July 6, 1990

Docket Nos. 50-317
and 50-318

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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. 76704) AND UNIT 2 (TAC NO. 76705)

The Commission has issued the enclosed Amendment No.144 to Facility Operating License No. DPR-53 and Amendment No.127 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 April 30, 1990.

These amendments include a footnote to Technical Specification 3/4.9.13, Spent Fuel Cask Handling Crane, which allows one-time movement of two separate special spent fuel shipping casks to and from the cask pit passing over spent fuel assemblies stored in the spent fuel pool. The footnote identifies the specific criteria to be met prior to movement of the casks over the fuel stored in the spent fuel pool.

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.144 to DPR-53
2. Amendment No.127 to DPR-69
3. Safety Evaluation

cc: w/enclosures
See next page

PDI-1	PDI-1	SRXB	OGC	PDI-1
CVogan	DMcDonald:rsc	RJones		RACapra
6/28/90	6/28/90	6/26/90	6-29-90	7/6/90

DOCUMENT NAME: ISSUANCE OF AMEND 76704/76705

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

Calvert Cliffs Nuclear Power Plant

cc:

Mr. William T. Bowen, President
Calvert County Board of
Commissioners
Prince Frederick, Maryland 20678

D. A. Brune, Esq.
General Counsel
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Mr. Jay E. Silberg, Esq.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW
Washington, DC 20037

Ms. G. L. Adams, Licensing
Calvert Cliffs Nuclear Power Plant
MD Rts 2 & 4, P. O. Box 1535
Lusby, Maryland 20657

Resident Inspector
c/o U.S.Nuclear Regulatory Commission
P. O. Box 437
Lusby, Maryland 20657

Mr. Richard McLean
Administrator - Radioecology
Department of Natural Resources
580 Taylor Avenue
Tawes State Office Building
PPER B3
Annapolis, Maryland 21401

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Joseph H. Walter
Engineering Division
Public Service Commission of Maryland
American Building
231 E. Baltimore Street
Baltimore, Maryland 21202-3486

Ms. Kirsten A. Burger, Esq.
Maryland People's Counsel
American Building, 9th Floor
231 E. Baltimore Street
Baltimore, Maryland 21202

Ms. Patricia Birnie
Co-Director
Maryland Safe Energy Coalition
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. 144
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 April 30, 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. 144, 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

Donald S. Brinkman

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: July 6, 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. 127
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 April 30, 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. 127 , 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

Donald S. Brinkman

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: July 6, 1990

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 144 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 127 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows:

Remove Page	Insert Page
3/4 9-16	3/4 9-16

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly. *

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. *

---

* These conditions are modified to permit shipping cask travel to and from the cask pit in the presence of fuel within one cask length radius of the pathway provided the boric acid concentration in the spent fuel pool is greater than or equal to 1000 ppm AND the following criteria are met by all assemblies within one cask length radius of the pathway: 1) Initial enrichment less than or equal to 4.1 w/o U-235, 2) Burnup greater than or equal to 28,000 MWD/MTU, and 3) Greater than 440 days elapsed from the shutdown of the last operating cycle in which the assembly was present in the core. Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly not satisfying the above criteria shall be demonstrated OPERABLE within 24 hours prior to using the crane for moving a cask within one length of fuel assemblies meeting the above criteria. These modifications are applicable only to the shipment of fuel rods supporting the EPRI sponsored hot-cell work for the shipment of a reactor vessel weld material surveillance capsule.

REFUELING OPERATIONS

SPENT FUEL CASK HANDLING CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 Crane travel of the spent fuel shipping cask crane shall be restricted to prohibit a spent fuel shipping cask from travel over any area within one shipping cask length of any fuel assembly. *

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly shall be demonstrated **OPERABLE** within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. *

---

* These conditions are modified to permit shipping cask travel to and from the cask pit in the presence of fuel within one cask length radius of the pathway provided the boric acid concentration in the spent fuel pool is greater than or equal to 1000 ppm AND the following criteria are met by all assemblies within one cask length radius of the pathway: 1) Initial enrichment less than or equal to 4.1 w/o U-235, 2) Burnup greater than or equal to 28,000 MWD/MTU, and 3) Greater than 440 days elapsed from the shutdown of the last operating cycle in which the assembly was present in the core. Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly not satisfying the above criteria shall be demonstrated OPERABLE within 24 hours prior to using the crane for moving a cask within one length of fuel assemblies meeting the above criteria. These modifications are applicable only to the shipment of fuel rods supporting the EPRI sponsored hot-cell work for the shipment of a reactor vessel weld material surveillance capsule.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 127 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 April 30, 1990, Baltimore Gas and Electric Company, the licensee for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, proposed Technical Specification (TS) changes to Appendix A of Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Units 1 and 2, respectively. The proposed TS changes were requested to allow one-time movements of two separate special spent fuel shipping casks to and from the cask pit passing over spent fuel assemblies stored in the spent fuel pool. The first cask (33 inches in diameter, 16 feet long, and approximately 18 tons in weight) will be used to ship 13 selected spent fuel rods to Chalk River, Canada. The second smaller cask will be used to ship a reactor vessel weld material surveillance capsule to Combustion Engineering, both in support of hot-cell work sponsored by the Electric Power Research Institute (EPRI). The staff's review and evaluation addresses the offsite radiological consequences and criticality concerns resulting from a postulated spent fuel shipping cask drop accident (the larger cask) over the spent fuel assemblies stored in the spent fuel pool.

2.0 BACKGROUND

The current Section 3.9.13 of the Calvert Cliffs TS, "Spent Fuel Cask Handling Crane," prohibits a spent fuel shipping cask from passing over any area within one shipping cask length of any stored fuel assembly. This TS requirement was the result of the licensee's response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The proposed TS changes would allow the present fuel, stored as the result of the extended Unit 2 outage, to remain in an area which currently prohibits storage when a fuel shipping cask is being moved into and away from the cask pit passing over that area. The weight of the spent fuel shipping casks described above are bounded by the weight of the heavy loads (cask drop accident) considered during the NUREG-0612 review.

The specific change requested will add the following footnotes to the Calvert Cliffs Units 1 and 2, TS Sections 3/4.9.13:

> "These conditions are modified to permit shipping cask travel to and from the cask pit in the presence of fuel within one cask length radius of the pathway provided the boric acid concentration in the spent fuel pool is greater than or equal to 1000 ppm AND the following criteria are met by all assemblies within one cask length radius of the pathway: 1) initial enrichment less than or equal to 4.1 w/o U-235, 2) Burnup greater than or equal to 28,000 MWD/MTU, and 3) greater than 440 days elapsed from the shutdown of the last operating cycle in which the assembly was present in the core. Crane interlocks and physical stops which restrict a spent fuel shipping cask from passing over any area within one shipping cask length of any fuel assembly not satisfying the above criteria shall be demonstrated OPERABLE within 24 hours prior to using the crane for moving a cask within one cask length of fuel assemblies meeting the above criteria. These modifications are applicable only for shipment of fuel rods supporting the EPRI sponsored hot-cell work and for the shipment of a reactor vessel weld material surveillance capsule."

Calvert Cliffs Units 1 and 2 share a common spent fuel pool that is located outside the containment in the auxiliary building and is divided by a dam into two identical halves, the north and south pools. The spent fuel shipping cask pit is located on the Unit 1 side, north of the dividing dam in the pool. The licensee estimated that a maximum of approximately 500 spent fuel assemblies could be located in the pool within the radius of one cask length from the cask pathway at any time during the cask movement.

3.0 EVALUATION

The staff evaluated postulated fuel handling cask drop accident using assumptions contained in Positions C.1.a through C.1.k of Regulatory Guide (RG) 1.25 and the procedures specified in Standard Review Plan (SRP) Section 15.7.5 (NUREG-0800). In addition, the specified assumptions postulate that the dropped fuel cask would damage all 500 spent fuel assemblies by falling onto its side and rolling in a direction in which it could break open all fuel rods in the 500 spent fuel assemblies.

Instantaneous puff release of noble gasses and radioiodine from the gaps of the broken rods occur as gas bubbles pass up through the water covering the fuel. All radioactivity reaching the auxiliary building atmosphere is exhausted within 2 hours through engineered safety feature filtered exhaust systems to the environment. As stipulated in the proposed TS change request, all radioactive material in the 500 spent fuel assemblies that were damaged by the dropped fuel cask is assumed to have decayed for a period of greater than 440 days.

The staff computed the offsite doses for the Calvert Cliffs Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary using the assumptions described above, the assumptions contained in Regulatory Guide 1.25, and the procedures specified in SRP Section 15.7.5. These computed offsite doses are well within the acceptance criteria given in Section 15.7.5 of the SRP and the exposure guidelines of 10 CFR Part 100. The attached tables provide the results of the staff's calculations of the offsite doses (Table 1) and the assumptions used in calculating offsite doses (Table 2).

Criticality concerns were also considered because a cask drop would cause a geometrical distortion of the fuel/rack system. Because the distortion is difficult to predict, assumptions were made to bound the most reactive configuration. For the calculations, the licensee assumed that the geometry of individual fuel assemblies was not deformed (this assumption maximized reactivity) and the storage racks were deformed to remove the inter-storage cell gap (neutron flux trap). Additionally, the poison material contained within the racks was ignored and replaced by pool water. Although NUREG-0612 states that criticality analysis for a dropped load may assume that the poison material integral to the rack remains in place, the licensee conservatively chose to ignore this benefit. The licensee also assumed an initial fuel assembly enrichment limit of 4.1 w/o U-235, a minimum assembly burnup of 28,000 MWD/MTU, and a minimum boron concentration in the spent fuel pool of 1000 ppm.

The staff finds the assumptions related to enrichment, burnup, and boron concentration, which are incorporated directly into the proposed TS change are consistent with NUREG-0612 and are, therefore, acceptable.

The restrictions imposed by the footnote will be confirmed by procedures which require that the serial numbers of each fuel assembly located within one cask length radius of the pathway be checked to assure that all the stored fuel within the arc conforms to the restrictions. The procedures also require verification that the boron concentration is greater than or equal to 1000 ppm prior to moving the cask into or out of the spent fuel pool.

A two-dimensional analysis using the previously approved DOT-IV computer code was performed for an infinite array of fuel assembly storage modules distorted as described above. The results yielded a Keff of 0.898. Because uncertainties are less than 0.03, the Keff value with uncertainties will be no greater than 0.928. Consequently, the results are well within the criticality limit of a Keff equal to or less than 0.95 specified in Criterion II of NUREG-0612, Section 5.1. The Keff of .095 is the currently approved limit in the TS for refueling operations as identified in the TS Bases 3/4.0, "Refueling Operations."

The staff has determined that the proposed TS changes are acceptable based on its evaluation discussed above. The offsite radiological consequences are well within the acceptance criteria based on the staff's independent analysis using the assumptions in RG 1.25 and SRP Section 15.7.5. The results of the licensee's criticality analysis, using previously approved methodologies, are well within the criticality limits of a Keff equal to or less than .095 specified in NUREG-0612 and SRP Section 9.1.2.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in 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 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.

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.

Dated: July 6, 1990

PRINCIPAL CONTRIBUTORS:

J. Lee
D. McDonald

Table 1

Calculated Radiological Consequences following fuel cask drop accident (rem)

	EAB	LPZ	SRP 15.7.5 Limits
Thyroid Dose	0.01	0.01	75
Whole Body Dose	0.05	0.01	6

Table 2

Assumptions used for estimating the radiological consequences following a postulated fuel cask drop accident in fuel pool

Parameter and Unit of Measure	Quantity
Power level, MWt	2700
Number of fuel assemblies damaged	500
Shutdown time, days	440
Inventory released from damaged rods, %	
Iodine	10
Noble Gases	30
Pool decontamination factors	
Iodine	100
Noble gases	1
Iodine forms in atmosphere above pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies for auxiliary building gas treatment system (spent fuel pool area), %	
Elemental	No filters assumed
Organic	No filters assumed
Atmospheric dispersion factor, sec/m³	1.3 E-4
Auxiliary building mixing efficiency, %	0