

PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

SPENT FUEL STORAGE CAPACITY MODIFICATION
SAFETY ANALYSIS REPORT
DOCKET NOS. 50-277 AND 50-278

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1.0 INTRODUCTION

1.1 CURRENT STATUS

Philadelphia Electric Company (PECO) is currently pursuing the design and manufacture of new spent fuel storage racks to be placed into the spent fuel pools of Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3. The purpose of these new racks is to increase the amount of spent fuel that can be stored in the existing spent fuel pools. The racks are designed so that they can store spent fuel assemblies in a high density array.

This Safety Analysis has been prepared to support PECO's request for NRC review and approval of new spent fuel racks in accordance with PBAPS Units 2 and 3 Facility Operating Licenses DPR-44 and DPR-56(1).

There are two spent fuel pools at PBAPS; one for each nuclear unit. The existing racks in each of these pools have 2608 total storage cells. In the 1987-88 time frame, these units will lose their full-core discharge reserve storage capacity (764 assemblies); and in the 1991-1992 time frame, they will no longer have the capacity to store fuel discharges from the operating units. Therefore, to ensure that sufficient capacity continues to exist at PBAPS to store discharged fuel assemblies, PECO plans to replace the existing storage racks with new spent fuel storage racks whose design will allow for more dense storage of spent fuel, thus enabling the existing pools to store more fuel in the same space as occupied by the current racks.

1.2 SUMMARY OF REPORT

This Safety Analysis Report follows the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978, as amended by the NRC letter dated January 18, 1979(2). Sections 3.0 through 5.0 of this report are consistent with the section/subsection format and content of the NRC position paper, Sections III through V.

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

The nuclear and thermal-hydraulic aspects of the report (Section 3.0) address the neutron multiplication factor, considering normal storage and handling of spent fuel as well as postulated accidents with respect to criticality and the ability of the spent fuel pool cooling system to maintain sufficient cooling. Temporary fuel storage considerations during rack removal and installation are also addressed.

Mechanical, material, and structural aspects (Section 4.0) involve the capability of the fuel assemblies, storage racks, and spent fuel pool structure 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.

1.3 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 PBAPS Units 2 and 3 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 PBAPS Updated FSAR [3] and Operating Licenses.

1.4 REFERENCES

1. PBAPS Units 2 and 3, Facility Operating Licenses DPR-44 and DPR-56, Docket Nos. 50-277 and 50-278.

2. Nuclear Regulatory Commission, Letter to all Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", as amended by the NRC letter dated January 18, 1979.
3. PBAPS Units 2 and 3, Updated Final Safety Analysis Report, Docket Nos. 50-277 and 50-278.

2.0 SUMMARY OF EXISTING RACK DESIGN

The existing spent fuel storage racks have the capacity to store 2608 spent fuel assemblies for Unit 2 and 2608 for Unit 3. The rack modules consist of an array of storage cavities having nominal center-to-center spacing of 7.0 inches; each storage cavity can accommodate one fuel assembly. The fuel assembly storage cavities are structurally connected to form 26 freestanding spent fuel assembly storage modules per spent fuel pool.

The existing high density spent fuel racks are free-standing, all-anodized aluminum construction. Each rack consists of six basic components:

- a. top grid casting
- b. bottom grid casting
- c. poison can assemblies
- d. side plates
- e. corner angle clips
- f. adjustable foot assemblies

Each component is anodized separately. The top and bottom grids are machined to maintain nominal fuel spacing of 7.00 inches center to center within the rack and a spacing of 9.75 inches between centers of cavities in adjacent racks. Poison cans nest in pockets which are cast in every other cavity opening of the grids. This arrangement ensures that no structural loads will be imposed on the poison cans. The poison cans consist of two concentric square tubes with four Boral plates located in the annular gap. The Boral is positioned so it overlaps the fuel pellet stack length in the fuel assemblies by 1 inch at the top and at the bottom. The outer can is formed into the inner can at each end and seal welded to isolate the Boral from the spent fuel pool (SFP) water. Each can is pressure and vacuum leak tested. The grid structures are bolted and riveted together during fabrication by four corner angles and four side shear panels. Leveling screws are located at the rack corners to allow adjustment for variations in pool floor level of up to ± 1 inch. To maintain a flat, uniform contact area, the bearing pad at the bottom of each leveling screw is free to pivot.

3.0 NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

The nuclear criticality design considerations and analysis methods used for PBAPS are the same as those used by Westinghouse for several other nuclear power plants which the NRC has previously reviewed and found acceptable. Those plants are listed below:

<u>UTILITY</u>	<u>PLANT</u>	<u>FUEL TYPE</u>
Arkansas Power and Light	Arkansas 1 & 2	PWR
Carolina Power and Light	Shearon Harris 1, 2, 3, & 4	PWR & BWR
Carolina Power and Light	H. B. Robinson	PWR
Duke Power	Oconee 1, 2, & 3	PWR
Duke Power	McGuire 1 & 2	PWR
Florida Power and Light	Turkey Point 3	PWR
Gulf States Utilities	River Bend 1	BWR
Public Service of New Hampshire	Seabrook	PWR

3.1 NEUTRON MULTIPLICATION FACTOR

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack. This is done by maintaining a minimum separation between assemblies and using a fixed neutron absorbing material between adjacent fuel assemblies.

The design basis for preventing criticality 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/ANS-57.2-1983^[1] and in NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979⁽²⁾.

The codes, standards, and regulations (or pertinent sections thereof) used to meet the above design basis are listed below:

USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Proposed Rev. 2, Dec. 1981.

USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, "Spent Fuel Storage," Rev. 3, July 1981.

USNRC Branch Technical Position, CPB 9.1-1, "Criticality in Fuel Storage Facilities."

USNRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978 and modification dated January 18, 1979.

ANSI/ANS-8.1-1983, "Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors."

ANSI/ANS-57.2-1983, "Design Requirements for LWR Spent Fuel Storage Facilities at Nuclear Power Plants."

ANSI/ANS-52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants."

ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety."

3.1.1 Normal Storage

- a. The fuel assembly contains the highest enrichment authorized without any control rods or any noncontained burnable poison and is at its most reactive point in life. The GE 7 x 7 fuel

assembly is more reactive than the GE 8 x 8 or 8 x 8(R) fuel assembly. Table 3-1 lists the fuel parameters used in the analysis.

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

- b. The moderator is demineralized 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. There is no dissolved boron present in the water.
- c. 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. However, neutron absorber plates are not necessary on the periphery of the modular array because calculations show that this finite array is less reactive than the nominal case infinite array. Therefore, the nominal case of an infinite array of cells with neutron absorber material is a conservative assumption.
- d. Mechanical uncertainties and biases due to mechanical tolerances during fabrication are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are:
 - Neutron absorber pocket thickness
 - Stainless steel thickness
 - Cell ID
 - Center-to-center spacing
 - Cell bowing

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

- e. 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.

3.1.2 Postulated Accidents

The criticality analysis includes postulated accidents so that the double contingency of ANSI/ANS 57.2-1983⁽¹⁾ is met and that the effective neutron multiplication factor (K_{eff}) is less than or equal to 0.95 under all conditions.

Accident conditions that will not result in an increase in K_{eff} of the rack 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 approximately thirteen 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. They include a dropped or misplaced fuel assembly outside the periphery of the racks. The rack neutron absorber loading is designed such that the nominal rack K_{eff} is approximately 0.90 since there is no soluble boron in the spent fuel pool. This will allow a 0.05 delta-K margin to the design limit for uncertainties, biases, and accident conditions. Thus, for postulated accidents, K_{eff} is less than or equal to 0.95.

The "optimum moderation" accident is not a problem in spent fuel storage racks because presence of poison plates removes the conditions necessary for "optimum moderation". The K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm³ to 0.0 gm/cm³ in poison rack designs.

The PBAPS Updated FSAR evaluates the potential for a cask drop over the spent fuel pool. The rerack program will not alter the results of that evaluation and will not increase the likelihood of a cask drop affecting storage rack criticality.

The results of earthquake loads on the deformation and position of the racks is described in Section 4.6. The racks are designed to withstand these loads and maintain a relative fuel position resulting in a K_{eff} less than or equal to 0.95.

3.1.3 Calculation Methods

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 ensures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes^(3,4) for cross-section generation and KENO IV⁽⁵⁾ for reactivity determination.

The 218 energy group cross-section library⁽³⁾ 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⁽⁴⁾ 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⁽⁴⁾ which is a one-dimensional S_N transport theory code. These multigroup cross-section sets are then used as input to KENO IV⁽⁵⁾ 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, water) that simulate LWR fuel shipping and storage conditions^(6,7) to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials⁽⁸⁾ (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method. Table 3-2 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 .

The fabrication tolerances for the racks allow for the nominal center-to-center spacing to be randomly reduced for individual cells. This change which results in an increase in K_{eff} , is included in the calculation of the worst case K_{eff} . The effect of the tolerances on material thicknesses also results in an increase in K_{eff} which will be treated conservatively as a bias.

The neutron absorbing material particle radius is used in the analysis to determine the neutron absorbing material self-shielding reactivity bias for the neutron absorbing plates. The particle size distribution is based on data supplied by the manufacturer.

The final result of the 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.

3.1.4 Criticality Analysis Results

The spent fuel storage rack is described in Section 4.0. The minimum B^{10} loading in the poison plates is $0.021 \text{ gm } B^{10}/\text{cm}^2$.

For normal operation and using the method in the above sections, the K_{eff} for the rack is determined in the following manner.

$$K_{\text{eff}} = K_{\text{worst}} + B_{\text{method}} + B_{\text{part}} + [(k_{\text{s nominal}})^2 + (k_{\text{s mech}})^2 + (k_{\text{s method}})^2]^{1/2}$$

where:

K_{worst} = worst case KENO K_{eff} which includes asymmetric fuel assembly position, material and mechanical

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

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

$k_{\text{s nominal}}$ = 95/95 uncertainty in the nominal case KENO K_{eff} .

$k_{\text{s method}}$ = 95/95 uncertainty in the method bias.

$k_{\text{s mech}}$ = 95/95 uncertainty to account for thickness, spacing and bowing tolerances.

The final K_{eff} from this analysis for all analyzed conditions is less than 0.95, including all uncertainties at a 95/95 probability/ confidence level. Therefore, the acceptance criteria for criticality is met.

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 DECAY HEAT CALCULATIONS FOR THE SPENT FUEL POOL

The Spent Fuel Pool Cooling System design is described in the PBAPS Updated FSAR⁽⁹⁾, Section 10.5. The heat load resulting from the presence of 3819 spent fuel assemblies is within the capabilities of the existing cooling system to maintain pool bulk water temperature at or below a design basis temperature of 150° F.

The analysis of the Spent Fuel Pool Cooling System capacity is based on the first assembly in each discharge being unloading to the spent fuel pool 120 hours after reactor shutdown. Fuel assemblies in normal discharges are conservatively assumed to have an exposure of 40,000 MWD/MTU. The fuel decay energy release rates are evaluated in accordance with the NRC Branch Technical Position APCS 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling".

For analysis in the normal condition, the spent fuel racks are considered to be filled with spent fuel discharged on 18 month refueling schedules. The spent fuel pool cooling system is in operation with the design cooling water flow rate and temperature on the tube side of the SFP heat exchangers and the design flow rate of pool water to the shell side of each heat exchanger at the calculated bulk pool temperature.

For analysis in the abnormal condition, all but 764 cavities (one full core) of the spent fuel racks are filled with spent fuel discharged according to the anticipated 18 month refueling schedule. With all other cooling system conditions the same as stated above, the spent fuel pool temperatures are calculated after a full core is discharged to fill the remaining spaces.

The maximum spent fuel pool heat load occurs after the full core discharge condition described above. In this case the Residual Heat Removal System is available if necessary, to supplement the Spent Fuel Pool Cooling System to maintain temperatures below 150° F. This heat load is well within the capability of the Residual Heat Removal System.

The results of a total loss of spent fuel pool cooling are presented in Section 3.4.3 assuming the normal water level is maintained.

3.3 LOCAL THERMAL-HYDRAULIC ANALYSIS FOR THE SPENT FUEL POOL

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 Peach Bottom spent fuel pools.

3.3.1 Criteria

The rack design must allow adequate cooling by natural circulation and by flow provided by the spent fuel pool cooling system.

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

- a) The coolant must remain subcooled at all points within the pool when the cooling system is operational.
- b) When the cooling system is postulated to be inoperable, the temperature of the fuel cladding must be sufficiently low that no structural failures would occur and that no safety concerns would exist.

3.3.2 Key Assumptions

- a) The nominal spent fuel pool water level is approximately 24 feet above the top of the fuel storage racks.
- b) The maximum fuel assembly decay heat output is 11.39 BTU/sec per assembly (based on Ref. 10) following 312 hours decay time (based on averaging decay time to first and last fuel assembly discharge) after shutdown.

- c. Bulk thermal-hydraulic analysis shows that the maximum pool temperature will not exceed 150°F when the spent fuel pool cooling system is in operation. However, 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.
- d) When the pool cooling system is not in operation, the maximum temperature at the inlet to the cells is assumed to be equal to the saturation temperature of 212°F at the top of the pool.

3.3.3 Analytical Methods and Calculations

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 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 sketched in Figure 3-1 where the flow paths are indicated by arrows. Note that each storage location shown in the sketch actually corresponds to a row of storage locations that is located at the same distance from the pool walls. This is more clearly shown in a plan view, Figure 3-2.

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.

Fuel assemblies from the most recently discharged batch ("hottest" fuel assemblies) are assumed to be located in various rows during different calculations in order to ensure 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 to ensure that conservatively accurate results are 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.

This is the approach that has been used to perform the analysis for the Peach Bottom spent fuel storage racks.

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 location has one large flow opening as shown in Figure 3-3. The use of these large flow holes virtually eliminates the possibility that all flow into the inlet of a given storage location 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.

3.3.3.1 Normal Operation Results

- Basis:
- a) Cooling System Operational (i.e. temperature of water at inlet to storage racks maintained at 150°F or below)
 - b) 312 hours (average decay time for discharged batch) after shutdown-Decay Heat = 11.39 BTU/second/assembly
 - c) Uniform decay heat loading in pool - No credit for lower actual heat input
 - d) The peak rod is assumed to have 60 percent more heat output than average rod
 - e) 5 mil thick crud layer on all rods
 - f) All storage locations filled with spent fuel resulting from normal refueling

- g) Minimum heat transfer coefficient in the storage cells is based on laminar forced flow. The analysis conservatively does not take credit for laminar or turbulent free or forced turbulent convection.

Results of the analysis show that no boiling occurs at any point within the storage racks when the normal cooling system is in operation or whenever pool temperature is maintained at or below 150°F.

3.4 POTENTIAL FUEL AND RACK HANDLING ACCIDENTS

The method for moving the racks into and out of the spent fuel pool is briefly mentioned in Section 4.7. The methods utilized ensure that postulated accidents do not result in a loss of cooling to either the spent fuel pool or the reactor, or result in a k_{eff} in the spent fuel pool exceeding 0.95.

3.4.1 Rack Module Mishandling

The potential for mishandling of rack modules during the rerack operation will be precluded through the use of heavy load handling procedures, load paths, and installation procedures. At no time will the cask handling crane carry a rack module over top of stored spent fuel.

3.4.2 Flow Blockage Analysis

The effects of a postulated flow blockage accident were calculated.

Basis: -

- a) Cooling system operational (i.e. temperature of water at inlet to storage racks maintained at 150°F).

- b) 312 hours (average decay time for discharged batch) after shutdown decay heat = 11.39 BTU/SEC/ASSY.
- c) Uniform decay heat loading in pool - no credit for lower actual heat.
- d) The peak rod is assumed to have 60 percent more heat output than average rod.
- e) 5 mil thick crud layer on all rods.
- f) All storage locations filled.
- g) Minimum heat transfer coefficient in the storage cells is based on laminar forced flow. The analysis conservatively does not take credit for laminar or turbulent free or forced turbulent convection.

Results of the analysis show that should up to 90% flow blockage occur, there would still be no local or bulk boiling inside the cells. Because of the large flow openings that are used in the storage racks, it is very improbable that a complete blockage could occur.

3.4.3 Accident Conditions

Under postulated accident conditions where all spent fuel pool cooling is assumed to be inoperative the spent fuel is cooled by allowing the pool water to boil to remove the decay heat from the spent fuel pool while maintaining pool water level. Although it is highly unlikely that a complete loss of cooling capability would occur, since the Residual Heat Removal System can be connected to the spent fuel pool, the racks are analyzed for this condition.

Basis:

- a) The temperature of water at the spent fuel racks' inlet is conservatively assumed to be 212°F which corresponds to the saturation temperature at the top of the pool.
- b) A spent fuel pool water level of approximately 24 feet above the top of the racks is maintained by providing makeup water to the pool from the installed makeup system, or other means.
- c) The assemblies that are evaluated are initially put into the pool at 312 hours after shutdown.
- d) The peak rods are assumed to have 60 percent greater heat output than average rods.
- e) All storage locations are filled and all downflow occurs in the peripheral gap.

Results of this analysis show that due to the effects of natural circulation, the fuel cladding temperatures are maintained sufficiently low to preclude structural failures. The maximum calculated fuel cladding temperature is 254°F.

Boiling of the water within the storage locations does not occur since the calculated maximum water temperature of 225°F is below the saturation temperature at the top of the racks of approximately 240°F.

3.5 TECHNICAL SPECIFICATIONS

The racks have been designed to ensure subcriticality based on the existing PBAPS Technical Specifications 5.5, B and D which requires the k_{eff} of the spent fuel storage pool to be less than or equal to 0.95, and an average fuel assembly loading not to exceed 17.3 grams U-235 per axial centimeter of total active fuel height of the assembly.

The spent fuel pool thermal load analysis is based on thermal loads resulting from the reactor being subcritical for at least 120 hours prior to movement of fuel assemblies from the reactor vessel to the spent fuel pool. One hundred twenty hours is the minimum time required to prepare for fuel transfer from the reactor to the spent fuel pool. This spent fuel decay time limits the decay heat load to within the limits analyzed.

The installation and use of new spent fuel storage racks does not necessitate the revision of any existing PBAPS Technical Specifications.

3.6 REFERENCES

1. ANSI/ANS-57.2-1983, "Design Requirements for LWR Spent Fuel Storage Facilities at Nuclear Power Plants."
2. USNRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978 and modification dated January 18, 1979.
3. 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).
4. N. M. George, et al, "AMPX: A Modular Code System for Generating Coupled Multi-group Neutron-Gamma Libraries from ENDF/B," ORNL/TM-3706 (March 1976).
5. L. M. Petrie and N. F. Cross, "KENO IV--An Improved Monte Carlo Criticality Program," ORNL-4938 (November 1975).
6. S. R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 2.35 wt percent ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2438 (October 1977).

7. S. R. Bierman, et al, "Critical Separation Between Subcritical Clusters of 4.29 wt percent ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories PNL-2615 (March 1978).
8. J. T. Thomas, "Critical Three-Dimensional Arrays of U (93.2) -- Metal Cylinders," Nuclear Science and Engineering, Volume 52, pages 350-359 (1973).
9. Philadelphia Electric Company "Peach Bottom Atomic Power Station Units 2 and 3, Final Safety Analysis Report (FSAR)", 1972 as amended by UFSAR, docket 50-277 and 50-278.
10. U. S. Nuclear Regulatory Commission (USNRC) Branch Technical Position APCSB 9.2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

TABLE 3-1

PARAMETERS FOR BWR FUEL ASSEMBLIES
TO BE STORED IN THE PEACH BOTTOM RACKS

FUEL BUNDLE

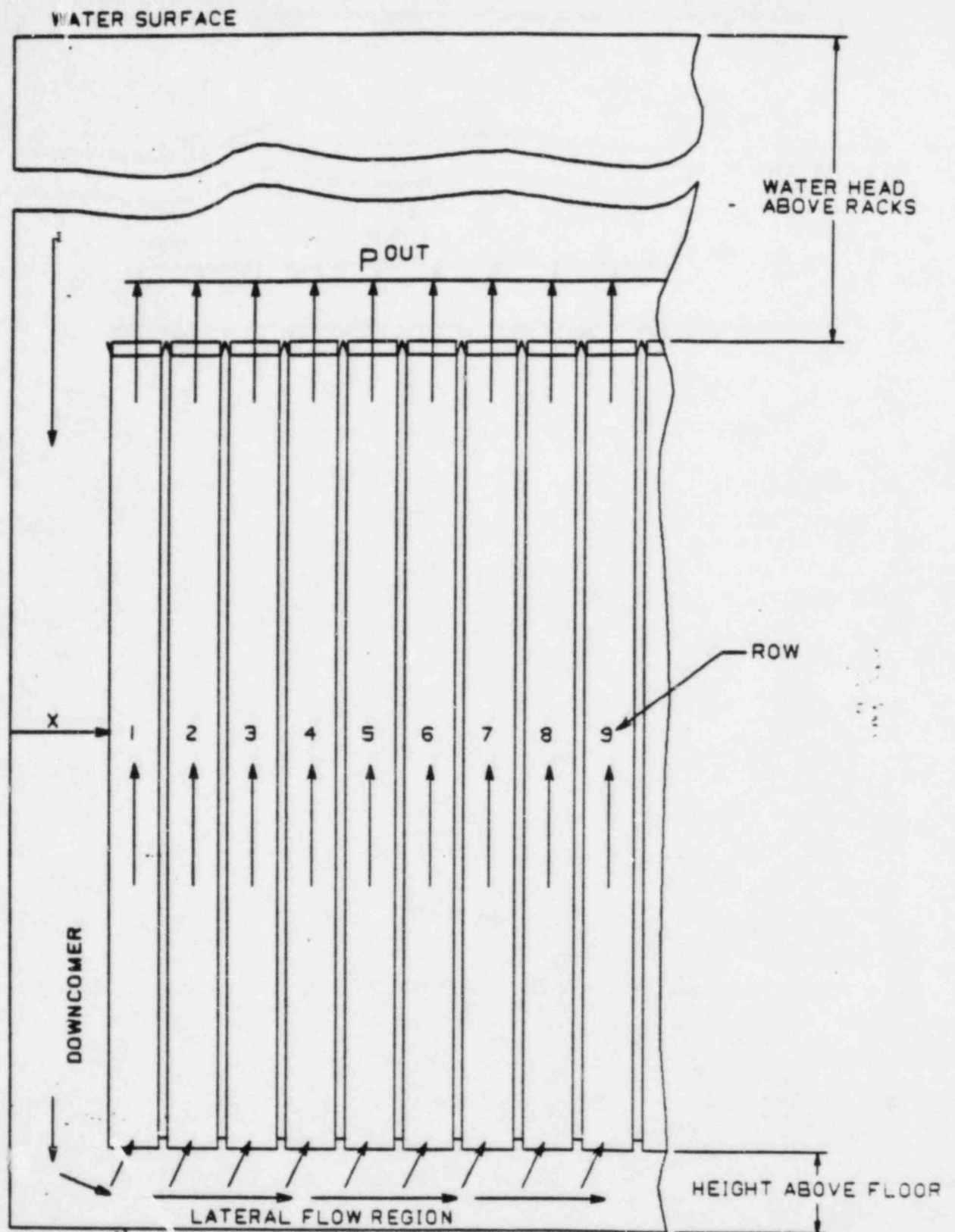
	<u>8 x 8R</u>	<u>8 x 8</u>	<u>7 x 7</u>
Enrichment in w/o U-235	3.5 w/o	3.5 w/o	3.5 w/o
Lattice Pitch	0.640"	0.640"	0.738"
Number of Fuel Rods/Assembly	62	63	49
Number of Water Rods/Assembly	2	1	0
Fuel Rod Pellet O.D.	0.410"	0.416"	0.487"
Fuel Rod Pellet Theoretical Density	95%	95%	95%
Fuel Rod Clad O.D.	0.483"	0.493"	0.563"
Fuel Rod Clad Thickness	0.032"	0.034"	0.032"
Fuel Rod Clad Material	Zr-2	Zr-2	Zr-2
Water Rod O.D.	0.591"	0.591"	---
Water Rod Tube Wall Thickness	0.030"	0.030"	---
Water Rod Material	Zr-2	Zr-2	---

FUEL CHANNEL

Material	Zr-4	Zr-4	Zr-4
Thickness	0.113"	0.113"	0.113"
Outside Square Dimension	5.84"	5.84"	5.83"

TABLE 3-2
BENCHMARK CRITICAL EXPERIMENTS^[6,7,8]

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	8.19	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	stainless steel	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	0.988 ± .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	23.48	1.005 ± .004
22. U metal cylinders	93.2	bare	plexiglas	15.74	1.010 ± .003
23. U metal cylinders	93.2	paraffin	plexiglas	24.43	1.006 ± .004
24. U metal cylinders	93.2	bare	plexiglas	21.74	0.999 ± .003
25. U metal cylinders	93.2	paraffin	plexiglas	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	plexiglas steel	16.67	1.005 ± .003



SPENT FUEL POOL NATURAL CIRCULATION MODEL
(ELEVATION VIEW)

FIGURE 3-1

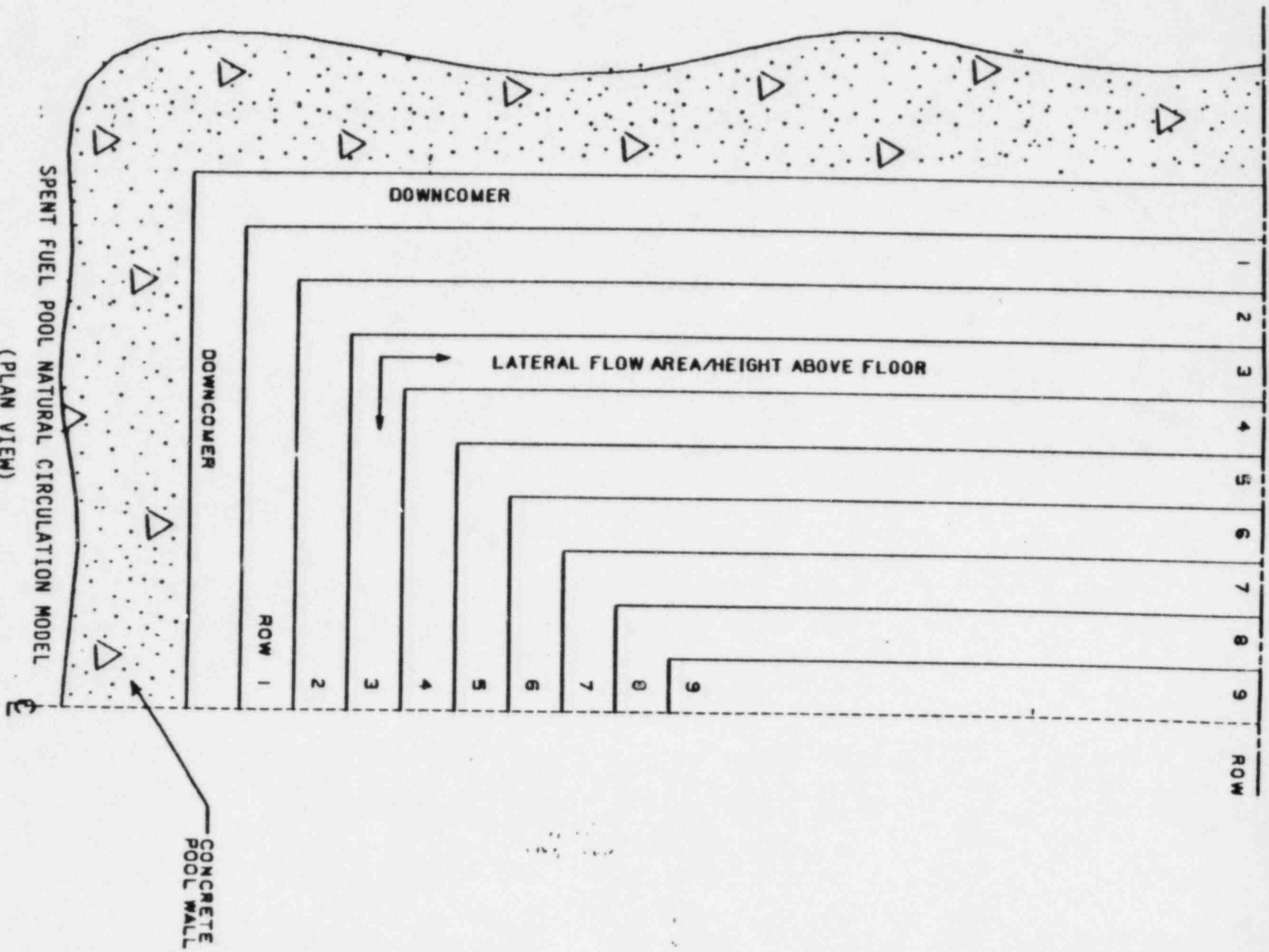
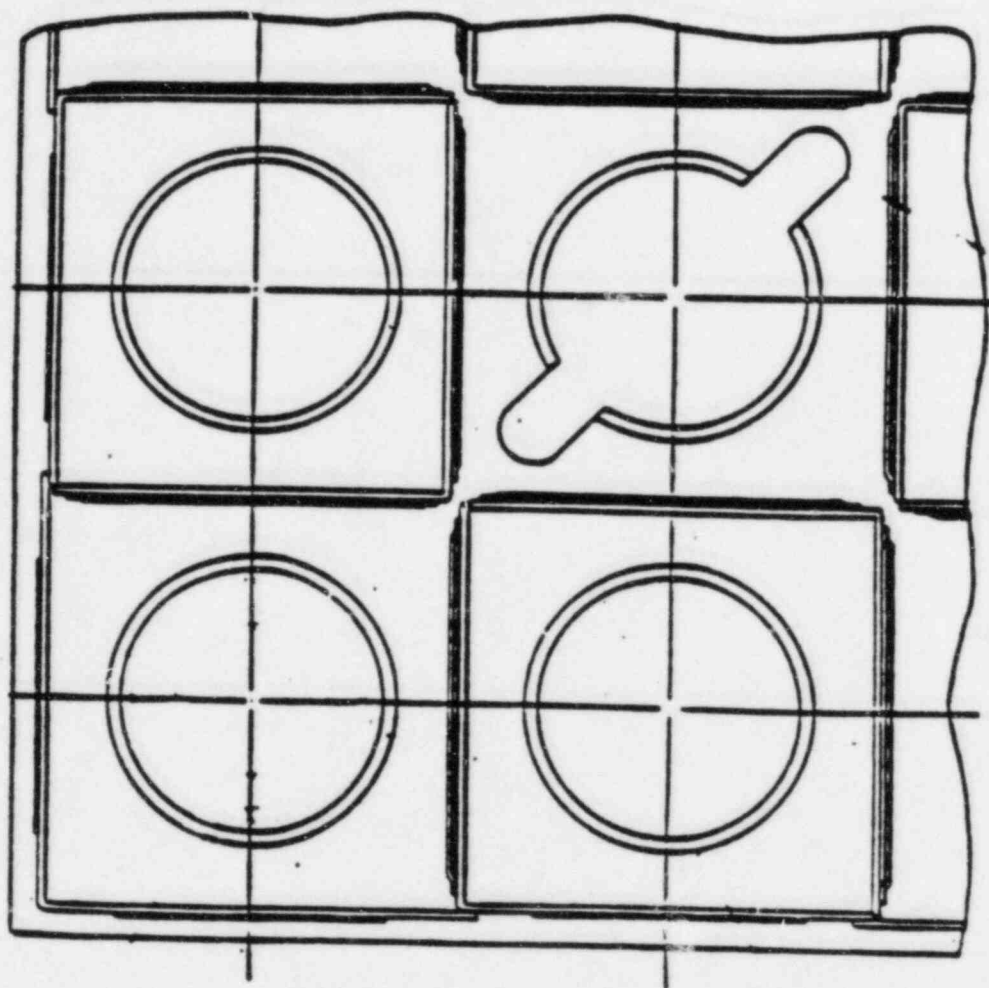


FIGURE 3-2



SPENT FUEL RACK INLET FLOW AREA
(PLAN VIEW)

FIGURE 3-3

4.0 MECHANICAL, MATERIAL AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 Description of Spent Fuel Pool Structure

A description of the spent fuel storage pool is provided in Section 10.3.4.2 of the PBAPS Updated FSAR.⁽¹⁾ The spent fuel pool has been designed to meet Seismic Category I requirements.

The walls and floor of the spent fuel pool are lined with an eight gage thick stainless steel liner. This liner serves only as a water tight boundary, it is not a structural member. Steel embedments are provided in the pool walls and slab to attach the liner to the pool structure. Monitoring trenches are provided behind the liner for detecting and collecting any leakage. Any leakage is directed to the liquid radwaste system. Individual liner plates are connected with full penetration welds.

The spent fuel pools are located inside the Reactor Buildings in an elevated position adjacent to the North (Unit 2) and South (Unit 3) sides of the drywell shield walls, shown on Figures 4-1 through 4-3.

The spent fuel pool has been evaluated structurally for the additional loading due to the increased number of fuel elements and modified rack design in accordance with Standard Review Plan (SRP) Section 3.8.4⁽²⁾ and the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as amended by the NRC letter dated January 18, 1979⁽³⁾, and other applicable codes and standards identified in Section 4.2.

The new spent fuel storage racks are designed so that the floor loading from racks filled with spent fuel assemblies will not exceed the structural capacity of the spent fuel pool structure. The analysis of the spent fuel pool slab for the effects of a cask drop as described in the PBAPS Updated FSAR is unchanged by the storage of additional spent fuel assemblies. Critical areas of the structure, such as the junction between the wall and the floor, have been checked for the additional loading.

4.1.2 Description of Spent Fuel Racks

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

A list of design criteria is given below:

1. The racks are designed in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 and revised January 18, 1979.
2. The racks are designed to meet the nuclear requirements of ANSI/ANS-57.2-1983, Sections 6.4.2.1 and 6.4.2.2. The effective multiplication factor, K_{eff} , in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions as described in Section 3.0.
3. The racks are designed to allow coolant flow such that boiling does not occur in the water channels between the fuel assemblies in the rack.

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 Table 4.2.
5. The racks are designed to withstand loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane without violating the criticality acceptance criterion.
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 storage locations.
8. The materials used in construction of the racks are compatible with the storage pool environment and do not contaminate the fuel assemblies.

The spent fuel pool storage rack arrangements for Peach Bottom Units 2 and 3 are shown in Figures 4-4 and 4-5. Each storage location is capable of storing 7 x 7, 8 x 8, and 8 x 8 (R) BWR fuel assemblies (with or without their fuel channels) at a design enrichment equal to or less than 3.50 maximum weight percent U_{235} .

4.1.2.1 Design of New Spent Fuel Racks

The high-density rack details are shown in Figure 4-6. The rack modules are free-standing and self supporting. The modules are neither anchored

to the floor nor braced to the pool walls. The storage cells within a rack module are assembled in a checkerboard pattern and welded to a common base plate. The vertical corners of adjacent cells are welded together to form an integral structure (Figures 4-6 and 4-7). Each rack module is provided with remotely adjustable leveling pads which are located at the center of the four corner cells and at interior module locations. These pads are used to level the racks during installation and distribute loads to the pool floor.

The rack module consists of two major sections; the base support assembly and the cell assembly. Figure 4-6 illustrates these sections. The racks are constructed from type 304LN stainless steel except for the leveling screws which are type 17-4 PH stainless steel.

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

The major components of the cell assembly are the fuel assembly cell, the Boraflex (neutron absorbing) material, and the wrapper. The wrapper is attached to the outside of the cell by intermediate spot welding along the entire length of the wrapper. The wrapper, which holds the Boraflex material in place, provides for venting of the Boraflex to the pool environment. Depending on the cell location in the rack module, and the associated criticality requirements, some cells have a Boraflex wrapper on all four sides, some on three sides, and some on two sides. Cells with four wrappers are located in the interior of the rack, cells with three wrappers are located on the periphery of the rack, and cells with two (adjacent) wrappers are located at the corners of the rack.

Rack module data for both units are described in Table 4.1.

4.1.2.2 Fuel Handling

The storage of additional spent fuel assemblies in the spent fuel pool will not affect the analysis and consequences of the design basis fuel handling accidents as presented in the PBAPS Updated FSAR. The spent fuel storage racks are 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 and 5.3, respectively, of this safety analysis report. The parameters of the postulated fuel assembly drop accident are contained in Section 4.6.

4.2 APPLICABLE CODES, STANDARDS, AND SPECIFICATIONS

The spent fuel storage racks are designed and fabricated to applicable provisions of the following codes, standards, and NRC Regulatory Guides:

- a. NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979.
- b. Code of Federal Regulations 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criteria 61 and 62. Code of Federal Regulations 10CFR21, "Reporting of Defects and Noncompliance."
- c. NRC Regulatory Guides
 - R.G. 1.13 Spent Fuel Storage Facility Design Basis, Rev. 2, Dec. 1981 (Draft)
 - R.G. 1.25 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, March 1972
 - R.G. 1.29 Seismic Design Classifications, Rev. 3, Sept. 1978.
 - R.G. 1.31 Control of Ferrite Content in Stainless Steel Welding Metal, Rev. 3, April 1978
 - R.G. 1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, Rev. 2, April 1973
 - R.G. 1.70 Standard Format and Content of Safety Analysis Report for Nuclear Power Plants, Rev. 3, Nov. 1978

- R.G. 1.92 Combining Modal Responses and Spatial Components
in Seismic Response Analysis, Rev. 1, Feb. 1976
- R.G. 1.142 Service Limits and Loading Combinations for Class
I Linear-Type Component Supports, Rev. 1, Jan.
1978

d. NRC Standard Review Plans - NUREG-0800

- SRP 3.7 Seismic Design, Rev. 1, July 1981
- SRP 3.8.4 Other Category I Structures, Rev. 1, July 1981
- SRP 3.8.5 Foundations, Rev. 1, July 1981
- SRP 9.1.2 Spent Fuel Storage, Rev. 3, July 1981
- SRP 9.1.3 Spent Fuel Pool Cooling and Cleanup System,
Rev. 1, July 1981
- SRP 9.2.5 Ultimate Heat Sink, Rev. 2, July 1981

e. Industry Codes and Standards

American Society of Mechanical Engineers, Boiler and Pressure
Vessel Code:

- a. Section II, "Material Specification."
- b. Section III, Division 1, "Nuclear Power Plant
Components," Subsection NF, "Component Supports," 1980
Edition through Summer 1982 Addendum
- c. Section V, "Nondestructive Examination."
- d. Section IX, "Welding Requirements."

American National Standards Institute, ANSI-57.2-1983,
"Design Objectives for Light Water Reactor Spent Fuel Storage
Facilities at Nuclear Power Stations."

American National Standards Institute, ANSI-52.1-1983,
"Nuclear Safety Criteria for the Design of Stationary Boiling
Water Reactor Plants."

American National Standards Institute, N45.2-1971, "Quality
Assurance Program Requirements for Nuclear Facilities."

American National Standards Institute, N45.2.1-1973,
"Cleaning of Fluid Systems and Associated Components for
Nuclear Power Plants."

American National Standards Institute, N45.2.2-1972,
"Packaging, Shipping, Receiving, Storage, and Handling of
Items for Nuclear Power Plants."

American National Standards Institute, ANSI-8.1-1983,
"Nuclear Criticality Safety in Operations with Fissionable
Materials Outside Reactors."

ACI Committee 318, "Building Code Requirements for Reinforced
Concrete (ACI 318-83)," American Concrete Institute, Detroit,
Mich., 1983.

Also, ACI Committee 318, "Commentary on Building Code
Requirements for Reinforced Concrete (ACI 318-83)," American
Concrete Institute, Detroit, Mich., 1983.

ACI Committee 349, "Code Requirements for Nuclear Safety
Related Concrete Structures (ACI 349-80)," American Concrete
Institute, Detroit, Mich., 1980.

AISC, "Specification for the Design, Fabrication, and
Erection of Structural Steel for Buildings," American
Institute of Steel Construction, New York, N.Y., 1978.

- AISC, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," American Institute of Steel Construction, New York, N.Y., Final Draft of initial issue, 9/14/83.

f. U.S. Nuclear Regulatory Commission (USNRC) Branch Technical Position:

- a. CPB 9.1-1, "Criticality in Fuel Storage Facilities."
- b. APCS89.2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

4.3 SEISMIC AND IMPACT LOADS

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

Earthquake loading is predicated upon an operating basis earthquake (OBE) at the site having a horizontal ground acceleration of 0.05 g. In addition, a safe shutdown earthquake (SSE), having a horizontal ground acceleration of 0.12 g, is used to check the design to ensure no loss of function. The site ground accelerations are provided in PBAPS Updated FSAR Appendix C.3. The OBE and SSE designations used herein correspond to FSAR designations of design earthquake (DE) and maximum credible earthquake (MCE) respectively.

Horizontal response spectra applicable for the spent fuel pool structure are applicable for both orthogonal horizontal directions. The vertical component of acceleration is taken as two-thirds of the horizontal ground acceleration. The seismic loads for the spent fuel pool structure are generated using the equivalent static load method.

Seismic analysis of the fuel storage racks is being performed by the time history method. Where time histories are used, the three orthogonal time histories are statistically independent. The time histories and response spectra 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 and 5 percent OBE and SSE, respectively. Increased damping due to submergence in the water has not been considered in this analysis.

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.

Deflection or movements of racks under earthquake loading is limited by design such that the racks do not touch each other or the spent fuel pool walls, and the fuel assemblies are not damaged.

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

4.4 LOADS AND LOAD COMBINATIONS

4.4.1 Spent Fuel Racks

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

The major structural loads are produced by the operational basis earthquake (OBE) and safe shutdown earthquake (SSE) events. The loads or stresses from the nonlinear seismic analysis 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. For each component, the maximum seismic stress is combined with the dead weight stress to produce the total stress.

4.4.2 Spent Fuel Pool Structure

All loading combinations required by USNRC Regulatory Guide 1.142 (4), USNRC Standard Review Plan 3.8.4 (2), ACI (5,6), and AISC (7,8) were evaluated. The number of combinations to be analyzed were reduced by eliminating combinations governed by others. Final governing equations for the SFP structure are shown in Table 4-3 for concrete structures using strength design methods and for structural steel using plastic design methods.

The dead load includes the weight of the spent fuel racks, stored fuel, spent fuel pool (SFP), and contributing weight of the adjacent floor slabs, roof, and walls.

The live load includes the roof snow load, distributed live loads on the adjacent floor slabs, crane loads, and a buoyant weight of a loaded spent fuel storage cask.

Hydrostatic loads consist of the lateral water pressure exerted on the SFP walls and slab.

Thermal loads are based on the pool water temperatures of 150 degrees F resulting from a full core discharge under normal operating conditions, and saturation temperatures for accident conditions varying from 252 degrees F at the bottom of the pool to 212 degrees F at the free water surface. In all cases, a conservative Reactor Building indoor ambient air temperature of 68 degrees F is used. A stress free temperature of 70 degrees F is used.

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 Design and Analysis Procedures for Spent Fuel Storage Racks

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

The rack 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 non-linear model using the dynamic capabilities of the Westinghouse Electric Computer Analysis (WECAN) Code ^(9,10). This is a general purpose finite element code with a great variety of elements which have static and dynamic capabilities.

The time history analysis is performed on a 3-D single cell non-linear model with the effective properties of an average cell within the rack module. The effective single cell structural properties are obtained from a 3-D structural model of the rack module, as shown in Figure 4-8. The details of the structural model and the seismic model are discussed in the following paragraphs.

The structural model, shown in Figure 4-8, is a quarter section 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 models, shown in Figures 4-9 and 4-10, are 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 which represent the structural and dynamic properties of the fuel rod bundle and grid support assemblies.

The cell assembly is represented by three-dimensional beam elements 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 a general mass matrix element connected between the fuel and cell.

The gaps between the fuel and cell are modeled by three-dimensional dynamic 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 sloshing movement of the pool water occurs above the top of the racks. Therefore, no sloshing loads are imposed on the rack structure.

The support pads are modeled by a combination of three-dimensional dynamic friction elements connected by a "rigid" base beam arrangement which produces the spacing of 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 in both directions on a plane. Vertical grounded springs at the support pad locations are used to model and account for the interaction between the racks and the spent fuel pool structure. The friction elements are capable of reversing the direction of the restraining force when sliding changes direction.

This 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 the 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. In addition the results are used to determine the rack response for full, partially filled, and empty rack module loading conditions.

4.5.2 Design and Analysis Procedures for Spent Fuel Pool Structure

The spent fuel pool (SFP) structure was analyzed to determine the maximum allowable fuel rack loads which could be imposed on the pool slab. A comprehensive structural analysis was performed and results were evaluated in accordance with current codes, standards and regulatory requirements.

4.5.2.1 Computer Program and Finite Element Model

The MSC/NASTRAN Version 62A (11) general purpose finite element program is used for this work. The three dimensional finite element model (FEM) includes the entire SFP structure plus adjacent key structural members to the extent where suitable boundary conditions can be assumed. See Figures 4-11 and 4-12 for the structural dimensions and the finite element model.

Figure 4-12 also shows the superelement configuration. Superelement 1 comprises the pool slab and the lower portion of the pool walls. Superelement 2 comprises the drywell shield structure and the balance of the model is in Superelement 3. The residual structure, or Superelement 0, consists of only the grid points on boundaries between the other superelements. A computer plot of the FEM is shown in Figure 4-13.

4.5.2.2 Geometry, Boundary Conditions, and Element Types

The modeled structure includes the walls and slab of the SFP, the North and West exterior walls of the Reactor Building between El. 165' and 234', drywell shield wall between El. 165' and 234', portions of the floor slabs at El. 195'-3" and 234'-0" and interior wall between El. 195'-3" and 234'-0". The steel superstructure of the Reactor Building and concrete floors and walls South of the drywell shield wall are not included in the FEM.

Floor slabs and walls immediately adjacent to the SFP are modeled to simulate the proper lateral restraint on the pool structure. Complete fixity against translation and rotation is assumed at the base of the drywell shield wall. Cut-off boundaries of adjoining walls and slabs were restrained with translational springs. These springs permit the model to

simulate the cantilever mode deflected shape of the Reactor Building under horizontal seismic loading. Translational springs simulate lateral stiffness of the remainder of the Reactor Building walls which were not included in the model. In-plane rotations of all interior grid points on slabs and walls are restrained.

A summary breakdown of the model connectivity is given below. The overall model contains an estimated 11,000 independent degrees of freedom.

MSC/NASTRAN FINITE ELEMENT MODEL DATA

<u>Element Type</u>	<u>Number</u>
ELAS2	134
QUAD4	1,836
BAR	192
TR1A3	479
RBAR, RTRPLT, RBE1, RBE2	<u>234</u>
Total =	2,875

<u>Superelement</u>	<u>No. of Grids</u>	<u>No. of Elements</u>
0	163	11
1	596	823
2	374	579
3	<u>1,133</u>	<u>1,462</u>
Total =	2,266	Total = 2,875

4.5.2.3 Analysis Method

External loads are applied incrementally to the FEM in order to establish reasonable internal force distribution patterns and cracking patterns. Concrete cracks under low tensile stresses at approximately 10 percent of its ultimate compressive strength. In reinforced concrete structures, localized concrete cracking reduces the overall structural stiffness. In structures such as this SFP, concrete stresses are usually tension with low

compression. Under these conditions concrete cracking is the major source of the nonlinearity as compared with nonlinear stress-strain relations and formation of plastic hinges. For this reason it is important to develop a logical procedure for establishing cracking patterns.

It is also important to note that as long as equilibrium conditions are satisfied and the yield criterion is not exceeded anywhere, a safe lower bound solution will result regardless of the loading history used in the analysis. Therefore, the sequence of loading and the resulting crack pattern is not critical for strength evaluations.

An iterative incremental approach is used to arrive at the final solutions. Checking stresses against the cracking criterion and adjustment of element material properties are done manually at the end of each iteration. Each iteration is a linear elastic analysis, however, the cumulative effect is a nonlinear analysis that has taken into account load redistribution after concrete cracking due to mechanical loads and thermal loads. At the end of each iteration, the cumulative load which includes the latest load increment, is applied to the structure.

4.5.2.4 Section Properties After Cracking

In order to arrive at a reasonable final condition with regard to cracking patterns under mechanical loads, cracking criteria must be implemented. Initially, i.e., prior to any load application all elements are considered to be uncracked with linear elastic properties. After load increments are applied, the cracking criteria are checked manually and local element elastic and shear moduli are modified accordingly to represent the cracked condition.

The cracking criteria are applied primarily to only those elements comprising the pool slab and the lower portions of the pool walls up to approximately 12 feet above the top of the pool slab. Cracking due to mechanical loads in the balance of the structure does not have significant effects on the results for the pool slab.

Extreme fiber tensile stresses in the element orthogonal local 1 and 2 directions are compared against the concrete modulus of rupture. Properties of elements having extreme fiber tensile stresses exceeding this value are modified. This is done by inputting the effective elastic modulus E_1 or E_2 based on the cracked section moment of inertia.

The inplane shear modulus G_{12} is calculated using the new E_1 and E_2 values from the principles of orthotropic plate behavior. In addition, the reduced transverse shear moduli G_{1z} and G_{2z} are used in the cracked directions.

4.5.2.5 Slab Shear Transfer to Pool Walls

A typical section through the intersection of the pool slab and wall is shown in Figure 4-14.

Transverse shear forces in the suspended slab are eventually developed by vertical axial tension forces in the pool walls. The critical section for slab shear and bending is taken at the face of the walls in accordance with ACI Code provisions. Similarly, the critical section in the wall is taken on the horizontal plane at the top of slab elevation.

Two additional critical sections at the wall/slab joint were evaluated in addition to the ones required by the ACI Code. These two sections are shown in Figure 4-15. Postulated concrete cracks along Critical Sections 1 and 2 shown in this figure represent the most severe conditions for load transfer through the wall/slab joint.

Calculations of concrete shear strength, V_c , and stirrup shear strength, V_s , are in compliance with Chapter 11 of ACI 349 and ACI 318. Concrete shear strengths are reduced to account for effects of axial tension forces. Also, increased concrete shear strength is gained from the presence of axial compressive forces using EQ. (11-4) of ACI 318 and ACI 349. Shear capacities of the steel beams and connections are determined in accordance with Part 2 of the AISC specifications (7) for plastic design.

4.5.2.6 Thermal Moment Relaxation

Thermal stresses are introduced by internal and external restraints. In reinforced concrete structures, cracking at a section reduces its rigidity in addition to the total stiffness of the member which relieves thermal stresses. Ideally an analysis would consider effects of concrete cracking on internal force distributions throughout the loading history. One alternate approach however, in compliance with ACI 349 Appendix A (6), is to assume the structure is uncracked for mechanical loads and cracked only for thermal loads. This second approach was used in this safety investigation for local areas of the SFP structure away from the pool slab. The method used to establish cracking patterns in lower areas of the SFP structure and the pool slab is described in Subsection 4.5.2.3.

4.5.2.7 Alternate Supporting Calculations

This section provides an estimate of the pool slab's ultimate load carrying capacity. The predicted slab strength is used to determine an overall margin of safety under certain loading conditions. In addition, a nonlinear finite element analysis of a simplified pool slab structure is made.

Design loads used in these alternate calculations are listed in Table 4-4. These loads are equal to the loads used for the MSC/NASTRAN finite element analysis and they are used to estimate the maximum allowable loading on the SFP.

The total capacity of the slab system to support vertical loads is controlled by shear which is comprised of the concrete shear capacity, stirrup capacity, and shear strength of the six W36 x 230 beams directly beneath the pool slab. Since thermal loads do not contribute to the gross shear forces around the perimeter of the slab, there are only two controlling loading combinations to consider for checking total shear capacity, as summarized below.

SUMMARY OF TOTAL LOADS VERSUS SLAB SHEAR CAPACITY

No.	Load Combination	Calculated Load (kip)	Maximum Allowable $\phi V_n + V_b$ (kip)
3	$U = 1.4D + 1.4F + 1.7L + 1.9E$	17,211	21,436
7	$Y = 1.7 (D + F + L + E)$	19,324	23,508

where, ϕ = ACI strength reduction factor
 V_n = Shear strength of concrete plus stirrups
 V_b = Shear capacity of six W36 X 230 beams

Simplified finite element models with nonlinear solution techniques can be very effective in taking advantage of the inherent strengths of structures such as the SFP. Also, it is possible to account for internal arching action if elements are modeled through the thickness of the slab. It had not been possible to account for the beneficial effects of arching action in hand calculations or in the MSC/NASTRAN finite element analysis. The purpose of this nonlinear finite element analysis is to predict the collapse load as accurately as possible.

The pool slab was modeled using the ADINA (12) finite element program for the purpose of demonstrating an overall load carrying capacity in excess of code requirements.

The slab is idealized as a system of twistless beam strips spanning across the pool in each direction at the slab centerlines. Each concrete strip is modeled using plane stress elements arranged in several layers through the thickness of the slab. Plane stress elements were chosen because of the nonlinear concrete material model available for this element. The material model describes the nonlinear stress-strain relations, tensile failure, compression crushing and post failure behavior of concrete. Reinforcement was modeled using nonlinear truss elements. The W36 X 230 beam is idealized using plane stress elements for the web and nonlinear truss elements for the flanges.

Boundary conditions are assumed fixed at the supports and a uniformly distributed load is applied incrementally to the top of the slab. No reinforcement yielding and very little concrete cracking were predicted at the allowable design load.

The analysis was halted when the applied load approached three times the factored design load. Final results indicated some concrete cracking at supports and at midspan and top bar yielding at the supports would occur, but collapse was not yet imminent. The ADINA model is used here as an alternate analysis to lend support and to increase the level of confidence in results of the more comprehensive analyses using the MSC/NASTRAN finite element model.

4.6 STRUCTURAL ACCEPTANCE CRITERIA

4.6.1 Structural Acceptance Criteria for Spent Fuel Storage Racks

The fuel racks are 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."⁽³⁾

The major normal and upset condition loads are produced by the operating basis earthquakes (OBE). The thermal stresses due to rack relative expansion are 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," (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 are within the acceptance limits identified in the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," (with clarifications as noted in Table 4.2).

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

4.6.2 Fuel Handling Crane Uplift Analysis

The objective of this analysis is to ensure that the rack can withstand the maximum uplift load of 4,000 pounds and a horizontal force of 1100 pounds of the fuel handling crane without violating the criticality acceptance criterion. The maximum uplift load is approximately two times the capacity of the fuel handling crane. In this analysis the loads are

assumed to be applied to a fuel cell. Resulting stresses are within acceptable stress limits, and there is no change in rack geometry of a magnitude which causes the criticality acceptance criterion to be violated.

4.6.3 Fuel Assembly Drop Accident Analysis

The objectives of this analysis are to ensure that, in the unlikely event of dropping a fuel assembly on a storage rack, accidental deformation of the rack will not cause the criticality acceptance criterion to be violated, and the spent fuel pool liner will not be perforated.

Three accident conditions are postulated. The first accident condition assumes that the weight of a fuel assembly and handling tool impacts the top end fitting of a stored fuel assembly or the top of a storage cell from a conservative drop height of 2 feet in a straight attitude. The second accident condition is similar to the first except the impacting mass is at an inclined attitude. The impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cells and the rack base plate assembly. Under these faulted conditions the criticality acceptance criterion is not violated and the pool liner is not perforated.

The third accident condition assumes that the dropped assembly falls straight through any empty cell and impacts the rack base plate from a conservative drop height of 2 feet above the top of the rack. The results of this analysis show that the impact energy is absorbed by the fuel assembly and the rack base plate. The spent fuel pool liner is not perforated. Criticality calculations show that $K_{eff} < 0.95$ and the criticality acceptance criterion is not violated.

In each of these accident conditions, the criticality acceptance criterion is not violated and the spent fuel pool liner is not perforated.

4.6.4 Fuel Rack Sliding and Overturning Analysis

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

The nonlinear model described in paragraph 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 low coefficient of friction ($\mu = 0.2$) produces maximum rack sliding while a high value ($\mu = 0.8$) produces maximum rack horizontal overturning force.

The fuel rack nonlinear time history analysis shows that the fuel rack slides a minimal distance. This distance combined with structural and thermal displacements is less than the rack-to-rack, rack-to-floor obstruction, or rack-to-wall clearances; thus, impact between adjacent rack modules, rack module and floor obstructions, and rack module and pool wall is prevented. Also, the factor of safety against overturning is well above the value permitted by Section 3.8.5.II.5 of the Standard Review Plan.

4.6.5 Fuel Bundle/Module Impact Evaluation

The nonlinear seismic model includes the gap between the fuel assembly and storage cell, and provides the fuel to cell impact load and the associated fuel storage rack response during the seismic event. The fuel assembly and fuel storage cell impact forces obtained from the nonlinear

analysis were used to evaluate the effects on the fuel rack structure and fuel assembly structure. These loads are within the allowable limits of the fuel rack module materials and fuel assembly materials. Therefore, there is no damage to the fuel assembly or fuel rack module due to impact loads.

4.6.6 Structural Acceptance Criteria for Spent Fuel Pool Structure

The available section strengths for reinforced concrete elements are calculated by the strength design method in accordance with ACI 318 and ACI 349 (5,6). Axial force/moment and axial force/shear interaction diagrams are generated for the entire SFP structure. These interaction diagrams were then used to manually check each critical section. The axial force/shear interaction diagram for the SFP floor includes the transverse shear strength of the steel beams. The available section strengths for structural steel members for axial loads plus bending are determined by plastic design methods in accordance with AISC (7).

The section strengths required to carry the increased loading are based on results from the MSC/NASTRAN finite element analyses. Required strengths in terms of shear forces and bending moments are determined for each element in the SFP structure and for each of the governing load combinations.

4.6.6.1 Pool Slab and Walls

The SFP load carrying capacity is limited by transverse shear capacity of the pool slab for Load Combination Nos. 3 and 7. Large reserve flexural capacities exist in the pool walls and slab for all load combinations. A reduced transverse shear capacity was used in the pool slab to reflect the small amount of membrane tension generated by the lateral fluid pressures on the pool walls. This shear capacity was compared against peak transverse shear forces from the MSC/NASTRAN finite element analysis results and is adequate.

The load transfer capacity of the wall/slab joints on the East and West sides of the pool were evaluated and found to be adequate.

4.6.6.2 Structural Steel Members

Simple beam moments due to factored slab dead load are manually superimposed on the FEM results. The resulting total bending moments were used to evaluate the beams for combined axial load and bending and all beams were found to be adequate.

4.6.6.3 Adjacent Reinforced Concrete Structural Members

Adjacent structural members evaluated in this safety investigation include a portion of the Reactor Building exterior wall and a section of the drywell shield wall. As discussed above, reracking will result in a relatively small increase in design loads over the currently licensed conditions. This small increase in the SFP design loads results in an even smaller percentage increase of design loads for adjacent structures as the loads are distributed throughout the Reactor Building. The changes in stress conditions of adjacent structures due to increased spent fuel storage capacity have been calculated to be minimal.

The shear stresses for the original design of the drywell shield wall are contained in FSAR Table C.4.5 and are summarized in Table 4-5. Original design concrete shear stresses at EL. 180'-0" are 89 percent of the allowable stress for dead load plus operating basis earthquake (OBE) and 69 percent of the allowable stress for dead load plus thermal load plus safe shutdown earthquake (SSE). All other reported stresses are less than 50 percent of the corresponding allowable stresses. Additional shear stresses due to increased spent fuel storage capacity are calculated to be 0.0020 kip/in² and 0.0032 kip/in² at EL. 180'-0" for OBE and SSE respectively. These shear stress increments are based on the MSC/NASTRAN finite element

analysis results. These increments represent increases in total shear stresses from 89 percent to 92 percent of the allowable for OBE and from 69 percent to 70 percent for SSE. The resulting total concrete shear stresses are less than the allowable shear stresses.

Local areas of the North exterior wall of the Reactor Building were also evaluated due to the increased loads. The areas checked are the support points of the East and West walls of SFP. These areas are adequate for combined axial load and bending. Shear forces are also less than the shear capacity.

4.6.7 Spent Fuel Pool Serviceability Checks

4.6.7.1 Deflections

Slab centerline deflections resulting from the MSC/NASTRAN analysis are used as the basis for checking code requirements on allowable deflections. Table 9.5(a) of ACI 349 indicates deflections that could be expected in Nuclear Safety Related concrete structures but which are not necessarily maximum allowable deflections. Maximum allowable deflections are normally specified in the plant design criteria. However, since specific deflection criteria are not available for this SFP structure, the ACI criteria are used as the basis for determining maximum allowable deflections. The ACI criteria apply to unfactored loads.

Effects of concrete creep on deflections were considered and the calculated long term deflections are less than the allowable values.

4.6.7.2 Concrete Crack Widths

Crack widths were calculated at the midspan and at the support edges of the SFP floor for unfactored loading conditions. An allowable crack width of 0.016 inch for operating conditions is recommended by ACI Committee 224 (13). Calculated crack widths are less than the allowable value.

4.6.8 Pool Slab/Fuel Rack Interface Loads

The actual total pool floor loads from the racks are calculated from the results of the rack nonlinear seismic model. The distribution of the loads on the individual support pads is calculated from the overall load from the seismic model times the peaking factor for the individual pad as determined from the structural model. The actual fuel rack/pool floor interface loads are within the maximum allowable limits specified by the spent fuel pool structural analysis.

Maximum allowable loads for the SFP floor are given in Table 4-6. Allowable local bearing pressures are calculated based on concrete code allowables defined in Section B.3.1 of Appendix B - "Alternate Design Method" of ACI 318. A one-third increase in allowable bearing stress is permitted for seismic and accident loading conditions.

4.6.9 Evaluation of Spent Fuel Pool Slab for Fuel Handling Accident

Impact loads resulting from a dropped fuel assembly do not control the design of the pool slab. Also, concrete scabbing off the bottom of the slab will not occur and concrete penetration is limited to the topping concrete above the structural slab.

The maximum leakage rate due to a postulated perforation of the liner is one-fifth of the available makeup water capacity. There is no danger of draining the spent fuel pool or uncovering the spent fuel. Analyses have shown that should the liner be perforated by a dropped consolidated fuel canister, the make-up water rate available is five times the leakage flow rate.

4.6.10 Spent Fuel Pool Liner Plate Evaluation

Typical wall and floor liner plates and anchorage details were evaluated. Effects of increased loading on the pool slab due to reracking were included in the liner evaluation in addition to normal and accident thermal effects.

The liner function is to retain the SFP fluid. The liner is not required to provide support for any member or loading, and therefore is not a structural

element. Acceptability criteria for the liner is the prevention of tearing or leakage. Consequently, the liner was evaluated for liner strain only.

Adequate margins of safety exist on the liner plates and anchorages to ensure the functionability of the liner system.

4.7 MATERIALS, QUALITY CONTROL, AND SPECIAL CONSTRUCTION TECHNIQUES

4.7.1 Construction Materials

Construction materials conform to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. All the materials used in the rack construction are compatible with the storage pool environment and do not contaminate the fuel assemblies or the pool water. The racks are constructed from type 304LN stainless steel except the leveling screws which are type 17-4 PH stainless steel. Plates placed on the spent fuel pool floor to avoid rack support pad interferences are Type 304 stainless steel.

4.7.2 Neutron Absorbing Material

The neutron material, Boraflex, used in the Peach Bottom spent fuel rack construction is manufactured by Brand Industrial Services, Inc., and fabricated to safety related nuclear criteria of 10CFR50, Appendix B. Boraflex is a silicone based polymer containing fine particles of boron carbide in a homogeneous, stable matrix. Boraflex contains a minimum B^{10} areal density of 0.021 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.⁽¹⁴⁾ 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.⁽¹⁵⁾

Long term borated water soak tests at high temperatures were also conducted.⁽¹⁶⁾ It was shown that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or

softening. 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 CO₂.

The testing performed to date verifies that Boraflex maintains a long-term material stability and mechanical integrity, and can be safely utilized as a poison material for neutron absorption in spent fuel storage racks.

4.7.3 Quality Assurance Program

The PBAPS spent fuel racks are "Q-Listed" items. PECO's Quality Assurance Program is applicable to the specification, procurement, design, fabrication, inspection, handling, and installation of the new storage racks.

The rack design, analysis, material procurement, and fabrication were performed under the Westinghouse Nuclear Components Division's Quality Assurance Program. This program conforms to the requirements of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section III & VIII (Subsection NCA-4000), 10CFR50, Appendix B, ANSI N45.2, MIL-I-45208, MIL-Q-9858A, and RDT F2-2.

The Westinghouse Nuclear Components Division holds the following ASME Boiler & Pressure Vessel Code certificates with the scope authorizations as indicated below:

Certificates

Scope Authorizations

N

Class 1, 2, 3 vessels, pumps, piping systems; Class 2, 3 storage tanks; and Class CS core support structures

NPT

Class 1, 2, 3 components parts and appurtenances, tubular parts welded with filler metal; and Class CS core support parts.

U

Pressure vessels

The preparation of the specification for rack design, and the analysis of the spent fuel pool structure was performed under the Gilbert/Commonwealth, Inc. (G/C) Nuclear Quality Assurance Program. This program, implemented through the G/C "Nuclear Quality Assurance Manual" is described in the G/C Corporate Topical Quality Assurance Report GAI-TR-106, Rev. 3, "Quality Assurance Program for Nuclear Power Plants." This program has been approved by the NRC and is in compliance with the requirements of 10CFR50, Appendix B, USNRC Reg. Guide 1.28, ANSI N45.2, and USNRC Reg. Guide 1.64.

4.7.4 Construction Techniques

4.7.4.1 Administrative Controls During Manufacturing and Installation

The Peach Bottom Units 2 and 3 new spent fuel storage racks will be manufactured at the Westinghouse Nuclear Components Division, Pensacola, Florida. This facility is a modern high-quality shop with extensive experience in forming, machining, welding, and assembling nuclear-grade equipment. Forming and welding equipment are specifically designed for

fuel rack fabrication and all welders are qualified in accordance with ASME Code Section IX.

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

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

4.7.4.2 Procedure

The following sequence of events is anticipated for the spent fuel storage rack replacement for PBAPS Units 2 and 3:

- 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.

The final configuration of the new rack modules in the spent fuel pool is shown in Figure 4-4 and 4-5. The installation of these racks will be accomplished in accordance with the following considerations and guidelines:

- o The removal of old racks and installation of new racks will be performed using plant approved procedures.
- o At no time will a rack module be carried directly over another module installed in the spent fuel pool, containing spent fuel.
- o 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"
- o 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

4.8.1 Initial Verification

The neutron absorber rack design includes a neutron absorber verification view-hole in the cell wall so that the presence of neutron absorber 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 additional on-site neutron absorber verification beyond normal receipt inspection activities.

4.8.2 Periodic Verification Surveillance

The neutron absorber 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. The sample coupons will be of a similar thickness as the neutron absorption material used within the storage system. Each 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 neutron absorption 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 specimen contained within.

A series of not less than 24 of the jacketed specimens shall be suspended from rigid straps so designed as to be hung on the outside periphery of a rack module. There are 2 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. 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 1.03×10^{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, PECO will evaluate this information and will modify the surveillance program accordingly.

PECo 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, PECO will determine the scheduling and extent of additional surveillances so as to assure acceptable material performance throughout the life of the plant.

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2. U. S. Nuclear Regulatory Commission, "Standard Review Plan 3.8.4, 'Other Seismic Category I Structures'," Rev. 1, NUREG-0800, July 1981.
3. U. S. Nuclear Regulatory Commission, letter from B. R. Grimes to All Power Reactor Licensees, 4/14/78, with enclosure entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," including Supplement dated 1/18/79.
4. U. S. Nuclear Regulatory Commission, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)," Regulatory Guide 1.142, Rev. 1, October 1981.
5. ACI Committee 318, "Building Code Requirements for Reinforced Concrete (ACI 318-83)," American Concrete Institute, Detroit, Mich., 1983.

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6. ACI Committee 349, "Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-80)," American Concrete Institute, Detroit, Mich., 1980.
7. AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction, New York, N.Y., 1978.
8. AISC, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," American Institute of Steel Construction, New York, N.Y., Final Draft of initial issue, 9/14/83.
9. WECAN, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252.
10. WECAN, "Benchmark Problem Solution Employed for Verification of the WECAN Computer Program," WCAP-8929.
11. C. W. McCormick (Ed.), MSC/NASTRAN, User's Manual, MacNeal-Schwendler Corp., Los Angeles, Calif., May 1983, 2 vol.
12. ADINA, A Finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis, ADINA Engineering, Inc., Watertown, Mass., Report AE 81-1, September 1981.
13. ACI Committee 224, "Control of Cracking in Concrete Structures," American Concrete Institute, Detroit, Mich., Report No. ACI 224R-80, October 1980.

14. J. S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Data," Brand Industries, Inc., Report 748-30-2 (August 1981).
15. J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1 (August 1981).
16. 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).

TABLE 4-1
RACK MODULE DATA (PER UNIT)

<u>QTY</u>	<u>ARRAY</u>	<u>STORAGE LOCATIONS</u>	<u>RACK ASSY DIMENSIONS (INCHES)</u>	<u>DRY WEIGHT (LBS) PER RACK ASSY</u>
1	9 x 14	126	54 x 89 x 180	10,000
2	10 x 14	280	64 x 89 x 180	11,200
1	11 x 14 Mod.	119	70 x 89 x 180	9,500
1	12 x 15	180	76 x 95 x 180	14,400
1	12 x 17	204	76 x 107 x 180	16,300
2	12 x 20	480	76 x 126 x 180	19,200
2	15 x 19	570	95 x 120 x 180	22,800
1	17 x 20	340	107 x 126 x 180	27,200
<u>4</u>	19 x 20	<u>1520</u>	120 x 126 x 180	30,400

15 racks 3819

Storage locations center-to-center spacing (inches) 6.28

Storage cell inner dimension (inches) 6.07

Intermediate storage location inner dimension (inches) 6.12

Type of Fuel BWR 8 x 8
BWR 8 x 8 (R)
BWR 7 x 7

TABLE 4-2
STORAGE RACK LOADS AND LOAD COMBINATIONS

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

Notes:

1. The abbreviations in the table above are those used in Standard Review Plan (SRP) Section 3.8.4 where each term is defined except for T_a , which is defined here as the highest temperature associated with the postulated abnormal design conditions. F_d is the force caused by the accidental drop of the heaviest load from the maximum possible height, and P_f is the upward force on the racks caused by a postulated stuck fuel assembly.
2. The provisions of NF-3231.1 of ASME Section III, Division I, shall be amended by the requirements of Paragraphs c.2, 3 and 4 of Regulatory Guide 1.124, entitled "Design Limits and Load Combinations for Class A Linear-Type Component Supports."
3. For the faulted load combination, thermal loads were neglected when they are secondary and self-limiting in nature and the material is ductile.

TABLE 4-3
SPENT FUEL POOL
GOVERNING DESIGN LOAD COMBINATIONS

Reinforced Concrete

1. $U = 1.4 + 1.4 + 1.7T_0$
2. $U = 1.4D + 1.4F$
3. $U = 1.4D + 1.4F + 1.7L + 1.9E$
4. $U = D + F + L + E' + T_a$
5. $U = D + F + L + E'$
6. $U = 1.05D + 1.05F + 1.3L + 1.43E + 1.3T_0$

Structural Steel

7. $Y = 1.7D + 1.7F + 1.7L + 1.7E$
8. $Y = 1.3D + 1.3F + 1.3L + 1.3E + 1.3T_0$
9. $Y = 1.1 (D + F + L + E' + T_a)$

Notation:

D = dead load

E = OBE (design earthquake)

E' = SSE (maximum credible earthquake)

L = live load

T_a = thermal load produced by accident condition

T_0 = thermal load during normal operation

U = section strength required to design loads based on the Strength Design method for reinforced concrete

Y = section strength required to resist design loads based on Plastic Design method for structural steel

TABLE 4-4
SLAB DESIGN LOAD SUMMARY

<u>MECHANICAL LOADS</u>	<u>TOTALS (KIP)</u>
Dead	1,475.5
Live	252.8
Hydrostatic	3,373.0
Fuel Rack Buoyant Weight	3,823.1
Vertical OBE	2,442.4
Vertical SSE	5,001.7
<u>THERMAL LOADS</u>	<u>TEMPERATURE (°F)</u>
Accident, mean	98.5
Accident, gradient	140.2 (±70.1)
Normal, mean	113.0
Normal, gradient	63.2 (±31.6)

Notes:

1. Dead load includes weight of slab and six W36 X 230 beams.
2. Live load includes weight of a buoyant loaded cask plus miscellaneous floor loads.
3. Seismic loads include structural, hydrodynamic, and fuel rack contributions.
4. Fuel rack buoyant weights and fuel rack seismic loads used in the SFP analyses represent the maximum allowable loads.

TABLE 4-5
COMPARISON OF POOL SLAB VERTICAL DESIGN LOADS

<u>DESIGN LOADS</u>		<u>WEIGHT (LB)</u>
<u>Original Design (pre-1978)</u>		
Water	2,387 lb/ft ² (40.0 x 35.33')	= 3,373,308
Other loads	2,000 lb/ft ² (40.0 x 35.33')	= 2,826,400
Slab (self weight + beams)	150 lb/ft ³ (6.7 x 40.0 x 35.33')	
	+ 230 lb/ft x 40.0' x 6 beams	= <u>1,475,500</u>
	Total	= 7,675,208
<u>Current Design (1978)</u>		
Water		= 3,373,308
Other loads	4,000 lb/ft ² (40.0 x 35.33')	= 5,652,800
Slab (self weight + beams)		= <u>1,475,500</u>
	Total	= 10,501,608
<u>New Safety Investigation (1984)</u>		
Water		= 3,373,308
Fuel rack buoyant weight		= 3,823,080
Seismic loads (structural, rack, hydrodynamic)		= 2,442,400
Slab (self weight + beams)		= <u>1,475,500</u>
	Total	= 11,114,268

Notes:

1. Seismic loads given above for this current investigation are OBE loads.
2. Total loads are listed here for comparison purposes and are not to be used as the basis for determining maximum allowable floor loads.

TABLE 4-6
 DRYWELL SHIELD WALL ORIGINAL DESIGN SUMMARY¹

<u>SECTION</u>	<u>LOAD COMBINATION</u>	<u>ALLOWABLE² STRESS (KSI)</u>	<u>MAXIMUM STRESS (KSI)</u>	<u>PERCENTAGE OF ALLOWABLE USED</u>
E1. 145'-0"	D + E	24.0	8.2	34
Circumferential tension reinforcement	D + T + E'	54.0	23.5	44
E1. 145'-0"	D + E	1.80	0.45	25
Concrete vertical compression	D + T + E'	3.40	1.57	46
E1. 180'-0"	D + E	0.07	0.0625	89
Concrete shear	D + T + E'	0.253	0.174	69

Notes:

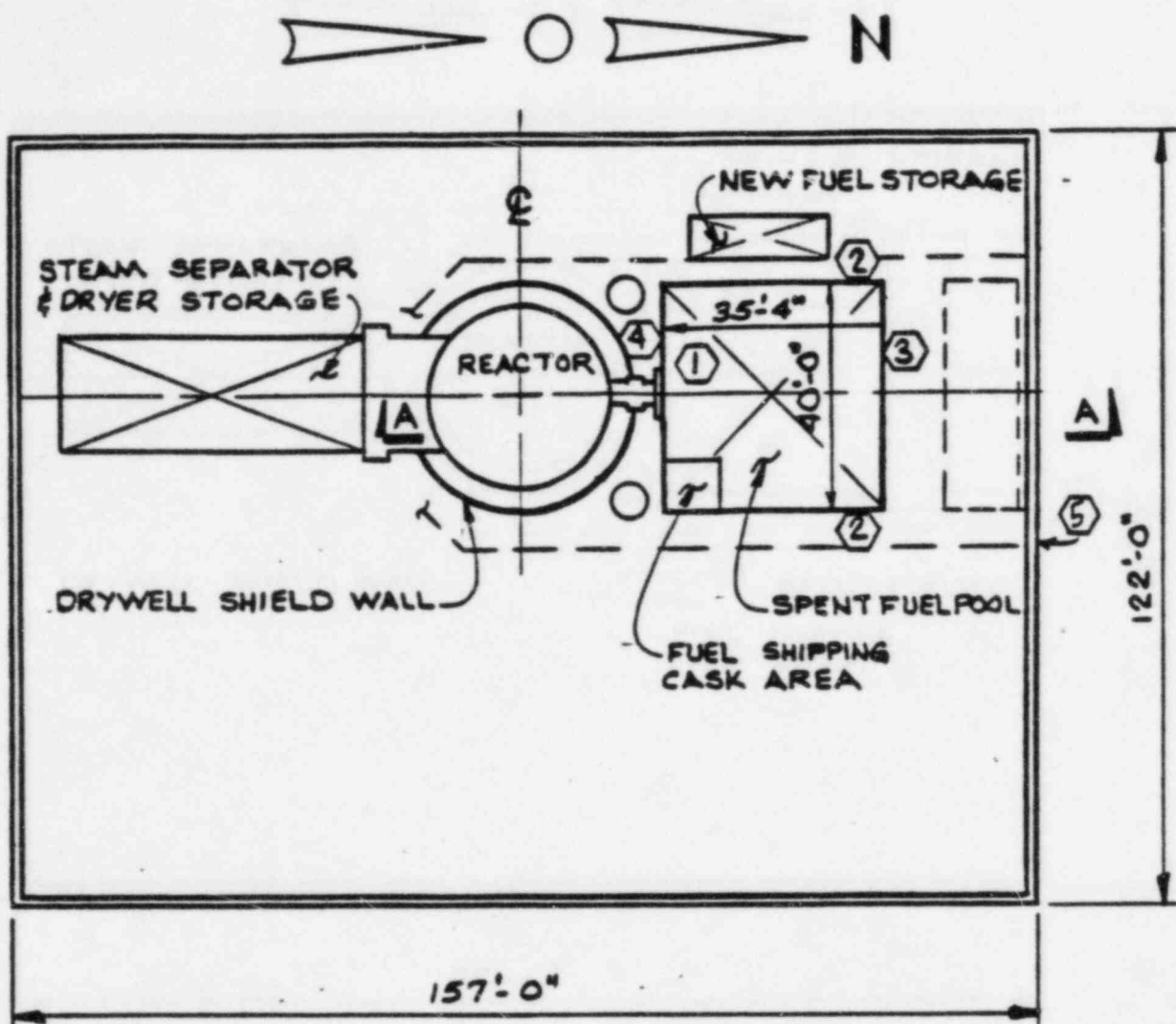
1. Based on FSAR Table C.4.5.
2. Based on ACI 318-63 Working Stress Design Methods.

TABLE 4-7
MAXIMUM ALLOWABLE FUEL RACK/POOL FLOOR INTERFACE LOADS

NO.	LOAD COMBINATION	TOTAL LOADS		LOCAL BEARING (KSI)
		VERTICAL (KIP)	HORIZONTAL (KIP)	
1.	D + L	3,900.0 ¹	N/A	2.4
2.	D + L + T _O	3,900.0 ¹	N/A	2.4
3.	D + L + T _O + E	5,700.0	1,900.0	2.4
4.	D + L + T _a + E	5,700.0	1,900.0	2.4
5.	D + L + T _O + P _f	5,700.0	N/A	3.2
6.	D + L + T _a + E'	8,000.0	3,000.0	3.2
7.	D + L + F _d	8,000.0	N/A	3.2
<u>Alternate¹</u>				
8.	1.4 (D + L + T _O) + 1.9E	8,900.0	3,600.0	See Note 2
9.	1.4 (D + L + T _a) + 1.9E	8,900.0	3,600.0	See Note 2
10.	1.7 (D + L + T _O + E)	9,700.0	3,200.0	See Note 2
11.	1.7 (D + L + T _a + E)	9,700.0	3,200.0	See Note 2

Notes:

1. Additional structural limits specified in Load Combination No. 8, 9, 10, and 11 shall be satisfied if total vertical loads calculated for Load Combination No. 1 and 2 are less than 3,700.0 kip. Otherwise, Load Combination No. 8, 9, 10, and 11 may be used in lieu of Load Combination No. 1, 2, 3, 4, and 5.
2. When total loads are evaluated using Load Combination No. 8, 9, 10, and 11, local bearing pressures shall satisfy Load Combination No. 1, 2, 3, 4, and 5.
3. Notations used in this table are the same as defined in SRP 3.8.4, Appendix D.



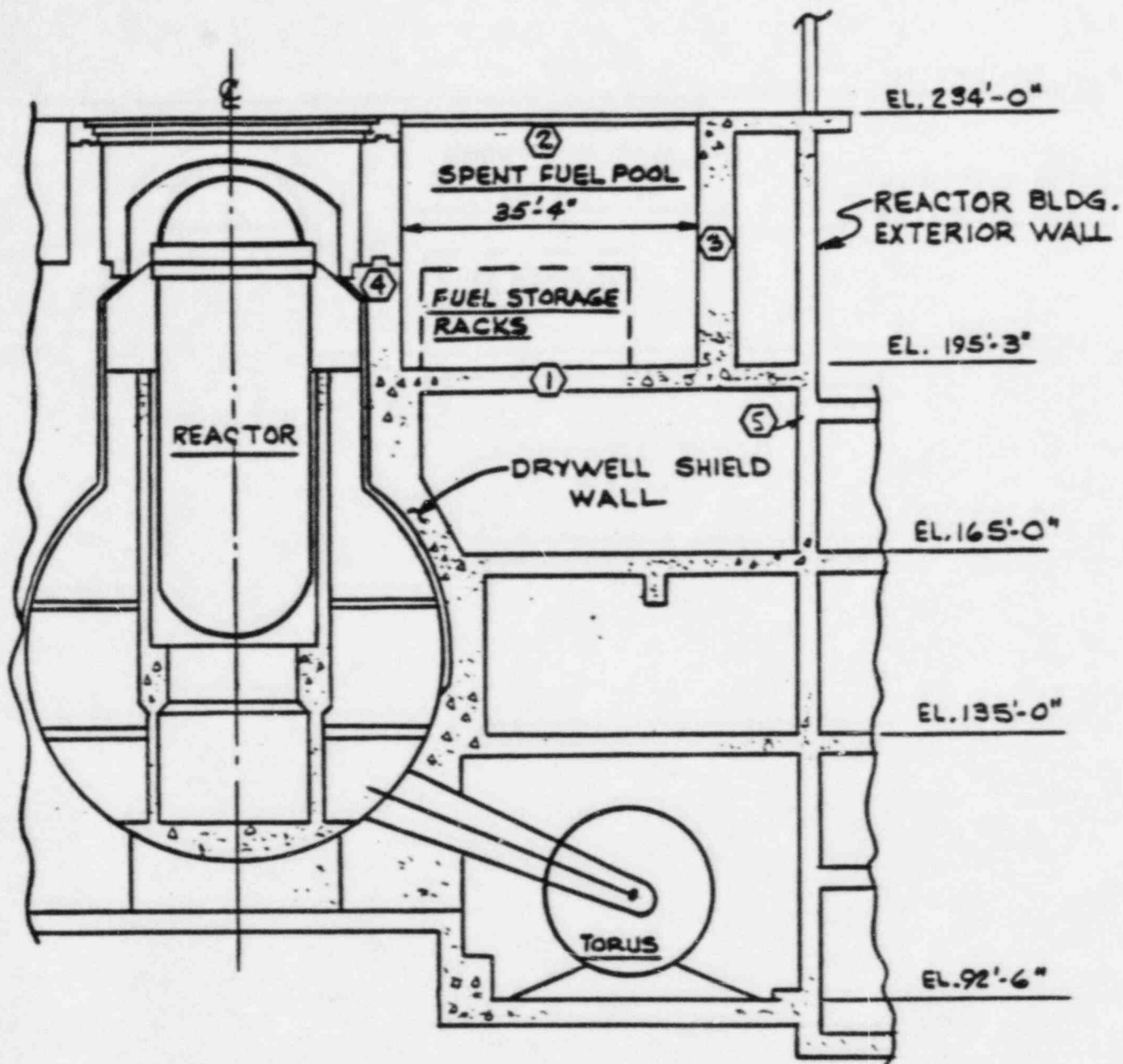
KEY STRUCTURAL ELEMENTS

- ① POOL SLAB
- ② POOL WALL
- ③ POOL CROSS WALL
- ④ SHIELD WALL
- ⑤ R.B. EXTERIOR WALL

REACTOR BUILDING

OPERATING FLOOR PLAN AT ELEVATION 234'-0"
UNIT 2 - SHOWN
UNIT 3 - OPP. HAND

FIGURE 4-1
 UNIT 2 SPENT FUEL POOL PLAN VIEW

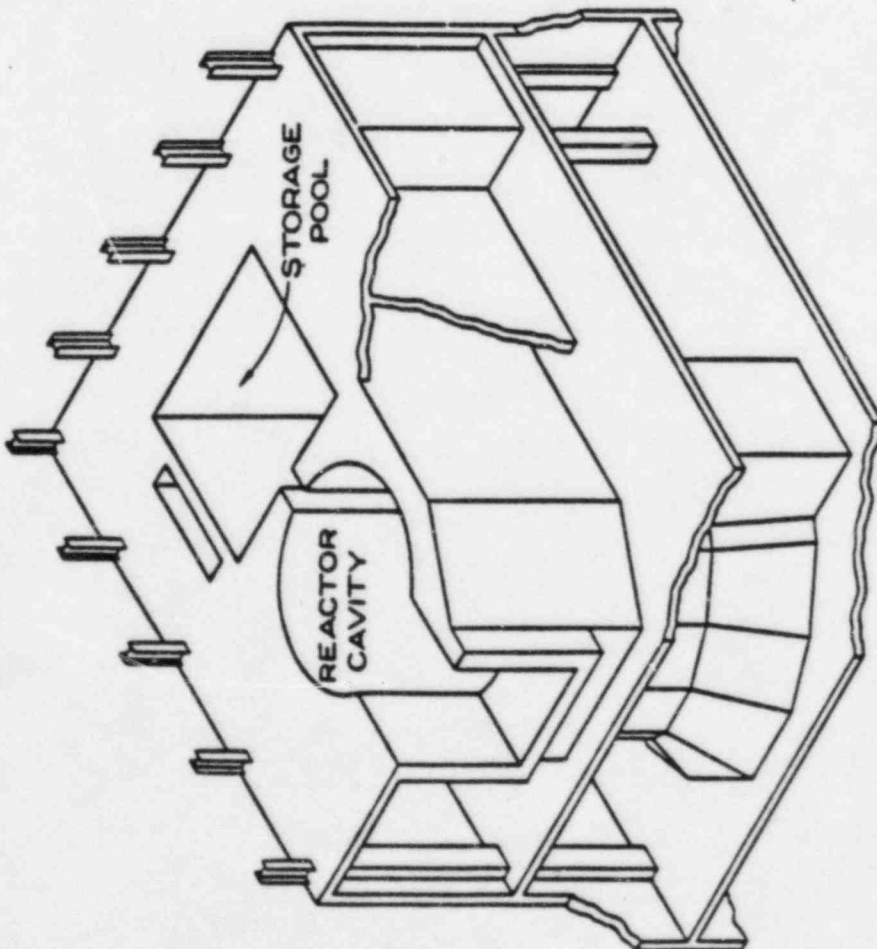


SECTION A-A

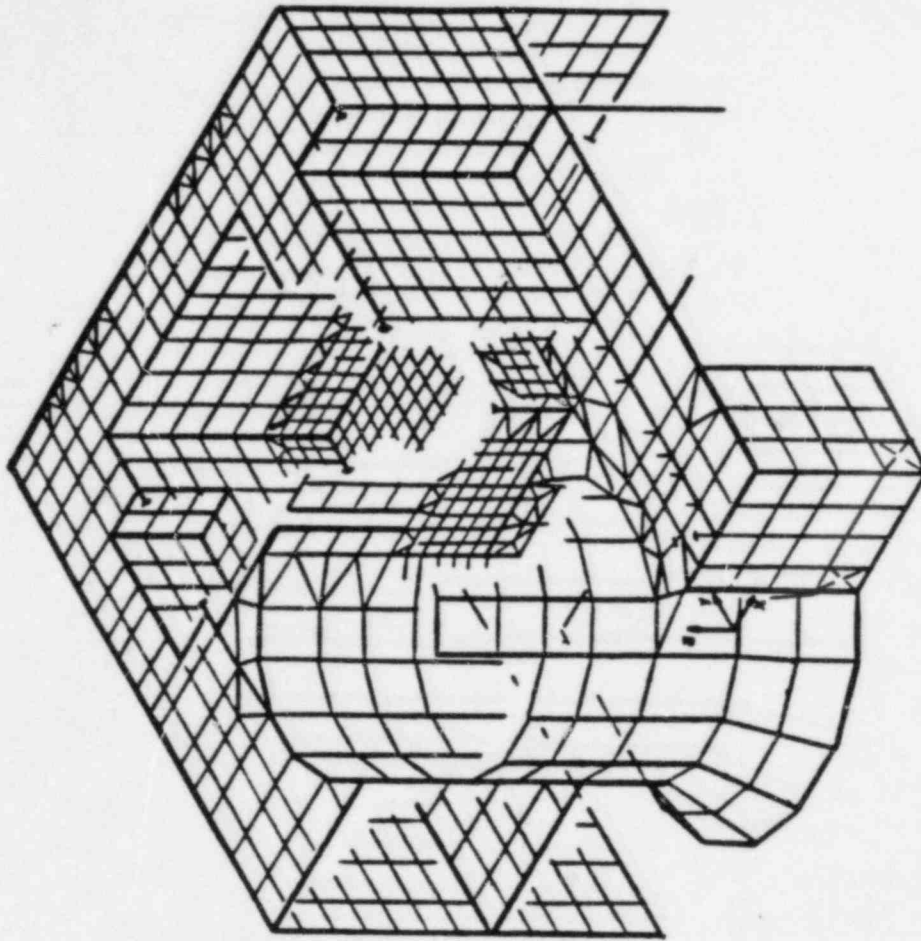
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- ⑤ R.B. EXTERIOR WALL

FIGURE 4-2
UNIT 2 SPENT FUEL POOL CROSS SECTION



(a) Structure



(b) Finite Element Model

FIGURE 4-3
UNIT 2 SPENT FUEL POOL ISOMETRIC VIEW

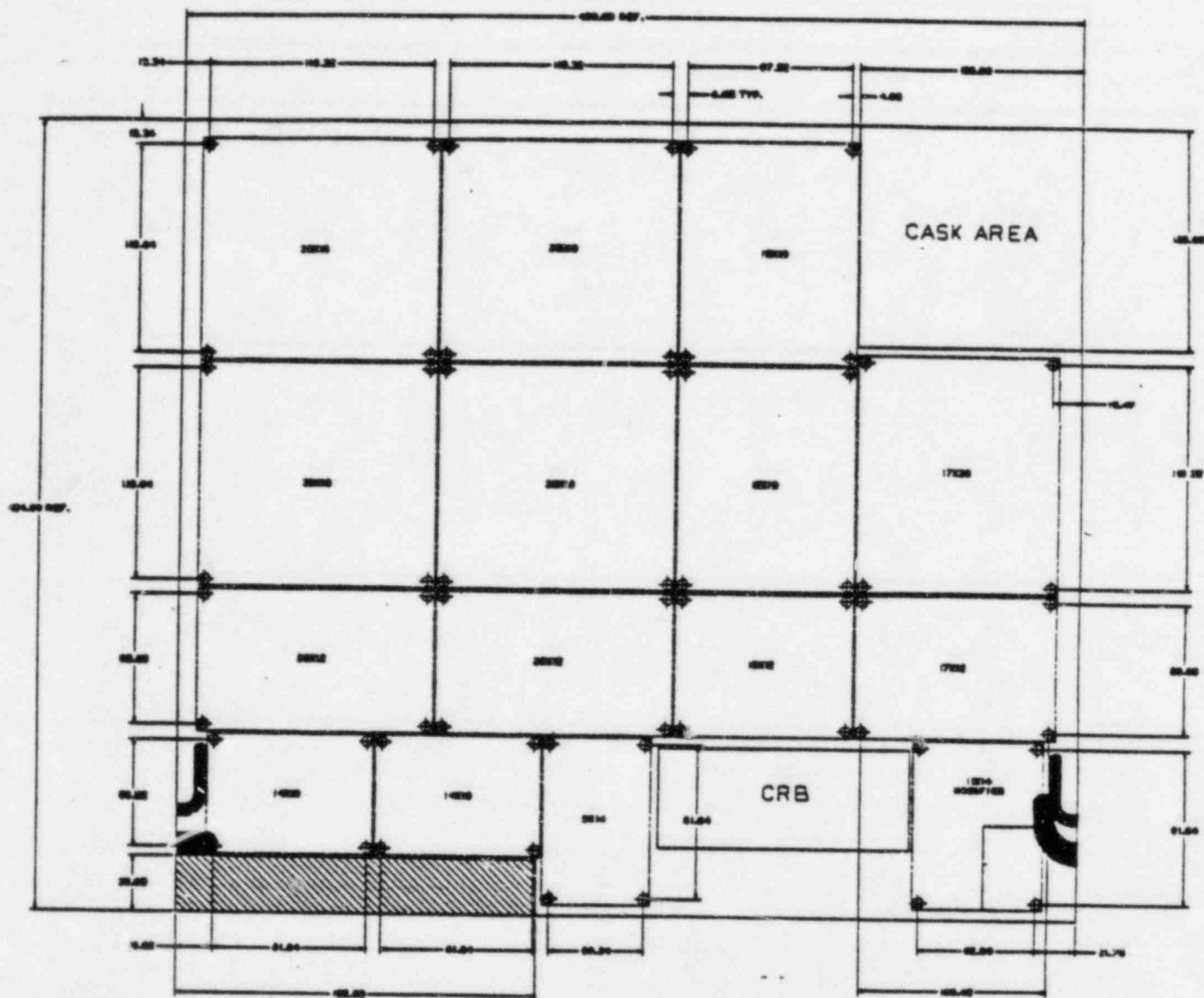


FIGURE 4-5
 SPENT FUEL POOL STORAGE RACK ARRANGEMENT UNIT 3
 TOTAL LOCATIONS = 3819
 (USABLE LOCATIONS = 3814)

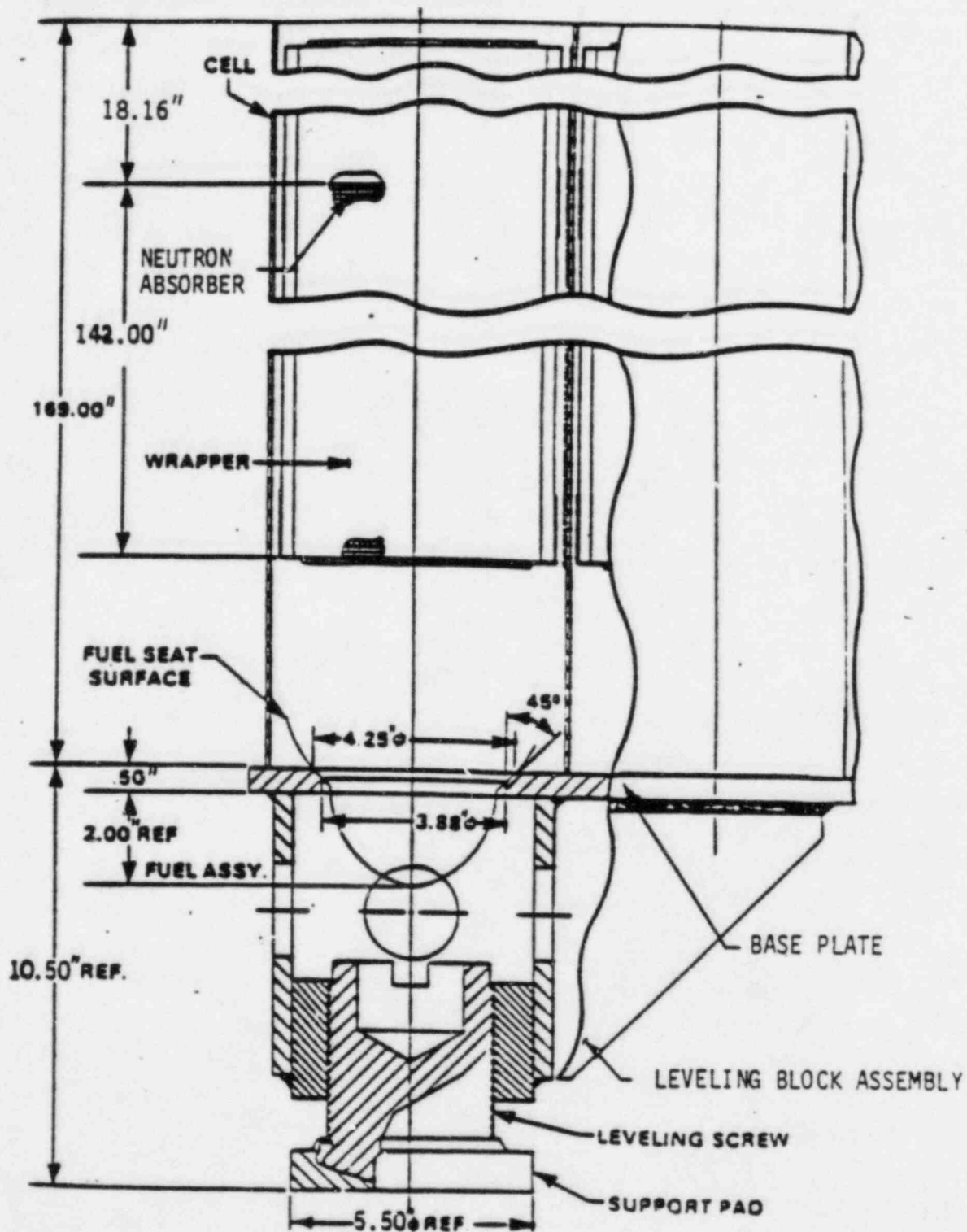


FIGURE 4-6
FUEL STORAGE RACK ASSEMBLY DETAILS

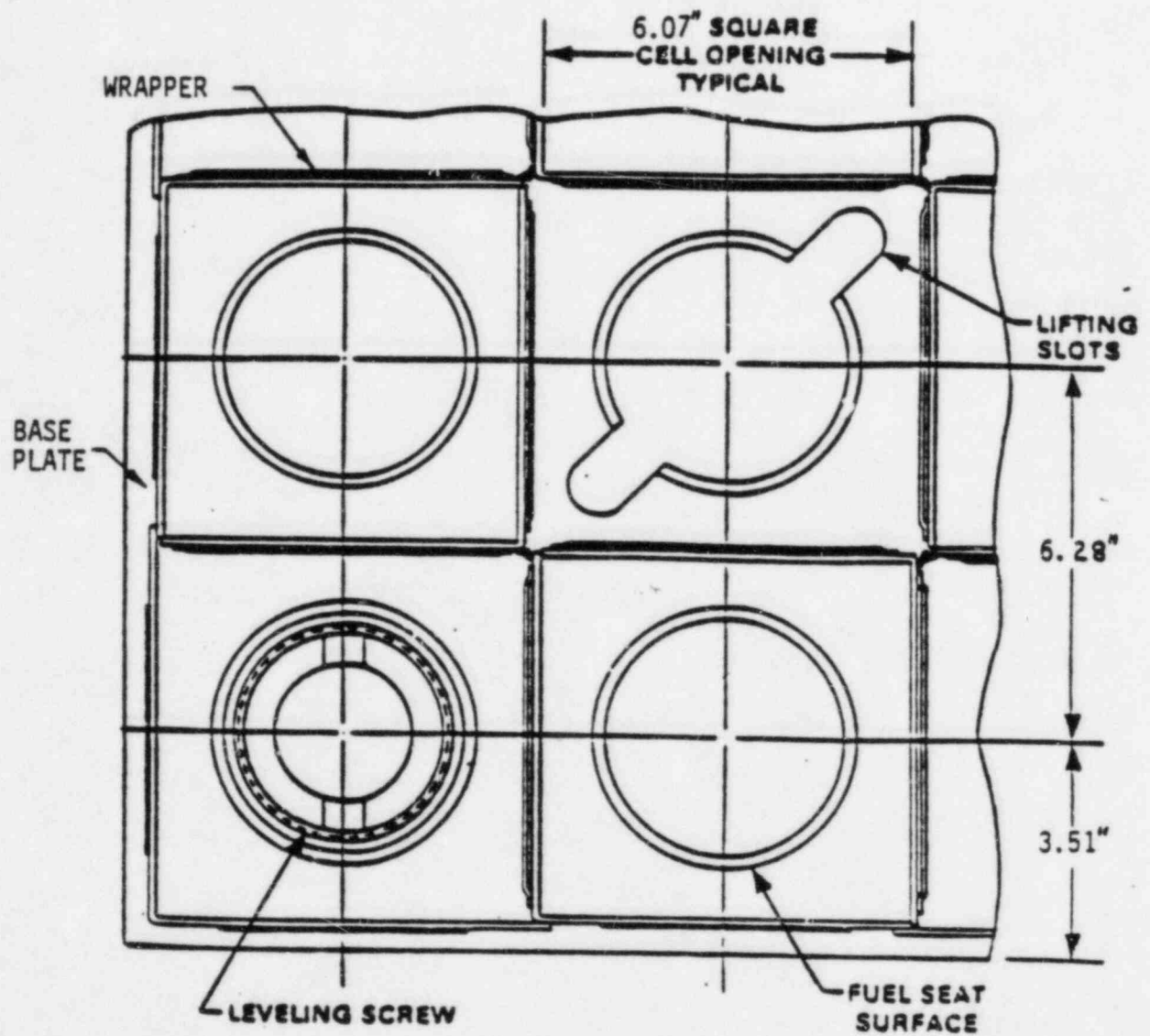


FIGURE 4-7
FUEL RACK PLAN VIEW

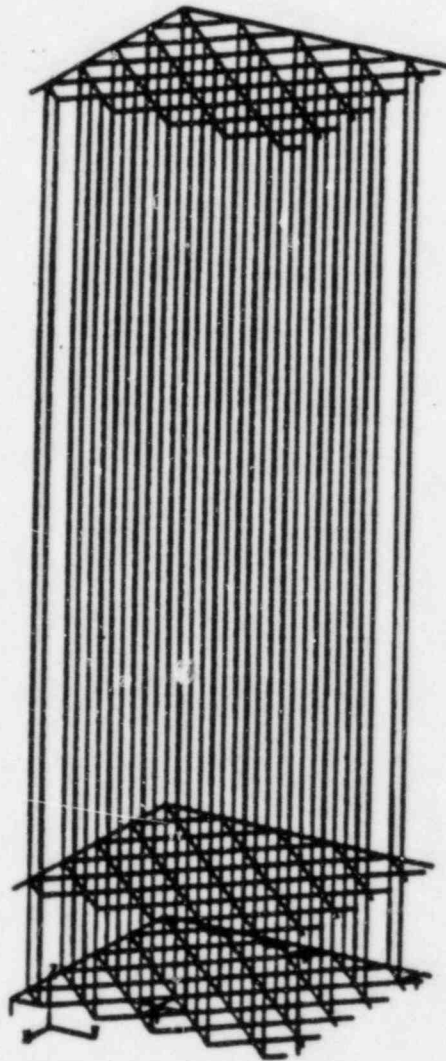


FIGURE 4-8
STRUCTURAL MODEL (QUARTER RACK)

TABLE 4-3
SPENT FUEL POOL
GOVERNING DESIGN LOAD COMBINATIONS

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7. $Y = 1.7D + 1.7F + 1.7L + 1.7E$
8. $Y = 1.3D + 1.3F + 1.3L + 1.3E + 1.3T_O$
9. $Y = 1.1 (D + F + L + E' + T_a)$

Notation:

D = dead load

E = OBE (design earthquake)

E' = SSE (maximum credible earthquake)

L = live load

T_a = thermal load produced by accident condition

T_O = thermal load during normal operation

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<u>THERMAL LOADS</u>	<u>TEMPERATURE (°F)</u>
Accident, mean	98.5
Accident, gradient	140.2 (±70.1)
Normal, mean	113.0
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Notes:

1. Dead load includes weight of slab and six W36 X 230 beams.
2. Live load includes weight of a buoyant loaded cask plus miscellaneous floor loads.
3. Seismic loads include structural, hydrodynamic, and fuel rack contributions.
4. Fuel rack buoyant weights and fuel rack seismic loads used in the SFP analyses represent the maximum allowable loads.

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 DRYWELL SHIELD WALL ORIGINAL DESIGN SUMMARY¹

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E1. 145'-0"	D + E	24.0	8.2	34
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E1. 180'-0"	D + E	0.07	0.0625	89
Concrete shear	D + T + E'	0.253	0.174	69

Notes:

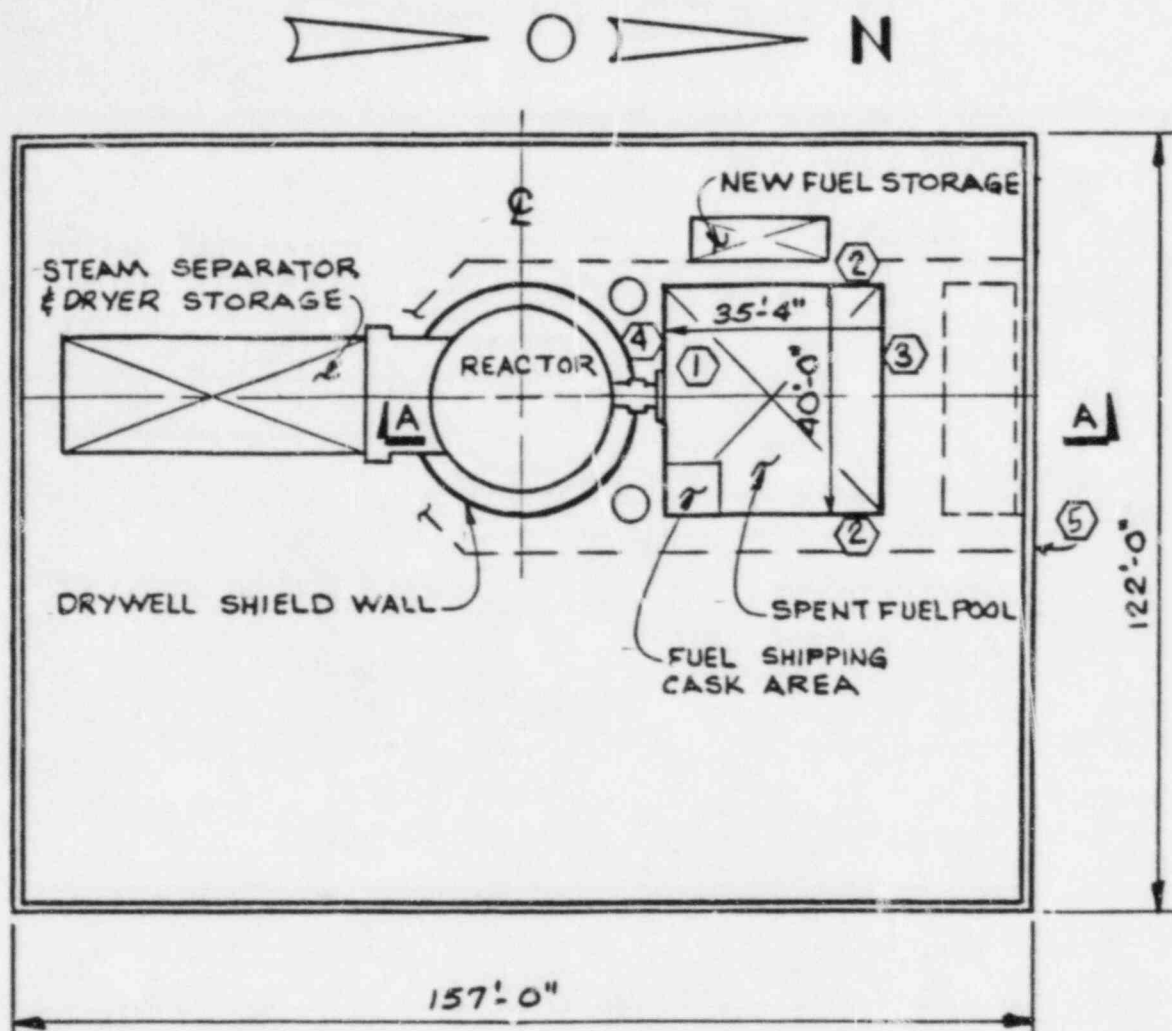
1. Based on FSAR Table C.4.5.
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TABLE 4-6
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4.	D + L + T _a + E	5,700.0	1,900.0	2.4
5.	D + L + T _O + P _f	5,700.0	N/A	3.2
6.	D + L + T _a + E'	8,000.0	3,000.0	3.2
7.	D + L + F _d	8,000.0	N/A	3.2
	<u>Alternate¹</u>			
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10.	1.7 (D + L + T _O + E)	9,700.0	3,200.0	See Note 2
11.	1.7 (D + L + T _a + E)	9,700.0	3,200.0	See Note 2

Notes:

1. Additional structural limits specified in Load Combination No. 8, 9, 10, and 11 shall be satisfied if total vertical loads calculated for Load Combination No. 1 and 2 are less than 3,700.0 kip. Otherwise, Load Combination No. 8, 9, 10, and 11 may be used in lieu of Load Combination No. 1, 2, 3, 4, and 5.
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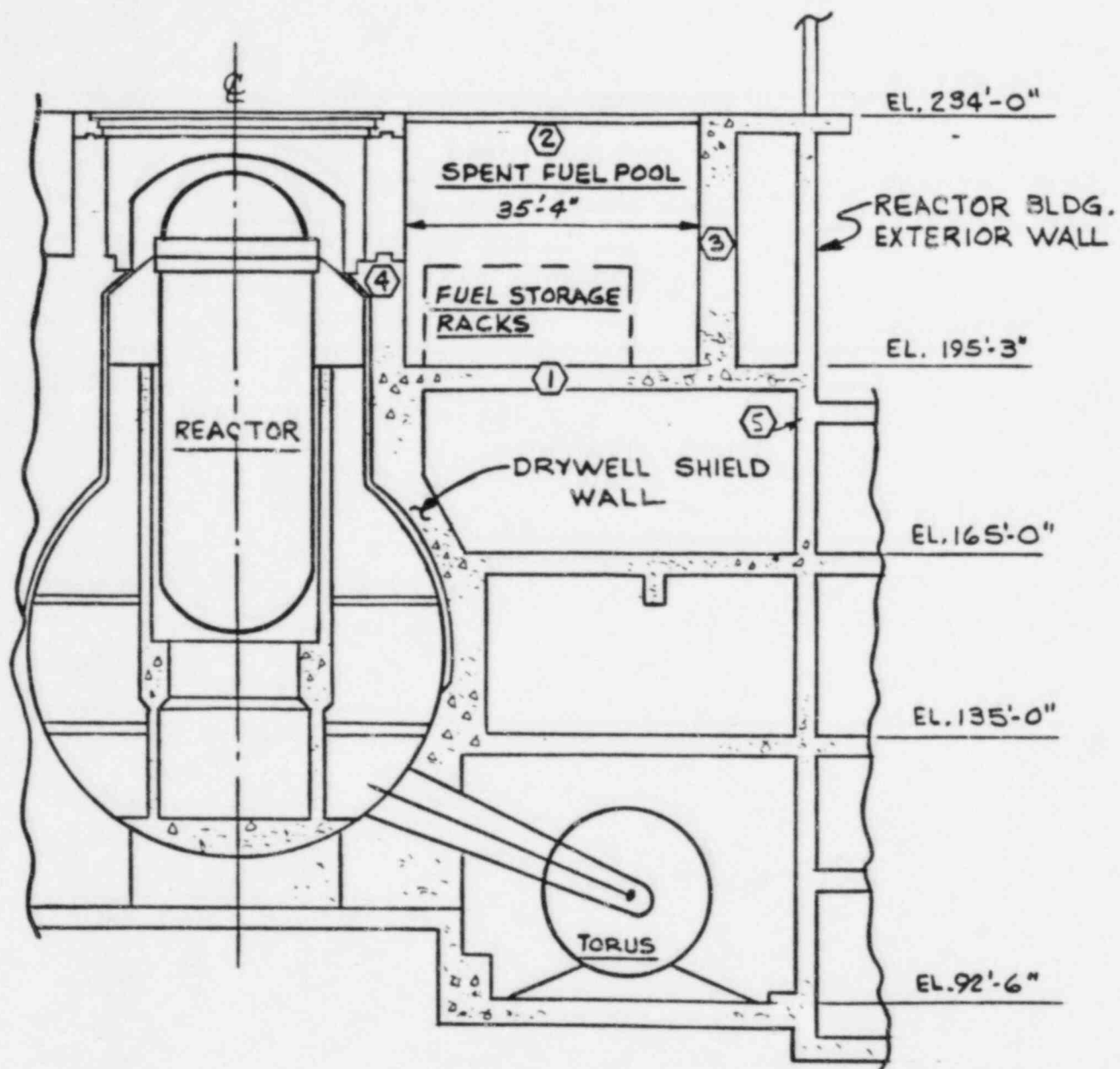
KEY STRUCTURAL ELEMENTS

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- ⑤ R.B. EXTERIOR WALL

REACTOR BUILDING

OPERATING FLOOR PLAN AT ELEVATION 234'-0"
UNIT 2 - SHOWN
UNIT 3 - OPP. HAND

FIGURE 4-1
UNIT 2 SPENT FUEL POOL PLAN VIEW

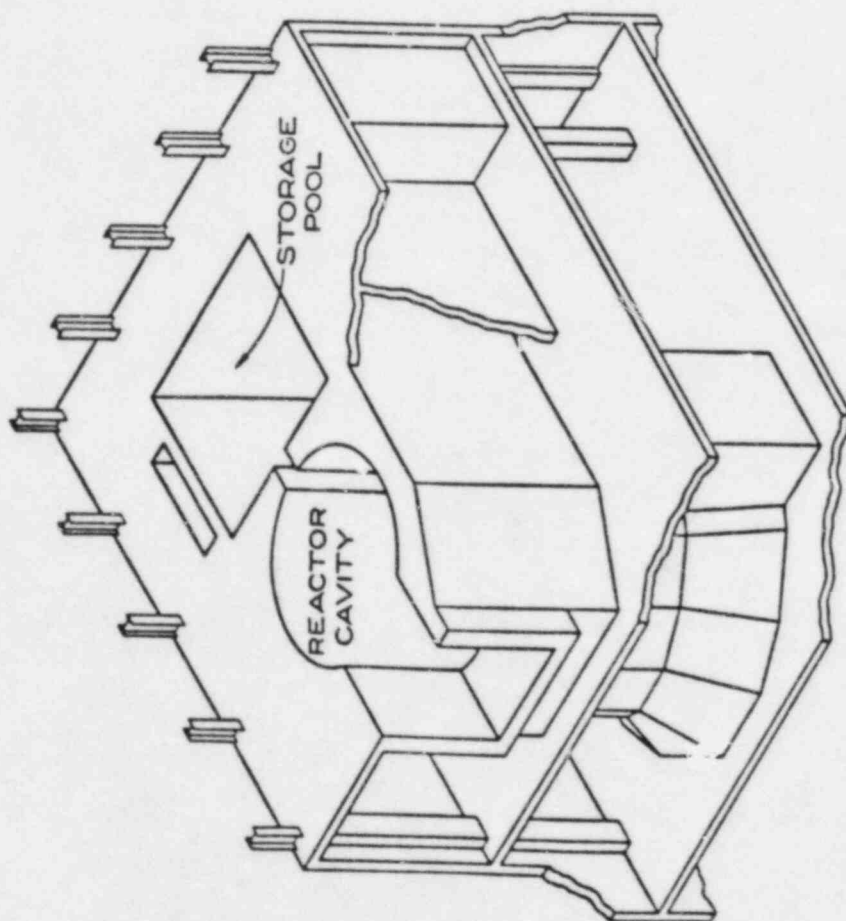


SECTION A-A

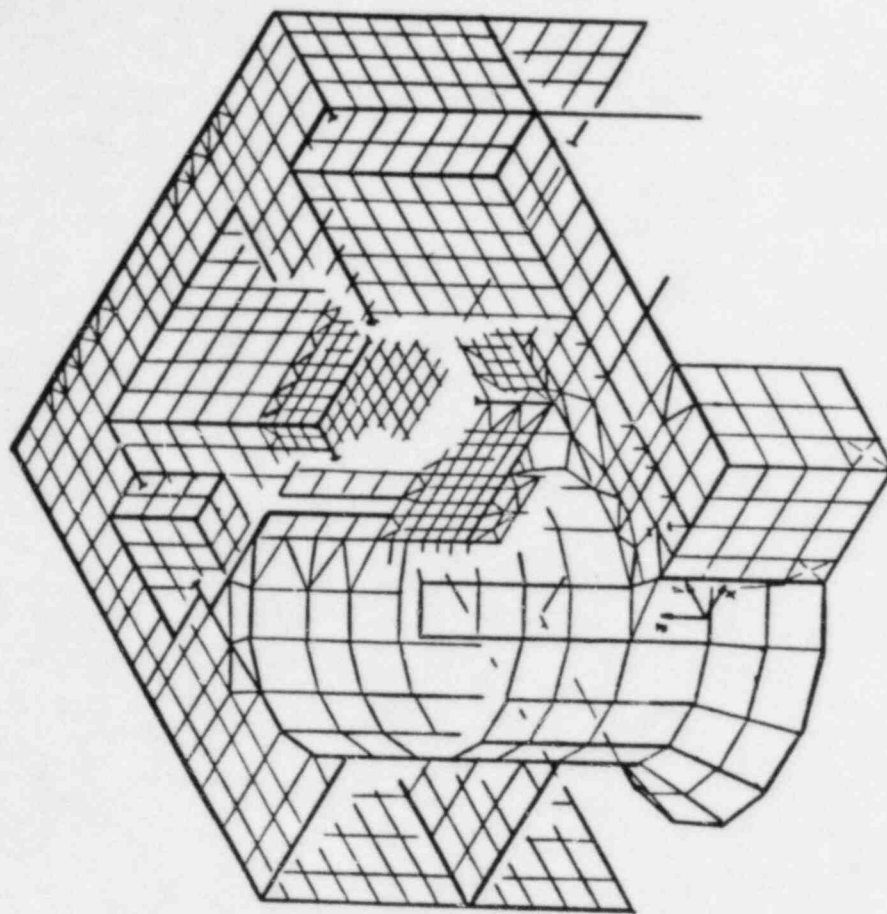
KEY STRUCTURAL ELEMENTS

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- ③ POOL CROSS WALL
- ④ SHIELD WALL
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FIGURE 4-2
UNIT 2 SPENT FUEL POOL CROSS SECTION



(a) Structure



(b) Finite Element Model

FIGURE 4-3
UNIT 2 SPENT FUEL POOL ISOMETRIC VIEW

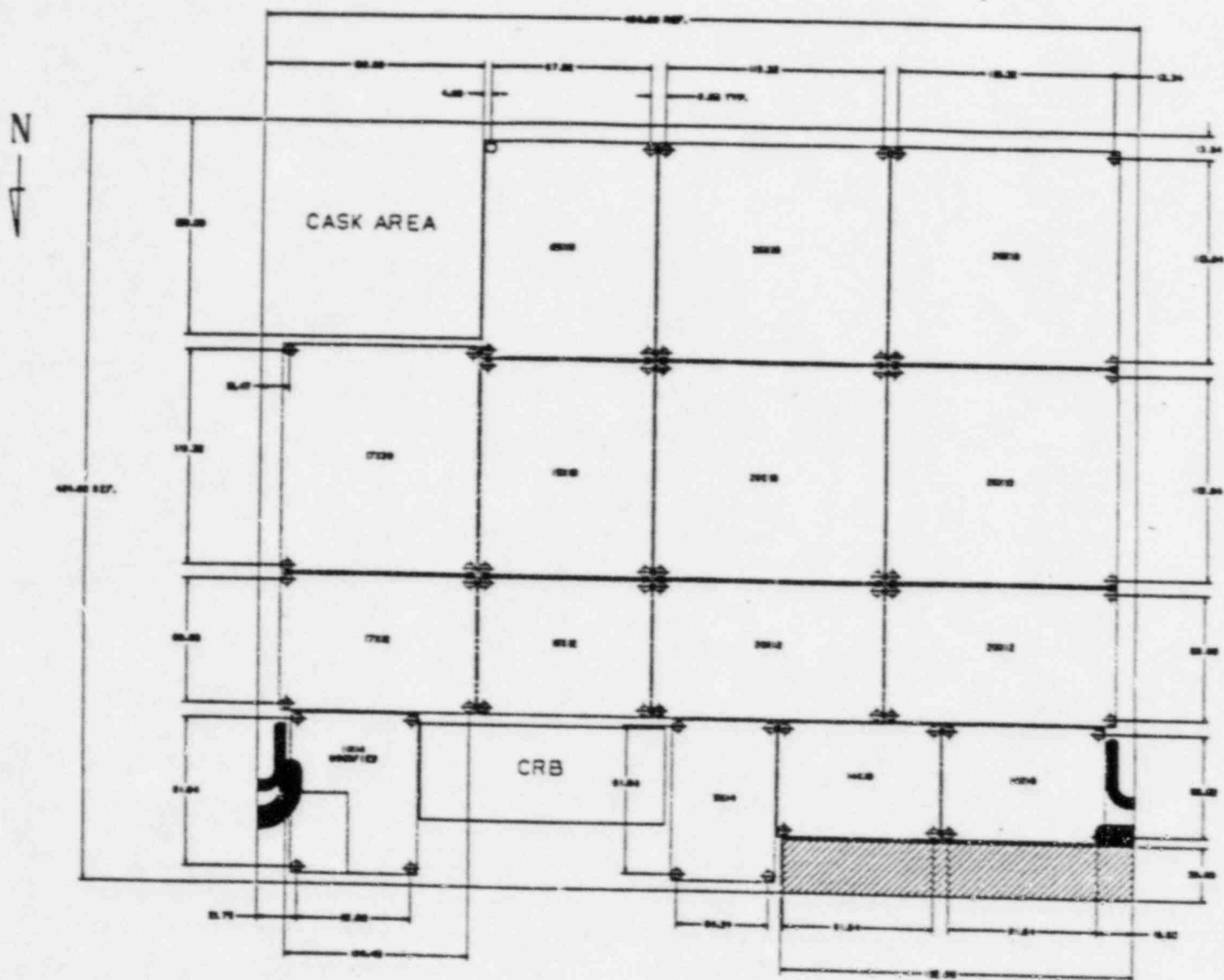


FIGURE 4-4
 SPENT FUEL POOL STORAGE RACK ARRANGEMENT UNIT 2
 TOTAL LOCATIONS = 3819
 (USABLE LOCATIONS = 3814)

N

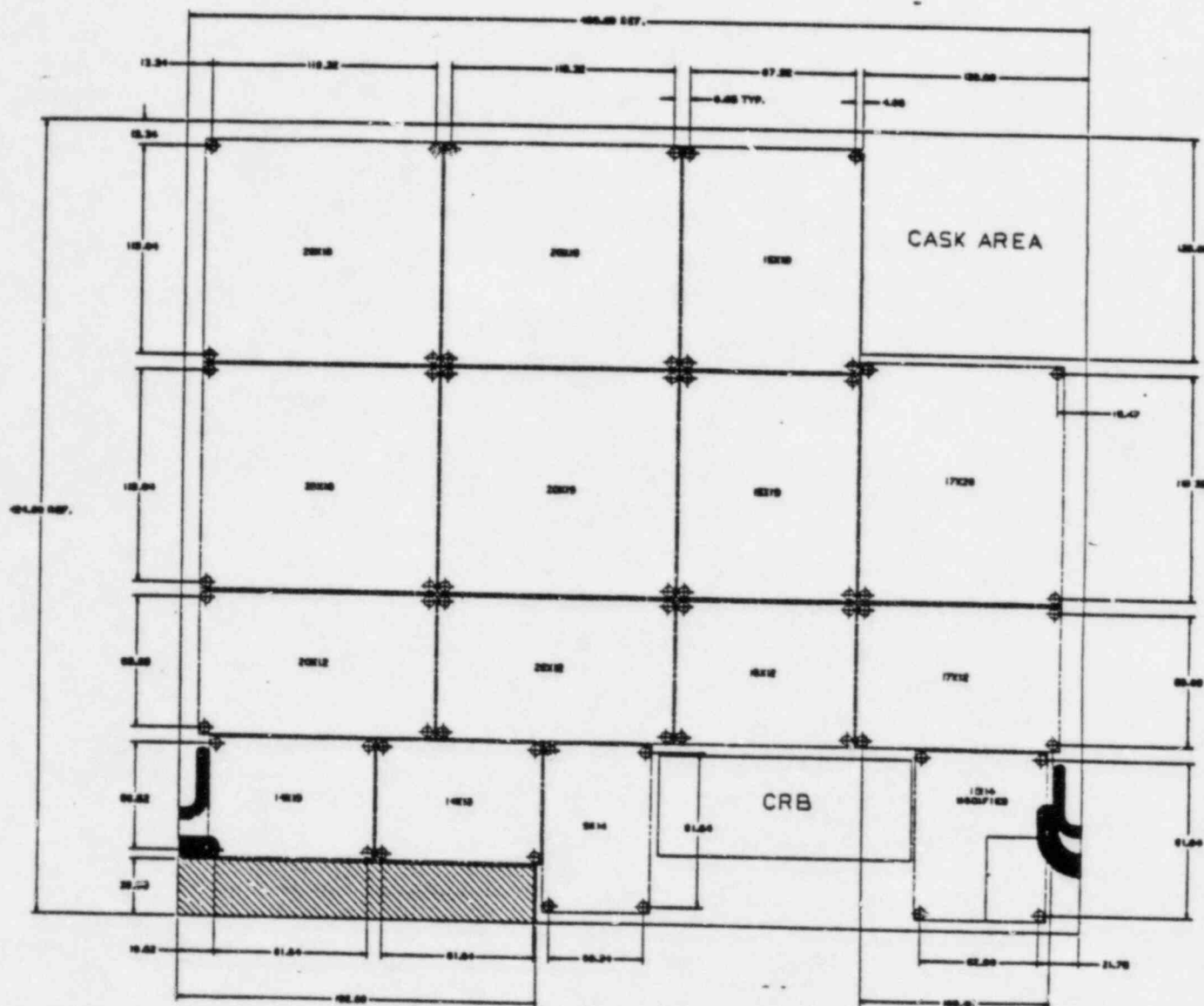


FIGURE 4-5
 SPENT FUEL POOL STORAGE RACK ARRANGEMENT UNIT 3
 TOTAL LOCATIONS = 3819
 (USABLE LOCATIONS = 3814)

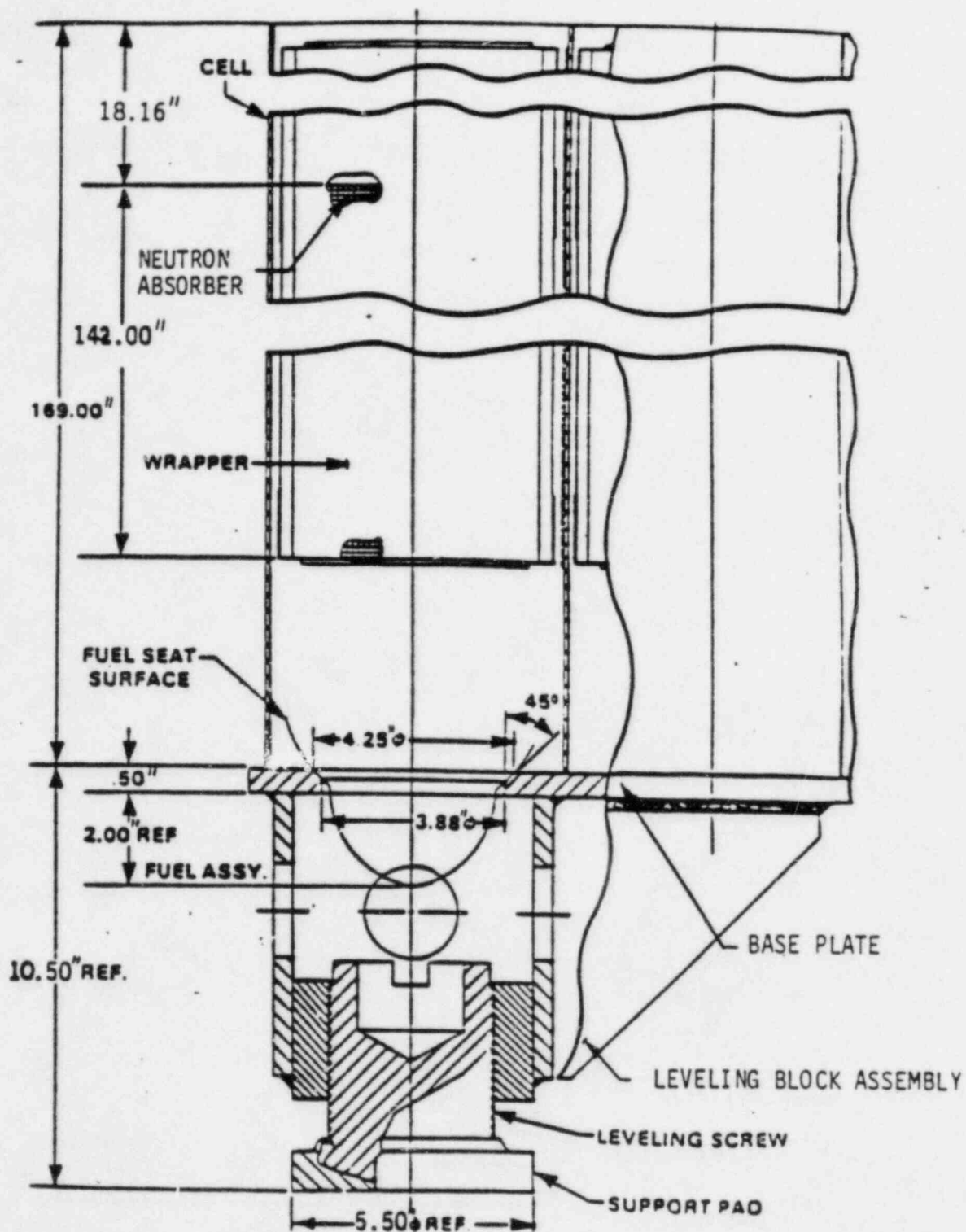


FIGURE 4-6
FUEL STORAGE RACK ASSEMBLY DETAILS

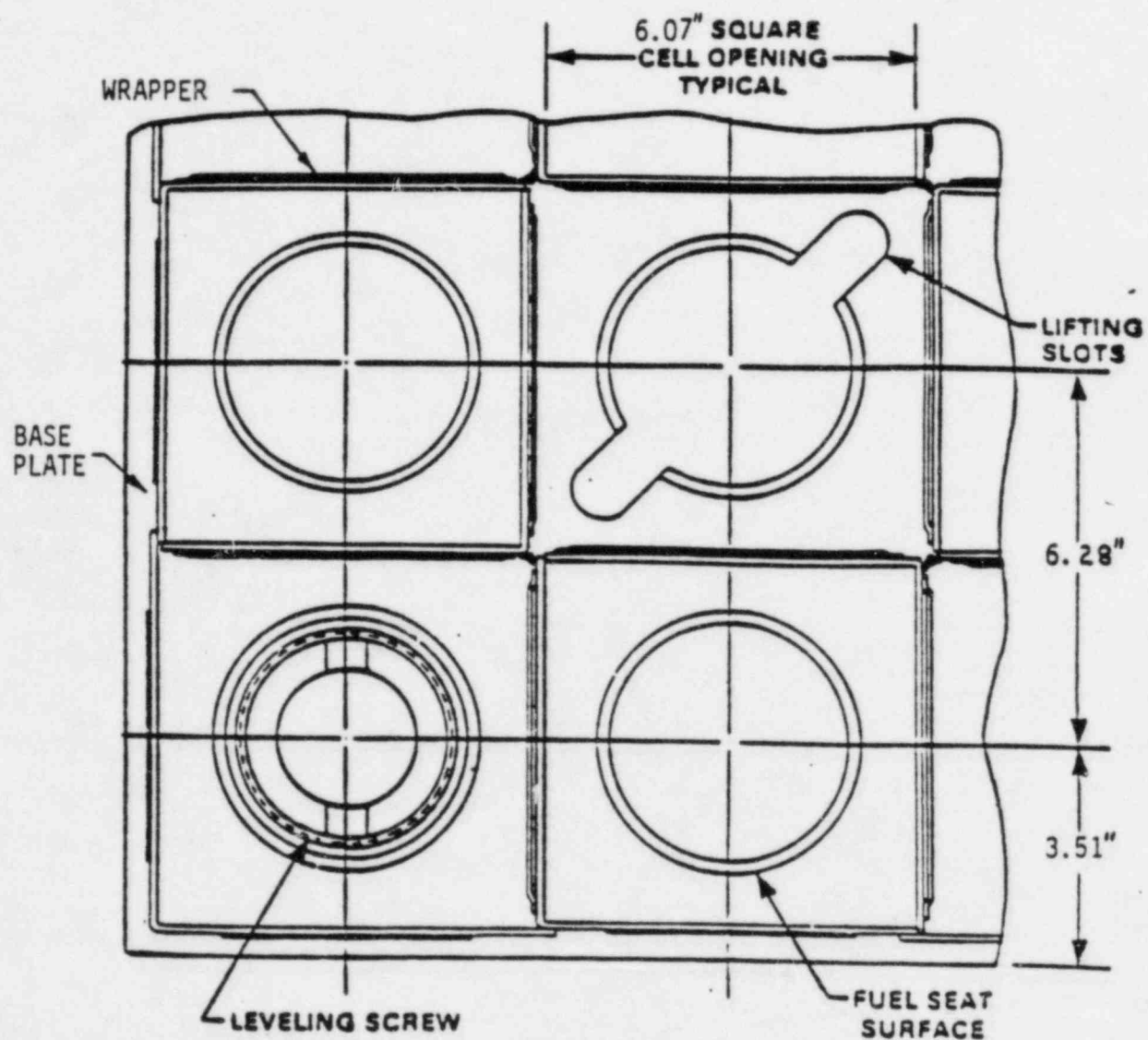


FIGURE 4-7
FUEL RACK PLAN VIEW

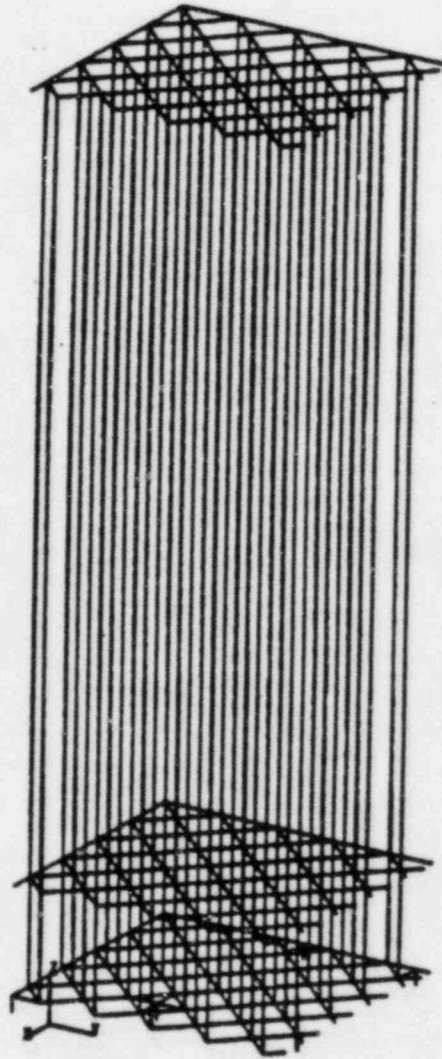


FIGURE 4-8
STRUCTURAL MODEL (QUARTER RACK)

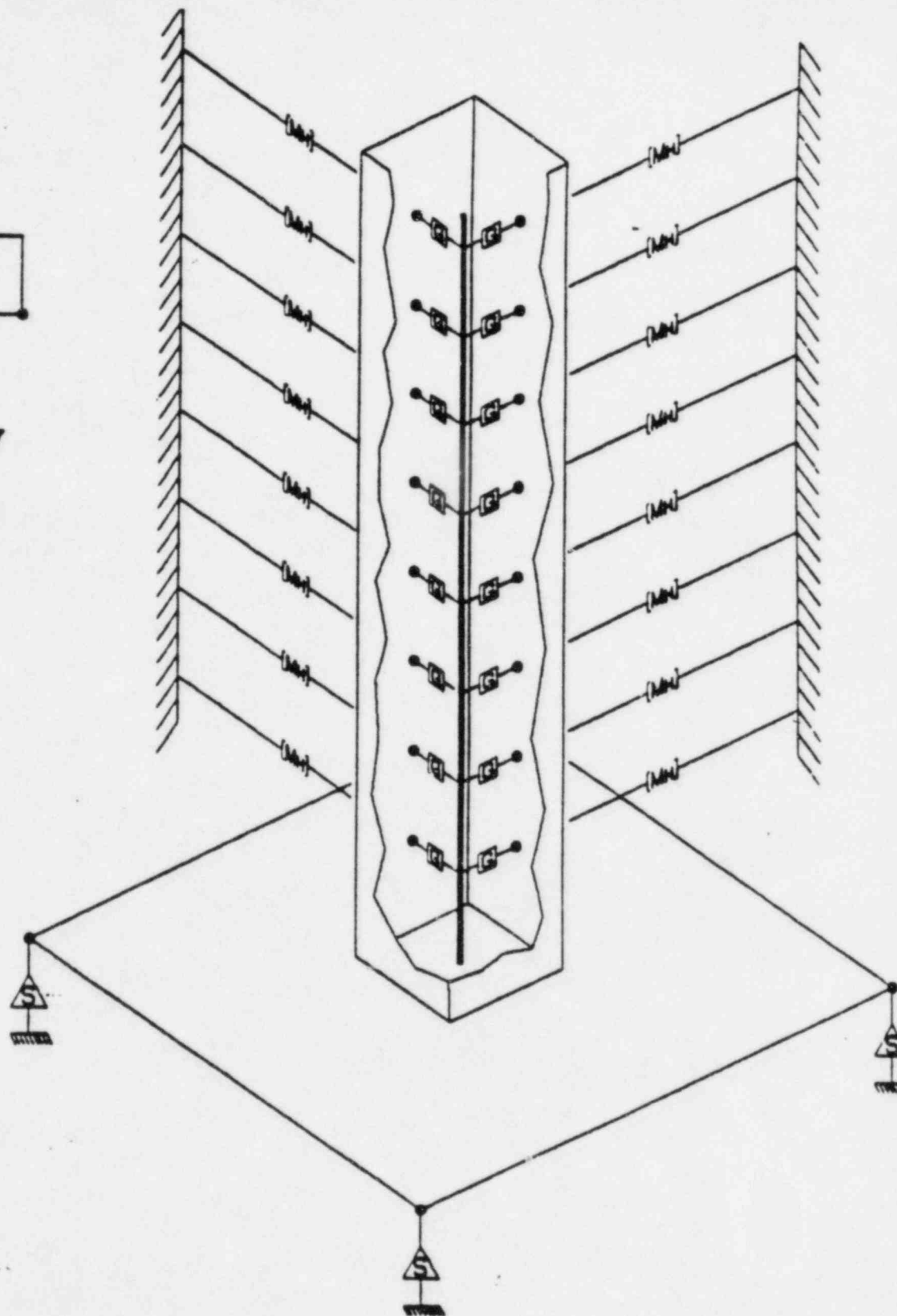
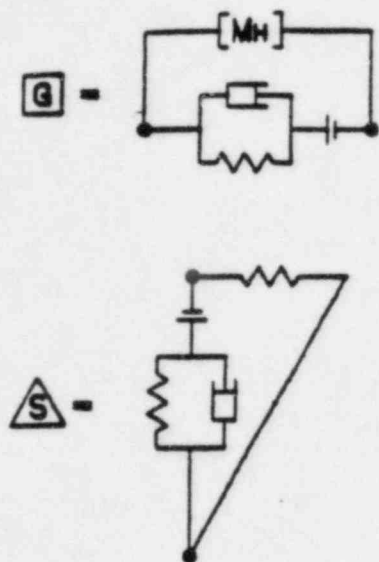


FIGURE 4-9

3-D NONLINEAR SEISMIC MODEL

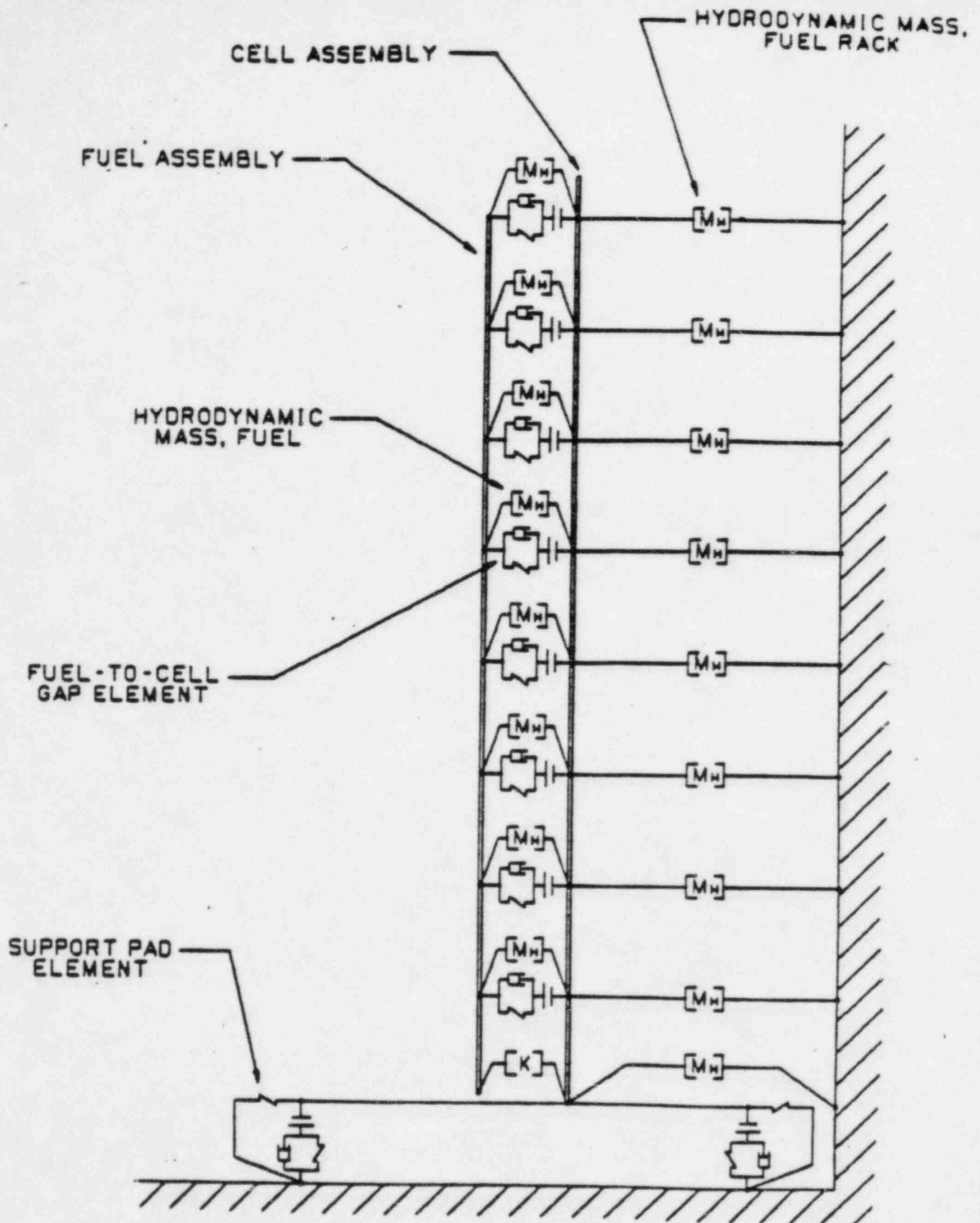


FIGURE 4-10

SECTION OF 3-D NON-LINEAR SEISMIC MODEL

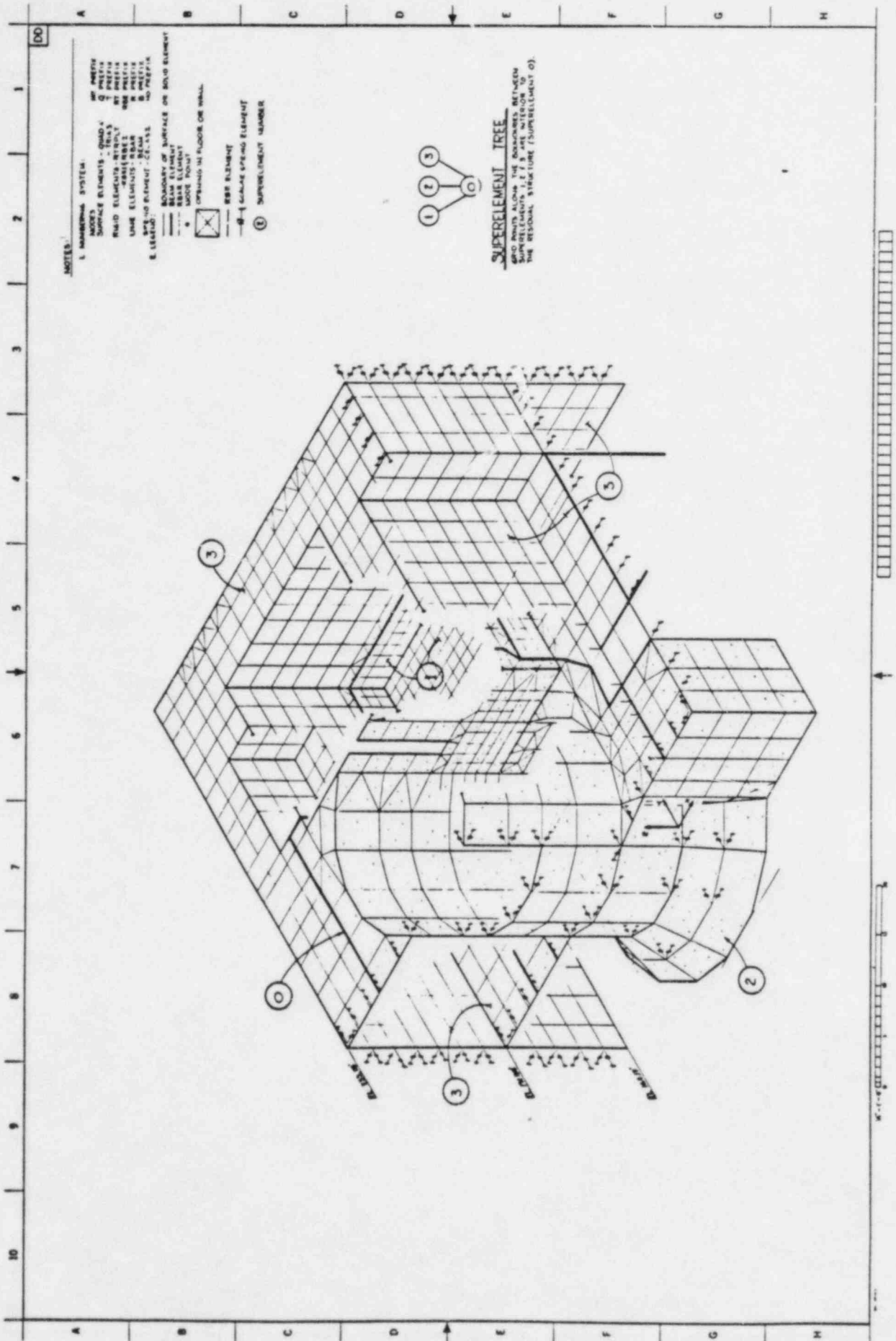


FIGURE 4-12
REACTOR BUILDING - FINITE ELEMENT MODEL

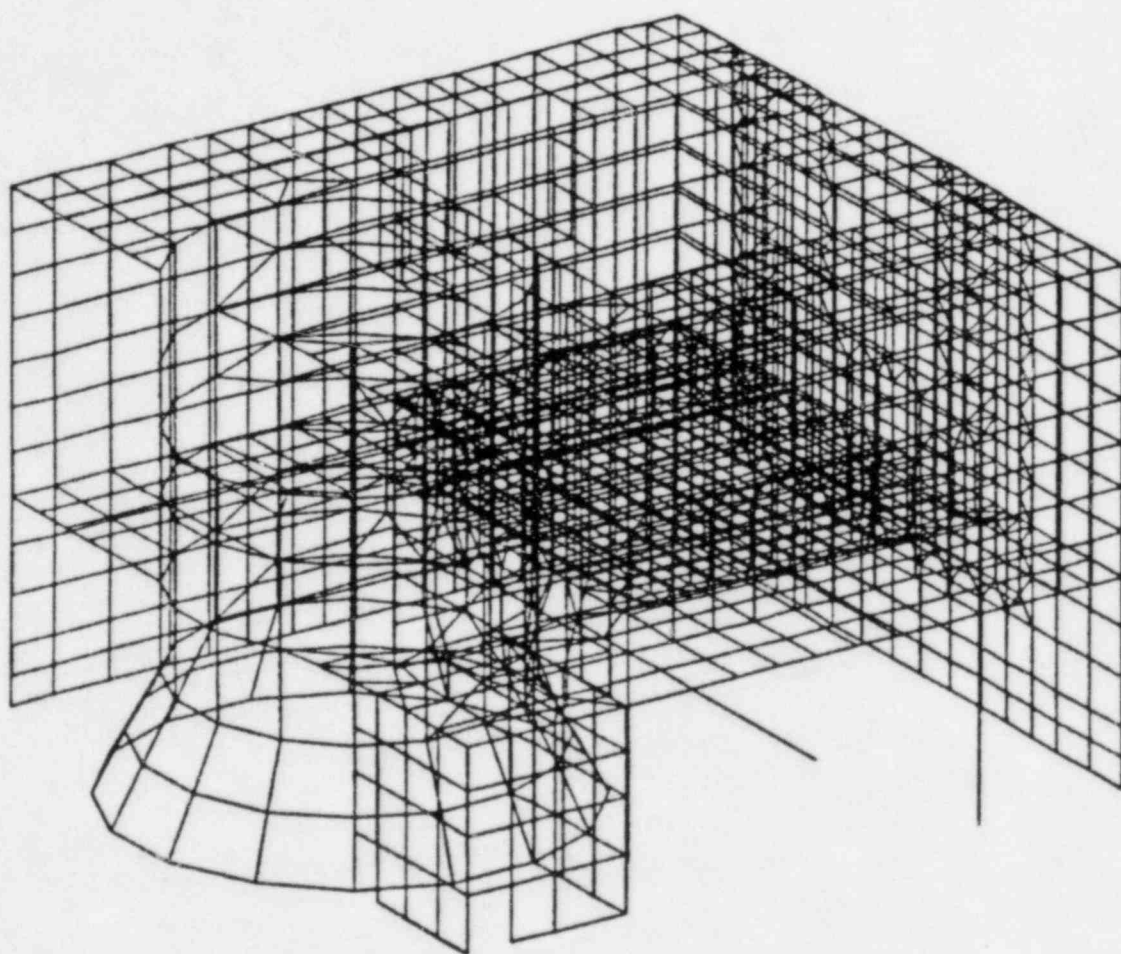


FIGURE 4-13
FEM UNDEFORMED SHAPE - ENTIRE STRUCTURE

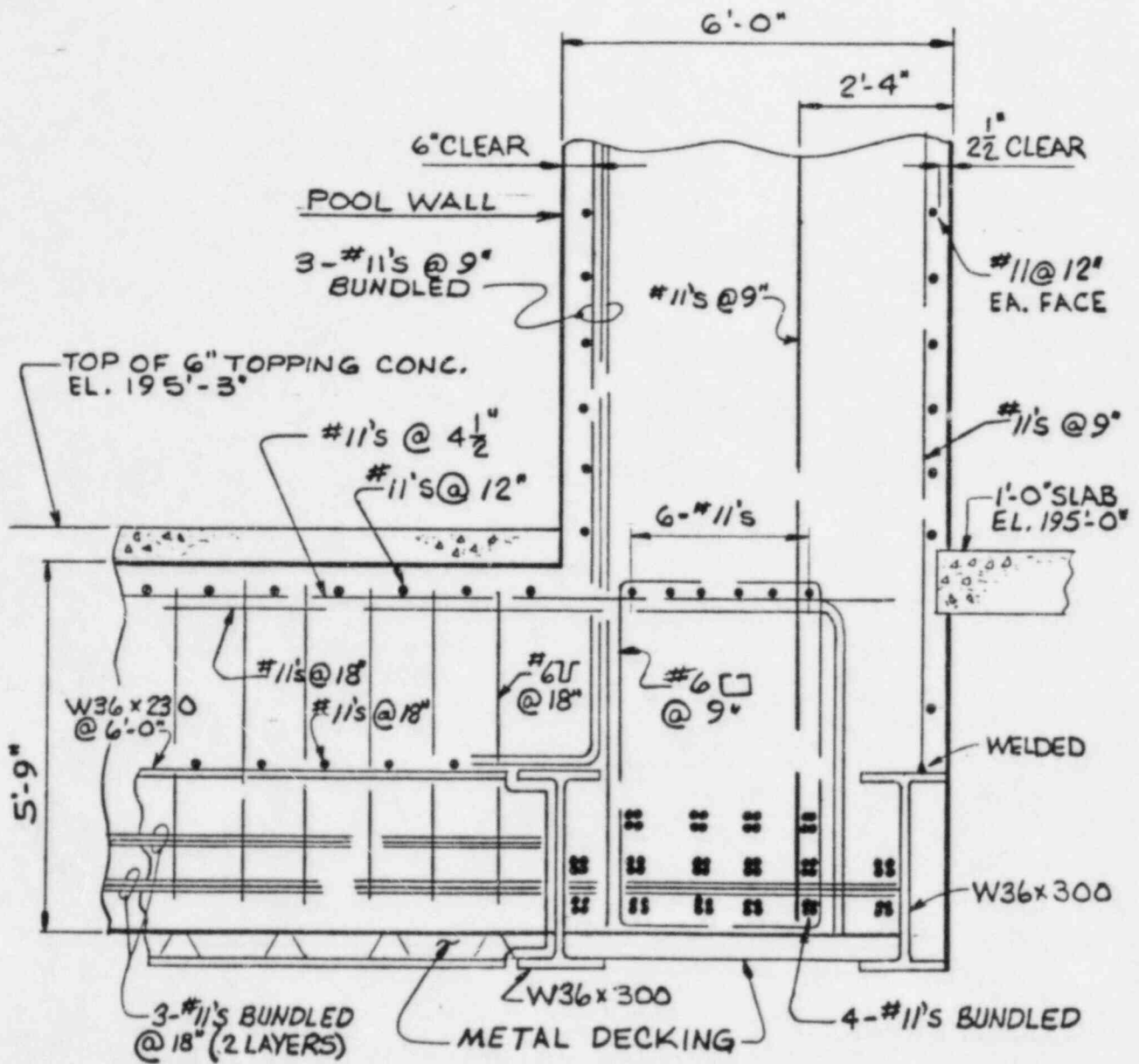


FIGURE 4-14
UNIT 2 TYPICAL SECTION LOOKING NORTH OR SOUTH

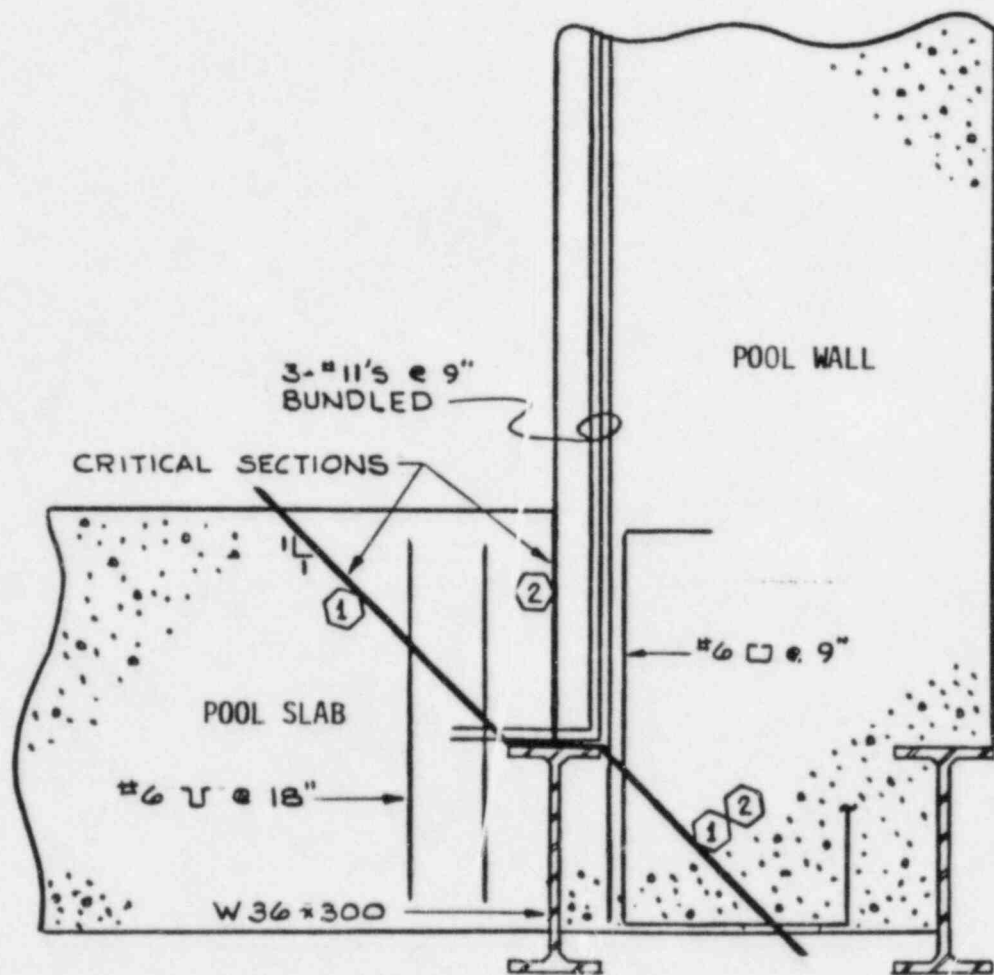


FIGURE 4-15
CRITICAL SECTION AT SLAB/WALL JOINT

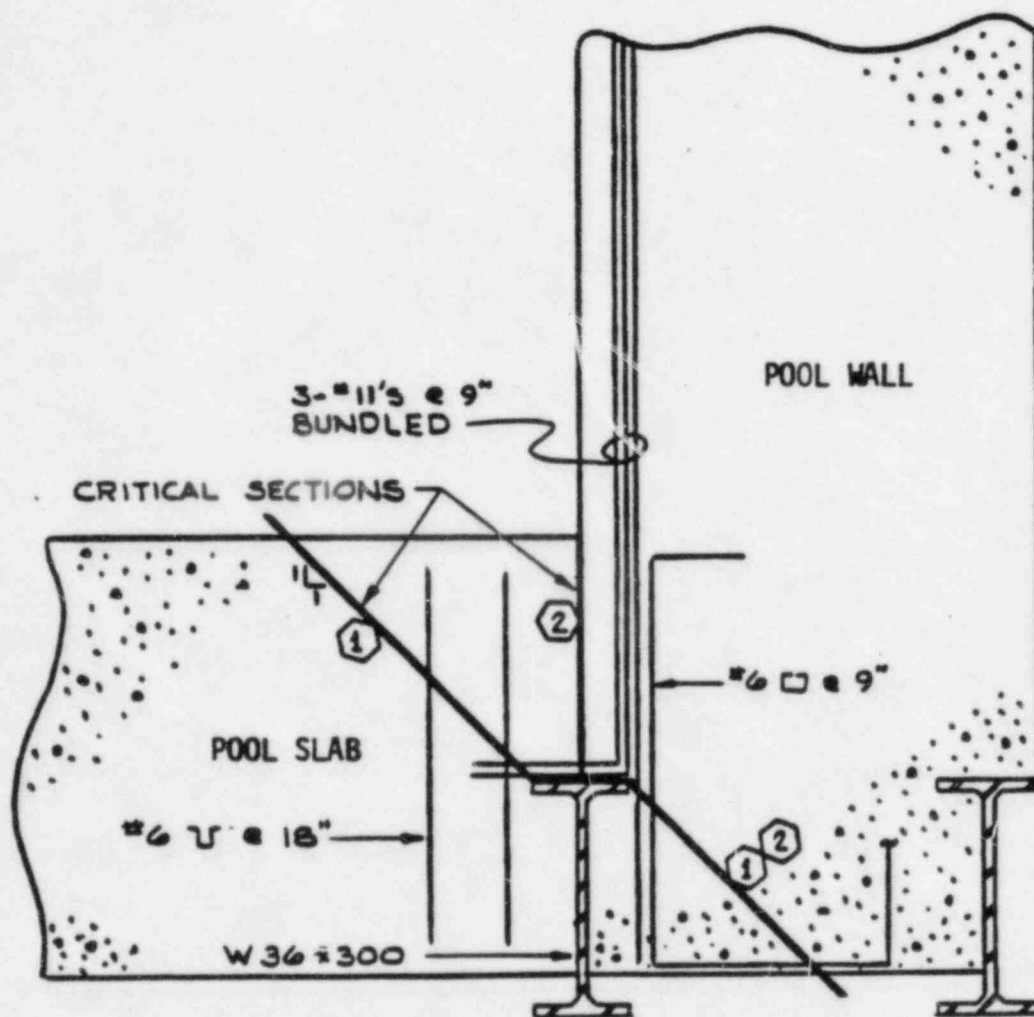


FIGURE 4-15
CRITICAL SECTION AT SLAB/WALL JOINT

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. PECO currently has no contractual arrangements with any fuel reprocessing facilities.
- b. At PBAPS, both the Unit 2 and Unit 3 spent fuel pools have been previously reracked to 2,608 cells each.

Table 5-1 includes proposed refueling schedules for both Unit 2 and Unit 3, and the expected number of fuel assemblies that will be transferred into the spent fuel pools at each refueling until full core discharge capability is lost.

- c. As of May 1985, the Unit 2 spent fuel pool contained 1464 spent fuel assemblies. The Unit 3 spent fuel pool contained 1,497 spent fuel assemblies and one new assembly.
- d. Adoption of this proposed spent fuel storage expansion would not necessarily extend the time period that spent fuel assemblies would be stored on site. Spent fuel could be sent off site for final disposition under existing legislation. The government facility is expected to become available in 1998. As matters now stand and until alternate storage facilities are available, spent fuel assemblies on site will remain there.
- e. Table 5-2 references the spent fuel storage capacity for both the PBAPS Unit 2 and 3 spent fuel pools after reracking. Based on the present PECO fuel management policy, the Unit 2 spent fuel pool will lose Full Core

Discharge Reserve (FCDR) in 1987. The Unit 3 pool will lose FCDR in 1988. In order to maintain full core discharge reserve capability during the new rack installation, the new racks require installation prior to the September 1986 refueling outage.

5.1.2 Estimated Costs

The costs associated with the Units 2 and 3 proposed spent fuel pool modification are estimated to be in the neighborhood of 9.1 million dollars. This figure includes items such as; 1) extensive engineering studies of spent fuel disposal alternatives, 2) design, engineering, manufacture, and installation of new spent fuel storage racks, 3) removal of offsite disposal of the existing spent fuel storage racks, 4) project management and licensing, and 5) allowance for funds used during construction. Estimated value of uncertainty and cost escalation are not included in this sum.

5.1.3 Consideration of Alternatives

- a. There are no operational commercial reprocessing facilities available for PECO's needs, nor are there expected to be any in the foreseeable future.
- b. At the present time, there are no existing available independent spent fuel storage facilities. There are no firm commitments by either commercial firms or government agencies to construct or operate an independent spent fuel storage facility. In addition, cost and/or schedule considerations make an independent spent fuel storage facility on site unacceptable to meet the spent fuel storage needs at PBAPS Units 2 and 3.
- c. At present PECO has no license to transship fuel between the Limerick and the PBAPS sites, nor are presently installed storage racks at Limerick licensed to store fuel generated at PBAPS. There are no plans at the present time for transshipment within the PECO system.

- d. Shutting down PBAPS would result in an economic hardship that would be imposed on PECO shareholders and customers. Moreover as indicated in NUREG-0575, "Final Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor fuel," the replacement of nuclear power by coal-generating capacity would cause excess mortality to rise from 0.59-1.7 to 15-120 per year for 0.8 GWY(e). Based on the above, shutting down PBAPS does not represent a viable alternative.

The subject of the comparative economics associated with various spent fuel options in the subject on Chapter 6 of NUREG-0575. Although the material presented is generic, it is of value in comparing the costs of the various operations. Of the options presented in Chapter 6 of NUREG-0575, high-density spent fuel storage at the site is the most economic option. Should the lack of spent fuel storage cause a shutdown, energy generated by PBAPS would need to be purchased from the Pennsylvania, New Jersey, Maryland Interconnection at a cost significantly higher than could be generated at PBAPS. In addition, PBAPS would need to be replaced with new generation facilities which would result in environmental effects and additional commitment of materials and resources.

Plant shut down would place a heavy financial burden on PECO and its customers which cannot be justified. The average daily cost to PECO for a plant shutdown is approximately \$500,000 per Unit.

5.1.4 Resources Committed

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

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

5.1.5 Thermal Impact on the Environment

Section 3.2 contains a description of the following considerations: the additional heat load and the anticipated maximum temperature of water in the SFP that would result from the proposed expansion, the additional heat load on component and/or plant cooling water systems. The proposed increase in storage capacity will result in an insignificant impact on the environment.

5.2 RADIOLOGICAL EVALUATION

The new spent fuel storage racks will allow storage of more spent fuel than is currently described in the FSAR. However, because the proposed racks will not affect such parameters as fuel burnup, the amount of fuel used by the reactors, or discharge schedules, there will be no appreciable change in the amount of radioactivity released from the station.

5.2.1 Solid Radwaste

Filter-demineralizer units are provided to minimize corrosion product buildup and control water clarity in the spent fuel pool. The filter-demineralizers, arranged so that one is designated for each reactor and a third is a common spare for use by either unit, are normally in continuous operation. The basis for regeneration of the filter-demineralizers is chemical exhaustion of the resin or high differential pressure across the filter unit. Presently, each filter-demineralizer is regenerated approximately every three weeks resulting in approximately 100 cubic feet of solid waste annually for each Unit.

Because most corrosion products are particulate matter which is introduced to the fuel pool during refueling and fuel transfer operations, the amount of solid waste generated by the cleanup system is not a function of spent fuel storage capacity. Therefore no significant increase in the quantity of solid radioactive wastes generated by the spent fuel pool cleanup system is anticipated. Replacement of the filter-demineralizer resins is accomplished remotely resulting in very low personnel doses.

5.2.2 Gaseous Radwaste

Gaseous radioactive releases from the station will not increase due to expansion of the spent fuel pool storage capacity. Nearly all releases from the pool/refueling floor area are the result of reactor pressure vessel head removal and maintenance operations. Within a week or two of off-loading of fuel, when the fuel has cooled, the gaseous releases from the pool area diminish to insignificant levels. Therefore there will be no significant increase in radioactive gaseous releases from long term storage of spent fuel. Historical data on gaseous release of Kr-85 from the plant vent are as follows:

Jan. to June 1982 = 0.72 Ci	July to Dec. 1982 = 0
Jan. to June 1983 = 0	July to Dec. 1983 = 0
Jan. to June 1984 = 0	July to Dec. 1984 = 26.7 Ci

5.2.3 Persc xposure

- a. Operating experience shows a dose rate of 2 mrem/hour at the edge and a dose rate of 5 to 10 mrem/hour above the center of the spent fuel pools exists regardless of the quantity of fuel stored. This is not expected to change with the proposed reracking because radiation levels above the pool are due primarily to radioactivity in the water. Experience shows that the radioactivity in the water varies during refueling and returns to a level of equilibrium regardless of the amount of fuel stored. Stored spent fuel is shielded by the water above the fuel such that dose rates at the top of the pool from this source are negligible.
- b. There have been negligible concentrations of airborne radioactivity from the spent fuel pools. The proposed reracking is not expected to significantly increase this activity.
- c. Operating plant experience with dense fuel storage has shown no noticeable increases in airborne radioactivity above the spent fuel pool or at the site boundary. No significant increases are expected from more

dense storage. Historical data of airborne radioactivity over the pool shows levels less than 10^{-10} microcuries/cc.

- d. As stated in Section 5.2.1 and based on operating experience with dense fuel storage racks, there is no significant increase in the radwaste generated by the spent fuel pool cleanup system. This is because operating experience has shown that with high density storage racks, there is no significant increase in the radioactivity levels in the spent fuel pool water. Operating experience with high density storage racks has shown no significant increase in the annual man-rem due to the increased fuel storage, including the changing of spent fuel pool cooling system resins and filters. Changing the racks to an even higher density will not change these conclusions.
- e. Most of the crud associated with spent fuel storage is released soon after fuel is removed from the reactor. Once fuel is placed into the pool storage positions, additional crud contribution is minimal.

Crud shaken loose during fuel handling and fission products released through defects in the fuel account for most of the radioactivity in the pool water. As the decay heat in the defective fuel decreases, the release of fission products is also reduced. Therefore, most releases from failed fuel occur soon after refueling. This radioactivity will either decay or be removed by the spent fuel pool cleanup system within a year such that there will be no increase in long term activity levels.

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

- f. During normal operation, the radiation zone designation of areas around the sides and underneath the pools will not change due to reracking. The depth of the water above the fuel is sufficient so there will be no measurable increase in dose rates above the pool due to radiation emitted

directly from the fuel. Sections 5.2.3.d and 5.2.3.e indicate that operating experience with high density racks has shown no significant increase in radioactivity levels in the water or dose rates above the pool. As stated in Sections 5.2.1 and 5.2.3.d operating experience with high density racks has shown no increase in the processing of solid radioactive waste or man-rem associated with it.

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

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

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

5.2.4 Radiation Protection During Re-Rack Activities

5.2.4.1 General Description of Protective Measures

The radiation protection aspects of the spent fuel pool modification will be controlled in accordance with existing health physics control procedure. Gamma radiation levels in the pool area are monitored by the station Area Radiation Monitoring System, which has a high level alarm feature. Additionally, periodic radiation and contamination surveys are conducted in work areas as necessary. Where there is a potential for significant airborne radionuclide concentrations, continuous air samplers can be used in addition to periodic grab sampling. Personnel working in radiologically controlled areas wear protective clothing and respiratory protective equipment, depending

on work conditions, as required by the applicable Radiation Work Permit (RWP). Personnel monitoring equipment is assigned to and worn by all personnel in the work area. At a minimum, this equipment consists of a thermoluminescent dosimeter (TLD) and self-reading pocket dosimeter. Additional personnel monitoring equipment, such as extremity badges, are utilized as required.

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

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

5.2.4.2 Anticipated Exposures During Re-Racking

Prior to re-racking, a detailed re-rack plan will be developed and approved by the PBAPS Plant Operation Review Committee encompassing, as a minimum, the following:

- a. Assurance that personnel exposure will be maintained as low as is reasonably achievable (ALARA), identifying the step-by-step operations, including the number of personnel involved in each step, the anticipated dose rate, the time involved and the estimated man-rem exposure.
- b. Assurance that spent fuel stored in the racks is not within the area of influence of a potential rack-drop accident during removal of existing rack modules or installation of new ones.

- c. Assurance that operations or potential accidents (e.g., rack-drop) will not adversely affect any plant equipment needed to mitigate consequences of a reactor accident or necessary to maintain safe shutdown.

Similar operations have been successfully accomplished by PECO and a number of utilities in the past, and a safe and acceptable re-racking plan will be developed. The Generic Environmental Impact Statement (NURGE-0575, August 1979) also suggests that re-racking may be safely accomplished subject to evaluation of specific rack designs and factors enumerated above.

5.2.5 Rack Disposal

The spent fuel storage rack modules that will be removed from the spent fuel pool weigh between 8,700 and 16,350 pounds each. When the racks are removed from the pool, they will be rinsed with demineralized water or spent fuel pool water to remove loose contamination. Depending on the levels of loose contamination, this rinse will be with either a low or high pressure spray. Personnel involved in the operation, and others in the immediate area, will wear appropriate protective clothing and respiratory protective equipment if needed.

The rinsing operation is expected to remove significant quantities of loose contamination from the racks while causing relatively low exposure to decontamination personnel. This procedure minimizes subsequent personnel exposures due to handling and packaging of rack sections for shipment and disposal.

Disposal options for the old racks include burial and decontamination. Depending on the effectiveness of decontamination, the rack sections will eventually either be sold as scrap, or if decontamination is not possible, buried at a low-level burial site.

5.3 ACCIDENT EVALUATION

5.3.1 Spent Fuel Handling Accidents

5.3.1.1 Fuel Assembly Drop Analysis

The proposed spent fuel pool modifications will not increase the radiological consequences of fuel handling accidents previously evaluated in the PBAPS Updated FSAR.

5.3.1.2 Cask Drop Analysis

The PBAPS Updated FSAR evaluates the potential for a cask drop over the spent fuel pool. The rerack program will not alter the cask handling procedures or the results of the cask drop evaluation. The cask handling crane meets the design and operational requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".

Existing technical specifications regarding allowable loads carried over spent fuel, and established heavy load paths will be followed during the removal and installation of storage racks.

5.3.2 Conclusions

The proposed spent fuel pool modifications will not increase the radiological consequences of a heavy load or cask drop accident previously evaluated.

Since there will be a negligible change in radiological conditions due to the increased storage capacity of the spent fuel pool, no change is anticipated in the radiation protection program. In addition, the environmental consequences of a postulated fuel handling accident in the spent fuel pool, described in Updated FSAR Section 14.0, remain unchanged. Therefore, there will be no change or impact to any previous determinations of the Final Environmental Statement. Based on the foregoing, the proposed modification will not significantly affect the quality of the human environment; therefore, under 10 CFR 51.5c, issuance of a negative declaration is appropriate.

TABLE 5-1
EXISTING SPENT FUEL STORAGE CAPACITY

PBAPS Unit 2

<u>Refuel Date</u>	<u>Number of Assm. Discharged</u>	<u>Cumulative Total Discharged</u>	<u>Storage Space Available</u>
To Date		1,464	1,144
9/86	276	1,740	868 *
12/87	276	2,016	592 **

PBAPS Unit 3

<u>Refuel Date</u>	<u>Number of Assm. Discharged</u>	<u>Cumulative Total Discharged</u>	<u>Storage Space Available</u>
To Date		1,497	1,111
3/87	276	1,773	835 *
9/88	276	2,049	559 **

* Storage Limit with Full Core Discharge Reserve (FCDR)

** FCDR Lost

TABLE 5-2
SPENT FUEL STORAGE CAPACITY AFTER RERACK

PBAPS Unit 2

<u>Refuel Date</u>	<u>Number of Assm. Discharged</u>	<u>Cumulative Total Discharged</u>	<u>Storage Space Available</u>
To Date		1,464	2,355
9/86	276	1,740	2,079
12/87	276	2,016	1,803
6/89	276	2,292	1,527
12/90	276	2,568	1,251
6/92	276	2,844	975 *
12/93	276	3,120	699 **

PBAPS Unit 3

<u>Refuel Date</u>	<u>Number of Assm. Discharged</u>	<u>Cumulative Total Discharged</u>	<u>Storage Space Available</u>
To Date		1,497	2,322
3/87	276	1,773	2,046
9/88	276	2,049	1,770
3/90	276	2,325	1,494
9/91	276	2,601	1,218
3/93	276	2,877	942 *
9/94	276	3,153	666 **

* Storage Limit with Full Core Discharge Reserve (FCDR)

** FCDR Lost