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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. 80 and 63 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 September 29, 1982.

These amendments revise the Technical Specifications to incorporate a revised pressurizer level band into the Pressurizer Limiting Condition for Operation and Surveillance Requirements.

A copy of the Safety Evaluation and 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. 80 to DPR-53
2. Amendment No. 63 to DPR-69
3. Safety Evaluation
4. Notice of Issuance

cc: See next page

No legal objection to form of amendment and SER known not requested

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OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD		
SURNAME	PMKreutzer	DJaffe/m	RAClark	GCLainas	CUTCHIN		
DATE	1/11/83	1/17/83	1/17/83	1/18/83	1/18/83		

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

DISTRIBUTION:
Docket File
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PMKreutzer

Docket No. 50-317/50-318

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: BALTIMORE GAS AND ELECTRIC COMPANY, Clavert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2

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

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

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

☐ Notice of Availability of Applicant's Environmental Report.

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

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

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

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

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

☒ Other: Amendment Nos. 80 and 63.
Referenced documents have been provided.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#3:DL					
SURNAME →	PMKreutzer/pr					
DATE →	1/26/82					

NRC FORM 102 7—79

Baltimore Gas and Electric Company

cc:
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Combustion Engineering, Inc.
Attn: Mr. P. W. Kruse, Manager
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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
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Nuclear Regulatory Commission, Region I
Office of Executive Director for Operations
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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. 80
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 September 29, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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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.80 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment if effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 25, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 80

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 4-5
B. 3/4 4-2

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 maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM maximum pressurizer level shall be limited to between 133 and 210 inches.

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 the above band 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

3/4.4.1 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 $46^{\circ}F$ ($34^{\circ}F$ 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 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

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. 63
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 September 29, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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. 63, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 25, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 63

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. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 4-5
B 3/4 4-2

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 maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM maximum pressurizer level shall be limited to between 133 and 210 inches.

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 the above band at least once per 12 hours.

STEAM GENERATORS

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.

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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

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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.

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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 $46^{\circ}F$ ($34^{\circ}F$ 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

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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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 80 AND 63 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

1.0 INTRODUCTION

In a letter dated September 29, 1982, the Baltimore Gas and Electric Company (BG&E) requested an amendment to operating licenses DPR-53 and DPR-69 for Calvert Cliffs Units 1 & 2. The amendment would change Technical Specification (TS) 3.4.4, "Pressurizer" Limiting Condition for Operation, and its associated surveillance requirements, to incorporate a revised pressurizer level band.

1.1 Discussion

A pressurizer provides pressure control for the reactor coolant system during reactor operation. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level ensures that the reactor coolant system is not a hydraulically solid system and that a steam bubble will be provided in the upper portion of the pressurizer to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief.

Pressurizer level is maintained by the action of the chemical and volume control system. A low pressurizer level signal functions to reduce letdown flow proportionally and to start the nonoperating charging pumps. A high level indication functions to increase letdown flow proportionally by opening the letdown control valves and stopping all but one charging pump.

The operating level of the pressurizer is programmed as a function of reactor power, as reflected by average coolant temperature (T_{avg}), to accommodate plant load changes and transients to minimize the changes in reactor coolant system volume. The level programmer establishes a program level which is directly proportional to average coolant temperature over the operating range of T_{avg}.

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The current TS 3.4.4 for Calvert Cliffs Units 1 and 2 states in part:

"The presurizer level shall be within ±5 percent of its program value."

Its program value is varying between 0% to 100% power operated within the Pressurizer Level setpoints 144" and 216".

Calvert Cliffs' operating experience has demonstrated that the present Limiting Condition for Operation (LCO) on pressurizer level is too restrictive resulting in an excessive number of Licensee Event Reports (LERs). In order to reduce the number of violations of the Pressurizer LCO, the licensee proposed a new pressurizer operating band with a maximum acceptable pressurizer level of 225 inches and a minimum acceptable pressure level of 133 inches for the plant's normal operations. The establishment of this new operating band was based on the results of reanalyses of three events, namely: the Loss of Load Event and the Excessive Charging Event which were analyzed to determine the maximum acceptable pressurizer level, and the Excess Load Event which was reanalyzed to determine the minimum acceptable pressurizer level. The third event, the Steam Generator Tube Rupture Event, was analyzed to determine the effect of the higher primary to secondary leak rate on site boundary doses.

2.0 EVALUATION

The changes to the TS proposed by BG&E in the September 29, 1982 application were reviewed with the consideration of providing adequate protection for the pressurizer and thus the primary system, i.e., to protect the pressurizer heaters from being energized uncovered and to protect the pressurizer code safety valves as well as power operated relief valves against water relief thus to ensure the operability of the pressurizer itself.

2.1 The Selection of the Limiting Events

By letter dated December 17, 1982, the licensee has provided additional information to justify that the three events analyzed are the limiting events. The criteria which BG&E has applied to select the limiting events for providing the basis for redefinition of the limits of acceptable pressurizer level are (1) that pressure limits were not exceeded by design basis events (DBEs) as a result of the change in pressurizer high-level TS (216"), and (2) reactor coolant circulation was not adversely affected as a result of the change in pressurizer low-level TS (144").

The BG&E evaluation was accomplished by reviewing all DBEs to determine which events would result in the largest increase or decrease in pressurizer level. The evaluation shows that the limiting DBEs sensitive to the proposed TS change are the Loss of Load Event, the Excess Load, and Steam Genarator Tube Rupture Events. The Excess Charging Event was determined

to be the event having the maximum potential for filling the pressurizer water solid. Therefore, it was analyzed as part of the process to redefine the pressurizer upper TS limit.

2.2 DBE Analyses

The BG&E analyses of the events presented below provide the basis for redefinition of the limits of acceptable pressurizer level.

2.2.1 The Excess Load Event

The Excess Load Event was analyzed to determine the minimum acceptable pressurizer level. The transient was initiated at a pressurizer level of 116 inches which is the level required to cover the pressurizer heaters. The 116-inch level includes 270 cubic feet of reactor coolant and the volume displacement of the pressurizer heaters. The depressurization was initiated by the instantaneous opening of steam dump and bypass valves which have a combined capacity of approximately 45% of the normal full power steam flow. This Excess Load Event persists until the steam generators are isolated on steam generator isolation signal due to low secondary pressure. The full power event maximizes primary cooldown, shrinkage and consequently reactor vessel upper head (RVUH) voiding.

The Excess Load Event was analyzed to assure that the draining of the pressurizer and subsequent RVUH voiding do not affect primary coolant circulation and do not result in exceeding fuel design limits.

The assumptions and parameters for analysis were selected to yield the maximum voiding in the RVUH. The computer code used for the analysis was the Combustion Engineering System's Excursion Code (CESEC), a staff-approved digital computer program which was developed to provide for the simulation of a Combustion Engineering Nuclear Steam Supply System (NSSS).

The analysis of the Excess Load Event predicts that at the time of maximum RVUH voiding approximately 63% of the head is occupied by steam. Since this steam bubble does not expand beyond the upper head and therefore will not affect primary coolant circulation, it will not result in exceeding fuel design limits.

We have reviewed the key parameters assumed and the results of the analysis, and have concluded that the licensee's proposed new TS limit on the minimum pressurizer level is acceptable. The licensee has further proposed that, taking into account the instrument uncertainties, the determined minimum pressurizer level of 116 inches is increased by 17 inches thus the proposed new pressurizer level lower limit will be 133 inches.

2.2.2 The Excessive Charging Event and the Loss of Load Event

These two events were analyzed to determine the maximum acceptable pressurizer level.

2.2.2.1 The Excessive Charging Event

The Excessive Charging Event initiated from a given maximum pressurizer level of 242 inches was analyzed to assure that the operator has at least fifteen minutes from initiation of a high pressurizer level alarm to take corrective action and terminate the event prior to filling the pressurizer solid. The analysis was performed for three combinations of charging and letdown flows: (1) all three charging pumps operating without letdown flow, (2) all three charging pumps operating with minimum letdown flow, and (3) two charging pumps operating without letdown flow.

The submitted analysis result (Table 1) shows that all three combinations of charging and letdown flows analyzed provide at least fifteen minutes after initiation of high level alarm for the operator to take corrective actions and terminate the event prior to filling the pressurizer solid.

We have reviewed the assumptions and the result of the analysis, and it is our judgment that the fifteen minutes is a sufficient amount of time to allow the operator to take corrective action to terminate the event prior to filling the pressurizer solid.

In the case of three charging pumps operating with less than 25 gpm letdown flow, the licensee has further proposed that the maximum pressurizer level shall be limited to 210 inches. This is the maximum initial pressurizer level of 227 inches assumed in the analysis (1) as shown in Table 1 reduced by 17 inches due to instrument uncertainties. This reduced pressurizer level is required to preserve the 15 minutes operator response time allowance.

2.2.2.2 The Loss of Load Event

The Loss of Load Event initiated from a given maximum pressurizer level of 242 inches was analyzed to assure that this transient does not cause the reactor coolant system (RCS) pressure to exceed the design limit of 2750 psia. The transient was initiated by the sudden closure of the turbine stop valves without a simultaneous reactor trip. The event was also analyzed to determine that the acceptable DNBR limit is not exceeded.

The assumptions and parameters for the analyses were selected to cause the greatest reduction in the rate of heat removal from the reactor coolant system and thus results in the most rapid increase in primary pressure. The CESEC computer program was used for the analysis of this event.

Table 1

Volume control assumptions		Maximum initial pressurizer level assumed in analysis		High level alarm analysis setpoint		Time to fill[1] pressurizer
Charging flow (GPM)	Letdown flow (GPM)	Liquid volume (ft^3)	level[(3)] (in)	Liquid volume (ft^3)	level[(3)] (in)	(minutes)
1. 132	0	915	227	920	228	15
2. 132	29	975[(2)]	242	1040	257	15
3. 88	0	975[(2)]	242	1100	272	15

(1) From time of initiation of high pressurizer level alarm

(2) Maximum limit based on Loss of Load event

(3) Referenced to the 1" level nozzle at the bottom of the pressurizer

The submitted analysis predicts that the transient results in a high pressurizer pressure trip signal at 6.2 seconds. At 10.1 seconds, the primary pressure reaches its maximum value of 2617 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 5.8 seconds at the setpoints of 1000 psia. The minimum transient DNBR calculated for the event is 1.34, compared to the design limit of 1.23. The results of the analysis demonstrate that during a Loss of Load Event, initiated from a pressurizer level of 242 inches, the peak RCS pressure and the minimum DNBR do not exceed their respective design limits.

We have reviewed the assumptions and result of this analysis, and have concluded that the licensee's analysis of this event to determine the new TS limit on the maximum pressurizer level is acceptable. Also, as discussed in Section 2.2.1 of this report, the licensee has further proposed to take into account the 17-inch instrument uncertainties, hence the proposed new upper limit will be 225 inches, 17-inches less than the parameter used in the event analysis.

2.3.3 The Steam Generator Tube Rupture Event

The Steam Generator Tube Rupture (SGTR) Event was analyzed to determine the effect of the higher primary to secondary leak rate on site boundary doses.

The analysis included the effects of manually tripping the reactor coolant pumps on a Safety Injection Actuation Signal due to low pressurizer pressure. The design basis SGTR is a double ended break of one steam generator U-tube.

For this event, the acceptable minimum DNBR did not exceed the design limit of 1.23. Also, the submitted analysis predicts that the doses resulting from the activity released as a consequence of a double-ended rupture of one steam generator tube are 0.34 rem for thyroid (DEQ I-131) and 0.18 rem for whole body (DEQ Xe-133) which are significantly below the guidelines of 10 CFR 100.

3.0 TECHNICAL SPECIFICATIONS

Based upon the review presented in Section 2.0, a change to TS 3/4.4.4, to incorporate a revised pressurizer level band of between 225 inches and 133 inches is consistent with the analyses of the affected DBEs, and is therefore acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments

involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 25, 1983

Principal Contributors:

P. Ting
D. Jaffe

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 September 29, 1982, (2) Amendment Nos. 80 and 63 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 25th day of January, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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. 80 and 63 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 are effective as of the date of issuance.

These amendments revise the Technical Specifications to incorporate a revised pressurizer level band into the Pressurizer Limiting Condition for Operation and Surveillance Requirements.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on December 8, 1982 (47 FR 55351). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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