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

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

RANCHO SECO NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-54

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

   A. The application for amendment by Sacramento Municipal Utility District (the licensee) dated December 17, 1984, as supplemented by letters dated March 14, 1985, and April 9, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-54 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 4, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-54

DOCKET NO. 50-312

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

Remove	Insert
xi	xi
xii	xii
2.1-1	2-1
2-2	2-2
2-3	2-3
Figure 2.1-2	Figure 2.1-2
Figure 2.1-3	Figure 2.1-3
2-7	2-7
Figure 2.3-2	Figure 2.3-2
3-17	3-17
3-18	3-18
3-33a	3-33a
3-33b	3-33b
Figure 3.5.2-1	Figure 3.5.2-1
Figure 3.5.2-2	Figure 3.5.2-2
Figure 3.5.2-3	Figure 3.5.2-3
Figure 3.5.2-4	Figure 3.5.2-4
Figure 3.5.2-5	Figure 3.5.2-5
Figure 3.5.2-6	Figure 3.5.2-6
--	Figure 3.5.2-7
--	Figure 3.5.2-8
Figure 3.5.2-9	Figure 3.5.2-9
Figure 3.5.2-10	--
Figure 3.5.2-11	--
Figure 3.5.2-12	--

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

LIST OF FIGURES

LIST OF FIGURES (Continued)

Figure

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 within the restricted region 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 (solid line) 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

The safety limits presented have been generated using the BAW-2 and BWC CHF correlations (1, 4) and the actual measured flow rate (2). The flow rate utilized is 104.9 percent of the design flow (369,600 gpm) based on four-pump operation (2, 3).

To maintain the integrity of the fuel cladding and to prevent fission product release to the primary coolant system, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling region 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 region 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 CHF correlation (1, 4). The BAW-2 and BWC correlations have 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 ratio 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 1.30 (BAW-2) or 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) 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 setpoints 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 DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 104.9 percent of 369,600 gpm). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

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 rod bowing.

1. The combinations of the radial peak, axial peak and position of the axial peak that yields a DNBR no less than the CHF correlation limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 KW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases 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.

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 the CHF correlation limit or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation.

The maximum permitted thermal power for three-pump operation depicted in Figure 2.1-2 is 87.8 percent due to a power level trip produced by the flux-flow ratio 1.06 times 74.4 percent design flow = 78.86 percent power plus the absolute value of the maximum calibration and instrumentation error. The maximum thermal power for other coolant pump conditions is produced in a similar manner. The actual maximum power levels are calculated by the RPS and will be directly proportional to the actual flow during partial pump operation.

REFERENCES

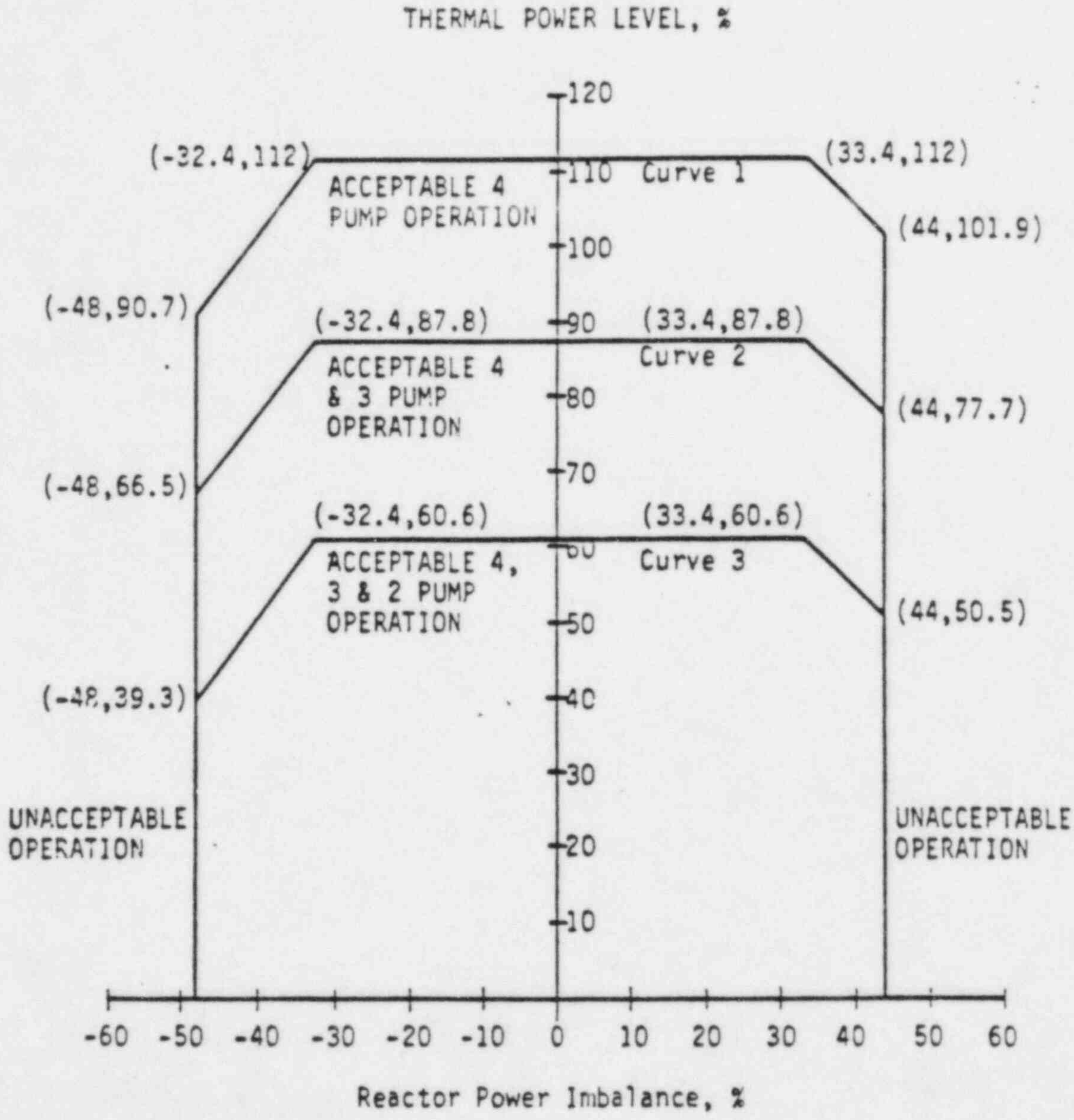
(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May 1976.

(2) Rancho Seco Unit 1, Cycle 2 Reload Report, BAW-1460, June 1977.

(3) Rancho Seco Unit 1, Cycle 3 Reload Report, BAW-1499, September 1978.

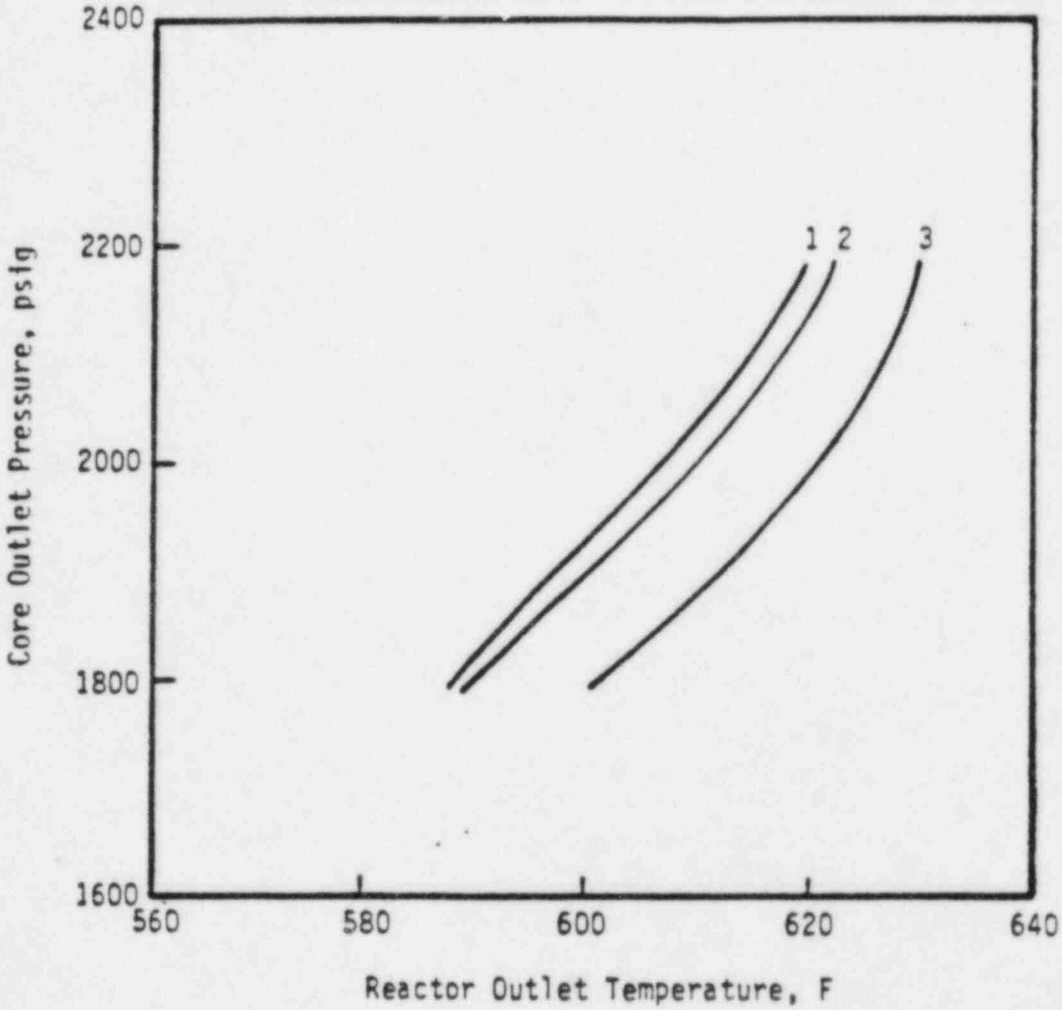
(4) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHR Data With the BWC Correlation, BAW-10143P, Part 2, Babcock and Wilcox, Lynchburg, Virginia, August 1981.

Figure 2.1-2 Core Protection Safety Limits, Reactor Power Imbalance, Rancho Seco 1, Cycle 7



Curve	Reactor Coolant Flow, % Design
1	104.9
2	78.0
3	50.9

Figure 2.1-3 Core Protective Safety Bases, Rancho Seco 1, Cycle 7



Curve	Reactor coolant flow, % design	Power, %	Pumps operating (type of limit)
1	104.9	112	Four (DNBR limit)
2	78.0	87.8	Three (DNBR limit)
3	50.9	60.6	One in each loop (quality limit)

B. Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below the CHF correlation limit by tripping the reactor due to (a) the 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. The pump monitors also restrict the power level to 55 percent for one reactor coolant pump operation in each loop.

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 (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient (1) and minimize the challenges to the EMOV and code safeties.

The low pressure (1900 psig) and variable low pressure ($12.96\ T_{out} - 5834$) trip set point shown in figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to the CHF correlation limit 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 ($12.96\ T_{out} - 5884$).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618 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.

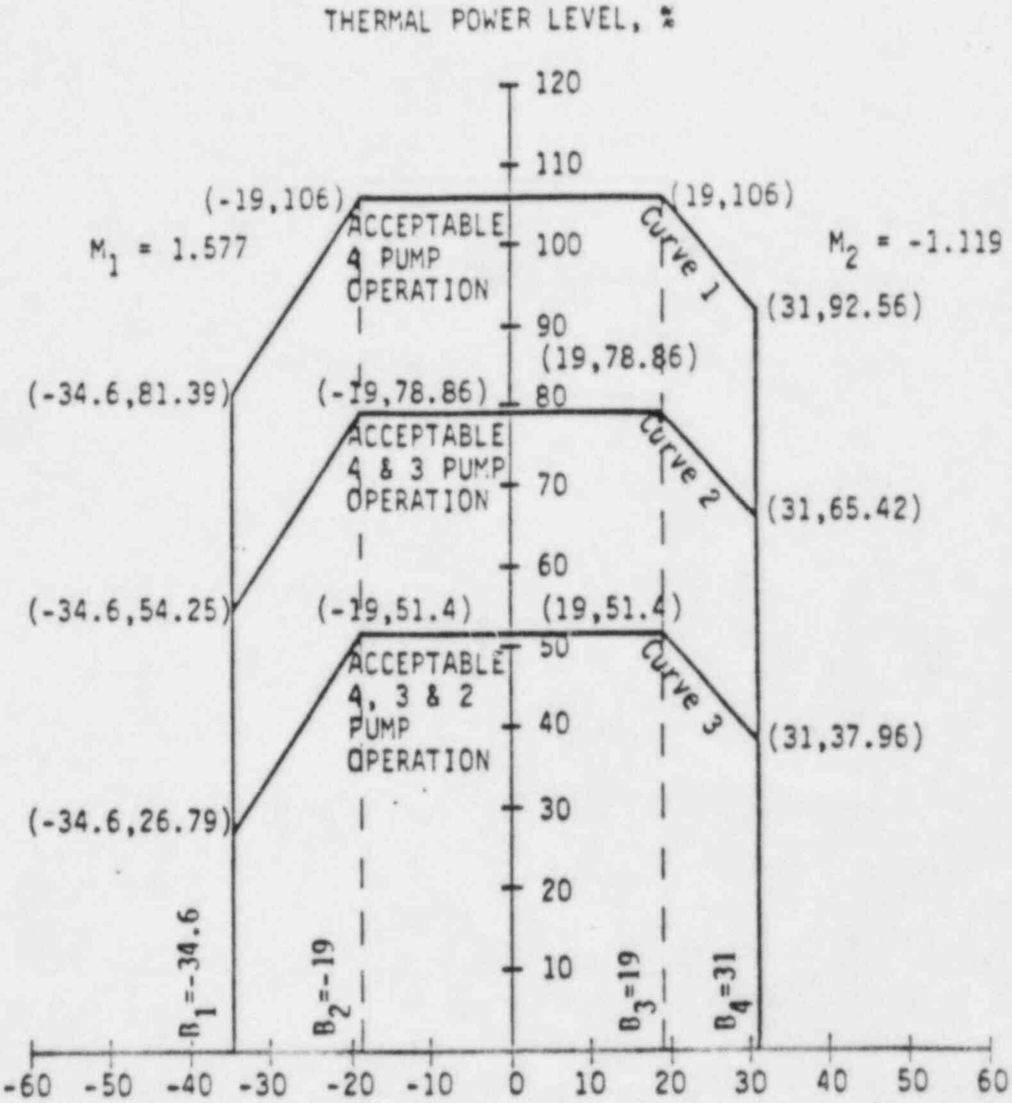
E. Reactor Building pressure

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

Figure 2.3-2 Protective System Maximum Allowable Setpoints, Reactor Power Imbalance, Rancho Seco 1, Cycle 7



Curve	Reactor Coolant Flow, % Design
1	104.9
2	78.0
3	50.9

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the operational status of high pressure injection and chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not remain critical unless the following conditions are met:

3.2.1 Two pumps capable of supplying high pressure injection are operable (also see Specification 3.3.2).

3.2.2 The borated water storage tank and its flow path to the reactor for high pressure injection are operable.

3.2.3 A source of concentrated boric acid solution in addition to the borated water storage tank is available and operable. This requirement is fulfilled by the concentrated boric acid storage tank. This tank shall contain at least the equivalent of 10,000 gallons of 7,100 ppm boron. System piping and valves necessary to establish a flow path for high pressure injection shall also be operable and shall have at least the same temperature as the boric acid storage tank. One associated boric acid pump is operable. The concentrated boric acid storage tank water shall not be less than 70F, and at least one channel of heat tracing shall be operable for this tank's associated piping. The concentrated boric acid storage tank boron concentration shall not exceed 8,500 ppm boron.

Bases

The makeup and purification system and chemical addition systems provide control of the reactor coolant system boron concentration.[1] This is normally accomplished by using either the makeup pump or one of the two high pressure injection pumps in series with a boric acid pump associated with the concentrated boric acid storage tank. The alternate method of boration will be the use of the makeup or high pressure injection pumps taking suction directly from the borated water storage tank.[2]

The quantity of boric acid in storage from either of the two above-mentioned sources is sufficient to borate the reactor coolant system to a 1 percent subcritical margin in the cold condition (70F) at the worst time in core life with a stuck control rod assembly. The maximum required is the equivalent of 9586 gallons of 7100 ppm boron. This requirement is satisfied by requiring a minimum volume of 10,000 gallons of 7100 ppm in the concentrated borated acid

storage tank during critical operations. The minimum volume for the borated water storage tank (390,000 gallons of 1800 ppm boron), as specified in section 3.3, is based on refueling volume requirements and easily satisfies the cold shutdown requirement. The specification assures that the two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The primary method of adding boron to the primary system is to pump the concentrated boric acid solution (7100 ppm boron, minimum) into the makeup tank using the 50 gpm boric acid pumps. Using only one of the two boric acid pumps, the required volume of boric acid can be injected in less than 3.5 hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the high pressure injection pumps.

Concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to ensure that a flow of boric acid is available when needed, this tank and its associated piping will be kept above 70F (30F above the crystallization temperature for the concentration present). Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures ensure boric acid solubility. The value of 70F is significantly above the crystallization temperature for a solution containing 12,200 ppm boron.

REFERENCES

1 FSAR subsections 9.2 and 9.3.

2 FSAR Figure 6.2-1.

3 Technical Specification 3.3.

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 envelope defined by Figures 3.5.2-7 through 3.5.2-9. If the imbalance is not within the envelope defined by Figures 3.5.2-7 through 3.5.2-9, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two 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 or his designated representative.

Bases

The power-imbalance envelope defined in Figures 3.5.2-7 through 3.5.2-9 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria.[3] Corrective measures will be taken should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundry. 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.**

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

The conservative application of the above peaking augmentation factors compensates for the potential peaking penalty due to Fuel rod bow.

The 25% ± 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:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping group)

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

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position.[1] The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis of hypothetical rod ejection accident.[2] A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than an 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6 and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

The Quadrant Power Tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 7.36%. The limits in Specification 3.5.2.4 are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The Quadrant Tilt and axial imbalance monitoring in Specifications 3.5.2.4F and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specifications 3.5.2.5.D.(1) and 3.5.2.5.D.(2) to prevent excessive power peaking by transient xenon. The xenon reactivity must either be beyond the "undershoot" region and asymptotically approaching its equilibrium value at rated power or the reactor must be operated in the range of 87% to 92% of the maximum allowable power for a period exceeding two hours in the soluble poison control mode so that the transient peak is burned out at a lower power level.

REFERENCES

(1) FSAR, Section 3.2.2.1.2
(2) FSAR, Section 14.2.2.4
(3) BAW-1850, October 1984, page 7-5

Figure 3.5.2-1 Rod Index Vs Power Level for Four-Pump Operation, 0 to 40 EFPD
-- Rancho Seco 1, Cycle 7

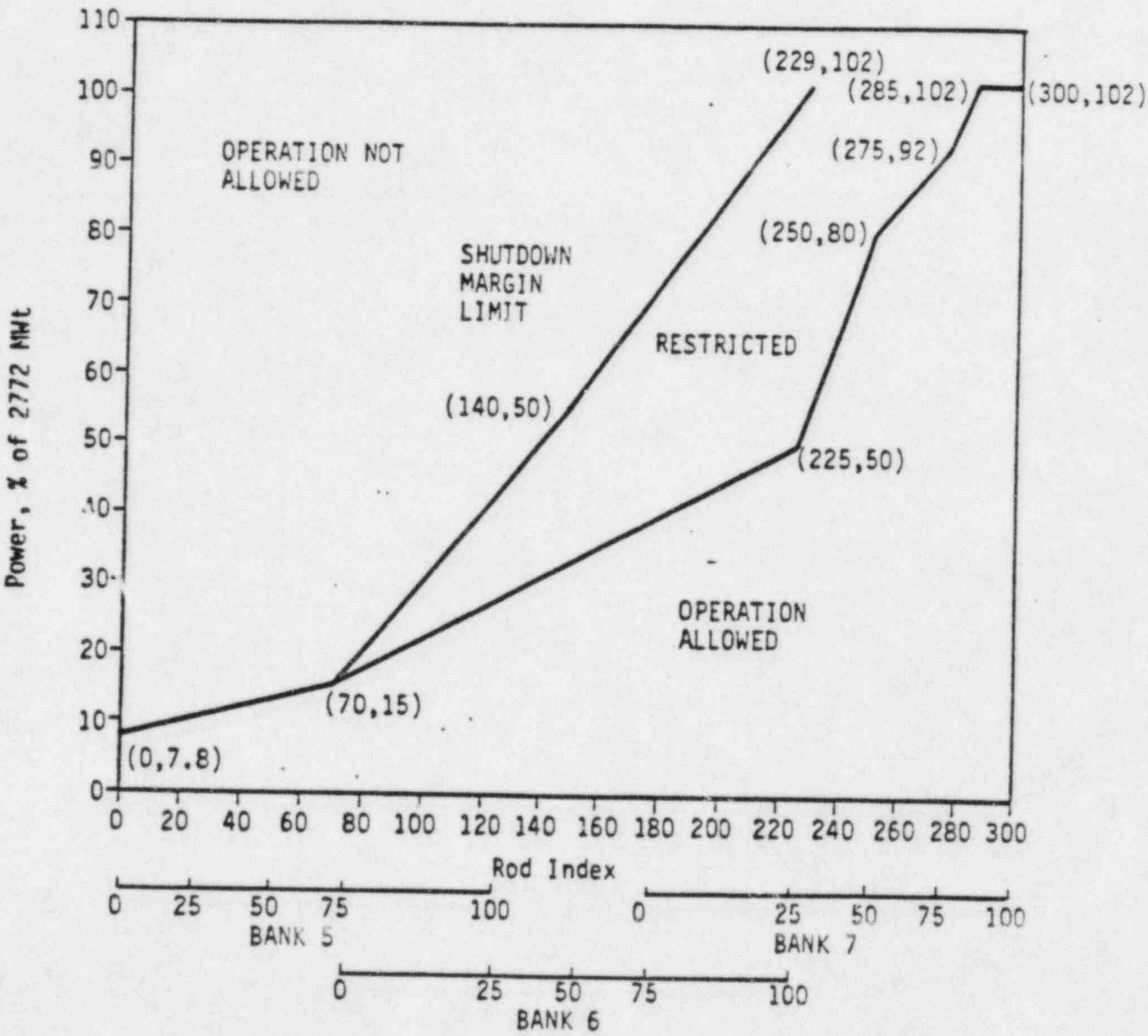


Figure 3.5.2-2 Rod Index Vs Power Level for Four-Pump Operation After 30 EFPD -- Rancho Seco 1, Cycle 7

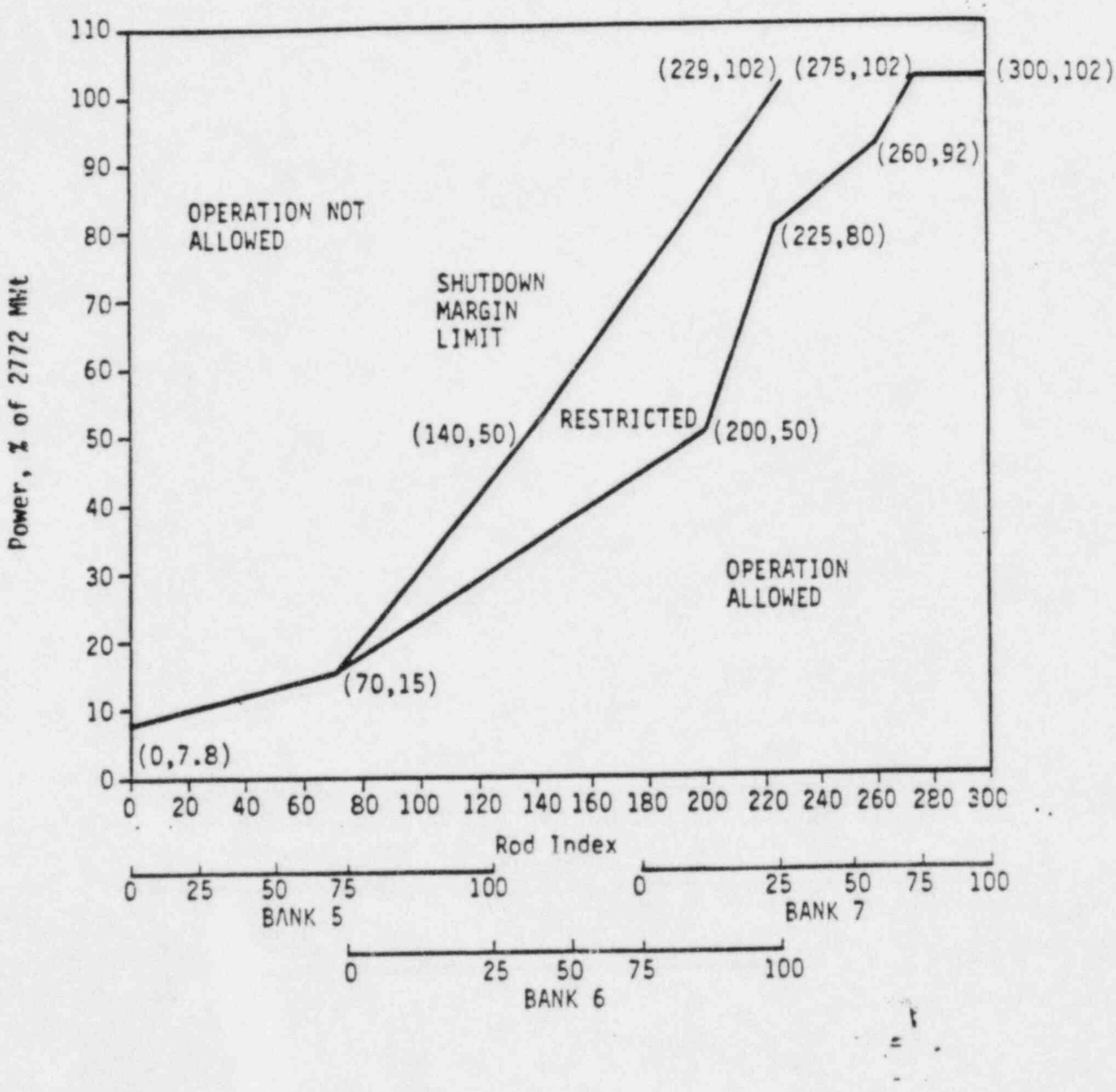


Figure 3.5.2-3 Rod Index Vs Power Level for Four-Pump Operation After 300 EFPD With APSRs Withdrawn -- Rancho Seco 1, Cycle 7

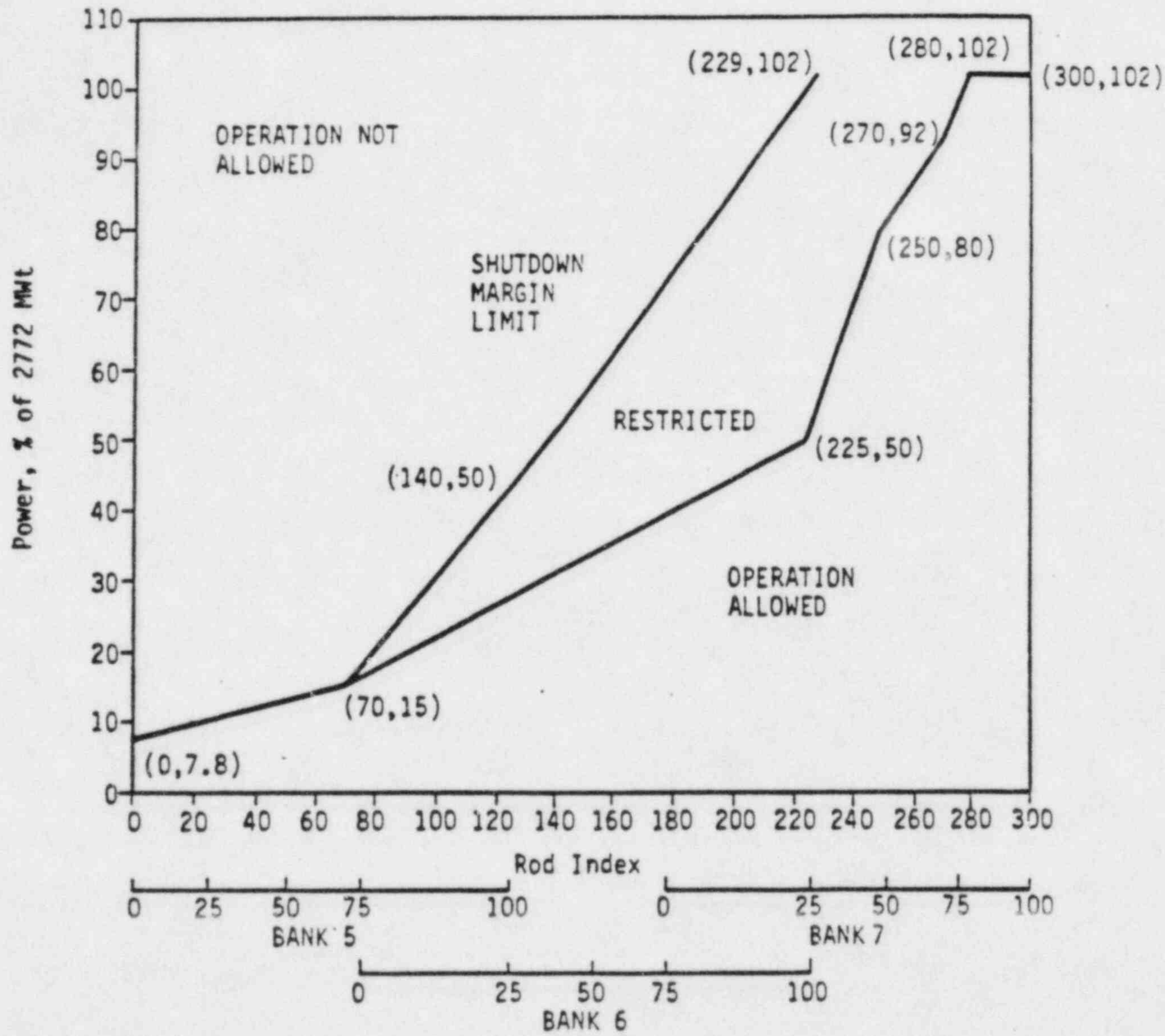


Figure 3.5.2-4 Rod Index Vs Power Level for Three-Pump Operation, 0 to 40 EFPD -- Rancho Seco 1, Cycle 7

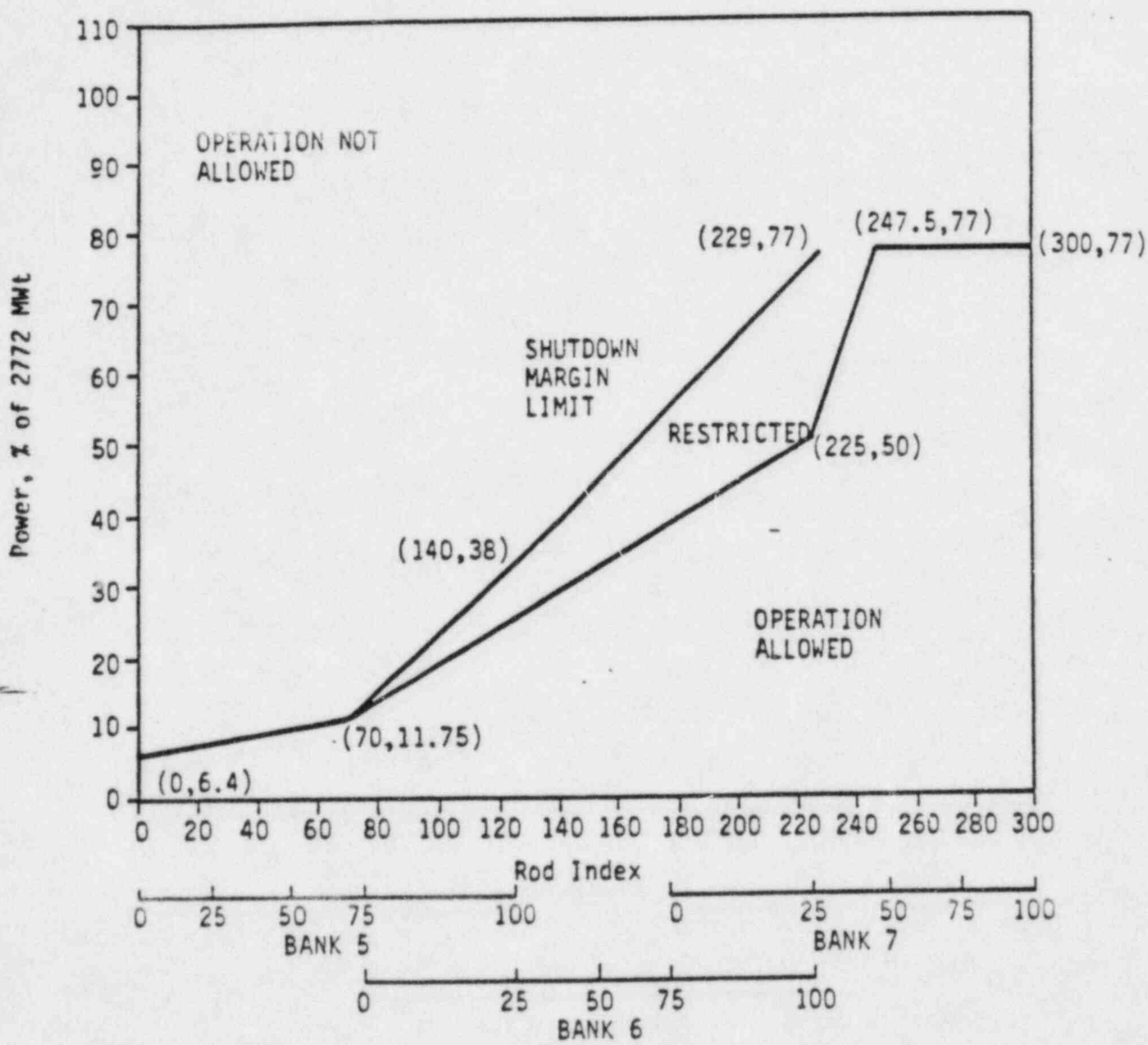


Figure 3.5.2-5 Rod Index Vs Power Level for Three-Pump Operation After 30 EFPD -- Rancho Seco 1, Cycle 7

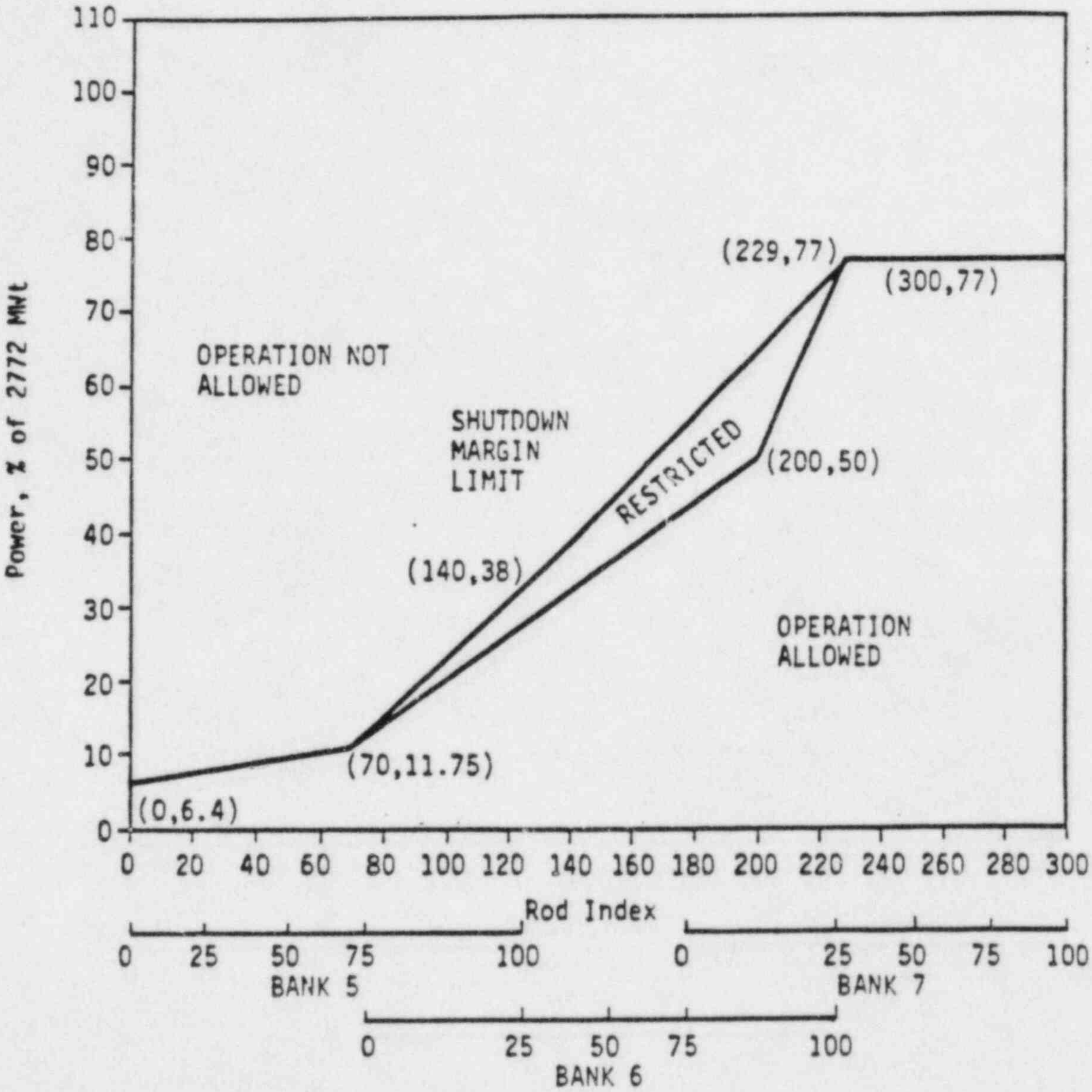


Figure 3.5.2-6 Rod Index Vs Power Level for Three-Pump Operation After 300 EFPD With APSRs Withdrawn -- Rancho Seco 1, Cycle 7

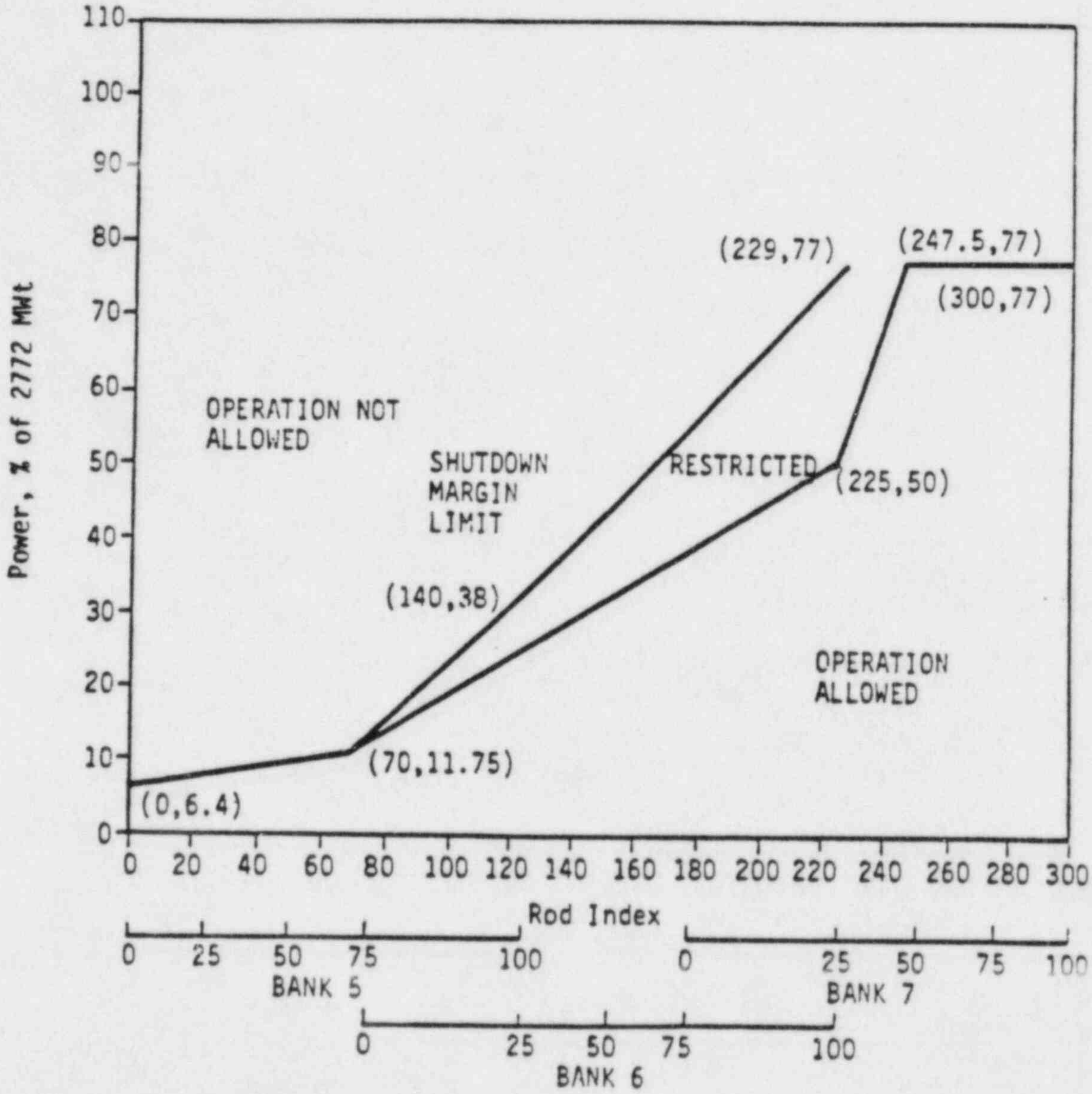


Figure 3.5.2-7 Core Imbalance vs Power Level, 0 to 40 EFPD -- Rancho Seco 1, Cycle 7

