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JAN 20 1972

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN. Mr. Edward J. Nelson
President
89 East Avenue
Rochester, New York 14604

Gentlemen.

A copy of the Notice of Proposed Issuance of Amendment to Provisional Operating License, which is being filed with the Office of the Federal Register for publication, is enclosed for your information. The proposed amendment (copy enclosed) would authorize operation of your R. E. Ginna Nuclear Power Plant No. 1 at power levels up to 1520 megawatts (thermal) and would incorporate revised Technical Specifications (as Change No. 6) to provide for such operation. A copy of the proposed revised Technical Specifications will be forwarded to you as soon as our review of all sections is completed. A copy of our related Safety Evaluation is also enclosed for your information.

Sincerely,

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

Enclosures:
1. Federal Register Notice
2. Proposed Amendment to License
3. Safety Evaluation

cc w/enclosures:
Mr. Eugene B. Thomas, Jr.
LeBoeuf, Lamb, Leiby & MacRae
1821 Jefferson Place, N. W.
Washington, D. C. 20036

Please revised in accordance in license manual to return to OGC

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Docket File

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 20, 1972

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The Atomic Energy Commission ("the Commission") is considering the issuance of an amendment to Provisional Operating License No. DPR-18 which presently authorizes the Rochester Gas and Electric Corporation to possess, use and operate the R. E. Ginna Nuclear Power Plant Unit No. 1 located on the south shore of Lake Ontario in Wayne County, New York, at steady state power levels up to a maximum of 1300 megawatts (thermal). The amendment would authorize Rochester Gas and Electric to operate its R. E. Ginna Nuclear Power Plant Unit No. 1 at steady state power levels up to a maximum of 1520 megawatts (thermal) in accordance with Rochester's application notarized February 2, 1971, and subsequent amendments thereto.

The Commission has found that the application, as amended, for the amendment complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations published in 10 CFR Chapter I. The license amendment will be issued after the Commission makes the findings relating to its review of the application, which are set forth in the proposed amendment, and concludes that the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Within thirty days from the date of publication of this notice in the FEDERAL REGISTER, the applicant may file a request for a hearing and any

person whose interest may be affected by this proceeding may file a petition for leave to intervene. Requests for a hearing and petitions to intervene shall be filed in accordance with the Commission's "Rules of Practice" in 10 CFR Part 2. If a request for a hearing or a petition for leave to intervene is filed within the time prescribed in this notice, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this amendment, see (1) the application for license amendment notarized February 2, 1971, Amendments 1 through 4 thereto, notarized February 17, September 30, October 7, and November 18, 1971, respectively; (2) the Report of the Advisory Committee on Reactor Safeguards dated December 17, 1971; (3) the proposed amendment to the provisional operating license; and (4) a related Safety Evaluation prepared by the Division of Reactor Licensing, all of which are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. A copy of each of items (3) and (4) above may be obtained upon request sent to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Director of Reactor Licensing. As soon as they are available, and prior to issuance of the license amendment, a copy of the proposed revised Technical Specifications will be made available for public inspection at the Commission's Public Document Room.

Dated at Bethesda, Maryland, this 20th day of January 1972.

FOR THE ATOMIC ENERGY COMMISSION

Donald J. Skovholt

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

PROPOSED AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 2
License No. DPR-18

1. The Atomic Energy Commission ("the Commission") has found that:

   A. The application complies with the requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the regulations of the Commission set forth in 10 CFR Chapter I;

   B. Construction of the facility has been substantially completed in conformity with Construction Permit No. CPPR-19, as amended, and the application, the provisions of the Act, and the rules and regulations of the Commission;

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

   D. There is reasonable assurance (i) that the facility can be operated at power levels up to 1520 megawatts (thermal) without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission;

   E. There are involved features and characteristics as to which it is desirable to obtain actual operating experience before the issuance of a full term operating license;

   F. The applicant is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   G. The applicant has furnished proof of financial protection that satisfies the requirements of 10 CFR Part 140; and

   H. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

2. Provisional Operating License No. DPR-18 issued to the Rochester Gas and Electric Corporation is hereby amended in its entirety to read:

A. This license applies to the Robert Emmett Ginna Nuclear Power Plant Unit No. 1, a closed cycle, pressurized, light-water-moderated and cooled reactor, and electric generating equipment (herein referred to as "the facility") which is owned by the Rochester Gas and Electric Corporation (hereinafter "the licensee" or "RG&E"). The facility is located on the licensee's site on the south shore of Lake Ontario, Wayne County, New York, about 16 miles east of the City of Rochester, and is described in license application Amendment No. 6, "Final Facility Description and Safety Analysis Report", and subsequent amendments thereto, and in the application for power increase notarized February 2, 1971, and Amendment Nos. 1 through 4 thereto (herein collectively referred to as "the application").

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

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50 - "Licensing of Production and Utilization Facilities", to possess, use and operate the facility as a utilization facility at the licensee's Brookwood site in Wayne County, New York, in accordance with the procedures and limitations described in the application and in this license;

(2) Pursuant to the Act and 10 CFR Part 70 - "Special Nuclear Material", to receive, possess and use at any one time up to 2300 kilograms of contained uranium 235 and 96 grams of plutonium in encapsulated form as plutonium-beryllium neutron sources, both in connection with operation of the facility;

(3) Pursuant to the Act and 10 CFR Part 30 - "Rules of General Applicability to Licensing of Byproduct Material", to

a. receive, possess and use up to 72.6 microcuries of neptunium 237 composed of six sealed sources contained in irradiation surveillance capsules,

b. receive, possess and use 600 curies of polonium encapsulated in the form of polonium-beryllium as primary source rods in neutron source assemblies,

c. receive, possess and use 65,000 curies of antimony encapsulated in the form of antimony-beryllium as a secondary source, and

d. possess, but not to separate, such byproduct materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

RG&E is authorized to operate the facility at steady state power levels up to a maximum of 1520 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A attached hereto, as Change No. 6, are hereby incorporated in this license as the Technical Specifications. These Specifications supersede those issued with the provisional license on September 19, 1969, and Changes 1 through 5 thereto. RG&E shall operate the facility at power levels up to 1520 megawatts (thermal) in accordance with the Technical Specifications, and may make changes therein only when authorized by the Commission in accordance with the provisions of Section 50.59 of 10 CFR Part 50.

(3) Reports

In addition to the reports otherwise required under this license and applicable regulations, RG&E shall make certain reports as required in accordance with the provisions of the Technical Specifications.

(4) Records

RG&E shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. RG&E shall observe such standards and requirements for the protection of the environment as are validly imposed pursuant to authority established under Federal and State law and as are determined by the Commission to be applicable to the facility covered by this operating license. This condition does not apply to radiological effects since such effects are dealt with in other provisions of this operating license.

E. This license is effective as of the date of issuance and shall expire at midnight, September 19, 1972, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Donald J. Skovholt
Assistant Director for
Reactor Operations
Division of Reactor Licensing

Attachment:
Appendix A - Technical Specifications

Date of Issuance:

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

POWER INCREASE

Date: January 20, 1972

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1.0 INTRODUCTION

On September 19, 1969, the Atomic Energy Commission issued Provisional Operating License (POL) No. DPR-18 to Rochester Gas and Electric Corporation (RG&E) authorizing operation of the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 (Ginna) facility at steady state power levels up to 1300 MWt. By petition notarized February 2, 1971, RG&E requested an amendment of its license to permit operation at steady state power levels up to 1520 MWt.

The Ginna facility was designed initially to operate at a power level of 1520 MWt, but the overall plant performance was evaluated by DRL for 1300 MWt since RG&E chose to request authorization to operate the plant at 1300 MWt. Although our POL review was based on operation at the 1300 MWt power level, the radiological consequences of accidents were evaluated at the stretch rating of 1520 MWt.

Initial criticality of Ginna was achieved on November 9, 1969, and full licensed power level (1300 MWt) was reached on March 12, 1970. The initial low power and full power testing was completed satisfactorily. The observed thermal, hydraulic and nuclear performance of the facility was within predictions. The 100-hour full power demonstration run was completed on March 16, 1970.

We have evaluated the RG&E facility for operation at power levels up to 1520 MWt with the present core loading. This evaluation is based on: review of "Technical Supplement Accompanying Application to Increase Power" dated February 1971 and four amendments thereto notarized February 17, September 30, October 7, and November 18, 1971, submitted by RG&E in support of its application to increase power; review of operation at power levels up to 1300 MWt; and review of the startup test program results.

Our examination of data from startup testing and power operation as reported by the licensee and by the Division of Compliance showed that the data confirmed design predictions and supported the expected capability to operate at the proposed power level of 1520 MWt. Facility operation was continued without serious problems resulting from the fuel cladding failures experienced shortly after the initial 100% power operation. Radioactivity releases have been within the limits of 10 CFR Part 20. Recent experimental hydraulic data regarding the high pressure safety injection system verified the adequacy of the injected flow rate and

flow distribution. We have evaluated the overall ECCS performance and its conformance with "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors" and have concluded that the ECCS as modified is acceptable.

We examined the reanalyses provided by the licensee of all anticipated operational transients affected by the power increase and that might be expected to result from any single operator error or equipment malfunction. The results show that the design and performance objectives will be satisfied during the proposed operation at 1520 MWt. In addition, the design basis accidents were reexamined for the higher power level using current calculational methods, assumptions and criteria. The presently calculated radiological doses resulting from these accidents remain acceptably low at the proposed power level of 1520 MWt. The program for monitoring coolant radioactivity will be improved.

The Advisory Committee on Reactor Safeguards (ACRS) completed its review of the application for operation at power levels up to 1520 MWt at its 140th meeting (December 9-11, 1971). A copy of the ACRS report is attached.

On the basis of our review, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the Ginna facility at steady state power levels up to a maximum of 1520 MWt.

2.0 SITE AND ENVIRONMENT

2.1 Meteorology

The meteorological data presented in the "Final Facility Description and Safety Analysis Report" were reviewed and appropriate diffusion parameters for accident and annual average conditions were calculated. Wind direction and speed at the 50-foot level and vertical temperature difference between the 150- and 10-foot levels were used in the evaluation.

Based on the joint frequency of on-shore occurrence of wind speed and vertical temperature difference, relative concentrations (X/Q) were calculated assuming a ground level point source with a building wake

correction (cA = 440 m^2) for the minimum exclusion distance (450 m) and for the Low Population Zone (LPZ) distance (4800 m). The relative concentrations are 5.3×10^{-4} sec/m^3 at the site boundary and 3.9×10^{-5} sec/m^3 at the LPZ distance. An analysis of the joint frequency of wind direction, wind speed and vertical temperature difference at the site boundary distance produced a maximum annual average relative concentration, (X/Q) of 5×10^{-6} sec/m^3, at the site boundary to the east which is 750 m from the plant. These calculated values differ from those used during the POL evaluation primarily because of the use of only on-shore occurrence of winds rather than all wind direction occurrences for the accident conditions and a more rigorous analysis of the joint frequency for the annual average conditions.

2.2 Environmental Monitoring

RG&E has followed the schedule that was presented in the Technical Specifications for taking and analyzing environmental samples. Data have not been taken over a sufficiently long period of time to determine if a trend has been established.

Current requirements for the monitoring and reporting of effluents will be incorporated in the Technical Specifications.

3.0 REACTOR DESIGN

3.1 Thermal and Hydraulic Analysis for 1520 MWt

Operation of the Ginna reactor at 1520 MWt with rated coolant flow results in thermal and hydraulic core parameters comparable to those of Point Beach Units 1 and 2 (see Table 1). Analysis of measurements obtained from operating Ginna at power levels near 1300 MWt for extended periods of time led to reducing the hot channel factors from those used in the analysis of initial operation to those found acceptable for Point Beach Unit Nos. 1 and 2. Point Beach and Ginna have similar core powers, heat transfer areas, coolant flow rates and coolant temperatures. Ginna was considered a prototype for Point Beach. The proposed design rating core parameter values for Ginna at 1520 MWt are equivalent to Point Beach Unit Nos. 1 and 2 at 1518.5 MWt and therefore represent no extension of accepted PWR thermal and hydraulic operating limits.

We have compared the transient analyses that were revised for 1520 MWt operation with the analyses presented for the 1300 MWt operation and

TABLE 1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Current	Proposed	Point Beach 1 & 2
Total Heat Output, MWt	1300	1520	1518.5
Linear power density at over power conditions, kW/ft	18.5	18.5	17.9
Nominal System Pressure, psia	2250	2250	2250
Hot Channel Factors			
Heat Flux			
Nuclear, F_q^N	3.28	2.72	2.72
Engineering, F_q^E	1.03	1.03	1.03
Total	3.38	2.80	2.8
Enthalpy Rise			
Nuclear, $F_{\Delta H}^N$	1.75	1.66	1.6
Coolant Flow			
Total Flow Rate, lbs/hr	67.3×10^6	68.0×10^6	66.7×10^6
Coolant Temperature, °F			
Nominal Inlet	551.9	544.5	552.5
Average Rise in Core	52	60.5	60
Heat Transfer			
Active Heat Transfer Surface Area, ft^2	28,715	28,715	28,715
Maximum Heat Flux, Btu/hr-ft^2	508,000	494,800	491,000
Linear power density, kW/ft	16.5	16.0	16.0
Fuel Central Temperatures, °F			
Maximum at 100% Power	3880	~3750*	~3750
DNB Ratio - Minimum Ratio During			
Nominal Operating Conditions	2.15	2.06	2.11
Transients	1.3	1.3	1.3

*Evaluated for pressurized fuel rods BOL in Region 4. These values are equal to or greater than for non-pressurized rods.

with comparable analyses for Point Beach. The limiting design and performance criteria were not changed from the 1300 MWt analyses and were met for the transients analyzed at 1520 MWt. The following abnormalities and transients were reanalyzed completely: (1) uncontrolled RCCA withdrawal at power, (2) loss of reactor coolant flow and (3) loss of external electric load. The reactor control and protective system prevents any anticipated transient from causing a DNB ratio of less than 1.3.

The predicted reactivity control and core nuclear characteristics were compared with results from startup physics test and found to be in agreement. The nuclear characteristics are not affected significantly by the proposed power increase. The shutdown margin and maximum control rod reactivity worth are within the Technical Specification requirements.

We have concluded that the Ginna reactor core can be operated safely at steady state power levels up to 1520 MWt on the basis of: the satisfactory core performance results from startup testing and operation for more than a year at power levels up to 1300 MWt and the consequences of the transients evaluated for this application.

3.2 Fuel Performance

Within the first two weeks of operation at 100% power (1300 MWt), a rise in coolant radioactivity was observed. A delayed neutron monitor and the coolant radioactivity measurements indicated about 20 fuel failure events occurred within a two-week period; no additional failures were detected during subsequent reactor operation. During a partial refueling in March 1971, RG&E unloaded the entire core and identified assemblies with clad failures. RG&E replaced 12 of the leaking assemblies with new fuel assemblies during the shutdown and made plans to replace the remaining leaking assemblies during the next refueling in the spring of 1972.

We have concluded that the fuel cladding will not be affected adversely by operation at 1520 MWt and that the fuel assemblies with clad leaks presently in the core may be operated at power levels up to 1520 MWt.

4.0 REACTOR COOLANT SYSTEM

4.1 Primary System Leakage

Design features were included in the Ginna design to facilitate rapid and accurate detection of primary coolant leakage into containment.

Monitoring of the coolant inventory and coolant makeup system has shown the leak rate to be 0.1 to 0.2 gpm. A doubling in this leak rate would be detected within a few hours by operators observing the increased frequency of makeup water addition. Air particulate and radioactive gas monitors have rapidly indicated coolant leakage or increases in coolant leak rates by changes in radioactivity measurements.

Monitoring of drain collection lines and the capability for personnel to inspect equipment in containment provides assurance that the source of leakage will be identified rapidly and that there will be a minimal amount of unidentified leakage. Thus, we have concluded that the primary coolant leakage monitoring capability in the Ginna facility is acceptable.

4.2 Inservice Inspection Program

The reactor vessel is designed to be inspectable from the inside surface by removing the vessel internals. Access has been provided for all other critical areas in the primary coolant boundary such as vessel nozzles and dissimilar metal welds. The applicant has identified the specific components to be inspected, the frequency of inspection and the type of inspections to be performed for the critical items listed under Examination Categories, and Component Parts and Methods of Inspection of the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Reactor Coolant Systems".

Development work is in progress to assure the availability of remotely operated equipment to perform inspections of areas of high radiation that cannot be inspected by normal means. This proposed program satisfies the provisions of the January 31, 1969 AEC document, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection".

We have concluded that the applicant's inservice inspection program is acceptable.

5.0 CONTAINMENT

5.1 Containment Leakage Rate

The containment structure is a steel-lined concrete vessel with vertical prestressing strands and circumferential steel reinforcing bars. The containment has a design pressure of 60 psig. In the POL application,

RG&E calculated the peak accident pressure to be about 52 psig for 1520 MWt operation. This peak pressure was confirmed by us and the containment design found acceptable for operation at 1520 MWt.

The structure was tested in April 1969 before beginning reactor operation. Containment was pressurized to 69 psig and then was reduced in steps to the test pressures of 60 psig and 35 psig. The respective leak rates were measured as 0.0219 ± 0.0168%/day and 0.0059 ± 0.0180%/day. These rates were less than the maximum allowable leakage rate of 0.25%/day at a 60 psig internal pressure. We have concluded that the containment building has a very low leakage rate that is within the accuracy of the measurements.

Analysis of the loss-of-coolant accident indicated that the allowable containment leakage rate should be reduced from 0.25%/day to 0.20%/day to reduce the calculated doses. The technical specification will be revised to incorporate the 0.20%/day leak rate limit at 60 psig. We have concluded that the Ginna containment with an allowable leak rate of 0.20%/day at 60 psig internal pressure is acceptable for reactor operation at 1520 MWt.

6.0 ENGINEERED SAFETY FEATURES

6.1 Emergency Core Cooling System (ECCS)

RG&E submitted an "Emergency Core Cooling Performance Report" in response to the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors", dated June 19, 1971. The Ginna Performance Report is a copy of the "Emergency Core Cooling Performance" for Point Beach Unit No. 2 and a justification to show that Point Beach Unit No. 2 is representative of the Ginna facility.

A DRL review of the Point Beach report and subsequent discussions between Wisconsin Electric Power (WEP), Westinghouse Electric Corporation and the DRL Staff disclosed that an uncertainty exists in the analysis. The acceptability of the injection point of the high pressure injection system was questioned. The analysis does not establish that the injected water would cool the core because at certain points of injection the water may be entrained with the high velocity steam coming from the core thereby prohibiting the cooling water from reaching the reactor core. The potential consequences of the problem have been evaluated

by RG&E and Westinghouse personnel. As the result of this evaluation, RG&E submitted Amendment No. 4 to the application proposing to delete hot leg high pressure injection of cooling water as a solution to the potential problem. We have reviewed this proposal and we have concluded that it is an acceptable solution to the problem. However, RG&E will continue to analyze this situation to determine the desirability of manually actuated high pressure injection of cooling water into the hot leg under specific accident conditions. The results of these analyses will be reviewed and approved by DRL before such manual injection of cooling water will be permitted.

On this basis, we have concluded that the ECCS operated in the manner proposed is acceptable for reactor operation at a power level up to 1520 MWt.

7.0 INSTRUMENTATION, CONTROL AND POWER SYSTEMS

7.1 Emergency Power

Our review of the Ginna facility indicates that the emergency power system satisfies "IEEE Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" (IEEE-308), and the Commission's General Design Criteria 17 and 18. Therefore, we have concluded that the design of the emergency power system is acceptable.

7.2 Instrumentation Changes

The level measuring instrumentation and associated valve logic circuits at the concentrated boric acid storage tanks have been modified to satisfy the single failure criterion. The modification was prompted by an experienced single failure which prevented the discharge valves from opening on demand.

8.0 AUXILIARY SYSTEMS

8.1 Radioactive Waste Collection and Monitoring

Operation at Ginna with failed fuel assemblies caused the entire radioactive waste collection and monitoring system to be reevaluated by RG&E. During operation, the gross radioactivity of the primary coolant was determined to be about 50% of the Technical Specification limit. Primary system leakage resulted in higher levels of radioactive gases

in the containment and increased the radioactivity in the liquid systems which in turn affected the waste gas system and the ventilation system. As the result of operating experience, numerous changes were made to improve the radwaste systems, and RG&E has expressed its intentions to continue to make modifications to improve the systems as additional operating experience is gained.

Liquid

Radioactive liquid leaking from the primary system collects in a coolant drain tank and/or a sump within containment and a sump and/or miscellaneous collection tanks outside of containment. The contents of the tanks are monitored to determine the level of radioactivity. Low level wastes discharged to Lake Ontario are within acceptable limits (10 CFR 20). High level wastes are sent to the waste evaporator to concentrate the radioactivity in a smaller volume for shipment offsite. The low level waste (condensate) from the evaporator is monitored to establish that it is within acceptable limits and then discharged to the Lake. The volume of concentrated waste (bottoms) from the evaporator has been larger than expected because the evaporator has not performed as designed. This design deficiency became apparent when the radioactivity content of the primary coolant increased due to fuel failures.

RG&E has conducted an extensive evaluation of the liquid waste collection and treatment facilities. The evaporator was modified by the supplier in an attempt to improve its performance. In addition, demineralizers were installed in the low level waste line from the evaporator to reduce further the activity level of the wastes before release to the Lake. Waste liquids were categorized according to the level of radioactivity, boric acid concentration, and contaminants such as detergents and chemicals. Physical changes to the Ginna facility were made to keep the different categories of liquids separated to permit special processing of each type of liquid. The evaluation and improvement of liquid radwaste facilities is a continuing effort. The results to date show that the evaporator performance was improved significantly with a reduced quantity of radioactivity being discharged to the Lake. Liquid releases are significantly less than 10 CFR Part 20 limits and the releases also meet the levels in proposed Appendix I to 10 CFR 50; therefore, we have concluded that the liquid waste treatment is acceptable and the releases are acceptably low.

Gaseous

During reactor operation with failed fuel assemblies, radioactive gases are released from the fuel and distributed in the primary coolant system. These radioactive gases (e.g., iodine, xenon and krypton) may escape the primary coolant system, and eventually reach the environs, by (1) being deliberately discharged to the plant vent header from tanks and other process components that contain radioactive gases, (2) leaking from these same components into the room or area, and (3) leaking into the secondary coolant system. The major mechanisms for release of radioactive gases from the facility have been observed to be the batch discharge of a waste gas decay tank, the batch purge of the containment atmosphere, and the continual release of other plant atmosphere via the auxiliary building ventilation system. There has been no detectable primary to secondary leakage. Measurements of iodine and noble gas concentrations at Ginna indicate that iodine is the most radiologically significant radioisotope. The review carefully considered the procedures used and the effect of modifications that were made to reduce the release of radioactive gases.

The plant vent header is a combination vent and distribution header for the radioactive gases. Gases discharged into the plant vent header may be withdrawn from the vent and used in another component as a cover gas, or if there is more gas available than required to keep the vent header filled, the gas may be compressed and stored in gas decay tanks. This compressed gas later may be withdrawn to maintain the supply of cover gas in the plant vent header, or discharged to the environs. Radioactive gas will be held in the decay tanks for the maximum decay period permitted by operating conditions prior to its discharge to the environs. The gases are sampled and analyzed prior to releasing the contents of the tank to establish that the releases are within the limits of the Technical Specification and are as low as practical.

The radioactive gases that leak from the primary system components in the containment will be retained in the containment until the atmosphere must be vented for personnel access. Prior to purging containment, the atmosphere is sampled and analyzed to determine the particulate and radiogas levels. The atmosphere may be recycled through "Kidney" charcoal adsorbers to reduce the iodine concentration and under one set of operating conditions, the radioactive iodine concentration was reduced by about a factor of 10. When containment is purged, the air flow is through full flow charcoal adsorbers.

A survey of the plant areas outside containment revealed points where radioactive gases were being released to the rooms. Leakage from equipment was reduced and some of the releases from vents were routed to an existing charcoal filter ventilation system that serves a portion of the auxiliary building. RG&E is evaluating the benefits of installing charcoal adsorbers elsewhere in the exhaust system. The combined ventilation exhaust air from the auxiliary building is monitored for particulates and radioactive iodine by a monitor containing a charcoal cartridge. A laboratory measurement of the charcoal cartridge indicated that the field monitor was reading low on at least one occasion. Thus, the technical specification limit for I-131 releases may have been exceeded on occasion, but the releases have been within the annual limit of 10 CFR 20. Since this occurrence, RG&E has established administrative controls whereby an iodine monitor reading of 80% of the allowable limit will be cause to initiate a procedure wherein the operator shuts off all plant exhaust system fans except those in the charcoal filter exhaust. We have concluded that significant improvements have been made to the plant design and operating procedures and that these changes demonstrate an awareness and an intent by RG&E to keep the radioactive releases as low as practicable.

9.0 ACCIDENT ANALYSIS

All of the previously evaluated accidents for operation of Ginna at 1300 MWt have been reanalyzed for 1520 MWt operation using current calculational methods, assumptions and criteria. Several of these methods and assumptions have been published as "Safety Guides for Water Cooled Nuclear Power Plants".

To determine compliance with established guidelines for this facility, we have evaluated the accident involving loss of coolant, the refueling accident, the steam generator tube rupture accident, steam line rupture accident inside and outside containment, and the control rod ejection accident. Other accidents, for which the consequences have not been changed by either the power increase or current calculational methods and criteria, have not been reanalyzed.

The doses that we have calculated for the various accidents are summarized in Table 2. These evaluations were made at the minimum exclusion distance of 450 meters for the 2-hour dose and the low population zone (LPZ) distance of 4800 meters (3 miles) for the course of the accident dose. Based upon the meteorological data presented in the Ginna FSAR, the

TABLE 2

SUMMARY OF CALCULATED DOSES

Accident	Exclusion Distance (450 m) - 2 hour dose		LPZ (4800 m) - Course of Accident	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
1. Loss of Coolant	155	6	36	1.2
2. Control Rod Ejection	50	1.9	6	0.2
3. Refueling Accident	17	1.7	1.2	0.1
4. Steam Generator Tube Rupture	1.5	0.3	0.12	0.02
5. Steam Line Rupture Inside Containment	12	1.0	2.0	0.1
6. Steam Line Rupture Outside Containment	18	negl.	1.2	negl.

appropriate X/Q values for the site boundary (exclusion distance) and the LPZ distance were determined to be 5.3×10^{-4} sec/m^3 and 3.9×10^{-5} sec/m^3, respectively.

As can be seen from the data in Table 2, the doses resulting from any of the design basis accidents are well within the guideline values of 10 CFR Part 100.

9.1 Loss-of-Coolant Accident

In calculating the consequences of a loss-of-coolant accident, we used the method and assumptions given in Safety Guide No. 4. A primary containment leak rate of 0.20 percent of the containment volume per day was assumed to remain constant for 24 hours and of 0.1 percent per day for the duration of the accident. As previously evaluated for the 1300 MWt operation and verified for 1520 MWt operation, we have used an overall iodine reduction factor of 5 for the initial two-hour period, 7.8 for the initial eight-hour period and 10 for the duration of the accident with both the spray and charcoal adsorbers operating simultaneously.

9.2 Control Rod Ejection Accident

Based upon recent Westinghouse analysis of a control rod ejection accident as presented in their Reference Safety Analysis Report (RESAR), we have assumed that 15 percent of the fuel rods could enter DNB causing cladding failure. We conservatively calculated that these rods have an equivalent fission product gap activity of 1.6 percent of the total iodine and noble gas (krypton and xenon) radioactivity in the reactor core.

In analyzing the consequences of the control rod ejection accident, we assumed that the primary to secondary coolant leakage is at the limit permitted by Technical Specifications, 1 gpm, and continues for three hours prior to isolation of the secondary system. A dose reduction factor of 20 was assumed for the iodines between the primary coolant and environs via the secondary coolant and through the air ejector. The primary coolant, excluding the above leakage, was assumed released to the containment volume and leaked to the environment at a rate of 0.20 percent per day for 8 hours. We assumed that the spray and the

filter system would be operated during this accident as stated in the LOCA. The effects of radiological decay during holdup were taken into account. The whole body doses include gamma and beta doses as discussed in Safety Guide No. 4.

9.3 Refueling Accident

The refueling accident is assumed to occur 80 hours after reactor shutdown; the effects of radiological decay prior to the assumed accident were considered. During fuel handling operation, a fuel assembly is assumed to drop with sufficient force to perforate all of the fuel rods (179) in the fuel assembly. The fuel assembly is assumed to be from a region of the core with a peaking factor of 1.5 and to contain a gap activity of 10 percent for the noble gases and 10 percent for the iodines - all of which are instantly released to the pool water. An overall dose reduction factor of 100 was assumed for the iodines in the pool water. All gases released from the refueling pool are discharged through the spent fuel pit charcoal adsorber system. Considering the organic iodine fraction, we assumed the dose reduction factor for the charcoal adsorber system to be 6.7 for iodine. The whole body doses include gamma and beta doses as discussed in Safety Guide No. 4.

9.4 Steam Generator Tube Rupture Accident

In analyzing the steam generator tube rupture accident, we assumed that the primary coolant gross radioactivity is $84/\bar{E}$ μCi/gm and the I-131 dose equivalent radioactivity is 3.0 μCi/gm in the primary system, and is 0.6 μCi/gm in the secondary coolant system.

Assuming a tube rupture within the steam generator in conjunction with a loss of offsite power, the primary coolant system will leak into the secondary system. We assumed that 25 percent of all the radioactivity contained in the primary coolant was transferred to the secondary system prior to isolation of the steam generator which occurs within one hour. The steam generated by the decay heat for the first hour after reactor scram (25 weight percent of secondary system) was assumed to be released to the environs through the steam relief valves with an overall dose reduction factor of 10 for the iodines. The whole body doses include gamma and beta doses as discussed in Safety Guide No. 4.

9.5 Steam Line Rupture Accident

A. Inside Containment

In analyzing the consequences of the steam line rupture accident at 1520 MWt, we assumed 10 percent fuel rod clad failures, 50 percent of the secondary coolant released within 2 hours to the containment volume which leaks at 0.20 volume percent per day, an iodine dose reduction factor of 50 for releases to the environ through the intact secondary system, and primary and secondary coolant radioactivity levels at Technical Specification limit. The maximum allowable primary to secondary system leakage of 1 gpm has been assumed. In the 10 percent failed fuel rods, we assumed that the highest power producing fuel rods were affected and that the gap activity is 1.15 percent of the total iodine and noble gas radioactivity in the reactor core. The effects of radiological decay during holdup were taken into account. The whole body doses included gamma and beta doses as discussed in Safety Guide No. 4.

B. Outside Containment

The same assumptions as discussed for the steam line rupture accident inside containment have been made except that 10% fuel clad failures were not assumed to occur and no iodine dose reduction factor is assumed for the releases from the secondary directly to the environ. These assumptions are equivalent to those given in Safety Guide No. 5 for a similar accident in a boiling water reactor.

10.0 CONDUCT OF OPERATIONS

10.1 Startup and Power Testing at 1300 MWt

The startup and power testing program results substantiated design predictions. The core thermal and hydraulic performances showed that the core operated within the specified thermal and hydrualic limits. Reactor system stability measurements were within applicable criteria. Control rod reactivity worth measurements and rod insertion scram times were satisfactory.

10.2 Plant Operation

Operation of the Ginna facility at 1300 MWt has been quite successful to date. Initial testing and operation were completed with satisfactory results. No equipment malfunctions have restricted power operation; however, while the turbine was removed from service for the warranty inspection some damaged blades were replaced. Plant availability has been 73% since commercial operation began; and 97% since refueling in March 1971.

10.3 1520 MWt Power Test Program

The licensee has proposed a testing program to be performed during the initial escalation in power to 1520 MWt. A set of base conditions will be measured at 1300 MWt before power escalation is initiated to serve as a basis for comparison with subsequent tests. These base conditions will include chemical and radioactivity levels at typical locations, radiation measurements, power distribution measurements and a core performance evaluation. These tests will be repeated at intermediate power levels of about 1380 MWt and 1455 MWt, before going to the full power level of 1520 MWt. Prior to going to each higher power level, the data from the previous level will be reduced and evaluated to determine that the calculated parameters at the next higher power level will not be exceeded or are within acceptable errors.

10.4 Technical Competence

The operating organization, its qualifications and responsibilities, operating procedures, records, maintenance, and review and audit functions have been improved as the result of experience acquired by RG&E since the POL was issued. The technical staff has increased in size and capability. Additional reactor operating experience gained since the issuance of the POL has made the staff of RG&E more alert to abnormal and significant events. The general technical performance of the RG&E staff has demonstrated its competence during the startup and power operations to date.

11.0 TECHNICAL SPECIFICATIONS

Several changes to the Technical Specifications are necessary for the proposed power increase. These changes involve references to power level or parameters associated with reactor power, allowable containment

leakage, primary to secondary coolant leakage, and primary coolant activity. In addition to these changes directly related to the power increase, the Technical Specifications will be updated by incorporation of current effluent release and environmental monitoring requirements.

12.0 STATUS OF ITEMS IN ACRS LETTER OF MAY 5, 1969

All but four of the items identified for future resolution in the Advisory Committee on Reactor Safeguards, ACRS, letter "Report on Robert Emmett Ginna Nuclear Power Plant Unit No. 1" of May 5, 1969, were resolved prior to issuance of the POL on September 19, 1968. Two more of the items have been resolved satisfactorily since the POL was issued. The remaining two items are discussed below.

A. Hydrogen Recombiner

A hydrogen recombiner system was installed at the Ginna facility during initial construction. Since the POL was issued, RG&E has checked out the system and successfully operated the recombiner system with test personnel. Procedures for operation of the equipment have been developed and we believe that the recombiner system could be operated to reduce the hydrogen concentration within the containment building if required after a LOCA.

In accordance with Part 50.109 of 10 CFR 50, RG&E has been asked to determine the doses that would result if purging of containment were used to control the hydrogen concentration in containment within acceptable levels following a LOCA. The results of these analyses will be evaluated to determine the acceptability of purging, and if substantial and additional protection of the public can be obtained by use of another method of hydrogen control, e.g., hydrogen recombiner.

B. Instrumentation

An evaluation of systematic instrument failures was prepared by Westinghouse and currently is being evaluated by DRL representatives and an ACRS Subcommittee.

13.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS has reviewed RG&E's application to increase the power level of the Ginna facility from 1300 MWt to 1520 MWt. The Committee completed its review during its 140th meeting held December 9-11, 1971.

The ACRS concluded in its letter dated December 17, 1971, that if due regard is given to its comments, the Robert Emmett Ginna Nuclear Station can be operated at power levels up to 1520 MWt without undue hazard to the health and safety of the public.

14.0 CONCLUSION

Based upon our review of the application, and of relevant information regarding facility operation to date as discussed in this evaluation, we have concluded that there is reasonable assurance that the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated at steady state power levels up to a maximum of 1520 MWt without endangering the health and safety of the public.

Donald J. Skovholt
Assistant Director for
Reactor Operations
Division of Reactor Licensing

Attachment:
ACRS ltr dtd 12/17/71

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 17, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards reviewed the request by Rochester Gas and Electric Corporation for an increase in the licensed power level of its Robert Emmett Ginna Nuclear Power Plant Unit No. 1 from 1300 MW(t) to 1520 MW(t). A Subcommittee had previously met with the licensee on December 6, 1971. During its review the Committee had the benefit of discussions with representatives of the licensee, the Westinghouse Electric Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed.

The Committee reported to the Commission on the operating license application on May 15, 1969, and a Provisional Operating License was issued on September 19, 1969, authorizing operation at steady-state power levels up to 1300 MW(t). New analyses have been submitted to show that the plant will perform satisfactorily at 1520 MW(t). The Ginna Unit is essentially the same as Point Beach Nuclear Plant, Units 1 and 2, which have been authorized for operation at 1518 MW(t).

Changes in Technical Specifications have been proposed to assure safe operation at the higher power. The licensee has also applied the Ginna operating experience to make improvements in the plant and mode of operation. The Committee believes the licensee should continue to work towards solutions of problems that have been identified by the Regulatory Staff and ACRS as common to large water reactors, including tolerance to anticipated transients with failure to scram. These matters can be resolved with the Regulatory Staff on an appropriate time schedule, not necessarily before commencing operation at the higher power.

The monitoring of iodine released with gaseous wastes at the Ginna plant has not provided reliable evidence of satisfactory performance of the iodine removal system. The Committee believes that attention should be given to improving iodine monitoring methods such that assurance can be provided that total offsite doses remain within appropriate limits.

The licensee will maintain a peak linear power density at full power not exceeding 16.0 kw/ft. Analyses of postulated loss-of-coolant accidents indicate acceptably low peak clad temperatures at the proposed power level of 1520 MW(t).

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, there is reasonable assurance that the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated at power levels up to 1520 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Chairman

References

1. Rochester Gas & Electric Corporation (RG&E) Proposed Technical Specifications and Bases, R. E. Ginna Nuclear Power Plant Unit No. 1, received April 25, 1969

2. RG&E Report, Significant Plant Problems, dated February 5, 1970

3. RG&E Petition Requesting Amendment of License and Extension of Expiration Date of Provisional Operating License, with Technical Supplement, received February 22, 1971

4. Amendments 1-4 to Petition and Technical Supplement

5. RG&E Performance Report for Ginna Plant No. 1, received September 13, 1971