Docket No. 50-244 February 23, 1988

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

SUBJECT: ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE NO. DPR-18

The Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated October 27, 1987 and January 11, 1988.

The amendment revises the requirements of the Technical Specifications related to steam generator tube plugging at 15 percent level. Also, you are requested to submit a proposed schedule within three months of the review of the modified methodology (including LOFTTR2) and revised results in WCAP-11668, reflecting the steam generator uncovery effect during the steam generator tube rupture event. The schedule should take into consideration the industry action that is underway to resolve this issue.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

Carl Stahle, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 25 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

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

February 23, 1988

Posted
Amdt. 25
to DPR-18

Docket No. 50-244

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Rochester Gas & Electric Corporation
89 East Avenue
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Sincerely,

Carl Stahle

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Project Directorate I-3
Division of Reactor Projects I/II

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Mr. Roger W. Kober
Rochester Gas and Electric Corporation

R. E. Ginna Nuclear Power Plant

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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 FACILITY OPERATING LICENSE

Amendment No. 25
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 October 27, 1987, and supplemented on January 11, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 25 , 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

Richard Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 23, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

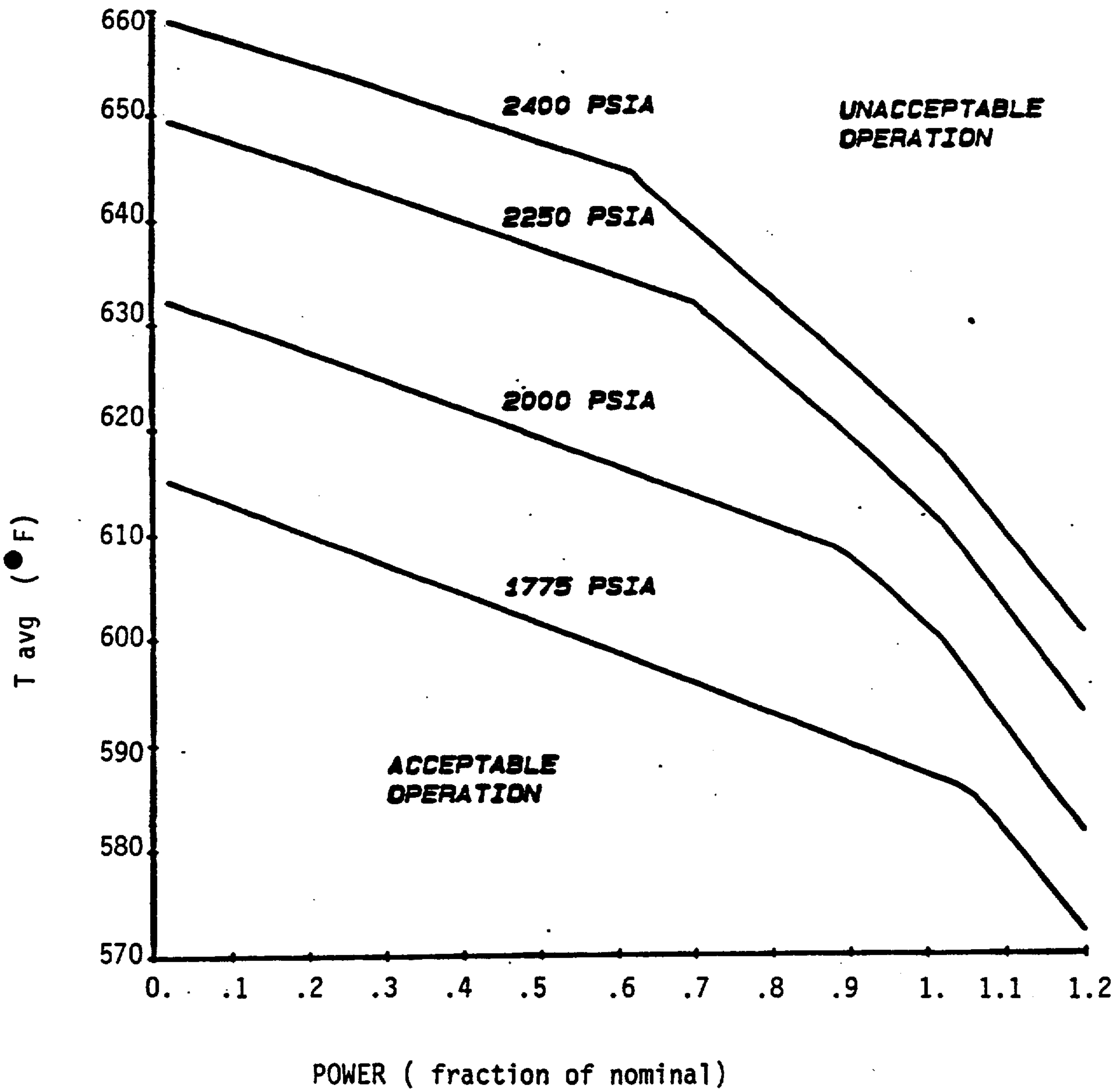
Revise Appendix "A" as follows:

Remove Pages	Insert Pages
2.1-4	2.1-4
2.3-2	2.3-2
2.3-3	2.3-3
2.3-6	2.3-6
2.3-7	2.3-7
2.3-8	2.3-8*
----	2.3-8a**
2.3-9	2.3-9

*No Change - Repositioned on page

**Indicates new page

FIGURE 2.1-1
CORE DNB SAFETY LIMITS
2 LOOP OPERATION



d. Overtemperature ΔT

$$\leq \Delta T_o \left[K_1 + K_2 (P-P^1) - K_3 (T-T^1) \left(\frac{1 + \tau 1 S}{1 + \tau 2 S}\right)\right] - f(\Delta I)$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T^1	=	573.5°F
P	=	pressurizer pressure, psig
P^1	=	2235 psig
K_1	=	1.20
K_2	=	.000900
K_3	=	.0209
$\tau 1$	=	25 sec
$\tau 2$	=	5 sec

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ less than +13 percent, $f(\Delta I) = 0$

(ii) for each percent that the magnitude of $q_t - q_b$ is more positive than +13 percent, the ΔT trip set point shall be automatically reduced by equivalent of 1.3 percent of rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5 (T-T^1) - K_6 \frac{\tau 3\, ST}{\tau 3S + 1}\right] - f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = indicated T avg at nominal conditions at rated power, °F

K_4 = 1.077

K_5 = 0.0 for $T<T^1$

= 0.0011 for $T \geq T^1$

K_6 = 0.0262 for increasing T

= 0.0 for decreasing T

$\tau 3$ = 10 sec

$f(\Delta I)$ = as defined in 2.3.1.2.d

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 the thermal capacity of the reactor coolant system to respond to power increases [1][2] 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 Figure 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.[4] 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 of the UFSAR. This setpoint 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.[1] 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 points specified are consistent with the value used in the accident analysis.[1] 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. A trip at this set point contains margin for both normal instrument error and transient overshoot of level beyond this trip setting. An additional 4% instrument error has been assumed to account for the effects of elevated temperatures on level measurement in accordance with IE Bulletin 79-21.(12) Therefore a trip setpoint of 88% prevents the water level from reaching the safety valves.(4)

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. A set point of 5% is equivalent to at least 40,000 lbs. of water and 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.(5) An additional 11% has been added to the set point to account for error which may be introduced into the steam generator level system at a containment temperature of 286°F as determined by evaluation performed for temperature effects on level measurements required by IE Bulletin 79-21.

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) UFSAR 15.0

(2) UFSAR 15.4

(3) UFSAR 15.6

(4) UFSAR 7.2

(5) UFSAR 15.2

(6) Deleted

(7) Deleted

(8) Deleted

(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.

(12) Letter from L.D. White, Jr. to B. Grier, USNRC dated September 14, 1979.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO STEAM GENERATOR TUBE PLUGGING AT 15 PERCENT LEVEL
SUPPORTING AMENDMENT NO.25 TO FACILITY OPERATING LICENSE NO. DPR-18
ROCHESTER GAS AND ELECTRIC CORPORATION
R. E. GINNA NUCLEAR POWER PLANT
DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated October 27, 1987 (Reference 1), Rochester Gas and Electric Corporation (the licensee) proposed an amendment to the Technical Specifications to request operation up to full power with steam generator tube plugging (SGTP) levels of 15 percent. The current NRC approved reload analyses for Ginna in Reference 3 were based on the assumptions of SGTP levels of 10 percent for transients and 12 percent for loss-of-coolant (LOCA) analyses. With the anticipated increase in the percentage of plugged steam generator tubes for plant operation beyond current fuel cycle, the licensee proposed Technical Specification changes to support a request of full power operation with SGTP levels of up to 15 percent. The Technical Specification changes include (1) Figure 2.1-1, core DNBR safety limits reflecting reduction in the RCS thermal design flow with the higher level of tube plugging, (2) the changes in the Overtemperature and Overpower Delta-T trip setpoints to provide protection for the changed core DNBR safety limits, and (3) changes to the bases to incorporate updated descriptions and references. The staff has reviewed the proposed Technical Specification changes and the supporting analytical results (References 1 and 2) and has prepared the following evaluation.

2.0 TRANSIENTS EVALUATION

Ginna is currently operating with a mixed core consisting of Westinghouse 14x14 OFA fuel and Exxon fuel. Since the Westinghouse 14x14 OFA fuel has operated for less fuel cycles than the Exxon fuel, the Westinghouse fuel assemblies have a higher power peaking factor, due to lower burndown effect, and were evaluated to be the limiting fuel from the DNB standpoint. The licensee, therefore, performed safety analyses based on the Westinghouse 14x14 OFA fuel assemblies to support the Technical Specification changes.

In Reference 1, the licensee indicated that the increase in SGTP from 10 percent to 15 percent level results in an approximately 2.2 percent reduction in RCS thermal design flow. For conservatism, the licensee assumed a 3 percent reduction in design flow for analyses supporting the request for the Technical Specifications changes. At the staff's request, the licensee evaluated the impact of operation at 3 percent reduction in design flow on thermal margin and documented the results in Reference 2.

The analyses show that a 3 percent flow reduction will result in a DNBR reduction of 3.3 percent. This result is obtained by using a previously approved sensitivity factor for the rate of change of DNBR with respect to flow reduction and is acceptable.

The licensee also calculated the rod bow penalty on DNBR by using the approved method (Reference 4) and obtained the maximum calculated rod bow penalty of 1.0 percent for the Westinghouse 14x14 OFA fuel in the core.

Since WRB-1 (Reference 5) correlation was used to establish the operating DNBR limit for the core, the generally approved DNBR margin of 12 percent is applicable to the core. This margin is sufficient to compensate for the 3.3 percent penalty associated with the reduced design flow, 1.0 percent rod bow penalty and 2.0 percent penalty generally accepted for the Westinghouse mixing fuel core.

The licensee has also reanalyzed the following transients and non-LOCA accidents: (1) uncontrolled RCCA withdrawal at power, (2) chemical and volume control system malfunction, (3) reactor coolant pump locked rotor, (4) loss of external electrical load, (5) excessive load increase incident and (6) rupture of a control rod mechanism housing RCCA ejection.

As a result of the thermal margin evaluation and transient and non-LOCA reanalyses, the licensee concluded that, even with an assumed 3 percent flow reduction no safety criteria will be violated during transients and non-LOCAs and the results differ insignificantly from that in the current reload analyses (Reference 3).

The licensee also reanalyzed the steam generator tube rupture event (SGTR) (Reference 7 and 8) by using the LOFTTR2 code, which is a modified version of the approved LOFTTR1 code. In the radiological analyses, the licensee assumed that element iodine, transferred from the primary side via tube leaks, will partition between the SG water and steam. As a result, the iodine concentration in the steam is a small fraction of the water concentration. The tube rupture occurred at location which is uncovered by water during the transient. This creates a direct activity release path to the environment and causes a maximum radiological release because this direct primary-to-environment leakage involves neither dilution by the secondary side water, nor partitioning of the activity carried by the leakage flow. Since LOFTTR2 cannot simulate the SGTR events with inclusion of the effect of the SG tube uncovery, the licensee assumed a factor of 3 for the accident initiated iodine spike and a factor of 6 for the preaccident iodine spike for the radiological release analyses. The calculated results, with consideration of penalty factors for dose increase accounting for the SG tube uncovery phenomenon, show that an increase in SGTP from 10 percent to 15 percent results in a slight increase in the calculated offsite radiation dose which still remain below the acceptance criteria of 10 CFR 100. The licensee indicated in Reference 7 that a methodology to calculate the doses due to the effect of SG tube uncovery will be developed by the Owners Group or a subgroup of utilities and the results in WCAP-11668 will be revised to reflect the effects of using the modified methodology. Based on the licensee's evaluation and the commitment to update the methodology, the staff concludes that the SGTR analyses are acceptable. However, the licensee is required to submit the modified methodology (including LOFTTR2) and revised results (in WCAP-11668) to account for the SG tube uncovery effect during the SGTR event for review and approval. The licensee should provide a schedule which is acceptable to the staff for the submittal within 3 months of the receipt of this evaluation.

Based on its review of the licensee's evaluation process and results, the staff concludes that the operation with SGTP level up to 15 percent is acceptable for the transients and non-LOCA responses.

2.2 LOSS-OF-COOLANT-ACCIDENT ANALYSIS EVALUATION

The licensee indicated that the sensitivity study (Reference 6) showed that the effect of an increase in tube plugging from 12 percent to 15 percent on the results of the small break LOCA is small. In addition, the current limiting small break results for 12 percent tube plugging demonstrated a margin of 1108°F to the acceptance limits of 10 CFR 50.46. Therefore, the licensee concluded and the staff agreed that the effect of an increase of 3 percent in SGTP on the results of the existing small break LOCA analyses is small.

The licensee provided an evaluation (Reference 1) on the effect of a 15 percent SGTP level on the large break LOCA. The licensee reanalyzed and evaluated the double ended cold leg guillotine (DECLG) break, with a discharge coefficient of 0.4, since this break was identified previously as the limiting case resulting in the highest peak cladding temperature. The DECLG break analysis was performed at 102 percent of rated core power of 1520 Mwt and at a total peaking factor of 2.32. Assumptions of a SGTP level of 15 percent and a reduced reactor coolant system loop flow rate of 85,000 gpm were made for the analysis.

The analysis was performed by using a modified version of the 1981 Westinghouse ECCS evaluation model (Reference 9). This evaluation model uses the SATAN-VI code (Reference 10) for the thermal-hydraulic transient analysis for the reactor coolant system during blowdown, the WREFLOOD code (Reference 11) for the analysis of the refill and reflood transient periods, the COCO code (Reference 12) for the containment pressure transient, and LOCTA-IV (Reference 13) for the calculation of the peak cladding temperature.

The staff has reviewed the large break analysis and found that (1) the calculated peak cladding temperature is 1887°F which is less than the safety limit of 2200°F, (2) the maximum local metal-water reaction is 2.8 percent which is below the limit of 17 percent, and (3) the total core metal-water reaction is less than 0.3 percent which is within the acceptable limit of 1.0 percent.

The staff, therefore, concludes that the results presented are acceptable since approved methods and computer codes were used and the analytical results show that the peak cladding temperature, metal-water reactions and clad oxidation are within the acceptance criteria of 10 CFR 50.46.

2.3 TECHNICAL SPECIFICATIONS CHANGES

The Technical Specifications changes submitted in Reference 1 for review and approval include: (1) changes in DNBR safety limits (Figure 2.1-1) reflecting the reduced RCS flow resulting from higher percentage of the plugged steam generator tubes, (2) changes in setpoints for Overtemperature and Overpower Delta-T trips (pages 2.3-2 and 2.3-3 of the Technical Specifications) reflecting the changed DNBR safety limits discussed in item (1) above, and (3) changes to bases (pages 2.3-6 to 2.3-9 of the Technical Specifications) to include updated descriptions and references.

The staff has reviewed the proposed Technical Specifications changes and found that the changes are acceptable since the changes discussed in items (1) and (2) are consistent with the assumptions used in revised safety analyses which demonstrate that the appropriate acceptance criteria are satisfied, and changes discussed in item (3) are editorial changes (Reference 3) and have no safety significance.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has evaluated the licensee's request to operate the plant up to full power with SGTP levels up to 15 percent and the associated Technical Specifications changes. Based on its review of the LOCA and transient analyses (Reference 1) provided by the licensee and additional information (Reference 2) requested by the staff, the staff has concluded that there is reasonable assurance that operation of the plant at full power with SGTP levels up to 15 percent does not violate the safety limits used for the current reload analysis (Reference 3) during transients and satisfies the performance requirements of 10 CFR 50.46 during LOCA.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from R. W. Kober (RGE) to C. Stahle (NRC), dated October 27, 1987.

2. Letter from R. W. Kober (RGE) to C. Stahle (NRC), dated January 11, 1988.

3. Letter from RGE to NRC, dated September 20, 1983, Reload Transition Safety Report.

4. WCAP-8691 (Revision 1) - Fuel Rod Bow Evaluation, July 1979.

5. WCAP-8762: New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Van Grids, July 1976.

6. Proceedings of Specialist Meeting on Small Break LOCA Analysis, Pisa, Italy, June 1985, Papers by S. Ciani and N. Lee.

7. Letter from R. Kober (RGE) to C. Stahle, dated December 24, 1987.

8. WCAP-11668: LOFTTR2 Analysis of Potential Radiological Consequences following a Steam Generator Tube Rupture at the Ginna Nuclear Power Plant.

9. WCAP-9221-A: Westinghouse ECCS Evaluation Model, 1981 Version, Revision 1, 1981.

10. WCAP-8306: SATAN-VI Program: Comprehensive Space-Time dependent Analysis of Loss-of-Coolant Accident (WREFLOOD code), June 1974.

11. WCAP-8171: Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD code), June 1974.

12. WCAP-8326: Containment Pressure Analysis Code (COCO), June 1974.

13. WCAP-8305: LOCTA-IV Program: Loss-of-Coolant Transient Analysis, June 1974.

Date: February 23, 1988

Principal Contributor: S. Sun, SRXB

February 23, 1988

MEMORANDUM FOR: Sholly Coordinator

FROM: Carl Stahle, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

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SUBJECT: REQUEST FOR PUBLICATION IN BI-WEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: October 27, 1987, as supplemented by letter dated January 11, 1988.

Brief Description of amendment: This amendment changes the requirements of the Technical Specifications related to steam generator tube plugging at 15 percent level.

Date of issuance: February 23, 1988

Effective date: February 23, 1988

Amendment No.: 25

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 2, 1987 (52 FR 45889).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 1988.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

NRC Project Director: Richard H. Wessman, Director

/S/

Carl Stahle, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

OFC	:PDI-3	:PDI-3	:PDI-3	:	:	:	:
NAME	:MRushbrook:ah	CStahle	:RHWessman	:	:	:	: :
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