DCS MS-016

APR 2 1 1983

Docket File NSIC
NRC PDR
Local PDR
ORB#3Rdg
PMKreutzer
DJaffe
OELD
LJHarmon
DBrinkman
LSchneider (1)
TBarnhart (8)
ACRS (10)
OPA (Clare Miles)
RFerguson
RDiggs
NSIC

Docket Nos. 50-317
and 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. 82 and 65 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 partial response to your application dated February 24, 1983. In our review of your requests, we found it necessary to make certain changes which were discussed with and agreed to by your staff.

These amendments revise the Technical Specifications to correct typographical errors, establish procedures limiting overtime for personnel involved in safety related activities, increase the steam generator minimum pressurization temperature, delete a requirement on the pressurizer safety valve acoustic monitor and change administrative requirements to provide for yearly audit and review of the safeguards contingency plan and the emergenc plan.

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

Sincerely,

Original signed by:

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

Enclosures:
1. Amendment No. 82 to DPR-53
2. Amendment No. 65 to DPR-69
3. Safety Evaluation
4. Notice of Issuance

cc: See next page

8305040783 830421
PDR ADOCK 05000317
P PDR

See previous concurrence page

OFFICE	ORB#3:DL*	ORB#3:DL*	ORB#3:DL*	DL:OR	OELD		
SURNAME	PKreutzer	DJaffe:dd	RAClark	GCLainas			
DATE	3/24/83	3/31/83	3/ /83	4/11/83	4/14/83		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

ASLAB

Docket File
NRC PDR
LOCAL PDR
ORB#3Rdg
DEisenhut
PMKreutzer
DJaffe
OELD
LJHarmon
TBarnhart (8)
LSchneider (1)
DBrinkman
ACRS (10)
OPA (Clare Miles)
RFerguson
RDiggs
NSIC

Docket Nos. 50-317
and 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. and 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 partial response to your application dated February 24, 1983. In our review of your requests, we found it necessary to make certain changes which were discussed with and agreed to by your staff.

These amendments revise the Technical Specifications to correct typographical errors, establish procedures limiting overtime for personnel involved in safety related activities, increase the steam generator minimum pressurization temperature, provide Limiting Conditions for Operation and Surveillance Requirements for the containment water level monitor, and change administrative requirements to provide for yearly audit and review of the safeguards plan and the emergency plan.

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

Sincerely,

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

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

cc: See next page

ORAB
T. IPPOLITO
4/11/83

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	DL:OR	OELD		
SURNAME	PMKreutzer	DJaffe:dd	RAClark	GCLainas			
DATE	3/24/83	3/31/83	4/11/83	3/ /83	3/ /83		

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

DISTRIBUTION:

Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-317
and 318

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: BALTIMORE GAS & ELECTRIC COMPANY, CCalvert Cliffs Units Nos. 1&2

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

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment No. 82 and 65
Referenced documents have been provided PDR.

Division of License
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer:dd					
DATE→	4/21/83					

NRC FORM 102 7—79

8305040789 830421
PDR ADOCK 05000317
P PDR

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

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. R. C. L. Olson, Principal Engineer
Nuclear Licensing Analysis Unit
Baltimore Gas and Electric Company
Room 922 - G&E Building
P. O. Box 1475
Baltimore, MD 21203

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. R. R. Mills, Manager
Engineering Services
P. O. Box 500
Windsor, CT 06095

Mr. R. M. Douglass, Manager
Quality Assurance Department
Baltimore Gas & Electric Company
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
Manager - Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, MD 20814

Mr. J. A. Tiernan, Manager
Nuclear Power Department
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

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

Mr. R. E. Denton, General Supervisor
Training & Technical Services
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

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. 82
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 February 24, 1983, 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' 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. 82, 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, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 82

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. Corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 3-41
3/4 3-42
3/4 7-13
3/4 7-22
B 3/4 4-1
B 3/4 7-3
6-11
6-13

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.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 80°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant < 80°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The ECCS pump room exhaust ventilation system shall be OPERABLE with one HEPA filter and charcoal adsorber train and two exhaust fans.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one ECCS pump room exhaust fan inoperable, restore the inoperable fan 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.

b. With the ECCS exhaust filter train inoperable, restore the filter train 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.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The ECCS pump room exhaust ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that each exhaust fan operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 3000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R. H.); or

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.195 during all normal operations and anticipated transients.

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 shutdown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{o}F$ 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 $46^{o}F$ ($34^{o}F$ 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 approximately 3×10^5 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 operating band for pressurizer level bounds the programmed level and ensures that RCS pressure remains within the bounds of an analyzed condition during the excessive charging event as well as during the limiting depressurization event, Excess Load. The operating band 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 offsite 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

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

The secondary water chemistry program is designed to provide maximum protection to both the steam generator and secondary system internals. The most damaging chemical reactants enter the system via condenser cooling water ingress. Accumulation of these impurities in the steam generators may lead to loss of metallurgical integrity and/or eventual component failure. The limits presented in Table 3.7-3 are those prescribed by the NSSS supplier as "limited-operation" chemistry parameters and are consistent with the most recent industry standards. By routine monitoring of these parameters, plant personnel are able to rapidly detect and limit the duration of ingress of chemically detrimental species and thereby maintain steam generator tube integrity.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 80°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 SALT WATER SYSTEM

The OPERABILITY of the salt water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the ECCS pump room exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the

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.

g. The amount of overtime worked by plant staff members performing safety related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

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.

AUDITS

6.5.2.8.1 Audits of facility activities shall be performed under the cognizance of the OSSRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualification of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. Deleted

f. The Safeguards Contingency Plan and implementing procedures at least once per 12 months in accordance with 10 CFR 73.40(d).

g. Any other area of facility operation considered appropriate by the OSSRC or the Vice President-Supply.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

6.5.2.8.2 Review of facility activities shall be performed under the cognizance of the OSSRC. These reviews shall encompass:

a. The Facility Emergency Plan and implementing procedures at least once per 12 months in accordance with 10 CFR Part 50.54(t).

AUTHORITY

6.5.2.9 The OSSRC shall report to and advise the Vice President-Supply on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of OSSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each OSSRC meeting shall be prepared, approved and forwarded to the Vice President-Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President-Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice President-Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the POSRC and submitted to the OSSRC and the Manager - Nuclear Power Department.

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
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 February 24, 1983, 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' regulations;

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

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

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

   2. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 65, 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, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 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. Corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 7-13
3/4 7-22
B 3/4 4-1
B 3/4 7-3
6-11
6-13

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 90°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant < 90°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The ECCS pump room exhaust ventilation system shall be OPERABLE with one HEPA filter and charcoal adsorber train and two exhaust fans.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one ECCS pump room exhaust fan inoperable, restore the inoperable fan 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.

b. With the ECCS exhaust filter train inoperable, restore the filter train 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.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The ECCS pump room exhaust ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that each exhaust fan operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 3000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R. H.); or

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.195 during all normal operations and anticipated transients.

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 shutdown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

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 approximately 3×10^5 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 operating band for pressurizer level bounds the programmed level and ensures that RCS pressure remains within the bounds of an analyzed condition during the excessive charging event as well as during the limiting depressurization event, Excess Load. The operating band 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 offsite 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

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

The secondary water chemistry program is designed to provide maximum protection to both the steam generator and secondary system internals. The most damaging chemical reactants enter the system via condenser cooling water ingress. Accumulation of these impurities in the steam generators may lead to loss of metallurgical integrity and/or eventual component failure. The limits presented in Table 3.7-3 are those prescribed by the NSSS supplier as "limited-operation" chemistry parameters and are consistent with the most recent industry standards. By routine monitoring of these parameters, plant personnel are able to rapidly detect and limit the duration of ingress of chemically detrimental species and thereby maintain steam generator tube integrity.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 90°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 SALT WATER SYSTEM

The OPERABILITY of the salt water system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the ECCS pump room exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the

AUDITS

6.5.2.8.1 Audits of facility activities shall be performed under the cognizance of the OSSRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. Deleted

f. The Safeguards Contingency Plan and implementing procedures at least once per 12 months in accordance with 10 CFR 73.40(d).

g. Any other area of facility operation considered appropriate by the OSSRC or the Vice President-Supply.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

6.5.2.8.2 Review of facility activities shall be performed under the cognizance of the OSSRC. These reviews shall encompass:

a. The Facility Emergency Plan and implementing procedures at least once per 12 months in accordance with 10 CFR 50.54(t).

AUTHORITY

6.5.2.9 The OSSRC shall report to and advise the Vice President-Supply on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of OSSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each OSSRC meeting shall be prepared, approved and forwarded to the Vice President-Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President-Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice President-Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the POSRC and submitted to the OSSRC and the Manager - Nuclear Power Department.

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.

g. The amount of overtime worked by plant staff members performing safety related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

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. 82 AND 65 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 amendment dated February 24, 1983, Baltimore Gas and Electric Company (BG&E) requested changes to the Technical Specifications (TS) for Calvert Cliffs Units 1 and 2. The proposed changes to the TS would (1) correct typographical errors as they appear in TS 4.7.7.1, "ECCS Pump Room Exhaust Air Filtration System" and TS Bases 3/4.4.2, "Safety Valves", (2) establish TS 6.8.1.g. to require procedures limiting overtime for personnel involved in safety related activities, (3) change TS 3/4.7.2, "Steam Generator Pressure/Temperature Limitation" and associated Bases to increase the steam generator minimum pressurization temperature (MPT), (4) delete a requirement on the pressurizer safety valve acoustic flow monitor, and (5) change the administrative requirements of Section 6 of the TS to provide for yearly audit and review of the safeguards contingency plan and the facility emergency plan, respectively.

In the course of reviewing the proposed TS submitted with the February 24, 1983 application, the staff found it necessary to make certain changes in the TS. These changes were discussed with and agreed to by BG&E.

Discussion and Evaluation

The February 24, 1983 application identifies two typographical errors in the TS and proposes the corrective wording. The first error appears in TS 4.7.7.1 in which the flow rate for ECCS pump room exhaust air filter bank testing is given as 2000 cfm $\pm$ 10%. BG&E has indicated that this value should be 3000 cfm $\pm$ 10%. A review of the TS indicates that the required system flow rate is shown in TS 4.7.7.1.b.4 as 3000 cfm $\pm$ 10%. This value is in agreement with Final Safety Analysis Report (FSAR) data shown in Table 9-19. Accordingly, the flow rate presently contained in TS 4.7.7.1, 2000 cfm $\pm$ 10%, is in error and should be changed to 3000 cfm $\pm$ 10%.

The second typographical error identified by BG&E appears in the Bases for TS 3/4.4.2. The Bases identifies the pressurizer safety valve relief flow rate as 7.6×10^5 lbs per hour. BG&E has indicated that the correct value should be "approximately 3×10^5 lbs per hour." This value is consistent with information contained in the Nuclear Steam Supply System (NSSS) design specifications. A review of the FSAR Table 4-19 indicates that two pressurizer safety valves are installed with relief flow rates of 2.96×10^5 and 3.02×10^5

lbs per hour. These values can be described as "approximately $3x10^5$ lbs per hour." Accordingly, the present TS bases value for the pressurizer safety valve relief rate, $7.6x10^5$ lbs per hour, should be changed to "approximately $3x10^5$ lbs per hour."

A second area of TS change involves the addition of an administrative TS requirement to establish written procedures to control "... the amount of overtime worked by plant staff members performing safety related functions" in accordance with NRC Generic Letter 82-12. Generic Letter 82-12 was issued on June 15, 1982 and contains the NRC position on limiting overtime for personnel involved in safety related activities. The limiting of overtime for personnel involved in safety related activities was established as TMI Action Item I.A.1.3.1 in NUREG-0737. A review of the proposed TS, as modified by the NRC and agreed to by BG&E, indicates that it is in agreement with the wording proposed by the NRC in its Generic Letter 82-16. This generic letter was issued on September 20, 1982 to licensees of Pressurized Water Reactors to provide guidance on acceptable wording for TS related to TMI Action Items. We conclude that the proposed wording of TS 6.8.1.g is in accordance with the NRC guidance contained in Generic Letter 82-16 and is therefore acceptable.

The third area of TS change involves the minimum pressurization temperature (MPT) for the steam generators. The establishment of an MPT assures that the steam generators will behave in a ductile fashion in response to transient pressure conditions. The MPT is presently specified as greater than 70°F at steam generator pressures greater than 200 psig per TS 3/4.7.2. BG&E has proposed that the MPT be increased to greater than 80°F for Unit 1 and greater than 90°F for Unit 2, for steam generator pressures greater than 200 psig. The proposed increase in MPT is based upon recommendations by the NSSS supplier. MPTs are established in recognition that the primary system materials undergo a transition from ductile to brittle behavior at low temperatures. Avoidance of operation of the steam generators in a temperature range where brittle failure could occur is important to prevent sudden failure of the reactor coolant pressure boundary represented by the steam generator tubes. Since brittle failure is a low temperature effect, the raising of the MPT is conservative in that it provides additional margin to the temperature range where brittle failure could occur. Accordingly, the increase in MPT as reflected in revised TS 3/4.7.2 and the associated Bases is acceptable.

A change to TS 3.3.3.6 is considered herein for deletion of a footnote in TS Table 3.3-10. This footnote would have allowed the inoperability of the acoustic flow monitor for Unit 1 Pressurizer Safety Valve PV-201 until June 1, 1981. Deletion of this relief is appropriate since the applicable date has passed. This is an administrative action with no safety significance and, accordingly, is acceptable.

The final TS change considered herein relates to the review and audit of emergency preparedness and safeguards contingency plans. On October 1, 1982 the NRC issued Generic Letter 82-17 which informed licensees and applicants of the requirements of 10 CFR 50.54(t) for an annual review of the facility emergency plan; a request was made for incorporation of this requirement in the TS. On October 30, 1982 the NRC issued Generic Letter 82-23 which informed licensees and applicants of the requirements of 10 CFR 73.40(d) for an annual audit of the safeguards contingency plan. A request was also made in Generic Letter 82-23 for incorporation of a requirement in the TS for annual audit of the safeguards contingency plan. BG&E has agreed to appropriate changes to TS 6.5.2.8 which would make the annual audit and review of the safeguards contingency plan (and implementing procedures) and the emergency plan (and implementing procedures) a responsibility of the BG&E Off-Site Review Committee (OSSRC). In this regard, we recognize the differences between a "review" and an "audit" as contained in ANSI Standard N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." An audit is defined in ANSI N18.7 as a methodical examination to determine conformance to requirements; a review represents a critical examination and evaluation to determine the adequacy of the requirements. We find the proposed changes to the TS acceptable since they meet the requirements contained in Generic Letters 82-17 and 82-23.

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 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: April 21, 1983

Pricipal Contributor:

D. H. Jaffe

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-317 AND 50-318

BALTIMORE GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 82 and 65 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, Unit Nos. 1 and 2. The amendments were effective as of the date of issuance.

The amendments revise the Technical Specifications to correct typographical errors, establish procedures limiting overtime for personnel involved in safety related activities, increase the steam generator minimum pressurization temperature, delete a requirement on the pressurizer safety valve acoustic monitor, and change administrative requirements to provide for yearly audit and review of the safeguards contingency plan and emergency plan.

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 the amendments will not result in any significant environmnetal 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 February 24, 1983, (2) Amendment Nos. 82 and 65 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, 1983

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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