May 3, 1988

Docket Nos. 50-317
and 50-318

Mr. J. A. Tiernan
Vice President-Nuclear Energy
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

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Dear Mr. Tiernan:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT UNITS 1 AND 2 (TACS 63096, 63097, 64598, 64599, 64600, 64601 AND 64602)

The Commission has issued the enclosed Amendment No. 129 to Facility Operating License No. DPR-53 and Amendment No. 111 to Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application transmitted by letters dated October 1, 1986 and January 20, 1987, as supplemented on February 16 and February 26, 1988.

These amendments (1) Modify the Unit 1 TS Limiting Condition For Operation (LCO) 3.3.3.2 for incore detectors by placing additional restrictions upon operability above those that were required for operation during the previous cycle (Cycle 8); (2) Change the surveillance periods of the Units 1 and 2 TS Surveillance Requirements (SRs) 4.1.3.4.c (demonstration of full length control element assembly (CEA) drop time) and 4.3.3.2.b (incore detector channel calibration) from at least once per 18 months to at least once per refueling interval, where a refueling interval shall be defined as 24 months; (3) Modify the Units 1 and 2 TS SR 4.7.11.1.1.f.3 for cycling fire suppression water system flow path valves that are not testable during plant operation; and 4.7.11.4.b, for the inspection, reracking and replacement of degraded coupling gaskets for fire hoses inside containment by extending their associated surveillance intervals from at least once every 18 months to at least once per refueling interval (24 months); (4) Renumber the Units 1 and 2 TS SR 4.7.11.1.1.f.3 as 4.7.11.1.1.g(2); TS SR 4.7.11.1.1.g as 4.7.11.1.1.g(1); and TS SR 4.7.11.1.1.f.4 as 4.7.11.1.1.f.3 and change the Units 1 and 2 TS SRs 4.7.11.1.1.g (fire suppression system flow test), 4.7.11.2.b and c (spray and sprinkler system functional test), and 4.7.11.4.c (containment fire hose stations operability and hydrostatic tests) by making administrative changes and more restrictive changes to the surveillance requirements; and (5) Change the Units 1 and 2 TS SR 4.4.10.1.2, "Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping," to update the required ASME Boiler and Pressure Vessel Code, Section XI, for Class 2 components, from the 1974 Edition with

Addenda through Summer 1975 to the 1983 Edition with Addenda through Summer 1983. In addition, TS SR 4.4.10.1.2.a would be deleted and TS SR 4.4.10.1.2.b would be renumbered as 4.4.10.1.2 and would be clarified to reflect a new 10-year inservice inspection interval.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Original signed by:

Scott Alexander McNeil, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 129 to DPR-53
2. Amendment No. 111 to DPR-69
3. Safety Evaluation

cc: w/enclosures
See next page

PDI-1* cv 5/3/88	PDI-1* 5/3/88 sam	OGC-WF*	RoC PDI-1
CVogan	SMcNeil:dlg	APHodgdon	RCapra
4/26/88	4/26/88	4/27/88	5/3 /88

* See previous concurrence.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Scott Alexander McNeil, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 129 to DPR-53
2. Amendment No. 111 to DPR-69
3. Safety Evaluation

cc: w/enclosures
See next page

PDI-1	PDI-1 SWM	OGC-WF	PDI-1
CVogan CV	SMcNeil:dlg		RCapra
4/26/88	4/26/88	4/21/88	4/ /88

Mr. J. A. Tiernan
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

cc:

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Calvert County Board of
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Prince Frederick, Maryland 20768

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
License No. DPR-69

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

   A. The applications for amendment by Baltimore Gas and Electric Company (the licensee) dated October 1, 1986 and January 20, 1987 as supplemented on February 16 and February 26, 1988, 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 applications, 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. 111, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 111

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 1-23	3/4 1-23
3/4 1-24*	3/4 1-24*
3/4 3-29*	3/4 3-29*
3/4 3-30	3/4 3-30
3/4 4-29	3/4 4-29
3/4 4-30*	3/4 4-30*
3/4 7-59	3/4 7-59
3/4 7-60*	3/4 7-60*
3/4 7-61	3/4 7-61
3/4 7-62	3/4 7-62
3/4 7-63	3/4 7-63
3/4 7-64*	3/4 7-64*
3/4 7-65	3/4 7-65
3/4 7-66*	3/4 7-66*

* Overleaf pages provided to maintain document completeness.

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515^{o}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per refueling interval.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AZIMUTHAL POWER TILT:

   At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

   1. At least 75% of all incore detector segments,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate:

   1. At least 75% of all incore detector locations,

   2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

   3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detector location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

LIMITING CONDITION FOR OPERATION

An **OPERABLE** quadrant symmetric incore detector segment group shall consist of a minimum of three **OPERABLE** rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the **AZIMUTHAL POWER TILT**,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the **UNRODDED PLANAR RADIAL PEAKING FACTOR**, the **UNRODDED INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated **OPERABLE**:

a. By performance of a **CHANNEL CHECK** within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the **AZIMUTHAL POWER TILT**.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the **UNRODDED PLANAR RADIAL PEAKING FACTOR**, the **UNRODDED INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

b. **At least once per** Refueling Interval by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

SURVEILLANCE REQUIREMENTS (Continued)

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

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition and Addenda through Summer 1983, for Class 2 components.

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

*Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed by June 1991 in conjunction with the reactor coolant pump motor overhaul program.

REACTOR COOLANT SYSTEM

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1.

ACTION:

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

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

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

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

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

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

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting the electric motor driven pump and operating it for at least 15 minutes. This test shall be performed on a STAGGERED TEST BASIS with the test required by 4.7.11.1.2.a.2.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

d. At least once per 12 months by performance of a system flush of the filled portions of the system.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each pump develops at least 2500 gpm at a discharge pressure of 125 psig,

   3. Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure $\geq$ 80 psig.

g. At least once per refueling interval by: (1) performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association, and (2) performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The diesel fuel oil day storage tank contains at least 174 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes. This test shall be performed on a STAGGERED TEST BASIS with the test required by Specification 4.7.11.1.1.b.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq$ 20 minutes while loaded with the fire pump.

4.7.11.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is $\geq$ 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems shown in Table 3.7-5 shall be **OPERABLE**:

APPLICABILITY: Whenever equipment in the spray/sprinkler protected areas is required to be **OPERABLE**.

ACTION:

a. With one or more of the required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant safe shutdown systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule forrestoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required spray and/or sprinkler systems shall be demonstrated **OPERABLE**:

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

b. At least once per 12 months by cycling each valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months

1. By performing a system functional test which includes simulated automatic actuation of the system, and verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the area in the vicinity of each nozzle(s) to verify the spray pattern will not be obstructed.

TABLE 3.7-5

FIRE PROTECTION SPRINKLERS

UNIT 2

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
Unit 2 Aux Feed Pump Room 605*	12'-0"
Unit 2 East Piping Area Room 408*	45'-0"
Unit 2 East Elec Pen Room 409*	45'-0"
Unit 2 West Elec Pen Room 414*	45'-0"
Cable Chase 2A*	45'-0"
Cable Chase 2B*	45'-0"
Unit 2 Main Steam Piping Room 309*	45'0"
Unit 2 Component Cooling Pp Room 201	5'-0"
Unit 2 East Piping Area 203*	5'-0"
Unit 2 Rad Exh Vent Equip Room 204*	5'-0"
Unit 2 Service Water Pp Room 205*	5'-0"
Unit 2 Boric Acid Tk and Pp Room 215*	5'-0"
Unit 2 Reactor Coolant Makeup Pump Room 216A*	5'-0"
Unit 2 Charging Pump Room 105*	(-)10'-0"
Unit 2 Misc Waste Monitor Tk Room 106*	(-)10'-0"
Unit 2 ECCS Pump Room 101*	(-)15'-0"
21 Diesel Generator	45'-0"
Unit 2 East Pipe Pen Room 206/310*	5'-0"

* Sprinklers required to ensure the OPERABILITY of redundant safe shutdown equipment.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

a. Cable spreading rooms total flood system, and associated vertical cable chase 1C, Unit 2.

b. 4160 volt switchgear rooms 27 & 45" elevation Unit 2.

