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March 26, 1981

Docket No. 50-244
LS05-81-03-063

Mr. John E. Maier
Vice President
Electric & Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

The Commission has issued the enclosed Amendment No. 38 to Provisional Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your submittals 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 separate submittals), December 19, 1979, and September 9, 1980 (transmitted by letter dated September 15, 1980). These submittals responded to the NRC staff's generic letter dated June 3, 1977. Additional submittals such as references (9), (10), and (11) of the enclosed Technical Specification page 2.3-9 provided further information to assist the staff in its review.

This amendment approves technical specifications related to degraded grid voltage protection for the Class 1E power system.

Copies of our related Safety Evaluation, which incorporates the Technical Evaluation Report on this subject by our NRC consultant, Lawrence Livermore Laboratory, and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

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

Enclosures and cc:
See next page

8105050115

P

E.K.

Enclosures:
1. Amendment No. 38 to
   License No. DPR-18
2. Safety Evaluation, including
   consultant's Technical
   Evaluation Report
3. Notice of Issuance

cc w/enclosures:
See next page

F.R. E.K.

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DATE	3/11/81	3/11/81	~~3/ /81~~	3/14/81	3/16/81	3/16/81	3/26/81

NRC FORM 318 (10/80) NRCM 0240 **OFFICIAL RECORD COPY** ☆ USGPO: 1980—329-824

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

March 26, 1981

Docket No. 50-244
LS05-81-03-063

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Vice President
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Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
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cc w/enclosures:
See next page

Mr. John E. Maier

cc

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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.

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

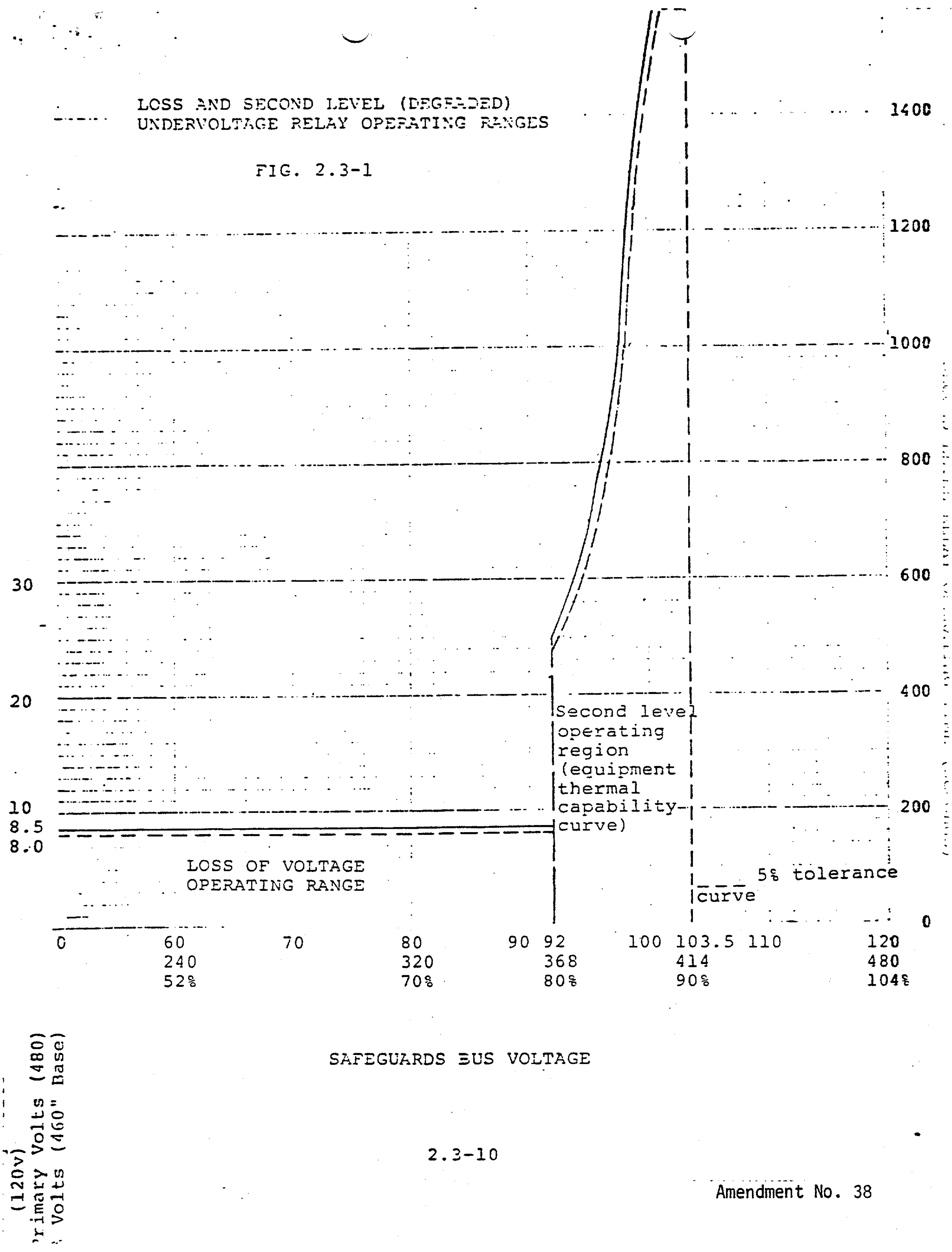
(9) Letter from L. D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977

(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 hr 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
D - Daily
B/W - Biweekly
Q - Quarterly
M - Monthly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
N.A. - Not applicable

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION AND DISCUSSION

The criteria and staff positions pertaining to degraded voltage protection were transmitted to Rochester Gas and Electric Corporation (RG&E) by NRC generic letter dated June 3, 1977. In response to this, by letters dated July 21, 1977, November 21, 1977, December 22, 1977, August 3, 1979, December 19, 1979 and September 9, 1980, the licensee proposed certain design modifications and changes to the Technical Specifications. A detailed review and technical evaluation of these proposed modifications and changes to the Technical Specifications was performed by Lawrence Livermore Laboratory (LLL), under contract to the NRC, and with general supervision by NRC staff. This work is reported by LLL in a report, "Technical Evaluation of the Proposed Design Modifications and Technical Specification Changes for the R. E. Ginna Nuclear Power Plant, Unit 1" (attached). We have reviewed this technical evaluation report and concur in its conclusions that the proposed design modification and Technical Specification changes are acceptable.

2.0 PROPOSED CHANGES

The proposed design changes consist of adding two undervoltage relays for each 480 volt Class 1E bus arranged in a 2-out-of-2 coincident logic. The setpoints including time delays have been selected such that a degraded grid condition will not persist long enough to exceed Class 1E equipment ratings, as defined by the equipment manufacturer. The limiting conditions for operation and surveillance requirements for proposed changes are documented in the licensee's proposed Technical Specification.

3.0 EVALUATION CRITERIA

The criteria used by LLL in its technical evaluation of the proposed changes include GDC-17, "Electric Power Systems," of Appendix A to 10 CFR 50; IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE Standard 308-1974, "Class 1E Power Systems for Nuclear Power Generating Stations," and the staff positions defined in NRC generic letter to RG&E dated June 3, 1977.

4.0 SUMMARY

We have reviewed the Lawrence Livermore Laboratory's Technical Evaluation Report and concur in its findings that (1) the proposed modifications will protect the Class 1E equipment and systems from a sustained degraded voltage of the offsite power source, and (2) the proposed changes to the Technical Specifications meet the criteria for periodic testing of protection systems and equipment. Therefore, we conclude that the proposed design modifications and changes to the Technical Specifications are acceptable.

5.0 ENVIRONMENTAL CONSIDERATIONS

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.

6.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the 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: March 26, 1981

Attached:
Report by Lawrence Livermore
Laboratory (UCID-18691)

UCID- 18691

TECHNICAL EVALUATION OF THE PROPOSED DESIGN MODIFICATIONS AND TECHNICAL SPECIFICATION CHANGES FOR THE R. E. GINNA NUCLEAR POWER PLANT, UNIT 1

James C. Selan

Lawrence Livermore Laboratory

This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

ABSTRACT

This report documents the technical evaluation of the proposed design modifications and technical specification changes on grid voltage degradation for the R. E. Ginna Nuclear Power Plant Unit 1. The review criteria are based on IEEE Std. 279-1971, IEEE Std. 308-1974, and General Design Criteria 17 of the Code of Federal Regulations, Title 10, part 50, Appendix A requirements for determining the acceptability of the proposed system to protect the Class 1E equipment from degradation of grid voltages.

FOREWORD

This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues (SEICSI) Program being conducted for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Operating Reactors, by Lawrence Livermore National Laboratory, Engineering Research Division of the Electronics Engineering Department.

The U. S. Nuclear Regulatory Commission funded the work under the authorization entitled "Electrical, Instrumentation and Control System Support," B&R 20 19 04 031, FIN A-0231.

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AND TECHNICAL SPECIFICATIONS
ON GRID VOLTAGE DEGRADATION
FOR THE
R. E. GINNA NUCLEAR POWER PLANT
UNIT 1

(Docket No. 50-244)

Lawrence Livermore National Laboratory, Nevada

1. INTRODUCTION

By letter dated June 3, 1977 [Ref. 1], the U. S. Nuclear Regulatory Commission (NRC) requested the Rochester Gas and Electric Corporation (RG&E) to assess the susceptibility of the R. E. Ginna Nuclear Power Plant, Unit 1, Class 1E electrical equipment to sustained degraded voltage conditions at off-site power sources and to the interaction between the offsite and onsite emergency power systems. In addition, the NRC requested that the licensee compare the current design of the emergency power systems at the plant facilities with the NRC staff positions as stated in the June 3, 1977 letter [Ref. 1], and that the licensee propose plant modifications, as necessary, to meet the NRC staff positions, or provide a detailed analysis which shows that the facility design has equivalent capabilities and protective features. Further, the NRC required that certain Technical Specifications be incorporated into all facility operating licenses.

In letters dated July 21, 1977 [Ref. 2], November 21, 1977 [Ref. 3], December 22, 1977 [Ref. 4], August 3, 1979 [Ref. 5], December 19, 1979 [Ref. 6], and September 9, 1980 [Ref. 7], RG&E proposed certain design modifications and additions to the licensee's Technical Specifications. These design modifications include the installation of a degraded voltage protection system for the Class 1E equipment. The proposed additions to the Technical Specifications are in regard to the setpoints, calibrations, and surveillance requirements associated with the proposed voltage protection system.

The purpose of this report is to evaluate the licensee's proposed design modifications and Technical Specification changes and to determine that they meet the criteria established by the NRC for the protection of Class 1E equipment from grid voltage degradation.

2. DESIGN BASIS CRITERIA

The design basis criteria that were applied in determining the acceptability of the system modification to protect the Class 1E equipment from degradation of grid voltages are as follows:

(1) General Design Criterion 17 (GDC 17), "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," in the Code of Federal Regulations, Title 10, Part 50 (10 CFR 50) [Ref. 8].

(2) IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" [Ref. 9].

(3) IEEE Std. 308-1974, "Class 1E Power Systems for Nuclear Power Generating Stations" [Ref. 10].

(4) NRC staff positions as stated in a letter dated June 3, 1977 [Ref. 1].

3. EVALUATION

3.1 EXISTING UNDERVOLTAGE PROTECTION

The present design uses undervoltage relays to sense the loss of offsite power. There are no Class 1E loads on the 4160-volt buses. This design consists of two relays per 480-volt Class 1E bus (two Class 1E buses per redundant load group) for the first level of undervoltage protection. An undervoltage condition (loss-of-voltage) will result in isolating the Class 1E buses from all offsite sources, initiating emergency diesel generator start and load shedding on the Class 1E buses, permitting closure of the diesel generator supply breakers, and lastly, the loads will be automatically time-sequenced onto the buses. Actuation begins with loss of voltage to 368 volts (77% of 480 volts). The existing system does not bypass the load-shedding feature once the emergency diesel generators are energizing the Class 1E buses.

