ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE

UNITS 1 & 2

SPENT FUEL POOL RERACK

LICENSING SUBMITTAL

November 1, 1982

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1.0 INTRODUCTION

This report is submitted in support of Arkansas Power and Light Company's application to amend the Arkansas Nuclear One operating licenses for reracking modifications to the Arkansas Nuclear One Unit-1 (ANO-1) and Unit-2 (ANO-2) spent fuel storage pools. This report is in accordance with the guidelines of the Nuclear Regulatory Commission's "Standard Review and Acceptance Plan for Spent Fuel Storage and Handling Applications."

These proposed modifications will increase spent fuel storage capacities by replacing the present spent fuel storage racks with storage racks as described in Section 2.0 of this report. Reracking the ANO spent fuel pools will increase the ANO-1 pool storage capacity from 589 spaces to 968 spaces and the ANO-2 pool storage capacity from 485 spaces to 988 spaces.

Arkansas Power and Light Company is responsible for the modification of the spent fuel storage pools. Westinghouse Electric Corp. is retained to design, analyze, fabricate, and provide technical assistance during the installation of the spent fuel racks. Structural Dynamics Technology, Inc. is retained to perform the spent fuel pool structural analysis and provide technical assistance.

2.0 GENERAL DESCRIPTION

The spent fuel pool and its associated systems are described in the ANO-1 FSAR Section 9.6 and in Section 9.1.3 of the ANO-2 FSAR. Figures 9-11 and 9.1-1 are reproductions of FSAR figures which show the general arrangement of the ANO-1 & 2 pools and the associated fuel handling equipment. This arrangement is not changed as a result of this modification. The spent fuel storage pool rack arrangement for ANO-1 is shown in Figure 2-1 and the arrangement for ANO-2 is shown in Figure 2-2.

Fuel storage is divided into two regions within each pool. Region 1 (approximately 200 assemblies), is of high density fuel assembly spacing obtained by utilizing a neutron absorbing material and is reserved for core off loading. Region 2 is also of a high density fuel assembly spacing providing normal storage for spent fuel assemblies. Region 1 is designed to accommodate non-irradiated fully enriched fuel. Region 2 is designed to accomodate irradiated fuel that has sustained approximately 80 percent of the design burnup. Placement of fuel in Region 2 is determined by burnup calculations and controlled administratively by Arkansas Power & Light. Fuel which does not meet this criterion may be placed in Region 2 in a checker board fashion. In these cases, vacant spaces surrounding the assembly being inspected will be controlled administratively to prevent inadvertent assembly insertion. No physical barrier is necessary between the two regions. The racks meet the requirements of the NRC Position for Review & Acceptance of Spent Fuel Storage and Handling Applications, dated April 14, 1978, and modified January 18, 1979, with the exception that, for Region 2 storage, credit is taken for fuel burnup based on the proposed Revision 2 of USNRC Regulatory Guide 1.13.

2.1 REGION 1 DESIGN

The Region 1 storage racks are composed of individual storage cells made of stainless steel. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. The cells within a module are interconnected by grid assemblies to form an integral structure as shown in Figure 2-3. Each rack module is provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The fuel rack assembly consists of three major sections which are the leveling pad assembly, the lower and upper grid assembly, and the cell assembly. Figure 2-4 illustrates these sections.

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

The lower grid attaches the cell assembly to the base plate. The lower grid consists of box-beam members and the base plate. The bottom of the cell assembly is welded to the lower grid. The upper grid consists of the box-beam members. The upper part of the cell assembly is welded to the upper grid. The upper and lower grid assemblies maintain the center-line to center-line spacing between the cells and provide the structural connections between the cells to form a fuel rack assembly.

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

The wrapper is attached to the outside of the cell by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of the Boraflex to the pool environment. Depending on the criticality requirements, some cells have a Boraflex wrapper on all four sides, some on three sides, and some on two sides.

2.2 REGION 2 DESIGN

The Region 2 storage racks consist of stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure. Each cell has attached to its outer wall a stainless steel wrapper plate creating a pocket opened at the top and bottom. This is referred to as a "spacer pocket" design and is shown in Figure 2-5. The spacer pockets are designed to accept poison inserts if future need arises.

This design is also provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The fuel rack assembly consists of two major sections which are the base support assembly and the cell assembly. Figure 2-6 illustrates these sections.

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

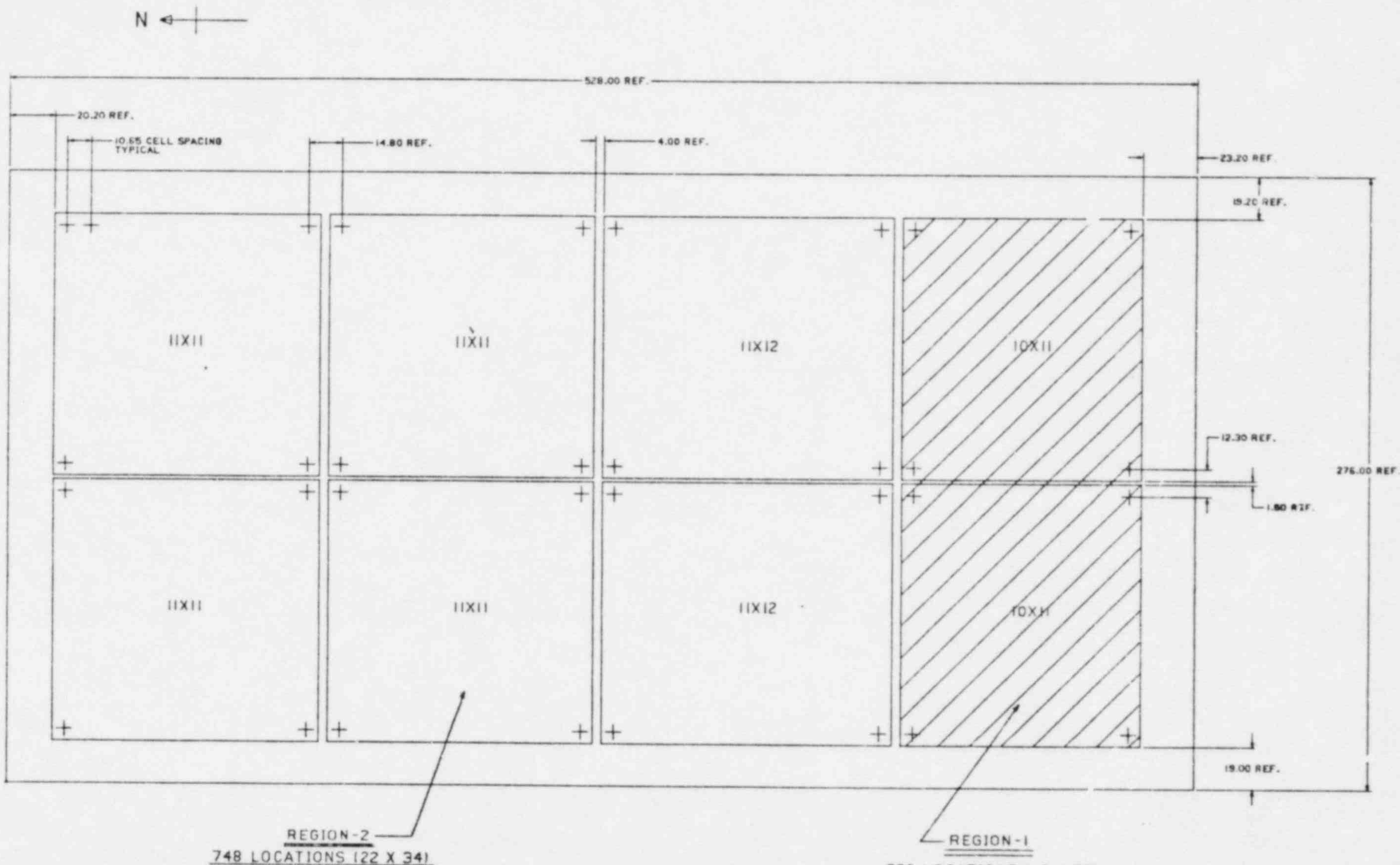
The components of the cell assembly are the cell enclosure and the wrapper plates.

The wrapper plates are attached to the outside of the cell by spot welding along the entire length of the wrapper. The wrapper forms the pocket and establishes the size of the noncell locations.

Rack module data for both units are described in Table 2-1.

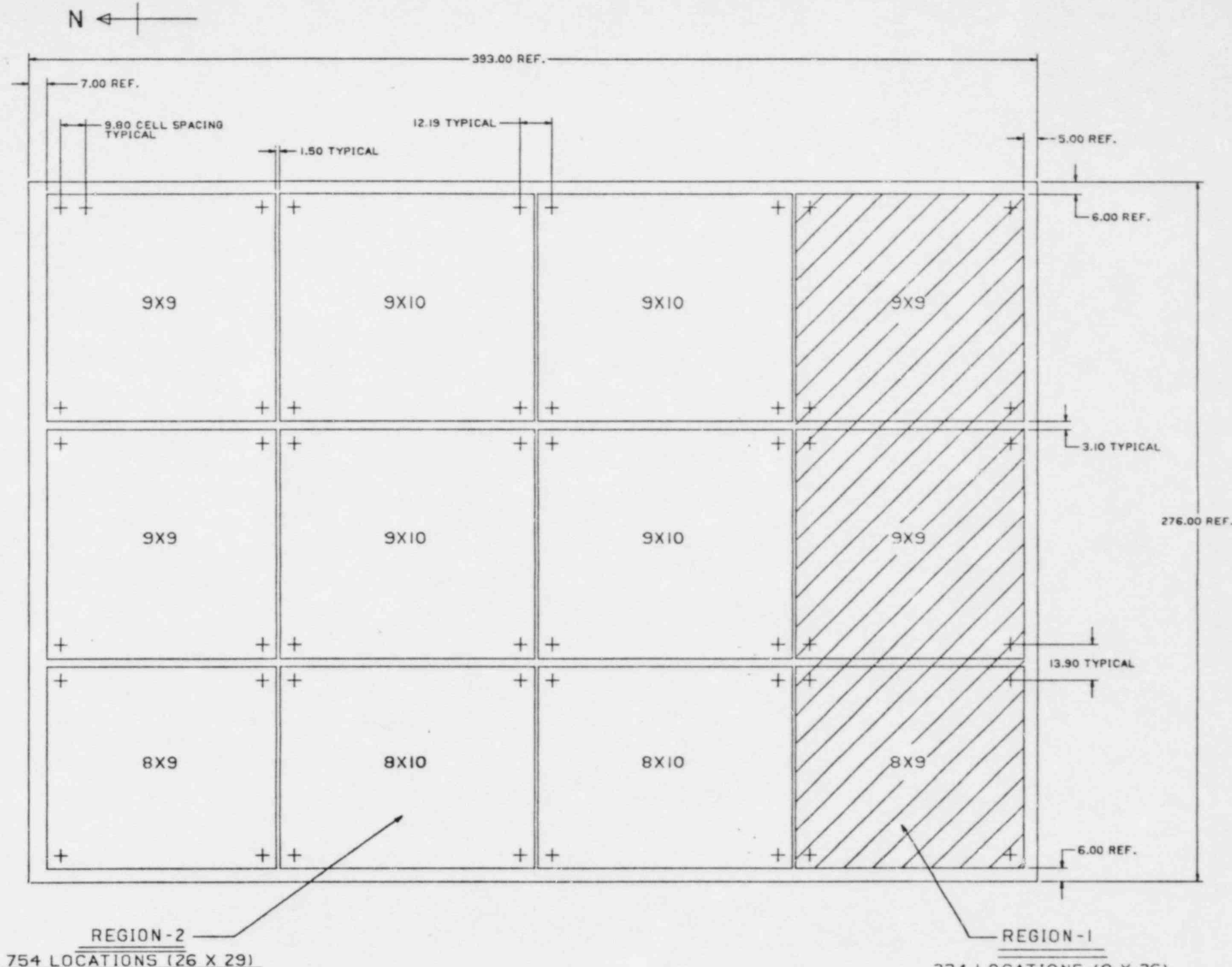
TABLE 2-1

	Unit 1		Unit 2	
	Region 1	Region 2	Region 1	Region 2
Number of Storage Locations	220	748	234	754
Number of Rack Arrays	2 (10x11)	2 (11x12) 4 (11x11)	2 (9x9) 1 (8x9)	4 (9x10) 2 (8x10) 2 (9x9) 1 (8x9)
Center-to-Center Spacing (Inches)	10.65	10.65	9.8	9.8
Cell I.D. (Inches)	8.97	8.97	8.58	8.58
Type of Fuel	B&W 15x15	B&W 15x15	C.E. 16x16	C.E. 16x16
Rack Assembly Dimensions (Inches)	(10x11) 106.7x117.4x169.0	(11x12) 118.2x128.8x169.0 (11x11) 118.2x118.2x169.0	(9x9) 89.2x89.2x188.875 (8x9) 79.4x89.2x188.875	(9x10) 89.2x99.0x188.875 (8x10) 79.4x99.0x188.875 (9x9) 89.2x89.2x188.875 (8x9) 79.4x89.2x188.875
Dry Weights (lbs)	27,500 (10x11)	19,500 (11x12) 18,000 (11x11)	20,300 (9x9) 18,000 (8x9)	16,000 (9x10) 14,200 (8x10) 14,400 (9x9) 13,000 (8x9)



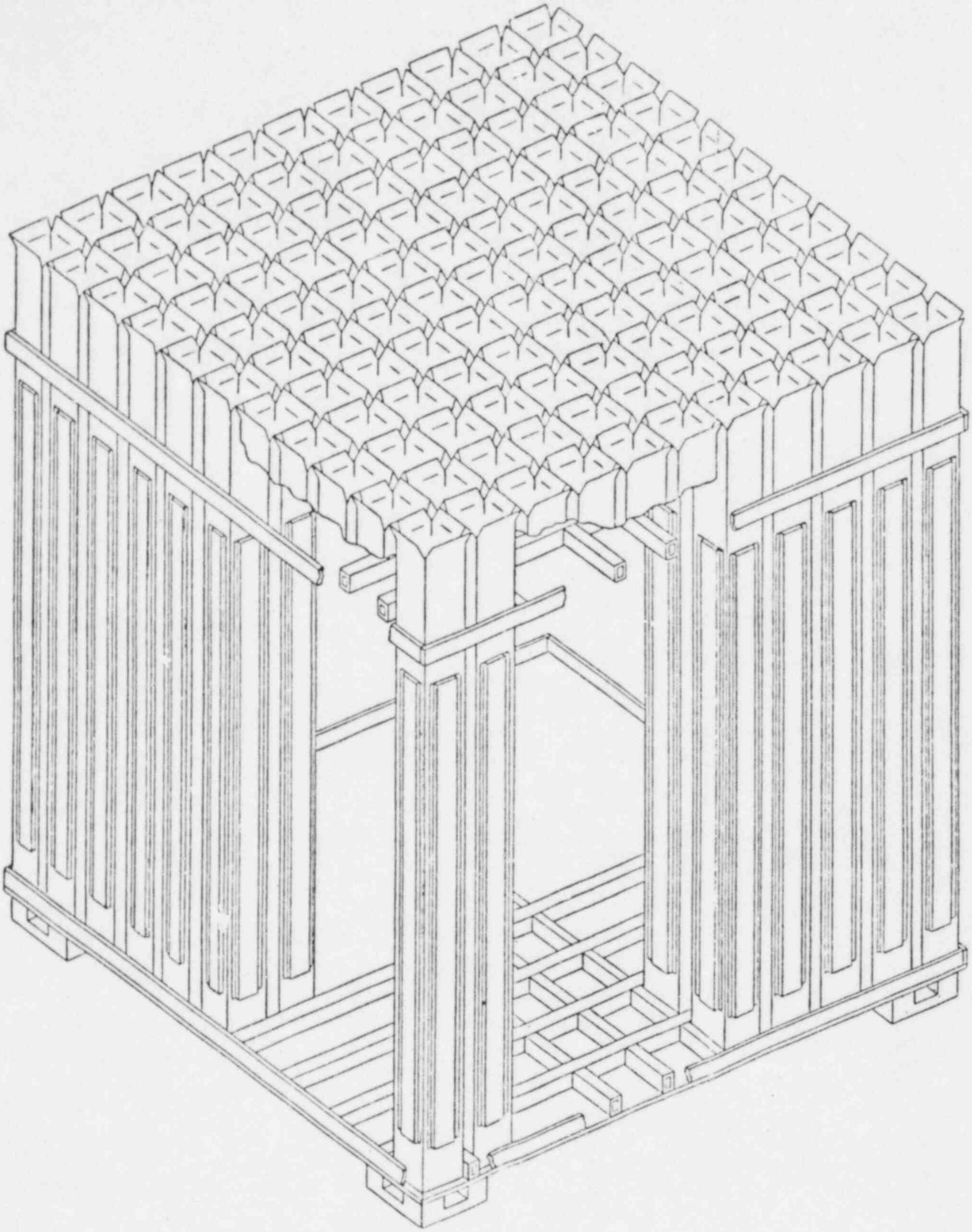
SPENT FUEL POOL ARRANGEMENT UNIT #1

FIGURE 2-1



SPENT FUEL POOL ARRANGEMENT UNIT #2

FIGURE 2-2



Region 1

Full Storage Rack Assembly

Figure 2-3

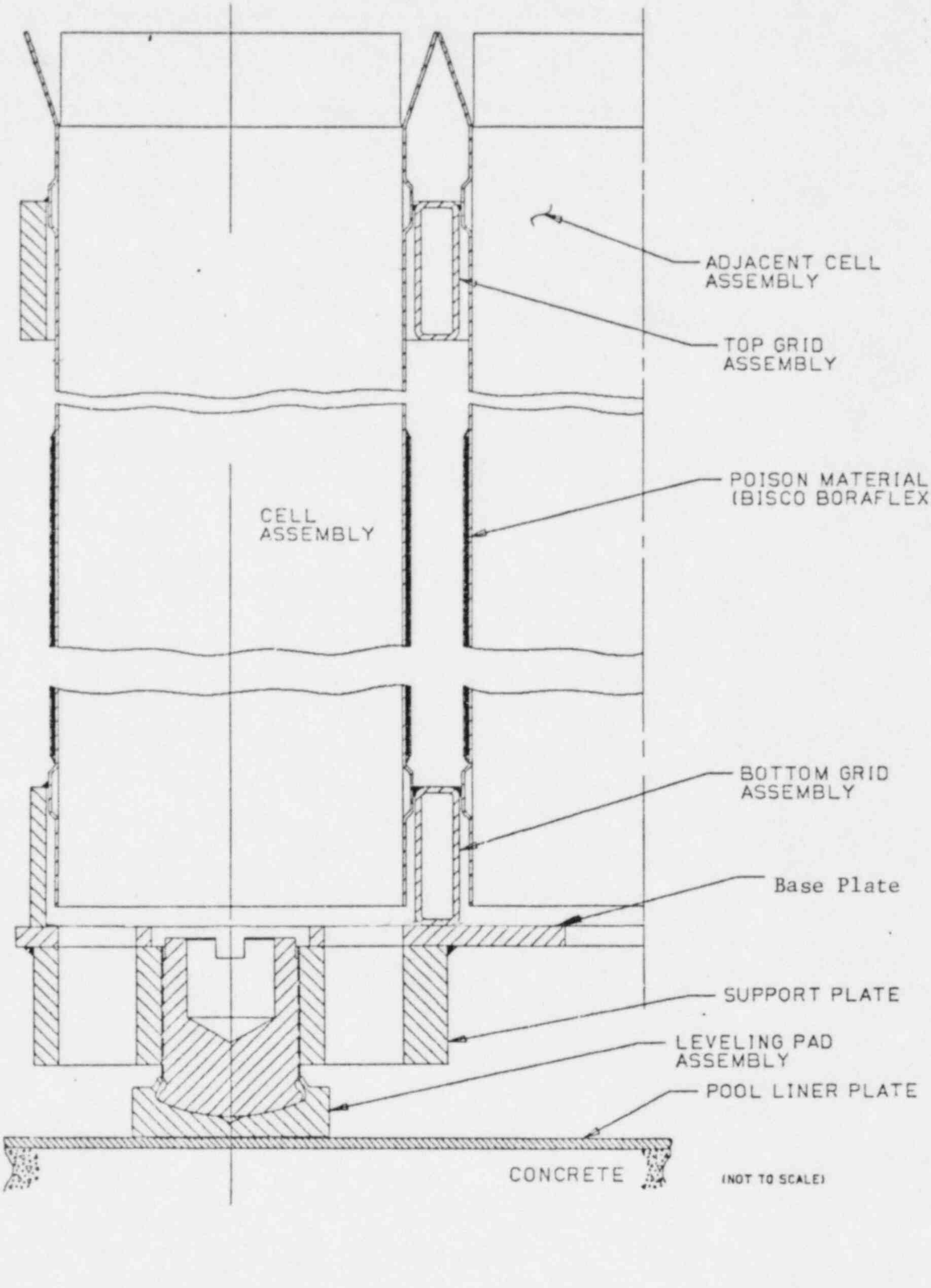
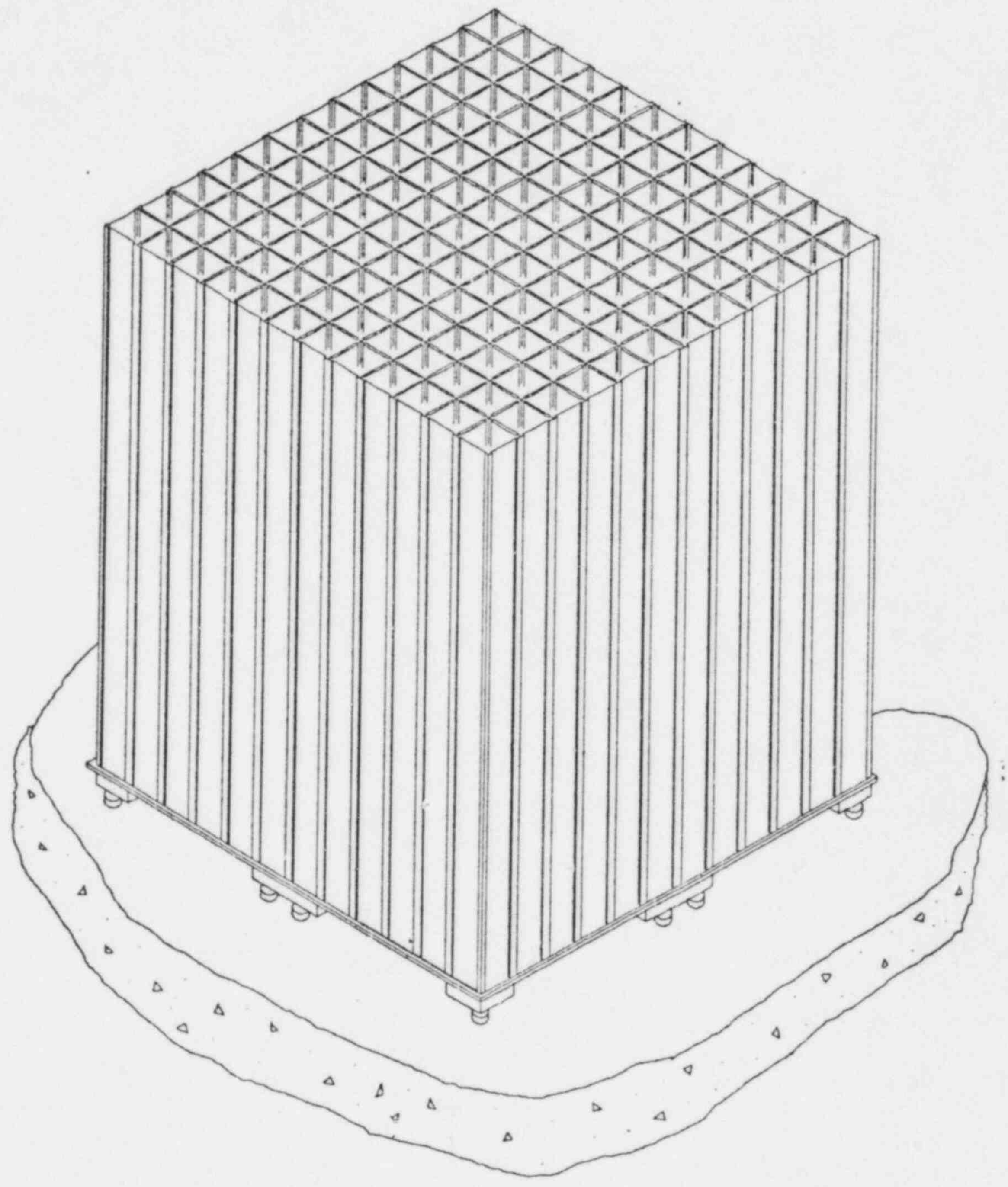

ADJACENT CELL ASSEMBLY
TOP GRID ASSEMBLY
POISON MATERIAL (BISCO BORAFLEX)
CELL ASSEMBLY
BOTTOM GRID ASSEMBLY
Base Plate
SUPPORT PLATE
LEVELING PAD ASSEMBLY
POOL LINER PLATE
CONCRETE
(NOT TO SCALE)

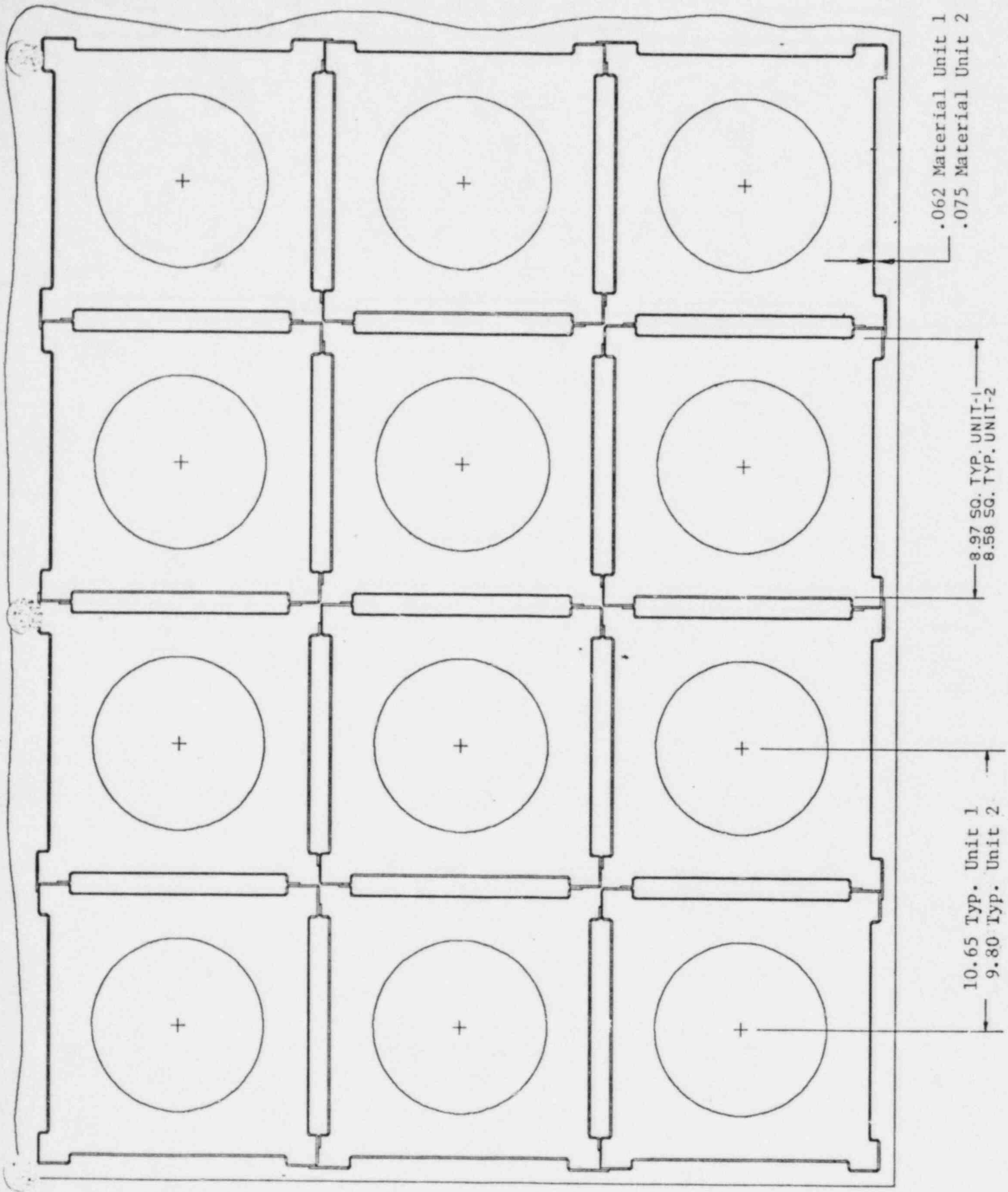
Figure 2-4



Region 2

"Spacer Pocket" Design

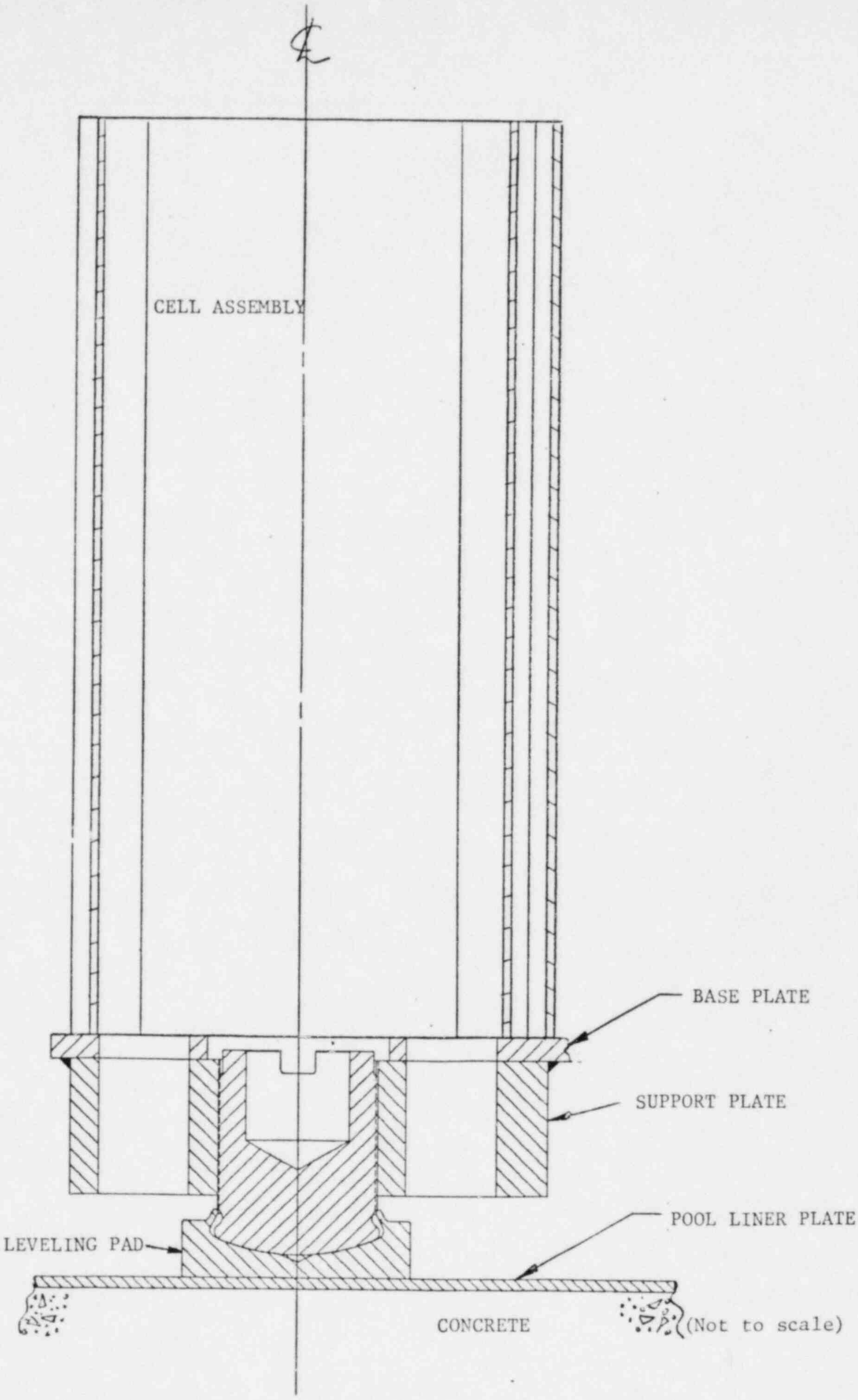
Figure 2-5



"Spacer Pocket" Design

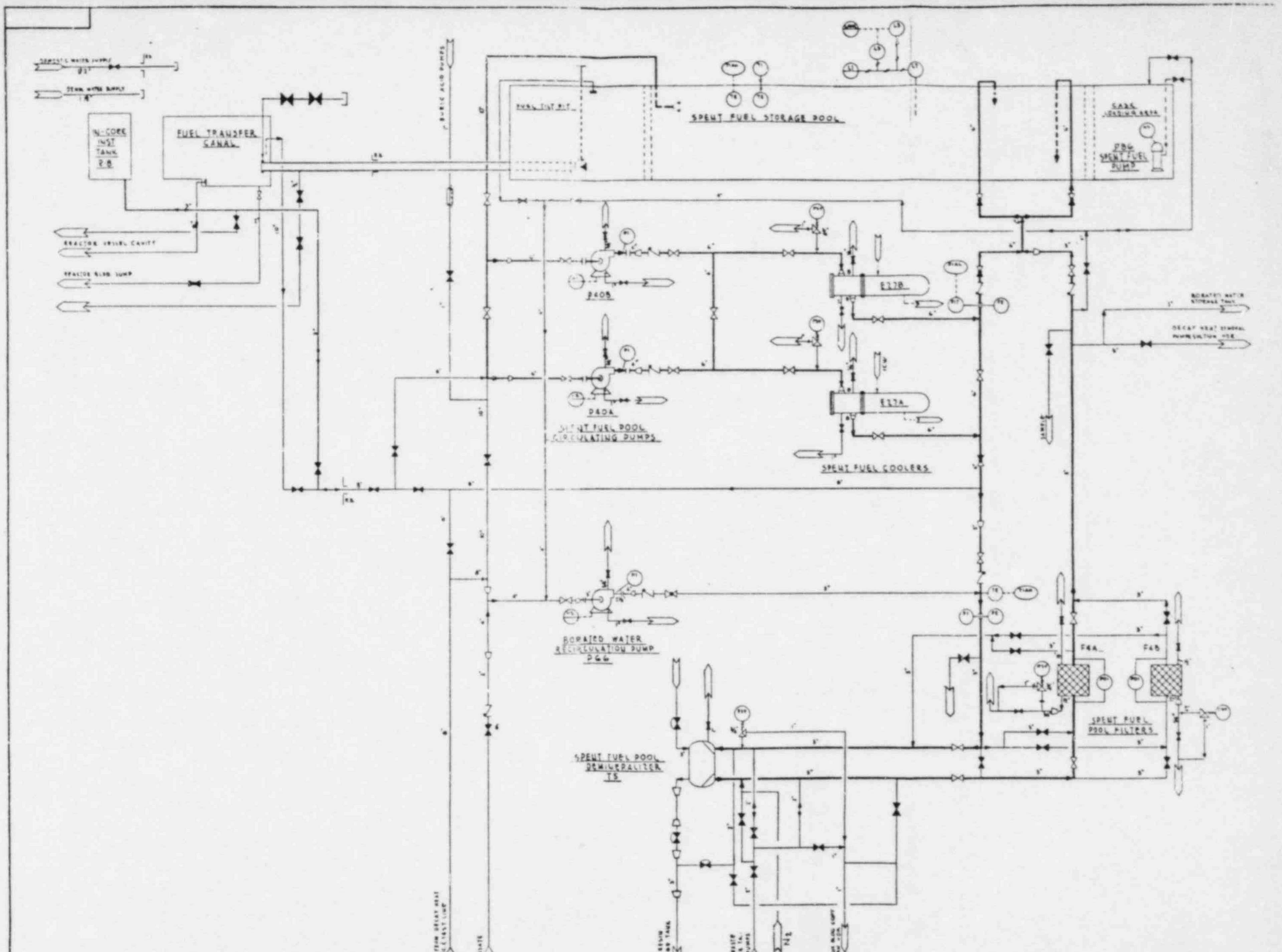
Plan View

(Figure 2-5 Cont.)



CELL ASSEMBLY
BASE PLATE
SUPPORT PLATE
POOL LINER PLATE
LEVELING PAD
CONCRETE
(Not to scale)

Figure 2-6

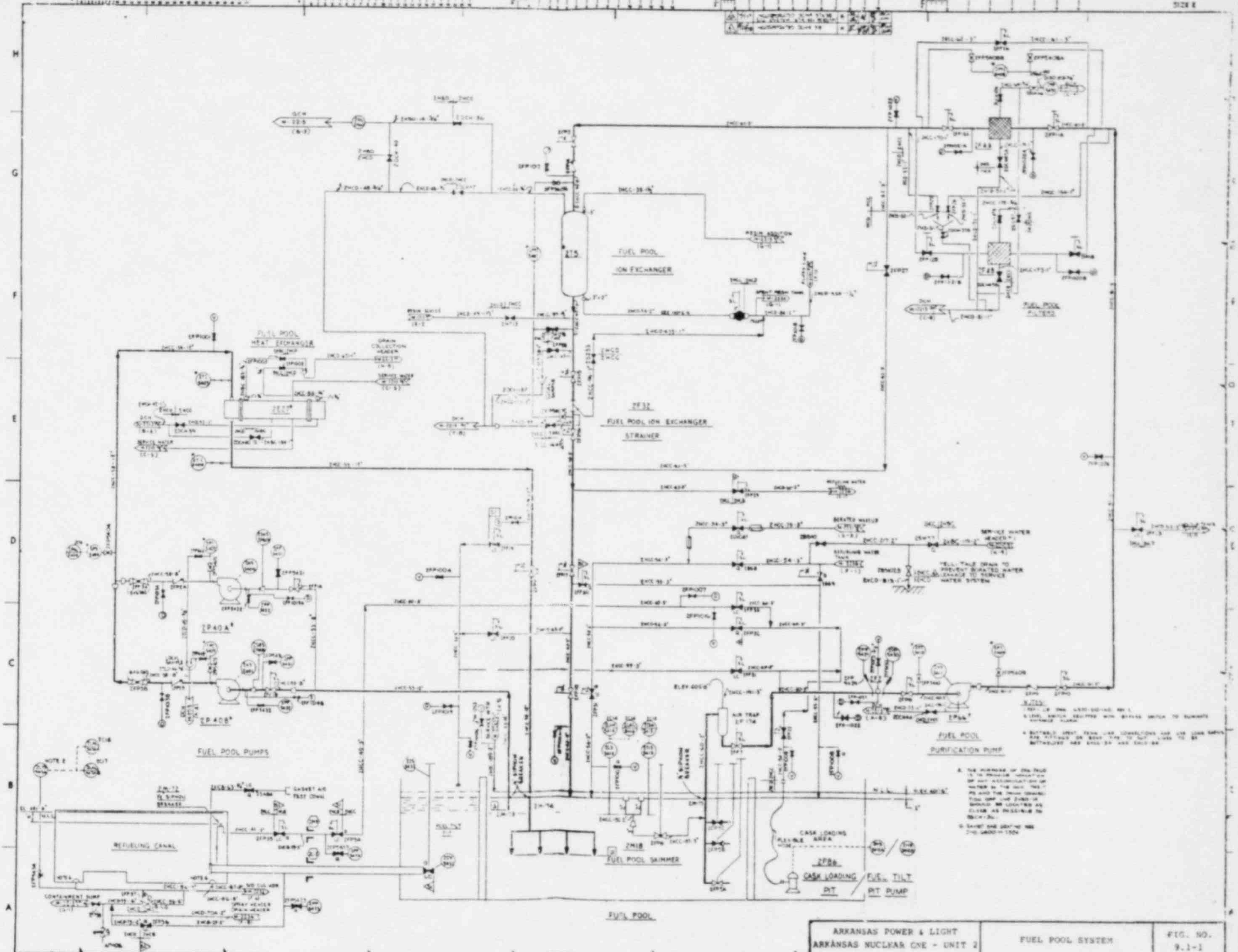


AMENDMENT No. 44

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 1	SPENT FUEL COOLING SYSTEM	FIGURE NO. 9-11

DLS ARE EXPLAINED ON
RES 1.7-1 AND 1.7-2.

S REFERENCES M-XXXX TO
FIGURE NUMBERS ARE
S IN TABLE 1.7-2.



3.0 DESIGN BASES

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

3.1 DESIGN LOADS

The dry weights for each type of specific module are listed in Table 2-1.

3.2 SPECIFIED LOADS AND DEFINITIONS

The following are load combinations specified for the modified racks:

Elastic Analysis	Acceptance Limits
(1) D + L	Normal Limits of NF 3231.1a
(2) D + L + E	Normal Limits of NF 3231.1a
(3) D + L $\pm$ T_o	Lesser of 2 S_y or S_u Stress Range
(4) D + L + T_o + E	Lesser of 2 S_y or S_u Stress Range
(5) D + L + T_a + E	Lesser of 2 S_y or S_u Stress Range
(6) D + L + T_a + E'	Faulted Condition Limits of NF 3231.1c

Definitions:

D - Dead loads or their related internal moments and forces including any permanent equipment and hydrostatic loads.

L - Live loads or their related internal moments and forces including any movable equipment loads.

E - Operational Basis Earthquake Loads

E' - Design Basis Earthquake Loads

T_o - Operating Thermal Loads

T_a - Accident Thermal Loads

3.3 APPLICABLE CODES, STANDARDS, AND REGULATIONS

The modified spent fuel storage racks are designed by Westinghouse Electric Corp. with the applicable provisions of the following codes, standards, and regulations:

"NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979, with the exception that credit is taken for burnup for all storage locations except for those reserved for full core discharge (in Region 1 of both Units).

3.3.1 NRC Regulatory Guides

R.G. 1.13 Proposed Revision 2 - Spent Fuel Storage Facility Design Basis

R.G. 1.29 Seismic Design Classifications

R.G. 1.44 Control of the Use of Sensitized Stainless Steel

R.G. 1.60 Design Response Spe. .ra for Seismic Design of Nuclear Power Plants

R.G. 1.61 Damping Values for Seismic Design of Nuclear Power Plants

R.G. 1.92 Combining Nodal Responses and Spatial Components in Seismic Response Analysis

R.G. 1.124 Service Limits and Loading Combinations for Class I Linear-Type Component Supports

3.3.2 NRC Standard Review Plans

SRP 3.7 Seismic Design

SRP 3.8.4 Other Category I Structures

SRP 9.1.2 Spent Fuel Storage

SRP 9.1.3 Spent Fuel Pool Cooling and Cleanup System

3.3.3 Industry Codes and Standards

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF.

American National Standards Institute, N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

American National Standards Institute, N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

4.0 NUCLEAR CONSIDERATIONS

4.1 INTRODUCTION

The spacer pocket spent fuel rack design is a high density rack design in which explicit credit is taken for the reduction in reactivity due to fuel burnup. The spent fuel rack design described herein employs two separate and different arrays which will be considered as two separate spent fuel racks. The smaller array, referred to as Region 1, is designed on the basis of the currently accepted NRC guidance on spent fuel rack design.[1] The larger array, Region 2, is designed to take into consideration the changes in fuel and fission product inventory resulting from depletion in the reactor core.

4.2 CRITICALITY ANALYSIS FOR REGION 1

4.2.1 Neutron Multiplication Factor

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

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI N210-1976 and in Reference 1.

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

4.2.2 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 following assembly parameters were modeled:

	Unit 1	Unit 2
Number of Fuel Rods per Assembly	208	236
Rod Zirc-4 Clad O.D. (inch)	0.430	0.382
Clad Thickness (inch)	0.0265	0.025
Fuel Pellet O.D. (inch)	0.3686	0.325
Fuel Pellet Density (% TD)	95	95
Fuel Pellet Dishing (%)	2.03	3.37
Rod Pitch (inch)	0.568	0.506

Number Zirc-4 Guide Tubes	17	5
Guide Tube O.D. (inch)	0.530	0.980
Guide Tube Thickness (inch)	0.016	0.04

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

b. The storage cell nominal geometry is shown on Figure 4-3.

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

d. The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the analytical model. The nominal case calculation is infinite in lateral and axial extent. Poison plates are not necessary on the periphery of the rack module except for the sides of the module adjacent to another rack module. Calculations for the racks with the poison removed on sides adjacent to the pool wall indicate a less reactive configuration than the nominal case of an infinite rack. Therefore, the nominal case of an infinite array of poison cells is a conservative assumption.

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

- poison pocket thickness
- stainless steel thickness
- can ID
- center-to-center spacing
- can bowing

The calculational method for treatment of uncertainty and bias is discussed in Section 4.2.4 of this report.

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

4.2.3 Postulated Accidents

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more than eight inches of water separating it from the rest of the stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity such as a theoretical example of an inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1976 is applied. This states that it shall require two unlikely, independent, concurrent events to produce a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition.

The presence of approximately 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. In perspective, this is more negative reactivity than is present in the poison plates (25% ΔK) so K_{eff} for the rack would be less than 0.95 even if the poison plates were not present. Thus, $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

For fuel storage applications, water is usually present. However, accidental criticality when fuel assemblies are stored in the dry condition is also accounted for. For this case, possible sources of moderation, such as those that could arise during fire fighting operations, are included in the analysis.

The "optimum moderation" accident is not a problem in poisoned fuel storage racks. The presence of poison plates removes the conditions necessary for "optimum moderation" so that K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 to 0.0 gm/cm^3 in poison rack designs.

4.2.4 Criticality Analytical Method

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. This benchmarking data is 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[2,3] for cross-section generation and KENO IV[4] for reactivity determination.

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

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

The results and some descriptive facts about each of the 27 benchmark critical experiments are given in Table 4-1. The average K_{eff} of the benchmarks is 0.9998 which demonstrates that there is no bias associated with the method. The standard deviation of the K_{eff} values is 0.0057 ΔK. The 95/95 one-sided tolerance limit factor for 27 values is 2.26. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.013 ΔK.

The total uncertainty (TU) to be added to a criticality calculation is:

$$TU = [(ks)^2_{method} + (ks)^2_{nominal}]^{1/2}$$

where $(ks)_{method}$ is 0.013 as discussed above and $(ks)_{nominal}$ is the statistical uncertainty associated with the particular KENO calculation being used. The most important effect on reactivity of the mechanical tolerances is the possible reduct[illegible] in the water gap between the poison plates of adjacent storage cells. For ANO-1 and 2, the worst combination of mechanical tolerances (i.e., sheet metal thickness, cell I.D. maximum, rack grid assembly, and cell bowing) will result in a reduction of the water gap between adjacent cells. For a single can it is found that reactivity does not increase significantly because the increase in reactivity due to the water gap reduction on one side of the can is offset by the decrease in reactivity due to the increased water gap on the opposite side of this can. The analysis, for the effect of mechanical tolerances, however, assumes a worst case of a rack composed of an array of groups of four cans where the water gap between the four cans is reduced. The reactivity increase of this configuration is found and is included as a bias term in calculating the K_{eff} of the rack.

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

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

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

4.2.5 Rack Modification

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

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

where:

$K_{nominal}$ = nominal case KENO K_{eff}

B_{mech} = K_{eff} bias to account for the fact that mechanical tolerances can result in water gaps between poison plates less than nominal.

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

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

$ks_{nominal}$ = 95/95 uncertainty in the nominal case KENO K_{eff}.

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

Substituting calculated values, K_{eff} is shown to be less than 0.95. Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criterion for criticality is met.

4.3 CRITICALITY ANALYSIS FOR REGION 2

4.3.1 Analytical Methods

The methods used in the analysis of Region 2 include NITAWL, XSDRNPM and KENO-IV for basic reactivity determination, along with LEOPARD[8], CINDER[9] and TURTLE[10] for reactivity equivalencing. LEOPARD and CINDER are used to calculate the isotopic compositions and cross-sections of the fuel as a function of burnup history and subsequent decay time. These cross-sections are then input to the TURTLE code, which is used to determine the reactivity equivalence (in the Region 2 rack) of assemblies with different initial enrichments and burnups. The reactivity equivalencing is extended back to an unirradiated assembly, which is then analyzed using NITAWL, XSDRNPM and KENO-IV in a fashion similar to the analysis for Region 1.

The accuracy of the burnup dependent isotopics is given in Table 4-2. These measurements were taken from the Saxton Core II[11] and the LEOPARD predictions show excellent agreement. The agreement between measurement and prediction not only verifies the accuracy of the isotopic predictions, it also verifies the accuracy of the cross-sections of the actinides and therefore indirectly the reactivity worth. In order to account for uncertainties in the prediction of the actinide number densities, an uncertainty of 5 percent of the worth of the actinides (.009 = .05 x .18) will be applied to the final rack multiplication factor. The accuracy of the reactivity calculations is shown in Table 4-3, giving the results of 101 critical experiments analyzed with LEOPARD[12].

When comparing the reactivities of spent and unirradiated fuel assemblies, an uncertainty arises in the ability to predict the depletion of the actinides in concert with the accumulation of fission products. Due to a lack of clean experimental information on the reactivity of spent fuel, the uncertainty must be inferred from comparisons of actual and calculated reactor reactivity lifetimes using LEOPARD and TURTLE. Lifetime calculations are typically within 2 percent of predictions. An extremely conservative estimate of the uncertainty due to depletion can be made by taking 5 percent of the change in reactivity due to burnup. As an example, take a B&W 15x15 assembly with a 3.10 w/o initial enrichment at 21000 MWD/MTU. The total reactivity change from fresh to spent is .188 ΔK. The corresponding uncertainty in the Region 2 multiplication factor is then .0094 ΔK (= .05x.188 ΔK).

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

4.3.2 Reactivity Equivalency

One of the basic principles behind the spacer pocket design is the concept of reactivity equivalencing. In this concept, a constant rack k_∞ contour is constructed in enrichment-burnup space using LEOPARD and TURTLE. The intersection point at zero burnup is then calibrated using KENO-IV. Figure 4-2 shows the constant k_∞ contour based on a high enrichment endpoint of 4.10 w/o and 36000 MWD/MTU. The advantage of this approach is that LEOPARD and TURTLE are used only to calculate relative reactivities as a function of burnup (a calculational ability qualified by many years of reactor design experience), while the actual rack reactivity determination is performed by the more powerful Monte Carlo method.

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

4.3.3 Reactivity Determination

The final K_{eff} for Region 2 is determined using the same methods described in Section 4.2 for Region 1. The actual conditions for this determination are defined by the zero burnup intercept point in Figure 4-2. In this instance, the intercept point is at 1.4 w/o U^{235}. The design model for Region 2 will therefore be an unirradiated assembly at 1.4 w/o initial enrichment.

The K_{eff} for Region 2 is also determined assuming a checker board storage configuration, i.e., alternate spacer pocket occupation, with unirradiated assemblies at 4.1 w/o enrichment.

Adjacent vacant spaces between stored assemblies in this case will be controlled administratively to prevent assembly introduction.

4.3.4 Postulated Accidents

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more than eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction).

However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

The presence of the approximately 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

For fuel storage applications, water is usually present. However accidental criticality when fuel assemblies are stored in the dry condition is also accounted for. For this case, possible sources of moderation, such as those that could arise during fire fighting operations, are included in the analysis.

This "optimum moderation" accident is not a problem in fuel storage racks because possible water densities are too low (≤ 0.01 gm/cm^3) to yield K_{eff} values higher than for full density water and the rack design prevents the preferential reduction of water density between the cells of a rack (e.g., boiling between cells).

4.3.5 Manufacturing Biases

The construction tolerances for the spacer pocket racks allow for the nominal center-to-center spacing to be randomly reduced for individual cells. This change will result in an increase in K_{eff} which will be treated conservatively as a bias. The effect of the tolerances on pocket height and material thicknesses also result in an increase in K_{eff} which will be treated conservatively as a bias.

Another center-to-center spacing reduction can be caused by the asymmetric assembly position within the storage cell. The inside dimensions of a nominal storage cell are such that if a fuel assembly is loaded into the corner of the cell, the assembly centerline will be displaced from the cell centerline. This means that adjacent asymmetric fuel assemblies would have their center-to-center distance reduced from the nominal. Analysis shows this reduction may increase reactivity. This will also be treated as a bias although the asymmetric positioning of assemblies within storage cells will in reality be random.

The final K_{eff} for Region 2 is constructed according to the following formula:

$$K_{eff} = K_{nom} + B_{meth} + B_{mech} + B_{asym} + [(ks_{meth})^2 + (ks_{nom})^2 + (ks_{pu})^2 + (ks_{bu})^2]^{\frac{1}{2}}$$

where:

K_{nom} is the eigenvalue from KENO for the nominal storage configuration,

B_{meth} is the bias in the method,

B_{mech} is the bias induced by material thickness variations and mechanical tolerances,

B_{asym} is the bias induced by the potential placement of the assemblies asymmetrically in the can,

ks_{meth} is the method uncertainty (95/95),

ks_{nom} is the uncertainty (95/95) on the nominal eigenvalue,

ks_{pu} is the uncertainty on the plutonium reactivity, and

ks_{bu} is the uncertainty on reactivity as a function of irradiation.

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

The final K_{eff} for Region 2 from this construction is less than 0.95, including all uncertainties at a 95/95 probability/confidence level. Therefore, the acceptance criterion for criticality is met.

4.4 ACCEPTANCE CRITERION 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.

Generally, the acceptance criterion for postulated accident conditions can be $K_{eff} \leqq 0.98$ because of the accuracy of the methods used coupled with the low probability of occurrence. For instance, in ANSI N210-1976, the acceptance criterion for the "optimum moderation" condition is $K_{eff} \leqq 0.98$. However, for storage pools which contain dissolved boron, the use of realistic initial conditions ensures that $K_{eff} \ll 0.95$ for postulated accidents as discussed in Sections 4.2.3 and 4.3.4. Thus, for simplicity, the acceptance criterion for all conditions will be $K_{eff} \leqq 0.95$.

TABLE 4-1

BENCHMARK CRITICAL EXPERIMENT[4,5,6]

	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.	"	"	"	"	8.39	0.093 ± .004
3.	"	"	"	"	6.39	1.005 ± .004
4.	"	"	"	"	4.46	0.994 ± .004
5.	"	"	"	Stainless steel	10.44	1.005 ± .004
6.	"	"	"	"	11.47	0.99[illegible] ± .004
7.	"	"	"	"	7.76	0.[illegible] ± .004
8.	"	"	"	"	7.42	1[illegible]4 ± .004
9.	"	"	"	boral	6.34	1.[illegible]5 ± .004
10.	"	"	"	"	9.03	0[illegible] ± .004
11.	"	"	"	"	5.05	1.001 ± .004
12.	"	4.29	"	water	10.64	0.999 ± .005
13.	"	"	"	Stainless steel	9.76	0.999 ± .005
14.	"	"	"	"	8.08	0.998 ± .006
15.	"	"	"	boral	6.72	0.998 ± .005
16.	U metal cylinders	93.2	bare	air	15.43	0.998 ± .003
17.	"	"	paraffin	air	23.84	1.006 ± .005
18.	"	"	bare	air	19.97	1.005 ± .003
19.	"	"	paraffin	air	36.47	1.001 ± .004
20.	"	"	bare	air	13.74	1.005 ± .003
21.	"	"	paraffin	air	23.48	1.005 ± .004
22.	"	"	bare	plexiglas	15.74	1.010 ± .003
23.	"	"	paraffin	plexiglas	24.43	1.006 ± .004
24.	"	"	bare	plexiglas	21.74	0.999 ± .003
25.	"	"	paraffin	plexiglas	27.94	0.994 ± .005
26.	"	"	bare	steel	14.74	1.000 ± .003
27.	"	"	bare	plexiglas steel	16.67	0.996 ± .003

TABLE 4-2

SAXTON CORE II ISOTOPICS ROD MY, AXIAL ZONE 6

Atom Ratio	Measured[a]	2σ Precision (%)	LEOPARD Calculation
U-234/U	4.65×10^{-5}	±29	4.60×10^{-5}
U-235/U	5.74×10^{-3}	±0.9	5.73×10^{-3}
U-236/U	3.55×10^{-4}	±5.6	3.74×10^{-4}
U-238/U	0.99386	±0.01	0.99385
Pu-238/Pu	1.32×10^{-3}	±2.3	1.222×10^{-3}
Pu-239/Pu	0.73971	±0.03	0.74497
Pu-240/Pu	0.19302	±0.2	0.19102
Pu-241/Pu	6.014×10^{-2}	±0.3	5.47×10^{-2}
Pu-242/Pu	5.81×10^{-3}	±0.9	5.38×10^{-3}
Pu/U[b]	5.938×10^{-2}	±0.7	5.970×10^{-2}
Np-237/U-238	1.14×10^{-4}	±15	0.86×10^{-4}
Am-241/Pu-239	1.23×10^{-2}	±15	1.08×10^{-2}
Cm-242/Pu-239	1.05×10^{-4}	±10	1.11×10^{-4}
Cm-244/Pu-239	1.09×10^{-4}	±20	0.98×10^{-4}

(a) Reported in reference 11

(b) Weight ratio

TABLE 4-3

BENCHMARK CRITICAL EXPERIMENTS

LEOPARD COMPARISONS

Description of Experiments[a]	Number of Experiments	LEOPARD k_{eff} Using Experimental Bucklings
UO^2		
Al clad	14	1.0012
SS clad	19	0.9963
Borated H_2O	7	0.9989
Subtotal	40	0.9985
U-Metal		
Al clad	41	0.9995
Unclad	20	0.9990
Subtotal	61	0.9993
Total	101	0.9990

(a) Reported in reference 12

FIGURE 4-1 SPACER POCKET LAYOUT

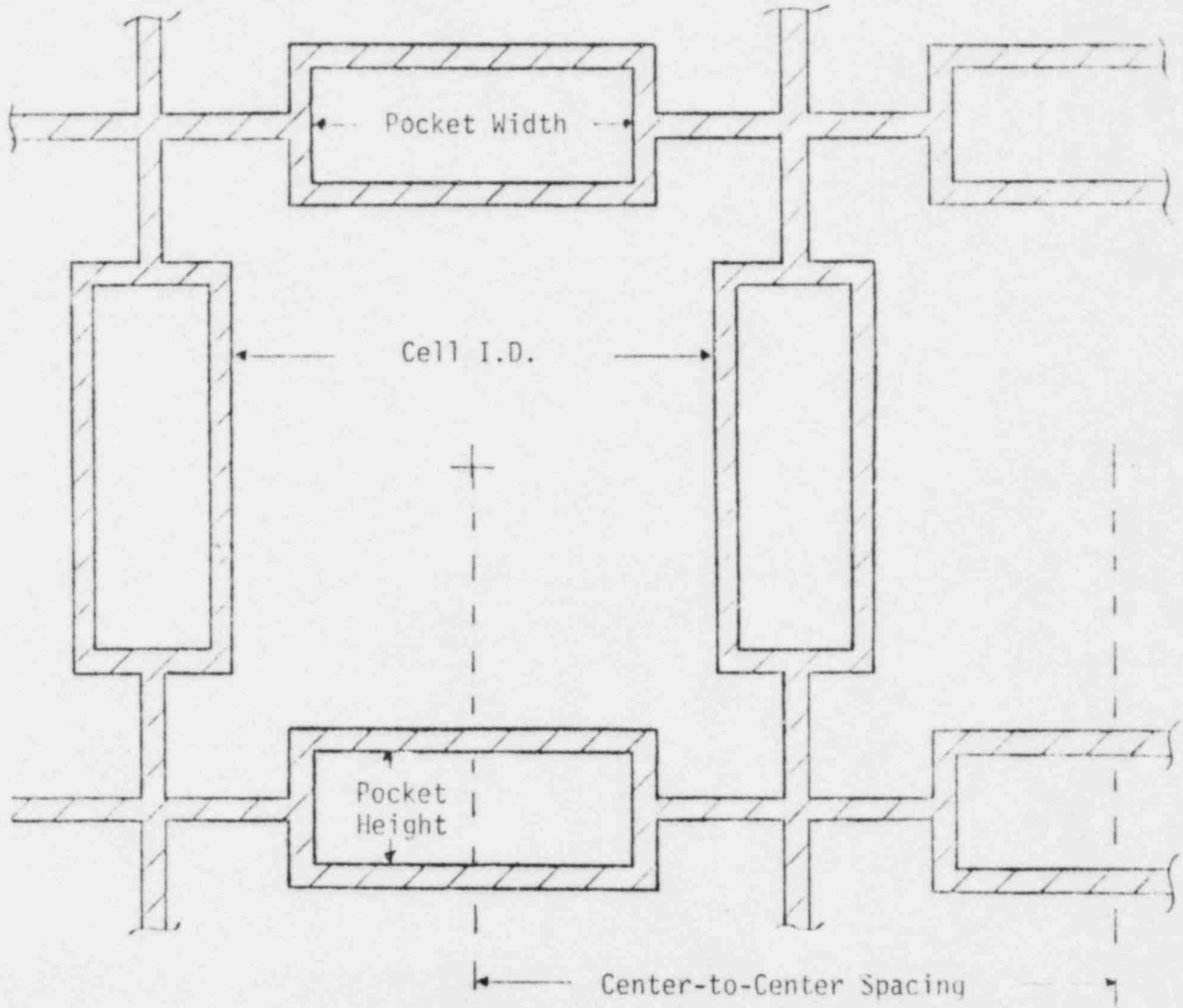


FIGURE 4-2

MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR REGION 2 STORAGE

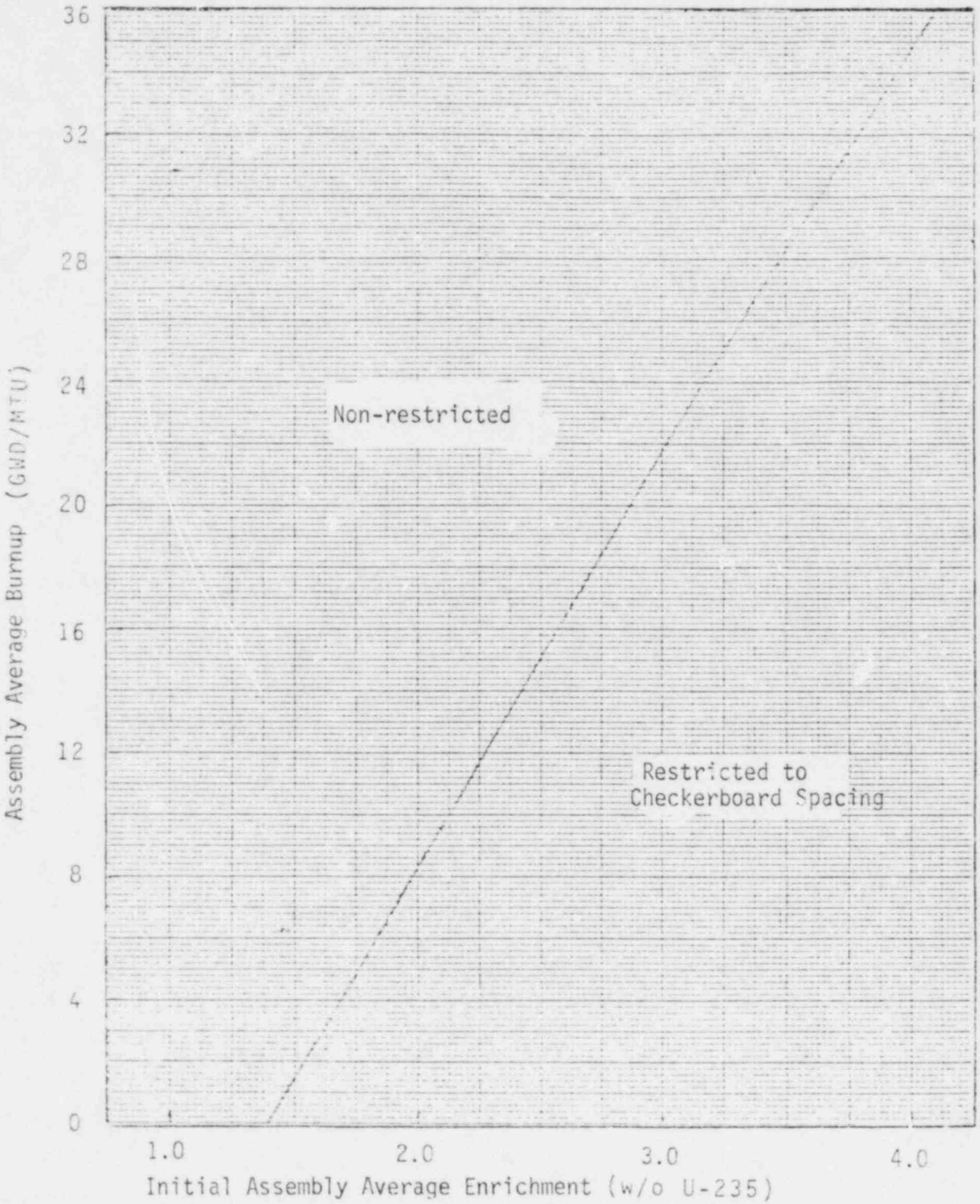
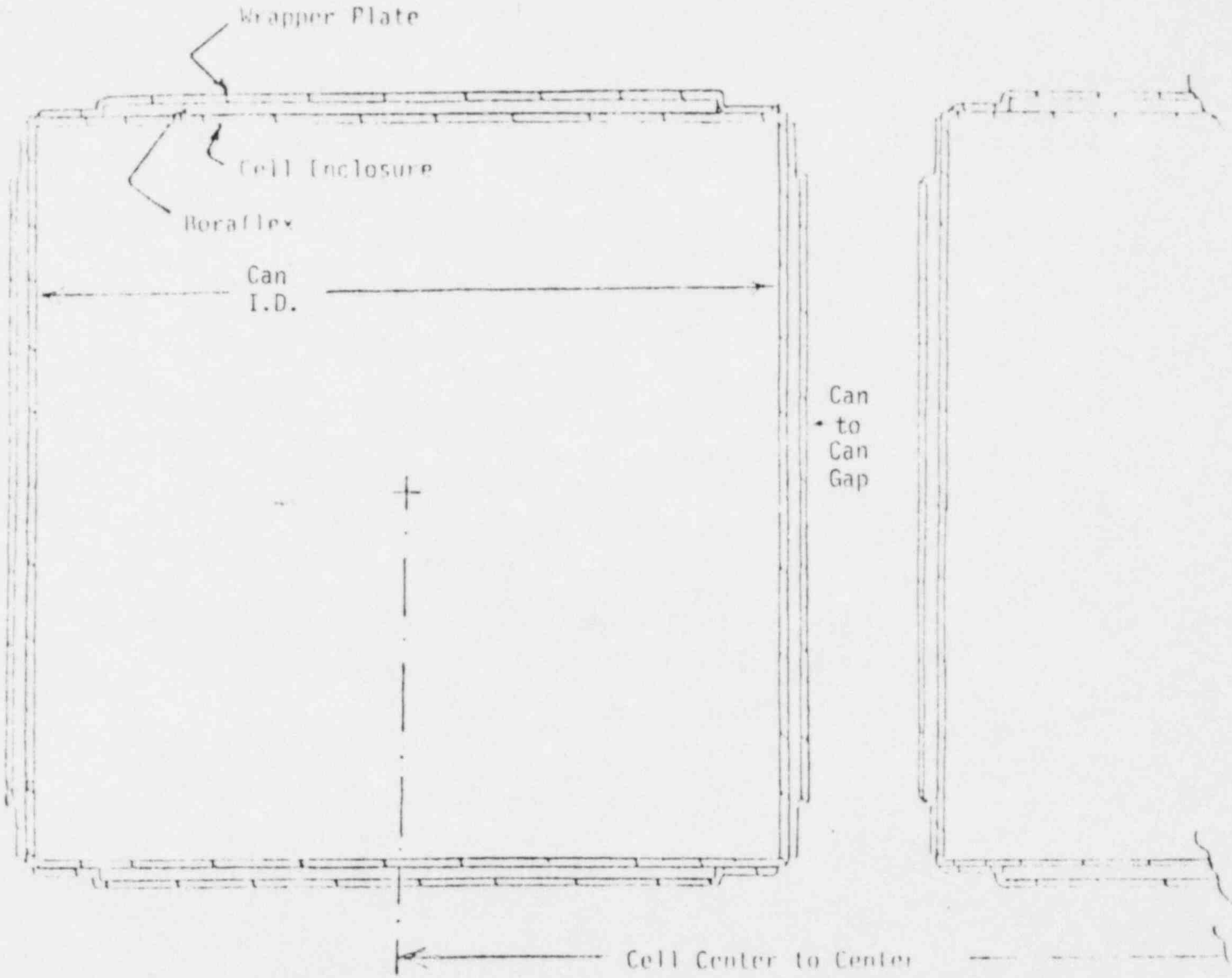


Figure 4-3 Region 1 Cell Layout



5.0 THERMAL-HYDRAULIC CONSIDERATIONS

The ANO spent fuel pool cooling and purification systems are designed to maintain water quality and clarity and to remove the decay heat from the stored fuel.

5.1 SPENT FUEL POOL COOLING SYSTEM DESIGN BASES

The ANO-1 cooling system was originally designed to maintain the spent fuel pool water at 150°F with a heat load based upon removing the decay heat generated from 1-1/3 cores. One batch irradiated for 930 days and cooled for 100 days, one batch irradiated for 720 days and cooled for 150 hours, one batch irradiated for 410 days and cooled for 150 hours, and one batch irradiated for 100 days and cooled for 150 hours. The ANO-2 cooling system was originally designed to maintain the pool temperature at 150°F with 2-3/4 cores in the pool assuming that one full core which contains three batches irradiated for one, two, and three years is placed in the pool seven days after reactor shutdown along with five batches of fuel from previous refuelings.

5.1.1 Heat Loads and Pool Temperatures

The ANO-1 and 2 pool cooling systems have the capability of maintaining the spent fuel pool at or below 150°F with a heat load from the typical combinations of spent fuel listed in Tables 5-1 and 5-2 that have been cooled for 168 hours or more.

This analysis was performed using the guidelines of APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

5.2 LOCAL FUEL BUNDLE THERMAL-HYDRAULIC ASSUMPTIONS AND CONSIDERATIONS

A local fuel bundle thermal-hydraulic analysis is performed to determine the maximum fuel clad temperatures which may occur as a result of using the spent fuel racks in the Arkansas spent fuel pools.

Key assumptions used in the analysis are:

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

- The maximum fuel assembly spent fuel pool decay heat output is 1.36×10^5 Btu/Hr.

- The maximum temperature of the water at the inlet to the storage cells is 150°F when the cooling system is operational.

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

5.3 DESCRIPTION OF ANALYTICAL METHOD AND TYPES OF CALCULATIONS PERFORMED

A natural circulation calculation is employed to determine the thermal-hydraulic conditions within the spent fuel storage cells. The model used assumes that all downflow occurs in the peripheral gap between the pools walls and the outermost storage cells and all lateral flow occurs in the space between the bottom of the racks and bottom of the pool. The effect of flow area blockage in the region is conservatively accounted, 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 5-1 where the flow paths are indicated by arrows. Each cell shown actually corresponds to a row of cells that are located at the same distance from the pool walls. This is more clearly shown in a plan view, Figure 5-2.

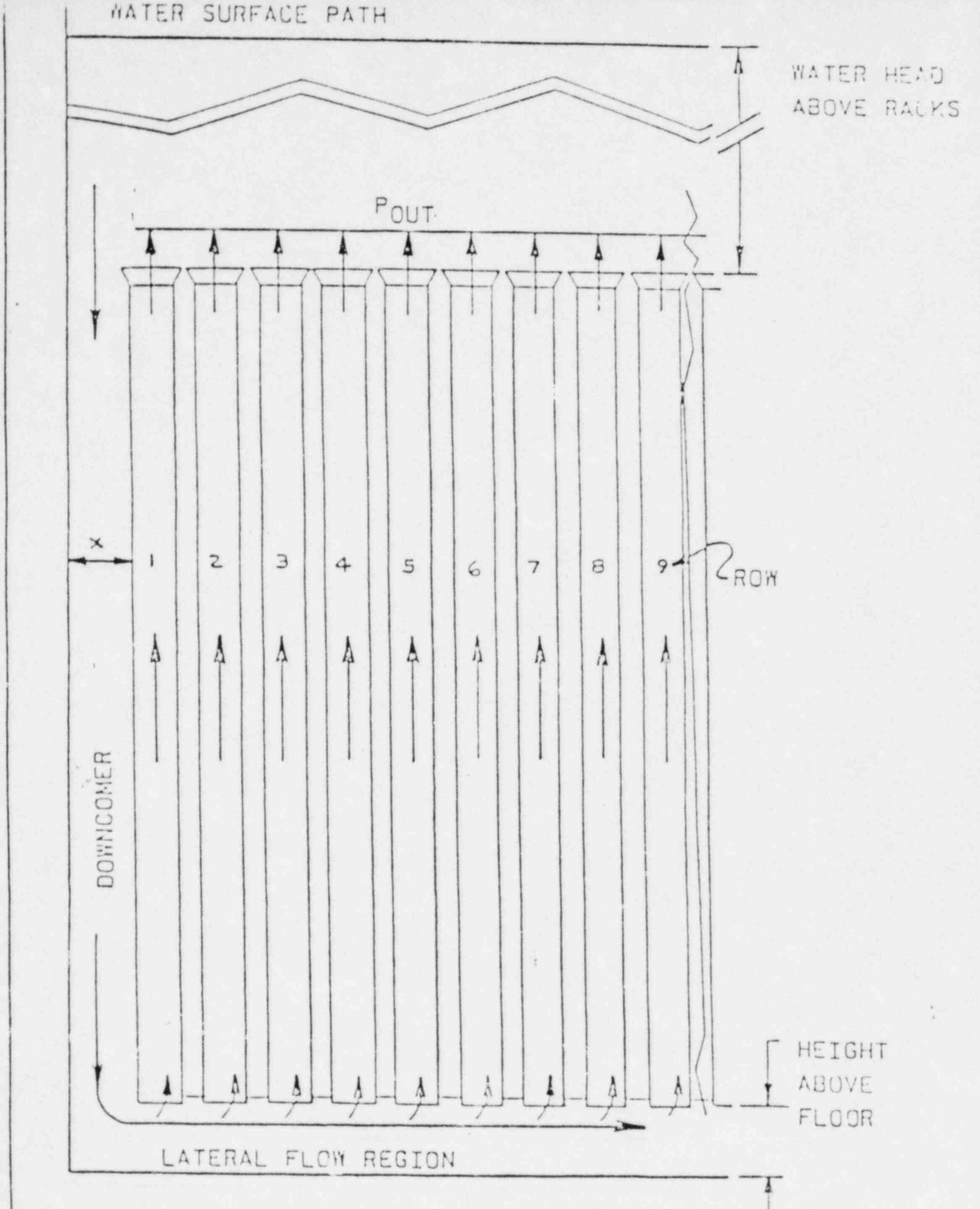

WATER SURFACE PATH
WATER HEAD
ABOVE RACKS
P OUT
x
1
2
3
4
5
6
7
8
9
ROW
DOWNCOMER
HEIGHT
ABOVE
FLOOR
LATERAL FLOW REGION

FIGURE
5-1

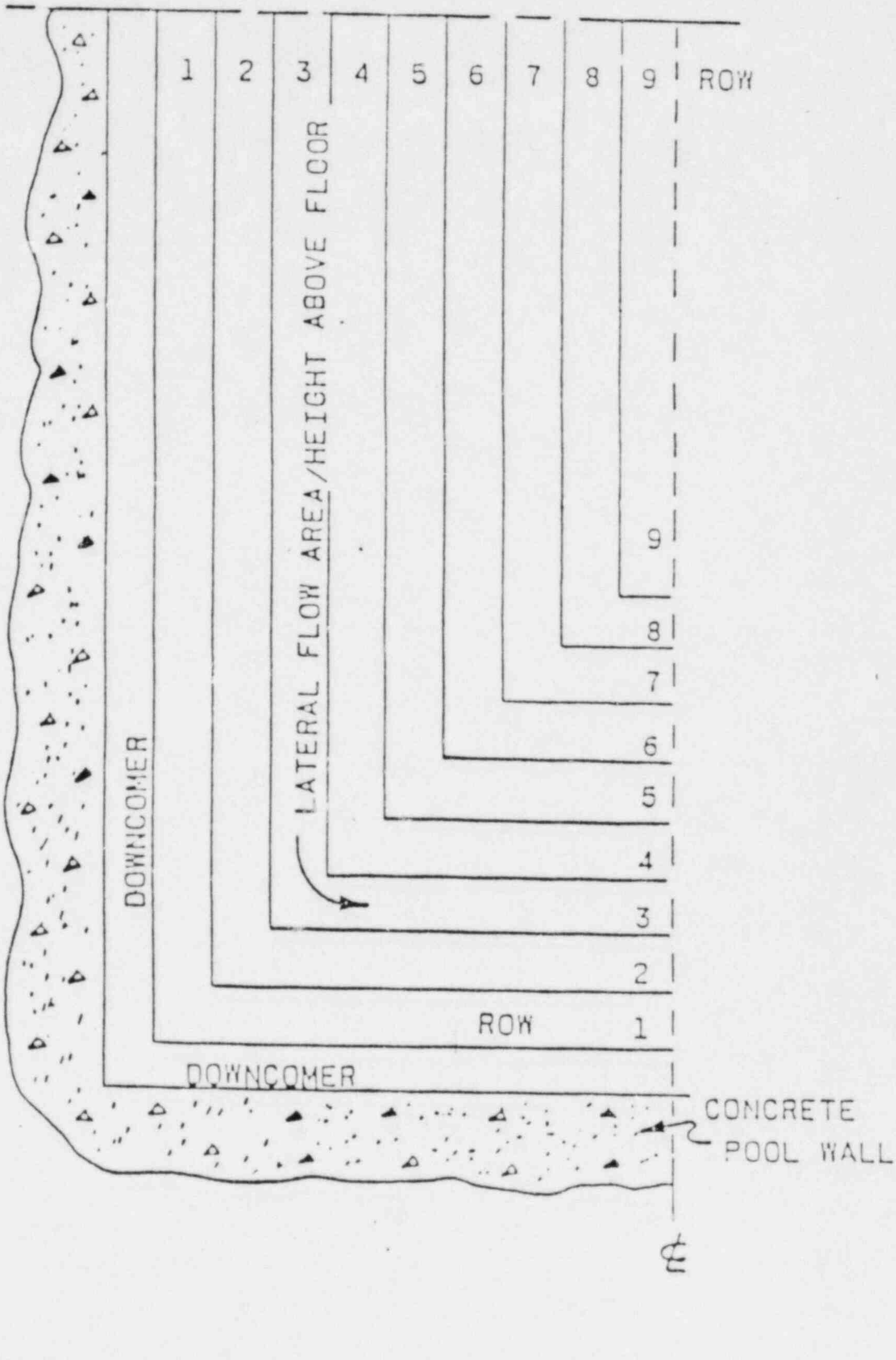

1
2
3
4
5
6
7
8
9
ROW
LATERAL FLOW AREA/HEIGHT ABOVE FLOOR
9
8
7
6
5
4
3
2
ROW
1
DOWNCOMER
DOWNCOMER
CONCRETE
POOL WALL
℄

FIGURE
5-2

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

The most recently discharged or "hottest" fuel assemblies are assumed to be located in various rows during different calculations in order to 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 decay heat rate. This maximizes the total natural circulation flow rate which leads to conservatively large pressure drops in the downcomer and lateral flow regions which reduces the driving pressure drop across the limiting storage location.

Since the natural circulation velocity strongly affects the temperature rise of the water and 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 such as to maximize this flow area. Each storage cell has either one large flow opening or multiple smaller openings. The use of large and/or multiple flow holes virtually eliminates the possibility that all flow into the inlet of a given cell can be blocked by debris or other foreign material that may get into the pool.

In order to determine the impact of a partial blockage on the thermal-hydraulic conditions in the cells, an analysis is also performed for various assumed blockages.

The analyses that have been described only address the flow through the storage cells. As noted in the discussion of criteria, it is also required that the flow and temperatures in the axial gap between adjacent storage locations be evaluated. In order to preclude the possibility of stagnant conditions in these gaps, flow relief areas are provided at the bottom of the cell assemblies. This flow area also ensures that air or steam cannot be trapped in the rack structure. The thermal-hydraulic conditions in the gap region are evaluated by using a parallel path thermal-hydraulic model of the gap and cell under consideration. This analysis considers the gamma heat generation in the cell enclosure and cell wrapper in addition to the decay heat input. Using the cell flow velocity and driving pressure differential obtained from the previously described pool analyses, the flow velocity in the gap and the axial temperature distributions of the coolant and structure are determined. The radial temperature distributions through the various components are also considered.

5.4 ACCEPTANCE CRITERIA

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

5.4.1 The design must allow adequate cooling by natural circulation and by flow provided by the spent fuel pool cooling system. The coolant should remain subcooled at all points within the pool when the cooling system is operational. When the cooling system is postulated to be inoperable, adequate cooling implies that the temperature of the fuel cladding should be sufficiently low that no structural failures would occur and that no safety concerns would exist.

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

5.4.3 The rack design must not allow trapped air or steam and direct gamma heating of the storage cell walls and the water between adjacent storage locations must be considered.

5.5 RACK/POOL THERMAL-HYDRAULICS

5.5.1 Design Bases

The Spent Fuel Pools and Pool Cooling Systems are designed to keep the pool water temperature below 120°F for normal refueling operations and 150°F for full core discharge situations. Under normal refueling conditions the fuel is discharged over a four day period after at least three days cooling inside the reactor vessel. The full core discharge is expected to take four days also with three days cooling in the reactor vessel prior to moving any fuel. The heat released from the fuel stored in the pool is determined in accordance with Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." Tables 5-1 and 5-2 show the expected loading in the pools. In the event mixed oxide fuel becomes available, the heat load in the pool will be slightly higher. The increase is apparent only in fuel which has decayed for a relatively long period of time and contributes little additional heat load to the pool.

5.5.2 System Description

The Spent Fuel Cooling Systems are described in the ANO FSAR Section 9.4.2 for ANO-1 and Section 9.1.3 for ANO-2.

5.5.3 Design Evaluation

During normal operation, the Spent Fuel Cooling System serves to maintain the pool water at temperatures below 120°F. Under normal conditions, the pool temperature is maintained at approximately 120°F as stated in the ANO FSAR Section 9.6.2.4.3.5 for Unit 1 and Section 9.1.3.2 for Unit 2 by recirculating spent fuel cooling water from the spent fuel pool through pumps and cooler(s) and back into the pool. For maximum normal conditions, full core offload, the pool temperature is maintained at approximately 150°F. An analysis of pool responses to loss of forced cooling is presented in Section 10.3 of this document.

The heat loads shown in Tables 5-1 and 5-2 represent the largest heat loads expected in the spent fuel pools. The calculations are based upon 18-month cycles for both units.

TABLE 5-1

ANO-UNIT 1 SPENT FUEL
OPERATING & COOLING TIMES

	To (EFPD)	Ts (Days)	Dis. Date	# Assemb.	Heat Load (10^6 Btu/Hr)
Batch 1	500	6310	1/77	60	0.03
Batch 2	772	5914	2/78	56	0.05
Batch 3	1101	5489	4/79	61	0.08
Batch 4	891	4849	1/81	60	0.06
Batch 5	1074	4150	12/82	68	0.09
Batch 6	1171	3512	9/84	64	0.10
Batch 7	1198	2965	3/86	68	0.12
Batch 8	1099	2417	9/87	72	0.12
Batch 9	1068	1870	3/89	68	0.13
Batch 10	1068	1322	9/90	64	0.16
Batch 11	1068	775	3/92	64	0.30
Batch 12	1068	227	9/93	64	1.19
Batch 13	862(1)	7	4/94	64	9.19
Batch 14	506	7	4/94	64	8.87
Batch 15	150	7	4/94	64	7.53

To = Operation Time

Ts = Cooling Time

(1) Full Core Discharge after 150 Days Operation into Cycle 15

TABLE 5-2

ANO-UNIT 2 SPENT FUEL
OPERATING & COOLING TIMES

	To (EFPD)	Ts (Days)	Dis. Date	# Assemb.	Heat Load (10^6 Btu/Hr)
Batch 1	325	6067	3/81	61	0.03
Batch 2	612	5520	9/82	60	0.05
Batch 3	873	5155	9/83	56	0.07
Batch 4	892	4607	3/85	60	0.08
Batch 5	961	4060	9/86	52	0.08
Batch 6	1056	3512	3/88	68	0.11
Batch 7	1068	2965	9/89	68	0.12
Batch 8	1068	2417	3/91	68	0.12
Batch 9	1068	1870	9/92	68	0.14
Batch 10	1068	1322	3/94	68	0.18
Batch 11	1068	775	9/95	68	0.34
Batch 12	1068	227	3/97	68	1.39
Batch 13	862 (1)	7	8/97	68	10.70
Batch 14	506	7	8/97	68	10.33
Batch 15	150	7	8/97	68	8.76

To = Operation Time

Ts = Cooling Time

(1) Full Core Discharge after 150 Days Operation into Cycle 15

6.0 STRUCTURAL AND SEISMIC CONSIDERATIONS

6.1 ASSUMPTIONS

The purpose of the seismic and stress analysis is to analyze the proposed modified spent fuel module under various loading conditions. The racks are evaluated for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) conditions and meet Seismic Category I requirements. A detailed stress analysis is performed to verify the acceptability of the critical load components and paths under normal and faulted conditions. The racks rest freely on the pool floor and are evaluated to ensure that under various loading conditions they do not impact each other, nor do they impact the pool walls.

6.2 SEISMIC ANALYSIS

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure. The dynamic response, internal stresses, and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. The second phase is a response spectrum analysis of a detail rack assembly finite element model. The damping values used in the seismic analysis are two percent damping for OBE and four percent damping for SSE as specified in NRC Regulatory Guide 1.61.

The simplified nonlinear finite element model is used to determine the fuel rack response for the structural characteristics of a submerged rack assembly. The nonlinearities of the fuel rack assembly which are accounted for in the model are in the gap between the fuel cell and the fuel assembly, the boundary conditions of the fuel rack support locations, and energy losses at the support locations.

The WECA' computer program [13,14] is used to determine the nonlinear time history response of the fuel assembly/fuel rack system. The fuel assembly to cell impact loads, and overall rack response are obtained from the nonlinear time history results.

The detail model is a three-dimensional finite element representation of a rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of model points. The results of the nonlinear time history model are incorporated in the detail model. Since the detail model does not account for the nonlinear effect of a fuel rack assembly, the internal loads and stresses for the rack assembly obtained from this model are corrected by load correction factors. The load correction factor is derived from the nonlinear model results and is applied to the components in the stress analysis. The responses of the model from accelerations in three directions are combined by the Square Root Sum of the Squares method in the stress analysis. The loads in two major components (support pad assembly and fuel cell) are examined, and the maximum loaded section of each of these components is found. These maximum loads from the detail model are corrected by the nonlinear load correction factors and used in the stress analysis to obtain the stresses within the rack assembly.

6.3 FUEL RACK STRUCTURAL ANALYSIS

The stress analysis for the racks is performed using the load combinations specified in the "NRC Position For Review and Acceptance of Spent Fuel Storage and Handling Applications," as described in Section 3.2.

The thermal loads due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction. The major seismic loads are produced by the operational basis earthquake (OBE) and safe shutdown earthquake (SSE) events.

It is noted from the seismic analysis that the magnitude of stresses vary considerably from one geometrical location to the other in the model. Consequently, the maximum loaded cell assembly, grid assembly, and the leveling pad assembly are analyzed. Such an analysis envelopes the other areas of the rack assembly.

Because of structural symmetry of the cell assembly about the x and y axes, the x and y direction horizontal seismic events produce identical loads. Consequently, the margins of safety for the multi-direction (x and y directions simultaneously) seismic event is computed by multiplying the undirectional loads by $\sqrt{2}$.

The loads described in the seismic analysis section are adjusted by load modification factors obtained from the nonlinear analysis. The computed stresses are below the allowable stresses as required by the NRC Position Paper.

6.4 FUEL HANDLING CRANE UPLIFT ANALYSIS

A fuel handling crane uplift analysis is performed to demonstrate that the rack can withstand the maximum 3000 pound uplift load of the fuel handling crane without violating the criticality acceptance criteria. In this analysis the uplift load is assumed to be applied to a fuel cell. Resulting stresses are within acceptable limits, and there is no change in rack geometry of a magnitude which causes the criticality acceptance criteria to be violated.

6.5 FUEL BUNDLE/MODULE IMPACT EVALUATION

An analysis is performed to evaluate the effect of an impact load due to fuel assembly and fuel storage cell interaction during a seismic event. The fuel rack system consists of an array of cells which form the fuel rack structure and fuel assemblies. The fuel rack system is located in the spent fuel pool and is submerged in water.

Since the fuel assembly is stored within the cell, the gap between the fuel assembly grid and cell changes (i.e., opens and closes) during a seismic event. From the equation of motion for such a system it is evident that the fuel rack system is nonlinear. This condition necessitates the performance of a transient dynamic analysis.

The mathematical features of the nonlinear fuel rack model facilitate the determination of the fuel assembly/cell interaction and hydrodynamic mass (fluid mass) effects on the fuel rack response during seismic excitation.

The effect of fuel assembly and fuel storage cell impact force on the rigid body displacements is obtained from the nonlinear analysis. The analysis is conducted with a minimum coefficient of friction of 0.2, and it is shown that the rigid body displacement is minimal. Thus, impact between adjacent rack modules or between a rack module and the pool wall is precluded.

The fuel assembly and fuel storage cell impact forces obtained from the nonlinear analysis are 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.

6.6 ACCEPTANCE CRITERIA

The fuel racks are analyzed for the normal and faulted load combinations of Section 3.2 in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The major normal and upset conditions loads are produced by the operational basis earthquakes (OBE). The thermal stresses due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction.

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

The stresses are below the allowable stresses as required by the ASME B&PV Code, Section III, Subsection NF.

In summary, the results of the seismic and structural analyses show that the ANO Units 1 and 2 spent fuel storage racks meet all the structural acceptance criteria adequately.

6.7 POOL STRUCTURAL ANALYSIS

The existing structures were analyzed for the modified fuel rack loads using the STARDYNE finite element computer program. The finite element models consisted of the pool walls and floor, foundation walls, cask laydown area, and fuel transfer canal area with the lowest elevation of 335'-0" assumed as a fixed boundary. Figures 6-1, 6-2 and 6-3 show details of the models prepared. The pool walls and floor slabs are modeled utilizing two layers of three dimensional solid "brick" elements with three degrees of freedom per node. In order to permit recovery of cube surface stresses, extremely thin quadrilateral membrane elements coincident with the nodes forming the surface of the solid elements were utilized. Recovery of these stresses along with cube centroidal stresses permitted the calculation of the required resultant forces and moments for an American Concrete Institute Code evaluation. The foundation walls were modeled using only one layer of brick elements through the thickness.

The pool liner was not modeled since it is not acceptable to count on its stiffness contribution to the overall pool capacity. However, stress evaluation of this component was conducted using results obtained from the computer analysis.

In order to correctly represent boundary conditions, modeling of the floor diaphrams and shear walls attached to the spent fuel pool was accomplished utilizing STARDYNE matrix elements.

6.7.1 Loads and Loading Combinations

Table 6-1 lists the individual load components to which the spent fuel pool facilities were subjected. In all cases, except for fuel rack loads, the loads are consistent with the original loads used in the analyses performed and documented in the FSAR for each unit.

Table 6-3 describes the final load combinations investigated for both normal and accident conditions. These combinations are based on the strength design method.

6.7.2 Seismic Loading

Original plant response spectra and damping values were used to analyze the spent fuel facilities. Since only one horizontal and one vertical spectra were originally generated, the horizontal spectra was assumed to act in two horizontal directions. NUREG 0800 states that the three directions of the earthquake must be combined in an SRSS fashion; however, a strict SRSS procedure does not maintain the signs of the force components which are of obvious importance for concrete structures. Therefore, the requirements of NUREG-0800 were met by formation of resultant earthquakes based on the three directions of earthquake applied simultaneously.

Components of seismic loads include vertical earthquake effects, which are factored dead weight, plus vertical rack forces and the horizontal effects, which are pool wall inertia forces, pool hydrodynamic forces, building seismic forces, and fuel rack horizontal forces. Table 6-2 shows the resulting factors used to produce a 0.1g effective OBE earthquake simulation.

6.7.3 Structural Acceptance Criteria

The resulting forces and moments for the controlling load combinations were compared against the ultimate strength of the concrete sections. This comparison was initially carried out without thermal relaxation. All load combinations not involving thermal loads must pass this initial criteria check. Any load combination or element involving thermal loads, which meets the criteria without consideration of thermal relaxation, was eliminated from further processing. The load combinations failing to qualify on this first pass were selectively investigated by relieving the thermal moment as the section cracks. This procedure for post-processing ensured that the controlling load combinations were addressed in the most efficient manner possible.

6.7.4 Results

Using the above methods, models, loads, loading combination, and acceptance criteria, the existing spent fuel facilities were determined to safely support the loads generated by the new fuel racks.

ARKANSAS NUCLEAR ONE-UNIT ONE

SPENT FUEL STORAGE FACILITY FINITE ELEMENT MODEL

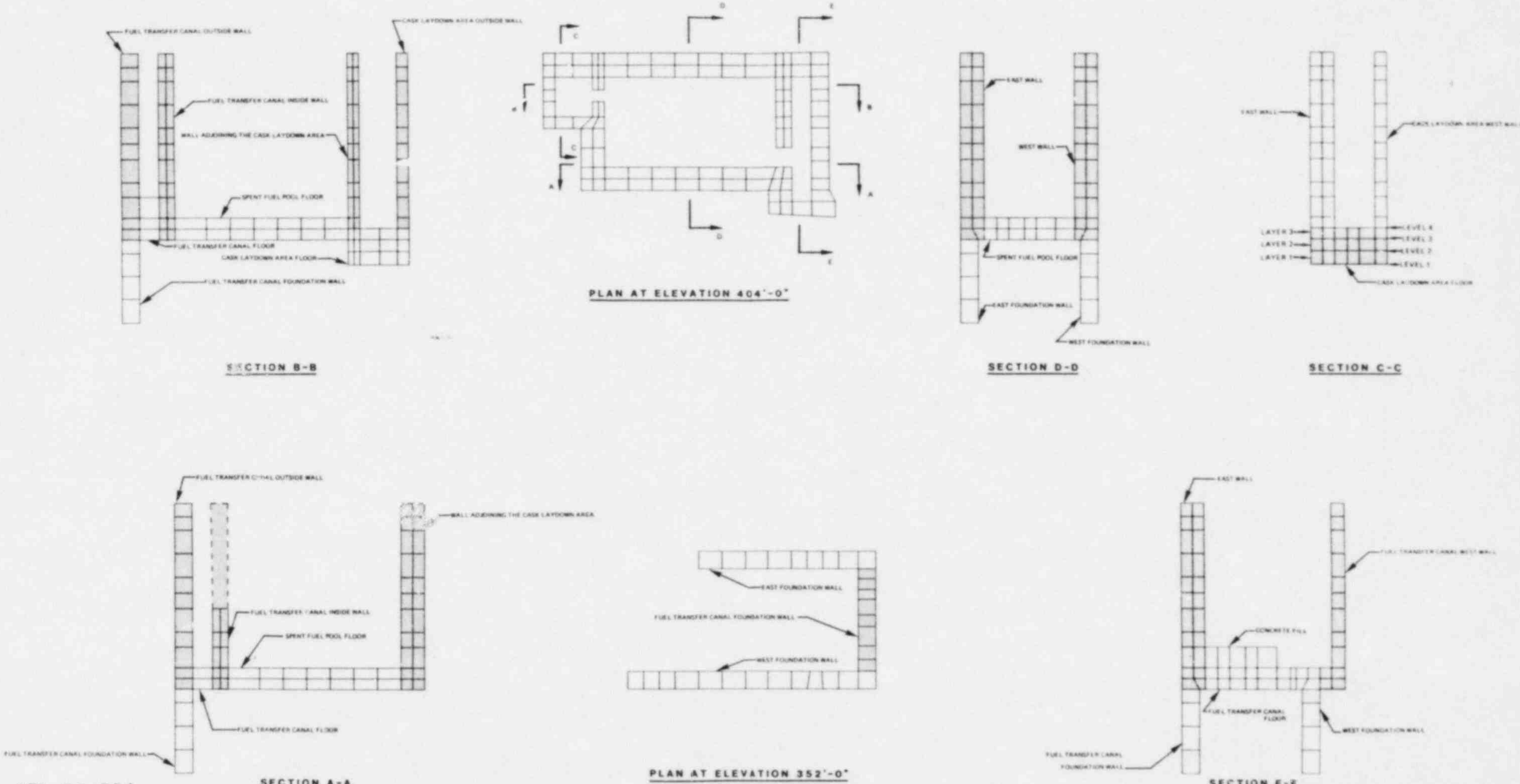


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FIGURE 6-1

PLANS AND SECTIONS

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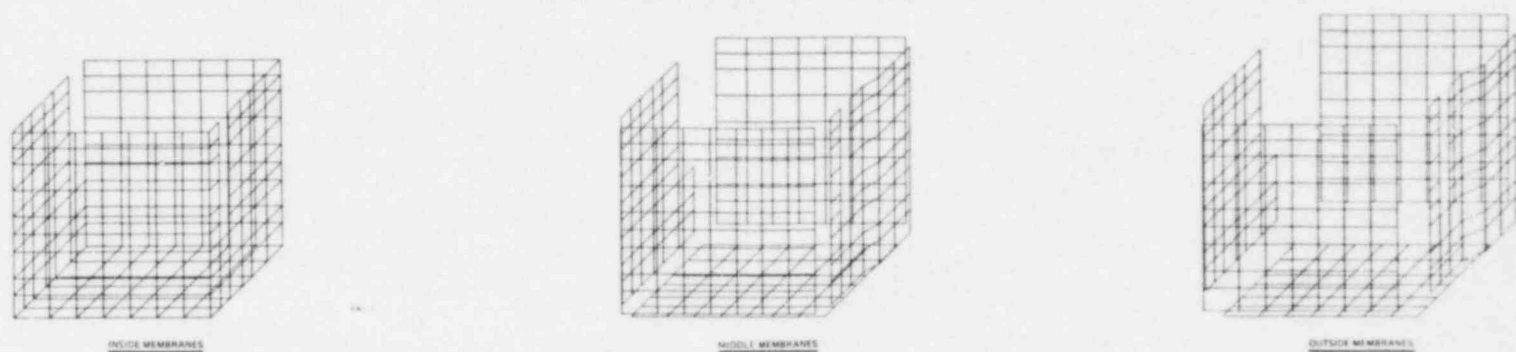
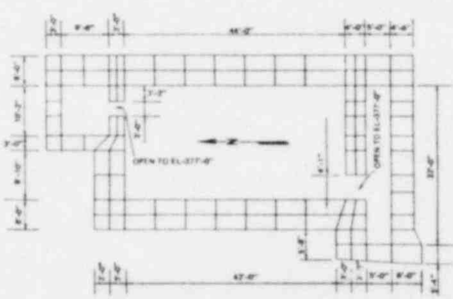
ARKANSAS NUCLEAR ONE-UNIT ONE

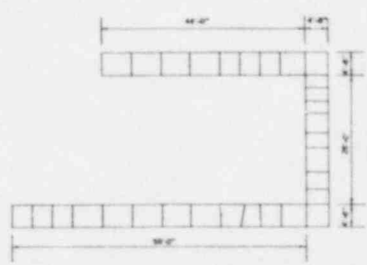
SPENT FUEL STORAGE FACILITY FINITE ELEMENT MODEL

ISOMETRIC VIEWS OF SURFACE MEMBRANE ELEMENTS

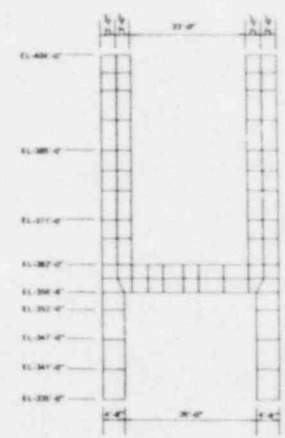




PLAN AT ELEVATION 404'-0"



PLAN AT ELEVATION 352'-0"



SECTION D-D

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FIGURE 6-2

KEY DIAGRAMS AND DIMENSIONS

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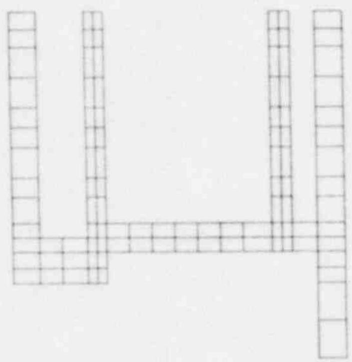
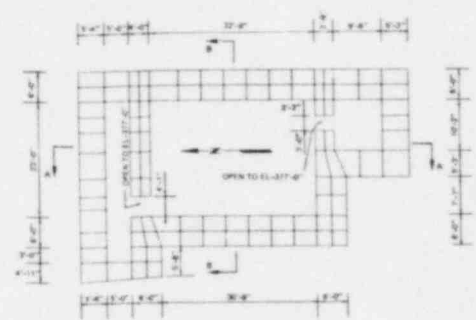
ARKANSAS NUCLEAR ONE-UNIT TWO

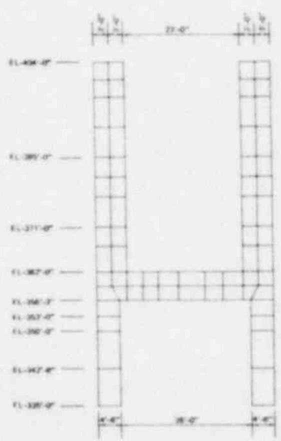
SPENT FUEL STORAGE FACILITY FINITE ELEMENT MODEL



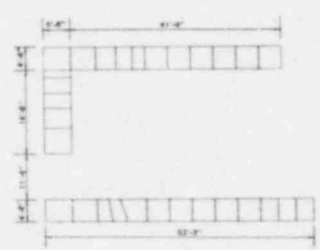
SECTION A-A



PLAN AT ELEVATION 404'-0"



SECTION B-B



PLAN AT ELEVATION 335'-0"

NOTES:

DRAWINGS OF ANO - UNIT 1 FROM SDT LETTER APL-01-004 ENCLOSURE SHALL BE USED FOR ANO - UNIT 2 FINITE ELEMENT MODEL, EXCEPT AS NOTED.

1. ALL UNIT 1 PLOTS REPRESENT THE MIRROR IMAGE OF THE ACTUAL PLOTS FOR ANO - UNIT 2 FINITE ELEMENT MODEL.
2. FOR DIMENSIONS, ELEVATIONS AND ORIENTATION OF UNIT 2 MODEL, SEE FIGURE 1.
3. EAST FOUNDATION WALL SHOWN IN FIGURE 2 WILL REPLACE THE PLOTS OF THIS WALL SHOWN IN FIGURES 10 AND 11 OF THE UNIT 1 DRAWINGS.
4. OPENING IN FUEL TRANSFER CANAL FOUNDATION WALL FOR UNIT 2 WAS REPRESENTED BY ADJUSTING ELEVATIONS OF NODES AS SHOWN IN FIGURE 1, SECTION B-B, AND REMOVING CUBE ELEMENTS 1464 THROUGH 1466, 1964 THROUGH 1966, AND THEIR CORRESPONDING SURFACE MEMBRANE ELEMENTS FROM THE UNIT 1 MODEL (SEE FIGURE 10 OF UNIT 1 DRAWINGS).
5. CONCRETE FILL IN FUEL TRANSFER CANAL HAS BEEN REMOVED.

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FIGURE 6-3

KEY DIAGRAMS AND DIMENSIONS

Structural Dynamics Technology, Inc.

TABLE 6-1
ARKASAS NUCLEAR ONE
SPENT FUEL STORAGE FACILITY STRUCTURAL EVALUATION
INDIVIDUAL LOAD CASE DESCRIPTION TABLE

Load Case No.	Notation	Description
1	D_c	Dead weight of the concrete.
2	H	Hydrostatic pressure due to water in the pool.
3	F	Accident flood load.
4	$T_o^{(1)}$	Normal operating thermal load.
5	$T_a^{(1)}$	Accident thermal load.
6	$E_{ew}^{(2)}$	Load generated by east-west 1g earthquake.
7	$E_{ns}^{(2)}$	Load generated by north-south 1g earthquake.
8	D_{fr}	Fuel rack dead weight load.
9	FR_v	Reaction load of fuel racks during 1g vertical earthquake.
10	FR_{ew}	Reaction load of fuel racks during 1g east-west earthquake.
11	FR_{ns}	Reaction load of fuel racks during 1g north-south earthquake.
12	C	Vertical crane reaction forces.

NOTE: (1) Includes effects of thermal moment on the east foundation wall due to 28° thermal gradient.

(2) Includes effect of pool hydrodynamic load, pool wall, horizontal inertial forces and building seismic response.

TABLE 6-2
ARKANSAS NUCLEAR ONE
SPENT FUEL STORAGE FACILITY STRUCTURAL EVALUATION
SUMMARY OF COMPOSITE LOADS

COMPOSITE LOADS	LOAD FACTORS												DESCRIPTION
	D_c	H	F	T_o	T_a	E_{ew}	E_{ns}	D_{fr}	FR_v	FR_{ew}	FR_{ns}	$C^{(1)}$	
D	1.0	1.0	-	-	-	-	-	-	-	-	-	-	Dead Load
L	-	-	-	-	-	-	-	1.0	-	-	-	1.0	Live Load
T_o	-	-	-	1.0	-	-		-	-	-	-	-	Operating Thermal Load
T_a	-	-	-	-	1.0	-	-	-	-	-	-	-	Accident Thermal Load
E_1	0.067	0.067	-	-	-	0.1	0.1	-	0.067	0.1	0.1	-	Loads Generated by
E_2	0.067	0.067	-	-	-	0.1	-0.1	-	0.067	0.1	-0.1	-	.1g Operating
E_3	-0.067	-0.067	-	-	-	0.1	0.1	-	-	0.1	0.1	-	Basis Earthquake
E_4	-0.067	-0.067	-	-	-	0.1	-0.1	-	-	0.1	-0.1	-	
F	-	-	1.0	-	-	-	-	-	-	-	-	-	Flood Load

(1) Crane load will be considered only if it is determined to contribute to live load in a conservative manner.

TABLE 6-3
ARKANSAS NUCLEAR ONE
SPENT FUEL STORAGE FACILITY STRUCTURAL EVALUATION
LOAD COMBINATION SUMMARY TABLE

No	Load Combination	Reference[1]
1	$1.4D + 1.7L + 1.9E$	Load Case 2
2	$.75\,(1.4D + 1.7L + 1.7T_o)$	Load Case 4
3	$.75\,(1.4D + 1.7L + 1.7T_o + 1.9E)$	Load Case 5
4	$D + L + T_o + E'$	Load Case a
5	$D + L + T_o + F$	Load Case b
6	$D + L + T_A$	Load Case c
7	$D + L + T_a + 1.25E'$	Load Case d

NOTES: (1) Reference (2) Section 3.8.4.

(2) E' = Represents a load, generated by .20g safe shutdown earthquake (SSE).
$E' = 2.0\,E$

Structural Dynamics Technology

7.0 MATERIAL CONSIDERATIONS

Construction materials conform to the requirements of ASME B and PV Code, Section III, Subsection NF. All the materials used in the 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 304 stainless steel.

The neutron absorbing material, Boraflex, used in the 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 homogenous, stable matrix.

Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material.[15] 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.[16]

Long term borated water soak tests at high temperatures were also conducted.[17] 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 CO_2.

The actual tests verify 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.

8.0 INSTALLATION CONSIDERATIONS

Installation of the proposed spent fuel storage racks is scheduled to begin March, 1983. All residing spent fuel assemblies will remain in the pools, while the spent fuel rack modules are replaced. This will be accomplished by shuffling the spent fuel assemblies to one area of the pool and replacing the empty modules with the new modules, then moving the spent fuel assemblies into the new rack modules and replacing the remaining empty rack modules.

8.1 RACK MODULE ASSEMBLY HANDLING CONSIDERATIONS

The overhead cranes in the auxiliary building at ANO will be used for removing the existing rack modules and lowering the new modules into the pool. No loads exceeding 2000 pounds will be allowed over the stored fuel assemblies at any time. All ANO fuel handling specifications and procedures will be observed at all times.

8.2 RADIATION PROTECTION CONSIDERATIONS

A complete review of all removal and installation procedures will be performed by the ANO Health Physics and ALARA committee as outlined in Section 9.0. All radiation protection measures will be followed at all times to insure minimum occupational exposures. Based upon ANO's experience with similar projects, a collective dose of 16 person-rem is estimated in replacing the ANO racks in Unit-1 and Unit-2.

9.0 RADIOLOGICAL CONSIDERATIONS

9.1 GENERAL DESCRIPTION OF RADIOLOGICAL ASPECTS

The radiological safety of the project is the responsibility of the ANO Health Physics Superintendent. He is assisted by his staff with support from the Corporate Health Physicist. The gamma radiation levels in the spent fuel pool area are monitored continuously by the area radiation monitoring systems. Additionally, radiation and contamination surveys are conducted in the spent fuel pool area on a weekly basis. In areas where a potential exists for significant airborne radioactivity, breathing zone air samples are taken and appropriate respiratory protective equipment is worn. Personnel working in radiologically controlled areas are required to wear protective clothing as specified by the applicable Radiation Work Permit (RWP). Contamination control measures are used to prevent the spread of contamination and to protect personnel from internal exposure from radioactive material. Risk level zones are established and posted. Movement of material, equipment, and personnel from a high risk zone to a lower risk zone requires specific contamination control measures to minimize the spread of contamination across boundaries.

Personnel monitoring devices are worn by all personnel working in the radiologically controlled area. The minimum requirement is a self-reading dosimeter and a thermoluminescent dosimeter (TLD).

Additional monitoring of the underwater divers will be by multiple whole body TLD's and extremity TLD's.

9.1.1 ALARA Considerations

The work to be performed will be reviewed by the ALARA committee, which reviews all work in radiologically controlled areas when the estimated collective dose for the job exceeds 1 person-rem. The committee will review the radiological survey data, job scope, steps to be taken to avoid unnecessary exposure, temporary shielding, and changes which may occur during the work which may require additional radiological safety practices.

9.1.2 Underwater Radiation Surveys

Underwater radiation surveys will be performed in all areas where divers must work or have the need for access to the work area. An underwater radiation monitoring instrument and/or thermoluminescent dosimeters will be used, when applicable, to perform dose rate measurements in the pool.

9.1.3 Diving Operations

Prior to diving operations, the spent fuel assemblies stored in the pool will be rearranged in such an array as to provide the lowest practicable dose rates to divers while minimizing the amount of rearrangement of the spent fuel. These two criteria should lessen the effects of radiation from both direct radiation from the fuel and from contaminated particles in the water that may be stirred up by fuel movement. Underwater work and access areas will be established for divers to ensure that exposures received are maintained ALARA. Health Physics Technicians will provide continuous coverage when divers are in the water. Their duties will be to provide health physics support, minimize personnel exposure, and enforce good radiation work practices and compliance with RWP requirements. The Health Physics Technicians and the diving supervisor will be in direct communication with the divers and will continually observe the divers while they are in the pool and warn them if they approach high radiation/exclusion zones. The divers will not be permitted to exceed dose rates of 1 rem/hour whole body.

Divers will wear protective clothing items inside their rubber diving suits to protect them from contamination when they remove the diving suits and exit the controlled area. TLD's will be worn inside the diving suits on the head, chest, back, legs, and extremities. Self reading dosimeters (SRD) will be sealed in plastic bags and worn inside the diving suits. The SRD's will be read and recorded after each dive. The TLD's will be read daily and a tabulation of each divers cumulative whole body and extremity doses will be prepared and reviewed by the diving supervisor and the cognizant Health Physics

Supervisor. This information will be used to maintain exposures within administrative limits and to allocate the exposure evenly among the divers.

9.1.4 Spent Fuel Pool Decontamination and Clean-up

The Spent Fuel Pool Purification Systems provide purification, clarification, and decontamination of pool water by recirculation through filters and a demineralizer. During the re-rack project the pool water will be sampled weekly to determine the concentrations of radionuclides in the pool. A portable filtered water vacuum system will be used as necessary to remove loosely deposited contamination from the fuel rack surfaces, pool floor, and walls near diver working areas to reduce the radiation exposures to the divers.

9.1.4.1 Solid Waste Disposal

The solid waste generated from the project will be in the form of demineralizer resins, filters, compactable trash, and non-compactable waste. These wastes will be disposed through normal plant methods for shipment to a licensed waste burial facility.

9.1.4.2 Decontamination of Removed Racks

The rack sections will be rinsed with a low pressure spray of demineralized water or spent fuel pool water as they are removed from the pool. If further decontamination is necessary the racks will be removed to an area with adequate containment to allow the use of hydrolasing equipment. Personnel performing decontamination on the racks will wear appropriate protective clothing and respiratory protection equipment. After decontamination, the racks will be moved to another area to be packed for disposal. The decontamination operations are expected to remove significant quantities of loose contamination while causing a relatively low exposure to the decontamination workers, but will reduce subsequent personnel exposures to workers packaging, handling, and shipping the rack sections.

9.1.5 Disposition of Old Racks

Four options are available for the old racks, as follows:

1. Burial without volume reduction.

2. Burial with volume reduction.

3. Decontaminated to releasable criteria of Regulatory Guide 1.86 and disposal.

4. Transfer to a licensed facility for decontamination and/or disposal.

An evaluation will be made for disposal of the racks based upon the contamination levels when removed, person-rem exposures estimated for decontamination, person-rem exposures estimated for volume reduction, and cost/benefit factors for each of the options. Ultimate disposal will be one of the four options listed after the evaluations have been performed.

9.1.6 Current Radiological Data From Spent Fuel Pools

The most recent radiological survey data from the spent fuel pool area indicates the following parameters:

DOSE RATES:

Unit I Fuel Transfer Bridge 4 mR/hr

Unit I Side of Pool 1 mR/hr

Unit II Fuel Transfer Bridge 4 mR/hr

Unit II Side of Pool 1 mR/hr

AIRBORNE RADIOACTIVITY:

Gross Particulate < 1.0 E-9 uCi/mL

Gross Iodine < 1.0 E-9 uCi/mL

Isotopic analysis not performed on air samples less than 1.0 E-9 uCi/mL.

Spent Fuel Pool Water. (July 1, 1982)

Isotope	uCi/mL
Cs-137	2.37 E-2
Cs-134	9.91 E-3
Co-60	8.73 E-4
Co-58	2.09 E-4
Zr-95	6.71 E-6
Sb-125	6.12 E-5
TOTAL	3.42 E-2

9.2 DESCRIPTION OF RADIOLOGICAL ASPECTS OF INCREASED SPENT FUEL STORAGE

As of July 1, 1982, 244 assemblies are stored in ANO-1 and 60 assemblies are stored in ANO-2 spent fuel pools, respectively. The current radiological data relating to the spent fuel pools are shown in Section 9.1.6.

Currently solid wastes are collected from the spent fuel pools and concentrated in two areas: (1) The spent fuel pool ion exchangers which remove ionic material from the pool water and (2) The particulate filters which remove particulates larger than 5 microns. The ANO-1 ion exchanger resin volume is approximately 20 ft^3 and the ANO-2 resin volume is approximately 32 ft^3 (useful). Both are currently changed about once per year. Both ANO-1 and ANO-2 particulate filter systems contain two filters which are changed approximately once per year.

9.2.1 Solid Waste Generation

The ANO-1 spent fuel pool demineralizer (ion exchanger) resin volume of approximately 20 ft^3 represents <10% of the total ANO-1 annual solid waste in the form of spent resins. ANO-2 spent fuel pool ion exchanger volume is approximately 32 ft^3 (useful) and represents about 10% of the total ANO-2 annual solid waste in the form of spent resin.

These resins are designed to be changed based on an increase in differential pressure rather than on the lack of ability to remove radioactive ionic material. Neither the frequency of fuel addition nor the annual amount of fuel to be added to the spent fuel pool will be changed due to the rack modifications. Therefore, the annual amounts of contaminants added to the pools are not expected to increase significantly. Since the resins are designed to be changed annually, and the annual amount of contaminants is not expected to increase, no appreciable increase in solid waste in the form of spent fuel pool ion exchanger resins is expected.

Similarly, the particulate filters are designed to be changed annually, based on differential pressure. As in the case of the ion exchanger resins, no significant change in the frequency of replacement due to the modified racks is expected.

The number of ANO-1 storage locations was increased in 1976, and no significant increase in radioactive solid waste generation in the form of resins or filters resulted.

9.2.2 Airborne Activity

The design of ANO-1 does not permit the measurement of radioactive gases released from individual ventilation systems, but data are available for overall plant releases. As shown in the ANO-1 FSAR Section 9.6.2.4.2, the data related to the 233 percent increase in fuel storage capacity in 1976 do not indicate an appreciable increase in the Kr-85 release rate. However, as a method of estimation, the FSAR analysis assumed a 233 percent increase in the annual Kr-85 release to predict a maximum release rate. The results indicate that even with this conservative treatment, the increase of total plant Kr-85 annual release would be less than 1.13 percent.

Since spent fuel storage activities are to remain basically unchanged, there are no data to predict nor reason to expect results different from those realized from the earlier modifications, i.e., no appreciable increase in release rate in either unit.

9.2.3 Personnel Exposure

The radionuclide concentrations in the spent fuel pools as presented in the applicable FSARs are based on one percent failed fuel and a crud burst model. Exposures based on this are presented in ANO-1 FSAR, Section 9.6.2.3. No significant increase in personnel doses was made due to the increase in storage capacity. As stated in Section 9.6.2.4.2 of the ANO-1 FSAR, "Fuel being transferred is the controlling contributor to the basic dose rates, not the stored fuel."

In regard to personnel exposure received during filter and resin changes, based on experience it is estimated that the annual exposure is <0.2 man-rem/unit. This annual exposure is not expected to increase, since the frequency of change is not expected to be altered.

10.0 ACCIDENT EVALUATIONS

The following analyses are related to postulated accidents identified in the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

10.1 CASK DROP ACCIDENT ANALYSIS

Arkansas Nuclear One Unit 1 and Unit 2 administrative procedures prevent the spent fuel cask from being moved over the spent fuel pools; hence, the cask drop accident is not credible.

10.2 FUEL ASSEMBLY DROP ACCIDENT ANALYSIS

A fuel assembly drop accident analysis is also performed to ensure that, in the unlikely event of dropping a fuel assembly, accidental deformation to the rack will not cause the criticality acceptance criteria to be violated, and that the spent fuel pool liner will not be perforated.

10.3 LOSS OF SPENT FUEL POOL COOLING

Under postulated accident conditions where all non-Category 1 spent fuel pool makeup systems become inoperative, alternate Seismic Class 1 methods for makeup to the spent fuel pool water are available in each unit. Section 9.6.1.3 of the ANO-1 FSAR describes these methods for Unit 1, and Section 9.1.3.3.1 of the ANO-2 FSAR details the methods for Unit 2.

Although it is highly unlikely that a complete loss of cooling capability could occur, the modified storage conditions are analyzed to this condition.

Basis:

a. No pool cooling implies that temperature of water at inlet to spent fuel racks is 212°F, which corresponds to the saturation temperature at the pool surface.

b. The nominal water level above the top of the racks is maintained.

c. Maximum fuel loading cases of Table 5-1 for Unit 1 and Table 5-2 for Unit 2 are assumed.

d. The assemblies that are evaluated are initially placed into the pool at seven days after shutdown.

e. The peak rods are assumed to have 60% greater heat output than average rods.

f. All storage cells are filled and all downflow occurs in the peripheral gaps.

The criterion for this condition is that adequate cooling must exist to maintain the temperature of the fuel cladding sufficiently low to prevent structural failures and safety concerns.

11.0 COST AND BENEFIT ASSESSMENT

Arkansas Nuclear One's present spent fuel rack capacities are 589 spaces for Unit-1 and 485 spaces for Unit-2. As Figure 11-1 and 11-2 indicate, ANO-1 and ANO-2 will lose full core discharge capacities in 1986 with these conditions. Currently, ANO does not have contracts or options for the use of any "away from reactor storage" or reprocessing facilities, nor do we expect to have these options available within the next 10 years.

11.1 ALTERNATIVES

Without the increased storage capacity provided by reracking the spent fuel pools, Arkansas Power and Light would be forced to shutdown Unit-1 and Unit-2 in 1989 due to the inability to refuel the units. This would cause a significant increase in electrical power cost to the citizens served by Arkansas Power & Light due to the cost of replacement power.

11.2 COMMITMENT OF MATERIAL RESOURCES

The proposed replacement rack modules will be fabricated from type 304 stainless steel material with these modules in Region 1 utilizing the neutron absorbing material, Boraflex. A total amount of approximately 347,000 pounds of stainless steel and approximately 9,600 pounds of Boraflex is expected to be used. The use of these materials will have no significant affect on the availability of these material resources.

Figure 11-1

SPENT FUEL STORAGE AT ANO - UNIT 1

REFUELING TIME	REFUELING NUMBER	DISCHARGED ASSEMBLIES	TOTAL IN POOL	SPACES REMAINING	
Spring 1977	1	56	56	533	
Spring 1978	2	56	112	477	
Spring 1979	3	64	176	413	
Spring 1981	4	68	244	345	
Fall 1982	5	72	316	273	
Fall 1984	6	68	384	205	*584
Spring 1986 (1)	7	64	448	141	*520
Fall 1987	8	64	512	77	*456
Spring 1989	9	64	576	13	*392
Fall 1990	10	64	640		*328
Spring 1992	11	64	704		*264
Fall 1993	12	64	768		*200
Spring 1995 (2)	13	64	832		*136
Fall 1996	14	64	896		* 72
Spring 1998	15	64	960		* 8

NOTE: * (Reracked)

(1) Full core discharge capability is lost with the present capacity of 589 spaces. (Core = 177 Assemblies)

(2) Full core discharge capability is lost assuming a reracked capacity of 968 spaces.

Figure 11-2

SPENT FUEL STORAGE AT ANO - UNIT 2

REFUELING TIME	REFUELING NUMBER	DISCHARGED ASSEMBLIES	TOTAL IN POOL	SPACES REMAINING	
Spring 1981	1	60	60	425	
Fall 1982	2	52	112	373	
Fall 1983	3	68	180	305	*808
Spring 1985	4	68	248	237	*740
Fall 1986 (1)	5	68	316	169	*672
Spring 1988	6	68	384	101	*604
Fall 1989	7	68	452	33	*536
Spring 1991	8	68	520		*468
Fall 1992	9	68	588		*400
Spring 1994	10	68	656		*332
Fall 1995	11	68	724		*264
Spring 1997	12	68	792		*196
Fall 1998 (2)	13	68	860		*128
Spring 2000	14	68	928		* 60

NOTE: * (Reracked)

(1) Full core discharge capability is lost with the present capacity of 485 spaces. (Core = 177 Assemblies)

(2) Full core discharge capability is lost assuming a reracked capacity of 988 spaces.

12.0 REFERENCES

1. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B.K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

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

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

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

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

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

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

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

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

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

11. Nodvik, R. J., "Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radiochemical Analysis of Irradiated Saxton Plutonium Fuel," July 1970.

12. Strawbridge, L. E. and Barry, R. F., "Criticality Calculation for Uniform Water-Moderated Lattices," Nuclear Science and Engineering 23. 58 (1965).

13. WECAN - "Documentation of Selected Westinghouse Structural Analyses Computer Codes", WCAP-8252.

14. WECAN - "Benchmark Problem Solution Employed for Verification of the WECAN Computer Program", WCAP-8929.

15. J.S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Data," Brand Industries, Inc., Report 748-30-1, (August 1979).

16. J.S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, (July 1979).

17. 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 748-2-1, (August 1978).