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

8010080 252

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 extern[illegible] 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