50-244

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 6, 1996

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: ISSUANCE OF AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M92188)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 60 to Facility Operating License No. DPR-18 for R.E. Ginna Nuclear Power Plant (Ginna). This amendment is in response to your application dated May 26, 1995, as supplemented May 5, 1995, and January 26, 1996. By letter dated May 26, 1995, you submitted a request for changes to the R. E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TSs) to revise the TSs in its entirety by converting to improved Standard Technical Specifications (STS), and other changes. Among the other changes requested in this letter was a proposal to allow the storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) Uranium-235 (U-235) in the new (fresh) and spent fuel storage racks and change the license to reflect changes related to the nuclear fuel cycle. You had previously submitted by letter dated May 5, 1995, an analysis to support this request. The NRC staff reviewed your analysis and found it acceptable. By letter dated August 30, 1995, the NRC staff transmitted the related fuel enrichment safety evaluation (SE) to you, and this SE, in its entirety, is enclosed.

By letter dated January 26, 1996, you requested that this portion of the improved STS amendment, which is presently under review by the NRC staff, be treated as an exigent amendment request owing to the imminent delivery of enriched fuel to Ginna. However, this request does not meet the requirements for an exigent amendment. Therefore, the request for exigency is denied, but the amendment is granted.

CP-1

A copy of the related SE is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Allen R. Johnson

Allen R. Johnson, Project Manager
Project Directorate I-1
Division of reactor Projects I/II
Office of Nuclear Reactor Regulations

Docket No. 50-244

Enclosures: 1. Amendment No. 60 to License No. DPR-18
2. Safety Evaluation

cc: See next page

A copy of the related SE is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

ORIGINAL SIGNED BY:

Allen R. Johnson, Project Manager
Project Directorate I-1
Division of reactor Projects I/II
Office of Nuclear Reactor Regulations

Docket No. 50-244

Enclosures: 1. Amendment No. 60 to License No. DPR-18
2. Safety Evaluation

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
License No. DPR-18

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

   A. The application for amendment filed by the Rochester Gas and Electric Corporation (the licensee) dated May 26, 1995, as supplemented May 5, 1995, and January 26, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 60 , 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Page 2 of License No. DPR-18
2. Changes to the Technical Specifications

Date of Issuance: February 6, 1996

ATTACHMENT 1 TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise License DPR-18 as follows:

Remove	Insert
2	2

facility") which is owned by the Rochester Gas and Electric Corporation (hereinafter "the licensee" or "RG&E". The facility is located on the licensee's site on the south shore of Lake Ontario, Wayne County, New York, about 16 miles east of the City of Rochester and is described in license application Amendment No. 6, "Final Facility Description and Safety Analysis Report," and subsequent amendments thereto, and in the application for power increase notarized February 2, 1971, and Amendment Nos. 1 through 4 thereto (herein collectively referred to as "the application").

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses RG&E:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the facility at the designated location in Wayne County, New York, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material or reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as amended, and Commission Safety Evaluations dated November 15, 1976, October 5, 1984, November 14, 1984, and August 30. 1995.

(a) Pursuant to the Act and 10 CFR Part 70, to receive and store four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979);

(b) Pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979), as supplemented February 20, 1980 and March 5, 1980;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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

Remove	Insert
5.4-1	5.4-1
5.4-2	5.4-2
5.4-3	5.4-3
5.4-4	5.4-4
5.4-5	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 ≤ 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 between January 1, 1984 and February 1, 1996 containing no more than 41.9 gms U-235 per axial cm, and (3) unirradiated fuel assemblies delivered after Feb. 1, 1996 containing no more than 49.8 grams U-235 per axial cm. All 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 that particular fuel assembly design, therefore assuring that Keff ≤ 0.95.

5.4.4 Canisters containing consolidated fuel rods may be stored in either Region 1 or 2 provided that:

a. the average burnup and initial enrichment of the fuel assemblies from which the rods were removed satisfy the requirements of 5.4.2 and 5.4.3 above, and

b. the average decay heat of the fuel assembly from which the rods were removed is less than 2150 BTU/hr

5.4.5 The requirements of 5.4.4a may be excepted for those consolidated fuel assemblies of Region RGAF2.

5.4.6 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.

