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

December 1, 1976

Docket Nos. 50-317
50-318

Baltimore Gas and Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President - Supply
Gas and Electric Building
Charles Center
Baltimore, Maryland 21203

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 18 and 3 to Facility Operating License Nos. DPR-53 and DPR-69 for the Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, respectively. The amendments consist of changes in the Technical Specifications in accordance with your application dated September 23, 1976.

These amendments modify the Technical Specifications to eliminate the operability and surveillance requirements of the facilities' Turbine Runback feature of the Engineered Safety Feature Actuation System. This feature reduces the turbine output to 70% when a control rod is dropped.

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

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 18 to License No. DPR-53
2. Amendment No. 3 to License No. DPR-69
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 23, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace pages 3.14-1, 3.14-3, 3.14-4, and 4.1-11 of the Appendix A portion of the Technical Specifications with the enclosed revised pages 3.14-1, 3.14-3, 3.14-4, and 4.1-11. Changed areas on the revised pages are indicated by a marginal line.

Note: The revised pages are printed on one side only. Therefore, the existing pages in the Technical Specifications should not be destroyed if the reverse side is an unrevised page.

3.14 ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENTATION SETTINGS

Applicability: Applies to the engineered safety features system initiation instrumentation settings.

Objective: To provide for automatic initiation of the engineered safety features in the event principal process variable limits are exceeded.

Specification: The engineered safety features system initiation instrumentation setting limits and permissible bypasses shall be as stated in Table 3.14.1.

Basis:

A. Containment High Pressure

The basis for the 4(+ 0.75, - 0.25) psig set point for the high-pressure signal is to establish a setting which would be reached immediately in the event of a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.(1)

B. Pressurizer Low Pressure

The pressurizer low-pressure signal, in conjunction with the containment high-pressure signal, provides a diverse input to the safety injection actuation signal. The 1600-psia setting includes an uncertainty of 22 psi and is the setting used in the FSAR Section 14 analysis.(2)

C. Steam Generator Low Pressure

A signal is provided, upon sensing low pressure in a steam generator, to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a 22-psi uncertainty; therefore, a value of 478 psia was used in the FSAR Section 14 analysis.(3)

D. Refueling Water Tank Low-Level Switches

Level switches are provided on the refueling water tank to actuate the valves in the injection pump suction lines

F. (Deleted)

References:

1. FSAR, Section 14.16
2. FSAR, Section 14.14
3. FSAR, Section 14.12
4. FSAR, Section 7.4.6

Table 3.14.1

Engineered Safety Features and Turbine Runback Systems Initiation Instrument Setting Limits

Functional Parameter	Channel	Setting Limit
1. Containment High Pressure	a. Safety Injection	4 (+ 0.75, - 0.25) psig
	b. Containment Spray	
	c. Containment Isolation	
	d. Containment Air Cooler LOCI Mode	
	e. Containment Iodine Removal System	
2. Pressurizer Low Pressure	Safety Injection	1600 (± 22) psia (a)
3. Steam Generator Low Pressure	Steam Line Isolation	500 (± 22) psia (b)
4. Refueling Water Tank Low Level	Recirculation Actuation	30 (+ 0, - 6) inches above tank bottom
5. Containment High Radiation	Containment Purge Isolation	200 (± 20) mr/hr
6. (Deleted)		
7. (Deleted)		

(a) May be bypassed below 1700 psia and is automatically reinstated above 1700 psia
(b) May be bypassed below 550 psia and is automatically reinstated above 550 psia

TABLE 4.1.3 (cont.)

Equipment Description	Surveillance Function	Frequency	Surveillance Method
5. (Cont.)	b. Test	R	b. Verify proper operation of all CEA drive control system interlocks.
6. (Deleted)			
7. Calorimetric and Feedwater Regulating Instrumentation	a. Calibration	R	a. Apply known d/p to feedwater flow sensors.
	b. Calibration	R	b. Apply known pressure to steam pressure sensors.
	c. Calibration	R	c. Known resistance substituted for feedwater temperature RTD. Calibration of RTD's.
	d. Calibration	R	d. Apply known d/p to level sensors.
8. Condensate Storage Tank Level Instrument	a. Check	R*	a. Apply known d/p to level sensors.
9. Control Room Ventilation	a. Test	R*	a. Verify proper damper operation during simulated high radiation conditions.
	b. Test	R*	b. Verify that the control room is under a positive pressure.
10. Pressurizer Level Instrument (Control Channels)	a. Check	S	a. Comparison of two separate level indications.
	b. Calibration	R	b. Apply known d/p to level sensors.
11. Pressurizer Level Instrument (Cold Calibrate)	a. Calibration	R	a. Apply known d/p to level sensors.

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. 3
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 September 23, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 3

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

3/4 3-13
3/4 3-16
3/4 3-18
3/4 3-23
3/4 3-24

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. CONTAINMENT PURGE VALVES ISOLATION					
a. Manual (Purge Valve Control Switches)	2/Penetration	1/Penetration	2/Penetration	1, 2, 3, 4	6
b. Containment Radiation - High Area Monitor	4	2	3	6	8
7. LOSS OF POWER 4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CVCS ISOLATION					
a. Manual (CVCS Isolation Valve Control Switches)	1/Valve	1/Valve	1/Valve	1, 2, 3, 4	6
b. West Penetration Room/Letdown Heat Exchanger Room Pressure - High	4	2	3	1, 2, 3, 4	7*

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1700 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1700 psia.

(c) Trip function may be bypassed in this MODE below 600 psia; bypass shall be automatically removed at or above 600 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

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

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

TABLE 3.3-3 (Continued)

ACTION 8 - With less than the Minimum Channels OPERABLE, operation may continue provide the containment purge valves are maintained closed.

ACTION 11 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
c. Pressurizer Pressure - Low	$\geq$ 1578 psia	$\geq$ 1578 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
3. CONTAINMENT ISOLATION (CIS)		
a. Manual CIS (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
4. MAIN STEAM LINE ISOLATION		
a. Manual (MSIV Hand Switches and Feed Head Isolation Hand Switches)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 478 psia	$\geq$ 478 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	≥ 24 inches above tank bottom	≥ 24 inches above tank bottom
6. CONTAINMENT PURGE VALVES ISOLATION		
a. Manual (Purge Valve Control Switches)	Not Applicable	Not Applicable
b. Containment Radiation - High		
Area Monitor	≤ 220 mr/hr	≤ 220 mr/hr
7. LOSS OF POWER		
4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	≥ 2450 volts	≥ 2450 volts

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Tank - Low	N.A.	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
6. CONTAINMENT PURGE VALVES ISOLATION				
a. Manual (Purge Valve Control Switches)	N.A.	N.A.	R	N.A.
b. Containment Radiation - High Area Monitor	S	R	M	6
7. LOSS OF POWER 4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	N.A.	R	M	1, 2, 3
9. CVCS ISOLATION West Penetration Room/ Letdown Heat Enchanger Room Pressure - High	N.A.	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The logic circuits shall be tested manually at least once per 31 days.

(3) SIAS logic circuits A-5, B-5, A-10 and B-10 may be exempted from testing during operation; however, these logic circuits shall be tested at least once per 18 months during shutdown.

(4) CIS logic circuits A-5 and B-5 may be exempted from testing during operation; however, these logic circuits shall be tested at least once per 18 months during shutdown.

(5) SGIS logic circuits A-1 and B-1 may be exempted from testing during operation; however, these logic circuits shall be tested at least once per 18 months during shutdown.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 18 AND 3 TO LICENSE NOS. DPR-53 AND DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

INTRODUCTION

By application for license amendment dated September 23, 1976, Baltimore Gas and Electric Company (BG&E) requested a change to the Technical Specifications for Calvert Cliffs Unit Nos. 1 and 2. The proposed change would eliminate the required use of the Turbine Runback feature of the Engineered Safety Feature Actuation System.

In reviewing the application for license amendment dated September 23, 1976, we found it necessary to modify the proposed Technical Specifications to meet our requirements. The changes in the proposed Technical Specifications were made with the concurrence of BG&E representatives.

DISCUSSION

The Turbine Runback feature is part of a safety system designed to mitigate the consequences of a control element assembly (CEA) drop incident. The CEA incident, analyzed in Section 14.4 of the Calvert Cliffs Nuclear Power Plant Final Safety Analysis Report (FSAR), assumes that the single most reactive CEA accidentally drops to its fully inserted position. The reactor, which was assumed to be operating at full power (plus 2% uncertainty), experiences a distorted radial power distribution, a rapid power reduction and an average reactor coolant temperature decrease. If the dropped CEA were to go undetected the auto-sequential mode of the reactor control system* would attempt

*Under auto-sequential control, the reactor control system automatically adjusts the control rods so that reactor power matches turbine demand via the preprogrammed linear relationship between average reactor coolant temperature, T avg., and turbine power demand. Thus, a decrease in reactor power, with consequent decrease in T avg., will cause the reactor to automatically withdraw CEA's, in a predetermined order, to increase T avg. to match turbine power. We refer to this as "the reactor following the turbine".

to return the reactor to full power by withdrawing available CEA's since the turbine demand would remain at full power. The coincidence of full power conditions and a distorted power distribution would effect the departure from nucleate boiling (DNB) design limit.

For the existing design and at analyzed operating conditions, a dropped CEA would not result in the violation of the DNB design limit since (1) a dropped CEA would be detected, (2) CEA withdrawal would be inhibited, and (3) the turbine would be automatically "runback" (power demand decreased) to 70% power. Although a skewed power distribution would still result, its occurrence at the decreased power level prevents violation of the DNB design limit.

Actual operating experience with the turbine runback feature has proven unsatisfactory. Spurious dropped rod signals have resulted in unnecessary turbine runbacks during normal full power operation. The resulting transients have curtailed power generation and have caused the nuclear steam supply system to undergo unnecessary thermal and pressure stresses. Therefore, BG&E has proposed that the required use of the Turbine Runback feature be deleted from the Technical Specifications. The proposed changes are similar to changes previously approved for Millstone Unit No. 2 and other reactors designed by Combustion Engineering.

EVALUATION

In support of their application for license amendment dated September 23, 1976, BG&E presented a new CEA drop incident analysis which updates the analysis contained in Section 14.4 of the FSAR. The old analysis presented in the FSAR indicated that with a turbine runback, the accidental drop of the most reactive CEA yields a minimum DNBR of 1.63. BG&E's revised analysis assumed that the reactor was initially operating under steady state conditions which yield the closest approach to the fuel design limits. In BG&E's analysis of the subsequent dropped CEA transient, which we evaluated and found to be acceptable, the minimum DNBR was calculated to be 1.34 as compared with the DNBR design limit of 1.30. It is significant that this transient yielded acceptable results (no violation of the DNB design limit) even though the Turbine Runback feature was not utilized.

Since the previous analysis presented in Section 14.4 of the FSAR did not consider a dropped CEA without a turbine runback, it is not possible to compare the decrease in DNBR between the two analyses. However, based upon the acceptable results presented in support of the application for license amendment dated September 23, 1976, it is appropriate to delete the operability and surveillance requirements for the Turbine Runback instrumentation channels, contained in Technical Specifications 3.14F and 4-1 for Unit No. 1 and Technical Specifications 3.3.2.1 and 4.3.2.1.1 for Unit No. 2.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendments do 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 amendments involve 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 these amendments.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 1, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-317 AND 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 18 and 3 to Facility Operating License Nos. DPR-53 and DPR-69 (respectively), issued to Baltimore Gas and Electric Company (the licensee), which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2 (the facilities) located in Calvert County, Maryland. The amendments are effective as of their date of issuance.

The amendments modified the Technical Specifications for the facilities to eliminate the operability and surveillance requirements of the Turbine Runback feature of the Engineered Safety Feature Actuation System. The Turbine Runback feature reduces the turbine output to 70% when a control rod is dropped.

The application for the amendments 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 amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the Federal Register on October 26, 1976 (41 F.R. 46915). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the application for amendment dated September 23, 1976, (2) Amendment No. 18 to License No. DPR-53 and Amendment No. 3 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 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 1st day of December, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors