Docket No. 50-289

METROPOLITAN EDISON COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT 1

SPENT FUEL POOL MODIFICATION

DESCRIPTION AND SAFETY ANALYSIS

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January 1977

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1.0 INTRODUCTION

Because of the uncertainties in the future availability of fuel reprocessing facilities, Metropolitan Edison Company (Met-Ed) plans to increase the storage capacity of Three Mile Island (TMI) Nuclear Station Unit 1 spent fuel pool "B" to prevent a shortage of space for storing spent fuel. The proposed method of accomplishing this increase is to install new storage racks with fuel assembly center-to-center spacing smaller than that of the present racks. These high density racks will utilize square tubes, made of stainless steel, to maintain the required shutdown margin and support the fuel assemblies.

The original unit design assumed a viable fuel reprocessing industry in the United States by the time the unit commenced operations. Therefore, the TMI-1 "A" and "B" spent fuel pools were sized to accommodate 2-1/3 cores which was thought to be a conservative approach. The assumption made during the design stage was that the one-third of a core discharged each year would be shipped to a reprocessing facility in a timely manner. Therefore, the pools would always have the capability to accept an entire core offload. However, the ability to reprocess fuel does not and will not exist for some time. Because of this situation, the spent fuel generated as a result of reactor operation cannot be disposed of and must be stored.

When it became apparent that fuel reprocessing capability would not be available for many years, Met-Ed decided to expand the storage capacity of the "B" pool. In preparation for this expansion the original spent fuel racks were removed from the "B" pool. This reduced the storage capacity from 430 assemblies to 256 assemblies which is the capacity of the "A" pool only.

At present 56 assemblies are being stored in the "A" pool. The unit is scheduled for shutdown for refueling in March 1977. This will increase total number of assemblies in storage to 104.

This situation will preclude the unloading of the entire core (177 assemblies) should it become necessary. Met-Ed finds this condition to be unacceptable and requests the approval of the NRC to increase the storage capacity of the "B" spent fuel pool to 496 elements. This increase allows the storage of spent fuel until 1986 and allows Met-Ed to retain the capability to offload an entire core up to that time. This report discusses in detail the various design features incorporated in this modification and demonstrates that these design features will have no detrimental effect on the health and safety of the public.

2.0 GENERAL DESCRIPTION

2.1 Present Design

Three Mile Island Nuclear Station Unit 1 is a 2535-MWt PWR (B&W) with a total of 177 fuel assemblies in the core. Its spent fuel storage complex consists of two Pools, "A" and "B", connected to each other by a canal and sliding gate, and a spent fuel pool cooling system. Water in the system contains approximately 1800 ppm boron. The cooling system includes two coolant pumps, two coolers, one borated water recirculation pump, and associated piping, valves, etc. The spent fuel cask loading pit is adjacent to Pool "B", and Pool "A" is connected to the reactor building fuel transfer canal by two fuel transfer tubes.

The major equipment components of the cooling system are located on the west side of the the Fuel Handling Building, a Class I structure hardened to withstand hypothetical aircraft impact as described in the TMI-1 FSAR. Part of the cooling system piping extends into the Reactor Building and into the Auxiliary Building. Both these structures are also Class I and hardened to withstand hypothetical aircraft impact.

The cooling system is designed to maintain 135°F in the pools with a heat load based on decay heat from one-third of a fully irradiated core that has been cooled for 150 hours, the postulated normal time between shutdown and removal of fuel from the core. This can be accomplished with one pump and one cooler. After an entire core offload with an additional one-third of a core already in the pool from a refueling 100 days earlier, the pool can be maintained at 153°F by using both pumps and both coolers. The design capacity of the cooling system is 9.5×10^6 Btu/hr during a normal refueling and 28.0×10^6 Btu/hr during an Entire Core Offload condition. The worst case heat generation rate will cause the spent fuel pools to heat up at a rate of 5.2°F/hr should all cooling be lost. During this Entire Core Offload condition,

sufficient time would exist to activate the Reclaimed Water System as an additional water source or to restore service to one of the spent fuel pool cooling chains. A purification loop is provided within the Radioactive Liquid Waste Disposal system for removing fission products and other contaminants from the water. A small flow from the spent fuel cooling pumps is diverted to a radiation monitor. The spent fuel cooling system is designed so that a line rupture will not cause a serious lowering of pool water level.

The present TMI-1 fuel storage capacity consists of:

a. 253 Wet fuel locations in Pool "A"
b. 3 Wet failed fuel locations in Pool "A"
c. 63 Wet fuel locations in the Reactor Building Transfer Canal (rack temporarily removed but available for reinstallation)
d. 1 Wet failed fuel detection location in the Reactor Building Transfer Canal (temporarily removed but available for reinstallation)
e. 66 Dry new fuel locations in New Fuel Storage Pool
f. Pool "B" is now empty, but was originally designed for wet storage of 171 assemblies and 3 failed fuel assemblies.

The spent and new fuel assemblies are stored in racks in parallel rows having a center to center distance of 21.125" in both directions. Control rod assemblies requiring removal from the reactor are stored in the spent fuel assemblies.

At present, Pool "A" contains spent fuel stored in already existing racks. Pool "B", on the other hand, has never been used, contains neither water nor spent fuel racks, and is free of radioactive contaminants. The proposed modification, discussed in the next section, is for new spent fuel racks to be installed in Pool "B".

2.2 Proposed Modification

The proposed fuel rack modifications, which conform in all respects to Safety Guide 13 (USNRC RG 1.13), will involve installing high density storage racks in the empty "B" pool.

A rack assembly consists of a rectangular array of storage cells with a 13.625" center-to-center spacing. Each storage cell consists of a 9.12" I.D. square stainless steel cell having a wall thickness of 0.187". The array size of each rack was chosen to maximize use of pool space as shown in Figure 2-1. The expanded storage capacity of Pool "B" is 496 elements. The new racks contain no materials installed purely for neutron absorption capability. Reactivity calculations do consider the nuclear properties of the stainless steel cells and water but do not take credit for the 1800 ppm boron in the pool water.

The Spent Fuel Pool Cooling System will maintain the fuel pools at a maximum of 135°F during Normal Refueling with one pump and one cooler, and 147°F following an Entire Core Offload with two pumps and two coolers in operation.

As the installation will be made in a dry uncontaminated pool, no radiological problems are anticipated. The installation will not require movement of the new racks over the spent fuel in the "A" pool or over the new fuel storage area.

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2.3 Schedule for Proposed Modification

The schedule for the proposed installation of spent fuel racks is presented in Table 2-1. In order to maintain an Entire Core Offload storage capability, the racks must be available following the 1977 refueling outage that is scheduled for completion in May 1977. In order for rack procurement and construction to begin in a timely manner, initial NRC review and comments will be necessary by March 11, 1977 with final approval by May 1, 1977.

