May 16, 1988

Docket No. 50-317

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Mr. J. A. Tiernan
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Tiernan:

SUBJECT: UNIT 1 CYCLE 10 TECHNICAL SPECIFICATION AMENDMENT (TAC 67143)

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-53 for the Calvert Cliffs Power Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated February 12, 1988 as supplemented on March 21, March 25 (2 letters) and April 14, 1988.

The amendment would make the following changes:

1. Modify Technical Specification (TS) Limiting Condition for Operation (LCO) 3.1.1.4 by adding a figure that provides the upper limits for moderator temperature coefficient (MTC) and increases this MTC limit for thermal power levels above 70% rated thermal power (RTP) from less positive than 0.2 E-4 delta k/k/°F to the linear equation where the MTC limit is less positive than +[(.9 + 4 (1-P))/3] E-4 delta k/k/°F where P is the fraction of RTP. Thus at 70% RTP, MTC must be less positive than +0.7 E-4 delta k/k/°F and at 100% RTP MTC must be less positive than +0.3 E-4 delta k/k/°F.

2. Increase the minimum required shutdown margin of TS LCO 3.1.1.1 above the currently required +3.5 delta k/k in accordance with the linear progression where the shutdown margin limit shall be greater than or equal to +[3.5 + 1.5(P)] delta k/k/ where P is the fraction of core cycle life. Thus at beginning of cycle, the shutdown margin limit is +3.5 delta k/k but at end of cycle the limit is +5.0 delta k/k.

3. Change the TS Figure 3.1-2, "CEA Group Insertion Limits vs. Fraction of Allowable Thermal Power for Existing RCP Combination," Bank 5 Transient Insertion Limit from the linear progression with values of 25% insertion at 90% RTP and 35% insertion at 100% RTP to a constant insertion limit of 35% between 90% and 100% RTP.

4. Reduce unnecessary Axial Shape Index (ASI) trips below 70% RTP and provide additional operation flexibility by:

   a. modifying TS Figure 2.2-1, "Peripheral Axial Shape Index vs. Fraction of Rated Thermal Power," by increasing the acceptable operation region below 70% RTP to the area bounded by the linear equations for the ASI limits, where:

(1) ASI limit = ±[.6 + 2/3 (.4-P)] (P is the fraction of RTP) between 40% and 100% RTP, and
(2) ASI limit = ±0.6 at powers below 40% RTP.

The current ASI limits are ±0.4 at power below 70% RTP;

b. expanding the acceptable operation region of TS Figure 3.2-2, "Linear Heat Rate Axial Flux Offset Control Limits," and TS Figure 3.2-4, "DNB Axial Flux Offset Control Limits," by increasing the negative ASI limit below 50% RTP from the current value of -0.3 to

(1) between 15% and 50% RTP, the linear equation for the negative ASI limit = -[0.3 + 3/7 (.5-P)], where P is the fraction of RTP;
(2) below 15% RTP, the negative ASI limit = -0.45.

5. Reflect the lowering of the departure from nucleate boiling ratio (DNBR) limit to 1.15 due to the incorporation of an extended statistical combination of uncertainties methodology through modifying Figures 2.2-2, "Thermal Margin/Low Pressure Trip Setpoint Part 1 (ASI v. A_1)," and 2.2-3, "Thermal Margin/Low Pressure Trip Setpoint Part 2 (Fraction of Rated Thermal Power v. QR_1)," by

a. changing the equation for the pressure variable trip from
P (TRIP VAR) = 2061 (Q_{DNB}) + 15.85 (T_{IN}) - 8915 to
P (TRIP VAR) = 2892 Q_{DNB} + 17.16 (T_{IN}) - 10682;

b. changing Q_{DNB}, which equals $QR_1 \times A_1$, by increasing QR_1 from the values of:
QR_1 = .235 + (628/7810)P between 0% and 78.1% RTP
QR_1 = .863 + (109/191)x(P-.781) between 78.1% and 97.2% RTP
QR_1 = P above 97.2% RTP
to
QR_1 = .3 + (11/12)P between 0% and 60% RTP
QR_1 = .85 + (3/8)x(P-.6) between 60% and 100% RTP
QR_1 = P above 100% RTP
where P is the fraction of RTP.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Scott Alexander McNeil, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 130 to DPR-53
2. Safety Evaluation
cc: See next page
* SEE PREVIOUS CONCURRENCE

w/noted revising ck SECY/issuance

PDI-1	PDI-1	OGC	PDI-1
CVogan 5/9/88	SMcNeil:mak sam	MYoung	RCapra RAC
5/02/88	5/9/88	5/12/88	5/16/88

Mr. J. A. Tiernan
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

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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:

(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

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	Insert Pages
2-11	2-11
2-12	2-12
2-13	2-13
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 1-27	3/4 1-27
3/4 2-3*	3/4 2-3*
3/4 2-4	3/4 2-4
3/4 2-11	3/4 2-11
3/4 2-12*	3/4 2-12*
B2-1	B2-1
B2-2*	B2-2*
B2-3	B2-3
B2-4*	B2-4*
B2-5	B2-5
B2-6	B2-6
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*

*Overleaf pages provided to maintain document completeness.

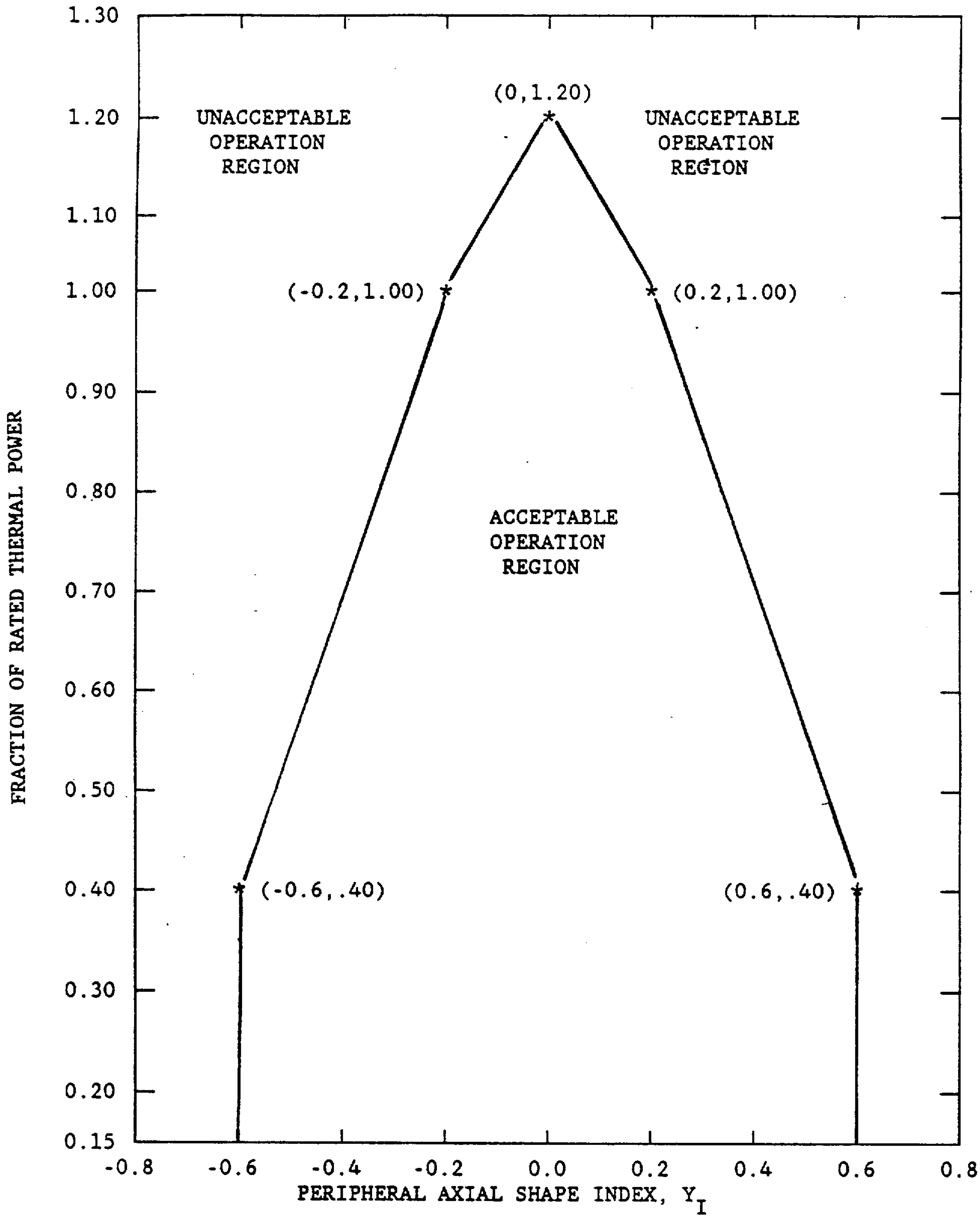

Figure 2.2-1
Peripheral Axial Shape Index, Y_I vs Fraction of Rated Thermal Power