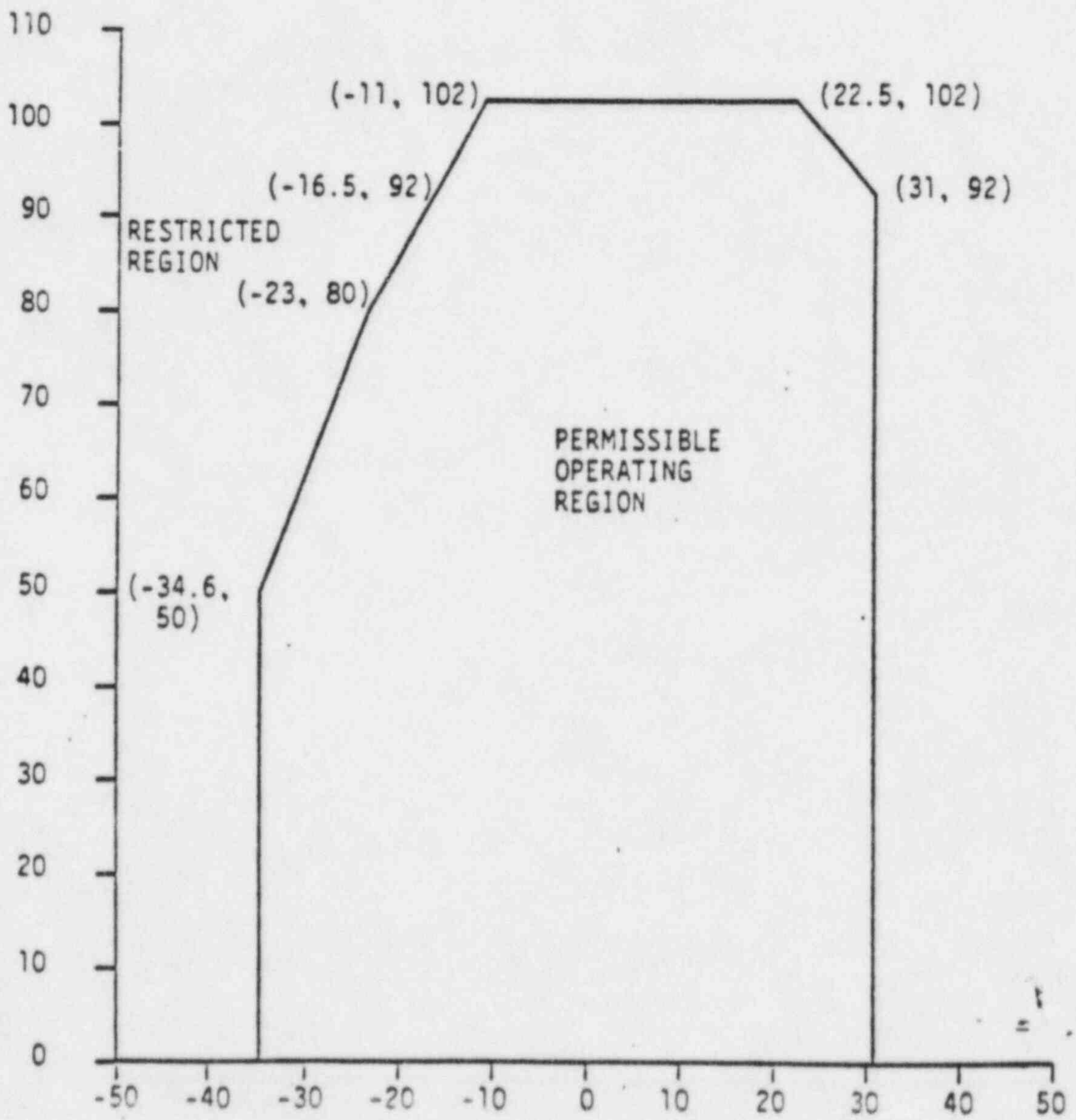


Figure 3.5.2-8 Core Imbalance Vs Power Level After 30 EFPD -- Rancho Seco 1, Cycle 7

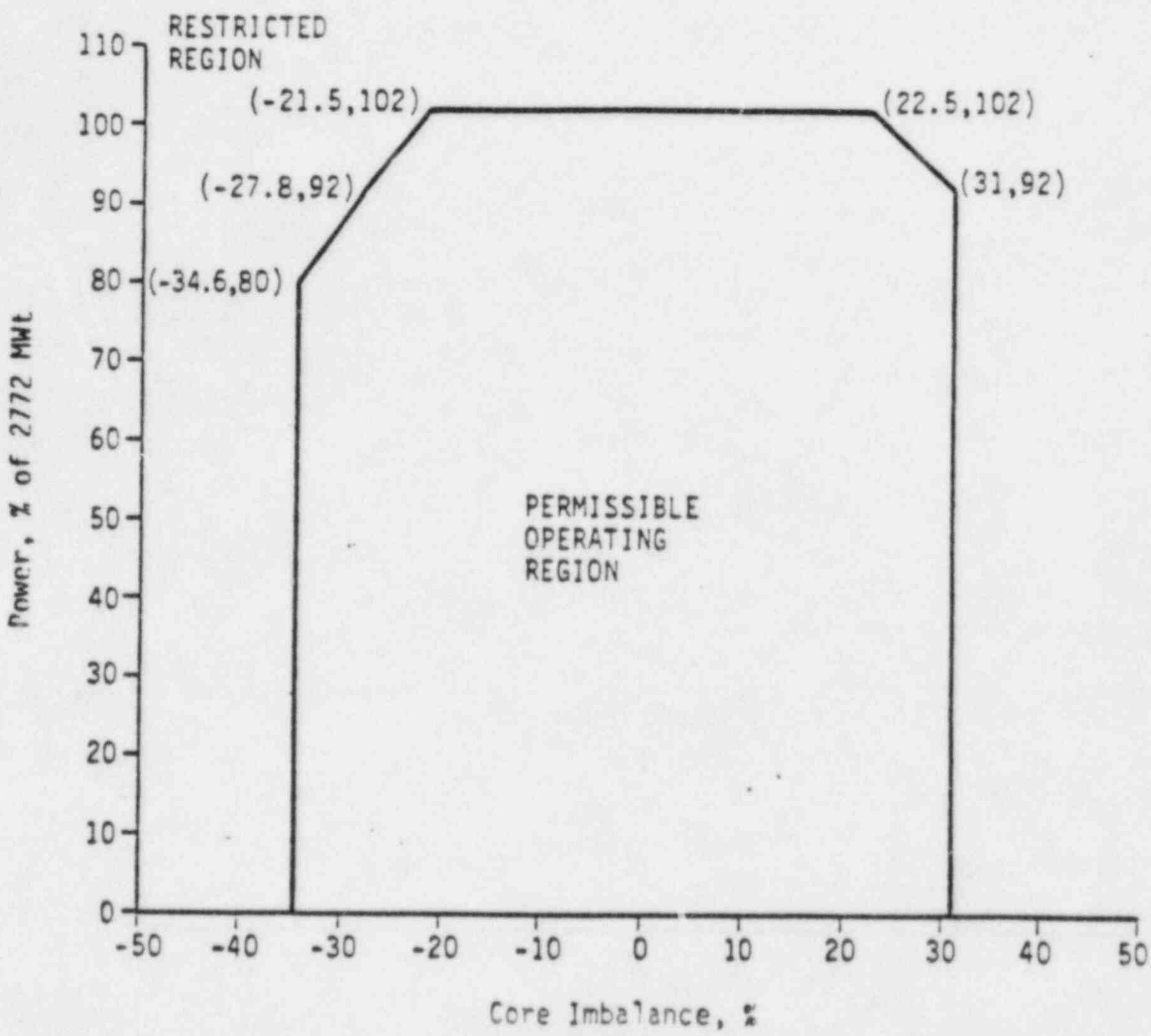


Figure 3.5.2-9 Core Imbalance Vs Power Level After 300 EFPD With APSRs Withdrawn -- Rancho Seco 1, Cycle 7