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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

GPU Nuclear Corporation (the licensee/GPUN) requested, in a November 14, 1990 license amendment request, modifications to the TMI-1 Technical Specifications (TSs) to reflect installation of new spent fuel storage racks. The new storage racks will provide an increase in spent fuel storage capacity from the present 749 assemblies to a total installed storage capacity of 1990 assemblies. The increased storage capacity will be accomplished by partially reracking pool A to provide 846 locations; complete reracking of pool A will provide an ultimate storage capacity of 1494 assemblies. Existing storage racks in pool B, which contain 496 locations, will continue to provide this storage capacity; ultimate spent fuel storage capacity would thus provide for a total of 1990 fuel assemblies. New racks would be installed to provide for storage of spent fuel assemblies with an initial enrichment of up to 4.6 weight percent (w/o) U-235 and burnup levels of up to 60,000 MWd/MTU.

TMI-1 has two spent fuel pools (pools A and B). The plan is to rerack pool A, changing the present pool A capacity from 253 to 1494 locations; pool B locations will not be affected by the planned reracking. The licensee plans to install 846 of the 1494 cells in pool A during fuel cycle 9; the remaining 648 cells will be installed at some later time. Pool B, containing old racks with the capacity for 496 spent fuel assemblies (SFAs) will not be changed. However, the gate used to separate pool A and B will be removed so as to permit access to both pools. The wall support brackets for the gate will also be removed in order to provide room for the new rack configuration. The storage racks in spent fuel pool (SFP) A will be divided into two regions. Region I will contain storage racks with 195 locations for storing fresh and irradiated fuel assemblies and Region II will have 648 locations for storing higher burnup and lower enrichment and 3 locations for storing failed fuel. There will be a difference in design of the storage racks in Regions I and II. The storage racks in both regions will be constructed from ASTM A240-Type 304 stainless steel with only adjustable support spindles made from A564-Type 630 precipitation hardened stainless steel. The neutron absorbing material will be Boral. Boral is a neutron absorber material manufactured by Brooks and Perkins and consists of a dispersed boron carbide in an 1100 aluminum alloy matrix and clad with 1100 aluminum alloy. In the storage racks in Region I, Boral panels are 7.5 inches wide, 0.081 inches thick, 138 inches long and have B-10 loading of 0.0211 gm/sq.cm. They are attached to an individual cell by picture frame sheathings which are welded to its outside walls. The sheathings have openings through which the gases which may be generated by radiolysis and/or water aluminum reaction can escape, thus preventing swelling and bulging due to pressure buildup. The individual cells are welded together into a storage rack module leaving a 3.4 inch wide gap between the individual cells. In Region II Boral panels are 7.5 inches wide, 0.091 inches thick, 144 inches long and have B-10 loading of 0.026 gm/sq.cm. The panels are similarly attached to the cell walls as in Region I, but the individual cells are welded together edge-to-edge, thus forming a honeycomb-structured storage rack. The storage racks will be exposed to air-saturated borated water of the spent fuel pool.

The licensee proposed a surveillance program to monitor performance of the Boral in the spent fuel pool. For that purpose, 16 specially designed test coupons will be used. Eight of them will be exposed to a random batch of spent fuel and will serve for long-term surveillance and the other eight will be placed in the center of freshly discharged fuel assemblies and will serve to evaluate accelerated exposure. Each coupon will have Boral specimen encased in a jacket of material identical to that used in the racks and the position and tolerances will be similar as that in the actual fuel cell. The jacket will have provisions for easy opening without disturbing the Boral specimen. The coupons will be removed at scheduled intervals and examined for loss of physical and neutron absorbing properties. In addition to the coupon surveillance, a direct surveillance of Boral panels in the fuel racks will be performed. This surveillance will consist of blackness testing followed by neutron radiography of suspected areas.

The licensee initially provided information regarding the reracking plan in a submittal dated November 14, 1990, from H. D. Hukill entitled "Technical Specification Change Request No. 201, Spent Fuel Reracks." Supplement 1 to the change request was forwarded in a submittal dated June 6, 1991 from T. G. Broughton. The licensee provided further information on June 14, September 18, November 27, December 12, 1991, and February 13, 1992, in response to staff requests for additional information.

2.0 EVALUATION

2.1 Radiological

2.1.1 Occupational Exposure Control

In the licensee's November 14, 1990 license amendment request, the licensee estimated that total occupational exposure for the reracking activities would be between 5 and 10 person rem. The SFP A rerack will be performed in accordance with TMI-1 approved written procedures. The majority of the spent fuel assemblies will be moved to SFP B during the rerack operation. The 24 to 80 assemblies that remain in SFP A will be moved to the north end of the pool. This will give an adequate shielding buffer of approximately 15-20 feet between the rerack work and the storage of the remaining fuel assemblies, in the unplanned event that divers would have to be used. The majority of the exposure from the rerack operation will be due to removing, decontaminating, and shipping the six spent fuel storage rack modules presently in SFP A and pool cleanup operations, including handling and processing of vacuum filters. The SFP racks will be approved for shipment per the requirements of 10 CFR Part 71 and 49 CFR Parts 171-178.

As part of the as low as is reasonably achievable (ALARA) planning, the licensee plans to make use of remote handling tools as much as possible for jobs such as removing nuts from existing rack retainer bolts and leveling operations. The tooling and handling equipment designed for the reracking work has been used in previous rerack projects, and has proven its effectiveness in helping the licensee to meet the set ALARA goals. To minimize possible contamination (e.g., from hot particles) to personnel and plant facilities from the existing SFP racks after removal, high pressure water decontamination of these racks will be conducted under water in accordance with approved procedures. The entire operation will be covered by the existing radiation protection program under the direction of the Radiological Engineer. All pool side and in-pool work activities will be surveyed by Radiological Controls (RC) personnel with specific radiological witness and hold points incorporated into written procedures. RC personnel will also have stop work authority in the event of any unsafe or questionable operations.

Past operations experience involving rerack operations at other facilities has shown that there are negligible increases in airborne radioactivity in the spent fuel pool area. This coupled with the licensee's experience involving fuel movements during refueling outages indicates neither the current health physics program nor area monitoring systems need significant modification. All plant personnel working on this job will be covered by applicable Radiation Work Permits. Also available will be appropriate protective clothing, respiratory protective and air sampling equipment, as needed, and personnel radiation monitoring equipment such as thermoluminescent dosimeters (TLDs), pocket dosimeters, and extremity badges.

An isotopic analysis of the SFP water indicates that the primary radionuclides are Co-58, Co-60, Ag-110m, Cs-134, and Cs-137 with concentrations in microcuries per milliliter of 1.5 E-5, 4.4 E-5, 4.6 E-5, 3.2 E-4, and 6.4 E-4 respectively. These radionuclides are the primary sources of radiation associated with the SFP water.

Operating experience has shown that typical dose rates of 1.0 mrem/hr are expected at the edge or above the pool center with levels of 2.5 to 3.0 mrem/hr during refueling operations. Further, there have been no noticeable increases in airborne radioactivity above the SFP and there has been no evidence of crud (e.g., Co-58, Co-60) buildup along the sides of the pool that might cause local areas of high radiation.

The major work effort to be expended for the rerack job will be for the removal of the old racks. This is expected to require 800 person-hours which excludes potential diving operations. The use of a diver is not planned at this time and would only be used if remote tooling fails to disconnect the rack-to-pool attachments. If needed, detailed procedures and radiological controls would be implemented to ensure minimum cumulative radiation dose to the diver. The total person-rem projected for the entire SFP A rerack modification is estimated to be between 5 to 10 person-rems. This estimate is consistent with the historical range of doses for SFP reracking operations and is a small fraction of the approximately 138 person-rem per year that TMI-1 has averaged over the past 3 years. The staff finds this estimate to be conservative.

Based the staff's review of the TMI-1 proposal, the staff concludes that the projected activities and estimated person-rem doses for this project appear achievable and that the licensee will be able to maintain individual occupational radiation exposures within the limits of 10 CFR Part 20 and maintain doses ALARA. Therefore, the proposed radiation protection aspects of the SFP A rerack are acceptable.

2.1.2 Solid Radioactive Waste

In the licensee's November 14, 1990 amendment request, GPUN noted that spent fuel racks removed would be decontaminated as much as possible by washing and wipedowns, packaged, and shipped to a licensed processing/disposal facility. The licensee further noted that although a small amount of additional spent resins may be generated by the pool cleanup system on a one-time basis, no significant increase in the volume of solid radioactive wastes is expected as a result of the expanded storage capacity.

Finally, GPUN noted that shipping containers and procedures will conform to U.S. Department of Transportation (DOT) regulations as well as the requirements of any State DOT office through which the shipment may pass.

Based on the staff's review, the staff finds that the licensee's plans for disposal of solid radioactive waste generated in connection with the planned reracking operation meet the staff's criteria and are, therefore, acceptable.

2.1.3 Design Basis Accidents

In GPUN's December 12, 1991 response to a November 1, 1991 request for additional information, the licensee noted that an NRC generic environmental assessment related to use of extended burnup up to 60 GWd/MTU and increased enrichment up to 5.0 w/o, which was published in the Federal Register (53 FR 30355), is applicable to TMI-1. Therefore, 10 CFR 51.52(b) or Table S-4 have not been separately addressed.

The staff has reviewed the licensee's submittals as well as a report prepared for the NRC entitled "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," NUREG/CR 5009 dated February 1988. In this report, prepared by Pacific Northwest Laboratory (PNL), the changes that could result in the NRC design basis accident (DBA) assumptions were examined to determine which assumptions contained in various Standard Review Plan (SRP) sections and/or Regulatory Guides might be changed as a result of extended burnup fuel up to 60,000 MWD/MTU.

The staff agrees with the PNL report's conclusion that the only DBA which could be affected by the use of extended burnup fuel would be the potential thyroid doses that could result from a fuel handling accident. The PNL report estimates that the calculated iodine gap-release fraction is 20% greater for some high-power fuel designs than the Regulatory Guide 1.25 assumed value of 0.10. Thus, the calculated thyroid doses resulting from a fuel handling accident with extended burnup fuel could be 20% higher than those estimated using Regulatory Guide 1.25.

Although no fuel handling accidents aving significant offsite radiological consequences have occurred, such accidents must be postulated and their potential consequences analyzed. In the licensee's analysis, an initial enrichment of 4.6 w/o in the isotope U-235 was assumed and a burnup of 60,000 MWd/MTU was assumed to have been attained by operating at a specific power of 30.97 MWd/KgU immediately prior to shutdown. In the licensee's analyses, it was assumed that the fuel handling accident resulted in the release of gaseous fission products in the pellet-clad gaps of the outer row of rods in one fuel assembly (56 rods assumed damaged). The licensee further evaluated two cases of pellet-clad gap activity, the escape rate method and using the assumptions identified in Regulatory Guide 1.25.

In the licensee's analysis, it was assumed that even with the higher burnup, the calculated doses will not differ appreciably from those of previous evaluations and that calculated doses would not differ appreciably from those of previous evaluations. However, since the licensee plans to store fuel enriched to 4.6 w/o U-235 and with a burnup of 60,000 MWd/MTU, the staff reanalyzed the fuel handling DBA for this case. As noted in NUREG/CR-5009 "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988, increased burnup could increase offsite doses from the fuel handling accident by a factor of 1.2 due to the fact that the calculated iodine gap-release fraction for some high power fuel designs is increased by 20%.

The staff conservatively assumed an increased gap fraction of 0.12 as compared t the previously assumed gap release fraction of 0.10 for iodines for all analyzed fuel handling accidents. The affected accidents include the spent fuel assembly drop, the spent fuel assembly cask drop, and the installation accident involving dropping of a rack onto fuel.

With regard to the spent fuel assembly drop accident, the staff assumed an increased radioiodine release of 20%, as described above. The spent fuel assembly drop consequences analyzed in the 1973 TMI-1 Operating License Safety Evaluation Report (OL-SER) were previously calculated by the staff to be 41 rem (thyroid) at the site boundary. With the 20% increase in radioiodine gap activity described in NUREG/CR-5009, the calculated radiological consequences at the site boundary would increase to 49 rem thyroid. This value is further increased by a factor of 2568/2535 to reflect the current licensed power level (2568 MWt) instead of the power level analyzed in the OL-SER (2535 MWt). The resultant calculated thyroid dose of 50 rem is well within the guideline values of 10 CFR Part 100 and meets the acceptance criterion of SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents" that calculated doses should be well within the guideline values of 10 CFR Part 100.

For the case of a hypothetical drop of a filled shipping cask conservatively assumed to result in the rupture of all fuel rods in 10 assemblies inside the cask, the licensee calculated an inhalation thyroid dose (using Regulatory Guide 1.25 assumptions) of 8.7 rads. In performing the analysis, the licensee assumed accident occurrence during cask transfer to a trailer at a time 120 days after plant shutdown. Further assuming no decontamination factor for water scrubbing and filtering, and a X/Q value of 1.2×10^{-3}, the licensee calculated an inhalation thyroid dose of 8.7 rads. Assuming the radionuclide inventories and constants listed in Table 9-1 of the licensee's submittal and increasing the radioiodine gap inventory available for release by 20% as noted in NUREG/CR 5009, the NRC staff independently calculated the radiological consequences of a cask drop accident 120 days after shutdown. With 10 assemblies assumed failed in the cask drop, the staff calculated a potential thyroid dose of 10.4 rem based on failure of 10 assemblies 120 days after shutdown. Consequently, the cask drop analysis for 120 day shutdown after operation satisfies the acceptance criteria noted in SRP Section 15.7.5-2 and is therefore acceptable.

Finally, the licensee considered the potential for installation accidents (e.g., dropping a cask onto spent fuel) and concluded that existing TSs and administrative controls are adequate to preclude movement of a rack directly over any fuel. The licensee noted that the potential for a heavy load drop is extremely small and that no heavy loads will be carried in the spent fuel pool area until all fuel in the pool has decayed for at least 72 hours, thereby limiting the assumed release from damage to all stored assemblies such that resultant doses are less than 10% of the 10 CFR Part 100 acceptance criteria. The staff has reviewed the information provided by the licensee related to the analysis of the potential for installation accidents (e.g., dropping a spent fuel storage rack onto spent fuel) and concludes that the analyzed consequences from this unlikely event meet the criteria set forth in the SRP and are therefore acceptable.

2.2 Material Properties/Corrosion

The austenitic stainless steel in the spent fuel pool liner and racks assemblies is compatible with the air saturated borated water and the radiation environment of the spent fuel pool. Borated water would have a pH higher than 4.5 and the oxygen dissolved in the water will help to passivate the stainless steel. In this environment austenitic stainless steel will exhibit only extremely low rates of corrosion. These corrosion rates are negligible for even the thinnest stainless steel elements of pool liner or rack assemblies. Galvanic attack between the stainless steel in pool liner or rack assemblies and Zircaloy in the fuel assemblies or Boral will not be significant since these materials are protected by passive oxide films. Concentration of chloride is maintained below the limit at which significant initiation of stress corrosion cracking could occur.

Boral 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. It has been qualified for 1.0E11 rads of gamma radiation while maintaining its neutron attenuation capability. Tests have shown that Boral does not possess leachable halogens that could be released into the pool environment in the presence of radiation. Similar findings have been made regarding the leaching of elemental boron from the Boral. In water with pH between 4.5 and 7, corrosion of Boral is insignificant. Surveillance coupons containing Boral and blackness testing of Boral panels will provide time-related information of the actual behavior of Boral in the spent fuel pool. The staff reviewed the proposed surveillance program for monitoring performance of the Boral panels in the spent fuel pool and concludes that the program will reveal deterioration that might lead to loss of neutron absorbing capability during the life of the spent fuel racks. The staff does not anticipate that such deterioration will occur, but in case it does, it would be gradual. In the unlikely event of Boral deterioration in the pool environment, the monitoring program will detect such deterioration and allow the licensee time to take suitable corrective actions.

2.3 Criticality Analysis

Two separate storage regions are provided in SFP A, with independent criteria defining the highest potential reactivity in each of the two regions. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.6 w/o U-235 or spent fuel regardless of its discharge burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated sufficient minimum burnups. The analysis of the reactivity effects of fuel storage in Region 1 and 2 was performed with the two-dimensional multi-group transport theory computer code, CASMO-2E. Independent verification calculations were also made with a Monte Carlo technique using the AMPX-KENO computer package with the 27-group SCALE cross section library. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. The staff concludes that the analysis methods used are acceptable.

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

(1) Unborated pool water at the temperature yielding the highest reactivity (68°F).

(2) Assumption of infinite array of storage cells in all directions.

(3) Neutron absorption effect of structural material is neglected.

The staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design in Region 1, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, and fuel enrichment and density tolerances were accounted for as well as eccentric fuel positioning and reduced boral length (cutback). These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations. The final Region 1 design when fully loaded with fuel enriched to 4.6 w/o U-235 resulted in a k-eff of 0.9285 when combined with all known uncertainties. This meets the staff's criterion of k-eff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

For Region 2. the same uncertainties were considered. In addition, an allowance for uncertainty in the burnup analyses and the axial burnup distribution as well as an adjustment to the CASMO-2E based on KENO calculations were included. A series of reactivity calculations were made to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield the equivalent k-eff. This method for obtaining the constant reactivity curve for required burnup as a function of enrichment is the standard one used for rack reactivity evaluations and is acceptable. The new TS Figure 5-4 shows the constant k-eff contour generated for the Region 2 racks. From this Figure, it can be seen that the reactivity of the racks containing fuel at 37,000 MWD/KgU burnup which had an initial enrichment of 4.6 w/o U-235 is equivalent to the rack reactivity with fresh fuel (zero burnup) having an initial enrichment of 1.75 w/o U-235. This configuration resulted in an acceptable maximum k-eff of 0.9390, including all appropriate uncertainties.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in Figure 5-4 or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of approximately 600 ppm of boron in the pool water required by TS 5.4.1 during fuel handling operations since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k-eff caused by the boron more than offsets the reactivity addition caused by credible accidents.

The following TS changes have been proposed as a result of the requested spent fuel pool reracking. The staff finds these changes acceptable.

(1) TS 5.4.1.a replaces the description of the nominal center-to-center distance of the existing SFP A racks with the description of the nominal center-to-center distance of the new Region 1 and 2 racks. This TS also clarifies that the new racks for SFP A are designed to maintain a k-eff of less than 0.95 based on fuel assemblies with an enrichment of 4.6 w/o U-235.

(2) TS 5.4.2.d is revised to identify the revised number of fuel assembly storage locations and the corresponding equivalent full core capacity for the new racks as initially installed in SFP A. Footnote "**" is revised to identify that an additional 648 storage cell locations can be installed to provide a total of 1494 storage locations or 8.44 cores. Footnote "***" is editorially revised to delete the word "reduced" from the description of the center-to-center spacing of the SFP B racks since its dimension is no longer the minimum spacing dimension for SFP racks.

(3) TS 5.4.1.b, 5.4.2.c, 5.4.2.d, and 5.4.2.e are revised to delete reference to the fuel assembly storage capability in the Fuel Transfer Canal as these racks will have been removed.

(4) TS 5.4.2.g and Figure 5-4 are added to provide administrative controls to limit storage of spent fuel assemblies in Region 2 of the SFP A storage racks based on initial enrichment and cumulative exposure.

2.4 Heavy Loads Concerns

2.4.1 Fuel Handling Building (FHB) Crane

The licensee noted that the FHB crane would be used to lift empty, old and new racks during the reracking process. Fuel in old racks would be transferred to new racks prior to removal of old racks where necessary.

The crane has a design rated capability of 110 tons while the heaviest racks the crane will be required to lift weigh 15 tons - a ratio of 8.46/1 with respect to design capability. Based on the crane's ultimate load capacity, the crane could lift a load 38.7 times greater than one rack before failure.

The present TS (specification 3.11.2) does not permit using this crane for lifting loads in excess of 15 tons unless the key operated travel interlock system is imposed. This assures that travel of any load in excess of 15 tons would be limited, in the event of a postulated load drop accident, to areas where damage to the spent fuel pool structure or damage to redundant trains of safety related components could not occur.

2.4.2 Special Lifting Devices

The licensee reported that the special lifting device used for handling the racks in TMI-1 pool A will have four independent load paths. The design would allow failure of one load bearing member without causing uncontrollable lowering of the load.

Further, the licensee noted that the special lifting device would comply with all provisions of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More For Nuclear Materials."

The licensee agreed to provide a plan for surveillance testing of the special lifting device, in compliance with the provisions of section 5.3.1 "Testing to Assure Continued Compliance," of ANSI N14.6-1978.

2.4.3 Lifting Devices Not Specifically Designed

The licensee committed to comply with the criteria of paragraph (1)(b) of section 5.1.6, "Single-Failure-Proof Handling Systems" of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," in the use of lifting devices not specifically designed, when reracking SFP A. Paragraph (1)(b) allows the use of alternatives for these devices (which includes slings) by permitting the use of either dual (redundant) devices or the use of a safety factor twice that normally required (10/1 versus 5/1) for such lifting devices. This paragraph also requires compliance with ANSI B30.9-1971, "Slings," including the consideration of dynamic loads in the installation and use of these devices.

2.4.4 Safe Load Paths

The licensee is committed to developing safe load paths for e use in carrying racks to and from SFP A. In addition, the licensee nas designed sacrificial impact shields to strengthen regions of the operating floor of the FHB. This includes a minimum clearance of 40 inches, laterally, between a rack or special handling device and spent fuel. The licensee proposes to follow structural floor members and beams, to the extent practical, during load movement, to maximizing the floor's ability to withstand the effect of a dropped load.

2.4.5 Operating Procedures

The licensee intends to provide procedures "...to cover the entire gamut of operations..." This will include mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. These procedures will cover handling of both old and new racks.

2.4.6 Training

The TMI-1 licensee plans to provide comprehensive training to the rack installation crew. This will include use of the lifting and upending equipment. A training seminar will use videotapes of the actual lifting and handling of the actual modules to be stored in the SFP. The licensee will require every crew member to pass a written examination in the use of the lifting and handling equipment.

2.4.7 Crane Inspection and Maintenance

The licensee noted that daily and monthly inspections of the FHB crane are required prior to use. This inspection includes checks of the functional operating mechanisms; electrical and hydraulic components and connections; the hoist hooks for visual cracks or deformations; the main and auxiliary hoist ropes for wear, twisting, stretching, broken strands; and the gear box oil levels. The licensee will perform a yearly inspection if more than 6 months have elapsed since the last yearly inspection. The yearly inspection checks the main switch contacts and connections; the drum rollers; contactors, relays and other electrical components; motors; lights and over-travel interlocks; structural integrity of rails, bridge, trolley and stops; bridge and trolley gear boxes, drive wheels, and line shaft; the main hoist drum gear box, mechanical and electrical load brake; and hooks and ropes.

The licensee plans to relubricate the moving parts of the fuel handling crane prior to the start of reracking. Finally, the licensee plans to load test the crane prior to use in reracking.

2.5 Thermal/Hydraulic Analysis

2.5.1 Decay Heat Load Calculations

The licensee reported that the decay heat calculations had been conducted in accordance with the provisions of Branch Technical Position (BTP) ASB 9-2. In the calculations, it was assumed that pool A was filled with a complement of 1640 spent fuel assemblies from 22 previous refuelings and that this fuel had accumulated 1460 full power days of exposure in the reactor prior to storage.

For the normal core offload of 80 SFAs, the licensee assumed that these were discharged at the end of a normal 2-year operating cycle, after having undergone a total reactor exposure time of 6 years. Discharge was assumed to occur at the rate of six assemblies per hour after a decay period of 150 hours after shutdown. For the fuel core reload, it was assumed that reload for Cycle 23 had been conducted prior to the full core offload. Thus, 1720 locations would be filled, leaving 278 empty spaces, including some spaces in SFP B which were not accessible for storage of fuel assemblies. This would leave room for only one full core offload.

For the full core offload the licensee considered two cases: (1) beginning-of-cycle (BOC) wherein a full core offload occurs after 36 days and

(2) end-of-cycle (EOC) wherein a full core offload occurs after 2 years of normal reactor operations.

2.5.2 Spent Fuel Pool Coolant Temperatures

The licensee made a comprehensive evaluation of the heat losses from the SFP during a normal or full core unloading. The evaluation considered heat lost to the FHB atmosphere as well as heat removed by the spent fuel pool cooling system. The heat lost to the building atmosphere includes heat lost due to convective heat transfer between the pool surface and the air as well as heat lost by evaporation of water at the surface of the pool.

In response to a staff inquiry the licensee provided information to show that the analysis of heat losses from the pool surface to the FHB atmosphere provided conservative results, i.e., the calculated pool temperature was higher than the actual temperature.

The licensee provided results for five cases, as follows:

Case	Number of Operating SFP Coolers	Offload	Maximum Bulk SFP Coolant Temperature, °F
1	1	normal[a]	158.4
2	2	normal[a]	129.7
3	2/1[b]	normal[a]	148.9
4	2	full core (BOC)[c]	156.1
5	2	full core (EOC)[d]	154.6

(a) normal offload - 80 SFAs
(b) two coolers initially, then one, to assure peak temperature below 160°F
(c) after 36 days operation
(d) after 2 years operation

NOTE: heat exchangers fouled to design values. Eighty heat exchanger tubes (of 328 total) in each cooler assumed to be plugged.

Some conservatism was applied to the calculation of heat lost to the SFP cooling system, including the assumption of cooler fouling and plugging, as noted above.

2.5.3 Fuel Element Cladding Temperature

The licensee calculated the temperature of fuel cladding by determining the maximum local water temperature in the SFP. The licensee also included the radial SFA peaking factors, the rod to bundle maximum power ratio and the maximum axial power factor of the rod in the calculation. Further, the calculation included the use of a crud deposit over the entire surface of the fuel pin.

Utilizing these conditions, the licensee calculated the maximum cladding temperature for two cases - one with no flow channel blockage, one with 50% blockage. In the former case a maximum cladding temperature of 251.9°F was obtained; in the latter case, 260.2°F.

2.5.4 Pool Boiling

The licensee assumed the worst cases in calculating the time for the SFP coolant to reach the boiling point of 212°F. Included was the assumption that all fuel assemblies are from the latest batch, discharged at the same time and having been operated in the reactor for the maximum irradiation time. It was also assumed that water flowed upward through the SFAs and downward between the rack modules and pool wall with no heat transfer to the surrounding walls and pool bottom. An idealized downcomer area was assumed to include the minimum gap between racks and wall. Five cases, corresponding to those above which were used to calculate the peak SFP coolant temperature during offloads, were calculated. The results showed that it would take 10.3 hours to 27.5 hours for the pool to start boiling once the normal SFP cooling system failed or was shut down.

The decay heat removal system, forced ventilation system and chemical addition system may be used to help cool the SFP as required. The forced ventilation system aids by improving the heat transfer from the pool to the FHB atmosphere and by aiding in the process of evaporation of SFP coolant into the FHB air. The chemical addition system may be used for SFP makeup by adding cool water from the reclaimed water storage tank into the SFP. The decay heat removal system may be used to add water to the SFP from the borated water storage tank.

2.5.5 Cleanup System

Part of the SFP coolant is diverted to the Liquid Waste Disposal System, after passing through the SFP heat exchangers to remove decay heat. There the diverted water comes into contact with the deionizer resin which removes fission products from the coolant. The water from the cooler also comes into contact with the precoat filters which are designed to remove particulates from the coolant. The licensee reported that, in the worst case, the temperature of the coolant leaving the SFP heat exchanger would be 135°F. The licensee noted that the ion exchange resins are capable of withstanding 150°F; the precoat filters are designed for 200°F; the rest of the liquid radwaste system is designed for a temperature of 150°F.

2.6 Structural Analysis

2.6.1 High Density Racks

The high density spent fuel storage racks are Seismic Category I equipment, and are required to remain functional during and after a Safe Shutdown Earthquake (SSE). The licensee used the finite element computer program, DYNARACK, for analysis to demonstrate the structural adequacy of the TMI-1 spent fuel rack design under earthquake loading conditions. A nonlinear dynamic model consisting of inertial mass elements, spring, gap, and friction elements as defined in the program was used to simulate three dimensional dynamic behavior of the rack including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

The seismic analysis was performed utilizing the time-history method. The seismic time histories were calculated from the plant floor response spectra as described in the TMI-1 Final Safety Analysis Report (FSAR). For stress and displacement analysis three rack geometries were considered: (1) 12 ft X 15 ft rack, (2) 8 ft X 13 ft rack, and (3) 8 ft X 12 ft rack. Each rack was considered fully loaded, partially loaded, and almost empty with three different coefficients of friction between the rack and the pool floor (μ=0.2, 0.5 and 0.8) to identify the worst case response for rack movement and for rack member stresses and strains.

The calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with allowable stresses specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. Table 6.5 of the licensee's Noveber 14, 1990 submittal presents the stress factors for various rack geometries, friction and loading configurations, where the stress factor is defined as the ratio of the calculated stress with respect to the allowable stress of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The results show that the stress factor varies from the 0.004 (minimum) to 0.41 (maximum), and most stress factors are below 0.10 indicating that the induced stresses in the rack due to the postulated loading conditions are very small when they are compared to the allowable stresses of the ASME Code.

Table 6.6 of the licensee's November 14, 1990, submittal shows the calculated horizontal displacements at the top and baseplate levels of the rack for different rack configurations described above. The displacements at the baseplate level are almost negligible (less than 0.01 inch) and the displacements at the top level vary from 0.02 inch (minimum) to 0.13 inch (maximum). These computed rack displacements show that rack-to-rack impacts and rack-to-wall impacts would not occur during an SSE event.

The licensee identified a rack geometry of 12 ft X 15 ft with full load to be the most critical rack configuration for over-turning analysis. The coefficient of friction of μ=0.8 between the rack and the pool floor was used. The horizontal displacement of 0.15 inch was calculated showing that the potential for overturning is minimal, if any.

Although the licensee pre..ted the structural design adequacy of the spent fuel rack by showing small induced stresses and displacements under SSE loading conditions, the staff made an independent assessment of safety margin against overturning and sliding of the rack in order to supplement the findings obtained from the DYNARACK analysis. The assessment is based on the principle of energy conservation whereby the kinetic energy resulting from the maximum velocity of the rack induced by an SSE loading is equated to the potential energy that is needed to raise the rack to a position where the center of gravity of the rack is about to move beyond the line connecting the two supporting legs of the rack. A conservative factor of safety is defined as the ratio of the potential energy needed to raise the rack to the point of tipping over with respect to the kinetic energy imparted to the rack by the SSE. The hydrodynamic effect is not considered in the analysis and the energy corresponding to the work done on the rack by buoyancy force is considered as a negative destabilizing factor.

The staff chose a rack geometry of 7 ft X 13 ft for overturning analysis since this geometry has a narrower width and would be more critical than the geometries used by the licensee. The factor of safety of 8.5 was calculated. This calculated factor of safety is larger than 1.5, provided in the SRP Section 3.8.5, and indicates that the overturning of the rack would not occur under an SSE loading condition.

The horizontal movement of the rack was calculated. A small coefficient of friction of 0.2 was used in order to allow a larger movement. The horizontal sliding displacement of the rack was calculated by dividing the kinetic energy imparted by the SSE to the rack by the friction force developed at the leg supports. Both buoyant and fluid resisting forces were considered in the analysis. Approximately 1.0 inch horizontal sliding movement of the rack was calculated. Although the calculated horizontal displacement of 1.0 inch is much larger than the horizontal displacement of the DYNARACK analysis, the calculated displacement is still smaller than the proposed minimum gap of 2.0 inches between racks and rack-to-wall at TMI-1, therefore, rack-to-rack impacts and rack-to-wall impacts would not occur during a SSE event.

Based on (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction from 0.2 to 0.8, different geometries and loading conditions of the rack), (2) the conservatism incorporated in the analysis by neglecting the hydrodynamic effects between racks, (3) large factor of safety of the induced stresses and displacements of the rack when they are compared to the allowable stresses provided in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, and (4) the staff's independent assessment based on simplistic but conservative assumptions, the staff concludes that the rack modules will maintain their functionality and structural integrity under postulated loading conditions and is acceptable.

However, it is quite likely that the racks will move during or after seismic events. Therefore, the licensee is required to institute a surveillance program that inspects and maintains rack gaps after an earthquake equivalent to or larger than an Operating Basis Earthquake (OBE), if any occurs. The licensee should assure that the racks are in the required positions after seismic events.

2.6.2 Spent Fuel Storage Pool

The spent fuel pool structure is a reinforced concrete structure and is designed as a Seismic Category I Structure. The dimension of the TMI-1 pool structure is approximately 24 ft. wide and 63 ft. long with 6 ft. thick reinforced concrete slab. The internal surface of the pool structure is lined with stainless steel to ensure water tight integrity.

The pool structure was analyzed by using the finite element computer program, ANSYS, to demonstrate the adequacy of the pool structure under fully loaded high density fuel racks with all storage locations occupied by fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in SRP Section 3.8.4.

The licensee identified the critical regions of the fuel pool slab and wall sections adjoining the pool slab. The ANSYS analysis calculated both moment and shear load demands at the critical regions, and compared the demands to the pool structure capacities as defined per the requirements of the American Concrete Institute (ACI) Code.

Tables 8.1 and 8.2 in the licensee's submittal of November 14, 1990, show the factors of safety for bending moments and for shear forces, respectively, at critical regions of the pool structure. The factors of safety vary from 1.04 to 2.28 for bending moments and from 1.25 to 4.13 for shear forces at different critical regions, and the factors of safety are acceptable.

In order to demonstrate the integrity of the pool liner, the liner was subjected to in-plane strains due to rack foot movements. The calculated cumulative damage factor was much smaller than the ASME Code limit of one and is therefore acceptable.

In view of the calculated factors of safety, the staff concludes that the licensee's structural analysis demonstrates the adequacy and integrity of the SFP structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the SFP design as presented is acceptable.

2.6.3 Fuel Handling Accident

Accidents evaluated by the licensee included (1) a fuel assembly being dropped from the height of 26" above a storage location and impacts the base of the module, (2) a fuel assembly being dropped from the height of 26" above the rack and hitting the top of the rack and (3) the same as (2) above except that the fuel assembly is assumed to be dropped in an inclined manner on the top of the rack.

The analysis results of Drop Case (1) above showed that the load transmitted to the liner through the support is properly distributed through the bearing pads located on the liner such that the liner is not damaged by impact. The analysis results of Drop Case (2) showed that the maximum local stress of the rack at the elevation of the top of the fuel region is less than material yield point, and no permanent deformation will be incurred. The analysis results of Drop Case (3) showed that the results of Drop Case (2) above bound the results of Drop Case (3) and no permanent deformation will be incurred.

The staff has reviewed the licensee's analysis results as submitted and concurs with its findings.

3.0 SUMMARY

Based on the review described above, the staff finds the proposed modifications to the TMI-1 spent fuel pool storage racks are acceptable and meet the requirements of General Design Criteria 61 and 62 for fuel storage and handling. The staff concludes that fuel from TMI-1 may be safely stored in Region 1 of SFP A provided that the U-235 enrichment does not exceed 4.6 w/o. Any of these fuel assemblies may also be stored in Region 2 of SFP A provided it meets the burnup and enrichment limits specified in Figure 5-4 of the TS. The surveillance program proposed by the licensee would reveal any deteriorations in the neutron absorbing capability of Boral and if significant degradation is found, the licensee would have sufficient time to take the appropriate corrective measures. The staff finds the licensee's plans to mitigate heavy loads concerns acceptable. Specifically, the heavy load handling system may be considered single-failure-proof because: (1) In lifting the heaviest rack the ratio of ultimate load capability to failure is 28.7/1; (2) For the remainder of the path from crane hook to rack the licensee has committed to employ either redundant load paths or 10/1 safety factor in compliance with the requirements of NUREG-0612; (3) The special lifting devices will be designed in compliance with ANSI N14.6-1978; and (4) The non-special handling devices will be installed and used in accordance with ANSI B30.9-1971. Paragraph (1)(b) of section 5.1.6 of NUREG-0612, which is referred to in section 2.4.3 above, recommends the use of ANSI B30.9-1971 as the criterion for these devices. In addition, the licensee plans to inspect and relubricate the FHB crane prior to the start of reracking. This includes visual inspection and provision of procedures for additional load testing of the special handling devices after any extended storage or layup period. The licensee has also committed to address other concerns, as noted above, in section 2.4.

The licensee stated that the calculation of the heat generated by the stored spent fuel, including normal and full core offloads, has been conducted in accordance with specified criteria. All fuel stored in the SFP prior to a normal or full core offload was assured to have been stored after 4 full power years in the reactor. The SFP was assumed to be filled with 1640 SFAs, the results of [illegible] refuelings.

The licensee used a calculation which included ambient heat losses and provided experimental results to show that the calculation of ambient heat losses was conservative. The ambient heat losses included both convective heat losses to the FHB atmosphere together with heat losses resulting from the evaporation of water at the pool surface. These losses were combined in a transient calculation which included the ambient losses, heat removed by the heat exchanger and heat generated by spent fuel, considering the offloaded SFAs together with the SFAs already in the pool. The maximum temperatures attained under normal refueling offload (80 SFAs) conditions varies from a high of 158.4°F, to a low of 129.7°F, as shown above in cases 1 and 2, respectively. The licensee intends to operate the SFP cooling system with two trains which will result in a maximum SFP bulk coolant temperature of 129.7°F (case 2, above). However, the recommendation of section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System" of the Standard Review Plan (NUREG-0800) specifies that under conditions of a normal offload the maximum SFP bulk coolant temperature should be no greater than 140°F. The calculated maximum bulk coolant temperature attained in the TMI-1 SFP under that condition is 158.4°F. The staff notes that the 140°F limitation was specified, in general, in order to protect parts of the SFP cleanup system, especially deionizing resins which could be sensitive to high temperatures. As seen from the discussion in section 2.5.5, above, the cleanup system, downstream of the outlet of the SFP cooling system heat exchanger, is not subject to temperatures as high as 140°F, even in the worst case.

Another concern with a high bulk SFP coolant temperature is its potential effect on the concrete constituting the bulk of the SFP. Long term temperatures in excess of 150°F could damage the concrete. This does not appear to be the case even in the event of failure of a cooling train, case 1, above. In such case the bulk coolant temperature is reduced from 158°F to 150°F in about 1½ - 2 weeks. This is not considered a problem because of the short time interval during which the SFP coolant temperature is in excess of 150°F. The temperatures calculated for a full core offload, cases 4, 5 above, with maximum bulk coolant temperatures of 156.1°F and 154.6°F, comply readily with the Standard Review Plan recommendations of "...without...boiling...," i.e., less than 212°F. Therefore, the staff finds acceptable the maximum SFP bulk coolant temperatures calculated for the TMI-1 SFP coolant pool under conditions of normal maximum (normal reload) and abnormal maximum (full core offload) conditions. The licensee assumed that loading of fuel into the SFP would start at 150 hours after shutdown. However, the present TSs permit unloading spent fuel from the reactor vessel after a 72 hour decay period. The licensee stated that tne apparent error was being corrected by the introduction of procedures which would require a decay period of 150 hours before fuel unloading would be initiated. The licensee committed to the preparation of the procedures no later than the completion of the first phase of the reracking process which is intended to place two Region I and four Region II racks in place.

Based on the review and evaluation of the GPUN's submittals and the staff's independent assessment, it is concluded that the submitted structural analysis and design of the spent fuel rack modules and the spent fuel pool structure are adequate to withstand the effects of the required loads. The analysis and design are in compliance with current licensing practice, thereby, are acceptable provided that GPUN commits to ensure that the design gaps between the racks/walls are maintained (1) during rack installation, (2) during fuel-handling operations, and (3) during and after seismic events, and institutes a surveillance program that inspects and maintains rack gaps after an earthquake equivalent to or larger than an operating-basis earthquake (OBE), if any occurs.

4.0 STATE CONSULTATION

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

5.0 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 in the Federal Register on January 13, 1992 (57 FR 1285). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

6.0 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.

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