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JUNE 1 4 1979

Docket No. 50-317

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

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. DPR-53 for Calvert Cliffs Nuclear Power Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications (TS) in accordance with your request dated February 23, 1979 and supplemental information dated January 12, February 7, March 13 and May 7, 29 and 31, 1979.

The amendment authorizes operation with modified (sleeved and reduced flow) guide tubes for the Control Element Assemblies (CEA) and with a high burnup demonstration fuel assembly installed in the core. The amendment also revises the Appendix A TS to incorporate changes resulting from the analyses of Cycle 4 reload fuel.

In your application for Part-Loop Operation, dated April 5, 1979, you requested that our review of this type of operation be handled concurrently with the reload amendment request. This has not been possible without a delay in the reload authorization.

Section 4.3 of your February 23, 1979 application, as supplemented by your letter dated March 5, 1979, requested authorization to use a prototype CEA in the core. Since these submittals, your staff has informed us that the prototype CEA will not be used in Unit 1 Cycle 4. We will complete this evaluation at a later date.

On March 28, 1979, Three Mile Island Unit No. 2 (TMI-2) experienced core damage which resulted from a series of events which were initiated by a Loss of Feedwater Event and apparently compounded by operational errors. We believe that several aspects of this accident have generic applicability to all light water power reactor facilities such as the Calvert Cliffs units. To identify corrective actions to be taken by all licensees, I&E bulletins have been issued since the TMI-2 Accident. The particular bulletin that applies to the Combustion Engineering facilities is Bulletin No. 79-06B.

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You provided your response to Bulletin No. 79-06B in letters from A. Lundvall to B. Grier dated April 26 and May 8, 1979. Our evaluation of the response indicates that the actions taken by your staff demonstrate understanding of the concerns

OFFICE➤						
SURNAME➤					7907260084	
DATE➤						

arising from the TMI-2 Accident in reviewing their implications on the Calvert Cliffs units operation, and provide added assurance for the protection of the public health and safety during plant operation. A separate safety evaluation will be issued documenting our review of your response to I&E Bulletin No. 79-06B and identifying certain areas where additional information or action is needed.

The problem you have experienced with the CEA guide tube sleeves during the current refueling outage confirms that sleeving cannot, at this time, be considered a permanent solution to this problem. The performance of the CEA sleeved and reduced flow guide tubes will need to be evaluated at the end of Cycle 4 operation. Your staff has agreed to provide an evaluation program (including the planned inspections) to determine the amount of guide tube wear experienced after two cycles of operation with sleeved fuel assemblies and one cycle with the reduced flow demonstration test. In addition, your staff has agreed to implement the guidance from Combustion Engineering that CEA movement should be restricted to system temperatures below 400 F except for normal movement associated with refueling operations.

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

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

Sincerely,

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

Enclosures:
1. Amendment No. 39 to DPR-53
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

STS G
Brinkman
6/5/79

OELD
CWoodhead
6/7/79

OFFICE	ORB#4:DOR	ORB#4:DOR	C-RSB:DOR	C-EB:DOR	C-ORB#4:DOR	AD-E&P:DOR
SURNAME	PKreutzer	MConner	FCheck	VNoonan	RReid	BGrimes
DATE	6/5/79	6/5/79	6/4/79	6/5/79	6/5/79	6/7/79

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

June 14, 1979

Docket No. 50-317

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

Dear Mr. Lundvall:

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You provided your response to Bulletin No. 79-06B in letters from A. Lundvall to B. Grier dated April 26 and May 8, 1979. Our evaluation of the response indicates that the actions taken by your staff demonstrate understanding of the concerns arising from the TMI-2 Accident in reviewing their implications on the Calvert Cliffs units operation, and provide added assurance for the protection of the public health and safety during plant operation. A separate safety evaluation will be issued documenting our review of your response to I&E Bulletin No. 79-06B and identifying certain areas where additional information or action is needed.

The problem you have experienced with the CEA guide tube sleeves during the current refueling outage confirms that sleeving cannot, at this time, be considered a permanent solution to this problem. The performance of the CEA sleeved and reduced flow guide tubes will need to be evaluated at the end of Cycle 4 operation. Your staff has agreed to provide an evaluation program (including the planned inspections) to determine the amount of guide tube wear experienced after two cycles of operation with sleeved fuel assemblies and one cycle with the reduced flow demonstration test. In addition, your staff has agreed to implement the guidance from Combustion Engineering that CEA movement should be restricted at system temperatures below 400 F except for normal movement associated with refueling operations.

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

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

Sincerely,

Robert W Reid

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

Enclosures:
1. Amendment No. 39 to DPR-53
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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U. S. Environmental Protection Agency
Region III Office
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Curtis Building (Sixth Floor)
Sixth and Walnut Streets
Philadelphia, Pennsylvania 19106

cc w/4 cys enclosures and 1 cy
of BG&E filings dtd: 2/23, 1/12, 2/7,
3/13, 5/7, 29 & 31/1979.
Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
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. 39
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated February 23, 1979 as supplemented, 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.

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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. 39, 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 14, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 39

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

IV
1-3
2-7
2-12
2-13
B 2-3
B 2-4
B 2-6
B 2-7
3/4 1-23
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INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals.

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High		
a. Four Reactor Coolant Pumps Operating	$\leq$ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.	$\leq$ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.
b. Three Reactor Coolant Pumps Operating	$\leq$ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of $\leq$ 80% of RATED THERMAL POWER.	$\leq$ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of $\leq$ 80% of RATED THERMAL POWER.
c. Two Reactor Coolant Pumps Operating - Same Loop	$\leq$ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of $\leq$ 46.8% of RATED THERMAL POWER.	$\leq$ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of $\leq$ 46.8% of RATED THERMAL POWER.
d. Two Reactor Coolant Pumps Operating - Opposite Loops	$\leq$ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of $\leq$ 51.1% of RATED THERMAL POWER.	$\leq$ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of $\leq$ 51.1% of RATED THERMAL POWER.

TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Reactor Coolant Flow - Low (1)		
a. Four Reactor Coolant Pumps Operating	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*
b. Three Reactor Coolant Pumps Operating	$\geq$ 72% of design reactor coolant flow with 4 pumps operating*	$\geq$ 72% of design reactor coolant flow with 4 pumps operating*
c. Two Reactor Coolant Pumps Operating - Same Loop	$\geq$ 47% of design reactor coolant flow with 4 pumps operating*	$\geq$ 47% of design reactor coolant flow with 4 pumps operating*
d. Two reactor Coolant Pumps Operating - Opposite Loops	$\geq$ 50% of design reactor coolant flow with 4 pumps operating*	$\geq$ 50% of design reactor coolant flow with 4 pumps operating*

*Design reactor coolant flow with 4 pumps operating is 370,000 gpm.

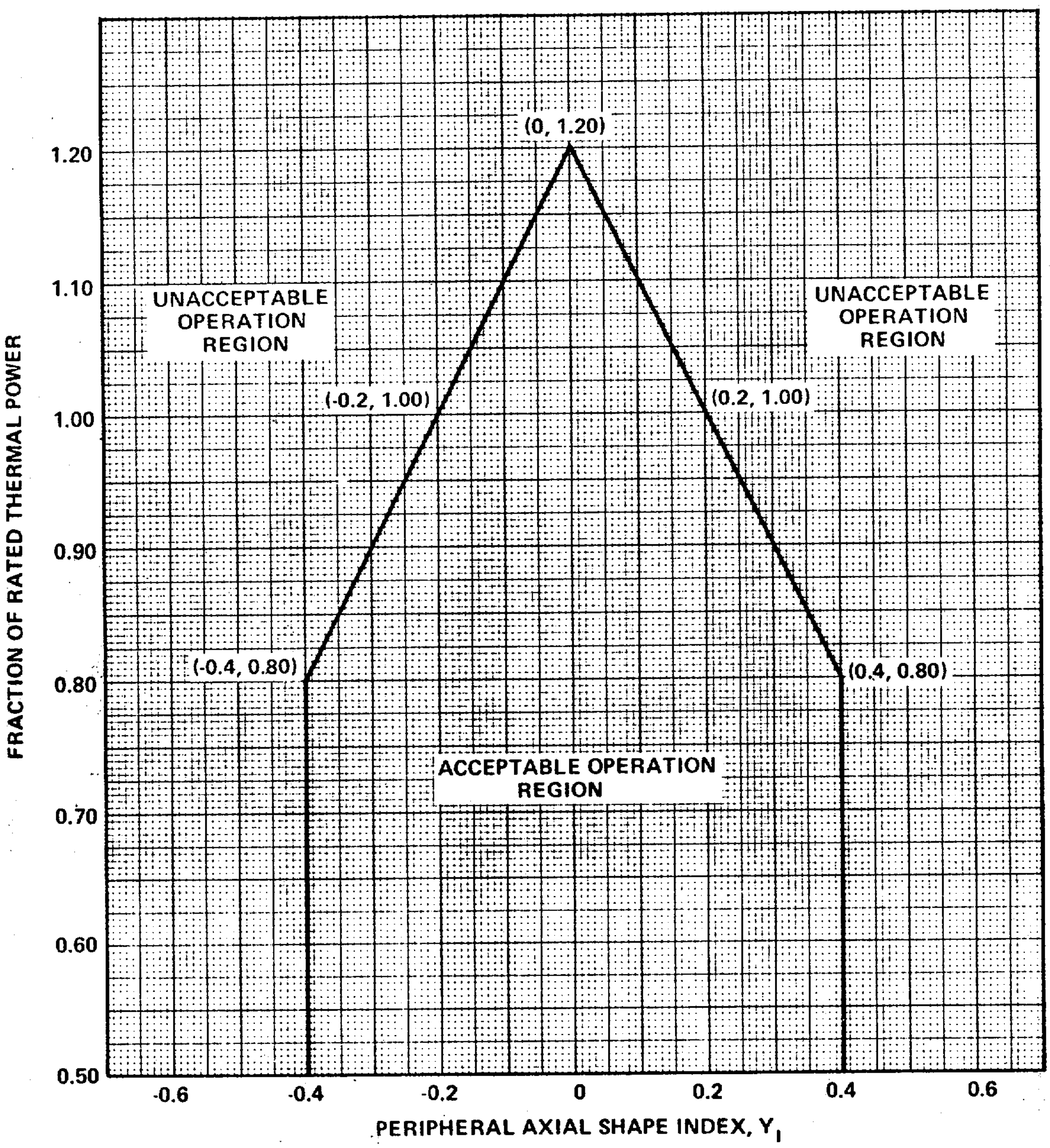

FIGURE 2.2-1
Peripheral Axial Shape Index, Y_I Versus Fraction of RATED THERMAL POWER

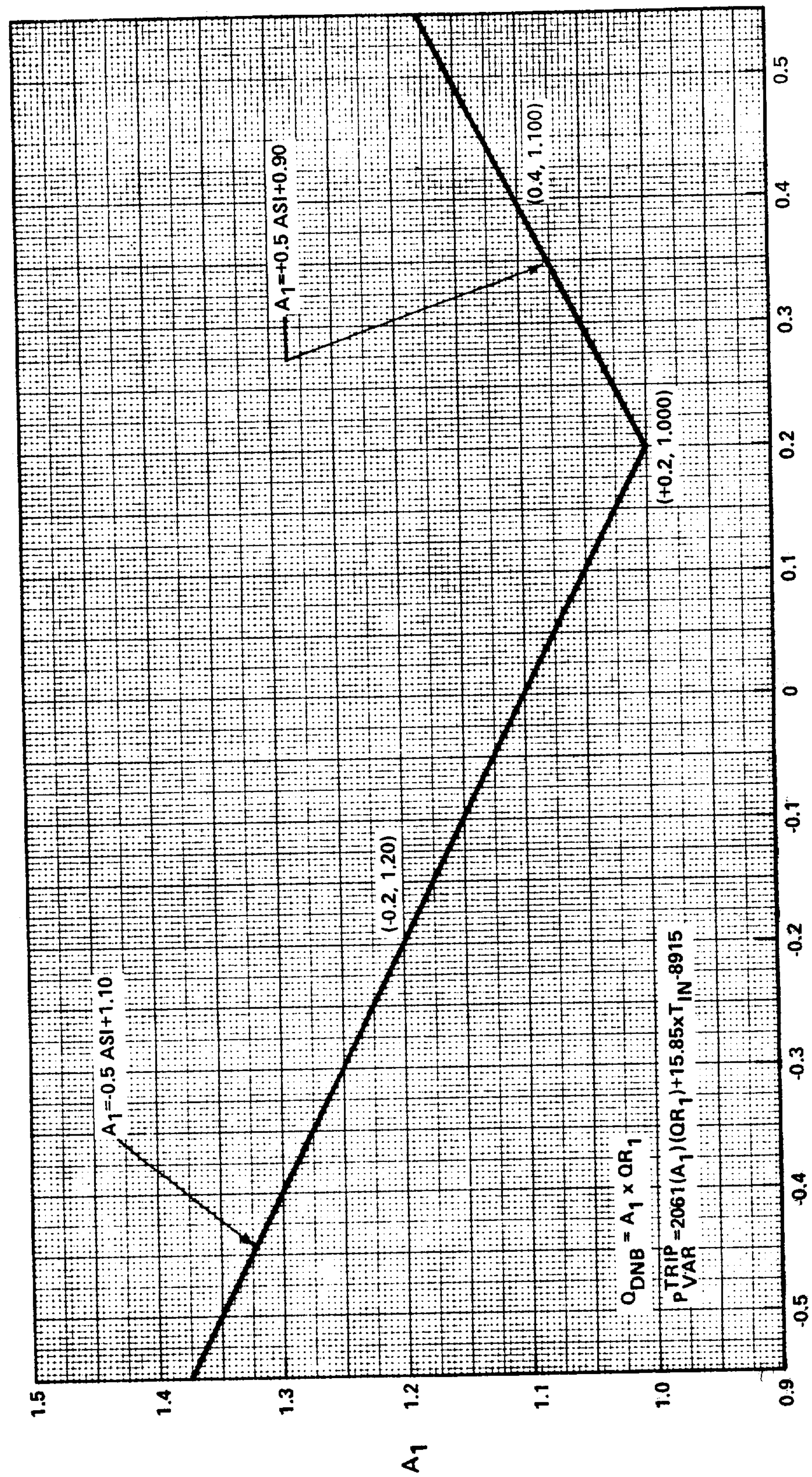

FIGURE 2.2-2

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

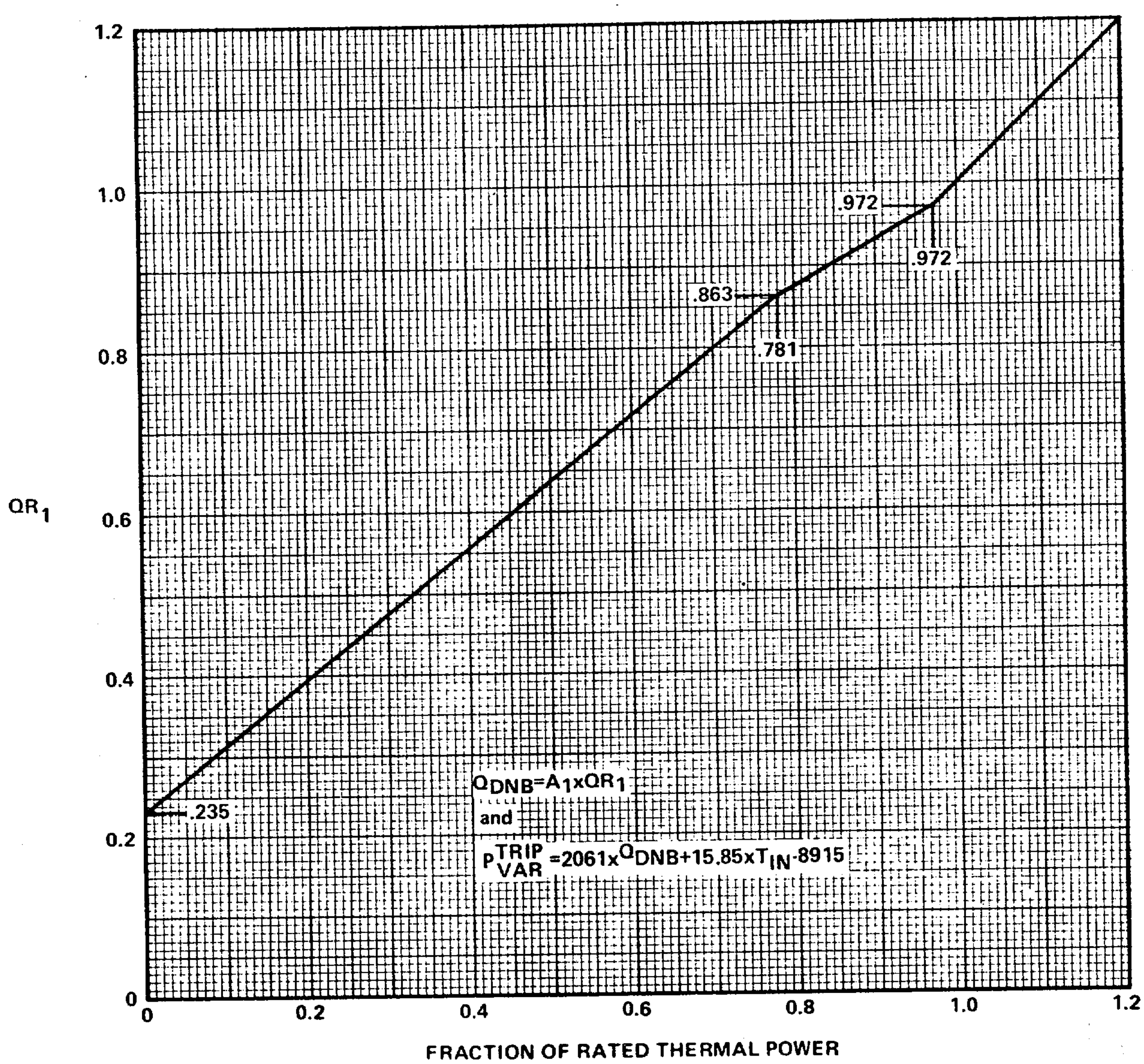

FIGURE 2.2-3

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

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of ECCS analysis for three pump operation.

Figure 2.2-4

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

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

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

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than 1.19 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 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SETPOINTS

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

Manual Reactor Trip

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

Power Level-High

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

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 30% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 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 $\pm$ 22 psi in the accident analyses.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint 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.

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.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time, measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 84 psia to compensate for pressure measurement error, trip system processing error, and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 84 psia allowance is made up of a 22 psia pressure measurement allowance and a 62 psia time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

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

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*#.

ACTION:

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

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

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

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

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^T_{xy} 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 1.070,

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.

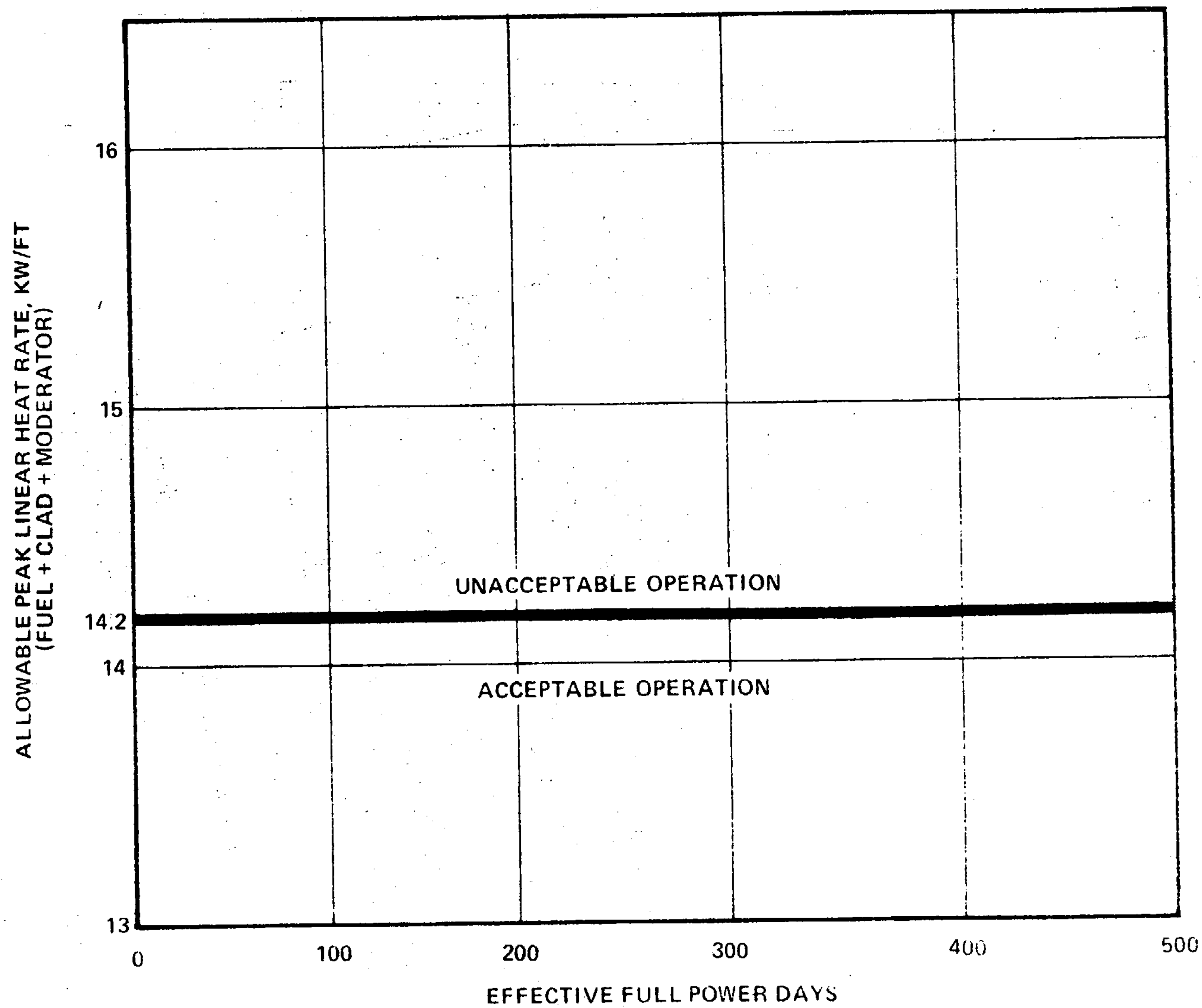

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

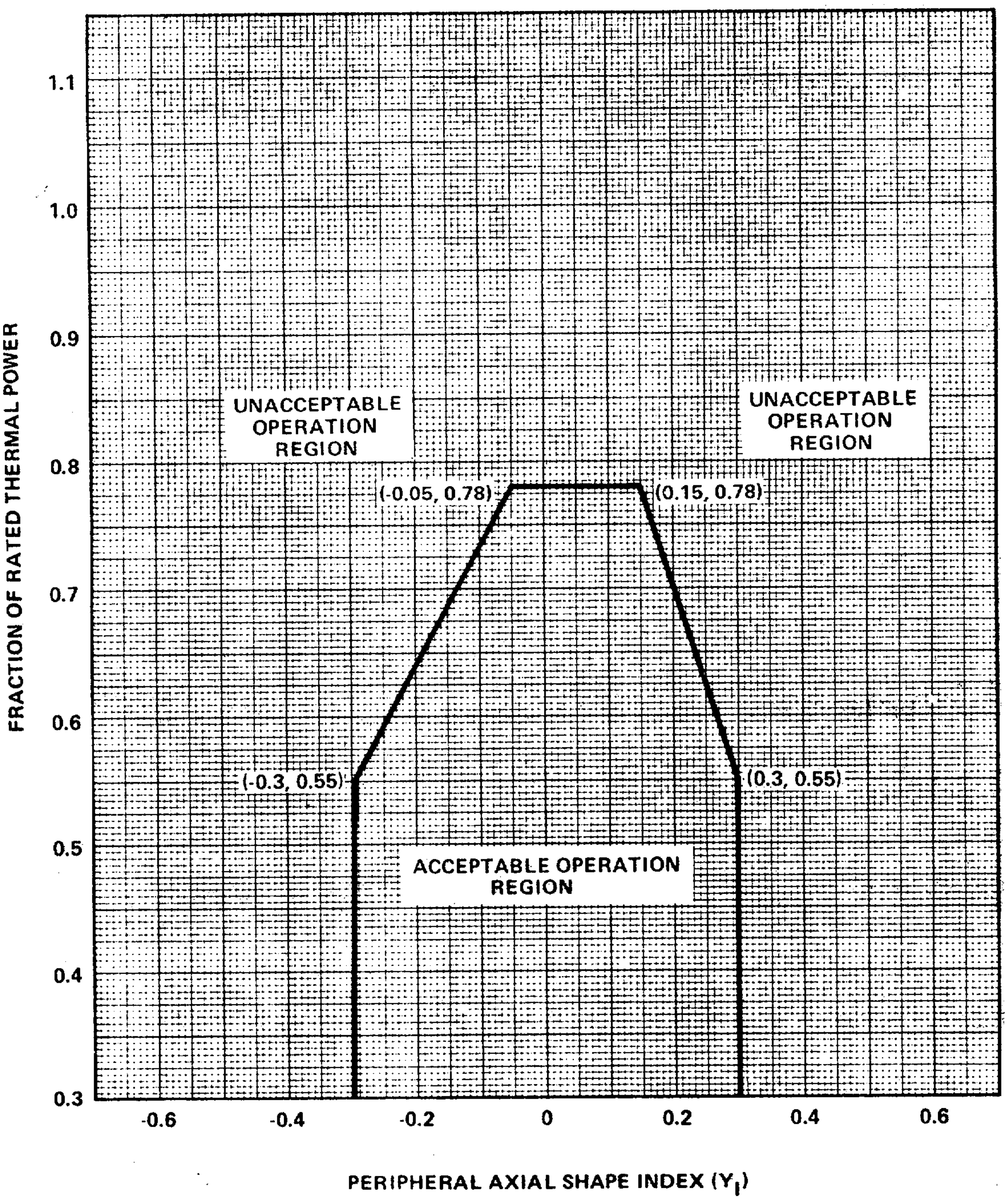

FIGURE 3.2-2
Linear Heat Rate
Axial Flux Offset Control Limits

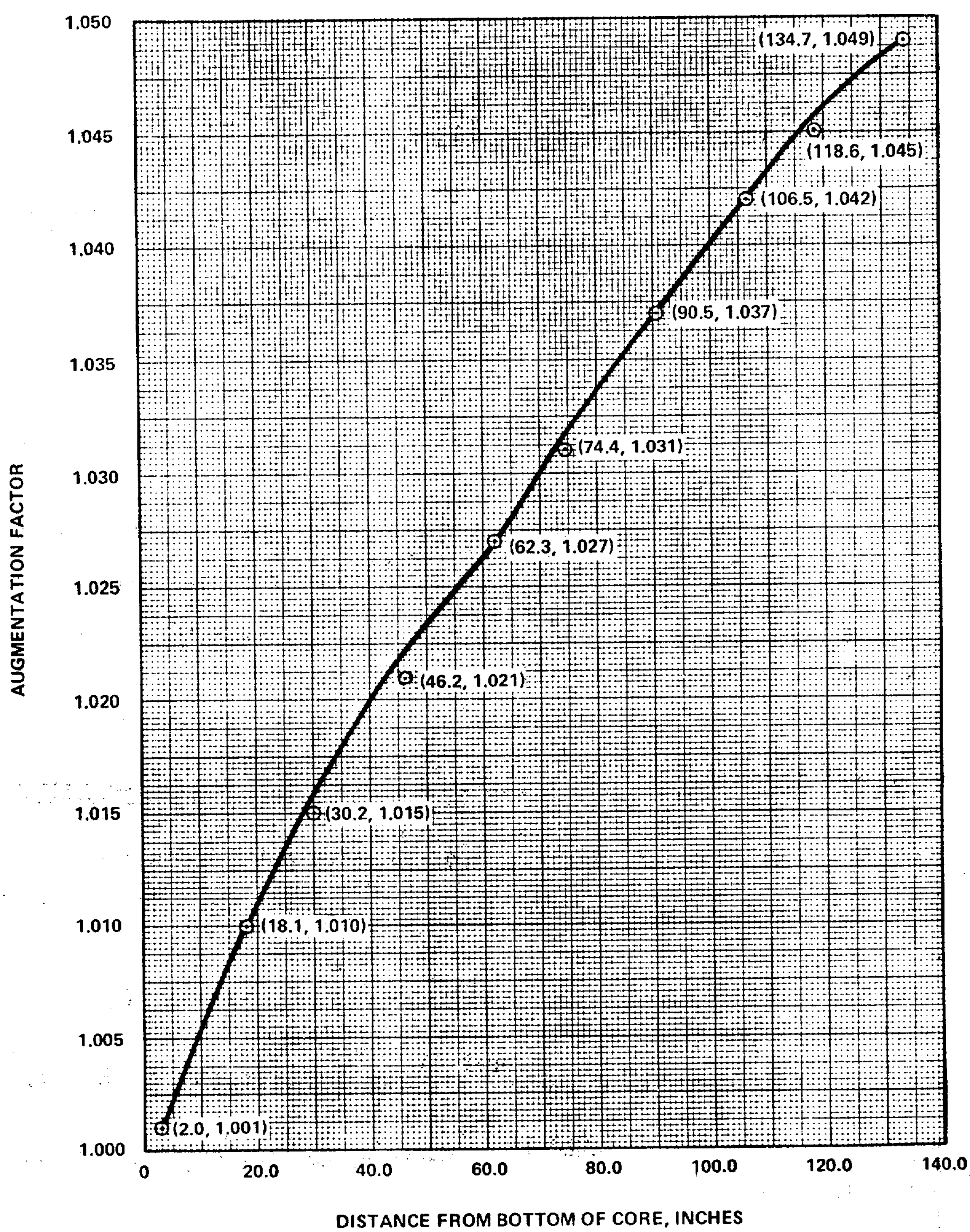

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

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

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_{xy}^T, defined as $F_{xy}^T = F_{xy}(1+T_q)$, shall be limited to ≤ 1.660.

APPLICABILITY: MODE 1*.

ACTION:

With $F_{xy}^T > 1.660$, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^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; 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_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}(1+T_q)$ and F_{xy}^T shall be determined to be within its limit at the following intervals:

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

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

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

*See Special Test Exception 3.10.2.

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.

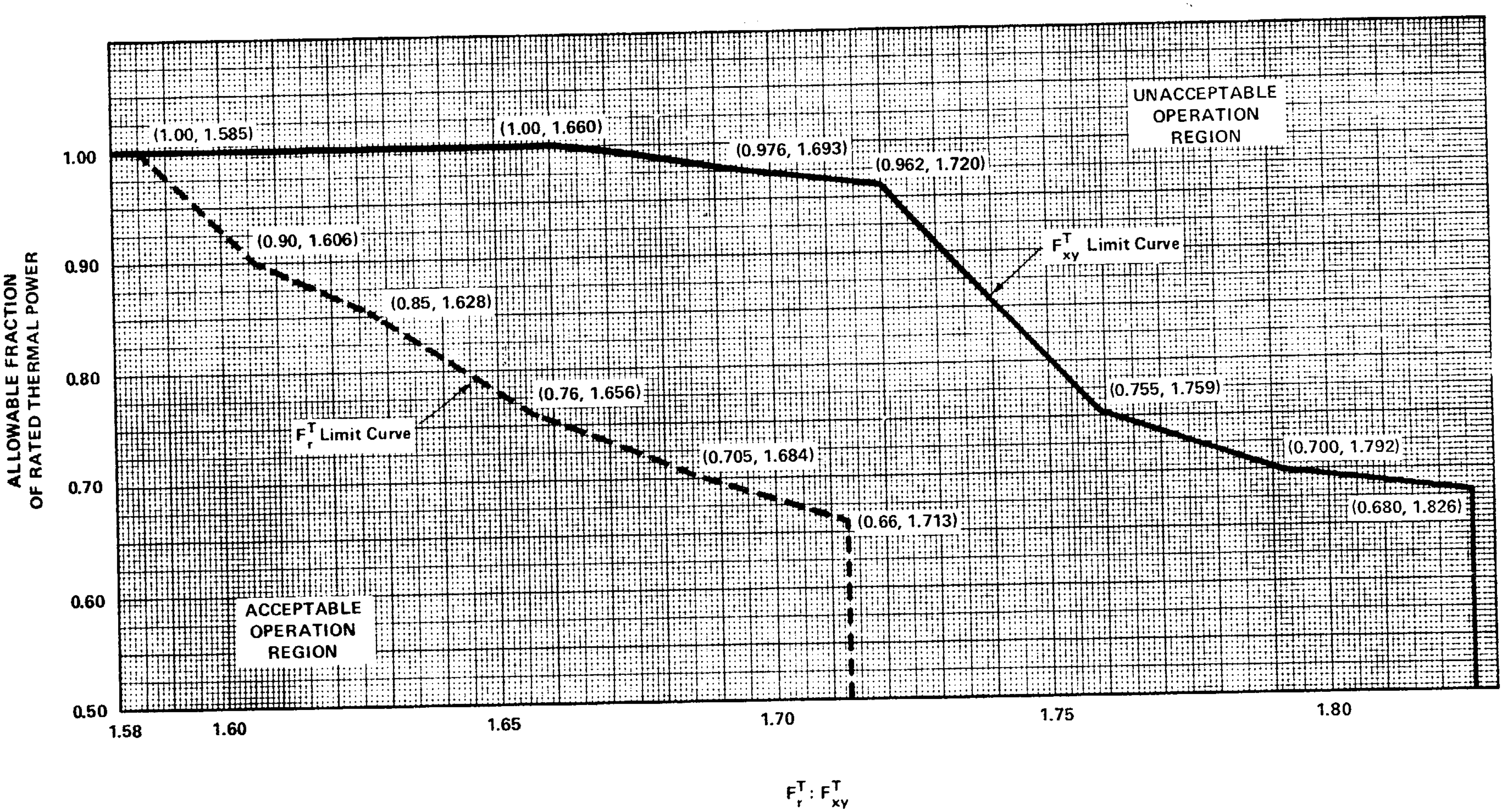

FIGURE 3.2-3
Total Radial Peaking Factors Versus Allowable Fraction of RATED THERMAL POWER

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 $\leq$ 1.571.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.571, 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.

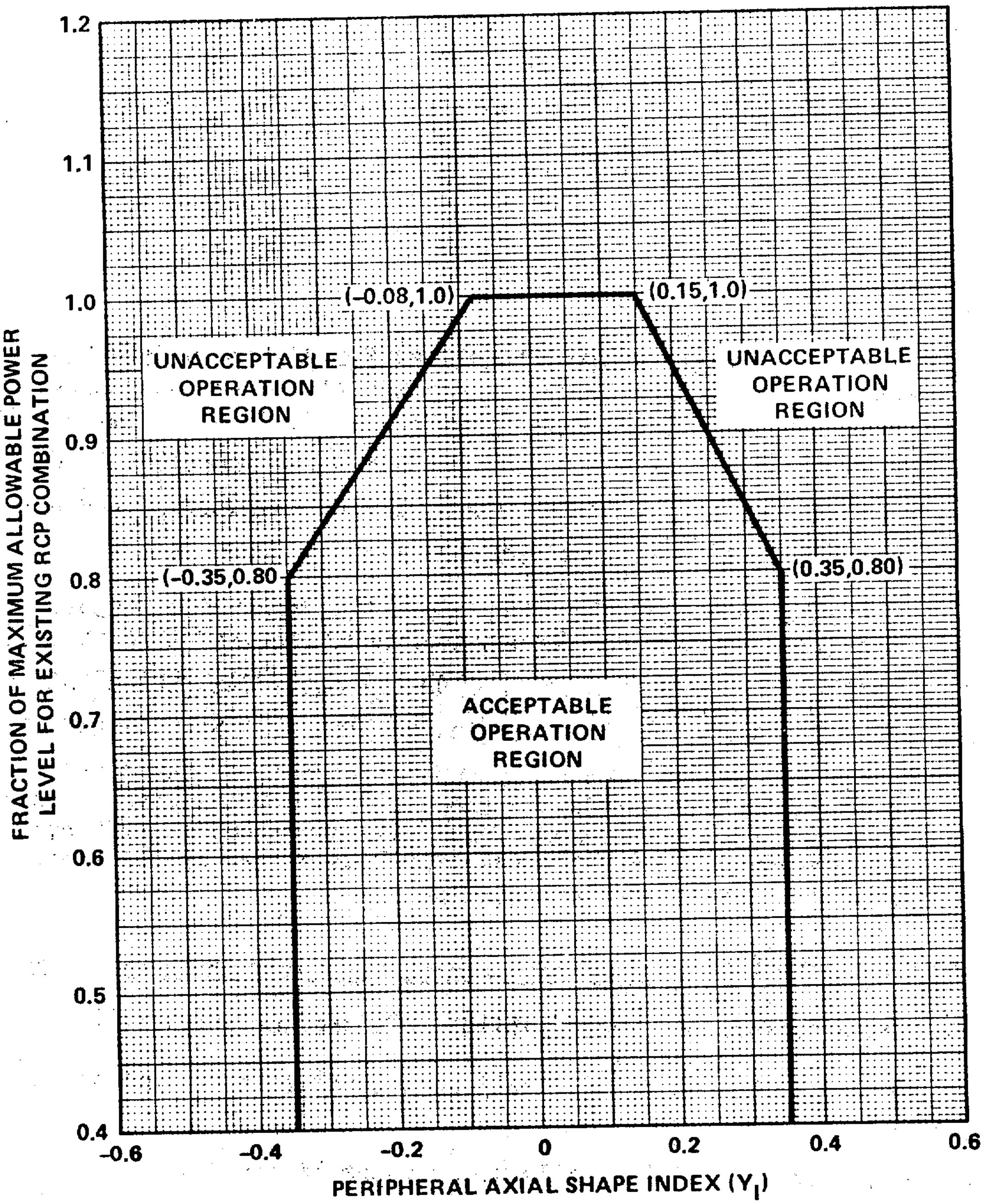

FIGURE 3.2-4
DNB Axial Flux Offset Control Limits

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.

POWER DISTRIBUTION LIMITS

FUEL RESIDENCE TIME

LIMITING CONDITION FOR OPERATION

3.2.5 This specification deleted.

SURVEILLANCE REQUIREMENTS

4.2.5 This specification deleted.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.2-1

DNB PARAMETERS

		LIMITS		
Parameter	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	$\leq$ 548°F	**	**	**
Pressurizer Pressure	$\geq$ 2225 psia*	**	**	**
Reactor Coolant System Total Flow Rate	$\geq$ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	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 $\leq$ 8.0 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 $\leq$ 8.0 seconds##
9. Thermal Margin/Low Pressure	$\leq$ 0.90 seconds*# and $\leq$ 8.0 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 1.070, 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^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, 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

This specification deleted.

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. 39 TO FACILITY OPERATING LICENSE NO. DPR-53

BALTIMORE GAS & ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 Introduction

By application dated February 23, 1979 and supplemental information dated January 12, February 7, March 5 and 13, May 7, 29 and 31, 1979, Baltimore Gas & Electric Company (BG&E or the licensee) requested an amendment to Facility Operating License No. DPR-53 for the Calvert Cliffs Nuclear Power Plant, Unit No. 1 (CCNPP-1). The amendment request consisted of:

- Technical Specification (TS) changes resulting from the analyses of Cycle 4 reload fuel;
- Approval to install a high burnup demonstration fuel assembly (SCOUT) and a prototype CEA; and
- Approval to operate another cycle with modified (sleeved and reduced flow) Control Element Assembly (CEA) guide tubes.

The associated specified TS changes are described in Section 4.0 of this Safety Evaluation (SE).

2.0 Background

In the Cycle 4 reload application for CCNPP-1 (Ref. 6), BG&E proposed to replace 40 Batch A and 32 Batch C fuel assemblies with 72 fresh Batch F fuel assemblies. The core related evaluations are presented in Sections 3.1 and 3.2 of this SE.

In December 1977, a severe CEA guide tube wear problem was identified at the Millstone Nuclear Power Station, Unit No. 2. Similar wear was subsequently found at CCNPP-1 and other facilities designed by Combustion Engineering (CE). The temporary repair for CCNPP-1 to allow Cycle 3 operation was to sleeve all fuel assemblies to be placed in CEA locations and the sleeving of other worn fuel assemblies in non-CEA locations to regain safety margins. Authorization for CCNPP-1 to operate for Cycle 3 in this mode was granted by Reference 1. As a result of the test program to evaluate the acceptability of the sleeves for a second cycle of operation, BG&E and CE found that some of the sleeves have become loose in the guide tubes (Ref. 14). The evaluation of the proposed repair and the entire CEA guide tube wear problem is presented in Section 3.3 of this SE.

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In the process of this review, we have requested and received additional information necessary for our evaluation (Refs. 10, 11).

CCNPP-1 is currently licensed to operate at 2700 MWt. The rated power level and all operating conditions remain the same for Cycle 4.

3.0 Evaluation

In this evaluation of a cycle reload for CCNPP-1, considerable use is made of generic reviews of various topical reports (See Topical References). Most of the topical reports have received formal NRC staff approval. In all cases where a topical report has not received approval, the report has been examined, its methods judged to be reasonable, and an appraisal has been made that a complete review will not reveal the methodology to be significantly in error. On this basis, all topicals referenced are judged to be acceptable for this reload evaluation.

3.1 Cycle 4 Fuel Design

The 217 fuel assembly Cycle 4 core will consist of:

BATCH IDENTIFICATION	WEIGHT & (w/o) ENRICHMENT	NUMBER OF FUEL ASSEMBLIES
B	#	1
D	#	48
D/	#	24
E	#	48
E/	#	24
F	3.03	48
F/	2.73	24

#Irradiated fuel from Cycle 3

As a result of the CEA guide tube wear problem, all fuel assemblies presently in Cycle 3 that will be placed in CEA locations in Cycle 4, with the exception of the Batch B test assembly and one other assembly, will have stainless steel sleeves installed in the CEA guide tubes in order to prevent guide tube wear. The Batch B test assembly was inspected during the current refueling outage and guide tube wear was found to be acceptable for another cycle of operation. The center core position occupied by the Batch B assembly is typically a low wear location for fuel assemblies. The other unsleeved fuel assembly in a CEA position is the result of a three way swap due to a problem sleeved fuel assembly as described in Reference 14. We find operation with two fuel assemblies unsleeved in CEA positions acceptable.

Of the new Cycle 4 fuel, eight Batch F assemblies and eight Batch F/ assemblies will be placed under dual CEAs and eight Batch F/ assemblies will be placed under single CEAs. These 24 new assemblies will have stainless steel sleeves installed in their CEA guide tubes.

BG&E has used the Cycle 3 reload analysis for CCNPP-1 as a "reference cycle" for the Cycle 4 reload analysis. Our original evaluation of Cycle 3 operation is presented in Reference 1. A reevaluation of Cycle 3 operation was necessary as a result of the reanalysis performed by BG&E in order to reach the licensed power level (Ref. 2). Analyses outside the envelope of the reference cycle have been reanalyzed.

3.1.1 Mechanical

In addition to the sleeving of fuel assemblies as described above and evaluated in Section 3.3 of this SE, the following other changes have been made to the mechanical design of the new fuel assemblies.

Upper End Fitting Assembly - The holddown plate in the upper end fitting has been thickened slightly. Since this reduces the holddown spring working length, the free length of the springs has been reduced by the same amount. Therefore, the holddown force has remained constant.

Lower End Fitting - The cross-bracing which connects the lower end fitting posts has been thickened and raised 1/8" from the lowermost surface of the fuel assembly.

Guide Tube Flow Holes - 16 Batch F assemblies have guide tube flow holes identical in size to the Batch E fuel. Another 16 assemblies have the reduced flow holes described in Reference 6. This modification is identical to that made to 16 fuel assemblies installed in the present cycle at CCNPP-2 and evaluated in Reference 3. The remaining forty fuel assemblies were modified to have slightly less flow than the normal Batch E fuel assemblies.

The effect of the modified cooling flow through the CEA guide tubes on the thermal hydraulics of the core will be evaluated in Section 3.1.3 of this safety evaluation.

An analytical prediction of the time of cladding creep collapse for all Cycle 2 fuel has been performed by CE using the CEPAN code which has been reviewed and approved by NRC. From this analysis, it has been concluded by CE that the collapse resistance of all the fuel rods is sufficient to preclude cladding collapse during its design lifetime. The design lifetime of this fuel will not be exceeded during Cycle 4 operation. The Batch B fuel which is the most limiting with regard to clad collapse will have accumulated 35,400 Effective Full Power Hours (EFPH) by the end of cycle (EOC). This is below the predicted time to clad collapse which has been calculated to be greater than 38,500 EFPH for any standard fuel rod in this assembly. We have reviewed this analysis and found it to be acceptable.

This cycle will also contain an additional change. This is the installation of a new fuel assembly called Scout which is a high burnup demonstration assembly that will provide information that will be useful in formulating a technical basis for the design, licensing and operation of fuel at high burnups for use in an extended fuel cycle.

The Scout high burnup demonstration assembly consists of 161 standard fuel rods and 15 demonstration rods. The mechanical design of the assembly components other than the 15 demonstration rods in this assembly is identical to the design of the other new fuel assemblies being loaded into the core. The 15 demonstration fuel pins are of two different mechanical designs. In one design, which is representative of six fuel pins, the spacer grid contacts the fuel pins at non-fueled regions. This could result in reduced grid/pin contact forces. To offset this possibility, the initial fill pressure in these rods was increased to decrease the magnitude of clad creepdown. A larger void volume exists in the rods with the greater initial pressurization which will result in no appreciable increase in the end of life internal pressure. CE has performed analytical predictions of the cladding creep collapse time for the demonstration fuel rods and has concluded that the collapse resistance of the demonstration fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 4 duration.

3.1.2 Nuclear Analyses Methodology

The Nuclear Design Model used in previous cycles has been PDQ, a two-dimensional diffusion code using four energy groups. PDQ has been accepted industry wide. For Cycle 4, CE performed the calculations of certain parameters using the ROCS code instead of PDQ. Using a higher order differencing methodology than PDQ and only one and a half energy groups, ROCS is able to compute many parameters nearly as accurately as PDQ in three dimensions with more reasonable computer run time.

For Cycle 4, the following safety parameters were computed using the ROCS code:

- Fuel Temperature Coefficients
- Moderator Temperature Coefficients
- Inverse Boron Worths
- Critical Boron Concentrations
- CEA drop distortion factors and reactivity worths
- Reactivity Scram Worths and Allowances
- Reactivity worth of regulating CEA banks
- Changes in 3-D core power distributions that result from inlet temperatures maldistributions (asymmetric steam generator transient)

None of these parameters require the detailed knowledge of pin powers normally computed by PDQ. BG&E states that in most cases, their parameters are calculated more accurately by ROCS because of its ability to account for three dimensional effects. BG&E has also stated that they observe guidelines to evaluate the adequacy of ROCS for computing these parameters on a case by case basis. If ROCS is judged to be not adequate for certain computation, then the computation is repeated using PDQ.

Based on our review, we find the use of ROCS to be acceptable for this reload.

3.1.3 Nuclear Parameters

In the Reference 1 SE, we found that introducing of stainless steel sleeves into the CEA guide tube had minimal effect on reactor physics. The operation of the CCNPP-1 for one cycle with all CEA guide tubes sleeved has borne out this conclusion.

In the SE supporting the Cycle 2 reload for CCNPP-2 (Ref. 3), we approved a demonstration test consisting of 16 fuel assemblies with reduced CEA guide tube flow. BG&E has also proposed a 16 fuel assembly demonstration test for Unit 1 Cycle 4. They anticipate no substantial change in axial and radial power distribution as a result of the decreased flow in the modified CEA guide tubes. This demonstration test will be discussed in Section 3.3 of this SE.

The licensee has stated that 40 Batch F assemblies have a flow hole configuration that presents a greater flow area and a consequent increase in guide tube flow over the standard Batch E assemblies. Since the flow area is greater than the standard assemblies by only 4%, the licensee has judged this to have an insignificant effect on axial and radial power distributions.

The Batch F reload fuel is comprised of two sets of assemblies with two enrichments as previously described in Section 3.1 of this safety evaluation. Cycle 4 burnup is expected to be between 10,000 Megawatt Days per Metric Ton Uranium (MWD/MTU) and 10,555 MWD/MTU. The licensee has examined the Cycle 4 performance characteristics for a Cycle 3 termination point of between 8950 and 10,000 MWD/MTU. The actual Cycle 3 burnup, as stated by the licensee, was 9465 MWD/MTU.

The Cycle 4 moderator temperature coefficient is calculated to be $-0.4\times10^{-4}\Delta\rho/°F$ at the EOC. The values for MTC are bounded by the values used in the reference cycle which are $-0.4\times10^{-4}\Delta\rho/°F$ at beginning of cycle (BOC) and $-2.1\times10^{-4}\Delta\rho/°F$ at EOC. We find these values of MTC to be acceptable.

Doppler coefficients calculated for Cycle 4 are $-1.50\times10^{-5}\Delta\rho/°F$ at BOC hot zero power (HZP), $-1.20\times10^{-5}\Delta\rho/°F$ at BOC hot full power (HFP) and $-1.37\times10^{-5}\Delta\rho/°F$ at EOC HFP. These values are slightly more negative at HFP for both BOC and EOC conditions. Changes of this magnitude, 5% more negative at HFP BOC and 10% more negative at HFP EOC have a minimal impact on the analysis of postulated Anticipated Operational Occurrences (AOOs) and accidents that result in a reactor cooldown. The slightly more negative values of the Doppler coefficient act to add additional conservatism to AOOs and accidents during which fuel temperature is tending to increase. We find the values of the Doppler coefficient calculated for Cycle 4 to be acceptable.

The total delayed neutron fraction for Cycle 4 has decreased slightly at EOC and increased slightly at BOC from that in the reference cycle. This would have a minor impact on the CEA ejection accident. The CEA ejection accident has been reanalyzed and is discussed in Section 3.5 of this safety evaluation.

At EOC 4, the reactivity worth of all CEAs inserted, less the highest worth CEA stuck allowance, is 7.7%$\Delta\rho$. The reactivity worth required to shut down the plant including power defect HFP to HZP, shutdown margin and safeguards allowance required to control the steam line break incident at EOC 4 is 6.2%$\Delta\rho$. The margin available in negative reactivity is 1.5%$\Delta\rho$ which is more than adequate to account for any uncertainty in nuclear calculations. We find these shutdown margins to be acceptable.

3.1.4 Thermal Hydraulics

The licensee states that the steady state Departure from Nucleate Boiling Ratio (DNBR) analyses of Cycle 4 at the rated power of 2700 MWT/MWt has been performed using the TORC code which employs the CE-1 DNBR correlation. The TORC code has been approved by Reference h for use in licensing and the CE-1 correlation has been approved with a 1.19 DNBR limit. TORC/CE-1 was also used in the generation of limiting conditions for operation (LCOs) on DNBR margin in the TS and all AOOs and postulated accidents which were reanalyzed for Cycle 4.

The fuel rod bowing effects on DNB margin for CCNPP-1 have been evaluated within the guidelines set forth in Reference g, as approved in the reference cycle SE (Ref. 1).

A total of 81 fuel assemblies will exceed the NRC-specified DNB penalty threshold burnup of 24,000 MWD/MTU, as established in Reference g, during Cycle 4. At the end of Cycle 4, the maximum burnup attained by any of these assemblies will be 42,800 MWD/MTU. From Reference g, the corresponding DNBR penalty for 42,800 MWD/MTU is 6.30 percent.

An examination of power distributions for Cycle 4 shows that the maximum radial peak at hot full power in any of the assemblies that eventually exceed 24,000 MWD/MTU is at least 10.30% less than the maximum radial peak in the entire core. Since the percent increase in DNBR has been confirmed to be never less than the percent decrease in radial peak, there exists at least 10.30% DNBR margin for assemblies exceeding 24,000 MWD/MTU relative to the DNBR limits established by other assemblies in the core. This margin is considerably greater than the Reference f reduction penalty of 6.30% imposed upon fuel assemblies exceeding 24,000 MWD/MTU in Cycle 4. Therefore, no power penalty for fuel rod bowing is required in Cycle 4.

The modifications to the fuel assemblies to alleviate the CEA guide tube wear problem have a small effect on their thermal hydraulic performance. As identified previously in this SE, Cycle 4 will have essentially two different modifications: 1) guide tube sleeving and 2) reduction in guide tube flow.

The flow characteristics of the assemblies with four 0.25" diameter hole and one 0.125" diameter hole and the assemblies with four 0.25" diameter holes and three 0.093" diameter holes are essentially equivalent.

The guide tube sleeving affects thermal hydraulic performance in three areas: core bypass flow, boiling in the guide tube sleeve annulus, and CEA cooling. As stated by the licensee, sleeving reduces the guide tube flow from 1400 lbm/hr to 700 lbm/hr. This change, however, compared to total core bypass flow is a minor effect which is in the conservative direction; i.e., it tends to increase the flow slightly through the core. Bypass flow must be maintained below 3.7% to preserve the design thermal margin. Sleeving improves this margin.

The second area of consideration is the potential for boiling in the guide tube-sleeve annulus. The licensee states that no boiling will occur in the region in which the sleeve is expanded into contact with the guide tube since the CEA linear heat rate of 3.68 KW/ft is below the boiling limit of 6.5 KW/ft. In the non-expanded region, axial peaks can be maintained such that CEA linear heat rates are below the 1.2 KW/ft boiling limit. Therefore, boiling is unlikely in this region. If boiling does occur, slots and holes in the sleeve assure that any expansion due to boiling is relieved and no mechanical damage will be caused. It is our opinion that limited boiling in this region is acceptable.

The criteria for adequate CEA cooling is that there is no bulk boiling in the guide tube during operation. The licensee states that cooling flow of 388 lbm/hr is required to meet this criteria. The cooling flow of 700 lbm/hr exceeds the minimum by a substantial margin. We find this to be acceptable.

The 16 fuel assemblies will have reduced guide tube cooling flow due to the reduction in number and size of the flow holes. The CEA cooling flow for this design has been stated by the licensee to be 565 lbm/hr. This exceeds the bulk boiling criteria of 388 lbm/hr and has a minimal impact in the conservative direction on total core bypass flow. However, for Cycle 4 none of these 16 assemblies will be in CEA locations.

The licensee has stated that the maximum peaking factor in any fuel rod in the Scout high burnup demonstration bundle is predicted to be more than 12% below the limiting pin peak in the core and the maximum pin peaking factor in any demonstration rod is predicted to be more than

15% below the limiting pin peak in the core. Considering that the bundle geometry of the Scout assembly is identical to the other Batch F assemblies and the Scout assembly power is well below the limiting core bundle the thermal hydraulic design of this assembly is acceptable.

3.2 Uncertainty in Nuclear Power Peaking Factors

In-core detector measurements are used to compute the core peaking factors using the INCA Code (Ref. c). The coefficients required to perform this data reduction are performed using the methodology described in the topical report.

For Cycle 4 operation, the licensee has proposed measurement uncertainties of 6% for the total integrated radial peaking factor (F_r) and 7% for the total power peaking factor (F_q) for base load operation and 8.0% and 10.0% for load follow operation.

The initial CE evaluation of peaking factor uncertainty was presented in References c and d. In a meeting with CE on March 6, 1979, data was presented showing measurement uncertainty of 6% in Fr and 7% in Fq to be conservative (Ref. 8). On this basis, we find these measurement uncertainties of 6% and 7% for Fr and Fq, respectively, to be acceptable without the load follow operation restrictions.

3.3 CEA Guide Tube Integrity

BG&E instituted an Eddy Current Testing (ECT) inspection program at CCNPP-1 to ascertain the condition of sleeves in assemblies located under CEA's during Cycle 3 (Ref. 4). No indications of sleeve wear were found in these assemblies, however several guide tube sleeves, when subjected to pull tests, did not exhibit the expected resistance to axial motion (Ref. 14). Because the CCNPP-1 wear inspection program showed ECT signals with widely varying magnitudes at the crimped regions of the sleeves, the inspection program was extended to assess the crimp size in a number of different type fuel assemblies. This inspection for crimp integrity was performed using the same probe and test procedure used in the wear inspection program.

The results of these inspections revealed a large number of sleeved fuel assemblies outside the ECT and pull test acceptance criteria used at other CE designed facilities. The explanations of CCNPP-1 results in comparison with the results from the other CE facilities were that the sleeving sequence used at other facilities in 1978 differed from that used at CCNPP-1 (the first facility where sleeving was performed). At the other facilities, pull tests were performed on the sleeves after the crimping step to verify the adequacy of the crimp. Following the "crimp verification" pull test, expanding steps were then performed on the sleeves. However, at CCNPP-1, the pull tests were not performed until after both the crimping and the expanding steps were completed. The licensees and CE have concluded that this sequence change added frictional resistance between the expanded sleeve and the guide tube wall to mask the presence of inadequate crimps that would have been identified by an intermediate "crimp verification" pull test.

In addition, the low ECT results at CCNPP-1, which indicate inadequate crimps, were unique to a particular fuel category. This fuel category consists of those assemblies that had been irradiated prior to sleeving in 1978. In this fuel category at CCNPP-1, the EC signals were low for approximately 50% of the 235 sleeves tested. The low signals for irradiated fuel were not evident at the other facilities. Thus, it appears that the increased yield strength of irradiated guide tubes reduced the displacement of the crimp.

To remedy the observed inadequacy of the crimps at CCNPP-1, a total of 28 assemblies were designated for recrimping, using the new style crimp over the previously made old style crimp. ECT was performed on each sleeve after recrimping to measure actual crimp size. The basis of selecting the 28 fuel assemblies was that these assemblies were in the category of those assemblies sleeved in 1978 in the irradiated condition and are to be under CEAs for Cycle 4 operation. Because the recrimp is positioned at some distance from the bottom of the sleeve, a second operation, in which the bottom is re-expanded against the guide tube wall, was also performed. This operation, together with a free path gauge check was used to insure that the end of the sleeve would not interfere with CEA insertion.

The licensee stated that bench tests were completed on sample guide tube and sleeves to determine effects on sleeve and guide tube geometry by installing a second crimp over a previously installed crimp. Results of these test samples showed that the new style crimp can be installed over the old style crimp without "rolling in" the end of the sleeve, or causing any other anomalies in geometry. The tests also indicated no need for an additional lower end expansion; however, this procedure was retained in field crimping operations to preclude any chance of sleeve edge protrusion. For the actual recrimps placed in the fuel assemblies in question, all sleeves have been ECT and shown to have crimp sizes sufficient to prevent axial motion (Ref. 14).

All other crimping and sleeving operations for this outage have used the new style crimping tools. The higher crimp pressure inherent with the new style crimp provides a greater force to locally deform (crimp) the higher strength irradiated guide tubes and likewise provides a more defined crimp geometry to resist axial motion of the sleeves.

We have reviewed the proposed crimping, and recrimping of the CEA guide tubes. and the results of the surveillance tests at CCNPP-1. Based on the information provided in Reference 14, we agree that the guide tube sleeving operations at CCNPP-1 provide acceptable repairs to the guide tubes for Cycle 4 operation,

In Reference 14, BG&E stated that CE recommended operational guidelines to reduce relaxation effects in the guide tube sleeves during Cycle 4 operation. This recommended guideline is to restrict movement of the CEAs at systems temperatures below 400 F except for normal movement associated with refueling operations. We find the recommended operational guideline reasonable. BG&E has agreed to implement this restriction on CEA movement.

Sixteen Batch F fuel assemblies have been modified by decreasing the number and size of the flow holes and the size of the bleed holes. Tests have indicated that the resulting decrease in guide tube flow was accompanied by less CEA flow-induced vibration and, therefore, less guide tube wear. The SE for CCNPP-2, Reference 3, found the demonstration test similar to that proposed for CCNPP-1 with 16 fuel assemblies to be acceptable. The increase in the CEA insertion time to 3.1 seconds was also found acceptable. We, therefore, conclude that the demonstration test of 16 modified fuel assemblies with reduced guide tube flow is acceptable for Cycle 4 operation of CCNPP-1.

BG&E has agreed to provide a Cycle 5 guide tube evaluation program, identifying changes from the Cycle 4 program at least 90 days prior to the CCNPP-1 shutdown for the Cycle 5 reload outage.

3.4 Analyses of Anticipated Operational Occurrences (AOOs)

Reference 5 discusses the safety analyses of postulated AOOs for CCNPP-1 Cycle 4. The licensee classifies the list of postulated AOOs into two categories. The first category includes those AOOs for which the Reactor Protection System (RPS) Limiting Safety System Settings (LSSS) as specified in the plant TS assure that the Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded. The second category includes those AOOs for which initial steady state overpower margins are maintained by adherence to the Limiting Conditions for Operation (LCOs) specified by the TS for the plant. Adherence to the LCOs assure that SAFDL limits are not exceeded.

The loss of flow transient causes the most rapid change in DNBR and both a reactor trip and steady-state overpower margin is required to maintain the SAFDLs. The LCOs and LSSSs for Cycle 4 TS were calculated using the methods described in Reference f. The required AOO reanalyses were done using the computer code CESEC (Ref. i).

The licensee stated in Reference 5 that the need for reanalysis of a particular AOO is determined by comparison of the key parameters for that AOO to those of the last cycle for which a complete analysis was performed. If the key parameters are within the envelope of the reference cycle data, no reanalysis is required. A reanalysis might also be performed in case it could lead to a significant relaxation of TS.

The results of that comparison show that the key parameters to all the AOOs and postulated accidents for Cycle 4 operation are the same as the specified reference cycle input parameters, except for the following:

1. CEA drop time to 90% inserted
2. Integrated radial peaking factor (F_r)
3. Seized rotor pin census
4. Core bypass flow fraction
5. RTD response time

For all AOOs and postulated accidents other than those reanalyzed, the licensee has stated that the CCNPP-1 safety analysis submitted either in the FSAR or in previous reload cycle license submittals bound the results that would be obtained for Cycle 4 and demonstrate continued safe operation of CCNPP-1 at 2700 MWt.

Since the CEA drop time to 90% insertion has increased for Cycle 4, the Loss of Flow Event, CEA Ejection Event, RCS Depressurization Event, Seized Rotor Event and the CEA Withdrawal Event were reanalyzed. These events are adversely impacted by the CEA drop time, since a reactor trip is necessary to terminate the event.

The sleeving of the CEA guide tubes has a negligible effect on CEA rod drop times but the reduction of the CEA guide tube flow holes does impact on the rod drop times. As previously stated, the Cycle 4 reload will have 16 fuel assemblies with reduced flow holes. The effect of these flow holes on rod drop times is to increase the time to 90% insertion from 2.5 to 3.1 seconds. BG&E has identified this as a proposed change to the TS 3.1.3.4 at this time, even though none of these assemblies are under CEAs during this cycle. To assess the impact of this change in rod drop time, the licensee has examined all the design basis events which could require a trip to prevent exceeding SAFDL limits. An evaluation of these design basis events showed that only five events may be adversely affected by increased scram time. For these evaluations, it was conservatively assumed that all the CEAs are inserted at the same insertion versus time characteristic curve as in the 16 fuel assemblies with the reduced guide tube flow. Those transients which were reanalyzed are discussed below.

BG&E has proposed a change to the TS Table 2.2-1 raising the high power level trip from 106.5% to 107.0% power. The safety analysis assumes a trip at 112% of rated power. A 5% power measurement uncertainty has always been applied in the process of generation LSSS limits. In the past, this uncertainty was applied in a multiplicative fashion (which yields the equivalent of a 5.5% of rated power uncertainty), but evaluations showed that application of the uncertainty in this fashion is conservative. In accordance with current methods (as described in Reference f), the power measurement uncertainty is now deducted algebraically. It is this difference in the manner in which the uncertainty is applied that leads to the 107% versus 106.5% LSSS limit. We have reviewed this change and find it to be acceptable.

3.4.1 CEA Withdrawal Event

The CEA Withdrawal event was reanalyzed for Cycle 4 due to the increase in the Resistance Temperature Detector (RTD) response time to envelope future cycles and the increase in the CEA drop time to 90% insertion from 2.5 seconds to 3.1 seconds. The CEA Withdrawal event was reanalyzed for reactor initial conditions of zero power and full power and the licensee has stated that the Departure from Nucleate Boiling (DNB) and fuel centerline melt Specified Acceptable Fuel Design Limits (SAFDLs) will not be exceeded during CEA Withdrawal transient.

The CEA Withdrawal transient initiated at rated thermal power results in the maximum pressure bias factor of 62.0 psia. This bias factor accounts for measurement system processing delays during the CEA Withdrawal event. The pressure bias factor for this cycle has increased from the reference cycle due to the increase in the RTD time constant and the increase in the CEA drop time to 90% insertion. This pressure bias factor is used in generating TM/LP trip setpoints to prevent the SAFDLs from being exceeded during a CEA Withdrawal Event. The TS have been changed to reflect the 62.0 psia pressure bias factor. We find this analysis and the change to the plant TS to be acceptable.

3.4.2 RCS Depressurization Event

The RCS Depressurization event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion from 2.5 seconds for Cycle 3 to 3.1 seconds for Cycle 4. As stated in Reference f, this is one of the events analyzed to determine a bias term input to the TM/LP trip. Hence, this event was analyzed for Cycle 4 to obtain a pressure bias factor. This bias factor accounts for measurement system processing delays during this event. The trip setpoints incorporating a bias factor at least this large will provide adequate protection to prevent the DNBR SAFDL from being exceeded during this event.

The analysis of this event shows that the pressure bias factor is 35 psia which is less than that required by the CEA Withdrawal Event. Hence, the use of the pressure bias factor determined by the CEA Withdrawal event will prevent exceeding the SAFDLs during an RCS Depressurization event.

3.4.3 Loss of Coolant Flow Event

The Loss of Coolant Flow event was reanalyzed for Cycle 4 to determine the impact on margin requirements that must be built into the LCOs due to the increase in the CEA drop time to 90% insertion.

The low flow trip setpoint is reached at 1.0 seconds and the CEAs start dropping into the core one second later. A minimum DNBR of 1.25 is reached at 2.3 seconds.

The low flow trip, in conjunction with the initial overpower margin maintained by the LCOs in the TS assure that the minimum DNBR will be greater than or equal to 1.19 for the Loss of Coolant Flow Event.

3.4.4 Conclusion

We have reviewed the licensee's analyses of AOOs for Cycle 4 operation of CCNPP-1 and conclude that they are acceptable.

3.5 Postulated Accidents Other Than LOCA

The licensee has reviewed the postulated accidents other than LOCA. Reference 5 discusses the safety analysis performed for this category of accident for CCNPP-1 Cycle 4. Postulated accidents as other plant events, need to be reanalyzed only if the key parameters influencing the event are not enveloped by the reference cycle data. Those accidents that were reanalyzed are discussed below.

3.5.1 CEA Ejection Event

The CEA Ejection Event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion and the increase in the augmentation factor in comparison to the reference cycle. In addition, the zero power case was analyzed due to the decrease in axial peak in comparison to the reference cycle. The reference cycle for this event is the analysis upon which the licensing of CCNPP-2 Cycle 2 was based. Our evaluation of this reload is found in Reference 3. Hence, this event was reanalyzed to demonstrate that the criterion for clad damage is not exceeded during Cycle 4 operation.

The licensee's analysis shows that for both the zero power and full power cases the clad damage pellet enthalpy threshold of 200 cal/gm is not violated. Therefore, no fuel rods are predicted to suffer clad damage.

3.5.2 Seized Rotor Event

The Seized Rotor event was reanalyzed for Cycle 4 due to the changes in the following key parameters.

- The increase in the CEA drop time to 90% insertion
- The decrease in core bypass flow, which increases the net core flow
- The decrease in the Radial Peaking Factor
- A more adverse (flatter) pin census.

The increase in the CEA drop time and the flatter pin census adversely impact the consequences of this event. Increasing the net core flow and decreasing the Radial Peaking Factor will decrease the consequences of this event. Hence, a reanalysis was performed for Cycle 4 to ensure that only a small fraction of fuel pins are predicted to fail during a Seized Rotor event.

A conservatively "flat" pin census distribution (a histogram of the number of pins with radial peaks in intervals of 0.1 in radial peak normalized to the maximum peak) was used to determine the number of pins that experience DNB.

The results indicate that increasing the core flow and decreasing the radial peaking factor offset the increase in the CEA drop time to 90% insertion. It was calculated that for Cycle 4, less than 0.5% of fuel pins will experience DNB for even a short period of time.

For the case of the loss of coolant flow arising from a seized rotor shaft, it is assumed that there is an instantaneous reduction to three pump flow. The low flow trip assures that less than 0.5% of fuel pins experience DNB. This is the same as that calculated for the reference cycle. Hence, the conclusions reached for reference cycle remain valid for Cycle 4.

3.5.3 Conclusions

We have reviewed the accident analyses for events other than LOCA for CCNPP-1 Cycle 4 and conclude that they are acceptable.

3.6 Cycle 4 LOCA Analysis

Reference 5 provides a comparison of the fuel specific parameters for the limiting fuels during Cycles 3 and 4.

The Cycle 4 core contains 216 high density fuel assemblies and one low density Batch B assembly. The highest power pin in the low density Batch B assembly will not achieve a power level greater than 75% of the highest power pin in the core. Therefore, a Batch B fuel pin will not be limiting in Cycle 4.

The remaining 216 high density fuel assemblies contain 72 partially depleted Batch D assemblies, 72 partially depleted Batch E assemblies and 72 fresh Batch F assemblies. Burnup dependent calculations were performed for the high density fuel assemblies with the FATES (Ref. b) and STRIKIN-II(Ref.a) codes. The results demonstrate that the most limiting fuel pin during Cycle 4 is located in one of the partially depleted Batch E assemblies.

The limiting high density fuel in Cycle 4 has a stored energy 268°F lower than the limiting fuel in Cycle 3. Consequently, the ECCS performance results reported for Cycle 3 conservatively bound the performance for Cycle 4. Therefore, the peak linear heat generation rate of 14.2 KW/ft which was demonstrated to be acceptable for Cycle 3 is also an acceptable limit for Cycle 4 operation.

In order to comply with 10 CFR 50, Appendix K, the LOCA analysis must demonstrate that the peak clad temperature (PCT) remains below 2,200 F and the maximum local cladding oxidation, which is a function of the time dependence of the PCT, remains below 17 percent.

During a LOCA, the cladding swells due to the decreased coolant pressure and the increased fuel temperature and gas pressure. The clad swelling is terminated if the cladding ruptures. The Rupture-Strain curve is a plot of clad strain (clad swelling) vs clad temperature at the point of clad rupture in a LOCA Event. The Rupture-Strain curve is an integral part of the CE ECCS flow blockage model. Recently the NRC staff has determined that, for clad rupture which occurs during the reflood phase of the LOCA, the Rupture-Strain curve used by CE is possibly nonconservative. However, this is not a problem for CCNPP-1,because clad rupture is predicted to occur during the blowdown phase and not the reflood phase. The staff review has found the CE analyses for the case of rupture during the blowdown phase to be acceptable.

We conclude, as a result of our review, that the CCNPP-1 Cycle 4 ECCS performance is in conformance with the criteria specified in 10 CFR 50.46(b) and is, therefore, acceptable.

4.0 Technical Specifications

The TS changes proposed for this amendment are summarized in the following statements.

Page 1-3

The definition of Shutdown Margin (Section 1.13) would be revised to eliminate the reference to part length CEAs.

Page 2-7

The Power Level-High RPS trip would be increased 0.5% to 107.0% as a result of the Cycle 4 analyses.

Pages 2-12 & 2-13

Figures 2.2-2 and 2.2-3, relating to the TM/LP trip setpoint, would be modified as a result of the Cycle 4 analyses.

Page 3/4 1-23

The CEA drop time, TS 3.1.3.4, would be increased from 2.5 seconds to 3.1 seconds as a result of the changed hydraulic characteristics of the 16 demonstration fuel assemblies.

Pages 3/4 2-4 & 3/4 2-5

New axial flux offset (Figure 3.2-2) and augmentation factors (Figure 4.2-1) would be added based on revised physics calculations.

Pages 3/4 2-8 & 3/4 2-9

These power distribution limit changes would be made based on revised physics calculations and application of the standard CE setpoint methodology.

Page 3/4 2-11

Figure 3.2-4 would include the increase in allowable azimuthal tilt.

Page 3/4 2-13

The old TS 3.2.5 would be eliminated since the core can not achieve a core exposure that would result in clad collapse.

Page 3/4 2-15

Table 3.2-1 would be revised to increase the cold leg temperature used in DNB calculations by 1 F to 548 F. Parameter values for less than four RCP operation would be eliminated pending NRC review of ECCS analyses for operation in that mode.

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Table 3.3-2 would be revised to increase the RTD response time from 5 to 8 seconds in accordance with the Cycle 4 analysis.

5.0 Physics Startup Testing

The physics startup test program as described in Reference 6 has been reviewed. The low power tests include CEA symmetry check, critical boron concentration measurements, isothermal temperature coefficient measurements and CEA group worth measurements. The power ascension tests include power coefficient and power distribution tests.

The staff discussed the CEA symmetry test and the review criteria for this test with the licensee. The licensee agreed to perform the CEA symmetry test on 2 shutdown banks and review criteria as stated in Reference 13. The review criteria for power distribution measurements are also given in Reference 13.

The staff finds the entire program including the acceptance and review criteria and the remedial actions acceptable.

6.0 Environmental Consideration

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.

7.0 Conclusion

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 14, 1979

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b. CENPD-139, "CE Fuel Evaluation Model", July 1974.

c. CENPD-145, "INCA: Method of Analyzing In-Core Detector Data in Power Reactors", April 1975.

d. CENPD-153, "Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems", August 1974.

e. CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

f. CENPD-199-P, "CE Setpoint Methodology", April 1976.

g. CENPD-225, "Fuel and Poison Rod Bowing", October 1976.

h. Evaluation of Topical Report CENPD-161-P, K. Kniel (NRC) to A. E. Scherer (CE), September 14, 1976.

i. CENPD-107, "CESEC-Digital Simulation of a CE Nuclear Steam Supply System," April 1974.

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3. NRC Amendment No. 18 for CCNPP-2, Cycle 2 Reload, R. W. Reid to A. E. Lundvall, October 21, 1978.
4. BG&E Sleeved CEA Guide Tube Inspection Program, A. E. Lundvall to R. W. Reid, January 12, 1979.
5. BG&E High Burnup Demonstration Program, A. E. Lundvall to R. W. Reid, February 7, 1979.
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10. NRC Request for Additional Information, R. W. Reid to A. E. Lundvall, April 13, 1979.
11. BG&E Reponse to Cycle 4 Reload Questions, A. E. Lundvall to R. W. Reid, May 7, 1979.
12. NRC Safety Evaluation - Small Break LOCA Analysis with No Credit for Charging Pump Flow, R. W. Reid to A. E. Lundvall, May 18, 1979.
13. BG&E Supplement 4 to Application for Cycle 4 Reload - Physics Startup Testing, A. E. Lundvall to R. W. Reid, May 29, 1979.
14. BG&E Supplement 5 to Application for Cycle 4 Reload - CEA Guide Tube Test Results and Repairs, A. E. Lundvall to R. W. Reid, May 31, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-317

BALTIMORE GAS & ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. DPR-53, issued to Baltimore Gas & Electric Company, 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 authorizes operation with modified guide tubes for the Control Element Assemblies with a high burnup demonstration fuel assembly installed in the core and revises the Technical Specifications to incorporate changes resulting from the analysis of Cycle 4 reload fuel.

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. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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.

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For further details with respect to this action, see (1) the application for amendment dated February 23, 1979 with supplemental information dated January 12, February 7, March 13 and May 7, 29 and 31, 1979, (2) Amendment No. 39 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 14th day of June 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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