Docket No. 50-244

August 8, 1991

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Dr. Mecredy:

SUBJECT: ISSUANCE OF AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-18 - R. E. GINNA NUCLEAR POWER PLANT (TAC NO. 67427)

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your letters dated January 19, 1988 as supplemented on October 5, 1989, March 28, June 29, July 11, August 19 and 28, September 17, December 6, 1990, and January 8 and 28, March 8, and April 15, 1991.

This amendment extends the Facility Operating License expiration date from April 25, 2006 to September 18, 2009.

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

Sincerely,

Original signed by Allen R. Johnson

Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 44 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

LA:PDI-3*	PM:PDI-3*	EMCB:**	ESGB*	SRXB*	OGC	PD:PDI-3
MRushbrook	AJohnson	CYCheng	GBagchi	RJones	CPW	SShankman
07/10/91	07/9/91	8/6/91	7/12/91	8/8/91	7/23/91	8/8/91

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GINNA AMEND.PKG.TAC NO.67427

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

August 8, 1991

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Vice President, Nuclear Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

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Enclosures:
1. Amendment No. 44 to
   License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
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 (RG&E) (the licensee) dated January 19, 1988 and as supplemented on October 5, 1989, March 28, June 29, July 11, August 19 and 28, September 17, December 6, 1990, and January 28, March 8, and April 15, 1991, 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 set forth in 10 CFR Chapter I;

   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 a change to paragraph 2F. of Facility Operating License No. DPR-18 and is hereby amended to read as follows:

   This license is effective as of the date of issuance and shall expire at midnight September 18, 2009.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Susan F. Shankman, Director (Acting)
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Amended Page 6 of license

Date of Issuance: August 8, 1991

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27827 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Robert Emmet Ginna Nuclear Plant Physical Security Plan," with revisions submitted through August 18, 1987; "Robert Emmet Ginna Nuclear Plant Guard Training and Qualification Plan" with revisions submitted through July 30, 1981; and "Robert Emmet Ginna Nuclear Plant Safeguards Contingency Plan" with revisions submitted through April 14, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is effective as of the date of issuance and shall expire at midnight, September 18, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

Attachment:
Appendix A - Technical Specifications

Date of Issuance: December 10, 1984

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 44 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 January 19, 1988, and as supplemented on October 5, 1989, March 28, June 29, July 11, August 19 and 28, September 17, December 6, 1990, and January 8 and 28, March 8, and April 15, 1991, Rochester Gas and Electric Corporation (RG&E) requested an amendment to Facility Operating License No. DPR-18 to extend the expiration date of the license from April 25, 2006 to September 18, 2009. The proposed amendment would extend the operating license (OL) for an additional three years and five months extending the operating license for a full 40 year period from date of issuance.

2.0 DISCUSSION

Section 103.c of the Atomic Energy Act (Act) of 1954 provides that a license is to be issued for a specific period not exceeding 40 years. Title 10 CFR 50.51 specifies that each license will be issued for a fixed period of time not to exceed 40 years from date of issuance. The current term for the R.E. Ginna Nuclear Power Plant is 40 years commencing with the April 25, 1966 issuance of the Construction Permit (CP). This represents an effective OL term of approximately 36 years and seven months. Consistent with the Act and 10 CFR 50.51 of the Commission's regulations, the licensee's proposed amendment of January 19, 1988 and as supplemented on October 5, 1989, March 28, June 29, July 11, August 19 and 28, September 17, and December 6, 1990, January 8 and 28, March 8, and April 15, 1991, seek an extension of the OL term for Ginna such that the fixed period of the license would be 40 years from the date of OL issuance.

A 40 year term commencing with the date of OL issuance would change the expiration date from April 25, 2006 to September 18, 2009 for an extension of three years and five months, the interval between issuance of the CP and OL.

3.0 EVALUATION

The NRC staff has evaluated the environmental impact and safety issues associated with issuance of the proposed license amendment which would allow approximately three additional years and five months of operation. In addressing the environmental impact the following was considered: 1) the need for proposed action; 2) radiological impact; 3) non-radiological impact; 4) alternate to the proposed action; 5) alternate use of resource; 6) other agencies and persons contacted; and 7) the basis for not preparing an environmental impact statement. This information is provided in the NRC staff's Environmental Assessment (EA) published in the Federal Register on April 24, 1991 (56 FR 18841). The following addresses the safety issues associated with the proposed amendment.

3.1 Reactor Vessel

3.1.1 Design and Fabrication

The R.E. Ginna Nuclear Power Plant was designed and constructed primarily on the basis of 40 years of plant operation. The reactor vessel was designed and fabricated for a 40 year life.

3.1.2 Vessel Materials Surveillance Program

A comprehensive vessel materials surveillance program is maintained in accordance with 10 CFR Part 50, Appendix H, ("Reactor Vessel Materials Surveillance Program Requirements.") The integrity and performance capability of the ferritic materials in the reactor vessel is assured when the fracture toughness is monitored with a surveillance program in conformance with these recommendations. The ferritic materials must meet the fracture toughness properties of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR 50, "Fracture Toughness Properties."

In accordance with the reactor vessel material surveillance program, specimens of the vessel base metal, the heat affected zone metal, the weld metal, and neutron monitor wires, are placed in capsules near the core beltline of the vessel. Selected specimen capsules are removed at intervals over the lifetime of the reactor and tested to determine the extent of the neutron embrittlement of the vessel materials.

Capsules V, R, and T in Ginna's surveillance program have been removed and tested. Capsule P will be removed at about 17 effective full power years (EFPY) (now targeted for the 1992 refueling outage) and capsules S and N are spares. The capsules have contained required number and types of beltline materials. The staff has determined that the material surveillance program at Ginna satisfies Appendix H to 10 CFR 50.

3.1.3 Fast Neutron Fluence for Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events (10 CFR 50.61)

Based on the NRC staff's evaluation of the surveillance capsule test data (paragraph 3.1.2 above), the nil-ductility transition reference temperature at end of life is within the screening criterion for pressurized thermal shock (PTS) events as required by 10 CFR 50.61. The pressure vessel service life is based on 32 EFPY of operation with regard to fast neutron induced brittle fracture conditions, at an assumed load factor of 0.80 equivalent to 40 calendar years of operation. The licensee submitted the required materials and fluence information by letter dated January 13, 1986, as required by 10 CFR 50.61. The NRC staff concluded after review of the information, that the Ginna pressure vessel meets the screening criteria of 10 CFR 50.61 for 40 calendar years of operation. The NRC staff therefore has found the licensee's amendment request for license extension acceptable.

3.1.4 Pressure-Temperature (P-T) Limits and Associated Low Temperature Overpressure (LTOP) Setpoint for Reactor Coolant System in the Technical Specifications

The nil-ductility transition reference temperature is also used in constructing the P-T limits and associated LTOP setpoint for the reactor coolant system in the Technical Specifications. Generic Letter 88-11 requires the licensee to use Regulatory Guide 1.99, Rev.2, to calculate the reference temperature. The staff concludes that the current P-T limits in the current Ginna Technical Specifications are acceptable to use up to the next refueling outage in March 1992. By letter dated, February 15, 1991, as supplemented on March 26 and May 14, 1991, the licensee requested an amendment to revise the P-T limits in response to Generic Letter 88-11. The staff has also evaluated this submittal and concluded that the revised P-T limits and associated LTOP setpoint for heatup, cooldown, leak test and criticality satisfy Generic Letter 88-11 and Appendix G to 10 CFR 50. The staff anticipates issuance of the requested license amendment under separate cover by October 1991.

3.1.5 Charpy Upper Shelf Energy

Section IV.B of Appendix G, 10 CFR 50, requires that the Charpy upper shelf energy (USE) at end of life be above 50 ft-lb. Presently the limiting USE in Ginna is above the 50 ft-lb limit. The licensee, however, predicts that the USE of the limiting beltline material, Linde 80 weld, at the end of 40 years will be below 50 ft-lb as shown in the Babcock & Wilcox (B&W) report, BAW-1803. The licensee has joined the Babcock & Wilcox Owners Group (B&WOG) which has initiated fracture mechanics studies of the low USE issue for all B&W fabricated reactors with Linde 80 welds. B&W will provide resolution to show whether the limiting USE of the reactor vessel material at Ginna will satisfy the 50 ft-lb requirement.

Should the B&WOG resolution indicate that the limiting USE at Ginna will be below the 50 ft-lb limit, Appendix G requires the licensee to show that the reactor vessel is designed to permit a thermal annealing treatment at a sufficiently high temperature to recover fracture toughness properties of beltline materials. In addition, the licensee will also perform the following to satisfy the requirements in Section V.C. of Appendix G: 1) a volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B.; 2) additional fracture toughness tests; and 3) an analysis that shows the existence of equivalent margin of safety for continued operation.

