Docket file

REGULATORY DOCKET FILE COPY

UNITED STATES NUCLEAR REGULATORY COMMISSION

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
September 19, 1980

Docket Nos. 50-317
and 50-318

Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

The Commission has issued the enclosed Amendments Nos. 47 and 30 to Facility Operating Licenses Nos. DPR-53 and DPR-69 for the Calvert Cliffs Nuclear Power Plant Units Nos. 1 and 2, respectively. The amendments are in accordance with your applications dated July 3, 1979, August 31, 1979, and January 15, 1980, and supplements thereto dated April 14 and 18, May 20 and 30, July 7, and September 12, 1980.

These amendments will allow an increase in the spent fuel storage capability up to a maximum of 1760 fuel assemblies in the spent fuel pool through the use of high density borated spent fuel racks. Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

Your letter of April 14, 1980 requested an increase in the spent fuel pool storage capacity from the previous application of 1760 to 1830 fuel assemblies. Subsequently, your letter of May 20, 1980, withdrew this request. However, your staff has indicated that subsequent application may be submitted to request an ultimate storage capacity of 1830 fuel assemblies. Our Safety Evaluation and Environmental Impact Appraisal were prepared considering this higher number of fuel assemblies except for the structure analysis review which was based on the 1760 fuel storage positions.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and Notice of Issuance and Negative Declaration are enclosed.

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures & cc:
See next page

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Enclosures:
1. Amendment No. 47 to License No. DPR-53
2. Amendment No. 30 to License No. DPR-69
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice and Negative Declaration

cc w/enclosures:
See next page

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cc w/4 cys enclosures and 1 cy of BG&E filings dtd.: 7/3/70; 8/31/79; 1/15/80 & supplements dtd 4/14&18/80; 5/20&30/80; 7/7/80; 9/12/80

Administrator, Power Plant Siting Program
Energy and Coastal Zone Administration
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

BALTIMORE GAS & ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-53

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

   A. The applications for amendment by Baltimore Gas & Electric Company (the licensee) dated July 3, 1979, August 31, 1979, and January 15, 1980, as supplemented by filings dated April 14 and 18, 1980, May 20 and 30, 1980, July 7, and September 12, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

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

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

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

   F. 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 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. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 19, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by Amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page 5-6 is also provided to maintain document completeness. No changes were made on 5-6.

Page

5-5

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,614 ± 460 cubic feet at a nominal T_{avg} of 532°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1 The spent fuel storage racks are designed and shall be maintained with a minimum 10 3/32" x 10 3/32" center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The k_{eff} of $\leq$ 0.95 includes the conservative allowances for uncertainties described in Section 9.7.2 of the FSAR. The maximum fuel enrichment to be stored in the fuel pool will be 4.1 weight percent.

CRITICALITY - NEW FUEL

5.6.2 The new fuel storage racks are designed and shall be maintained with a nominal 18 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having a maximum enrichment of 4.0 weight percent U-235 is in place and aqueous foam moderation is assumed. The k_{eff} of $\leq$ 0.98 includes the conservative allowance for uncertainties described in Section 9.7.2 of the FSAR.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 63 feet.

CAPACITY

5.6.4 The fuel storage pool is designed and shall be maintained with a combined storage capacity, for both Units 1 and 2, limited to no more than 1760 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 heatup and cooldown cycles	70°F to 532°F to 70°F
	400 reactor trip cycles	100% to 0% RATED THERMAL POWER
	10 Primary Hydrostatic Tests	3125 psia and 60°F > NDTT
	320 Primary Leak Tests	2500 psia and 60°F > NDTT
Steam Generator	10 Secondary Hydrostatic Tests	1250 psia Secondary Side and temperature ≥ 100°F
	320 Secondary Leak Tests	1000 psia Secondary Side With Primary - Secondary Δp of 820 psi and shell side temperature between 100°F and 200°F

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

BALTIMORE GAS & ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. DPR-69

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

   A. The applications for amendment by Baltimore Gas & Electric Company (the licensee) dated July 3, 1979, August 31, 1979, and January 15, 1980, as supplemented by filings dated April 14 and 18, 1980, May 20 and 30, 1980, July 7, and September 12, 1980, comply 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 1.

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

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

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

   F. 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 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. 30, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 19, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by Amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page 5-6 is also provided to maintain document completeness. No changes were made on 5-6.

Page

5-5

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,614 $\pm$ 460 cubic feet at a nominal T_{avg} of 532°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1 The spent fuel storage racks are designed and shall be maintained with a minimum 10 3/32" x 10 3/32" center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The k_{eff} of $\leq$ 0.95 includes the conservative allowances for uncertainties described in Section 9.7.2 of the FSAR. The maximum fuel enrichment to be stored in the fuel pool will be 4.1 weight percent.

CRITICALITY - NEW FUEL

5.6.2 The new fuel storage racks are designed and shall be maintained with a nominal 18 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having a maximum enrichment of 4.0 weight percent U-235 is in place and aqueous foam moderation is assumed. The k_{eff} of $\leq$ 0.98 includes the conservative allowance for uncertainties described in Section 9.7.2 of the FSAR.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 63 feet.

CAPACITY

5.6.4 The fuel storage pool is designed and shall be maintained with a combined storage capacity, for both Units 1 and 2, limited to no more than 1760 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component	Cyclic or Transient Limit	Design Cycle or Transient
Reactor Coolant System	500 heatup and cooldown cycles	70°F to 532°F to 70°F
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Steam Generator	10 Secondary Hydrostatic Tests	1250 psia Secondary Side and temperature $\geq$ 100°F
	320 Secondary Leak Tests	1000 psia Secondary Side With Primary - Secondary Δp of 820 psi and shell side temperature between 100°F and 200°F

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 47 AND 30 TO

FACILITY OPERATING LICENSE NOS. DPR-53 AND DPR-69

RELATING TO MODIFICATION OF THE SPENT FUEL POOL

BALTIMORE GAS & ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT

UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 Introduction

By letters dated July 3, 1979, August 31, 1979, and January 15, 1980, Baltimore Gas and Electric Company (BG&E) proposed to change the spent fuel pool (SFP) storage design for the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit Nos. 1 and 2. The presently approved design was reviewed and approved in Amendment Nos. 27 and 12 to Facility Operating License Nos. DPR-53 and DPR-69 issued January 4, 1978. The present installed storage capacity is 200 spent fuel assemblies in the Unit 2 (South) side of pool (unmodified), and 528 assemblies in the Unit 1 (North) side of pool (modified). The proposed modification will permit the storage of 830 fuel assemblies in the North half of the pool and 930 fuel assemblies in the South half of the pool. In response to our questions, BG&E submitted supplemental information by letters dated April 14 and 18, May 20 and 30, July 7, and September 12, 1980.

2.0 Background

The Calvert Cliffs Nuclear Power Plant (CCNPP) spent fuel pool (SFP) was originally designed with the storage capacity of 1-2/3 cores, (410 fuel assemblies) felt to be adequate for the storage of the discharge (72 assemblies per unit per year) from each reactor for one year prior to its shipment off-site for reprocessing, plus 217 storage locations for core unloading whenever it became necessary.

