ATTACHMENT III

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

SPENT FUEL STORAGE
MODIFICATION

SAFETY ANALYSIS REPORT

February 20, 1986

94 Pages

PALSFP-1-NL02

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF FIGURES

Chapter 3

Chapter 4

Chapter 5

1.0 INTRODUCTION

1.1 LICENSE AMENDMENT REQUESTED

Consumers Power Company is currently pursuing the design and manufacture of new spent fuel storage racks to be placed into the spent fuel pool and the spare (north) tilt pit of the Palisades Plant. The purpose of these new racks is to increase the amount of spent fuel that can be stored in the existing spent fuel pool and spare (north) tilt pit. The racks are designed to store spent fuel assemblies in a high density array. Therefore, Consumers Power Company is requesting that a license Amendment be issued to the Palisades Unit Facility Operating License DPR-20, 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 a license amendment.

1.2 CURRENT STATUS

The spent fuel pool, located in the auxiliary building adjacent to the containment, is lined with stainless steel and has reinforced concrete walls and floor varying in thickness from 4-1/2 feet to 6 feet.

The original fuel racks were stainless steel with a center-to-center spacing of 11-1/4 inches. There were two 1/4-inch stainless steel plates between each pair of fuel assemblies. At design temperature, with no credit taken for soluble boron in the pool water, the maximum k_{eff} was less than 0.95. A recessed area was provided in the pool for a spent fuel shipping cask.

The spent fuel pool cooling system (see Section 3-2) is a closed loop system consisting of two half-capacity pumps, a full-capacity heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves and instrumentation.

The spent fuel pool cooling system has a heat removal capability of 23×10^6 Btu/h. The spent fuel pool cooling system is conservatively designed to maintain a pool average temperature at less than 125°F with 1/3 core of fully burned up fuel in the pool, 36 hours after reactor shutdown. A single failure of the cooling system would increase the pool temperature by only 3°F. The water in the spent fuel pool is normally borated to 1,720 ppm. The entire spent fuel pool cooling piping system is tornado protected and is located in a CP Co Design Class 1 structure as defined in Chapter 5 of the Palisades FSAR Update.

Fuel pool makeup water is supplied from the Safety Injection and Refueling Water (SIRW) Tank. A secondary backup supply of water is available from the fire system. This would be utilized to replenish the fuel pool water inventory in the event of considerable loss of pool water.

Two fuel tilt pits are located in the fuel handling area adjacent to the spent fuel pool and connected to it by canals which can be closed off by dam blocks. The south tilt pit is used for normal fuel transfer activities. The spare (north) tilt pit originally was provided to accommodate an additional unit on the site and is now considered an area of the spent fuel pool and is capable of storing 110 fuel assemblies. The dam block is not installed between the north (spare) tilt pit and the main pool when spent fuel is stored in the north tilt pit.

In 1978, due to the lack of fuel reprocessing facilities, spent fuel pool storage capacity was increased from a capacity of 276 assemblies to a capacity of 798 assemblies. This increase in capacity was achieved by removing the formerly existing fuel and control rod racks and replacing them with racks which have smaller center-to-center spacing.

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].

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

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

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

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

1.4 CONCLUSIONS

On the basis of the design requirements presented in this report, operating experience with high density fuel storage, and material referenced in this report, it is concluded that the proposed modification of the Palisades 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 Palisades FSAR Update and Operating License.

1.5 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. Palisades Plant Final Safety Analysis Report Update, Docket No. 50-255.

2.0 SUMMARY OF EXISTING RACK DESIGN

The existing spent fuel storage racks have the capacity to store 798 spent fuel assemblies. Each individual storage location consists of two concentric 1/8-inch austenitic Type 304 stainless steel square cans with the annular space occupied by B_4C neutron absorber plates to ensure subcriticality.

A rack assembly consists of a rectangular array of storage cans with a minimum 10-1/4 inches center-to-center spacing of the fuel assemblies. The array size of each rack was chosen to optimize the use of the pool space.

The racks are Seismic Category I per NRC Regulatory Guide 1.29 and can contact the pool wall and adjacent racks through pads mounted at the top and bottom of each rack to prevent excessive movement of the racks under postulated seismic accelerations. Provisions are made in the design to accommodate thermal expansion.

The cask laydown area can contain two 50-element racks which may be used to store fuel during full core off-loads. These two racks may be removed to allow placement of the spent fuel shipping cask or to allow the use of fuel inspection and repair equipment. An antitipping device provides antitipping protection and acts as a seismic restraint for the remaining racks.

The spare (north) tilt pit is used for spent fuel and control rod storage. Control rods and dimensionally abnormal fuel assemblies may be stored in one rack with slightly larger cans than those used in other racks. To minimize the heat generation in the tilt pit, normally only fuel decayed for at least one year will be stored there. When fuel with a shorter decay time is stored in the tilt pit, thermal conditions are monitored so that the design criteria are not exceeded.

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 57.2-1983 and in the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications"[1].

The spent fuel rack design will employ two separate and different arrays. The Region I racks have been previously licensed and installed in the Palisades spent fuel pool and are described in Section 2.0. Since these racks are being reused, criticality concerns for them will not be addressed. The new racks in Region II are designed to maintain $k_{eff} < 0.95$ for Combustion Engineering and Exxon fuel which has an initial enrichment/burnup combination in the acceptable area of Figure 3-2 with utilization of every cell committed.

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 II.

b. 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.

   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 8.1-1983 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 inadvertent drop of an assembly between the outside periphery of the rack 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, is a realistic initial condition.

The presence of approximately 1,720 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 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 METHOD VALIDATION

The calculation method and cross-section values 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 0.0014 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 0.0032 Δk.

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 57.2-1983, "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 as discussed in Section 3.1.3.1.

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 [13].

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.

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 3-2 shows the constant k_∞ contour based on a high enrichment end point of 3.26 w/o and 25,000 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 3.1.4.1.

The actual conditions for this determination are defined by the zero burnup intercept point in Figure 3-2. In this instance the intercept is at 1.50 w/o U^{235}. The design model for Region II is therefore on an unirradiated assembly at 1.50 w/o enrichment. Studies have shown that the axial burnup distributions of depleted fuel assemblies have no impact on the fuel rack reactivity.

3.1.4 RACK MODIFICATION

3.1.4.1 REGION II

The Region II spent fuel storage rack design is described in Section 4.1.2.1. The minimum B^{10} loading in the Region II poison plates is .006 gm B^{10}/cm^2.

3.1.4.1.1 SPENT FUEL STORAGE

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

$$k_{eff} = k_{worst} + B_{meth} + B_{part} + [ks_{meth}^2 + ks_{worst}^2 + ks_{re}^2]^{1/2}$$

where

k_{worst} = worst case KENO k_{eff} that includes centered fuel assembly position, material tolerance, and mechanical tolerances which result in spacings between assemblies less than nominal

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

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

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

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

ks_{re} = 95/95 uncertainty in the reactivity equivalence methodology

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.

3.1.4.1.2 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. Figures 3-3 and 3-4 illustrate the results of these Region II sensitivity studies.

3.1.5 ACCEPTANCE CRITERIA FOR CRITICALITY

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

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

3.2 SPENT FUEL POOL (BULK) COOLING SYSTEM EVALUATION

An evaluation of the adequacy of the existing spent fuel pool cooling system with the increased number of storage cells will be performed. The particulars of the method of this evaluation are discussed in what follows.

The first step in the analysis will involve the determination of the heat removal capability of the existing heat exchanger unit. Once this has been calculated, a plot of bulk pool exit temperature versus pool heat load will be generated assuming the minimum to maximum number of pumps in service.

Utilizing the decay heat method provided in NRC Branch Technical Position ASB 9-2 [14], the maximum pool heat load assuming all storage cells to be filled will be determined for both normal refueling and a full core offload just before normal refueling. With the pool heat loads for the above cases the bulk pool exit temperature will be determined utilizing the capability plot generated in the first step above assuming various numbers of pumps in service.

If in any of the above cases an exit pool temperature greater than 150°F is obtained, then a calculation of the time to reach 150°F will be determined assuming no change in the number of pumps in service. The same calculation will be performed if 212°F is exceeded.

In addition to the above, for those cases where 150°F is exceeded, it is important to know how soon after the beginning of pool loading this temperature will be reached assuming no corrective action to be taken. This will also be calculated utilizing a constant length of time to load each assembly.

Finally, an estimate of the time to reach 150°F following the beginning of pool loading assuming a complete loss of cooling system capability will be calculated, again conservatively assuming no corrective action to be taken.

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 Palisades 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 must be considered.

3.3.2 KEY ASSUMPTIONS

1. The nominal water level is 23 feet above the top of the fuel storage racks.

2. The maximum average fuel assembly decay heat output is 50.5 BTU/sec following 150 hours decay after shutdown. For conservatism, this value will be used.

3. For normal operations, the pool temperature is maintained ≦ 125°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 3-5 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 3-6.

As shown in Figure 3-6, 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 3-7. 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.

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 the rack modules during the rerack operation is being evaluated. The procedures and administrative controls governing the rerack operation will ensure the safe handling of rack modules. Applicable structures meet the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [15].

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 evaluation described in Section 3.1.2.

3.4.2 TEMPORARY CONSTRUCTION CRANE DROP

During the rerack operation, a temporary personnel platform will be installed in the spent fuel pool. This installation will be performed using lift rigs and shall meet the design and operational requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

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

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

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

SECTION 3

TABLES

TABLE 3-1

BENCHMARK CRITICAL EXPERIMENTS

	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 UO2 CRITICAL EXPERIMENTS

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 UO2 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.

SECTION 3 FIGURES

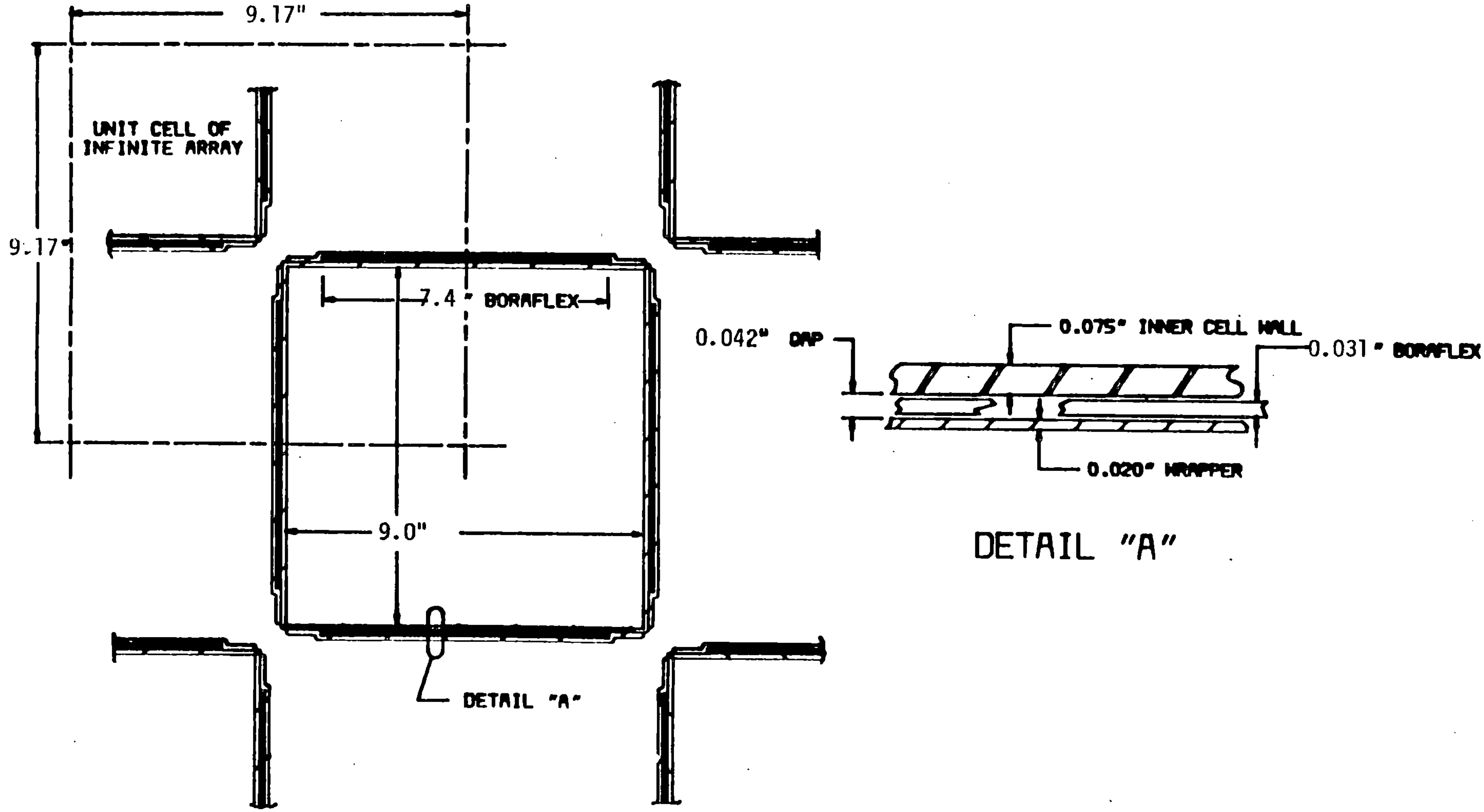
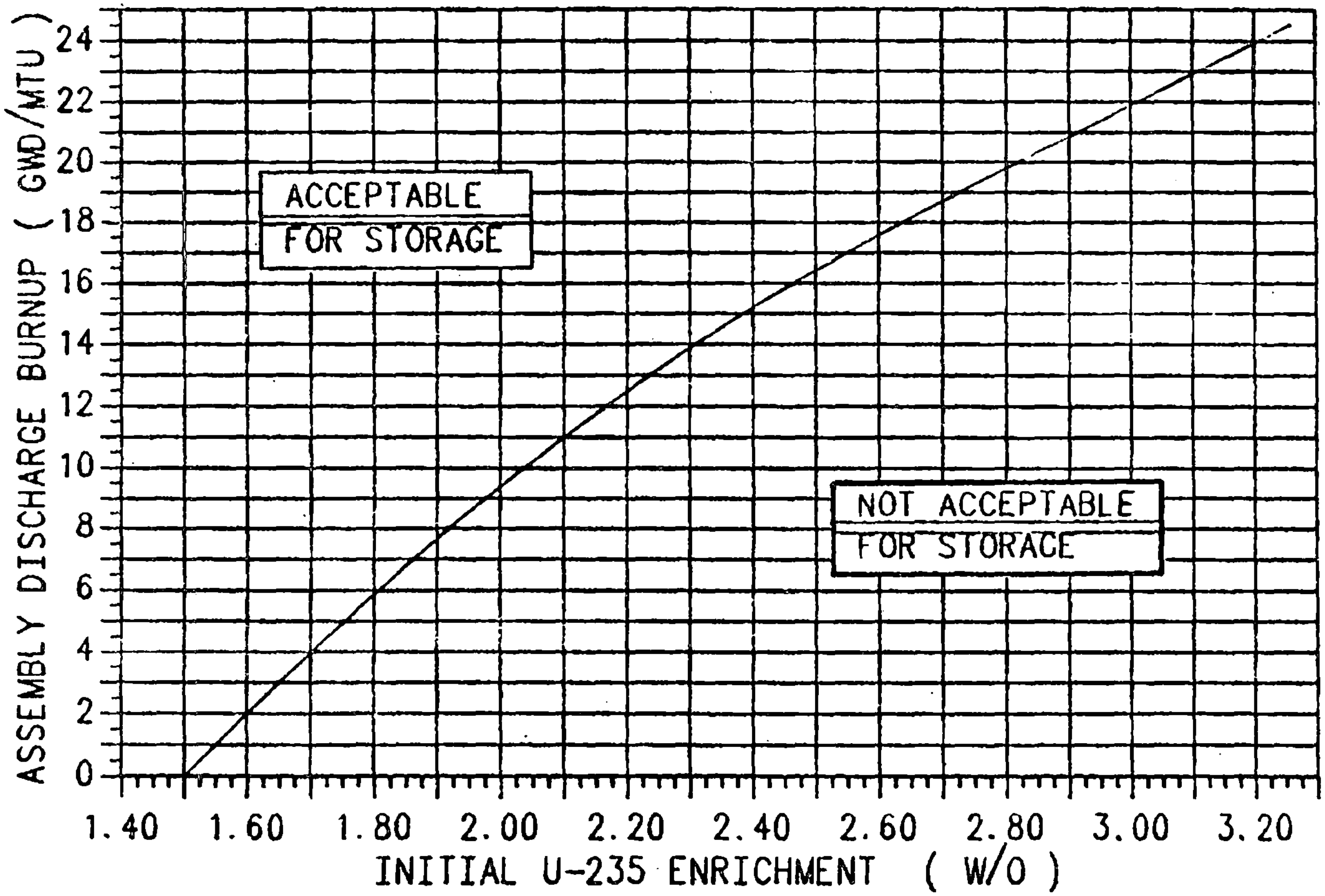


Figure 3-1

NOMINAL DIMENSIONS FOR THE REGION II STORAGE CELLS

FIGURE 3-2
Minimum Burnup vs. Initial Enrichment

MINIMUM BURNUP vs INITIAL U-235 ENRICHMENT FOR STORAGE IN SPENT FUEL RACKS



Palisades Region 2 Preliminary Burnup Credit Curve

WESTINGHOUSE PROPRIETARY CLASS 2

Final curve to be provided when analysis is complete.

