DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION REVISION

(To be submitted at a later date by Duke Power Company)

8007090276

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

UNITS 1 AND 2 SPENT FUEL POOL

POISON RERACK LICENSING SUBMITTAL

July, 1980

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1.0 INTRODUCTION

The Oconee Nuclear Station was designed and constructed with two spent fuel storage pools-one associated with Units 1 and 2 and one with Unit 3. The design was such that the pools would be capable of storing 1 2/3 and 1 1/3 cores respectively. Both the Oconee Nuclear Station Final Safety Analysis Report and Technical Specifications address the adequacy of this to "accommodate a full core of irradiated fuel assemblies in addition to concurrent storage of the largest quantity of new and spent fuel assemblies predicated by the fuel management program" (References 1 and 2). The actual designed capacity for each pool was 336 and 216 locations.

In 1975 it was deemed prudent to increase the storage capacity at the Oconee site. The Unit 1 and 2 pool contained spent fuel from the initial Unit 1 refueling in 1974. The Unit 3 pool was empty of any spent fuel. Thus, it was decided to increase the capacity of the Unit 3 pool. A request to amend the Unit 3 Operating License, DPR-55, was submitted on September 12, 1975 and was approved, as Amendment No. 17, on December 22, 1975. The completed modification increased its capacity to 474 locations (including failed fuel). It was considered at that time that the resulting combined on-site capacity (810 locations) would be sufficient to store spent fuel until such time as shipment to the Allied General Nuclear Services reprocessing plant could begin. The modification of the Unit 1 and 2 pool in this time frame would have had to have been done "wet" (with spent fuel present in pool). Such operations were considered to exceed state-of-the-art technical capabilities at that time.

On April 17, 1977, President Carter issued a policy statement on commercial reprocessing of spent nuclear fuel which effectively eliminated reprocessing as an end to the nuclear fuel cycle, at least in the near future. On Ocotober 18, 1977, the Department of Energy accepted ultimate responsibility for storing spent nuclear fuel. On December 23, 1977, the GESMO proceedings were deferred indefinitely. The combined effect of this national policy was to leave operating nuclear plants, like Oconee, without a repository for the spent fuel previously generated or being generated, other than expanded storage provided by the owner/operator. Two plausible alternatives are left for Oconee: (a) ship the spent fuel to other Company nuclear plants; or (b) expand onsite storage pool.

By letter dated March 9, 1978, Duke Power Company requested approval of option (a) using McGuire Nuclear Station as a temporary storage site for some Oconee fuel until another site can be identified or constructed. Proceedings in this are ongoing.

By letter dated February 2, 1980, Duke requested approval of expanding the capacity of the Unit 1, 2 pool utilizing high-capacity non-power rack. The expansion of the Unit 1 and 2 pool capacity as approved, allowed the storage of up to 750 assemblies in that pool and 1224 on-site (including "failed-fuel" locations).

The continued delays in the McGuire proceedings, as well as the lack of national policy decisions which would help ameliorate the situation at Oconee, have forced Duke Power to once again rerack the Oconee pools with poison racks providing further storage capacity. Therefore, Duke is hereby requesting approval for

using Westinghouse designed/constructed poison racks in the Oconee 1 and 2 pool. The expanded storage capacity (to 1212 spaces) will allow storage of 462 additional assemblies.

The following chapters are provided with intent to provide information necessary for review and approval of the request for amendment to the Oconee Nuclear Station Technical Specifications (Attachment 1). It is considered that the modification is not inimical to the health and safety of personnel or the general public and that it represents an environmentally acceptable alternative which meets the requirements of NEPA and the guidance provided by the Commission on such applications (References 5 and 6).

References

1. Oconee Nuclear Station Technical Specification Section 5.4.2.1
2. Oconee Nuclear Station Final Safety Analysis Report Section 9.7.1.3
3. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications"
4. September 16, 1975 Federal Register Notice (FR-42801)

2.0 RACK DESIGN

2.1 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.

A list of design criteria is given below:

a. The racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 and revised January 18, 1979.

b. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor, K_{eff}, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions as described in section 2.3.2.

c. The racks are designed to allow coolant flow such that boiling in the water channels between fuel assemblies does not occur. Maximum fuel cladding temperatures are calculated for various pool cooling conditions as described in section 2.3.3.

d. The racks are designed to Seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support structures. The structural evaluation and seismic analyses are performed using the specified loads and load combinations in section 2.1.1.

e. 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.

f. 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.

g. The racks are designed to preclude the insertion of a fuel assembly in other than design locations.

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

2.1.1 SPECIFIED LOADS AND DEFINITIONS

The following are load combinations specified for racks:

ELASTIC ANALYSIS	ACCEPTANCE LIMITS
(1) D + L	Normal Limits of NF 3231.1a
(2) D + L	Normal Limits of NF 3231.1a
(3) D + L + 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.

T_o - Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

T_a - Thermal effects and loads resulting from highest temperature associated with the postulated abnormal design condition.

E - Loads generated by the operating basis earthquake.

E' - Loads generated by the safe shutdown earthquake.

Analyses were performed to ascertain the acceptability of the critical load components and paths under the load combinations given above.

2.1.2 APPLICABLE CODES AND STANDARDS

"NRC Position For Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and revised January 18, 1979.

NRC Regulatory Guides

R.G. 1.13 Spent Fuel Storage Facility Design Basis

R.G. 1.29 Seismic Design Classifications

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

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

R.G. 1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis
R.G. 1.124	Service Limits and Loading Combinations for Class I Linear-Type Component Supports

NRC Standard Review Plan

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 Cooling and Cleanup System

Industry Codes and Standards

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

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 and Fissionable Materials Outside Reactors."

2.2 DESIGN DESCRIPTION

The spent fuel storage rack is composed of individual storage cells made of stainless steel. Each cell has a lead-in opening which is symmetrical and is blended smooth. This opening precludes insertion of the fuel assemblies in other than prescribed locations. These racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. The cells within a module are interconnected to form an integral structure as shown in Figure 2-1. 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 following information applies to Oconee Units 1 and 2 which share a common spent fuel pool.

Number of Cells	1312
Number Rack Arrays	4 - 8 x 11 10 - 8 x 12
Poison Material	Boraflex 0.02 gm $^{10}B/cm^2$ Vented to pool environment

Center-to-Center Spacing	10.65 in.
Type of Fuel	B&W 15 x 15, 4.3 weight percent enrichment (maximum)
Rack Assembly Dimension & Weights	8 x 11 - 85.5 x 117 x 176 - 24,200 lbs. 8 x 12 - 85.5 x 128 x 176 - 26,400 lbs.

The pool outline and rack arrangement is shown in Figure 2-2.

2.2.1 DESIGN LOADS

D - Weight of 8 x 11 rack dry . . . . . 24,200 lbs.
Weight of 8 x 12 rack dry . . . . . 26,400 lbs.

L - Live loads are negligible since the fuel assemblies are lowered very slowly into the cells.

T_o - Service expansion temperature range $\Delta T = 20°F$

T_a - A postulated design condition that would cause T_a is failure of the spent fuel pool cooling system. The water will gradually heat up and boiling could theoretically occur; however, since this process is slow, it is predicted to remain in the T_o ΔT range.

E - OBE loads

E' - SSE loads

2.3 DESIGN EVALUATION

Evaluations and anlayses were performed in the following areas to verify the ability of the rack design to perform its required functions.

1. Structural and Seismic
2. Nuclear Criticality
3. Thermal-Hydraulic
4. Poison Material

2.3.1 STRUCTURAL AND SEISMIC

The purpose of the structural analysis is to analyze the critical components/load paths under various loading conditions. The structural analysis also determines the margin of safety against overturning due to loads from an SSE. 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. Sufficient clearance is also provided to prevent racks from sliding into the pool clip angles. Figure 2-3 shows the general arrangement of a typical fuel rack assembly and its pool floor leveling pad.

2.3.1.1 Component Description

The complete fuel rack assembly is divided into three major sections for stress analysis purposes:

a. Leveling pad assembly
b. Lower and upper grid assembly
c. Cell assembly

The following paragraphs describe each assembly.

Leveling Pad Assembly

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

Lower and Upper Grid Assembly

The lower grid attaches the cell assembly to the support plate. The lower grid consists of box-beam members, the spacer plates and support plate. The cell assembly at bottom is welded to the lower grid through the spacer plate. The upper grid consists of the box-beam members and the spacer plates. The cell assembly at top is welded to the upper grid through the spacer plate. The upper and lower grid assembly maintains the precise center line to center line spacing between the cells and provides the structural connections between the cells to form a fuel rack assembly.

Cell Assembly

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

The ID of the cell is 8.965 with a 0.075 inch wall. The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. The wrapper is attached to the outside of the cell through a seam welding along the entire length of the wrapper. Thus, the wrapper surrounds the Boraflex material, and also provides for venting to the pool environment. The spacer plates are welded to the upper and lower grid assembly, thus providing a structural connection between the cell and the grid assembly.

Depending upon the criticality requirements, some cells have a Boraflex wrapper on all four sides, some on three sides and some on two sides.

2.3.1.2 Seismic Analysis Models

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 and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. The second phase is a response spectrum analysis of a detail rack assembly finite element model shown in Figure 2-4. The damping values used in the seismic analysis are two percent damping for OBE and four percent damping for SSE as specified in RG 1.61.

The simplified nonlinear finite element model is used to determine the fuel rack response for full, partially filled and empty fuel assembly loading conditions. This nonlinear model has the structural characteristics of an individual cell within a submerged rack assembly. The nonlinearities of the fuel rack assembly which are accounted for in the model are due to changes in the gap between the fuel cell and fuel assembly, the balancing conditions of the fuel rack support locations and energy losses at the support locations. The WECAN computer program was used to determine the nonlinear time history response of the fuel assembly/fuel rack system. The effective fuel mass, fuel assembly to cell impact loads, and overall rack loading are obtained from the time history response of these results.

