FLORIDA POWER & LIGHT COMPANY

TURKEY POINT UNITS 3 AND 4

SPENT FUEL STORAGE FACILITY MODIFICATION

SAFETY ANALYSIS REPORT

DOCKET NOS. 50-250 AND 50-251

Rev. 0

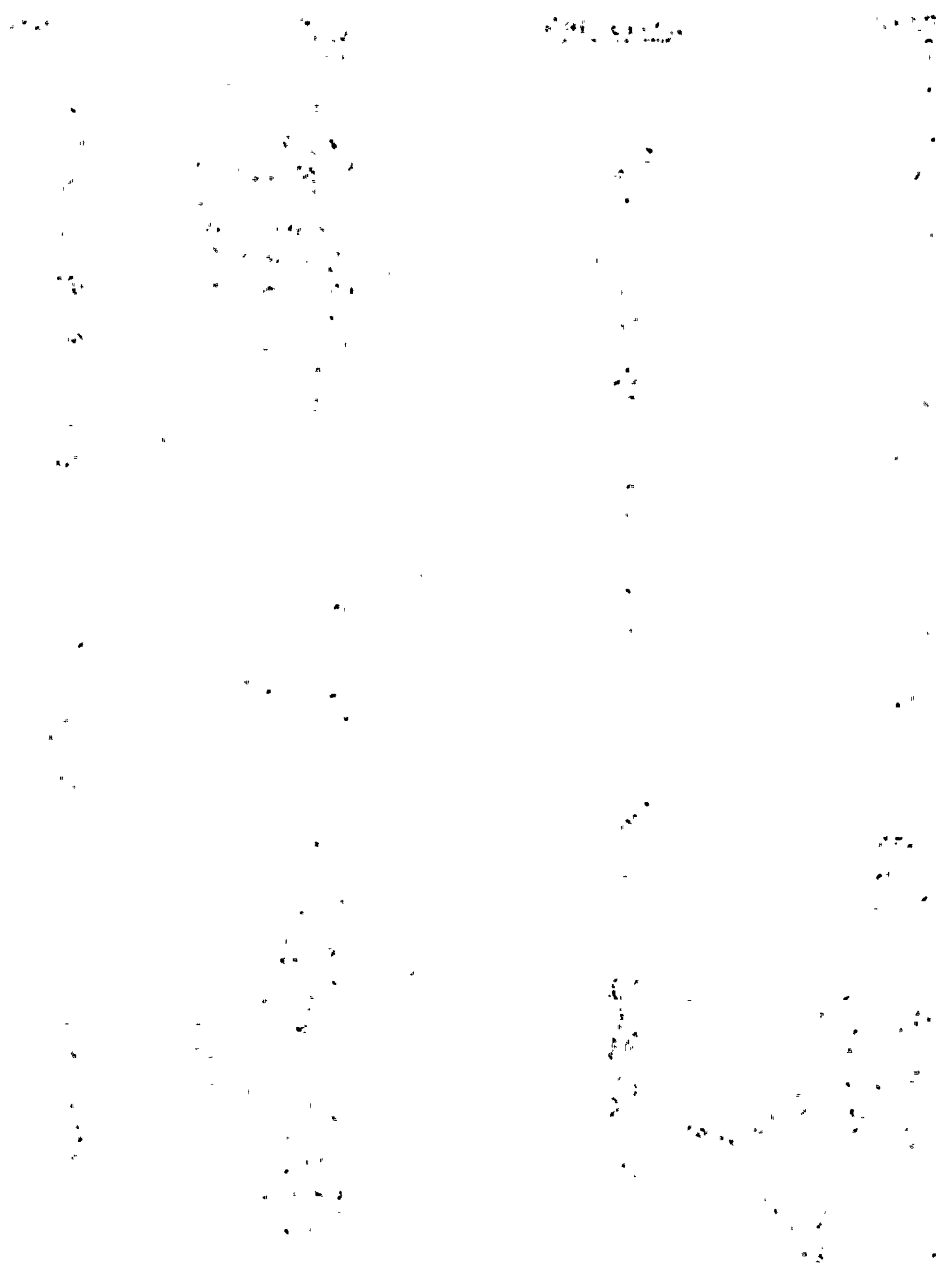
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4

SPENT FUEL STORAGE FACILITY MODIFICATION
SAFETY ANALYSIS REPORT
DOCKET NOS. 50-250 AND 50-251

LIST OF EFFECTIVE PAGES

Page	Revision	Page	Revision
i	0	Table 3-4	0
ii	0	Table 3-5	0
iii	0	Figure 3-1	0
iv	0	Figure 3-2	0
v	0	Figure 3-3	0
vi	0	Figure 3-4	0
vii	0	Figure 3-5	0
viii	0	Figure 3-6	0
ix	0	Figure 3-7	0
1-1	0	Figure 3-8	0
1-2	0	Figure 3-9	0
2-1	0	Figure 3-10	0
3-1	0	Figure 3-11	0
3-2	0	4-1	0
3-3	0	4-2	0
3-4	0	4-3	0
3-5	0	4-4	0
3-6	0	4-5	0
3-7	0	4-6	0
3-8	0	4-7	0
3-9	0	4-8	0
3-10	0	4-9	0
3-11	0	4-10	0
3-12	0	4-11	0
3-13	0	4-12	0
3-14	0	4-13	0
3-15	0	4-14	0
3-16	0	4-15	0
3-17	0	4-16	0
3-18	0	Table 4-1	0
3-19	0	Table 4-2	0
(T.S.3.12-1)	0	Figure 4-1	0
(B3.12-1)	0	Figure 4-2	0
(T.S.3.17-1)	0	Figure 4-3	0
(T.S.3.17-2)	0	Figure 4-4	0
(B3.17-1)	0	Figure 4-5	0
(T.S.Table 4.1-2, Sht. 2)	0	Figure 4-6	0
(T.S.5.4-1)	0	Figure 4-7	0
Table 3-1	0	Figure 4-8	0
Table 3-2	0	Figure 4-9	0
Table 3-3	0		

[illegible]



LIST OF EFFECTIVE PAGES (Continued)

Page	Revision	Page	Revision
Figure 4-10	0	5-9	0
Figure 4-11	0	5-10	0
Figure 4-12	0	Table 5-1	0
5-1	0	Table 5-2	0
5-2	0	Table 5-3	0
5-3	0	Table 5-4	0
5-4	0	Table 5-5	0
5-5	0	Table 5-6	0
5-6	0	Table 5-7	0
5-7	0	Table 5-8	0
5-8	0		

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

1.1 LICENSE AMENDMENT REQUESTED

Florida Power & Light Company (FPL) is currently pursuing the design and manufacture of new spent fuel storage racks to be placed into the spent fuel pools of Turkey Point Units 3 and 4. The purpose of these new racks is to increase the amount of spent fuel that can be stored in the existing spent fuel pools. The racks are designed so that they can store spent fuel assemblies in a high density array. Therefore, FPL hereby requests that License Amendments be issued to the Turkey Point Units 3 and 4 Facility Operating Licenses DPR-31 and DPR-41 [1], respectively, 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 amendments.

1.2 CURRENT STATUS

There are two spent fuel pools at Turkey Point; one for each nuclear unit. The existing racks in each of these pools have 636 total storage cells. Because of the specific piping and lighting interferences in some areas of each pool, there are some unusable cells in each pool. There are 621 cells which are usable to store spent fuel assemblies in Unit 3 and 614 in Unit 4. In the 1986-87 time frame, these units will lose their full-core reserve storage capacity (157 assemblies), and in the 1990-1991 time frame they will no longer have the capacity to store fuel discharged from the operating units. (This takes into account the capability of transferring spent fuel between the spent fuel pools of both units.) Therefore, to ensure that sufficient capacity continues to exist at Turkey Point to store discharged fuel assemblies, FPL plans to replace the existing storage racks with new spent fuel storage racks whose design will allow for more dense storage of spent fuel, thus enabling the existing pools to store more fuel in the same space as occupied by the current racks.

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 [2]. Sections 3.0 through 5.0 of this report are consistent with the section/subsection format and content of the NRC position paper, Sections III through V.

This report contains the nuclear, thermal-hydraulic, mechanical, material, structural, and radiological design criteria to which the new racks are designed. In anticipation of possible future spent fuel storage needs, the new racks are designed and analyzed to envelop the fuel characteristics of

each nuclear plant within the FPL system. However, as further discussed in Section 5.0, there are no plans at the present time for fuel transshipment within the FPL system.

The nuclear and thermal-hydraulic aspects of the 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. Temporary fuel storage considerations during rack removal and installation are also addressed.

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 Turkey Point Units 3 and 4 spent fuel storage facilities will continue to provide safe spent fuel storage, and that the modification is consistent with the facility design and operating criteria as provided in the Turkey Point Updated FSAR [3] and Operating Licenses.

1.5 REFERENCES

1. Turkey Point Plant Units 3 and 4, Facility Operating Licenses DPR-31 and DPR-41, Docket Nos. 50-250 and 50-251.

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. Turkey Point Plant Units 3 and 4, Updated Final Safety Analysis Report, Docket Nos. 50-250 and 50-251.

2.0 SUMMARY OF EXISTING RACK DESIGN

The existing spent fuel storage racks have the capacity to store 621 spent fuel assemblies for Unit 3 and 614 for Unit 4. The racks consist of an array of nominal center-to-center spacing of 13.7 inches; each storage cavity can accommodate one fuel assembly. The fuel assembly storage cavities are structurally connected to form 12 freestanding spent fuel assembly storage modules per spent fuel pit. Each spent fuel module is made up of square storage cells joined together by an egg crate structure. The egg crate structure limits structural deformations to maintain the required minimum storage cell spacing for all design conditions, including the safe shutdown earthquake (SSE). The storage cells have lead-in surfaces at the top to provide guidance for insertion of fuel assemblies. The racks are supported on stainless steel embedments in the bottom slab of the pool structure. Further information on the existing rack design is provided in Appendix 14E of the Turkey Point Updated FSAR.

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, including uncertainties, 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 as recommended in ANSI N210-1976 and in the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" [1].

The spent fuel rack design described herein employs two separate and different arrays which will be considered as two separate spent fuel racks. The Region I design, a standard high density poison rack, is based on maintaining $k_{eff} < 0.95$ for Westinghouse 15x15 OFA fuel (which is the most limiting from a criticality standpoint) at 4.5 w/o U-235. Region II is designed to maintain $k_{eff} < 0.95$ for Westinghouse 15x15 OFA fuel which has:

1. An initial enrichment/burnup combination in the acceptable area of Figure 3-3 with utilization of every cell committed, or,

2. An enrichment of 4.5 w/o U-235 stored in a rack in a checkerboard arrangement with two out of every four cells containing a fuel assembly.

The following are the conditions that are assumed in meeting this design basis.

3.1.1 Normal Storage

a. As described in Section 4.1.2.1, spent fuel storage is divided into two regions. The storage cell nominal geometry is shown on Figure 3-1 for Region I and Figure 3-2 for Region II.

b. For Region I, the fuel assembly contains the highest enrichment anticipated (4.5 w/o) without any control rods or any noncontained burnable poison and is at its most reactive point in life.

   Storage of fuel in Region II assumes burnup of U-235 has occurred. Suitability for storage of irradiated fuel in Region II is determined utilizing a minimum fuel burnup

versus enrichment curve calculated for the rack design. The actual fuel assembly conditions are defined by the zero burnup enrichment (1.5 w/o U-235).

c. The assembly is conservatively modeled with water replacing the assembly grid volume and no U-234 or U-236 in the fuel pellet. No U-235 burnup is assumed.

d The moderator is pure water at the temperature within the design limits of the pool which yields the largest reactivity. A conservative value of 1.0 gm/cm^3 is used for the density of water. No dissolved boron is included in the water.

e. The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design. The nominal case calculation is infinite in lateral and axial extent.

f. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are:

- Poison pocket thickness
- Stainless steel thickness
- Cell ID
- Center-to-center spacing
- Cell bowing

The calculated method uncertainty and bias are discussed in Section 3.1.3.

g. Credit is taken for the neutron absorption in full length structural materials and in solid materials added specifically for neutron absorption. A minimum poison loading is assumed in the poison plates and B_4C particle self-shielding is included as a bias in the reactivity calculation.

Methods for initial and long-term verification of poison material stability and mechanical integrity are discussed in Section 4.8.

3.1.2 Postulated Accidents

The criticality analysis includes postulated accidents so that the double contingency principle of ANSI N16.1-1975 is met and that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 under all conditions.

Most postulated accident conditions will not result in an increase in k_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water

density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than eight inches of water separating it from the active fuel height of stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity. These would include the inadvertant drop of an assembly between the outside periphery of the rack and the pool wall, a cask drop accident, or damage to the fuel racks when empty rack modules are being installed. Therefore, for accident conditions, the double contingency principle of ANSI N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to provide for protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water, as specified in proposed Technical Specification 3.17, is a realistic initial condition.

The presence of approximately 1950 ppm boron in the pool water will decrease reactivity by about 30 percent Δk. In perspective, this is more negative reactivity than is present in the poison plates (25 percent Δk), so k_{eff} for the rack would be less than 0.95 even if the poison plates were not present. Thus, for postulated accidents, should there be a reactivity increase, k_{eff} would still be less than or equal to 0.95 due to the combined effects of the dissolved boron and the poison plates.

The "optimum moderation" accident is not a problem in spent fuel storage racks because the presence of poison plates removes the conditions necessary for "optimum moderation". The k_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 in the poison rack design.

3.1.3 Calculation Methods

3.1.3.1 Criticality Analysis for Region I

The calculation method and cross-section values for Region I are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. These benchmarking data are sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which provides for the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes [2,3] for cross-section generation and KENO IV[4] for reactivity determination.

The 218 energy group cross-section library[2] that is the common-starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-IV data. The NITAWL program[3] includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program[3] which is a one-dimensional S_N transport theory code. These multigroup cross-section sets are then used as input to KENO IV[4] which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 27 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various materials (Boral, steel and water) that simulate LWR fuel shipping and storage conditions [5,6] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials [7] (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method. Table 3-1 summarizes these experiments.

The average k_{eff} of the benchmarks is 0.9998 which demonstrates that there is no bias associated with the method. The standard deviation of the k_{eff} values is 0.0057 Δk. The 95/95 one-sided tolerance limit factor for 27 values is 2.26. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.013 Δk. The total uncertainty to be added to a criticality calculation is:

$$TU = \left[(ks)^2_{\text{method}} + (ks)^2_{\text{nominal}} + (ks)^2_{\text{mech}} \right]^{1/2}$$

where $(ks)_{method}$ is 0.013 as discussed above, $(ks)_{nominal}$ is the statistical uncertainty associated with the particular KENO calculation being used and $(ks)_{mech}$ is the statistical uncertainty associated with mechanical tolerances, such as thicknesses and spacings.

The most important effect on reactivity of the mechanical tolerances is the possible reduction in the water gap between the poison plates. The worst combination of mechanical tolerances is that which results in the maximum reduction in the water gap. For a single cell it is found that reactivity does not increase significantly because the increase in reactivity due to the water gap reduction on one side of the cell is offset by the decrease in reactivity due to the increased water gap on the opposite side of this cell. The analysis for the effect of mechanical tolerances, however, assumed a worst case of a rack composed of an array of groups of four cells

with the minimum water gap between the four cells. The reactivity increase of this configuration is included as a bias term in calculating the k_{eff} of the rack. It is included as a bias term since cells can be welded to a common grid during manufacturing which is the likely cause of the water gap reduction.

An additional reactivity consideration is due to cell bowing. The individual cell bowing tolerance could also result in a reduction of the water gap between poison plates. Again an array of groups of four fuel/pin assemblies is assumed with the minimum water gap between the four cells. The resulting reactivity increase is included as an uncertainty because cell bowing will be random as opposed to the cells welded to a common grid. Also, since this common grid effect is already included in the analysis, it is as equally likely that cell bowing will cause a reactivity decrease as it does an increase.

Some mechanical tolerances are not included in the analysis because worst case assumptions are used in the nominal case analysis. An example of this is eccentric assembly position. Calculations are performed which show that the most reactive condition is the assembly centered in the cell which is assumed in the nominal case. Another example is the reduced width of the poison plates. No bias is included here since the nominal KENO case models the reduced width explicitly.

The final result of the uncertainty analysis is that the criticality design criterion is met when the calculated effective multiplication factor, plus the total uncertainty (TU) and any biases, is less than 0.95.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 5.1.12; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety," NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"; and the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

3.1.3.2 Criticality Analysis for Region II

3.1.3.2.1 Analytical Methods

The methods used in the analysis of Region II include NITAWL, XSDRNPM and KENO-IV for basic reactivity determination, along with PHOENIX[8] and CINDER[9] for reactivity equivalencing. Because of the Boraflex poison present in the spent fuel racks, multi-group transport theory is considered a more qualified and accurate approach to use than two-group diffusion theory for reactivity equivalencing. PHOENIX is used to

calculate the isotopic compositions and cross-sections of the fuel as a function of irradiation history and subsequent decay time. It then determines the reactivity equivalence (in the Region II rack) of assemblies with different initial enrichments and burnups. The reactivity equivalencing is extended back to an unirradiated assembly, which is then analyzed using NITAWL, XSDRNPM and KENO-IV in a fashion similar to the analysis for Region I.

The accuracy of the burnup dependent isotopics is given in Table 3-2. These measurements were taken from the Yankee Core[10] and the PHOENIX predictions show excellent agreement. The agreement between measurement and prediction not only verifies the accuracy of the isotopic predictions, it also verifies the accuracy of the cross-sections of the actinides and therefore indirectly the reactivity worth. In order to account for uncertainties in the prediction of the actinide number densities, an uncertainty of 5 percent of the worth of the actinides (0.009 = 0.05 x 0.18) will be applied to the final rack multiplication factor. The accuracy of the reactivity calculations is shown in Table 3-3 giving the results of 81 critical experiments (described in Table 3-4) analyzed with PHOENIX[19].

Predicted burnups required to produce an equivalent reactivity in nonpoison spent fuel racks from PHOENIX and LEOPARD[11]/ TURTLE[12] (a calculational ability qualified by many years of reactor design experience) are shown in Table 3-5.

When comparing the reactivities of a spent and an unirradiated fuel assembly, an uncertainty arises in the ability to predict the depletion of the actinides in concert with the accumulation of fission products. Due to a lack of clean experimental information on the reactivity of spent fuel, the uncertainty must be inferred from comparisons of actual and calculated reactor reactivity lifetimes using PHOENIX. An extremely conservative estimate of the uncertainty due to depletion can be made by taking 5 percent of the change in reactivity due to irradiation. As an example, take a Westinghouse 15 x 15 OFA assembly with a 3.0 w/o initial enrichment at 22,000 MWD/MTU. The total reactivity change from fresh to spent is 0.227 Δk. The corresponding uncertainty in the Region II multiplication factor is then 0.0114 Δk (0.05 x 0.227 Δk).

In order to verify the applicability of these calculations for long-term storage, fission product decay after discharge was taken into account using CINDER. The fission products were permitted to decay for 30 years after discharge, and the time at which the cell reactivity peaked was chosen for the design basis. The maximum reactivity occurs at approximately 100 hours after shutdown (primarily due to the decay of Xe^{135}), at which point it begins to decrease, continuing throughout the 30-year time span.

3.1.3.2.2 Reactivity Equivalency

One of the basic principles behind the Region II rack design is the concept of reactivity equivalencing. In this concept, a constant rack k_∞ contour is constructed in enrichment-burnup space using PHOENIX. The intersection point at zero burnup is then calibrated using KENO-IV. Figure 3-3 shows the constant k_∞ contour based on a high enrichment endpoint of 4.50 w/o and 39,000 MWD/MTU. The advantage of this approach is that PHOENIX is used only to calculate relative reactivities as a function of irradiation while the actual rack reactivity determination is performed by the more powerful Monte Carlo method.

The principal motivation behind reactivity equivalencing is the relationship between assembly k_∞ and rack k_∞ as a function of initial enrichment. If a constant assembly k_∞ contour is constructed in enrichment/burnup space, the rack k_∞ increases as the enrichment increases. If the rack is designed to contain assemblies with high initial enrichments, a substantial amount of usable margin at lower enrichments would be lost by using the assembly k_∞ contour rather than the rack k_∞ contour. Reactivity equivalencing eliminates this unnecessary conservatism and permits more flexible storage capability at lower burnups.

3.1.3.2.3 Reactivity Determination

The final k_{eff} for Region II is determined using the same analytical methods and treatment of mechanical uncertainties as described in Section 3.1.3.1 for Region I. The actual conditions for this determination are defined by the zero burnup intercept point in Figure 3-3. In this instance the intercept is at 1.5 w/o U^{235}. The design model for Region II is therefore on an unirradiated assembly at 1.5 w/o enrichment. Studies have shown that the axial burnup distributions of depleted fuel assemblies have no impact on the fuel rack reactivity.

During the actual installation of the new spent fuel storage racks it will be necessary to temporarily store some Region I fuel assemblies in the Region II racks. Therefore, as discussed in Section 3.1.4.2.2, the k_{eff} for Region II is also determined assuming a checkerboard storage configuration, i.e., alternate cell locations, with unirradiated assemblies at a maximum assumed 4.5 w/o enrichment. Adjacent vacant spaces between stored assemblies in this temporary configuration will be controlled and maintained administratively to prevent insertion of an assembly.

3.1.4 Rack Modification

3.1.4.1 Region I

3.1.4.1.1 Fuel Storage

The Region I spent fuel storage rack design is described in Section 4.1.2.1.1. The minimum B^{10} loading in the Region I poison plates is 0.02 gm B^{10}/cm^2. Fuel assembly configuration and dimensions are shown in Figure 3.2.3-9 of the Turkey Point Updated FSAR.

For normal operation and using the method described in Section 3.1.3.1, the k_{eff} for the Region I rack is determined in the following manner.

$$k_{eff} = k_{nominal} + B_{mech} + B_{method} + B_{part} + \left[\left(ks_{nominal}\right)^2 + \left(ks_{method}\right)^2 + \left(ks_{mech}\right)^2\right]^{1/2}$$

where:

$k_{nominal}$	=	nominal case KENO k_{eff}
B_{mech}	=	bias to account for material thickness and construction tolerances
B_{method}	=	method bias determined from benchmark critical comparisons
B_{part}	=	bias to account for poison particle self-shielding
$ks_{nominal}$	=	95/95 uncertainty in the nominal case KENO k_{eff}
ks_{method}	=	95/95 uncertainty in the method bias
ks_{mech}	=	95/95 uncertainty to account for material thickness and construction tolerances.

The final k_{eff} for Region I from this analysis will be less than 0.95, including all uncertainties at a 95/95 probability/confidence level. Therefore, the acceptance criteria for criticality is met.

3.1.4.1.2 Sensitivity Analysis

To show the dependence of k_{eff} on fuel storage cell parameters, sensitivity studies are performed in which the poison loading, the fuel enrichment, and the storage cell center-to-center spacing are varied. Figure 3-4 illustrates the results of these Region I sensitivity studies.

3.1.4.2 Region II

The Region II spent fuel storage rack design is described in Section 4.1.2.1.2. The minimum B^{10} loading in the Region II poison plates is 0.012 gm B^{10}/cm^2. Fuel assembly configuration and dimensions are shown in Figure 3.2.3-9 of the Turkey Point Updated FSAR.

3.1.4.2.1 Spent Fuel Storage

The final k_{eff} for Region II with spent fuel is constructed according to the following formula:

$$k_{eff} = k_{nom} + B_{meth} + B_{mech} + B_{part} + \left[\left(ks_{meth}\right)^2 + \left(ks_{mech}\right)^2 + \left(ks_{nom}\right)^2 + \left(ks_{pu}\right)^2 + \left(ks_{bu}\right)^2\right]^{1/2}$$

where:

k_{nom} = nominal case KENO k_{eff}

B_{meth} = method bias determined from benchmark critical comparisons

B_{mech} = bias to account for material thickness and construction tolerances

B_{part} = bias to account for poison particle self-shielding

ks_{meth} = 95/95 uncertainty in the method bias

ks_{mech} = 95/95 uncertainty to account for material thickness and construction tolerances

ks_{nom} = 95/95 uncertainty in the nominal case KENO k_{eff}

ks_{pu} = 95/95 uncertainty in the plutonium reactivity

ks_{bu} = 95/95 uncertainty in reactivity as a function of irradiation

While it may be argued that ks_{bu} and ks_{pu} are not independent and should not be combined statistically, it should be considered that the reactivity of fuel as a function of burnup depends implicitly on the production rate of plutonium. The two uncertainties are so closely related that accounting for them twice is a conservative form of double accounting.

The final k_{eff} for Region II from this analysis will be less than 0.95, including all uncertainties at a 95/95 probability/confidence level. Therefore, the acceptance criteria for criticality is met.

