ARKANSAS 50-313

PROPOSED TECHNICAL SPECIFICATIONS CHANGE

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## 2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

50-313

### 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 is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is 374,880 gpm, which is 106.5% of the design flow rate for four operating reactor coolant pumps. 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.

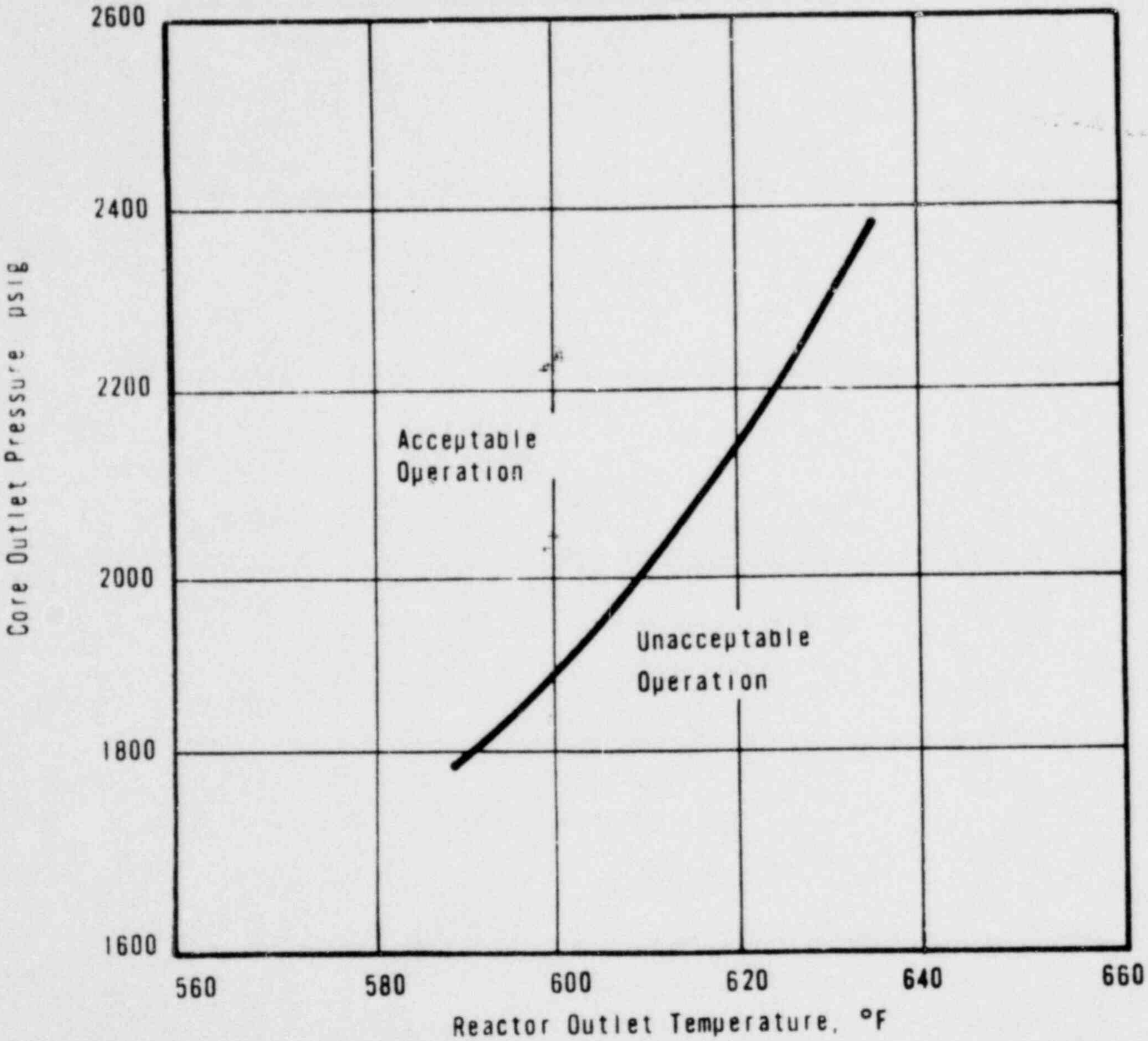
The maximum thermal power for three pump operation is 86.4 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.07 = 79.9 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

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

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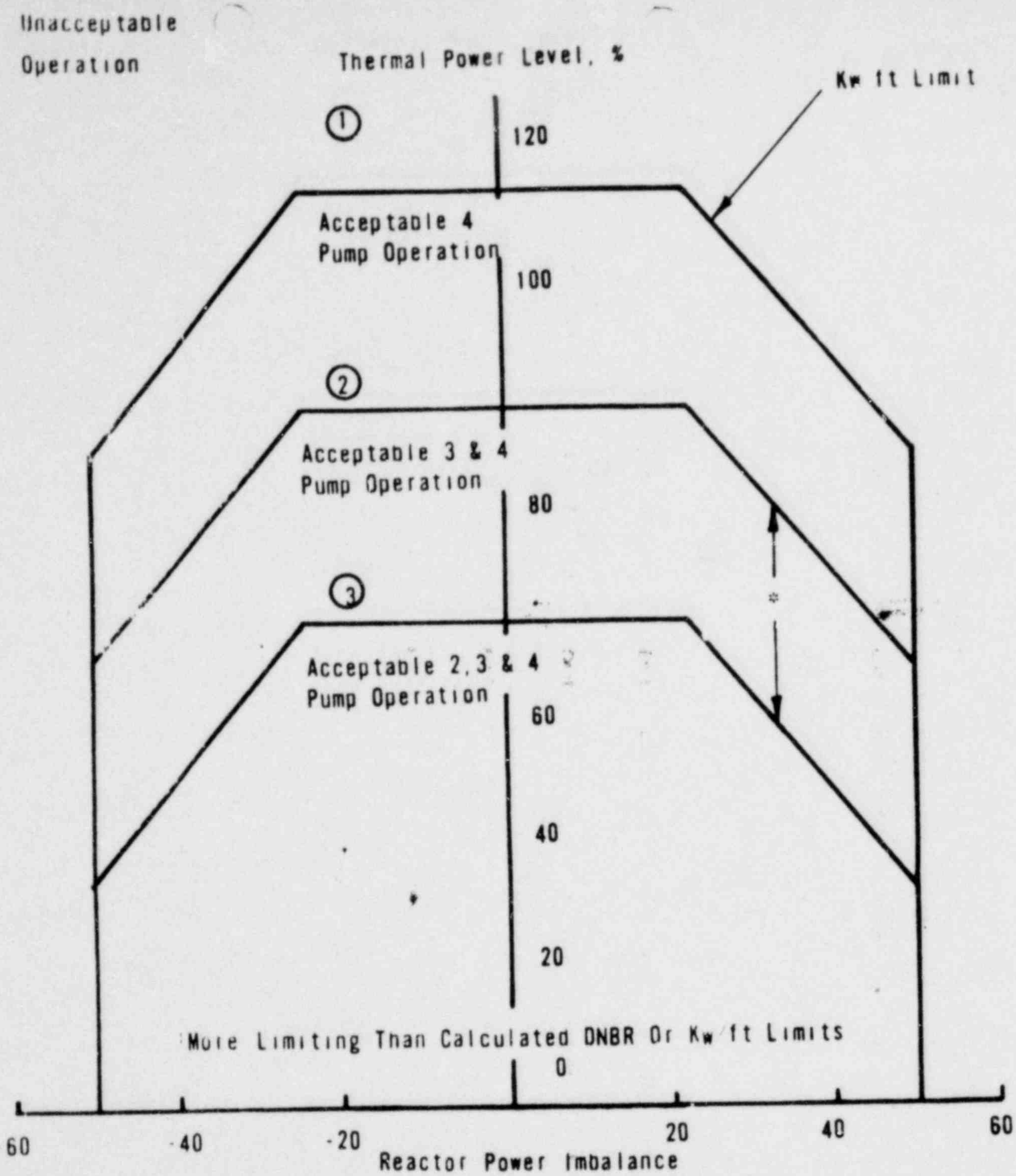
ARKANSAS NUCLEAR ONE-UNIT 1
CYCLE 2

