June 30, 1978

Docket No.: 50-317

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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. 33 to Facility Operating License No. DPR-53 for the Calvert Cliffs Nuclear Power Plant Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated May 8, 1978, as supplemented by May 30 and June 21, 1978.

The amendment changes the Technical Specifications based on the reanalysis of the Cycle 3 core thermal hydraulic characteristics using new Combustion Engineering computer codes.

In the enclosed Safety Evaluation, we have concluded that your submittals to date have not justified uncertainties in F_r^T and F_q^T of less than 8 and 10%, respectively. You have justified certain conservatisms in analytical methods which have allowed the use of uncertainties in F_r^T of 5.2% and F_q^T of 5.8% for Cycle 3 of Calvert Cliffs Unit 1.

In the future, your analyses should either: (1) be based on the uncertainty factors of 8 and 10%, (2) provide a fuel cycle specific justification for the use of conservatisms to offset uncertainty factors of less than 8 and 10%, or (3) be based on the use of lower uncertainty factors as justified by submittals of additional responses to our questions transmitted to you on April 4, 1978. Please inform us within 30 days of the date of this letter as to which of the above alternatives you plan to use in the future.

GD

Construct 1

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 33 to DPR-53
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

STS:DOR
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DATE	6/29/78	6/29/78	6/ /78	6/ /78	6/30/78	

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Department of Natural Resources
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Annapolis, Maryland 21401

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

BALTIMORE GAS & ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated May 8, 1978, as supplemented May 30 and June 21, 1978, 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. 33, 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 W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 33

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

B 2-1
B 2-3
B 2-5
B 2-6
3/4 1-27
3/4 2-1
3/4 2-2
3/4 2-4
3/4 2-6
3/4 2-8
3/4 2-9
3/4 2-11
3/4 2-15
3/4 3-6
B 3/4 2-1
B 3/4 2-2

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 would 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 21 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 1.19. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the minimum DNBR is no less than 1.19 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 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in

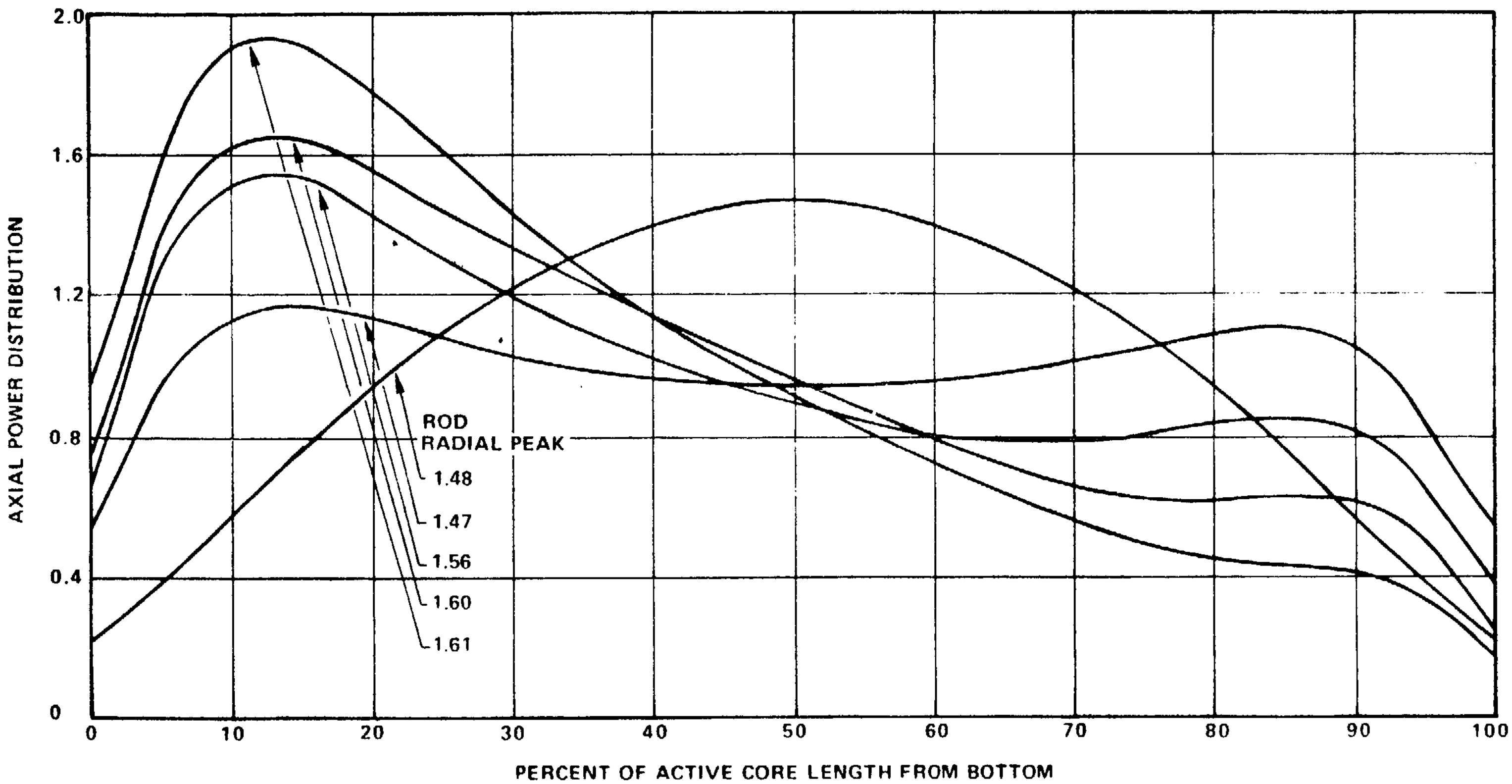

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

Table 2.1-1. The area of safe operation is below and to the left of these lines. For both 2-pump configurations, the limiting condition is void fraction rather than DNBR. The void fraction limits assure stable flow and maintenance of DNBR greater than 1.19.

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 1.19 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 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

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

Manual Reactor Trip

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

Power Level-High

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

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 106.5% 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 112% 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 1.19 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 1.19 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 concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

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 500 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 ± 22 psi in the accident analyses.

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 design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 13 minutes before auxiliary feedwater is required.

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 1.19 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 1.19, or when a void fraction limit is exceeded which could result in local flow instability.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1750 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

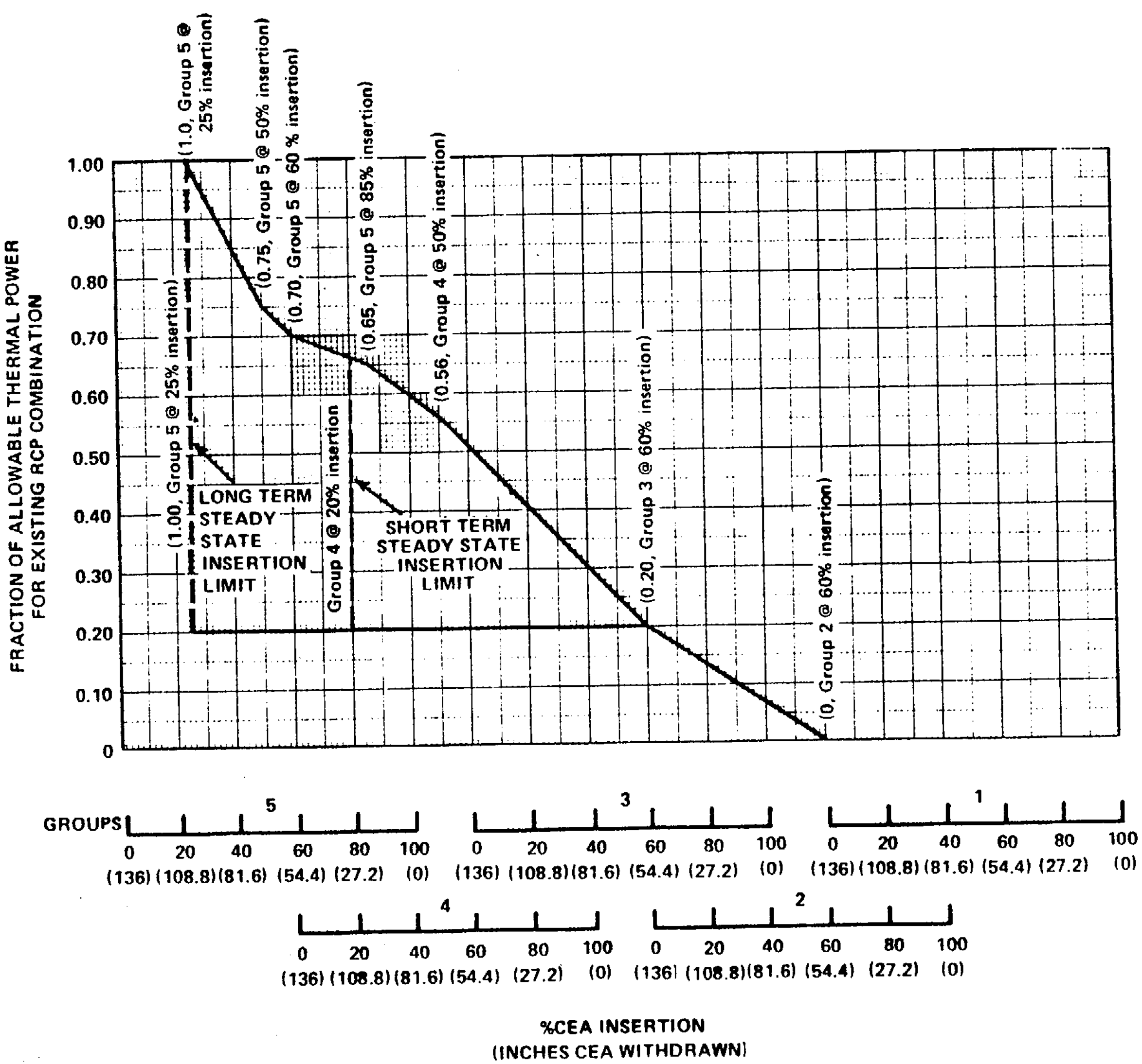

FIGURE 3.1-2 CEA Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination

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 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. 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:

$$M \times N$$

where:

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

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_{xy}^{T} curve of 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 1.058,

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.