3.1.4.2.2 Checkerboard Fuel Storage

During the installation of the new spent fuel storage racks, it will be necessary to temporarily store Region I fuel without the necessary burnup in the Region II racks. During this temporary fuel storage condition, a checkerboard storage configuration, i.e., alternate cell occupation, will be maintained in the Region II storage racks. Strict administrative controls will be utilized to ensure adjacent spaces between stored assemblies are unoccupied.

The final k_{eff} for Region II with checkerboard loading is constructed according to the following formula:

$$k_{eff} = k_{nom} + B_{meth} + B_{mech} + B_{part} + \left[\left(ks_{meth}\right)^2 + \left(ks_{mech}\right)^2 + \left(ks_{nom}\right)^2\right]^{1/2}$$

where:

k_{nom} = nominal case KENO k_{eff}

B_{meth} = method bias determined from benchmark critical comparisons

B_{mech} = bias to account for material thickness and construction tolerances

B_{part} = bias to account for poison particle self-shielding

ks_{meth} = 95/95 uncertainty in the method bias

ks_{mech} = 95/95 uncertainty to account for material thickness and construction tolerances

ks_{nom} = 95/95 uncertainty in the nominal case KENO k_{eff}

The final k_{eff} for Region II with a checkerboard fuel loading will be less than 0.95 including all uncertainties at a 95/95 probability/confidence level. Therefore the acceptance criteria for criticality is met.

3.1.4.2.3 Sensitivity Analysis

To show the dependence of k_{eff} on fuel storage cell parameters, sensitivity studies are performed in which the poison loading, the fuel enrichment, and the storage cell center-to-center spacing are varied. Figures 3-5 and 3-6 illustrate the results of these Region II sensitivity studies.

3.1.5 <u>Acceptance Criteria for Criticality</u>

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

Methods for initial and long-term verification of poison material stability and mechanical integrity are discussed in Section 4.8.

3.2 DECAY HEAT CALCULATIONS FOR THE SPENT FUEL POOL (BULK)

3.2.1 <u>Spent Fuel Pool Cooling System Design</u>

The existing spent fuel pit cooling system removes residual heat from fuel stored in the spent fuel pit. This existing system cools one-third of the core freshly discharged from the reactor plus up to 4-1/3 cores of previously discharged fuel, but it can accommodate the heat load from 1-1/3 freshly discharged cores, plus the heat load from the previously discharged fuel (i.e., 3-1/3 cores). When one-third of a freshly discharged core is present, the pump and spent fuel heat exchanger will handle the load and maintain a pool water temperature less than 127° F. When a freshly discharged core is stored in addition to the one-third of a recently discharged core, the pool water temperature is maintained below 150° F.

As described in Turkey Point Updated FSAR Section 9.3[13], the spent fuel pit cooling loop consists of a pump, heat exchanger, filter, demineralizer, piping, and associated valves and instrumentation. The pump draws water from the pit, circulates it through the heat exchanger, and returns it to the pit. Component Cooling Water cools the heat exchanger. As stated in Updated FSAR Section 9.3 and SER Section 9.5[14], redundancy of this equipment is not required because of the large heat capacity of the pit and its corresponding slow heat-up rate. Nonetheless, a 100-percent-capacity spare pump which is permanently piped into the spent fuel pit cooling system has been installed. This pump is capable of operating in place of the originally installed pump, but not in parallel with the original pump. Also, alternate connections are provided for connecting a temporary pump to the spent fuel pit loop.

3.2.2 Decay Heat Analyses

For the proposed reracking, the spent fuel pit cooling system has been evaluated to determine the effect on the system of increasing the spent fuel stored in the pit from 4-2/3 cores to 9 cores (1413 assemblies which exceeds the actual maximum storage capacity of 1404 assemblies). This expansion of the spent fuel storage in the pit increases the decay heat load for each pool from 8.82×10^6 Btu/hour to 16.98×10^6 Btu/hour and the pool peak transient water temperature after refueling to less than 141° F. When a freshly discharged core is stored in addition to the one-half of a recently discharged core, plus the heat load from the previously discharged fuel (i.e., 7-1/2 cores), the pool water temperature is maintained less than 180° F.

The preceding discussions are based on the following assumptions:

a. One-half core unloaded 150 hours after reactor shutdown, plus 8-1/2 cores in storage from 1-1/2 years to 20 years, assuming irradiation times for the fuel of 38,000 effective full power hours. The 9 cores is 9 assemblies more than the maximum capacity which the spent fuel pool can accommodate. Actual refuelings may be less than one-half cores as indicated in Table 5-3. However, the assumption of one-half core offloads is conservative in that it results in a higher peak heat load for the pool than assuming smaller offloads over a longer period of time.

b. Spent fuel cooling system flow of 2200 gpm, (Ref. Updated FSAR Appendix 14E.)

c. Component cooling water flow to spent fuel pit heat exchanger of 2800 gpm at a temperature of 100 F, (Ref. Updated FSAR Appendix 14E and Section 9.3, respectively.)

d. Credit is only taken for heat removal via the spent fuel pool cooling system.

The spent fuel decay heat calculations were performed in accordance with the method provided in NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light-Water Reactors for Long-Term Cooling [15].

In view of the effect on heat load by increasing the storage of spent fuel, the existing spent fuel pit cooling system is considered to be adequate. The spent fuel pit is designed to withstand stresses associated with a steady-state gradient of 150° F. As shown on Figure 3-7 for the final 10 refuelings, the pool peak transient water temperature after refueling is less than 141° F.

The addition of storage capacity to accommodate spent fuel from normal refuelings, as discussed above, results in a change in the design basis heat load from 8.82 x 10^6 Btu/hour to 16.98 x 10^6 Btu/hour for each pool. This increases the heat-up rate for the pool in the event of a complete loss of cooling capability; however, this is still considered sufficient time to provide an alternate means for cooling in the event of a failure in the cooling system as shown below:

	Existing	Augmented
Design basis heat load, Btu/hr	8.82 x 10^6	16.98 x 10^6
Time for pool water to heat from 127° to 180° F (Existing)/141° to 180°F (Augmented), Hours	15	4

The total increase in heat load rejected to the environment through the cooling systems due to the increased spent fuel storage over the current heat load rejected is 8.16 x 10^6 Btu/hour (16.98 - 8.82) from either Units 3 or 4. This represents an increase of approximately 0.3 percent of the total heat rejected to the cooling reservoir from either Units 3 or 4. The cooling reservoir is a closed loop system which rejects heat from the circulating water and intake cooling water systems to the environment by evaporation and direct radiation. There is no interconnection between the cooling reservoir and offsite bodies of water. Therefore, the increase in heat rejected will have negligible impact on the environment.

The increase in heat load does not alter in any way the existing facility design bases. Thus, the heat load increase is acceptable. This decay heat analysis is also bounding for the temporary fuel storage configuration (see Section 4.7.4) that will be utilized during rack installation.

3.2.3 Spent Fuel Pool Makeup

The increased spent fuel storage capability does not affect the design basis or functional requirements for makeup to the spent fuel pool. Makeup requirements for unexpected leakage are as summarized in the AEC (NRC) Final Safety Evaluation Report, Section 9.6.

Normal makeup is supplied from the demineralized water system or from the refueling water storage tank. Alternate means of makeup also include temporary connections from the fire water system or the primary water storage tank. The discussions in Section 5.2 (Loss of Makeup) of Appendix 14E of the Updated FSAR are not changed by the proposed rerack.

As a result of the increased spent fuel storage capacity, the makeup required due to evaporation will increase less than 1 gpm over current requirements. This makeup requirement is based on the design basis heat load due to normal refueling being present in the pool. This increase is well within the capacity of the normal makeup systems.

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

The purpose of 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 Turkey Point spent fuel pool.

3.3.1 Criteria

The criteria used to determine the acceptability of the design from a thermal-hydraulic viewpoint is 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 when the cooling system is operational. When the cooling system is postulated to be inoperable, adequate cooling implies that the temperature of the fuel cladding should be sufficiently low that no structural failures would occur and that no safety concerns would exist.

2. The rack design must not allow trapped air or steam. Direct gamma heating of the storage cell walls and the intercell water must be considered.

3.3.2 Key Assumptions

1. The nominal water level is 25.0 feet above the top of the fuel storage racks.

2. The maximum fuel assembly decay heat output is 50.5 BTU/sec following 150 hours decay after shutdown. For conservatism, this value will be used for both Region I and Region II fuel assemblies.

3. For normal operations, the maximum pool temperature shall not exceed 150°F. For conservatism, the temperatures of the storage racks and the stored fuel are evaluated assuming that the temperature of the water at the inlet to the storage cells is 150°F during normal operation.

4. Under postulated accident conditions, when no pool cooling systems are operational, the maximum temperature at the inlet to the cells is assumed to be equal to the saturation temperature at atmospheric pressure or 212°F.

3.3.3 Description of Analytical Method and Types of Calculations Performed

A natural circulation calculation is employed to determine the thermal-hydraulic conditions within the spent fuel storage cells. The model used assumes that all downflow occurs in the peripheral gap between the pool walls and the outermost storage cells and all lateral flow occurs in the space between the bottom of the racks and the bottom of the pool. The effect of flow area blockage in the region is conservatively accounted for and a multi-channel formulation is used to determine the variation in axial flow velocities through the various storage cells. The hydraulic resistance of the storage cells and the fuel/pin assemblies is conservatively modeled by applying large uncertainty factors to loss coefficients obtained from various sources. Where necessary, the effect of Reynolds Number on the hydraulic resistance is considered, and the variation in momentum and elevation head pressure drops with fluid density is also determined.

The solution is obtained by iteratively solving the conservation equations (mass, momentum and energy) for the natural circulation loops. The flow velocities and fluid temperatures that are obtained are then used to determine the fuel cladding temperatures. An elevation view of a typical model is shown in Figure 3-8 where the flow paths are indicated by arrows. Note that each cell shown in that sketch actually corresponds to a row of cells that is located at the same distance from the pool walls. This is more clearly shown in a plan view, Figure 3-9.

As shown in Figure 3-9, the lateral flow area underneath the storage cells decreases as the distance from the wall increases. This counteracts the decrease in the total lateral flow that occurs because of flow that branches up and flows into the cells. This is significant because the lateral flow velocity affects both the lateral pressure drop underneath the cells and the turning losses that are experienced as the flow branches up into the cells. These effects are considered in the natural circulation analysis.

The most recently discharged or "hottest" fuel assemblies are assumed to be located in various rows during different calculations in order to verify that they may be placed anywhere within the pool without violating safety limits. In order to simplify the calculations, each row of the model must be composed of storage cells having a uniform decay heat level. This decay heat level may or may not correspond to a specific batch of fuel, but the model is constructed so that the total heat input is correct. The "hottest" fuel assemblies are all assumed to be placed in a given row of the model in order that conservatively accurate results can be obtained for those assemblies. In fact, the most conservative analysis that can be performed is to assume that all assemblies in the pool (or

rows in the model) have the same maximum decay heat rate. This maximizes the total natural circulation flowrate which leads to conservatively large pressure drops in the downcomer and lateral flow regions which reduces the driving pressure drop across the limiting storage locations.

Since the natural circulation velocity strongly affects the temperature rise of the water and the heat transfer coefficient within a storage cell, the hydraulic resistance experienced by the flow is a significant parameter in the evaluation. In order to minimize the resistance, the design of the inlet region of the racks has been chosen to maximize this flow area. Each storage cell has one or more flow openings as shown in Figure 3-10. The use of these large or multiple flow holes substantially minimizes the possibility that all flow into the inlet of a given cell can be blocked by debris or other foreign material that may get into the pool. In order to determine the impact of a partial blockage on the thermal-hydraulic conditions in the cells, an analysis is also performed for various assumed blockages.

The analyses that have been described only address the flow through the storage cells. As noted in the discussion of criteria, it is also required that the flow and temperatures in the axial gap between adjacent storage cells for Region I racks be evaluated. In order to preclude the possibility of stagnant conditions in these gaps, flow relief areas are provided at the location of the grid support structures as shown in Figure 3-11. This flow area also avoids the trapping of air or steam in the rack structure. The thermal-hydraulic conditions in the gap region are evaluated by using a parallel path thermal-hydraulic model of the gap and cell under consideration. This analysis considers the gamma heat generation in the cell enclosure, poison material, and cell wrapper in addition to the decay heat input. Using the cell flow velocity and driving pressure differential obtained from the previously described pool analyses, the flow velocity in the gap and the axial temperature distributions of the coolant and structure are determined. The radial temperature distributions through the various components are also considered.

3.4 POTENTIAL FUEL AND RACK 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 methods utilized ensure that postulated accidents do not result in a loss of cooling to either the spent fuel pool or the reactor, or result in a k_{eff} in the spent fuel pool exceeding 0.95.

3.4.1 Rack Module Mishandling

The potential for mishandling of rack modules during the rerack operation has been evaluated. At no time will the cask handling crane or the temporary construction crane carry a

rack module directly over spent fuel stored in the pool. The procedures and administrative controls governing the rerack operation will ensure the safe handling of rack modules. Both the temporary construction crane and the cask handling crane meet the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [16].

In the unlikely event that a rack should strike the side of another rack module containing fuel assemblies, the consequences of this postulated accident would be bounded by the cask drop evaluations described in Sections 3.1.2 and 5.3.1.2.

3.4.2 Temporary Construction Crane Drop

During the rerack operation, a temporary construction crane will be installed in the spent fuel pool. This installation will be performed using lift rigs which meet the design and operational requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". The consequences of a postulated accident during this installation are bounded by the cask drop evaluations described in Sections 3.1.2 and 5.3.1.2.

3.4.3 Loss of Pool Cooling

During the re-racking operation, it will be necessary to raise and maneuver the old racks out of the spent fuel pool in order to install the new spent fuel racks (See Section 4.7.4). The handling of these heavy loads will be accomplished by the use of a temporary construction crane and the cask handling crane. Both of these cranes meet the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", to prevent accidental dropping.

In the event that a rack should drop on the floor, the potential for loss of pool cooling could be postulated. An analysis has previously been submitted and accepted by the NRC (Reference [17]) for dropping of the spent fuel cask. The results of this analysis demonstrated that the pool floor would remain elastic during impact and that a crack would not develop. This cask weighs substantially more than a single rack assembly and has a smaller cross sectional area for load distribution. The loss of pool water inventory from a rack drop is bounded by this previous analysis for loss of pool water inventory from a cask drop. Therefore, loss of spent fuel cooling from loss of pool water inventory will not occur as a result of a rack drop.

3.5 TECHNICAL SPECIFICATIONS

Proposed revisions to existing Turkey Point Technical Specifications [18] are shown on the following Technical Specification pages 3.12-1, 5.4-1, B3.12-1 and Table 4.1-2 (Sheet 2) as barred,

and proposed Technical Specification 3.17 including Table 3.17-1 and page B3.17-1. These Technical Specification revisions do not involve a significant reduction in any margin of safety.

3.6 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. W. E. Ford III, et al, "A 218-Group Neutron Cross-Section Library in the AMPX Master Interface Format for Criticality Safety Studies," ORNL/CSD/TM-4 (July 1976).

3. N. M. Greene, et al, "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL/TM-3706 (March 1976).

4. L. M. Petrie and N. F. Cross, "KENO IV--An Improved Monte Carlo Criticality Program," ORNL-4938 (November 1975).

5. S. R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 2.35 wt percent ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2438 (October 1977).

6. S. R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 4.29 wt percent ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2615 (March 1978).

7. J. T. Thomas, "Critical Three-Dimensional Arrays of U (93.2) -- Metal Cylinders," Nuclear Science and Engineering, Volume 52, pages 350-359 (1973).

8. A. J. Harris, et al, "A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors," WCAP-10106, June 1982.

9. T. R. England, "CINDER - A One-Point Depletion and Fission Product Program," WAPD-TM-334, August 1962.

10. J. B. Melehan, "Yankee Core Evaluation Program Final Report," WCAP-3017-6094, January 1971.

11. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September 1963.

12. S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758-A, January 1975.

13. Turkey Point Plant Units 3 and 4, Updated Final Safety Analysis Report, Docket Nos. 50-250 and 50-251.

14. Turkey Point Plant Units 3 and 4, Safety Evaluation Report, Docket Nos. 50-250 and 50-251.

15. Nuclear Regulatory Commission, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling", Branch Technical Position ASB 9-2, NUREG-0800, July 1981.

16. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants", NUREG-0612, July 1980.

17. Letter from G. Lear, NRC, to R. E. Uhrig, FPL, dated July 9, 1976.

18. Turkey Point Plant Units 3 and 4, Technical Specifications, Docket Nos. 50-250 and 50-251.

19. L. E. Strawbridge and R. F. Barry, "Critical Calculations for Uniform Water-Moderated Lattices", Nuclear Science and Engineering, Volume 23, 1965.

3.12 CASK HANDLING

Applicability: Applies to limitations during cask handling.

Objective: To minimize the possibility of an accident during cask handling operations that would affect the health and safety of the public.

Specifications: During cask handling operations:

1. The spent fuel cask shall not be moved into the spent fuel pit until all the spent fuel in the pit has decayed for a minimum of 1525 hours.**

2. Only a single-element cask may be moved into the spent fuel pit.

3. A fuel assembly shall not be removed from the spent fuel pit in a shipping cask until it has decayed for a minimum of 120 days.*

*The Region 10 fuel that was in the Unit 3 reactor during the period of April 19, 1981 through April 24, 1981, may be removed from the Unit 3 spent fuel pit in a shipping cask after a minimum decay period of 95 days.

**The spent fuel cask can be moved into the Unit 4 Spent Fuel Pit after a minimum decay of 1000 hours until the new two-region high density spent fuel racks are installed.

B3.12 BASIS FOR LIMITING CONDITIONS FOR OPERATION, CASK HANDLING

Requiring spent fuel decay time to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit will keep potential offsite doses well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four.

Requiring that spent fuel decay time be at least 120 days prior to moving a fuel assembly outside the fuel storage pit in a shipping cask will ensure that potential offsite doses are a fraction of 10 CFR 100 limits should a dropped cask and ruptured fuel assembly release activity directly to the atmosphere.

3.17 SPENT FUEL STORAGE

Applicability: Applies to limitations on the storage of spent fuel assemblies.

Objective: To minimize the possibility of exceeding the reactivity design limits for storage of spent fuel.

Specifications:

1. Fuel assemblies containing more than 43.9 grams of U-235 per axial centimeter shall not be placed in the single region spent fuel storage racks. After installation of the two-region high density spent fuel racks, the maximum loading for fuel assemblies in the spent fuel racks is 57.7 grams of U-235 per axial centimeter.

2. The minimum boron concentration while fuel is stored in the spent fuel pit shall be 1950 ppm.

3.* Storage in Region II of the spent fuel pit shall be further restricted by burnup and enrichment limits specified in Table 3.17-1.

4.* During the re-racking operation only, fuel that does not meet the burnup requirements for normal storage in Region II may be stored in Region II in a checkerboard arrangement (i.e., no fuel stored in adjacent spaces).

*This Technical Specification is applicable only after installation of the new two-region high density spent fuel racks.

Table 3.17-1
Spent Fuel Burnup Requirements
for Storage in Region II of
the Spent Fuel Pit

Initial w/o	Discharge Burnup GWD/MT
1.5	0
1.75	5.0
2.0	9.0
2.2	12.0
2.4	14.8
2.6	17.6
2.8	20.1
3.0	22.6
3.2	25.0
3.4	27.4
3.6	29.6
3.8	31.8
4.0	34.0
4.2	36.1
4.5	39.0

Linear interpolation between two consecutive points will yield conservative results.

B3.17 BASES FOR LIMITING CONDITIONS FOR OPERATION, SPENT FUEL STORAGE

1. The spent fuel storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure k_{eff} is equal to or less than 0.95 for normal operations and postulated accidents.

2.* The spent fuel racks are divided into two regions. Region I racks have a 10.6-inch center-to-center spacing and the Region II racks have a 9.0-inch center-to-center spacing. Because of the larger center-to-center spacing and poison (B^{10}) concentration of Region I cells the only restriction for placement of fuel is that the initial fuel loading is equal to or less than 57.7 grams of U-235 per axial centimeter of fuel assembly. This assures the fuel enrichment limit assumed in the spent fuel safety analyses will not be exceeded. Prior to placement in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.17-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.

*This Technical Specification is applicable upon installation of the new two-region high density spent fuel racks.

PROPOSED TECHNICAL SPECIFICATION

TABLE 4.1-2 (Sheet 2 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Check	Frequency	Max. Time Between Tests (Days)
5.	Control Rods (cont'd)	Partial movement of full length rods	Biweekly while critical	20
6.	Pressurizer Safety Valves	Set Point	Each refueling shutdown	NA
7.	Main Steam Safety Valves	Set Point	Each refueling shutdown	NA
8.	Containment Isolation Trip	Functioning	Each refueling shutdown	NA
9.	Refueling System Interlocks	Functioning	Prior to each refueling	NA
10.	Accumulator	Boron Concentration	At least once per 31 days and within 6 hours after each solution volume increase of ≧1% of tank volume.†	
11.	Reactor Coolant System Leakage	Evaluate	Daily	NA
12.	Diesel Fuel Supply	Fuel inventory	Weekly	10
13.	Spent Fuel Pit	Boron Concentration	Monthly	45
14.	Secondary Coolant	I-131 Concentration	Weekly*†	10
15.	Vent Gas and Particulates	I-131 and Particulate Activity	Weekly*	10
16.	Fire Protection Pump and Power Supply	Operable	Monthly	45
17.	Turbine Stop and Control Valves, Reheater Stop and Intercept Valves	Closure	Monthly***	45
18.	LP Turbine Rotor Inspector (w/o rotor disassembly)	V, MT, PT	Every 5 years	6 years
19.	Spent Fuel Cask Crane Interlocks	Functioning	Within 7 days	7 days when crane is being used to maneuver spent fuel cask.

Rev. 0

5.4 FUEL STORAGE

1. The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class 1 structures. Each spent fuel pit has a stainless steel liner to ensure against leakage.

2. The racks for new fuel storage are designed to store fuel in a safe subcritical array. The fuel is stored vertically in an array with sufficient center-to-center spacing to assure k_{eff} is equal to or less than 0.95 for normal operations or accident conditions. Fuel containing more than 43.9 grams of U-235 per axial centimeter of fuel assembly shall not be placed in the New Fuel Storage Area.

3. Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks.* Strict administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

   *During rack installation, it will be necessary to temporarily store Region I fuel in the Region II spent fuel racks. Strict administrative controls will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks.

TABLE 3-1

BENCHMARK CRITICAL EXPERIMENTS [5,6,7]

	General Description	Enrichment w/o U235	Reflector	Separating Material	Characterizing Separation (cm)	k_{eff}
1.	UO_2 rod lattice	2.35	water	water	11.92	1.004 ± .004
2.	UO_2 rod lattice	2.35	water	water	8.39	0.993 ± .004
3.	UO_2 rod lattice	2.35	water	water	6.39	1.005 ± .004
4.	UO_2 rod lattice	2.35	water	water	4.46	0.994 ± .004
5.	UO_2 rod lattice	2.35	water	stainless steel	10.44	1.005 ± .004
6.	UO_2 rod lattice	2.35	water	stainless steel	11.47	0.992 ± .004
7.	UO_2 rod lattice	2.35	water	stainless steel	7.76	0.992 ± .004
8.	UO_2 rod lattice	2.35	water	stainless steel	7.42	1.004 ± .004
9.	UO_2 rod lattice	2.35	water	boral	6.34	1.005 ± .004
10.	UO_2 rod lattice	2.35	water	boral	9.03	0.992 ± .004
11.	UO_2 rod lattice	2.35	water	boral	5.05	1.001 ± .004
12.	UO_2 rod lattice	4.29	water	water	10.64	0.999 ± .005
13.	UO_2 rod lattice	4.29	water	stainless steel	9.76	0.999 ± .005
14.	UO_2 rod lattice	4.29	water	stainless steel	8.08	0.998 ± .006
15.	UO_2 rod lattice	4.29	water	boral	6.72	0.998 ± .005
16.	U Metal Cylinders	93.2	bare	air	15.43	0.998 ± .003
17.	U Metal Cylinders	93.2	paraffin	air	23.84	1.006 ± .005
18.	U Metal Cylinders	93.2	bare	air	19.97	1.005 ± .003
19.	U Metal Cylinders	93.2	paraffin	air	36.47	1.001 ± .004
20.	U Metal Cylinders	93.2	bare	air	13.74	1.005 ± .003
21.	U Metal Cylinders	93.2	paraffin	air	13.74	1.005 ± .004
22.	U Metal Cylinders	93.2	bare	plexiglass	15.74	1.010 ± .003

TABLE 3-1 (Continued)

	General Description	Enrichment w/o U235	Reflector	Separating Material	Characterizing Separation (cm)	k_{eff}
23.	U Metal Cylinders	93.2	paraffin	plexiglass	24.43	1.006 ± .004
24.	U Metal Cylinders	93.2	bare	plexiglass	21.74	0.999 ± .003
25.	U Metal Cylinders	93.2	paraffin	plexiglass	27.94	0.994 ± .005
26.	U Metal Cylinders	93.2	bare	steel	14.74	1.000 ± .003
27.	U Metal Cylinders	93.2	bare	plexiglass steel	16.67	1.006 ± .005

TABLE 3-2

COMPARISON OF PHOENIX ISOTOPIC PREDICTION

TO YANKEE CORE 5 MEASUREMENTS

Quantity (Atom Ratio)	% Difference
U235/U	-0.67
U236/U	-0.28
U238/U	-0.03
PU239/U	+3.27
PU240/U	+3.63
PU241/U	-7.01
PU242/U	-0.20
PU239/U238	+3.24
MASS(PU/U)	+1.41
FISS-PU/TOT-PU	-0.02

Percent difference is average difference of ten comparisons for each isotope.

TABLE 3-3

BENCHMARK CRITICAL EXPERIMENTS

PHOENIX COMPARISONS

Description of Experiments	Number of Experiments	PHOENIX k_{eff} Using Experimental Bucklings
UO_2		
Al clad	14	0.9947
SS clad	19	0.9944
Borated H_2O	7	0.9940
Subtotal	40	0.9944
U-Metal		
Al clad	41	1.0012
Total	81	0.9978

TABLE 3-4
DATA FOR U METAL AND U02 CRITICAL EXPERIMENTS[19]

Case Number	Cell Type	A/O U-235	H2O/U Ratio	Fuel Density (G/CC)	Pellet Diameter (CM)	Material Clad	Clad OD (CM)	Clad Thickness (CM)	Lattice Pitch (CM)	B-10 PPM
1	Hexa	1.328	3.02	7.53	1.5265	Aluminum	1.6916	.07110	2.2050	0.0
2	Hexa	1.328	3.95	7.53	1.5265	Aluminum	1.6916	.07110	2.3590	0.0
3	Hexa	1.328	4.95	7.53	1.5265	Aluminum	1.6916	.07110	2.5120	0.0
4	Hexa	1.328	3.92	7.52	.9855	Aluminum	1.1506	.07110	1.5580	0.0
5	Hexa	1.328	4.89	7.52	.9855	Aluminum	1.506	.07110	1.6520	0.0
6	Hexa	1.328	2.88	10.53	.9728	Aluminum	1.1506	.07110	1.5580	0.0
7	Hexa	1.328	3.58	10.53	.9728	Aluminum	1.1506	.07110	1.6520	0.0
8	Hexa	1.328	4.83	10.53	.9728	Aluminum	1.1506	.07110	1.8060	0.0
9	Square	2.734	2.18	10.18	.7620	SS-304	.8594	.04085	1.0287	0.0
10	Square	2.734	2.92	10.18	.7620	SS-304	.8594	.04085	1.1049	0.0
11	Square	2.734	3.86	10.18	.7620	SS-304	.8594	.04085	1.1938	0.0
12	Square	2.734	7.02	10.18	.7620	SS-304	.8594	.04085	1.4554	0.0
13	Square	2.734	8.49	10.18	.7620	SS-304	.8594	.04085	1.5621	0.0
14	Square	2.734	10.38	10.18	.7620	SS-304	.8594	.04085	1.6891	0.0
15	Square	2.734	2.50	10.18	.7620	SS-304	.8594	.04085	1.0617	0.0
16	Square	2.734	4.51	10.18	.7620	SS-304	.8594	.04085	1.2522	0.0
17	Square	3.745	2.50	10.27	.7544	SS-304	.8600	.04060	1.0617	0.0
18	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	0.0
19	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	0.0
20	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	456.0
21	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	709.0
22	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1260.0
23	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1334.0
24	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1477.0
25	Square	4.069	2.55	9.46	1.1278	SS-304	1.2090	.04060	1.5113	0.0
26	Square	4.069	2.55	9.46	1.1278	SS-304	1.2090	.04060	1.5113	3392.0
27	Square	4.069	2.14	9.46	1.1278	SS-304	1.2090	.04060	1.4500	0.0
28	Square	2.490	2.84	10.24	1.0297	Aluminum	1.2060	.08130	1.5113	0.0
29	Square	3.037	2.64	9.28	1.1268	SS-304	1.1701	.07163	1.5550	0.0
30	Square	3.037	8.16	9.28	1.1268	SS-304	1.2701	.07163	1.1980	0.0
31	Square	4.069	2.59	9.45	1.1268	SS-304	1.2701	.07163	1.5550	0.0
32	Square	4.069	3.53	9.45	1.1268	SS-304	1.2701	.07163	1.6840	0.0
33	Square	4.069	8.02	9.45	1.1260	SS-304	1.2701	.07163	1.1980	0.0
34	Square	4.069	9.90	9.45	1.1268	SS-304	1.2701	.07163	2.3810	0.0
35	Square	2.490	2.84	10.24	1.0297	Aluminum	1.2060	.08130	1.5113	1677.0
36	Hexa	2.096	2.06	10.38	1.5240	Aluminum	1.6916	.07112	2.1737	0.0
37	Hexa	2.096	3.09	10.38	1.5240	Aluminum	1.6916	.07112	2.4052	0.0
38	Hexa	2.096	4.12	10.38	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
39	Hexa	2.096	6.14	10.38	1.5240	Aluminum	1.6916	.07112	2.9891	0.0
40	Hexa	2.096	8.20	10.38	1.5240	Aluminum	1.6916	.07112	3.3255	0.0
41	Hexa	1.307	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
42	Hexa	1.307	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
43	Hexa	1.307	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
44	Hexa	1.307	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
45	Hexa	1.307	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0

TABLE 3-4
DATA FOR U METAL AND U02 CRITICAL EXPERIMENTS
(Continued)

Case Number	Cell Type	A/O U-235	H2O/U Ratio	Fuel Density (G/CC)	Pellet Diameter (CM)	Material Clad	Clad OD (CM)	Clad Thickness (CM)	Lattice Pitch (CM)	B-10 PPM
46	Hexa	1.160	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
47	Hexa	1.160	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
48	Hexa	1.160	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
49	Hexa	1.160	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
50	Hexa	1.160	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0
51	Hexa	1.040	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
52	Hexa	1.040	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
53	Hexa	1.040	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
54	Hexa	1.040	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
55	Hexa	1.040	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0
56	Hexa	1.307	1.00	18.90	.9830	Aluminum	1.1506	.07112	1.4412	0.0
57	Hexa	1.307	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
58	Hexa	1.307	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
59	Hexa	1.307	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
60	Hexa	1.307	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
61	Hexa	1.160	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
62	Hexa	1.160	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
63	Hexa	1.160	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
64	Hexa	1.160	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
65	Hexa	1.160	1.00	18.90	.9830	Aluminum	1.1506	.07112	1.4412	0.0
66	Hexa	1.160	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
67	Hexa	1.160	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
68	Hexa	1.160	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
69	Hexa	1.160	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
70	Hexa	1.040	1.33	18.90	19.050	Aluminum	2.0574	.07620	2.8687	0.0
71	Hexa	1.040	1.58	18.90	19.050	Aluminum	2.0574	.07620	3.0086	0.0
72	Hexa	1.040	1.83	18.90	19.050	Aluminum	2.0574	.07620	3.1425	0.0
73	Hexa	1.040	2.33	18.90	19.050	Aluminum	2.0574	.07620	3.3942	0.0
74	Hexa	1.040	2.83	18.90	19.050	Aluminum	2.0574	.07620	3.6284	0.0
75	Hexa	1.040	3.83	18.90	19.050	Aluminum	2.0574	.07620	4.0566	0.0
76	Hexa	1.310	2.02	18.88	1.5240	Aluminum	1.6916	.07112	2.6160	0.0
77	Hexa	1.310	3.01	18.88	1.5240	Aluminum	1.6916	.07112	2.9900	0.0
78	Hexa	1.159	2.02	18.88	1.5240	Aluminum	1.6916	.07112	2.6160	0.0
79	Hexa	1.159	3.01	18.88	1.5240	Aluminum	1.6916	.07112	2.9900	0.0
80	Hexa	1.312	2.03	18.88	.9830	Aluminum	1.1506	.07112	1.7250	0.0
81	Hexa	1.312	3.02	18.88	.9830	Aluminum	1.1506	.07112	1.9610	0.0

TABLE 3-5

COMPARISON OF LEOPARD/TURTLE AND

PHOENIX EQUIVALENT REACTIVITY BURNUPS

FUEL TYPE	INITIAL U-235 w/o	EQUIVALENT BURNUP (GWD/MT) LEOPARD/TURTLE	PHOENIX	ΔBU(GWD/MT)
W-17x17	1.558	0	0	-
	2.750	17.09	16.80	-0.29
	4.5	36.00	35.73	-0.27
W-15x15	1.360	0	0	-
	2.250	15.52	15.57	0.05
	4.0	36.00	36.46	0.54
W-14x14	1.183	0	0	-
	1.750	13.01	13.01	0
	3.5	36.00	36.52	0.52

Fuel assemblies depleted under reactor conditions and modeled in cold non-poisoned spent fuel racks to determine constant reactivity burnups.

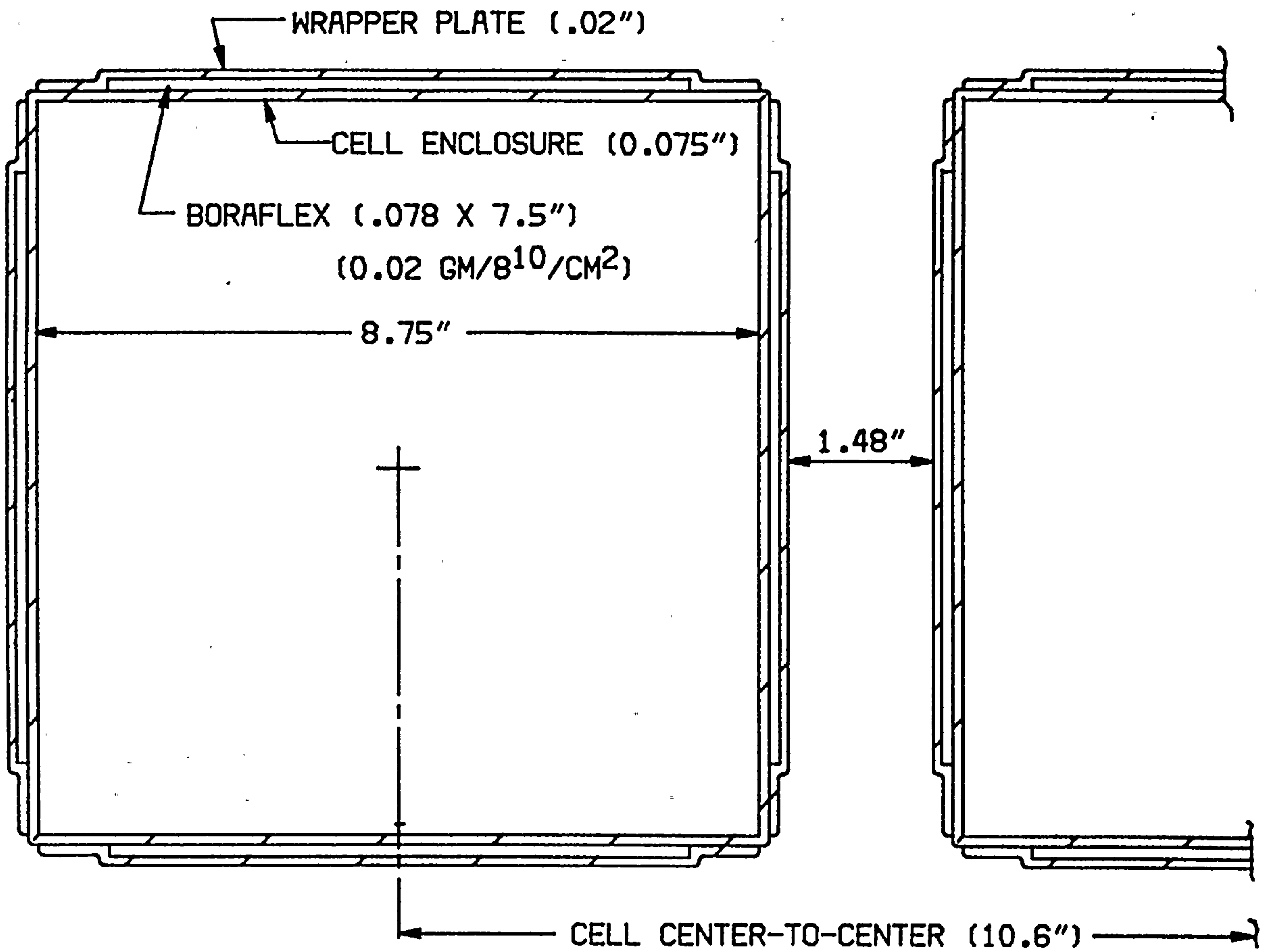


Figure 3-1
NOMINAL DIMENSIONS FOR THE REGION I STORAGE CELLS

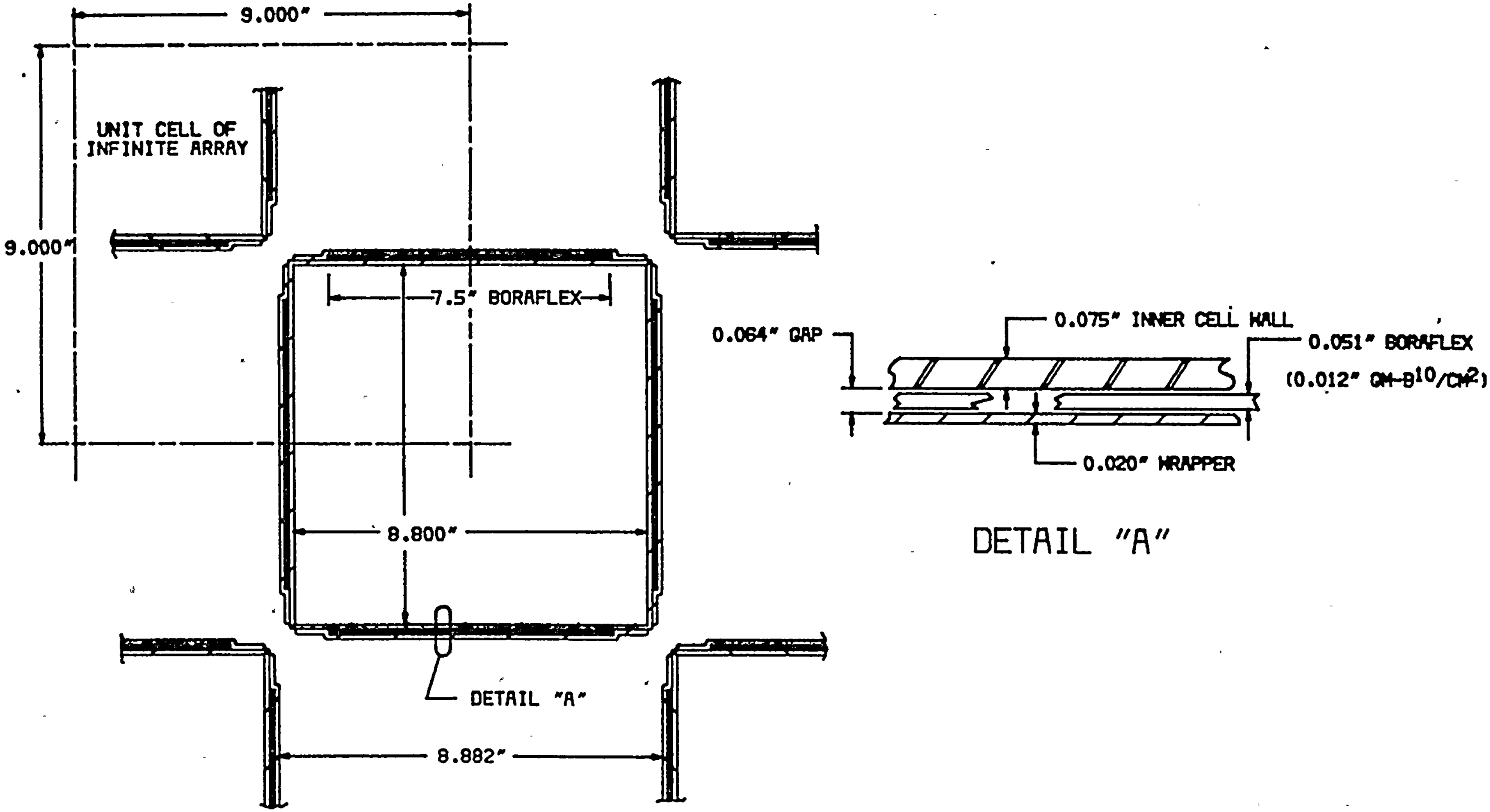


Figure 3-2

NOMINAL DIMENSIONS FOR THE REGION II STORAGE CELLS

Rev. 0

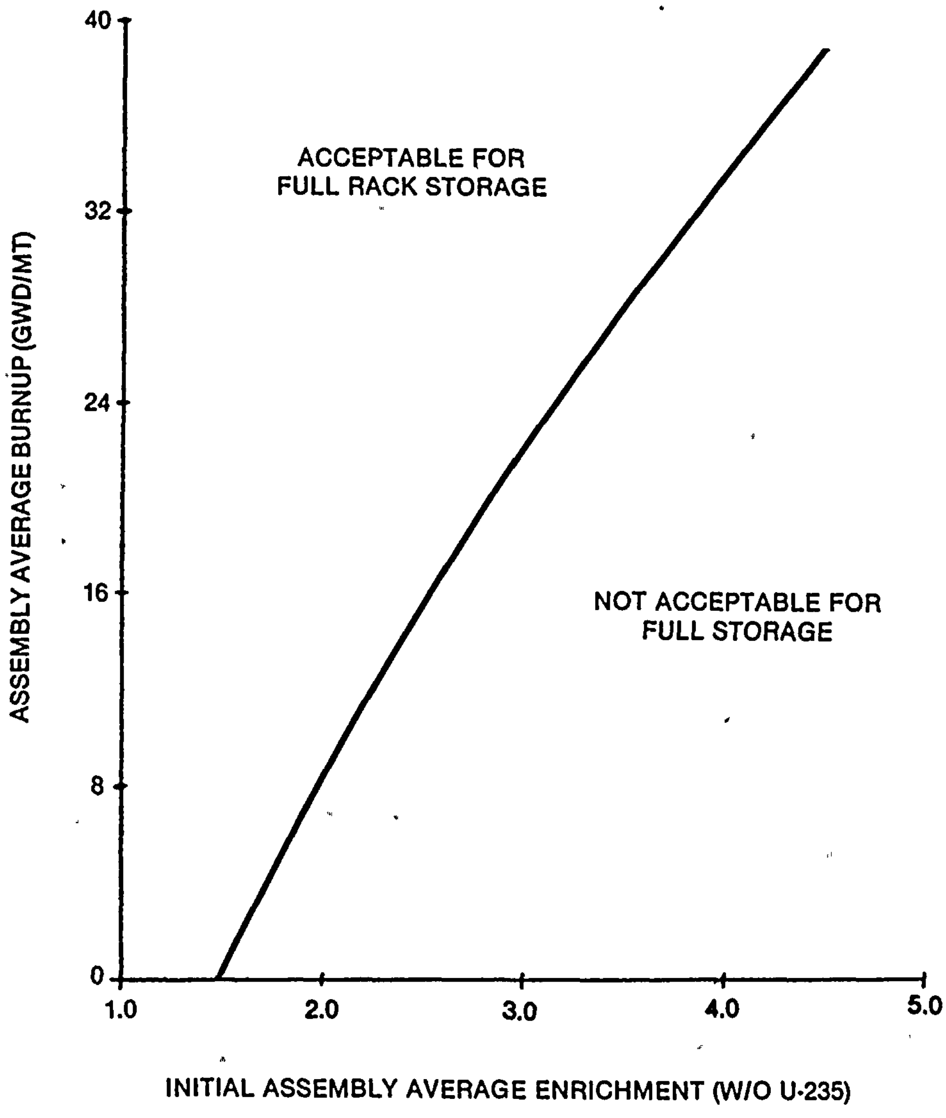
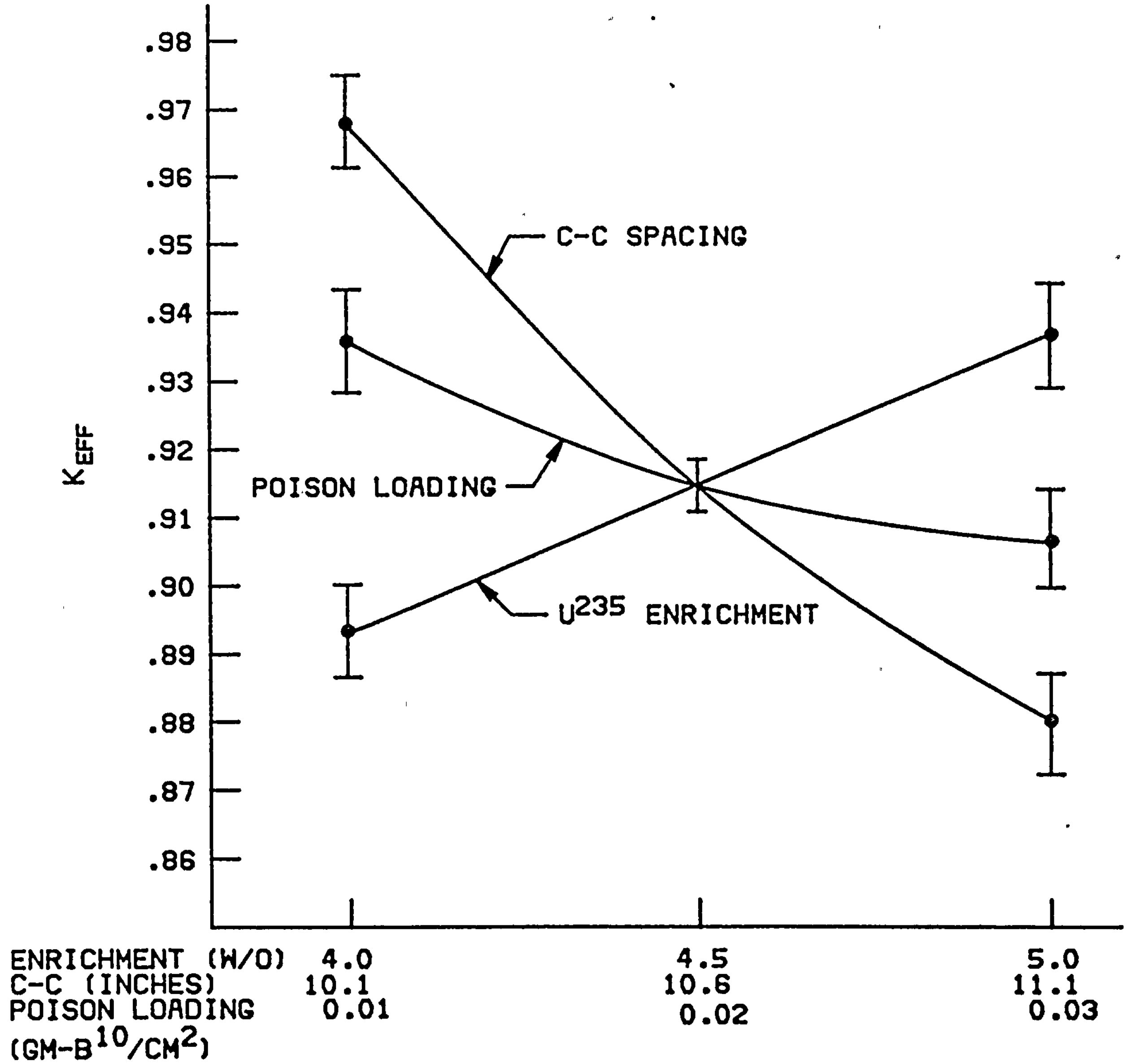


Figure 3-3

MINIMUM BURNUP VS. INITIAL ENRICHMENT FOR REGION II STORAGE



NOTES:

