September 15, 2000

Mr. Ronald DeGregorio
Vice President Oyster Creek
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, NJ 08731

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: INCREASED SPENT FUEL POOL CAPACITY (TAC NO. MA5965)

Dear Mr. DeGregorio:

The Commission has issued the enclosed Amendment No. 215 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated June 18, 1999 , as supplemented on June 22 and December 10, 1999, and February 10, and May 2, 2000.

On the date of the June 18, 1999, application, GPU Nuclear, Inc. (GPUN) was the licensed operator for Oyster Creek. On August 8, 2000, GPUN's ownership interest in Oyster Creek was transferred to AmerGen Energy Company, LLC (AmerGen). By letter dated August 10, 2000, AmerGen requested that the Nuclear Regulatory Commission continue to review and act upon all requests before the Commission which had been submitted by GPUN. Accordingly, the staff has completed its review of the requested amendment.

The amendment revises the Technical Specifications (TSs) to reflect the installation of additional spent fuel pool storage racks. The additional new racks will provide 390 additional spent fuel assembly storage locations.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Helen N. Pastis, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 215 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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On the date of the June 18, 1999, application, GPU Nuclear, Inc. (GPUN) was the licensed operator for Oyster Creek. On August 8, 2000, GPUN's ownership interest in Oyster Creek was transferred to AmerGen Energy Company, LLC (AmerGen). By letter dated August 10, 2000, AmerGen requested that the Nuclear Regulatory Commission continue to review and act upon all requests before the Commission which had been submitted by GPUN. Accordingly, the staff has completed its review of the requested amendment.

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AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 215
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear, Inc. et al., (the licensee), dated June 18, 1999 , as supplemented on June 22 and December 10, 1999, and February 10, and May 2, 2000, as adopted by AmerGen Energy Company, LLC, pursuant to a letter dated August 10, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 215 , are hereby incorporated in the license. AmerGen Energy Company, LLC Nuclear, Inc. shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 215

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.3-1	5.3-1
5.3-2	5.3-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 215

TO FACILITY OPERATING LICENSE NO. DPR-16

AMERGEN ENERGY COMPANY, LLC

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated June 18, 1999, as supplemented on June 22, and December 10, 1999, and February 10, and May 2, 2000, the GPU Nuclear, Inc. (the licensee) submitted a request for revisions to the Oyster Creek Nuclear Generating Station Technical Specifications (TSs).

On the date of the June 18, 1999, application, GPU Nuclear, Inc. (GPUN) was the licensed operator for Oyster Creek. On August 8, 2000, GPUN's ownership interest in Oyster Creek was transferred to AmerGen Energy Company, LLC (AmerGen). By letter dated August 10, 2000, AmerGen requested that the Nuclear Regulatory Commission continue to review and act upon all requests before the Commission which had been submitted by GPUN. Accordingly, the staff has completed its review of the requested amendment.

The amendment would revise the TSs to reflect the installation of additional spent fuel pool (SFP) storage racks. The additional new racks would provide 390 additional spent fuel assembly storage locations.

The additional information provided in letters June 22, December 10, 1999, February 10, and May 2, 2000, did not affect the staff's proposed finding of no significant hazards consideration, and was within the scope of the amendment application as noticed.

2.0 EVALUATION

The current TSs allow 2645 spent fuel assemblies to be stored in the SFP. The licensee is proposing to install four additional high density spent fuel storage racks. These new racks would provide an additional 390 spent fuel assemble storage locations and an ultimate storage capacity of 3,035 fuel assemblies in the Oyster Creek SFP. The new storage capacity would restore temporarily the full-core discharge capability at the plant. The licensee plans to install the new racks in existing available SFP floor space.

The licensee also proposed to revise TS Section 5.3 Basis to clarify that the existing reference to containment air temperature applies to the reactor building because this is the location of the fuel pool. The licensee considers this proposed revision to be editorial in nature.

The licensee also proposed to revise the Oyster Creek TS Section 5.3 Basis to include a reference to the Holtec Licensing Report HI-981983, "Licensing Report for Storage Capacity Expansion of Oyster Creek SFP," Revision 4, dated June 15, 1999, which provides the design basis and safety analysis for installation and use of the new high density SFP storage racks.

The NRC reviewed the licensee's application under seven areas: (1) criticality, (2) thermal-hydraulics, (3) structural integrity, (4) heavy loads, (5) materials, (6) radiological effects, and (7) off-site dose analysis. Each of these seven areas is discussed in detail below.

2.1 Criticality

The new high density racks are designed by Holtec International. The Holtec racks consist of an egg-crate structure with 0.070-inch thick fixed neutron absorber material (Boral) of 0.0162 g/cm^2 boron-10 areal density positioned between the fuel assembly storage cells in a 0.077-inch channel. The 0.075-inch thick stainless steel boxes have a nominal 5.93-inch inside opening. This provides a nominal center-to-center lattice spacing of 6.106 inches. The Oyster Creek spent fuel storage racks are presently designed and maintained with a k-eff of less than or equal to 0.95, including all calculational uncertainties.

The analysis of the reactivity effects of fuel storage in the Oyster Creek racks was performed with the CASMO4 two-dimensional transport theory code. Independent check calculations were made with the continuous energy MCNP Monte Carlo code and the KENO5a three-dimensional Monte Carlo code package using the 238-group SCALE cross-section library. CASMO4 was used to determine the peak reactivity over burnup and to evaluate small reactivity increments associated with manufacturing tolerances. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Oyster Creek spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, the two independent methods of analysis (MCNP and KENO5a) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Oyster Creek storage racks with a high degree of confidence.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include the following:

(1) Racks contain most reactive fuel authorized to be stored at Oyster Creek without any control rods or any uncontained burnable absorber and with the fuel at the burnup corresponding to the highest planar reactivity during its burnup history.

(2) Unborated pool water at the temperature yielding the highest reactivity ($4\,^{o}C$) over the expected range of water temperatures.

(3) Assumption of infinite array (no neutron leakage) of storage cells in all directions except for assessment of certain abnormal/accident conditions where neutron leakage is inherent.

(4) Neutron absorption in minor structural material is neglected (i.e., spacer grids are analytically replaced by water).

The staff concludes that appropriately conservative assumptions were made.
The following General Electric Company fuel assembly types were used for the criticality analyses assuming a uniform initial enrichment of 4.6 weight percent (w/o) U-235:

(1) GE 8x8R (8x8 rod array with two water rods replacing two fuel rods)

(2) GE-9B (8x8 rod array with large water rod replacing four fuel rods)

(3) GE-11 (9x9 rod array with two large water holes replacing seven fuel rods)

The design basis reactivity calculations accounted for uncertainties from manufacturing tolerances, flow channel bulging, and fuel enrichment and density. Also, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations.

In boiling-water reactor (BWR) fuel, there is a need for distributed enrichments to avoid power peaking problems, and, because 5.0 w/o is the maximum enrichment allowed for any single fuel rod, it is not likely that a BWR assembly will exceed an average enrichment of about 4.6 w/o U-235. Therefore, calculations were made for the fuel designs at Oyster Creek assuming an average enrichment of 4.6 w/o U-235 in both the spent fuel storage rack configurations and the Oyster Creek core geometry (6.0-inch assembly pitch, 20 °C). The results indicate that any of the fuel types with an average initial enrichment of 4.6 w/o or less and a k-inf in the standard core geometry less than or equal to 1.32 would result in a rack k-eff of less than 0.95, including all appropriate uncertainties at a 95 percent probability, 95 percent confidence (95/95) level. This meets the staff’s criterion for SFP storage and is, therefore, acceptable.

Calculations for fuel initially enriched to less than 3.2 w/o U-235 show that the maximum k-eff in the storage rack would be less than 0.95, including all uncertainties regardless of gadolinia content and, thus, the k-inf in the standard cold core geometry need not be considered for fuel below this enrichment.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events due to temperature and water density effects, abnormal or eccentric fuel assembly positioning, and the drop of a fuel assembly on top of the storage rack which could lead to an increase in reactivity. However, such events were found to have a negligible effect and the resulting reactivity would remain below the 0.95 design basis for the Oyster Creek storage racks.

Based on the above evaluation, the proposed change to TS 5.3.1.E increasing the maximum number of spent fuel assemblies stored in the SFP to 3035 is acceptable. Based on the review described above, the staff finds the criticality aspects of the proposed expansion of the Oyster

Creek spent fuel storage racks is acceptable and meets the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

2.2 Thermal-Hydraulics

The Oyster Creek SFP has a current storage capacity of 2645 spent fuel assemblies.[1] As a result of Oyster Creek's current spent fuel limitations and the lack of an off-site spent fuel storage facility, Oyster Creek is currently operating in cycle 17 with insufficient storage capacity to offload the entire reactor core. The addition of the four new racks will temporarily restore their full core offload capability. No changes are currently planned to the existing racks in the SFP, and the new racks will fit into existing available floor space in the SFP. The new racks are high density racks manufactured by Holtec International that are constructed of type 304L stainless steel sheet. The racks are free standing, using Boral as a neutron absorber, and are seismically qualified.

In their June 18, 1999 letter, the licensee provided Holtec Report HI-981983, "Licensing Report for Storage Capacity Expansion of Oyster Creek SFP," dated June 15, 1999. This report provides the design basis and safety analysis supporting the installation and use of the new high density SFP storage racks at Oyster Creek. The report also provides supporting information needed by the NRC staff to perform its review in accordance with NRC guidance on SFP modifications which include:

1) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 and its subsequent addendum dated January 18, 1979.

2) NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition," dated July 1981.

The Holtec report includes thermal hydraulic analyses that are consistent with Oyster Creek's current licensing basis and the guidance of SRP Section 9.1.3, "SFP Cooling and Cleanup System." Specifically, Holtec calculated decay heat loads and maximum SPF pool temperatures for Oyster Creek's normal refueling offload (1/3 of the core) scenario with a single failure, as well as an abnormal full core offload (single failure not considered). Accordingly, the staff has reviewed the licensee's proposed amendment against the SRP acceptance criteria. However, it should be noted that originally, a full core offload was thought to be necessary only for a small number of occurrences. Examples of the need for a full core offload included periodic in-service inspection of the reactor vessel, an emergency offload due to a fuel failure or core damage event, and for removing the fuel from the reactor after its final operating cycle. Since a limited number of occurrences were anticipated, the full core offload was categorized in the SRP as an abnormal (unplanned) event with different acceptance criteria (e.g., single failure not considered) than for the normal or planned refueling scenario. In Oyster Creek's licensing basis, the normal scenario is subject to single failure criteria and accident conditions. If the

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[1] With 2420 fuel assemblies already stored in the SFP, Oyster Creek's current SFP rack configuration has room for 225 additional fuel assemblies before they run out of storage space.

licensee were to decide to routinely perform full core offloads during refueling outages, then this change would need to be evaluated under the requirements of 10 CFR 50.59 to determine if the NRC's review and approval is required prior to making this change.

2.2.1 Description of SFP and Cooling Systems

The SFP dimensions are 27 feet by 39 feet with a water depth of approximately 37 feet 9 inches. The depth of the water over the top of the stored fuel is approximately 25 feet which provides approximately 200,000 gallons of water above the fuel.

The SFP is designed to prevent any inadvertent draining of the pool water below the level of 1 foot above the top of the stored fuel. All lines below this level are equipped with valving to prevent backflow. Level monitoring switches are located in the SFP and in the SFP surge tank which will alarm in the control room should a loss of water level be detected. A low-low level switch in the SFP will automatically shut down the SFP cooling system (SFPCS) pumps if the low-low setpoint is reached. Low water level may also be indicated by an increase in radiation level around the SFP. Oyster Creek is equipped with radiation monitors on the operating floor near the SFP. Reactor building ventilation and the standby gas treatment systems are automatically initiated if high radiation levels are detected.

Make up water to the SFP is normally provided by the condensate system from the condensate storage tank (CST) which has a nominal capacity of 525,000 gallons. The condensate pumps can provide 250 gallons per minute (gpm) with one pump operating or 420 gpm with two pumps. Additional makeup can be provided from the demineralized water storage tank (nominal capacity 30,000 gallons) by connecting the demineralized water transfer pumps to the SFP with hoses. The fire protection system can also provide makeup from the fire pond to the CST using the 2,000 gpm diesel driven fire pumps through a permanent connection.

The SFPCS removes decay heat from fuel stored in the SFP through its associated heat exchangers to the reactor building closed cooling water (RBCCW) system. The SFP water is maintained within its TS limits by these systems. The SFPCS consists of two SFP pumps, two SFP shell and tube heat exchangers, two augmented fuel pool pumps, and one augmented fuel pool plate and frame heat exchanger. In addition, the SFPCS also includes interconnections with the condensate demineralizers and the condensate systems which filter and demineralize the SFP water as well as provide makeup water to the SFP. The SFPCS operates continuously to maintain the SFP water temperature at or below the Oyster Creek TS limit (maximum of 125 degrees Fahrenheit (°F)).

The storage of additional fuel assemblies in the SFP will raise the heat load to be cooled by the SFPCS. Holtec performed an analysis to verify that the existing SFPCS could maintain the SFP temperature within its required limits. The Holtec report includes thermal-hydraulic analyses that confirm that the existing SFPCS is adequate to demonstrate that, for planned (normal) conditions, the SFPCS can maintain the bulk pool water temperature within SFP limits required by the Oyster Creek TS.

When maintaining the SFP temperature at 125 °F, the cooling capacity of each fuel pool heat exchanger is 2.75 million British Thermal Units per hour (MBTUH) with an RBCCW flowrate of 500 gpm and the RBCCW heat exchanger water inlet temperature of 90 °F. The total SFP water flowrate through each heat exchanger is 500 gpm. Fuel pool water outlet temperatures are monitored for each heat exchanger and high temperatures are annunciated in the control room. A portion of these two cooling trains is seismic class I qualified. The portion that is not seismic class I can be isolated using a gate valve. As noted above, these pumps are operated continuously to maintain the SFP temperature at or below 125 °F. When maintaining the SFP temperature at 125 °F, the cooling capacity of the augmented heat exchanger is 12.2 MBTUH at 800 gpm fuel pool water flowrate, 2000 gpm RBCCW water flowrate, and an RBCCW water inlet temperature of 90 °F. Only one pump is needed to meet the augmented heat exchanger operational needs. The augmented portion of the SPFCS is seismic class I. The augmented systems are normally operated during refueling to handle the decay heat of the off loaded fuel. The system can be operated independently or in parallel with the spent fuel pump/heat exchangers. However, if SFP water cleanup capability is needed, the spent fuel pumps must operate to provide the cleanup capability.

2.2.2 Decay Heat Load

Oyster Creek's decay head load calculations were performed in accordance with the provisions of the NRC's Standard Review Plan, Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Revision 2, dated July 1981. With regards to the portions of the analyses related to SFP cooling, Holtec performed calculations for three different fuel discharge conditions. Specifically, they calculated for each discharge condition: 1) the decay heat load in the SFP, 2) the maximum bulk SFP water temperature, 3) the time after shutdown to reach the maximum SFP temperature, 4) the time to boil after a loss of SFP cooling capability, and 5) the rate of water inventory loss if the SFP were to boil following a loss of SFP cooling capability. These calculations were performed to ensure that the existing SFPCS is adequate to maintain the SFP temperatures within their current TS limits without the need for any modifications, and to ensure that there is adequate water makeup capability in the event of a complete loss of SFP cooling capability.

Holtec performed a conservative analysis to determine the bounding case for the maximum decay heat generation using the following assumptions:

- The fuel inventory was assumed to be 3168 fuel assemblies which is greater than the 3035 assembly storage capacity of the SFP with the addition of four new racks requested in this license amendment.
- The decay heat calculations assume 2,053 days of in-core irradiation at full power for the fuel assemblies being offloaded to conservatively bound the decay heat calculation.
- The SFP ambient air conditions were assumed to be 104 °F with 100% relative humidity. This assumption limits credit for evaporation heat losses.
- Rejection of heat to areas surrounding the SFP through the pool walls and floor are neglected. This results in a conservative calculation of the SFP temperature.
- All heat exchangers were assumed to be fouled to their design basis level. This assumption reduces the postulated heat exchanger effectiveness to its lowest postulated value.
- The temperature of the RBCCW water supply to the heat exchanger inlets is assumed to be 90 °F.

Three discharge scenarios were considered for the Oyster Creek bulk pool thermal-hydraulic evaluation:

- Case 1 - an abnormal[2] partial core (188 fuel assemblies) transferred to the SFP beginning 6 days after reactor shutdown at a rate of 5.8 fuel assemblies per hour. Decay heat removal is provided by one SFPCS pump and one SFPCS heat exchanger.
- Case 2 - a normal partial core (188 fuel assemblies) transferred to the SFP beginning 6 days after reactor shutdown at a rate of 5.8 fuel assemblies per hour. Decay heat removal is provided by the augmented heat exchanger with one pump in operation.
- Case 3 - a full core (560 fuel assemblies) transferred to the SFP beginning 31 days after reactor shutdown at a rate of 5.8 fuel assemblies per hour. Decay heat removal is provided by the augmented heat exchanger with one pump in operation.

SRP Section 9.1.3 states that "For the maximum normal heat load with normal cooling systems in operation, and assuming a single active failure, the temperature of the pool should be kept at or below 140 °F and the liquid level should be maintained." Oyster Creek's TS 125 °F limit is more conservative than the SRP requirement, and is, therefore, part of the basis used by the licensee to determine the acceptability of their normal discharge scenario. In addition, SRP Section 9.1.3 further states that "For the abnormal maximum heat load (full core unload) the temperature of the pool water should be kept below boiling and the liquid level maintained with normal systems in operation. A single active failure need not be considered for the abnormal case." This criterion was used by the licensee for the Oyster Creek unplanned (abnormal) discharge scenario.