By our Amendment Nos. 27 and 12 dated January 4, 1978, we approved BG&E's request to expand their SFP capacity to 1056 fuel assemblies, 528 for each unit, through the use of high density spent fuel racks. The South pool was modified as planned. Before racks were designed for the North side of the pool, which has the installed capacity of 200 fuel assemblies, BG&E realized that a further increase in SFP capacity would likely be necessary before any reprocessing facility is ready. By letter dated July 3, 1979, BG&E amended their request to expand the North pool capacity to 840 assemblies with high capacity poison racks. In a subsequent letter dated January 15, 1980, BG&E requested that the South part of the pool also be included in our review. The proposed total capacity would be 1760 assemblies, 830 for the North pool and 930 for the South pool. Furthermore, BG&E again amended the application to increase the SFP capacity from 1760 to 1830 assemblies in their letter of April 14, 1980. They have, however, subsequently withdrawn this request in the letter of May 20, 1980 due to the need to proceed with the modification to the North side. Our reviews, except for the structure analysis, were completed before May 20, 1980 and were based on a capacity of 1830 assemblies. The review of the structure analysis was based on a capacity of 1760 assemblies.

BG&E states in their July 3, 1979 submittal that it is responsible for the modification to the spent fuel storage pool. Nuclear Energy Services is retained to design the spent fuel racks, contract for fabrication, perform analysis pertinent to the modification, and provide technical assistance during installation. Bechtel Power Corporation provided engineering assistance in reviewing the spent fuel pool structural considerations.

3.0 Discussion and Evaluation

In reviewing the SFP modification for CCNPP Unit Nos. 1 and 2, we considered: (1) criticality analysis, (2) spent fuel cooling, (3) installation of racks and fuel handling, (4) structure design, (5) fuel handling, (6) occupational radiation exposure, (7) radioactive waste treatment, and (8) Material acceptability.

3.1 Criticality Analysis

Two modification factors, fuel loading limit and high density racks, were considered in the evaluation of criticality analysis.

Fuel Loading Limit

The Nuclear Services Corporation (NSC) performed the criticality analyses for increasing the uranium-235 enrichment from 3.7 to 4.1 weight percent for fuel assemblies that are to be placed in the present racks. This corresponds to an increase in the fuel loading limit from 44.0 to 48.5 grams of uranium-235 per axial centimeter of fuel assembly. For these calculations NSC used the CHEETAH computer program to obtain four energy group cross sections for diffusion theory calculations with the CITATION program. The accuracy of this diffusion theory method was checked by comparison with several series of critical experiments.

Parametric calculations were made for the maximum possible reduction in storage lattice pitch, eccentric fuel assembly placement, and an increase in fuel pool water temperature to 212°F. A calculation was also made for the inadvertent placement of a fuel assembly adjacent to a filled rack. This resulted in a maximum neutron multiplication factor of 0.94 for fuel assemblies with 48.5 grams of uranium-235 per axial centimeter of assembly.

High Density Racks

The proposed new higher density racks are to be made up of individual double-walled containers which are about fourteen feet long. The inner wall of each of these containers will be made from a 0.060 inch thick sheet of 304 L stainless steel which will be formed into an indented, square cross section container with an inside dimension of 8.56 inches. The outer, or external, wall will also be a sheet of 0.060 inch thick stainless steel. Borated, neutron absorbing plates, which are 6.5 inches wide and 0.090 inches thick, will be placed in each of the four spaces between the two walls, which are formed by the indentations in the inner wall. Thus each of the four sides of every container will have a borated plate in it which, as BG&E states in its January 15, 1980 submittal, will initially contain at least 0.024 grams of boron-ten per square centimeter of plate. BG&E also states in this submittal that the average center-to-center spacing between all containers will be maintained at 10.09375 ± 0.03125 inches by the external sheets and by welded spacers. For an overall fuel region dimension of 8.13 inches, as shown in the July 3, 1979 submittal, this results in a fuel region volume fraction of 0.65.

Nuclear Energy Services, Incorporated (NES) performed the criticality analyses for BG&E for the proposed borated plate racks. For these calculations NES assumed a uniform distribution of unirradiated fuel with a maximum enrichment of 4.1 weight percent uranium-235 in the Unit 1 fuel assemblies, no burnable poisons, and pure, i.e., unborated, water in the pool.

NES made parametric calculations by using the HAMMER computer program to obtain four-group cross sections for EXTERMINATOR diffusion theory calculations. This calculational method was used to determine the nominal k_∞ and then the effects of design and fabrication tolerances, changes in temperature, and abnormal dislocations of fuel assemblies in the racks. NES also did verification calculations with the KENO Monte Carlo program. When using the 123 group NITAWL cross sections in a KENO-IV calculation of the nominal reference configuration, NES obtained a neutron multiplication factor of 0.92 ± 0.006. This included the effect of having discrete particles of boron in the plates rather than a uniform distribution of boron atoms. From its parametric calculations NES found that all of the possible manufacturing tolerances, such as those in cell pitch and in the thickness of the stainless steel walls, and all possible variations during the life of the racks, such as a reduction in the boron loading from 0.024 to 0.0194 grams of boron-ten per square centimeter of plate, could increase the neutron multiplication factor by 0.01 Δk. NES also found from its parametric calculations that eccentric positioning of fuel assemblies in the racks or increasing the pool temperature would not increase the neutron multiplication factor. In its January 15, 1980 submittal, BG&E states that accidental placement of fuel between the fuel racks or the racks and pool wall will be prevented by structural material. This will preclude an increase in the neutron multiplication factor due to a misplaced fuel assembly. From the above, the maximum possible neutron multiplication factor in the modified pool is 0.936.

In its April 14, 1980 submittal, BG&E states that neutron attenuation tests, to verify onsite that there is a sufficient amount of boron in the racks to maintain the k_{eff} below 0.95, will be performed after the fuel racks are installed in the pool.

A test fixture containing a neutron source and suitably shielded detectors will be lowered into each fuel storage location in each rack, one cell at a time. The backscattered neutron flux will be measured to confirm the existence of a neutron poison material.

Also in its April 14, 1980 submittal, BG&E states that verification that the boron remains in place throughout the life of the racks will be accomplished by placing samples in the high gamma areas of the spent fuel pool and then periodically removing them throughout the life of the fuel racks for various tests.

In case of a fuel handling accident, it is conceivable that an assembly could be laid across the tope of a ruel rack. In this case, the distance between the tops of the stored fuel and the bottom of the misplaced fuel will be greater than 25 inches which, according to NES's calculations, effectively separate the two groups of fuel. No increase in K_{eff} will result from this accident.

We find the above cited licensee's results agree well with results of parametric calculations made with other methods for similar fuel pool storage lattices. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the nominal fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

Since this neutron multiplication factor will increase if the boron loading in the plates is decreased below the stated minimum, an onsite neutron attentuation test is required to verify the presence of the boron ten in the racks and a surveillance program is required to verify continuously that the boron loading in any plate will not decrease below 0.024 grams of boron ten per square centimeter of plate. In this regard we find the tests proposed by BG&E in its April 14, 1980 submittal acceptable.

With these two tests and the limit on fuel loading, we find that all factors that could affect the neutron multiplication factor in this pool have been conservatively accounted for and that the maximum neutron multiplication factor in this pool with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margin to preclude criticality in the fuel. Accordingly, there is a Technical Specification which limits the effective neutron multiplication factor in the spent fuel pool to 0.95.

We find that when any number of the fuel assemblies, which BG&E described in these submittals and which have no more than 48.5 grams of uranium-235 per axial centimeter of fuel assembly, are loaded into the present and the proposed racks, the neutron multiplication factor will be less than 0.95.

On this basis, we conclude that when the plant's Technical Specifications are amended to prohibit the storage of fuel assemblies that contain more than 48.5 grams of uranium-235 per axial centimeter of fuel assembly, there is reasonable assurance that the health and safety of the public will not be endangered by the use of the present and proposed racks.

3.2 SPENT FUEL COOLING

The spent fuel pool at the Calvert Cliffs Nuclear Power Plant is located in the auxiliary building, and it is divided into two halves, i.e., one for each unit. Each of these halves of the pool has a volume of about 2.9×10^4 cubic feet. When it is filled with spent fuel assemblies, each half will hold more than 1.9×10^5 gallons of water.

The licensed thermal power for each of the two reactors is 2700 MWth. BG&E plans to refuel these reactors annually at which times about 72 of the 217 fuel assemblies in each core will be offloaded. To calculate the maximum heat load for a normal refueling, BG&E assumed a 7 day time interval between the shutdown of one reactor and the time when 72 of its fuel assemblies are placed in the spent fuel pool. This is assumed to occur 67 days after the offloading of one third of the other reactor into the spent fuel pool. On this basis BG&E calculated the maximum heat load for the twenty first annual refueling to be 17.3×10^6 BTU/hr.

The cooling system for the Calvert Cliffs spent fuel pool has two pumps and two heat exchangers. These are cross connected so that any combination of a pump and heat exchanger can be used to cool either half of the spent fuel pool. Additional cooling can be obtained by connecting the shutdown cooling of either unit to the spent fuel pool cooling system. Each spent fuel cooling pump is designed to pump 1390 gallons of water per minute. With both pumps and heat exchangers in operation, the spent fuel pool cooling system is designed to remove 20×10^6 BTU/hr while maintaining the fuel pool outlet water temperature at 127°F with 95°F service water cooling the heat exchangers. The shutdown cooling system, when connected to the spent fuel pool, is designed to remove 27×10^6 BTU/hr while maintaining the fuel pool outlet temperature at 130°F with 95°F service water cooling the heat exchanger.

Section 9.4.5 of the FSAR states that the spent fuel pool cooling system supplemented by the shutdown cooling system is capable of removing 38.7×10^6 BTU/hr. From Table 9-14 of the FSAR it is seen that the shutdown cooling system acting alone would be capable of removing 27.3×10^6 BTU/hr while maintaining the fuel pool outlet temperature at 130°F with 95°F service water.

BG&E states that alarms are provided to insure the maintenance of the water level in the spent fuel pool and to call attention to a high temperature condition. BG&E also states that the water in the Refueling Water Tanks or the Demineralized Water System can be used for make up to the spent fuel pool water. This can be supplied at flow rates or between 300 and 1390 gpm. Each of the two Refueling Water Tanks holds about 4×10^5 gallons of water.

Using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^3 seconds, and assuming a seven day cooling time, as used by BG&E, we calculate that the peak heat loads in the spent fuel pools after the twenty fourth annual refueling (i.e., 1728 fuel assemblies in the pools) could be 20×10^6 BTU/hr. We also calculate that the peak heat loads for a full core offload, which takes place 67 days after the twenty first annual refueling, could be 38.6×10^6 BTU/hr. For this we find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from 1056 to 1760 is 2.4×10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pool.

We find that the two trains of the present fuel pool cooling system can remove 20×10^6 BTU/hr while maintaining the fuel pool outlet water temperature at 127°F. We also find that in the case of a postulated single failure, which effectively shuts down one loop immediately after any normal refueling offload, the fuel pool outlet water temperature will not exceed 155°F. We also find that when these two trains are supplemented by the shutdown cooling system the 38.6×10^6 BTU/hr heat load can be removed with a spent fuel pool outlet water temperature of no more than 130°F. We find this acceptable since these heat loads are less than the heat removal capacity specified in Section 9.4.5 of the FSAR.

In the unlikely event that both spent fuel pool cooling loops were to fail when a full core that fills the racks had just been offloaded into the spent fuel pool the maximum possible heat up rate of the water would be 24°F/hr. Assuming that the average water temperature in the pool is initially 120°F, about four hours would elapse before there would be bulk boiling. After this, if the condensed steam was not returned to the spent fuel pool, the water level in the pool would start to drop. The maximum possible rate that it could drop would be 0.8 ft/hr. The alarms would call operator's attention to use makeup water from the Refueling Water Tanks or the Demineralized Water System. From this we find that, if this unlikely event took place, there would be sufficient time (several hours for operators to take action) to establish the 80 gpm flow of water that would be required at that time to maintain the water level in the pool.

We find that the present cooling capacity in the spent fuel pool of the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, will be sufficient to handle the incremental heat load that will be added by the proposed modifications. We also find that this incremental heat load will not alter the safety considerations of spent fuel pool cooling from those which we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed design.

3.3 INSTALLATION OF RACKS AND FUEL HANDLING

In its January 15, 1980 submittal, BG&E states that the North half of the pool is scheduled for rack removal and new installation in the summer of 1980. Under this schedule all the fuel residing in the spent fuel pool can be moved to the South half of the pool. The North pool can then be drained and the modification can be accomplished in a dry pool. The South half of the pool will likewise be modified under a schedule such that all the stored fuel can be transferred to the North pool. The modification will then be performed in a dry pool.

By taking advantage of the split-pool design, the licensee can install the new racks without having to move a rack close to or over spent fuel. After the new racks are installed, the fuel handling procedures in and around the pool will be the same as those that were in effect prior to the proposed modifications.

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

3.4 Structure Design

The inner wall of each storage cell is made up of a 0.060 inch thick sheet of 304L stainless steel, formed into a square with an inner dimension of 8-9/16 inches. On the outside of each of the four (4) sides of this inner wall, a poison sheet 6-1/2 inches wide is sandwiched between the inner wall and an external 0.060 inch thick stainless steel sheet.

The spent fuel pool is a reinforced concrete structure with a 3/16 inch thick stainless steel liner plate for leak tightness. The pool is 92 feet long, 25 feet wide, and 39 feet deep, with a 2 foot wall dividing the two halves. A slot in the wall has removable gates allowing for the movement of fuel between the two halves of the pool. The pool is an integral part of the auxiliary building and designed as a Seismic Category I structure, in accordance with the Calvert Cliffs Nuclear Power Plant FSAR.

The proposed modification for the spent fuel storage capacity expansion program has been reviewed in accordance with the NRC report "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 1978. The structural review consisted of an examination of the following areas: the proposed design criteria, the design loads and load combinations, methods of analysis, the dropped fuel accident, the material properties, the hydrodynamic effects, the fabrication and installation provisions, and the effect of increased loads on the floor slab and liner.

The material properties for structural components of the spent fuel racks used in the analyses were taken from Section III of the ASME Code. Load combinations and acceptance limits are in conformance with the NRC Standard Review Plan, Section 3.8.4 and ASME Section III, Subsection NF.

The Calvert Cliffs Nuclear Power Plant Units 1 and 2 high density spent fuel storage racks have been designed to meet the requirements for Seismic Category I structures. Detailed linear seismic analyses have been performed to verify the adequacy of the design to withstand the loadings encountered during the severe and extreme environmental conditions of the Operating Basis and Design Basis Earthquakes. Detailed non-linear time history seismic analyses have been performed to evaluate the maximum sliding of the storage racks and to determine the maximum frictional resistance load transmitted by the storage racks to the pool floor liner plate during the Design Basis Earthquake.

The effects of damping have not been considered in the non-linear sliding analysis. Excluding the effects of damping provides conservative analysis results because the portion of the external energy that would normally be absorbed in the damping element is available to increase the flexural deformation and the sliding of the fuel storage rack.

The natural frequency and the mode shape for each of the natural modes of vibration are calculated by using the Lanczos Modal Extraction Methods. The seismic response analyses are performed by the response spectrum modal superposition methods using the applicable response spectra curves. Individual modal responses of the system are combined in accordance with Section 1.2 of Regulatory Guide 1.92. The maximum responses (deflection, acceleration, velocity, shear forces, moments, stresses reaction loads) of the system for the three orthogonal spatial components (two (2) horizontal and one vertical) of an earthquake are combined on a square root of the sums of the squares (SRSS) bases (Regulatory Guide 1.92). For the non-linear time history seismic analysis of the spent fuel assembly/storage cell structure, a 10 x 10 storage rack and the stored fuel assemblies have been represented by a two dimensional lumped mass finite element model. The model consists basically of two coincident finite element cantilever beams, one representing the 100 storage cells and the other representing the 100 stored fuel assemblies attached to a "floor" mass by means of a non-linear sliding element. The non-linear time history seismic analyses are performed by step-by-step integration techniques (Houbolt Method) using the ANSYS computer program.

For the accidental fuel assembly drop condition, 1300 pound weight (fuel assembly) was postulated to drop on the rack from a height of 24 inches above the top of the rack. Three cases were considered: 1) a direct drop on the top of a 2 x 2 module, 2) a subsequent tipping of the fuel assembly and 3) a straight drop through the storage cell with impact to the rack base structure.

Linear and non-linear analysis techniques using energy balance methods were used to evaluate the structural damage resulting from a fuel assembly drop into the rack.

The acceptance criteria for the accidental fuel assembly drop on the rack are: (1) the resulting impact will not adversely affect the overall structural integrity of the rack and the leak-tightness integrity of the fuel pool floor and liner plate, and (2) the deformation of the impacted storage cells will not affect the ability to cool adjacent fuel elements.

The evaluation demonstrated that the energy developed by a freely falling fuel assembly from a height extending 24 inches (limited by the maximum lifting height of the crane) above a module would not cause liner plate perforation.

All materials, fabrication, installation, and quality control of the spent fuel racks are controlled in accordance with an effective quality assurance program meeting the requirements of 10 CFR 50, Appendix B and Subsection NF of the ASME Code.

The spent fuel pool structure was re-evaluated based on the increased loads caused by the new high density spent fuel storage racks using ACI-318-63 Code "Building Code Requirements for Reinforced Concrete," with the factored loads specified in Standard Review Plan 3.8.4. The licensee has calculated stresses at critical sections and found that these stresses are within the allowable stresses specified in the FSAR.

The structural aspects of the spent fuel storage racks have been evaluated based upon NRC guidance provided in the report entitled, OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 1978. Based upon our review of the analyses and the design done by the licensee, we conclude that the rack structure itself, the supporting pool liner and slab, are capable of supporting the applied loads without exceeding relevant stresses of Subsection NF or the FSAR Design Criteria. As previously stated, we find the material fabrication, installation, and quality control criteria acceptable. We conclude that the proposed modification of the Calvert Cliff spent fuel storage pool to the capacity of 1760 storage positions is in conformance with NRC requirements.

3.5 Fuel Handling

The NRC staff has published the results and recommendations of their generic review of the handling of heavy loads in the vicinity of spent fuel pools in NUREG-612. As a result of these recommendations, a program to review operating plants against the guidelines developed in this report is under way by the staff. Because Calvert Cliffs 1/2 is required to prohibit loads greater than the nominal weight of a fuel assembly and handling tool to be transported over spent fuel in the SFP, we have concluded that the likelihood of any other heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary during our review.

The potential consequences of fuel handling accidents in the spent fuel pool area presented in the Safety Evaluation Report (SER) dated August 1972 are not changed because the new high density racks increase the storage capacity of the SFP since, at worst, the number of fuel assemblies that could be damaged from a fuel handling accident is two (from a direct hit by a dropped assembly) under both the old and new storage rack designs and configurations.

3.6 Occupational Radiation Exposure

We have reviewed the licensee's plans for the removal and disposal of the close center high density racks and the installation of high density borated racks with respect to occupational radiation exposure. The occupational radiation exposure for this operation is estimated by the licensee to be about 10 man-rem. We consider this to be a conservative estimate. This estimate represents a small fraction of the total man-rem burden from occupational exposure at the plant.

This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job was being performed.

The licensee is planning on performing the modification in two stages. First the fuel residing in the North half of the pool will be moved to the South half. The North pool will then be drained and decontaminated and the modification can proceed in the dry pool with as low as is reasonably achievable background radiation. During decontamination of the racks the pool background radiation level is expected to be about 1.5 mrem/hr with the dose rate in the proximity of the racks averaging about 7.5 mrem/hr. Upon completion of the modification and refilling of water in the North pool, the fuel will be transferred from the South pool into the North pool, and the South pool will be likewise modified.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel pool area from radionuclide concentrations in the pool water and the spent fuel assemblies. The spent fuel assemblies themselves will contribute a negligible fraction of the dose rates in the pool area because of the depth of water shielding the fuel. Consequently, the occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.7 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material from both units. The waste treatment systems were evaluated in the Safety Evaluation Report (SER) for both units dated August 1972. There will be no change in the waste treatment systems or in the conclusions of the evaluation of these systems as described in Section 3.1.7 of the SER because of the proposed modification.

3.8 Material

The fuel storage racks are primarily fabricated from Type 304-L stainless steel with poison elements on each side of the storage cell. Based on our review of previous operating experience with similar stainless steel racks approved and in use, we have concluded that there is reasonable assurance that no signficant corrosion of the stainless steel will occur over the lifetime of the plant.

The poison elements consist of boron carbide (B_4C) powder in a fiberglass matrix fabricated by Carborundum Company. The material has been corrosion tested for eight months at Oak Ridge at a boron concentration of 2500 ppm (a value more than the CCNPP SFP boron concentration). No signficant corrosion occurred. We, therefore, would expect no accelerated corrosion of the rack materials. Although the B_4C composite material is subject to off-gasing under irradiation, the racks are of a vented design to prevent swelling of the can. The binder material in the B_4C composite does not decompose significantly and, therefore, the B_4C particles are held in place during irradiation. The irradiation data has been submitted to us previously on the Haddam Neck and Millstone Unit 1 Dockets Nos. 50-218 and 50-245, respectively, in the form of CBO-N-78-299 dated October 1978. We have licensed this poison for use in Spent Fuel Racks at these facilities and at LaCrosse having found their use acceptable. We find that the B_4C poison material is similarly acceptable for use at Calvert Cliffs.

4.0 Technical Specification

As indicated in the criticality analysis of this safety evaluation, the Uranium -235 enrichment would need to be increased from 44.0 to 48.5 grams per axial centimeter of fuel assembly. This corresponds to an increase from 3.7 to 4.1 weight percent. In conformance to the Technical Specification format, the enrichment in section 5.6.1 is in terms of weight percent (w/o) rather than grams per axial centimeter of fuel. The 4.1 w/o in section 5.6.1 is different from the 4.0 w/o in section 5.6.2 because they correspond to different types of storage (wet compared to dry) with different center-to-center distance between fuel assemblies. Specification 5.6 will need to be changed to relate the capacity of the combined pool to a limit of 1760 fuel assemblies.

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

Dated: September 19, 1980

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 47 AND 30 TO LICENSES NOS. DPR-53 AND DPR-69

RELATING TO MODIFICATION OF THE SPENT FUEL POOL

BALTIMORE GAS & ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNITS NOS. 1 AND 2

DOCKETS NOS. 50-317 AND 50-318

1.0 DESCRIPTION OF PROPOSED ACTION

By letters dated July 3, 1979, August 31, 1979 and January 15, 1980, Baltimore Gas and Electric Company (BG&E) proposed to change the spent fuel pool (SFP) storage design for Calvert Cliffs Nuclear Power Plant Units Nos. 1 and 2 (CCNPP) from the design which was reviewed and approved in Amendment Nos. 27 and 12 to Facility Operating License Nos. DPR-53 and DPR-69 issued January 4, 1978. This approved spent fuel storage capacity is 1056 fuel assemblies. The proposed change consists of increasing the existing spent fuel storage capacity for both units from 728 fuel assemblies (only half of the pool has been modified as authorized) to 1760 fuel assemblies. In response to our questions, BG&E submitted supplemental information by letters dated April 14 and 18, 1980, May 20 and 30, 1980, July 7, 1980, and September 12, 1980.

The modification evaluated in this environmental impact appraisal is the proposal by the licensee to replace the existing spent fuel storage racks with high density borated storage racks. This appraisal is being performed for a total capacity of 1830 fuel assemblies.

2.0 NEED FOR INCREASED STORAGE CAPACITY

The CCNPP SFP was originally designed with the storage capacity of 410 fuel assemblies (1-2/3 cores). The first refueling of CCNPP Unit 1 was in January 1977 at which time 72 fuel assemblies were replaced and stored in the SFP. The first 72 spent fuel assemblies from Unit 2 were placed in the SFP in September 1978. At that rate, 144 assemblies per year from both units would be discharged from the reactor to the SFP.

By letter dated January 4, 1978, we approved BG&E's request to expand their SFP capacity to 1056 fuel assemblies which would extend the storage capability of the pool through 1982 and leave room for a complete core discharge.

Spent fuel is not currently being processed on a commercial basis in the United States and storage capacity away from reactor sites is available only on an emergency basis. Additional spent fuel storage capacity is expected to become eventually available at facilities provided by the Department of Energy (DOE). Various options are being considered which could result in shipments to such interim facilities in 1984 and to long-term disposition facilities commencing in the 1995-2000 time frame. However, these dates are uncertain since the Congress has not yet authorized or funded these facilities.

Based on the above information, there is clearly a need for additional onsite spent fuel storage capacity to assure continued operation of the CCNPP units, with full core off-load capability, after the Spring of 1983. The proposed expansion of the total SFP capacity to 1760 assemblies would provide this capability until the Fall of 1987 using annual refueling cycles. If longer refueling cycles (such as the 18-months fuel cycle currently proposed by BG&E and under staff review for the next reloads of both units) begin as planned, operation of CCNPP Unit No. 1 could continue until the Spring of 1992 and Unit No. 2 could operate until the Fall of 1992 with full core off-load capability remaining.

3.0 THE FACILITY

The CCNPP units are described in the Final Environmental Statement (FES), issued by the Commission in April 1973, related to the section on operation of the facilities. Each unit is a Pressurized Water Reactor (PWR) which produces 2700 megawatts thermal (MWt) and has a gross electrical output of 835 megawatts (MWe). Pertinent descriptions of principal features of the plant as it currently exists are summarized below to aid the reader in following the evaluations in subsequent sections of this appraisal.

3.1 Fuel Inventory

Each CCNPP reactor contains 217 fuel assemblies. The fuel assemblies are a cluster of 176 fuel rods or sealed tubes arranged in a 14 by 14 array. The weight of the fuel, as UO_2, is approximately 207,200 pounds. About one-third of the assemblies are removed from the reactor and replaced with new fuel each year. Present scheduling is for the refueling outage to be in the first few months for Unit No. 2 and the last few months of each year for Unit No. 1.

The proposed modification of the SFP would not change the quantity of uranium fuel used in the reactor over the anticipated operating life of the facility and would not change the rate at which spent fuel is generated by the facility. The added storage capacity would increase the number of spent fuel assemblies that could be stored in the SFP and the length of time that some of the fuel assemblies could be stored in the pool.

3.2 Purpose of the SFP

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core and they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

3.3 SFP Cooling System

The SFP for CCNPP is provided with a cooling loop which removes decay heat from fuel stored in the SFP. The cooling system for the SFP has two pumps and two heat exchangers. These are cross-connected so that any combination of a pump and heat exchanger can be used to cool the SFP for either Unit No. 1 or No. 2. There is also additional cooling available from valving the shutdown cooling system of either unit to the SFP cooling system. Each SFP cooling pump is designed to pump 1390 gallons of water per minute. With both pumps and heat exchangers in operation the spent fuel cooling system is designed to remove $20x10^6$ BTU/hr while maintaining the fuel pool outlet water temperature at 127°F with 95°F service water cooling the heat exchangers. The shutdown cooling system when connected to the SFP is designed to remove $27x10^6$ BTU/hr while maintaining the fuel pool outlet temperature at 130°F with 95°F service water cooling the heat exchanger. After the SFP modification, the maximum possible total heat load including uncertainties will be $17.3x10^6$ BTU/hr, within the capacity of the SFP cooling system. Our Safety Evaluation finds the maximum possible temperatures of 127°F and 155°F, for both SFP loops operating and single failure leaving one SFP loop operating, respectively, to be acceptable.

3.4 SFP Purification System

The SFP purification loop consists of a cartridge filter, a mixed bed demineralizer and the required piping, valves and instrumentation. The SFP cooling system pumps draw water from the pool or the refueling cavity. A fraction of this flow is passed through the SFP purification loop. The water is returned to the pool or the refueling cavity.

Because we expect only a small increase in the radioactivity released to the pool water as a result of the proposed modification as discussed in Section 4.4 of this environmental impact appraisal, we conclude the SFP filtering system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels which have existed prior to the modification.

3.5 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems are evaluated in the FES dated April 1973. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

4.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

4.1 Land Use

The external dimensions of the SFP will not change because of the proposed expansion of its storage capacity; therefore, no additional commitment of land is required. The SFP is intended to store spent fuel assemblies under water for a period of time to allow shorter-lived radioactive isotopes to decay and to reduce their thermal heat output. This type of use will remain unchanged by the modification but the additional storage capacity would provide for an additional nine normal refuelings. Thus, the proposed modification would result in more efficient use of the land already designed for spent fuel storage.

4.2 Water Use

There will be no significant change in plant water usage as a result of the proposed modification. As discussed subsequently, storing additional spent fuel in the SFP will increase the heat load on the SFP cooling system which is transferred to the service water system and to the plant salt water system. The modification will not change the flow rate within these cooling systems. Since the temperature of the SFP water during normal refueling operations will remain below 127°F presented in the FSAR and evaluated in the FES, the rate of evaporation and thus the need for makeup water will not be significantly changed by the proposed modification.

4.3 Nonradiological Effluents

There will be no change in the chemical or biocidal effluents from the plant as a result of the proposed modification.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be additional discharge of heat to the atmosphere and to the Chesapeake Bay. Storing spent fuel in the SFP for a longer period of time will add more heat to the SFP water. The SFP heat exchangers are cooled by the service water system which in turn is cooled by the salt water system. As discussed in the staff's Safety Evaluation, the maximum incremental heat load resulting from the SFP modification is 2.4×10^6 BTU/hr. Compared with the existing heat load (210×10^6 BTU/hr) on the plant salt water cooling system, this small additional heat load from the SFP cooling system will be negligible.

4.4 Radiological Impacts

4.4.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is fuel which has decayed at least three years. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90 are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

4.4.2 Effect of Fuel Failure on the SFP

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

Based on the operational reports submitted by the licensee and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

Experience indicates that there is little radionuclide leakage from Zircaloy-clad spent fuel stored in pools for over a decade. Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel elements which developed defects during reactor exposure, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden Units Nos. 1 and 2. Based on the operational reports submitted by licensees and discussions with the operators, there has not been any significant leakage of fission products from spent reactor fuel stored in the MO pool or the NFS pool. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the MO pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity.

A recent Battelle Northwest Laboratory (BNL) report, "Behavior of Spent Nuclear Fuel in Water Pool Storage: (BNWL-2256 dated September 1977), states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of 10^{-3} to 10^{-4} μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the basin are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been determined in laboratory studies and also by casual observations of pellet behavior when broken rods are stored in pools.

4.4.3 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing the additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defected fuel. However, we have conservatively estimated that an additional 150 curies per year of Krypton-85 may be released which would result in an additional total body dose at the site boundary to an individual of less than 0.001 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.001 man-rem/yr. This is less than the natural fluctuations in the dose this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of less than 0.5% of the exposures from the plant evaluated in the FES for the individual (Table V-5) and the population (Table V-6). Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Storing additional spent fuel assemblies in the pool is not expected to increase the bulk water temperature above 127°F during normal refuelings used in the design analysis. Since the temperature of the pool water will normally be maintained below 127°F, it is not expected that there will be any significant change in evaporation rates or the release of tritium or iodine as a result of the proposed modifications from that previously evaluated.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than from the SFP. Therefore, even if there were a slightly higher evaporation rate from the SFP, the increase in tritium and iodine released from the plant as a result of the increase in the stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. The plant radiological effluent technical specifications, which are not being changed by this action, restrict the total releases of gaseous radioactivity from the plant including the SFP.

4.4.4 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the cartridge filter and the demineralizer and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filter and demineralizer. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste due to the modification, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by 64 cubic feet of resin a year from the demineralizer (2 resin beds/year). Because Unit 1 has operated for 5 years and Unit 2 has operated for about 4 years, we have estimated the annual average amount of solid waste shipped from both units from the volume of solid waste shipped from a representative number of pressurized water reactors during 1973 to 1976. This is 18,300 cubic feet per year for both units. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 64 cubic feet (ft^3) per year, the increase in total waste volume shipped would be less than 0.4% and would not have any significant environmental impact.

In addition to the above, there are also the present spent fuel racks to be removed from the SFP from both units and disposed of. They will be hydrolazed to remove all loose contamination, crated whole and stored on site. At some time in the future they will be electropolished to remove all surface contamination and sold as clean scrap. If the racks cannot be cleaned to the extent that they can be sold as clean scrap, then the crated 22,000 ft^3 volume of SFP racks would be shipped to a low level waste disposal site as additional solid waste. Averaged over the lifetime of the plant, this would increase total waste shipped from the plant by about 3% and would not have any significant environmental impact.

The activity in the electropolishing solution, as described above, will be deposited on demineralizer resins and will add a total of about 10 ft^3 of resin to the radwaste inventory of the plant. This will have a negligible impact.

4.4.5 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the station as a result of the proposed modification. The amount of radioactivity on the SFP cartridge filter and demineralizer might slightly increase due to the additional spent fuel in the pool but this increase of radioactivity should not be released in liquid effluents from the station.

The cartridge filter removes insoluble radioactive matter from the SFP water. This is periodically removed to the waste disposal area in a shielded cask and placed in a shipping container. The insoluble matter will be retained on the filter or remain in the SFP water.

The resins are periodically flushed with water to the spent resin tank. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system. With respect to leaks in the SFP liner, no water leaks have been observed from the SFP.

4.4.6 Occupational Exposures

We have reviewed the licensee's plan for the removal, disassembly and disposal of close center high density racks and the installation of high density borated racks for both units with respect to occupational radiation exposure. The occupational radiation exposure for this operation is estimated by the licensee to be about 10 man-rem. We consider this to be a reasonable estimate. This operation is expected to be performed only once during the lifetime of the station and will therefore represent a very small fraction of the total man-rem burden from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in the stored fuel assemblies on the basis of information supplied by the licensee and by utilizing realistic assumptions for occupancy times and for dose rates in the SFP area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification will add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

4.4.7 Impacts of Other Pool Modifications

As discussed above, the additional environmental radiological impacts in the vicinity of CCNPP-1&2 resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the CCNPP-1&2 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that an SFP modification at any other facility should not significantly contribute to the environmental impact of CCNPP-1&2 and that the CCNPP-1&2 SFP modification should not contribute significantly to the environmental impact of any other facility.

4.4.8 Impacts on the Community

The new storage racks were fabricated offsite and shipped to the CCNPP, where they are stored. Only a few truck or rail shipments would be involved in shipment of these racks and disposal of the present ones. The impacts of dismantling the present racks and installing the new ones will be limited to those normally associated with metal working activities. During fuel handling operations, the impacts will be confined to the refueling floor of the reactor building. Consequently, no significant impact on the community is expected to result from the fuel rack conversion or subsequent operation with increased storage of spent fuel in the SFP.

4.5 Evaluation of Radiological Impact

As discussed above, the proposed modification does not significantly change the radiological impact evaluated in the FES.

5.0 ENVIRONMENTAL IMPACT OF POSTULATED ACCIDENTS

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES for CCNPP dated April 1973.

Additionally, the NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because CCNPP has the TS requirement to prohibit the movement of loads in excess of 1600 pounds over fuel assemblies in the SFP (TS 3.9.7), we have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is under way.

6.0 ALTERNATIVES

The staff has considered the following alternatives to the proposed expansion of the SFP storage capacity at CCNPP-1&2: (1) reprocessing the spent fuel; (2) shipment of spent fuel to a separate fuel storage facility; (3) shipment of spent fuel to another reactor site; (4) reduced plant operation; and (5) shutdown of facility. These alternatives are discussed below.

6.1 Reprocessing of Spent Fuel

As discussed earlier, none of the three commercial reprocessing facilities in the United States is currently operating. The MO has not been licensed and NFS informed the NRC on September 22, 1976, that it was "withdrawing from the nuclear fuel reprocessing business". The NFS facility is on land owned by the State of New York and leased to NFS through 1980. The Allied-General Nuclear Services (AGNS) reprocessing plant at Barnwell, South Carolina, received a construction permit on December 18, 1970. In October 1973, AGNS applied for an operating license for the reprocessing facility; construction of the reprocessing facility is essentially complete but no operating license has been granted. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU of spent fuel in the onsite storage pool, on which construction has also been completed but hearings with respect to this application have not been held and no license has been granted.

In 1976, Exxon Nuclear Company, Inc. submitted an application for a proposed Nuclear Fuel Recovery and Recycling Center (NFRRC) to be located at Oak Ridge, Tennessee. The plant would include a storage pool that could store up to 7,000 MTU in spent fuel. However, licensing review of this application was discontinued in 1977 as discussed below.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the U. S. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U. S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling".

On December 23, 1977, the NRC terminated the fuel cycle licensing actions involving mixed oxide fuel (GESMO) (Docket No. RM-50-5), the AGNS' Barnwell Nuclear Fuel Plant Separation Facility, Uranium Hexafluoride Facility and Plutonium Product Facility (Dockets Nos. 50-332, 70-1327 and 70-1821), the Exxon Nuclear Company, Inc. NFRRC (Docket No. 70-1432), and the NFS West Valley Reprocessing Plant (Docket No. 50-201). The Commission also announced that it would not at this time consider any other applications for commercial facilities for reprocessing spent fuel, fabricating mixed-oxide fuel, and related functions. Consideration of these or comparable facilities has been deferred indefinitely. Accordingly, the Staff considers that shipment of spent fuel to such facilities for reprocessing is not a feasible alternative to the proposed expansion of CCNPP SFP storage capacity, especially when considered in the revelant time frame - i.e., 1983 and at least several years thereafter - when the expanded capacity will be needed. Even if the government policy were changed tomorrow to allow reprocessing of spent fuel, the present backlog of spent fuel at various plants and the time it would take to bring adequate reprocessing capacity on line would require that the current spent fuel be stored somewhere for up to another ten years.

6.2 Independent Spent Fuel Storage Facility

An alternative to expansion of onsite SFP storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1,000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. The fuel storage pools at MO and NFS are functioning as smaller ISFSIs although this was not the original design intent. The license for the General Electric (GE) facility was amended on December 3, 1975 to increase the storage capacity to about 750 MTU; and, as of August 30, 1978, 310 MTU was stored in the pool in the form of 1196 spent fuel assemblies. An application for an 1100 MTU capacity addition is pending and the present schedule calls for completion in 1980 if approved. However, by a motion dated November 8, 1977, GE requested the Atomic Safety and Licensing Board to suspend indefinitely further proceedings on this application. This motion was granted.

The staff has discussed the status of storage space at MO with GE personnel. We were informed that GE is primarily operating the MO facility to store either fuel owned by GE (which had been leased to utilities on an energy basis), or fuel which GE has previously contracted to reprocess. We were also informed that the present GE policy is not to accept spent fuel for storage except fuel for which GE has a previous commitment. There is no such commitment for CCNPP spent fuel. Storage of the CCNPP spent fuel at the existing reprocessing facilities is not a viable alternative to the expansion of the CCNPP spent fuel pools.

The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool at West Valley. Although the storage pool is not full, NFS has indicated that it is not accepting additional spent fuel, even from the reactor facilities with which it had reprocessing contracts.

If the receiving and storage station at Barnwell is eventually licensed to accept spent fuel, as discussed in Section 6.1, it would be functioning as an ISFSI until the reprocessing facilities there are licensed to operate. The pool has unused space for about 400 MTU, but AGNS has indicated that it does not wish to operate the storage facility without reprocessing.

With respect to construction of new ISFSIs, on October 6, 1978 the NRC proposed a new Part 72 of its regulations specifying procedures and requirements for the issuance of relevant licenses, along with requirements for the siting, design, operation and record keeping activities of the facilities (43 FR 46309). The staff has estimated that at least five years would be required for completion of an ISFSI. This estimate assumes one year for preliminary design; one year for preparation of the license application, environmental report, and licensing review in parallel with one year for detail design; two and one-half years for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for additional independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merrill Lynch, Pierce, Fenner and Smith, Inc. issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975 (ANS Transactions, 1975 Winter Meeting, Vol. 22, TANSAO 22-1-836, 1975). In 1974, E. R. Johnson Associates estimated the construction cost would be equivalent to approximately $9,000 per spent fuel assembly.

Several licensees have evaluated construction of an ISFSI and have provided cost estimates. In 1975, Connecticut Yankee, for example, estimated that an independent facility with a storage capacity of 1,000 MTU (BWR and/or PWR assemblies) would cost approximately $54 million and take about five years to put into operation. The Commonwealth Edison Company estimated the construction cost of an ISFSI in 1975 at about $10,000 per fuel assembly. To this would be added the costs for maintenance, operation, safeguards, security, interest on investment, overhead, transportation and other costs. These costs are significantly larger than the estimated cost of the increased storage capacity which will be obtained by expanding the present reactor pools (approximately $5,000/assembly).

For the long term, DOE is modifying its program for nuclear waste management to include design and evaluation of a long term repository to provide Government storage of unreprocessed spent fuel rods in a retrievable condition. It is estimated that the long term storage facility will start accepting commercial spent fuel in the time frame of 1995 to 2000. The criteria for acceptance is that the spent fuel must have decayed a minimum of ten years so it can be stored in dry condition without need for forced air circulation.

As an interim alternative to the long term retrievable storage facility, on October 18, 1977, DOE announced a new "spent nuclear fuel policy". DOE will determine industry interest in providing interim fuel storage services on a contract basis. If adequate private storage services cannot be provided, the Government will provide interim fuel storage facilities. These interim facilities would be designed for storage of the spent fuel under water. DOE, through its Savannah River Operations Office, is preparing a conceptual design for an interim spent fuel storage pool of about 5000 MTU capacity. Congressional authorization has been requested to borrow $300 million for design and construction of this facility.

Based on recent DOE testimony before Congress, it appears that the earliest DOE's interim storage pool would be licensed to accept spent fuel would be about 1984. However, DOE has also stated its intent not to accept any spent fuel for interim storage that has not decayed for a minimum of five years.

Based on the above information, neither an independent spent fuel storage installation nor a Government interim storage facility appears to be a feasible alternative to meet the licensee's needs. The staff does not regard the alternative of storing spent fuel at MO or Barnwell as offering a significant environmental advantage over construction and use of an expanded storage facility at CCNPP. The availability of this alternative is speculative and it also would be considerably more expensive. Furthermore, constructing a new ISFSI or a Governmental interim storage facility would clearly have a greater environmental impact than the proposed action. It would require additional land and considerable equipment and structures, whereas installing new racks at CCNPP requires only the small amount of material necessary to construct the racks and minor personnel exposure during installation, if the present racks are contaminated prior to their removal.

6.3 Storage at Another Reactor Site

BG&E does not have another nuclear plant other than the CCNPP in their system that is operating or under construction. According to a survey conducted and documented by the former Energy Research and Development Administration, up to 27 of the operating nuclear power plants will lose the ability to refuel during the period 1977-1986 without additional spent fuel storage pool expansions or access to offsite storage facilities. Thus, the licensee cannot assuredly rely on any other power facility to provide additional storage capability except on a short term emergency basis. If space were available in another reactor facility, the cost would probably be comparable to the cost of storage at a commercial storage facility.

6.4 Reduced Plant Output

Nuclear plants are usually base-loaded because of their lower costs of generating a unit of electricity compared to other thermal power plants on the system. Therefore, reducing the plant output to reduce spent fuel generation is not an economical use of the resources available. The total production costs remain essentially constant, irrespective of plant output. Consequently, the unit cost of electricity is increased proportionately at a reduced plant output. If the plant is forced to substantially reduce output because of spent fuel storage restrictions, the licensee would be required to purchase replacement power or operate its higher cost fossil-fired units, if available, without any accompanying environmental advantage. The cost of electricity would therefore be increased without any likely reduction of environmental impact.

6.5 Shutdown of Facility

Storage of spent fuel from the CCNPP units in the existing racks is possible but only for a short period of time. As discussed above, if expansion of the SFP capacity is not approved, if an alternate storage facility is not located, and even if 18-month fuel cycles are used, BG&E would have to shut down Unit No. 1 in late 1987 and Unit No. 2 in late 1986 due to a lack of spent fuel storage facilities, resulting in the cessation of at least 1630 MWe net electrical energy production.

According to the licensee, the levelized annual fixed charge on investment is $101,300,000/yr and on fuel is $10,000,000 for a total of $111,300,000/yr. BG&E states that if a forced shutdown from lack of fuel storage capabilities occurred, they would keep the majority of their 380-man staff over the short term for possible restart. This size crew would cost about $10,000,000/yr.

If CCNPP terminated operations, replacement power would be derived principally from operation of fossil fuel plants. Monthly replacement power would cost about $47.5 million at current rates. In addition to the cost of replacement power, the real cost could be a power curtailment and resultant hardships in the BG&E service area.

6.6 Comparison of Alternatives

In Section 4 of this environmental impact appraisal the incremental environmental impacts of the proposed expansion of the SFP storage capacity were evaluated and were found to be insignificant. Therefore, none of the alternatives to this action offers a significant environmental advantage. Furthermore, alternatives (1), reprocessing, and (2), storage at an independent spent fuel storage facility, are not presently available to the licensee and are not likely to become available in time to meet the licensee's need. Alternative (3), shipment to another reactor site, would be a short term emergency solution but would eventually involve shipment to another temporary storage facility. Alternatives (4), reducing the plant output, and (5), shutdown of the facility, would both entail substantial additional expense for replacement electrical energy which may not be available for prolonged periods of time.

Table 1 presents a summarized comparison of the alternatives, in the order presented in Subsections 6.1 through 6.5. From inspection of the table, it can be seen that the most cost effective alternative is the proposed SFP modification, which is included as alternative 6. The SFP modification would provide the required storage capacity, while minimizing environmental effects, capital cost and resources committed. The staff therefore concludes that expansion of the CCNPP SFP storage capacity is superior to the alternatives available or likely to become available within the necessary time frame.

TABLE 1

COMPARISON OF ALTERNATIVES

	Alternative	Cost	Benefit
1.	Reprocessing of Spent Fuel	>$10,000/assembly	Continued production of electrical energy by Units 1 & 2. This alternative is not available either now or in the foreseeable future.
2a.	Storage at Reprocessor's Facility	$3,000 to $6,000/assembly per yr* plus shipping costs of $12,000 per assembly.	Continued production of electrical energy by Units 1 & 2. This alternative is not available now or in the foreseeable future.
2b.	Storage at a new Independent Facility	$20,00-$40,000/assembly plus operating and transportation costs, and environmental impacts related to development of a new facility.	Continued production of electrical energy by Units 1 & 2. This alternative could not be available for at least 4 years.
3.	Storage at Other Nuclear Plants	Cost of shipment to other facility plus cost for subsequent shipment to an ISFSI; increased environmental costs of extra shipping and handling.	Continued production of electrical energy. However, this alternative is unlikely to be available.
4.	Reduction in Plant Output	See below for replacement electricity costs. Amount of replacement required would be equivalent to at least 50% reduction in rated output of Units 1 and 2.	Continued production of electrical energy by Units 1 and/or 2 - but at much higher unit cost. The generation of replacement electricity elsewhere would probably create no less impacts.

* Since NFS and MO are not accepting spent fuel for storage, the cost range reflects prices that were quoted in 1972 to 1974.

TABLE 1

COMPARISON OF ALTERNATIVES

	Alternatives	Cost	Benefit
5.	Reactor Shutdown	Replacement electricity costs are estimated to be as much as $1,560,000/day if both units are shut down, plus the costs of maintenance and security of the plant.	Environmental impacts associated with plant operation would cease but the generation of replacement electricity elsewhere would probably create no less impacts.
6.	Increased Storage Capacity of CCNPP SFP	$5,000/added assembly storage space	Continued production of electrical energy by CCNPP Units 1 & 2

NOTE: This cost-benefit analysis was commenced prior to the issuance of NUREG-0575, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel dated August 1979, and is provided in lieu of a reference to the generic statement.

7.0 EVALUATION OF PROPOSED ACTION

7.1 Unavoidable Adverse Environmental Impacts

7.1.2 Radiological Impacts

Expansion of the storage capacity of the SFP will not create any significant additional adverse radiological effects. As discussed in Section 4.4, the additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.001 mrem/yr and 0.001 man-rem/yr, respectively, and is less than the natural fluctuations in the dose this population would receive from background radiation. The total dose to workers during removal of the present storage racks and installation of the new racks is estimated to be about 10 man-rem. Operation of the plant with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the present total annual occupational exposure at this facility.

7.2 Relationships Between Local Short Term Use of Man's Environment and the Maintenance and Enhancement of Long Term Productivity

Expansion of the SFP storage capacity would permit more efficient use of the land already committed to this purpose. There would be no other changes from the evaluation in the FES.

7.3 Irreversible and Irretrievable Commitments of Resources

7.3.1 Water, Land and Air Resources

The proposed action will not result in any significant change in the commitments of water, land and air resources as identified in the FES. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings between fuel assemblies.

7.3.2 Material Resources

Under the proposed modification, the present spent fuel storage racks will be replaced by new racks that will increase the storage capacity of the SFP by 704 spent fuel assemblies. The new spent fuel storage racks consist of type 304-L stainless steel square box with an inner dimension of 8-9/16 inches approximately 15.2 feet long with a 0.06 inch wall thickness. The largest storage rack consists of a 10x10 array of individual storage boxes, a base with four legs, and various bracing and support members. The fuel assemblies sit on bars across the bottom of each storage box. The top of the storage boxes are flared to form a lead-in funnel. Each rack is estimated to weigh approximately 29,000 lbs. empty. A total of 19 of these racks will be used in each section of the SFP, approximately weighing 551,000 lbs.

Thus, the resources to be committed for fabrication of the new spent fuel storage racks total approximately 551,000 pounds of stainless steel. The amount of stainless steel used annually in the U. S. is about 2.82×10^{11} lbs. The material is readily available in abundant supply. The amount of stainless steel required for fabrication of the new racks is a small amount of this resource consumed annually in the U. S. and therefore can be ignored in this Appraisal. The amount of boron required in the borated rack is insignificant. We conclude that the amount of material required for the new racks at CCNPP is insignificant and does not represent a significant irreversible commitment of material resources.

8.0 BENEFIT-COST BALANCE

This section summarizes and compares the cost and the benefits resulting from the proposed modification to those that would be derived from the selection and implementation of each alternative. Table 1 presents a tabular comparison of these costs and benefits. The first three alternatives are not possible at this time or in the foreseeable future except on a short term emergency basis. Alternatives 4 and 5 have higher cost and no less environmental impacts than that of increasing storage capacity of CCNPP SFP.

From examination of the table, it can be seen that the most cost-effective alternative is the proposed spent fuel pool modification. As evaluated in the preceding sections, the environmental impacts associated with the proposed modification would not be significantly changed from those analyzed in the Final Environmental Statement for CCNPP Units 1 and 2 issued in April 1973.

9.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines 40 CFR 1500.6. We have determined that the proposed license amendment will not significantly affect the quality of the human environment and that there will be no significant environmental impact attributable to the proposed action other than that which has already been predicted and described in the Final Environmental Statement for CCNPP dated April 1973. Therefore, the staff has found that an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Dated: September 19, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION 7590-01

DOCKETS NOS. 50-317 AND 50-318

BALTIMORE GAS & ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 47 and 30 to Facility Operating Licenses Nos. DPR-53 and DPR-69, respectively, issued to Baltimore Gas & Electric Company (the licensee), which revised the licenses and their appended Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 (the facilities) located in Calvert County, Maryland. The amendments are effective as of their date of issuance.

The amendments authorize replacement of the existing racks in both sides of the spent fuel pool of the facilities with borated racks of a design capable of accommodating up to 830 assemblies for Unit 1 and 930 assemblies for Unit 2. The modification and subsequent use of the two-section pool permits a total of 1760 fuel assemblies to be stored instead of the previously authorized total of 1056 assemblies.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Consideration of Proposed Modification to Facilities Spent Fuel Storage Pool in connection with this action was published in the Federal Register on March 7, 1980 (46 FR 14981). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal of the action being authorized and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action significantly greater than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility dated April 1973, and the action will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the applications for amendments dated July 3 and August 31, 1979, January 15, 1980, as supplemented April 14 and 18, May 20 and 30, July 7, and September 12, 1980, (2) Amendment Nos. 47 and 30 to License Nos. DPR-53 and DPR-69, (3) the Commission's concurrently issued Safety Evaluation, and (4) the Commission's concurrently issued Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Calvert County Library, Prince Frederick, Maryland 20678. A single copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 19th day of September, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing