December 7, 2000

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

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: REVISION TO THE STORAGE CONFIGURATION REQUIREMENTS WITHIN THE EXISTING STORAGE RACKS AND TAKING CREDIT FOR A LIMITED AMOUNT OF SOLUBLE BORON (TAC NO. MA8443)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No.79 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 8, 2000, as supplemented April 5, 2000, and October 25, 2000.

The amendment revises the Technical Specifications through revision to the storage configuration requirements within the existing storage racks and taking credit for a limited amount of soluble boron.

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

This amendment involved a particularly complex issue in which you provide an excellent application. Your staff should be commended for a thorough and complete submittal which required no additional information for the NRC staff review.

Sincerely,

/RA/

Guy S. Vissing, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 79 to License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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DISTRIBUTION: See next page

ACCESSION NO. ML003761578
**See previous concurrence

*SE Provided on October 2, 2000, and no major changes were made

OFFICE	PM:PDI-1	LA:PDI-1	OGC**	SC:PDI-1	SRXB*
NAME	GVissing	SLittle	11/28/00	P. Tam for MGamberoni	LKopp
DATE	12/6/00	12/6/00	AHodgdon	12/6/00	10/2/00

cc:

Ho K. Nieh, Jr., 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 Law
120 Broadway
New York, NY 10271

Dan Stenger
Hopkins and Sutter
888 16th St.
Washington, DC 20006-4103

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

DATED: December 7, 200

AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. DPR-18-GINNA NUCLEAR POWER PLANT

PUBLIC
PDI-1 Reading
J. Zwolinski/S. Black
M. Gamberoni
OGC
G. Hill (2), T-5 C3
W. Beckner, 013/H15
ACRS
L. Doerflein, RI
L. Kopp
F. Akstulewicz
cc: Plant Service list

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
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 8, 2000, as supplemented April 5, 2000, and October 25, 2000, 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. 79 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to June 30, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Peter S. Tam for/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 7, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.7-29	3.7-29
3.7-30	3.7-30
3.7-31	3.7-31
3.7-31a	3.7-31a
3.7-31b	3.7-31b
3.7-31c	3.7-31c
3.7-31d	3.7-31d
3.7-31e	3.7-31e
3.7-31f	3.7-31f
3.7-31g	3.7-31g
3.7-31h	3.7-31h
3.7-31i	3.7-31i
4.0-2	4.0-2
4.0-3	4.0-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 79 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 March 8, 2000, as supplemented April 5, 2000, and October 25, 2000, the Rochester Gas and Electric Corporation (RG&E or the licensee) submitted a request for changes to the R. E. Ginna Nuclear Power Plant Technical Specifications (TSs). The requested changes would allow use of credit for soluble boron in the spent fuel pool criticality analysis. This criticality analysis was performed using methodology analogous to that developed by the Westinghouse Owners Group (WOG) and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology." The April 5, 2000, and October 25, 2000, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The Ginna spent fuel storage pool contains two regions with four rack types. Region 1 consists of five type 3 storage racks nominally employing borated stainless steel panels in each storage cell as neutron absorbers. Region 2 consists of six type 1 racks containing the neutron absorber, Boraflex, within two of the walls of each cell, two type 2 racks employing borated stainless steel panels similar to the type 3 racks, and six type 4 racks similar to the type 2 racks (not yet installed).

RG&E has previously notified the Nuclear Regulatory Commission (NRC) that testing of Boraflex panels in Region 2 indicated degradation such that certain portions of the spent fuel pool licensing basis criticality analysis may no longer be conservative. This included the ability to maintain k_{eff} less than or equal to 0.95 if flooded with unborated water. Interim compensatory measures were approved by the NRC which included a requirement in TS 3.7.12 to maintain 2300 ppm of boron in the pool with a surveillance frequency of 7 days and a footnote to TS 4.3.1.1(b) allowing the k_{eff} requirement to be met with the pool borated to at least 2300 ppm.

The purpose of this license amendment request is to reflect the revision of the criticality analysis and rack utilization schemes for Regions 1 and 2 assuming no credit for Boraflex and taking partial credit for the presence of soluble boron in the pool water. Region 1, with 294 storage cells, is designed to accommodate new (unirradiated) or spent fuel utilizing a checkerboard arrangement. Region 2, with 1075 storage cells, is designed to accommodate

fuel of various initial enrichments which have various accumulated burnups and decay times. Decay time refers to the time period for which the fuel assembly has been residing since irradiation from power operation in the reactor. The Ginna spent fuel storage racks were analyzed using criticality methodology analogous to that which has been reviewed and approved by the NRC (Ref. 3). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analysis and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level; and

2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The analysis of the reactivity effects of fuel storage in the Ginna spent fuel racks was performed with SCALE-PC, a personal computer version of the SCALE-4.3 code package (which includes KENO-Va, NITAWL, CSAS-2, and BON-AMI), with the 44-group ENDF/B-V neutron cross section library. Since the KENO-Va code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code, DIT, which uses an 89-group structure collapsed from the ENDF/B-VI library. The SCALE-PC models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the Ginna racks are designed and have been found to adequately reproduce the critical values. The selected critical experiments included the Babcock & Wilcox experiments carried out in support of close proximity storage of power reactor fuel and the Pacific Northwest Laboratory program carried out in support of the design of fuel shipping and storage configurations. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to Ginna storage rack conditions. The DIT code and its cross section set have been used in the design of reload cores and extensively benchmarked against operating reactor history and test data. 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.

The following assumptions were used for the fuel assemblies in the rack reactivity calculations. The design basis fuel assembly for the fresh storage cells in Region 1, type 3 racks was taken to be a representation of the Westinghouse optimized fuel assembly, a 14x14 fuel assembly having a nominal enrichment of 5 weight percent (w/o) U-235, no integral fuel burnable absorbers (IFBAs), and the instrument tube location replaced by a fuel rod. The design basis fuel assembly for the irradiated fuel storage cells in both Region 1 and 2 was taken to be an approximation to the Westinghouse standard 14x14 fuel assembly wherein the control rod guide tubes were represented as Zircaloy-4 and the instrument tube was replaced by a fuel rod. The staff concludes that these are conservative assumptions with respect to reactivity.

A methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of spent fuel pool water temperatures (50 °F to 212 °F) were included. Uncertainties due to fuel assembly manufacturing tolerances, rack fabrication tolerances, and KENO methodology were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC guidelines and are, therefore, acceptable.

The following regions and rack types were analyzed to assure a maximum value of k_{eff}, including all biases and uncertainties, less than 1.0 with no credit for soluble boron (0 ppm).

The Region 1, type 3 racks consist of a checkerboard array of fresh fuel (5.0 w/o U-235) with no burnable absorbers alternating with irradiated assemblies with a fresh equivalent enrichment equal to 2.22 w/o U-235. The burnup versus initial enrichment curve for burned fuel assemblies checkerboarded with fresh fuel assemblies in Region 1, type 3 cells is shown in proposed TS Figure 3.7.13-1. Fuel assemblies with minimum burnups above the curve (area A) may be stored in any location in Region 1. Assemblies with minimum burnups below the curve (area B) may only be stored in cells with lead-in funnels.

For storage in the Region 2, type 1 cells, the acceptable combination of initial enrichment, burnup and decay time are shown in proposed TS Figures 3.7.13-2 through 3.7.13-6. Fuel assemblies with initial enrichments, burnups and decay times within domain A1 of the figures may be stored in any type 1 cell. Assemblies with initial enrichments, burnups and decay times, within domain A2 must be stored face-adjacent to a type A1 or A2 assembly, or a water cell (empty cell). Face-adjacent means that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. Fuel assemblies with initial enrichments, burnups and decay times, within domain B of these figures must be stored face-adjacent to a type A1 assembly, or a water cell. Fuel assemblies with initial enrichments, burnups and decay times, within domain C must be stored face-adjacent to a water cell only.

For storage in the type 2 and type 4 cells in Region 2, the acceptable combination of initial enrichment, burnup and decay time are shown in proposed TS Figures 3.7.13-7 through 3.7.13-11. Assemblies with initial enrichments, burnups and decay times within domain A1 may be stored in any location in Region 2 type 2 and type 4 cells. Fuel assemblies with initial enrichments, burnups and decay times within domain A2 of the figures must be stored face-adjacent to a type A1 or A2 assembly or a water cell. Assemblies with initial enrichments, burnups and decay times within domain B of Figures 3.7.13-7 through 3.7.13-11 must be stored face-adjacent to a type A1 assembly or a water cell. Assemblies with initial enrichments, burnups and decay times within domain C must be stored face-adjacent to a water cell only.

Consolidated rod storage canisters may also be stored in either region provided that the minimum burnup requirements of proposed Figures 3.7.13-1 through Figure 3.7.13-11 are met.

Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including 95/95 uncertainties. To maintain this 5 percent subcritical margin, 377 ppm of soluble boron is required. This boron value is well below the minimum spent fuel pool boron concentration value of 2300 ppm required by TS 3.7.12 and is, therefore, acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to define the conditions under which fresh and irradiated fuel assemblies are interchangeable on an overall reactivity basis. The NRC has previously accepted the use of reactivity equivalencing to equate an array of fresh fuel assemblies and their enrichments, that have been shown to be acceptable for storage, into an array of irradiated assemblies with different initial enrichments, decay times, and burnable absorber concentrations. To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies with enrichments higher than those acceptable for storage of fresh assemblies, a series of reactivity calculations were performed to generate a set of enrichment versus fuel assembly discharge burnup ordered pairs which all

yield an equivalent k_{eff} when stored in the spent fuel storage racks. For conservatism, the nuclide inventory assumed no Xe-135, peak Sm-149, and peak Pu-239, thereby maximizing the assembly reactivity.

The analysis also included spent fuel decay time credit, which results from the radioactive decay of isotopes in the spent fuel to daughter isotopes. The loss in reactivity due to the radioactive decay of the spent fuel results in reducing the minimum burnup needed to meet the reactivity requirements. The reactivity of an irradiated fuel assembly will decrease following its discharge from the reactor and the decay of short lived fission products due to the decay of actinides and long half-life fission products. For long cooling periods, the decay of Pu-241, with a half-life of approximately 14 years, to Am-241 is the most important contribution to a reduction in fuel assembly reactivity. The minimum required burnups for initial assembly enrichments up to 5.0 w/o U-235 for Region 2 are shown in TS Figures 3.7.13-2 through 3.7.13-11. These burnup-enrichment pair data also include credit for Pu-241 decay from 0 to 20 years.

Uncertainties associated with burnup credit include a reactivity uncertainty of 0.005 Δk at 30,000 MWD/MTU (megawatt days per metric ton uranium) applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties. An uncertainty of 5 percent was also applied to the calculated burnup to account for burnup measurement uncertainty. The NRC staff concludes that these uncertainties conservatively reflect the uncertainties associated with burnup calculations and are acceptable.

Since a majority of spent fuel assemblies discharged from Ginna have a burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative, an eight node axial model was implemented for the spent fuel pool burnup calculations. Axial burnup profiles provided by RG&E were used in the analyses.

The amount of additional soluble boron that is needed to account for these reactivity equivalencing uncertainties is 207 ppm. Adding this to the soluble boron credit of 377 ppm required for k_{eff} to be less than or equal to 0.95 results in a total soluble boron credit of 584 ppm.

An evaluation of various fuel mishandling accidents indicated that the misplacement of a fresh fuel assembly enriched to 5.0 w/o U-235 in a type 1 location results in the highest reactivity increase. However, the minimum spent fuel pool boron concentration value of 2300 ppm required by TS 3.7.12 is more than sufficient to maintain k_{eff} less than or equal to 0.95 for this reactivity increase. In fact, an additional 381 ppm of soluble boron is sufficient to maintain k_{eff} less than or equal to 0.95 for this reactivity increase. By virtue of the double contingency principle, which has been endorsed by the staff, two unlikely independent and concurrent events are beyond the scope of the required analysis. Therefore, credit for the presence of the entire 2300 ppm of soluble boron may be assumed in evaluating other accident conditions such as a fuel misplacement. However, for conservatism, an additional 381 ppm of boron was added to the 584 ppm determined above, resulting in a total required boron concentration of 965 ppm. In addition, this soluble boron concentration was increased by 1 percent due to the difference in the B-10 atom percent used in the analysis (19.9) and that measured at Ginna (19.7), resulting in a total soluble boron concentration of 975 ppm.

The NRC safety evaluation report for crediting soluble boron (Ref. 3) states that potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit should be identified. In addition, the available time span of these dilution events should be quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

Deterministic plant-specific dilution event calculations were performed for Ginna in order to define the dilution times and volumes necessary to dilute the spent fuel pool from the minimum TS boron concentration of 2300 ppm to a soluble boron concentration where a k_{eff} of 0.95 would be approached (975 ppm). The water volume in the pool is approximately 213,600 gallons. In order to dilute 213,600 gallons in the spent fuel pool from the TS limit of 2300 ppm to 975 ppm, 183,000 gallons of unborated water is needed. The largest source of unborated water would be the condensate storage tanks, with a combined volume of 160,000 gallons. This is less than the dilution volume of 183,000 gallons and is, therefore, not capable of diluting the pool to 975 ppm.

The licensee's evaluation of potential dilution scenarios concluded that the largest possible flow rate for dilution of the pool is 290 gpm due to a through wall crack in the 8-inch fire main. This line does not run above the pool, but it is conservatively assumed that the break flow jet impinges on the pool surface. The fire system can draw an essentially limitless supply of water from the lake. At a flow rate of 290 gpm, over 10 hours would be required to dilute the pool from the TS limit of 2300 ppm to 975 ppm. A rupture of this fire main would result in many alarms which would alert the operators to this potential dilution event in minutes. The expected alarms include auto start of the fire pumps and high spent fuel pool level. Ten hours is more than sufficient for the operators to identify the location of the pipe break and stop the fire pumps or isolate the affected piping.

The licensee concluded that an event that could dilute the spent fuel pool boron concentration from 2300 ppm to 975 ppm is not a credible event. The NRC staff concurs that the combination of the large volume of water required for a dilution event, TS required minimum boron concentration, weekly sampling requirement, spent fuel pool alarms, plant personnel rounds, and other administrative controls and procedures, will adequately detect a dilution event prior to k_{eff} reaching 0.95 (975 ppm). Therefore, the proposed boron dilution analysis is acceptable.

Additionally, the criticality analysis for the spent fuel pool shows that k_{eff} remains less than 1.0 at a 95/95 probability/confidence level even if the pool were completely filled with unborated water. Therefore, even if the spent fuel storage pool were diluted to zero ppm, the stored fuel is expected to remain subcritical, thereby meeting the requirements of Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), Appendix A, General Design Criterion (GDC) 62.

2.1 Conclusion

Based on the review described above, the staff finds the criticality aspects of the proposed Ginna license amendment requesting credit for soluble boron are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The criticality analysis conforms to the NRC guidance for the assurance of criticality safety in spent fuel storage pools (Ref. 4). Based on the above evaluation, the proposed changes to TSs 3.7.13 and 4.3 are acceptable.

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

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 17918). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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.

6.0 REFERENCES

1) Robert C. Mecredy, RG&E, letter to U.S. Nuclear Regulatory Commission, “Application for Amendment to Facility Operating License, Credit for Soluble Boron in Spent Fuel Pool, Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant, Docket No. 50-244,” March 8, 2000.

2) Robert C. Mecredy, RG&E, letter to U.S. Nuclear Regulatory Commission, “Corrected Pages Associated with the Credit for Soluble Boron in Spent Fuel Pool, Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Power Plant, Docket No. 50-244,” April 5, 2000.

3) Newmyer, W. D., “Westinghouse Spent Fuel Rack Criticality Analysis Methodology,” Westinghouse Electric Corporation, WCAP-14416-NP-A, November 1996.

4) L. Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” NRC Memorandum to T. Collins, August 19, 1998.

Principal Contributor: L. Kopp

Date: December 7, 2000