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. 38
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 September 9, 1980 (transmitted by letter dated September 15, 1980), as supported by filings dated July 21, 1977, November 21, 1977, December 16, 1977, (transmitted by letter dated December 22, 1977), July 31, 1979 (transmitted August 3, 1979 - two separated submittals) and December 19, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 26, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 38

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 a vertical line which indicates the area of change.

REMOVE	INSERT
2.3-4	2.3-4
2.3-5	2.3-5
2.3-6	2.3-6
2.3-7	2.3-7 (reformat only)
2.3-8	2.3-8
2.3-9	2.3-9
--	2.3-10
3.5-4a	3.5-4a
4.1-7	4.1-7

f. Low reactor coolant flow - $\geq$ 90% of normal indicated flow.

g. Low reactor coolant pump frequency - $\geq$ 57.5 Hz.

2.3.1.3 <u>Other reactor trips</u>

a. High pressurizer water level - $\leq$ 92% of span

b. Low-low steam generator water level - $\geq$ 5% of narrow range instrument span

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 Remove bypass of "at power" reactor trips at high power (low pressurizer pressure and low reactor coolant flow) for both loops:

Power range nuclear flux - $\leq$ 8.5% of rated power

(1) (Note: During cold rod drop tests, the pressurizer high level trip may be bypassed.)

2.3.2.2 Remove bypass of single loss of flow trip at high power:

Power range nuclear flux - $\leq$ 50% rated power

2.3.3 Relay operating will be tested to insure that they perform according to their design characteristics which must fall in within the ranges defined below:

2.3.3.1 Loss of voltage relay operating time $\leq$ 8.5 seconds for 480 volt safeguards bus voltages $\leq$ 368 volts. Measured values shall fall at least 5% below the theoretical limit. This 5% margin is shown as the 5% tolerance curve in Figure 2.3-1.

2.3.3.2 Acceptable degraded voltage relay operating times and setpoints, for 480 volt safeguards bus voltages $\leq$ 414 volts and > 368 volts are defined by the safeguard equipment thermal capability curve shown in Figure 2.3-1. Measured values shall fall at least 5% below the theoretical limit. This 5% margin is shown as the 5% tolerance curve in Figure 2.3-1.

Basis:

The high flux reactor trip (low set point) provides redundant protection in the power range for a power excursion beginning from low power. This trip value was used in the safety analysis.(1) In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident.(3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds),[4] and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown in Fgure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[5][6] The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur as described in Section 7.2.3 of the FSAR and in WCAP-8058, "Fuel Densification, R. E. Ginna Nuclear Power Plant Unit 1, Cycle 3" and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[2] The overpower and overtemperature protection set points include consideration of the effects of fuel densification on core safety limits.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or both reactor coolant pumps. The set point specified is consistent with the value used

in the accident analysis.[7] The underfrequency reactor trip protects against a decrease in flow caused by low electrical frequency. The specified set point assures a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 700 $ft.^3$ of water corresponds to 92% of span. The specified set point contains margin for both instrument error and transient overshoot of level beyond this trip setting, and therefore the trip function prevents the water level from reaching the safety valves.[2]

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point equivalent to at least 40,000 lbs. of water assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[8] The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Operation with one pump will not be permitted above 130 MWT (8.5%). An orderly power reduction to less than 130 MWT (8.5%) will be accomplished if a pump is lost while operating between 130 MWT (8.5%) and 50%. Automatic protection is provided so that a power-to-flow ratio is maintained equal to or less than one, which insures that the minimum DNB ratio increases at lower flow

because the maximum enthalpy rise does not increase. For this reason the single pump loss of flow trip can be bypassed below 50% power.

The loss of voltage and degraded voltage trips ensure operability of safeguards equipment during a postulated design basis event concurrent with a degraded bus voltage condition.[9][10][11]

The undervoltage set points have been selected so that safeguards motors will start and accelerate the driven loads (pumps) within the required time and will be able to perform for long periods of time at degraded conditions above the trip set points without significant loss of design life. All control circuitry or safety related control centers and load centers, except for motor control centers M and L, are d.c. Therefore, degraded grid voltages do not affect these control centers and load centers. Motor control centers M and L, which supply the Standby Auxiliary Feedwater System, are fully protected by the undervoltage set points. Further, the Standby System is normally not in service and is manually operated only in total loss of feedwater and auxiliary feedwater.

The 5% tolerance curve in Figure 2.3-1 and the requirements of specifications 2.3.3.1 and 2.3.3.2 include 5% allowance for measurement error. Thus, providing the measurement error is less than 5%, measured values may be directly compared to the curve. If measurement error exceeds 5%, appropriate allowance shall be made.

References:

(1) FSAR 14.1.1

(2) FSAR, Page 14-3

(3) FSAR 14.3.1

(4) FSAR 14.1.2

(5) FSAR 7.2, 7.3

(6) FSAR 3.2.1

(7) FSAR 14.1.6

(8) FSAR 14.1.9

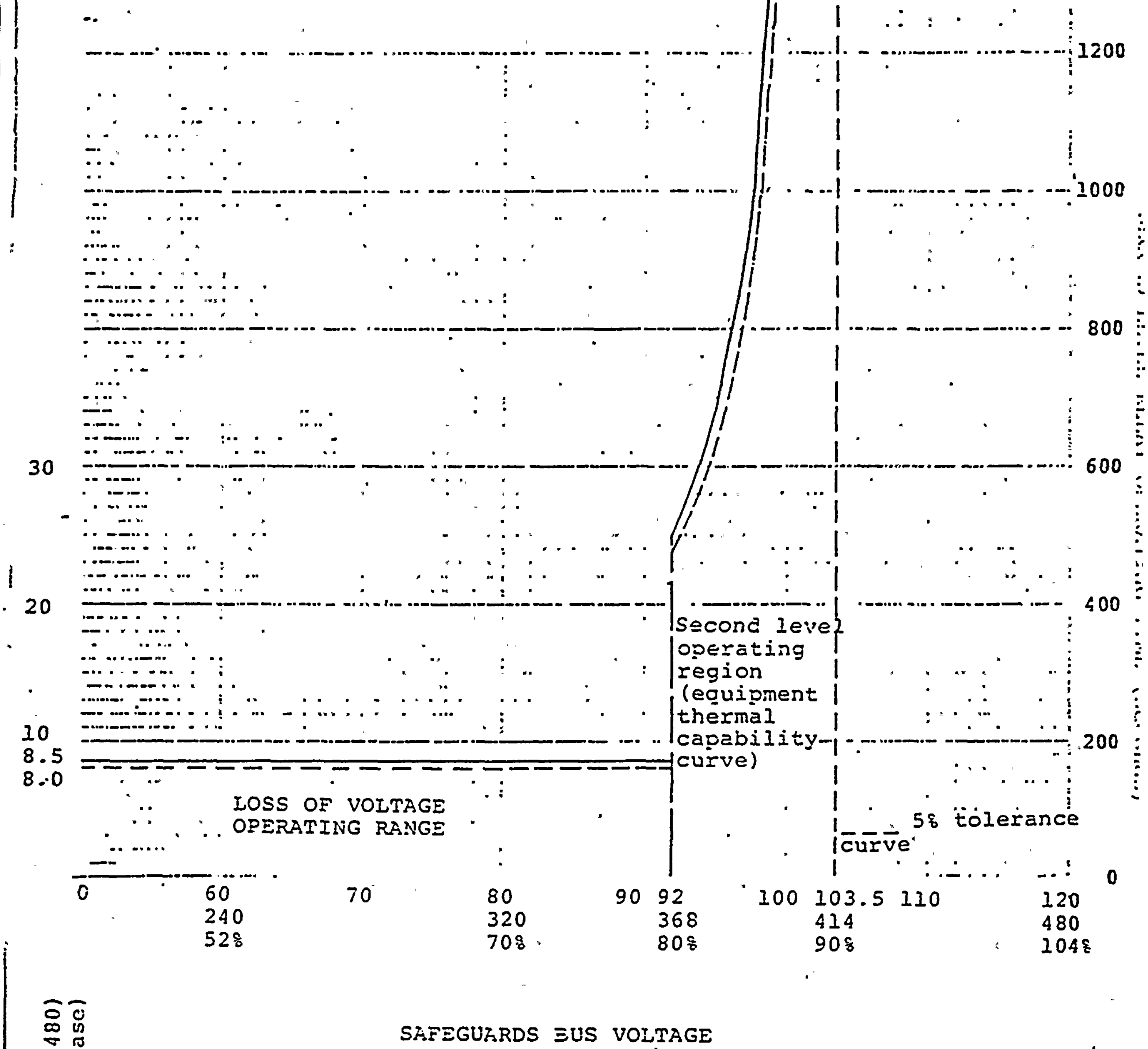
(9) Letter from L. D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977

(10) Letter from L. D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977

(11) Letter from L. D. White, Jr. to D. Ziemann NRC, dated July 24, 1978

LOSS AND SECOND LEVEL (DEGRADED)
UNDERVOLTAGE RELAY OPERATING RANGES

FIG. 2.3-1



	1 NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5. PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 5 CANNOT BE MET
17. Circulating Water Flood Protection						
a. Screenhouse	2	1	2+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs with two channels inoperable.
b. Condenser	2	1	2+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs. with two channels inoperable.
18. Loss of Voltage/ Degraded Voltage 480 Volt Safeguards Bus	4/bus	2/bus	2/bus	*		Maintain hot shutdown or place bus on diesel generator.

NOTE 1: When block condition exists, maintain normal operation.

F.P. = Full Power
* Not Applicable
** If both rod misalignment monitors (a and b) inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of rated power in addition to the increased surveillance noted.
*** If a functional unit is operating with the minimum operable channels, the number of channels to trip the reactor will be column 3 less column 4.
\+ A channel is considered operable with 1 out of 2 logic or 2 out of 3 logic.

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
25.	Containment Pressure	S	R	M	Narrow range containment pressure (-3.0, +3 psig excluded)
26.	Steam Generator Pressure	S	R	M	
27.	Turbine First Stage Pressure	S	R	M	
28.	Emergency Plan Radiation Instruments	M	R	M	
29.	Environmental Monitors	M	N.A.	N.A.	
30.	Loss of Voltage/Degraded Voltage 480 Volt Safeguards Bus	N.A.	R	M	

S - Each Shift M - Monthly

D - Daily P - Prior to each startup if not done previous week

B/W - Biweekly R - Each Refueling Shutdown

Q - Quarterly N.A. - Not applicable