Enclosure 1 to Safety Evaluation
Spent Fuel Storage Facility Modification
Safety Analysis Report

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286
DPR-64

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1.0 INTRODUCTION

1.1 LICENSE AMENDMENT REQUESTED

The Power Authority of the State of New York (the Authority) has contracted for the design and fabrication of new spent fuel storage racks to be placed into the spent fuel pool of Indian Point 3 Nuclear Power Plant. The purpose of the new racks is to increase the amount of spent fuel that can be stored in the existing spent fuel pool. The racks are designed so that they can store spent fuel assemblies in a high density array. Therefore, the Authority hereby requests that a License Amendment be issued to the Indian Point 3 Facility Operating License DPR-64 (Reference 1) to include installation and use of new storage racks that meet the criteria contained herein. This Safety Analysis Report (SAR) has been prepared to support this request for license amendment.

1.2 CURRENT STATUS

The existing racks in the spent fuel pool at Indian Point 3 have 840 total storage cells. With the presently available storage cells, Indian Point 3 is expected to lose the full-core reserve storage capability in 1994. In order to provide sufficient capacity at Indian Point 3 to store additional fuel assemblies, the Authority plans to replace the existing storage racks with new high density spent fuel storage racks. The design of the new racks will allow for more dense storage of spent fuel, thus enabling the existing pool to store more fuel. The new high density racks have a usable storage capacity of 1345 cells, extending the full-core-reserve storage capability until the year 2005.

An objective of this reracking project is to satisfy the requirement in the Nuclear Waste Policy Act of 1982 that licensees exhaust all means of storing spent fuel on the plant site before the U.S. Department of Energy (DOE) can take spent fuel for storage offsite.

1.3 SUMMARY OF REPORT

This Safety Analysis Report follows the guidance of the NRC position paper entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as amended by the NRC letter dated January 18, 1979 (Reference 2).

This report contains the nuclear, thermal-hydraulic, mechanical, material, structural, and radiological design criteria to which the new racks are designed.

The nuclear and thermal-hydraulic aspects of this report (Section 3.0) address the neutron multiplication factor considering normal storage and handling of spent fuel as well as postulated accidents with respect to criticality and the ability of the spent fuel pool cooling system to maintain sufficient cooling.

Mechanical, material, and structural aspects (Section 4.0) involve the capability of the fuel assemblies, storage racks, and spent fuel pool system to withstand effects of natural phenomena and other service loading conditions.

The environmental aspects of the report (Section 5.0) concern the thermal and radiological release from the facility under normal and accident conditions. This section also addresses the occupational radiation exposures, generation of radioactive waste, need for expansion, commitment of material and nonmaterial resources, and a cost-benefit assessment.

1.4 CONCLUSIONS

On the basis of the design requirements presented in this report, operating experience with high density fuel storage, and material referenced in this report, it is concluded that the proposed modification of the Indian Point 3 fuel storage facilities will continue to provide safe spent fuel storage, and thus the modification is consistent with the facility design and operating criteria as provided in the Indian Point 3 FSAR Update and Operating License (Reference 1).

1.5 REFERENCES

1. Indian Point 3 Nuclear Power Plant, Facility Operating License DPR-64, Docket No. 50-286.

2. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC letter dated January 18, 1979.

3. Indian Point 3 Nuclear Power Plant, Updated Final Safety Analysis Report, Docket No. 50-286.

2.0 SUMMARY OF RACK DESIGN

2.1 EXISTING RACKS

The spent fuel pool at Indian Point 3 presently contains spent fuel assembly storage racks which are designed to provide storage locations for up to 840 fuel assemblies. The racks are designed to maintain the stored fuel in a safe, coolable, and subcritical configuration during normal and abnormal conditions.

The present storage racks consist of structural grid frames supporting storage cells for spent fuel assemblies. The storage cells are of rectangular cross section and are formed from a type 304 SS sheet of 0.150 inch minimum thickness with borated SS poison plates welded to the cell at specified locations. Each spent fuel assembly is supported on a 1/4 inch thick support plate. Each storage cell has a 6-inch diameter hole at the bottom to allow natural convection cooling. Adequate space between storage cells is provided for downflow.

The storage receptacles are arranged in modules. Twelve rack modules of seven different sizes are used. In seven of the twelve modules, the center-to-center spacing of the fuel cells is 12 inches in either direction; in five of the twelve modules, the center-to-center spacing of the fuel cells is 12 inches in the north-south direction and 11.25 inches in the east-west direction. Borated plates are attached to the sides of each cell which are adjacent to another cell; for those cells with an 11.25 inch pitch an additional boron plate is attached in the 11.25 inch pitch direction. Each borated plate is 145" x 7" x 1/8" thick type 304 SS containing 1.0% minimum, 1.2% maximum by weight boron. No borated plates are placed on the outside faces of the cells in any module.

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

For further information on the existing spent fuel storage racks see the previous Spent Fuel Pool Modification - Description and Safety Analysis (Reference 1).

2.2 NEW MAXIMUM DENSITY RACKS

The new high density spent fuel storage racks are free-standing, Boral poison design with two storage regions, designated Region 1 and Region 2. Region 1 provides storage for unirradiated fuel with an enrichment up to 4.5 w/o U-235 and provides space for storage of partially burned fuel and a full core unload. Region 2 provides storage for irradiated fuel with an initial enrichment up to 4.5 w/o U-235 which has achieved the specified burnup as discussed in this report. For example, corresponding to the 4.5 w/o initial enrichment, the minimum required burnup for safe storage in Region 2 is 36,000 MWD/MTU.

Region 1 consists of three 80-cell racks, providing 240 storage spaces. Region 2 consists of nine racks ranging in size from 104 to 132 cells, providing 1105 storage cells. Figure 2-1 shows the arrangement of the rack modules in the spent fuel pool. Tables 2-1 and 2-2 give the relevant design data for each region and physical data for each module type, respectively.

The spent fuel storage rack design is a welded honeycomb array of stainless steel boxes which has no grid frame structure. Each cell has a welded-in bottom plate, either 1/2" or 3/4" thick, to support the fuel assembly. A central hole in the bottom plate provides for cooling water flow. All storage cells are bounded on four sides by Boral poison sheets, except on the periphery of the pool rack array.

Region 1 consists of square storage cells which are spaced in both directions by a narrow rectangular water box (see Figure 2-2). The Boral poison sheets are captured between adjacent walls of the storage cells and water boxes. The required space for the poison is provided by local round raised areas coined in the box walls to half the thickness of the poison sheets. All boxes are fusion welded together at these local raised areas. The poison sheets are scalloped along their edges to clear the raised areas, which also serve to retain the sheets laterally. The sheets are retained axially by a short stainless sheet, the same thickness as the poison, welded at the bottom of the poison to one of the two adjacent box walls.

Region 2 consists of square storage cells with a poison sheet captured between adjacent boxes (see Figure 2-3) in the same manner as described in Region 1.

At a rack-to-rack interface, poison sheets are captured on the outer face of one of the two racks. Each poison sheet is captured under a thin stainless sheet, whose four edges are bent the thickness of the poison and intermittently welded to each box on the outer face. The racks are installed with no gap because the very strong hydrodynamic coupling forces the racks to move together even when a full and empty rack are adjacent to each other.

Each rack is supported and leveled on four screw pedestals which bear directly on the pool floor. These free-standing racks are free to move horizontally. However, with only a .2 friction factor, there is no wall impact even assuming 5 OBE and 1 SSE earthquake events all add up in the same direction.

2.3 REFERENCES

1. Spent Fuel Pool Modification - Description and Safety Analysis provided as Attachment A to Authority's September 1, 1977 Letter IPO-26 to the NRC.

TABLE 2-1

DESIGN DATA

Region	Cell Pitch (nominal inch)	Min. B-10 Loading (areal density)	Flux Trap Gap (nominal inch)
1	10.76	.020 gm/cm^2	1.44
2	9.075	.020 gm/cm^2	0.00

TABLE 2-2

MODULE DATA

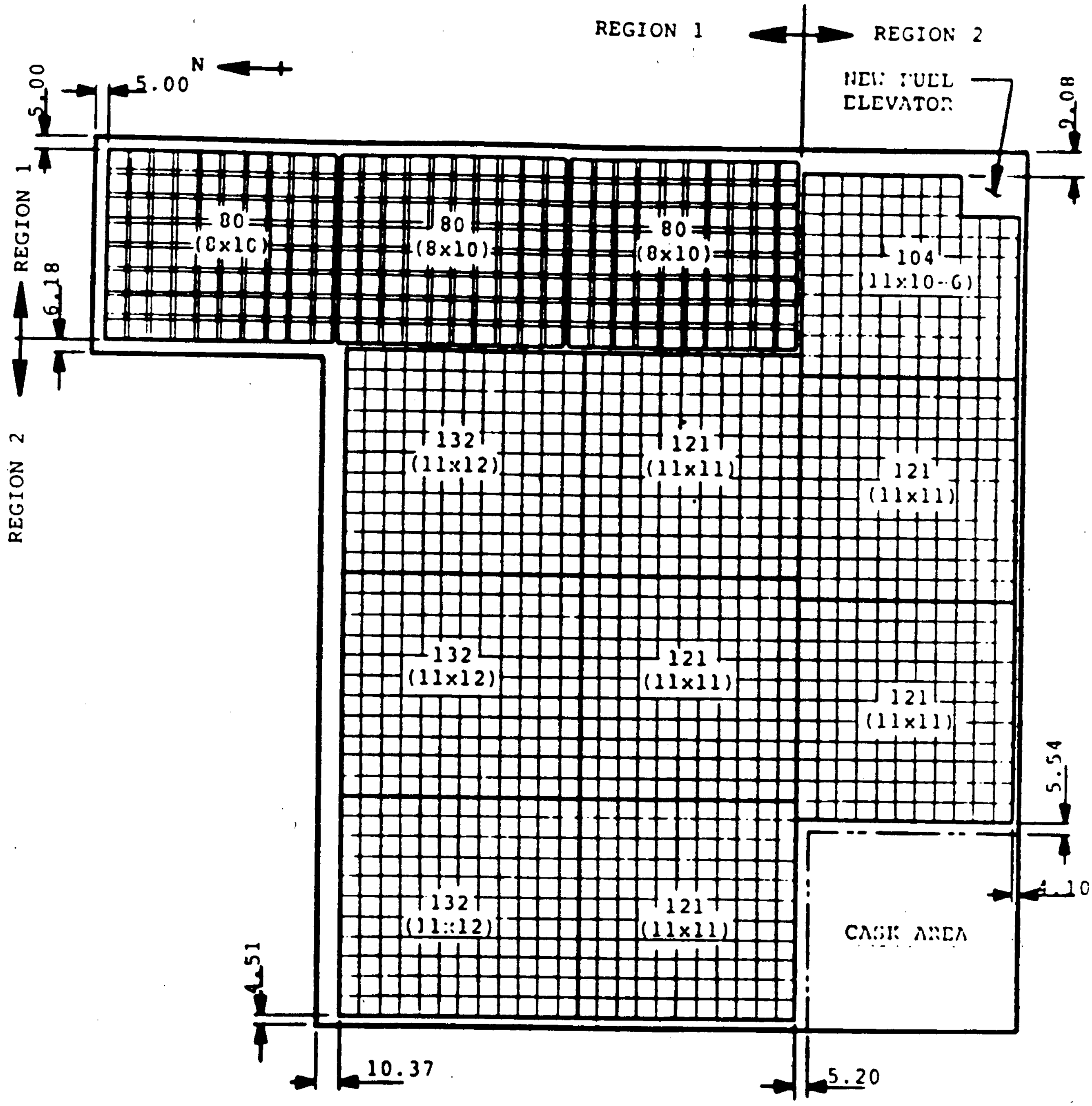
Module I.D.	No. of Modules	No. Cells In N-S Direction	No. Cells In E-W Direction	Total No. of Cells Per Module	Est. Dry Wt. (lbs) Per Module
Region 1 8721-2,-3,-4	3	10	8	80	27,880
Region 2 8721,-6,-9,-12	3	12	11	132	23,870
Region 2 8721-7,-8,-10 -11,-13	5	11	11	121	22,150
Region 2 8721-5	1	11*	10*	104	19,000

* Cells missing in this module in area of new-fuel elevator.

SECTION 2 FIGURES

FIGURE 2-1

POOL LAYOUT



REGION 1 STORAGE = 240
REGION 2 STORAGE = 1105
TOTAL = 1345

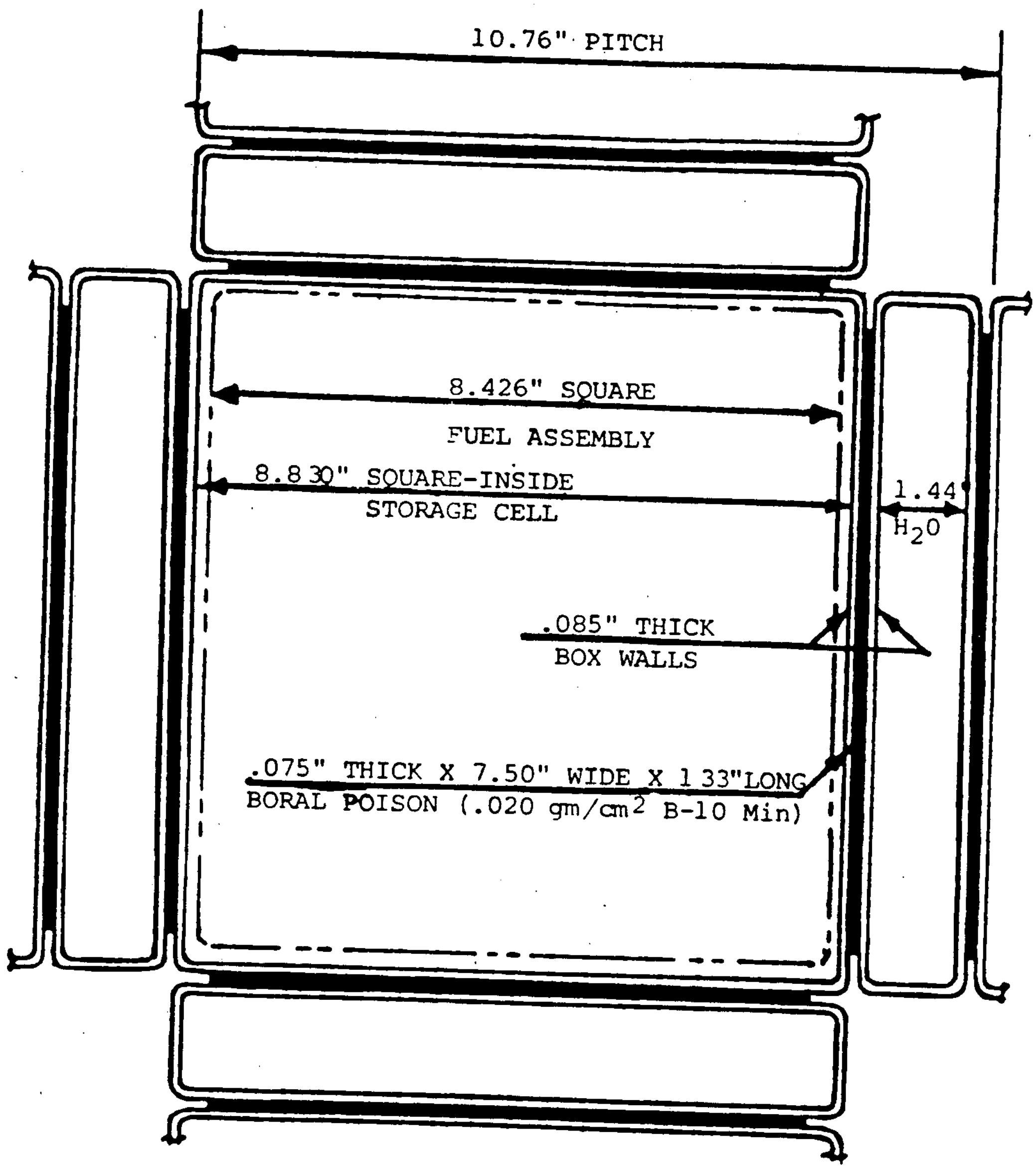


FIGURE 2-2

REGION 1 STORAGE CELL GEOMETRY

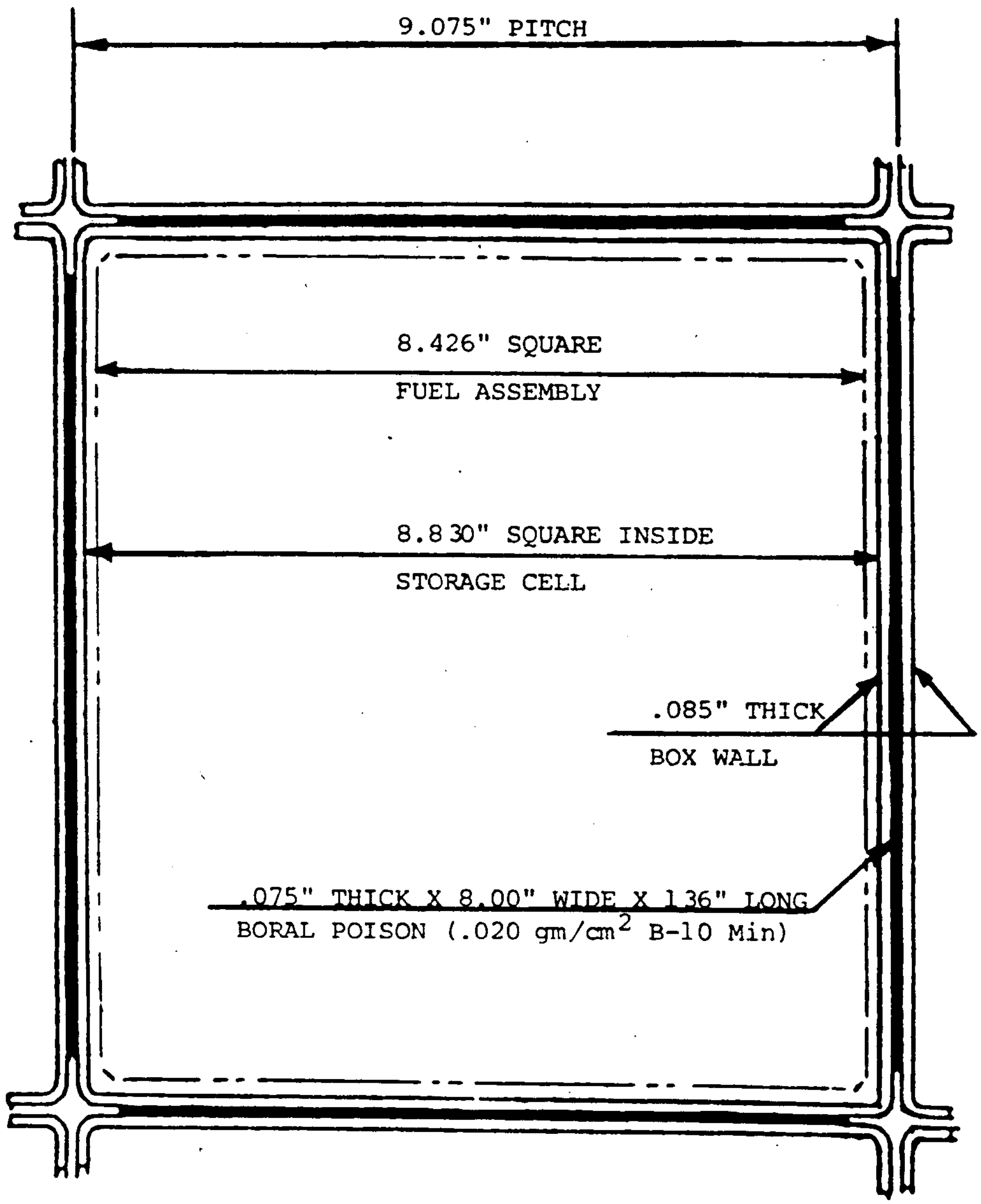


FIGURE 2-3

REGION 2 STORAGE CELL GEOMETRY

3.0 NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

3.1 NEUTRON MULTIPLICATION FACTOR

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies.

The design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.95, including uncertainties, as recommended in ANSI 57.2-1983 and in the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 9).

The spent fuel rack design will employ two separate and different geometries. They are designed to maintain $k_{eff} < 0.95$ for storage of Westinghouse 15x15 PWR Standard and Optimized fuel (or equivalent) up to a maximum enrichment of 4.5 w/o U-235. Region 1 is designed for safe storage of new fuel and spent fuel of any burnup. Region 2 is designed for safe storage of fuel which has accumulated a minimum burnup based on initial enrichment.

The following subsections describe the conditions in the spent fuel pool which are assumed in calculating the effective neutron multiplication factor, the analysis methodology, and the analysis results.

3.1.1 Normal Storage

(a) The spent fuel storage is divided into two regions per the layout shown on Figure 2-1. The Region 1 nominal geometry is shown on Figure 2-2 and the Region 2 nominal geometry is shown on Figure 2-3.

(b) Storage of fuel in Region 1 assumes no credit for burnup, i.e., new fuel up to 4.5 w/o U-235. Storage of irradiated fuel in Region 2 is limited by determination of minimum burnup versus enrichment, which is shown on Figure 3-22.

(c) The calculational approach is to use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and to account for any deviations of the actual spent fuel rack array from this assumed infinite array as perturbations on the calculated reactivity of the basic cell.

(d) The basic cell calculation is performed with nominal dimensions. Tolerances on the geometric array representing the racks are treated as perturbations. Variations in the fuel characteristics and operating conditions are also treated as perturbations. The reactivity effects of all positive perturbations are then combined statistically to determine a single reactivity perturbation which is added to the calculated basic cell multiplication factor (including biases) to determine the final conservative evaluation of the spent fuel rack maximum possible multiplication factor.

(e) Credit is taken for the neutron absorption in the rack structural material and for the neutron absorption material specifically added to the rack for that purpose. A means for inspection is provided.

(f) No credit is taken for the negative reactivity effect of soluble boron in the pool water. Pure water at 68°F temperature (which results in maximum reactivity) is assumed as the moderator.

3.1.2 Postulated Accidents

The criticality analysis addresses postulated accidents and shows that the k_∞ in both regions is less than 0.95. The double contingency principle of ANSI 8.1-1983 is applied which states that it requires two unlikely, independent, concurrent events to produce a criticality accident.

A loss of pool cooling accident is addressed in the criticality analysis by a sensitivity study that shows that the spent fuel storage rack multiplication factor decreases with decrease in water density. No credit for soluble boron or axial leakage was taken in this sensitivity analysis.

The Region 1 fuel racks are designed such that a dropped fuel assembly cannot occupy a position in the racks other than a normal fuel storage location. The only positive effect of a dropped fuel assembly on the reactivity of the rack would be by virtue of reduction in axial neutron leakage from the rack. Since no credit is taken for axial neutron leakage, a dropped fuel assembly would not increase the reactivity of the spent fuel storage rack above the maximum calculated normal storage value.

The positive reactivity effect of an unirradiated fuel assembly located adjacent to the fully loaded Region 1 spent fuel storage rack was analyzed and found to be more than compensated for by the significant negative reactivity effect of 1000 ppm (minimum) of soluble boron contained in the pool water and therefore would not increase the reactivity of the spent fuel storage rack above the maximum calculated normal storage value.

The Region 2 fuel racks do not contain any vacant spaces, other than unused fuel storage locations, into which a dropped fuel assembly could fall. The only positive reactivity effect of a dropped assembly would thus be due to reduced axial leakage from the rack. Since no credit is taken for axial leakage, this accident cannot increase the multiplication factor above the maximum calculated normal storage value.

The positive reactivity effect of an unirradiated fuel assembly located adjacent to the fully loaded Region 2 spent fuel storage rack was analyzed and found to be more than compensated for by the negative reactivity effect of 1000 ppm (minimum) of soluble boron. The other postulated accident that was analyzed is that involving an unirradiated fuel assembly at the maximum enrichment, or an insufficiently depleted fuel assembly, being incorrectly transferred to Region 2. This is also considered to be an abnormal Region 2 condition and appropriate credit is taken for the 1000 ppm (minimum) of soluble boron present in the pool water. The resulting Region 2 k_∞ is then calculated to be less than 0.95. Thus, in all cases, the Region 2 spent fuel storage rack design ensures that the multiplication factor is less than the 0.95 limit.

3.1.3 Calculation Methods

The following discussion summarizes the criticality safety analysis of the Indian Point 3 spent fuel storage racks. The analytical techniques described here are identical to those used to successfully license spent fuel storage racks for several other plants.

The spent fuel pool is divided into two regions. Since the analytical methods used for the two regions are similar, but not identical, the methods used are discussed in separate sections.

3.1.3.1 Criticality Analysis for the Region 1 Spent Fuel Storage Racks

New fuel and spent fuel assemblies of any burnup may be stored in Region 1. For purposes of analysis, all fuel stored in Region 1 is assumed to be new fuel at the maximum allowable enrichment.

3.1.3.1.1 Analytical Technique

The LEOPARD (Reference 1) computer program is used to generate macroscopic cross-sections for input to four energy group diffusion theory calculations which were performed with the PDQ-7 (Reference 2) program. LEOPARD calculates the neutron energy spectrum over the entire energy range from thermal up to 10Mev and determines averaged cross-sections over appropriate energy groups. The fundamental methods used in the LEOPARD program are those used in the MUFT (Reference 3) and SOFOCATE (Reference 4) programs which were developed under the Naval Reactor Program where they were extensively tested. In addition, Westinghouse Electric Corporation, the developers of the original LEOPARD program, demonstrated the accuracy of these methods by extensive analysis of measured critical assemblies consisting of slightly enriched UO_2 fuel rods (Reference 5).

In addition, Pickard, Lowe and Garrick, Inc. (PLG) has made a number of improvements to the LEOPARD program to increase its accuracy for the calculation of reactivities in systems which contain significant amounts of plutonium mixed with UO_2. PLG has tested the accuracy of these modifications by analyzing a series of UO_2 and PuO_2-UO_2 critical experiments. These benchmarking analyses not only demonstrate the improvements obtained for the analysis of PuO_2-UO_2 systems, but also demonstrate that these modifications have not adversely affected the accuracy of the PLG-modified LEOPARD program for calculations of slightly enriched UO_2 systems.

The UO_2 critical experiments chosen for benchmarking include variations in H_2O-UO_2 ratios, U-235 enrichments, pellet diameters and cladding materials. Although the LEOPARD model also accurately calculates the reactivity effects of soluble boron, these experiments have not been included in the LEOPARD benchmarking criticals since most of the spent fuel rack analyses do not include soluble boron.

Neutron leakage was represented by using measured buckling input to infinite lattice LEOPARD calculations to represent the critical assembly. A summary of the results is shown in Table 3-1 for the 27 measured criticals chosen as being directly applicable for benchmarking the LEOPARD model for generating group average cross-sections for spent fuel rack criticality calculations. The average calculated k_{eff} is 0.9980 and the standard deviation from this average is 0.00080Δk. Reference 5 raised questions concerning the accuracy of the measured buckling reported for the experiments number 12 through 19. If these data are excluded, the average calculated k_{eff} for the remaining 19 experiments is 1.0006 with a standard deviation from this value of 0.0065Δk. In all of these experiments there are significant uncertainties in the measured bucklings which are necessary inputs to the LEOPARD analysis. These uncertainties are the same order of magnitude as the indicated errors in the LEOPARD results, and therefore a more definitive set of experimental data is used to establish the accuracy of the combined LEOPARD/PDQ-7 model used for the criticality analysis of the spent fuel racks.

The PDQ series of programs have been extensively developed and tested over a period of 20 years, and the current version, PDQ-7, is an accurate and reliable model for calculating the subcritical margin of the proposed spent fuel rack arrangement. This code or a mathematically equivalent method is used by many of the U.S. suppliers of light water reactor cores and reload fuel. In addition, this code has received extensive utilization in the U.S. Naval Reactor Program.

As a specific demonstration of the accuracy of the calculational model used for the calculations, the combined LEOPARD/PDQ-7 model has been used to calculate 14 measured just critical assemblies (Reference 6 and 7). The criticals are high neutron leakage systems with a large variation in U/H_2O volume ratio and include parameters in the same range as those applicable to the proposed rack designs. Experiments including soluble boron are included in this demonstration since the ability of PDQ-7 to calculate neutron leakage effects is of primary interest. The use of soluble boron allows changes in the neutron leakage of the assembly while maintaining a uniform lattice and thus allows a better test of the accuracy of the model. Furthermore, it eliminates the error associated with the measured bucklings which is inherent in the LEOPARD benchmarks, thus permitting determinations of the actual calculational uncertainty which must be accounted for in the spent fuel rack criticality analysis.

These combination LEOPARD/PDQ-7 calculations result in a calculated average k_{eff} of 0.9928 with a standard deviation about this value of 0.0012Δk. These results, as shown in Table 3-2, demonstrate that the proposed LEOPARD/PDQ-7 calculational model can calculate the reactivity of the spent fuel rack arrangements with an accuracy of better than 0.0104Δk with 95 percent probability at the 95 percent confidence level.

The cross-sections of the Boral neutron absorbing material which is an integral part of the design are calculated using fundamental techniques that have been successfully applied in the past to thin heavily absorbing mediums such as control rods.

Experimental tests of the theory for rods formed from slabs and inserted in light-water-moderated cores indicate that it is capable of predicting the k_{eff} of a core with control rods inserted to about the same degree of accuracy as is possible with rods withdrawn (Reference 22).

The procedure is straightforward and is comprised of several well defined steps:

(1) The B^{10} from the thin Boral sheets is homogenized in an appropriate amount of water and LEOPARD is used to obtain unshielded macroscopic B^{10} cross-sections.

(2) Integral transport theory is applied in slab geometry using They's method for calculating flux depressions and shielding factors to determine an appropriate B^{10} number density. This approach is similar to that of Amouyal and Benoist as reported in Reference 8.

(3) The B^{10} number density calculated in Step 2 is homogenized in water and LEOPARD is used to obtain corrected microscopic cross-sections.

(4) Blackness theory is applied to obtain macroscopic cross-sections which will produce the required boundary conditions at the surface of the Boral sheets.

In addition to the fourteen critical assemblies in Table 3-2, the LEOPARD/PDQ model was used to calculate the k_{eff} for twelve additional critical assemblies, seven of which incorporated thin, heavily-absorbing materials for which the procedure just described was used to determine the macroscopic cross-sections.

These twelve criticals were performed by Battelle Pacific Northwest Laboratories specifically for the purpose of providing benchmark critical experiments in support of spent fuel criticality analysis. They are described in detail in detail in Reference 23. The results of these critical experiments are summarized in Table 3-3. The first seven of these twelve experiments include fixed neutron poison absorber plates, and the average k_{eff} calculated for these just critical assemblies was 0.9935, with a standard deviation around this value of 0.0007Δk. The other five critical experiments in this series do not include absorber plates and the average k_{eff} calculated for these just critical assemblies was 0.9944, with a standard deviation around this value of 0.0007Δk. The overall average k_{eff} calculated for these twelve just critical assemblies was 0.9939 with a standard deviation around this value of 0.0009Δk.

For the 26 measured criticals in Tables 3-2 and 3-3, the mean calculated k_{eff} was 0.9933 and the standard deviation was 0.0012. This results in a model bias of +0.0067Δk and a 95/95 uncertainty of 2.28σ or 0.0027Δk.

As a result of this approach to separately benchmark both the cross-sections and the diffusion theory calculations against applicable critical assemblies, the "transport theory correction factor" is implicitly included in the derived calculational uncertainty factor.

3.1.3.1.2 Calculational Approach

The PDQ-7 program is used in the final predictions of the multiplication factor of the spent fuel storage racks. The calculations are performed in four energy groups and take into account all of the significant geometric details of the fuel assemblies, fuel boxes and major structural components. The geometry used for most of the calculations is a basic cell representing one quarter of a repeating array of two different types of identical stainless steel boxes. The specific geometry of this basic cell is shown in Figure 3-1, and the assumed fuel assembly characteristics (corresponding to a 15 x 15 Westinghouse OFA assembly) are listed in Table 3-4.

The calculational approach is to use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and to account for any deviations of the actual spent fuel rack array from this assumed infinite array as perturbations on the calculated reactivity of the basic cell. The fuel assemblies are assumed to be unirradiated with a uniform U-235 enrichment of 4.5 w/o. The calculated k_{∞} of this basic cell is 0.9040. Most of the calculations are performed at a uniform pool temperature of 68°F with full density water, but the reactivity effects of pool temperature and water density are also taken into account as a perturbation on the basic cell calculations. No credit is taken for the soluble boron in the base case. This removes the need for a separate analysis of a boron dilution accident. Similarly, no credit is taken for axial neutron leakage (axial buckling) removing the need to analyze the case of a dropped fuel assembly. However, credit may be taken for both soluble boron and the axial buckling when accident situations are considered, since only a single failure need be assumed.

The basic cell calculation is performed with nominal dimensions on all the stainless steel boxes and results in the k_{∞} values shown in Table 3-5 and Figure 3-2 for fuel assemblies with U-235 enrichments of 4.25 w/o, 4.5 w/o, 4.75 w/o. Tolerances on the geometric array representing the racks are treated as perturbations on this basic cell calculation at an enrichment of 4.5 w/o, with a minimum B^{10} loading in the Region 1 Boral of .020g/cm^2 in .075 inches of thickness.

3.1.3.1.3 Perturbations to the Basic Cell

In order to determine the effects of possible variations in the fuel characteristics, dimensional tolerances of the rack, and operating conditions, various perturbations in the basic cell were considered. All cases were run for unirradiated 4.5 w/o fuel.

The k_{∞} of the basic cell as a function of temperature is shown in Figure 3-3. Based on this figure, the basic cell temperature of 68°F, which corresponds to the lower limit for the pool, is the most reactive temperature.

The sensitivity of the spent fuel storage rack multiplication factor to the simultaneous and uniform variation of water density in both the fuel box and Boral box is illustrated in Figure 3-4. No credit for soluble boron or axial leakage was taken in this sensitivity analysis.

Based on the results of the benchmarking of the combined LEOPARD/PDQ-7 analysis model, the bias in the calculated multiplication factor compared to the 26 measured just critical arrays is +.0067Δk, and this bias must be added to the calculated basic cell reactivity. The 2.28σ uncertainty in the model, which corresponds to the 95/95 confidence level, is .0027Δk which is added as a calculational uncertainty.

A geometric modeling effect bias was introduced to account for mesh spacing and smeared stainless steel - water composition effects on the PDQ base model. Most of the calculations with the basic cell geometry utilize a 37 x 37 two-dimensional array of mesh points. To test the adequacy of this mesh description, a calculation was run with a 72 x 72 mesh size. The resulting perturbation on the base cell k_{∞} due to mesh spacing effects is - .0003Δk. The use of explicit stainless steel cross sections, rather than a smeared stainless steel - water composition intended primarily for adequate modeling of low water density configurations, results in a perturbation on the base cell k_{∞} of +.0025Δk. Thus, the overall geometric modeling effect bias is +.0022Δk.

The basic cell model assumes that the entire active fuel length is shielded with Boral. An axial perturbation case was run using the actual Boral panel length, which is designed to leave 5.5 inches of active fuel uncovered at the top and bottom of the fuel assembly, resulting in a bias of +.0039Δk. Figure 3-5 plots the Δk as a function of inches of active fuel uncovered at the top and bottom of the fuel assembly for Region 1.

There are also a number of tolerances and uncertainties which result in perturbations which must be considered in the criticality analysis. The reactivity effects of all such positive perturbations are then combined statistically in accordance with Reference 9 to determine a single reactivity perturbation which is added to the calculated basic cell multiplication factor (including biases) to determine the final conservative evaluation of the spent fuel rack maximum possible multiplication factor.

Normal variations in manufacturing tolerances may result in variation in box wall thickness of ± .004 inches and in Boral thickness of ± .005 inches. These result in uncertainties of ± .0005Δk and ±.0036Δk, respectively. Variations in Boral panel length of ±.25 inches results in an uncertainty of ±.0016Δk. Variations on the fuel box dimension and Boral box dimension result in uncertainties of ± .0009Δk and ± .0033Δk, respectively. These are listed in Table 3-6, which provides a summary of the biases and uncertainties on the basic cell.

The reactivity effects of the fuel position within the fuel box were also analyzed. Calculations have confirmed that the fuel assemblies are located in their most reactive position when centered within the fuel boxes.

The nominal density of the UO_2 pellets contained in the fuel assemblies is 95 percent of theoretical density. Increasing this to the maximum fabrication tolerance of 96 percent results in a positive reactivity perturbation of .0011Δk.

Results of reactivity perturbations to the basic cell due to biases, tolerances and uncertainties are summarized in Table 3-6. The total reactivity perturbation to be added to the basic cell is 0.0167Δk. This results in a final conservatively calculated spent fuel rack Region 1 multiplication factor of 0.9207 for 4.5 w/o fuel.

As an additional calculational conservatism, it should be noted that the spent fuel pool water contains a minimum concentration of 1000 ppm soluble boron at all times. When the reactivity effect of this minimum soluble boron concentration is included in the calculations, the spent fuel pool multiplication factor including all biases, tolerances, and uncertainties is found to be 0.8215.

3.1.3.2 Critically Analysis for the Region 2 Spent Fuel Storage Racks

Fuel assemblies to be stored in Region 2 must have accumulated the minimum burnup requirement based on the intitial enrichment of the fuel assembly. The Region 2 analysis takes fuel burnup into account, and a curve is developed which specifies the minimum burnup for fuel to be stored in Region 2. Since the criticality analyses for Region 2 are subject to different uncertainties than those applicable to Region 1, the uncertainties applicable to the Region 2 analyses are independently derived.

3.1.3.2.1 Analytical Technique

The isotopic composition of the irradiated fuel is calculated as a function of assembly average burnup and subsequent decay using the LEOPARD (Reference 1) and CINDER (Reference 10) computer programs. Once the isotopic composition of the fuel assemblies is known, the subsequent criticality calculations for the spent fuel racks in Region 2 are performed in a manner that is analogous to the calculations for Region 1. Consequently, the analytical methods used for criticality analysis of Region 1 are also incorporated into the criticality analysis of Region 2.

The accuracy of the burnup independent isotopic concentrations calculated with the LEOPARD program is demonstrated in Figures 3-6 through 3-16. Figures 3-6 through 3-13 show comparisons of LEOPARD calculated data with measured data from a UO_2 fuel assembly irradiated in the Yankee-Rowe reactor while Figures 3-14 through 3-16 show corresponding data for a mixed oxide (PuO_2-UO_2) fuel assembly irradiated in the SAXTON reactor.

Except for the data labeled PLG calculation, the data and curves on Figures 3-6 through 3-13 and Figures 3-14 through 3-16 are taken directly from References 11 and 12, respectively. In all cases, the accuracy of the calculations labeled PLG is within the uncertainty in the measured data.

In addition to the 26 critical array benchmarks referenced in the Region 1 analysis, 11 critical arrays of mixed oxide fuel rods which contain high concentrations of the plutonium isotopes are used to demonstrate the accuracy of reactivity calculations for irradiated fuel. Tables 3-7 and 3-8 show results of criticality analyses for the SAXTON (Reference 13) and ESADA (Reference 14) sets of experiments which cover a wide range of water-to-oxide volume ratios. A summary of these data is shown in Table 3-9. For the mixed oxide criticals, the calculated mean multiplication factor is 0.9969 with a standard deviation about this value of 0.0066Δk. Using the 95 percent probability at 95 percent confidence level criterion (one-sided) with 37 data points, this implies a total calculational uncertainty of $2.17\sigma = 0.0086\Delta k$ with a bias of +.0057Δk.

The other major uncertainty in the calculation of k_∞ in Region 2 is associated with the calculated reduction in fuel assembly reactivity associated with the depletion of the heavy metals and the accumulation of fission products as a function of fuel assembly exposure. The calculations were done for the base case with fuel at an enrichment of 4.5 w/o and a burnup of 36,000 MWD/MTU. For this fuel, the total reactivity loss from the fresh, unirradiated state is .2243Δk/k, of which less than 50 percent is attributed to the buildup of fission products. The relative change, $\Delta k/k$, is used to provide a consistent basis for comparing perturbations at different values of k_∞. Calculations of reactor reactivity lifetimes using the same analytical methods used in this analysis demonstrate an accuracy of better than 5 percent for exposures which have approximately 50 percent of the reactivity loss due to fission products. Thus the uncertainty due to burnup is no more than 10 percent of the reactivity loss due to fission products. Therefore, the resulting uncertainty in the calculated fuel assembly k_∞ associated with the fuel depletion would be conservatively estimated at 0.0112Δk/k (= 0.10 x 0.5 x 2243Δk/k). The corresponding uncertainty in the calculated Region 2 multiplication factor is 0.0104Δk on a base case Region 2 k_∞ of 0.9241 (4.5 w/o at 36,000 MWD/MTU).

In order to provide further assurance of the conservative nature of these calculations, the decay of all fission products following discharge of the fuel assembly was taken into account. This was accomplished with the aid of the CINDER (Reference 10) code which treats a total of 186 nuclides in 84 linear chains. The fission product inventory for each fuel assembly was decayed for 40 years following its removal from the reactor core, and the time point of minimum fission product absorption within that 40 year period was used as the basis for determining the fission product macroscopic absorption cross-sections for that particular fuel assembly at that specific exposure. That minimum occurs at less than 100 days into the decay and from then on continues to increase. Reduction in the fission product inventory due to leakage or escape to the plenum has been found to be negligible (Reference 15).

3.1.3.2.2 Calculational Approach

Reactivity calculations, using the LEOPARD and PDQ-7 models described previously for the Region 1 analysis, were performed for Region 2. The geometry used for these calculations is that of a basic quarter cell representing one quarter of a stainless steel box with fuel assembly and associated Boral. The specific geometry of this basic cell is shown in Figure 3-17.

3.1.3.2.3 Perturbations to the Basic Cell

The effect of perturbations in manufacturing and thermal parameters were analyzed for the Region 2 configuration. This analysis assumed a Westinghouse 15x15 design, characterized by a fuel assembly with an initial enrichment of 4.50 w/o U-235 at a burnup of 36,000 MWD/MTU, and a minimum B^{10} loading in the Region 2 Boral of .020 g/cm^2 in .075 inches of thickness. This fuel assembly and burnup were selected as being typical of the irradiated fuel being considered for storage in Region 2.

A sensitivity analysis was performed to determine the variation of Region 2 k_∞ with Boral panel length. Based on the results of the analysis, as shown in Figure 3-18, it was determined that the Region 2 rack design would incorporate 136 inch Boral panels. This results in a bias of +0.0010Δk with respect to the base case even under the very conservative assumption that the rod stack grows by almost 1.4% during irradiation to 146 inches, leaving 5.0 inches of fuel exposed at each end.

An analysis was also performed to determine the reactivity effect associated with the Region 1 - Region 2 rack interface. The results of this analysis show that there is a negative reactivity effect associated with the change in geometry at the interface boundary and therefore there is no increase in the calculated k_∞ of Region 2.

Specific calculations were performed on the irradiated fuel for variations in all relevant physical dimensions, temperature, water density and pellet density. Table 3-10 provides a summary of the reactivity perturbations to the Region 2 spent fuel storage racks. Detailed results of these calculations for Region 2 are presented in Figures 3-19 and 3-20 for the effects of temperature and water density, respectively.

As an additional calculational conservatism, it should be noted that the spent fuel pool coolant contains a minimum concentration of 1000 ppm soluble boron at all times. When the reactivity effect of this minimum soluble boron concentration is included in the calculations, the spent fuel pool multiplication factor including all biases, tolerances, and uncertainties is found to be 0.8439.

3.1.3.2.4 Required Minimum Burnup as a Function of Initial Enrichment for Region 2 Spent Fuel

The maximum combined bias and uncertainty were determined for 4.50 w/o fuel at a burnup of 36,000 MWD/MTU. The biases were added and the uncertainties were combined by summing the squares of the individual uncertainties, and taking the square root of the sum. These calculations are shown in Table 3-10. The bias and uncertainty is $0.0238\Delta k$, giving a value of $\Delta k/k$ of $0.0258\Delta k/k$ (= 0.0238/0.9241). The relative change, $\Delta k/k$, is used to provide a consistent basis for comparing uncertainties at different value of k_∞. Thus, the k_∞ with uncertainties will be less than 0.95 if the computed k_∞ satisfies the relationship

$$k_\infty + k_\infty\ (\Delta k/k) < 0.95$$

or, using $\Delta k/k = 0.0258$, if the computed k_∞ is less than 0.9261. However, in order to allow for possible interpolation errors, a target k_∞ of .9245 was used. The values of k_∞ as a function of initial enrichment and burnup given in Table 3-11 are plotted in Figure 3-21. The burnup which results in a Region 2 k_∞ of .9245 for each initial enrichment is obtained from the appropriate curve. These results are tabulated in Table 3-12. Figure 3-22 was prepared directly from the information presented in Table 3-12 and shows the required minimum exposure as a function of initial enrichment to assure that the value of k_∞ in Region 2 of the spent fuel rack is less than 0.95 with a probability of 95 percent at the 95 percent confidence level.

3.1.4 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95 including all uncertainties, under all conditions.

Criticality is precluded by spacing of fuel assemblies acceptable for storage, which ensures that a subcritical array of k_{eff} less than or equal to 0.95 is maintained, assuming unborated pool water. The pool, however, will always contain boric acid at the refueling concentration of 1000 ppm (minimum) whenever there is irradiated fuel in the pool.

Methods for initial and long term verification of poison material stability and mechanical integrity are discussed in Section 4.

3.2 THERMAL-HYDRAULIC ANALYSES FOR THE SPENT FUEL POOL (BULK)

The purpose of the bulk fuel pool thermal-hydraulic analyses is to demonstrate the adequacy of the existing spent fuel pool cooling system for utilization of the increased number of storage cells.

3.2.1 Spent Fuel Pool Cooling System Design

The spent fuel pool cooling system consists of pumps (main and standby), heat exchanger, filters, demineralizer, piping and associated valves and instrumentation. The operating pump draws water from the pool, circulates it through the heat exchanger and returns it to the pool. In the event of a failure of the main spent fuel pump, a standby pump can be put into operation immediately from a local startup pushbutton station.

The spent fuel pool heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pool water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

The clarity and purity of the spent fuel pool water are maintained by using a second pumping system to pass approximately 5 percent of the cooling system flow through a filter and demineralizer. The spent fuel pool pump suction line, which is used to drain water from the pool, penetrates the spent fuel pool wall above the fuel assemblies. The penetration location prevents loss of water as a result of a possible suction line rupture.

The primary source of makeup water to the spent fuel pit is the Primary Makeup Water Storage Tank, which is a seismic Class I component. The pumps and most of the piping associated with this tank are also seismic Class I. The makeup water to the spent fuel pool is seismic Class II, as is the spent fuel pool cooling and cleanup loop. Additional backup can be provided through a temporary connection from the plant demineralizers or from the Fire Water Tank.

In addition to the second spent fuel pool cooling system pump to provide standby pool cooling capacity, there is also a provision for adding a portable cooling pump.

3.2.2 Decay Heat and Bulk Pool Temperature Analyses

3.2.2.1 Basis

The Indian Point Unit 3 reactor is rated at 3025 megawatts thermal (MWt). The core contains 193 fuel assemblies. Thus, the average operating power per fuel assembly, P_o, is 15.67 MW. The fuel discharge can be made in one of the following two modes:

- Normal refueling discharge
- Full core discharge

Table 3-13 and Table 3-14 provides the parameters for the decay heat and bulk pool temperature analyses.

3.2.2.2 Model Description

NUREG-0800 Branch Technical Position ASB 9-2, "Residual Decay Energy For Light Water Reactors For Long Term Cooling" (Reference 24) is utilized to compute the heat dissipation requirements in the pool in accordance with Standard Review Plan 9.1.3.

The operating power, P_O, is taken equal to the rated power, even though the reactor may be operating at less than its rated power during much of the exposure period for the batch of fuel assemblies. The computations and results reported here are based on the discharge taking place when the inventory of fuel in the pool will be at its maximum, resulting in an upper bound on the decay heat rate.

Having determined the heat dissipation rate, the next task is to evaluate the time-dependent temperature of the pool water. Table 3-14 identifies the assumed heat transfer data for the Spent Fuel Pool Heat Exchanger, consistent with the Updated FSAR (Reference 25), plant component installation information and previous reracking analysis (Reference 26). A number of simplifying assumptions are made which render the analysis conservative, including:

- Additions of fuel to the spent fuel pool at the end of the in-reactor cooldown period are assumed to occur instantaneously.

- The Heat Removal Effectiveness for the Spent Fuel Pool Heat Exchanger is assumed to be 90 percent of design.

- No credit is taken for the improvement in the film coefficients of the heat exchanger as the operating temperature rises due to monotonic reduction in the water kinematic viscosity with temperature rise. Thus, the film coefficient used in the computations are lower bounds.

- No credit is taken for heat loss by evaporation of the pool water.

- No credit is taken for heat loss to pool walls and pool floor slab.

In addition, a sensitivity analysis was performed to assess the effects of alternative heat transfer data assumptions regarding the Component Cooling Water Inlet Temperature to the Spent Fuel Pool Heat Exchanger. A worst case scenario was investigated assuming a Component Cooling Water Inlet Temperature of 100°F, corresponding to an infrequent river water temperature of 87.8°F (hot summer conditions combined with cooling water discharge conditions from the upstream Indian Point 2 plant). Refueling discharges or full core removal would not normally be planned during such a period, but the results for both cases are calculated in order to envelope the potential pool temperatures.

The time until pool boiling occurs and the boil-off rate (assuming a complete loss of fuel pool cooling with no corrective action) is determined next, using the maximum decay heat rates and the spent fuel pool thermal inertia. The thermal inertia is calculated based on the volume and heat capacity of the pool water and its contained racks and fuel, but conservatively ignoring the pool liner and concrete, piping and contained water external to the pool boundaries, and pool water in the transfer canal.

3.2.2.3 Bulk Pool Temperature and Pool Heat-Up Results

The following maximum pool bulk temperatures are calculated to result from the Table 3-13 and Table 3-14 assumptions:

Normal Batch Discharge Case:	138°F
Full Core Discharge Case:	188°F

Under the worst case component cooling inlet extreme temperature condition described in Section 3.2.2.2, the corresponding maximum pool bulk temperatures are calculated to be:

Normal Batch Discharge Case:	150°F
Full Core Discharge Case:	200°F

Based on the conservatisms in the Branch Technical Position ASB 9-2 decay heat methodology, the full power irradiation time, the pool temperature modeling assumptions and the worst case extreme assumptions, these results are considered acceptable.

For the worst case assumptions, resulting pool heat-up rates, times until pool boiling begins, and resulting boil-off rates for a complete loss of pool cooling (starting at the time of the above maximum pool bulk temperatures with the corresponding maximum fuel pool decay heat release rates) are as follows:

Case	Pool Heat-Up Rate - °F/Hr	Time Until Pool Boiling Begins,Hr	Pool Boil-off Rate, Gpm
Normal Batch Discharge	7.30	8.5	37.4
Full Core Discharge	14.6	0.82	75.0

The temperature and level indicators in the spent fuel pool would warn the operator of a loss of cooling. Thus, there is sufficient time to take any necessary action to provide adequate cooling and makeup while the cooling capability of the spent fuel pool cooling is being restored.

The total increase in heat rejected to the environment through the cooling systems due to the increased spent fuel storage over the current rejected heat load is 1.64 MBTU/hr. This represents an increase of less than 0.03 percent of the total heat rejected to the environment during normal plant operation. This increase in rejected heat will have negligible impact on the environment.

3.3 THERMAL-HYDRAULIC ANALYSES FOR THE SPENT FUEL POOL (LOCALIZED)

The primary purpose of the localized thermal-hydraulic analysis is to determine the maximum fuel clad temperatures which may occur as a result of using the spent fuel racks in the Indian Point 3 spent fuel pool. In addition, maximum water temperatures due to gamma heating of rack walls, poison material, and Region 1 water boxes are determined.

3.3.1 Criteria

The criteria used to determine the acceptability of the design from a thermal-hydraulic viewpoint are summarized as follows:

1. The design must allow adequate cooling by natural circulation and by flow provided by the spent fuel pool cooling system. The coolant should remain subcooled at all points within the pool whether or not the cooling system is operational.

2. The rack design must not allow trapped air or steam. Direct gamma heating of the storage cell walls must not result in boiling of the adjacent water.

3.3.2 Key Assumptions

- A conservatively hot assembly is assumed based on a time after reactor shutdown of 120 hours and a peak to average clad heat flux ratio of 1.57.

- All decay energy is assumed to be absorbed in the fuel and surrounding coolant for the hot assembly or natural circulation analysis. (In reality, some gamma radiation will be absorbed in the adjacent cell boxes and poison.)

- For the gamma heating of rack walls, poison, and the Region 1 water boxes, the decay heat absorbed is taken to be proportional to the mass densities of the materials in the spent fuel pool. (In reality, most of the gamma radiation never leaves the fuel assembly due to strong uranium attenuation.) Gamma heating proportional to the mass fraction is roughly equivalent to the assumption of uniform gamma flux in the repeating unit cell.

- A circulation flow path from the South wall or downcomer to a position along the North wall is assumed for the hottest assembly. This derates the flow to the hottest assembly since there will also be flow down the three remaining walls.

- The dominant pressure drops are over estimated by factors of 1.5 for the fuel assembly pressure loss and 2.0 for under rack pressure losses.

3.3.3 Description of Analytical Method and Types of Calculations Performed

The methods used for analyzing the localized thermal-hydraulic aspects of the spent fuel pool involve relatively uncomplicated correlations for friction factors, loss coefficients, and heat transfer coeffficients that make a detailed computer analysis unnecessary. Further simplifying but conservative assumptions reduce the mathematical complexity to the point where hand calculations or programmable calculators are all that are required.

