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. 62
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 April 13, 1983 as supplemented on August 12, 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 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. 62, 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

Walter A. Paulson

Walter A. Paulson, Acting Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 31, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 62

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 contain the captioned amendment number and marginal lines which indicate the area of changes.

REMOVE	INSERT
3.10-3	3.10-3
6.2-1	6.2-1
6.9-3	6.9-3
--	6.9-3a*
6.9-6	6.9-6
--	6.9-6a*

*This page is included for pagination purposes only; there are no changes to the provisions contained thereon.

average power tilt ratio shall be determined once a day by at least one of the following means:

a. Movable detectors

b. Core-exit thermocouples

3.10.2.2 Power distribution limits are expressed as hot channel factors. At all times, except during low power physics tests the hot channel factors must meet the following limits:

$F_Q(Z) = (2.32/P)*K(Z)$ for $P \geq .5$

$F_Q(Z) = 4.64*K(Z)$ for $P \leq .5$

$F^N_{\Delta H} = 1.66\ [1 + .3(1-P)]$ for $0 \leq P \leq 1.00$

Where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q shall be increased by three percent to yield F_Q. If the measured F_Q or $F\Delta_H^N$ exceeds the limiting value, with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced one percent for each percent with $F\Delta_H^N$ or F_Q exceeds the limiting value, whichever is more restrictive. If the hot channel factors cannot be reduced below the limiting values within one day, the Overpower ΔT trip setpoint and the Overtemperature ΔT trip setpoint shall be similarly reduced.

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or.

c. Limit power to 75% of rated power.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips caused by transients or emergencies.

d. All core alterations shall be directly supervised by either a licensed Senior Reactor Operator or Senior reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

f. A Fire Brigade of 5 members shall be maintained on site at all times.* This excludes the two members of the minimum shift crew necessary for safe shutdown.

g. Adequate shift coverage shall be maintained without routine heavy use of overtime. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions including senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Changes to the guidelines for the administrative procedures shall be submitted to the NRC for review.

*Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to the minimum requirements.

yield a calculated dose or dose commitment greater than those forming the basis of Specifications 4.12.2.2 or 3.16.1. The report shall also contain a discussion which identifies the causes of the unavailability of milk or leafy vegetable samples and identifies locations for obtaining replacement samples in accordance with Specification 3.16.1.4.

The radioactive effluent release report shall include a discussion which identifies the circumstances which prevent any required detection limits for effluent sample analyses from being met.

The radioactive effluent release reports shall include any changes made during the reporting period to the ODCM as specified in Section 6.15, and to the Process Control Program as specified in Section 6.16. The radioactive effluent release reports shall also include a discussion of any major changes to radioactive waste treatment systems in accordance with Specification 6.17.2.1.

6.9.1.5 <u>Pressurizer Relief and Safety Valve Challenges</u>

Challenges to the pressurizer power operated relief valves or safety valves shall be reported no less frequently than on an annual basis.

6.9.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

Note: This item is intended to provide for reporting of potentially generic problems.

(10) Failure of the pressurizer PORVs or safety valves.

b. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of adminstrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limitations for identified leakage set forth in technical specifications need not be reported under this item.