Regulatory File Cy

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	Docket No. 50-286
(Indian Point 3 Nuclear Power Plant)	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Sections 50.59 (c) and 50.90 of the regulations of the U. S. Nuclear Regulatory Commission, Power Authority of the State of New York as sole owner and co-holder of Facility Operating License No. DPR-64, hereby requests that portions of Technical Specification 3.8 set forth in Appendix A to that license be amended.

In addition, the Power Authority requests the Commission to review and approve a proposed modification to the Indian Point 3 Nuclear Power Plant storage facility pursuant to Section 50.59 (a) (2) (ii) of the Commission's regulations.

The proposed modification is described and evaluated in Attachment A to this Application. The proposed Technical Specification changes consist of the specific revisions set forth in Attachment B to this Application, and a safety evaluation of the proposed changes is set forth in Attachment C. This evaluation demonstrates that the proposed changes do not involve a significant change in the types or an increase in the amounts of effluents or any change in the authorized power level.

POWER AUTHORITY OF THE STATE
OF NEW YORK

By George T. Berry

George T. Berry
General Manager and
Chief Engineer

Sworn to before me this
1st day of September 1977

Helen J. McCormick
Notary Public

HELEN J. McCORMICK
Notary Public, State of New York
No. 01MC 2607500
Qualified in Kings County
Term Expires March 30, 1979

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ATTACHMENT A

SPENT FUEL POOL MODIFICATION

DESCRIPTION

AND

SAFETY ANALYSIS

Power Authority of the State of New York
Indian Point 3 Nuclear Power Plant

August 1977

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1.0 INTRODUCTION

Uncertainties in the future availability of fuel reprocessing facilities have prompted the Power Authority of the State of New York to initiate plans to increase the spent fuel storage capacity of the Indian Point 3 Nuclear Power Plant (IP3). The proposed modification would replace the existing spent fuel storage racks with a new rack design arranged in a more closely spaced lattice array, thereby increasing the spent fuel storage capacity of IP3 from 264 to 837 spent fuel assemblies plus three failed fuel assemblies. This increased capacity would provide storage space for all spent fuel discharges including the 10th refueling outage (with full core reserve).

The modification is scheduled to be complete prior to April 1978, which is the earliest estimated date of the first refueling. It is intended that the necessary support modifications be completed, and that the fuel storage racks be installed while the fuel storage pool is dry. Installation in the dry condition will permit the activities to be carried out without the additional procedures and safety considerations that would be necessary if there were irradiated fuel stored in the spent fuel pool.

2.0 GENERAL DESCRIPTION

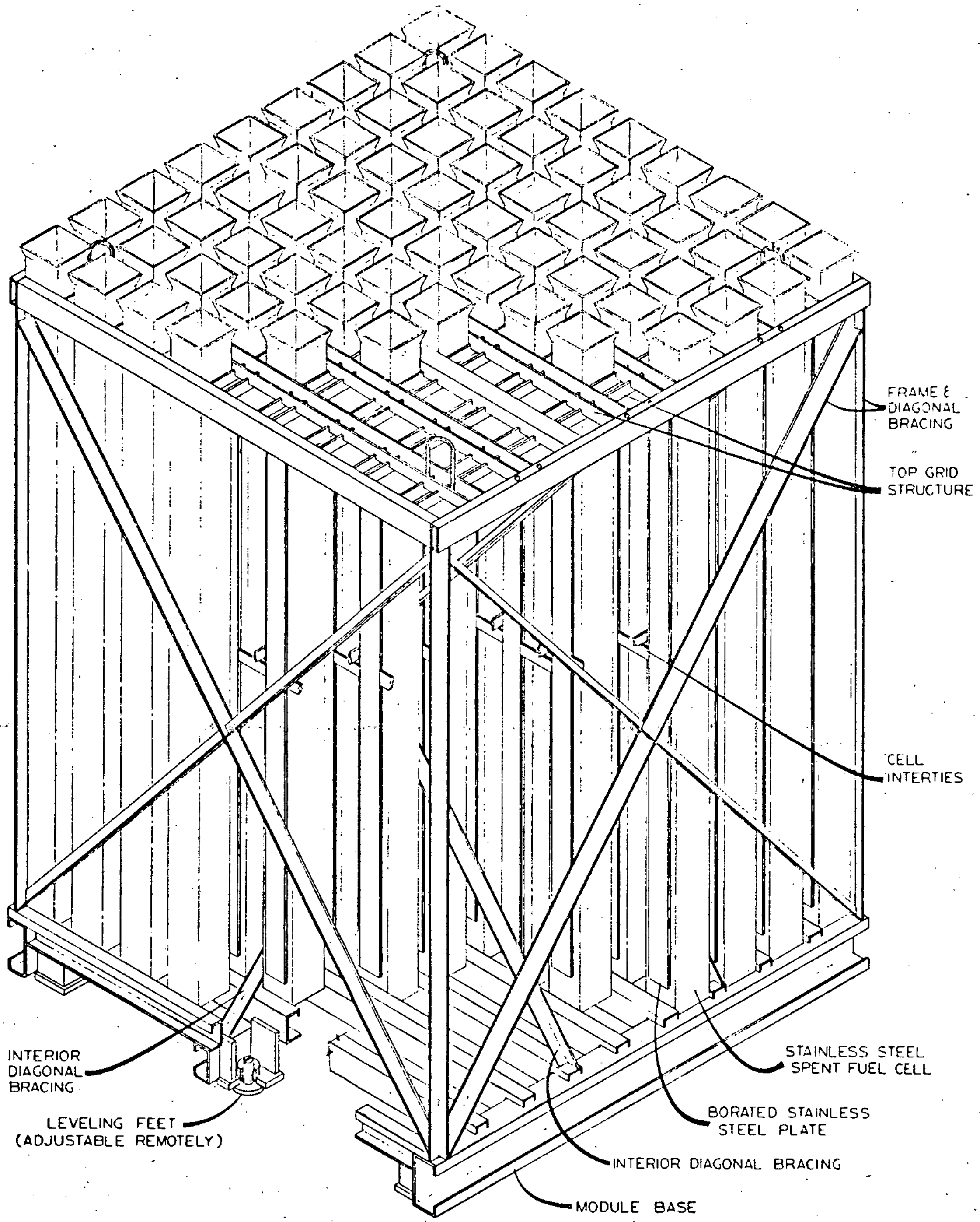
The spent fuel storage racks for Indian Point 3 consist of structural grid frames supporting storage receptacles (fuel storage cells) for spent fuel as shown in Figure 2.0.1. The storage cells, which hold each spent fuel assembly, are square tubes formed from Type 304 stainless steel sheet of 0.150 inches minimum thickness with borated stainless steel poison plates welded to the cell in specified locations. The spent fuel bundle is located inside the fuel storage cell and is supported on a 1/4 inch thick support plate. Each storage cell has a 6" diameter hole at the bottom to allow natural convection cooling. Adequate space between fuel storage cells is provided for downflow.

Twelve (12) rack modules of seven (7) different sizes are used in the available space of the Indian Point 3 spent fuel pool. In seven (7) of the twelve (12) modules, the center-to-center spacing of the fuel cells is 12 inches each way with two (2) borated plates attached to each fuel cell as shown in Figure 2.0.2 and designated Type A. In the other five (5) modules, the center-to-center spacing of fuel cells is 12 inches in the North-South direction, and 11-1/4 inches in the East-West direction, with three (3) borated plates attached to each fuel cell as shown in Figure 2.0.2 and designated Type B. Each borated plate is 145" x 7" x 1/8" thick Type 304 stainless steel containing 1.0% minimum, 1.2% maximum by weight of boron. No borated plates are placed on the outside faces of the cells in any module. Specifically, the rack modules include:

- Three (3) 9 x 7 rack modules with cell spacing at 12" x 12"
- One (1) 6 x 6 rack module with cell spacing at 12" x 12"
- Two (2) 9 x 9 rack modules with cell spacing at 12" x 12"
- One (1) 8 x 9 rack module with cell spacing at 12" x 12"
- Two (2) 9 x 9 rack modules with cell spacing at 12" x 11-1/4"
- One (1) 8 x 9 rack module with cell spacing at 12" x 11-1/4"
- Two (2) 9 x 8 rack modules with cell spacing at 12" x 11-1/4"

The fuel storage modules will be supported and leveled by remotely adjustable feet. The feet will bear directly on the pool floor. All modules are also connected to the existing 4-1/2" diameter pool floor embedments, which provide location of the modules and in combination with friction resist the horizontal seismic loads. Adjacent rack modules are interconnected by bolted interties which interlock the modules to prevent tipping. The rack module interties are designed to permit free thermal expansion of adjacent modules, while retaining the vertical and horizontal loads resistance required to prevent rack overturning.

The replacement spent fuel storage racks will provide storage for a total of 837 spent fuel assemblies plus three (3) failed fuel assemblies. The racks are specifically designed to store spent fuel assemblies from the Indian Point 3 Reactor. However, the racks may also be used for the storage of new fuel or partially spent fuel that has been removed temporarily form the reactor.



TYPICAL HIGH DENSITY POISON SPENT FUEL MODULE

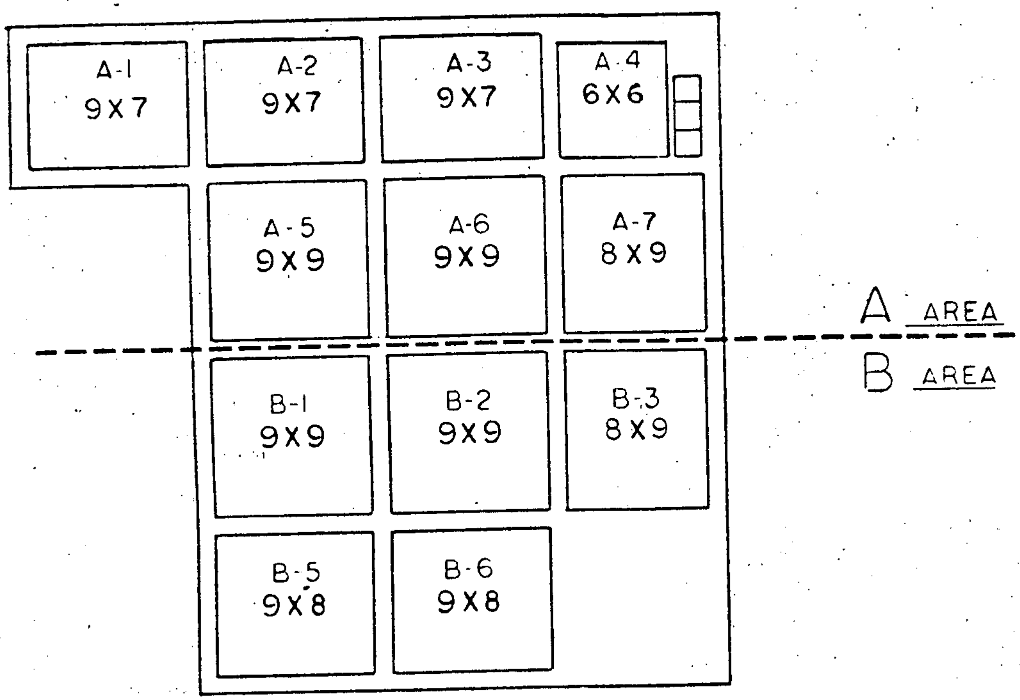
FIGURE 2.0.1

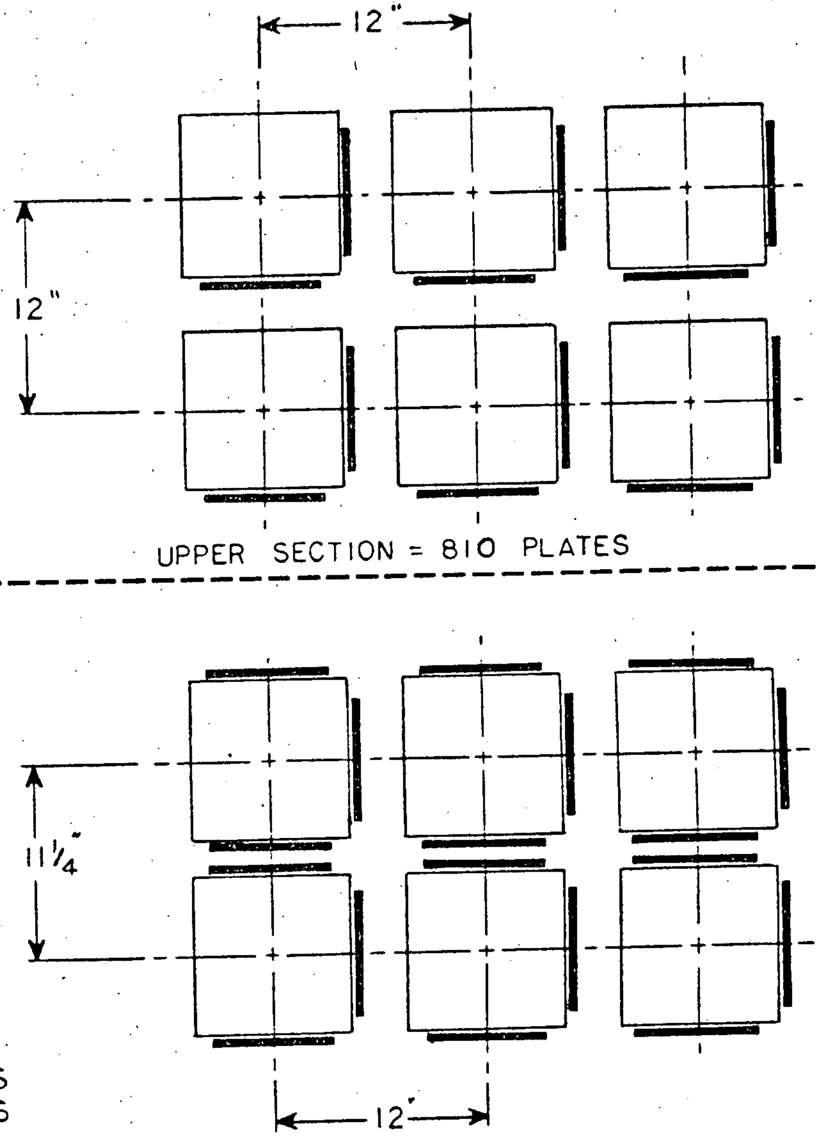
EXXON NUCLEAR COMPANY, Inc.

TYPICAL HIGH DENSITY SPENT FUEL STORAGE MODULE

TJN

NONE





PLAN-GENERAL ARRANGEMENT

TOTAL SPENT FUEL STORAGE CAPACITY, 837 POSITIONS
TOTAL FAILED FUEL STORAGE CAPACITY 3 POSITIONS

INDIAN POINT UNIT Nº 3

LOWER SECTION = 1003 PLATES

TOTAL PLATES, 1819

FIGURE 2.0.2

2.1 STRUCTURAL AND MECHANICAL DESIGN

2.1.1 Design and Analysis

A comprehensive structural evaluation of the high density spent fuel storage racks for Indian Point 3 has been performed. This evaluation included calculation of static and dynamic seismic loads, stress analysis for all applicable loading combinations, and determination of structural adequacy of all load carrying members.

2.1.2 Design Criteria

Structural design criteria for spent fuel storage racks have been developed to assure conformance with recognized codes and applicable U.S. NRC Regulatory Guides, as follows:

1. The fuel storage racks have been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class III Linear Supports.

2. Regulatory Guide 1.13 - The design conforms with the stated provisions for spent fuel storage equipment.

3. Regulatory Guide 1.29 - The spent fuel storage racks have been designed as Category I structures.

4. Regulatory Guide 1.92 - Seismic load combinations of vibrational modes and three (3) orthogonal component motions [two (2) horizontal and one (1) vertical] are in accordance with the provisions of the Regulatory Guide.

5. U.S. NRC - Standard Review Plan, Section 3.8.4, Other Seismic Category I Structures - Load combinations and structural acceptance criteria for steel structures are as follows:

D = Dead load of fuel rack structure and stored fuel.

Live Loads (L)

L_a = Load from force of lowering a fuel assembly at maximum crane speed.

L_b = Crane uplift force.

L_c = Load from accidental release of a fuel assembly while handling.

L_d = Rack shipping loads.

T_o = Thermal loads during maximum normal conditions.

E = Loads generated by Operating Basis Earthquake (OBE).

E^1 = Loads generated by Safe Shutdown Earthquake (SSE).

T_a = Thermal load resulting from maximum pool temperature.

Load Combination	Limit
$D + E$	$S^{(1)}$
$D + L_d$	S
$D + L_a + T_o$	1.5S
$D + L_b + T_o$	1.5S
$D + L_c + T_o$	$1.5S^{(2)}$
$D + T_o + E$	1.5S
$D + T_o + E^1$	1.6S
$D + T_a^{(4)} + E$	$1.6S^{(3)}$
$D + T_a^{(4)} + E^1$	$1.7S^{(3)}$

NOTES:

(1) S is the required section strength based on the elastic design methods and the allowable stresses defined in NF-3400 for linear type supports.

(2) Localized damage to fuel cells is permitted provided that the spacing of fuel bundles is not affected.

(3) For these combinations, in computing the required section strength, S, the plastic section modulus of steel shapes may be used, except where spacing of fuel bundles would be affected.

(4) Self-limiting stresses due to T_a may be neglected.

6. The spent fuel storage racks have been designed for the Indian Point 3 response spectrum at ground level as presented in Figures A.1 and A.1-2 of the Final Safety Analysis Report (FSAR). The ground level response spectrum was used since the fuel storage pool floor concrete is poured directly on bed rock.

7. A damping coefficient of 1% was used for the seismic analysis for both OBE and SSE cases. This is in accordance with the original plant design and is conservatively less than the 2% (OBE) and 4% (SSE) values permitted for welded structures by Regulatory Guide 1.61.

8. No credit was taken for any structural contribution from the borated stainless steel poison plates welded to the fuel storage cells. However, the effects of the imposed stresses resulting from the static and dynamic deflections of the fuel cells were evaluated for the poison plates and their attachment welds.

9. Consideration was given to the worst possible loading condition of the fuel storage racks varying from empty to fully loaded.

2.1.3 Static and Dynamic Analysis

The SAP-4[1] computer program was used for static and dynamic analysis of the fuel storage rack structure. Figure 2.1.1 shows the structural analysis model. The structural analysis model is for a 9 x 9, Type B storage rack which is the most limiting structural case. The analytical model is for a complete rack structure and comprises 120 nodes and 223 structural elements. The model includes the rack-to-rack interties which are included in the actual structure to ensure structural stability.

The total mass of water enclosed in the fuel storage rack, properly proportioned between nodes was lumped together with the masses of the fuel assemblies, poison plates and storage cells in the lumped parameter SAP-4 model. A description of the additional analysis performed to support the lumping of the fuel assembly mass is given in Section 2.1.4 (c).

Static and seismic loads obtained from the SAP-4 model were combined together and with other loads as required by the criteria outlined above to calculate stresses in the structural members. The calculated stresses were then compared with the applicable allowable stresses to confirm the structural adequacy.

Computer plots of the primary vibrational mode shape in each direction are presented in Figures 2.1.2, 2.1.3, and 2.1.4. The calculated natural frequencies and participation factors are given in Table 2.1.1.

The limiting load combinations and stress values for the rack members are presented in Table 2.1.2. A summary of the loads on the pool walls and floor is included in Table 2.1.3.

2.1.4 Non-Linear Effects

Time history analysis of a single fuel storage cell/fuel assembly to account for the effects of the clearance gap between the storage cell wall and the fuel assembly were performed. The method of analysis was identical to that submitted by Arkansas Power and Light Company in its letters dated October 18, 1976 and November 11, 1976, and as approved by the NRC in its Safety Evaluation Report for the Arkansas Nuclear One, Unit 1 Spent Fuel Rack Modification dated December 17, 1976.

The analysis was performed using the same data as for ANO-1 except as follows:

(a) The time history used was that used in the original design of the Indian Point Unit 3 plant, Reference FSAR, Figure A-III.

(b) The linear elastic portion of the analysis was performed using a damping coefficient of 1%, the same as that used in the overall rack analysis described in Section 2.1.3. The non-linear portion of the analysis was performed using the same representative damping values used for the ANO-1 analysis.

(c) The additional mass of two (2) borated poison plates was included in the model.

The results of the analysis were that the maximum combined support reactions calculated by the non-linear time history analysis was only 52% of the maximum combined reactions calculated by the linear elastic time history analysis.

It is concluded that modeling the Indian Point Unit 3 spent fuel storage rack fuel cell/fuel assembly units assuming that the fuel cell and fuel assembly are coupled, will result in greater storage cell reaction forces than explicitly modeling the initial clearance between the storage cell walls and the fuel assembly. Therefore, application of coupled fuel storage cell/ fuel assembly lumped mass modeling assumption in the complete spent fuel storage rack module linear elastic analysis, with an impact factor of unity, will result in conservative storage cell reaction loads to the fuel rack frame.

It is concluded, therefore, that application of the coupled fuel storage cell/fuel assembly lumped mass modeling assumptions in the overall rack analysis with an impact factor of unity is a conservative and acceptable procedure.

(d) Dropped Fuel Assembly Accident

An evaluation of the effects of a postulated dropped fuel assembly accident has been completed to confirm that there would be no effect on the spacing of fuel assemblies stored in the racks. The method of design and analysis is identical to that submitted by Omaha Public Power District in its letter dated June 2, 1976, and as approved by the NRC in its Safety Evaluation Report for the Fort Calhoun Station Unit No. 1 Spent Rack Modification dated July 2, 1976.

The compression test data developed for Fort Calhoun was scaled to the proper fuel cell size and thickness using the methods given in Reference 2. The fuel cell sizes for the two (2) plants are:

Fort Calhoun - 8 5/8" Inside Square, .135" Wall

Indian Point 3 - 8 31/32" Inside Square, .150" Wall

The maximum possible fuel drop height is 15 inches from the top of the fuel cells, since fuel bundles will not be moved over the fuel storage racks at a higher elevation. The maximum load imposed on the storage cell by this drop is conservatively calculated to be 36,600 pounds resulting from a kinetic energy at the point of impact of 1,788 ft-lbs. Water drag effects were conservatively neglected in calculating the maximum impact energy. This load was combined with dead weight loading to show compliance with the load combination criteria outlined above.

In the case where the fuel assembly is dropped inside the storage cell, the fuel assembly would impact the 1/4 inch support plate at the bottom of the cell. The welds attaching this plate to the storage cell are weaker than the connection of the cell to the rack frame members. Damage would, therefore, be limited to failure of the support plate attachment welds, in which case the fuel assembly would fall a further 16 inches to the pool floor.

The effects of a dropped assembly accident in which the assembly rotates as it drops, was also evaluated. In this case, the assembly impacts a row of storage cells and comes to rest laying on top of the rack modules. The maximum kinetic energy of impact on one (1) cell is conservatively calculated to be 1,150 ft-lbs resulting in lower loads than the simple vertical drop case discussed above.

2.1.5 Structural Adequacy of Fuel Pool

The bottom of the Indian Point 3 spent fuel pool is a 3'-0" thick reinforced concrete mat poured directly on bedrock as described in the Indian Point 3 FSAR. The racks are supported on adjustable screw feet with an 8 inch diameter bearing area as shown in Fig. 2.0-1.

The maximum vertical load on the adjustable screw feet included the following:

1. The dead load of the completely filled rack structure.
2. Vertical downward seismic force.
3. The overturning moment caused by horizontal seismic forces.

The analysis showed that the resulting bearing stresses on the concrete are less than the allowable bearing stresses of ACI-318-71.

The proposed fuel racks design impose loads on the pool wall only in the northeast corner of the fuel pool. The force from the top of the racks against the 6'-3" thick pool wall was found to produce less than 1 ksi in the pool wall reinforcing steel. Since the minimum yield strength of the reinforcing steel is approximately 60 ksi, it was found that the fuel pool is adequately designed.

Therefore, it is concluded that the Indian Point 3 spent fuel pool structure has sufficient capability to accommodate the proposed fuel storage racks.

Section 2.1 References

1. Bathe, K., E. Wilson and F. Paterson, "SAP IV, A Structural Analysis Program for Static and Dynamic Response of Linear Systems," University of California; June, 1973.

2. Hopkins, Design and Analysis of Shafts and Beams, McGraw Hill, 1970.

TABLE 2.1.1

NATURAL FREQUENCIES AND MODAL PARTICIPATION FACTORS

Mode #	Frequency (M_Z)	Participation Factor x	y	z	Notes
1	5.46	.01	20.38	.43	Primary Y Mode
2	5.54	19.88	.02	.21	Primary X Mode
3	7.75	.006	.006	.0001	
4	7.91	.0004	.01	.0001	
5	10.95	.0001	1.84	.002	
6	11.69	1.42	.0001	.002	
7	12.60	.004	.001	.0001	
8	13.97	.01	.02	.0001	
9	16.45	4.64	.0004	.003	
10	16.65	.0001	7.22	.001	
11	17.41	.0001	6.44	.001	
12	18.33	6.76	.00802		
13	21.14	1.31	.05	.003	
14	21.74	.14	.10	.003	
15	23.73	.02	2.23	1.88	
16	24.33	.08	3.16	1.39	
17	24.71	1.13	.20	.70	
18	25.25	2.48	.06	.51	
19	25.73	.48	.54	.05	
20	26.19	.69	.15	.04	
21	26.33	.082	.22	1.61	
22	26.45	.009	.10	1.59	
23	27.51	.01	.04	13.24	Primary Z Mode
24	28.00	.18	.007	.03	

TABLE 2.1.2

COMPARISON OF MOST LIMITING STRESSES

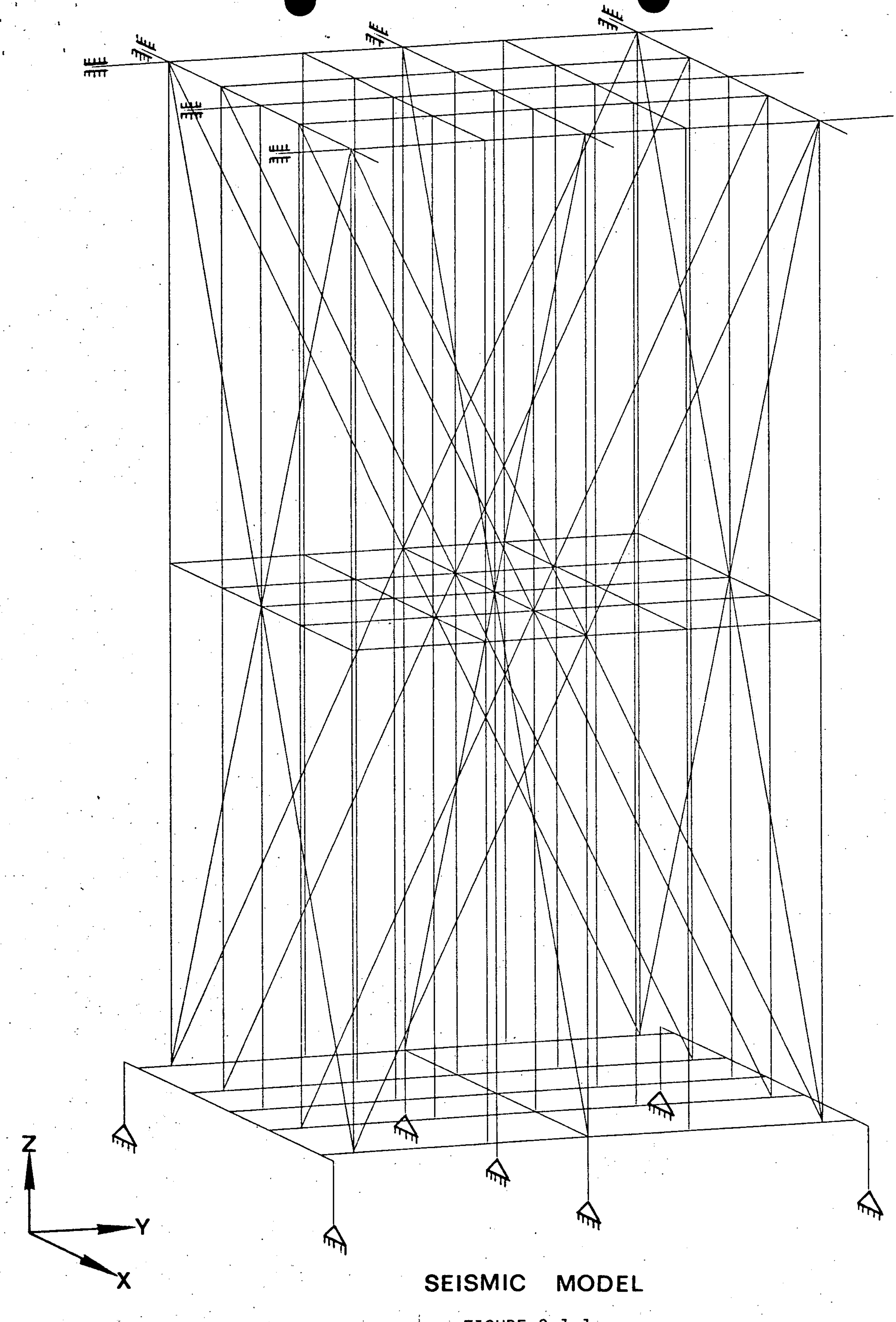
AND ALLOWABLE STRESSES ON STRUCTURAL MEMBERS

Structural Member	Most Limiting Load Combination	Type Of Stress	Most Limiting Combined Stress Ratio	Allowable Limit
Cross Bracing	D + E	Axial	0.884	1.0
Spacer Bar	D + E	Axial	0.744	1.0
Fuel Cell	$D + L_c + T_o$	Bending + Axial	0.501	1.0
12½" Channel	D + E	Bending + Axial	0.59	1.0
8" Channel	D + E	Bending + Axial	0.828	1.0
6" x 2-5/8" Channel	D + E	Bending + Axial	0.835	1.0
6" x 2-1/16" Channel	D + E	Bending + Axial	0.914	1.0
6" x 2-1/16" Tubes	D + E	Bending + Axial	0.948	1.0
2-3/8" x ¼" Bars	D + E	Bending + Axial	0.831	1.0
1-7/8" Channel	D + E	Bending + Axial	0.892	1.0
2-5/8" Channel	D + E	Bending + Axial	0.869	1.0
Interties	D + E	Bending	0.949	1.0
Feet	D + E	Bending + Axial	1.0	1.0

TABLE 2.1.3

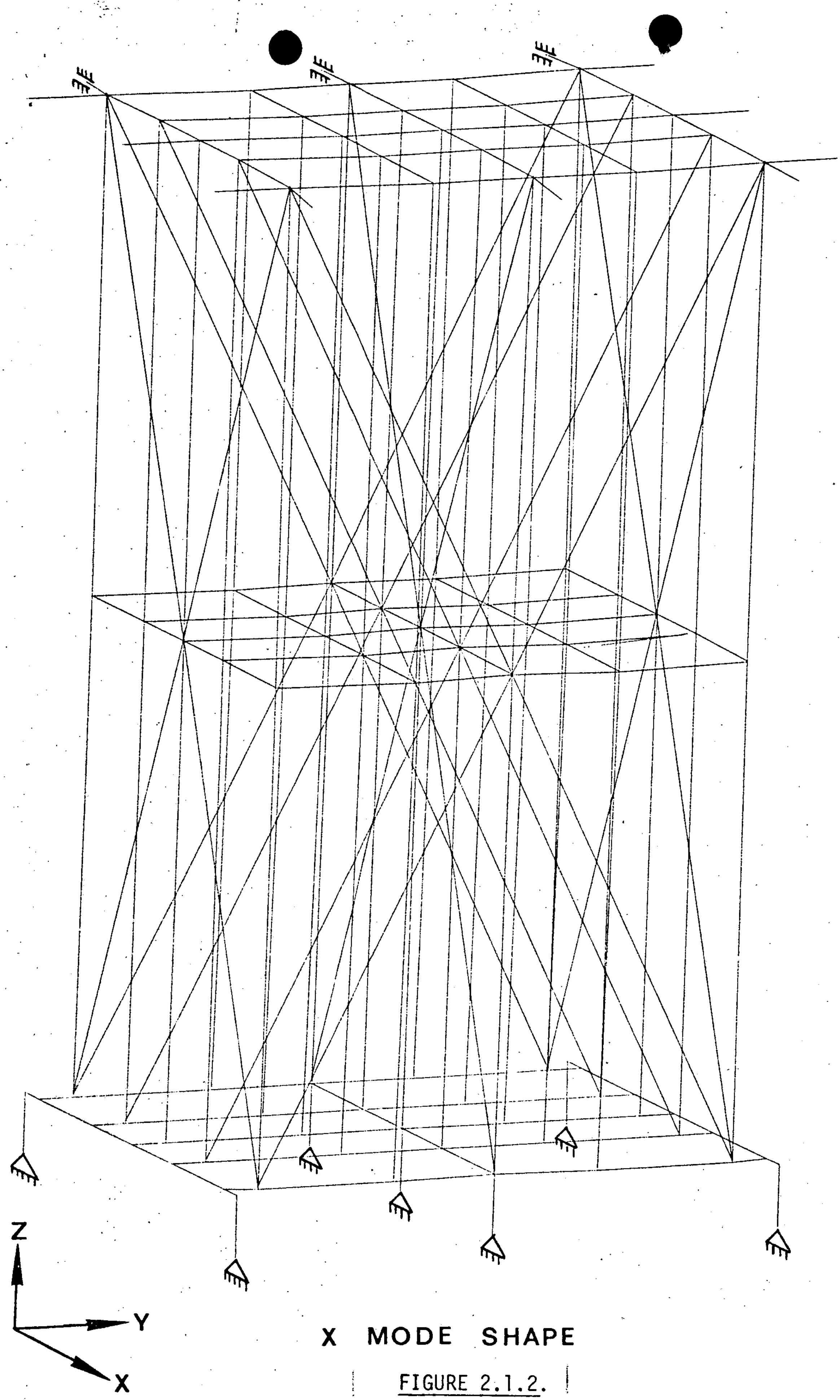
SUMMARY OF LOADS ON THE SPENT FUEL POOL FLOOR AND WALLS

	OBE	SSE
Maximum load per rack foot (vertical)	52,190 lbs.	67,000 lbs.
Shear loads on dowel pins	22,900 lbs.	46,000 lbs.
Load per wall restraint (Northeast corner of pool only)	8,310 lbs.	20,600 lbs.



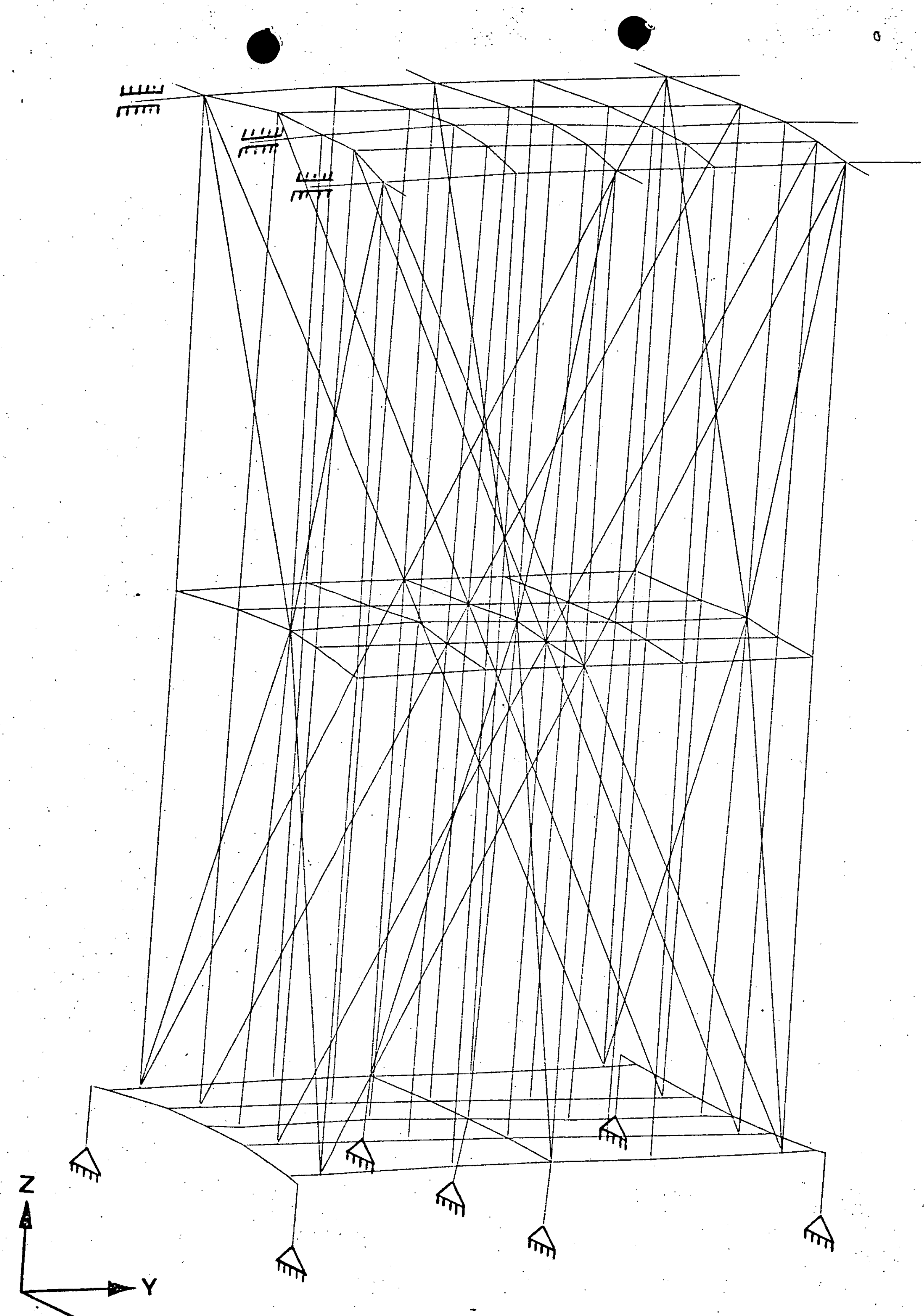
SEISMIC MODEL

FIGURE 2.1.1.



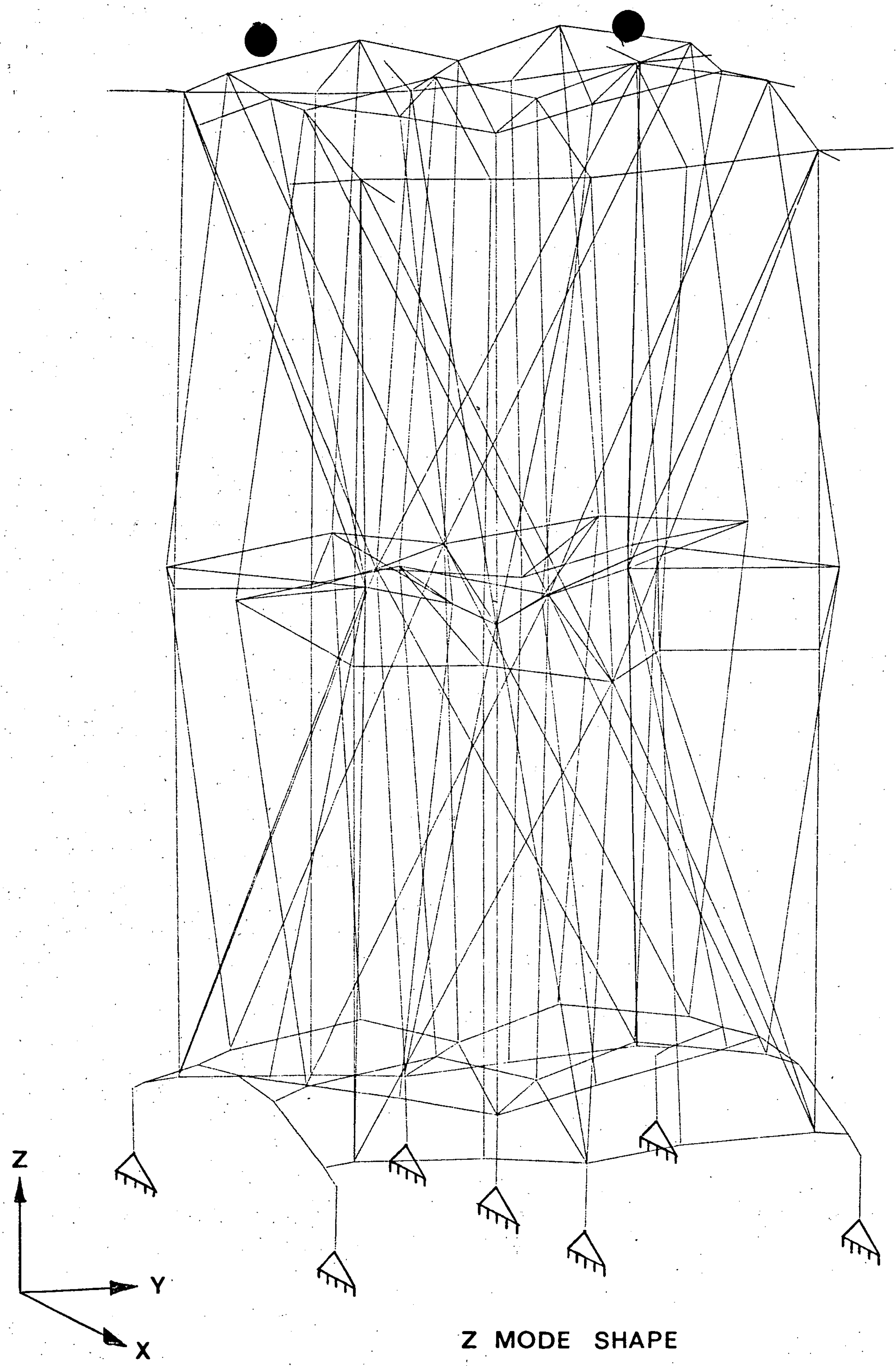
X MODE SHAPE

FIGURE 2.1.2.



Y MODE SHAPE

FIGURE 2.1.3.



Z MODE SHAPE

FIGURE 2.1.4.

2.2 NUCLEAR ANALYSIS

2.2.1 Criticality Considerations

An analysis was performed of the potential maximum reactivity of the fuel stored in the proposed fuel assembly storage facility. This analysis considered the minimum possible spacing under normal and earthquake conditions, the maximum fuel enrichment level, the most reactive conditions of fuel density, and the most reactive water temperature. No credit was taken for any boron present in the storage pool water.

2.2.2 Criticality Criteria

The spent fuel storage racks are designed so k_{eff} is limited to a value of less than 0.95 under all credible circumstances. The calculated value is less than 0.95 by a margin sufficient to account for calculational uncertainties. Credit was not taken for any burnable poison that may be contained in the fuel. Credit was taken for neutron absorption of the 1% borated stainless steel plates and stainless steel storage cells. No administrative procedures are required to space or arrange fuel assemblies in the racks.

The k_{eff} calculations are based on a maximum fuel enrichment level in new unburned fuel of 3.5% U-235. Criticality calculations considered reductions in fuel bundle center-to-center spacing resulting from dimensional tolerances and clearance between the fuel bundle and its storage cell. The calculations also considered variances in boron loadings within the borated plates and deformations under structural loads and from abnormal events.

2.2.3 Calculational Methods

The KENO-III Monte Carlo code was utilized to calculate the reactivity of the Indian Point Unit No. 3 fuel storage array. Multigroup cross section data (18 energy groups) utilized in these calculations were averaged using the CCELL[3], BRT-1[4], and GAMTEC-II[5] computer codes. Specifically, the cross section data for various regions within the storage array were obtained as follows:

CCELL - Utilized to obtain cell averaged multigroup cross section data for fuel rod-water lattices. Such calculations included both the bundle averaged cell parameters and the actual lattice cell parameters (see Table 2.2.1). In addition, CCELL was used to 1) examine the effects of UO_2 pellet density, moderator temperature, and fuel temperature on the infinite media multiplication factor of the fuel assembly, and 2) calculate epithermal multigroup cross section data for stainless steel and boron (E 0.683 ev) averaged in a neutron energy spectrum characteristic of the water regions within a fuel assembly.

BRT-1 - Thermal group (0.683 ev) cross section data for the borated stainless steel plates and stainless steel fuel guides were averaged using the Battelle revised THERMOS code. Such data were averaged assuming a 0.150-inch thick slab of stainless steel next to a 0.125-inch slab of borated stainless steel (1.0 wt.% $B_{nat.}$), both of which were separated from rod-water lattices (characteristic of the fuel assembly) by appropriate slab thicknesses of water, stainless steel, and/or borated stainless steel, depending on the module type.

GAMTEC-II - Multigroup cross section data for water were averaged over a neutron energy spectra characteristic of an infinite media.

In addition to the codes identified above, the KENO-IV Monte Carlo code was utilized to verify the accuracy of the above mentioned calculational technique for poisoned, rod-water lattices. Cross section data used in KENO-IV were generated over 123 energy groups using the NITAWL and XSDRNPM codes.

2.2.4 Storage Array Description

The Indian Point Unit No. 3 spent fuel storage pool will accommodate twelve (12) specially designed storage modules as shown in Figure 2.2.1. Each module contains a specific number of fuel assembly locations (e.g., 81 locations for a 9x9 module) and installation calls for a 14.0-inch nominal center-to-center fuel assembly separation between modules.

The upper section of modules, i.e., those located above the dashed line in Figure 2.2.1 will locate a maximum of 459 fuel assemblies on 12.0-inch nominal centers. (This excludes three locations provided for failed fuel assemblies.) Each guide tube within the rack will have a nominal inside square of 9.0 inches and a minimum stainless steel wall thickness of 0.150 inches. In addition, two 0.125-inch thick borated (1.0 wt.% $B_{nat.}$ minimum) stainless steel plates (7.0 inches wide by 145.0 inches long) will be attached on two adjacent sides of each guide tube as shown in Figure 2.2.2. These are designated Type A modules.

The lower section consisting of Type B modules will provide storage locations for 378 fuel assemblies on 12.0 inch nominal centers in one direction and 11.25-inch nominal centers in the other direction, as indicated in Figure 2.2.2. The individual Type B guide tubes have the same dimensions and material structures as the Type B module guide tube, except that one additional borated stainless steel plate per guide tube is added in the 11.25 -inch direction. In both the upper and lower sections, those guide tubes located on the periphery of each module will not have poison plates on the sides facing the 5.0-inch water gap between modules.

In addition to the modules described above, three locations are provided for failed fuel assemblies adjacent to the 6x6A module. Metal framework is used to locate failed fuel assemblies on a 19.5 inch nominal center-to-center fuel assembly spacing and assures a center-to-center assembly spacing of not less than 14.4 inches. Failed fuel assemblies will have a minimum center-to-center separation from fuel assemblies in the 6x6A module of 15.9 inches

In addition to the nominally spaced array, the minimum spacing between fuel assemblies has also been considered. Specifically, the minimum center-to-center separation between adjacent guide tube will be "gauged" to assure a minimum center-to-center separation of adjacent fuel assembly guide tubes of not less than 0.1875 inches (S) less than the nominal value. All tolerances are from a true position within the storage pool or rack and are not cumulative across an array of adjacent fuel guides. As a consequence, the worst credible spacing in the pool array occurs as a cluster of four adjacent assemblies with other fuel guides being spaced the nominal center-to-center distance from the cluster. Hence, the worst credible spacing arrangement within the array is as shown in Figure 2.2.3. This arrangement also assumes that fuel assemblies in the cluster are in contact with the inside of each respective fuel guide tube.

Placement of fuel assemblies within or on the perimeter of the array outside of the planned storage locations is precluded by rack design and/or fuel handling procedures.

2.2.5 Storage Array Reactivity Calculations

The KENO-III Monte Carlo code was used to compute storage pool reactivities for assumed worst credible conditions. The bundle averaged fuel assembly parameters given in Table 2.2.1 were used to describe the effective fuel assembly. Reactivity calculations were performed using and effectively infinite representation of the storage array. Due to the asymmetrical geometry of the array, specular reflection conditions could not be used to simulate an infinite array. Thus, a 6x6 guide tube array model, of infinite length, was used to determine the maximum reactivity of the array.

In evaluating the overall reactivity of the "as designed" storage array, assumptions were made with regards to the worst credible conditions (from the standpoint of neutronics) that could exist in the pool. Conditions assumed in the "worst case" reactivity calculations include:

1. 3.5 wt.% ^{235}U enriched fresh UO_2 fuel;
2. Bundle-averaged fuel assembly parameters;

3. Minimum guide tube center-to-center spacing of 0.1875 (S) inches less than the nominal values. (This accounts for limits on installation tolerances and guide tube deflection due to load stresses, etc.) This represents the worst case geometry for an array as described in Figure 2.2.3.
4. Temperature variances (20-100^{o}C) in the pool water;
5. No credit taken for any soluble boron in the pool water.

For the nominal case reactivity calculations, only assumptions 1, 2 and 5 are utilized; the pool water temperature is assumed to be 20^{o}C.

Table 2.2.2 lists final results for the storage pool as it will be installed. All worst credible conditions are concurrently considered in the reactivity calculations for the Type A and B modules. In addition to these assumptions, the noncredible condition of assuming the fuel assembly to have a fuel-moderator temperature of 20^{o}C and the water between fuel assemblies to be at 100^{o}C was made. This assumption maximizes both the reactivity of the fuel assembly and the interaction between adjacent assemblies. For this non-credible boundary case, the reactivity was calculated to be 0.929±.004 for the Type A module and 0.940±.004 for Type B.

The KENO-IV Monte Carlo code with 123 energy group cross section data generated using the NITAWL and XSDRNPM codes was utilized to calculate the Type B module array reactivity as described in Table 3.3. This calculation indicates a conservative bias of 0.015±.004 $\frac{\Delta k}{k}$ in the KENO-III calculations when using 18 energy group cross section data averaged as previously described.

2.2.6 Systematic Uncertainties and Benchmark Calculations

Theory-experiment comparisons have been made for small water-moderated critical arrays of fuel rods. Such critical experiments have been evaluated using the KENO Monte Carlo code with cross section data averaged as for this criticality safety evaluation.

The results of these calculations are shown in Table 2.2.3. Inspections of the results indicate that the calculational method yields conservative results relative to the experimental data. In addition, the KENO calculated reactivities are in agreement with previously performed DTF-IV [6] transport theory calculations within the statistical uncertainty of the Monte Carlo calculations.

As a basis for additional calculational model verification, the results of unpublished ORNL critical experiments employing 4.95 w/o ^{235}U metal rods were obtained from E. B. Johnson and G. E. Whitesides[7] OF ORNL. Five of these experiments were chosen for which reactivity calculations were performed. Table 2.2.4 describes the physical makeup for each of the experimental criticals. In performing the actual experiments, an infinite water reflector (6 inches) was present in all directions except on the top of the lattice. The water height above the lattice was varied to control the reactivity.

The three dimensional KENO-II Monte Carlo computer code was used to calculate k_{eff} for each array. The multigroup cross section data (18 energy group) used in KENO-II was averaged using the CCELL, BRT-1, and GAMTEC-II codes. Specifically, the cross section data for each critical (by region) was obtained as follow:

<u>FUEL</u> (rod-water lattice) - the CCELL code was used to obtain

cell averaged multigroup cross section data. CCELL handles both the epithermal (0.683 ev - 10 Mev) and thermal (0.683 ev) energy ranges for the fuel.

WATER (in the reflector) - GAMTEC-II (with GAM-I library data) was utilized to generate multigroup cross section data for water averaged over a neutron energy spectrum characteristic of an infinite media.

DEPLETED URANIUM BLOCK - The GAMTEC-II (ENDF/B III library) code was again utilized for the cross section data generation of U (0.185) metal.

BORAL SHEET - the BRT-1 code was used to generate thermal group (0.683 ev) cross section data for the boral sheet in a neutron energy spectrum characteristic of the actual physical slab geometry (0.25 inches thick). The epithermal cross section data were obtained using CCELL in a neutron energy spectrum described by the fuel pin cell media.

The results of the KENO-II (18 group) reactivity calculations are given in Table 2.2.5. At a 95% confidence level, Cases 1, 2 and 3 are from a statistical standpoint represented by the experimentally determined critical value (k_{eff}=1.000). For cases 4 and 5, which include the boral sheet, results would suggest a conservatism in the KENO-II calculation of not less than ~ 2.5% in k_{eff}.

In addition to these calculations, the reactivities of lattices 2A, 3B, and 4C were calculated using the KENO-IV computer code with 123 energy group cross section data generated by the NITAWL and XSDRNPM codes. These results are also given in Table 2.2.5.

2.2.7 Conclusions

This analysis conservatively demonstrates that the reactivity of the proposed Indian Point Unit No. 3 fuel storage array is less than 0.95 under assumed worst credible array conditions.

Section 2.2 References

3. W. W. Porath, "CCELL Users Guide", BNW/JN-86, Pacific Northwest Laboratories, February, 1972.

4. C. L. Bennett and W. L. Purcell, "BRT-1: Battelle Revised THERMOS", BNWL-1434, Pacific Northwest Laboratories, June, 1970.

5. L. L. Carter, C. R. Richey and L. E. Hushey, "GAMTEC-II: A code for Generating Consistent Multigroup Constants Utilized in Diffusion and Transport Theory Calculations", BNWL-35, Pacific Northwest Laboratories, March, 1965.

6. K. D. Lathrop, "DTF-IV - A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering", LA-3373, Los Alamos Scientific Laboratory, July, 1965.

7. Personal Communication with E. B. Johnson and G. E. Whitesides, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

TABLE 2.2.1

DESIGN BASE INDIAN POINT UNIT NO. 3
FUEL ASSEMBLY PARAMETERS

	NOMINAL LATTICE CELL PARAMETERS	BUNDLE AVERAGED CELL PARAMETERS
Lattice Pitch	0.563"	0.5913"
Clad OD	0.422"	0.4256"
Clad Material	Zr-4	Zr-4
Clad Thickness	0.0243"	0.0261"
UO_2 Pellet Diameter	0.3659"	0.3659"
Pellet Density (% TD)	95.0%	94.0%
Enrichment (wt % U-235)	3.5 (specified)	3.5
Active Fuel Rods	204	204
Rod Array	15 x 15	15 x 15
Effective Array Dimensions	8.445" x 8.445"	8.445" x 8.445"
Control Rod Guide Tubes (Zr-4)	0.545"OD x 0.030"wall(upper) 0.484"OD x 0.030"wall(lower)	N/A
Instrument Tube (Zr-4)	0.463"OD x 0.016"wall(lower)	N/A
Max. U-235 Assembly Loading	44.4557	N/A

TABLE 2.2.2

FINAL DESIGN REACTIVITY CALCULATIONS

INDIAN POINT UNIT NO. 3

W 15 x 15 3.5 w/o

9.00" GT ID

0.15" SS GT Thickness

CCELL-KENO III Calculations; unless specified otherwise

Array	Description	$k_{eff} \pm \sigma$
Type A (12.0" x 12.0")	Worst case geometry Pool temperature - 100°C Fuel assembly temperature - 20°C	0.929 ± .004
Type B (11.25" x 12")	Worst case geometry Pool temperature - 100°C Fuel assembly temperature - 20°C	0.940 ± .004
Type B	Worst case geometry Pool temperature - 100°C Fuel assembly temperature - 20°C	0.926 + .003 NITAWL-XSDRNPM-KENO IV (123 group)
Mixed Module Corner*	Worst case geometry (Δs = 0.50")	0.914 ± .004

*The mixed module corner is an array calculation performed with initial fission neutrons started in the region where two Type A and two Type B modules meet.

TABLE 2.2.3

THEORY - EXPERIMENT CORRELATIONS

EXP'T. NO.	FUEL DENSITY (g/cm^3)	FUEL WT % ^{235}U	FUEL PELLET DIA.(IN)	CLADDING MAT'L.	CLADDING THICK. (IN)	SQUARE LATTICE SPACING(IN)	MODERATOR-TO-FUEL VOLUME RATIO	EXP'TL. RESULTS CYLIND. CORE RADIUS CM	CCELL-DTF-IV CALCULATED REACTIVITY (k_{eff})	CCELL-KENO-II CALCULATED REACTIV[illegible] (k_{eff})
1	10.18	2.70	0.300	304 SS	0.0161	0.435	1.405	26.820	1.016	1.008 $\pm$ .006
2	10.18	2.70	0.300	304 SS	0.0161	0.470	1.853	24.294	1.015	1.014 $\pm$ .005
3	10.18	2.70	0.300	304 SS	0.0161	0.573	3.357	23.600	1.011	1.003 $\pm$ .005
4	10.18	2.70	0.300	304 SS	0.0161	0.615	4.078	24.771	1.009	1.010 $\pm$ .005
5	10.18	2.70	0.300	304 SS	0.0161	0.665	4.984	27.172	1.005	1.005 $\pm$ .005

TABLE 2.2.4

FUEL ROD-WATER LATTICE

Rod Material	Uranium Metal
Enrichment	4.95 wt.% ^{235}U
% Theo. Density	99 (18.9 $^{g}/cm^{3}$ U)
Rod OD, cm	0.762 (unclad)
Rod Height, cm	30.0
Pitch, cm	2.05 (square)
$Vm/_{Vf}$ Ratio	8.22

DEPLETED URANIUM BLOCK

Material	Uranium Metal U (0.185)
% Theo. Density	100 (19.04 $^{g}/cm^{3}$ U)
Length, cm	60.4
Width, cm	21.7
Height, cm	25.9 (centered)

BORAL SHEET (Brooks and Perkins)

Core Material	B_4C and Al
Wt.% B_4C in Core	38.9 (ORNL measurement)
Core Density, g/cm^3	2.63
Core Thickness, cm	0.429
Clad Material (Type)	1100 Al
Clad Thickness, cm	0.104
Length, cm	47.114
Height, cm	25.876 (centered)
Total Width, cm	0.637

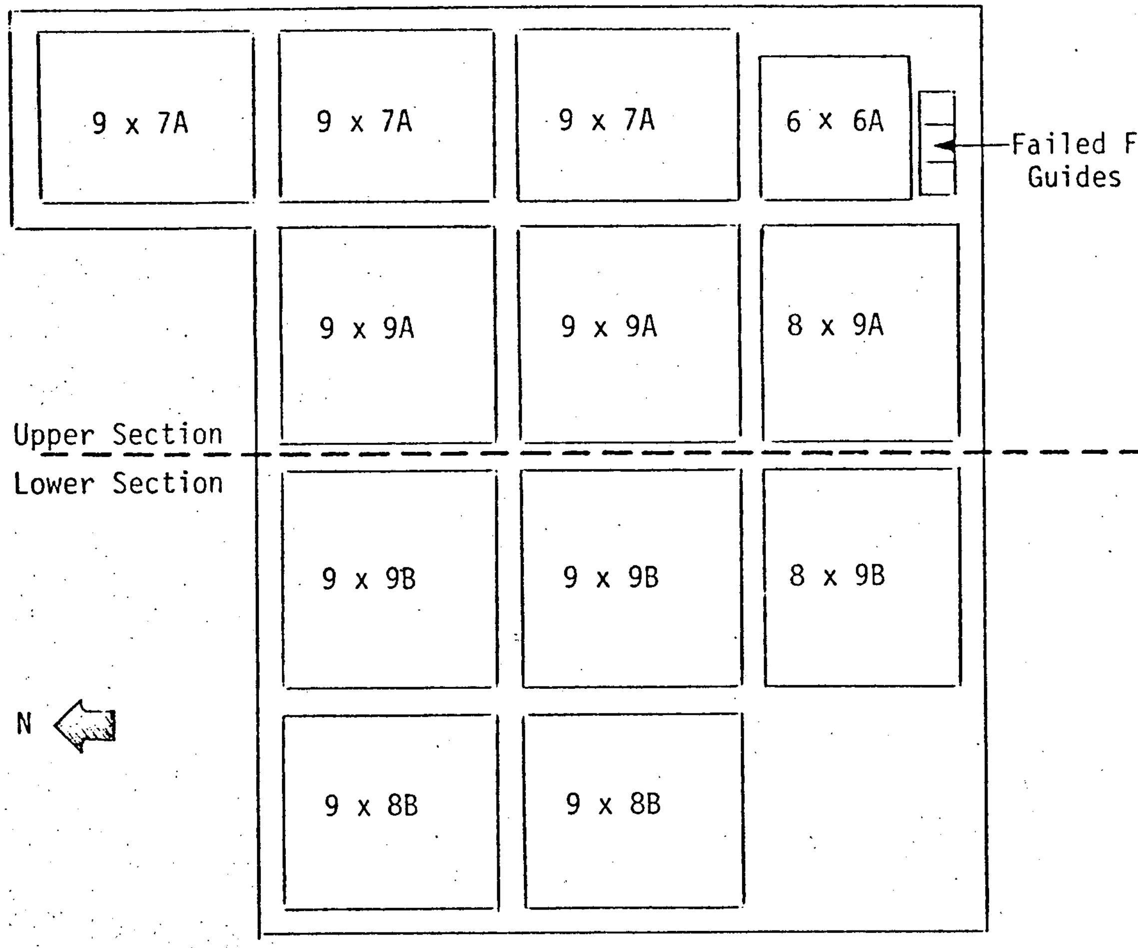
TABLE 2.2.5

CALCULATED K_{EFF} VALUES FOR ORNL CRITICAL LATTICES

Case	Lattice Number	Number of Rods	Critical Water Height Above Lattice, cm	CCELL-KENO II (18 group) $k_{eff} \pm \sigma$	CCELL-KENO II (18 group) 95% Confidence Level	NITAWL-XSDRNPM-KENO IV (123 group) $k_{eff} \pm \sigma$	NITAWL-XSDRNPM-KENO IV (123 group) 95% Confidence Level
			Rod Water Lattice Only				
1	1A	203	7.1	0.988 ± .006	0.976 - 1.000	-	-
2	2A	195	15.24	0.998 ± .006	0.986 - 1.010	0.999 ± .006	0.987 - 1.011
			Rod Water Lattice + U (0.185) Block				
3	3B	245	9.5	1.001 ± .006	0.989 - 1.013	0.993 ± .006	0.981 - 1.005
			Rod Water Lattice + U (0.185) Block + Boral Sheet				
4	4C	324	15.24	1.038 ± .005	1.028 - 1.048	1.000 ± .005	0.990 - 1.010
5	5C	359	11.94	1.037 ± .006	1.025 - 1.049	-	-

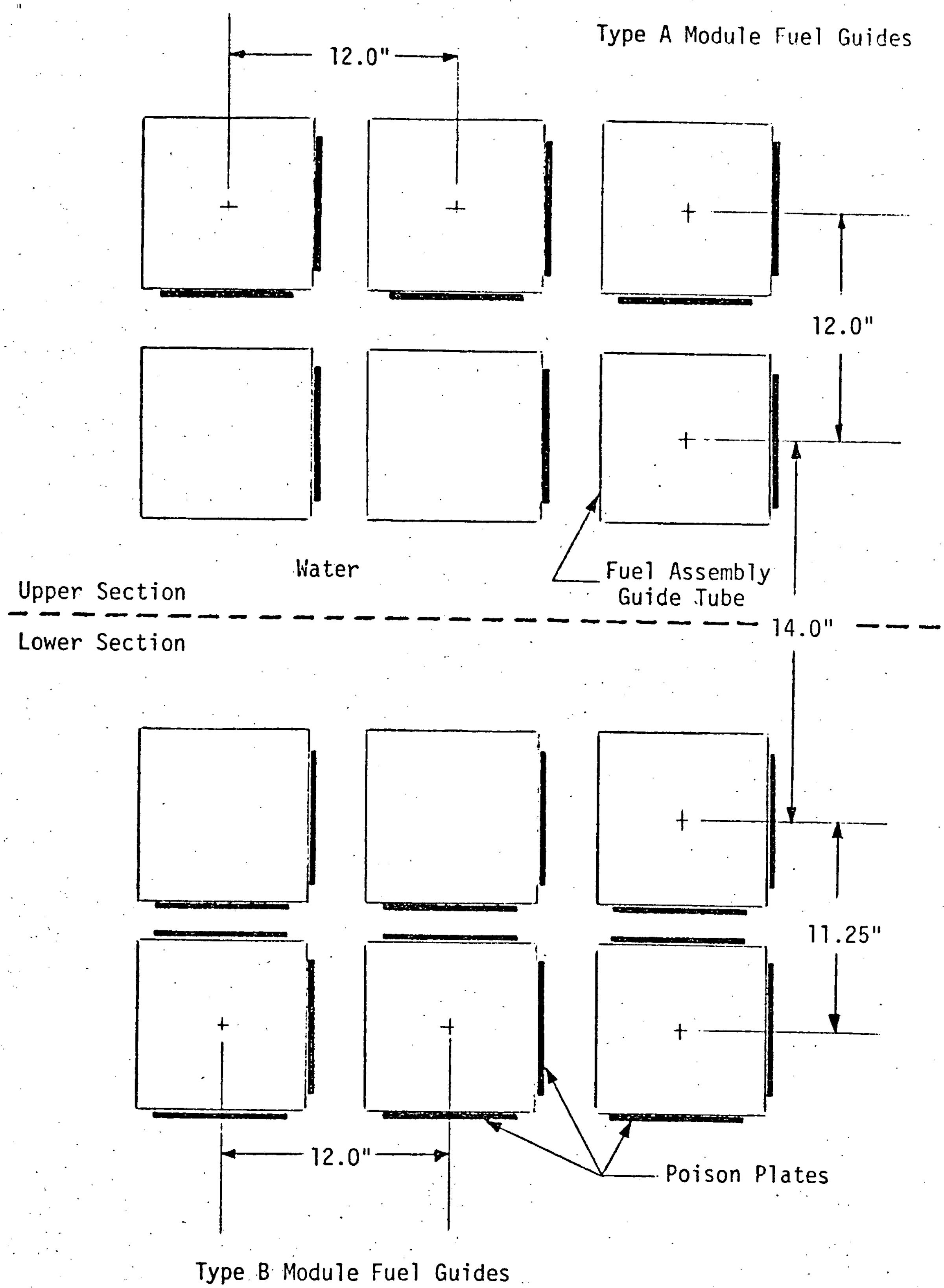
FIGURE 2.2.1

INDIAN POINT UNIT NO. 3 FUEL STORAGE POOL
MODULE ARRANGEMENT (TYPES A & B)



Fuel Storage Locations	837
Failed Fuel Storage Locations	3
TOTAL	840

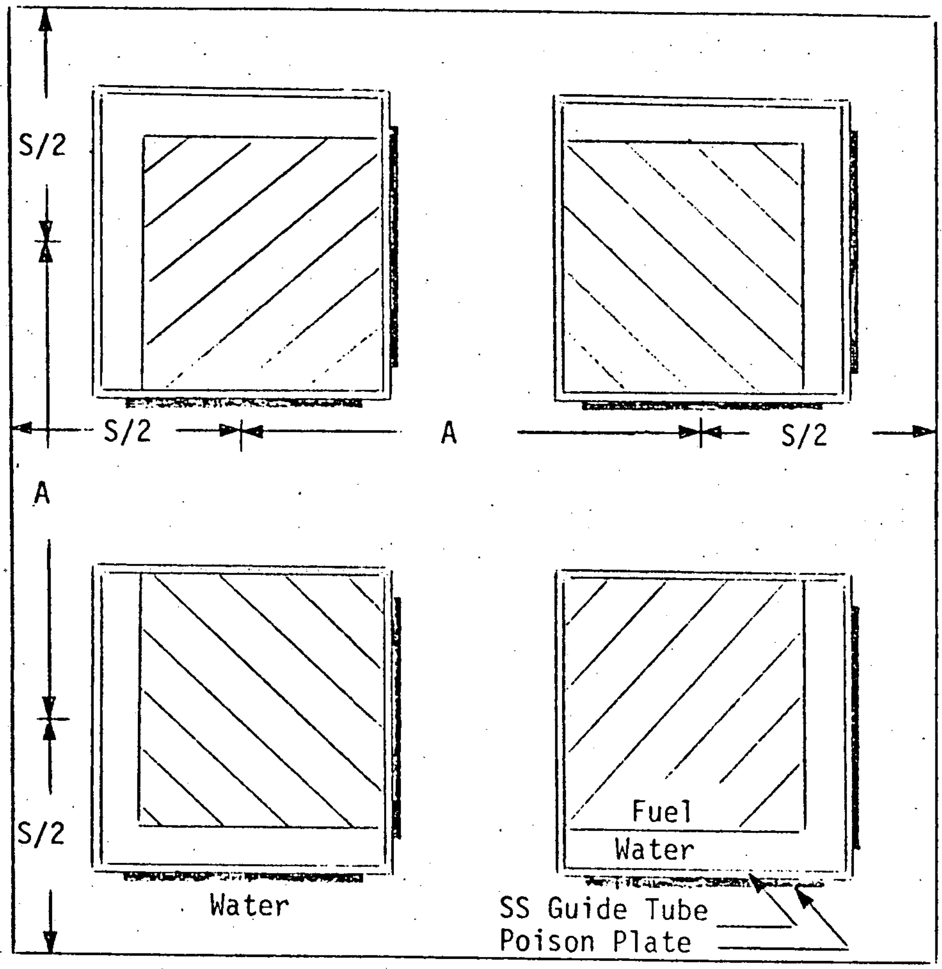
FIGURE 2.2.2



Type A Module Fuel Guides
12.0"
12.0"
Water
Fuel Assembly Guide Tube
Upper Section
Lower Section
14.0"
11.25"
12.0"
Poison Plates
Type B Module Fuel Guides

FIGURE 2.2.3

WORST CREDIBLE FUEL ASSEMBLY ARRANGEMENT
Type A Fuel Storage Module



DIMENSIONS (Guide Tube Center-to-Center Spacing)

S = 12.0"
Δs = 0.1875"
A = 11.8125"

Analysis of the natural convection cooling of spent fuel assemblies placed in the high density Exxon Nuclear Company, Inc. (ENC) spent fuel racks was conducted to demonstrate that acceptable fuel clad temperatures would be maintained. Supplemental analysis was also conducted to determine if the high density rack design would result in any significant increases to the spent fuel pool and cooling system heat loads, and ventilation system heat loads.

2.3.1 Natural Convection Cooling of a Single Fuel Assembly

The high density spent fuel storage rack design developed by ENC for the Indian Point Unit 3 Nuclear Power Plant provides 837 spent fuel assembly storage cells, plus 3 failed fuel assembly storage locations. The original fuel storage design provided 264 fuel storage cells. Because of the high density storage (compared to the original design) the design was reviewed to determine if adequate natural convection cooling is available during normal operation to (a) maintain fuel rod clad temperatures at acceptable levels; and (b) preclude boiling within the fuel assemblies. Fuel rod clad temperatures were also evaluated under hypothetical loss of forced coolant circulative conditions where the pool surface is assumed to reach a saturation temperature of 212°F.

Under normal refueling conditions, 1/3 of a core (64 assemblies) is discharged into the pool every 15 months. Under normal conditions, fuel assemblies are cooled within the reactor for 100 hours following reactor shutdown prior to insertion into the fuel storage racks. The heat generation rate for the spent fuel assembly 100 hours after reactor shutdown is calculated at 58.8 KW where the fuel assembly has spent 1,368 effective full power days (45 months) within the reactor. This calculation is based on NRC Branch Technical Position APCSB 9-2 and a reactor thermal power rating of 3,025 MWT for a core of 193 fuel assemblies. Decay heat from fission products is 51.8 KW and decay heat from heavy elements is 7.0 KW.

Thermal hydraulics analysis of the natural convection cooling of a single fuel assembly indicates that there is adequate cooling even under hypothetical conditions where a loss of forced coolant circulation is assumed to occur. This result is based on the two cases presented in Table 2.3.1.

The first case is the normal situation where the heat generation rate is 58.8 KW per assembly and the fuel cell inlet temperature is taken as 150°F. The fuel pin peak cladding temperature is 182°F, therefore there can be no boiling within the fuel assembly and the flow is single phase.

The second case is similar to the first except for the assumed inlet temperature, 239.2°F. This is the saturation temperature corresponding to the hydrostatic pressure at the top of the fuel cell. This is the maximum temperature that water flowing towards the fuel assembly inlet can attain under the hypothetical conditions where forced coolant circulation is assumed lost -- and the surface of the pool is assumed to reach 212°F which is the saturation condition at that location. Under these assumed conditions, boiling does occur in the upper portion of the fuel assembly. Maximum cladding temperature under this case is calculated at 247.1°F.

The effect on cooling of a partial flow blockage due to a dropped fuel assembly lying horizontally on top of the racks was also considered and found to be of no significance.

In summary, thermal hydraulics analysis indicates that even under hypothetical extreme conditions, peak clad temperatures are well below conditions where any degradation of the clad would occur. The spent fuel storage racks are considered safe from a thermal hydraulics standpoint based on this analysis.

A discussion of the methods used to arrive at the above result follows.

In order to perform conservative calculations for defining fuel rod cladding temperature, the following information was developed:

1. Maximum fuel heat generation rates per assembly and maximum local fuel rod heat generation rates;
2. Fuel assembly inlet temperature;
3. Flow resistance within the fuel assembly, and worst case flow resistance within the storage rack;
4. Heat flow resistance between the fuel clad and coolant.

Briefly, this information was obtained in the following way. Maximum heat generation rates per fuel assembly were based on NRC Branch Technical Position APCSB 9-2. Maximum local fuel rod heat generation rates were obtained by subsequently applying a conservative peaking factor of 1.6.

Fuel assembly inlet temperature was set at 150°F for the normal operating condition. Under the assumed loss of forced coolant circulation conditions, the inlet temperature was taken as local saturation temperature at the top of the fuel storage racks, 239.2°F. Flow paths and flow resistances were identified and computed on a worst case basis.

Heat flow resistance was calculated from classical and experimental relations developed for laminar, turbulent, and boiling flow regimes.

Using the above information, simultaneous solution of the continuity, momentum, and energy equations using a version of the COBRA code and subsequent hand calculations provided local and maximum spent fuel clad temperature, coolant temperature rise, mass flow rates, local pressures, and pressure drops within the fuel assembly. For the two phase flow situations which can occur during a loss of forced coolant circulation, two additional parameters, void fraction and quality, were also computed. The COBRA thermal hydraulics computer code is used extensively throughout the nuclear industry for thermal hydraulics analyses. Hand checks of the results of this code have been made on spent fuel storage rack thermal hydraulics analysis.

2.3.2 Spent Fuel Pool and Cooling System Heat Loads

The ENC high density fuel storage racks increase spent fuel storage capacity from 264 storage positions to 840 positions. This increase in storage capacity will bring an increase in the maximum spent fuel pool total decay heat load under normal and abnormal conditions. If heat losses due to evaporation and condensation through the pool walls are ignored, then heat loads to the spent fuel cooling system are the same as the spent fuel pool heat loads. Heat loads to the spent fuel cooling system for the ENC design have been calculated on this basis. Spent fuel pool heat loads and pool discharge temperatures for the original and high density spent fuel pool rack designs are provided in Table 2.3-2.

The original design pool heat load values are taken from Table 9.3-3, DOCKET-50286-29, Indian Point Nuclear Generating Unit No. 3, FSAR. The high density design heat load values were calculated using NRC Branch Technical Position APCSB 9-2 and the following assumptions:

1. High density design pool capacity -- 840 spent fuel positions (13 fuel regions).
2. Fuel discharge schedule.
   (a) Normal -- one region of 64 fuel assemblies (1/3 core) discharged every 15 months until pool is full, 100 hrs. cooling.
   (b) Abnormal -- ten regions discharged at the rate of one every 15 months, followed by a full core discharge after 400 hrs. cooling.

2.3.3 Spent Fuel Pool Cooling System Performance

The heat removal capability of the spent fuel cooling system has been calculated as a function of the spent fuel pool water temperature and is presented in Figure 2.3-3, "Bulk Temperature Versus Cooling Rate". The analysis is based on a spent fuel heat exchanger design heat transfer capability of 12.6×10^6 Btu/hr, and values of 88.2°F for the component cooling water temperature and 1.4 million pounds per hour for the component cooling water flow rate through the spent fuel heat exchanger.

Based on a worst case heat load of $26x10^6$ Btu/hr, with an instantaneous full core unload after a 400-hr. decay, and an initial pool temperature of 150°F, the bulk pool temperature will reach a peak temperature of 152.8°F in approximately twelve (12) hours and then decay gradually (see Figure 2.3-1). For a normal refueling operation where it is assumed that 1/3 of the core is discharged instantaneously after a 100-hr. decay, the bulk spent fuel pool temperature will reach a peak temperature of 128°F in approximately 20 hours and then decay gradually as shown in Figure 2.3-2.

2.3.4 Pool Heat Up Characteristics

Under the hypothetical conditions where a loss of forced circulation is assumed to occur, the pool water will experience a rise in temperature until boiling occurs.

Assuming no heat losses due to evaporation at the surface or conduction through the pool walls, the time required for the bulk temperature to reach 212°F, or the boiling condition at the surface, under maximum normal and abnormal heat load conditions is:

Discharge Condition	Initial Pool Bulk Temperature	Final Pool Bulk Temperature	Approximate Heat Up Time
Normal	128°F	212°F	11 hrs.
Abnormal	152.8°F	212°F	5 hrs.

At the present time the IP3 FSAR describes alternate connections to hook up a temporary pump in the event the principal fuel pool cooling pump should fail. It is our intention to permanently install a redundant cooling pump of sufficient capacity to maintain adequate pool cooling. This standby pump can be activated in sufficient time to maintain the integrity of fuel.

2.3.5 <u>Ventilation System Heat Load</u>

The heat loads to the ventilation system are dependent upon the surface temperature of the spent fuel pool and related evaporation rate of water from the pool. As indicated above, the maximum pool discharge temperature under the abnormal conditions described previously is calculated at 152.8^{O}F. The design spent fuel pool water temperature is 200^{O}F. The original fuel pool design indicated a peak temperature of 145.8^{O}F (refer Table 2.3-2) as compared with the proposed high density design peak temperature of 152.8^{O}F. This small increase in pool water temperature will result in an insignificant increase in the ventilation system heat loads.

<u>Section 2.3 References</u>

A. Indian Point Nuclear Generating Unit No. 3, Final Safety Analysis Report, Amendment 13, Consolidated Edison Co., December 4, 1970, Docket-50286-29, 635 p.

B. License Amendment Application to Permit Modification of the Indian Point Unit No. 3 Fuel Storage Pool. <u>Attachment</u> A. <u>Spent Fuel Pool Proposed Modifications.</u> Consolidated Edison of New York, Inc. Power Authority of the State of New York. Indian Point Station, Unit No. 3, Docket 50-286, June, 1976.

TABLE 2.3.1

THERMAL HYDRAULIC PARAMETERS FOR 58.8 Kw FUEL ASSEMBLY HEAT LOAD IN PROPOSED IP-3 SPENT FUEL RACK

FLOW TYPE	SINGLE PHASE	TWO PHASE
System Parameter	Case 1	Case 2
Cooling Loop Operational	Yes	No
Fuel Assembly Heat Generation rate, Kw	58.8	58.8
Fuel Assembly Coolant Bulk Inlet Temperature	150	239.2
Fuel Assembly Coolant Mass Flow Rate lb/hr.	11,210	46,774
Fuel Assembly Coolant Bulk Discharge Temperature, °F*	168	239.2
Bundle Coolant Bulk Max. Temp. °F	168	242.2
Fuel Pin Film Temp. Drop °F, Max.	14	4.9
Fuel Pin Peak Cladding Temp. °F	182	247.1
Equilibrium Quality*	0	.004
Void Fraction*	0	.297

* At top of assembly

TABLE 2.3-2
SPENT FUEL POOL HEAT LOADS
ORIGINAL DESIGN

Pool Inventory	Pool Heat Load--Btu/hr	Pool Discharge Temperature
1/3 core - normal discharge of one region	10×10^6	113.6°F
1-1/3 cores - normal discharge of one region, followed by full core discharge	23×10^6	145.8°F

HIGH DENSITY ENC DESIGN

Pool Inventory	Pool Heat Load--Btu/hr	Pool Discharge Temperature
4-1/3 cores - normal discharge of 13 regions, 100 hrs. cooling	17×10^6	128°F
4-1/3 cores - normal discharge of 10 regions, followed by full core discharge at the end of the 11th cycle	26×10^6	152.8°F

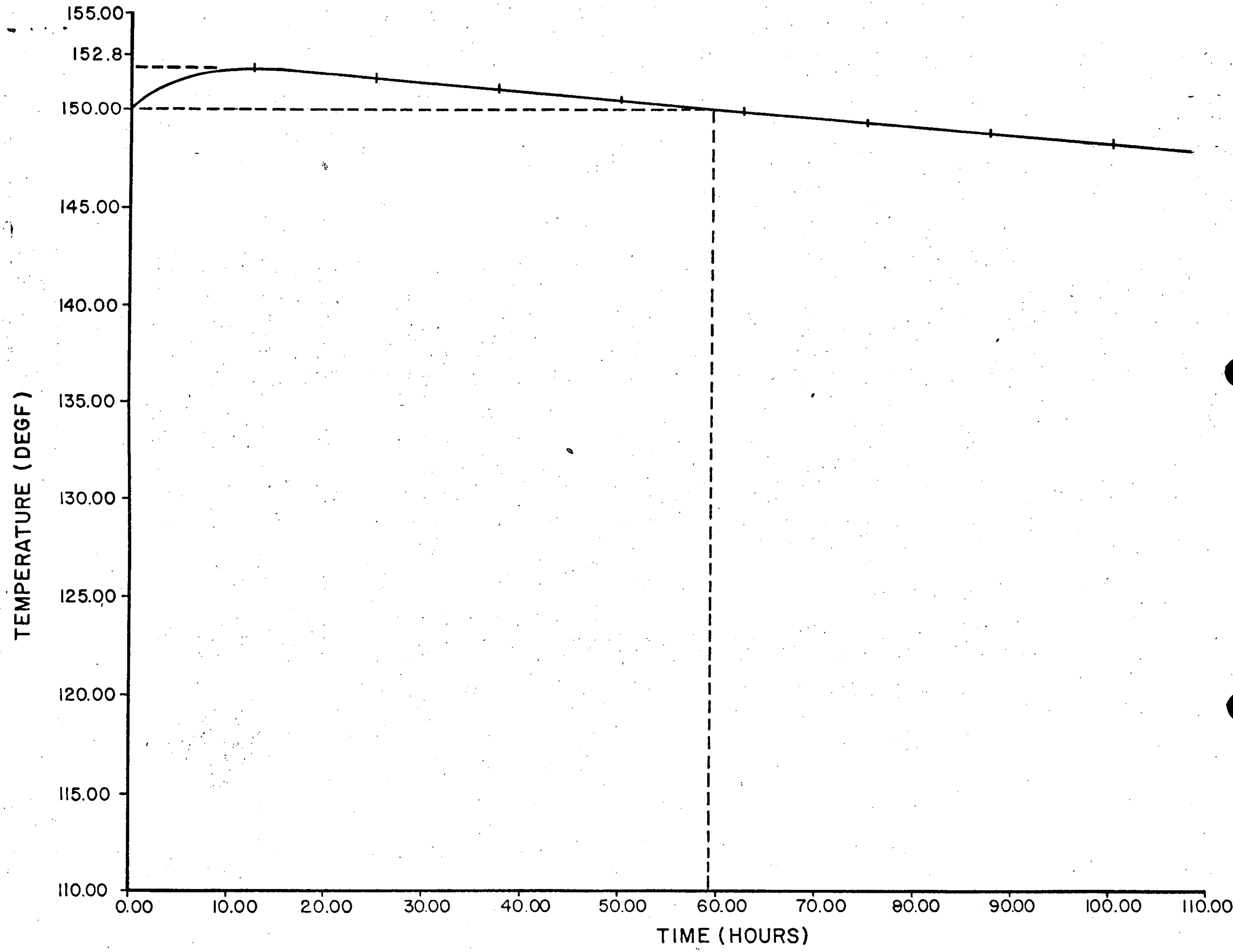


FIGURE 2.3-1 BULK POOL TEMPERATUTE TRANSIENT FOR INSTANTANEOUS CORE INSERTION (POOL TEMP. AT 150° F INITIALLY)

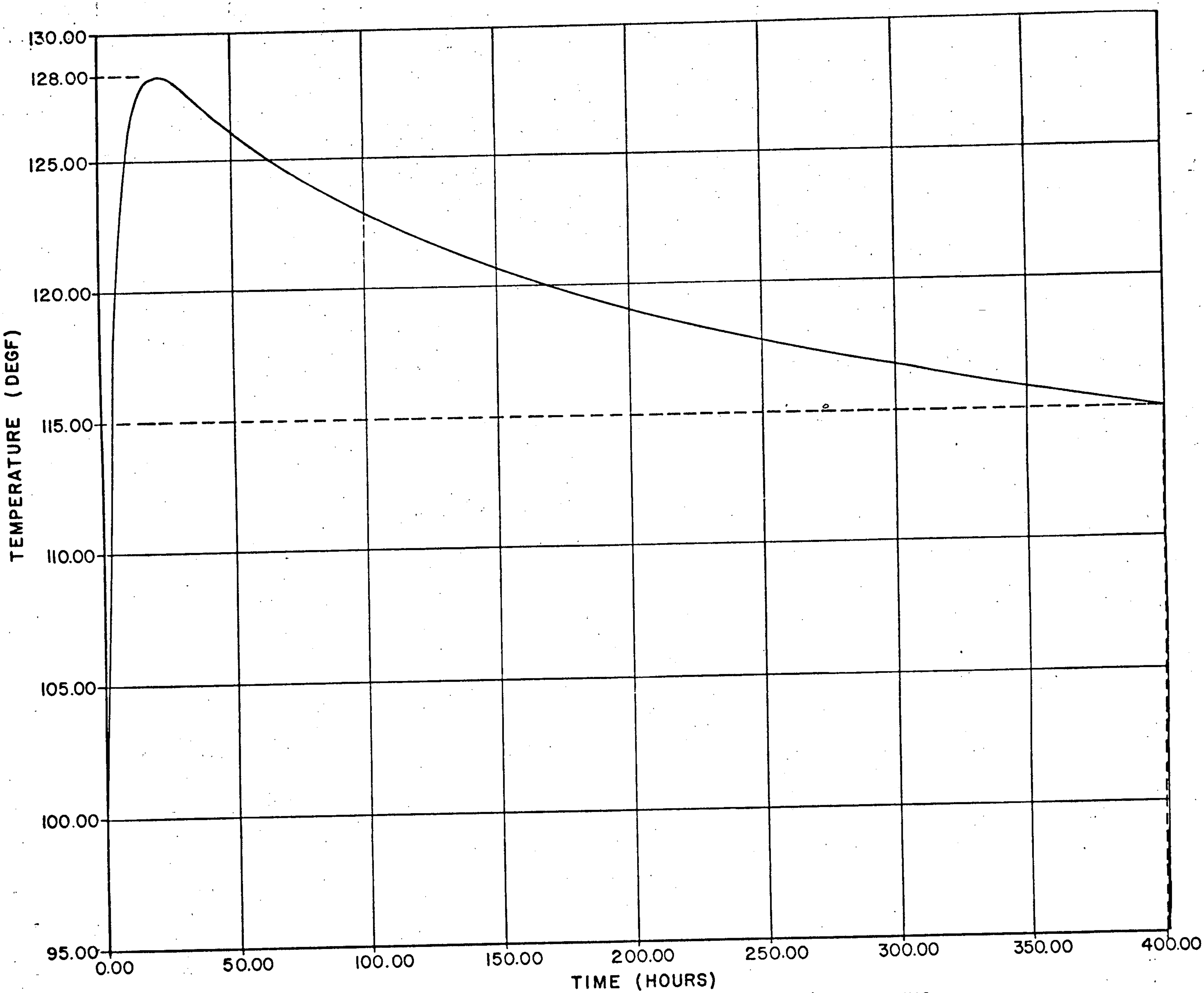


FIGURE 2.3-2 - BULK POOL TEMPERATURE TRANSIENT FOLLOWING THE 10th REFUELING

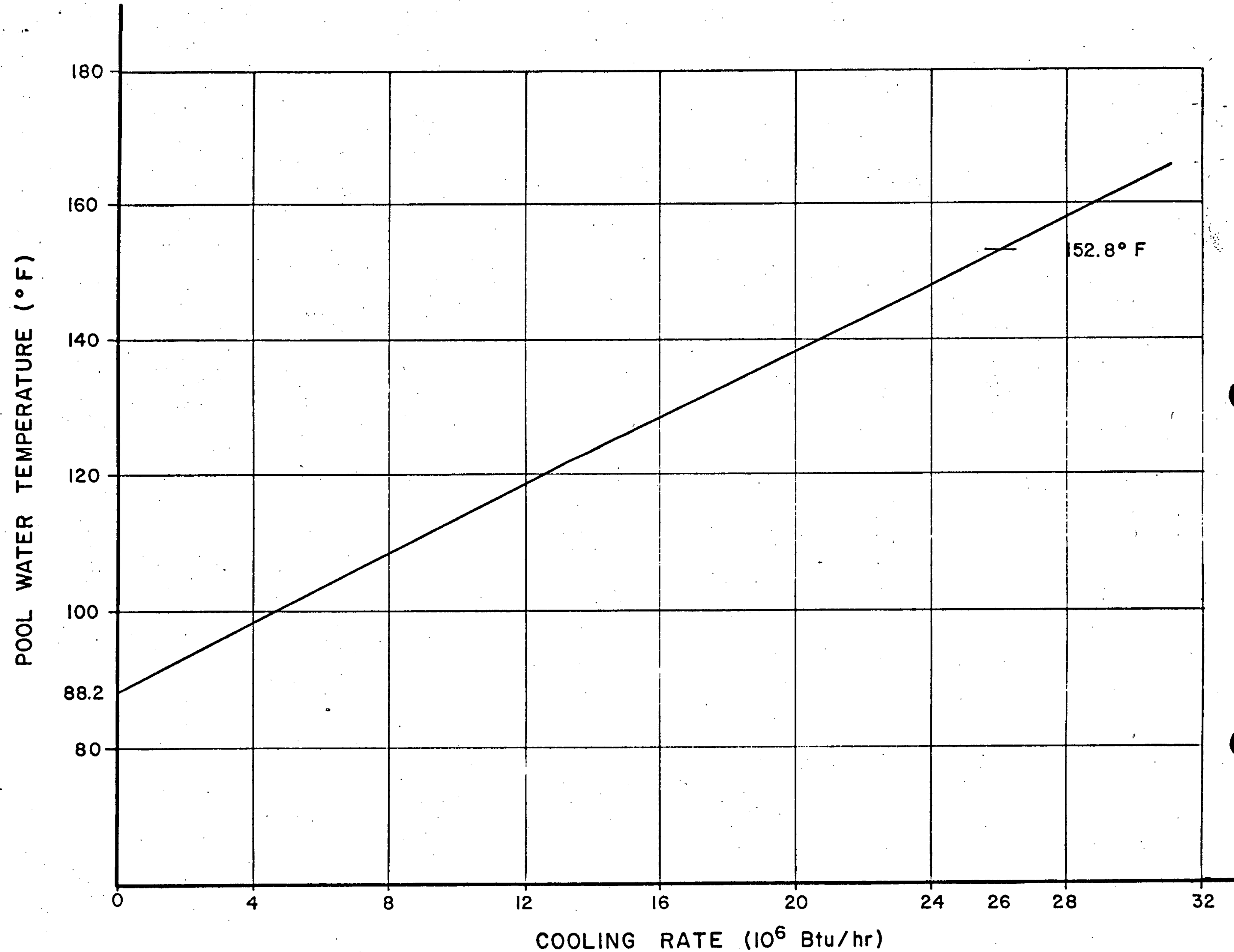


FIGURE 2.3.3 BULK POOL TEMPERATURE VERSUS COOLING RATE

2.4 Quality Assurance

The Power Authority of the State of New York has provided adequate controls for all the activities relative to the proposed modifications by means of a Quality Assurance Program. This program is identical to the one proposed to and accepted by NRC for the J.A. FitzPatrick Nuclear Power Plant, Docket No. 50-333.

3.0 SAFETY ANALYSIS

3.1 Radiological Effect

Radionuclide concentrations in the spent fuel pool were computed assuming reactor coolant activity corresponding to 1% and .2% failed fuel. These concentrations were based on a reactor coolant inventory contained in Table 9.2.5 of the Indian Point United No. 3 FSAR and a fuel element inventory based on Tables 14.2.1-2 and 14.2.1-4 of the Indian Point Unit No. 3 FSAR and Westinghouse document WCAP-7664. The analyses assumed 100 hour cleanup of the primary water prior to commencing refueling activities, instantaneous-uniform mixture of refueling water and reactor coolant, and an activity decay time of 100 hours. These concentrations are not expected to change significantly as a result of the proposed expansion of fuel element storage capabilities. Expected dose rates resulting from fuel-handling operations were computed using the above-mentioned radionuclide concentrations and treating the fuel pool as a unformly distributed gamma ray source. The radiation transport analyses and shielding calculations were made using the HDOSE and SHIELD Computer Codes. These codes have been utilized in a similar analysis for the Carolina Power & Light Brunswick Steam Electric Plant. Dose rates at the refueling platform three (3) feet above the pool surface have been computed for two cases. Case 1 determined the dose rates to personnel directly above the fuel pool during normal plant operation and Case 2 determined the dose rates to the crane operator on the refueling bridge while moving a fuel assembly. The results for Case 1 are tabulated as follows:

Summary of Exposure Rates (millirem/hr.)

Source	1% Failed Fuel	.2% Failed Fuel
Pool Water	2.5	0.5
Stored Fuel	<0.1	$\ll 0.1$
Fuel Assembly Under Transfer Operation	--	--
Total	2.5	0.5

The dose rates through the fuel pool wall were determined to be negligible.

It is expected that 3 to 6 man-shifts per day would be required in the fuel storage building during normal fuel-handling operations. Thus, the maximum exposure received by the general personnel due to 1% failed fuel during the expected three-week refueling period would be approximately 1.25 to 2.50 rem, respectively.

For Case 2 involving the dose rate to the crane operator during actual fuel assembly movement, the calculated dose rates for 1% and 0.2% failed fuel were 14.3 and 12.3 millirem/hr., respectively. Credit was not taken for the attenuation properties of structural materials such as the fuel assembly, end plug, grid plate and the crane platform. Furthermore, the attenuation by the uranium dioxide was conservatively substituted for in the analysis with water attenuation. Further, since the fuel assembly is actually in transit for such a short duration of time, it is not expected that the crane operators' maximum exposure would be significantly more than that received by the other workers in the fuel storage building. As a result of the conservative models used in both cases, coupled with the conservative assumptions used in the analyses, the total occupational exposures are basically unchanged from similar previous analyses done for the existing spent fuel storage racks at Indian Point 3 Nuclear Power Plant. The tritium release and hence airborne dose to the environment around the spent fuel pool is expected to change only as a function of the water evaporation rate. Since it has been demonstrated that the temperature of the fuel pool water remains essentially unchanged (approximately 150^{o}F), it can be concluded that the anticipated airborne dose rate due to tritium will not significantly increase.

3.1.1 Environmental Impact Evaluation

The beneficial and adverse environmental effects resulting from the implementation of the proposed modification in the spent fuel pool will not alter the results of the cost-benefit analysis originally performed by the US Nuclear Regulatory Commission for the Indian Point Nuclear Generating Plant Unit No. 3 (NUREG-75/002). No further environmental impact evaluation is deemed necessary.

The fuel storage contribution to off-site doses (10CFR50 App. I), are minimal and the expanded fuel storage is not expected to significantly affect the off-site release.

3.2 Accident Analysis

In addition to evaluating the proposed modification with respect to criticality and cooling considerations, postulated accidents involving spent fuel have been reviewed.

The Indian Point 3 FSAR Section 14.2.1 describes an analysis of four fuel-handling accidents:

a) a fuel assembly becomes stuck inside the reactor vessel

b) a fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pit

c) a fuel assembly becomes stuck in the penetration valve

d) a fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck

Accidents (a), (c) and (d) are not relevant to the design of the spent fuel racks. The accidental dropping of a fuel assembly into the spent fuel pit was found to be different in the case of the proposed spent fuel racks from that reported in Section 14.2.1 of the FSAR.

The IP3 Safety Evaluation Report, Section 15.3 considered the case of a fuel assembly dropped into the pool with the assumption that all fuel rods of that assembly were damaged. Calculated doses resulting from the release of fission product bases for these accidents were within the guideline values of 10CFR Part 100. The proposed spent fuel racks do not affect those analyses, thus the calculations remain valid and the conclusions applicable.

Possible fuel assembly damage due to cask drop accidents was not evaluated in the IP3 FSAR or the Safety Evaluation Report. However, the IP3 Technical Specification prohibits movement of spent fuel casks over spent fuel and require that all irradiated fuel stored in the spent fuel pool be in a subcritical condition for at least ninety days before a cask may be moved over any region of the pool. The latter restriction assures that, even in the event of an unlikely sideways cask drop resulting in damage to the maximum possible number of assemblies, the exposure limits of 10 CFR Part 100 would not be exceeded.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, T_{avg} shall be $\leq 140^{\circ}F$ and the minimum boron concentration sufficient to maintain the reactor subcritical by at least $10\% \Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

6. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

7. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 100 hours. In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before a continuous interval of 400 hours has elapsed after shutdown.

9. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

10. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test and prior to fuel handling. A test of interlocks shall also be performed.

11. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 100-hour decay time following the subcritical condition and the 23 feet of water above the top of the irradiated fuel assemblies are consistent with the assumptions used in the dose calculation for the fuel-handling accident.

The waiting time of 400 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR.

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite doses to within acceptable limits in the event of a fuel-handling accident. The system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel handling and is in a standby basis.

The minimum spent fuel pit boron concentration and the 90-day restriction of the movement of the spent fuel cask to allow the irradiated fuel to decay were specified in order to minimize the consequences of an unlikely sideways cask drop.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated on the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building crane.

ATTACHMENT B

APPLICATION FOR AMENDMENT TO
OPERATING LICENSE

Power Authority of the State of New York

Indian Point 3
Docket No. 50-286
Facility Operating License No. DPR-64

August, 1977

ATTACHMENT C

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Power Authority of the State of New York

Indian Point 3
Docket No. 50-286
Facility Operating License No. DPR-64

August, 1977

Safety Evaluation

Item 8:

Add, "In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before a continuous interval of 400 hours has elapsed after shutdown.

Safety Evaluation

For the case of a single region discharge, the existing waiting time requirement of 100 hours assures that the pool water temperature is well below the design objective. For a full-core discharge, the added requirement of 400 hours total waiting time will limit the decay heat generation rate in the spent fuel pool so that the pool water temperature will not exceed the FSAR design objective.

The decay heat calculation was performed in accordance with the NRC branch position paper (Auxiliary and Power Conversion Systems Branch Position, Section 9.2.5, Appendix A, Residual Decay Energy for Light-Water Reactors for Long-Term Cooling).

The proposed changes have been reviewed by the Plant Operating Review Committee and the Power Authority's Safety Review Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents produced.