



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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130  
License No. DPR-53

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated February 12, 1988, as supplemented on March 21, March 25 (2 letters) and April 14, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-53 is hereby amended to read as follows:

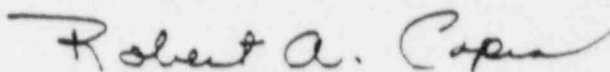
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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 130, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Robert A. Capra, Director  
Project Directorate I-1  
Division of Reactor Projects, I/II

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: May 16, 1988



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

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 130 FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Revise Appendix A as follows:

Remove Pages

2-11  
2-12  
2-13  
3/4 1-1  
3/4 1-2\*  
3/4 1-2a  
3/4 1-3\*  
3/4 1-4\*  
3/4 1-5  
3/4 1-5a  
3/4 1-6\*  
3/4 1-27  
3/4 2-3\*  
3/4 2-4  
3/4 2-11  
3/4 2-12\*  
B2-1  
B2-2\*  
B2-3  
B2-4\*  
B2-5  
B2-6  
B3/4 1-1  
B3/4 1-1a  
B3/4 1-2\*  
B3/4 1-3\*  
B3/4 1-4\*  
B3/4 1-5\*

Insert Pages

2-11  
2-12  
2-13  
3/4 1-1  
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3/4 1-2a  
3/4 1-3\*  
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3/4 1-5a  
3/4 1-6\*  
3/4 1-27  
3/4 2-3\*  
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B2-4\*  
B2-5  
B2-6  
B3/4 1-1  
B3/4 1-1a  
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B3/4 1-4\*  
B3/4 1-5\*

\*Overleaf pages provided to maintain document completeness.

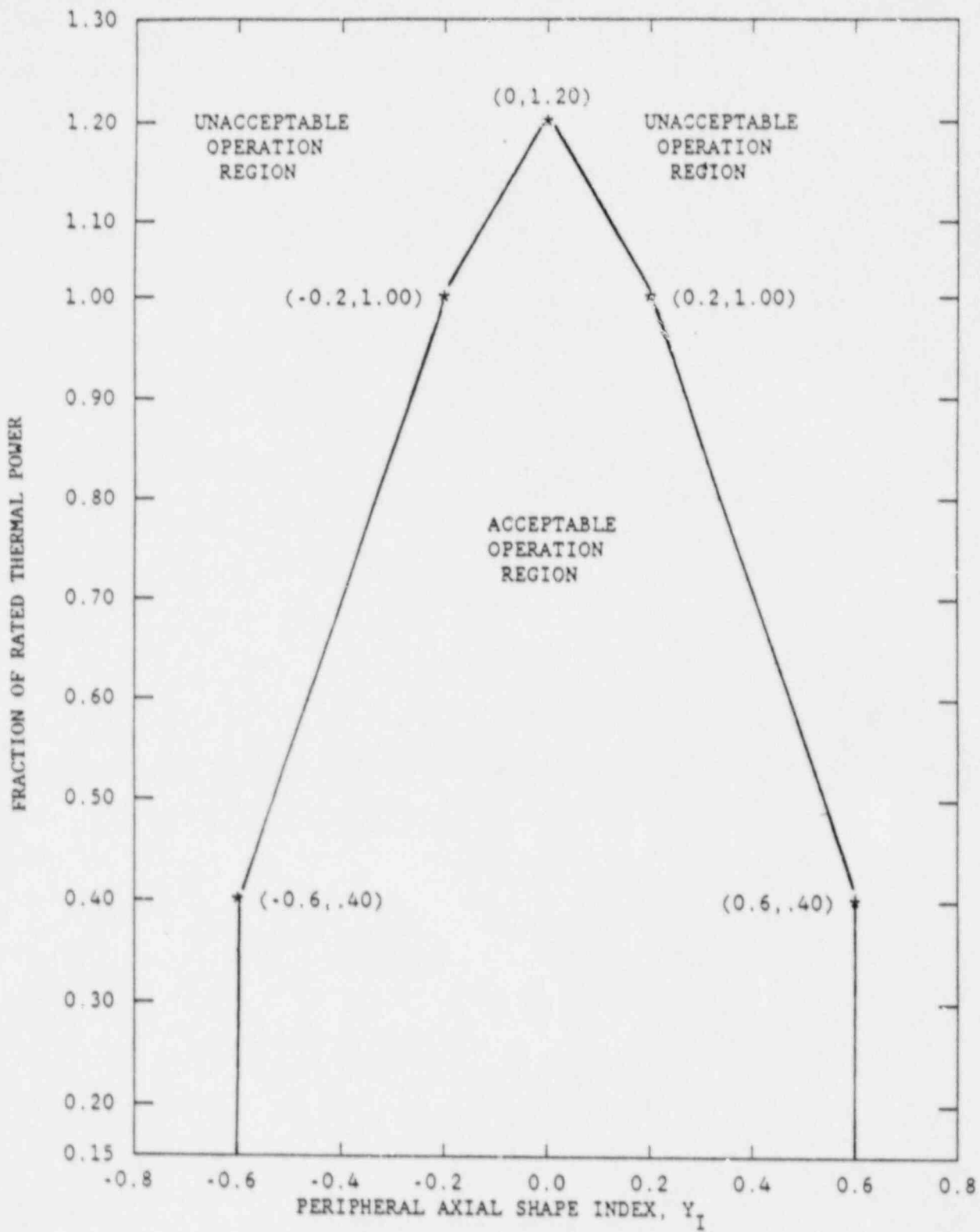


Figure 2.2-1  
Peripheral Axial Shape Index,  $Y_I$  vs Fraction of Rated Thermal Power

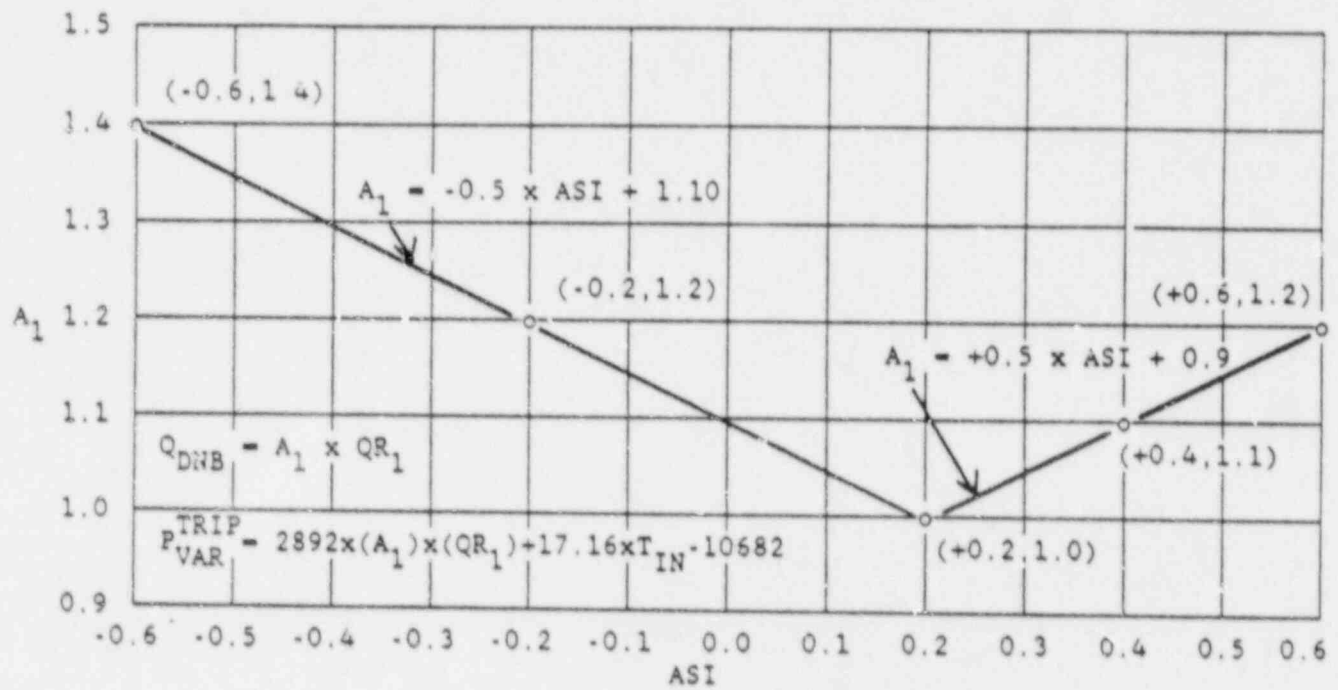


FIGURE 2.2-2

Thermal Margin/Low Pressure Trip Setpoint  
Part 1 (ASI versus A<sub>1</sub>)

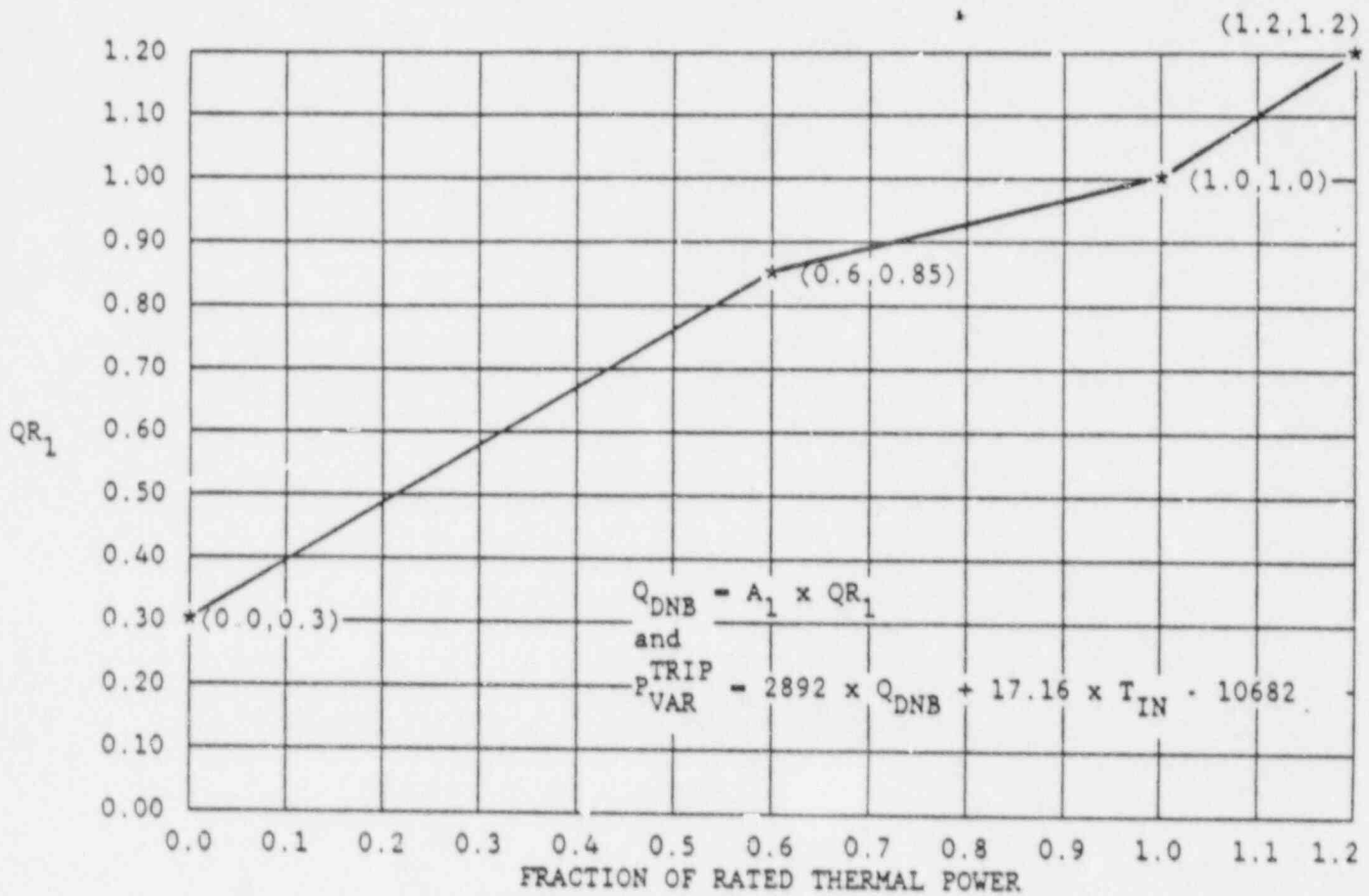


FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint

Part 2 (Fraction of RATED THERMAL POWER versus QR<sub>1</sub>)

### 3/4.1 REACTIVITY CONTROL SYSTEMS

#### 3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN -  $T_{avg} > 200^{\circ}F$

#### LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be equal to or greater than the limit line of Figure 3.1-1b.

APPLICABILITY: MODES 1, 2\*\*, 3, and 4.

