April 17, 1981

Docket No. 50-244
LS05-81-04-025

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

Dear Mr. Maier:

SUBJECT: CONTROL ROD POSITION INDICATION

The Commission has issued the enclosed Amendment No. 40 to Provisional Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment responds to your application notarized August 29, 1980 (submitted by letter dated September 3, 1980). Your application is in response to our letter dated November 5, 1979.

The amendment authorizes technical specifications regarding control rod position indication and control rod misalignment.

Changes have been made to your submittal as mutually agreed upon during telephone conversations with your staff during the weeks of March 9 and March 16, 1981. Also, several technical specification pages have been modified in format only, as discussed with and agreed upon by your staff.

Copies of our Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Dennis M. Crutchfield

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

RECEIVED APR 27 1981 U.S. NUCLEAR REGULATORY COMMISSION

Enclosures:
1. Amendment No. 40 to License No. DPR-18
2. Safety Evaluation
3. Notice

P

8104300525

LIC or Amendment + F.R. notice

cc w/enclosures:
See next page

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| OFFICE | ORB#5:DL | ORB#5:DL | ORB#3:DL | OELD | C/ORB#5:DL | AD-SA:DL | |
| SURNAME | RSnaider:dr | HSmith | PWagner | KETCHEN | DCrutchfield | GCLainas | |
| DATE | 4/2/81 | 3/31/81 | 4/2/81 | 4/7/81 | 4/16/81 | 4/17/81 | |

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

April 17, 1981

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LS05-81-04-025

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Vice President
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   License No. DPR-18
2. Safety Evaluation
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cc w/enclosures:
See next page

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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. 40
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Company (the licensee) dated August 29, 1980 (transmitted by letter dated September 3, 1980), 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 by changing paragraph 2.C(2) of Provisional Operating License No. DPR-18 to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 40 , 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 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: April 17, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 40

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 |
|---|---|
| 3.5-4 | 3.5-4 |
| 3.5-4a | 3.5-4a |
| 3.10-3 | 3.10-3 |
| 3.10-5 | 3.10-5 |
| 3.10-6 | 3.10-6 |
| 3.10-7 | 3.10-7 |
| 3.10-8 | 3.10-8* |
| 3.10-8a | 3.10-8a |
| 3.10-8b | 3.10-8b* |
| 3.10-8c | 3.10-8c* |
| 3.10-9 | 3.10-9 |
| 3.10-10 | 3.10-10 |
| Fig. 3.10-1 | 3.10-11 |
| Fig. 3.10-2 | 3.10-12 |
| Fig. 3.10-3 | 3.10-13 |
| --- | 3.10-14 |
| 4.1-5 | 4.1-5 |
| 4.1-6 | 4.1-6 |
| 4.1-8 | 4.1-8 |
| 4.1-9 | 4.1-9* |
| 4.1-10 | 4.1-10 |

*These pages are included for pagination purposes only.

| NO. | FUNCTIONAL UNIT | 1<br>NO. OF CHANNELS | 2<br>NO. OF CHANNELS TO TRIP | 3<br>MIN. OPERABLE CHANNELS | 4<br>MIN. DEGREE OF REDUNDANCY | 5<br>PERMISSABLE BYPASS CONDITIONS | 6<br>OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 5 CANNOT BE MET |
|---|---|---|---|---|---|---|---|
| 11. | Turbine Trip | 3 | 2 | 2 | 1 | | Maintain 50% of rated power |
| 12. | Steam Flow Feedwater flow mismatch with Lo Steam Generator Level | 2/loop | 1/loop | 1/loop | 1/loop | | Maintain hot shutdown |
| 13. | Lo Lo Steam Generator Water Level | 3/loop | 2/loop | 2/loop | 1/loop | | Maintain hot shutdown |
| 14. | Undervoltage 4 KV Bus | 2/bus | 1/bus | 1/bus | * | | Maintain hot shutdown |
| 15. | Underfrequency 4 KV Bus | 2/bus | 1/bus (both busses) | 1/bus | * | | Maintain hot shutdown |
| 16. | Quadrant power tilt monitor (upper & lower ex-core neutron detectors) | 1 | * | 1 or Log individual upper & lower ion chamber currents once/hr & after a load change of 10% or after 30" of control rod motion | * | | Maintain hot shutdown |

| NO. | FUNCTIONAL UNIT | 1<br>NO. OF CHANNELS | 2<br>NO. OF CHANNELS TO TRIP | 3<br>MIN. OPERABLE CHANNELS | 4<br>MIN. DEGREE OF REDUNDANCY | 5<br>PERMISSABLE BYPASS CONDITIONS | 6<br>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. |

NOTE 1: When block condition exists, maintain normal operation

F.P. = Full Power
* Not Applicable
*** 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.

