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December 12, 1991
C311-91-2146

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Response to Request for Additional Information
Regarding Spent Fuel Pool Rerack

Enclosed is the GPU Nuclear response to a request for additional information contained in the NRC letter dated November 1, 1991, regarding rerack of the spent fuel pool.

If any additional information is required please advise.

Sincerely,

T. G. Broughton

T. G. Broughton
Vice President and Director, TMI-1

DJD/amk

Enclosure: TMI-1 Response to NRC Request for Additional Information Regarding Spent Fuel Pool Reracking

cc: TMI Senior Resident Inspector
Region I Administrator
TMI-1 Senior Project Manager

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ENCLOSURE

TMI-1 Response to NRC Request for Additional Information
Regarding Spent Fuel Pool Rerack

Question 450.1

With respect to the environmental impacts of transportation of high burnup fuels, the staff position is that 10 CFR 51.52(b) calls for a detailed analysis of the environmental effects of transportation of fuel and waste for reactors using fuels exceeding 4% enrichment and/or 33,000 MwD/T burnup. Since your amendment request would permit storage of fuel which substantially exceed these values, GPU Nuclear must either adopt the staff's assessment of the environmental effects of transportation (53 FR 30335) with a statement that it is in fact properly applicable to TMI-1 and its fuel use or the licensee should provide its own statement under 51.52(b).

RESPONSE

TMI-1 Technical Specification Section 5.3.1.6 restricts reload fuel assemblies and rods to a maximum enrichment of 4.3 weight percent of U-235. Technical Specification Section 5.4.2.f restricts fuel in the storage pool to less than or equal to 57.8 grams of U-235 per axial centimeter of fuel assembly, which corresponds to an enrichment of 4.3 weight percent U-235. Enrichment upgrade to 4.3 weight percent U-235 was approved in Technical Specification Amendment No. 138 issued April 25, 1988. The NRC generic assessment (53 FR 30355) indicates that the environmental impact of extended irradiation up to 60 GWD/MT and increased enrichment up to 5 weight percent are bounded by the impacts reported in Table S-4 of 10 CFR Part 51. This generic assessment is applicable to TMI-1; therefore 10 CFR 51.52(b) or Table S-4 have not been separately addressed.

The proposed Technical Specification change to accommodate reracking of Pool A does not change the above restrictions. The new high density storage racks have been designed and analyzed assuming a higher enrichment in order to provide additional margins.

QUESTION 450.2

In evaluating the environmental impacts of the use of extended burnup fuel, the staff position (53 FR 30355) is that the calculated iodine-131 gap release fraction is 20% greater than the Regulatory Guide 1.25 assumed value of 0.10. The licensee should provide its analysis, using assumptions from Regulatory Guide 1.25, to demonstrate that offsite radiological consequences from fuel handling accidents are within staff acceptance criteria (i.e., "well within" the guideline values of 10 CFR 100).

RESPONSE

The offsite radiological consequences from fuel handling accidents in the Fuel Handling Building and the Reactor Building are described in TMI-1 FSAR Section 14.2.2.1. The postulated fuel handling accident in the Reactor Building is the bounding analysis and has been analyzed with more conservative assumptions using Regulatory Guide 1.25. As tabulated in FSAR Table 14.2-5, the radiological consequences are well within the guidelines of 10 CFR 100.

The Licensing Report for Pool A Reracking, TMI-1 (Holtec Report HI-89407), Section 9.1.1 describes the fuel handling accident analysis which has been made for fuel of 4.6% wt. initial enrichment burned to 60,000 MWD/MTU, 2568 MWT reactor power, and using the conservative methods and assumptions of Regulatory Guide 1.25. Since the off-site radiological consequences are dominated by the short-lived radionuclides, the calculated doses do not differ appreciably from those of previous evaluations and are nearly independent of enrichment or burnup. Results are tabulated in Section 9.1.2 of the Licensing Report and confirm that the doses are essentially the same as those previously reviewed and accepted.

QUESTION 450.3

In LER 91-014-00, Georgia Power identified a discrepancy between Unit 2 Technical Specification (TS) 3.9.10 and the design of the High Density Fuel Storage System (HDFSS) racks in the Hatch Unit 2 spent fuel pool. Essentially, this deficiency occurred as a result of a less-than-adequate design change safety evaluation in that the design change did not address compliance with T.S. 3.9.10, which requires that 23 feet of water be maintained over the top of irradiated fuel assemblies seated in the storage racks. Verify that the planned rack installation at TMI-1 will permit the maintenance of a minimum of 23 feet of water above irradiated fuel assemblies seated in the storage racks.

RESPONSE

The TMI-1 Technical Specifications do not address minimum water level in the spent fuel pools. As stated in TMI-1 FSAR Section 9.7.2.3, the existing fuel storage racks provide a minimum of 23 feet of water shielding over stored assemblies. The TMI-1 spent fuel pool water level is maintained at a depth of approximately 39 feet. The top of a fuel assembly stored in the new racks will be approximately 172 inches above the bottom of the pool liner resulting in a depth of 24.7 feet of water shielding. Therefore, the new high density storage racks will continue to maintain a minimum of 23 feet of water shielding over stored irradiated assemblies seated in the storage racks.

QUESTION 450.4

The calculated short-term (accident) diffusion estimate utilized by the staff is discussed in Section 2.3.4 of the staff's Safety Evaluation Report related to operations of the Three Mile Island Unit 1 facility. The licensee should reanalyze this accident using the currently licensed thermal power level of 2568 megawatts, and the assumptions utilized by the staff in the SER related to Unit 1 licensing.

RESPONSE

The more conservative short-term accident diffusion estimate discussed in the NRC Safety Evaluation Report Sections 2.3.4 and 15.0, was utilized by NRC to perform an independent conservative evaluation of the loss-of-coolant and fuel handling accidents at the time of original plant licensing. As stated in the Pool A rerack Licensing Report (HI-89407) the fuel handling accident analysis for the new high density storage racks assumes a short-term $X/Q = 6.8 \times 10^{-4}$ sec/m^3 and a core power level of 2568 MWT. This short term diffusion value is the current licensing basis for TMI-1 (FSAR Section 14.2.2.1) and is based on a two year period of onsite meteorological data (7-1-76 to 6-30-78) as identified in Met-Ed letter to NRC dated May 8, 1979.

QUESTION 450.5

Table 14.2-5 of the TMI-1 FSAR presents information related to radioactive releases for the postulated fuel handling accident - Reg. Guide 1.25 Analysis (in the reactor building). Provide your analysis of the radiological consequences of a fuel handling accident in the reactor building. Your analysis should use staff approved assumptions, criteria, and methodology as set forth in Regulatory Guide 1.25 and Standard Review Plan 15.7.4.

RESPONSE

TMI-1 FSAR Section 14.2.2.1 describes the existing Regulatory Guide 1.25 analysis of the postulated fuel handling accident in the reactor building. The fuel handling accident in the reactor building has been evaluated for the maximum core thermal power and fuel enrichment currently allowed by Technical Specifications. The proposed amendment request for reracking of Spent Fuel Pool A with new high density storage racks is unrelated to this postulated accident scenario since these assumptions are not being affected. The design basis accident analyses are reevaluated for each reload core design to ensure that the existing design and licensing basis remains bounding.

QUESTION 450.6

If fuel handling operations inside containment occur when the containment is open to the environment, describe the measures provided for prompt radiation detection by use of radiation monitors and for automatic isolation of the reactor building.

RESPONSE

As described in TMI-1 FSAR Section 14.2.2.1, a Reactor Building purge is assumed to be in progress during fuel handling operations inside containment. The gas channel of radiation monitor RM-A9 is provided with an interlock to close the purge valves on high radiation. The interlock is required to be verified operable within one week prior to refueling operations by Technical Specification 3.8.9. Since the radiation monitor and interlock are not single failure proof, the FSAR analysis assumes an instantaneous release through the purge exhaust filters. The calculated dose rates are well within the limits of 10 CFR 100.