The detail model is a three-dimensional finite element representative of an 8 x 12 rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of nodal points.

The results of the single cell nonlinear time history model are incorporated in the detail model. Since the detail model does not account for the nonlinear effect of a fuel assembly impacting the cell and the hydrodynamic restoring force, 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 single cell nonlinear model results and is applied to the components in the structural analysis. The responses of the model from accelerations in three directions are combined by the SRSS method in the structural analysis. The loads in four major components (support pad assembly, bottom grid, top grid, and fuel cell) are examined, and the maximum loaded section of each of these components was found. These maximum loads from the detail model are used in the structural analysis to obtain the stresses within the rack assembly.

2.3.1.3 Loads and Load Combinations for Structural Analysis

The loads and load combinations to be considered are those given in NRC Standard Review Plan, Section 3.8.4-II.3. 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 uni-direction loads by the square root of 2.

The grid assembly margins of safety for the multi-direction seismic event are produced by combination x-direction and y-direction loads of SRSS.

The margins of safety, due to the multi-direction shock for the leveling pad assembly, are the same as the uni-direction because maximum stresses, due to x-direction and y-direction seismic acceleration, do not occur at the same point. The multi-direction seismic event stresses for the weld of the leveling pad assembly are properly corrected.

The loads summarized in the seismic analysis section are corrected by load correction factors obtained from the nonlinear analysis.

2.3.1.4 Fuel Handling Crane Uplift Analysis

The objective of this analysis is to ensure that the rack can withstand the maximum uplift load of 3000 lbs. of the fuel handling crane without violating the criticality acceptance criteria.

Two accident loading conditions are postulated. The first condition assumes that the uplift load is applied to a fuel cell. The second condition assumes that the load is applied to the top grid. Calculations show that for either condition, the resulting stresses are within acceptable stress limits. There is no change in rack geometry and the criticality acceptance criteria is not violated.

2.3.1.5 Fuel Assembly Drop Accident Analysis

The objectives of this analysis are to ensure that, in the unlikely event of dropping a fuel assembly, accidental deformation to the

rack will not cause the criticality criteria to be violated, and the spent fuel pool liner will not be perforated.

Two accident conditions are postulated. The first accident condition assumes that the weight of a fuel assembly, control rod assembly and handling mechanism (3,000 lbs) impacts on the top of the rack. Calculations show that the impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cell funnels and the section of cell above the upper grid structure and the rack base plate/lower grid assembly. If in the unlikely event that two adjacent cells are crushed together for their full length, criticality calculations show that $K_{eff} < 0.95$. Under these faulted conditions, credit is taken for dissolved boron in the water, and criticality acceptance criteria is not violated for the Oconee poison spent fuel racks. The pad liner is not perforated. A radiological evaluation is provided in Section 6.3.

The second accident condition assumes that the fuel assembly falls straight through an empty cell and impacts the rack base plate from a drop height of 234 inches. 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 will not be perforated and the margin of safety is positive. Critically calculations show that $K_{eff} < 0.95$ and the criticality acceptance criteria is not violated for the Oconee posion spent fuel racks.

In both these accident conditions, the criticality acceptance criteria is not violated and the spent fuel pool liner is not perforated.

2.3.1.6 Structural Acceptance Criteria

The fuel racks are analyzed for the normal and faulted load combinations of Section 2.1.1 in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The major normal and upset condition 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 allowable 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 analysis show that the Oconee spent fuel storage racks meet all the structural acceptance criteria adequately.

2.3.2 NUCLEAR CRITICALITY

2.3.2.1 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 poisons 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 that 0.95 as recommended in ANSI N210-1976 and in "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

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

2.3.2.2 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 enrichment of the fuel assembly is 4.3 w/o U-235 with no depletion or fission product buildup.

b. 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.

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, poison plates are not necessary on the periphery of the rack and on one side of the long axis between modules because calculations show that this finite rack is less reactive than the normal case infinite rack. Therefore, the nominal case of an infinite array of poison cells is a conservative assumption.

d. Mechanical uncertainties are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties. The uncertainties included in the analysis are:

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

The calculational method uncertainty is discussed in Section 2.3.2.4.

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 self shielding is included as a bias in the reactivity calculations.

2.3.2.3 Postulated Accidents

Most accident conditions will not result in an increase 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 dropped assembly has more than eight inches of water separating it from the rest of the rack which precludes interaction).

However, accidents can be postulated which would increase reactivity. Therefore, for accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that one is not required to assume protection against 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 is assumed as a realistic initial condition, since not assuming its presence would be a second unlikely event.

The presence of approximately 2000 ppm boron in the pool water will decrease reactivity by about 30 percent Δk. In perspective, this is more negative reactivity than is present in the poison plates, so K_{eff} for the rack would be less than 0.95 even if the poison plates were not present. Thus, for postulated accidents, should there be a reactivity increase, K_{eff} would be less than or equal to 0.95 due to the combined effects of the dissolved boron and the poison plates.

This "optimum moderation" accident is not a problem in spent fuel storage racks because possible water densities are too low ($\leq$ 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). Further, the presence of poison plates removed 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.

2.3.2.4 Criticality Analysis

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 insures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system of codes [1,2] for cross-section generation and KENO IV [3] for reactivity determination.

The 218 energy group cross-section library [1] 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 [2] 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 XSDRNPH program [2] which is a one-dimensional S_N transport theory code. These multi-group cross-section sets are then used as input to KENO IV [3] which is a three-dimensional Monte Carlo 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 (4,5) to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials (6) (Plexiglas, steel and air) that demonstrate the wide range of applicability of the method. (See Table 2.3-1 for summary of these experiments.)

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

The total uncertainty to be added to a criticality calculation is:

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

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

The most important effect on reactivity of the mechanical tolerances is the possible reduction in the water gap between the poison plates. The worst combination of mechanical tolerances are those

that result in the maximum reduction in the water gap. 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, assumed a worst case of a rack composed of an array of groups of four cans with the minimum water gap between the four cans. The reactivity increase of this configuration is included as a bias term in calculating the K_{eff} of the rack. It is included as a bias term since cans can be welded to a common grid during manufacturing which is the likely cause of the water gap reduction.

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

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 Pressurizer 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."

2.3.2.5 Rack Modification

The spent fuel storage rack is described in Section 2.0. The minimum ^{10}B loading in the poison plates is 0.02 gm $^{10}B/cm^2$.

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

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

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_{plate} = 95/95 uncertainty in the bias associated with the poison width reduction.

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

KS_{mech} = 95/95 uncertainty to account for thickness, spacing and bowing tolerances which are assumed to reduce the water gap between groups of four cells by 0.5 inches.

Substituting calculated values, the result is:

$K_{eff} = 0.9475$

Since this K_{eff} is less than 0.95 including uncertainties at 95/95 probability/confidence level, the acceptance criteria for criticality is met.

2.3.2.6 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.

Generally, the acceptance criteria for postulated accident conditions can be $K_{eff} \leq 0.98$ because of the accuracy of the methods used coupled with the low probability of occurrence. For instance, in ANSI N210-1976 the acceptance criteria for the "optimum moderation" condition is $K_{eff} \leq 0.98$. However, for storage pools, which contain dissolved boron, the use of realistic initial conditions ensures that $K_{eff} << 0.95$ for postulated accidents as discussed in Section 2.3.2.3. Thus, for simplicity, the acceptance criteria for all conditions will be $K_{eff} \leq 0.95$.

2.3.3 THERMAL-HYDRAULIC

The purpose of thermal-hydraulic analysis is to determine the maximum fuel clad temperatures which may occur as a result of using the poison spent fuel racks in the Oconee spent fuel pool.

2.3.3.1 Criteria

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

a. The design must allow adequate cooling by natural circulation and by flow provided by 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.

b. 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.

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

2.3.3.2 Key Assumptions

a. The nominal water level is 24 feet above the top of the fuel storage racks.

b. The maximum fuel assembly decay heat output is 7.92×10^4 watts.

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

d. 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.

2.3.3.3 Description of Analytical Method and Types of Calculations Performed

A natural circulation calculation is employed to determine the thermohydraulic 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 and the flow velocities and fluid temperatures. An elevation view of a typical model is sketched in Figure 2-5 where the flow paths are indicated by arrows. Note that each cell shown in that sketch actually corresponds to a row of cells that are located at the same distance from the pool walls. This is more clearly shown in a plan view, Figure 2-6.

As shown in that sketch, 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 ciruclation 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 flowrate which leads to conservatively large pressure drops in the downcomer and lateral flow regions which reduces the driving pressure drop across the limiting storage locations.

Since the natural circulation velocity strongly affects the temperature rise of the water and the heat transfer coefficient within a storage cell, the hydraulic resistance experienced by the flow is a significant parameter in the evaluation. In order to minimize the resistance, the design of the inlet region of the racks has been chosen such as to maximize this flow area. Each storage cell has at least three separate flow openings as shown in Figure 2-7. The use of these 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 cells be evaluated. In order to preclude the possibility of stagnant conditions in these gaps, flow relief areas are provided at the location of the grid support structures as shown in Figure 2-8. 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, poison material and cell wrapper in addition to the decay heat input. Using the cell flow velocity and driving pressure differential obtained from the previously described pool analyses, the flow velocity in the gap and the axial temperature distributions of the coolant and structure are determined. The radial temperature distributions through the various components are also considered.

2.3.3.4 Results

Normal Operation

Basis:

a. Cooling System Operational

b. Three days after shutdown-Decay Heat = 75.1 BTU/second/assembly

c. Uniform decay heat loading in pool - No credit for lower actual heat input

d. Peak Rod has 60% more heat output than average rod

e. All storage cells filled.

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 within its allowable limits. Water temperatures in the gap between cells are lower than inside the cells, and boiling does not occur in the inter-cell gaps. Although the normal water level is 24 feet above the top of the racks, a level of only 10 feet is required for a saturation temperature of 225°F which is greater than the cell outlet temperature, and no boiling occurs.

Flow Blockage Analysis

Basis:

a. Three days after shutdown

b. Temperature of water at inlet to storage racks = 150°F

Results of the analysis show that should up to 75% flow blockage occur, there would be no boiling in the water channels between the cells or in the cells. Because of the multiple flow openings that are used in the Westinghouse storage racks, it is very improbable that a complete blockage could occur.

Abnormal Condition

Under postulated accident conditions where all non-Category 1 spent fuel pool cooling systems become inoperative, there is an alternative method for cooling the spent fuel pool water. Although it is highly unlikely that a complete loss of cooling capability could occur, the racks 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 of 24 feet above the top of the racks is maintained.

c. A conservative fuel loading case is assumed. The pool is completely filled with fuel from both Oconee units based on simultaneous full core discharges at one month following a normal refueling of both units. Previous refuelings of one-third core from each unit are assumed to have occurred at one year intervals.

d. The assemblies that are evaluated are initially put into the pool at three 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 gap.

Results of this analysis show that due to the effects of natural circulation, the fuel cladding temperatures are sufficiently low to preclude structural failures. No boiling in the water channels between the fuel assemblies and within the storage cells occurs.

Since the saturation temperature is approximately 239°F and the maximum cell outlet temperature at three days after shutdown is about 236°F, boiling does not occur in the water channels between fuel assemblies. As decay heat decreases the cell outlet temperatures also continue to decrease.

2.3.4 NEUTRON ABSORBING MATERIAL

The neutron material, Boraflex, used in the Oconee 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 contains a minimum ^{10}B areal density of 0.02 gm/cm^2.

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.(7) 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.(8)

Long term borated water soak tests at high temperatures were also conducted.(9) 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 long-term material stability and mechanical integrity, and can be safely utilized as a poison material for neutron absorption in spent fuel storage racks.

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

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

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

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

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

9. 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 2.3-1

BENCHMARK CRITICAL EXPERIMENTS [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	0.993 ± .004
3.	"	"	"	"	6.39	1.005 ± .004
4.	"	"	"	"	4.46	0.994 ± .004
5.	"	"	"	stainless steel	10.44	1.005 ± .004
6.	"	"	"	"	11.47	0.992 ± .004
7.	"	"	"	"	7.76	0.992 ± .004
8.	"	"	"	"	7.42	1.004 ± .004
9.	"	"	"	boral	6.34	1.005 ± .004
10.	"	"	"	"	9.03	0.992 ± .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	0.988 ± .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	0.006 ± .003

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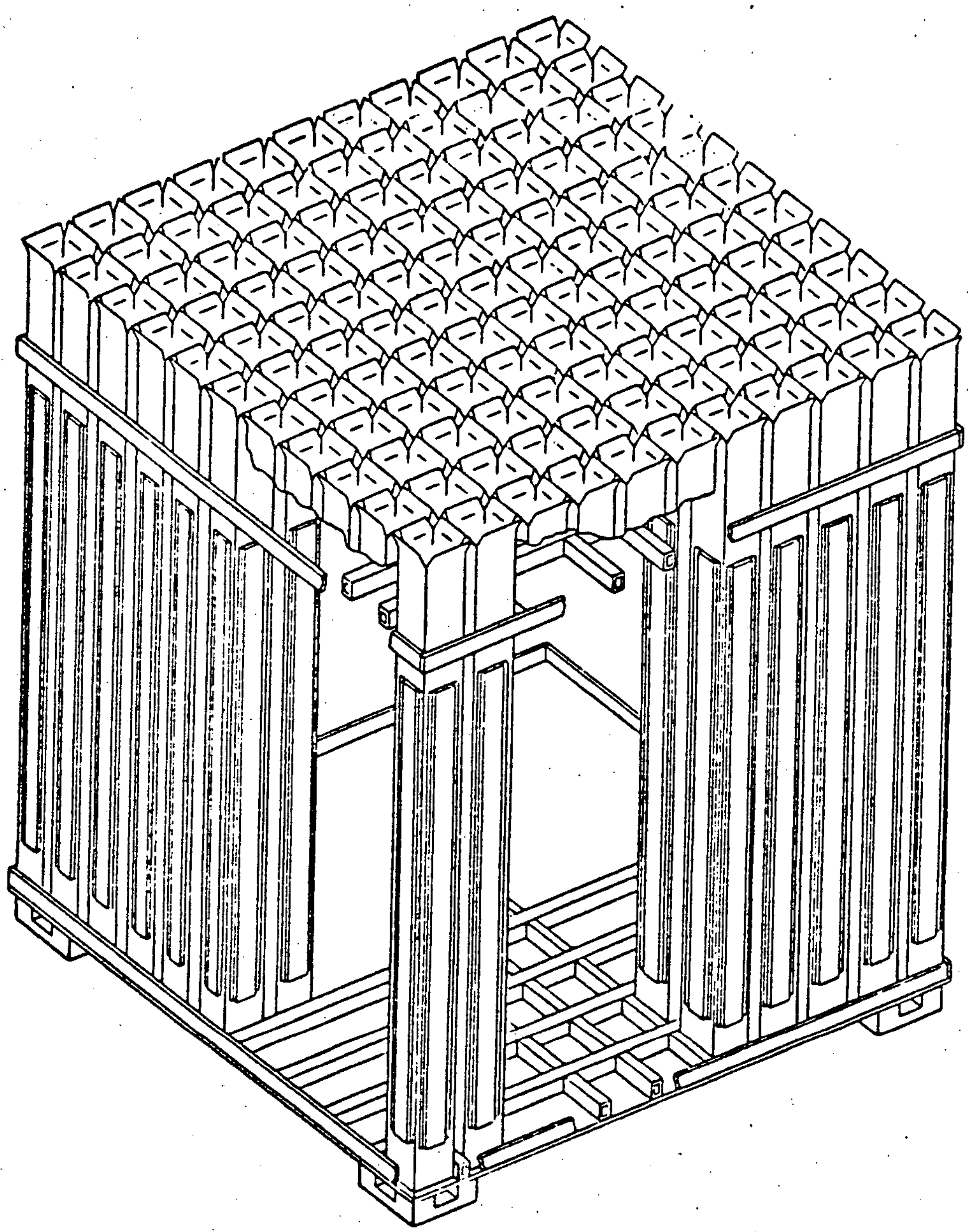
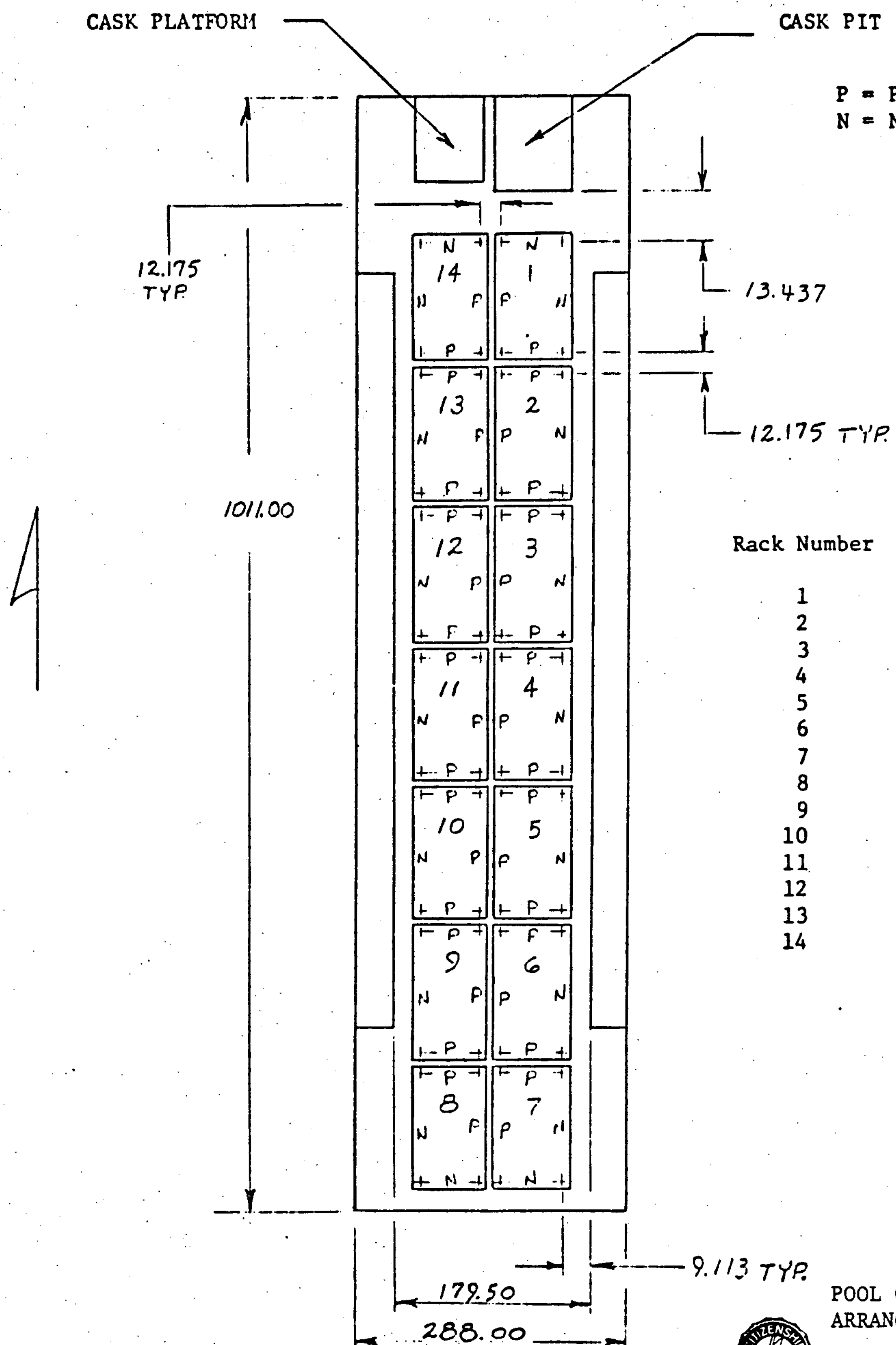


FUEL STORAGE RACK
MODULE
OCONEE NUCLEAR STATION

Figure 2-1



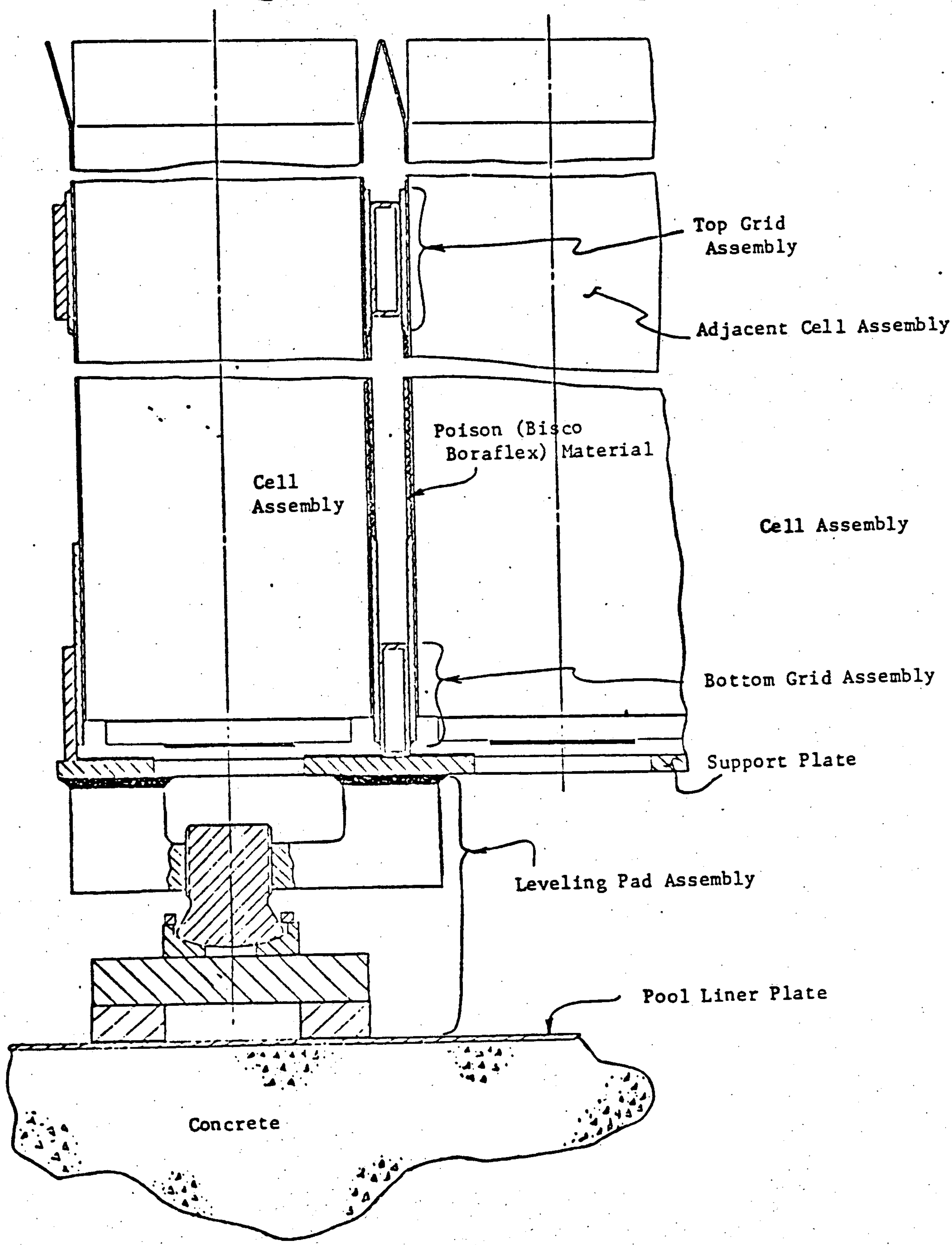
P = Peripheral Poison
N = No Peripheral Poison

Rack Number	Array
1	8 x 12
2	8 x 12
3	8 x 12
4	8 x 12
5	8 x 12
6	8 x 11
7	8 x 11
8	8 x 11
9	8 x 11
10	8 x 12
11	8 x 12
12	8 x 12
13	8 x 12
14	8 x 12

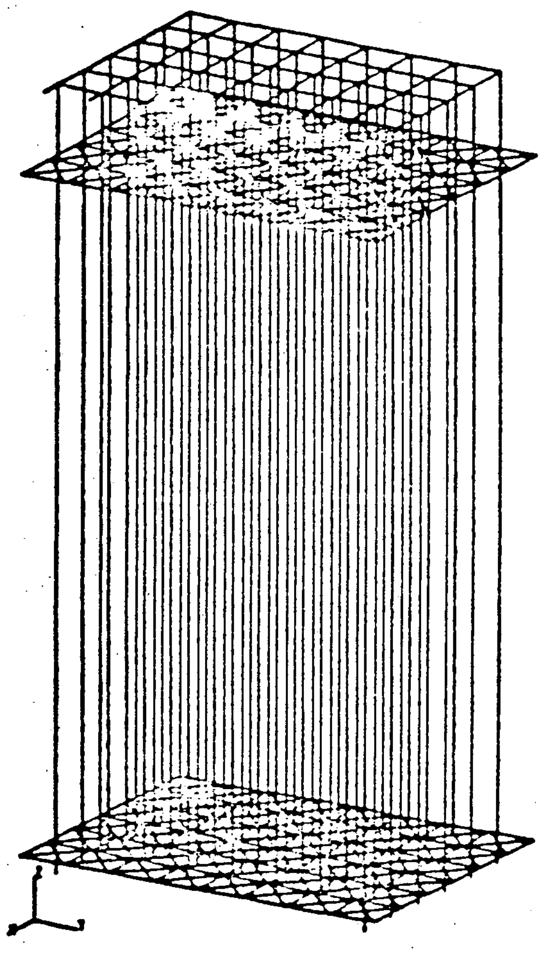
POOL OUTLINE AND RACK ARRANGEMENT

OCONEE NUCLEAR STATION

Figure 2-2



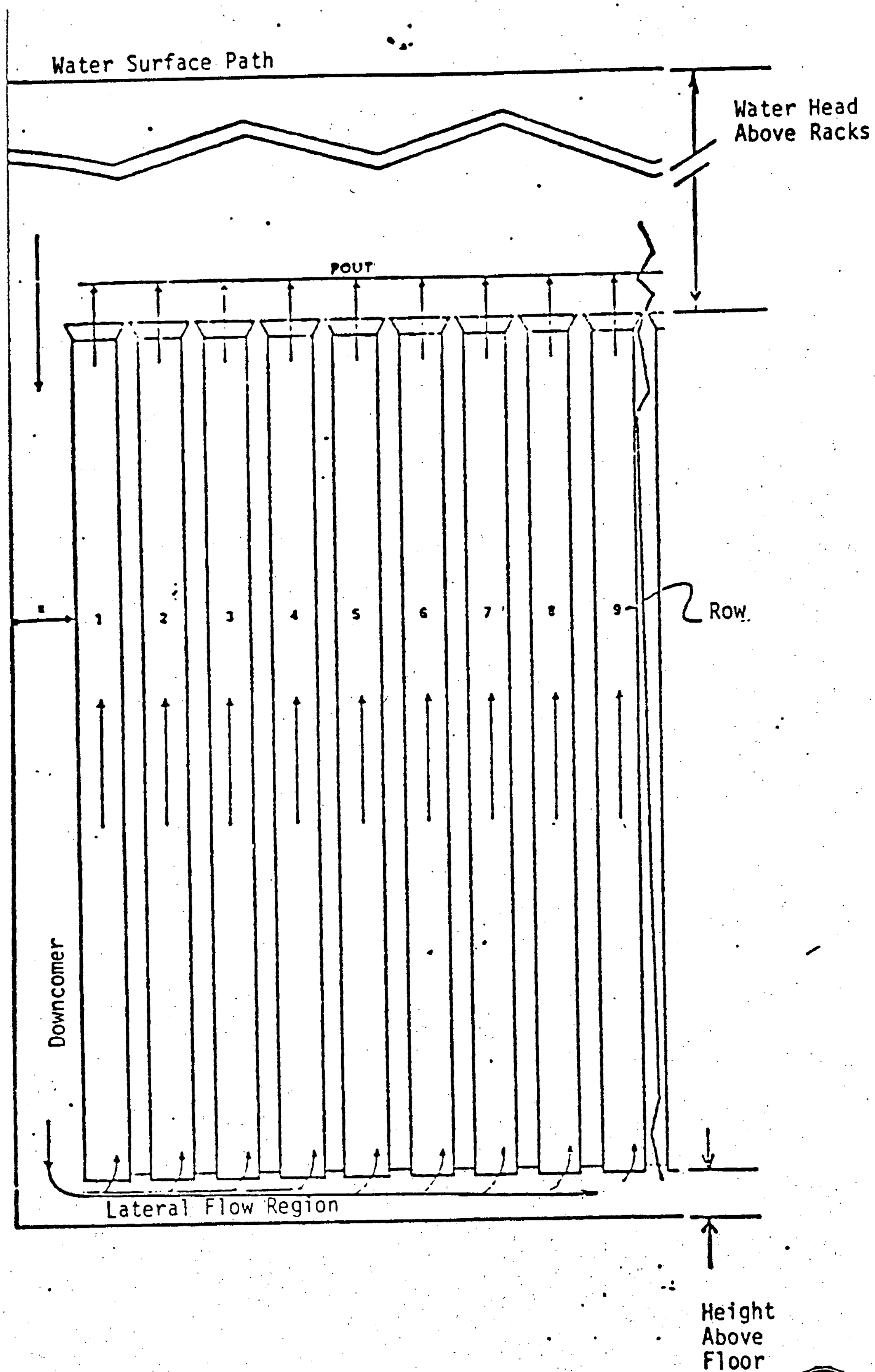
FUEL RACK ASSEMBLY
OCONEE NUCLEAR STATION
Figure 2-3



