UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 24, 1994

Docket Nos. 50-317
and 50-318

Mr. Robert E. Denton
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

Dear Mr. Denton:

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M88164) AND UNIT NO. 2 (TAC NO. M88165)

The Commission has issued the enclosed Amendment No. 192 to Facility Operating License No. DPR-53 and Amendment No. 169 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 November 3, 1993.

The amendments modify the surveillance requirements to reflect the removal of the auto-closure interlock from the shutdown cooling system and revises the setpoint for the open permissive interlock.

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

Sincerely,

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. 192 to DPR-53
2. Amendment No. 169 to DPR-69
3. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Robert E. Denton
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2

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DATED: August 24, 1994

AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-53-CALVERT CLIFFS UNIT 1
AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-69-CALVERT CLIFFS UNIT 2

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
C. Miller, 14/A/4
P. T. Kuo
C. Vogan
D. McDonald
OGC
D. Hagan, 3302 MNBB
C. Liang, 8/E/23
G. Hill (4), P1-22
C. Grimes, 11/F/23
H. Balukjian, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 192
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 November 3, 1993, 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. 192 , 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

P. T. Kuo, Acting 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: August 24, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 November 3, 1993, 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. 169, 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

P. T. Kuo, Acting 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: August 24, 1994

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 192 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 169 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 5-5	3/4 5-5
B 3/4 5-2	B 3/4 5-2
B 3/4 5-3	B 3/4 5-3

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per **REFUELING INTERVAL** by:

1. Verifying the Shutdown Cooling System open-permissive interlock prevents the Shutdown Cooling System suction isolation valves from being opened with a simulated or actual RCS pressure signal of $\geq$ 309 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4.0 $\pm$ 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 $\pm$ 0.1 liters of 77 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

f. At least once per **REFUELING INTERVAL,** during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection Pump.

   b. Low-Pressure Safety Injection Pump.

pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

Portions of the Low Pressure Safety Injection (LPSI) System flowpath are common to both subsystems. This includes the LPSI flow control valve, CV-306, the flow orifice downstream of CV-306, and the four LPSI loop isolation valves. Although the portions of the flowpath are common, the system design is adequate to ensure reliable ECCS operation due to the short period of LPSI System operation following a design basis Loss of Coolant Incident prior to recirculation. The LPSI System design is consistent with the assumptions in the safety analysis.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to $\geq$ 7.0. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

The Surveillance Requirements provided to ensure **OPERABILITY** of each component ensure that as a minimum, the assumptions used in the safety analyses are met and the subsystem **OPERABILITY** is maintained. The surveillance requirement for flow balance testing provides assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Minimum HPSI flow requirements for temperatures above 365°F are based upon small break LOCA calculations which credit charging pump flow following an SIAS. Surveillance testing includes allowances for instrumentation and system leakage uncertainties. The 470 gpm requirement for minimum HPSI flow from the three lowest flow legs includes instrument uncertainties but not system check valve leakage. The **OPERABILITY** of the charging pumps and the associated flow paths is assured by the Boration System Specification 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance is consistent with safety analysis assumptions.

The surveillance requirement for the Shutdown Cooling (SDC) System open-permissive interlock provides assurance that the SDC suction isolation valves are prevented from being remotely opened when the RCS pressure is at or above the SDC System design suction pressure of 350 psia. The suction piping to the LPSI pumps is the SDC System component with the limiting design pressure rating. The interlock provides assurance that double isolation of the SDC System from the RCS is preserved whenever RCS pressure is at or above the SDC System design pressure. The 309 psia value specified for this surveillance is the actual pressurizer pressure at the instrument tap elevation for PT-103 and PT-103-1 when the SDC System suction pressure is 350 psia. The Surveillance Test Procedure for this surveillance will contain the required compensation to be applied to this value to account for instrument uncertainties. This test is performed using a simulated RCS pressure input.

