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

September 24, 1984

Docket Nos: 50-369
and 50-370

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No.35 to Facility Operating License NPF-9 and Amendment No. 16 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No.35 to Facility Operating License NPF-9 and Amendment No.16 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated February 17, 1984.

The amendments change the Technical Specifications to permit an expansion of the spent fuel pool storage capacity.

A copy of the related safety evaluation supporting Amendment No.35 to Facility Operating License NPF-9 and Amendment No.16 to Facility Operating License NPF-17 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No.35 to NPF-9
2. Amendment No.16 to NPF-17
3. Safety Evaluation

cc w/encl:
See next page

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McGuire -

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated February 17, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

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

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 35, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: September 24, 1984

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated February 17, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

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

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 16, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: September 24, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
5-7	5-8
5-7a	
3/4 9-16	
3/4 9-17	

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 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and 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, which includes a conservative allowance for uncertainties as described in Section 9.1.2.3.1 of the FSAR, and

b. A nominal 21-inch center-to-center distance between fuel assemblies placed in the new fuel storage vault racks, and

c. A nominal 10.4-inch and 9.125-inch center-to-center distance between fuel assemblies placed in Region 1 and Region 2 storage racks, respectively, in the spent fuel storage pool.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft. 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 spaces in Region 1 and 1177 spaces in Region 2) having an initial enrichment less than or equal to 4.0 weight percent U-235.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/hr and 200 cooldown cycles at $\leq$ 100°F/hr (pressurizer cooldown at $\leq$ 200°F/hr for 200°F $\leq$ T pressurizer $\leq$ 650°F).	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 551°F. Cooldown cycle - T_{avg} from $\geq$ 551°F to $\leq$ 200°F.
	80 loss of load cycles.	Without immediate Turbine or Reactor trip.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical power source supplying the Onsite Class 1E Distribution System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	100% to 0% of RATED THERMAL POWER with steam dump.

3/4.9.12 FUEL STORAGE - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.12 Fuel is to be stored in the spent fuel storage pool with:

a. Storage in Region 2 restricted to the following:

1) Fuel which has been qualified in accordance with Figure 3.9-2 or unqualified fuel stored in a checkerboard configuration. In the event checkerboard storage is used, one row between normal storage locations and checkerboard storage locations will be vacant; and

2) Irradiated fuel which has decayed at least 16 days; and

b. The boron concentration in the spent fuel pool maintained at greater than or equal to 2000 ppm.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in the incorrect Region until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

b. Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 2000 ppm, until such time as the boron concentration is increased to 2000 ppm or greater.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12a. Verify all fuel assemblies to be placed in Region 2 of the spent fuel pool are within the enrichment and burnup limits of Figure 3.9-2 by checking the assemblies' design and burnup documentation.

b. Verify at least once per 31 days that the spent fuel pool boron concentration is greater than 2000 ppm.

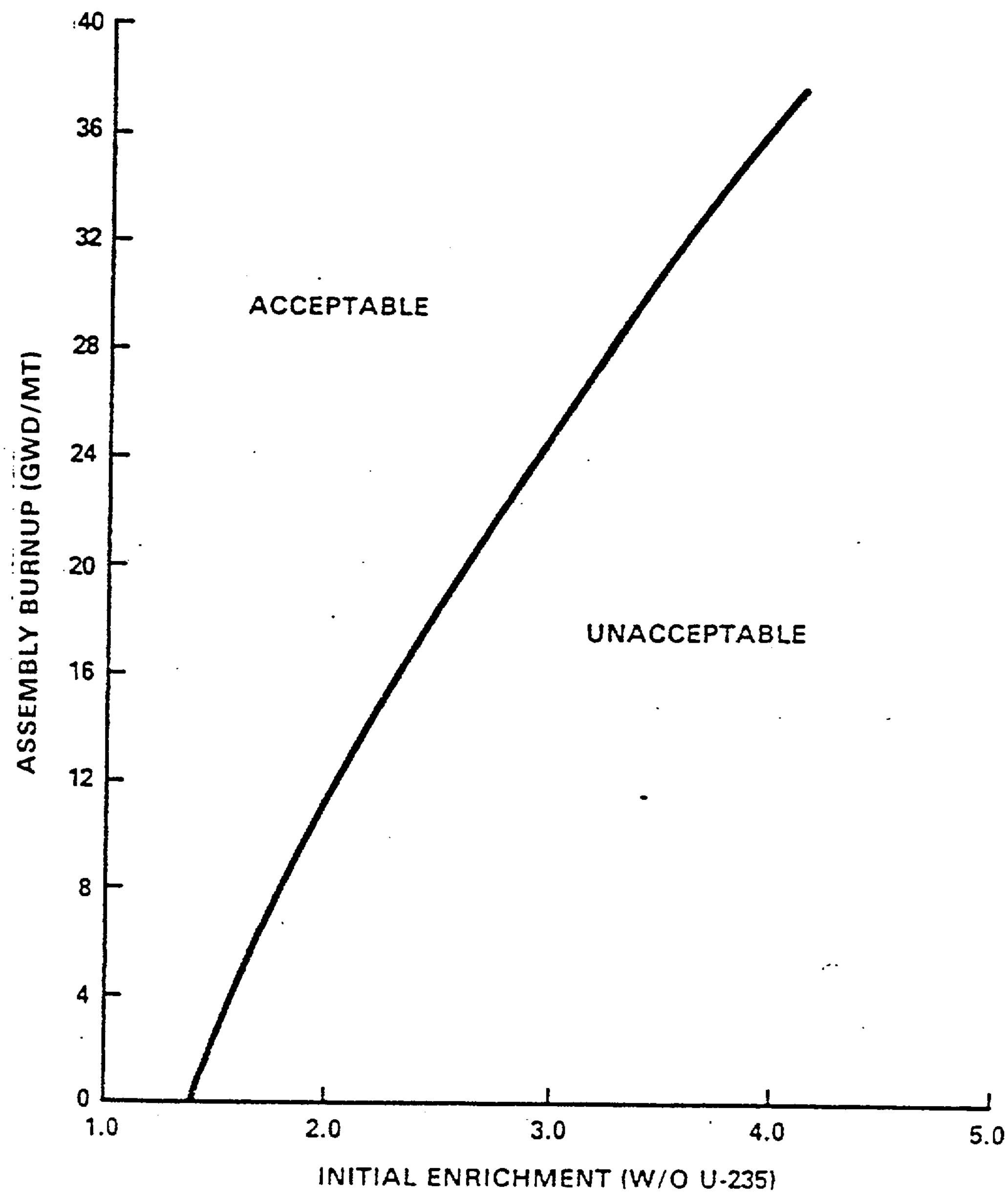

Figure 3.9-2
Minimum Burnup vs. Initial Enrichment
for Region 2 Storage

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

SAFETY EVALUATION REPORT

RELATED TO AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NPF-9

AND TO AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated February 17, 1984, Duke Power Company (the licensee) made application to amend the operating licenses for the McGuire Nuclear Station, Units 1 and 2, to allow an expansion of the spent fuel pool storage capacity at each unit from 500 to 1463 storage spaces. The proposed expansion is to be achieved by reracking each spent fuel pool with two-region poisoned racks. This expansion will provide storage until 2010 with space for offloading a full core assuming reloads of a third of a core. The licensee identified in the FSAR and in the spent fuel pool storage capacity submittal that spent fuel from the Oconee reactor may be stored in the McGuire spent fuel pool. The storage of Oconee fuel at McGuire was evaluated and reported in Supplement No. 2 of the Safety Evaluation Report (NUREG-0422). Our evaluation of the licensee's capacity expansion submittal does not change the conclusion in the SER Supplement No. 2 related to the storing of Oconee spent fuel at McGuire.

EVALUATION

In order to support this amendment application, the licensee by letter dated March 20, 1984, submitted a "Spent Fuel Pools Rerack Modification Safety and Environmental Analysis" which served as a keystone for the staff evaluation. Supplemental information was provided by the licensee as reflected in the following evaluation summary.

1.0 Criticality Aspects

Description of Racks

Each pool will contain racks that provide 1463 designated locations for the storage of reactor fuel. The storage racks will be divided between two regions - one containing 286 locations and one containing 1177. The smaller region, having sufficient capacity for approximately 1½ full cores, will be used for the storage of fresh fuel and fuel not suitable for Region 2. The larger region will normally be restricted to fuel having a specified minimum burnup. The licensee proposes that, if at some future date Region 1 becomes filled, storage of high reactivity fuel (up to fresh 4.0 percent enrichment) be permitted in Region 2 in a checkerboard array with every other location empty. Physical barriers will be used to prevent storage in the empty locations.

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The Region 1 racks will consist of stainless steel cans of 8.75 inches square interior dimension and 0.075 inch wall thickness. On the outer surface of each side of the cans Boraflex sheets having a minimum areal density of 0.02 gram per square centimeter of B-10 are held in place by a thin-walled stainless steel wrapper plate. The rack structure maintains these cans on a 10.40 inches center-to-center spacing.

The Region 2 rack design consists of stainless steel cans welded together to form a honeycomb type structure. The cans have an interior square dimension of 8.93 inches and are made of stainless steel. All four sides of interior cans have Boraflex sheets containing 0.006 gram of B-10 per square centimeter of surface area that are held in place by a stainless steel wrapper which is spot welded to the can. The resulting structure maintains the stored fuel assemblies at a center-to-center spacing of 9.125 inches.

On the outer boundary of each rack the Boraflex sheet is omitted. Neighboring racks are maintained at sufficient separation from each other to preclude an increase in pool reactivity from this cause.

Calculation Methods

The calculation of the effective multiplication factor, k-eff, for Region 1 makes use of the AMPX system of codes for cross-section preparation and the Monte-Carlo Code KENO-IV for reactivity. This code set has been verified against a set of 27 critical experiments that simulate various features of the rack design. A calculational method bias of zero and uncertainty of 0.013 k-eff (95/95) was inferred from these comparisons.

The calculation of the criterion for acceptable burnup for storage in Region 2 makes use of the concept of reactivity equivalence. Since the KENO-IV code cannot handle burned fuel assemblies, it is necessary to obtain the fresh fuel assembly enrichment which yields the same pool k-eff as the burned assembly. Because of the presence of the poison in the Region 2 racks a multigroup transport theory code is more appropriate than diffusion theory for this calculation. The PHOENIX code was used.

The calculation proceeds as follows:

1. An end-point of 36.5 GWD/MT burnup for a bundle having an initial enrichment of 4.0 weight percent U-235 is chosen

2. PHOENIX is used to calculate the k_∞ of such an assembly in the rack geometry (including can and Boraflex absorber)

3. The burnup required to produce the same k_∞ is calculated for a number of smaller enrichments

4. The enrichment required to produce the same k without burnup is obtained (in the present case the value is 1.4 percent)

5. KENO-IV is used to calculate the rack multiplication factor for the 1.4 percent enrichment assembly.

The advantage of this procedure is that only relative multiplication factors are computed by PHOENIX. The final value of the rack multiplication factor is obtained from the more powerful KENO-IV code.

Treatment of Uncertainties

For the Region 1 analysis the total uncertainty is the statistical combination of the method uncertainty, the uncertainty in the particular KENO calculation, and mechanical uncertainties due to tolerances, spacing, etc. The mechanical uncertainties were treated either by making worst case assumptions (e.g., using the minimum rather than nominal value of the boron loading) or by performing sensitivity studies and obtaining a value for the uncertainty in rack multiplication factor due to uncertainty in dimensions.

In the Region 2 analysis the same uncertainties are considered along with others that are unique to the rack design and usage. These include uncertainty due to particle self-shielding in the boron (actually bias), uncertainty in the plutonium reactivity and uncertainty in the reactivity as a function of burnup. Including both the plutonium and burnup reactivity uncertainties is conservative since the latter includes the former as one of its components. The particle self-shielding bias is important for Region 2 because of the low boron loading relative to Region 1 (0.006 vs. 0.02 gm/cm^2 of B-10).

The PHOENIX code was qualified for burnup calculations by comparing calculated isotopic ratios to measurements made in Yankee-Rowe Core 5, and by comparison of equivalent reactivity burnup between PHOENIX and the LEOPARD/TURTLE code. A set of 81 critical experiments was analyzed to qualify the code for zero burnup conditions. Conservative uncertainties of 5 percent of the reactivity worth of the actinides and 5 percent of the reactivity change due to burnup have been assigned to these parameters.

Results of Analysis

Normal Storage

For Region 1, the rack multiplication factor is calculated to be 0.944, including uncertainties at the 95/95 level, when fuel having an enrichment of 4.0 weight percent U-235 is stored therein. Fuel of either the

Westinghouse standard or OFA design may be stored. Pure water at 1.0 gram per cubic centimeters is assumed.

For Region 2, the rack multiplication factor is 0.940 for the most reactive irradiated fuel permitted to be stored in the racks, i.e., fuel with the minimum burnup permitted for each initial enrichment. For fresh fuel (4.0 percent enrichment) stored in a checkerboard array in the racks, the effective multiplication factor is 0.866. These multiplication factors include all uncertainties and are obtained for pure water at a density of one gram per cubic centimeter. Burned fuel of the Westinghouse standard or OFA design or of the Babcock and Wilcox 15x15 design may be stored in Region 2. Analyses were performed for all three fuel types and the proposed curve of burnup vs. initial enrichment bounds the results of the calculation.

Abnormal Storage Conditions

Most abnormal storage conditions will not result in an increase in k-eff of the racks. For example, loss of a cooling system will result in an increase in pool temperature but this causes a decrease in the k-eff value. It is possible to postulate events (e.g., a seismic event) which could lead to an increase in pool reactivity. However, for such events, credit may be taken for the approximately 2000 ppm of boron in the pool water. The reduction in the k-eff value caused by the boron (approximately 0.25) more than offsets the reactivity addition caused by credible accidents.

Summary

The following discussion summarizes our evaluation of the proposed re-racking of the McGuire spent fuel storage pools related to criticality aspects.

We have reviewed the assumptions made in the performance of the criticality analyses. These include use of the highest permitted reactivity bundle, pure water moderator at a density of 1.0 gram per cubic centimeter, and an infinite array of assemblies. These are consistent with NRC guidelines and are acceptable.

We have reviewed the uncertainties which have been included. For Region 1 these include variation in poison pocket thickness, stainless steel thickness, cell interior dimensions, center-to-center spacing and cell bowing. Other parameters, such as boron loading, are taken at their most conservative limits. For Region 2 additional uncertainties due to burnup calculations and calculations of plutonium worth are included. For both regions calculational uncertainties and biases are included. These uncertainties meet our requirements and are acceptable.

We have reviewed the verification of the calculational methods. The KENO-IV code is widely used in the industry for the purpose of calculating fuel rack criticality. The set of benchmark critical experiments used to verify the calculational method encompasses the enrichment, separation distance and separating material used in the racks. The set of experiments used to verify the PHOENIX code for the reactivity equivalence calculations is adequate and encompasses the pellet size and enrichment of the fuel proposed for storage in the McGuire racks. The uncertainties in the burnup and plutonium worth are verified against Yankee Core 5 isotopics and comparisons with the Westinghouse design LEOPARD/TURTLE code package. We find that adequate verification of the codes used in the criticality analyses has been performed.

The technique of using reactivity equivalencing to define the storage criterion (burnup as a function of initial enrichment) is, in some form, in widespread use in the industry and is acceptable.

For Region 1 racks we have compared the results of the McGuire calculation to a generic study and found them to be compatible. Finally the results of the calculation for Region 1 and 2 meet our acceptance criterion of less than or equal to 0.95 including all uncertainties at the 95/95 level.

We have reviewed the proposed Technical Specification 3/4.9.12 and find that it is consistent with the assumptions in the safety analysis and is acceptable.

Conclusions

Based on our review, which is described above, we find the criticality aspects of the design of the spent fuel racks to be acceptable. We conclude that fresh Westinghouse 17x17 fuel of either the standard or OFA design may be safely stored in Region 1 so long as its enrichment does not exceed 4.0 w/o U-235. We further conclude that either type of Westinghouse fuel or fuel of the 15x15 Babcock and Wilcox design may be stored in Region 2 provided it falls in the acceptable region of Figure 2.4-3 of the Safety and Environmental Analysis report. Fuel which does not meet this criterion may be stored in Region 2 provided it is stored in a checkerboard arrangement with every other location vacant. The licensee has committed to provide physical barriers to prevent storage in the empty locations.

2.0 Systems Aspects

Decay Heat Loads

The licensee's calculated spent fuel discharge heat load to the pool, which was determined in accordance with the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term

Cooling," and the Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," indicates that the expected maximum normal heat load following the last refueling is 18.0 MBTU/Hr. This heat load results in a maximum bulk pool temperature of 133°F. The expected maximum abnormal heat load following a full core discharge is 41.6 MBTU/Hr. This abnormal heat load results in a maximum bulk pool temperature of 178°F, or a maximum bulk pool temperature of 120°F with both cooling trains operating. Assuming the loss of all cooling, boiling would occur after 13.8 hours for the normal heat load condition and after 4.7 hours for the maximum heat load condition. This provides reasonable time to initiate makeup to the spent fuel pool.

