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

November 14, 1984

Docket No. 50-244
LS05-84-11-012

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

SUBJECT: INCREASE OF THE SPENT FUEL POOL STORAGE CAPACITY

Re: R. E. Ginna Nuclear Power Plant

The Commission has issued the enclosed Amendment No. 65 to Provisional Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated April 2, 1984 as supplemented on June 12, 1984.

The amendment authorizes you to increase the storage capacity of the spent fuel pool from 595 to 1016 fuel assemblies by modifying six of the rack modules in the spent fuel pool. The unmodified racks (Region I) are capable of storing existing fuel assemblies as well as the recently approved Westinghouse Optimized Fuel Assemblies with initial enrichments up to 4.25% U-235. Prior to storing fuel assemblies in the modified racks (Region II), 60 days must have elapsed since the reactor reached hot shutdown and the combination of the assembly average burnup and the initial U-235 enrichment must satisfy certain criteria as discussed in Section 1 of our Safety Evaluation.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 27, 1984 (49 FR 30261). No requests for hearing and no public comments were received. A Notice of Issuance of Environmental Assessment of Finding of No Significant Impact was published in the Federal Register on November 14, 1984 (49 FR 45086).

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A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Sincerely,

Original signed by

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 65 to
   License No. DPR-18
2. Safety Evaluation

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See next page

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 65
License No. DPR-18

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

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated April 2, 1984 and supplemented by letter dated June 12, 1984 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.

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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 Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 65, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 65

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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

REMOVE	INSERT
5.4-1	5.4-1
---	5.4-2
---	5.4-3
---	5.4-4
---	5.4-5

5.4 Fuel Storage

Specification

5.4.1 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water.

5.4.2 The new and spent fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations. The spent fuel storage racks are divided into two regions as depicted on Figure 5.4-1. The fuel is stored vertically in an array with sufficient center to center distance between assemblies to assure Keff $\leq$ 0.95 for (1) unirradiated fuel assemblies delivered prior to January 1, 1984 (Region 1-15) containing no more than 39.0 gms U-235 per axial cm, and (2) unirradiated fuel assemblies delivered after January 1, 1984 containing no more than 41.9 gms U-235 per axial cm. Both cases assume unborated water used in the pool.

5.4.3 In Region 2 of the spent fuel storage racks, fuel is stored in a close packed array utilizing fixed neutron poisons in each of the stored locations. For discharged fuel assemblies to be stored in Region 2, (1) 60 days must have elapsed since the core reached hot shutdown prior to discharge and (2) the combination of assembly average burnup and initial U-235 enrichment must be such that the point identified by these two parameters on Figure 5.4-2 is above the line applicable to the particular fuel assembly design, therefore assuring that Keff $\leq$ 0.95.

Amendment No. 11, 60, 65

5.4.4 The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit.

Basis

The center to center spacing of Region 1 insures that Keff $\leq$ 0.95 for the enrichment limitations specified in 5.4.2[1], and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100[2].

In Region 2, Keff $\leq$ 0.95 is insured by the addition of fixed neutron poison (boraflex) in each of the Region 2 storage locations, and a minimum burnup requirement as a function of initial enrichment for each fuel assembly design. The 60 day cooling time requirement insures that for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100.

The two curves of Figure 5.4-2 divide the fuel assembly designs into two groups. The first group is all fuel delivered prior to January 1, 1984. This incorporates all Exxon and Westinghouse HIPAR designs used at Ginna[4] The second curve is for the Westinghouse Optimized Fuel Assembly design delivered to Ginna beginning in February 1984[3]

The assembly average burnup is calculated using INCORE generated power sharing data and the actual plant operating history. The calculated assembly average burnup should be reduced by 10% to account for uncertainties. An uncertainty of 4% is associated with the measurement of power sharing. The additional 6% provides additional margin to bound the burnup uncertainty associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.[3]

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The records of these calculations shall be kept for as long as fuel assemblies remain in the pool.

References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RG&E March 15, 1984.

FIGURE 5.4-1

SPENT FUEL STORAGE RACKS

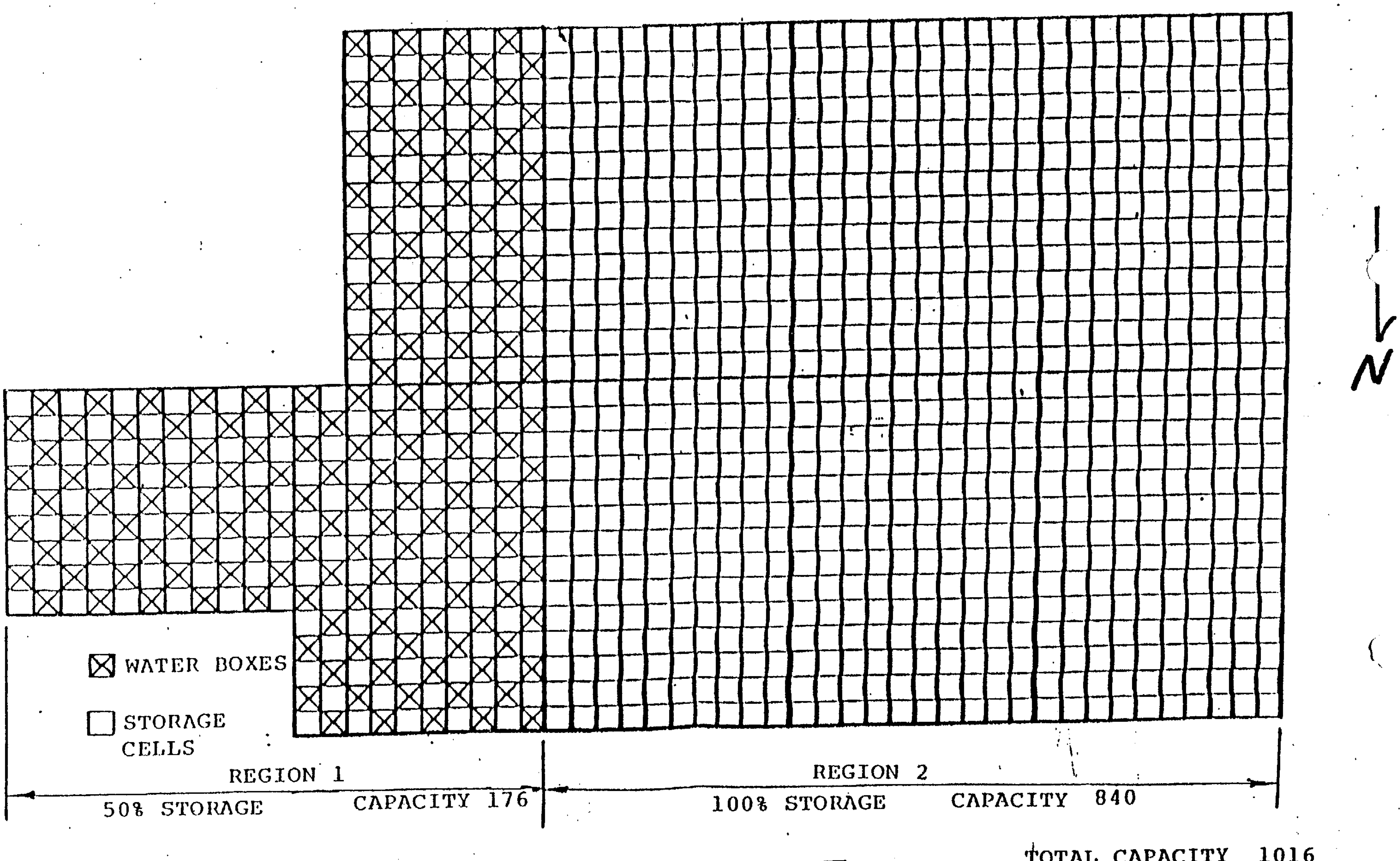
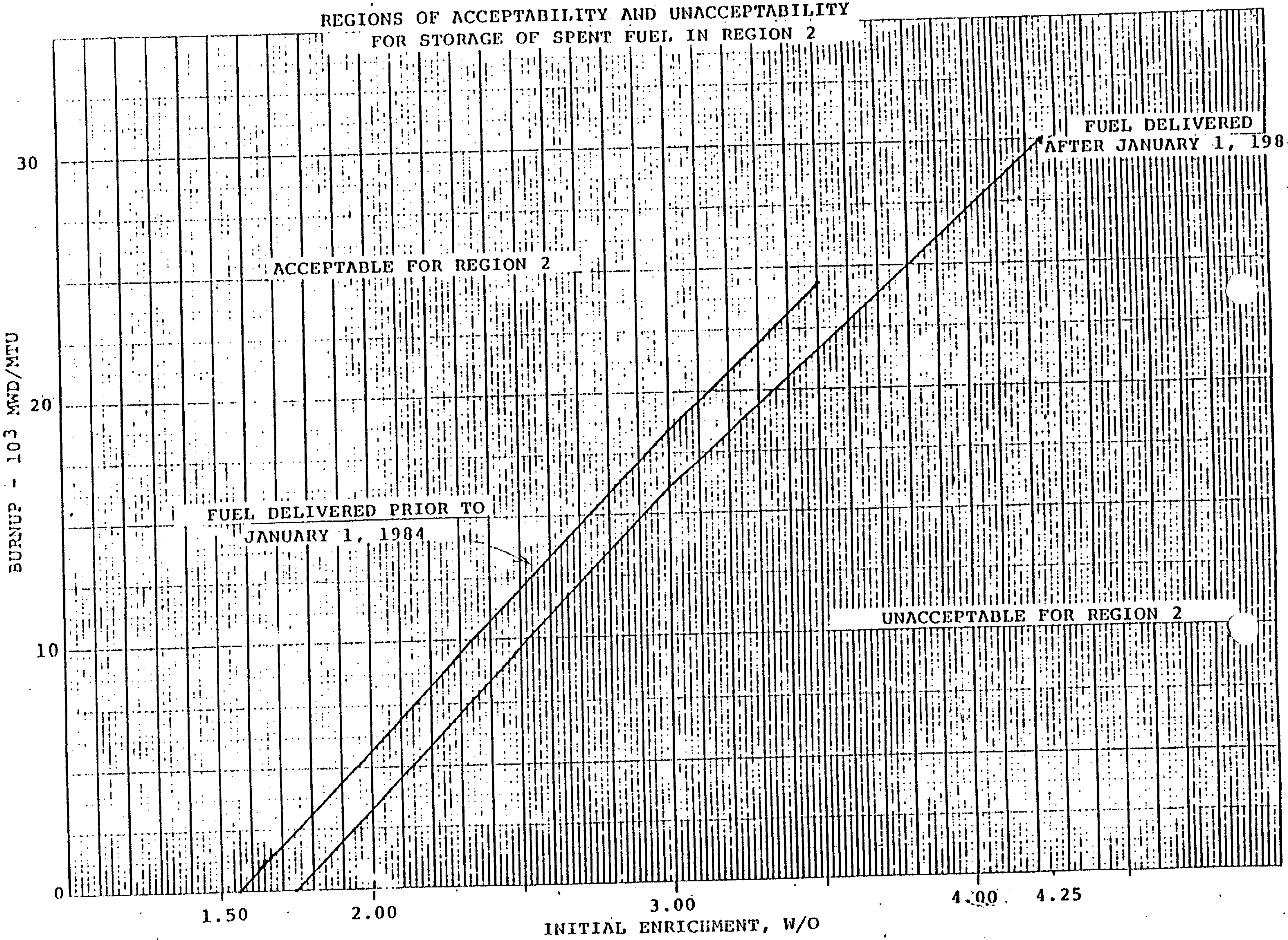


