September 28, 1990

Docket Nos. 50-317
and 50-318

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OC/LFMB	Plant File
MWaterman	JLinville

Mr. G. C. Creel
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
MD Rts. 2 & 4
P. O. Box 1535
Lusby, Maryland 20657

Dear Mr. Creel:

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1 (TAC NO. 54523) AND UNIT 2 (TAC NO. 54524)

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. DPR-53 and Amendment No. 148 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 in response to your application transmitted by letter dated July 10, 1987, as supplemented on August 31, 1989, and August 3, 1990.

These amendments add Technical Specifications (TS) relating to surveillance, operability, and reporting requirements for the reactor vessel level monitoring system (RVLMS) to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, TS 3/4.3.3.6, "Post Accident Instrumentation," including TS Tables 3.3-10 and 4.3-10, and TS 6.9.2, "Special Reports." In addition, the applicable Bases sections of the TS are updated to reflect the proposed changes.

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:

Daniel G. McDonald, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 147 to DPR-53
2. Amendment No. ~~148~~ 128 to DPR-69
3. Safety Evaluation

cc w/enclosures:
See next page

PDI-1	PDI-1	OGC	PDI-1
CVogan	DMcDonald:rsc	R Bachmann	for RACapra
9-6-90 9/10	9/28/90	9/19/90	9/25/90

DOCUMENT NAME: AMEND 54523/524 RVLMS

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

September 28, 1990

Docket Nos. 50-317
and 50-318

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Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
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Sincerely,

Daniel G. McDonald Jr

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Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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1. Amendment No. 147 to DPR-53
2. Amendment No. 148 to DPR-69
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. G. C. Creel
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

cc:

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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. 147
License No. DPR-53

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

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated July 10, 1987, as supplemented on August 31, 1989, and August 3, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 147, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel McDonald Jr

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 28, 1990

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. 148
License No. DPR-69

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

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated July 10, 1987, as supplemented on August 31, 1989, and August 3, 1990, 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. 148, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald Jr

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 28, 1990

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 147 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 148 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-41	3/4 3-41
3/4 3-41a	3/4 3-41a
3/4 3-42	3/4 3-42
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4*	B 3/4 3-4*
6-18*	6-18*
6-18a	6-18a

*Pages that did not change, but are overleaf

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure	2	31
2. Wide Range Logarithmic Neutron Flux Monitor	2	31
3. Reactor Coolant Outlet Temperature	2	31
4. Pressurizer Pressure	2	31
5. Pressurizer Level	2	31
6. Steam Generator Pressure	2/steam generator	31
7. Steam Generator Level (Wide Range)	2/steam generator	31
8. Auxiliary Feedwater Flow Rate	2/steam generator	31
9. RCS Subcooled Margin Monitor	1	31
10. PORV/Safety Valve Acoustic Flow Monitoring	1/valve	31
11. PORV Solenoid Power Indication	1/valve	31
12. Feedwater Flow	2	31
13. Containment Water Level (Wide Range)	2	32, 33
14. Reactor Vessel Water Level	2*	34, 35

* A channel has eight sensors in a probe. A channel is operable if four or more sensors, one or more in the upper three and three or more in the lower five, are operable.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of **OPERABLE** post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to **OPERABLE** status within 30 days or be in **HOT SHUTDOWN** within the next 12 hours.

ACTION 32 - With the number of **OPERABLE** post-accident monitoring channels one less than the minimum channel operable requirement in Table 3.3-10, operation may proceed provided the inoperable channel is restored to **OPERABLE** status at the next outage of sufficient duration.

ACTION 33 - With the number of **OPERABLE** post-accident monitoring channels two less than required by Table 3.3-10, either restore one inoperable channel to **OPERABLE** status within 30 days or be in **HOT SHUTDOWN** within the next 12 hours.

ACTION 34 - With the number of **OPERABLE** Post-Accident Monitoring Channels one less than the minimum Channel **OPERABLE** requirement in Table 3.3-10, either restore the system to **OPERABLE** status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

ACTION 35 - With the number of **OPERABLE** Channels two less than required by Table 3.3-10, either restore the inoperable channel(s) to **OPERABLE** status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring for core and reactor coolant system voiding;
2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status; and
3. Restore the system to **OPERABLE** status at the next scheduled refueling.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

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

The subcooled Margin Monitor (SMM), the Heated Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to, and recovery from, ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These instruments are not required by the accident analysis, nor to bring the plant to **HOT STANDBY** or **COLD SHUTDOWN**.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during an extended **COLD SHUTDOWN**. This is because the sensors are accessible only after the plant has been cooled down and drained, and the missile shield has been moved. If only one channel is inoperable, it should be restored to **OPERABLE** status in accordance with the schedule outlined in a Special Report. If both channels are inoperable, the system shall be restored to **OPERABLE** status in the next refueling outage.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

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

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

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.11.2.1.a based on average annual X/Q. The **OPERABILITY** and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/tip will occur prior to exceeding the limits of 10 CFR Part 20. The **OPERABILITY** and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed **MEMBER OF THE PUBLIC** from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977, and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants".

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Solidification agent or absorbent (e.g., cement).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to **UNRESTRICTED AREAS** of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the **PROCESS CONTROL PROGRAM** (PCP), and to the **OFFSITE DOSE CALCULATION MANUAL** (ODCM), as well as a listing of new locations for dose calculations identified by the annual land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents - Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program, Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

r. Post-Accident Instrumentation, Specification 3.3.3.6

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

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4. Pressurizer Pressure	2	31
5. Pressurizer Level	2	31
6. Steam Generator Pressure	2/steam generator	31
7. Steam Generator Level (Wide Range)	2/steam generator	31
8. Auxiliary Feedwater Flow Rate	2/steam generator	31
9. RCS Subcooled Margin Monitor	1	31
10. PORV/Safety Valve Acoustic Flow Monitoring	1/valve	31
11. PORV Solenoid Power Indication	1/valve	31
12. Feedwater Flow	2	31
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2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status; and

3. Restore the system to **OPERABLE** status at the next scheduled refueling.

TABLE 4.3-10

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6. Steam Generator Pressure	M	R
7. Steam Generator Level (Wide Range)	M	R
8. Auxiliary Feedwater Flow Rate	M	R
9. RCS Subcooled Margin Monitor	M	R
10. PORV/Safety Valve Acoustic Monitor	N.A.	R
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12. Feedwater Flow	M	R
13. Containment Water Level (Wide Range)	M	R
14. Reactor Vessel Water Level	M	N.A.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

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3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.11.2.1.a based on average annual X/Q. The **OPERABILITY** and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/tip will occur prior to exceeding the limits of 10 CFR Part 20. The **OPERABILITY** and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed **MEMBER OF THE PUBLIC** from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977, and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants".

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Solidification agent or absorbent (e.g., cement).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to **UNRESTRICTED AREAS** of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the **PROCESS CONTROL PROGRAM (PCP)**, and to the **OFFSITE DOSE CALCULATION MANUAL (ODCM)**, as well as a listing of new locations for dose calculations identified by the annual land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents - Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program, Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

r. Post-Accident Instrumentation, Specification 3.3.3.6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 148 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

1.0 INTRODUCTION

By letter dated July 10, 1987, as supplemented on August 13, 1989, and August 3, 1990, Baltimore Gas and Electric Company (BG&E) requested amendments to the Technical Specifications (TS) for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (CC-1/2). The proposed amendments address surveillance, operability, and reporting requirements for the Reactor Vessel Level Monitoring System (RVLMS).

The RVLMS is one of three systems used to indicate the potential for inadequate core cooling. The other two systems are the Subcooling Margin Monitor (SMM) and the Core Exit Thermocouples (CETs). As stated by the licensee, the purpose of the RVLMS is to help the operator 1) track an event, 2) assess the functioning of automatic equipment, and 3) detect the consequences of mitigating actions.

The RVLMS uses a Combustion Engineering Heated Junction Thermocouple (HJTC) system to detect the presence of highly voided coolant above the top of the reactor core. The presence of highly voided coolant is inferred from the differential temperatures between vertically adjacent HJTC probes. If two HJTCs indicate approximately the same temperature, the operator can infer that the same thermodynamic conditions exist at both HJTC locations. If adjacent HJTCs indicate significantly different temperatures, the operator can infer that the hotter HJTC is in a highly voided region of the vessel, and the cooler HJTC is still in a region of the vessel that has low-void-fraction coolant.

The July 10, 1987, request was initially noticed on May 18, 1988 (53 FR 17777). Subsequently, by letter dated August 3, 1990, BG&E responded to the NRC staff's request for additional information in relation to inoperable RVLMS channels and calibration requirements. The August 3, 1990, BG&E letter modified the request by changing TS Table 3.3-10 to require that the system be restored to operable status at the next refueling instead of the initial request which required only a single channel be restored to operable status at the next refueling.

2.0 BACKGROUND

The NRC staff provided guidance which included a TS action statement that required a plant shutdown within seven days if less than two channels of RVLMS were OPERABLE, and within 48 hours if less than one channel was OPERABLE. The NRC staff and the Combustion Engineering Owners Group (CEOG) discussed the safety significance of not having the RVLMS available, and concluded that the proposed TS was too restrictive. The CEOG submitted a less restrictive TS, which was then proposed for Palo Verde Unit 1, which the NRC staff reviewed and approved.

The Palo Verde Unit 1 TS requires that, with one RVLMS channel inoperable, either the system is restored to OPERABLE Status within seven days or a Special Report must be submitted to the NRC within 30 days detailing the cause of the inoperability and the schedule for restoring the system to OPERABLE status. With the number of OPERABLE channels two less than required (no OPERABLE channels), operation may continue until the next scheduled refueling provided an alternative method of monitoring for core and reactor coolant system voiding is available, and the licensee submits a Special Report to the NRC within 30 days detailing the cause of the inoperability and the schedule for restoring the system to OPERABLE status. The BG&E justification for the requested TS amendments is based on the staff's approval of the generic CEOG TS for Palo Verde Unit 1.

3.0 EVALUATION

The staff accepts the licensee's proposed footnote at the bottom of TS Table 3.3-10. This footnote defines an OPERABLE RVLMS channel as eight sensors in a probe, consisting of one or more operable sensors in the upper three, and three or more operable sensors in the lower five. This footnote is consistent with the previously-approved TS definition of an operable RVLMS channel.

The licensee proposes changing the phrase "reactor coolant system inventory" to "reactor coolant system voiding" in Action Statement 35.1 of the NRC staff's guidance RVLMS Technical Specifications, resulting in: "Initiate an alternative method of monitoring for reactor coolant system voiding." The licensee indicates that this statement is more consistent with the Emergency Operating Procedures (EOPs) that give guidance in determining whether voids exist in the reactor vessel and reactor coolant system. The staff finds this change acceptable.

The licensee initially proposed to substitute the words "at least one channel" for "the system" in Action Statement 35.3 of the TS, resulting in: "Restore at least one channel to OPERABLE status at the next refueling." The licensee claims that the use of "system" in Action Statement 35.3 is ambiguous because it does not define whether the "system" is comprised of one or both RVLMS channels. This statement is not consistent with the licensee's description of the RVLMS on Page 2 of their July 10, 1987, submittal, which states, "The RVLMS [Reactor Vessel Level Monitoring System] consists of two independent safety channels. Each channel consists of a probe assembly, support tube, pressure boundary modifications, signal processing equipment and an operator interface." The licensee's letter dated August 3, 1990, modified the initial request from "at least one channel" to the "system." This is consistent with its definition of "system," as discussed above, and is also consistent with the generic guidance provided. Therefore, the NRC staff finds this acceptable.

In an NRC staff request for additional information dated August 3, 1989, the licensee was asked to address return to 100% plant power with both RVLMS channels inoperable, based on the location of the failure (inside the reactor vessel, inside the containment, or outside the containment) for operating Modes 1-5 prior to refueling, Mode 6 (refueling), and Modes 1-5 post-refueling.

The licensee's August 31, 1989, submittal states that in Modes 1-5 prior to refueling or in Mode 6, the plant cannot return to 100% power. This action is consistent with the NRC staff's guidance and is acceptable.

The licensee states that in Modes 1-5 after refueling, the plant can return to 100% power if the failure is inside the reactor vessel, because replacement of a channel is not feasible during post-refueling Modes 1-5, and there is sufficient redundant instrumentation to monitor reactor coolant system void indications. Given the diversity of instrumentation available to detect reactor coolant system void indications and the necessity to cool down the plant prior to replacing a failed channel, the licensee's response is acceptable.

The licensee states that the plant can return to 100% power if a RVLMS failure occurs inside containment when the plant is in Modes 1-5 after refueling. The licensee's justification is based on minimizing man-rem doses and short term parts availability considerations. The licensee qualifies their position by stating, "All attempts will be made to repair at least one channel. If repairs are not feasible, the Units can be operated until the next refueling shutdown as long as alternate methods of void detection are initiated." The staff notes that TS action statements concerning loss of an alternate method of monitoring voiding in the reactor coolant system (either SMM or CET) that cannot be restored to OPERABLE will force a plant shutdown. Therefore, the staff accepts the licensee's proposed criteria.

The licensee states that if a RVLMS channel becomes inoperable due to failures outside the containment during post-refueling Mode 1-5 operations, the units could return to 100% power. The licensee commits to making all attempts to repair at least one channel as soon as possible. This response is only qualitative, but does indicate a commitment on the part of the licensee to address channel inoperability when it occurs. Given the number of diverse systems for indicating the potential for inadequate core cooling, and the licensee's commitment to repair failed RVLMS channels as soon as possible, the staff finds this portion of the licensee's requested amendment acceptable.

The licensee states that the calibration frequency for the HJTC probes should be changed from R (Refueling interval) to N/A (Table 4.3-10, "Post-Accident Monitoring Instrumentation Surveillance Requirements") because, once installed, the RVLMS sensors cannot be recalibrated. Consequently, a channel check is the only surveillance performed on the HJTCs. Based on the system design and the alternate methods of monitoring voids, as previously discussed, the staff finds this response acceptable.

4.0 SUMMARY

The NRC staff has determined that the proposed TS surveillance, operability, and reporting requirements for the RVLMS at the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, are acceptable based on the above discussion.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change 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 Section 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.

6.0 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 wiht 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: September 28, 1990

PRINCIPAL CONTRIBUTORS:

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