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 $P \leq .5$

$F_{\Delta H}^{N} = 2.22 - .56P$ for $P \geq .75$

$F_{\Delta H}^{N} = 1.80$ for $P \leq .75$

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 $F_Q^N$ 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 which $F_{\Delta H}^N$ 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 $\Delta T$ trip setpoint and the Overtemperature $\Delta T$ trip setpoint shall be similarly reduced.

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio, exceeds1.02 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.

3.10.3 Control Rod Drop Time

3.10.3.1 While critical, the individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 1.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg}$ greater than or equal to 540°F, and

b. All reactor coolant pumps operating.

3.10.3.2 With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to criticality.

3.10.4 Control Rod Group Height

3.10.4.1 While critical, and except for physics testing, all full length (shutdown and control) rods shall be operable and positioned within ± 12 steps (indicated position) of their group step counter demand position.

3.10.4.2 With any full length rod inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untripable, determine that the shutdown margin requirement of Specification 3.10.1.1 is satisfied within 1 hour and be in hot shutdown within 6 hours.

3.10.4.3 With one full length rod inoperable due to causes other than addressed by 3.10.4.2, above, or misaligned from its group step counter demand position by more than ± 12 steps (indicated position), operation may continue provided that within one hour either:

3.10.4.3.1 The rod is restored to operable status within the above alignment requirements, or

3.10.4.3.2 The rod is declared inoperable and the shutdown margin requirement of Specification 3.10.1.1 is satisfied. Operations may then continue provided either:

a. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour, while maintaining the limit of Specification 3.10.1.3; or

b. The power level is reduced to less than or equal to 75% of rated power within the next one hour, and the high neutron flux trip setpoint is reduced to less than or equal to 85% rated power within the next four hours (total of six hours) and the following evaluations are performed:

(i) The shutdown margin requirement of Specification 3.10.1.1 is determined at least once per 12 hours.

(ii) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours.

(iii) A reevaluation of each accident analysis of Table 3.10-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

c. If power has been restricted in accordance with (b) above, then following completion of the evaluation identified in (b), the power level and high neutron flux trip setpoint may be readjusted based on the results of the evaluation provided the shutdown margin requirement of Specification 3.10.1.1 is determined at least once per 12 hours.

3.10.4.4 With two or more full length rods inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in hot shutdown within 6 hours.

3.10.5 <u>Control Rod Position Indication Systems</u>

3.10.5.1 While critical, the analog rod position indication system and the step counters shall be operable and capable of determining the control rod positions within ± 12 steps.

3.10.5.2 With a maximum of one analog rod position indicator per bank inoperable either:

a. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

b. Reduce the power to less than 50% of rated power within 8 hours.

3.10.5.3 With a maximum of one step counter per bank inoperable either:

a. Verify that all analog rod position indicators for the affected bank are operable and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

b. Reduce the power to less than 50% of rated power within 8 hours.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis.[1] In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors.

The lines shown on Figure 3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end-of-cycle life and is based on the value used in analysis of the hypothetical steam break accident. Early in cycle life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to that which is required at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-3 has been determined from extensive analyses considering operating maneuvers consistent with the Technical Specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope demonstrate compliance with the Final Acceptance Criteria limit for Emergency Core Cooling Systems.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading pattern. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomolies which might, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as described in Specification 3.10.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution limits which are given in terms of flux difference limits and control bank insertion limits are observed. Flux difference is $q_T - q_B$ as defined in Specification 2.3.1.2d.

The permitted relaxation in $F^N_{\Delta H}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In specification 3.10 $F_Q$ is arbitrarily limited for $P < 0.5$ (except for low power physics tests).

The limits on axial power distribution referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = $\Delta I$/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies primarily with burnup.

