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Posted
Amdt. 120
to DPR-56

Docket Nos. 50-277
and 50-278

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
P.O. Box 8699
Philadelphia, Pa. 19101

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment No. 116to Facility Operating License No.DPR-44 and Amendment No. 120 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. The amendments consist of changes to the Technical Specifications (TSs) in response to your letter dated June 13, 1985, as supplemented.

These amendments allow spent fuel pool storage capacity expansion from 2,608 to 3,819 for each Unit's spent fuel pool. The expansion is to be achieved by reracking with newer higher density racks.

The request for these amendments was individually noticed on December 12, 1985 (50 FR 50873). No comments were received relevant to these amendments.

By letter dated December 26, 1985, you provided revised thermal-hydraulic calculations based upon a new estimate of the number of fuel assembly moves made per day. These revised calculations are currently under staff review and will be the subject of a future NRC action. The current license amendments authorizing the spent fuel pool storage capacity expansion from 2,608 to 3,819 storage spaces for each unit have been issued based on the assumed thermal loadings and the thermal-hydraulics calculations contained in your June and August 1985 submittals.

Mr. Edward G. Bauer, Jr.

Copies of the Safety Evaluation and Environmental Assessment related to this action are also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by/

Gerald E. Gears, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:

1. Amendment No. 116 to DPR-44
2. Amendment No. 120 to DPR-56
3. Safety Evaluation

cc: w/enclosures
See next page

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

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated June 13, 1985, as supplemented by letters dated August 1, 1985, October 9, 1985 and January 30, 1986 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 2.C.(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 116, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
242	242
	242a

5.5 FUEL STORAGE

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Spent fuel shall only be stored in the spent fuel pool in a vertical orientation in approved* storage racks.

D. The average fuel assembly loading shall not exceed 17.3 grams U-235 per axial centimeter of total active fuel height of the assembly.

5.6 SEISMIC DESIGN

The station Class I structures and systems have been designed for ground accelerations of 0.05g (design earthquake) and 0.12g (maximum credible earthquake)

*By letter dated February 19, 1986, The Commission's granted approval limited to certain specfic high density storage racks and methods of storage for Unit 2 spent fuel pool.

5.5.C BASES

This approval is limited to those storage racks and methods of storage described in Licensee's application dated August 1, 1985, October 9, 1985 and January 30, 1986 and in Commission staff documents "Safety Evaluation by Office of Nuclear Reactor Regulation Supporting Amendment Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 116 and 120 to Facility Operating Licenses Nos.DPR-44 and DPR-56" and "Environmental Assessment by the Office of Nuclear Reactor Regulation Relating to the Modification of the Spent Fuel Storage Racks, Facility Operating Licenses Nos.DPR-44 and DPR-56" for Peach Bottom Atomic Power Station Units 2 and 3, Docket Nos. 50-277 and 50-278, dated February 18, 1986.

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

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated June 13, 1985, as supplemented by letters dated August 1, 1985, October 9, 1985 and January 30, 1986 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 2.C.(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 120, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
242	242
	242a

5.5 FUEL STORAGE

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Spent fuel shall only be stored in the spent fuel pool in a vertical orientation in approved* storage racks.

D. The average fuel assembly loading shall not exceed 17.3 grams U-235 per axial centimeter of total active fuel height of the assembly.

5.6 SEISMIC DESIGN

The station Class I structures and systems have been designed for ground accelerations of 0.05g (design earthquake) and 0.12g (maximum credible earthquake)

*By letter dated February 19, 1986, The Commission's granted approval limited to certain specfic high density storage racks and methods of storage for Unit 3 spent fuel pool.

5.5.C BASES

This approval is limited to those storage racks and methods of storage described in Licensee's application dated August 1, 1985, October 9, 1985 and January 30, 1986 and in Commission staff documents "Safety Evaluation by Office of Nuclear Reactor Regulation Supporting Amendment Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 116 and 120 to Facility Operating Licenses Nos.DPR-44 and DPR-56" and "Environmental Assessment by the Office of Nuclear Reactor Regulation Relating to the Modification of the Spent Fuel Storage Racks, Facility Operating Licenses Nos.DPR-44 and DPR-56" for Peach Bottom Atomic Power Station Units 2 and 3, Docket Nos. 50-277 and 50-278, dated February 18, 1986.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENTS NOS. 116 AND 120 TO FACILITY OPERATING LICENSES NOS. DPR-44 AND DPR-56

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

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1.0 INTRODUCTION

By letter dated June 13,1985, Philadelphia Electric Company (the licensee or PECo) made application for approval to install and use new high density spent fuel racks at Peach Bottom Atomic Power Station, Units 2 and 3. Revision 1 to the application was submitted by letter dated August 1, 1985 in order to include some confirmatory calculations. Further information in response to staff questions was provided in letters dated October 9, 1985 and January 30, 1986. The proposed action would increase the spent fuel pool storage capacity in each unit from 2608 to 3819 storage cells by replacing existing storage racks with higher density storage racks.

1.1 Discussion

There are two spent fuel pools (SFPs) at Peach Bottom; one for each unit. The existing racks in each of these pools have 2608 total storage cells. Amendment Nos 49 and 48 for Units 2 and 3, respectively, dated November 30, 1978, increased the original SFP storage capacity from 1110 fuel assemblies to the present design of 2608 assemblies per pool. In the 1987-1988 time frame, these SFP units will lose their full-core discharge reserve storage capacity (764 fuel assemblies); and in the 1991-1992 time frame, they will no longer have the capacity to store additional fuel discharges from the operating units. The licensee, therefore, is proposing to replace the existing spent fuel storage racks with new spent fuel storage racks whose design will allow for more fuel in the same space as occupied by the current racks. The new rack structures will increase the existing spent fuel storage capacity from 2,608 to 3,819 storage cells for each unit. The following general description of the proposed action is based upon the licensee's August 1, 1985 submittal.

The proposed new racks are being designed and fabricated by Westinghouse Electric Company located in Pensacola, Florida. The new racks, designed to be free standing, will be installed by setting them on the spent fuel pool floor as the old racks are removed.

As in the previous storage rack replacement at Peach Bottom in 1978, some of the pool floor swing bolts (no longer functional) will be removed to within one inch of the fuel pool liner to avoid interference with the support feet on the new racks. Also, to avoid rack feet interference with the pool liner seam welds, leak detection trenches, sparger support brackets and support bases of removed swing bolts, some stainless steel plates will be set in place to span these items and provide a surface for the rack feet to rest. Also the end sections and diffusers of the spent fuel pool (SFP) cooling discharge piping will be removed.

2.0 EVALUATION

The "Spent Fuel Storage Capacity Modification Safety Analysis Report" provided by the licensee on June 13, 1985 and revised on August 1, 1985, in support of this application for approval was the basis for the NRC staff evaluation. Supplemental information provided by the licensee is also reflected in the following Safety Evaluation which summarizes the NRC staff effort.

2.1 <u>Criticality Considerations</u>

The rack design consists of square stainless steel cylinders which are fastened together in an egg crate-like structure. A Boraflex sheet is located on each outer surface and held in place by a stainless steel wrapper which is welded to the cylinder.

The calculations of rack reactivity (K-effective) were performed by the licensee with the KENO-IV Monte Carlo code. Cross sections were generated with the AMPX system of codes using the ENDF/B-IV data base. This code package has been used in numerous fuel rack calculations and the NRC staff finds it acceptable.

The licensee's fuel rack designer (Westinghouse) has verified the application of the code by calculating a number of critical experiment configurations and comparing calculated results with the experiment. These comparisons showed essentially zero bias for the calculations with an uncertainty of 0.0032 at the 95 percent level with 95 percent confidence interval. We conclude that the calculation procedure has been suitably qualified. Calculations of the K-effective value of the racks were performed for the three types of BWR fuel assemblies to be stored in the racks - 7x7, 8x8 and 8x8R. Calculations were done for an enrichment of 3.5 w/o U-235 for each type. It was determined that 7x7 assembly was the most reactive.

Uncertainties were treated either by assuming worst case conditions or by performing sensitivity studies and obtaining appropriate values. Worst case assumptions were made for asymmetric fuel assembly position and material properties (e.g., boron loading). Uncertainty values were obtained for material thickness, and spacing and bowing tolerances. Poison particle self-shielding effects were treated as a bias in the calculations. This treatment of uncertainties meets our requirements and is acceptable.

Postulated accidents which were considered include the loss of cooling systems, dropping a fuel assembly on top of the racks and dropping of an assembly outside the periphery of the racks. These accidents either do not cause an increase in the K-effective value or the increase is small compared to the margin between the nominal K-effective and the acceptance criterion of 0.95. We conclude that proper analyses of the accident conditions have been performed.

The maximum value of K-effective for normal storage or a postulated accident condition is 0.936 including uncertainties at a 95/95 probability/confidence level. This meets our acceptance criterion of 0.95 for this quantity and is acceptable.

We conclude that the proposed high density spent fuel storage racks are acceptable with respect to criticality. This conclusion is based on the following:

1. Calculations are performed for the fuel having the maximum reactivity.
2. The calculation method has been verified against experiment.

3. Uncertainties in the calculations have been properly treated.
4. Credible accidents have been analyzed.
5. The results of the analyses meet NRC acceptance criterion for K-effective.

Finally, the Technical Specifications (TSs) for Peach Bottom limit storage in the pool to fuel having less than 17.3 grams of U-235 per centimeter of assembly length. The licensee has confirmed that this is equivalent to 3.5 w/o U-235 enrichment in the most reactive (7x7) assembly. We conclude that the proposed rack design is acceptable for storage of assemblies meeting the TS requirements.

2.2 Spent Fuel Pool Cooling and Makeup

The licensee calculated 13.14 MBTU/hr as the maximum "normal" heat load, to the pool (all spent fuel storage locations full with fuel from successive cyclic discharge) following the last refueling. The staff performed an independent calculation for the maximum "normal" heat load to the pool in accordance with the guidelines of Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," and Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System" which resulted in a value of 16.69 MBTU/hr. The licensee indicated that two of the three existing spent fuel pool cooling heat exchanger trains have a combined heat removal capability of 17.33 MBTU/hr when maintaining the bulk pool temperature at 150°F. The 150°F pool temperature is the upper limit previously approved by the staff. Thus, the heat load calculated by the licensee results in a maximum bulk pool temperature of 135°F. This value is based on assuming a single failure in the spent fuel pool cooling system which leaves two fuel pool cooling system heat exchangers in operation. This value is below the 150°F upper limit for bulk pool temperature for normal storage conditions. The pool temperature will also be below 150°F based on the staff calculated maximum "normal" heat load. Thus, we have verified that the pool temperature is maintained within acceptable limits for the maximum "normal" heat load condition. In addition, the licensee has concluded based on their analysis that no boiling would occur within the storage racks when the normal fuel pool cooling system is in operation or whenever the pool temperature is maintained at or below 150°F. The licensee calculated 23.12 MBTU/hr as the maximum "abnormal" heat load following a full core discharge, with the remaining storage spaces full with fuel from successive cyclic discharges. This "abnormal" heat load results in a maximum bulk pool temperature of 143°F, with all cooling train heat exchangers operating. Assuming the loss of all cooling, boiling would occur after 82 hours for the maximum "abnormal" heat load condition. This is a substantial time period for actions to be taken such as initiating makeup to the spent fuel pool. No upper limit for the maximum "abnormal" storage condition is established in the staff criteria, and therefore, the above temperatures are acceptable.

The spent fuel pool cooling system is normally cooled by the service water system. The licensee proposed no modifications to this system as part of this spent fuel pool expansion project.

