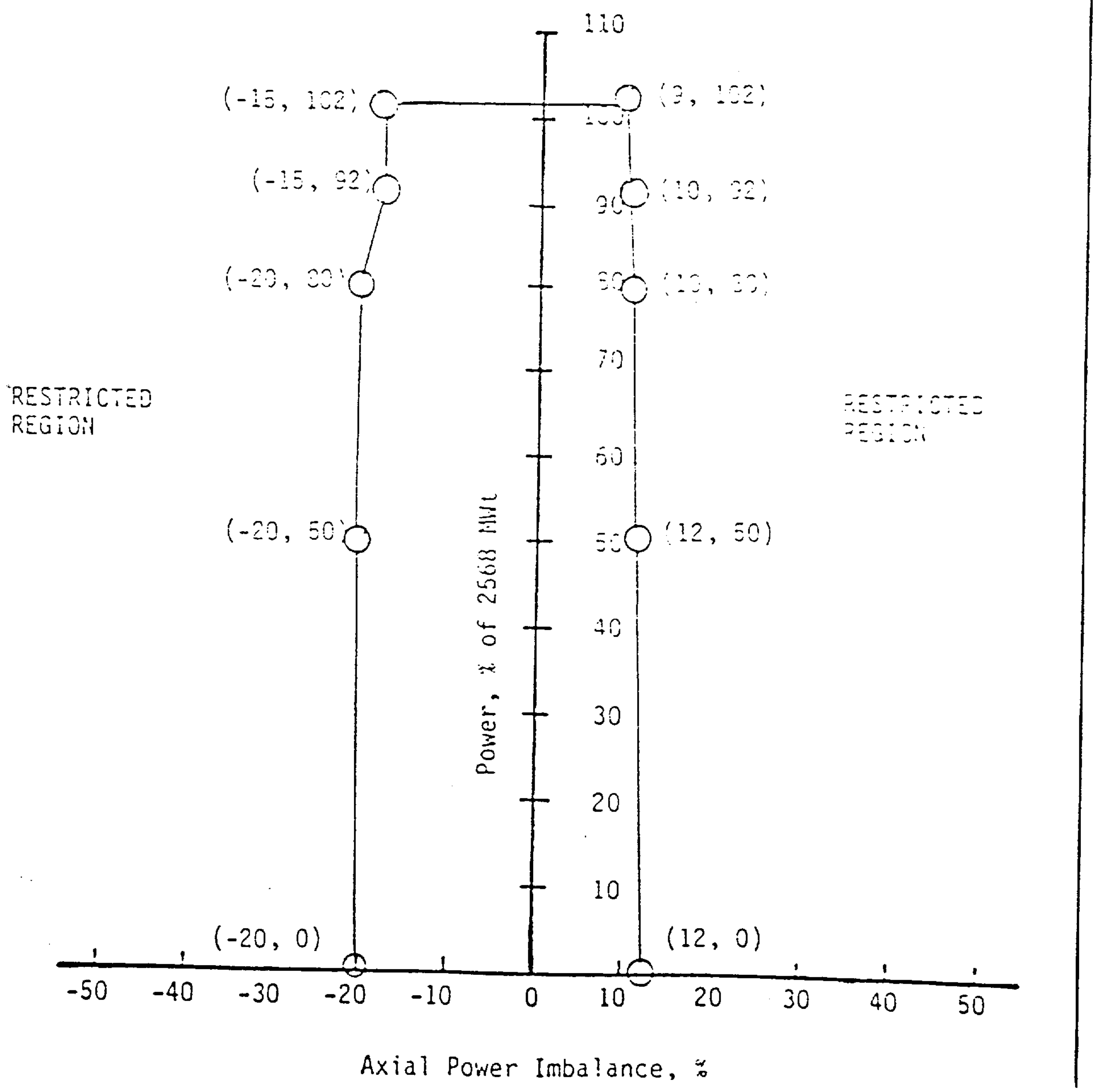
Rod Position Limits for 3-Pump Operation
From 0 EFPD TO EOC ---- ANO-1
Figure 3.5.2-2A



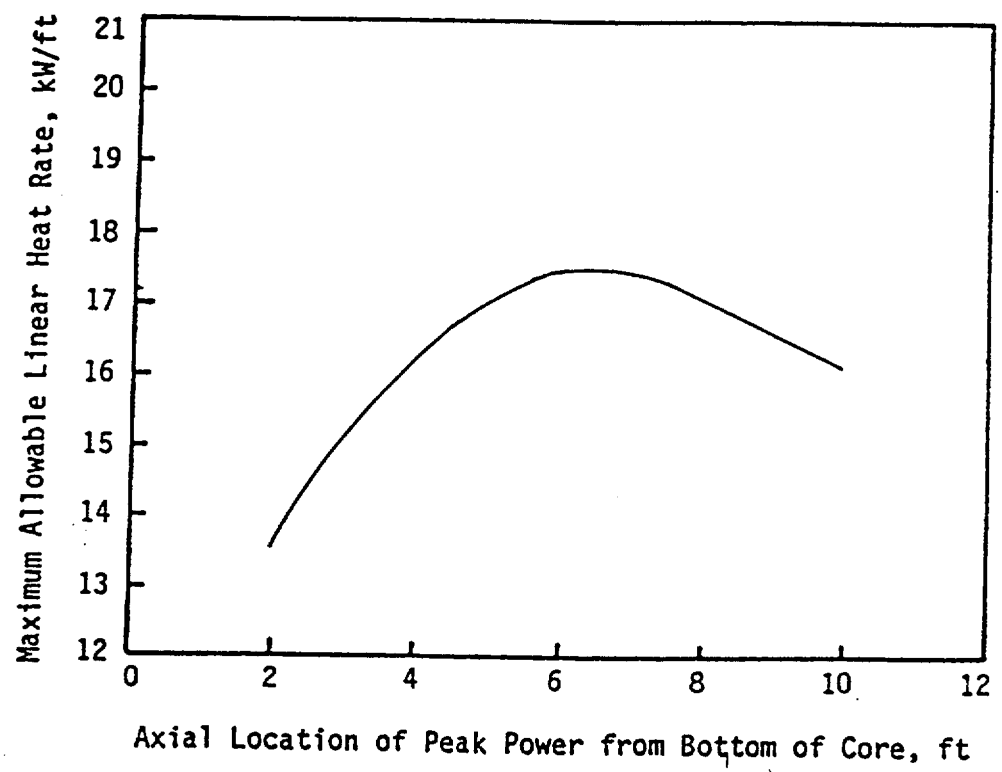
ROD POSITION LIMITS FOR 2 - PUMP OPERATION
FROM 0 EFPD TO EOC ---- ANO-1
Figure 3.5.2-2B



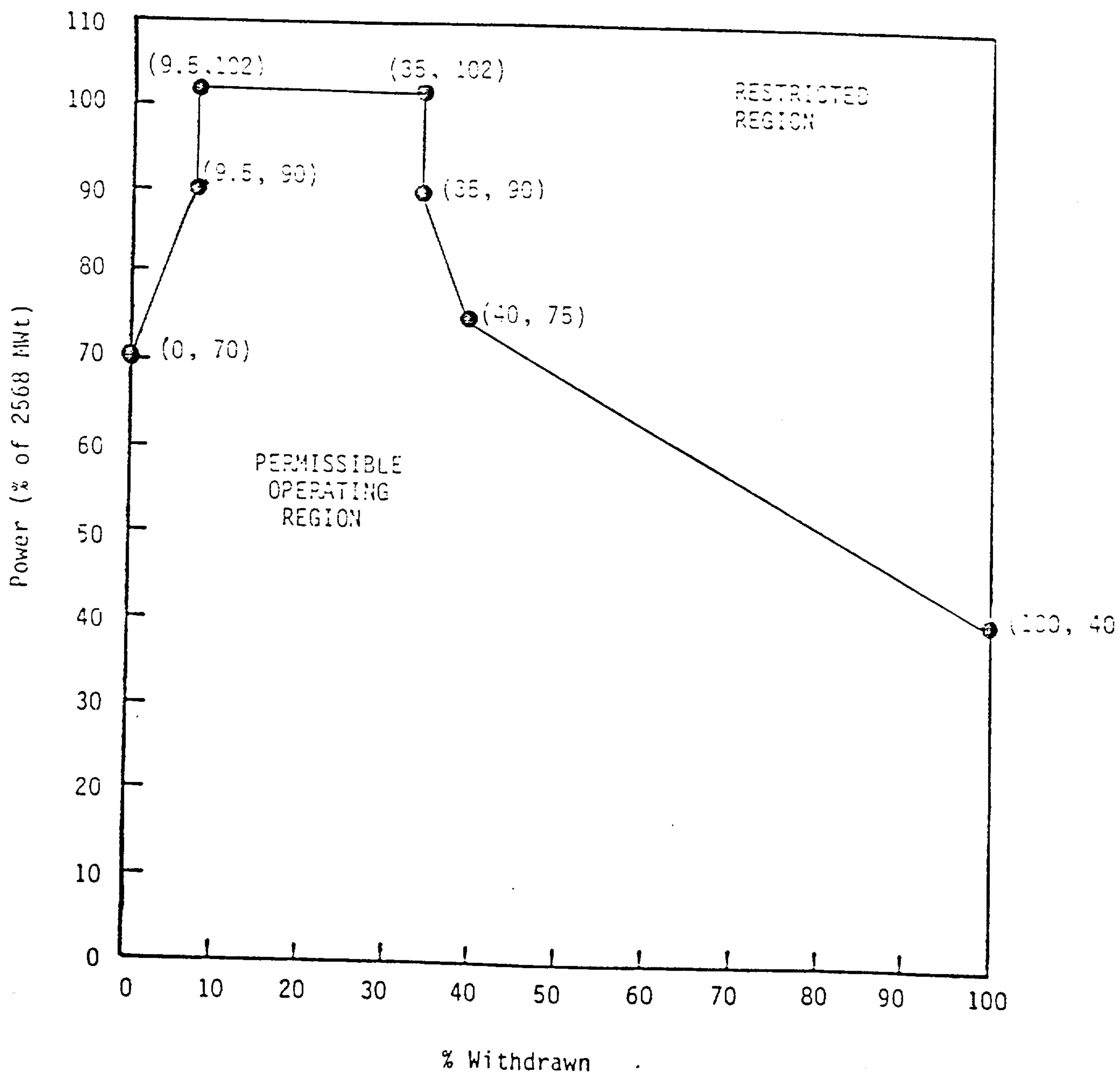
Operational Power Imbalance Envelope for Operation
From 0 EFPD TO EOC EFPD ---- ANO-1
Figure 3.5.2-3



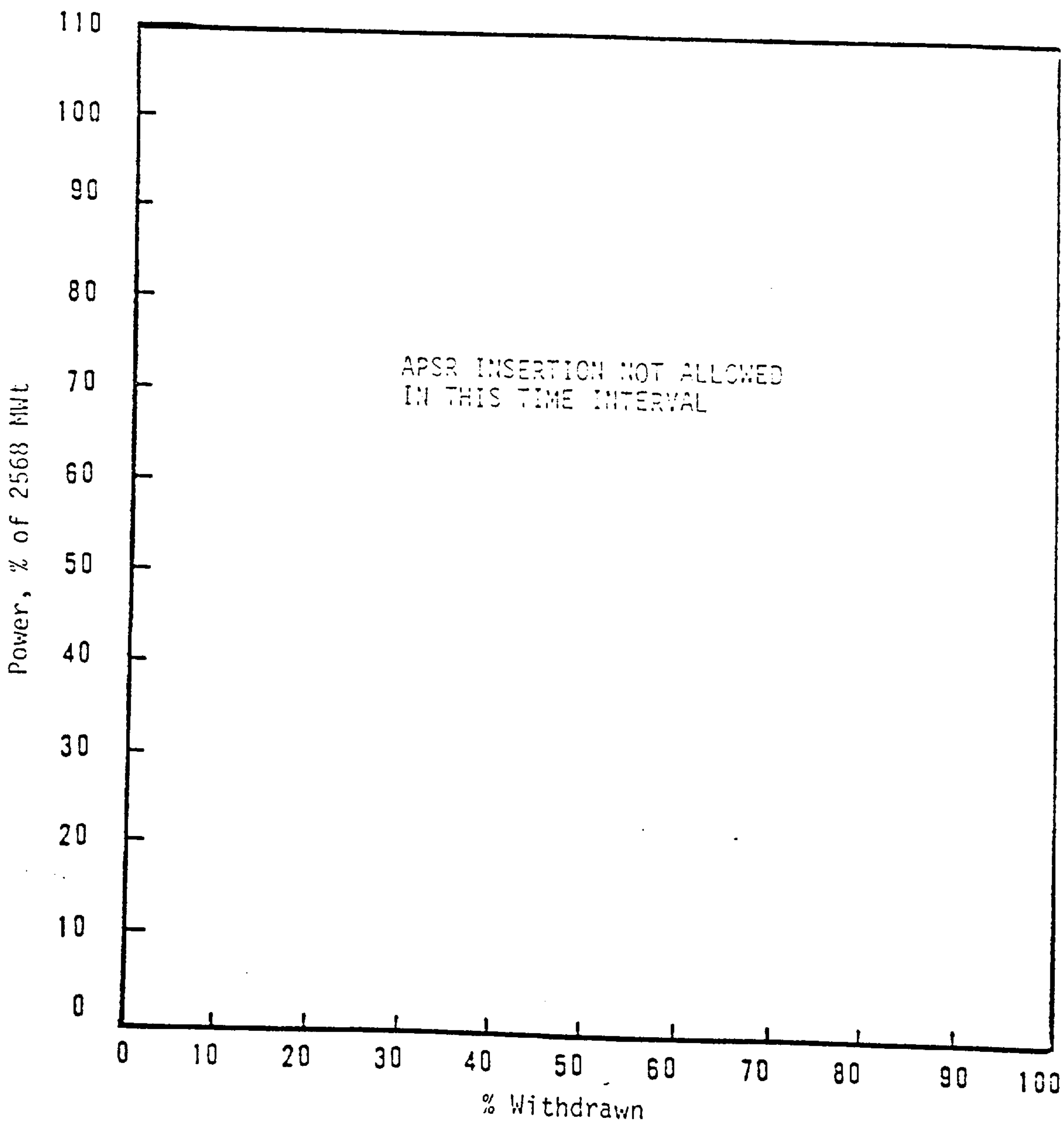
LOCA Limited Maximum Allowable Linear Heat Rate
Figure 3.5.2-4



APSR Position Limits for Operation From
0 EFPD to APSR Withdrawal ---- ANO-1
Figure 3.5.2-4A



APSR Position Limits for Operation After
APSR Withdrawal ---- ANO-1
Figure 3.5.2-4B



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated September 26, 1984 (Ref. 1), supplemented by letter dated October 31, 1984 (Ref. 2), Arkansas Power and Light Company (AP&L or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed changes would modify the TSs to permit operation for the seventh cycle. The safety analyses performed and the resulting modifications to the plant TSs are described in the Cycle 7 Reload Report (Ref. 3). Additional supporting information was provided by letter dated December 6, 1984 (Ref. 4).

The safety analysis for the previous sixth cycle of operation at ANO-1 is being used by the licensee as a reference for the proposed seventh cycle of operation. Where conditions are identical or limiting in the sixth cycle analysis, our previous evaluation (Ref. 5) of that cycle continues to apply.

1.1 Description of the Cycle 7 Core

The ANO-1 core consists of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 7 will operate in a feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies composed of silver-indium-cadmium alloy clad in stainless steel. In addition to the full length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level is 2568 MWt.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

2.1 Fuel Assembly Mechanical Design

The 72 Babcock and Wilcox (B&W) Mark B-4 15x15 fuel assemblies to be loaded as Batch 9 fuel for Cycle 7 operation are mechanically interchangeable with Batch 8 fuel assemblies previously loaded at ANO-1. The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for ANO-1 or were analyzed specifically for Cycle 7 using methods and limits previously reviewed and approved by the NRC.

2.2 Fuel Rod Design

All batches in the ANO-1 Cycle 7 core utilize the same B&W Mark B-4 fuel design, and the Batch 9 fuel parameters are virtually identical to the previously loaded Batch 8 fuel except for enrichment, which has been increased from 3.21 to 3.30 wt/% U.

There has been a change in the pellet design for Batch 9 fuel rods. The fuel pellet length/diameter ratio has been decreased from 1.63 to 1.18. The licensee claims this change will not affect fuel performance, and at high burnups it is expected to decrease local cladding strains.

Four fuel assemblies in the highest burnup Batch 7B are extended burnup lead test assemblies (LTAs), which are scheduled for a third cycle of irradiation in Cycle 7. These assemblies, which are described in Reference 7, are similar in design to the standard Mark B-4 fuel assemblies except for changes to the fuel rod and fuel assembly structure to extend their burnup capability. We previously concluded (Ref. 6) that the irradiation of the four LTAs in ANO-1 was acceptable.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 8) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. Our current criterion (SRP 4.2, Section II.A.1(f)) is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified.

AP&L has stated that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 7. This analysis is based on the use of the approved B&W TACO2 code (Ref. 9). We conclude that the rod internal pressure limits have been adequately considered for Cycle 7 operation.

2.3 Fuel Thermal Design

There are no major changes between the thermal design of the new Batch 9 fuel and previous batches reinserted in the Cycle 7 core. The licensee presented results of the thermal design evaluation of the Cycle 7 core. These are based upon analyses performed with the TACO2 code. The Cycle 7 core protection limits were based on a linear heat rate to centerline fuel melt of 20.5 kW/ft. The results of the thermal design evaluation show no difference between Batch 9 fuel and the Batch 7 and 8 fuel already approved for use in the core. We have reviewed the fuel design parameters for normal operation and find them acceptable.

2.3.1 Loss of Coolant Accident (LOCA) Initial Conditions

In addition to the steady-state conditions, the average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA analysis (Section 7.2 of the reload submittal) are also calculated with the TACO2 code (Ref. 9). The reload report stated that the fuel temperature and pin pressure data used in the generic LOCA analysis (Ref. 10) are conservative compared to those calculated for Cycle 7 at ANO-1.

2.4 Conclusion

We have reviewed the fuel system design and analysis for ANO-1 Cycle 7 operation and find the application acceptable.

3.0 EVALUATION OF THE NUCLEAR DESIGN

To support Cycle 7 operation of ANO-1, the licensee has provided analyses using analytical techniques and design bases established in reports that have been approved by the NRC staff. The licensee has provided a comparison of the core physics parameters for Cycles 6 and 7 as calculated with these techniques. There are slight differences in these parameters. This is to be expected since the core has not yet reached an equilibrium cycle. All of the accidents analyzed in the Final Safety Analysis Report (FSAR) were reviewed for Cycle 7 operation. We note that the Cycle 7 characteristics were conservative compared to those analyzed for previous cycles and no new accident analyses were performed.

We find the predicted characteristics acceptable because they use approved techniques, the validity of which has been reinforced through a number of cycles of predictions for this and other reactors. As a result of our review of the characteristics compared to previous cycles, we agree with the licensee's conclusions regarding Cycle 7 accident analysis.

The licensee's calculations took into account ejected rod worths and their adherence to accident analysis criteria in development of rod position limits for Cycle 7 Technical Specifications. The licensee has provided predictions of rod worths and control requirements demonstrating adequate shutdown margin throughout the cycle. Startup tests of control rod worth will provide a verification of the accuracy of these predictions.

Withdrawal of the APSRs is planned near the end of Cycle 7, as in Cycle 6. This results in a calculated stability index of -0.052 per hour, which ensures the axial stability of the core.

Core design changes for Cycle 7 are the transition to a very low leakage design and the use of "short-stack" lumped burnable poison rods. For this transition cycle, 12 twice-burned assemblies are located on the core periphery to reduce fluence levels on the reactor vessel. The analytic techniques used by B&W to predict physics parameters adequately account for the effects of such changes in the process of performing a reload analysis.

The lumped burnable poison used in Cycle 7 has a 9 inch shorter stack than that used in the standard Mark B design, i.e., 117 versus 126 inches of Al_2O_3- B_4C. The top 9 inches of the poison stack are replaced by a Zircaloy tubular spacer. This design produces a lower axial peak at the beginning of the cycle and increases operational flexibility. We reviewed the effects of this design and its impact on the calculation of the Technical Specification changes at a meeting with the licensee and B&W on November 27, 1984. The calculations conservatively account for such changes and, therefore, the short stack burnup poison design is acceptable.

4.0 EVALUATION OF THE THERMAL-HYDRAULIC DESIGN

The objective of the thermal-hydraulic review is to confirm that the design of the reload core has been accomplished using acceptable methods and that acceptable safety margin is available from conditions which would lead to fuel damage during normal operation and anticipated transients.

The thermal-hydraulic analysis for Cycle 7 was performed with a 1.71 design radial - local ($F\Delta_H^N$) power peak with a 1.65 symmetric chopped cosine design axial flux shape. This is in comparison with the 1.71 radial - local and 1.5 axial flux shape used in Cycle 6. The changed shape results in an allowable increase in the total peak for Cycle 7 to 2.83 from the Cycle 6 value of 2.57. The selection of the Cycle 7 peaking was made to increase flexibility in the determination of operating limits (i.e., rod insertion limits), and is appropriately accounted for in the safety analysis.

The thermal-hydraulic models and methodology used for Cycle 7 are the same as used for Cycle 6, except for the implementation of crossflow modeling with the LYNX1, LYNX2, and LYNXT computer codes (References 11-13, respectively). The crossflow modeling is described in Reference 14 submitted as part of the Cycle 7 reload package. LYNX1 and LYNX2 are approved codes. Our review of LYNXT is not yet complete, but it has progressed sufficiently to allow its use for this application. We reviewed the crossflow modeling described in Reference 13, and find it acceptable for Cycle 7.

Departure from Nucleate Boiling (DNB) margin improvement gained with crossflow modeling would support an increase of the flux/flow reactor trip setpoint up to 1.08 for Cycle 7. The licensee, however, has elected to use a value of 1.07 for this setpoint. This, and the other Technical Specification changes for Cycle 7 have been conservatively selected to permit the potential application of these limits to future cycles without the need for additional Technical Specification changes. Since the changes have been chosen conservatively, this approach is acceptable.

The important thermal-hydraulic parameters are the same for both Cycles 6 and 7 as summarized in Table 1. Based on the similarities of Cycles 6 and 7, we find the operation of Cycle 7 acceptable.

Table 1. Maximum Design Conditions, Cycles 6 and 7

	Cycle 6	Cycle 7
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Vessel inlet/outlet coolant temp at 100% power, F	555.6/602.4	555.6/602.4
DNBR modeling	Closed-channel	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.5 cosine	1.65 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	140.7[a]	141.8
Avg. heat flux at 100% power, 10^3 Btu/h-ft^2	175[a]	174
Max heat flux at 100% power, 10^3 Btu/h-ft^2	450[a]	492
CHF correlation	B&W-2	B&W-2
Minimum DNBR		
At 112% power	2.05	2.08
At 100% power	2.39	2.43

(a) Based on densified length.

5.0 ACCIDENT AND TRANSIENT ANALYSIS

The licensee has examined each FSAR accident analysis with respect to changes in Cycle 7 parameters to determine their effect on the plant thermal performance during the analyzed accidents and transients. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, thermal-hydraulic parameters, and physics and kinetics parameters. Fuel thermal analysis values are listed in Table 4-2 of Reference 3 for all fuel batches in Cycle 7. Table 1 compares the thermal-hydraulic parameters for Cycles 6 and 7. These parameters are the same for both cycles. A comparison of the key kinetic parameters from the FSAR and Cycle 7 is provided in Table 7-2 of Reference 3. These comparisons indicate no significant changes or changes in the conservative direction. The effects of fuel densifictaion on the FSAR accident analyses have also been evaluated.

A generic LOCA analysis for the B&W 177-fuel assembly, lowered loop nuclear steam supply system (NSSS) has been performed using the final acceptance criteria emergency core cooling system (ECCS) evaluation model (Reference 10). That analysis used the limiting values of key parameters for all plants in the 177-FA lowered-loop category, and therefore is bounding for the ANO-1 Cycle 7 operation.

We conclude from the examination of Cycle 7 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the ANO-1 plant's ability to operate safely during Cycle 7.

6.0 TECHNICAL SPECIFICATIONS

As indicated in our review of Section 3, the operating characteristics for Cycle 7 were calculated with well-established, approved methods. In addition, we agreed with the licensee's evaluation of control rod worths and their role in the establishment of Cycle 7 control rod position limits. The proposed Technical Specifications are the result of the cycle-specific analyses for power peaking, control rod worths, and quadrant tilt allowance. We discussed the Specification of the flux/flow reactor trip setpoint in Section 4.

With the above modification, we therefore conclude that the Technical Specification changes proposed by the licensee in Reference 1 and repeated in Section 8 of the Cycle 7 Reload Report (Ref. 3) are acceptable.

At our request, in Reference 2 the licensee withdrew credit for use of the FLECSET heat transfer correlation in the LOCA analysis contained in the original submittal. This affected only the linear heat rate limits. Figure 3.5.2-4, "LOCA Limited Maximum Allowable Linear Heat Rate," as revised in Reference 2, is the correct figure to use. The licensee proposed ~~to~~ delete this figure, but also provided the corrected figure if we did not agree to the deletion. Since this figure defines the "maximum peaking factor allowed by the Technical Specifications" mentioned in Part 50, Appendix K, we do not approve of its deletion.

7.0 STARTUP TESTING

We reviewed the startup testing program for ANO-1 presented in Reference 3. We find that this program will acceptably verify the cycle design and provide data required by the Technical Specifications.

8.0 EVALUATION FINDINGS

We have reviewed the fuels, physics, thermal-hydraulic and transient information presented in the ANO-1 reload report. We find the proposed reload and the associated modified Technical Specifications acceptable.

9.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

10.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: December 20, 1984

Principal Contributors: M. Dunenfeld and G. Vissing

REFERENCES

1. John M. Griffin (AP&L) letter to J. F. Stolz (NRC) on "Cycle 7 Reload Report and Proposed Technical Specification Change Request" dated September 26, 1984.

2. John M. Griffin (AP&L) letter to J. F. Stolz (NRC) on "Revision to ANO-1 Cycle 7 Reload Report" dated October 31, 1984.

3. "Arkansas Nuclear One, Unit 1, Cycle 7 Reload Report," Babcock and Wilcox Company Report BAW-1840 dated August 1984 (transmitted with Reference 1 above).

4. J. Ted Enos (AP&L) letter to John F. Stolz (NRC) on ANO-1 Technical Specification Revision - Cycle 7 Reload, dated December 6, 1984.

5. John F. Stolz (NRC) letter to John M. Griffin (AP&L) on Amendment No. 71, Cycle 6 Reload, dated March 10, 1983.

6. Robert W. Reid (NRC) letter to William Cavanaugh III (AP&L) on Amendment No. 52, Cycle 5, dated March 9, 1981.

7. D. C. Trimble (AP&L) letter to R. W. Reid (NRC) dated November 6, 1980 and transmitting BAW-1626.

8. Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U. S. Nuclear Regulatory Commission Report NUREG-0800 (formerly NUREG-75/087, July 1981.

9. Y. Hsii, et al., "TACO2-Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10141P-A, Rev. 1, June 1983.

10. W. L. Bloomfield, et. al., "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, Revision 1, September 1975.

11. B. R. Hao and J. M. Alcorn, "LYNX1: Reactor Fuel Assembly Thermal-Hydraulic Analysis Code," BAW-10129, October 1976.

12. "LYNX2: Subchannel Thermal-Hydraulic Analysis Program," BAW-10130, October 1976.

13. J. H. Jones, et al, "LYNXT - Core Transient Thermal-Hydraulic Program," BAW-10156, February 1984.

14. R. L. Harne and J. H. Jones, "Thermal-Hydraulic Crossflow Applications," BAW-1829, April 1984.