2.2.3 Analysis Results

Based on the Holtec analysis, the licensee concludes in their June 18, 1999, letter that the "Thermal Hydraulic Analysis confirms that SFP bulk temperatures are kept below 125 °F, as required by existing [Oyster Creek Nuclear Generation Station] Technical Specification Section 5.3.1.D, during normal refueling offload (Case ii - normal refueling batch transferred to pool six days after reactor shutdown using augmented fuel pool heat exchanger with one pump in operation) and full-core offload (Case iii - full core transferred to pool thirty-one days after reactor shutdown using augmented fuel pool heat exchanger with one pump in operation) discharge scenarios."[3] To confirm this conclusion, the staff performed an independent set of calculations to verify the calculated maximum SFP decay heat loads and the maximum expected SFP temperature. For our calculations, the staff used the same conservative assumptions that were used in the Holtec analyses (described in Section 3.2 above). These assumptions are clearly conservative and would provide a bounding calculation of SFP decay heat loads and maximum temperatures. Table 1 summarizes the results of the Holtec analyses:

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2 This condition is calculated to demonstrate the worst-case SFP temperature. It is based on all SFP cooling being provided by one of the SFPCS pumps only without use of the augmented SFP cooling system.

3 The Holtec calculation assumes a single failure of one augmented fuel pump.

Table 1
Summary of Holtec's SFP Thermal Hydraulic Analyses Results

Case[4]	In-core Hold Time (hours) Before Move to SFP	Maximum SFP Temperature (°F)	Coincident Time[5] (Hours after Shutdown)	Coincident[6]/Max[7] Heat Exchanger Loads (MBTUH)	SFP Boiling[8]	
					Time to Boil (hours)	Boil-off Rate[9] (gpm)
1	144	168.35[10]	205	6.16/8.7	7.75	16.6
2	144	114.10	187	8.4/8.7	17.21	17.8
3	744	124.62	853	12.06/12.5	10.49	26.4

Our independent calculations conservatively estimated the maximum decay heat loads to be 8.1 MBTUH for Case 1, 8.1 MBTUH for Case 2, and 11.8 MBTUH for Case 3. As can be seen from the table above, Holtec's calculations are higher than our results; therefore, the staff concluded that Oyster Creek's analyses conservatively define the maximum decay heat loads for their SFP cooling system. In addition, the staff calculated maximum expected SFP bulk water temperatures of 166 °F for Case 1, 113 °F for Case 2, and 123 F for Case 3. These results are consistent with Holtec's analyses for Oyster Creek. This leads to our conclusion that Holtec's analyses conservatively predict the maximum SFP temperatures. Therefore, the staff agreed with the licensee's conclusion that, for both the planned (normal) partial core offload and unplanned full core offload refueling conditions, their existing SFPCS would be sufficient to maintain the SFP temperature.

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4 As described in Section 3.2 above.

5 The time after shutdown at which the SFP water reaches its calculated peak temperature.

6 The decay heat load at the time the SFP reaches its maximum temperature.

7 The maximum decay heat load estimated from Figures 5.8.4 and 5.8.5 in Holtec's report. This load is not coincident with the maximum SFP water temperature due to thermal inertia of the pool.

8 Resulting from the unlikely event of a complete loss of SFP cooling.

9 The boil-off rate values in this table are estimated from Figure 5.8.6 in Holtec's report.

10 This case demonstrates that the SFP water will not boil even with one SFPCS and both augmented fuel pool cooling pumps unavailable.

at or below 125 °F consistent with their existing Technical Specifications. Since their TS limit is more conservative than the acceptance criteria in SRP Section 9.1.3, the staff finds their decay heat load and SFP temperature analyses to be acceptable.

2.2.4 Effects of SFP Boiling

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise and eventually will reach the boiling temperature (if cooling is not restored). Holtec's analysis, taking the proposed rerack into account, shows that the minimum time from the loss-of-cooling at peak pool water temperature (168.35 °F) until the pool boils is 7.75 hours with a maximum boil-off rate of 17 gpm for the most severe scenario (partial core offload with one SFPCS pump and both augmented cooling system pumps unavailable). For the case of an unplanned (abnormal) full core offload and a complete loss of SFP cooling, it would take 10.49 hours until the pool boils with a maximum boil-off rate of 26 gpm. In both cases, the make up water from any of the available make up water sources would easily maintain water level in the pool. In the unlikely event that there is a complete loss of SFP cooling during planned (normal) refueling conditions, Holtec's analysis showed that there would be approximately 50 hours between the time the high SFP temperature alarm (120 °F setpoint) annunciates in the control room and the time that the pool temperature would reach the analyzed maximum (168.35 °F). It would then take an additional 7.75 hours for the pool to begin boiling. The licensee concludes in their June 18, 1999, letter that "sufficient time is available to respond to existing SFP water temperature alarms (120 °F) to restore pool cooling with either the shell-and-tube heat exchanger or the augmented heat exchanger." The staff concurs with the licensee's conclusion, and find this to be acceptable.

2.2.5 Reactor Building Heating Ventilation and Air Conditioning (HVAC)

Holtec evaluated the potential effects of the rerack on the Reactor Building's HVAC to make sure that the humidity burden on the HVAC system did not exceed its design basis. They noted that the original design basis of the SFP is 125 °F. The 125 °F design basis exceeds the predicted maximum temperatures calculated based on the four additional spent fuel racks. Therefore, they concluded that the design basis humidity burden on the HVAC system would not be exceeded with the proposed change. The staff agrees with their evaluation.

2.2.6 Conclusions

The staff has reviewed the June 18, 1999, amendment request from the licensee and conclude that the proposed addition of four new racks to the SFP to increase the storage capacity of the pool to 3035 fuel assemblies is acceptable. Our conclusion is based on the analyses submitted by the licensee which confirm that the SFP temperature will not exceed Oyster Creek's TS limit of 125 °F. In addition, for the worst-case single failure (loss of the augmented SFPCS), their analyses confirm that boiling would not occur in the SFP.

2.3 Structural Integrity

The staff evaluated the adequacy of the seismic and structural aspects of the SFP. The staff reviewed the procedures and the results of the structural analyses performed by the licensee to demonstrate the integrity of the free-standing spent fuel racks and the SFP structure under the postulated loads for normal, seismic, and accident conditions at Oyster Creek.

2.3.1 Storage Racks

The four new racks will use the new neutron absorbing material, Boral, whereas the existing ten (10) racks will continue to utilize Boraflex. SFP racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE) under all applicable loading conditions. The licensee's consultant, Holtec International, performed the design, fabrication, and safety analysis of the new high density SFP storage racks. All storage racks are made of Type 304L austenitic stainless steel. The overall design of the new racks at Oyster Creek is similar to Holtec racks that the NRC has approved for service at many other nuclear power plants such as FitzPatrick, Zion, and Duane Arnold (Reference1). All SFP racks are free-standing and self-supporting equipment, and are not anchored to the floor of the storage pool. The key design criteria of the Oyster Creek SFP racks are described in Section 2.2 of Reference 1. Briefly, the following criteria are applicable from the structural safety point of view: (1) all free-standing rack modules are required to be kinematically stable (against tipping or overturning) if a seismic event (which is 150% of the postulated operating basis earthquake or 110% of the postulated safe shutdown earthquake) is imposed on any module; (2) all primary stresses in the rack modules must satisfy the limits postulated in Section III, Subsection NF of the 1995 American Society of Mechanical Engineers Boiler and Pressure Vessel Code; (3) the spatial average bulk pool temperature is required to remain under 125 °F in the wake of normal refueling, and (4) the reinforced concrete structure of the SFP should be able to withstand the effects of the load combinations set forth in SRP Section 3.8.4.

At the time of the previous re-racking of the Oyster Creek SFP in 1983-84, the seismic evaluation of the racks was performed using single-rack (SR) three-dimensional (3-D) simulations. However, for the current SFP expansion, both SR and whole pool multi-rack (WPMR) analyses were performed to simulate the dynamic behavior of the high density rack structures (Reference 1). Holtec used a computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under the earthquake loading conditions. The DYNARACK program (which can perform simultaneous simulation of all racks in the pool for the WPMR analysis) has been accepted by the NRC in previous re-rack analyses for several nuclear power plants.

The DYNARACK program utilizes a nonlinear analytical model consisting of inertial mass elements, spring elements, gap elements and friction elements to simulate the three-dimensional dynamic behavior of the rack and the fuel assemblies including the frictional and hydrodynamic effects (Reference 1). In response to a staff question, the licensee explained that, instead of using a "stick" model to analyze the rack, the DYNARACK computer code utilizes the "component element method" (CEM) that can simulate the friction, impact, and other nonlinear dynamic events accurately (Reference 3). The code models the beam characteristics of the rack including shear, flexibility, and torsion effects appropriately, by modeling each rack as a 3-dimensional structure having the support pedestals and the fuel assemblies in proper locations. The potential rattling between the fuel and storage cells is simulated by permitting the impact at any of the four facing walls followed by rebound and impact at the opposite wall. Further, the rack pedestals can lift off, or slide, to satisfy the instantaneous dynamic equilibrium of the system throughout the seismic event. The rack structure can undergo overturning, bending, twist, and other dynamic motion modes as dictated by the interaction between the seismic inertia, impact, friction, and fluid coupling forces (Reference 3). The DYNARACK code calculates the nodal forces and displacements at the nodes, and then obtains the detailed stress field in the rack elements from the calculated nodal forces.

The lateral motion of the rack due to earthquake ground motion is resisted by the pedestal-to-pool slab interface friction, and is amplified or retarded by the fluid coupling forces produced by the close position of the rack to other structures. The seismic analyses of the racks were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time histories) was generated from the Oyster Creek reactor building (RB) response spectra at elevation 75' 3". The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying the requirements of both enveloping design response spectra and matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in the SRP Section 3.7.1 (References 1 and 3).

Using the results of the DYNARACK analysis, the licensee performed the structural evaluation of the spent fuel rack design, using the design criteria based on NRC’s former Office of Technology position paper dated April 14, 1978 (Reference 4), now incorporated in SRP Sections 3.8.4 and 3.8.5. These criteria include the maximum required safety factors against rack overturning of 1.5 and 1.1 during OBE and SSE events, respectively (Reference 1, Section 6.6.1). The licensee considered the applicable loads and their combinations in the seismic analysis of the rack modules, and performed parametric simulations for both the Single Rack and Whole Pool Multi Rack analyses. The parameters, which were varied in the different computer runs, consisted of the rack/pool interface coefficient of friction, the extent of storage locations occupied by spent fuel (ranging from nearly empty to full) and the type of seismic input (SSE or OBE). For the parametric simulations, the licensee performed a total of nineteen 3-D SR model analyses and two WPMR model analyses (Reference 1, Section 6.7). The results of these analyses (discussed in Section 6.8 of Reference 1) show the maximum rack displacement to be 0.469 inches (for SSE condition). The licensee then performed a rack overturning evaluation, and found the factor of safety against overturning to be 105 which is much higher than the prescribed limit of 1.1 for SSE condition. These results indicate that there are large safety margins against overturning of the racks as evidenced by the small rack movements and, thereby, the structural integrity and stability of the racks and fuel assemblies are maintained.

From the large number of computer runs of parametric evaluations, the licensee computed the maximum values of pedestal vertical forces, pedestal friction forces, pedestal thread shear stresses, displacements and stress factors (Reference 1, Section 6.8). Using these data, the licensee performed the rack impact evaluation, as well as the stress limit evaluation of the rack structure using the ASME Code, Section III, Subsection NF, for normal and upset conditions (Level A or Level B), and Section F-1334 (ASME Section III, Appendix F) for Level D condition. The calculated results show that there are no rack-to-wall impacts, and no rack-to-rack impacts at higher rack elevations under any conditions. However, there are some impacts between adjacent racks at the baseplate level, and some impacts between fuel assemblies and fuel cell walls (Reference 1, Section 6.8.4). The licensee evaluated the stresses imposed by the instantaneous impacts on the steel baseplate, and found that all stresses were well below the corresponding "NF" limits. In addition, the licensee calculated the weld stresses of the rack at the connections (e.g., baseplate-to-cell, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions, and demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code Section III, Subsection NF, indicating that the weld connection design of the rack is adequate.

Based on (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the large factor of safety of the induced stresses of the rack when compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, Section III, (3) a reasonable assurance that there is no rack-to-wall and rack-to-rack impacts (except some insignificant impact at the rack baseplate level as described above), and (4) the licensee's overall structural integrity conclusions supported by both SR analyses and WPMR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and are, therefore, acceptable.

2.3.2 Spent Fuel Storage Pool

The SFP, situated above ground, is located in the RB, between elevation 75' 3" and elevation 119' 3". The inside (plan) dimensions of the SFP pool structure are 27'-0" wide, 39'-0" long and 39'-1“ deep. The contents of the pool are supported by a two-way, reinforced concrete slab and the underlying beams. The minimum thickness of the pool slab (excluding the grout) is 54". The thickness of the reinforced concrete walls is generally 72", except that the thickness of the west wall is reduced to 54" above elevation 95' 3" (Reference 3). A fully-welded stainless steel liner covers the interior surface of the pool. The floor consists of an array of 1/4" thick plates, while the wall covering is made of 1/8" thick material. The liner anchorage in the floor consists of a series of 6" x 6" x 1/4" beams, which are embedded in the grout layer beneath the liner (Reference 3).

The licensee’s calculation indicates the maximum bulk pool temperature during a partial core discharge (Case (i) in Table 5.8.1 of Reference 1) will be 168.35 °F which exceeds the allowable value of 150 °F (per ACI Code 349). In response to a staff request to justify exceeding the allowable temperature of 150 °F, the licensee has stated that the case (i) in which this calculated exceedance occurs is an abnormal condition, when both pumps to augmented heat exchanger are unavailable (Reference 3). However, the licensee states that Oyster Creek Technical Specification (TS) 5.3.1D limits the bulk pool temperature to 125 °F, and that plant procedures require core discharge activities to cease when the pool temperature reaches 115 °F. The licensee has stated further that measures will be taken to prevent exceeding the TS limit of 125 °F by either commissioning the augmented cooling system to service, or returning the recently irradiated fuel assemblies to the reactor vessel. This explanation by the licensee adequately addresses the staff’s concern related to the maximum bulk pool temperature.

The structural analysis of the SFP was performed by using the finite element computer program, STARDYNE, to demonstrate the adequacy of the pool structure under fully-loaded fuel racks with all storage locations occupied by 3035 fuel assemblies (which is the maximum pool capacity after rack installation). The fully loaded pool structure was subjected to the load combinations specified in SRP 3.8.4, Rev. 1, 1981 (Reference 1). Table 8.7.1 of Reference 1 gives the predicted minimum factors of safety for the reinforced concrete ranging from 1.06 to 2.06 for bending moments of the concrete beams and slab, and 1.09 to 2.96 for shear of the same structural elements. In view of the calculated factors of safety, the staff concludes that the licensee's structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

2.3.3 Fuel Handling Accident

The following fuel handling accident cases were evaluated by the licensee: (1) one case for the drop of a fuel assembly (with its handling tool) impacting the top of a rack ("shallow drop" scenarios), and (2) two cases for the drop of a fuel assembly (with its handling tool) impacting the baseplate ("deep drop" scenarios).

The "shallow drop" event produces localized plastic deformation of the top of the impacted region, but the maximum depth of this plastic deformation is limited to 7.43", which is below the design limit of 18" (Reference 1, Section 7.2.3.1). The impact region of one of the two "deep drop" events, located above the support leg, produces negligible baseplate deformation. The maximum stresses produced by this impact in the liner, and in the pedestal cylinder at the contact surface with the bearing pad, are well below the failure limits, thus resulting in no damage to the SFP liner (Reference 1, Section 7.2.3.2). The concrete stratum directly beneath the pedestals sustains a localized, low compressive stress. The second "deep drop" condition through an interior cell produces some local deformation of the baseplate. While the baseplate does not fracture during the impact, the welds connecting the adjacent cells to the baseplate are severed. The licensee determined that this "deep drop" event produces a maximum baseplate deflection of 2.26", which is less than the distance of 8.1875" from the baseplate to the liner (Reference 1, Section 7.2.3.2). The maximum calculated stress of 44.3 ksi in the baseplate is below the failure stress of 71 ksi for this material (Table 7-2 in Reference 1). Therefore, the licensee concluded that the pool liner will not be damaged. The staff reviewed the licensee's fuel drop analysis results in Reference 1 and concurs with its findings.

Based on the review and evaluation of the licensee's submittal (Reference 1) and its subsequent response (Reference 3) to the staff's request for additional information, the staff concludes that the structural analyses of the spent fuel storage rack modules and SFP under seismic and accident loading conditions are in compliance with the acceptance criteria specified in the Final Safety Analysis Report and are consistent with the current licensing practice.

2.4 Heavy Loads

2.4.1 Background

The staff evaluated the considerations for moving heavy loads during the rack installation. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either (1) the potential for a load drop is extremely small or (2) the potential hazards of load drops do not exceed acceptable limits. The guidelines provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, and design, testing, inspection, and maintenance of cranes and lifting devices and analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increasing handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the guidelines of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety.

Oyster Creek's Updated Final Safety Analysis Report Table 9.1-10 defines a heavy load as 800 lbs. or more. According to the licensee, the maximum load to be lifted during the rack installation is 11,000 lbs. This includes the rack, lift rig, rigging, and the temporary hoist.

TS 5.3.1 establishes the maximum amount of spent fuel assemblies stored in the spent fuel storage pool at 2645 fuel assemblies. TS 5.3.1.B.1 states, in part, that "loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility..." The licensee did not propose any changes to TS 5.3.1.B.1. However, to assure that the potential for a rack drop during the rack installation is reduced and racks are not moved over fuel in the SFP or safety-related equipment along the safe load path, the licensee proposes to use defense-in-depth guidelines as provided in NUREG-0612.

2.4.2 Crane Hoisting System/Lifting Device

The addition of the new racks will not involve removal of any SFP storage racks from the SFP. The licensee states that activities involved in installing the new additional racks will be performed in accordance with NUREG-0612 and ANSI N14.6 -1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." The 100-ton Reactor Building overhead crane will be used to move the racks into the SFP. UFSAR Section 9.1.4.2.3, "Cranes and Hoists," states that the reactor building crane has been reviewed and approved for conformance with NUREG-0612. Therefore, the crane is designed to meet the requirements of EOCI-61, "Specifications for Electric Overhead Traveling Cranes - 1961 and the crane design has been reviewed and found acceptable in meeting the intent of the requirements in the Crane Manufacturers Association of America (CMAA) - "Specification No. 70 for Electric Overhead Traveling Cranes," and American National Standard Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)."

A temporary hoist (lifting rig) will be attached to the overhead crane to avoid submerging and contaminating the crane hook in the water in the SFP. The lifting rig will be remotely engaged and interposed between the crane hook and the rack and is specifically designed to lift the new spent fuel rack modules. It is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 -1978. It consists of four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. Both the stress design and the load testing of the lifting rig satisfies guidelines in Section 5.1.6(1) of NUREG-0612 and ANSI N14.6 (1978), respectively. Accordingly, the lift rods are designed as follows: (1) with the appropriate stress design factor as specified in ANSI N14.6 (safety factor of 5 to 1); (2) load tested to 300% of the maximum weight to be lifted and suspended for 1 hour; and (3) after load testing, the integrity of the critical weld joints is examined using a liquid penetrant. The rack lift will involve non-customized lifting devices (i.e., slings) that are in accordance with NUREG-0612 and ANSI B30.9-1971, “Slings.” Accordingly the slings will be proof tested at a minimum of 1.5 times their rated capacity in accordance with Section 9.3.3 in ANSI B30.9.

The staff finds that the use of the 100-ton reactor building crane coupled with the design and testing of the single-failure proof lifting rig, and use of the other lifting devices provides a large factor of safety that will enable the licensee to handle heavy loads with little to no risks of dropping the racks during rack installation.

2.4.3 Load Paths

The licensee states that safe load paths will be developed for moving the racks into the reactor building and the SFP. Review of UFSAR Section 9.1.4.2 “System Description” indicates that the racks will be brought into the building along a path similar to that for new fuel. Therefore, using the reactor building crane, the new racks will be lifted from the ground level of the reactor building equipment hatch, el. 23', to the SFP operating floor at elevation 119', then into the SFP. The licensee plans to shuffle the spent fuel in racks in the SFP into racks that are not in the travel path of the lifted racks. Therefore, the new racks will not be carried over any region of the pool containing fuel. In addition, crane stop blocks will be temporarily installed to prevent crane travel/load movement over fuel. Also, the racks will be lifted such that the center of gravity of the lift points will be aligned with the center of gravity of the load to allow better control of the lifted load. The licensee stated that the crane and bridge operators would be trained in accordance with ANSI B30.2-1996. Also, plant-specific training would be provided, including training in the use of the lifting system, upending equipment, and all other aspects of the rack installation.

2.4.4 Heavy Loads Handling Accident Analysis

Although heavy loads analyses are not expected as a part of compliance with Generic Letter 85-11, “Completion of Phase II of ‘Control of Heavy Loads At Nuclear Power Plants,’ NUREG-0612,” the licensee’s submittal did address the possibility of a drop of the heaviest rack module (11,000 lbs.). This is in accordance with the staff’s recommendations in NRC Bulletin (NRCB) 96-02, “Movement of Heavy loads over Spent Fuel, Over Fuel in the Reactor Core, or over Safety-Related Equipment.” The licensee evaluated a drop of the empty fuel rack from a height of 40 feet to the bottom of the SFP. The evaluation of the rack drop indicated that the pool liner would be pierced causing potential liner leakage. Also, the concrete would be indented sufficiently (about 2.7 inches deep) to cause local cracking of the concrete material. Gross failure of the SFP slab does not occur. As stated in the UFSAR Section 9.1.2.2.1 “Spent Fuel Storage Pool,” the plant’s drainage system beneath the stainless steel pool liner detects and collects leakage between the liner and the concrete. Licensee response to NRC Request for Additional Information, dated May 2, 2000, states that the SFP drainage system has drain lines with manual isolation valves that would be closed prior to installation of the new racks to prevent SFP inventory loss if a rack drop occurs. Furthermore, any SFP water that is collected by the drainage system goes into the Reactor Building Equipment Drain Tank where it can be recycled via the liquid radwaste system to the Condensate Storage Tank (CST) enabling the licensee to make up the loss of inventory to the pool. Makeup from the CST can be supplied at a rate of 250 gpm to 420 gpm. Additional makeup to the SFP can be obtained at a rate of 150 gpm from the Demineralized Water Storage Tank. Also, the Fire Protection System can be used to provide SFP makeup. Accordingly, the licensee could cope with and manage damage to the SFP caused by the drop of a rack. A cask drop accident is not a consideration in this evolution; therefore, it was not analyzed.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee plans to implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. They will provide the following: (1) comprehensive training to the rack installation crew, (2) use of redundantly designed lifting rigs, (3) inspection and maintenance checks on the cranes and lifting devices prior to the rack operation, and (4) specific procedures that cover the entire rack installation effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement.

The staff accepts the licensee’s finding that they can cope with and manage the damage to the SFP liner and concrete slab and maintain water over the fuel if a rack drop was to occur. Also, the staff agrees with the licensee that the use of the crane in conjunction with administrative procedures and controls focused on, but not limited to, the areas noted above will enable the licensee to maintain safety during the rack installation.

Based on the preceding discussions, the staff finds that the aforementioned considerations of heavy loads to support the proposed changes to TS 5.3.1.E and the increase in the storage of spent fuel assemblies in the SFP are acceptable. The licensee’s use of the reactor building crane in conjunction with administrative controls that are in accordance with NUREG-0612 will help to maintain safety during the installation of the additional new spent fuel storage racks in the SFP. These considerations for moving heavy loads which will enable the licensee to move the racks during installation while preventing any damage to spent fuel and the SFP structure, are acceptable to the staff.

2.5 Materials

The staff evaluated the compatibility of structural materials and Boral with the SFP environment. The existing ten racks utilize Boraflex as the neutron absorber material. The new Holtec racks contain Boral as the active neutron absorber. The Boral neutron absorber is stable, strong, durable, and corrosion resistant. These free-standing, self-supporting racks are designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel Code.

2.5.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304L for all sheet metal stock and internally threaded support legs, ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100 °F) for externally threaded support spindle, and ASME specification SFA 5.9 ER308L for weld material.

These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the SFP environment. Therefore, they are acceptable for use in this application.

2.5.2 Poison Material

The Holtec racks employ Boral™ as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in SFP environments, where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

Based on its evaluation, the staff finds the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec International are compatible with the SFP environment at the Oyster Creek Nuclear Generating Station. The type of degradation exhibited by the racks does not affect their neutron absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

2.6 Radiological Effects

2.6.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the Oyster Creek spent fuel racks with respect to occupational radiation exposure. As stated above, for this modification the licensee plans to install 4 new fuel rack modules in the SFP. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed for between 0.7 and 1.4 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. The Radiation Protection department will prepare Radiation Work Permits for the various jobs associated with the reracking operation. Each member of the project team will receive radiation protection training on the reracking operation, as required by 10 CFR Part 19. Daily pre-job briefings will be used to inform workers of job scope and techniques. Personnel will wear protective clothing and will be required to wear personal dose monitoring equipment as required by approved plant procedures and 10 CFR Part 20.

Prior to the start of rack installation, radiation surveys will be conducted for direct radiation levels and smearable contamination levels. Previous historical experience during similar reracking shows that radioactive airborne material level increases in the above-pool work area should be negligible. In order to minimize contamination and airborne problems, all equipment removed from the pool will be rinsed off and wiped down, and surveyed.

In accordance with 10 CFR Part 20.1602, if divers were to enter the pool, the Oyster Creek SFP would have to be controlled as a Very High Radiation Area (VHRA), since divers could be able to gain access to the spent fuel stored in the pool. However, the licensee does not intend to use underwater divers during the installation of the new fuel racks. If emergent, unusual circumstances do occur and the licensee needs to use divers for the installation of the new SFP rack modules, then all diving operations will be governed by special procedures. These procedures will meet the intent of Regulatory Guide 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,” Appendix A. This Appendix, “Procedures for Diving Operations In High and Very High Radiation Areas,” summarizes good operating practices for divers. These operating practices include survey, remote dose monitoring, radiation protection stop-work authority, use of physical barriers, communication with divers, and emergency procedures. Additionally, if divers are used for the rack installation, detailed radiation surveys in and around the dive area will be performed, and sources greater than 1 rem/h will be identified and controlled. Thermoluminescent dosimeters will be used to verify underwater survey instrument results. Divers will be trained and use calibrated underwater radiation survey instruments for confirmatory surveys of their work area.

The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase due to the expanded SFP storage capacity. However, the licensee will operate continuous air monitors in areas where there is a potential for significant airborne activity during the fuel reracking operation. In addition, the Reactor Ventilation Radiation Monitor will be used to monitor airborne activity.

An underwater vacuum system will be used as necessary to supplement the installed SFP filtration system. Reducing contamination levels before diving and after pressure washing of equipment to assist in pool clarity restoration are some activities where supplemental underwater cleaning may be necessary. The licensee will use the existing SFP filtration system during fuel rack installation to maintain water clarity in the SFP.

The storage of additional spent fuel assemblies in the SFP will result in negligible increases in the dose rates on the refueling floor and in adjacent accessible areas to the SFP. Maximum dose rates increases outside the concrete wall of the SFP will be less than 1 mr/hr. The increased fuel storage will have a negligible impact on existing accessible areas below the concrete floor of the SFP (in the overhead of the shutdown cooling room, an existing high radiation area).

On the basis of our review of the Oyster Creek Station's proposal, the staff concludes that the SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds the projected dose for the project of about 1 to 2 person-rem to be in the range of doses for similar SFP modification activities at other plants and therefore acceptable.

2.6.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP purification system. The licensee predicts that only a very small amount of addition resin will be generated from the new rack installation; therefore, the change-up frequency of the SFP purification system may be very slightly increased, but only temporarily during the reracking operation. In order to maintain the SFP water reasonably clear and clean, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP, as necessary, to remove any radioactive crud, sediment, and other debris before the new fuel rack modules are installed. Filters from this underwater vacuum will be a minor source of solid radwaste. Overall, however, the licensee does not expect that increasing the storage capacity of the SFP will result in a significant change in the generation of solid radwaste at Oyster Creek.

2.7 Off-Site Dose Consequences

In its application, the licensee evaluated the possible offsite dose consequences of a fuel handling accident (FHA) in the SFP. The proposed reracking of the Oyster Creek SFP will not affect any of the assumptions or inputs used in evaluating the dose consequences of the FHA.

The staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the licensee's calculated doses. In performing these calculations, the staff used the assumptions and guidance given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The licensee also followed this guidance. For an FHA occurring in the SFP, the licensee and staff assumed that the cladding of all the fuel rods in the dropped assembly fails. The damaged fuel rods are assumed to contain freshly off-loaded fuel with a minimum of 100 hours of decay.

The staff determined the licensee's assumptions and calculational method are acceptable, and the staff's calculations gave results that confirmed the licensee's offsite dose results for an FHA in the SFP. The results of the licensee's calculations are given in Table 1. The licensee's results meet the acceptance criteria for offsite dose given in SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents" of well within (25% of) 10 CFR Part 100 guidelines, in other words, 75 rem thyroid and 6.25 rem for the whole body dose.

Table 1
Licensee Projected Offsite Dose for the Fuel Handling Accident in the SFP

	EAB Dose (rem)	SRP Acceptance Criteria (rem)
Thyroid	0.487	75
Whole Body	0.196	6.25

The staff has determined that the radiological dose analyses performed by the licensee in support of the proposed SFP reracking are acceptable. The staff also finds the licensee calculated radiological consequences of a fuel handling accident within the SFP meet acceptance criteria given in 10 CFR Part 100. The current FSAR analysis for the fuel handling accident inside the reactor building remains bounding. Therefore, the staff finds the proposed installation of additional racks at Oyster Creek to be acceptable with regard to potential radiological consequences of a hypothetical fuel handling accident in the SFP.
Therefore, the staff concludes that the proposed revisions to the TSs for the proposed SFP rerack are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published (65 FR 55061) in the Federal Register on September 12, 2000. Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter dated June 18, 1999, from GPU Nuclear to USNRC, Subject: “Oyster Creek Nuclear Generating Station Technical Specification Change Request No. 261 - SFP Expansion.”

2. Letter dated November 22, 1999, from Helen N. Pastis, NRC, to GPU Nuclear, Subject: "Request for Additional Information on your proposed License Amendment concerning the SFP Expansion."

3. Letter dated January 6, 2000, from GPU Nuclear to USNRC, Subject: "Oyster Creek Nuclear Generating Station -- Response to Request for Additional Information re: Proposed License Amendment for SFP Expansion."

4. USNRC Office of Technology Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, dated April 14, 1978, as modified by amendment dated January 18, 1979.

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