SEP 9 1977

Docket No. 50-317

DISTRIBUTION
Docket
NRC PDR
Local PDR'
ORB #2 Reading
VStello
KRGoller
RMDiggs
DDavis
MConner
OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
HSaltzman
BGrimes
BHarless
DEisenhut
ACRS (16)
OPA (C. Miles)
DRoss
TBAbernathy
JRBuchanan
B Hardin
J Donohew
E Reeves
G Zech
Z Olshan

Baltimore Gas & Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President - Supply
P. O. Box 1475
Baltimore, Maryland 21203

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-53 for Unit No. 1 of the Calvert Cliffs Nuclear Power Plant. The amendment is in response to your application dated March 24, 1977, and supplements thereto dated June 10 and 30 and August 8, 1977, and earlier filings dated October 1, 1976, and November 5 and 30, 1976.

The amendment authorizes operation of the facility at power levels up to 2700 megawatts (thermal).

Copies of the related Safety Evaluation, Environmental Impact Appraisal, and the Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Marshall Grotenhuis

for ~~Don K.~~ Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to License DPR-53
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/enclosures: See next page

Notified: BG&E's R. Olson
I&E J. Sniezak
and D. Haverkamp
1515 on 9/9/77
Const. 1

	TS	DOR:AD/OR
	JMcGough	KRGoller
	8/31/77	9/9/77

*NOTE: SEE ATTACHED YELLOW FOR CONCURRENCE.

OFFICE	DOR:ORB #2	DOR:ORB #2	DOR:RS/OT	DOR:EEB/OT	OELD	DOR:ORB #2
SURNAME	RMDiggs:nm	MConner	RBaer	BGrimes	Tourtellotte	DKDavis
DATE	8/29/77	8/29/77	8/30/77	9/2/77	9/9/77	9/9/77

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Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

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	OELD						
	8/ /77						
OFFICE	ORB#2:DOR	ORB#2:DOR	OT/RS:DOR	OT/EEB:DOR	AD:DOR	ORB#2:DOR
SURNAME	RDiggs	MConner:nm	BBear	BGrimes	KGoller	DDavis
DATE	8/ 29/77	8/ 26/77	8/ /77	8/ /77	8/ /77	8/ /77

NRC FORM 318 (9-76) NRCM 0240

☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

September 9, 1977

Docket No. 50-317

Baltimore Gas & Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President - Supply
P. O. Box 1475
Baltimore, Maryland 21203

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Division of Operating Reactors

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2. Safety Evaluation
3. Environmental Impact Appraisal
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cc w/enclosures: See next page

cc
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Power Plant Siting Program
Department of Natural Resources
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U. S. Environmental Protection Agency
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Sixth & Walnut Strs.
Philadelphia, Pennsylvania 19106

Mr. Bernard Fowler
President, Board of County
Commissioners
Prince Frederick, Maryland 20768

cc w/enclosures and BG&E filings
referred to in first paragraph
of this letter:

Dr. Paul Massicot, Director
Department of Natural Resources
Power Plant Siting Program
Energy & Coastal Zone Admin.
Tawes State Office Bldg.
Annapolis, Maryland 21401

cc w/3 enclosures and 1cy of BG&E
filings dtd 6/10 & 30 & 8/8/77

Director, Department of State
Planning
301 West Preston Street
Baltimore, Maryland 21201

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Power Company (the licensee) dated March 24, 1977, as supplemented by filings dated June 10 and 30, and August 8, 1977, and earlier filings dated October 1, 1976 and November 5 and 30, 1976, 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;

   E. The licensee has satisfied the requirements of 10 CFR Part 170.21 on payment of license fee of power increase, and

   F. 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 by amending Section 2.C to revise paragraphs (1) and (2) of Facility Operating License No. DPR-53 to read as follows

   (1) Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24, 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

Joe W. Rue for

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Sepcifications

Date of Issuance: September 9, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 24

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

1-1
2-2
2-11
2-12
2-13
3/4 1-27
3/4 2-2
3/4 2-4
3/4 2-6
3/4 2-8
3/4 2-9
3/4 2-11

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other required auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

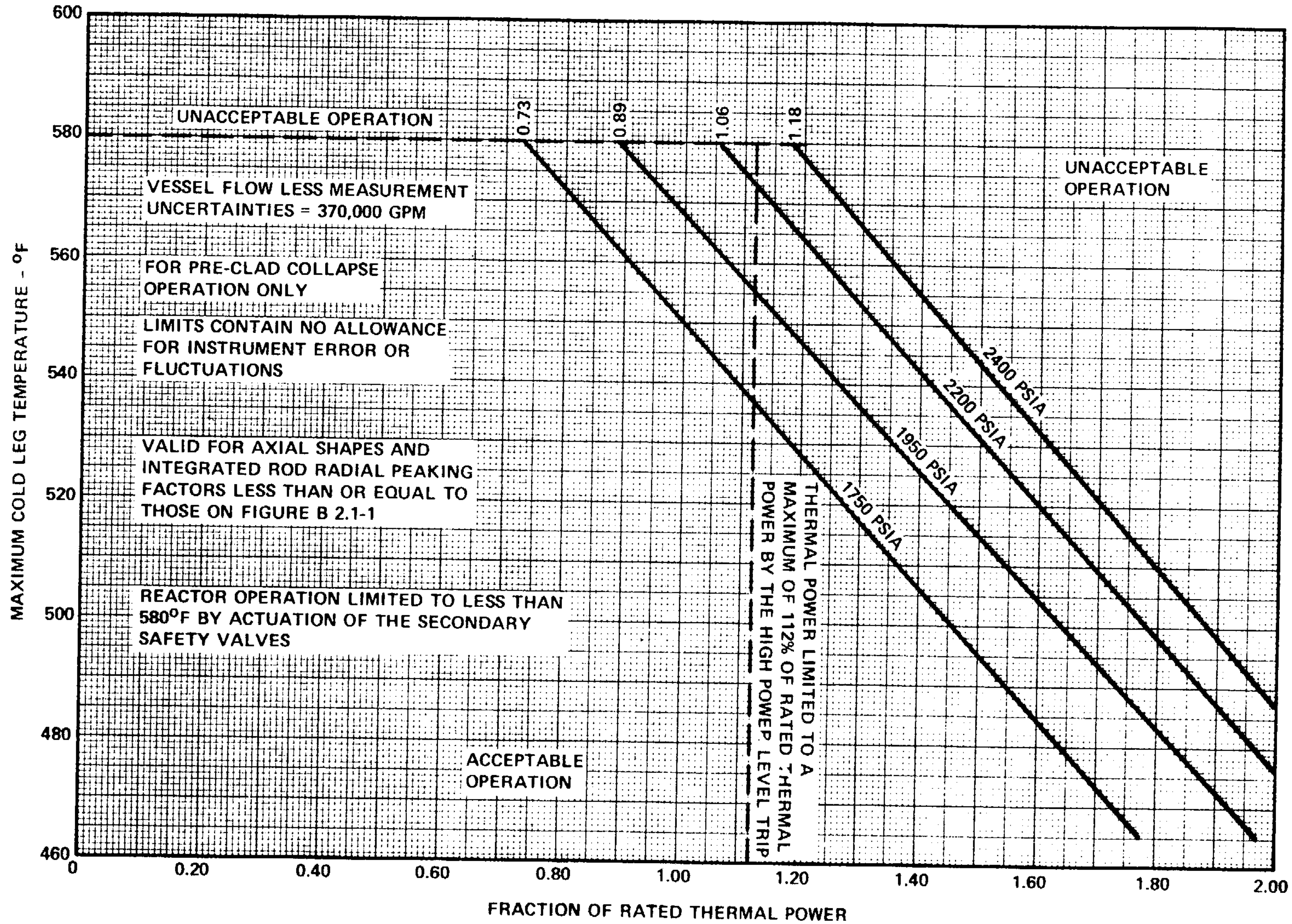

Figure 2.1 - 1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

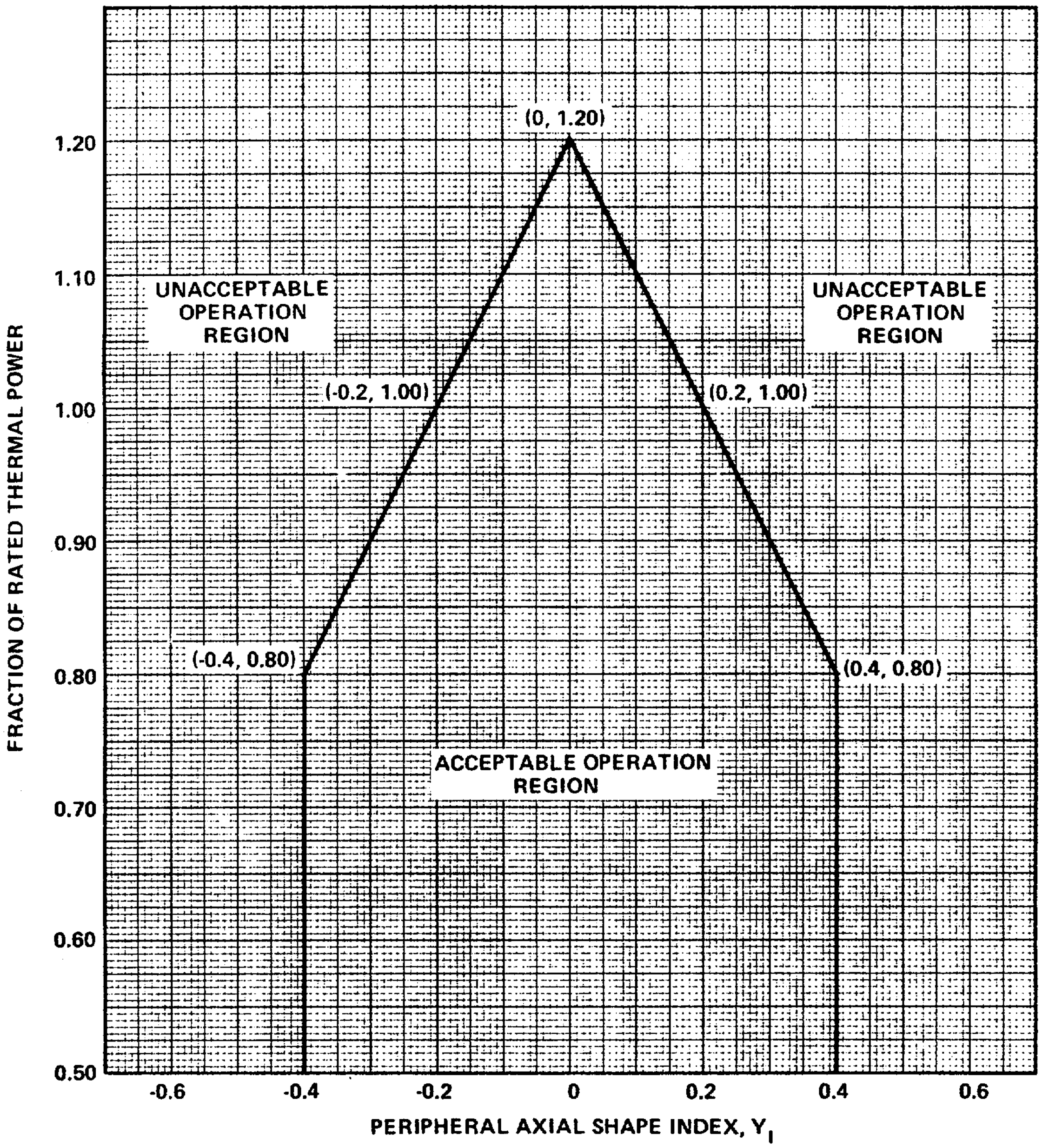

FIGURE 2.2-1
Peripheral Axial Shape Index, Y_I Versus Fraction of RATED THERMAL POWER

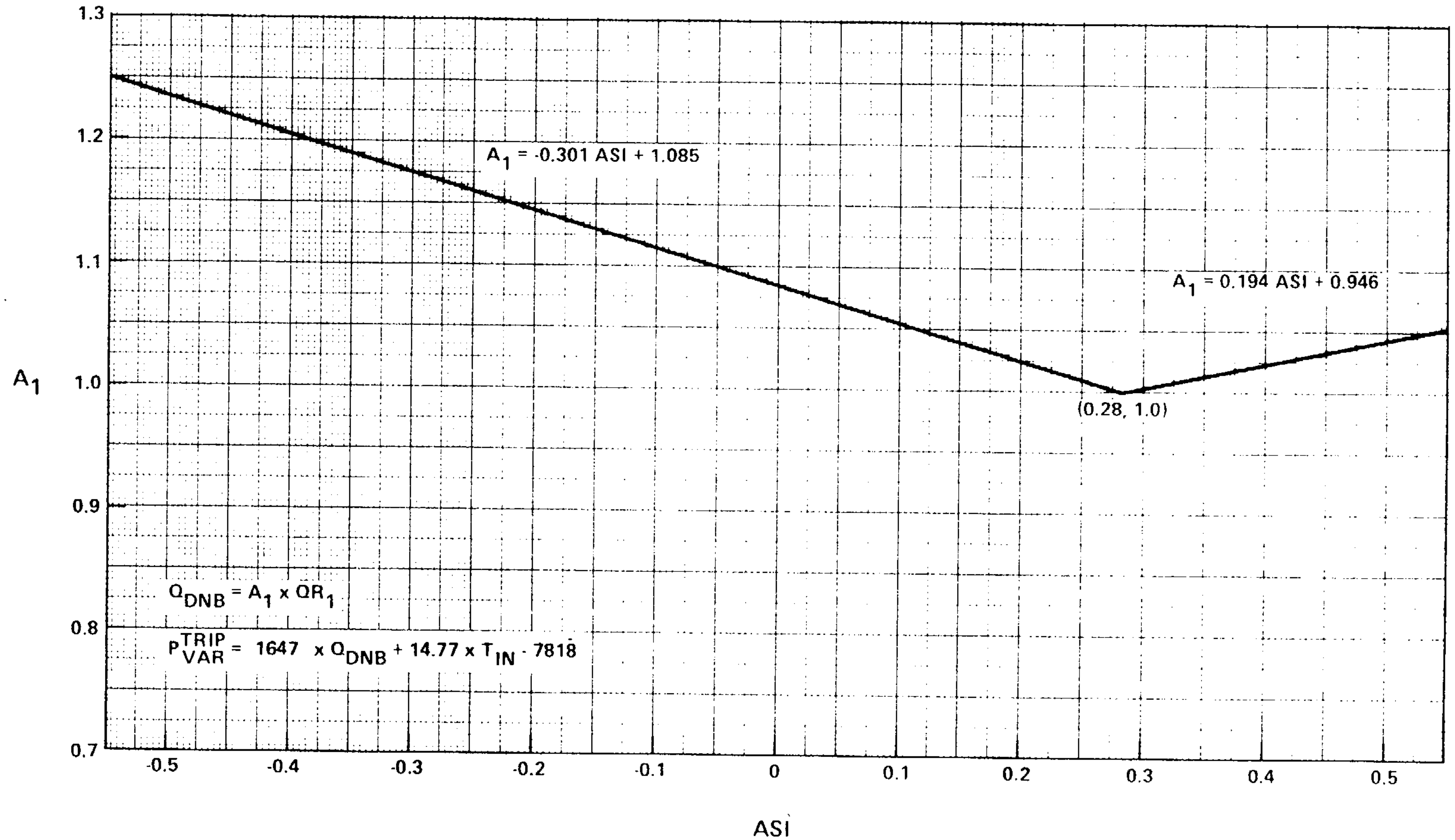

FIGURE 2.2-2

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

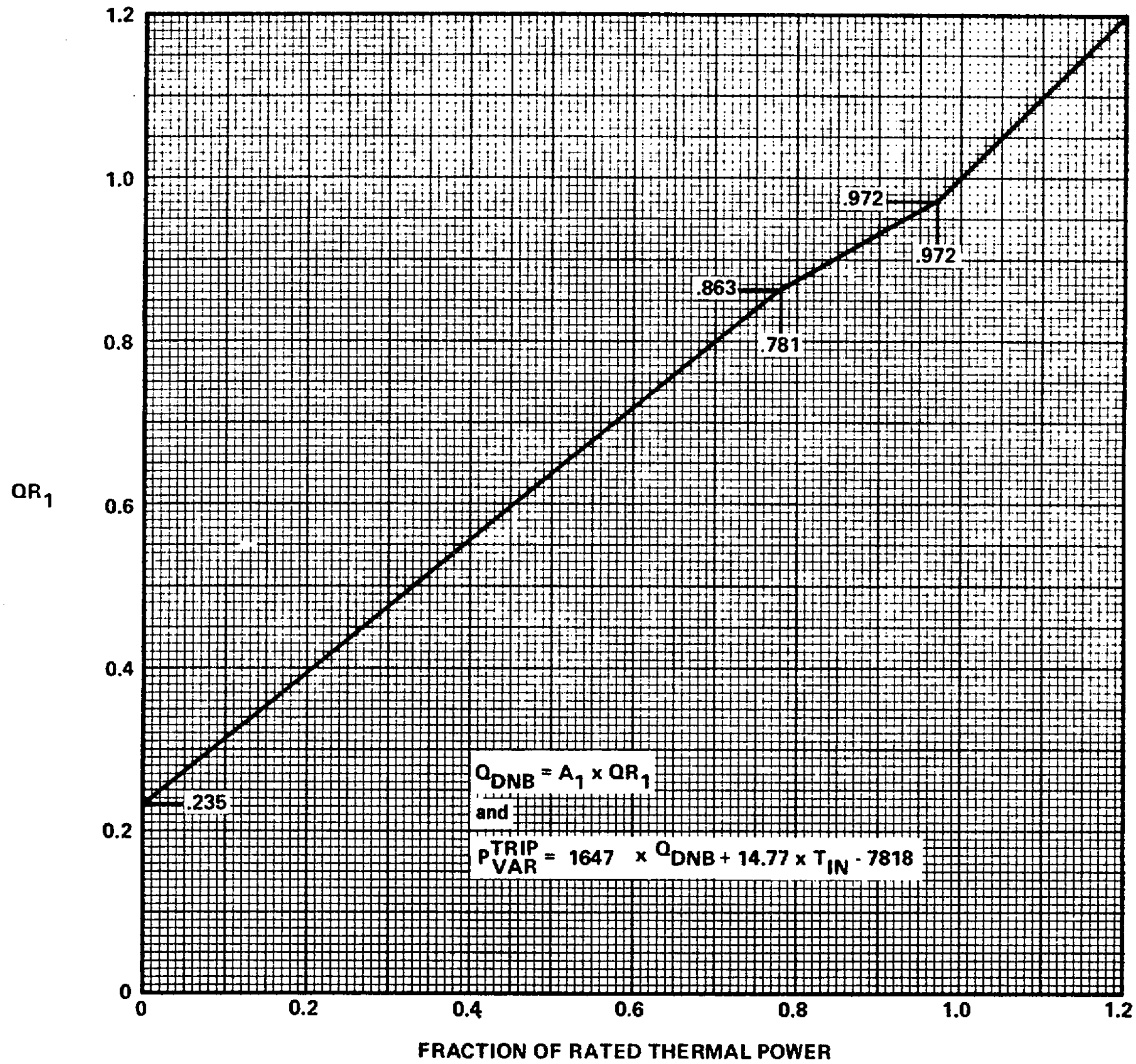

FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER versus QR_1)

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Figure 2.2-4

Thermal Margin/Low Pressure Trip Setpoint-Part 1
Three Reactor Coolant Pumps Operating

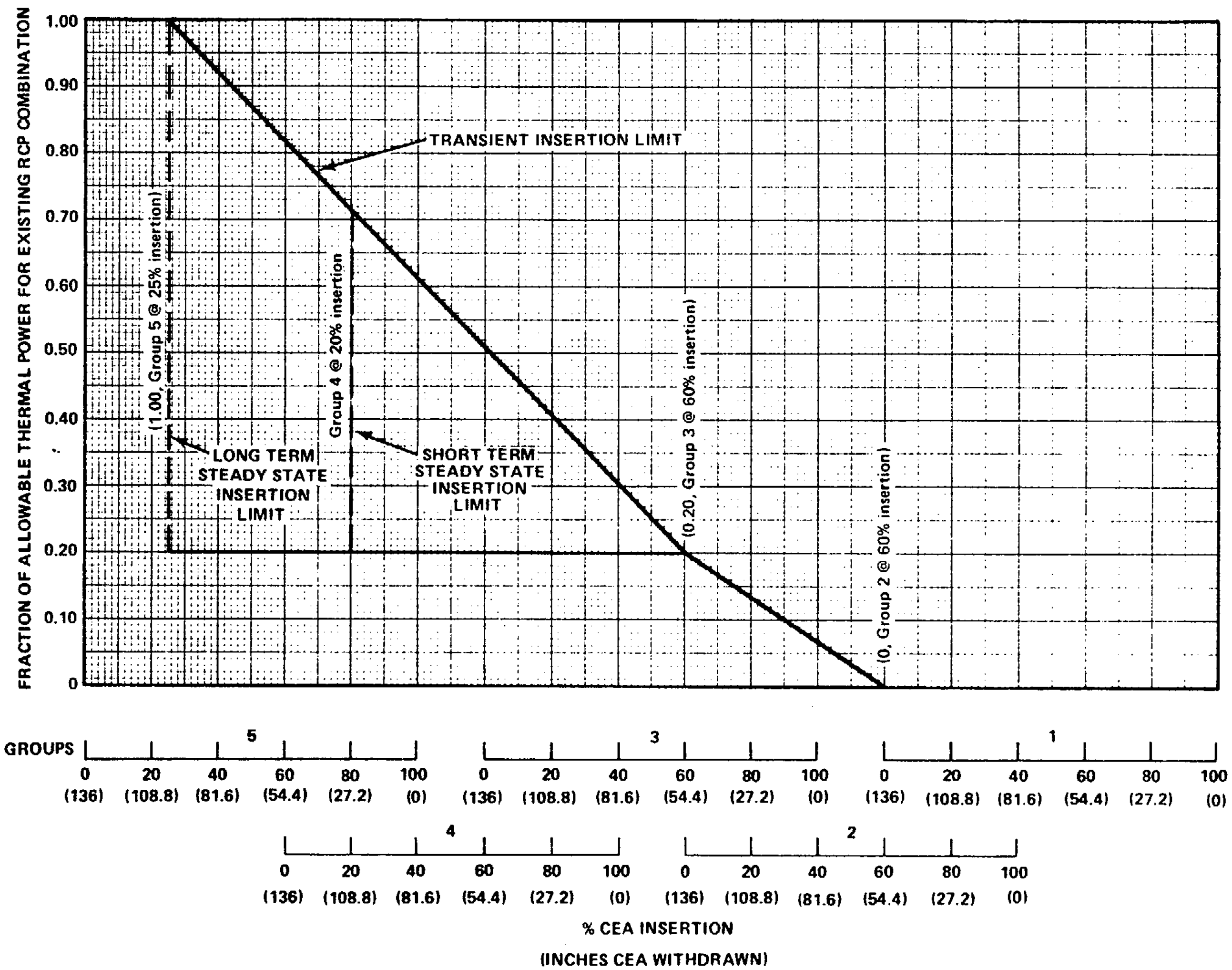

FIGURE 3.1-2

CEA Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the limits of Figure 3.2-2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the following expression:

SURVEILLANCE REQUIREMENTS (Continued)

$$\frac{L}{17.0} \times M \times N$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

3. N is the maximum allowable fraction of RATED THERMAL POWER as determined by Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.10,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

