See Correction letter of 8/7/87

July 24, 1987

Docket No. 50-255

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Mr. Kenneth W. Berry
Director, Nuclear Licensing
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. Berry:

SUBJECT: AMENDMENT TO PROVISIONAL OPERATING LICENSE NO. DRP-20 (TAC NO. 60844)

The Commission has issued the enclosed Amendment No. 105 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated February 20, 1986, as supplemented by submittals dated April 16 and 24, July 24, October 16, December 19, 1986 and April 23, 1987.

This amendment revises the Technical Specifications for the storage of spent fuel and authorizes you to increase the storage capacity of the spent fuel pool from 798 to 892 fuel assemblies.

By letter dated July 1, 1987, we requested information from you regarding anomalies found in the Boraflex neutron absorbing material used in spent fuel pools of the Point Beach Nuclear Plant and Quad Cities Station. By letter dated July 6, 1987, you responded. The long-term stability of the Boraflex has not yet been resolved. However, since you plan to maintain full core off-load capability requiring 204 vacant spent fuel storage cells and the largest of the Region II racks has 121 cells, you thereby maintain the capability of to remove and modify or replace these racks if this should prove necessary in the long-term. We will be corresponding with you further on this issue.

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed. The Environmental Assessment related to this action was transmitted to you on July 14, 1987 . The Notice of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on July 20, 1987 (52 FR 27267).

Sincerely,

Original signed by

Thomas V. Wambach, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

PD III-1*	PD III-1*	D/PD III-1*	OGC*
R. Ingram	TWambach:lt	MVirgilio	
5/6/87	5/7/87	6/29/87	6/30/87

Enclosures:
1. Amendment No. 105 to
   License No. DPR-20
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Mr. Kenneth W. Berry
Consumers Power Company

Palisades Plant

cc:
M. I. Miller, Esquire
Isham, Lincoln & Beale
51st Floor
Three First National Plaza
Chicago, Illinois 60602

Mr. Thomas A. McNish, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Jerry Sarno
Township Supervisor
Covert Township
36197 M-140 Highway
Covert, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Palisades Plant
ATTN: Mr. David P. Hoffman
Plant General Manager
27780 Blue Star Memorial Hwy.
Covert, Michigan 49043

Resident Inspector
c/o U.S. NRC
Palisades Plant
27782 Blue Star Memorial Hwy.
Covert, Michigan 49043

Nuclear Facilities and
Environmental Monitoring
Section Office
Division of Radiological
Health
P.O. Box 30035
Lansing, Michigan 48909

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 105
License No. DPR-20

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consumers Power Company (the licensee) dated February 20, 1986, as supplemented April 16 and 24, July 24, October 16, December 19, 1986, and April 23, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 105 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective upon completion of installation of the storage racks described in the revised Technical Specifications.

FOR THE NUCLEAR REGULATORY COMMISSION

M. J. Virgilio

Martin J. Virgilio, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 24, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 105

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
v	v
4-14b	4-14b
5-4	5-4
	5-4a
	5-4b
	5-4c

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 105 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated February 20, 1986, Consumers Power Company (the licensee) requested an amendment to the operating license for the Palisades Plant that would allow expansion of the fuel storage capacity of the spent fuel pool. The amendment consists of changing the Design Features (Section 5.4) of the Technical Specifications to describe and provide appropriate limitations for the expanded storage capability. Additional submittals supplementing the information provided with the application were dated April 16 and 24, July 24, October 16, and December 19, 1986 and April 23, 1987. This expansion is to be accomplished by replacing storage racks in approximately one-half of the spent fuel pool with racks having a closer center-to-center spacing. The proposed modification will increase the spent fuel storage capacity of Palisades from 798 to 892 fuel assemblies, thus allowing a full core discharge capability for two fuel cycles (Cycle 8 and Cycle 9) longer than with the existing racks. The spent fuel storage pool will be divided into two regions. Region I contains the existing storage racks which have a nominal center-to-center spacing of 10.25 inches and is designed to accommodate non-irradiated fuel enriched to a maximum of 3.27 weight percent. Region II will contain the new racks which have a nominal center-to-center spacing of 9.17 inches. Placement of fuel in Region II is restricted by burnup and enrichment limits.

2.0 EVALUATION

2.1 Criticality Considerations

The Region I racks, which have been previously licensed and installed in the Palisades spent fuel pool, are being reused and, therefore, criticality concerns for them will not be addressed.

The Region II storage racks consist of stainless steel canisters welded together to form a honeycomb type structure. The canisters have an interior square dimension of 9.0 inches. The neutron absorbing material, Boraflex, is attached to the sidewall of each canister and is held in place by a stainless steel wrapper which is spot welded to the can. The resulting structure maintains the stored fuel assemblies at a center-to-center spacing of 9.17 inches.

2.1.1 Calculation Methods

The calculation of the effective multiplication factor, K_{eff}, makes use of the AMPX system of codes for neutron cross-section preparation and the KENO-IV and PHOENIX codes for reactivity. KENO-IV is a Monte Carlo code and has been verified against a set of 27 critical experiments that simulate various features of the storage rack design. A calculational method bias of 0.0 and a 95/95 probability /confidence uncertainty of 0.0032 was inferred from these comparisons.

The calculation of the criterion for acceptable burnup for storage in Region II makes use of the concept of reactivity equivalencing, which has been reviewed and approved by the NRC in numerous other spent fuel storage applications. Since the KENO-IV code cannot treat irradiated fuel assemblies, it is necessary to obtain the fresh (unirradiated) fuel assembly enrichment which yields the same pool K_{eff} as the irradiated assembly. Because of the presence of the neutron absorber Boraflex, a multigroup transport theory code is more appropriate than diffusion theory for this calculation. Therefore, the PHOENIX code was used to calculate the reactivity of fuel assemblies in the Region II racks as a function of initial enrichment and burnup. Reactivity equivalencing is then used to extend back to an unirradiated assembly. The advantage of this proceudre is that only relative multiplication factors are computed by PHOENIX. The final value of the rack multiplication factor is obtained from the more powerful KENO-IV code.

2.1.2 Treatment of Uncertainties

For the Region II analysis, the total uncertainty applied to the calculated multiplication factor is the statistical combination of the method uncertainty, the uncertainty in the particular KENO calculation, mechanical uncertainties due to tolerances, spacing, etc., uncertainty due to particle self-shielding in the boron (actually bias), and uncertainty due to the reactivity equivalence methodology. Some of the mechanical tolerances such as centered fuel assembly position, reduced Boraflex plate width, reduced spacing between fuel assemblies, and minimum boron loading in the Boraflex plates, are treated by making worst case assumptions in the nominal calculated value of K_{eff}.

The PHOENIX code was qualified for burnup calculations by comparing calculated isotopic ratios to measurements made in Yankee Rowe Core 5 and by comparison of equivalent reactivity burnup between PHOENIX and the LEOPARD/TURTLE codes. A set of 81 critical experiments was analyzed to qualify the code for zero burnup conditions. Based on the results, a value of 1% is used for the uncertainty associated with the burnup dependent reactivities and applied to the final rack multiplication factor.

2.1.3 Results of Analysis

For Region II, the rack multiplication factor is 0.9155 for the most reactive irradiated fuel permitted to be stored in the racks, i.e., fuel with the minimum burnup permitted for each initial enrichment. All calculations are obtained for pure water at a density of 1.0 gm/cc at the temperature, within the design limits of the pool, which yields the highest reactivity. No dissolved boron is included in the water.

2.1.4 Accident Analyses

Most abnormal storage conditions will not result in an increase in the reactivity of the racks. For example, loss of a cooling system will result in an increase in pool temperature but this causes a decrease in the K_{eff} value.

It is possible to postulate events which could lead to an increase in pool reactivity such as the inadvertent drop of an assembly between the outside periphery of the rack. However, for such events credit may be taken for the approximately 1720 ppm of boron in the pool water by application of the double contingency principle of ANSI N16.1-1975. This states that one is not required to assume two unlikely, independent, concurrent events to provide for protection against a criticality accident. The reduction in K_{eff} caused by the boron (approximately 0.25) more than offsets the reactivity addition caused by credible accidents.

2.1.5 Summary of Evaluation

The staff has reviewed the assumptions made in the performance of the criticality analyses. These include use of the highest permitted reactivity fuel, pure water moderator at a density of 1.0 gm/cc, and an infinite array of assemblies. These are consistent with the NRC guidelines and are acceptable.

The staff has reviewed the uncertainties and biases included in the licensee's analysis. These are treated by either using worst case conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are poison pocket thickness, stainless steel thickness, fuel cell ID, and center-to-center spacing. In addition, a minimum poison (boron) loading is assumed in the Boraflex plates and particle self-shielding in the boron. These uncertainties were determined at least at a 95% probability 95% confidence level, thereby meeting the NRC requirements, and are acceptable.

The staff has reviewed the verification of the calculational methods. The KENO-IV code is widely used in the industry for the purpose of calculating fuel rack criticality. The set of benchmark critical experiments used to verify the calculational method encompassess the enrichment, separation distance and separating material used in the racks.

The set of experiments used to verify the PHOENIX code for the reactivity equivalence calculations is adequate and encompasses the pellet size and enrichment of the fuel proposed for storage in the Palisades racks. The conservatism of the uncertainty in the burnup dependent reactivities is verified by Yankee Rowe Core 5 isotopics and comparisons with the design LEOPARD/TURTLE code package. The staff finds that adequate verification of the codes used in the criticality analyses has been performed.

The results of the criticality calculations meet the staff's acceptance criterion of K_{eff} less than or equal to 0.95 including all uncertainties at the 95/95 probability/confidence level.

2.1.6 Technical Specifications

The staff has also reviewed the proposed changes to Technical Specifications 4.2.1 and 5.4.2 described in the Technical Specification Change Request Spent Fuel Pool Storage Capacity Expansion dated February 20, 1986, and October 16, 1986, and finds that they are consistent with the assumptions in the safety analysis and are acceptable.

The specific changes to the Technical Specifications are:

Specification 4.2.1, Table 4.2.1 - A reference is added to new Specification 5.4.2f with regard to sampling of the boron concentration in the spent fuel pool. This ties the sampling requirement of Table 4.2.1 to the boron concentration requirement in 5.4.2f and is acceptable.

Specification 5.4.2b - This specification is deleted because there is only a single rack of this type and it is included in the new Specification 5.4.2c.

Specification 5.4.2c - Defines the design features of the Region I racks and the single Type E rack. These features are unchanged from the previous Technical Specifications.

Specification 5.4.2d - Defines the design features of the Region II racks and limits the loading of spent fuel assemblies to those meeting the burnup requirements of Table 5.4-1. These limits were found acceptable as discussed in previous sections of this evaluation.

Specification 5.4.2e - Specifies the maximum loading in any storage location.

Specification 5.4.2f - This specification of boron concentration is the previous specification 5.4.2d with editorial clarification .

Specification 5.4.2g is the same as the previous 5.4.2e.

Specification 5.4.2h is the same as the previous 5.4.2f.

The previous specification 5.4.2g is deleted since the new analysis evaluated as acceptable in Section 2.8 of this evaluation justifies free-standing racks without lateral support.

2.1.7 Conclusion

Based on the review described above, the staff finds the criticality aspects of the design of the Palisades spent fuel racks to be acceptable. The staff concludes that the new racks in Region 2 are adequately designed to maintain K_{eff} less than 0.95 for Combustion Engineering and Exxon fuel used in the Palisades core as long as the spent fuel burnup requirements of Technical Specification Table 5.4-1 are adhered to.

2.2 Spent Fuel Pool (Bulk) Cooling System

The spent fuel pool (bulk) cooling system is a closed loop system which is designed to remove 23 X 10^6 BTUs/hr. while maintaining the pool outlet temperature at no more than 125°F. It consists of two pumps, two heat exchangers and associated piping and valves.

In its SER dated June 30, 1977, the staff accepted the system for cooling 798 stored fuel assemblies. Increasing the storage capacity to 892 fuel assemblies as proposed by the licensee provides for two additional refueling cycles. The increased heat load associated with these additional cycles is less than one percent of the design heat load (136 fuel assemblies that have decayed for over twelve years). This increased heat load is negligible and we, therefore, conclude that the existing spent fuel pool cooling system can adequately handle this increase without a significant increase in bulk pool water temperature and with no impact on safety.

The licensee reanalyzed the spent fuel cooling system capability for both normal refueling (1/3 core) and a full core offload. The methods described in NRC Branch Technical Position ASB 9-2 were used for establishing the decay heat loads. The results meet the acceptance criteria of the NRC Standard Review Plan 9.1.3. We, therefore, conclude that the capability of the spent fuel pool cooling system is acceptable for the proposed additional storage.

2.3 Load Handling During Installation

The proposed modification will involve the movement of fuel assemblies presently installed in the pool, removal of some of the existing spent fuel storage racks, and the installation of new racks. The licensee has committed to use procedures to prevent the movement of the fuel racks over stored spent fuel assemblies and to conduct load handling operations in accordance with the criteria of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Based on these commitments, we conclude that the licensee can adequately conduct the necessary load handling operations during installation without a significant impact on safety.

2.4 Spent Fuel Shipping Cask Drop Accident

In its June 30, 1977, SER, the staff stated that the licensee would have a Technical Specification which prohibits the movement of any cask inside the fuel storage facility. The Technical Specification is still in effect, and our previous evaluation remains unchanged. We, therefore, conclude that a shipping cask drop accident is precluded by the present Technical Specifications.

2.5 Fuel Assembly Drop Accident

In its June 30, 1977, SER, the staff stated that it had examined the consequences of fuel damage resulting from the postulated drop of a fuel assembly and determined that the consequences of such an accident would not be increased above those presented in the original Palisades Safety Evaluation Report dated February 7, 1967. This conclusion remains valid for this proposed modification as well since there are no changes in the assumption in the fuel assembly drop accident analysis. Based on the above, we conclude that the consequences of a fuel assembly drop accident are not increased by this modification and the previously approved accident analysis remains valid.

2.6 Radioactive Wastes

The plant contains radioactive waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The radioactive waste treatment systems have been previously evaluated by the staff and found acceptable. There will be no change in the radioactive waste treatment systems or in the conclusions given regarding the evaluation of these systems as a result of the proposed spent fuel rerack. Our evaluation of the radiological considerations supports the conclusions that the proposed installation of new spent fuel storage racks at Palisades is acceptable because the conclusions of the previous evaluation of the radioactive waste treatment systems are unchanged by the installation of new spent fuel storage racks.

2.7 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for expansion of the spent fuel pool storage capacity with respect to occupational radiation exposure and concludes that the ALARA policy, design, and operational considerations are acceptable. This conclusion is based on the licensee having considered the requirements of 10 CFR Parts 20.11 and 20.103, and the guidelines of Regulatory Guides 8.8 and 8.10. The occupational exposure for this operation is estimated by the licensee to be 3.3 man-rem. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job is being performed. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. One potential source of radiation is radioactive activation of corrosion products called crud. Crud may be released to the pool water because of fuel movements during the proposed spent fuel pool modification. This could increase radiation levels in the vicinity of the pool. During refuelings, when the spent fuel is first moved into the fuel pool, the addition of crud to the pool water from the fuel assembly and from the introduction of primary coolant to the pool water is greatest. However, significant releases of crud to the pool water during rack replacement is not expected since the new racks are cleaned prior to installation. The purification system for the pool, which has kept radiation levels in the vicinity of the pool to low levels, includes a filter to remove crud and will be operating during the modification of the pool.

By letter dated July 24, 1986, the licensee provided information describing actions to be taken during the spent fuel pool modification. Some of the ALARA activities directed to the reduction of occupational radiation exposure include: (a) vacuum cleaning of the spent fuel pool floor; (b) calibrated alarming dosimeters and personnel monitoring dosimeters; (c) hydrolasing and cleaning of old spent fuel racks; (d) using remote operations for rack removal and replacement operations; and (e) utilizing the spent fuel pool cavity filtration system to maintain clean water in the pool.

The licensee also has provided a description of contained and airborne radioactivity sources which are related to the spent fuel pool water and may become airborne as a result of failed fuel and evaporation. The staff has reviewed these source terms and finds them acceptable.

Based on the above, we conclude that the projected activities and estimated man-rem doses for the proposed spent fuel pool expansion appear reasonable. Further, we conclude that the licensee intends to take ALARA considerations into account, to implement reasonable dose-reducing activities, and thus, will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and meet the guidelines of Regulatory Guide 8.8. We, therefore, find the proposed occupational radiation protection aspect of the spent fuel pool modification program to be acceptable.

2.8 Structural Considerations

The NRC staff and its consultant, Franklin Research Center (FRC), performed an audit of the assumptions, methodology, and details of structural analyses that were used for the new spent fuel racks and existing pool structure at Westinghouse facilities, Pensacola, Florida. Consumers Power Company submitted Revision 1 to the February 20, 1986, Technical Specification Change Request, dated October 16, 1986. By letter dated December 19, 1986, Consumers Power Company provided responses to the inquiries made by the NRC staff during a November 6, 1986, telephone conference call. A Technical Evaluation Report (TER) on the spent fuel expansion project has been written by FRC (Attachment 1). The NRC staff has reviewed the TER and agrees with the conclusion that the new spent fuel racks and existing pool structure have met the NRC criteria and are acceptable. The geometry and analysis of the new spent fuel racks and existing pool structure are briefly described below.

2.8.1 Description and Evaluation

The spent fuel pool is constructed of reinforced concrete and lined with 3/16 inch stainless steel to ensure against leakage. The spent fuel pool is supported by a series of walls that rest on a mat foundation which is physically isolated from other structures. The pool has an inside dimension of 38 feet 9 inches by 14 feet 8 inches and a depth of 38 feet. The pool is divided into two fuel storage regions. Region I (422 locations) consists of existing racks with high density fuel assembly spacing normally used for core off-loading. Region II (470 locations) contains the new storage racks which consist of stainless steel cells assembled in a checkerboard pattern with a 9.17 inch center-to-center spacing. Placement of fuel in Region II is determined by spent fuel burnup calculations and controlled administratively. The new storage racks use a neutron absorbing material, Boraflex, which is attached to each cell sidewall by a stainless steel wrapper. The cells are welded to a base support assembly and to one another to form an integral structure resting on the pool freely, that is neither anchored to the floor nor braced to the pool walls. Because the racks rest freely on the floor and because these racks have not been designed to accommodate impact, it is necessary to determine that during seismic events the racks do not impact each other, the walls, or the existing Region I racks, and are capable of maintaining their integrity. Thus, displacement and stress calculations of the racks are required. They were calculated by the licensee using three computer programs.

Effective structural properties of an average fuel cell within the rack assembly were obtained through a three-dimensional linear structural model that represents the rack assembly. These structural properties were then used in a two-dimensional nonlinear seismic model to perform seismic calculations. In addition to the structural properties, hydrodynamic mass of the fuel, the gap between the fuel and cell, the support pad boundary condition of the free-standing rack, and the assumed coefficient of friction between the support pad and pool floor were also input to the computer program. A coefficient of friction equal to 0.2 was assumed to obtain the maximum sliding distance of the base of a rack. A coefficient of friction equal to 0.8 was assumed to obtain the maximum load in the rack and maximum structural deflection of the rack. The maximum loads thus obtained were then input to a three-dimensional structural analysis program to obtain local stresses in the rack. The licensee indicated that all stresses were within the allowables set by the NRC criteria. The licensee also indicated that the maximum single rack sliding displacement was 0.0053 inches, the maximum single rack deformation including elastic distortion and tipping was 0.258 inches, and the maximum relative displacement between adjacent racks was 0.439 inches. Since the minimum clearance space available is 1.50 inches, the racks will not impact each other or the walls during earthquakes.

The licensee performed an anlaysis for the spent fuel pool structure including the load of new racks and found that the maximum stresses in the reinforcing steel and concrete were less than the allowables set by the ACI 318 code. Since the total weight of the new racks was only slightly more than the total weight of the old racks they replaced, the licensee's analysis result was anticipated and is acceptable to the staff.

The licensee has also performed an analysis of dropping a fuel assembly straight through a storage location and the results indicated that the pool liner would not be perforated. For other fuel assembly drops, the licensee stated that, with the 1720 ppm boron in the pool water, there would be no deformation that could reasonably be achieved by the drop of a fuel assembly that would cause the criticality acceptance criterion to be violated.

2.8.2 Conclusion

Based on result of the NRC audit review, the TER by FRC, and review of the licensee's submittals, the staff has concluded that the licensee has adequately and satisfactorily addressed all the structural issues related to the expansion of storage capacity of the spent fuel pool at Palisades Plant.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment dated July 14, 1987, was issued for this amendment. Notice of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on July 20, 1987 (52 FR27267).

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 24, 1987

Principal Contributors:

Thomas Wambach, Project Manager
John Ma, Engineering Branch
Robert Ferguson, Plant Systems Branch
John Minns, Plant Systems Branch
Charles Nichols, Plant Systems Branch
Laurence Kopp, Reactor Systems Branch

Attachment 1: "Evaluation of Spent Fuel Racks Structural Analysis of Palisades Plant," January 12, 1987, Franklin Research Center. (TER C5506-650)

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS (Continued)

 Amendment No. 37, 48, 63, 67, 68, 85,

Table 4.2.1

Minimum Frequencies for Sampling Tests

(1) A daily sample shall be obtained and analyzed if fission product monitor is out of service

(2) After at least 2 EFPD and at least 20 days since the last shutdown of longer than 48 hours.

(4) When iodine or particulate radioactivity levels exceed 10 percent of limit in Specification 3.9.6 and 3.9.9, the sampling frequency shall be increased to a minimum of once each day.

(5) If the air ejector gas monitor is out of service, the secondary coolant gross radioactivity shall be measured once per day to evaluate steam generator leak tightness.

(6) Reference Specification 3.8.5 for maximum bulk water temperature and monitoring requirements.

(7) Reference Bases section of Specification 3.8 and Section 5.4.2f of the Design Features for minimum boron concentration (≥ 1720 ppm).

5.4 FUEL STORAGE

5.4.1 New Fuel Storage

a. The pitch of the new fuel storage rack lattice is $\geq$ 9.375 inches, and every other position in the lattice shall be permanently occupied by an 8" x 8" structural steel box beam or core plugs such that the minimum center-to-center spacing of new fuel assemblies in the alternating storage array is 13.26". This distance in the alternating storage lattice is sufficient so that K_{eff} will not exceed 0.98 where fuel which contains not more than 41.24 grams of U-235 per axial centimeter of active fuel assembly is in place and optimum (i.e., aqueous foam) moderation is assumed, and the K_{eff} will not exceed 0.95 when the storage area is flooded with unborated water. The calculated K_{eff} includes a conservative allowance for uncertainties as described in CPC letters of 12/18/78 and 1/12/79.

b. New fuel may also be stored in shipping containers.

c. The new fuel storage racks are designed as a Class I structure.

5.4.2 Spent Fuel Storage

a. Irradiated fuel bundles will be stored, prior to off-site shipment in the stainless steel-lined spent fuel pool.

b. (Deleted)

c. The spent fuel storage pool and spare (north) tilt pit are divided into two regions identified as Region I and Region II as illustrated in Figure 5.4-1. Region I racks are designed and shall be maintained with a nominal 10.25" center-to-center distance between fuel assemblies with the exception of the single Type E rack which has a nominal 11.25" center-to-center distance between fuel assemblies. The Region I spent fuel storage racks are designed such that fuel having a maximum U-235 loading of 3.27 w/o of U-235 placed in the racks would result in a K_{eff} equivalent to ≤ 0.95 when flooded with unborated water. The K_{eff} of ≤ 0.95 includes a conservative allowance for uncertainties.

d. Region II racks have a 9.17 inch center-to-center spacing. Because of this smaller spacing, strict controls are employed to evaluate burnup of the fuel assembly prior to its placement in Region II cell locations. Upon determination that the fuel assembly meets the burnup requirements of Table 5.4-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.

e. After installation of the two-region high density spent fuel racks, the maximum loading for fuel assemblies in the spent fuel racks is 3.27 w/o of U-235.

f. The minimum spent fuel pool water boron concentration shall be 1720 ppm. Boron concentration shall be verified at least once monthly.

g. The spent fuel racks are designed as a Class I structure.

h. Spent fuel shipping casks shall not be moved into the fuel storage building until such time as the NRC has reviewed and approved the spent fuel cask drop evaluation.

i. Storage in Region II of the spent fuel pool and spare (north) tilt pit shall be restricted by burnup and enrichment limits specified in Table 5.4-1.

NOTE: Until needed for fuel storage, one Region II rack in the northeast corner of the spent fuel pool may be removed and replaced with the cask anti-tipping device.

References

FSAR Update Chapter 5
FSAR Update Chapter 9

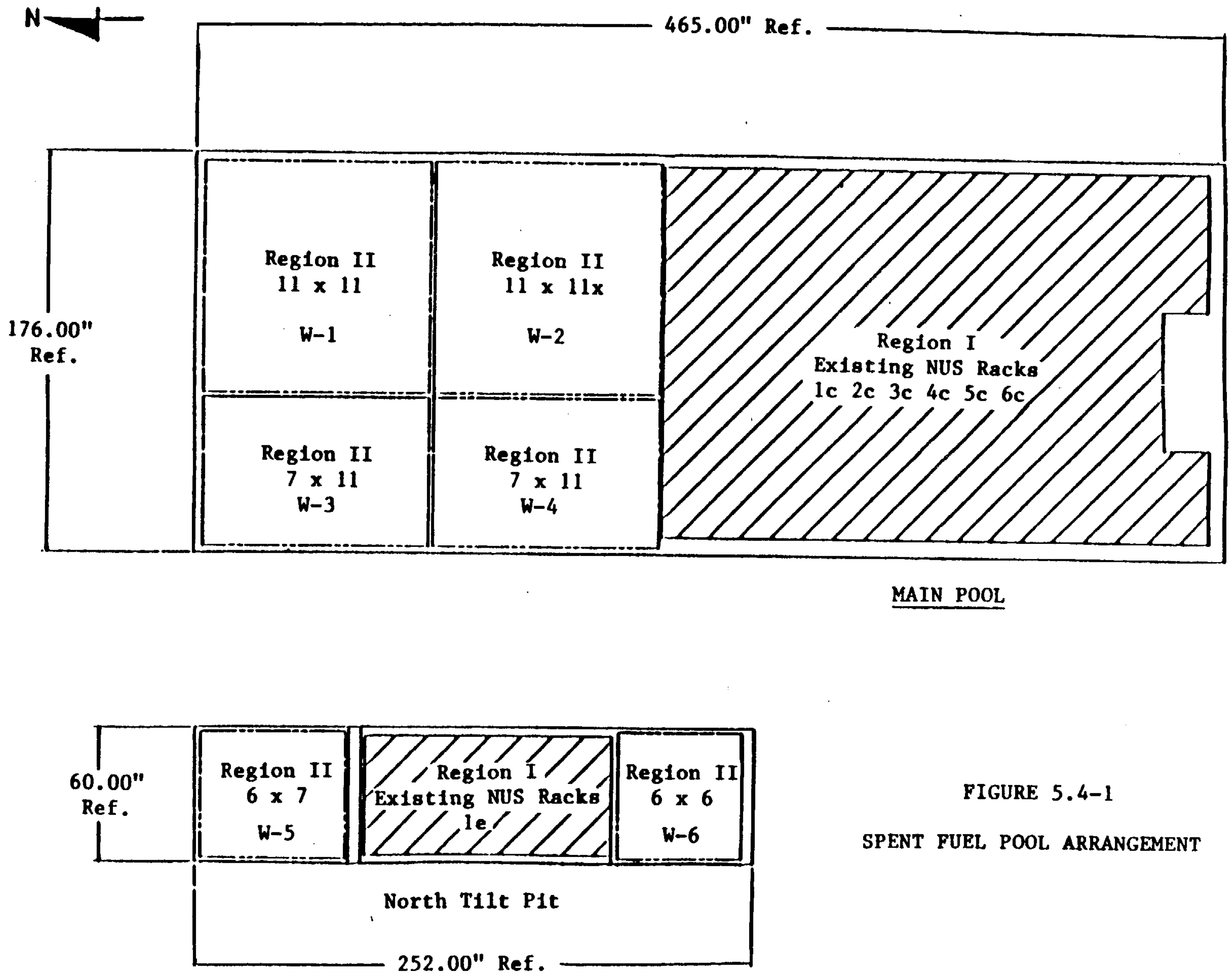

FIGURE 5.4-1

SPENT FUEL POOL ARRANGEMENT

TABLE 5.4-1

Spent Fuel Burnup Requirements
for Storage in Region II of
the Spent Fuel Pit

Initial w/o	Discharge Burnup GWD/MT
1.5	0
1.6	1.9
1.8	5.2
2.0	8.5
2.2	11.5
2.4	14.1
2.6	16.6
2.8	18.8
3.0	20.9
3.2	22.9
3.27	23.5

Linear interpolation between two consecutive points will yield conservative results.

FRANKLIN RESEARCH CENTER

DIVISION OF ARVIN/CALSPAN

TECHNICAL REPORT

20TH & RACE STREETS PHILADELPHIA, PA 19103 TWX 710-670-1889 TEL. (215) 448-1000

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EVALUATION OF SPENT FUEL RACKS STRUCTURAL ANALYSIS

CONSUMERS POWER COMPANY
PALISADES PLANT

TER-C5506-650

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

FRC Group Leader: A. Okaily
NRC Lead Engineer: J. Ma

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Prepared by:	Reviewed by:	Approved by:
Aly A. Okaily	Stephen J. Triolo	P. Carfagno
Principal Author		Department Director
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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, PWR Licensing-B) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION

1.1 PURPOSE OF THE REVIEW

This technical evaluation report (TER) covers an independent review of the Consumers Power Company's licensing report [1] on spent fuel storage modification for the Palisades Plant with respect to the evaluation of the spent fuel racks' structural analyses, the fuel racks' design, and the pool's structural analysis. The objective of this review was to determine the structural adequacy of the Licensee's high-density spent fuel racks and spent fuel pool.

1.2 GENERIC BACKGROUND

Many licensees have entered into a program of introducing modified fuel racks to their spent fuel pools that will accept higher density loadings of spent fuel in order to provide additional storage capacity. However, before the new higher density racks may be used, the licensees are required to submit rigorous analysis or experimental data verifying that the structural design of the fuel rack is adequate and that the spent fuel pool's structure can accommodate the increased loads.

The analysis is complicated by the fact that the fuel racks are fully immersed in the spent fuel pool. During a seismic event, the water in the pool, as well as the rack structure, will be set in motion, resulting in fluid-structure interaction. The hydrodynamic coupling between the fuel assemblies and the rack cells, as well as between adjacent racks, plays a significant role in affecting the dynamic behavior of the racks. In addition, the racks are free-standing. Since the racks are not anchored to the pool floor or the pool walls, the motion of the racks during a seismic event is governed by the static/dynamic friction between the rack's mounting feet and the pool floor, and by the hydrodynamic coupling to adjacent racks and the pool walls.

Accordingly, this report covers the review and evaluation of analyses submitted for the Palisades Plant by the Licensee, wherein the structural analysis of the spent fuel racks under seismic loadings is of primary concern due to the nonlinearity of gap elements and static/dynamic friction, as well as fluid-structure interaction. In addition to the evaluation of the dynamic structural analysis for seismic loadings, the design of the spent fuel racks and the analysis of the spent fuel pool structure under the increased fuel load are reviewed.

2. APPLICABLE DESIGN CODES AND STRUCTURAL ACCEPTANCE CRITERIA

2.1 APPLICABLE DESIGN CODES

The design and fabrication of the new high-density spent fuel racks as well as the structural analysis of the spent fuel pool are performed in accordance with applicable portions of the following NRC Regulatory Guides, Standard Review Plan Sections, and published standards:

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

b. NRC Regulatory Guides

1.13, Rev. 2 Dec. 1981 (Draft)	Spent Fuel Storage Facility Design Basis
1.29, Rev. 3 Sept. 1978	Seismic Design Classification
1.92, Rev. 1 Feb. 1976	Combining Model Responses and Spatial Components in Seismic Response Analysis
1.124, Rev. 1 Jan. 1979	Service Limits and Load Combinations for Class 1 Linear-Type Component Supports

c. Standard Review Plan - NUREG-0800

Rev. 1, July 1981	Section 3.7, Seismic Design
Rev. 1, July 1981	Section 3.8.4, Other Seismic Category I Structures
Rev. 3, July 1981	Section 9.1.2, Spent Fuel Storage
Rev. 1, July 1981	Section 9.1.3, Spent Fuel Pool Cooling System

d. Industry Codes and Standards

ANSI N210-76	Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

ASME Section III-80 Nuclear Power Plant Components (through Summer 1982 Addendum)

ACI 318-63	Building Code Requirements for Reinforced Concrete

e. Palisades Final Safety Analysis Report (FSAR) Update, Rev. 1

2.2 STRUCTURAL ACCEPTANCE CRITERIA

The principal acceptance criteria [1] for the evaluation of the new spent fuel racks and the existing spent fuel pool structures for the Palisades Plant are set forth by the NRC's OT (Operating Technology) Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (OT Position Paper) [2] and Palisades Final Safety Analysis Report (FSAR Update).

The main safety function of the spent fuel pool and the new spent fuel racks, as stated in OT Position Paper [2], is "to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling."

2.2.1 Structural Acceptance Criteria for Spent Fuel Pool Structure

As stated in the licensing report [1], the spent fuel pool structure was designed for ductile behavior (i.e., with reinforcing steel stresses controlling the design). The acceptance criteria are stated in Chapter 5, Appendix A of the FSAR Update [3]. These criteria apply in the structural reanalysis. Acceptance is based on maintaining structural integrity and ductile behavior of the pool structure. The pool structure includes the pool walls and mat and the supporting soil beneath the mat. Stresses in concrete and reinforcing steel components required to maintain structural integrity should be within the allowable stresses corresponding to the load combinations described in Section 3.5.3 of this TER and the ultimate strength design portion specified in the ACI 318-71 code.

2.2.2 Structural Acceptance Criteria for Spent Fuel Storage Racks

Section IV of the NRC OT Position paper [2] describes the mechanical, material, and structural considerations for the new fuel racks and their analysis.

Applicable codes, standards, and specifications for construction materials are provided by Section IV-2 of the OT Position Paper [2] as follows:

"Construction materials should conform to Section III, Subsection NF of the ASME* Code. All materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel materials may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for the stainless steel alloy used may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code."

Criteria for seismic and impact loads are provided by Section IV-3 of the OT Position Paper, which requires the following:

- o Seismic excitation along three orthogonal directions should be imposed simultaneously.
- o The peak response from each direction should be combined by the square root of the sum of the squares. If response spectra are available for vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.
- o Increased damping of fuel racks due to submergence in the spent fuel pool is not acceptable without applicable test data and/or detailed analytical results.
- o Local impact of a fuel assembly within a spent fuel rack cell should be considered.

Temperature gradients and mechanical load combinations are to be considered in accordance with Section IV-4 of the OT Position Paper [2]. The design and analysis procedures are specified in Section IV-5 as follows:

"Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes;

*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.
**American Institute of Steel Construction, Latest Edition.

the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified."

The structural acceptance criteria are provided by Section IV-6 of the OT Position Paper. For sliding, tilting, and rack impact during seismic events, Section IV-6 of the OT Position Paper [2] provides the following:

"For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.9.5.II.5 of the Standard Review Plan

(b) it can be shown that any sliding and titling motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated."

3. TECHNICAL REVIEW

3.1 INTRODUCTION

The technical materials and evaluation presented in this section are based on the Licensee's revised safety analysis report dated October 16, 1986 [1] and its response to the NRC's request for additional information [3]. On October 8 and 9, 1986, a structural analysis audit of the new spent fuel racks and existing pool was performed by FRC and NRC staff at Westinghouse facilities, Pensacola, Florida. The audit served the technical evaluation purpose of determining the adequacy of the structural analysis assumptions, methodology, and details performed by the Licensee.

3.2 DESCRIPTION OF STRUCTURES

3.2.1 Description of Existing Spent Fuel Pool

Figures 3-1 through 3-7 show the physical configuration of the spent fuel pool structure.

The spent fuel pool and the new fuel storage facilities are located between column rows F and G and column lines 22 and 28 of the auxiliary building. The pool has a depth of 38 ft; the floor is at elevation 611 ft, rising to the operating deck at elevation 649 ft. The portion of the auxiliary building housing the spent fuel pool structures is founded on a separate mat and is physically isolated from other structures.

The spent fuel pool is constructed of reinforced concrete and is oriented in the north-south direction in the auxiliary building. The main pool floor is at elevation of 611 ft, and the tilt pit floors are at elevation 610 ft. The spent fuel pool is supported by series of walls which bear on the foundation mat at 590 ft. Thus, the pool structure extends upward from the mat at elevation 590 ft to operation floor elevation 649 ft. The pool walls also serve as support for adjacent floors in addition to their primary function of resisting the hydrostatic and hydrodynamic pressures.

The entire interior face of the spent fuel pit has a 3/16-in stainless steel liner to ensure against leakage. The inside dimensions of the pool are 38 ft 9 in by 14 ft 8 in. A 9-ft x 9-ft area in the northeast corner of

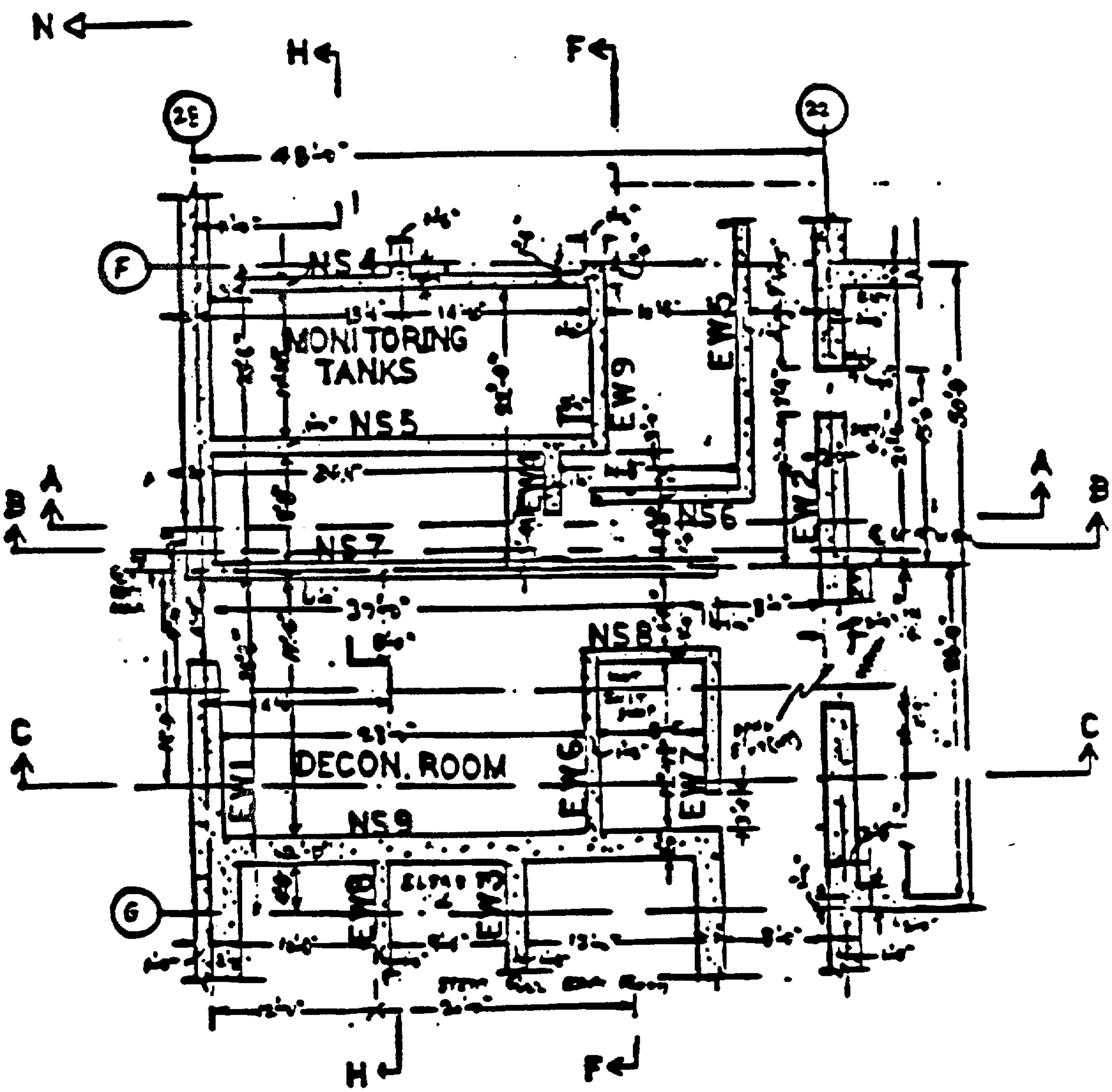


Figure 3-1. Plan at Elevation 590 ft

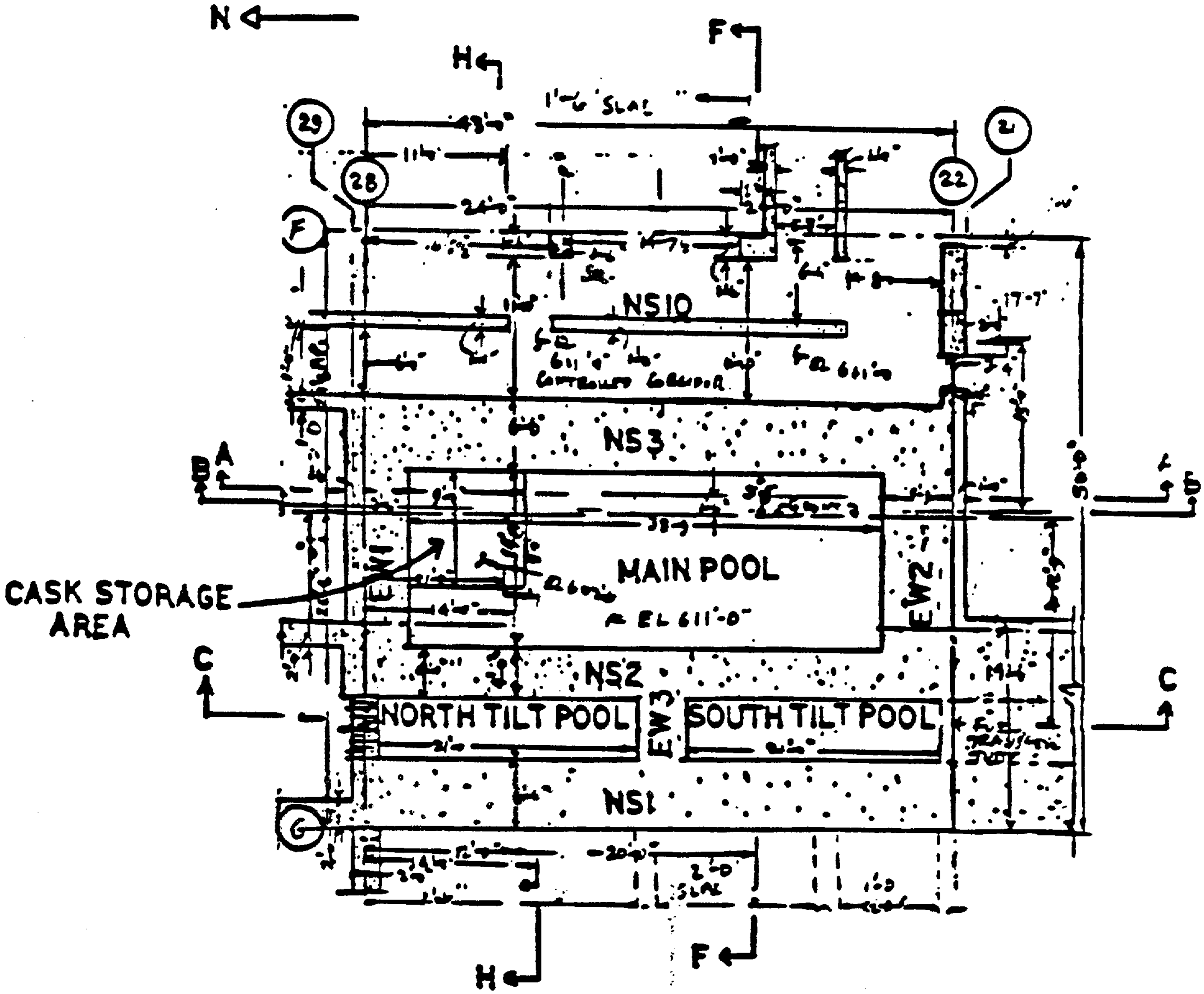


Figure 3-2. Plan at Elevation 611 ft

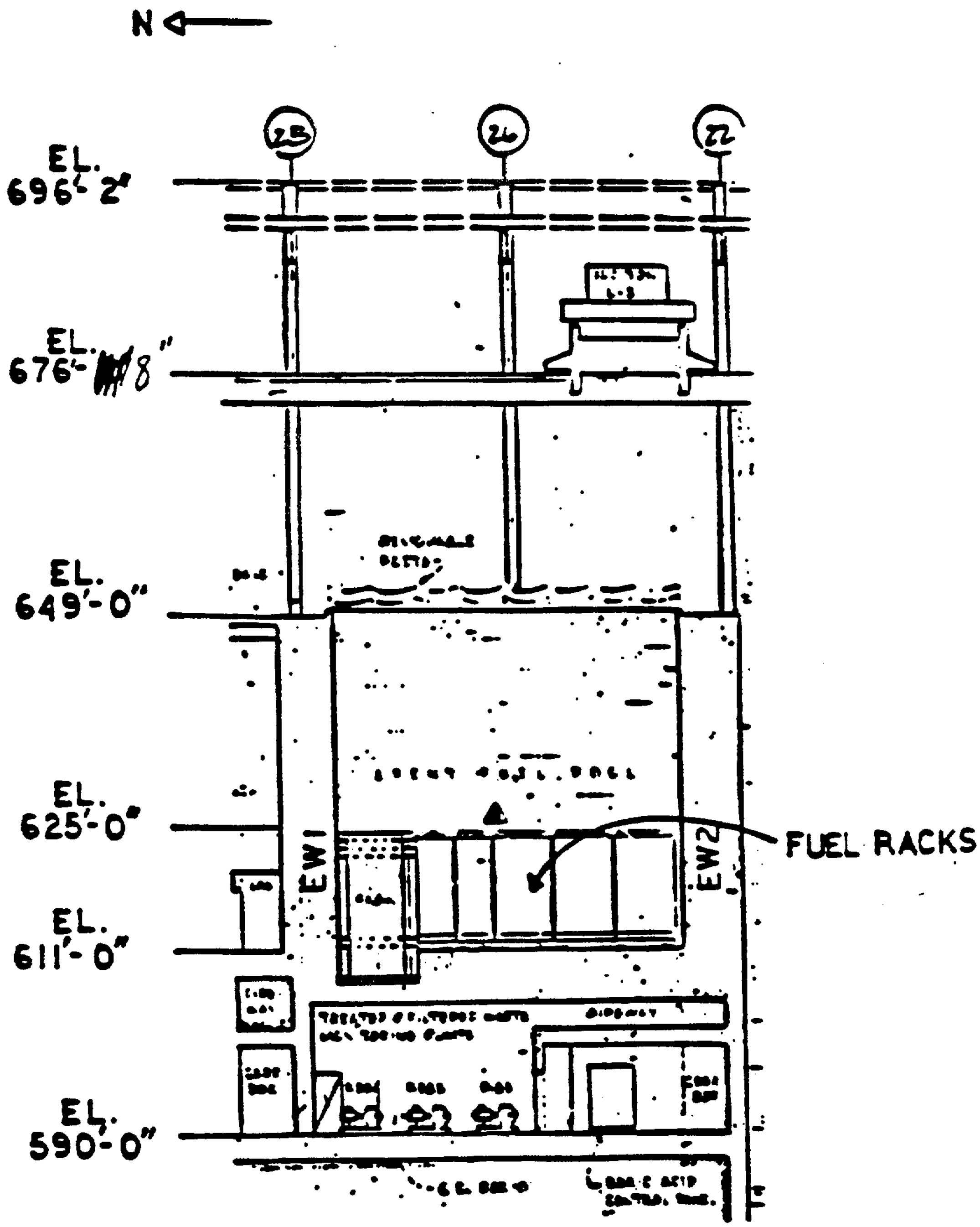


Figure 3-3. Section A-A - Elevation 590 ft to 696 ft

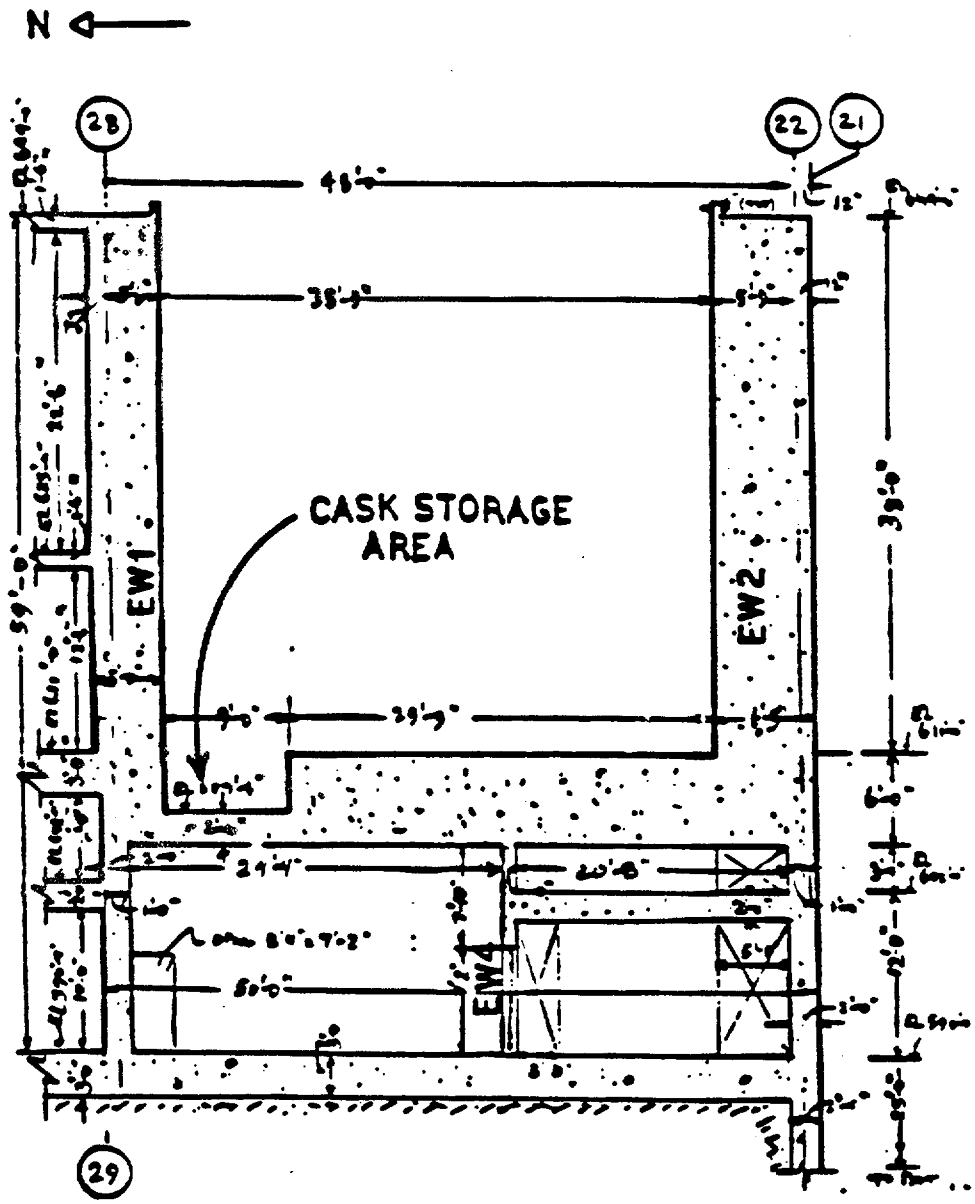


Figure 3-4. Section B-B - Elevation 590 ft to 649 ft

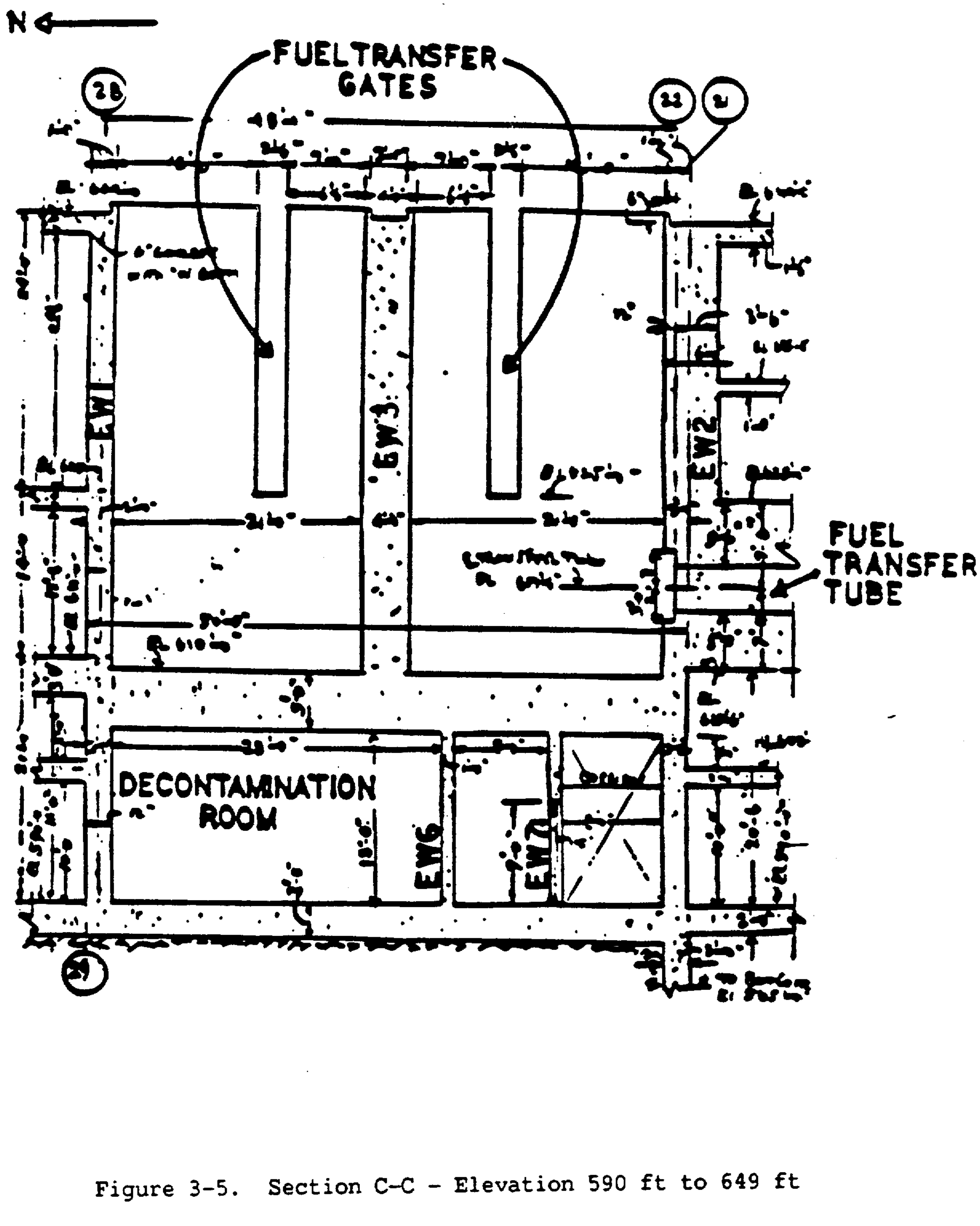


Figure 3-5. Section C-C - Elevation 590 ft to 649 ft

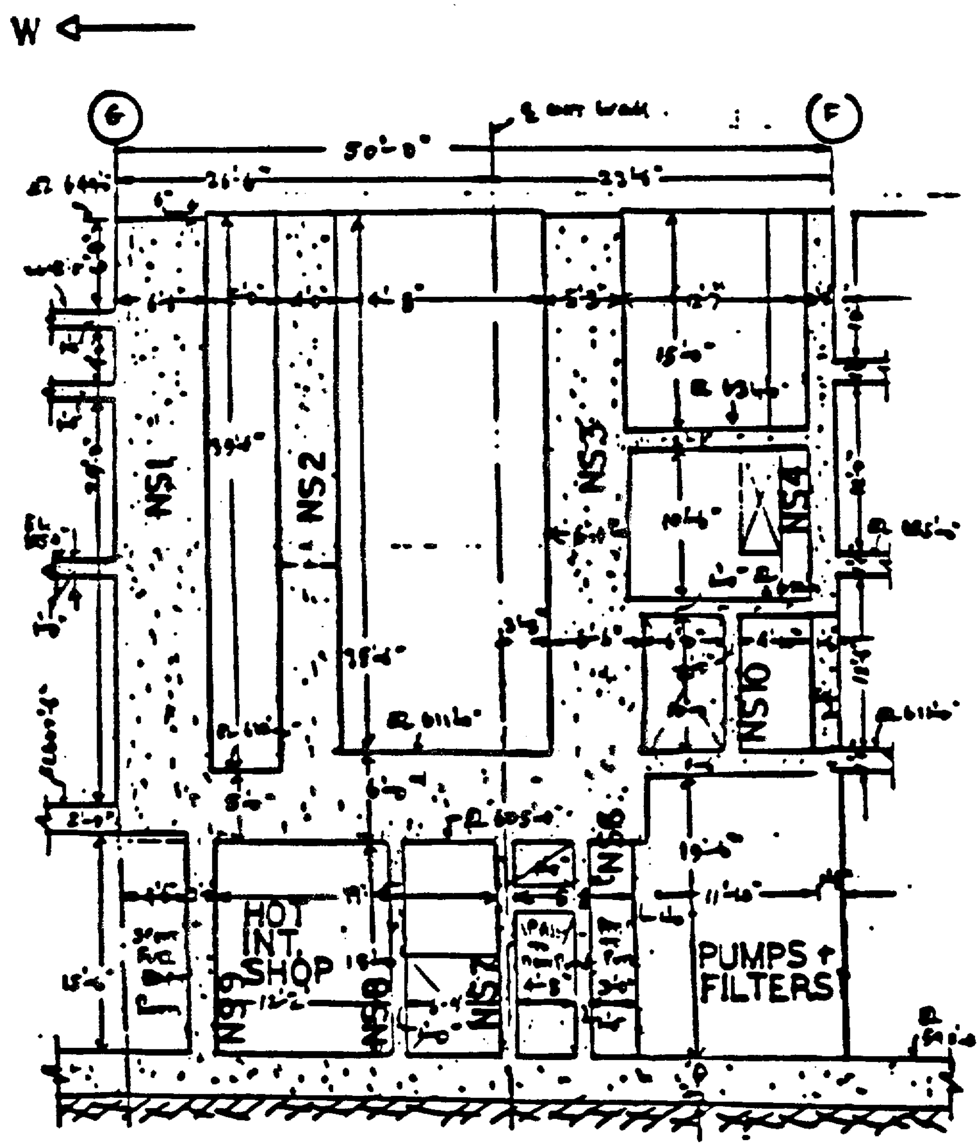


Figure 3-6. Section F-F - Elevation 590 ft to 649 ft

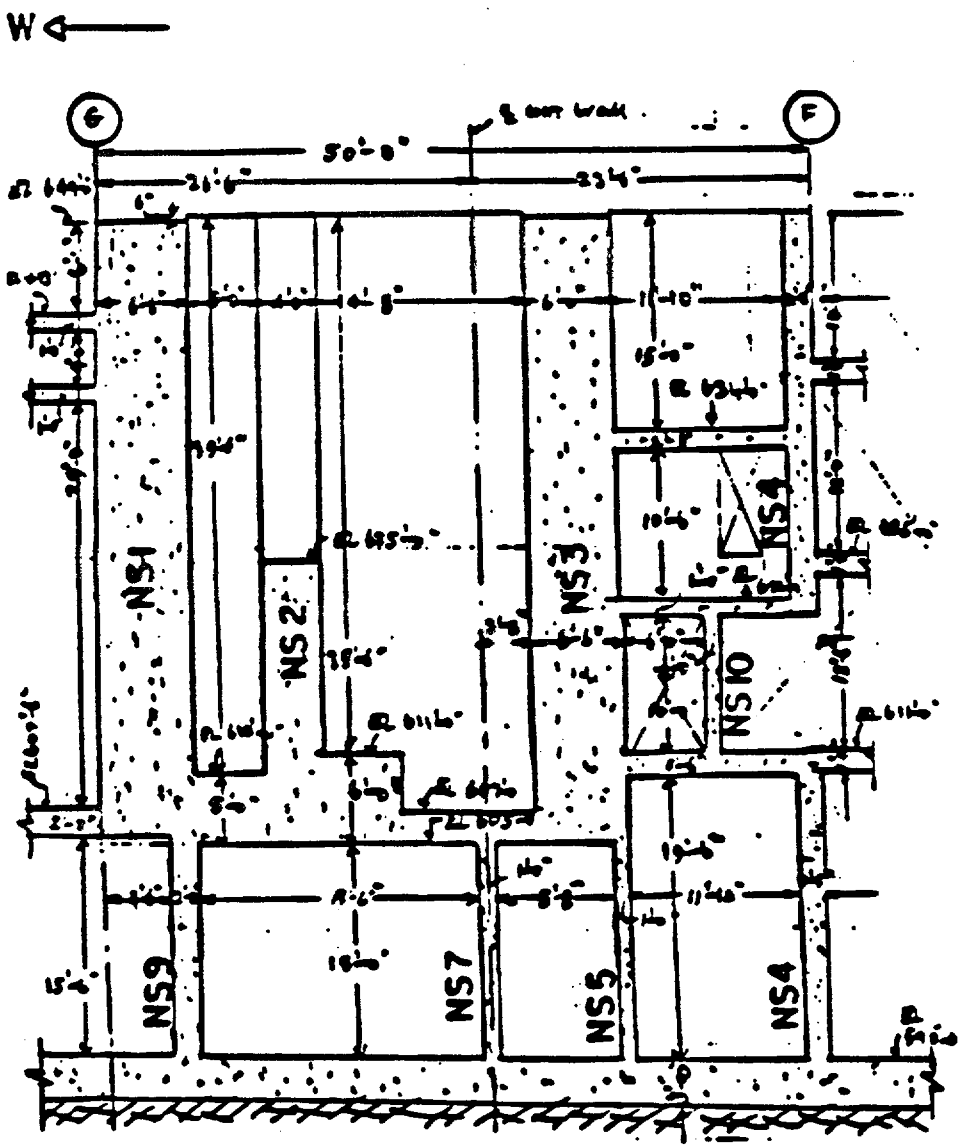


Figure 3-7. Section H-H - Elevation 590 ft to 649 ft

the pool is recessed to accommodate a shipping cask. Adjacent to the spent fuel pool and on the west side are two tilt pits measuring 21 ft x 5 ft on the inside, separated from the main pool by a 4-ft-thick reinforced concrete wall. A cutout in this wall approximately 2 ft 6 in wide and extending down from the operating floor elevation to elevation 625 ft serves the purpose of a gate to transfer spent fuel bundles from the south tilt mechanism to the spent fuel pool. The north tilt pit is now used for storing additional spent fuel. The gate between the north tilt pit and the main pool is always open when spent fuel is stored in the north tilt pit.

3.2.2 Spent Fuel Pool Racks Arrangement

The spent fuel storage pool and north tilt pit rack arrangement is shown in Figure 3-8. Fuel storage is divided into two regions. Region I (422 locations) consists of existing racks with high density fuel assembly spacing obtained by utilizing a neutron absorbing material and is normally used for core off-loading. Region II (470 locations) consists of new racks with high density fuel assembly spacing and provides normal storage for spent fuel assemblies meeting required burnup considerations. Region I is designed to accommodate irradiated and nonirradiated fully enriched fuel. Region II is designed to accommodate irradiated fuel. Normal placement of fuel in Region II is determined by burnup calculations and is controlled administratively.

3.2.3 Description of the New (Region II) Spent Fuel Racks

The new (Region II) storage racks consist of stainless steel cells assembled in a checkerboard pattern with a 9.17-in centerline-to-centerline spacing, producing a honeycomb-type structure as shown in Figure 3-9. These racks use a neutron absorbing material, Boraflex, which is attached to each cell sidewall by a stainless steel wrapper. The cells are welded to a base support assembly and to one another to form an integral structure. This design is provided with leveling screws which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The fuel rack assembly consists of two major sections which are the base support assembly and the cell assembly. Figures 3-10 through 3-12 illustrate

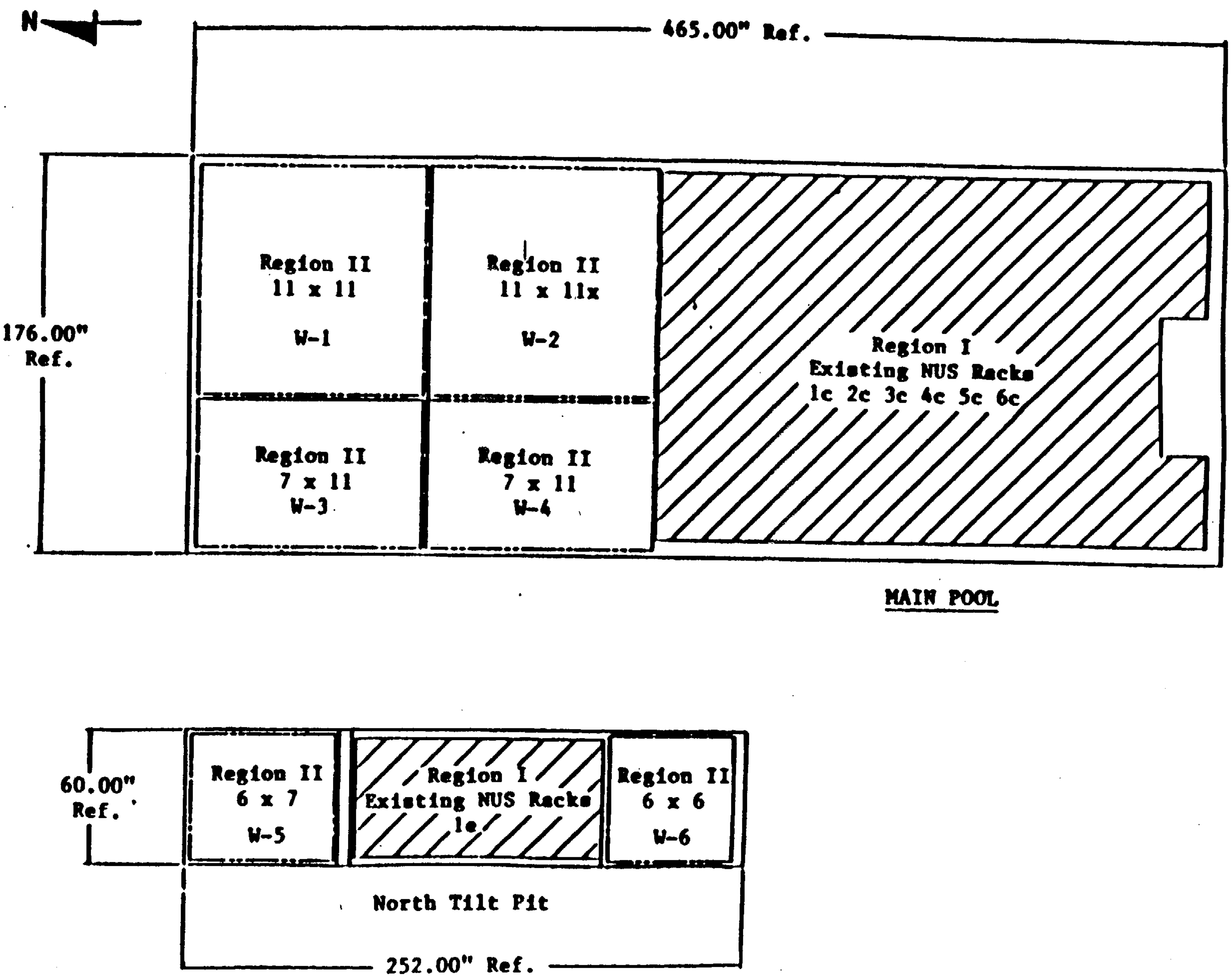


Figure 3-8. Spent Fuel Pool Arrangement

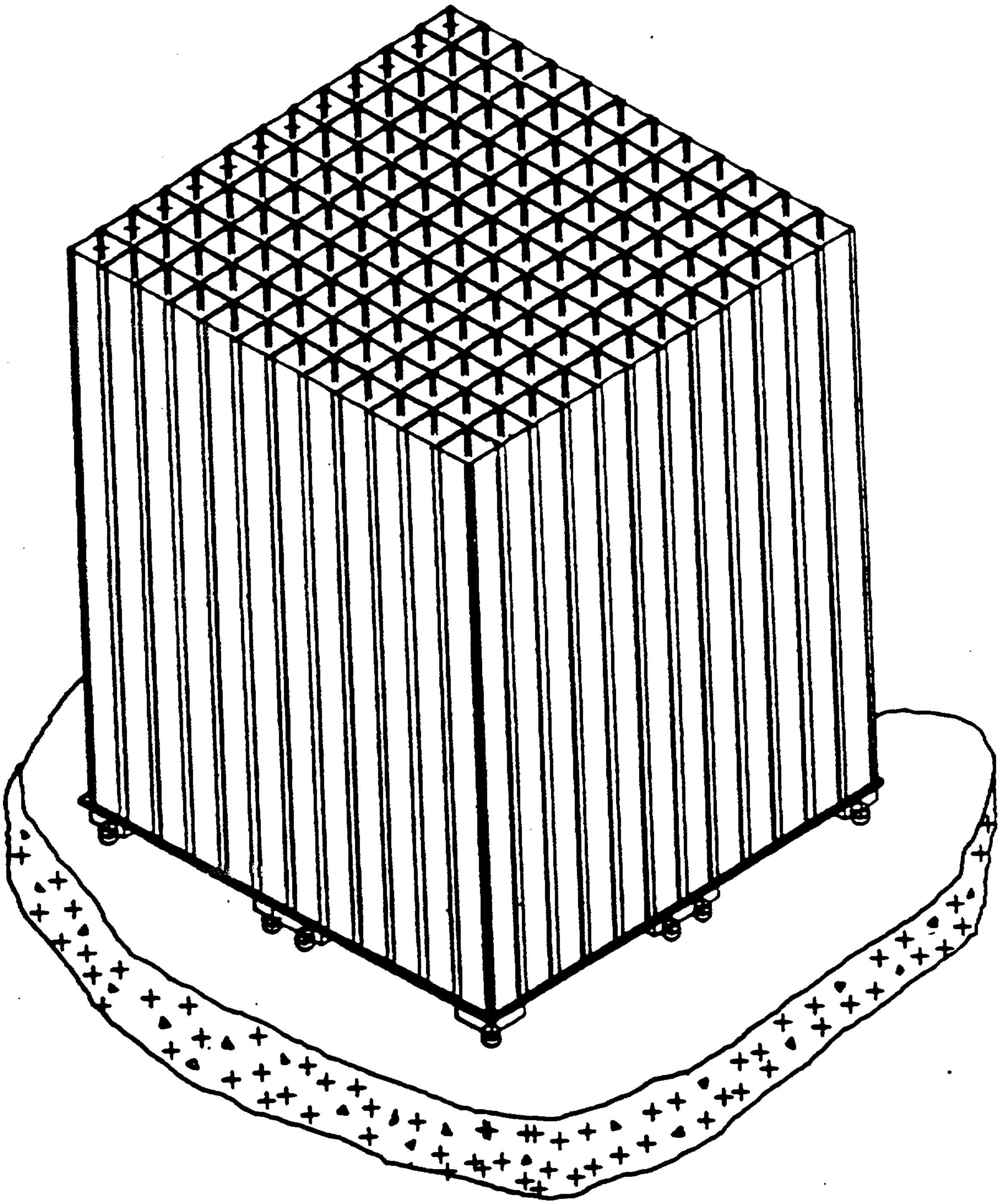


Figure 3-9. Region II Fuel Storage Rack Module

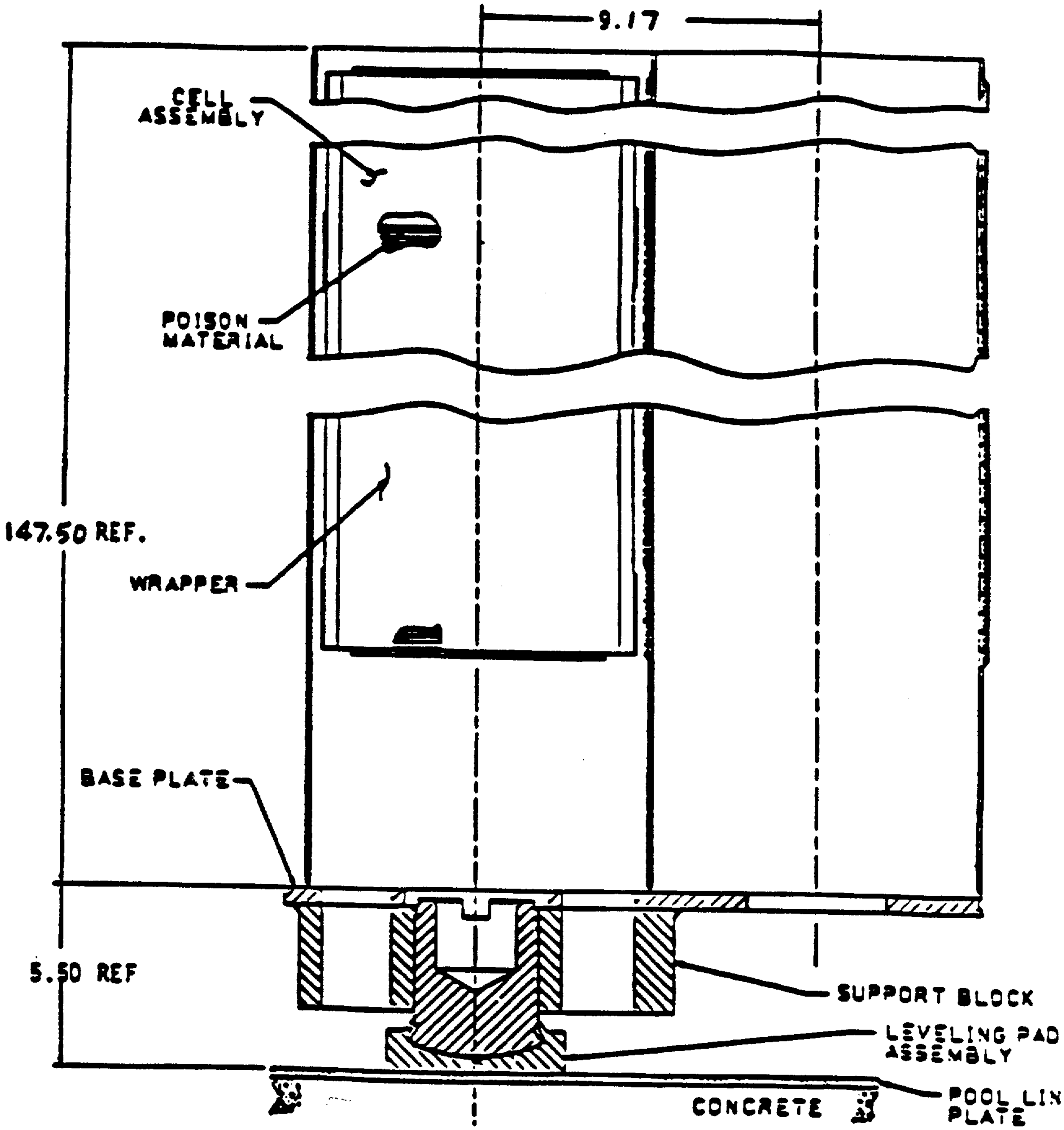


Figure 3-10. Region II Module Cross Section

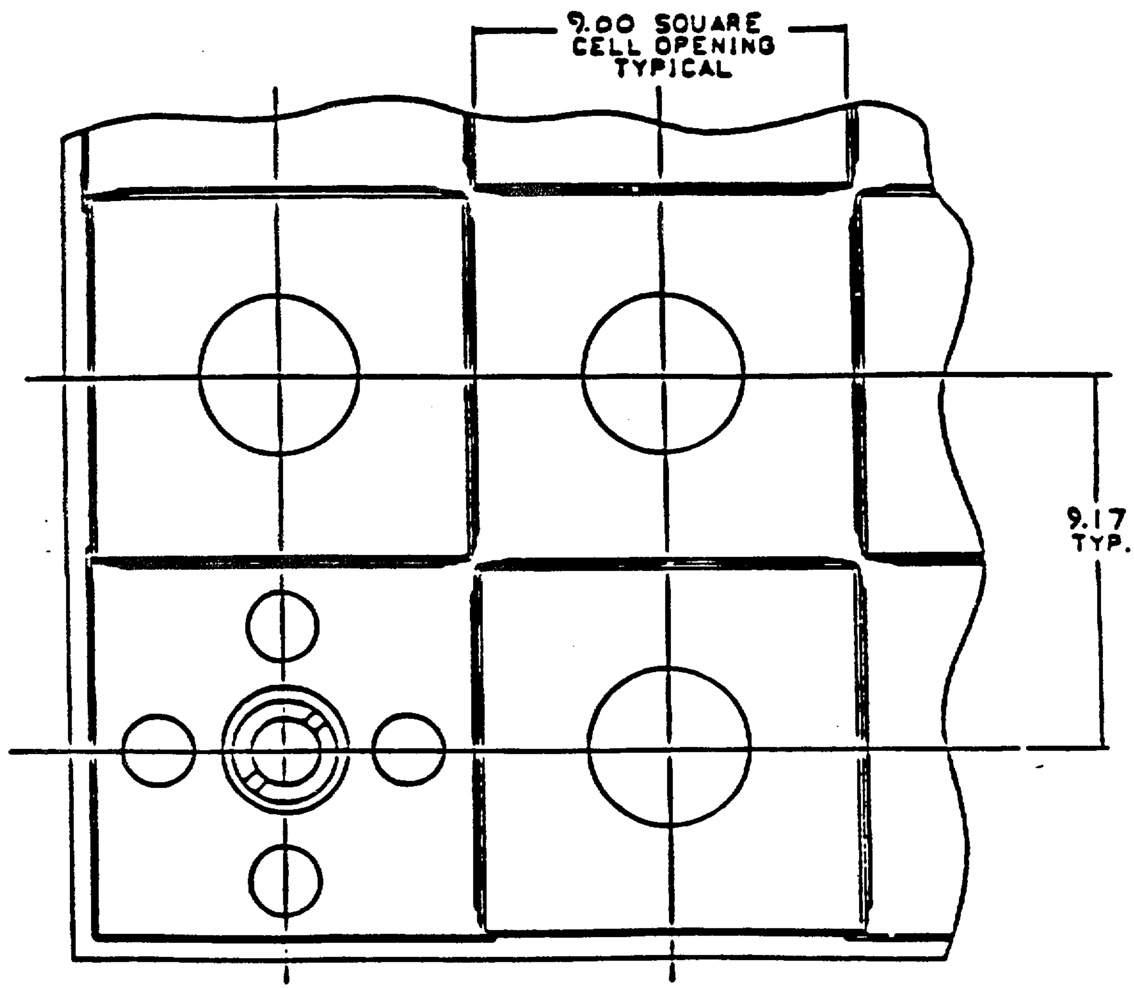


Figure 3-11. Region II Module Top View

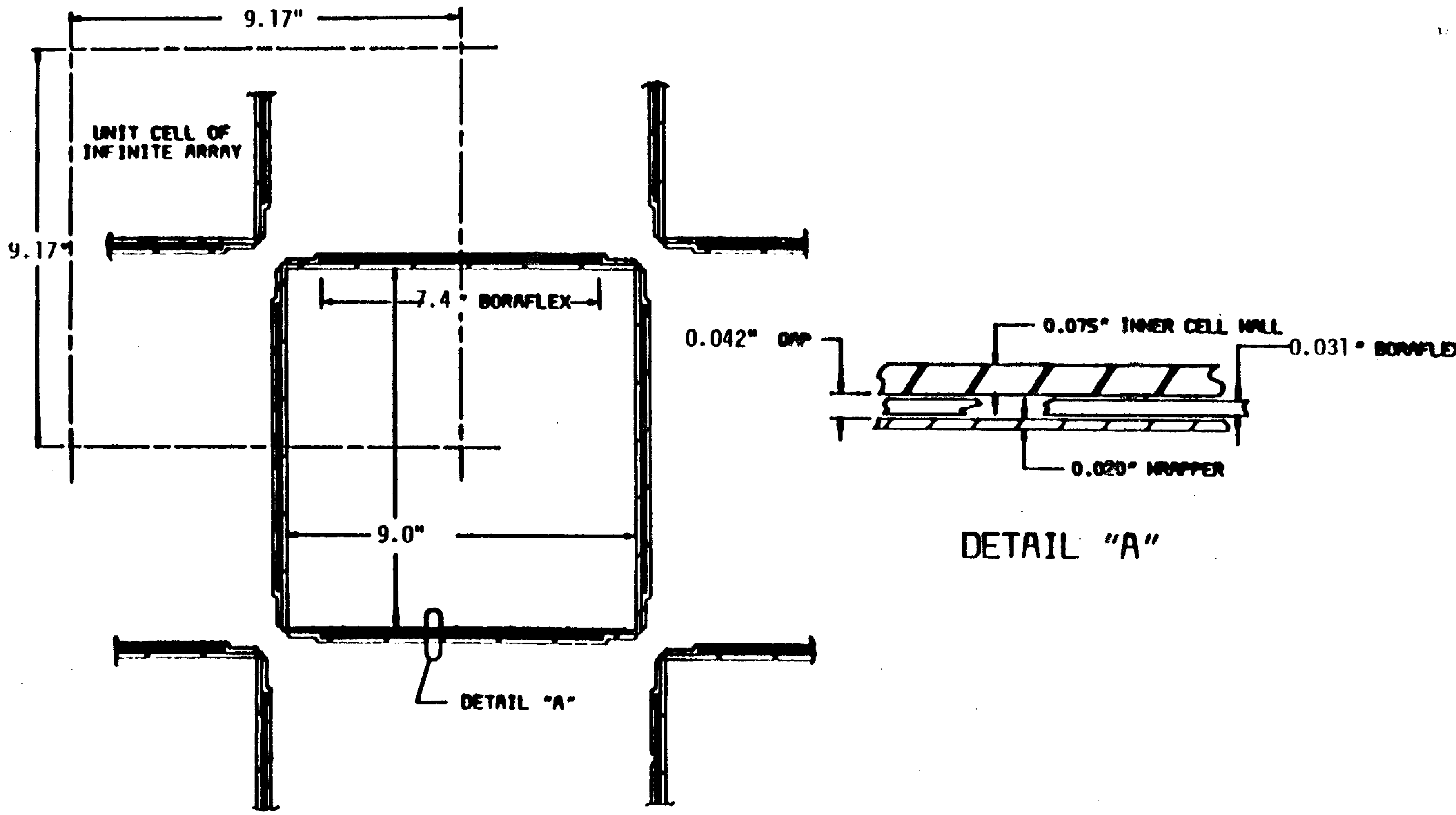


Figure 3-12. Nominal Dimensions for the Region II Storage Cells

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

The stainless steel wrapper is attached to the cell sidewall by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of the Boraflex to the pool environment. Depending on the criticality requirements and location within the rack array, some cells have a Boraflex/wrapper assembly on four sides, three sides, or two sides, as required by the analysis. The new rack module data are presented in Table 3-1.

3.3 DESIGN CRITERIA OF NEW SPENT FUEL RACKS

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

A list of design criteria for the new racks is given below:

a. The racks are designed in accordance with the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," [2] and Standard Review Plan (SRP) Section 3.8.4.

b. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor k_{eff} is $\leq$ 0.95 including all uncertainties and under all credible conditions.

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

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 loads and load combinations specified in Section IV-4 of the OT Position Paper [2].

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

Table 3-1. Rack Module Data

	Region II
Number of Storage Locations	470*
Number of Rack Arrays	2 (11 x 11) 2 (11 x 7) 1 (7 x 6) 1 (6 x 6)
Center-to-Center Spacing (inches)	9.17
Cell Inner Diameter (in)	9.00
Type of Fuel	CE 15 x 15 Exxon 15 x 15
Rack Assembly Dimensions (in)	(11 x 11) 102 x 102 x 153 (11 x 7) 102 x 65 x 153 (7 x 6) 65 x 56 x 153 (6 x 6) 56 x 56 x 153
Dry Weights (lb) Per Rack Assembly	13,3000 (11 x 11) 8,500 (11 x 7) 4,600 (7 x 6) 4,000 (6 x 6)

*Plus four locations inaccessible due to water inlet pipe.

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 within the rack array. There is no space between storage locations since the cells are welded to each other. Therefore, a fuel assembly can only be inserted in designated storage locations.

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

3.4 FINITE ELEMENT MODELING AND SEISMIC ANALYSIS OF SPENT FUEL RACK MODULES

The seismic and stress analysis of the spent fuel rack modules considered the various conditions of full, partially filled, and empty fuel assembly loadings. The racks were evaluated for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) conditions and meet Seismic Category I requirements. A detailed stress analysis was 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 were evaluated to determine that under all loading conditions they do not impact each other, the pool walls, or the existing Region I racks.

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

The time-history analysis was performed on a single cell nonlinear model with the effective properties of an average cell within the rack module. The nonlinear model is shown in Figure 3-13.

The effective single-cell properties were obtained from a structural model of the rack modules, as shown in Figure 3-14.

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

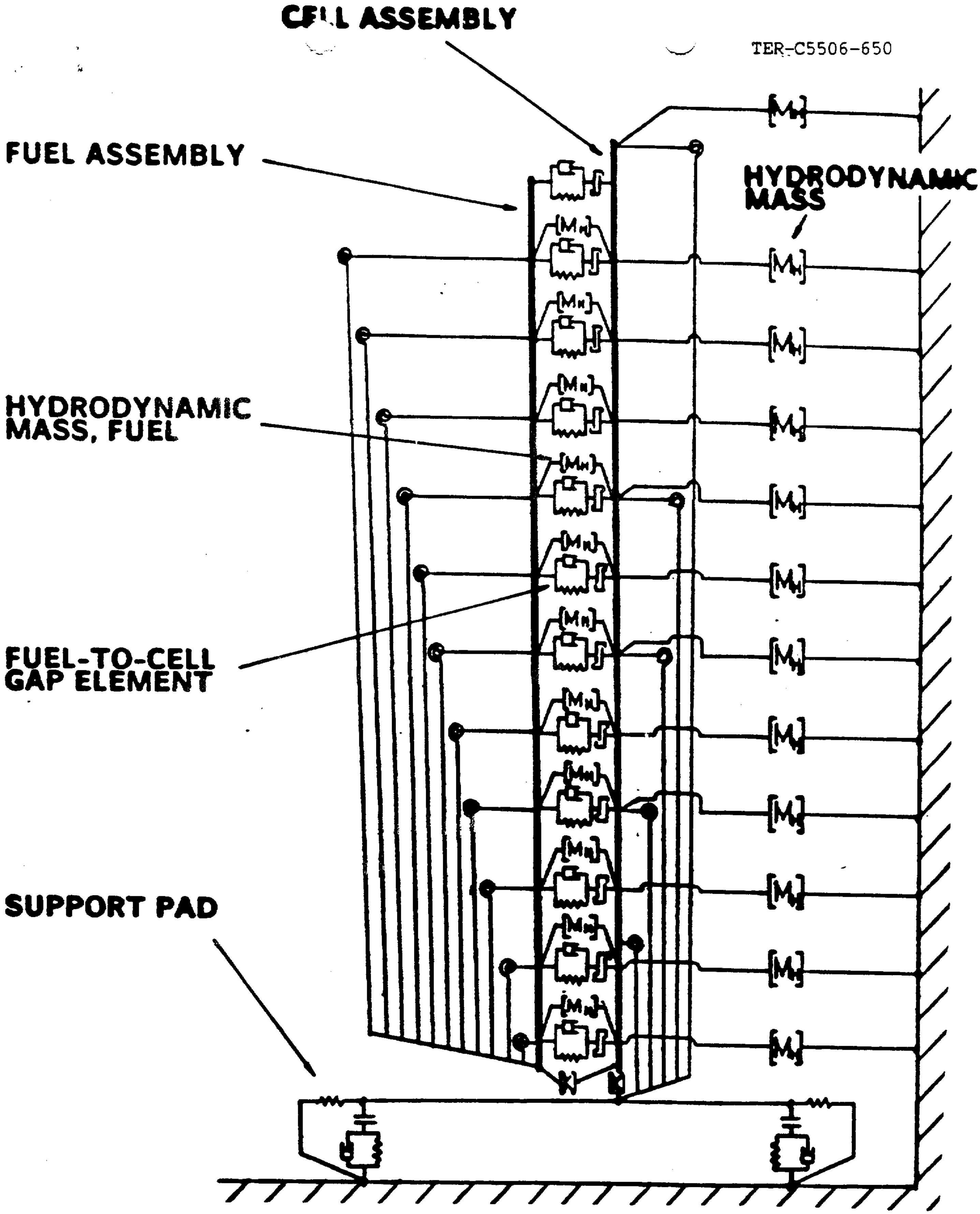


Figure 3-13. Nonlinear Seismic Model

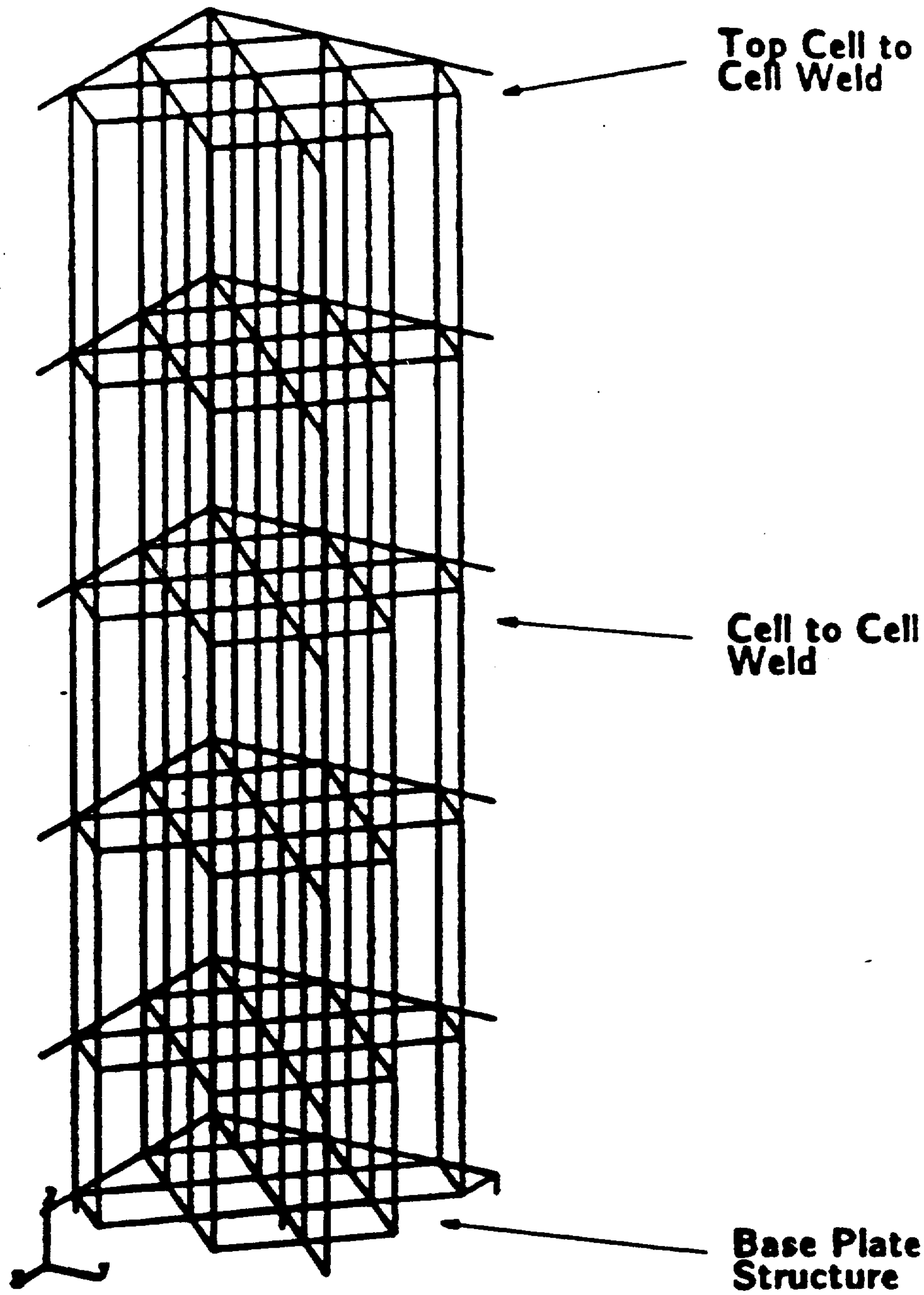


Figure 3-14. Structural Model of Typical Fuel Rack

3.4.1 Three-Dimensional Linear Structural Model

The structural model, shown in Figure 3-14, is a finite element representation of the rack assembly consisting of beam elements interconnected at a finite number of nodal points, and general mass matrix elements. The beam elements represent the beam action of the cells, the stiffening effect of the cell to cell welds, and the supporting effect of the support pads. The general mass matrix elements represent the hydrodynamic mass of the rack module. The beams which represent the cells are loaded with equivalent seismic loads and the model produces the structural displacements and internal load distributions necessary to calculate the effective structural properties of an average cell within the rack module. In addition, the stiffness properties and the internal load and stress distributions of this model are used to calculate stress peaking factors to account for the load gradients within the rack modules.

3.4.2 Two-Dimensional Nonlinear Seismic Model

3.4.2.1 Model Description

The nonlinear seismic model, shown in Figure 3-13, is composed of the effective properties from the structural model with additional elements to account for hydrodynamic mass of the fuel, the gap between the fuel and cell, and the support pad boundary conditions of a free-standing rack. The elements of the nonlinear model are as follows:

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

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

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

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

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

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

3.4.2.2 Assumptions Used in the Seismic Analysis

As stated in the licensing report [1] and the Licensee's response (dated July 24, 1986) [3] to the NRC's request for additional information, the following basic assumptions were used in the seismic analysis of the spent fuel racks two-dimensional nonlinear model:

- The nonlinear model was run with simultaneous inputs of the vertical and the most limiting horizontal acceleration time-history values.
- A structural damping value of 2% was used for both OBE and SSE seismic loading conditions.
- Analysis was performed using lower and upper limits of static friction coefficients (0.2 and 0.8, respectively) between rack support pads and pool floor.
- The fluid damping was conservatively neglected.
- The analysis included effects of water in the pool, such as fluctuation of pressure due to acceleration and sloshing.
- The seismic analysis treated the racks as if they were hydrodynamically coupled (move in phase).
- The internal loads and stresses from the seismic model were adjusted by peaking factors from the structural model to account for the stress gradients through the rack module.
- The maximum stresses from each of the three seismic events were combined by the square-root-of-the-sum-of-the-squares (SRSS) method.
- The minimum gap (clearance space) between each adjacent rack module was 1.50 in. The minimum gap between the rack modules and the pool walls was 1.80 in.

The assumptions listed above were found to be acceptable in general. It should be noted, however, that effects of torsional moments due to partially loaded racks were not captured by the two-dimensional nonlinear seismic model. Based on seismic analysis results and using best engineering judgment, it has been concluded that ignoring torsional moment effects would not influence the overall conclusions.

3.4.2.3 Calculation of Hydrodynamic Mass

As stated in the Licensee's response [3], the hydrodynamic mass between the rack cells and the pool wall was calculated by evaluating the effects of the gap between the rack modules and the pool wall using a method outlined by R. J. Fritz [4]. The adjacent racks were considered to respond in phase during earthquake events due to the small clearance (or gap) between racks and the high ratio of rack-to-gap size. Therefore, the seismic analysis treated the racks as if they were hydrodynamically coupled (moved in phase), which yields the maximum displacements of the racks. The hydrodynamic mass between the fuel assembly and the cell walls was based upon the fuel rod array size and cell inside dimensions using the technique of potential flow and kinetic energy. The hydrodynamic mass was calculated by equating the kinetic energy of the hydrodynamic mass with the kinetic energy of the fluid flowing around the fuel rods. The concept of kinetic energy of the hydrodynamic mass is discussed in a paper by D. F. DeSanto [5].

The applications of Fritz's method [4] for hydrodynamic coupling effects between rack modules and a pool wall is considered acceptable as long as the vibratory seismic displacements of the racks remain small compared to the fluid cavity (clearance or gap dimension).

3.4.2.4 Evaluation of Impact Spring Stiffness and Impact Damping

The impact spring stiffness and impact damping values used to model impacting between a fuel assembly and the storage cell walls were determined by testing [3]. The tests were performed conservatively in air since water tends to increase the damping effects from those of air. During tests, a weight was dropped onto a fuel assembly spacer grid mounted vertically to a load cell. The top end of the spacer grid was free. Sections of fuel rod cladding were inserted into the spacer grid to simulate the fuel's effects on stiffness and damping. A displacement transducer was attached to the drop weight to measure the relative deformation between the spacer and the drop weight. The results of this test, including the spacer impact stiffness and damping, are summarized in Table 3-2. The spacer impact stiffness and damping values were used to determine the properties of the fuel-to-cell gap elements of the nonlinear seismic model (Figure 3-13). The methodology and values used by the Licensee are acceptable.

Table 3-2. Summary of Impact Spring Stiffness and Impact Damping Between Fuel Assembly and Cell Wall

Drop Height of Weight (in)	0.25		0.50	
Direction Relative to Spacer Orientation	X	Y	X	Y
Natural Frequency (Hz)	31.6	21.0	26.2	21.2
Spacer Impact Stiffness (lb/in)	14,544	6,402	9,970	6,510
Spacer Impact Damping (% of Critical Damping)	15.8	12.3	19.0	17.7

3.4.2.5 Friction Coefficient Between Rack Support Pads and the Pool Liner

Two static friction coefficients were used by the Licensee in the seismic analysis to simulate possible relative displacement between rack support pads and the pool liner. The maximum sliding distance (rack base horizontal displacement) of the rack module was obtained using a minimum friction coefficient of 0.2. The maximum rack loads and structural deflections were obtained using a maximum friction coefficient of 0.8. Based on numerous experimental tests on stainless steel/stainless steel water-lubricated sliding systems, Rabinowicz [6] concluded that the mean friction coefficient anticipated is 0.523, and the lowest friction coefficient likely to be encountered is 0.349. The range (0.2 and 0.8) of friction coefficient used by the Licensee, however, appears to be sufffficient to cover all eventualities and therefore is acceptable [7].

3.4.3 Seismic Loading

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

An operating basis earthquake (OBE) at the site having a peak horizontal ground acceleration of 0.10 g, and a safe shutdown earthquake (SSE), having a peak horizontal ground acceleration of 0.20 g, were used in the seismic analysis.

The acceleration time histories applied to the fuel rack models were obtained by synthesizing the 1940 El Centro earthquake such that the resulting response spectra envelop the Palisades floor response spectra [3]. The Palisades floor response spectra employed are those of the original design of the plant.

3.4.4 Finite Element Computer Code

As stated by the Licensee [3], analyses of the racks were performed on the Westinghouse Electric Computer Analysis (WECAN) Code, which has been developed over many years by Westinghouse. It is a general purpose finite element code with a great variety of static and dynamic capabilities.

The general WECAN code has been audited by the NRC Vendor Program Branch [8].

3.4.5 Integration Time Step

To determine if the solution was fully converged, a time increment study was performed. Different time increments were used, and it was shown that the results were the same for the time increments of 0.0013 seconds and 0.0025 seconds. Thus, for the seismic analysis, the time step chosen was 0.0025 seconds [3]. The time step chosen by the Licensee is acceptable.

3.4.6 Load and Load Combinations

Table 3-3 (from Reference 3) presents different load combinations and the corresponding acceptable limits (allowables) to be considered in the analysis of the spent fuel racks including those given in the NRC's OT Position Paper [2].

The loads used in the structural analysis to calculate maximum stresses in the racks were those from the nonlinear seismic model adjusted by peaking factors from the structural model to account for the stress gradients through the rack module.

The multi-direction seismic effect was considered by combining x-direction, y-direction, and z-direction loads by the SRSS method. This loading and stress analysis methodology were reviewed and found to be acceptable.

3.4.7 Evaluation of Seismic Stress Analysis Results

The Licensee's response to the NRC's request for additional information (RAI) [3] provides the main source of information for the seismic stress analysis results.

The main spent fuel pool has two 11 x 11 rack modules and two 7 x 11 rack modules, while the tilt pool has a 6 x 6 rack module and a 6 x 7 rack module. Seismic analyses were performed for both the 11 x 11 and 7 x 11 racks in the main pool. For racks in the tilt pool, a seismic analysis was performed for the 6 x 7 rack, which enveloped the response of the 6 x 6 rack. The seismic stress analysis results are discussed in the following subsections.

3.4.7.1 Evaluation of Fuel Rack Sliding, Lift-Off, and Overturning

The Licensee indicated that the maximum single rack displacement including elastic distortion and tipping is 0.2579 in, and the maximum single rack sliding displacement is 0.0053 in. The maximum relative displacement between adjacent racks is 0.439 in. This value is much less than the minimum available 1.50-in clearance space. Thus, impact between adjacent rack modules or between a rack module and the pool will not occur.

The maximum pad (mounting foot) lift-off from the pool floor is 0.342 inches. This pad was modeled using an impact/gap element (see Figure 3-13) which allows impact to be accounted for in the dynamic analysis. The loads developed from this dynamic analysis were, in turn, used in the stress analysis.

Table 3-3. Loads and Load Combinations [3]

Load Combination(1)	Acceptance Limit(2)
D + L	Normal limits of NF 3231.1a
D + L + P_f	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + T_o	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_o + E	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_a + E	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_o + P_f	Lesser of $2S_y$ or S_u stress range (see Note 3)
D + L + T_a + E'	Faulted condition limits of NF 3231.1c (see Note 4)
D + L + F_d	The functional capability of the fuel racks shall be demonstrated

Notes:

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

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

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

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

For the evaluation of rack stability, the rack was evaluated for both partially and fully loaded conditions. It was determined that the partial loading of two rows of fuel, coupled with the limiting condition of the six-cell direction of the rack (i.e., the side of the rack comprised of six storage cells), yielded a minimum safety factor against overturn of 32. This value is much greater than the 1.5 minimum required by the OT Position Paper [2].

3.4.7.2 Evaluation of Maximum Rack Stresses

The stress analysis results of the nonlinear seismic model were combined according to Table 3-3 (loads and load combinations) to determine the minimum margin of safety of each structural component of the new spent fuel racks. Table 3-4 (from Reference 3) provides a summary of the maximum computed stresses in the rack structure (cell assembly) and support structure (support pad assembly) along with the corresponding allowable values and their margins of safety for the controlling normal and upset (OBE) load conditions. Evaluation of the reported margin of safeties indicate, that for those particular rack modules investigated, the seismic stress analysis results are acceptable.

3.5 REVIEW OF SPENT FUEL POOL STRUCTURAL ANALYSIS

3.5.1 Finite Element Model of the Spent Fuel Pool

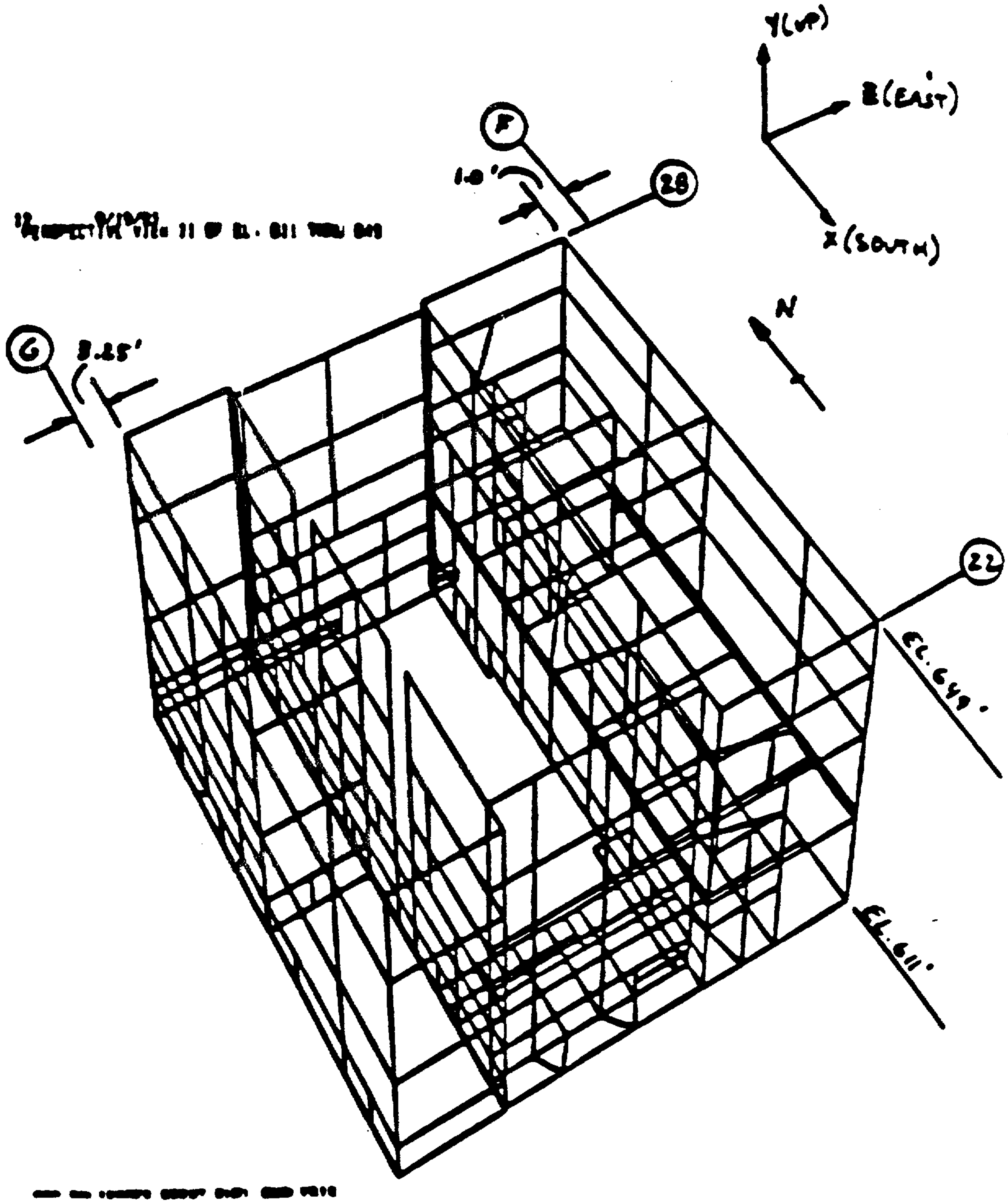
The spent fuel pool structure was analyzed using a 3-dimensional static finite element model. The model included soil, foundation mat, building structural elements, and the boundary condition to reflect structure/structure interaction. A selected perspective view of the model from elevation 611 ft through 649 ft is given in Figure 3-15. No dynamic analysis model was used to analyze the spent fuel pool structure. The finite element model was used with the NASTRAN program version 64 developed and documented by Macneal-Schwendler Corporation.

The geometry of the existing spent fuel pool structure, as used in modeling and analysis, is depicted in Figures 3-1 through 3-7.

