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

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 <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. 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 Service Water System

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 Control Room Emergency Air Treatment System

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°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 $\leq 330^{\circ}F$. 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 $\leq 50^{\circ}F$ 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.