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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 76
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated August 20, 1982 as supplemented, September 2 and December 20, 1983 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 2.B(2) and 2.C(2) of Provisional Operating License No. DPR-16 are hereby amended to read as follows:

   2.B(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Facility Description and Safety Analysis Report, as supplemented and amended;

   2.C(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76 , are hereby incorporated in the license. GPU Nuclear Corporation 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

Walter A. Paulson

Walter A. Paulson, Acting Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 17, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 76

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
5.3-1	5.3-1
5.3-2	5.3-2

5.3 AUXILIARY EQUIPMENT

5.3.1 Fuel Storage

A. Normal storage for unirradiated fuel assemblies is in critically safe new fuel storage racks in the reactor building storage vault; otherwise, fuel shall be stored in arrays which have a K_{eff} less than 0.95 under optimum conditions of moderation or in NRC-approved shipping containers.

B. The spent fuel shall be stored in the spent fuel storage facility which shall be designed to maintain fuel in a geometry providing a K_{oo} less than or equal to 0.95.

C. The fuel to be stored in spent fuel storage facility shall not exceed a maximum average planar enrichment of 3.01 w/o U-235.

D. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility.

E. The 30 ton spent fuel shipping cask shall not be lifted more than 6 inches above the top plate of the cask drop protection system. Vertical limit switches shall be operable to assure the 6 inch vertical limit is met when the cask is above the top plate.

F. The temperature of the water in the spent fuel stored pool, measured at or near the surface, shall not exceed 125°F.

G. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 2600.

BASIS

The specification of K_{oo} less than or equal to 0.95 in the spent fuel storage facility assures an ample margin from criticality. Criticality analysis was performed on the poison racks to insure that a K_{oo} of 0.95 would not be exceeded. The basis for this analysis assumed an average planar lattice enrichment of 3.01 w/o U-235 and includes manufacturing tolerances.

The effects of a dropped fuel bundle onto stored fuel in the spent fuel storage facility have been analyzed. This analysis shows that the fuel bundle drop would not cause doses resulting from ruptured fuel pins that exceed 10 CFR 100 limits (1,2,3) and that dropped waste cans will not damage the pool liner.

The elevation limitation of the spent fuel shipping cask to no more than 6 inches above the top plate of the cask drop protection system prevents loss of the pool integrity resulting from postulated drop accidents. An analysis of the effects of a 100 ton cask drop from 6 inches has been done (4) which showed that the pool structure is capable of sustaining the loads imposed during such a drop. Limit switches on the crane restrict the elevation of the cask to less than or equal to 6 inches when it is above the top plate.

Detailed structural analysis of the spent fuel pool was performed using loads resulting from the dead weight of the structural elements, the building loads, hydrostatic loads from the pool water, the weight of fuel and racks stored in the pool, seismic loads, loads due to thermal gradients in the pool floor and walls, and dynamic load from the cask drop accident. Thermal gradients result in two loading conditions; normal operating and the accident conditions with the loss of spent fuel pool cooling. For the normal condition, the containment air temperature was assumed to vary between 65°F and 110°F while the pool water temperature varied between 85°F and 125°F. The most severe loading from the normal operating thermal gradient results with containment air temperatures at 65°F and the water temperature at 125°F. Air temperature measurements made during all phases of plant operation in the shutdown heat exchanger room, which is directly beneath part of the spent fuel pool floor slab, show that 65°F is the appropriate minimum air temperature. The spent fuel pool water temperature will alarm in the control room before the water temperature reaches 120°F.

Results of the structural analysis show that the pool structure is structurally adequate for the loadings associated with the normal operation and the condition resulting from the postulated cask drop accident (5) (6). The floor framing was also found to be capable of withstanding the steady state thermal gradient conditions with the pool water temperature at 150°F without exceeding ACI Code requirements. The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F (5).

Since the cooled fuel pool water returns at the bottom of the pool and the heated water is removed from the surface, the average of the surface temperature and the fuel pool cooling return water is an appropriate estimate of the average bulk temperature; alternately the pool surface temperature could be conservatively used.

