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

December 23, 1975

Docket No. 50-317

Posted Amn-12 DPR-53 Ch-11

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

Gentlemen:

We have reviewed your letters of November 18, December 8, and December 22, 1975, relating to your request for an exemption from the provisions of Appendix J of 10 CFR 50 and the requirements of 4.5.C.5 of the Technical Specifications for Calvert Cliffs Unit 1. Our letter of December 4, 1975, confirmed the telephone conversation on November 28, 1975, in which I granted a 30-day exemption. The limited exemption was to allow continued plant operation pending completion of our assessment of your November 18 request and for you to provide additional justification for the six-month exemption.

By letter dated December 8, 1975, you provided additional information and requested an extension until May 30, 1976, of the exemption which we granted until December 30, 1975. You stated that the request for extension is in the public interest and that operating under the requested exemption will not endanger life or property or the common defense and security or the health and safety of the public. Our review confirms your conclusion.

Therefore, an extension is granted to the exemption. Calvert Cliffs Unit 1 operation may be continued until the next shutdown when the plant is cooled down but not later than May 30, 1976, without completing the Type C tests on certain containment isolation valves.

The appropriate modification of your Technical Specifications to reflect this exemption is enclosed as Amendment No. 12 to License No. DPR-53 with Change No. 11 to the Technical Specifications. A copy of the Notice of this amendment which has been submitted to the Federal Register for publication and a copy of our Safety Evaluation are enclosed.

Sincerely,

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Enclosures and cc:
See next page

Enclosures:
1. Amendment No. 12
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
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Calvert County Library
Prince Frederick, Maryland 20678

Mr. Bernard Fowler
President, Board of County
Commissioners
Prince Frederick, Maryland 20678

(w/4 cys of encls. to this letter
and 1 cy of BG&E's filings dtd.
12/8/75 and 12/22/75:
Mr. Warren D. Hodges, Director
Department of State Planning
301 West Preston Street
Baltimore, Maryland 21201

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

(EXEMPTION FROM APPENDIX J)

Amendment No. 12
License No. DPR-53

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

   A. The applications for amendment by Baltimore Gas and Electric Company (the licensee) dated November 18, 1975 and December 8, 1975, are acceptable as requests for exemption from Section III.D.3 of Appendix J of 10 CFR Part 50 of the Commission's rules and regulations set forth in 10 CFR Chapter I, and comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act);

   B. Granting the temporary exemption from Appendix J of 10 CFR Part 50 is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest;

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

   D. 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;

   E. 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

   F. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C(2) of Facility License No. DPR-53 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 11."

3. This license amendment is effective as of December 30, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Attachment:
Change No. 11 to the
Technical Specifications

Date of Issuance: December 23, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 12

CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace pages 4.5-9, 4.5-25, 4.5-26 and 4.5-27 of the Appendix A portion of the Technical Specifications with the attached revised pages bearing the same number. The changed areas on the pages are shown by a marginal line.

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

5. Periodic Retest Schedule

Type C tests shall be performed during each reactor shutdown for refueling, but in no case at intervals greater than 2 years.* 11

D. Special Testing Requirements

Any major modification, replacement of a component which is part of the containment boundary, or resealing of a seal-welded door, performed after the preoperational leakage rate test, shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage rate from this test shall be included in the report to the Commission required by E.2, below. The acceptance criterion of A.5.b, B.4, and C.4, as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test, do not require a separate test.

E. Inspection and Reporting of Results

1. Containment Inspection

A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leaktightness. If there is evidence of structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, non-destructive examinations, and tests as specified in the applicable code specified in 10 CFR 50, 55a. Such structural deterioration and corrective actions taken shall be reported as part of the test report submitted in accordance with E.2, below.

* An exemption granted on December 23, 1975, allows the remaining Type C tests (asterisks on Table 4.5.2) to be performed during the next plant shutdown and cooldown following issuance of this change but no later than May 30, 1976. 11

TABLE 4.5.2

LOCAL LEAK TESTING OF

CONTAINMENT ISOLATION VALVES

Pene. No.	P&ID (FSAR Figure)	Isolation Valve Identification No.	Location WRT Containment	Type of Valve	Test Type (a)
1A*	M-66 (9-10)	CV-5465	Inside	Globe	3
		CV-5466	Inside	Globe	3
		CV-5467	Inside	Globe	3
		CV-5464	Outside	Globe	1
1B*	M-78 (11-2)	CV-2180	Outside	Globe	3
		CV-2181	Outside	Globe	1
1C*	M-73 (9-3)	CV-506	Inside	Globe	3
		CV-505	Outside	Globe	1
1D	M-463 (9-11A)	Sv-6529	Outside	Globe	1
2A*	M-73 (9-3)	CV-515	Inside	Globe	3
		CV-516	Inside	Globe	1
		7M3-1	Outside	Gate	1
		7M3-1	Outside	Gate	1
2B*	M-73 (6-1)	CV-517	Inside	Globe	5 (b)
		CV-518	Inside	Globe	5 (b)
		CV-519	Inside	Globe	5 (b)
		SP-210M3-2	Inside	Check	1
		210M3-2	Outside	Check	1
7A	M-65 (9-20A)	Blind Flange	Inside	---	2
		19-1	Outside	Gate	1
7B	M-65 (9-20A)	Blind Flange	Inside	---	2
		19-1	Outside	Gate	1
8 *	M-76 (5-10)	MOV-5462	Outside	Gate	1
		MOV-5463	Outside	Gate	1
9 *	M-74 (6-1)	238M3-1	Inside	Check	1
		238M3-2	Outside	Check	1
10 *	M-74 (6-1)	238M3-1	Inside	Check	1
		238M3-2	Outside	Check	1
13	M-65 (9-20)	CV-1412	Inside	Butterfly	4 (c)
		CV-1413	Outside	Butterfly	1
14	M-65 (9-20)	CV-1410	Inside	Butterfly	4 (c)
		CV-1411	Outside	Butterfly	1

NOTES:

(a) See Legend at end of table.

(b) Penetration in use during accident conditions

(c) Accident pressure seats and test pressure unseats these valves

* See asterisk note in Specification 4.5.C.5.

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TABLE 4.5.2 (Cont.)

Pene. No.	P&ID (FSAR Figure)	Isolation Valve Identification No.	Location WRT Containment	Type of Valve	Test Type (a)
15	M-98 (11-4)	CV-5291	Inside	Globe	3
		CV-5292	Outside	Globe	1
16*	M-51 (9-6)	CV-3832	Outside	Butterfly	4 (d)
18*	M-51 (9-6)	CV-3833	Outside	Butterfly	4 (d)
19A*	M-53 (9-23)	223-1	Inside	Check	1
		MOV-2080	Outside	Gate	1
19B	M-479 (9-23A)	*19-2	Inside	Gate	1
		130-1	Outside	Globe	1
20A	M-68 (5-10)	223-1	Outside	Check	1
		*CV-612	Inside	Gate	2
		*CV-622	Inside	Gate	2
		*CV-632	Inside	Gate	2
		*CV-642	Inside	Gate	2
20B*	M-68 (5-10)	223-1	Outside	Check	1
		223-2	Inside	Check	1
20C	M-68 (5-10)	223-1	Outside	Check	1
		*223-2	Inside	Check	1
23	M-77 (11-1)	CV-4260	Outside	Globe	1
24*	M-463 (9-11A)	SV-6531	Outside	Globe	1
37	M-479 (9-23A)	*29-1	Inside	Gate	1
		142-1	Outside	Globe	1
38	M-72 (4-1	CV-5460	Outside	Globe	1
39	M-74 (6-1)	130M3-1	Outside	Globe	5 (d)
		130M3-2	Outside	Globe	1
41*	M-74 (6-1)	MOV-652	Inside	Gate	6 (e)
		MOV-651	Outside	Gate	1
44	M-56 (9-22)	*238-1	Inside	Check	1
		238-1	Outside	Check	1
		MOV-6200	Outside	Gate	1
47A	M-463 (9-11A)	SV-6540A	Inside	Globe	1
		SV-6507A	Outside	Globe	1
47B	M-463 (9-11A)	SV-6540E	Inside	Globe	1
		SV-6507F	Outside	Globe	1
47C	M-463	SV-6540F	Inside	Globe	1
		SV-6507F	Outside	Globe	1
47D	M-463 (9-11A)	*SV-6540F	Inside	Globe	1
		SV-6507G	Outside	Globe	1

NOTES:
(d) Closed system
(e) Two valves, normally closed, connected to reactor coolant loop
* See asterisk note in Specification 4.5.C.5.

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TABLE 4.5.2 (Cont.)

Pene. No.	P&ID (FSAR Figure)	Isolation Valve Identification No.	Location WRT Containment	Type of Valve	Test Type (a)
48A	M-65 (9-20A)	MOV-6900	Inside	Gate	6 (f)
		MOV-6901	Outside	Gate	1
48B	M-65 (9-20A)	*238-1	Inside	Check	1
		MOV-6903	Outside	Gate	1
49A	M-463 (9-11A)	SV-6540B	Inside	Globe	3
		SV-6507B	Outside	Globe	1
49B	M-463 (9-11A)	SV-6504C	Inside	Globe	3
		SV-6507D	Outside	Globe	1
49C	M-463 (9-11A)	SV-6540D	Inside	Globe	3
		SV-6507D	Outside	Globe	1
50	M-65 (9-20A)	Blind Flange	Inside	---	2
		Blind Flange	Outside	---	1
59	M-58 (9-7)	29M3-1	Inside	Gate	6 (f)
		29M3-1	Outside	Gate	1
60	M-77 (11-1)	130-1	Inside	Globe	3
		19-2	Outside	Globe	1
61	M-58 (9-7)	76Y-1	Inside	Gate	6 (f)
		293M-1	Inside	Gate	6 (f)
		293M-1	Inside	Gate	6 (f)
		293M-1	Outside	Gate	1
62*	M-71 (5-10)	MOV-6579	Outside	Gate	1
64	M-71 (5-10)	238-1	Outside	Check	1

NOTES:

(f) Two valves normally closed.
* See asterisk note in Specification 4.5.C.5.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO FACILITY LICENSE NO. DPR-53

CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1

DOCKET NO. 50-317

INTRODUCTION

By letter dated November 18, 1975, the Baltimore Gas and Electric Company (BG&E) requested an amendment to Facility License No. DPR-53 for the Calvert Cliffs Nuclear Plant Unit 1. The request was for a change to the Technical Specifications extending the interval for testing some Type C containment penetration valves from two years to two years and six months. However, we only granted a 30-day exemption during which BG&E could present additional information in support of their six-month request or to plan for shutdown, cooldown and testing. BG&E provided supporting information on December 8 and December 22 and requested an extension of the exemption until May 30, 1976, based on 10 CFR Part 50, Section 50.12.

DISCUSSION

The Type C tests were initially performed in November 1973 for Calvert Cliffs Unit 1 prior to fuel loading. The plant electrical output was connected to the grid in January 1975 about five months after the initial fuel loading. The initial core will operate for about 21 months. Therefore, the first retest interval[1/] of less than two years after the initial tests does not coincide with the first refueling interval scheduled in December 1976, about 36 months after the initial Type C testing. The current regulation and Technical Specifications do not take into account that the first Type C test occurs prior to the initial fuel loading; therefore, the first retest interval is longer than subsequent retest intervals. In order to comply with Appendix J, an additional reactor shutdown, cooldown and depressurization cycle would be required at significant costs to BG&E and its customers.

1/ Section III.D.3 of Appendix J of 10 CFR Part 50 and Section 4.5.C.5 of Calvert Cliffs Unit 1 Technical Specifications

About 50% of the containment isolation valves are still to be tested. On December 22 BG&E advised us that the remaining tests will be completed the next time the reactor is shutdown and cooled down but not later than May 30, 1976.

EVALUATION

The Type C tests of containment isolation valves are performed periodically to assure that a combined leakage rate of all penetrations and valves (Type B and Type C tests) are less than 60% of the total containment allowable leakage. The required interval was selected to coincide with the refueling intervals which are normally not more than two years after the first refueling. Since Calvert Cliffs' first refueling interval will be about 36 months from the time of the initial tests, a shutdown and cooldown for only the Type C tests would be an unnecessary plant thermal cycle. Such thermal cycles are limited by design to minimize the effects of thermal and mechanical stresses. On this basis it is desirable to combine the Type C valve leakage tests with some other scheduled shutdown event, such as refueling.

BG&E has satisfactorily tested 50% of the containment isolation valves, and has analyzed the test results. We previously concluded in our evaluation of December 4, 1975, that the actual measured leakage (10 percent of allowable, including Type B tests) demonstrates only minor valve degradation. Valve seat degradation is not a rapid phenomenon. Thus, the 90% leakage margin for the remaining valves provides a degree of confidence that the technical specification leakage limits would not be exceeded due to valve degradation by extending the two year test interval by six months. Furthermore, BG&E predicted total leakage by extrapolating the actual test results. The predicted total Type B and Type C leakage is between 33% and 45% of allowable. We have reviewed the test results and the techniques used by BG&E to predict valve leakage. The actual measured leakage demonstrates only minor valve degradation from the test results of November 1973. The leakage prediction techniques, although not proven, indicate that acceptable results would occur if valve seat degradation on each type of valve remains constant for each type of valve.

BG&E estimates that five and one-half days of plant downtime would be required to perform the remaining Type C testing. Included is about three and one-half days for tests and two days to shutdown and to return to power. Experience has shown that other problems appear during shutdown periods which normally delay startup by 24 hours or more. Therefore, the five and one-half day outage estimate is conservative.

During a five and one-half day outage BG&E stated that fossil fueled units will operate to replace the lost electrical generation of approximately 108,400,000 KWH from Calvert Cliffs. Approximately 192,500 barrels of oil at a cost of about 2.2 million dollars could be used to replace the nuclear generated power. Between 1.5 and 1.8 million dollars of the added

costs would be borne by the power consumer. Such costs are not in the public interest. On the basis of the foregoing discussion, the extension of the limited exemption granted by our letter of December 4 from 30 days to six months is acceptable.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power levels and will not result in any significant negative 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 statement, negative declaration, or 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) granting the exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest, (2) 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, and (3) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Date: December 23, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-317

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 12 to Facility Operating License No. DPR-53, issued to Baltimore Gas and Electric Company (the licensee), which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant Unit 1 (the facility) located in Calvert County, Maryland. The amendment is effective as of December 30, 1975.

The amendment incorporates an exemption from the requirements of Section III.D.3 of Appendix J of 10 CFR Part 50. It changes the Technical Specifications for the facility to extend the first retest interval of approximately 50% of the containment isolation valves until the next plant shutdown and cooldown following issuance of this amendment, but no later than May 30, 1976.

The applications comply 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. The Commission also concluded that the granting of the exemption from the requirements of Section III.D.3 of Appendix J for the above-referenced test is authorized by law and will not endanger life or property or the common defense and security and is

otherwise in the public interest. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the applications for amendment dated November 18, and December 8 and 22, 1975, (2) the letter from the Director of the Nuclear Reactor Regulation to Baltimore Gas and Electric Company dated December 4, 1975, and the letter issued concurrently with this Notice, (3) Amendment No. 12 to License No. DPR-53, with Change No. 11, and (4) 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), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 23rd day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard D. Silver

Richard D. Silver, Acting Chief
Operating Reactors Branch #2
Division of Reactor Licensing