At indicated RCS temperatures of 365°F and less, HPSI injection flow is limited to less than or equal to 210 gpm except in response to excessive reactor coolant leakage. With excessive RCS leakage (LOCA), make-up requirements could exceed an HPSI flow of 210 gpm. Overpressurization is prevented by controlling other parameters, such as RCS pressure and subcooling. This provides overpressure protection in the low temperature region. An analysis has been performed which shows this flow rate is more than adequate to meet core cooling safety analysis assumptions. HPSI pumps are not required to auto-start when the RCS is in the MPT enable condition. The Safety Injection Tanks provide immediate injection of borated water into the core in the event of an accident, allowing adequate time for an operator to take action to start a HPSI pump.

Surveillance testing of HPSI pumps is required to ensure pump **OPERABILITY**. Some surveillance testing requires that the HPSI pumps deliver flow to the RCS. To allow this testing to be done without increasing the potential for overpressurization of the RCS, either the RWT must be isolated or the HPSI pump flow must be limited to less than or equal to 210 gpm or an RCS vent greater than 2.6 square inches must be provided.

3/4.5.4 REFUELING WATER TANK (RWT)

The **OPERABILITY** of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per **REFUELING INTERVAL** by:

1. Verifying the Shutdown Cooling System open-permissive interlock prevents the Shutdown Cooling System suction isolation valves from being opened with a simulated or actual RCS pressure signal of $\geq$ 309 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 100 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4.0 $\pm$ 0.1 grams of TSP from a TSP storage basket is submerged, without agitation, in 3.5 $\pm$ 0.1 liters of 77 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

f. At least once per **REFUELING INTERVAL,** during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection Pump.

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Portions of the Low Pressure Safety Injection (LPSI) System flowpath are common to both subsystems. This includes the LPSI flow control valve, CV-306, the flow orifice downstream of CV-306, and the four LPSI loop isolation valves. Although the portions of the flowpath are common, the system design is adequate to ensure reliable ECCS operation due to the short period of LPSI System operation following a design basis Loss of Coolant Incident prior to recirculation. The LPSI System design is consistent with the assumptions in the safety analysis.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to ≥ 7.0. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.

The Surveillance Requirements provided to ensure **OPERABILITY** of each component ensure that at a minimum, the assumptions used in the safety analyses are met and the subsystem **OPERABILITY** is maintained. The surveillance requirement for flow balance testing provides assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. Minimum HPSI flow requirements for temperatures above 305°F are based upon small break LOCA calculations which credit charging pump flow following a SIAS. Surveillance testing includes allowances for instrumentation and system leakage uncertainties. The 470 gpm requirement for minimum HPSI flow from the three lowest flow legs includes instrument uncertainties but not system check valve leakage. The **OPERABILITY** of the charging pumps and the associated flow paths is assured by the Boration System Specifications 3/4.1.2. Specification of safety injection pump total developed head ensures pump performance is consistent with safety analysis assumptions.

The surveillance requirement for the Shutdown Cooling (SDC) System open-permissive interlock provides assurance that the SDC suction isolation valves are prevented from being remotely opened when the RCS pressure is at or above the SDC System design suction pressure of 350 psia. The suction piping to the LPSI pumps is the SDC System component with the limiting design pressure rating. The interlock provides assurance that double isolation of the SDC System from the RCS is preserved whenever RCS pressure is at or above the SDC System design pressure. The 309 psia value specified for this surveillance is the actual pressurizer pressure at the instrument tap elevation for PT-103 and PT-103-1 when the SDC System suction pressure is 350 psia. The Surveillance Test Procedure for this surveillance will contain the required compensation to be applied to this value to account for instrument uncertainties. This test is performed using a simulated RCS pressure input.

At temperatures of 305°F and less, HPSI injection flow is limited to less than or equal to 210 gpm except in response to excessive reactor coolant leakage. With excessive RCS leakage (LOCA), make-up requirements could exceed a HPSI flow of 210 gpm. Overpressurization is prevented by controlling other parameters, such as RCS pressure and subcooling. This provides overpressure protection in the low temperature region. An analysis has been performed which shows this flow rate is more than adequate to meet core cooling safety analysis assumptions. HPSI pumps are not required to auto-start when the RCS is in the MPT enable condition. The Safety Injection Tanks provide immediate injection of borated water into the core in the event of an accident, allowing adequate time for an operator to take action to start an HPSI pump.

Surveillance testing of HPSI pumps is required to ensure pump operability. Some surveillance testing requires that the HPSI pumps deliver flow to the RCS. To allow this testing to be done without increasing the potential for overpressurization of the RCS, either the RWT must be isolated or the HPSI pump flow must be limited to less than or equal to 210 gpm or an RCS vent greater than or equal to 2.6 square inches must be provided.

3/4.5.4 REFUELING WATER TANK (RWT)

The **OPERABILITY** of the RWT as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated November 3, 1993, the Baltimore Gas and Electric Company (BG&E, the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would modify portions of the surveillance requirements in TS 4.5.2.e.1 and its associated TS Bases. The purpose of the requested TS changes is to support removal of the Autoclosure Interlock (ACI) which provides automatic isolation of the Shutdown Cooling (SDC) System suction line isolation valves. In addition, the setpoint for the Open Permissive Interlock (OPI) would also be revised. The associated TS Bases would be updated to reflect the proposed changes.

Removal of the ACI is expected to reduce the incidence of events involving loss of SDC System capability to cool during nonpower operations due to inadvertent closure of the suction valves. The proposed TS changes are in accordance with the recommendations of Generic Letter (GL) 88-17, "Loss of Decay Heat Removal." GL 88-17 recommended changes to existing TSs which restrict or limit overall safety benefit. Since the ACI is a contributor to the loss of SDC system events, its proposed removal is consistent with the recommendations in GL 88-17.

The OPI setpoint was reviewed by BG&E as part of this TS change request. It was determined that the setpoint should be the actual reactor coolant system (RCS) pressure at the instrument tap location. The requested change for the OPI setpoint from 300 pounds per square inch absolute (psia) to 309 psia reflects the actual pressure at the instrument tap when the RCS pressure is 350 psia.

2.0 BACKGROUND

The SDC system is based on a design pressure of 350 psig while the RCS is designed for a maximum pressure of 2,500 psia. During normal operating conditions, a double barrier between the high pressure RCS and the low pressure SDC system is provided by two motor-operated suction valves. These

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valves are closed when the RCS is hot and pressurized (normal operating conditions) and open when the SDC system is in operation (cooldown or refueling). Each of these suction valves is provided with manual controls on the main control board.

There are two automatic interlocks associated with the control circuitry; the ACI and the OPI.

The OPI prevents the suction valves from being opened when RCS design suction pressure equals or exceeds 350 psia. The OPI function and the associated TSs are not affected by the proposed amendment except for a revision of the OPI setpoint at which the opening of the SDC suction isolation valves is allowed. This revision corrects the setpoint of the actual RCS pressure at the instrument tap elevation from 300 psia to 309 psia which is the RCS pressure at the tap elevation when the SDC system suction pressure is 350 psia.

The purpose of the SDC system ACI is to ensure that the low pressure piping of the SDC system is properly isolated from the RCS pressure during startup operations. When the valves are in the open position, the ACI causes the valves to close automatically when RCS pressure increases to a value above the predetermined autoclosure setpoint. Although the ACI protects the low pressure piping of the SDC system, spurious actuation could terminate decay heat removal during shutdown cooling operations.

The Commission and industry have previously recognized the safety benefits from removing the ACI circuitry from the SDC system. A disadvantage of the autoclosure feature is the possibility of an inadvertent valve closure during SDC system operation resulting in the loss of decay heat removal capability. The safety benefits of removing the ACI circuitry were stated in the Commission's case study on long term decay heat removal, Case Study Report AEOD/C503, "Decay Heat Removal Problems at U.S. Pressurized Water Reactors," December 1985 and also in a study performed for the Commission by Brookhaven National Laboratory, NUREG/CR-5015, "Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99," May 1988. In GL 88-17, "Loss of Decay Heat Removal," the Commission requested that TSs which restrict or limit the safety benefit of actions discussed in GL 88-17 should be identified and that appropriate changes should be submitted. One of the items listed by GL 88-17 that could limit such safety benefits was the ACI.

In parallel with the Commission's activities, Combustion Engineering (CE) completed a report, CE NSPD-550, "Risk Evaluation of Removal of Shutdown Cooling System Auto-Closure Interlock," September 1989, that documented the results of a generic analysis of the impact of removing the ACI from the SDC system. This report is generally applicable to Calvert Cliffs, Units 1 and 2, and was supplemented by a plant specific evaluation CE NSPD-548,"Requirements for the Removal of the Shutdown Cooling Suction Valve Auto-Closure Interlock," September 1989. The evaluation was performed to determine the change in an interfacing system loss-of-coolant accident (ISLOCA) frequency, the change in

SDC system unavailability, and the impact on mitigating low-temperature overpressure events due to the removal of ACI.

3.0 EVALUATION

In support of its requested TS changes, BG&E referenced the CE reports discussed in Section 2.0 above. The CE reports included a probabilistic risk analysis (PRA) regarding the removal of the SDC system ACI and a plant specific evaluation for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2. BG&E described how improvements identified by the above reports will be implemented at the Calvert Cliffs units. These results take into account the impact of the removal of the ACI feature on the SDC system inlet isolation valves. BG&E concluded that the implementation of their proposed design, TSs, administrative control, and procedure changes will reduce the frequency of a SDC system overpressurization event and increase the SDC system availability at the Calvert Cliffs units.

The staff reviewed BG&E's proposal against the recommendations and guidance in the CE NSPD-550 report and the plant specific evaluation in the CE NSPD-548 report. The hardware change proposed for the Calvert Cliffs units is the removal of the ACI function from the SDC system suction valves. The OPI will remain intact with a proposed setpoint change.

For Calvert Cliff, Units 1 and 2, BG&E evaluated the following items to support the removal of the ACI.

1. The means available to prevent Event V concerns:

   Event V is a LOCA event outside of containment. The Calvert Cliffs design provides for a double barrier between the RCS and the SDC system. The design provides a very high confidence that at least one barrier can be established and maintained under postulated conditions. This is accomplished through the use of separate power supplies, independent valve position indication, and the separation of control and indication power sources. Procedural controls, personnel training, automatic audible and visual alarms, and the OPI function minimize the potential for operator error when establishing double isolation or attempting to defeat it once it is established. The OPI will prevent opening of the SDC system suction isolation valves when the RCS pressure exceeds the interlock setpoint.

2. The alarms to alert the operator of an improperly positioned SDC system isolation valve:

   Prior to the removal of ACI, automatic visual and audible alarms will be provided in the main control room to inform the operator if any one of the SDC system suction isolation valves is not fully closed when RCS pressure is above the alarm setpoint. The alarms will be tested at each refueling outage.

3. Verification of the adequacy of relief valve capacity:

   A review by the licensee of the original design basis of the SDC system relief valve indicated that the ACI has not been credited in the selection of the limiting events or mitigation of the resulting transients. The use of the SDC system relief valve along with administrative controls provide overpressure protection on the SDC system. As a result, the removal of the ACI will have no adverse impact on SDC system overpressure protection provisions.

4. Means other than ACI to ensure that both isolation valves are closed:

   Closure of both isolation valves prior to pressurizing the RCS remains primarily an operator function. The alarms provide backup if the operator fails to close both valves. In addition to the alarms described in Item 2 above, the proposed modification will use position indication, operating procedures, and personnel training to ensure that the double barrier is established when needed.

5. Assurance that the OPI is not affected by ACI removal:

   The OPI function will be maintained in its present form. Assurance that the OPI function is not affected will be confirmed by testing the operability of the OPI function after the ACI is removed.

6. Assurance that valve position indication will remain available in the control room after ACI removal:

   There is continuous valve position indication on the main control board. The indication for the valve position utilizes DC control power. Power for valve position indication has been separated from control power. Circuit operability will be verified after the ACI is removed.

7. Assessment of the effect of ACI removal on SDC system availability and low-temperature overpressure (LTOP) event:

   The effect of ACI removal on LTOP protection for the reactor vessels was not assessed since no portion of the SDC system at Calvert Cliffs is used for this purpose. The LTOP system at Calvert Cliffs makes use of a lowered setpoint on the Power-Operated Relief Valves (PORVs) located on the pressurizer, along with administrative controls, to provide protection against brittle fracture of the RCS pressure boundary. As a result, ACI removal has no effect on LTOP. The use of the SDC system relief valve along with administrative controls provide protection against overpressurizing the SDC system.

   A plant specific PRA evaluation was performed, as previously noted, to evaluate the affect of the proposed change on the probability of ISLOCA, SDC system availability, and potential mitigation of slow acting pressure transients. The use of ACI for isolation of the SDC system from the RCS during slow acting pressure transients was not part of the original plant design basis. However, considering that the potential for using the ACI for this purpose exist, BG&E evaluated the impact of ACI removal on this type of overpressure protection.

The results of the risk evaluation show that: (1) the frequency of an interfacing system LOCA decreases with the removal of the ACI circuitry from the SDC system when accompanied by the addition of a control room alarm and procedural enhancements, (2) removal of the ACI increases SDC system availability, and (3) the net effect of ACI deletion from the SDC system is an overall improvement in safety.

The setpoint for the OPI function was reviewed as part of this proposed change by BG&E. The surveillance requirement for the SDC system OPI provides assurance that the SDC system suction isolation valves are prevented from being remotely opened when the RCS pressure is at or above the SDC system design suction pressure of 350 psia. It was determined that the value in the TSs should be the actual RCS pressure at the instrument pressure tap location which is 309 psia when the SDC system suction pressure is 350 psia. Revising the OPI actuation from 300 psia to 309 psia is a result of establishing a clear basis for this value and takes into effect the correct elevation at the point of the measurement for measuring the RCS pressure. The TS surveillance test procedure will contain the necessary compensation to be applied to this value to account for instrument uncertainties.

4.0 Summary

The removal of the ACI from the SDC system and the other actions, as detailed above, is consistent with the recommendations of GL 88-17 and has an overall positive impact on safety. Therefore, BG&E's proposal to remove the ACI capability and to modify the TS surveillance associated with TS 4.5.2.e.1 by removing the requirement to verify that the ACI isolates the SDC system is acceptable.

Inclusion of the requirement in TS 4.5.2.e.1 to verify that the SDC system OPI prevents the SDC system suction isolation valves from being opened with a simulated or actual RCS pressure equal to or greater than 309 psia is acceptable in that it provides reasonable assurance that the SDC system will not be over pressurized. Revising the OPI setpoint to 309 psia is acceptable sense it reflects the measured pressure at the instrument tap location when the SDC system pressure is 350 psia and the surveillance testing will take into account instrument uncertainties.

The revisions to TS Bases Section B 3/4.5 are acceptable in that they reflect the changes to the TSs detailed above.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 64600). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 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 the amendments.

7.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
H. Balukjian

Date: August 24, 1994

Docket Nos. 50-317
and 50-318

August 24, 1994

Mr. Robert E. Denton
Vice President -- Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

Dear Mr. Denton:

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M88164) AND UNIT NO. 2 (TAC NO. M88165)

The Commission has issued the enclosed Amendment No. 192 to Facility Operating License No. DPR-53 and Amendment No. 169 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 November 3, 1993.

The amendments modify the surveillance requirements to reflect the removal of the auto-closure interlock from the shutdown cooling system and revises the setpoint for the open permissive interlock.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly <u>Federal</u> <u>Register</u> 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. 192 to DPR-53
2. Amendment No. 169 to DPR-69
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1:PM	OGC	PDI-1:D	
NAME	CVogan CV	DMcDonald:avl	EHoller	PTKuo	
DATE	8/4/94	08/04/94	8/19/94	8/24/94	/ /

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