~~APR 1 2 1981~~

Docket No. 50-317
50-318

APR 2 1 1981

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. 53 and 36 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 November 10, 1980 as supplemented by letters dated November 25, 1980 and January 23, 1981.

The changes to the TS incorporate certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident in direct response to our request dated July 2, 1980.

Certain modifications to your proposed TS changes were necessary to meet our criteria. These modifications have been discussed with and agreed to by your staff. One such modification involved the education/background requirement for the Shift Technical Advisor (STA) in TS 6.3.1. Your staff expressed concern about the definition of "equivalent" in our requirement that "the STA shall have a Bachelor's Degree or equivalent in scientific or engineering discipline with...". We have determined that the definition of equivalent may, on an interim basis until our review is completed, be as defined in your submittals of November 9, 20 and 23 and December 14, 1979. This definition may need to be revised pending the staff review of your total STA program.

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

Sincerely,

Original signed by
Robert A. Clark

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

Enclosures:
1. Amendment No. 53 to DPR-53
2. Amendment No. 36 to DPR-69
3. Safety Evaluation
4. Notice of Issuance

OFFICE						
SURNAME						
DATE						

50-317

~~APR 1 1981~~

APR 21 1981

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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PMKreutzer

Docket No. 50-317 and 50-318

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: BALTIMORE GAS AND ELECTRIC COMPANY, Calvert Cliffs Nuclear Power Plants, Units 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: ~~Amendment Nos. 53 and 36.~~
  Referenced documents have been provided PDR.

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer/pn					
DATE→	4/22/81					

NRC FORM 102 7—79

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

April 21, 1981

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. 53 and 36 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 November 10, 1980 as supplemented by letters dated November 25, 1980 and January 23, 1981.

The changes to the TS incorporate certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident in direct response to our request dated July 2, 1980.

Certain modifications to your proposed TS changes were necessary to meet our criteria. These modifications have been discussed with and agreed to by your staff. One such modification involved the education/background requirement for the Shift Technical Advisor (STA) in TS 6.3.1. Your staff expressed concern about the definition of "equivalent" in our requirement that "the STA shall have a Bachelor's Degree or equivalent in scientific or engineering discipline with...". We have determined that the definition of equivalent may, on an interim basis until our review is completed, be as defined in your submittals of November 9, 20 and 23 and December 14, 1979. This definition may need to be revised pending the staff review of your total STA program.

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

Sincerely,

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

Enclosures:
1. Amendment No. 53 to DPR-53
2. Amendment No. 36 to DPR-69
3. Safety Evaluation
4. Notice of Issuance

Baltimore Gas and Electric Company

cc:
James A. Biddison, Jr.
General Counsel
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203

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Baltimore Gas and Electric Company
Lusby, MD 20657

Bechtel Power Corporation
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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
Quality Assurance Department
301 West Preston Street
Baltimore, MD 21201

Mr. R. M. Douglass, Manager
Quality Assurance Department
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Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203

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

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

Mr. Ralph E. Architzel
Resident Reactor Inspector
NRC Inspection and Enforcement
P. O. Box 437
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4853 Cordell Avenue, Suite A-1
Bethesda, MD 20014

Mr. Bernard Fowler
President, Board of County Commissioners
Prince Frederick, MD 20768

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

Mr. W. J. Lippold, Supervisor
Nuclear Fuel Management
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
P. O. Box 1475
Baltimore, Maryland 21203

cc w/enclosure(s) and incoming
dated: 11/10/80, 11/25/80, 1/25/81

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

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. 53
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 November 10, 1980 and supplemented November 25, 1980 and January 23, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53, 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: April 21, 1981

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. 36
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 November 10, 1980 and supplemented November 25, 1980 and January 23, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   2. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 36, 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: April 21, 1981

ATTACHMENT TO LICENSE AMENDMENT NOS. 53 AND 36

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

DOCKET NOS. 50-317 AND 50-318

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.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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BASES

ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	7*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	7*
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure -- High	4	2	3	1, 2, 3	11
3. CONTAINMENT ISOLATION (CIS)#					
a. Manual CIS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	7*

Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

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
4. MAIN STEAM LINE ISOLATION					
a. Manual (MSIV Hand Switches and Feed Head Isolation Hand Switches)	1/valve	1/valve	1/valve	1, 2, 3, 4	6
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	7*
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	7*

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					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	7*

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b, and 1.c).

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
8. 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 3.3-4

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	$\geq$ 570 psia	$\geq$ 570 psia

Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

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	$\geq$ 24 inches above tank bottom	$\geq$ 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	$\leq$ 220 mr/hr	$\leq$ 220 mr/hr
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	2450 ± 105 volts with a 2 ± 0.2 second time delay	2450 ± 105 volts with a 2 ± 0.2 second time delay
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	3628 ± 25 volts with a 8 ± 0.4 second time delay	3628 ± 25 volts with a 8 ± 0.4 second time delay

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b, and 1.c).

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	$\leq$ 6.9
b. Feedwater Isolation	$\leq$ 80
7. Refueling Water Tank-Low	
a. Containment Sump Recirculation	$\leq$ 80
8. Reactor Trip	
a. Feedwater Flow Reduction to 5%	$\leq$ 20
9. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$ 2.2***
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$ 8.4***

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

***Response time measured from the incidence of the undervoltage condition to the diesel generator start signal.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure -- High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIS) #				
a. Manual CIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)(4)	1, 2, 3
4. MAIN STEAM LINE ISOLATION (SGIS)				
a. Manual SGIS (MSIV Hand Switches and Feed Head Isolation Hand Switches)	N.A.	N.A.	R	N.A.
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)(5)	1, 2, 3

Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

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				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	M	1, 2, 3
8. CVCS ISOLATION West Penetration Room/ Letdown Heat Exchanger Room Pressure - High	N.A.	R	M	1, 2, 3, 4

Containment purge valve isolation is also initiated by SIAS (functional units 1.a, 1.b and 1.c).

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.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Power Range Nuclear Flux	2
2. Containment Pressure	2
3. Wide Range Logarithmic Neutron Flux Monitor	2
4. Reactor Coolant Outlet Temperature	2
5. Reactor Coolant Total Flow	2
6. Pressurizer Pressure	2
7. Pressurizer Level	2
8. Steam Generator Pressure	2/steam generator
9. Steam Generator Level	2/steam generator
10. Feedwater Flow	2
11. Auxiliary Feedwater Flow Rate	1/steam generator
12. RCS Subcooled Margin Monitor	1
13. PORV/Safety Valve Acoustic Flow Monitoring	1/valve
14. PORV Solenoid Power Indication	1/valve

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Nuclear Flux	M	Q
2. Containment Pressure	M	R
3. Wide Range Logarithmic Neutron Flux Monitor	M	N.A.
4. Reactor Coolant Outlet Temperature	M	R
5. Reactor Coolant Total Flow	M	R
6. Pressurizer Pressure	M	R
7. Pressurizer Level	M	R
8. Steam Generator Pressure	M	R
9. Steam Generator Level	M	R
10. Feedwater Flow	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. RCS Subcooled Margin Monitor	M	R
13. PORV/Safety Valve Acoustic Monitor	N.A.	R
14. PORV Solenoid Power Indication	N.A.	N.A.

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.

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.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. The pressurizer level shall be within ± 5 percent of its programmed value.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within ± 5 percent of its programmed value at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident", December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, Thermal Margin/Low Pressure and Axial Flux Offset trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{\circ}$F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 46°F (34°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 7.6×10^{5} lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The programmed level also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valves function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of off-site power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage

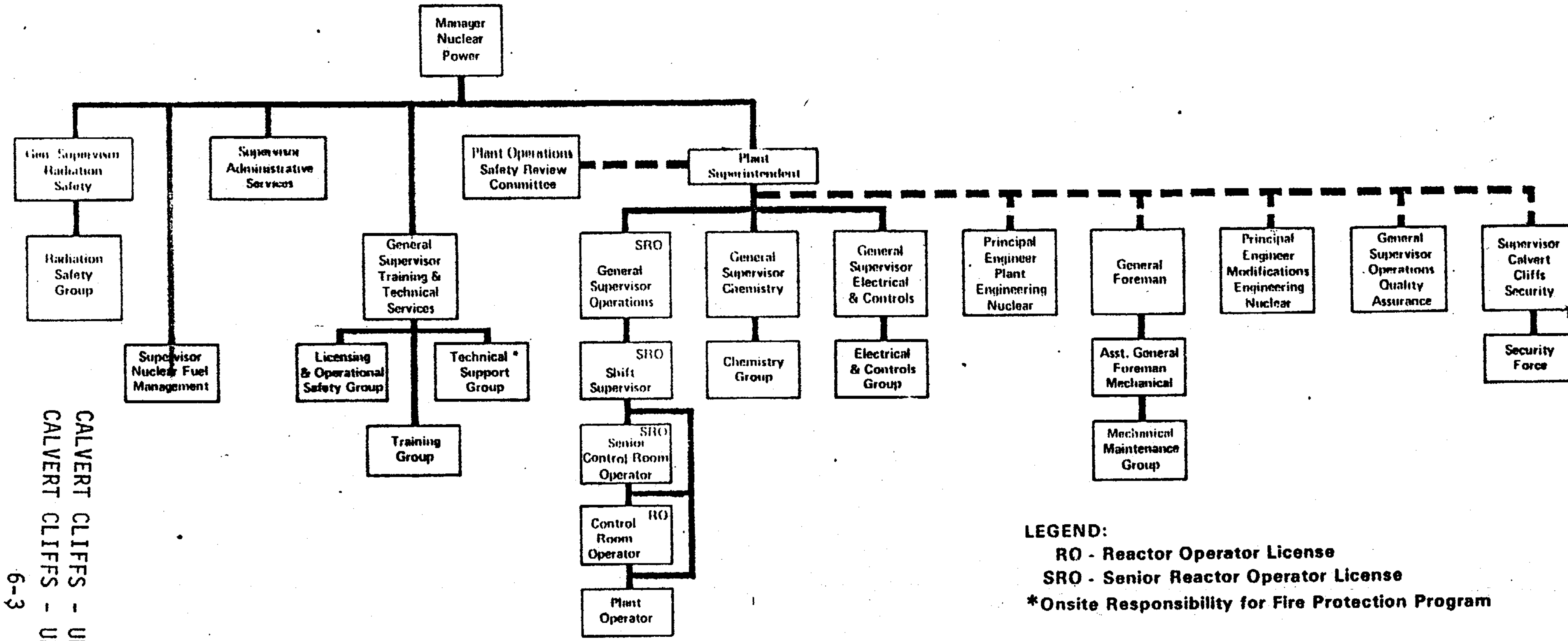

FIGURE 6.2.2 Organization Chart (Two Unit Operation) - Calvert Cliffs Nuclear Power Plant
Baltimore Gas and Electric Company

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION #

Condition of Unit 2 - No Fuel in Unit 1

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advisor	1##	0

Condition of Unit 2 - Unit 1 in MODES 1, 2, 3 or 4

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL**	2	2*
OL**	3	2
Non-Licensed	3	3
Shift Technical Advisor	1##	1##

Condition of Unit 2 - Unit 1 in MODES 5 or 6

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL**	2	1*
OL**	2	2
Non-Licensed	3	3
Shift Technical Advisor	1##	0

TABLE 6.2-1 (Continued)

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS during fuel reloading.

**Assumes each individual is licensed on each unit.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.1.

##With one unit in MODE 5 or 6, and the other unit in MODE 1, 2, 3 or 4, the SOL holder other than the Shift Supervisor may serve as STA. With one unit defueled and the other unit in MODE 1, 2, 3 or 4, the STA must be an SOL holder in addition to the one SOL required. With both units in MODE 1, 2, 3 or 4, the STA must be an SOL holder in addition to the two SOL's required.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Safety Engineer who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the General Supervisor - Training and Technical Services and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the General Supervisor - Training and Technical Services and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS AND SAFETY REVIEW COMMITTEE (POSRC)

FUNCTION

6.5.1.1 The POSRC shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The POSRC shall be composed of the:

Chairman:	Plant Superintendent
Member:	General Supervisor - Operations
Member:	General Supervisor - Electrical and Controls
Member:	General Supervisor - Chemistry
Member:	Principal Engineer - Plant Engineering Nuclear
Member:	General Foreman - Maintenance and Modifications
Member:	Supervisor - Nuclear Fuel Management
Member:	General Supervisor - Radiation Safety
Member:	General Supervisor - Training and Technical Services

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the POSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in POSRC activities at any one time.

b. A high radiation area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a, above, and in addition locked barricades shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of Licenses DPR-53 and DPR-69 dated October 24 , 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 SYSTEM INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.15 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 53 AND 36 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 letter dated November 10, 1980 as supplemented by your letters of November 25, 1980 and January 23, 1981, Baltimore Gas and Electric Company (BG&E) proposed changes to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Unit Nos. 1 and 2. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

BACKGROUND INFORMATION:

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to BG&E dated April 7, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

EVALUATION:

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in post-accident functioning of these components.

The pressurizer level instruments were originally designed to be powered from a vital instrument bus, their power sources are assured by the existing TS 3.8.2.1 and 3.8.2.2. The "functional test" for PORV's in the model specifications was deleted since the initiating channel is part of the Reactor Protective System and is tested under existing TS Table 4.3-1 item 4. The submitted specifications for pressurizer heaters and pressurizer relief and block valves include provisions for the operability and testing of emergency power supplies.

We have reviewed these proposed TS and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components. However the surveillance requirement of PORV block valve in the submitted TS 4.4.3.2 of "once per cold shutdown" is not adequate; the standard TS requires "once per 92 days". BG&E has agreed to change their proposal to meet our requirements.

2.1.3.a Direct Indication of PORV and SV Flow

BG&E has provided an acoustic monitoring system downstream of the pressurizer power-operated relief valves (PORVs) and safety valves (SVs) to provide direct indication of flow through any of these valves in the control room. These indications are a diagnostic aid for the operator and provide no automatic action. This system was previously reviewed and accepted as documented in our April 7, 1970 safety evaluation. BG&E has proposed TS consistent with our requirements. These TS are, therefore, acceptable.

2.1.3.b Instrumentation for Inadequate Core Cooling

BG&E has installed an instrument system to detect the effect of inadequate core cooling. This instrument system, a subcooling meter, receives and processes data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated April 7, 1980. The licensee submitted TS with a 31-day channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TS are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.4 Diverse Containment Isolation

The licensee has reviewed the containment isolation system to ensure that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. These parameters are pressurizer pressure low and containment pressure high. We have previously reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated April 7, 1980. BG&E has made modification such that reset does not result in the automatic loss of containment isolation after the containment isolation signal is removed. Reopening of containment isolation would require deliberate operator action.

The existing TS 3.6.4.1 and Tables 3.3-3, 3.3-4, 4.3-2 and 3.6-1 list actuation parameters, instrumentation channels, appropriate surveillance and actions in the event of component inoperability. By letter dated January 23, 1981, BG&E proposed to add footnotes to TS Tables 3.3-3, 3.3-4 and 4.3-2 for clarification of how the safety injection actuations signal effects containment isolations. We find the existing TS as clarified by the January 23, 1981 proposal acceptable.

2.1.4 Integrity of Systems Outside Containment

Our request indicated that all licensees should propose a license condition to require a periodic system integrity measurement program to prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems which are located outside the reactor containment. BG&E's present program includes provisions for a preventive maintenance program and periodic visual inspections. The program also includes system leak test measurements at frequencies not to exceed refueling cycle intervals.

In lieu of a license condition, BG&E has agreed to place such a requirement in TS Section 6. Based on our review, we find that inclusion of this requirement in the Administrative Controls Section of the TS satisfies our requirement and is, therefore, acceptable.

2.1.7.a Auto Initiation of Auxiliary Feedwater System

BG&E first installed a control grade circuit to automatically initiate the auxiliary feedwater system (AFWS) flow. This circuitry has now been upgraded to safety grade per our long-term requirement. Our Safety Evaluation of this modification will be issued under separate cover.

2.1.7.b Auxiliary Feedwater Flow Indication

The licensee has an installed auxiliary feedwater flow indication that meets our vital power requirement. We reviewed this system in our Safety Evaluation dated April 7, 1980 and found it acceptable. The licensee has proposed an 18-month channel calibration requirement. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

2.1.9.c Iodine Monitoring

Our request indicated that the licensee should implement a program which will ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. BG&E's present program includes training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

BG&E has agreed to place the requirement in TS Section 6 instead of the license. Based on our review, we find that inclusion of this requirement in the Administrative Controls Section of the TS satsifies our requirement and is, therefore, acceptable.

2.2.1.b Shift Technical Advisor

Our request indicated that the TS related to minimum shift manning should be revised to reflect the augmentation of a Shift Technical Advisor (STA). The licensee's application would add one STA to each shift to perform the function of accident assessment during reactor operation. We require that the individual performing this function have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. BG&E expressed concern about the definition of "equivalent" in the above requirement. We have determined that the definition of equivalent may, on an interim basis until our review is completed, be as defined in the BG&E submittals dated November 9, 20 and 23 and December 14, 1979. The licensee should be made aware that this definition may need to be revised pending our review of their total STA program.

Based on our review, we find the STA proposed TS satisfy our requirements and are, therefore, acceptable.

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.

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 accidents previously considered and do not involve a significant decrease in a safety margin, 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 21, 1981

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. 53 and 36 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.

The amendments incorporate certain of the Lessons Learned Category "A" requirements related to the Three Mile Island Accident.

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.

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 amendment dated November 10, 1980, (2) Amendment Nos. 53 and 36 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 21st day of April, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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