Docket No. 50-244

JUL 30 1985

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

See correction letter of 8/9/85

Dear Mr. Kober:

SUBJECT: SEP RELATED TECHNICAL SPECIFICATIONS

Re: R. E. Ginna Nuclear Power Plant

The Commission has issued the enclosed Amendment No.11 to Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated August 1, 1983.

The amendment approves changes to Technical Specifications relating to overpressure protection system operability, minimum refueling water storage tank volume, process-to-actuator response time testing and service water pump power alignment. The proposed change for battery discharge tests which was revised by your October 26, 1983 submittal will be addressed by separate correspondence.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on November 22, 1983 (50 FR 52824). No public comments or requests for hearing were received.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

Original signed by

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 11 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

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

July 30, 1985

Docket No. 50-244
LS05-85-07-046

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Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

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Mr. Roger W. Kober
Rochester Gas and Electric Corporation

R. E. Ginna Nuclear Power Plant

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July 30, 1985

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY 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 August 1, 1983, 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 Facility 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. 11, 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

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 30, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY 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 marginal lines indicating the area of change.

REMOVE	INSERT
3.3-2	3.3-2
3.3-4	3.3-4
3.3-7	3.3-7
3.15-1	3.15-1
--	3.15-2
4.4-11	4.4-11
4.8-2	4.8-2

a. The refueling water tank contains not less than 300,000 gallons of water, with a boron concentration of at least 2000 ppm.

b. Each accumulator is pressurized to at least 700 psig with an indicator level of at least 50% and a maximum of 82% with a boron concentration of at least 1800 ppm. Neither accumulator may be isolated.

c. Three safety injection pumps are operable.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

g. A.C. Power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D, accumulator injection valves 841 and 865, and refueling water storage tank delivery valve 856. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. As soon as appropriate modifications are complete, D.C. control power shall be removed from refueling water storage tank delivery valves 896A and B with the valves open. In the meantime, single failure protection for valves 896A and B will be provided by locking out A.C. power, remote from the control room, with operating personnel assigned specifically to restore A.C. power when the valves are required to function in the event of a loss-of-coolant accident.

h. Revisions to procedures for post-LOCA long term cooling as described in letters to the Nuclear Regulatory Commission from Rochester Gas and Electric Corporation dated April 1, 1975, April 30, 1975, and May 13, 1975, shall be implemented prior to reactor startup following the shutdown of March 10, 1975.

i. Check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.1 are still applicable.

d. One residual heat exchanger may be out of service for a period of no more than 72 hours.

e. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours. Prior to initiating valve repairs, all valves in the system 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.

g. Those check valves specified in 3.3.1.1 i may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed 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 overpressure protection system is required to be operable.

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 Containment Cooling and Iodine Removal

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. Four fan cooler units are operable.

in the hot shutdown condition. If the requirements of 3.3.3.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition.

a. One component cooling pump may be out of service provided the pump is restored to operable status within 24 hours.

b. One heat exchanger or other passive component may be out of service provided the system may still operate at 100% capacity and repairs are completed within 24 hours.

3.3.4 <u>Service Water System</u>

3.3.4.1 The reactor shall not be made critical unless the following conditions are met:

a. At least two service water pumps, one on bus 17 and one on bus 18, and one loop header are operable.

b. All valves, interlocks, and piping associated with the operation of two pumps are operable.

3.3.4.2 Any time that the conditions of 3.3.4.1 above cannot be met, the reactor shall be placed in the cold shutdown condition.

3.3.5 <u>Control Room Emergency Air Treatment System</u>

3.3.5.1 The reactor shall not be made critical unless the control room emergency air treatment system is operable.

3.15 Overpressure Protection System

Applicability

Applies whenever the temperature of one or more of the RCS cold legs is $\leq 330^{\circ}F$, or the Residual Heat Removal System is in operation.

Objective

To prevent overpressurization of the reactor coolant system and the residual heat removal system.

Specification

3.15.1 Except during secondary side hydrostatic tests in which RCS pressure is to be raised above the PORV setpoint, 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 an RCS or RHRS pressure transient shall be reported in accordance with 6.9.2.

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 $\leq 330^{o}F$. This relief capacity will also ensure that no overpressurization of the RHR system could occur. Either PORV has adequate relieving capability to protect the RCS and RHRS 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 $\leq 50^{o}F$ above the RCS cold leg temperature or (2) the start of a safety injection pump and its injection into a water solid RCS.[1,2]

References:

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

(2) SER for SEP Topics V-10.B, V-11.B, VII-3, "Safe Shutdown," dated September 29, 1981

the tendon containing 6 broken wires) shall be inspected. The acceptance criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.

b. Before reseating a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each isolation valve specified in Table 3.6-1 shall be demonstrated to be operable in accordance with the Ginna Station Pump and Valve Test Program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.4.6.2 The RESPONSE TIME of the containment isolation valves, as listed in Table 3.6-1, shall be demonstrated to be within the limit at least once per 18 months. This response time includes only the valve travel times for all valves that change position.