1. Fuel Cladding Temperatures

In this analysis, two recirculation paths are identified for the natural circulation cooling of the Indian Point 3 spent fuel assemblies. A local path where coolant is convectively driven up the hottest assembly and down a "cold" assembly is studied first. A second path flowing under the spent fuel racks, up the hot assemblies, into the mixing region above the racks, and finally down the South wall of the pool to complete the path is then modeled and analyzed. For the local path, the fuel assembly inlet temperature is taken to be the hottest pool bulk temperature of 200°F for full core unload condition. For the second path, the inlet temperature is taken to be the average for the pool. Apart from the estimation of the coolant inlet temperatures to the hot batch of spent fuel, these flow paths are decoupled from the cooling loop and spent fuel pool heat exchanger.

Results including peak clad and coolant temperatures calculated for each path are provided in Table 3-15. For all cases, peak temperatures are well below corresponding saturation temperatures, so no local boiling will occur.

2. Gamma Heating of Rack Walls, Poison and Region 1 Water Box

Conservative estimates of gamma heating in the rack walls, poison and Region 1 water box are made.

The flow rate in the water box is determined by equating the driving head to the loss head. The coolant temperature corresponding to this flow rate is then computed as 220.2 °F for the hottest pool bulk temperature condition (200°F) during full core unloading. This is significantly below the boiling point of the adjacent water.

3.4 POTENTIAL FUEL AND RACK HANDLING ACCIDENTS

Procedures for fuel handling are not different for the new maximum density racks than for the existing racks, thus no potential exists for unreviewed fuel handling accidents.

The method for moving the racks into and out of the spent fuel pool is briefly discussed in Section 4.7.4.2. The sequence of installation of the new racks and removal of the old racks is required to provide paths for empty racks (new or existing) or other heavy loads (2000 lb.) not to be moved over racks storing spent fuel.

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TABLE 3-1

SUMMARY OF LEOPARD RESULTS FOR MEASURED CRITICALS

Case** Number	Reference Number	Enrichment (atom %)	H_2O/U Volume	Fuel Density (g/cm^3)	Pellet Diameter (cm)	Clad Diameter (cm)	Clad Thickness (cm)	Lattice Pitch	Critical Buckling m^{-2}	Calculated k_{eff}
1	16	2.734	2.18	10.18	0.7620	0.8594	0.04085	1.0287	40.75	1.0015
2	16	2.734	2.93	10.18	0.7620	0.8594	0.04085	1.1049	53.23	1.0052
3	16	2.734	3.80	10.18	0.7620	0.8594	0.04085	1.1938	63.28	1.0043
4	17	2.734	7.02	10.18	0.7620	0.8594	0.04085	1.4554	65.64	1.0098
5	17	2.734	8.49	10.18	0.7620	0.8594	0.04085	1.5621	60.07	1.0118
6	17	2.734	10.13	10.18	0.7620	0.8594	0.04085	1.6891	52.92	1.0072
7	18	2.734	2.50	10.18	0.7620	0.8594	0.04085	1.0617	47.5	1.0008
8	18	2.734	4.51	10.18	0.7620	0.8594	0.04085	1.2522	68.8	0.9987
9	18	3.745	2.50	10.37	0.7544	0.8600	0.0406	1.0617	68.3	1.0010
10	18	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.1	1.0025
11	19	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.68	1.0009
12	20	4.069	2.55	9.46	1.1278	1.2090	0.0406	1.5113	88.0	0.9889
13	20	4.069	2.14	9.46	1.1278	1.2090	0.0406	1.450	79.0	0.9830
14	21	4.069	2.59	9.45	1.1268	1.2701	0.07163	1.555	69.25	0.9999
15	21	4.069	3.53	9.45	1.1268	1.2701	0.07613	1.684	85.52	0.9958
16	21	4.069	8.02	9.45	1.1268	1.2701	0.07163	2.198	92.84	1.0040
17	21	4.069	9.90	9.45	1.1268	1.2701	0.07163	2.381	91.79	0.9872
18	21	3.037	2.64	9.28	1.1268	1.2701	0.07163	1.555	50.75	0.9946
19	21	3.037	8.10	9.28	1.1268	1.2701	0.07163	2.198	68.81	0.9809
20	13	0.714 *	1.68	9.52	0.8570	0.9931	0.0592	1.3208	108.8	0.9912
21	13	0.714 *	2.17	9.52	0.8570	0.9931	0.0592	1.4224	121.5	1.0029
22	13	0.714 *	4.70	9.52	0.8570	0.9931	0.0592	1.8669	159.6	0.9944
23	6	0.714 *	10.76	9.52	0.8570	0.9931	0.0592	2.6416	128.4	1.0008
24	14	0.729 *	1.11	9.35	1.2827	1.4427	0.0800	1.7526	89.1	0.9902
25	14	0.729 *	3.49	9.35	1.2827	1.4427	0.0800	2.4785	104.72	1.0055
26	14	0.729 *	3.49	9.35	1.2827	1.4427	0.0800	2.4785	79.5	0.9948
27	14	0.729 *	1.54	9.35	1.2827	1.4427	0.0800	1.9050	90.0	0.9878

* These are PuO_2 in Natural UO_2

** Cases 1 through 19 are with stainless steel clad, Cases 20 through 27 are zircaloy.

TABLE 3-2

WESTINGHOUSE UO_2 Zr-4 CLAD CYLINDRICAL CORE CRITICAL EXPERIMENTS

Experiment	Pitch In	Boron Concentration (ppm)	Material Buckling (for LEOPARD CM-2)	Critical No. of Pins	Radius of Fuel Region (cm)	k_{eff} (LEOPARD/PDQ-7)
1	0.600	0	.008793	489.4	19.021	0.9912
2	0.690	0	.009725	317.0	17.605	0.9941
3	0.848	0	.008637	251.6	19.276	0.9927
4	0.976	0	.006458	293.0	23.935	0.9935
5	0.600	306.	.007177	659.9	22.088	0.9927
6	0.600	536.4	.006244	807.2	24.429	0.9937
7	0.600	727.7	.005572	950.2	26.504	0.9940
8	0.600	104.	.008165	546.3	20.097	0.9919
9	0.600	218.	.007599	607.1	21.186	0.9917
10	0.600	330.	.007106	669.5	22.248	0.9916
11	0.600	446.	.006661	735.3	23.315	0.9909
12	0.600	657.1	.005809	895.3	25.727	0.9944
13	0.848	104.	.007320	321.0	21.772	0.9938
14	0.848	218.	.006073	420.5	24.919	0.9925
						0.9928 Mean
						0.0012 Std

Notes

(a) Fuel Region Data

Enrichment = 2.719 w/o U-235
Fuel Density = 10.41 g/cm^3
Pellet Radius = 0.20 in
Clad IR = 0.2027 in
Clad OR = 0.23415 in

(b) Thickness of water reflector is that required to attain total radius of 50 cm for model.

(c) $B_Z{}^{2}$(PDQ-7) = .000527 cm^{-2}

TABLE 3-3

BATTELLE FIXED NEUTRON POISON CRITICALS

Case	Length Times Width*	No. of Assemblies In Array	Absorber Type	Absorber Thickness	Distance To Fuel Cluster	Critical Separation of Clusters	k_{eff} LEOPARD/PDQ
020	20 x 17	3	Boral	.713 cm	.645 cm	6.34 cm	0.9932
017	22.21 x 16[X]	3	Boral	.713	.645	5.22	0.9944
002	20 x 18.88+	1	Boral	.713	2.732	oo	0.9925
028	20 x 16	3	S.S.	.485 cm	.645 cm	6.88 cm	9.9946
027	20 x 16	3	S.S.	.302	.645	7.43	0.9935
032	20 x 17	3	S.S. 1.1 w/o B	.298 cm	.645 cm	7.56 cm	0.9933
038	20 x 17	3	S.S. 1.6 w/o B	.298	.645	7.36	0.9931
002B	20 x 18.075	1	None	-	-	∞	0.9956
015	20 x 17	3	None	-	-	11.92 cm	0.9942
013	20 x 16	3	None	-	-	8.39	0.9945
022	20 x 15	3	None	-	-	6.39	0.9933
021	20 x 16	3	None	-	-	4.46	0.9946

Statistical Summary:

Series	Number	mean k_{eff}	σ
Boral	3	0.9934	0.0008
S.S.	2	0.9941	0.0006
S.S. (Borated	2	0.9932	0.0001
Fixed Poison Total	7	0.9935	0.0007
Non-Poison Total	5	0.9944	0.0007
Overall	12	0.9939	0.0009

* This is in units of pitch (Pitch = 2.032 cm)

X Center assembly was 20x16 and the outer two were elongated at 22.21x16.

\+ 20x18.88 was one assembly with a boral sheet on two sides.

Fuel region date: Enrichment = 2.35 w/o, Pellet radius = 0.5588 cm, Clad OR = .635 cm, Wall thickness = .0762 cm, Pitch = 2.032 cm.

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TABLE 3-4

WESTINGHOUSE 15 x 15 FUEL ASSEMBLY CHARACTERISTICS (OFA)

Description	
Numbers of rods containing UO_2	204
Rod pitch (in)	0.563
Assembly width (in)	8.426
Active fuel length (in)	144
Fuel Storage Rack design enrichment, w/o	4.5
Instrument tube	
Material	Zr-4
O.D. (in)	0.545
I.D. (in)	0.515
Guide tubes	
Material	Zr-4
O.D. (in), above dashpot	0.545
O.D. (in), in dashpot	0.484
I.D. (in), above dashpot	0.515
I.D. (in), in dashpot	0.454
Fuel pellet	
Material	UO_2
Density (% theoretical)	95
O.D. (in)	.3659
Cladding	
O.D. (in)	.4220
I.D. (in)	.3734
Spacer Grids	
Number	7
Weights of materials	
Inconel grids lbs (2)	4.0
Zircalloy grids lbs (5)	17.55

TABLE 3-5

REGION 1 BASIC CELL k_∞ AS A FUNCTION OF ENRICHMENT

Enrichment, w/o	k_∞
4.25	0.8945
4.50	0.9040
4.75	0.9128

TABLE 3-6

SUMMARY OF PERTURBATIONS TO THE
MULTIPLICATION FACTOR OF THE BASIC CELL FOR THE REGION 1 RACKS

Description	Δk Effect	k_∞
Basic cell at 68°F, 4.5 w/o U-235 W 15 x 15 fuel assembly, 02g B^{10}/cm^2 in .075" Boral, 10.76" rack pitch		0.9040
Calculational Biases		
Most reactive temperature in the range of 68°F to 212°F	Note 1	
Most reactive water density	Note 1	
LEOPARD/PDQ model bias	+0.0067	
Geometric modeling effect	+0.0022	
Axial leakage	-0.0021	
Reduced length Boral panel	+0.0039	
Total Bias	+0.0107	
Basic cell including biases		0.9147
Tolerances and Uncertainties		
Tolerance on SS wall thickness	±0.0005	
Tolerance on Boral thickness	±0.0036	
Tolerance on Boral panel length	±0.0016	
Tolerance on fuel box dimensions	±0.0009	
Tolerance on Boral box dimensions	±0.0033	
Fuel position uncertainty	Note 1	
Tolerance on fuel pellet density	±0.0011	
Calculational uncertainty (95/95)	±0.0027	
Total Uncertainty (statistical)	±0.0060	
Maximum k_∞ including biases and uncertainties for 4.5 w/o fuel		0.9207

Note 1: This reactivity perturbation is negative and ignored for conservatism.

TABLE 3-7

SAXTON PuO_2-UO_2 CRITICAL EXPERIMENTS

Expt.	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (Inches)	k_{eff}	$k_{eff}-1$
1	0	1.68	.520	.9912	-.0088
2	0	2.17	.560	1.0029	+.0029
3	337	2.17	.560	1.0084	+.0084
4	0	4.70	.735	.9944	-.0056
5	0	10.76	1.040	1.0008	+.0008

TABLE 3-8

ESADA PuO_2-UO_2 CRITICAL EXPERIMENTS

Expt.	Boron (ppm)	Pu-240 (%)	H_2O/UO_2 (Volume)	Pitch (Inches)	k_{eff}	$k_{eff}-1$
1	0	8	1.11	.690	.9902	-.0098
2	0	8	3.49	.9758	1.0055	+.0055
3	526	8	3.49	.9758	.9949	-.0051
4	0	24	3.49	.9758	.0948	-.0052
5	0	8	1.54	.750	.9878	-.0122
6	526	8	1.11	.690	.9945	-.0055

TABLE 3-9

SUMMARY OF PREDICTIONS FOR k_{eff}
IN CRITICALITY EXPERIMENTS

Experiment	Cases	k_{eff}
Saxton PuO_2-UO_2	5	0.9995 ± .0068
Esada PuO_2-UO_2	6	0.9946 ± .0061
All PuO_2-UO_2	11	0.9969 ± .0066

TABLE 3-10

SUMMARY OF PERTURBATIONS TO THE
MULTIPLICATION FACTOR OF THE BASIC CELL FOR THE REGION 2 RACKS

Description	Δk Effect	k_∞
Basic cell at 68°F, 4.50 w/o U-235, 36,000 MWD/MTU, .02g B^{10}/cm^2 in .075" Boral, 9.075" rack pitch		0.9241
Calculational Biases		
LEOPARD/PDQ model bias	+0.0057	
Geometric modeling effect	+0.0050	
Most reactive water density	Note 1	
Most reactive temperature over operating range	Note 1	
Axial leakage	-0.0022	
Reduced length Boral panel	+0.0010	
Total Bias	+0.0095	
Basic cell including biases		0.9336
Tolerances and Uncertainties (95/95)		
Depleted fuel reactivity uncertainty	±0.0104	
Tolerance on box dimensions	±0.0010	
Tolerance on stainless steel wall thickness	±0.0003	
Tolerance on Boral thickness	±0.0043	
Tolerance on Boral panel length	±0.0005	
Tolerance on fuel pellet density	±0.0017	
Calculations uncertainty (2.17σ)	±0.0086	
Total Uncertainty (statistical combination)	+0.0143	
Maximum k_∞ including biases and uncertainties		0.9479
Total Biases, Tolerances, and Uncertainties	0.0238	

Note 1: This reactivity perturbation is negative and ignored for conservatism.

TABLE 3-11

REGION 2 k_∞ AS A FUNCTION OF
INTITAL ENRICHMENT AND BURNUP

Initial Enrichment, w/o	Burnup MWD/MT	Region 2 k_∞
2.25	5,000	.9513
2.25	9,000	.9138
2.25	15,000	.8639
3.00	12,000	.9689
3.00	18,000	.9221
3.00	24,000	.8785
3.75	21,000	.9655
3.75	27,000	.9247
3.75	33,000	.8853
4.50	30,000	.9609
4.50	36,000	.9241
4.50	42,000	.8877

TABLE 3-12

REGION 2 MINIMUM BURNUP AS A FUNCTION OF ENRICHMENT TO OBTAIN A k_{∞} OF 0.9245 PRIOR TO THE ADDITION OF TOLERANCES AND UNCERTAINTIES

Initial Enrichment, w/o	Minimum Burnup MWD/MT
2.25	7,900
3.00	17,700
3.75	27,100
4.50	36,000

TABLE 3-13

DECAY HEAT ANALYSIS ASSUMPTIONS

1. Normal Batch Discharge Case:

   - Irradiation time: 1050 Days

   - Addition of the most recent batch : 145 hours after shutdown

   - Batch size: 76 assemblies

   - Total assemblies in the pool: 1152 (1345-193 Full Core Reserve)*

2. Full Core Discharge Case

   - Irradiation time: 76 assemblies 1050 Days
     76 assemblies 666 Days
     41 assemblies 666 Days

   - Addition of full core: 268 hours after shutdown.

   - Total assemblies in the pool: 1345*

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* The pool has a total storage capacity of 1345 storage cells. It is conservatively assumed that 14 batches of 76 assemblies each have been previously discharged at 20 month intervals with an additional 12 assemblies included in the first discharge. For the Full Core Discharge case an additional batch of 76 assemblies with a time out of the reactor of 36 days is assumed prior to the reactor shutdown for addition of the full core. Each assembly in these previous discharges has had 1050 days of exposure at full power (15.67 MWt).

TABLE 3-14

ASSUMED HEAT TRANSFER DATA FOR THE SPENT FUEL POOL HEAT EXCHANGER

Type:	Shell and U-Tube
Tube Side (Spent Pool Fuel Water) Flow Rate, lb/hr:	1.1×10^6
Shell Side (Component Cooling Water) Flow Rate lb/hr:	1.4×10^6
Design Heat Transfer Rate, Btu/hr:	7.96×10^6 *
Component Cooling Water Inlet Temperature, °F:	88.2 **

* Value per FSAR Update and component installation information and value used in previous reracking analysis. The Heat Exchanger Effectiveness is assumed to be 90 percent of design for the current reracking analysis.

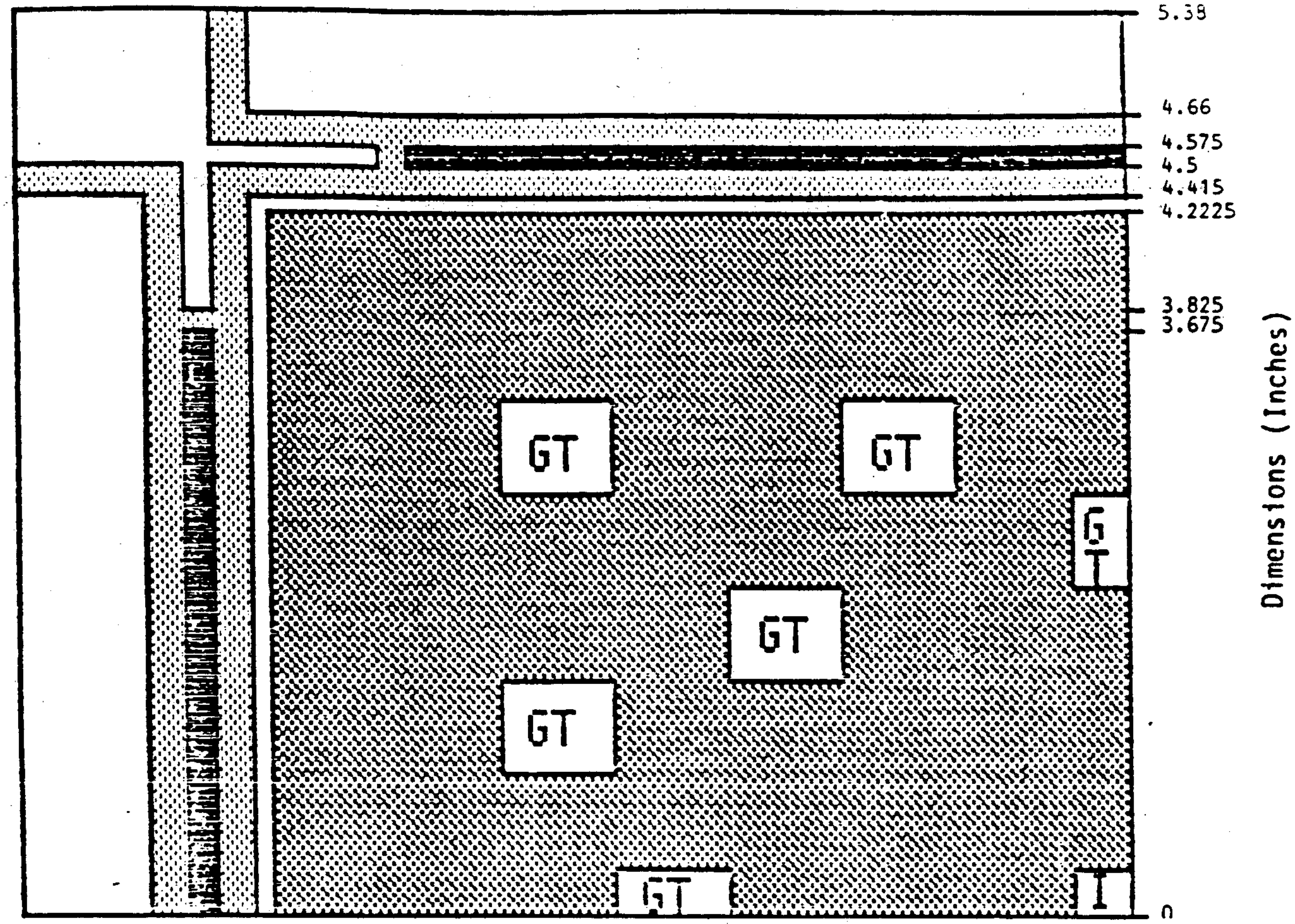
** Value for operating outlet temperature on shell side (component cooling water) of Component Cooling Heat Exchanger per FSAR Update and value used in previous reracking analysis. Value corresponds to a operating inlet temperature of 75°F on tube side (service water) of Component Cooling Heat Exchanger.

TABLE 3-15

PEAK COOLANT AND CLAD TEMPERATURES RESULTS DATA

		Path 1	Path 2
Volume flow rate in the hottest assembly	ft^3 /sec (GPM)	.0362 16.2	.0318 14.2
Position of Clad Hot Spot - ft		7.75	7.16
Peak Coolant Temperature, Normal Batch Discharge, Worst Case Condition - °F		172.9	168.1
Peak Clad Temperature, Normal Batch Discharge, Worst Case Condition - °F		187.2	180.9
Peak Coolant Temperature, Full Core Unload, Worst Case Condition - °F		222.9	210.1
Peak Clad Temperature, Full Core Unload, Worst Case Condition - °F		237.2	222.9
Saturation Temperature at Top of Racks - °F		241.8	241.8
Saturation Temperature at Position of Clad Hot Spot - °F		245.	245.

SECTION 3 FIGURES



LEGEND

GT	Guide Tube		Fuel Pin Cell
I	Instrument Tube		Stainless Steel
	Boral [a]		Water

---

[a] Boral width is conservatively modeled to be 98% of the actual width to account for the cutouts for welds along the edges.