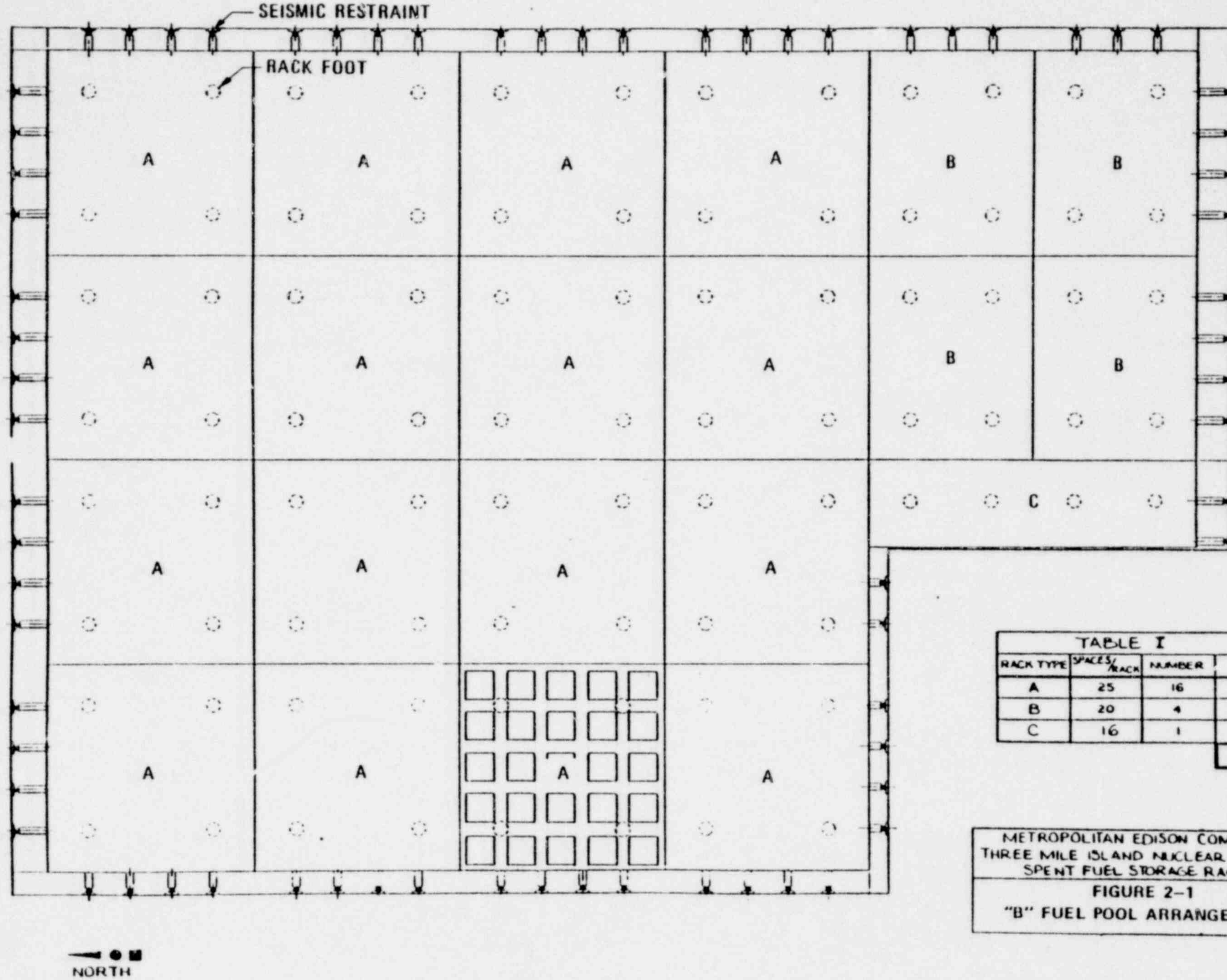

TABLE I			
RACK TYPE	SPACES/RACK	NUMBER	TOTAL
A	25	16	400
B	20	4	80
C	16	1	16
			496

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THREE MILE ISLAND NUCLEAR STATION
SPENT FUEL STORAGE RACKS

FIGURE 2–1
"B" FUEL POOL ARRANGEMENT

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TABLE 2-1

SCHEDULE FOR PROPOSED MODIFICATION

Item	Date
Submittal of Safety Analysis Report and Environmental Impact Evaluation	February 4, 1977
Initial NRC Review and Comments	March 11, 1977
Final NRC Approval	May 1, 1977
Rack Installation	July-September 1977

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3.0 MECHANICAL DESIGN

3.1 Spent Fuel Storage Cells

Each fuel assembly is stored in a stainless steel cell 13.5' long and having a square cross section with a 9.12" interior dimension. These storage cells are fabricated of 0.187" type 304 stainless steel sheet and are of an all welded construction. The cells are flared at the top to an 11" square cross section to facilitate insertion and removal of fuel assemblies. Each cell is partially closed at the bottom by welding two 0.187" stainless steel bars across the bottom of the cell on opposite sides. These bars provide two 2" wide support ledges for the fuel assembly. This method of support leaves a 45 square inch rectangular opening at the bottom of the cell. This opening allows cooling water to flow upward through the fuel assembly to provide for removal of decay heat. The cell manufacturing and rack assembly process are controlled to ensure that there will be no binding during insertion or removal of a fuel element.

3.2 Fuel Rack Assemblies

The individual cells are assembled into racks having a 13.625" pitch between adjacent cells by being welded to two lattices of structural stainless steel channels (see Figure 3.1). One lattice is located near the top of the rack and one near the bottom. Each lattice consists of two sets of continuous channels, one set laid over the other and running the length or width of the rack. The channels are welded together at the bottom support points to provide structural rigidity.

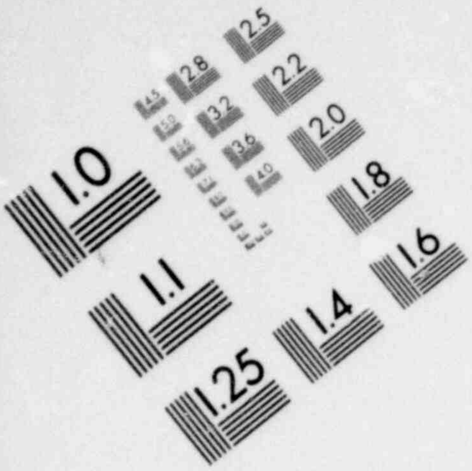
Lifting lugs are attached at four intersection points of the upper lattice of each rack. To provide structural rigidity, the overlapping channels are welded to each other under the lifting points as well as to the cells.

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



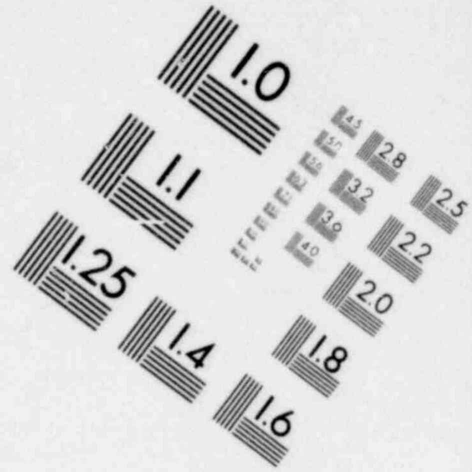
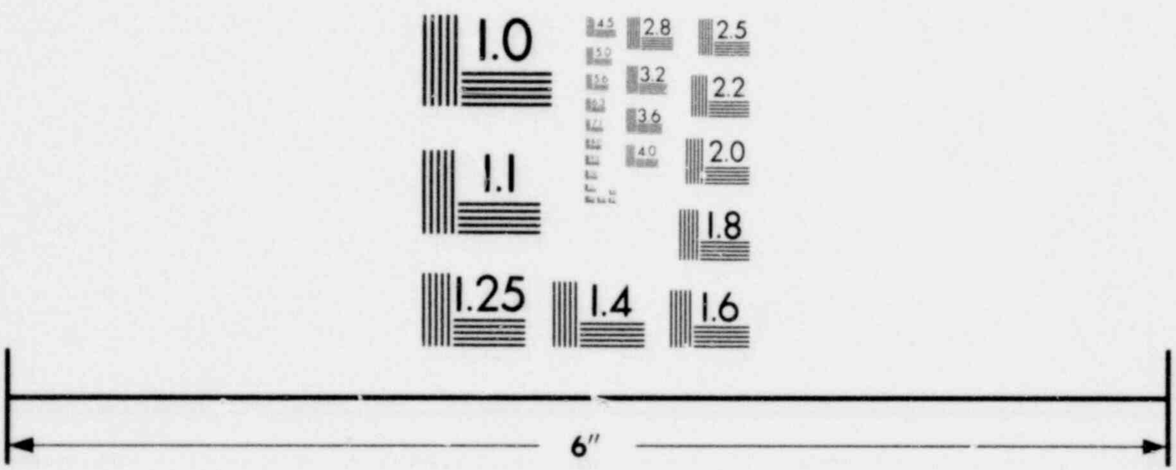
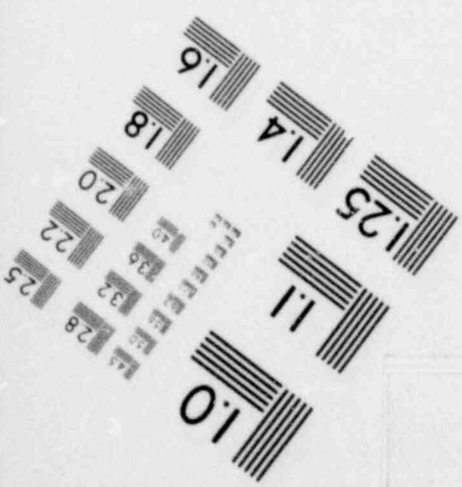


IMAGE EVALUATION TEST TARGET (MT-3)





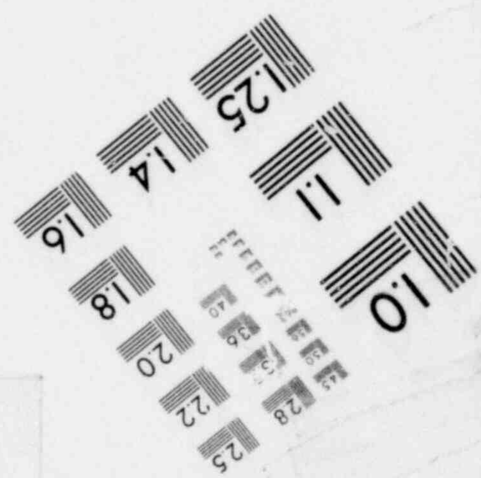


IMAGE EVALUATION
TEST TARGET (MT-3)

6″

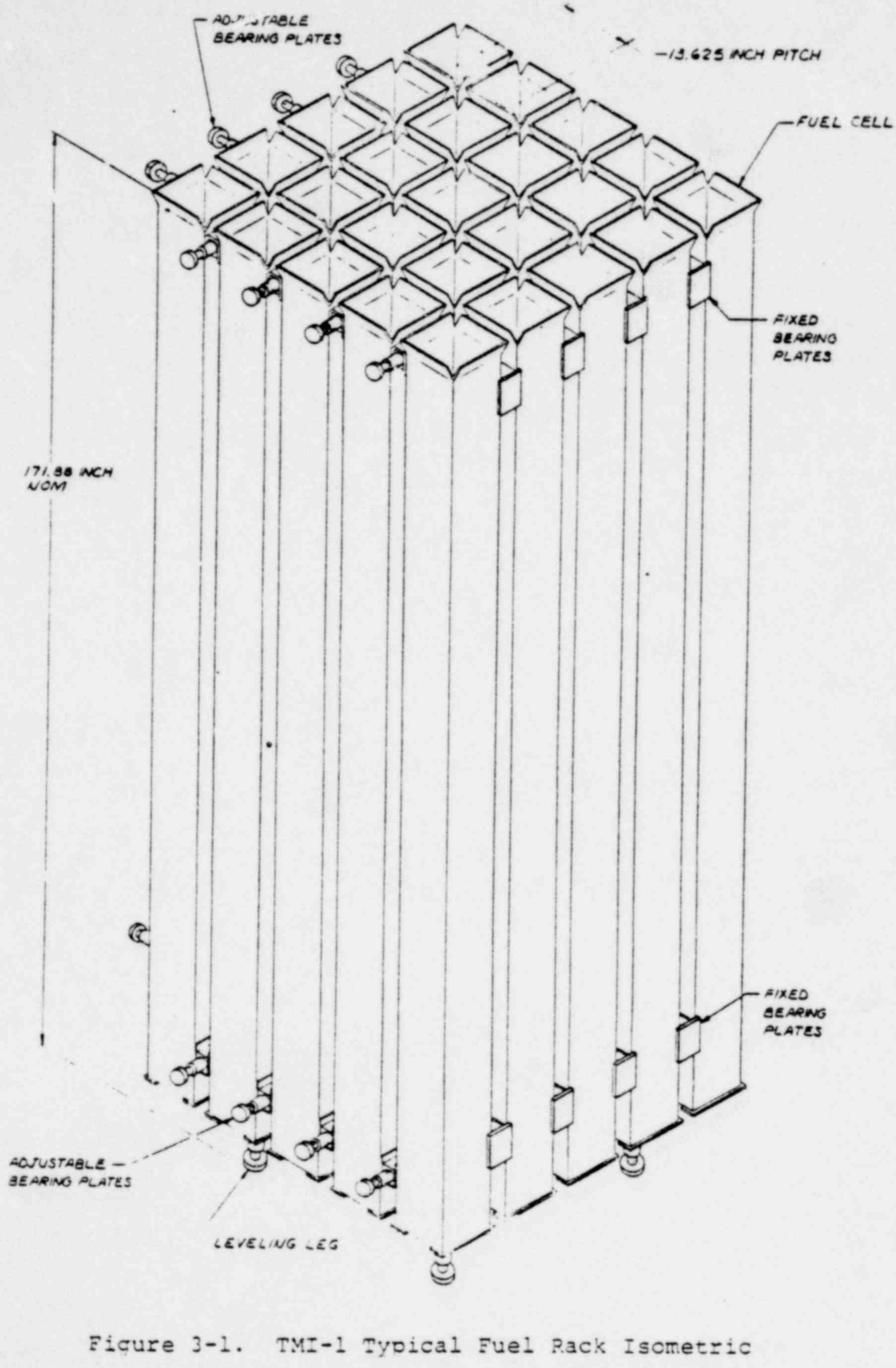


Figure 3-1. TMI-1 Typical Fuel Rack Isometric

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Each rack is supported at the bottom by four individually adjustable leveling legs. The legs are located between cells at the intersections of two channels and are welded directly to the channels. Gussets are welded between the channels and the legs to provide additional strength. The bearing pads of the support legs are sized to ensure that stresses on the pool liner and underlying concrete are within acceptable limits.

At the periphery of each rack, where one rack touches another, there is a system of fixed or adjustable bearing plates attached to the upper and lower lattice structure to transmit thermal and seismic loads between adjacent racks. Where the racks face the pool wall, compression-type seismic restraint devices are provided on both the upper and lower lattice structure. These seismic restraint devices (see Fig. 3.2) are adjustable and a gap will be provided for thermal growth of the racks resulting from expected temperature variations in the pool. The bearing pads of the seismic restraint devices are sized to ensure that wall and liner stresses are within acceptable limits.

There are three different sizes of rack, each having been chosen to maximize the storage capacity of the pool. The rack types are as follows:

Type	Array
A	5 x 5
B	5 x 4
C	8 x 2

Within each type the racks are identical except for the array size and the arrangement of fixed and adjustable restraints, which depend on the rack's specific location within the pool.

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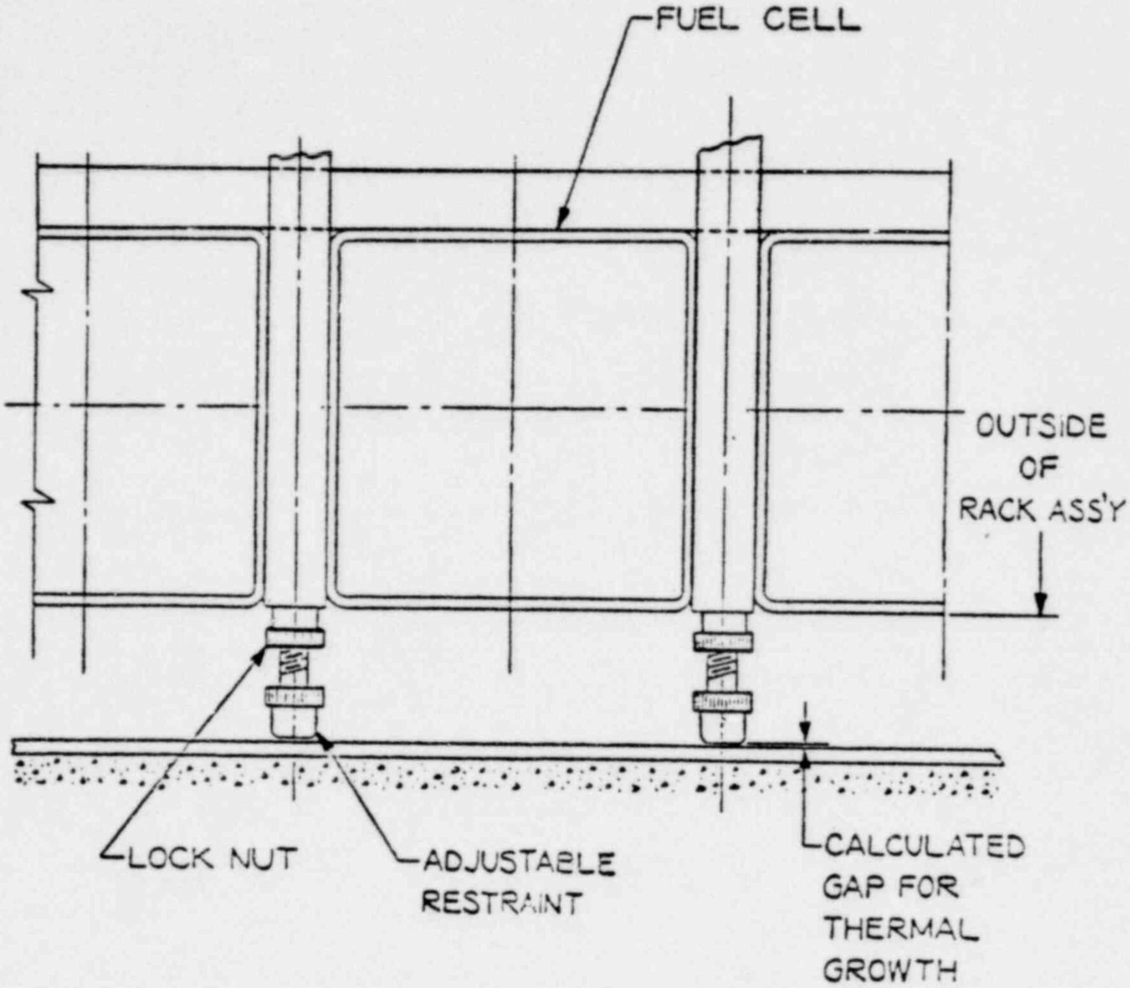


Figure 3-2. Compression Type Restraint

3.3 Codes, Standards, and Practices for Fuel Assembly Rack Design, Construction, and Assembly

The following are the codes, standards, and practices to which the fuel assembly racks will be designed, constructed, and assembled (revisions utilized are those in effect as of November 1, 1976). Other applicable codes, standards, and regulatory guides are identified elsewhere in this document.

1. Design Codes

   a. AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1969, including Supplements 1, 2 and 3.

   b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components (Tables I-7.0 and I-8.0 are used for yield values for materials of construction).

2. Material Codes

   a. ASTM Specification A-240-74a, Specification for Stainless and Heat-Resisting Chromium and Chromium-Nickel Steel Plate Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

   b. ASTM Specification A-320-74, Specification for Alloy Steel Bolting Materials for Low Temperature Service.

   c. AWS Specification A-5.9-69, Corrosion-Resisting Chromium and Chromium-Nickel Steel Welding Rods and Bare Electrodes.

   d. ASTM Specification A-276-73, Specification for Stainless and Heat-Resisting Steel Bars and Shapes.

e. AWS Specification A-5.4-69, Specification for Corrosion-Resisting Chromium and Chromium-Nickel Steel Covered Welding Electrodes.

3. Welding Codes

a. ASME Boiler and Pressure Vessel Code, Section IX-1974, Welding and Brazing Qualifications.

4. Quality Assurance, Cleanliness, and Packaging Requirements

a. 10CFR50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants

b. RG 1.28 Quality Assurance Program Requirements - Design and Construction (Safety Guide 28), 6/7/72

c. RG 1.37 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, 3/16/73

d. RG 1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, 3/16/73

e. RG 1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev. 2, 6/76

f. RG 1.88 Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev. 2, 10/76

3.4 Fuel Rack Installation Procedure

The proposed racks will be installed in Pool "B" which at present is dry and has no racks installed. The racks are however designed for either wet or dry installation. This pool has never been filled with water having any radioactive contamination and therefore the installation will involve no potential health physics problems. The installation procedures will preclude handling the racks over Pool "A" which has spent fuel in it, or over the new fuel storage area.

3.5 Material Compatibility

Because the replacement racks, their associated hardware, and the seismic restraints are of all stainless steel construction, as is the spent fuel pool liner, there is no potential for galvanic corrosion. Material compatibility between the fuel assemblies and the new storage racks is also not a problem as stainless steel has been shown to be compatible with both fuel assemblies and with borated water in the pool.

4.0 CRITICALITY CONSIDERATIONS

The racks are designed for a 13.625" center-to-center spacing between storage cells. The results of the criticality analyses are as follows:

1. The center-to-center spacing of 13.625" results in a k_∞ of 0.892 under nominal conditions.

2. The worst case situations, considering maximum variations in the position of fuel assemblies within the storage rack, variations in cell dimensions, the most reactive temperature, calculational uncertainties, and worst case accidents result in a k_∞ of 0.934 with a confidence level of 95%.

4.1 Assumptions and Method of Analysis

The referenced set of calculations were based upon the following assumptions:

1. New fuel of 3.50 wt% ^{235}U nominal average enrichment equivalent to 45.90 grams of ^{235}U per centimeter of height.

2. Water temperature of $68^{o}F$.

3. No credit taken for soluble poison.

4. Fuel racks are infinite in three dimensions.

5. Control rods and other fixed poisons are not present in the fuel assembly.

The majority of the calculations were performed with methods commonly employed in light water reactor design, i.e., four-group

diffusion theory cell calculations using PDQ-07. The cross sections for these calculations are generated with NUMICE, the NUS version of LEOPARD. This code uses the same cross section library tape and calculational techniques as LEOPARD.

Selected cases were checked and the final design multiplication factors were verified with Monte Carlo criticality calculations using KENO with 123-group cross sections. The 123-group cross section library is generated from the basic GAM-THERMOS library using XSDRN (P_3, S_8).

4.2 Results of Analysis

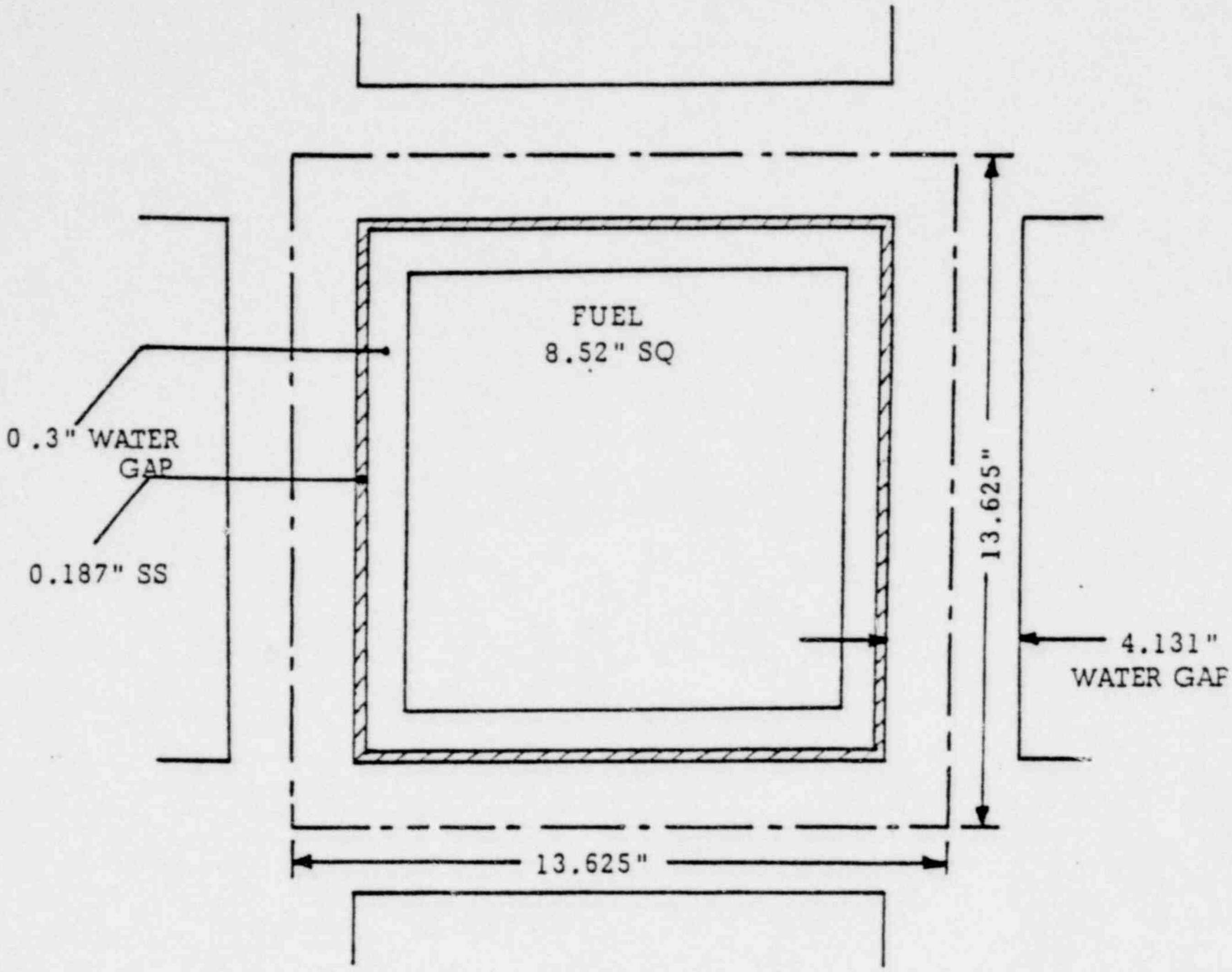
Figure 4-1 shows the geometry of the reference storage cell used in the design calculations. Section 4.3 summarizes the results of both the diffusion theory and Monte Carlo calculations. In general, the four-group PDQ diffusion calculations produce k_∞ values about 0.015 lower than the Monte Carlo calculations.

Calculational uncertainties in the use of PDQ with cross sections based on the LEOPARD library have been obtained by comparing the results of a series of benchmark calculations with critical experiments. These comparisons[1] have shown that the average difference between the calculations and experimental results was 0.009 Δk_∞.

The KENO code, using the 123-group GAM-THERMOS cross section library, has been extensively benchmarked also. For a series of ten critical experiments reported[2] the average k_{eff} as

(1) WCAP-3269-25, "Calculation of Lattice Parameters and Criticality for Uranium Water Moderated Lattices," L. E. Strawbridge, Westinghouse Electric Corporation, September 1963.

(2) "Validation of Monte Carlo Calculations of Shipping Cask Systems," L. M. Petrie and P. G. McCarty, ORNL, CONF-731101-14, 1973.



FUEL LOADING: 45.90 GM U-235/cm Height

Figure 4-1. Geometry of the Reference Storage Cell

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calculated using KENO and 123-group cross sections was 0.9914 $\pm$ 0.0020. Using the same method, NUS has performed another benchmark on one of the Yankee critical experiments[3] with Ag-In-Cd cruciform control rods banked at 26.37 cm from the bottom of the fuel. The calculated k_{eff} was 1.008 $\pm$ 0.006. On the basis of the above comparisons with criticals, a calculational uncertainty of 0.008 Δk_∞ was assigned to the KENO calculations.

Statistical analysis of the Monte Carlo results shows a standard deviation of $\pm$ 0.004, giving a 2σ uncertainty of 0.008 Δk_∞. Thus, an additional 0.008 Δk_∞ uncertainty is assigned to the KENO calculations.

The worst-case criticality condition was obtained by using the maximum tolerances for the positioning of the fuel assemblies within the storage cell as well as the relative cell-to-cell positioning and cell dimensions.

4.3 Worst-Case Analysis of Tolerances and Calculational Uncertainties

The following are the results of the KENO analysis of the worst-case tolerances and calculational uncertainties:

Nominal Conditions, k_∞ 0.892

Enrichment, 3.5 wt%
Mechanical Spacing, 13.625"
Pool Temperature, 68°F

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(3) "Yankee Critical Experiments - Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water," P. W. Davison et al., YAEC-94, April 1959, page 82.

Worst Tolerances, Δk_∞

Enrichment, 102% of nominal	0.003
Most Reactive Temperature	0.003
SS Composition	0.002
Eccentric Fuel Loading in Cell	0.007
Mechanical Design	0.009
TOTAL	0.024

Calculational Uncertainties, Δk_∞

KENO Benchmark	0.008
Statistics (2σ)	0.008
Total Calculational Uncertainties	0.016

Maximum, k_∞

Nominal, k_∞		0.892
Worst Tolerances	0.024	
Calculational Uncertainties	0.016	
	0.040	
MAXIMUM, k_∞ (without accident case)		0.932
MAXIMUM, k_∞ (with accident case - see Section 4.5)		0.934

4.4 Parametric Studies

The ba e case, as established in the preceding sections, refers to the rack design with 13.625" spacing, 3.50 wt% nominal enrichment and 68°F pool water temperature. The k_∞ of the base case is 0.892 based on the 123-group KENO calculation. Parametric studies were performed to determine the effect on k_∞ of varying the base case conditions one at a time. The results are presented below:

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1. k_∞ vs. Center-to-Center Spacing (PDQ)

	13.375"	$+0.007\ \Delta k_\infty$
(Nominal)	13.625"	(Base)
	13.875"	$-0.0067\ \Delta k_\infty$

2. k_∞ vs. Enrichment (PDQ)

	3.40 wt% U-235	$-0.0052\ \Delta k_\infty$
(Nominal)	3.50 wt% U-235	(Base)
	3.60 wt% U-235	$+0.0045\ \Delta k_\infty$

3. k_∞ vs. Cell Wall Thickness (PDQ)

	0.173"	$+0.0023\ \Delta k_\infty$
(Nominal)	0.187"	(Base)
	0.201"	$-0.0021\ \Delta k_\infty$

4. k_∞ vs. Water Temperature (PDQ)

	$40^{o}F$	$-0.0022\ \Delta k_\infty$
(Nominal)	$68^{o}F$	(Base)
	$100^{o}F$	$+0.0011\ \Delta k_\infty$
	$153^{o}F$	$+0.0019\ \Delta k_\infty$
	$212^{o}F$	$+0.0019\ \Delta k_\infty$

4.5 Fuel Handling Accident Analysis

The worst accident during spent fuel handling involves dropping a fuel assembly that would land horizontally on top of the storage racks. Inadvertent positioning of an assembly vertically next to the rack is not possible for the following two reasons. First, all the racks will be installed in the pool before fuel storage commences and thus there will not be any open water region except for the gaps between the racks and the pool walls. Second, a permanent barrier will be installed in each gap between the racks

and the pool walls, as necessary, to prevent insertion of an assembly. In any assembly drop accident the minimum distance between the active fuel and the accident assembly is greater than 12". Calculations show that the multiplication factor is not penalized more than 0.002 Δk_∞ if this separation is 5" or more. Therefore, this situation has an insignificant effect on the pool criticality.

5.0 STRUCTURAL ANALYSIS

5.1 Loads and Loading Criteria

In accordance with Regulatory Guide 1.29, the spent fuel storage racks were designated Seismic Category I. Structural integrity of the fuel racks when subjected to normal and abnormal loads, as well as the OBE and DBE, has been demonstrated with respect to the NRC Standard Review Plan Section 3.8.4. In accordance with the Review Plan, the following loads, load combinations, and structural acceptance criteria were considered:

5.1.1 Loads

a. Normal Loads

   i. Dead Loads - dead weight of rack and fuel assemblies and hydrostatic loads

   ii. Live Loads - effect of lifting empty rack during installation

   iii. Thermal Loads - uniform thermal expansion of racks due to change in average pool temperature from 70 to 147°F and a thermal gradient between adjacent storage boxes of 20°F.

b. Severe Environmental Load - Operating Basis Earthquake (OBE)

c. Extreme Environmental Load - Design Basis Earthquake (DBE)

d. Accidental drop of a spent fuel assembly from a height of 2.675' above the top of the racks, which is conservative with regard to fuel handling operations.

e. Postulated stuck fuel assembly which causes an upward force of 300 lb, or a downward force of 350 lb.

5.1.2 Load Combinations

The spent fuel storage racks were analyzed using elastic working stress design methods for the following applied loads:

a. Dead Loads Plus Live Loads
b. Dead Loads Plus Impact Loads Plus OBE
c. Dead Loads Plus Thermal Loads Plus Impact Loads Plus OBE
d. Dead Loads Plus Thermal Loads Plus Impact Loads Plus DBE (SSE)
e. Dead Loads Plus Fuel Assembly Drop
f. Dead Loads Plus Stuck Fuel Assembly

Live loads were not included in load combinations b through f, since the only live load on the rack was that due to lifting, and lifting of the racks is performed with the racks empty.

5.1.3 Structural Acceptance Criteria

The following were the strength limits for each of the above load combinations:

Load Combination	Strength Limit
a	1.0 S
b	1.0 S
c	1.5 S
d	1.6 S
e	1.6 S (except as noted below)
f	1.6 S (except as noted below)

where S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of

the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969, including Supplement Numbers 1, 2 and 3. (Supplement 3 was effective June 12, 1974.) Yields for major structural members were obtained from ASME Boiler and Pressure Vessel Code Section III. For load combinations e and f, local stresses might exceed the limits, provided there was no loss of function of the fuel rack. In addition the fuel cell was checked for buckling to ensure that no collapse occurs due to compressive loading.

5.2 Seismic Analysis

The individual fuel racks described in Section 3.0 will be of all-welded construction. The racks will rest on the floor and butt against one another at the top and bottom. At the perimeter of the pool there will be clearance between the pool wall and the upper and lower seismic restraints sufficient to allow for thermal expansion of the racks.

The seismic loading of a typical fuel rack was determined from a response spectrum modal dynamic analysis in which the stiffness of the fuel assembly was neglected. However, the mass of the fuel assemblies and an effective mass of water were considered to be uniformly distributed along the storage cells.

The response spectrum modal analysis assumed that all gaps were closed by thermal expansion. The assumption that the restraint was in contact with the wall (no gap) was necessary in order to consider the worst case of fuel/cell interaction (see discussion below). The assumption used in the analysis of the rack was consistent with this requirement and provided a basis for combining the results of the modal analysis and the fuel/cell interaction analysis. A consequence of the closed gaps is that each rack is touching the one next to it and the whole array must be modeled. The detailed modal rack in Figure 5-1 is part of an array of racks supported at the walls. The analysis considered the entire pool

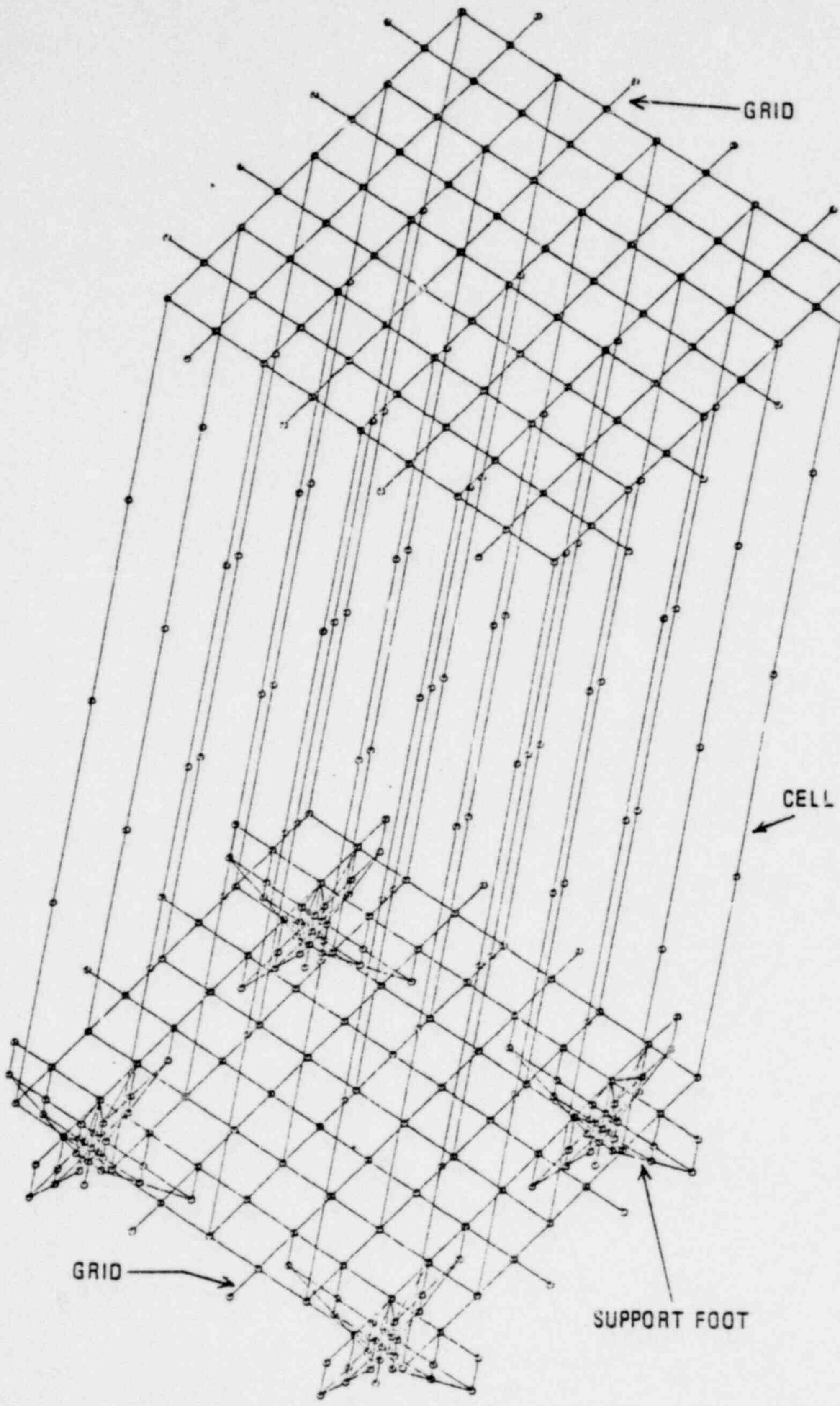

Figure 5-1 TMI 5 x 5 Rack Model

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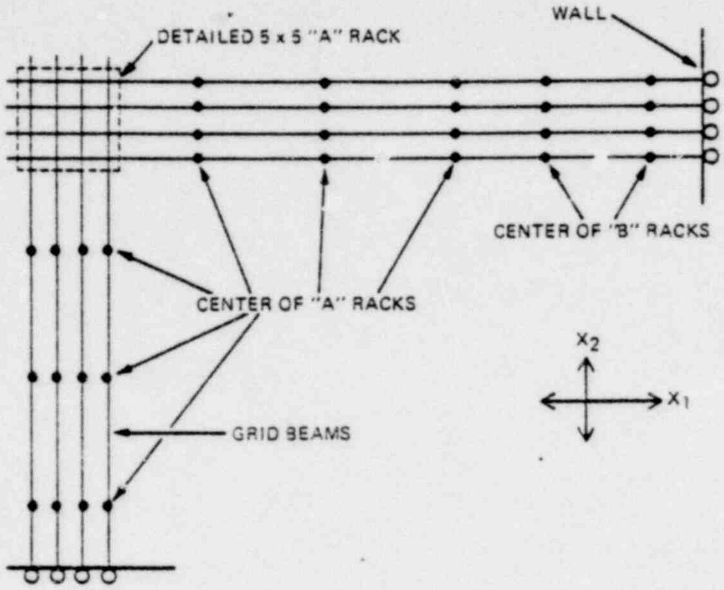

Figure 5-2 Entire Pool in Lateral Vibration (Plan View)

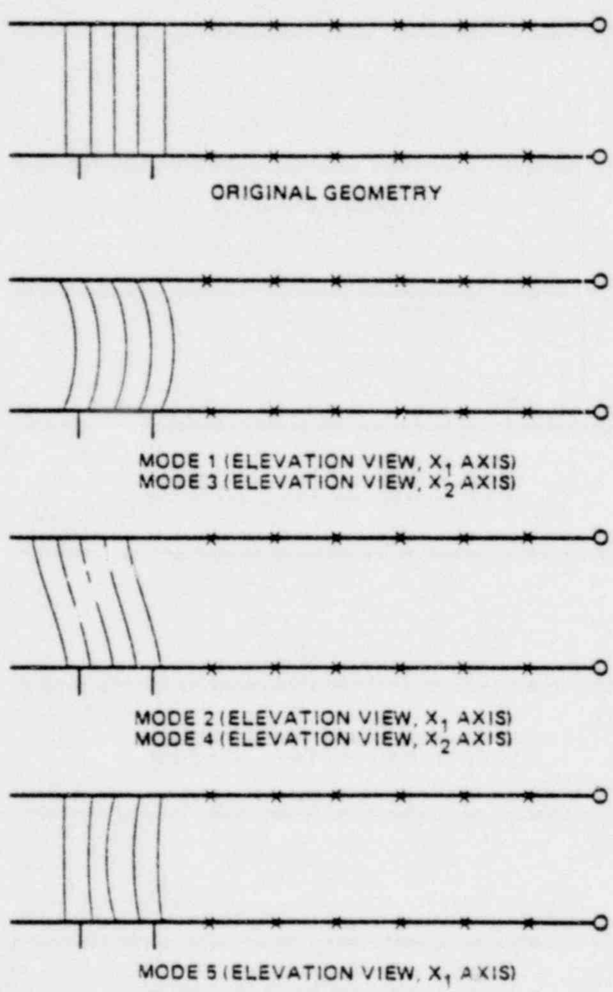

Figure 5-3 Mode Shapes

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in lateral vibration, as shown in Figure 5-2. Thus, when the rack vibrated in the X_1 or X_2 direction, the detailed 5 x 5 "A" rack portion of the model showed the cell bending and cell/grid interaction. The portion of the model with only grid beams and concentrated weights (at the positions shown as centers of "A" or "B" racks in Figure 5-2) showed the spring effect of all the racks acting together through the grids. The detailed portion was supported in the vertical direction at the feet.

The appropriate response spectra for the OBE and DBE, respectively, were employed. To obtain the appropriate response spectra, the response spectra at elevations 302.5' and 329' were linearly interpolated to obtain the maximum response spectrum which occurs at the upper seismic restraint location. This spectrum was then used at each frequency of interest. These frequency response spectra were developed using the methods reported in the TMI-1 FSAR. The damping factors of 0.5% and 2% for the OBE and DBE, respectively, utilized for the initial unit design were increased to 2% and 4% to account for the additional damping afforded submerged structures.

The STARDYNE computer program was used to perform the structural analysis of the racks. Storage racks were modeled in detail using beam and plate finite elements. The three-dimensional finite element model for the 5 x 5 rack is shown in Figure 5-1. This is considered representative of the rack design.

To determine the earthquake modal response, STARDYNE was first run to determine the natural frequencies, mode shapes, and participation factors. For the five analyzed modes the significant frequencies, participation factors, and corresponding accelerations are given in Table 5-1. Figure 5-3 provides a sketch of modes 1, 2, 3, 4, and 5. Modes 3 and 4 are similar to modes 1 and 2, respectively, but are in the X_2 direction.

TABLE 5-1

MODAL DATA

Mode Shape	Frequency (Hz)	Participation Factors: Horizontal X_1	Horizontal X_2	Vertical X_3	Horizontal Acceleration (g): OBE	DBE
1	11.5	1.56	0	0	0.85	1.27
2	13.9	0.23	0	0	0.44	0.65
3	15.6	0	1.84	0	0.32	0.51
4	19.7	0	0.17	0	0.39	0.62
5	29.1	0.95	0	0	0.20	0.34

The seismic loads were then determined using the above modal data. Since the static vertical direction had no participation below 33 Hz, a static seismic analysis was run for the vertical direction using a "g" load of 0.08g (OBE) and 0.19g (DBE). The results for the horizontal earthquakes were determined using a response spectrum analysis. Since no modes were closely spaced (as defined by USNRC RG 1.92), the results of each mode were combined with the other modes in an SRSS fashion. The results of the three directions of earthquake were then combined in an SRSS fashion, per USNRC RG 1.92.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly was assumed to be uniformly distributed along the length of each of the fuel storage cells. This assumption was conservative in that lower rack fundamental frequencies were calculated which, due to the relatively stiff rack design, result in higher seismic amplified acceleration loading on the rack. Since a gap on the order of 0.6" will exist between the sides of a fuel assembly and the cell (fuel leaning on one side of cell), the fuel will move within the cell during a seismic event. The effect of this motion, termed fuel-cell interaction,

was analyzed using the ANSYS computer program. A nonlinear dynamic analysis of a single cell and fuel assembly was performed to determine the maximum shear force and bending moment that might occur at critical sections of the cell as a result of the fuel assembly impacting the cell at maximum velocity. The cell and fuel assembly were modeled by beam finite elements and are separated by nonlinear gap elements. The cell was restrained at the upper and lower grid elevations by a spring that represents the grid stiffness. The fuel, which was assumed to be pinned at its base, was given an initial velocity relative to the cell. The impact velocity is taken to be the SRSS sum of the maximum rack velocity and the maximum floor velocity. This approach is conservative in that the rack and fuel vibration must be out of phase if impact is to occur. Impact loads were determined as a function of time and were included in load combinations b, c, and d.

5.3 Structural Adequacy

Using the previously listed loads and load combinations, stresses were calculated at critical sections of the rack. The results of the structural and seismic analyses demonstrate that the spent fuel racks are structurally adequate and will meet the design criteria. Critical stresses together with locations and margins to allowable are gi\ n in Table 5-2.

5.4 Pool Wall and Floor Loading

The ability of the fuel pool floor and walls to withstand the loads imposed by the modified fuel racks will be analyzed in accordance with Section 5.4 of the TMI-1 FSAR. All loads, including the thermal and water sloshing loads will be combined in compliance with NRC Standard Review Plan 3.8.4.

TABLE 5-2

SUMMARY OF CRITICAL STRESS RESULTS

Location	Limiting Load Combination*	Calculated Stress (psi)	Allowable Stress (psi)	Margin
Cell to grid welds	d	14,000	17,600	1.26
Grid beams	d	23,700	26,400	1.11
Leg to gusset welds	b	9,770	11,000	1.13
Seismic restraint bearing pad to grid weld	b	9,670	11,000	1.14
Lifting lug	a	12,800	16,500	1.29

*See Section 5.1.2 for definition.

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6.0 COOLING CONSIDERATIONS

6.1 General Description

The Spent Fuel Pool (SFP) Cooling System removes decay heat generated by spent fuel stored in the TMI-1 Pools "A" and "B". The system consists of two cooling water pumps, two heat exchangers, a borated water recirculation water pump, and associated piping that connect this system to the Decay Heat Removal and the Radioactive Liquid Waste Disposal (RLWD) Systems. The Cooling System is redundant in that either of the two pump/cooler combinations or both pumps and coolers can be used to cool Pool "A", Pool "B", or both pools. The system is further described in Section 9.4 of the TMI-1 FSAR.

6.1.1 Normal Refueling

The SFP Cooling System maintains Pools "A" and "B" at a maximum temperature of 135°F under a Normal Refueling condition with one pump and one cooler in operation. A Normal Refueling condition is defined as the decay heat generated from eleven (11) yearly refuelings with the eleventh being placed in the spent fuel pool within 150 hours of reactor shutdown.

6.1.2 Entire Core Offload

The SFP Cooling System has the additional capacity to maintain the spent fuel pools at a maximum temperature of 147°F under an Entire Core Offload condition with two pumps and two coolers in operation . An Entire Core Offload condition is defined as the decay heat generated from eleven (11) yearly refuelings and an entire core offload with the entire core being placed in the spent fuel pool within 150 hours of reactor shutdown. Following the Entire Core Offload, both "A" and "B" spent fuel pools will be full, resulting in the worst case heat generation condition.

6.2 Cooling System Performance

The adequacy of the Spent Fuel Pool Cooling System has been analyzed in view of the expanded fuel storage capacity. Table 6-1 summarizes the cooling system performance for the Normal Refueling and Entire Core Offload conditions.

TABLE 6-1

SPENT FUEL POOL COOLING
HEAT LOADS AND OPERATING TEMPERATURES

	Heat Load (Btu/hr)	Pool Temperature (^{o}F)
Normal Refueling	9.7×10^6	135
Entire Core Offload	2.57×10^7	147

The decay heat loads were calculated using the computer code ORIGEN developed at Oak Ridge National Laboratory. ORIGEN is a point depletion code that solves the equations of radioactive buildup and decay for large numbers of isotopes with arbitrary coupling. The methods of analysis used in this evaluation are considerably more advanced than those used in the TMI-1 FSAR decay heat analysis. That analysis was based on infinite irradiation time of fuel assemblies whereas this analysis is based on finite irradiation time and the actual projected refueling schedule.

The average design burnup of spent fuel is 30,120 MWD/MTU. This burnup is an average of fuel discharged after 3 and 4 cycles of core residence.

The temperature of the Nuclear Services Closed Cycle Cooling Water going into the Spent Fuel Pool (SFP) Cooling Heat Exchangers

is the controlling factor in establishing the heat transfer capability of the Spent Fuel Pool Cooling System. A review of recorded temperatures since unit startup in 1974 shows that using a temperature of 95°F conservatively assumes worst case heat exchanger cooling conditions.

6.2.1 Decay Heat Loads

In evaluating the impact of increasing the storage capacity of TMI-1 spent fuel pools, two decay heat loads have been analyzed:

1. Normal Refueling

   a. 521 spent fuel assemblies in storage accumulated from ten successive yearly refueling outages.

   b. 53 spent fuel assemblies discharged to the storage pools during the eleventh TMI-1 refueling outage scheduled in 1986. It was conservatively assumed that this discharge is completed 150 hours following reactor shutdown from full power.

   c. It was conservatively assumed that all eleven refuelings occurred following reactor operation for 272 effective full power days for an entire year.

The results of this analysis are as follows:

Pool Heat Generation Rate: 9.7×10^6 Btu/hr
Maximum Pool Temperature: 135°F

2. Entire Core Offload

   a. 574 spent fuel assemblies in storage accumulated from eleven successive yearly refueling outages.

b. 177 fuel assemblies (entire core) discharged to the storage pools during the twelfth TMI-1 refueling outage scheduled in 1987. It was conservatively assumed that this discharge is completed 150 hours following reactor shutdown from full power.

c. It was conservatively assumed that all eleven refuelings occurred following reactor operation for 272 effective full power days for an entire year.

The results of this analysis are as follows:

Pool Heat Generation Rate: 2.57×10^7 Btu/hr
Maximum Pool Temperature: $147^{o}F$

6.2.2 Failure Analysis

Two conditions of failure analysis have been analyzed:

1. Single-failure analysis

   Although single failure is unlikely, Table 6-2 is a single-failure analysis of the TMI-1 Spent Fuel Pool Cooling System.

2. Loss of all spent fuel pool cooling

   It is highly unlikely that a complete loss of spent fuel pool cooling will occur due to the following reasons:

   a. Seismic Class I System.

   b. Redundant cooling pumps.

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TABLE 6-2

SPENT FUEL POOL COOLING (SFPC) SYSTEM
SINGLE ACTIVE FAILURE ANALYSIS

		Consequence	
Component	Failure Mode	Normal Refueling (9.7×10^6 Btu/hr)	Entire Core Offload (2.57×10^7 Btu/hr)
Spent Fuel Pool Cooling Pump	Mechanical	1. Redundant pump is operational. No effect on SFPC.	1. Redundant pump is operational. 2. Inventory of spares is available for rapid repairs. 3. Supplemental cooling methods available as discussed in Section 6.2.3. 4. Maximum pool temperature: 199°F.
Nuclear Services Closed Cycle Cooling Water (NSCCCW) Pump (cools SFPC heat exchangers)	Mechanical	1. No effect on SFPC. 2. Redundant NSCCCW pump is available for 100% system capacity.	1. No effect on SFPC. 2. Redundant NSCCCW pump is available for 100% system capacity.
River Water (RW) Pump (cools NSCCCW heat exchangers)	Mechanical	1. No effect on SFPC. 2. Redundant RW pump is available for 100% system capacity.	1. No effect on SFPC. 2. Redundant RW pump is available for 100% system capacity.
Offsite Power	Electrical	1. Emergency power is available. 2. Manual starting of SFPC, NSCCCW, and RW pumps is possible. 3. No effect on SFPC.	1. Emergency power is available. 2. Manual starting of SFPC, NSCCCW, and RW pumps is possible. 3. No effect on SFPC.
SFPC Air-Operated Valves (suction of pumps and outlet of heat exchangers)	Loss of Air and/or Electrical Control Power	1. No effect on SFPC. 2. Valve position remains "as is" upon failure.	1. No effect on SFPC. 2. Valve position remains "as is" upon failure.
NSCCCW Motor-Operated Valves (NS-V16A and NS-V16B at inlet of SFPC heat exchangers)	Loss of Electrical Power	1. No effect on SFPC. 2. Valve position remains "as is" upon failure.	1. No effect on SFPC. 2. Valve position remains "as is" upon failure.

c. Cooling pumps are supplied from separate electrical sources, each with the ability of being powered by separate emergency diesels.

d. Redundant heat exchangers.

e. The systems that provide the ultimate heat sink for the spent fuel pool cooling heat exchangers are Seismic Class I, redundant systems.

Since the heat generation rates as a result of this modification will be essentially the same as those reported in Section 9.4 of the TMI-1 FSAR, the heatup rates resulting from a loss of all cooling will be essentially unchanged. Therefore, commencing with a pool temperature of 147°F the minimum time for the pool(s) to reach 212°F will be 12.5 hours.

6.2.3 Supplemental Cooling

There are four supplemental means of providing for cooling the spent fuel pools in addition to the Spent Fuel Pool Cooling System.

1. The Decay Heat Removal System can be used to cool the pools.

2. The forced ventilation system can be used to improve the cooling effects of pool surface evaporation.

3. Reclaimed water can be used for pool water makeup as well as for its cooling effect.

4. Time delays can be imposed on the transfer of fuel assemblies into the fuel pool. The decay heat analysis very conservatively assumes that an entire core will be discharged into the spent fuel pools within 150 hours

of reactor shutdown. The first refueling outage at TMI-1 required an entire core offload. From reactor shutdown to removal of the entire core to the spent fuel pool, 436.5 hours (18.17 days) elapsed. Subtracting delays and learning experience, a realistic minimum time for this operation is on the order of 240 hours (10 days). Monitoring the spent fuel pool water temperature and imposing time delays is undesirable, but as an emergency method it is simple and reliable.

6.3 Fuel Element Heat Transfer

The bottom of a fuel cell is elevated above the floor to assure adequate flow under the rack to each fuel assembly. There is also sufficient space between the rack complex and the pool walls to provide adequate downcomer clearance. Analyses have been performed which show that sufficient flow is induced by natural convection to preclude local boiling in the hottest storage location.

The analyses were based on the following assumptions:

1. The element inlet temperature is the mixed hot temperature of the pool. This temperature is 147^{o}F and applies to the thermally limiting condition of a no failure full-core offload.

2. A hot assembly peaking factor of 1.78 is applied to a limiting batch average assembly energy release rate. The average assembly energy release rate is 1.43×10^5 Btu/hr corresponding to 150 hours after shutdown.

3. The maximum local peaking factor is 2.67 giving a maximum local heat flux of 1360 Btu/hr-ft^2.

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4. A film coefficient of 36 Btu/hr-ft^2-^{o}F is based on pure conduction through a stagnant boundary layer at the fuel rod surface.

5. A one-dimensional fluid flow analysis is used.

6. For the hottest storage location the downcomer region on the periphery of the pool feeds ten assemblies in a row, each assumed to be generating the maximum heat rate defined in Assumption 2.

With the spent fuel pool cooling system operational, the local coolant temperature will not exceed 170^{o}F for an Entire Core Offload. For this condition the maximum surface temperature of a fuel rod is less than 207^{o}F providing more than 31^{o}F margin to local boiling. The margin to bulk boiling is greater than 68^{o}F. This represents the limiting thermal condition in the pool. Under the preceding conditions the hottest fuel rod surface temperature is below the local saturation temperature of 238^{o}F and thus precludes local boiling. During normal refueling the thermal conditions in the pool are less limiting than those given above.

6.4 Spent Fuel Pool Chemistry Control

Water purity and clarity are maintained by the Spent Fuel Pool Cooling (SFPC) System and the Radioactive Liquid Waste Disposal (RLWD) System. The borated water recirculation pump takes a suction on the skimmers located at the top of Pool "A" or Pool "B" and discharges to the RLWD Systems. Water passes through one of two precoat filters in the RLWD System. The filter uses stainless steel wire wound elements with a powdered resin. When the pressure drop across the filter becomes too great, it is backwashed and recoated. In this manner, floating debris is removed from the pool surface and water quality is maintained. The volume of Pools "A" and "B" is 649,000 gallons. The borated water

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recirculation pump has a capacity of 180 gpm. This system, therefore, has the capacity to filter the pools every 60 hours.

Radioactive contaminant levels in the fuel pools are primarily a function of failed fuel fraction and reactor operating level with the highest levels during and shortly following refuelings. The RLWD System additionally processes pool water to control radioactive contaminant levels. Besides the precoat filters which utilize ion exchange as well as a filtering mechanism, the RLWD System contains demineralizers and waste evaporators for processing pool water. Besides these means, natural radioactive decay helps reduce levels of radioactivity. Operating experience has demonstrated that the SFPC and RLWD Systems are highly effective in maintaining water purity and clarity during storage periods and refueling operations. Throughout 1976, the water in Pool "A" was circulated through the RLWD cleanup system for a total of 177 hours, 104 hours of this cleanup occurred prior to placing any fuel in the spent fuel pool; 73 hours of cleanup occurred while an entire core was temporarily stored in Pool "A". That cleanup proved more than satisfactory to maintain pool chemistry conditions and minimize pool radioactive contaminant levels. For the last eight months of 1976, there has been no need to circulate pool water through the cleanup system, due to the thoroughness of cleanup during the refueling outage.

7.0 RADIOLOGICAL CONSIDERATIONS

7.1 Fuel Handling Building Dose Rates

The additional spent fuel assemblies in Pool "B" resulting from this rack modification will have in insignificant impact on the radiological effects discussed in the TMI-1 FSAR. A QAD computer code was utilized for this analysis. Calculations indicate that the dose rates in the fuel handling building as reported in Sections 11.3.1 and 11.3.2.6 of the TMI-1 FSAR will be essentially unchanged. Levels will normally be less than 1.5 mR/hr with certain refueling manipulations causing short term levels in excess of 1.5 mR/hr. As stated in the TMI-1 FSAR, radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for unit personnel in order not to exceed the integrated doses specified in 10 CFR 20.

Additionally, a QAD computer analysis indicates that the radiation levels discussed in Section 11.3.2.6 of the TMI-1 FSAR will remain unchanged. That is, during spent fuel transfer, the dose rate at the pool surface with a minimum of 7 feet of water shielding between the top of an assembly and the surface will result in approximately 15 mR/hr. The dose contribution attributed to the increased fuel storage is negligible.

The radiological consequences of a fuel handling incident are discussed in Section 14 of the TMI-1 FSAR. The bases for the analysis are unaffected by enlarging the capacity of the spent fuel storage pool and therefore the analysis and results are still applicable.

7.2 Cask Handling

The spent fuel cask drop analysis was filed with the NRC on February 14, 1976. This submittal is currently under NRC review.

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This fuel rack modification has no effect on the evaluation, since the possibility of a cask drop in the spent fuel pool has been minimized.

8.0 CONCLUSION

Based on the above analyses and description, Met-Ed concludes that the described modification can be accomplished without undue hazard to the health and safety of the public and that it conforms to applicable regulations.