CORE PROTECTION SAFETY LIMIT
Figure 2.1-1




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

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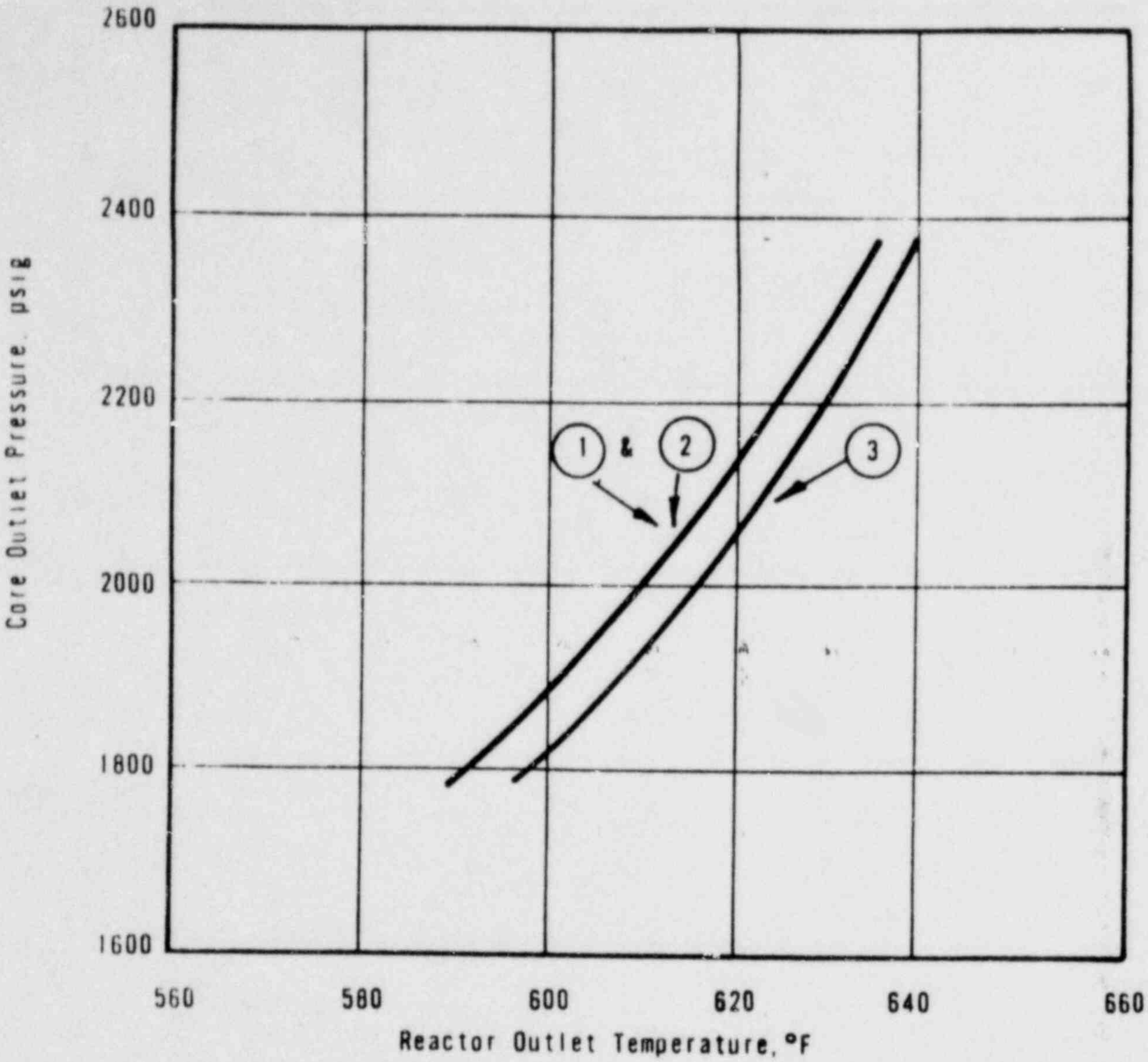
CORE PROTECTION SAFETY LIMITS
Figure 2.1-2




REACTOR COOLANT FLOW

| CURVE | GPM | POWER | PUMPS OPERATING (TYPE OF LIMIT) |
|---|---|---|---|
| 1 | 374,880 (100%)* | 112% | FOUR PUMPS (DNBR LIMIT) |
| 2 | 280,035 (74.7%) | 86.4% | THREE PUMPS (DNBR LIMIT) |
| 3 | 184,441 (49.2%) | 59.1% | ONE PUMP IN EACH LOOP (QUALITY LIMIT) |

*106.5% OF CYCLE 1 DESIGN FLOW

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CYCLE 2

CORE PROTECTION SAFETY LIMITS
Figure 2.1-3

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 79.9 percent and reactor flow rate is 74.7 percent or flow rate is 70.1 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 52.4 percent and reactor flow rate is 49.0 percent or flow rate is 45.8 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 the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio 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 1.07 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 pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit 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 ($11.72\ T_{out}$ $-5114.7$) 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 ($11.72\ T_{out}$ $-5154.7$).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F, 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 620 F.

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.

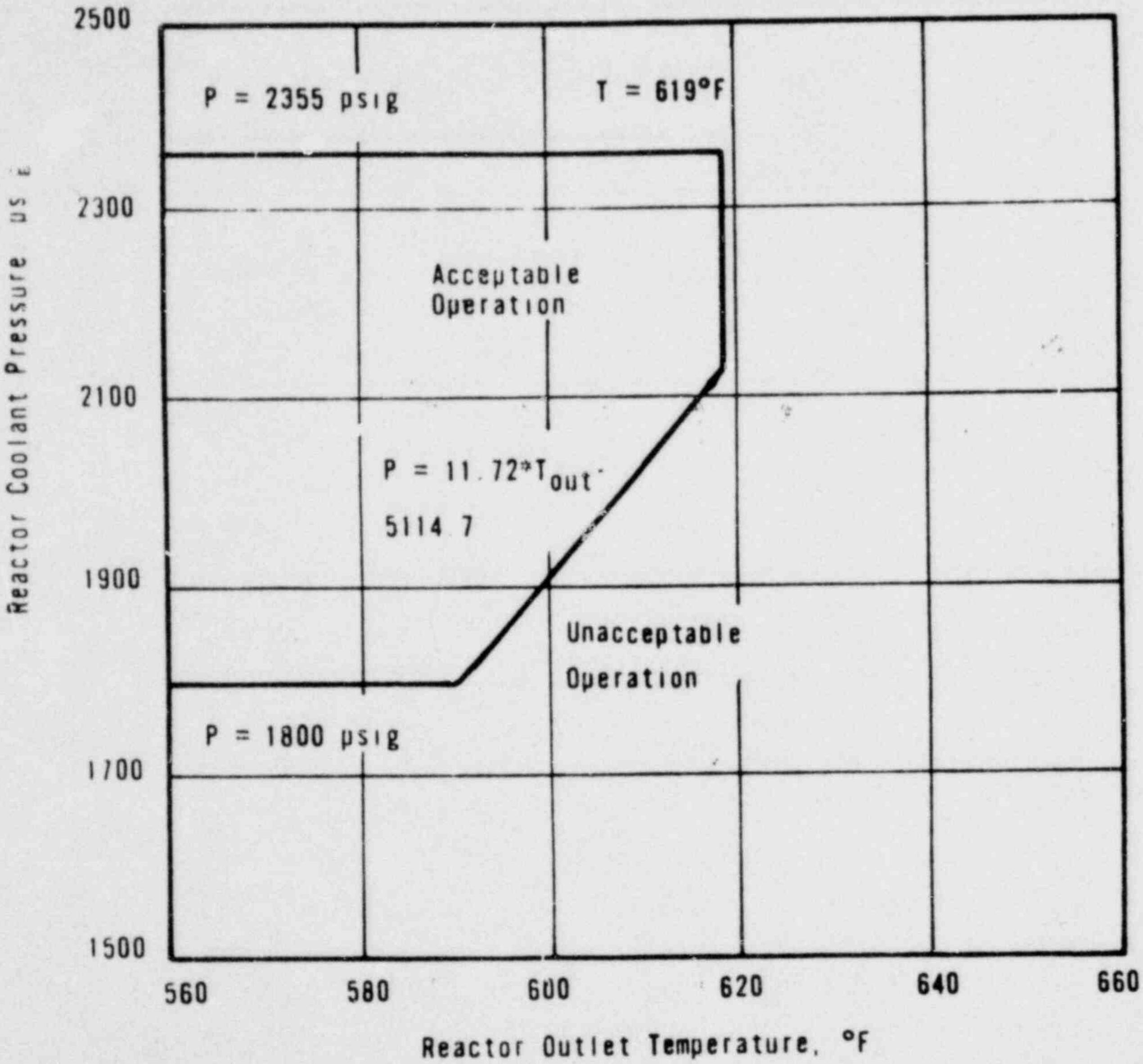
F. Shutdown bypass

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 ≤ 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. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

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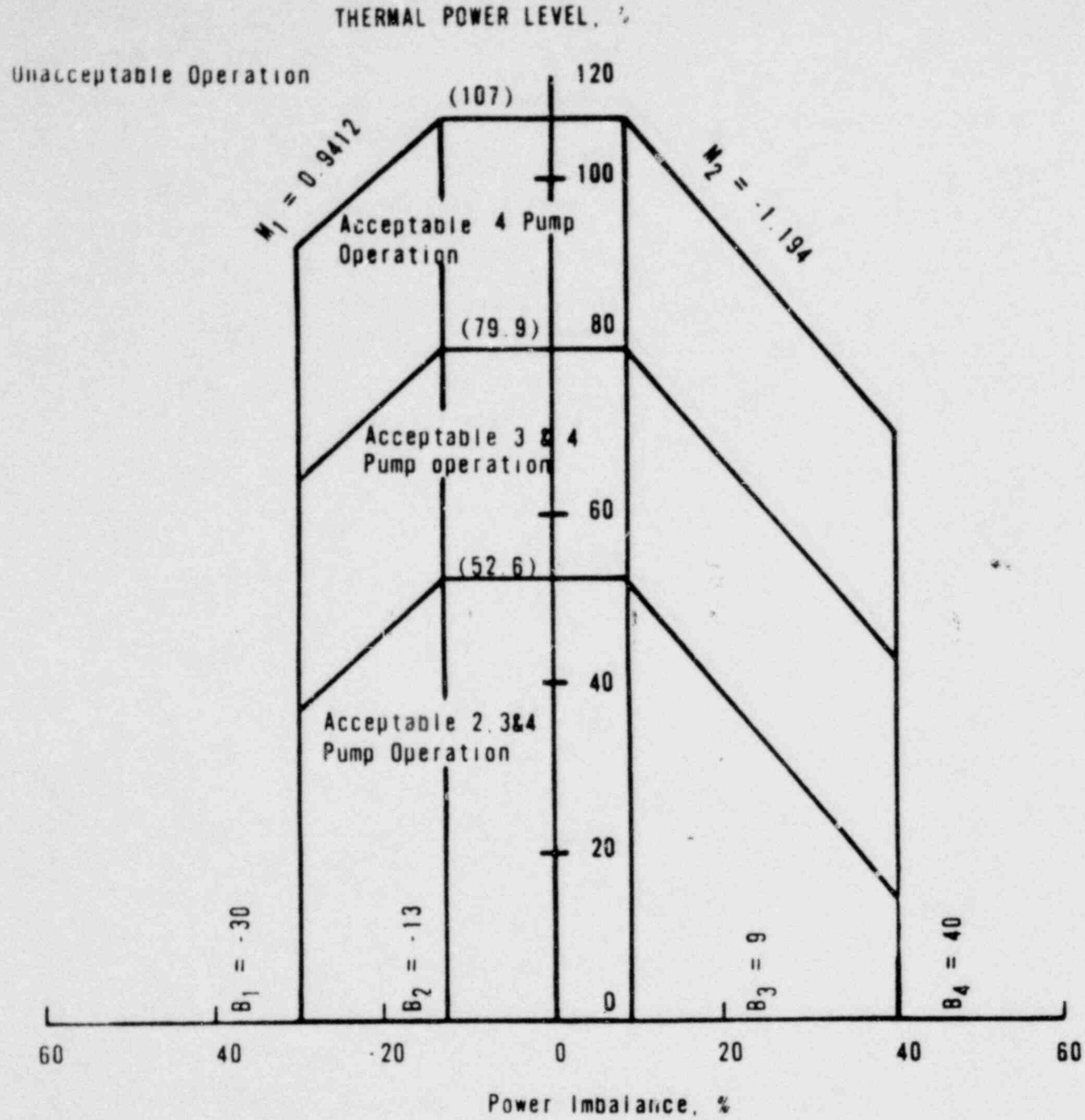
PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SET POINT

Figure 2.3-1

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CYCLE 2

PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
Figure 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 |
|---|---|---|---|---|
| uclear power, % of ated, max | 105.5 | 105.5 | 105.5 | 5.0[3] |
| uclear power based on low[2] 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 |
| uclear power based on ump monitors, % of ated, max [4] | NA | NA | 55% | Bypassed |
| igh reactor coolant ystem pressure, psig, ax | 2355 | 2355 | 2355 | 1720[3] |
| ow reactor coolant sys- em pressure, psig, min | 1800 | 1800 | 1800 | Bypassed |
| ariable low reactor oolant system pressure, sig, min | $(11.72\ T_{out}-5114.7)^{(1)}$ | $(11.72\ T_{out}-5114.7)^{(1)}$ | $(11.72\ T_{out}-5114.7)^{(1)}$ | Bypassed |
| eactor coolant temp, , max | 619 | 619 | 619 | 619 |
| igh reactor building ressure, 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. 2

## 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 $\pm$ 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.

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. 6

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% if power of 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% $\pm$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 3.52-3A, 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.

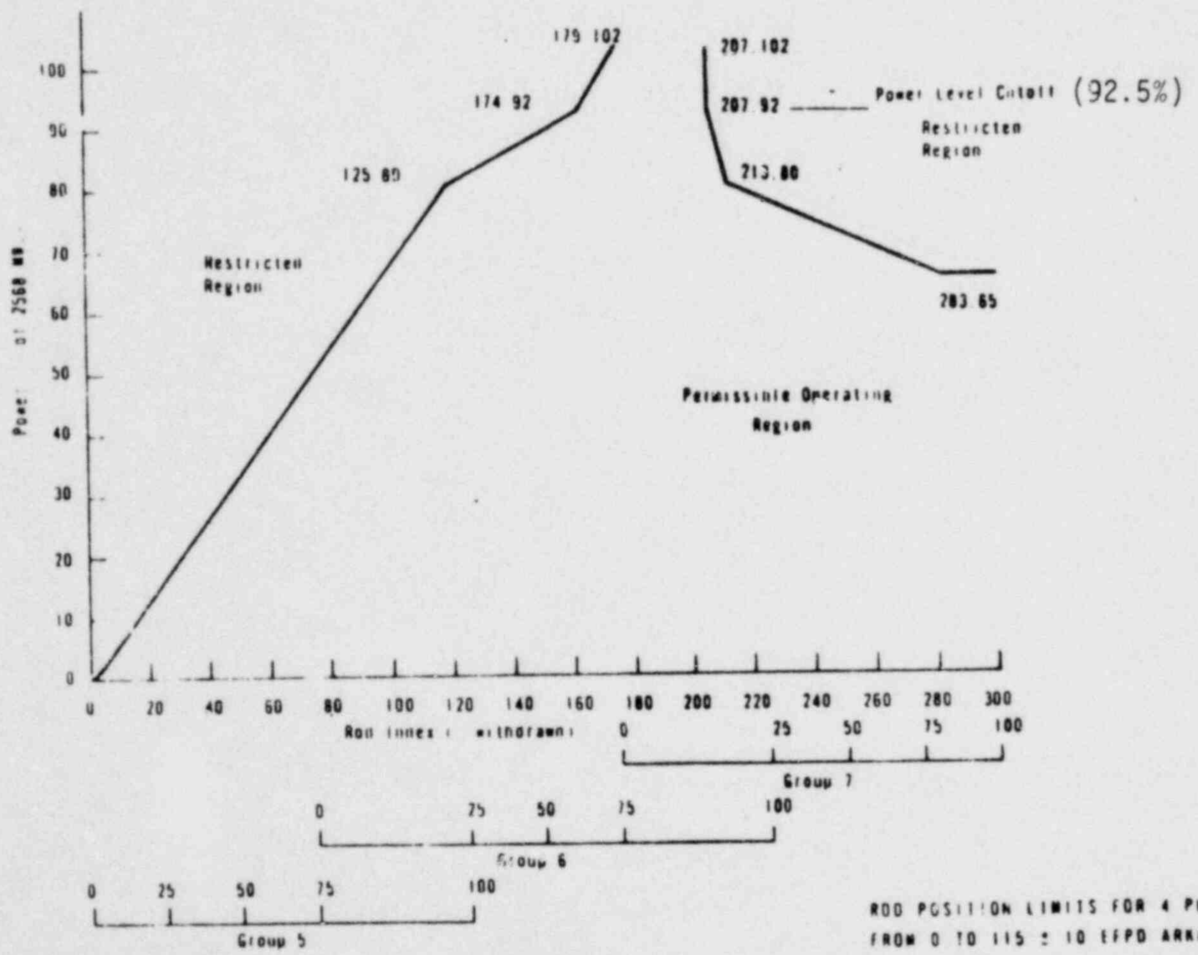
Bases

The power-imbalance envelopes defined in Figures 3.5.2-3A, 3.5.2-3B and 3.5.2-3C is 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

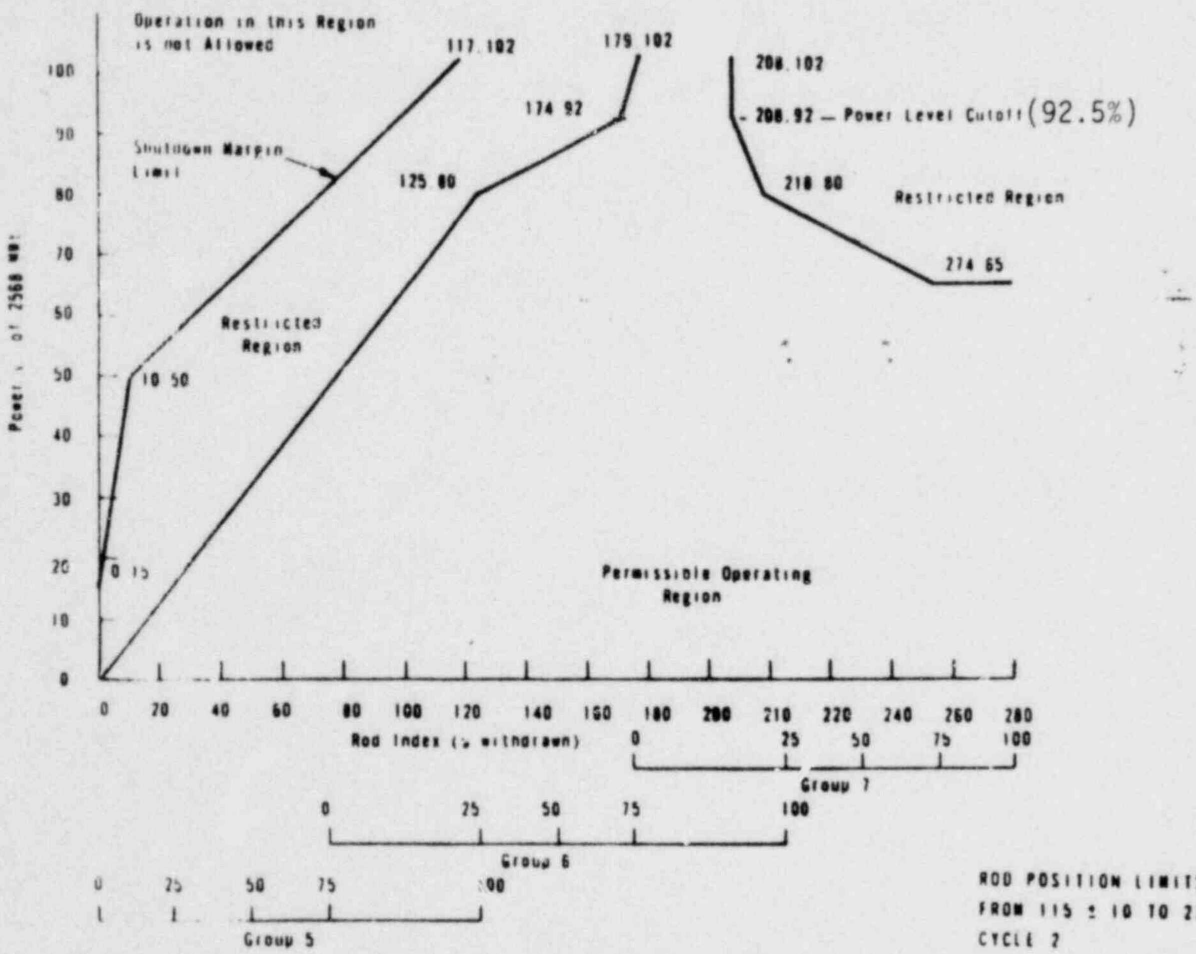
The 25 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

