December 4, 1990

Docket Nos. 50-317
and 50-318

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

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Dear Mr. Creel:

SUBJECT: ISSUANCE OF AMENDMENTS FOR CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1 (TAC NO. 72075) AND UNIT 2 (TAC NO. 72076)

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-53 and Amendment No. 130 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 January 20, 1989, as supplemented on June 30, 1989, and October 4, 1990.

These amendments revise the auxiliary feedwater (AFW) actuation delay time in Technical Specification (TS) Table 3.3.-5, Item 10, of the Engineered Safety Features Response Times. The amendments change the response (delay) time from less than or equal to 54.5 seconds to less than or equal to 180 seconds. The new response time is applicable to the steam-driven and motor-driven AFW pumps.

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. 149 to DPR-53
2. Amendment No. 130 to DPR-69
3. Safety Evaluation

cc w/enclosures:
See next page

| PDI-1 | PDI-1 | OGC | PDI-1 |
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| CVogan | DMcDonald:avl | | RACapra |
| 10-17-90 | 11/09/90 | 11/19/90 | 12/04/90 |

DOCUMENT NAME: CC AMEND AFW TAC NO. 72075/076

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

December 4, 1990

Docket Nos. 50-317
and 50-318

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. 72075) AND UNIT 2 (TAC NO. 72076)

The Commission has issued the enclosed Amendment No.149 to Facility Operating License No. DPR-53 and Amendment No.130 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 January 20, 1989, as supplemented on June 30, 1989, and October 4, 1990.

These amendments revise the auxiliary feedwater (AFW) actuation delay time in Technical Specification (TS) Table 3.3.-5, Item 10, of the Engineered Safety Features Response Times. The amendments change the response (delay) time from less than or equal to 54.5 seconds to less than or equal to 180 seconds. The new response time is applicable to the steam-driven and motor-driven AFW pumps.

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,

Daniel G. McDonald

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.149 to DPR-53
2. Amendment No.130 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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P. O. Box 902
Columbia, Maryland 21044

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. 149
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 January 20, 1989, as supplemented on June 30, 1989 and October 4, 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. 149, 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

Robert A. Capra

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: December 4, 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. 130
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 January 20, 1989, as supplemented on June 30, 1989, and October 4, 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. 130, 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

Robert A. Capra

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: December 4, 1990

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 149 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 130 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 3/4 3-21 | 3/4 3-21 |
| 3/4 3-22* | 3/4 3-22* |

*Pages that did not change, but are overlief

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 6. Steam Generator Pressure-Low | |
| a. Main Steam Isolation | $\leq$ 6.9 |
| b. Feedwater Isolation | $\leq$ 80 |
| 7. Refueling Water Tank-Low | |
| a. Containment Sump Recirculation | $\leq$ 80 |
| 8. Reactor Trip | |
| a. Feedwater Flow Reduction to 5% | $\leq$ 20 |
| 9. Loss of Power | |
| a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage) | $\leq$ 2.2* |
| b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage) | $\leq$ 8.4* |
| 10. Steam Generator Level-Low | |
| a. Steam Driven AFW Pump | $\leq$ 180 |
| b. Motor Driven AFW Pump | $\leq$ 180 |
| 11. Steam Generator $\Delta$P-High | |
| a. Auxiliary Feedwater Isolation | $\leq$ 20.0 |

TABLE NOTATION

* Response time measured from the incidence of the undervoltage condition to the diesel generator start signal.

(1) Header fill time not included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. SAFETY INJECTION (SIAS) | | | | |
| a. Manual (Trip buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure - High | S | R | M | 1, 2, 3 |
| c. Pressurizer Pressure - Low | S | R | M | 1, 2, 3 |
| d. Automatic Actuation Logic | N.A. | N.A. | M(1)(3) | 1, 2, 3 |
| 2. CONTAINMENT SPRAY (CSAS) | | | | |
| a. Manual (Trip buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure-High | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(6) | 1, 2, 3 |
| 3. CONTAINMENT ISOLATION (CIS)# | | | | |
| a. Manual CIS (Trip buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure-High | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(4) | 1, 2, 3 |
| 4. MAIN STEAM LINE ISOLATION (SGIS) | | | | |
| a. Manual SGIS (MSIV Hand Switches and Feed Head Isolation Hand Switches) | N.A. | N.A. | R | N.A. |
| b. Steam Generator Pressure-Low | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(5) | 1, 2, 3 |

#Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 6. Steam Generator Pressure-Low | |
| a. Main Steam Isolation | ≤ 6.9 |
| b. Feedwater Isolation | ≤ 80 |
| 7. Refueling Water Tank-Low | |
| a. Containment Sump Recirculation | ≤ 80 |
| 8. Reactor Trip | |
| a. Feedwater Flow Reduction to 5% | ≤ 20 |
| 9. Loss of Power | |
| a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage) | ≤ 2.2* |
| b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage) | ≤ 8.4* |
| 10. Steam Generator Level - Low | |
| a. Motor Driven AFW Pump | ≤ 180 |
| b. Steam Driven AFW Pump | ≤ 180 |
| 11. Steam Generator ΔP-High | |
| a. Auxiliary Feedwater Isolation | ≤ 20.0 |

TABLE NOTATION

* Response time measured from the incidence of the undervoltage condition to the diesel generator start signal.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. SAFETY INJECTION (SIAS) | | | | |
| a. Manual (Trip Buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure - High | S | R | M | 1, 2, 3 |
| c. Pressurizer Pressure - Low | S | R | M | 1, 2, 3 |
| d. Automatic Actuation Logic | N.A. | N.A. | M(1)(3) | 1, 2, 3 |
| 2. CONTAINMENT SPRAY (CSAS) | | | | |
| a. Manual (Trip Buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure -- High | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(6) | 1, 2, 3 |
| 3. CONTAINMENT ISOLATION (CIS) # | | | | |
| a. Manual CIS (Trip Buttons) | N.A. | N.A. | R | N.A. |
| b. Containment Pressure - High | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(4) | 1, 2, 3 |
| 4. MAIN STEAM LINE ISOLATION (SGIS) | | | | |
| a. Manual SGIS (MSIV Hand Switches and Feed Head Isolation Hand Switches) | N.A. | N.A. | R | N.A. |
| b. Steam Generator Pressure - Low | S | R | M | 1, 2, 3 |
| c. Automatic Actuation Logic | N.A. | N.A. | M(1)(5) | 1, 2, 3 |

# Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-53

# AND AMENDMENT NO. 130 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 January 20, 1989, as supplemented on June 30, 1989, and October 4, 1990, the Baltimore Gas and Electric Company (the licensee) proposed to amend the Technical Specifications (TS) of the Calvert Cliffs Nuclear Power Plant, Units 1 and 2. The proposed change would increase the response time, upon an initiating signal, of the steam and motor-driven auxiliary feedwater (AFW) pumps. The licensee provided a responses to our requests for additional information by letters dated June 30, 1989, and October 4, 1990.

## 2.0 EVALUATION

The current TS, items 10a and b in TS Table 3.3-5, have a response time of 54.5 seconds for the steam-driven and motor-driven AFW pumps. The TS value is based upon the response time of the steam-driven pumps to an initiation signal as detailed in the Updated Final Safety Analysis Report (UFSAR), Chapter 14, which includes: 50 seconds to open the steam admission valves and 4.5 seconds for the pumps to accelerate to full speed. The travel time, 3.5 seconds, required for the water to travel through the piping to the steam generators is not included in the TS.

The licensee has stated that an increase in the response time for the steam-driven AFW pumps would allow for modifications necessary to prevent or minimize dynamic damage to the governor linkages. Also, the present emergency diesel generators (EDGs) loading is approaching the machine's capacity limits and an increase in the response time for the motor-driven AFW pumps would provide greater flexibility with regard to the loading of the EDGs. The modifications to the AFW systems and changes in the load sequences for the EDGs will provide an overall enhancement to the reliability of the AFW systems. The licensee indicated in its October 4, 1990, response that post modification testing will be performed to determine the actual AFW systems response times. Future surveillance testing will include trending of the AFW systems response times and an evaluation of any adverse trending will be performed so that appropriate corrective actions can be taken.

The major concern associated with the proposed TS change is that the steam generators could go dry, thereby causing their loss as a heat sink. This could occur during a loss of feedwater event. Combustion Engineering (CE), the Nuclear Steam Supply System vendor, reanalyzed the event for the licensee using the NRC-approved CESEC computer code. Major assumptions were introduced such as new low steam generator level trip setpoints and an increased delay time (218.5 seconds) for the delivery of AFW flow. The results demonstrated that the steam generator inventories were maintained without loss of the steam generators as a heat sink.

The licensee proposed to change the TS AFW response time to 180 seconds, which is much lower than the delay time used in the CE analysis. As no change in the level setpoints have been requested, the licensee's proposal is more conservative than the CE analysis. The proposed TS response time, however, is based on Table 2 in the January 20, 1989, submittal and includes the 3.5 second water travel time. Thus, the proposed TS change and Table 2 are inconsistent. However, due to the large margin demonstrated by the CE analysis, the staff finds the proposed TS value acceptable. The licensee should revise the UFSAR, Chapter 14, to reflect consistency in the application of the water travel time.

## 3.0 SUMMARY

We have reviewed the results of the supporting analyses for the proposed TS changes and have concluded that the changes are acceptable. However, as noted, the UFSAR should be updated to reflect consistency with TS Table 3.3-5 in the application of water travel time.

## 4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of the facilities' components located within the restricted areas as defined in 10 CFR Part 20. 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 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.

## 5.0 CONCLUSION

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

PRINCIPAL CONTRIBUTORS:

D. Katze
D. McDonald

Dated: December 4, 1990

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

November 15, 1990

MEMORANDUM FOR: Sholly Coordinator

FROM: Daniel G. McDonald, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II

SUBJECT: REQUEST FOR PUBLICATION IN BI-WEEKLY FR NOTICE - NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR A HEARING (TAC NOS. 79005 AND 79006)

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendment request: November 5, 1990

Description of amendment request: The proposed Technical Specifications (TS) changes are requested in response to the Nuclear Regulatory Commission's Generic Letter (GL) 88-17, "Loss of Decay Heat Removal," dated October 17, 1988. One of the actions requested in the GL was for licensees to identify any TS for their facility that would restrict or limit the safety benefits of the actions identified in the GL and to request appropriate TS changes. High flow in the Shutdown Cooling (SDC) system during reduced Reactor Coolant System (RCS) inventory could result in air ingestion into the RCS. The air in the RCS could lead to vortexing in the SDC system pumps; thus, resulting in damage or failure of the pumps and subsequent loss of decay heat removal capability.

The proposed changes to TS 3.9.8.1 for Units 1 and 2 will delete the flow rates currently specified for Mode 6 (Refueling) operation. The current



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flow requirement for Mode 6 operation is equal to or less than 3000 gpm, or equal to or less than 1500 gpm when the RCS is drained to a level below mid-plane of the hot leg. The current flow requirements are applicable for the above conditions regardless of the decay heat level of the core. The proposed deletion of the specified flow rates will allow operation at lower flow rates when the RCS is at intermediate inventory levels. The reduced flow rates will decrease the likelihood of air ingestion into the RCS resulting in SDC pump vortexing which could lead to pump failure and subsequent loss of the decay heat removal capability.

The licensee also proposes changes to the TS Bases 3/4.9.8 to support the deletion of the specified flow rates for SDC during Mode 6 operation. The Bases also indicate that shutdown cooling flow must provide sufficient heat removal to match core decay heat generation and maintain the core exit temperature within the Mode 6 limit.

<u>Basis for proposed no significant hazards consideration determination</u>: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92.

The licensee has evaluated the proposed amendment against the standards provided above and has supplied the following information:

[1] involve a significant increase in the probability or consequences of an accident previously evaluated;

Previously evaluated accidents which could be impacted by SDC flow changes include a (1) boron dilution event, and a (2) loss of coolant flow. Sufficient flow for mixing will continue to be provided and the assumptions and conclusions of the Boron Dilution Event analysis presented in the FSAR [Final Safety Analysis Report] (Section 14.3) will be preserved. Also, two SDC loops will continue to provide the level of protection previously established by the safety analysis when there is less than 23 feet of water above the core, thereby ensuring that a single failure of the operating SDC loop will not result in a complete loss of decay heat removal capability.

This change will allow a variable SDC flow to be established when the RCS is partially drained to prevent vortexing. The established flow will provide SDC System performance commensurate with its design functions of removing decay heat and maintaining RCS temperature [less than or equal to] 140°F in MODE 6. Further, this change will provide a net improvement in SDC System reliability by reducing the probability of common mode failure due to vortexing during partially drained RCS conditions. Therefore, this change will not increase the probability or the consequences of an accident previously evaluated.

[2] create the possibility of a new or different type of accident from any accident previously evaluated;

This change does not represent a significant change in the configuration or operation of the plant. Specifically, no new hardware is being added to the plant as part of the proposed change, no existing equipment is being modified, nor are any significantly different types of operations being introduced. Variable flow of SDC is currently allowed in MODE 5 and will be similarly controlled in MODE 6.

The SDC System will still be operated in the same manner as before with the exception that the LPSI [Low Pressure Safety Inspection] pump flow will be throttled to match decay heat removal requirements. The system will maintain the same capacity for decay heat removal as before. No new or different kinds of accidents than any previously evaluated are being created. This change will actually help prevent a possible common mode failure of both LPSI pumps caused by vortexing and air entrainment while at partially drained conditions.

[3] involve a significant reduction in a margin of safety.

This change will ensure that the margin of safety is maintained. The system configuration will remain the same, and it will be operated in a manner less likely to cause vortexing. The system's capability for decay heat removal and mixing will be maintained, as will system redundancy. Administrative control minimum flow in MODE 6 is consistent with the philosophy of control currently applied in MODE 5 and promoted in Generic Letter 88-17, Enclosure 2, Section 3.5.2. Therefore, the proposed change will not reduce the margin of safety associated with this system.

The staff has reviewed and agrees with the licensee's analysis of the significant hazards consideration determination. Based on the review and the above discussion, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick Maryland.

Attorney for licensee: Jay E. Silbert, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Project Director: Robert A. Capra

Daniel G. McDonald Jr

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

The staff has reviewed and agrees with the licensee's analysis of the significant hazards consideration determination. Based on the review and the above discussion, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick Maryland.

Attorney for licensee: Jay E. Silbert, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Project Director: Robert A. Capra

ORIGINAL SIGNED BY:

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

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