

August 18, 1986

Docket No.: 50-416

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

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Dear Mr. Kingsley:

SUBJECT: REVISION TO TECHNICAL SPECIFICATIONS - FUEL STORAGE AND
SPENT FUEL STORAGE POOL TEMPERATURE

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 6, 1985 as revised and supplemented by letters dated July 29, August 15, August 30, September 11, September 12, November 1, and December 18, 1985; and March 14, March 15, June 5, June 9, and July 25, 1986.

This amendment revises Technical Specifications Section 5.6, "Fuel Storage," to allow increased upper containment pool capacity and increased spent fuel storage pool capacity. This amendment also revises Specification 3/4.7.9, "Spent Fuel Storage Pool Temperature," to limit the pool temperature to 140°F and require plant shutdown if pool temperature cannot be maintained below this limit.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:

1. Amendment No. 17 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

PD#4/LA
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8/18/86

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

MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that
 - A. The application for amendment by Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association, (the licensees) dated May 6, 1985 as revised and supplemented by letters dated July 29, August 15, August 30, September 11, September 12, November 1, and December 18, 1985; and March 14, March 15, June 5, June 9, and July 25, 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. NPF-29 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 17, are hereby incorporated into this license. Mississippi Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 18, 1986

PD#4/PA
MButler
8/18/86

JK
PD#4/PM
LKintner:1b
8/17/86

*revised version to SE
check state & SEC and
EA published before
issuance*
OGC/HM
MYoung
8/11/86

PD#4/D
WButler *WB*
8/17/86

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf page(s) provided to maintain document completeness.*

Remove

5-5
5-6
3/4 7-33
3/4 7-34

Insert

5-5*
5-6
3/4 7-33*
3/4 7-34

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a design nominal active fuel length of 150 inches. The initial core loading shall have a design nominal enrichment of 1.708 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 193 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing a design nominal 143.7 inches of boron carbide, B_4C , powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

- a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
- b. For a pressure of:
 1. 1250 psig on the suction side of the recirculation pump.
 2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
 3. 1550 psig from the discharge shutoff valve to the jet pumps.
- c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal T_{ave} of 533°F.

DESIGN FEATURES

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.2-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

- a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 4.3 of the FSAR.
- b. A nominal 6.26-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 202'5 1/4".

CAPACITY

5.6.3 The spent fuel storage capacity is designed and shall be maintained with a storage capacity limited to:

- a. No more than 2324* spent fuel assemblies in the spent fuel pool, and
- b. No more than 800 spent fuel assemblies in the upper containment pool.

Placement of fuel in the upper containment pool is limited to temporary storage of fuel during refueling operations. Prior to return to reactor criticality, all spent fuel shall be removed from the upper containment pool.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

*The physical limit is 4348. The 2324 limit reflects the number of spent fuel assemblies that can be stored in the spent fuel pool without excessive reliance on RHR supplement cooling; i.e., for a time period in excess of a normal refueling duration.

TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

<u>AREA</u>	<u>TEMPERATURE LIMIT (°F)</u>
a. <u>Containment</u>	
Inside Drywell	150
CRD Cavity	185
Outside Drywell	105
Steam Tunnel	125
b. <u>Auxiliary Building</u>	
General	104
ECCS Rooms	150
ESF Electrical Rooms	104
Steam Tunnel	125
c. <u>Control Building</u>	
ESF Switchgear and Battery Rooms	104
Control Room	90
d. <u>Diesel Generator Rooms</u>	125
e. <u>SSW Pumphouse</u>	104*

*For this area, the limit shall be the greater of 104°F or outside ambient temperature plus 20°F, not to exceed 122°F for greater than one hour.

PLANT SYSTEMS

3/4.7.9 SPENT FUEL STORAGE POOL TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.7.9 The spent fuel storage pool temperature shall be maintained at less than or equal to 140°F.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel storage pool.

ACTION: With the spent fuel storage pool temperature greater than 140°F but less than 210°F, perform the following:

- a. Restore the pool temperature to less than or equal to 140°F within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and
- b. If at any time after exceeding 140°F an extrapolated temperature plot indicates that the pool temperature will exceed 210°F in less than 20 hours, be in at least STARTUP within 6 hours and in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.9.1 The spent fuel storage pool temperature shall be verified to be less than or equal to 140°F at least once per 12 hours.

4.7.9.2 Start each fuel pool cooling and cleanup pump not already running at least once per 92 days and run each pump for at least 15 minutes.

File Copy
w/Enclosures!!

Mr. Oliver D. Kingsley, Jr.
Mississippi Power & Light Company

Grand Gulf Nuclear Station

cc:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. NPF-29
MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
GRAND GULF NUCLEAR STATION, UNIT 1
DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated May 6, 1985, the Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association (the licensees) submitted an application for a license amendment to increase the storage capacity of the spent fuel pool and the upper containment pool for Grand Gulf Nuclear Station (GGNS) Unit 1 by replacing the originally installed fuel racks with new high density racks. By letters dated July 29, August 15, August 30, September 11, September 12, November 1, and December 18, 1985, and March 14, March 15, June 5, June 9, and July 25, 1986, the licensees revised and supplemented their application. The notice of consideration of issuance of this license amendment was published in the Federal Register before the licensees' July 25, 1986, submittal. The July 25, 1986, submittal contained supplemental information provided in response to the staff's questions regarding plant procedures that would be used during refueling in the event of a loss of offsite power and a subsequent single failure and a commitment to provide increased cooling capacity for the spent fuel pool. This supplemental information did not change the proposed Technical Specifications described in the notice, and the potential need for increased cooling capacity was described in the notice. The notice of consideration accurately describes the license amendment request, and the supplemental information does not affect the substance of the requested amendment.

The GGNS Unit 1 is a boiling water reactor with a Mark III containment. The spent fuel pool is located in the auxiliary building, which is similar to spent fuel pool arrangements for pressurized water reactors. Above the GGNS reactor, and within the containment, there is an upper containment pool with racks for holding new fuel to be placed in the reactor and spent fuel removed from the reactor during refueling; however, before reactor startup after refueling, all spent fuel is transferred to the spent fuel pool for storage.

The amendment would revise Section 5.6, "Fuel Storage," of the Technical Specifications to allow increased upper containment pool capacity and increased spent fuel storage capacity. This increased capacity would be obtained by replacing the fuel racks in the upper containment pool and in the spent fuel storage pool with high density fuel racks. The center-to-center distance between fuel assemblies would be changed from 12 inches to 6.26 inches. This reracking would increase the upper containment pool capacity from 170 to 800 fuel assemblies in order to hold a complete core unloading, if necessary, and increase the spent fuel pool storage capacity from 1270 to 4348 fuel assemblies. This would provide spent fuel storage capability until the year 2003, assuming reloads of a third

of a core. The capability to off-load the entire core would be available until the year 2000. However, the number of fuel assemblies to be stored in the spent fuel pool would be limited by Technical Specifications to 2324 until spent fuel pool cooling capability is increased. The Technical Specifications would be changed to limit the spent fuel pool water temperature to 140°F rather than 150°F and require plant shutdown rather than a special report if the limiting temperature were exceeded.

The licensees have removed the originally installed spent fuel racks, which were not used to store spent fuel assemblies, and have installed the new high density spent fuel racks during a planned outage in the fall of 1985. The licensees determined, pursuant to 10 CFR 50.59, that removal of the old racks and installation of the new racks would not involve an unreviewed safety question. A condition in the operating license prohibits the storage of spent fuel in the spent fuel pool until the standby service water system is modified. Modification of the standby service water system will be completed during the first refueling outage.

2.0 EVALUATION

2.1 Criticality Considerations

The high density spent fuel storage racks consist of double-walled stainless steel boxes with Boraflex neutron absorber sheets in the space between the walls. The inner dimension of the square boxes is 6.0 inches, and the boxes are arranged in an array having a 6.26-inch center-to-center spacing. The Boraflex sheets are 144 inches in length. As a result, 3 inches of the active fuel length of the assemblies extend beyond the Boraflex on each end. This fact was accounted for in the analysis. The Boraflex absorber contains 0.0204 gram of B-10 per square centimeter of surface area.

2.1.1 Calculation Methods

The nuclear criticality analysis of the spent fuel racks was performed with the AMPX-KENO computer package using the 123-group GAM-THERMOS cross-section set with the NITAWL treatment of U-238 resonance effects. This calculation procedure is widely used for fuel rack criticality analyses and is acceptable. It has been verified by Southern Science, who did the analysis, by comparison with critical experiments. A calculational bias has been determined from the comparisons. The nominal design case assumes an 8x8R assembly having fuel rods with a uniform enrichment of 3.5 weight percent (w%) U-235. The following conservative assumptions are made:

- (1) The moderator is pure water at the temperature yielding the maximum reactivity.
- (2) The racks are assumed to be infinite in extent in the lateral and vertical directions.
- (3) The fuel is assumed to be fresh and no credit is taken for burnable poison. Such fuel is more reactive than that which has burned out to its highest reactivity point.

- (4) No credit is taken for absorption in minor structural members (spacers, etc.), and the cladding is assumed to be pure zirconium.

Uncertainties treated in the analysis include those due to Boraflex thickness, width, and B-10 concentration; fuel enrichment, density, and diameter; lattice pitch; stainless steel thickness; and flow channel distortion. The reactivity of the racks is maximum at the lowest temperature (39°F). The maximum reactivity occurs with the assembly centered in the storage box. These uncertainties are the ones usually considered in spent fuel reactivity analyses and are acceptable.

2.1.2 Results and Conclusions

Abnormal and accident situations analyzed included heating the pool water to boiling and reducing the pool water density to 0.1 gram per cubic centimeter, closing of the water gap between racks, dropping of a fuel assembly onto the racks, and positioning of a fuel assembly outside the racks. The analyses showed that in no case was the k -effective of the racks greater than that of the design case. The staff concludes that the full range of accident and abnormal situations has been considered.

The results of the analyses show that, for 8x8R assemblies with uniform fuel enrichment of 3.5 w% U-235, the k -effective of the racks is 0.937 including all uncertainties (taken at a 95% probability with 95% confidence). Since a uniform enrichment distribution bounds the results for anticipated enrichment distributions, the staff concludes that storage of fuel assemblies having average planar enrichments of less than or equal to 3.5 w% U-235 enrichment may be safely stored in the GGNS Unit 1 high density storage racks. This conclusion is based on the following:

- (1) Acceptable state-of-the-art methods verified by comparison with experiment were used in the analysis.
- (2) Acceptable, conservative assumptions were used in the analysis of the design case.
- (3) An acceptable set of uncertainties was considered.
- (4) Acceptable abnormal and accident situations were analyzed.
- (5) The results meet the staff's acceptance criterion of 0.95 for the k -effective value of the racks, including uncertainties.

2.2 Materials

The safety function of the spent fuel pool and storage rack system is to maintain the spent fuel assemblies in a subcritical array under all credible storage conditions. The staff has reviewed the compatibility and chemical stability of the materials, except the fuel assemblies, wetted by the pool water.

The high density fuel racks are constructed of Type 304 stainless steel, except for the neutron absorber material. The existing spent fuel pool liner is stainless steel. The high density spent fuel storage racks 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

consists of individual storage cells interconnected to form an integral structure.

The space that 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 contains oxygen-saturated demineralized water. The water chemistry control of the spent fuel pool was previously reviewed and found to meet NRC recommendations.

2.2.1 Corrosion and Materials Compatibility

The pool liner, rack lattice structure, and fuel storage racks are stainless steel, which is compatible with the storage pool environment. In this environment of oxygen-saturated high purity water, the corrosive deterioration of the

Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inch 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 storage tubes, and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex is composed of nonmetallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron-absorbing material. The tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion (Anderson, 1979). During tests performed at the University of

Michigan (Anderson, 1981), Boraflex was exposed to 1×10^{11} rads of gamma radiation with substantial concurrent neutron flux in deionized water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to breakup of the Boraflex. The annulus space that contains the Boraflex is vented to the pool at each storage tube assembly. 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 or swelling of the inner stainless steel wrapper.

The tests (Anderson, 1979) have shown that neither irradiation, environment, nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions were reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally in the Boraflex will typically contain 0.1 w% of soluble boron. The test 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 licensees have

committed to conduct a long-term fuel storage cell inservice surveillance program. Surveillance samples are removable stainless steel clad Boraflex sheets, which are prototypical of the fuel storage cell walls. These specimens will be removed and examined periodically over the expected service life.

2.2.2 Conclusion

From the evaluation above, the staff concludes that corrosion of the high density spent fuel racks in the spent fuel storage pool environment will be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys that 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.

The staff further concludes that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool are adequate on the basis of the test data cited above and actual service experience in operating reactors.

The staff has reviewed the surveillance program and concludes that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensees, 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 licensees will reveal any deterioration of the Boraflex that might lead to the loss of neutron-absorbing capability during the life of the spent fuel racks. The staff expects that significant deterioration will not occur. However, should deterioration occur, this monitoring program will ensure that the licensees will be aware of it in sufficient time to take corrective action.

The staff, therefore, finds that the implementation of an inservice surveillance program and the selection of appropriate materials of construction by the licensees meet the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 61, regarding a capability to permit appropriate periodic inspection and testing of components, and GDC 62, regarding the prevention of criticality by maintaining the structural integrity of components and of the boron neutron absorber and are, therefore, acceptable.

2.2.3 References

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

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

2.3 Structural Design

The staff's evaluation of the high density racks is based on a review performed by NRC's consultant, Franklin Research Center (FRC). The FRC Technical Evaluation Report, TER-C5506-579, is appended to this Safety Evaluation as an appendix.

2.3.1 Description of the High Density Racks and Spent Fuel Pool

The new high density racks are stainless steel "egg-crate" structures. Each cell contains a spent fuel assembly, and a typical rack consists of approximately 300 cells. Weight of the rack and fuel is transmitted to the floor of the pool through supporting legs. Each rack is free standing on the pool floor, and a gap is provided between the racks and between the racks and the pool wall so as to preclude impact during earthquake.

GGNS Unit 1 has an upper containment pool containing fuel storage racks additional to those in the spent fuel storage pool. The upper containment pool is adjacent to the reactor cavity inside the containment. This upper containment pool was designed for temporary storage of spent or new fuel during refueling activities until the fuel could be moved to the spent fuel storage pool or replaced in the reactor vessel during core reload. The spent fuel pool is in the auxiliary building and is designed for long-term storage of spent fuel during reactor operation. Both the upper containment pool and the spent fuel pool are reinforced concrete structures.

2.3.2 Applicable Codes, Standards, and Specifications

The staff found that the licensees' load combinations and acceptance criteria were consistent with those 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 staff evaluated the existing concrete pool structure for the new loads in accordance with the requirements of Grand Gulf Nuclear Station Final Safety Analysis Report (FSAR) Section 3.8.4, which was approved by the staff during the plant operating license review.

2.3.3 Seismic and Impact Loads

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 plant design basis is a 0.075g operating basis earthquake (OBE) and a 0.15g safe shutdown earthquake (SSE). The seismic loads were applied to the model in three orthogonal directions. Loads resulting from a fuel bundle drop accident were considered in a separate analysis.

The postulated loads from these events were found acceptable. Further details are provided in the appendix.

2.3.4 Analyses of the Racks and Pool Structures

The dynamic response and internal stresses and loads for the racks are obtained from a seismic analysis that is performed in two phases. The first phase is a time history analysis on a simplified nonlinear lumped mass model. The second phase is a stress analysis of a detailed linear three-dimensional finite element model. The methodology is discussed further in the appendix. 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 the results were considered to be 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 acceptance limits.

The existing pool 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. Design criteria, including loading combinations and allowable stresses, are in compliance with the Grand Gulf FSAR, and it has been determined that the existing spent fuel pools can safely support the loads generated by the new fuel racks.

2.3.5 Conclusion

On the basis of its review of the structural aspects of the information submitted by the licensees in support of their request to allow installation of high density spent fuel racks in the existing spent fuel pool, the staff concludes that the high density spent fuel racks are structurally acceptable.

2.4 Installation of Racks and Load Handling

Since GGNS Unit 1 is in its first fuel cycle, the originally installed spent fuel racks were not used for storing spent fuel. The licensees removed the old racks and installed the new high density spent fuel racks in a planned outage in the fall of 1985 in which other work was also accomplished.

The licensees performed a safety analysis pursuant to 10 CFR 50.59 to determine whether the removal of the old racks and installation of the new racks (without placing spent fuel in them) would involve an unreviewed safety question. The licensees concluded that it would not. A license condition prohibits the storage of spent fuel in the spent fuel pool until the standby service water system is modified. This work on the standby service water system is scheduled to be completed during the first refueling outage.

The rerack was completed using plant procedures for handling heavy loads that were developed from the guidelines in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980. The licensees' compliance with the criteria of Phase I of NUREG-0612 was found acceptable by the staff in Supplement No. 5 to the Safety Evaluation Report for the Grand Gulf Nuclear Station (NUREG-0831), dated August 1984, and compliance with Phase II of NUREG-0612 was found acceptable by the staff in its letter to the licensees dated April 4, 1985. The licensees stated that no heavy loads were dropped during the removal of the old racks and installation of the new racks.