*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




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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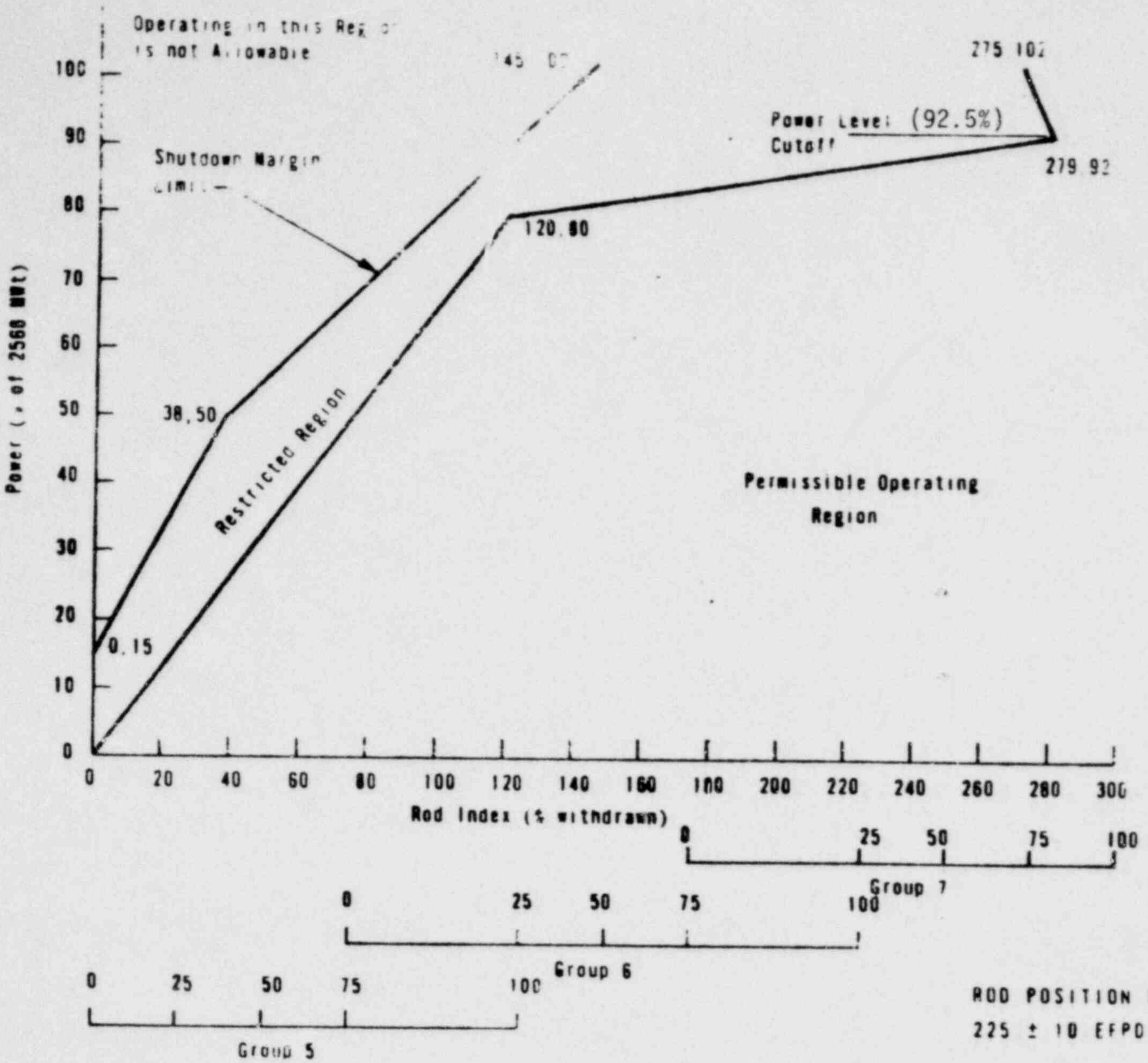
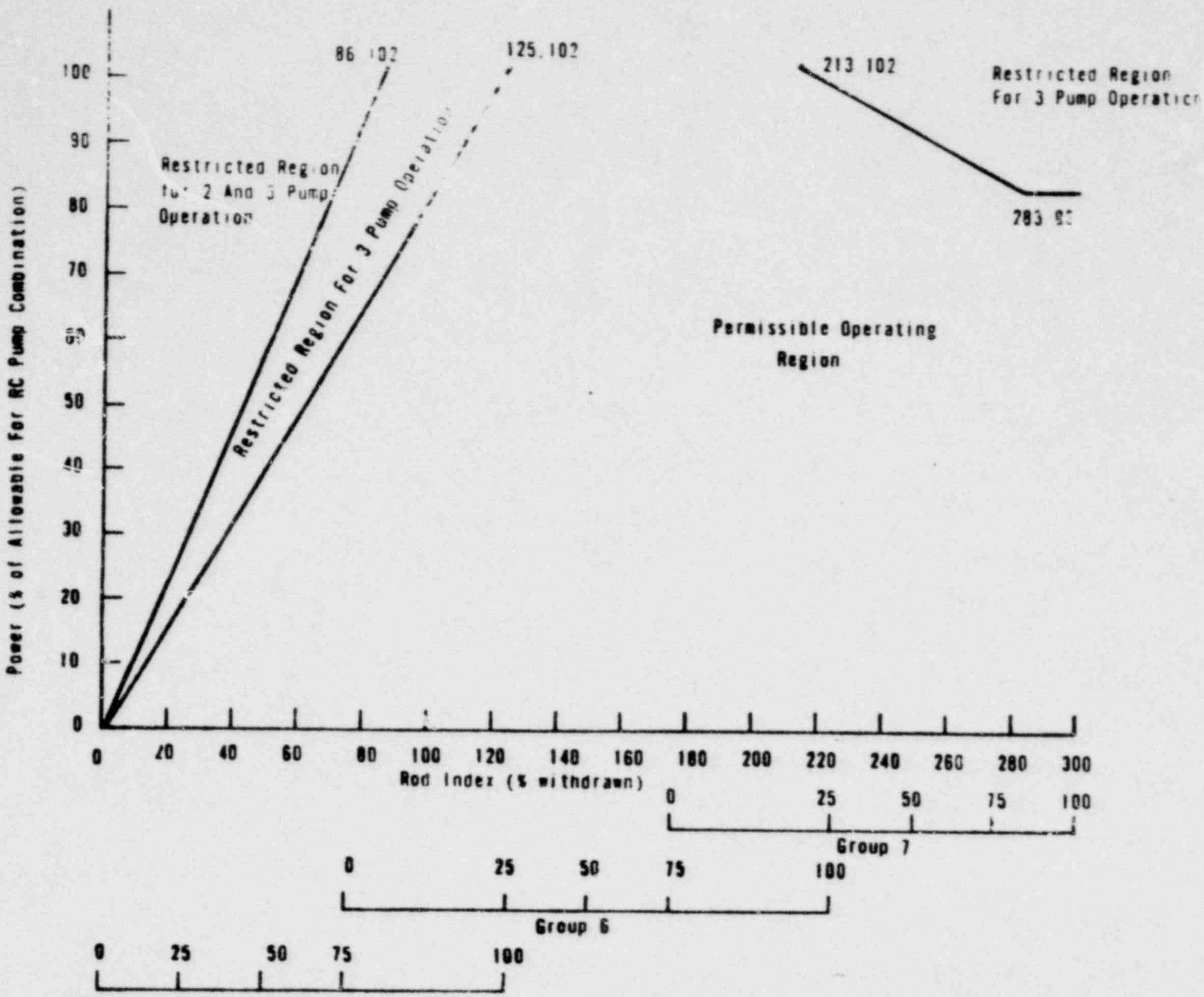



ROD POSITION LIMITS FOR 4 PUMP OPERATION AFTER 225 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-1C




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

Figure 3 5 2 2A

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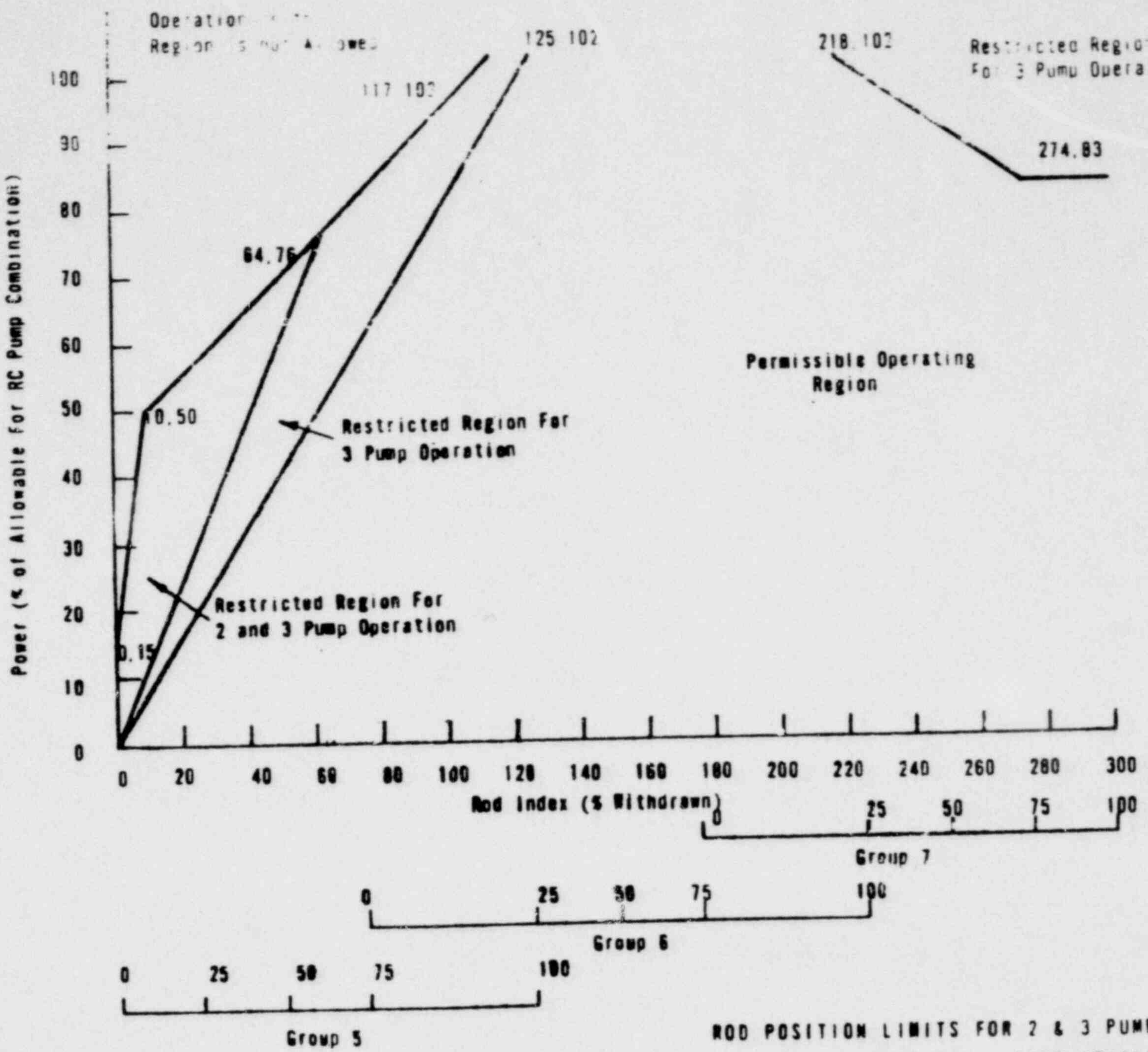
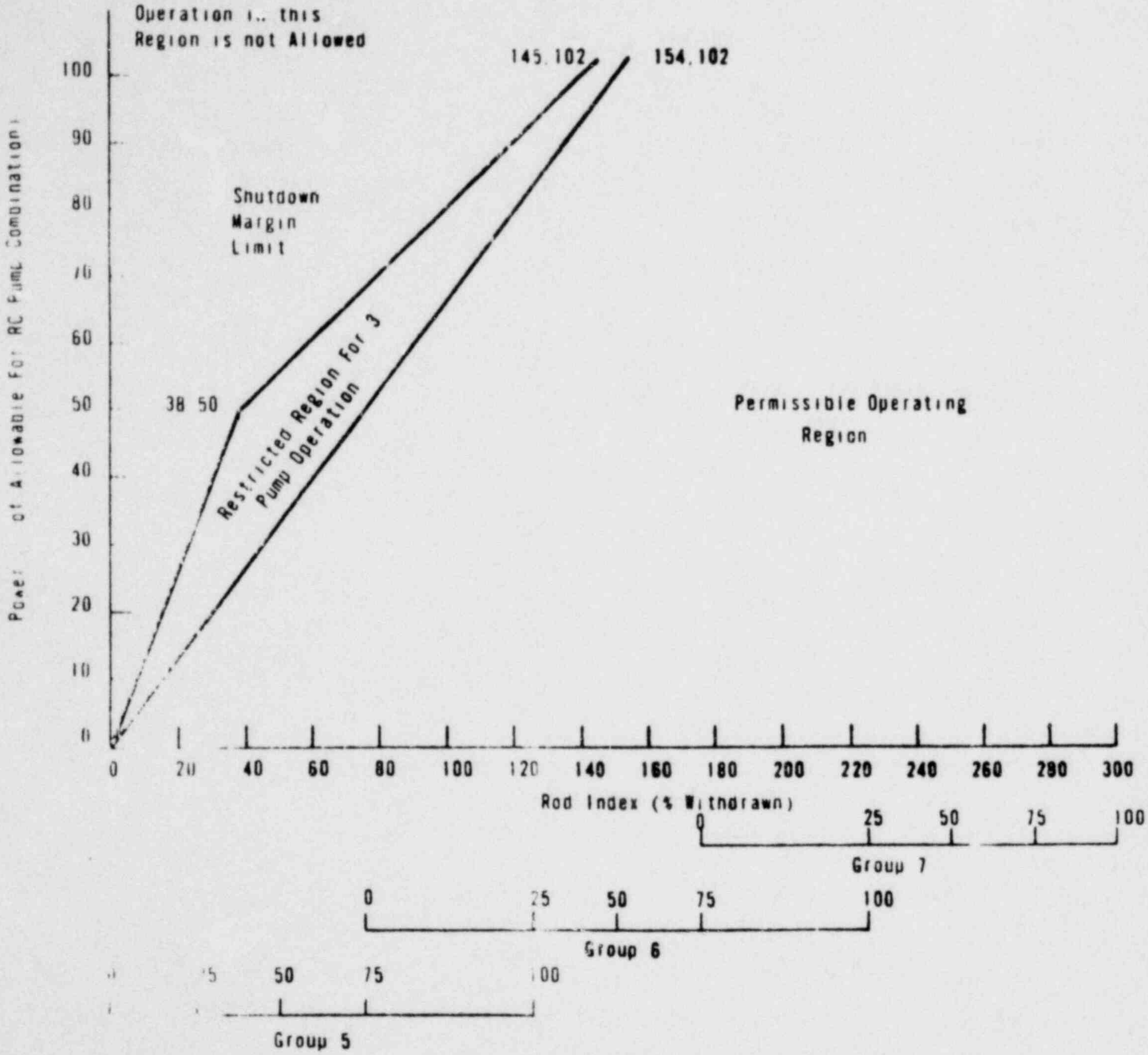
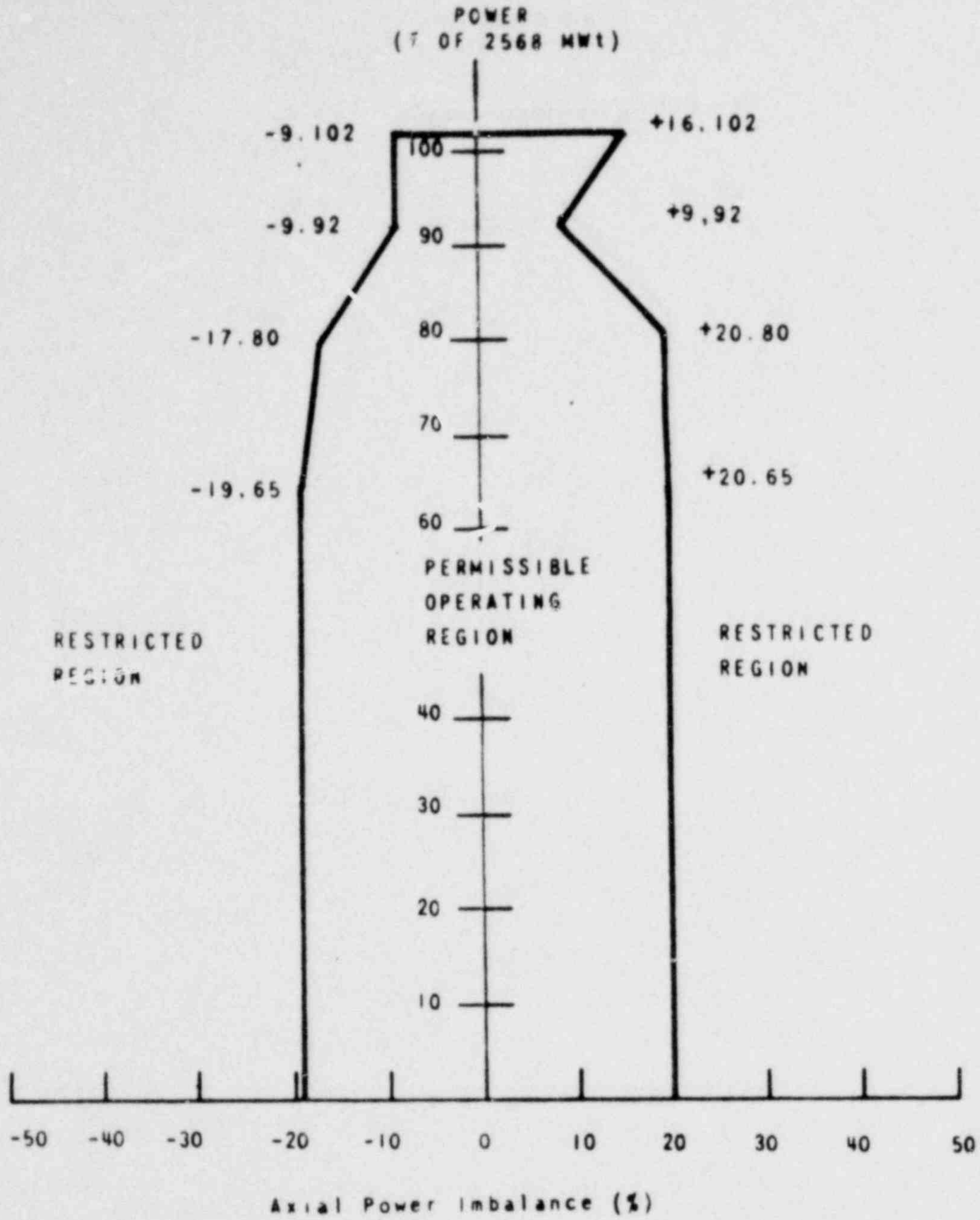



ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION
FROM 115 ± 10 TO 225 ± EFPD ARKANSAS CYCLE 2
Figure 3.5.2-2B



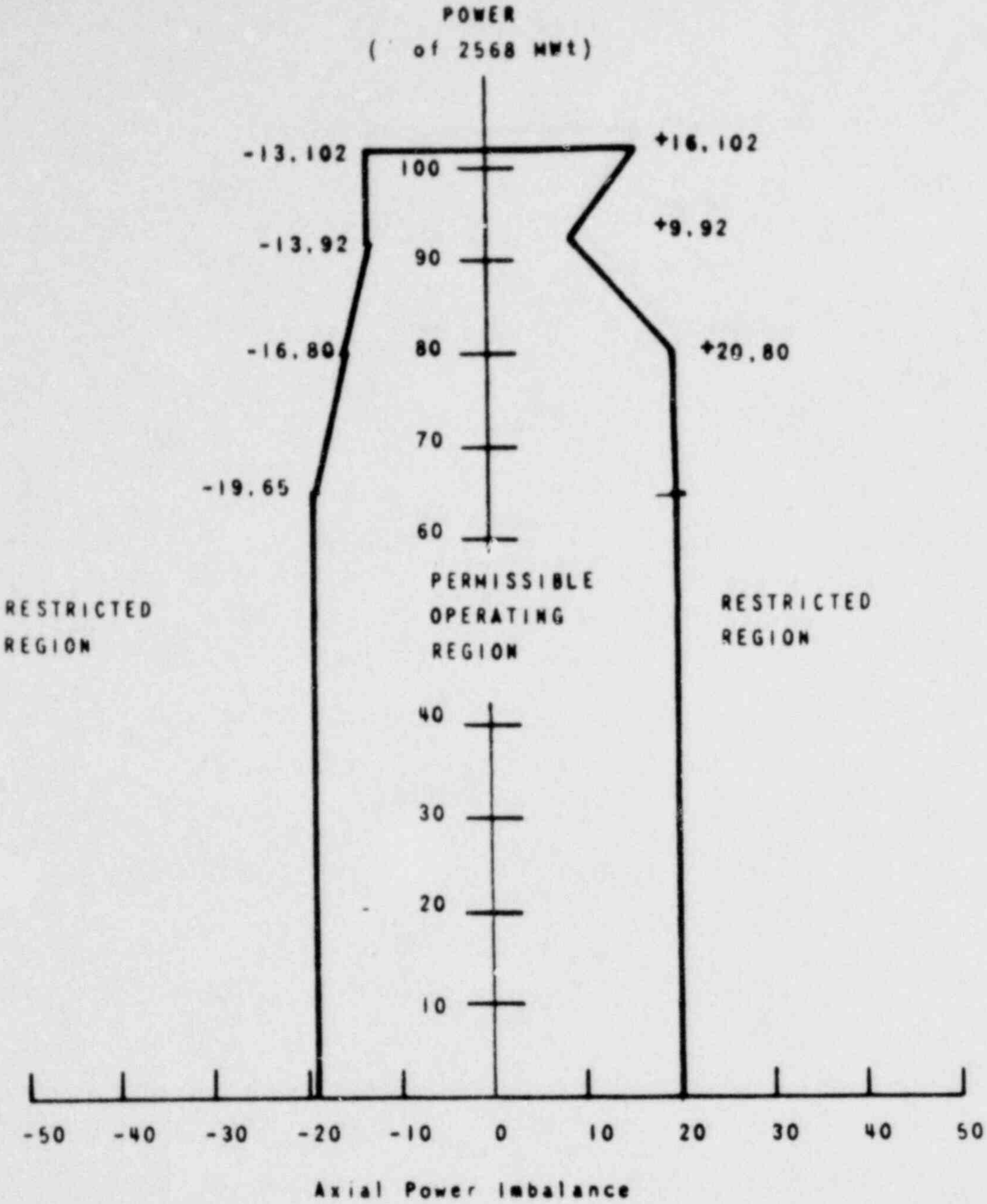

**ROD POSITION LIMITS FOR 2 & 3 PUMP OPERATION AFTER 225 ± 10 EFPD ARKANSAS CYCLE 2**

Figure 3.5.2-2C



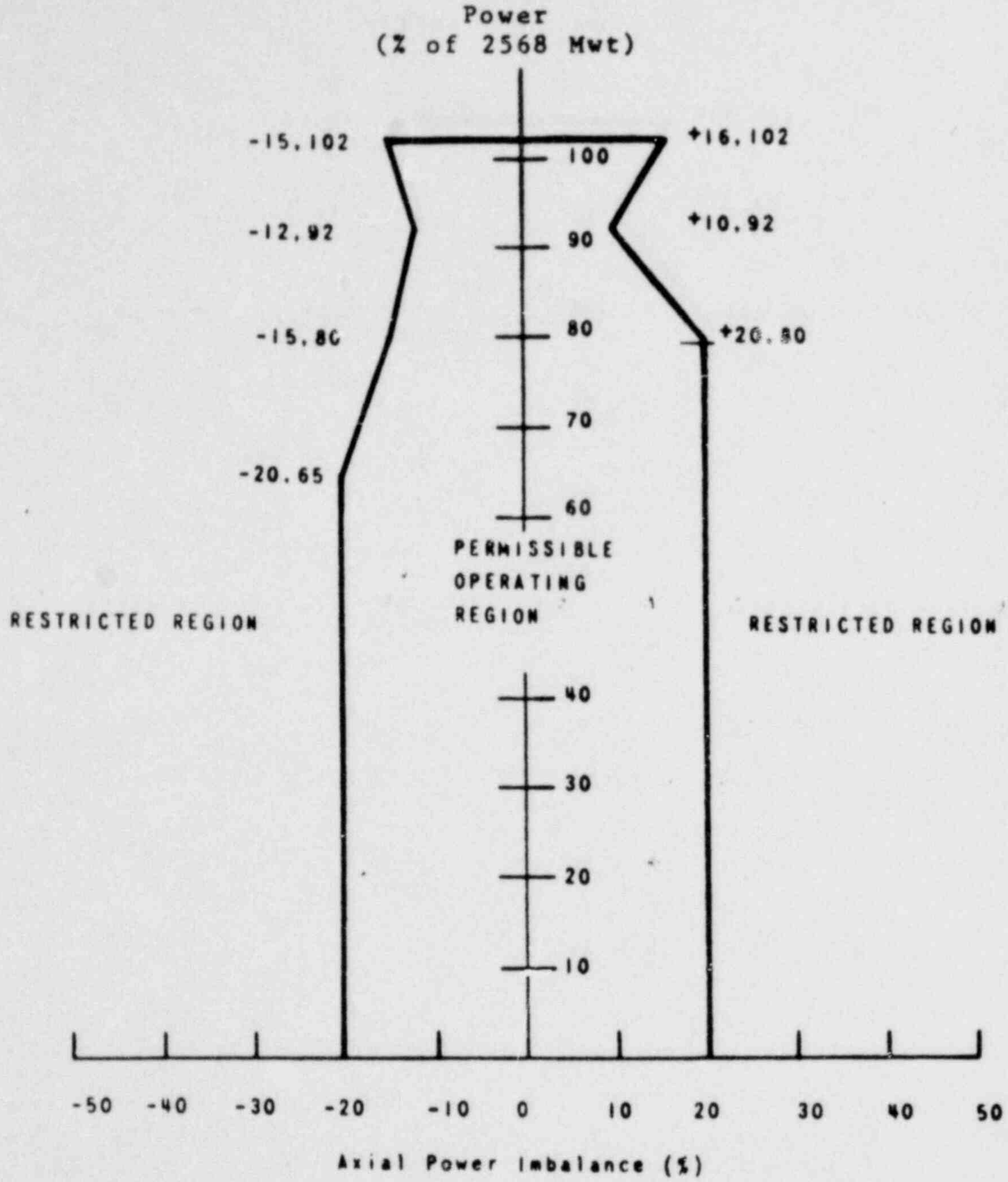

OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 115 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-3A



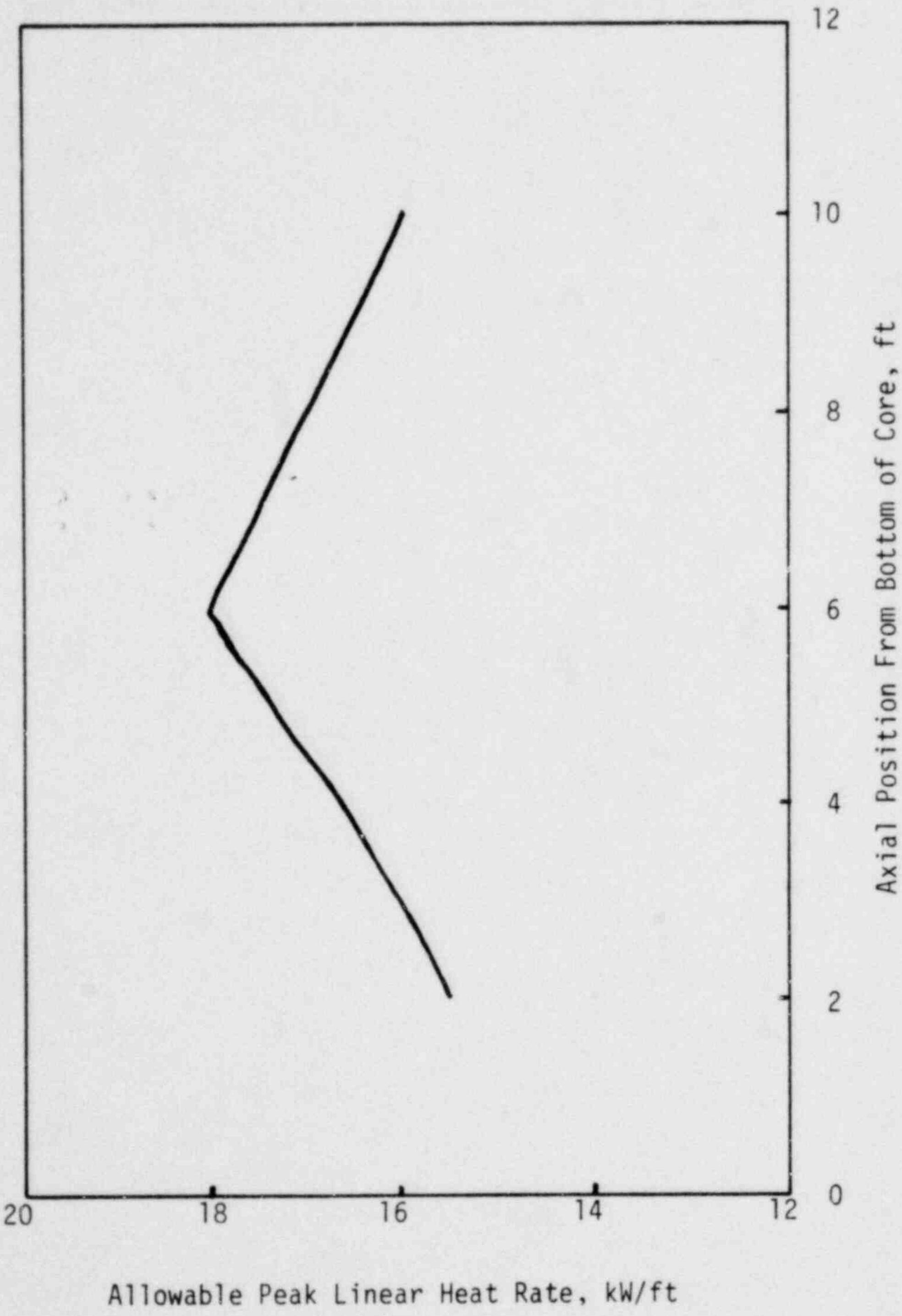

OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 115 ± 10 TO 225 ± 10 EFPD ARKANSAS CYCLE 2

Figure 3.5.2-3C




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

Figure 3.5.2-3C




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

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:<br>a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities.<br>b. Verifying that the valve is not stuck in an open position, and<br>c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs (applied vertically upward). | Each refueling shutdown. |

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