Docket No. 50-313

MAR 3 1 1977

Arkansas Power & Light Company
ATTN: Mr. J. D. Phillips
Senior Vice President
Production, Transmission and Engineering
Sixth and Pine Streets
Pine Bluff, Arkansas 71601

Gentlemen:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. DPR-51 for Arkansas Nuclear One - Unit No. 1. This amendment revised the provisions of the Technical Specifications in response to your requests dated July 9, 1975, and December 1, 1976, as supplemented by letters dated August 8 and 22, 1975, October 15, 1975, and December 31, 1975, January 13, 1977, February 7, 17, 22 and 24, 1977, and March 1, 9 and 17, 1977.

This amendment: (1) incorporates revised core protection limits in response to the plant specific analysis for Reload 1; (2) incorporates revised fuel rod bow analyses and Technical Specification changes pertinent thereto; (3) incorporates revised and NRC-approved Babcock & Wilcox Company model for nucleate boiling heat transfer correlation during blowdown; (4) incorporates new technical specification limiting conditions for operation and surveillance requirements regarding core internal vent valves; and (5) incorporates modified operating limits in the Technical Specifications based upon an evaluation of emergency core cooling (ECCS) performance calculated in accordance with an acceptable ECCS evaluation model that conforms with the requirements of Appendix K of 10 CFR Part 50 and as required by the Commission's Order for License Modification dated December 27, 1974, with the following exception. The Commission's analysis of electrical single failure criterion is still under consideration and will be the subject of a separate review.

The incorporation of modified operating limits for ECCS (item 5 above) supersedes the restrictions imposed by the Commission's Order for License Modification dated December 27, 1974.

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The portion of your January 13, 1977 letter related to examination of primary nozzle-to-vessel welds has been previously evaluated and resulted in issuance of Amendment No. 20, dated March 15, 1977.

This amendment constitutes approval for operation of Arkansas Nuclear One - Unit No. 1 for fuel cycle 2.

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

Sincerely,

Original signed by

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 21 to License No. DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Notified AP&L (Don Rueter) that this package was signed. 3/31/77 1700.

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SURNAME	RMDiggs	RPSnaider:rd	RBaer	WButler	Davis	DKDavis
DATE	3/30/77	3/30/77	3/31/77	3/31/77	3/30/77	3/31/77

For further details with respect to this action, see (1) the applications for amendment dated July 9, 1975, and December 1, 1976, as supplemented by letters dated August 8 and 22, 1975, October 15, 1975, December 31, 1975, January 13, 1977, February 7, 17, 22, and 24, 1977, and March 1, 9, and 17, 1977, (2) Amendment No. to Facility Operating 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 single 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

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

OFFICE ➤						
SURNAME ➤						
DATE ➤						

cc w/enclosures:
Horace Jewell, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Phillip K. Lyon, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Mr. Donald Rueter
Manager, Licensing
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72201

Arkansas Polytechnic College
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

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

cc w/enclosures and copy of AP&L filings referenced in first paragraph of this letter, except for proprietary information:
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.
License No. DPR-51

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

   A. The applications for amendment by Arkansas Power & Light Company (the licensee) dated July 9, 1975 and December 1, 1976 (as supplemented by letters dated August 8 and 22, 1975, October 15, 1975, December 31, 1975, January 13, 1977, February 7, 17, 22 and 24, and March 1, 9 and 17, 1977), comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.   , are hereby incorporated in this 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

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

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DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Accomplish page changes to the Appendix A portion of the Technical Specifications as noted below. The changed areas on the revised pages are identified by a marginal line.

Remove Existing Page	Add Revised Page
7	7
8	8
9	9
9a	9a
9b	9b
9c	9c
11	11*
12	12
13	13
14a	14a
14b	14b
15	15
16	16
17	17
29	29*
30	30
47	47
48	48
48b	48b
- -	48bb
- -	48bbb
48c	48c
- -	48cc
- -	48ccc
48d	48d
48dd	48dd
48ddd	48ddd
48e	48e
48f	- -
73a	73a
101	101*
102	102

*There were no changes on these pages. They are included as a matter of convenience in updating the Technical Specifications.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the pressure/temperature line the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/reactor-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation. (1) The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the radio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A

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 of 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;

$$F_q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_z^N = 1.50$$

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

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 and fuel rod bowing:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ 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 19.4 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 specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 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 of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent[1], whichever condition is more restrictive.

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.

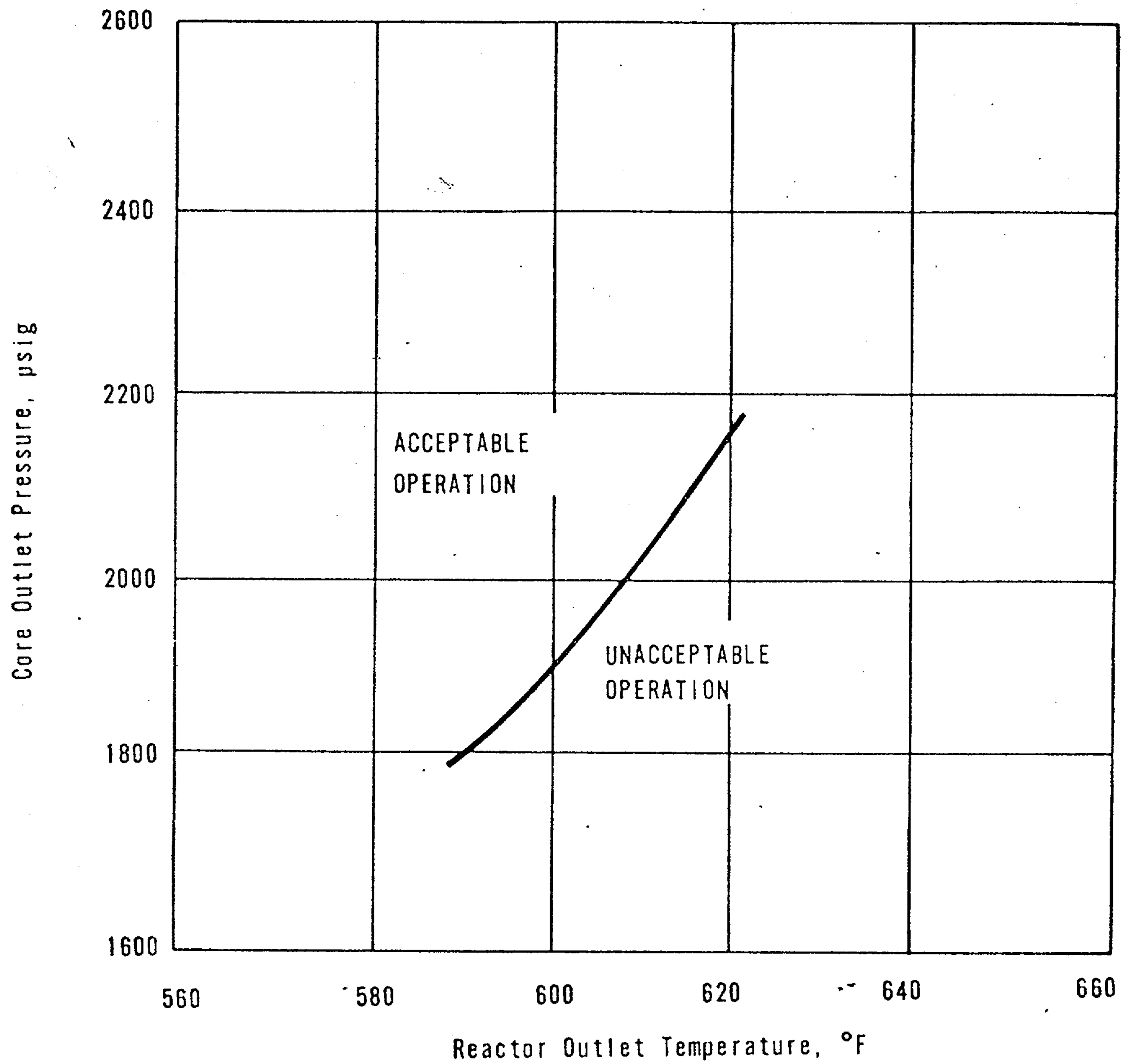
The maximum thermal power for three pump operation is 86.0 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.065= 79.6 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced 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.

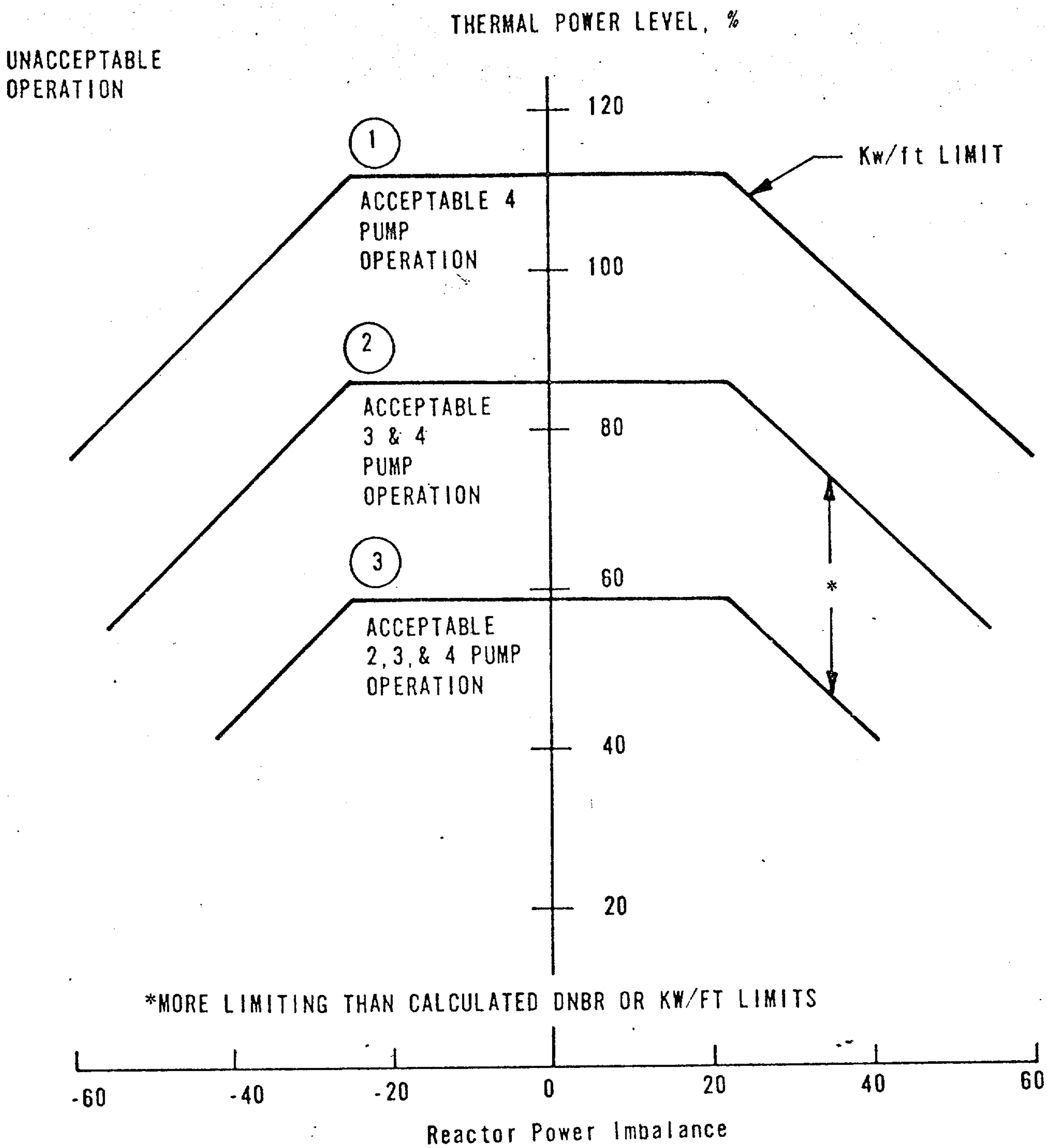
REFERENCES

(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

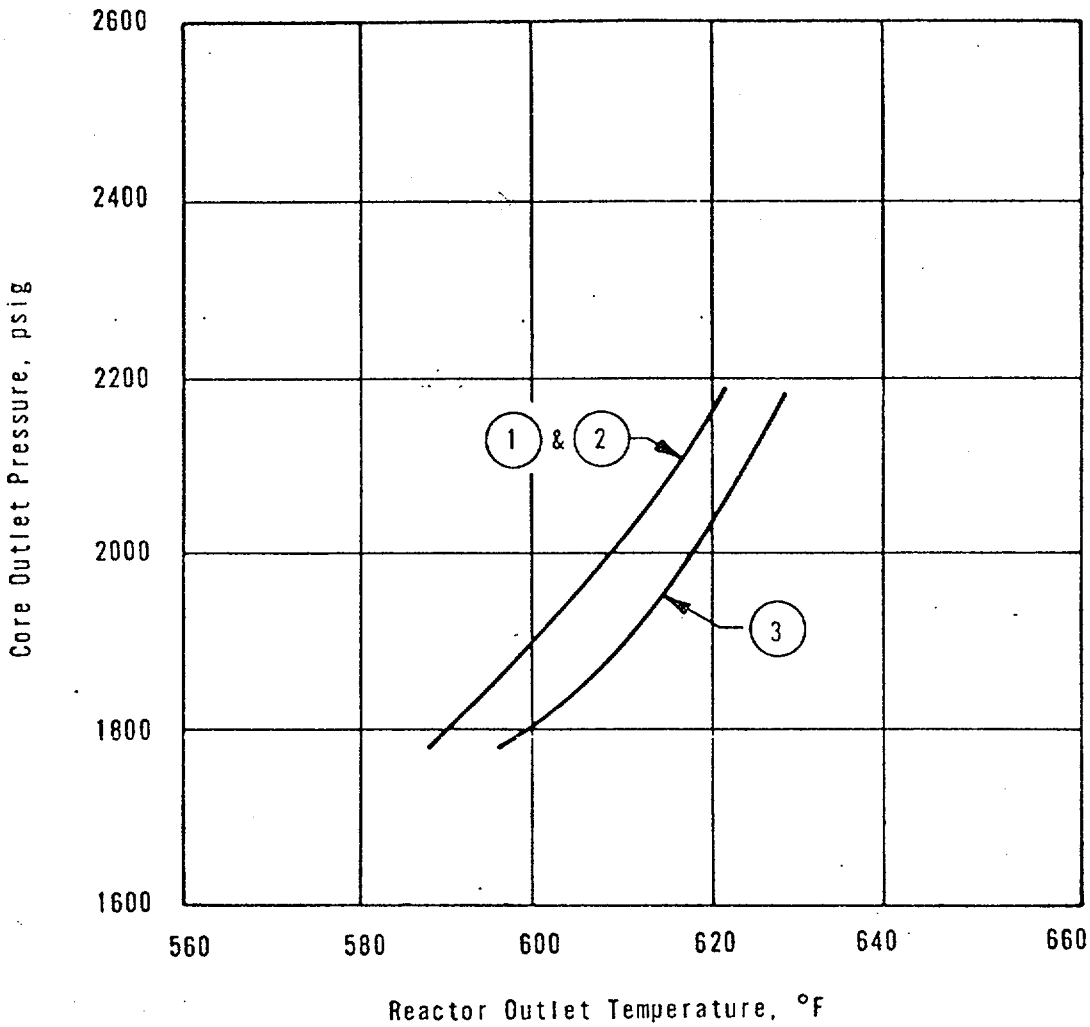


ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CORE PROTECTION SAFETY LIMIT	FIG. NO. 2.1-1



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

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 1	CORE PROTECTION SAFETY LIMITS	FIG. NO. 2.1-2



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%)*	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	86.7%	THREE PUMPS (DNBR LIMIT)
3	184,441 (49.2%)	59.0%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

*106.5% OF DESIGN FLOW

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	CORE PROTECTION SAFETY LIMITS	FIG. NO. 2.1-3

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 more conservative than 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 106.5 percent and reactor flow rate is 100 percent or flow rate is 93.9 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.5 percent and reactor flow rate is 74.7 percent or flow rate is 70.4 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 46.0 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 1.065 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. 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 shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (2) The low pressure (1800 psig) and variable low pressure ($11.75T_{out}$ -5103) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of ($11.75T_{out}-5143$).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620F.

E. Reactor building pressure

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

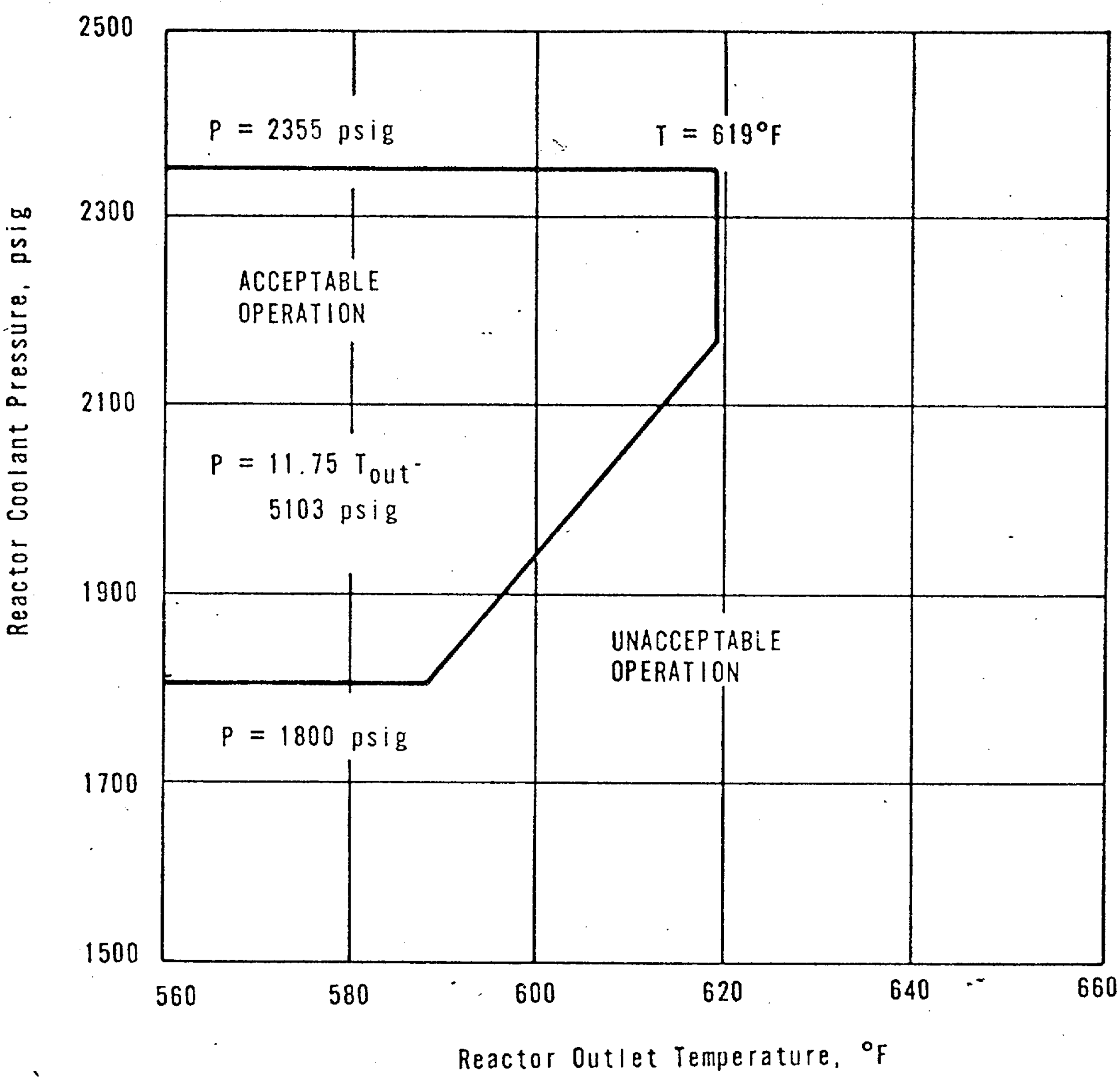
F. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of $\leq$5.0 percent of rated power is automatically imposed during reactor shutdown.