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With both the primary and backup Halon systems protecting the areas inoperable, within one hour establish an hourly fire watch with backup fire suppression equipment for those areas protected by the inoperable Halon system. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 12 months by performing a visual inspection of the nozzle(s) and visible flow paths for obstructions.

d. At least once per 18 months by verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

e. Following completion of major maintenance or modifications on the system(s), within 72 hours by performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour. Restore the fire hose station(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the fire hose station(s) to OPERABLE status.

b. The provision of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE;

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station. Hose stations located in the containment shall be visually inspected on each scheduled reactor shutdown, but not more frequently than every 31 days.

b. At least once per 18 months for hose stations located outside the containment and once per refueling interval for hose stations inside the containment by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years for hose stations located outside the containment and once per refueling interval for hose stations inside the containment by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater that the maximum pressure available at that hose station or replacement with a new hose.

TABLE 3.7-6

FIRE HOSE STATIONS

UNIT 2

LOCATION	ELEVATION	NUMBER OF HOSE STATIONS
1. Containment	10'	2
	45'	2
	69'	2
2. Auxiliary Building	-15'*	1**
	-10'*	2**
	5'	3
	27'	2
	45'	4
	69'*	3
3. Turbine Building, Heater Bay Outside Service Water Pump Rooms and Aux Feedwater Pump Rooms	12'	2
Outside Switchgear Room	27'	1
Outside Switchgear Room	45'	2
4. Intake Structure	10'*	1

*Fire Hose Stations required for primary protection to ensure the OPERABILITY of safety related equipment.

**Hose Stations which serve both Units 1 and 2.

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

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. DPR-53

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

   A. The applications for amendment by Baltimore Gas and Electric Company (the licensee) dated October 1, 1986 and January 20, 1987, as supplemented on February 16 and February 26, 1988, 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 1-23	3/4 1-23
3/4 1-24*	3/4 1-24*
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 4-27*	3/4 4-27*
3/4 4-28	3/4 4-28
3/4 7-67	3/4 7-67
3/4 7-68*	3/4 7-68*
3/4 7-69	3/4 7-69
3/4 7-70	3/4 7-70
3/4 7-73	3/4 7-73
3/4 7-74*	3/4 7-74*

* Overleaf pages provided to maintain document completeness.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515^{\circ}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per refueling interval.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be **OPERABLE** with at least one **OPERABLE** detector segment in each core quadratn on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the **AZIMUTHAL POWER TILT**:

   At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient **OPERABLE** detector segments in these detector groups to computer at least two **AZIMUTHAL POWER TILT** values at each of the four axial elevations containing incore detectors.

b. For recalibration of the incore neutron flux detector system:

   1. At least 75% of all incore detector segments,
   2. A minimum of 9 **OPERABLE** incore detector segments at each detector segment level, and
   3. A minimum of 2 **OPERABLE** detector segments in the inner 109 fuel assemblies and 2 **OPERABLE** segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the **UNRODDED PLANAR RADIAL PEAKING FACTOR**, the **UNRODDED INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate:

   1. At least 75% of all incore detector segments,
   2. A minimum of 9 **OPERABLE** incore detector segments at each detector segment level, and
   3. A minimum of 2 **OPERABLE** detector segments in the inner 109 fuel assemblies and 2 **OPERABLE** segments in the outer 108 fuel assemblies at each segment level.

An **OPERABLE** incore detector segment shall consist of an **OPERABLE** rhodium detector constituting one of the segments in a fixed detector string.

An **OPERABLE** incore detector location shall consist of a string in which at least three of the four incore detector segments are **OPERABLE**.

LIMITING CONDITION FOR OPERATION (Continued)

An OPERABLE quadrant symmetric incore detector segment group shall consist of a minimum of three OPERABLE rhodium incore detector segments in 90^{0} symmetric fuel assemblies.

APPLICABILITY: When the incore detection system is used for:

a. Monitoring the **AZIMUTHAL POWER TILT**,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the **UNRODDED PLANAR RADIAL PEAKING FACTOR**, the **UNRODDED INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated **OPERABLE**:

a. By performance of a **CHANNEL CHECK** within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Monitoring the **AZIMUTHAL POWER TILT**.

2. Recalibration of the excore neutron flux detection system.

3. Monitoring the **UNRODDED PLANAR RADIAL PEAKING FACTOR**, the **UNRODDED INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

b. At least once per refueling interval by performance of a **CHANNEL CALIBRATION** operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: ALL MODES.

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Per the requirements of Specification 4.0.5, and

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

SURVEILLANCE REQUIREMENTS (Continued)

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

4.4.10.1.2 Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping - The unencapsulated welds greater than 4 inches in nominal diameter in the main steam and main feedwater piping runs located outside the containment and traversing safety related areas or located in compartments adjoining safety related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria, and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition and Addenda through Summer 1983, for Class 2 components.

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

*Reactor coolant pump flywheel inspections for the first inservice inspection interval may be completed by June 1990 in conjunction with the reactor coolant pump motor overhaul program.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated **OPERABLE:**

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a **STAGGERED TEST BASIS** by starting the electric motor driven pump and operating it for at least 15 minutes. This test shall be performed on a **STAGGERED TEST BASIS** with the test required by 4.7.11.1.2.a.2.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

d. At least once per 12 months by performance of a system flush of the filled portions of the system.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each pump develops at least 2500 gpm at a discharge pressure of 125 psig,

   3. Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure $\geq$ 80 psig.

g. At least once per refueling interval by: (1) performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association, and (2) performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The diesel fuel oil day storage tank contains at least 174 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes. This test shall be performed on a STAGGERED TEST BASIS with the test required by Specification 4.7.11.1.1.b.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq$ 20 minutes while loaded with the fire pump.

4.7.11.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is $\geq$ 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioriation, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems shown in Table 3.7-5 shall be **OPERABLE**:

APPLICABILITY: Whenever equipment in the spray/sprinkler protected areas is required to be **OPERABLE**.

ACTION:

a. With one or more of the required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant safe shutdown systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required spray and/or sprinkler systems shall be demonstrated **OPERABLE**:

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

b. At least once per 12 months by cycling each valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months

    1. By performing a system functional test which includes simulated automatic actuation of the system, and verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the area in the vicinity of each nozzle(s) to verify the spray pattern will not be obstructed.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-6 shall be **OPERABLE**.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be **OPERABLE**.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an **OPERABLE** hose station within 1 hour. Restore the fire hose station(s) to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the fire hose station(s) to **OPERABLE** status.

b. The provision of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated **OPERABLE**;

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station. Hose stations located in the containment shall be visually inspected on each scheduled reactor shutdown, but not more frequently than every 31 days.

b. At least once per 18 months for hose stations located outside the containment and once per refueling interval for hose stations inside the containment by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years for hose stations located outside the containment and once per refueling interval for hose stations inside the containment by:

   1. Partially opening each hose station valve to verify valve **OPERABILITY** and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater that the maximum pressure available at that hose station or replacement with a new hose.

TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	NUMBER OF HOSE STATIONS
1. Containment	10'	2
	45'	2
	69'	2
2. Auxiliary Building	-15'*	1**
	-10'*	2**
	5'	6
	27'	3
	45'	5
	69'*	4
3. Turbine Building, Heater Bay Outside Service Water Pump Rooms and Aux Feedwater Pump Rooms	12'	3
Outside Switchgear Room	27'	2
Outside Switchgear Room	45'	3
4. Intake Structure	10'*	1

*Fire Hose Stations required for primary protection to ensure the OPERABILITY of safety related equipment.