3.2 MODIFICATIONS

The licensee has proposed a design change which includes automatic degraded voltage protection. This modification consists of the addition of two time-delayed, undervoltage relays on each 480-volt Class 1E bus, to provide the second level of undervoltage protection. After a preset time delay, according to the relay-tripping characteristics defined in Figure 1, the second-level protection scheme will automatically monitor and initiate both the offsite source disconnection and the onsite source connection when the voltage degrades below the safe operating voltage level. The limiting conditions for operation and surveillance requirements for the proposed design changes presented in this evaluation are documented in the licensee's proposed Technical Specifications.

3.3 DISCUSSION

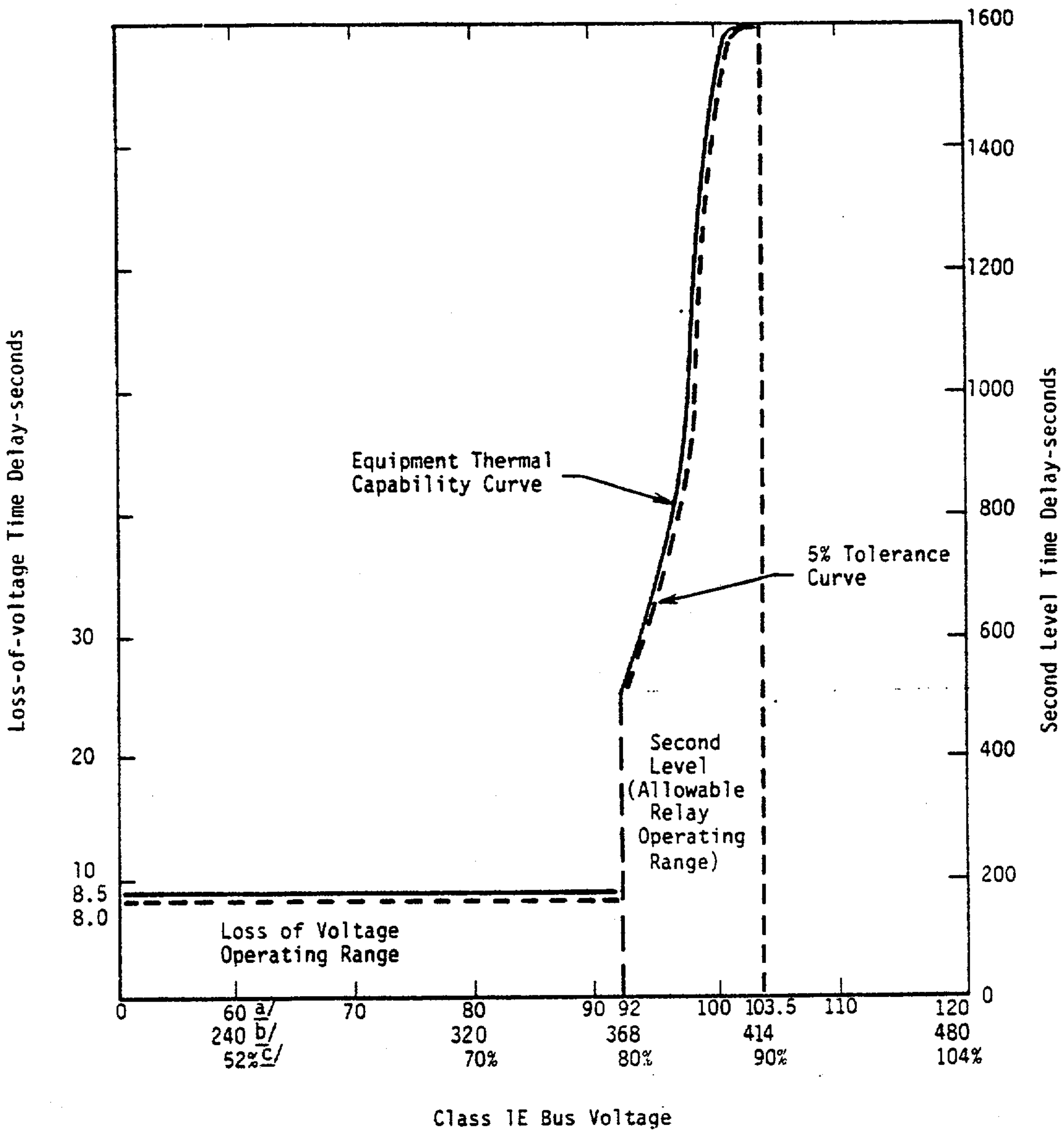
This section presents a statement on the NRC staff positions from their June 3, 1977 letter [Ref. 1] followed by an evaluation of the licensee's design.

3.3.1 NRC Staff Position 1: Second Level of Undervoltage or Overvoltage Protection with a Time Delay.

This position is to be met by the licensee meeting certain criteria. Each criterion has been evaluated against the licensee's proposal and is addressed below.

(1) "The selection of voltage and time setpoints shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels."

The licensee's second level of undervoltage protection setpoints (voltage and time delay) are defined by the curves shown in Figure 1. The solid-line curve defines the maximum time (determined by equipment manufacturers) that the Class 1E equipment can operate for a specific degraded voltage without causing equipment damage, loss of equipment life, or affecting the ability of the equipment to perform a required function. The maximum time delay at the 414-volt setpoint is 1600 seconds. The dotted-line curve (5% tolerance band) defines the maximum allowable time delay before protective relaying actuation is initiated. This tolerance band was determined by all the accuracies of the relay test instrumentation. The relays will be tested to insure that they perform according to their design operating characteristics, which must fall within the area defined by the dotted-line curve in Figure 1.



a/ Secondary volts (120 volts)
b/ Primary volts (480 volts)
c/ Percent volts (460 volt base)

NOTE: This figure has been reproduced from the information in Figure 2.3-1 in the Proposed Technical Specifications (Ref. 7) for the R.E. Ginna Nuclear Power Station, Unit 1.

Figure 1. Loss-of-voltage and second-level undervoltage relay operating ranges.

Any deviation outside the limits defined in Figure 1 will result in recalibration of the relay. The licensee's analysis has been reviewed and shows that this protection range is satisfactory for the 480-volt Class 1E loads and other components whose functional performance would be inadequate because of undervoltage.

(2) "The voltage protection shall include coincidence logic to preclude spurious trips of the offsite power sources."

The second-level protection scheme will be designed using two-out-of-two coincident logic. The integration of the second-level protection system into the first level of undervoltage protection can be seen in Figure 2. If a loss-of-voltage condition exists, relays 27 and 27B will drop out and initiate the automatic voltage-restoring scheme. For a degraded voltage condition, relays 27SL and 27BSL will drop out after the time-delay setpoints are exceeded and will initiate the automatic voltage-restoring scheme.

(3) "The time delay selected shall be based on the following conditions."

(a) "The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the FSAR accident analysis."

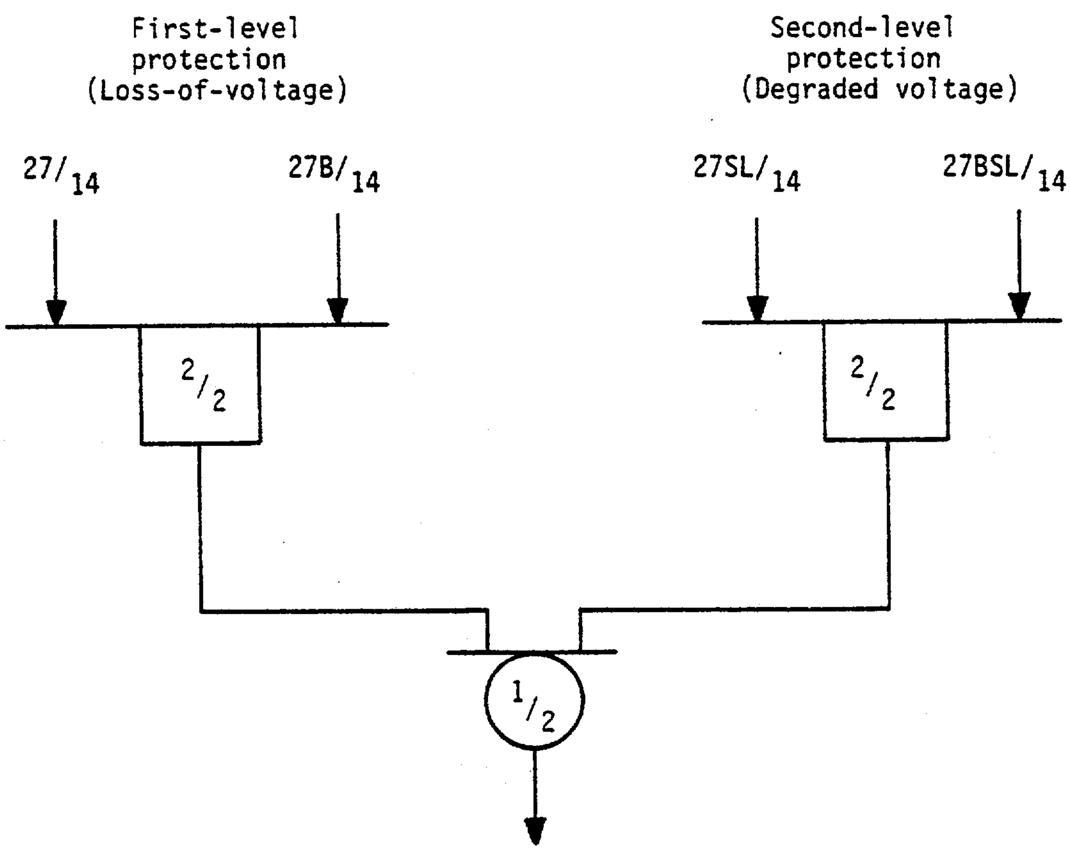
The second-level undervoltage relay setpoints defined in Figure 1 are such that the relay operating characteristics will protect the Class 1E equipment from sustained degraded voltage and also insure that all Class 1E motors will start successfully and be loaded onto the diesel generator within the time assumed in the FSAR accident analysis.

(b) "The time delay shall minimize the effect of short-duration disturbances from reducing the unavailability of the offsite power sources."

The licensee's proposed time delay defined in Figure 1 is long enough to override any short grid disturbances.

(c) "The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components."

A review of the licensee's voltage analysis indicates that the time delay will not cause any failure of any equipment connected to and associated with the Class 1E emergency power system.



Trip Logic and Voltage Restoring Scheme

NOTE: This logic diagram also applies to buses 16, 17, and 18.

Figure 2. Coincident trip logic for bus 14.

(4) "The undervoltage monitors shall automatically initiate the disconnecton of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded."

The two-out-of-two coincidence logic automatically disconnects offsite power from the Class 1E buses experiencing degraded voltage and initiates the voltage-restoring scheme.

(5) "The voltage monitors shall be designed to satisfy the requirements of IEEE Standard 279-1971."

The licensee states that the relays and relaying scheme is in compliance with IEEE Std. 308-1974, and my review confirms that the requirements of IEEE 279-1971 are met.

(6) "The Technical Specifications shall include limiting conditions for operations, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second-level voltage protection monitors."

Limiting conditions for operation and surveillance requirements, as well as trip setpoints for allowable values for degraded voltage protection, are included in the licensee's proposed Technical Specifications.

3.3.2 NRC Staff Position 2: Interaction of Onsite Power Sources with Load-Shed Feature.

The second position requires that the system be designed to prevent load shedding of the emergency buses automatically once the onsite sources are supplying power to all sequenced loads. If an adequate basis can be provided for retaining the load-shed feature, the licensee must assign maximum and minimum values to the setpoint of the load-shed feature. These setpoints must be documented in the Technical Specifications. The load-shedding feature must be reinstated if the onsite source supply breakers are tripped.

The licensee is retaining the load-shed feature once the emergency buses are being supplied by the onsite sources on the basis that the load-shed feature is to protect the Class 1E equipment from unsatisfactory bus voltages. To meet the requirements of NRC Staff Position 2, the licensee has proposed in the Technical Specifications the maximum setpoint values of 368 volts and 8.5 seconds to the loss-of-voltage (load-shed feature) relay. These maximum limits on the voltage and time-delay values of the load-shed feature will ensure that relay operating drift will not cause spurious trips of the onsite source while the Class 1E loads are being sequenced onto the buses.

3.3.3 NRC Staff Position 3: Onsite Power Source Testing

The third position requires that certain test requirements be included in the Technical Specifications. These tests are to "...demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown." The tests are to simulate loss of offsite power in conjunction with a safety injection actuation signal and to simulate interruption and subsequent reconnection of onsite power sources. These tests will verify the proper operation of the load-shed system, the load-shed bypass when the emergency diesel generators are supplying their respective buses, and that there is no adverse interaction between the onsite and offsite power sources.

The licensee will satisfy the requirements of the NRC by testing the system by initiating loss of offsite power in conjunction with a simulated safety injection signal. The tests sequence will be bus de-energization, load shedding, voltage restoration, and load sequencing. The operating time with full load on emergency onsite power will be at least five minutes.

3.4 TECHNICAL SPECIFICATIONS

The changes proposed by Rochester Gas and Electric Corporation to the R. E. Ginna Unit 1 Nuclear Power Plant Technical Specifications reflect the proposed design modifications. Specifically, the proposed changes:

(1) Include the trip setpoints for the degraded voltage protection sensors and the associated time delays (Figure 1).

(2) Provide the required coincidence logic (two-out-of-two).

(3) Incorporate action statements regarding limiting conditions for operations when the number of operable channels for degraded voltage protection is reduced.

(4) Provide the surveillance requirements for channel calibration during refueling shutdown and the monthly channel functional test.

(5) Provide surveillance requirements to demonstrate at least once per 18 months that the loss of offsite power in conjunction with a safety injection actuation signal will provide the sequence of Class 1E bus de-energization, load shedding, voltage restoration, and load sequencing.

4. CONCLUSION

Based on the information provided by Rochester Gas and Electric Corporation, it has been determined that the proposed modifications comply with NRC Staff Position 1. All of the staff's requirements and design basis criteria have been met. The voltage setting and time delays will protect the Class 1E equipment from a sustained degraded voltage condition of the offsite power source.

The licensee is retaining the load-shed feature while the onsite sources are supplying the Class 1E buses, and has submitted in the proposed Technical Specifications the maximum limits to the setpoint values of the loss-ofvoltage (load-shed feature) relay. A review of the setpoint values, limits, and logic circuitry has determined that there will be no adverse interaction of the onsite sources with the load-shed feature during load sequencing, thus the requirements of NRC Staff Position 2 are met.

The proposed additions to the Technical Specifications and the method of testing the logic circuitry have been reviewed and found to meet the requirements of NRC Staff Position 3.

Accordingly, I recommend that the NRC approve the proposed design modifications and proposed Technical Specifications changes.

REFERENCES

1. NRC letter (A. Schwencer) to RG&E (L. D. White, Jr.), dated June 3, 1977.

2. RG&E letter (L. D. White, Jr.) to NRC (A. Schwencer), dated July 21, 1977.

3. RG&E letter (LeBoeuf, Lamb, Leiby and Macrae) to NRC (E. Case), dated November 21, 1977.

4. RG&E letter (LeBoeuf, Lamb, Leiby and Macrae) to NRC (E. Case), dated December 22, 1977.

5. RG&E letter (LeBoeuf, Lamb, Leiby and Macrae) to NRC (H. Denton), dated August 3, 1979.

6. RG&E telecopy (G. Daniels) to NRC (J. Shea), dated December 19, 1979.

7. RG&E letter (L. D. White, Jr.) to NRC, dated September 9, 1980.

8. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), General Design Criterion 17 (GDC 17), "Electric Power Systems" of Appendix A, "General Design Criteria for Nuclear Power Plants."

9. IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

10. IEEE Std. 308-1978, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

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. 38 to Provisional Operating License No. DPR-18, to Rochester Gas and Electric Corporation (the licensee), which revised the technical specifications for operation of the R.E. Ginna Plant (facility) located in Wayne County, New York. This amendment is effective as of the date of its issuance.

The amendment incorporates limiting conditions for operation and surveillance requirements regarding degraded grid voltage protection for the Class 1E power system.

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 September 9, 1980 (transmitted by letter dated September 15, 1980), as preceded and supported by submittals 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 separate submittals), and December 19, 1979, (2) Amendment No. 38 to License No. DPR-18, and (3) 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 (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 26th day of March, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

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