For carrying heavy loads over spent fuel in the high density racks, the procedures developed from the guidelines of NUREG-0612 are applicable. In addition, because a postulated drop of the spent fuel pool gate onto the racks containing fuel could damage the fuel, administrative procedures will be used to prevent moving the gate over racks that contain spent fuel.

The staff concludes that procedures for handling heavy loads over the spent fuel stored in the high density racks are acceptable.

2.5 Radiological Consequences of Accidents

The review of postulated accidents was conducted according to the guidance of Standard Review Plan Section 15.7.4 (NUREG-0800), NUREG-0554 ("Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979), and NUREG-0612 with respect to accident assumptions.

2.5.1 Cask Drop Accidents

The licensees' submittal has indicated that no change would occur to the equipment used in cask handling or transport operations as a result of the proposed spent fuel pool and upper containment pool modifications. Specifically, the potential cask drop distance is limited to 30 feet and the cask handling crane is single failure proof. These cask handling conditions are acceptable without calculation of radiological consequences. The staff concludes, therefore, that, with respect to a cask drop accident, the assumptions and conclusions reported in Section 15.3.3 of NUREG-0831, the Grand Gulf Nuclear Station Safety Evaluation Report (SER) dated September 1981, remain valid and no additional analyses are necessary for the proposed modification.

2.5.2 Construction Accidents

Because the old racks were moved and the new racks were installed before any spent fuel was stored in the originally installed racks, there was no potential for a construction accident involving stored spent fuel.

2.5.3 Fuel Handling Accidents

The licensees have proposed to expand the storage capacity of the spent fuel pool from 1270 spent fuel assemblies to 4348 spent fuel assemblies and the storage capacity of the upper containment pool from 170 spent fuel assemblies to no more than 800 spent fuel assemblies. The maximum weight of the loads that may be transported over spent fuel in either of the pools is limited to less than 1140 pounds by Technical Specification 3/4.9.7 and would not be changed by the proposed amendment. The spent fuel cask handling crane rails do not extend over any portion of the spent fuel pool. The proposed license amendment does not, therefore, increase the radiological consequences of a postulated fuel handling accident considered in the SER of September 1981 because this accident would still result in, at most, release of the gap activity of one fuel assembly because of the limitations on available impact kinetic energy.

2.5.5 Conclusion

The staff concludes that the assumptions and conclusions for the fuel handling accidents and cask drop accidents presented in the SER dated September 1981 for the originally installed spent fuel racks remain valid for the high density spent fuel racks. Therefore, the staff concludes that the high density spent fuel racks are acceptable with respect to fuel handling accidents because the calculated doses for the cask drop and fuel handling accidents remain unchanged and are within the NRC dose criterion.

2.6 Occupational Radiation Exposure

2.6.1 Evaluation

The staff has reviewed the licensees' removal and disposal of the low density racks and the installation of the high density racks, with respect to occupational radiation exposures. Because the spent fuel pool for GGNS Unit 1 has never had spent fuel stored in it and is currently clean and uncontaminated, the dose to workers resulting from the spent fuel pool modification itself is estimated to be less than 1 person-rem. Thus, the staff concludes that exposure to workers resulting from the spent fuel pool modification is as low as is reasonably achievable (ALARA) and is acceptable.

The staff has estimated the increment in the onsite occupational doses resulting from the proposed future increase in stored fuel assemblies at GGNS Unit 1. The estimate is based on information supplied by the licensees and on assumed occupancy times and estimated dose rates in the spent fuel pool area from radionuclide concentrations in the spent fuel pool water. The licensees have developed a loading pattern for the high density spent fuel racks in the spent fuel pool that will maintain occupational dose rates from spent fuel assemblies at less than 2.5 mr per hour. On the basis of present and projected operations in the spent fuel pool area, the staff estimates that the proposed modification should add less than 1% to the total annual occupational radiation dose at the plant. This small increase in the radiation dose in the spent fuel pool area should not affect the licensees' ability to maintain individual occupational doses at ALARA levels and within the limits of 10 CFR Part 20.

2.6.2 Conclusion

The staff finds that storing additional fuel in the Unit 1 spent fuel pool, in accordance with the proposed loading pattern, will not result in any significant increase in doses received by plant personnel and should not affect licensees' ability to maintain individual occupational doses at ALARA levels and within the limits of 10 CFR Part 20.

2.7 Spent Fuel Pool Cooling System

This section of the Safety Evaluation deals with the acceptability of the capability to provide adequate cooling to the spent fuel in the spent fuel pool, the proposed Technical Specifications, and the predicted decay heat generation rates from the spent fuel.

2.7.1 Evaluation

The licensees calculated the spent fuel pool decay heat generated in the spent fuel pool from normal refuelings and considering a full core offload in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," and Standard Review Plan (NUREG-0800) Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The licensees' calculations showed that the heat generation rate for the spent fuel pool with normal refuelings (normal heat load) is 18.93 MBTU per hour, and the heat generation rate for the full spent fuel pool with the last 800 fuel assemblies a core offload (abnormal heat load) is 47.81 MBTU per hour. The staff has performed an independent calculation of the two cases, and the results confirm that the licensees have used the appropriate methods for determining the heat generation rates.

The staff also performed an analysis of the spent fuel pool water temperature based on the normal and abnormal heat load conditions. The analysis indicated that the bulk pool water temperature for the normal heat load case would be 171°F, and for the abnormal heat load case the water temperature would be 212°F. Since the pool water temperature for the normal heat load case is higher than the 140°F pool water temperature identified in Section 9.1.3 of the Standard Review Plan, the licensees have committed, in a June 5, 1986, submittal, to limit the amount of spent fuel stored in the spent fuel pool to a maximum of 2324 spent fuel assemblies. This represents 10 reloads. The spent fuel pool water temperature was calculated for storing 2324 fuel assemblies and using the residual heat removal (RHR) system to remove the decay heat from the pool for the first 35 days following power operation. This is anticipated to

represent approximately 30 days following the removal of the reactor head and the beginning of the transfer of the first fuel assembly to the spent fuel pool. On the basis of limiting the maximum number of spent fuel assemblies to 2324 and the commitment to use the RHR system for the first 35 days, the spent fuel pool water temperature will be less than the 140°F identified in the Standard Review Plan and is, therefore, acceptable.

In the submittal dated June 5, 1986, the licensees committed to propose an acceptable engineering solution to the current inadequacy of the spent fuel pool cooling system to bring the plant into conformance with the Standard Review Plan for the physical limit of the high density spent fuel racks (4348 fuel assemblies) before startup following the third refueling. The licensees also committed to implement the solution before startup following the fifth refueling. In a submittal dated July 25, 1986, the licensees briefly identified two potential engineering solutions that would provide adequate spent fuel pool cooling system capacity. These solutions were (1) increase the spent fuel pool pumping capacity or (2) increase the heat exchanger capacity either by replacing an existing heat exchanger or by adding new heat exchangers. Either of these solutions appears to be an acceptable approach to increasing the capability of the spent fuel pool cooling system so that the full capacity of the high density spent fuel racks could be utilized. Such utilization would require a future Technical Specification change in addition to the currently proposed change.

The spent fuel pool water temperature, based on the abnormal heat load case, is estimated to be 212°F with the storage of 2324 fuel assemblies and only using the spent fuel pool cooling system. In this case, one loop of the RHR system is available to provide adequate cooling of the spent fuel pool.

The licensees have proposed a Technical Specification that would limit the maximum spent fuel pool water temperature to 140°F in accordance with Standard Review Plan Section 9.1.3. In addition, if the water temperature exceeds the 140°F limit, the licensees have committed to restore the water temperature to less than 140°F in less than 8 hours, or to begin shutdown of the plant and to achieve hot shutdown within 12 hours and cold shutdown within the following 24 hours. Furthermore, the licensees committed to determine if the spent fuel pool water temperature could be expected to exceed 210°F within 20 hours of exceeding the Technical Specification temperature limit of 140°F. If the water temperature is projected to exceed 210°F within 20 hours, the action specified in Technical Specification 3.0.3 would be applicable. This represents the most rapid and orderly shutdown with the minimum transient on the reactor and related systems, in order to achieve safe shutdown before removing one loop of the RHR system from the reactor service mode and placing it into the spent fuel pool cooling mode of operation. Because the proposed Technical Specifications conform to the Standard Review Plan and the evaluation in the GGNS Safety Evaluation Report, NUREG-0831, the staff concludes that the proposed Technical Specifications are acceptable.

Proposed Technical Specification 5.6.3 reflects the new physical storage capacity of the spent fuel storage facility with the limitation of a maximum usable storage of 2324 fuel assemblies. In addition, it reflects the limit of 800 fuel assemblies in the upper containment pool. In a submittal dated June 9, 1986, the licensees modified the proposed Technical Specifications to require removal of all spent fuel from the upper containment pool before returning the reactor to a critical condition following a refueling. This conforms to the

standard practice of not storing spent fuel inside the containment during reactor operation and is, therefore, acceptable.

The spent fuel pool cooling system, with the exception of the cleanup portion, is designed to Quality Group C and seismic Category I requirements. The spent fuel pool cooling system can be powered from redundant divisions of the Class 1E power system. In case of a seismic event, a seismic Category I bypass line and redundant seismic Category I isolation valves have been provided at the cleanup system connections to the fuel pool cooling lines to isolate the nonseismic Category I portion of the system to ensure that failure in that portion of the system has no adverse effect on safety-related equipment. This design satisfies the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification."

The nonsafety-related component cooling water system provides cooling water to the fuel pool heat exchangers under normal operating conditions. Backup cooling is provided by the seismic Category I standby service water system (SSWS), which transfers the spent fuel pool heat loads to the ultimate heat sink. During testing of the system before the plant was licensed, the licensees determined that the SSWS pumps were undersized and unable to provide design flows for safety-related components and the full spent fuel storage facility. Accordingly, License Condition 2.C.(20) was imposed, which prohibits the storage of spent fuel in the spent fuel pool until the SSWS is modified to provide design cooling water flows to all safety-related components, including the spent fuel pool cooling system heat exchangers. This modification of the SSWS will be completed during the first refueling outage. On the basis of its independent analysis, the staff concludes that once the SSWS has been properly modified, there should be adequate cooling water flow to the spent fuel pool cooling system heat exchangers to remove the decay heat generated by 2324 spent fuel assemblies with the reloading pattern specified in Table 1.1 of the licensees' May 6, 1985, submittal. Thus, the requirements of GDC 44, "Cooling Water," have been satisfied for the storage of 2324 spent fuel assemblies, subject to satisfactory modification of the SSWS.

The staff requested that the licensees discuss the redundancy of components so that the spent fuel can be adequately cooled assuming a single active failure concurrent with the loss of offsite power, as specified in the Standard Review Plan. In a response dated July 25, 1986, the licensees described their procedure to provide alternative cooling for spent fuel in the spent fuel pool and the reactor in the event of a loss of offsite power (LOOP) concurrent with a single failure for normal plant operating conditions. In particular, the licensees addressed the operational conditions of cold shutdown and refueling with only the equipment and systems available that are required by the plant Technical Specifications. For these two operational conditions, the worst single failure is the failure of the one required diesel generator. The licensees committed to revise emergency procedures to include the operator actions necessary in the event of a LOOP with the failure of a diesel generator under cold shutdown or refueling conditions. The operator actions would require the use of the station fire truck to pump water from the fire water storage tanks via fire hoses through stairwells and into secondary containment and

primary containment to maintain the water level in the spent fuel pool and in the reactor. Manual operation of the valves in the spent fuel pool cooling system, the low pressure core spray systems, and the RHR system will allow a controlled flow of water from the reactor to the suppression pool to prevent bulk boiling in the pools. The licensees have evaluated the water flow requirements and have determined that the required flow rate of 720 gpm is well within the capability of the fire truck's 1000-gpm capacity, and the two fire water storage tanks, which have a total capacity of 600,000 gallons, will provide more than 8 hours of cooling for the spent fuel. The licensees committed to provide these procedures for the staff's review and approval before entering the first refueling outage. Because the licensees have a defined method of providing adequate cooling for the spent fuel in the event of a LOOP and a single failure and have committed to provide adequate procedures, the staff concludes that the design features of the plant in conjunction with the procedures are acceptable to provide alternative cooling for spent fuel in the spent fuel racks and in the reactor during refueling operations in the event of a loss of offsite power and the worst single active failure.

2.7.2 Conclusion

The staff concludes that the spent fuel pool storage capacity modifications are acceptable for the storage of 2324 fuel assemblies (out of a total capacity of 4348 fuel storage locations) with respect to the rack storage capacity, the developed heat loads, the pool water temperatures, and the capability of the spent fuel pool cooling and support systems.

The staff further concludes that the approach described by the licensees for cooling spent fuel in the spent fuel pool, upper containment pool, and the reactor in the event of a loss of offsite power and a single failure during refueling is acceptable.

2.8 Radioactive Waste Treatment

GGNS Unit 1 contains radioactive waste treatment systems designed to collect and process the gaseous, liquid, and solid waste that might contain radioactive material. The staff evaluated and found acceptable the radioactive waste treatment systems in its Safety Evaluation Report (NUREG-0831) dated September 1981, in support of the issuance of the operating license. There is no change in the conclusions regarding the evaluation of these systems because of the use of the high density spent fuel racks. Therefore, the staff concludes that the radioactive waste treatment systems are acceptable for use with the high density spent fuel racks.

3.0 ENVIRONMENTAL CONSIDERATION

A separate Environmental Assessment has been prepared pursuant to 10 CFR 51. The Notice of Availability of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on August 18, 1986 (51 FR 29527).

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration, which was published in the Federal Register

(51 FR 26078) on July 18, 1986, and consulted with the State of Mississippi. No public comments were received, and the State of Mississippi did not have any comments.

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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APPENDIX
TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION REPORT

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EVALUATION OF SPENT FUEL RACKS STRUCTURAL ANALYSIS

MISSISSIPPI POWER AND LIGHT COMPANY

GRAND GULF NUCLEAR STATION UNIT 1

TER-C5506-579

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, 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 Mississippi Power and Light Company licensing report [1] on high-density spent fuel racks for the Grand Gulf Nuclear Station Unit 1 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 freestanding. Since the racks are not anchored to the pool floor or the pool walls, the motion of the racks during a seismic event is governed by the static/dynamic friction between the rack's mounting feet and the pool floor and by the hydrodynamic coupling to adjacent racks and the pool walls.

Accordingly, this report covers the review and evaluation of analyses submitted for the Grand Gulf plant by the Licensee, wherein the structural analysis of the spent fuel racks under seismic loadings is of primary concern

due to the nonlinearity of gap elements and static/dynamic friction, as well as fluid-structure interaction. In addition to the evaluation of the dynamic structural analysis for seismic loadings, the design of the spent fuel racks and the analysis of the spent fuel pool structure under the increased fuel load are reviewed.

2. 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:

- o OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, U.S. Nuclear Regulatory Commission, January 18, 1979 [2]
- o 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
- o ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers
 - Section III, Subsection NF, Component Supports
- o 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
- o Other Industry Codes and Standards
 - American National Standards Institute, N210-76
 - American Society of Civil Engineers, Suggested Specification for Structures of Aluminum Alloys 6061-T6 and 6067-T6.

2.2 PRINCIPAL ACCEPTANCE CRITERIA

The principal acceptance criteria for the evaluation of the structural analysis of the spent fuel racks for the Grand Gulf Unit 1 plant 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."

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.

* American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, 1980 Edition.

** American Institute of Steel Construction, Latest Edition.

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

Temperature gradients and mechanical load combinations are to be considered in accordance with Section IV-4 of the OT Position Paper.

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

As described in the Licensee's report [1], the spent fuel rack modules are totally immersed in the spent fuel pool, wherein the water in the pool produces hydrodynamic coupling between the fuel assembly and the rack cell, as well as between the fuel rack module and adjacent modules. The hydrodynamic coupling significantly affects the dynamic motion of the structure during seismic events. The modules are freestanding, that is, they are not anchored to the pool floor or connected to the pool walls. Thus, frictional forces between the rack base and the pool liner act together with the hydrodynamic coupling forces to both excite and restrain the module in horizontal and vertical directions during seismic events. As a result, the modules exhibit highly nonlinear structural behavior under seismic excitation, for which it is necessary to adopt time-history analysis methods to generate accurate and reliable analytical estimates.

Pool slab acceleration data used in the analysis were derived from the original pool floor response spectra. Structural damping of 4% for the racks was assumed for the safe shutdown earthquake (SSE) condition.

A lumped mass dynamic model was formulated by the spent fuel racks' vendor in accordance with computer code DYNAHIS to simulate the major structural dynamic characteristics of the modules. Two sets of lumped masses were used, one to represent the fuel rack module and another to represent the fuel assemblies. The lumped masses of these racks were connected by beam elements. The lumped masses of fuel assemblies were linked to those of the rack by gap elements (nonlinear springs). Frictional elements (springs) were used to represent the frictional force between the rack base and pool liner. Hydrodynamic masses were included in the model to approximate the coupling effect between the water and the structure. The model was subjected to the simultaneous application of three orthogonal components of seismic loads derived from a stated earthquake with one vertical and one horizontal component.

An elastostatic model was first used to evaluate element stiffness characteristics for use in the dynamic model. The results generated from the dynamic model, in terms of nodal displacements and forces at nodes and elements, were then introduced to the elastostatic model to compute the detailed stresses and corner displacements in the module.

The resulting stresses at potentially critical locations of the module were examined for design adequacy in accordance with the acceptance criteria. The possibilities of impact between adjacent racks and the tipping of the module were also evaluated.

3.2 EVALUATION OF THE ELASTOSTATIC MODEL

3.2.1 Element Stiffness Characteristics

An analytic approach for stressed-skin models was adopted to evaluate the stresses and deformations in the rack modules [1]. Essentially, the module was represented by lumped masses linked by beam elements possessing equivalent bending, torsional, and extensional rigidities and shear deformation coefficients. These properties were used to determine the stiffness matrix for the elastic beam elements.

Impact springs were used between the lumped masses of the fuel assemblies and those of the fuel rack to simulate the effect of impact between them. The spring rates of these impact springs were determined from the local stiffness of a vertical panel and computed by finding the maximum displacement of a 6.0-in-diam circular plate built in around the bottom edge and subjected to a specified uniform pressure. The Licensee did not mention the corresponding compliance of the fuel assembly in determining the value of the impact springs. The effect of neglecting the compliance of the fuel assembly is conservative in that it would sharpen the impact force, i.e., produce a higher force for a shorter time.

Linear frictional springs in two orthogonal directions were placed at four corner positions on the rack base to represent the effect of the static frictional force between each mounting pad and the pool liner. Angular

frictional springs about the vertical axis of each pad representing the distribution of pad friction under angular motion were not provided in the model. Review of the application of angular frictional springs indicated that their contribution to the displacement solution would be negligible.

3.2.2 Stress Evaluation and Corner Displacement Computation

Computer code "EGELAST", a proprietary code of the Joseph Oat Corporation, was used to compute critical stresses and displacements in the rack module and its support. Nine critical locations were identified on the cross section of rack chosen for stress evaluation, including the four corners of the cross section, the midpoint of each of the four sides, and its center. Results from the dynamic model were input to "EGELAST" for computation. Stresses were evaluated at each of the nine critical locations at each selected cross section of the rack. Displacements were calculated at each of the four corners of the cross section. Maximum stresses and corner displacements were determined for all time steps.

With respect to the computed values used from the nonlinear dynamic displacement analysis, the Licensee provided the following [3]:

"The loads in the bending, shear and extensional springs in the dynamic model are transferred to the post-processor EGELAST which computes the maximum bending and shear stresses in the rack using the principles mentioned in Section 6.3.1. EGELAST has been benchmarked on numerous problems and has been used for licensing several rack projects."

3.3 EVALUATION OF THE NONLINEAR DYNAMIC MODEL

3.3.1 Assumptions Used in the Analysis

The following assumptions were used in the analysis:

- a. Adjacent rack modules were assumed to have motions equal and opposite to the rack module being analyzed. This defined a plane of symmetry in the fluid of each space between the module being analyzed and the adjacent modules and permitted the analysis of an isolated rack module.

- b. All fuel rod assemblies in a rack module were assumed to move in phase. This was necessary for the lumped mass model and was assumed to produce the maximum effects of the fuel assembly/storage cell impact loads.
- c. The effect of fluid drag was conservatively omitted.

Assumption "a" was made to reduce the collection of fuel racks in the spent fuel pool to a manageable three-dimensional problem--that of one rack module. The assumption offers a degree of conservatism in that it reduces the available clearance space between rack modules for dynamic displacement without impact to one-half the initial clearance. A further discussion of its effects upon hydrodynamic coupling is presented in Section 3.3.3 of this report.

Assumption "b", said to offer conservatism, is not necessarily conservative. Regardless of the initial position of each individual fuel assembly, all fuel assemblies within a fuel rack module will settle into in-phase motion soon after the rack module is set in motion. This is because each fuel assembly is a long vertical column which pivots about its base and moves within a small clearance space within the rack cell.

With respect to Assumption "c", review indicates that fluid drag is a complex issue [4, 5, 6]. The OT Position Paper [2], which forms the principal basis of acceptance criteria for this plant, indicates from a previous study [5] that viscous damping is generally negligible and that increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results. However, a more recent paper [6] indicates that the hydrodynamic damping of a perforated plate vibrating in water is comprised of two regimes, the smaller of which is proportional to the kinematic viscosity, while the larger is "a non-linear regime where the log decrement is proportional to the vibrational velocity and is independent of viscosity." Thus, even for the small displacements of a vibrating perforated plate where hydrodynamic flow about the plate is not developed, Reference 6 indicates that fluid damping independent of viscosity is present. This is supported by Fritz [4], who, in addition to developing relationships for coupled hydrodynamic mass in submerged flexible body vibration, developed the

associated damping relationships based upon Darcy friction factors that also show damping to be proportional to velocity as well as to fluid density. Although Fritz's relationships indicate the damping magnitude to be very small, the motion of a fuel assembly throughout its clearance from the cell walls is sufficient to promote some hydrodynamic flow about, and through, the fuel assembly that is more fully developed than for the case of vibrating bodies.

As the Licensee has not taken any credit for impact structural damping of the limber fuel assembly, it appears that a small amount of damping could be justified as either impact damping of the fuel assembly or equivalent fluid drag without compromising the conservatism of the analysis.

3.3.2 Lumped Mass Model

The lumped mass approach was used in the dynamic model, wherein the mass of the fuel rack was lumped at five equidistant locations as shown in Figure 1. For horizontal motion, the rack mass was proportioned at one-quarter of the total mass for each of the three middle mass nodes and at one-eighth of total mass each for the top and the bottom nodes. The mass of the base plate and support structure was lumped with the bottom node. For the fuel assemblies, five lumped masses were used in a similar pattern of distribution. For vertical motion, two-thirds of the racks' dead weight acted at the bottom mass node, with the remaining one-third applied at the top node. All of the dead weight (gravitational force) of the fuel assembly was at the bottom node.

3.3.3 Hydrodynamic Coupling Between Fluid and Rack Structure

When an immersed fuel rack is subject to seismic excitation, hydrodynamic coupling forces act between the fuel assembly and fuel rack masses, as well as between the fuel rack and adjacent structures. The Licensee applied the linear model of Fritz [4] to estimate these coupling effects. In evaluating the hydrodynamic coupling between adjacent racks, the Licensee also assumed that the rack was surrounded on all four sides by rigid boundaries separated from the rack module by an equivalent gap. As discussed previously in Section 3.3.1, the Licensee chose to model the dynamic condition wherein adjacent rack modules were assumed to have motions equal and opposite to the module being

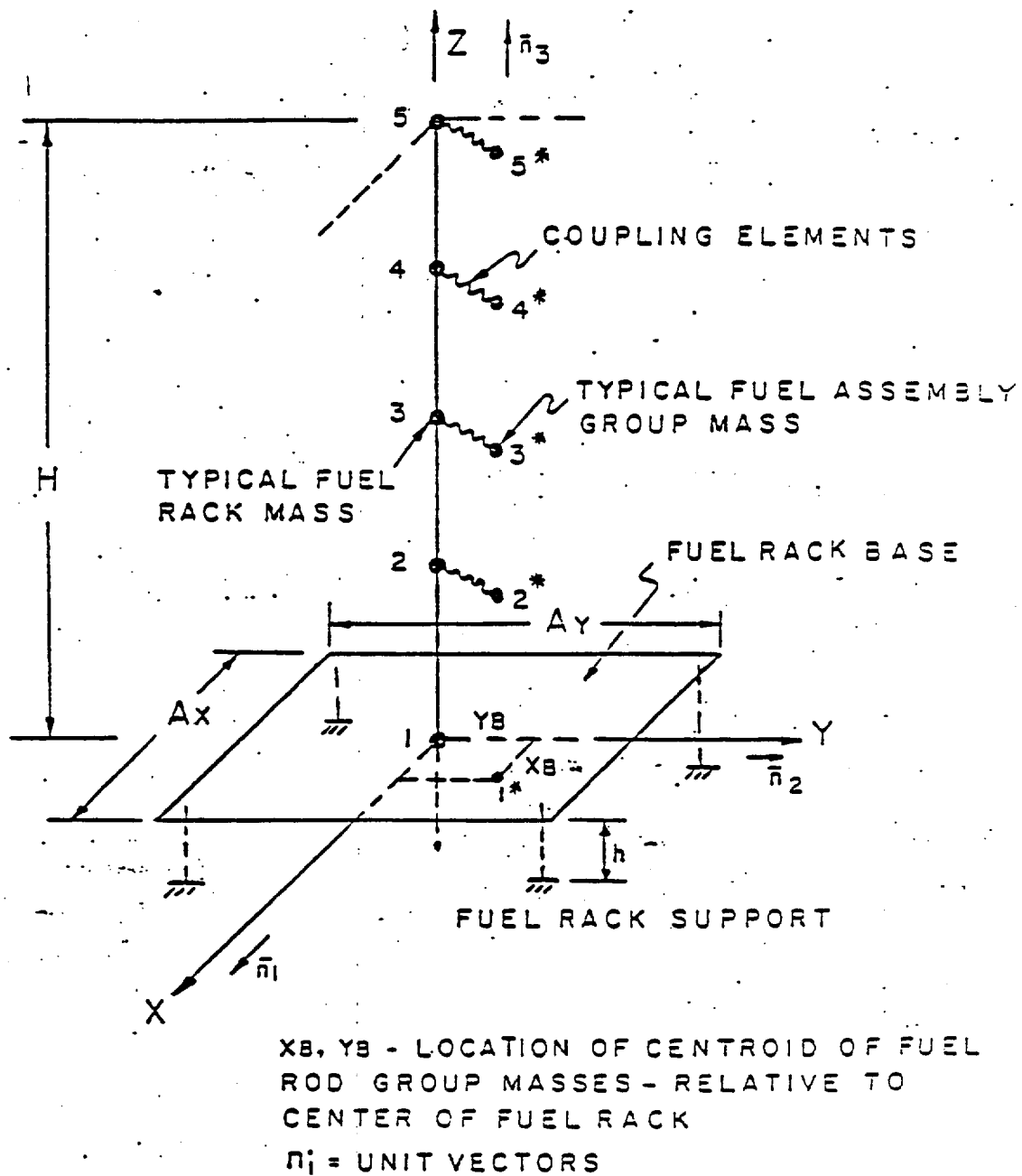


Figure 1. Dynamic Model

analyzed. Although this assumption neglects the fact that adjacent rack modules may have quite different dynamic response characteristics, such as to interact and respond as a global system, it does provide a very manageable reduction in the analytic modeling of the problem while addressing the case in which the available space for dynamic rack displacement is at a minimum. Review and evaluation of this assumption has indicated that, although the associated conservatism cannot be evaluated directly within the scope of this review, the assumption is considered to provide an adequate modeling technique so long as the resulting dynamic displacements remain relatively small compared with the available displacement space.

Fritz's [4] 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 [4] 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 fuel assemblies and the rack cell walls, as well as between adjacent fuel rack modules or 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 assembly within a rack cell, as well as 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. However, the method is acceptable where the rack displacements are not large compared with the available displacement space.

3.3.4 Equations of Motion

The Licensee included 32 degrees of freedom in the three-dimensional lumped mass model [1]. All rack mass nodes were free to translate and rotate about two orthogonal horizontal axes. The top and bottom rack mass nodes had additional freedom for translation and rotation with respect to the vertical axis. The bottom fuel assembly mass node was assumed fixed to the base plate, whereas the remaining four fuel assembly mass nodes were free to translate along the two horizontal axes.

The structural behavior of the lumped mass model was completely described in terms of 32 equations of motion, one for each degree of freedom, which were obtained through the Lagrange equations of motions. Review and evaluation have confirmed the acceptance of this approach.

3.3.5 Seismic Inputs

The time history accelerations of the seismic motion used as input data for the dynamic equations were stated [1] to have been developed by the Bechtel Corporation for the Grand Gulf plant. The history acceleration plots of input data included by the Licensee were as follows [1]:

Auxiliary Pool

- East-West acceleration
- North-South acceleration
- Vertical acceleration

Containment Pool

- East-West acceleration
- North-South acceleration
- Vertical acceleration

Because the Licensee provided a full three-dimensional dynamic analysis, input to the dynamic equations was reported to include two simultaneous orthogonal components of horizontal acceleration concurrently with the vertical seismic acceleration.

3.3.6 Integration of the Dynamic Equations

Because the equations of motion include nonlinear parameters that change value suddenly for the simulation of fuel assembly impacts and racks lifting off the floor, the integration procedures employed must include additional precautions to assure that the integration remains stable and that the solution reached is a fully converged solution. Since the magnitude of the integration time step (ΔT) is critical to both stability and convergence, a well-accepted technique is to repeat the solution of the set of equations using a range of values for the integration time step and to compare the results.

If the computed displacements compare closely in value for a wider range of integration time step values, then the integration is generally accepted as representing both a stable and fully converged solution.

The Licensee [7] performed five solutions with integration time steps ranging from 0.75×10^{-5} sec to 0.3×10^{-4} sec with computed displacements as shown in Table 1. This indicates that the time step value of 0.2×10^{-4} sec used in the Licensee's report [1] is acceptable.

Table 1. Displacement Convergence Study

<u>Time Step, T (sec)</u>	<u>Maximum Displacement (inch)</u>	<u>Coincident Time Instant (sec)</u>
0.3×10^{-4}	1.402	11.84
0.2×10^{-4}	1.27	11.31*
0.15×10^{-4}	1.29	8.66
0.1×10^{-4}	1.33	8.44
0.75×10^{-5}	1.33	8.43

3.3.7 Frictional Force Between Rack Base and Pool Surface

The Licensee used the maximum value of 0.8 and the minimum value of 0.2 to cover the range of static coefficients of friction between rack base and pool liner.

Rabinowicz, in a report to the General Electric Company [9], focused attention on the mean and the lowest coefficient of friction. Rabinowicz also discussed the behavior of static and dynamic friction coefficients, indicating that the dynamic, or sliding, coefficient of friction is inversely proportional to velocity. Thus, the use of static and dynamic coefficients of friction could produce larger rack displacements; that is, the higher value of static friction could permit the buildup of energy that may require a larger displacement at a lower value of dynamic friction to dissipate.

A key to the importance of the complicating consideration of static and dynamic friction appears to be whether significant rack energy is dissipated

*Per Reference 8.

in sliding friction. If only minimal rack energy is dissipated in sliding friction, then more complete methods of modeling friction would make very little difference in the resulting computed displacement.

3.3.8 Impact with Adjacent Racks

One of the Licensee's structural acceptance criteria [1] is the kinematic criterion which seeks to ensure that adjacent racks will not impact during seismic motion. As shown in Figures 2 and 3, gaps between racks and between the racks and walls vary from rack to rack.

In response to a request for additional information regarding the gap between the racks and the walls, the Licensee [3] confirmed that the shaded areas of Figures 2 and 3 denote the gap between the fuel racks and the pool walls. Thus, clearance is provided between the racks and walls and between adjacent racks.

For the Licensee's mathematical model, the no-collision-of-adjacent-racks criterion generally requires that the maximum rack displacement be smaller than half of the gap between racks. If both adjacent racks are analyzed and out-of-plane rack motion is considered, then the sum of their displacements should be less than the rack clearance. Although it is acceptable to use an average, or equivalent, gap for the purpose of assessing the contribution of fluid action around a fuel module with unequal spacing from other modules, the actual minimum operating gap must be used for comparison with the computed displacements. Although the module may, under the influence of seismic excitation and induced fluid forces, move toward the position of equal gaps from its initial position, repeated collision with adjacent modules could take place before any minimum gap is widened. Thus, comparison of the computed fuel module displacements with the minimum proportioned operating gap is essential.

The Licensee's response follows [3]:

"The reviewer is correct in stating that out-of-phase motion of neighboring racks is possible. Therefore, it is necessary to model the racks and the associated virtual and coupling hydrodynamic masses assuming a plane of symmetry midway in the gap region. All seismic analyses carried out for Grand Gulf Nuclear Station Unit 1 racks are based on this assumption. The computed maximum rack displacements are

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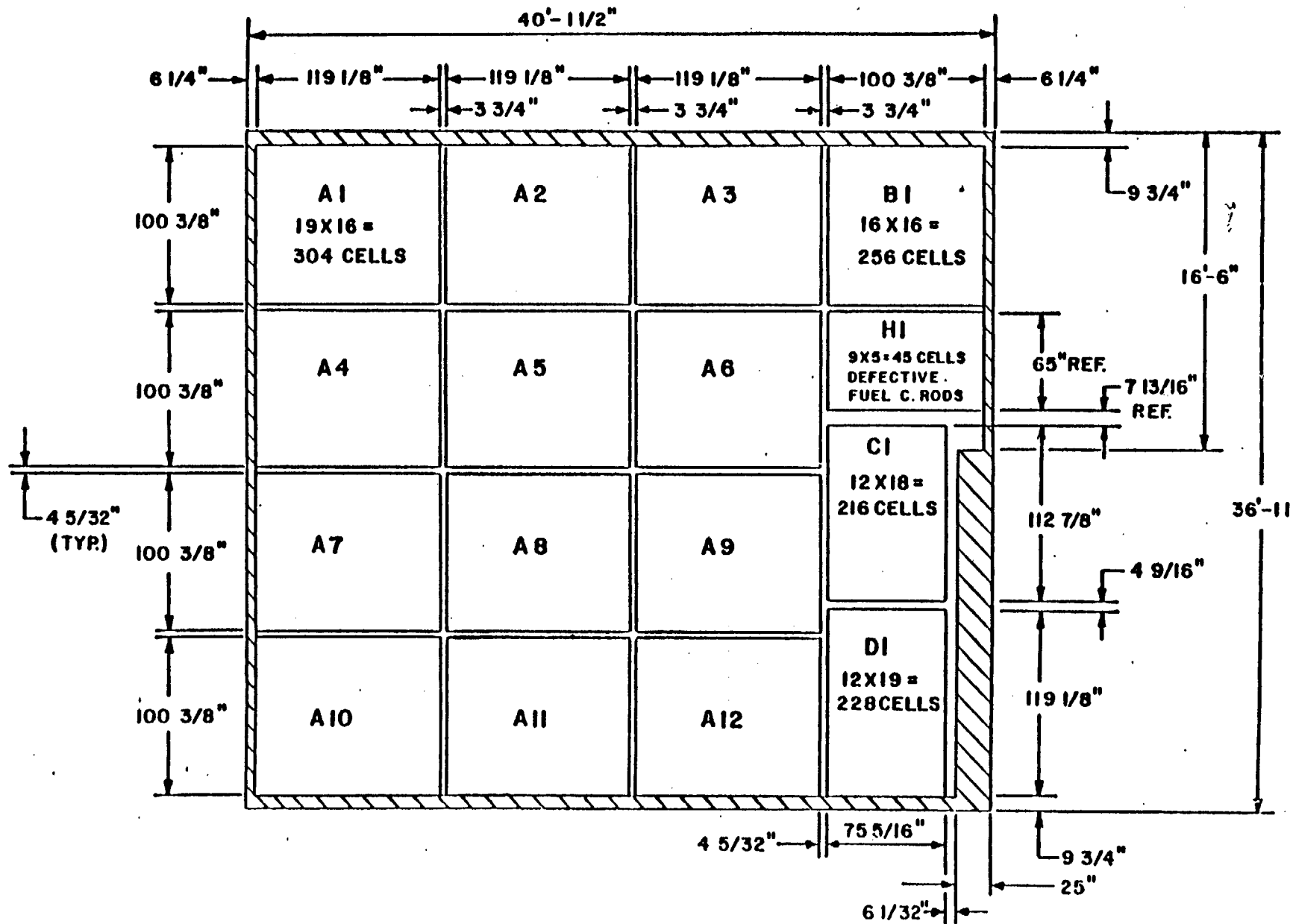


Figure 2. Racks' Arrangement in Spent Fuel Pool

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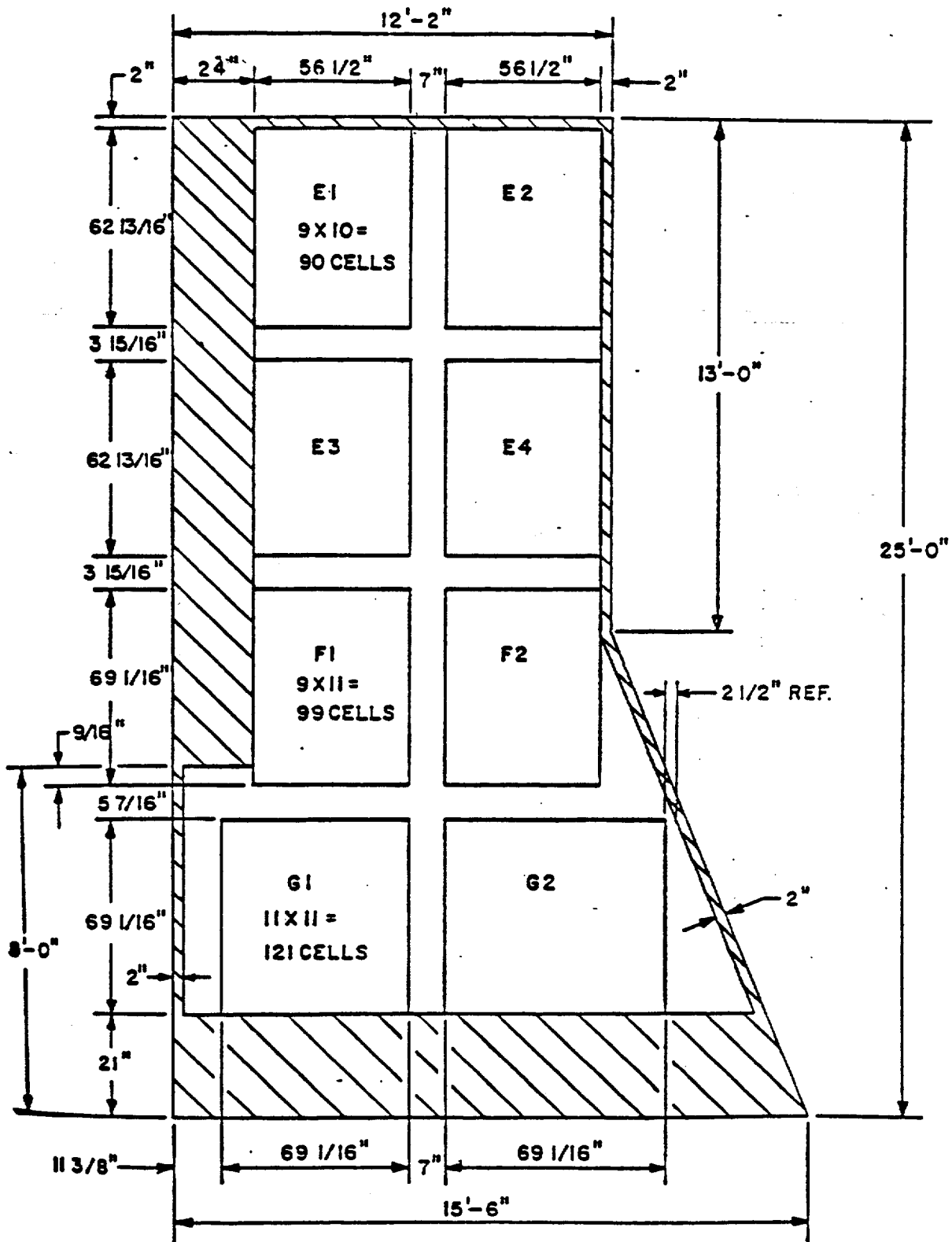


Figure 3. Racks' Arrangement in Containment Pool

required to be less than 50% of the nominal inter-rack gap. Data presented in Table 6.4 [1] shows that this requirement is met in all cases."

The Licensee's plan for proportioning gaps between racks for comparison with computed rack displacements is satisfactory.

3.3.9 Rack Displacements and Stresses

The Licensee provided tables of selected computed displacements representing the maximum rack movement at the top of the rack [1]. Displacements and stresses were stated to be for the safe shutdown earthquake (SSE) event for all loading cases except case 11, which considered the operating basis earthquake (OBE). The Licensee's description of the cases considered follows:

<u>Case Number</u>	<u>Description</u>
1	Full rack, damping > 2%, u = 0.8
2	Tripping analysis (1.5 SSE horizontal quake) 2% damping, u = 0.8
3	Full rack, u = 0.8, 2% damping
4	Full rack, u = 0.2, 2% damping
5	Half load, diag. fill, u = 0.8, 2% damping
6	Half load, diag. fill, u = 0.2, 2% damping
7	Half load, positive X quadrant, u = 0.8, 2% damping
8	Empty rack, u = 0.8, 2% or 4% damping
9	Empty rack, u = 0.2, 2% or 4% damping
10	Half load; positive X quadrant, u = 0.2, 2% damping
11	Full rack, u = 0.8, 2% damping, OBE quake

As discussed in Section 3.3.8, the Licensee correctly compared the computed rack displacements with the available space between the rack and a wall and with half the space between adjacent racks. Table 6.4 of the

Licensee's report [1] provides a listing of the displacements for each rack. With the exception of case 2 for rack G, all displacements are well within the allowable space. The larger displacements for rack G are associated with the tipping analysis (case 2), for which the earthquake excitation amplitudes were taken as 1.5 times the SSE earthquake. The higher earthquake excitation amplitude was used to show that the racks remain stable with respect to tipping. The criterion that the racks' displacement be less than half the space between racks does not apply.

As discussed in Section 3.3.6 of this report, there is evidence of a stable and sufficiently converged solution to the dynamic equations; thus, the rack displacements reported by the Licensee were found to be acceptable.

The Licensee reports the ratios of computed stresses to allowable stresses in Table 6.5 of the Licensee's report [1]. The Licensee reported that the stress ratios were computed using the SSE conditions such that the allowable ratio value for the OBE condition is 1.0 and 2.0 for the SSE condition. Review of the reported stress ratios indicated that all values are below their allowable values.

3.4 REVIEW OF SPENT FUEL POOL STRUCTURAL ANALYSIS

3.4.1 Assumptions

Grand Gulf Nuclear Station Unit 1 has two fuel pools that may be loaded with spent fuel. These are the upper containment pool and the spent fuel pool in the auxiliary building. In the course of analyzing the pool floors for both pools, the Licensee recognized that the upper containment pool has considerably higher bending and shear strength than the spent fuel pool (auxiliary building) although its floor loading was less. Consequently, the Licensee proceeded with the analysis of the spent fuel pool with the intent that the results be applicable to both pools.

Assumptions used in performing the analysis were:

- o The pool floor was modeled as a simply supported composite rectangular plate for the dynamic analysis. No credit was taken for structural resistance offered by the pool walls.

- o The stiffness and strength properties of the concrete floor were based upon the complete cracking of the concrete in tension over the entire floor.
- o Floor loading for the analysis assumed that all racks are fully loaded with channeled fuel assemblies.

3.4.2 Dynamic Pool Floor Analysis

With the pool floor modeled as a simply supported rectangular orthotropic plate, a dynamic time history load was applied and the equations were integrated to determine the maximum floor displacements. The results of the pool floor dynamic analysis were scanned to determine the maximum floor deformation computed. This maximum floor deformation was then used as the input value in a detailed static finite element analysis of the floor to determine the highest stresses in the beams and concrete associated with the dynamic loading.

It was noted that the dynamic loading was obtained from the results of the dynamic analysis of a fully loaded Type A spent fuel rack, and represented the sum of the fuel racks acting concurrently.

3.4.3 Pool Floor Analysis Conclusions

The Licensee summarized the loadings used for the spent fuel pool analysis in Table 8.1 [1] and provided samplings of the computed results in Table 8.2 [1] that indicate ample safety margins. Review of the pool floor modeling, loading, analysis, and computed results indicated that the methods used and the conclusions drawn by the Licensee are satisfactory.

3.5 REVIEW OF HIGH-DENSITY FUEL STORAGE RACKS' DESIGN

3.5.1 Jammed Fuel Handling Condition

With respect to the forces associated with jammed fuel handling equipment, the Licensee provided the following [1]:

"A 4000-pound uplift force and a 1000-pound horizontal force are applied at the top of the rack at the 'weakest' storage locations. The force is assumed to be applied on one wall of the storage cell boundary as an upward edge force over length l . It is shown that if the length l is over 2.46" then no yielding will occur. If l is smaller than 2.46", the damage is limited to the region above the top of the active fuel. Horizontal force of 1000 pounds applied at the top edge of a cell wall produces plastic deformation over 2" depth - well removed from the zone of active fuel."

This statement was reviewed and found to be acceptable.

3.5.2 Dropped Fuel Accident I

The Licensee [1] considered the accidental drop of a 600-pound fuel assembly from a position wherein the nose of the fuel assembly is 36 inches above the rack top lead-in to a storage position. The impact of the fuel assembly dropping through the storage position and hitting the baseplate was calculated by the Licensee to be absorbed without full penetration of the baseplate and without excessive loads transmitted through the rack mounting feet to the pool liner.

Review of this dropped fuel accident indicated that the results were acceptable.

3.5.3 Dropped Fuel Accident II

Section 7.1.3 of the report [1] discusses the effect of a fuel assembly dropping from a position 36 in above the rack and hitting the top of the rack. The report indicates that the maximum local stress is limited to 21 ksi, which is less than the 25-ksi yield stress of the material. Although no details were given in the report [1] about the possibility of local buckling that could alter the cross-sectional geometry of the racks, under these stresses it is understood that any buckling would be confined to a local region above the fuel, which is acceptable.

3.5.4 Liner Integrity Analysis

Section 7.4 of the Licensee's submittal addresses the analysis of stresses in the pool liner produced by the mounting feet of the fuel racks

under dynamic seismic conditions. Rack type A, the heaviest rack module, was used for the analysis.

The analysis for tearing of the liner by shear forces between the liner and the rack mounting feet was performed by assuming that the horizontal forces from the rack mounting feet were distributed over the cross-sectional area of liner adjacent to the mounting foot. That is, the stressed area of the liner is the liner thickness (0.25 inch) multiplied by the mounting foot width (approximately 14 inches).

Rack loading cases were:

- o Case 1, full rack, damping > 2%, $u = 0.8$
- o Case 5, half full rack (diagonal fill), 2% damping, $u = 0.8$

Note that the higher value of friction coefficient ($u = 0.8$) was used because it produced the highest frictional forces and highest liner stress.

In Table 7.5 [1], the Licensee summarized the computed liner stresses. Comparison with the minimum tensile strength of the liner (type 304 stainless steel) indicated that the minimum factor of safety relative to minimum tensile strength is greater than 3.8.

Review of these analyses indicated that the Licensee's methods and resulting values are acceptable.

3.5.5 Dropped Gate

Section 7.4 of the Licensee's submittal [1] covers the investigation of the consequences of dropping the transfer canal gate on loaded fuel racks. The 4-ft-wide, 7,000-lb gate was assumed to be dropped from a height of 15 inches above the spent fuel racks, with impact taking place along a lineal edge of the gate.

Other assumptions used for the analysis included:

- o Vertical walls of the spent fuel rack cells were modeled as long, ribbed plate columns 169 in high with a 0.063-in wall. Ribs were assumed to be 3 in wide and 0.063 in thick with a pitch spacing of 6.26 inches.

- o Virtual mass of the gate in water was assumed to be equal to its displaced mass.
- o Form and viscous drag were neglected.
- o The top 1.25 inch of the spent fuel rack walls was assumed to be crashed by the impact of the gate.

The analysis considered the elastic strain energy of the column and the resulting elastic deflection. The remaining energy and the spring constant of the column were used to calculate a pseudo-static force on the column.

Comparison of the pseudo-static force with calculated critical buckling loads indicated that the impact force was below the force that would produce buckling in the spent fuel rack cells.

Review of the Licensee's analysis indicated that the methods, assumptions, and results are acceptable.

In addition, the Licensee indicated that administrative controls would be developed for control of the gate movement across the rack areas.

4. CONCLUSIONS

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

- o The Licensee's mathematical model for structural dynamics of the spent fuel rack modules under seismic loadings simulates three-dimensional dynamics of the rack module, representing a state-of-the-art approach.
- o The seismic dynamic model considers only the case of fluid coupling to adjacent rack modules wherein the motion of each adjacent module normal to the boundary is assumed to be equal and opposite in its displacement to the module being analyzed. Although this assumption neglects the fact that adjacent fuel rack modules may have quite different dynamic response characteristics, it does provide a very manageable reduction in the analytical modeling of the problem while addressing the case in which the available space for dynamic rack displacement is at a minimum.
- o 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 known accuracy only for the condition where the displacements are small compared with the available clearance space. However, the solutions provided appear to become upper bounds where the displacements are not small, and are therefore acceptable.
- o The Licensee took no credit for damping between the fuel assemblies and the rack cell walls, whereas the properties of the limber fuel assembly may permit the use of structural impact damping.
- o The spent fuel pool was considered to have sufficient capacity to sustain the loadings from the high-density fuel racks.

It is concluded that structural analysis of the spent fuel rack modules and spent fuel pool meets the acceptance criteria.

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