References

1. Amendment No. 78 to the FDSAR (Section 7)
2. Supplement No. 1 to Amendment No. 78 to the FDSAR (Question 12)
3. Supplement No. 1 to Amendment No. 78 of the FDSAR (Question 40)
4. Supplement No. 1 to Amendment No. 68 of the FDSAR.
5. Revision No. 1 to Addendum 2 to Supplement No. 1 to Amendment No. 78 of FDSAR (Questions 5 and 10)
6. FDSAR Amendment No. 79

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated August 20, 1982, as supplemented September 2 and December 20, 1983, GPU Nuclear Corporation (GPU) submitted an application to increase the storage capacity of the spent fuel pool (SFP) by replacing the existing racks with new storage racks ("reracking"). By letters dated May 30, June 4, and June 13, 1984 GPU provided additional clarification in response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information. This would be the second rerack for Oyster Creek, the first being authorized by Amendment No. 22 on March 30, 1977 which increased the capacity of the SFP from its original capacity of 840 to 1800 fuel elements.

The present amendment would authorize the licensee to increase the storage capacity of the SFP from the current capacity of 1800 fuel assemblies to 2600 fuel assemblies with average planar enrichments no greater than 3.01 weight percent U-235. This request includes Amendment No. 79 to the Facility Description and Safety Analysis Report (FDSAR).

At the present time, there are 980 spent fuel assemblies in the SFP. The licensee estimates that full-core reserve in the SFP would be lost following the 1985 refueling. Since this date is earlier than the date a federal depository should be available for spent fuel [1998-Nuclear Waste Policy Act of 1982, Sec. 302(a)(5)] additional spent fuel capacity is needed.

Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on October 8, 1982 (47 FR 44647). No request for hearing and no comments were received.

In the August 20, 1982 letter, GPU stated that they would provide supplemental information in 1983 which would address the areas of reactivity considerations, pool structural adequacy, and heat load. This intent was also noted in the October 8, 1982 Federal Register notice. The supplemental information was provided in letters dated September 2 and December 20, 1983.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

The SFP criticality calculations are based on unirradiated fuel assemblies with no burnable poisons which have a maximum average planar enrichment of 3.01 weight percent U-235.

2.1.1 Analysis Methods - Southern Science (a division of Black and Veatch) performed the criticality analyses for the spent fuel racks. The reference method for the nuclear criticality analyses is the AMPX-KENO computer package, using the 123 group GAM-THERMOS cross-section set and the NITWAL subroutine for U-238 resonance shielding effects. The licensee's submittal referenced a number of benchmark calculations against critical experiments for this code package. Results of these calculations indicate a calculational bias of 0, with an uncertainty of ± 0.0028Δk corresponding to a 95 percent probability at a 95 percent confidence level. In addition, a small correction of 0.0036Δk in the calculational bias was necessary to account for the slightly greater gap thickness between fuel assemblies in the Oyster Creek spent fuel rack compared to the corresponding thickness in the benchmark critical experiments. For investigation of mechanical tolerance effects, the CASMO code and a four-group diffusion/blackness theory method of analysis were used to evaluate trends and the small incremental reactivity effects that would otherwise be lost in the KENO statistical variation.

The staff finds the analysis methods and uncertainty allowances used for the high density storage racks acceptable.

2.1.2 Spent Fuel Rack Storage - The criticality of fuel assemblies in the Oyster Creek SFP is prevented by maintaining a minimum separation of 6.198 inches between rows of fuel assemblies and by inserting the neutron absorber, Boraflex, between rows of fuel assemblies. Several spent fuel racks using Boraflex have received NRC approval. The NRC acceptance criterion for spent fuel storage is that there is a 95 percent probability at a 95 percent confidence level (including uncertainties) that K_{eff} of the fuel assembly array will be less than 0.95 for all storage conditions.

In addition to the calculational method uncertainty mentioned previously, uncertainties and biases due to fuel cell dimensions, pitch between rows of fuel cells, Boraflex loading, fuel pellet density, fuel position, and pool water temperature are included either by using worst case initial conditions or by performing sensitivity studies to obtain the appropriate values. All uncertainties were at least 95/95 probability/confidence values.

Using these methods and assumptions, the nominal k_{eff} of the spent fuel racks is calculated as 0.9295. The fuel is assumed to be unirradiated with no burnable poison at a maximum average planar enrichment of 3.75 weight percent U-235. The basic storage rack cell used for the analysis included a fuel bundle wherein the average planar enrichment of each of the fuel rods was 3.01 weight percent U-235. In reality, a fuel bundle will have a distribution of fuel rod enrichments rather than a uniform rod enrichment. Independent calculations with distributed boiling water reactor (BWR) enrichments typical of BWR fuel assemblies confirm that the uniform enrichment case yields the higher criticality for the same average enrichment and is therefore the limiting case for criticality safety evaluations.

The pool water temperature was conservatively taken to be approximately 39°F. Increasing temperature was shown to decrease reactivity. With the calculational bias and all uncertainties added, the reactivity (k_{∞}) of the storage racks will always be less than 0.947 with 95 percent probability at a 95 percent confidence level.

2.1.3 Accident Analysis - The effects of water density (temperature), positioning fuel assemblies outside of the storage rack, mispositioning fuel assemblies in the storage rack, fuel channel distortions, dropped fuel assembly (reactivity effect) and lateral movement of fuel racks were considered with acceptable results.

2.1.4 Technical Specifications - The Technical Specifications for Section 5.3.1 proposed by the licensee specify the maximum average planar enrichment of 3.01 weight percent U-235, and the maximum number of spent fuel assemblies (2600) to be stored in the pool. These are in conformance with the analysis and are therefore acceptable.

2.1.5 Conclusions - Based on the review, the staff concludes that the storage racks meet the requirements of General Design Criterion 62 as regards criticality. Also, the staff concludes that 2600 fuel assemblies of maximum average planar enrichment of 3.01 weight percent U-235 may be stored in the poisoned high density racks in the fuel pool. These conclusions are based on the following considerations:

1. Calculational methods which have been verified by comparison with experiment have been used.

2. Conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions.

3. Credible accidents have been considered.

4. Suitable uncertainties have been considered in arriving at the final value of the multiplication factor.

5. The final effective multiplication factor value meets our acceptance criterion of less than or equal to 0.95.

2.2 Spent Fuel Pool Cooling and Makeup

The increase in the total decay heat load resulting from the expansion will amount to only a few percent of the total heat load due to the longer decay times of the oldest fuel assemblies. The licensee therefore concluded that the existing spent fuel cooling capability could adequately remove the additional decay heat without exceeding the pool water temperature presented in Standard Review Plan (SRP) Section 9.1.3 (NUREG-0800). Information was also provided to demonstrate that the available source of makeup water provides adequate assurance that the fuel would not become uncovered in the event all pool cooling was lost.

2.2.1 Decay Heat Loads - The Oyster Creek reactor is rated at 1930 MWT and contains 560 fuel assemblies. Based on information contained in submittals made during the first pool expansion review, it appears that the licensee's current calculated maximum normal and maximum abnormal decay heat loads were calculated in a similar manner to the earlier values.

The maximum abnormal heat load (full core offload plus the pool full from successive normal refueling discharges) 10 days after shutdown is stated by the licensee to be 17.845×10^6 BTU/hr and that an additional 125 days of decay would be required before the heat load would be less than the capacity of the original SFP cooling loop (5.5×10^6 BTU/hr, refer to Section 2.2.2 of the Safety Evaluation (SE)). Similarly, the maximum normal heat load (pool full from successive normal refueling discharges) 10 days after shutdown is stated by the licensee to be 6.392×10^6 BTU/hr and that between 15 and 20 additional days of decay would be required before the heat load would be less than the capacity of the original cooling loop.

Using the licensee's current information and conservative assumptions regarding the discharge history of the previously discharged fuel assemblies, the staff independently calculated the maximum abnormal heat loads in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." The staff's maximum abnormal heat load 10 days after shutdown was calculated to be 19×10^6 BTU/hr. With this value, an additional 187 days of decay would be required before the decay heat load would be within the capacity of the original cooling loop. The staff's maximum normal heat load 10 days after shutdown was calculated to be 8.5×10^6 BTU/hr. With this value, an additional 45 days of decay would be required before the decay heat load would be within the capacity of the original cooling loop.

The differences in the staff's calculated heat loads compared to the licensee's are not significant since they do not exceed the capacity of the added cooling loop (19×10^6 BTU/hr, described in Section 2.2.2 of this SE). The difference in the additional decay times, before the heat loads are equal to or less than the original cooling loop capacity, is also not significant since the added cooling loop can be reactivated to maintain the pool water temperature below the Technical Specification limit of 125°F should the original cooling loop not be capable of accomplishing this. Discussion of the effects of possible thermal cycling on the fuel pool structure is provided in Section 2.4 of the SE, Appendix A.

Based on the above, the staff concludes that the maximum normal and abnormal heat loads are within the capacity of the SFP cooling system and are, therefore, acceptable.

2.2.2 Spent Fuel Pool Cooling System - The Oyster Creek SFP cooling system as initially licensed is shown on Figure X-3-2 of the Facility Description and Safety Analysis Report (FDSAR). It consisted of one cooling loop containing two parallel trains each with a pump and heat exchanger. This loop was rated at 5.5×10^6 BTU/hr with the pool water bulk temperature at 125°F, the Reactor Building Closed Cooling Water inlet temperature at 90°F, assuming 10 percent of the heat exchanger tubes are plugged and assuming no fouling. Additionally, a pool water temperature limitation of 125°F has been imposed in the Technical Specifications due to structural considerations of the pool.

During the first pool expansion (1977), the licensee committed to the installation of an additional cooling train in parallel with the above described parallel cooling trains. The new cooling train consists of two parallel full capacity pumps in series with one heat exchanger. This train is rated at $19 \pm 1 \times 10^6$ BTU/hr when the pool water temperature is at 125°F and is designed to withstand a safe shutdown earthquake (SSE) and the loss-of-offsite power coincident with a single active component failure. The installation of this additional pool cooling train was made in lieu of a previously proposed modification to cross connect the SFP cooling system to the shutdown cooling system train A heat exchanger. The proposed cross connect scheme would have only resulted in doubling the capacity of the SFP cooling system while the cooling capacity of the new additional cooling train would be approximately that of the maximum abnormal heat load (following a full core offload) without assistance of the existing cooling system.

The licensee indicates that it is anticipated that the new cooling train will be operated a limited period of time, i.e., only when pool heat load exceeds 5.5×10^6 BTU/hr due to a recent discharge. In addition, prior to placing the new cooling train in operation, surveillance will be performed to verify that its performance is satisfactory.

The licensee calculated the length of time the new cooling train would be required to be in operation. This is the period before the total decay heat load is reduced to 5.5×10^6 BTU/hr (the capacity of the original cooling loop) for both the maximum normal and abnormal heat loads. The results indicate that for maximum abnormal decay heat load, operation of the new cooling train would be required for 125 days after shutdown. In the case of the maximum abnormal heat load, the licensee calculated the new cooling train would be required to operate for between 15 and 20 days. From the staff's calculated maximum abnormal and normal decay heat loads, it has been determined that the length of time operation of the new cooling train would be required is 187 and 45 days, respectively, before the total heat load in the pool would decay to 5.5×10^6 BTU/hr., i.e., the rated capacity of the originally licensed SFP cooling system capacity.

The staff also notes that in previous submittals, the licensee stated that with proper valve line-up it was possible to obtain 8.9×10^6 BTU/hr of pool cooling by recirculating 500 gpm of fuel pool water through one main condenser. In its evaluation of the current SFP expansion the staff did not consider this method of cooling because there was insufficient information presented to perform an evaluation. Further, the licensee did not take credit for this method of cooling.

The licensee has provided the results of analysis of the potential for local boiling in the SFP. The results indicate that the exit water temperature from the most choked flow storage cell containing fuel with only 7 days decay following shutdown would be 173.4°F. The corresponding saturation temperature at the top of the storage racks would be 240°F. Therefore, the margin between local boiling and maximum water exit temperature is 66.6°F. The staff concludes from this that there is reasonable assurance that local boiling would not occur.

Based on the above, the staff concludes that the existing SFP cooling system provides sufficient decay heat removal capability to assure safe storage of spent fuel in the proposed expanded pool and is, therefore, acceptable.

2.2.3 Boiloff Rate - Assuming all fuel pool cooling is lost with the maximum abnormal heat load in the pool and a pool water temperature of 90°F, the licensee calculated it would take 14.5 hours for the pool water temperature to reach the boiling temperature. At this time the boiloff rate would be 41.2 gpm. Further, the licensee calculated that boiling would have to continue for 83.5 hours before the top of the storage racks would begin to be uncovered.

The staff performed a similar boiloff calculation but assumed that the pool water temperature is initially at 125°F. This assumption is made because the abnormal heat load closely approximates the rated capacity of the added cooling train when the pool water temperature is 125°F. Further, the indicated time to discharge the core would allow the pool water to rise to this temperature. The staff calculates that the pool would reach boiling in 11.4 hours with a boiloff rate of 39.4 gpm and the boiling time required before the top of the storage racks would begin to be uncovered is 77 hours. From the above, the staff concludes that there is adequate time to provide SFP makeup and maintain an acceptable pool water level in the unlikely event of loss of SFP cooling capability.

2.2.4 Makeup Water - The licensee states that there are three different sources of makeup water for the SFP. The normal source of makeup water is the 5.25×10^5 gallon condensate storage tank. The makeup rate is 250 gpm when using either one of the two condensate transfer pumps. Makeup water can also be provided at the rate of 150 gpm from the 3×10^4 gallon demineralized water storage tank using the demineralized water transfer pump and hose connections in the pool area. The third source of makeup water are the two skimmer surge tanks. These tanks normally contain about 3500 gallons. Using the SFP cooling pumps, a makeup rate of 100 gpm is possible.

A pool water level monitoring system has been provided. It will alarm in the control room and give local indication whenever the water level deviates from a nominal elevation of 118'-1 1/2" by more than 2 1/4". Therefore, the staff concludes that the operator will be adequately informed should makeup water be needed. The makeup rate from the above sources exceeds the maximum boiloff rate indicated previously in Section 2.2.3 of this evaluation. Based on the above, and on the staff's previous Systematic Evaluation Program (SEP), the staff concludes that the condensate and demineralized makeup water systems provide acceptable sources of fuel pool makeup water.

2.2.5 Conclusion - The staff has reviewed the proposed second SFP expansion program for Oyster Creek and concludes the following:

- The design of the previously added fuel pool cooling train is adequate for removal of the maximum abnormal heat load and it is capable of withstanding a single active component failure.

- The licensee has demonstrated that there is reasonable assurance that local pool boiling will not occur.

- The capability of the described makeup water systems have sufficient inventory and are in excess of the maximum boiloff rate and thus provide assurance that stored spent fuel will not become uncovered.

- The calculated time to reach boiling assuming loss of pool cooling is sufficient to establish pool makeup and maintain an adequate pool water level.

- The pool water level monitoring system provides reasonable assurance that the operator will be alerted to take action should the pool water level drop.

In summary, based on this review, the staff concludes that the Oyster Creek proposed SFP expansion meets the guidelines of SRP Sections 9.1.2, 9.1.3, 9.1.4, and 9.1.5, and is therefore, acceptable.

2.3 Rack Installation and Load Handling

The steps and procedures required to accomplish reracking the SFP will be developed so as to eliminate the need for carrying loads over stored spent fuel and will ensure that reasonable protective measures will be taken to preclude load drops during reracking.

2.3.1 New Storage Rack Design - The licensee proposes to replace the five existing spent fuel storage racks with ten free standing, fixed poison high density storage racks that have been designed and fabricated by the Joseph Oat Corporation. This will increase the number of storage cells from 1800 to 2600. These storage racks, except for the four support spindles, will be fabricated from type 304 stainless steel sheet, plate and forgings and sheets of Boraflex fixed poison. Boraflex is a patented product consisting of a dispersion of B_{10} enriched boron carbide in a silicon polymer. The support spindles are fabricated from SA564-Alloy 630.

The storage cells in the storage racks are assembled from preformed stainless steel sheets to form a series of double wall square storage cells. During the assembly, strips of Boraflex sheet are sandwiched between the double walls. The nominal interior dimension of the storage cells is 6 inches and the nominal center distance between storage cells in 6.198 inches. Therefore, the storage cells will accommodate the fuel channels which have a nominal outside dimension of 5.438 x 5.438 inches.

The storage capacity of the new racks will range between 176 to 320 fuel assemblies and their weight will range from 18 000 pounds to 38 400 pounds. The bottom end of the assembled storage cells will be welded to a 5/8 inch thick stainless steel base plate which has coolant flow holes in it on the same lattice spacing as the storage cells. The storage rack base plate is supported above the pool floor by four support legs. This forms a lower plenum to permit coolant to flow laterally over the pool floor and to enter the bottom of the storage cells. The vertical dimension of the support legs on eight of the storage racks is 6 inches. The height of the support legs on the two remaining storage racks is 11 1/2 inches.

The new storage racks will be designed, constructed, and assembled in accordance with ANSI N210-1976 (ANS 57.2), ASME Section III, Subsection NF, ASTM A240, ASME Section II parts A and C and ASME Section IX. The storage racks will be seismic Category I as identified in Paragraph 6.4 of ANSI N210-1976, and in the criteria of SRP Section 9.1.2. The nominal and maximum gap between storage racks is 1 1/2 inches and 4 inches, respectively, which assures that a fuel assembly cannot be inadvertently inserted into a nondesignated space within the storage rack array.

The licensee stated, in their reponse to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," that the refueling platform auxiliary hoists have been derated from their current rating of 1000 pounds to 750 pounds. Considering that the new free-standing storage racks weigh a minimum of 18 000 pounds, the staff concludes that the maximum uplift force developed by the refueling platform auxiliary hoist cannot cause damage to the storage racks or the pool liner.

The licensee also analyzed a vertical and horizontal dropped fuel assembly event. The results indicate that for two vertical assemblies separated by water that the reactivity (k_{∞}) will be less than 0.90 for any water gap spacing greater than 2.5 inches. For a dropped assembly lying horizontally on top of the rack, the separation distance is about 14 inches and will not constitute a criticality hazard.

Based on its review, the staff concludes that the new storage racks will adequately support and protect the spent fuel assemblies during normal and accident conditions, and are therefore acceptable.

2.3.2 Load Handling - There will be a total of 980 stored spent fuel assemblies in the pool when the reracking operations take place. To provide assurance that unacceptable consequences will not occur as a result of the reracking operations, the licensee states that procedures will be prepared which will include organization and administrative responsibilities as well as the detailed work practice. Each step will require multiple signatures before proceeding to the next step.

The reracking operations consist of removing the stored spent fuel from the rack to be removed and placing it outside of the area of influence of the load handling operations before the removal of the old storage rack and the installation of the new storage rack. This series of steps will be repeated for each rack being removed or inserted. Precautions will be taken to prevent the movement of fuel racks over other fuel racks containing stored spent fuel. Appropriately designed equipment will be utilized during the racking operation. The special handling equipment for the new storage racks will be designed and constructed in accordance with ANSI N14.6-1978. The lifting device employed in removing the old storage racks will be qualified by load testing at twice the maximum load being lifted. All slings utilized in the installation and removal of storage racks will be qualified to the requirements of ANSI B30.9-1971. The loads will be handled by the Reactor Building Crane which was designed in accordance with EOCI-61. The staff's heavy loads handling review (NUREG-0612) concluded that EOCI-61 substantially complies with the criteria specified in Guideline 5.1.1 (7).

From the above, the staff concludes that reasonable measures will be taken to prevent damage to the stored spent fuel during reracking operations and thus the potential for offsite radiological release will be minimized.

2.3.2 Conclusion - The described seismic Category I spent fuel storage racks will safely support and protect the stored spent fuel assemblies because:

- The arrangement of the storage racks within the pool is such that it is not possible to inadvertently insert a fuel assembly into a nondesignated space within the storage rack array.

- The maximum uplift force of the refueling platform auxiliary hoists is not sufficient to cause damage to the free-standing storage racks or the pool liner.

- The dropping of a fuel assembly will not lead to an unacceptable criticality accident.

The described reracking operations provide reasonable assurance that dropping of a storage rack will not occur, and in the unlikely event a rack drop should occur, the consequences will be acceptable.

In summary, based on its review, the staff concludes that the Oyster Creek proposed SFP expansion meets the guidelines of SRP Sections 9.1.2, 9.1.3, 9.1.4, and 9.1.5, and is therefore, acceptable.

2.4 Structural Design

The Safety Evaluation (SE) of structural aspects of the proposed modification is based on a review performed by NRC's consultant, Franklin Research Center (FRC). The FRC Technical Evaluation Report (TER) C5506-525 revised August 15, 1984 is appended to this SER as Appendix A.

2.4.1 Description of the Spent Fuel Pool and Racks - The pool is a reinforced concrete structure which is approximately 20'-0" by 39'-0". Wall thicknesses are 6'-0" on three sides and the fourth side is shared with the reactor building wall. The floor is supported by girders and walls. The pool is lined with a welded stainless steel watertight liner plate.

The new racks are stainless steel "egg-crate" structures. The fuel assembly storage cells are supported on a heavy welded base. The racks are each free-standing on the pool floor.

2.4.2 Applicable Codes, Standards, and Specifications - Load combinations and acceptance criteria were compared with those found in the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979. The existing concrete pool structure was evaluated for the new loads in accordance with the requirements of the American Concrete Institute Code, ACI 349.

2.4.3 Loads and Load Combinations - Loads and load combinations for the racks and the pool structure were reviewed and found to be in agreement with the applicable portions of the NRC Position.

2.4.4 Seismic and Impact Loads - Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. The seismic loads were applied to the model in three orthogonal directions simultaneously. Damping values for the seismic analysis of the racks and the pool structure were taken as 2 percent for OBE and 4 percent for SSE. Rack/fuel bundle interactions were considered in the structural analysis.

Loads due to a fuel bundle drop accident were considered in a separate analysis for such an occurrence. The postulated loads from these events described above were found to be acceptable.

2.4.5 Design and Analysis Procedures

a. Design and Analysis of the Racks - A non-linear 3-dimensional time-history analysis of the rack module was performed. The model included mass, spring, damping, and gap elements and accounts for sliding, tipping and potential rack-to-rack interaction. A detailed finite-element model of the racks was also constructed in order to determine stresses and strains within the racks. Partial as well as fully loaded racks were analyzed with a range of sliding friction coefficients between 0.8 and 0.2.

Calculated stresses for the racks components were found to be well within allowable limit. The racks were found to have adequate margins against sliding and tipping.

An analysis was conducted to assess the potential effects of a dropped fuel bundle on the racks and results were considered satisfactory.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension in the fuel assembly. Resulting stresses were found to be within acceptance limits.

b. Analysis of the Pool Structure - The Oyster Creek fuel pool is a reinforced concrete structure. The floor is essentially a plate structure and is supported by concrete walls and girders. The licensee performed both static and dynamic analysis and found that moments and shear of the pool floor and supporting girders and walls are lower than the code allowable value by factor ranging from approximately 1.5 to 3.0.

2.4.6 Conclusions - The staff concludes that the proposed rack installation will satisfy the requirements of 10 CFR 50 Appendix A, GDC 2, 4, 61 and 62, as applicable to structures, and is therefore acceptable.

2.5 Materials

The staff has reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water, in accordance with SRP Section 9.1.2 and "Review and Acceptance of Spent Fuel Storage and Handling Application, April 1978."

The spent fuel racks will be constructed of type 304-L stainless steel, except for the nuclear poison material. The spent fuel pool liner is constructed of stainless steel. The high density spent fuel storage racks utilize Boraflex sheets as a neutron absorber. Boraflex consists of boron carbide powder in a rubber-like silicone polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The major components of the assembly are the fuel assembly cells, the Boraflex material, the wrapper and the upper and lower spacer assemblies.

Boraflex neutron absorber surrounds each cell on all four sides, sandwiched in between an inner and outer angular subelement. The design ensures coverage of the active length of each fuel assembly, except for approximately 2 inches at each end. Venting is provided through the roof openings of the storage cell compartment corners to prevent gas entrapment. Stainless steel spacer straps hold the Boraflex in position.

2.5.1 Evaluation - The pool liner, rack lattice structure, and fuel storage tubes are stainless steel which is compatible with the storage pool environment.

The corrosion rate of type 304-L stainless steel in this water is sufficiently low to defy our ability to measure it. No instances of corrosion of this material in SFPs containing pure water have been observed (Ref. 1).

Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex is composed of non-metallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material (Ref. 2). The evaluation tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex to 1.03×10^{11} rads of gamma radiation with substantial concurrent neutron flux of borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex. Long-term borated water soak tests at high temperatures were also conducted (Ref. 3). The tests show that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space which contains the Boraflex is vented to the pool at each storage tube assembly. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the inner stainless steel tube.

The tests (Ref. 4) have shown that neither irradiation, environment, nor Boraflex composition has a discernable effect on the neutron transmission of the Boraflex material. The tests also show that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally in the Boraflex will typically contain 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble specie from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long-term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically.

2.5.2 Conclusion - From the evaluation as discussed above, the staff concludes that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life.

The staff further concludes that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate based on the test data cited above and actual service experience in operating reactors.

The staff has reviewed the surveillance program and concludes that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. The material surveillance program spelled out by the licensee will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power during the life of the spent fuel racks. The staff does not anticipate that such deterioration will occur. This monitoring program will ensure that, in the unlikely situation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

The staff therefore finds the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison and are, therefore, acceptable.

2.5.3 References

1. J.R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-23021, July 1977.

2. J.S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, July 1979.

3. J.S. Anderson, "A Final Report of the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Brand Industries, Inc., Report 748-21-1, August 1978.

4. J.S. Anderson, "Boraflex Neutron Shielding Material--Product Performance Data," Brand Industries, Inc., Report 748-30-1, August 1979.

2.6 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the removal and disposal of the low density racks and installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to result in approximately 25 person-rem. 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 is being performed.

2.6.1 Evaluation - 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. One potential source of radiation is radioactive activation or corrosion products called crud. Crud may be released to the pool water because of fuel movements during the proposed SFP modification. This could increase radiation levels in the vicinity of the pool. During refuelings, when the spent fuel is first moved into the fuel pool, the addition of crud to the pool water from the fuel assembly and from the introduction of primary coolant to the pool water is greatest. However, the licensee does not expect to have significant releases of crud to the pool water during modification of the pool. The purification system for the pool, which has kept radiation levels in the vicinity of the pool to low levels, includes a filter to remove crud and will be operating during the modification of the pool.

The racks will be individually lifted from the pool water and decontaminated by "hydrolasing" (a high pressure water spray technique) to remove any loose radioactivity prior to movements to a receiving area for preparation for disposal. The decontaminated old racks will be shipped for burial or the bulk of the decontaminated racks could be disposed of as clean scrap. In any event, the disposal methodology will follow as low as reasonably achievable (ALARA) guidelines.

Divers will not be used. The new racks will be handled and installed using remote handling devices.

2.6.2 Conclusion - Based on the manner in which the licensee will perform this modification, the radiation protection program, including area and airborne radioactivity monitoring, and relevant experience from other operating reactors that have performed similar SFP modifications, the s ff concludes that the Oyster Creek SFP modification can be performed in a manner that will limit exposures to workers to ALARA levels.

The staff has estimated the increment in onsite occupational dose during normal operations after the pool modification resulting from the proposed increase in stored fuel assemblies. This estimate is based on information supplied by the licensee for occupancy times and for dose rates in the spent fuel 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. Based on present and projected operations in the SFP area, the staff estimates that the additional spent fuel should add less than 0.1 percent increase to personnel occupational radiation exposure in the vicinity of the pool. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to ALARA levels and within the limits of 10 CFR Part 20. Thus, the staff concludes that storing additional fuel in the pool will not result in any significant increase in doses received by workers.

2.7 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. The waste treatment systems were evaluated in the SER in support of the issuance of the Operating License in 1969 and in supplements thereto. There will be no change in the waste treatment systems or in the conclusions given regarding the evaluation of these systems because of the proposed modification. The staff's evaluation of the radiological considerations supports the conclusion that the proposed modification to the Oyster Creek Nuclear Generating Station SFP is acceptable because the conclusions in the evaluation of the waste treatment systems, as found in the SER supporting the issuance of the operating license are unchanged by the modification of the SFP.

2.8 Radiological Consequences of Accidents Involving Postulated Mechanical Damage to Spent Fuel

2.8.1 Cask Drop Accidents - In an SER on an earlier SFP expansion dated October 27, 1976, the staff concluded that the spent fuel cask travel will be limited to the specified travel path and that the licensee's cask drop protection was adequate for the prevention of cask tip accidents. Since that SER, the licensee has added a technical specification (T.S.5.3.1(d)) which prohibits the movement of loads greater than the weight of one fuel assembly over irradiated fuel in the fuel pool. Based upon the information presented above, the staff concludes that the likelihood of a cask drop onto irradiated fuel is sufficiently small that the offsite radiological consequences for such an accident need not be considered.

2.8.2 Spent Fuel Pool Gate Drop Accidents - In a submittal on the control of heavy loads (Phase I), the licensee stated that lifting procedure 756.1.004 which establishes the "safe paths" for moving the fuel pool gates would be used for the removal and installation of the SFP gates. The staff concluded (SE dated June 21, 1983) that this procedure met the requirements of Guideline 2, Sections 5.1.1 of NUREG-0612. With the use of this procedure, coupled with the current plant technical specification 5.3.1(d), the staff concludes that the likelihood of a fuel pool gate drop onto irradiated fuel is sufficiently small that the offsite radiological consequences of such an accident need not be considered.

2.8.3 Fuel Handling Accidents - The licensee has proposed to expand the storage capacity of the SFP from 1800 spent fuel assemblies to 2600 assemblies. During the action, the maximum weight of loads which may be transported over spent fuel in the pool will be limited to that of a single assembly by plant technical specification 5.3.1(d). Because this accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitations on available impact kinetic energy, the proposed SFP modification does not, therefore, increase radiological consequences of fuel handling accidents above that considered in the staff Safety Evaluation contained in the Oyster Creek SEP TOPIC XV-20, Radiological Consequences of Fuel Handling Accidents, May 1982.

2.8.4 Conclusion - Based upon the above evaluations, the staff concludes that the likelihood of either a cask drop or a fuel pool gate drop onto irradiated fuel is sufficiently small that the offsite radiological consequences for these accidents need not be calculated. Additionally, the offsite radiological consequences from a postulated fuel handling accident would remain unchanged from that which was reported in the staff SE referenced in Section 2.8.2 of this evaluation. The staff's present analysis indicates a 0-2 hr Exclusion Area Boundary (EAB) thyroid dose of 0.6 rem and whole body dose of 0.3 rem given an atmospheric transport and diffusion Relative Concentration value of 7.6×10^{-4} sec/m^3. These conservatively estimated doses are well within the 10 CFR Part 100 guideline values. Therefore, the staff concludes that the proposed modifications are acceptable.

3.0 OVERALL CONCLUSION

Based on the review, the staff concludes that the licensee's proposed SFP modification to increase the storage capacity of the SFP to 2600 fuel assemblies is acceptable. In addition, the proposed Technical Specifications and license conditions are acceptable.

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

4.0 ACKNOWLEDGEMENT

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