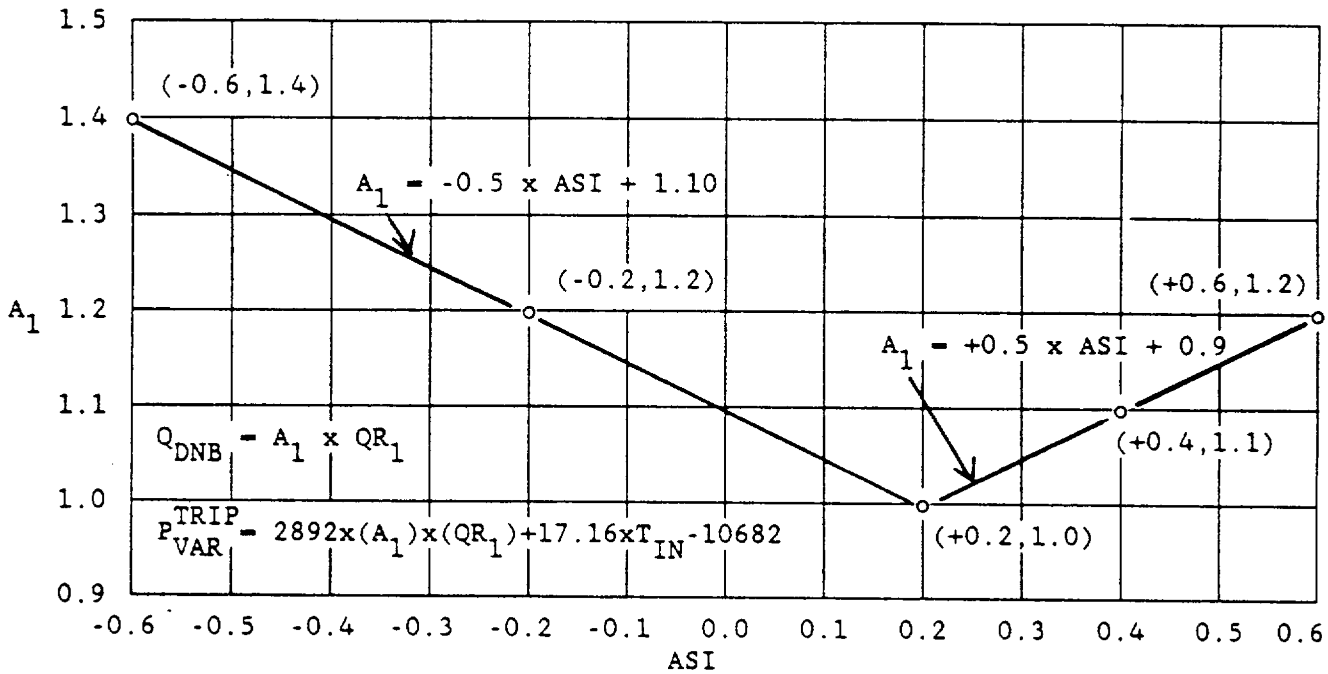

FIGURE 2.2-2

Thermal Margin/Low Pressure Trip Setpoint
Part 1 (ASI Versus A_1)

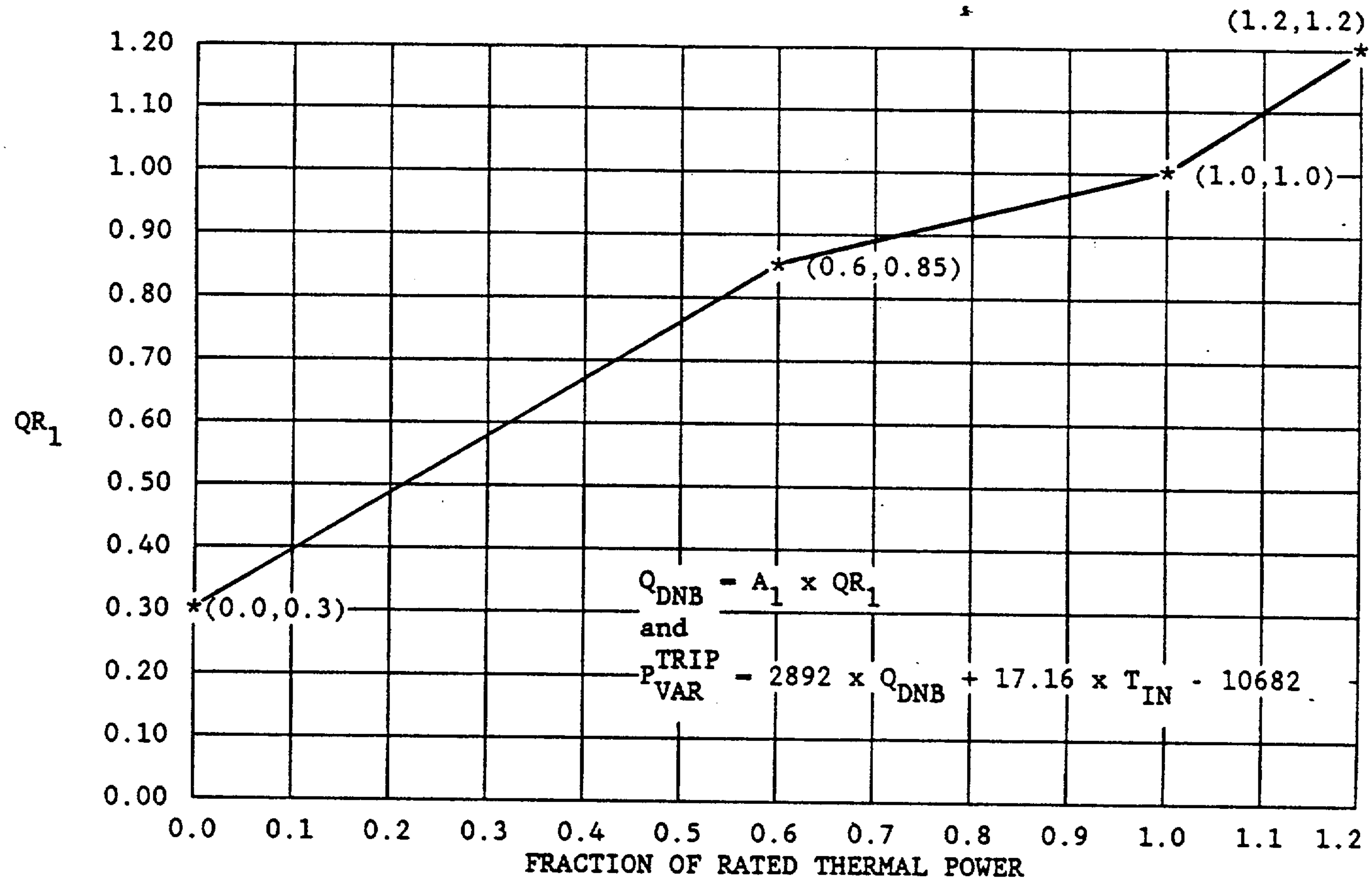

FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint

Part 2 (Fraction of RATED THERMAL POWER versus QR_1)

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - Tavg > 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 $\geq$ 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 $K_{eff} < 1.0$

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.