FIGURE 3-3

Keff as a Function of Fuel Enrichment and Poison Loading for Fuel in a Region 11 Spent Fuel Storage Rack

(To be Provided when Analysis is complete)

FIGURE 3-4

Keff as a Function of Storage Cell Center to Center Spacing in a Region of Spent Fuel Storage Rack

(To be provided when analysis is complete)

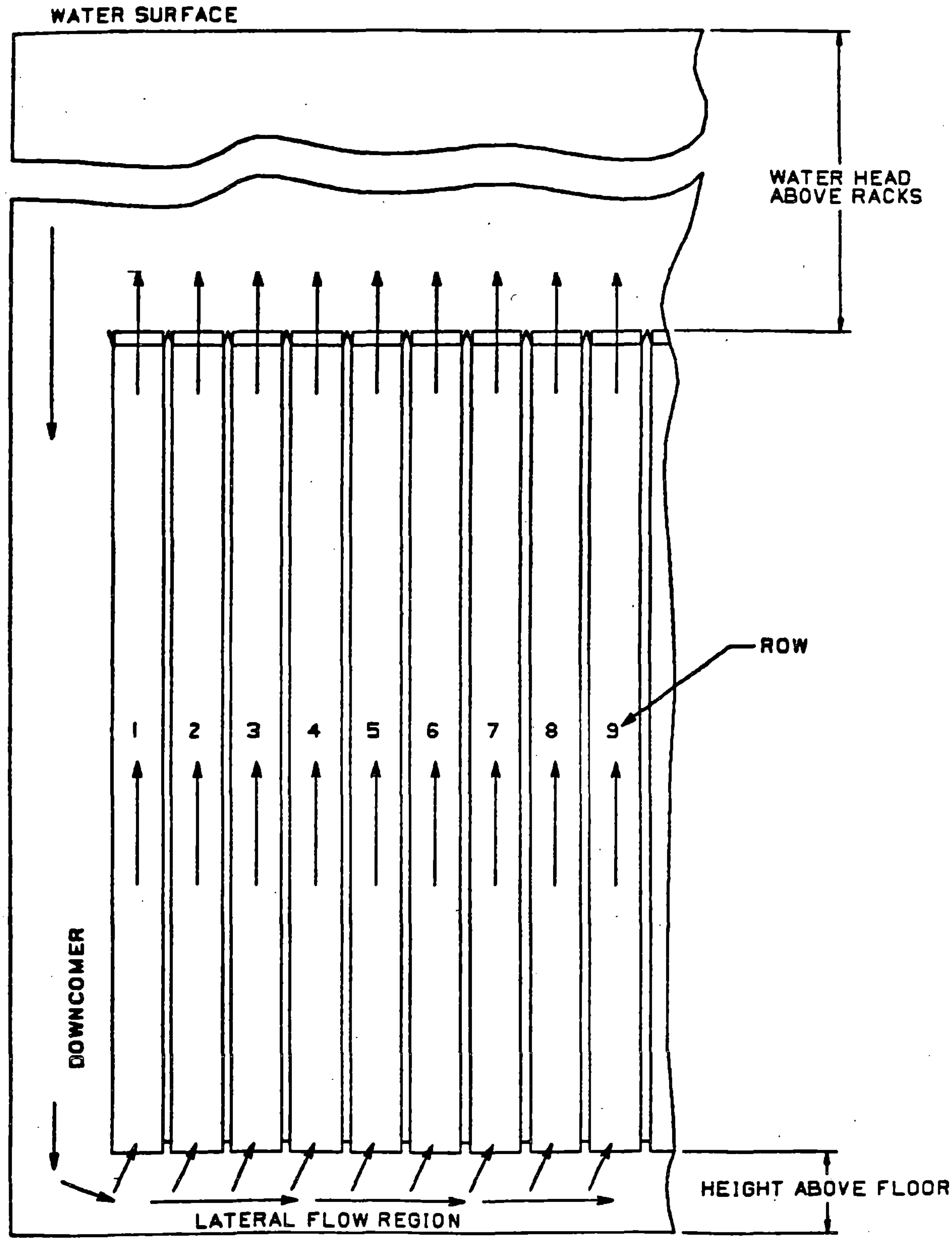


FIGURE 3-5

SPENT FUEL POOL NATURAL CIRCULATION MODEL (ELEVATION VIEW)

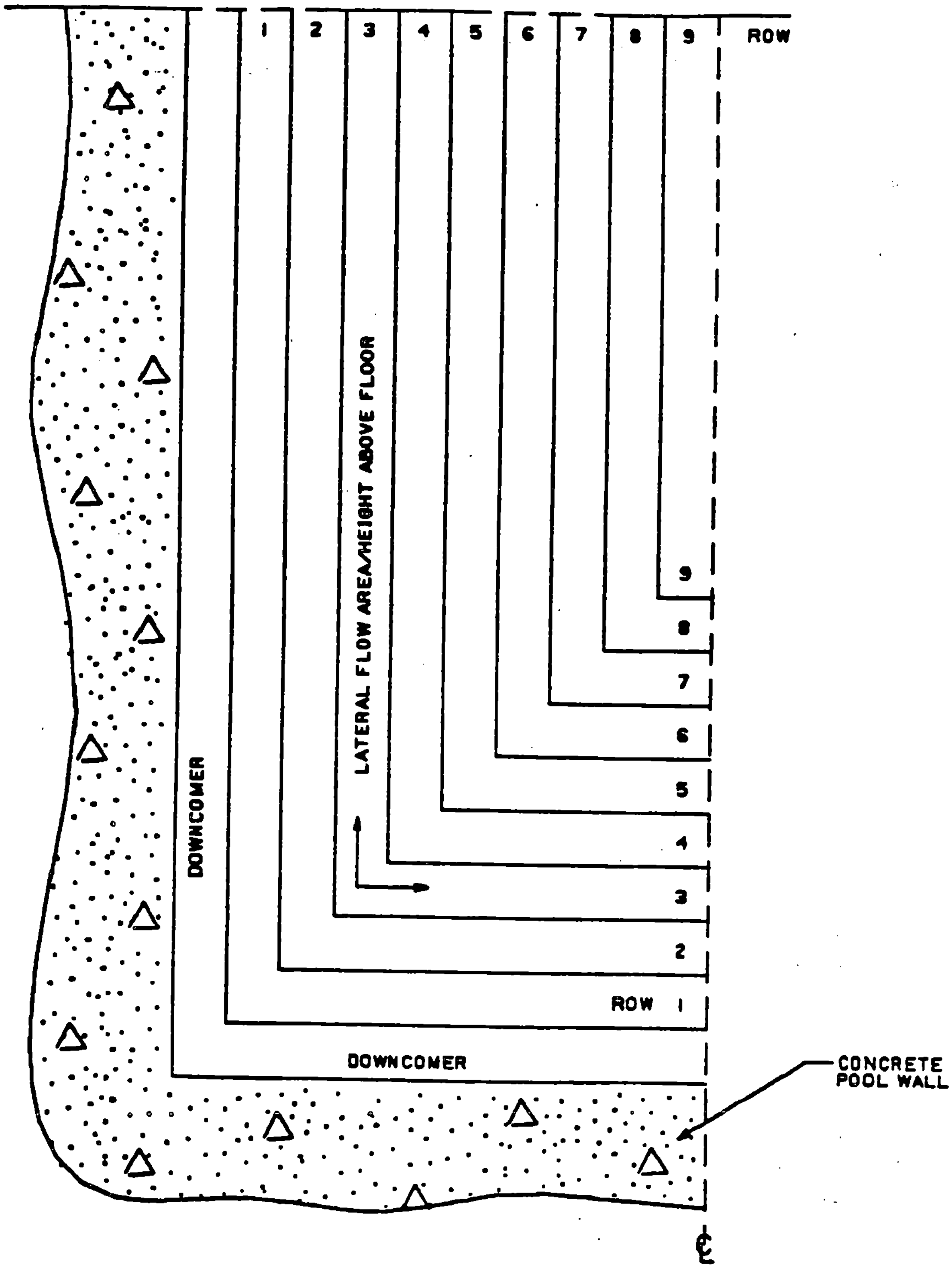


FIGURE 3-6

SPENT FUEL POOL NATURAL CIRCULATION MODEL
(PLAN VIEW)

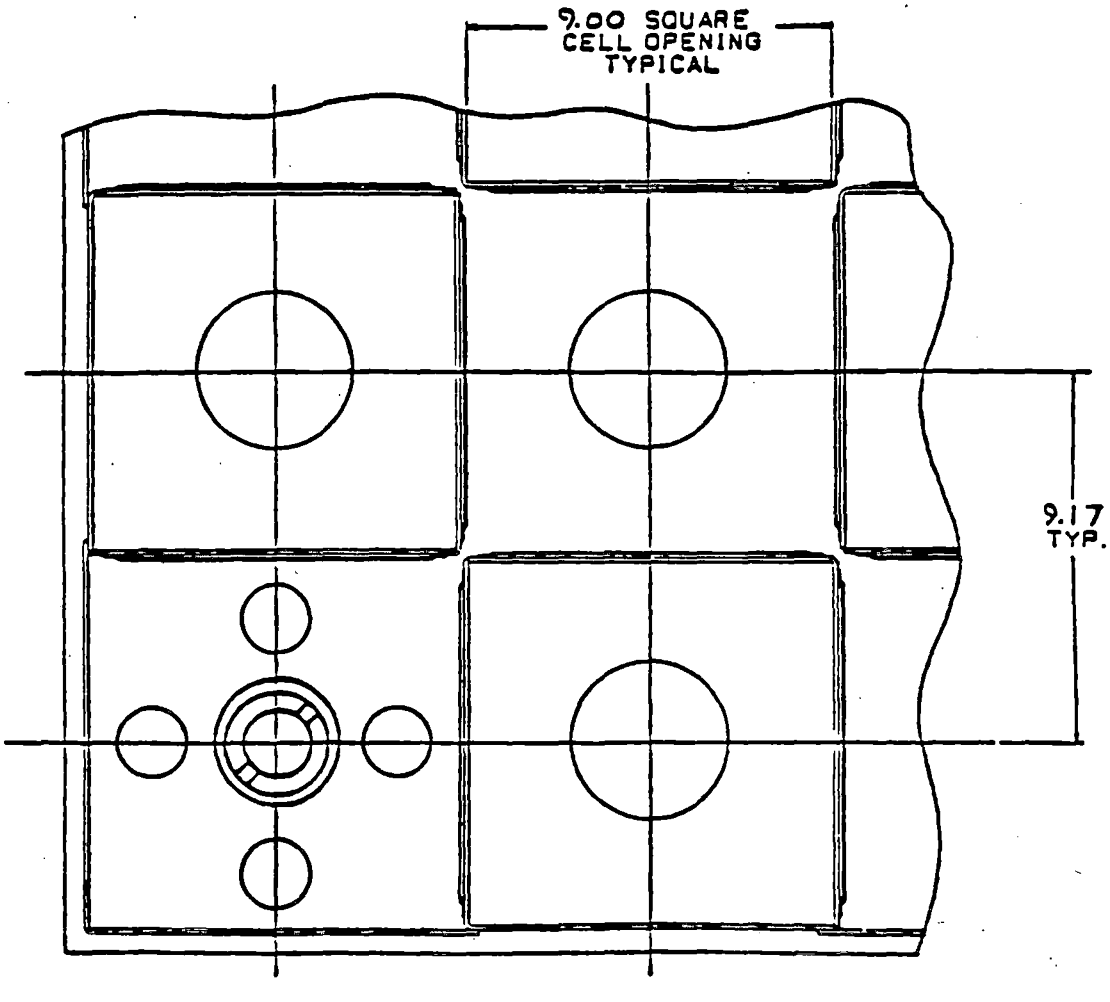


FIGURE 3-7

SPENT FUEL RACK INLET FLOW AREA TYPES
REGION II

4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 DESCRIPTION OF EXISTING FACILITIES: SPENT FUEL POOL AND BUILDING

The Palisades Plant is comprised of three main structures, namely the containment, turbine and auxiliary buildings. Addition to the auxiliary building on its north end was built in 1971 to accommodate radwaste processing equipment. In 1983, a second addition to the auxiliary building was added above the Baler Room to serve as a Technical Support Center. The spent fuel pool and the new fuel storage facilities are located between column rows F and G, and column lines 22 and 28 of the auxiliary building. The pool has a depth of 38 feet with the floor at elevation 611 feet rising to the operating deck at elevation 649 feet. Fuel transfer to and from the containment occurs through a fuel transfer tube. The pool structure has a refueling canal at the southwestern end and extends upward from pit elevation 610 feet to elevation 649 feet. The portion of the auxiliary building housing the spent fuel pool structures is founded on a separate mat and is physically isolated from other structures, namely, the containment building, the turbine building and the auxiliary building addition.

Figures 4-1 through 4-7 show the physical configuration of the spent fuel pool structure.

The spent fuel pool is constructed of reinforced concrete and is oriented in the north-south direction in the auxiliary building. The main pool floor is at elevation of 611 feet with the tilt pit floors at elevation 610 feet. The spent fuel pool is supported by series of walls which bear on the foundation mat at 590 feet. The decontamination rooms, waste monitoring tanks and pumps, heating and ventilating pipeways are located in compartments below the fuel pool floor at elevation 590 feet. Thus, the pool structure extends upward from the mat at elevation 590 feet to operation floor elevation 649 feet. The pool walls also serve as support for adjacent floors in addition to their primary function to resist the hydrostatic pressure and fuel rack loads.

The entire interior face of the spent fuel pit has 3/16-inch stainless steel liner to ensure against leakage. The inside dimensions of the pool are 38 feet-9 inches by 14 feet-8 inches. A 9 foot x 9 foot area in the northeast corner of the pool is recessed to accommodate a shipping cask. Adjacent to the spent fuel pool and on the west side are two tilt pits measuring 21 feet x 5 feet on the inside, separated from the main pool by a 4 foot thick reinforced concrete wall. A cutout in this wall approximately 2 feet-6 inches wide and extending down from the operating floor elevation to elevation 625 feet serves the purpose of a gate to transfer spent fuel bundles from the south tilt mechanism to the spent fuel pool. The gate to the south tilt pit is normally closed and the south pit is flooded with water only in case of a fuel transfer. The tilt pit at the north end of the pool was hitherto reserved for a future tilt mechanism which was to serve a second reactor unit. This north tilt pit is now used to store additional spent fuel. The gate between the north tilt pit and the main pool is always open when spent fuel is stored in the north tilt pit.

The spent fuel racks are supported on the floor of the pool and extend up to elevation 624 feet. Lateral restraints to the fuel pool walls at about 1 foot and 13 feet above the floor provide lateral support to the spent fuel racks and help resist any lateral loads such as earthquake loads that the racks may be subjected to. A small gap is provided between the ends of the lateral restraints and the face of the walls to accommodate any thermal expansion due to temperature excursion of the water contained in the pool. Thus, the bottom 14 foot section of the pool is the storage area, the middle 13 foot section is the transfer zone and the top 10 foot section of water is the shielding zone.

The spent fuel pool superstructure extends from elevation 649 feet to the roof at elevation 698 feet-2 inches. The electric overhead traveling bridge crane structure is supported at elevation 676 feet-8 inches. The spent fuel assemblies are handled and manipulated by an operator standing on a movable bridge called a fuel handling machine which is located over the pool at elevation 649 feet-8 inches. The fuel handling machine is also used to transfer fuel from the tilt mechanism to the storage racks.

4.1.2 DESCRIPTION OF SPENT FUEL RACKS

The function of the spent fuel storage racks is to provide storage space for fuel assemblies in a flooded pool while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excessive mechanical and 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 [1].

2. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor k_{eff} is $\leq .95$ including all uncertainties and under all credible conditions.

3. The racks are designed to allow coolant flow such that boiling in 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. 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 and north tilt pit rack arrangement is shown in Figure 4-8. Fuel storage is divided into two regions. Region I (422 locations) consists of existing racks with high density fuel assembly spacing obtained by utilizing a neutron absorbing material and is normally used for core off-loading. Region II (470 locations) consists of new racks with high density fuel assembly spacing and provides normal storage for spent fuel assemblies meeting required burnup considerations. Region I is designed to accommodate irradiated and 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.

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

Rack module data for the new racks is presented in Table 4-1.

4.1.2.1.1 NEW RACK (REGION II) DESIGN

The Region II storage racks consist of stainless steel cells assembled in a checkerboard pattern with a 9.17-inch centerline-to-centerline spacing, producing a honeycomb type structure as shown in Figure 4-9. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell sidewall by a stainless steel wrapper. The cells are welded to a base support assembly and to one another to form an integral structure. This design is provided with leveling screws which contact the spent fuel pool floor 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 4-10 and 4-11 illustrate these sections.

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

The stainless steel wrapper is attached to the the cell sidewall 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. 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, 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. 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 Radioactive Waste Containing Components of Nuclear Power Plants
1.29, Rev. 3 Sept. 1978	Seismic Design Classification

1.92, Rev. 1 Feb. 1976	Combining Model 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

c. 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, ASB 9-2, Residual Decay Energy for Light BTP, Water Reactors for Long-Term Cooling, Rev. 1, July 1981

d. 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

e. Palisades FSAR Update, Rev. 1

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 0.10 g. In addition, a safe shutdown earthquake (SSE), having a horizontal ground acceleration of 0.20 g is used to check the design to ensure no loss of function.

Seismic analysis of the fuel storage racks is being performed by the time-history method. 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 analysis includes the effects of the water in the pool, such as fluctuation of pressure due to acceleration, and sloshing.

Deflections or movements of racks under earthquake loading are 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 magnitudes that there is not structural damage to the fuel assemblies.

4.4 LOADS AND LOAD COMBINATIONS

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 envelops 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 seismic model which have been adjusted by peaking factors from the structural model to account for the stress gradients through the rack module.

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 DESIGN AND ANALYSIS PROCEDURE FOR SPENT FUEL STRUCTURE

The spent fuel pool structure with augmented storage capability is analyzed using the three-dimensional finite element method. The model includes soil, foundation mat, building structural elements, and the boundary condition to reflect structure/structure interaction. A selected perspective view of the model from elevation 611 ft through 649 ft is given in Figure 4-12.

The following loads were considered in the evaluation of the pool integrity:

- Dead load, includes pool structure self-weight, racks and fuel assemblies, and hydrostatic loads. In addition, all floor live loads, dead loads of adjacent structures and superstructure crane loads are included.
- Operating basis earthquake
- Safe shutdown earthquake
- Operating temperatures
- Hydrostatic loads are considered for a water level at elevation 648 feet in the spent fuel pool and tilt pits
- Sloshing effects of water - hydrodynamic loads
- Thermal Loads
- Increased loading due to the additional spent fuel elements to be stored in the pool.

To determine the adequacy of the structure, the criteria outlined in Section 5.9.1 of Palisades FSAR Update were adopted.

Based on the Palisades FSAR Update, the following critical load combinations were considered in the analysis of the pool structure.

1.25D + 1.25T + 1.25E (Normal Operating Condition)

1.0D + 1.0T + 1.0E' (Abnormal Operating Condition)

where

D = Dead load defined above including hydrostatic loads

E = Seismic (OBE) load including hydrodynamic (sloshing) loads

E' = Seismic (SSE) load including hydrodynamic (sloshing) loads

T = Thermal gradient load

The seismic loading used in the pool analyses was in accordance with the response spectra for the pool structure in the east-west (E-W) and north-south (N-S) as given in Chapter 5.2 of the FSAR Update.

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 and are being evaluated to determine that under all loading conditions they do not impact each other nor do they impact the pool walls or the existing Region I racks. Additional analysis is being performed to determine if modification to the Region I racks is required to prevent their impacting the Region II racks.

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure. The seismic analysis of a free-standing fuel rack is a time-history analysis performed on a nonlinear model.

The time history analysis is performed on a single cell nonlinear model with the effective properties of an average cell within the rack module. The nonlinear model is shown in Figure 4-13.

The effective single-cell properties are obtained from a structural model of the rack modules, as shown in Figure 4-14.

The details of the structural model and the seismic model are discussed in the following paragraphs.

The structural model, shown in Figure 4-14, is a finite element representation of the rack assembly consisting of beam elements interconnected at a finite number of nodal points, and general mass matrix elements. The beam elements model the beam action of the cells, the stiffening effect of the cell to cell welds, and the supporting effect of the support pads. The general mass matrix elements represent the hydrodynamic mass of the rack module. The beams which represent the cells are loaded with equivalent seismic loads and the model produces the structural displacements and internal load distributions necessary to calculate the effective structural properties of an average cell within the rack module. In addition to the stiffness properties, the internal load and stress distributions of this model are used to calculate stress peaking factors to account for the load gradients within the rack module.

The nonlinear seismic model, shown in Figure 4-13, is composed of the effective properties from the structural model with additional elements to account for hydrodynamic mass of the fuel, the gap between the fuel and cell, and the support pad boundary conditions of a free-standing rack. The elements of the nonlinear model are as follows:

The fuel assembly is modeled by beam elements and rotational spring elements which represent the structural and dynamic properties of the fuel rod bundle and grid support assemblies.

The cell assembly is represented by beam elements and rotational springs which have structural properties of an average cell within the rack structure.

The water within the cell and the hydrodynamic mass of the fuel assembly are modeled by general mass matrix elements connected between the fuel and cell.

The gaps between the fuel and cell are modeled by dynamic gap elements which are composed of a spring and damper in parallel, coupled in series to a concentric gap. The properties of the spring are the impact stiffness of the fuel assembly grid or nozzle and cell wall. The properties of the damper are the impact damping of the grid or nozzle. The properties of the concentric gap are the clearance per side between the fuel and cell.

The hydrodynamic mass of a submerged fuel rack assembly is modeled by general mass matrix elements connected between the cell and pool wall.

The support pads are modeled by a combination of dynamic friction elements connected by a "rigid" base beam arrangement which produces the spacing of corner support pads. The cell and fuel assemblies are located in the center of the base beam assembly and form a model which represents the rocking and sliding characteristics of a rack module.

The nonlinear model is run with simultaneous inputs of the vertical and the most limiting horizontal acceleration time history values. The damping values used in the seismic analysis are 2 percent damping for OBE and SSE. In addition, the model is run for a range of friction coefficients (0.2 and 0.8) to obtain the maximum values. The results from these runs are fuel to cell impact loads, support pad loads, support pad liftoff, rack sliding, and fuel rack structure internal loads and moments. These values are searched through the full time in order to obtain maximum values. The internal loads and stresses from the seismic model are adjusted by peaking factors from the structural model to account for the stress gradients through the rack module. Consequently, the maximum loaded rack components of each type are analyzed. Such an analysis envelops the other areas of the rack assembly. The maximum stresses from each of the three seismic events are combined by the SRSS method. In addition, the results are used to determine the rack response for full, partially filled, and empty rack module loading conditions.

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 the (i.e., with reinforcing steel stresses controlling the design). The acceptance criteria are stated in Chapter 5, Appendix A of the FSAR Update. 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 load combinations previously described and the ultimate strength design portion of the ACI 318-63 code.

The strength requirements for load combinations considered for evaluation are:

1. $y = \frac{1}{\emptyset}\ (1.25D + 1.25T + 1.25E)$

2. $Y = \frac{1}{\emptyset}\ (1.0D + 1.0T + 1.0E')$

where

D, T, E and E' are as defined in Section C.1, and

Y = Required yield strength of the structure

$\emptyset$ = Yield capacity reduction factor per ACl 318-71

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 earthquake (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" [1], (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" [1], (with clarifications as noted in Table 4-2).

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

4.6.3 FUEL HANDLING MACHINE UPLIFT ANALYSIS

An analysis will be performed to demonstrate that the rack can withstand a maximum uplift load of 4,000 pounds. This load can be applied to a postulated stuck fuel assembly without violating the criticality acceptance criterion. Resulting stresses will be within acceptable stress limits, and there will be no change in rack geometry of a magnitude which causes the criticality acceptance criterion to be violated.

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 and its handling tool of 1,500 pounds impacts the top of the fuel rack from a drop height of 3 feet. Calculations will show that the impact energy is absorbed by the dropped fuel assembly, the cells and rack base plate assembly. 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 (1,500 lbs) falls straight through an empty cell and impacts the rack base plate from a drop height of 183 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 will show that $k_{eff} \leq 0.95$ and the acceptance criterion is not violated.

4.6.5 FUEL RACK SLIDING AND OVERTURNING ANALYSIS

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 is produced by friction. A range of friction coefficients ($\mu = 0.2$ and 0.8) are used in this analysis. A low coefficient of friction ($\mu = 0.2$) produces maximum rack base 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 rack-to-rack or rack-to-wall clearances. Thus, impact between adjacent rack modules or between a rack module and the pool is prevented. 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 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 304 stainless steel.

4.7.2 NEUTRON ABSORBING MATERIAL

The neutron absorbing material, Boraflex, used in the Palisades 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 and the Quality Assurance Plan of the Westinghouse Water Reactor Divisions. Boraflex is a silicone-based polymer containing fine particles of boron carbide in a homogeneous, stable matrix. The Boraflex used in the new racks at Palisades contains a minimum B^{10} areal density of 0.006 gm/cm^2.

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[3]. 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[4].

Long-term borated water soak tests at high temperatures were also conducted[5]. 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, fabrication and installation of the new high density spent fuel storage racks comply with the pertinent Quality Assurance requirements of Appendix B to 10 CFR 50 and the Westinghouse Nuclear Service Integration Division Quality Assurance Plan WCAP 9245[8] which complies with the Westinghouse Water Reactor Divisions' Quality Assurance Plan as described in WCAP 8370[6] which is approved by the NRC.

Project auditing, source surveillance, plant surveillance, plant QC support, plant fuel and rack movement and plant health physics support shall conform with the Consumers Power Company Quality Assurance Program [10].

4.7.4 CONSTRUCTION TECHNIQUES

4.7.4.1 ADMINISTRATIVE CONTROL DURING MANUFACTURING AND INSTALLATION

The Palisades 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 prevent 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.

4.7.4.2 PROCEDURE

4.7.4.2.1 PREINSTALLATION

The following sequence of preinstallation events is planned for the spent fuel storage rack replacement project:

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

- A temporary platform for personnel access to perform rerack operations will be installed and used in conjunction with the existing fuel handling bridge.

- Handling of the lift rig for removing and installing racks will be from an intermediate hoist suspended from the 100-ton capacity overhead crane. The intermediate hoist in conjunction with the overhead crane will provide sufficient lift height to permit removal and installation of fuel racks. Reference Figure 4-15 for intermediate hoist and personnel platform location.

- 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"[7].

- 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 onsite poison verification.

The poison coupons used in the surveillance program will be representative of the material used. 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 12 of the jacketed poison specimens shall be suspended from rigid straps so designed as to be hung on the outside periphery of a rack module. There are two sets of these straps. 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[4]. 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 x 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, CP Co will evaluate this information and will modify the surveillance program as determined warranted and justified.

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

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

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

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

5. 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).

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

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

8. WCAP 9245, The Westinghouse Electric Corporation Nuclear Energy Systems Nuclear Service Division Quality Assurance Program Plan, Revision 8, December 1984.

9. Palisades FSAR Update.

10. Consumers Power "Quality Assurance Program Description for Operational Nuclear Power Plants," CPC-2A, Rev 4.

SECTION 4

TABLES

TABLE 4-1

Rack Module Data

	Region II
Number of Storage Locations	470*
Number of Rack Arrays	2 (11 x 11) 2 (11 x 7) 1 (7 x 6) 1 (6 x 6)
Center-to-Center Spacing (Inches)	9.17
Cell I.D. (Inches)	9.00
Type of Fuel	CE 15 x 15 Exxon 15 x 15
Rack Assembly Dimensions (Inches)	(11 x 11) 102 x 102 x 153 (11 x 7) 102 x 65 x 153 (7 x 6) 65 x 56 x 153 (6 x 6) 56 x 56 x 153
Dry Weights (lbs) Per Rack Assembly	13,300 (11 x 11) 8,500 (11 x 7) 4600 (7 x 6) 4000 (6 x 6)

*Plus 4 locations inaccessible due to water inlet pipe.

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.

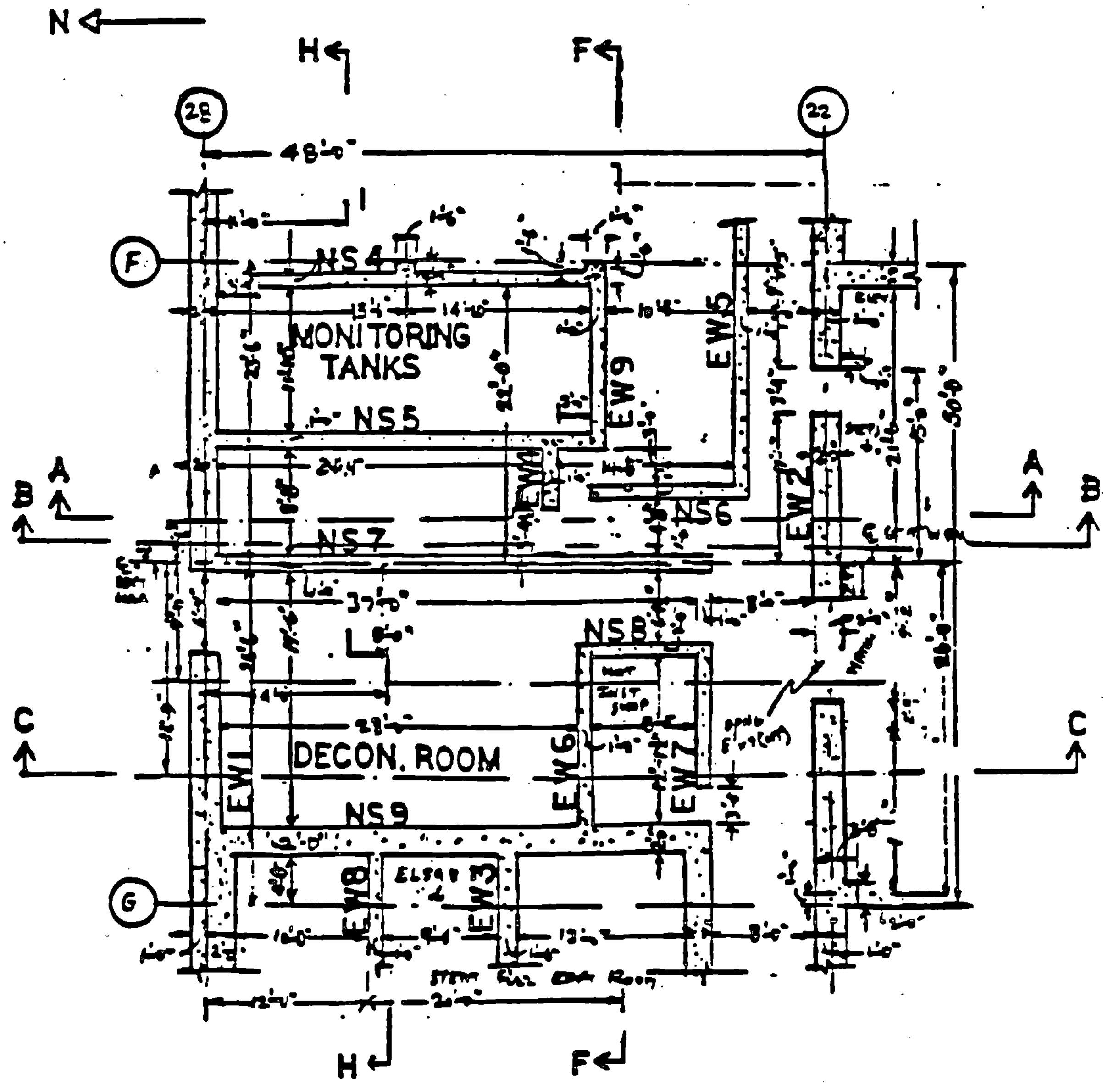
2. The provisions of NF-3231.1 of ASME Section III, Division I, shall be amended by the requirements of Paragraph c.2.3 and 4 of Regulatory Guide 1.124, entitled "Design Limits and Load Combinations for Class A Linear-Type Component Supports."

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.

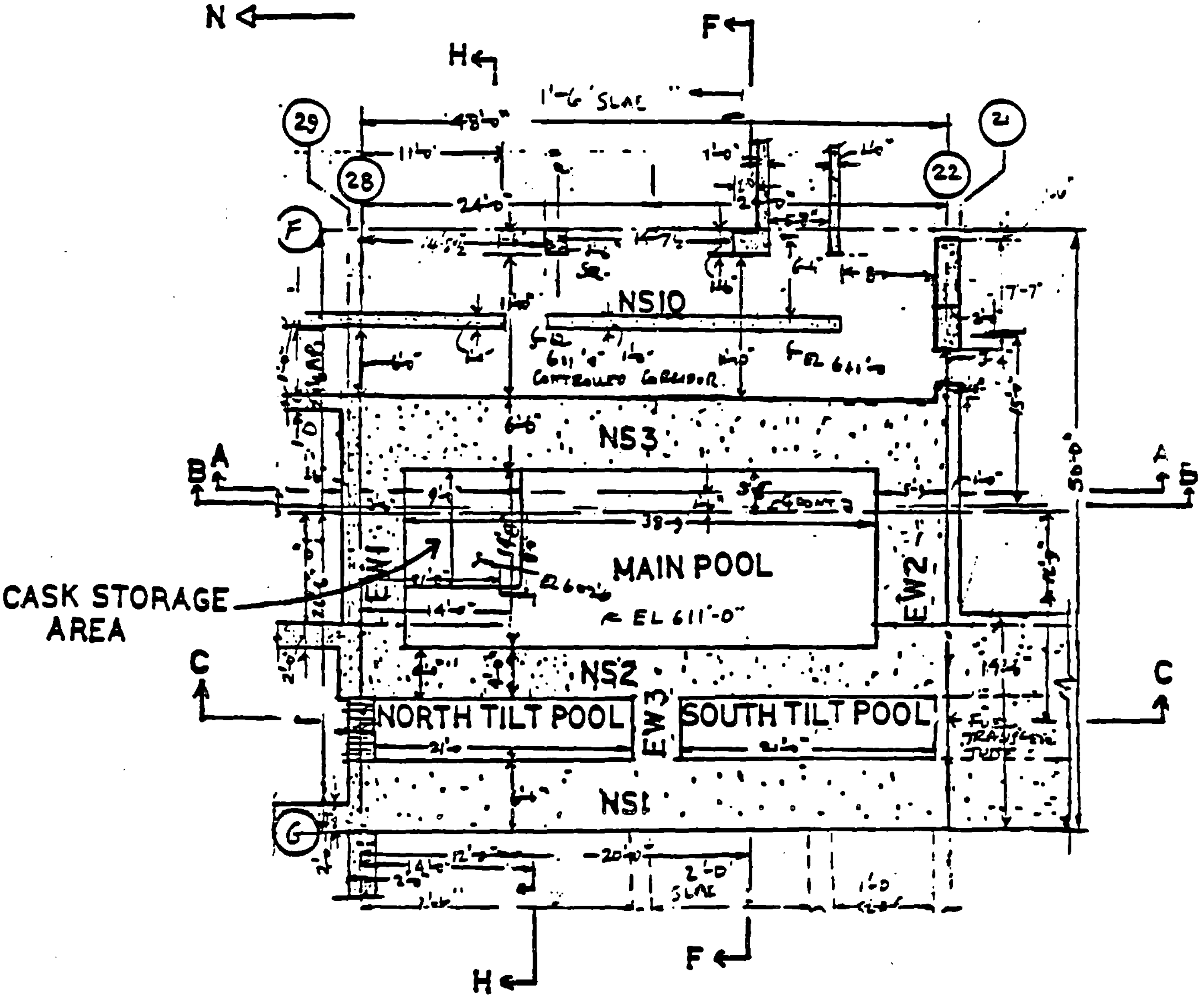
SECTION 4

FIGURES



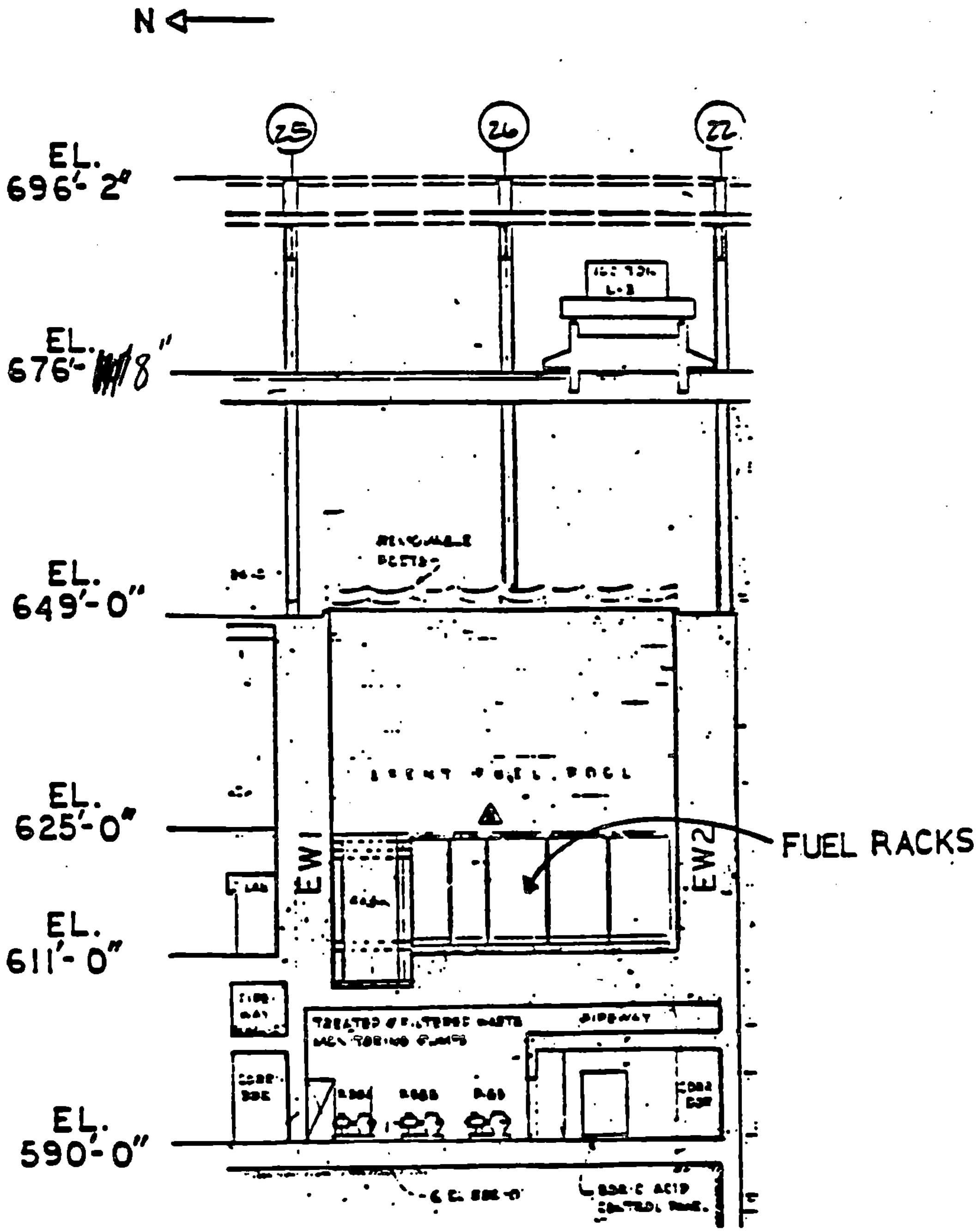
PLAN AT EL. 590 FT

FIGURE 4-1



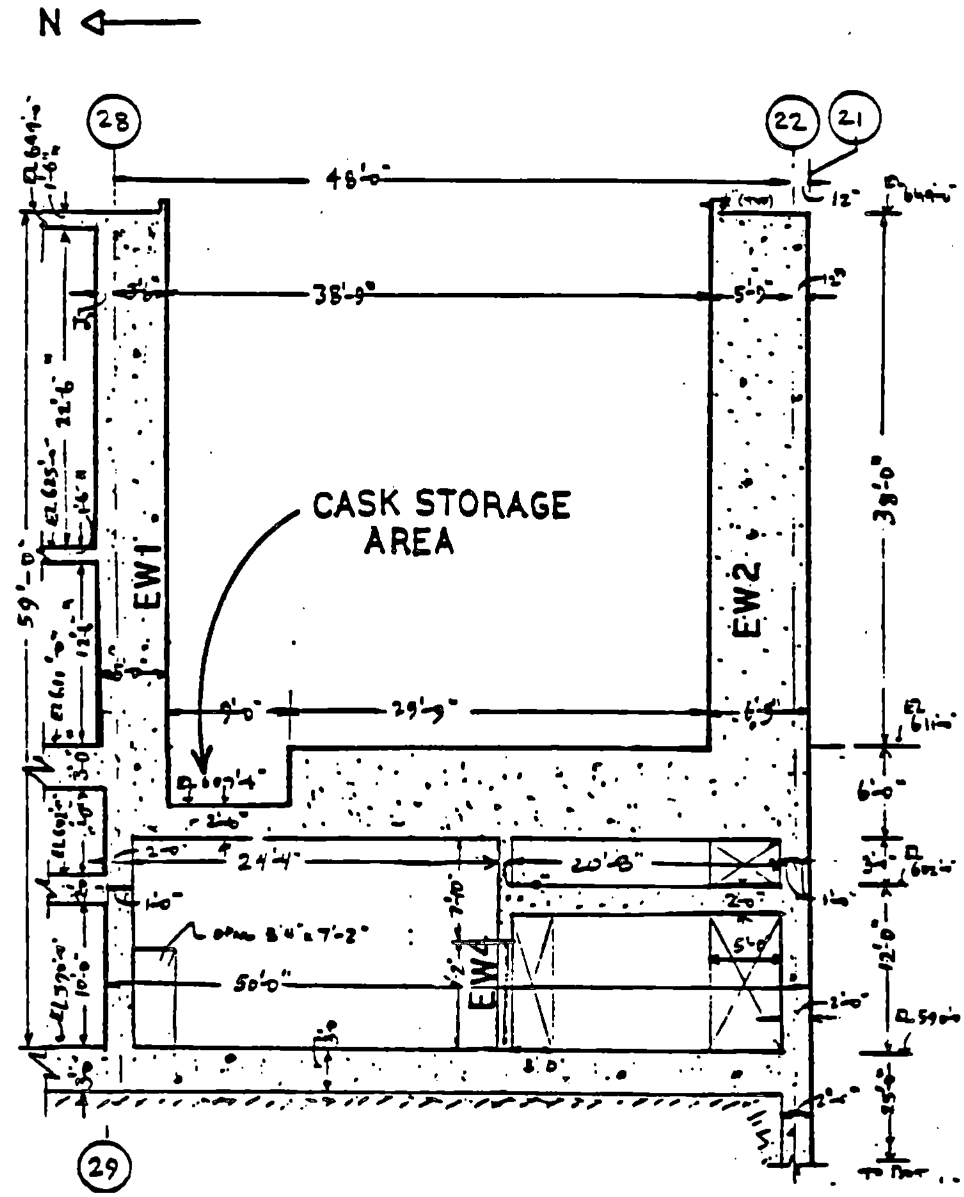
PLAN AT EL. 611 FT

FIGURE 4-2



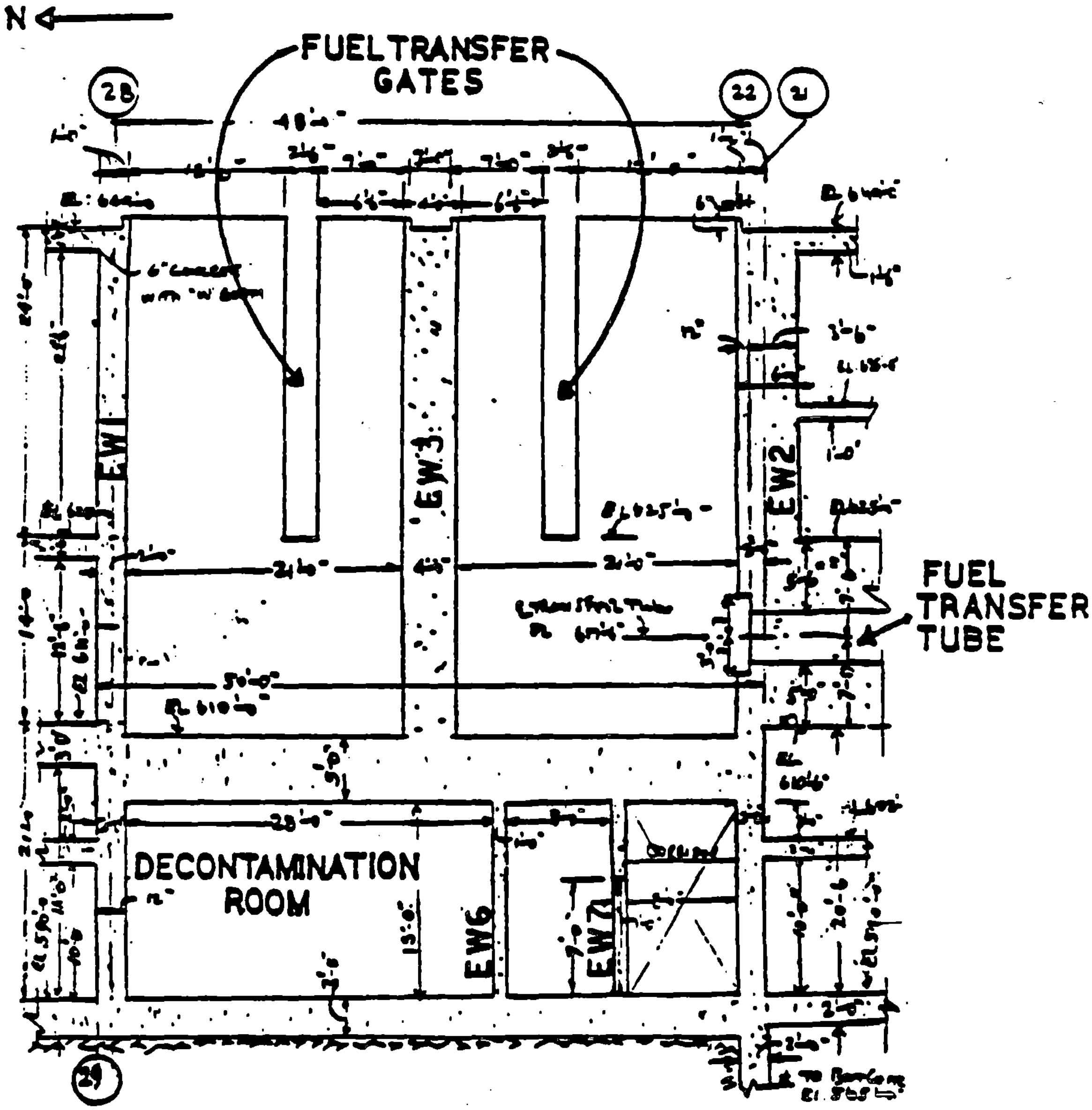
SECTION A-A - EL. 590 FT TO 696 FT

FIGURE 4-3



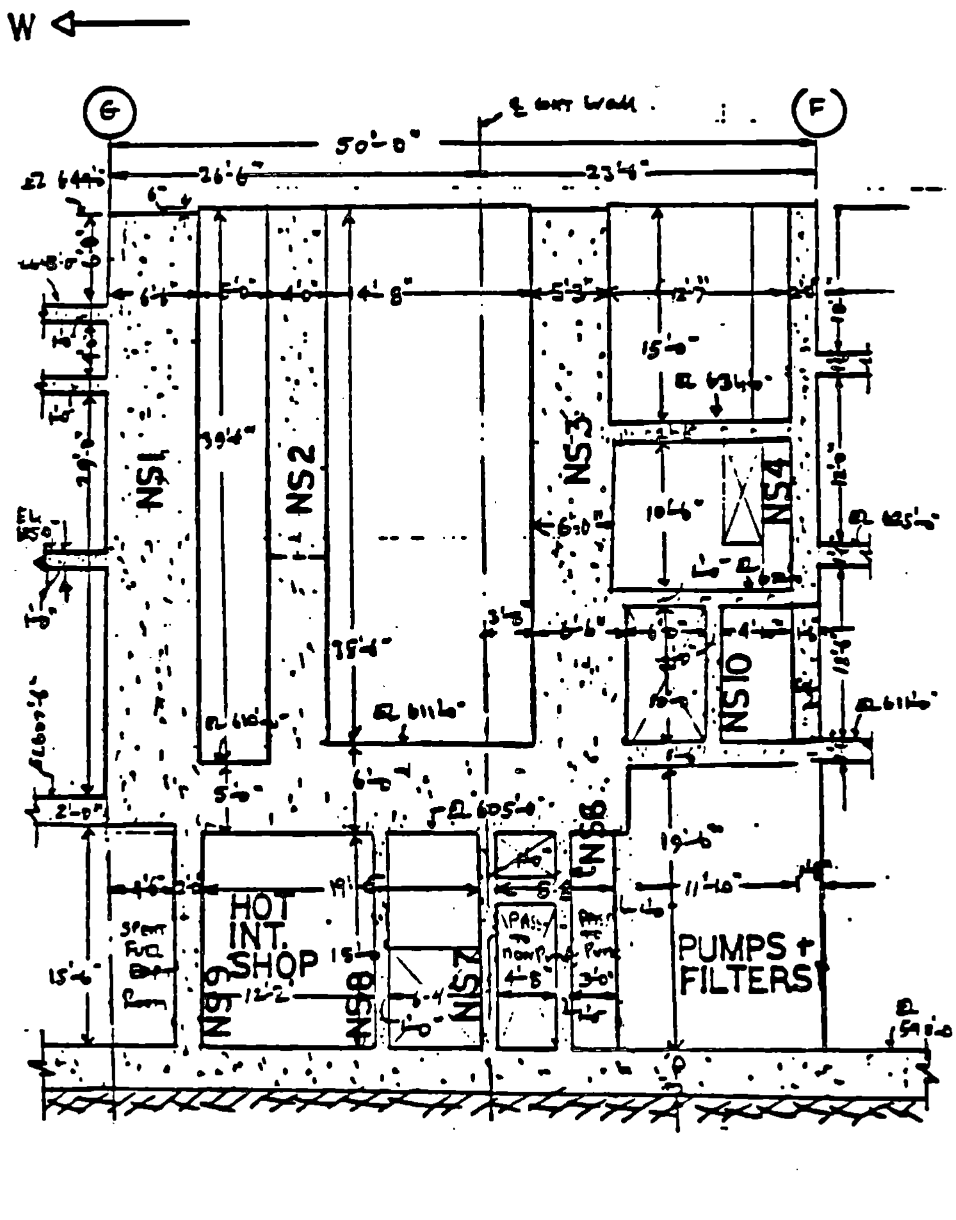
SECTION B-B - EL. 590 FT TO 649 FT

FIGURE 4-4



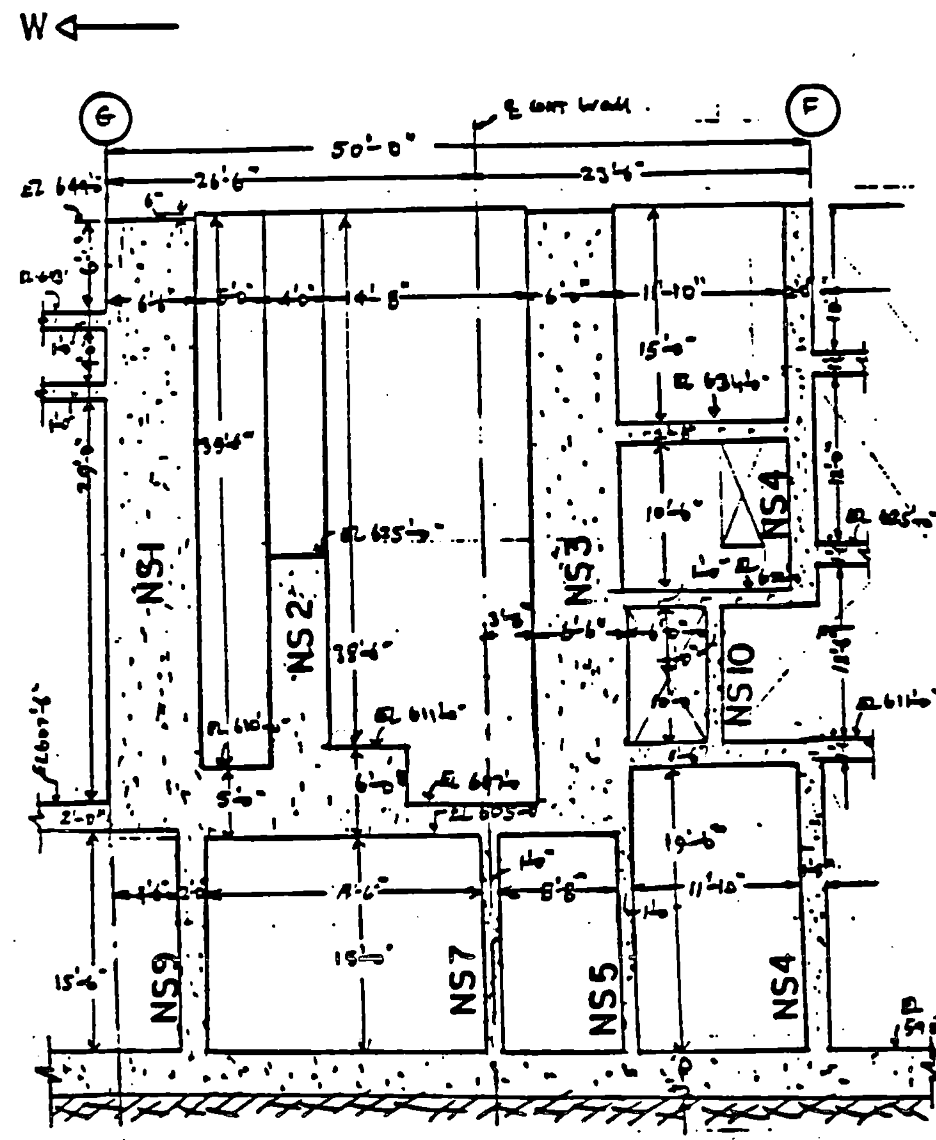
SECTION C-C - EL. 590 FT TO 649 FT

FIGURE 4-5



SECTION F-F - EL. 590 FT TO 649 FT

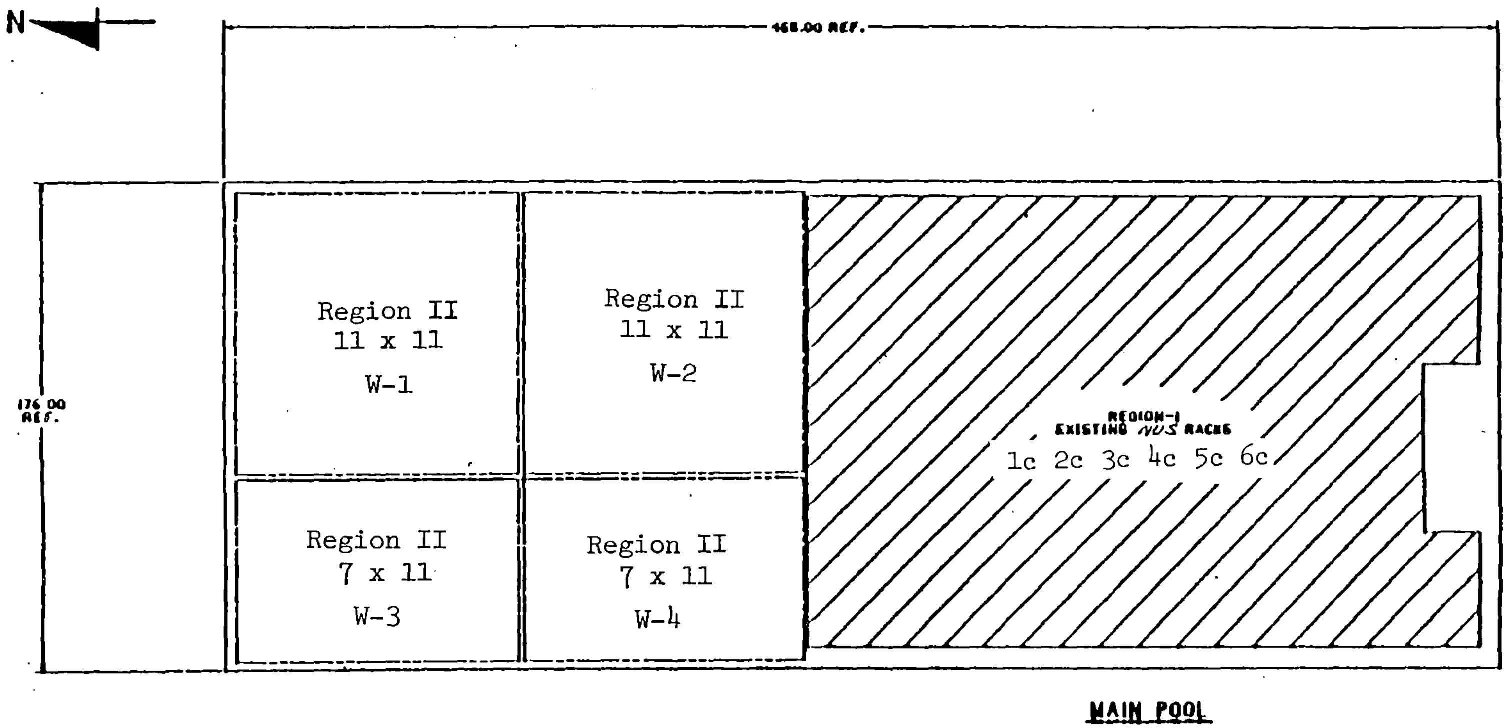
FIGURE 4-6



SECTION H-H - EL. 590 FT TO 649 FT

FIGURE 4-7

FIGURE 4-8



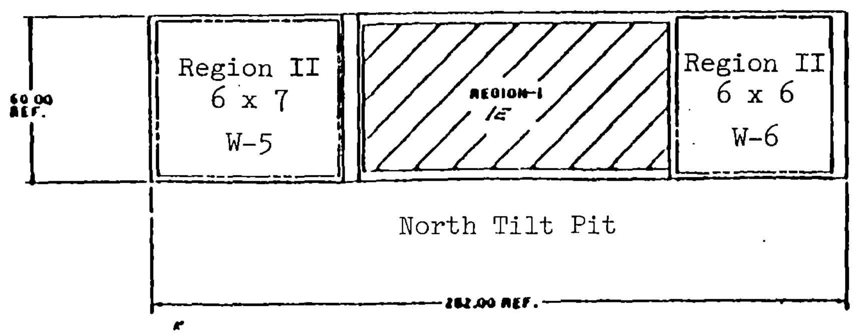
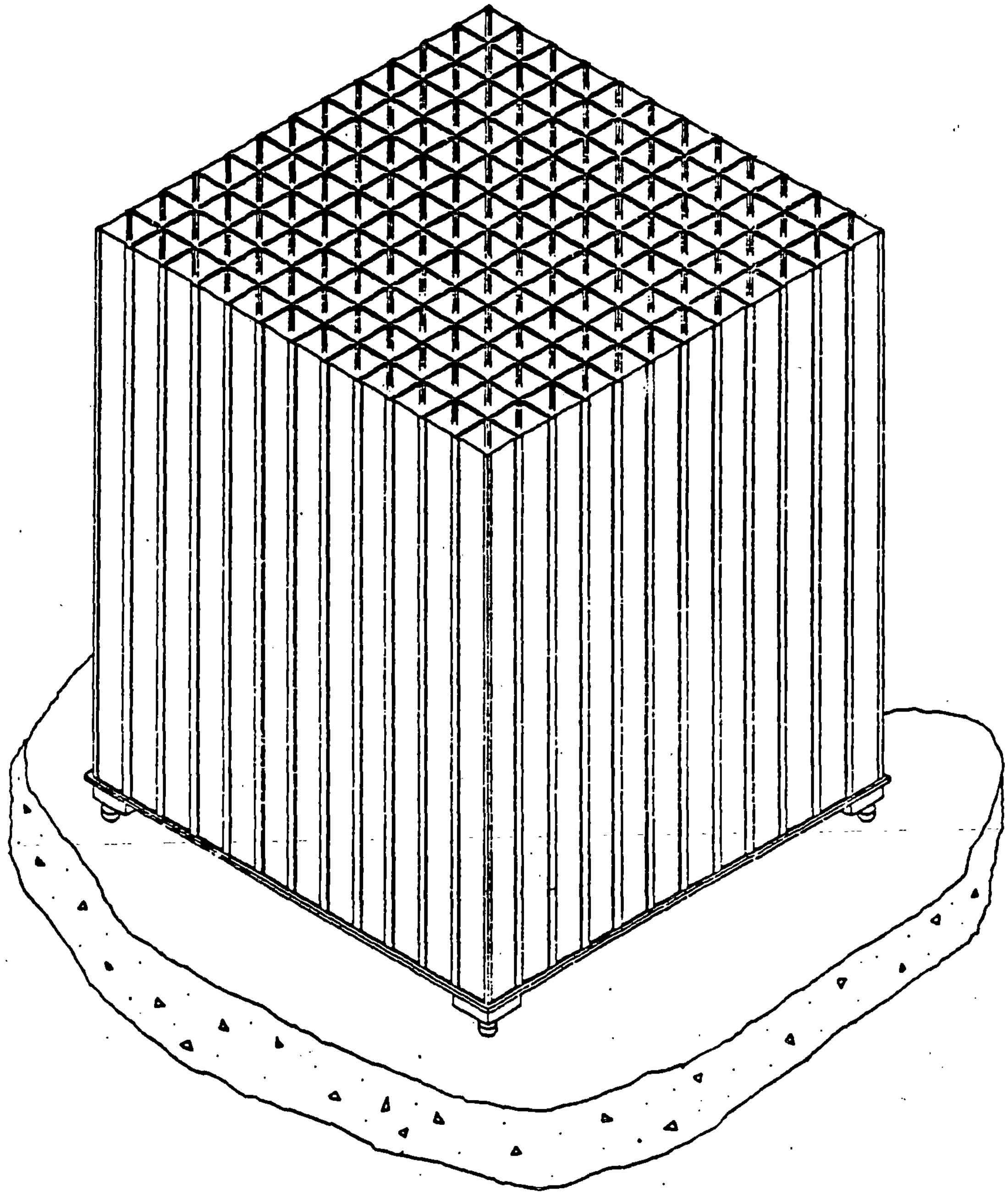
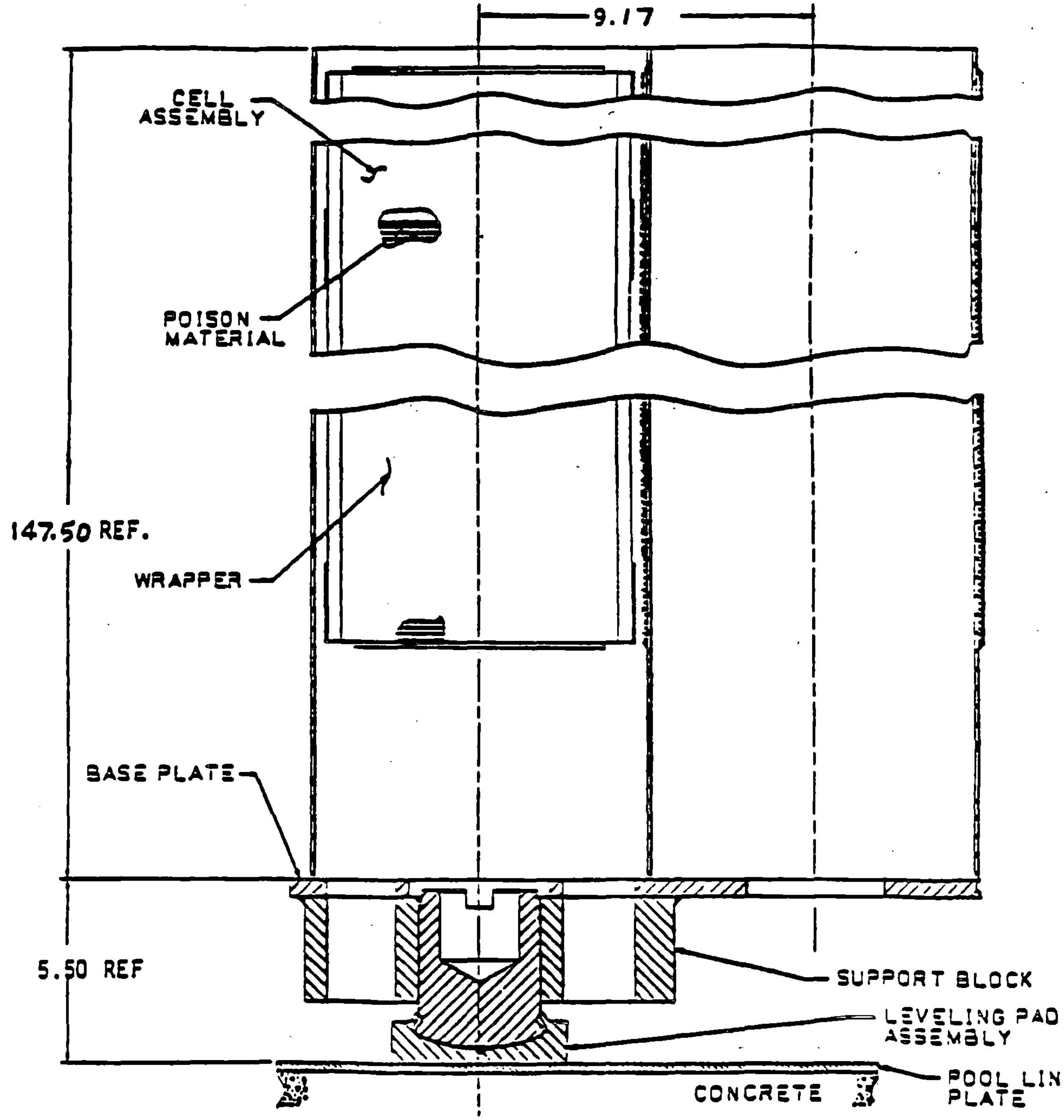


FIGURE 4-8
SPENT FUEL POOL ARRANGEMENT



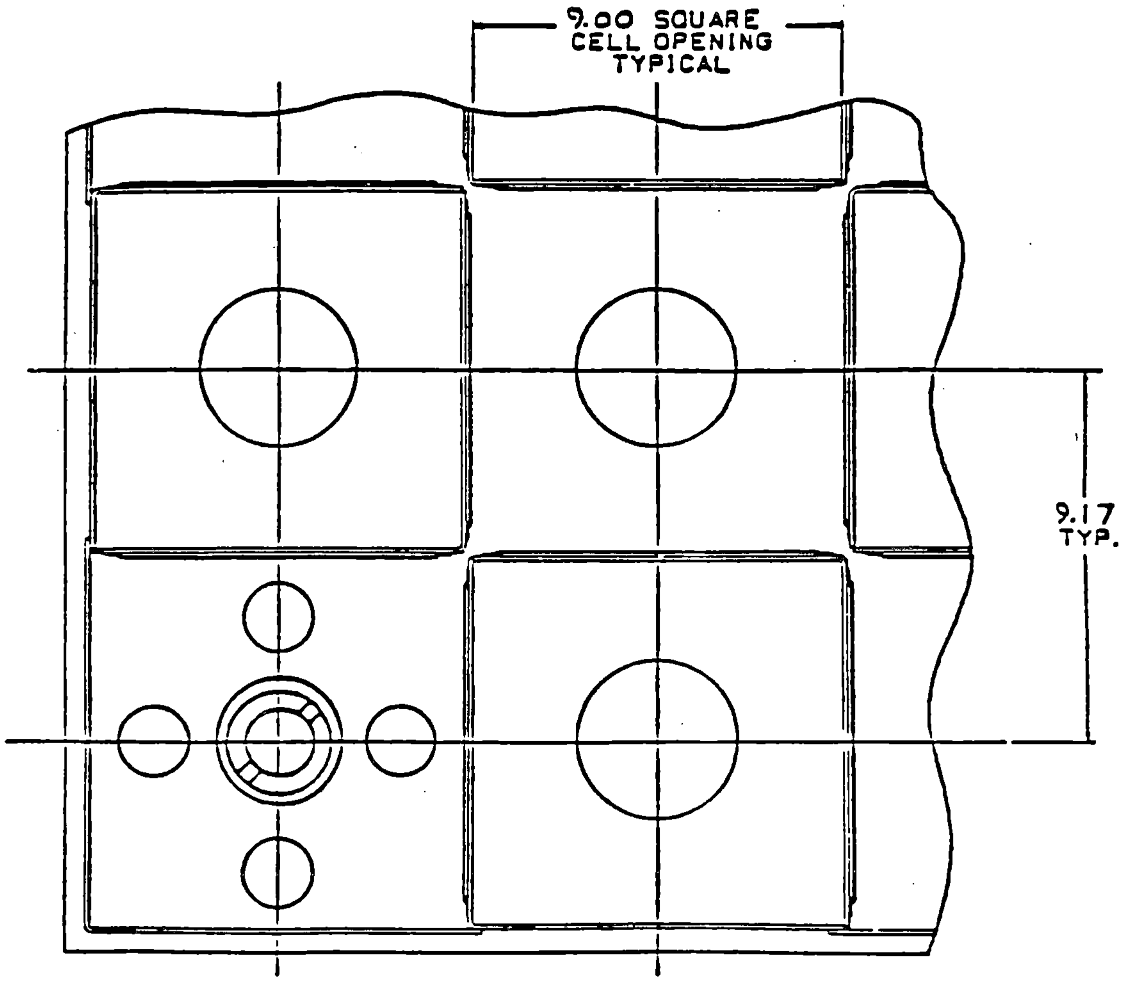
REGION II
FUEL STORAGE RACK MODULE

FIGURE 4-9



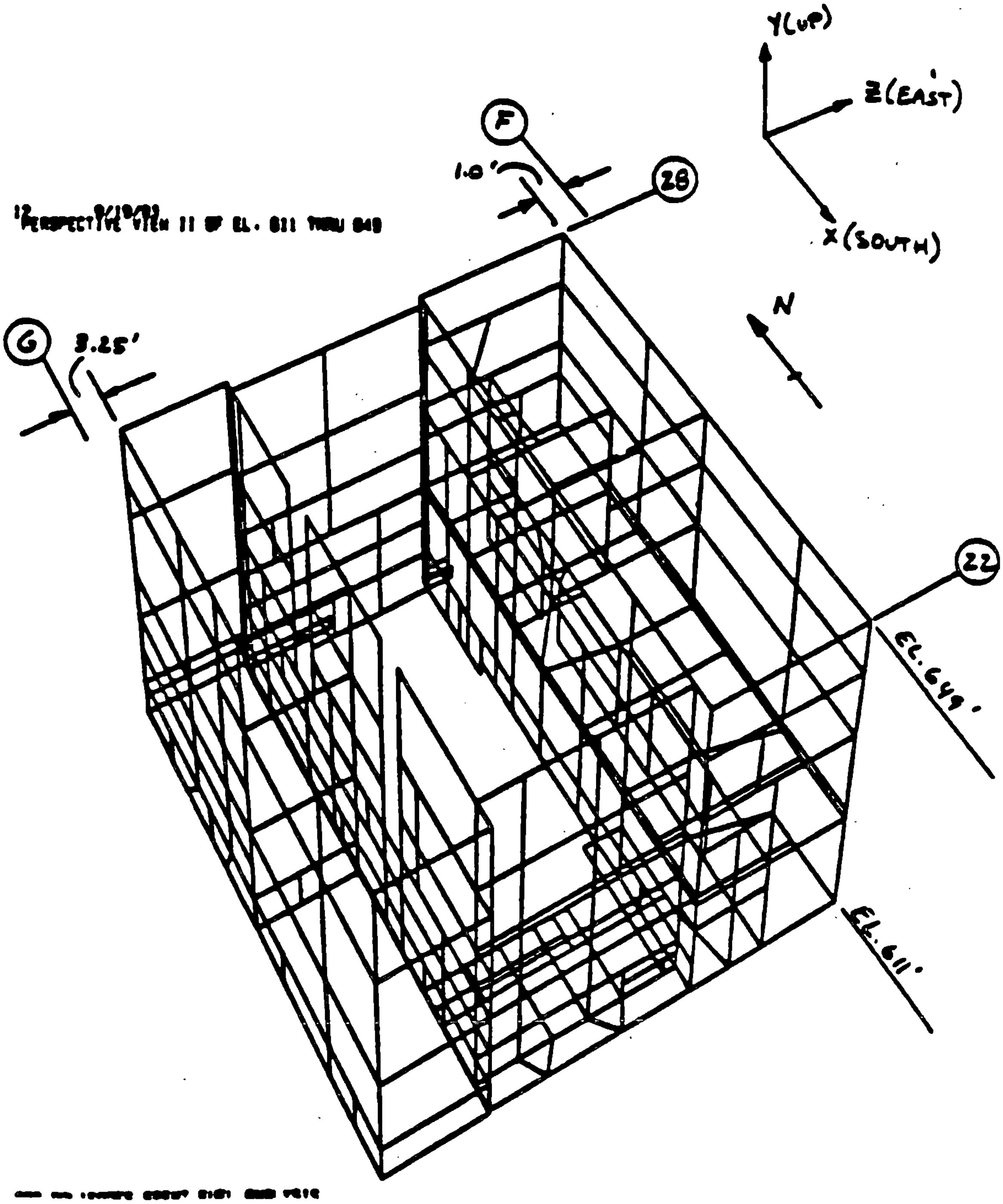
REGION II
MODULE CROSS-SECTION

FIGURE 4-10



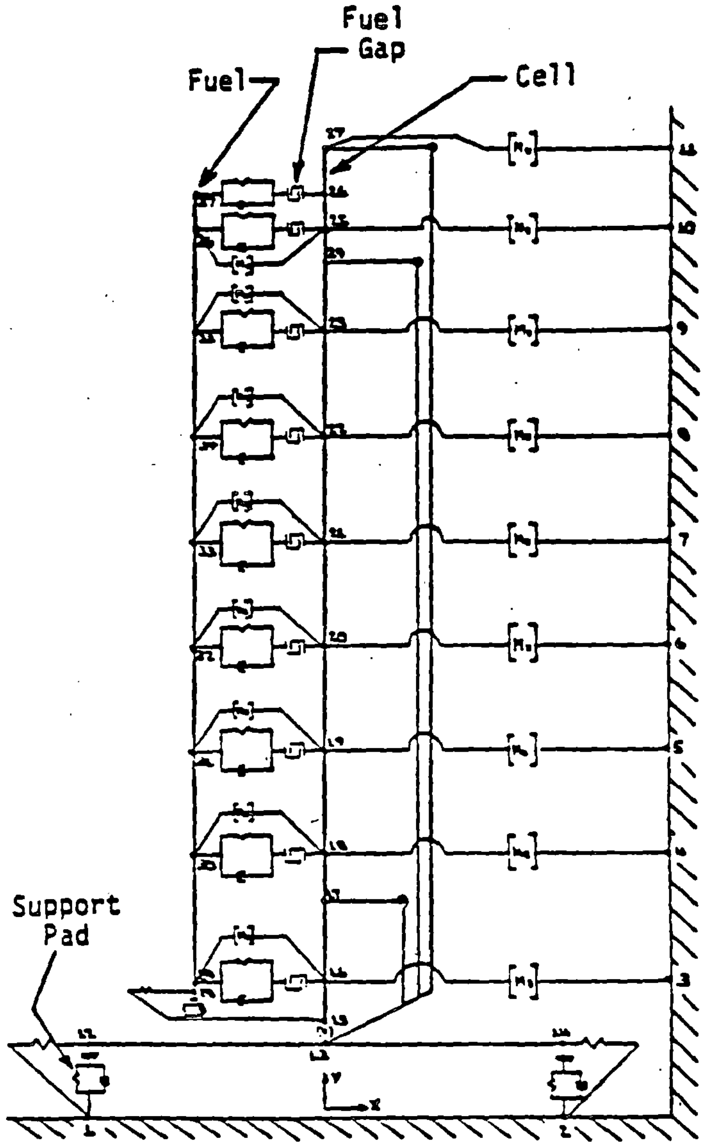
REGION II
MODULE TOP VIEW

FIGURE 4-11



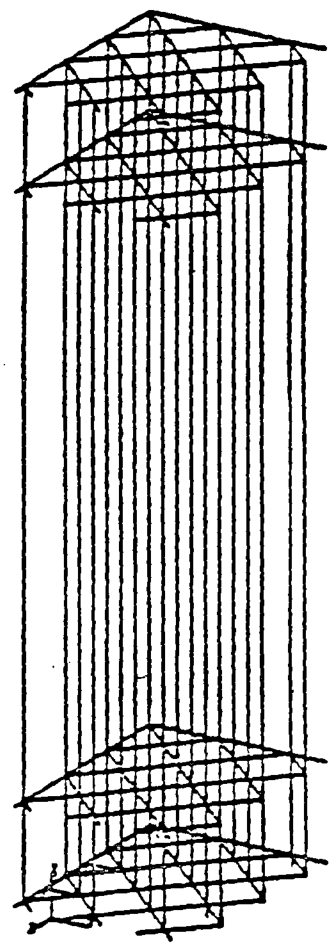
PERSPECTIVE VIEW - EL. 611 FT THRU 649 FT.

FIGURE 4-12



NONLINEAR SEISMIC MODEL

FIGURE 4-13



STRUCTURAL MODEL

FIGURE 4-14

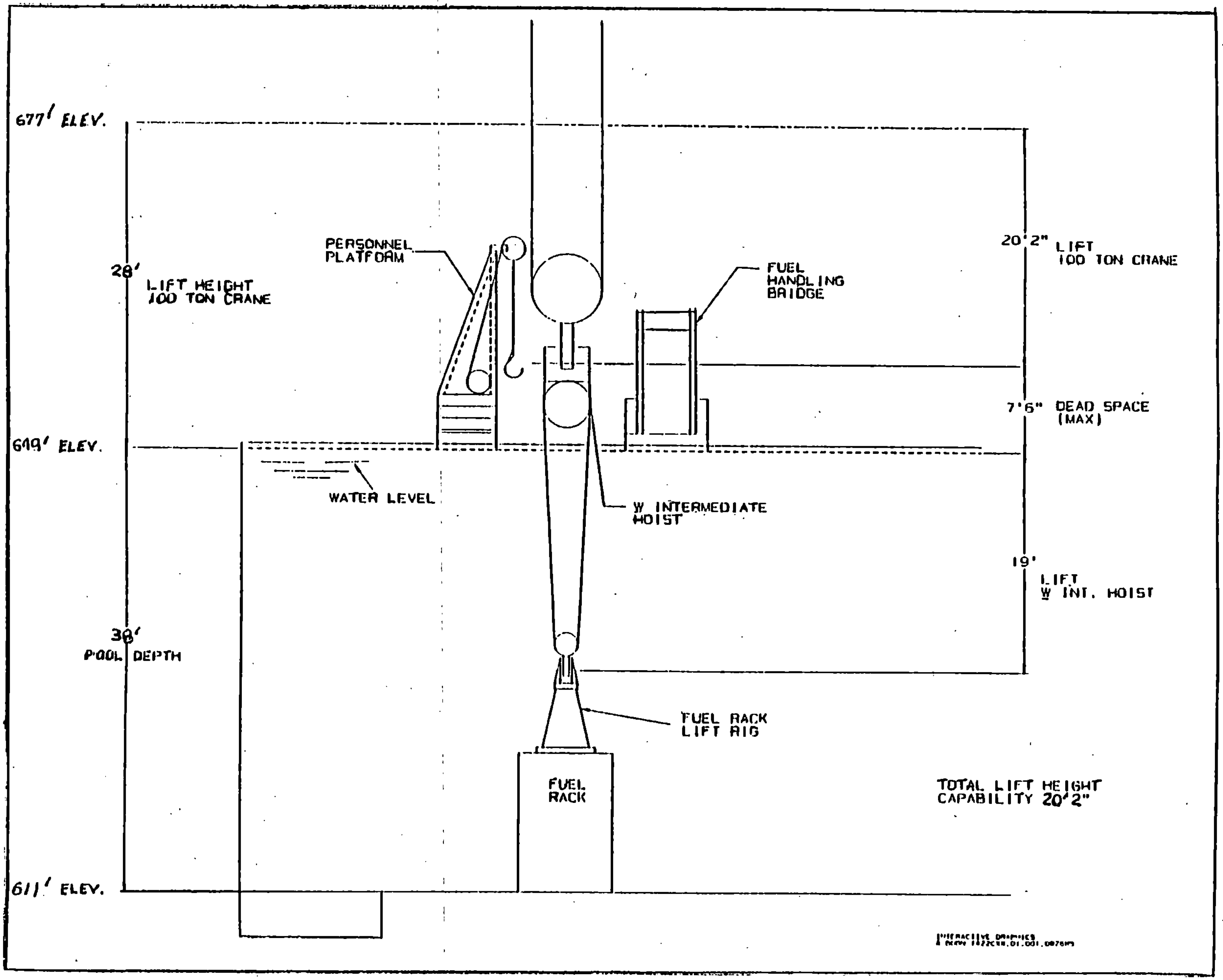


FIGURE 4-15
Intermediate Hoist and Platform Location

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. CPC currently has no contractural arrangements with any fuel reprocessing facilities.

B. At Palisades, the spent fuel pool has been previously reracked to 798 cells. Four of these cells are not usable for spent fuel because they contain other components (Table 5-2).

   Table 5-1 includes proposed refueling schedule and the expected number of fuel assemblies that will be transferred into the spent fuel pool at each refueling until the total existing capacity is reached. Full Core Reserve (FCR) is based on the number of usable cells.

C. As of December 31, 1985, the Palisades spent fuel pool contained 545 spent fuel assemblies.

D. At present, the storage of components other than fuel has affected the total number of available storage locations in the pool. The pool contains cells (defined in item B) that are usable for fuel storage, but are used to store any components other than fuel. An itemized list of components stored in each cell 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 onsite. Spent fuel could be sent offsite 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 onsite will remain there.

F. Table 5-3 references the spent fuel storage capacity for the Palisades spent fuel pool after reracking. Based on the present CPC fuel management policy, the Palisades spent fuel pool will lose FCR when Cycle 10 begins in 1989. (NOTE: This is a conservative estimate based on 12-month operating cycles, 3-month refueling outages, and assuming no other major plant modifications take place which would extend one or more refueling periods.)

5.1.2 ESTIMATED COSTS

The cost associated with the proposed Palisades spent fuel pool modification is estimated to be in the neighborhood of four and one-quarter million dollars. This figure includes items such as (1) design, engineering, manufacture, and installation of new spent fuel storage racks, (2) removal and offsite disposal (as low level radioactive waste) of the existing spent fuel storage racks, and (3) allowance for funds used during construction. Estimated cost escalation is included in this sum.

5.1.3 CONSIDERATION OF ALTERNATIVES

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

B. At the present time, there are no existing independent spent fuel storage facilities available to store Palisades spent fuel. There are no firm commitments by either firms or government agencies to construct or operate an independent spent fuel storage facility. In addition, cost and/or schedule considerations and State of Michigan laws make an independent spent fuel storage facility onsite unacceptable to meet the spent fuel storage needs at Palisades.

C. Replacement power costs, if the Palisades plant were to be shut down due to lack of spent fuel space, are indicated in Table 5-4. Plant shutdown would place a heavy financial burden on Michigan residents within CPC's service area and cannot be justified.

5.1.4 RESOURCES COMMITTED

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

The materials used for new rack fabrication are discussed in 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 INTRODUCTION

To evaluate the radiological impact of expanding the capacity of the Spent Fuel Pool (SFP), it is illustrative to look at the historical trends. During the 14-year operating history of the plant, the population of the Spent Fuel Pool was as illustrated in Table 5-5.

The dates in Table 5-5 represent the end of fuel movements in and out of the SFP during refueling outages and special maintenance outages. Two of these outages were full core offloads: late 1973 and late 1983. In 1976 the entire core was placed in the SFP for fuel inspection. Since then, one-third of the core (68 bundles) have been added to the SFP inventory with each refueling outage. In 1983, the 10-year Inservice Inspection program required a full core offload but the net increase in SFP population following refueling was 68 bundles.

5.2.2 SOLID RADIOACTIVE WASTE

During the time since 1976, the resin in the SFP demineralizer has been changed approximately on an annual basis. About 50 cubic feet of dewatered resin is produced as a result of this operation.

The rate at which the resin is replaced and the volume of solid radioactive waste produced have been constant even as the SFP inventory has increased. This trend is expected to continue with future additions to the SFP because of the chemistry controls applied to the primary coolant system and the high integrity of the fuel cladding - factors which limit the amount of activation and fission products entering the Spent Fuel Pool water.

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

5.2.3 GASEOUS RADWASTE

KRYPTON-85

Gaseous effluents specific to the spent fuel building are mixed with those from the auxiliary building during normal operations and with those from containment during outages. Monitoring for radioactivity occurs after the gases are mixed. As a result, only the total amount of krypton-85 released is measured.

In 1983, a total of 3.124 curies of krypton-85 were released. In 1984, 0.0265 curies of krypton-85 were released. By way of comparison, the Palisades capacity factor in 1983 was 0.678 and 0.146 in 1984.

5.2.3.1 WATER ANALYSIS

The most recent (3/10/85) analysis of radionuclides in the SFP water is given in Table 5-6.

These values are typical of those formed shortly before the SFP demineralizer resin is changed.

Figure 5-1 shows the historical trend of Cesium-137 concentration in the SFP. If adding fuel to the SFP has an impact on the ability to clean the water, then a long-term increase in the concentration of Cesium-137 would be expected. Instead, it is apparent that the ability to clean the water is independent of the number of fuel bundles in the SFP. It is dependent instead on the demineralizer resin cycle. The Cesium-137 concentration increases from 1 E-5 μCi/ml to 5 E-3 μCi/ml as the demineralizer resin is exhausted. After the resin has been changed, the concentration drops to 1 E-5 μCi/ml again.

The other isotopes behave similarly although the concentrations are one to two orders of magnitude lower than Cesium-137.

5.2.4 EXTERNAL DOSE RATE

To monitor dose rates in the SFP area, thermoluninescent dosimeters (TLD) have been mounted on a wall adjacent to the SFP since the beginning of plant operations. Table 5-7 lists the doses recorded each quarter from 1979 through 1984.

Comparison of Table 5-7 and 5-5 gives the following conclusions:

1. Increasing the number of fuel bundles stored from 273 to 613 has not produced any upward trend in quarterly exposures.

2. Only three calendar quarters had recorded exposures greater than 2 Rem. These quarters (1979-4, 1981-3, 1983-4) are all associated with refueling outages. These exposures can be attributed to refueling operations and special maintenance operations not performed during normal operations.

3. The average exposure during nonoutage quarters is 1.232 Rem which corresponds to an average rate of 0.56 millirem per hour. The average exposure during refueling outage quarters is 3.257 Rem corresponding to an exposure rate of 1.49 millirem per hour. The overall average exposure is 1.508 Rem or 0.69 millirem per hour.

The dose rate directly above the SFP has been measured during routine area surveys on the service platform. Survey sheets were examined for the periods of time between 1975 and 1983 during which the plant was operating. Thirteen surveys were found with a record of the dose rates on the service platform directly above the SFP. These measurements ranged from 0.2 to 3.5 millirem per hour. The average dose rate was 1.5 millirem per hour. As with the TLD results, there is no correlation between the dose rate and the number of fuel bundles in the SFP. Table 5-8 gives the dates of the surveys and the survey results.

5.2.5 AIR ANALYSIS

Table 5-9 gives the most recent (3/11/85) analysis of airborne radionuclides in the SFP area.

Figure 5-2 shows the gross beta-gamma activities from SFP airborne samples during the normal operating periods from 1979 through 1983. As with the other parameters examined thus far, there is no correlation between the gross airborne activities and the number of fuel bundles in the SFP.

5.2.6 WHOLE BODY DOSE RATE FROM AIRBORNE RADIONUCLIDES

The whole body dose rate from a semi-infinite cloud of radionuclides is given in Meteorology and Atomic Energy - 1968 as

$$D_\gamma = 0.25\ \bar{E}_\gamma\ \chi$$

where

D_γ = dose rate in rads/sec

$\bar{E}_\gamma$ = average gamma energy in MeV/disintegration

χ = radionuclide concentration in μCi/ml

To convert the dose rate to millirad per hour, the results from the equation above are multiplied by 3.6E6 millirad per hour/1 rad per sec.

5.2.6 WHOLE BODY DOSE RATE

The whole body dose rate in the SFP area from the radionuclides given in Table 5-9 is:

$$\begin{aligned} D_\gamma &= (0.25)(3.6E-6)[(0.0453)(2.54E-10)+(0.248)(7.06E-11)+(0.598) \\ &\quad (6.53E-11)] \\ &= 6.2E-5 \text{ millirad per hour.} \end{aligned}$$

Clearly, this dose rate is negligible compared to the background whole body dose rate of 0.69 millirem per hour stated above.

The dose rate from airborne radionuclides at the site boundary is also negligible.

5.2.7 RADIATION PROTECTION DURING RERACK ACTIVITIES

The radiation protection aspects of the spent fuel pool modification are the responsibility of the Plant Superintendent of Health Physics. 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 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.

Contamination control measures are used to protect persons from internal exposures to radioactive material and to prevent the spread of contamination by the work process, personnel traffic and the movement of material and equipment contamination problems. Material and equipment will be monitored and appropriately decontaminated and/or wrapped prior to removal from the spent fuel pool area. The plant radiation protection staff 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 as can be practically achieved.

5.2.8 ANTICIPATED EXPOSURES DURING RERACKING

Table 5-10 (Table provided later) is a summary of expected exposures for each phase of the reracking 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 reracking operation is conservatively estimated to be approximately 3.3 person-rem.

5.2.9 RACK DISPOSAL

The spent fuel storage and rack modules that will be removed from the spent fuel pool will be decontaminated and disposed of as radioactive waste in accordance with existing Palisades Radwaste Procedures. The type of disposal will be based on the capability to decontaminate.

5.3 ACCIDENT EVALUATION

5.3.1 FUEL HANDLING ACCIDENT

The consequences of a fuel handling accident have been evaluated.

Fuel failure during reracking as a result of inadvertent criticality or overheating during transfer is highly improbable. Similarly, damage to a fuel bundle as a consequence of external forces is also improbable. Operating procedures prohibit the handling of heavy objects above the fuel storage racks. Inadvertent disengagement of the fuel bundle from the fuel handling machine is prevented by interlocks; consequently, the probability of dropping and damaging a fuel bundle is low.

For the purpose of defining the upper limit of the consequences of a fuel handling accident, it is assumed that the fuel bundle is dropped during handling. Because of interlocks and procedural and administrative controls, such an event is unlikely. However, if the bundle is damaged to the extent that a number of fuel rods fail, the accumulated fission gases and iodines in the fuel rod gap could be released into the surrounding water.

The fuel bundles are stored in the spent fuel storage rack which are at the bottom of the spent fuel pool. Because of the configuration and construction of the fuel storage racks, a dropped fuel bundle can strike no more than four fuel bundles in the storage racks. The most limiting fuel bundle drop onto stored fuel would be that where a bundle is dropped directly onto one stored bundle. Impact can occur only between the ends of the involved fuel bundles (the bottom end fitting of the dropped fuel bundle striking the top end fitting of a stored fuel bundle). The results of analyses of the energy absorption capability of the fuel bundles show that a fuel bundle is capable of absorbing the kinetic energy of the drop without causing any fuel rod failures. The worst fuel handling accident with respect to radioactive release that could occur in the spent fuel pool is when a fuel bundle is dropped onto the spent fuel pool floor. After striking the pool floor, the bundle would rotate from the vertical position into a horizontal attitude. During this rotation, it is postulated that the bundle strikes a protruding structure member of the fuel storage rack. The fuel storage rack is designed without such protruding structures and, hence, the shape and nature of the assumed member is indeterminate.

To estimate the number of fuel rod failures for this mode, the energy required to crush a fuel rod sufficiently to cause failure is determined. The additional energy required to bend the entire assembly is also determined. Using the computed energy to crush an individual fuel rod and the energy absorbed by bending of the entire fuel assembly, the energy required to fail one complete row of fuel rods is determined. The point of impact is assumed to occur at the most effective location for fuel rod damage; i.e., center of percussion, and the load is assumed to be a line load. Resistance to crushing by the fuel pellet is considered in the analysis. The crushing failure mode for the fuel tube is considered to require the least energy absorption; hence, the model results in a conservative upper limit for the number of fuel rod failures. Further, resistance offered by the guide bars is neglected.

Failure by bending is not a credible mode of failure for the fuel rods. More bending energy is required to fail a fuel rod than is available. However, to fail more than one layer of fuel rods requires that the layers subsequent to the outer row of fuel rods requires that the layers subsequent to the outer row of fuel rods fail by bending rather than crushing since it is not possible to apply the line load to layers of fuel rods beyond the first row.

Approximately 43,500 in-lb of kinetic energy from rotation must be absorbed. The energy required to bend the assembly and crush the outer row of fuel rods to failure is 15,500 in-lb. To fail the second row of fuel rods, more than 60,000 in-lb of energy is required, which is greater than the kinetic energy originally available. Hence, it may be concluded that as much as one complete outer row of fuel rods (13 fuel rods) may fail in the event of a fuel handling incident but that insufficient kinetic energy is available to cause further failures.

The fission product activity in the fuel rod gap was determined for the average fuel rod having a residence time of three full power years at 2,650 MWt. The results were then multiplied by 1.65 to accommodate maximum potential radial peaking for the highest power fuel rods.

The fuel assembly was assumed dropped at two days after reactor shutdown and all the gap activity was released to the water. A credit was taken for partial retention of the iodines in the water by the application of an effective decontamination factor of 100.

The spent fuel area of the auxiliary building is exhausted via a charcoal filter. With a decontamination factor of 10 (90% efficient), the thyroid dose would be reduced to 0.62 rem.

The potential offsite doses resulting from a credible fuel handling accident in the spent fuel pool area are less than the guidelines of 10CFR100.

5.3.2 HEAVY LOAD DROP INTO THE SPENT FUEL POOL

Although carrying heavy loads over the main spent fuel pool is prohibited by Administrative Control, analyses in Reference 4 and in Section 3.1.2 of this Safety Analysis indicate that K_{eff} would remain less than or equal to .95 for postulated load drop accidents.

Analysis of the release of radioactivity caused by a worst case heavy load drop on the main fuel pool shows that the increase in release of radioactivity due to increasing the storage capacity, as described in this Safety analysis, is less than or equal to .02 Rem.

5.4 REFERENCES

1. Palisades FSAR Update.

2. Consumers Power Company Standard Reference Data Book - March 1984.

3. Palisades Technical Specification 3.21.

4. Consumers Power Company letter dated November 1, 1976, D A Bixel to A Schwencer, Spent Fuel Pool Modifications

5. NRC letter dated June 30, 1977, A Schwencer to D A Bixel, and the attached SER - Safety Evaluation by the Office of NRR Supporting Amendment No. 29 to Provisional Operating License No. DPR-20

6. NRC letter dated November 9, 1983, D M Crutchfield to D J VandeWalle, Control of Heavy Loads (Phase I) - NUREG-0612

SECTION 5

TABLES

TABLE 5-1

ESTIMATED SPENT FUEL CAPACITY REQUIREMENTS

PALISADES PLANT

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 - 7	--	545	204	753	45	0
8	5/15/87	613	204	821	0	23
9	8/15/88	681	204	889	0	91

During Cycle 9, and thereafter, an average of at least 68 bundles per cycle will be consolidated or removed from the pool for Dry Cask Storage. This process will be continued through Cycle 15 which will end in 1998. Capability for storage of a full core discharge will be maintained during this time period.

*Including 4 spaces filled with miscellaneous material.
**Conservatively based on a 12-month Operating cycle and a 3-month Refueling Outage.

TABLE 5-2

COMPONENTS STORED IN SPENT FUEL POOL

COMPONENT
1 Cannister Containing Fuel Rods
1 Dummy Bundle
1 Fuel Assembly Skeleton (No Fuel Rods)
1 Surveillance Capsule Carriage

TABLE 5-3

SPENT FUEL POOL CAPACITY AFTER RERACK

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 - 7	--	545	204	753	139	0
8	5/15/87	613	204	821	71	0
9	8/15/88	681	204	889	3	0

*Including 4 spaces filled with miscellaneous material.

TABLE 5-4

PALISADES PLANT

FORCED OUTAGE ANNUAL NET REPLACEMENT POWER COST

Year	Year Cost (Millions)
1990	$ 175.2
1991	209.2
1992	318.4
1993	274.9
1994	344.3
1995	494.6
1996	464.6
1997	537.1
1998	590.3
Total	$ 3,408.6

TABLE 5-5

SPENT FUEL POOL POPULATION HISTORY

DATE	# BUNDLES IN SFP
1/ 1/72	0
10/17/73	204
4/ 5/74	1
3/23/76	205
2/22/78	273
10/15/79	341
10/15/81	409
9/29/83	613
4/15/84	477
2/ 3/86	545

As of 2/3/86, with the reactor re-fueled for Cycle 7 operation, the number of bundles in the SFP remains at 545.

TABLE 5-6

SPENT FUEL POOL WATER ANALYSIS

ISOTOPE	CONCENTRATION (μCi/ml)
Manganese-54	1.4E-5
Cobalt-58	1.7E-5
Cobalt-60	2.1E-4
Cesium-134	7.1E-4
Cesium-137	2.1E-3

TABLE 5-7

THERMOLUMINESCENT DOSIMETER RESULTS

YEAR	QUARTER	REM	YEAR	QUARTER	REM
1979	1	1.700	1982	1	1.500
	2	1.950		2	N/A
	3	1.500		3	0.620
	4	3.100		4	0.970
1980	1	1.500	1983	1	1.575
	2	N/A		2	0.740
	3	1.000		3	0.730
	4	1.950		4	2.320
1981	1	1.450	1984	1	1.220
	2	0.700		2	0.800
	3	4.350		3	1.035
	4	1.350		4	1.125

TABLE 5-8

DOSE RATES ON THE SFP SERVICE PLATFORM

DATE	MREM/HR	DATE	MREM/HR
11/12/75	0.5	12/18/77	3.0
9/ 9/77	< 1.0	6/ 8/79	1.0
9/21/77	≦ 1.5	8/29/79	1.0
10/12/77	0.2	10/26/79	1.0
10/26/77	2.0	3/26/81	0.7
11/21/77	0.5	1/31/83	2.0
12/ 9/77	3.5		

TABLE 5-9

SFP AIR ANALYSIS

RADIONUCLIDE	CONCENTRATION (μCi/ml)	1 SIGMA ERROR (%)	$\bar{E}_{\gamma}$ MeV)
Xenon-133	2.54E-10	8.7	0.0453
Xenon-135	7.60E-11	18.3	0.248
Cesium-137	6.53E-11	29.6	0.598*

Sample Volume - 2.03E-6 Milliliters

*From Barium-137m

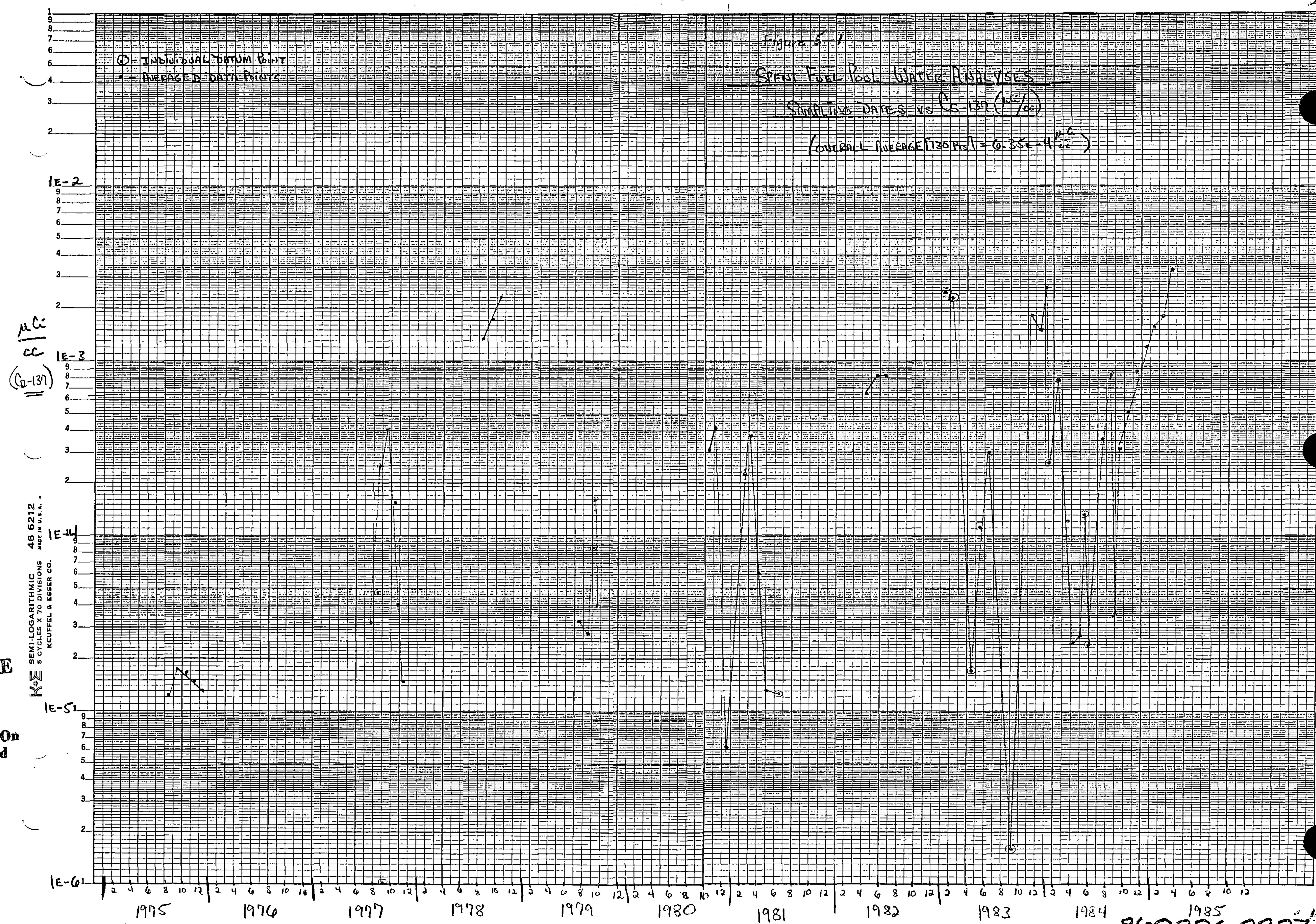
TABLE 5-10

ESTIMATED ALARA DOSES DURING RE-RACKING

		Person-Rem
Removal of existing racks (including decontamination and loading for shipment)		0.688
Prerequisite and cleanup (including fuel shuffle)		0.840
Installation of new racks		1.748
	Total	3.276

SECTION 5

FIGURES



TI APERTURE CARD

Also Available On Aperture Card

8602260327-01

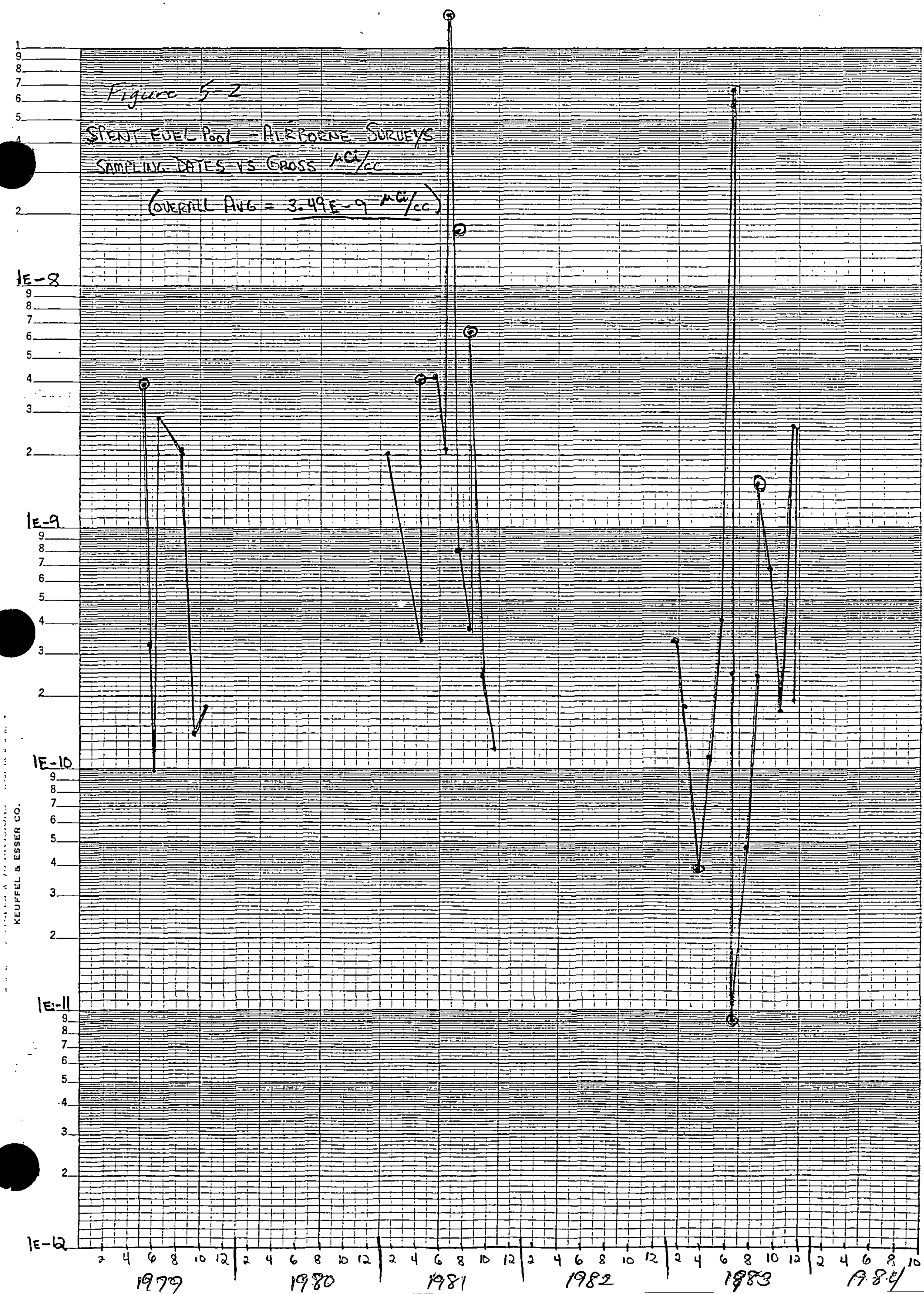


Figure 5-2
SPENT FUEL POOL - AIRBORNE SURVEYS
SAMPLING DATES VS GROSS µCi/cc
(OVERALL AVG = 3.49E-9 µCi/cc)
1E-8
1E-9
1E-10
1E-11
1E-12
1979
1980
1981
1982
1983
1984
KEUFFEL & ESSER CO.