   5. A THERMAL POWER measurement uncertainty factor of 1.02.

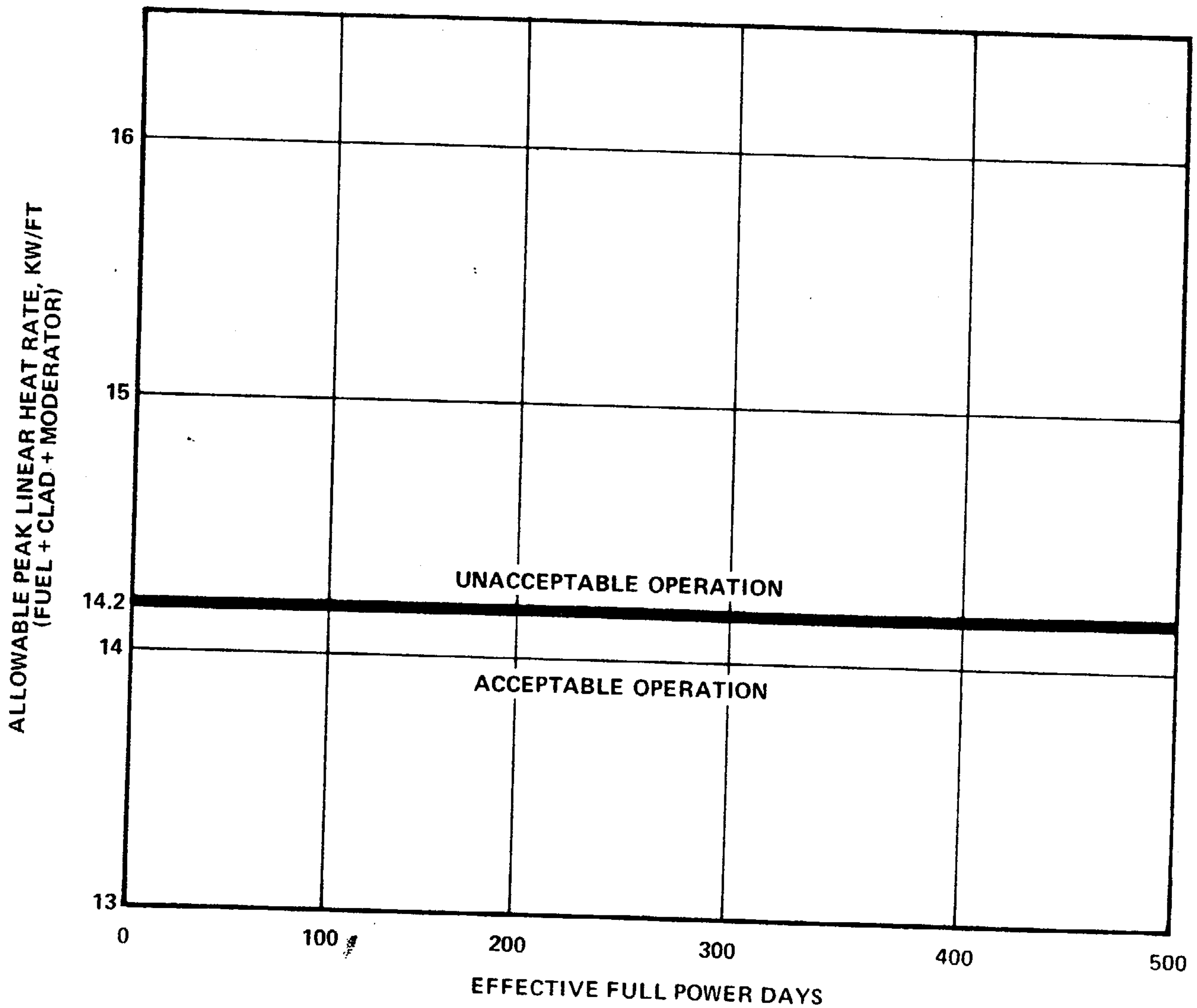

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

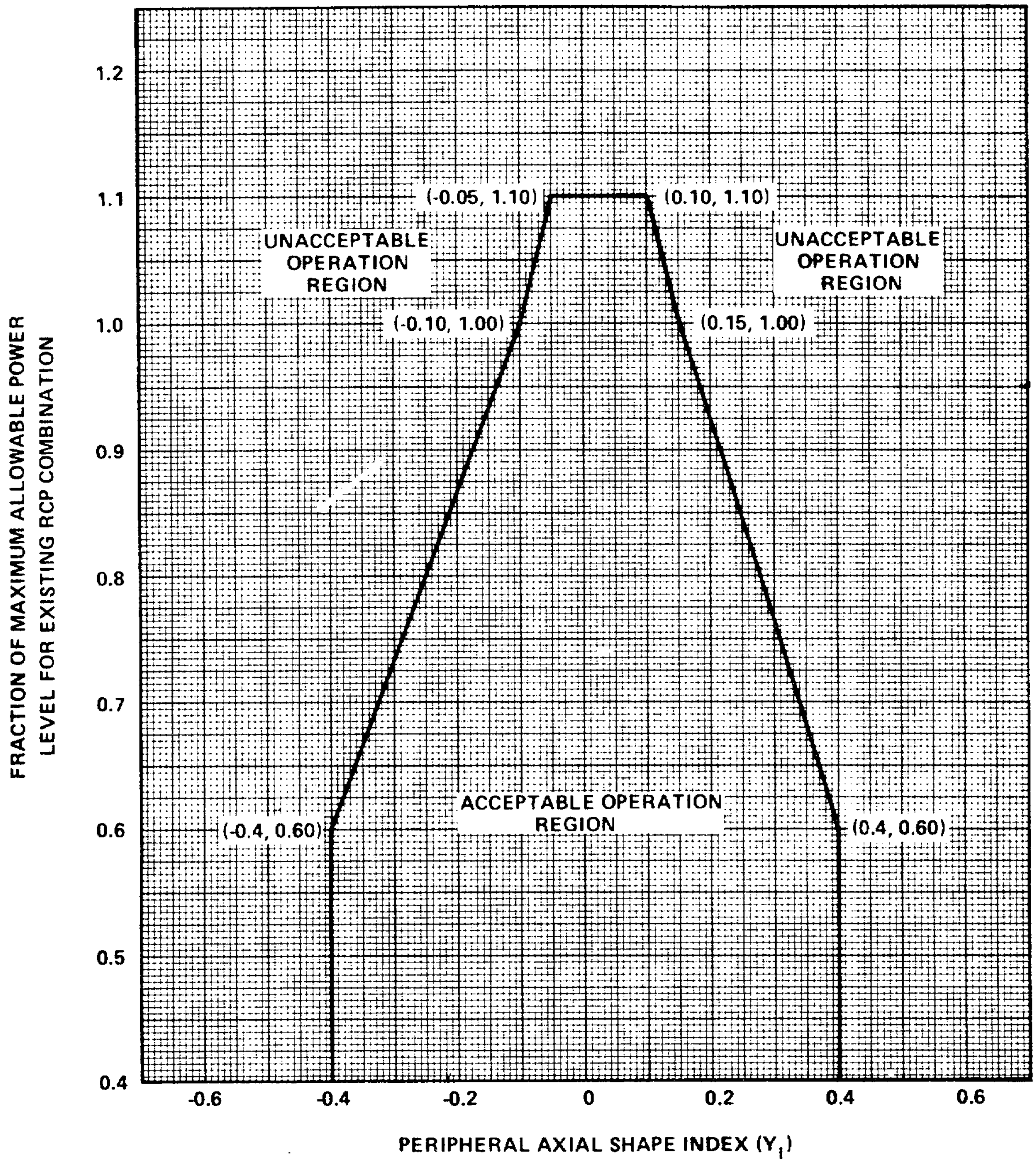

FIGURE 3.2-2
Linear Heat Rate
Axial Flux Offset Control Limits

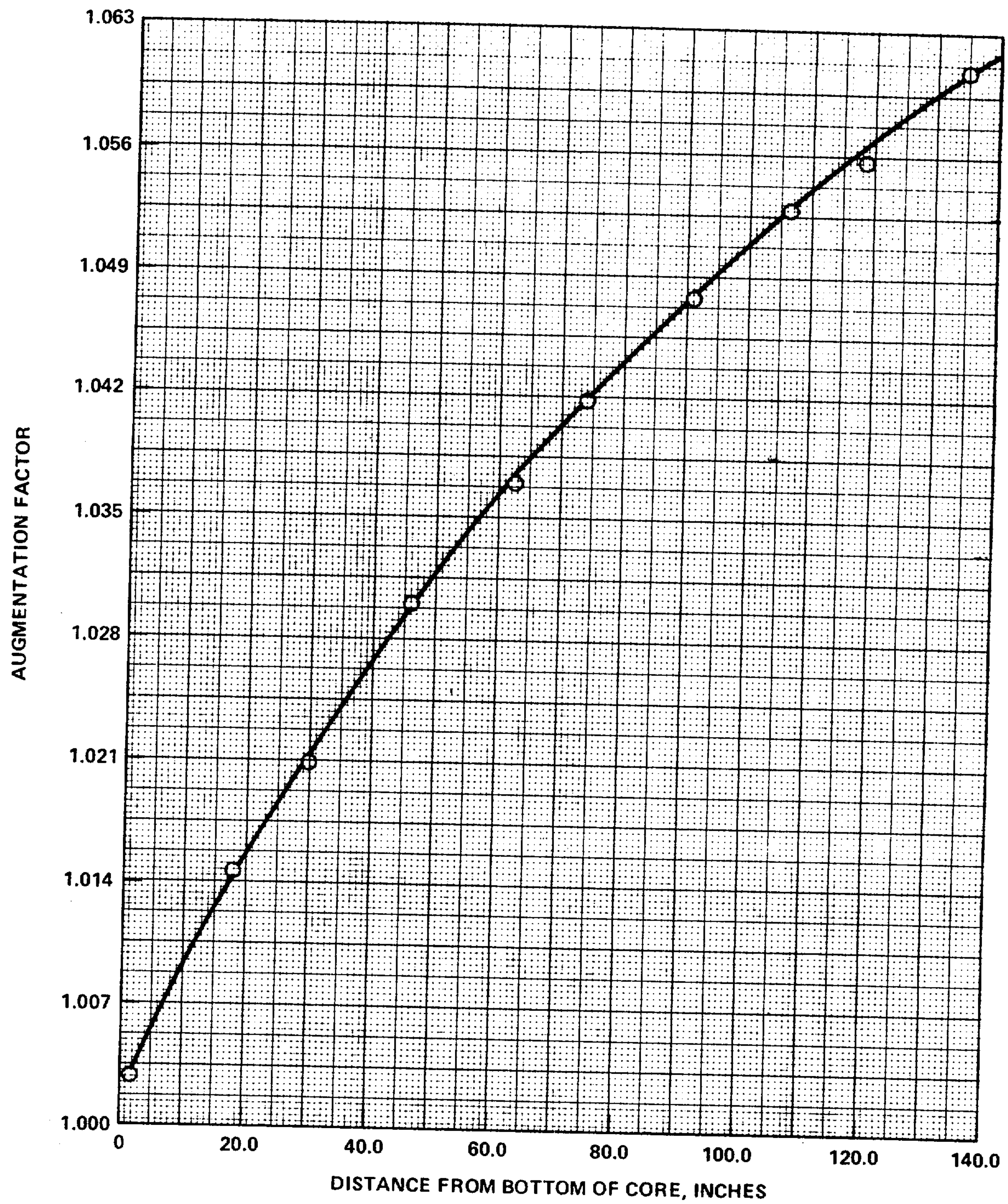

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.50.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.50, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3, fully withdraw the PLCEAs and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}\ (1+T_q)$ and F^T_{xy} 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 > 0.020.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F_{xy}^T is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and 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.

4.2.2.4 T_q shall be determined each time a calculation of F_{xy}^T is required and the value of T_q used to determine F_{xy}^T shall be the measured value of T_q.

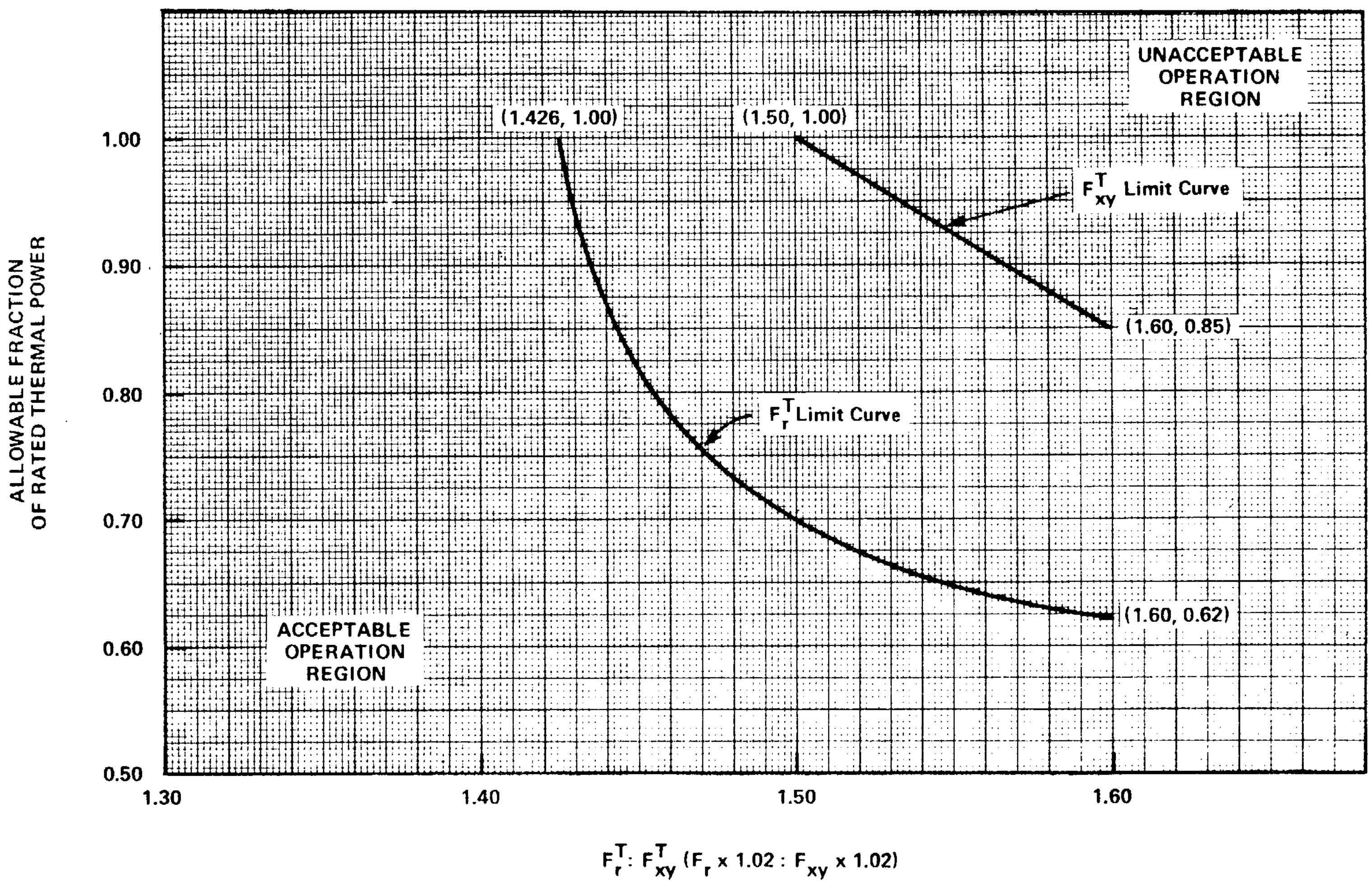

FIGURE 3.2-3

Total Radial Peaking Factor Versus Allowable Fraction of RATED THERMAL POWER

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to ≤ 1.42.

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.42$, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3, fully withdraw the PLCEAs and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

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 $F_r^T = F_r(1+T_q)$ and 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 > 0.020.

*See Special Test Exception 3.10.2.

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and 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 required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

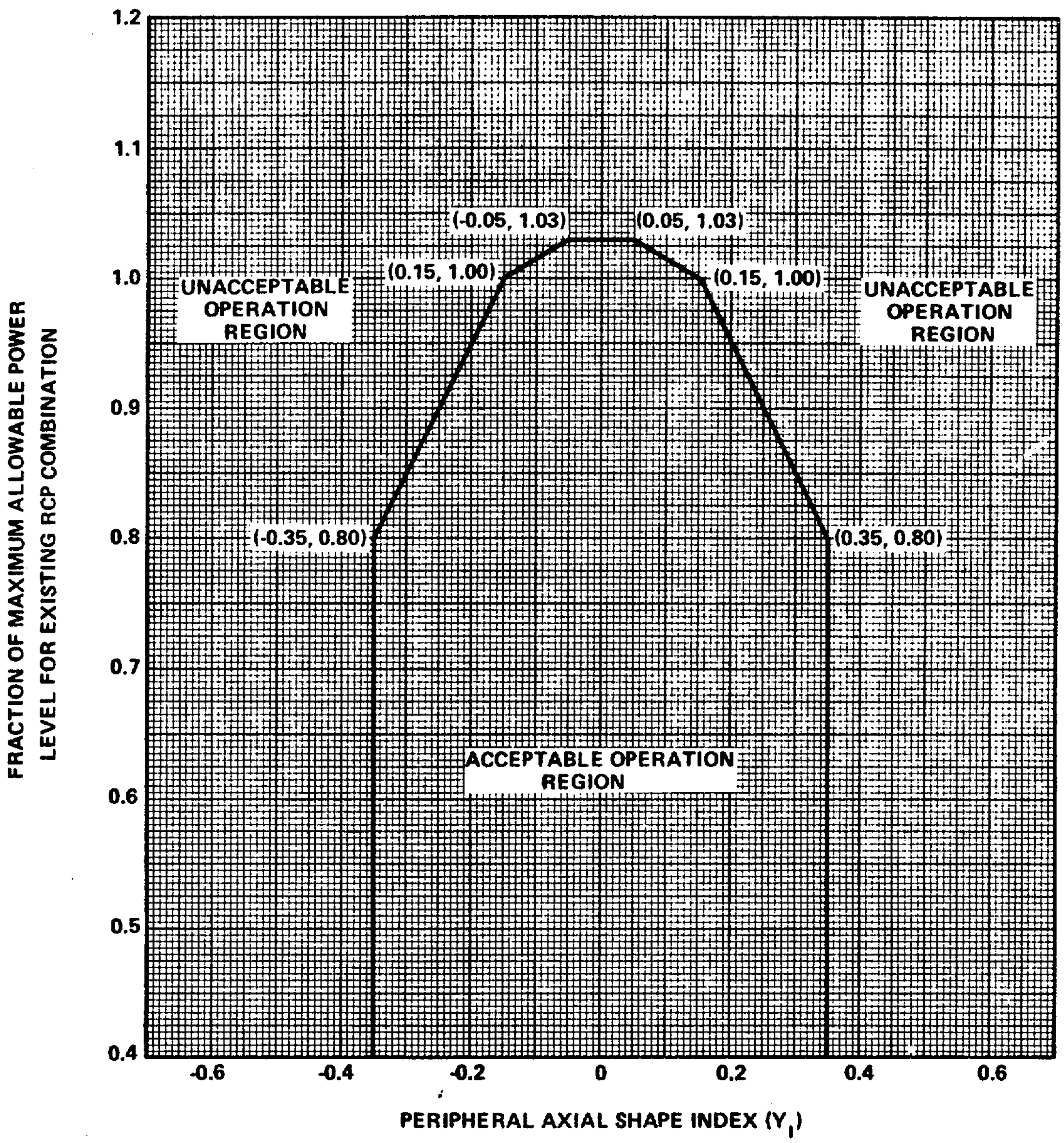

FIGURE 3.2-4
DNB Axial Flux Offset Control Limits

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.020.

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

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.020 but $\leq$ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F_{xy}^T) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) are within the limits of Specifications 3.2.2 and 3.2.3.

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

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

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

a. Calculating the tilt at least once per 12 hours, and

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

*See Special Test Exception 3.10.2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS UNIT NO. 1

DOCKET NO. 50-317

1.0 INTRODUCTION

In letters dated October 1 (Reference 1), November 5 (Reference 2), and November 30 (Reference 3), 1976, the licensee, Baltimore Gas and Electric Company, submitted information to support operation of Calvert Cliffs Unit 1 for Cycle 2. Included was information (safety analyses and technical specifications) regarding operation at a stretch power rating of 2700 Mwt representing about a 5.5% increase over the proposed initial Cycle 2 power level of 2560 Mwt. At the time of the Cycle 2 reload application, the licensee did not propose operation at the stretch power level. At the request of the staff, the licensee provided (Reference 4) technical specifications specifically limiting operation of Cycle 2 at 2560 Mwt. (The initial technical specifications as proposed in Reference 3 were provided for a Rated Thermal Power of 2700 Mwt, and it was intended that Thermal Power be administratively restricted to 94.8% of Rated Thermal Power or 2560 Mwt). Based upon its review of the above information, the staff issued Amendment No. 21 to License No. DPR-53 approving operation with Cycle 2 fuel at 2560 Mwt (Reference 5).

In a letter dated March 24, 1977 (Reference 6), the licensee requested a license amendment allowing operation with Cycle 2 fuel at the stretch power level of 2700 Mwt. Reference was made to applicable information regarding safety analyses, the LOCA analysis, and operating technical specifications which was previously submitted in References 1, 2 and 3. Additional modifications to the technical specifications and responses to questions from the staff with regard to the stretch power application were subsequently submitted in References 7, 8 and 9.

Our safety review of the licensee's stretch power application focused mainly on the impact of the proposed power increase on the safety analyses, physics startup tests, and the technical specifications. The discussions included in the staff's Safety Evaluation Report for Cycle 2 (Reference 5) in the areas of nuclear characteristics, fuel design, and effects of rod bowing and burnable poison rods, apply equally well for operation at stretch power and are not discussed further here. The steady-state thermal-hydraulic performance and safety analyses evaluations included in Reference 5 are also applicable, however, due to the relative importance of certain safety analyses in considering a power increase, these areas are discussed further below.

It should be noted that the analyses of incidents for which the primary consequence of interest is the possible dose at the site boundary or which provide the design bases for the engineered safety features systems were analyzed for a core power level of 2700 Mwt in the FSAR. Within this group, the steam line rupture, steam generator tube rupture, and LOCA incidents were reanalyzed specifically for the proposed Cycle 2 operating conditions (also at 2700 Mwt) and submitted in Reference 1. The FSAR analyses for the fuel handling and waste gas incidents remain applicable for stretch power operation during Cycle 2.

2.0 SAFETY ANALYSES (Other than LOCA)

2.1 Evaluation of Anticipated Operational Occurrences

A majority of the postulated Anticipated Operational Occurrences analyzed in the Calvert Cliffs 1 FSAR were reanalyzed using approved Combustion Engineering analysis methods for Cycle 2 at an assumed power level of 102% of 2700 Mwt (2754 Mwt), and the staff concluded that the new analyses were acceptable (Reference 5). Since the conditions assumed in the FSAR analyses of the Boron Dilution and Excess Load Incidents were shown to be more limiting than the 2700 Mwt Cycle 2 conditions, these cases were not reanalyzed. The Startup of an Inactive Coolant Pump was not reanalyzed for Cycle 2 since the current Technical Specifications do not permit operation with less than all reactor coolant pumps in operation.

2.2 Evaluation of Analyses of Postulated Accidents

Incidents classified as accidents are those events with a low probability of occurrence which are analyzed to evaluate the protection afforded by the plant design and characteristics. A discussion of potential radiological consequences due to these accidents is presented in Section 4.

The following accidents were reanalyzed using standard Combustion Engineering analysis methods for Calvert Cliffs Unit No. 1, Cycle 2 at an assumed core power of 2754 MWt.

Loss of Coolant
CEA Ejection
Steam Line Rupture
Steam Generator Tube Rupture
Seized Rotor

The Loss-of-Coolant Accident is discussed separately in Section 3.0 of this report. A brief discussion of the other accidents follows.

2.2.1 CEA Ejection

Rapid ejection of a control element assembly (CEA) produces a large insertion of reactivity which results in a large power increase which is terminated by the Doppler effect. For beginning-of-cycle conditions, 0.2% of the fuel rods could experience cladding perforations and 4% of the fuel rods could experience centerline melting following a rapid CEA ejection. The corresponding values reported in the FSAR for Cycle 1 were 5.4% and 3.6%. The analysis of the CEA Ejection Accident is acceptable to the staff.

2.2.2 Steam Line Rupture

The Steam Line Rupture Accident was reanalyzed for Cycle 2 because of three changes from the previous analysis:

1. Increased Doppler feedback
2. Increased fuel power
3. Increased core inlet temperature

Reanalysis showed that for the Cycle 2 core, there would be no return to criticality. A return to criticality was predicted for Cycle 1. For the case of a Steam Line Rupture the limiting case is the No Load One Loop Steam Line Rupture for which the minimum shutdown margin is $3.2\%\Delta\rho$. For this case no fuel damage is predicted. The analysis of the Steam Line Rupture Accident is acceptable.

2.2.3 Steam Generator Tube Rupture

The integrity of the barrier between the reactor coolant system and the main steam system is significant because a leaking steam generator tube allows the transfer of reactor coolant from the primary system to the main steam system. Radioactivity contained in the primary system coolant then has a path to the environment.

The results of the analysis for Cycle 2, done at an assumed core power of 2754 Mwt are less severe than those predicted in the FSAR analysis because the amount of reactor coolant transported to the steam system in 30 minutes is 49,800 pounds for the Cycle 2 analysis as compared with 56,800 pounds in the FSAR analysis. The reason for the difference in the amount of coolant transported to the main steam system is a change in the analytical technique used by the licensee. For the Cycle 2 analysis the Combustion Engineering CESEC computer code was used to model the plant in greater detail than that used in the FSAR analysis.

The analysis of the Steam Generator Tube Rupture Accident is acceptable.

2.2.4 Seized Rotor

The Seized Rotor Accident is a postulated instantaneous seizure of the pump shaft due to mechanical failure. The effect is a rapid reduction in the reactor coolant flow to the three pump flow value. The Low Flow Trip will trip the reactor. At a reactor power of 2754 Mwt the licensee predicts a minimum DNBR of 1.11 at 1.4 seconds after seizure of one reactor coolant pump. The maximum reactor coolant system pressure is 2280 psi. This analysis is acceptable. Comparable values for DNBR and system pressure are not available from the FSAR analysis of Cycle 1, however, the analyses for Cycle 1 indicate that 2.5% of the fuel rods would have a DNBR less than 1.3.

3.0 ECCS PERFORMANCE (LOCA ANALYSIS)

By letter dated November 5, 1976 (Reference 2), the licensee submitted an ECCS performance analysis for Cycle 2 operation. In anticipation of a future application for stretch power, the analysis submitted for Cycle 2 was performed at the anticipated stretch power level of 2754 Mwt (102% of 2700 Mwt). Correspondingly, in the licensee's request for approval to operate at stretch power (Reference 6), the Cycle 2 ECCS analysis was referenced as being applicable.

In our Safety Evaluation Report (Reference 5), we concluded that the ECCS performance analysis submitted by the licensee was acceptable for operation at 2560 Mwt.

We have re-reviewed the ECCS performance analysis discussed above with regard to the proposed increase in power from 2560 Mwt to the stretch power level of 2700 Mwt and conclude that the analysis is equally acceptable for operation at the higher power level.

Recently the staff has noted that in LOCA calculations for some PWRs, a decrease in primary coolant inlet temperature has resulted in a predicted increase in peak clad temperature. In discussions with the PWR vendors we have learned that they have all observed this trend while performing LOCA calculations with their individual approved evaluation models. In the past, it has been widely accepted that it was conservative to assume the highest possible initial coolant temperature for LOCA calculations (typically maximum full power operating temperature plus $4^{o}F$ for measurement uncertainty). The apparent cause of this behavior stems from the fact that a reduction in coolant inlet temperature results in a reduction in the coolant saturation pressure. This decreases the flow rate from the vessel side of the break after the short period of subcooled blowdown. This reduced flow, for the postulated cold leg break, decreases the magnitude of the downward flow rate through the core that exists for a large portion of the blowdown period. This decreases the heat transfer coefficient and consequently less stored energy is removed during blowdown.

Reducing the coolant inlet temperature also changes the flow rate from the top of the vessel to the hot leg and out of the break through the steam generator and reactor coolant pump. The changes in hot leg flow caused by a reduction in inlet temperature tend to decrease the core flow rate during the period of positive core flow. This also leads to the removal of less stored energy during blowdown. Thus, the fuel temperature is higher at the end of bypass. Most PWRs exhibit peak clad temperature during reflood, and entering the reflood period with a greater fraction of stored heat remaining after blowdown may cause an increase in the peak clad temperature. It has also been observed that the decreased negative core flow may extend the time to end of bypass. Then in the evaluation model more accumulator water is assumed to spill out of the break. If, as a result, there is insufficient accumulator water remaining to fill the downcomer, reflood will be delayed. This will also contribute to the increase in peak clad temperature.

However, a reduction in coolant inlet temperature may not always result in an increase in peak clad temperature. It has been observed that if the clad rupture location changes to a different elevation where the core power is less, peak clad temperature may decrease. Also, lowering inlet temperature causes a change in the steam generator secondary side steam conditions which tends to reduce peak clad temperature. This effect is discussed further below.

In addition to the predicted changes in blowdown core flow and heat transfer, reducing coolant inlet temperature also causes a slight reduction in containment back pressure during reflood. Reducing this pressure is known to result in lower reflood rates with correspondingly higher clad temperatures. However, the effect due to containment back pressure is minor compared to blowdown core flow and heat transfer effects.

At this time the staff believes that nominal values of inlet temperatures and steam generator secondary side steam conditions should be used in all LOCA calculations since the effects of variations in inlet temperatures and steam conditions on peak clad temperature are not consistent and are at best second order effects.

Although the sensitivity of calculated peak clad temperature to coolant inlet temperature has not been determined specifically for Calvert Cliffs Unit 1, there is some information available which indicates that ECCS performance for this plant would not be significantly affected at reduced coolant inlet temperatures. In Reference 9, the licensee addressed the question of the effect of coolant inlet temperature on Calvert Cliffs ECCS performance. The LOCA analysis which was submitted in support of the plant's stretch power application shows that clad rupture occurs very near to the time that the blowdown core flow reverses in direction from positive to negative. It is also shown that the peak clad temperature and oxidation occur at the rupture node for all cases analyzed. In Reference 9 the licensee has presented an argument stating that for a blowdown such as that exhibited by Calvert Cliffs 1, a reduction in coolant inlet temperature could in fact result in a peak clad temperature decrease due to a shift in the clad rupture location to a higher position in the core where the local power is less. Upon reviewing the licensee's submittal concerning this effect, the staff agrees that although the above argument is plausible, it should be confirmed by additional calculations. The licensee has agreed to perform the calculations necessary to confirm the overall insensitivity of Calvert Cliffs Unit 1 to reduced coolant inlet temperature. The licensee has proposed (Reference 9) to administratively restrict coolant inlet temperature to within the range of 537° to 550°F (includes ± 2°F for measurement uncertainty) whenever reactor power exceeds the current licensed level of 2560 Mwt. The ECCS calculations submitted in support of stretch power were performed assuming a coolant inlet temperature of 550°F. Applying the maximum reported sensitivity of 4°F increase in peak clad temperature for each 1°F decrease in coolant inlet temperature to the permitted 13°F variation in inlet temperature, results in a maximum estimated peak clad temperature increase of 52°F. The present margin shown in the Calvert Cliffs calculations to the 10 CFR 50 limit on peak clad temperature (2200°F) is 55°F which is sufficient to offset the effect of a reduction in coolant inlet temperature. In addition, a reduction in coolant inlet temperature results in a corresponding reduction in core average temperature and steam generator secondary side steam pressure. A reduction in steam generator secondary pressure results in lower peak clad temperatures which would reduce the increase in peak clad temperature from the 52°F value estimated above. As noted above, the licensee will perform a confirmatory calculation to determine the specific sensitivity of Calvert Cliffs Unit 1 to changes in inlet temperature.

In summary, we conclude from our review that the ECCS performance for Calvert Cliffs 1 conforms to the acceptance criteria stated in 10 CFR 50.46 for operation at the stretch power level of 2700 Mwt provided that the peak linear heat generation rate does not exceed 14.2 kw/ft. (The licensee has proposed to use the 14.2 kw/ft limit for all of the fuel although 72 of the total 217 fuel assemblies are loaded with pre-densified fuel and are capable of operation at 16.5 kw/ft).

4.0 RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS

We have reviewed the evaluation of the potential radiological consequences of the postulated loss of coolant accident, fuel handling accident, steam line failure accident, steam generator tube failure accident and radioactive gas storage tank accident in the Safety Evaluation Report (SER) - Reference 5.

The consequences of the steam line failure accident and steam generator tube failure accident are controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations and were performed at 2700 MWt. These consequences are not significantly affected by performing the calculation at 102% of 2700 MWt (2754MWt). The consequences of a radioactive gas storage tank accident are controlled by limiting the permissible inventory of radioactivity in a gas storage tank and are not a function of power. We have reviewed the limits in the Appendix A Technical Specification and find that they are acceptable to keep potential consequences of these three accidents appropriately within the 10 CFR Part 100 guidelines.

The analyses for the loss of coolant accident and the fuel handling accident in the SER (reference 5) were performed at 2700 MWt. Neither the rod ejection accident nor the fuel handling accident inside containment were reviewed and evaluated in the SER. The fuel handling accident in the SER is for a postulated accident in the spent fuel building. On March 21, 1977, the licensee submitted an analysis of the fuel handling accident inside containment for Calvert Cliffs Unit 1. We have reviewed this analysis. The assumptions for this accident are the same as those for the fuel handling accident in the SER except there is no iodine removal factor of 6.67 for a charcoal filter as there is no engineered safety feature ventilation filtration system to reduce the consequences of the fuel handling accident inside containment.

The consequences of the loss of coolant accident, fuel handling accident, steam line failure accident, steam generator tube failure accident, and radioactive gas storage tank accident at 2700 MWt are given in the SER. The consequences of the fuel handling accident inside containment and the rod ejection accident at 2700 MWt are given in Table 2. The assumptions used in the evaluation of the rod ejection accident are presented in Table 1. The potential consequences of these accidents at 2700 MWt, assuming all the parameters presented in the SER are not changed, are significantly less than the guidelines of 10 CFR Part 100. The potential consequences at 102% of 2700 MWt would be at most directly proportional to the power level and still significantly less than the guidelines of 10 CFR Part 100.

5.0 RADIOACTIVE WASTE MANAGEMENT SYSTEMS

We expect that increasing the thermal power level of Calvert Cliffs Unit 1 from 2560 Mwt to 2700 Mwt will increase the concentration of activity in the reactor primary coolant and in water entering the radwaste treatment systems. This increase should be less than the percentage increase in the thermal power level which is 5.5%. This small increase in the concentration of activity will not affect the performance of equipment in the radwaste treatment systems. There is also no change in the flows and volumes of liquids and gases in these systems. Therefore, we expect the increase in radwaste effluents due to the change in thermal power level to also be less than the percentage increase in the thermal power level. This small increase in radioactive effluents does not change our conclusion in the SER (Reference 5) that the radwaste treatment system at Calvert Cliffs Unit 1 will be capable of limiting radioactive releases to values which are a small fraction of 10 CFR Part 20 limits. (The **statement** would be true at 102% of the licensed power level.)

The proposed amendment does not include changes to Section 2.3 of the Appendix B Technical Specifications. This section restricts releases of radioactive materials in gaseous and liquid effluents from the plant. The proposed amendment will not allow the licensee to discharge concentrations greater than the maximum allowed (Specifications. 2.3.A.1, 2.3.B.1 and 2.3.B.2) nor to discharge more activity in a year than the maximum allowed (Specifications 2.3.A.2 and 2.3.B.3). Therefore, although the licensee under the proposed amendment may be expected to release more radioactivity, compliance with specification 2.3.A.1, 2.3.A.2, 2.3.B.1, 2.3.B.2, and 2.3.B.3 will maintain concentrations of radioactive materials in unrestricted areas to a small fraction of 10 CFR Part 20, Standards for Protection Against Radiation. Consequently, there will be no appreciable effect on the environment or health and safety of the public from this action.

By letters dated June 4 and October 15, 1976, BG&E provided additional information pursuant to Appendix I to 10 CFR Part 50. After we complete our evaluation of these submittals we intend to revise the Technical Specifications to reflect the requirements of Appendix I.

6.0 PHYSICS TESTS

The licensee has described his confirmatory test program incident to increasing rated thermal power to 2700 Mwt in Reference 9.

Reactor power will be increased slowly (approximately 1% per hour) from the present licensed level of 2560 Mwt to, or just below, 2700 Mwt. The following physics related tests will then be performed:

i) Isothermal Temperature Coefficient Measurement
ii) Power Coefficient Measurement
iii) Power Distribution Measurement

The test methods employed will be similar to those described in the Calvert Cliffs Unit No, 1 Startup Test Report (Reference 5).

Test results and comparison with prediction and acceptance limits will be reported to NRC within 45 days of completion of the above tests.

We conclude that the licensee's plan for confirmatory testing and documentation is acceptable.

7.0 TECHNICAL SPECIFICATIONS

The results of the steady-state and transient safety analyses performed for an assumed power level of 2754 Mwt as described above have been used to define Limiting Conditions for Operation (LCO) and Limiting Safety System Setpoints (LSSS). The LCO and LSSS assure that the initial steady-state overpower margin and the action of the Reactor Protective System will prevent a violation of the Specified Acceptable Fuel Design Limits during Anticipated Operational Occurrences. They also assure that radioactive material releases during postulated accidents will remain within the Commission's guidelines of 10 CFR 100.

References 3, 7, and 8 include proposed Technical Specification modifications applicable to operation at the stretch power level of 2700 Mwt. While most of the proposed modifications are specifically related to an increase in power, some others are not. The proposed modifications are to the Technical Specifications which were adopted from the Combustion Engineering Standard Technical Specifications at the start of Cycle 2.

Reference 8 was submitted to correct an inconsistency in Standard Technical Specifications 3.2.1 and 3.2.2 for the current Cycle 2 operation at 2560 Mwt. The correction involved changing the maximum allowed value for the total planar radial peaking factor (F_{xy}^T) from 1.58 to 1.50. The 1.58 limit was recently found to be non-conservative, if the ex-core system is used to monitor linear heat rate. This determination resulted from a Combustion Engineering review of the development of that limit. The proposed change will assure that Technical Specification Figure 3.2-2 is conservative for all planar radial peaks allowed by Figure 3.2-3 under the conditions of ex-core monitoring of linear heat rate. In addition, the limit for F_{xy}^T is conservative for linear heat rate monitoring by either ex-core or in-core detectors. The Technical Specification changes increase the existing margins to the Limiting Safety System Settings, provide the new Figure 3.2-3 that requires more conservative reactor operation, and result in a more restrictive surveillance requirement in paragraph 4.2.1.3 when monitoring linear heat rate with the ex-core detector system.

Reference 7 noted that the above correction in F_{xy}^T is also applicable to operation at stretch power and that the Technical Specifications proposed in Reference 3 must be modified accordingly.

Major technical specification modifications related specifically to the proposed increase in power include the following:

Sec. 1.3 (page 1-1) RATED THERMAL POWER is changed to 2700 Mwt.

Sec. 2.1 (Figure 2.1-1) REACTOR CORE THERMAL MARGIN SAFETY LIMIT modified for 2700 Mwt operation.

Sec. 2.2 (Table 2.2-1) AXIAL FLUX OFFSET TRIP SETPOINT modified for 2700 Mwt operation by revising Figure 2.2-1.

Sec. 3.1 (Figure 3.1-2) POWER DEPENDENT INSERTION LIMITS modified for 2700 Mwt operation.

Sec. 3.2 (Figure 3.2-2) LINEAR HEAT RATE AXIAL FLUX OFFSET CONTROL LIMITS modified for 2700 Mwt operation.

Sec. 3.2 (Figure 3.2-4) AXIAL FLUX OFFSET DNB OPERATING LIMITS modified for 2700 Mwt operation.

Sec. 3.2 (Figure 3.2-3) ALLOWABLE COMBINATIONS OF THERMAL POWER AND F_r^T modified for 2700 Mwt operation.

The above proposed modifications to the LCO and LSSS have been made using standard Combustion Engineering methods.

Proposed technical specification modifications not specifically related to the proposed power increase include improvements in figures and wording and the change in F_{xy}^T described above.

Based upon our review of the modifications to the Technical Specifications proposed in Reference 3 and amended and supplemented by References 7 and 8, we conclude that the proposed modifications are acceptable.

8.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 9, 1977

TABLE 1

ASSUMPTIONS USED BY NRC REGULATORY STAFF IN CALCULATIONS OF POTENTIAL OFFSITE DOSES FOR THE ROD EJECTION ACCIDENT

1. Regulatory Guide 1.77, Appendix B and Appendix to Standard Review Plan 15.4.8 assumptions, plus those given below.

2. Power Level 2754 Mwt

3. CEA ejection results in clad perforation in 0.2% of core fuel pins and center line melting in 4% of core fuel pins and resultant fission product release.

4. Exclusion area boundary (1150 meters) X/Q of 1.8E-4 sec/m^3 (0-2 hrs) and LPZ boundary (3218 meters) X/Q of 5.6E-5 sec/m^3 (0-8 hrs) and 1.2E-5 sec/m^3 (8-24 hrs) from Figure 2(A) of Regulatory Guide 1.77, divided by 2, per Calvert Cliffs SER (dated 8/28/72), Section 3.1.3.

5. Iodine Decontamination Factor between steam and water = 10.

6. Mass of primary coolant = 460,000 lbm.

7. Initial primary system pressure = 2250 psia.

8. Primary and secondary system pressure at 30 minutes after accident = 1000 psia.

9. Primary and secondary system temperature at 3 hours after accident = 300^{o}F.

10. Normal primary to secondary leak rate = 1 gpm total.

11. Steam generator secondary side volume = 7939 cubic feet.

12. Steam generators remain unisolated for 24 hours after the accident.

TABLE 2

POTENTIAL OFFSITE DOSES CALCULATED BY STAFF

FOR DESIGN BASIS ACCIDENTS

Accident	EXCLUSION BOUNDARY TWO HOUR (1150 Meters) Thyroid (Rem)	EXCLUSION BOUNDARY TWO HOUR (1150 Meters) Whole Body (Rem)	LOW POPULATION ZONE COURSE OF THE ACCIDENT (2 Miles) Thyroid (Rem)	LOW POPULATION ZONE COURSE OF THE ACCIDENT (2 Miles) Whole Body (Rem)
Fuel Handling Accident Inside Containment	33	<7	13	<7
Rod Ejection Accident	28	<1	17	<1

REFERENCES

1. Calvert Cliffs Unit 1 Second Cycle License Application, letter to B. C. Rusche from A. E. Lundvall, Jr., October 1, 1976.

2. Calvert Cliffs Unit 1 Supplement 1 to Second Cycle License Application (ECCS Analysis), letter to B. C. Rusche from A. E. Lundvall, Jr., November 5, 1976.

3. Calvert Cliffs Unit 1 Supplement 2 to Second Cycle License Application, letter to B. C. Rusche from A. E. Lundvall, Jr., November 30, 1976.

4. Calvert Cliffs Unit 1 Supplement 3 to Second Cycle License Application, letter to B. C. Rusche from A. E. Lundvall, Jr., March 2, 1977.

5. Safety Evaluation and Environmental Impact Appraisal Supporting Amendment No. 21 to Operating License No. DPR-53 Calvert Cliffs Unit 1, March 14, 1977.

6. Calvert Cliffs Unit 1 Stretch Power Request for Amendment to Operating License, letter to D. L. Ziemann from A. E. Lundvall, Jr., March 24, 1977.

7. Calvert Cliffs Unit 1 Supplement to Request for Amendment to Operating License Allowing Stretch Power Operation, letter to D. K. Davis from A. E. Lundvall, Jr., June 10, 1977.

8. Calvert Cliffs Unit 1 Proposed Modification to Standard Technical Specifications 3.2.1 and 3.2.2, letter to E. G. Case from A. E. Lundvall, Jr., June 10, 1977.

9. Calvert Cliffs Unit 1 Response to Questions Regarding Stretch Power Application, letter to D. K. Davis from J. W. Gore, Jr., June 30, 1977.

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS UNIT NO. 1

DOCKET NO. 50-317

1.0 Description of Proposed Action

By letter dated March 24, 1977 (Reference 1), Baltimore Gas & Electric Company (BG&E) requested an amendment to License No. DPR-53 to allow operation of Calvert Cliffs Unit No. 1 at a stretch power rating of 2700 MWt. Additional responses to questions from the staff with regard to the environmental impact of stretch power operation were submitted on August 8, 1977 (Reference 2).

The licensee is presently licensed to operate Calvert Cliffs Unit No. 1 located in the State of Maryland, County of Calvert, on the western shore of the Chesapeake Bay at a maximum power level of 2560 MWt. The proposed cange will increase the thermal power level by 5.5%. We have reviewed this matter and our conclusions are set forth below.

2.0 Environmental Impact of Proposed Action

The potential environmental impacts associated with the proposed action are those that are significantly greater for operation at 2700 MWt than those evaluated in the Final Environmental Statement dated April 1973 (Reference 4) for issuance of the initial operating licensee (DPR-53) for operation at 2560 MWt, and thus could be considered to significantly affect the quality of the human environment.

2.1 The Site

Because the power level increase will result in no adverse change in the site features of Calvert Cliffs Unit No. 1, there will be no impact on the location of the plant, the regional demography and land use, the historic significance or the environmental features of site and environs.

2.2 The Plant

2.2.1 External Appearance

No change in plant appearance will result from stretch power operation.

2.2.2 Transmission Lines

BG&E states in Reference 2 that the existing transmission lines have been designed to carry the additional power generated by the increase to 2700 MW and still retain a margin of about 10%. This is true even with both units operating at stretch power. Therefore, no adverse impact due to transmission lines is created.

2.2.3 Reactor and Steam-Electric System

In response to a staff question relating to the modified fuel management required for operation at 2700 MWt, BG&E indicated that a small increase in reload fuel batch average enrichment will be required to maintain planned refueling schedules and projected Unit capacity factors (Reference 2). The remainder of the FES in this area remains unaffected.

2.2.4 Effluent Systems

2.2.4.1 Heat

The current Appendix B Technical Specifications limit the condenser ΔT to 10°F (5.56°C). BG&E expects that the increase in power level from 2560 to 2700 MWT may theoretically result in an average ΔT increase of about 0.6°F. They also state their intention to not exceed the 10°F temperature rise and restrict operation accordingly. Since the FES is based on the 10°F ΔT and the condenser flow rate is not changed, the maximum heat rejected rate to the bay is as analyzed by the FES.

However, since the normal ΔT for 2560 MWT operation has been below 10°F, this change allowing operation at 2700 MWT will increase the heat output to the bay to the maximum allowed. The environmental impact of this discharge of heat has been previously analyzed and approved by Maryland State Department of Natural Resources, Water Resources Administration, in issuing the NPDES Permit (Reference 5) and by the NRC in issuing the Appendix B Technical Specification (Reference 6).

2.2.4.2 Radioactive Waste

We have reviewed the FES (Reference 4) related to the operation of Calvert Cliffs Nuclear Power Plant Units 1 and 2. The radiological consequences of the proposed amendment are not different from those reported in the FES dated April 1973 except for those related to effluents from the radioactive waste treatment system. The evaluation of the radioactive waste treatment systems of Units 1 and 2 was performed for a thermal power level for both plants of 2560 MWt not for 2700 MWt. Increasing the thermal power level by 5.5% can be expected to increase the estimated releases of radioactive materials and the estimated radiological impact of Calvert Cliffs Unit 1 in the FES.

BG&E has pointed out (Reference 2) that the FES was prepared assuming operation at rated power (2560 MWt) and .25% failed fuel. Environmental Statements today are prepared in accordance with Regulatory Guide 1.112 and ANSI N-237. These references require that one assume 0.125% failed fuel. Therefore, if the FES were redone today using present guidelines, the estimated releases of radioactive material to the environment would be nearly 50% less than those assumed in the original FES. The FES was prepared assuming that Lithium was not added to the RCS. The addition of Lithium increases the production of Tritium in the RCS during power operation. Combustion Engineering's present best estimate is a production rate of 826 curies per year and a maximum rate of 1508 curies per year. The FES estimates 1000 curies per year. They conclude that the FES is conservative in addressing radioactive discharges from Calvert Cliffs while operating at stretch power. Actual isotopic concentrations should be less than those reported in these two documents. We expect that the increase in radioactive waste will be no more than the percentage increase in the thermal power level, 5.5%, of the estimates given in the FES.

2.2.4.3 Chemical Waste and Make-up Waste

BG&E has stated that these quantities are not expected to increase due to an increase in power level of 5.5% since they are largely a function of leakage rates and blowdown. The recently installed blowdown recovery system deletes blowdown as a significant source of water loss, and system leakage is not expected to increase measurably due to the higher rated power. We agree with the licensee analysis.

2.2.4.4 Sanitary and Other Wastes

No change in sanitary waste or storm drain discharge will result from stretch power operation.

2.2.5 Transportation of Fuel and Radwaste

BG&E states in Reference 2 that the increase in design power level to 2700 MWT will result in an additional approximate 1500 megawatt days per ton batch average discharge burnup in the present cycle. In subsequent cycles, BG&E is planning a small increase in the reload fuel batch average enrichment to maintain the planned refueling schedules. The increase to 2700 MWT will not change the average number of fuel assemblies discharged at each refueling nor will it cause more frequent refuelings. Therefore, transportation requirements for new and spent fuel will not change.

The handling of solid radioactive wastes will not change from the description presented in the FES.

2.3 Environmental Impact of Plant Operation

2.3.1 Land Use

The subjects covered in the FES relating to land use are not adversely affected by the stretch power application. The increased power level will have the beneficial effect of increasing the use of land, in that more electrical energy could be produced from the same land area.

2.3.2 Water Use

Water use will not change from the assumptions used in the FES. Since the temperature rise across the condensers will not exceed the 10°F limit except as allowed by the Technical Specifications and NPDES Permit, and since a rise of 10°F was assumed in the FES, the analysis does not change. Circulating water flow rate is the same.

2.3.3 Biological Impact

The only potential biological impact from stretch power operation would be due to increased temperature. This item has been considered previously in section 2.2.4.1 of this impact appraisal.

2.3.4 Radiological Impact on Man

The radiological consequences of the proposed amendment are not different from those reported in the FES (Reference 4) except for those related to effluents from the radioactive waste treatment system. Therefore, the implementation of the proposed amendment will not significantly increase normal radiological effluents from the plant. Implementation will also not allow the licensee to discharge concentrations greater than the maximum allowed nor to discharge more activity in a year than the maximum allowed. Compliance with the present Technical Specifications will adequately control releases such that there will be no appreciable effect on the environment due to operation under these proposed changes.

2.4 Environmental Impact of Postulated Accidents

Estimates of potential radiological consequences due to postulated accidents are presented in the FES (Reference 4). Those postulated accidents which are power level dependent were evaluated in the FES at 2700 MWT. Permitting Calvert Cliffs Unit No. 1 to operate at 2700 MWT will not change those estimated consequences nor the FES conclusions as to environmental impact due to these postulated accidents.

2.5 Summary of Remaining Topics

The remaining FES topics are unaffected by the 5.5% increase in reactor power level. BG&E states (Reference 2) that the need for power can best be illustrated by stating the potential savings to customers in replacement power cost. It is estimated that an average increase of 45 MWe output per unit will result in an annual savings of $7.5 million per unit. The cost of the additional generating capacity typically would be about 10% of the savings.

3.0 Conclusion

We have concluded, based on the considerations discussed above, that the environmental impact of operation at 2700 MWt will not be substantially greater than that evaluated in the Final Environmental Statement dated April 1973 for the facility and will not significantly affect the quality of the human environment. Therefore, an environmental impact statement need not be prepared for the power increase and that a negative declaration to this effect is appropriate.

Date: September 9, 1977

REFERENCES

1. Calvert Cliffs Unit 1 Stretch Power Request for Amendment to Operating License, letter to D. L. Ziemann from A. E. Lundvall, Jr., March 24, 1977.

2. Calvert Cliffs Unit 1 Stretch Power Answer to NRC Staff Questions, letter to D. K. Davis from A. E. Lundvall, Jr., August 8, 1977.

3. Calvert Cliffs Unit 1 and 2 Appendix I Submittals, letters to B. C. Rusche from J. W. Gore, Jr., June 4 and October 15, 1976.

4. Final Environmental Statement Related to Operation of Calvert Cliffs Nuclear Power Plant Units 1 and 2, April 1973.

5. State of Maryland NPDES Discharge Permit to Baltimore Gas and Electric Company from A. Schiffman, June 1, 1976

6. Environmental Impact Appraisal Supporting Amendment Nos. 23 and 7 to Operating License Nos. DPR-53 and DPR-69 Calvert Cliffs Unit Nos. 1 and 2, July 29, 1977.

7. Safety Evaluation of the BG&E's Calvert Cliffs Nuclear Power Plant Units 1 and 2, August 28, 1977

8. Calvert Cliffs Units Nos. 1 and 2 Fuel Handling Incident Inside Containment, letter to D. L. Ziemann from A. E. Lundvall, Jr. March 21, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-317

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND

NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. DPR-53, issued to Baltimore Gas and Electric Company (the licensee), which revised the license and its appended Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit No. 1 (the facility) located in Calvert County, Maryland. The amendment is effective as of the date of its issuance.

The amendment authorized the licensee to operate the facility at a power level of 2700 MWt which is an increase from the previously authorized level of 2560 MWt.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of the Amendment to Facility Operating License in connection with this action was published in the Federal Register on June 23, 1977 (42 F.R.31844). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the authorized power increase and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action significantly greater than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility dated April 1973, and the action will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the appliation for amendment dated March 24, 1977, and supplements dated June 10 and 3, and August 8, 1977, and earlier filings by the licensee dated October 1, 1976 and November 5 and 30, 1976, (2) Amendment No. 24 to License No. DPR-53, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Calvert County Library, Prince Frederick, Maryland 20678. A single copy of itmes (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 9th day of September, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors