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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 Introduction

By letter dated February 3, 1977, Metropolitan Edison Company (MEC) proposed to modify the spent fuel pool (SFP) storage arrangement for SFP "B" at the Three Mile Island Nuclear Station Unit No. 1 (TMI-1) from the design which was reviewed and approved during the operating license review and which is described in the TMI-1 Final Safety Analysis Report and Technical Specifications. The proposed modification would replace the storage racks presently approved for SFP "B", which provide storage capacity for 174 fuel assemblies, with new racks which would provide storage capacity for 496 fuel assemblies. With this modification the total storage capacity for SFP's "A" and "B" would be increased from 430 assemblies to 752 assemblies.

This modification was requested by MEC based on its projections of the nonavailability of offsite spent fuel storage or reprocessing facilities prior to filling the presently authorized storage capacity.

The new storage racks will be constructed from stainless steel and are designed to seismic Category I criteria. The new racks consist of a rectangular array of storage cells welded to lattices of structural stainless steel channel located near the top and bottom of the cells. The lattice formed from the stainless steel channels limits structural deformation and maintains a nominal center-to-center spacing of 13.625 inches between adjacent storage cells. The cells have a square cross section with a 9.12 inch I.D. and a .187 inch wall thickness. The new racks are supported by the existing "B" pool floors and walls and utilize compression type restraints with pads at the points of contact with the pool liner. The racks will be fabricated in modules consisting of 5X5, 5X4, and 8X2 cells.

We have reviewed the proposed modification and by letter dated April 8, 1977, requested additional information. This additional information was provided by MEC in a letter dated May 24, 1977, and by GPU Service Corporation (consultant to MEC) in a letter dated July 21, 1977.

Our review addressed the following considerations: criticality, fuel pool cooling, structural and mechanical considerations, material considerations, fuel handling, rack installation, occupational radiation exposure and radioactive waste treatment.

2.0 Evaluation

2.1 Criticality Analysis

In its February 3, 1977, submittal MEC states that its criticality calculations are based on fuel assemblies with fresh (i.e., unirradiated) fuel with a nominal 3.5 weight percent uranium-235 content and containing no burnable poison or control rods. MEC also states that the 3.5 percent enrichment corresponds to a fuel loading of 45.9 grams of uranium-235 per axial centimeter of fuel assembly.

NUS Corporation, a consultant to MEC, performed the criticality analyses assuming the racks to be fully loaded. For parametric calculations, NUS used their version of the LEOPARD computer program, called NUMICE, to obtain four group cross sections for PDQ-7 diffusion theory calculations. The accuracy of this method was checked by using it to calculate water-moderated, uranium lattice experiments. NUS states that the calculated neutron multiplication factors obtained from NUMICE/PDQ-7 deviated from the experimental values by an average of $\pm$ 0.009. In order to ensure that the results of these four group calculations for the storage lattice were accurate, NUS used the KENO Monte Carlo program with 123 group cross sections from the XSDRN program with the GAM-THERMOS library to check selected cases and to verify the neutron multiplication factor of the final design. This method was checked by using it to calculate critical experiments of shipping cask configurations. This series of calculations showed that this GAM-THERMOS/KENO method yielded neutron multiplication factors that are within $\pm$ 0.008 of the experimental values. However, there is an additional statistical uncertainty of $\pm$ .008 in these calculations.

The use of these computer programs gives a neutron multiplication factor of 0.89 for an infinite array of these spent fuel assemblies located in the nominal storage lattice, which is assumed to be at a temperature of 20°C, with no soluble poison present.

Because the rack design allows a free space of 0.3 inches between a centered fuel assembly and each of the container walls, it would be possible for assemblies to be located off center (i.e., eccentrically) in the storage containers. Eccentric loading will increase the neutron multiplication factor. Other factors that could increase the neutron multiplication factor in the spent fuel storage pool are: (1) mechanical design tolerances; (2) increased U-235 content (assumed 102% of nominal); (3) possible variations in stainless steel composition; and (4) increased water temperature. NUS calculated that all of these factors acting together could increase the neutron multiplication factor by 0.024.

MEC states that it will not be possible to inadvertently bring a transient fuel assembly up to the outside of a fully loaded rack because: (1) all of the racks will be installed before fuel storage commences; (2) after the racks are installed, there will not be any open water regions except between the racks and pool walls; and (3) a permanent barrier will be installed in each gap between the racks and the pool walls, as necessary, to prevent the insertion of an assembly.

By summing the maximum calculational deviation of 0.008 from experiment, the statistical uncertainty of 0.008 in the KENO calculations, and the 0.024 effect of the worst tolerances and conditions, NUS finds the maximum neutron multiplication factor for this storage lattice to be 0.93.

The staff has reviewed the analytical model used by NUS to perform their calculations and has concluded that the model is capable of accurately predicting the maximum neutron multiplication factor. In addition, the above results compare favorably with the results of parametric calculations made with another method for a similar fuel pool storage lattice. Accordingly, we conclude that the NUS calculations are substantially accurate.

By assuming new unirradiated fuel with no burnable or soluble poison, the NUS calculations give the maximum neutron multiplication factor that could be obtained throughout the core life of the nominal fuel assembly. This includes the effect of the plutonium which is generated during the fuel cycle. Therefore, we find the maximum neutron multiplication factor in the pool to be 0.93.

To conform with the assumptions in the criticality analysis, MEC has agreed that the station's Technical Specifications should be modified to prohibit the storage of fuel assemblies that contain more than 46.8 grams of uranium-235 per axial centimeter of assembly. This corresponds to 102% of the nominal U-235 loading and is considered in the calculations cited above.

We find that when any number of fuel assemblies, which have no more than 46.8 grams of uranium-235 per axial centimeter of fuel assembly, are loaded into the proposed racks, the neutron multiplication factor will be less than 0.93. Since this factor is less than our acceptance criterion of 0.95, we conclude that based on criticality considerations, the proposed design is acceptable.

2.2 Spent Fuel Cooling

MEC plans to refuel annually. This will require the replacement of about 52 of the 177 fuel assemblies in the core every year.

In its February 3, 1977 submittal MEC assumed a 150 hour time interval after the reactor is shutdown prior to moving fuel during the annual refueling and during any full core off-loading into the spent fuel pool. For this cooling time, MEC stated that the heat load on the SFP cooling system for any annual refueling will not exceed 9.7×10^6 BTU/hr (2.8 MWt) and that the heat load for the full core off-load, which fills the capacity of the racks, will not exceed 27.7×10^6 BTU/hr (7.5 MWt). MEC stated that these heat loads were calculated with the ORIGEN point depletion program which was developed at the Oak Ridge National Laboratory.

In Section 9 of the Three Mile Island FSAR, MEC stated that the SFP cooling system consists of two pumps, each rated for a flow of 1000 gallons per minute, and two heat exchangers each rated for removing 6.0×10^6 BTU/hr. At these rated conditions, the spent fuel cooling system will reduce the SFP outlet water temperature by 12°F. In addition, MEC stated in its February 3, 1977 submittal that the seismic Category I Decay Heat Removal System (DHRS) is connected to the SFP cooling system so that it can be used to cool the pool during reactor shutdown periods when there is an excess core cooling capability.

In its February 3, 1977 submittal, MEC states that the SFP outlet water temperature will be maintained at or below 135°F during any annual refueling and at or below 147°F during a full core off-load.

By using the conservative method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, we find that about ten days of cooling, rather than MEC's 150 hours (6.25 days), would be required for the heat loads to decrease to those stated in MEC's February 3, 1977 submittal. However, we find that MEC's 147°F value for the maximum fuel pool outlet water temperature to be sufficiently conservative for the rated flow rates, and that the 147°F outlet water temperature will not be exceeded even with only 150 hours of cooling time.

If, shortly after placing the 52 fuel assemblies from an annual refueling in the SFP, one of the two SFP cooling pumps were to fail, the fuel pool outlet water temperature would not exceed 147°F. If both pumps were to fail, the excess capacity of the seismic Category I DHRS could be used to keep the outlet water temperature below 147°F, as long as the reactor was shutdown. By the time the reactor is started up after a refueling operation, only one SFP cooling pump will be needed to maintain the outlet water temperature below the stated 135°F; so the other pump will provide for redundancy.

When a full core is off-loaded into the fuel pool, the DHRS, which is designed to engineered safety feature criteria and seismic Category I criteria, will be available for cooling the SFP if it is needed. We find that a single failure in this system will not cause the SFP outlet water temperature to increase above 147°F.

We therefore conclude that the present cooling capacity in TMI-1 will be sufficient to accommodate the incremental heat load that will be added by the proposed modifications. We also find that this incremental heat load will not alter the safety considerations of SFP cooling from that which we previously reviewed and found to be acceptable.

2.3 Installation of Racks and Fuel Handling

MEC states that the proposed fuel rack modifications will be made in a dry, empty pool which has not previously contained spent fuel assemblies. It is further stated that the installation will not require movement of the new racks over the other SFP or over the storage area for new fuel.

Since there will be no fuel assemblies in SFP "B" while it is being modified, it will not be possible for an accident in this pool to result in any increased neutron multiplication factor.

The NRC staff has underway a generic review of load handling operations in the vicinity of SFPs to determine, among other considerations, the likelihood of a heavy load impacting fuel in the pool. As an interim measure pending completion of this review and to facilitate installation of the modified fuel storage racks, MEC has agreed to amendment of the TMI-1 Technical Specifications to provide administrative limits on the handling of loads weighing in excess of 3000 pounds. These limits have been selected to prohibit handling such loads over irradiated fuel or in such a manner that a dropped load which tipped over could damage spent fuel. The limits also require that such loads be handled at the minimum practicable height.

MEC has also agreed to an amendment of the TMI-1 Technical Specifications which would prohibit the presence of the Spent Fuel Cask in the Unit 1 Fuel Handling Building pending completion of the review of load handling operations.

We conclude that with these additional limitations, as set forth in the amended Technical Specifications, installation of the modified racks will not significantly affect the probability or consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory.

2.4 Structural and Mechanical

The new spent fuel storage rack designs are designated Seismic Category I and were reviewed for the following in accordance with the applicable portions of Sections 3.7 and 3.8 of the Standard Review Plan: structural design and analysis procedures for all loads including seismic and impact loadings; supporting arrangements for the racks including their restraints; loading combinations and structural acceptance criteria and quality control for design, fabrication and installation. Seismic analyses of the fuel storage racks were performed using a response spectrum modal dynamic analysis, enveloped over the elevation changes, in the two horizontal directions and a static seismic analysis in the vertical direction in accordance with

Section 3.7 of the Standard Review Plan. The modal responses for each horizontal direction and the combination of each of the independent direction results were arrived at in accordance with Regulatory Guide 1.92. The effective mass of the water and the fuel-cell interaction were also included in the seismic analyses. The existing pool structure was analyzed, using a finite element model, for the increased loading conditions imposed by the new high density storage racks and all loadings and load combinations were in accordance with Section 3.8.4 of the Standard Review Plan. Welding is to be performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code.

Based on the above, we find that the analysis, design, fabrication, and installation of the proposed racks are in accordance with accepted criteria, and are in conformance with the rules of ASME Boiler and Pressure Vessel Code and AISC "Specification for Design, Fabrication and Erection of Structural Steel for Buildings" including supplements 1, 2, and 3.

Accordingly, we conclude that the effect of the additional loads imposed on the existing pool structure by the proposed modification are within acceptable limits and therefore that the proposed modification is acceptable with respect to structural and mechanical considerations.

2.5 Material

The SFP racks, their associated hardware, the seismic restraints, and the pool liner are all constructed of stainless steel. Based on our review and operating experience to date, we conclude that, considering the pool temperature and the quality of the demineralized pool water, and taking no credit for inservice inspection, there is reasonable assurance that no significant corrosion of the racks, fuel cladding or pool liner will occur

over the lifetime of the plant. This issue, however, is under generic review by the NRC Staff. If the future results of this investigation indicate that additional protective measures are needed, we will at that time require implementation of appropriate measures.

2.6 Occupational Radiation Exposure

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

2.7 Radioactive Waste Treatment

The station contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems are evaluated in the TMI-1 Safety Evaluation Report (SER) dated July 1973. There will be no change in the waste treatment systems described in Section 11.0 of the SER and no change in the conclusions of the evaluation of these systems in Section 11.0 of the SER because of the proposed modifications.

3.0 Technical Specifications

By letter dated February 3, 1977, MEC proposed an amendment to the TMI-1 Technical Specifications, Section 5.4.2.d, which would revise the description of the facility's design features to reflect the proposed increased spent fuel storage capacity. During our review we found it necessary to include three additional provisions to the Technical Specifications. These

were: (1) a limit of 46.8 grams per axial centimeter of fuel element stored in the SFP, (2) limits on handling loads weighing in excess of 3000 pounds, and (3) a prohibition on the presence of spent fuel handling casks in the TMI-1 Fuel Handling Building pending completion of our review of load handling operations in that building. These additional revisions have been discussed with and accepted by MEC.

4.0 Summary

Our evaluation supports the conclusion that the proposed modification to the SFP at TMI-1 is acceptable because:

(1) The physical design of the new storage racks will preclude criticality for any credible moderating condition with the limits to be stated in the Technical Specifications.

(2) The SFP cooling system has adequate cooling capacity.

(3) No shielded cask movement will be permitted within the Fuel Storage Building prior to the completion of the cask drop analysis review and no movement of loads in excess of 3000 pounds will be allowed over or near irradiated fuel assemblies in the SFP's.

(4) The structural design and the materials of construction are adequate to function normally for the duration of the plant lifetime and to withstand the seismic loading of the design basis earthquake.

(5) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

(6) The installation and use of the new fuel racks does not alter the probability or consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 19, 1977