The technical specifications on power distribution assure that the $F_Q$ upper bound envelope of 2.32 times Figure 3.10-3 is not exceeded and xenon distributions are not developed which, at a later time, could cause greater local power peaking even though the flux difference is then within the limits.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with control Bank D more than 190 steps withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 percent ΔI is permitted from the indicated reference value. During periods where extensive load following is required, it may be impossible to establish the required core conditions for measuring the target flux difference every month. For this reason, two methods are permissible for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly differen_ from those resulting from _eration within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent of each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\pm$ 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainity in $F_q$ is depleted. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in $F_Q^N$ and $F_{\Delta H}^N$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction on peaking factors.

The two hours in 3.10.2.3 are acceptable since complete rod misalignment (full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident.

If instead of determining the hot channel factors, the operator decides to reduce power, the specified 75% power maintains the design margin to core safety limits for up to a 1.12 power tilt, using the 2 to 1 ratio. Reducing the overpower trip set point ensures that the protection system basis is maintained for sustained plant operation. A tilt ratio of 1.12 or more is indicative of a serious performance anomaly and a plant shutdown is prudent.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with $T_{avg}$ greater than or equal to 540°F and with both reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The various control rod banks (shutdown banks, control banks A, B, C, and D are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position.(2) These are known as the step counters and analog rod position indication, respectively.

Operability of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The 12 step (7.5 inches) permissible indicated misalignment ensures that the 15 inch misalignment assumed in the safety analysis is met.

The action statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in power; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

References:

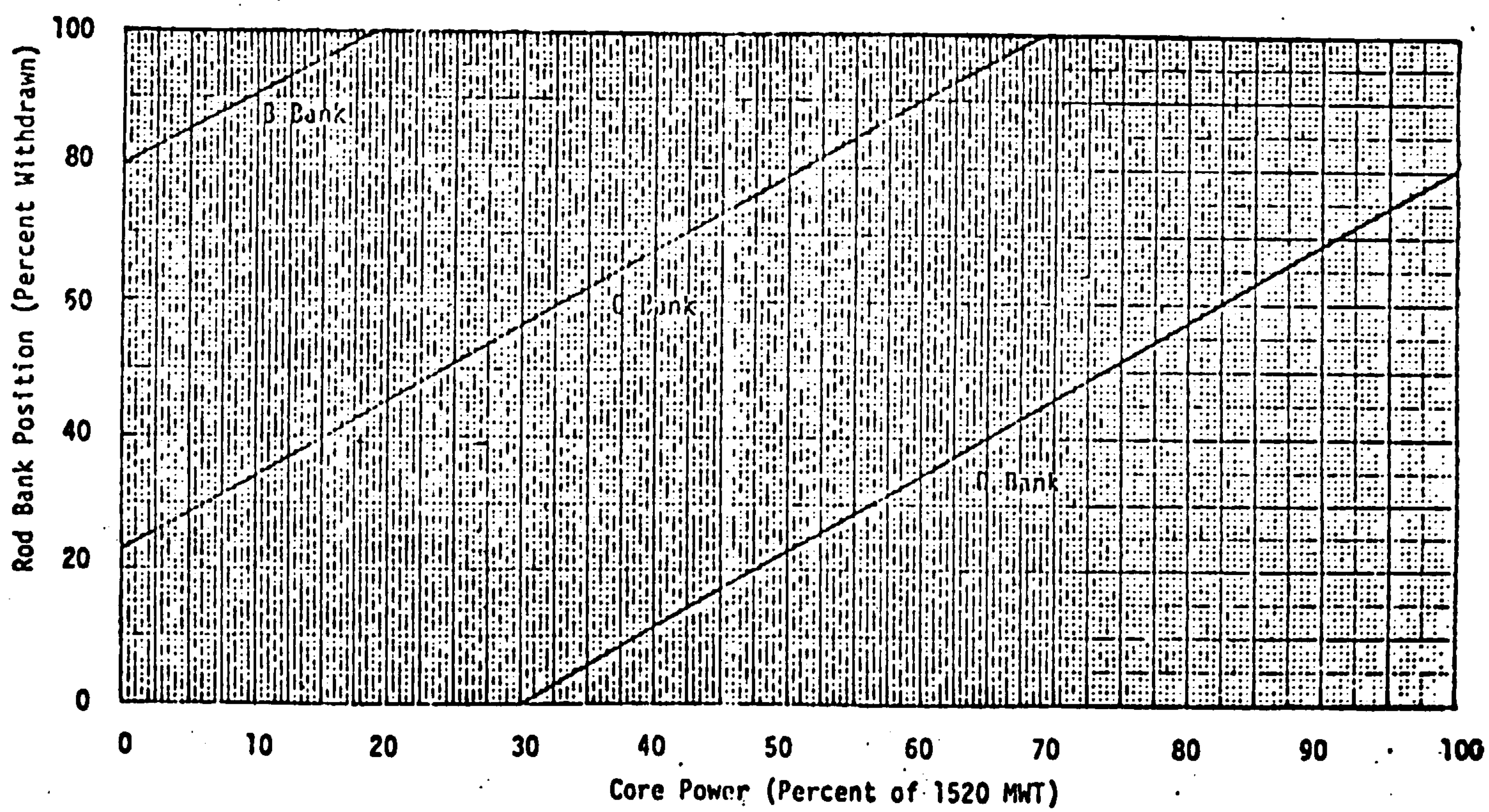
(1) Technical Supplement Accompanying Application to Increase Power - Section 14

(2) FSAR, Section 7.3

**FIGURE 3.10-1**

**CONTROL ROD INSERTION LIMITS VERSUS CORE POWER**

**FOR BOL THROUGH EOL**



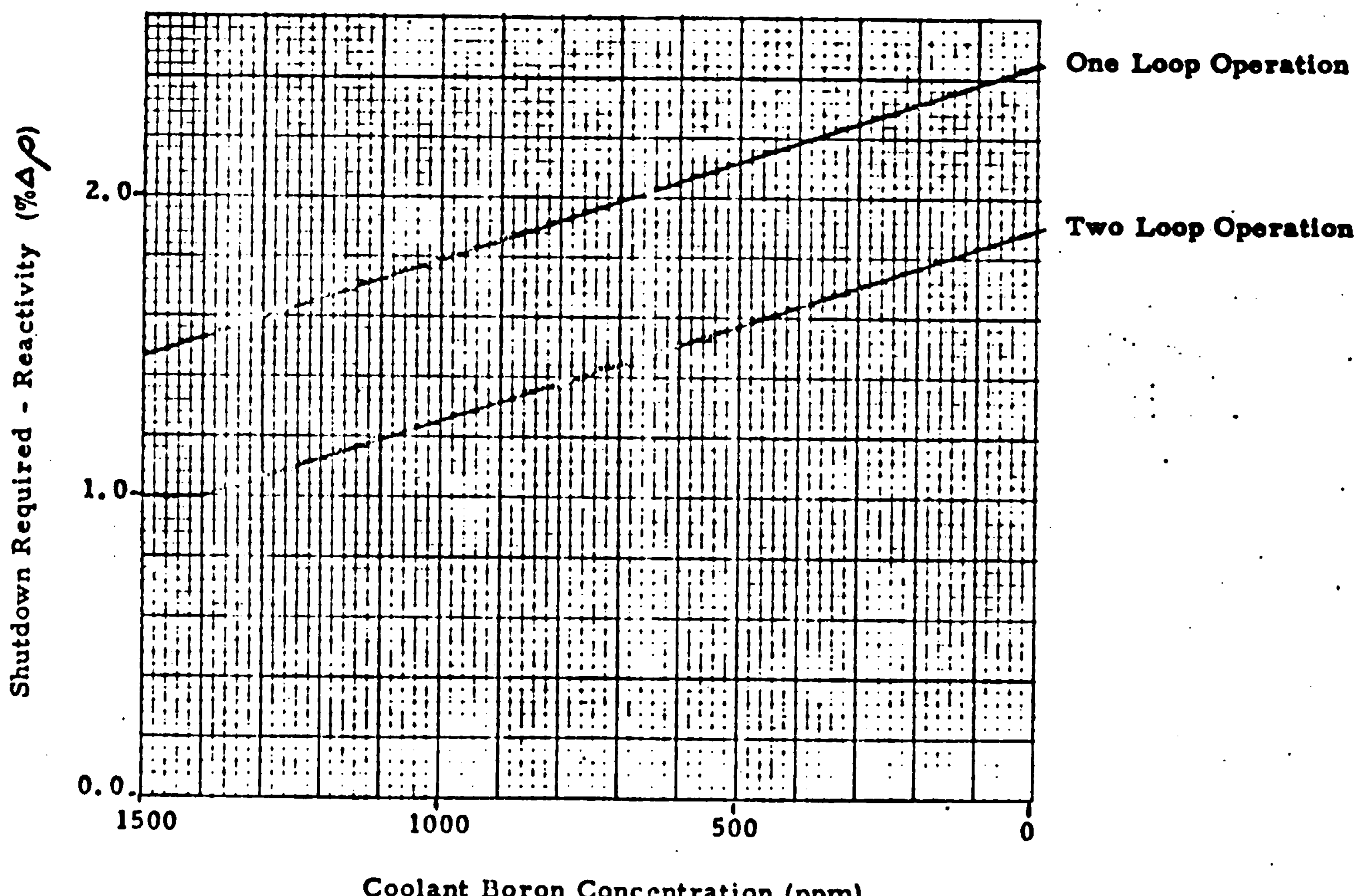



Figure 3.10-2
Required Shutdown Margin

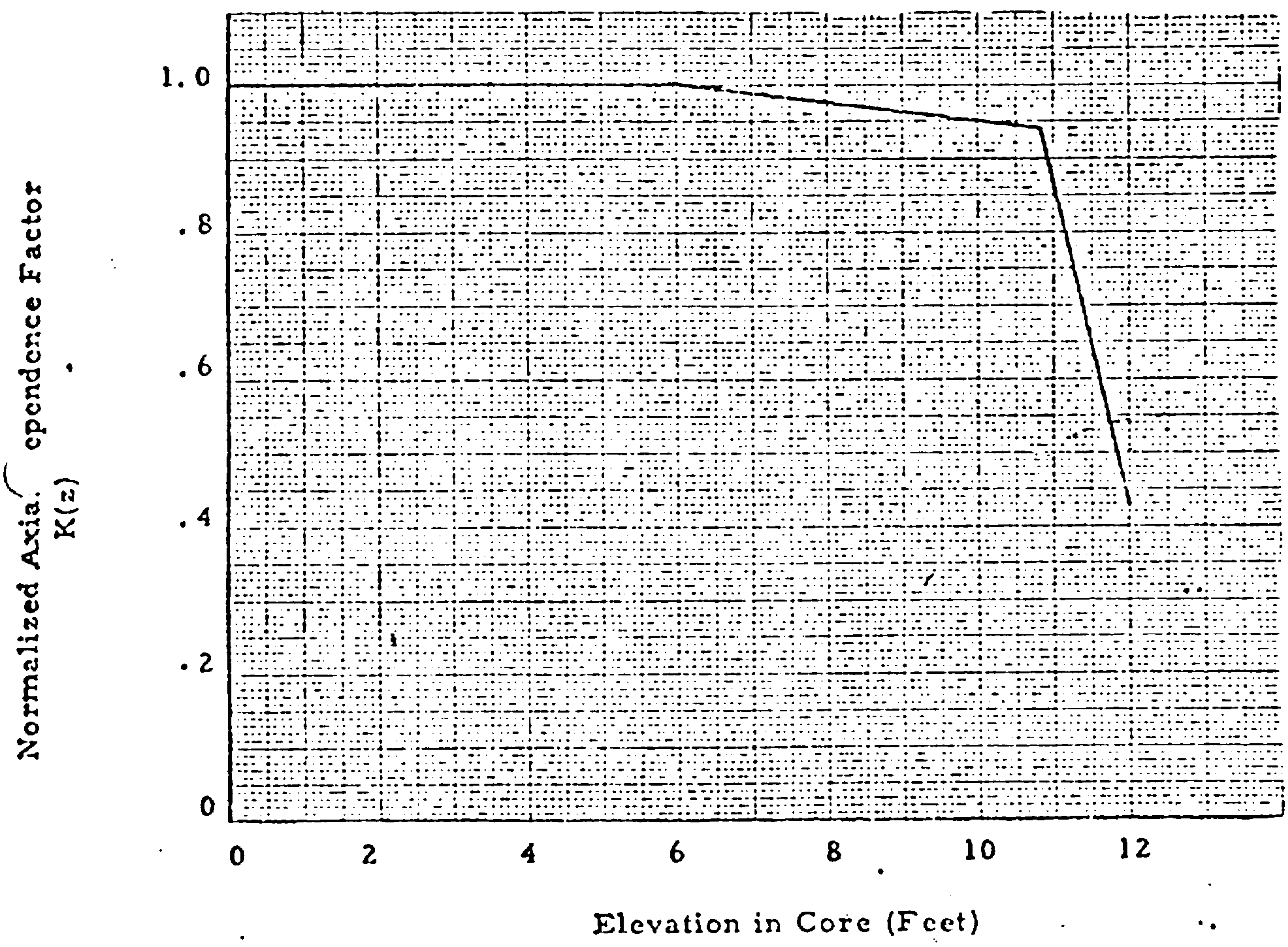



Figure 3.10-3 Normalized Axial Dependence Factor for $F_Q$ versus Elevation

Amendment No. ~~7~~,40

## Table 3.10-1

### ACCIDENT ANALYSIS REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE CONTROL ROD

Rod Insertion Characteristics

Rod Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Rod Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Steam Line Break

Rod Ejection

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

| Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|
| 1. Nuclear Power Range | S<br>M*(3) | D(1)<br>Q*(3) | B/W(2)(4)<br>P(2)(5) | 1) Heat balance calculation**<br>2) Signal to ΔT; bistable action (permissive, rod stop, trips)<br>3) Upper & lower chambers for axial offset**<br>4) High setpoint ($\leq$ 109% of rated power)<br>5) Low setpoint ($\leq$ 25% of rated power) |
| 2. Nuclear Intermediate Range | S(1) | N.A. | P(2) | 1) Once/shift when in service<br>2) Log level; bistable action (permissive, rod stop, trip) |
| 3. Nuclear Source Range | S(1) | N.A. | P(2) | 1) Once/shift when in service<br>2) Bistable action (alarm, trip) |
| 4. Reactor Coolant Temperature | S | R | M(1) | 1) Overtemperature-Delta T<br>2) Overpower - Delta T |
| 5. Reactor Coolant Flow | S | R | M | |
| 6. Pressurizer Water Level | S | R | M | |
| 7. Pressurizer Pressure | S | R | M | |
| 8. 4 Kv Voltage & Frequency | N.A | R | M | Reactor Protection circuits only |
| 9. Analog Rod Position | S(1,2) | R | M | 1) With step counters<br>2) Log analog rod positions each 4 hours when rod deviation monitor is out of service |

* By means of the movable in-core detector system.

** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

TABLE 4.1-1 (CONTINUED)

| Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|
| 10. Rod Position Bank Counters | S(1,2) | N.A. | N.A. | 1) With analog rod position 2) Log analog rod positions each 4 hours when rod deviation monitor is out of service |
| 11. Steam Generator Level | S | R | M | |
| 12. Charging Flow | N.A. | R | N.A. | |
| 13. Residual Heat Removal Pump Flow | N.A. | R | N.A. | |
| 14. Boric Acid Tank Level | D | R | N.A. | Bubbler tube rodded weekly |
| 15. Refueling Water Storage Tank Level | N.A. | R | N.A. | |
| 16. Volume Control Tank Level | N.A. | R | N.A. | |
| 17. Reactor Containment Pressure | D | R | M(1) | 1) Isolation Valve signal |
| 18. Radiation Monitoring System | D | R | M | |
| 19. Boric Acid Control | N.A. | R | N.A. | |
| 20. Containment Drain Sump Level | N.A. | R | N.A. | |
| 21. Valve Temperature Interlocks | N.A. | N.A. | R | |
| 22. Pump-Valve Interlock | R | N.A. | N.A. | |
| 23. Turbine Trip Set-Point | N.A. | R | M(1) | 1) Block trip |
| 24. Accumulator Level and Pressure | S | R | N.A. | |

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

| | Test | Frequency | FSAR Section Reference |
|---|---|---|---|
| 1. Reactor Coolant Samples | Gross Radioactivity Concentration (beta-gamma) | 3 times/weekly and at least every third day (1) (7) | |
| | Radio-chemical (2)(4) | Monthly (6) | |
| | E Determination (2) | Monthly (6) | |
| | Tritium Concentration | Weekly (6) | |
| | Chloride and Fluoride | 3 times/week and at least every third day | |
| | Oxygen | 5 times/week and at least every second day except when below 250°F | |
| | Gross Radioiodine Concentration | Weekly (3) (6) | |
| 2. Reactor Coolant Boron | Boron concentration | Weekly | |
| 3. Refueling Water Storage Tank Water Sample | Boron concentration | Weekly | |
| 4. Boric Acid Tank | Boron concentration | Twice/week | |
| 5. Control Rods | Rod drop times of all full length rods | After vessel head removal and at least once per 18 months (8) | 7 |
| 6. Full Length Control Rod | Movement of at least 10 steps in any one direction for any rod not fully inserted | Monthly | 7 |
| 7. Pressurizer Safety Valves | Set point | Each Refueling shutdown | 4 |
| 8. Main Steam Safety Valves | Set point | Each Refueling shutdown | 10 |

Amendment No. ~~6~~, ~~15~~, ~~23~~, 40

Table 4.1-2 (Continued)

| | | Test | Frequency | FSAR Section Reference |
|---|---|---|---|---|
| 9. | Containment Isolation Trip | Functioning | Each Refueling Shutdown | 5 |
| 10. | Refueling System Interlocks | Functioning | Prior to Refueling Operations | 9.4.5 |
| 11. | Service Water System | Functioning | Each Refueling Shutdown | 9.5.5 |
| 12. | Fire Protection Pump and Power Supply | Functioning | Monthly | 9.5.5 |
| 13. | Spray Additive Tank | NaOH Concent. | Monthly | 7 |
| 14. | Accumulator | Boron Concentration | Bi-Monthly | 6 |
| 15. | Primary System Leakage | Evaluate | Daily | 4 |
| 16. | Diesel Fuel Supply | Fuel Inventory | Daily | 8.2.3 |
| 17. | Spent Fuel Pit | Boron Concentration | Monthly | 9.5.5 |
| 18. | Secondary Coolant Samples | Gross activity | 72 hours (5)(6) | |
| 19. | Circulating Water Flood Protection Equipment | Calibrate | Each Refueling Shutdown | |

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1.a or increases by

10 μ Ci/gm from the previous measured level, the sampling frequency shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of E. A radiochemical analysis and calculation of E and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μ Ci/gm from the previous measured level.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μ Ci/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give equivalent I-131 activity.

(5) An isotopic analysis for DOSE EQUIVALENT I-131 concentration is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.

(6) Not required during a cold or refueling shutdown.

(7) During a cold or refueling shutdown, primary coolant Gross Radioactivity will be determined weekly.

(8) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 40 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

## ROCHESTER GAS AND ELECTRIC CORPORATION

## R. E. GINNA NUCLEAR POWER PLANT

## DOCKET NO. 50-244

### 1. Background

The NRC staff has conducted a review of the Licensee Event Reports (LERs) and Technical Specification requirements related to the Control Rod Position Indication Systems (RPI) at Westinghouse PWRs and determined that a wide variation exists in the number of LERs received and the Technical Specification requirements.

### 2.0 Discussion and Evaluation

Westinghouse has performed safety analyses for control rod misalignment up to 15 inches or 24 steps (one step equals 5/8 inch). Since analysis of misalignments in excess of this amount have not been submitted, we have imposed an LCO restricting continued operation with a misalignment in excess of 15 inches. Because the analog control rod position indication system has an uncertainty of 7.5 inches (12 steps), when an indicated deviation of 12 steps exists, the actual misalignment may be 15 inches. This is because one of the coils, spaced at 3.75 inches, may be failed without the operator's knowledge. The Standard Technical Specifications were written to eliminate any confusion about this, and restrict deviations to 12 indicated steps. Surveillance requirements, on the indication accuracy of 12 steps, were also prepared to ensure that the 15 inch LCO is met. Since there is no difference intended in requirements issued for any Westinghouse reactor, plants with Technical Specifications written in different terms of misalignment should consider the 12 step instrument inaccuracy when monitoring rod position.

A related problem is that the installed analog control rod position indicating system equipment may not, in some areas, be adequate to maintain the control rod misalignment specification requirement because of drift problems in the calibration curves. This is evidenced by numerous LERs concerning rod position indication accuracy. In these cases, the uncertainty may be more than 12 steps.

Rochester Gas and Electric Corporation (RG&E) was requested, by letter dated November 5, 1979, to review the technical specifications for the R. E. Ginna Nuclear Power Plant to ensure that the control rods are required to be maintained within ± 12 steps indicated position and that the rod position indication system is accurate to within ± 12 steps.

By letter notarized August 29, 1980 (transmitted September 3, 1980), RG&E responded to the NRC request and provided proposed technical specification changes to incorporate the staff's requirements.

Based on our review of the licensee's submittal, we find that the proposed changes are in conformance with the staff's request and are, therefore, acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATION

We have determined that the proposed amendment does not authorize a change in effluent types, increase in total amounts of effluents, or an increase in power level, and will not result in any significant environmental impact. Having made this determination, we have 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.

## 4.0 CONCLUSION

We also conclude, 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 the health and safety of the public.

Date: April 17, 1981

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. 40 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 its date of issuance.

The amendment incorporates technical specifications regarding control rod position indication.

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 notarized August 29, 1980 (transmitted by letter dated September 3, 1980), (2) Amendment No. 40 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 17th day of April, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

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

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(10-80) U.S. NUCLEAR REGULATORY COMMISSION

# REACTOR FACILITY FEE DETERMINATION

1 FEE FORM TYPE (Check one)
- [X] PRELIMINARY
- [X] FINAL
- [ ] AMENDED

**INSTRUCTIONS.** Fill-in items 1 through 14, as applicable, and send the original copy to the License Fee Management Branch.

2. DOCKET NUMBER(S): 50-244

3. ACCESSION NUMBER: 8009040413

4. LICENSEE: ROCHESTER GAS AND ELECTRIC CORPORATION

5. PLANT NAME AND UNIT(S): R. E. GINNA UNIT NO. 1

6. DATE OF APPLICATION: 9/3/80

7. FEE REMITTED: [X] YES [ ] NO

8. LICENSEE FEE DETERMINATION

| CLASS I | CLASS II | CLASS III | CLASS IV | CLASS V | CLASS VI | EXEMPT | NONE |
|---|---|---|---|---|---|---|---|
| | | ✓ | | | | | |

9. SUBJECT: TECH. SPEC. AMENDMENT - CONTROL ROD MISALIGNMENT

10. TAC NUMBER ASSIGNED (If available): 43244

11. APPROVAL: [✓] LETTER [ ] ORDER

DATE OF ISSUANCE: 4/17/81

AMENDMENT NUMBER(S): 40

12. NRC FEE DETERMINATION

- [✓] The above application has been reviewed in accordance with Section 170.22 of Part 170 and is properly categorized.
- [ ] The above application has been reviewed in accordance with Section 170.22 of Part 170 and is incorrectly classified

Fee should be class(es):

JUSTIFICATION FOR CLASSIFICATION OR RECLASSIFICATION

- [ ] This application is a Class ______ type of action and is exempt from fees because it is:
  - [ ] Filed by a nonprofit educational institution.
  - [ ] Filed by a Government agency and is not for a power reactor.
  - [ ] For a Class I, II, or III amendment which results from an NRC request dated ______ for the application and the amendment is to simplify or clarify License or Technical Specifications; has only minor safety significance, and is being issued for the convenience of NRC *(must meet all of the criteria)*.
  - [ ] Other *(State reason therefor)*

13. SIGNATURE *(Branch Chief)*: Dennis M. Crutchfield — DATE: 12/15/80

KS 12/1

14. FINAL CERTIFICATION The preliminary fee determination has been reassessed and is hereby affirmed
SIGNATURE *(Project Manager or Branch Chief)*: Dennis M. Crutchfield — DATE: 12/15/80

**FOR LICENSE FEE MANAGEMENT BRANCH USE ONLY** *(All others do not write below this line)*

The above exemption request has been reviewed and is hereby accepted as being exempt
SIGNATURE *(Chief LFMB)* — DATE

**DISTRIBUTION BY LFMB**

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| [ ] | Records Services Branch | [ ] | DL Branch Chief | [ ] | LFMB Exemption File | [ ] | LFMB Reactor File | [ ] |