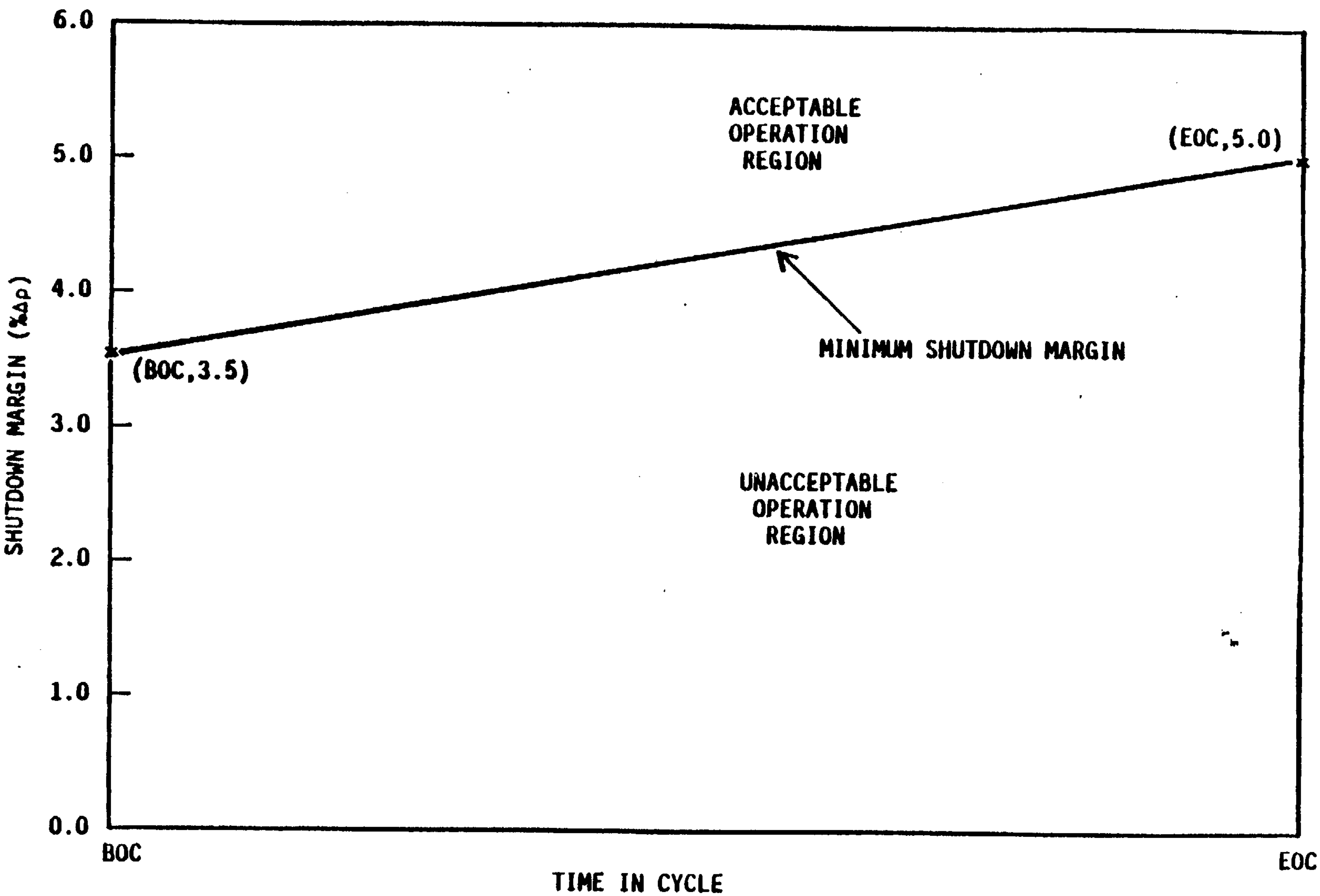
6.0
5.0
4.0
3.0
2.0
1.0
0.0
SHUTDOWN MARGIN (%Δρ)
ACCEPTABLE OPERATION REGION
(EOC,5.0)
(BOC,3.5)
MINIMUM SHUTDOWN MARGIN
UNACCEPTABLE OPERATION REGION
BOC
EOC
TIME IN CYCLE

Figure 3.1-1b

SHUTDOWN MARGIN - $T_{avg} \leq 200^{0}F$

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 ≥ 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 ≥ 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 ≥ 3000 gpm through the reactor coolant 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.

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

\# See Special Test Exception 3.10.2.

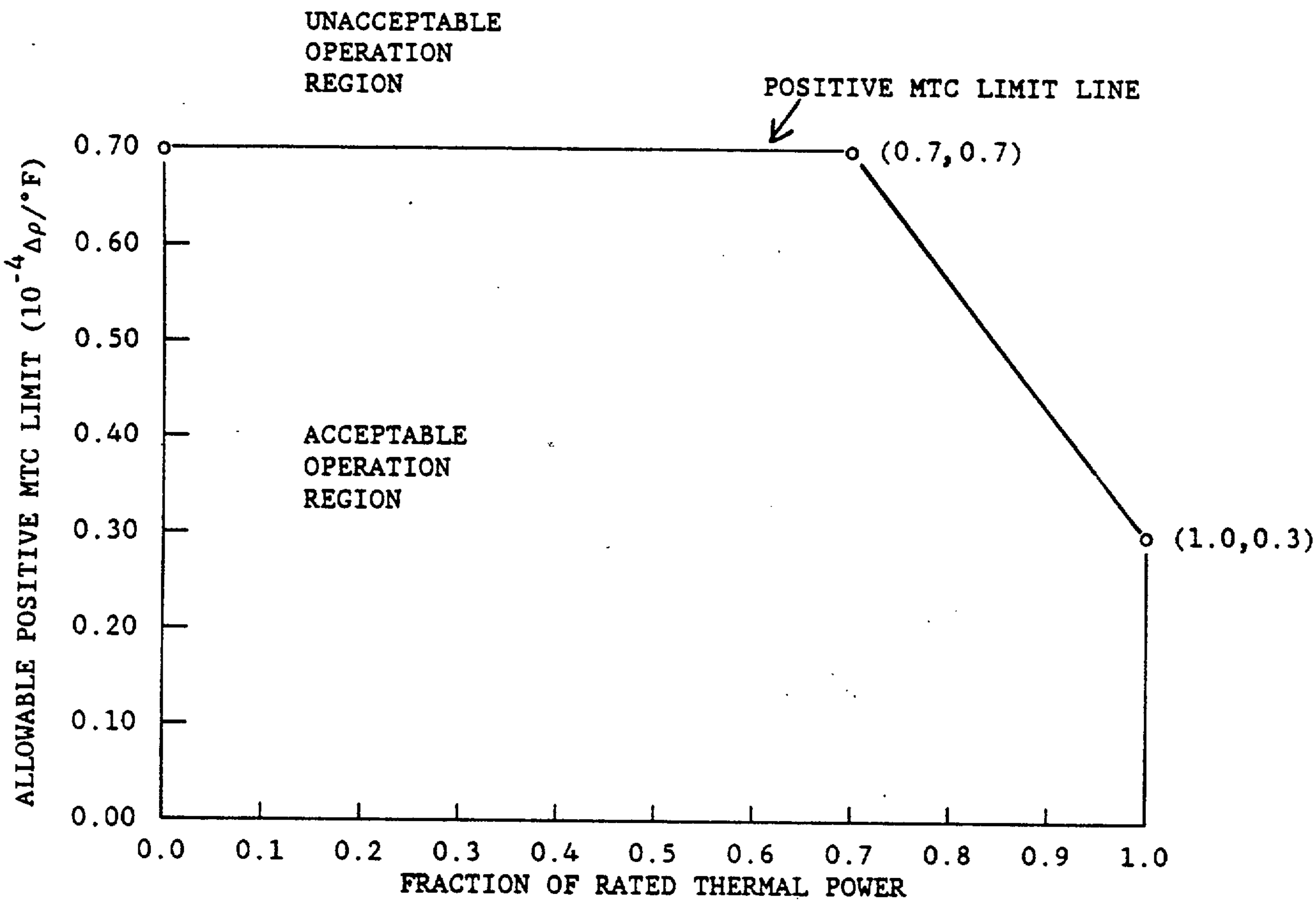

FIGURE 3.1-1a

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

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.

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

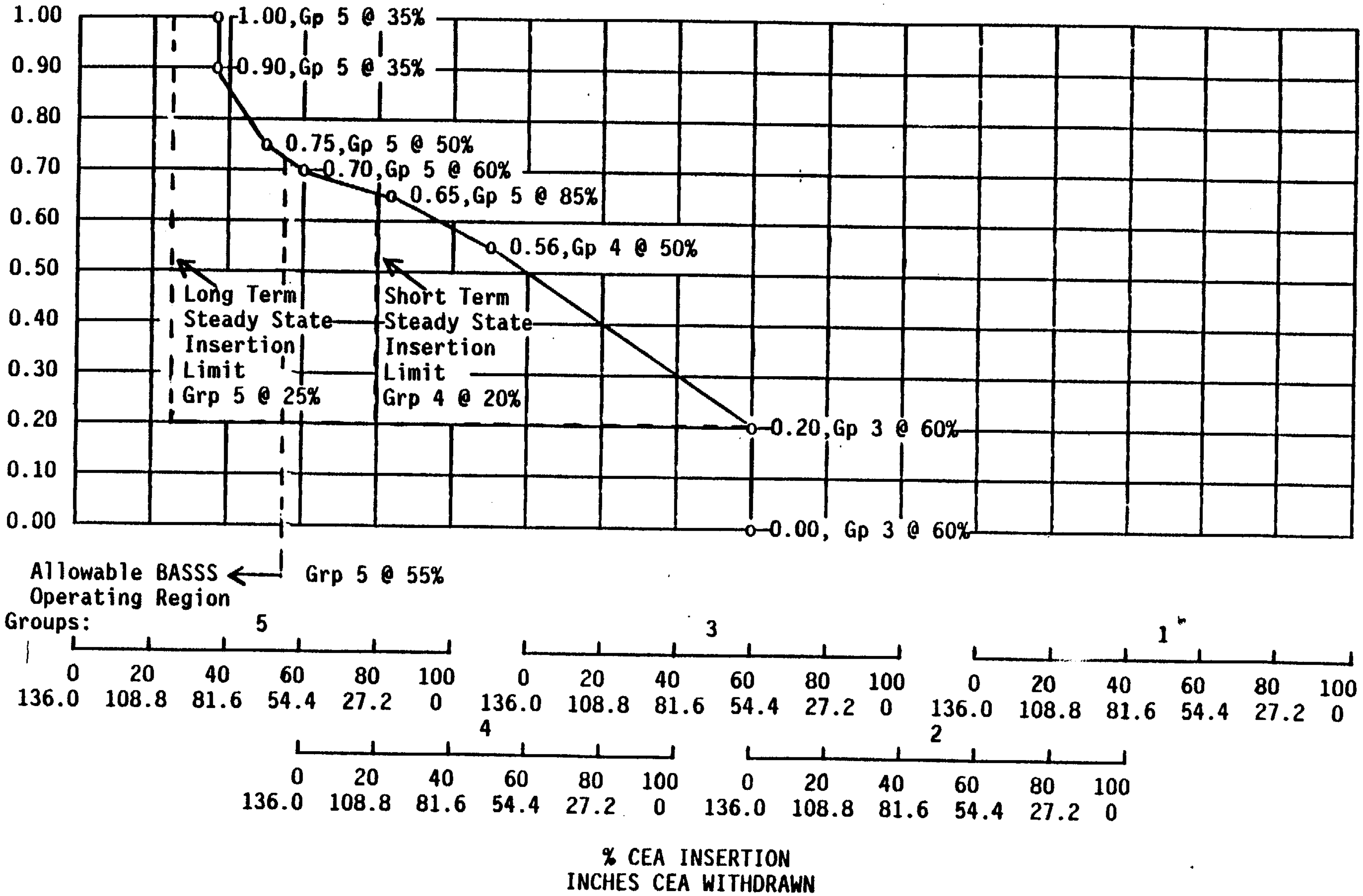

Figure 3.1-2
CEA GROUP INSERTION LIMITS VS. FRACTION OF ALLOWABLE THERMAL POWER
FOR EXISTING RCP COMBINATION

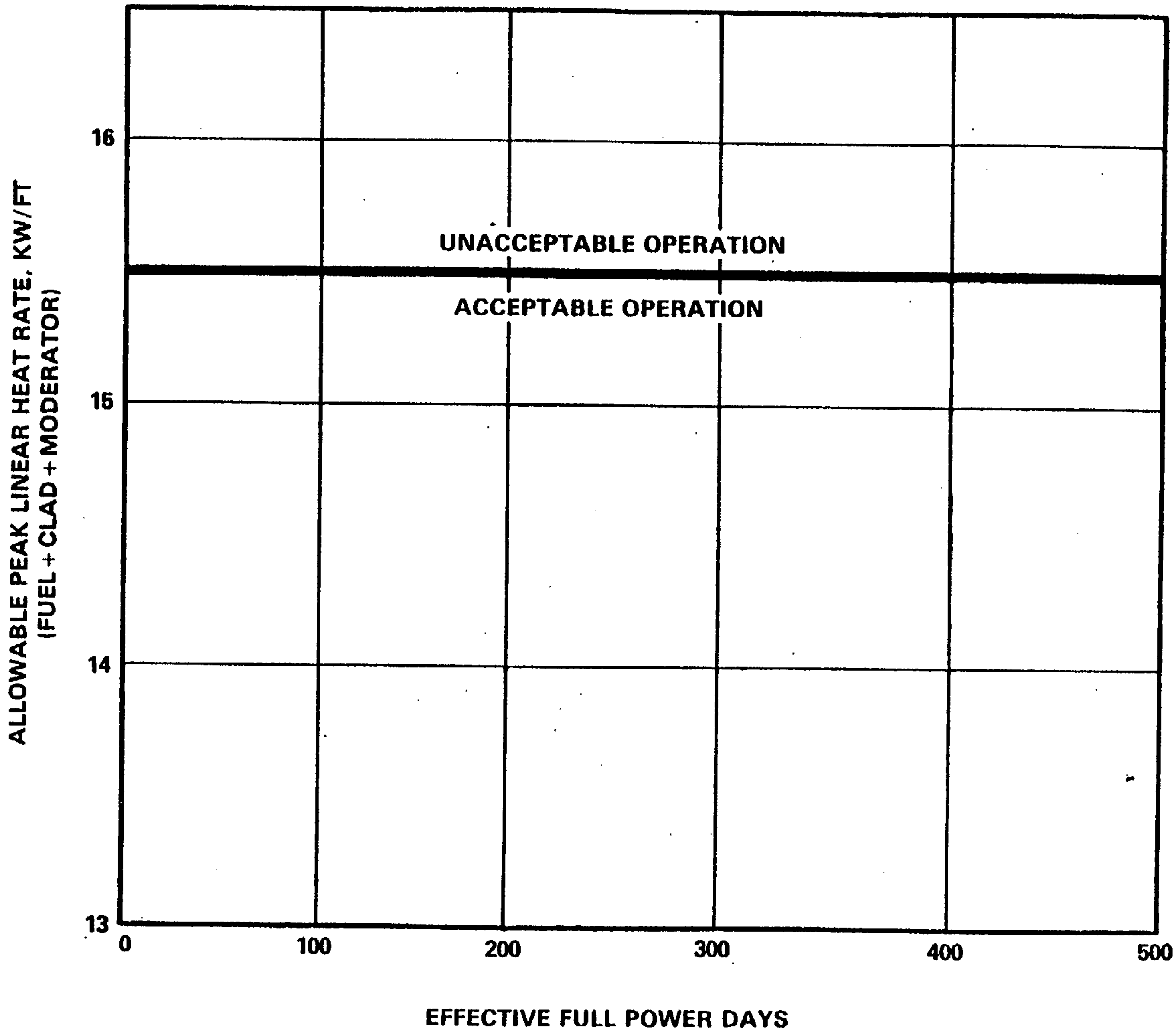

FIGURE 3.2-1

Allowable Peak Linear Heat Rate vs Burnup

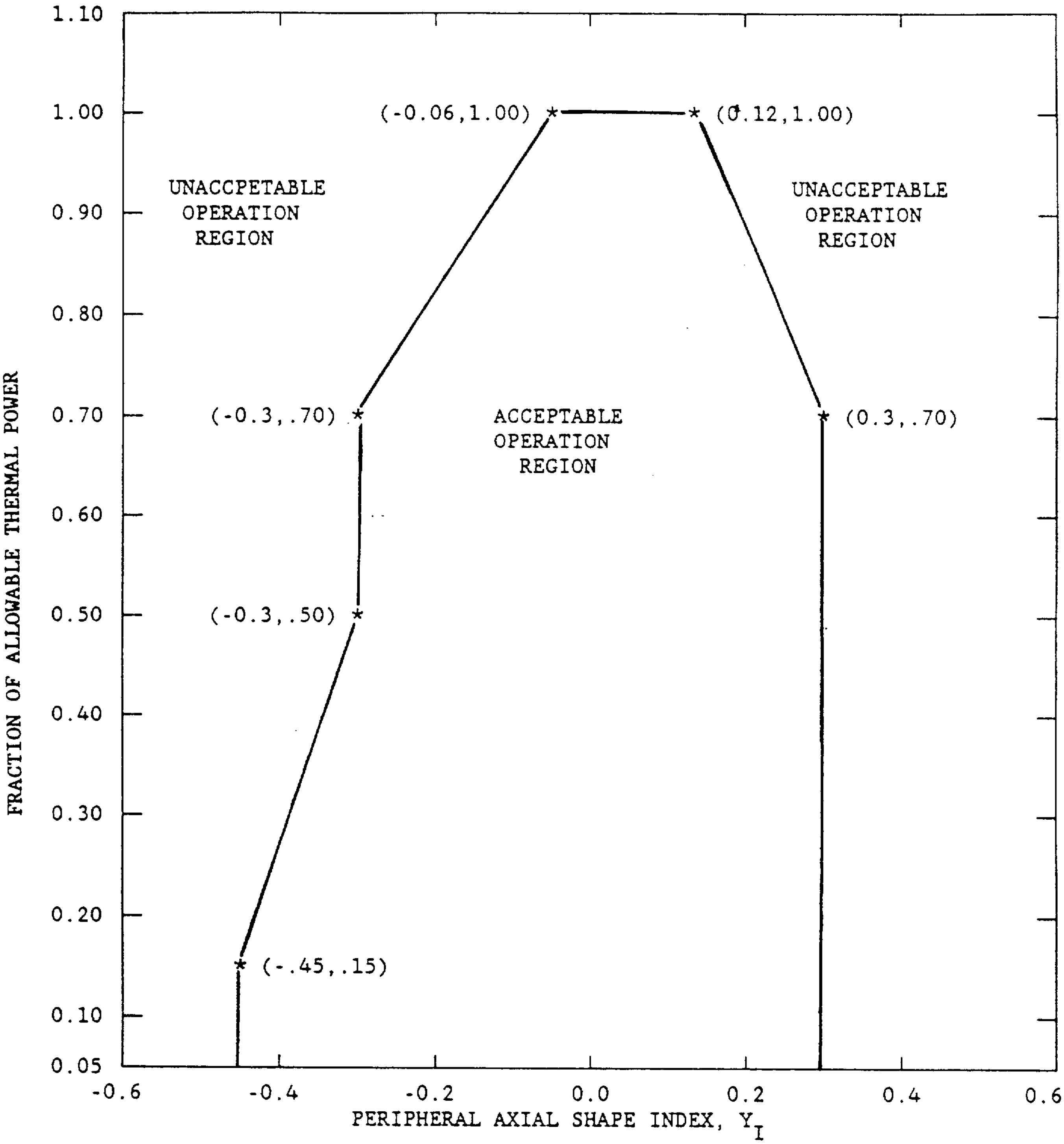

Figure 3.2-2
Linear Heat Rate Axial Flux Offset Control Limits

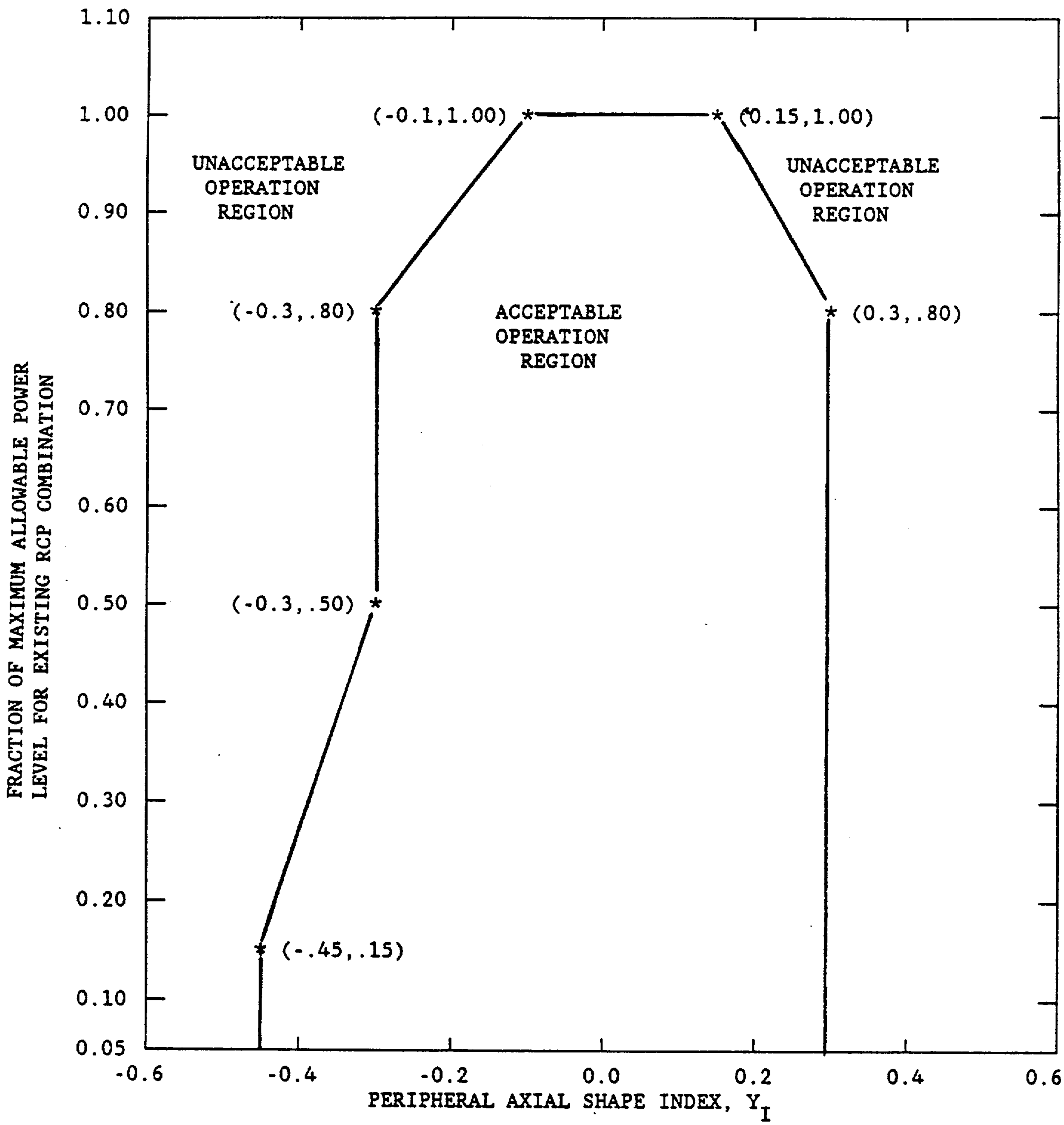

Figure 3.2-4
DNB Axial Flux Offset Control Limits

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^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 $\leq$ 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 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

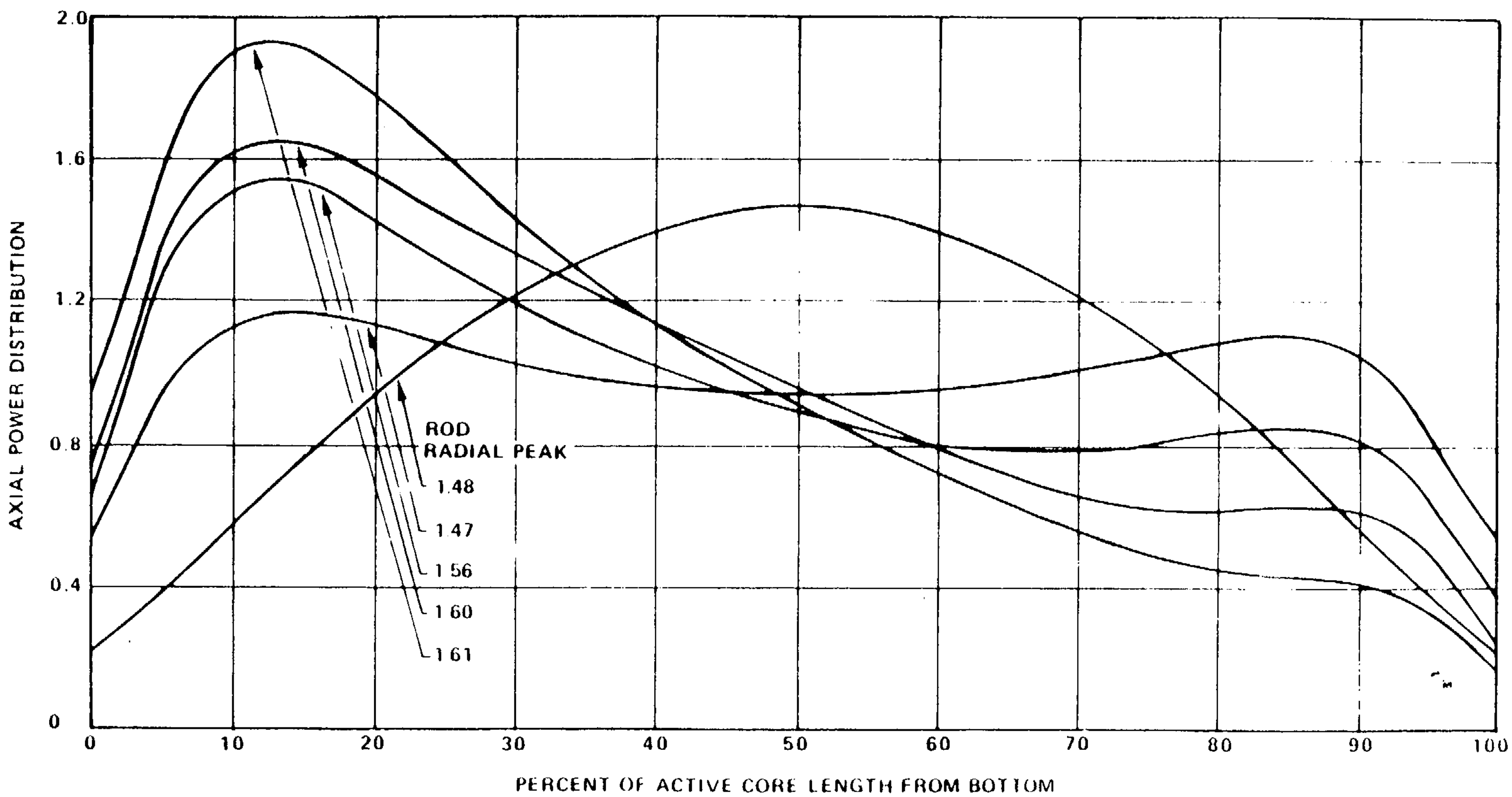

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

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 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

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 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 ± 85 psi which was based on the main steam line break event inside containment.

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 Δ 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

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

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 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 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 ($\geq$ 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_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 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 safety analyses. Measurement with $T_{avg} \geq 515°F$ 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-317

1.0 INTRODUCTION

By letter dated February 12, 1988, as supplemented on March 21, March 25 (2 letters) and April 14, 1988, the Baltimore Gas and Electric Company (BG&E or the licensee) submitted a request for an amendment to its operating license for Calvert Cliffs Unit No. 1 to allow operation for a tenth cycle at a 100% rated core power of 2700 MWt (Ref. 1). The licensee also submitted proposed modifications to the Technical Specifications (TS) for Cycle 10. Cycle 10 will have a 24 month cycle length as compared to 18 months for the previous cycle.

The licensee submitted a final camera-ready copy of the previously requested TS on April 14, 1988.

The supplements to the February 12, 1988 submittal did not affect the proposed TS change noticed in the Federal Register on April 15, 1988, with correction on April 29, 1988, and did not affect the staff's proposed no significant hazards determination.

The NRC staff has reviewed the application and the supporting documents (Refs. 2 & 3) and has prepared the following evaluation of the fuel design, nuclear design, thermal-hydraulic design, and TS changes.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Assembly Description

The Cycle 10 core consists of 217 fuel assemblies. Ninety-six fresh (unirradiated) Batch M assemblies will replace previously irradiated assemblies. Of these 96 fresh assemblies, 92 will be manufactured by Combustion Engineering (CE) and four by Advanced Nuclear Fuels (ANF) Corporation, and are placed in the Cycle 10 core as an aid in qualifying ANF fuel for 24 month cycle operation. The 92 fresh CE assemblies will consist of 16 unshimmed Batch M assemblies and 76 Batch M* assemblies each containing 12 B_4C rods for neutronic shimming and having an initial assembly average enrichment of 4.08 weight percent (w/o) U-235. The four ANF Batch MX demonstration assemblies contain 12 fuel bearing Gd_2O_3 rods for shimming and have an initial assembly average enrichment of 3.85 w/o U-235.

2.2 Mechanical Design

The mechanical design of the CE Batch M reload fuel is identical to the Batch K fuel previously inserted in Calvert Cliffs Unit 1. All CE fuel to be loaded for the Cycle 10 core was reviewed to ascertain that adequate shoulder gap clearance exists. Analyses were performed with approved models and the licensee concluded that all shoulder gap and fuel assembly length clearances are adequate for Cycle 10. The replacement control element assembly (CEA) to be used in the center location of the core will have the same reconstituted features as the replacement CEA installed in the reference cycle.

The mechanical design features of the ANF lead fuel assemblies are described in Reference 3. Most of the assembly and core interface dimensions are identical to the CE fuel assemblies. Differences in the upper and lower end fitting height and overall assembly height should not affect the performance of either fuel assembly. Experience with similar ANF fuel designs co-residing adjacent to CE reload fuel in the Maine Yankee, Fort Calhoun, and St. Lucie Unit 1 cores have caused no unexpected problems or operational difficulties. Therefore, the staff finds the ANF lead assemblies to be mechanically compatible with the co-resident CE fuel during Cycle 10.

2.3 Thermal Design

The thermal performance of the CE fuel in Cycle 10 was evaluated using the FATES3B fuel evaluation model (Ref. 4). The staff issued an SER (Ref. 5) approving the use of FATES3B for BG&E licensing submittals. The licensee analyzed a composite, standard fuel pin that enveloped the various CE fuel batches in Cycle 10. The analysis modeled the power and burnup levels representative of the peak pin at each burnup interval. Although the burnup range analyzed for the peak pin was greater than that expected at the end of Cycle 10, approximately 0.3% of the fuel pins will achieve burnups greater than the 52,000 MWD/T value approved for CE fuel (Ref. 6) if Cycles 9 and 10 are operated to their maximum burnups. In response to the staff's request, the licensee confirmed that these few high burnup pins will be in low power regions of the Cycle 10 core and the maximum pressure within these pins will not reach the nominal reactor coolant system pressure of 2250 psia (Ref. 7).

Evaluations have been performed to show that the four ANF lead assemblies are thermally compatible with the existing CE fuel assemblies and meet the appropriate fuel thermal design criteria required by the staff (Ref. 3).

Based on its review of the information discussed above, the staff concludes that the evaluation of the thermal design of the CE and ANF fuel for Cycle 10 is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The Cycle 10 core consists of 217 fuel assemblies, each having a 14 by 14 fuel rod array. A general description of the core loading is given in Section 2.1 of this SER. The highest U-235 enrichment occurs in the CE Batch M fuel assemblies which contain an assembly average enrichment of 4.08 w/o U-235. The Calvert Cliffs fuel storage facilities have been approved for storage of fuel of maximum enrichment of 4.10 w/o U-235 and, therefore, the fresh Batch M assemblies are acceptable from a fuel storage aspect.

The Cycle 10 core will use a low-leakage fuel management scheme. With the proposed loading, the Cycle 10 reactivity lifetime for full power operation is expected to be 21,400 MWD/T based on a Cycle 9 length of 11,800 MWD/T. The analyses presented by the licensee will accommodate a Cycle 10 length between 20,600 MWD/T and 21,800 MWD/T based on Cycle 9 lengths between 9,800 MWD/T and 11,800 MWD/T.

3.2 Power Distribution

Hot full power (HFP) fuel assembly relative power densities are given in the reload analysis report for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) unrodded configurations. Radial power distributions at BOC and EOC are also given for control element assembly (CEA) Bank 5, the lead regulating bank, fully inserted. These distributions are characteristic of the high burnup end of the Cycle 9 shutdown window and tend to increase the radial power peaking in the Cycle 10 core. The four ANF lead test assemblies were calculated to have maximum pin power peaking at least 10% lower than the maximum pin peaking in the core under all expected Cycle 10 operating conditions. The distributions were calculated with approved methods and include the increased power peaking which is characteristic of fuel rods adjacent to water holes. In addition, the safety and setpoint analyses conservatively include uncertainties and other allowances so that the power peaking values actually used are higher than those expected to occur at any time in Cycle 10. Therefore, the predicted Cycle 10 power distributions are acceptable.

3.3 Reactivity Coefficients

In order to accommodate 24 month cycles, the moderator temperature coefficient (MTC) limit above 70% power is raised from $+0.2\times10^{-4}$ delta rho/° F to a value which varies linearly from $+0.3\times10^{-4}$ delta rho/° F at 100% power to $+0.7\times10^{-4}$ delta rho/° F at 70% power. The staff has previously expressed concern about the positive MTC effect on the generic anticipated transients without scram (ATWS) assumptions and BG&E has stated that they will address the generic ATWS implications, if any, in the future. In the interim, the staff has approved operation for core designs with allowable positive MTC values provided that the MTC becomes negative at 100% power and equilibrium xenon conditions. The licensee has predicted a negative MTC at hot full power, equilibrium xenon conditions of -0.2×10^{-4} delta rho/° F for Cycle 10 and has committed to a full power negative value at equilibrium xenon conditions (Ref. 7).

The Doppler coefficient for Cycle 10 is a best estimate value expected to be accurate to within 15%. These reactivity coefficient values are bounded by the values used in the safety analyses for the reference cycle (Calvert Cliffs Unit 2 Cycle 8). The staff, therefore, finds the values of the MTCs and Doppler coefficients to be acceptable.

3.4 Control Requirements

The CEA worths and shutdown margin requirements at the most limiting time for the Cycle 10 nuclear design. that is, for the EOC, are presented in Reference 7. These values are based on an EOC, hot zero power (HZP), steamline break accident. At EOC 10, the reactivity worth with all CEAs inserted is 9.0% delta rho. An allowance of 1.1% delta rho is made for the stuck CEA which yields the worst results for the EOC HZP steamline break accident. An allowance of 2.0% delta rho is made for CEA insertion in accordance with the power dependent insertion limit (PDIL). The calculated scram worth is the total CEA worth less the worth of the stuck CEA and less the worth of CEA insertion to the PDIL and is 5.9% delta rho. Deducting 0.8% delta rho for physics uncertainty and bias yields a net available scram worth of 5.1% delta rho. Since the TS EOC shutdown margin at zero power is 5.0% delta rho, a margin of 0.1% delta rho exists in excess of the TS shutdown margin. Therefore, sufficient CEA worth is available to accommodate the reactivity effects of the steam line break event at the worst time in core life allowing for the most reactive CEA stuck in the full withdrawn position. The staff concludes that the licensee's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming a stuck CEA that results in the worst reactivity condition for an EOC, HZP steamline break accident. Thus, the control requirements are acceptable.

3.5 Safety Related Data

Other safety related data such as limiting parameters of dropped CEA reactivity worth and the maximum reactivity worth and planar power peaks associated with an ejected CEA for Cycle 10 are identical to the values used in the reference cycle and are, therefore, acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

4.1 DNBR Analysis

Steady state thermal-hydraulic analysis of CE fuel for Cycle 10 is performed using the approved core thermal-hydraulic code TORC and the CE-1 critical heat flux correlation (Ref. 8). The core and hot channel are modeled with the approved method described in CENPD-206-P-A (Ref. 9). The design thermal margin analysis is performed using the fast running variation of the TORC code, CETOP-D (Ref. 10), which has been approved for Calvert Cliffs with the appropriate hot assembly inlet flow starvation factors to assure its conservatism with respect to TORC. The engineering hot channel factors for heat flux, heat input, rod pitch and cladding diameter are combined statistically with other uncertainty factors using the approved extended statistical combination of uncertainties (ESCU) method described in CEN-348(B)-P (Ref. 11) to arrive at an equivalent departure from nucleate boiling ratio (DNBR) limit of 1.15 at a 95/95 probability/confidence level.

DNBR analyses were also performed to assess the performance of the ANF lead assemblies (Ref. 3) using the XCOBRA-III code (Ref. 12) and the ANF approved thermal-hydraulic methodology for mixed fuel cores (Ref. 13). The XNB departure from nucleate boiling correlation (Ref. 14) has been shown to be applicable to co-resident CE and ANF fuel (Refs. 14 & 15) and the staff concludes that it is acceptable to apply it to the mixed Cycle 10 core containing the four ANF lead fuel assemblies. The results indicate that the ANF lead assemblies exhibit higher MDNBRs than the hot CE assembly due to the 5% lower assembly power at which the ANF lead assemblies were simulated. Since the insertion of the ANF lead assemblies does not significantly affect the minimum DNBR (MDNBR) of the hot CE assembly, which establishes the core MDNBR, the staff concludes that the core MDNBR is essentially unchanged by insertion of the four ANF lead assemblies and thus the design criterion on DNBR is satisfied by the mixed core containing ANF lead assemblies. Thus, the results of the DNBR analysis are acceptable.

4.2 Fuel Rod Bowing

The fuel rod bow penalty accounts for the adverse impact on MDNBR of random variations in spacing between fuel rods. The methodology for determining rod bow penalties for Calvert Cliffs was based on the NRC approved methods presented in the CE topical report on fuel and poison rod bowing (Ref. 16). The penalty at 45,000 MWD/T burnup is 0.006 in MDNBR. This penalty is included in the ESCU uncertainty allowance discussed above. For those assemblies with average burnup in excess of 45,000 MWD/T, sufficient margin exists to offset rod bow penalties. The staff, therefore, concludes that the analysis of fuel rod bow penalty is acceptable.

5.0 EVALUATION OF SAFETY ANALYSES

5.1 Non-LOCA Events

For the non-LOCA safety analyses, the licensee has determined that the key input parameters for the transient and accident analyses lie within the bounds of those of the reference cycle (Unit 2 Cycle 8). As noted in Section 6.0, the shutdown margin TS is being changed from a singular value to a variable ranging from 3.5% delta rho at BOC to 5.0% delta rho at EOC. The EOC shutdown margin requirement is determined by the steam line rupture event and a reevaluation of this event at EOC 10 with the revised shutdown margin has indicated that it is less limiting than the reference analysis. The staff, therefore, concludes that the non-LOCA transient and accident events for Cycle 10 are bounded by the reference analyses and, therefore, the results of the non-LOCA safety analysis are acceptable.

5.2 LOCA Events

The large break loss of coolant accident (LOCA) has been reanalyzed for Cycle 10 to demonstrate that a peak linear heat generation rate (PLHGR) of 15.5 kw/ft complies with the acceptance criteria of 10 CFR 50.46 for emergency core cooling systems (ECCS) for light water reactors. The Cycle 10 analysis, as the reference cycle analysis, was performed with the 1985 CE evaluation model which was approved in Reference 17. The Cycle 10 analysis showed that the double ended guillotine pipe break at the pump discharge with a discharge coefficient of 0.6 (0.6 DEG/PD) gave the highest peak clad temperature. Table 8.1-1 of the reload report provides the input parameters for the fuel for Cycle 10 and the reference cycle. Table 8.1-2 presents the results of the analysis for the limiting break for Cycle 10 and the reference cycle. The results for the limiting Cycle 10 break show that (1) the peak clad temperature is 1983° F which is well below the acceptance criterion of 2200° F and (2) the maximum local and core wide oxidation values are 4.14% and less than 0.51%, respectively, and these are well below the acceptance criteria of 17% and 1%, respectively. The analysis considered up to 500 plugged tubes per steam generator and a 40 second safety injection pump response time. Since the Cycle 10 large break LOCA ECCS analysis has shown that both the peak clad temperature and clad oxidation meet the acceptance criteria of 10 CFR 50.46, the operation of Cycle 10 at an allowable PLHGR of 15.5 kw/ft is acceptable.

The licensee reports that analyses have confirmed that small break loss of coolant accident (SBLOCA) results for Calvert Cliffs Unit 1 Cycle 8, which is the reference cycle for SBLOCA, bound the Calvert Cliffs Unit 1 Cycle 10 results. Unlike the large break LOCA analysis, the SBLOCA considered only 100 plugged tubes per steam generator. The increased safety injection pump response time considered in the large break analysis also was not evaluated for the SBLOCA analysis. Since the acceptance criteria for the SBLOCA are met, the operation of Cycle 10 at an allowable PLHGR of 15.5 kw/ft, with up to 100 plugged tubes per steam generator, is acceptable.

6.0 TECHNICAL SPECIFICATIONS

As indicated in the staff's evaluation of the nuclear design, provided in Section 3, the operating characteristics of Cycle 10 were calculated with approved methods. The proposed TS are the results of the cycle specific analyses for, among other things, power peaking and control rod worths. The analyses performed include the implementation of a low-leakage fuel shuffle pattern with fuel enrichments and burnable poison loadings and distributions chosen to provide a cycle length of 24 months. Some of the requested TS changes involve changes to both Unit 1 and Unit 2 TS. Each proposed change is discussed below.

6.1 Figure 2.2-2 Thermal Margin/Low Pressure Trip Setpoint-Part 1

Figure 2.2-2 is modified due to a revision in the curve fit for the TM/LP trip setpoint to accommodate the implementation of the extended statistical combination of uncertainties methodology. The setpoint analysis uses this methodology and the licensee has determined that acceptable results are obtained for Cycle 10. The changes to Figure 2.2-2 are, therefore, acceptable.

6.2 Figure 2.2-3 Thermal Margin/Low Pressure Trip Setpoint-Part 2

Figure 2.2-3 is modified for the same reason as Figure 2.2-2 and the change is acceptable for the same reason.

6.3 Bases 2.1.1 and 2.2.1

The text is modified to replace a specific MDNBR value with the phrase DNB SAFDL. The use of a phrase in place of a specific MDNBR value was recommended in the extended SCU methodology (Ref. 11) and approved by the staff (Ref. 18). The change is, therefore, acceptable.

6.4 Technical Specification 3.1.1.1 Shutdown Margin

Two modifications are proposed for this TS. First, the shutdown margin is changed from a constant value to text which refers to a new Figure 3.1-1b which presents shutdown margin as a function of time in cycle. Since the required shutdown margin varies throughout the cycle due to fuel depletion, boron concentration and moderator temperature and this variation with cycle time has been incorporated in all the appropriate safety analyses for Cycle 10, this change is acceptable.

The shutdown margin at EOC is increased from 3.5% delta k/k to 5.0% delta k/k. The analysis of the Cycle 10 steam line rupture analysis, which is limiting at hot zero power EOC conditions, supports this change and it is, therefore, acceptable.

6.5 Technical Specification 3.1.1.4 Moderator Temperature Coefficient

The MTC limit above 70% power is being raised from $+0.2\times10^{-4}$ delta rho/° F to a value which varies linearly from $+0.3\times10^{-4}$ delta rho/° F at 100% power to $+0.7\times10^{-4}$ delta rho/° F at 70% power. This change is being implemented to accommodate 24 month cycles and to facilitate initial reactor startup at the beginning of the cycle. The licensee has committed to a negative MTC at hot full power, equilibrium xenon conditions. As mentioned in Section 3.3, this value has been predicted to be -0.2×10^{-4} delta rho/° F. The feedline break analysis which supports this change is applicable to Cycle 10 and, therefore, the proposed change is acceptable.

6.6 Figure 3.1-2 CEA Group Insertion Limits

The transient insertion limit between 90% and 100% power is being increased from an allowed insertion limit which varies linearly from 35% for Bank 5 at 90% power to 25% at 100% power, to a constant value of 35%. This change, which is being made to enhance the ability to control axial oscillations near EOC, has been incorporated into all of the Cycle 10 physics, safety and setpoint analyses and is, therefore, acceptable.

6.7 Figure 2.2-1 Axial Power Distribution Trip LSSS

Figure 2.2-1 is modified to increase the positive and negative axial shape index (ASI) regions below 70% power. The setpoint analysis uses the modified results given by Figure 2.2-1 and the licensee has determined that acceptable results are obtained for Unit 1 Cycle 10 and Unit 2 Cycle 8. The changes to Figure 2.2-1 are, therefore, acceptable for both units.

6.8 Figure 3.2-2 Linear Heat Rate Axial Flux Offset Control Limits And Figure 3.2-4 DNB Axial Flux Offset Control Limits

These Figures are modified to increase the negative ASI limits below 50% power. The licensee has evaluated the effect of the proposed new limits on the Unit 1 Cycle 10 and Unit 2 Cycle 8 transient analyses, margin to fuel centerline melt limits, margin to DNB limits, margin to LOCA PLHGR limit, core power versus planar radial peaking factor LCO, TM/LP LSSS, and core power versus integrated radial peaking factor LCO and has determined that acceptable results are obtained. The changes are, therefore, acceptable for Unit 1 Cycle 10 and Unit 2 Cycle 8.

7.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design, and the transient and accident analysis information presented in the Calvert Cliffs Unit 1 Cycle 10 reload submittals. Based on this review, which is described above, the staff concludes that the proposed Cycle 10 reload and associated modified TS are acceptable. This conclusion is further based on the following: (1) previously reviewed and approved methods were used in the analyses; (2) the results of the safety analyses show that all safety criteria are met; and (3) the proposed TS are consistent with the reload safety analyses.

8.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities' components located within the restricted areas as defined in 10 CFR 20 and changes in surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

9.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 16, 1988

PRINCIPAL CONTRIBUTOR:

L. Kopp

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