**Hose Stations which serve both Units 1 and 2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-317 AND 50-318

INTRODUCTION

By the applications for license amendments dated October 1, 1986 and January 20, 1987, as supplemented on February 16 and February 26, 1988, the Baltimore Gas and Electric Company (BG&E, the licensee) requested changes to the Technical Specifications (TS) for Calvert Cliffs, Units 1 and 2. The TS changes proposed are as follows: (1) Modify the Unit 1 TS Limiting Condition For Operation (LCO) 3.3.3.2 for incore detectors by placing additional restrictions upon operability above those that were required for operation during the previous cycle (Cycle 8); (2) Change the surveillance periods of the Units 1 and 2 TS Surveillance Requirements (SRs) 4.1.3.4.c (demonstration of full length control element assembly (CEA) drop time) and 4.3.3.2.b (incore detector channel calibration) from at least once per 18 months to at least once per refueling interval, where a refueling interval shall be defined as 24 months; (3) Modify the Units 1 and 2 TS SR 4.7.11.1.1.f.3 for cycling fire suppression water system flow path valves that are not testable during plant operation, and 4.7.11.4.b, for the inspection, reracking and replacement of degraded coupling gaskets for fire hoses inside containment by extending their associated surveillance intervals from at least once every 18 months to at least once per refueling interval (24 months); (4) Renumber the Units 1 and 2 TS SR 4.7.11.1.1.f.3 as 4.7.11.1.1.g(2); TS SR 4.7.11.1.1.g as 4.7.11.1.1.g(1); and TS SR 4.7.11.1.1.f.4 as 4.7.11.1.1.f.3 and change the Units 1 and 2 TS SRs 4.7.11.1.1.g (fire suppression system flow test), 4.7.11.2.b and c (spray and sprinkler system functional test), and 4.7.11.4.c (containment fire hose stations operability and hydrostatic tests) by making administrative changes and more restrictive changes to the surveillance requirements; and (5) Change the Units 1 and 2 TS SR 4.4.10.1.2, "Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping," to update the required ASME Boiler and Pressure Vessel Code, Section XI, for Class 2 components from the 1974 Edition and addenda through Summer 1975 to the 1983 Edition with Addenda through Summer 1983. In addition, TS SR 4.4.10.1.2.a would be deleted and TS SR 4.4.10.1.2.b would be renumbered as 4.4.10.1.2 and would be clarified to reflect a new 10-year inservice inspection interval.

The February 16, and February 26, 1988 submittals provided camera-ready copies of the proposed TS changes as were requested by the licensee on January 20, 1987. The supplement to the January 20, 1987 submittal did not affect the proposed TS changes noticed in the Federal Register on January 13 1987 and did not affect the staff's proposed no significant hazards determination.

DISCUSSION AND EVALUATION

Change No. 1 proposed in the January 20, 1987 submittal to modify the Unit 1 TS LCO 3.3.3.2 for incore detector operability by making its provisions more restrictive than those required for Unit 1 Cycle 8 operation. During startup for Unit 1 Cycle 8, an unexpectedly large number of incore detector strings failed thereby placing Unit 1 close to its operability limits. To provide increased operational flexibility for Unit 1 during Cycle 8 operations, the requirements of TS LCO 3.3.3.2 were relaxed for one cycle only. In order to restore LCO 3.3.3.2 to its pre-cycle 8 requirements, the following modifications were proposed:

(1) LCO 3.3.3.2.a would require at least eight operable symmetric incore detector segment groups, with at least two of these detector segment groups at each of the four axial elevations containing incore detectors, to have sufficient operable detector segments to compute at least two azimuthal power tilt values at each of these four axial elevations. During Cycle 8, eight symmetric incore detector segment groups of no specified elevation were required with sufficient operable detector segments to compute at least two azimuthal power tilt values at three of the four axial elevations.

(2) LCO 3.3.3.2.b would require that at least 75% of all incore detector segments be operable for recalibration of the excore neutron flux detection system rather than the 50% required during Cycle 8.

(3) LCO 3.3.3.2.c would require, for monitoring the unrodded planar radial peaking factor, the unrodded integrated radial peaking factor, or the linear heat rate, that at least 75% of all incore detector locations be operable rather than the 50% required during Cycle 8.

As these proposed changes to TS LCO 3.3.3.2 are all more restrictive in nature, restoring requirements that were relaxed for Unit 1 Cycle 8, and as these proposed changes improve incore detector system performance, the NRC staff has deemed these proposed changes to be acceptable.

In the January 20, 1987 submittal, Change No. 2 proposed to extend the surveillance periods from 18 to 24 months for the Units 1 and 2 TS SRs for demonstrating full length CEA drop time (TS SR 4.1.3.4.c) and for performing incore detector channel calibrations (TS SR 4.3.3.2.b).

The current surveillance period for these test is 18 months which corresponds to the current refueling cycle. The extension in the surveillance interval to 24 months is requested to facilitate a 24-month operating cycle.

According to the current TS SR 4.1.3.4, the drop time of each full length CEA must be verified to be less than or equal to 3.1 seconds (1) following each removal of the reactor vessel head (TS SR 4.1.3.4.a), (2) following maintenance or modification of the CEA drive system which could affect specific CEA drop times (TS SR 4.1.3.4.b), and (3) at least once per 18 months (TS SR 4.1.3.4.c). The CEA drop time is measured from the time that electrical power is interrupted to a fully withdrawn CEA to the time required for the CEA to be at its 90% insertion position. This drop time testing is performed at a reactor coolant system average temperature greater than or equal to 515° F and with all four reactor coolant pumps operating. These conditions are representative of reactor conditions for reactor trips from operating conditions. The purpose of the CEA drop time testing is to ensure that scram insertion times are consistent with those used in the safety analyses.

To justify changing TS SR 4.1.3.4.c to state "at least once per refueling interval" (24 months) instead of "at least once per 18 months", BG&E analyzed CEA drop time measurements from 15 hot functional sets of test data. Eight sets of measurements were from Unit 1 and seven from Unit 2. The licensee found that the average CEA drop time for standard fuel assemblies is approximately 2.3 seconds. The maximum standard deviation for drop times from any fuel cycle is 0.094 seconds. The 15 sets of test data included data from both 12 month and 18 month fuel cycles. The licensee concluded that the data indicate that no increase in drop time trend is observed for either longer fuel cycles or due to increased periods between surveillance testing.

Factors which could adversely affect the CEA drop times when the surveillance interval is increased are (1) changes in component clearances, (2) changes in the physical configuration of the CEA or guide tubes, and (3) the buildup of corrosion products and suspended material in the coolant system that could interfere with CEA motion. The licensee stated that changes to component clearances and changes in the physical configuration of the CEA or guide tubes are more likely to occur when the reactor vessel head is removed and when maintenance is performed on the CEAs (including replacement) and that portion of the drive system directly interfacing with a fuel assembly. For these two factors, TS SRs 4.1.3.4.a and 4.1.3.4.b are applicable and not affected by the proposed change in the testing interval of TS SR 4.1.2.3.c. The licensee stated that corrosion products and suspended material in the coolant system are minimized by coolant chemistry requirements and other controls on the reactor coolant system. In addition, each full-length CEA is exercised at least once per 31 days in accordance with TS SR 4.1.3.1.2. The testing required by this TS SR should detect sticking CEAs. Each planned or unplanned reactor trip that may occur during extended 24 month fuel cycles would provide additional information on CEA drop times and operability.

The staff concurs with the licensee's assessment that extending the interval of TS SR 4.1.3.4.c from 18 months to at least once per refueling interval is acceptable. This concurrence is based on the licensee's analysis of previous fuel cycles CEA drop time measurements which do not exhibit any adverse effects for 18-month cycles as compared to 12-month cycles and on a review of other relevant factors which could adversely affect CEA drop times but are covered by other TS SRs.

Currently, the incore detection system must be demonstrated to be operable at least once per 18 months by performance of a channel calibration in accordance with TS SR 4.3.3.2.b. This channel calibration excludes the neutron detectors but includes all electronic components. The channel calibration consists of two parts: (1) a resistance check of the cable from the computer termination to the reactor core, and (2) a check of the ability of the computer to read a known voltage level. The resistance check verifies cable integrity. The licensee has reviewed tests performed since the initial startup of Calvert Cliffs Units 1 and 2. No evidence of cable degradation was found. The licensee is, however, in the process of replacing the in-containment cable with environmentally qualified cable. The design specification for the new cable will ensure that it is at least as reliable as the cable it replaces.

The second part of the channel calibration checks the computer's ability to read a know voltage level. Three known inputs are input into the computer: (1) a short circuit, (2) a 150 millivolt signal, and (3) a 250 millivolt signal. Proper computer readings are verified for each test with the voltages being between ± 2 millivolts. Other checks to verify proper computer operation are also performed and include CRT and alarm printer verification. The licensee reviewed test data from initial plant startup to the present time and reports that this test has been consistently performed satisfactorily.