FIGURE 5.4-2

REGIONS OF ACCEPTABILITY AND UNACCEPTABILITY
FOR STORAGE OF SPENT FUEL IN REGION 2



SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 65 TO

PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

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Dated: November 14, 1984

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1.0 INTRODUCTION

By letter dated April 2, 1984, as supplemented June 12, 1984, Rochester Gas and Electric Corporation (RG&E, the licensee) submitted an application to increase the storage capacity of the spent fuel pool (SFP) by modifying the six west-most rack modules in the spent fuel pool. By letters dated July 6, July 31, August 10, August 13, August 27, September 27, and October 23, 1984 the licensee provided additional clarification in response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information. This would be the second rerack for Ginna, the first being authorized by Amendment No. 11 on November 15, 1976 which increased the capacity of the SFP from its original capacity of 210 to 595 fuel elements.

The present amendment would authorize the licensee to increase the storage capacity of the SFP from the current capacity of 595 fuel assemblies to 1016 fuel assemblies with average planar enrichments no greater than 4.25 weight percent U-235.

At the present time, there are 332 spent fuel assemblies in the SFP. The licensee also has 81 fuel assemblies stored at what was formerly the NSF at West Valley, New York. These assemblies will be transferred to the Ginna SFP by September 1985. The licensee estimates that full-core reserve in the SFP would be lost following the 1987 refueling. Since this date is earlier than the date a federal depository should be available for spent fuel [1998- Nuclear Waste Policy Act of 1982, Section 302(a)(5)] additional spent fuel capacity is needed.

RG&E proposes to increase the storage capacity of the R. E. Ginna storage pool by modifying six of the nine existing "flux trap" type storage racks currently in the storage pool to high density "fixed poison" type storage racks. This change will double the storage capacity of the six modified racks from 420 to 840 storage cells. The storage capacity of the three remaining "flux trap" type storage racks (176 storage cells) will remain unchanged. Therefore, the total storage capacity of the pool will be increased from 595 to 1016 storage cells. Since the pool will contain two different types of storage racks, it will be divided into two regions. Region 1 will consist of the three "flux trap" type storage racks and Region 2 will consist of the six modified "fixed poison" type storage racks.

Previously, RG&E proposed and received NRC staff approval for a possible increase in the U-235 enrichment of the fuel assemblies from 3.5 to 4.25 weight percent. The licensee also received approval for the use and storage of the Westinghouse Optimized Fuel Assemblies (OFA). Table 2-1 of the licensee's submittal of April 2, 1984 shows that the Region 1 storage racks are capable of safely storing the previously existing R. E. Ginna fuel assemblies as well as the Westinghouse OFA. However, prior to storing fuel assemblies in the new fixed poison (Region 2) storage racks, the fuel assemblies must meet the following conditions:

1. 60 days must have elapsed since the reactor reached hot shutdown.

2. The combination of the assembly average burnup and the initial U-235 enrichment must be such that the point identified by the two parameters on Figure 5.4-2 of the April 2, 1984 submittal is above the line applicable for the particular fuel assembly design. This will assure that k_{eff} for the stored fuel is equal to or less than 0.95. To assure that the burnup has been properly established, the licensee indicates that the burnup of each assembly will be established using the Nuclear Fuel Accountability Code that was started in 1970. This code establishes the isotopic content of the fuel and other parameters such as burnup. This information along with the curves on Figure 5.4-2 of the submittal will be used to determine if an assembly can acceptably be stored in Region 2. The seismic analysis of the modified spent fuel storage racks incorporated higher loadings which would be expected for the case of rod consolidation. However, the licensee request of April 2, 1984 as supplemented June 12, 1984 requested approval only for storage of unconsolidated fuel.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 27, 1984 (49 FR 30261). No requests for hearing and no public comments were received.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

The storage racks have been analyzed for two groups of fuel assembly designs. The first group consists of all fuel delivered prior to 1984 and incorporates all Exxon and Westinghouse HIPAR designs used at Ginna containing no more than 39.0 gm U-235 per axial cm (3.5 weight percent U-235). The second group consists of the Westinghouse OFA design delivered to Ginna beginning in February 1984 containing no more than 41.9 gm U-235 per axial cm (4.25 weight percent U-235).

The Region 2 design consists of six racks, each containing 140 stainless steel cells for a total of 840 fuel assembly storage locations. There is a 8.43 inch center-to-center spacing between assemblies and a neutron absorbing material, Boraflex (Ref. 1), is attached to the stainless steel walls of each storage cell. Boraflex consists of boron carbide powder in a rubber-like silicone polymeric matrix. The minimum boron-10 density in the Boraflex is 0.020 gm/cc.

The design is intended to contain any of the Exxon or Westinghouse HIPAR or OFA 14x14 fuel assemblies used in Ginna with an initial enrichment of up to 4.25 weight percent U-235 at an assembly average exposure of 30,000 MWD/MTU. For lower initial enrichments, the amount of exposure required for storage in Region 2 will be reduced. For 3.00 weight percent U-235, for example, it is 15,960 MWD/MTU and for 1.75 weight percent U-235, even fresh fuel can be stored in Region 2 as seen from Technical Specification Figure 5.4-2.

2.1.1 Analysis Methods

The criticality aspects of the storage of Westinghouse and Exxon fuel assembly designs used at Ginna in the burnup-dependent region (Region 2) of the spent fuel storage pool have been analyzed using the PDQ-7 computer program for reactivity determination with four energy group neutron cross sections generated by the LEOPARD program as modified by Pickard, Lowe and Garrick, Incorporated (PLG). These codes have been benchmarked against both Westinghouse and Battelle Pacific Northwest Laboratories critical experiments with pellet diameters, water-to-fuel ratios and U-235 enrichments encompassing those in the Ginna fuel rack design. In addition, a series of PuO_2 - UO_2 critical experiments were analyzed to determine the accuracy of calculations of systems containing significant amounts of plutonium mixed with UO_2 and, therefore, the accuracy of reactivity calculations for irradiated fuel. These latter results led to the conclusion that the calculational model is capable of determining k_{eff} of the Ginna spent fuel racks with a combined LEOPARD/PDQ-7 model bias of +0.0031 and a 0.0186 Δk uncertainty corresponding to a 95 percent probability at a 95 percent confidence level (95/95).

In order to establish burnup criteria for storage in Region 2, a constant storage rack infinite multiplication factor (with minimum post-shutdown fission product inventory) contour is constructed as a function of burnup and initial enrichment using LEOPARD and PDQ-7. Since the calculations use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and axial leakage is not accounted for, the calculated multiplication factor is, in reality, K_∞, which will be larger (more conservative) than k_{eff}. This contour is based on a high enrichment endpoint of 4.25 weight percent U-235 and 30,000 MWD/MTU as shown in the attached Figure 5.4-2 from the Ginna Technical Specifications. This is representative of the Westinghouse OFA fuel delivered after January 1, 1984. A similar curve for Exxon and Westinghouse HIPAR fuel delivered prior to 1984 is also shown.

2.1.2 Spent Fuel Rack Storage

The basic rack cell at 20°C, 4.25 weight percent U-235, and 30,000 MWD/MTU results in a reactivity of 0.9072. Including all the appropriate calculational biases and 95/95 uncertainties results in a maximum reactivity change of 0.0390, giving a maximum reactivity of 0.9462, which meets the staff acceptance criterion of less than or equal to 0.95. For lower enrichments with the same computed multiplication factor, the amount of exposure will be reduced, reducing the reactivity uncertainties due to depletion of fuel and buildup of fission products. The total uncertainty is, therefore, reduced making the rack cell reactivity calculated at 4.25 weight percent U-235 and 30,000 MWD/MTU conservative for all lower enrichments. For additional conservatism, a constant multiplication of 0.9050 is used to generate the final burnup verus enrichment curves in the Technical Specifications.

2.1.3 Accident Analyses

Since the maximum possible reactivity of the Region 2 spent fuel rack is based on infinite array calculations both laterally and vertically, the effect of a dropped fuel assembly on top of the rack would not exceed the calculated maximum reactivity value. In addition, the racks are designed to prevent a dropped fuel assembly from occupying a position other than a normal fuel storage location. Procedures exist to assure that assemblies discharged from the core are first moved to Region 1. After the refueling operation is complete and the suitability of each spent fuel assembly for movement and storage into Region 2 is verified, this fuel will be moved into Region 2. Therefore, administrative procedures exist to help preclude a fuel misloading event. However, even if it occurs, the spent fuel storage pit is filled with borated water at a concentration sufficient to maintain k_{eff} below 0.95. NRC review policy permits credit for this boron.

2.1.4 Technical Specifications

The staff concludes that the modifications to the Ginna Technical Specifications submitted by licensee letters dated April 2, 1984, and June 12, 1984 are acceptable to allow operation with the proposed expansion of SFP storage capacity.

2.1.5 Conclusions

The staff concludes that the proposed storage racks meet the requirements of General Design Criterion 62 as regards criticality. This conclusion is based on the following considerations:

(1) Acceptable calculation methods which have been verified by comparison with experiment have been used.

(2) Conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions.

(3) Credible accidents have been considered.

(4) Suitable uncertainties have been considered in arriving at the final value of the multiplication factor.

(5) The final effective multiplication factor value meets the staff acceptance criterion.

2.2 Spent Fuel Pool Cooling and Makeup

2.2.1 Decay Heat Load and Spent Fuel Pool Cooling System

In 1981, the staff reviewed and approved a proposed SFP cooling system modification for Ginna (Ref. 2). This modification will be implemented in 1986, and will consist of the addition of a new cooling loop in parallel with the existing loop which is sized to accommodate the maximum normal and abnormal heat loads should the storage capacity be increased to 1360 fuel assemblies at some future date. Since the present proposal calls for an increase in the total storage capacity of the pool to 1016 fuel assemblies, the staff concludes that the previously approved SFP cooling system will acceptably handle the maximum normal and abnormal heat loads for this proposed expansion.

The modified SFP cooling system could accommodate the full core discharge and normal refueling heat loads through the year 2010. On those occasions where a full core discharge takes place, the licensee has committed to incrementally increase the decay time in the reactor vessel from 8 days in the year 1981 to 14 days in the year 2010 in order to assure that the maximum pool water temperature will not exceed the Technical Specification limit of 150°F. The licensee has also indicated that fuel consolidation may be proposed in the future, however this is not included in the currently proposed fuel pool expansion and is not a part of the staff review of the SFP cooling system adequacy.

Based on the above, the staff concludes that the maximum normal and abnormal heat loads resulting from the proposed expansion will not exceed the anticipated heat loads used in the previous evaluation of the SFP cooling system modifications and, therefore, the SFP cooling system is acceptable.

2.2.2 Boiloff Rate and Makeup Systems

As indicated in the SFP cooling system discussion above, the decay heat loads will not exceed those previously considered and approved during the pool cooling system modification review in 1981. Therefore, the staff concludes that the associated boiloff rate also will not exceed that which was previously accepted. Similarly, the staff concludes that the demands on the pool water makeup system will not exceed those previously reviewed and approved and, therefore the makeup capability is acceptable.

2.2.3 Local Boiling

At the time of the previous storage rack expansion review, the licensee provided an analysis to determine the difference in temperature of the water exiting from the top of the storage cells with respect to the corresponding water saturation temperature. It was assumed in this analysis that a recently discharged batch of fuel assemblies were grouped together in the original storage cells (Region 1 arrangement) in a location as far away from the cooling system cold water inlet as possible. Under these conditions, it was found that the temperature of the water exiting from the hottest fuel assembly is less than 155°F and the corresponding saturation temperature is over 235°F. There is therefore a margin of about 80°F to prevent local boiling from occurring.

In the case of the modified storage racks (Region 2 arrangement), fuel will not be moved into these storage racks until at least 60 days of decay has taken place. Therefore, the decay heat load would have decreased to about 60 percent of that of recently discharged fuel. This combined with the enlarging of the flow holes in the former water boxes indicates that the exit temperature of the water from the Region 2 storage cells will be less than previously reviewed and approved for the Region 1 storage cells. Therefore, the staff concludes that adequate margin to local boiling has been demonstrated for the Region 2 storage racks and they are therefore acceptable in this regard.

2.2.4 Conclusion

The staff has reviewed the spent fuel cooling and makeup as it relates to the second SFP expansion program for R. E. Ginna and concludes the following:

(1) The resulting decay heat loads in the pool are less than those assumed in the proposed SFP cooling system modification which was approved by the staff in 1981.

(2) The boiloff rate assuming the loss of all pool cooling is less than that assumed in the staff's 1981 review, and therefore the makeup systems previously approved by the staff will provide assurance that the fuel will not be uncovered.

(3) The margin between the temperature of the water exiting from the Region 2 storage cells will be approximately 80°F less than the corresponding pool water saturation temperature, thus providing adequate assurance that local boiling will not occur.

In summary, based on its review, the staff concludes that the R. E. Ginna proposed second SFP expansion meets the guidelines of SRP Section 9.1.3, and is therefore, acceptable.

2.3 Rack Modification and Load Handling

The steps and procedures required to accomplish reracking the SFP will be developed so as to eliminate the need for carrying loads over stored spent fuel and will ensure that reasonable protective measures will be taken to preclude load drops during reracking.

2.3.1 Modified Storage Racks

RG&E engaged US Tool & Die to perform the mechanical, structural and material analysis of the modifications to the existing Wachter storage racks. The nuclear analysis was performed by Pickard, Lowe, and Garrick Inc.

The rack modification program will consist of sequentially removing and modifying one storage rack at a time. The steps involved in the modifications will be as follows:

(1) The 332 fuel assemblies presently in the pool will be moved as far as practical from the rack to be removed.

(2) A diver will remove the four mounting bolts that attach the storage rack to its support base.

(3) Using the lifting rig, the storage rack will be raised clear of the pool surface and partially decontaminated using high pressure water before it is moved to the decontamination pit.

(4) Following additional decontamination in the decontamination pit, the guide funnels and guide angles will be cut free of the storage rack.

(5) The existing lifting attachments will be removed, and four modified bottom plates with the new lifting slots will be installed.

(6) The flow holes in the bottom plates will be enlarged and ½ inch bottom plates will be installed in the former water boxes.

(7) The right-angled poison assemblies will be installed and welded in place in each storage cell.

(8) Divers will install appropriate shims at the four corners of the support base in the pool.

(9) The existing jack screws on the racks will be retracted so that the weight of the rack will bear in the support base shims.

(10) The modified rack will be lifted, transported and lowered onto the support base shims.

(11) The above steps will be repeated for the remaining five storage racks to be modified.

(12) All seismic support between the rack bases will be removed.

The right-angled poison assemblies to be installed in the storage cells will consist of a nominal 0.062 inch thick preformed sheet of stainless steel and two nominal 0.075 inch thick-by-7 5/8 inch wide strips of Boraflex are sandwiched between the cell walls and the preformed stainless sheets. This installation will reduce the internal dimensions of the storage cells from a nominal 8.280 x 8.280 to 8.143 x 8.143 inches.

During the 1977 spent fuel expansion when the "flux trap" type storage racks were installed, the staff determined that the racks met seismic Category I criteria. Since RG&E proposes in this submittal to convert these racks to provide twice the number of storage cells, the effective weight of the stored fuel in a given rack will be doubled. Further, RG&E has requested the staff to evaluate the adequacy of the storage racks if at sometime in the future they decide to implement a rod consolidation program. This will effectively increase the weight of the stored fuel assemblies over that previously approved by the staff during the 1977 review. A separate structural evaluation of the seismic design capability of the racks which accounts for the increased weight of the stored fuel is reported in Section 2.4 of this Safety Evaluation.

Following the installation of the right-angled poison assemblies in each storage cell, a gauge will be inserted into the cell to verify that the fuel assemblies will not experience unacceptable frictional forces during their insertion or withdrawal. Westinghouse guidance in this regard states that a drag force of 50 pounds is not to be exceeded. Further, based on previous experience, the licensee stated that a drag force of approximately 400 pounds is required before damage to the fuel assemblies will occur. RG&E has committed to evaluate all drag forces in excess of 50 pounds on a case-by-case basis. The licensee has stated that in no case will the developed drag force be accepted if it is sufficient to threaten the integrity of a fuel assembly. The modified storage racks will have an estimated weight of 28,000 pounds each. From this, and the sturdiness of the rack construction, the staff concludes that the vertical frictional force of 400 pounds exerted by the fuel handling crane will not cause damage to the storage rack. Further, as a result of having removed the lead-in funnels on the storage cells, the licensee has committed to provide a portable lead-in funnel to aid the operator in properly aligning fuel assemblies during their insertion in the Region 2 storage cells. The gaps between the storage racks are a small fraction of the cross sectional dimensions of a fuel assembly. Therefore, the staff concludes that a fuel assembly cannot inadvertently be placed in any location other than the designated storage areas within the lattice array of the racks.

In RG&E's letter of June 12, 1984 the licensee indicated that fuel rod consolidation may be proposed at some time in the future. However, the licensee requested that the staff not consider consolidation as part of the rack modification and load handling review.

With regard to a postulated vertically dropped fuel assembly accident, the licensee states that if the assembly were to drop 14 feet onto a flat surface, the resulting impact stresses would be acceptably low and no significant damage would be expected in any fuel rods. If the dropped assembly were to strike a sharp object, the licensee conservatively assumed that one row of fuel rods would fail. In the case of a tipped fuel assembly drop, the resultant kinetic energy would be much less than for the vertical drop. Therefore, aside from the postulated damage to a row of fuel rods, the licensee concluded that the crush strength of the storage cells will protect the stored fuel from damage from dropped fuel assemblies.

Based on the above, the staff concludes that the Region 2 storage racks will adequately support and protect the stored fuel assemblies and are, therefore, acceptable.

2.3.2 Load Handling

There are currently 332 fuel assemblies stored in the pool. The licensee has indicated that for each reracking operation, the stored spent fuel assemblies will be moved away from the area where the load handling operations are to take place in order to minimize the consequences should a load drop occur and minimize the radiological exposure to the divers who attach the lifting device to the storage racks.

The load handling operations associated with reracking will be conducted in accordance with Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" as it relates to safe load paths, procedures, crane operator training and qualifications, and crane inspection and maintenance. Further, the special lifting device interposed between the storage racks and the crane hook will consist of two redundant spreader bars, slings, and vertical lifting adapters. Both spreader bars are located such that the center of gravity of the storage rack is directly below the crane hook. Therefore, should a failure occur in one of the spreader bars, the load will remain stable and would not swing. The calculated stresses for the special lifting device are also less than that prescribed in the guidelines of ANSI 14.6.

As in the previous SFP expansion, the auxiliary building crane will be used for handling the storage racks. This crane was procured to EOCI-61 specifications.

Based on the manufacturer's (Whiting Corporation) evaluation of the crane provided by the licensee in response to the criteria of NUREG-0612, the staff concludes that the crane meets the intent of Guideline 7 of NUREG-0612, Section 5.1.1.

The range of travel of the crane is such that the hook cannot be placed directly over the center of gravity of the two most westward storage racks. To enable these storage racks to be lifted vertically without encountering mechanical interference with the adjacent storage racks, a chainfall or cable winch will be attached to the main hook block. The chainfall or winch will be anchored by means of a temporary holding beam attached to three auxiliary building columns in a fashion similar to that previously done during the 1976 reracking operations. The licensee acknowledged that this operation may cause some accelerated wear of the auxiliary building crane cable drum. However, due to the limited time of use for conducting this operation, the wear should not become significant. Further, the licensee states that the cable drum is due to be replaced as part of the overall upgrade of the crane to satisfy the criteria of NUREG-0554.

Based on the above, the staff concludes that the reracking operations will be performed in accordance with the guidelines of NUREG-0612 as applicable and that all reasonable measures will be taken to preclude unacceptable consequences in the unlikely event of a load drop. Therefore, the described reracking operations are acceptable.

2.3.3 Conclusion

The staff has reviewed the proposed modification of the SFP racks and load handling as it relates to the SFP expansion program for R. E. Ginna and concludes that the modified racks are designed such that:

(1) The maximum uplift friction forces developed by the crane will not damage the storage racks.

(2) The postulated dropping of a fuel assembly will not lead to unacceptable consequences.

(3) The arrangement of the storage racks within the pool is such that it is not possible to inadvertently insert a fuel assembly into a nondesignated space within the storage rack array.

(4) The racks satisfy seismic Category I criteria for unconsolidated fuel.

(5) Adequate load handling precautions will be taken during the reracking operations.

In summary, based on its review, the staff concludes that Ginna proposed SFP expansion meets the guidelines of SRP Sections 9.1.2, 9.1.4, and 9.1.5, and is therefore, acceptable.

2.4 Structural Design

The Safety Evaluation (SE) of structural aspects of the proposed modification is based on a review performed by NRC's consultant, Franklin Research Center (FRC). The FRC Technical Evaluation Report (TER) C5506-531 is appended to this SER as Appendix A.

2.4.1 Description of the Spent Fuel Pool and Racks

The spent fuel storage pool is designed for the underwater storage of spent fuel assemblies, failed fuel cans and control rods after their removal from the reactor. The pool is constructed of reinforced concrete having thick walls and is Class I seismic design. The slab of the pool is founded on bedrock. In addition, the entire interior basin face is lined with stainless steel plate.

The racks are stainless steel egg-crate structures. Original design of the racks is composed of three major components.

(1) The rack modules, which are rectangular arrays of cells of which one out of two are storage cells. The others are water boxes.

(2) The support bases, on which the rack modules rest, are rectangular construction of I beams.

(3) Seismic support between the bases and the pool walls provides a means to transmit horizontal loads from the racks to the walls.

Structural modifications for the proposed amendment are as follows:

(1) Using a special cutting machine remove guide tunnels and guide angles over the water bases so that spent fuel assemblies can be stored.

(2) Remove all (both Region 2 and Region 1) seismic supports between the rack base and the pool walls.

The seismic analysis was performed for both the standard and consolidated fuels.

2.4.2 Applicable Codes, Standards and Specifications

Load combinations and acceptance criteria were compared with those found in the "Staff Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979. The existing concrete pool structure was evaluated for the new loads in accordance with the requirements in the Ginna FSAR.

2.4.3 Loads and Load Combinations

Loads and load combinations for the racks and the pool structure were reviewed and found to be in agreement with the applicable portions of the Staff Position.

2.4.4 Seismic 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 seismic loads were applied to the model in three orthogonal directions. Damping values for the seimsic analysis of the racks were taken in accordance with the Regulatory Guide 1.61. Rack/fuel bundle interactions were considered in the structural analysis.

2.4.5 Design and Analysis Procedures

(1) Design and Analysis of the Racks - Horizontal seismic analysis was performed using the time history method. This accounts for the non-linearities inherent in the spent fuel storage racks which include fuel-to-rack wall impacts, rack sliding, and vertical impact due to rack tipping. The vertical seismic analysis was performed using spectra method. The vertical reaction loads were combined with the horizontal seismic loads using the square root of sum of the squares method.

Calculated stresses for the rack components were found to be well within the allowable limit. 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 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.

(2) Analysis of the Pool Structure - The floor of the SFP is a stainless steel lined, 3-foot thick, reinforced concrete slab. The slab is founded on bedrock. The structure of the pool was evaluated for the original FSAR and again for the floor loads associated with subsequent rack replacement. Because the rack will be modified to a free-standing design, only the increased concrete bearing stresses on the floor were evaluated. These were found to be acceptable.

2.4.6 Conclusion

The staff concludes that the proposed rack installation will satisfy the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 2, 4, 61, and 62, as applicable to structures, and is therefore acceptable.

2.5 Materials

The staff has reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water.

The only new material or components to be added during the proposed modification are the nuclear absorber strips. The existing spent fuel racks to be adapted in the proposed expansion are constructed entirely of Type 304 stainless steel, except for the nuclear poison material. The existing SFP liner is constructed of stainless steel. The high density spent fuel storage racks will utilize Boraflex sheets as a neutron absorber. 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, and the L-shaped stainless steel sheaths.

During modification, the flow holes in the bottom plates of the existing fuel storage cells will be enlarged and additional bottom plates will be added to the former water boxes. Each cell will contain an insert consisting of two Boraflex sheets at right angles to one another and an L-shaped stainless steel insert to hold them in place. The Boraflex absorber will not be sealed within the storage cell and vent paths for any gas generated during exposure will be available to the pool. The pool contains oxygen saturated demineralized water containing boric acid. The water chemistry control of the SFP has been evaluated and reported in the SER supporting Amendment No. 11 to the operating license and found to meet NRC recommendations. The increased storage capacity of the pool does not change this evaluation.

2.5.1 Evaluation

The pool liner, rack lattice structure and fuel storage tubes are stainless steel, which is compatible with the storage pool environment. 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 (Ref. 3), exposing Boraflex to 1.103×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water.

These materials are being used in many operating SFPs. The licensee committed to monitor the SFP surveillance program at Point Beach, which the staff has found acceptable. The materials in the Point Beach program are identical to the materials in this SFP and thus the monitoring of this surveillance is acceptable to meet the surveillance program requirement.

2.5.2 Conclusion

From the evaluation as discussed above, the staff concludes 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.

The staff further concludes 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.

The staff has reviewed the surveillance programs at the reactors cited by the licensee and concludes that the monitoring of materials in these spent fuel storage pools will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the SFP. The materials surveillance program in these cited units will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power well before comparable radiation exposures have been reached in the licensee's spent fuel racks. The staff does not anticipate, however, that such deterioration will occur. The monitoring program will ensure that in the unlikely situation that the Boraflex will deteriorate in the SFP environments, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

The staff, therefore, finds that the commitment to follow the monitoring program at the other PWR SFPs and the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components. The staff also finds that the licensee meets Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison. The staff therefore concludes that the materials to be used in the proposed modification are acceptable.

2.6 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the Ginna SFP racks with respect to occupational radiation exposure. The licensee estimates that the exposure for this operation will be approximately 78 man-rems. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job is being performed. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel.

2.6.1 Evaluation

One potential source of radiation is radioactive activation of corrosion products, termed "crud." Crud may be released to the pool water because of fuel movement during the proposed SFP rack modifications. This could increase radiation levels in the vicinity of the pool. During refuelings, when the spent fuel is first moved into the fuel pool, the addition of crud to the pool water from the fuel assembly and from the introduction of primary coolant to the pool water is greatest. However, the licensee, based on experience from plant's performing similar modifications, does not expect to have significant releases of crud to the pool water during modification of the SFP racks. In addition, the purification system for the pool, which has maintained radiation levels in the vicinity of the pool at low levels during normal operations, will be operating during the modification of the SFP racks. The staff has evaluated the licensee's proposed crud reduction program in the SFP and finds it acceptable.

The presently installed racks will be individually lifted from the SFP and while suspended over the SFP, will be rinsed using high pressure water to remove any loose radioactivity. The racks will then be moved to a receiving area for modification. The licensee has proposed decontaminating most of the components removed from the racks during the modification and then disposing the clean material as industrial waste. Material that cannot be decontaminated will be packed into drums and disposed of as normal radioactive waste. The disposal methods used will follow ALARA guidelines.

Divers will be used during the SFP rack modification. The licensee has developed specific procedures using the recommendations of Regulatory Guide 8.8 to ensure that doses to the divers will be within the requirements of 10 CFR Part 20 and ALARA guidelines. The ALARA procedures for divers include: reshuffling of the spent fuel to provide zones around the divers' work areas where no fuel will be stored; radiation survey after the fuel is reshuffled to map radiation zones; instruction to divers on their travel limits within the pool; and constant monitoring of divers' radiation dose by the use of remote readout dosimetry.

2.6.2 Conclusion

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

The staff has estimated the increment in onsite occupational dose during normal operations after the pool modification resulting from 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 SFP area, the staff estimates that the proposed modification should add less than one percent to the total annual occupational radiation exposure at the plant. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational dose to ALARA levels and within the limits of 10 CFR Part 20. Thus, the staff concludes that storing additional fuel in the SFP will not result in any significant increase in dose received by workers.

2.7 Radioactive Waste Treatment

The plant contains radioactive waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The radioactive waste treatment systems were evaluated in the SER for the full-term operating license dated October 1983 (NUREG-0944), in support of the issuance of Operating License No. DPR-18. There will be no change in the radioactive waste treatment systems or in the conclusions given regarding the evaluation of these systems because of the proposed modification of the SFP racks. The staff evaluation of the radiological considerations supports the conclusion that the proposed installation of new spent fuel storage racks at Ginna is acceptable because the conclusions of the evaluation of the radioactive waste treatment systems, as found in the Ginna SER for the full-term operating license, are unchanged by the modification of spent fuel storage racks.

2.8 Radiological Consequences of Accidents Involving Postulated Mechanical Damage to Spent Fuel

For evaluation of accidents involving the SFP, three types of accidents were considered; a cask drop or tip, a tornado missile impact, and a fuel assembly drop while handling fuel.

2.8.1 Cask Drop/Tip Accidents

Technical Specification 3.11.6 states that "The spent fuel shipping cask shall not be carried by the auxiliary building crane, pending the evaluation of the spent fuel cask drop accident and the crane design by RG&E, and NRC review and approval." Since the shipping cask cannot presently be carried by the auxiliary building crane by this administrative control, because the crane design evaluation has not yet been completed by the staff, a cask drop/tip accident is precluded for the proposed Technical Specification amendment.

2.8.2 Tornado Missile Accidents

The design values for tornado wind speed and missile characteristics were those established in the staff review of Systematic Evaluation Program (SEP) Topics III-2, Wind and Tornado Loadings, and III-4.A, Tornado Missiles. The design missile is stated to be a 1490 lb wooden pole, 35 feet in length and 13.5 inches in diameter, which could impact the racks with a vertical velocity of 70 ft/sec. The staff judges that the worst position for impact of this missile would be that centered on a fuel storage location where, because of the 13.5 inch missile diameter compared to a diagonal dimension of the spent fuel storage box of 11.9 inches, a total of nine fuel storage cells could be damaged in the reracked six sections of the SFP. This relative impact orientation of missile and storage cell configuration would have a low likelihood of occurrence, however. It is thus judged that a conservative estimate of damage to stored spent fuel assemblies from impact of the design missile is sufficient damage to nine assemblies in reracked pool sections, or two assemblies in the unreracked sections to result in the release of their concomitant volatile gap activities. In performing the accident radiological consequence analysis, it is assumed that the fuel has been discharged from the reactor after operation at a steady-state power level of 1551 MW_{th} for an extended period of time. The assumptions in the staff analysis are listed in Table 1 below. The calculated (0-2 hr) offsite accident radiological consequences are estimated to be 63 rem thyroid and 0.1 whole body at the Exclusion Area Boundary, for impact with unreracked assemblies. For impact with reracked assemblies, the corresponding offsite radiological consequences are 2 rem thyroid and 0.1 rem whole body. Both sets of consequences are well within the guidelines of 10 CFR Part 100.

Table 1: Assumptions in Staff Offsite Radiological Consequence Analysis of Postulated Tornado Missile Accident

	Unreracked Section	Reracked Section
Reactor Power Level	1551 MW_{th}	1551 MW_{th}
Effective Pool Decontamination Factor for Iodine	100	100
Radial Power Peaking Factor	1.2	1.2

	Unreracked Section	Reracked Section
Fuel Exposure for Impacted Spent Fuel Assembly	30,000 MWD/MTU	30,000 MWD/MTU
Number of Equivalent Impacted Spent Fuel Assemblies	2	9
Cooldown time for Impacted Spent Fuel Assembly	100 hr	60 d
Diffusion and Transport Atmospheric Relative Concentration, 0-2 hours, @ Exclusion Area Boundary	2.2×10^{-4} sec/m^3	2.2×10^{-4} sec/m^3
Filters	none assumed operational	none assumed operational

2.8.3 Fuel Handling Accident

In performing the radiological consequence analysis for the fuel handling accident, it was assumed that the fuel has been discharged from the reactor after operation at a steady-state power level of 1551 MW_{th} for an extended period of time. The assumptions in the staff analysis are listed in Table 2 below. The calculated (0-2 hr) offsite accident radiological consequences are estimated to be 44 rem thyroid and 0.1 rem whole body at the Exclusion Area Boundary, well within the guidelines of 10 CFR Part 100.

Table 2: Assumptions in Staff Offsite Radiological Consequences Analysis of Postulated Fuel Handling Accident

	Unreracked Section	Reracked Section
Reactor Power Level	1551 MW_{th}	1551 MW_{th}
Effective Pool Decontamination Factor for Iodine	100	100
Radial Power Peaking Factor	1.65	1.65
Fuel Exposure for Impacted Spent Fuel Assembly	30,000 MWD/MTU	30,000 MWD/MTU
Number of Equivalent Impacted Spent Fuel Assembly	1	1
Cooldown Time for Impacted Spent Fuel Assembly	100 hr	60 d

	Unreracked Section	Reracked Section
Diffusion and Transport Atmospheric Relative Concentration, 0-2 hours, @ Exclusion Area Boundary	2.2×10^{-4} sec/m^3	2.2×10^{-4} sec/m^3
Filters	none assumed operational	none assumed operational

2.8.4 Conclusion

Since the spent fuel shipping cask may not be carried by the auxiliary building crane, cask drop/tip accidents need not be considered.

The staff also concludes that a tornado missile accident resulting in damage to either two 30,000 MWD/MTU spent fuel assemblies in the unreracked pool section, or nine similar assemblies in the reracked sections, with at least 100 hours or 60 days of cooldown time, respectively, will result in atmospheric radionuclide releases with consequences which are well within the guidelines of 10 CFR Part 100.

Additionally, the staff concludes that a fuel handling accident resulting in damage to either a recently discharged 30,000 MWD/MTU spent fuel assembly in the unreracked pool area, or a more substantially decayed assembly in the reracked area, will result in atmospheric radionuclide releases which are well within the guidelines of 10 CFR Part 100.

The staff therefore concludes that the proposed modifications as acceptable.

3.0 OVERALL CONCLUSION

Based on the review, the staff concludes that the licensee's proposed SFP modification to increase the storage capacity of the SFP to 1016 fuel assemblies is acceptable. In addition, the proposed Technical Specifications are acceptable.

The staff concludes, 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 or to the health and safety of the public.

4.0 ACKNOWLEDGEMENT

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5.0 REFERENCES

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

2. D. M. Crutchfield (NRC) to J. E. Maier (RG&E), SUBJECT: Spent Fuel Pool Cooling System Modifications (Ginna), dated November 3, 1981.

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

TECHNICAL EVALUATION REPORT

EVALUATION OF SPENT FUEL RACKS STRUCTURAL ANALYSIS

ROCHESTER GAS AND ELECTRIC CORPORATION
R. E. GINNA NUCLEAR POWER PLANT

NRC DOCKET NO. 50-244

NRC TAC NO. 54762

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 26

FRC TASK 531

Prepared by

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Prepared for

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Washington, D.C. 20555

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October 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, and Aly A. Okaily.

1. INTRODUCTION

1.1 PURPOSE OF THE REVIEW

This technical evaluation report (TER) covers an independent review of the Rochester Gas and Electric Company licensing report [1] on high-density spent fuel racks for the R. E. Ginna Nuclear Station 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 Ginna 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:

- 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 the Ginna 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.

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* 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 - Standing inside a fuel cell, the fuel assembly repeatedly impacts the four inside walls of the cell under earthquake loadings. These impacts are nonlinear in nature and when componded with the hydrodynamic coupling effect will significantly affect the dynamic responses of the modules in seismic events.

b. Rack sliding on the 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 restrained horizontally by virtue of the frictional forces between the module base and the pool liner. The module will slide when these frictional forces are not large enough to overcome the horizontal seismic loads.

c. Vertical impact due to rack tipping - When the overturning moment generated by horizontal seismic loads becomes exceedingly large, some of the module supports may lift off momentarily from the pool liner. Although the rack tipping occurs in very short duration only, it will significantly affect the stress distribution of the module as well as the pool liner.

Only the six modules in Region 2, shown in Figure 1, are subjected to rerack modification [1]. All of these modules have identical cross-sectional dimensions, 84.3 in x 118.02 in. Modules having this design of nearly square cross sections generally behave in three-dimensional fashion in seismic events. Hence, the modules will exhibit three-dimensional nonlinear structural behavior under earthquake loadings, and all seismic analyses of modules should therefore focus on characterizing this behavior.

A time history analysis of the modules was performed by the Licensee using a special purpose computer program RACKOE [1]. The RACKOE model, shown in Figure 2, is a two-dimensional, nonlinear, finite element model representative of the module. Both OBE and SSE loading conditions were

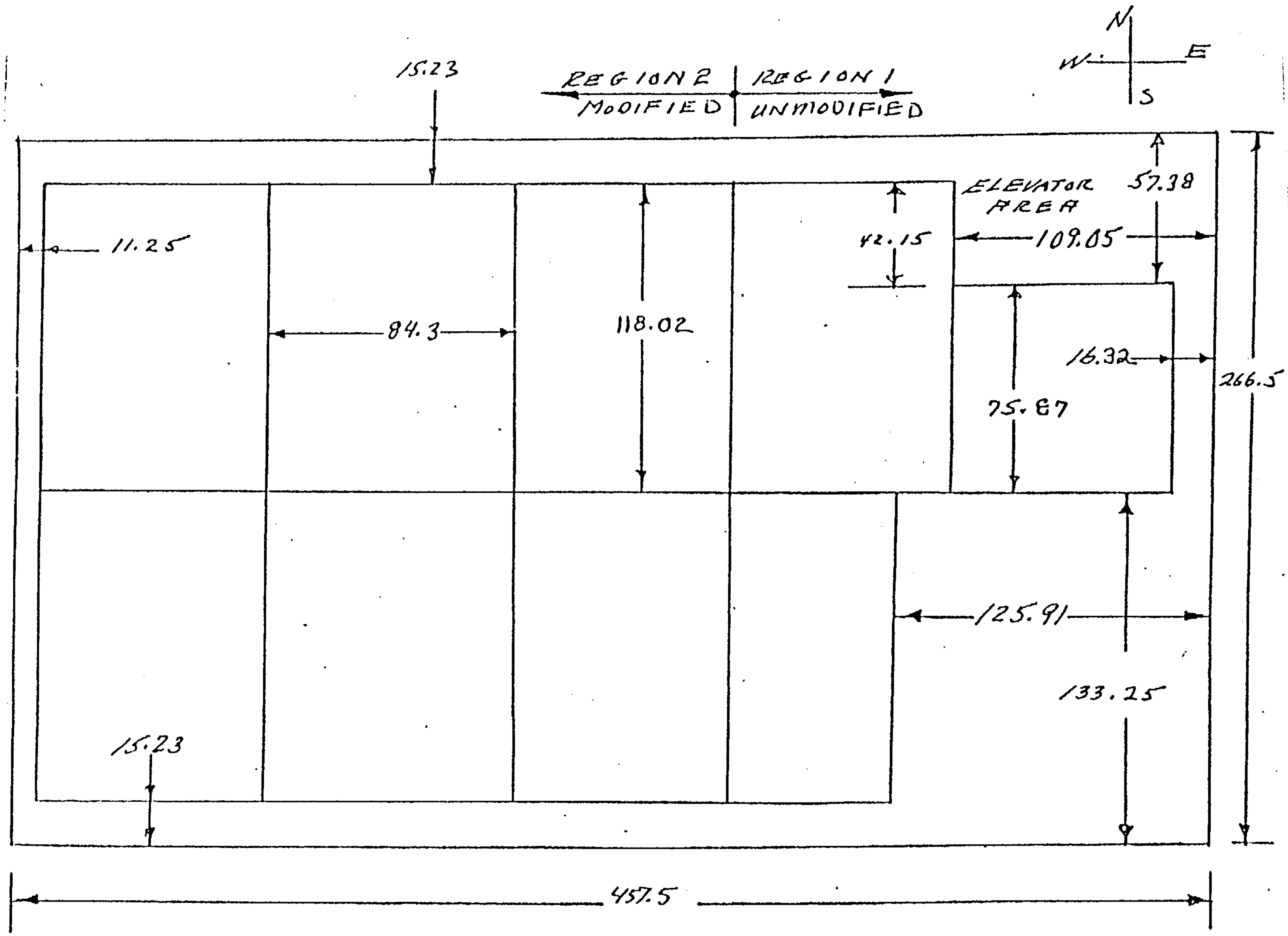

Figure 1. Pool Layout

evaluated by the Licensee. The OBE time history data was obtained by dividing the SSE time history data by two. The seismic analysis was performed for both the standard and the consolidated fuels.

The description and the evaluation of the RACKOE model are addressed in detail in Section 3.2. The displacement and stress results are discussed in appropriate subsections.

3.2 EVALUATION OF THE RACKOE MODEL

3.2.1 Description of the Model

A two-dimensional nonlinear model was developed in accordance with the special purpose finite element program, RACKOE. This program was designed to solve the equations of motion explicitly using Euler's extrapolation formula. A schematic view of the RACKOE model is shown in Figure 2.

The masses of the fuel cells and fuel assemblies are discretized in the RACKOE model. There are six concentrated mass nodes used to represent the fuel cells, with one node at the base of the module and the other five nodes at equal distance along the fuel cells. The nodes are linked by flexible elements. Similar arrangements are made to simulate fuel assemblies at five mass nodes. The mass nodes of fuel cells and fuel assemblies are connnected via (1) gap elements, to simulate impact between them, and (2) hydrodynamic masses, to represent hydrodynamic coupling between them. The friction between the module and the pool support stand is handled by friction elements which can only carry compressive loads. A horizontal spring is also used to represent frictional resistance of the module against sliding.

Separate analyses were conducted for the standard and the consolidated fuels. Individual analyses were performed for vertical and horizontal seismic loads. After determining the vertical natural frequency of the model to be greater than 33 Hz, an equivalent static response spectra method was used to perform the vertical seismic analysis. The horizontal seismic analysis was conducted using the time history method of analysis. Two different boundary conditions were considered in this analysis:

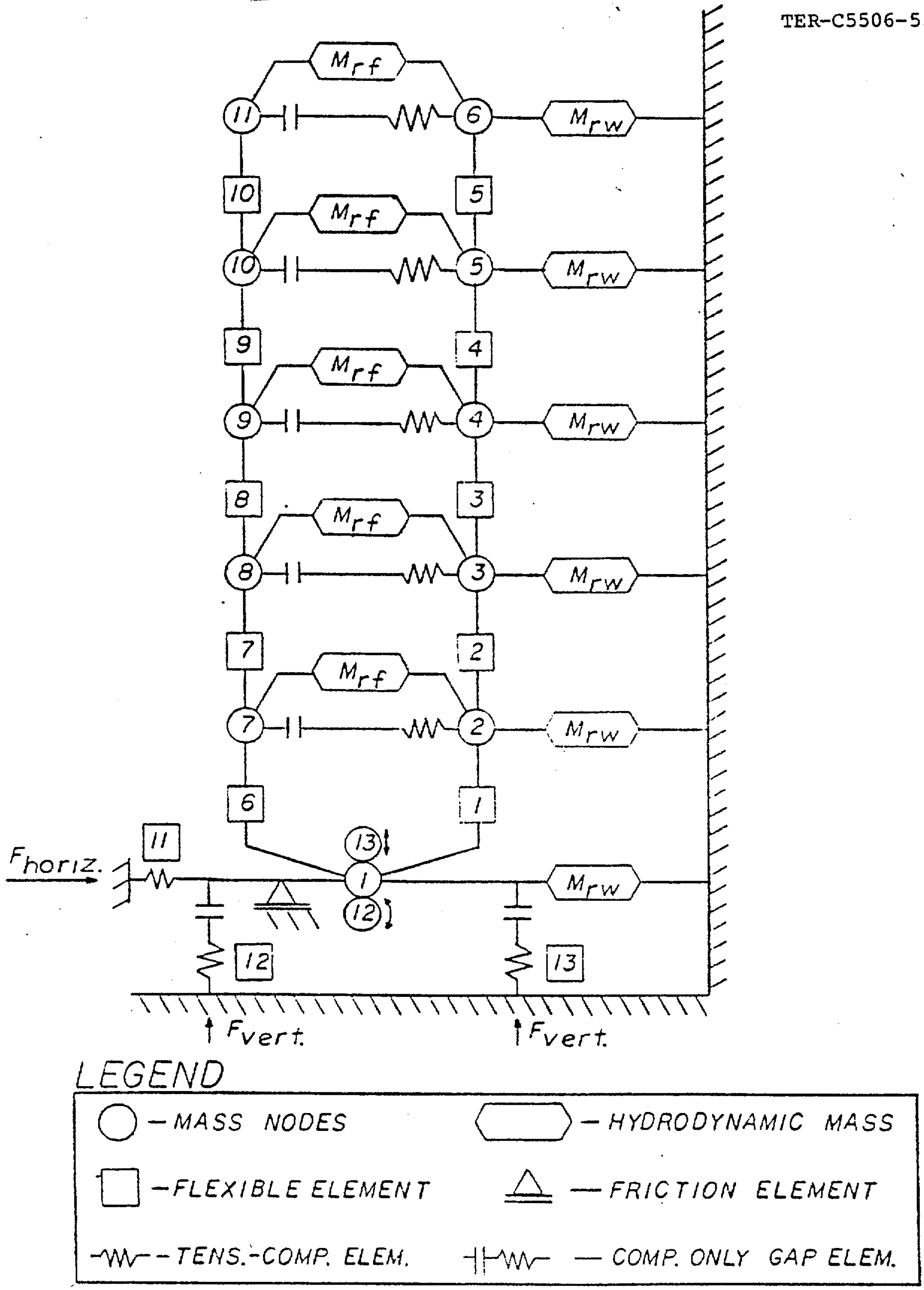

Figure 2. RACKOE Model of Spent Fuel Racks

1. The coefficient of friction between the module rack and the support stand is assumed to be 0.2. This is the minimum anticipated friction factor [3]. The results in this case will yield the maximum distance the module may slide in a seismic event.

2. The differential motion between the module base and the support stand is prevented. This boundary condition corresponds to the case when the coefficient of friction is greater than 0.5 [4]. Maximum stresses will be developed in the model in this case.

Different horizontal seismic analyses were performed for the east-west and north-south directions to account for the differences in structural configuration of the modules and seismic loadings in these two directions. The final results were obtained by combining the vertical reaction loads with the horizontal seismic loads using the square root of the sum of the squares method.

3.2.2 Assumptions Used in the Analysis

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

a. The damping values used for this analysis were taken from Regulatory Guide 1.61 [5]. They are 2% for OBE and 4% for SSE events.

b. Only a constant value of friction coefficient was considered in each seismic analysis. The coeffcient of friction remained unchanged whether the module was stationary or in motion.

c. Adjacent modules would move in phase in seismic events.

d. The modules were installed very deeply in the fuel pool. Consequently, the sloshing effect is negligible.

The assumption in Item c may be valid when adjacent modules are identically loaded, but an out-of-phase response will most likely occur for differently loaded modules, either empty, partially, or fully loaded.

3.2.3 Impact Between Adjacent Modules

The pool layout shown in Figure 1 indicated that there is no gap between adjacent modules in the pool. The Licensee stated that, because of the strong hydrodynamic coupling effects in the case of no gap, adjacent modules were forced to vibrate in phase, thus precluding any impact between adjacent modules [5]. This claim is generally true for identically loaded modules, but out-of-phase vibration will most likely occur when the modules are loaded in different patterns, either empty, partially, or fully loaded. The out-of-phase motion will probably result in some form of impact between adjacent modules. In light of this probability, an impact analysis is needed in order to demonstrate that the impact does not cause any damage to the module structure or its contents [2].

The Licensee responded [5] by performing an impact analysis. The RACKOE model shown in Figure 2 was modified to include the compression-only springs at the top of the module to represent the presence of the adjacent module. The compressive force obtained in these springs was used to calculate the impact area on the wall of fuel cells based on the allowable compressive stress requirement. The length of wall required to resist the compressive force was calculated to the 16.0 inches in the east-west direction and 16.9 inches in the north-south direction. These impact wall lengths are acceptable since there is not much space between adjacent modules. The Licensee also demonstrated that the impact between adjacent modules would not adversely affect the stress distribution within the module structure [6].

3.2.4 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 response of the module in seismic events. The Licensee applied the linear model of Fritz [7] to estimate these coupling effects. This modeling technique assumes that the hydrodynamic coupling mass between two vibratory structures is inversely proportional to the gap between them. This assumption will generate an infinite coupling mass when there is no gap between adjacent modules. In light of this virtual impossibility, a 1-in gap was assumed between adjacent modules in evaluating the coupling mass between them. This approach is more realistic and also serves a conservative purpose.

Fritz's [7] 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 [7] when employed within the conditions upon which it was based, that of vibratory displacements which are very small compared with 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 only 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.

The limitations of Fritz's [7] modeling technique for hydrodynamic coupling of fuel assemblies within a rack cell, and of rack modules adjacent to other rack moduels or a pool wall, would indicate that the Licensee's fuel rack hydrodynamic coupling is accurate only for dynamic displacements that are small relative to the available displacement clearance.

3.2.5 Solution Stability and 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. The study was conducted using consolidated fuel model with maximum friction in the north-south direction for the SSE condition [3]. The following results were obtained for three time steps: 0.001, 0.0005, and 0.00025 second.

	Time Step (sec)		
	0.001	0.0005	0.00025
Max. Vertical Reaction (lb)	549,000	456,000	427,000
Max. Horizontal Reaction (lb)	293,000	293,000	240,000
Max. Vertical Liftoff (in)	0.042	0.017	0.05

The time step of 0.0005 second was chosen to be used throughout the seismic analysis.

3.2.6 Liftoff Analysis

A liftoff analysis was performed by the Licensee to study the effect of the liftoff of module upon the stress distribution within the module structure. A modified RACKOE model, shown in Figure 3, was used in this analysis [8]. This simplified model used a single mass to simulate the module and its contents. This approach basically assumes a stiff beam to represent the entire module. This assumption is reasonably valid because the module is very stiff in the vertical direction. Furthermore, the Licensee demonstrated that identical results were obtained from a model containing five concentrated mass nodes to represent the module structure and its contents [6].

3.2.7 Displacement and Stress Results

For the operating life of the plant, the Licensee predicted that the maximum distance that the modules can slide is 0.95 in [4]. The closest obstruction, excluding adjacent modules, is the west wall which has an installed gap of 11.25 in. Consequently, the module sliding and tilting will not impact the pool wall. Since there is no gap between adjacent modules, this predicted sliding of modules will probably cause some form of impact between adjacent modules. As discussed in Section 3.2.3, an impact analysis was performed to insure that no damage was caused by the impact to the module structure and its contents.

During the review, the Licensee submitted a revised stress analysis report [9] providing detailed analyses of stress in the spent fuel rack module. While the stress report [9] incorrectly addressed acceptance criteria based upon Appendix D to Section 3.8.4 of NRC's Standard Review Plan, the report's transmittal letter [10] referenced a separate stress summary that compared the rack's stresses to the correct acceptance criteria provided by the OT Position Paper [2]. This separate stress summary, comparing calculated stresses to allowable values, indicated that the maximum design margins for base metal and weld stresses are greater than 0.47 for standard fuel and 0.25 for consolidated fuel. A detailed review of the stress report indicated that the methodology and level of stresses are satisfactory.

3.2.8 Eccentrically Loaded Modules

The Licensee allowed the modules to be eccentrically loaded as the situation demanded. An analysis was performed by the Licensee to study the

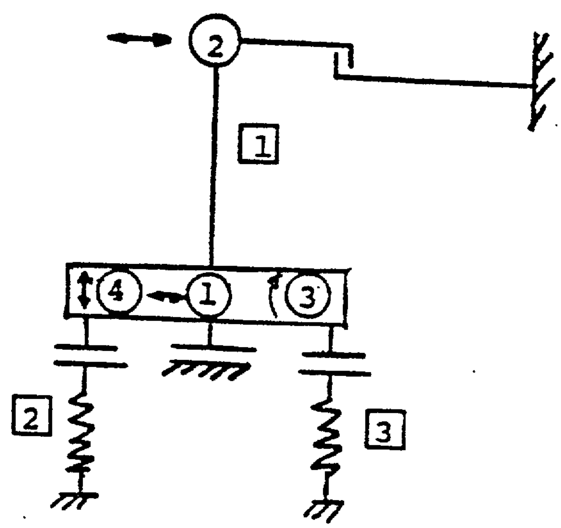

Figure 3. Simplified RACKOE Model for Liftoff Analysis

effect of such loading configurations upon the stress distribution within the module structure [4]. The RACKOE model was modified by inputting a different stiffness matrix of pedestals to reflect the eccentric loading pattern. The Licensee identified the worst eccentric loading case as when the module was loaded with two rows of consolidated fuel and subjected to seismic excitations in the east-west direction. The analysis results showed that this loading configuration produced a slightly greater liftoff distance of the pedestal than a fully loaded module, but it yielded a lower horizontal seismic load, vertical pedestal reaction, and horizontal displacement of the module top than did the fully loaded module.

3.3 REVIEW OF SPENT FUEL POOL STRUCTURAL ANALYSIS

The Licensee's justification for not performing a structural analysis of the spent fuel pool under the anticipated increased loads of the modified spent fuel storage racks is based on the following:

a. For the pool walls, the overall loads are reduced significantly compared to the original design loads due to removal of the seismic restraint supports. Meanwhile, there are relatively large dimensions between the free-standing racks and the walls compared to the maximum sliding distance of 0.5 in which consequently cause only very small hydrodynamic forces.

b. The floor of the spent fuel pool is a stainless steel lined, 3-ft-thick, reinforced concrete slab. The slab is founded on bedrock (Ginna FSAR, Section 2.8.3). The structure of the pool was evaluated for the original FSAR and again for the higher loads associated with a subsequent rack replacement (Reference 1 of April 2, 1984).

c. Because the rack will be modified to a free-standing design, only the increased concrete bearing stresses of the floor were evaluated. These were found to be acceptable (maximum concrete bearing stress is 2337 psi and the allowable is 3570 psi).

3.4 REVIEW OF HIGH DENSITY FUEL STORAGE RACKS' DESIGN

With respect to an accidental drop of a fuel assembly from above the rack module and through a rack cell, the Licensee stated [9] that the impact of the fuel assembly on the fuel support plates for that cell would damage it so that the particular cell could not be used for storage of spent fuel until repairs were completed. The Licensee indicated that spent fuel in other cells would not be adversely affected.

The Licensee assured that the spent fuel pool liner would not be perforated as follows [11]:

> "We have determined the fuel assembly velocity required to perforate the stainless steel liner using methodology developed for tornado missile impact analysis. Using the submerged weight of a fuel assembly dropped from 30 inches above the top of the rack, but neglecting all drag forces due to water or impact with cell walls or bottom plate, the velocity of the fuel assembly on impact is not sufficient to perforate the liner."

4. CONCLUSIONS

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

- The Licensee's analysis assumes that the fuel rack modules are positioned within the spent fuel pool without clearance space between the modules. Without clearance, the rack modules will impact to some extent. However, an impact analysis indicated that stresses associated with impacting are satisfactory.

- The review of the Licensee's stress analysis indicated that the analysis and level of stresses are acceptable.

- The review of the spent pool structure is satisfactory for the higher density fuel loading.

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