#### ACTION:

With the SHUTDOWN MARGIN less than the limit line of Figure 3.1-1b immediately initiate and continue boration at  $\geq 40$  gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

#### SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than the limit of Figure 3.1-1b:

- a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).
- b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.
- c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.
- d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

\* Adherence to Technical Specification 3.1.3.6 as specified in Surveillance Requirements 4.1.1.1.1 assures that there is sufficient available shutdown margin to match the shutdown margin requirements of the safety analyses.

\*\* See Special Test Exception 3.10.1.

# With  $K_{eff} \geq 1.0$

## With  $K_{eff} < 1.0$

## REACTIVITY CONTROL SYSTEMS

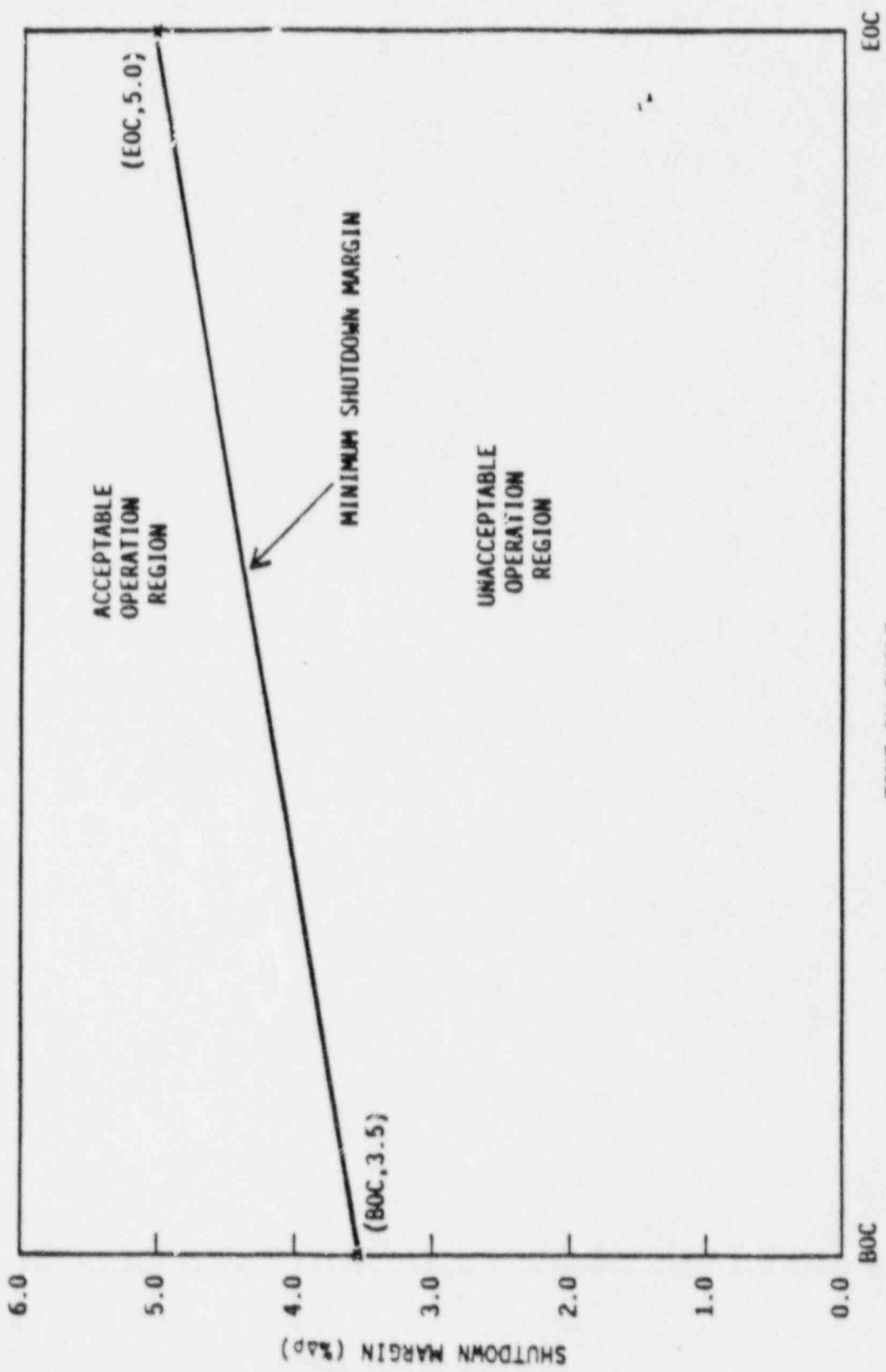
### SURVEILLANCE REQUIREMENTS (Continued)

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- e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:
1. Reactor coolant system boron concentration,
  2. CEA position,
  3. Reactor coolant system average temperature,
  4. Fuel burnup based on gross thermal energy generation,
  5. Xenon concentration, and
  6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within  $\pm 1.0\% \Delta k/k$  at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.





TIME IN CYCLE  
Figure 3.1-1b

## REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN -  $T_{avg} \leq 200^{\circ}F$

### LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be  $\geq 3.0\%$   $\Delta k/k$ .

#### APPLICABILITY: MODE 5

- a. Pressurizer level  $\geq 90$  inches from bottom of the pressurizer.
- b. Pressurizer level  $< 90$  inches from bottom of the pressurizer and all sources of non-borated water  $\leq 88$  gpm.

#### ACTION:

- a. With the SHUTDOWN MARGIN  $< 3.0\%$   $\Delta k/k$ , immediately initiate and continue boration at  $\geq 40$  gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.
- b. With the pressurizer drained to  $\leq 90$  inches and all sources of non-borated water  $> 88$  gpm, immediately suspend all operations involving positive reactivity changes while the SHUTDOWN MARGIN is increased to compensate for the additional sources of non-borated water or reduce the sources of non-borated water to  $\leq 88$  gpm.

### SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be  $\geq 3.0\%$   $\Delta k/k$ :

- a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).
- b. At least once per 24 hours by consideration of the following factors:
  1. Reactor coolant system boron concentration,
  2. CEA position,
  3. Reactor coolant system average temperature,
  4. Fuel burnup based on gross thermal energy generation,
  5. Xenon concentration, and
  6. Samarium concentration.

4.1.1.2.2. With the pressurizer drained to  $\leq 90$  inches determine:

- a. Within one hour and every 12 hours thereafter that the level in the reactor coolant system is above the bottom of the hot leg nozzles, and
- b. Within one hour and every 12 hours thereafter that the sources of non-borated water are  $\leq 88$  gpm or the shutdown margin has compensated for the additional sources.

## REACTIVITY CONTROL SYSTEMS

### BORON DILUTION

#### LIMITING CONDITION FOR OPERATION

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3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be  $\geq 3000$  gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

#### ACTION:

With the flow rate of reactor coolant through the reactor coolant system  $< 3000$  gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

#### SURVEILLANCE REQUIREMENTS

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4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be  $\geq 3000$  gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

- a. Verifying at least one reactor coolant pump is in operation,  
or
- b. Verifying that at least one low pressure safety injection pump is in operation and supplying  $\geq 3000$  gpm through the reactor coolant system.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

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3.1.1.4 The moderator temperature coefficient (MTC) shall be:

- a. Less positive than the limit line of Figure 3.1-1a, and
- b. Less negative than  $-2.7 \times 10^{-4} \Delta k/k/^{\circ}F$  at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2\*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

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\* With  $K_{eff} \geq 1.0$ .

# See Special Test Exception 3.10.2.

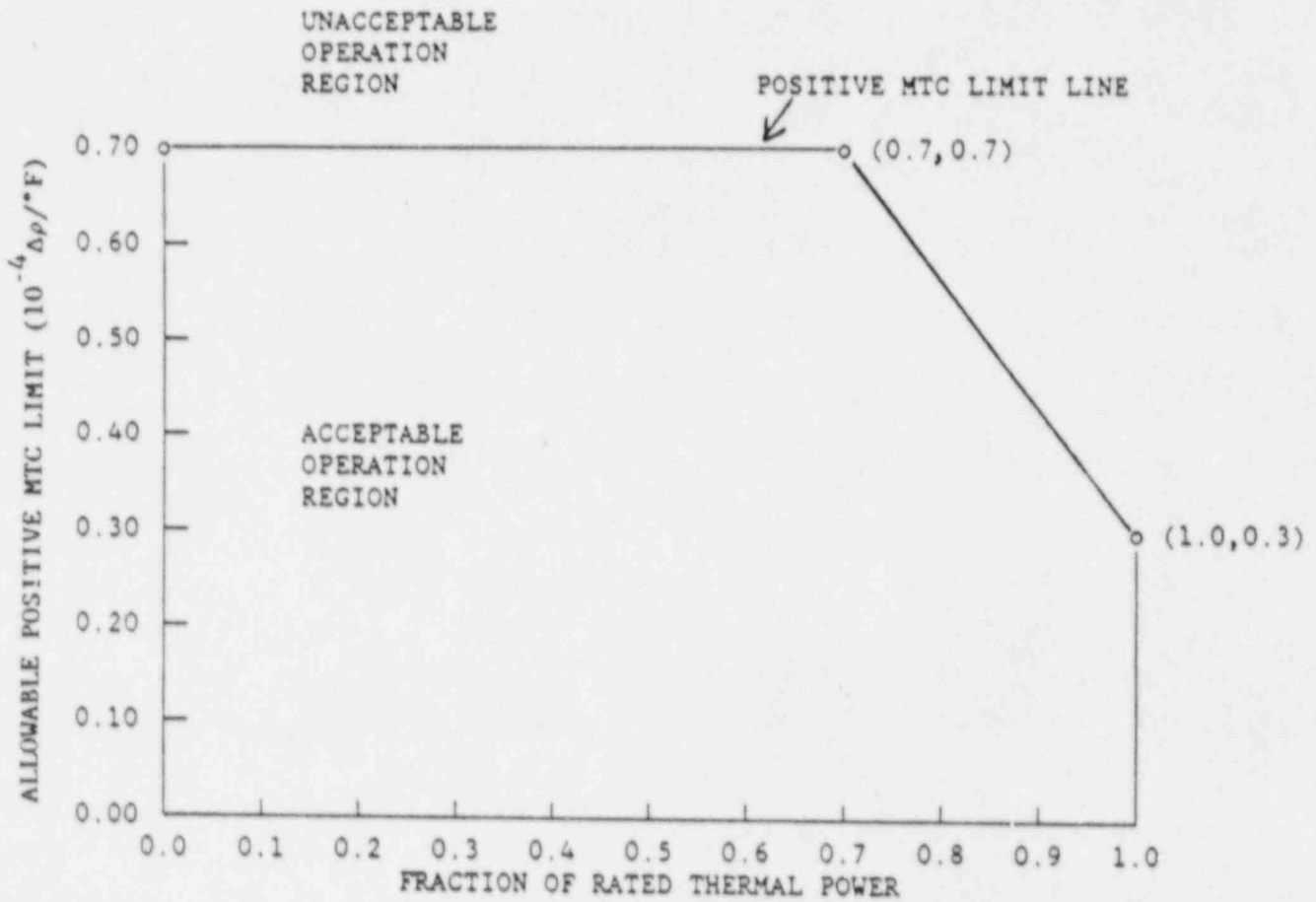


FIGURE 3.1-1a

Fraction of Rated Thermal Power  
 vs. Allowable Positive MTC Limit ( $10^{-4} \Delta\rho/^\circ\text{F}$ )

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

- a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.
- b. At any THERMAL POWER above 90% of RATED THERMAL POWER, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER.
- c. At any THERMAL POWER, within 7 EFPD of reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

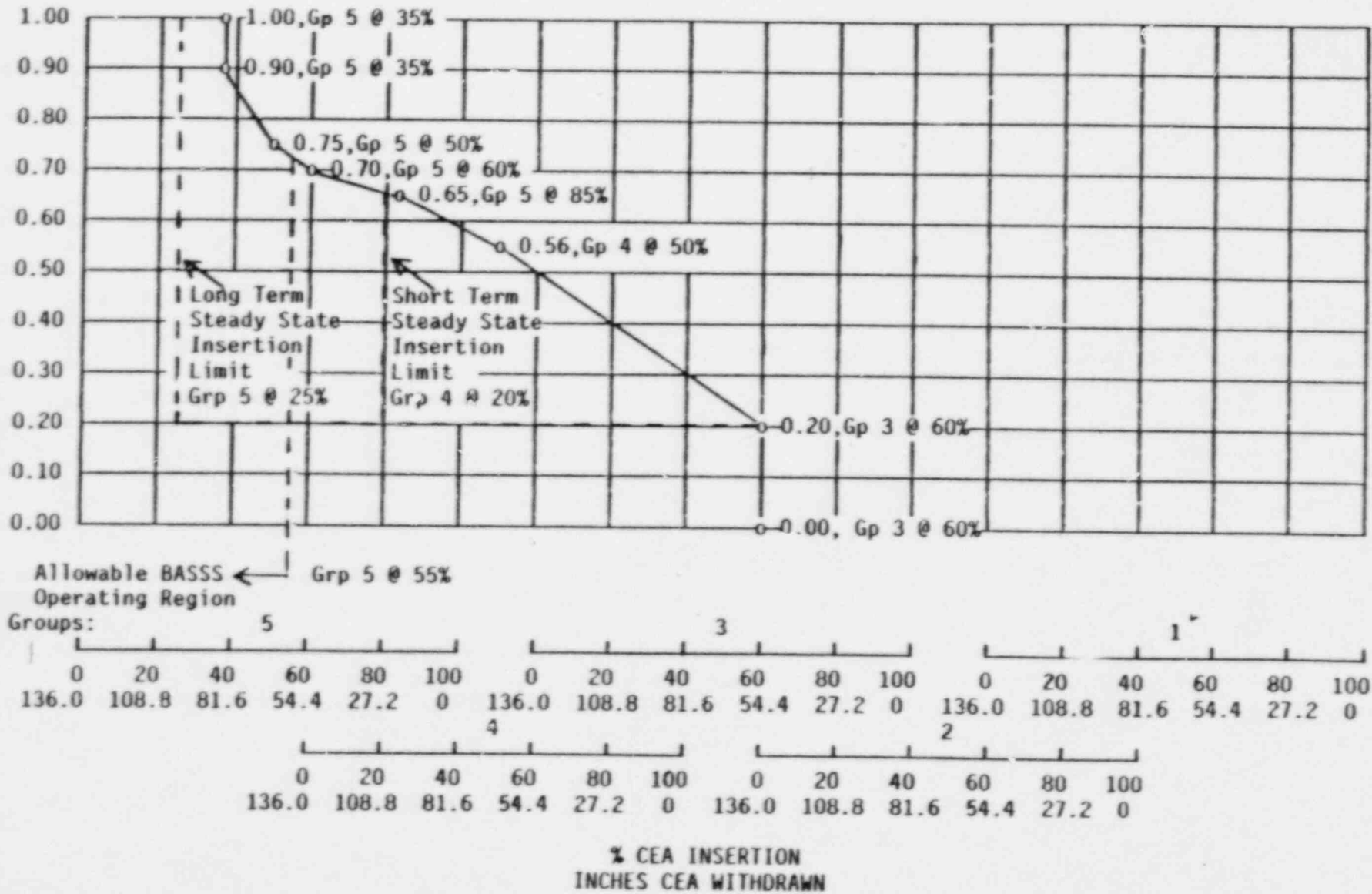


Figure 3.1-2  
 CEA GROUP INSERTION LIMITS VS. FRACTION OF ALLOWABLE THERMAL POWER  
 FOR EXISTING RCP COMBINATION

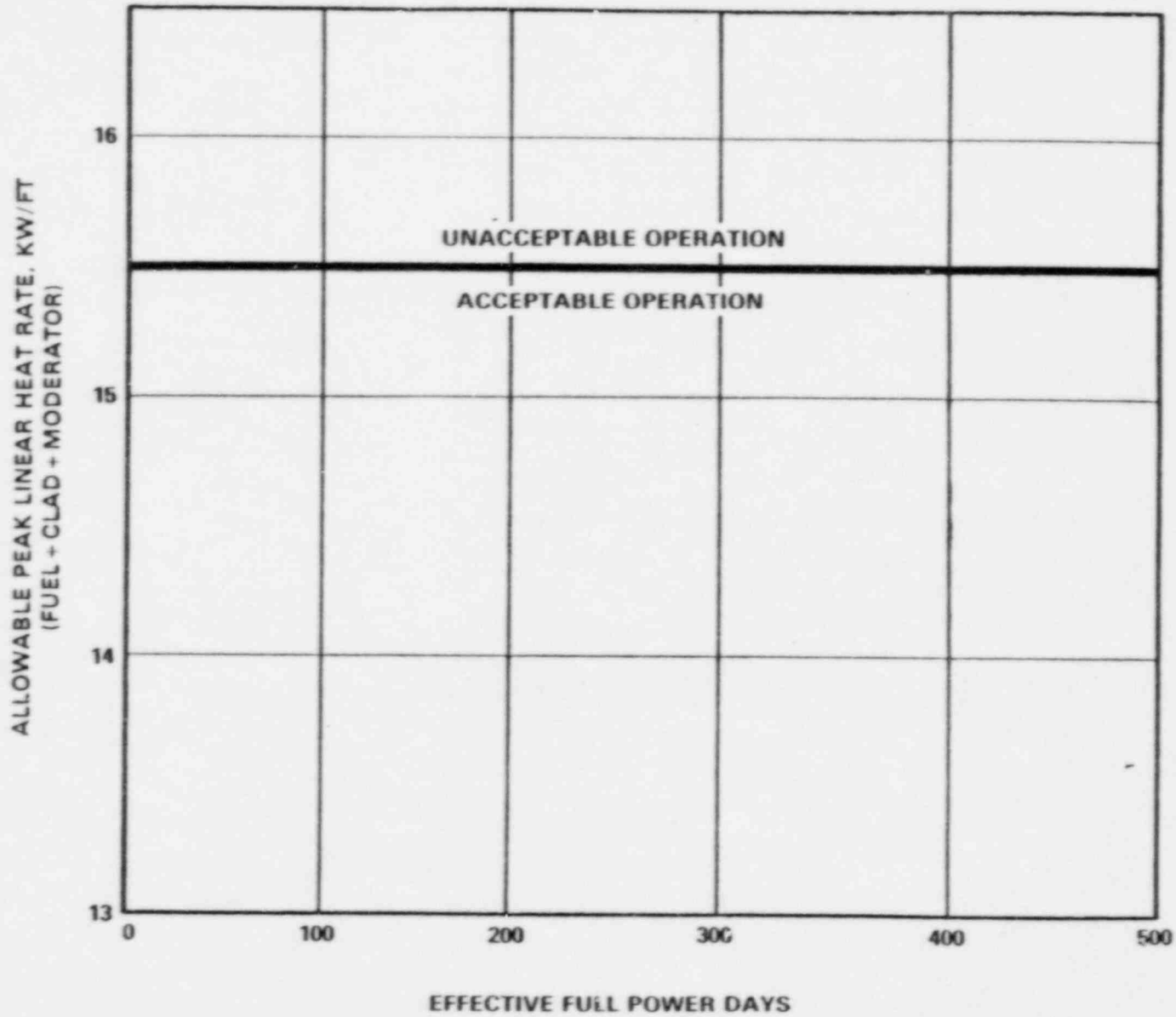


FIGURE 3.2-1  
Allowable Peak Linear Heat Rate vs Burnup



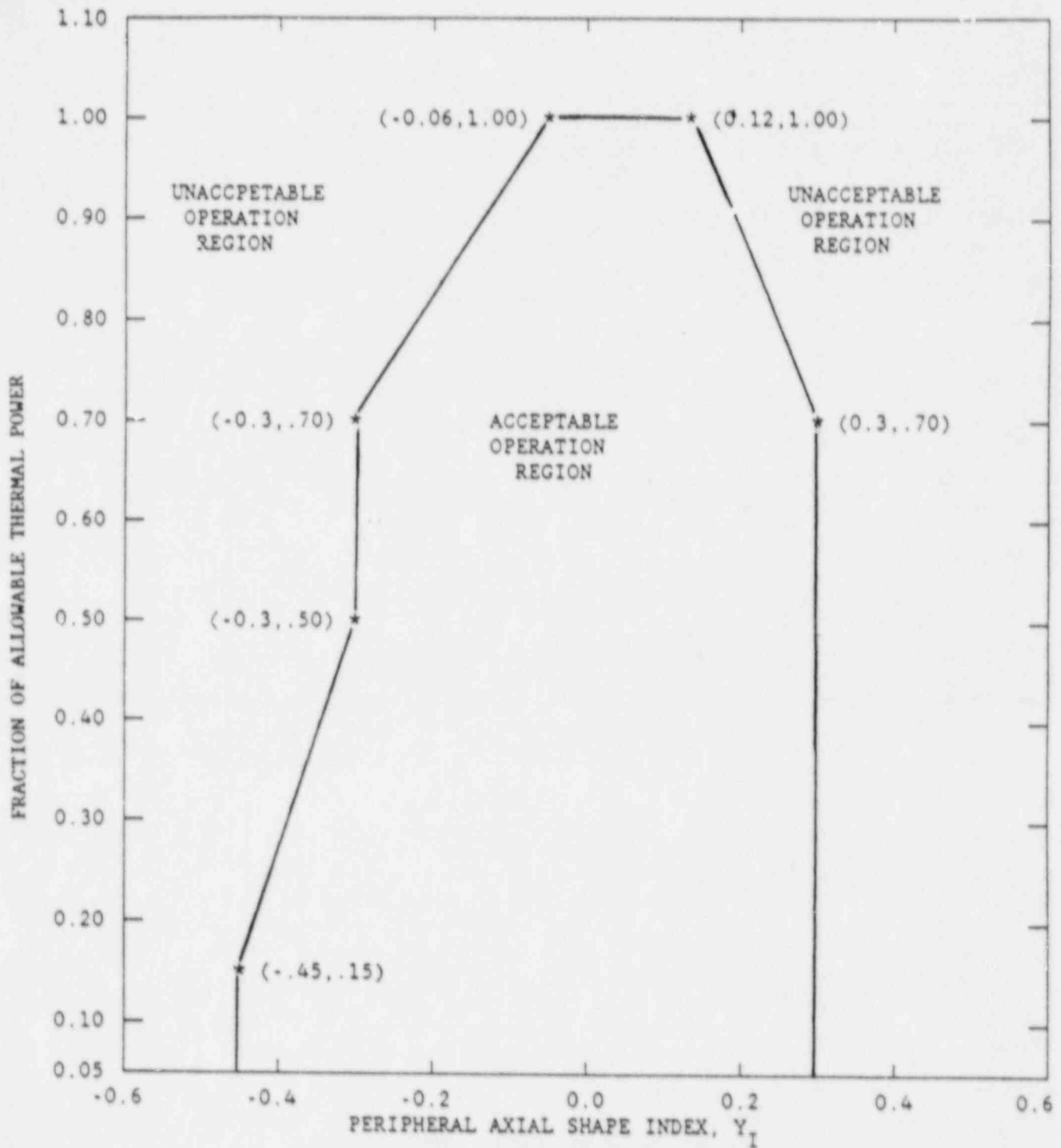


Figure 3.2-2  
 Linear Heat Rate Axial Flux Offset Control Limits

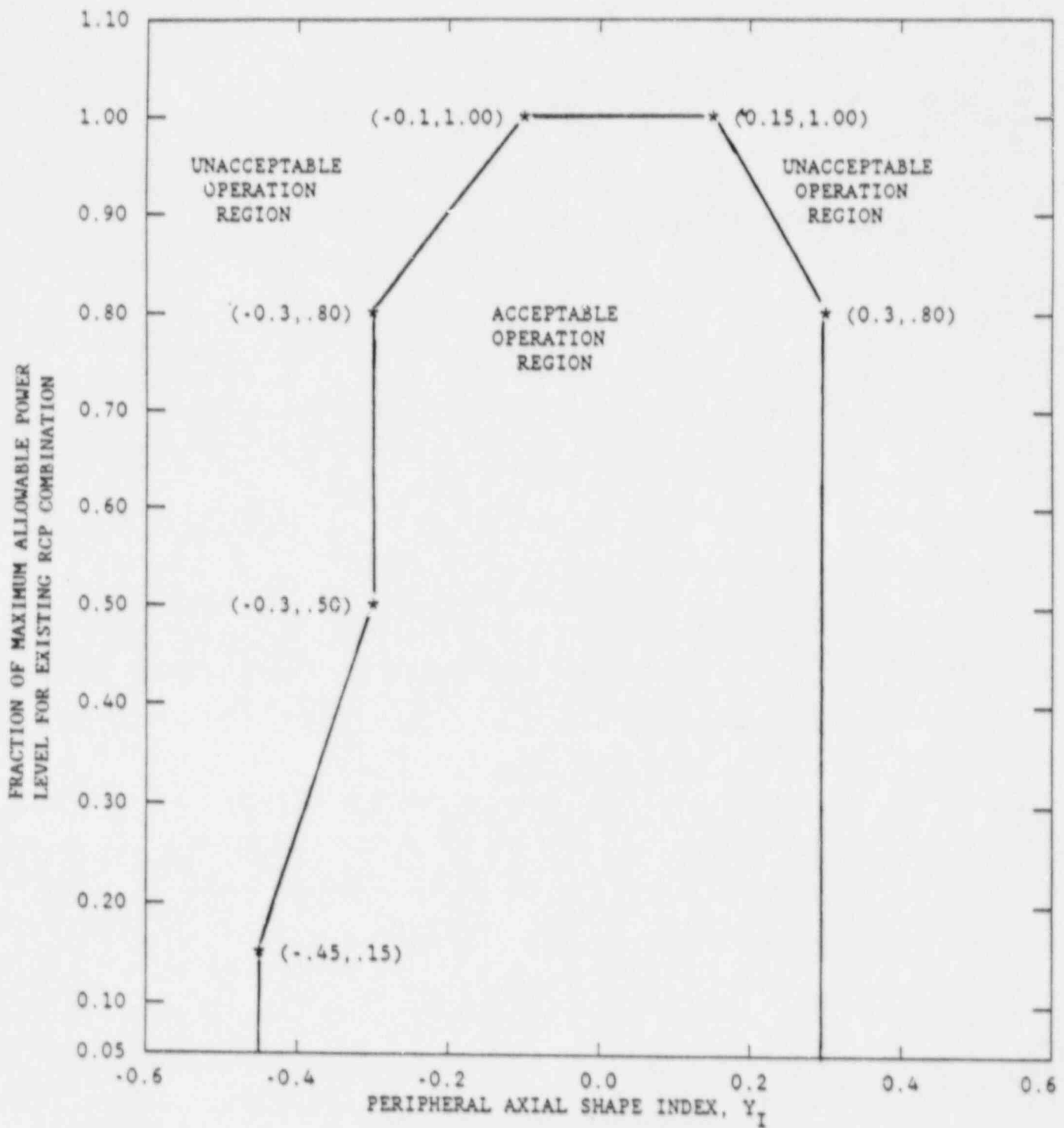


Figure 3.2-4  
DNB Axial Flux Offset Control Limits

## POWER DISTRIBUTION LIMITS

### AZIMUTHAL POWER TILT - $T_q$

#### LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT ( $T_q$ ) shall not exceed 0.030.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER.\*

#### ACTION:

- a. With the indicated AZIMUTHAL POWER TILT determined to be  $> 0.030$  but  $\leq 0.10$ , either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR ( $F_{xy}^T$ ) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR ( $F_r^T$ ) are within the limits of Specifications 3.2.2 and 3.2.3.
- b. With the indicated AZIMUTHAL POWER TILT determined to be  $> 0.10$ , operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR ( $F_r^T$ ) and TOTAL PLANAR RADIAL PEAKING FACTOR ( $F_{xy}^T$ ) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to  $\leq 20\%$  of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

#### SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

- a. Calculating the tilt at least once per 12 hours, and
- b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER IS  $> 75\%$  of RATED THERMAL POWER.

\* See Special Test Exception 3.10.2.

## 2.1 SAFETY LIMITS

### BASES

#### 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which could result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 22.0 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and, therefore, **THERMAL POWER** and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local 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 the DNB SAFDL of 1.15 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This DNB SAFDL assures with at least a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show conservative loci for points of **THERMAL POWER**, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the DNB SAFDL is not violated for the family of axial shapes and corresponding radial peaks shown in Figure 82.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3, and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at **THERMAL POWER** levels higher than 110% of **RATED THERMAL POWER** is prohibited by the high power level trip setpoint specified in

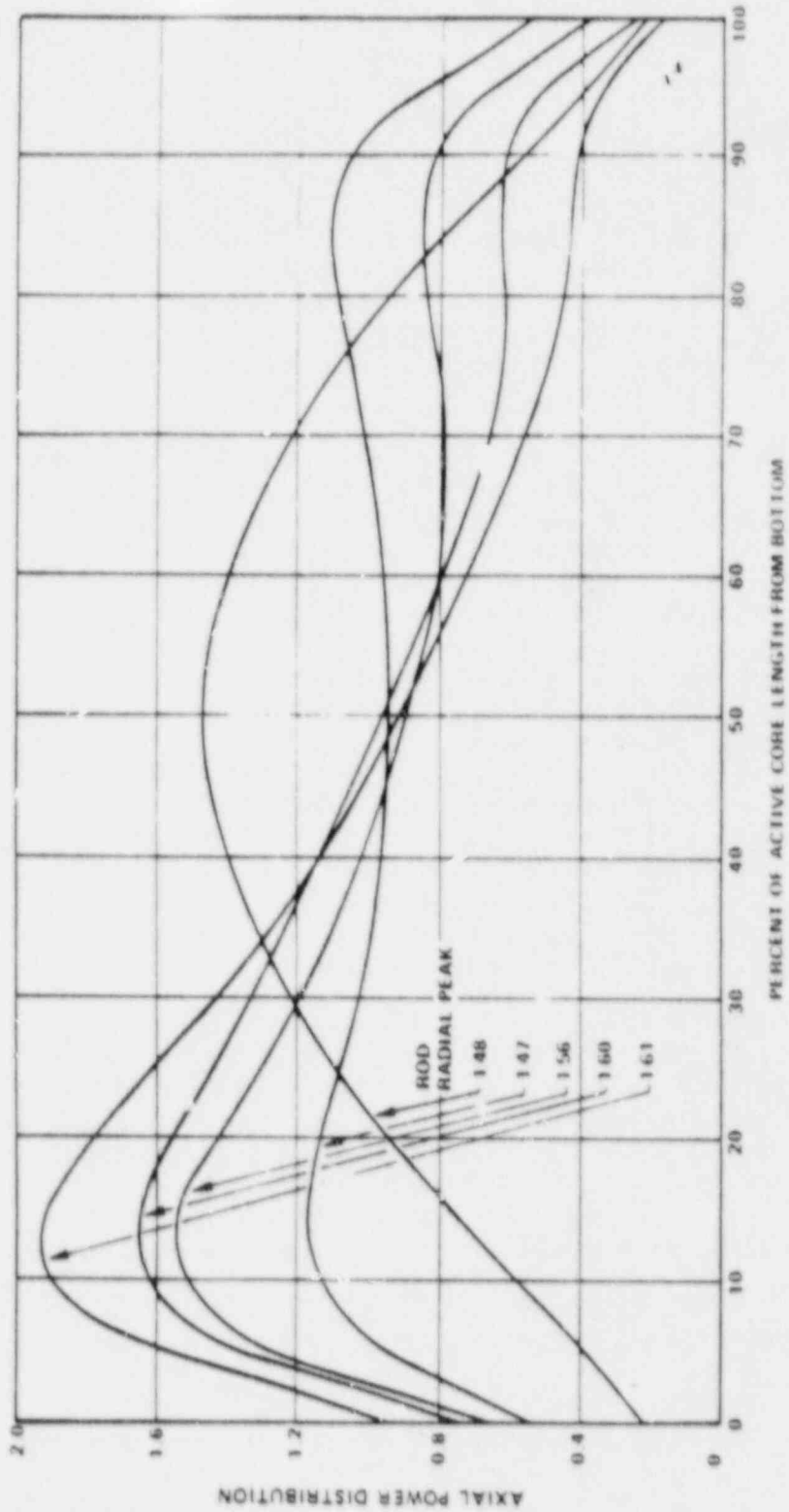


Figure B2.1.1 Axial Power Distribution for Thermal Margin Safety Limits

## SAFETY LIMITS

### BASES

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Table 2.1.-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figures 2.1-1, 2.1-2, 2.1-3, and 2.1-4 to be valid are shown on the figures.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than 1.15, in conjunction with the ESCU methodology, and preclude the existence of flow instabilities.

#### 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I, 1969 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is, therefore, consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3215 psia to demonstrate integrity prior to initial operation.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### BASES

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#### 2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between the trip setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

##### Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

##### Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 30% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 110% of RATED THERMAL POWER, which is the value used in the safety analyses.

##### Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit



## LIMITING SAFETY SYSTEM SETTINGS

### BASES

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above the DNB SAFDL of 1.15, in conjunction with the ESCU methodology, under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two-or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below DNB SAFDL of 1.15, in conjunction with the ESCU methodology, during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

#### Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

#### Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

#### Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 685 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of  $\pm 85$  psi which was based on the main steam line break event inside containment.



## LIMITING SAFETY SYSTEM SETTINGS

### BASES

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#### Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

#### Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

#### Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of  $\Delta T$  power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### BASES

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#### 3/4.1.1 BORATION CONTROL

##### 3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting SHUTDOWN MARGIN requirement at beginning of cycle is determined by the requirements of several transients, including Boron Dilution and Steam Line Rupture. The SHUTDOWN MARGIN requirements for these transients are relatively small and nearly the same. However, the most limiting SHUTDOWN MARGIN requirement at end of cycle comes from just one transient, the Steam Line Rupture event. The requirement for this transient at end of cycle is significantly larger than that for any other event at that time in cycle and, also, considerably larger than the most limiting requirement at beginning of cycle.

The variation in the most limiting requirement with time in cycle has been incorporated into Technical Specification 3.1.1.1, in the form of a specified SHUTDOWN MARGIN value which varies linearly from beginning to end of cycle. This variation in specified SHUTDOWN MARGIN is conservative relative to the actual variation in the most limiting requirement. Consequently, adherence to Technical Specification 3.1.1.1 provides assurance that the available SHUTDOWN MARGIN at anytime in cycle will exceed the most limiting SHUTDOWN MARGIN requirement at that time in cycle.

In MODE 5, the reactivity transients resulting from any event are minimal and do not vary significantly during the cycle. Therefore, the specified SHUTDOWN MARGIN in MODE 5 via Technical Specification 3.1.1.2 has been set equal to a constant value which is determined by the requirement of the most limiting event at any time during the cycle, i.e., Boron Dilution with the pressurizer level less than 90 inches and the sources of non-borated water restricted. Consequently, adherence to Technical Specification 3.1.1.2 provides assurance that the available SHUTDOWN MARGIN will exceed the most limiting SHUTDOWN MARGIN requirement at any time in cycle.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### BASES

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#### 3/4.1.1 BORATION CONTROL

##### 3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

##### 3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

## REACTIVITY CONTROL SYSTEMS

### BASES

#### 3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum  $RT_{NDT}$  temperature.

#### 3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The system also provides coolant flow following an SIAS (e.g., during a Small Break LOCA) to supplement flow from the Safety Injection System. The Small Break LOCA analyses assume flow from a single charging pump, accounting for measurement uncertainties and flow mal-distribution effects in calculating a conservative value of charging flow actually delivered to the RCS. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUT-DOWN MARGIN from all operating conditions of 3.0%  $\Delta k/k$  after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6500 gallons of 7.25% boric acid solution from the boric acid tanks or 55,627 gallons of 2300 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

## REACTIVITY CONTROL SYSTEMS

### BASES

The boron capability required below 200°F is based upon providing a 3%  $\Delta k/k$  SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 737 gallons of 7.25% boric acid solution from the boric acid tanks or 9,844 gallons of 2300 ppm borated water from the refueling water tank.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

### 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA and to a large misalignment ( $\geq 15$  inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments ( $\leq 15$  inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.



## REACTIVITY CONTROL SYSTEMS

### BASES

Overpower margin is provided to protect the core in the event of a large misalignment ( $\geq 15$  inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. The ACTION statement associated with a large CEA misalignment requires prompt action to realign the CEA to avoid excessive margin degradation. If the CEA is not realigned within the given time constraints, action is specified which will preserve margin, including reductions in THERMAL POWER.

For a single CEA misalignment, the time allowance to realign the CEA (Figure 3.1-3) is permitted for the following reasons:

1. The margin calculations which support the power distribution LCOs for DNBR are based on a steady-state  $F_T^I$  as specified in Technical Specification 3.2.3.
2. When the actual  $F_T^I$  is less than the Technical Specification value, additional margin exists.
3. This additional margin can be credited to offset the increase in  $F_T^I$  with time that will occur following a CEA misalignment due to xenon redistribution.

The requirement to reduce power level after the time limit of Figure 3.1-3 is reached offsets the continuing increase in  $F_T^I$  that can occur due to xenon redistribution. A power reduction is not required below 50% power. Below 50% power there is sufficient conservatism in the DNB power distribution LCOs to completely offset any, or any additional, xenon redistribution effects.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

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### BASES

Operability of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the OPERABILITY and the ACTION statements applicable to inoperable CEA position indicators permit continued operations when positions of CEAs with inoperable indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The surveillance requirements affecting CEAs with inoperable position indication channels allow 10 minutes for testing each affected CEA. This time limit was selected so that 1) the time would be long enough for the required testing, and 2) if all position indication were lost during testing, the time would be short enough to allow a power reduction to 70% of maximum allowable thermal power within one hour from when the testing was initiated. The time limit ensures CEA misalignments occurring during CEA testing are corrected within the time requirements required by existing specifications.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with  $T_{avg} \geq 515^{\circ}\text{F}$  and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Transient Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.