The licensee has a thermal annealing plan for the reactor vessel, prepared by Westinghouse and EPRI as shown in EPRI Report No. NP-6113-M, "Thermal Annealing of an Embrittled Reactor Vessel." Should the B&WOG resolution show that USE falls below 50 ft-lb, the licensee has committed to perform an analysis to show the existence of margins of safety for continued operation. The staff has determined that the licensee is satisfying the provisions of Appendix G.

3.1.6 Reactor Vessel Inspection

During the 1989 refueling outage, the licensee completed the Second 10-year Interval (1980-1989) inservice inspection of the reactor vessel according to the requirements of the ASME Boiler and Pressure Vessel Code (Code), Section XI (1974 edition including addenda through Summer 1975) and Regulatory Guide 1.150, Rev.1. The licensee has performed a 100 percent examination of the reactor vessel beltline weld with no recordable indications found.

The licensee reexamined an existing flaw indication in the weld of the inlet nozzle N2B of the reactor vessel. The staff concluded that the flaw indication was an embedded volumetric reflector resulting from vessel fabrication. The staff also concluded that flaw growth would be negligible during the remaining life of the reactor vessel. A visual examination of the internal surfaces, nozzles and reactor internals and an ultrasonic examination of all accessible welds were also performed.

The results of this inspection were found acceptable by the staff.

3.2 Inservice Inspection (ISI) Program

The licensee submitted the Third 10-Year Interval (1990-1999) ISI Program on July 21, 1989, and January 16, 1990. The NRC staff reviewed and evaluated the program for compliance with 10 CFR 50.55a(g) and Technical Specification 4.2.1.5.

The ISI Program Plan has been evaluated for (a) application of the correct Section XI Code edition and addenda, (b) compliance with examination and test requirements of Section XI, (c) acceptability of the examination sample, (d) compliance with prior ISI commitments made by the licensee, (e) correctness of the application of system or component examination exclusion criteria, and (f) adequate information in support of requests for relief from impractical Section XI Code requirements.

The staff concluded that the Ginna Third 10-year Interval ISI Program, constitutes a basis for compliance with 10 CFR 50.55a(g) and Technical Specification 4.2.1.5, and was therefore found acceptable.

3.3 Inservice Testing (IST) Program for Pumps and Valves

The licensee's Third 10-year Interval (1990-1999) IST Program for pumps and valves, to meet the 1986 Edition of ASME Code, Section XI, was approved by the NRC on April 15, 1991. The IST Program approved 39 relief requests and 31 cold shutdown justifications. The licensee's Third 10-year Interval (1990-1999) IST Program was found acceptable by the NRC staff.

3.4 Potential Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

The licensee has submitted a response to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." Bulletin 88-02 requests that licensees for plants with Westinghouse steam generators employing carbon steel support plates take certain actions (specified in the bulletin) to minimize the potential for a steam generator tube rupture event caused by a rapidly propagating fatigue crack such as occurred at North Anna Unit 1 on July 15, 1987. Based on the review of the Westinghouse reports, WCAP-11802 and WCAP-11803, the staff concluded that the actions taken by the licensee resolved the issues identified in Bulletin 88-02 and are, therefore, acceptable.

3.5 Containment Structural Integrity

In June 1990, the NRC staff performed an inspection of the exterior containment structure of the Ginna plant. At the inspection, a number of concerns were raised by the NRC. In particular the concerns focused on the intrusion of groundwater through the construction layers of the concrete foundation mat and its presence on the cylinder-to-foundation ring beam connection. The licensee has determined that the ground water intrusion is not through the concrete foundation mat and and has taken measures to control the water accumulation. Since these measures including the physical work have been completed, no standing water is present on the foundation joint. The staff accepts the licensee assessment that the water accumulation on the ring-beam near the containment base is not due to the intrusion through the concrete foundation mat. The licensee has committed to monitor this condition to prevent any occurrence of conditions that may affect the structural integrity of the containment.

The behavior of the cylinder-to-ring beam connection has been extensively analyzed by the licensee. The results indicate that under certain boundary conditions, the meridional moments under some loading combinations can be higher than what was considered in the original design near the base. The licensee claims that such hypothetical conditions do not represent the actual behavior. The staff is currently reviewing various aspects of this issue in a separate activity. However, based on the information submitted by the licensee (including the results of surveillance of prestressing tendons) and staff judgement, the staff concludes that the sections near the base can withstand the postulated Loss-of-Coolant Accident (LOCA) and Safe Shutdown Earthquake (SSE) loads without jeopardizing the containment function.

The licensee has also evaluated the effects of higher than the originally postulated ground water level on Category I structures. The results of the evaluations indicate that the Category I structural foundations can accommodate the higher hydrostatic loads without jeopardizing their safety functions.

The staff has found no evidence of any accelerated degradation mechanism that could affect the life expectancy of the containment structure at least up to 40 years beyond the date of issuance of the operating license. The staff, however, will continue to monitor the licensee's efforts to assess the most likely behavior of the containment at the base.

3.6 Boric Acid Corrosion

On October 1989, the NRC staff audited the Ginna Nuclear Power Plant against their program to prevent boric acid-related corrosion and their commitment to an acid leakage monitoring and corrosion prevention program. Ginna is adequately implementing a program for monitoring small primary coolant leakage through carbon steel components caused by boric acid corrosion.

3.7 Electric Equipment

Aging analysis and type testing have been performed on all safety-related Class 1E electrical equipment, identified at the Ginna Station for meeting postulated accident conditions in addition to a full 40-year service life, in accordance with 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." Qualified lifetimes for scheduled component replacement purposes have been identified for equipment as part of the 10 CFR 50.49 program. Since all safety-related electrical equipment and components have scheduled maintenance and replacement procedures, the NRC staff concludes that all electrical equipment used in safety-related applications at the Ginna Station is ensured to be qualified for the life of the plant including the extension period of this amendment.

3.8 Licensee Nuclear Improvement Programs

The licensee has demonstrated, by its Nuclear Improvement Programs and additional resource expenditures in its fiscal 1990-1991 budget, a willingness for plant improvements and projected staffing additions. One such program involves an RG&E Configuration Management Program which implements reconstitution of 1) design bases documents; 2) safety classification and setpoint verification; 3) procedure updates and change control; 4) piping and instrument and electrical drawings; 5) vendor manuals; and (6) document control and information access. In addition, the licensee has implemented an effective RG&E Reliability Centered Maintenance Program.

3.9 Licensee Maintenance and Surveillance Programs

The licensee's corporate and site management strongly support the Ginna Station maintenance and surveillance programs. Major initiatives include the above RG&E Configuration Management Program, involving procedure upgrading, reliability centered maintenance, and the replacement of aged and worn equipment. An upgrade of calibration procedures has been completed with a strong program to upgrade the remaining maintenance procedures. Overall maintenance is being well-planned and adequately accomplished by competent workers with corrective actions accomplished in timely fashion. The NRC concludes that the RG&E maintenance and surveillance programs are adequate, capably performed, and improving.

3.10 NRC Team and Special Inspections

NRC team and special inspections have been performed periodically to provide additional assurance that abnormal or unanticipated degradation will not occur in components or systems required for safe and reliable plant operation.

In September 1988, an NRC Integrated Performance Assessment Team (IPAT) inspection identified several areas of weakness in the licensee's performance. The licensee developed a performance improvement program to address these issues. In December 1989, the licensee had addressed all key initiatives and agreement now exists between the NRC and the licensee with regard to the corrective actions.

In July 1990, an NRC Regulatory Effectiveness Review (RER) was conducted in which the review team concluded that the NRC approved Security Plan is a sound program, well maintained, and reflected a diligent and proactive approach by security personnel.

In October 1989, an NRC emergency operating procedure (EOP) team inspection found the procedures and operators use of them excellent.

In December 1989, an NRC safety safeguards functional inspection (SSFI) team reviewed the operational readiness of the RHR system. The system was found fully operational.

In April 1990, an NRC maintenance team inspection (MTI) documented that the licensee was implementing an effective maintenance program. Upgrades in process included a Reliability Centered Maintenance (RCM) and establishing a broad based Configuration Management Program.

In September 1990, an NRC augmented inspection team (AIT) verified that the licensee's actions, following a reactor trip in which the turbine did not trip immediately, were comprehensive to establish root cause and identify corrective actions.

In December 1990 and January 1991, the NRC conducted a special safety inspection at the plant as a result of a safety significant event involving the disabling of safeguards logic instrumentation, during the conduct of maintenance activity, in which the plant was shutdown. The staff assessed the significance of the event, the adequacy of plant personnel and organizational performance, reviewed the licensee's actions to review the event, and assessed the adequacy of the licensee proposed corrective actions primarily those planned with respect to restart.

In June 1991, the NRC conducted an electrical distribution system functional inspection (EDSFI) in which no significant deficiencies were found in the electrical distribution system.

The NRC staff concludes that the above reported team inspection activities form a bases which assures that the licensee's request for license extension is reasonable.

3.11 Systematic Evaluation Program

The Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The review provides 1) an assessment of the significance of differences between current NRC technical positions on safety issues and those that existed when a particular plant was licensed; 2) a basis for deciding on how these differences should be resolved in an integrated plant review; and 3) a documented evaluation of plant safety. The Ginna Station is a participant with nine other plants in the SEP in that it's provisional operating license was issued in September 1969 before a comprehensive set of licensing criteria for older plants had been developed. Safety evaluation reports (SERs) for 92 topics were issued and a Final Integrated Safety Assessment Report (NUREG-0821, December 1982, and Supplement 1, August 1983) published for the Ginna Station. The SEP evaluation of all 92 topics led to the conclusion that 34 topics were not necessarily consistent with NRC licensing conditions at that time. Since that time safety improvements and upgrades have continually been implemented as a result of this SEP topic review.

In April 1983 an RG&E Structural Upgrade Program (SUP) subsequently windowed select SEP topics to which the SUP was developed to integrate plant improvements for extreme environmental conditions.

The NRC reviewed the SUP and issued an SER in March 1987, summarizing RG&E commitments for improving the Ginna safety-related structures, systems, and components in withstanding a safe shutdown earthquake (SSE). The status was reported in an SER, November 1989, where reanalyses, redesign and resupport was conducted. An NRC audit with regard to load analyses and testing methods of the Ginna plant structures was performed in August 1989 in which the SEP/SUP was completed.

RG&E has also been a participant in a Seismic Qualification Utility Group (SQUG) that has completed a pilot program to explore an alternate method for seismically qualifying selected nuclear plant components based on experience with equipment during earthquakes. Additional work on this program is now underway by SQUG to which RG&E is committed to SQUG methodology.

The NRC staff concludes that the RG&E SEP/SUP program commitments in these areas are acceptable and supports the amendment request for license extension.

3.12 Reactor Operator and Senior Reactor Operator Licensing Examinations

Periodically the NRC administers requalification examinations to employees of the licensee who operate the Ginna Nuclear Power Plant. The evaluations are conducted in accordance with NUREG-1021, Operator Licensing Examiner Standards, ES-601, and has two purposes 1) to evaluate individual operator competency; and 2) to evaluate the effectiveness to the licensed operator requalification training program. Based on the criteria of ES-601, the performance of the operators on examinations indicate that the requalification program is satisfactory, and the facility developed examination material which continues to meet regulatory requirements for continuing operator license renewal.

4.0 SUMMARY AND FINDINGS

The NRC staff concluded in the Environmental Assessment, published in the Federal Register on April 24, 1991 (56 FR 18841), that the annual radiological effects during the additional three years and five months of operation, that would be authorized by the proposed license amendment, are not more than were previously predicted in the original Final Environmental Statement, dated December 1973, and subsequent Environmental Evaluation, dated June 17, 1983.

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register on March 23, 1988 (53 FR 9513).

The staff concludes from its considerations of the design, operation, testing, and monitoring of mechanical and electrical equipment, structures, and reactor vessel that an extension of the operating license for the R.E. Ginna Nuclear Power Plant to a 40-year service life from the date of the Full-Term OL issuance, is consistent with the UFSAR, NRC safety evaluations, supporting amendments, and licensing submittals made by the licensee. Therefore, there is reasonable assurance that the Ginna plant will continue to operate safely for the additional period authorized by this amendment.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

A Notice of Issuance of an Environmental Assessment and Finding of No Significant Impact relating to the proposed extension of the Facility Operating License expiration date for the R.E. Ginna Nuclear Power Plant was published in the Federal Register on April 24, 1991 (56 FR 18841).

7.0 CONCLUSION

The staff 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; and 2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Allen Johnson
C. Y. Cheng
Robert Jones
Goutam Bagchi

Date: August 8, 1991

AMENDMENT NO. 44 TO DPR-18 - R. E. GINNA NUCLEAR POWER PLANT DATED August 8, 1991

DISTRIBUTION:
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