The spent fuel pool water is cooled by the component cooling water system, which in turn is cooled by the service water system. The licensee proposed no modifications to these two systems as part of this spent fuel pool expansion project. Although the spent fuel pool heat exchanger only has a design capacity of 15.0 MBTU/Hr, an independent pool water temperature calculation was performed which verified that the pool water temperature will remain within acceptable limits. Thus our review of these systems as to their adequacy to remove the additional heat load indicates that they are capable of removing the additional heat.

Control of Heavy Loads

Presently, there is no spent fuel in the McGuire Unit 2 spent fuel pool. There is spent fuel in the McGuire Unit 1 spent fuel pool and the licensee has stated that no spent fuel racks will be carried over spent fuel. However, the empty spent fuel storage racks will be carried over an empty portion of two racks which will contain spent fuel. The utility has performed a load drop analysis which indicated that there would be no damage to the stored fuel and therefore no radiological consequences. The postulated rack drop would not change the separation distance between the stored fuel assemblies or the concentration of boron. Therefore, the margin of safety to criticality will not be affected by a rack drop accident. The racks will enter and exit the fuel building by means of the outdoor cask handling crane. The racks will be maneuvered inside the fuel building by means of the fuel building crane. There are no safety related components on or above the fuel handling floor. Therefore, a drop of a spent fuel rack will not have any adverse consequences as identified in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is, therefore, acceptable.

Summary and Conclusions

Based on the above, we conclude that the proposed overall spent fuel pool storage capacity modification program is acceptable for storage of 1463 spent fuel assemblies per reactor unit with respect to the storage rack capacity, the developed heat loads and pool water temperatures, the load handling, and the spent fuel pool cooling and support system capabilities.

3.0 Material Considerations

Description

The safety function of the spent fuel pool and storage rack system is to maintain the spent fuel assemblies in a subcritical 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 stainless steel, except for the nuclear poison material. The existing spent fuel pool liner is constructed of stainless steel. The high density spent fuel storage racks will utilize Boraflex[1] 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 major components of the assembly are the fuel assembly cells, the Boraflex material, the wrapper and the upper and lower spacer plates.

The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. The wrapper surrounds the Boraflex material, but is open at the top and bottom to provide for venting of any gases that are generated. The Boraflex sheets sit in a square annular cavity formed by the square inner stainless steel tube and the outer wrapper. Each sheet is supported by lower spacer plate.

Corrosion & Materials Compatibility

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} 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 Inconel 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 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. Tests were performed at the University of Michigan,[2] exposing Boraflex to 1.103×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex. Long term borated water soak tests at high temperatures were also conducted.[3] The tests show that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space which 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 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 in the Boraflex will typically contain 0.1 wt percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble specie 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 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 (approximately 5 year intervals).

Summary and Conclusion

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 borated 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 actual service experience in operating reactors.

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, 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 spelled out 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. We do not anticipate that such deterioration will occur. 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 a monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 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 poison and is therefore acceptable.

References

1. J. S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Date," 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).

3. J. S. Anderson, "A Final Report on the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Brand Industries, Inc., Report 748-21-1, (August 1978).

4. Structural Considerations

General

The structural review of the proposed spent fuel storage pool expansion was performed by our consultant, the Franklin Research Center (FRC). The results of their review are described in the FRC Technical Evaluation Report TER-C5506-526, revised August 10, 1984, which is incorporated by reference in this safety evaluation.

The spent fuel pools are constructed of reinforced concrete lined with stainless steel plates. The Unit 1 pool is a flat, reinforced concrete slab 4.5 feet thick, supported on reinforced concrete beams, 6.5 feet deep by 4.5 feet wide, and by 4.0 feet thick walls at the perimeter. The beams are spaced on 20.0 foot centers, spanning across the pool in the short direction. Floor loads are transmitted to the structural foundation from the floor slab, to the deep beams, to the perimeter walls, then to the bedrock foundation. The Unit 2 pool floor is supported continuously on a bedrock foundation. The floor slab is 11.0 feet thick. The additional thickness as compared to Unit 1 is due to construction considerations. All dead, live and seismic loads are transmitted directly through the floor slab then to the bedrock foundation.

Fuel storage is divided into two regions within each pool. The Region 1 storage racks are composed of individual storage cells made of stainless steel. The cells within a module are interconnected by grid assemblies to form an integral structure. Each rack module is provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above throughout the cells at installation. The modules are free-standing and are not anchored to the floor nor braced to the pool walls. The fuel rack assembly consists of three major sections which are the leveling pad assembly, the lower and upper grid assemblies, and the cell assembly.

The Region 2 storage racks consist of stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure. The cells are welded to a base support assembly and to one another to form an integral structure without use of grids as used in Region 1 racks. This design is also provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are free standing and are not anchored to the floor nor braced to the pool walls. The fuel rack module consists of two major sections which are the base support assembly and the cell assembly.

Codes, Standards and Design Loads

Load combinations and acceptance criteria were compared with those found in the "NRC 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 McGuire FSAR Section 3.8.4. 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 NRC Position.

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.

Design and Analysis Procedures by Licensee

a. Design and Analysis of the Racks

The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. The second phase is a response spectrum analysis of a detailed linear three dimensional rack assembly finite element model. Two percent damping is used in the seismic analysis for both the OBE and SSE. Further details on the methodology is discussed in Franklin Research Technical Evaluation Report TER-C5506-526, revised August 10, 1984.

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 bundle on the racks and results were considered satisfactory.

An analysis was also 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 acceptable.

b. Analysis of the Pool Structure

The slab, beams and walls are reinforced to meet all FSAR criteria. The existing structures were analyzed for the modified fuel rack loads using the STRUDL 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 McGuire FSAR Section 3.8.4 and the existing spent fuel pools are determined to safely support the loads generated by the new fuel racks.

Conclusion

It is concluded that the proposed rack installation will satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 2, 4, 61 and 62 as applicable to structures.

5.0 Radiation Protection

Occupational Radiation Exposure

The occupational exposure estimated by the licensee for the rerack modification is 15 person-Rem. This estimate is based on a detailed breakdown of occupational exposure for each phase of the rerack modification, and considers the number of individuals performing a specific job, the average dose rates in the area where the job is being performed, and the time spent by workers performing the job in these areas. This estimate represents a small fraction of the annual dose estimated for the station - less than 1.5%. The rerack modification for Unit 2 will involve no occupational exposure, since the operation will be carried out in a dry, radiologically "clean" pool.

During the modification process and during projected operations, the fuel assemblies themselves (including the additional assemblies) will contribute a negligible amount to pool area dose rates due to the depth of water shielding the fuel.

Radioactive activation and corrosion products (crud) may be released to the pool water from fuel surfaces during fuel movements during the modification. This could increase radiation levels in the vicinity of the pool, however, the Spent Fuel Pool Cooling System, in conjunction with water vacuuming of the pool floor, walls and fuel rack surfaces, will filter and purify the pool water. This will remove the crud and minimize the dose contribution from crud in the pool water to workers and divers in the pool area.

The licensee has considered burial, decontamination, and long-term on-site storage as means to dispose of old racks from the Unit 1 Spent Fuel Pool. Following removal from the pool, the racks will be rinsed by low pressure spray or hydrolased to reduce contamination levels and subsequent handling doses. Protective clothing and respiratory protection will be utilized as needed to keep exposures to contamination and airborne radioactivity ALARA. The racks will be decontaminated, if possible, and sold as scrap or shipped to a burial site if decon is not practical. Unit 2 racks will be radiologically clean and will be sold as scrap.

Doses to divers will be minimized by rearrangement of stored assemblies to give the lowest practical dose rates. Additionally, diver paths will be marked, and health physics personnel will monitor their work. Radiation protection controls for divers include protective clothing, multiple - TLD's and self-reading dosimeters, underwater surveys, management dose tracking, and direct communications with divers. Divers will be warned if they approach high radiation/exclusion zones, and will be kept at least 10 feet from spent fuel.

Periodic radiation and contamination surveys will be conducted in work areas, and grab sampling and/or continuous sampling performed where there is a potential for airborne radioactivity.

Work will be controlled by Radiation Work Permit, posting, use of stay time, zoning, access control, and health physics personnel to assure that doses are kept ALARA.

Additionally, we have estimated the increment in onsite occupational dose during normal operations after the pool modification as a result of the proposed increase in stored fuel assemblies. This estimate is based on information supplied by the licensee for occupancy times and for dose rates in the spent fuel area from radionuclide 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 spent fuel pool area, we estimate that the proposed modification should add less than one percent of the total annual occupational radiation exposure at both units. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable levels and within the limits of 10 CFR Part 20.

Conclusion

Based on the manner in which the licensee will perform the modification; our previous evaluation of their radiation protection/ALARA program during the licensing process; the radiation protection measures proposed for the modification task, including radiation, contamination, and airborne radioactivity monitoring; and relevant experience from other operating reactors that have performed similar spent fuel pool modifications, the staff concludes that adequate radiation protection measures have been taken to assure worker protection, and the McGuire Spent fuel pool modification can be performed in a manner that will ensure that doses to workers and the general public will be ALARA and that storing additional fuel in the two pools will not result in any significant increase in doses received by workers.

SAFETY CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (49 FR 27225) on July 2, 1984, and consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

In conclusion the staff finds the proposed changes to the plant technical specifications to be acceptable and 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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... : September 24, 1984

September 24, 1984

AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION
AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-17 - McGUIRE NUCLEAR STATION

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— BIRKEL —

TECHNICAL EVALUATION REPORT

EVALUATION OF SPENT FUEL RACKS STRUCTURAL ANALYSIS

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION UNITS 1 AND 2

NRC DOCKET NO. 50-369, 50-370

NRC TAC NO. 53531, 53532

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 26

FRC TASK 527

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July 17, 1984
Revised August 10, 1984

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Revised August 10, 1984

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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.

The following staff of the Franklin Research Center contributed to the technical preparation of this report: Vu N. Con, Maurice Darwish, R. Clyde Herrick, Vincent K. Luk, Balar S. Dhillon (consultant), and T. B. Belytschko (consultant).

1. INTRODUCTION

1.1 PURPOSE OF THE REVIEW

This technical evaluation report (TER) covers an independent review of the Duke Power Company's licensing report [1] on high-density spent fuel racks for the McGuire Nuclear Station Units 1 and 2 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 the McGuire Units 1 and 2 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, Subsection NF, Component Supports
  Subsection NB, Typical Design Rules

- 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

  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 spent fuel racks' structural analysis for McGuire Units 1 and 2 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, Latest Edition.

** American Institute of Steel Construction, Latest Edition.

- 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.

- 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.

- 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

The submerged spent fuel rack modules exhibit highly nonlinear structural 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.

All modules at McGuire Nuclear Station have nearly square cross sections across the axes of fuel cells [1]. Modules of this design geometry generally behave in three-dimensional fashion under earthquake loadings. Hence, the modules will exhibit three-dimensional nonlinear structural behavior in seismic events, and all seismic analyses of modules should therefore focus on characterizing this behavior.

There are two types of modules at McGuire Units 1 and 2 [1]. The modules in Region 1 have a center-to-center storage cell spacing of 10.4 in. They are reserved for temporary core off-loading, temporary storage of new fuel, and storage of spent fuel above specified levels of reactivity. The modules in Region 2, with 9.125-in center-to-center spacing, are used to store irradiated fuel below specific reactivity levels. The designs of modules in Regions 1 and 2 are shown in Figures 1 and 2, respectively.

The Licensee conducted the seismic analysis of modules in two parts. The first part was a time history analysis of a simplified two-dimensional nonlinear finite element model of an individual fuel cell shown in Figure 3.

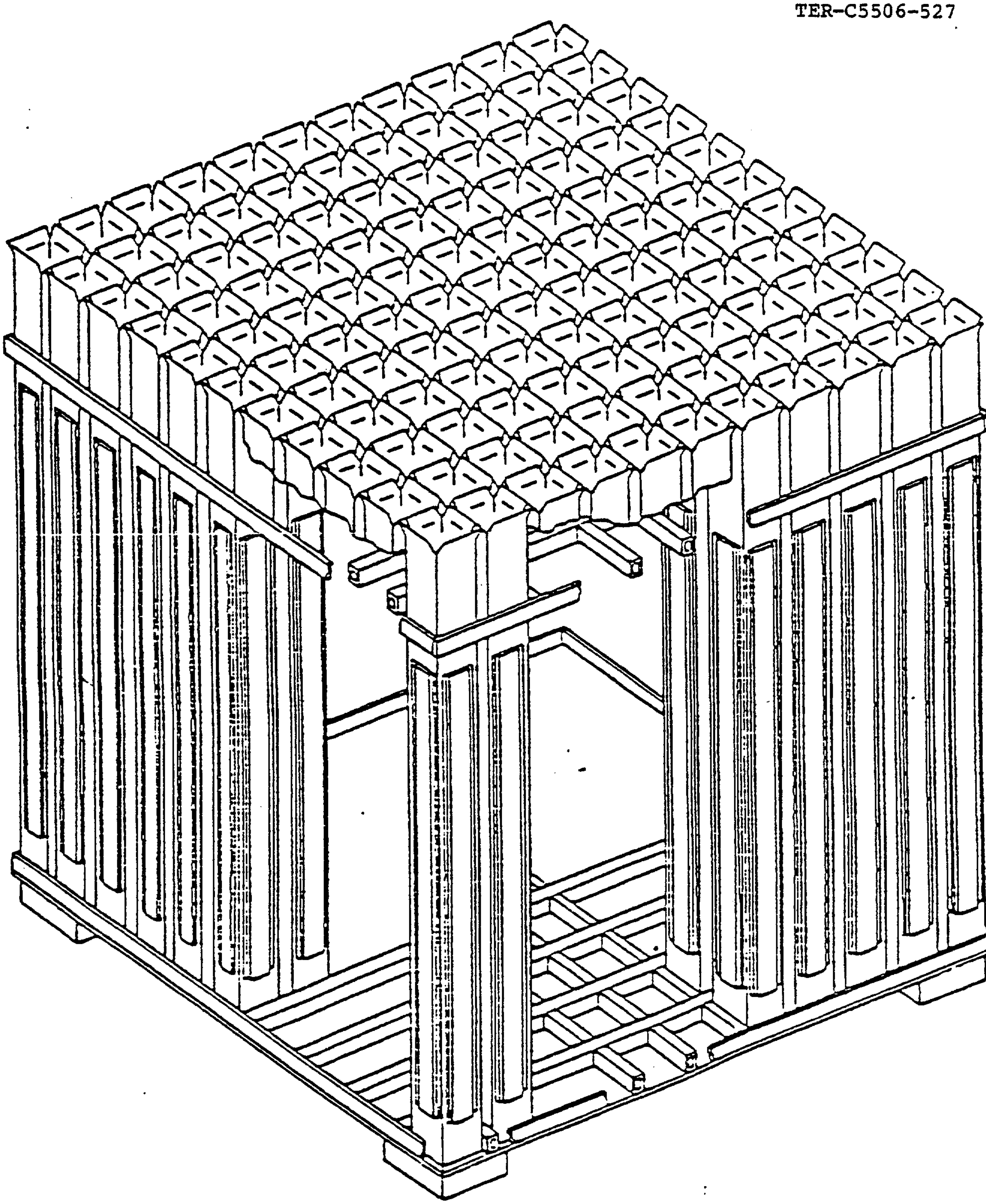


Figure 1. Fuel Storage Rack Assembly in Region 1

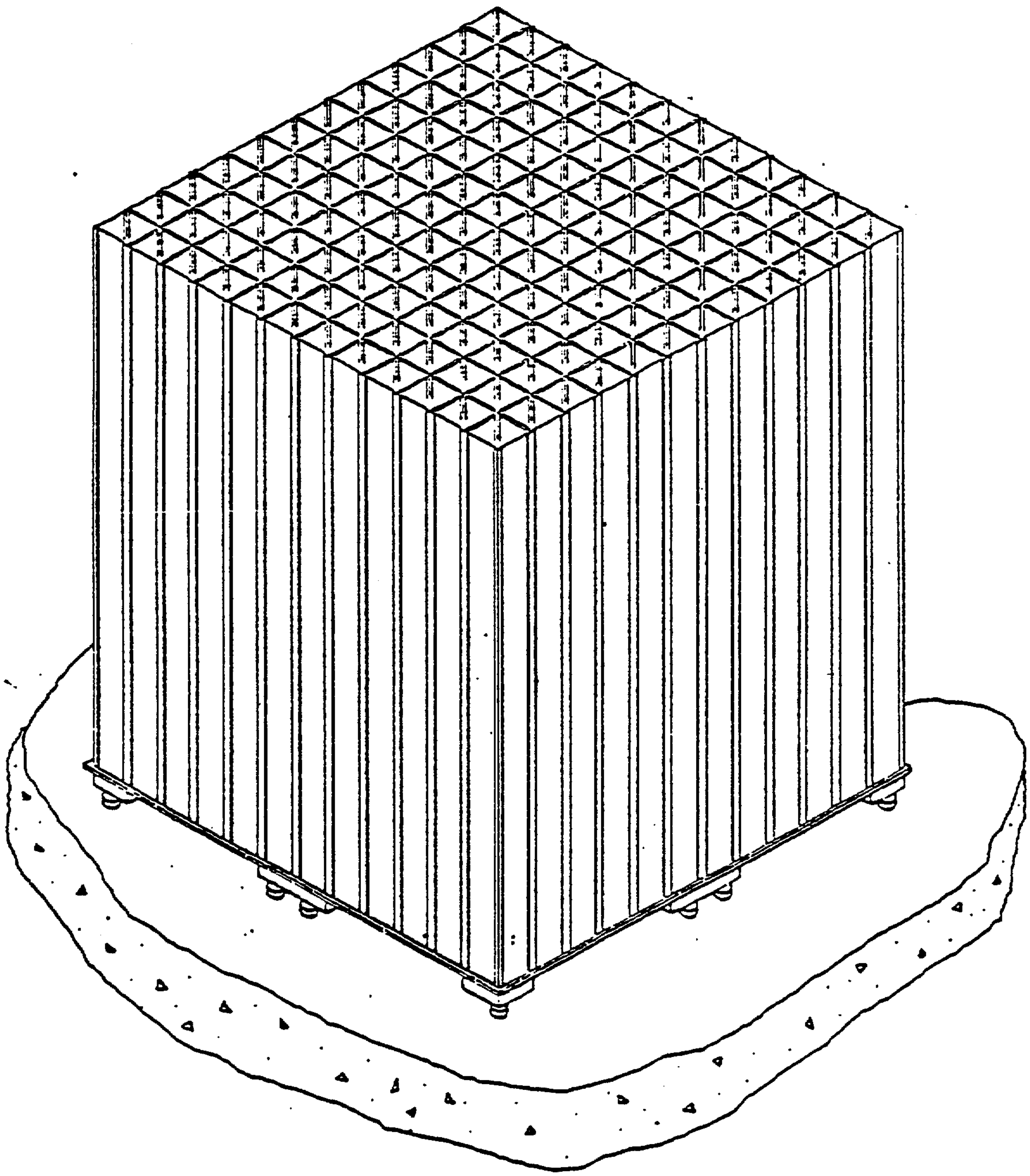


Figure 2. Fuel Storage Rack Assembly in Region 2

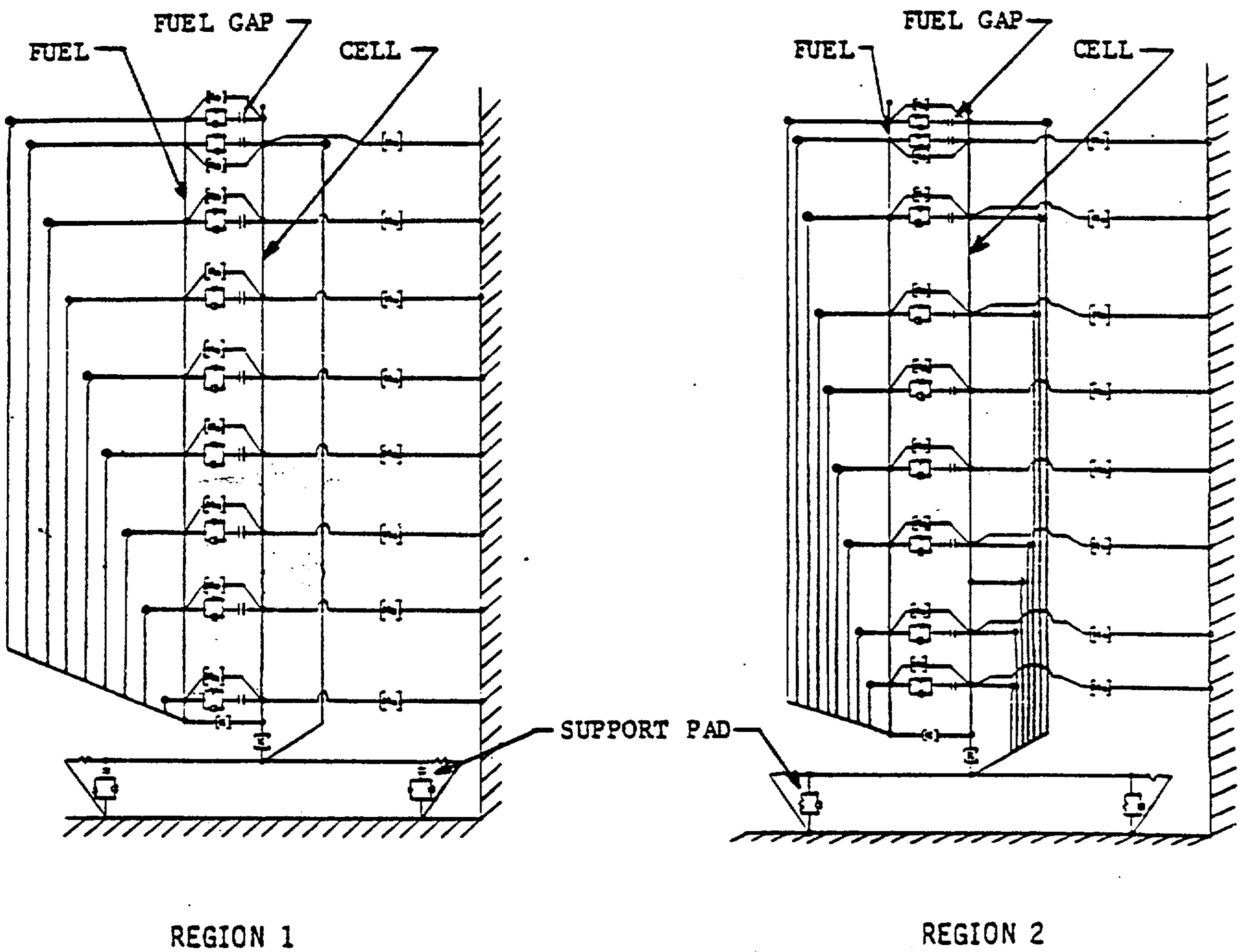


Figure 3. Two-Dimensional Nonlinear Model

The second part was a response spectrum analysis of a detailed three-dimensional linear finite element model of a rack assembly shown in Figure 4. Both modules consisted of two models to reflect the two different designs of modules in Regions 1 and 2. Structural damping of 2% was used in the seismic analysis for both the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE).

With regard to the models used in the analysis, the following issue was discussed at a meeting with the Licensee [3]:

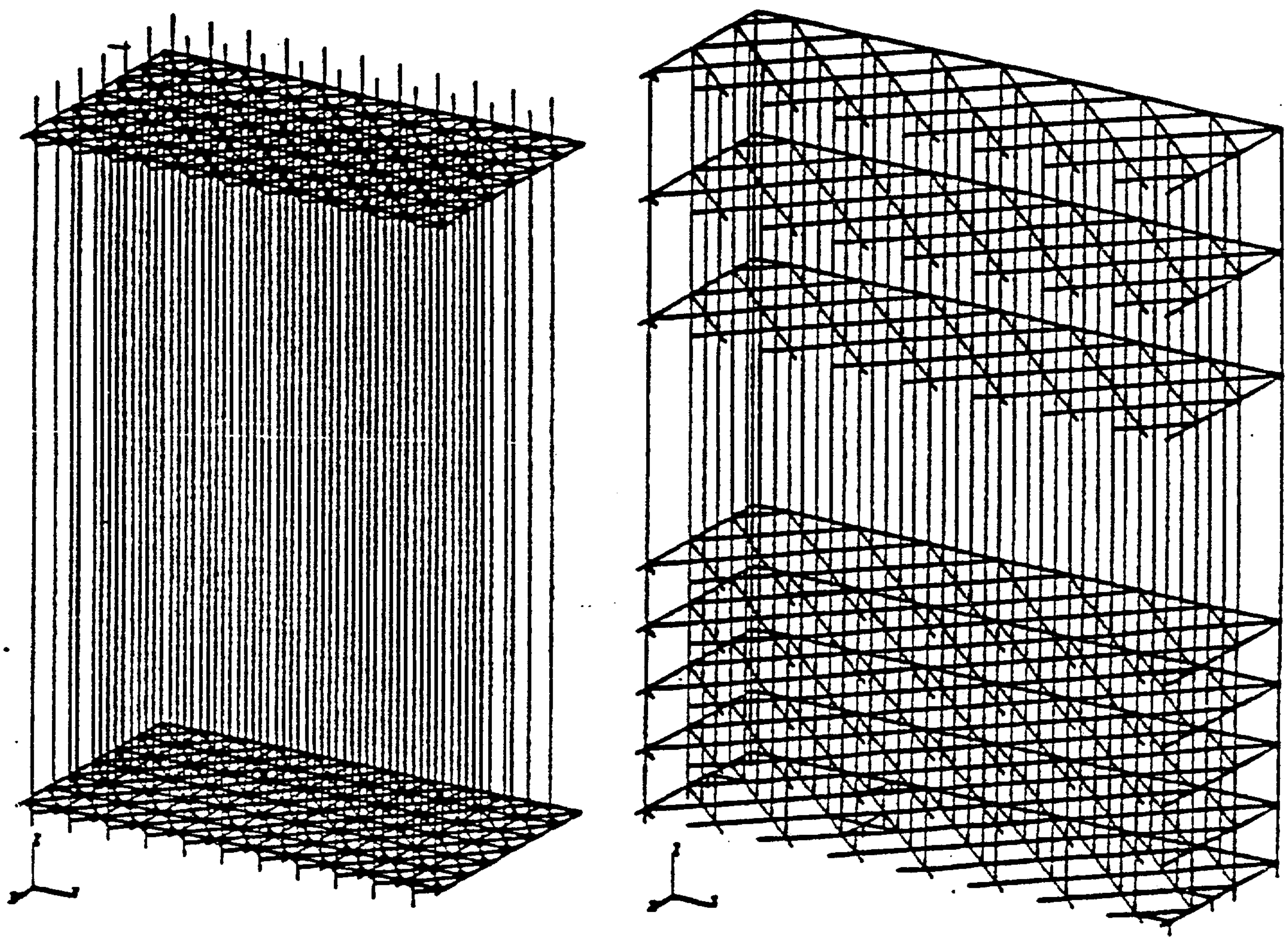
> The simplified two-dimensional model does not fully simulate the more complicated three-dimensional structure behavior exhibited by the modules. The two-dimensional model essentially uncouples the two mutually perpendicular horizontal motions which are nonlinearly interrelated under seismic loadings. Thus, an approach using two models (nonlinear, two-dimensional and linear, three-dimensional model) may have difficulty in resolving peak stresses.

The value of impact damping (15%) used in the analysis was questioned when documentation of the damping values provided by the Licensee confirmed a range of only 10% to 15% [4]. However, the Licensee has submitted the following response which cites test data performed by Babcock & Wilcox Company (fuel suppliers) which is stated to be on file with the USNRC [5]:

> "In determining the fuel assembly impact damping, B & W performed a series of tests. The upper and lower bounds for the tests are reported as .1462 and .1650 respectively with a median value for all tests of .1565. B & W Topical Report 10133P Rev. 1, filed with the NRC on 5/3/79, gives a fuel assembly impact damping value of 16% for a Mark C assembly. The report also notes that B & W Mark C charactristics are similar to the B & W Mark B assembly characteristics which are stored at McGuire, thus, the results are directly comparable. The Applicant maintains that use of a damping value of 15% is appropriate and the conservatisms of the analytical results used in the design of the proposed racks are preserved."

Since damping influences the amplitude of dynamic response, it is important to use values that do not overestimate the energy loss of the system.

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.



REGION 1 REGION 2

Figure 4. Three-Dimensional Linear Model

3.2 EVALUATION OF THE SIMPLIFIED TWO-DIMENSIONAL NONLINEAR MODEL

3.2.1 Description of the Model

The simplified two-dimensional model was developed to simulate the major structural characteristics of an individual fuel cell within a submerged rack assembly. Two versions of this model are shown in Figure 3 to reflect two different module designs in Regions 1 and 2. The model was developed in accordance with the WECAN (Westinghouse Electric Computer Analysis) code.

A time history analysis of the model was performed by the Licensee with the simultaneous application of a vertical and a horizontal component of seismic loads. Nonlinear gap elements were used in the model to represent the possible impact between the fuel cell and the fuel assembly, as well as the friction between the module base and the pool liner. The hydrodynamic coupling effect between fuel cell and fuel assembly, as well as between fuel cell and rigid wall, is simulated by appropriate coupling springs. A damping value of 15% was used to represent the impact damping of the fuel assembly manufactured by Babcock & Wilcox (B&W) Company. Justification of 15% impact damping was discussed in Section 3.1 of this report.

3.2.2 Assumptions Used in the Analysis

The following assumptions were used in the seismic analysis of the model:

a. A structural damping value of 2% was used for both OBE and SSE events.

b. The fluid damping was conservatively neglected.

c. Only a constant value of friction coefficient was considered in each seismic analysis. The coefficient of friction remained unchanged whether the module was stationary or in motion. Analysis was performed for static friction coefficients of u = 0.2 and 0.8. These two cases would envelop the values of intermediate friction coefficients.

d. The initial status of the gap between fuel cells and fuel assembly is immaterial because all fuel cells would move in phase soon after an earthquake occurred. Adjacent modules would also move in phase in seismic events.

e. The modules stand on the pool liner occupying the bottom one-third of the water body in the fuel pool. Therefore, the sloshing effect is negligible.

The assumption in Item d may be valid when adjacent modules are fully loaded, but the out-of-phase response will most likely occur when some modules are either partially loaded or empty.

3.2.3 Hydrodynamic Coupling Between Fluid and Cell Structure

The hydrodynamic coupling effect between adjacent modules and between the fuel cell and fuel assembly plays a significant role in affecting the dynamic responses of the module in seismic events. As stated in Section 3.2.2, the modules were assumed to move in phase. 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 as an equivalent representation of the entire rack assembly. 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 [6].

The technique of potential flow and kinetic energy was used in assessing the hydrodynamic coupling mass between the fuel cell and the fuel assembly. This mass, which depends on the size of fuel assembly and the inside dimensions of the fuel cell, was calculated by equating the kinetic energy of the hydrodynamic coupling mass to that of the fluid flowing around the fuel assembly within the fuel cell. The concept of this method was discussed in a paper by De Santo [7].

Fritz's [6] 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 [6] 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 only 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 [6] modeling technique for hydrodynamic coupling of rack modules adjacent to other rack modules or a pool wall reinforce the position of this review that the Licensee's fuel rack dynamic model be considered conservative only for dynamic displacements that are small relative to the available displacement clearance.

3.2.4 Seismic Loading

The model was subject to a simultaneous application of a vertical and a horizontal component of seismic loads. The hydrodynamic coupling mass in the same horizontal direction is also incorporated in the analysis. In a meeting at Westinghouse in Pensacola, Florida, the Licensee stated that there were two distinct horizontal seismic response spectra as well as two different sets of hydrodynamic coupling masses in these two horizontal directions [3]. However, only one time history corresponding to one of the two horizontal response spectra was used in the analysis. Subsequent to the meeting, the Licensee provided the following [5]:

> "Of the two horizontal seismic response spectra, the E-W spectrum has larger acceleration values than the N-S spectrum in the frequency range of the fuel rack (4-8 H_z). Thus, the seismic analysis was conservatively performed with the E-W response spectrum, the E-W hydrodynamic mass (maximum hydrodynamic mass), and the minimum support and spacing (N-S in region 2 and E-W in region 1), to obtain the maximum fuel rack response."

This statement documents the use of response spectra providing conservative analysis.

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. It was found that the convergence of solution occurred at a time step of 0.00125 sec [4]. This time step is much greater than the 2.0×10^{-4} sec reported by Gilmore of Westinghouse in a similar analysis [8]. The Licensee explained that the wide range of time step for convergence might be responsible for these differing values.

3.2.6 Displacement and Stress Results

The Licensee claimed that the displacement of the module would be the same as that of the individual cell found in this model because of the in-phase motion assumption used in this analysis. The Licensee determined that the module slides a maximum distance of 0.10 in at $u = 0.2$ [1]. While this result may not be conservative because the two-dimensional model used in this analysis uncouples the two horizontal responses under seismic loadings, it does indicate that the displacements are relatively small.

The moments and shear forces generated from this model were used to calculate the load correction factors. The load results from the detailed model were then multiplied by these factors to yield the stress results in the structural analysis of the module, as discussed in Section 3.3 of this report. A detailed review of this method is given in Section 3.4 of this report.

3.3 EVALUATION OF THE DETAILED THREE-DIMENSIONAL LINEAR MODEL

3.3.1 Description of the Model

A model was developed to simulate the major structural characteristics of the entire module submerged in the fuel pool. Two versions of the model are shown in Figure 4 to represent two different module designs in Regions 1 and 2. The WECAN code was used to develop these two models. Three-dimensional beam elements were used to construct the models.

According to Reference 4, the seismic analysis was done on the 11x13 module in Region 1 and the 12x16 module in Region 2. The model of the module in Region 1 has two fine meshes of elements, one on the top and the other on the bottom of the model to represent the top and the bottom grip assembly of the module, respectively. There are eight horizontal meshes of elements in the model of the module in Region 2 to simulate the eight skip weld locations along the length of cells.

A response spectrum analysis of the three-dimensional models was performed. The three components of the seismic loads were applied to the models, one component at a time.

3.3.2 Assumptions Used in the Analysis

All the assumptions except the initial status of the gap between fuel cell and fuel assembly used in the analysis of the two-dimensional model are applicable here. A few additional assumptions used in this analysis are described below:

a. A composite distributive mass density was used in the analysis to embody the masses of the fuel cell, the fuel assembly, the poison material, and the hydrodynamic coupling mass.

b. No impact between the fuel cell and the fuel assembly was considered.

c. The module base was stationary with respect to the pool liner at all times.

3.3.3 Load Correction Factor

Since the detailed model did not account for the nonlinear effect of a fuel assembly impacting a fuel cell and the support pad movements, the internal loads and stresses for the module assembly obtained from this model were modified by load correction factors. The calculation was focused on the bending moments and shear forces obtained at the base plate of this detailed model. The bending moment load correction factor was defined as the ratio of the bending moment obtained at the base of the simplified model to the average bending moment derived at the base of the detailed model. Similar definition was used for the shear force load correction factor. The maximum loads from this detailed model were multiplied by these load correction factors and were used in the structural analysis to obtain the streses within the module assembly. Further discussion is provided in Section 3.4.

3.3.4 Module Assembly Lift-Off Analysis

Both partially and fully loaded modules were evaluated for module stability. The support pad vertical displacement was used as the parameter for this study. The Licensee found that the maximum lift-off was produced by partial loading of three rows of fuel [4]. This condition yielded a factor of safety against overturn much larger than the 1.5 minimum requirement.

3.3.5 Stress Results

The maximum responses of the detailed model from the seismic components in three directions were combined by the SRSS model in the structural analysis. The maximum loads experienced by the modules were obtained when u = 0.8 [4]. According to Reference 1, the stresses at most locations of the modules had margins of safety higher than 7% with the exception of the weld stresses tabulated in the following:

Description	Margin of Safety
1. Weld shear at leveling pad assembly, Region 1 modules	1%
2. Weld shear at top grid member, Region 1 modules	3%
3. Weld shear at leveling pad assembly, Region 2 modules	3%

The margins of safety for these weld stresses are very small.

3.4 SUMMARY EVALUATION OF THE SEISMIC ANALYSIS OF FUEL RACK MODULES

3.4.1 Background

During the initial review of the seismic dynamic response analysis of the rack modules, concern developed as to the ability of the methods employed to relate two-dimensional nonlinear displacement analysis to the three-dimensional linear mathematical model used to compute peak rack stresses, especially when stress safety margins in the spent fuel racks were reported as low as 1%, 3%, and 7%. Coupled with this concern was a lack of information about the detailed analysis procedures of a unique, proprietary analysis method.

In order to resolve these concerns, a visit was made to the facilities of the Licensee's spent fuel rack vendor to review the analysis procedures in detail and to evaluate the character and magnitude of the intermediate results transferred from one part of the analysis to another. A detailed review was performed. Appropriate discussion and conclusions are provided in the following sections.

3.4.2 Seismic Analysis Method

The Licensee's description of the analysis is as follows [1]:

"The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the structure. The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model shown in Figure 2.3-1. The second phase is a response spectrum analysis of a detailed rack assembly finite element model shown in Figure 2.3-2. Two percent damping is used in the seismic analysis for both the OBE and SSE.

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

The fuel assembly to cell impact loads, support pad lift off, rack sliding, and overall rack response are obtained from the nonlinear time history model. In determining the maximum fuel rack response, the response value for each item of interest is searched for maximum values.

The detailed model is a three-dimensional finite element representative of a rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of nodal points.

The results of the single cell nonlinear time history model are incorporated in the detailed model. Since the detailed model does not account for the nonlinear effect of a fuel assembly impacting the cell and the support pad movements, the internal loads and stresses for the rack assembly obtained from this model are corrected by load correction factors. The load correction factors are derived from the single cell nonlinear model results and are applied to the components in the structural analysis. The responses of the model from accelerations in three directions are combined by the SRSS method in the structural analysis. The loads in the major components are examined and the maximum loaded section of each of the components is found. These maximum loads from the detailed model are used in the structural analysis to obtain the stresses within the rack assembly."

3.4.3 Review of the Analysis Method

The focus of the meeting was on the methods used to relate the two-dimensional nonlinear analysis to the three-dimensional linear analysis for the purpose of evaluating stresses. Displacements were obtained directly from the nonlinear model where no conversion factors were needed.

A detailed review was made of the two-dimensional nonlinear dynamic model and of the source and formulation of parameters used therein. The two-dimensional model incorporates the nonlinear impacting parameters of a fuel assembly within a fuel rack cell and includes sliding and lift-off of the rack module mounting feet. Because the two-dimensional analysis must employ a time history solution to resolve the effects of the nonlinear elements, the two-dimensional model is a limited model not well suited for detailed stress analysis.

The three-dimensional linear model is a comprehensive model that is suitable for predicting stresses in all regions of the rack module. However, it is a linear model in which the fuel assembly masses are included directly with the rack masses without any consideration for their moving through clearance spaces and impacting the rack structure. Neither are the mounting feet considered to slide or impact vertically following a lift-off, should it occur. However, as a linearized model of the spent fuel rack module for internal stress analysis purposes, the three-dimensional model was reviewed and found to be acceptable.

The remaining review was directed to the methods of incorporating the dynamic loading effects from the two-dimensional nonlinear impacting model to the three-dimensional linear stress analysis model. This was accomplished by comparing two selected loading parameters of the two-dimensional and three-dimensional mathematical models and using these parametrs to establish a load correction factor with which to correct dynamic response analyses made using the three-dimensional linear model. A detailed review of the magnitudes of the load correction factors resulting from analysis of the spent fuel racks for the McGuire Nuclear Station revealed that the load correction factors actually reduced both the base moment and shear of the region 1 racks, but only the base moment of the region 2 racks. The values of the factors are:

	Region 1	Region 2
Base moment factor	0.805	0.708
Base shear factor	0.98	1.287

While the thought that a load correction factor, which relates impacting two-dimensional behavior to loads and stresses predicted by a linear, non-impacting, three-dimensional model, would actually reduce the load and stress predicted by the three-dimensional model may be a little surprising, this can be true. When one considers the fact that the linear three-dimensional model assumes that the fuel assumbly mass is rigidly fixed within the rack structure, then all of the mass in the rack participates in the linear response of the rack. In the two-dimensional nonlinear model, the rack structure responds first to the seismic stimuli, followed by an increment of time later impacting of the fuel assembly within the rack cell (should it occur) after the clearance gap between the fuel assembly and the rack cell walls closes. Thus, the actual combined dynamic response of the rack and the fuel assembly is highly dependent upon the dynamic parameters of the system, including the case where the base moments and shears could be reduced over that of a linear model. Also, in the nonlinear model, a spring-damper model was used for the cell-fuel composite. This model provided a more compliant coupling which is more representative of reality. This serves to reduce the stress in the nonlinear model.

In summary, although the Licensee's analysis method appears to provide an approach toward providing an estimate of the stresses where separate nonlinear, impacting displacement solutions and linear stress analysis must be combined, there are a number of criticisms that limit its acceptance:

- o The analysis method is unconventional and is therefore not widely used in seismic analyses to permit extensive experience in many applications.

- o No examples validated by alternate analysis methods were provided to confirm the analysis method.

- o The three-dimensional stress analysis, the results of which are ratioed by the load correction factors from the two-dimensional, nonlinear analyses, still analyzes one earthquake direction at a time. Thus, the maximum stress at a point must be computed as the square root of the sum of the squares of the separate values for each earthquake direction. Peak stresses due to sharp impact forces may not be resolved well.

For these reasons, the method cannot be accepted as a reliable generalized method without further validation.

3.4.4 Detailed Review of Rack Stress Analysis

Although the methodology is not acceptable at this time as a general analysis method, this does not preclude the acceptability of this particular stress analysis. An essential element of the stress analysis is that it include all forces associated with the seismic excitation. These include (1) horizontal forces which arise from the acceleration of the racks and (2) vertical and horizontal forces on the mounting pads which result from a combination of horizontal reactions and from rocking motions of the rack modules. The vertical forces generated by the mounting pads lifting off and impacting the floor are of particular importance.

Any stress analysis that includes all of the loads sustained in a seismic event in which any resultant computed loadings are equal to or greater than the actual seismic loadings will provide a conservative analysis. In response to a request [9] during the review, the Licensee provided summations [10] of the vertical forces in an effort to show that the equivalent maximum mounting pad forces of the linear three-dimensional stress model equal or exceed the maximum mounting pad forces of the two-dimensional nonlinear model.

The data supplied by the Licensee [10] were as follows:

- o Region 1, 11 x 13 Rack, East/West Seismic

	Linear Model	Nonlinear Model
Total load	242,84*	240,100
Dead weight	120,00*	121,100
Ratio (total/dead wt.)	2.023	1.98

*Data assumed to be reported in error and to represent 242,840 and 120,000, respectively.

o Region 1, 11 x 13 Rack, North/South Seismic

	Linear Model	Nonlinear Model
Total load	218,088	200,057
Dead weight	118,779	121,100
Ratio (total/dead wt.)	1.84	1.65

o Region 2, 12 x 16 Rack, East/West Seismic

	Linear Model	Nonlinear Model
Total load	345,400	335,600
Dead weight	154,000	155,500
Ratio (total/dead wt.)	2.24	2.16

o Region 2, 12 x 16 Rack, North/South Seismic

	Linear Model	Nonlinear Model
Total load	413,200	457,500
Dead weight	156,400	155,500
Ratio (total/dead wt.)	2.64	2.94

In reviewing these results, a recognized consultant retained for the review of the nonlinear analysis methods offered the following [11]:

"The results presented for the Region 1 racks clearly meet this condition; (it is assumed that the linear forces given for East/West Seismic are mistyped and should read 242840 and 120000 rather than 24284 and 12000).

The results presented for Region 2 racks do satisfy this requirement for the East/West component but for the North/South the loads in the linear model are only 90.3% of those predicted by the nonlinear model: 413200 for linear vs 457500 for nonlinear. There are several factors which contribute to the conservatism of the stress analysis of the North/South component in Region 2: the East/West component, which is more severe, was used for the nonlinear calculation; the Region 2 linear model includes the mass of a consolidated fuel canister which is much heavier than the standard fuel assembly which was used in the nonlinear model; East/West support spacing, which is smaller, is used in generating the nonlinear forces. There are also other elements of conservatism in the fuel rack analysis which apply to all of the components-item 3 on page 3 of letter of Mr. H. B. Tucker.

It would have been desirable for the Licensee to provide some numerical estimates that show that these factors of conservatism are sufficient to negate the loss of 9.7% of the vertical load in Region 2. Obviously, if no stresses exceed the allowable with Rm scaled up enough to yield a total load of 457500 pounds, there is no problem. Similarly, a nonlinear calculation which demonstrates that the factors of conservatism are sufficient to reduce the nonlinear load to 413200 would eliminate any questions. However, in view of the many factors of conservatism, this small underestimate of one of the loads may not require further substantiation. In Region 1, everything is fine."

In summary, while the methodology used for the stress analysis cannot be accepted without further validation, the detailed review of the stress analysis for these particular rack modules coupled with the conservatisms seen to be present indicate that the stress analysis is acceptable.

3.5 REVIEW OF SPENT FUEL POOL STRUCTURAL ANALYSIS

3.5.1 Spent Fuel Pool Floor Analysis

The McGuire Nuclear Station fuel pool slab is a reinforced concrete plate structure lined with stainless steel. The Unit 1 floor consists of a 4-ft 6-in reinforced concrete slab supported on 6-ft 6-in deep by 4-ft 6-in wide reinforced concrete beams, and has 4-ft 0-in wide walls at the perimeter.

The Unit 2 floor is a 11-ft 0-in thick reinforced concrete slab supported on a bedrock foundation. The additional thickness in Unit 2 is due to construction considerations.

The analysis was presented to demonstrate structural integrity of the floor systems for the postulated loading conditions with the new high-density racks.

3.5.2 Licensee's Assumptions

The Licensee made the following assumptions for the analysis:

1. The slab is modeled as a mesh composed of beam elements.
2. The stiffness of the pool liner was ignored in the analysis of concrete slab.
3. Original plant response spectra are used in consideration of seismic loadings.
4. Dynamic rack loads are taken from the Westinghouse rack module tables.

3.5.3 Analysis Procedure

The Unit 1 pool slab was modeled by beam elements with boundaries at centerlines of walls and deep beams. The slab area with the greatest clear span and largest load/area ratio was analyzed as the most critical case.

The static loads were gravity loads from water, concrete, and racks. Dynamic loads were obtained from OBE and SSE spectra curves for McGuire Nuclear Station, and from dynamic rack loads given in the Westinghouse rack module tables. The Licensee provided additional information [5] as follows:

> "The McGuire Auxiliary Building is a poured in place reinforced concrete structure as stated in Section 3.8.4.1.1 of the McGuire FSAR. Contained in this building are auxiliary systems, control rooms, and spent fuel pools for both units along with related piping and electrical cables. The mass added as a result of fuel densification is negligible compared to the mass of the structures and equipment comprising the Auxiliary Building, thus, the seismic response spectra applicable to the spent fuel pool floor slab is not altered. The method of dynamic analysis is described in Section 3.7.2.1 of the McGuire FSAR."

A thermal gradient was imposed across the pool slab during normal operation. A loading condition with pool temperature reaching 212°F was also evaluated.

The analysis was performed by using the "STRUDL" computer code.

The results of the analysis were summarized by the Licensee in Table 3.1-1 [1], which indicates that the fuel pool floor system has sufficient capacity to sustain the loading or the new rack conditions with a design margin of 1.3.

The Unit 2 slab system is a different type of structure than Unit 1. In a recent response [5], the Licensee provided the following information regarding Unit 2:

> "As stated in Section 3.1, paragraph 4, of the license submittal, the Unit 2 pool floor is supported continuously on bedrock. All dead, live and seismic loads are transmitted directly through the floor to the bedrock foundation. In response to an earlier question concerning the model and loading system used in the analysis of the spent fuel pool floor (reference response to Question No. 1, letter dated June 19, 1984), reference was made only to the Unit 1 pool floor slab. The Unit 2 pool floor slab analysis was not addressed since the Unit 1 pool floor represented the limiting condition."

The Licensee indicated that the pool floor system has significant design margin to sustain the additional floor loading.

3.6 REVIEW OF HIGH-DENSITY FUEL STORAGE RACKS' DESIGN

Comments and conclusions following the review of Sections 2.3.1.3 and 2.3.1.4 of Section 2.3 [1] entitled "Design Evaluation" are contained in the following subsections:

3.6.1 Fuel Handling Crane Uplift Analysis

In Section 2.3.1.3 [1], the Licensee stated that the rack can withstand the maximum uplift load of 3000 lb of the fuel handling crane without violating the criticality acceptance criteria. The uplift load is assumed to be applied to fuel cell. The Licensee stated that the resulting stresses are within the acceptable stress limits, and there is no change in rack geometry of a magnitude which causes the criticality acceptance criteria to be violated.

However, the reviewed report [1] does not provide the stress level or the extent of the rack deformation under the uplift load.

3.6.2 Fuel Assembly Drop Accident Analysis

In Section 2.3.1.4 [1], the Licensee discussed the unlikely event of dropping a fuel assembly, wherein two accident conditions are postulated.

The first accident condition considers that the weight of the fuel assembly, control rod assembly and handling mechanism (3000 lb) impacts the top end fitting of a stored fuel assembly from a drop height of 6 ft. Although the Licensee did not provide the analysis or analysis methods, the Licensee stated [1] that "calculations show that the impact energy is absorbed by the dropped fuel assembly, the cells and the rack base plate assembly." Although independent analysis was not performed, rack modules under fabrication were inspected at the Licensee's vendor facilities during the review. Based upon this inspection of the rack construction, the rack modules are considered to be acceptable for a 6-ft fuel assembly drop as described above.

The second accident condition considered by the Licensee is that of a dropped assembly (3000 lb) falling straight through an empty cell and impacting upon the base plate from a drop height of 234 in. For this case, the Licensee provided the following additional information [5]:

> "Analysis has been performed which shows that the 234 in fuel assembly accidental drop satisfies the design criteria of not resulting in perforation of the pool liner. In the analysis it is shown that the energy of the falling fuel assembly is satisfactorily absorbed by the crushing of the fuel rack base plate and the deformation of the lower portion of the fuel assembly (lower fitting and lower portion of the guide tubes and instrument tube). The load tranmitted to the pool liner is such that the stress developed in the liner does not result in perforation. It should be noted that the analysis performed is conservative in that the fuel assembly is assumed to be under free fall (water resistance within the cell is neglected), and it is assumed that no energy is dissipated by the breaking of welds which hold the base plate to the rest of the rack."

For both accidents above, the Licensee indicated that the spent fuel pool liner would not be perforated and that the criticality acceptance criterion [1] was not violated.

4. CONCLUSIONS

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

- 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 in which the displacements are small compared to the available clearance space. As the Licensee's reported displacements are small, an acceptable use of the hydrodynamic coupling was employed.

- o Computed displacements are small relative to clearance between rack modules or between rack modules and the spent fuel pool walls. Thus, the use of two-dimensional dynamic rack module analysis was satisfactory for displacement.

- o While the methodology employing two-dimensional nonlinear models and linear three-dimensional models correlated by load correcting factors to introduce the nonlinear impacting load characteristics to the three-dimensional linear model was not considered to be acceptable without further validation as a stress analysis method, a detailed step-by-step review of the stress analysis coupled with additional load tabulations requested and supplied indicates that, with the conservatisms noted to be present, the stress analysis is acceptable.

- o The spent fuel pool floor system has design margin to sustain the additional floor loadings.

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