Table 3-4. Summary of Design Stresses and Minimum Margin of Safety for New (Region II) Racks Normal and Upset Conditions (OBE) [3]

		Computed Stress (psi)	Allowable Stress (psi)	Margin of Safety
1.0	Support Pad Assembly			
1.1	Support Pad Shear	2801	11000	2.93
	Axial and Bending	11538	16500	0.43
	Bearing	9805	24750	1.52
1.2	Support Pad Screw			
	Shear	8030	11000	0.37
1.3	Support Plate			
	Shear	2802	11000	2.93
	Weld Shear	16100	24000	0.49
2.0	Cell Assembly			
2.1	Cell			
	Axial and Bending	0.86	1.0**	0.16
2.2	Cell to Base Plate Weld			
	Weld Shear	17695	24000	0.36
2.3	Cell to Cell Weld			
	Weld Shear	22652	27500*	0.21
2.4	Cell to Wrapper Weld			
	Weld Shear	9053	11000	0.21
2.5	Cell Seam Weld			
	Weld Shear	19173	24000	0.25
2.6	Cell to Cover Plate Welds			
	Weld Shear	20431	24000	0.18

*Thermal plus OBE stress is limiting.
**Allowable per Appendix XVII-2215, Eq. (24), ASME III



PERSPECTIVE VIEW - EL. 611 FT THRU 649 FT.

Figure 3-15. Finite Element Model of Spent Fuel Pool Structure

Plate elements (isoparametric quadrilateral and triangular elements) were used to represent the mat, walls, and floors. Beam elements were used for beam and column structural elements. In the absence of well-defined expansion joints between the pool building and adjacent structures, elastic springs were incorporated in the modeling to reflect the adjacent structure interaction. At the base mat, each node has six soil springs (2 horizontal, 1 vertical, 2 rocking, 1 rotation about the vertical axis) to represent the soil structure interaction effect. The structural model consists of 772 nodes, 1045 elements, and 4632 static degrees of freedom (6 degrees of freedom per node).

3.5.2 Load Combinations

As stated in the licensing report [1], the following loads were considered in the evaluation of the pool integrity:

- Dead load, includes pool structures' self-weight, racks and fuel assemblies, and hydrostatic loads. In addition, all floor live loads, dead loads of adjacent structures, and superstructure crane loads are included.
- Operating basis earthquake (OBE)
- Safe shutdown earthquake (SSE)
- Operating temperatures
- Hydrostatic loads are considered for a water level at elevation 648 feet in the spent fuel pool and tilt pits.
- Sloshing effects of water - hydrodynamic loads
- Thermal loads
- Increased loading due to the additional spent fuel elements to be stored in the pool. The structural model of the pool was loaded assuming that all the individual racks were responding in phase.

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

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

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

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

where

D = Dead load defined above including hydrostatic loads

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

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

T = Thermal gradient load

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

Two additional load combinations [3] were considered to evaluate the isolated effects of the mechanical loads and to evaluate the abnormal event of a full core off-load case. The additional load combinations are:

$1.25D + 1.25E$
$1.0D + 1.0T_{ab}$

where

T_{ab} = Thermal gradient for abnormal operating condition.

3.5.3 Design Allowable Stress Limits

The design allowable stress limits outlined in "Building Code Requirements for Reinforced Concrete" (ACI 318-71) were considered the basis of evaluation for the spent fuel structure [3].

To determine the adequacy of structure, the stress criterion outlined in FSAR Update Appendix A was adopted. The allowable stresses for different load combinations considered for evaluation are:

1. $Y = \frac{1}{\phi}(1.25D + 1.25T + 1.25E)$

(Normal Operating Condition)

2. $Y = \frac{1}{\phi}(1.25D + 1.25E)$

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

(Abnormal Operating Condition)

4. $Y = \frac{1}{\phi} (1.0D + 1.0T_{ab})$

where:

D, T, T_{ab}, E, and E' are defined in Section 3.5.2

Y = Required yield strength of the material

ϕ = Yield capacity reduction factor per ACI 318-71 for both reinforcement and concrete.

3.5.4 Evaluation of Spent Fuel Pool Stress Analysis

The maximum reinforcement and concrete stresses of the critical sections in the pool walls and slabs, in the substructure walls, and in the foundation mat were identified for different load combinations. The maximum reinforcement and concrete stresses at different locations of the spent fuel pool are presented in Tables 3-5 and 3-6, respectively [3]. The reported maximum reinforcement and concrete stresses are less than the corresponding code allowables; therefore, the stress analyses are acceptable.

3.6 FUEL HANDLING CRANE UPLIFT ANALYSIS

Section 4.6.3 of the licensing report [1] states:

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

It should be noted that the reviewed report [1] does not provide the analysis stress results or the extent of the rack deformation due to the specified maximum uplift load. The main emphasis of the analysis seems to have been to demonstrate that the criticality acceptance criteria were not violated.

Table 3-5. Maximum Reinforcement Stresses [3]

LOCATION	Direction[1] Reinf Stress (ksi)	Direction[1] Element[3] No.	Direction[1] Load[4] Comb.	Direction[2] Reinf Stress (ksi)	Direction[2] Element[3] No.	Direction[2] Load[4] Comb.
MAT & SLABS						
590' (MAT)	30.00	13	2	10.44	13	2
607' - 6"	17.30	54	1	15.8	53	1
610' - 0"	35.1	72	1	12.6	70	1
611' - 0"	34.9	128	1	15.5	128	2
EAST-WEST WALLS						
EW 1	19.3	618	2	28.9	618	2
EW 2	14.5	664	1	18.8	664	2
EW 3	4.0	683	3	31.9	683	4
NORTH SOUTH WALLS						
NS 1	24.8	311	1	20.3	311	4
NS 2	17.0	357	3	22.2	357	2
NS 3	37.4	429	1	16.6	428	1
NS 4	1.1	480	1	17.0	480	2

All reinforcement stresses are below the allowable stress of 40 ksi (yield strength of ASTM-A-615, Grade 40).

1. For Mat and Slabs: Direction 1 = NS, Direction 2 = EW
2. For Walls: Direction 1 = Horizontal, Direction 2 = Vertical
3. See Attachment A of Reference 3 for element locations.
4. Load combinations are defined in Section 3.5.2.

Table 3-5. (Cont.)

LOCATION	Direction 1[1] Reinf Stress (ksi)	Direction 1[1] Element[3] No.	Direction 1[1] Load[4] Comb.	Direction 2[2] Reinf Stress (ksi)	Direction 2[2] Element[3] No.	Direction 2[2] Load[4] Comb.
SUPPORT WALLS BELOW						
NS 4	20.30	466	2	28.19	466	2
NS 5	32.53	501	1	7.51	494	2
NS 6	29.0	513	2	2.0	513	2
NS 7	18.9	526	1	2.0	526	2
NS 8	6.1	536	1	6.1	536	2
NS 9	20.7	546	1	3.9	546	1
NS 10	35.7	561	2	18.1	561	2
EW 4	10.5	690	3	2.0	690	3
EW 5	16.6	696	2	28.0	696	2
EW 6	23.9	705	4	2.0	705	1
EW 7	35.2	715	1	2.0	715	2
EW 8	29.8	718	2	3.9	718	3
EW 9	21.1	720	2	23.7	720	2
EW 3	26.0	677	3	2.0	677	1

All reinforcement stresses are below the allowable stress of 40 ksi (yield strength of ASTM-A-615, Grade 40).

1. For Mat and Slabs: Direction 1 = NS, Direction 2 = EW
2. For Walls: Direction 1 = Horizontal, Direction 2 = Vertical
3. See Attachment A of Reference 3 for element locations.
4. Load combinations are defined in Section 3.5.2.

Table 3-6. Maximum Concrete Stresses [3]

LOCATION	Direction[1] Conc. Stress (ksi)	Direction[1] Element[3] No.	Direction[1] Load[4] Comb.	Direction[2] Conc. Stress (ksi)	Direction[2] Element[3] No.	Direction[2] Load[4] Comb.
MAT & SLABS						
590' (MAT)	0.5	13	2	0.2	13	2
607' - 6"	0.3	53	1	0.3	54	4
610' - 0"	1.3	71	2	0.5	71	1
611' - 0"	0.5	128	1	0.3	128	2
EAST-WEST WALLS						
EW 1	0.3	618	2	0.1	618	1
EW 2	0.7	664	1	0.6	664	1
EW 3	1.4	685	1	0.1	685	3
NORTH SOUTH WALLS						
NS 1	0.4	311	1	0.2	311	4
NS 2	0.6	360	1	0.6	353	4
NS 3	0.6	428	1	0.4	428	1
NS 4	0.1	480	1	0.1	480	2

All reinforcement stresses are below the allowable stress of 3 ksi (concrete stress at 28 days).

1. For Mat and Slabs: Direction 1 = NS, Direction 2 = EW
2. For Walls: Direction 1 = Horizontal, Direction 2 = Vertical
3. See Attachment A of Reference 3 for element locations.
4. Load combinations are defined in Section 3.5.2.

Table 3-6. (Cont.)

LOCATION	Direction[1] Conc. Stress (ksi)	Direction[1] Element[3] No.	Direction[1] Load[4] Comb.	Direction[2] Conc. Stress (ksi)	Direction[2] Element[3] No.	Direction[2] Load[4] Comb.
SUPPORT WALLS BELOW						
NS 4	0.1	466	2	0.3	465	3
NS 5	0.1	503	2	0.8	503	3
NS 6	0.3	512	2	0.8	512	1
NS 7	0.3	518	2	1.5	526	1
NS 8	0.1	536	1	1.4	536	2
NS 9	0.2	545	3	1.4	545	3
NS 10	0.1	561	3	0.1	561	2
EW 4	0.1	686	2	1.2	686	2
EW 5	0.1	692	2	1.0	692	2
EW 6	0.1	705	3	1.2	705	1
EW 7	0.1	715	3	0.7	715	3
EW 8	0.1	718	3	0.9	718	3
EW 9	0.1	720	2	0.5	720	2
EW 3	0.1	677	3	1.0	677	3

All reinforcement stresses are below the allowable stress of 3 ksi (concrete stress at 28 days).

1. For Mat and Slabs: Direction 1 = NS, Direction 2 = EW
2. For Walls: Direction 1 = Horizontal, Direction 2 = Vertical
3. See Attachment A of Reference 3 for element locations.
4. Load combinations are defined in Section 3.5.2.

3.7 FUEL ASSEMBLY DROP ACCIDENT ANALYSIS

The licensing report [1] states in Section 4.6.4 that:

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

For the analysis of a dropped fuel assembly, three accident conditions were postulated. The first accident condition conservatively assumed that the weight of a fuel assembly and its handling tool of 1,500 pounds impacted the top of the fuel rack from a drop height of 3 feet. Calculations showed that the impact energy is absorbed by the dropped fuel assembly, the cells and rack base plate assembly. Under these faulted conditions, credit was taken for dissolved boron in the water, and the criticality acceptance criterion is not violated.

The second accident condition was inclined drop on top of the rack. Results were the same as for the first condition.

The third accident condition assumed that the dropped assembly (1,500 lbs) fell straight through an empty cell and impacted the rack base plate from a drop height of 183 inches. The results of this analysis showed that the impact energy is absorbed by the fuel assembly and the rack base plate. Criticality calculations show the $k_{eff} \leq 0.95$ and the acceptance criterion is not violated."

Similar to the fuel handling crane uplift analysis, the licensing report [1] does not provide any structural analysis details or results of the three postulated fuel drop accidents. It appears that the main emphasis of the Licensee's analysis was to demonstrate that the criticality acceptance criteria were not violated (i.e., $k_{eff} \leq 0.95$) due to accidental deformation of the rack.

4. CONCLUSIONS

The following conclusions were reached after review and evaluation of the Licensee's submittals [1, 3] and the applicable referenced documents.

- The seismic analysis performed using the two-dimensional nonlinear model did not capture the torsional response modes of eccentric partially loaded racks. However, the stress analysis results of the new spent fuel racks indicated that the calculated margins of safety coupled with the conservative assumptions used are likely to offset the effects of ignoring the torsional modes of response.

- Impacting between the new (Region II) spent fuel rack modules and/or between a rack module and adjacent walls of the spent fuel pool is not likely to occur. The maximum computed displacements from the seismic analysis results are smaller than the existing clearances.

- Stability against overturning of the new spent fuel racks under seismic loadings appears to be assured with a large margin of safety

- The new spent fuel racks are capable of resisting internal stresses due to specified loading conditions with acceptable margins of safety.

- For the spent fuel pool concrete structure and its stainless steel liner, the maximum computed stresses including those imposed by the new rack modules are within the specified allowables.

- For the fuel assembly drop accident analysis and the fuel handling crane uplift analysis, no details pertinent to structural analysis methodology and results were submitted by the Licensee. However, in both cases, the Licensee stated that the criticality acceptance criteria were not violated (i.e., $k_{eff} \leq 0.95$) due to accidental deformation of the rack.

5. REFERENCES

1. Revised Safety Analysis Report, "Spent Fuel Storage Modifications," Consumers Power Company, Palisades Plant, Docket No. 50-255 October 16, 1986 including Amendment dated December 19, 1986

2. OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, U.S. Nuclear Regulatory Commission, January 18, 1979

3. Consumers Power Company, "Spent Fuel Pool Storage Capacity Expansion, Response to Questions Transmitted by NRC Letter Dated April 25, 1986," July 24, 1986

4. R. J. Fritz "The Effect of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, pp. 167-173, February 1972

5. D. F. DeSanto "Added Mass and Hydrodynamic Damping of Perforated Plates Vibrating in Water," ASME Journal of Pressure Vessel Technology, Vol. 103, p. 175, May 1981

6. E. Rabinowicz "Friction Coefficient Value for a High Density Fuel Storage Stystem," Report to General Electric Nuclear Energy Programs Division, November 23, 1977

7. E. Rabinowicz "Friction Coefficients of Water Lubricated Stainless Steels for a Spent Fuel Rack Facility," Report to Boston Edison Company, January 24, 1977

8. Letter from G. Zech, Chief, Vendor Program Branch (USNRC) to J. Gallagher, General Manager (Westinghouse Electric Corp.) dated January 7, 1985 (Docket No. 99900404/84-03)

7590-01

U.S. NUCLEAR REGULATORY COMMISSION

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

NOTICE OF ISSUANCE OF AMENDMENT TO

PROVISIONAL OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 105 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised the license for operation of the Palisades Plant (the facility) located in Van Buren County, Michigan. The amendment is effective as of the date of its issuance.

The license amendment provides Technical Specifications applicable to an expanded storage capability for spent fuel at Palisades Plant. This expansion is to be accomplished by installing new storage racks in approximately one-half of the spent fuel pool. The modification will increase the spent fuel storage capacity of Palisades from 798 to 892 fuel assemblies, thus allowing a full core discharge capability for two fuel cycles (Cycle 8 and Cycle 9), longer than with existing racks. The spent fuel storage pool will be divided into two regions. Region I contains the existing storage racks which have a nominal center-to-center spacing of 10.25 inches and is designed to accommodate non-irradiated, fuel. Region II will contain the new racks which have a nominal center-to-center spacing of 9.17 inches. Placement of fuel in Region II is restricted by burnup and enrichment limits.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulation. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on November 18, 1986, 51 FR 41711. No request for a hearing or petition for leave to intervene was filed following this notice.

Also, in connection with this action, the Commission prepared an Environmental Assessment dated July 24, 1987, and Notice of Environmental Assessment and Finding of No Significant Impact was published in the FEDERAL REGISTER on July 20, 1987 (52 FR 27267).

For further details with respect to this action, see (1) the application for amendment dated February 20, 1986, supplemented by submittals dated April 16 and 24, July 24, October 16 and December 19, 1986 and April 23, 1987, (2) Amendment No. 105 to License No. DPR-20, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Van Zoeren Library, Hope College, Holland, Michigan 49423. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Bethesda, Maryland, this 24th day of July, 1987.

Thomas V. Wambach

Thomas V. Wambach, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects