APR 18 1979

Docket No. 50-244

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Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. White:

The Commission has issued the enclosed Amendment No. 26 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated October 11, 1978 (which was transmitted by letter dated October 18, 1978), as supported by the analysis submitted by your letter dated July 29, 1977.

The amendment revises the Technical Specifications to include requirements for the Reactor Coolant System overpressurization protection modification.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 26 to DPR-18
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Const ccp

7905090205

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE➤ | DOR:ORB#2 | DOR:ORB#2 | OELD | DOR:AD/S&P | DOR:ORB#2 | |
| SURNAME➤ | HSmith:sah | JShea | KETCHEN | RVollmer | DLZiemann | |
| DATE➤ | 4/6/79 | 3/19/79 | 3/ /79 4/12/79 | 4/18/79 | 4/18/79 | |

Re: Overpressure Protection

William O. Miller, Chief
License Fee Management Branch, ADM

Hazel

Date: 11/1/78
Amended Form Date:

FACILITY AMENDMENT CLASSIFICATION - DOCKET NO(S). 50-244

Licensee: Rochester Gas

Plant Name and Unit(s): Ginna

License No(s): DPR-18

Mail Control No: 7810190068

Request Dated: 10/18/78 (amd 10/11/78)

Fee Remitted: Yes ✓ No ____

Assigned TAC No: 6740

Licensee's Fee Classification: Class I ____, II ____, III ✓, IV ____, V ____, VI ____, None ____.

Amendment No. 26

Date of Issuance 4/18/79

[X] 1. This request has been reviewed by DOR/DPM in accordance with Section 170.22 of Part 170 and is properly categorized.

[ ] 2. This request is incorrectly classified and should be properly categorized as Class ____. Justification for classification or reclassification: ____

[X] 3. ~~Additional information is required to properly categorize the request:~~

* We have reassessed our preliminary fee determination and have determined that it remains valid, ~~until such time that a different position is agreed upon between LFMB and DOR.~~

J. Shea D. Ziemann Date 4/16/79

[ ] 4. This request is a Class ____ type of action and is exempt from fees because it:

(a) ____was filed by a nonprofit educational institution,

(b) ____was filed by a Government agency and is not for a power reactor,

(c) ____is for a Class ____ (can only be a I, II, or III) amendment which results from a written Commission request dated ____ for the application and the amendment is to simplify or clarify license or technical specifications, has only minor safety significance, and is being issued for the convenience of the Commission, or

(d) ____other (state reason therefor): ____

KS 10/30/78

PO'C
PO'Connor

~~* verified w/ J Carter on 3/6/79. KS~~

R. Silver D L Ziemann
Division of Operating Reactors/Project Management

[ ] The above request has been reviewed and is exempt from fees.

Attached: incoming

William O. Miller, Chief
License Fee Management Branch

Date

LFMB 6/78

cc
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LeBoeuf, Lamb, Leiby & MacRae
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Mr. Michael Slade
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Webster, New York 14580

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U. S. Environmental Protection
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ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

*(w/cpy of application dtd 10/11/78)

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. 26
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 October 11, 1978 (transmitted by letter dated October 18, 1978), as supported by the analysis submitted by letter dated July 29, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of 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. 26 , 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 L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 26

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| i | i |
| ii | ii |
| 3.1-2 | 3.1-2 |
| 3.1-4 | 3.1-4 |
| 3.2-3 | 3.2-3 |
| 3.2-4 | 3.2-4 |
| 3.3-4 | 3.3-4 |
| 3.3-12 | 3.3-12 |
| -- | 3.15-1 |
| -- | 4.16-1 |
| 6.9-10 | 6.9-10 |

# TABLE OF CONTENTS



Amendment No. 15, 26

TABLE OF CONTENTS (cont.)

3.1.1.1 d. 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.

e. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures ($\leq$ 330°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°F above each of the RCS cold leg temperatures.

3.1.1.2 Steam Generator

a. One steam generator shall be capable of performing its heat transfer function whenever the average coolant temperature is above 350°F.

b. The temperature difference across the tube sheet shall not exceed 100°F.

3.1.1.3 Safety Valves

a. At least one pressurizer code safety valve shall be operable whenever the reactor head is on the vessel.

b. Both pressurizer code safety valves shall be operable whenever the reactor is critical.

Bases:

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.

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.[2] 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.

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

References

(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

Amendment No. 26

c. One boric acid tank may be out of service provided a minimum of 2,000 gallons of a 12% to 13% by weight boric acid solution at a temperature of at least 145°F is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.

d. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.

3.2.4 Whenever the reactor coolant system is ≥ 200°F and is being cooled by the RHR system and the overpressure protection system is not operable, at least one charging pump shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull-stop position.

Basis:

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] This is normally accomplished by using either one of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents (12% concentration of boric acid) to the charging pumps.

(2) The charging pumps can take suction from the refueling water storage tank. (2,000 ppm boron solution)

(3) The safety injection pumps can take their suctions from either the boric acid tanks or the refueling water storage tank.

The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life. Approximately 1800 gallons of the 12% to 13% solution of boric acid are required to meet cold shutdown conditions.[2] Thus, a minimum of 2000 gallons in the boric acid tanks is specified. An upper concentration limit of 13% boric acid in the tank is specified to maintain solution solubility at the specified low temperature limit of 145°F. Two channels of heat tracing are installed on lines normally containing concentrated boric acid solution to maintain the specified low temperature limit.

Placing a charging pump in pull-stop whenever the reactor coolant system temperature is $\geq$ 200°F and is being cooled by RHR without the overpressure protection system operable will prevent inadvertant overpressurization of the RHR system should letdown be terminated.(3)

References:

(1) FSAR, Section 9.2

(2) FSAR, Page 9.2-37

(3) L. D. White, Jr. letter to A. Schwencer, NRC, dated February 24, 1977

(ii) The two reactor coolant drain tank pumps shall be tested and their operability demonstrated prior to initiating repairs of the inoperable residual heat removal pump.

d. One residual heat exchanger may be out of service for a period of no more than 24 hours.

e. Any valve required for the functioning of the safety injection or residual heat removal systems may be inoperable provided repairs are completed within 12 hours. Prior to initiating repairs, all valves in the systems that provide the duplicate function shall be tested to demonstrate operability.

f. Power may be restored to any valve referenced in 3.3.1.1 g for the purposes of valve testing providing no more than one such valve has power restored and provided testing is completed and power removed within 12 hours.

3.3.1.3 Except during diesel generator load and safeguard sequence testing or when the vessel head is removed or the steam generator manway is open no more than one safety injection pump shall be operable whenever the temperature of one or more of the RCS cold legs is $\leq$ 330°F.

3.3.1.3.1 Whenever only one safety injection pump may be operable by 3.3.1.3 at least two of the three safety injection pumps shall be demonstrated inoperable a minimum of once per twelve hours by verifying that the control switches are in the pull-stop position.

3.3.2 <u>Containment Cooling and Iodine Removal</u>

3.3.2.1 The reactor shall not be made critical except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 4500 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. At least two containment spray pumps are operable.

c. At least three fan cooler units are operable.

until repairs were effected. (6)(7)

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident. (8)

The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet.

The limitation of no more than one safety injection pump to be operable and the surveillance requirement to verify that two safety injection pumps are inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.


References

(1) FSAR Section 9.3

(2) FSAR Section 6.2

(3) FSAR Section 6.3

(4) FSAR Section 14.3.5

(5) FSAR Section 1.2

(6) FSAR Section 9.3

(7) FSAR Section 14.3

(8) FSAR Section 9.4

3.15 **Overpressure Protection System**

**Applicability**

Applies whenever the temperature of one or more of the RCS cold legs is $\leq$ 330°F.

**Objective**

To prevent overpressurization of the reactor coolant system.

**Specification**

3.15.1 At least one of the following overpressure protection systems shall be operable:

a. Two pressurizer power operated relief valves (PORVs) with a lift setting of $\leq$ 435 psig, or

b. A reactor coolant system vent of $\geq$ 1.1 square inches.

3.15.1.1 With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the RCS through a 1.1 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.2 With both PORVs inoperable, depressurize and vent the RCS through a 1.1 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.3 Use of the overpressure protection system to mitigate a RCS pressure transient shall be reported in accordance with 6.9.3.

**Basis**

The operability of two pressurizer PORVs or an RCS vent opening of greater than 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq$ 330°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq$ 50°F above the RCS cold leg temperature or (2) the start of a safety injection pump and its injection into a water solid RCS.(1)

**References:**

(1) L. D. White, Jr. letter to A. Schwencer, NRC, dated July 29, 1977

## 4.16 Overpressure Protection System

Applicability:

Applies to the reactor coolant system overpressure protection system.

Objective:

To verify that the overpressure protection system will function properly if needed.

Specification

4.16.1 Each PORV shall be demonstrated operable by:

a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

b. Performance of a channel calibration on the PORV actuation channel at least once per 18 months.

c. Verifying the PORV isolation valve is open at least once per 72 hours when the overpressure protection system is required to be operable.

4.16.2 The RCS vent(s) shall be verified to be open at least once per 12 hours when the vent(s) is being used for overpressure protection except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position. Then verify these valves open at least once per 31 days.

(2) Annually: A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 200 mrem/yr and their associated man rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

(NOTE: This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.)

d. Reactor Overpressure Protection System Operation

In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any other corrective action necessary to prevent recurrence.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 26 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

## Introduction

Our letter of August 11, 1976 (Reference 1) requested an analysis of the Reactor Coolant System (RCS) response to pressure transients that could occur during startup or shutdown and recommended the inclusion of design modifications determined to be necessary to preclude exceeding the limits specified in 10 CFR Part 50, Appendix G. By letter dated July 29, 1977, (Reference 9) Rochester Gas and Electric Corporation (RG&E) submitted a plant-specific analysis in support of the proposed reactor vessel over-pressure protection system (OPS) for the R. E. Ginna Nuclear Power Plant (Ginna), which supplemented other documentation previously submitted by RG&E (References 2-4, 6-8). The OPS has been designed to protect the primary system coolant pressure boundary from the effects of operating errors during cold shutdown when the primary system is solid, which could otherwise produce primary pressure excursions above allowable limits (References 6, 7, and 9). RG&E submitted the proposed Technical Specifications for the OPS by application dated October 11, 1978, (which was transmitted by letter dated October 18, 1978, Reference 12).

During the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. The term "pressure transients," as used in this report, refers to events during which the temperature pressure limits of the reactor vessel, as shown in the Ginna Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperature (less than 200°F) when the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG 0138 (Reference 10) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors.

## 1.0 Discussion

### 1.1 Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, since reactor vessel steels are less tough and could possibly fail in a brittle manner if subjected to high pressures at low temperatures, power reactors have always operated with restrictions on the pressure allowed during startup and shutdown operations.

At operating temperatures, the pressure allowed by 10 CFR Part 50 Appendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating PWRs, including Ginna, did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

### 1.2 Regulatory Actions

By letter dated August 11, 1976, (Reference 1) the NRC requested that RG&E begin to design and install systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. Satisfactory administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

RG&E responded (References 2, 3, and 4) with preliminary information describing interim measures to prevent these transients along with some discussion of proposed hardware. Installation of a low pressure actuation setpoint on the pressurizer air operated relief valves was proposed.

RG&E participated as a member of a Westinghouse user's group formed to support the analysis effort required to verify the adequacy of the proposed system to prevent overpressure transients. Using input data generated by the user's group, Westinghouse performed transient analyses (Reference 11) which were used as the basis for plant-specific analysis.

The NRC requested additional information concerning the proposed procedural changes and the proposed hardware changes (Reference 5). RG&E provided the required responses (References 6 and 7). Reference 9 transmitted the plant-specific analysis for Ginna.

## 1.3 Design Criteria

Through a series of meetings and correspondence with PWR vendors and licensees, the staff developed a set of criteria for an acceptable overpressure mitigating system. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of those allowed by Appendix G. Specific criteria for system performance are:

1. Operator Action: No credit can be taken for operator action for ten minutes after the operator is aware of a transient.

2. Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3. Testability: The system must be testable on a periodic basis consistent with the system's employment.

4. Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

   The staff also requested the licensee to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, along with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

## 1.4 Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators, and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

On Westinghouse designed plants, the most common cause of the overpressure transients to date has been isolation of the letdown path. Letdown during low pressure operations is via a flowpath through the RHR system. Thus, isolation of RHR can initiate a pressure transient if a charging pump is left running. Although other transients occur with lower frequency, those which result in the most rapid pressure increases were identified by the staff for analysis. The most limiting mass input transient identified by the staff is inadvertant injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a 50°F temperature difference between the water in the reactor vessel and the water in the steam generator.

Based on the historical record of overpressure transients and the imposition of more effective administrative controls, we consider the limiting events identified above an acceptable basis for analyses of the proposed OPS.

## 2.0 System Description and Evaluation

RG&E adopted the "Reference Mitigating System" concept developed by Westinghouse and the user's group. RG&E proposed to modify the actuation circuitry of the existing air operated pressurizer relief valves to provide a low pressure setpoint at 435 psig during startup and shutdown conditions. The new low pressure Power Operated Relief Valve (PORV) actuation circuitry uses multiple pressure sensors, power supplies and logic trains to improve system reliability. Each of the two PORV's is manually enabled using two keylock switches, one to line up the air supply and the other to enable the low pressure setpoint. When the reactor vessel is at low temperatures with the Overpressure Protective System (OPS) enabled, a pressure transient is terminated below the Appendix G limit by automatic opening of the PORV's. An enabling alarm monitors the RCS temperature, the position of the keylock switches (2 per channel), and the upstream isolation valve position. The OPS is enabled at a temperature of 330°F during plant cooldown and is disabled at the same temperature during plant heatup. The enabling alarm alerts the operator in the event the RCS temperature is below 330°F and OPS valve or switch alignment has not been completed. On this basis, we consider the pressurizer relief valves with a manually enabled low pressure setpoint to be an acceptable concept for an overpressure mitigating system.

## 2.1 Air Supply

The Ginna PORV's are gate valves that are spring closed and nitrogen opened. Each of the two PORV's receives actuating nitrogen, ($N_2$), from either the plant instrument air (nitrogen) system or a backup nitrogen accumulator. The accumulators are sized to provide sufficient actuating $N_2$ for ten minutes of PORV operation (about 150 cycles) without operator action during the most limiting transient and a loss of the plant instrument air system. Low pressure alarms are installed in the control room to alert the operator to a low nitrogen accumulator pressure condition. The staff therefore finds the Ginna OPS normal and alternate nitrogen supplies acceptable.

## 2.2 Electrical Instrumentation and Control

### 2.2.1 Instrumentation and Alarms Available to Operator

In addition to narrow range pressurizer pressure indication, reactor coolant system wide range pressure indication and recording (0-3000 psig) and low pressure indication (0-700 psig) are provided on the main control board. This pressure indication is provided by PT-420, PT-429, PT-430, PT-431, and PT-449 shown on drawing 33013-424. An overpressure alarm which incorporates two setpoints is also provided. One setpoint is variable and follows the Technical Specification limit. The other setpoint alarms at a given differential pressure, determined by the operator, below the Technical Specification limit. Both setpoints alarm and light on the plant computer.

Indication of pressurizer relief valve operation are valve light indication and "pressurizer relief line high temperature 20°F above ambient."

The installed pressure and temperature instrumentation at Ginna will provide a permanent record over the full range of both pressure and temperature.

### 2.2.2 Disabling Components

When power is removed from valve motor operators under administrative control provisions, the status of the lights and indicators available to verify their proper alignment and the administrative controls for removing power from a valve motor operator or a pump are as follows:

a. Valves are provided with red and green control board status lights. All safeguards valves also have safeguards white light indication. Deenergized valves retain normal status light indication since indication is provided by the D. C. control circuitry. Indication is lost only if the D. C. control power fuses are removed at the motor control center breaker panel.

b. Removing power from a valve motor operator or pump motor is accomplished at a motor control center or 480 volt bus. A pump may be put in "pull stop" at the main control board.

### 2.2.3 Testability

Testability has been provided. RG&E has stated that operability will be verified prior to solid system, low temperature operation by use of the remotely operated isolation valve, enable/disable switches and normal electronics surveillance methodology. Additionally, the actuation circuitry logic will be tested during each refueling outage. Testing requirements will be incorporated in the Technical Specifications as discussed in Section 4.2 of this evaluation.

### 2.2.4 Conclusion

The design of the Ginna low temperature overpressure protection system in the areas of electrical, instrumentation and control (EI&C) is in accordance with those design criteria orginally prescribed by the NRC and later expanded during subsequent discussions with RG&E.

We find the EI&C aspects of the proposed design acceptable on the basis that: (1) the proposed overpressure protection system complies with IEEE Std 279-1971, and seismic criteria as identified in Section 2.0; (2) the system is redundant and satisfies the single failure criterion; (3) the design requires no operator action prior to ten minutes after the operator receives an overpressure action alarm; (4) the system is testable on a periodic basis; and (5) the proposed changes to the Technical Specifications would reduce the probability of overpressurization events to acceptable levels.

## 2.3 Appendix G

The Appendix G curve submitted by RG&E for purposes of overpressure transient analysis is based on 10.6 effective full power years irradiation. The zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions. Margins of 60 psig and 10°F are included for possible instrument errors. The Appendix G limit at 100°F according to this curve is 535 psig. We therefore conclude that use of this curve is acceptable as a basis for overpressure mitigating system performance.

## 2.4 Setpoint Analysis

The one loop version of the LOFTRAN (Reference WCAP 7907) code was used to perform the mass input analyses. The four loop version was used for the heat input analysis. Both versions require some input modeling and initialization changes. LOFTRAN is currently under review by the staff and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV setpoint overshoot. The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the setpoint magnitude. The PORV setpoint is adjusted so that given the setpoint overshoot, the resultant pressure is still below that allowed by Appendix G limits.

RG&E presented the following Ginna plant characteristics to determine the pressure reached for the design basis pressure transients:

| | |
|---|---|
| SI Pump Flowrate @ 435 psig | 60 lb/sec |
| RCS Volume | 6065 $ft^3$ |
| PORV Opening Time | 3 sec |
| S G Heat Transfer area | 44,000 $ft^2$ |
| Relief Valve setpoint | 435 psig |

Westinghouse identified certain assumptions used in LOFTRAN that are conservative, and tend to overpredict the peak RCS pressure in the design base transients. These are listed below, along with some plant parameters Westinghouse has assumed in the generic analysis that RG&E has identified to be conservative relative to the actual Ginna values.

1) One PORV was assumed to fail.

2) The RCS was assumed to be rigid with respect to metal expansion.

3) No credit was taken for the reduction in reactor coolant bulk modulus at RCS temperatures above 100°F (constant bulk modulus at all RCS temperatures).

4) No credit was taken for the shrinkage effect caused by low temperature SI water added to higher temperature reactor coolant.

5) The entire volume of water of the steam generator secondary was assumed available for heat transfer to the primary. In reality, the liquid immediately adjacent and above the tube bundle would be the primary source of energy in the transient.

6) The overall steam generator heat transfer coefficient, U, was assumed to be the free convective heat transfer coefficient of the secondary $h_{sec}$. The forced convective heat transfer coefficient of the primary, $h_{pri}$ and the tube metal resistance have been ignored thus resulting in a conservative (high) coefficient.

7) The RCP startup time assumed in the heat input analysis was 9-10 sec whereas the actual RCP startup time is about 22 sec.

8) The SI pump startup time assumed in the mass input analysis was 1.64 sec whereas the actual SI pump startup time is about 3.0 sec.

Based on the above, we find these assumptions acceptable.

2.4.1 Mass Input Case

The inadvertant start of a safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case.

Westinghouse provided RG&E with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates PORV setpoint overshoot for this transient system volume, relief valve opening time and relief valve setpoint. These sensitivity analyses were then applied to the Ginna plant parameters to obtain a conservative estimate of the PORV setpoint overshoot. We find this method of analysis acceptable.

Using the Westinghouse methodology, the Ginna PORV setpoint overshoot was determined to be slightly less than 100 psi. With a relief valve setpoint of 435 psig, a final pressure of 535 psig is reached for the worst case mass input transient. Since the 10.6 EFPY Appendix G limit at temperatures above 100°F is above 535 psig, we have concluded that the system performance is acceptable with a 435 psig low pressure relief valve setpoint.

2.4.2 Heat Input Case

Inadvertant startup of a reactor coolant pump with a primary to secondary temperature differential across the steam generator of 50°F, and with the plant in a water solid condition, was selected as the limiting heat input case. For the heat input case, Westinghouse provided RG&E with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV setpoint overshoot as a function of RCS volume, steam generator UA and initial RCS temperature. For this transient, the reference relief valve selected was assumed to have a total opening time of three seconds from the instant the signal to open is received until the valve reached the full open position.

The calculated final pressure for the heat input transient for a fixed $\Delta T$ of 50°F depends on the initial RCS temperature and is given here:

| RCS Temperature | Maximum Pressure |
|---|---|
| 100°F | 457 psig |
| 140°F | 480 psig |
| 180°F | 508 psig |
| 250°F | 554 psig |

In all these cases, for the given RCS temperature, the Appendix G limits are not exceeded.

We find that the analyses of the limiting mass input and heat input cases show a maximum pressure transient below that allowed by Appendix G limits and are therefore acceptable.

## 2.5 Plant Modification

RG&E installed most of the equipment comprising the final OPS during the 1978 refueling outage.

$N_2$ supply valves of the proper seismic qualification are not currently available. RG&E has proposed using non-seismically qualified valves until the proper valves can be installed. Since the PORV $N_2$ supply system operability is not affected by the installation of non-seismically qualified valves, and the liklihood of a seismic event is low, we conclude that the use of non-seismically qualified valves in the PORV $N_2$ supply system during this interim period is acceptable.

The OPS enabling alarm installed during the 1978 refueling outage does not monitor the PORV upstream MOV position. However, the alarm (one per PORV) will monitor RCS temperature and the position of the enabling switches (2 per PORV). RG&E has agreed to install the equipment necessary for the monitoring of the MOV's position during the first shutdown of sufficient duration after the equipment becomes available. Also, RG&E has agreed to ensure the proper positioning of the MOV's should the OPS be required in the interim period. This interim arrangement is acceptable pending completion of this modification. Should any delay be encountered which could impact these unfinished modifications, RG&E should promptly notify the NRC.

## 3.0 Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by RG&E, a defense in-depth approach is adopted using procedural and administrative controls. Specific conditions required to assure that the plant is operated within the bounds of the analysis are adequately described in the Technical Specifications.

## 3.1 Procedures

A number of provisions for prevention of pressure transients are contained in the Ginna operating procedures. These procedures require that an acceptable RCS temperature profile be achieved prior to startup (and jogging) of a reactor coolant pump (RCP) with the RCS in a water-solid condition. In addition, plant shutdown and cooldown procedures call for one RCP to be run until the RCS temperature has been lowered to 150°F, thus reducing the possibility of a significant RCS temperature assymetry.

Also, RG&E has modified plant procedures to restrict water solid operations to only those times when absolutely necessary. For example, the plant must be maintained in a water-solid condition during RCS filling and venting operations, during hydrostatic testing of the RCS, and during plant heatup prior to bringing the RCS within water chemistry specifications.

The cooldown procedures require the safety injection signal associated with the pressurizer and steam line low pressure be blocked at approximately 2000 psig. At less than 1800 psig, the high head safety injection discharge valves to the RCS loops are shut. At approximately 1500 psig the high head SI pumps are de-energized by placing their control switches in the "pull-stop" position. In the "pull-stop" position the SI pumps cannot automatically start. The SI pumps are not re-energized while the RCS is in a cold and shutdown condition unless special surveillance testing is in progress or a SI accumulator is to be filled (only one SI pump is energized).

The diesel generator load and safeguards sequence test conducted during cold or refueling shutdown operates each safeguard train (2 pumps). However, the pump discharge valves are closed, the valve power supply breakers are open and the breaker DC control fuses are removed. During other tests the SI pumps are prohibited from starting and except during valve cycling tests, the discharge valves are shut.

We consider the procedural and administrative controls acceptable. However, we believe certain procedural and administrative controls should be included in the Technical Specifications. These are listed in the following section.

3.2 Technical Specifications

RG&E has submitted for our review, Technical Specifications (Reference 1) to be incorporated into the Ginna license. These specifications are consistent with the intent of the statements listed below.

1. Both PORVs must be operable whenever the RCS temperature is less than 330°F, except one PORV may be inoperable for seven days. If these conditions are not met, the primary system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

2. Operability of the overpressure protection system requires that the low pressure setpoint will be selected (two switches per train), the upstream isolation valves open and the backup air supply charged.

3. No more than one high head SI pump may be energized at RCS temperature below 330°F, except during the diesel generator load and safeguards sequence test.

4. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or if the SG/RCS $\Delta T$ in both loops is verified to be less than 50°F.

5. The overpressure mitigating system must be tested on a periodic basis consistent with the need for its use.

6. Failure of the Overpressure Protection System to operate when required is a reportable item.

4.0 Conclusions

The administrative controls and hardware changes made by RG&E provide additional protection for the Ginna Plant from pressure transients at low temperatures by reducing further the probability of initiation of a transient and by limiting the pressure, if such a transient should nevertheless occur, to levels less than the limits set by Appendix G.

We have concluded that the overpressure mitigating system and the proposed revisions to the Technical Specifications satisfy our requirements, are similar to those proposed and accepted by us for other PWRs, and on this basis are acceptable to NRC.

Environmental Consideration

We have determined that 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because this amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: April 18, 1979

REFERENCES


1. NRC (Schwencer) to Rochester Gas and Electric Corporation (RG&E) dated August 11, 1976.
2. RG&E (White) to NRC (Schwencer) dated September 3, 1976.
3. RG&E (White) to NRC (Schwencer) dated October 15, 1976.
4. RG&E (White) to NRC (Schwencer) dated December 8, 1976.
5. NRC (Schwencer) to RG&E (White) dated January 10, 1977.
6. RG&E (White) to NRC (Schwencer) dated February 24, 1977.
7. RG&E (White) to NRC (Schwencer) dated March 31, 1977.
8. RG&E (White) to NRC (Schwencer) dated April 26, 1977.
9. RG&E (White) to NRC (Schwencer) dated July 29, 1977.
10. "Staff Discussion of Fifteen Technical Issues listed in Attachment G November 3, 1976 Memorandum from Director NRR to NRR Staff." NUREG-0138, November 1976.
11. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, dated July 1977.
12. RG&E (LeBoeuf, Lamb, Leiby & MacRae) to NRC (Denton) dated October 18, 1978, and attached application dated October 11, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 26 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Plant (the facility) located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to include requirements for the Reactor Coolant System overpressurization protection modification.

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 dated October 11, 1978 (which was transmitted by letter dated October 18, 1978), (2) the analysis submitted by the licensee's letter dated July 29, 1977, (3) Amendment No. 26 to License No. DPR-18, and (4) the Commission's related Safety Evaluation. 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 Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of items (3) and (4) 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 18th day of April, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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Operating Reactors Branch #2
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