NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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DOCKET NUMBER: 50-289

FILE NUMBER:

TO: Mr. R. W. Reid

FROM: Metropolitan Edison Company, Reading, Pa., J. G. Herbein

DATE OF DOCUMENT: 5/24/77

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DESCRIPTION

(1-P)

ENCLOSURE

Consists of requested additional information regarding their tech spec change request No. 47 to permit an increase in the storage capacity of the TMI-1 spent fuel pool........

(9-P)

PLANT NAME:

Three Mile Island Unit No. 1

RJL

FOR ACTION/INFORMATION

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May 24, 1977
GQL 0703

Director of Nuclear Reactor Regulation
Attn: R. W. Reid, Chief
Operating Reactor Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Docket No. 50-289
Operating License No. DPR-50

Attached please find our response to your request for additional information regarding our Technical Specification Change Request No. 47 to permit an increase in the storage capacity of the TMI-1 spent fuel pool. Feel free to call Mr. J. M. Cajigas (Ext. 164) should you have any questions regarding this matter.

Sincerely,

JG Herbein

J. G. Herbein
Vice President



JGH:JMC:kl
Attachment

cc: Ms. Margaret Reilly
Chief Division of Reactor Review
PaDER
Fulton Building
Harrisburg, PA 17120

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THREE MILE ISLAND NUCLEAR STATION UNIT 1
SPENT FUEL POOL MODIFICATION
REQUEST FOR ADDITIONAL INFORMATION

QUESTION #1:

Provide a detailed summary of the stress margins due to the increased loading of the fuel pool walls and floor for the critical load combinations. Include a discussion of the possibility of shear failures in the areas of contact of the rack supports with the floor and walls. Compare numerically these results to those for the previous rack structure.

RESPONSE:

Since the spent fuel pool was reanalyzed using the strength design method instead of the working stress design method, our answers are in the form of required section capacity versus existing section capacity instead of stress margins. The loading combinations used are in accordance with U. S. NRC Standard Review Plan 3.8.4 as follows:

a. $U = 1.4D + 1.4F + 1.7L$

b. $U = 1.4D + 1.4F + 1.7L + 1.9E$

c. $U = 0.75\ (1.4D + 1.4F + 1.7L + 1.7\ T_o)$

d. $U = 0.75\ (1.4D + 1.4F + 1.7L + 1.7\ T_o + 1.9E)$

e. $U = 1.2D + 1.9E + 1.2F$

f. $U = 0.9D + 1.4F$

g. $U = D + L + T_o' + E' + F$

h. $U = D + L + T_o' + I_a + F$

where:

U = section strength required to resist design loads based on the strength design methods described in ACI 318-71

D = dead loads including permanent equipment

L = live loads including movable equipment

F = hydrostatic loads

$T_o$ = loads generated by temperature with full capacity of pool cooling system operable

$T_o'$ = loads generated by temperatures resulting from partial failure of pool cooling system

E = loads due to OBE with maximum ground acceleration of 0.06g. One horizontal acceleration component combines additively with the vertical acceleration component.*

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E' = loads due to SSE with maximum ground acceleration of 0.12g. One horizontal acceleration component combines additively with the vertical acceleration component.*

$I_a$ = loads due to hypothetical aircraft transmitted to the pool from exterior walls or roof by interconnecting members.

* Seismic forces consist of the summation of the following individual loads:

1. Structural Seismic
2. Fuel Rack Seismic Loading
3. Hydrodynamic Loading

The critical sections, loading combinations required capacity, existing capacity and discussions of the fuel pool walls and floor are shown in Table 1.

Our analysis of the concrete structure of the spent fuel pools A and B is a conservative linear elastic analysis. The resultant forces and moments of all sections except local areas as shown in Table 1 are within the existing capacity.

For critical localized sections 2, 3, 4 the amount of required section capacity in excess of the existing capacity is primarily caused by the temperature effects. As indicated in Table 1 discussion, the thermal stress is self-limiting and secondary in accordance with ASME Boiler and Pressure Vessel Code. ACI Standard 359-74, Section III, Division 2, 1975 ed., Section CC-3136.4, pp. 183-194. Hence, the excessive required section capacities should not be a cause of concern.

The maximum estimated crack width of spent pool walls and floor above the bottom of the pool slab of the localized critical sections 2, 3, 4 is 0.0029 in., this compares favorably with the limiting crack width of 0.013 in. specified in Commentary on Building Code Requirements for Reinforced Concrete (ACI 318-71).