Basis:

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286°F.

4.8.5 Except during cold or refueling shutdowns, the suction, discharge, and cross-over motor operated valves for the Standby Auxiliary Feedwater pumps shall be exercised at intervals not to exceed one month.

4.8.6 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. These tests shall be performed prior to exceeding 5% power during a startup if the time since the last test exceeds one month.

4.8.7 At least once per 18 months, control of the standby auxiliary feed system pumps and valves from the control room will be demonstrated.

4.8.8 At least once per 18 months during shutdown

a. Verify that each automatic valve in the flow path for each auxiliary feedwater pump actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

b. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

4.8.9 Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.8.10 The RESPONSE TIME of each pump and valve required for the operation of each "train" of auxiliary feedwater shall be demonstrated to be within the limit of 10 minutes at least once per 18 months.

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet design. The flow rates will be measured at a simulated steam generator pressure of 1100 psia. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Proper functioning of the steam turbine admission valve and the feedwater pumps start will demonstrate the integrity of the steam driven pump.

Monthly testing of the Standby Auxiliary Feedwater pumps by supplying water from a condensate supply tank to the steam generators will verify their ability to meet design. The flow rate will be measured at a simulated steam generator pressure of 1100 psia.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated August 1, 1983, Rochester Gas and Electric Corporation (the licensee, RG&E) requested an amendment to the Ginna Technical Specifications (TS) which consisted of five parts: (1) revise the Overpressure Protection System (OPS) operability requirements such that the OPS will be made operable whenever the Residual Heat Removal (RHR) system is placed in operation; (2) revise the minimum refueling water storage tank (RWST) volume requirements from 230,000 gallons to 300,000 gallons; (3) delete the process-to-actuator response time testing requirement for auxiliary feedwater and containment isolation; (4) revise the service water pump class 1E power alignments to include the requirement that at least one of the pumps be aligned to each of the two redundant class 1E power supplies; and (5) revise the battery testing requirements to include the requirement for a battery discharge test.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on November 22, 1983 (50 FR 52824). No public comments or requests for hearing were received.

The proposed change on battery testing requirements was revised by a letter from RG&E dated October 26, 1983. This proposed change will be addressed in separate correspondence. The other four changes are discussed below.

2.0 EVALUATION

2.1 Overpressure Protection System Operability

The overpressure relief capacity was reviewed by the staff as part of the Systematic Evaluation Program (SEP). The results of the review were reported in Section 4.21.1 of the Integrated Plant Safety Assessment Report (IPSAR) for Ginna (NUREG-0821).

Overpressure relief capacity is required by 10 CFR Part 50 (General Design Criteria 19 and 43), as implemented by Standard Review Plan (SRP) Section 5.4.7, BTP ASB 5-1, and Regulatory Guide 1.139, for the RHR system when it is in operation; that is, when it is not isolated from the reactor coolant system (RCS). The OPS fulfills this function. At the time of the SEP review there was no procedural requirement in the TS that ensured that the OPS was in service whenever the RHR system is in service. During cooldown, the procedures placed the RHR system into service at 350°F and 360 psi, whereas the OPS was not required to be in service until 330°F. TS 3.15 "Overpressure Protection System" requires that the OPS be operable whenever the temperature of one or more of the RCS cold legs is $<$330°F. The licensee has proposed to further specify that the OPS be operable whenever the RHR system is in operation. Use of the OPS for RHR system protection is also reflected in the objective, reporting requirements section and basis of this TS. In addition, it is proposed that the wording for TS 3.3.1.3 be changed from "whenever the temperature of one or more of the RCS cold legs is 300°F" to "whenever the overpressure protection system is required to be operable." This change would make this TS consistent with the proposed change to TS 3.15. The proposed TS changes provide added assurance of protection from overpressure events, are responsive to staff requests and are therefore considered acceptable. The TS changes to be incorporated by this amendment for the OPS are consistent with the staff position and are acceptable.

2.2 Minimum Refueling Water Storage Tank Requirements

The Engineered Safety Feature (ESF) Switchover Procedures were reviewed by the staff as part of the SEP. The results of the review were reported in Section 4.23.1 of the Ginna IPSAR.

Item 19 of SRP Section 6.3 states that the complete sequence of emergency core cooling system (ECCS) operation from injection to long-term core cooling (recirculation) should be examined to see that minimal manual action is required, and that, where manual action is needed, sufficient time (generally 20 minutes) is available for the operator to respond. The time for individual operators actions suggested by ANSI standard N660 is one minute per action. Parallel actions, such as switching off both RHR pumps, are counted as one action.

The Ginna procedures for switchover from injection to recirculation did not meet current NRC criteria with respect to time for operator action. In addition, the staff noted that the procedure required that all injection flow to the core be terminated while pump suction was realigned to the containment sump. The staff therefore recommended that the switchover procedure be evaluated for improvement. As discussed in a letter from RG&E dated June 25, 1982, the licensee has developed a revised switchover approach to address these concerns.

As part of this approach, the minimum initial RWST level would be increased from 230,000 gallons to 300,000 gallons (88% level). From this level (with a 3% allowance for instrument error) it takes over 20 minutes to reach the 28% low level alarm assuming that all ESF pumps operate at runout flow rates. At the 28% low level alarm the operator must shut off one safety injection pump, one containment spray pump, and both RHR pumps. The RHR pump suction is then realigned to draw from the containment building sump. RG&E has stated that analyses show that there would be sufficient water in the sump at this time to provide adequate net positive suction head (NPSH) for the RHR pumps. During the time that the RHR pumps are not running, RWST water will be injected into the RCS by the safety injection pumps. Analyses show that one safety injection pump will maintain a sufficient flowrate to compensate for coolant boil-off and maintain vessel coolant inventory.

Once the RHR pump suction has been switched from RWST injection to the recirculation mode, the operator would shut off the remaining operating containment spray pump and safety injection pumps at the 15% low-low RWST level signal. If the RCS pressure is above the shutoff head for the RHR pumps, the operator would "piggy-back" the safety injection pump suction to the RHR pump discharge to draw water from the sump.

RG&E has provided an analysis of its revised procedure for ESF switchover following a loss-of-coolant accident. No operator action for switchover is required before 20 minutes and sufficient time is available to complete the necessary actions while maintaining adequate pump NPSH. The staff therefore finds the Ginna method for ECCS switchover from injection to recirculation mode acceptable.

The emergency operating procedures for the revised switchover method are being implemented in coordination with TMI Action Plan item I.C.1 "Short-term Accident and Procedures Review." The staff noted that there is a possibility that the 15% low-low level alarm will actuate before the RHR pump switchover sequence is complete. The staff recommends that the procedures be written with an explicit precaution to complete RHR switchover before shutting off other pumps at the low-low level alarm. This precaution would provide further assurance that core cooling will be maintained by the RHR system when the safety injection pumps are secured. The licensee should address the staff recommendation when developing the emergency operating procedures discussed above.

The licensee has proposed to increase the minimum RWST volume to be maintained per TS 3.3.1.1a from 230,000 gallons to 300,000 gallons. This change results in more water being available for core cooling and more time for the operator to respond to a loss-of-coolant accident. Therefore, the staff finds this proposed TS change acceptable.

2.3 Deletion of Response Time Testing of Selected Isolation Initiation Circuits

The licensee has proposed to delete the requirement to perform response time testing of the initiating circuits (sensor to bistable) for containment isolation (TS 4.4.6.2) and for the auxiliary feedwater system (TS 4.8.10).

The licensee has conducted response time testing from the sensor through the bistable devices, during 1981 and 1982 refueling outages, and found that the response time testing of the initiating circuits does not appear to be beneficial. This particular portion of the overall system response time is a very small fraction of the total system response time (milliseconds vs. 1 to 10 minutes). The licensee has proposed that functional testing of the actuation logic and relays be retained, while the sensor to actuated equipment bistable response time testing would be deleted.

The response time testing for ESF system was a technical assessment topic in the SEP. NUREG-0820, IPSAR for Palisades, Section 4.22 concluded that the response time testing of time-critical components (for example, diesel generator load-sequencer timing, diesel generator start times, and stroke times of important valves) are considered adequate to detect circuit problems that could contribute to degraded response time. Backfitting was not required to include the initiating circuits in the response time testing. The requirements for the response time of containment isolation valve travel and auxiliary feedwater train operation is still included in these TS.

The staff finds therefore that the proposed TS changes are consistent with NUREG-0820 findings, and the licensee's proposed changes are acceptable.

2.4 Service Water Pump Class 1E Power Alignments

The proposed change to TS 3.3.4.1a will clarify the electrical power alignment of the service water pumps. During the review of SEP Topic IX-3, "Station Service and Cooling Water Systems," it was noted that, during power operation, the Ginna TS required that two service water pumps be operable. It did not specify that at least one of these pumps be aligned to each of the two redundant Class 1E power supplies. This proposed TS change does require that the pumps be aligned on redundant Class 1E bases. This will assure that a single failure of one train of power will not affect both operable service water pumps. The staff considers this proposed change acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ACKNOWLEDGEMENT

H. Li, E. McKenna and C. Miller prepared this Safety Evaluation.

Dated: July 30, 1985