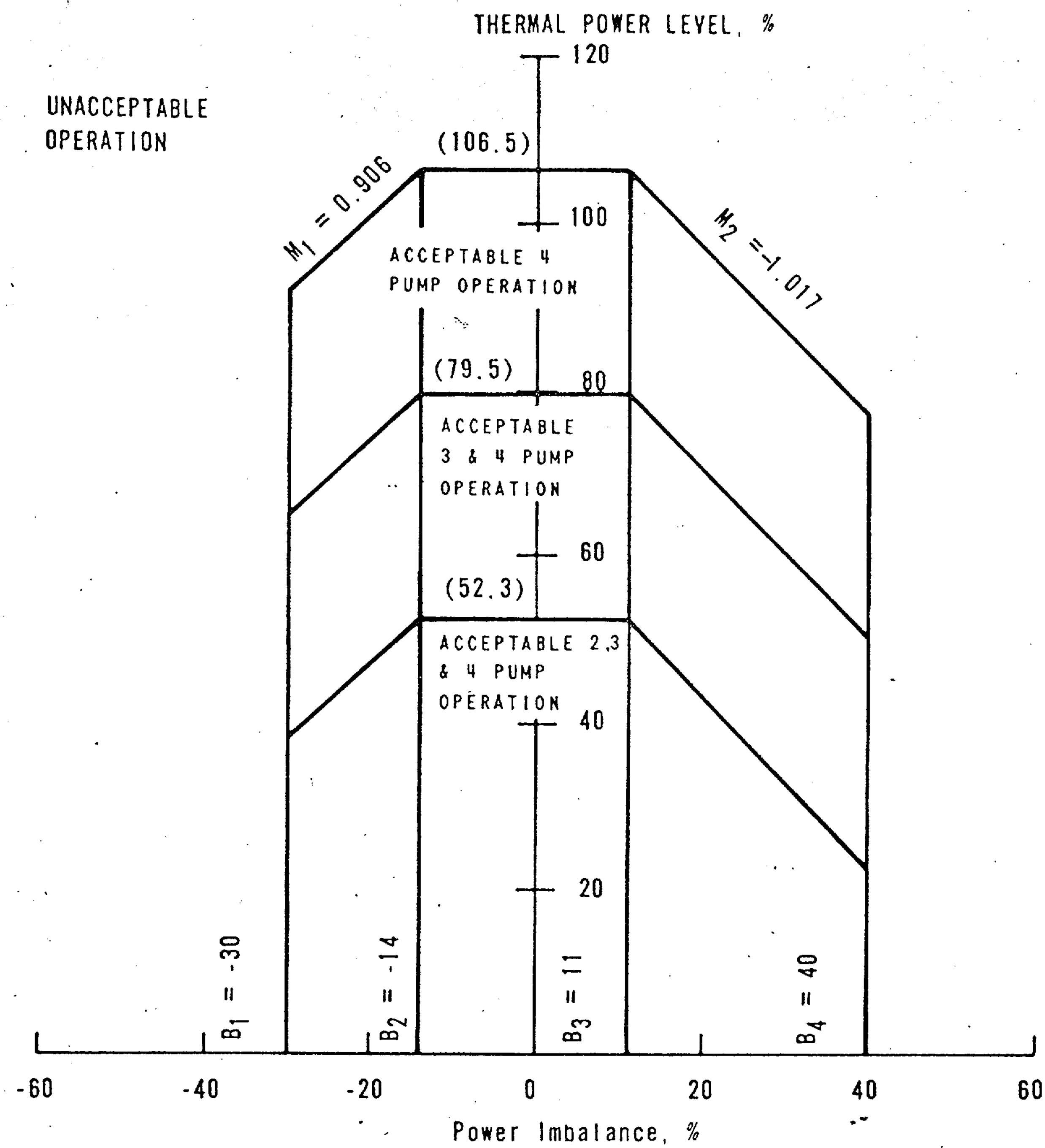
2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of <5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficent natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SET POINT	FIG. NO. 2.3-1

Amendment No.



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 1	PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS	FIG. NO. 2.3-2

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	1.065 times flow minus reduction due to imbalance(s)	1.065 times flow minus reduction due to imbalance(s)	1.065 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.

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

A. One steam generator shall be operable whenever the reactor coolant average temperature is above 280 F.

3.1.1.3 Pressurizer Safety Valves

A. The reactor shall not remain critical unless both pressurizer code safety valves are operable.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1.

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less.[1]

The decay heat removal system suction piping is designed for 300 F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. [4] Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.[5] The pressurizer code safety valve lift set point shall be set at 2500 psig ± 1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7.

(2) FSAR, Section 4.2.5.1 and 9.5.2.3.

(3) FSAR, Section 4.2.5.4.

(4) FSAR, Section 4.3.10.4 and 4.2.4.

(5) FSAR, Section 4.3.7.

b. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, reactor coolant temperature, pressurizer water level and reactor coolant makeup tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the reactor coolant makeup tank resulting in a tank level decrease. The reactor coolant makeup tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 2 inches of tank height. This inventory monitoring method is capable of detecting changes on the order of 62 gallons. A 1 gpm leak would therefore be detectable within approximately 1.1 hours.

   As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on different principles, i.e., activity, sump level and reactor coolant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

c. The reactor building gaseous monitor is sensitive to low leak rates if expected values of failed fuel exist. The rates of reactor coolant leakage to which the instrument is sensitive are discussed in FSAR Section 4.2.3.8.

The upper limit of 30 gpm is based on the contingency of a hypothetical loss of all AC power. A 30 gpm loss of water in conjunction with a hypothetical loss of all AC power and subsequent cooldown of the reactor coolant system by the atmospheric dump system and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore both electrical power to the station and makeup flow to the reactor coolant system.

References

FSAR Section 4.2.3.8

3.1.7. Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

Bases

A non-positive moderator coefficient 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 $+0.5 \times 10^{-4}\ \Delta k/k/^{o}F$ 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 $+0.5 \times 10^{-4}\ \Delta k/k/^{o}F$.

When the hot zero-power value is corrected to obtain the hot 95% value, the following corrections will be applied.

1. Uncertainty in isothermal measurement — The measured moderator temperature coefficient will contain uncertainty owing to (a) $\pm 0.2^{o}F$ in the ΔT of the base and perturbed conditions, and (b) uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4}\ \Delta k/k$.

   Proper corrections will be added for these conditions to provide a conservative moderator coefficient.

2. Doppler coefficient at hot zero power — During measurement of the isothermal moderator coefficient at hot zero power, the fuel temperature will increase by the same amount as for the moderator. The measured temperature coefficient must be increased by $0.21 \times 10^{-4}\ (\Delta k/k)/^{o}F$ to obtain a pure moderator temperature coefficient.

3. Moderator temperature change — The hot zero-power measurement must be reduced by $0.08 \times 10^{-4}\ \Delta k/k/^{o}F$. This corrects for the difference in water temperature from zero power (532F) and 15% power (580F). Above this power, the average moderator temperature remains 580F.

4. Fuel temperature interaction (power effect) — The moderator coefficient must be adjusted to account for the interaction of an average moderator temperature with increasing fuel temperatures (as power increases). This correction is $-0.0022 \times 10^{-4}\ \Delta\alpha_m/\Delta\%$ power. It adjusts the moderator coefficient at 15% power to the coefficient at any power level above 15%. For example, the power effect adjustment from a 15% coefficient to 100% power is

$$(-0.0022 \times 10^{-4})(100\% - 15\%) = -0.187 \times 10^{-4}\ \Delta\alpha_m.$$

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 delcared 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 3.41%, 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 3.41% 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 3.41%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 3.41% 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 withdrawal limits (Figures 3.5.2-1A, 3.5.2-1B and 3.5.2-1C shall be reduced 2% in power for each 1% tilt in excess of 3.41%.

   c. The operational imbalance limits (Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C) shall be reduced 2% in power for each 1% tilt in excess of 3.41%.

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, 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. If the 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.

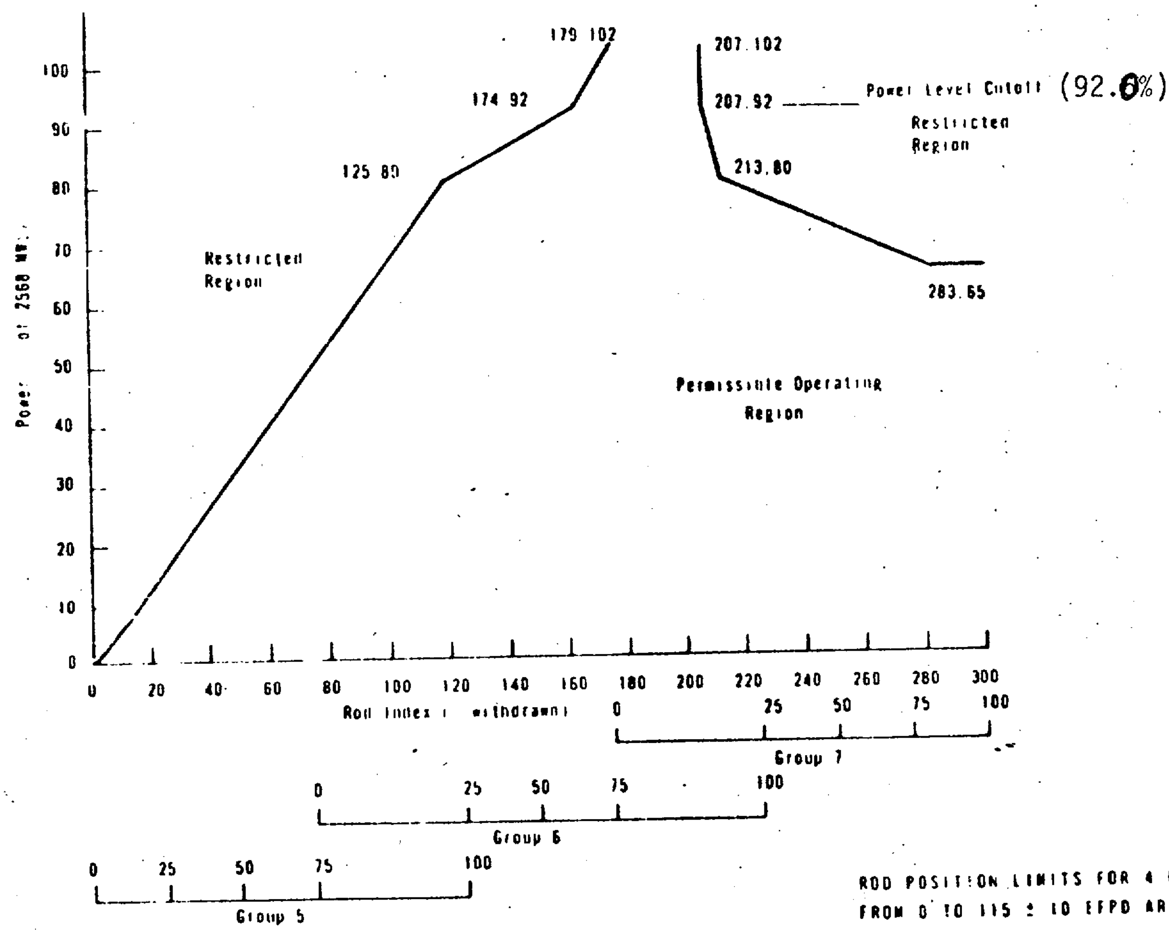
Bases

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

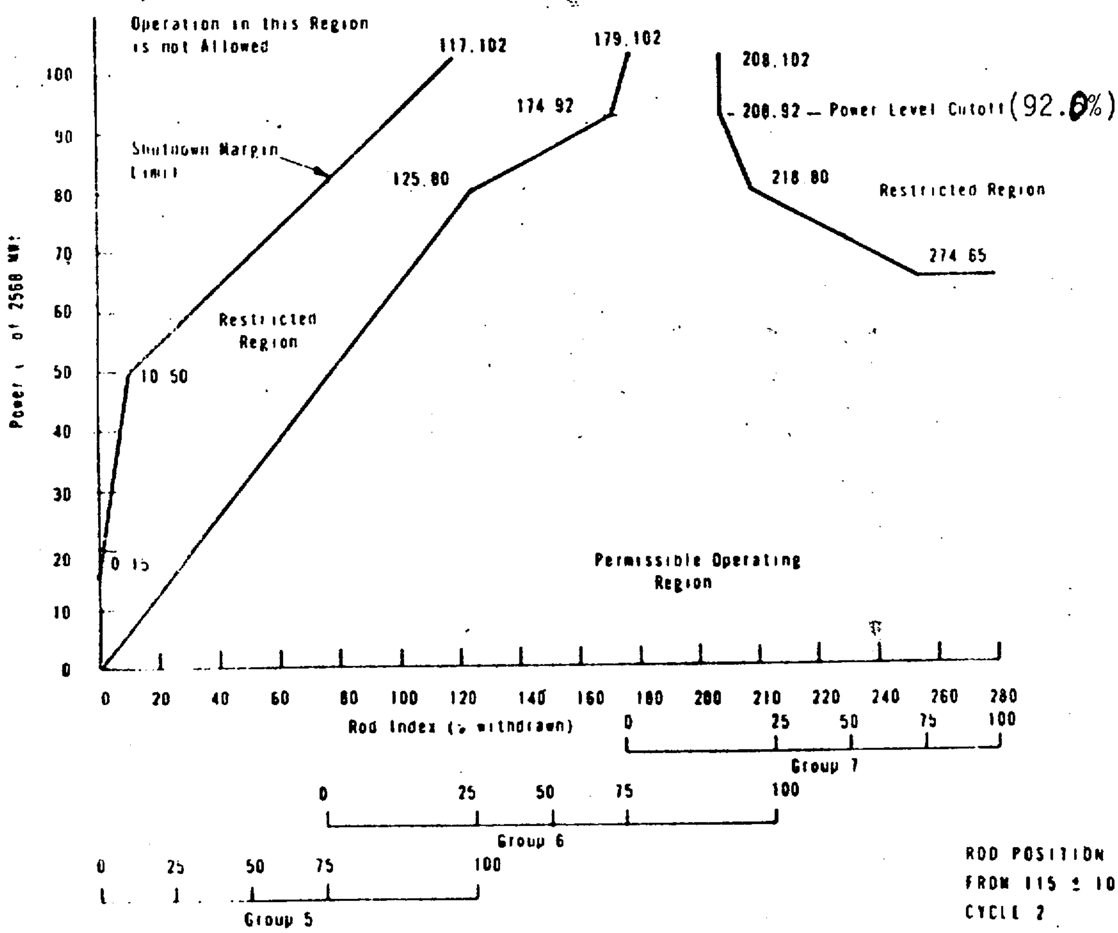
The 25 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

*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.



ROD POSITION LIMITS FOR 4 PUMP OPERATION
FROM 0 TO 115 ± 10 EFPD ARKANSAS CYCLE 2
Figure 3.5.2-1A

Amendment No. ~~6~~,



ROD POSITION LIMITS FOR 4 PUMP OPERATION
FROM 115 ± 10 TO 225 ± 10 EFPD ARKANSAS
CYCLE 2

Figure 3.5.2-1B

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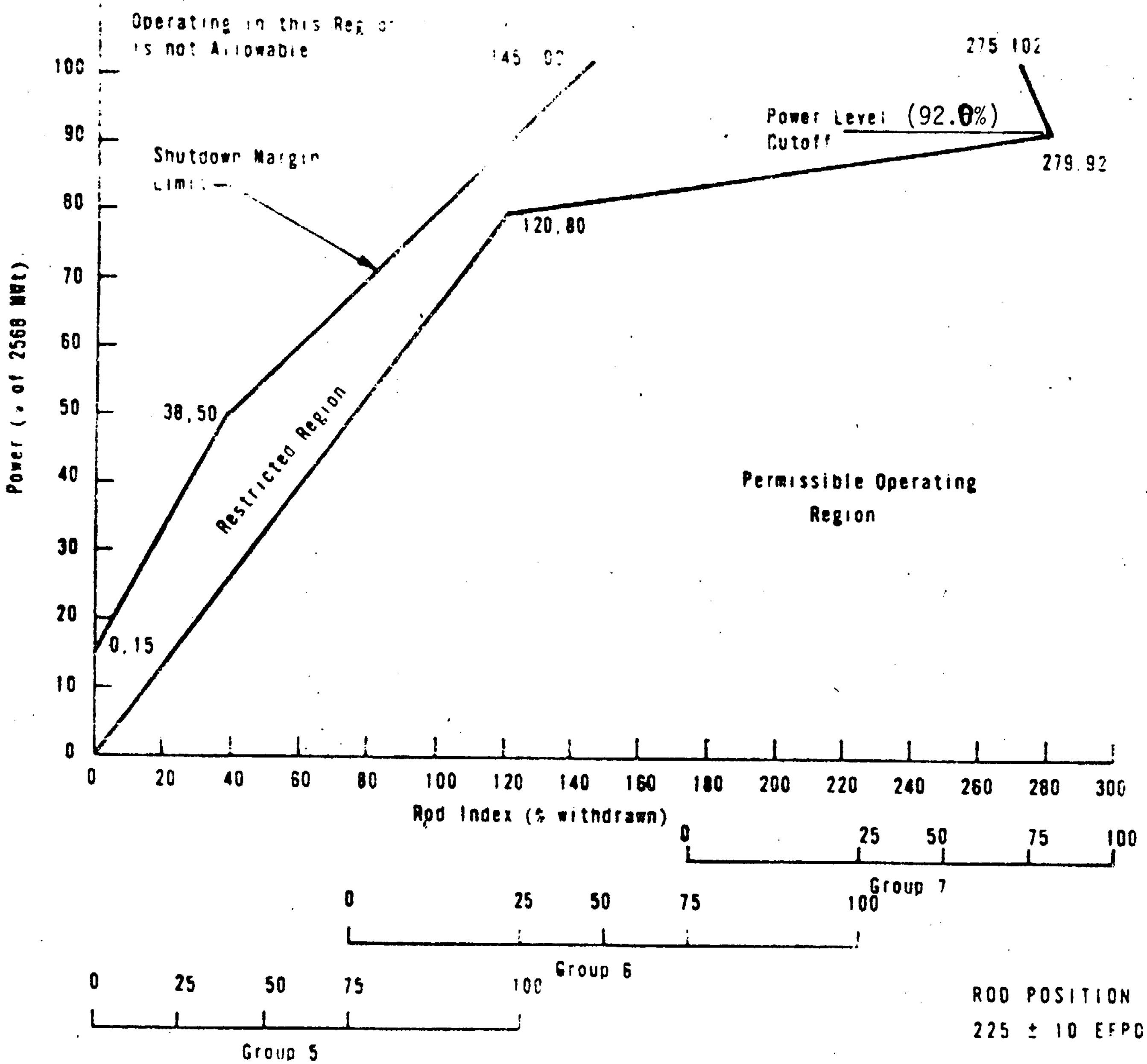
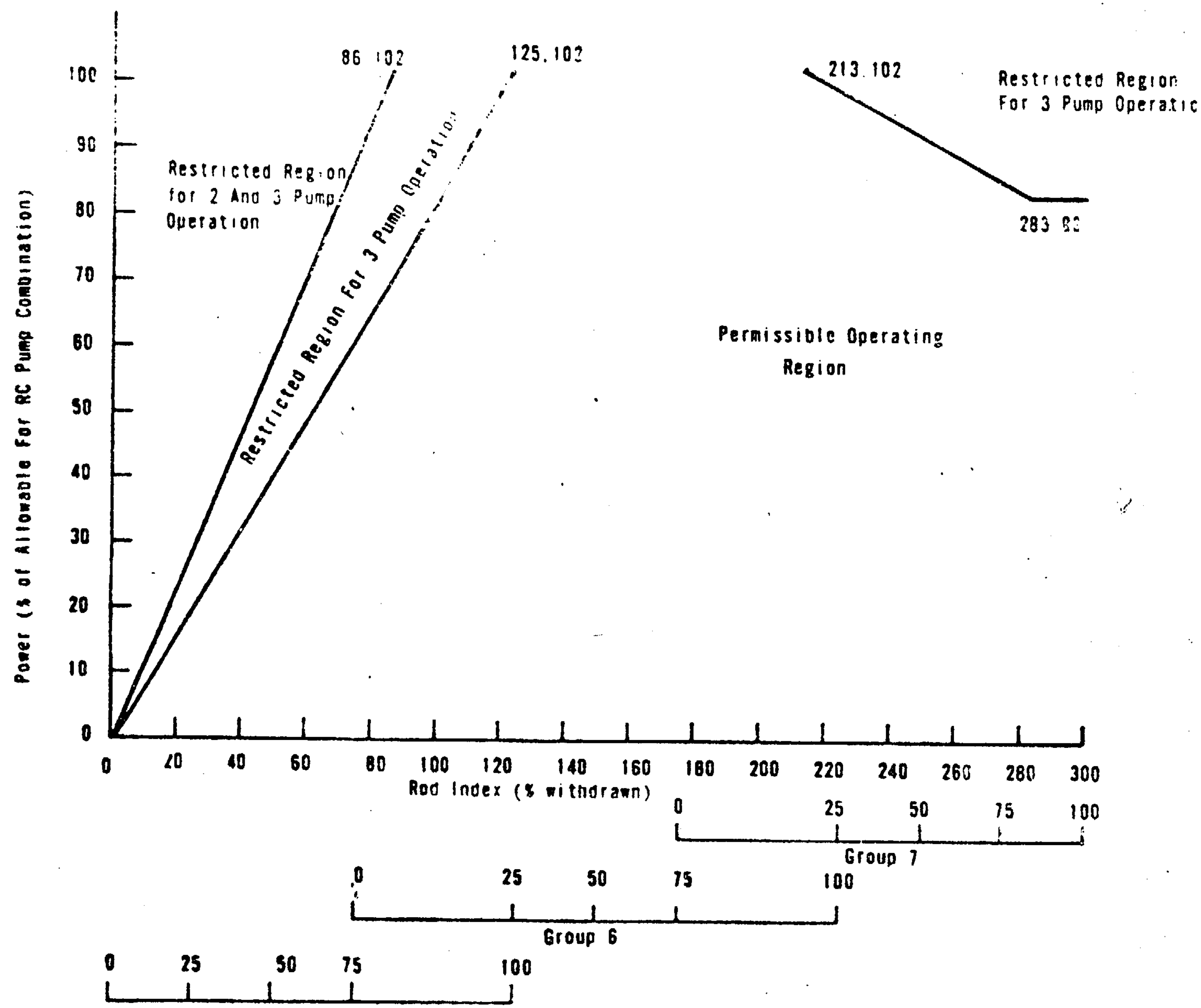
Amendment No.



ROD POSITION LIMITS FOR 4 PUMP OPERATION AFTER
225 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-1C

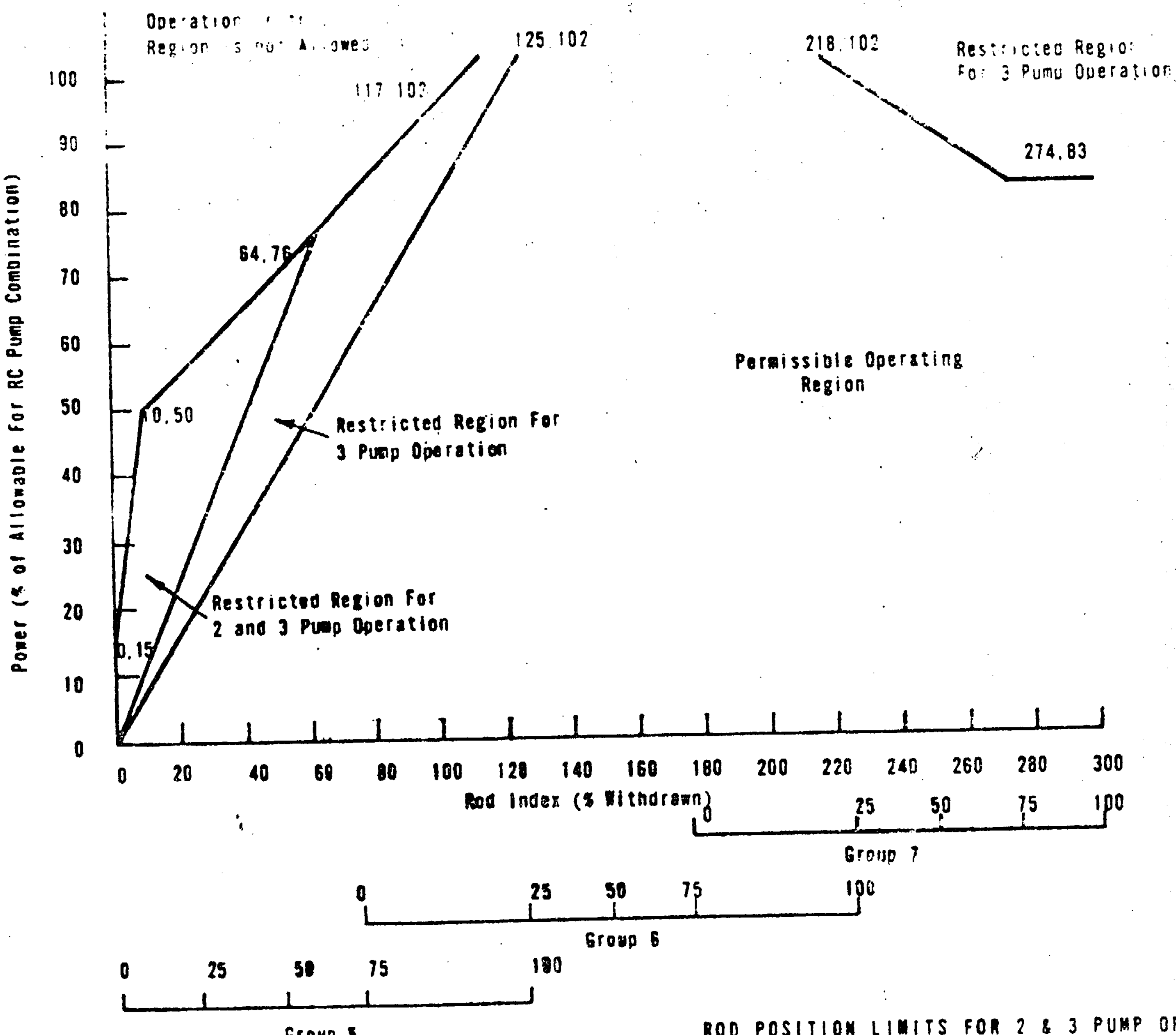
48bbb



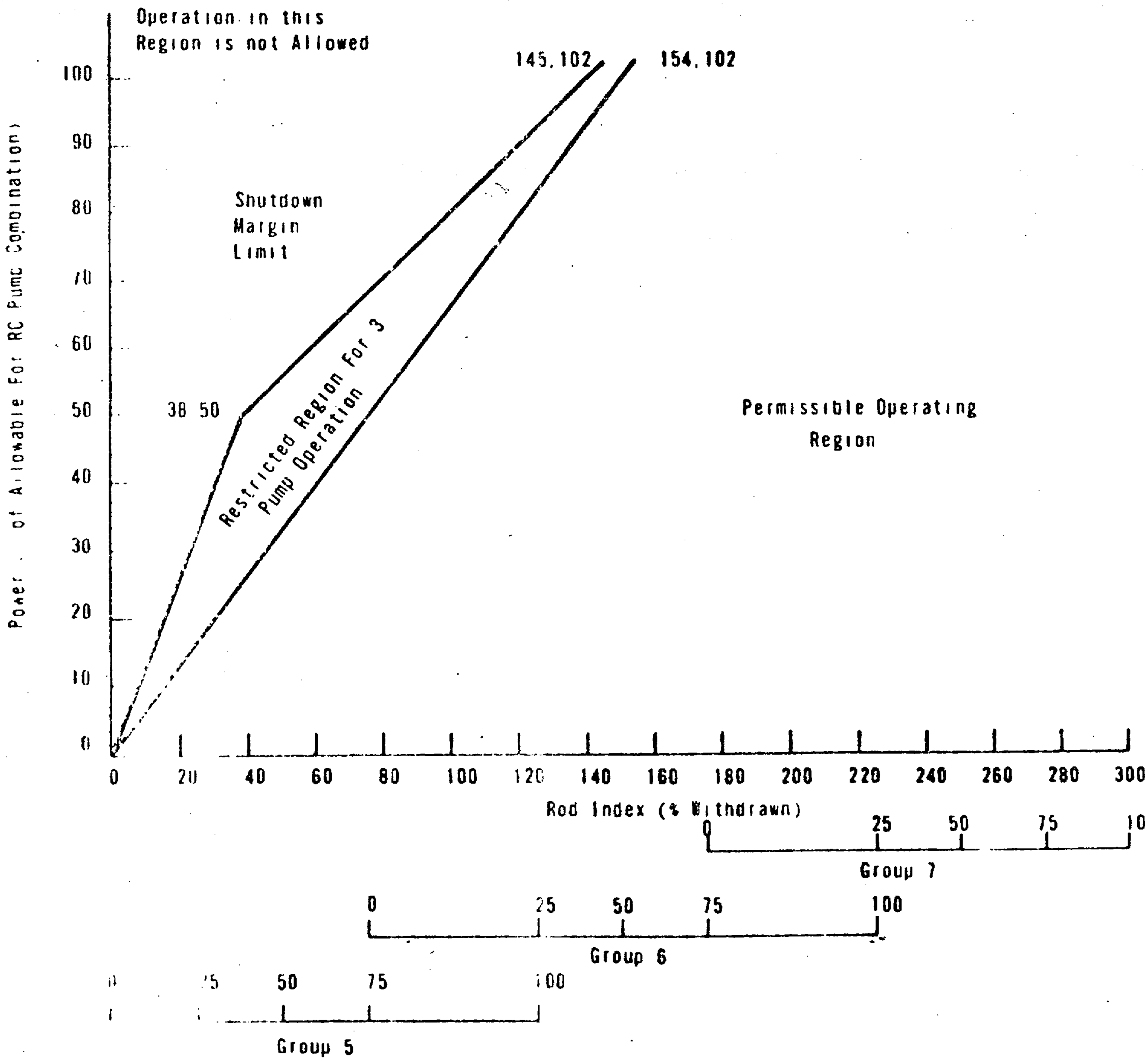
ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION
FROM 0 TO 115 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3 5 2-2A

48c



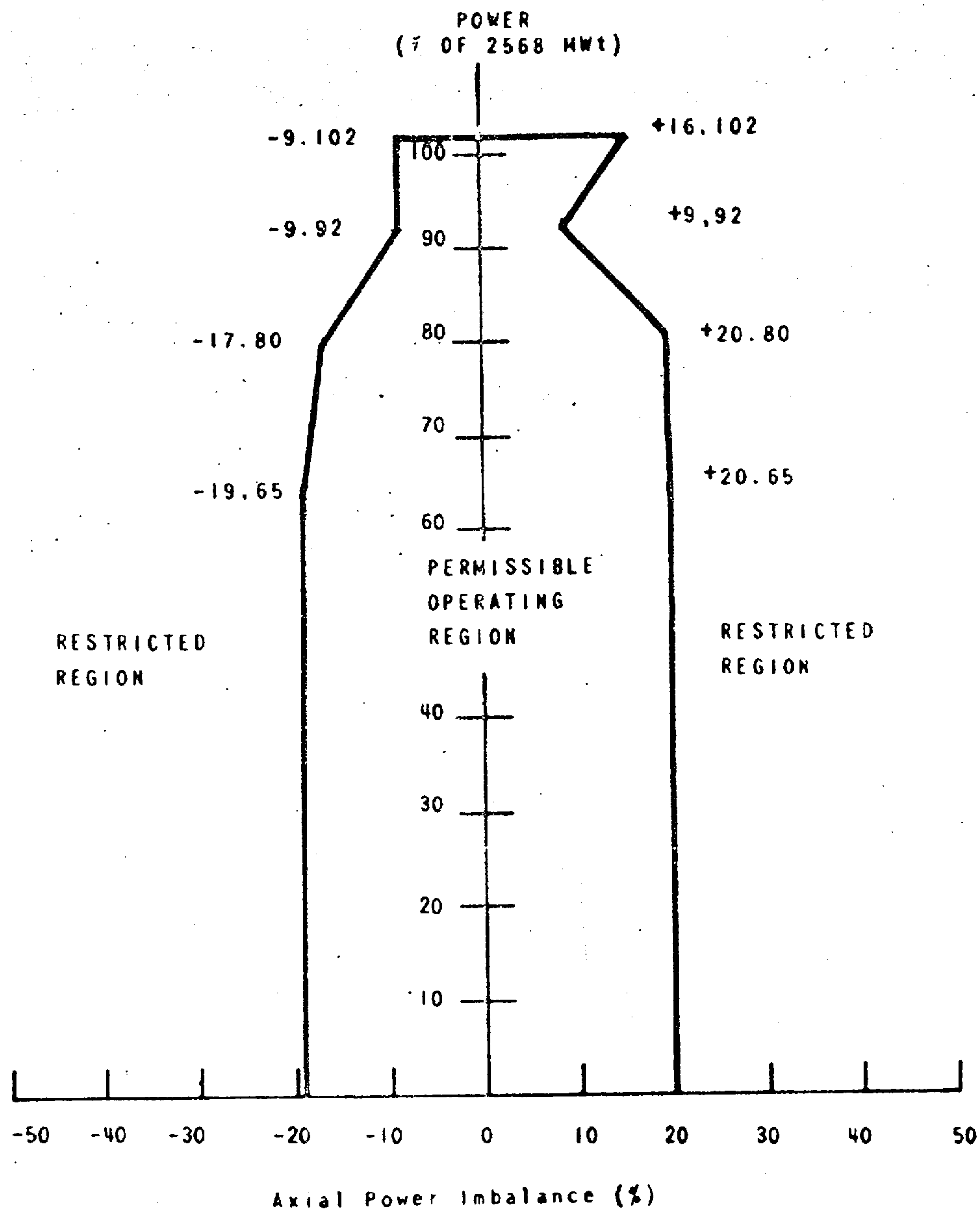
ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION
FROM 115 ± 10 TO 225 ± EFPD ARKANSAS CYCLE 2
Figure 3.5.2-2B



ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION AFTER 225 ± 10 EFPD ARKANSAS CYCLE 2

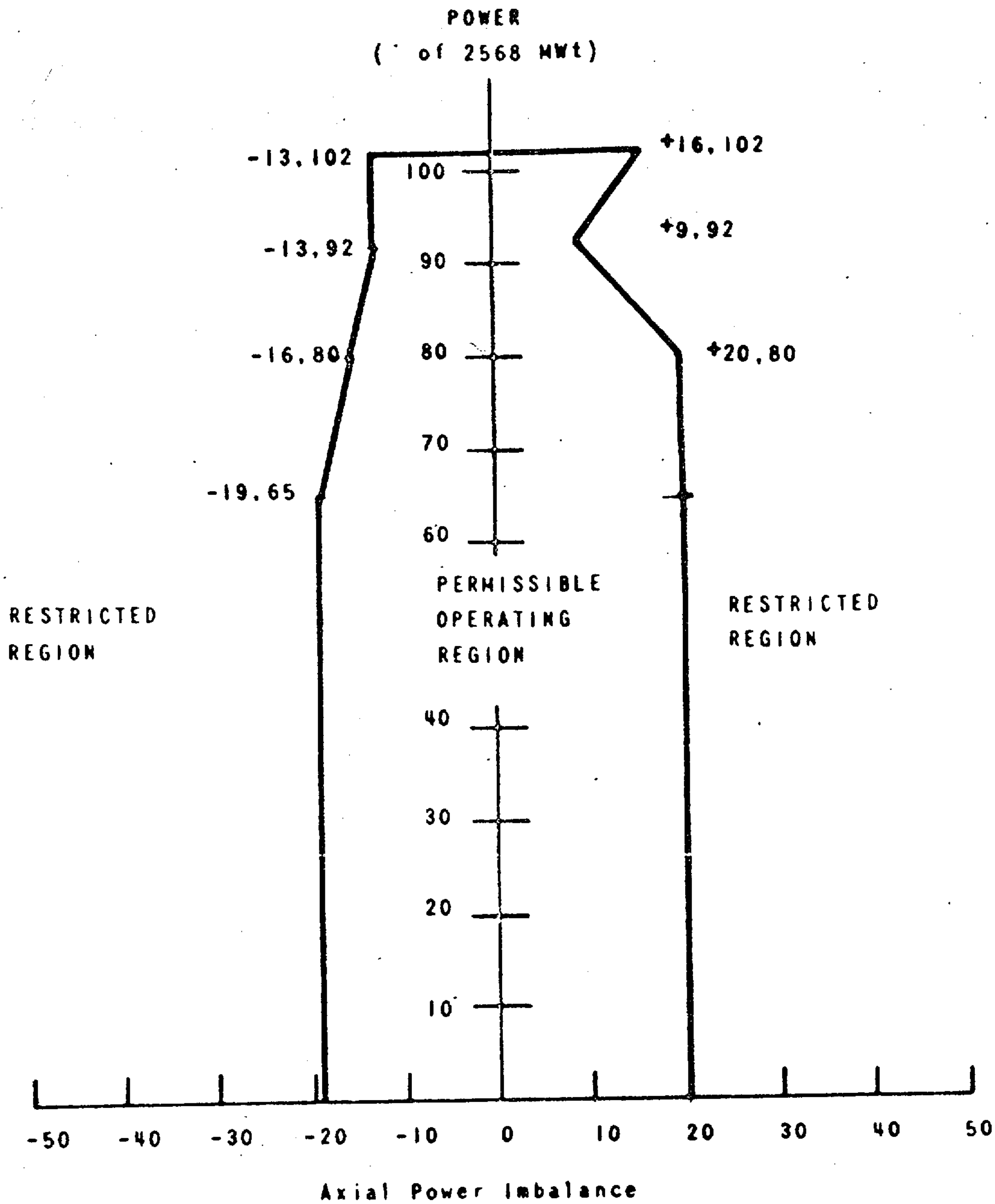
Figure 3.5.2-2C

Amendment No.



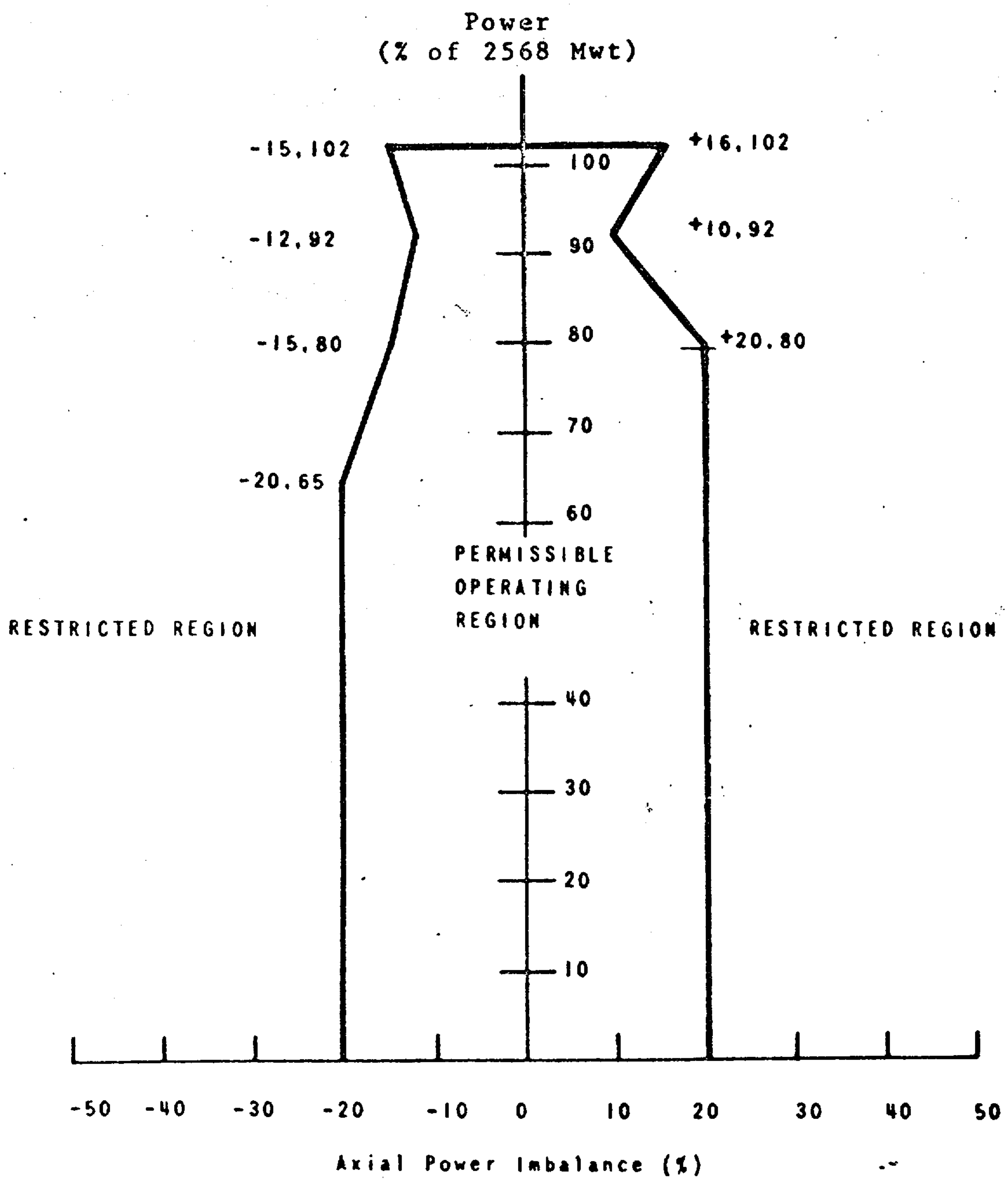
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION
FROM 0 TO 115 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-3A



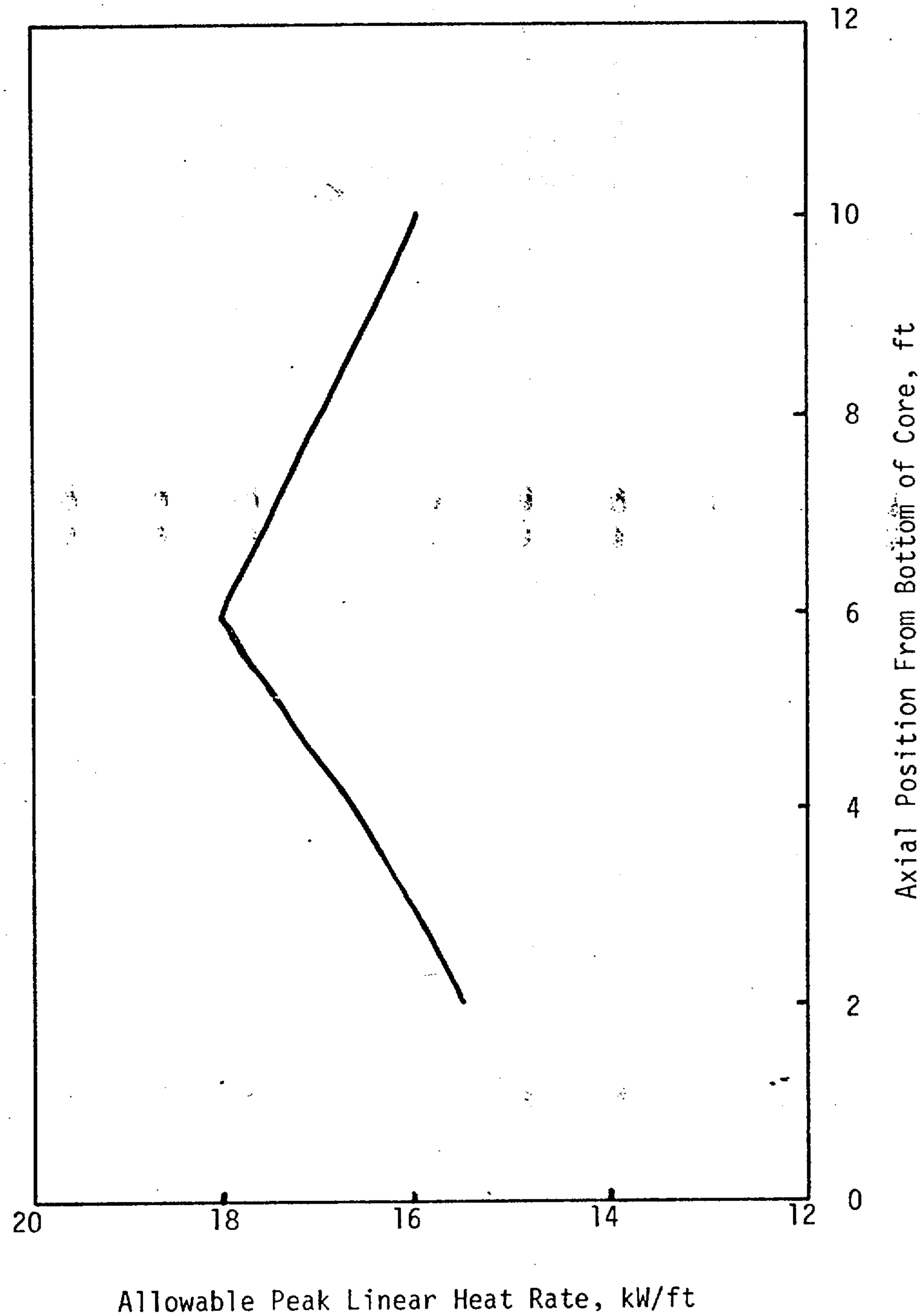
OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 115 ± 10 TO 225 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-3B



OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 225 ± 10 EFPD ARKANSAS, CYCLE 2

Figure 3.5.2-3C



ARKANSAS POWER & LIGHT COMPANY ARKANSAS NUCLEAR ONE-UNIT 1	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	FIG. NO. 3.5.2.4

Amendment No.

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

Item	Test	Frequency
12. Flow Limiting Annulus on Main Feedwater Line at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.
13. SLBIC Pressure Sensors	Calibrate	Each Refueling Period
14. Main Steam Isolation Valves	a. Excercise Through Approximately 10% Travel	a. Quarterly
	b. Cycle	b. Each Refueling Shutdown.
15. Main Feedwater Isolation Valves	a. Exercise Through Approximately 5% Travel	a. Quarterly
	b. Cycle	b. Each Refueling Shutdown.
16. Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs (applied vertically upward).	Each refueling shutdown.

Amendment No. 4,

Bases

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required engineered safety features from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and diesel starter motor air pressure are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests at intervals not to exceed each refueling shutdown period.

Considering system redundancy, the specified testing intervals for the station batteries should be adequate to detect and correct any malfunction before it can result in system malfunction. Batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

Routine battery maintenance specified by the manufacturer includes regularly scheduled equalizing charges in order to retain the capacity of the battery. A test discharge should be conducted to ascertain the capability of the battery to perform its design function under postulated accident condition. An excessive drop of voltage with respect to time is indicative of required battery maintenance or replacement.

Testing of the emergency lighting is scheduled annually and is subject to review and modification if experience demonstrates a more effective test schedule.

References

FSAR, Section 8

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.46 seconds at reactor coolant full flow conditions or 1.20 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14.

Each control rod drive mechanism shall be exercised by a movement of approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

Introduction

Arkansas Power & Light Company (AP&L) proposed to reload Arkansas Nuclear One - Unit 1 for Reload 1 (Cycle 2) and requested amendment to Facility Operating License No. DPR-51 by letter dated December 1, 1976 and supplements thereto dated January 13, February 7, 17, 22 and 24, and March 1, 9 and 17, 1977. By filing dated July 9, 1975, as supplemented by letters dated August 8 and 22, October 15, December 13, 1975, and the December 1, 1976 reload request, AP&L submitted their emergency core cooling system (ECCS) performance reevaluation as required by Appendix K of 10 CFR Part 50 of the Commission's regulations and the Commission's Order for License Modification dated December 27, 1974.

The amendment would modify the license and Technical Specifications to allow operation of the facility with:

(1) revised core protection limits in response to the plant specific analysis for reload 1;

(2) revised limits in response to modified fuel rod bow analyses;

(3) revised limits to reflect the modified Babcock and Wilcox (B&W) Company model for nucleate boiling heat transfer correlation during blowdown,

(4) new technical specification limiting conditions for operation and surveillance requirements governing core internal vent valves; and

(5) modified operating limits based upon an evaluation of emergency core cooling system (ECCS) performance calculated in accordance with an acceptable ECCS evaluation model that conforms with the requirements of Appendix K of 10 CFR Part 50 and as required by the Commission's Order for License Modification dated December 27, 1974, with the following exception. Our analysis of the electrical

single failure criterion is still under consideration and will be the subject of a separate review. The incorporation of the modified operating limits relating to ECCS supersede the restrictions imposed by the Commission's Order dated December 27, 1974.

During our review of the proposed technical specifications, we determined that certain changes were necessary to conform with regulatory requirements. These changes have been accepted by AP&L. That portion of the January 13, 1977 letter related to examination of primary nozzle-to-vessel welds was authorized by Amendment No. 20 issued on March 15, 1977.

Discussion and Evaluation

Fuel Reload

The ANO-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The reload in preparation for cycle 2 operation consists of the removal of all 56 batch 1 fuel assemblies, the relocation of some of the partially-spent batch 2 and batch 3 fuel assemblies, and the placement of the new batch of fuel assemblies in 8 positions in the interior of the core and the remaining 48 in the periphery of the core. Tables 4-2 and 4-3 of reference 1 summarize the reload core fuel assemblies parameters.

Fuel Mechanical Design

The outside dimensions and configuration of the new Mark B-4 (Batch 4) fuel assemblies and once-burned Mark B-3 fuel assemblies are identical except that the Mark B-4 have spring-type flexible spacers and the Mark B-3 have corrugated-type flexible spacers. This new fuel rod spacer has been previously reviewed and found acceptable by the NRC staff on the basis of no significant mechanical or material change to the reactor operation[9] and has been successfully operating in similar cores for a substantial time (Reference Section 4.5 and Table 4-1 of Reference 1). The new Mark B-4 fuel assemblies, therefore, do not represent any unreviewed or untested change in mechanical design from the reference cycle and are therefore acceptable.

This mechanical design change has been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that this fuel design difference in the ANO-1 core is of negligible effect and that the once burned fuel assemblies, batches 2 and 3, are limiting.

Fuel rod cladding creep collapse analyses were performed for the cycle 2 core. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse.[1] The calculational methods, assumptions, and data have been previously reviewed and approved by the NRC staff.[10] The analysis assumed a 2000 hour densification time which maximizes creep; no fission gas production which maximizes differential pressure; and lower tolerance limit on clad thickness and upper tolerance limit on clad ovality, both of which maximize cladding creep deformation. Also, to be conservative, the most restrictive as-built fuel density was applied to the worst power region in the core. The actual operating history along with the most restrictive future power histories to which the partially-spent fuel assemblies may be exposed were used in the analyses of Batch 2 and Batch 3 fuels. The Batch 4 fuel analysis was not specifically performed because for cycle 2 operation Batch 3 fuel has been determined to have the most restrictive power level and will therefore be most limiting. An analysis of the Batch 4 fuel will be performed for cycle 3. Based on the analyses performed, no cladding creep collapse is predicted during the life of the fuel.

From the viewpoint of cladding stress due to differential pressure, thermal stress due to fuel temperature gradients, and bending stress, neither the yield stress nor the B&W 1% total strain criterion for the cladding is predicted to be exceeded in the cycle 2 core. The cladding stress estimated for cycle 1 core will envelope the limiting stresses for the cycle 2 core because of the lower prepressurization and lower fuel pellet density of the cycle 1 Batch 1 fuel. The B&W fuel design criterion for cladding circumferential plastic strain was shown to be satisfied for ANO-1 fuel.[3] This analysis used the maximum fuel pellet diameter burnup and density, and the minimum cladding inside diameter.

The Batch 4 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previously successful operations with equivalent fuel the staff concludes that the fuel mechanical design for cycle 2 operation is acceptable and its application to cycle 2 operation will not endanger the health and safety of the public.

Fuel Thermal Design

The fuel thermal design analysis was conducted with the TAFY-3 computer code, as discussed in reference 2. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in the "Fuel Densification Report".[3] Modifications to the "Fuel Densification Report" on the fuel pellet void probability, F_q, and fuel grain size distribution, F_k, have been previously reviewed and approved by the NRC staff.[11]

As part of our evaluation of the TAFY code, the following modifications to the code were approved for use in reference 4:

1. The code option for no restructuring of fuel has been used;

2. The calculated gap conductance was reduced by 25%.

During cycle 2 operation, the highest power levels are predicted to occur in Batch 3 fuel. The fuel temperature analysis for this fuel, as documented in reference 5, is applicable for cycle 2 and is based on limiting Beginning-of-Cycle (BOC) conditions. Based on the analyses presented in reference 1 and comparison with allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations,(12) the fuel thermal design for the ANO-1 cycle 2 core is acceptable and can be applied with reasonable assurance that the health and safety of the public will not be endangered.

Fuel Material Design

The fuel material design for cycle 2 operation is not significantly different from that of cycle 1 operation. The only difference is that Zircaloy-4 is used as the fuel assembly tubular spacer material in Batch 4 fuel instead of zirconium dioxide (ZrO_2), which is used in Batch 2 and Batch 3 fuel. This change does not affect the primary coolant system chemistry. This change has been reviewed and has a substantial amount of previous experience (Section 4.5 and Table 4-1 of reference 1). Therefore, the fuel material design for ANO-1 cycle 2 operation is acceptable.

Nuclear Analysis

The reactor physics parameters for ANO-1 cycle 2 core were calculated with PDQ07. Since the ANO-1 core has not yet reached an equilibrium cycle, minor differences in the physics parameters between the initial cycle and cycle 2 cores are expected but are not significant. These insignificant differences include the technical specification basis change to α_T* due to cycle dependent parameters. In view of this and the fact that the startup tests which will be conducted prior to power operation will verify that the significant aspects of the core performance, e.g., control rod drive tests, scram times, shutdown margin, criticality checks, power symmetry, and instrument calibration are within specified acceptance criteria, the staff finds AP&L's nuclear analysis for cycle 2 to be acceptable.

*The change represents a cycle-dependent correction to the moderator temperature coefficient in going from hot zero power to 95% of rated conditions and accounts for the difference in fuel temperature.

Thermal-Hydraulic Analysis

Major acceptance criteria for the thermal-hydraulic design are specified in the NRC's Standard Review Plan Section 4.4 ("Thermal and Hydraulic Design"). These criteria establish the acceptable limits for DNBR (Departure from Nucleate Boiling Ratio). The thermal-hydraulic analyses for the ANO-1 cycle 2 reload core were made with previously approved models and methods, as stated in the ANO-1 Final Safety Analysis Report (Docket No. 50-313).

The effect of fuel rod bow was evaluated with consideration given to both the hot channel power spike due to concave bowing away from the hot rod and the effect on DNBR of flow area reduction due to convex bowing toward the hot rod. These phenomena were evaluated separately since they are mutually exclusive. In the submittal dated January 13, 1977, AP&L summarized the methods and results of the rod bow analysis. This original rod bow analysis was performed with an as yet unapproved B&W model. Therefore, AP&L was requested to provide analyses with the NRC approved rod bow model. However, by letters of March 9, 1977 and March 17, 1977, AP&L was able to show sufficient available margin in the analyses in order to offset the difference between models without reducing any margins of safety.

The effect of rod bow on DNBR must be considered for both the variable pressure-temperature setpoint, quadrant tilt specifications, and the flux-flow trip. For the variable pressure-temperature setpoint and the quadrant tilt technical specifications, removal of the densification power spike and the flow area reduction penalties, as approved in reference 6, combine to provide adequate margin for the difference between the submitted and approved model without reducing any margins of safety. In the case of the flux-flow analysis, AP&L has proposed thermal margins from comparison of test to analytical assumptions for the reactor scram time, i.e., time from breaker trip to 3/4 rod insertion level. For this analysis, ANO-1 had previously used scram times which were related to Technical Specification values. However, testing resulted in scram times that were substantially lower. Thus, by decreasing the Technical Specification value by the time interval which corresponds to difference between the submitted and approved rod bow model, and without reducing any margins of safety AP&L has shown that the thermal analysis is equivalent to that with the approved rod bow model. All other Technical Specification setpoints were established with the NRC approved model and justified on that basis.

The reactor coolant flow rate was accurately measured during cycle 1 operation and a minimum measured value of 109.7% of the system design flow was determined. AP&L has proposed to take credit in the cycle 2 thermal-hydraulic analysis for the fact that the actual system flow is greater than the design flow rate, and has also included uncertainties and conservatisms in this analysis.

In the past, a 4.6% reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for ANO-1. This penalty is associated with the potential of a core internal vent valve being stuck open during normal operation. The core internal vent valves are incorporated into the design of the reactor internals to preclude potential vapor lock during a postulated cold-leg break Loss-of-Coolant Accident (LOCA). The NRC staff has concluded that by application of a surveillance program the vent valve flow penalty may be removed. The surveillance requirements demonstrate that the vent valves are not stuck open and that the vent valves operate freely.

AP&L's proposed surveillance program has been reviewed. The program differs from previously approved surveillance programs in that: (1) it tests on a force equivalent basis for full open position, whereas the NRC recommended program suggests a start to open and a full open pressure differential across the vent valves; and (2) the proposed force equivalent corresponds to a larger pressure differential than recommended. By letter of February 22, 1977, AP&L has shown that the force equivalent method is applicable. By letter of March 9, 1977, AP&L has also shown that not testing for the start to open case and the greater force equivalent has a negligible effect on the limiting LOCA, i.e., less than 3°F increase in the peak cladding temperature (PCT) for the limiting LOCA analysis, and PCT remains less than 2200°F. Therefore, the NRC staff concluded that AP&L has proposed a surveillance program that adequately meets the NRC staff's concerns and requirements, and the core internal vent valve penalty was properly eliminated. The ANO-1 Technical Specifications are being modified to add the new surveillance specification.

There are differences in the flow resistance between the Mark B-3 fuel assemblies of Batches 2 & 3 and the Mark B-4 fuel assemblies of Batch 4. The flow resistance for a Mark B-4 fuel assembly is slightly less than that for the Mark B-3 assemblies. These differences have been analyzed and from this analysis it was concluded that the Mark B-3 assemblies are limiting for the ANO-1 cycle 2 operation. This phenomenon also results in cross flow which has been calculated and demonstrated from previous operating experience to be of negligible effect.[1]

In summary, AP&L has proposed that: (1) a reactor coolant flow rate based on a conservative adjustment of the actual measured flow rather than the design flow be used; (2) the 4.6% core vent valve flow penalty be eliminated by establishment of an acceptable surveillance program; and (3) the DNBR fuel densification power spike removal, flow area reduction credit, and rod bow penalty be incorporated. Because of the analyses mentioned above, we have found the thermal-hydraulic analysis to be acceptable and the proposed Technical Specifications related to thermal hydraulic analysis also acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by AP&L demonstrates that the ANO-1 FSAR analyses conservatively bound the predicted condition for cycle 2 operation except for the items discussed below.

A. Loss-of-Coolant Flow

The analysis of this transient resulted in a setpoint reduction for the flux-flow-power imbalance trip. The overall reduction in trip setpoint resulted from a combination of credits as established in reference 6 and a penalty for rod bow power spike as discussed in the thermal hydraulics section of this report. The applicable analysis has been reviewed and found acceptable by the staff and the ANO-1 Technical Specifications are being modified to reflect the reduction in trip setpoint.

B. Loss-of-Coolant Accident (LOCA) Analysis

The previously applied W-3 Critical Heat Flux (CHF) correlation was replaced with the BAW-2 correlation. Both of these have been previously approved[13] for use in the LOCA analysis. The following modifications form the basis and substance of this change: (1) An extension downward from 2000 psia to 1750 psia for the applicable pressure range based on a review of rod bundle CHF data taken in the range of interest; and (2) A reduction in DNBR from 1.32, which represents a 99% confidence level that 95% of the rods will not experience DNB, to 1.30, which represents a 95% confidence level that 95% of the hot rods will not experience DNB. This is consistent with the Standard Review Plan[12] and industry practice. A revision to B&W's ECCS evaluation model was proposed[7] and has been approved by the NRC staff.[8] This change is to use a nucleate boiling heat transfer correlation during blowdown after critical heat flux (CHF) is first predicted. By letter dated February 17, 1977, AP&L submitted the approved generic B&W analysis[7] using the revised ECCS model.

The staff has reviewed these modifications as identified above and has concluded that they are in compliance with Appendix K of 10 CFR 50 and are acceptable for use in the ANO-1 analysis. This LOCA analysis submitted for the ANO-1 reload analysis meets the criteria of 10 CFR 50.46 and is acceptable on that basis.

The ECCS analyses submitted by the licensee (letter of July 9, 1975, as supplemented by letters dated August 8, August 22, October 15, and December 31, 1975, and the AP&L reload report of December 1, 1976, with its associated supplements) and reviewed by the NRC fulfilled the requirements of the Commission's December 27, 1974 Order for Modification of License and Appendix K of 10 CFR 50. The remaining exception is the completion of the ongoing NRC review of the ECCS electrical single failure criteria in response to the NRC letter of May 7, 1976. Based on findings of the ANO-1 licensing safety evaluation report dated June 6, 1973, no single failure has yet been identified which would require further modification to the technical specifications. Completion of this ongoing review is scheduled for June 1, 1977, and will be documented subsequently. Therefore, operation in the proposed manner does not endanger the public health and safety and is in compliance with the Commission's regulations.

The change in control rod position limits specifically mentioned in the ECCS-related Notice of Proposed Issuance of Amendment to License dated July 22, 1975, has been incorporated as part of this ECCS re-evaluation.

Startup Physics Tests

The proposed startup physics test program for ANO-1 has been reviewed. The program was discussed with AP&L for clarification of the number of measurements for critical boron concentration and moderator temperature coefficient. At least two of each of these measurements will be performed and the results compared with predictions. The acceptance criterion for the control rod reactivity worth measurements is being changed to require additional measurements if the initial acceptance criterion is not met.

The proposed startup physics test program with these clarifications and additions has been reviewed and found acceptable.

Technical Specifications

The proposed Technical Specifications changes for ANO-1 cycle 2 operation include:

1. incorporation of revised core protection limits in response to analyses mentioned above.

2. incorporation of new technical specification limiting conditions for operation and surveillance requirements regarding core vent valves.

3. changes to Technical Specification Bases to reflect the modifications of 1 and 2 above, and

4. modified operating limits related to ECCS.

Some modifications to the proposed Technical Specifications were necessary to meet NRC staff requirements. The staff finds that the proposed Technical Specifications, as modified, are acceptable and consistent with the information submitted by the licensee.

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.

Conclusions

Based on our review of the items identified as (1) through (4) in the introduction to this evaluation, and the considerations discussed in this evaluation, we have concluded that (1) because the items do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in safety margin, they do not involve a significant hazards consideration. We also have concluded, based on the considerations discussed in this evaluation, that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and the issuance of an amendment to the license will not be inimical to the common defense and security or to the health and safety of the public, and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Date:

References

1. "Arkansas Nuclear One - Unit 1, Cycle 2 - Reload Report," BAW-1433 November, 1976.

2. Morgan, C. A., and Kao, H. S., "TAFY-Fuel Pin Temperature and Glass Pressure Analysis," BAW-10044, May, 1972.

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

4. "Technical Report on Densification on Babcock & Wilcox Reactor Fuel," July 6, 1973.

5. "Arkansas Nuclear One, Unit 1 - Fuel Densification Report," BAW-1391, June, 1973.

6. Letter from D. L. Ziemann (NRC) to J. D. Phillips (AP&LCo) re: ANO-1 Rod Bow Penalty and Associated Credits, dated December 30, 1976.

7. Letter from K. E. Shurke (B&W) to S. A. Varga (NRC), Re: D. F. Ross December 2, 1976 letter to Shurke on B&W ECCS Evaluation Model, dated January 24, 1977.

8. Letter to D. B. Vassallo from D. F. Ross, Re: Topical Report Evaluation BAW-10104, ECCS Evaluation Model, Revised Nucleate Boiling Lockout Model, dated February 2, 1977.

9. SER on Oconee Nuclear Station, Units 1,2,&3, dated June 30, 1976, Amendment Nos. 27, 27 and 23 for License Nos. DPR-38, DPR-47 and DPR-55.

10. Letter from A. Schwencer (NRC) to J. F. Malloy (B&W), on January 29, 1975.

11. Memorandum from R. Lobel to D. F. Ross, "Present Status of B&W Power Spike Model," July 23, 1974.

12. Standard Review Plan, Section 4.4, pg. 4.4-2 and 4.4-3.

13. Letter dated April 15, 1976 from J. Stolz (NRC) to K. E. Shurke, on BAW-10000 A Topical Reporting of May, 1976.

14. Safety Evaluation Report By The Directorate of Licensing U. S. Atomic Energy Commission In The Matter of Arkansas Power & Light Company, Arkansas Nuclear One - Unit No. 1 Nuclear Power Plant, Pope County, Arkansas, Docket No. 50-313, June 6, 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.   to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised Technical Specifications for operation of 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 revised the Technical Specifications for the facility to authorize operation with: (1) revised core protection limits ~~to~~ in response to plant specific analysis for cycle 2, (2) modified fuel rod bow analyses, (3) the revised Babcock and Wilcox Company model for nucleate boiling heat transfer correlation during blowdown, (4) new limiting conditions for operation and surveillance requirements regarding core internal vent valves, and (5) modified operating limits based upon an evaluation of emergency core cooling system (ECCS) performance calculated in accordance with an acceptable ECCS evaluation model that conforms with the requirements of Appendix K of 10 CFR Part 50 of the Commission's Order for License Modification dated December 27, 1974, with the following exception. The Commission's analyses of the electrical single failure criterion is still under consideration and will be the subject of a separate review. The incorporation of the modified operating limits relating to ECCS supersedes the restrictions imposed by the Commission's Order dated December 27, 1974.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (5) above was published in the FEDERAL REGISTER on July 30, 1975 (40 F.R. 31996). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action on item (5) above. Prior public notice of items (1) through (4) above was not required since these actions do not involve a significant hazards consideration.

The Commission has determined that the issuance of the 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.