July 30, 1998

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M95759)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998, and May 22, 1998.

This amendment changes the Technical Specifications to accommodate the modification of the spent fuel pool by replacing the three Region 1 rack modules with seven new borated stainless steel rack modules scheduled for implementation in 1998. Six new peripheral modules would be added at some future date. Two of the seven new modules planned to be installed in 1998 are to be designated as part of Region 2, effectively increasing the Region 2 area. The other five new modules compose Region 1, resulting in a total of 294 storage positions in Region 1. Region 2, with 1075 storage positions, consists of three rack types, Type 1, Type 2, and Type 4. Type 1 cells are the Boraflex cells that form Region 2 for the existing license. Two racks of Type 2 cells, containing borated stainless steel (BSS) absorber plates are be added to increase the storage capacity of Region 2. In addition, the capacity of Region 2 could be increased in the future by the addition of Type 4 racks, which also contain BSS absorber plates. The amendment increases the boron concentration from 300 ppm to 2300 ppm.

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

Sincerely, Original Signed by:
Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 72 to License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: See next page

DFO1

DOCUMENT NAME: G:\GINNA\M95759.AMD *See previous concurrence
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PDI-1	E	LA:PDI-1		D:PDI-1		SRXB*		EMEB*		EMBC*		OGC*	
NAME	GVissing:lcc		SLittle		SBajwa		TCollins		RWessman		ESullivan		AHodgdon	
DATE	07/31/98		07/15/98		07/17/98		06/22/98		06/29/98		06/24/98		07/02/98	

Official Record Copy

9808030279 980730
PDR ADOCK 05000244
P PDR

DATED: July 30, 1998

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-18-GINNA NUCLEAR POWER PLANT

Docket File
PUBLIC
PDI-1 Reading
J. Zwolinski (A)
S. Bajwa
S. Little
G. Vissing
OGC
G. Hill (2), T-5 C3
W.Beckner, 0-11/E/22
ACRS
L. Kopp, 0-8E23
K. Parczewski, 0-7D4
C. Hinson, 0-12H2
D. Shum, 0-8D1
Y. Kim, 0-7H15
B. Thomas, 0-8D1
C. Hehl, Region I
J. Linville, Region I

cc: Plant Service list

DATED: July 30, 1998

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-18-GINNA NUCLEAR POWER PLANT

Docket File
PUBLIC
PDI-1 Reading
J. Zwolinski (A)
S. Bajwa
S. Little
G. Vissing
OGC
G. Hill (2), T-5 C3
W.Beckner, 0-11/E/22
ACRS
L. Kopp, 0-8E23
K. Parczewski, 0-7D4
C. Hinson, 0-12H2
D. Shum, 0-8D1
Y. Kim, 0-7H15
B. Thomas, 0-8D1
C. Hehl, Region I
J. Linville, Region I

cc: Plant Service list

July 30, 1998

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M95759)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998, and May 22, 1998.

This amendment changes the Technical Specifications to accommodate the modification of the spent fuel pool by replacing the three Region 1 rack modules with seven new borated stainless steel rack modules scheduled for implementation in 1998. Six new peripheral modules would be added at some future date. Two of the seven new modules planned to be installed in 1998 are to be designated as part of Region 2, effectively increasing the Region 2 area. The other five new modules compose Region 1, resulting in a total of 294 storage positions in Region 1. Region 2, with 1075 storage positions, consists of three rack types, Type 1, Type 2, and Type 4. Type 1 cells are the Boraflex cells that form Region 2 for the existing license. Two racks of Type 2 cells, containing borated stainless steel (BSS) absorber plates are be added to increase the storage capacity of Region 2. In addition, the capacity of Region 2 could be increased in the future by the addition of Type 4 racks, which also contain BSS absorber plates. The amendment increases the boron concentration from 300 ppm to 2300 ppm.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,
Original Signed by:
Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 72 to License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION</u>: See next page

DOCUMENT NAME: G:\GINNA\M95759.AMD *See previous concurrence

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PDI-1	E	LA:PDI-1		D:PDI-1		SRXB*		EMEB*		EMBC*		OGC*	
NAME	GVissing:lcc		SLittle		SBajwa		TCollins		RWessman		ESullivan		AHodgdon	
DATE	07/31/98		07/15/98		07/17/98		06/22/98		06/29/98		06/24/98		07/02/98	

Official Record Copy

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

July 30, 1998

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M95759)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998, and May 22, 1998.

This amendment changes the Technical Specifications to accommodate the modification of the spent fuel pool by replacing the three Region 1 rack modules with seven new borated stainless steel rack modules scheduled for implementation in 1998. Six new peripheral modules would be added at some future date. Two of the seven new modules planned to be installed in 1998 are to be designated as part of Region 2, effectively increasing the Region 2 area. The other five new modules compose Region 1, resulting in a total of 294 storage positions in Region 1. Region 2, with 1075 storage positions, consists of three rack types, Type 1, Type 2, and Type 4. Type 1 cells are the Boraflex cells that form Region 2 for the existing license. Two racks of Type 2 cells, containing borated stainless steel (BSS) absorber plates are be added to increase the storage capacity of Region 2. In addition, the capacity of Region 2 could be increased in the future by the addition of Type 4 racks, which also contain BSS absorber plates. The amendment increases the boron concentration from 300 ppm to 2300 ppm.

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

Sincerely,

Guy S. Vissing

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 72 to License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

cc:

Peter D. Drysdale, Sr. Resident Inspector
R.E. Ginna Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, NY 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Lalw
120 Broadway
New York, NY 10271

Nicholas S. Reynolds
Winston & Strawn
1400 S Street N.W.
Washington, DC 20005-3502

Ms. Thelma Wideman, Director
Wayne County Emergency Management
Office
Wayne County Emergency Operations Center
7336 Route 31
Lyons, NY 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
111 West Falls Road, Room 11
Rochester, NY 14620

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

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. 72
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 March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1998, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998 and May 22, 1998 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.72 , 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

S. Bajwa

S. Singh Bajwa, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 30, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 72

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
3.7-27	3.7-27
3.7-28	3.7-28
3.7-29	3.7-29
3.7-30	3.7-30
3.7-31	3.7-31
	3.7-31a
4.0-2	4.0-2
4.0-3	4.0-3

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.12 The SFP boron concentration shall be ≥ 2300 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP boron concentration not within limit.	-------------NOTE------------- LCO 3.0.3 is not applicable. ------------------------------	
	A.1 Suspend movement of fuel assemblies in the SFP.	Immediately
	AND	
	A.2 Initiate action to restore SFP boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the SFP pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Storage

LCO 3.7.13 Fuel assembly storage in the spent fuel pool shall be maintained as follows:

a. Fuel assemblies in Region 1 shall have a K-infinity of $\leq$ 1.458 and shall have initial enrichment and burnup within the acceptable area of Figure 3.7.13-1; and

b. Fuel assemblies in Region 2 shall have initial enrichment and burnup within the acceptable area of the Figure 3.7.13-2.