FIGURE 3-1
REGION 1 PDQ QUARTER CELL CALCULATIONAL MODEL

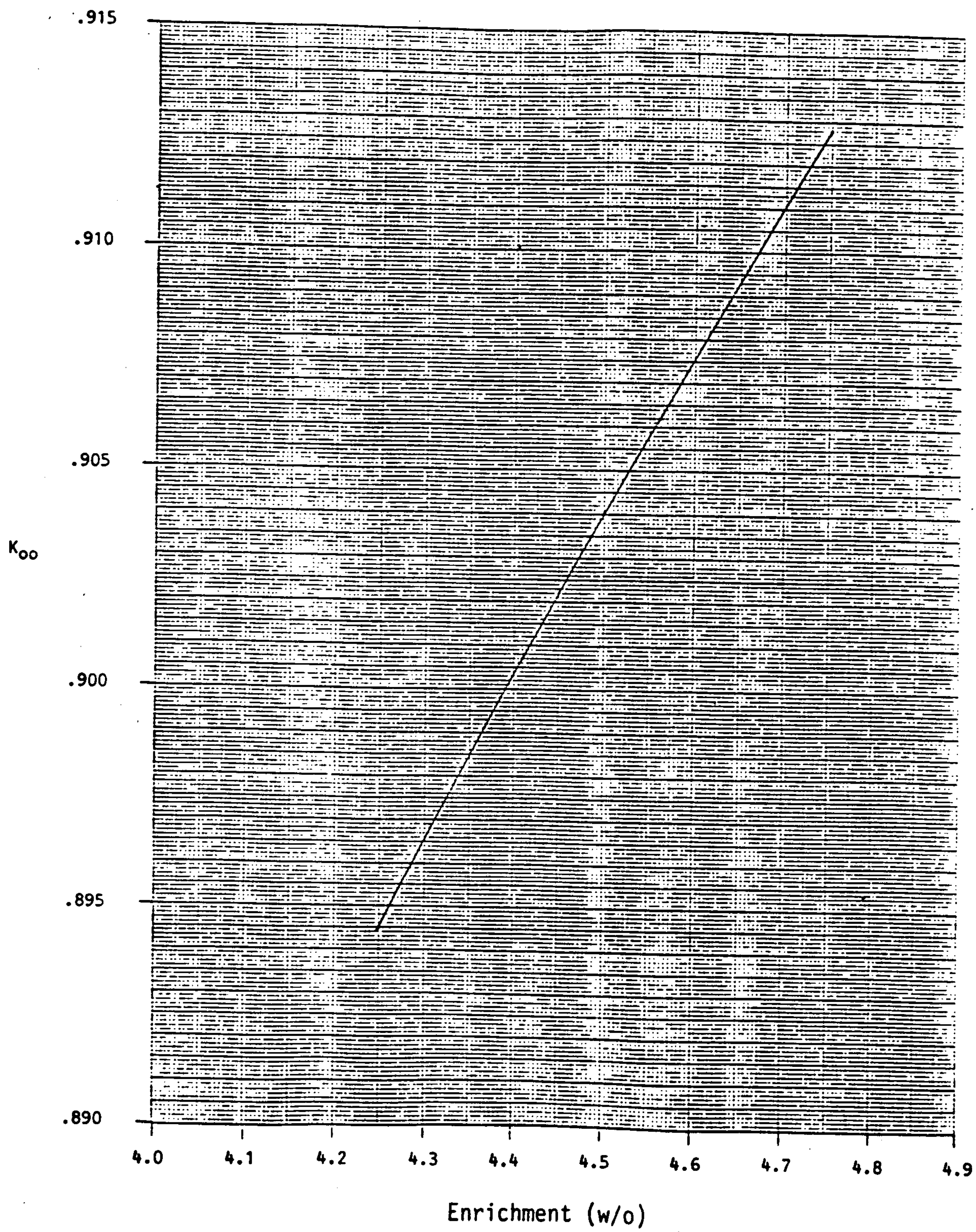


FIGURE 3-2
REGION 1 k_{∞} AS A FUNCTION OF ENRICHMENT

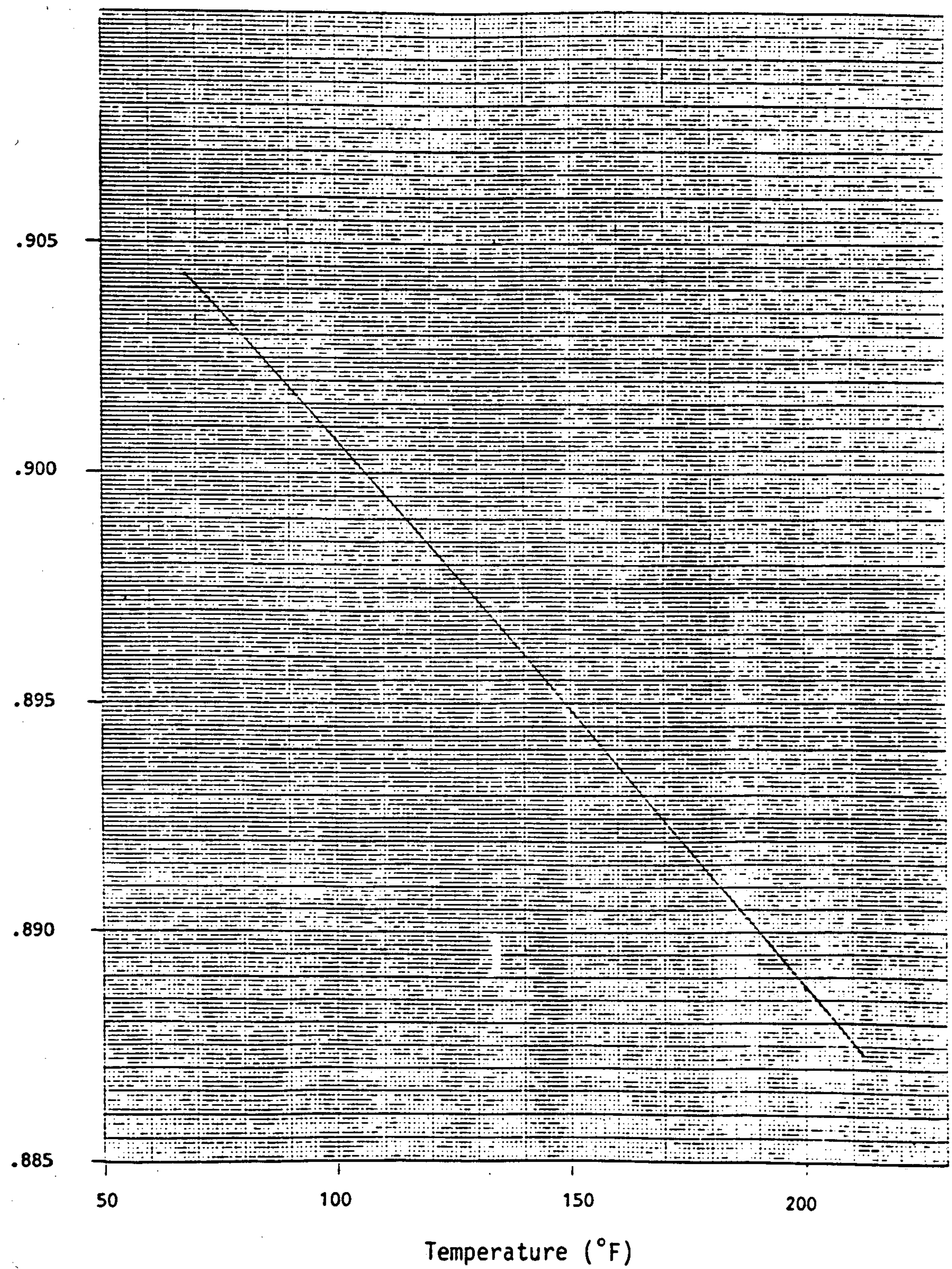


FIGURE 3-3

REGION 1 VARIATION OF k_{∞} WITH WATER TEMPERATURE

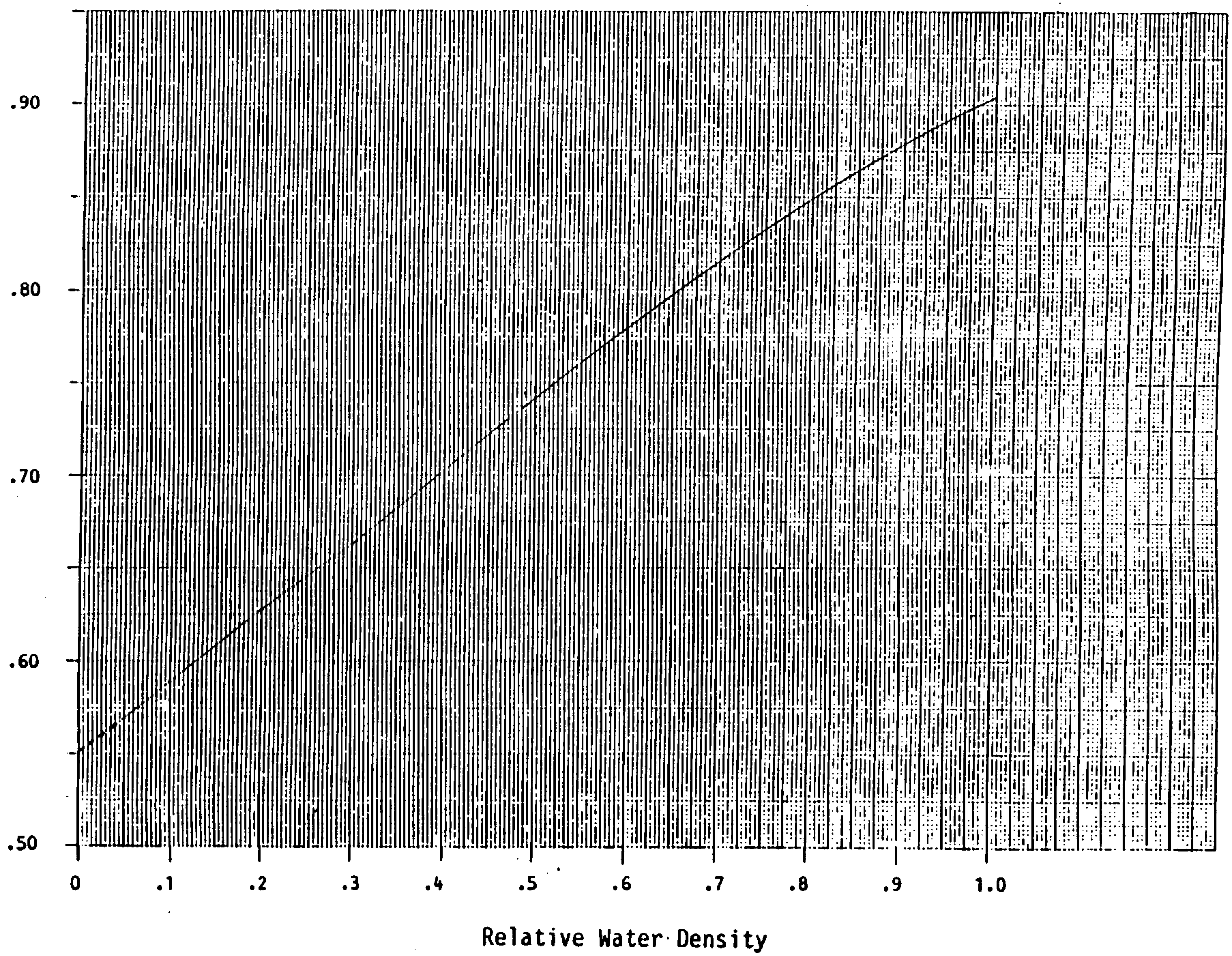


FIGURE 3-4
REGION 1 VARIATION OF k_{∞} WITH WATER DENSITY

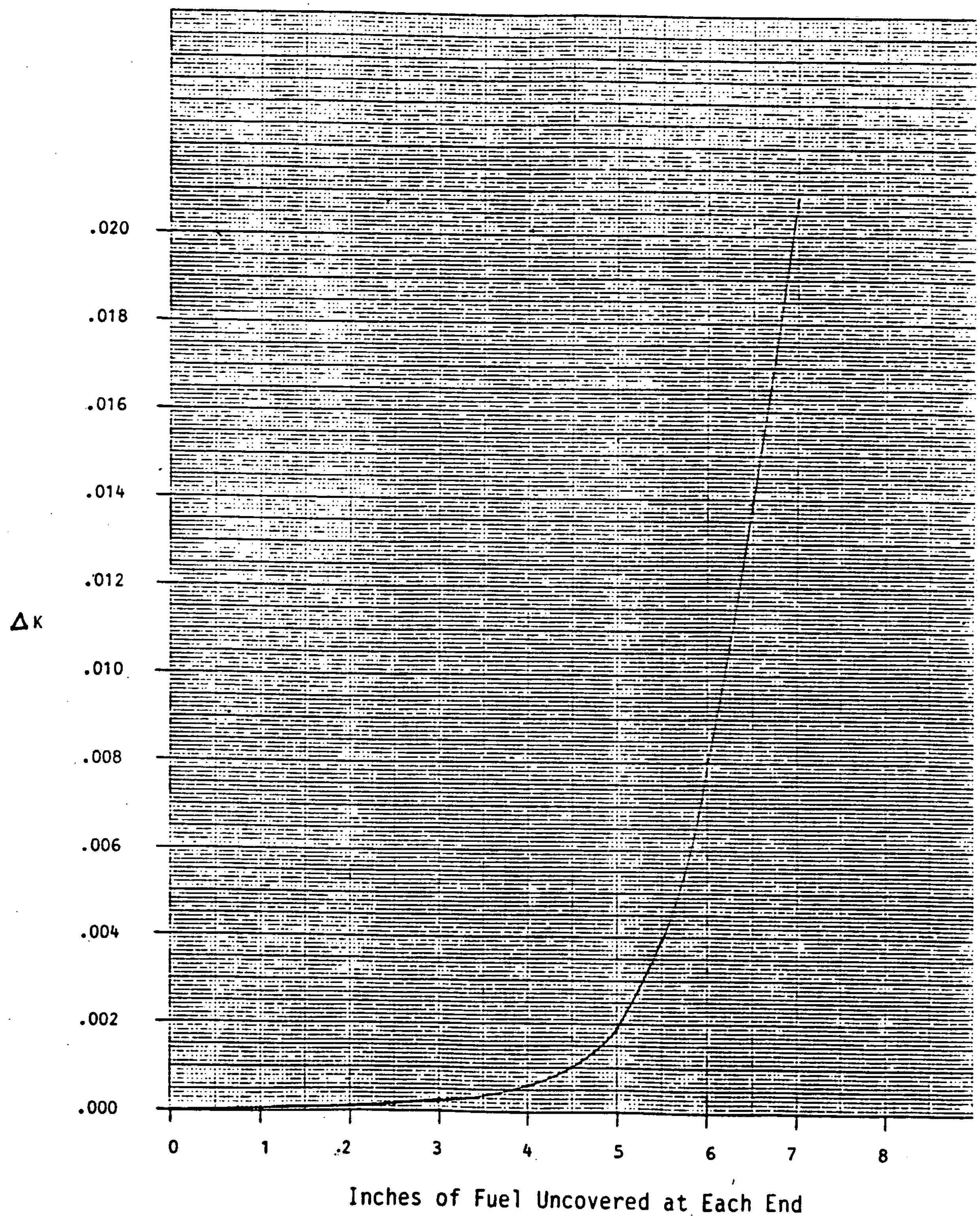


FIGURE 3-5
REGION 1 VARIATION OF k_{∞} WITH BORAL PANEL LENGTH

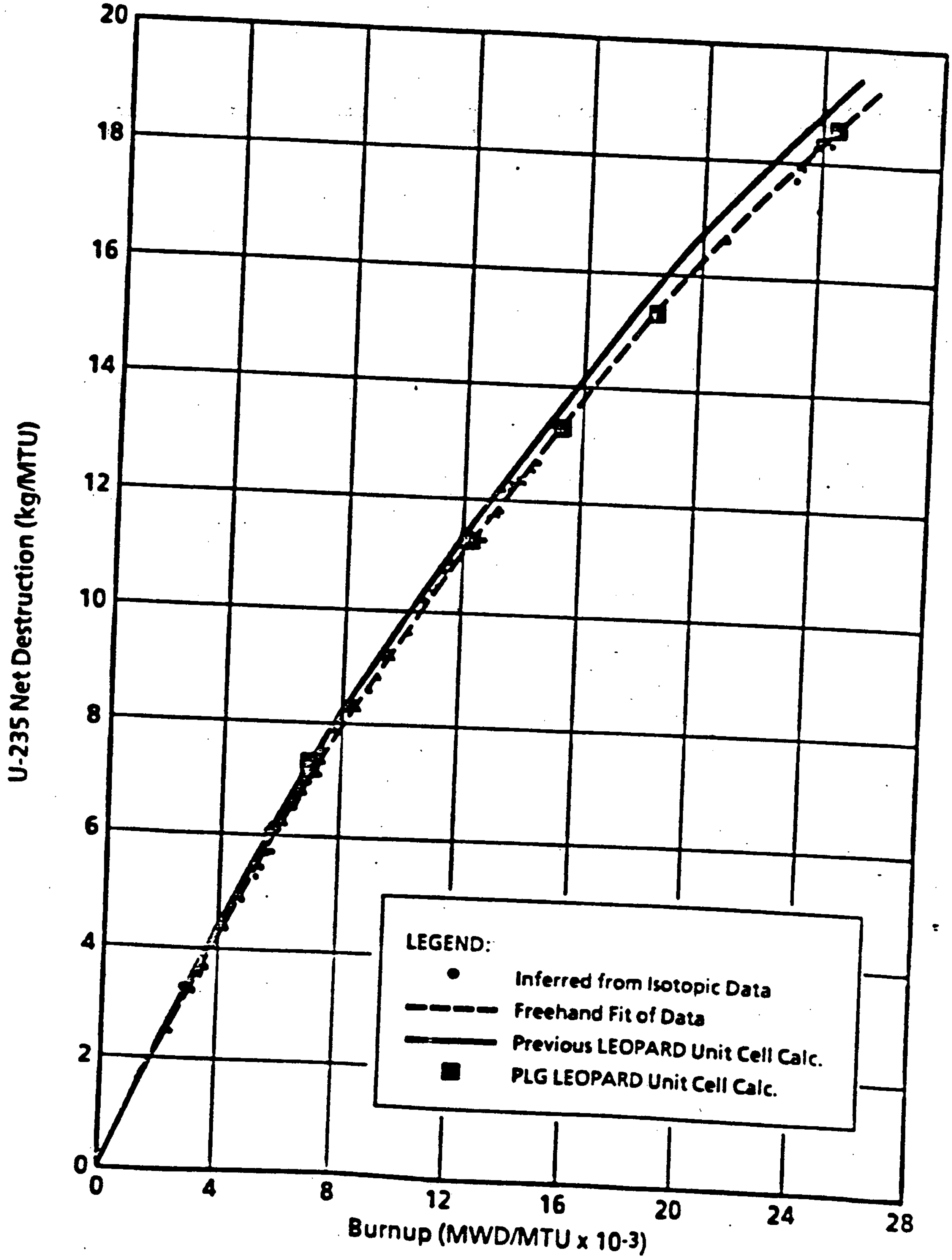


FIGURE 3-6
NET DESTRUCTION OF U-235 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

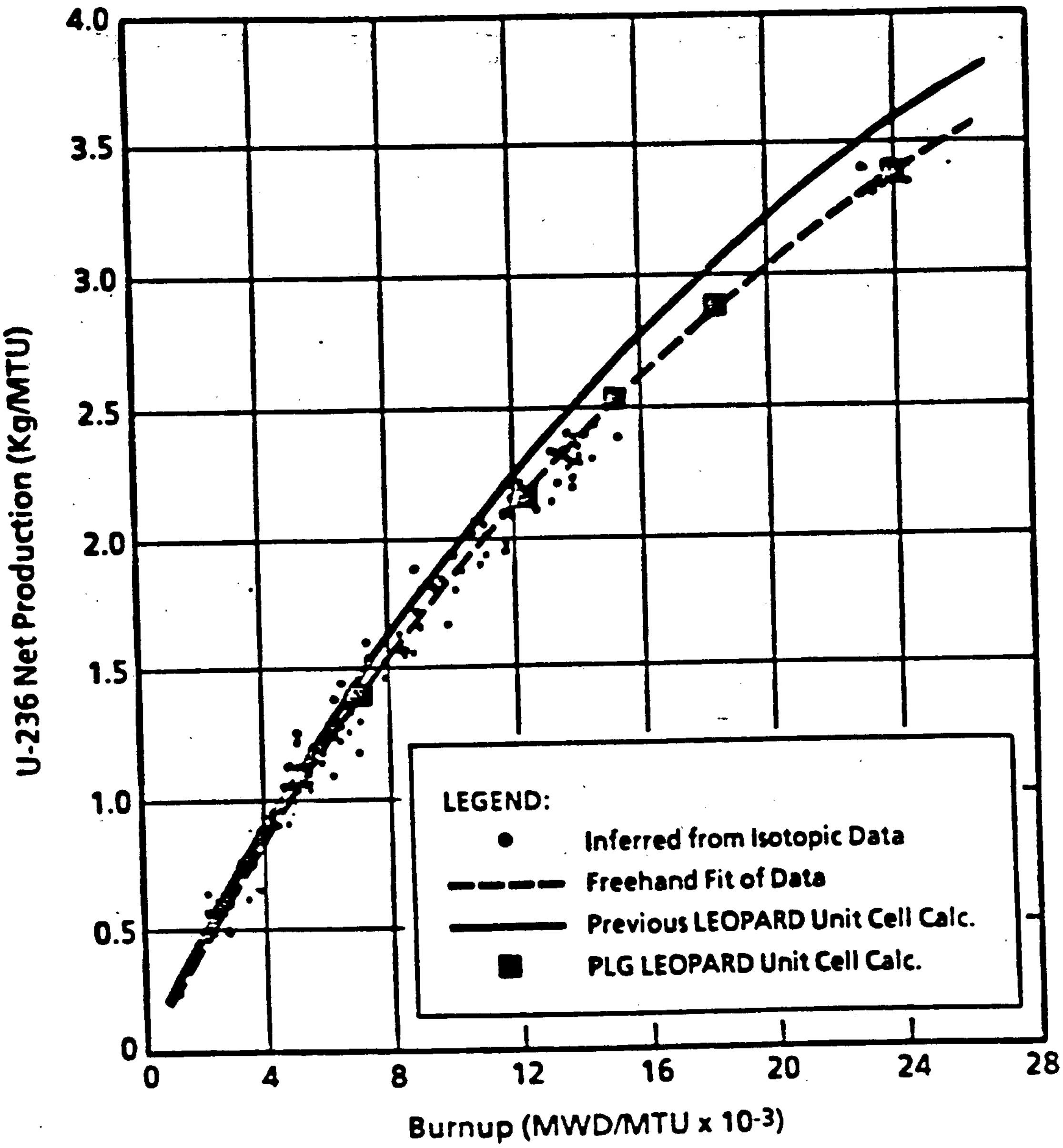


FIGURE 3-7
SPECIFIC PRODUCTION OF U-236 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

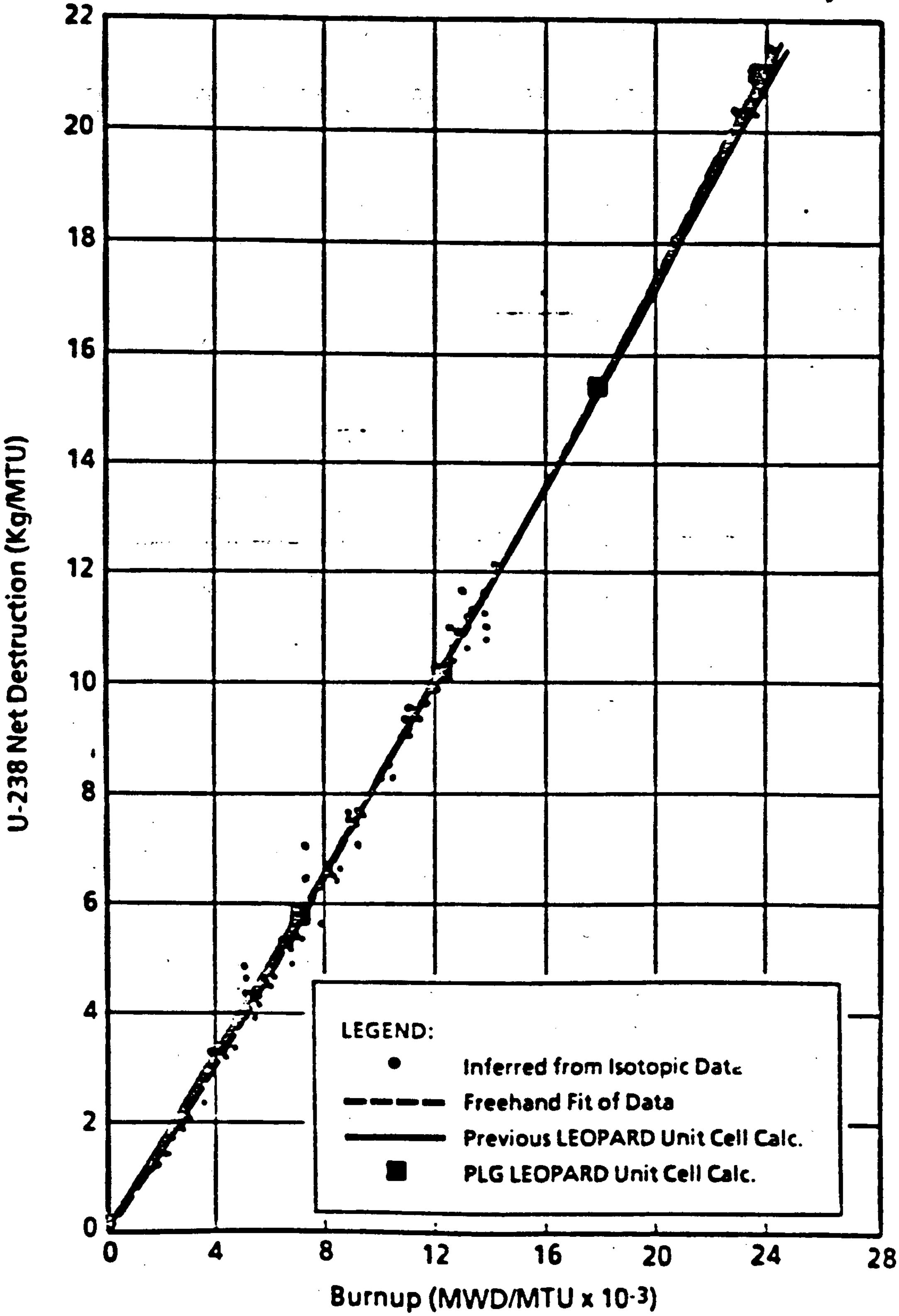


FIGURE 3-8
NET DESTRUCTION OF U-238 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

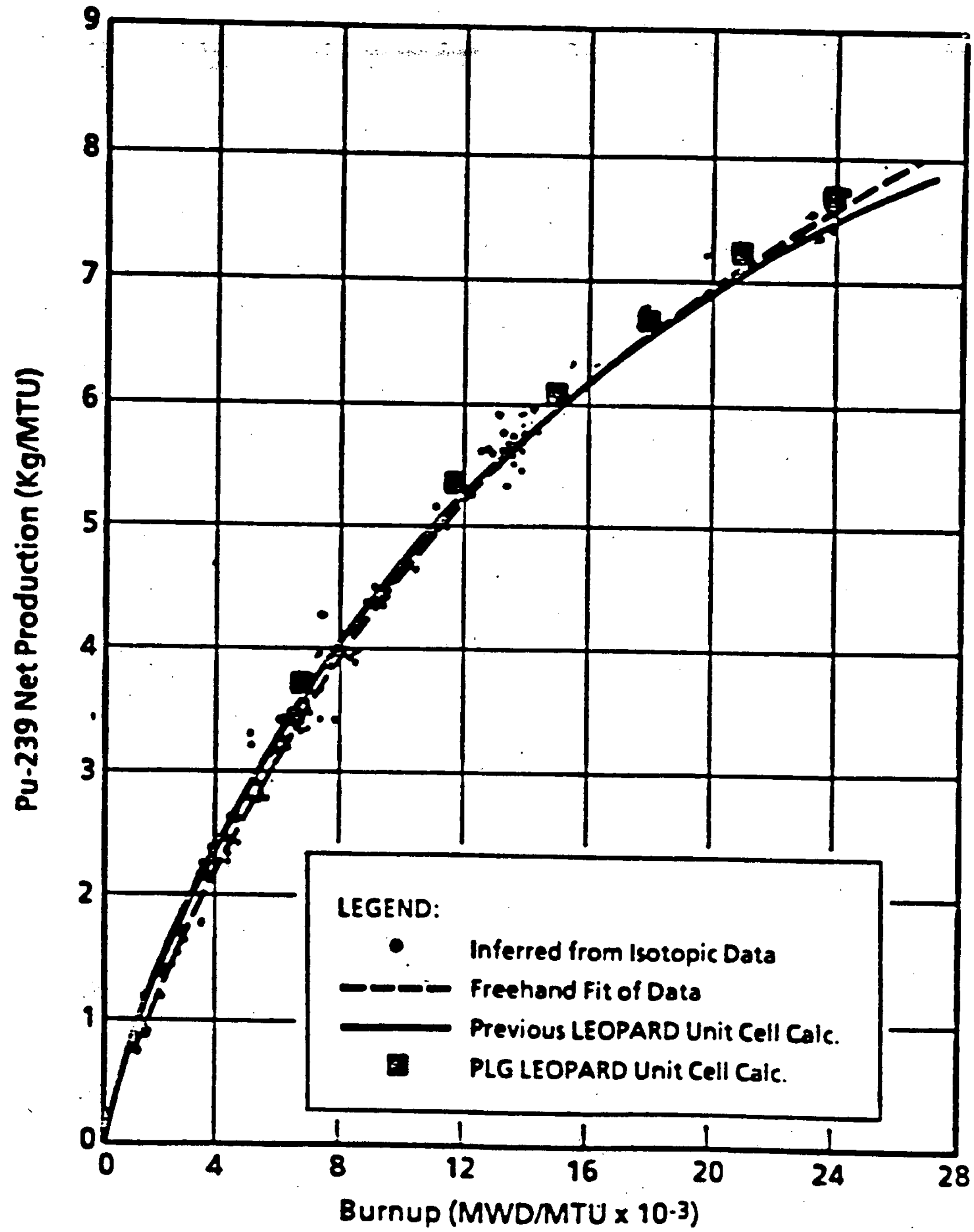


FIGURE 3-9
SPECIFIC PRODUCTION OF PU-239 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

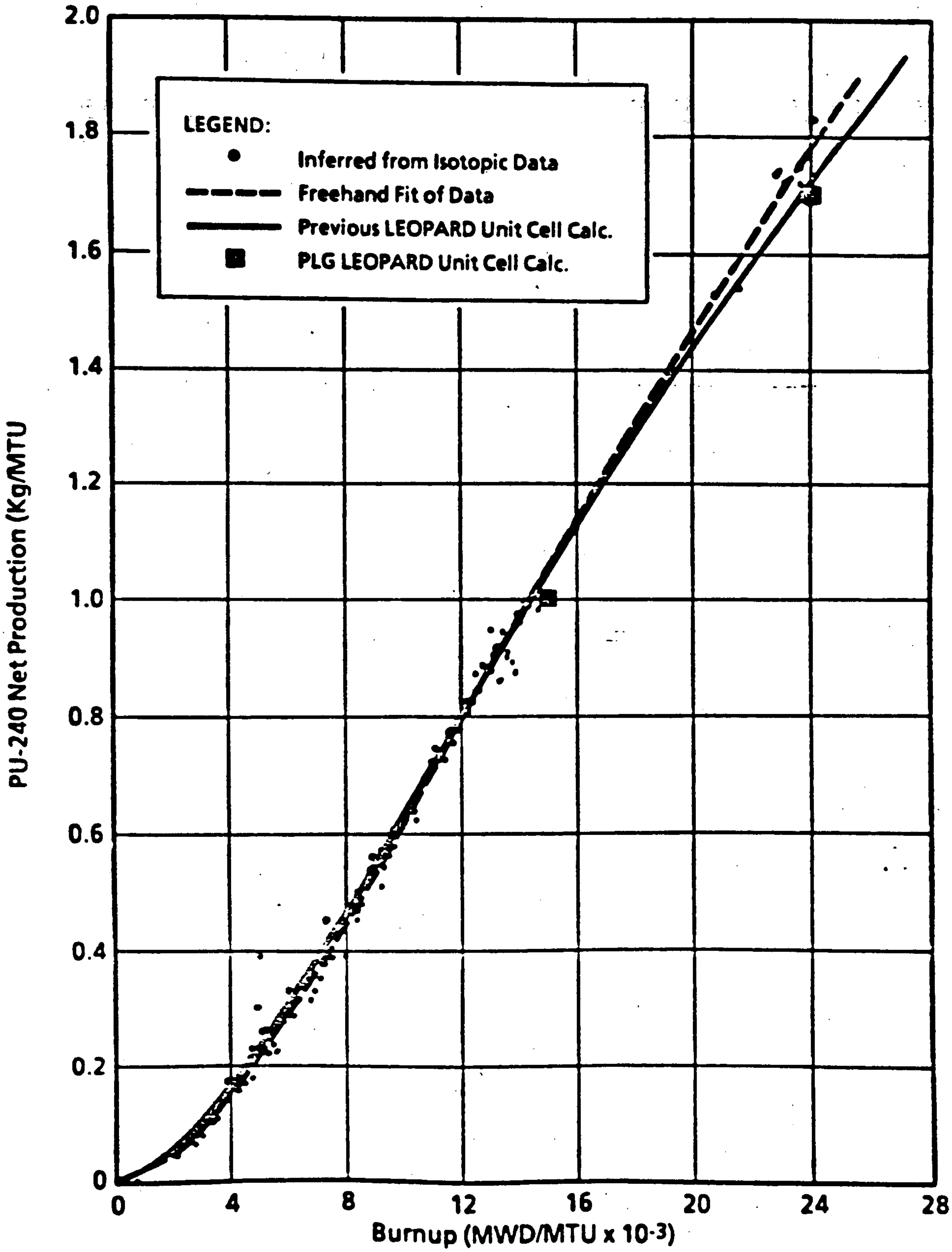


FIGURE 3-10
SPECIFIC PRODUCTION OF PU-240 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

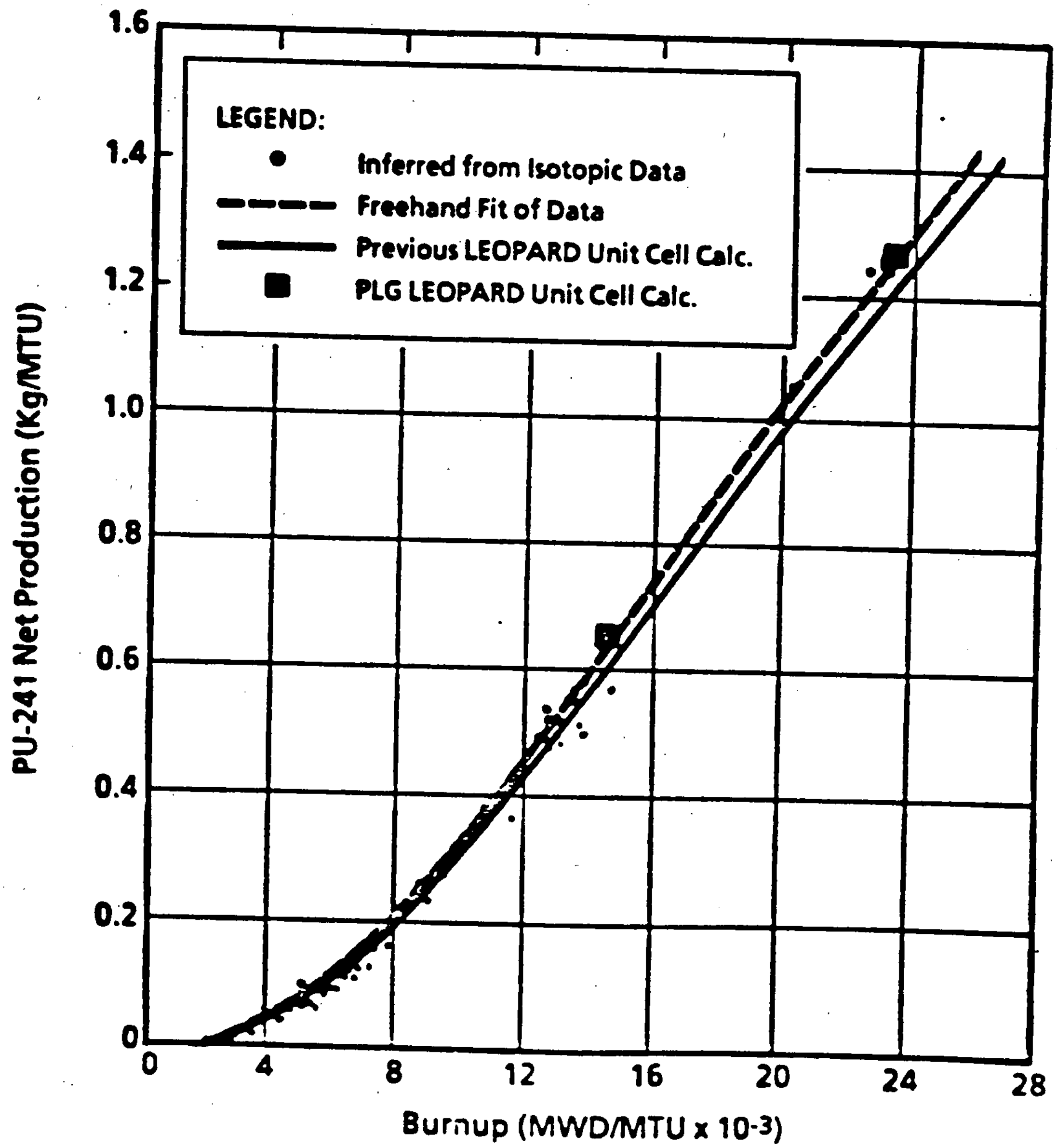


FIGURE 3-11
SPECIFIC PRODUCTION OF PU-241 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

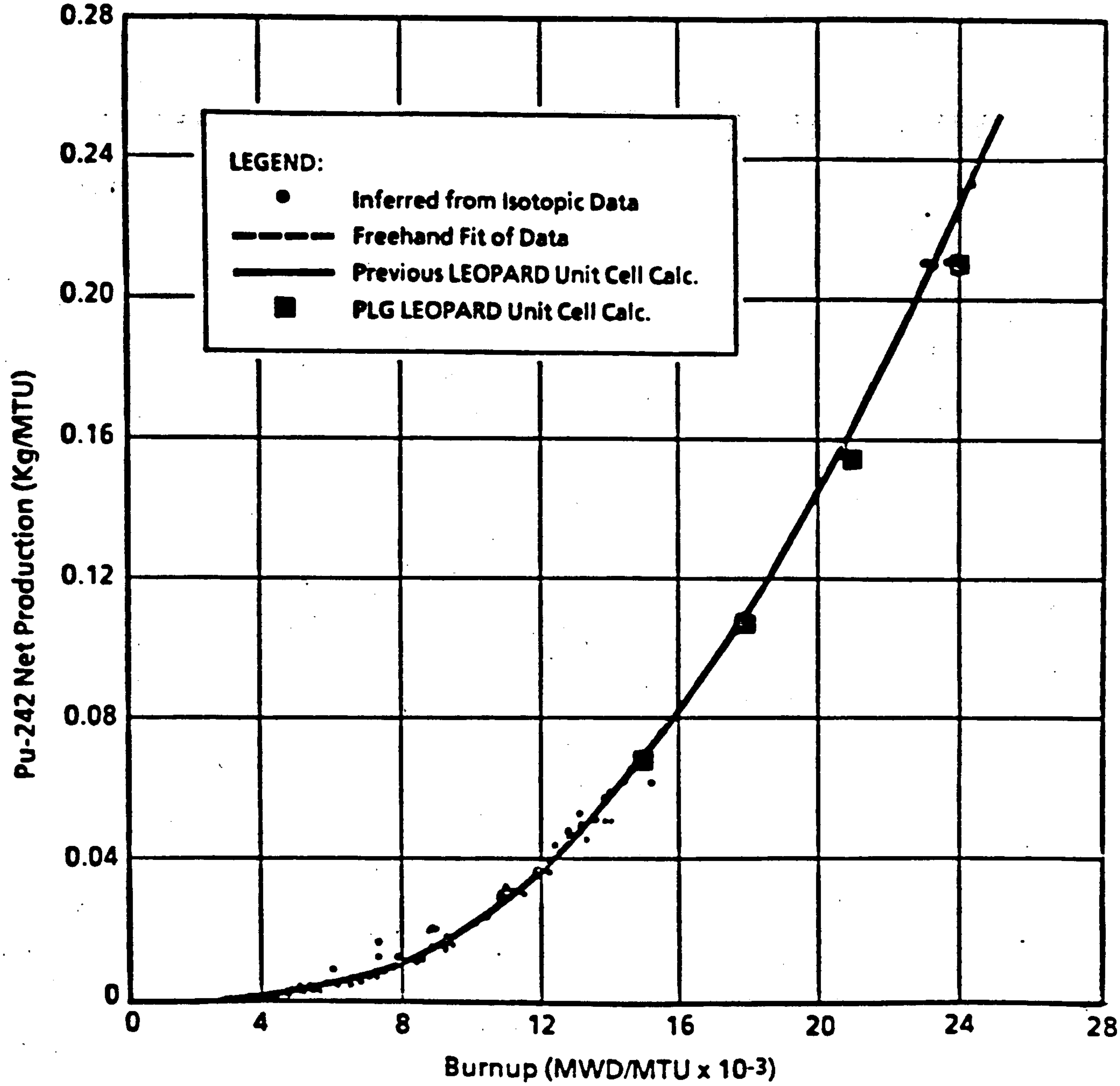


FIGURE 3-12
SPECIFIC PRODUCTION OF PU-242 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

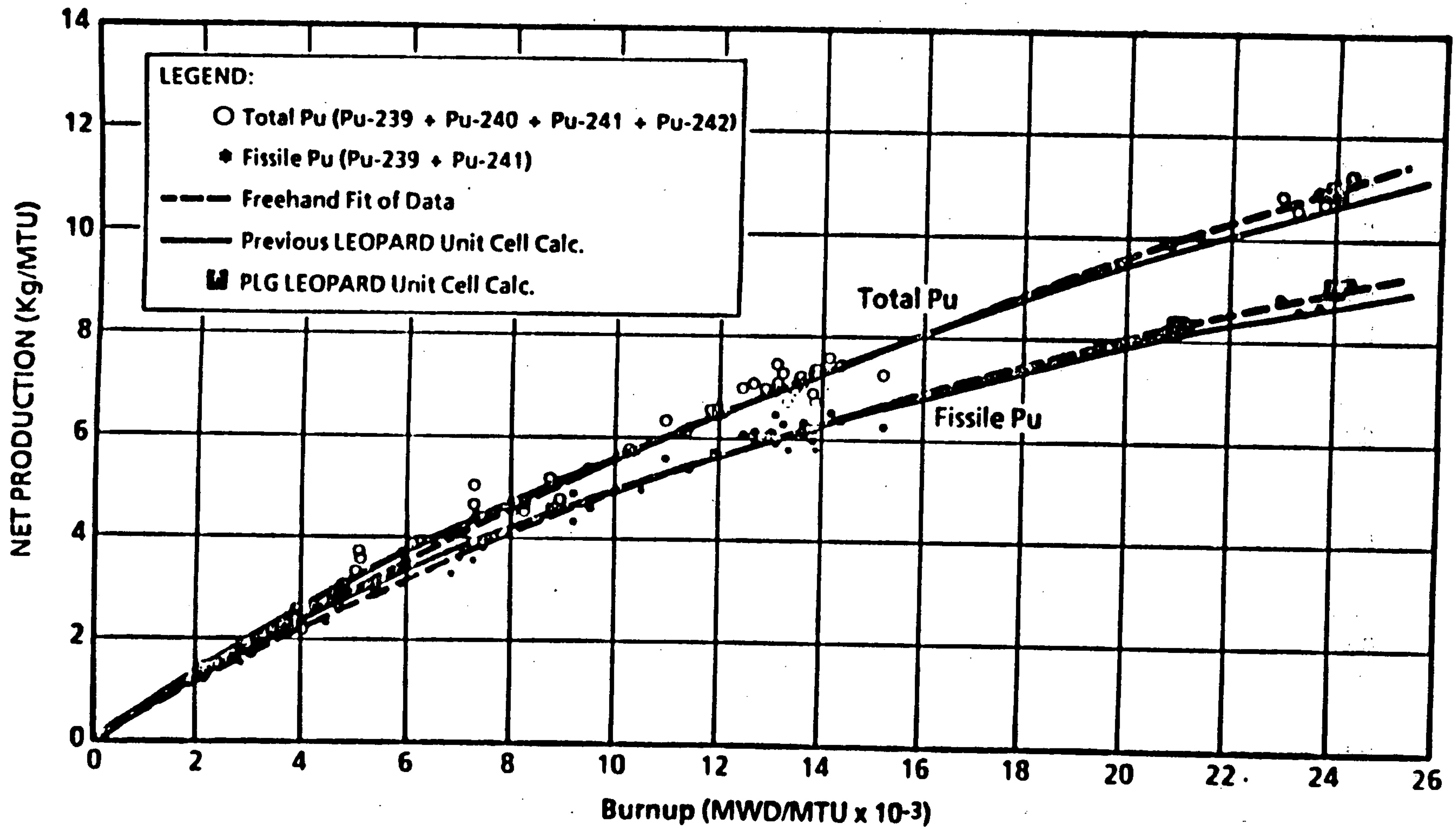


FIGURE 3-13
SPECIFIC PRODUCTION OF TOTAL PU AND FISSILE PU BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

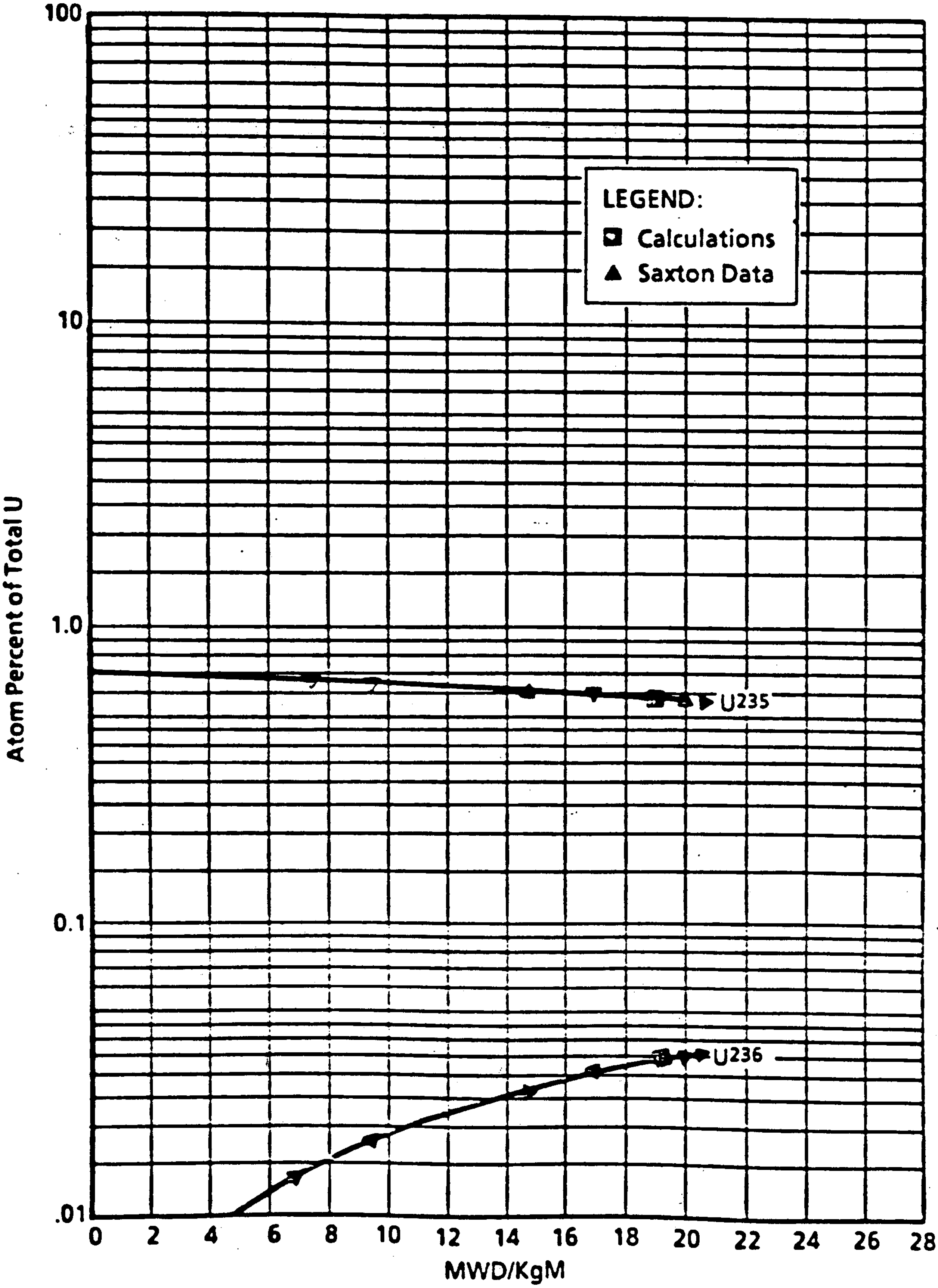


FIGURE 3-14
ATOM PERCENT OF TOTAL U VERSUS EXPOSURE

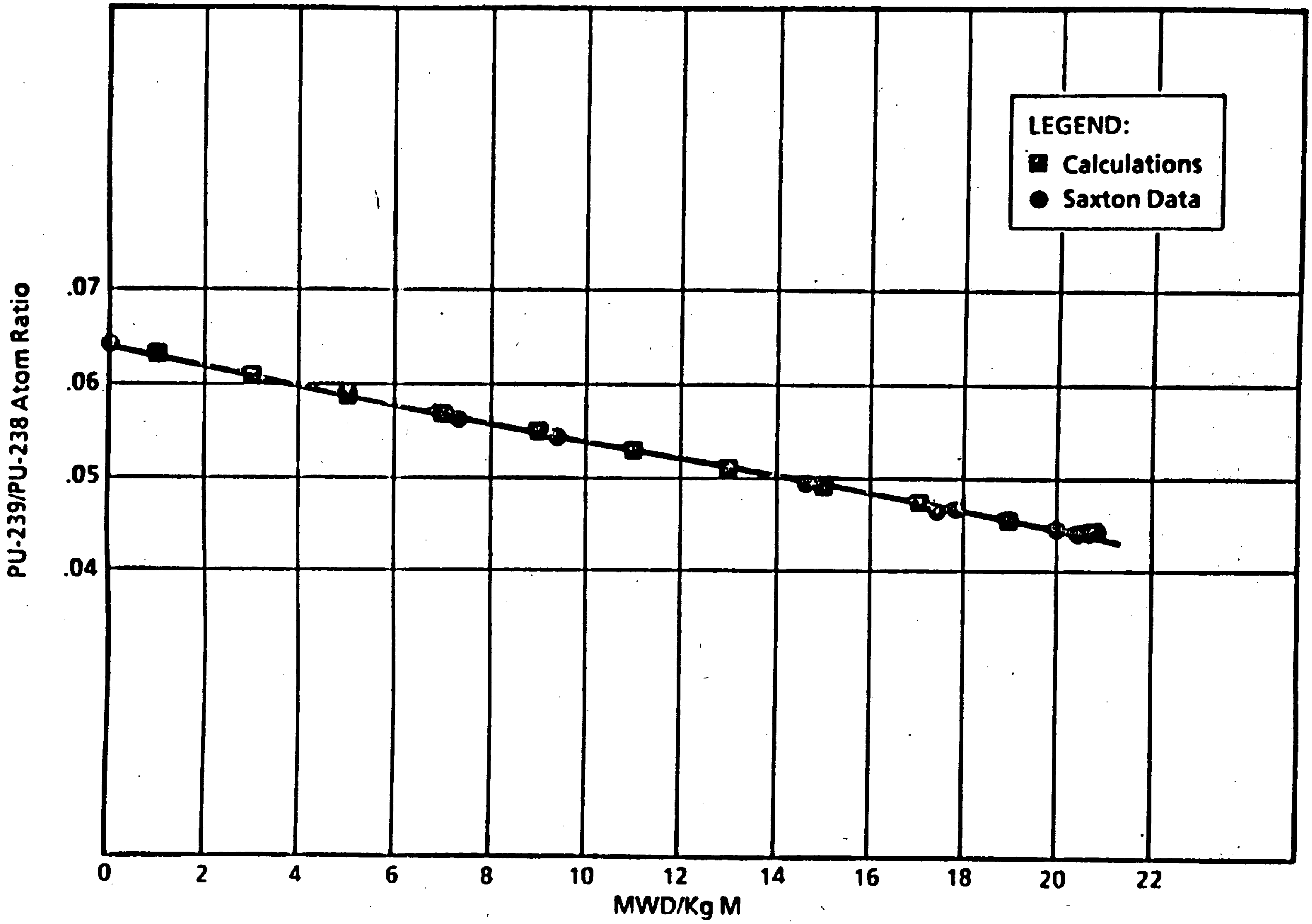


FIGURE 3-15
PU-239/U-238 ATOM RATIO VERSUS EXPOSURE

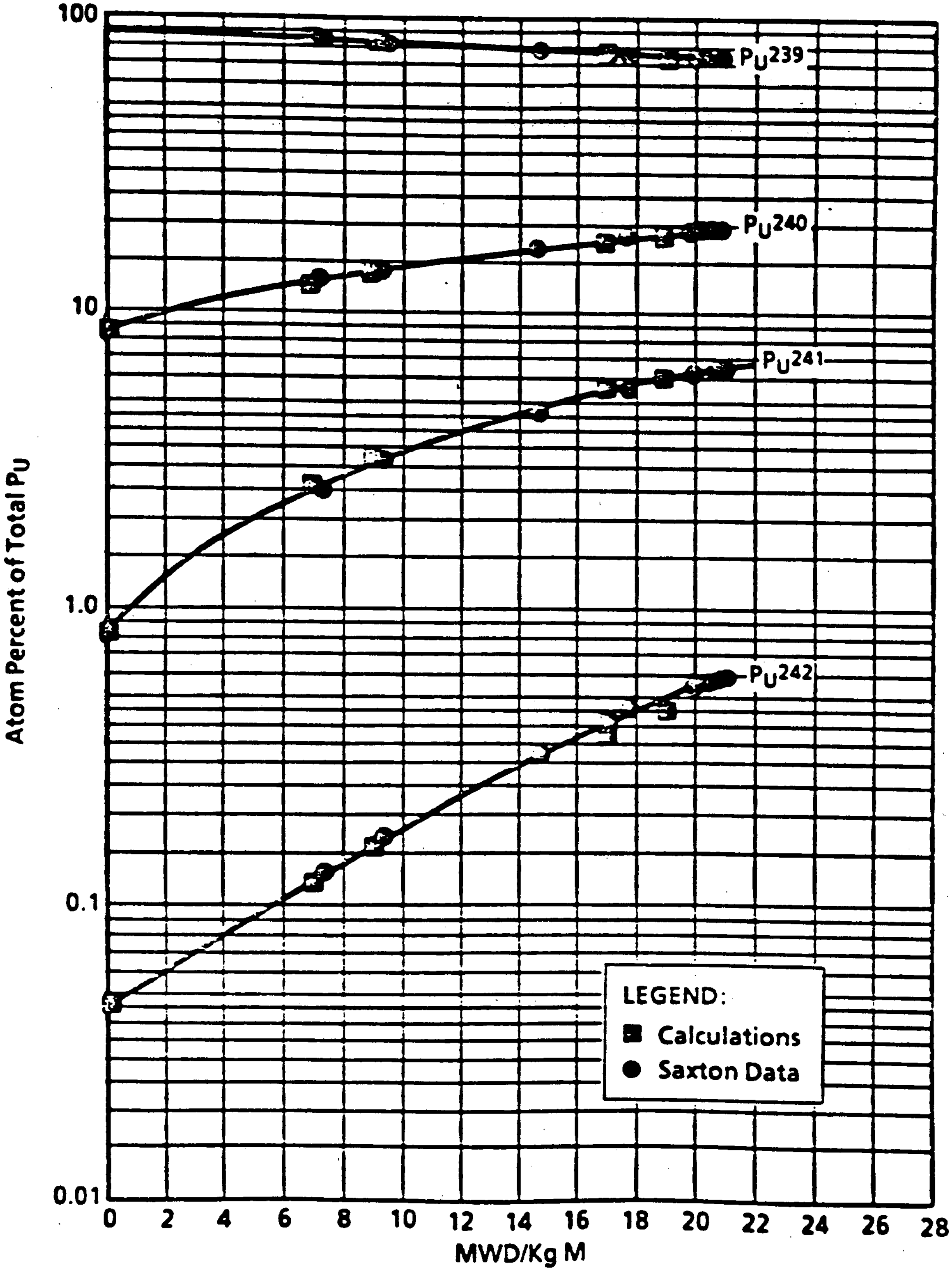
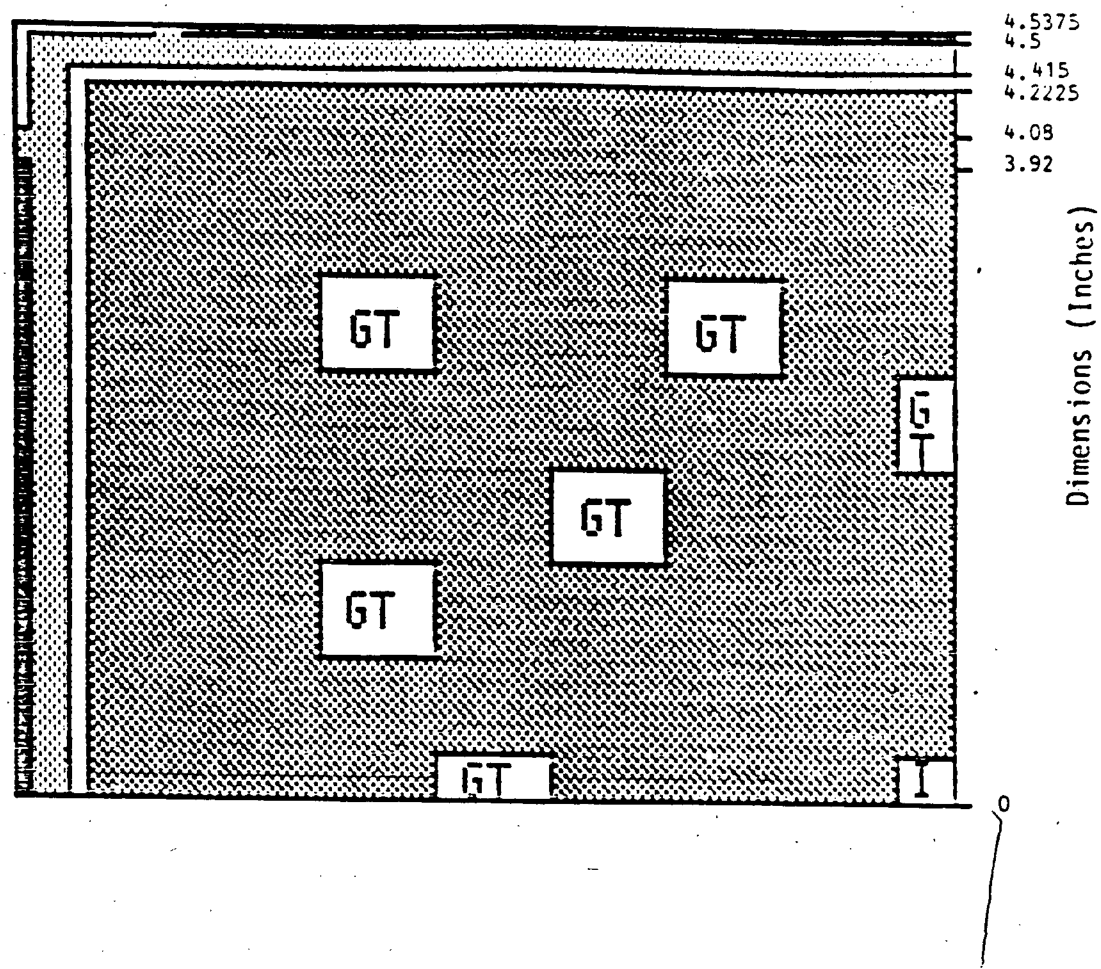


FIGURE 3-16
ATOM PERCENT OF TOTAL PU VERSUS EXPOSURE



LEGEND

GT Guide Tube

I Instrument Tube

Boral [a]

Fuel Pin Cell

Stainless Steel

Water

[a] Boral width is conservatively modeled to be 98% of the actual width to account for the cutouts for welds along the edges.

FIGURE 3-17
REGION 2 PDQ QUARTER CELL CALCULATIONAL MODEL

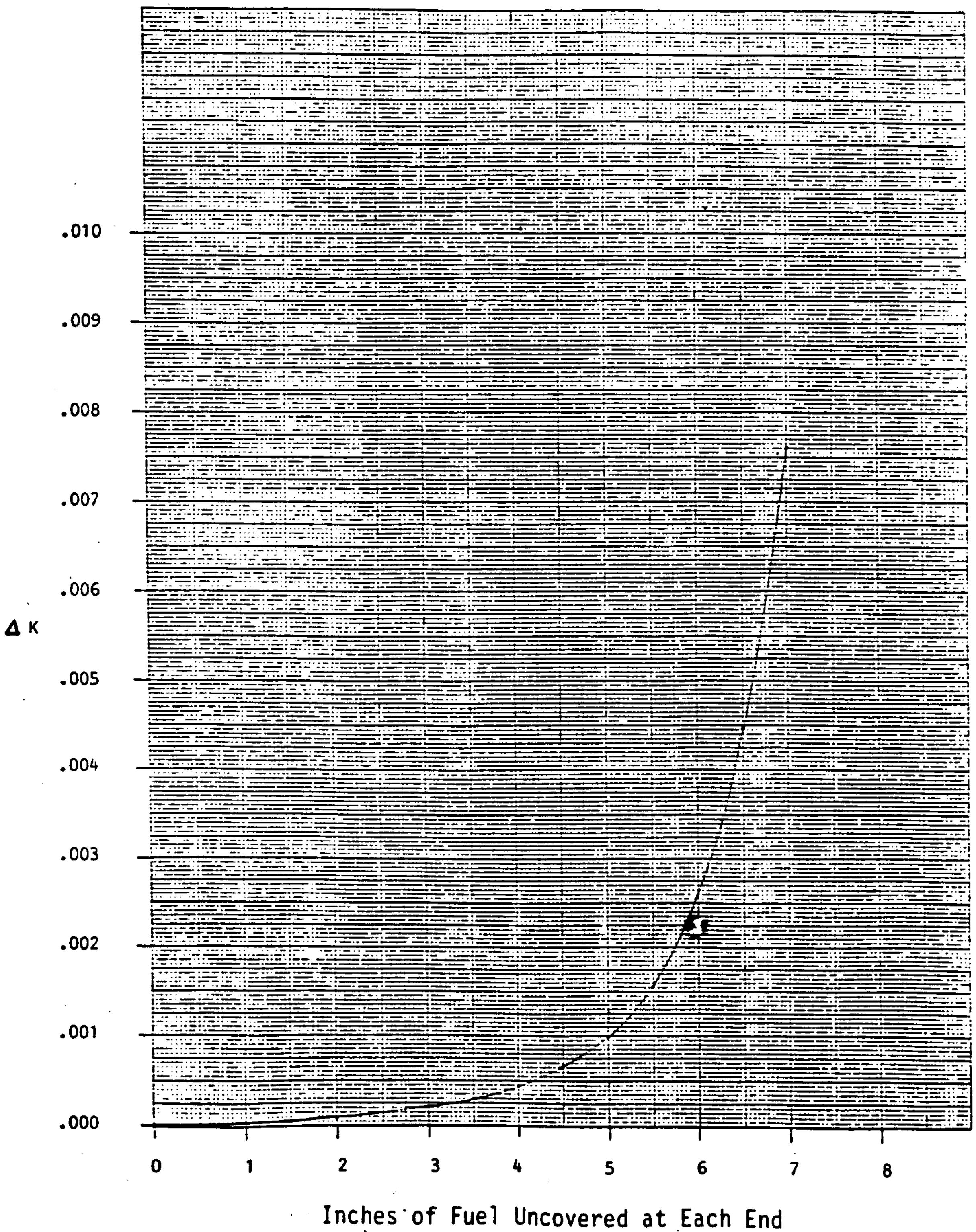


FIGURE 3-18
REGION 2 VARIATION OF k_{∞} WITH BORAL PANEL LENGTH
(4.5 w/o FUEL AT 36,000 MWD/MTU)

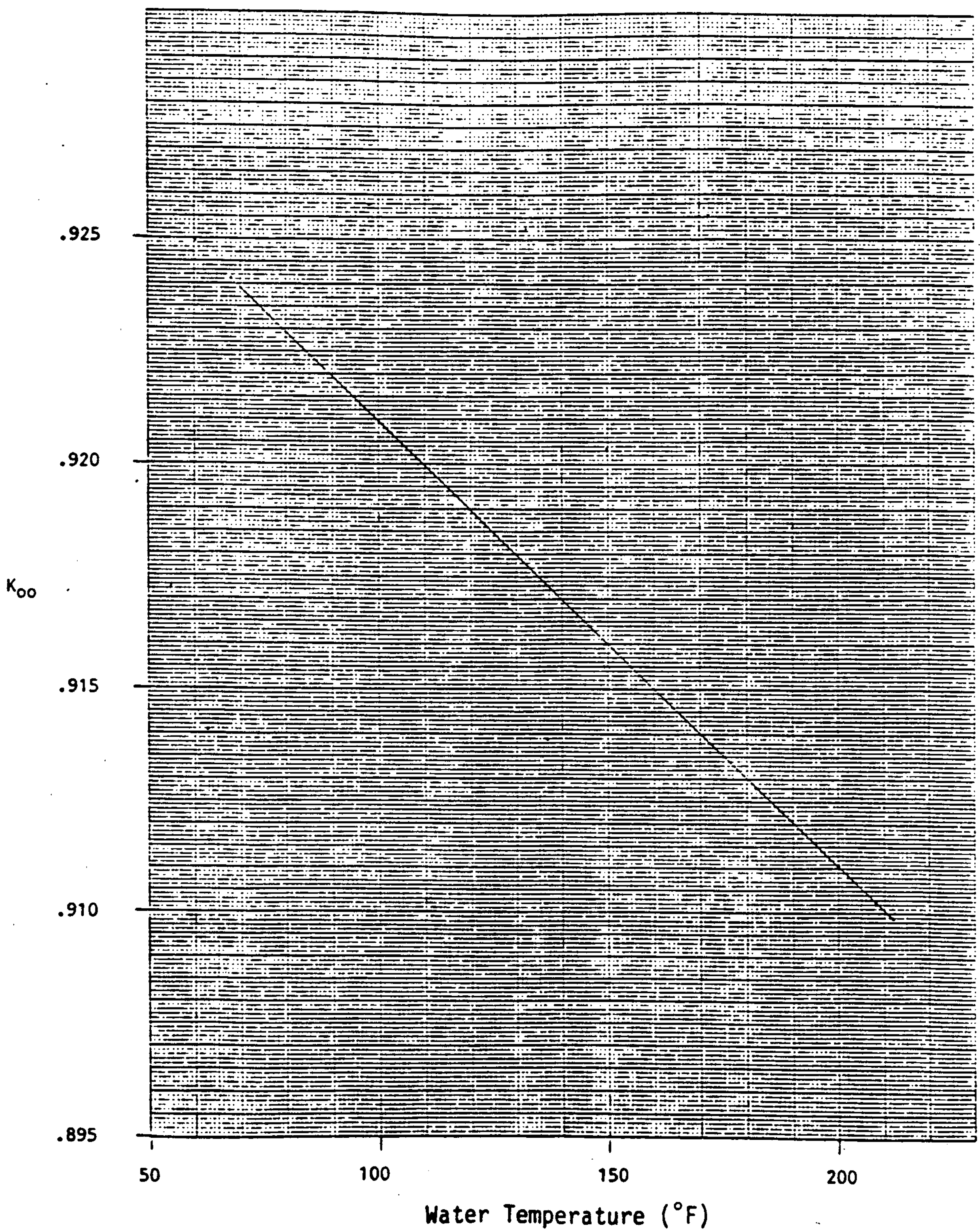


FIGURE 3-19
REGION 2 VARIATION OF k_{∞} WITH POOL TEMPERATURE
(4.5 w/o FUEL AT 36,000 MWD/MTU)

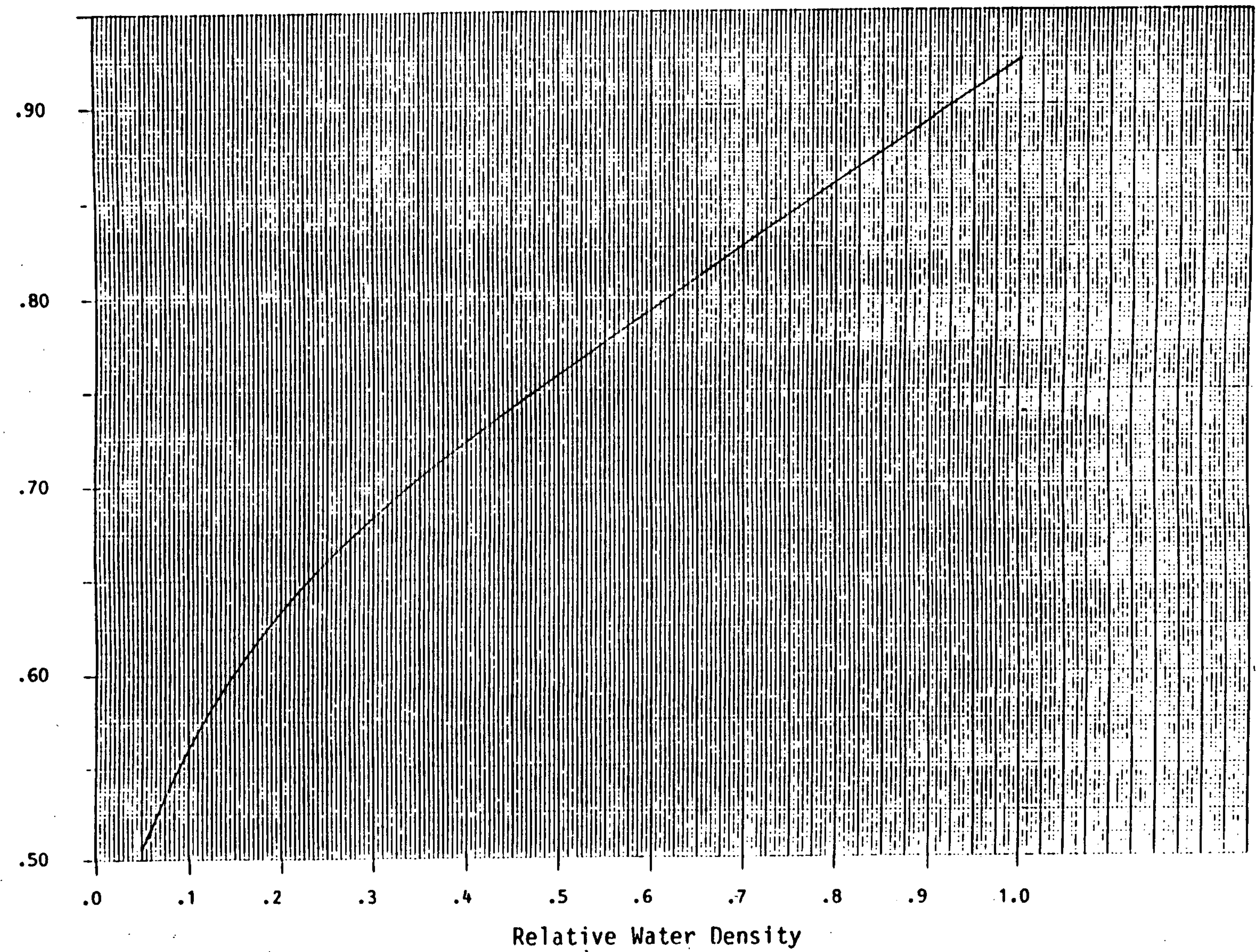


FIGURE 3-20
REGION 2 VARIATION OF k_∞ WITH WATER DENSITY

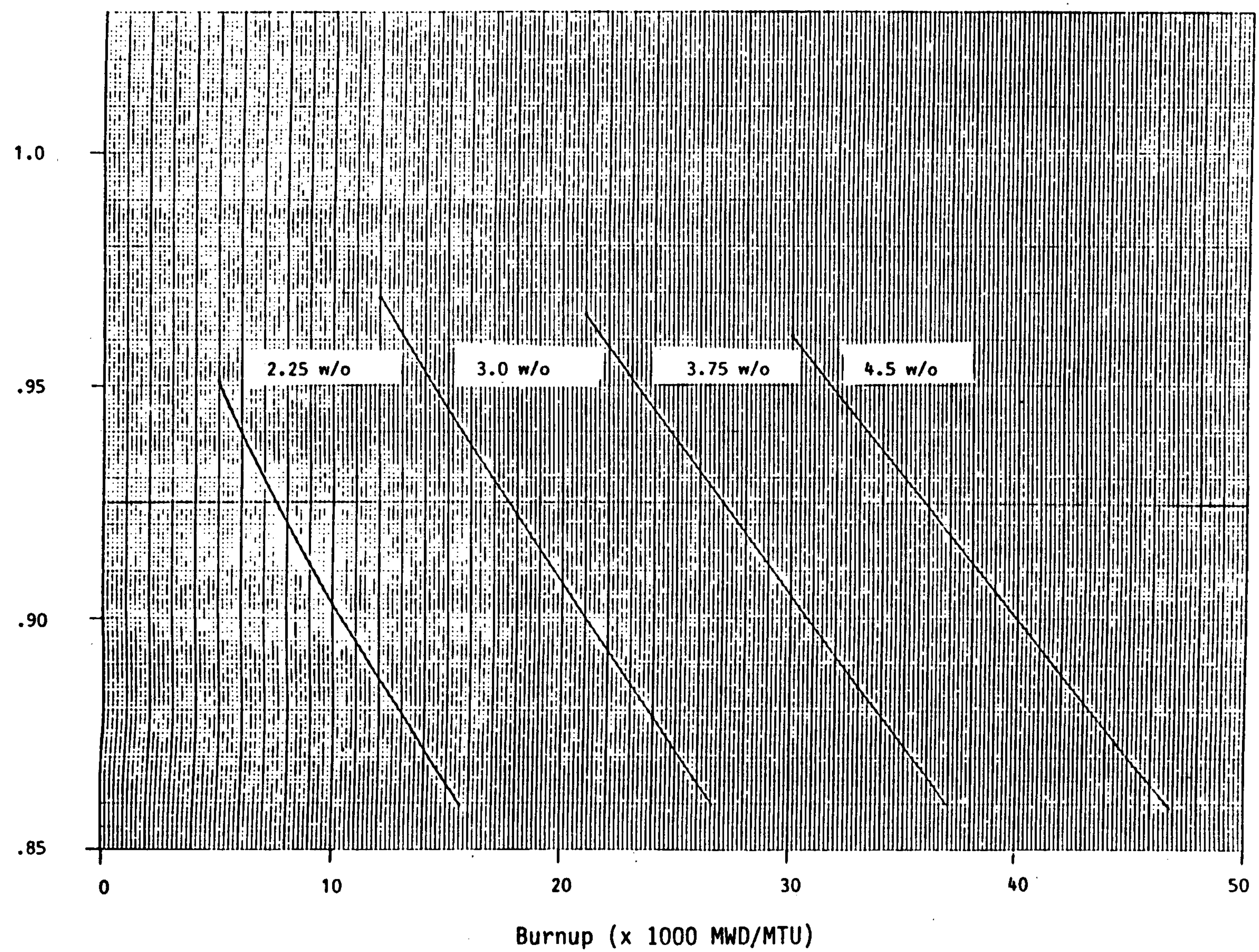


FIGURE 3-21
REGION 2 k_{∞} VERSUS BURNUP AT VARIOUS ENRICHMENTS

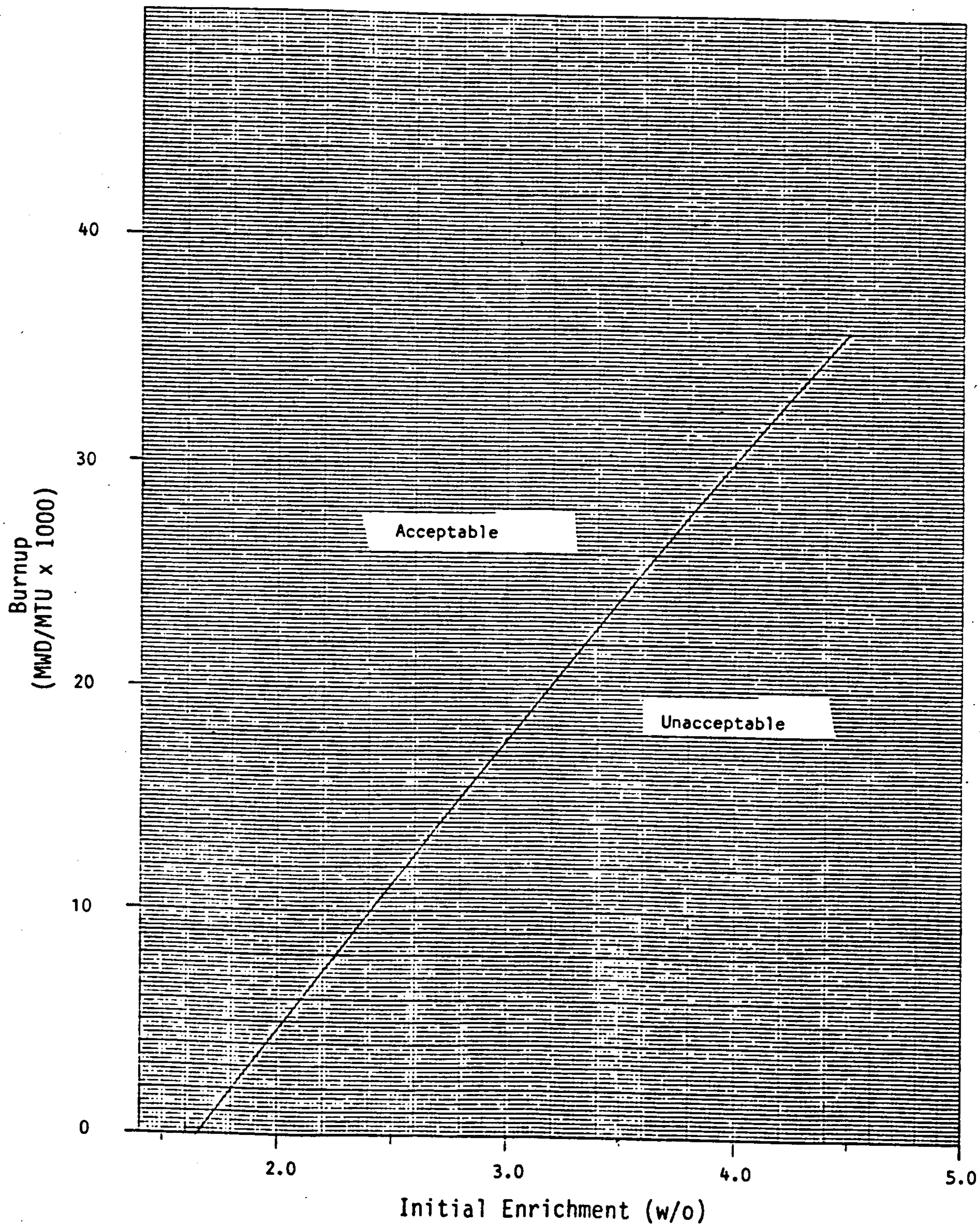


FIGURE 3-22
REGION 2 REQUIRED BURNUP AS A FUNCTION OF INITIAL ENRICHMENT

4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 Description of Fuel Storage Building

The Fuel Storage Building (FSB) consists of cast-in-place reinforced concrete interior and exterior walls. It is completely isolated from all other structures.

The FSB has been designed as a seismic Class I structure. The building exterior walls, floors and interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and protect the equipment inside from the effects of adverse environmental conditions including tornado and hurricane winds, temperature, external missiles and flooding.

The spent fuel pool is a steel lined reinforced concrete structure that provides space for storage of spent fuel assemblies and control rods inserted in the fuel assemblies. The pool is located at the north end of the Fuel Storage Building, and adjacent to the east side of the Containment Building. The fuel pool is 33.0 feet wide, 36.0 feet long and 40.42 feet deep. The fuel transfer canal is separated from the pool by a five foot thick wall. The fuel pool and the fuel transfer canal area are surrounded by 6 foot 3 inch thick reinforced concrete walls. The thickness of the reinforced concrete mat is 3 foot 7 inch. In the fuel transfer canal area the thickness of the mat varies from 3 foot 1 inch to 2 foot 10 inch.

4.1.2 Description of Spent Fuel Racks

The function of the spent fuel storage racks is to provide safe storage for spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excessive mechanical or thermal loadings.

A list of design criteria is as follows:

1. The racks are designed in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978 as amended by the NRC letter dated January 18, 1979 (Reference 1), and Appendix D to Standard Review Plan 3.8.4.

2. The racks are designed to meet the design objectives for light water reactor spent fuel storage facilities at nuclear power stations as specified by ANSI N 210. The effective multiplication factor, k_{eff}, is <.95 including all uncertainties and under all credible conditions.

3. The racks are designed for adequate cooling such that boiling will not occur in the fuel assemblies.

4. The racks are designed to Seismic Category 1, and classified as ASME Code Class 3 component support structures.

5. The racks are designed with appropriate neutron absorbing material, Boral, to permit safe storage of fuel with an initial enrichment up to 4.5 w/o U-235.

6. The racks are designed to provide maximum storage capacity within the spent fuel pool at Indian Point 3.

7. The racks are designed to provide smooth continuous lateral guidance along the length of each cell to prevent damage during insertion or removal of fuel assemblies.

8. The racks are designed to be free-standing on the pool floor with no lateral supports to the pool walls. Sliding is minimal and the racks will not impact the walls or floor appurtenances. There is no rack-to-rack impact since the very strong hydrodynamic coupling forces the racks to move together even when a full and empty rack are adjacent to each other.

9. The racks are designed to preclude storage of a fuel assembly in other than design locations within the rack array. Accidental placement of a fuel assembly between the rack array and pool walls is treated as a credible accident in the Criticality Analysis and will not violate the safe critical configuration of the racks.

10. The materials used in the construction of the racks are compatible with the storage pool environment and will not contaminate the fuel assemblies or the pool water.

4.1.2.1 Design of Spent Fuel Storage Racks

The spent fuel storage rack arrangement in the pool is shown in Figure 2-1. Fuel storage is divided into two regions. Region 1 (240 locations) provides for storage of unirradiated fuel with an initial enrichment up to 4.5 w/o U-235 and for partially burned fuel and a full core unload. Region 2 (1105 locations) provides for storage of irradiated fuel that has achieved a specified burnup. Placement of fuel in Region 2 is determined by burnup calculations and is controlled administratively.

The new racks meet the requirements of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and amended January 18, 1979, with the exception that credit is taken for fuel burnup based on the proposed Revision 2 of NRC Regulatory Guide 1.13 (Reference 2).

The rack module data is presented in Table 4-1.

4.1.2.1.1 Rack Design (Region 1)

As shown in Figure 2-2, the Region 1 rack design is a welded honeycomb array of identical square stainless steel boxes spaced in both directions by a narrow stainless steel water box. The long cross-sectional dimension of this narrow rectangular box is the same as the square box. A sheet of Boral poison material is captured between all adjacent walls of the square and rectangular boxes and on the outside box walls of each of the two racks at rack-to-rack interfaces. A double row of mating flat round raised areas are coined into the walls of all the square boxes and into the two cross-sectional long walls of the narrow rectangular boxes. The raised dimension of each of these local coined areas is half the thickness of the Boral poison sheet. Thus the space provided by the mating raised areas on adjacent box walls is the thickness of the poison sheet. With the poison installed, the boxes are welded together by fusing them at these local coined areas, using a proprietary process, which has been used to fasten together at least 5,000 storage cells. The poison sheets are axially centered on the active fuel region. These sheets are approximately 11" shorter than the active fuel, 5 1/2" at each end, to take advantage of the reduced flux at the ends of the active fuel region. The sheets are scalloped along the two long edges to clear the raised areas on the box walls. They are thus contained axially and laterally by these raised areas. Also, each sheet is contained axially at the bottom by a stainless strip, of the same thickness as the poison sheet, which is welded to the wall of one of the two adjacent boxes. On the outside wall of each of the two racks at a rack-to-rack interface a sheet of poison is captured on each box under a thin sheet of stainless. All four edges of this stainless are bent the thickness of the poison sheet and these bent edges are intermittently welded to the box wall. All of these square and rectangular boxes have a welded-in bottom plate. In the square boxes, which are the fuel storage cells, this bottom plate serves to support the fuel assembly. It has a center hole for coolant flow around and through the fuel assembly. The water box has an orifice plate to control the coolant flow.

Each rack is supported on four corner screw adjustable pedestals welded to the bottom of the rack. The pedestal structure is provided with holes and passages for flow to the holes in the cell bottom plates which are covered by the pedestal structure. Pedestal adjustment is accomplished with a tool through the cell over the pedestal centerline. Inverted V-shaped lead-in guides, which span the space between storage cells, are welded to the top edges of the storage cells.

4.1.2.1.2 Rack Design (Region 2)

As shown in Figure 2-3, the Region 2 rack design is a welded honeycomb array of identical square stainless steel boxes. There are no intermediate water boxes in Region 2. A sheet of Boral poison material is captured between all adjacent walls of the square boxes and on the outside wall of one of the two racks at a rack-to-rack interface. At rack-to-rack interfaces between Region 1 and Region 2, a sheet of Boral poison is captured on the outside box walls of each of the two racks. The Region 2 rack construction is the same as Region 1 racks, where all box walls are coined and fusion welded together at the mating local coined areas. A bottom plate with a central hole is welded into each box. Four corner screw adjustable pedestals are welded to the bottom of the rack. There are however, no lead-in-guides in Region 2 because there is not space for them between cells. A portable lead-in funnel is provided to aid in the fuel assembly insertion. The top elevation of the cells in Region 2 is the same as the top elevation of the lead-in guides in Region 1.

4.1.2.2 Fuel Handling

The storage of additional spent fuel assemblies in the spent fuel pool will not affect the analysis and consequences of the design basis fuel handling accident. The spent fuel storage racks are designed to safely withstand the effects of the design basis fuel handling accident. The resulting criticality and radiological consequences of a postulated fuel assembly drop are addressed in Sections 4.6.2 and 5.3.1, respectively.

4.2 APPLICABLE CODES, STANDARDS, AND SPECIFICATIONS

The design and fabrication of the spent fuel racks and the analysis of the spent fuel pool have been performed in accordance with the applicable portions of the following NRC Regulatory Guides, Standard Review Plan Sections, and published standards (deviations from the guidance provided by these documents are noted in the appropriate sections of this Safety Analysis Report):

a. <u>CFR - Code of Federal Regulations</u>

10CFR21	-	Reporting Safety Related Defects and Noncompliance
10CFR50 Appendix A	-	General Design Criteria
10CFR50 Appendix B	-	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

b. <u>NRC - Nuclear Regulatory Commission</u>

Regulatory Guides

Guidance	Staff Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, sent to Power Reactor Licensees by letter dated April 14, 1978, as amended January 18, 1979
R.G. 1.13	Spent Fuel Storage Facility Design Basis
R.G. 1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors
R.G. 1.28	(ANSI N45.2) Quality Assurance Program Requirements
R.G. 1.29	Seismic Design Classification
R.G. 1.38	(ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants
R.G. 1.44	Control of the Use of Sensitized Stainless Steel
R.G. 1.58	(ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel
R.G. 1.64	(ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants
R.G. 1.74	(ANSI N48.2.10) Quality Assurance Terms and Definitions
R.G. 1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis
R.G. 1.123	(ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

c. <u>Branch Technical Position</u>

CPB 9.1-1	Criticality in Fuel Storage Facilities
ASB 9-2	Residual Decay Energy for Light-Water Reactors for Long-Term Cooling

d. <u>Standard Review Plans</u>

SRP 3.7.1	Seismic Design Parameters
SRP 3.7.2	Seismic System Analysis
SRP 3.7.2	Seismic Subsystem Analysis
SRP 3.8.4	Other Seismic Category I Structures
SRP 9.1.2	Spent Fuel Storage
SRP 9.1.3	Spent Fuel Pool Cooling System

e. <u>ANSI - American National Standards Institute</u>

N16.1	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
N18.2	Pressurized Water Reactor Criteria
N45.2	Quality Assurance Program Requirements for Nuclear Facilities
N45.2.2	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase).
N45.2.6	Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
N45.2.9	Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants
N45.2.10	Quality Assurance Terms and Definitions
N45.2.11	Quality Assurance Requirements for the Design of Nuclear Power Plants
N45.2.13	Quality Assurance Requirements for Control of Procurement of Equipment, Materials and Services for Nuclear Power Plants
N210	Design Objective for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

f. <u>NFPA - National Fire Protection Association</u>

NFPA Handbook

g. ASNT - American Society for Non-destructive Testing

TC-1A Recommended Practice for Non-destructive Testing Personnel Qualification and Certification

h. ASME - American Society of Mechanical Engineers (1980 Code with Addendum)

Boiler and Pressure Vessel Code:

Section III - Nuclear Power Plant Components Subsection NF

Section V - Non-destructive Examination

Section IX - Welding and Brazing Qualifications

i. ASTM - American Society for Testing and Materials

E165 Standard Methods for Liquid Penetrant Inspection

A193 Stainless Steel Bolting

A240 Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels

A262 Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels

A276 Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes

A479 Steel Bars for Boilers and Pressure Vessels

j. ACI - American Concrete Institute

ACI-349-80 Code Requirements for Nuclear Safety Related Concrete Structures and Commentary

ACI-ASME Section III - Code for Concrete Reactor Vessels and Containments, Subsection CC

k. Final Safety Analysis Report (FSAR): Indian Point 3 Nuclear Power Plant.

4.3 SEISMIC AND IMPACT LOADS

The maximum density fuel racks were designed, and the spent fuel pool structure evaluated, using the seismic loading described in this section.

Earthquake loading was predicted based on the safe shutdown earthquake (SSE) at the site having a horizontal ground acceleration of .15g. In addition an operating basis earthquake (OBE), 2/3 of SSE, or .10g, was also analyzed. The maximum vertical acceleration was taken as 2/3 the maximum horizontal acceleration. Damping values of 5% for SSE and OBE were used for the Fuel Storage Building (FSB), consistent with the original plant design.

A conservative damping value of 4% for SSE and OBE (supported by test data from the University of Akron for the Indian Point 3 rack design) was used for seismic analysis of the spent fuel storage racks. Similar tests (documented in "Experimental and Finite Element Evaluation of Spent Fuel Rack Damping and Stiffness," by Scavummo, et al., September 1986) demonstrate that the unique sandwich construction of U.S. Tool & Die racks provide a seismically designed structure with built-in damping to absorb earthquake energy.

The seismic analysis of the spent fuel storage racks was performed to determine the rack behavior and ensure no loss of function resulting from these seismic disturbances. A non-linear finite element computer program was used to analyze the horizontal disturbances, using time-histories synthesized from the floor response spectra. The vertical disturbances were analyzed by the equivalent static method using the peak response spectra.

The seismic analysis determined the rack loads, sliding and lift-off in the three orthogonal directions. The loads were combined using the square root sum of the squares (SRSS) method. Sliding and lift-off results indicate that the racks will not impact the walls.

4.4 LOADS AND LOAD COMBINATIONS

4.4.1 Spent Fuel Pool

4.4.1.1 Loads

The following design loads were considered in the spent fuel pool analysis:

a) Structural Dead Load (D)

   Dead loads consist of the dead weight of the spent fuel racks and their contained fuel and control rods, plus the pool water, concrete, grout, and steel liner structure, and the superstructure walls and miscellaneous building items within the Fuel Storage Building.

b) Live Load (L)

   Live loads are random temporary load conditions for maintenance or special operations which include the spent fuel cask dead weight up to 40 tons.

c) Seismic Loads (E and E')

   Seismic loads include the loads induced by Safe Shutdown Earthquake (E') and Operating Basis Earthquake (E). The hydrodynamic load during the earthquake events was also considered.

d) Normal Operating Thermal Loads (To)

   These thermal loads are generated under normal operating or shutdown conditions.

Normal Operating Condition

- o Pool Water Temperature = 200 °F *
- o Room Temperature = 70 °F
- o Outside Temperature = 0 °F above grade
  50 °F below grade

* Maximum bulk pool temperature following full core unload.

e) Accident (Loss of Fuel Pool Cooling) Thermal Load (Ta)

The thermal accident temperature for the spent fuel pool water is 212°F throughout the pool.

f) Wind Loads (W)

The load generated by the design wind velocity specified for the plant; i.e., 90 mph.

4.4.1.2 Load Combinations

In the spent fuel pool analysis, the following load combinations were considered for the concrete pool structure:

a) Service Load Conditions

1) 1.4 D + 1.7 L
2) 1.4 D + 1.7 L + 1.9 E
3) 1.4 D + 1.7 L + 1.7 W
4) (0.75)(1.4 D + 1.7 L + 1.7 To)
5) (0.75)(1.4 D + 1.7 L + 1.9 E + 1.7 To)
6) (0.75)(1.4 D + 1.7 L + 1.7 W + 1.7 To)
7) 1.2 D + 1.9 E
8) 1.2 D + 1.7 W

b) Factored Load Conditions

9) D + L + To + E'
10) D + L + Ta
11) D + L + Ta + 1.25 E
12) D + L + Ta + E'

For the evaluation of the liner and liner anchors, the following load combinations are applicable:

a) Service Load Conditions

1) D + E + To

b) Factored Load Conditions

2) D + E' + Ta

4.4.2 Spent Fuel Racks

4.4.2.1 Loads

The following loads were considered in the rack design:

Dead Load	(D) =	Dead loads or their related internal moments and forces including any permanent equipment loads.
Live Load	(L) =	Live loads or their related internal moments and forces including any movable equipment loads and other loads which vary with intensity and occurrence.
Fuel Drop Accident Load	(Fd) =	Force caused by the accidental drop of the heaviest load from the maximum possible height.
Crane Uplift Load	(Pf) =	Upward force on the racks caused by a postulated stuck fuel assembly (2000 lbs).
Seismic Loads	(E) =	Loads generated by the Operating Basis Earthquake (OBE).
	(E') =	Loads generated by the Safe Shutdown Earthquake (SSE).
Thermal Loads	(To) =	Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.
	(Ta) =	Thermal effects and loads due to the highest temperature associated with the postulated abnormal design conditions.

4.4.2.2 Load Combinations

The load combinations considered in the analysis of the spent fuel racks are shown below and include those given in the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 1) and Appendix D to Standard Review Plan 3.8.4.

D + L

D + L + Pf

D + L + E

D + L + To

D + L + To + E

D + L + Ta

D + L + To + Pf

D + L + Ta + E'

D + L + Fd

All the rack loads were derived in the seismic analysis due to earthquake motions combined with the rack and fuel weights.

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 <u>Design and Analysis Procedures for Spent Fuel Pool</u>

4.5.1.1 Spent Fuel Pool Structure Finite Element Analysis

In this analysis, the EBS/NASTRAN program, developed by Ebasco and linked to the commercially available NASTRAN program, was used. Various layers of concrete and reinforcing bars were used to determine the effects of concrete cracking. The nonlinear analysis scheme based on the combination of stiffness iteration and load iteration methods, available in the EBS/NASTRAN program, was used to automatically determine the stresses in the concrete and reinforcing bars after the concrete cracks. Since the effect of the fuel rack load on the pool floor is limited to the mat in the pool area, the upper portion of the pool walls is not required for the re-evaluation. Therefore, the finite element model included the lower portion of walls, the pool floor (mat) and the underlying soil rock.

A computer plot of the finite element model is presented in Figure 4-1 which shows the overall view of the model.

4.5.1.2 Liner and Anchorage Analysis

The liner and its anchors were evaluated for the temperature load, the strain induced load due to the deformation of the floor, and the horizontal seismic load. The program POSBUKF developed by Ebasco was used for the liner buckling analysis due to the temperature and strain induced loads. This program is capable of determining the post-buckling stress/strain if the liner plate buckles. The effect of the hydrostatic pressure was considered in this analysis. In calculating the in-plane shear due to the horizontal seismic loads transmitted from the fuel rack to the liner, the maximum assumed friction coefficient of 0.8 was used.

The liner anchors were evaluated for the unbalanced liner in-plane force due to the temperature and strain induced loads, as well as the horizontal seismic in-plane shear force.

4.5.2 Design and Analysis Procedures for Spent Fuel Racks

Seismic analyses were performed using a non-linear finite element computer program for the horizontal earthquake motions and a conservative static analysis for the vertical earthquake motions. A mechanical stress analysis, based on the results from the seismic analysis, was performed to show the adequacy of the rack structure.

The racks were evaluated for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) conditions to meet Seismic Category 1 requirements. A non-linear finite element computer program was used to generate loads, sliding and lift-off for the two horizontal earthquake motions. The non-linear model is shown on Figure 4-2. The equivalent static load method was used for the vertical analysis since the calculated rack vertical natural frequency is less than 33 Hertz. The resultant maximum loads are combined by the square root sum of the squares (SRSS) method.

In the computer program the fuel is considered to rest on the bottom of the storage cell with freedom to move laterally. With a seismic disturbance, the clearance between the fuel and storage cell walls may lead to impacts, thus making the analysis non-linear.

The space between the fuel and the storage cell walls is filled with water so that as the fuel and cell move relative to each other, hydrodynamic forces are set up due to acceleration of the water. These forces are exerted on the fuel and rack structure. There is also movement between the racks and the pool wall where hydrodynamic forces are set up and are exerted on the rack structure and the pool wall.

4.6 STRUCTURAL ACCEPTANCE CRITERIA

4.6.1 Structural Acceptance Criteria for Spent Fuel Pool Structure

4.6.1.1 Criteria

The stresses/strains resulting from the loading combinations described in Section 4.4.1 satisfy the following acceptance criteria:

a) Spent Fuel Pool Concrete Structure

The design stress limits described in ACI 349-80 and 80R were used for the evaluation of the spent fuel pool reinforced concrete structural components. The capacity of all sections was computed based on the Ultimate Strength Design.

b) Liner and Liner Anchors

The acceptance criteria for the liner and liner anchors are in accordance with the requirements specified in Paragraph CC-3720 and CC-3730 of ACI-ASME Section III, Division 2, Subsection CC and can be summarized as follows:

i) Liner

The strain in the liner induced by thermal loads and the deformation of the pool structures is limited to the allowables presented in Table CC-3720-1 of ACI-ASME Section III Code.

ii) Liner Anchors

The displacement of the liner anchors induced by thermal loads and deformation of the pool structures is limited to the allowable presented in Table CC-3730-1 of ACI-ASME Section III Code.

4.6.1.2 Material Properties

The following material properties were used in the analysis of the spent fuel pool structure:

a) Concrete - (f'c = 3,000 psi)
Young's Modulus = 3.12×10^6 psi
Poison's Ratio = 0.156
Thermal Expansion Coeff = 5.5×10^{-6} $1/^{o}F$

b) Rebar Steel -
Young's Modulus = 29×10^6 psi
Poison's Ratio = 0.30
Thermal Expansion Coeff = 6.5×10^{-6} $1/^{o}F$
Yield Strength = 60,000 psi

c) Liner Plate -
Young's Modulus = 28.0×10^6 psi
Poison's Ratio = 0.3
Thermal Expansion Coeff = 9.4×10^{-6} $1/^{o}F$

4.6.1.3 Results

a) Spent Fuel Pool Floor

For the nonlinear analysis of the critical load combination of Section 4.4.1.2, the maximum stress results in the concrete and rebars are summarized in Table 4-2. The results show that the spent fuel pool structure adequately meets all structural acceptance criteria.

b) Liner and Anchorage

The critical loading case producing maximum compressive stress in the liner plate was evaluated. This compressive stress was due to temperature and the deformation of the mat. The buckling analysis result indicated that the liner plate would not buckle, due to the stability effect of the hydrostatic pressure.

Two loading conditions were considered necessary in the liner anchor evaluation; one was the strain-induced load which produced the unbalanced in-plane force at the edge of the pool area, and the other was the horizontal seismic load transmitted through the friction between the rack support and the liner. This horizontal seismic load was assumed to be uniformly distributed at the liner anchors. A maximum friction coefficient of 0.8 was used in calculating this horizontal force.

The results of the liner and liner anchor evaluation indicated minimum safety factors for liner and liner anchor of 4.71 and 3.29, respectively.

4.6.2 Structural Acceptance Criteria for Spent Fuel Racks

4.6.2.1 Criteria

The calculated stresses for the fuel racks are based on load and load combinations in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 1) and Appendix D to Standard Review Plan 3.8.4, which also specifies the acceptance criteria.

The load combinations used are those shown except that temperature effects are not considered. Any thermal load due to the highest postulated pool temperature, or thermal transients, do not affect the results because the allowable stresses for all conditions are low. There are no temperature gradients in the racks greater than a few degrees Fahrenheit which would occur between an empty and occupied storage cell.

4.6.2.2 Results

a) Summary

Table 4-3 shows the maximum stress results compared to the acceptance criteria (allowable stress) and resulting safety factors for the Region 1 rack module and for the largest (i.e., highest load) Region 2 rack module, respectively. The results show that the racks adequately meet all the structural acceptance criteria. Particular accident, sliding and lift-off loads are discussed below.

b) Fuel Handling Crane Uplift Analysis

The rack stress analysis demonstrates that the rack can withstand a maximum uplift load of 2,000 pounds. This load can be applied to a postulated stuck fuel assembly without violating the criticality acceptance criterion. Resulting stresses will be within acceptable stress limits, and there will be no change in rack geometry of a magnitude which causes the criticality acceptance criterion to be violated.

c) Fuel Assembly Drop Accident Analysis

In the unlikely event of dropping a fuel assembly on a rack, the resulting deformation will not alter the criticality safe array of the rack.

Three accident conditions are postulated for the drop analysis in accordance with the requirements of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 1).

The first accident condition is a straight drop on top of the rack. This may result in local rack deformation but does not alter the criticality safe array of the rack.

The second accident condition is an inclined drop on the rack. This will result in a much less severe impact force and rack damage.

The third accident condition assumes a straight drop through an empty cell. This will result in high energy absorption in the cell bottom plate welds possibly leading to bottom plate weld failure, but no change in rack geometry of a magnitude which causes criticality acceptance criterion to be violated.

d) Fuel Rack Sliding and Overturning Analysis

Non-linear seismic analysis shows that sliding of the free-standing racks is minimal, using a low friction coefficient of 0.2. In accordance with Regulatory Guide 3.7.3 Section II.2.b, it is assumed that five OBE and one SSE seismic events can occur. The total sliding is assumed to be additive in the same direction. This results in a small decrease in the rack to wall gaps around the periphery of the rack array in the pool. Therefore there will be no rack to pool wall impacts.

The non-linear seismic analysis also shows rack lift-off using a high friction coefficient of 0.8 is a minimal momentary condition and will not cause overturning.

The analysis indicates that, with virtually no rack to rack gap as installed in the pool, the racks will vibrate in phase under various loading conditions of full, partially filled and empty. This is due to the very strong hydrodynamic coupling between racks. Analysis shows that rack to rack impacting will not occur through the full range of realistically expected gaps between installed racks.

4.7 MATERIALS, QUALITY CONTROL, AND SPECIAL CONSTRUCTION TECHNIQUES

4.7.1 Construction Materials

Stainless steel construction material for the racks is Type 304, ASTM A-240. This material is compatible with the storage pool environment and will not contaminate the fuel assemblies or the pool water.

4.7.2 Neutron Absorbing Material

The neutron absorbing material, Boral, is manufactured at the AAR Brooks & Perkins facilities in Livonia, Michigan. Boral is manufactured under the control and surveillance of a computer aided Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B. Boral material is composed of boron carbide and the 1100 alloy aluminum. The material used in the Indian Point 3 racks contains a minimum B-10 areal density of 0.020gm/cm^2.

Boral has been subjected to accelerated irradiation tests. Test specimens have been exposed to cumulative doses of 3 x 10^{11} rads gamma and 16 x 10^{19} neutrons per sq. cm. in demineralized water and borated water. Tests were performed at the Phoenix Memorial Laboratory of the University of Michigan using the Ford Nuclear Reactor. The neutron absorption properties of Boral was unaffected after the above exposures.

During irradiation, some gas may be generated. Water in contact with aluminum will release hydrogen chemically until the aluminum surface is passivated, and water will disassociate through hydrolysis from gamma radiation. Any gas generated is free to escape since all the Boral poison material is vented in the rack.

The tests on Boral verify that it will maintain long-term material stability and mechanical integrity and that it can be safely utilized as the poison material for neutron absorption in the spent fuel racks.

4.7.3 Quality Assurance

The design, procurement, fabrication and delivery of the new high density spent fuel racks comply with the pertinent Quality Assurance requirements of Appendix B to 10CFR50 and the U.S. Tool & Die, Inc. Quality Assurance Program.

Project auditing, source surveillance, plant surveillance, plant QC support, plant fuel and rack movement and plant health physics support shall conform to New York Power Authority Quality Assurance Program.

4.7.4 Construction Techniques

4.7.4.1 Administrative Controls During Manufacturing

The Indian Point 3 spent fuel racks will be manufactured at the U.S. Tool & Die, Inc., facilities, in Allison Park, Pennsylvania. These modern, high quality facilities have extensive stainless steel experience in forming, welding, machining and assembling nuclear-grade equipment. Forming and welding equipment are specifically designed for fuel rack fabrication. All welders are qualified in accordance with ASME Code Section IX.

Throughout the fabrication process, from procurement to delivery, all work is in accordance with approved drawings and procedures and is controlled throughout by the U.S. Tool & Die, Inc. Quality Assurance Program. Project auditing and source surveillance of the fabrication process is conducted in accordance with the New York Power Authority Quality Assurance Program.

4.7.4.2 New Rack Installation and Old Rack Removal

To avoid damage to the stored spent fuel during rack replacement, all work on the racks in the spent fuel pool area will be performed using written and approved procedures. These procedures will preclude the movement of the fuel racks over the stored spent fuel assemblies.

Radiation exposures during the removal of the old racks from the pool will be controlled by procedure. Water levels will be maintained to afford adequate shielding from the direct radiation of the spent fuel. Prior to rack replacement, the cleanup system will be operated to reduce the activity of the pool water to as low a level as can be practically achieved.

The new maximum density rack modules are designed to be free-standing, i.e., any single rack or combination of racks installed in the spent fuel pool is capable of withstanding a design basis seismic event without overturning or causing damage to fuel assemblies inserted within them. The existing racks, on the other hand, are not free-standing and are provided with interties that contribute to the necessary support to prevent overturning or damage to fuel during a design basis seismic event. Previous analysis of the existing racks shows that four (4) rack modules in a square configuration, connected with interties, is sufficiently stable to be designated as free-standing.

The rack removal/installation sequence will be designed with the aforementioned restrictions in mind. Specifically, no existing interties will be removed until the rack to which they are attached is ready to be removed. Furthermore, the four-module configuration of the existing racks will be maintained as much as possible during the removal sequence. If, at any time during reracking, there exists a rack configuration less stable than the four-module configuration, it will be analyzed and any additional restraints required for assurance of safety during a design basis seismic event will be provided accordingly.

4.8 TESTING AND IN-SERVICE SURVEILLANCE

The rack design includes poison verification view-holes in the cell walls so that the presence of poison material may be visually confirmed at any time over the life of the racks. Upon completion of rack fabrication, such an inspection is performed. This visual inspection, coupled with the U.S. Tool & Die, Inc. Quality Assurance Program controls and the use of qualified Boral neutron absorbing material, satisfies an initial verification test to assure that the proper quantity and placement of material is achieved during fabrication of the racks.

A poison surveillance program to verify the Boral poison material long-term stability and mechanical integrity is provided in compliance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 1).

Poison coupons used in the surveillance program will be taken from the production lot poison. Each poison specimen will be encased in a type 304 stainless steel jacket. The jacket will be mechanically closed without welding in such a manner as to retain its form yet allow rapid and easy opening without contributing mechanical damage to the poison specimen contained within.

Two jacketed full-length poison sheets and two jacketed strings of shorter length specimens will be furnished and installed in Region 1 where exposure to gamma radiation can be manipulated. Each will be suspended at the proper axial location, from a removable lead-in guide in a water box. Appropriate tools will be furnished to remove and re-install them.

The full-length specimens will be examined periodically and returned to the pool. The short-length specimens will be subject to removal, one or two at a time, and examined for physical properties and neutron transmissability. An appropriate number of control specimens, which are not to be irradiated, will be furnished.

Initial surveillance will be performed after a pre-determined interval of exposure in the pool environment which depends on the placement of irradiated fuel assemblies alongside the specimens. This initial surveillance will be implemented after an exposure interval of five years or less. Based on the results of this initial surveillance, determination will then be made for the future scheduling.

4.9 REFERENCES

1. Nuclear Regulatory Commission, Letter to all Power Reactor Licensees, from B.K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC letter dated January 18, 1979.

2. Nuclear Regulatory Commission, "Spent Fuel Storage Facility Design Basis," Proposed Revision 2 to Regulatory Guide 1.13, December 1981.

