

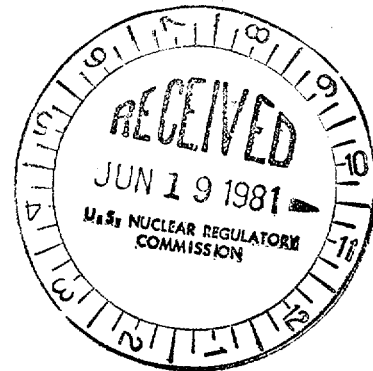


UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

June 3, 1981

Docket No. 50-244  
LS05-81-06-017

Mr. John Maier  
Vice President  
Electric and Steam Production  
Rochester Gas & Electric Corporation  
89 East Avenue  
Rochester, New York 14649



Dear Mr. Maier:

SUBJECT: DECAY HEAT REMOVAL CAPABILITY

The Commission has issued the enclosed Amendment No. 43 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment responds in part to your application notarized November 12, 1980 (submitted by letter dated November 17, 1980). Your application was in response to our letter dated June 11, 1980 to all pressurized water reactor licensees.

The NRC staff letter of June 11, 1980 requested proposed technical specification changes to provide for redundancy in decay heat removal capability in all modes of operation. Model Standard Technical Specifications were included for you to use as guidance in developing your plant-specific specifications.

We reviewed your proposed specifications and determined that changes were necessary in order that they conform with the intent of our model specifications. These changes were mutually agreed upon during telephone conversations with members of your staff. The proposed specifications, thus amended, provide added redundancy in the operability of decay heat removal capability in all modes of operation. They also add surveillance requirements to ensure operability of the reactor coolant loops when the reactor is in any of the various modes of operation. Because the proposed specifications are more conservative than the existing specifications, we have determined that they are acceptable.

The amendment does not authorize a change in effluent types or total amounts nor an increase in power level, and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

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The amendment does not involve significant new safety information of a type not considered by a previous Commission safety review of the facility. It does not involve a significant increase in the probability or consequences of an accident, does not involve a significant decrease in a safety margin, and, therefore, does not involve a significant hazards consideration. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by this action and that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

The remainder of your application notarized November 12, 1980, which concerned technical specifications regarding minimum water level to be maintained above irradiated fuel assemblies during refueling operations, was resolved with the issuance of Amendment No. 36 to Provisional Operating License DPR-18. Amendment 36 was dated March 2, 1981.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

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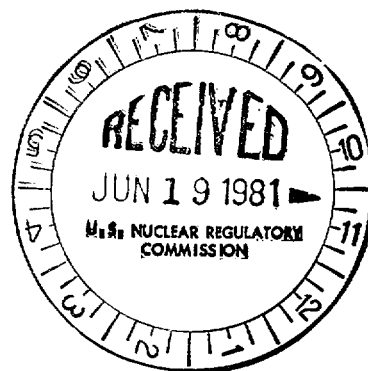
Dennis M. Crutchfield, Chief  
Operating Reactors Branch #5  
Division of Licensing

Enclosures:

- 1. Amendment No. to License No. DPR-18
- 2. Notice of Issuance

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*J. GARNER*  
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*See minor comment, p. 1. of ltr. - changed request in second # to post tense. AS 6/3/81*

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SURNAME	HSmith:cc	DSnaider	Ketchen	DCrutchfield	Glathas		
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Mr. John E. Maier

- 3 -

June 3, 1981

cc

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 43  
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) notarized November 12, 1980 (submitted by letter dated November 17, 1980), 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 43, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch #5  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 3, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 43

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Change the Technical Specifications contained in Appendix A of License No. DPR-18 as indicated below. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

<u>REMOVE</u>	<u>INSERT</u>
1-1	1-1
3.1-1	3.1-1
3.1-2	3.1-2
3.1-3	3.1-3
3.1-4	3.1-4
--	3.1-4a*
--	3.1-4b
--	3.1-4c
--	3.1-4d*
--	3.1-4e*
3.8-2a	3.8-2a
4.3-3	4.3-3
--	4.3-4
4.11-2	4.11-2

\* Included for pagination purposes only.

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## TECHNICAL SPECIFICATIONS

### 1.0. DEFINITIONS

The following terms are defined for uniform interpretation of the specification.

#### 1.1 Thermal Power

The rate that the thermal energy generated by the fuel is accumulated by the coolant as it passes through the reactor vessel.

#### 1.2 Reactor Operating Modes

<u>Mode</u>	Reactivity	Coolant
	<u><math>\Delta k/k\%</math></u>	<u>Temperature</u> (°F)
Refueling	$\leq -10$	$T_{avg} \leq 140$
Cold Shutdown	$\leq -1$	$T_{avg} \leq 200$
Hot Shutdown	$\leq -1$	$T_{avg} \geq 540$
Operating	$\geq 0$	$T_{avg} \sim 580$

#### 1.3 Refueling

Any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted.

#### 1.4 Operable

Capable of performing all intended functions in the intended manner.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Coolant System

Applicability:

Applies to the operating status of the Reactor Coolant System when fuel is in the reactor.

Objective:

To specify those conditions of the Reactor Coolant System which must be met to assure safe reactor operation.

Specification:

3.1.1 Operational Components

3.1.1.1 Reactor Coolant Loops

- a. When the reactor power is above 130 MWT (8.5%), both reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation.
- b. If the conditions of 3.1.1.1.a are not met, then immediate power reduction shall be initiated under administrative control. If the shutdown margin meets the one loop requirements of Figure 3.10-2, then the power shall be reduced to less than 130 MWT. If the one loop shutdown margin of Figure 3.10-2 is not met, the plant shall be taken to the hot shutdown condition and the one loop shutdown margin shall be met.



- c. Except for special tests, when the average coolant temperature is above 350°F, or when the reactor is at hot shutdown or is critical with the reactor power less than or equal to 130 MWT (8.5%), at least one reactor coolant loop and its associated steam generator and reactor coolant pump shall be in operation. The other loop and its associated steam generator must be operable so that heat could be removed via natural circulation. However, both reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.
- d. If the conditions of 3.1.1.1.c are not met, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.
- e. When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F, at least two of the following coolant loops shall be operable:
- (i) reactor coolant loop A and its associated steam generator and reactor coolant pump.
  - (ii) reactor coolant loop B and its associated steam generator and reactor coolant pump.

- (iii) residual heat removal loop A.\*
  - (iv) residual heat removal loop B.\*
- f. Except during steam generator crevice cleaning operations, while at cold shutdown or when the average coolant temperature is between 200°F and 350°F, at least one of the coolant loops listed in paragraph 3.1.1.1.e shall be in operation. However, both reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.
- g. If the conditions of 3.1.1.1.e are not met, immediately initiate corrective action to return the required loops to operable status, and if not in cold shutdown already, be in cold shutdown within 24 hours.
- h. If the conditions of 3.1.1.1.f are not met, then suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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\* The normal or emergency power source may be inoperable while in cold shutdown.

- i. At least one reactor coolant pump or the residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.
- j. At least one reactor coolant pump shall be in operation for a planned transition from one Reactor Operating Mode to another involving an increase in the boron concentration of the reactor coolant, except for emergency boration.
- k. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures  $\leq 330^{\circ}\text{F}$  unless 1) the pressurizer water volume is less than 324 cubic feet (38% level) or 2) the secondary water temperature of each steam generator is less than  $50^{\circ}\text{F}$  above each of the RCS cold leg temperatures.

3.1.1.2 Steam Generator

- a. The temperature difference across the tube sheet shall not exceed  $100^{\circ}\text{F}$ .

3.1.1.3 Safety Valves

- a. During cold shutdown or refueling when the reactor head is bolted on the vessel, at least one pressurizer code safety valve shall be operable with a lift setting of 2485 psig  $\pm 1\%$ .
- b. If the conditions of 3.1.1.3.a are not met, immediately suspend all operations involving positive reactivity changes and place an operable RHR loop into operation in the shutdown cooling mode.

- c. Whenever the reactor is at hot shutdown or critical, both pressurizer code safety valves shall be operable with a lift setting of 2485 psig  $\pm$  1%.
- d. If one pressurizer code safety valve is not operable while the reactor is at hot shutdown or critical, then either restore the inoperable valve to operable status within 15 minutes or be in at least hot shutdown within 6 hours and below a Tavg of 350°F within an additional 6 hours.

3.1.1.4 Relief Valves

- a. Both pressurizer power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor is at hot shutdown or critical.
- b. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.
- c. With one or more block valves(s) inoperable, within 1 hour either restore the block valves(s) to operable status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

3.1.1.5 Pressurizer

Whenever the reactor is at hot shutdown or critical the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hrs. or have the RHR system in operation within an additional 6 hrs.

Bases:

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above 1.30 during all normal

Change No. ~~12~~  
Amendment No. ~~23, 26, 42~~, 43

operations and anticipated transients. Heat transfer analyses<sup>(1)</sup> show that reactor heat equivalent to 130 MWT (8.5%) can be removed by natural circulation alone. Therefore operation with one operating reactor coolant loop while below 130 MWT provides adequate margin.

The specification permits an orderly reduction in power if a reactor coolant pump is lost during operation between 130 MWT and 50% of rated power.<sup>(2)</sup> Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than one which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase.

When the reactor coolant system average temperature is above 350°F, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require one loop be in operation and the other loop be capable of removing heat via natural circulation.

When the reactor coolant system average temperature is between 200°F and 350°F or while in cold shutdown, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity

changes in the reactor. Mixing of the reactor coolant will be sufficient to prevent a sudden increase in reactivity if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant. When the boron concentration of the reactor coolant system is to be increased, the process must be uniform to prevent sudden reactivity increases in the reactor during subsequent startup of the reactor coolant pumps. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump is running while the change is taking place. Emergency boration without a reactor coolant pump in operation is not prohibited by this specification.

Prohibiting reactor coolant pump starts without a large void in the pressurizer or without a limited RCS temperature differential will prevent RCS overpressurization due to expansion of cooler RCS water as it enters a warmer steam generator. A 38% level in the pressurizer will accommodate the swell resulting from a reactor coolant pump start with a RCS temperature of 140°F and steam generator secondary side temperature of 340°F, or the maximum temperature which usually exists prior to cooling the reactor with the RHR system.

Temperature requirements for the steam generator correspond with measured NDT for the shell and allowable thermal stresses in the tube sheet.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against overpressurization.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The requirement that 100 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown and during cooldown. (3)

#### References

- (1) FSAR Section 14.1.6
- (2) FSAR Section 7.2.3
- (3) Letter from L. D. White, Jr. to D. L. Ziemann, USNRC, dated October 17, 1979



flange. If this condition is not met, all operations involving movement of fuel or control rods in the reactor vessel shall be suspended.

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; no operations which may increase the reactivity of the core shall be made.

3.8.3 If the conditions of 3.8.1.d are not met, then in addition to the requirements of 3.8.2, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

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

4.3.4 Relief Valves

4.3.4.1 Each PORV shall be demonstrated operable at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.3.4.2 Except during cold and refueling shutdown each block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel unless the valve is already closed.

4.3.5 Reactor Coolant Loops

4.3.5.1 When reactor power is above 130 MWt (8.5%), the reactor coolant pumps shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.2 When the average coolant temperature is above 350°F but the reactor is not critical, when the reactor is at hot shutdown, or when the reactor is critical but reactor power is less than or equal to 130 MWt (8.5%):

- a) the operating reactor coolant pump(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours, and
- b) if a reactor coolant pump is not operating, but must be operable, it shall be demonstrated operable once per 7 days by verifying correct breaker alignments and indicated power availability.

4.3.5.3 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F, and fuel is in the reactor, the following shall be performed to demonstrate a loop is operable. Tests need not be performed if a loop is not relied upon to satisfy the requirements of Specification 3.1.1.1.e.

- a) to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.
- b) to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.

4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

Basis:

This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to neutron irradiation and the thermal environment. The test data obtained from this program will be used to determine the conditions under which the reactor vessel can be operated with adequate margins of safety against fracture throughout its service life.

The surveillance requirements on pressurizer equipment will assure proper performance of the pressurizer function and give early indication of malfunctions.

- d. Flow shall be maintained through the system using either the filter or bypass flow path for at least 15 minutes each month.
- 4.11.1.2 After each replacement of a charcoal filter drawer or after any structural maintenance on the charcoal housing for the spent fuel pit charcoal adsorber system, the condition of Specification 4.11.1.1.b shall be demonstrated for the affected portion of the system.
- 4.11.2 Residual Heat Removal and Coolant Circulation
- 4.11.2.1 When the reactor is in the refueling mode and fuel is in the reactor, at least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 4 hours.
- 4.11.2.2 When the water level above the top of reactor vessel flange is less than 23 feet, both RHR pumps shall be verified to be operable by performing the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a.
- 4.11.3 Water Level - Reactor Vessel
- 4.11.3.1 The water level in the reactor cavity shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods in containment.

Basis

The measurement of the air flow assures that air is being withdrawn from the spent fuel pit area and passed through the adsorbers. The flow is measured prior to employing the adsorbers to establish that

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-244ROCHESTER GAS AND ELECTRIC CORPORATIONNOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL  
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Provisional Operating License No. DPR-18, to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant (the facility) located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment modifies the provisions of the Technical Specifications to provide for redundancy in decay heat removal capability in all modes of operation.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment notarized November 12, 1980 (transmitted by letter dated November 17, 1980), and (2) Amendment No. 43 to License No. DPR-18, including the Commission's letter of transmittal. 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 Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of item (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 3rd day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

  
Dennis M. Crutchfield, Chief  
Operating Reactors Branch #5  
Division of Licensing