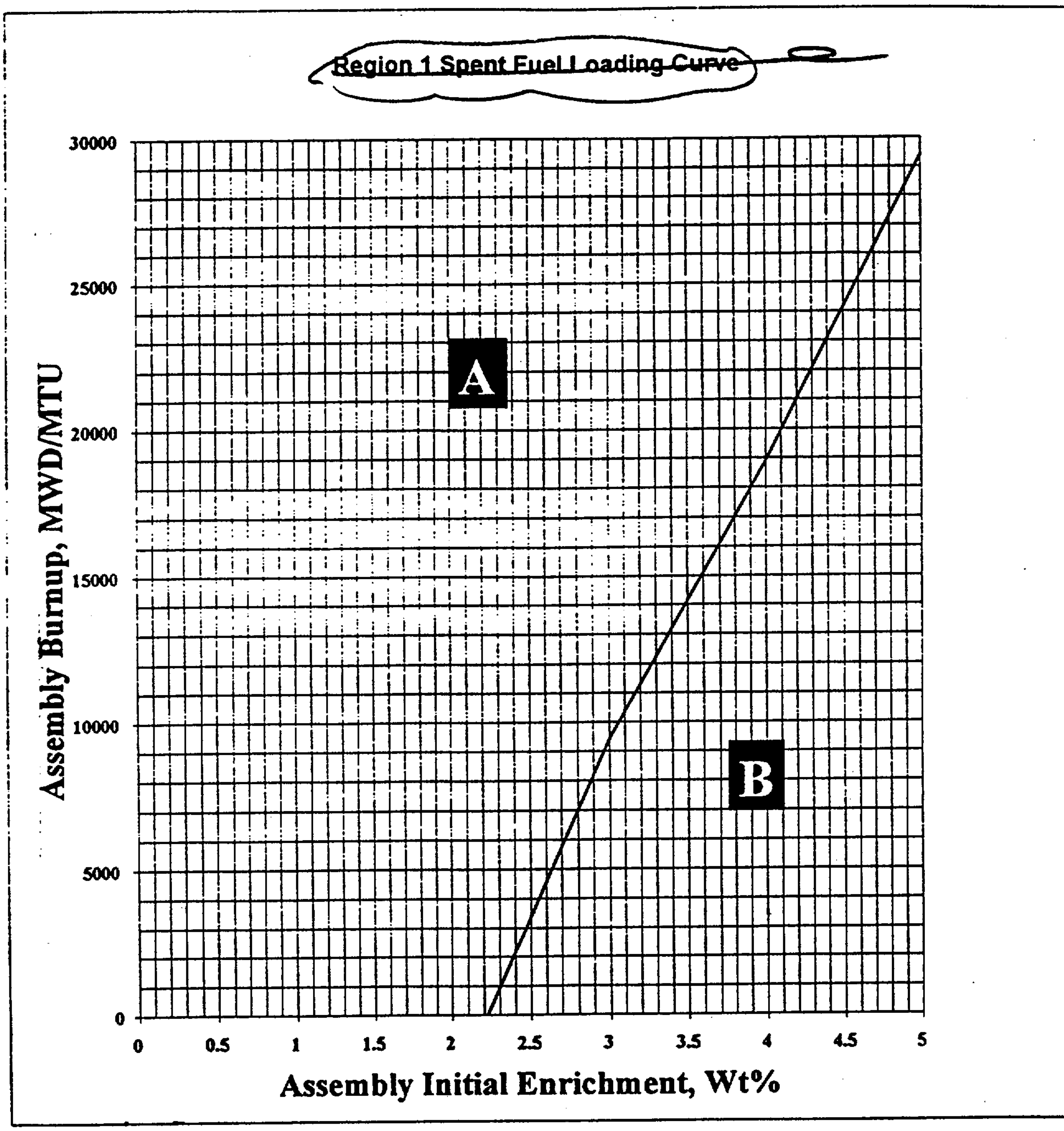
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met for either region.	A.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

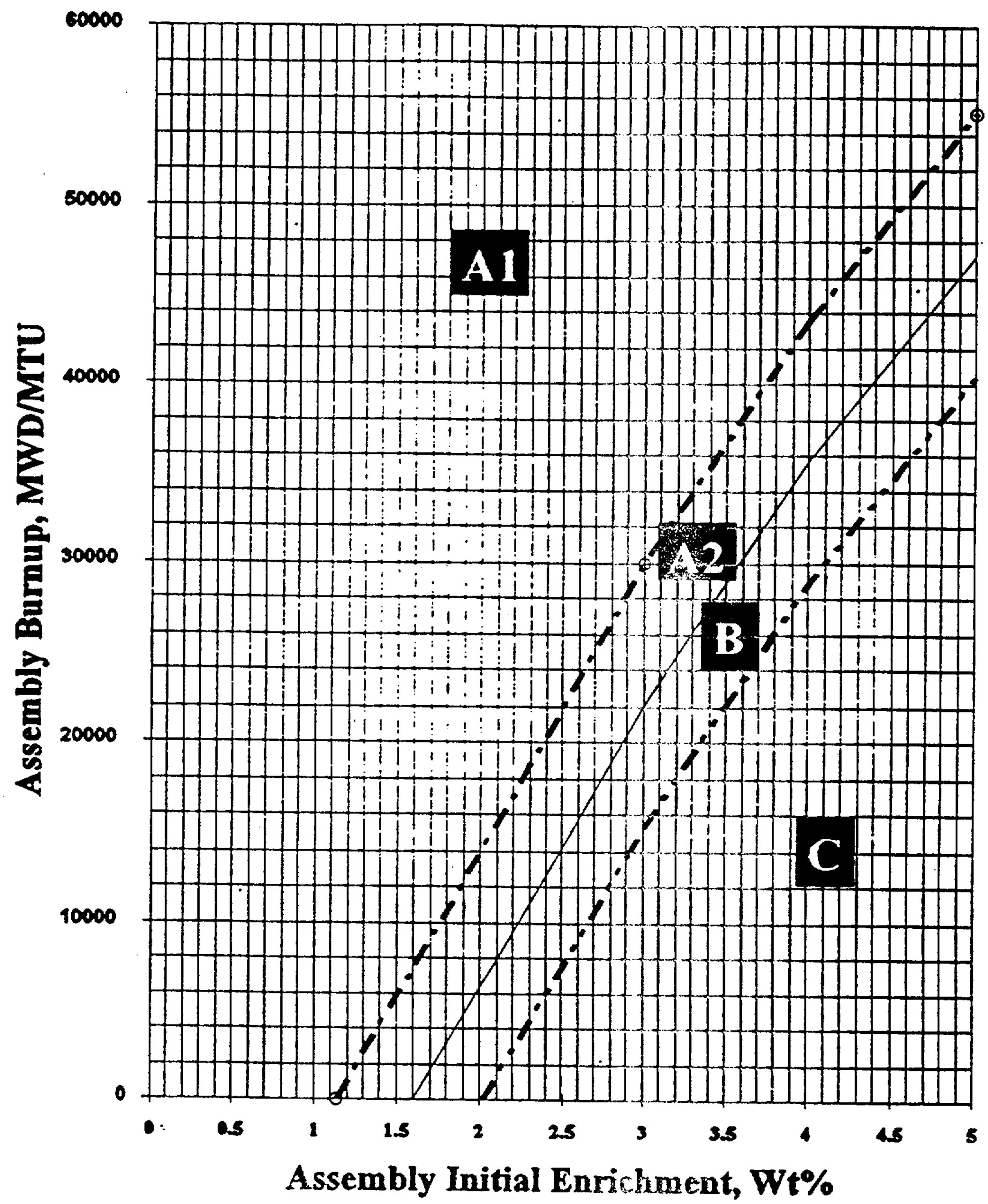
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify by administrative means the K-infinity of the fuel assembly is ≤ 1.458 and that the initial enrichment and burnup is in accordance with Figure 3.7.13-1.	Prior to storing the fuel assembly in Region 1
SR 3.7.13.2	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-2.	Prior to storing the fuel assembly in Region 2



A - Acceptable burnup domain for storage in any location within Region 1.
B - Acceptable burnup domain for storage in cells with lead-in funnels only.

Figure 3.7.13-1
Fuel Assembly Burnup Limits in Region 1



A1 - Acceptable burnup domain for storage in any location within Region 2.
A2 - Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell.
B - Assembly burnup domain for storage face-adjacent to a Type A1 assembly or a water cell.
C - Acceptable burnup domain for storage face-adjacent to a water cell only.

Figure 3.7.13-2
Fuel Assembly Burnup Limits in Region 2

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water*, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be $\leq$ 2150 BTU/hr.

4.3.1.2 The new fuel storage dry racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR.

* Until December 31, 1999, the spent fuel storage racks shall be maintained with a $k_{eff} \leq 0.95$ when flooded with water containing $\geq$ 2300 ppm soluble boron

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 257'0" (mean sea level).

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1879 fuel assemblies and 1369 storage locations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72TO 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 March 31, 1997, as supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, April 27, 1998, May 8, 1998, and May 22, 1998, the Rochester Gas and Electric Corporation (the licensee or RG&E) submitted a request for changes to the R. E. Ginna Nuclear Power Plant Technical Specifications (TSs). The requested changes would change the TS to accommodate reracking its spent fuel pool (SFP) in order to increase its capacity. The present SFP can accommodate 1016 fuel assemblies in two regions. The proposed modification would consist of replacing three fuel racks in Region 1 with seven new fuel racks, with an option to add six additional peripheral modules at a later date. Two of the seven new modules will be designated Region 2 and the remaining five modules will compose Region 1. After modification, all five fuel racks in Region 1 will be of a new design, containing borated stainless steel plates for neutron attenuation (poison) and being able to accommodate a total of 294 fuel assemblies of fresh fuel and spent fuel in a checkerboard pattern. The new Region 2 will be comprised of six modules from the old Region 2, containing Boraflex neutron attenuation panels and capable of accommodating 828 spent fuel assemblies, two new fuel racks containing borated stainless steel poison plates and accommodating 187 fuel assemblies and, to be added later, six peripheral modules containing borated stainless steel poison plates and accommodating an additional 60 spent fuel assemblies. Total number of locations in the modified SFP will be increased to 1369. This will allow storage of 1879 fuel assemblies by using consolidated rod canisters in some spent fuel locations. The amendment would increase the boron concentration from 300 ppm to 2300 ppm and change the surveillance interval from 31 days to 7 days. The May 8 and 22, 1998, letters provided clarifying information that did not change the proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Criticality Considerations

2.1.1 Evaluation

The analysis of the reactivity effects of fuel storage in the Ginna SFP was performed with the three-dimensional KENO V.a computer code using the 44 group cross section set processed by the SCALE 4.2 code system. Since the KENO V.a code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code, CASMO-3. CASMO-3 was also used for the determination of small reactivity increments due to 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 spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and neutron absorber worth. A sufficient number of neutron histories (at least 1 million) were accumulated in each calculation to minimize the statistical uncertainty of the KENO V.a calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Ginna storage racks with a high degree of confidence.

General Design Criterion 62 of Appendix A to 10 CFR Part 50 states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. This requirement is met by conforming to the NRC acceptance criterion for criticality, which states that the effective neutron multiplication factor (k_{eff}) in the spent fuel pool storage racks, if fully flooded by unborated water, shall be no greater than 0.95, including uncertainties at a 95/95 probability/confidence level.

Region 1 contains Type 3 storage racks and is designed to accommodate fresh fuel with initial nominal enrichments up to 4.0 w/o U-235 for fuel without integral fuel burnable absorbers (IFBAs), and up to 5.0 w/o U-235 for fuel with IFBAs. IFBAs consist of neutron absorbing material applied as a thin ZrB_2 coating on the outside of the UO_2 fuel pellet. As in the previous licensing analysis, the concept of reactivity equivalencing for storage of fuel assemblies with nominal enrichments between 4.0 w/o and 5.0 w/o U-235 was used. This concept is based on the reactivity decrease associated with the presence of the IFBAs and has been previously approved by the NRC.

The infinite multiplication factor, k_∞, was used as an alternative method for determining the acceptability of fresh fuel assembly storage in the Region 1 racks. A reference k_∞ of 1.458 was determined for a nominal fresh Westinghouse 14x14 Optimized Fuel Assembly (OFA) enriched to 4.0 w/o U-235 in the Ginna core geometry moderated by pure water at a temperature of 68°F. Thus, any assembly with a k_∞ no greater than 1.458 will result in a $k_{eff} \leq 0.95$ if stored in the Region 1 racks.

Fresh assemblies are stored in a checkerboard configuration so that fresh fuel is not directly adjacent to other fresh fuel. The positions adjacent to the fresh fuel are filled with depleted assemblies or left empty. All interior Region 1 cells are formed by four Borated Stainless Steel (BSS) sheets. Each cell containing a spent fuel assembly also contains a stainless steel casing which surrounds the BSS plates. Fresh fuel assemblies are only placed in the cells without stainless steel casings which contain lead-in funnels. A Westinghouse Optimized Fuel Assembly (OFA) was found to be the most reactive fuel type for fresh fuel while a Westinghouse Standard (STD) assembly was found to be the most reactive for spent fuel assemblies.

The concept of burnup reactivity equivalencing was used in order to store fuel with nominal enrichments up to 5.0 w/o U-235 in Region 1 checkerboarded with fresh 4.0 w/o U-235 (or up to 5.0 w/o U-235 with sufficient IFBAs). This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in pressurized-water reactor (PWR) fuel storage analysis. A series of reactivity calculations was performed to generate a set of enrichment versus burnup ordered pairs, which yields an equivalent k_{eff} for fuel stored in the Ginna racks. The results of the burnup reactivity equivalencing is shown in TS Figure 3.7.13-1 and shows that fuel with an initial U-235 enrichment of 5.0 w/o and irradiated to at least 29,400 MWD/MTU is equivalent to fresh fuel enriched to approximately 2.25 w/o U-235. Fuel with initial enrichment versus burnup values which meets the burnup requirements in Figure 3.7.13-1

(Domain A) results in a maximum k_{eff} of less than 0.95, including biases and 95/95 uncertainties, when stored in any location or when checkerboarded with fresh 4.0 w/o U-235 assemblies in Region 1. Fuel assemblies with minimum burnups below this value (Domain B) do not meet the 0.95 acceptance criterion and may only be stored in cells with lead-in funnels designated for fresh fuel.

For the Region 1 analysis, biases due to the calculational method and penalties for pool temperature and off-center assembly placement were included. Uncertainties due to the KENO V.a statistics and the KENO Va method were statistically combined with mechanical tolerance uncertainties. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. Where appropriate, an uncertainty was applied to the calculated burnups to account for burnup measurement uncertainties. We find this uncertainty to be consistent with previously approved Westinghouse practice and acceptably conservative. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable. Region 2 contains Type 1 storage racks with Boraflex and Type 2 racks with BSS plates. Type 4 racks containing BSS absorber plates can be added to the north and south periphery of the Type 1 racks. Because of recent industry-wide experience, which has indicated Boraflex degradation in the form of shrinkage and gap formation as well as dissolution and thinning, these effects were evaluated for the Type 1 rack criticality analysis. The analysis included the assumption of a 12-inch axial gap randomly distributed on each Boraflex panel and an 8.3% shrinkage over the width of each panel. Possible dissolution was accounted for by reducing the Boraflex thickness by 50%. However, based on Boraflex testing performed at Ginna in early 1998, as described in LER 1998-001, Boraflex dissolution was discovered in certain locations resulting in gaps larger than those assumed in the criticality evaluation. In order to account for this non-conservatism, RG&E has taken prompt corrective action by removing spent fuel assemblies from the affected locations and establishing administrative controls to prevent storage of spent fuel assemblies in these designated cells. In addition, TS 3.7.12 has been revised to increase the minimum required boron concentration in the pool to 2300 parts per million (ppm), monitored on a weekly basis. This is equivalent to the refueling boron concentration required by TS 3.9.1 during Mode 6 and to the minimum reactor water storage tank (RWST) concentration required by TS 3.5.4. Westinghouse calculations have shown that this amount of soluble boron is more than sufficient to compensate for a complete loss of all Boraflex in Region 2, while maintaining $k_{eff} \leq$ 0.95 under all postulated normal and accident conditions. RG&E anticipates that these interim compensatory measures will remain in effect during the spent fuel pool rerack modification, A future licensing amendment request will be submitted for NRC review, detailing required TS changes that will form the basis for a final resolution of this issue. RG&E plans to have the permanent solution implemented by December 31, 1999. This reflects the time needed to evaluate, design, and implement necessary modifications and to obtain NRC approval. The NRC concurs that these interim measures adequately compensate for the non-conservative assumptions in the criticality analysis and finds the actions acceptable.

The use of fixed neutron absorbers allows a closer pitch but requires burnup credit to satisfy the 0.95 k_{eff} criticality criterion. TS Figure 3.7.13-2 defines the burnup versus initial enrichment requirements for the Region 2 racks. Fuel assemblies with initial enrichments and burnups within Domain A1 result in a $k_{eff} \leq 0.95$ if stored in any location in Region 2. Assemblies within Domain A2 must be stored face-adjacent to a Type A1 assembly, another A2 assembly, or an empty cell in order to satisfy the 0.95 k_{eff} criterion. In order to allow some flexibility and to preclude filling the Region 1 rack with lower burned assemblies, evaluations were made to allow storage of assemblies with burnups up to about 15% below the normal curve, defined as Domain B in Figure 3.7.13-2. Fuel assemblies with initial enrichments and burnups within this domain can be stored face-adjacent to a Type A1 assembly or an empty cell. Assemblies below this limit are designated as Domain C fuel and must be stored face-adjacent to an empty cell only. In fact, fuel with any enrichment/burnup combination can be stored face-adjacent to an empty cell. The phrase "face-adjacent" means that the flat surface of a fuel assembly in one cell faces the flat surface of the assembly in the next cell.

As in the Region 1 analysis, for the Region 2 analysis, biases due to the calculational method and penalties for pool temperature and off-center assembly placement were included as well as an uncertainty in the depletion calculations. Uncertainties due to the KENO V.a statistics and the KENO V.a methodology bias were statistically combined with mechanical tolerance uncertainties. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. For the Type 1 rack analysis, an additional uncertainty due to B-10 self-shielding in Boraflex was included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The results of these analyses, using the acceptable methods discussed above, meet the NRC criterion of k_{eff} no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, and are therefore, 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 inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable areas or an assembly drop, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of at least 2300 ppm of soluble boron required in the pool by TS 3.7.12, 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 the boron more than offsets the reactivity addition caused by credible accidents, including Boraflex degradation.

Reactivity calculations were also performed for consolidated fuel containers designed to accommodate additional fuel consolidation. The calculations show that these canisters can also be stored in either region of the pool provided that the minimum burnups of TS Figure 3.7.13-1 and 3.7.13-2 are met.

The following Technical Specification changes have been proposed as a result of the requested spent fuel pool modifications. Based on the above evaluation, the staff finds these changes acceptable as well as the associated Bases changes.

(1) In TS 3.7.12, the requirement for the minimum boron concentration in the spent fuel pool would be increased from 300 to 2300 ppm and the Mode of Applicability would be changed to whenever any fuel assembly is stored in the spent fuel pool. The surveillance frequency would be reduced from 31 days to 7 days.

(2) In TS 3.7.13, the requirements for storage in Region 1 would be revised to include restrictions on initial enrichment and accumulated burnup as identified in new Figure 3.7.13-1. The required action for not satisfying the storage requirements would be revised to allow movement of the noncomplying assembly to any acceptable storage location regardless of storage region. Figure 3.7.13-1 would be added to provide the initial enrichment and burnup restrictions for storage in specified locations in Region 1. Previous Figure 3.7.13-1 would be renumbered to 3.7.13-2 and revised to provide additional restrictions on acceptable storage locations for Region 2.

(3) In SR 3.7.13.1, the Note providing an exemption to the SR when moving a fuel assembly from Region 2 to Region 1 would be removed. The additional restriction on initial enrichment and burnup given by Figure 3.7.13-1 would be added prior to storing a fuel assembly in Region 1.

(4) TS 4.3.1.1 (b) would be changed to add a footnote stating that $k_{eff} \leq 0.95$ in the spent fuel storage racks would be maintained with a soluble boron concentration ≥ 2300 ppm until December 31, 1999.

(5) TS 4.3.1.1 (c) would be changed to remove the statement concerning RGAF2 fuel storage canister not satisfying the requirements for initial enrichment and burnup of LCO 3.7.13.

(6) In TS 4.3.3, the spent fuel pool storage capacity would be increased from 1016 to 1879 fuel assemblies and 1369 locations (assuming consolidation).

2.1.2 Conclusions on The Criticality Considerations

RG&E plans to have a permanent solution to the Boraflex degradation concern implemented by December 31, 1999. This reflects the time needed to evaluate, design, and implement necessary modifications and to obtain NRC approval. In the interim, the staff has allowed temporary credit for the soluble boron in the pool water to maintain $k_{eff} \leq 0.95$. Preliminary calculations by Westinghouse have shown that there is a large margin (approximately 850 ppm) between the boron concentration required to maintain $k_{eff} \leq 0.95$ (1450 ppm) and the proposed minimum value of 2300 ppm. During this interim period, surveillances of boron concentration will be required every 7 days. Due to the large inventory within the spent fuel pool, dilution of the soluble boron within the pool from 2300 ppm to 1450 ppm is very unlikely during a 7-day period without being detected by operations personnel or by available water level detection systems.

Based on this, and on the review described above, the staff finds the criticality aspects of the proposed increase in the storage capacity of the Ginna SFP storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

2.2 Spent Fuel Pool Cooling Considerations

2.2.1 Evaluation

2.2.1.1 Spent Fuel Pool Cooling System

The SFP cooling system is designed to maintain the SFP water temperature at or below 150 °F during all modes of plant operation including full core off-load outages. The licensee stated that this design SFP water temperature is not imposed as an SFP water temperature limit in the GINNA TS; however, it is specified as an SFP water temperature limit in the technical requirements manual (TRM), which is an extension of the Updated Final Safety Analysis Report (UFSAR). Changes to the TRM are performed under 10 CFR 50.59.

The SFP cooling system consists of three loops. The primary loop (loop 2), which is Seismic Category 1 designed, contains SFP heat exchanger B, SFP pump B and piping; Loop 1 contains SFP heat exchanger A, SFP pump A and piping; and Loop 3, which is a skid-mounted loop, contains a skid mounted SFP pump, a standby SFP heat exchanger and hoses. Loops 1 and 3 are not Seismic Category 1 designed. Heat is removed from the SFP heat exchangers by the service water system. Electrical power to all SFP pumps is supplied from safety buses. Normally, either Loop 1 or Loop 2 is operated alone to maintain the desired temperature. The hoses and power supply of Loop 3 are disconnected and the fuel pool water supply and discharge hose connections are closed off with threaded caps. Loop 3 will be connected and operated only when high heat loads are expected. In the event of a full core off-load, Loop 2 will be operated alone with Loop 1 and 3 available for backup cooling. The licensee stated that, prior to a full core off-load, the skid-mounted Loop 3 is placed in position (i.e., electrical power from safety bus and hoses are connected, the loop is leak checked, etc.,) to support operation in parallel with Loop 1.

The SFP cooling system is designed with only Loop 2 in operation at Lake Ontario (ultimate heat sink) water temperature of 80 °F to maintain the SFP water temperature at or below 150 °F with a SFP heat load of 16.0×10^6 Btu/hr resulting from a planned or unplanned full core off-load. In the May 8, 1998, submittal, the licensee stated that Loops 1 and 3, which are designed as backup to LOOP 2, are each capable of removing 9.3×10^6 Btu/hr with a pool temperature of 150 °F and Lake Ontario water temperature of 80 °F. Thus, for a full core off-load, 100% backup cooling can be provided by Loop 1 with Loop 3 operating in parallel.

The SFP cooling system heat loads (1/3 or full core off-load) will increase as the number of spent fuel assemblies stored in the SFP increases. In order to maintain the SFP temperature below the TRM temperature limit of 150 °F, the fuel must be held in the core for a minimum shutdown duration to ensure that the total SFP heat load is less than the heat removal capability of the existing SFP cooling system. In any event, spent fuel assemblies may not be off-loaded from the core prior to a minimum shutdown duration of 100 hours. Since the heat removal capability of the SFP cooling system is a function of Lake Ontario temperature, the licensee performed analyses for the following discharge scenarios to determine/establish required shutdown time to maintain the SFP temperature less than 150 °F for Lake Ontario water temperatures of 40 °F and 60 °F as well as the design Lake Ontario water temperature of 80 °F.

2.2.1.2 Routine Refueling Outage With 1/3 Core Off-load[1]

To determine the SFP heat load for the limiting case, the licensee assumed that beginning in 1997 through the end of plant life, a bounding 44 spent fuel assemblies of 18-month fuel cycle were discharged to the SFP after a core shutdown duration of 100 hours. The calculated maximum SFP heat load, which includes the heat load from all previous discharged batches, is 11.3×10^6 Btu/Hr. This calculated maximum SFP heat load of 11.3×10^6 Btu/Hr is well within the 16.0×10^6 Btu/Hr heat removal capability of either the Loop 2 SFP heat exchanger or its backup Loop 1 and 3 heat exchangers at the highest Lake Ontario water temperature of 80 °F. Consequently, the licensee concluded that a normal 1/3 core off-load after 100-hours decay will not result in the SFP approaching its design temperature of 150 °F.

The following table compares SFP cooling system Loop 2 (primary loop) heat removal capability to the decay heat load after a 100-hour decay time at various Lake Ontario water temperatures:

---

1 **Routine refueling is a 1/3 core (approximately 40 fuel assemblies) off-load.**

Lake Water Temperature (°F)	Primary Loop Heat Removal Capability (MBtu/hr)	SFP Heat Load (MBtu/hr)	Core Shutdown Duration (hours)
40	24.6	11.3	100
60	20.4	11.3	100
80	16.0	11.3	100

Based on our review, we concur with the licensee that a normal 1/3 core off-load after 100-hours decay during routine refueling outage will not result in the SFP approaching its design temperature of 150 °F.

2.2.1.3 Full Core Off-load

The licensee performed an analysis to determine the required core shutdown duration for ensuring that the design SFP water temperature limit of 150 °F is not exceeded during a planned or unplanned full core off-load with a full SFP inventory of spent fuel assemblies and at Lake Ontario temperatures of 40 °F, 60 °F and 80 °F.

The following summarizes the SFP heat loads with various core shutdown times and their corresponding SFP cooling system Loop 2 (primary loop) heat removal capability for Lake Ontario temperatures of 40 °F, 60 °F and 80 °F:

Lake Water Temperature (°F)	Primary Loop SFP Heat Exch. Capacity (MBtu/hr)	SFP Heat Load (MBtu/hr)	Core Shutdown Time Required (hours)
40	24.6	21.7	100
60	20.4	20.4	132
80	16.0	16.0	280

As indicated in the above table, maintaining the SFP temperature limit of 150 °F is based on two primary parameters. The first is the Lake Ontario temperature, since the lake provides the ultimate heat sink for the SFP heat exchangers. The second is the in-core or in-vessel decay time following reactor shutdown, since this determines the heat load within the SFP. Therefore, in the November 11, 1997, submittal, the licensee stated that the TRM will be modified prior to the next full core off-load with the new racks installed to ensure 100% backup for all SFP cooling scenarios at various Lake Ontario water temperatures, in-reactor decay time, and associated SFP heat loads. Also, in the May 8, 1998, submittal, the licensee stated that the interpretation for this section of the TRM would be to chose the most conservative lake temperature (i.e., the next higher temperature) for the applicable scenario.

Also, the SFP has a water temperature monitor, which alarms in the control room when the SFP water temperature reaches 115 °F. Annunciator response instruction lists the probable causes and corrective actions to be taken when the high temperature alarm is received. This will provide an additional measure to prevent the SFP water temperature from being exceeded.

Based on our review and the licensee's commitment to impose 100% backup for all SFP cooling scenarios at various Lake Ontario water temperatures, in-vessel decay time, and associated SFP heat loads in the TRM, we find that the design and operation of the SFP cooling system meets the intent of the guidance described in Standard Review Plan for SFPs. Therefore, the design and operation of the SFP cooling system is acceptable.

2.2.1.4 Effects of SFP Boiling

In the unlikely event that there is a complete loss of cooling to the SFP, the SFP water temperature will begin to rise and eventually will reach the boiling temperature. The licensee performed analysis to demonstrate the time to boil and the boil off rate based on the various heat loads for the full core off-load scenarios. The following table summarizes the results of the analysis:

SFP Heat Load (MBtu/hr)	Core Shutdown Time Required for 150°F TRM Temp. Limit (hours)	Time to Boil 150°F-212°F (hours)	Boil Off Rate (gpm)
21.7	100	5.7	47.0
20.4	132	6.1	44.0
16.0	280	7.7	35.0

As indicated in the above table, the calculated minimum time from the loss-of-pool cooling until the pool boils is 5.7 hours with a maximum boil-off rate of 47.0 gpm. The licensee performed an evaluation to demonstrate that there will be sufficient time to restore SFP cooling or to establish makeup water, if required, to the pool from various qualified sources. The licensee stated that 60 gpm of water from the refueling water storage tank can be made available as makeup in less than 15 minutes. As an alternative, 50 gpm of water from the CVCS hold-up tanks can also be made available in approximately 15 minutes.

Based on our review, we find that in the unlikely event of a complete loss of cooling, the licensee is capable of aligning makeup to the pool before boiling begins and that the makeup will be supplied at a rate which exceeds the boil-off rate. We conclude that cooling the SFP at GINNA by adding makeup water during an unlikely event of a complete loss of SFP cooling conforms with the guidance described in the SRP; therefore, it is acceptable.

2.2.2 Conclusions on Spent Fuel Pool Cooling Considerations

Based on its review of the licensee's rationale and commitments to impose 100% backup for all SFP cooling scenarios at various Lake Ontario water temperatures, in-vessel decay time, and associated SFP heat loads in the TRM and provided that the plant's FSAR and TRM will be updated to reflect the above information regarding the SFP, we conclude that the licensee's proposed plan to rerack the SFP during the 1999 refueling outage to allow an increase in the spent fuel storage capacity from 1016 to 1879 fuel assemblies is acceptable.

2.3 Heavy Loads Considerations

2.3.1 Introduction

The heavy load handling activities involved in the proposed rerack operation consist of: removal and installation of the spent fuel rack modules; handling of the gate separating the spent fuel pool from the cask loading pit; transfer of spent fuel assemblies; use of the hoisting system including a single failure proof crane and lifting devices; safe load paths; procedures; trained operators; and evaluation of postulated load drop accidents.

2.3.2 Evaluation

2.3.2.1 Hoisting System

The hoisting system used during the rerack operations consists of the Auxiliary Building 30-ton single failure proof crane and single failure proof lifting devices. This crane will be used to lift the consolidated fuel, the pool canal gate, storage racks, and the spent fuel shipping casks.

As stated by the licensee, the crane was upgraded to meet single-failure-proof criteria in NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants, " dated May 1979, and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The upgraded crane was evaluated by the staff in an SE dated November 14, 1984, in which the staff found that the crane satisfied guideline 7 of NUREG-0612, Section 5.1.1. Therefore, the crane is designed to meet criteria and guidelines in Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes" and in CMAA-70, "Specifications for Electric Overhead Traveling Cranes."

The crane is rated at 32.5 tons with a maximum critical service load of 30 tons and will be used to lift the spent fuel racks and fully loaded spent fuel shipping casks. The single failure proof design of the crane enables it to stop and hold the load under all conditions including a safe shutdown earthquake. The licensee plans to use redundant and nonredundant lifting devices as determined to be needed throughout the lifts. The lifting devices will also be in accordance with guidance in NUREG-0612 and ANSI B30.2. Therefore, both the crane and the lifting devices are to be load tested at 125% of the maximum service load (rated load) prior to use in accordance with ANSI B30.2.

2.3.2.2 Postulated Load Drop Accidents

Although NUREG-0612 states that licensees using single failure proof cranes to transfer heavy loads do not have to perform heavy load drop analyses, R. E. Ginna presented analyses for load drops during all lift operations. Load drop analyses are presented for the storage rack modules, the spent fuel assembly, the canal gate, and the spent fuel shipping casks in accordance with the guidelines in NUREG-0612. Also presented is an analysis for a postulated drop of the consolidated canisters. The canisters will be used to move all of the fuel rods from two assemblies. Based on the analyses, the licensee found that because the 30-ton single failure proof crane will be used to lift all loads, the loads will be prevented from dropping. The licensee also found that because the lifting devices for each of the loads except the spent fuel pool gate will be single failure proof, load drops due to failure of the lifting devices are very unlikely. The licensee committed to upgrade the lifting mechanism for the spent fuel pool gate to the guidelines in NUREG-0612 to prevent a drop of the gate.

The licensee committed to use electrical interlocks on the Auxiliary Building Crane to prevent the movement of casks over storage racks and to implement procedures and administrative controls to preclude load travel over spent fuel. The use of the single failure proof crane and upgraded lifting devices with the crane interlocks, procedures and administrative controls to prevent crane travel over fuel enables the licensee to conclude that a load drop involving the release of fission products is highly unlikely. Based on this finding, the licensee stated that radiological consequences of a release need not be determined and that the drop of a cask or a tip-over accident need not be postulated. The licensee did examine the potential consequences of straight drop of a fuel assembly and noted that it would not result in any damage to the integrity of the pool nor uncover the spent fuel. These methods of assuring safer handling of heavy loads are acceptable to the staff.

2.3.3.3 Conclusions on Heavy Loads Considerations

Based on the preceding discussion, the staff finds acceptable the licensee's methods of handling heavy loads during the rerack operation, including the licensee's commitment to abide by the guidelines in NUREG-0612. Accordingly, the licensee will follow guidelines concerning establishing safe load paths, using procedures and controls for the movement of heavy loads, training crane operators, design and testing of lifting devices, and inspection, testing, and maintenance of cranes. These commitments will enable the licensee to perform its rerack operation in a safe manner.

The licensee's evaluation of the consequences of postulated load drops of spent fuel storage racks, spent fuel assemblies, the canal gate, and shipping transfer casks satisfy the guidelines in Section 5.1 of NUREG-0612. The licensee has committed to use procedures, administrative controls, and redundant rigging, as applicable, to prevent load drops.

Based on the above, the staff concludes that the single failure proof crane, the upgraded lifting devices, testing the hoisting system, operator training, and procedures for inspection will reduce the probability of a load drop in the SFP to an acceptable level. Therefore, the proposed changes to the SFP capacity are acceptable.

2.4 Structural Design Considerations

2.4.1 Evaluation

The primary purpose of this review is to assure the structural integrity and functionality of the rack modules and the stored fuel assemblies subject to the effects of the postulated loads (Appendix D of SRP Section 3.8.4) and fuel handling accidents.

2.4.1.1 Storage Racks

The 1879 fuel assemblies will be contained in nineteen (19) fuel storage racks, which are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). Among 19 storage racks, six (6) racks are currently being used and the other thirteen (13) racks will be added to the SFP at Ginna. Among those 13 new racks, seven (7) new racks are planned to be installed in 1998 and the remaining six (6) new racks will be added at some time in the future. All 13 new racks will be designed and manufactured by Societe Atlantique de Techniques Avancees (ATEA). RG&E with its contractor, Framatome Technologies, Inc. (FTI), performed structural analyses for the racks for the requested license amendment.

RG&E used a computer program, ANSYS, for dynamic analysis to demonstrate the structural adequacy of the Ginna spent fuel rack design under the combined effects of the earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor and walls of the storage pool. A nonlinear dynamic model consisting of mass elements, fluid elements, spring elements, gap elements and friction elements, as defined in the program, was used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two model analyses were performed: the 3-D single-rack (SR) model analysis and the 3-D whole pool multi-rack (MR) analysis. In these 3-D model analyses, each rack was considered fully loaded, half loaded and almost empty loaded with three different coefficients of friction between the rack and the pool floor (μ=0.2, 0.5 and 0.8) to identify the worst case response for rack movement and for rack member stresses and strains.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time histories) were generated from the design response spectra defined in the final safety analysis report (FSAR) (Reference 4). RG&E demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in Standard Review Plan (SRP) Section 3.7.1.

Section 3.5 of the submittal report (Reference 1) shows the SR and MR analysis results. The results of the SR analysis show that the maximum displacements of the racks at the top and the baseplate corners are about 0.7 inch and 0.2 inch, respectively, assuring that there are no rack-to-wall or rack-to-rack impacts under the service, upset and faulted loading conditions (Level A, B and D Service Limits). The analysis results show that the uplift rack movement is very small (less than 0.2 inch) indicating that there are large safety margins against overturning of the racks. The analysis results demonstrate that structural integrity and stability of the racks and fuel assemblies are maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Subsection NF. The results show that all induced stresses under the service, upset and faulted loading conditions (Level A, B and D Service Limits) are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the rack design is adequate.

In the 3-D MR analyses, thirteen (13) free standing racks were considered to investigate the fluid-structure interaction effects between racks and pool walls as well as those among the racks. The results of the MR analysis indicate that all calculated stresses are smaller than the corresponding allowable stresses of the ASME Code. In addition, the results show that there are no rack-to-wall or rack-to-rack impacts as the result of an SSE, assuring that the structural integrity and stability of the racks are maintained.

RG&E also calculated the weld stresses of the rack at the connections under the dynamic and thermal loading conditions. RG&E demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on: (1) the RG&E's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the large safety margins of the induced stresses of the rack when they are compared to the corresponding allowables provided in the ASME Code, Section III, (3) a reasonable assurance that there is no rack-to-wall and rack-to-rack impacts, and (4) RG&E's overall structural integrity conclusions supported by both 3-D SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under the service, upset and faulted loading conditions (Level A, B and D Service Limits) and, therefore, are acceptable.

2.4.1.2 Fuel Handling Accident

The following two refueling accident conditions were evaluated by RG&E: (1) two cases for drop of a fuel assembly with its handling tool, which impacts the baseplate and support leg (deep drop scenario) and (2) two cases for drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of the first accident condition (deep drop scenario) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads and induces stress in the liner. The induced stress is smaller than the allowable and, therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of the second accident condition (shallow drop scenario) show that damage will be restricted to a depth of 0.14 inch below the top of the rack, which is well above the active fuel region. The staff reviewed RG&E's analysis results provided in Reference 1 and concurs with its findings. This is acceptable based on the RG&E's structural integrity conclusions supported by the parametric studies.

2.4.2 Conclusions on Structural Design Considerations

Based on the review and evaluation of RG&E's submittal (Reference 1), and additional information and analysis provided by RG&E (References 2 and 3), the staff concludes that RG&E's structural analysis and design of the spent fuel rack modules are adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR and applicable provisions of the SRP. They are, therefore, acceptable.

2.4.3 References Related to Structural Design Considerations

1. "Application for Amendment to Facility Operating License, Revised Spent Fuel Pool Storage Requirements, Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant, Docket No. 50-244," Letter dated March 31, 1997, from RG&E to U.S. NRC.

2. "Response to Request for Additional Information - Spent Fuel Pool (SFP) Modification - Structural Design Considerations (TAC No. M95759), R.E. Ginna Nuclear Power Plant, Docket No. 50-244," Letter dated October 20, 1997, from RG&E to U.S. NRC.

3. "Response to Request for Additional Information on the Structural Aspects of the Spent Fuel Pool Storage Rack Modification at Ginna Nuclear Power Plant (TAC No. M95759)," Letter dated January 27, 1998, from RG&E to U.S. NRC.

4. R. E. Ginna Nuclear Power Plant, Final Safety Analysis Report.

2.5 Materials and Chemical Engineering Considerations

2.5.1 Evaluation

The new spent fuel rack will be manufactured by Societe Atlantique de Techniques Avancees (ATEA) in their Nantes, France facility. The licensee has indicated that the materials used for structural and neutron attenuation (poison) components in the fuel racks have been successfully used in other nuclear plants and are not expected to show any significant degradation in the SPF environment. A detailed description of materials used in individual components was provided in the submittal.

2.5.1.1 Structural Materials

The licensee has indicated that all the selected structural materials conformed to the ASTM specifications and meet the intent of the ASME Section III, Subsection NF requirements.

The cell walls, base support plate and support pads were made either from ASTM-A240 or A479 Type 304L stainless steel and the perimeter rack connection from ASTM A240 Type 304 stainless steel. The choice of Type 304 stainless steel for fabrication of the rack assembly is reasonable. The high chromium content imparts corrosion resistance to the SFP environment. Similarly, the use of ASTM A564 Type 630 precipitation hardened stainless steel, heat treated to 1100°F to increase its resistance to stress corrosion cracking, will ensure its compatibility with this environment. However, it should be recognized that in an acidic environment such as exists in the SFP, presence of chloride or bromide ions may lead to some corrosion of stainless steel. It is important, therefore, that the purity of water chemistry in the SFP is well controlled to ensure that the presence of these ions or other impurities that can cause corrosion is reduced to the minimum. The compositions of different materials used in construction of the fuel racks, including Grade 308L weld material, place them sufficiently close in galvanic series so that, when in contact, no galvanic coupling, which could be a cause of corrosion, will occur.

2.5.1.2 Neutron Attenuation (Poison) Material

2.5.1.2.1 Borated Stainless Steel (BSS)

In the new racks the licensee used for a neutron attenuating (poison) material borated stainless steel. The steel was produced using processes designed to prevent formation of residual stresses. The rack manufacturer has developed a special design by which BSS panels could be attached to the spent fuel racks without welding. They were also never used as load-bearing components. Consequently, during their service the BSS components will not be exposed to any internal or external stresses. The sizes of the BSS panels varied slightly in different fuel racks, but in all cases the material was identical. It consisted of grade 304 B6/B7, Type B meeting the ASTM-A887-89 and A-480 standards. This austenitic stainless steel contained minimum 1.7 percent of natural boron (B10) and maximum 0.04 percent of carbon. Boron is in a form of very small iron boride (Fe_2B) particles uniformly distributed in the steel matrix. This produces macroscopically uniform material. Iron boride is very resistant to oxidation and chemical attack. It is not expected, therefore, that it will be lost from the borated stainless steel plates during their lifetime. The steel itself has corrosion characteristics very similar to those of a regular 304 austenitic stainless steel, which is known to exhibit corrosion resistance in the SFP environment. However, the licensee will institute a surveillance program involving examination of the test coupons exposed to the SFP environment. Mechanical properties of the borated stainless steel are similar to those of a regular austenitic steel, although in the present application they were of no special concern because the steel is not used in any load bearing components.

2.5.1.2.2 Boraflex

Boraflex is used as a neutron absorbing material in the existing spent fuel racks. These racks will be retained in Region 2 of the modified SFP. Boraflex panel consists of a sheet of silicon rubber material containing boron carbide particles imbedded in a polymer matrix. The panels of Boraflex are attached to the walls of fuel cells. Although, in general, the material exhibits fairly good stability in the SFP environment, after a long exposure to radiation it tends to show some signs of degradation. There are two types of degradation: dimensional changes and loss of material. Because in radiation fields polymer chains tends to cross-link, Boraflex sheets shrink causing gap formation. Usually, this effect occurs at radiation doses of less than approximately 10^{10} rads. After that dose is reached, no more shrinking occurs. The second type of degradation consists of a partial decomposition of the polymer due to polymer chain scission and release of boron carbide particles. The Boraflex in the existing rack has already been exposed to the limiting radiation doses and no more shrinkage is expected. In the safety analysis the licensee made a conservative assumption that 8.3 percent of shrinkage has occurred and included the resulting gap formation in its criticality analysis. Since loss of boron carbide caused by polymer degradation will be continuously occurring as polymer receives more radiation, a program to monitor neutron attenuation capability of the Boraflex panels is required.

2.5.1.2.3 Water in Spent Fuel Pool

Proposed increase of the number of fuel assemblies stored in the spent fuel pool from 1016 to 1369 and increase of boron concentration from 300 ppm to 2300 ppm may have some effect on the spent fuel pool's cleaning system. However, the licensee has demonstrated that the effect will be relatively small and will not impact its performance to any significant degree.

2.5.3 Conclusions on Materials and Chemical Engineering Considerations

Based on its evaluation, the staff finds that the structural materials used in the new racks and borated stainless steel, used for neutron attenuation, are compatible with the environment of the SFP operating with increased boric acid concentration. These materials are not expected to undergo degradation that could affect the ability of storage racks to safely store spent fuel. In addition, as a safety precaution, the licensee will monitor the conditions of the borated stainless steel by means of surveillance coupons. Behavior of Boraflex in the spent fuel pool environment has already been demonstrated in the existing fuel racks. Its shrinkage and consequential gap formation were conservatively accounted for in the criticality analysis. The licensee will monitor the effects of radiation on its integrity by a special monitoring program. Based on these considerations, the staff concludes that the structural and poison materials used in the modified SFP will not experience any degradation that would negate the capability to perform their design function.

2.6 Radiation Protection, Solid Radioactive Waste, and Accident Analylsis Considerations

2.6.1 Evaluation

2.6.1.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the Ginna spent fuel racks with respect to occupational radiation exposure. As stated above, for this modification the licensee plans to remove three SFP rack modules and replace them with seven new rack modules. The licensee will then decontaminate the three removed modules prior to disposing of them. The previous SFP reracking, performed in 1984-85, involved a dose of 14 person-rem. On the basis of past experience, the licensee estimates that they can perform the proposed reracking for between 8 and 12 person-rem.

In order to achieve this reduction in dose, the licensee plans to incorporate lessons learned from previous SFP rerackings performed at Ginna. The licensee will track personnel doses using their automated electronic dosimetry program, which provides the licensee with a continuous updating of worker doses.

In addition to wearing multiple electronic and TLD dosimetry to ensure accurate recording of their doses, all divers used to perform work in the SFP will be equipped with teledosimetry capable of monitoring high dose rates. This teledosimetry system will have a continuous readout monitored at the top of the pool. An underwater T.V. system will be used to monitor the movements of divers to ensure that they do not stray into areas of high dose rates. The licensee will use a Continuous Air Monitor, capable of monitoring for noble gases and iodides, in the SFP area during the modifications to monitor for any gaseous releases. In addition, the plant effluent radiation monitoring system will monitor any gaseous releases.

The licensee will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and generation of radioactive wastes. To the extent feasible, the licensee will use long-handled tools to facilitate SFP rack module removal and installation. Those tools having hollow handles will be designed to permit water to enter the handle during use to prevent any potential radiation streaming to the person using the tool.

During reracking operations, there is the potential for an increase in radioactivity concentrations in the SFP due to spalling of crud from spent fuel assemblies during movement. In order to minimize the effects of spalling in the SFP, the licensee plans to minimize the number of fuel assembly shuffles during the removal of existing racks and installation of new racks. Any changes in the radioactivity levels in the SFP will be monitored by the two underwater radiation probes that will be used during the reracking operation. The licensee also plans to use an underwater vacuum and fine pore filters to minimize any potential radiological effects of spalling and to maintain water clarity in the SFP.

The licensee calculated the expected dose rates for the areas adjacent to the sides of the SFP and determined that the the increased fuel storage will have a negligible effect on dose rates in accessible areas. The calculated dose rate at the surface of the SFP with the increased fuel storage is estimated to be 8E-10 R/hr. Dose rates on the fuel pool level are primarily due to radionuclides in the pool water. During normal operations, dose rates in this area are generally 1.0 to 2.0 mrem/hr. These dose rates usually increase slightly during refueling operations, when the fuel is being moved. The staff finds these dose rates to be acceptable and in accordance with SFP dose rates at other plants.

The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase due to the expanded SFP storage capacity. However, there will be a continuous airborne radioactivity monitor available in the area to monitor for airborne radioactivity levels.

On the basis of our review of the Ginna proposal, the staff concludes that the Ginna SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained as low as is reasonable achievable (ALARA). The staff finds the projected dose for the project of 8 to 12 person-rem to be in the range of doses for similar SFP modifications at other plants and, therefore, acceptable.

2.6.1.2 Solid Radioactive Waste

Spent resins are generated by the spent fuel pool purification system. In order to minimize the generation of spent resins, the licensee will clean the floor of the SFP before any work is begun and after each of the old Region 1 fuel rack modules is removed. On the basis of experience gained following the 1984-1985 SFP modification, the licensee concludes that the additional fuel storage made possible by the increased storage capacity will not result in a significant change in the generation of solid radwaste.

Following the proposed reracking operation, the three fuel rack modules removed from Region 1 of the SFP will be decontaminated. The old racks will then either be cut up and stored onsite or packaged and shipped by truck to a facility licensed for the processing of low-level radioactive waste. If shipped, the licensee has stated that the shipping containers and procedures will conform to all applicable regulations set forth by the U.S. Department of Transportation (DOT) as well as the requirements of any State DOT office through which the shipment may pass and the requirements of the American Association of State Highway and Transportation Officials.

2.6.1.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of six hypothetical accidents involving fuel in the SFP. Because the licensee uses single failure proof cranes for the lifting of heavy loads in the vicinity of the SFP, four of these accidents are deemed not plausible. The licensee evaluated the other two hypothetical accidents, the fuel handling accident and the tornado missile accident, to determine the thyroid and whole-body doses at the Exclusion Area Boundary (EAB), Low-Population Zone (LPZ), and Control Room. The proposed reracking of the Ginna SFP will not affect any of the assumptions or inputs used in evaluating the dose consequences of either of these hypothetical accidents.

The staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the licensee's doses. In performing these calculations, the staff used the assumptions of RG 1.25, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The staff performed two separate assessments. For the fuel handling accident, the staff assumed that the cladding of all of the fuel rods (179 rods) in a single fuel assembly would be perforated if the fuel assembly were dropped during handling. The damaged fuel assembly is assumed to contain freshly off-loaded fuel with a minimum of 100 hours of decay. The tornado missile accident assumes that a hypothetical tornado missile (a 1,490 pound wooden pole, 35 feet in length and 13.5 inches in diameter) could impact and damage the fuel in either region of the SFP. Region 1 contains freshly off-loaded assemblies with a minimum of 100 hours of decay interspaced with older fuel assemblies from Region 2. Region 2 contains fuel assemblies with minimum decay times of at least 60 days. Therefore, a tornado missile accident to the Region 1 part of the SFP will provide the limiting dose consequences. A tornado missile is assumed to damage nine fuel assemblies. Since freshly off-loaded fuel assemblies are alternated with Region 2 assemblies in the Region 1 part of the SFP, a tornado missile is assumed to damage five freshly off-loaded assemblies and four Region 2 assemblies. The parameters which the staff utilized in its assessment are presented in Table 1.

The staff's calculations confirmed that the thyroid doses at the EAB, LPZ, and Control Room from either a fuel handling accident or a tornado missile accident meet the acceptance criteria and that the licensee's calculations are acceptable. The results of the staff's calculations are presented in Table 2. For a fuel handling accident, the staff calculated a dose of 13.1 rem thyroid at the EAB and 0.82 rem thyroid at the LPZ. For the tornado missile accident, the staff calculated a dose of 37.6 rem thyroid at the EAB and 18.8 rem thyroid at the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is contained in SRP Section 15.7.4 of NUREG-0800 and is 75 rem thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem). For the fuel handling accident involving the damage of a single fuel assembly, the staff calculated a dose of 0.63 rem thyroid to the control room operator. This calculation assumed no intake of unfiltered air into the control room. In actual conditions for this accident, filtered recirculation flow of control room air would not start until 30 seconds into the accident, resulting in a higher dose to the control room operator. The staff has reviewed the licensee's calculation for the resulting dose to the control room operator assuming 30 seconds of unfiltered air intake at the beginning of the accident and found it to be acceptable. The licensee calculated a dose to the control room operator for the fuel handling accident (involving a single fuel assembly) of 23 rem thyroid. For the tornado missile accident, the staff calculated a dose of 14.5 rem thyroid for the control room operator. The acceptance criterion for the control room operator dose is 30 rem thyroid (SRP Section 6.4 of NUREG-0800). The staff, therefore, finds the proposed reracking at Ginna to be acceptable with respect to potential radiological consequences as a result of a hypothetical fuel handling or tornado missile accident.

Table 1

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING AND TORNADO MISSILE ACCIDENTS

Parameters	
Power Level, Mwt	1551
Number of Fuel Rods Damaged (Single Assembly)	179
Number of Fuel Rods Damaged (Nine Assemblies)	1611
Total Number of Rods in Core	21,659
Shutdown Time, hours	
Region 1 Assemblies	100
Region 2 Assemblies	1440
Power Peaking Factor	
Fuel Handling Accident	1.75
Tornado Missile Accident	1.2
Fission-Product Release Fractions (%)*	
Iodine	10
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
Filter Efficiencies for Auxiliary Building (%)*	
Fuel Handling Accident	
. . . Elemental	90
. . . Organic	70
Tornado Missile Accident	
. . . Elemental	0
. . . Organic	0
Filter Efficiencies for Control Room (%)	90
Core Fission Product Inventories per TID-14844	
Atmospheric Dispersion Factors, X/Q (sec/m^3)	
Exclusion Area Boundary (0-2 hours)	
. . . Fuel Handling Accident	4.8×10^{-4}
. . . Tornado Missile Accident	6.0×10^{-5}
Low Population Zone (0-8 hours)	3.0×10^{-5}
Control Room (0-8 hours)	6.95×10^{-4}

* Regulatory Guide 1.25

TABLE 2

THYROID DOSES FROM FUEL HANDLING AND TORNADO MISSILE ACCIDENTS AT GINNA (VALUES CALCULATED BY NRC STAFF)

	DOSE (REM-THYROID)	
	FUEL HANDLING ACCIDENT	TORNADO MISSILE ACCIDENT
EAB*	13.1	37.6
LPZ*	0.82	18.8
Control Room**	0.63***	14.5

*Acceptance Criterion = 75 rem thyroid
**Acceptance Criterion = 30 rem thyroid
***See Safety Evaluation for discussion of Control Room dose for fuel handling accident

3.0 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.

4.0 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 July 22, 1998, (63 FR 39296). 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.

5.0 CONCLUSION

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 Contributors: L. Kopp
K. Parczewski
C. Hinson
D. Shum
Y. Kim
B. Thomas

Date: July 30, 1998