November 15, 1976

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Docket No. 50-244

Rochester Gas & Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
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89 East Avenue
Rochester, New York 14604

Gentlemen:

The Commission has issued the enclosed Amendment No. 11 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment consists of a change to the license and revises the provisions of the Technical Specifications in response to your request of January 30, supplemented by letters dated May 10, June 3, August 5 and September 22, 1976.

This amendment authorizes changes in the design of Ginna spent fuel storage pool from that reviewed and approved in the operating license review and as described in the R. E. Ginna Nuclear Power Plant Final Safety Analysis Report. The changes will increase spent fuel storage capacity from 210 to 595 assemblies.

Copies of the Joint Safety Evaluation and Environmental Impact Appraisal and Federal Register Notice are also enclosed.

Sincerely

/s/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 11 to DPR-18
2. Joint Safety Evaluation and Environmental Impact Appraisal
3. Federal Register Notice

cc w/enclosures:
See next page

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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 PROVISIONAL OPERATING LICENSE

Amendment No. 11
License No. DPR-18

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

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated January 30, supplemented by letters dated May 19, June 3, August 5 and September 29, 1976, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.B(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   "B.(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material or reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as amended and the Application for License Amendment dated January 30, 1976, supplemented by letters dated May 19, June 3, August 5 and September 29, 1976."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 11

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

1. Remove the following pages:

   3.8-2
   3.8-4
   3.11-2
   3.11-3
   3.11-4
   5.4-1

2. Insert the following revised pages:

   3.8-2
   3.8-4
   3.11-2
   3.11-3
   3.11-4
   5.4-1

3. Insert the following new page:

   3.11-5

ast one source range neutron x monitor shall be in service.

d. At least one residual heat removal pump and heat exchanger shall be in operation.

e. Immediately before reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 2000 ppm shall be maintained in the primary coolant system and checked by sampling twice each shift.

f. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

g. The spent fuel pool temperature shall be limited to 150°F.

3.8.2 If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease; work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Basis:

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time. In addition interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing spent fuel.

The spent fuel pool temperature is limited to 150°F because if the spent fuel pool cooling system is lost at that temperature, sufficient time (approximately 7 hours) is available to provide back-up cooling, assuming the maximum anticipated heat load (full core discharge & previously stored fuel), until a temperature of 180°F is reached, the temperature at which the structural integrity of the pool was analyzed and found acceptable.

References:

(1) FSAR - Section 9.5.2

(2) Table 3.2.1-1

(3) FSAR - Section 9.3.1

(4) ANS - 5.1 (N18.6), October 1973

e. Charcoal adsorbers shall be installed in the ventilation system exhaust from the spent fuel storage pit area and shall have the following operating requirements:

(1) The total air flow rate from the charcoal adsorbers shall be at least 75% of that measured with a complete set of new adsorbers.

(2) The bypass flow for the entire set of charcoal adsorbers shall be less than 1%.

(3) The charcoal adsorbent shall be determined to have an iodine removal efficiency of at least 99.5% for an elemental iodine concentration and a carrier gas residence time equivalent to operating conditions (about 1 to 10 mg/cc and 0.2 sec., respectively),

3.11.2 Radiation levels in the spent fuel storage area shall be monitored continuously.

3.11.3 The trolley of the auxiliary building crane shall never be stationed or permitted to pass over storage racks containing spent fuel.

3.11.4 Fuel assemblies with less than 60 days since irradiation shall not be placed in storage positions with less spacing between them than that indicated in Figure 3.11-1 by the designation RDF.

3.11.5 The sp[illegible]t fuel shipping cask shall not b[illegible] carried by the auxiliary building crane, pending the evaluation of the spent fuel cask drop accident and the crane design by RG&E and NRC review and approval.

Basis:

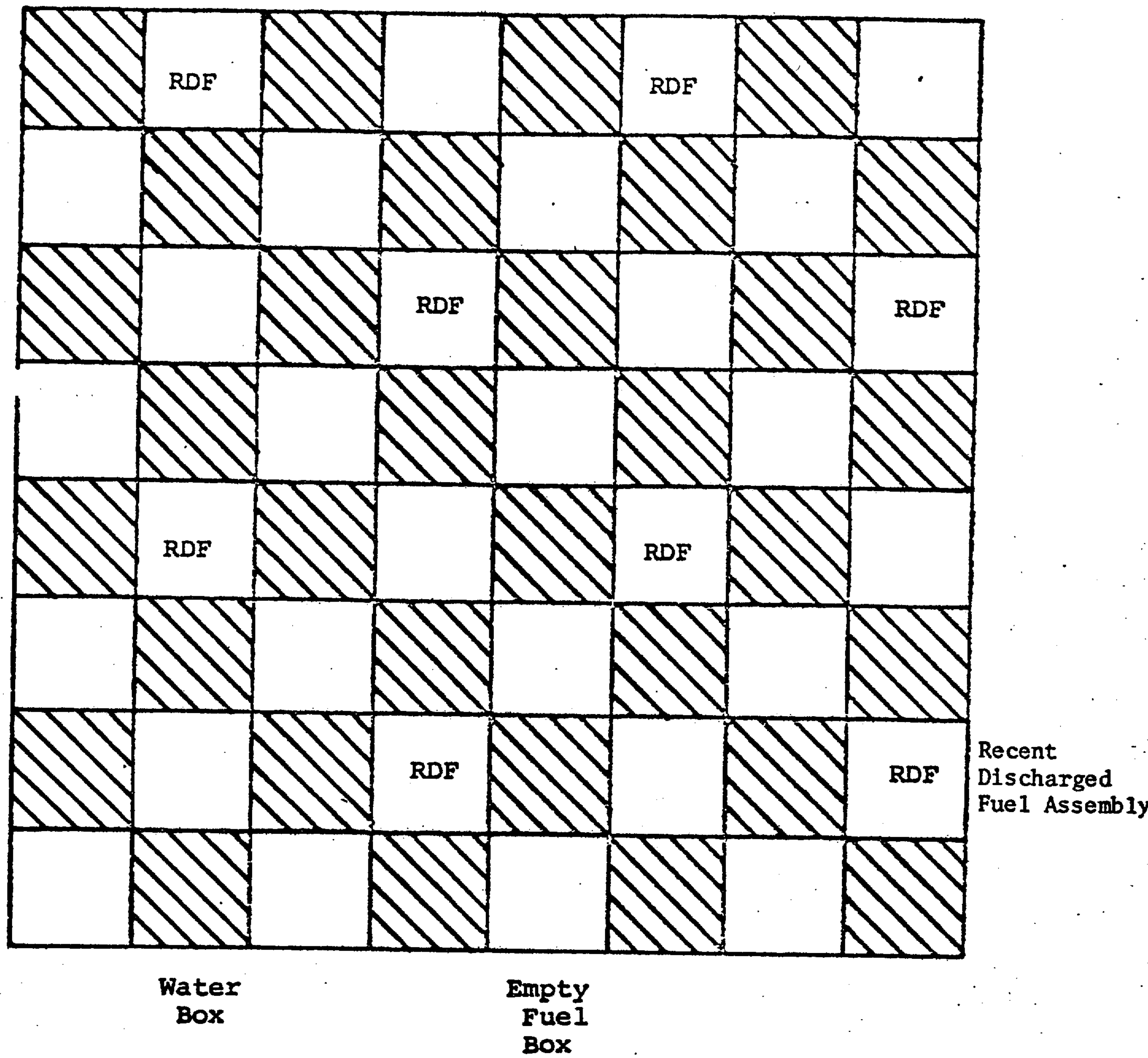
Charcoal adsorbers will reduce significantly the consequences of a refueling accident which considers the clad failure of a single irradiated fuel assembly. Therefore, charcoal adsorbers should be employed whenever irradiated fuel is being handled. This requires that the ventilation system should be operating and drawing air through the adsorbers.

The desired air flow path, when handling irradiated fuel, is from the outside of the building into the operating floor area, toward the spent fuel storage pit, into the area exhaust ducts, through the adsorbers, and out through the ventilation system exhaust to the facility vent. Operation of a main auxiliary building exhaust fan assures that air discharged into the main ventilation system exhaust duct will go through a HEPA and be discharged to the facility vent. Operation of the exhaust fan for the spent fuel storage pit area causes air movement on the operating floor to be towards the pit. Proper operation of the fans and setting of dampers would result in a negative pressure on the operating floor which will cause air leakage to be into the building. Thus, the overall air flow is from the location of low activity (outside the building) to the area of highest activity (spent fuel storage pit). The exhaust air flow would be through a roughing filter and charcoal before being discharged from the facility. The roughing filter protects the adsorber from becoming fouled with dirt; the adsorber removes iodine, the isotope of highest radiological significance, resulting from a fuel handling accident. The effectiveness of charcoal for removing iodine is assured by having a high throughput and a high removal efficiency. The throughput is attained by operation of the exhaust fans. The high removal efficiency is attained by minimizing the amount of iodine that bypasses the charcoal and having charcoal with a high potential for removing the iodine that does pass through the charcoal. A 99% throughput with a removal efficiency of 99.5% will result in an overall iodine removal efficiency greater than 98%. The difference between 98% and the percentage assumed in the evaluation of the fuel handling accident provides adequate safety margin for degradation of the filter after the tests.

The minimum spacing specified for fuel assemblies with less than 60 days decay is based on maintaining the potential release of fission products that could occur should an object fall on and damage stored fuel to less than that which could have occurred with fuel stored in the original fuel storage racks.

Figure 3.11-1

Proposed Spent Fuel Rack With Recent Discharge Stored As Illustrated



Amendment No. 11

5.4 Fuel Storage

Specification

5.4.1 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water.

5.4.2 The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.90$ for new fuel assemblies containing no more than 39 gms U^{235} per axial cm, and assuming unborated water were used in the pool.

5.4.3 The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit.

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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO PROVISIONAL LICENSE NO. DPR-18

ROCHESTER GAS & ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By letter dated January 30, 1976, the Rochester Gas and Electric Corporation (RG&E) submitted an application for a license amendment to increase the storage capacity of the R. E. Ginna spent fuel pool (SFP) from 210 to 595 fuel assemblies. This application was subsequently supplemented by additional information provided by letters dated May 19, June 3, August 5 and September 29, 1976.

Discussion

The present storage capacity of the SFP is 210 fuel assemblies and there are currently 92 spent fuel assemblies stored in the pool. The modification evaluated is the proposal by RG&E to replace the existing fuel storage racks with closer spaced racks. The new racks would increase the storage capacity of the SFP to 595 fuel assemblies. The current fuel storage capacity including ability to accommodate an emergency discharge of a full core, can accommodate refueling only through the spring of 1977. The proposed modification would increase the spent fuel storage to accommodate refueling plus a full core unloading through 1985. In our evaluation we considered the impacts which may result from storing an additional 385 spent fuel assemblies in the SFP for an additional eight years.

The proposed modification does not alter the external physical geometry of the spent fuel pool or involve changes to the SFP cooling or purification systems. The proposed modification will not affect in any manner the quantity of uranium fuel utilized in the reactor over the anticipated operating life of the facility and thus in no way will it affect the generation of spent uranium fuel by the facility. The rate and total quantity of spent fuel generated and stored in the SFP during the anticipated operating lifetime of the facility remains unchanged as a result of the proposed expansion. However, the modification will increase the number of spent fuel assemblies stored in the SFP at one time and the storage time of some of the fuel assemblies will be increased.

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant in New York was shut down in 1972 for alterations and expansions; on September 22, 1976, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Service (AGNS) proposed plant is under construction in South Carolina, and this facility is not licensed to operate. The General Electric Company's (GE) Midwest Fuel Recovery Plant in Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the GE and NFS facilities are licensed for storing spent fuel and applications have been filed for permission to expand these facilities. Also, AGNS has applied for a license to receive and store irradiated fuel assemblies prior to a decision on the licensing action relating to the separation facility. Construction of the AGNS receiving storage station itself is complete.

The NRC Staff is preparing a generic environmental impact statement on spent fuel storage of light water power reactor fuel and is expected to complete this statement by the fall of 1977. The proposed expansion of the SFP capacity at the Ginna Plant will afford RG&E operational flexibility by providing storage space for spent fuel discharges through 1985 with storage space for an emergency full core discharge.

I - SAFETY EVALUATION

Reactivity Considerations Discussion

The proposed high density fuel assembly storage racks are of a Wachter Associates design which uses square, type 304 stainless steel tubes to hold the fuel assemblies in a checkerboard pattern, i.e., fuel assemblies located in every other storage lattice position with the alternate positions filled only with water. Even though the square fuel assembly and water tubes are to be the same size, (i.e., of an outside dimension of 8.43 inches) it will not be possible to insert a fuel assembly into a water tube because the opening at the top of the water tube will be restricted by the lead-in guides associated with the adjacent fuel storage positions. This will result in an opening which is too small to admit a fuel assembly. The nominal wall thickness of all of the stainless steel tubes is 0.090 inches. The thickness of stainless steel between all storage lattice positions is 0.180 inches (two tube walls). The 8.43 inches square tubes are to be held in a close packed array. This will result in a mean distance between fuel assembly centers of 11.92 inches and a fuel assembly volume fraction in the storage rack of 0.426.

RG&E based its criticality analyses for this array on an enrichment of 3.5 weight percent U^{235}. Assuming a UO_2 density at 95 percent of the theoretical density, the resulting fuel loading will be slightly less than 39.0 grams of U^{235} per axial centimeter of fuel assembly.

Pickard, Lowe and Garrick, Inc. (PLG) of Washington, D. C., performed the nuclear analysis for RG&E. PLG used its version of the LEOPARD computer program to generate macroscopic cross sections for input to four energy group, diffusion theory calculations. These were made with the PDQ-7 program. The LEOPARD program is a derivative of the MUFT and SOFOCATE programs, which were developed for the Atomic Energy Commission in the late 1950's along with the PDQ diffusion theory program. RG&E's report provides the results of criticality calculations which were made with these methods. PLG first calculated the K_{eff} for the nominal storage lattice cell and then made perturbation theory calculations to account for possible variations and uncertainties.

In view of the potential long life of these storage racks and the likelihood that the NRC criterion of a maximum neutron multiplication factor of 0.95 will be applicable to fuel assemblies with mixed oxide (PuO_2 - UO_2) fuel, we requested RG&E to make sufficient allowance for it in the subcriticality of the loaded fuel pool or to provide a commitment to modify the facility in the future if it does not satisfy NRC's subcriticality requirement. In response to this request, RG&E stated that if mixed oxide fuel assemblies are used, criticality calculations for the mixed oxide fuel will be submitted for review by the NRC and that the NRC's subcriticality requirements will be met. We find this commitment to be acceptable.

Evaluation of Reactivity Considerations

For a fuel loading of 38.7 grams of U^{235} per axial centimeter of fuel assembly and for the nominal dimensions as specified on the drawings, PLG calculated the infinite neutron multiplication factor, k_∞ of this storage lattice to be 0.878. On comparing the PLG calculational method with another standard calculational method, we found that the PLG method will yield somewhat lower values for the neutron multiplication and consequently is less conservative. Further comparison revealed that the reactivity worth of the stainless steel assumed in the PLG calculational method is about 25 percent higher than assumed in the other standard method. Decreasing the reactivity worth of the stainless steel by this 25 percent would increase the calculated k_∞ to 0.89 which is consistent with the results of the other standard method.

PLG made perturbation theory calculations to determine the possible variations and uncertainties in the neutron multiplication in the storage pool. Since nominal values were used in the base calculation, factors that would increase the neutron multiplication would be (1) the use of minimum values for the thickness of the stainless steel, (2) the use of minimum spacing between fuel assemblies and (3) the use of a pool water temperature of 200°F. Factors that would decrease k_∞ would be (1) to account for the axial neutron leakage and (2) include the effect of the inconel spacer grids, neither of which were included in the nominal calculation. PLG also found that increasing the number of mesh points in the PDQ diffusion calculations decreased calculated neutron multiplication. The magnitude of the combined negative effects is almost the same as the magnitude of the combined positive effects. We find the results of these perturbation theory calculations acceptable and conclude that these considerations do not significantly alter the maximum k_∞ value of .89 which is well below our requirement of less than 0.95 and therefore acceptable. We have modified RG&E's proposed Technical Specifications by adding a maximum limit of 39.0 grams of U^{235} per axial centimeter of fuel assembly for stored fuel assemblies. RG&E has agreed to this modification since it represents the bounding value for the RG&E criticality calculations.

A potentially significant increase in neutron multiplication factor in this array of stored fuel assemblies attributable to gamma heating, could be obtained if, somehow, the water in the water boxes were to be displaced with steam while the fuel assemblies remained filled with water.

RG&E analyzed the effect of the gamma heating of the water in the water boxes. The results of this analysis show that, in order for the temperature differential in the water flowing through the water boxes to be the same as that for the fuel boxes, the holes in the bottom of the water boxes must be enlarged from 1/4 inch to 3/4 inches in diameter.

With the enlarged holes there will be more water flowing through the water boxes and less through the fuel boxes. Thus, RG&E recalculated the thermal-hydraulic problem to determine the effect of enlarging the water box holes on the maximum fuel element clad temperature in the hottest fuel assembly. This analysis showed that there would be only a minimal increase in the maximum clad temperature and that it would still be less than 160°F. The associated pool water temperature, both in the fuel storage box and the adjacent water boxes would be less than 160°F and, therefore, there would be no steam formation in either the fuel box or the water box. On this basis, we find that 3/4 inch diameter holes in the water boxes are acceptable.

Spent Fuel Cooling Discussion

The spent fuel pool cooling system includes one 610 gallon per minute pump and one heat exchanger. Approximately ten percent of the flow is by-passed around the heat exchanger and purified by a demineralizer and a filter. For this flow rate, the heat exchanger is designed to transfer 5.3×10^6 Btu/hr to 80^oF service water when the spent fuel pool (SFP) temperature is 120^oF. When the SFP temperature is 150^oF, the heat exchanger will be able to transfer 9.3×10^6 Btu/hr to the service water. The service water inlet temperature for this heat exchanger has not exceeded 80^oF since the plant became operational in 1969.

The quantity of decay heat, that will be generated in the spent fuel, has been calculated in accordance with the requirements of ANS-5.1 (N18.6) October 1973 plus 20 percent, where the fuel assemblies are assumed to have been irradiated at rated core power for the average burnup of the discharged fuel. The results of these calculations show (Table VI-3) that up until the 1982 refueling, the heat load will be less than 5.3×10^6 Btu/hr if the core is allowed to cool for 15 days before transferring these fuel assemblies to the fuel pool. Similarly, up until 1978, the total heat load on the spent fuel pool after a full core discharge will be less than 9.3×10^6 Btu/hr, if the core is allowed to cool for 30 days prior to transferring fuel assemblies to the fuel pool. After the normal 1979 refueling, this heat load would be increased by about 2% to 9.46×10^6 Btu/hr.

In its original submittal, RG&E stated that the time for the water in the pool to reach 180^oF following a failure in the spent fuel pool cooling system would be 24 hours, starting from an initial pool temperature of 120^oF in the normal, one third core refueling case, and 6.8 hours starting from an initial pool temperature of 150^oF in the full core discharge case. RG&E also stated that in the event either of the above conditions were to occur, equipment maintenance could be accomplished or backup cooling could be obtained. In response to our request for more details on how the loss of either the spent fuel cooling system pump or the heat exchanger would be handled, RG&E presented the results of analyses showing that the pool water temperature could be maintained within acceptable limits if either the pump or heat exchanger become unavailable. Should the pump be lost, approximately one and one half hours would be required to install and commence operations with a portable pump. The pool water temperature would increase 4^oF above 120^oF when in the normal refueling mode before the portable pump would become operational and 7^oF above 150^oF when in the full core discharge mode before the portable pump would become operational. Should the spent fuel pool cooling system heat exchanger be lost, one of the two component cooling system heat exchangers could be temporarily connected to the loop in about three hours. During this interval, the water temperature of the SFP would have increased to 128°F, when in the normal refueling mode before the component cooling water heat exchanger would become operational for SFP cooling. During normal full power operation and shutdown operation, only one of the two component heat exchangers is essential for the removal of the heat loads associated with reactor operation. The Technical Specifications require both heat exchangers to be operable when the reactor is critical. However, one heat exchanger is permitted to be taken out of service provided it is returned to operation within 24 hours. Should the component heat exchanger be required for cooling the spent fuel pool water beyond 24 hours, a cold shutdown condition could be attained and maintained with the remaining equipment.

In response to our concern regarding the possibility of inadequate cooling of recently off-loaded fuel should fuel assemblies be closely grouped in the new storage racks, RG&E performed a natural circulation heat transfer analysis assuming the discharged batch was grouped together in the storage rack furthest from the SFP cold water inlet. The results were 160°F for the maximum cladding temperature of the hottest fuel assembly and 242°F for the corresponding saturation temperature. RG&E states that this is sufficient margin between the maximum cladding temperature and the saturation temperature. From this information, RG&E concludes that there is no limiting thermal requirement which would prevent the grouping of the entire off-loaded batch of fuel assemblies together in the storage racks.

Evaluation of Spent Fuel Cooling

RG&E's calculated heat rates are appropriately conservative. Consequently, if the spent fuel pool cooling system operates at rated design, the pool's bulk water temperature will be less than 120°F during a normal refueling procedure, which involves 15 days of incore cooling prior to any off-loading of fuel. For a full core transfer to the pool after 30 days of incore cooling, the pool outlet water temperature should not exceed 150°F until after the normal refueling in 1979. Prior to the time when additional cooling capacity is needed, RG&E states that if necessary they will modify the spent fuel cooling system to maintain conformance to the Technical Specifications. We find this to be acceptable.

We find that the actions that would be taken by RG&E, as described in its plant procedures, in the event of failure of the spent fuel pool cooling system, are acceptable. A limit of 150°F on the SFP water temperature has been incorporated in the Technical Specifications which provides sufficient margin to allow time for the corrective actions to be taken prior to exceeding the temperature at which the structural analysis was performed.

We have examined RG&E's analysis assumptions and have independently calculated both the maximum heat loads on this system and the rate of increase of fuel pool water temperature under various conditions. We have concluded that the present cooling system is adequate for the increased storage of spent fuel assemblies until 1979. RG&E has committed to modifying the spent fuel cooling system prior to 1979 if modifications are needed to maintain conformance to the requirements of the Technical Specifications.

Evaluation of Structural, Mechanical and Material Design

All design, analyses fabrication, and installation of the new spent fuel racks are being performed under the direction of Wachter Associates, Inc. Each rack assembly is made of a repeating array of square boxes. Each box in the assembly is welded to adjacent boxes to form a honeycomb box structure arrangement. Alternate boxes in a checker board pattern are designed to contain spent fuel assemblies. The remaining boxes will contain pool water. The boxes are approximately 13-1/2 feet long, 8.25 inches square and 0.09 inches thick. Each rack assembly is supported on and bolted to a rack base assembly fitted with four adjustable leveling pads. Nine rack assemblies will be placed in the pool. The assembly bases are interconnected to each other by means of key blocks and shear blocks and are laterally supported off the wall by means of large bearing pads attached to the rack base. All material used in the fabrication and construction of the racks consists of 304 stainless steel.

The new spent fuel racks are built to meet Section VIII of the 1974 ASME Boiler and Pressure Vessel Code. All applicable structural steel items were designed to the AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings, revised 7th edition, in conjunction with the material allowables from the B&PV Code.

The seismic design of the racks is based on the response spectra and damping values presented in the R. E. Ginna FSAR. No benefit is taken for the damping effect of the water. The water contained within the individual boxes was assumed to move with the box itself and the actual mass of the contained water was accordingly distributed over the box length. In the initial design of the racks a horizontal acceleration of o.2g was applied simultaneously with normal gravity plus or minus a vertical acceleration of o.2g. The direction of the horizontal seismic component was assumed to be in the worst-case direction which results in the maximum loads at any fuel rack corner joint. As an independent check on the adequacy of the design, additional calculations were performed wherein the seismic excitations along three orthogonal directions were imposed simultaneously as recommended in Regulatory Guide 1.92.

The fuel racks and supporting structures were designed for the extreme environmental conditions occurring simultaneously with the abnormal plant conditions (i.e., fully-loaded spent-fuel racks in a hot bath (200°F) undergoing a safe shutdown earthquake). The racks were also analyzed for normal operating conditions, severe environmental conditions and extreme environmental conditions. Normal code stress limits were used as acceptance criteria for all of the above postulated load conditions. In addition, RG&E considered the loads from a dropped fuel assembly and found that the racks have adequate structural strength to withstand the effects of such an accident.

RG&E also performed a review of the load carrying ability of the spent fuel pool floor and walls and found that the existing concrete structure is capable of supporting the proposed increased fuel assemblies and restraining the spent fuel racks during a seismic event. All stresses were found to be in accordance with ACI 318-71. The temperature limits established in the FSAR for the spent fuel pool are not being changed with the present modification, therefore the effects of temperature grandients on the pool structure will remain unchanged. However, the wall separating the spent fuel pool from the refueling canal was also analyzed for the case wherein the canal is drained. RG&E's evaluation indicated that the pool temperature should be maintained within approximately 30°F of the temperature in the canal. RG&E has committed to prepare operating procedures which will implement the requirement that the temperature difference across the wall is not to exceed 30°F with the refueling canal drained.

During installation of the new racks, the Seismic Category I capability of any existing racks which will temporarily contain spent fuel will be maintained. Temporary struts or supports will be added as required. The newly installed racks necessary to temporarily contain spent fuel also will be restrained to maintain their Seismic Category I capability during the installation procedures.

The criteria used in the analysis, design, and construction of the new spent fuel racks to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff. The use of these criteria provide reasonable assurance that the new fuel pool structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. We, therefore, find the structural, mechanical, and material aspects of the design acceptable.

Evaluation of Potential Accidents

Fuel Handling Accidents

Although the new storage racks provide accommodation for a larger inventory of spent fuel, the radiological consequences of a fuel handling accident are not more severe than those previously reported in the Ginna FSAR. As discussed in the earlier section of this safety evaluation dealing with the structural evaluation of the new storage racks, the stored fuel is protected by the rack structure from being impacted by a dropped fuel assembly. Therefore, the radiological consequences remain those previously evaluated for the damage to the dropped assembly itself.

Construction Accident

In their August 5, 1976 submittal, RG&E responded to our concern for the safety related plant equipment located adjacent to and below the travel path for the replacement fuel racks. The response to Item B in this submittal, states that a transporter, which in this case will be a flat-bed trailer and truck, will take the replacement storage racks through the east door of the Auxiliary Building down the truck alley on the operating floor to the spent fuel pool area. The first rack brought in will be lifted from the transporter and placed on the operating floor adjacent to the spent fuel pool canal wall. In the unlikely event of a load drop during the removal of the old storage racks and the insertion of the new racks, the empty rack on the floor will serve as an energy absorber and thereby protect the operating floor and the plant equipment which is located below. From the description of the sequence of steps to be followed and the other precautions that will be taken by RG&E, we conclude that adequate plans have been made to prevent damage to the plant's safety equipment in the event of a load drop accident during these construction operations.

Since there are now 92 spent fuel assemblies in the pool, the storage racks will be replaced while the pool is full of water. In this regard, RG&E states that, "During installation of the new racks the Seismic Category I capability of any existing racks which will temporarily contain spent fuel will be maintained. The existing spent fuel will be stored in the pool in a planned location pattern which will allow the sequence of installation work to be performed without crane loads being carried over any area where spent fuel is stored. Load lifting, lowering and lateral transfer will be controlled by guide lines to prevent accidental contact with stored fuel. Necessary relocation of the stored fuel within the pool during installation will be planned to assure minimum handling." We have determined that suitable precautions will be taken to satisfy applicable safety and design criteria. We find that the proposed construction activities can be performed with reasonable assurance that no damage to stored fuel or any safety-related equipment or structures will occur.

Missiles or Dropped Objects

Even though the new storage racks will increase the storage capacity for fuel assemblies from 210 to 595, the outer envelope of these new racks will be within the envelope of the present fuel storage racks. Therefore, the probability of a missile or dropped object strike has not been increased. However, since there is a higher density of stored fuel, the potential for an increase in the radiological consequences of such an occurrence does exist. RG&E, therefore, proposed a spent fuel storage pattern that would limit the minimum distance between freshly discharged fuel. The storage spaces within this minimum distance could be filled with spent fuel that has decayed more than 60 days. In this manner, the density of fission product inventory in any local area is maintained less than that which could have been stored in the existing storage racks and the radiological consequences of a postulated missile strike or dropped object would not be increased. This limit on minimum spacing of freshly discharged fuel (less than 60 days decay time) has been added to the Technical Specifications.

To preclude the possibility of heavy loads being dropped from the Auxiliary Building crane, the Technical Specifications state "the trolley of the Auxiliary Building crane shall never be stationed or permitted to pass over storage racks containing spent fuel." In addition, existing interlocks on the crane will be modified to limit the horizontal motion of the crane bridge and trolley so that the crane cannot pass over the stored spent fuel. A bypass mode is provided to permit the insertion and removal of the spent fuel cask. However, since RG&E's analysis of the spent fuel cask drop accident has not been completed, a Technical Specification has been added that prohibits handling the spent fuel cask until RG&E's analysis has been completed and accepted by the NRC.

We find, based on the foregoing consideration, that the proposed modifications will not result in an increase in probability of occurrence or consequences of an accident previously evaluated, nor in the creation of the possibility of an accident or malfunction of a different type than any previously evaluated, nor in the reduction of the margin of safety as defined in the basis for any Technical Specification.

Radiation Protection for Workers

In addition to their own personnel, RG&E will be using contractor personnel as divers for underwater work. We have reviewed the plans for radiation protection measures including the use of the divers, the precautions to be taken, and the criteria to be used for personnel protection. We find that these plans, properly implemented, will ensure that the requirements of 10 CFR Part 20 will be met and are acceptable.

Conclusion on Safety

We have 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.

II - ENVIRONMENTAL IMPACT APPRAISAL

On September 16, 1975, the Commission announced (40 F.R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement. The Commission directed that in the consideration of any such proposed licensing action, the following five specific factors should be applied, balanced, and weighted in the context of the required environmental statement or appraisal.

a. Is it likely that the licensing action here proposed would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?

The Ginna reactor core contains 121 fuel assemblies and 33 rod cluster type control rods. The facility was licensed in September 1969 and commenced operation in March 1970. The Ginna SFP was designed on the basis that a fuel cycle would be in existence that would only require storage of spent fuel for a year or two prior to shipment to a reprocessing facility. Therefore, a pool storage capacity for 210 assemblies (about 1 2/3 cores) was considered adequate. This provided for complete unloading of the reactor even if the spent fuel from two refuelings were in the pool. The normal refueling schedule for Ginna is an annual refueling cycle. Under current fuel management planning, RG&E expects to load approximately 36 to 40 fresh assemblies each year.

RG&E currently has a reprocessing agreement with Nuclear Fuel Services, Inc. to cover fuel discharged from Ginna through 1981. Under this contract, RG&E shipped the initial core loading of fuel-121 fuel assemblies to NFS in the spring of 1973. Because of the lack of additional storage space at NFS and the announced withdrawal of NFS from the reprocessing business, the ability of NFS to meet terms and period of the contract is uncertain. Because of refuelings since 1973, there are currently 92 spent fuel assemblies stored in the Ginna SFP. With the existing storage racks, full core discharge would no longer be possible after the refueling in the spring of 1977. If 36 to 40 fuel assemblies are discharged each year; the SFP would be filled after the spring 1979 refueling. If at that time fuel could not be shipped off site, operation of the reactor would have to be terminated.

Since spent fuel reprocessing facilities cannot assuredly be available to RG&E prior to the mid 1980's (and, therefore, no spent fuel can be shipped for reprocessing), spent fuel discharges subsequent to 1979 will have to be stored or the facility shut down. The proposed licensing action (i.e., installing new racks of a design that permits storing more assemblies in the same space) would provide the licensee with additional operating flexibility which is desirable even if adequate offsite storage facilities hereafter become available to the licensee.

We have concluded that a need for additional spent fuel storage capacity exists at Ginna which is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

b. Is it likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?

With respect to this proposed licensing action, we have considered commitment of both material and nonmaterial resources. The material resources considered are those to be utilized in the capacity expansion of the SFP.

Under the proposed modification, the present spent fuel racks will be replaced by new spent fuel racks that will increase the storage capacity to 595 assemblies. The new racks are modular in design. The total quantity of stainless steel to be utilized in the new spent fuel racks is approximately 200,000 pounds. The racks do not use a poison material such as boron impregnated stainless steel, B_4C plates or boral. The amount of stainless steel used annually in the U.S. is about 2.82×10^{11} lbs. The material is readily available in abundant supply. The amount of stainless steel required for fabrication of the new racks is a small amount of this resource consumed annually in the United States. We conclude that the amount of material required for the racks at Ginna is insignificant and does not represent an irreversible commitment of natural resources. This licensing action would not constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or other actions that might be taken by the industry in the future to alleviate fuel storage problems. No other resources need be allocated because the other design characteristics of the SFP remain unchanged. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings among fuel assemblies.

The increased storage capacity at the Ginna SFP was considered as a nonmaterial resource and was evaluated relative to proposed similar licensing actions within a two year period (the time we estimate is necessary to complete the generic environmental statement) at other nuclear power plants, fuel reprocessing facilities and fuel storage facilities. We have determined that the proposed expansion in the storage capacity of the SFP is only a measure to allow for continued operation and to provide operational flexibility at the facility, and will not affect similar licensing actions at other nuclear power plants.

We conclude that the expansion of the SFP at the Ginna facility prior to the preparation of the generic statement does not constitute a commitment of either material or nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

c. Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overlooking any cumulative environmental impacts?

The SFP at Ginna was designed principally to store spent fuel assemblies prior to shipment to a reprocessing facility. These assemblies may be transferred from the reactor core to the SFP during a core refueling, or to allow for inspection or modification to core internals which may require the removal and storage of certain fuel assemblies or a full core. The assemblies are initially intensely radioactive due to their fission product content and have a high thermal output. Thus they are stored in the SFP to allow for radioactive and thermal decay. The major proportion of decay occurs during the 150 day period following removal from the reactor core. After this period, the assemblies may be withdrawn and placed into a heavily shielded fuel cask for offsite shipment. Space permitting, the assemblies may be stored for an additional period allowing additional preshipment fission product decay and thermal cooling.

Since the additional capacity of the SFP is proposed for fuel from this site alone, all the environmental impacts can be assessed within the context of this application. Potential impacts, both nonradiological and radiological relative to the fuel rack conversion and subsequent operation of the expanded SFP at this facility were considered by the NRC staff. No environmental impacts on the environs outside the spent fuel storage building were identified that would be associated with the proposed construction of the expanded storage capacity of the SFP. The impacts within this building are expected to be limited to those normally associated with metal working activities.

No significant environmental impacts, either onsite or offsite, could be identified as resulting from operation of an expanded SFP at this facility. The only potential offsite nonradiological environmental impact that could arise from this proposed action would be an additional discharge of heat to Lake Ontario. Both RG&E and the staff have evaluated the existing SFP cooling system and have concluded that there is adequate cooling capacity to maintain the pool water temperature below 120°F with the normal refueling schedule. The SFP heat exchanger is cooled by the service water system. The heat transferred to the service water system from the SFP is a small amount of the total heat load on this system. Only 700 gpm out of a total service water flow rate of about 11,700 gpm is nominally required to cool the SFP. Compared to the existing heat load on the service water system and the total heat rejected to Lake Ontario by the once-through circulating water system, the small additional heat load from the SFP cooling system (attributable to the longer storage of additional spent fuel) will be negligible.

The potential offsite radiological environmental impact associated with this expansion resulting from an incremental addition in the longlived radioactive effluents released at the facility was evaluated and has been determined to be environmentally insignificant as discussed below.

The expansion of the SFP will allow spent fuel to be stored for an additional eight-year period without shipment offsite and maintain space to off-load a full core. During storage both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the surface material would consist of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. Radionuclides that could be released to the water through cladding defects, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive material is its contribution to local radiation to which workers in and near the SFP would be exposed. The volatile radionuclides of most concern that might be released through cladding defects are xenon, krypton, tritium and the iodine isotopes.

About 60 gpm of SFP water is circulated through a purification system consisting of a demineralizer and filter. This system is designed to remove the nonvolatile corrosion and fission product nuclides and to control water chemistry and optical clarity. The demineralizer is a flushable type containing 20 cu. ft. of mixed bed resin. The filter is a disposable synthetic cartridge type rated to remove particles greater than 5 microns in size. To remove surface dust and debris, the SFP is equipped with a skimmer system consisting of a pump, strainer and filter. The latter is a replaceable type unit rated to remove particles larger than 5 microns. Since the SFP contains about 255,000 gallons of water, at least 70 hours would be required at 60 gpm for one purification turnover. While the SFP purification system is operated continuously, half of the time it is used to purify the SFP water and half of the time to purify the refueling water storage tank.

Storing additional spent fuel in the SFP may increase the amount of corrosion and fission product nuclides introduced into the SFP water. The purification system is capable of removing the increased radioactivity so as to maintain acceptable radiation levels above and in the vicinity of the pool. Redesign of the SFP racks increases only the storage capacity of the pool and not the frequency or the amount of the core to be replaced for each fuel cycle. Thus, the amount of corrosion product nuclides released into the pool during any year will be about the same regardless of the length of time or number of assemblies stored in the pool. Expansion of the capacity does increase the potential for increasing the amount of fission products released into the SFP water from clad defects. This could increase the amount of radioactivity accumulated on the filter and demineralizer which are disposed of as solid waste.

RG&E does not expect to change the frequency of operation of the SFP purification system as a result of the fuel storage rack modification; therefore, the frequency of filter changes and resin changes may increase. The SFP filter and the skimmer filter are replaced when the pressure drop across the filter exceeds 20 psi or, if this limit is not reached, the filter cartridge is replaced during each refueling. The SFP demineralizer is replaced when the decontamination factor approaches unity. Based on Ginna's present fuel experience,the percentage of leaking fuel assemblies is very small and the proposed increase in storage capability represents longer term storage of well cooled fuel. Therefore, RG&E predicts that the increase in frequency of filter and resin changes is not expected to be significant.

According to RG&E, operating experience indicates that the SFP demineralizer generates approximately 40 cu. ft. of solid radioactive wastes per year; of this, approximately 50% can be attributed to storage of spent fuel. Experience also indicates that the SFP purification filter and the filter in the skimmer system each generate about 4 cu. ft. of solid radioactive wastes per year.

Ginna has a temporary Spent Fuel Pool Leakoff Return System (SFPLRS). The purpose of this system is to demineralize the water that collects between the SFP stainless steel liner and concrete pool structure, approximately 1.08 gpm and return it to the SFP. If there were no SFPLRS this water would go into the liquid radioactive waste processing system. The SFPLRS was installed to reduce the waste processed by the Liquid Waste Processing System. Based on information compiled during the first half of 1976, it appears that the 1.5 cu. ft. of SFPLRS demineralizers resin is being changed, on the average, 29 times per year.

The total solid radioactive wastes from the SFP purification system attributable to the SFP is approximately 0.75 percent of the average solid radioactive wastes generated each year. At present, there are 92 fuel assemblies in the SFP. As an upper limit on the amount of additional solid radioactive waste that might result from the proposed modification, RG&E has predicted that if the generated solid radioactive wastes increase linearly, with the number of fuel assemblies in the SFP, which is unlikely, the solid waste would increase by a factor of 5.15 with 474 fuel assemblies in the SFP. (To maintain full core discharge capability only 595 - 121 = 474 fuel assemblies can be stored in the SFP.)

We have conservatively assumed that due to the expansion the amount of solid radwaste generated each year by the SFP purification system may double. This would increase the volume of solid waste to be shipped from the facility by about 70 cu. ft. per year. If the increased storage of spent fuel does eventually increase the amount of solid waste by 70 cu. ft. per year, the increase in total waste volume would be less than 1% and would not have any significant environmental impact.

We have reviewed RG&E's plan for removal, disassembly and offsite shipment of the old racks and installation of the new racks using utility and contractor personnel (including divers). The total occupational radiation exposure for this operation is estimated to be about 40 to 45 man-rem. We consider this to be a reasonable estimate.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by RG&E and by utilizing realistic assumptions for radionuclide concentrations in the SFP water and for occupancy times. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because they are under 26 feet of water. Our analysis indicates that the occupational radiation exposure resulting from the proposed action represents less than one percent of the present total annual occupational exposure at this facility. The small increase in radiation exposure will not affect the RG&E's ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers. RG&E predicted that while increased fuel storage may result in an increased frequency of changing the demineralizer resin, it is not expected to result in any increase in the radionuclide concentrations or in subsequent radiation levels at the surface of the water.

With respect to gaseous releases, since short lived noble gases would have decayed to negligible amounts, the only significant noble gas isotope remaining in the SFP and attributable to storing additional assemblies for a longer period of time would be Krypton-85. Based on operating experience for Zircaloy clad fuel (see NUREG-0017), we have assumed that 0.12% of all fuel rods will have cladding defects which permit the escape of fission product gases. It is assumed that the fission product gases escape on a relatively linear basis with time. On this basis, we have conservatively estimated that an additional 20 curies per year of Krypton-85 will be released when the modified pool is filled to capacity. For comparison, RG&E concluded that increasing the fuel storage from 210 to 595 assemblies (a factor of 2.83) will not increase the Krypton-85 release rate, since fuel discharge will continue on a 1/3 core per year rate and the release of Krypton-85 is most likely to occur during the initial handling and the first year of storage when the fuel is hotter. RG&E states that increasing the pool capacity represents a longer storage of well cooled fuel without the thermal driving forces required to cause Krypton-85 to diffuse from the defective fuel assembly. RG&E concluded that the increased fuel storage will have essentially no impact on concentrations of radioactivity in the air of the auxiliary building.

The fuel storage pool area, which is within the auxililary building, is continuously ventilated. Normally, this air is released through the plant vent. For comparison, the Ginna facility has reported an average release of 2400 curies per year of noble gases from the entire facility for the last five years of operations. RG&E has conducted a continuous environmental radioactivity monitoring program starting prior to facility operation. Sampling and analyses were performed on air particulates, gamma dose rate, surface water, well water, fruit, bottom sediment, milk, algae and fish. The results are published in the annual reports. Based on the data obtained, there is no significant radioactivity in the environment that can be attributed to facility operation.

The additional 20 Ci/yr of Krypton-85 that we have conservatively estimated may be released as a result of the proposed modification would be less than 1% of the total noble gas release from the facility and would not have any significant impact on radiation levels or personnel exposures offsite.

Assuming that the spent fuel will be stored onsite for several years (rather than shipped offsite after 6 to 24 months storage as originally planned), Iodine-131 releases will not be significantly increased by the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between each annual refueling. Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above the 120°F used in the design analysis. The analysis of cooling capability of the SFPCS heat exchanger was conservative, particularly since a service water flow of 700 gpm at 80°F was assumed. Since the temperature of the pool water will normally be maintained below 120°F, it is not expected that there will be any significant change in evaporation rates or in the release of tritium as a result of the proposed modification.

The staff will determine the acceptability of the spent fuel cask tip accident evaluation prior to cask use. A Technical Specification has been added, with RG&E's agreement, to prohibit the handling of spent fuel casks above the spent fuel pool or near its edge until we have reviewed and accepted the spent fuel cask tip evaluation. On the basis of previous analyses of cask tip accidents, the staff concludes that such an event can be precluded by physical restraints or the consequences of such an event minimized by allowing only fuel which has decayed for several months to be in the pool area vulnerable to a cask tip accident. Thus the staff has determined that the modification to the SFP to increase its capacity can be accomplished without creating the possibility for an accident or malfunction of a different type than evaluated previously, and that neither the probability nor the consequences of a spent fuel handling accident would be increased. We therefore conclude that the modification is acceptable.

We have considered the potential cumulative environmental impacts associated with the expansion of the SFP and have concluded that they will not result in radioactive effluent releases that significantly affect the quality of the human environment during either normal operation of the expanded SFP or under postulated fuel handling accident conditions.

d. Have all technical issues which have arisen during the review of this application been resolved within that context?

This report points out that all questions concerning health, safety and environmental concerns have been answered.

e. Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?

In regard to this licensing action, we have considered the following alternatives; (1) shipment of spent fuel to a fuel reprocessing facility, (2) shipment of spent fuel to a separate fuel storage facility, (3) shipment of spent fuel to another reactor site, and (4) ceasing operation of the facility. These alternatives are considered in turn. For comparison, the cost of expanding the SFP as proposed by RG&E is estimated to be $1,800,000. This is a capital expense which equates to a yearly cost of the increased storage capability of approximately $2/kgU.

(1) RG&E currently has a reprocessing agreement with Nuclear Fuel Services, Inc. (NFS) to cover fuel discharge from Ginna through 1981. However, on September 22, 1976, NFS announced that they were withdrawing from the fuel reprocessing business. As discussed earlier, there are no storage and/or reprocessing facilities in the U. S. that are presently able to contract for the storage and reprocessing of spent fuel. With the present spent fuel storage and reprocessing situation, it appears unlikely that shipment of spent fuel to any such facilities could be made within the next several years.

(2) Although it is not anticipated that any storage will be available in the foreseeable future based on inquiries by other licensees to potential spent fuel storage facilities, we estimate the costs associated with storage at another facility to be $3000 to $3800 per year for each PWR type fuel assembly. This would be based on a minimum storage commitment of seven to ten years and would equate to about $10/kgU per year. The cost of shipping the spent fuel to the storage facility could add $2/kgU per year.

An independent storage facility cannot with any certainty be licensed and built in time to meet RG&E's needs. Even if off site storage were available the cost based on best estimates available would be about $10 to $15 per kgU per year for reserved storage space. The additional investment in transportation for off site storage could add $2/kgU per year.

(3) According to a survey conducted and documented by the Energy Research Development Administration, as much as 46 percent of the operating nuclear power plants will lose the ability to refuel during the period 1975-1984 should there not be any additional spent fuel storage pool expansions or commitments to utilize offsite storage facilities. Thus, RG&E cannot rely on any other power facility to provide additional storage capability except on a short-term emergency basis.

Shipping to another reactor would cost the $2/kgU per year for transportation. In addition, there would be the cost of handling within the receiving reactor and the cost of engineering, licensing and contracting for such a capability.

(4) With the existing storage racks and the fact that there are currently 92 spent fuel assemblies stored in the pool, Ginna would not be able to transfer a full core into the pool after the refueling scheduled for the spring of 1977. If the normal refueling schedule is followed, the SFP will be filled after the spring 1979 refueling. If at that time, fuel could not be shipped off site and if the storage space in the pool has not been expanded, operation of the reactor would have to be terminated.

Terminating operation of the Ginna facility would impact the customers of RG&E very heavily. The Ginna facility generated electrical energy in 1975 equal to 67 percent of the requirements of RG&E customers. This power was supplied to their customers at a fuel cost of .251¢/kwh. Replacement energy if available from an existing fossil fuel plant would average over the year on the order of six times that amount or an additional dollar cost of 38 million dollars for the year, averaging $138 per customer. Replacement power (a replacement fossil fuel plant of equal capacity if completed now and operated at today's prices) would cost on the order of 95 million dollars per year.

In summary, the alternatives described above do not offer the operating flixibility of the proposed action nor could most of them be completed as rapidly as the proposed action. The alternatives of shipping the spent fuel to a reprocessing facility, an independent storage facility or to another reactor would be more expensive than the proposed action and either might pre-empt storage space needed by another utility. The alternative of ceasing operation of the facility also would be more expensive than the proposed action because of the need to provide fossil fuel replacement power. In addition to the economic advantages of the proposed action, we have determined that the expansion of the SFP would have a negligible environmental impact. Accordingly, deferral or severe restriction of the action here proposed would result in substantial harm to the public interest.

Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Date: November 15, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS & ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

AND NEGATIVE DECLARATION

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation, which revised Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant located in Wayne County, New York. The amendment is effective as of its date of issuance.

This amendment authorizes changes in the design of Ginna spent fuel storage pool from that reviewed and approved in the operating license review and as described in the R. E. Ginna Nuclear Power Plant Final Safety Analysis Report. The changes will increase spent fuel storage capacity from 210 to 595 assemblies.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on June 14, 1976 (41 F. R. 24006). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action.

For further details with respect to this action, see (1) the application for amendment dated January 30, as supplemented by letters dated May 19, June 3, August 5 and September 29, 1976, (2) Amendment No. 11 to Provisional License No. DPR-18 and (3) the Commission's related Safety Evaluation and Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Lyons Public Library, 67 Canal Street, Lyons, New York 14489 and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 15th day of November 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors