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Docket No. 50-317
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 Nos. 75 and 56 to Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated August 2, 1982.

These amendments correct typographical errors involving inservice inspection and valve testing; allow start up of the reactors with one or more inoperable but isolated power operated relief valves; incorporate numerical values of containment leakage rates and incorporate by reference the latest applicable version of Regulatory Guide 1.33.

A copy of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Monte Conner for

David H. Jaffe, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 75 to DPR-53
2. Amendment No. 56 to DPR-69
3. Safety Evaluation
4. Notice of Issuance

cc: See next page

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|---|---|---|---|---|---|---|---|
| SURNAME | PMKreutzer | DJaffe | GHolahan | RAClark | GCLainas | Woodhead | |
| DATE | 9/1/82 | 9/1/82 | 9/7/82 | 9/8/82 | 9/9/82 | 9/14/82 | |

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

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PMKreutzer

Docket No. 50-317/50-318

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: BALTIMORE GAS AND ELECTRIC COMPANY, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 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: Referenced documents have been provided PDR.
Amendments Nos. 75 and 56.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE→ | ORB#3:DL | | | | | |
| SURNAME→ | PMKreutzer/pn | | | | | |
| DATE→ | 9/21/82 | | | | | |

NRC FORM 102 7—79

Baltimore Gas and Electric Company

cc:
James A. Biddison, Jr.
General Counsel
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Baltimore, MD 21203

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Mr. R. C. L. Olson, Principal Engineer
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Mr. Leon B. Russell
Plant Superintendent
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

Bechtel Power Corporation
Attn: Mr. J. C. Ventura
Calvert Cliffs Project Engineer
15740 Shady Grove Road
Gaithersburg, MD 20760

Combustion Engineering, Inc.
Attn: Mr. P. W. Kruse, Manager
Engineering Services
P. O. Box 500
Windsor, CT 06095

Public Document Room
Calvert County Library
Prince Frederick, MD 20678

Director, Department of State Planning
301 West Preston Street
Baltimore, MD 21201

Mr. R. M. Douglass, Manager
Quality Assurance Department
Fort Smallwood Road Complex
P. O. Box 1475
Baltimore, MD 21203

Mr. S. M. Davis, General Supervisor
Operations Quality Assurance
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

Ms. Mary Harrison, President
Calvert County Board of County Commissioners
Prince Frederick, MD 20768

U. S. Environmental Protection Agency
Region III Office
Attn: Regional Radiation Representative
Curtis Building (Sixth Floor)
Sixth and Walnut Streets
Philadelphia, PA 19106

Mr. Ralph E. Architzel
Resident Reactor Inspector
NRC Inspection and Enforcement
P. O. Bos 437
Lusby, MD 20657

Mr. Charles B. Brinkman
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Mr. R. E. Denton, General Supervisor
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Lusby, MD 20657

cc w/enclosure(s) and incoming
dated:

Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
Department of Natural Resources
Tawes State Office Building
Annapolis, MD 21204

Regional Administrator
Nuclear Regulatory Commission, Region I
Office of Executive Director for Operations
631 Park Avenue
King of Prussia, Pennsylvania 19406

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated August 2, 1982, 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.

DESIGNATED ORIGINAL

Certified By Patricia J. Noonan

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 75, 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 A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
  Technical Specifications

Date of Issuance: September 20, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. 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 4-4
3/4 4-28
3/4 5-5a
3/4 6-2
3/4 6-3
3/4 6-4
3/4 6-18
6-13

LIMITING CONDITION FOR OPERATION

3.4.2.1 The following pressurizer code safety valves shall be OPERABLE:

| Valve | Lift Settings (±1%) |
|---|---|
| RC-200 | 2500 psia |
| RC-201 | 2565 psia |

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours..

3.4.2.2 At least one of the above pressurizer code safety valves shall be OPERABLE:*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* Both valves may be removed in MODE 5 provided at least one valve is replaced by a spool piece which allows the pressurizer to relieve directly to the quench tank.

## RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.3 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more block valve(s) closed and power removed from the block valve(s) to satisfy a. or b. above, the provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, in accordance with Table 4.3-1, Item 4.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed and power removed to meet Specification 3.4.3 a. or b.

## 3/4.4.10 STRUCTURAL INTEGRITY

### ASME CODE CLASS 1, 2 AND 3 COMPONENTS

### LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1975, for Class 2 components.

a. System integrity and baseline data shall be established by performing a 100% volumetric examination of each weld prior to exceeding 18 months of operation.

b. Each weld shall be examined in accordance with the above ASME Code requirements, except that 100% of the welds shall be examined, cumulatively, during each 10 year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back to the first 10 year inspection program. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4.0 ± 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 ± 0.1 liters of 77 ± 10°F borated water from the RWT, the pH of the mixed solution is raised to ≥ 6 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

g. By verifying the correct position of each electrical position stop for the following Emergency Core Cooling System throttle valves:

1. During each performance of valve cycling required by Specification 4.0.5 by observation of valve position on the control boards.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 4 hours following completion of maintenance on the valve or its operator by measurement of stem travel when the ECCS subsystems are required to be OPERABLE.

HPSI SYSTEM

| Valve Number | Valve Number |
|---|---|
| MOV-616 | MOV-617 |
| MOV-626 | MOV-627 |
| MOV-636 | MOV-637 |
| MOV-646 | MOV-647 |

h. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates:

HPSI System
Single Pump

170 ± 5 gpm to each injection leg.

## 3/4.6.1 PRIMARY CONTAINMENT

## CONTAINMENT INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.1, and

2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

## LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$ (346,000 SCCM), 0.20 percent by weight of the containment air per 24 hours at $P_a$, 50 psig, or

2. $\leq L_t$ (61,600 SCCM), 0.058 percent by weight of the containment air per 24 hours at a reduced pressure of $P_t$, 25 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ (207,600 SCCM), for all penetrations and valves subject to Type B and C tests when pressurized to $P_a$.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 $L_a$ (259,500 SCCM) or 0.75 $L_t$ (46,200 SCCM), as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 $L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

## SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either $P_a$ (50 psig) or at $P_t$ (25 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 $L_a$ (259,500 SCCM) or .75 $L_t$ (46,200 SCCM), the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 $L_a$ (259,500 SCCM) or .75 $L_t$ (46,200 SCCM), a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 $L_a$ (259,500 SCCM) or .75 $L_t$ (46,200 SCCM) at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 $L_a$ (86,500 SCCM) or 0.25 $L_t$ (15,400 SCCM).

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at $P_a$ (50 psig) or $P_t$ (25 psig).

d. Type B and C tests shall be conducted with gas at $P_a$ (50 psig) at intervals no greater than 24 months except for tests involving air locks.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

## CONTAINMENT AIR LOCKS

### LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ (17,300 SCCM) at $P_a$, 50 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With an air lock inoperable, except as a result of an inoperable door gasket, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an air lock inoperable due to an inoperable door gasket:

1. Maintain the remaining door of the affected air lock closed and sealed, and

2. Restore the air lock to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a.* After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours by verifying that the seal leakage is $< 0.0002\ L_a$ (69.2 SCCM) as determined by precision flow measurement when the volume between the door seals is pressurized to a constant pressure of 15 psig,

* Exemption to Appendix "J" of 10 CFR 50.

## 3/4.6.4 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.4.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.4.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on each containment isolation Channel A or Channel B test signal, each required isolation valve actuates to its isolation position.

b. Verifying that on each Containment Radiation-High Test Channel A or Channel B test signal, both required containment purge valves actuate to their isolation position.

c. Verifying that on each Safety Injection Actuation Channel A or Channel B test signal, each required isolation valve actuates to its isolation position.

4.6.4.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Technical Specification 4.0.5.

4.6.4.1.4 Containment purge isolation valves shall be demonstrated OPERABLE at least once every 6 months by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Technical Specification 4.6.1.2.d for all other Type B or C penetrations, the combined leakage rate is less than or equal to 0.60 $L_a$ (207,600 SCCM). The leakage rate for the containment purge isolation valves shall also be compared to the previously measured leakage rate to detect excessive valve degradation.

## 6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager - Nuclear Power Department and the OSSRC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the OSSRC and the Manager - Nuclear Power Department, within 14 days of the violation.

## 6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the POSRC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the originial procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the POSRC and approved by the Plant Superintendent within 14 days of implementation.

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

### STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas & Electric Company (the licensee) dated August 2, 1982, 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.

DESIGNATED ORIGINAL

Certified By Patricia J. Noonan

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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56, 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 A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
  Technical Specifications

Date of Issuance: September 20, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 56

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the follwing pages of the Appendix A Technical Specifications with the enclosed pages as indicated. 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 4-4
3/4 4-29
3/4 5-5a
3/4 6-2
3/4 6-3
3/4 6-4
3/4 6-18
6-13

REACTOR COOLANT SYSTEM

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The following pressurizer code safety valves shall be OPERABLE:

| Valve | Lift Settings (±1%) |
|---|---|
| RC-200 | 2500 psia |
| RC-201 | 2565 psia |

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

3.4.2.2 At least one of the above pressurizer code safety valves shall be OPERABLE:*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* Both valves may be removed in MODE 5 provided at least one valve is replaced by a spool piece which allows the pressurizer to relieve directly to the quench tank.

## RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.3 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more block valve(s) closed and power removed from the block valve(s) to satisfy a. or b. above, the provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, in accordance with Table 4.3-1, Item 4.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed and power removed to meet Specification 3.4.3 a. or b.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1975, for Class 2 components.

a. System integrity and baseline data shall be established by performing a 100% volumetric examination of each weld prior to exceeding 18 months of operation.

b. Each weld shall be examined in accordance with the above ASME Code requirements, except that 100% of the welds shall be examined, cumulatively, during each 10 year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back to the first 10 year inspection program. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected.

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD and/or SA exceeding their applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours,

   2. SA shall be measured at least once per 24 hours and shall be processed at least once per 7 days, and

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

b. With the APD and/or SA exceeding their applicable Action Levels, measure and process APD and SA data within 24 hours to determine if the core barrel motion is exceeding its limits. With the core barrel motion exceeding its limits, reduce the core barrel motion to within its Action Levels within the next 24 hours or be in HOT STANDBY within the following 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of $4.0 \pm 0.1$ grams of TSP from a TSP storage basket is submerged, without agitation, in $3.5 \pm 0.1$ liters of $77 \pm 10^{\circ}F$ borated water from the RWT, the pH of the mixed solution is raised to $\geq 6$ within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

g. By verifying the correct position of each electrical position stop for the following Emergency Core Cooling System throttle valves:

1. During each performance of valve cycling required by Specification 4.0.5 by observation of valve position on the control boards.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 4 hours following completion of maintenance on the valve or its operator by measurement of stem travel when the ECCS subsystems are required to be OPERABLE.

HPSI SYSTEM

| Valve Number | Valve Number |
|---|---|
| MOV-616 | MOV-617 |
| MOV-626 | MOV-627 |
| MOV-636 | MOV-637 |
| MOV-646 | MOV-647 |

h. By performing a flow balance test during shutdown following completion of HPSI system modifications that alter system flow characteristics and verifying the following flow rates:

HPSI System
Single Pump

170 ± 5 gpm to each injection leg.

## 3/4.6 CONTAINMENT SYSTEMS

## 3/4.6.1 PRIMARY CONTAINMENT

### CONTAINMENT INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.1, and

   2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

## LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$ (346,000 SCCM), 0.20 percent by weight of the containment air per 24 hours at $P_a$, 50 psig, or

2. $\leq L_t$ (44,600 SCCM), 0.042 percent by weight of the containment air per 24 hours at a reduced pressure of $P_t$, 25 psig.

b. A combined leakage rate of $\leq$ 0.60 $L_a$ (207,600 SCCM) for all penetrations and valves subject to Type B and C tests when pressurized to $P_a$.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 $L_a$ (259,500 SCCM), or 0.75 $L_t$ (33,400 SCCM), as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 $L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

## SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either $P_a$ (50 psig) or at $P_t$ (25 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 $L_a$ (259,500 SCCM) or .75 $L_t$ (33,400 SCCM), the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 $L_a$ (259,500 SCCM), or .75 $L_t$ (33,400 SCCM), a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 $L_a$ (259,500 SCCM) or .75 $L_t$ (33,400 SCCM) at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 $L_a$ (86,500 SCCM) or 0.25 $L_t$ (11,100 SCCM).

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at $P_a$ (50 psig) or $P_t$ (25 psig).

d. Type B and C tests shall be conducted with gas at $P_a$ (50 psig) at intervals no greater than 24 months except for tests involving air locks.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

## CONTAINMENT AIR LOCKS

### LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ (17,300 SCCM), at $P_a$, 50 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With an air lock inoperable, except as a result of an inoperable door gasket, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an air lock inoperable due to an inoperable door gasket:

1. Maintain the remaining door of the affected air lock closed and sealed, and

2. Restore the air lock to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a.* After each opening, except when the airlock is being used for multiple entries, then at least once per 72 hours by verifying that the seal leakage is $< 0.0002\ L_a$ (69.2 SCCM) as determined by precision flow measurement when the volume between the door seals is pressurized to a constant pressure of 15 psig,

* Exemption to Appendix "J" of 10 CFR 50.

## 3/4.6.4 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.4.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.4.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on each containment isolation Channel A or Channel B test signal, each required isolation valve actuates to its isolation position.

b. Verifying that on each Containment Radiation-High Test Channel A or Channel B test signal, both required containment purge valves actuate to their isolation position.

c. Verifying that on each Safety Injection Actuation Channel A or Channel B test signal, each required isolation valve actuates to its isolation position.

4.6.4.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Technical Specification 4.0.5.

4.6.4.1.4 Containment purge isolation valves shall be demonstrated OPERABLE at least once every 6 months by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Technical Specification 4.6.1.2.d for all other Type B or C penetrations, the combined leakage rate is less than or equal to 0.60 $L_a$ (207,600 SCCM). The leakage rate for the containment purge isolation valves shall also be compared to the previously measured leakage rate to detect excessive valve degradation.

## 6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager - Nuclear Power Department and the OSSRC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the OSSRC and the Manager - Nuclear Power Department within 14 days of the violation.

## 6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A: of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the POSRC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the originial procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the POSRC and approved by the Plant Superintendent within 14 days of implementation.

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

### STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 75 AND 56 TO

FACILITY OPERATING LICENSES NOS. DPR-53 AND DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NOS. 1 & 2

DOCKET NOS. 50-317 AND 50-318

## Introduction

By application for license amendments dated August 2, 1982, Baltimore Gas and Electric Company (BG&E) requested changes to the Technical Specifications (TS) for Calvert Cliffs Units 1 and 2. The proposed changes address the following: (1) typographical errors are corrected involving inservice inspection (TS 4.5.2.g.2) and valve testing (TS 3.4.3.c); (2) a change to TS 3.4.3.c to allow startup of the reactor with one or more inoperable, but isolated, power operated relief valves (PORVs); (3) changes to several TS to incorporate numerical values of containment leakage rates $L_a$ and $L_t$; and (4) a change to TS 6.8.1.a to incorporate by reference the latest applicable version of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)".

## Discussion and Evaluation

### 1. Correction of Typographical Errors

TS 4.4.10.1.2, "Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping", requires an "...augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition and Addenda through Summer 1976, for Class 2 components." A review of the inservice inspection program requirements for Calvert Cliffs indicates that the Summer 1976 Addenda is not applicable to Calvert Cliffs. The correct reference is to the Summer 1975 Addenda. Accordingly, it is appropriate to correct TS 4.4.10.1.2 by changing "...Addenda through Summer 1976", to "...Addenda through Summer 1975".

The second typographical error corrected herein involves TS.4.5.g.2, "Emergency Core Cooling Systems". TS 4.5.g.2 requires the verification of the correct position of the auxiliary high pressure safety injection (HPSI) loop isolation valves. One such valve, MOV-646, is repeated twice in the list of valves to be checked. A review of drawing M-74, "Calvert Cliffs Safety Injection and Containment Spray System" indicates that auxiliary HPSI loop isolation valve MOV-647

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is missing from the list of valves in TS 4.5.g.2. From the above, we conclude that one of the duplicate entries for valve MOV-646 was, most likely, intended to be for valve MOV-647. Accordingly, it is appropriate to change one of the entries for valve MOV-646 to an entry for valve MOV-647 in TS 4.5.g.2.

2. Startup of the Reactor with an Isolated PORV

The pressurizers for Calvert Cliffs 1 and 2 are each equipped with two ASME Code approved safety valves and two power operated relief valves (PORVs). Each PORV is capable of being isolated by its associated block valve. The Code Safety Valves, alone, are sufficient to provide over pressure protection from the most severe pressure transient initiated from full power. At the present time, TS 3.4.3, "Reactor Coolant Relief Valves", allows full power operation with failed PORV(s) provided that the associated block valves are closed and electrical power removed for the valve. Startup of the reactor with isolated PORVs is not permitted per TS 3.0.4*, however, since a TS "Action Statement" would have been initiated. By application dated August 2, 1982, BG&E has requested that the reactor be allowed to start up, thereby changing operation modes, with inoperable PORVs that have been properly isolated by closing their associated block valve(s) and removing electrical power to the valve(s).

As indicated above, full power operation with failed PORVs that are properly isolated is permited under TS 3.4.3. The Code Safety Valves are sufficient to protect the reactor coolant system under the most severe, full-power, overpressure, conditions.** Since there are no safety considerations associated with the startup of the reactor with isolated PORV(s) more restrictive than those already considered for full power operation it is appropriate to allow startup of the reactor with isolated PORV(s). Accordingly, the following words should be added to TS 3.4.3 as item "c":

> "With one or more block valve(s) closed, and power removed from the block valve(s) to satisfy a or b above, the provisions of Specification 3.0.4 are not applicable."

In addition to the above change, the licensee has requested relief from TS 4.4.3.2 as it relates to PORV block valve testing. The licensee has requested that, in the event that the PORV block valve has been closed and isolated in response to an inoperable PORV, block valve testing should be suspended. We concur in the licensee's request in that block valve testing with subsequent failure could prevent the isolation of a failed PORV. Accordingly, we find this change acceptable.

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* TS 3.0.4 does not permit changing operational modes when a TS "Action Statement" has been initiated.

** TS 3.4.3 applies only to operation modes 1, 2 and 3 (power operation, startup and hot standby). Overpressure protection with the reactor coolant temperature below 275°F (modes 4, 5 and 6) is addressed separately in TS 3.4.9.3 and is not affected by this proposed TS change.

3. Numerical Values of $L_a$ and $L_t$

By letter dated July 22, 1982, BG&E forwarded LER 82-22/IT, Revision 1. This LER reported an administrative inadequacy whereby a personnel error, during revision of containment local leak rate surveillance test procedures, resulted in the total Type B* and C* containment leak rate to be compared to $L_a$* rather than .6 $L_a$. The corrective measures proposed in LER 82-22/IT, Revision 1, included the introduction of numerical values for $L_a$ and $L_t$* into TS 3/4 6.1.2, "Containment Leakage", 3/4 6.1.3, "Containment Air Lock", and 4.6.4, "Containment Isolation Valves". The introduction of these numerical values, and fractions of these values as appropriate, would help assure correct comparison of measured leakage rates with limiting values in the TS.

The addition of numerical values of $L_a$ and $L_t$ to the TS, and fractions of these values as appropriate, in no way changes the associated requirements. The introduction of these numerical values is administrative in nature and has no effect on the safety of the facility.

4. Updated Reference to "Appendix A" of Regulatory Guide 1.33

Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" was part of the original licensing basis for Calvert Cliffs Units 1 and 2. The requirement to establish, implement and maintain procedures in accordance with Appendix A of Regulatory Guide 1.33 is specified in TS 6.8.1a. The commitment to Appendix A of Regulatory Guide 1.33 was subsequently updated to include Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. By application dated August 2, 1982, BG&E requested a change to TS 6.8.1a to reflect their updated commitment.

Appendix A to Regulatory Guide 1.33, Revision 2, represents the latest approved guidance from the NRC regarding the safety related activities to be controlled by procedures. This document is referenced in the BG&E Operational Quality Assurance Program as described in Section I.B.2 of the Calvert Cliffs Updated Final Safety Analysis Report (FSAR). Accordingly, it is appropriate to change TS 6.8.1a to incorporate Appendix A to Regulatory Guide 1.33, Revision 2, by reference.

Environmental Consideration

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.

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* These terms are defined in 10 CFR Part 50, Appendix J, Section II.A.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 20, 1982

Principal Contributor:

Dave Jaffe

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-317 AND 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. 75 and 56 to Facility Operating Licenses Nos. DPR-53 and DPR-69, issued to Baltimore Gas and Electric Company, which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2. The amendments are effective as of the date of issuance.

These amendments correct typographical errors involving inservice inspection and valve testing; allow start up of the reactors with one or more inoperable but isolated power operated relief valves; incorporate numerical values of containment leakage rates and incorporate by reference the latest applicable version of Regulatory Guide 1.33.

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. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

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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 the amendments.

For further details with respect to this action, see (1) the application for amendments dated August 2, 1982, (2) Amendment Nos. 75 and 56 to License Nos. DPR-53 and 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 Licensing.

Dated at Bethesda, Maryland, this 20th day of September, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing