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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. DPR-69

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

   A. The applications for amendment by Baltimore Gas and Electric Company (the licensee) dated February 12, 1988 and February 7, 1989, as supplemented on March 30, April 21 and April 25 and May 8, 1989, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the 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-69 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 123 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 10, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 123 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-11	2-11
2-12	2-12
2-13	2-13
B2-1	B2-1
B2-2*	B2-2*
B2-3	B2-3
B2-4*	B2-4*
B2-5	B2-5
B2-6	B2-6
3/4 1-1	3/4 1-1
3/4 1-2*	3/4 1-2*
3/4 1-2a	3/4 1-2a
3/4 1-3*	3/4 1-3*
3/4 1-4*	3/4 1-4*
3/4 1-5*	3/4 1-5*
3/4 1-5a*	3/4 1-5a*
3/4 1-6*	3/4 1-6*
3/4 1-9*	3/4 1-9*
3/4 1-10*	3/4 1-10*
3/4 1-11*	3/4 1-11*
3/4 1-12*	3/4 1-12*
3/4 1-17*	3/4 1-17*
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-19A*	3/4 1-19A*
3/4 1-19B	3/4 1-19B
3/4 1-20*	3/4 1-20*
3/4 1-23*	3/4 1-23*
3/4 1-24*	3/4 1-24*
3/4 1-27	3/4 1-27
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-4a	3/4 2-4A
3/4 2-5*	3/4 2-5*
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a*	3/4 2-7A*
3/4 2-8	3/4 2-8
3/4 2-8 (cont.)	3/4 2-8A
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10A
3/4 2-11	3/4 2-11
3/4 2-12*	3/4 2-12*
3/4 2-13	3/4 2-13
3/4 2-14	3/4 2-14

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 123 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows: (CONTINUED)

Remove Pages	Insert Pages
B3/4 1-1	B3/4 1-1
B3/4 1-1a	B3/4 1-1A
B3/4 1-2*	B3/4 1-2*
B3/4 1-3*	B3/4 1-3*
B3/4 1-4	B3/4 1-4
B3/4 1-5*	B3/4 1-5*
B3/4 2-1*	B3/4 2-1*
B3/4 2-2	B3/4 2-2
B3/4 7-2a	B3/4 7-2A
5-3*	5-3*
5-4	5-4

* Overleaf pages provided for continuity purposes only.

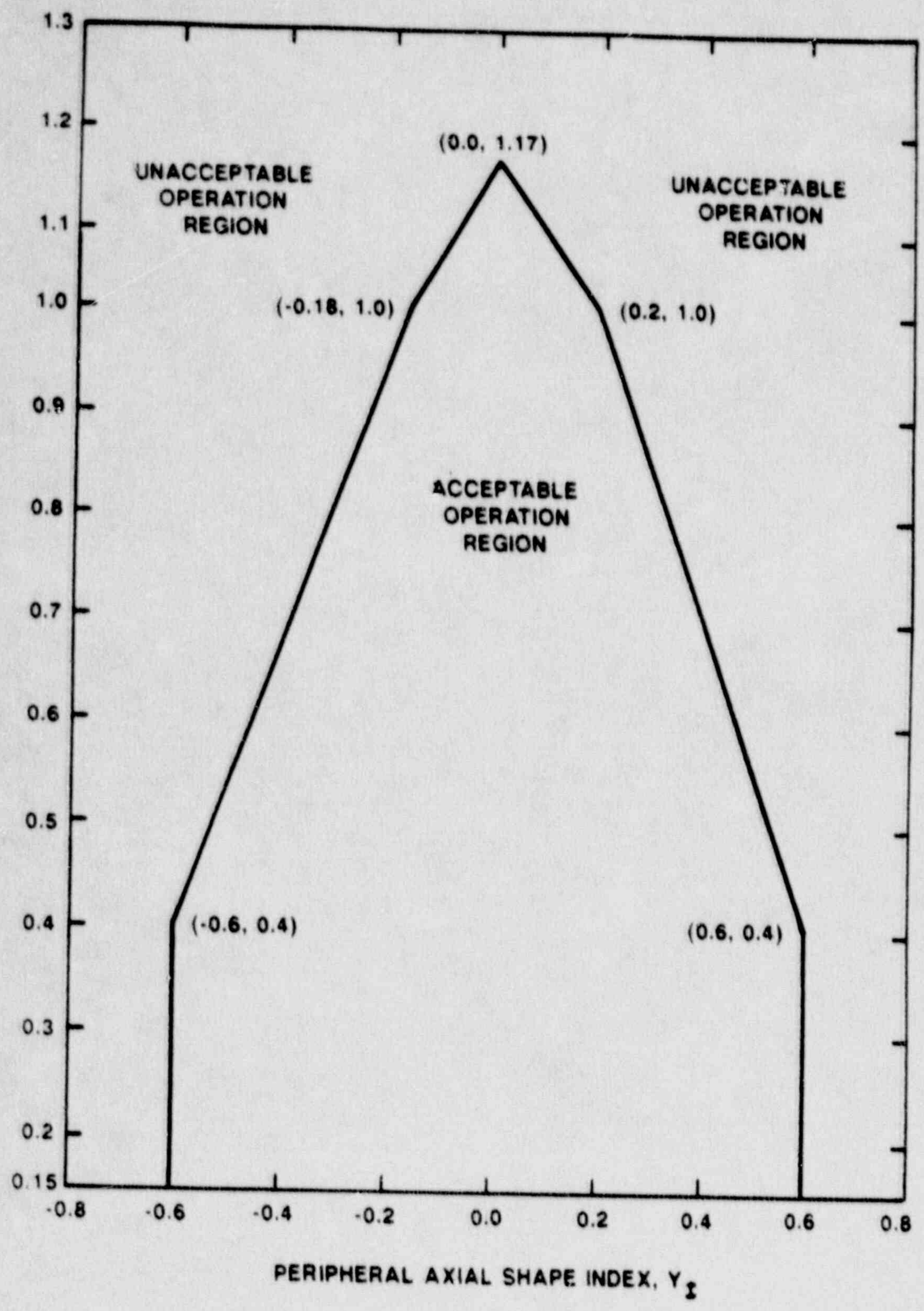

FIGURE 2.2-1
PERIPHERAL AXIAL SHAPE INDEX, Y_I
vs. FRACTION OF RATED THERMAL POWER

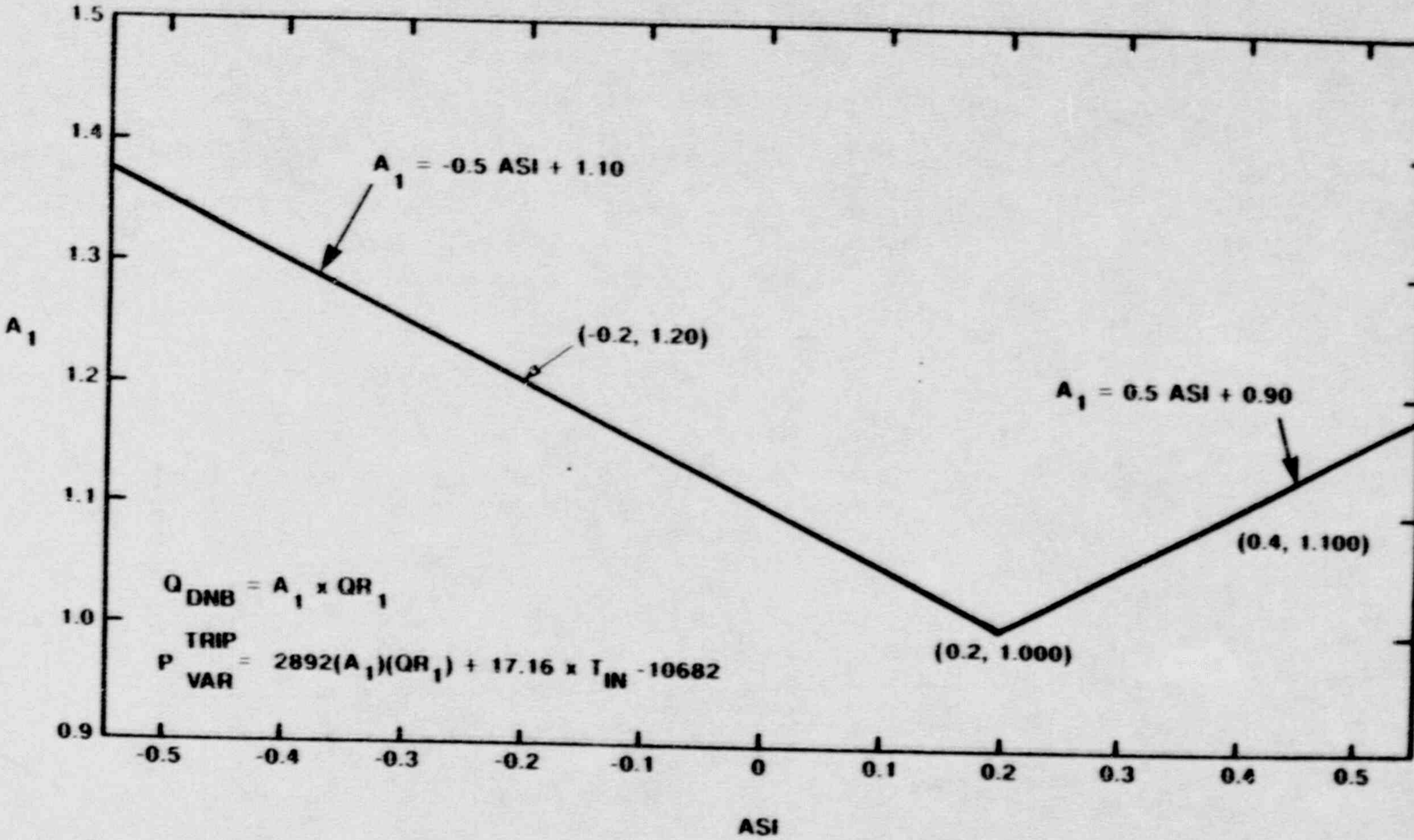

FIGURE 2.2-2

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (ASI VERSUS A_1)

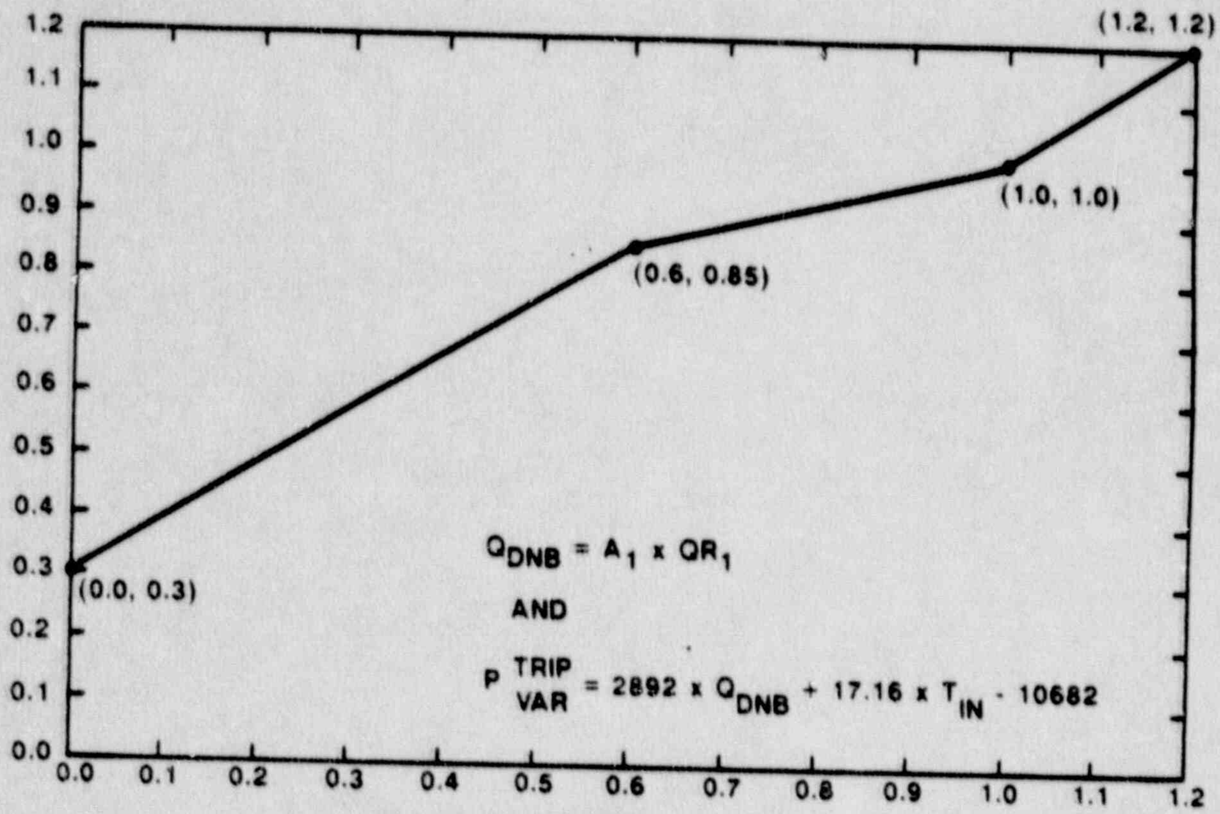

FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT

PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

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 of 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 B2.1-1. 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

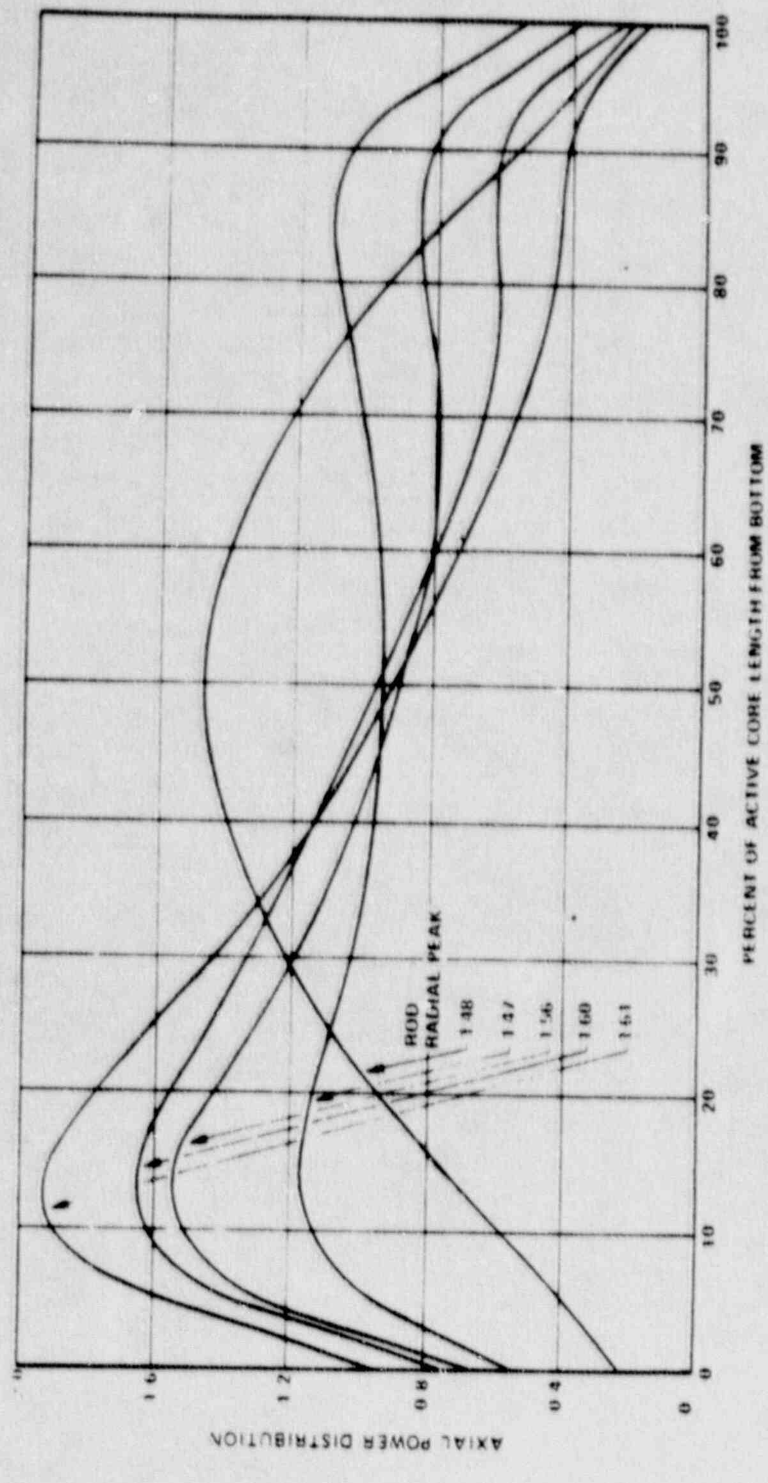

Figure B2.1.1 Axial Power Distribution for Thermal Margin Safety Limits

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 over pressurization 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 3125 psia to demonstrate integrity prior to initial operation.

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 each 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 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 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 the DNB SAFDL of 1.15, in conjunction with 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 over pressurization 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 ± 85 psi in the accident analyses which was based on the Main Steam Line Break event.

LIMITING SAFETY SYSTEM SETTINGS

BASES

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 ΔT 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

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°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 ≥ 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 line 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.

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* 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 $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

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.

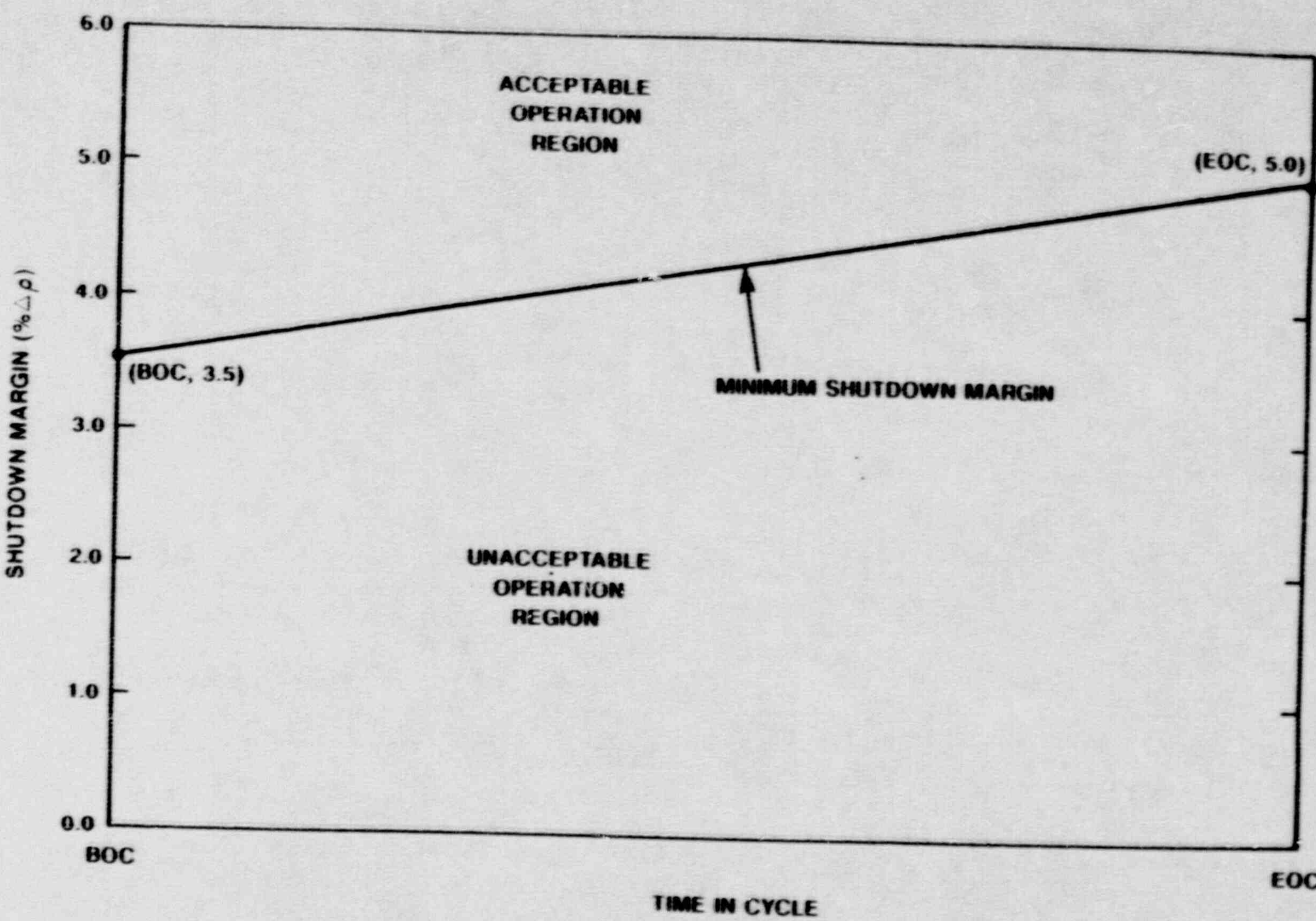
6.0
5.0
4.0
3.0
2.0
1.0
0.0
SHUTDOWN MARGIN (%Δρ)
ACCEPTABLE
OPERATION
REGION
(EOC, 5.0)
(BOC, 3.5)
MINIMUM SHUTDOWN MARGIN
UNACCEPTABLE
OPERATION
REGION
BOC
EOC
TIME IN CYCLE

FIGURE 3.1-1b

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.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

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

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 SYSTEM

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

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 $-2.7 \times 10^{-4}\ \Delta k/k/^{o}F$ 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

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.

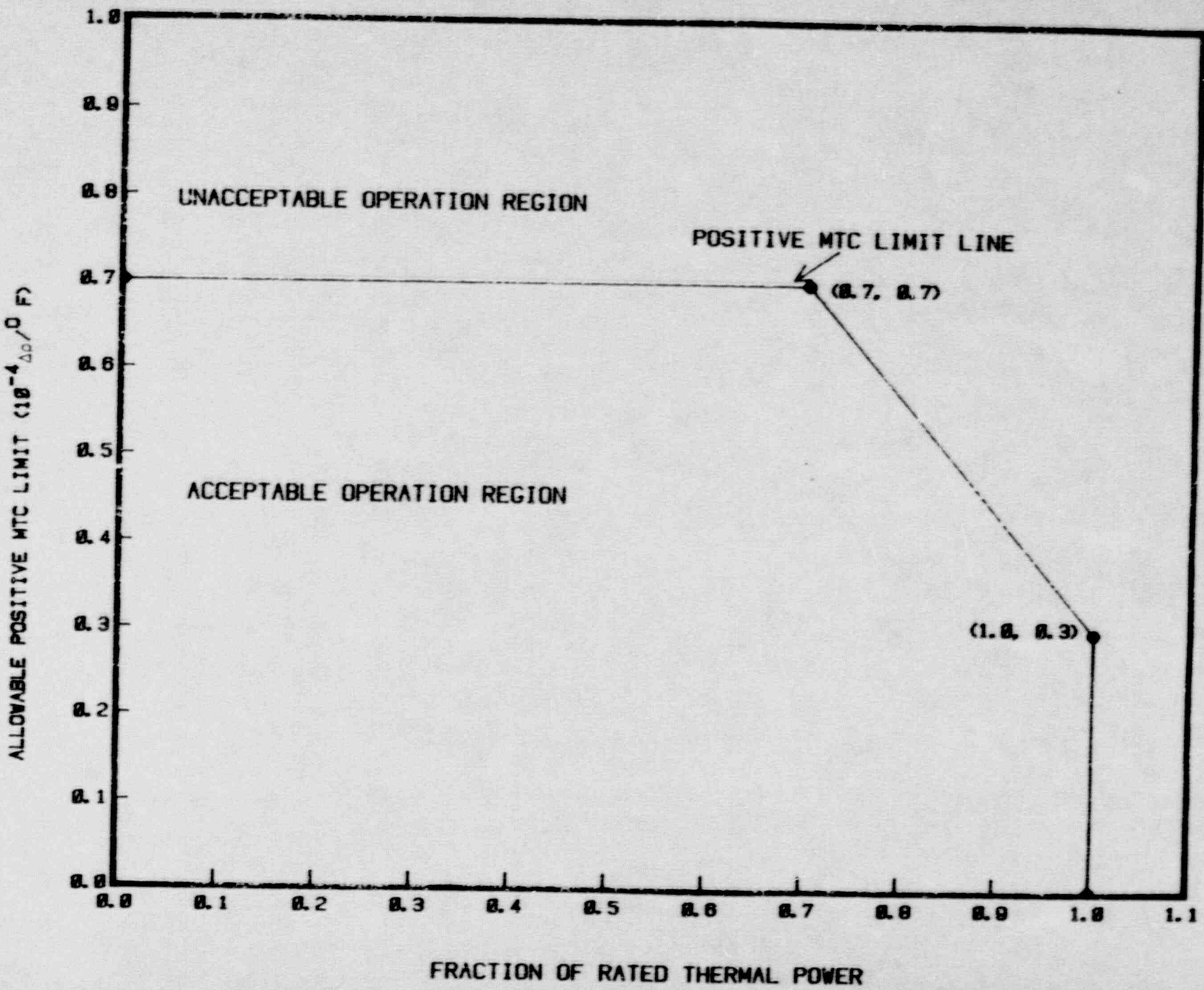
UNACCEPTABLE OPERATION REGION
POSITIVE MTC LIMIT LINE
(0.7, 0.7)
ACCEPTABLE OPERATION REGION
(1.0, 0.3)
ALLOWABLE POSITIVE MTC LIMIT (10^{-4} $\Delta\rho$/°F)
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
FRACTION OF RATED THERMAL POWER

Figure 3.1-1a

SURVEILLANCE REQUIREMENTS (Continued)

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 after each fuel loading.

c. At any THERMAL Power, within 7 EFPD of reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. Two flow paths from the boric acid storage tanks required to be **OPERABLE** pursuant to Specifications 3.1.2.8 and 3.1.2.9 via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System **OPERABLE**, restore at least two boron injection flow paths to the Reactor Coolant System to **OPERABLE** status within 72 hours or be in at least **HOT STANDBY** and borated to a **SHUTDOWN MARGIN** equivalent to at least 3% Δ k/k at 200°F within the next 6 hours; restore at least two flow paths to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated **OPERABLE**:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the concentrated boric acid tanks is above the temperature limit line shown on Figure 3.1-1.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per refueling interval by verifying on a SIAS test signal that:

(1) each automatic valve in the flow path actuates to its correct position, and

(2) each boric acid pump starts.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be **OPERABLE**.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump **OPERABLE**, restore at least two charging pumps to **OPERABLE** status within 72 hours or be in at least **HOT STANDBY** and borated to a **SHUTDOWN MARGIN** equivalent to at least 3% Δ k/k at 200°F within the next 6 hours; restore at least two charging pumps to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated **OPERABLE**:

a. At least once per refueling interval by verifying that each charging pump starts automatically upon receipt of a S fety Injection Actuation Test Signal.

b. No additional Surveillance Requirements other than those required by Specification 4.0.5.

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* Above 80% **RATED THERMAL POWER** the two **OPERABLE** charging pumps shall have independent power supplies.

BORIC ACID PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Motion Inhibit and all shutdown and regulating CEAs shall be **OPERABLE** with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: **MODES** 1* and 2*

ACTION:

a. With one or more CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, be in at least **HOT STANDBY** within 6 hours.

b. With the CEA Motion Inhibit inoperable, within 6 hours either:

1. Restore the CEA Motion Inhibit to **OPERABLE** status, or

2. Place and maintain the CEA drive system mode switch in either the "Off" or any "Manual Mode" position and fully withdraw all CEAs in groups 3 and 4 and withdraw the CEAs in group 5 to less than 5% insertion, or

3. Be in at least **HOT STANDBY**.

c. With one CEA inoperable due to causes other than addressed by **ACTION** a, above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in **MODES** 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year.

d. With one CEA inoperable due to causes other than addressed by **ACTION** a, above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in CEA group 5, operation in **MODES** 1 and 2 may continue.

---

* See Special Test Exceptions 3.10.2 and 3.10.4.

e. With one or more CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

   1. Restored to **OPERABLE** status within its above specified alignment requirements, or

   2. Declared inoperable. After declaring the CEA inoperable, operation in **MODES** 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

      a. The **THERMAL POWER** level shall be reduced to ≤ 70% of the maximum allowable **THERMAL POWER** level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used.

      b. Within one hour after reducing the **THERMAL POWER** as required by (a) above, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the **THERMAL POWER** level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

f. With one CEA misaligned from any other CEA in its group by 15 inches or more, operation in **MODES** 1 and 2 may continue, provided that the misaligned CEA is positioned within 7.5 inches of the other CEAs in its group in accordance with the time allowance determined by the Better Axial Shape Selection System (BASSS) or, if the BASSS time allowance is unavailable, the time allowance shown in Figure 3.1-3. If Figure 3.1-3 is used, the pre-misaligned F_r^T value used to determine the allowable time to realign the CEA from Figure 3.1-3 shall be the latest measurement taken within 5 days prior to the CEA misalignment. If no measurements were taken within 5 days prior to the misalignment, a pre-misaligned F_r^T of 1.70 shall be assumed.

g. With one CEA misaligned from any other CEA in its group by 15 inches or more at the conclusion of the permitted time allowance, immediately start to implement the following actions:

   1. If the **THERMAL POWER** level prior to the misalignment was greater than 50% of **RATED THERMAL POWER, THERMAL POWER** shall be reduced to less than the greater of:

a) 50% of **RATED THERMAL POWER**

b) 75% of the **THERMAL POWER** level prior to the misalignment

within one hour after exceeding the permitted time allowance.

2. If the **THERMAL POWER** level prior to the misalignment was ≤ 50% of **RATED THERMAL POWER**, maintain **THERMAL POWER** no higher than the value prior to the misalignment.

If negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used. Within one hour after establishing the appropriate **THERMAL POWER** as required above, either:

1. Restore the CEA to within the above specified alignment requirements, or

2. Declare the CEA inoperable. After declaring the CEA inoperable, **POWER OPERATION** may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the **THERMAL POWER** level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

h. With more than one CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in at least **HOT STANDBY** within 6 hours.

i. For the purposes of performing the CEA operability test of TS 4.1.3.1.2, if the CEA has an inoperable position indication channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. If a direct position indication (full out reed switch or voltage dividing network) cannot be restored within ten minutes from the commencement of CEA motion, or CEA withdrawal exceeds the surveillance testing insertion by > 7.5 inches, the position of the CEA shall be assumed to have been > 15 inches from its group at the commencement of CEA motion.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each CEA not fully inserted shall be determined to be **OPERABLE** by inserting it at least 7.5 inches at least once per 31 days.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated **OPERABLE** at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

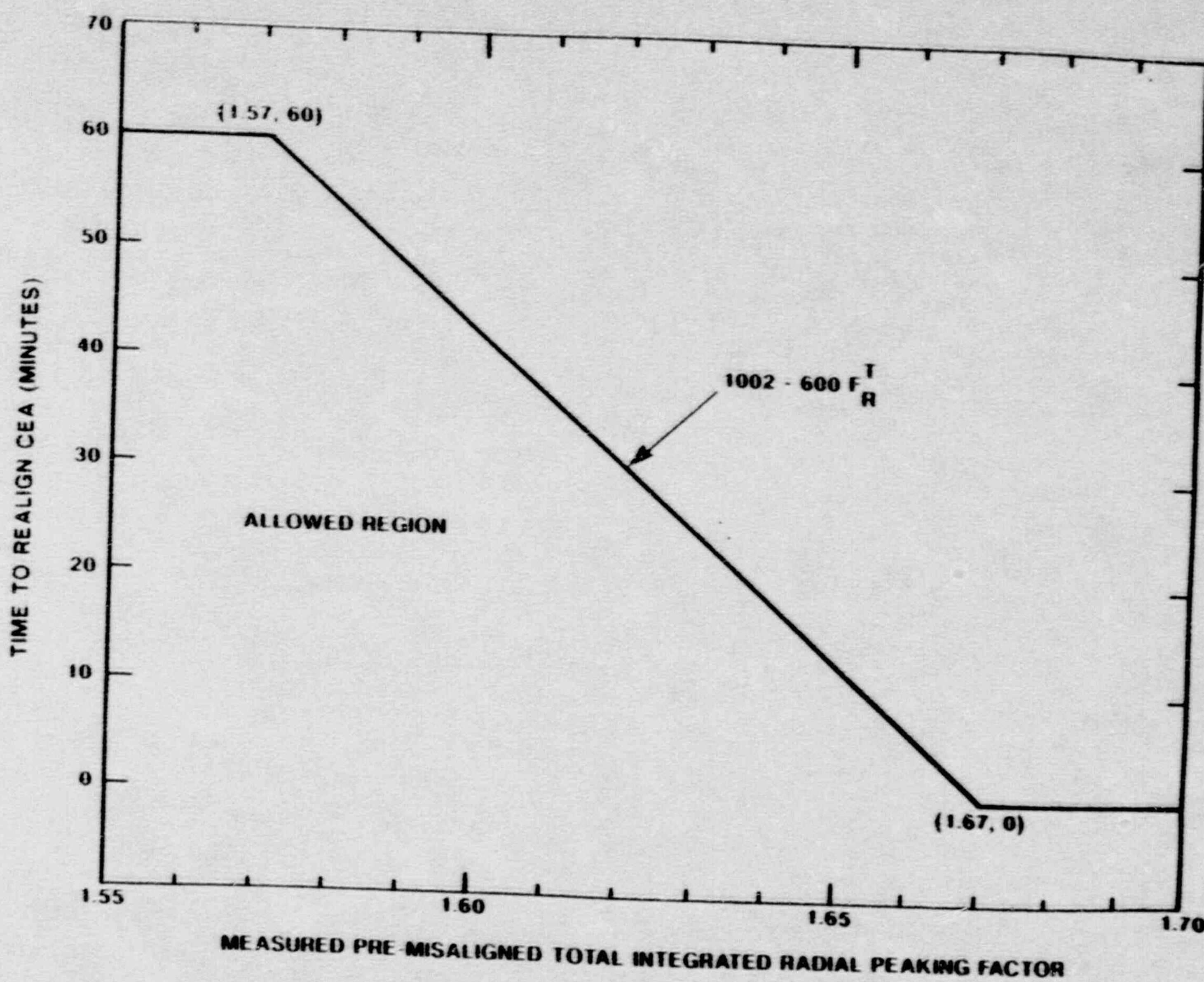

FIGURE 3.1-3

ALLOWABLE TIME TO REALIGN CEA VERSUS
INITIAL TOTAL INTEGRATED RADIAL PEAKING FACTOR

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REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be ≤ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515^{o}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per refueling interval.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

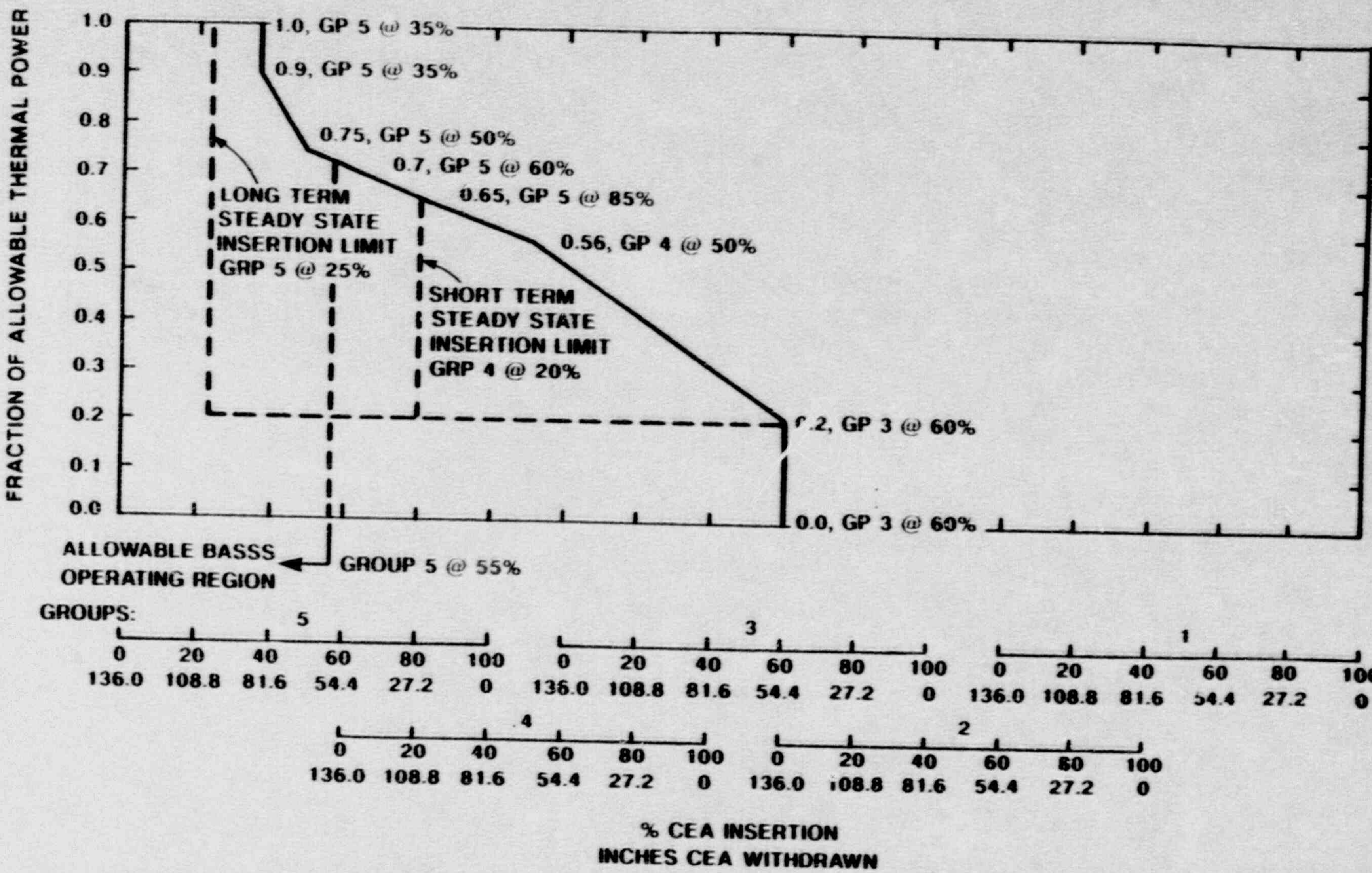


FIGURE 3.1-2

CEA GROUP INSERTION LIMITS vs. FRACTION OF ALLOWABLE THERMAL POWER FOR EXISTING RCP COMBINATION

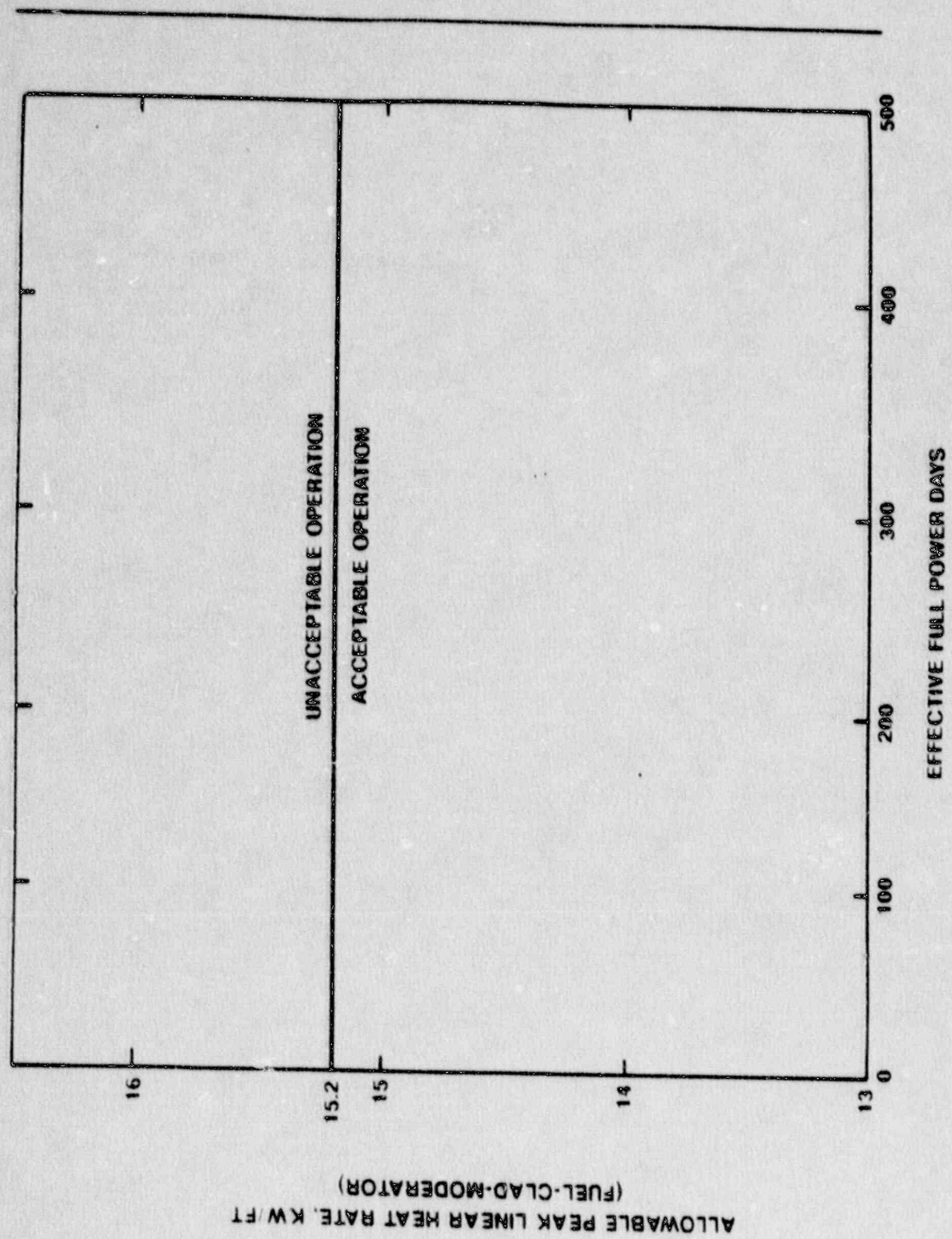


FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE vs. BURNUP

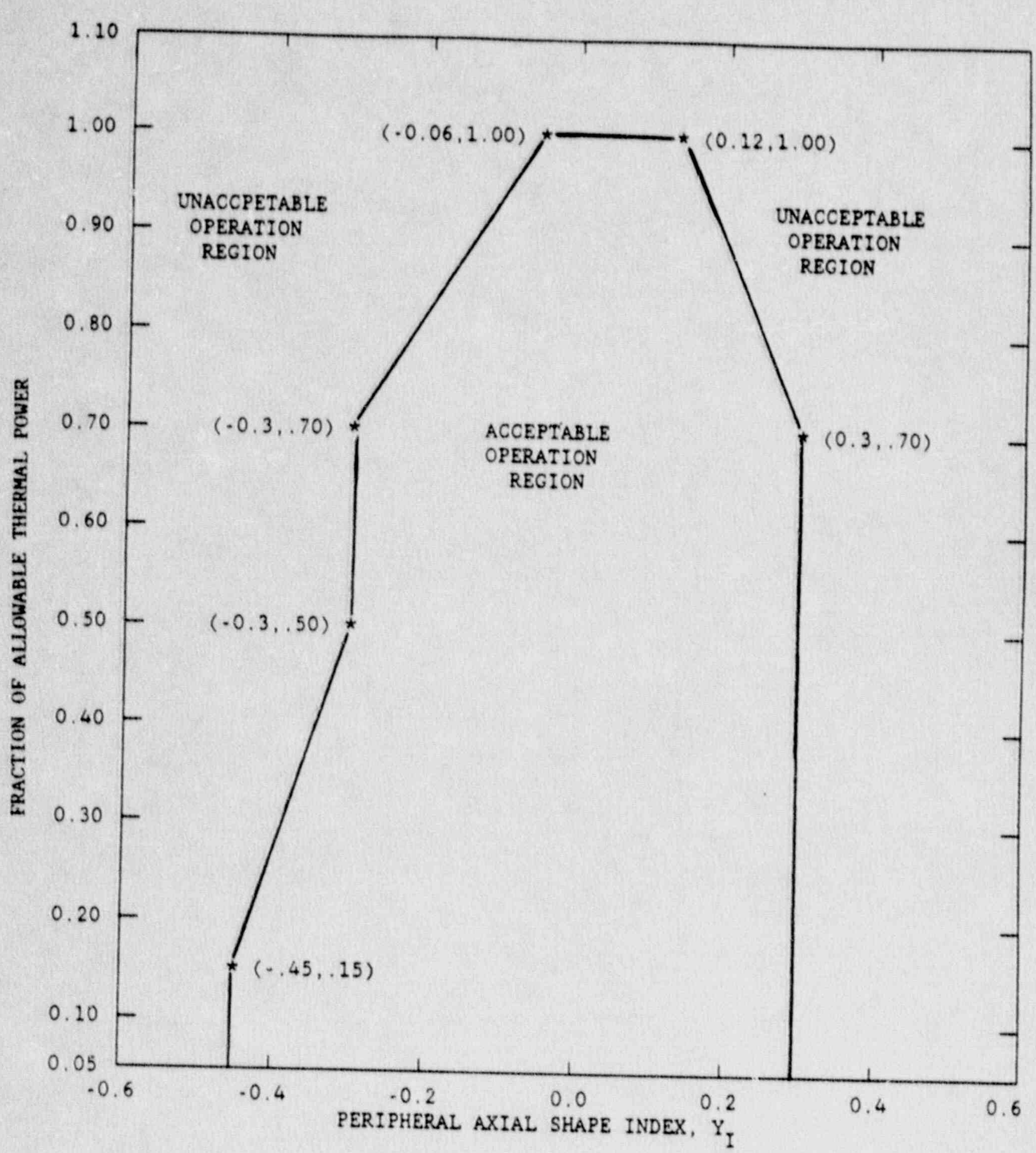


Figure 3.2-2
Linear Heat Rate Axial Flux Offset Control Limits

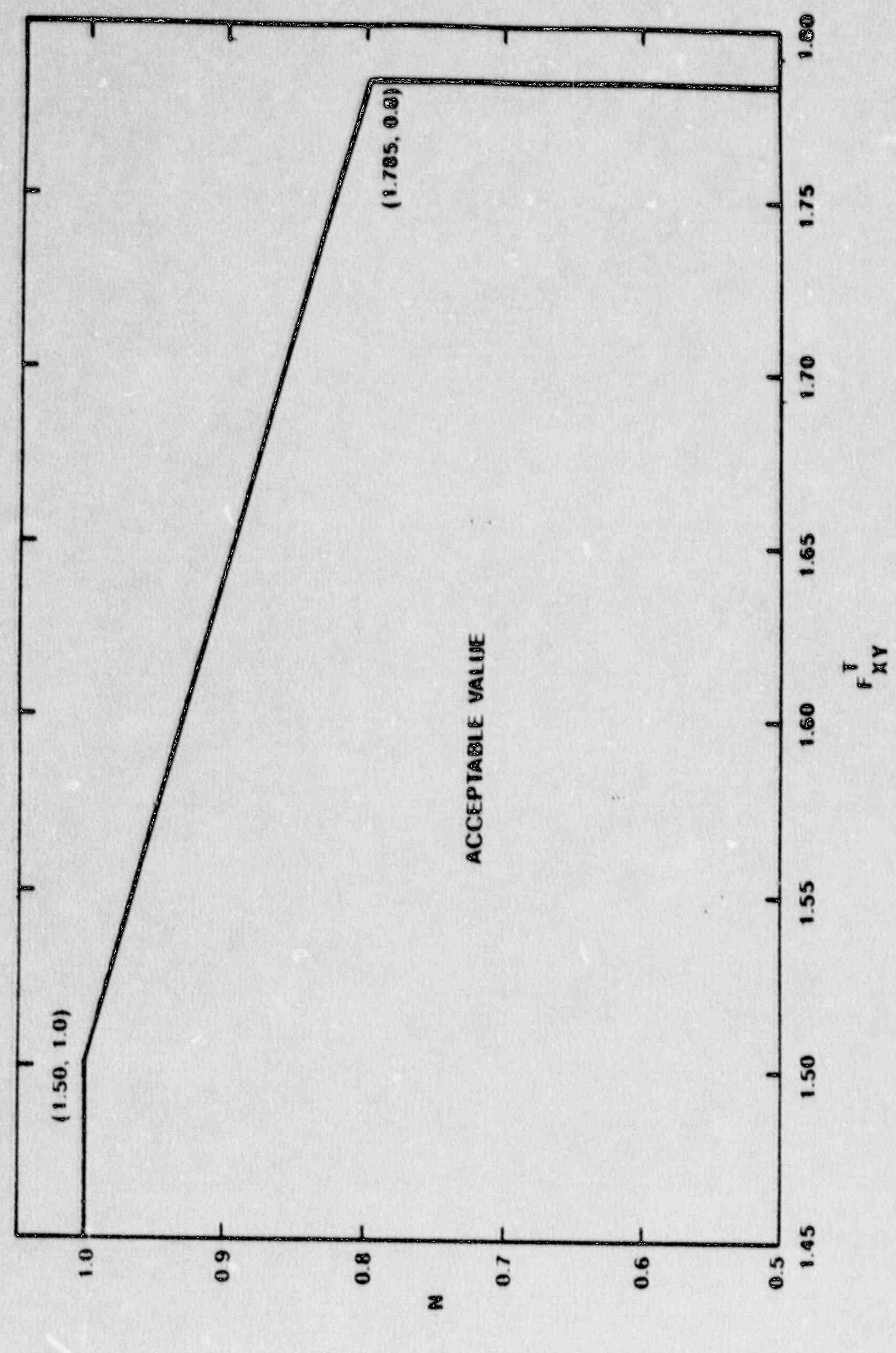


FIGURE 3.2-3b

TOTAL PLANAR RADIAL PEAKING FACTOR vs. N

DELETED

POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F_{xy}^T shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F_{xy}^T > 1.70, within 6 hours either:

a. Withdraw and maintain full length CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce THERMAL POWER as follows:

   1. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^T within the limits of Figure 3.2-3a, or

   2. Reduce THERMAL POWER to less than or equal to the limit established by the Better Axial Shape Selection System (BASSS) as a function of F_{xy}^T; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 F_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}(1+T_q)$ when F_{xy} is determined with a non-full core power distribution mapping system and shall be calculated as $F_{xy}^T = F_{xy}$ when determined with a full core power distribution mapping system. F_{xy}^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

---

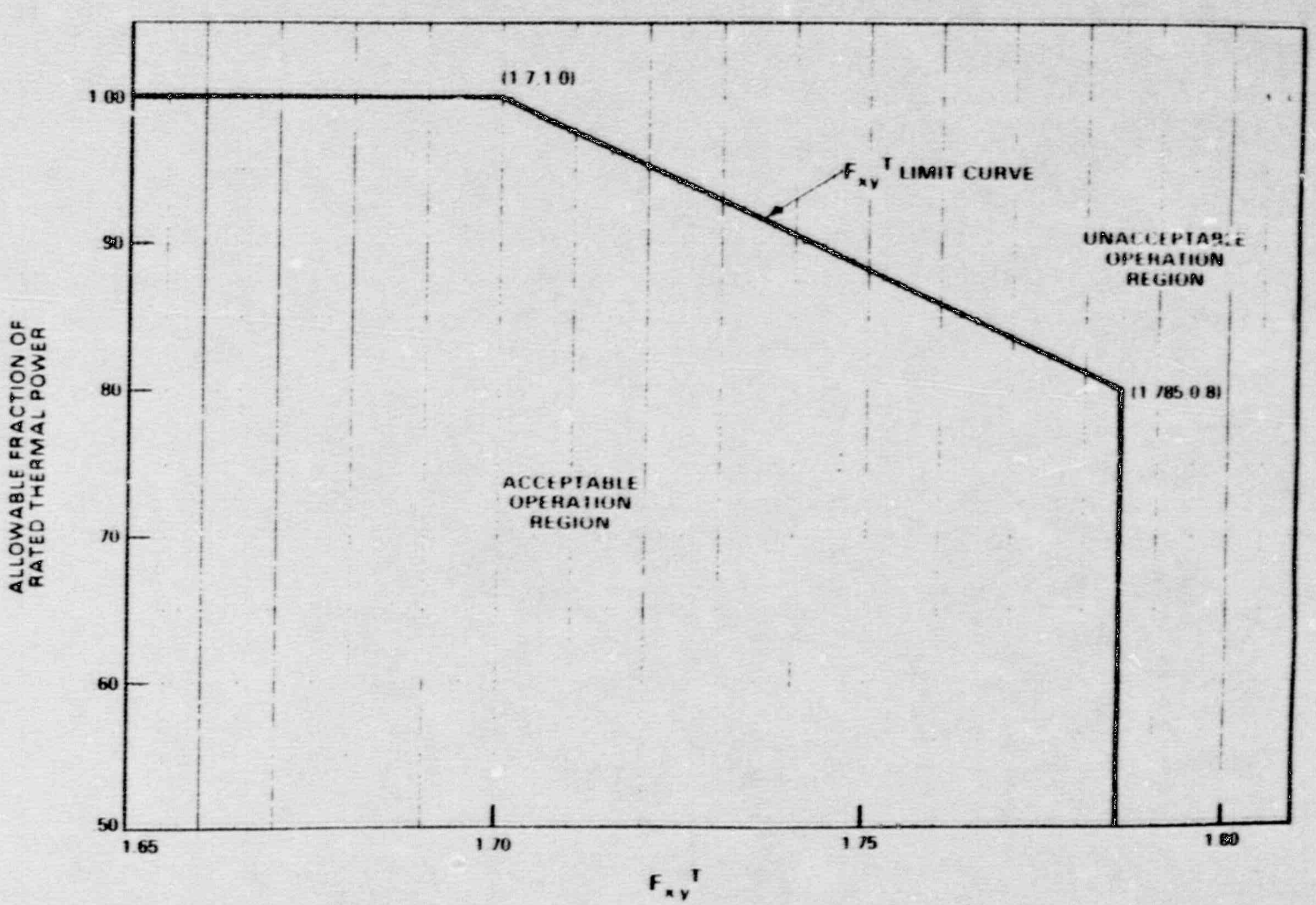
*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.1.3 F_{xy}^{T} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.1.4 T_q shall be determined each time a calculation of F_{xy}^{T} is made using a non-full core power distribution mapping system and the value of T_q used to determine F_{xy}^{T} shall be the measured value of T_q.



TOTAL PLANAR RADIAL PEAKING FACTOR vs ALLOWABLE FRACTION OF RATED THERMAL POWER

Figure 3.2-3a

POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

LIMITING CONDITION FOR OPERATION

3.2.2.2 The value of N presently used in Specification 4.2.1.3 shall be in accordance with Figure 3.2-3b.

APPLICABILITY: MODE 1 when operating in accordance with Specification 4.2.1.3.

ACTION:

With the value of N presently used in Specification 4.2.1.3 exceeding the limit shown in Figure 3.2-3b, within 6 hours either:

a. Reduce the value of N used in Specification 4.2.1.3 to within the limits of Figure 3.2-3b; or

b. Be in at least **HOT STANDBY**.

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}\ (1+T_q)$ when F_{xy} is determined with a non-full core power distribution mapping system and shall be calculated as $F_{xy}^T = F_{xy}$ when determined with a full core power distribution mapping system. N shall be determined to be within its limit by monitoring F_{xy}^T at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

b. At least once per 3 days of accumulated operation in **MODE** 1.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2.3 F_{xy}^{T} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.2.4 T_q shall be determined each time a calculation of F_{xy}^{T} is made using a non-full core power distribution mapping system and the value of T_q used to determine F_{xy}^{T} shall be the measured value of T_q.

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.70, within 6 hours either:

a. Be in at least **HOT STANDBY**, or

b. Withdraw and maintain the full length CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

   1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F_r^T within the limits of Figure 3.2-3c, or

   2. Reduce **THERMAL POWER** to less than or equal to the limit established by the Better Axial Shape Selection System (BASSS) as a function of F_r^T.

   When the **THERMAL POWER** is determined from Figure 3.2-3c, it shall be used to establish a revised upper **THERMAL POWER LEVEL** limit on Figure 3.2-4 (i.e., Figure 3.2-4 shall be truncated at the allowable fraction of **RATED THERMAL POWER** determined by Figure 3.2-3c). Subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r\ (1+T_q)$ when F_r is determined with a non-full core power distribution mapping system and shall be calculated as $F_r^T = F_r$ when determined with a full core power distribution mapping system. F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

b. At least once per 31 days of accumulated operation in **MODE** 1, and

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r^T shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is made using a non-full core power distribution mapping system and the value of T_q used to determine F_r^T shall be the measured value of T_q.

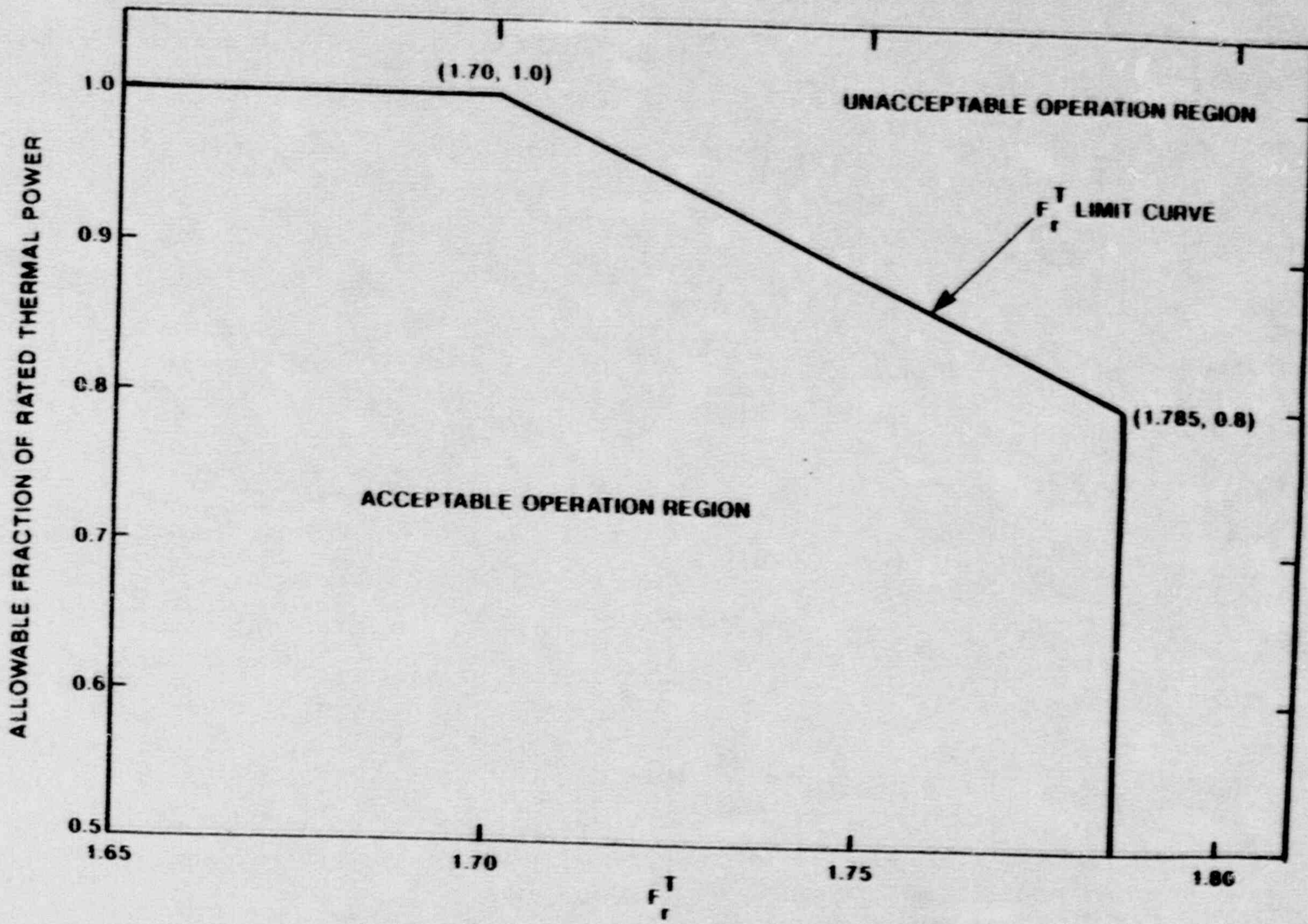


FIGURE 3.2-3c

TOTAL INTEGRATED RADIAL PEAKING FACTOR vs. ALLOWABLE FRACTION OF RATED THERMAL POWER

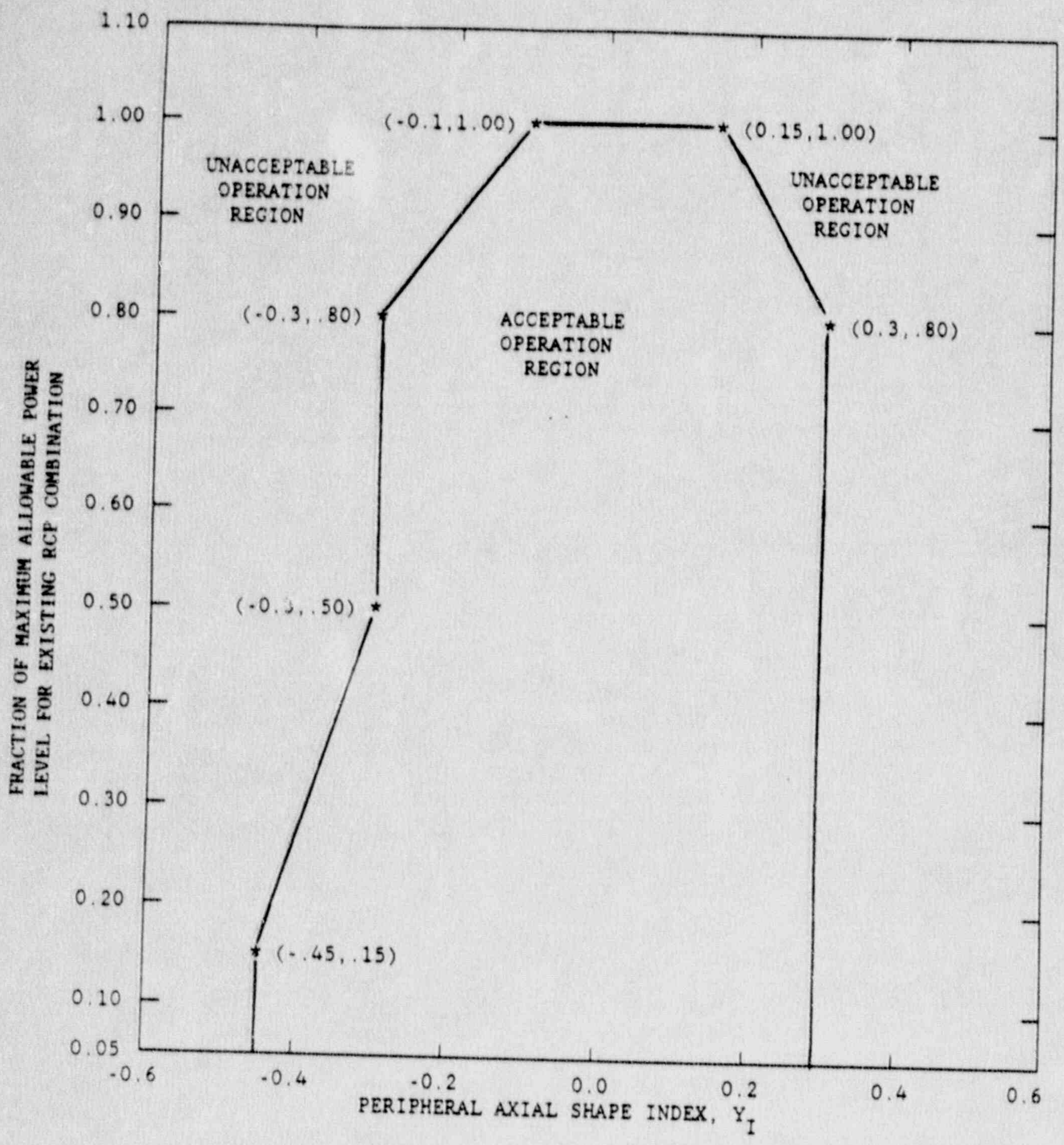


Figure 3.2-4
DNB Axial Flux Offset Control Limits

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 ≤ 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^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) 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^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) 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 ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENTS

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.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. **AXIAL SHAPE INDEX, THERMAL POWER**

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce **THERMAL POWER** to less than 5% of **RATED THERMAL POWER** within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

4.2.5.3 The Better Axial Shape Selection System (BASSS) may be used for monitoring **THERMAL POWER** as a function of **AXIAL SHAPE INDEX**. BASSS monitoring shall be limited to CEA insertions of the lead bank ≤ 55%.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING	THREE REACTOR COOLANT PUMPS OPERATING	TWO REACTOR COOLANT PUMPS OPERATING - SAME LOOP	TWO REACTOR COOLANT PUMPS OPERATING-OPPOSITE LOOP
Cold Leg Temperature	≤ 548° F	**	**	**
Pressurizer Pressure	≥ 2200 psia*	**	**	**
Reactor Coolant System Total Flow Rate	≥ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX, THERMAL POWER	***	**	**	**

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

*** The AXIAL SHAPE INDEX, THERMAL POWER shall be maintained within the limits established by the Better Axial Shape Selection System (BASSS) for CEA insertions of the lead bank of ≤ 55% when BASSS is OPERABLE, or within the limits of FIGURE 3.2-4 for CEA insertions specified by FIGURE 3.1-2.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

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 any time 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

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.

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 a 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 maldistribution 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 SHUTDOWN 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.

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.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 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 or as determined by BASSS) 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_r^T as specified in Technical Specification 3.2.3.

2. When the actual F_r^T is less than the Technical Specification value, additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^T 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 or the time limit determined by BASSS is reached offsets the continuing increase in F_r^T 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.

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 accident analyses. Measurements with $T_{avg} \geq 515°$ 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.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAP INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 3) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) a measurement-calculational uncertainty factor of 1.062, 2) an engineering uncertainty factor of 1.03, 3) an allowance of 1.002 for axial fuel densification and thermal expansion, and 4) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy} or F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An **AZIMUTHAL POWER TILT** > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy} (1 + T_q)$ and $F_r^T = F_r (1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of **RATED THERMAL POWER** provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNB SAFDL of 1.15 throughout each analyzed transient.

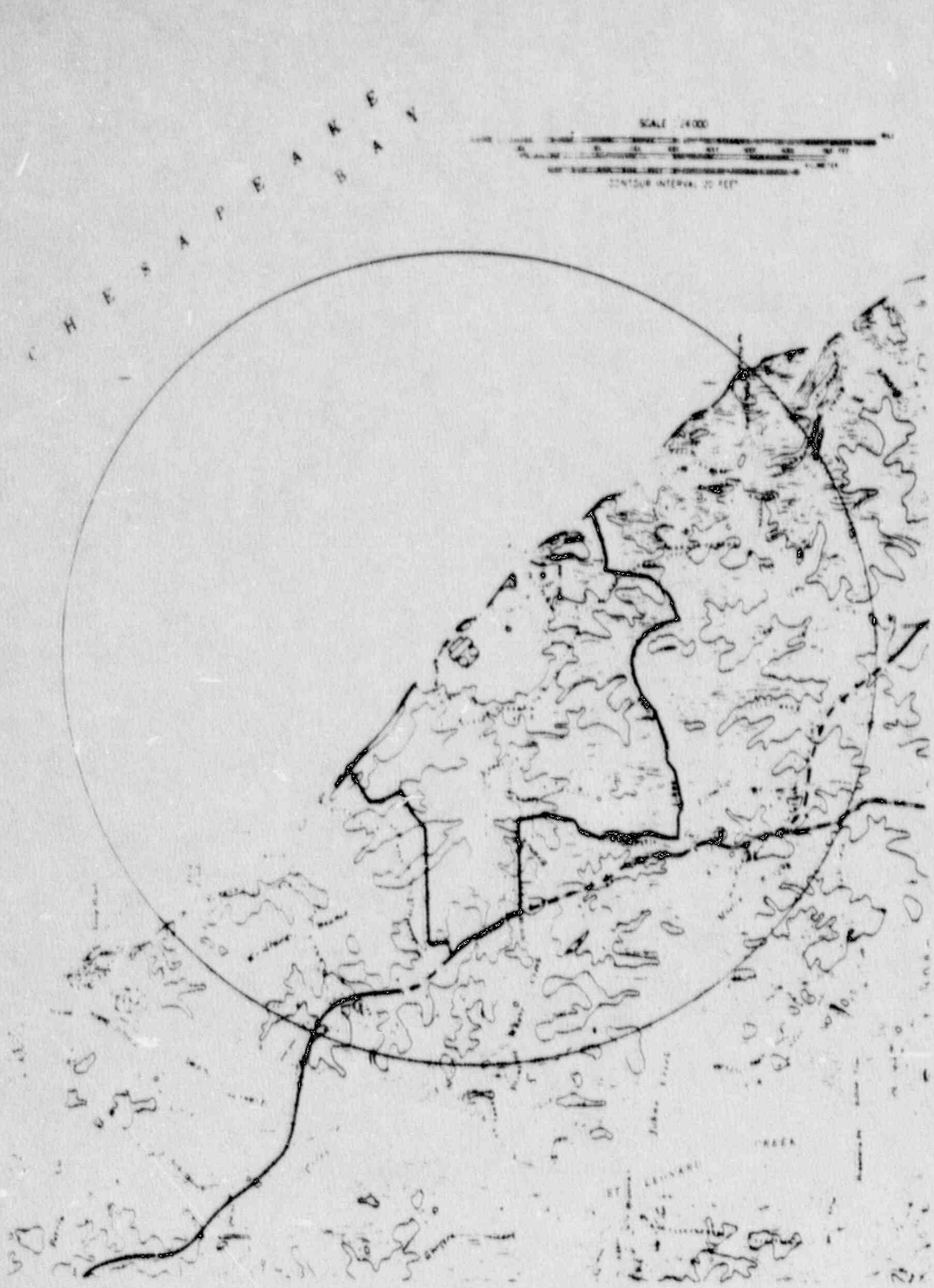
In addition to the DNB criterion, there are two other criteria which set the specification in Figure 3.2-4. The second criterion is to ensure that the existing core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative **AXIAL SHAPE INDEX** value at full power. The third criterion is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). Figure 3.2-4 is used to assure the LHR criterion for this condition because the linear heat rate LCO, for both ex-core and in-core monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.7.1.2 (Continued)

(3) Main Steam Line Break — 1550 gpm Auxiliary Feedwater Flow (this being the maximum flow through the AFW suction line, with one unit requiring flow, prior to pump cavitation due to low NPSH).

At 10 minutes after an Auxiliary Feedwater Actuation Signal, the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required by the existing plant condition.



LOW POPULATION ZONE
FIG. 5.1-2

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.35 weight percent U-235.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.3 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer, which is 700°F.