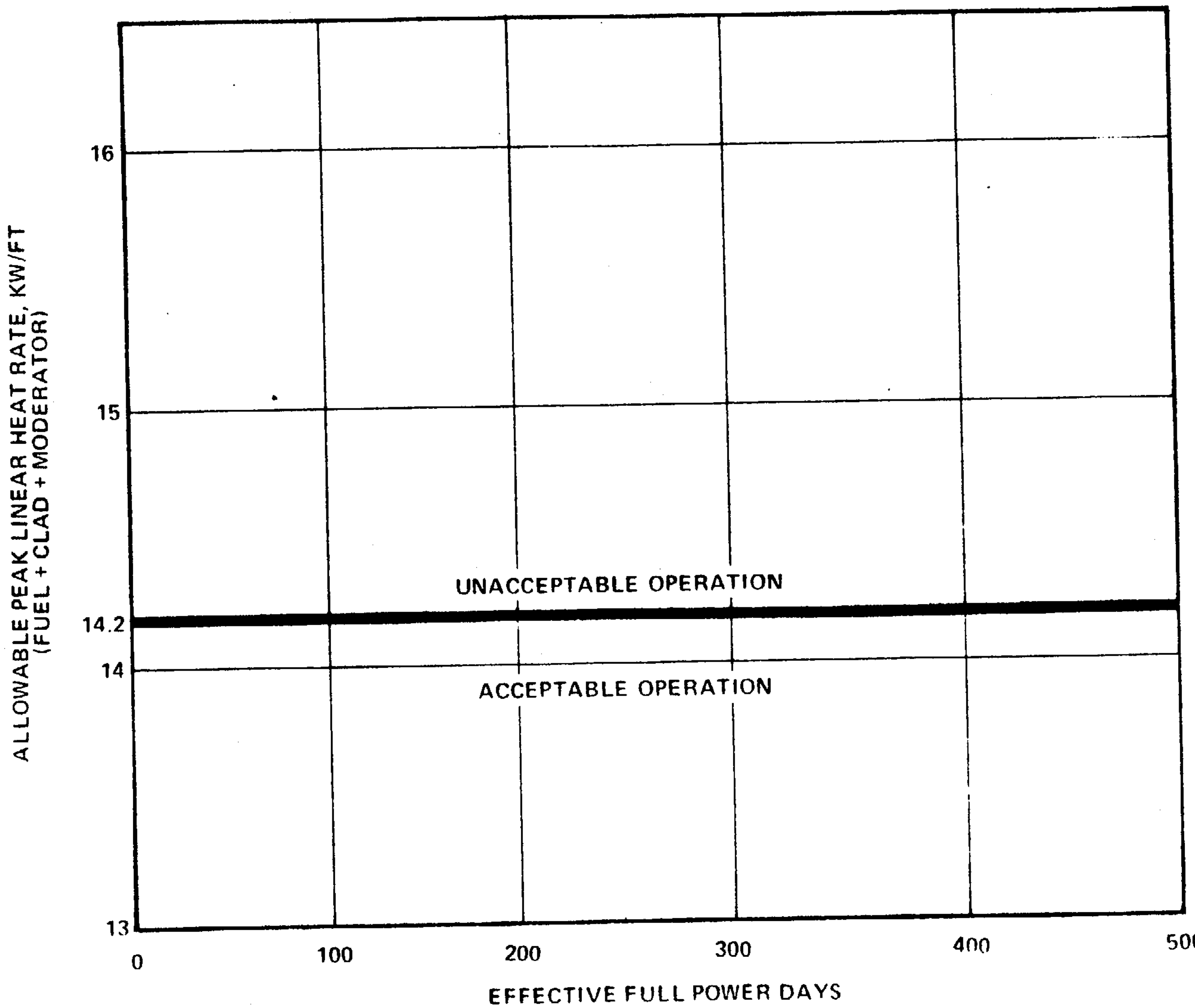

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

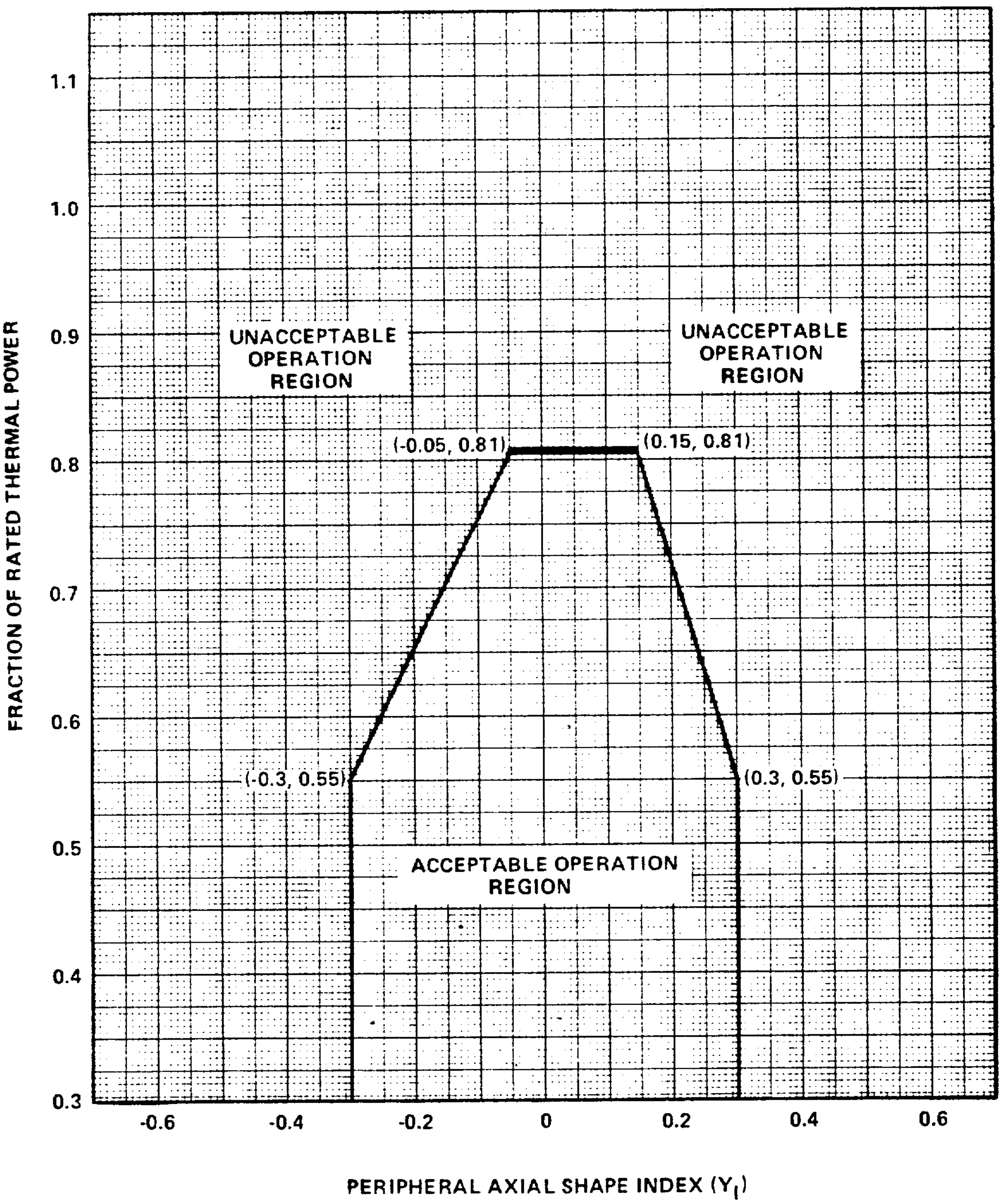

FIGURE 3.2-2
Linear Heat Rate
Axial Flux Offset Control Limits

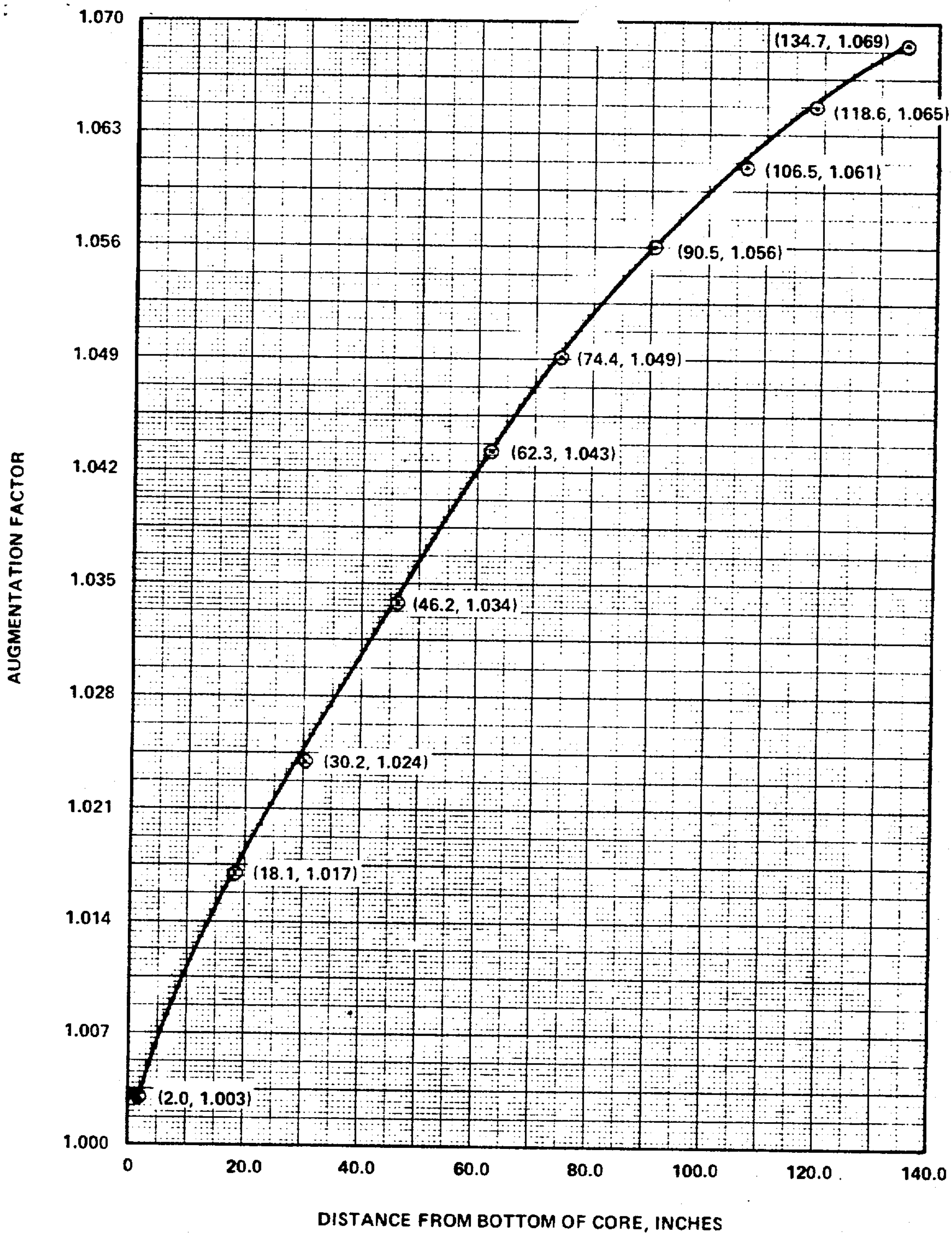

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

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 $\leq$ 1.660.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.660, 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 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 > 0.030.

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

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

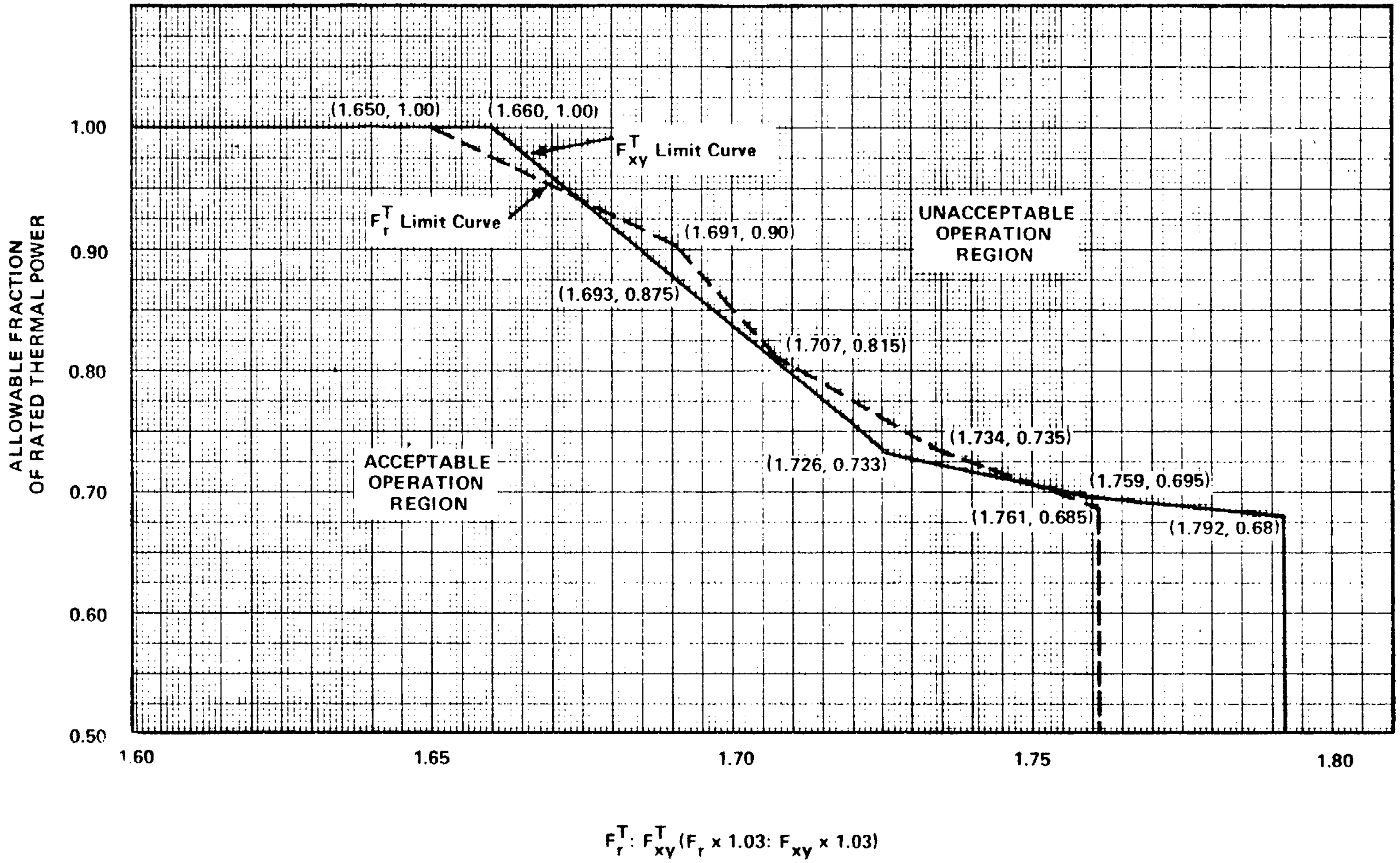

FIGURE 3.2-3

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 $\leq$ 1.650.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.650, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ 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 > 0.030.

*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 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.

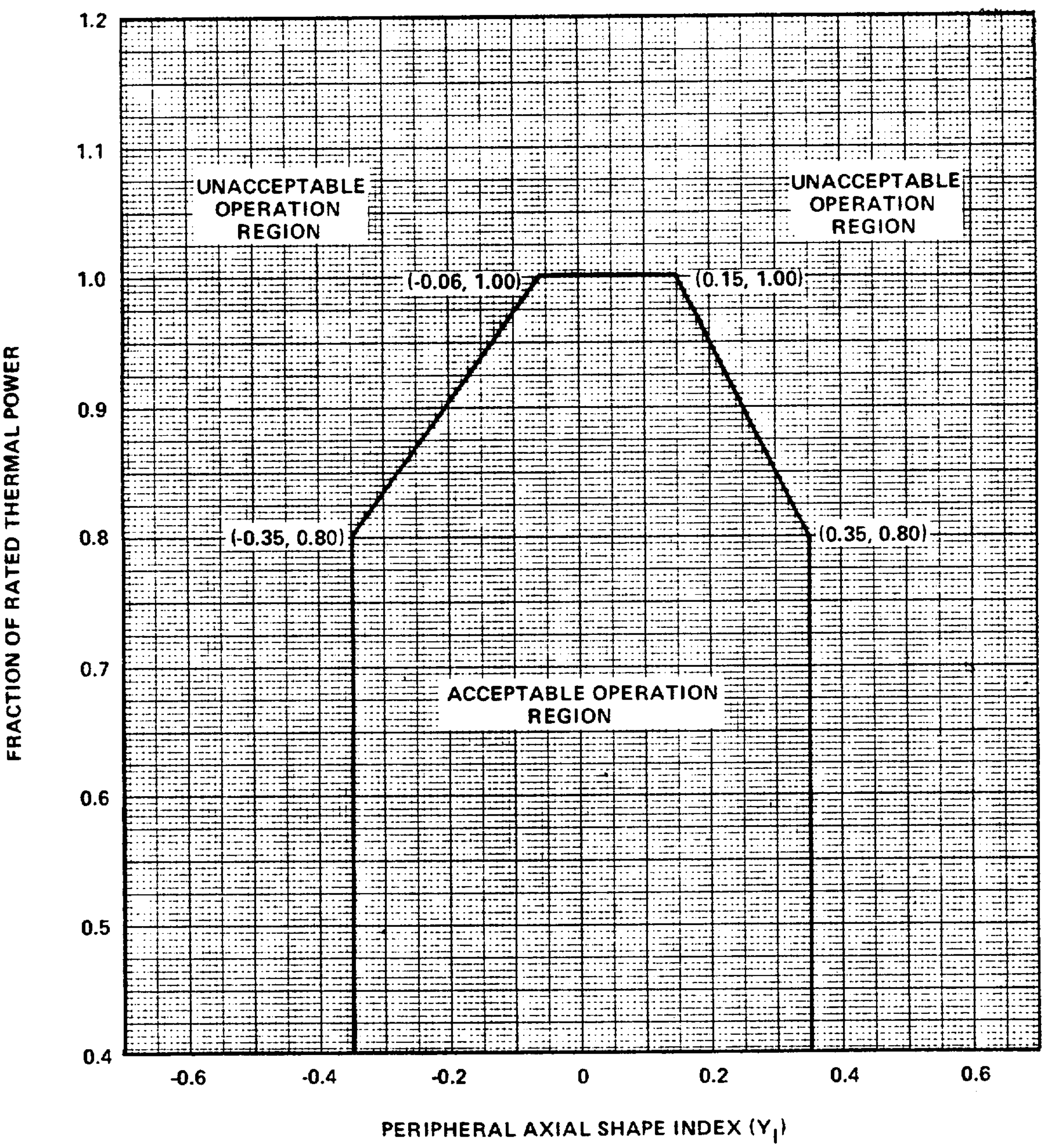

FIGURE 3.2-4
DNB Axial Flux Offset Control Limits

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

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

SURVEILLANCE 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.

TABLE 3.2-1

DNB PARAMETERS

Parameter	LIMITS			
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	≤ 547°F	≤ 547°F	≤ 547°F	≤ 547°F
Pressurizer Pressure	≥ 2225 psia*	≥ 2225 psia*	≥ 2225 psia*	≥ 2225 psia*
Reactor Coolant System Total Flow Rate	≥ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	Figure 3.2-4	Figure 3.2-4	Figure 3.2-4	Figure 3.2-4

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	$\leq$ 0.40 seconds*# and $\leq$ 5.0 seconds##
3. Reactor Coolant Flow - Low	$\leq$ 0.50 seconds
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Containment Pressure - High	$\leq$ 0.90 seconds
6. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
7. Steam Generator Water Level - Low	$\leq$ 0.90 seconds
8. Axial Flux Offset	$\leq$ 0.40 seconds*# and $\leq$ 5.0 seconds##
9. Thermal Margin/Low Pressure	$\leq$ 0.90 seconds*# and $\leq$ 5.0 seconds##
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

#Response time does not include contribution of RTDs.

##RTD response time only. This value is equivalent to the time interval required for the RTDs output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

BASES

3/4.2.1 LINEAR HEAT RATE

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

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.058, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

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

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

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

3/4.2.4 FUEL RESIDENCE TIME

The limitation on fuel burnup during the third fuel cycle insures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur in the limiting batch until well beyond the proposed third cycle of operation. However, operation beyond the specified third cycle fuel burnup limitation will require further analyses.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.19 throughout each analyzed transient.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 Introduction

By letter dated May 8, 1978[1], as supplemented by letters dated May 30, 1978[2] and June 21, 1978[3], the Baltimore Gas and Electric Company (BG&E or the licensee) requested an amendment to Facility Operating License No. DPR-53. The amendment would:

(1) Modify the operating limits in the Technical Specifications (TS) for Calvert Cliffs Nuclear Power Plant (CCNPP), Unit No. 1 for Cycle 3 operation. These changes include the following:

(a) Increase the allowed Integrated Radial Peaking Factor, F_r^T.

(b) Modify the Departure from Nucleate Boiling Ratio (DNBR) Axial Shape Index (ASI) Control Limit Tent.

(c) Increase the allowed Total Planar Radial Power Peaking Factor, F_{xy}^T.

(d) Modify the Linear Heat Rate (LHR) ASI Control Limit Tent.

(e) Modify the Control Element Assembly (CEA) Power Dependent Insertion Limit (PDIL).

(f) Modify the CEA Insertion Limit when monitoring LHR with Excore Detectors.

(g) Reduce Core Inlet Temperature.

(h) Reduce Low Flow Trip Response Time.

(2) Provide changes in the safety analysis methodology as follows:

(a) Compute DNBR by using the CE-1 correlation for which the minimum allowable DNBR during steady state operation and transient conditions is 1.19.

(b) Utilize the Combustion Engineering (CE) two stage TORC code to replace the COSMO-INTHERMIC code for thermal margin analysis.(5)

(c) Revise the Thermal Margin/Low Pressure (TM/LP) setpoint methodology.

(d) Introduce a relationship between Peaking Factors and speed of shutdown into the Loss of Flow (LOF) analysis.

(e) Modify the statistical method for combining uncertainties.

2.0 Background and Summary

By letter of March 31, 1978[4], we issued Amendment No. 32 to BG&E's operating license for CCNPP, Unit No. 1 which permitted the commencement of Cycle 3 operation. Although the plant is licensed for 100% of rated power, two circumstances have combined to reduce the present CCNPP, Unit No. 1 power producing capability to less than rated power. These are the following:

(1) Due to CEA guide tube wear during the previous two cycles, the fuel was shuffled in such a manner that only one special test assembly with worn guide tubes was placed under a CEA. This shuffle scheme resulted in an assembly pattern which produced higher power peaking than would have been achieved without the guide tube restriction.

(2) The original safety analysis (OSA) did not include a correction for what is now known to be a nonconservative prediction of fuel pin power peaking in pins adjacent to CEA guide tubes (water holes). A 2.8% penalty was required to accommodate this correction.

In order to alleviate the limits causing the power reduction, BG&E submitted a new safety analysis (NSA) for CCNPP Unit 1 Cycle 3 operation employing certain new analytical methodologies and imposing operating conditions which are more restrictive in some cases and less restrictive in other cases[1].

We are reviewing Reference 6 which develops the concept of nuclear peaking factors, F_r and F_q. From the data presented therein, we have determined that the F_r and F_q uncertainties of 5.2% and 5.8% used by CE in the CCNPP Unit 1 OSA are nonconservative. At the present stage of review, we feel that the greater uncertainties of 8% and 10% respectively can be justified. We previously accepted the use of 5.2% and 5.8% for other cases only because of the overall conservatism identified by CE in the calculational model, which we judged would offset the nonconservatism in the peaking factor uncertainties. In the NSA, BG&E describes a revised calculational model which, although conservative, removes some of the credits that had been applied to offsetting the nonconservatism in the peaking factor uncertainties. However, BG&E has identified areas of the NSA methodology which contain additional credits which in the interim can be used to offset the nonconservatism in the peaking factor uncertainties.

3.0 Evaluation

3.1 Changes to Operating Restrictions

The following changes in operating restrictions have been proposed:

(1) Reduce Power Dependent Insertion Limit (PDIL).

(2) Change DNB Axial Shape Index (ASI) Control Limits.

(3) Change LHR ASI Control Limits.

(4) Reduce Core Inlet Temperature.

(5) Withdraw CEA's to their long term insertion limit for LHR monitored on excore detectors only.

These operating restriction changes were the basis for input changes to the NSA.

Each of these changes to operating restrictions is discussed in the following sections.

3.1.1 DNB Limiting Safety System Setting (LSSS) - Thermal Margin/Low Pressure (TM/LP) Trip Setpoints

The PDIL would be changed to reduce the allowed CEA insertion at any given power level, hence reducing the axial power peaking factor and allowing an increase in radial peaking factor and a higher TM/LP trip setpoint. This change is a re-arrangement in allowable peaking factor, therefore, results in no change in margins.

3.1.2 LHR LSSS - Local Power Density Trip (Axial Shape Index (ASI) Trip Tent)

As just stated the proposed PDIL change would reduce axial power peaking, and hence allow greater radial power peaking and higher power trip points.

3.1.3 DNB LCO - ASI Control

3.1.3.1 ASI Control Limits

The allowed ASI power level control limit as a function of F_r^T (TS Figure 3.2-3) would change to allow larger values of F_r^T. In particular, for 100% power, the allowed value of F_r^T would be increased from 1.48 for the OSA to 1.65 for the NSA. This change would accommodate the fact that Cycle 3 experience has demonstrated that 1.65 is the lowest F_r^T that is achievable during steady state operation at or near the beginning of the cycle. Since this operating change was incorporated in the NSA, it is acceptable.

3.1.3.2 Core Inlet Temperature

The TS limit on core inlet temperature would be reduced from 548 to 547 F which serves to increase the DNBR.

3.1.4 LHR LCO - ASI Control

3.1.4.1 ASI Control Limits

The allowed ASI power level alarm limit as a function of F_{xy}^T (TS Figure 3.2-3) would change to allow larger values of F_{xy}^T. In particular, for 100% power the allowed value of F_{xy}^T would be increased from 1.54 for the OSA for Cycle 3 operation to 1.66 for the NSA for Cycle 3 operation. This change would accommodate the fact that Cycle 3 experience has demonstrated that 1.66 is the lowest F_{xy}^T that is achievable during steady state operation at or near the beginning of the cycle. Since this change was incorporated in the new analysis, it is acceptable.

3.1.4.2 CEA Insertion

The TS allow operation with LHR monitored by either the Continuoue Incore Detector Alarms or the Excore Detector ASI Alarms. For monitoring with the Excore Alarms, the licensee has proposed a restriction that the CEA's must be withdrawn to their long term insertion limits. This would reduce power peaking when monitoring with the Excore Detector ASI Alarms, and thus would allow higher LHR Alarm power setpoints.

3.1.5 Conclusion on Operating Restrictions

As stated previously, the proposed changes in operating restrictions have been incorporated in formulating the input for the NSA so that the operating restrictions and NSA form a consistent package. On this basis, we conclude that the use of the revised operating restrictions is acceptable and does not result in any reduction in safety margin.

3.2 Changes in the Analysis for Cycle 3 Operation - Computional Methodology

3.2.1 TORC Reanalysis of Thermal Margin

The three stage TORC thermal hydraulics computer code[5] has been developed to replace COSMO-INTHERMIC. TORC employs the CE-1 DNBR correlation whereas COSMO-INTHERMIC employs the W-3 DNBR correlation. The three stage TORC code has been approved for use in licensing and the CE-1 correlation has been approved with a 1.19 DNBR limit.[12] The NSA was performed with the two stage production version of TORC.[8] While the review of two stage TORC has not been completed by the NRC staff, the review has progressed to the point that we judge the use of the two stage TORC to be acceptable. The two stage TORC has been benchmarked against the more accurate three stage TORC and is conservative relative to the three stage TORC. TORC/CE-1 produces results in better agreement with experiment than COSMO-INTHERMIC/W-3. Based on these considerations, we find the use of two stage TORC with a CE-1 DNBR limit of 1.19 to be acceptable.

Although the use of TORC/CE-1 involves a change in the DNBR safety limit from 1.30 to 1.19, there is no change in the acceptance criteria with respect to fuel damage. The change is a result of new experimental data and the CE-1 correlation derived from the data on file. The previous limit of 1.30 was for the W-3 correlation and the limit of 1.19 is for the CE-1 correlation. Either of these limits, when considered in conjunction with its DNBR correlation, corresponds to a 95% probability at a 95% confidence level that DNB will not occur. The use of TORC impacts the DNBR LSSS and LCO.

3.2.2 Statistical Combination of Uncertainties

In the OSA the following uncertainties were combined multiplicatively:

F_Q^E = Engineering Factor

F_Q^N = Nuclear Factor

F_Q^F = Fuel Rod Bowing Factor

F_Q^P = Poison Rod Bowing Factor

In Reference 7, CE proposed that these uncertainties could appropriately be combined Root-Sum-Square (RSS). We concur with this method. The RSS combination has been used in the NSA. The RSS combination impacts the DNBR LSSS, LHR LSSS, DNBR LCO and LHR LCO.

3.2.3 TM/LP Cross-Plotting Methodology

In the previous TM/LP methodology described in Reference 9, the computational technique is simplified by replacing certain curves with conservative linear approximations. In Reference 7, CE indicated that the linear approximation introduced was overly conservative in the TM/LP setpoint calculation (from 3% to 15% in power) and indicated how the computation could be performed in a more realistic manner. Responses to our concerns about the new TM/LP methodology were addressed in Reference 11. Based on the information in References 9, 7 and 11, we conclude that the new TM/LP methodology used in the NSA for Cycle 3 operation is conservative and is acceptable. The TM/LP methodology impacts the DNBR LSSS only.

3.2.4 LHR LSSS (ASI Trip Tent)

The Trip Tent that is included in current TS Figure 2.2-1 was constructed to lie within the LSSS Tent computer in the OSA. It was found that the current TS Figure 2.2-1 was also within the LSSS Tent computer in the NSA and hence TS Figure 2.2-1 was not revised. TS Figure 2.2-1 does not lie as far below the NSA LSSS Tent as it did below the OSA LSSS Tent, but the ASI Trip Tent in the proposed TS is still conservative. As will be indicated again in Section 3.3.2, the degree of conservatism in the new ASI Trip Tent is still at least 5%.

3.2.5 Reduction of the DNBRLCO ASI Monitoring Limits

The DNB ASI Control Limit Tent for the new Cycle 3 analysis (proposed TS Figure 3.2-4) was constructed to be consistent with the changes indicated in sections 3.1.1. 3.1.3. 3.2.1, 3.2.2, and 3.2.3.

3.2.6 Reduction of Required Overpower Margin (ROPM) for the Loss of Flow (LOF) Transient

The calculated ROPM values for the LOF transient are factored into the development of the DNBR LCO (proposed TS Figures 3.2-3 and 3.2-4). The reduction in ROPM of 3.5% resulted from two considerations:

(1) Faster response time for Low Flow Trip; and

(2) More consistent treatment of overpower margin from initially rodded and unrodded cases.

The faster response time for the Low Flow Trip is based on the response times observed during the periodic surveillance. Previously, a response time of 0.65 seconds had been assumed in the OSA. In the periodic surveillance, the measured response time was 0.38 ± 0.03 seconds. The time response assumed in the NSA is 0.5 seconds, which is conservative relative to the measured value. Use of the 0.5 second response time instead of 0.65 second response time resulted in a reduction in ROPM of 1.5%.

BG&E has proposed a more consistent treatment of overpower margin for CCNPP, Unit 1 as follows:

(a) the largest power peaking factors and the fastest shutdown reactivity insertion are postulated to occur when the LOF transient begins with the CEA's inserted at the PDIL; and

(b) the smallest power peaking factors and the slowest shutdown reactivity insertion are postulated to occur when the LOF transient begins with the CEA fully withdrawn.

Previously, BG&E had assumed the highest peaking factors were combined with the slowest shutdown reactivity insertion. We agree that BG&E's proposed change is a more logical approach, and is adequately conservative. This change would result in a 2% reduction of ROPM.

3.2.7 Reduction of the LHR LCO ASI Monitoring Limits

The LHR ASI Control Limit Tent for the Cycle 3 NSA (proposed TS Figure 3.2-2) was constructed to be consistent with the changes proposed in sections 3.1.2, 3.1.4, 3.2.2 and 3.2.4.

3.2.8 Conclusions on the New Analysis for Cycle 3 Operation

We conclude that none of the above changes in the methodology of analysis introduces nonconservatism in the new analysis for Cycle 3 operation and on this basis we find that these changes are acceptable.

3.3.0 Uncertainty in Nuclear Power Peaking Factors

3.3.1 Documentation of Uncertainties

Reference 6, which is still under review by the NRC staff, documents the assumed uncertainties in F_r^T and F_q^T of 5.2% and 5.8% and Reference 7 documents the 4.6% water hole power peaking bias. At the present stage of review, we conclude that the 5.2% and 5.8% are nonconservative. In addition, the 4.6% should have an uncertainty associated with it which the licensee has not factored into their analysis. However, whatever uncertainty is inherent in the 4.6% could logically be applied to the 5.2% and 5.8%, and our present position is to accept the 4.6% fully and assign any uncertainties in the water hole peaking to the uncertainties in F_r^T and F_q^T.

3.3.2 Status of NRC Staff Review of CE Uncertainties

We have submitted to BG&E an extensive list of questions concerning the justification for the 5.2% and 5.8% (Reference 10). These questions were answered in part in Reference 11. However, a number of responses were not complete, and many of the questions of Reference 10 have not yet been addressed. With the available data, we conclude that uncertainties of 8% for F_r^T and 10% for F_q^T can be justified, but not the uncertaintes of 5.2% and 5.8%.

3.3.3 Uncertainty in F_r and F_q Used in NSA

In Reference 7, which documents the water hole peaking bias of 4.6%, certain computational conservatisms are cited which could act as credits to mitigate the effects of the additional 4.6% peaking. Based on these credits, we agree that an additional penalty of 2.8% rather than 4.6% could be justified. With the adoption of BG&E's revised computational model, this reduction in penalty can no longer be justified and BG&E has assumed the full 4.6% peaking bias penalty in the new analysis for Cycle 3 operation of CCNPP Unit 1.

There is no specific mechanism for introducing the 4.6% bias into the NSA calculations. Rather than revamp the calculational model to incorporate the 4.6% bias, the bias is simply added to the uncertainty in F_r^T and F_q^T that is used in the NSA. This gives an uncertainty in F_r^T of 5.2% + 4.6% = 9.8% and an uncertainty in F_q^T of 5.8% + 4.6% = 10.4%. These 9.8% and 10.4% uncertainties were used throughout the NSA.

3.3.4 Conservatisms in the NSA to Offset Nonconservatism in Uncertainties

In lieu of justifying the 5.2% and 5.8%, BG&E has cited a number of known conservatisms in the NSA methodology for CCNPP Unit 1 for which they do not take credit[2,3], which would offset the nonconservatism in the 5.2% and 5.8%. These are enumerated in the following Sections 3.3.4.1 and 3.3.4.2.

3.3.4.1 DNBR Conservatism

There are four areas in which conservatisms are known to exist in the DNBR NSA. In the first two of these, the degree of conservatism is known, since comparisons with a more exact model have been performed. In the latter two, the degree of conservatism has not been made, however it is clear that they are conservative. These four areas of conservatism are:

3.3.4.1.1 Conservatism in the Production TORC

As stated previously, two versions of TORC exist, the detailed more accurate three stage TORC, and the fast running two stage production TORC (TORC/CE-1) which is used in the NSA. The production TORC has been benchmarked against the three stage TORC and is known to be conservative relative to it. In addition, the fuel pin power distribution input to TORC/CE-1 is more homogeneous than expected, and thus predicts less turbulent fluid interchange than the actual power distribution produces. The combined effect of the two above factors results in a conservatism of up to 3% depending on plant conditions. BG&E has stated that for most plant conditions a conservatism of 2% to 3% exists. However, under some unlikely plant conditions, the degree of conservatism becomes much smaller and therefore we will not allow credit for this conservatism.

3.3.4.1.2 Conservatism in TM/LP Setpoint Due to Statistical Combinations

BG&E has identified a 6.0% credit from the statistical combinations used in the TM/LP analysis [3]. In the TM/LP analysis, the assumed uncertainties in various measured parameters are not combined in a single equation, but are factored into functional relationships as biases at various points in the analysis. BG&E states that this biasing of functional relationships throughout the analysis is tantamount to multiplying the relative power uncertainties equivalent to the uncertainties in the various measured parameters and applying the total power uncertainty to the best estimate calculation. We conclude that the biasing of functional relationships is actually equivalent to adding the absolute power uncertainties equivalent to the measured parameters uncertainties. This difference is not significant in the present context in that the difference in the credit computed in the two manners is less than 1%. The specific uncertainties, along with our estimate of their equivalent power uncertainties, are as follows:

Parameter	Measurement Uncertainty (Reference II, Table 5-3) (Roughly 2σ level)	Equivalent Power Uncertainty (Staff estimate) (% Power)
ASI	0.06 ASIU	2.2 %
Pressure	22 PSI	0.8 %
Temperature	2 DEGF	0.9 %
Flow	4 % FlOW	5.0 %
Power	5 % Power	3.5 %

The sum of these uncertainties is 12.4% power. Since these measurement uncertainties are statistically independent, the proper method for combining them is RSS. The RSS combination yields 6.6% power giving a net conservatism in the analysis of 5.8% power. This compares well with the 6.0% relative power credit BG&E is claiming for CCNPP Unit 1. We conclude that the 6.0% credit for this Cycle is acceptable. The conversion factors from the various measurement uncertainties are plant and cycle dependent, and thus this credit must be recomputed if it is to be applied in other cases.

The 6.0% credit cited here is for the DNBR LSSS. CE states that the Equivalent Power Uncertainty corresponding to the various Measurement Uncertainties would be greater for the DNBR LCO than the DNBR LSSS, so that a greater credit is available for the DNBR LCO. Of the four conservatisms cited here the Statistical Combination Conservatism is the only one for which we are allowing credit. The credit required is 2.8% and the credit demonstrated is 6.0%.

3.3.4.1.3 Conservatism in Pseudo-Hot- Pin Synthesis

In computing F_r^T the pseudo-hot-pin synthesis is used in INCA[6]. In this technique, the axially integrated hot pin power is computed by integrating the hottest pin in each axial region. If a single pin is the hottest in every axial region, then this technique produces correct results, and if the hot pin is different in different axial regions, as BG&E states is realistically expected, the results will be conservative. We will not allow credit for this conservatism because it has not been quantified.

3.3.4.1.4 Conservatism in Axial Flux Shapes

In the multiplicity of QUIX axial flux shapes (9) used in the NSA, more severe shapes are calculated than are expected to occur during actual operation. This results in conservatism in the NSA. We will not allow credit for this conservatism because it has not been quantified.

3.3.4.2 LHR Conservatisms

For the Loss of Coolant Accident (LOCA) NSA assumes that the fuel LHR must be limited to 14.2 kw/ft for all fuel. The fresh fuel, in fact, can withstand 16.0 kw/ft. Since the old fuel power peaks are less than 85% of the new fuel power peaks, there is conservatism in the NSA which amounts to approximately 5% in LHR LCO and LHR LSSS. The credit required is 4.2% and the credit demonstrated is 5%.

The TS ASI Trip Tent is constructed to lie within the NSA LSSS Tent. As indicated in section 3.2.4, the power level at which a trip will occur has been increased to be closer to the NSA predicted power level, but the construction still contains at least 5% conservatism. The NRC staff's acceptance criteria for the ASI Trip Tent has not changed and a TS ASI Trip Tent identical to the NSA ASI LSSS Tent would be acceptable to us. Therefore, we find it acceptable to use the 5% margin to offset nonconservatisms in the LHR LSSS. Since more credit than is required was identified in the last paragraph it is not necessary to invoke this additional 5% credit.

3.3.4.3 Staff Evaluation of Credits

The required credits to offset the nonconservatism in the peaking factors are the following:

DNBR LCO & LSSS	LHR LCO & LSSS
8.0 - 5.2 = 2.8%	10.0 - 5.8 = 4.2%

Since we have just demonstrated credit in excess of these figures, we find it acceptable to use the identified margin to offset the nonconservatism in F_r^T and F_q^T.

4.0 Conclusions

We conclude that with the proposed operating restrictions, the changes in computational methodology, and the credits applied to the uncertainties, the fuel performance will be well within the limits of the acceptance criteria previously applied and, therefore, result in no significant reduction in safety margins.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: June 30, 1978

REFERENCES

1. Request for Amendment to Operating License DPR-53 for Calvert Cliffs Unit No. 1, letter dated May 8, 1978, from A. E. Lundvall, Jr., Baltimore Gas and Electric Company to R. W. Reid, NRC.

2. Response to Additional Information Request, letter dated May 30, 1978, from A. E. Lundvall, Jr., Baltimore Gas and Electric Company, to R. W. Reid, NRC.

3. Response to Additional Information Request, letter dated June 21, 1978, from A. E. Lundvall, Jr., Baltimore Gas and Electric Company, to R. W. Reid, NRC.

4. Baltimore Gas and Electric Company, Docket No. 50-317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Amendment No. 32 to Facility Operating License No. DPR-53, letter from Robert W. Reid, NRC, to A. E. Lundvall, Jr., BG&E, March 31, 1978.

5. CENPD-161-P, TORC Code: A computer Code for Determining the Thermal Margin of a Reactor Core, Combustion Engineering, July 1975.

6. CENPD-145, INCA: Method of Analyzing In-Core Detector Data in Power Reactor, Ober, Terney, and Marks, Combustion Engineering, April 1975.

7. CEN-85(B)-P, Solution to Increased Water Hole Peaking on Operating Reactors, Combustion Engineering, February 16, 1978.

8. CENPD-206-P, TORC Code Verification and Simplified Modeling Methods, Combustion Engineering, January 1977.

9. CENPD-199, CE Setpoint Methodology, Combustion Engineering, April 1976.

10. Request for Additional Information, letter from R. W. Reid, NRC, to A. E. Lundvall, Jr., BG&E, April 4, 1978.

11. Response to Request for Additional Information on CEN-85(B)-P, letter from A. E. Lundvall, Jr., BG&E, to R. W. Reid, NRC, April 28, 1978.

12. Evaluation of Topical Report CENPD-161-P, letter from K. Kniel, NRC, to A. E. Scherer, CE, September 14, 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-317

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. DPR-53, issued to Baltimore Gas & Electric Company (the licensee), which revised 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 its date of issuance.

The amendment changes the Technical Specifications based on the reanalysis of the Cycle 3 core thermal hydraulic characteristics using new Combustion Engineering computer codes.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 8, 1978, as supplemented May 30 and June 21, 1978, (2) Amendment No. 33 to License No. DPR-53, and (3) the Commission's related Safety Evaluation. 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. A copy of items (2) and (3) 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 30th day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors