SPENT FUEL POOL PROPOSED MODIFICATIONS

Consolidated Edison Company of New York, Inc.
Indian Point Station, Unit No. 2
Docket No. 50-247

May 9, 1975

TABLE OF CONTENTS

I. Introduction

The purposes of this report are to provide a detailed description of the proposed Indian Point Unit No. 2 spent fuel storage pool modification, and to furnish the additional information requested in the NRC letter of April 2, 1975.

II. Description of Proposed Modification

A. Reason for Modification

To provide increased spent fuel storage capacity for Indian Point Unit No. 2, the present spent fuel storage racks in Unit No. 2 will be replaced with new racks that store the fuel in a more closely spaced lattice. With the new racks, the storage capacity will be increased from 264 to 482 assemblies, an increase of 83%.

There will be no spent fuel in the Indian Point Unit No. 2 pool before February 16, 1976, the earliest expected time of the first scheduled discharge. Completion of the proposed modification before this date will permit the modification to be performed without the additional procedures and safety considerations that would be necessary if the pool contained spent fuel assemblies.

B. New Spent Fuel Rack Design

1. General Criteria

The new spent fuel racks will meet all relevant design criteria of ANSI Standard N18.2-1973

(Revised, August 1974) and draft ANSI Standard N210 (Revised, January 1975).

2. Description of Racks

The new racks will be similar in design to the present racks with the following exceptions:

a. The center-to-center spacing of the storage locations is reduced from the present 20.5 inches to 14 inches.

b. To insure adequate subcriticality margin with the reduced spacing, 1/8-inch thick boron-stainless steel plates, running the full-length of the active fuel region of an assembly, are installed on the sides of each storage location between storage positions.

The new rack concept is shown in Figure 1.

The racks rest on the stainless steel liner plate at the bottom of the storage pool. Each rack is seismically restrained at the bottom by two 4.5-inch diameter guide pins and at the top by removable plates which connect the rack to the adjacent racks. The sides of racks adjacent to the storage pool walls have kicker plates at the top to provide additional seismic stability.

3. Structural Design

The racks are designed to withstand the combined loadings of the dead weight of the rack structure, the weight of the spent fuel assemblies, and seismic loads. The racks are considered to be Class I (seismic).

All design is in accordance with the AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings, 1970. Stresses are within AISC working stress allowable for normal loading conditions (dead-load plus weight of spent fuel assemblies) and within 0.9Fy for the faulted condition (dead load plus weight of spent fuel assemblies plus safe shutdown earthquake).

4. Nuclear Criticality Design

The present Indian Point Unit No. 2 spent fuel storage racks maintain subcriticality by providing a center-to-center spacing of 20.5 inches between assemblies. In the proposed new spent fuel storage racks, the center-to-center spacing will be reduced to 14.0 inches. The increase in reactivity caused by the reduction in spacing will be offset by using fixed neutron absorber plates consisting of equivalent 304 stainless steel with a natural boron content of 1.0 to 1.2 percent by weight. All the fuel storage locations, except the outermost which are adjacent to the pool liner, will have a neutron absorber plate welded to each side. The outermost storage locations will have neutron absorber plates only on those sides facing other storage locations. The neutron absorber plates will run the full length of the active fuel region.

Nuclear criticality analyses were carried out to ensure that the new spent fuel racks will remain subcritical by an adequate margin even under conservative assumptions. The results demonstrate that the nuclear design of the proposed racks meets the current Technical Specification limit (Reference 1) and satisfies the current ANSI standard (Reference 2).

The conservative assumptions used for the design case calculations are:

a) Fresh unirradiated 3.5 w/o U-235 fuel (present Technical Specifications limit is 3.4 w/o).

b) Water temperature of 68°F

c) Minimum boron content of the boron stainless steel (1.0 w/o).

d) Minimum dimensions allowed in the fabrication of the boron stainless steel plates (1/8" x 7" x 145").

e) Center-to-center spacing of 13.875 inches which includes fabrication tolerance (instead of 14.0 inches which is the design specification).

f) Zircaloy guide thimbles instead of the stainless steel guide thimbles used in the present fuel assembly design.

g) No axial or radial neutron leakage (infinite medium calculation).

h) No soluble boron.

The criticality analysis was performed using a two-dimensional discrete ordinate transport theory computer code, DOT (Reference 3). This code employed three broad group cross-sections, two fast and one thermal, which were obtained using GAM II (Reference 4) and THERMOS (Reference 5), respectively. The calculated k_∞ for the above design case is 0.874.

A separate and independent calculation by Westinghouse provided a k_∞ of 0.87, agreeing with the DOT result. The Westinghouse analysis used two-dimensional diffusion theory (Reference 6) and blackness theory for the neutron absorber plates.

C. References

1. Docket No. 50-247, Technical Specifications, Final Facility Description and Safety Analyses Report, Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2.

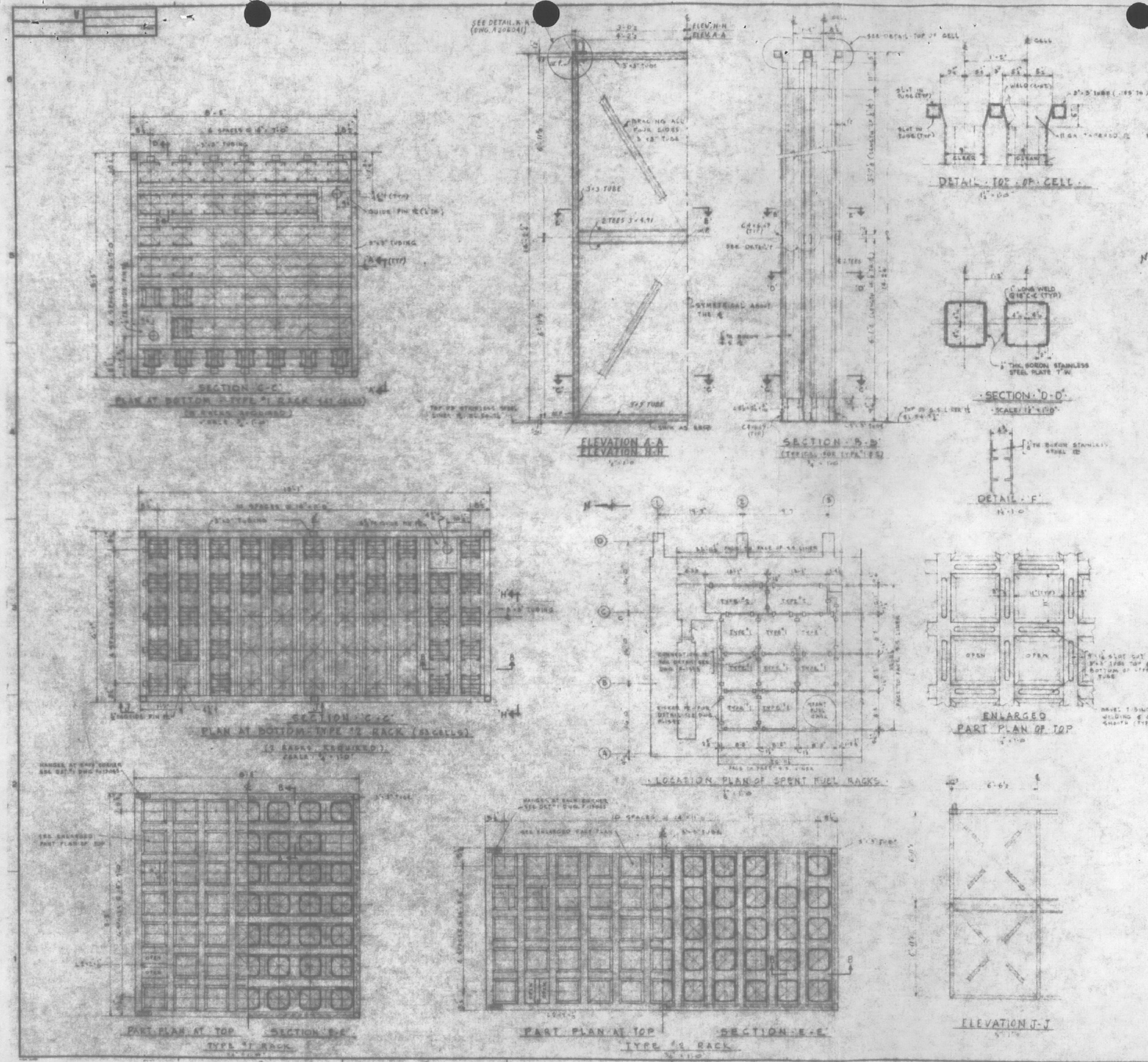
2. ANSI N210, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", (Revised January 1975).

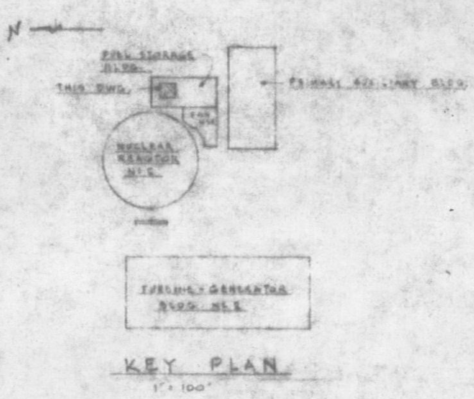
3. K-1694, "A User's Manual for DOT (a Two-Dimensional Discrete Ordinate Transport Code With Anisotropic Scattering, "F. R. Mynatt, (1967).

4. GA-4265, "GAMII, A B_3 Code for the Calculation of Fast Neutron Spectra and Associated Multi-group Constants", G. D. Joanou and J. S. Dudek, (1963).

5. BNL-5826, "A Thermalization Transport Theory Code for Reactor Lattice Calculations", H. Honeck, (1962).

6. WAPD-TM-678, "PDQ-7 Reference Manual", W. R. Cadwell, (January 1967).





NOTES:

1- RACK UNITS TO BE COMPLETELY ASSEMBLED IN FABRICATORS SHOP.
2- RACKS TO BE ALL WELDED CONSTRUCTION. ALL WELDING SHALL CONFORM TO THE LATEST REVISION OF THE A.W.S. CODE PERTAINING TO STAINLESS STEEL.
3- RACKS AND CELLS TO HAVE A TOLERANCE OF ±1/16" IN ALL DIMENSIONS AND ALIGNMENT TOTAL PLUMB 0'-05'.
4- WHERE LIGHTS OCCUR OVER CELLS THESE CELLS CANNOT BE USED FOR SPENT FUEL STORAGE.
5- FIXED SUPPORTS FOR LIGHTS LIMITS VERTICAL MOVEMENT OF RACKS NO 1, 2, 9 & 10 ONLY. PRIOR TO HANDLING SUBJECT RACKS:
   A- REMOVE RACKS NO 3, 4, 5, 6, & 7
   B- LIFT RACKS NO 1, 2, 9 OR 10 VERTICALLY A MIN. OF 9", MAX OF 14" PRIOR TO MOVING RACKS LATERALLY TO CLEAR LIGHT GUIDE & SUPPORT.
6- INSIDE FACE OF FUEL RACKS SHALL BE FREE OF ALL BURRS, ROUGH EDGES, AND TO BE SMOOTH AND CLEAN.
7- ALL MATERIAL SHALL BE STAINLESS STEEL TYPE 304, ASTM GRADE:
   A) A-240 PLATES, SHEETS
   B) A-276 ANGLES, SPECIAL SECTIONS
   C) A-554 TUBING
8- FOR COVER PLATE DETAILS SEE DWG. F-15085
9- EXIST GUIDE PINS FOR TYPE "2 RACKS MUST BE RELOCATED AND WELDED TO STAINLESS STEEL LINER

"CLASS A" ITEMS

QA-300-1

REFERENCE DWGS.

U.E.&C. DWG No 9321-F-1304 FUEL STORAGE BLDG. EXISTING SPENT FUEL RACKS - SHEET #1
9321-F-1305 FUEL STORAGE BLDG. EXISTING SPENT FUEL RACKS - SHEET #2
9321-F-1348 - SPENT FUEL RACKS - SHEET #3
9321-F-1593 - SPENT FUEL RACKS - MANIPULATOR TOOL - ARRGT & DETAILS
9321-F-5083 - SPENT FUEL RACKS - SHEET No 4

A206040

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT
INCREASED STORAGE CAPACITY SPENT FUEL STORAGE RACKS

JOB 4642

III. Additional Information Requested by Division of Reactor Licensing Letter Dated April 2, 1975

A. Criticality Considerations

1. K_{eff} vs. spacing of assemblies should be tabulated or graphed in the region of the 14-inch design spacing to show that movement of assemblies in the racks will not create a criticality problem.

Response

The results of the parametric study performed with diffusion theory and blackness theory are tabulated as follows:

Center-to-Center Spacing (Inches)	k_∞ (1.0 w/o Boron)
13.875	0.87
13.275	0.88
12.000	0.92

The 13.875 inch spacing is representative of the design 14-inch center-to-center spacing with allowance made for fabrication tolerances. The 13.275 inch spacing is the minimum center-to-center spacing possible between two assemblies based on allowance for fabrication tolerances and assembly movement within the storage locations. Realistically, if an assembly moves towards one neighbor, it must lean away from some other neighbor. Neglecting this realism for conservative reasons, an infinite array with a center-to-center spacing of 13.275 inches produces the k_∞ of 0.88 that is cited above. It should be noted that, even at a spacing of 12 inches, the calculated reactivity is still less than the value of 0.95 which is the current ANSI standard (ANSI N210, Revised January 1975).

A. Criticality Considerations

2. K_{eff} vs. boron loading in the structural steel of the new racks should be graphed or tabulated using conservative assumptions such that any fabrication deviations in boron concentrations will be adequately considered.

Response

The parametric study of the effect of boron concentration on reactivity employed transport theory. The results are tabulated as follows:

Natural Boron Content (Weight Percent)	k_∞ (at 13.875" Spacing and 68°F)
1.0	0.874
0.9	0.878
0.75	0.885
No Neutron Absorber Plates	0.950

The boron stainless steel will be produced with a natural boron content of 1.0 to 1.2 percent by weight. A strict quality assurance program will ensure that any 5 gram sample contains at least 1.0 weight percent of natural boron.

A. Criticality Considerations

3. K_{eff} vs. water temperature from room temperature up to the maximum expected temperatures in the pool should be graphed or tabulated.

Response

The temperature effect on reactivity has been analyzed using transport theory and the results are tabulated as follows:

Water Temperature (°F)	k_∞ (at 13.875" Spacing, and 1.0 w/o Boron
68	0.874
120	0.872
150	0.869
200	0.864

This clearly demonstrates the presence of a negative moderator temperature coefficient.

A. Criticality Considerations

4. An analysis of uncertainties in the calculations should be provided to identify or demonstrate the worst-case situations.

Response

Uncertainties and tolerances in enrichment, rack material, and rack fabrication were all considered in the reactivity calculation of the design case. A computational uncertainty of 0.1% $\Delta\rho$ and a standard deviation of 0.85% $\Delta\rho$ were reported by Westinghouse for the k_∞ calculations obtained using two-dimensional diffusion theory and blackness theory. These values are based upon the Westinghouse analysis of critical experiments involving poisoned as well as unpoisoned cases. Total calculational uncertainty based upon the arithmetical sum of these component uncertainties is equal to 0.95% $\Delta\rho$.

A. Criticality Considerations

5. An analysis of failures and/or accidents involving spent fuel must be provided. As an example, a re-analysis of the design basis fuel-handling accident should be provided.

Response

The Indian Point Unit No. 2 FSAR Section 14.2.1 describes an analysis of four fuel-handling accidents:

a) a fuel assembly becomes stuck inside 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. Accident (b), the accidental dropping of a fuel assembly into the spent fuel pit, is no different with the proposed spent fuel racks from that reported in Section 14.2.1 of the FSAR.

The Indian Point Unit No. 2 Safety Evaluation Report Sections 11.3 and 11.4 considered the case of a fuel assembly dropped into the pool with the assumption that all fuel rods of that assembly were damaged. This document reported that the calculated doses resulting from the release of fission product gases were within the guideline values of 10CFR Part 100. The new proposed spent fuel racks in the pool do not affect the analysis of the dropped fuel assembly. The calculation remains valid and the conclusions remain applicable.

B. Seismic Considerations

1. Provide an analysis which shows that the fuel storage racks can withstand a design basis earthquake with all the storage spaces filled with fuel assemblies and with damage to fuel assemblies at an acceptably low level.

Response

The seismic design of each spent fuel rack is based on limiting stresses in the structural elements of the rack to 0.9Fy under the combined loading of the dead weight of the rack, the weight of the spent fuel assemblies and the Safe Shutdown Earthquake.

The weight of the fuel assemblies stored in the rack is conservatively calculated assuming that control rods are inserted in each assembly.

The racks sit on the bottom of the Spent Fuel Storage Pool which is a 3' thick reinforced concrete mat poured directly on bedrock. The seismic acceleration applied to the racks therefore is the same as the site ground acceleration, and the ground response spectrum curves shown on Figures A.1-1 and A.1-2 of the Indian Point Unit No. 2 FSAR are used in the vertical and horizontal directions, respectively. The vertical and horizontal earthquake forces are assumed to act simultaneously.

The seismic forces on the storage racks, were determined by URS/John A. Blume and Associates. Of special concern was the effects of water pressure on the racks since the racks will be fully submerged under approximately 25 feet of water when functioning.

Based on a review of the literature, Blume advised that there is an added mass effect due to the water pressure acting on the rack structure. This effect is significant (in proportion to the weight of the structure) in the horizontal direction, and is relatively insignificant in the vertical direction.

Blume also noted that the effects of the water will increase the effective damping of the rack structure. Whereas 1% damping would normally be applicable for a welded steel structure, the damping value would increase to 2% on the same structure underwater.

Therefore, to determine lateral forces on the rack, the 2% damping curve was used considering the total weight of the rack structure, the fuel assemblies and the added mass effect of the water. Vertical forces were determined using the actual weight of the rack structure and the fuel assemblies ignoring the insignificant added mass effects of the water. The 1% damping curve was used in the vertical direction. The added mass effect was computed by Blume on the information given in References 1, 2 and 3. The increased damping effect is based on Reference 2.

To obtain natural periods, the rack structure was analyzed as a single degree of freedom system for both directions. The racks are basically considered as a cantilever structure supported laterally at the base by friction and by two 4.5-inch diameter guide pins.

The fundamental mode period in the horizontal direction is less than 0.15 seconds. Higher mode periods are 1/3 of the fundamental period and less. The stiffness of the fuel assembly is neglected but the weight of the assembly is considered to be uniformly distributed in the cell.

In the vertical direction, the fuel cells and the perimeter frames are very rigid. The horizontal diaphragms are the only flexible parts. Conservatively considering only the bottom diaphragm and applying the entire weight of the rack to it, the fundamental period obtained is 0.18 second.

Based on the above, it was conservatively assumed that the acceleration response of the rack structure as a whole is equal to the peak of the response spectrum - i.e. 0.35g horizontally (peak of 2% curve) and 0.30g vertically (peak Of 1% curve).

The static working stress analysis of the racks assumed each row of cells to be behaving as a Vierendeel Truss. While the rack is actually a 14'-2" deep truss with a top, middle and bottom chord, only the lower half of the rack was used in determining the vertical load capacity of the structure. The bottom chord of the truss is comprised of the horizontal channels at the base of the rack and the top chord is the horizontal channels at the mid-height of the rack. (See Figure B-1).

Using the STRESS computer program (Ref.4), forces and moments in the truss due to the weight of the rack and fuel assemblies were determined. Stresses were limited to less than 18,000 psi for non-compact shapes and 20,000 psi for compact shapes.

The vertical seismic load will increase the moments, forces and stresses by 30%.

The horizontal component of the earthquake transmitted to the top of the structure is carried by the tubing (which acts as a diaphragm) to the diagonal bracing down to the base of the structure. The total horizontal shear can be resisted at the base by friction and by the two 4.5-inch guide pins.

Combined stresses in members due to simultaneous vertical dead, live and seismic forces and horizontal seismic forces are limited to 0.9Fy.

Overturning forces on the racks consist of the horizontal seismic force applied at the center of gravity and a 0.3g vertical force upward. The 0.3g force is multiplied by the actual, dry weight of the rack.

The stabilizing force is taken as the buoyant weight of the rack and fuel assemblies.

As a result, there is a net overturning force on the rack. Stability is achieved by tying the tops of each rack to one another using removable plates. The sides of the racks adjacent to the storage pool walls have kicker plates at the top to brace the racks against the wall after the tops have displaced laterally by more than 0.125 inches.

The spent fuel assemblies will be confined in individual cells which comply with the fuel manufacturer's recommended arrangement, and the maximum seismic loads imparted to the fuel assemblies by the new racks will be no greater than calculated for the original racks. Therefore, the new storage racks will provide the same protection against damage to the fuel due to the design basis earthquake as intended by the original design.

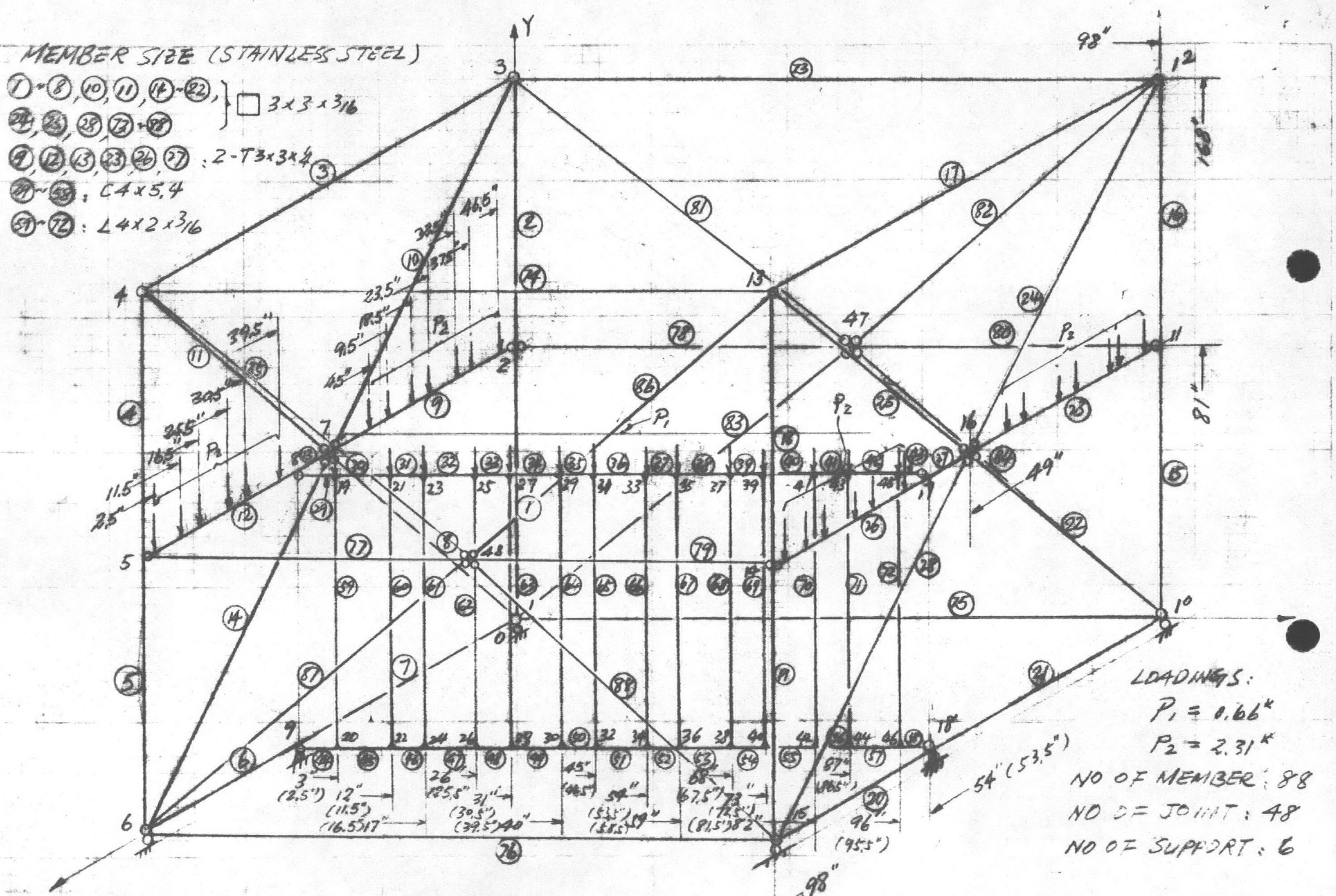
References

1. Stetson, T.E. and Mavis, "Virtual Mass and Acceleration in Fluids" Trans. ASCE, Volume 122, 1957

2. Clough, Ray W., "Effects of Earthquakes on Underwater Structures" Proceedings II WCEE, Tokyo and Kyoto, Japan, July 1960.

3. Chandrasckaran, A.R. et al, "Virtual Mass of Submerged Structures" Proc. ASCE, Volume 98, No. HY5, May 1972

4. "STRESS" - A structural program by S.J. Fenjes, R.D. Logcher, S.P. Mauch, And R.F. Reinschmidt, M.I.T.



SPACE FRAME

FIGURE No. B-1

-20-

B. Seismic Considerations

2. Provide an analysis of the storage pool with respect to the design basis earthquake, assuming that all storage spaces are filled with fuel assemblies.

Response

As noted in the answer to Question B.1, the bottom of the Spent Fuel Storage Pool is a 3-foot thick reinforced concrete mat poured directly on bedrock.

The total weight of a completely filled rack structure is approximately 92 kips. Adding 28 kips due to a vertical downward seismic force, a total load of 120 kips is generally distributed over an area 8'-5" x 8'-5" (70 square feet). The shims provided to level each rack are of sufficient size and number to prevent local crushing of the concrete immediately beneath the exterior frame of the rack where the shims are located. The resulting compressive force carried through the mat into the bedrock is less than 2 kips/square foot.

The impact of the top of the racks against the 4'-6" thick pool walls was found to produce stresses less than 3.5 KSI in the wall reinforcing steel, which has a minimum yield strength of 60 KSI.

C. Cooling Considerations

1. Provide an analysis which shows that the total heat load of the spent fuel can be dissipated by the pool cooling system. Assume that all fuel storage spaces are occupied and show that a conservative assumption of fuel heat is used in the calculations.

Response

The maximum spent fuel heat loads have been calculated for conservatively selected normal and abnormal cases, and are presented in Figures C-1 and C-2. The normal case corresponds to the discharge of one region at approximately fifteen month intervals until all storage locations are occupied. The abnormal case refers to the discharge of a full core with four regions of spent fuel already present in the pool, at which time all locations will then be occupied. The heat loads were calculated using the latest ANS decay heat curves (Reference 1). The calculations assume fifteen months of 100% power operation per cycle and include actinide decay heat. For the normal case it was conservatively assumed that the individual region discharge takes place one hundred hours after reactor shutdown. For the abnormal case, it was conservatively assumed that the full core discharge takes place at the end of the cycle when all three regions can contribute decay heat, with the fuel being moved into the pool four hundred hours after reactor shutdown.

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 C-3. The analysis is based on the conservative FSAR values of 91.3°F for the component cooling water temperature and 1.1 million pounds/hr for the component cooling water flow rate through the spent fuel heat exchanger.

With maximum heat loads, the maximum pool water temperature will be 125°F and 139°F for the normal and abnormal cases, respectively.

Reference

1. ANS standard (proposed) N18.6, Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors (Revised October, 1973)

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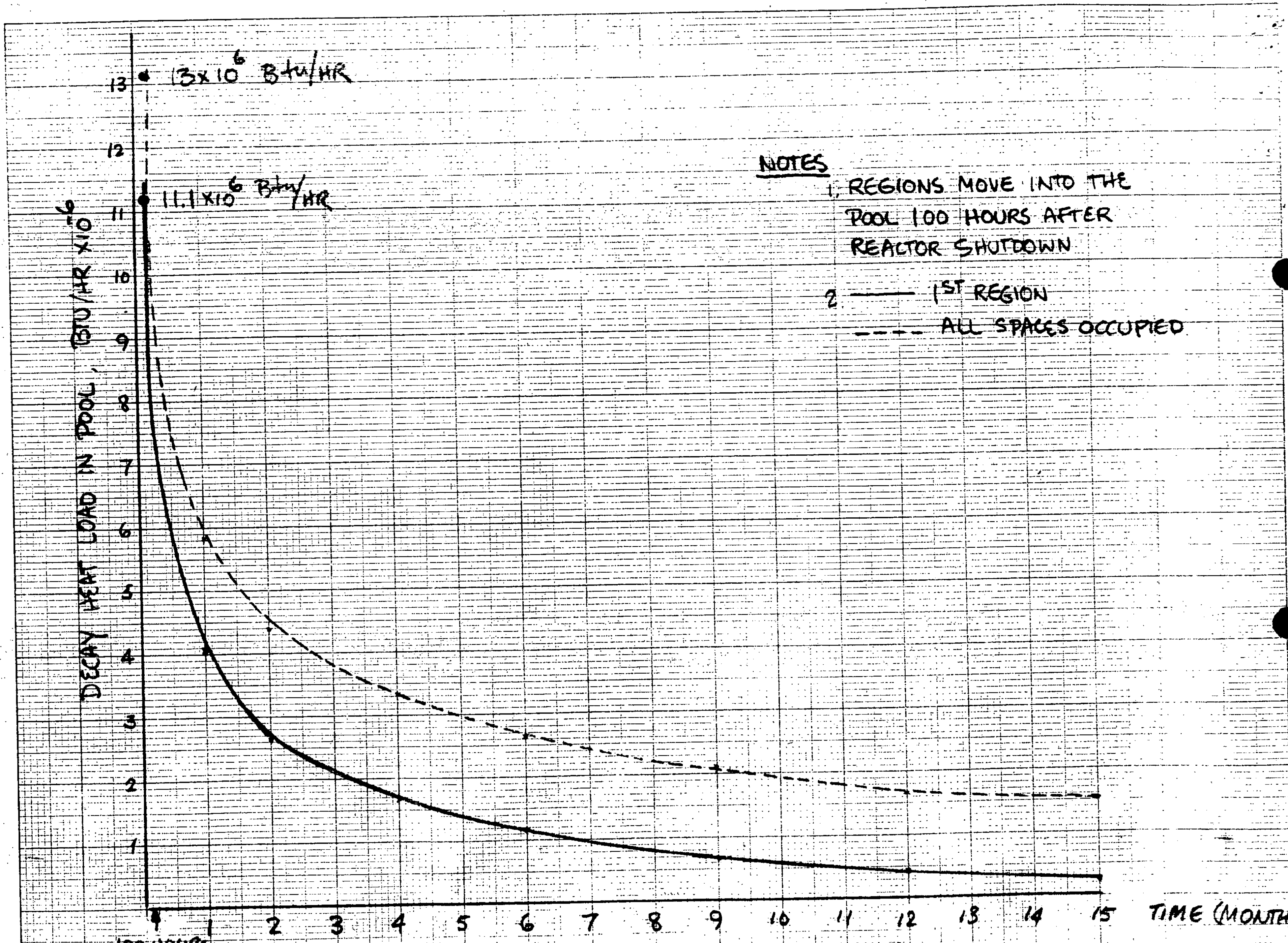

FIG. C-1. HEAT LOAD FOR THE NORMAL CASE.

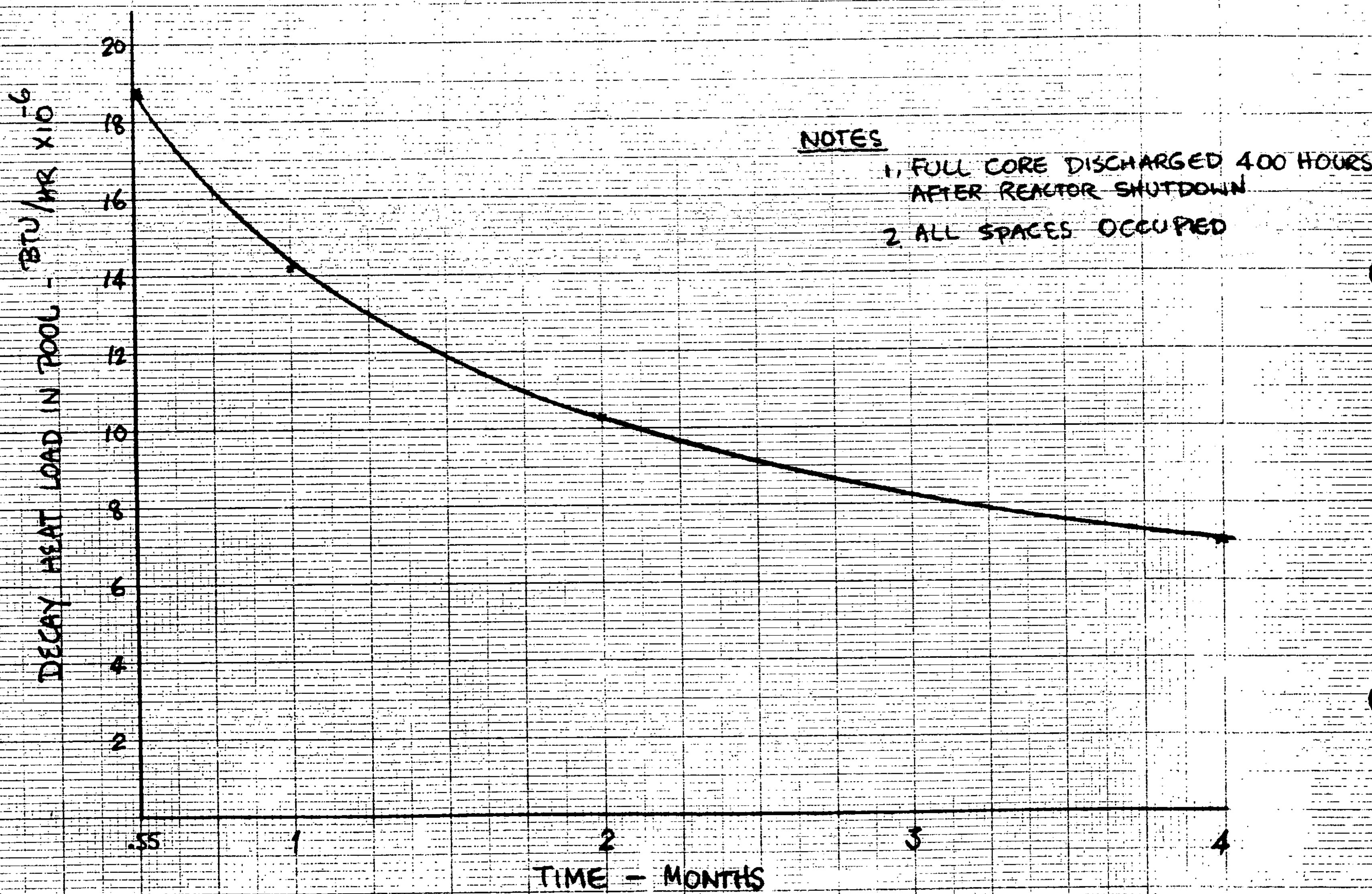

FIG. C-2. HEAT LOAD FOR THE ABNORMAL CASE.

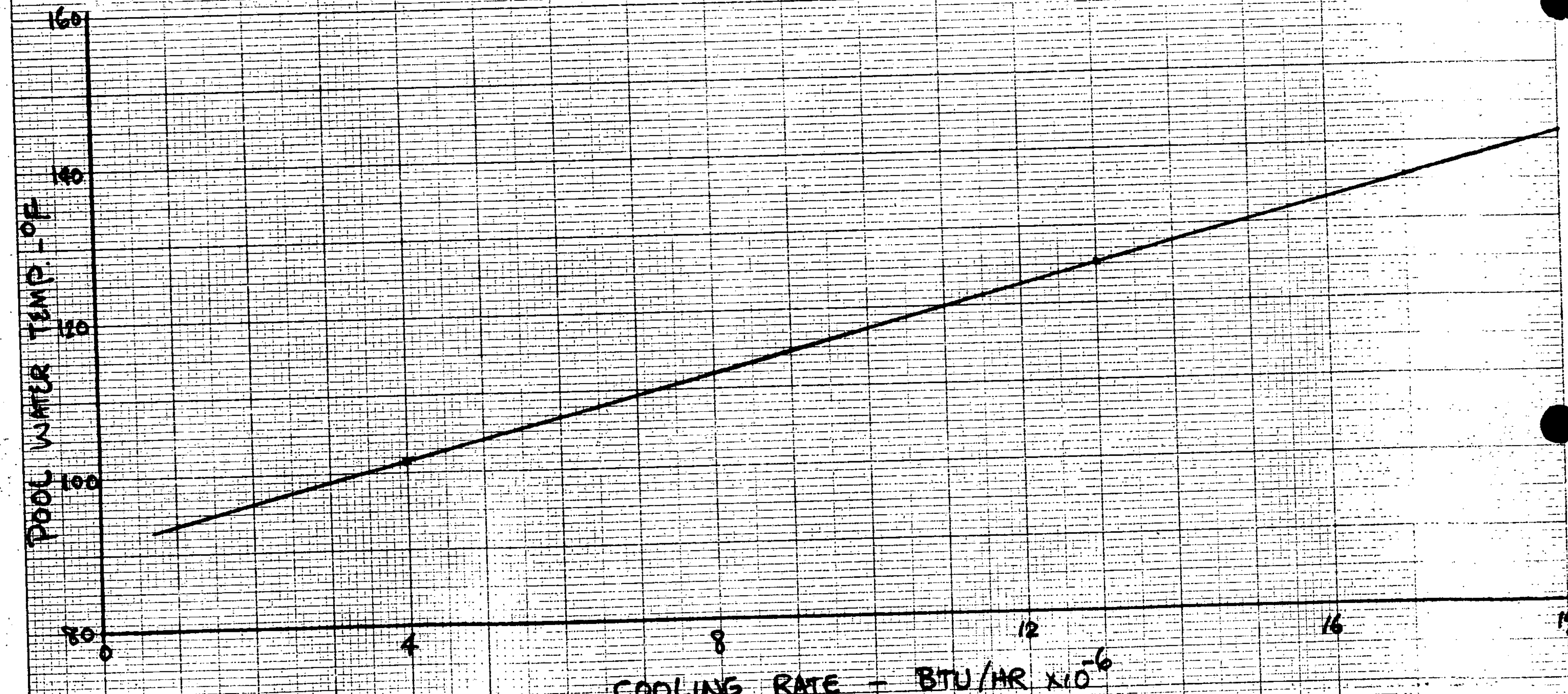

FIG. C-3. COOLING CAPABILITY OF THE SPENT FUEL COOLING SYSTEM

C. Cooling Considerations

2. Provide an analysis that shows that the spent fuel will be adequately cooled for the design conditions and for abnormal conditions.

Response

At the maximum heat load for the normal and abnormal conditions, the pool water temperature will not exceed 125°F and 139°F, respectively.

To assure adequate cooling of each fuel assembly, natural flow paths were considered in the rack design. Sufficient downcomer area exists between storage locations and at the top of the racks between lead-in funnels, and ample inlet area has been provided at the bottom of each storage location to permit adequate flow to each fuel assembly.

C. Cooling Considerations

3. Provide an analysis of the spent fuel pool heatup time using conservative fuel heat load assumptions. List alternate means of cooling, state the times needed to activate the alternative cooling systems, describe their heat removal capability and show that the alternate cooling systems would adequately cool the spent fuel.

Response

Pool temperature as a function of time in the absence of external cooling is presented in Figures C-4 and C-5 for the normal and abnormal cases described in response to Item C(1). These times are calculated for the same conservative assumptions as in C(1). Figures C-4 and C-5 show that the pool water temperature would rise to 180°F in 9 1/2 hours for the normal case and 5 hours for the core discharge case.

At the present time, the Unit No. 2 FSAR describes alternate connections to hook up a temporary pump in the event the spent fuel pool cooling pump should fail. It is our intention to permanently install a standby pump of sufficient capacity to maintain the maximum pool water temperature within 150°F for either heat load case. This standby pump can be activated within one hour following failure of the normal pump.

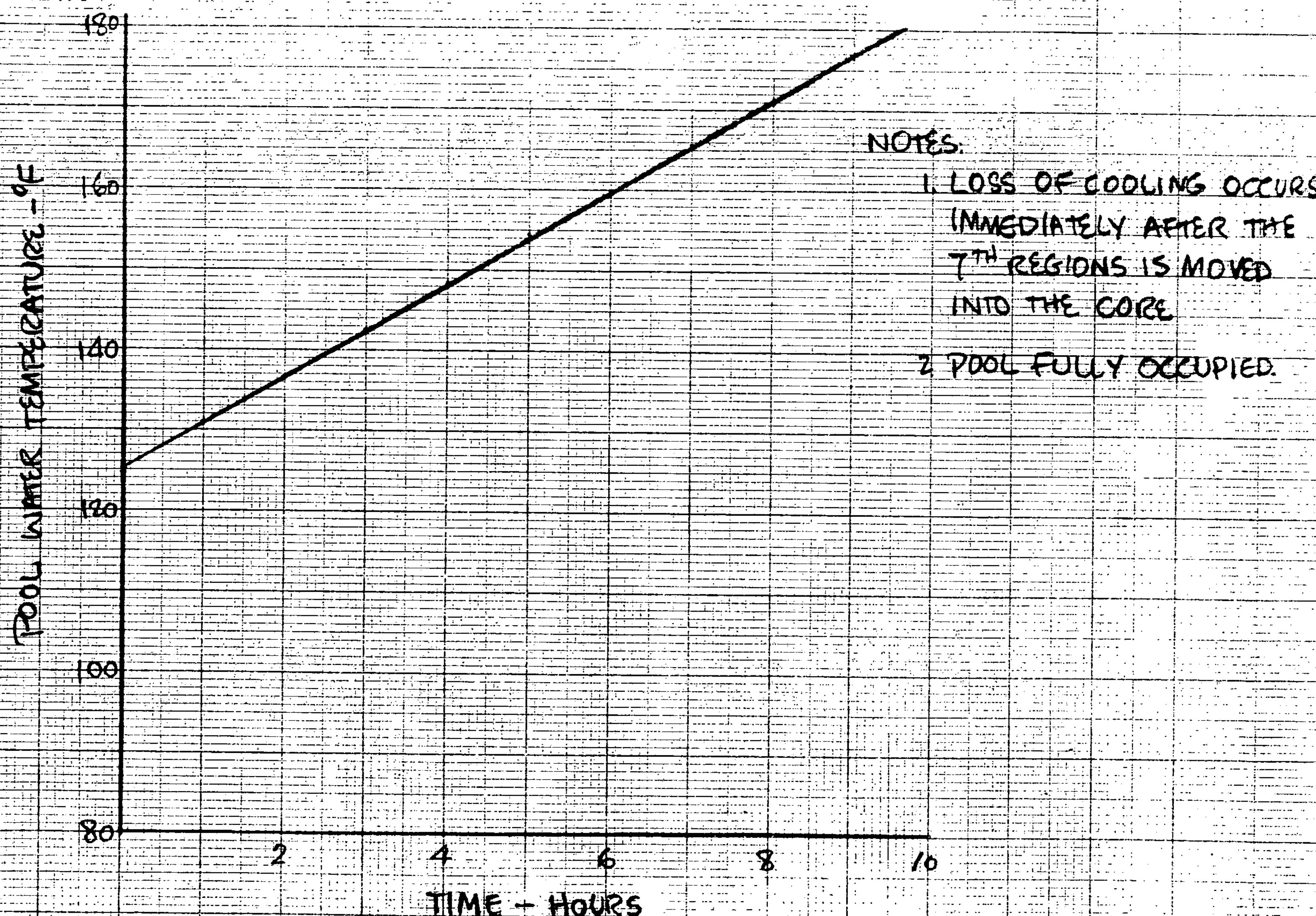

FIG. C-4. TRANSIENT HEAT-UP OF POOL WATER TEMPERATURE WITH NO COOLING FOR THE NORMAL CASE

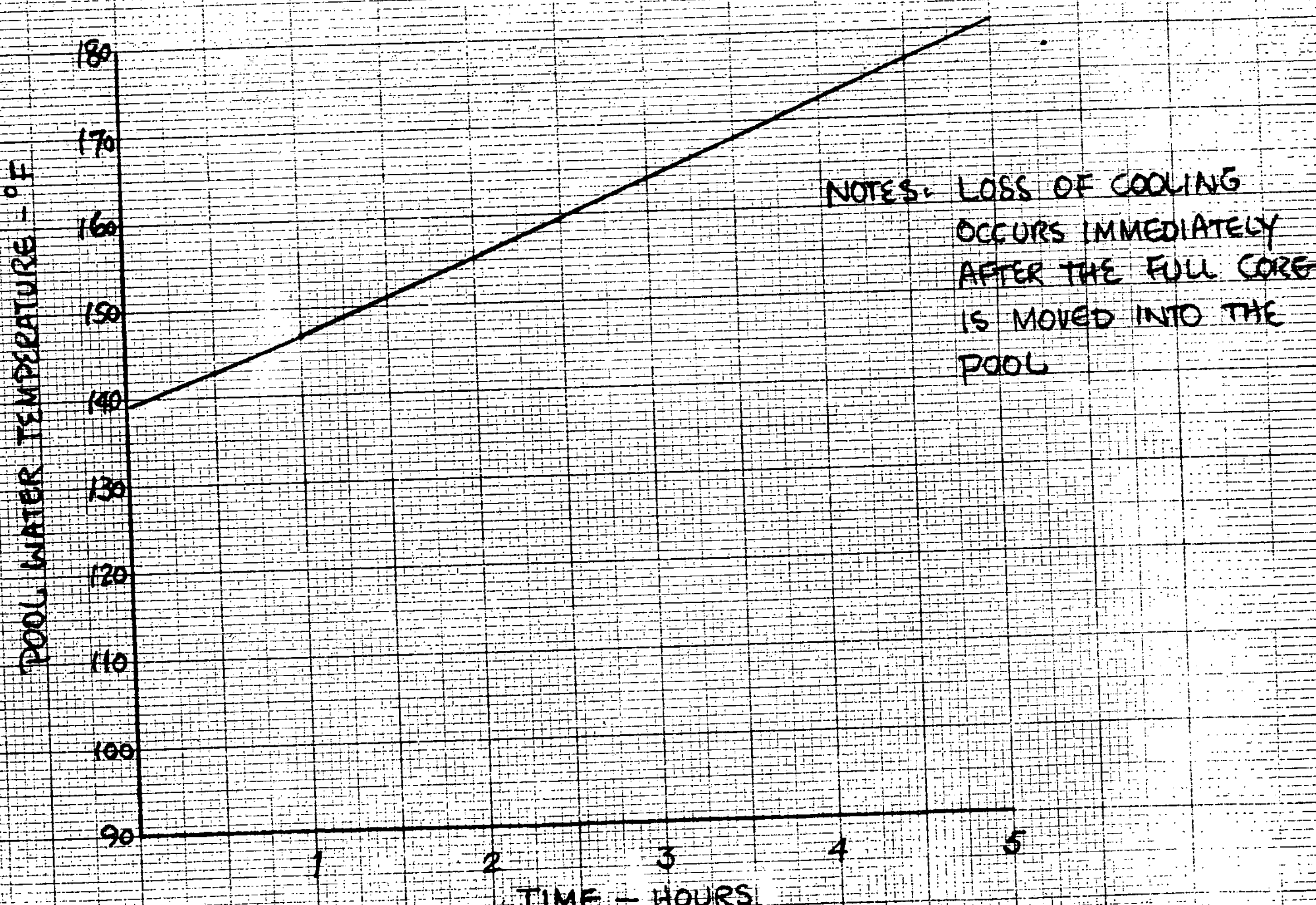

FIG. C-5. TRANSIENT HEAT-UP OF POOL WATER TEMPERATURE WITH NO COOLING FOR THE ABNORMAL CASE.