DETAIL SEISMIC MODEL

OCONEE NUCLEAR STATION

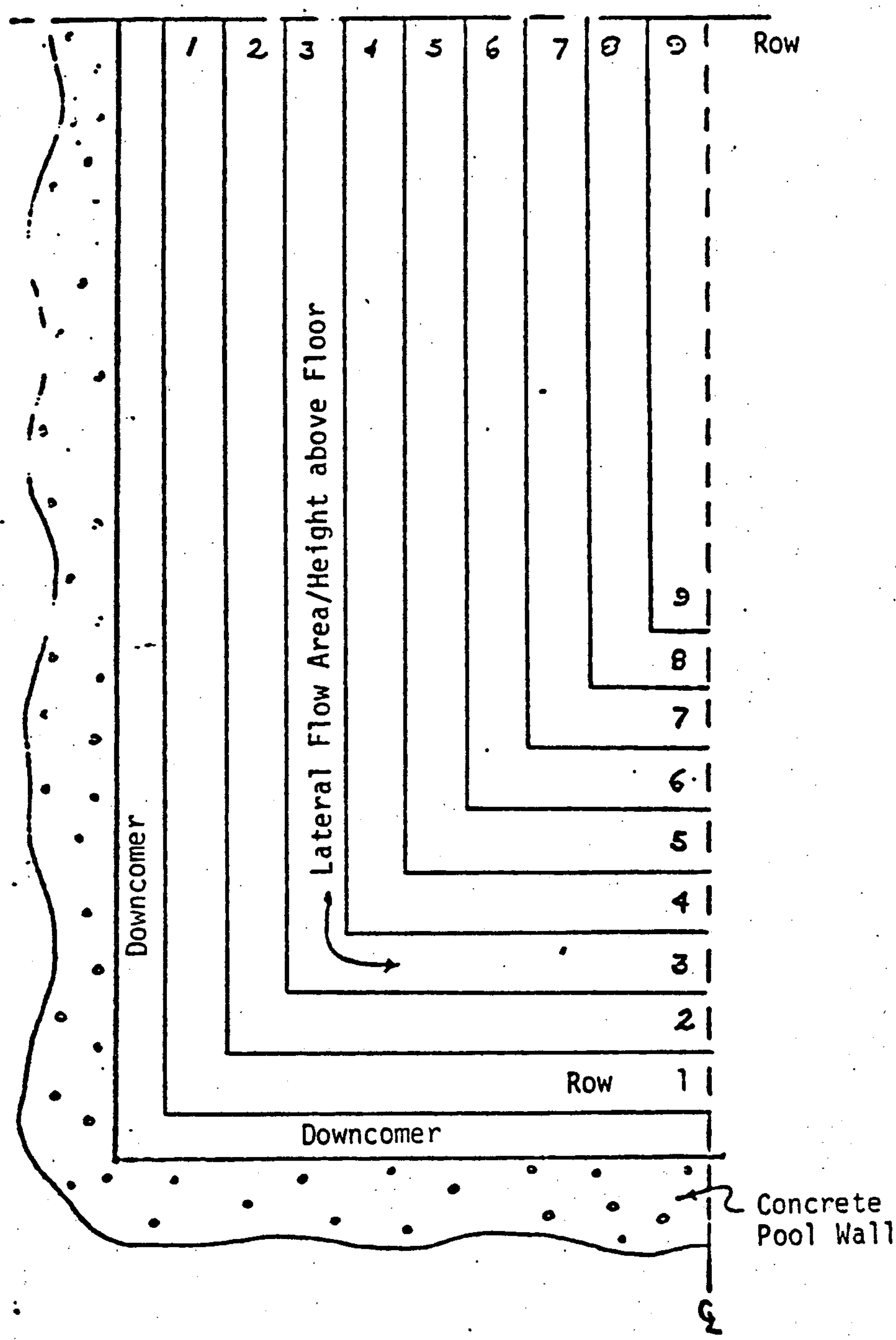
Figure 2-4



SPENT FUEL POOL
NATURAL CIRCULATION
MODEL
(ELEVATION VIEW)

OCONEE NUCLEAR STATION

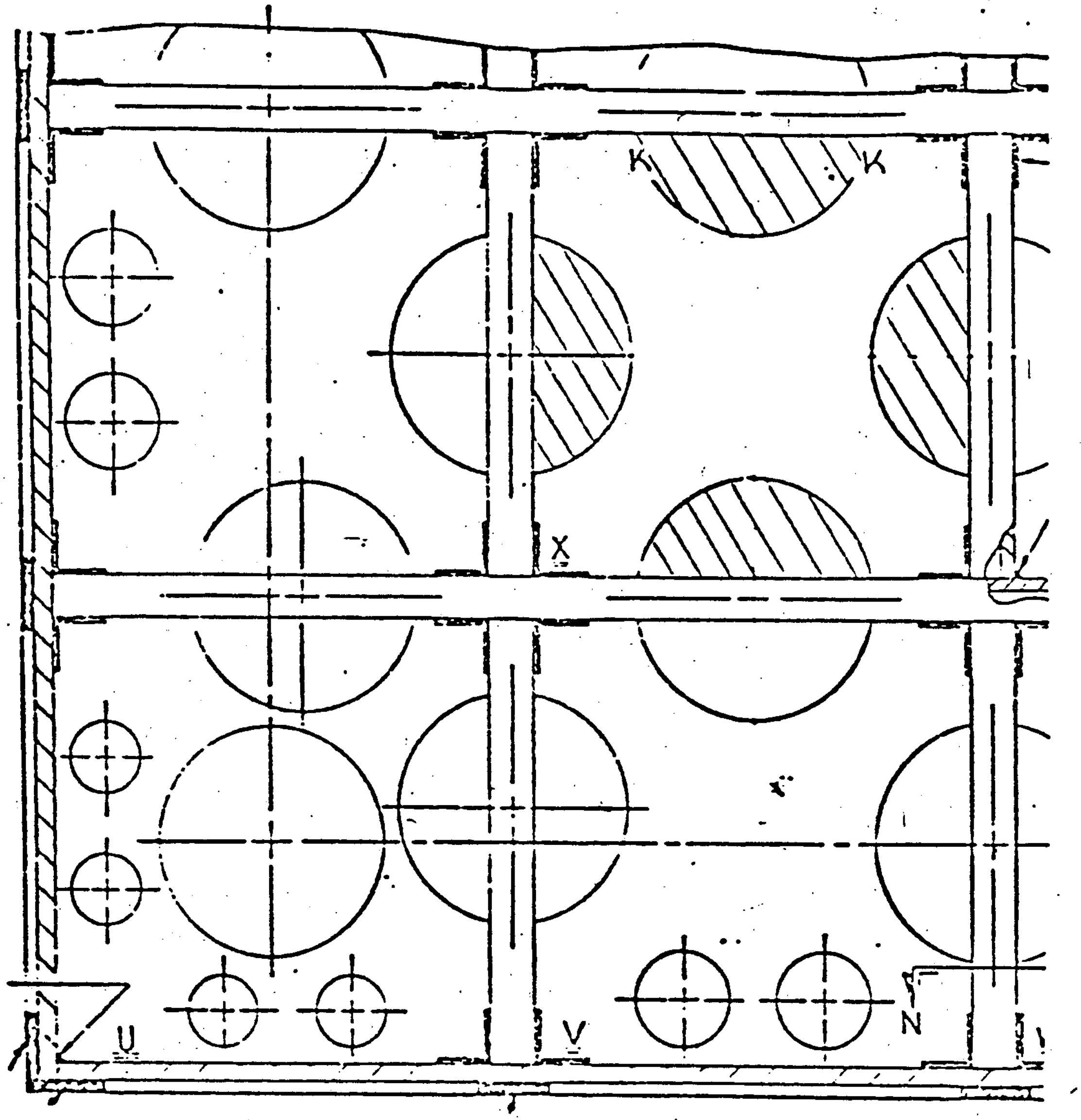
Figure 2-5



SPENT FUEL POOL
NATURAL CIRCULATION
MODEL
(PLAN VIEW)