In comparison with the code permissible shear stresses (ACI 318-71) carried by the reinforced concrete, the normal shear and punching shear stresses are very small in the areas of contact of the proposed rack supports with the floor and walls. The forces due to the response of the new spent fuel storage racks are small, while the floor and wall thicknesses are relatively larger. Therefore, it is not possible to have shear failures in these areas.

1469 090

TABLE 1

| CRITICAL SECTIONS | LOADING COMBINATIONS | REQUIRED & EXISTING CAPACITY | DISCUSSIONS |
|---|---|---|---|
| 1. Localized region, bottom slab of Pool A at the south end near the middle wall extends 24'-0" north from the middle wall. | a, b, e, f | The required capacity slightly exceeds the existing capacity. | The north-south is the weak direction of the two way reinforced slab, and is slightly under capacity; the structure will maintain its integrity after local stress redistribution. |
| 2. Localized region, South wall below Elev. 305'-0", a horizontal strip of 5'-0" wide. | c, d, g, h | The required capacity exceeds the existing capacity. | Since this wall is below the spent fuel pool and the thermal stress is self limiting and secondary, the existing design is adequate. Also, the required capacity at localized critical section is caused by the abrupt change of input temperature, and it will be reduced when refined temperature gradients are applied. |
| 3. Localized region, North end of east wall at Elev. 348'-0" of Pool A . | c, d, g, h | The required capacity exceeds the existing capacity. | Again, the large thermal stress is self limiting and secondary, and with refined modeling, the high moment would be reduced; the design is adequate. |
| 4. Localized region, West wall at the junction of the bottom slab and middle wall . | c, d, g, h | The required capacity exceeds the existing capacity. | See 2 above. |

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QUESTION #2:

Provide the components of the stress value given in Table 5-2 for load combination "d" (as defined in Section 5.1.2) at grid beam location.

RESPONSE:

The components of stress at the worst grid beam location for load combination "d" are as follows:

| | Axial Stress (psi) | Bending Stress (psi) |
|---|---|---|
| Dead Load | 270 | 2270 |
| Thermal Load | 40 | 3910 |
| Impact Load | 0 | 0 |
| SSE | 10760 | 6470 |

The contribution due to impact is zero because the loads due to fuel weight acting with the cans are higher than the impact loads due to fuel striking the cans.

QUESTION #3:

Provide justification for neglicting any amplification of the seismic loads, transferred to the rack analyzed, due to the flexibility of the fuel cans in the adjacent racks.

RESPONSE:

The fuel cans in adjacent racks were assumed to be infinitely flexible in that they provided no restraint between the upper and lower grids. Modeling in this manner tended to amplify the dynamic response since the significant mode shapes are governed by grid deflection rather than can deflection. Some of this grid deflection could have been attenuated had the cans in surrounding racks been modeled.

QUESTION #4:

What has been the amount of solid wastes shipped from the plant in the last year?

RESPONSE:

Between January 1, 1976 and December 31, 1976; 15,700 $ft^3$ of solid waste was shipped from Three Mile Island Unit 1.

QUESTION #5:

On page 3-4 of your submittal of February 3, 1977, you state that it is "impossible" to predict the amount of waste generated from the precoat filter. If the volume cannot be "upperbounded," there is no basis for you or us to reach a conclusion that the volume is negligible. It is requested that you reevaluate the first paragraph of page 3-4, discussing the projected frequency of operation of the filters, the basis for their replacement, the cubic feet of powdered resin used to precoat the filters and an estimate of the volume of solid waste presently attributable to the SFP operations.

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RESPONSE:

As stated in Section 3.3 of the Environmental Impact Evaluation of the TMI-1 Fuel Rack Licensing Submittal, the "A" Spent Fuel Pool was purified by the RLWD System for a total of only 73 hours during the period March 1976 to March 1977. During this time, a full core off load into "A" Pool occurred. The precoat filter used for this purification contains 1.33 $ft^3$ of powdered resin per charge. The criteria for recharging the precoat filter is based on high differential pressure developing across the filter. Following the 73 hours of operation of purifying the spent fuel pool, the precoat filter was used for 48 hours to purify the Borated Water Storage Tank. After use on this tank, the resin was discharged, then solidified. Based on a full year of experience, the submittal stated that the amount of waste generated by the Spent Fuel Pool is negligible. To be extremely conservative, less than 10 $ft^3$ of solid waste can be attributed to the SFP operations over a year's time.

QUESTION #6:

What has been the release of radioactive noble gases and tritium from the SFP Building in the last three years? What is the expected increase in the release of radioactive noble gases and tritium from the facility due to the SFP modification?

RESPONSE:

The SFP and Auxiliary Building Ventilation exhausts are combined at TMI-1. Continuous monitoring of the combined exhaust began following plant startup. Within the sensitivity of the instrumentation, no tritium or radioactive noble gases have ever been detected. The Minimum Detectable Activity (MDA) for radioactive noble gases is approximately $5 \times 10^{-8}$ μc/cc and tritium is approximately $1 \times 10^{-8}$ μc/cc.

Similarly as discussed in Question #8, it is anticipated that the radionuclide concentrations in the fuel pool water will witness little change as a result of this modification. Evaporation rates will be the same since the fuel pool temperatures will remain essentially unchanged from original calculations. Therefore, it is anticipated that the release tritium and radioactive noble gases will remain unchanged.

QUESTION #7:

What is the weight of any material (e.g., racks) that will be removed from the SFP due to the modification? What will be done with this material?

RESPONSE:

The "B" Spent Fuel Pool presently is a dry, empty pool. There will be no material removed from this pool due to the fuel rack modification. The original fuel racks are stored in an open field next to Three Mile Island Nuclear Station Unit 1. Since the racks, weighing approximately 15,000 pounds, are uncontaminated, they are scheduled to remain in storage until a use for the aluminum is found or disposed as ordinary scrap metal.

1469 093

QUESTION #8:

Provide a discussion of increase in occupational man-rem exposure to personnel in the Spent Fuel Pool area from radionuclide concentration in the Spent Fuel Pool due to the expansion of the capacity of the pool including the following:

(a) Identify the principal radionuclides and their respective concentrations in the spent fuel pool found by gamma isotopic analysis during all operations. Identify the sample with respect to a specific operation (i.e., refueling, fuel handling, etc.).

(b) Provide an estimate of the man-rem exposure that will be received during removal of the old racks and installation of new ones.

(c) Provide an estimate of the dose rates above the spent fuel pool from the concentrations of the radionuclides identified in (a) and the concomitant occupational exposure, in annual man-rem, due to all operations associated with fuel handling in the spent fuel pool area. Describe the impact of the proposed modifications on these estimates. Include in your analysis the expected exposure from more frequent changing of the demineralizer resin and filter cartridges.

RESPONSE:

(a) The TMI-1 "A" Spent Fuel Pool has stored fuel for fourteen (14) months and has experienced two (2) yearly refuelings including a full core offload in 1976. Based on various pool samples, the following table identifies the principal radionuclides and their respective concentrations:

| Principal Radionuclide | Concentration (μc/ml) | Operation |
|---|---|---|
| Co$^{58}$ | $2.0 \times 10^{-2}$ | |
| Co$^{60}$ | $1.8 \times 10^{-4}$ | |
| Cs$^{134}$ | $2.2 \times 10^{-3}$ | |
| Cs$^{136}$ | $1.6 \times 10^{-4}$ | Refueling |
| Cs$^{137}$ | $2.0 \times 10^{-3}$ | |
| I$^{131}$ | $1.7 \times 10^{-3}$ | |
| Mn$^{54}$ | $1.2 \times 10^{-4}$ | |
| Co$^{58}$ | $1.3 \times 10^{-3}$ | |
| Co$^{60}$ | $4.0 \times 10^{-5}$ | |
| Cs$^{134}$ | $4.1 \times 10^{-4}$ | Midway Between Refuelings |
| CS$^{137}$ | $5.0 \times 10^{-4}$ | |
| Mn$^{54}$ | $2.0 \times 10^{-5}$ | |

1469 094

RESPONSE: (Continued)

(b) There will be no man-rem exposure resulting from the removal of the old racks because the old racks were removed two years ago before plant startup. An estimated exposure of 0.15 man-rem is anticipated during the installation of the new fuel racks. This estimate is based on actual fuel pool surveys and takes into account conservative rack installation requirements.

(c) Based on radiation surveys during fourteen (14) months of fuel storage and two (2) refuelings, the following table provides typical dose rates:

| Dose Rates (mR/hr) | Operation | Annual Occupational Exposure (Man-Rem) |
|---|---|---|
| 20 | Refueling | 11.5 |
| 0.4 | Reactor Operating | 1.5 |

The predominant contribution of radionuclide concentrations in the Spent Fuel Pool comes from the mixing of the pool water with primary coolant during refuleing. Therefore, it is anticipated that storing additional spent fuel will not significantly increase the radionuclide concentrations. Consequently, the total annual man-rem exposure attributed to the fuel pool will be unchanged as a result of this rack modification.

It is not anticipated that the frequency of use of the fuel pool water purification system precoat filters will increase as a result of this modification; therefore, the man-rem exposure from changing powdered resin will be unchanged.

QUESTION #9:

During the first refueling, 56 fuel asemblies were transferred into the SFP. The submittal stated that during the current refueling, 48 fuel assemblies will be replaced. The submittal infers on page 5-3 that on the average, you plan to replace 52 fuel assemblies per year. Based on your current fuel management plans, discuss the projected refueling schedules, including the number of the fuel assemblies that will be transferred into the SFP at each refueling.

1469 095

RESPONSE: The projected refueling schedule is as follows:

| Year of Refueling | Net Number of Fuel Assemblies Discharged to the Spent Fuel Pools | Total Number of Spent Fuel Assemblies in the Spent Fuel Pools |
|---|---|---|
| 1976 | 56 | 56 |
| 1977 | 48 | 104 |
| 1978 | 52 | 156 |
| 1979 | 52 | 208 |
| 1980 | 52 | 260 |
| 1981 | 52 | 312 |
| 1982 | 53 | 365 |
| 1983 | 52 | 417 |
| 1984 | 52 | 469 |
| 1985 | 52 | 521 |
| 1986 | 53 | 574 |

The refueling cycle continues in a re-occurring pattern of ... 52, 52, 52, 53 ... assemblies.

QUESTION #10:

The submittal (p. 5-3) states that the replacement cost of energy and capacity would be approximately $159 million per year. Discuss whether reserves are such that replacement power for TMI-1 would likely be available within the General Public Utilities Corporation System or from other utility systems after 1980. If TMI-1 were forced to shut down due to lack of storage space for spent fuel, discuss the source and cost of replacement power of system reserves are not expected to be adequate without TMI-1. If TMI-1 were to be shut down, there still would be certain costs associated with the facility such as interest on investment, physical protection, etc., apart from the costs associated with maintaining TMI-1 in a "shutdown" condition.

RESPONSE:

If TMI-1 were forced to shut down in 1980, the Pennsylvania, New Jersey, Maryland (PJM) Interconnection System reserves would be reduced by approximately 2% from values which range between 24% and 29% in the succeeding six (6) years. Though this will not precipitate PJM System breakdown, it will, however, lead to increased risk of conditions during which available capacity will be insufficient to meet load. PJM currently aims for a reliability index of one such occasion in ten (10) years and the planned reserves are generally able to meet this objective. A reduction of 2 percent will halve the reliability index to approximately one (1) occasion in five (5) years.

1469 096

RESPONSE: (Continued)

If TMI-1 were shut down, GPU reserves in this period would be totally inadequate to provide reliable service without assistance from PJM. As presently planned, GPU reserves in the 1980's range from a low of 16.0 percent to a high of 27.5 percent with an average of about 22 percent. Shutdown of TMI-1 would reduce these values by approximately 12, at which level reliability would be so seriouly impaired that frequent involuntary load interruptions would occur if GPU were obliged to depend solely upon its own resources. This evaluation assumes that no other nuclear units in the PJM are shut down for the same or similar reasons. If such shutdowns were to be considered, the statement of adequate PJM System reserves would have to be re-examined.

If TMI-1 were to shut down in 1980, the cost per year to maintain TMI-1 in this condition is approximately \$75 million. This cost includes:

Physical Protection Requirements
Routine Custodial Maintenance
Necessary Decontamination
Loss of Nuclear Fuel Investment
Depreciation
Federal and State Income Tax
Franchise, Property, and Other Taxes
Return on Investments

QUESTION #11:

Discuss the number of spent fuel assemblies that could be impacted in the proposed compact arrangement by the cask and associated lifting gear if the cask and lifting gear should tip and fall while in or near the spent fuel pool.

RESPONSE:

A cask drop analysis for Three Mile Island Nuclear Station Unit 1 was submitted to the Nuclear Regulatory Commission on February 14, 1976 by Metropolitan Edison Company letter GQL-0215. In that analysis, specific consideration was given to the integrity of spent fuel assemblies stored in the spent fuel pools "A & B) during handling of the cask.

It was demonstrated that the cask transfer path will be limited so that the cask will be tipped in a direction away from the "B" spent fuel pool in the event of a cask drop. Therefore, there are no assemblies that could be impacted by a dropped cask since the cask will not tip into the pool.