TABLE 4-1

SPENT FUEL RACK MODULE DATA

	Region 1	Region 2
Number of Storage Locations	240	1105
Number of Rack Arrays	3 (8x10)	3 (11x12) 3 (11x11) 1 (11x10)-(6)
Center-to-Center Spacing (Inches)	10.76	9.075
Cell I.D. (Inches)	8.83	8.83
Type of Fuel	(W) 15x15 Optimized	(W) 15x15 Optimized
Rack Assembly Dimensions (Inches) Height 177 1/2 All Racks	(8x10) 84-7/16 x 106-1/16	(11x12) 99-7/8 x 108-7/8 (11x11) 99-7/8 x 99-7/8 (11x10)-(6) 99-7/8 x 90-3/4
Dry Weights (lbs)	(8x10) 27,880	(11x12) 23,870 (11x11) 22,150 (11x10)-(6) 19,000

6471j

TABLE 4-2

MAXIMUM STRESS RESULTS OF POOL STRUCTURE ANALYSIS

Selected Load Combination	Location	Maximum Average Shear (Kips/In Width) Actual	Allowable	Safety Factor	Maximum Bending Moment (In-Kips/In Width) Actual	Allowable	Safety Factor
1.4D + 1.7L	Mat	0.96	3.68	3.83	59	135	2.29
	Ext Wall	1.58	6.75	4.27	131	213	1.63
	Int Wall	1.97	5.25	2.66	73	148	2.03
	Canal Mat	2.19	8.72	3.98			
1.4D + 1.7L + 1.9E	Mat	1.70	3.68	2.16	95	135	1.42
	Ext Wall	2.95	6.75	2.29	166	303	1.83
	Int Wall	3.02	5.25	1.74	82	148	1.80
	Canal Mat	3.60	8.72	2.42			
0.75 (1.4D + 1.7L + 1.9E + 1.7 To)	Mat	2.60	3.68	1.38	352	423	1.20
	Ext Wall	4.87	6.75	1.39	368	513	1.39
	Int Wall	7.55	8.24	1.09	911	1017	1.12
	Canal Mat	6.71	8.72	1.30			
D + L + Ta + E'	Mat	3.40	3.68	1.08	295	430	1.46
	Ext Wall	4.35	6.75	1.55	255	553	2.17
	Int Wall	7.34	8.24	1.12	466	832	1.79
	Canal Mat	7.40	8.72	1.18			

TABLE 4-3

MAXIMUM STRESS RESULTS OF RACK STRUCTURAL ANALYSIS

Location	Cond.	Type	Region 1 Actual (KSI)	Region 1 Allowable (KSI)	Region 1 Safety Factor	Region 2 Actual (KSI)	Region 2 Allowable (KSI)	Region 2 Safety Factor
Maximum Cell to Cell Fusion Weld Stress:	OBE	Shear	10.0	21.00	2.10	9.3	21.00	2.25
		Bend	13.6	18.00	1.32	12.2	18.00	1.47
	SSE	Shear	13.3	29.06	2.19	11.66	29.06	2.49
		Bend	18.0	36.00	2.00	15.47	36.00	2.33
Cell Bottom Plate to Box Wall Weld Stress:	OBE	Shear	13.0	21.00	1.61	13.54	21.00	1.55
	SSE	Shear	16.5	29.06	1.76	20.30	29.06	1.43
Top Pedestal Plate to Cell Bottom Plate Weld:	OBE	Shear	12.7	21.00	1.65	3.12	21.00	6.7
	SSE	Shear	19.0	29.06	1.52	4.69	29.06	6.2
Pedestal Thread Stress Internal:	OBE	Shear	6.43	8.58	1.33	5.55	8.58	1.55
	SSE	Shear	8.15	10.73	1.31	6.63	10.73	1.61
External:	OBE	Shear	7.66	8.58	1.12	6.55	8.58	1.31
	SSE	Shear	9.71	10.73	1.10	7.83	10.73	1.37

SECTION 4 FIGURES

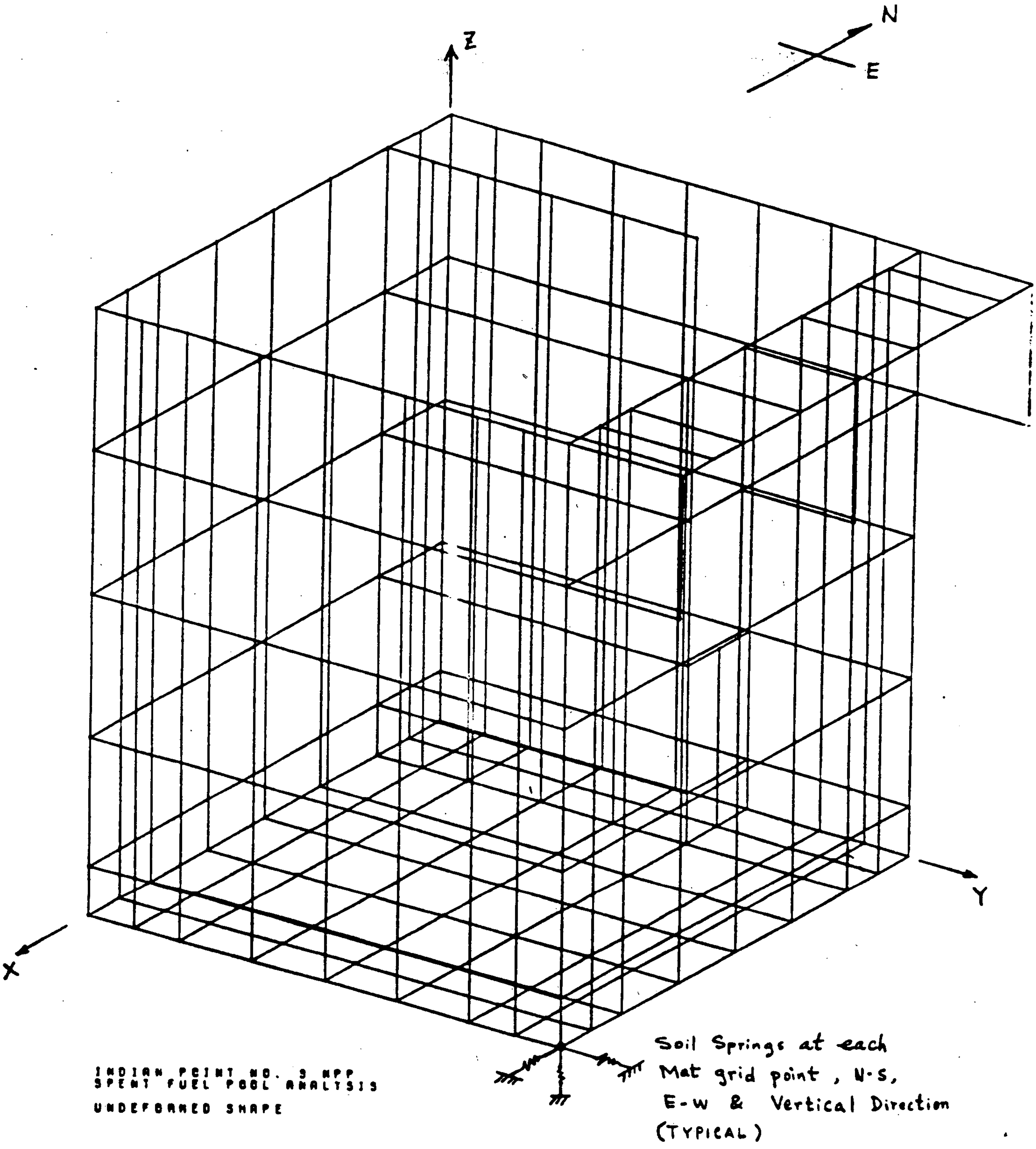


FIGURE 4-1
OVERALL FINITE ELEMENT MODEL FOR SPENT FUEL POOL

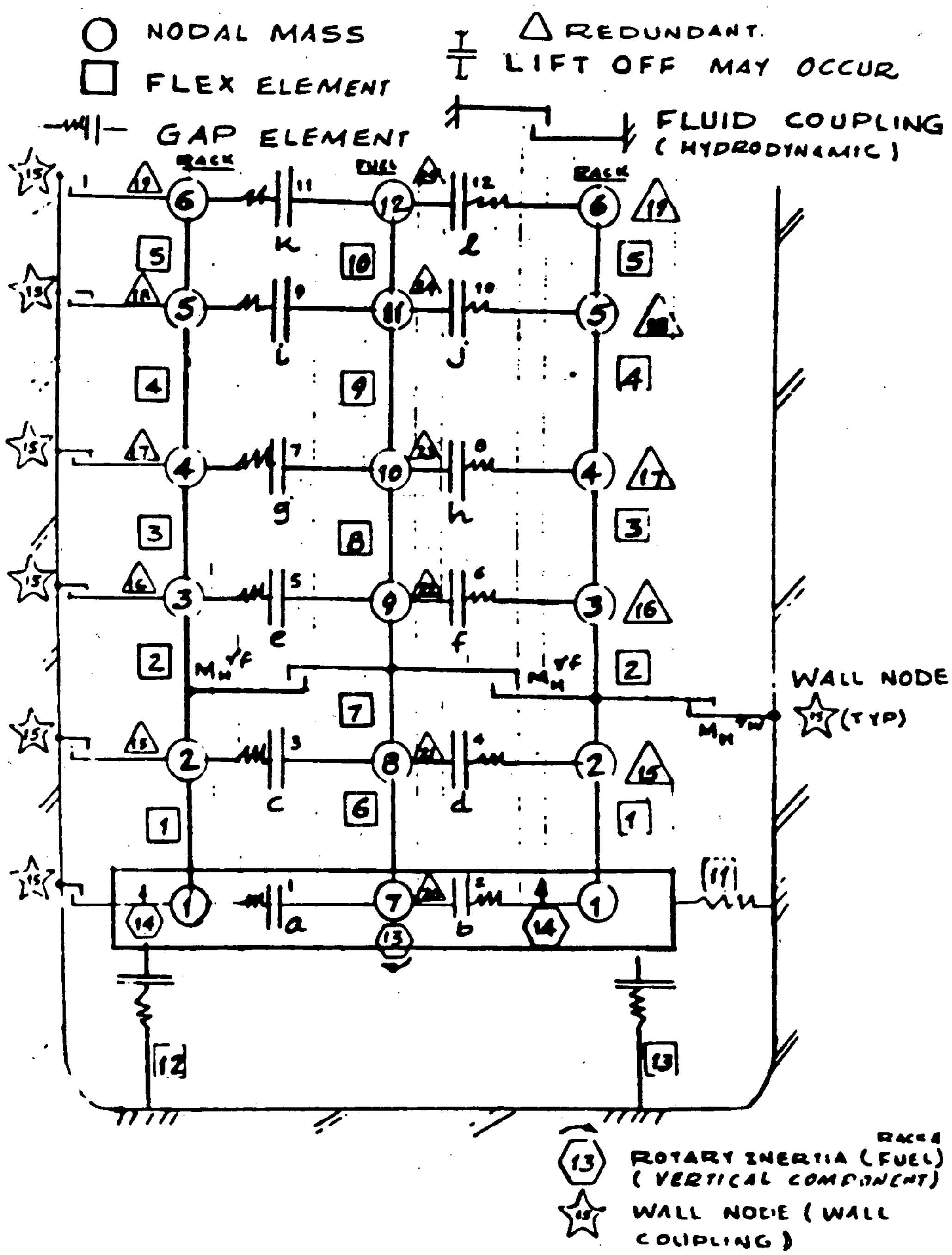


FIGURE 4-2

NON-LINEAR RACK/FUEL MODEL IN SPENT FUEL POOL

5.0 COST/BENEFIT AND ENVIRONMENTAL ASSESSMENT

5.1 COST/BENEFIT AND THERMAL ASSESSMENT

The cost/benefit of the reracking modification is demonstrated in the following sections.

5.1.1 Need for Increased Storage Capacity

a. The Authority currently has no contractual arrangements with any fuel reprocessing facility. There are no operating or planned fuel reprocessing facilities available in the U.S.

   The Authority has executed contracts with the U.S. Department of Energy (DOE) pursuant to the Nuclear Waste Policy Act of 1982. However, the disposal facilities are not expected to be available for spent fuel any earlier than 1998, if a monitored retrievable storage (MRS) facility is constructed, or 2003 for construction of a permanent repository (Reference 1).

b. Table 5-1 includes a projected refueling schedule for Indian Point 3 and the expected number of fuel assemblies that will be transferred into spent fuel pool at each refueling until the ability to maintain a full core reserve is lost in 1994. At present the licensed capacity is 840 storage cells. All calculations in the table for loss of full core reserve (FCR) are based on the number of licensed total cells in the pool. The table is then continued assuming the installation of 1345 replacement cells which lengthens the time of loss of FCR to the year 2005.

c. The Indian Point 3 spent fuel pool is expected to contain 368 to 444 fuel assemblies at the time of reracking. It is best to minimize the inventory of spent fuel in the pool at the time of reracking in order to minimize fuel handling and radiation exposure.

d. Adoption of this proposed spent fuel storage expansion would not necessarily extend the time period that spent fuel assemblies would be stored on site. Spent fuel will be removed from the site for disposal under the provisions of the Nuclear Waste Policy Act of 1982, but a government facility is not expected to be available to accept full reload quantities of spent fuel from Indian Point 3 before 2003 (Reference 2).

5.1.2 Estimated Costs

Total construction cost associated with the proposed modification is estimated to be approximately four (4) million dollars. This figure includes the cost of designing and fabricating the spent fuel racks; engineering costs; installation and support costs at the site; and removal and offsite disposal of the existing racks.

5.1.3 Consideration of Alternatives

a. There are no operational commercial reprocessing facilities available for Authority's needs in the United States, nor are there expected to be any in the foreseeable future.

b. While plans are being formulated by DOE for construction of spent fuel disposal facilities per the Nuclear Waste Policy Act of 1982, a facility is not expected to be available to accept spent fuel any earlier than the 1998 to 2003 time frame (Reference 1).

c. The Authority does not own or control any facility where it could transfer spent fuel from Indian Point 3. The James A. FitzPatrick nuclear plant, owned by the Authority, is a Boiling Water Reactor (BWR) with BWR spent fuel racks that could not accept Pressurized Water Reactor (PWR) fuel from Indian Point 3.

d. There are no existing available independent spent fuel storage facilities. Transfer of Indian Point 3 spent fuel to other utility facilities would only compound storage problems there and is not a viable option.

e. Licensed at-reactor spent fuel storage alternatives involving dry cask/vault storage were evaluated and excluded from consideration at this time due to technical and overall economic reasons. The existing crane capacity plus the limited land space available at the Indian Point 3 site were key considerations in favor of expanding at-reactor storage through reracking over the alternatives of dry cask/vault storage.

f. Estimates for costs of replacement power were calculated in Table 5-2 based on the New York Public Service Commission's avoided capacity and energy costs as per cases no. 28962, 28793 and 28689 dated January 14, 1987. Annual and cumulative replacement power costs are given starting in 1999, the first year spent fuel in the reactor could not be removed due to lack of storage capacity in the existing racks, through the year 2003. This scenario anticipates that the U.S. Department of Energy will be removing fuel from Indian Point 3 at a rate equal to the generation rate by the year 2003.

   Indian Point 3 power is now used by the transportation agencies of the New York metropolitan area and many other public institutions such as schools and hospitals. Plant shutdown would place a heavy financial burden on New York residents served by the Authority and cannot be justified.

5.1.4 Resources Committed

Reracking of the spent fuel pools will not result in any irreversible and irretrievable commitments of water, land, and air resources. The land area now used for the spent fuel pool will be used more efficiently by safely increasing the density of fuel storage.

The materials used for new rack fabrication are discussed in Sections 4.7.1 and 4.7.2. These materials are not expected to significantly foreclose alternatives available with respect to any other licensing actions designed to improve the capacity for storage of spent fuel.

5.1.5 Thermal Impact on the Environment

Section 3.2 considered the following: the additional heat load and the anticipated maximum temperature of water in the spent fuel pool that would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems, and whether there will be any significant increase in the amount of heat released to the environment. As discussed in Section 3.2, the proposed increase in storage capacity will result in an insignificant impact on the environment.

5.2 RADIOLOGICAL EVALUATION

5.2.1 Solid Radwaste

Currently, resins are generated by the spent fuel pool purification system. Current frequency of resin change out is approximately once every two years. No significant increase in volume of solid radioactive wastes is expected due to the new racks based on operating plant experience with high density fuel storage. It is estimated that a minimal amount of additional resins will be generated by the spent fuel pool cleanup system during reracking.

5.2.2 Gaseous Releases

Gaseous releases from the Fuel Storage Building (FSB) are combined with other plant ventilation systems prior to sampling. The plant gaseous releases are reported semi-annually per NRC Regulatory Guide 1.21. The gaseous releases from the FSB comprise less than one percent of the total radioactivity released through the plant vent. No significant increases are expected as a result of the reracking.

5.2.3 Personnel Exposure

a. The range of values for recent gamma isotopic analyses of spent fuel pool water is shown on Table 5-3.

b. Operating experience shows dose rates of less than 2.5 mrem/hour either at the edge or above the center of the spent fuel pool regardless of the quantity of fuel stored. This is not expected to change with the proposed reracking because radiation levels above the pool are due primarily to radioactivity in the water, which experience shows to return to a level of equilibrium. Stored spent fuel is so well shielded by the water above the fuel that dose rates at the top of the pool from this source are negligible.

c. There have been negligible concentrations of airborne radioactivity from the spent fuel pool. Operating plant experience with high density fuel storage has shown no noticeable increases in airborne radioactivity above the spent fuel pool or at the site boundary. Recent air samples taken above the spent fuel pool have shown less than detectable levels of airborne radioactivity. No significant increases are expected from the more dense storage of spent fuel.

d. As stated in Section 5.2.1, reracking and utilization of the new racks will result in no significant increase in the radwaste generated by the spent fuel pool cleanup system. This is because operating experience has shown that with high density storage racks, there is no significant increase in the radioactivity levels in the spent fuel pool water, and no significant increase in the annual person-rem due to the increased fuel storage, including the changing of spent fuel pool cooling system resins and filters.

e. A small amount of primary coolant corrosion product (crud) deposited on the fuel assembly surface may spall off during emplacement in the spent fuel pool from the reactor. Once fuel is placed into a pool storage position, additional crud spalling is minimal.

The highest possible water level is maintained in the spent fuel pool to keep exposure as low as reasonably achievable. Should crud building ever be detected on the spent fuel pool walls around the pool edge, it could easily be washed down.

f. There is no access underneath the spent fuel pool. During normal operation, the radiation dose rate around the outside of the pool could increase locally up to 0.6 mrem per hour should freshly discharged fuel be located in the cells adjacent to the pool liner. This dose rate decreases rapidly with time, and is acceptable. The depth of the water above the fuel is sufficient so there will be no measurable increase in dose rates above the pool due to radiation emitted directly from the fuel.

Operating experience has shown a negligible increase in person-rem due to the increased fuel storage with high density racks. Therefore, a negligible increase in the annual person-rem is expected at Indian Point 3 as a result of the increased storage capacity of the spent fuel pool with the higher density storage racks.

The existing Indian Point 3 health physics program did not have to be modified as a result of the previous increase in storage of spent fuel. It is not anticipated that the health physics program will need to be modified for this increase in storage capability.

5.2.4 <u>Radiation Protection During Rerack Activities</u>

5.2.4.1 General Description of Protective Measures

The radiation protection aspects of the spent fuel pool modification are the responsibility of IP-3 Radiological and Environmental Services with the support of corporate staff. Gamma radiation levels in the pool area are constantly monitored by the station Area Radiation Monitoring System, which has a high level alarm feature. Additionally, periodic radiation and contamination surveys are conducted in work areas as necessary. Where there is a potential for significant airborne radionuclide concentrations, continuous air samplers can be used in addition to periodic grab sampling. Personnel working in radiologically controlled areas will wear protective clothing, and when required by work area conditions, respiratory protective equipment, as required by the applicable Radiation Work Permit (RWP). Personnel monitoring equipment is assigned to and worn by all personnel in the work area. At a minimum, this equipment consists of a thermoluminescent dosimeter (TLD) and self-reading pocket dosimeter. Additional personnel monitoring equipment, such as extremity badges, are utilized as required.

Contamination control measures are used to protect persons from internal exposures to radioactive material and to prevent the spread of contamination. Work, personnel traffic, and the movement of material and equipment in and out of the area are controlled so as to minimize contamination problems. Material and equipment will be monitored and appropriately decontaminated and/or wrapped prior to removal from the spent fuel pool area. The plant radiation protection staff will closely monitor and control all aspects of the work so that personnel exposures, both internal and external, are maintained as low as reasonably achievable (ALARA).

Water levels in the spent fuel pool will be maintained to provide adequate shielding from the direct radiation of the spent fuel. Prior to rack replacement, the spent fuel pool cleanup system will be operated to reduce the activity of the pool water to as low a level as can be practically achieved.

5.2.4.2 Anticipated Exposures During Reracking

Total occupational exposure for the reracking operation is conservatively estimated to be between 3 and 9 person-rem. These estimates are based on an assumed three month installation period using an average of five persons per shift and two shifts per day.

5.2.5 Rack Disposal

The spent fuel storage rack modules that will be removed from the spent fuel pool weigh up to 34,000 pounds each. The total weight of these racks is approximately 176 tons. They will be cleaned of loose contamination, packaged and shipped to a licensed radioactive waste processing facility.

Shipping containers will meet the requirements of DOT regulations pertaining to radioactive waste shipments, including limitations with respect to the waste surface dose and radionuclide activity distribution. Shipping containers will be certified to meet all requirements for a strong tight package. The maximum weight of a loaded shipping container will be in accordance with the American Association of State Highway and Transportation Officials (AASHTO). Trucks and drivers used for rack and waste transportation will have all permits and qualifications required by the Federal DOT and the DOT for each State through which the truck will pass.

At the waste processing facility, the racks will be decontaminated to the maximum extent possible. Remaining portions of the racks and contaminated waste generated from decontamination will be buried at a licensed radioactive waste burial site. In preparing non-decontaminatable waste for shipment and subsequent burial, volume reduction methodologies will be employed such as compaction, combining metallic materials with "soft waste" to minimize void space, and super compaction where feasible.

5.3 ACCIDENT EVALUATION

5.3.1 <u>Spent Fuel Handling Accidents</u>

5.3.1.1 Fuel Assembly Drop Analysis

For a fuel assembly drop on top of the rack, maximum expected deformation will be locally limited to less than the top six inches of the rack walls and will not reduce minimum spacing between the stored fuel assemblies. Consequently, fuel assembly drop accidents will not result in a significant increase in reactivity. Furthermore, soluble boron in the pool water would substantially reduce the reactivity and assure that the true reactivity is always less than the limiting value for any conceivable dropped fuel accident.

Radiological consequences of a worst case fuel assembly drop in the spent fuel pool will be bounded by the radiological consequences of fuel handling accidents previously evaluated in the FSAR (Reference 3), Section 14.2.1, with potential resulting thyroid and whole body doses at the site boundary well within 10 CPR Part 100 limits and NRC Standard Review Plan 15.7.5 acceptance criteria.

5.3.1.2 Cask Drop Analysis

Current Technical Specifications for Indian Point 3 (Reference 4) require that a spent fuel cask shall not be moved over any region of the spent fuel pool which contains irradiated fuel. This restriction effectively precludes a spent fuel cask being handled over the spent fuel pool and, consequently, a cask drop analysis is not necessary.

5.3.1.3 Rack Drop Accident Analysis

Sections 5.1.1, 5.1.2 and 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", provide guidance for heavy load handling operations.

Section 5.1.2 provides four alternatives for assuring the safe handling of heavy loads during a fuel storage rack replacement. Alternative (1) of Section 5.1.2 provides guidelines that the control of heavy loads can be satisfied by establishing that the potential for a heavy load drop is extremely small as demonstrated by meeting the single-failure-proof crane criteria. Alternative (1) is satisfied during the subject application.

NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", provides guidance for design, fabrication, installation and testing of new cranes that are of a high reliability design. For operating plants, NUREG-0612, Appendix C, "Modification of Existing Cranes," provides guidelines on the implementation of NUREG-0554 for operating plants. An evaluation of storage rack movements by the Fuel Storage Building crane for conformance with the NUREG-0612, Appendix C guidelines demonstrated that alternative (1) above is satisfied. The Fuel Storage Building crane has a rated capacity of 40 tons, which incorporates a factor of safety of five. The maximum weight of any existing or replacement storage rack is 17 tons. Therefore, the minimum safety factor is 11.8 for movements of the storage racks by the Fuel Storage Building crane. This applies to non-redundant load-bearing components. Redundant special lifting devices, which have a rated capacity sufficient to maintain the required safety factor, will be utilized in the movements of the storage racks. As per NUREG-0612, Appendix B, this ensures that the probability of a load drop is extremely low.

The existing mechanical stops will be removed so that the Fuel Storage Building crane will have access to any location over the spent fuel pool. However, administrative controls, which incorporate predetermined safe load pathways, will ensure that at no time will any storage rack be moved directly over an irradiated fuel assembly. In addition, no heavy loads will be carried in the spent fuel pool area until all fuel in the pool has been subcritical a minimum of 120 days and has had sufficient time for decay of gaseous radionuclides in the fuel (gap activity) such that accidental release of all these gases would result in potential offsite doses less than 10 percent 10 CFR 100 limits.

5.3.1.4 Abnormal Location of a Fuel Assembly

The abnormal location of a fresh unirradiated fuel assembly or insufficiently depleted fuel assembly could result in a positive reactivity effect. This could occur if the assembly were to be inadvertently loaded into a Region 2 storage cell. Soluble poison, however, is present in the spent fuel pool water (for which credit is permitted under these conditions) and would maintain the reactivity substantially less than the design reactivity limitation (k_{eff} of<0.95).

5.3.2 Conclusions

Since a spent fuel cask will not be handled over or in the vicinity of spent fuel as discussed in Section 5.3.1.2, the proposed modification does not result in a significant increase in the probability of the cask drop accident. Furthermore, by imposing a minimum decay time of 120 days for all fuel in the spent fuel pool prior to heavy load handling associated with the rack replacement operations, potential offsite doses are less than 10 percent of 10 CFR 100 limits should a dropped heavy load caused damage to stored spent fuel. Since there will be a negligible change in radiological conditions due to the increased storage capacity of the spent fuel pool, no change is anticipated in the radiation protection program. In addition, the environmental consequences of a postulated fuel handling accident in the spent fuel pool, described in Updated FSAR Section 14 (Reference 3), remain unchanged. Therefore, there will be no change or impact to any previous determinations of the Final Environmental Statement (Reference 5). Based on the foregoing, the proposed amendments will not significantly affect the quality of the human environment; therefore, under 10 CFR 51, issuance of a negative declaration is appropriate.

5.4 REFERENCES

1. U.S. Department of Energy, "OCRWM Mission Plan Amendment," June 1987.

2. U.S. Department of Energy, "Annual Capacity Report," June 1987

3. Indian Point 3 Nuclear Power Plant, Updated Final Safety Analysis Report, Docket No. 50-286.

4. Indian Point 3 Nuclear Power Plant, Technical Specifications, Facility Operating License DPR-64.

5. Indian Point 3 Nuclear Power Plant, Final Environmental Statement, Docket No. 50-286.

TABLE 5-1

NUCLEAR FUEL DISCHARGE INFORMATION
INDIAN POINT 3

Cycle No	Shutdown Dates	Number Assemblies Discharge	Cumulative Total of Spent Fuel Assemblies in the Pool
01	6/1978	64	64
02	9/1979	76	140
03	3/1982	76	216
04	6/1985	76	292
05	5/1987	76	368
		840 CURRENTLY INSTALLED CELLS	
(ACTUAL CYCLE INFORMATION THROUGH CYCLE FIVE, PROJECTED THEREAFTER)			
06	1/1989	76	444
07	9/1990	76	520
08	5/1992	76	596
09	1/1994	76	672 (1)
10	9/1995	76	748
11	5/1997	76	824
12	1/1999	76	900
13	9/2000	76	976
14	5/2002	76	1052
15	1/2004	76	1128
16	9/2005	76	1204 (2)
17	5/2007	76	1280
18	1/2009 (3)	76	1356
19	9/2010	76	1432
20	5/2012	76	1508
21	1/2014	76	1584
END OF LIFE	9/2015	193 FINAL OFFLOAD	1777

(1) FULL CORE RESERVE (FCR) LOST AT 647 CELLS WITH CURRENT RACKS; RERACK REQUIRED TO REGAIN FCR

(2) FCR LOST AT 1152 CELLS WITH RERACK (1345 AVAILABLE STORAGE LOCATIONS)

(3) CURRENT END OF LIFE = 8/2009, PROJECTED EXTENDED LIFE TO 2015

TABLE 5-2

ANNUAL REPLACEMENT POWER COSTS ATTRIBUTED TO
INDIAN POINT 3

Year	Energy Production (GWH) 1/	Nominal Net Replacement Costs 2/ ($000)	($/MWH)	Cumulative Cost ($000)	Present Value 1987 Dollars ($000) 3/	Cumulative Present Value 1987 Dollars ($000)
1999	6340	725,930	114.50	725,930	288,277	288,277
2000	6340	756,996	119.40	1,482,926	278,346	566,623
2001	6340	788,696	124.40	2,271,622	268,520	835,143
2002	6340	822,298	129.70	3,093,920	259,223	1,094,366
2003	6340	857,168	135.20	3,951,088	250,199	1,344,565

1/ Based on: Plant rating of 965 MW and annual capacity factor of 75%.

2/ Calculated based on statewide avoided capacity and energy costs in the Con Edison franchise area prepared and issued by the New York Public Service Commission on 1/14/87 as per Cases No. 28962, 28793 and 28689. Reflects gross replacements costs (excludes any offset for avoided variable costs such as fuel and operation and maintenance expenses).

3/ Based on a discount rate of 8%.

NOTE: Indian Point 3 assumed to be out of service all year(s).

TABLE 5-3

GAMMA ISOTOPIC ANALYSIS
SPENT FUEL POOL WATER

RADIONUCLIDE	ACTIVITY
Co-57	9.533 E-6 uCi/ml
Co-58	1.072 E-3 uCi/ml
Co-60	2.855 E-3 uCi/ml
Cs-137	2.880 E-5 uCi/ml