Docket No. 50-318

OCT 19 1977

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

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. DPR-69 for Unit No. 2 of the Calvert Cliffs Nuclear Power Plant. The amendment is in response to your application dated July 13, 1977, and supplements thereto dated September 30, 1977 and October 5, 1977.

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

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

Sincerely,

/S/

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

Notified Rich Olson at BG&E at 1505 on 10/19. Notified Rich Conte (Region I) and Frank Nolan of I&E at 1510 on 10/19/77. EZConner

Enclosures:
1. Amendment No. 9 to License No. DPR-69
2. Safety Evaluation and Environmental Impact Appraisal
3. Notice/Negative Declaration

cc w/enclosures:
See next page

NOT NECESSARY SINCE NO CHANGES IN OT SER INPUT

OFFICE	DOR:AD/OT				
SURNAME	DEisenhut				
DATE	10/ /77				

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SURNAME	RMDiggs	MConner:esp		DKDavis	KRGoller KRG	VStello
DATE	10/18/77	10/13/77	10/17/77	10/18/77	10/19/77	10/19/77

10/19/77 ESP

Const. 1

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Power Company (the licensee) dated July 13, 1977, as supplemented by filings dated September 30 and October 5, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

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

   F. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and by amending Section 2.C to revise paragraphs 1 and 2 of Facility Operating License No. DPR-69 to read as follows:

   1. <u>Maximum Power Level</u>

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

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 9, 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

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 19, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 9

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

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	Pages
I	3/4 2-1
IV	3/4 2-2
IX	3/4 2-3
1-1	3/4 2-4
1-3	3/4 2-5
1-5	3/4 2-6
1-6	3/4 2-7
2-2	3/4 2-8
2-9	3/4 2-9
2-11	3/4 2-10
2-12	3/4 2-11
2-13	3/4 2-12
2-14	3/4 2-13
2-15	3/4 2-14 (added)
2-16 (added)	3/4 2-15 (added)
2-17 (added)	3/4 5-7
2-18 (added)	B 3/4 1-2
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1.0 DEFINITIONS

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1.0 DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

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

OPERATIONAL MODE

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

ACTION

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

OPERABLE - OPERABILITY

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

REPORTABLE OCCURRENCE

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

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

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

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

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

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

CHANNEL CALIBRATION

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

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:

a. 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, and

b. No change in part length control element assembly position.

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."

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad\qquad Y_I = AY_E + B$$

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.24 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.26 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.28 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

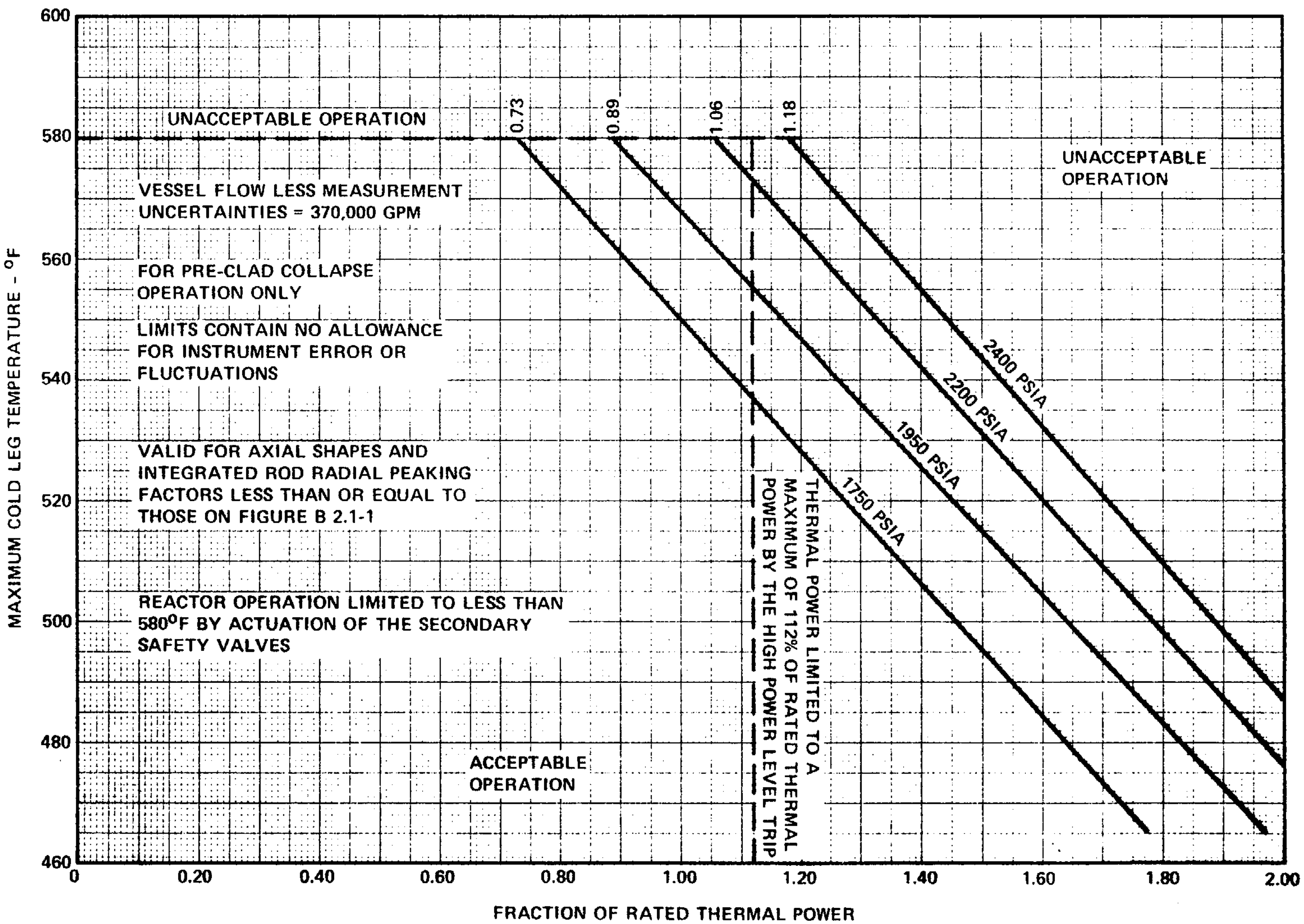

Figure 2.1 - 1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2400 psia
5. Containment Pressure - High	$\leq$ 4 psig	$\leq$ 4 psig
6. Steam Generator Pressure - Low (2)	$\geq$ 500 psia	$\geq$ 500 psia
7. Steam Generator Water Level - Low	$\geq$ 10 inches below top of feed ring.	$\geq$ 10 inches below top of feed ring.
8. Axial Flux Offset (3)	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.
9. Thermal Margin/Low Pressure (1)		
a. Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.
b. Three Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-4 and 2.2-5.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-4 and 2.2-5.
c. Two Reactor Coolant Pumps Operating - Same Loop	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-6 and 2.2-7.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-6 and 2.2-7.
d. Two Reactor Coolant Pumps Operating - Opposite Loops	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-8 and 2.2-9.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-8 and 2.2-9.
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 45 psig	$\geq$ 45 psig

TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
11. Rate of Change of Power - High (4)	$\leq$ 2.6 decades per minute	$\leq$ 2.6 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

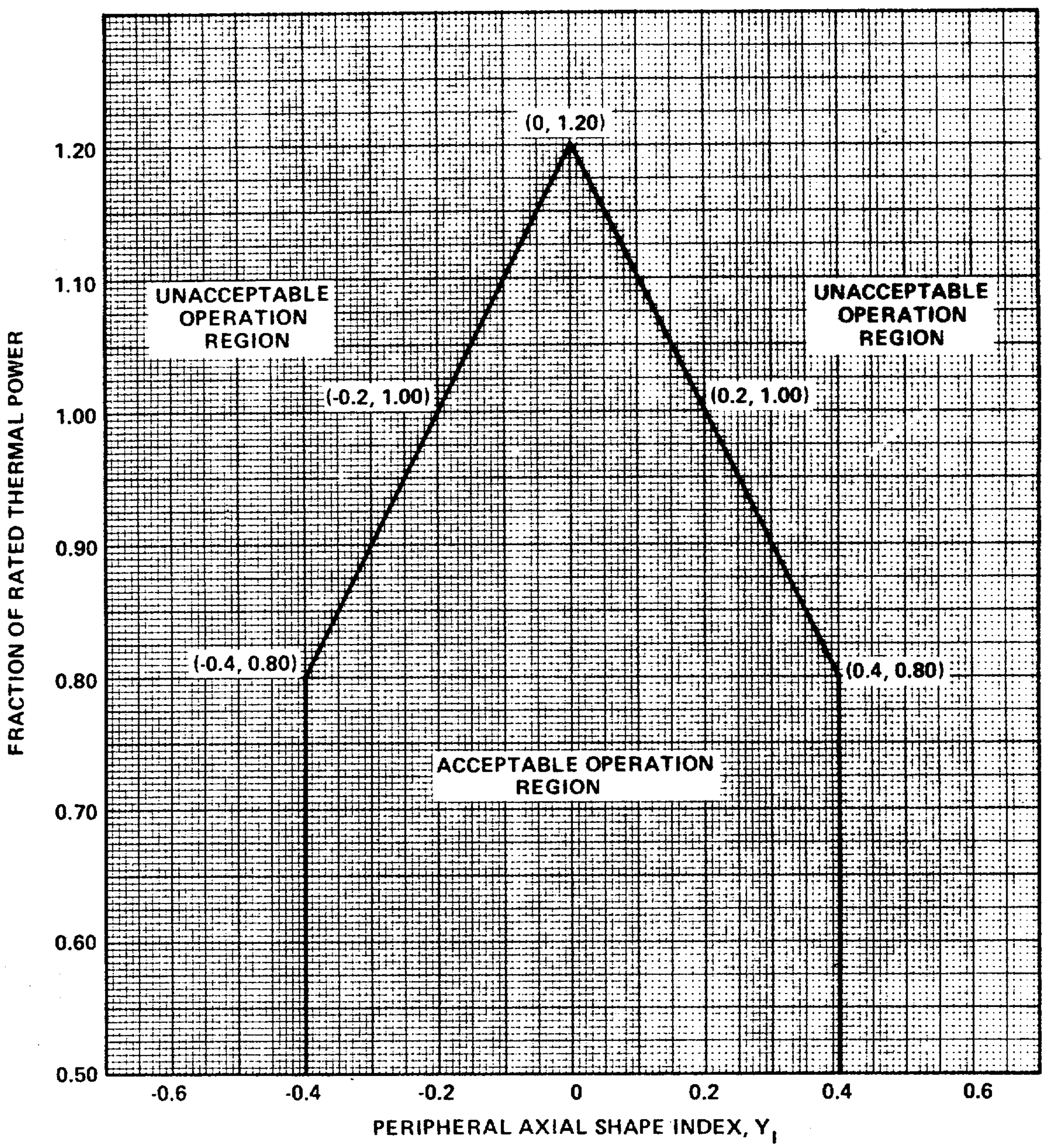

FIGURE 2.2-1

Peripheral Axial Shape Index, Y_I Versus Fraction of RATED THERMAL POWER

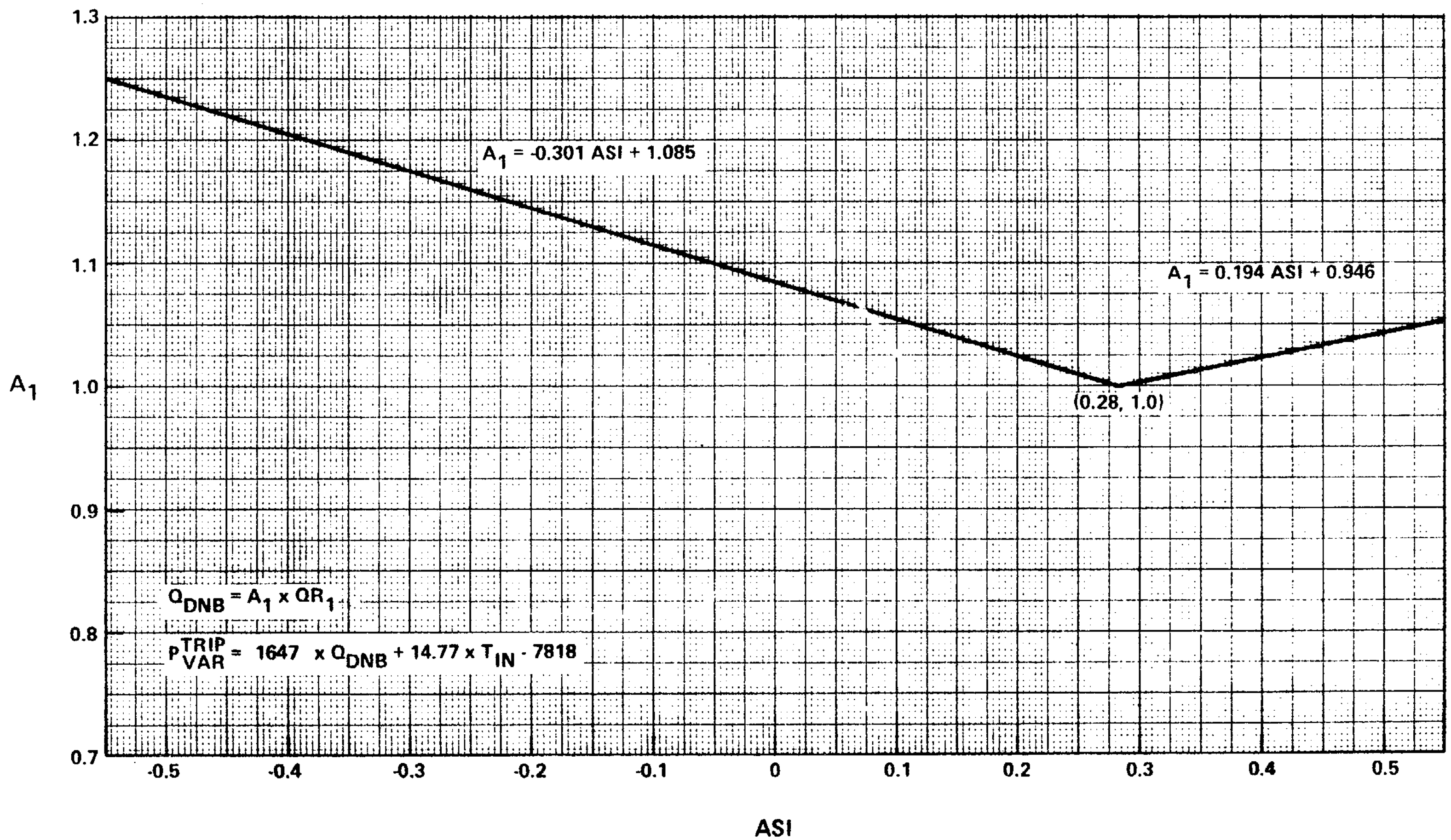

FIGURE 2.2-2

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

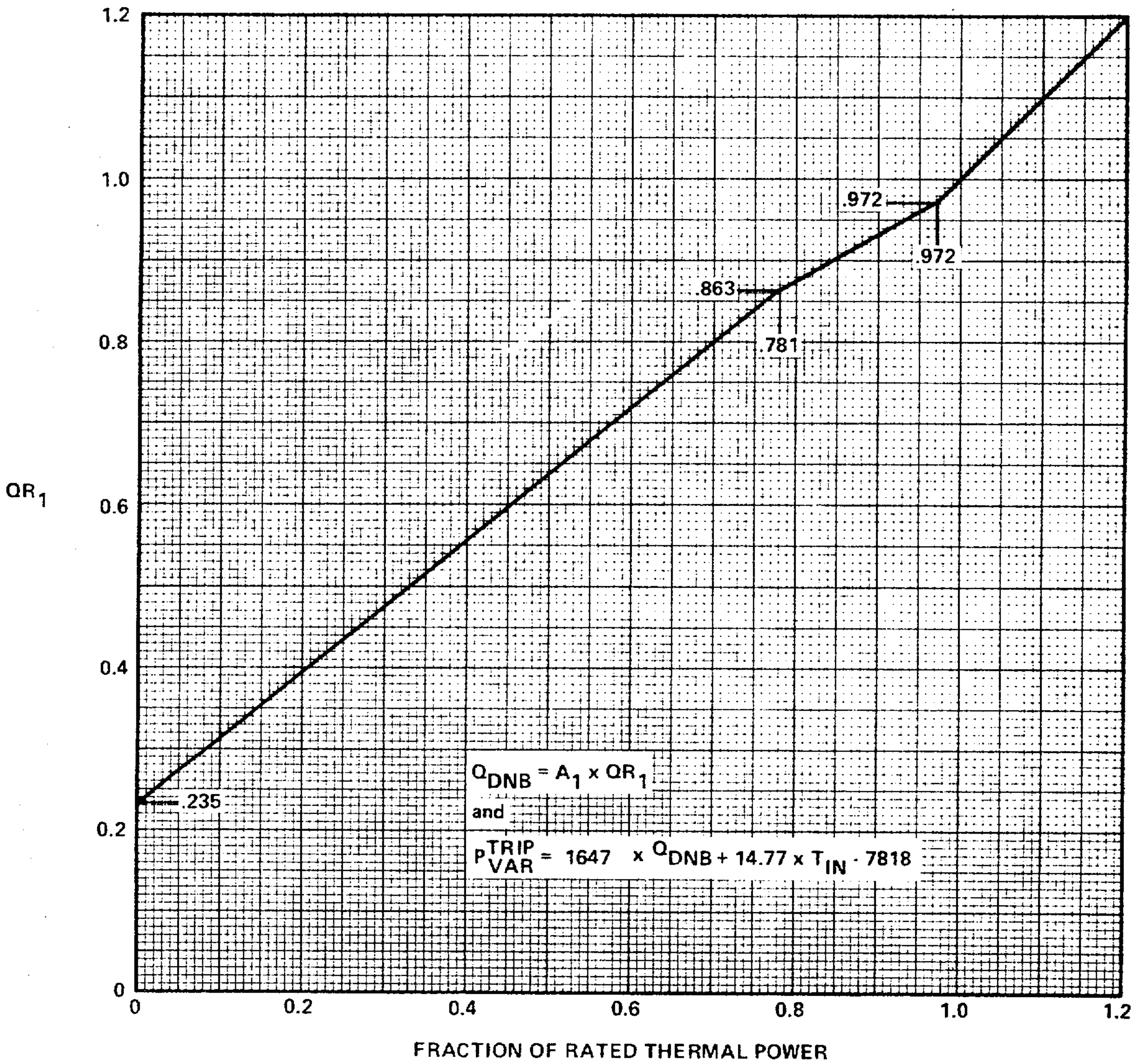

FIGURE 2.2-3

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

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Figure 2.2-4

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

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Figure 2.2-5

Thermal Margin/Low Pressure Trip Setpoint - Part 2
Three Reactor Coolant Pumps Operating

This page left blank pending NRC approval of ECCS analysis for two pumps (same loop) operation.

Figure 2.2-6

Thermal Margin/Low Pressure Trip Setpoint - Part 1
Two Reactor Coolant Pumps Operating - Same Loop

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Figure 2.2-7

Thermal Margin/Low Pressure Trip Setpoint - Part 2
Two Reactor Coolant Pumps Operating - Same Loop

This page left blank pending NRC approval of ECCS Analysis for two pumps (opposite loops) operation.

Figure 2.2-8

Thermal Margin/Low Pressure Trip Setpoint - Part 1
Two Reactor Coolant Pumps Operating - Opposite Loops

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Figure 2.2-9

Thermal Margin/Low Pressure Trip Setpoint - Part 2
Two Reactor Coolant Pumps Operating - Opposite Loops

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the minimum DNBR is no less than 1.30 for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in

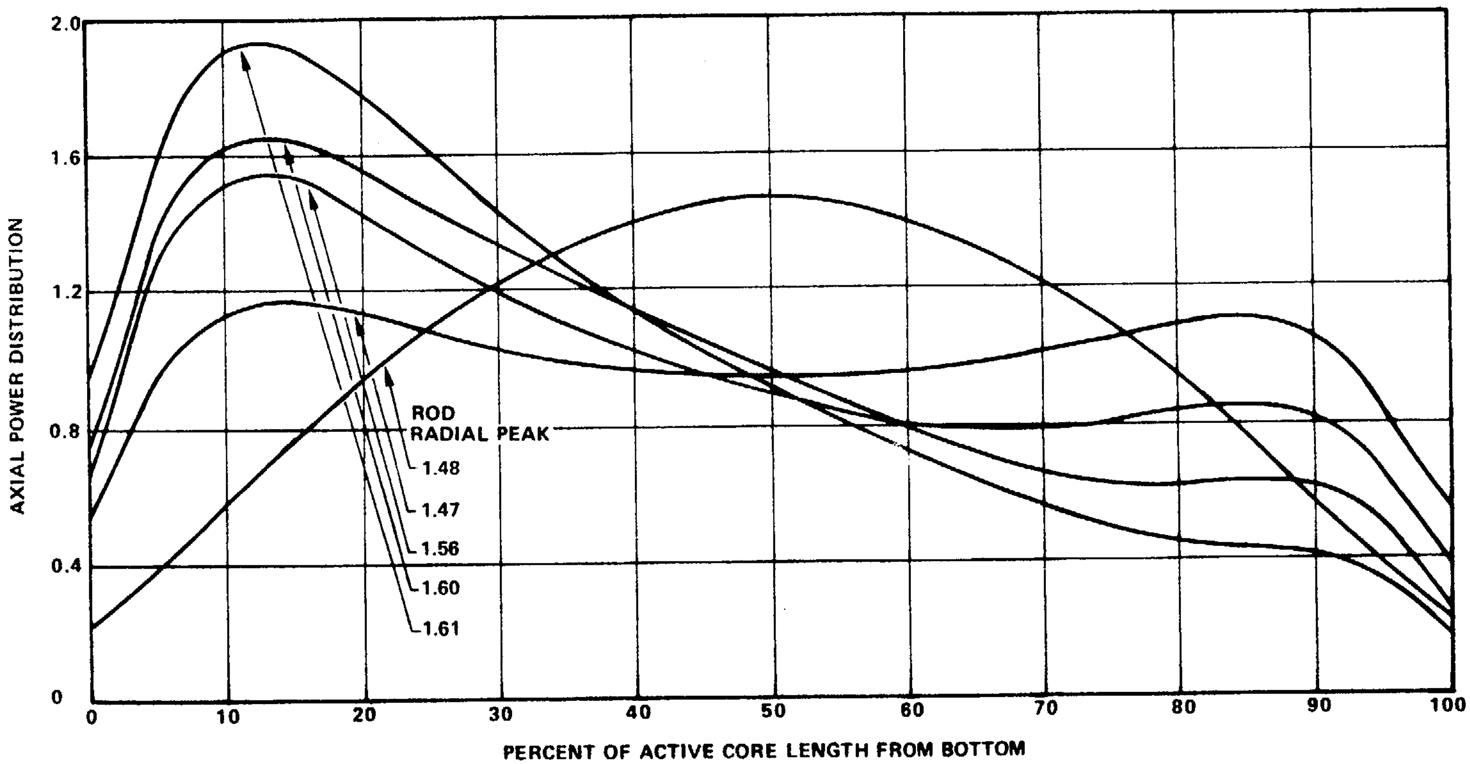

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

Thermal Margin/Low Pressure (Continued)

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 41 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 41 psia allowance is made up of a 22 psia pressure measurement allowance, a 5 psia trip system processing allowance and a 14 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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3.2% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 3.2% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.2% $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

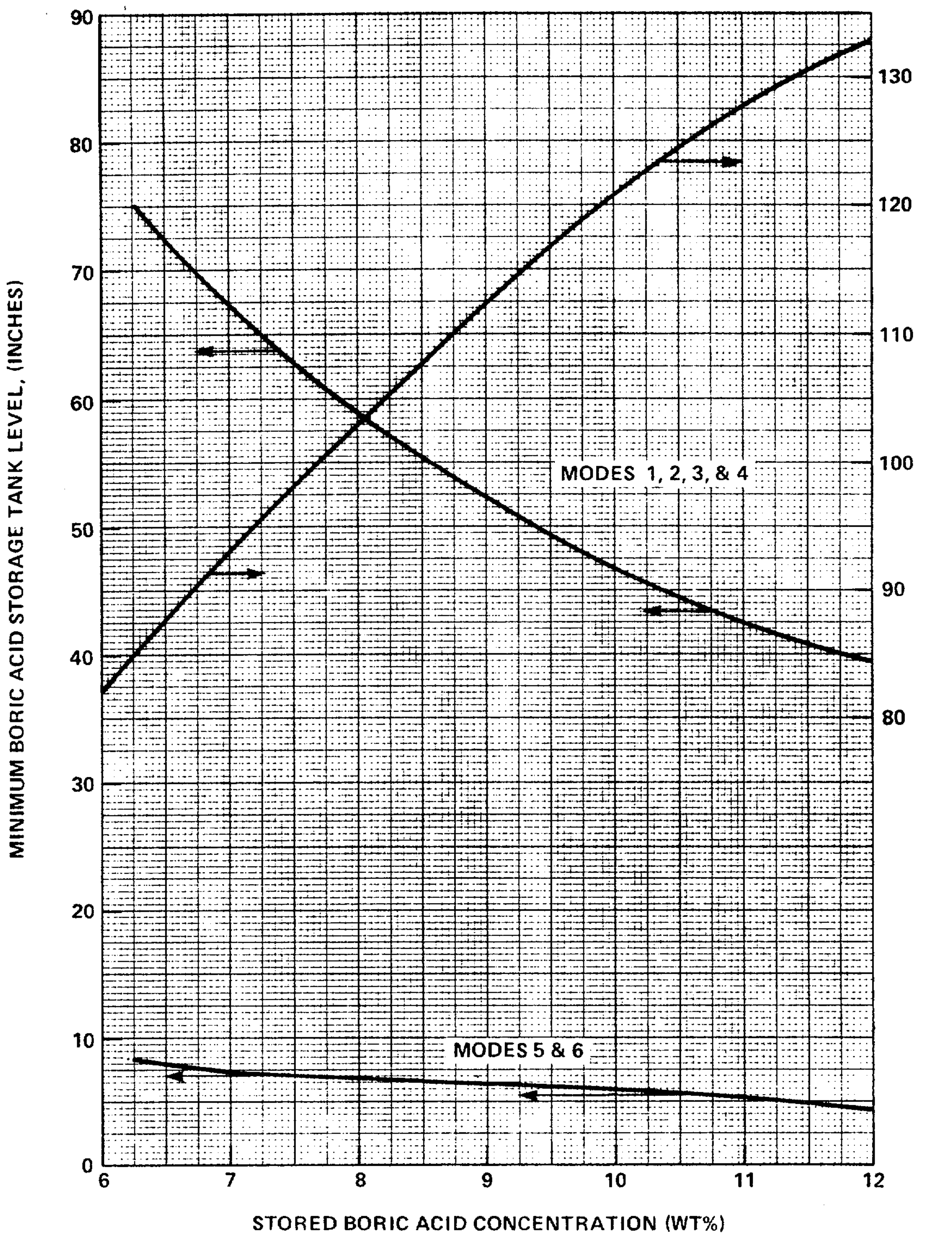

FIGURE 3.1-1

Minimum Boric Acid Storage Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following three borated water sources shall be OPERABLE:

a. Two boric acid storage tank(s) and one associated heat tracing circuit per tank with the contents of the tanks in accordance with Figure 3.1-1 and the boron concentration limited to $\leq$ 8%, and

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,

2. A boron concentration of between 1720 and 2700 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 100°F in MODE 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least one per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume in each water source, and

3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

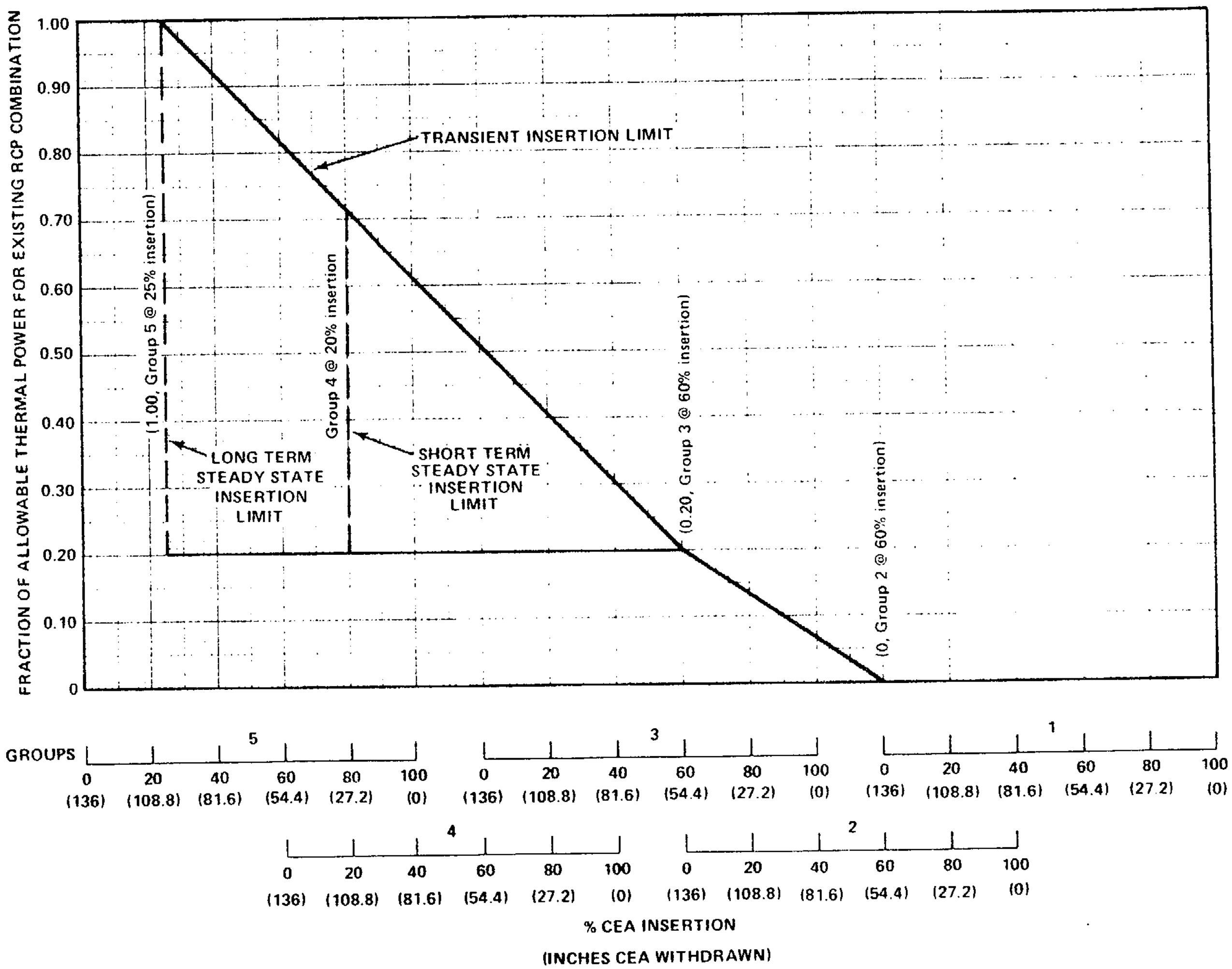

FIGURE 3.1-2

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

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

where:

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

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

3. N is the maximum allowable fraction of RATED THERMAL POWER as determined by Figure 3.2-3 of Specification 3.2.2.

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

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

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

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

   2. A measurement-calculational uncertainty factor of 1.10,

   3. An engineering uncertainty factor of 1.03,

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

   5. A THERMAL POWER measurement uncertainty factor of 1.02.

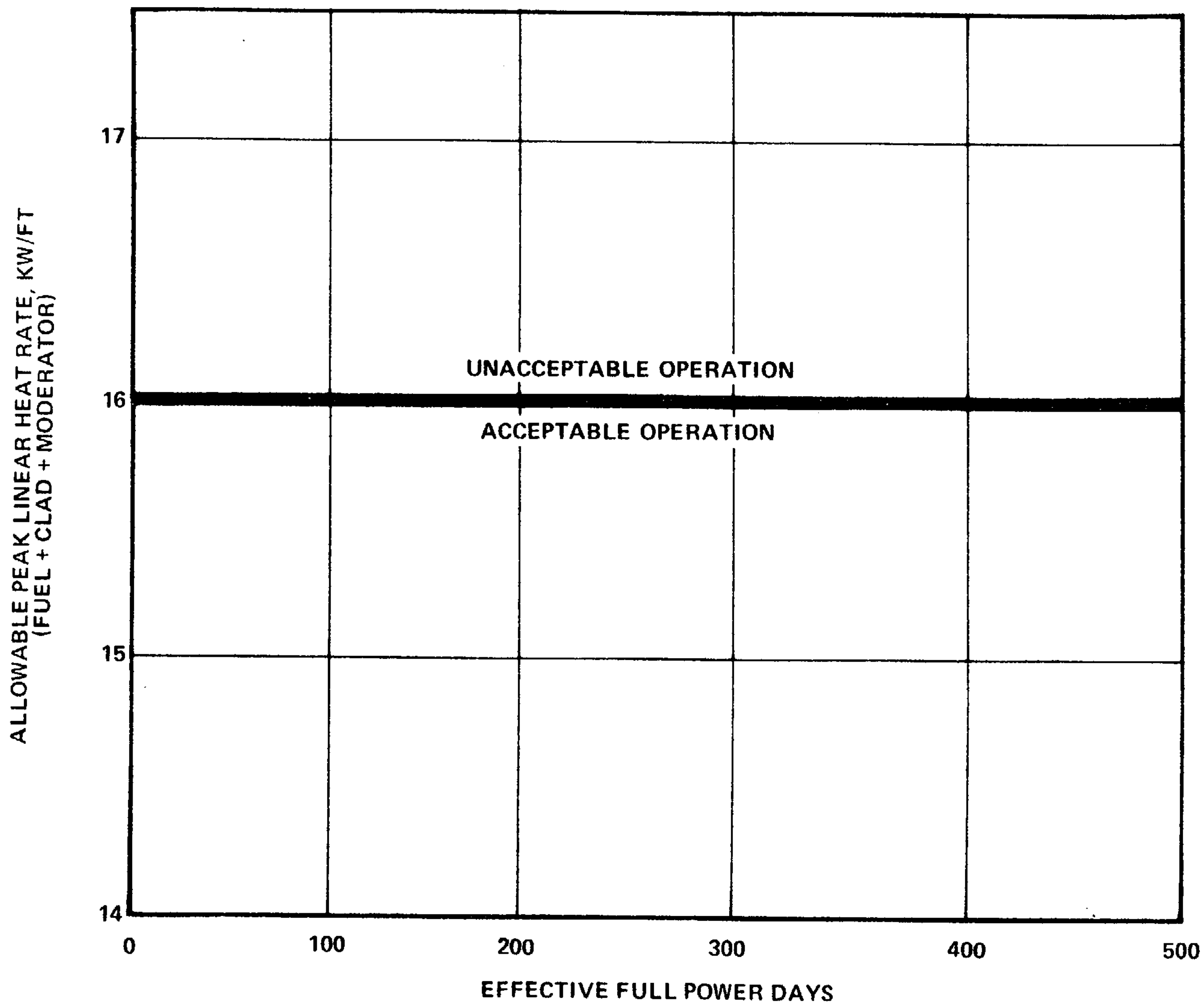

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

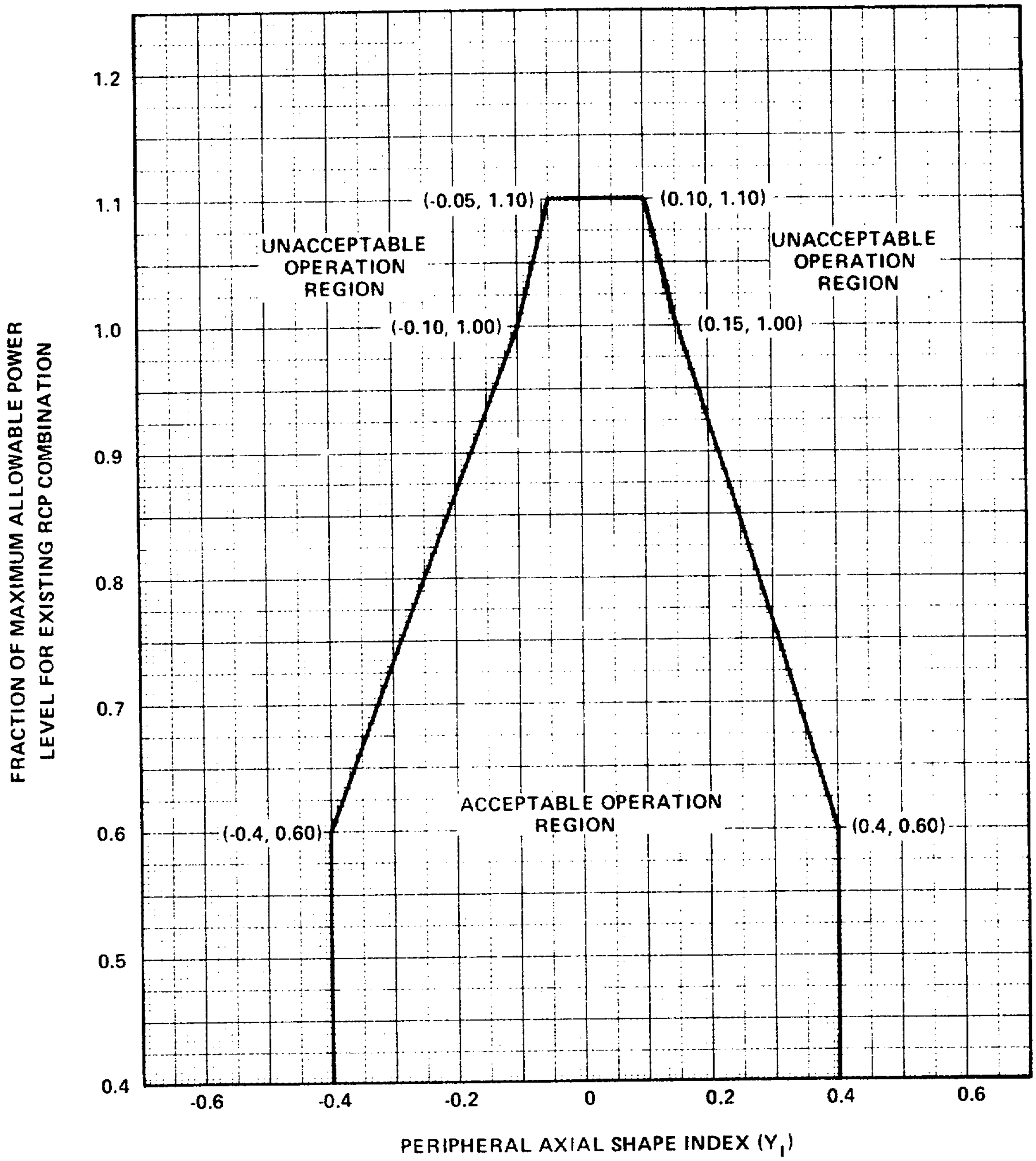

FIGURE 3.2-2
Linear Heat Rate
Axial Flux Offset Control Limits

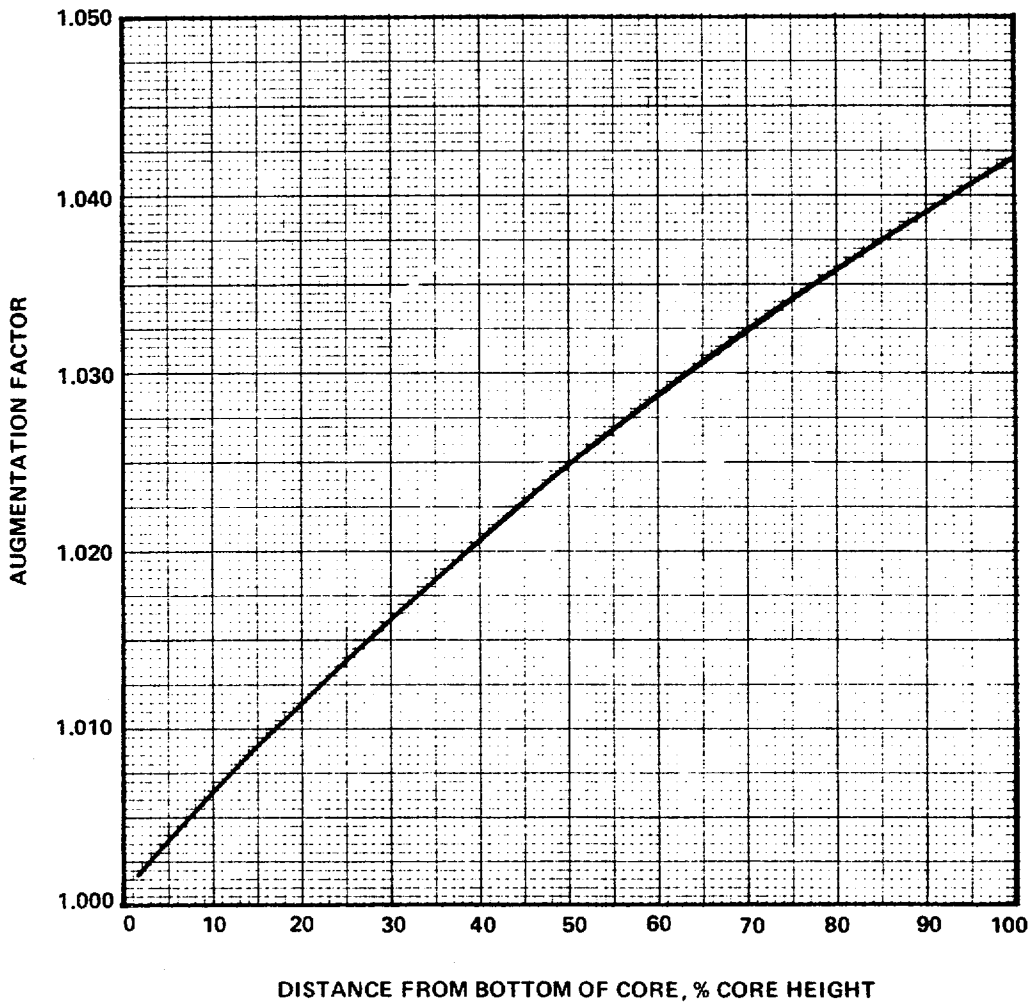

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

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

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to ≤ 1.43.

APPLICABILITY: MODE 1*.

ACTION:

With $F^T_{xy} > 1.43$, within 6 hours either:

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

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+Tq)$ and F^T_{xy} shall be determined to be within its limit at the following intervals:

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

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

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

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

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

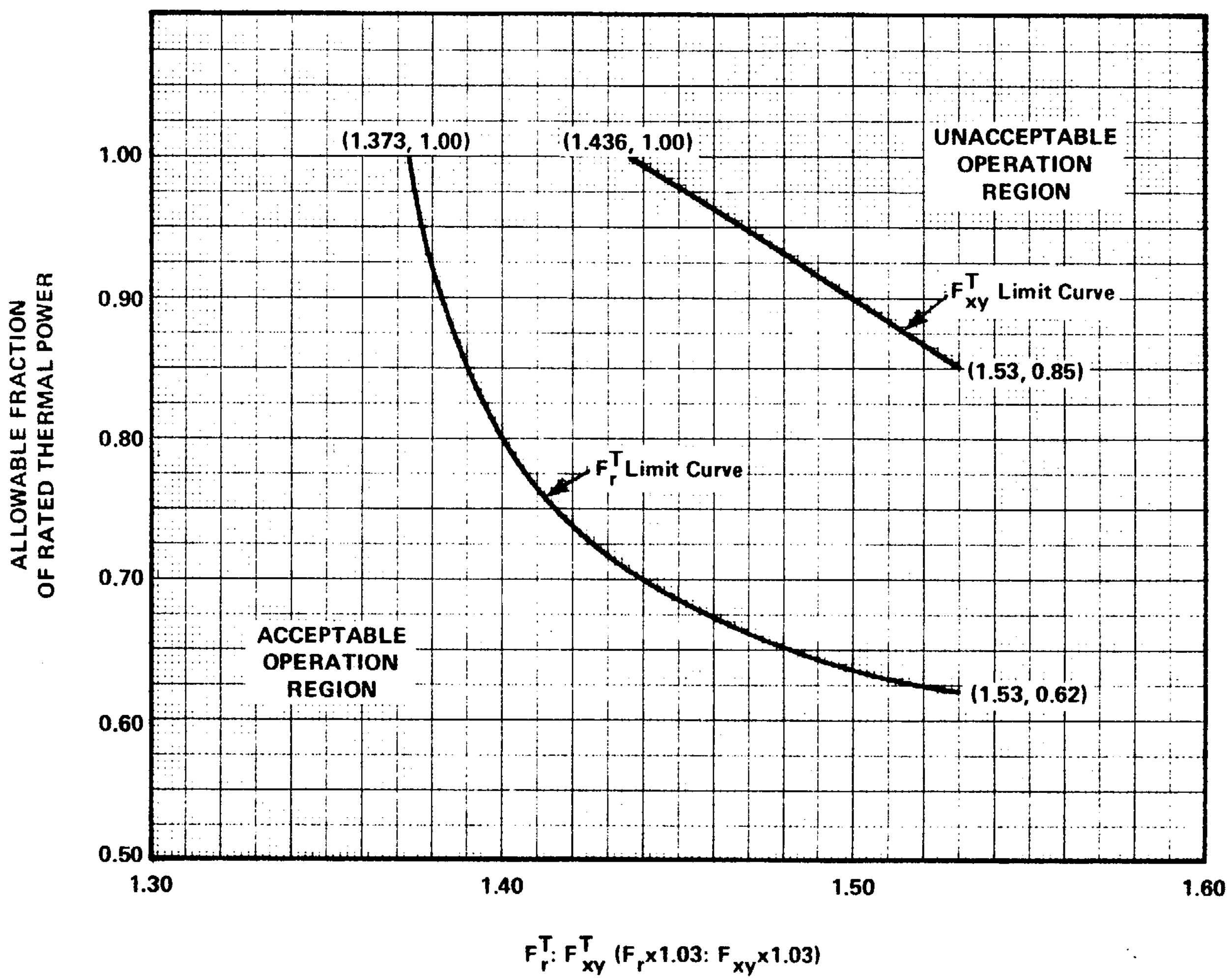

FIGURE 3.2-3
Total Radial Peaking Factor Versus Allowable Fraction of RATED THERMAL POWER

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to ≤ 1.37.

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.37$, within 6 hours either:

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

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

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

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

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

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

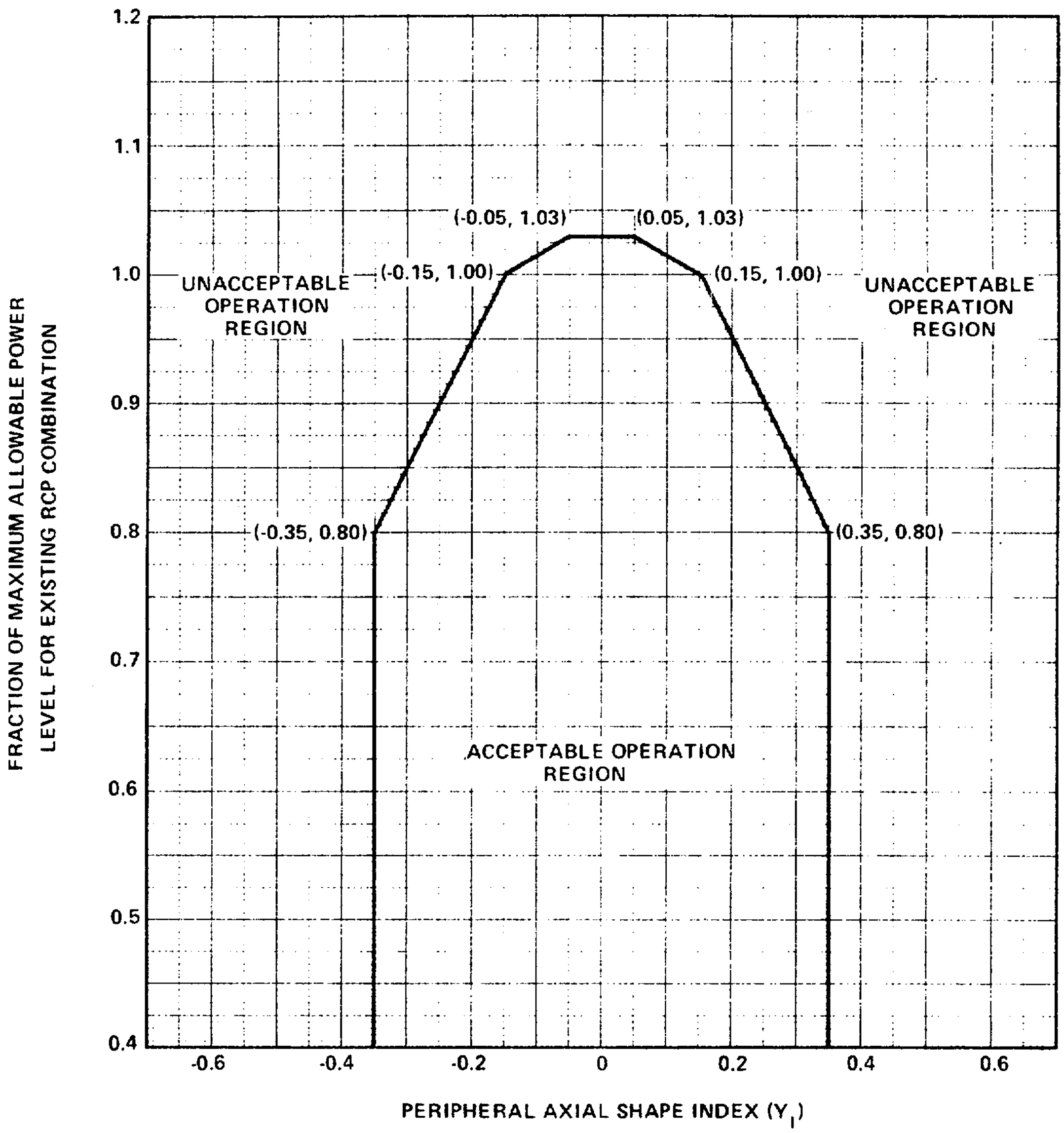

FIGURE 3.2-4

DNB Axial Flux Offset Control Limits

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

*See Special Test Exception 3.10.2.

FUEL RESIDENCE TIME

LIMITING CONDITION FOR OPERATION

3.2.5 The core average fuel burnup shall be limited to $\leq$ 525 Effective Full Power Days during the initial fuel cycle.

APPLICABILITY: MODE 1.

ACTION:

With the core average fuel burnup determined to exceed 525 Effective Full Power Days, be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The core average fuel burnup, based on gross thermal energy generation, shall be determined by calculation at least once per 31 days.

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

Parameter	LIMITS Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	$\leq 548^{\circ}F$	$\leq 548^{\circ}F$	$\leq 548^{\circ}F$	$\leq 548^{\circ}F$
Pressurizer Pressure	$\geq$ 2225 psia*	$\geq$ 2225 psia*	$\geq$ 2225 psia*	$\geq$ 2225 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	Figure 3.2-4	Figure 3.2-4	Figure 3.2-4	Figure 3.2-4

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

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

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons,

b. A boron concentration of between 1720 and 2700 ppm,

c. A minimum water temperature of 40°F, and

d. A maximum water temperature of 100°F in MODE 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 2.4% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200^{o}F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

BASES

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 3813 gallons of 7.25% boric acid solution from the boric acid tanks or 47,204 gallons of 1720 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.2 POWER DISTRIBUTION LIMITS

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 3.2.3 are satisfied, and 4) the TOTAL 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 1.10, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

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

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

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

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

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

3/4.2.4 FUEL RESIDENCE TIME

The limitation on fuel burnup during the initial fuel cycle insures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur until well beyond the proposed first cycle of operation which is about 525 Effective Full Power Days. However, operation beyond the first cycle will require further analyses.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 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 AND ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

DOCKET NO. 50-318

INTRODUCTION

By application dated July 13, 1977[1], as supplemented by filings dated September 30, 1977 and October 5, 1977, the Baltimore Gas and Electric Company (BG&E) requested an amendment to Facility Operating License No. DPR-69 for the Calvert Cliffs Unit No. 2. The request is to allow operation of Cycle 1 fuel at a stretch power level of 2700 MWt, a 5.5% increase above the currently licensed power level of 2560 MWt.

The licensee's application made reference to certain safety analyses and Technical Specification changes which were previously submitted in support of stretch power operation of Unit 1[2-5]. Also included were new analyses and Technical Specifications where required. In addition, the licensee has addressed the applicabilty of the Unit 1 analyses and Technical Specification changes to Unit 2. The effect of differences existing in the two plants, regarding their operation at stretch power, is evaluated in the application.

Based on our review of the above referenced submittals for Calvert Cliffs Unit 1, as documented by our Safety Evaluation[6], the Commission issued Operating License Amendment No. 24, dated September 9, 1977, to allow operation of that unit at 2700 MWt.

Similarly, as for Unit 1, the staff review of the licensee's stretch power application for Unit 2 focused mainly on the impact of the proposed power increase on the safety analyses, physics tests, and operating Technical Specifications and environmental considerations. Our review of these areas is described further below with additional discussion regarding the effects of rod bowing and burnable poison rods.

I. DISCUSSION AND SAFETY EVALUATION

The licensee has provided information in Reference 1 to show that most of the Unit 1 stretch power (2700 MWt) Safety Analyses, the Limiting Safety System Settings (trip points) and Limiting Conditions for Operation Settings conservatively bound the corresponding values for 2700 MWt operation of Unit 2, after its burnup exceeds 6000 MWD/MTU. This fuel exposure was achieved prior to this review. After this burnup, the 3-D peaking factors in Unit 2 are enveloped by the peaking factors assumed in the Unit 1 safety analyses provided that the radial peaking factor limits are reduced in the Technical Specifications as discussed below.

Table 1 shows a comparison of the important key core parameters which are different for Calvert Cliffs Units 1 and 2 for stretch power operation at 2700 MWt. Most of the other parameters which are important to the safety analyses or determination of Technical Specifications are either the same for the two plants or have values for Unit 1 which conservatively bound those for Unit 2.

The remaining potentially non-conservative differences in core and system parameters for the two plants occur in the values of: boron worth and critical boron concentration, CEA drop radial distortion factors for an unrodded core, and reactor coolant pump coastdown characteristics. These latter differences have resulted in the reanalysis of certain transients specifically for stretch power operation of Unit 2, and these are discussed further in the next section. The LOCA is also discussed in a separate section.

TABLE 1

STRETCH POWER PARAMETERS

Parameter	Unit 1 (Cycle 2)	Unit 2 (Cycle 1)
Allowable Peak Linear Heat Generation Rate	14.2 KW/ft	16.0 KW/ft
F_{xy}^{t}	1.50	1.43
F_{r}^{t}	1.42	1.36
Maximum Augmentation Factor	1.063	1.042
Azimuthal Power Tilt	1.02	1.03
Axial Peak Upper Limit	1.35	1.42

TABLE 1 (Cont'd)

STRETCH POWER PARAMETERS

Parameter	Unit 1 (Cycle 2)	Unit 2 (Cycle 1)
Core Average Linear Heat Rate	6.430 KW/ft	6.436 KW/ft
Number of Shims in Core	1260	1296

The slight increase in core average linear heat rate for Unit 2 indicated in Table 1 is due to the greater number of poison shim rods present in that core. Units 1 and 2 originally had an identical number of shim rods but as was described in Reference 2, 36 damaged shim rods were removed from Unit 1 during the first refueling outage because of blisters caused by hydriding. As a result of this experience on Unit 1, the poison shim rods in Unit 2 were reworked prior to startup to prevent similar damage. The staff's approval of the rework is summarized in Reference 7.

The decrease in augmentation factor indicated for Unit 2 is a result of increased fuel pellet density with a subsequent decrease in the adverse effects of in-pile densification. The corresponding improvement in gap conductance is responsible for the increase in allowable peak linear heat generation rate shown in Table 1.

To demonstrate that operation of Unit 2 at stretch power as proposed will be generally conservative relative to Unit 1, the licensee has submitted a comparison of the maximum permitted peak linear heat rate (not including uncertainties) but including augmentation factors and a tilt allowance. While some of the core parameters which contribute to peak linear heat rate are seen in Table 1 to be greater for Unit 2 than for Unit 1, the combined effect of all of the pertinent parameters is such that the local power for Unit 2 will be less than that for Unit 1. The combined effect of the core parameters may be determined by calculating the product of the core average linear heat rate, the augmentation factor, and the axial and radial peaking factors. The result is the maximum allowed peak linear heat rate (maximum local power). This calculation has been performed taking the values from Table 1, and the results are shown in Table 2.

TABLE 2

CALCULATED MAXIMUM ALLOWABLE PEAK LINEAR HEAT GENERATION RATES

Calvert Cliffs Plant	Calculated Peak Linear Heat Generation Rate
Unit 1	13.84 KW/ft
Unit 2	13.62 KW/ft

The results shown in Table 2 confirm that most of the stretch power operating conditions and Technical Specifications for Unit 2 are conservatively bound by those for Unit 1. This is true provided that the reduction in the radial peaking factor (F_{xy}^t) as indicated in Table 1 is incorporated into the Unit 2 Technical Specifications. In further support of the above, the licensee states that he has reviewed the input parameters assumed in the safety analyses of Unit 1[2] for stretch power operation and confirmed that they are generally more limiting than the corresponding values for Unit 2. We conclude that this is acceptable. Exceptions to the above statements have been addressed in additional safety analyses as described below.

It should be noted that the values of maximum linear heat rate calculated above are nominal, as the parameters in Table 1 do not include uncertainties. Since the uncertainties are identical for both plants, they do not affect the comparison results. Including uncertainties would increase the calculated values of linear heat rate, however, in reality, this increase is offset by the fact that the axial peaking factors shown in Table 1, and included in the Technical Specifications, are conservatively high. This has been demonstrated by incore power distribution measurements made at both plants.

Based on our review, we conclude that most of the Calvert Cliffs Unit 1 safety analyses and Technical Specifications (with modifications as proposed) are applicable and acceptable for stretch power operation of Unit 2.

The remaining safety analyses including LOCA are discussed below.

ADDITIONAL SAFETY ANALYSES

As noted above, it has been shown that most of the safety analyses for Unit 2 are bounded by the Unit 1 stretch power analyses provided that the radial peaking factor limit for Unit 2 is reduced as indicated.

However, other plant differences have made it necessary to reanalyze certain transients specifically for stretch power operation of Unit 2. The plant differences and the affected transients are:

1. Reactor coolant pump coastdown characteristics (Loss of Coolant Flow Incident)

2. CEA drop radial distortion factor (Full Length CEA Drop Incident)

3. Boron worth and critical boron concentration (Boron Dilution Incident)

Following is a brief review of each of the affected transients. LOCA is discussed in a separate section.

Loss of Coolant Flow Incident

The Loss of Coolant Flow incident was reanalyzed for Unit 2 at 2700 MWt since the Unit 2 reactor coolant pumps coast down faster than the Unit 1 pumps. In addition to the difference in coastdown, the radial peaking factor was assumed to be the lower value shown in Table 1 above for Unit 2, and the coolant inlet temperature was reduced slightly below the Unit 1 analysis temperature in accordance with the current value of this parameter. Other than the changes noted, the same standard Combustion Engineering analysis methods were used in the reanalysis as were used for Unit 1.

The analysis results indicated that the change in pump coastdown characteristics caused low flow trip and the time of minimum DNBR to occur sooner relative to the Unit 1 analysis. The minimum DNBR was 1.54, while for Unit 1 this value was 1.3. The maximum increase in reactor coolant system pressure was the same as for the Unit 1 analysis (initially at 2200 psia - increasing to 2289 psia). We conclude that the reanalysis is acceptable.

Full Length CEA Drop Incident

The Full Length CEA Drop Incident was reanalyzed for Unit 2 due to an increase in the all rods out radial distortion factor and increase in dropped CEA worth in comparison with the Unit 1 stretch power analysis. Otherwise, the same standard Combustion Engineering analysis method was used for Units 1 and 2.

The results indicate that the increase in unrodded distortion factor for Unit 2 is compensated for by the increase in negative CEA drop worth resulting in a less severe DNBR transient for Unit 2. The minimum DNBR was calculated to be 1.42 for Unit 2 while the corresponding value for Unit 1 was 1.30. We conclude that the reanalysis is acceptable.

Boron Dilution Incident

To account for differences in boron worth and critical boron concentration between Calvert Cliffs Units 1 and 2 at stretch power, a reanalysis of the boron dilution incident was performed for Unit 2. For conditions of hot-zero-power-critical, and operation at power, this transient produces a slow power and temperature increase which causes an approach to both the DNBR and Linear Heat Rate Specified Acceptable Fuel Design Limits (SAFDL). For these cases, the power excursion is terminated by one or more of the following reactor trips: Axial Flux Offset, Thermal Margin-Low Pressure, or Variable High Power. For boron dilutions initiated from refueling and startup conditions, alarms and indications in the control room alert the operators and reduce the probability of a sustained dilution. In the event the operators do not take the appropriate action, the above listed reactor trips again assure that the SAFDL will not be exceeded.

The analysis results indicate that the time periods available for operator action for the above cases are adequate, and we conclude that the reanalysis is acceptable.

Loss of Coolant Accident

Due to the similarity of the Calvert Cliffs Units 1 and 2 reactors, common LOCA blowdown and refill-reflood calculations have been performed which are applicable to stretch power operation (2700 MWt) of each plant[3]. These calculations have been referenced in the Unit 2 application. However, burnup dependent calculations have been performed specifically for each plant to determine the most limiting burnup during stretch power operation. Also, hot rod thermal transient calculations were performed separately for each plant to account for the sensitivity of the thermal behavior to the unique features of the fuel. As noted above, Calvert Cliffs Unit 2 contains fuel having an increased fuel pellet density resulting in a decrease in the adverse effects of densification. The corresponding increase in gap conductance has a strong influence on the LOCA results. This is reflected in the calculated allowable peak linear heat generation rate (PLHGR) of 16.0 KW/ft for Unit 2 at stretch power, while the stretch power PLHGR for Unit 1 is 14.2 KW/ft.

Both the referenced Unit 1 LOCA calculations and those performed specifically for Unit 2 were done using standard, approved Combustion Engineering ECCS evaluation models.

The results of the stretch power LOCA analyses are summarized below in Table 3.

TABLE 3

STRETCH POWER LOCA ANALYSIS RESULTS

Parameter	Unit 1 Low Density Fuel	Unit 1 High Density Fuel	Unit 2
Linear Heat Generation Rate	14.2 KW/ft	16.5 KW/ft	16.0 KW/ft
Peak Clad Temperature	2145°F	2120°F	1957°F
Local Clad Oxidation	7.2%	13.4%	8.9%
Overall Clad Oxidation			
(Hydrogen Generation)	<0.54%	<0.591%	<0.49%
Worst Break	*0.8DES/PD	1.0 DES/PD	1.0 DES/PD

*Signifies that the worst break (highest peak clad temperature) occurred for a break having a Moody discharge coefficient of 0.8, was of the double-ended-split (DES) type, and was located on the reactor coolant pump discharge (PD) side of the cold leg.

As indicated in Table 3, the predicted values of peak clad temperature, local clad oxidation, and hydrogen generation are below their respective limits of 2200°F, 17 percent, and 1 percent which are specified in 10 CFR 50.46(b).

The effect of fuel rod bowing on fuel rod and poison shim rod behavior has not been explicitly included in the Calvert Cliffs LOCA analyses. However, the subject of the effects of fuel rod bowing on Combustion Engineering 14x14 fuel, such as that used in Units 1 and 2, is discussed generically in a letter submitted to the Nuclear Regulatory Commission by Combustion Engineering[(8)]. In the letter, Combustion Engineering states its position that the uncertainty factors which are presently applied to the Combustion Engineering 14x14 fuel are sufficiently large to account for the effects of rod bowing. These uncertainty factors are the 8% factor applied for nuclear power distribution measurement uncertainty and the 3% engineering factor uncertainty.

We have reviewed the generic rod bowing information submitted by Combustion Engineering. It is our conclusion that the uncertainty factors which are presently included in the LOCA analyses for the Calvert Cliffs plants, and which are described above, are sufficient to account for rod bowing effects.

In our review of the LOCA analysis for Unit 2 at stretch power we have considered the possible effects of reducing the value assumed for the reactor coolant inlet temperature. This effect was discussed in detail in the Calvert Cliffs Unit 1 safety evaluation report for stretch power[6] where it was noted that reducing coolant inlet temperature has resulted in increases in the predicted peak clad temperatures for some PWRs. In the Unit 1 safety evaluation, the staff concluded that the margin to the 10 CFR 50 limit on peak clad temperature (2200°F), which was indicated by the LOCA analysis results, was sufficient to offset the possible increase in predicted peak clad temperature caused by assuming a reduction in coolant inlet temperature. The Unit 2 analysis results show even a greater margin to the 2200°F peak clad temperature limit (243°F margin as compared to 55°F for Unit 1), and we conclude that the indicated margin is sufficient to account for variations in coolant inlet temperature. In Reference 9, the licensee addressed the effect of inlet temperature on ECCS performance of Calvert Cliffs Unit 2. The licensee's submittal cites the large indicated margin to the Acceptance Criteria limits and adds that the version of the PARCH code [10] which was used during the reflood portion of the analysis did not include a staff approved benefit which would substantially increase the calculated margins yet further. Similarly as for Unit 1, we will require that the licensee perform a confirmatory calculation to determine the specific sensitivity of Unit 2 to changes in inlet temperature. A schedule for submitting the confirmatory calculation is presently being defined.

As a result of our review, we conclude that the Calvert Cliffs Unit 2 ECCS performance for stretch power operation at 2700 MWt and a peak linear heat generation rate no greater than 16.0 KW/ft will conform to the peak clad temperature, maximum oxidation, hydraogen generation, coolable geometry, and long term cooling criteria of 10 CFR 50.46(b) and is therefore acceptable.

Radiological Consequences of Postulated Accidents

We have reviewed the evaluation of the potential radiological consequences of the postulated loss-of-coolant accident, fuel handling accident, steam line failure accident, steam generator tube failure accident and radioactive gas storage tank accident in the Safety Evaluation Report (SER)[12].

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

The analyses for the loss-of-coolant accident and the fuel handling accident in the SER(12) were performed at 2700 MWt. Neither the rod ejection accident nor the fuel handling accident inside containment were reviewed and evaluated in the SER. The fuel handling accident in the SER is for a postulated accident in the spent fuel building. On March 21, 1977, the licensee submitted an analysis of the fuel handling accident inside containment for Calvert Cliffs Units 1 and 2(13). We have reviewed this analysis. The assumptions for this accident are the same as those for the fuel handling accident in the SER except there is no iodine removal factor of 6.67 for a charcoal filter as there is no engineered safety feature ventilation filtration system to reduce the consequences of the fuel handling accident inside containment.

The consequences of the loss of coolant accident, fuel handling accident, steam line failure accident, steam generator tube failure accident, and radioactive gas storage tank accident at 2700 MWt are given in the SER(12). The consequences of the fuel handling accident inside containment and the rod ejection accident at 2700 MWt were determined by the NRC staff and reported in the safety evaluation for Unit No. 1 stretch power(6). The potential consequences of these accidents at 2700 MWt, assuming all the parameters presented in the SER are not changed, are significantly less than the guidelines of 10 CFR Part 100.

We found during our review of this action that the control room operator doses resulting from a LOCA were never analyzed by the licensee or by the staff. We have requested that BG&E supply the additional information necessary for our evaluation of this concern by separate letter dated October 17, 1977. If the analysis indicates that plant changes are necessary to reduce these doses, such changes would most likely involve control room ventilation system modifications, not a reduction in the authorized reactor power level. Therefore, we have determined that this amendment allowing stretch power can be authorized independent of our review of the control room operator LOCA doses.

Radioactive Waste

We expect that increasing the thermal power level of Calvert Cliffs Unit 2 from 2560 MWt to 2700 MWt will increase the concentration of activity in the reactor primary coolant and in water entering the radwaste treatment systems. This increase should be less than the percentage increase in the thermal power level which is 5.5%. This small increase in the concentration of activity will not affect the performance of equipment in the radwaste treatment systems. There is also no change in the flows and volumes of liquids and gases in these systems. Therefore, we expect the increase in radwaste effluents due to the change in thermal power level to also be less than the percentage increase in the thermal power level. This small increase in radioactive effluents does not change our conclusion in the SER[12] that the radwaste treatment system at Calvert Cliffs Unit 2 will be capable of limiting radioactive releases to values which are a small fraction of 10 CFR Part 20 limits.

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

By letters[14], BG&E provided additional information pursuant to Appendix I to 10 CFR Part 50. After we complete our evaluation of these submittals we intend to revise the Technical Specifications to reflect the requirements of Appendix I.

Physics Tests

The licensee has described his confirmatory test program incident to increasing rated thermal power to 2700 MWt in Reference 1.

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

i) Isothermal Temperature Coefficient Measurement

ii) Power Coefficient Measurement

iii) Power Distribution Measurement

The test methods employed will be similar to those described in the Calvert Cliffs Unit No. 1 Startup Test Report (Reference 11).

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

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

Technical Specifications

The results of the steady-state and transient safety analyses performed for a power level of 2700 MWt as described above have been used to define Limiting Conditions for Operation (LCO) and Limiting Safety System Setpoints (LSSS). The LCO and LSSS assure that the initial steady-state overpower margin and the action of the Reactor Protective System will prevent a violation of the Specified Acceptable Fuel Design Limits during Anticipated Operational Occurrences. They also assure that radioactive material releases during postulated accidents will remain within the Commission's guidelines of 10 CFR 100.

Reference 1 includes proposed Technical Specification modifications applicable to operation of Calvert Cliffs Unit 2 at a stretch power level of 2700 MWt.

As noted above, it has been shown that most of the 2700 MWt Technical Specifications approved for Unit 1 are also applicable to stretch power operation of Unit 2. The major items requiring change for stretch power operation which are identical for Units 1 and 2 are listed below by Technical Specification Section Number.

Section 1.3 (page 1-1) RATED THERMAL POWER is changed to 2700 MWt.

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

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

Section 3.1 (Figure 3.1-2) POWER DEPENDENT INSERTION LIMITS modified for 2700 MWt operation.

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

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

Technical Specifications which are different for Unit 2 than for Unit 1 at stretch power include:

Section 2.1 (Figure B2.1-1) AXIAL POWER DISTRIBUTION FOR THERMAL MARGIN SAFETY LIMITS modified to reflect higher upper limits for axial peaking in Unit 2.

Section 3.2 (Figure 3.1-1) ALLOWABLE PEAK LINEAR HEAT RATE modified to reflect allowable limit for Unit 2 at stretch power.

Section 4.2 (Figure 4.2-1) AUGMENTATION FACTOR modified to reflect improved densification characteristics of Unit 2 fuel.

Section 3.2.2 and Figure 3.2-3 TOTAL PLANAR RADIAL PEAKING FACTOR modified to reflect reduced value for Unit 2.

Section 3.2.3 and Figure 3.2-3 TOTAL INTEGRATED RADIAL PEAKING FACTOR modified to reflect reduced value for Unit 2.

Section 3.2.2, 3.2.3, 3.2.4 AZIMUTHAL POWER TILT modified to reflect increased value for Unit 2.

Other less important changes have been proposed to make the Unit 2 Technical Specifications consistent with those for Unit 1.

Based upon our review of the Technical Specification modifications proposed in Reference 1 for stretch power operation of Calvert Cliffs Unit 2, we conclude that the proposed modifications are acceptable.

SAFETY CONCLUSION

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

II. ENVIRONMENTAL IMPACT APPRAISAL

Prior to issuing Amendment 24, allowing Stretch Power for Calvert Cliffs Unit No. 1, we reviewed the Final Environmental Statement (FES)[15] related to the operation of both Calvert Cliffs Units. We addressed the sections of the FES that were affected by the power level increase in the Environmental Impact Appraisal (EIA) for Amendment 24[16].

In this review, we have confirmed that the potential environmental impacts associated with the proposed action are those identified in the EIA for Amendment 24 [16]. Increasing the Unit 2 power level to 2700 MWt will increase the heat output to the Chesapeake Bay and the quantity of radioactive waste.

Heat Output

The current Appendix B Technical Specifications limit the condenser ΔT to $10^{o}F$ ($5.56^{o}C$). BG&E expects that the increase in power level from 2560 to 2700 MWt may theoretically result in an average ΔT increase of about $0.6^{o}F$[17]. They also state their intention to not exceed the $10^{o}F$ temperature rise and restrict operation accordingly. Since the FES is based on the $10^{o}F$ ΔT and the condenser flow rate is not changed, the maximum heat rejected rate to the bay is as analyzed by the FES.

However, since the normal ΔT for 2560 MWt operation has been below $10^{o}F$, this change allowing operation at 2700 MWt will increase the heat output to the bay to the maximum allowed. The environmental impact of this discharge of heat has been previously analyzed and approved by Maryland State Department of Natural Resources, Water Resources Administration, in issuing the NPDES Permit[18] and by the NRC in issuing the Appendix B Technical Specification[19].

Radioactive Waste

The FES evaluation of the radioactive waste treatment systems of Units 1 and 2 was performed for a thermal power level for both plants of 2560 MWt not for 2700 MWt[15]. Increasing the thermal power level by 5.5% can be expected to increase the estimated releases of radioactive materials and the estimated radiological impact of Calvert Cliffs Unit 2 in the FES.

BG&E has pointed out that the FES was prepared assuming operation at rated power (2560 MWt) and 0.25% failed fuel[17]. Environmental Statements today are prepared in accordance with Regulatory Guide 1.112 and ANSI N-237. These references require that one assume 0.125% failed fuel. Therefore, if the FES were redone today using present guidelines, the estimated releases of radioactive material to the environment would be nearly 50% less than those assumed in the original FES. The FES was prepared assuming that Lithium was not added to the Reactor Coolant System (RCS). The addition of Lithium increases the production of Tritium in the RCS during power operation. Combustion Engineering's present best estimate is a production rate of 826 curies per year and a maximum rate of 1508 curies per year. The FES estimates 1000 curies per year. They conclude that the FES is conservative in addressing radioactive discharges from Calvert Cliffs while operating at stretch power. Actual isotopic concentrations should be less than those reported in these two documents. We expect that the increase in radioactive waste will be no more than the percentage increase in the thermal power level, 5.5% of the estimates given in the FES.

ENVIRONMENTAL CONCLUSION

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

Date: October 19, 1977

REFERENCES

1. Calvert Cliffs Unit 2 Request for Amendment to Operating License Allowing Stretch Power Operation, letter to D. K. Davis from A. E. Lundvall, Jr., July 13, 1977.

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

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

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

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

6. Safety Evaluation Supporting Amendment No. 24 to Operating License No. DPR-53 Calvert Cliffs Unit 1, September 9, 1977.

7. Poison Rod Rework for Calvert Cliffs 2, NRC Memorandum for H. Rood from R. O. Meyer, November 1, 1976.

8. Fuel and Poison Rod Bowing Effects in Combustion Engineering Fuel, letter to D. F. Ross from A. E. Scherer, July 16, 1976.

9. Inlet Temperature Sensitivity of Calvert Cliffs Unit Two ECCS Performance, September 30, 1977.

10. CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model," August, 1974.

    CENPD-132, Supplement 1, December, 1974.

    CENPD-132, Supplement 2, July, 1975.

11. Calvert Cliffs Nuclear Power Plant Unit No. 1 Startup Test Report, August 29, 1975.

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

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

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

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

16. Environmental Impact Appraisal Supporting Amendment No. 24 to DPR-53, September 9, 1977.

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

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

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND

NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 9 to Facility Operating License No. DPR-69, issued to Baltimore Gas and Electric Company (the licensee), which revised the license and its appended Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit No. 2 (the facility) located in Calvert County, Maryland. The amendment is effective as of the date of its issuance.

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

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of the Amendment to Facility Operating License in connection with this action was published in the

Federal Register on August 22, 1977 (42 F.R. 42264). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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

For further details with respect to this action, see (1) the application for amendment dated July 13, 1977, and supplements dated September 30, 1977 and October 5, 1977, (2) Amendment No. 9 to License No. DPR-69, and (3) the Commission's combined Safety Evaluation and Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Calvert County Library, Prince Frederick, Maryland 20678. A single copy of 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 19th day of October, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

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