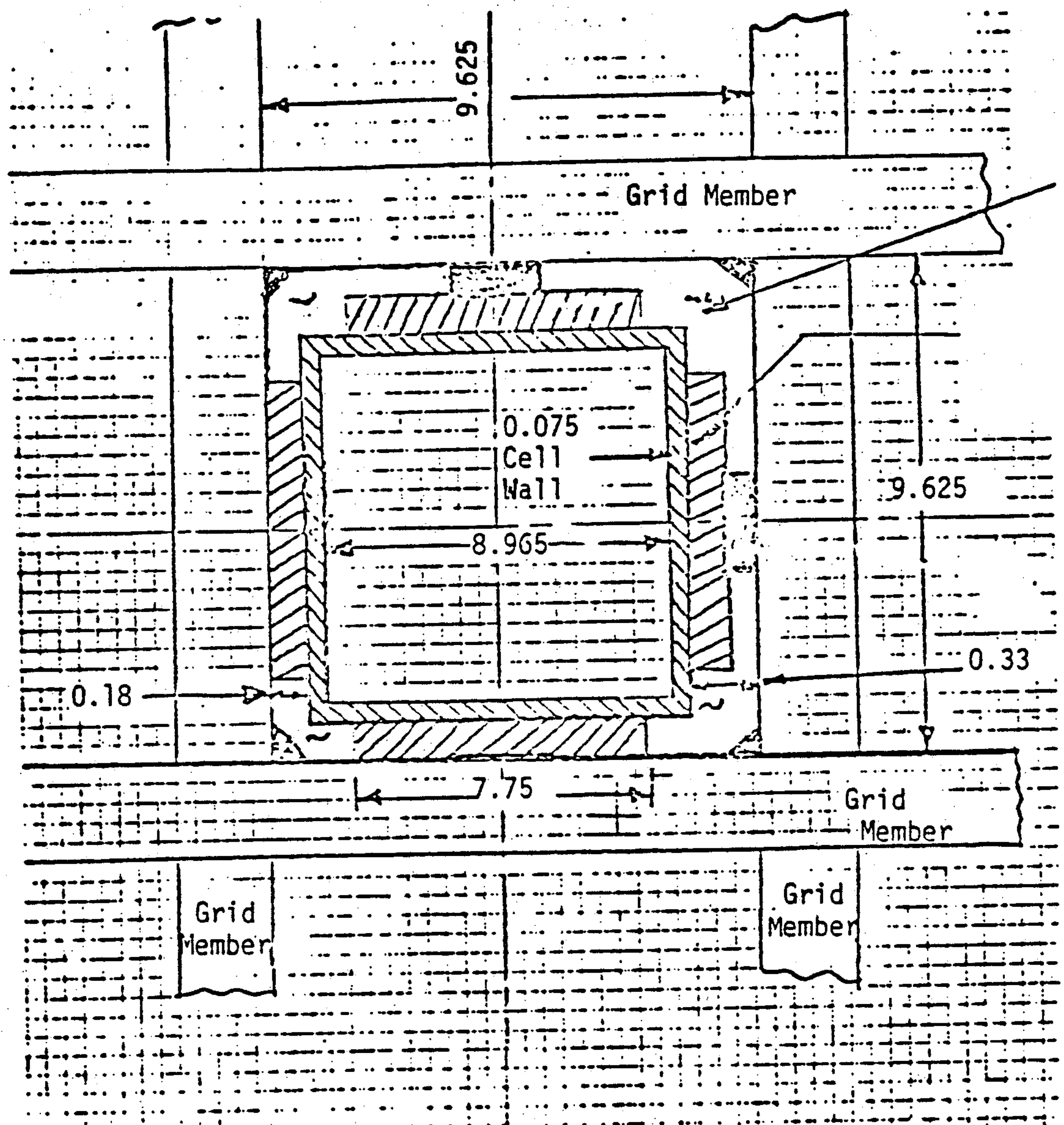
OCONEE NUCLEAR STATION

Figure 2-6



SPENT FUEL RACK
INLET FLOW AREAS
OCONEE NUCLEAR STATION

Figure 2-7



Total flow area A = 2.6 in.^{2} nominal. (In Gap)
At Support Grid

INTERCELL FLOW AREA
REGION
OCONEE NUCLEAR STATION

Figure 2-8

3.0 SPENT FUEL INTERFACE

3.1 STRUCTURAL

The spent fuel pool and its cooling system are described in the Oconee Nuclear Station Final Safety Analysis Report Section 9.4. Figure 3-1 is a reproduction of FSAR Figure 9-11. It shows the general arrangement of the Units 1 and 2 pool and the associated fuel handling equipment. This arrangement is not changed as a result of this modification. However an additional cooling train will be provided before the Unit 1 refueling in 1981. This matter has been addressed in other correspondence.

The spent fuel pool is constructed of reinforced concrete lined with stainless steel clad plate. The fuel pool concrete reinforcing steel, linear plate and welds connecting the inner plate to the fuel pool floor concrete embedments are analyzed based on consideration of the new racks and additional fuel. Design criteria including loading combinations and allowable stresses are in compliance with Oconee FSAR Appendix 5A for Class I structures. The determination of Ta (abnormal thermal load condition to be used in combination with E') is based on the failure of one pump or cooler during normal operating conditions.

The rack/spent fuel pool interface is described in Section 2.2.

3.2 THERMAL

3.2.1 DESIGN BASES

The Spent Fuel Pool and Pool Cooling System are designed to keep the pool water temperature below 163°F for normal refueling operations and full core discharge situations with any two pump-cooler configurations in operation. The addition of a third spent fuel cooler and pump will keep the pool water temperature below 150°F. 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". Table 3.2.1 shows the expected loading in the pool. The heat loading is shown in Table 3.2.2. 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.

3.2.2 SYSTEM DESCRIPTION

The Spent Fuel Cooling System is described in the Oconee FSAR Section 9.4.2. This system will be augmented by the addition of a third spent fuel cooler and pump which will take suction from the existing

spent fuel pool coolant piping. Heat exchanger cooling water will be drawn from the recirculating water system. The added cooler and pump are described in Table 3.2.3.

3.2.3 DESIGN EVALUATION

During normal operation the Spent Fuel Cooling System serves two main functions. The first is to maintain the pool water at temperatures below 150°F. The second function is to provide purification of the spent fuel pool coolant for clarity during fuel handling operations. Under normal conditions, with three pump-cooler configurations in operation, the pool temperature is maintained under 125°F as stated in the Oconee FSAR Section 9.4.2.1 by recirculating spent fuel cooling water from the spent fuel pool through the pumps and coolers and back into the pool. The pumps and coolers are arranged in parallel. The purification function is performed as described in the Oconee FSAR Section 9.4.2.1.1.

The heat loads shown in Tables 3.2.1 and 3.2.2 represent the largest heat loads expected in the spent fuel pool. The normal case assumes that Units 1 & 2 are refueled consecutively and the pool is filled with previous discharges except those spaced reserved for full core discharge. The full core discharge case assumes consecutive refueling followed by a full core discharge after a short period of operation. In this case all the spaces in the pool are filled. The calculations assume that 18 month cycles are used on both units.

Pool temperature is maintained below 150°F by operation of any two pump-cooler configurations for the normal heat load and by operation of all three pumps and coolers for the maximum heat load. Upon failure of one pump or cooler for either of these conditions sufficient cooling capacity remains to maintain bulk pool temperature below 205°F. An analysis of pool response to loss of all forced cooling is presented in Section 6.4 of this document.

3.3 WATER QUALITY

Operating experience has shown that concentrations of radionuclides are greatest during periods of fuel movement in the pool (i.e., refueling) and are not directly related to the number of assemblies stored in the pool. Therefore, the increased load on the Spent Fuel Pool Purification System will be small and the existing system will adequately maintain water chemistry, clarity, and activity within acceptable levels.

TABLE 3.2-1

Normal Heat Loads for Oconee 1 & 2 Spent Fuel Pool

Number Of Assemblies	Irradiation (EFPD)	Decay Time	Heat Output (10^6 BTU/hr)
72	1263	7 days	10.7
72	1263	35 days	5.44
144	1263	1.5 yrs	1.56
144	1263	3.0 yrs	.879
144	1263	4.5 yrs	.704
144	1263	6.0 yrs	.644
144	1263	7.5 yrs	.613
144	1263	9.0 yrs	.589
144	1263	10.5 yrs	.568
42	1263	12.0 yrs	.160
			21.857

TABLE 3.2-2

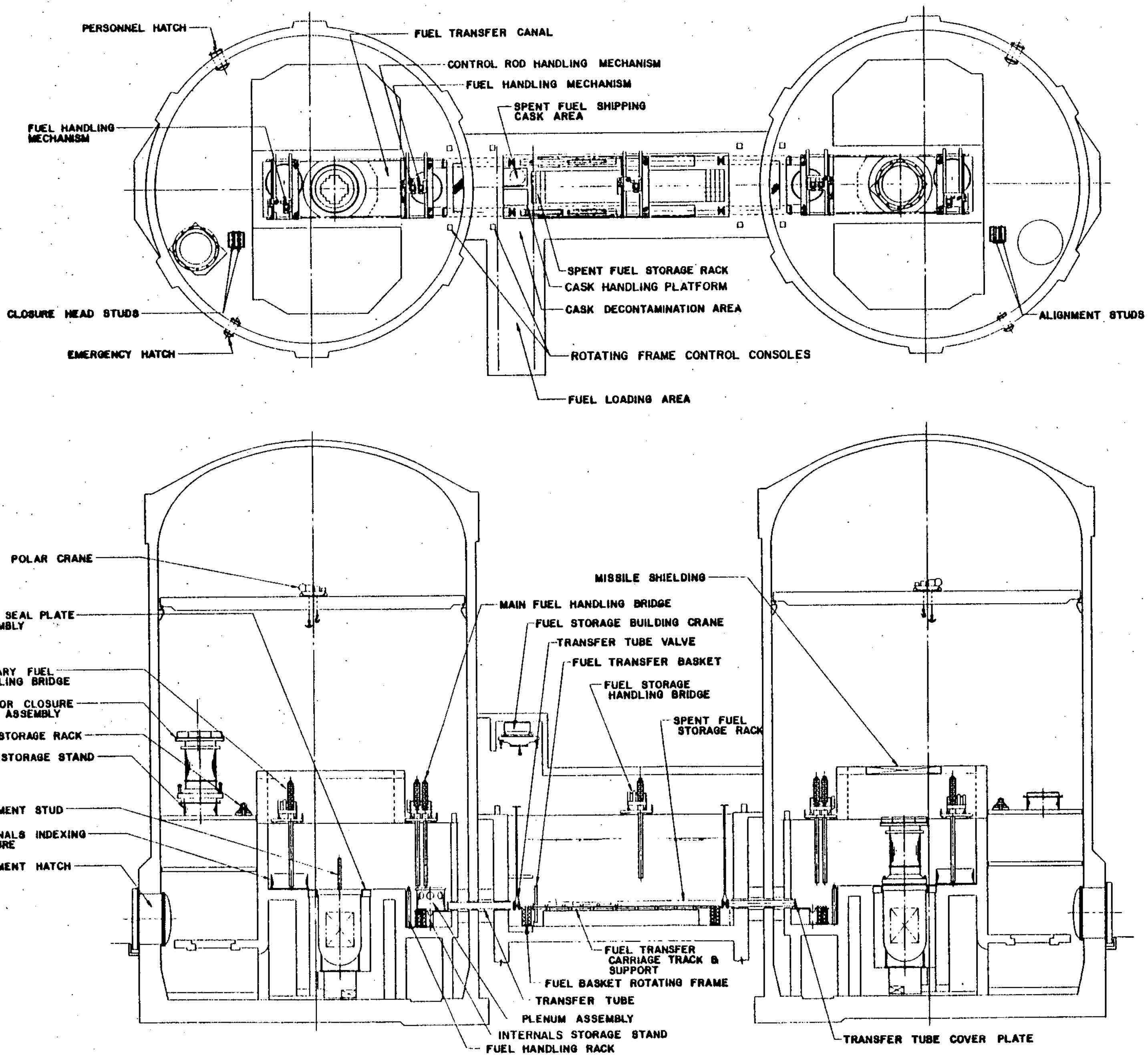
Maximum Heat Loads for Oconee 1 & 2 Spent Fuel Pool

Number Of Assemblies	Irradiation (EFPD)	Decay Time	Heat Output (10^6 BTU/hr)
72	11	7 days	4.1
72	432	7 days	10.1
33	853	7 days	4.8
72	1263	35 days	5.4
72	1263	60 days	4.1
144	1263	1.5 yrs	1.6
144	1263	3.0 yrs	.9
144	1263	4.5 yrs	.704
144	1263	6.0 yrs	.644
144	1263	7.5 yrs	.613
144	1263	9.0 yrs	.589
127	1263	10.5 yrs	.500
1294			34.050

TABLE 3.2-3

Components to be Added to Oconee Units 1 & 2 Spent Fuel Pool Cooling System

Spent Fuel Cooler	
Type	Tube & Shell
Material Tube/Shell	CS/SS
Capacity, BTU/hr/cooler	7.75×10^6
Code	ASTME V111/111-C
Spent Fuel Pump	
Type	Horizontal Centrifugal
Material	SS
Flow, gpm	1000
Head, ft. H_2O	100
Motor, hp	40



FUEL HANDLING SYSTEM
OCONEE NUCLEAR STATION
Figure 3-1

4.0 RACK INSTALLATION

The installation plan is based on the following objectives:

a) Maintaining installation exposure levels As Low As Reasonably Achievable (ALARA).

b) Avoiding transport of rack modules over stored fuel.

c) Achieving acceptable tolerances on module verticality, levelness, and positioning.

4.1 QUALITY ASSURANCE

The quality assurance aspects of the removal of the existing racks and the installation of the new racks will be carried out in such a manner as to meet the applicable requirements of the Duke Power Company Quality Assurance Program as described in Topical Report DUKE-1A.

4.2 REMOVAL OF EXISTING RACKS

The existing storage racks are Combustion Engineering, Inc., supplied High Capacity (Hi-Cap) Fuel Assembly Racks constructed of stainless steel. The configuration of these racks is shown in Figure 4-1.

Removal of the existing modules will be accomplished by first lifting them to the cask platform by a temporary construction crane. Then, the modules will be rerigged to the 100 Ton Cask Handling Crane and moved to the fuel handling area for packaging.

Disposal of the existing racks is addressed in Section 5.

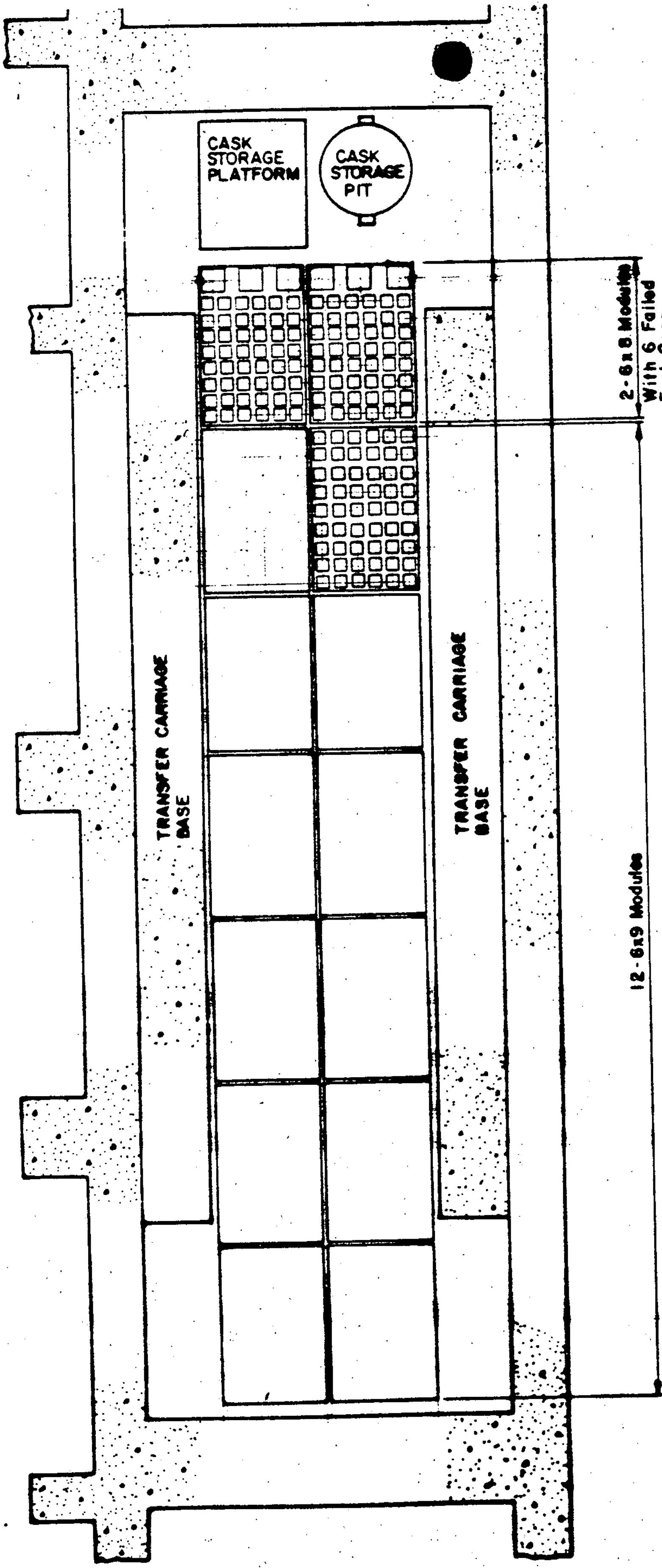
4.3 INSTALLATION OF NEW RACKS

The final configuration of the 14 new modules, supplied by Westinghouse Electric Corporation, is shown in Figure 4-2. All new modules will be brought preassembled into the fuel handling area by truck and lifted to the cask platform by the 100 Ton Cask Handling Crane. The modules will then be rerigged to a temporary construction crane. The construction crane will move all modules underwater to their final position. Existing rack removal and new rack installation will be sequenced such that the modules will be lowered to an elevation below the top of any stored fuel, but above the fuel transfer carriage. Where it is necessary to lift a module over an empty module, the distance of travel with the module in the raised position will be minimized. At no time will construction crane loads be moved over fuel stored in the pool.

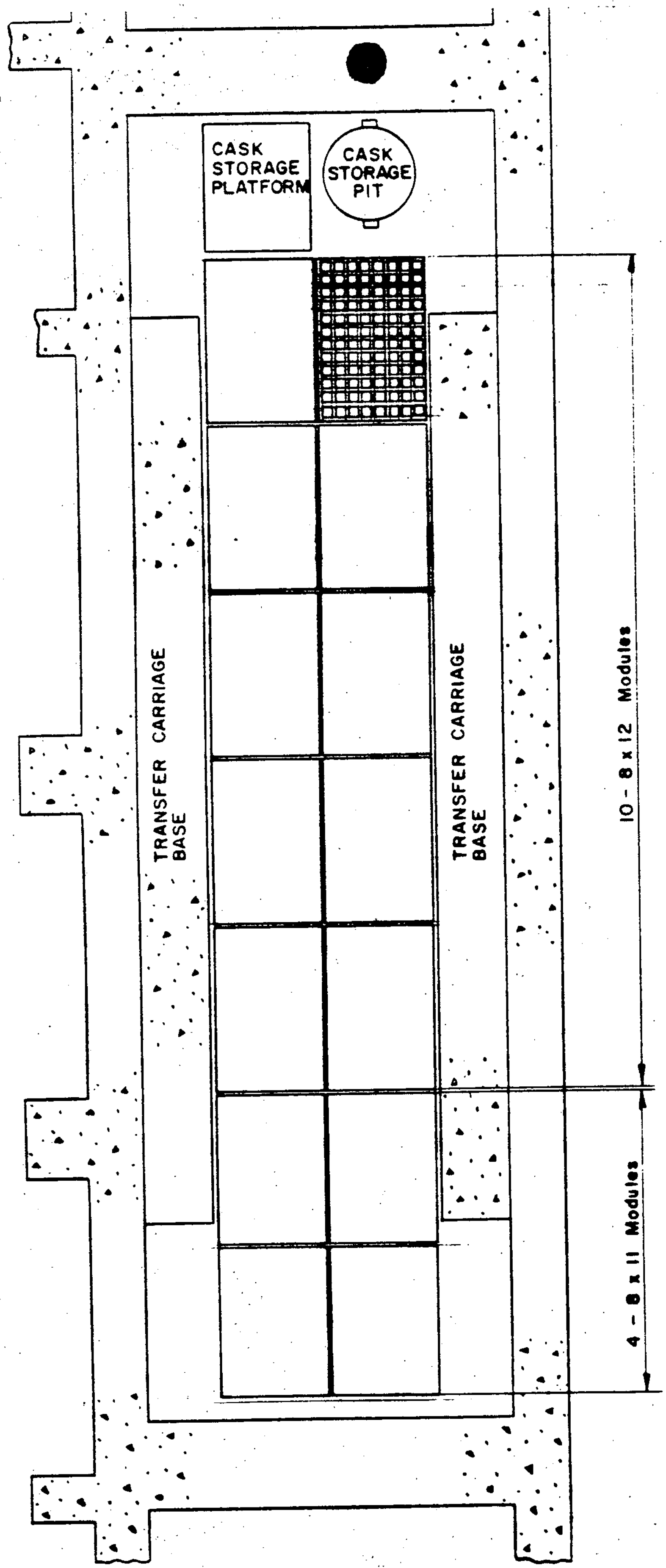
Approximately 342 spent fuel assemblies will be moved to various storage locations in the pool during the reracking operation. The distance between divers performing the work and stored fuel will be restricted as defined in Section 5.

The base of the original racks consisting of bent plates welded to the bottom liner will remain in place unless unexpected circumstances arise during reracking that require their removal. The rack supports will position the new racks above the existing bent plates.

New rack verticality and levelness tolerances will be achieved by the use of screw-adjustable supports. Module to module positioning will be accomplished by the use of spacer tools.



EXISTING SPENT FUEL RACK CONFIGURATION
OCONEE NUCLEAR STATION
Figure 4-1



NEW RACK FINAL
CONFIGURATION
OCONEE NUCLEAR STATION

Figure 4-2

5.0 RADIATION PROTECTION

5.1 GENERAL DESCRIPTION OF RADIOLOGICAL ASPECTS OF PROJECT

The radiation protection aspects of the spent fuel pool modification are the responsibility of the Station Health Physicist, who is assisted by his staff, with the support of the System Health Physicist and his staff. Gamma radiation levels in the pool area are constantly monitored by the station Area Radiation Monitoring System, which has a high level alarm feature. Additionally, periodic radiation and contamination surveys are conducted in work areas as necessary. Where there is a potential for significant airborne radionuclide concentrations, continuous air samplers are used in addition to periodic grab sampling. Personnel working in radiologically controlled areas must 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, will be worn by divers working in the pool.

Contamination control measures are used to protect persons from internal exposure to radioactive material and to prevent the spread of contamination. Radiation Control Zones (RCZ's) are established around the work area. 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 removed from the SFP will be rinsed, decontaminated further if necessary, and wrapped and/or tagged as necessary. Divers exiting the pool water will also be rinsed off to minimize personnel and area contamination problems. The station radiation protection staff closely monitors and controls all aspects of the work to ensure that personnel exposures, both internal and external, are maintained as low as reasonably achievable (ALARA).

5.1.1 UNDERWATER RADIATION SURVEY

In addition to periodic measurement of dose rates around and above the pool, underwater surveys will be conducted to determine the dose rates in areas where divers must work or pass through.

A low and high range underwater radiation monitoring instrument will be used, when applicable, to perform dose rate measurement underwater.

5.1.2 POOL DECONTAMINATION AND CLEAN-UP

The Spent Fuel Pool Cooling System provides purification and clarification of pool water by reciruclating it through a demineralizer and filters. This system operates in this mode to minimize radiation exposure to personnel from the amount of dis-

solved and suspended radionuclides in the pool water. The water will be sampled weekly to monitor the concentration of the radionuclides in the pool. In addition, a portable filtered water vacuum system will be used, as necessary, to clean loosely deposited contaminants from the pool floor, walls and fuel rack surfaces around diver working areas to minimize radiation exposures. A floating skimmer will be used to minimize exposure due to floating crud.

5.1.3 DIVING OPERATIONS

Prior to all diving operations, the spent fuel assemblies stored in the pool will be arranged in such a manner as to yield the lowest practicable dose rates to divers and still minimize the amount of movement or rearrangement of the assemblies. These two criteria should lessen the combined effects of radiation directly from the fuel and radiation from activated particles in the water that are stirred up by fuel movement. Marked underwater travel paths will be established for divers to ensure that exposures received going to and from the work areas are maintained ALARA. Health Physics personnel will be in the immediate area at all times when divers are in the water. Their duties will be to provide health physics support to minimize personnel exposure and to enforce good radiological work practices and adherence to RWP requirements. They, along with the diver supervisor who will be in direct communication with the divers, will continually observe the divers while they are in the pool and ensure that the divers are warned if they approach high radiation/exclusion zones. The divers will not be permitted to approach within ten (10) feet of spent fuel.

Divers will wear protective clothing items inside their rubber diving suits to protect them from contamination when they remove their diving suits and exit the SFP area. TLD's will be worn inside the diving suits on the chest, back, and extremities. Self-reading pocket dosimeters will be sealed in plastic bags and also worn inside the diving suit. The self-reading pocket dosimeters will be read and recorded after each dive. A daily tabulation of each individual's cumulative whole body and extremity doses will be prepared on each diver and will be reviewed by the diving supervisor and the cognizant Health Physics Supervisor. This information will be used in part (1) to maintain doses ALARA within the limits and (2) to efficiently allocate exposure among the divers working the the pools.

5.1.4 DECONTAMINATION OF REMOVED RACK SECTIONS

When the racks are removed from the spent fuel pool, they will be rinsed with a low pressure spray using demineralized water or spent fuel pool water. Personnel involved in this operation or others in the immediate area will wear appropriate protective clothing and respiratory protective equipment, if needed. The rack sections will be allowed to drip dry prior to movement to the new fuel receiving area to be boxed for disposal. This

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

5.1.5 ANTICIPATED EXPOSURES DURING RE-RACKING

Table 5.1-1 is a summary of expected exposures for each phase of the re-racking operation and for each group of workers. These estimates are based upon a task by task comparison of the man hours and dose rates observed in the 1979 re-racking with the man hours and dose rates anticipated for this re-racking.

5.2 DISPOSITION OF OLD RACKS

Burial with and without compaction, and long term storage on-site of racks until reuse or plant decommissioning have been evaluated for the disposal of the eleven contaminated racks. The racks will be sent off site for burial unless, upon removal from the pool, contamination levels on the racks are so low that decontamination and decay to acceptable levels appear feasible.

TABLE 5.1-1

ESTIMATED ALARA DOSES DURING RERACKING
(All Doses in Person REM)

	Shift Temporary Crane	Vacuuming Pool For Reracking	Rack Removal And Replacement	Total
Operations	0.0	0.1	1.0	1.1
Maintenance	4.5	0.1	3.8	8.4
Health Physics	0.4	0.3	1.9	2.6
Engineering	0.1	0.0	0.7	0.8
Divers	0.0	1.4	8.4	9.8
Total	5.0	1.9	15.8	22.7

6.0 SAFETY ANALYSIS

The following analyses are related to postulated accidents associated with operations in and around the spent fuel pool.

6.1 CONSTRUCTION ACCIDENT

The approximately 342 fuel assemblies in the fuel pool during the installation of the new racks, will have decayed at least days prior to reracking. Therefore, the possibility of the failure of a temporary crane or other construction accident is considered less severe than a cask drop accident. As an additional precaution, the rack modules will not be moved over stored spent fuel during the installation process.

6.2 CASK/HEAVY-LOAD ACCIDENT

In order to calculate the consequences of a cask drop accident, it is necessary to determine the maximum number of fuel assemblies which could be contacted. The worst case is considered to be a hoist cable failure when the cask is positioned over the fuel pool wall and the cask has an eccentric drop into the wall. In this case, yoke and load block could be deflected onto the spent fuel.

There are 138 cans under the projected cask, yoke, and block impact area. These cans buckle and deflect into adjacent cans until the total energy of the falling cask is absorbed. In total, 576 cans can potentially suffer a total loss of integrity during a cask drop accident.

The radiological consequences of the cask drop accident will be mitigated by limiting the age of fuel stored in the first 36 rows. No case movement will be allowed if fuel in these locations has decayed less than 55 days. The worst radiological consequences experienced would result from 100% of the activity contained in the fission gases trapped in gaps in the fuel stored in the locations being released into the pool water. The exclusion area boundary dose, taking no credit for ventilation system filtration, would be 0.1 rem whole body and 232 rem to the thyroid. These doses are well below 10CFR Part 100 limits.

6.3 LOSS OF FORCED COOLING

The large volume of water in the spent fuel pool takes several hours to heat up to boiling if all cooling capacity is lost. There is ample time to effect repairs to the cooling system or arrange alternate cooling should adequate cooling capacity be lost. The amount of time before the pool begins to boil is dependent on both the heat load and the initial pool temperature. With three pump-cooler configurations in operation prior to loss of forced cooling, the time to adiabatically heat up to boiling from the normal operating temperature of 125°F or less given in the FSAR and from the maximum temperature of 150°F or less is shown in Table 6.3-1. With any two pump-cooler configurations in operation prior to loss of forced cooling, the time to adiabatically heat up to boiling from normal operating temperatures of 140°F or less and from the maximum temperature of 163°F is shown in Table 6.3-2.

TABLE 6.3-1

Time to Boiling in the Unit 1 & 2 Spent Fuel Pool
Three pump-cooler configurations in operation prior to loss of F.C.

Heat Load (10^6 BTU/hr)	Initial Pool Temperature (°F)	Heatup Time (hrs)
21.857	125	15.14
34.050	150	6.68

TABLE 6.3-2

Time to Boiling in the Unit 1 & 2 Spent Fuel Pool
Any two pump-cooler configurations in operation prior to loss of F.C.

Heat Load (10^6 BTU/hr)	Initial Pool Temperature (°F)	Heatup Time (hrs)
21.857	140	12.30
34.050	163	5.10