<u>Basis</u>

The center to center spacing of Region 1 insures that Keff $\leq$ 0.95 for the enrichment limitations specified in 5.4.2[1,6], and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100[2]. Fuel assemblies with an enrichment of $\leq$ 4.05 w/o can be stored in any available location. Fuel assemblies with an enrichment > 4.05 w/o can also be stored in Region I provided that integral burnable poisons are present in the assemblies such that k-infinity is $\leq$ 1.458.

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 except Exxon fuel delivered prior to January 1, 1984, which incorporates all Westinghouse HIPAR designs used at Ginna.[4] The second curve is for all Exxon fuel, as well as 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 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 record of these calculations shall be kept for as long as fuel assemblies remain in the pool.

The fuel storage canisters are designed so that, normally, they can contain the equivalent number of fuel rods from two fuel assemblies in a close packed array, and can be stored in either Region 1 or Region 2 rack locations. The close packed array will insure the K_∞ of the rack configuration containing any number of canisters will be less than that for stored fuel assemblies at the same burnup and initial enrichment. The exception of paragraph 5.4.5 is possible because the consolidated configuration is substantially less reactive than that of a fuel assembly. The maximum decay heat requirement will ensure that local and film boiling will not occur between the close packed fuel rods

if the pool temperature is maintained at or below 150°F. The decay heat of the assembly will be determined using ANS 5.1, ASB 9-2 or other acceptable substitute standards.

With the addition of the storage of consolidated fuel canisters, the theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2x1016). Moreover, due to limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1016 fuel assemblies.[5]

References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Safety Evaluation from John Zwolinski to Roger Kober, November 14, 1984, "Increase of the Spent Fuel Pool Storage Capacity."
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.
5. Letter, D.M. Crutchfield to J.E. Maier, November 5. 1981.
6. Safety Evaluation from Allen Johnson to Dr. Robert C. Mecredy, August 30. 1995, "Proposed Criticality Analysis of Ginna New and Spent Fuel Racks/Consolidated Rod Storage Canisters.

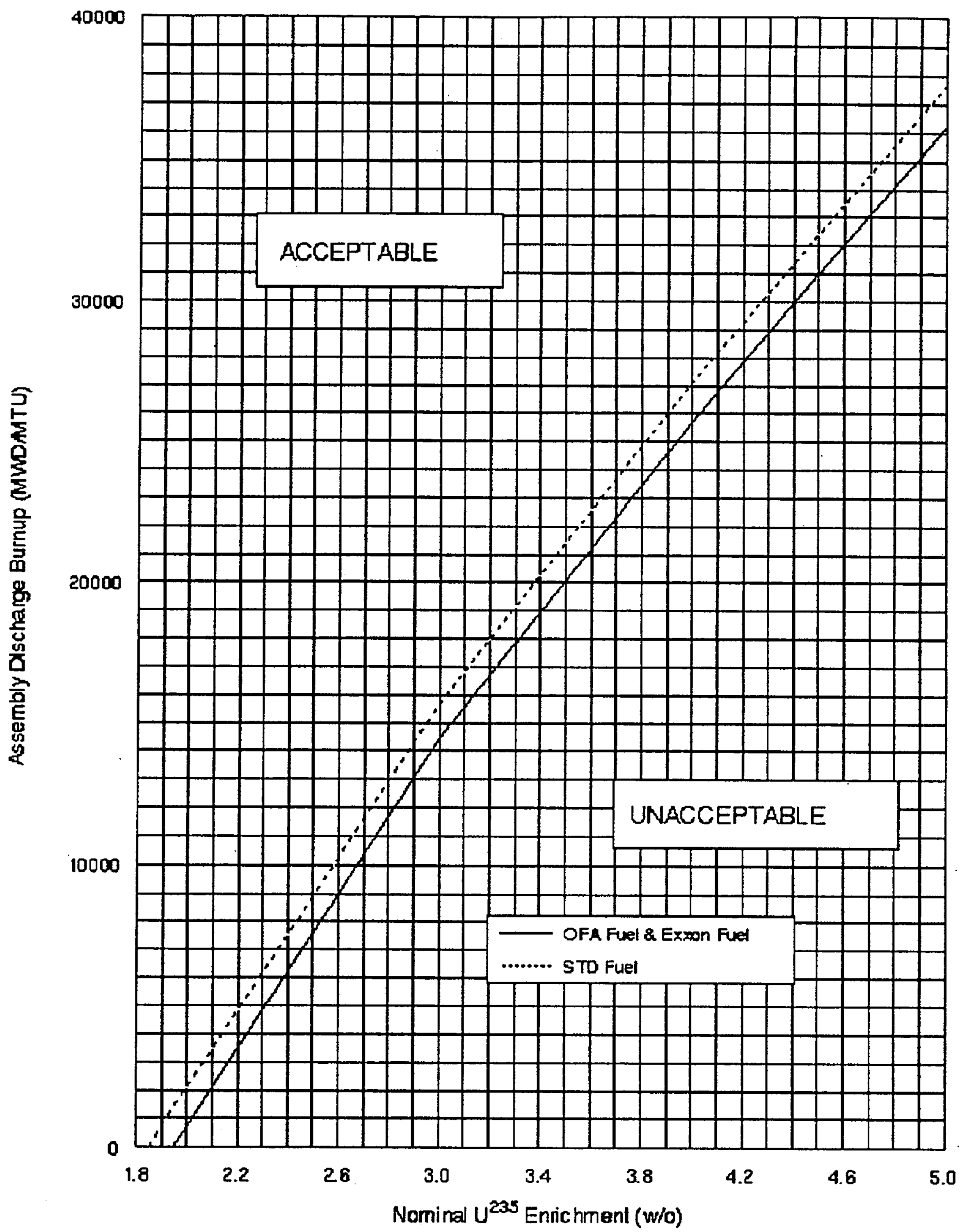

Figure 5.4-2
Fuel Assembly Burnup Limits in Region 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated May 26, 1995, Rochester Gas and Electric Corporation (RG&E or the licensee) submitted a request for changes to the R. E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TSs) to revise the TSs in its entirety by converting to improved Standard Technical Specifications (STS), and other changes. Among the other changes requested in this letter was a proposal to allow the storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) Uranium-235 (U-235) in the new (fresh) and spent fuel storage racks and change the license to reflect changes related to the nuclear fuel cycle. The licensee had previously submitted by letter dated May 5, 1995, an analysis to support this request. The NRC staff reviewed the licensee's analysis and found it acceptable. By letter dated August 30, 1995, the NRC staff transmitted the related fuel enrichment safety evaluation (SE) to the licensee, and this SE is included in its entirety, below.

By letter dated January 26, 1996, the licensee requested that this portion of the improved STS amendment, which is presently under review by the NRC staff, be treated as an exigent amendment request owing to the imminent delivery of enriched fuel to Ginna. However, this request does not meet the requirements for an exigent amendment. Therefore, the request for exigency is denied, but the amendment is granted.

2.0 EVALUATION

The analysis of the reactivity effects of fuel storage in the new and spent fuel storage racks was performed with the three-dimensional multi-group Monte Carlo KENO-5a (computer code), using neutron cross sections generated by the NITAWL (computer code) package from the 227 energy group library. Since the KENO-5a code package does not have depletion capability, burnup analyses were performed with the two-dimensional transport theory, PHOENIX (computer code). PHOENIX was also used to determine the reactivity effects of material and manufacturing tolerances. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Ginna fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber worth. The intercomparison between two independent methods of analysis (KENO-5a and

PHOENIX) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a reactivity calculations, 270,000 neutron histories were typically accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO-5a reactivity calculations. Based on the above, the NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Ginna new and spent fuel storage racks with a high degree of confidence.

The fresh fuel storage vault is intended for the receipt and storage of fresh fuel under dry (air) conditions. However, to assure the criticality safety under normal and accident conditions and to conform to the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling, two separate criteria must be satisfied as defined in NRC Standard Review Plan (SRP), Section 9.1.1. These criteria state that the maximum reactivity of the fully loaded fuel racks shall not exceed a K_{eff} (effective multiplication factor) of 0.95 if fully flooded with unborated water or a K_{eff} of 0.98 assuming the optimum hypothetical low density moderation (e.g., fog or foam). The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true K_{eff} will not exceed the calculated maximum value at a 95% probability, 95% confidence level (95/95).

The maximum K_{eff} for a fully loaded vault of Westinghouse Optimized Fuel Assembly (OFA) fuel enriched to 5.0 w/o U-235 was calculated to be 0.9146 under fully flooded conditions. For the hypothetical low-density optimum moderation condition, the maximum calculated K_{eff} was 0.6666 at a moderator density of approximately 6% of full density for a fully loaded vault of OFA fuel. The calculations included a calculational bias and uncertainty derived from benchmark calculations, as well as uncertainties due to KENO-5a statistics, cell wall thickness and fuel enrichment at the 95/95 probability/confidence level. The results conform to the acceptance criteria of SRP 9.1.1 and are, therefore, acceptable.

The storage racks in the spent fuel pool are divided into two regions. Region 1 contains 351 stainless steel storage cells spaced 8.43 inches apart and contains no Boraflex or other neutron absorber. Region 2 consists of 840 storage cells and contains 0.075-inch thick Boraflex panels. The cells are also stainless steel arranged on a 8.43-inch center-to-center spacing. The spent fuel racks are normally fully flooded by borated water. However, to meet the criterion stated in Section 9.1.2 of the NRC SRP, K_{eff} must not exceed 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true K_{eff} will not exceed 0.95 at a 95/95 probability/confidence level.

Initial calculations for Region 1 have shown that OFA fuel was the most reactive type. The spent fuel storage racks in Region 1 were evaluated for 4.0 w/o U-235 enriched fuel moderated by pure water at 68 °F with a density of 1.0 gm/cc. The fuel assemblies were arranged in a two-out-of-four checkerboard pattern. For the nominal storage cell design in Region 1, uncertainties due to tolerances in fuel enrichment and density, fuel pellet dishing, storage cell I.D. and pitch, and stainless steel thickness were accounted for as well as eccentric fuel positioning. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, calculational and methodology biases and uncertainties due to benchmarking and water temperature range were included. The resulting K_{eff} was 0.9487, meeting the 0.95 acceptance criterion.

As an alternative method for determining the acceptability of fuel storage in the Region 1 racks, the K_e (infinite multiplication factor) of a nominal fresh 4.0 w/o U-235 fuel assembly in the Ginna core geometry was determined to be 1.458. Therefore, fuel with a reference K_e no greater than 1.458 can be stored in a checkerboard configuration in Region 1 and meet the 0.95 rack reactivity acceptance criterion.

To enable the storage of fuel assemblies with nominal enrichments greater than 4.0 w/o U-235, the concept of reactivity equivalencing was used. In this technique, which has been previously approved by the NRC, credit is taken for the reactivity decrease due to the integral fuel burnable absorber (IFBA) material coated on the outside of the UO_2 (Uranium oxide) pellet. Based on these calculations, the reactivity of the fuel rack array, when checkerboarded with fuel assemblies enriched to 5.0 w/o U-235 with each containing 64 IFBA rods, was found to be equivalent to the rack reactivity when checkerboarded with 4.0 w/o fuel with no IFBA rods. The calculation assumed the standard IFBA patterns used by Westinghouse with the minimum standard loading of 1.675 mg/inch of Boron-10 per rod. Since the worth of individual IFBA rods can change depending on position within the fuel assembly, additional margin was included in the IFBA requirement to account for this. In addition, the IFBA requirements also include a 10% margin on the total number of IFBA rods for 5.0 w/o enriched assemblies to account for calculational uncertainties. The staff concludes that the IFBA requirement calculations contain sufficient conservatism to account for manufacturing and calculational uncertainties.

The Region 2 spent fuel storage racks were analyzed for storage of Westinghouse 14x14 OFA fuel assemblies with nominal enrichments up to 1.95 w/o U-235, and Westinghouse 14x14 STD (standard) assemblies with nominal enrichments up to 1.85 w/o U-235. The same initial assumptions, biases and uncertainties as used for the Region 1 analyses were included, except for the effects of Boraflex shrinkage and gaps.

Since the Region 2 racks contain Boraflex, the reactivity calculations also considered the effects of Boraflex shrinkage and gap formation. All Boraflex panels were modeled with 4% shrinkage. Five different scenarios were examined ranging from all of the Boraflex panels experiencing random gap formation to all of the panels experiencing shrinkage from the bottom end. Since the

bottom end results in more active fuel exposure than the top end, the latter assumption is conservative. The worst-case assumption was where 75% of the panels experience non-uniform shrinkage (random gaps) and the remaining 25% of the panels experience uniform shrinkage (pull-back) from the bottom end. This scenario was used to perform the criticality analysis for Region 2. Based on the results of blackness testing performed at other storage facilities, and on upper bound values recommended by Electric Power Research Institute (EPRI), the staff concurs that these assumptions bound the current measured data and future development of additional shrinkage and gaps. The final Region 2 design, when fully loaded with Westinghouse OFA fuel enriched to 1.95 w/o U-235, resulted in a K_{eff} of 0.9469 when combined with all known uncertainties. This meets the staff's criterion of K_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

In order to store Westinghouse 14x14 OFA as well as Westinghouse STD and Exxon 14x14 assemblies with enrichments up to 5.0 w/o U-235, the concept of burnup credit reactivity equivalencing was used. This is predicted upon the reactivity decrease associated with fuel depletion and has been previously accepted by the staff for spent fuel storage analysis. For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment versus fuel assembly discharge burnup ordered pairs which all yield an equivalent K_{eff} less than 0.95 when stored in the spent fuel storage racks. The results indicate that a fresh OFA 1.95 w/o enriched fuel assembly yields the same rack reactivity as an initially enriched 5.0 w/o OFA depleted to approximately 36,200 MWD/MTU. Since the Westinghouse 14x14 STD is more reactive than the Westinghouse 14x14 OFA fuel at the low enrichment limit for the Region 2 rack, a separate burnup credit curve was determined for the STD fuel. In addition, since the Exxon 14x14 fuel is less reactive than the Westinghouse OFA, the Westinghouse 14x14 OFA burnup credit curve may be used for the Exxon 14x14 fuel. A reactivity uncertainty of 0.0121 K (reactivity increase) was applied to the burnup credit curves. This is consistent with current practice and is acceptable.

To allow for possible future storage of Ginna fuel rods in consolidated rod storage canisters (CRSC), analyses were performed to determine the acceptable range of the number of consolidated rods. The fuel rods were assumed to be randomly dispersed in the canister and the same uncertainties and biases used for the Region 2 rack analysis were applied. The results indicate that an acceptable range of the number of consolidated rods is no greater than 144 or no less than 256 rods. The storage of a canister which contains between 144 and 256 consolidated rods is not acceptable.

Most abnormal storage conditions will not result in an increase in the K_{eff} of the racks. However, it is possible to postulate events, such as the misloading of an assembly with an enrichment and burnup (or IFBA) combination outside of the acceptable area or pool temperatures exceeding 180 °F in Region 1 (heatup event) or decreasing below 50 °F in Region 2 (cooldown event), which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of boron in the pool water

required by proposed improved Technical Specification 3.7.12 (current Technical Specification 5.4), since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in K_{eff} caused by 300 ppm of boron is sufficient to mitigate the worst postulated accident in any pool region. Therefore, the staff criterion of K_{eff} no greater than 0.95 for any postulated accident is met.

3.0 CONCLUSION

Based on the review described above, the NRC staff finds the criticality aspects of the proposed enrichment increase to the Ginna new and spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the above-mentioned fuel is acceptable for storage in the Ginna fuel storage racks, evaluations of reload core designs (using any enrichment) will be necessary, to be performed on a cycle by cycle basis, as part of the reload safety evaluation process. Each reload design is to be evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and Technical Specifications to ensure that reactor operation is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on January 23, 1996 (61 FR 1785). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment

6.0 SUMMARY

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Laurence Kopp

Date: February 6, 1996