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Docket No. 50-318

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

REGULATORY DOCKET FILE COPY

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2 (CCNPP-2). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 27, 1979, as supplemented October 1, 1979.

The amendment changes the Appendix A TS of CCNPP-2 to increase the measurement/calculational uncertainties for peaking factors F_p and F_q from 5.1 and 5.8 percent to 6.0 and 7.0 percent, respectively.

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

Copies of the Safety Evaluation and the 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. 24 to DPR-69
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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DATE	11/24/79	11/29/79	12/4/79	12/3/79	12/5/79



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

December 11, 1979

Docket No. 50-318

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. 24 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2 (CCNPP-2). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 27, 1979, as supplemented October 1, 1979.

The amendment changes the Appendix A TS of CCNPP-2 to increase the measurement/calculational uncertainties for peaking factors F_r and F_q from 5.1 and 5.8 percent to 6.0 and 7.0 percent, respectively.

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

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

Sincerely,

A handwritten signature in cursive script, appearing to read "Robert W. Reid".

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

Enclosures:

1. Amendment No. 24 to DPR-69
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Baltimore Gas and Electric Company

cc w/enclosure(s):

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Commissioners
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(AW-459)
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Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
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Philadelphia, Pennsylvania 19106

cc w/4 cys enclosures and 1 cy
of BG&E filings dtd: 8/27/79 & 10/1/79

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. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated August 27, 1979 as supplemented October 1, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 24

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.

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DEFINITIONS

Ē - AVERAGE DISINTEGRATION ENERGY

1.20 \bar{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.

$$Y_E = \frac{L-U}{L+U}$$

$$Y_I = AY_E + B$$

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.

DEFINITIONS

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.

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.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

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

$$M \times N$$

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.
2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_{xy} curve shown up 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 1.07,
 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.

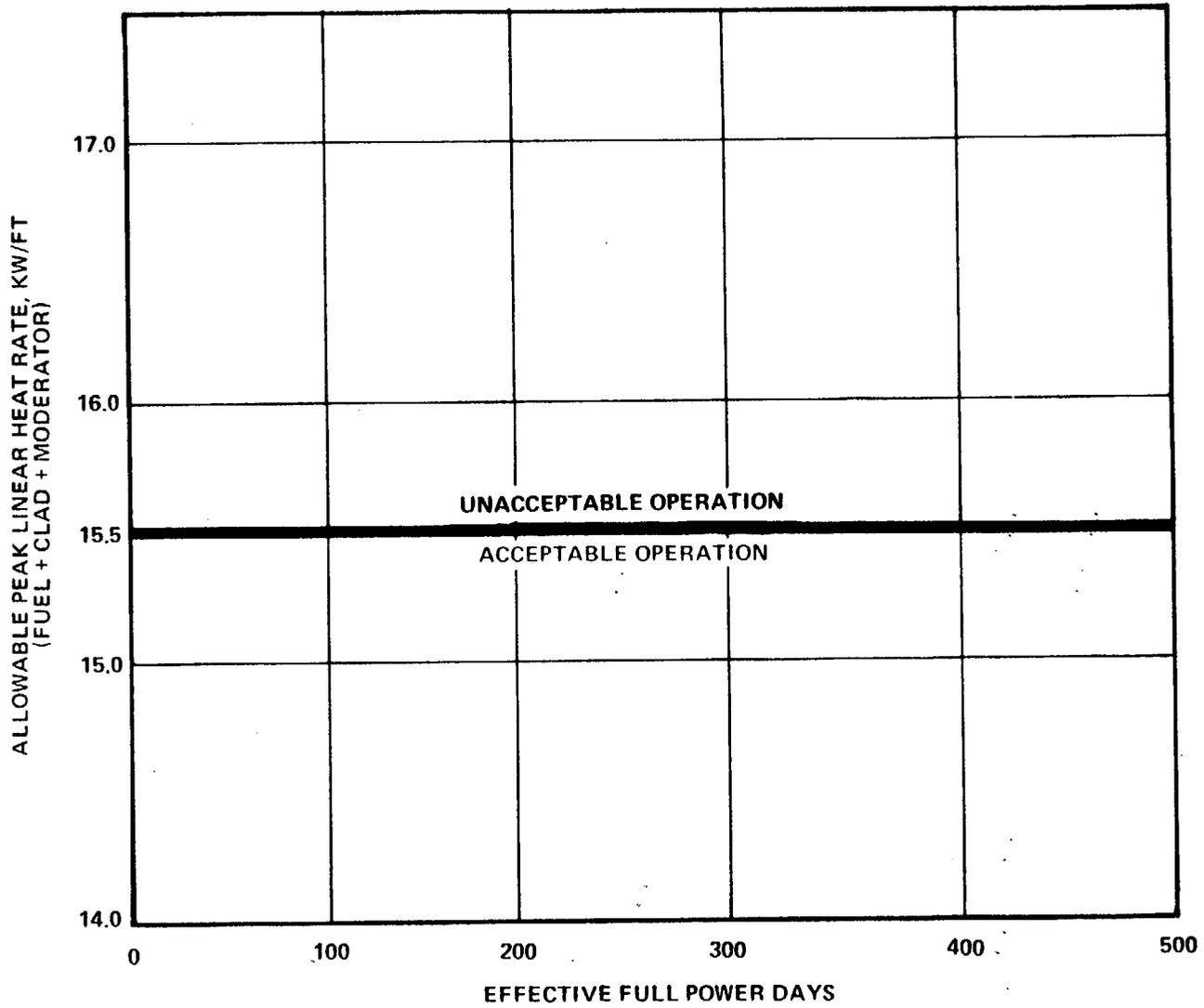


Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

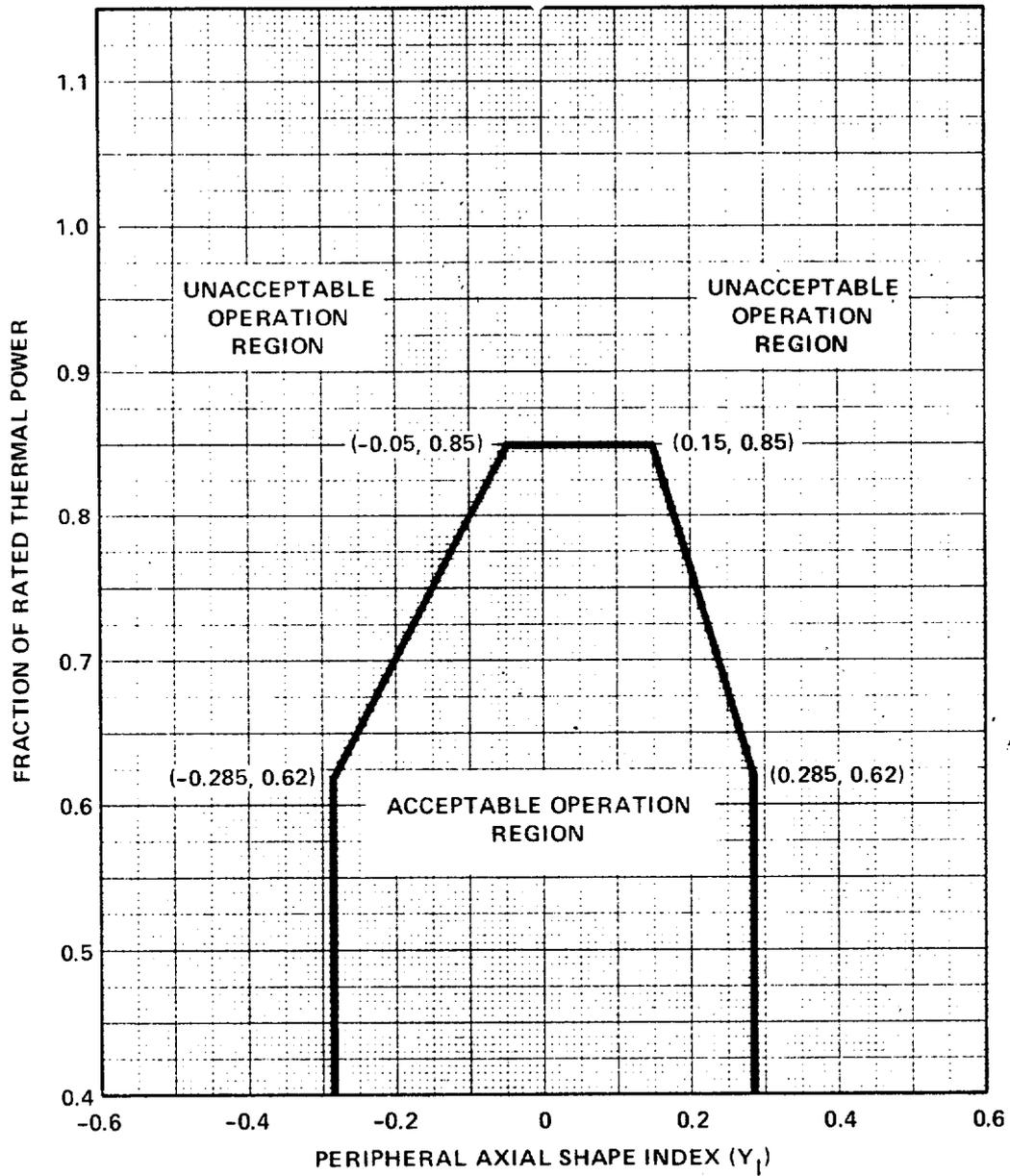


FIGURE 3.2-2
 Linear Heat Rate
 Axial Flux Offset Control Limits

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 1.070*, 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_{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

POWER DISTRIBUTION LIMITS

BASES

Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy} , F_r or T 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 $F_{xy}^T = F_{xy} (1 + T_q)$ and $F_r^T = F_r (1 + T_q)$ is the measured tilt.

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

3/4.2.5 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 CE-1 calculated 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. 24 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 August 27, 1979, and supplemental information dated October 1, 1979, Baltimore Gas and Electric Company (BG&E or the licensee) requested an amendment to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit No. 2 (CCNPP-2). The amendment request consisted of changes to the Technical Specifications (TS) increasing the measurement/calculational uncertainties for peaking factors F_r and F_q from 5.1 and 5.8 percent to 6.0 and 7.0 percent, respectively.

Discussion and Evaluation

In the CCNPP-2 Cycle 2 analysis performed for BG&E by Combustion Engineering (CE), measurement/calculational uncertainties of 5.1 and 5.8 percent for F_r and F_q were used. These uncertainties had previous values of 8 and 10 percent. Because of unanswered questions, the NRC staff could not approve the proposed change in values of these uncertainties. BG&E and CE proposed two modes of operation: (1) load following operation where the 8 and 10 percent uncertainties would apply; and (2) non-load following operation where the lower uncertainties would apply. Using this approach, they were able to identify conservatisms in the safety analysis methodology where no credit was taken. Our evaluation of the acceptability of taking credit for these conservatisms is documented in the Cycle 2 reload Safety Evaluation of October 21, 1978.

As a result of meetings between the NRC staff and CE prior to and during our review of the Cycle 4 reload analysis for CCNPP-1, uncertainties of 6 and 7 percent for F_r and F_q , respectively, were found to be acceptable for all modes of operation. This is documented in our Safety Evaluation of June 14, 1979.

The requested change, increasing the uncertainty of F_r from 5.1 to 6 percent and F_q from 5.8 to 7 percent during normal (non-load follow) operation, is in the conservative direction, i.e., increases the constraints on plant operation. However, for load following operation, a mode not planned for use at CCNPP, the reduction in the uncertainty of F_r from 8 to 6 percent and F_q from 10 to 7 percent would be less conservative. Since this requested change for CCNPP-1 was

found acceptable (June 14, 1979, Safety Evaluation) and is fully applicable for Unit No. 2, we conclude that the same change is acceptable for CCNPP-2.

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.

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: December 11, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-318BALTIMORE GAS AND ELECTRIC COMPANYNOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. DPR-69 issued to Baltimore Gas & Electric Company, which revised 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 its date of issuance.

The amendment revises the Appendix A Technical Specifications of the facility to increase the measurement/calculational uncertainties for peaking factors F_r and F_q from 5.1 and 5.8 percent to 6.0 and 7.0 percent, respectively.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

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For further details with respect to this action, see (1) the application for amendment dated August 27, 1979, as supplemented October 1, 1979, (2) Amendment No. 24 to License No. DPR-69, 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 11th day of December 1979.

FOR THE NUCLEAR REGULATORY COMMISSION



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



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

Docket No. 50-318

DISTRIBUTION:
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Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: CALVERT CLIFFS UNIT NO. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s); Time for Submission of Views on Antitrust Matters.
- Notice of Availability of Applicant's Environmental Report.
- Notice of Proposed Issuance of Amendment to Facility Operating License.
- Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- Notice of Availability of NRC Draft/Final Environmental Statement.
- Notice of Limited Work Authorization.
- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).
- Notice of Issuance of Facility Operating License(s) or Amendment(s).
- Other: Amendment No. 24
Referenced documents have been provided - PDR

Division of Operating Reactors, ORB4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#4: DOR				
SURNAME →	RIngram/cb				
DATE →	12/13/79				

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-318BALTIMORE GAS AND ELECTRIC COMPANYNOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.24 to Facility Operating License No. DPR-69 issued to Baltimore Gas & Electric Company, which revised 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 its date of issuance.

The amendment revises the Appendix A Technical Specifications of the facility to increase the measurement/calculational uncertainties for peaking factors F_p and F_q from 5.1 and 5.8 percent to 6.0 and 7.0 percent, respectively.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated August 27, 1979, as supplemented October 1, 1979, (2) Amendment No. 24 to License No. DPR-69, 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 11th day of December 1979.

FOR THE NUCLEAR REGULATORY COMMISSION



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