To justify changing TS SR 4.3.3.2.b to state "at least once per refueling interval" (24 months) instead of "at least once per 18 months", the licensee stated that no adverse trends have been observed for test data either over time or due to the shift from 12-month to 18-month fuel cycles. In addition, performance of the power distribution TS SRs 4.2.2.1.2 and 4.2.3.2, conducted at least once per 31 Mode 1 days, provides further assurance of incore detection system operability in that an inoperable incore detector segment would probably be apparent due to the resultant skew of the peaking factors calculated through these surveillances. The licensee stated that, with the incore detector system inoperable, other methods are employed to carry out its monitoring and calibration functions.

The staff concurs with the licensee's assessment that extending the interval of TS SR 4.3.3.2.b from 18 months to at least once per refueling interval is acceptable. This concurrence is based on the licensee's analysis of previous fuel cycles' incore detection system calibration data which do not exhibit any adverse trends for 18 month fuel cycles as compared to 12 month fuel cycles and on power distribution that are imposed at least once every 31 Mode 1 TS SRs days, which will provide a check of anomalous incore detector readings.

The proposal to modify TS SR 4.7.11.1.1.f.3 affects only two fire suppression water system valves inside containment. LCO 3.7.11.1.c requires at all times an operable fire suppression water system flow path that takes a suction from the water storage tanks and transfers the water through the distribution system up to the first valve before the water flow alarm device on each sprinkler, hose standpipe or spray system riser. All valves in this flow path can be tested during unit operation with the exception of the two valves inside containment (the motor operated containment isolation valve and a manual block valve). TS SR 4.7.11.1.1.f.3 requires these two valves to be tested by cycling and verifying flow. The licensee's results from a review of plant history indicate that there has never been a failure of either valve to perform adequately. The licensee further states that there is no evidence that a 6-month extension in this surveillance interval between valve cycles would adversely impact valve operation. Hence, the probability or consequences of previously evaluated accidents would not be significantly increased by the proposed 6-month extension of the surveillance interval of TS SR 4.7.11.1.1.f.3.

The proposed modification of TS SR 4.7.11.4.b would affect only the inspection and reracking of fire hoses inside containment. A review of previously conducted containment fire hose inspections revealed no failures of the fire hoses. The licensee stated that these results were expected as it has been a licensee policy to replace all fire hoses inside containment on a three-year frequency. The licensee intends, for the 24-month operating cycle, to hydrostatically test or replace all containment fire hoses every two years.

Furthermore, test results have shown that the hose coupling gasket material has not degraded significantly over the three-year interval between hose replacements. Finally, during hose inspection, there has never been evidence of hose mildew, rot or similar damage due to chemicals, abrasion, moisture or normal wear. Thus, it is unlikely that the containment fire hoses would experience any significant degradation over the proposed 6-month surveillance interval extension.

The licensee has requested that the surveillance interval of only those tests that could not be performed during unit operation (i.e., testing and inspecting fire hoses and fire suppression water system valves inside containment) be extended to a 24-month cycle. These containment fire protection components to be tested are generally inaccessible during unit operation due to ALARA consideration, and so, will be tested during refueling outages. However, the likelihood of a fire inside containment during unit operation is much smaller than during outage work periods. Thus, the likelihood of a fire occurring inside containment, that would damage safety and safety-related systems, will not be significantly increased by this proposed 6-month test interval extension. Therefore, the margins of safety provided by these safety and safety-related systems will not be significantly reduced.

Finally, the new surveillance frequencies of these TS requirements conform with the guidance provided in National Fire Protection Association Standards Nos. 13A and 1962.

For all of the above reasons, the staff concludes that the changes proposed for TS SRs 4.7.11.1.1.f.3 and 4.7.11.4.b are acceptable.

Change No. 4 proposes to renumber the Units 1 and 2 TS SR 4.7.11.1.1.f.3 as 4.7.11.1.1.g(2); TS SR 4.7.11.1.1.g as 4.7.11.1.1.g(1); and TS SR 4.7.11.1.1.f.4 as 4.7.11.1.1.f.3 and to modify the Units 1 and 2 TS SRs 4.7.11.1.1.g, 4.7.11.2.b & c and 4.7.11.4.c by making more restrictive changes to the current surveillance requirements. These changes were requested in the January 20, 1987 submittal. The proposed restrictive changes to the surveillance requirements are as follows:

(1) the surveillance interval for performing a fire suppression water system flow test in accordance with TS SR 4.7.11.1.g would be changed to "at least once per refueling interval" (24 months) from the currently required "at least once per 3 years,"

(2) the spray and sprinkler system cycling test of each flow path valve would be conducted at least every 12 months. Currently, only testable valves are required to be cycled at least every 12 months by TS SR 4.7.11.2.b, whereas TS SR 4.7.11.2.c.1.b requires the cycling of those not testable during plant operation at least every 18 months. All of these valves, however, are testable during plant operation, making TS 4.7.11.2.c.1.b superfluous. Consequently, the licensee has proposed deletion of TS 4.7.11.2.c.1.b and of the word "testable" from the phrase "by cycling each testable valve" in TS 4.7.11.2.b,

(3) fire hose station valve operability and hose hydrostatic tests currently are required by TS 4.7.11.4.c to be performed at least once per 3 years. The licensee has proposed that these tests on fire hose stations inside containment be required to be performed at least once refueling interval (24 months).

These proposed administrative and restrictive changes, as described above, will provide equivalent or improved fire protection and suppression capability as compared to the current TS requirements. In addition, these proposed TS surveillance requirements conform with the fire protection guidance provided in National Fire Protection Association Standards Nos. 13A and 1962. Therefore, the NRC staff finds these changes, as proposed, to be acceptable.

As provided in the October 1, 1986 amendment request, Change No. 5 proposes to update the Units 1 and 2 TS SR 4.4.10.1.2 to the requirements of the 1983 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, with Addenda through Summer 1983. Currently, the licensee is required to comply with the 1974 Edition with Addenda through Summer 1975.

The first 10-year ASME Code Section XI inservice inspection (ISI) interval ended on April 1, 1987 for Calvert Cliffs Unit 1 and on July 3, 1987 for Unit 2. Section 50.55a(g)(4)(ii) of 10 CFR requires that inservice examinations during successive 120-month (10 year) ISI intervals comply with the requirements of the latest edition of the ASME Code that was incorporated in 10 CFR 50.55a(b)(2) twelve months prior to the start of the ISI interval. The latest edition and addenda of the ASME Code in 10 CFR 50.55a(b)(2) on April 1, 1986 was the 1983 Edition with Addenda through Summer 1983.

When 10 CFR 50.55a(b)(2) was changed to require use of the 1983 Edition of the ASME Code for ISI, the NRC staff determined that the use of the 1983 Edition vice use of the previously required 1974 Edition was preferential as adoption of the 1983 Code Edition would permit the use of improved methods for inservice inspection of nuclear power plants.

The changes to TS SR 4.4.10.1.2 that result from shifting to the 1983 Edition of the ASME Code provide requirements that are at least as conservative as those provided by the 1974 Edition. Furthermore, this change does not impact any TS SRs other than those specifically set forth in TS SR 4.4.10.1.2.

In addition, the licensee has proposed the deletion of TS SR 4.4.10.1.2.a and the renumbering and clarification of TS SR 4.4.10.1.2.b. These changes are purely administrative. TS SR 4.4.10.1.2.a was provided to establish baseline data during the first 18 months of plant operation. This one-time requirement has been satisfied as this baseline data has been established. Thus, this surveillance requirement is moot. TS SR 4.4.10.1.2.b was clarified to reflect entry into 10-year intervals that are subsequent to the first 10-year interval. This change has no other practical impact upon this surveillance requirement.

Therefore, for the reasons give above, the NRC staff has determined that the changes to TS SR 4.4.10.1.2 proposed to reflect the licensee's update to the 1983 ASME Code are acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change to requirements with respect to the installation or use of the facilities' components located within the restricted areas as defined in 10 CFR 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 3, 1988

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