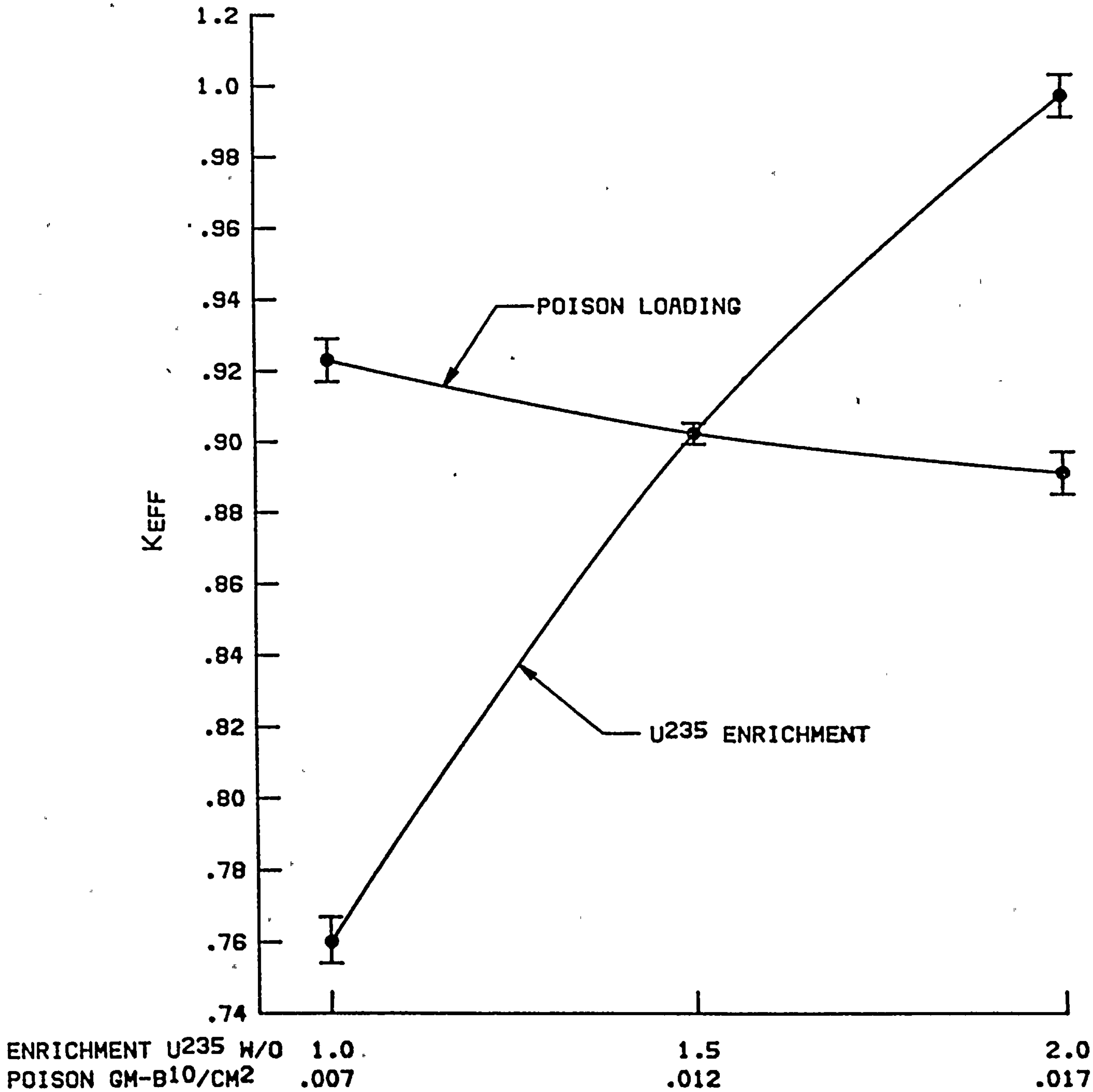
FOR ENRICHMENT CURVE, C-C = 10.6", POISON LOADING = 0.02 $GM\text{-}B^{10}/CM^2$

FOR C-C CURVE, W/O = 4.5, POISON LOADING = 0.02 $GM\text{-}B^{10}/CM^2$

FOR POISON LOADING CURVE, W/O = 4.5, C-C = 10.6"

Figure 3-4

K_{EFF} AS A FUNCTION OF C-C SPACING, POISON LOADING, AND ENRICHMENT FOR FRESH WESTINGHOUSE 15 X 15 OFA FUEL IN REGION I STORAGE RACK

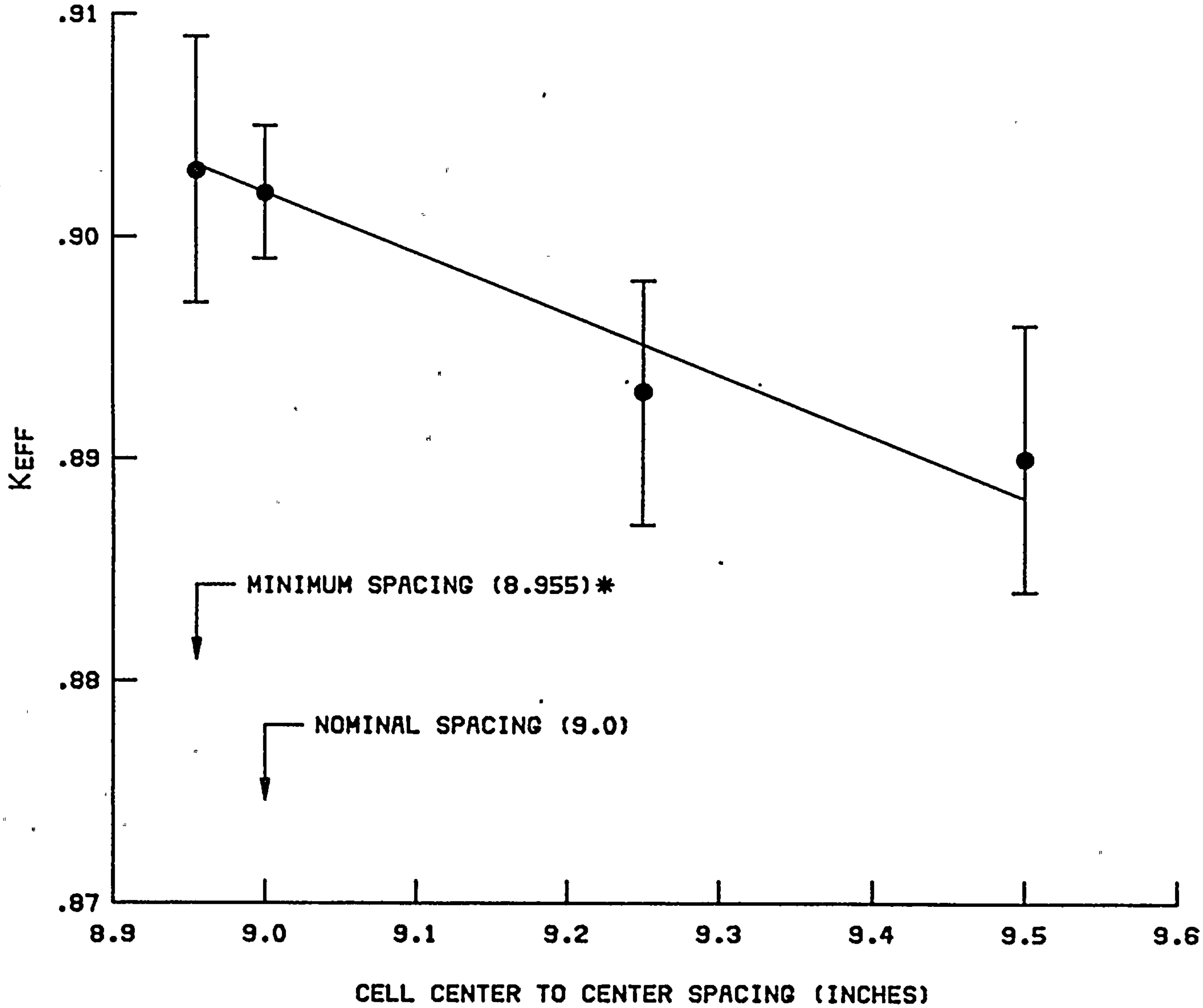


NOTES: FOR ENRICHMENT CURVE, B-10 LOADING = .012
FOR POISON LOADING CURVE, ENRICHMENT = 1.5
FOR BOTH CURVES, THE CENTER TO CENTER SPACING
OF FUEL ASSEMBLIES = 9.0 INCHES

Figure 3-5

K_{EFF} AS A FUNCTION OF FUEL ENRICHMENT AND POISON LOADING FOR WESTINGHOUSE 15 X 15 OFA FUEL IN REGION II SPENT FUEL STORAGE RACK

Rev. 0



NOTES: FUEL ENRICHMENT = 1.5 W/O U^{235}
POISON LOADING = .012 GM-B^{10}/CM^2
* DUE TO "CHECKERBOARD" DESIGN MINIMUM
CELL SPACING = 8.955 INCHES

Figure 3-6

K_{EFF} AS A FUNCTION OF STORAGE CELL CENTER-TO-CENTER SPACING FOR WESTINGHOUSE 15 X 15 OFA FUEL IN REGION II SPENT FUEL STORAGE RACK

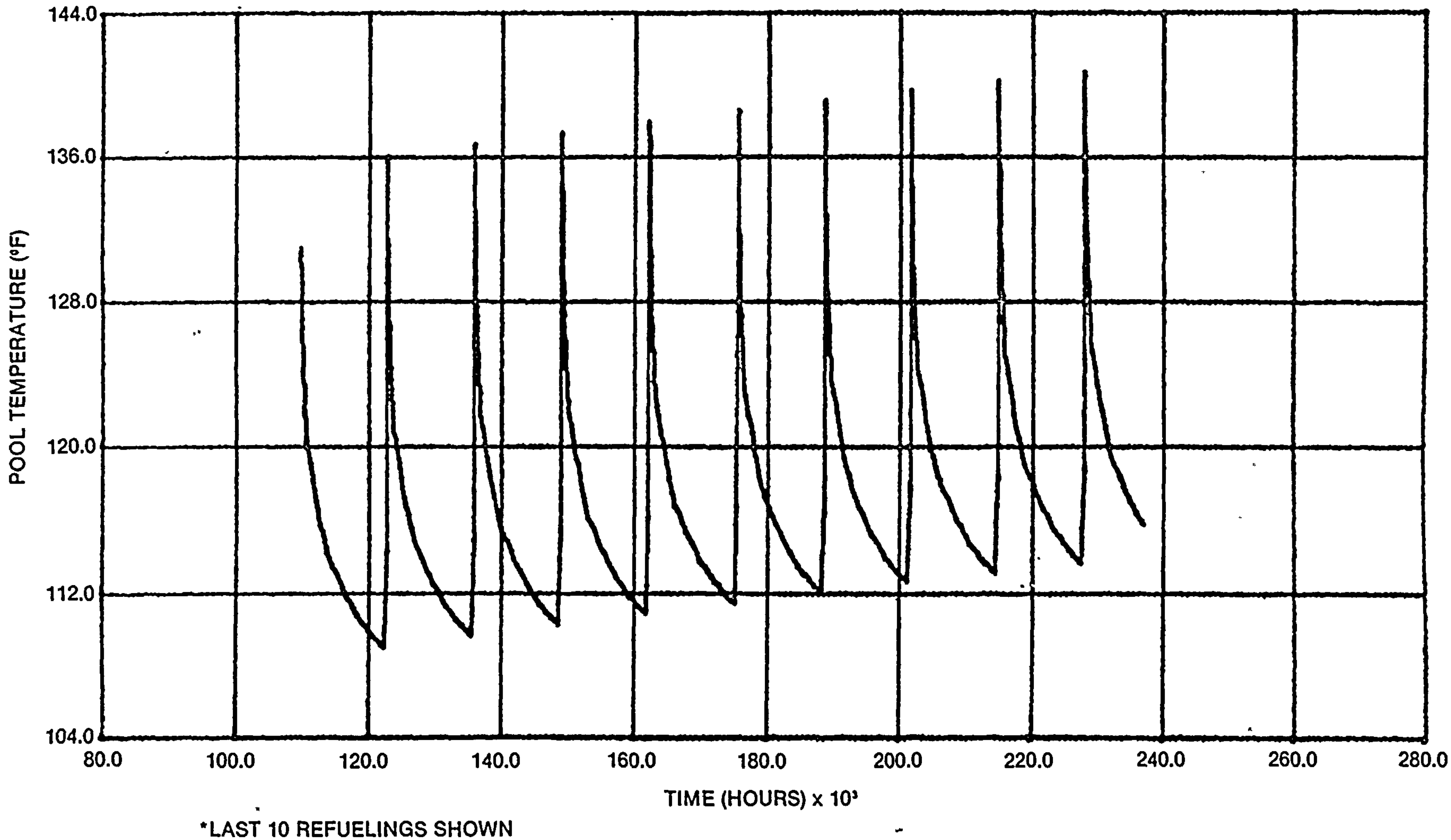


Figure 3-7

TURKEY POINT SFP HEATUP — NORMAL FUEL CYCLE*

Rev. 0

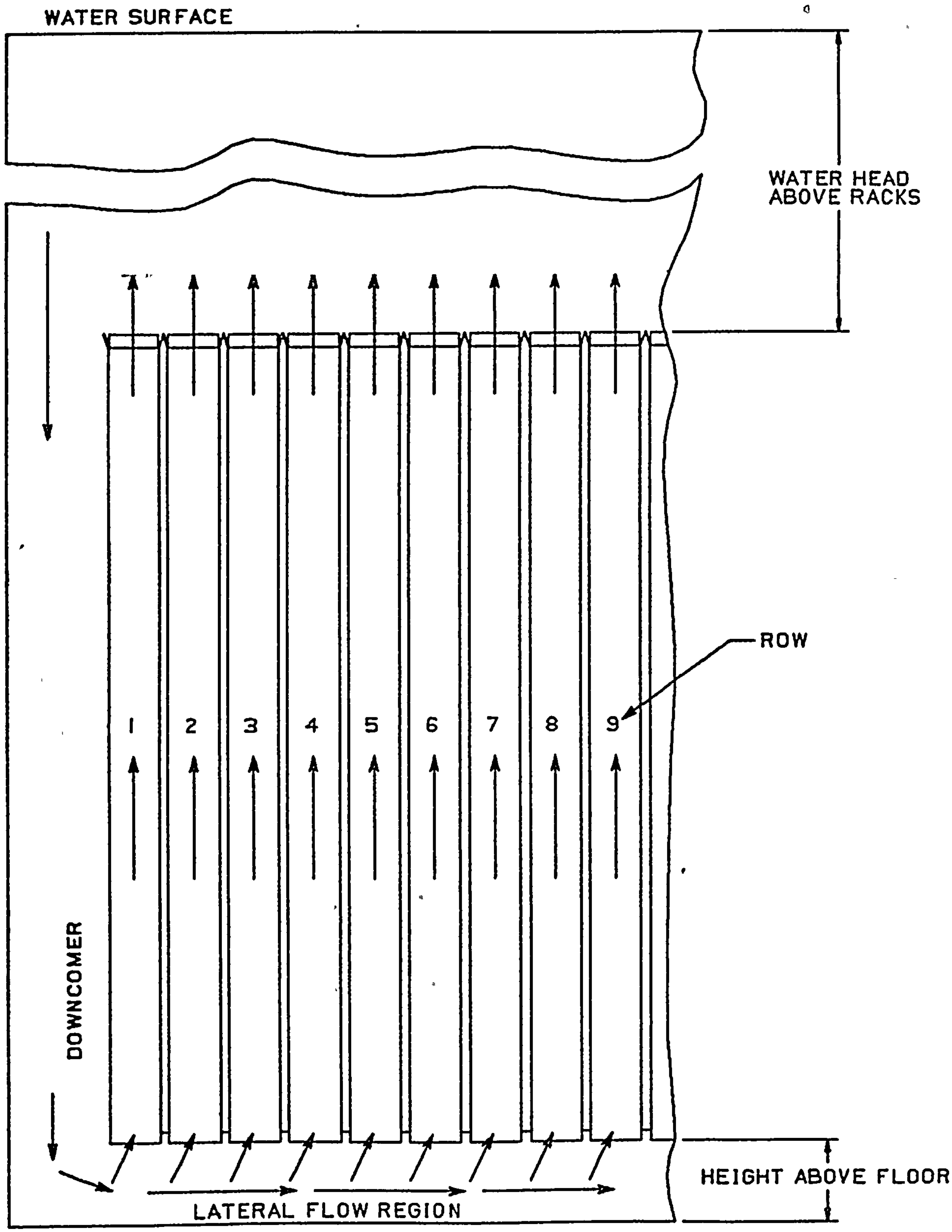


Figure 3-8

SPENT FUEL POOL NATURAL CIRCULATION MODEL
(ELEVATION VIEW)

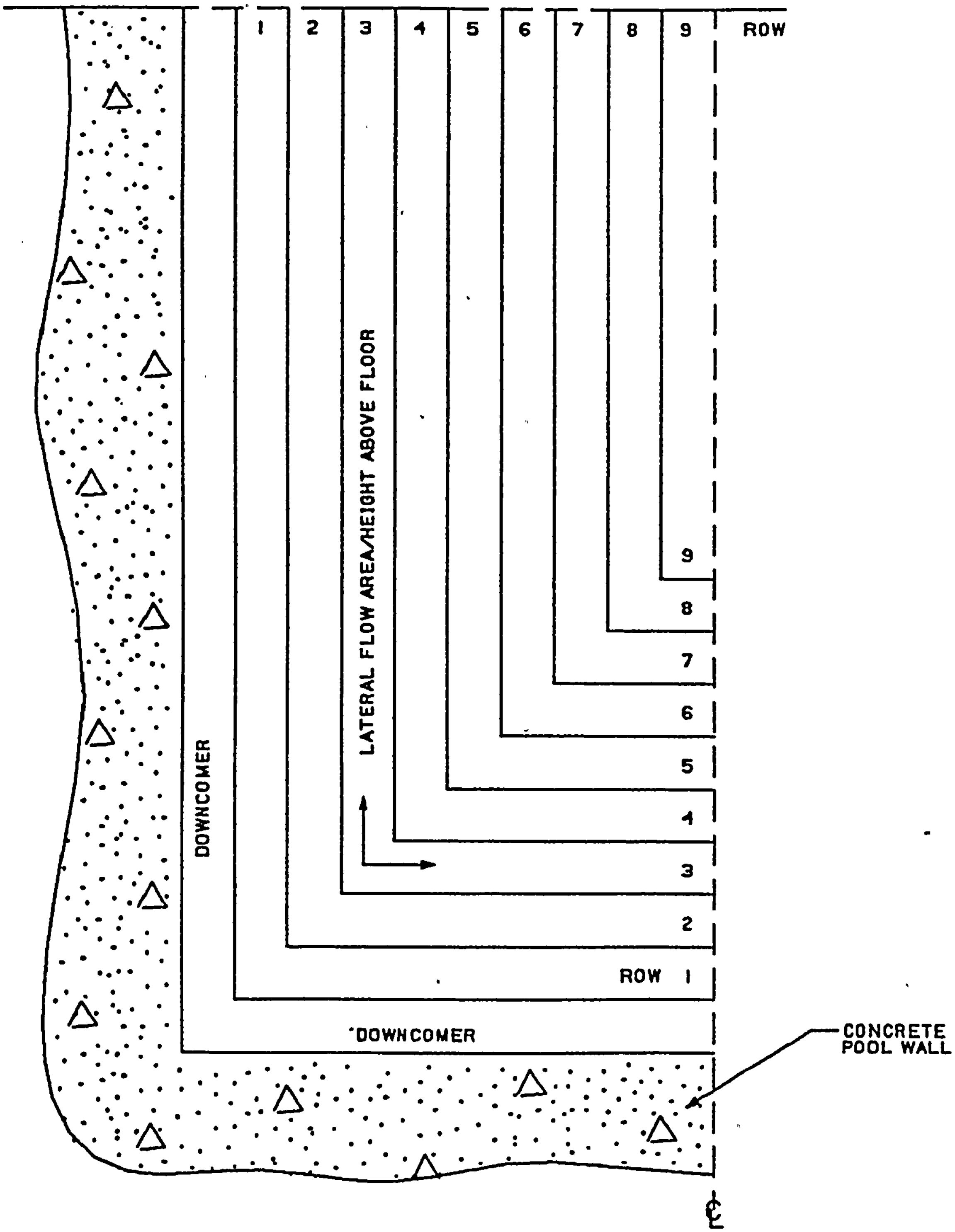


Figure 3-9

SPENT FUEL POOL NATURAL CIRCULATION MODEL
(PLAN VIEW)

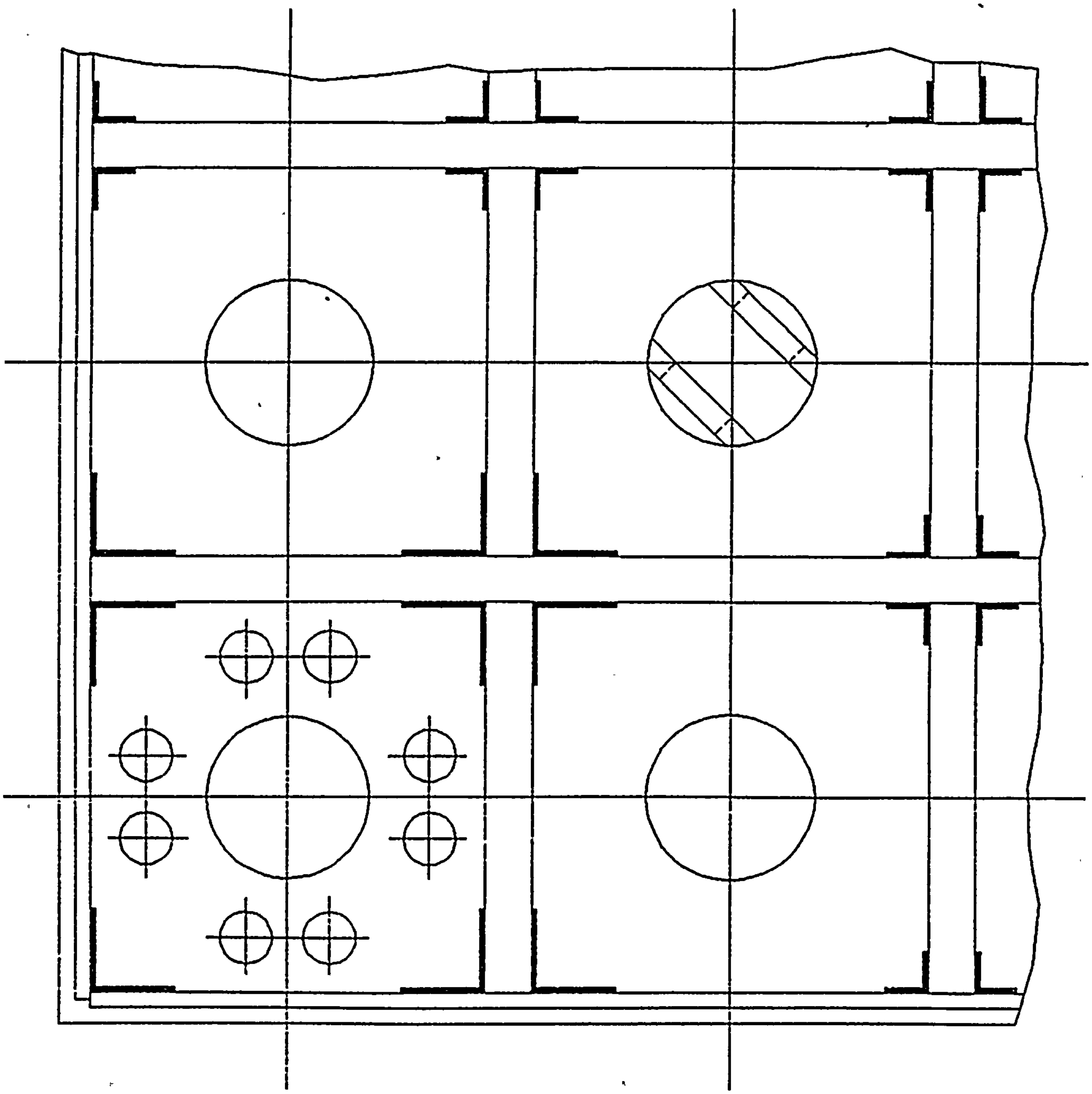


Figure 3-10
SPENT FUEL RACK INLET FLOW AREA TYPES
REGION I (REGION II SIMILAR)

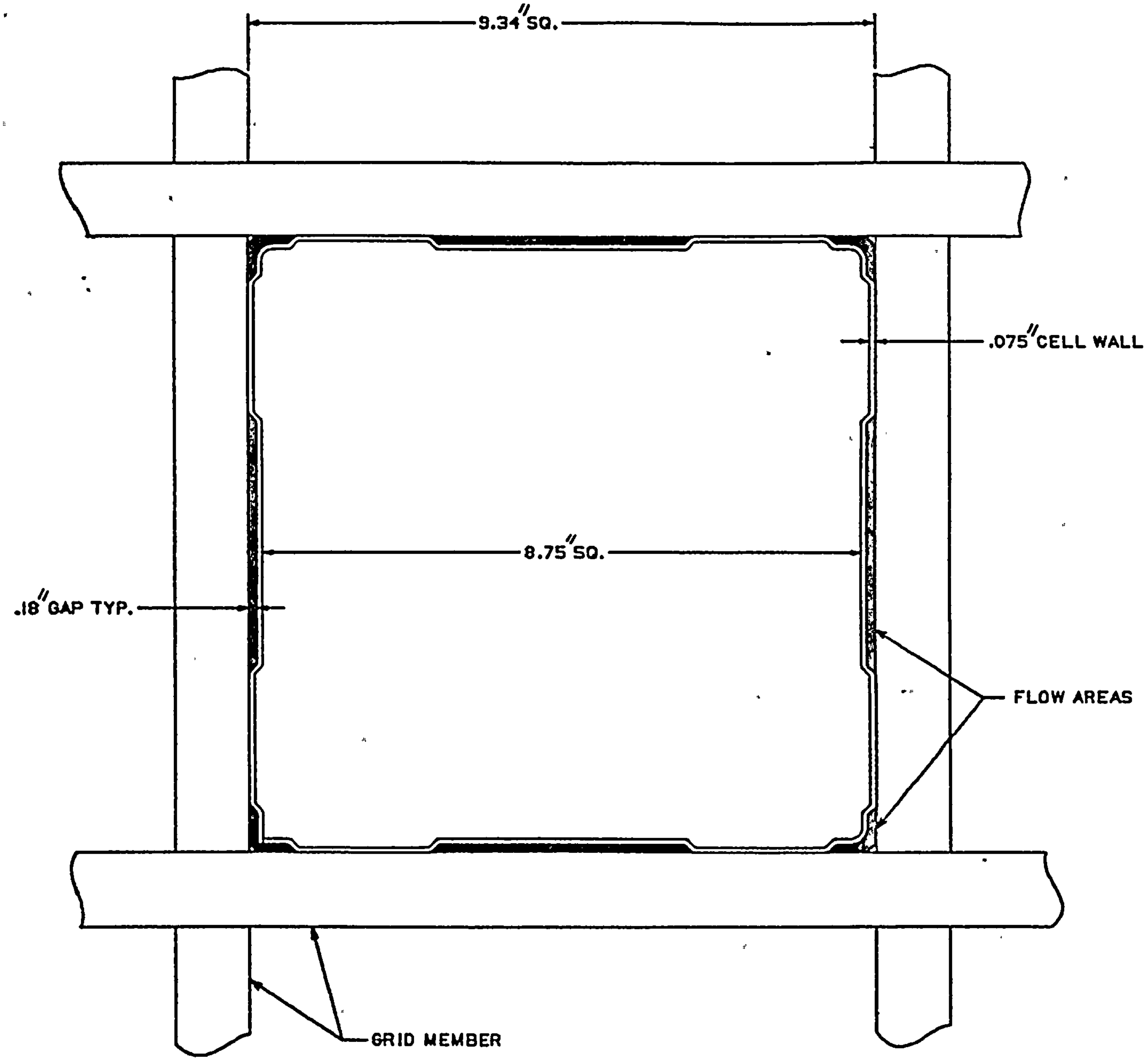


Figure 3-11

INTERCELL FLOW AREA
REGION I

4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 Description of Spent Fuel Building

A description of the spent fuel storage pool is provided in Sections 5.2.3 and 5.2.4 of the Turkey Point Updated FSAR[1]. The spent fuel pool structure and building have been designed in accordance with the criteria outlined in Section 5.2.2 and Appendix 5A of the Updated FSAR as accepted in Section 5.0 of the AEC (NRC) Final Safety Evaluation Report [2].

The walls and floor of the spent fuel pit are lined with a 1/4-inch-thick stainless steel liner. Monitoring trenches are provided behind the liner for detecting and collecting any leakage. Any leakage is directed to the waste disposal drainage system, thus preventing uncontrolled leakage of fuel pool water. Figures 4-1 through 4-3 show the existing configuration of the spent fuel pool.

The spent fuel pool is being evaluated structurally for the additional loading due to the increased number of fuel elements and modified rack design in accordance with Turkey Point Updated FSAR Appendix 5A, Standard Review Plan (SRP) Section 3.8.4[3], and 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[4].

The new spent fuel storage racks will be designed so that the floor loading from racks filled with spent fuel assemblies will not exceed the structural capacity of the spent fuel building. The bearing pressure is being reviewed to ensure that the pressure is within the allowable bearing capacity of the soil and rock underneath. Analysis of the spent fuel pool mat will include the effects of a 25-ton cask drop as described in Turkey Point Updated FSAR Appendix 14E. Critical areas of the structure, such as the junction between the wall and the floor, are being checked for the additional loading.

4.1.2 Description of Spent Fuel Racks

The function of the spent fuel storage racks is to provide for storage of 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 given below:

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) and SRP Section 3.8.4 [3].

2. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor, k_{eff}, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions.

3. The racks are designed to allow coolant flow such that boiling in the water channels between the fuel assemblies in the rack does not occur. Maximum fuel cladding temperatures are calculated for various pool cooling conditions as described in Section 3.3.

4. The racks are designed to Seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support Structures. The structural evaluation and seismic analyses are performed using the specified loads and load combinations in Section 4.4.

5. The racks are designed to withstand loads without violating the criticality acceptance criteria which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane.

6. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.

7. The racks are designed to preclude the insertion of a fuel assembly in other than design locations within the rack array. For Region I racks, this is accomplished by the close spacing of storage cells (less than 2 inches) and the upper grid structure. For Region II racks, there is no space between storage locations since the cells are welded to each other. Therefore, a fuel assembly can only be inserted in designated storage locations.

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

4.1.2.1 Design of Spent Fuel Racks

The spent fuel storage pool rack arrangement is shown in Figure 4-4. Fuel storage is divided into two regions within each

pool. Region I (286 locations) consists of high density fuel assembly spacing obtained by utilizing a neutron absorbing material and is normally used for core off-loading. Region II (1118 locations) also consists of high density fuel assembly spacing and provides normal storage for spent fuel assemblies meeting required burnup considerations. Region I is designed to accommodate nonirradiated, fully enriched fuel. Region II is designed to accommodate irradiated fuel. Normal placement of fuel in Region II is determined by burnup calculations and is controlled administratively. No physical barrier is necessary between the two regions. However, as described in Section 4.7.4, during rack installation it will be necessary to temporarily store Region I fuel without the necessary burnup in the Region II racks utilizing a checkerboard storage configuration.

The 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 modified January 18, 1979, with the exception that, for normal Region II storage, credit is taken for fuel burnup based on the proposed Revision 2 of USNRC Regulatory Guide 1.13[5].

Rack module data is presented in Table 4-1.

4.1.2.1.1 Region I Design

The Region I storage racks are composed of individual storage cells made of stainless steel. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. The cells within a module are interconnected by grid assemblies to form an integral structure as shown in Figure 4-5. Each rack module is provided with leveling screws which contact the spent fuel pool floor embedments and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The fuel rack assembly consists of three major sections which are the leveling screw, the lower and upper grid assemblies, and the cell assembly. Figure 4-6 illustrates these sections. The tops of the support plates are welded to the fuel rack base plate. The leveling screws transmit the loads to the pool floor embedments, provide a sliding contact, and provide for the leveling adjustment of the rack.

The lower grid consists of box-beam members, side plates and the base plate. The bottom of the cell assembly is welded to the lower grid. The upper grid consists of box-beam members and side plates. The upper part of the cell assembly is welded to the upper grid. The upper and lower grid assemblies maintain the 10.6-inch centerline-to-centerline spacing between the cells and provide the structural connections between the cells to form a fuel rack assembly.

The major components of the cell assembly are the fuel assembly cell, the Boraflex (neutron absorbing) material, and the wrapper.

The wrapper is attached to the outside of the cell by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of the Boraflex to the pool environment. Depending on the criticality requirements and location within the rack array, some cells have a Boraflex/wrapper assembly on four sides, three sides or two sides, as required by the analysis.

4.1.2.1.2 Region II Design

The Region II storage racks consist of stainless steel cells assembled in a checkerboard pattern with a 9.0-inch centerline-to-centerline spacing, producing a honeycomb type structure as shown in Figure 4-7. Each cell is of the same basic design as described for Region I; i.e., the major components are the cell, the Boraflex (neutron absorbing) material, and the wrapper. The cells are welded to a base support assembly and to one another to form an integral structure without use of grids as used in Region I racks. This design is also provided with leveling screws which contact the spent fuel pool floor embedments and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The fuel rack assembly consists of two major sections which are the base support assembly and the cell assembly. Figures 4-8 and 4-9 illustrate these sections.

The major components of the base support assembly are the leveling screw, the support plate, and the base plate. The top of the support plate is welded to the fuel rack base plate. The leveling screws transmit the loads to the pool floor embedments, provide a sliding contact, and permit the leveling adjustment of the rack.

The wrapper is attached to the outside of the cell by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of the Boraflex to the pool environment. Depending on the criticality requirements and location within the rack array, some cells have a Boraflex/wrapper assembly on four sides, three sides, or two sides, as required by the analysis.

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 accidents as presented in Turkey Point Updated FSAR Section 14.2.1 or the AEC (NRC) Final Safety Evaluation Report, Section 10.3. The spent fuel storage

racks are being designed to withstand the design basis fuel handling accident. The resulting criticality and radiological consequences of a postulated fuel assembly drop are addressed in Sections 4.6.4 and 5.3.1.1, respectively.

4.2 APPLICABLE CODES, STANDARDS, AND SPECIFICATIONS

The racks are being designed and fabricated to applicable portions of the following NRC Regulatory Guides, Standard Review Plan Sections, and published standards.

a. April 14, 1978 NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, as amended by the NRC letter dated January 18, 1979.

b. Turkey Point Plant Units 3 and 4, Updated Final Safety Analysis Report, Docket Nos. 50-250 and 50-251.

c. NRC Regulatory Guides

1.13, Rev. 2 Dec. 1981 (Draft)	Spent Fuel Storage Facility Design Basis
1.25, March 1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors
1.26, Rev. 3 Feb. 1976	Quality Group Classifications and Standards for Water Steam and Radio-active Waste Containing Components of Nuclear Power Plants
1.29, Rev. 3, Sept. 1978	Seismic Design Classification
1.92, Rev. 1 Feb. 1976	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.124, Rev. 1, Jan. 1978	Service Limits and Load Combinations for Class 1 Linear - Type Component Supports

d. Standard Review Plan - NUREG-0800

Rev. 1, July 1981	Section 3.7, Seismic Design
Rev. 1, July 1981	Section 3.8.4, Other Seismic Category I Structures
Rev. 3, July 1981	Section 9.1.2, Spent Fuel Storage

Rev. 1, July 1981	Section 9.1.3, Spent Fuel Pool Cooling System
NRC Branch Technical Position Rev. 2, July 1981	ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling

e. Industry Codes and Standards

ANSI N16.1-75	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
ANSI N16.9-75	Validation of Calculational Methods for Nuclear Criticality Safety
ANSI N210-76	Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations
ASME Section III-80 (through Summer 1982 Addendum)	Nuclear Power Plant Components
ACI 318-63	Building Code Requirements for Reinforced Concrete

4.3 SEISMIC AND IMPACT LOADS

The new spent fuel racks are being designed, and the spent fuel pool structure reevaluated, using the seismic loading described in this section.

Earthquake loading is predicated upon an operating basis earthquake (OBE) at the site having a horizontal ground acceleration of 0.05 g. In addition, a safe shutdown earthquake (SSE), having a horizontal ground acceleration of 0.15 g, is used to check the design to ensure no loss of function.

Horizontal response spectra applicable for the spent fuel pool structure are provided in Turkey Point Updated FSAR Appendix 5A and are applicable for both orthogonal horizontal directions. The vertical component of acceleration is taken as two-thirds of the horizontal ground acceleration in accordance with Updated FSAR Appendix 5A.

Seismic analysis of the fuel storage racks is being performed by the time history method and the response spectrum method. Where time histories are used, the three orthogonal time histories are statistically independent. The time histories and response spectrum utilized in these analyses represent the responses of the pool structure to the specified ground motion. The seismic analysis of the racks is being performed with a damping value of 2 percent for both OBE and SSE.

Maximum dynamic forces and stresses are being calculated for the worst condition as determined by combination with forces and stresses computed in accordance with Section 4.4. The initial and subsequent installations of rack modules up to the full capacity are being considered in determining the worst possible condition.

The analysis includes the effects of the water in the pool, such as fluctuation of pressure due to acceleration, and sloshing.

Deflection or movements of racks under earthquake loading is limited by design such that the racks do not touch each other or the spent fuel pool walls, the racks are not damaged to the extent that nuclear parameters outlined in Section 3.1 are exceeded, and the fuel assemblies are not damaged.

The interaction between the fuel elements and the rack is being considered, particularly gap effects. The resulting impact loads are of such small magnitudes that there is no structural damage to the fuel assemblies.

4.4 LOADS AND LOAD COMBINATIONS

The analysis of the spent fuel pool for structural adequacy, when loaded with the new high density spent fuel racks, considers the loads and load combinations of Turkey Point Updated FSAR Appendix 5A.

The Table 4-2 loads and load combinations to be considered in the analysis of the spent fuel racks include those given in 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.

It is noted from the seismic analysis that the magnitude of stresses varies considerably from one geometrical location to the other in the model. Consequently, the maximum loaded major rack components will be analyzed. Such an analysis envelopes the other areas of the rack assembly.

The margins of safety for the multi-direction seismic event are produced by combining x-direction, y-direction, and z-direction loads by the square-root-of-the-sum-of-the-squares (SRSS) method.

The loads used in the structural analysis are loads from the linear seismic analysis multiplied by load correction factors to account for the nonlinear effects.

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 Design and Analysis Procedures for Spent Fuel Pool Structure

The spent fuel pool structure with the augmented storage capability is being analyzed by the three-dimensional finite element method. The finite element model, shown in Figure 4-10, includes the brick elements for the foundation mat and the walls and solid elements to represent the foundation support. The loads and loading combinations are the same as those listed in Turkey Point Updated FSAR Appendix 5A for Class I structures. The increased loading due to the additional spent fuel elements to be stored in the pool is included in the analysis.

4.5.2 Design and Analysis Procedures for Spent Fuel Storage Racks

The seismic and stress analysis of the spent fuel rack modules will consider the various conditions of full, partially filled, and empty fuel assembly loadings. The racks are being evaluated for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) conditions and meet Seismic Category I requirements. A detailed stress analysis is being performed to verify the acceptability of the critical load components and paths under normal and faulted conditions. The racks rest freely on the pool floor embedments and are being evaluated to determine that under all loading conditions they do not impact each other nor do they impact the pool walls.

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces governing loads and stresses on the structure. The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. These models, one for Region I and one for Region II, are shown in Figure 4-11. The second phase is a response spectrum analysis of a detail rack assembly finite element model. These models, one for Region I and one for Region II, are shown in Figure 4-12. When modes with closely spaced (within 10 percent of each other) modal frequencies exist, the system response is obtained by taking the absolute sum of the responses of the closely spaced modes and combining this sum with the other remaining modal responses using the SRSS rule. The damping values used in the seismic analysis are 2 percent damping for OBE and SSE.

The simplified nonlinear finite element model is used to determine the fuel rack response. This nonlinear model has the structural characteristics of an individual cell within a submerged rack assembly. The nonlinearities of the fuel rack assembly which are accounted for in the model are due to

changes in the gap between the fuel cell and the fuel assembly, the boundary conditions of the fuel rack support locations and energy losses at the support locations.

The fuel assembly to cell impact loads, leveling screw lift off, rack sliding, and overall rack response are obtained from the nonlinear time history model. In determining the maximum fuel rack response, the response value for each item of interest is searched for maximum values.

The detail seismic model is a three-dimensional finite element representation of a rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of nodal points.

The results of the single cell nonlinear time history model will be incorporated in the detail model. Since the detail model does not account for the nonlinear effect of a fuel assembly impacting the cell or the leveling screw movements, the internal loads and stresses for the rack assembly obtained from this model are corrected by load correction factors. The load correction factors are derived from the single cell nonlinear model results and will be applied to the rack component loads in the structural analysis. The responses (loads and displacements) of the model from accelerations in three directions are combined by the SRSS method in the structural analysis. The loads in the major components are examined, and the maximum loaded section of each of these components is found. These maximum loads from the detail model are used in the structural analysis to obtain the stresses within the rack assembly.

The hydrodynamic mass of the submerged racks is calculated using the specific rack and pool geometry. This mass is used in the seismic analysis. The sloshing movement of the water is in the upper elevations of the pool above the top of the racks. Thus, no sloshing loads are imposed on the rack structure.

4.6 STRUCTURAL ACCEPTANCE CRITERIA

4.6.1 Structural Acceptance Criteria for Spent Fuel Pool Structure

The spent fuel pool structure was designed for ductile behavior (i.e., with reinforcing steel stresses controlling the design). The original acceptance criteria are stated in Updated FSAR Appendix 5A. These criteria apply in the structural reanalysis. Acceptance is based on maintaining structural integrity and ductile behavior of the pool structure. The structural components which define the pool structure used here include the pool walls and mat and the supporting soil beneath the mat. Stresses in concrete and reinforcing steel components required to maintain structural continuity will be within the allowables

calculated using the loading combinations previously described and the ultimate strength design portion of the ACI 318-63 code. The bearing pressures on the soil supporting the pool base mat and rock below the soil will remain within the allowable bearing pressures.

4.6.2 Structural Acceptance Criteria for Spent Fuel Storage Racks

The fuel racks will be analyzed for the normal and faulted load combinations of Section 4.4 in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The major normal and upset condition loads are produced by the operating basis earthquakes (OBE). The thermal stresses due to rack relative expansion will be calculated and combined with the appropriate seismic loads in accordance with the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" [4], (with clarifications as noted in Table 4-2).

The faulted condition loads are produced by the safe shutdown earthquakes (SSE) and a postulated fuel assembly drop accident.

The computed stresses will be within the acceptance limits identified in the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" [4], (with clarifications as noted in Table 4-2).

In summary, the results of the seismic and structural analysis show that the Turkey Point spent fuel storage racks meet all the structural acceptance criteria adequately.

4.6.3 Fuel Handling Crane Uplift Analysis

An analysis will be performed to demonstrate that the rack can withstand a maximum uplift load of 4,000 pounds. This load, approximately two times the capacity of the fuel handling crane, can be applied to any point of the fuel rack 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.

4.6.4 Fuel Assembly Drop Accident Analysis

In the unlikely event of dropping a fuel assembly, accidental deformation of the rack will not cause the criticality acceptance criterion to be violated.

For the analysis of a dropped fuel assembly, three accident conditions are postulated. The first accident condition conservatively assumes that the weight of a fuel assembly, control rod assembly and handling mechanism of 3,000 pounds

impacts the top end fitting of a stored fuel assembly from a drop height of 3 feet. Calculations will show that the impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cells and rack base plate assembly. If in the unlikely event that two adjacent cells are crushed together for their full length, criticality calculations show that $k_{eff} \leq 0.95$. Under these faulted conditions, credit is taken for dissolved boron in the water, and the criticality acceptance criterion is not violated.

The second accident condition is an inclined drop on top of the rack. Results will be the same as for the first condition.

The third accident condition assumes that the dropped assembly (3,000 lbs) falls straight through an empty cell and impacts the rack base plate from a drop height of 201 inches. The results of this analysis will show that the impact energy is absorbed by the fuel assembly and the rack base plate. Criticality calculations show that $k_{eff} \leq 0.95$ and the criticality acceptance criterion is not violated.

4.6.5 <u>Fuel Rack Sliding and Overturning Analysis</u>

Consistent with the criteria of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," the racks will be evaluated for overturning and sliding displacement due to earthquake conditions under the various conditions of full, partially filled, and empty fuel assembly loadings.

The nonlinear model described in Section 4.5 is used in this evaluation to account for fuel-to-rack impact loading, hydrodynamic forces, and the nonlinearity of sliding friction interfaces.

The horizontal resistive force at the interface between the rack module and pool floor embed is produced by friction. A range of friction coefficients ($\mu = 0.2$ to 0.8) are used in the analysis. A low coefficient of friction ($\mu = 0.2$) produces maximum rack horizontal displacement or sliding while a high value ($\mu = 0.8$) produces maximum rack horizontal overturning force.

The fuel rack nonlinear time-history analysis will show that the fuel rack slides a minimal distance. This distance combined with the rack structural deflection and thermal growth is less than the rack-to-rack, rack-to-wall or rack-to-embedment edge clearances. Thus, impact between adjacent rack modules or between a rack module and the pool wall is prevented and the leveling screws will not slide off the embedment plates. The factor of safety against overturning will be well within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

4.7 MATERIALS, QUALITY CONTROL, AND SPECIAL CONSTRUCTION TECHNIQUES

4.7.1 Construction Materials

Construction materials will conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All the materials used in the construction are compatible with the storage pool environment and will not contaminate the fuel assemblies or the pool water. The plates, sheets, strips, bars and structural shapes used for rack construction are Type 304L stainless steel.

4.7.2 Neutron Absorbing Material

The neutron absorbing material, Boraflex, used in the Turkey Point spent fuel rack construction is manufactured by Brand Industrial Services, Inc., and fabricated to the safety-related nuclear criteria of 10 CFR 50, Appendix B. Boraflex is a silicone-based polymer containing fine particles of boron carbide in a homogeneous, stable matrix. Boraflex contains a minimum B^{10} areal density of 0.020 gm/cm^2 for Region I racks and 0.012 gm/cm^2 for Region II racks.

Boraflex has undergone extensive testing to study the effects of gamma and neutron irradiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material[6]. Tests were performed at the University of Michigan exposing Boraflex to 1.03×10^{11} rads gamma radiation with a substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities before and after being subjected to an environment of borated water and 1.03×10^{11} rads gamma radiation[7].

Long-term borated water soak tests at high temperatures were also conducted[8]. Boraflex maintains its functional performance characteristics and shows no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

During irradiation, a certain amount of gas may be generated. A conservative evaluation of the effect of gas generation on the spent fuel pool building atmosphere indicates that the maximum gas generation would be less than 0.01 percent of the total room volume. Additionally, the majority of gas generation is nitrogen, oxygen and CO_2, therefore no combustible hazard will exist.

The actual tests verify that Boraflex maintains long-term material stability and mechanical integrity and that it can be safely utilized as a poison material for neutron absorption in spent fuel storage racks.

4.7.3 Quality Assurance

The design, procurement, and fabrication of the new high density spent fuel storage racks comply with the pertinent Quality Assurance requirements of Appendix B to 10 CFR 50 as implemented through FPL's Topical Quality Assurance Report FPL-NQA-100A[9]; the Westinghouse Water Reactors Division Quality Assurance Plan as described in WCAP 8370[10]; and the Bechtel Quality Assurance Program for Nuclear Plants, BQ-TOP-1[11], all approved by the NRC.

4.7.4 Construction Techniques

4.7.4.1 Administrative Controls During Manufacturing and Installation

The Turkey Point Units 3 and 4 new spent fuel storage racks will be manufactured at the Westinghouse Nuclear Components Division, Pensacola, Florida. This facility is a modern high-quality shop with extensive experience in forming, machining, welding, and assembling nuclear-grade equipment. Forming and welding equipment are specifically designed for fuel rack fabrication and all welders are qualified in accordance with ASME Code Section IX.

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 by written procedures. These procedures 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 written procedures. 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.

During rack installation, it will be necessary to temporarily store some Region I fuel assemblies in the Region II spent fuel racks. Proposed Technical Specifications 3.17 and 5.4 (see Section 3.5) describe the administrative controls that will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks.

4.7.4.2 Procedure

4.7.4.2.1 Preinstallation

The following sequence of preinstallation events is anticipated for the spent fuel storage rack replacement for Units 3 and 4:

a. Design and fabricate new spent fuel storage racks.

b. Prepare modification procedure.

c. Fabricate and test all special tooling.

d. Receive and inspect new spent fuel storage racks.

4.7.4.2.2 Installation

The final configuration of the 12 new rack modules in the spent fuel pool is shown in Figure 4-4. The installation of these racks will be accomplished in accordance with the following considerations and guidelines:

- A temporary construction crane will be installed and operated in the spent fuel pool area to move new and existing rack modules within the spent fuel pool.
- At no time will this temporary crane carry a rack module directly over another module already installed in the spent fuel pool.
- A work platform will be installed in the cask laydown area in the spent fuel pool. This platform will be used as a staging area for the transfer of new and existing rack modules between the temporary construction crane and the fuel cask crane.
- The temporary construction crane, work platform, and all rack modules will be installed and/or removed from the spent fuel pool area with the fuel cask crane.
- All load handling operations in the spent fuel pool area will be conducted in accordance with the criteria of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"[12].
- Spent fuel relocations within the pool will be performed as required to maintain separation between the stored fuel and the rerack operations.

4.8 TESTING AND IN-SERVICE SURVEILLANCE

The neutron absorber rack design includes a poison verification view-hole in the cell wall 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 Westinghouse quality assurance program controls and the use of qualified Boraflex neutron absorbing material, satisfies an initial verification test to assure that the proper quantity and placement of material was achieved during fabrication of the racks. This precludes the necessity for on-site poison verification.

The poison coupons used in the surveillance program will be representative of the material used within Region I and Region II locations. They will be of the same composition, produced by the same method, and certified to the same criteria as the production lot poison. The sample coupons will be of a similar thickness as the poison used within the storage system. Each poison specimen will be encased in a stainless steel jacket of an identical alloy to that used in the storage system, formed so as to encase the poison material and fix it in a position similar to that designed into the storage system. The jacket will be mechanically closed without welding in such a manner as to retain its form throughout the use period yet allow rapid and easy opening without contributing mechanical damage to the poison specimen contained within.

A series of not less than 24 of the jacketed poison specimens shall be suspended from rigid straps so designed as to be hung on the outside periphery of a Region I or Region II rack module. There are 4 sets of these straps, two for each region. The specimens will be located in the spent fuel pool such that they will receive a representative exposure of gamma radiation. The specimen location will be adjacent to a designated storage cell with design ability to allow for removal of the strap, providing access to a particular specimen.

As discussed in Section 4.7.2, irradiation tests have been previously performed to test the stability and structural integrity of Boraflex in boric acid solution under irradiation [7]. These tests have concluded that there is no evidence of deterioration of the suitability of the Boraflex poison material through a cumulative irradiation in excess of 1×10^{11} rads gamma radiation. As more data on the service life performance of Boraflex becomes available in the nuclear industry in the coming years through both experimentation and operating experience. FPL will evaluate this information and will modify the surveillance program as determined warranted and justified.

FPL plans to perform an initial surveillance of the specimens after approximately five years of exposure in the pool environment. During this surveillance, several specimens will be removed from the pool and examined. This examination is expected to include visual inspection as well as other tests determined necessary to verify that the performance of the Boraflex is consistent with the reported test results. Based on the results of this initial surveillance, FPL will determine the scheduling and extent of additional surveillances so as to assure acceptable material performance throughout the life of the plant.

4.9 REFERENCES

1. Turkey Point Plant Units 3 and 4, Updated Final Safety Analysis Report, Docket Nos. 50-250 and 50-251.

2. Turkey Point Plant Units 3 and 4, Safety Evaluation Report, Docket Nos. 50-250 and 50-251.

3. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Revision 1, July 1981.

4. 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.

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

6. J. S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Data," Brand Industries, Inc., Report 748-30-2 (August 1981).

7. J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1 (August 1981).

8. J. S. Anderson, "A Final Report on the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Brand Industries, Inc., Report 748-21-1 (August 1978).

9. FPL-NQA-100A, FPL Topical Quality Assurance Report.

10. WCAP 8370, The Westinghouse Electric Corporation Quality Assurance Plan, Revision 9, Amendment 1, February 1978.

11. Bechtel Power Corporation, "Bechtel Quality Assurance Program for Nuclear Power Plants", Topical Report BQ-TOP-1, Revision 3A, October 1980.

12. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980.

TABLE 4-1

Rack Module Data

	Unit 3 or 4	
	Region I	Region II
Number of Storage Locations	286	1118
Number of Rack Arrays	2 (8x11) 1 (10x11)	1 (9x13 MOD) 3 (9x13) 3 (10x13) 1 (9x14) 1 (10x14)
Center-to-Center Spacing (Inches)	10.60	9.00
Cell I.D. (Inches	8.75	8.80
Type of Fuel	W 15x15 W 15x15 OPTIMIZED, CE 14x14, Exxon 14x14, CE 16x16	Same as Region I
Rack Assembly Dimensions (Inches)	(8x11) 84.75x116.63x168.38 (10x11) 106x116.63.168.38	(9x13 & 9x13 MOD) 81.63x117.63x168.38 (9x14) 81.63x126.63x168.36 (10x13) 90.63x117.63x168.39 (10x14) 90.63x126.63x168.38
Dry Weights (lbs) Per Rack Assembly	21,700 (8x11) 27,100 (10x11)	12,200(9x13 MOD) 13,000 (9x13) 14,300 (10x13) 13,900 (9x14) 15,400 (10x14)

TABLE 4-2

LOADS AND LOAD COMBINATIONS

Load Combination	Acceptance Limit
D + L	Normal limits of NF 3231.1a
D + L + P_f	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + T_o	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_o + E	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_a + E	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_o + P_f	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_a + E'	Faulted condition limits of NF 3231.1c (see Note 4)
D + L + F_d	The functional capability of the fuel racks shall be demonstrated

Notes:

1. The abbreviations in the table above are those used in SRP Section 3.8.4 where each term is defined except for T_a, which is defined here as the highest temperature associated with the postulated abnormal design conditions. F_d is the force caused by the accidental drop of the heaviest load from the maximum possible height, and P_f is the upward force on the racks caused by a postulated stuck fuel assembly.

2. The provisions of NF-3231.1 of ASME Section III, Division I, shall be amended by the requirements of Paragraphs c.2.3 and 4 of Regulatory Guide 1.124, entitled "Design Limits and Load Combinations for Class A Linear-Type Component Supports."

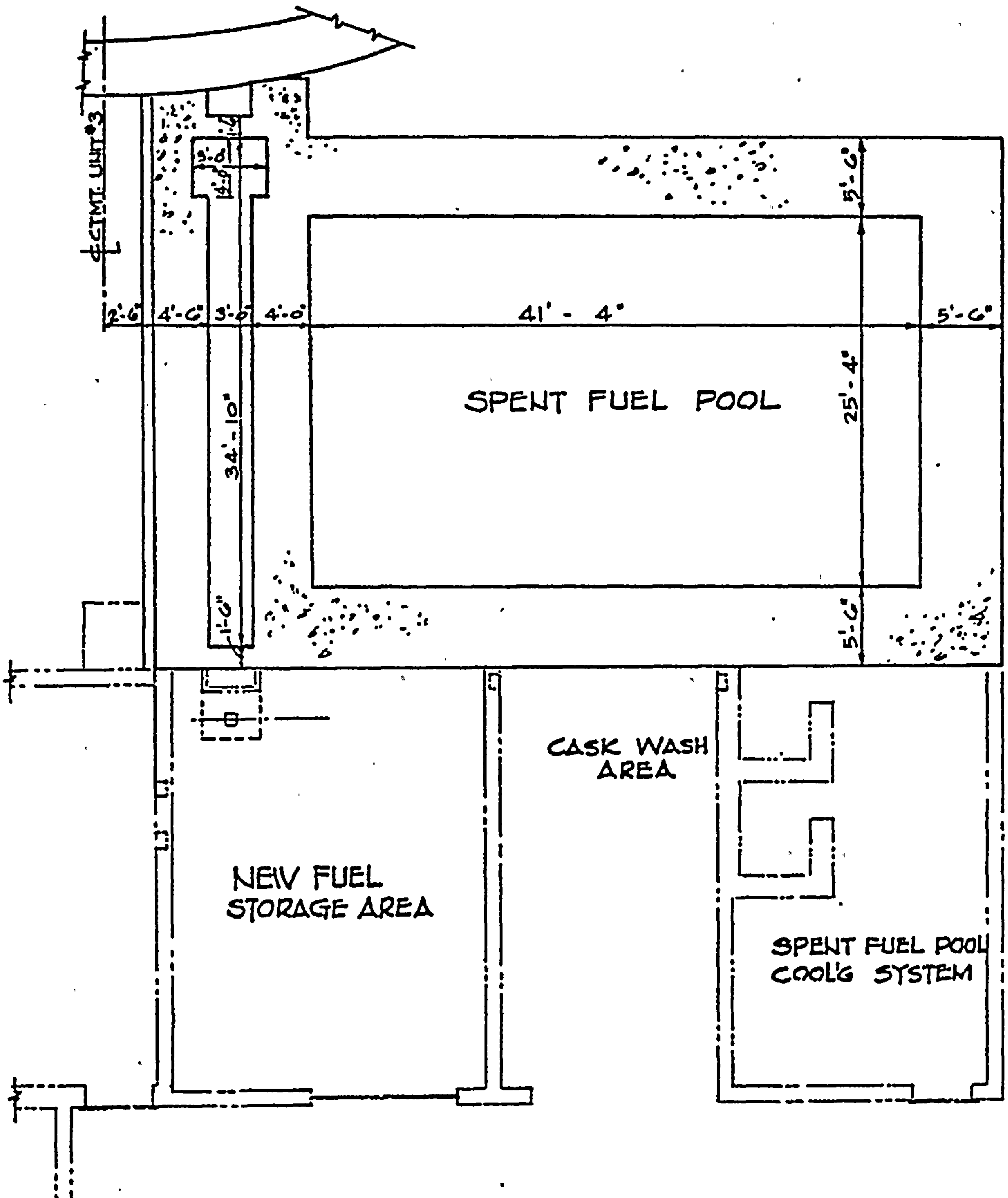
TABLE 4-2 (Continued)

Notes (Continued):

3. The application of this acceptance limit for the combination of primary and thermal stresses will typically limit the stresses to S_y. However, when proper justification is provided to show that the thermal stresses are self-limiting, the combined stresses may exceed S_y provided the lesser of 2 S_y or S_u stress range limit is met.

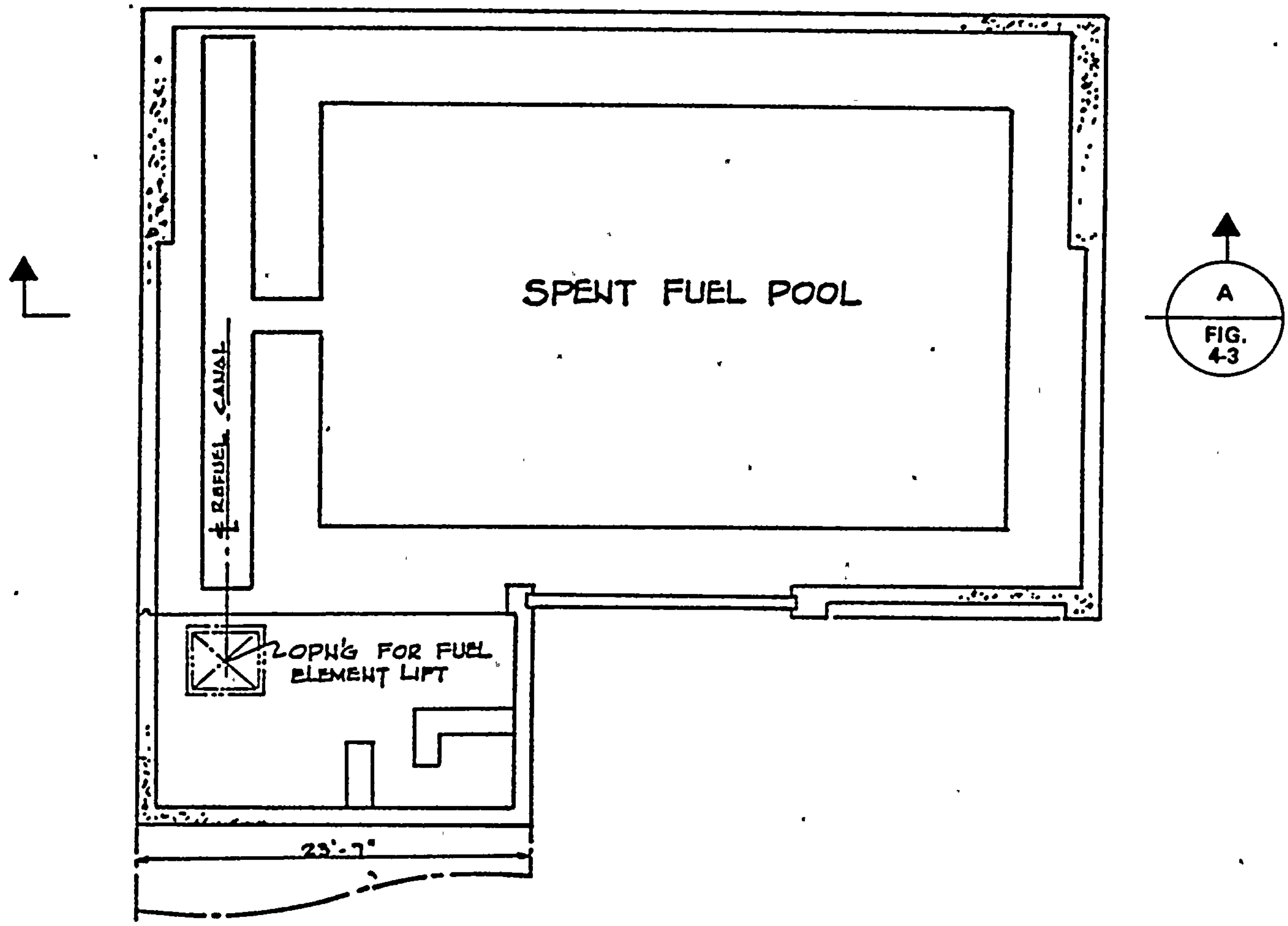
4. For the faulted load combination, thermal loads will be neglected when they are secondary and self-limiting in nature and the material is ductile.

FIGURE 4-1
SPENT FUEL POOL PLAN AT ELEVATION 18'-0"



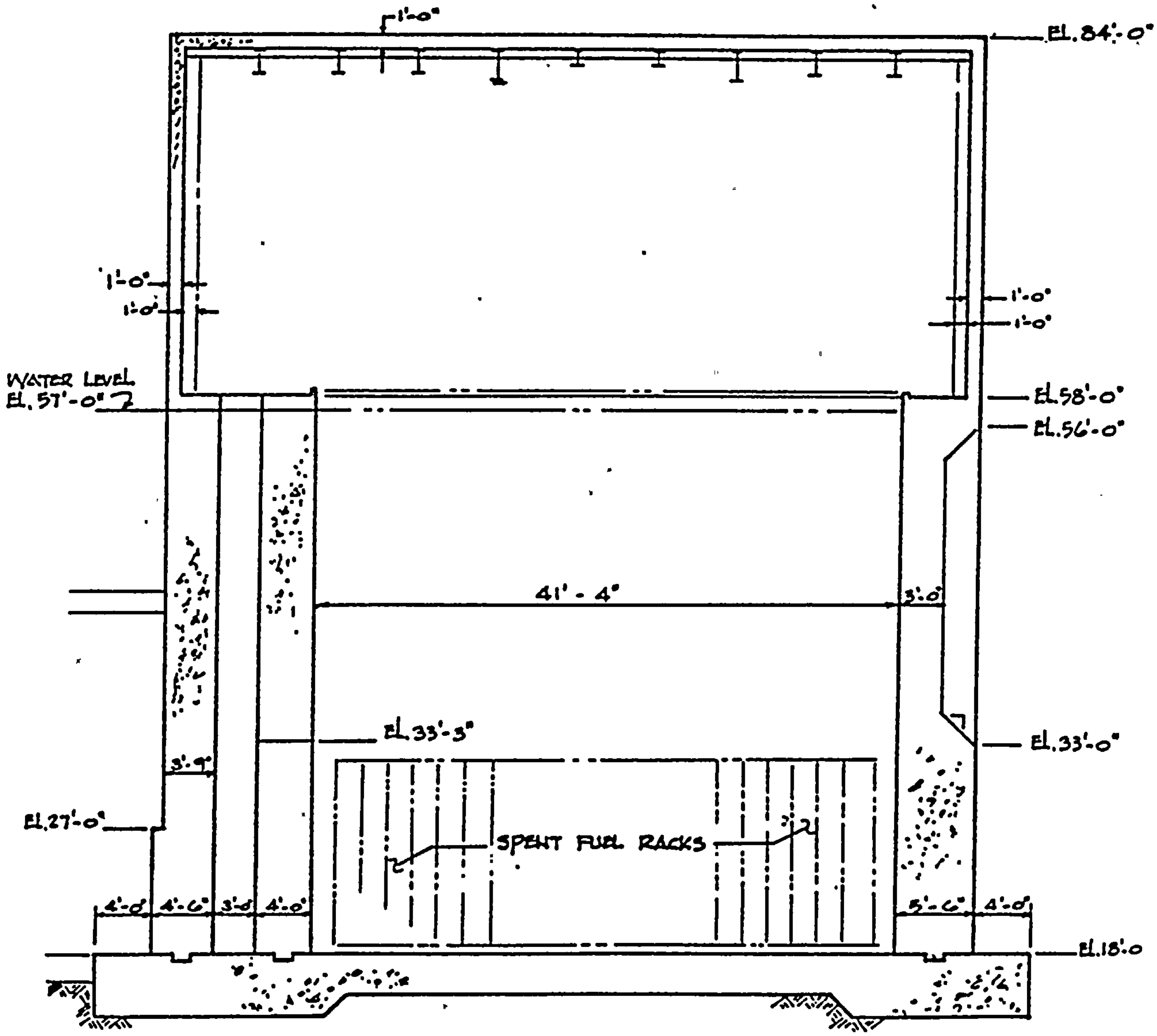
UNIT 3 — AS SHOWN
UNIT 4 — OPPOSITE HAND

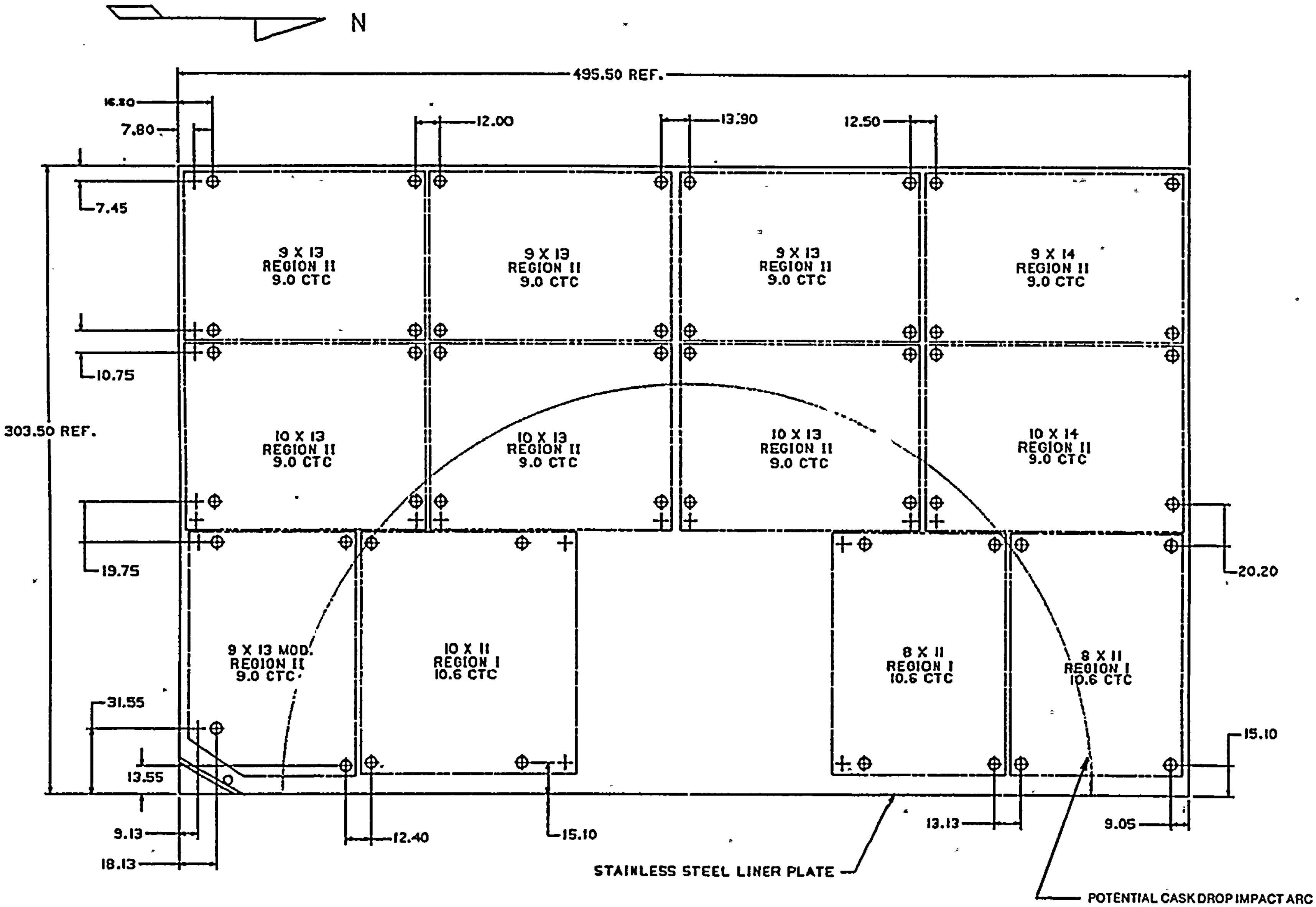
FIGURE 4-2
SPENT FUEL POOL PLAN AT ELEVATION 58'-0"



UNIT 3 — AS SHOWN
UNIT 4 — OPPOSITE HAND

FIGURE 4-3
SECTION A-A OF SPENT FUEL POOL





UNIT 3 — AS SHOWN
UNIT 4 — OPPOSITE HAND

ALL DIMENSIONS IN INCHES

Figure 4-4

SPENT FUEL POOL ARRANGEMENT

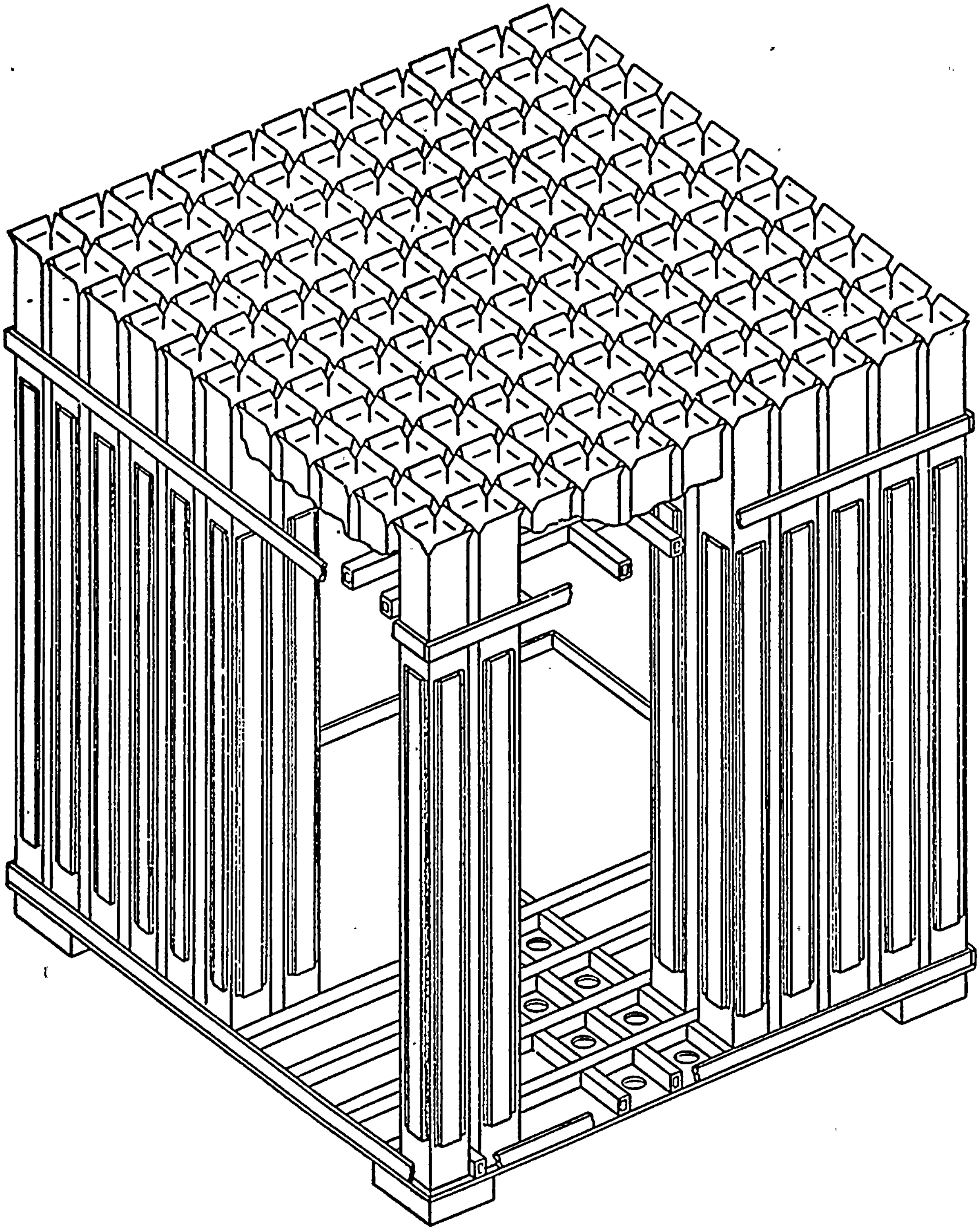


Figure 4-5

REGION I
FUEL STORAGE RACK MODULE

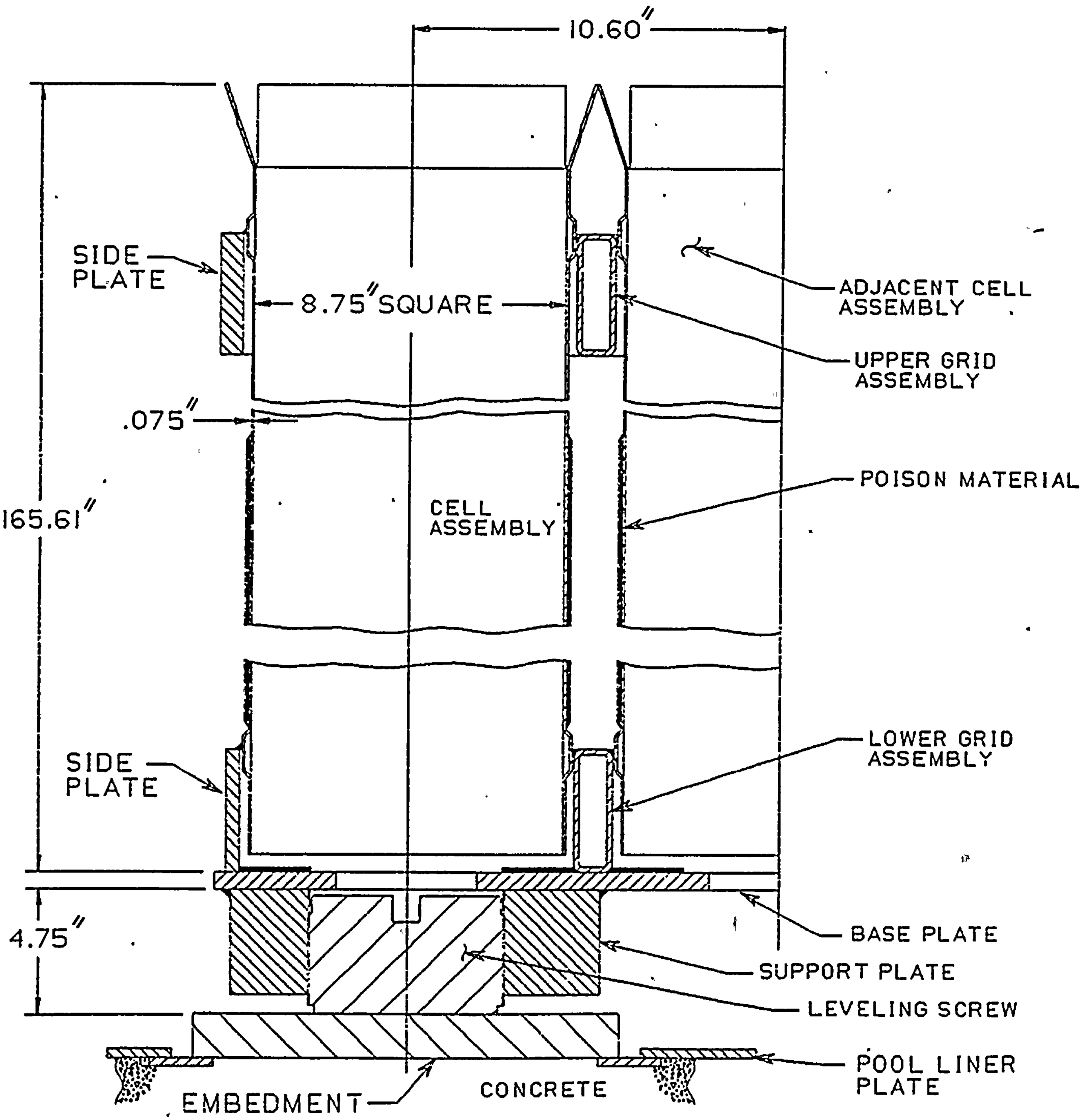


Figure 4-6

REGION I
MODULE CROSS-SECTION

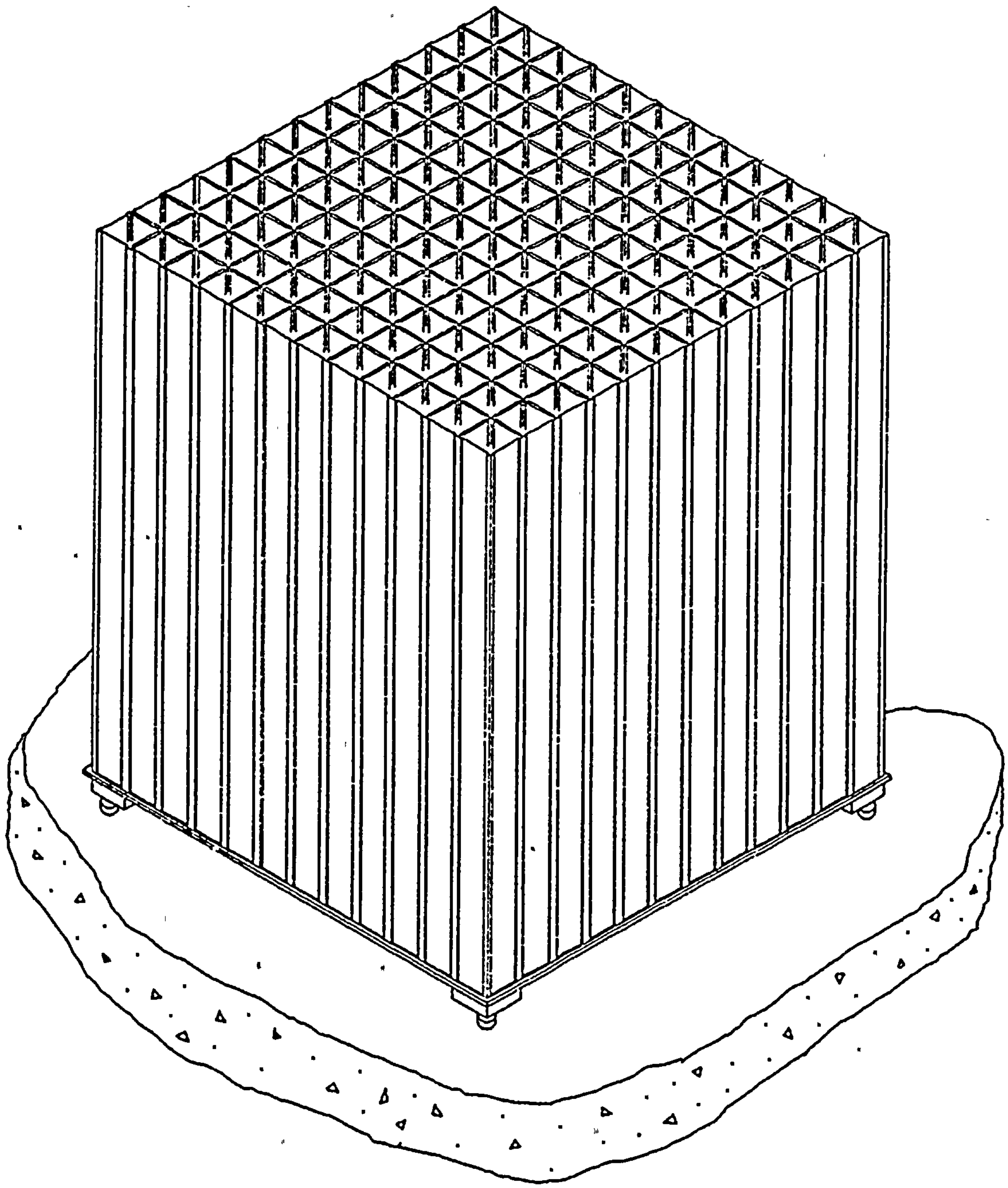


Figure 4-7

REGION II
FUEL STORAGE RACK MODULE

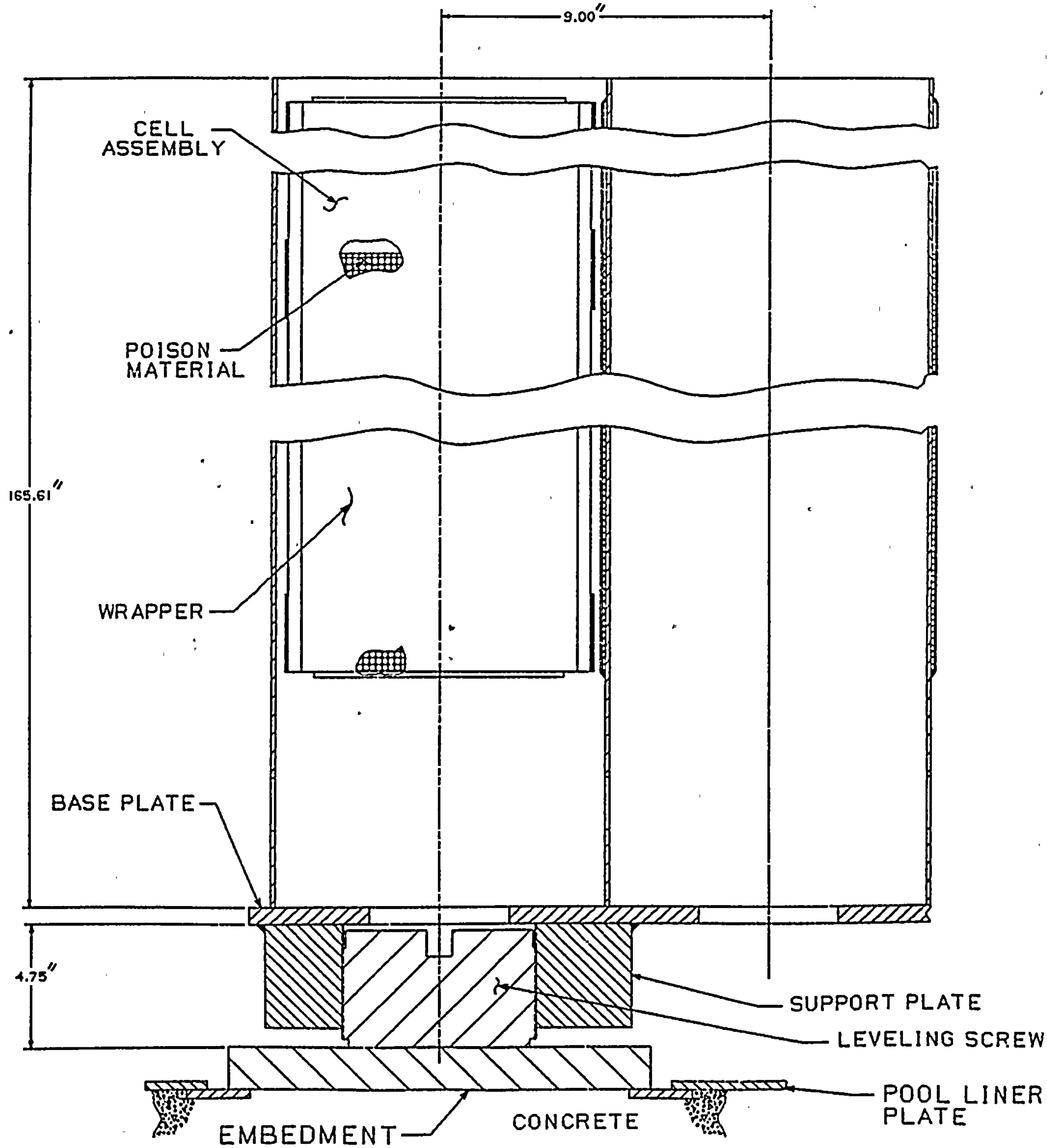


Figure 4-8

REGION II
MODULE CROSS-SECTION

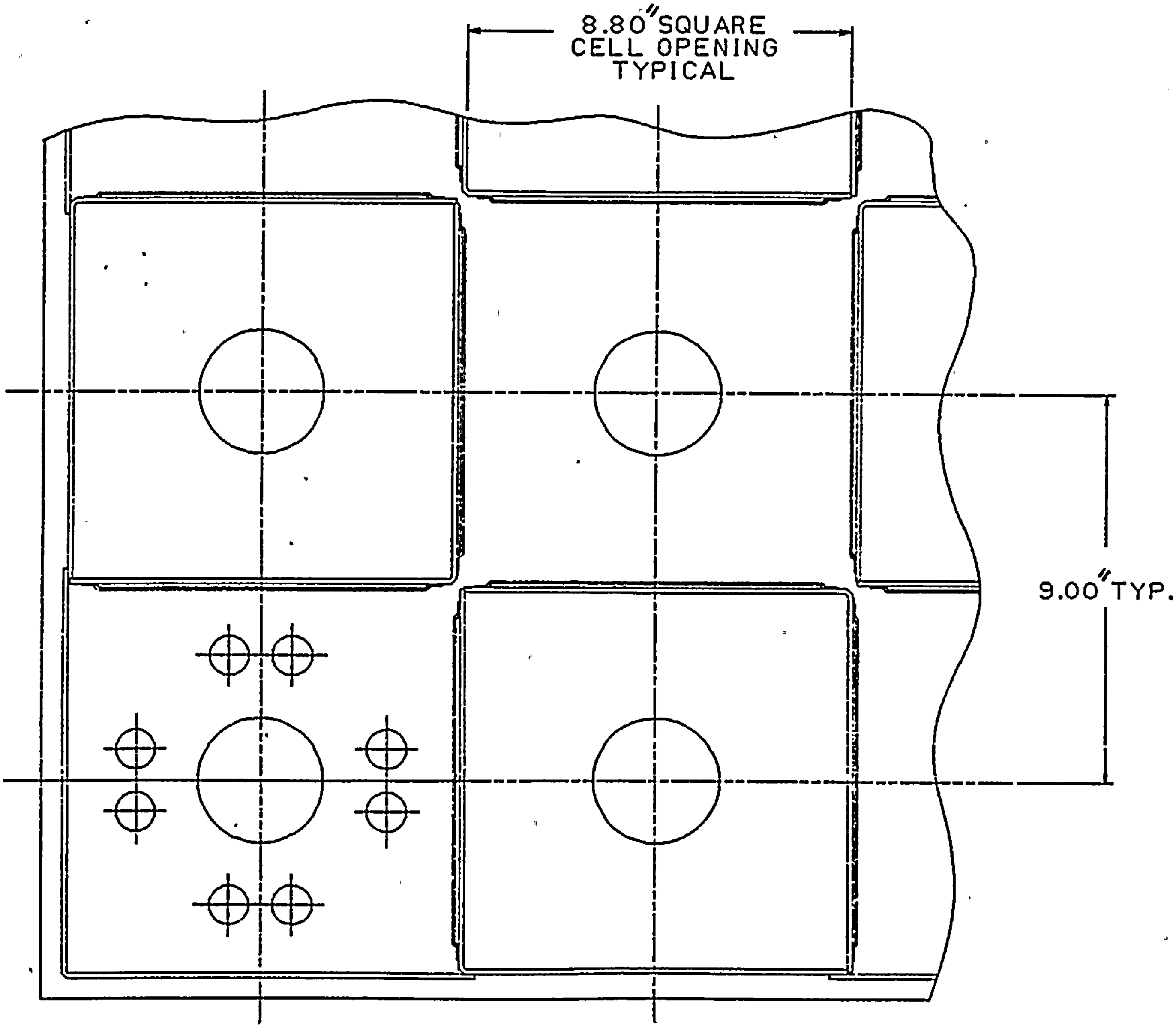


Figure 4-9

REGION II
MODULE TOP VIEW

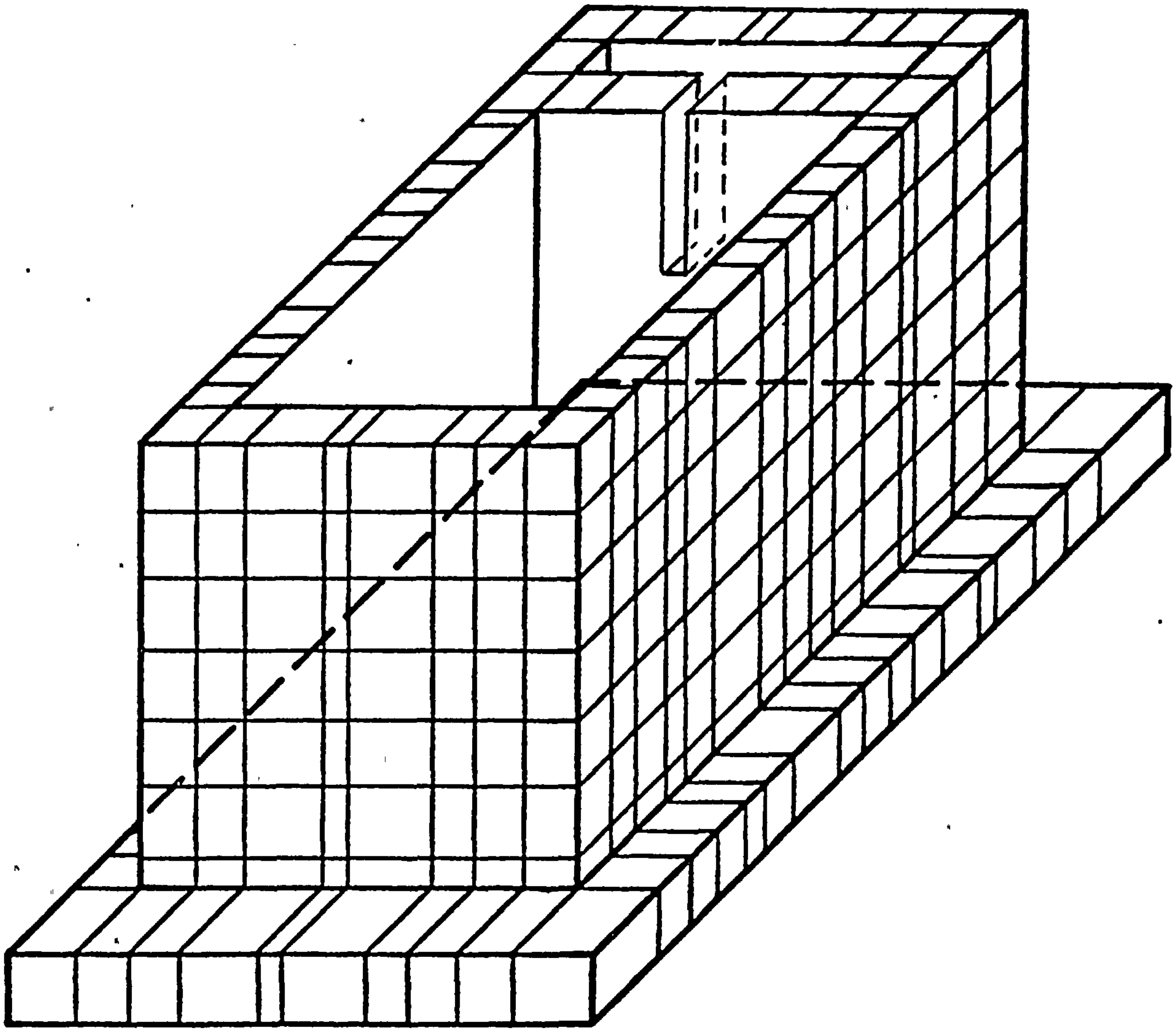


Figure 4-10

FINITE ELEMENT MODEL OF SPENT FUEL POOL

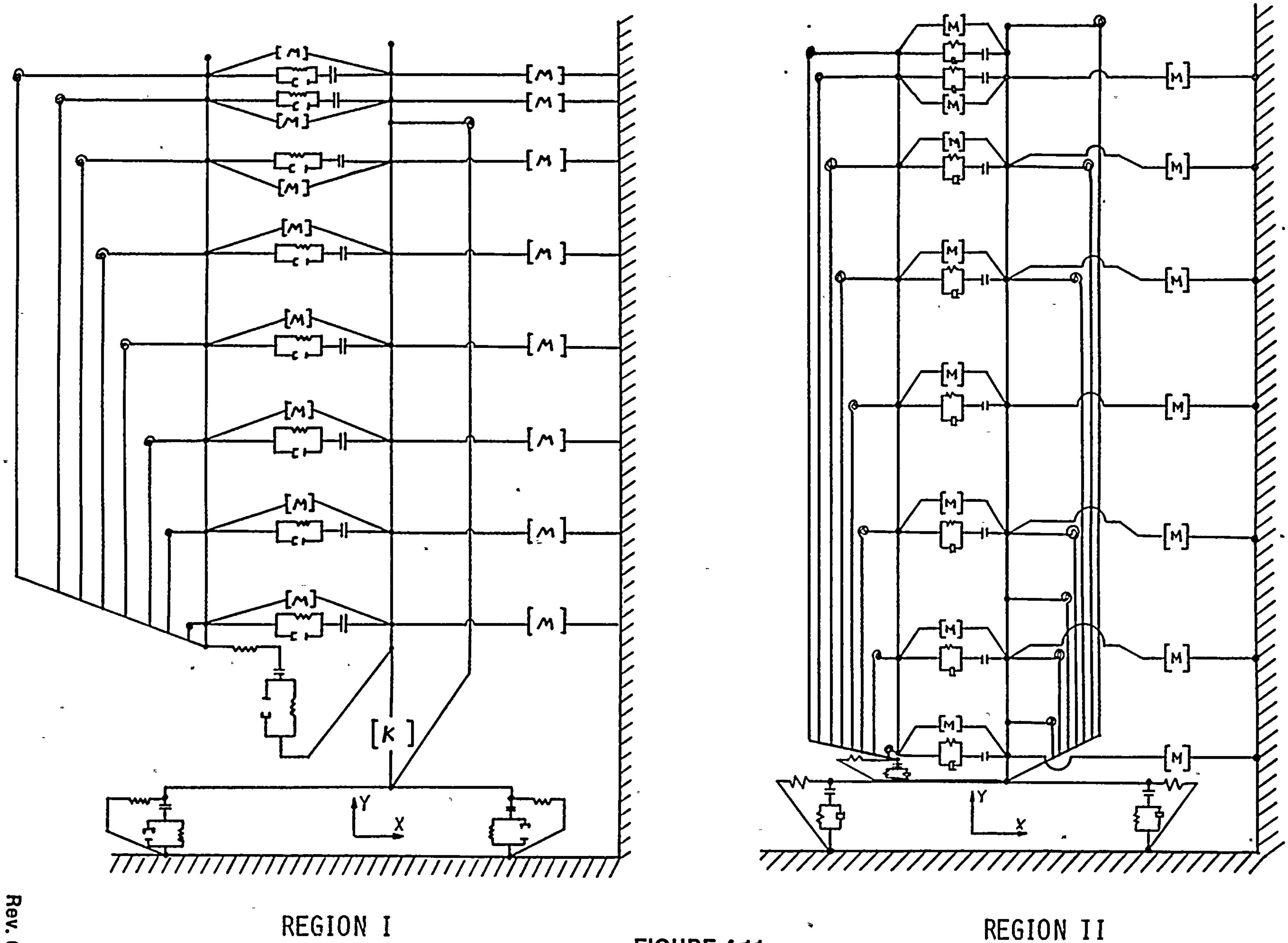
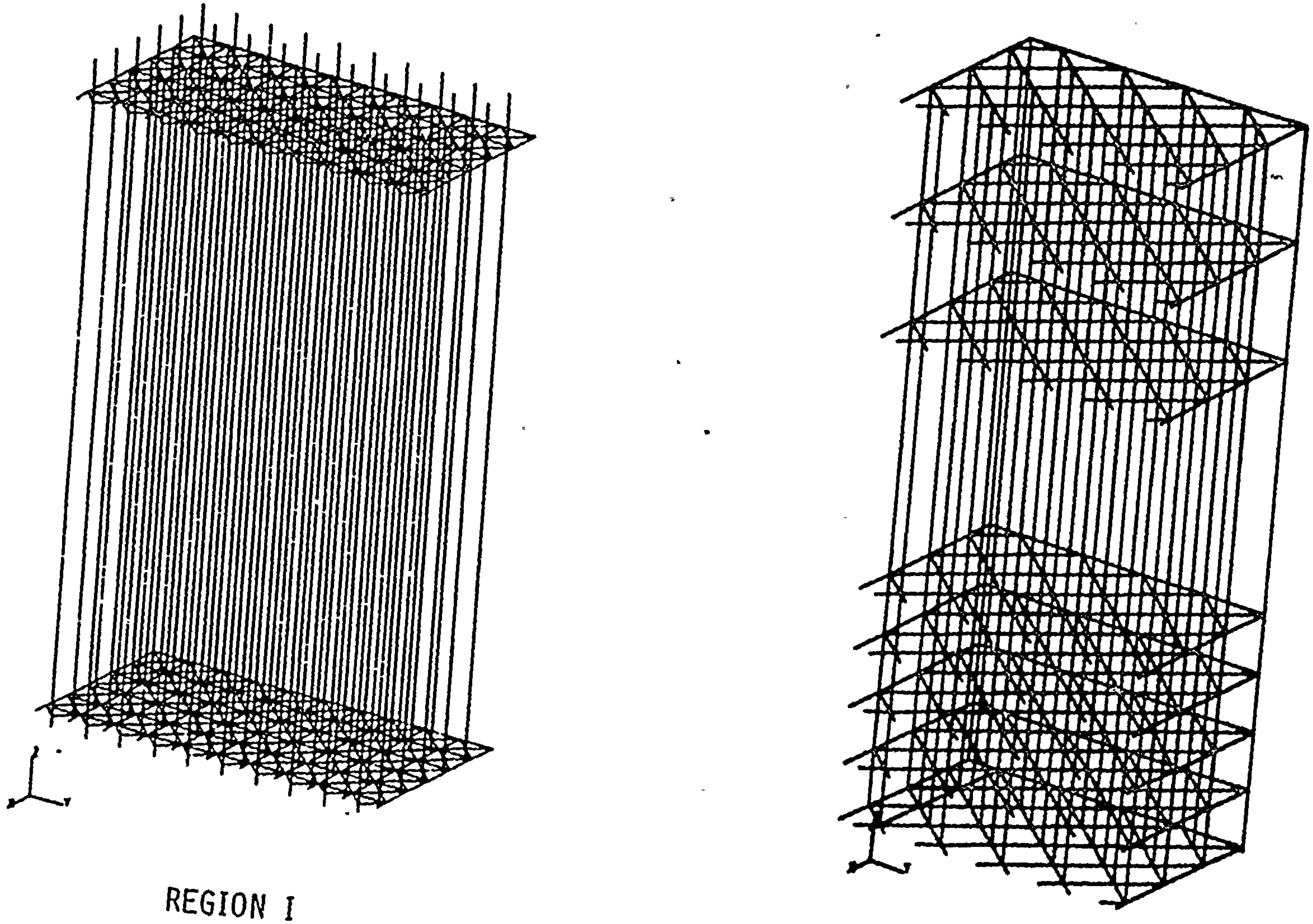


FIGURE 4-11

NONLINEAR SEISMIC MODELS



REGION I

REGION II

Figure 4-12

DETAIL SEISMIC MODELS

5.0 COST/BENEFIT ASSESSMENT

5.1 COST/BENEFIT ASSESSMENT

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

5.1.1 Need for Increased Storage Capacity

a. FPL currently has no contractual arrangements with any fuel reprocessing facilities.

b. At Turkey Point, both the Unit 3 and Unit 4 spent fuel pools have been previously reracked to 636 cells each. Because of specific piping and lighting interference in some areas of each pool, there are some unusable cells in each pool. For Unit 3, 621 storage cells are usable for spent fuel storage and for Unit 4, the number of usable cells is 614.

   Table 5-1 includes proposed refueling schedules for both Unit 3 and Unit 4, and the expected number of fuel assemblies that will be transferred into the spent fuel pools at each refueling until the total existing capacity is reached. All calculations in the table for loss of full core reserve (FCR) are based on the number of usable cells, not the number of total cells in each pool.

c. As of February 15, 1984, the Unit 3 spent fuel pool contained 372 spent fuel assemblies. The Unit 4 spent fuel pool contained 313 spent fuel assemblies and one new assembly.

d. At present, the storage of components other than fuel has not affected the total number of available storage locations in each pool because, for the most part, these components are inserts for the fuel. Each pool contains cells (defined in item b) that are unusable for fuel storage, but which can be used to store any components other than fuel. An itemized list of components stored in each pool is provided in Table 5-2.

e. 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 could be sent off site for final disposition under existing legislation. The government facility is expected to become available in 1998. As matters now stand and until alternate storage facilities are available, spent fuel assemblies on site will remain there.

f. Table 5-3 references the spent fuel storage capacity for both the Turkey Point Unit 3 and 4 spent fuel pools after reracking. Based on the present FPL fuel management

policy, the Unit 4 spent fuel pool will lose FCR during the Cycle 24 refueling in 2005. The Unit 3 pool will lose FCR during its Cycle 24 refueling in 2006. (Note: For calculation purposes, this table assumes all cells are usable).

5.1.2 Estimated Costs

The costs associated with the Unit 3 proposed spent fuel pool modification are estimated to be in the neighborhood of six and three-quarter million dollars. This figure includes items such as; 1) extensive engineering studies of spent fuel disposal alternatives, 2) design, engineering, manufacture, and installation of new spent fuel storage racks, 3) removal and offsite disposal (as low level radioactive waste) of the existing spent fuel storage racks, and 4) allowance for funds used during construction. Estimated value of uncertainty and cost escalation are not included in this sum.

5.1.3 Consideration of Alternatives

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

b. At the present time, there are no existing available independent spent fuel storage facilities. There are no firm commitments by either commercial firms or government agencies to construct or operate an independent spent fuel storage facility. In addition, cost and/or schedule considerations make an independent spent fuel storage facility on site unacceptable to meet the spent fuel storage needs at Turkey Point Units 3 and 4.

c. At present, FPL has no license to transship fuel between the St. Lucie and Turkey Point sites, nor are presently installed storage racks at either St. Lucie 1 or 2 licensed to store fuel generated at Turkey Point. There are no plans at the present time for transshipment within the FPL system.

d. The system production costs from the November 1983 updated version of the PROMOD [1] long term base case simulation, that assumes continued operation of the Turkey Point Nuclear Plants, were compared with separate PROMOD simulation costs that assume loss of either Turkey Point Unit 3 or 4. Cycles 13 for Units 3 and 4 were considered the last cycles before refueling capability is lost for this comparison. Table 5-4 indicates the average yearly system cost increases for Units 3 and 4 for 3 years after reactor shutdown. Plant shutdown would place a heavy financial burden on Florida residents within FPL's service area 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 pools will be used more efficiently by safely increasing the density of fuel storage.

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

5.1.5 Thermal Impact on the Environment

Section 3.2 contains a description of the following considerations: the additional heat load and the anticipated maximum temperature of water in the SFP 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, approximately 120 cubic feet of resins are generated per unit, per year, by the SFP purification system. No significant increase in volume of solid radioactive wastes is expected due to the new racks. It is estimated that an additional 30 cubic feet of resins will be generated by the spent fuel pool cleanup system per unit only during reracking.

Operating plant experience with high density fuel storage has not indicated any noticeable increase in the solid radioactive wastes generated by the increased fuel storage.

5.2.2 Gaseous Radwaste

The following are the Kr-85 release data for 1981, 1982, and 1983. These data are for the Units 3 and 4 plant vent because the Unit 4 spent fuel pool building vent releases through the plant vent. Kr-85 analyses are not performed on the Unit 3 spent fuel pool building vent which discharges to the atmosphere separately.

January through December 1981 = 2.79 Ci (Kr-85)
January through December 1982 = 16.84 Ci (Kr-85)
January through November 1983 = 50.8 Ci (Kr-85)

5.2.3 Personnel Exposure

a. The range of values for recent gamma isotopic analyses of spent fuel pool water is shown on Table 5-5. Unit 3 was refueled in October and November 1983; Unit 4 was refueled in October 1982. Earlier data have shown that the concentrations in the Unit 4 spent fuel pool rose temporarily during refueling and then returned to a level of equilibrium.

b. Operating experience shows dose rates of 10 to 40 mrem/hour either at the edge or above the center of the spent fuel pools 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 pools. The latest airborne analysis is provided in Table 5-6. The proposed reracking is not expected to significantly increase this activity.

d. Operating plant experience with dense fuel storage has shown no noticeable increases in airborne radioactivity above the spent fuel pool or at the site boundary. No significant increases are expected from more dense storage.

e. As stated in Section 5.2.1 and based on operating experience with dense fuel storage racks, there is 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. Operating experience with high density storage racks has shown no significant increase in the annual man-rem due to the increased fuel storage, including the changing of spent fuel pool cooling system resins and filters. Changing the racks to an even higher density will not change these conclusions.

f. Most of the crud associated with spent fuel storage is released soon after fuel is removed from the reactor. Once fuel is placed into the pool storage positions, additional crud contribution is minimal.

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

g. There is no access underneath the spent fuel pool. During normal operation, the radiation zone designation of areas around the sides of the pools will not change due to reracking. 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. Sections 5.2.3.e and 5.2.3.f indicate that operating experience with high density racks has shown no significant increase in radioactivity levels in the water or dose rates above the pool. As stated in Sections 5.2.1 and 5.2.3.e operating experience with high density racks has shown no increase in the processing of solid radioactive waste or the man-rem associated with it.

As discussed in Sections 5.2.3.c and 5.2.3.d, operating experience has shown no noticeable increases in airborne radioactivity for high density storage racks.

Operating experience has shown no increase in man-rem due to the increased fuel storage with high density racks. Therefore, no increase in the annual man-rem is expected at Turkey Point as a result of the increased storage capacity of the spent fuel pools with the higher density storage racks.

The existing Turkey Point 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 Re-Rack 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 the Plant Health Physicist, who is assisted by his staff, with the support of the Corporate Health Physicist and his 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 shall wear protective clothing and respiratory protective equipment, depending on work conditions, 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 station radiation protection staff closely monitors and controls 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 Re-Racking

Table 5-7 is a summary of expected exposures for each phase of the Unit 3 re-racking operation. These estimates are made based on the proposed installation plan, including fuel transfers, the use of long-handled tools, and the onsite decontamination of the old storage racks. Also, current pool radioactivity levels were conservatively increased in calculating these exposures. The total occupational exposure for the Unit 3 re-racking operation is conservatively estimated to be between 88 and 130 person-rem.

5.2.5 <u>Rack Disposal</u>

The spent fuel storage rack modules that will be removed from the spent fuel pool weigh between 32,200 and 36,200 pounds each. They will be decontaminated and disposed of as radioactive waste for burial, or decontaminated and disposed of as nonradioactive waste in accordance with existing Turkey Point health physics procedures. The type of disposal will be based on the capability to decontaminate.

5.3 ACCIDENT EVALUATION

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

5.3.1.1 Fuel Assembly Drop Analysis

As discussed in Section 4.1.2.2, the proposed spent fuel pool modifications will not increase the radiological consequences of fuel handling accidents previously evaluated in Section 14.2.1 of the Turkey Point Updated FSAR.

5.3.1.2 Cask Drop Analysis

5.3.1.2.1 Cask Handling

Appendix A to Appendix 14E of the Turkey Point Updated FSAR [2] evaluates the potential for a cask drop over the spent fuel pit. As discussed in Appendix A, limit switches prevent movement of the cask beyond the laydown area in the bottom of the pit; prevent interference of the cask crane bridge, trolley, and hoist with fuel racks or building structures; and restrict vertical lift of the cask to an elevation of about 6 inches above the top of the pit wall. The crane also has hurricane latches. The rerack program will not alter the cask handling procedures described in Updated FSAR Section 9.5. The cask handling crane meets the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [3].

5.3.1.2.2 Radiological Consequences

Figure 4-4 shows the loci of possible impact points for the unlikely, postulated event of a cask tip into the spent fuel pool. For this loci of points, it is assumed that the cask hits the Spent Fuel Building walls as it is being moved, the yoke disengages, and the cask tips and falls horizontally into the pool. This cask drop impact arc was previously presented in Figure A-1 of Updated FSAR Appendix 14E.

For the calculation of radiological consequences potentially resulting from a cask drop accident, two cases were evaluated regarding the number of fuel assemblies that are assumed to suffer a loss of integrity:

Case 1: The number of assemblies damaged is equal to the number offloaded during a normal refueling plus the remainder of the pool filled with discharged assemblies from previous refuelings.

Case 2: The number of assemblies damaged is equal to a full-core offload plus the remainder of the pool filled with discharged assemblies from previous refuelings.

The model for calculating the thyroid and whole-body site boundary doses incorporated the conservative assumptions specified in Standard Review Plan (SRP) Section 15.7.5[4] and Regulatory Guide 1.25[5] with the exception that a 1.0 Radial Peaking Factor (RPF) was utilized for Case 2. An RPF of 1.65 as specified in Regulatory Guide 1.25 is intended to represent the highest burnup fuel assembly to which all the impacted fuel assemblies are to be equated. While this value may be appropriate for the analysis of a postulated accident involving a single assembly, it is grossly overconservative when applied

to an analysis of a full core whose fuel assemblies have various exposure histories. An RPF of 1.0 has been determined as being more representative for the offload of a full core and has been applied to each assembly in the Case 2 analysis. The use of a 1.0 RPF for the calculation of cask drop radiological consequences has been previously submitted to the NRC for FPL's St. Lucie Unit 1 plant (see Reference [6].)

Table 5-8 lists the thyroid doses for the two cases evaluated. (The whole-body doses are not listed since the thyroid doses are limiting for both cases.) The results of the analysis demonstrate that by requiring the decay time of spent fuel in the pool to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses will be less than the guidelines of SRP Section 15.7.5 should a dropped cask strike the stored fuel assemblies. These doses are well within 10 CFR Part 100 limits. Accordingly, Technical Specification 3.12[7] has been revised to require a decay time of 1525 hours for all fuel in the spent fuel pool prior to cask handling operations (see Section 3.5). This is conservative since not all spent fuel storage modules located in the pool are susceptible to impact from any single cask drop. Thus, the proposed spent fuel pit modifications will not increase the radiological consequences of a cask drop accident previously evaluated.

5.3.1.2.3 Overhead Cranes

Except for the area described in Section 5.3.1.2.1, the spent fuel cask crane is not capable of traveling over or into the vicinity of the spent fuel pool. A complete cask crane component description, cask handling description, and cask crane design evaluation are provided in Updated FSAR Section 9.5 and Appendix 14E and will not be affected as a result of the rerack program.

5.3.1.2.4 Acceptability

The accident aspects of review establish acceptability with respect to Sections 5.3.1.2.1 and 5.3.1.2.2 of this report.

Requiring spent fuel decay time to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit will keep potential offsite doses well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

5.3.2 <u>Fuel Decay</u>

Prior to cask handling operations, proposed Technical Specification 3.12 (see Section 3.5) requires a decay time of 1525 hours for all fuel in the pool. Thus, with the increased storage capacity, the radiological consequences of a cask drop will be well within the requirements of 10 CFR Part 100.

5.3.3 Loads Over Spent Fuel

A technical specification which limits the maximum weight of loads that may be transported over spent fuel is presently being pursued by FPL as part of the resolution of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".

5.3.4 Conclusions

Since the spent fuel cask will not be handled over or in the vicinity of spent fuel except as provided for in Section 5.3.1.2.1, the proposed modification will not result in a significant increase in the probability of the cask drop accident previously evaluated in the Turkey Point Updated FSAR or Safety Evaluation Report [8]. Furthermore, as shown in Section 5.3.1.2.2, by requiring the decay time of spent fuel to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses will be well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies. The proposed spent fuel pit modifications will not increase the radiological consequences of a cask drop accident previously evaluated.

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 14.0, remain unchanged. Therefore, there will be no change or impact to any previous determinations of the Final Environmental Statement [9]. Based on the foregoing, the proposed amendments will not significantly affect the quality of the human environment; therefore, under 10 CFR 51.5c, issuance of a negative declaration is appropriate.

5.4 REFERENCES

1. PROMOD III Computer Code, Version 22.8, Energy Management Associates.

2. Turkey Point Plant Units 3 and 4, Updated Final Safety Analysis Report, Docket Nos. 50-250 and 50-251.

3. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants", NUREG-0612, July 1980.

4. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, July 1981.

5. Nuclear Regulatory Commission, "Assumption Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, March 1972.

6. St. Lucie Plant Unit 1, Final Safety Analysis Report, Section 9.1, Docket No. 50-335.

7. Turkey Point Plant Units 3 and 4, Technical Specifications, Docket Nos. 50-250 and 50-251.

8. Turkey Point Plant Units 3 and 4, Safety Evaluation Report Supporting Amendments 23 and 22 to Licenses DPR-31 and DPR-41, respectively, Docket Nos. 50-250 and 50-251.

9. Turkey Point Plant Units 3 and 4, Final Environmental Statement, Docket Nos. 50-250 and 50-251.

TABLE 5-1

ESTIMATED SPENT FUEL POOL CAPACITY REQUIREMENTS

Turkey Point Unit 3

Cycle No.	Approx. Cycle Startup Date	Total No. Assemblies in Pool from all Previous Cycles	Spaces Required for Full Core Reserve	Total No. Spaces Needed During This Cycle	Excess Storage Available	Augmented Storage Required
1 to 8	--	309	157	466	155	0
9	12/10/83	372	157	529	92	0
10	05/18/85	440	157	597	24	0
11	12/15/86	492*	157	649	0	28
12	05/01/88	560	157	717	0	96
13	12/15/89	612**	157	768	0	148
14	05/01/91	680	157	837	0	216

Turkey Point Unit 4

Cycle No.	Approx. Cycle Startup Date	Total No. Assemblies in Pool from all Previous Cycles	Spaces Required for Full Core Reserve	Total No. Spaces Needed During This Cycle	Excess Storage Available	Augmented Storage Required
1 to 8	--	277	157	434	180	0
9	05/16/83	314	157	471	143	0
10	05/12/84	382	157	539	75	0
11	12/01/85	446	157	603	11	0
12	05/01/87	510†	157	667	0	53
13	12/15/88	574††	157	731	0	117
14	05/01/90	638	157	795	0	181

*FCR lost at 464 stored assemblies; pool capacity = 621 assemblies.
**Cycle 13 is last refueling possible with existing storage racks - unit must be shut down due to inability to refuel after this cycle.
†FCR lost at 457 stored assemblies; pool capacity = 614 assemblies.
††Cycle 13 is last refueling possible with existing storage racks - unit must be shut down due to inability to refuel after this cycle.

TABLE 5-2

COMPONENTS STORED IN SPENT FUEL POOL

COMPONENT	Unit 3	Unit 4
Thimble plugs - standard	63	9
Thimble plugs - OFA	8	0
Burnable poison rod assemblies	154	146
Sources (primary/secondary)	4/0	3/1
Control rod assemblies	2	1
"Dummy" test fuel assemblies	0	1
Control rod mechanisms (part length)	8	8

TABLE 5-3

SPENT FUEL POOL CAPACITY AFTER RERACK

Unit-3 Cycle No.	Approx. Cycle Startup Date	Estimated Total No. Assemblies in Pool from all Previous Cycles	Spaces Required for Full Core Reserve	Estimated Total No. Spaces Needed During This Cycle	Excess Storage Available	Augmented Storage Required
1-14	05/01/91	680	157	837	567	0
15	12/15/92	732	157	889	515	0
16	05/01/94	800	157	957	447	0
17	12/05/95	852	157	1009	395	0
18	05/01/97	920	157	1077	327	0
19	12/15/98	972	157	1129	275	0
20	05/01/2000	1040	157	1197	207	0
21	12/15/01	1092	157	1249	155	0
22	05/01/03	1160	157	1317	87	0
23	12/15/04	1212	157	1369	35	0
24	05/01/06	1280*	157	1437	0	33
**	EOL	1332	157	1489	0	85

Unit-4 Cycle No.	Approx. Cycle Startup Date	Estimated Total No. Assemblies in Pool from all Previous Cycles	Spaces Required for Full Core Reserve	Estimated Total No. Spaces Needed During This Cycle	Excess Storage Available	Augmented Storage Required
1-14	05/01/90	638	157	795	609	0
15	12/15/91	702	157	859	545	0
16	05/01/93	766	157	923	481	0
17	12/15/94	830	157	987	417	0
18	05/01/96	894	157	1051	353	0
19	12/15/97	958	157	1115	289	0
20	05/01/99	1022	157	1179	225	0
21	12/15/2000	1086	157	1243	161	0
22	05/01/02	1150	157	1307	97	0
23	12/15/03	1214	157	1371	33	0
24	05/01/05	1278*	157	1435	0	31
25	12/15/06	1342	157	1499	0	95
**	EOL	1406	157	1563	0	159

With the rerack, the number of storage cells will be 1404 for each pool.

*FCR is lost at 1247 assemblies in each spent fuel pool.

**Unit end-of-licensed life (EOL) = 4/27/07 for both Units 3 and 4.

TABLE 5-4

SYSTEM PRODUCTION COSTS
($000)

	TP3 Outage Case	Base Case	Cost of TP3 Outage
1990	3,513,108.0	3,120,124.0	392,984.0
1991	4,028,913.0	3,681,888.0	347,025.0
1992	4,355,845.0	3,982,701.0	373,144.0

	TP4 Outage Case	Base Case	Cost of TP4 Outage
1989	3,164,936.0	2,789,172.0	375,764.0
1990	3,440,211.0	3,120,124.0	320,087.0
1991	4,039,903.0	3,681,888.0	358,015.0

TABLE 5-5

GAMMA ISOTOPIC ANALYSIS OF SPENT FUEL POOL WATER

Unit #3 January 6, 1983 - January 5, 1984

Isotope	Concentration (μCi/ml)
Co - 57	$(1.0 - 2.9) \times 10^{-5}$
Co - 58	$(0.06 - 4.6) \times 10^{-3}$
Co - 60	$(0.02 - 7.2) \times 10^{-2}$
Cs - 134	$(1.1 - 8.5) \times 10^{-4}$
Cs - 137	$(0.04 - 1.2) \times 10^{-3}$
I - 131	1.9×10^{-5}

Unit #4 September 23, 1982 - January 5, 1984

Isotope	Concentration (μCi/ml)
Co - 57	$(0.06 - 1.8) \times 10^{-4}$
Co - 58	$(0.09 - 6.7) \times 10^{-2}$
Co - 60	$(0.14 - 7.5) \times 10^{-2}$
Cs - 134	$(1.7 - 3.6) \times 10^{-4}$
Cs - 137	$(0.34 - 5.7) \times 10^{-4}$
I - 131	$(0.29 - 1.7) \times 10^{-4}$

TABLE 5-6

AIRBORNE RADIOACTIVITY ANALYSIS OF THE UNIT 3
SPENT FUEL POOL AREA (CURIES)

January to December 1983

Month	Co-58	Co-60	Cs-137	I-131	I-133	Mn-54	Ru-103
Jan	N.D.	1.8×10^{-6}	N.D.	9.8×10^{-7}	N.D.	N.D.	N.D.
Feb	N.D.	3.0×10^{-7}	N.D.	N.D.	N.D.	N.D.	N.D.
Mar	N.D.	1.6×10^{-6}	N.D.	N.D.	N.D.	N.D.	N.D.
Apr	N.D.	3.8×10^{-7}	N.D.	N.D.	N.D.	N.D.	N.D.
May	N.D.	2.9×10^{-7}	N.D.	2.4×10^{-6}	N.D.	N.D.	N.D.
Jun	N.D.	4.9×10^{-7}	N.D.	6.3×10^{-7}	N.D.	N.D.	N.D.
Jul	N.D.	7.6×10^{-7}	N.D.	3.5×10^{-7}	N.D.	2.5×10^{-7}	N.D.
Aug	N.D.	4.5×10^{-7}	N.D.	3.7×10^{-6}	N.D.	N.D.	N.D.
Sept	N.D.	N.D.	1.5×10^{-7}	4.7×10^{-6}	N.D.	N.D.	N.D.
Oct*	1.4×10^{-7}	N.D.	N.D.	2.9×10^{-3}	8.5×10^{-6}	N.D.	9.4×10^{-8}
Nov	3.1×10^{-7}	N.D.	N.D.	3.8×10^{-4}	N.D.	N.D.	5.7×10^{-7}
Dec	N.D.	N.D.	N.D.	2.9×10^{-5}	N.D.	N.D.	N.D.

*Fuel movement performed for Unit #3 Refueling

N.D. - Not Detectable

NOTE: Data were taken from Unit 3, which has a separate SFP area vent. The Unit 4 SFP area exhaust goes through the plant vent and cannot be sampled individually.

TABLE 5-7

ESTIMATED ALARA DOSES DURING RE-RACKING UNIT 3

		Person-Rem
Removal of Old Racks (including decontamination and disposal)		48 - 68
Spent Fuel Shuffles		4 - 6
Installation of New Racks		22 - 34
Support Services		14 - 22
	Total	88 - 130

TABLE 5-8

SITE BOUNDARY DOSES DUE TO A

SPENT FUEL CASK DROP [1]

	Number of Damaged Assemblies Assumed[3]	Decay Time Prior to Cask Handling (hours)	Site Boundary Thyroid Dose (rem)[2]
Case 1	80 (Refueling)	1475	27
Case 2	Full-Core Offload	1525	27

Notes

1. Based on assumptions specified in Standard Review Plan Section 15.7.5 and Regulatory Guide 1.25 with the exception that a 1.0 Radial Peaking Factor was used for Case 2. See the discussion in Section 5.3.1.2.1.

2. Whole-body doses are not listed since the thyroid doses are limiting for both cases.

3. Remainder of pool filled with discharged assemblies from previous refuelings.