Under emergency conditions, the reactor building cooling water heat exchangers can be manually connected to provide cooling to the spent fuel pool cooling system. It is in turn cooled by the emergency service water system. The residual heat removal system can also be utilized to supplement the spent fuel pool cooling system under abnormal heat load conditions.

The licensee has also analyzed the effects of spent fuel pool boiling on the outside environment. The licensee utilized a model similar to that previously employed for a comparable analysis on the Limerick Station to determine the offsite radiological consequences of pool boiling. The results indicated that the resulting offsite dose was a very small fraction of 10 CFR Part 100 limits and was a negligible offsite contribution. We find this analysis and it conclusion to be acceptable.

2.3 Installation of Racks and Load Handling

Currently, there is spent fuel in the Peach Bottom Units 2 and 3 spent fuel pools. However, the licensee has stated that at no time will the cask handling crane carry a spent fuel storage rack module over stored spent fuel. The licensee has committed to employ heavy load handling procedures, safe load paths and installation procedures as part of the administrative controls to preclude the potential for the mishandling of rack modules and miscellaneous heavy load items during the rerack operation over the spent fuel pool.

The licensee has also performed a load drop analysis for the rack module. The results of that analysis indicate that the proposed spent fuel pool modifications will not result in fuel damage and that the resulting radiological consequences will not be in excess of the fuel handling accident previously evaluated in the updated Peach Bottom FSAR. The postulated rack drop also would not change the minimum separation distance between the stored fuel assemblies or the concentration of fixed neutron absorbing material between the adjacent fuel assemblies. Therefore, the margin of safety to criticality will also not be affected by the postulated rack drop accident.

The licensee has committed to use the main hook of the reactor building crane for lifting the existing spent fuel storage racks and the new storage racks. The main hook and its associated load lifting system on the reactor building crane are of a single failure proof design, such that a single failure will not result in dropping the load. The auxiliary hook on the reactor building crane will be used only for lifting small miscellaneous items whose weights are less than that of a fuel assembly. This will ensure that the consequences of their being dropped is bounded by the existing FSAR fuel handling accident analysis. The refueling bridge crane will be used for lifting fuel assemblies and transferring them within the pool in accordance with the existing station approved procedures.

The licensee has stated that the cask handling crane meets the design and operational criteria of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants." We have verified that the staff's previous safety evaluation report for NUREG-0612 has found this crane in compliance with the applicable guidelines for the control of heavy loads. Therefore, we conclude that the handling of heavy loads during the spent fuel pool expansion modification will be in conformance with staff criteria and is acceptable.

Based upon the discussions in Sections 2.2 and 2.3 above, we conclude that the proposed SFP modifications for each SFP with respect to the developed heat loads, pool water temperatures, and load handling practices are in accordance with applicable criteria and are, therefore, acceptable.

2.4 Structural Design

Our evaluation of the structural aspects of the proposed modifications are based on a review performed by the staff's consultant, Franklin Research Center (FRC). The FRC Technical Evaluation Report (TER) is appended to this safety analysis and provides additional details relating to the structural evaluation.

The SFPs are reinforced concrete structures located inside the Reactor Building in an elevated position adjacent to the North (Unit 2) and South (Unit 3) sides of the drywell shield walls. The walls and floor of the SFP are lined with a stainless steel liner. This liner serves only as a water tight boundary, and it is not a structural member.

The new high density racks are stainless steel "egg-crate" structures. Each cell would contain a spent fuel assembly. Weight of the rack and fuel is transmitted to the floor of the pool through supporting legs. The racks are each free-standing on the pool floor and a gap is provided between the racks and between racks to pool wall so as to preclude impact during an earthquake.

Load combinations and acceptance criteria were compared with those found in the "Staff Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979. The existing concrete pool structure was evaluated for the new loads in accordance with the requirements of the applicable portions of NRC Regulatory Guides 1.12, 1.142, and Standard Review Plan 3.8.4. The pool structure re-analysis also uses ACI 318-83, ACI 349-80, and AISC Standards.

Loads and load combinations for the racks and the pool structure were reviewed and found to be in agreement with the applicable portions of the staff position. Additional details are provided in the appended TER.

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. The seismic loads were applied to the model in three orthogonal directions. Loads due to a fuel bundle drop accident were considered in a separate analysis. The postulated loads from these events were found to be acceptable.

The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases . The first phase is a time history analysis on a nonlinear finite element model. The second phase is a response spectrum analysis of a detailed linear three dimensional finite element model of the rack assembly. Further details on the methodology may be found in the appended TER.

Calculated stresses for the rack components were found to be within allowable limits. The racks were found to have adequate margins against sliding and tipping.

An analysis was conducted to assess the potential effects of a dropped fuel assembly on the racks and results were considered satisfactory.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension load in the fuel assembly. Resulting stresses were found to be within acceptable limits by the staff.

The existing structures were analyzed for the modified fuel rack loads using a finite element computer program. Original plant response spectra and damping values were used in consideration of the seismic loadings, and the existing SFPs are determined to safely support the loads generated by the new fuel racks.

We, therefore, conclude that the proposed rack installation will satisfy the requirements for 10 CFR Part 50, Appendix A (General Design Criteria 2, 4, 61, and 62), as applicable to structures.

2.5 Materials

The safety function of the SFPs and storage rack system is to maintain the spent fuel assemblies in a sub-critical array during all credible storage conditions. We have reviewed the compatibility and chemical stability of the materials, except the fuel assemblies, wetted by the pool water.

The spent fuel racks in the proposed expansion would be constructed entirely of Type 304 LN stainless steel, except for leveling screws which are Type 17-4 PH stainless steel and the neutron absorber material. The high density spent fuel storage racks will utilize Boraflex sheets as a neutron absorber. Boraflex consists of boron carbide powder in a rubber-like silicone polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure.

The space which contains the Boraflex is vented to the pool. Venting will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the stainless steel tube.

The pool liner, rack lattice structure and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storate tubes, and the Zircaloy in the spent fuel assemblies will not be significant because the materials are either similar or the materials are protected by highly pessivating oxide films and are therefore at similar potentials. The Boraflex is composed of non-metallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion (1).

Tests were performed at the University of Michigan (2), exposing Boraflex to 1×10^{11} rads of gamma radiation with substantial concurrent neutron flux in deionized water. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex.

The annulus space in each cell assembly which contains the Boraflex is vented to the pool. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging and swelling of the inner stainless steel tube.

The tests (1) have shown that neither irradiation, environment nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also show that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally present in the Boraflex will typically contain 0.1 wt percent of soluble boron. The tests results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble species from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell inservice surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically over the expected service life.

From our evaluation as discussed above, we conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in deionized water indicate that the Boraflex material will not undergo significant degradation during the expected service life.

We further conclude that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool is adequate based on the test data cited above, and the actual service experience in operating reactors. We have reviewed the surveillance program and we conclude that the monitoring of the materials in the SFPs, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. The materials surveillance program delineated by the licensee will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. This monitoring program will ensure that, in the unlikely situation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

We, therefore, find that the implementation of an inservice surveillance program and the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR 50 Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron neutron absorber and is, therefore, acceptable.

2.6 Spent Fuel Pool Cleanup System

The SFP cleanup system is part of the pool cooling system. It consists of a full flow (550 gpm) filter-demineralizer composed of a filter precoat powdered ion exchange resin. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels. The staff expects only a small increase in radioactivity released to the pool water as a result of the proposed modification. We, therefore, conclude that the spent fuel pool cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

2.7 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the Peach Bottom SFP racks with respect to occupational radiation exposure. The licensee estimates that the exposure for this operation will be approximately 36 man-rems. This estimate is based on the licensee's 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, termed "crud". Crud may be released to the pool water because of fuel movement during the proposed SFP rack modifications. 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, the licensee, based upon previous experience from performing similar modifications, does not expect to have significant releases of crud to the pool water during modification of the SFP racks. In addition, the purification system for the pool (SFP Cleanup System), which has maintained radiation levels in the vicinity of the pool at low levels during normal operations, will be operating during the modification of the SFP racks. The staff has evaluated the licensee's proposed crud reduction program in the SFP and finds it acceptable.

The presently installed racks will be individually lifted from the SFP and will be rinsed either with low or high pressure water to remove any loose radioactivity. The racks will then be moved to a receiving area for appropriate disposal. Currently, the licensee has proposed decontaminating the racks and then disposing of the clean material as industrial waste. Material that cannot be decontaminated will be disposed of as normal radioactive waste. Either disposal method used will follow ALARA (as-low-as-reasonably-achievable) guidelines.

Divers will be used during the SFP rack modification. The licensee has developed specific procedures using the recommendations of Regulatory Guide 8.8 to ensure that doses to the divers will be within the requirements of 10 CFR Part 20 and ALARA guidelines. The ALARA procedures for divers include: reshuffling of the spent fuel; radiation surveys after the fuel is reshuffled to map radiation zones; instruction to divers on their travel limits within the pool; and constant monitoring of divers' radiation dose.

The staff's evaluation of the Peach Bottom's proposed SFP rack modification includes a review of the manner in which the licensee will perform the modification, the radiation protection program, including the use of area and airborne radioactivity monitoring, and the use of relevant experience from other operating reactors that have performed similar SFP rack modifications. Based on this review, the staff concludes that the Peach Bottom SFP rack modification can be performed in a manner that will ensure ALARA exposures to workers.

In addition, the staff has estimated the increment in onsite occupational dose during normal operations after the pool modifications resulting from the proposed increase in stored fuel assemblies. This estimate is based upon information supplied by the licensee for occupancy times and for dose rates in the SPF area from radionuclides concentrations in the SFP water. 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. Based on present and projected operations in the SFP area, the staff estimates that the proposed modification should add less than one (1) percent to the total annual occupational radiation exposure at the plant. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational dose to ALARA levels and within the limits of 10 CFR Part 20. Thus, the staff concludes that storing additional fuel in the SFP will not result in any significant increase in dose received by workers.

2.8 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems were evaluated in the Final Environment Statement (FES) dated April 1973. There will be no change in the waste treatment systems described in Section III.2 of the FES because of the proposed modifications. There will be an expected modest increase in the loadings on the Spent Fuel Cleanup system (refer to Section 2.6-Spent Fuel Pool Cleanup System).

2.9 Radiological Consequences of Cask Drop and Fuel Handling Accidents

This portion of the staff's review was conducted in accordance with the guidance in NUREG-0800, "Standard Review Plan", Sections 15.7.4 and 15.7.5, Regulatory Guide 1.25 and NUREG-0612 with respect to accident assumptions.

The licensee has committed to follow existing technical specifications regarding allowable loads carried over stored spent fuel during the reracking procedure, and during normal operation after its completion. The staff agrees with the licensee that the change in radiological conditions which can influence accident conditions in the SFP after the increase in storage capacity will be negligible compared with that prior to the modifications. The fuel burn-up (assumed to be 40,000 MwD/MTU), pool water level, and iodine decontamination factor will remain unchanged. The Peach Bottom Safety Evaluation Report, dated August 1972, was evaluated for a less tightly packed pool. However, even though more assemblies could possibly be impacted in a dropped assembly accident with more dense arrangement, the radiological consequences of this accident will not significantly increase. Therefore, the radiological analysis of the cask drop, fuel assembly, and heavy load accident is unchanged from that previously analyzed for the existing spent fuel pool configuration (Safety Evaluation Report for Peach Bottom Units 2 and 3, August 1972). In addition, the staff has performed an independent bounding analysis based upon this modification which shows that the doses at the Exclusion Boundary and Low Population Zone will be well within the SRP 15.7.4 dose guidelines. Therefore, the staff concludes that proposed modification is acceptable.

3.0 Summary

Our evaluation supports the conclusion that the proposed modification to the Peach Bottom SFP is acceptable because:

(1) The physical design of the new storage racks will preclude criticality for any credible moderating condition.

(2) The SFP cooling system has adequate cooling capacity.

(3) The installation and use of the proposed fuel handling racks can be accomplished safely with the limit that no rack modules will be moved over any spent fuel assemblies.

(4) The installation and use of the new spent fuel racks can be done safely and will not alter the consequences of the design basis accident for the SFP, i.e., the dropping and rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory within the gap.

(5) The likelihood of an accident involving heavy loads in the vicinity of the SFP is negligible.

(6) The structural design and materials of construction are adequate to function normally for the duration of the plant lifetime and to withstand the seismic loading of the design basis earthquake.

(7) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

4.0 Environmental Considerations

A separate Environmental Assessment has been prepared pursuant to 10 CFR Part 51.

5.0 Conclusions

We have concluded, based on the consideration 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 activity will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 19, 1986

The following NRC personnel have contributed to this Safety Evaluation: W. Brooks, J. Raval, S.Kim, R.Fell, M. Lamastra, H.Gilpin, F. Witt and G.Gears

6.0 References

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

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

APPENDIX A

TECHNICAL EVALUATION REPORT

NRC DOCKET NO. 50-277, 50-278

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FRC PROJECT C5506

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FRC TASK 585

EVALUATION OF SPENT FUEL RACKS STRUCTURAL ANALYSIS

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM UNITS 2 AND 3

TER-C5506-585

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

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January 15, 1986

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CONTENTS (Cont.)

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, Division of Operating Reactors) 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 Philadelphia Electric Company's licensing report [1] on high-density spent fuel racks for Peach Bottom Units 2 and 3 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 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 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 Peach Bottom Units 2 and 3 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. ACCEPTANCE CRITERIA

2.1 APPLICABLE CRITERIA

The criteria and guidelines used to determine the adequacy of the high-density spent fuel racks and pool structures are provided in the following documents:

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

- Standard Review Plan, NUREG-0800, U.S. Nuclear Regulatory Commission

  Section 3.7, Seismic Design
  Section 3.8.4, Other Category I Structures
  Appendix D to Section 3.8.4, Technical Position on Spent Fuel Pool Racks
  Section 9.1, Fuel Storage and Handling

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

- Regulatory Guides, U.S. Nuclear Regulatory Commission

  1.29 - Seismic Design Classification

  1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

  1.61 - Damping Values for Seismic Design of Nuclear Power Plants

  1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

  1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Component Types

- Other Industry Codes and Standards

  American National Standards Institute, N210-76.

2.2 PRINCIPAL ACCEPTANCE CRITERIA

The principal acceptance criteria for the evaluation of the spent fuel racks' structural analysis for Peach Bottom Units 2 and 3 are set forth by the

NRC's OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (OT Position Paper) [2]. Section IV of the document describes the mechanical, material, and structural considerations for the fuel racks and their analysis.

The main safety function of the spent fuel pool and the fuel racks, as stated in that document, 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."

Specific applicable codes and standards are defined 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 stainless steel base metal 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.
>
> Other materials, design procedures, and fabrication techniques will be reviewed on a case-by-case basis."

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.

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

** American Institute of Steel Construction, Latest Edition.

- 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.

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 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 3.8.5.II.5 of the Standard Review Plan

(b) it can be shown that any sliding and tilting 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 MATHEMATICAL MODELING AND SEISMIC ANALYSIS OF SPENT FUEL RACK MODULES

Submerged spent fuel rack modules exhibit highly nonlinear structural dynamic behavior under seismic excitation. The sources of nonlinearity can generally be categorized by the following:

a. The impact between fuel cell and fuel assembly: The fuel assembly standing inside a fuel cell will impact its four inside walls repeatedly under earthquake loadings. These impacts are nonlinear in nature and when compounded with the hydrodynamic coupling effect will significantly affect the dynamic responses of the modules in seismic events.

b. Friction between module base and pool liner: The modules are free-standing on the pool liner, i.e., they are neither anchored to the pool liner nor attached to the pool wall. Consequently, the modules are held in place by virtue of the frictional forces between the module base and pool liner. These frictional forces act together with the hydrodynamic coupling forces to both excite and restrain the module during seismic events.

Peach Bottom Units 2 and 3 plan to utilize high density fuel racks comprising nine variations in storage capacity that are arranged in the spent fuel pools as shown in Figures 1 and 2 [1]. Data pertaining to the rack module designs are provided in Table 1. Note that the clearance space between the rack modules and the pool structure is shown in Figures 1 and 2 by the boxed dimensions. The minimum rack module to rack module clearance is 1.68 inches, as reported by the Licensee [3].

The rack modules for each unit ranged in capacity (and size) from 9 x 14 cells to 19 x 20 cells. These largest and smallest racks were chosen by the Licensee for structural dynamics analysis. Since experience indicates that, for a given rack height, the rack module with the smallest horizontal dimensions will usually yield the highest rack displacements (tipping), the Licensee's choice of modules for analysis is acceptable.

The seismic analysis was performed by the Licensee in two parts. The first part was a three-dimensional, nonlinear, time-history analysis of dynamic rack displacements employing a mathematical model of a spent fuel rack module, modeled as shown in Figure 3, to include the fuel assemblies and hydrodynamic

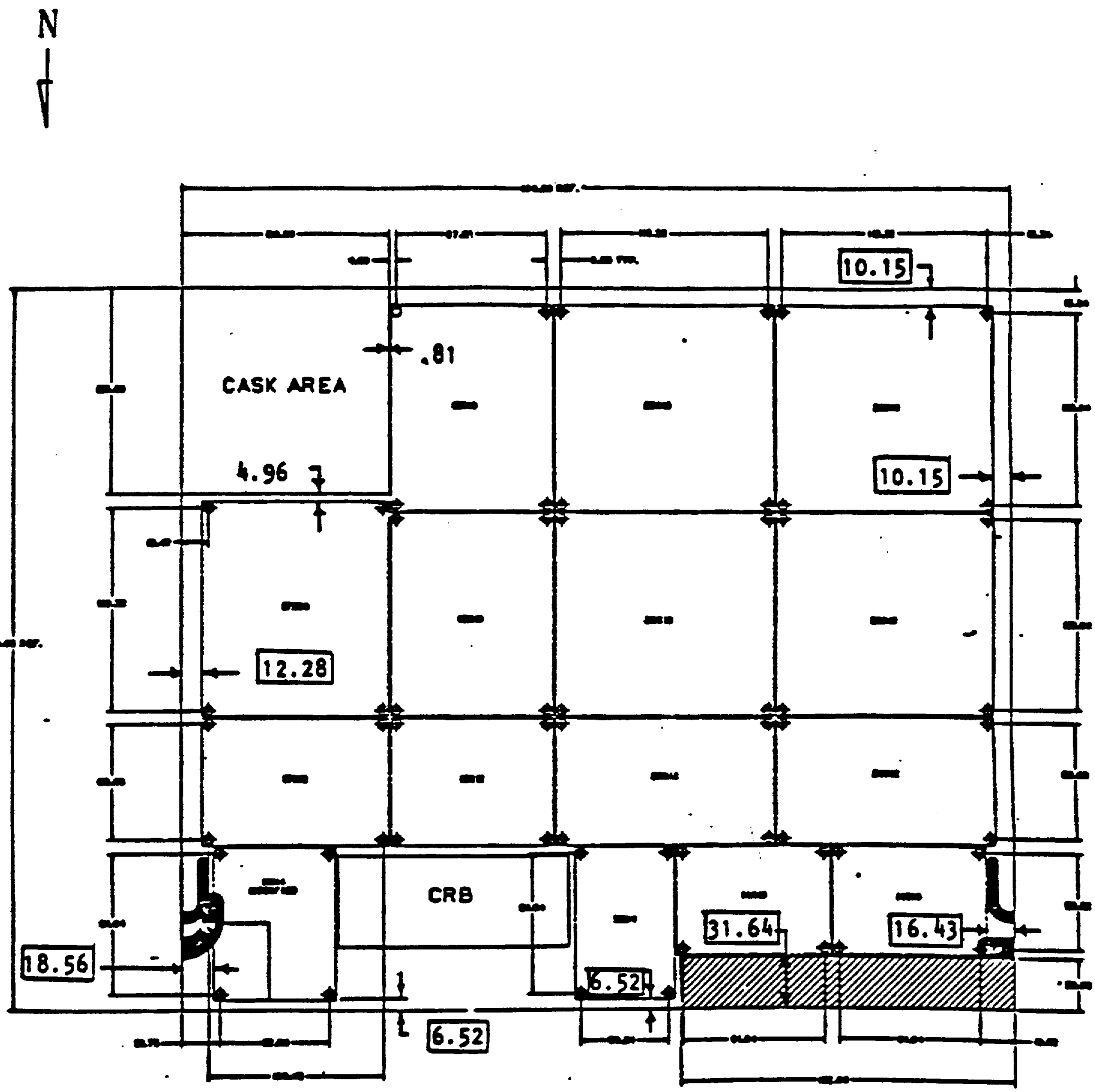

Figure 1. Spent Fuel Storage Rack Arrangement Unit 2

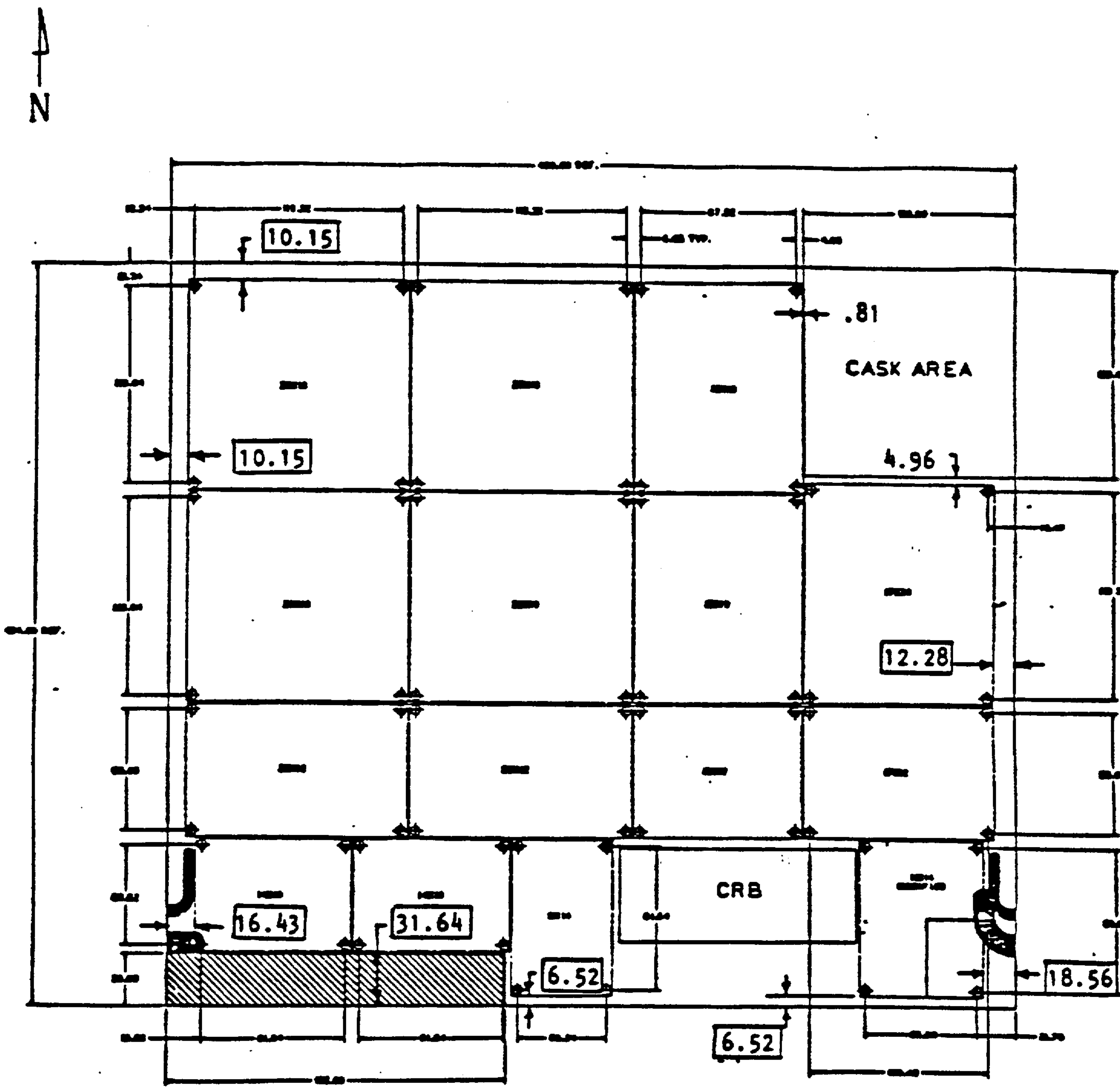

Figure 2. Spent Fuel Storage Rack Arrangement Unit 3

Table 1. Rack Module Data (Per Unit)

Qty	Array	Storage Locations	Rack Assembly Dimensions (inches)	Dry Weight (lb) Per Rack Assembly
1	9 x 14	126	54 x 89 x 180	10,000
2	10 x 14	280	64 x 89 x 180	11,200
1	11 x 14 Mod.	119	70 x 89 x 180	9,500
1	12 x 15	180	76 x 95 x 180	14,400
1	12 x 17	204	76 x 107 x 180	16,300
2	12 x 20	480	76 x 126 x 180	19,200
2	15 x 19	570	95 x 120 x 180	22,800
1	17 x 20	340	107 x 126 x 180	27,200
4	19 x 20	1,520	120 x 126 x 180	30,400
15 racks		3,819		

Storage locations center-to-center spacing (inches)	6.28
Storage cell inner dimension (inches)	6.07
Intermediate storage location inner dimensions (inches)	6.12
Type of fuel	BWR 8 x 8 BWR 8 x 8 (R) BWR 7 x 7

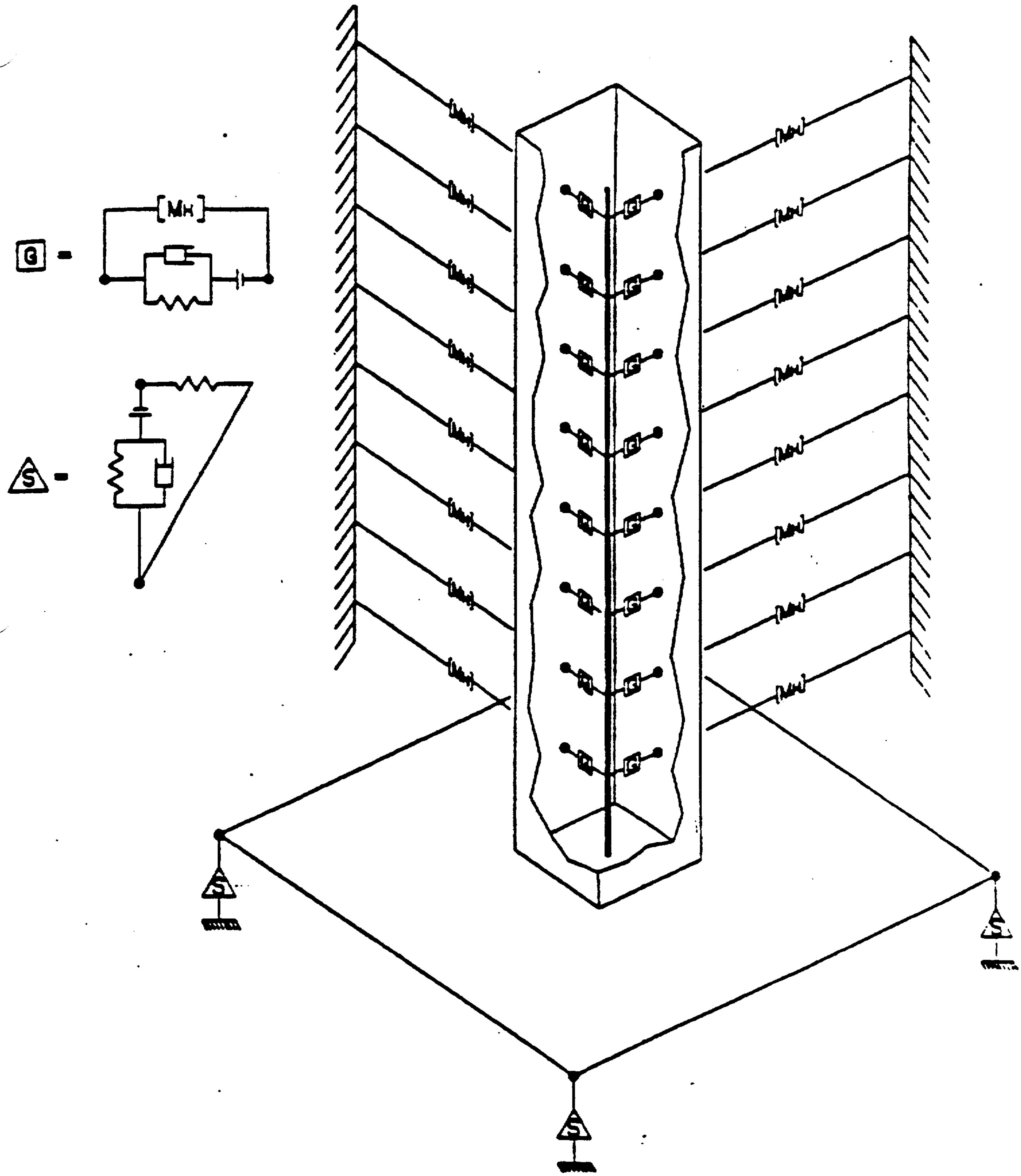

Figure 3. Three-Dimensional Nonlinear Seismic Model

coupling to other rack modules and/or the pool wall. The second part of the seismic analysis used a linear, three-dimensional, finite element model of the fuel rack, as shown in Figure 4, for the dual purposes of computing rack stresses and determining the rack module structural properties for use in the nonlinear dynamic displacement analysis.

The Licensee's seismic and stress analysis of the spent fuel rack modules considered full, partially filled, and empty rack modules.

The description and evaluation of the two models are addressed in detail in Sections 3.2 and 3.3. The displacement and stress results are discussed in appropriate subsections.

3.2 EVALUATION OF THE NONLINEAR DYNAMIC DISPLACEMENT ANALYSIS

3.2.1 Description of the Model

The Licensee performed seismic displacement analyses of the free-standing fuel rack modules with the use of the Westinghouse Electric Computer Analysis (WECAN) Code [1]. The analysis was performed as a time-history analysis using the three-dimensional mathematical model shown in Figures 3 and 5, with simultaneous application of three orthogonal, independent, acceleration time-histories (two horizontal and one vertical).

The effective structural properties of the single cell model shown in Figure 3 were modeled by three-dimensional beam elements and were derived from linear three-dimensional analysis of the fuel rack to which the hydrodynamic mass of the water was added. The fuel assembly, modeled by beam elements and represented in Figure 3 by the heavy vertical line, was connected to the cell walls through springs, dampers, gap elements, and hydrodynamic mass of the water in the cell. This model enabled the simulation of fuel assembly motion in the clearance space between the fuel assembly and the rack cell walls, as well as impact with the cell walls.

Hydrodynamic mass coupling of the rack module to adjacent rack modules and to the spent fuel pool walls is shown in Figures 3 and 5, and is discussed in Section 3.2.3.

The Licensee provided the following description of the modeling of support pads [1]:

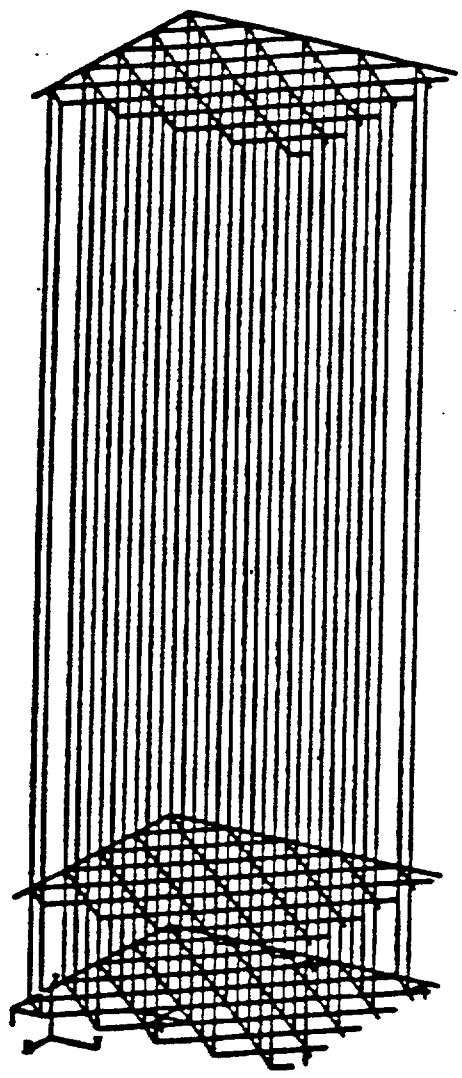

Figure 4. Structural Model (Quarter Rack)

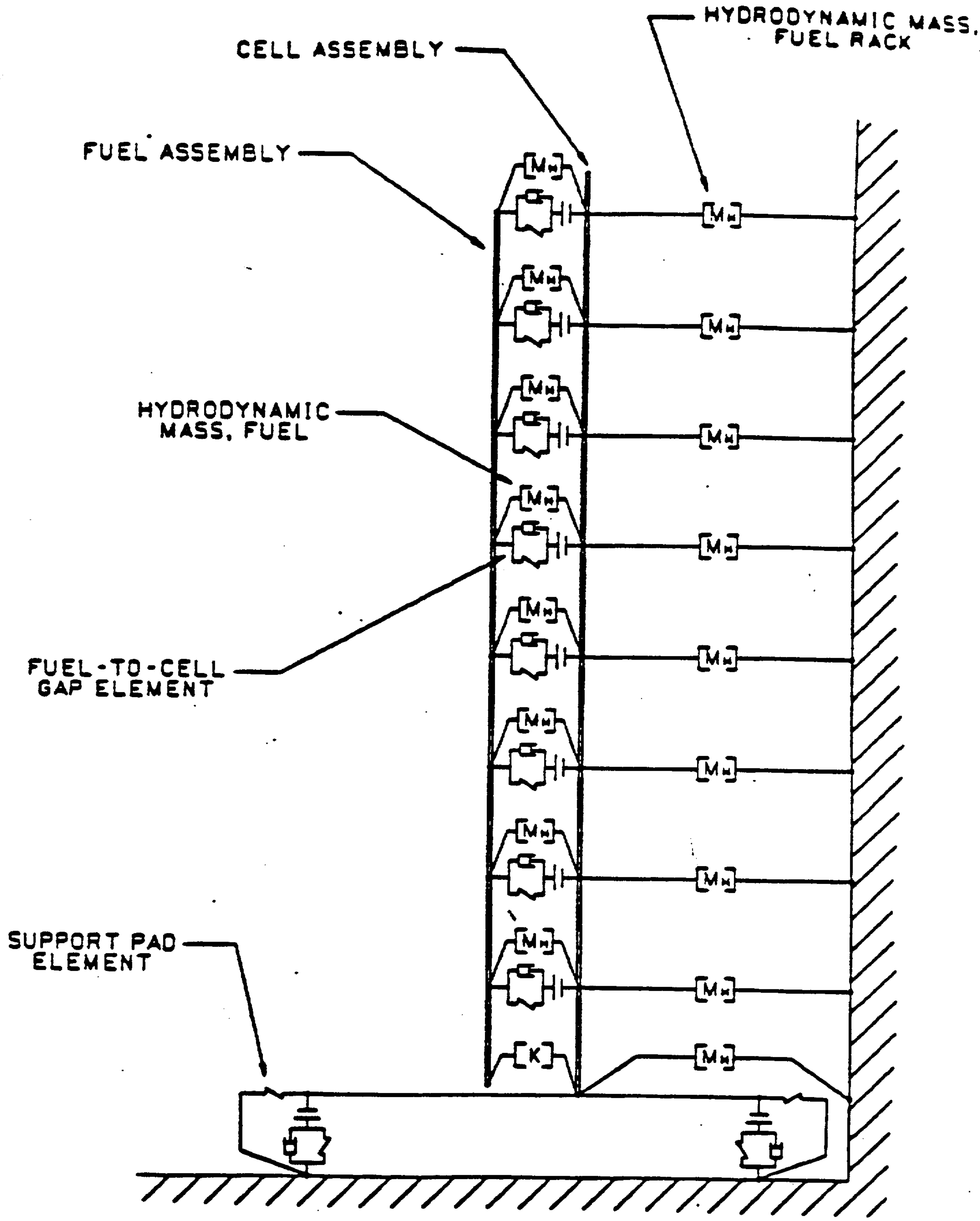

Figure 5. Section of Three-Dimensional Nonlinear Seismic Model

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

Structural damping used in the analysis, with the exception of damping unique to fuel assembly impact, was 2% for the OBE event and 5% for SSE. Added damping due to submergence in the pool water was not considered.

Damping of the impact between the limber fuel assemblies and the walls of the storage cells requires consideration beyond that of usual structural damping. In response to a request for additional information, the Licensee provided the following [3]:

"Impact damping between the fuel assembly and the rack cell was included in the analysis. A damping ratio of 0.04 was used for both the top and bottom fittings of the fuel assembly and is a conservative value for impact damping of rigid structures since higher damping ratios are used in the seismic analysis for the reactor vessel and piping supports.

For the intermediate fuel grid assemblies a damping ratio of 0.25 was used. The grid assembly is a flexible structure with frictional connections at the fuel rods which produces large impact damping values. A review of GE fuel information by the Westinghouse Nuclear Fuel Division has determined that a grid assembly damping ratio of 0.25 is appropriate. This damping value is consistent with the grid damping ratio that has been determined for Westinghouse fuel by tests performed by the Westinghouse Nuclear Fuel Division using a fuel assembly in air impacting on a rigid surface."

The Licensee's modeling of the rack modules and use of fuel assembly impact damping is acceptable.

3.2.2 Frictional Force Between Rack Support Pads and the Pool Liner

The Licensee used a maximum value of 0.8 and a minimum value of 0.2 for the range of static friction coefficient between the rack support pads and the pool liner [1]. Rabinowicz, in a report to the General Electric Company [4], focused attention on the mean and the lowest coefficient of friction to be used in these circumstances. While Rabinowicz supported the range of static coefficient used by the Licensee, he also indicated that the dynamic, or sliding, coefficient of friction is inversely proportional to velocity. The Licensee did not indicate whether the analysis used an initial static coefficient of friction and a lower dynamic coefficient of friction once sliding motion began. While the use of a lower dynamic coefficient of friction may have yielded somewhat larger sliding displacements, the Licensee's computed sliding displacement was sufficiently small to dismiss further consideration of dynamic coefficients of friction. Thus, the Licensee's use of friction coefficient between the support pads and the pool liner is acceptable.

3.2.3 Hydrodynamic Coupling Between Fluid and Cell Structure

Hydrodynamic coupling acts between adjacent rack modules, between a rack module and the pool walls, and between fuel assemblies and the cells in which they are inserted. Hydrodynamic coupling can have a significant effect upon the dynamic response of a rack module during seismic events.

In response to a request for additional information, the Licensee indicated that the motion of adjacent racks may be out of phase or unrelated [3]. This assumption led to consideration of the motion of an individual cell surrounded on all four sides by rigid boundaries which are separated from the cell by equivalent gaps. The hydrodynamic coupling mass between the rack module and the pool wall, as shown in Figure 3, was calculated by evaluating the effects of the gap between the modules and the pool wall using the method outlined in the paper by Fritz [5].

Fritz's [5] method for hydrodynamic coupling is widely used and provides an estimate of the mass of fluid participating in the vibration of immersed mass-elastic systems. Fritz's method has been validated by excellent agreement with experimental results [5] when employed within the conditions upon which it was based, that of vibratory displacements which are very small compared to the dimensions of the fluid cavity. Application of Fritz's method for the evaluation of hydrodynamic coupling effects between rack modules and a pool wall has been considered by this review to serve as an approximation of the actual hydrodynamic coupling forces. This is because the geometry of a fuel rack module in its clearance space is considerably different than that upon which Fritz's method was developed and experimentally verified.

Thus, the limitations of Fritz's [5] modeling technique for hydrodynamic coupling of rack modules adjacent to other rack modules or a pool wall indicate that the Licensee's fuel rack dynamic model should be considered conservative only for dynamic displacements that are small relative to the available displacement clearance.

3.2.4 Seismic Loading

The Licensee indicated that the earthquake loading was predicated upon an operating basis earthquake (OBE) at the site having a horizontal ground acceleration of 0.05 g, and that a safe shutdown earthquake (SSE) with a horizontal ground acceleration of 0.12 g was used to check the design to assure no loss of function [1]. The Licensee indicated further that these OBE and SSE designations correspond to FSAR designations of design earthquake (DE) and maximum credible earthquake (MCE), respectively [1].

In response to a request for additional information, the Licensee described the procedure used to determine the two orthogonal horizontal and one vertical simulated earthquake acceleration time-histories as follows [3]:

> "Simulated earthquake acceleration time histories in two orthogonal horizontal directions were generated from the Reactor Building seismic response spectra at the spent fuel pool floor evaluation using the SIMQKE* computer program. The results were evaluated to ensure that statistical independence was achieved and that the resulting response spectra adequately enveloped the original Reactor Building floor response spectra.
>
> The two horizontal acceleration time histories are generated from a single seismic floor response spectra which represented the worst case for the structure. Therefore, seismic analyses of the fuel racks are conservatively based on the worst case horizontal seismic loading applied in both horizontal directions simultaneously."

*SIMQKE, A program for Artificial Motion Generation, User's Manual and Documentation, Department of Civil Engineering, Massachusetts Institute of Technology, November 1976.

The Licensee has stated further that one of the two orthogonal, horizontal, acceleration time-histories was directed across the short dimension of the rack module in the analysis of the 9 x 14 cell rack module [6].

Evaluation indicated that the Licensee's development and application of simulated acceleration time-histories is acceptable.

3.2.5 Integration Time Step

The Licensee performed a time step study in an effort to find the correct integration time step to yield a converged solution [3]. Solutions using different time steps showed that the results were the same for time increments of 0.0025 sec and 0.00125 sec. The Licensee then performed the final analysis using a time step of 0.0025 sec.

3.2.6 Rack Displacements

The Licensee's analysis indicated that the maximum sliding displacement occurred with the minimum friction coefficient of 0.2, whereas the maximum rack displacement at the top of the rack due to bending and tipping occurred with the maximum friction coefficient of 0.8 [3].

The Licensee also noted that the maximum rack module displacements occurred for full racks and that the displacement of the 9 x 14 cell rack module in the 9-cell direction was the largest [3]. These largest displacements are presented in Table 2.

Maximum liftoff of a support pad from the pool liner was reported by the Licensee to be 0.0129 inch under the SSE event, and to occur on the 9 x 14 cell rack in the 9-cell direction [3].

The maximum computed displacements due to sliding, elastic deformation, and tipping are shown in Table 2, which provides the data supplied with the Licensee's response [3] to a request for additional information.

It is noted in Table 2 that each occurrence of sliding is relatively small with the sum of five OBE occurrences amounting to 0.049 inch.

Table 2. Rack Displacements: SSE Seismic + Maximum Normal Thermal

			SSE Seismic + Normal Thermal Displacements	
	Symbol	Units	Rack Top	Rack Base
Max. Sliding Distance, $\mu = 0.2$ $\Delta_s = (0.0098)5$*	Δ_s	in	0.049	0.049
Max. Structural Defl., $\mu = 0.8$	δ	in	0.647	0.0
Total Displacement One Rack $\Delta = \Delta_s + \delta$	Δ	in	0.696	0.049
SRSS Combined Displacement 2 Racks with Only 1 Sliding $\Delta_{max} = \Delta 2 + \delta 2$	Δ_{max}	in	0.950	0.049
Max. Normal Thermal Displacement	δ_T	in	0.087	0.087
Max. Combined Thermal & Seismic Displacements $\Delta = \delta_T + \Delta_{max}$	Δ	in	1.037	0.136
Nominal Rack to Rack Gap	Δ	in	1.68	1.03

*This accounts for five OBE events.

Maximum structural deflection at the top of the rack was reported to be 0.647 inch which, when combined with accumulated sliding, yielded 0.696 inch [3]. For the case of adjacent dissimilar rack modules whose responses may be out of phase, the Licensee combined the displacement of the two rack modules by the square root of the sum of the squares to yield a combined displacement of 0.950 inch. After including the maximum normal thermal growth, the Licensee compared the maximum combined displacement of 1.037 inches to the installed clearance of 1.68 inches between racks (shown in Table 2). With the combined displacement of the two adjacent rack modules less than the available clearance space, the Licensee indicated that impact of the racks would not occur and that impact analysis of the rack modules is not necessary.

While the use of the square root of the sum of the squares is a reasonable approach to combining out-of-phase displacements of adjacent rack modules for comparison to the available clearance space, the worst possible case is that of direct summation of the rack's displacement. This worst case would represent the point in time when the responses are 180 degrees out of phase. Thus, using the Licensee's displacement data as shown in Table 2, it can be seen that even the direct sum of two total displacements is less than the clearance space of 1.68 inches. Note that the clearance space between the rack modules and pool structure, as shown by the boxed dimensions in Figures 1 and 2, is much larger.

The evaluation of the Licensee's computed maximum displacements and their comparison with the installed clearance space indicated that they are acceptable, and that rack module impacts with other rack modules and the pool structure is unlikely.

3.3 EVALUATION OF THE DETAILED THREE-DIMENSIONAL LINEAR MODEL

3.3.1 Description of the Model

The Licensee used a finite element model of the rack module to determine the stresses in the module. The Licensee's description of the procedure follows [1]:

> "The structural model, shown in [Figure 4], is a quarter section representation of the rack assembly consisting of beam elements interconnected at a finite number of nodal points and general mass matrix elements. The

beam elements model the beam action of the cell, the stiffening effect of the cell to cell welds, and the supporting effect of the support pads. The general mass matrix elements represent the hydrodynamic mass of the rack module. The beams which represent the cells are loaded with equivalent seismic loads and the model produces the structural displacements and internal load distributions necessary to calculate the effective structural properties of an average cell within the rack module. In addition to the stiffness properties, the internal load and stress distributions of this model are used to calculate stress peaking factors to account for the load gradients within the rack module."

The results of the seismic displacement analyses were searched throughout the full analysis time to obtain the maximum response forces. These maximum values were then adjusted by peaking factors from the structural model to account for stress gradients through the rack module [1].

Load combinations and acceptance stress limits used in the Licensee's stress analysis were in accordance with the NRC's OT Position Paper [2] and are shown in Table 3. The Licensee's computed stresses, allowable stresses, and safety margins are shown in Table 4 [1]. Note that the safety margins, computed in accordance with the following formula, are all greater than zero, thereby indicating acceptable conditions:

$$\text{Safety Margin} = \frac{\text{Allowable Stress}}{\text{Design Stress}} - 1$$

3.3.2 Review of Stress Levels

Evaluation of the rack module stresses indicated that the analysis, level of stresses, and acceptability criteria are satisfactory.

3.4 REVIEW OF SPENT FUEL POOL STRUCTURAL ANALYSIS

3.4.1 Spent Fuel Pool Structural Analysis

The spent fuel pool (SFP) structure was analyzed using linear and nonlinear finite element models to determine the maximum allowable fuel rack loads that could be imposed on the pool slab.

Table 3. Storage Rack Loads and Load Combinations

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

Notes:

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

2. The provisions of NP-3231.1 of ASME Section III, Division I, shall be amended by the requirements of Paragraphs c.2, 3, and 4 of Regulatory Guide 1.124, entitled "Design Limits and Load Combinations For Class A Linear-Type Component Supports."

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

Table 4. Summary of Design Stresses and Minimum Margins of Safety
Normal and Upset Conditions

		Design Stress (psi)	Allowable Stress (psi)	Margin of Safety
1.	Support Pad Assembly			
1.1	Support Pad			
	Shear	1595	11000	5.90
	Axial and Bending	10479	16500	.57
	Bearing	13645	27500*	1.02
1.2	Support Pad Screw			
	Shear	7958	11000	.38
1.3	Support Structure			
	Axial and Bending	17626	27500*	.56
	Shear	1233	11000	7.92
	Weld Shear	19072	275000*	.44
2.0	Cell Assembly			
2.1	Cell			
	Axial and Bending	.816	1.0**	.23
2.2	Cell to Base Plate Weld			
	Weld Shear	19082	24000	.26
2.3	Cell to Cell Weld			
	Weld Shear	16286	21000	.29
	Pin Shear	7384	9260	.25
2.4	Cell to Wrapper Weld			
	Weld Shear	8300	11000	.33
2.5	Cell Seam Weld			
	Weld Shear	3501	4516***	.29
2.6	Cell to Cover Plate Welds			
	Weld Shear	11854	24000	1.03

* Thermal Plus OBE Stress is Limiting
** Allowable per Appendix XVII -2215 Eq (24)
*** Design Load and Allowable Load in Lbs is shown

Loading combinations required by USNRC Regulatory Guide 1.142, USNRC Standard Review Plan 3.8.4, the American Concrete Institute, and the American Institute of Steel Construction were satisfied. These were consolidated into the set of load combination requirements shown in Table 5, and were satisfied using strength design methods for the concrete structures and plastic design methods for structural steel [1].

Thermal loads were based on pool water temperatures of 150°F resulting from a full core discharge under normal operating conditions, and saturation temperatures for accident conditions varying from 250°F at the bottom of the pool to 212°F at the free water surface. A conservative ambient air temperature of 68°F was used. A stress free-temperature of 70°F was assumed.

3.4.2 Analysis Procedures

3.4.2.1 Method of Analysis

The Licensee employed the MSC/NASTRAN general purpose finite element program to investigate the spent fuel pool structure, using a three-dimensional finite model that included the entire spent fuel pool structure as well as adjacent key structural members. The model is shown in Figure 6. The Licensee provided the following additional features of the model [1]:

> "Floor slabs and walls immediately adjacent to the SFP are modeled to simulate the proper lateral restraint on the pool structure. Complete fixity against translation and rotation is assumed at the base of the drywell shield wall. Cut-off boundaries of adjoining walls and slabs were restrained with translational springs. These springs permit the model to simulate the cantilever mode deflected shape of the Reactor Building under horizontal seismic loading. Translational springs simulate lateral stiffness of the remainder of the Reactor Building walls which were not included in the model. In-plane rotations of all interior grid points on slabs and walls are restrained."

The overall model was estimated to contain 11,000 independent degrees of freedom [1].

While this was a linear mathematical model, the Licensee applied the external loads in increments to perform a piecewise linear solution to the nonlinear problem of cracking in the concrete under tensile stresses. Checking of the computed stresses against the concrete cracking criterion and

Table 5. Spent Fuel Pool Governing Design Load Combinations

Reinforced Concrete

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

Structural Steel

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

Notation:

D = dead load

E = OBE (design earthquake)

E'= SSE (maximum credible earthquake)

L = live load

T_a= thermal load produced by accident condition

T_o= thermal load during normal operation

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

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

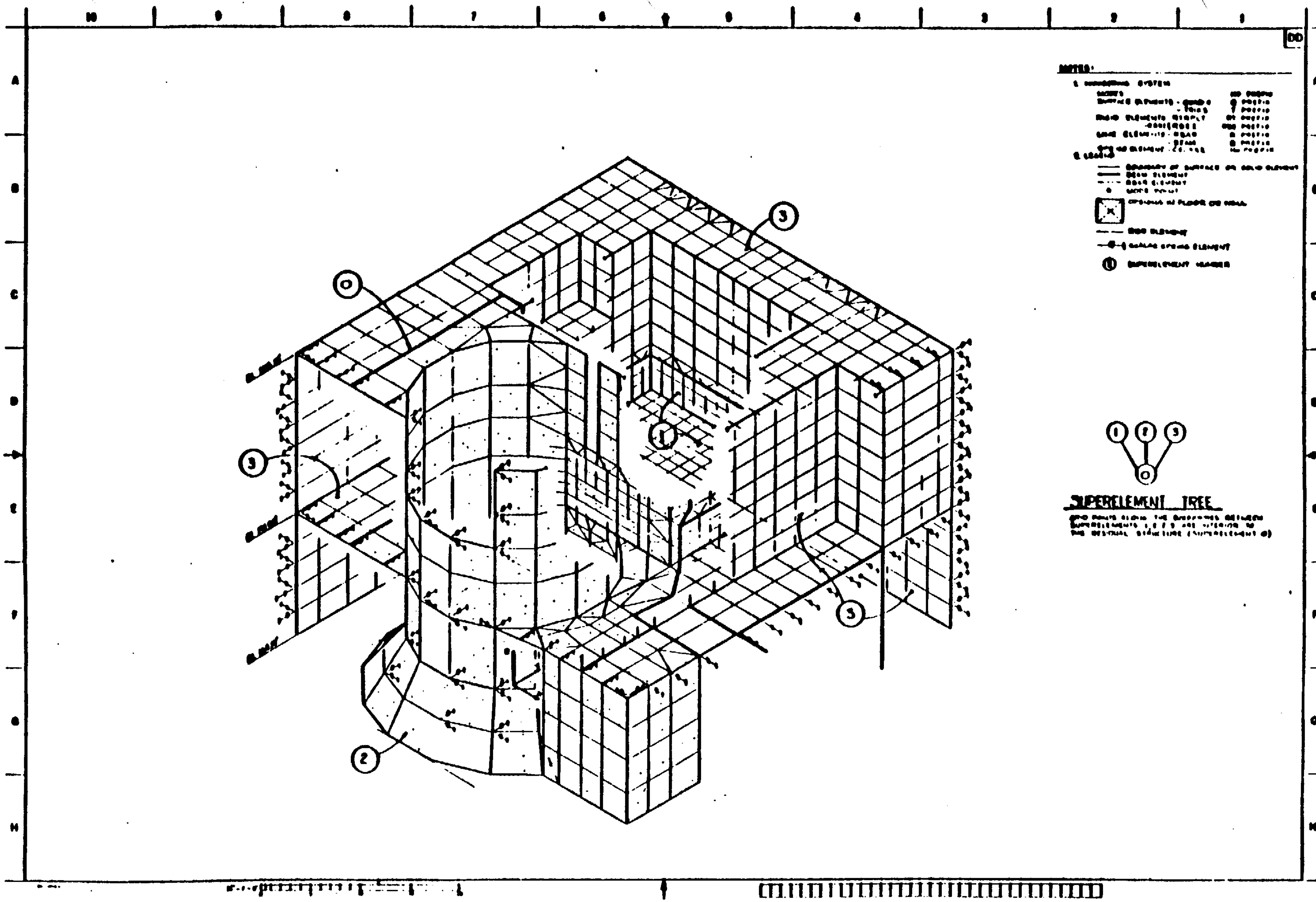

Figure 6. Reactor Building - Finite Element Model

the adjustment of material properties to reflect crack development was reported to have been performed manually at the end of each iteration. Thus, each new iteration was begun using the accumulated load that included the new load increment as well as stiffness properties reflecting crack development to that point.

Cracking criteria were applied primarily to the elements comprising the pool slab and lower portions of the pool walls. Application of the cracking criteria was carried out by comparing the local orthogonal tensile stresses against the modulus of rupture and adjusting the respective elastic modulus to reflect crack development.

The critical section for slab shear and bending was taken at the face of the walls in accordance with ACI Code provisions. The critical section in the wall was taken on the horizontal plane at the top of the slab elevation [1]. Shear capacities of the steel beams and connections were determined in accordance with Part 2 of the AISC specifications for plastic design.

With respect to thermal moment relaxation of local areas away from the pool slab, the approach used for the investigation was, in accordance with ACI 349 Appendix A, to assume the structure is uncracked for mechanical loads and cracked for thermal loads.

3.4.2.1 Supporting Analysis

In addition to the piecewise linear analysis described above, the Licensee performed a nonlinear finite element analysis of a simplified pool slab structure to provide an estimate of the pool slabs' ultimate load carrying capacity. The pool slab was modeled using the ADINA finite element program by which it was possible to compute the collapse load of the slab considering the beneficial effects of arching [1].

The Licensee reported that the nonlinear analysis indicated no reinforcement yielding and very little concrete cracking at the design load.

The Licensee halted the nonlinear analysis when the applied load approached three times the factored design load. At this point, the analysis indicated that some cracking at supports and at midspan would occur, that the top bar at supports would yield, but that collapse was not imminent [1].

3.4.3 Results of the Analysis

The Licensee reported the following [1]:

- "Reduced transverse shear capacity was used in the pool slab to reflect the small amount of membrane tension generated by the lateral fluid pressure on the pool walls. This shear capacity was compared against peak transverse shear forces from the MSC/NASTRAN finite element analysis results and is adequate."
- "The load transfer capacity of the wall/slab joints on the East and West sides of the pool were evaluated and found to be adequate."
- "Additional shear stresses due to increased spent fuel storage capacity are calculated to be 0.0020 kip/in^2 and 0.0032 kip/in^2 at EL. 180'-0" for OBE and SSE respectively. These shear stress increments are based on the MSC/NASTRAN finite element analysis results. These increments represent increases in total shear stresses from 89 percent to 92 percent of the allowable for OBE and from 69 percent to 70 percent for SSE. The resulting total concrete shear stresses are less than the allowable shear stresses."
- "Local areas of the North exterior wall of the Reactor Building were also evaluated due to the increased loads. The areas checked are the support points of the East and West walls of SFP. These areas are adequate for combined axial load and bending. Shear forces are also less than the shear capacity."

The Licensee's maximum allowable fuel rack/pool floor interface loads and stresses are reproduced in Table 6. The Licensee's comparison of the pool floor interface loads and stresses with allowable values is shown in Table 7.

Evaluation of the spent fuel pool analysis indicated that the analysis is satisfactory and that the spent fuel pool structure is adequate for the increased density of fuel storage.

3.5 FUEL HANDLING ACCIDENT ANALYSIS

3.5.1 Fuel Handing Crane Uplift

The Licensee provided the following with respect to crane uplift of a fuel assembly [1]:

"The objective of this analysis is to ensure that the rack can withstand the maximum uplift load of 4,000 pounds and a horizontal force of 1,000 pounds of the fuel handling crane without violating the critically acceptance criterion. The maximum uplift load is approximately two times

Table 6. Maximum Allowable Fuel Rack/Pool Floor Interface Loads

NO.	LOAD COMBINATION	TOTAL LOADS VERTICAL (KIP)	TOTAL LOADS HORIZONTAL (KIP)	LOCAL BEARING (KSI)
1.	D + L	3,900.0[1]	N/A	2.4
2.	D + L + T_o	3,900.0[1]	N/A	2.4
3.	D + L + T_o + E	5,700.0	1,900.0	2.4
4.	D + L + T_a + E	5,700.0	1,900.0	2.4
5.	D + L + T_o + P_f	5,700.0	N/A	3.2
6.	D + L + T_a + E'	8,000.0	3,000.0	3.2
7.	D + L + F_d	8,000.0	N/A	4.76
	Alternate[1]			
8.	1.4 (D + L + T_o) + 1.9E	8,900.0	3,600.0	See Note 2
9.	1.4 (D + L + T_a) + 1.9E	8,900.0	3,600.0	See Note 2
10.	1.7 (D + L + T_o + E)	9,700.0	3,200.0	See Note 2
11.	1.7 (D + L + T_a + E)	9,700.0	3,200.0	See Note 2

Notes:

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

2. When total loads are evaluated using Load Combination No. 8, 9, 10, and 11, local bearing pressures shall satisfy Load Combination No. 1, 2, 3, 4, and 5.

3. Notations used in this table are the same as defined in SRP 3.8.4, Appendix D.

Table 7. Pool Floor Loads

Load Combination	Condition*	Design Stress or Load	Allowable Stress or Load	Margin of Safety
1. D + L	Local Bearing	1.76	2.4	.36
2. D + L + To	Local Bearing	1.76	2.4	.36
3. D + L + To + E	Local Bearing	1.94	2.4	.24
4. D + L + Ta + E	Local Bearing	1.94	2.4	.24
5. D + L + To + Pf	Local Bearing	1.76	3.2	.82
6. D + L + Ta + E'	Vertical	6180	8000	.29
	Horizontal	1670	3000	.80
	Local Bearing	2.63	3.2	.22
7. D + L + Fd	Vertical	4130	8000	.94
	Local Bearing	4.39	4.76	.08
8. 1.4(D + L + To) + 1.9E	Vertical	7730	8900	.15
	Horizontal	1590	3600	1.26
9. 1.4(D + L + Ta) + 1.9E	Vertical	7730	8900	.15
	Horizontal	1590	3600	1.26
10. 1.7(D + L + To + E)	Vertical	8760	9700	.11
	Horizontal	1420	3200	1.25
11. 1.7(D + L + Ta + E)	Vertical	8760	9700	.11
	Horizontal	1420	3200	1.25

*Vertical refers to total pool floor vertical load in kips. Horizontal refers to total pool floor horizontal load in kips. Local bearing refers to pool floor bearing stress under the highest loaded support pad in ksi.

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

3.5.2 Accidental Fuel Assembly Drop

The Licensee provided the following [1]:

"Three accident conditions are postulated. The first accident condition assumes that the weight of a fuel assembly and handling tool impacts the top end fitting of a stored fuel assembly or the top of a storage cell from a conservative drop height of 2 feet in a straight attitude. The second accident condition is similar to the first except the impacting mass is at an inclined attitude. The impact energy is absorbed by the dropped fuel assembly, the stored fuel assembly, the cells and the rack base plate assembly. Under these faulted conditions the criticality acceptance criterion is not violated and the pool liner is not perforated. The third accident condition assumes that the dropped assembly falls straight through any empty cell and impacts the rack base plate from a conservative drop height of 2 feet above the top of the rack. The results of this analysis show that the impact energy is absorbed by the fuel assembly and the rack base plate. The spent fuel pool liner is not perforated. Criticality calculations show the k_{eff} <0.95 and the criticality acceptance criterion is not violated.

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

4. CONCLUSIONS

Based upon the review and evaluation, the following conclusions were reached:

- The Licensee used three-dimensional, nonlinear dynamic displacement analyses with three simultaneous, independent, orthogonal, earthquake acceleration time histories to provide greater resolution of the rack module displacements than is possible with two-dimensional analyses combined by the square root of the sum of the squares method.

- The limitations of the modeling technique employed for hydrodynamic coupling of fuel assemblies within a fuel rack cell and of fuel rack modules to other rack modules and the pool walls indicate that the modeling technique contributes experimentally verified results only for displacements which are small compared with the available clearance space. While the Licensee's reported rack module displacements are not small relative to the clearance space, the techniques used are acceptable in association with the conservative assumptions employed.

- The spent fuel pool structure has design margin to sustain the higher density floor loadings.

5. REFERENCES

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

ENVIRONMENTAL ASSESSMENT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE

SPENT FUEL STORAGE RACKS

FACILITY OPERATING LICENSES NOS. DPR-44 AND DPR-56

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

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1.0 INTRODUCTION

The present storage capacity of the spent fuel pools at Peach Bottom Atomic Power Station, Units 2 and 3, is 2,608 fuel assemblies for each spent fuel pool for each unit. These limited storage capacities were in general in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a few years and then shipped offsite for reprocessing and recycling.

Commercial reprocessing of spent fuel has not developed as had been originally anticipated. In 1975 the Nuclear Regulatory Commission directed the staff to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement was to consider alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Reactor Fuel (NUREG 0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August 1979. In the FGEIS, consistent with long range policy, the storage of spent fuel is considered to be interim storage, to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite storage by modification of the existing spent fuel pools. Since the issuance of the FGEIS, numerous applications have been received and approved. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage designs and limitations caused by the spent fuel already stored in pools, the FGEIS recommended that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pools, the FGEIS discusses in detail other spent fuel storage alternatives. The finding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of various alternatives reflects the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the existing spent fuel storage capacity is filled, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Assessment (EA) addresses only the specific environmental concerns related to the proposed expansion of the Peach Bottom Atomic Power Station, Units 2 and 3, spent fuel storage capacity. This EA consists of three major parts, plus a summary and conclusion. The three parts are: (1) descriptive material, (2) an appraisal of the environmental impact of the proposed action, and (3) an appraisal of the environmental impact of postulated accident. Additional discussion of the alternatives to increasing the storage capacity of existing spent fuel pool is contained in the FGEIS.

1.1 Description of the Proposed Action

By application dated June 13, 1985 and supplemented by letters dated August 1, 1985, October 9, 1985 and December 26, 1985, Philadelphia Electric Company (the licensee or PECo) requested approval to permit increases in the storage capacity of the Peach Bottom Atomic Power Station, Units 2 and 3, spent fuel pools (SFPs) from 2,608 to 3,819 storage cells. The increases are to be accomplished by use of new rack structures and removal of the SFPs cooling piping and diffusers.

The environmental impacts associated with the operations of Peach Bottom Atomic Power Station, Units 2 and 3, were considered in the Final Environmental Statement (FES) issued in April 1973(1). The purpose of this EA is to evaluate any additional environmental impacts which are attributable to the proposed increases in the SFPs storage capacity at both Peach Bottom units.

1.2 Need for Increased Storage Capacity

Each unit at Peach Bottom Atomic Power Station is a boiling water reactor (BWR). The licensee's projected SFP capacity requirements indicate that both units will lose their full-core discharge reserve storage capacity (764 assemblies) in the 1987-88 time frame; and, in the 1991-1992 time frame, they will no longer have the capacity to store any additional fuel discharges from the operating units. Therefore, to ensure that sufficient capacity continues to exist for Peach Bottom to store discharged fuel assemblies, PECo plans to replace the existing storage racks with new spent fuel storage racks whose design will allow for more dense storage of spent fuel, thus enabling the existing pools to store more fuel in the same place as occupied by the current racks.

1.3 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for alterations and expansions; in September, 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York are licensed to store spent fuel. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that have contractual arrangements with NFS. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris operation was renewed for another 20 years, however, GE is also not accepting any additional spent fuel for storage at this facility.

2.0 FACILITY

The principal features of the spent fuel storage and handling at Peach Bottom, Units 2 and 3, as they relate to the proposed modifications are described here to aid understanding of the evaluations provided in subsequent sections of this EA.

2.1 Spent Fuel Pool (SFP)

Initially spent fuel assemblies are intensely radioactive due to their fresh fission product content when removed from the core; also, they have a high thermal output. The SFP is designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them offsite. Space permitting, assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling. The SFPs structures structures are reinforced concrete lined with an eight gage thick stainless steel liner.

2.2 Spent Fuel Pool Cooling and Cleanup System

Each Peach Bottom unit has an independent spent fuel pool and spent fuel pool cooling and cleanup system. The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The Peach Bottom spent fuel cooling system is composed of three fuel pool cooling pumps, three heat exchangers, a filter-demineralizer,and two skimmer surge tanks. The filter-demineralizers, which collect radioactive corrosion products, are so arranged that one is designated for each reactor unit, and the third is a common spare for use by either unit when either of the other two units is taken out of service for precoating.

The pumps circulate the pool water in a closed loop, taking suction from the skimmer surge tanks through the heat exchangers, circulating the water through the filter-demineralizers, and discharging through diffusers at the bottom of the pool fuel. The cooled water traverses the pool picking up heat and debris before starting a new cycle by discharging over the skimmer weirs into the skimmer surge tanks. Makeup water for the system can be transferred from the condensate storage tank to the skimmer surge tanks. Pool water clarity and purity are maintained by a combination of filtration and ion exchange. Alarms, differential pressure indicators, and flow indicators monitor the condition of the filter-demineralizers.

2.3 Radioactive Waste Treatment System

Each unit contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) for Unit Nos. 2 and 3, dated April 1973. The proposed modifications will not result in any significant additional radwaste that will need to be processed. Therefore, there will be no changes in the waste treatment systems described in Section 3.0 of the FES because of the proposed modifications.

3.0 NON-RADIOLOGICAL ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

The non-radiological environmental impacts associated with the operations of Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, as designed were considered in the FES. The proposed modifications of the SFPs will not cause any new non-radiological environmental impacts which were not previously considered based on the following:

1) The proposed modifications will alter only the spent fuel storage racks. They will not alter the external physical geometry of the SFP structures. In addition, construction of the new racks will be done offsite and transported

to the facility. No unusual terrestrial effects are anticipated or considered likely.

2) Additional storage will not result in a measurable increase in non-radiological chemical waste discharges to the receiving water. The licensee does not propose any changes in chemical usage or change to the NPDES permit.

3) Additional SFP heat output will not cause measurable thermal effects to the receiving water. The increase in the heat load due to this modification is less than five (5) percent for a 18-month reload and less than ten (10) percent for full-core discharge as compared with the present SFP design heat load. These calculated decay heat discharges to the plant water and to the Susquehanna River due to the proposed modifications do not significantly exceed the design values used by the NRC in its 1978 evaluation of the non-radiological environmental impact due to spent fuel increased storage at Peach Bottom.(2)

We conclude, based on the above evaluations, that the SFP modifications will not result in non-radiological environmental effects significantly greater or different from those already reviewed and analyzed in the FES for Peach Bottom, Units 2 and 3.

4.0 RADIOLOGICAL ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

4.1 Introduction

The potential radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90 are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in or near the SFPs would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates, however, that there is little radionuclides leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP.

During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. A few weeks after refueling, the spent fuel is cooled in the SFP and the fuel clad temperature becomes relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lifes and decay to insignificant levels within a few months. Based on the operational reports submitted by licensees and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear Fuel Services (NFS) storage pool at West Valley, New York. Some spent fuel assemblies which have significant leakage while in operating reactors have been stored in these two pools. After storage in the onsite SFP, these fuel assemblies were later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from these fuel assemblies in the offsite storage facility.

4.2 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton (Kr85). Experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products, including Kr-85, from stored fuel containing cladding defects.

The proposed Peach Bottom Unit Nos. 2 and 3 SFP modifications will increase the overall capability for each unit from 2608 to 3819 cells per unit. An average of 276 fuel assemblies are expected to be stored following each refueling. Since space must be reserved to accommodate a complete reactor core discharge (764 fuel assemblies), the useful pool capacity after the proposed modification will be 3055 fuel assemblies per unit. For the Peach Bottom site, at least one full core storage capability will be maintained for both units until 1993.

We assumed that all of the Kr-85 that is going to leak from defected fuel will do so in the interval between refuelings. The assumption is conservative and maximizes the amount of Kr-85 to be released. Our calculations summarized in Table 1 show that the maximum expected release of Kr-85 from one refueling cycle (276 assemblies) is approximately 144.3 curies. Spent fuel discharges from both units are expected to yield an annual release of 199 curies/year of Kr-85. This is not significant when compared to the estimated 300,000 curies/year of noble gas releases for the combined units from all other sources (1). Accordingly, the enlarged capacity of the pool has no significant effect on the greatest release rate of Kr-85 to the atmosphere. Thus, we conclude that the proposed modifications will have an insignificant effect on offsite exposures.

Iodine-131 release from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the SFP storage capacity because the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

A relatively small amount of tritium is contributed during reactor operation by fission of reactor fuel and subsequent diffusion of tritium through the fuel and Zircaloy cladding. Almost all of the tritium release from the fuel occurs while the fuel is hot, that is, during operations and, to a limited extent, shortly after shutdown. Thus, expanding SFP capacity will not increase the tritium activity in the SFP.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above 150°F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES.

Assuming the loss of all SFP cooling, boiling could occur after 83 hours for the maximum "abnormal" heat load condition (full core discharge with all remaining storage spaces full with fuel from successive cyclic discharges). This is a substantial period for actions to be taken such as initiating pool makeup water for the SFP. The licensee has analyzed the effects of SFP boiling on the outside environment. The licensee utilized a model similar to that previously employed for a comparable analysis on the Limerick Generating Station to determine the offsite radiological consequences of SFP boiling. The results indicate that the potential offsite dose would be a very small fraction of 10 CFR Part 100 limits and was a negligible offsite contributor. We find this analysis and its conclusion to be acceptable.

4.3. Solid Radioactive Waste

The concentration of radionuclides in the SFP water is controlled by the filters and the demineralizers, and by the decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the SFP, and decreases as the SFP water is processed through the filters and demineralizers. The increase of radioactivity, if any, due to the proposed modifications, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the SFP water to acceptable levels. The licensee states that the amount of solid waste presently being generated by the spent fuel pool cleanup system is approximately 100 cubic feet per unit every year. The licensee does not expect that these SFP modifications will result in any significant increase in this amount of solid waste generated from the spent fuel pool cleanup system. While we agree with the licensee, we note that should there be an increase in spent fuel pool resin waste generation, the total waste, however, would still be within those values estimated in the FES.

The present spent fuel pool racks will be removed from the pool. The disposal method has not been determined by the licensee. However, should the present racks be shipped to an ultimate burial site, the additional quantity of solid waste is not expected to be environmentally burdensome because the volume is small compared to the annual waste generation rate.

4.4 Radioactivity Released to Receiving Waters

Since the SFP cooling and cleaning systems operate as a closed system, only water originating from cleanup of the SFP floors and resin sluice water need be considered as potential sources of radioactivity. It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of the proposed SFP modifications. The SFP demineralizer resin removes soluble radioactive material from the SFP water. These resins are periodically sluiced with water to the SFP resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the SFP, but the soluble radioactive material would be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water it will be removed by processing through the liquid radwaste system. Therefore, because the liquid radwaste processing system captures radioactive material, it is not expected that any additional radioactivity will be released to the environment resulting from the proposed SFP modifications.

4.5 Occupational Radiation Exposures

The staff has reviewed the licensee's plan for the modification of the Peach Bottom SFP racks with respect to occupational radiation exposure involving the removal and disposal of the current racks, and the installation of the proposed higher density racks. The licensee estimates that the exposure for this operation will be approximately 36 man-rems. This estimate is based upon the licensee's 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. This exposure is a small fraction (less than one percent) of the total annual person-rem from occupational exposure.

We have estimated the increment in onsite occupational dose during normal operations after the proposed SFP modifications have been completed with the proposed increase in stored fuel assemblies. Our estimate is based on information supplied by the licensee for occupation times and for dose rates in the SFP area from radionuclides concentrations in the SFP water. 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. Based on the present and projected operations in the SFP area, the staff estimates that the proposed modifications should add less than one percent to the total annual occupational radiation exposure at the plant. This small projected increase in radiation should not affect the licensee's ability to maintain individual occupational dose to ALARA levels and within the limits of 10 CFR Part 20. Thus, the staff concludes that the storing of additional fuel in the SFP will not result in any significant increase in dose received by workers.

5.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

5.1 Rack Module Assembly Drop Accident

The overhead cranes in the auxiliary building at Peach Bottom will be used for removing the existing rack modules and lowering the new modules into the pool. The licensee has stated in Section 4.7.4.2, Procedure, of its August 1, 1985 submittal that all load handling operations for the new high density fuel storage racks in the SFP area will be conducted in accordance with the criteria of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". In this same Section, the licensee has committed that at no time will a rack module be carried directly over another module installed in the SFP, and containing spent fuel. Therefore, the assessment of the radiological consequences of a replacement rack drop accident is not required.

5.2 Fuel Handling Accident

The staff has reviewed the licensee's proposed SFP storage capacity increase as it relates to changes in the radiological consequences of a postulated fuel handling accident as compared to those reported in the FES (1). A bounding calculation performed by the staff shows that the radiological consequences of a cask drop/tip accident are well within the NRC Standard Review Plan (SRP) dose guidelines (SRP 15.7.5). The staff, therefore, concludes that the proposed SFP modifications are acceptable.

5.3 Conclusion

Based upon the above evaluation, the staff concludes that the likelihood of a rack module assembly drop accident is sufficiently small because rack module assemblies will not be carried directly over other fuel-containing modules installed in the SFP and, therefore, the staff concludes that this accident need not be considered. Also, a fuel handling accident involving a dropped assembly or cask would not be expected to result in radionuclide releases leading to offsite radiological consequences exceeding those of the fuel handling accident evaluated in the staff's FES of April 1973; that is, the doses would be well within 10 CFR Part 100 values. We conclude, therefore, that the proposed modifications are acceptable and will not result in radiological environmental effects that differ significantly from those previously evaluated.

6.0 ALTERNATIVE USE OF RESOURCES

This action involves no use of resources not previously considered in the FES(1) for Peach Bottom Units 2 and 3. In addition, because we have not identified any significant environmental impacts which would result from this action, we have not considered alternatives to the proposed action or assessed the impacts of alternative beyond that considered in the FGEIS.

7.0 OTHER PERSONS CONSULTED

The NRC staff evaluated the licensee's proposal and consulted the FGEIS but did not consult other agencies or persons in preparing this environmental assessment.

8.0 SUMMARY

The Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel concluded that the environmental impact of interim storage of spent fuel was negligible and the cost of various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in SFP designs the FGEIS recommended licensing SFP expansion on a case-by-case basis. For Peach Bottom Atomic Power Station, Units 2 and 3, expansion of the storage capacity of the SFPs does not significantly change the radiological impact evaluated in the April 1973 FES (1). As discussed in Sections 2.0 and 4.0, the proposed reracking and added fuel are well within the capability of the SFP cleanup system and this system will keep the concentrations of radioactivity in the SFP water well within acceptably low levels. Operation of the proposed SFP with additional spent fuel in the SFPs is not expected to increase the occupation radiation exposure by more than one percent of the total annual occupational exposure at Peach Bottom. We conclude that there are no significant radiological or nonradiological impacts associated with the proposed license amendments and that the amendments will not have a significant effect on the quality of the human environment.

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9.0 REFERENCES

1. Final Environmental Statement (FES) related to Peach Bottom Atomic Power Station, Units 2 and 3, April 1973

2. Letter from J.F. Stolz (NRC) to E.G.Bauer (Philadelphia Electric Company) dated November 30, 1978.

TABLE I

SFP MODIFICATION
ESTIMATE RELEASE RATE OF KR-85

DATA

Peach Bottom, Units 2 and 3

Core = 764 fuel assemblies

Single Refueling = 276 core assemblies per unit per 18 months

Cladding = Zircalay-4

Burnup = approx. 40,000 MWd/MTu

Weight of UO_2 in Core = 164.3 MT of UO_2 or 144.7 MTu

Escape Rate Coeff. of Kr-85 = 6.5×10^{-8}/sec

Fission Yield of Kr-85 = 0.0034

Failed Fuel Fraction (NUREG-0017) = .0012

Half-life (Kr-85) = 10.7 years

$$\text{Amt Kr-85 in fuel} < \frac{\text{Production rate}}{>\text{decay} + >\text{leakage}}$$

atoms/f   f/MWsec

$$\text{Production Rate} = \frac{0.0034 \times 3.12 \times 10^{16} \times 3293\ \text{MWt}}{144/8\ \text{MTu}}$$

$$= 2.4 \times 10^{15}\ \frac{\text{atoms}}{\text{MTu sec}}$$

$= \underline{2.4 \times 10^{15}}$ atoms/MTu sec

($>$decay $= 2.05 \times 10^{-9}$/sec, $>$leak $= 6.5 \times 10^{-8}$/sec)

Amt KR-85 in fuel $\underline{<3.60 \times 10^{22}}$ atoms/MTu
$\underline{<2380}$ Curies/MTu

The following model assumes that all Kr-85 that can leak out from the failed fuel assemblies will be released before the spent fuel is removed from the pool.

Simple case: All Kr-85 escape between refueling =

$$2880 \text{ curie/MTu} \times \frac{244.7 \text{ MTu}}{764 \text{ ass.}} \times \frac{276 \text{ ass.}}{\text{refuel}} \times .0012 = 149.3 \text{ curies/refueling}$$

For the two units, the average spent fuel input yields

$$144.3 \text{ curies/refuel} \times \frac{2 \text{ refuelings}}{18 \text{ months}} \times \frac{12 \text{ months}}{\text{year}} \quad 199.0 \text{ curies/year}$$