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. 19
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 January 6, 1978, as supplemented by letters dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, 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 Appendices A and B, as revised through Amendment No. 19 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

Darrell G. Eisenhut

Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance May 1, 1978

cc
Lex K. Larson, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Mr. Michael Slade
1250 Crown Point Drive
Webster, New York 14580

Rochester Committee for Scientific Information
Robert E. Lee, Ph.D.
P. O. Box 5236 River Campus Station
Rochester, New York 14627

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

Director, Technical Development Programs - (w/cys of 4/7/77, 1/6/78, 1/10/78, 3/27/78, 4/6/78, 4/17/78, and 4/25/78 filings by RG&E)
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14627

Supervisor of the Town of Ontario
107 Ridge Road West
Ontario, New York 14519

Chief, Energy Systems Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

ATTACHMENT TO LICENSE AMENDMENT NO. 19

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Change the Technical Specifications contained in Appendix A of License No. DPR-18 as indicated below. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

Remove	Insert
3.10-2	3.10-2
----	3.10-2a
3.10-4	3.10-4
3.10-8c	3.10-8c

3.10.1.2 When the reactor is critical except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and control rod exercises, each group of control rods shall be inserted no further than the limits shown by the lines on Figure 3.10-1 and moved sequentially with a 100 ($\pm$5) step overlap between successive banks.

3.10.1.4 During control rod exercises indicated in Table 4.1-2, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.1.5 The part length control rods will not be inserted except for physics tests or for axial offset calibration performed at 75% power or less.

3.10.1.6 During measurement of control rod worth and shutdown margin, the shutdown margin requirement, Specification 3.10.1.1, need not be observed provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion and all part length control rods are fully withdrawn. Each full length control rod not fully inserted, that is, the rods available for trip insertion, shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the shutdown margin to less than the limits of Specification 3.10.1.1. The position of each full length rod not fully inserted, that is, available for trip insertion, shall be determined at least once per 2 hours.

3.10.2 Power Distribution Limits and Misaligned Control Rod

3.10.2.1 The movable detector system shall be used to measure power distribution after each fuel reloading prior to operation of the plant at 50% of rated power to ensure that design limits are not exceeded.

If the core is operating above 75% power with one excore nuclear channel out of service, then the quadrant to

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip "set point is reduced by 50%".

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by interpolation using the most recent measured value and the predicted value at the end of the cycle life. The target flux difference shall be between +5.0 and -7.5% at the beginning of cycle life and between +2.0 and -7.5% at the end of cycle life. Linear interpolation shall be used to determine values at other times in cycle life.

3.10.2.8 Except during physics tests, control rod excercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within ± 5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

different from those resulting from operation 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 ± 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 berify 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 and $F_{\Delta H}$ 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~ peaking factors.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By application dated January 6, 1978, as supplemented by letters dated January 10, March 27, April 6, April 17, and April 25, 1978, Rochester Gas and Electric Corporation (the licensee) requested authorization to operate the R. E. Ginna Nuclear Power Station in Cycle 8 with reload fuel supplied by Exxon Nuclear Company, Inc., and requested a change to the Technical Specifications involving power distribution control limits.

Discussion

The R. E. Ginna Nuclear Power Station has operated seven fuel cycles with fuel supplied by Westinghouse Corporation. Cycle 8 will involve the first use of fuel from a different vendor, Exxon Nuclear Company, Inc. (ENC). The loading for Cycle 8 will consist of 32 new ENC fuel assemblies loaded at the periphery of the core and 89 exposed Westinghouse assemblies scatter loaded in the center of the core. All assemblies are of similar design with the ENC assemblies designed to be compatible with the other fuel assemblies. Reactor power level, core average linear heat rate and primary coolant system temperature and pressure for Cycle 8 will remain the same as for the previous cycle.

The licensee has stated that all technical specification limits for the previous cycle are applicable to Cycle 8, with the exception of one limit involving power distribution control. The licensee also proposed a change to the bases of the specifications involving power distribution control to reflect a revised methodology used in the reactor physics analyses for Cycle 8.

The licensee's analyses for Cycle 8 also include the first use of ENC analytical methods to verify the acceptability of Ginna operating limitations and safety margins.

The staff evaluation which follows, addresses the acceptability of the use of the ENC assemblies in Cycle 8 and the acceptability of the proposed changes in Technical Specification. The evaluation includes the staff's review of nuclear, thermal-hydraulic and accident analyses for Cycle 8 operation.

Evaluation

1. Design of the New Fuel

The new fuel assemblies for the core periphery were designed by Exxon Nuclear Corporation to be compatible with the Westinghouse depleted fuel assemblies that are to remain in the Ginna core.

The Exxon fuel design is similar to the Westinghouse fuel bundle design (References 1 and 2).

The Exxon fuel design criteria and fuel design calculations are discussed in Exxon reports submitted with the application for Fuel Cycle 8 operation. Those aspects of the fuel design important to safety have been reviewed by the staff and found acceptable. Those aspects are: (1) the fuel performance during LOCA; (2) fuel clad collapse and fuel densification; (3) fretting wear; and (4) the effect of fuel rod bowing on the departure from nucleate boiling ratio (DNBR).

The GAPEX code (Reference 3) was used to calculate stored energy for LOCA calculations. GAPEX has been reviewed and approved by the staff for fuel temperature and internal pressure calculations in PWR fuel (Reference 4).

Reference 1 presents calculations which show that the cladding will not collapse during Cycle 8. These calculations utilize the RODEX and COLAPX codes. The RODEX code (Reference 5) calculates the cladding temperature and fuel rod internal pressure while COLAPX (Reference 7) calculates the collapse time using the RODEX input. COLAPX has been reviewed by the staff and found acceptable for cladding collapse calculations. RODEX has not been approved by the staff but the models in RODEX affecting clad temperature and internal pressure are similar to those in the GAPEX code, which has been approved. Moreover, since the clad collapse analyses for the Westinghouse fuel does not predict collapse during Cycle 8, and since the cladding for the Exxon fuel is thicker than that of the Westinghouse fuel (Reference 2) which makes it more resistent to clad collapse, we have concluded, with reasonable assurance, that the results of the RODEX analysis are acceptable.

Exxon tests to determine the magnitude of fretting at the fuel rod axial spacer contact points due to flow induced vibration revealed no active fretting corrosion and negligible difference in wear observed between 500, 1000, and 1500 hours. Based on these test results and the larger diameter - thicker clad of the Exxon fuel rods in the 14 x 14 fuel assemblies for Ginna and therefore greater stiffness, we have concluded that fuel rod integrity with respect to flow induced vibration and fretting wear is acceptable.

The effect of fuel rod bowing on Departure from Nucleate Boiling Ratio (DNBR) has been a subject of continuing discussion between the staff and Exxon. An Exxon analysis considered the fuel rod bowing penalties for the most limiting transients and attempted to show that there is sufficient margin to offset the calculated penalties. These results are presented in Reference 2. The staff has concluded that these analyses are not completely acceptable because the heat flux and pressure used to calculate the bowing penalties were for minimum DNBR conditions and do not represent the worst conditions for calculating the rod bowing penalties. However, Reference 2 shows that there is an 8.5 percent margin to the safety limit which offsets this nonconservatism. On this basis, we have concluded that there is adequate thermal margin to assure safe plant operation without violating the minimum DNBR safety limit.

Based on successful irradiation experience of Exxon fuel assemblies in other PWR cores and the analyses which have been done for Ginna Fuel Cycle 8, we have concluded that the Exxon fuel assemblies for Cycle 8 will perform in a safe and acceptable manner. The licensee has agreed (RG&E telecon 4/14/78) to submit plans for inspection of the Exxon fuel assemblies to NRC for concurrence at least 90 days prior to the end of Fuel Cycle 8 to enable additional NRC review of the fuel prior to its use in Cycle 9.

2. <u>Thermal Hydraulic Design</u>

The new Exxon fuel assemblies are designed to have thermal hydraulic characteristics equivalent to those of the existing fuel. Therefore, there will not be any major differences in the thermal hydraulic behavior of the core.

The licensee has shown that at 118 percent of rated power, the calculated minimum DNBR is 1.47. The corresponding value for the Westinghouse fuel assemblies is 1.43. The fuel and cladding temperature analysis uses Exxon calculational methods (Reference 7), assuming maximum power peaking and engineering tolerances. The calculated maximum fuel and cladding temperatures are well below the design limits. We, therefore, conclude that the Exxon fuel assemblies are compatible with the Westinghouse fuel assemblies in the Ginna core and that the thermal hydraulic criteria will not be exceeded during plant operation.

3. Nuclear Design

The Fuel Cycle 8 loading will consist of 89 fuel assemblies with burnups ranging from 7,178 MWD/MTU to 23,813 MWD/MTU and 32 fresh ENC fuel assemblies.

The licensee has specified new values for the target flux difference. They are between +5.0 and -7.5% for the beginning of cycle life and between +2.0 and -7.5% for the end of cycle life. For the intermediate times the values are obtained by linear interpolation. The licensee has compared the neutronic characteristics of the Cycle 8 and Cycle 7 cores and concluded that they are approximately the same. The reactivity coefficients of the Cycle 8 core are bounded by the coefficients used in the safety analyses and we have concluded that the coefficients are acceptable.

Justification of the assumed total rod worth uncertainty of 10% used in the determination of shutdown margin has not been presented. Confirmatory tests are therefore included in the startup physics tests for fuel Cycle 8.

The physics startup test program for Ginna Cycle 8 presented in the March 27, 1978 submittal (Reference 2), was reviewed with the licensee. Several changes to the rod worth and power coefficient measurements were made. These changes are documented in the Reference 17 submittal. As part of this test program, control rod reactivity worth will be measured for banks D, C, B and A in order to verify that adequate shutdown margin is available. If any one bank worth differs from the predicted value by more than 15% or the sum of the worths of these banks differs from the predicted value by more than 10%, the first shutdown bank should be measured. If the sum of the five measured banks differs from the predicted value by more than 10%, additional shutdown bank measurements will be performed to verify the technical specification shutdown margin.

We have concluded that the total physics startup test program as modified is acceptable. However, there are areas in the licensee's safety analysis that warrant verification in the physics startup test program. Therefore, a summary report as described in the March 27th submittal (Reference 2) will be submitted to the NRC. The licensee has agreed to submit the report within 45 days of completion of the program.

4. Steady State and Load Follow Operation

Compliance with F_Q and $F\Delta H$ limiting conditions for operation is ensured by adherence to previously approved constant axial offset control strategy and core monitoring with incore and excore flux monitors. Incore monitoring is achieved using travelling fission chambers. Data from the fission chambers and calculated coefficients

(Reference 9) are processed by the computer code INCORE to obtain power distribution maps. Extensive comparisons of predicted and measured core power distributions have been performed by Exxon for 3 and 4 loop cores. In general, the results of these comparisons are favorable. However, R. E. Ginna is a two loop plant and there is only a single set of measured and calculated power distributions for R. E. Ginna, Cycle 7, at hot full power, 1000 MWD/MTU. The results of this comparison show good agreement between measurement and calculation and add credibility to the licensee's assertion that an F_Q uncertainty factor of 5% is appropriate for Cycle 8. However, additional data will be obtained during the fuel cycle 8 startup physics tests.

5. Safety Analyses

The licensee has analyzed the anticipated operating occurrences and postulated accidents using the plant transient simulator code PTSPWR (Reference 15). The results of these analyses are presented in Reference 14. Our review of this code has progressed sufficently to allow us to conclude that analyses using PTSPWR provide acceptable margins to peak linear heat generation rate and departure from nucleate boiling design limits. The reactivity coefficients assumed in the safety analyses are to be confirmed during the physics startup tests.

a. Steam Line Break Analyses

The Steam Line Break (SLB) accident analysis (Reference 14) is of particular concern. SLB analysis methods have not been generically approved. The licensee asserts that should a large SLB occur the plant would return to criticality, reaching a peak average core power of 22% of rated power at approximately 90 sec after accident initiation. The minimum DNBR at this condition, using the Macbeth critical heat flux correlation, would be 1.58. Even if DNB were to occur during a steam line break accident, DNB would be restricted to a small region of the core in the vicinity of the assumed stuck rod. It is noted that DNB anywhere in the core is unlikely if all control rods scram as expected. Of the fuel rods which might experience DNB in the vicinity of the stuck rod, some fraction would release their fission gas inventory. The fission gas would have to be transported to the secondary side of the coolant system (primary to secondary steam generator leakage) in order to represent a potential hazard. The potential release to the atmosphere would be significantly less than 10 CFR Part 100 limits. Accordingly, we have concluded that the consequences of a steam line break are acceptable.

b. ECCS Analysis

The licensee has submitted ECCS performance analyses for the Westinghouse (Reference 19) and new ENC fuels (Reference 1). The Westinghouse analysis was performed for Cycle 7 fuel which the staff believes is a conservative evaluation for the Westinghouse fuel during Cycle 8. The ENC analysis was performed for Cycle 8 using the ENC WREM-II ECCS evaluation model (Reference 7) which is described in References 8 and 9. The applicability of the model to two-loop Westinghouse PWR plants was evaluated by ENC in Reference 10. The ENC evaluation model has been reviewed and approved conditionally by the NRC (Reference 16). The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of the Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghouse model did not consider interaction between UPI water and steam flow. (References 11 and 12). After plant specific submittals by the licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable as an interim basis for continued safe operation of the Westinghouse two loop plants, while long term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered (Reference 20). In the interim, before these models are developed, the licensee has provided a modification to the current Westinghouse model which accounts for UPI-core interaction (Reference 13). It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion and agreed to submit within 30 days, calculational results to confirm the validity of this conclusion. (Reference 21).

The ECCS analyses have been performed with the upper head fluid temperature equal to the fluid outlet (hot leg) temperature and assuming 10 percent of steam generator tubes plugged. The analyses included a spectrum of breaks which consisted of guillotine double ended cold leg (DEGCL) breaks with discharge coefficients of 1.0, 0.6 and 0.4 and split breaks with break areas of 8.25, 4.9 and 3.30 ft^2. No small break analysis was performed. The licensee has demonstrated, by showing analogy between the present analysis and the analyses performed previously for other plants, that the small break LOCA is not limiting (Reference 2). The critical break size was determined to be DEGCL with C_D=0.4.

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection (unless the plant is modified), analyses at the specific operating conditions applicable to the Ginna plant demonstrate that the effect of disregarding upper plenum injection interaction on refill and reflood conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which the models are applied for conditions at the Ginna plant, the models do not deviate from the requirements of 10 CFR 50 Appendix K item I.D.3, and the calculations are acceptable.

On March 23, 1978 Westinghouse informed the NRC that an error in the West-ECCS evaluation model had been found which had resulted in incorrectly calculated peak clad temperatures in all LOCA analyses previously submitted by their customers. For several plants preliminary estimates indicated that they would not meet the 2200°F limit of 10 CFR 50.46 at their present maximum overall peaking factor limits. Westinghouse and several of their customers met with the NRC staff on March 29, 1978 in Bethesda to discuss the error and its impact on specific plant analyses. Subsequent to that meeting, Westinghouse provided information through the licensees of operating reactors to justify continued operation at the interim peaking factor Technical Specification limits proposed by the NRC staff on April 3, 1978.

On April 17, 1978 (Reference 19) RG&E submitted a letter indicating that continued operation at their present Technical Specification limit of 2.32 (total peaking factor) was justified on the basis of additional generic Westinghouse analyses. Westinghouse had determined that the impact of correcting the error on the peak cladding temperature for the RE Ginna plant was significant but within the presently existing margin (228°F) to the 2200°F° acceptance criteria limit. The NRC Staff confirmed the conservatism of that and all other plant evaluations and on April 18, 1978 published a Safety Evaluation Report (Reference, attachment to Exemption). Since the Westinghouse and ENC fuels were analyzed using the respective Westinghouse and ENC evaluation models, and since there is no zirconium-water error in the ENC calculational model, the error in zirconium-water reaction in the Westinghouse calculational model has no effect on the Exxon calculations. The Zirconium-water reaction error in the Westinghouse model is the subject of an exemption request by the licensee dated April 25, 1978, (Reference 21) and a separate exemption action by NRC.

6. Technical Specification Changes

The proposed addition to the Technical Specifications restricts the permissable range of the target flux difference i.e. the ratio of the flux in the top half of the core minus the flux in the lower half of of the core to the total flux measured at 100% power, equilibrium conditions. The addition, Technical Specification 3.10.2.7, assures that axial power distributions realized in the reactor will be no more limiting with respect to linear heat generation rate than the axial power distributions used by Exxon to analytically confirm (Reference 18) that, limiting values of linear heat generation vs core height, Technical Specification 3.10.2.2, will not be violated. The restriction has been reviewed and approved on a generic basis and has recently been incorporated in the Technical Specifications of PWR's using Exxon Nuclear fuel.

The change to Technical Specification 3.10.1.4 and the addition of specification 3.10.1.6 are required to permit the physics testing program as discussed in part 3 of our evaluation. The change and the addition are in accordance with the Standard Technical Specifications for Westinghouse PWR's which we have already reviewed and approved.

The changes to the basis of the Technical Specification related to core power distribution are in accordance with the Standard Technical Specification which we have approved and are therefore acceptable also.

Environmental Consideration

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.

Conclusion

We have 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 1, 1978

REFERENCES

(1) Letter from LeBoef, Lamb, Leiby and MacRae (Counsel for Rechester Gas and Electric Corporation) to E. G. Case (NRC), dated January 6, 1978.

(2) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated March 27, 1978.

(3) XN-73-25, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients", June 1975.

(4) USNRC Report, "Technical Report on Densification of Exxon Nuclear PWR Fuel", February 27, 1975.

(5) XN-76-8(P), "RODEX: Fuel Rod Design Evaluation Code", February 1977.

(6) XN-72-23, "Clad Collapse Calculational Procedure", November 1, 1972.

(7) XN-NF-77-58, "ECCS analysis for the R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model", December 1977.

(8) XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", Vol I through III, July-August 1975 and Supplements 1 through 7, August-November 1975.

(9) XN-76-27, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II", July 1976 and Supplements 1 and 2, September-November 1976.

(10) XN-NF-77-25, "Exxon Nuclear Company ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment using the ENC WREM-II ECCS Model - Large Break Example Problem," August 1977.

(11) Letter from E. G. Case (NRC) to L. D. White, Jr. (Rochester Gas and Electric Corporation), dated December 16, 1977.

(12) Letter RG&E to NRC, Development of a New Model to Account for Upper Plenum Injection, dated March 5, 1978.

(13) Letter from L. D. Amish (Rochester Gas and Electric Corporation) to A. Schwencer (NRC), dated February 1978.

(14) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", November 1977.

(15) XN-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Revision 1, May 1975.

(16) USNRC Topical Report Evaluation, Exxon Nuclear Company Report XN-NF-77-25, April 1978.

(17) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated April 6, 1978.

(18) Exxon Nuclear Power Distribution Control for Pressurized Water Reactors XN-76-40, September 1976.

(19) Letter from L. D. White, Jr., (RG&E) to A. Schwencer (NRC) dated April 7, 1977.

(20) Letter to RG&E dated April 28, 1978 transmitting staff SER of UPI model evaluation.

(21) Letter from RG&E to NRC dated April 25, 1978, related to ENC UPI calculations.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter ROCHESTER GAS AND ELECTRIC CORPORATION (R. E. Ginna Nuclear Power Plant	))))))	Docket No. 50-244

EXEMPTION

I.

The Rochester Gas and Electric Corporation (the licensee), is the holder of Provisional Operating License No. DPR-18 which authorizes the operation of the nuclear power reactor known as R. E. Ginna Nuclear Power Plant (the facility) at steady reactor power levels not in excess of 1520 megawatts thermal (rated power). The facility consists of a Westinghouse Electric Company designed pressurized reactor (PWR) located at the licensee's site in Wayne County, New York.

II.

In accordance with the requirements of the Commission's ECCS Acceptance Criteria 10 CFR 50.46, the licensee submitted on April 7, 1977 and January 6, 1978 ECCS evaluations for proposed operation using 14 x 14 fuel manufactured by the Westinghouse Electric Company and the Exxon Nuclear Company (ENC). These evaluations established limits on the peaking factor based upon ECCS evaluation models developed by the Westinghouse Electric Company (Westinghouse), the designer of the Nuclear Steam Supply System for this facility, and by Exxon, the supplier of the reload fuel. The Westinghouse and ENC ECCS evaluation models had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46 and Appendix K. The evaluations indicated that with the peaking factor limited as set forth in the evaluations and with other limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On March 23, 1978 Westinghouse informed the Nuclear Regulatory Commission (NRC) that an error had been discovered in the fuel rod heat balance equation which resulted from the incorrect use of only half of the volumetric heat generation due to metal-water reaction in calculating the cladding temperature. Thus, the LOCA analyses previously submitted to the Commission by licensees of Westinghouse reactors were in error. The error identified would result in an increase in calculated peak clad temperature, which, for some plants, could result in calculated temperatures in excess of 2200°F unless the allowable peaking factor was reduced somewhat. Westinghouse identified a number of other areas in the approved model which Westinghouse indicated contained sufficient conservatism to offset the calculated increase in peak clad temperature resulting from the correction of the error noted above. Four of these areas were generic, applicable to all plants, and a number of others were plant specific. As outlined in the NRC Staff's Safety Evaluation Report (SER) of April 18, 1978 (attached). the staff determined that some of these modifications would be appropriate to offset to some extent the penalty resulting from correction of the error. The attached SER of April 18, 1978 sets forth the value for each modification applicable to each facility.

As part of the proposed change to the technical specifications the licensee has submitted information and analyses to permit Cycle 8 operation with reshuffled Westinghouse fuel and with 32 Westinghouse fuel assemblies replaced with fresh fuel assemblies manufactured by the Exxon Nuclear Company (ENC) and loaded on the periphery of the core. Based on an analysis of the information presented by the licensee, the staff has concluded that the new fuel manufactured by Exxon Nuclear Company (ENC) is both similar to and compatible with the fuel previously supplied by Westinghouse. The ENC calculations for the ENC fuel for the Ginna Core are not affected by the Westinghouse error. (Safety Evaluation for the reload application dated May 1, 1978). The staff's evaluation determined that the impact of correcting the Westinghouse Zirc [illegible] cladding temperature for the [illegible] margin (22[illegible]°F) to the 2200°F acceptance criteria limit. The NRC Staff has confirmed that the impact of correcting the error in the Westinghouse ECCS evaluation model as it relates to the use of Westinghouse fuel is conser[illegible] April [illegible] Safety Evaluation Report.

Although revised computer calculations correcting the error, noted above, and incorporating the modifications described in the Staff's April 18, 1978 SER have not been run for each plant, the various parametric studies that have been made for various aspects of the approved Westinghouse model over the course of time provide a reasonable basis for concluding that when final revised calculations for the facility are submitted using the revised and corrected Westinghouse model, they will demonstrate that operation will conform to the criteria of 10 CFR 50.46(b), when operated at the peaking factors set forth in the SER of April 18, 1978. Such revised calculations fully conforming to 10 CFR 50.46 are to be provided for the facility as soon as possible.

Operation of the facility would nevertheless be technically in non-conformance with the requirements of §50.46, in that specific computer runs for the particular facility employing revised models with the Westinghouse metal-water error corrected and with the proposed model changes considered, as a complete entity will not be complete for some time. However, operation as proposed in the licensee's application dated January 6, 1978, and at the peaking factor limit specified in this Exemption will assure that the ECCS system will conform to the performance criteria of §50.46. Accordingly, while the actual computer runs for the specific facility are carried out to achieve full compliance with 10 CFR §50.46, operation of the facility will not endanger life or property or the common defense and security.

In the absence of any safety problem associated with operation of the facility during the period until the computer computations are completed, there appears to be no public interest consideration favoring restriction of the operation of the captioned facility. Accordingly, the Commission has determined that an exemption in accordance with 10 CFR §50.12 is appropriate. The specific exemption is limited to the period of time necessary to complete computer calculations.

IV.

Copies of the Safety Evaluation Report dated April 18, 1978, and the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D. C. 20555, and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

(1) Licensee submittals dated April 7, 1977, January 6, 1978, and April 25, 1978.

(2) Amendment No. 19 to License No. DPR-18 and the related Safety Evaluation for the reload application, and

(3) This Exemption in the matter of RE Ginna Nuclear Power Plant.

Wherefore, in accordance with the Commission's regulations as set forth in 10 CFR Part 50, the licensee is hereby granted an exemption from the requirements of 10 CFR §50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without errors discussed herein. This exemption is conditioned as follows:

(1) As soon as possible, the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, and approved by the NRC staff and corrected for the errors described herein.

(2) Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_Q) for the facility shall be limited to 2.32.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Attached:
Safety Evaluation Report,
dated April 18, 1978

Dated at Bethesda, Maryland
this 1st day of May, 1978

April 18, 1978

Safety Evaluation Report

Error in Westinghouse ECCS Evaluation Model

Introduction

Westinghouse was informed on March 21, 1978 by one of their licensees that an error had been discovered in their ECCS Evaluation Model. This error was common to both the blowdown and heatup codes. Westinghouse determined by analyses that the fuel rod heat balance equation in the LOCTA IV & SATAN VI codes was in error and that the LOCA analyses previously submitted by their customers were incorrect and predicted PCT's which were too low. Westinghouse determined that only half of the volumetric heat generation due to metal-water reaction was used in calculating the cladding temperatures and that an unreviewed safety question existed since preliminary estimates indicated that some plants would not meet the 2200^{o}F limit of 10 CFR 50.46 without a reduction in overall peaking factor limit. Westinghouse notified their customers and NRC on March 23, 1978 while the utilities notified NRC through the regional I&E Offices.

Promptly upon notification by Westinghouse, the staff assessed the immediate safety significance of this information. The staff noted certain points that indicated no immediate action was required to assure safe operation of the plants. First, most plants operate at peaking factors significantly below the maximum peaking factor used for safety calculations. By making safety computations at factors higher than actual operating levels, the facility has a wide range of flexibility, without the need for hour to hour recomputations of core status. The difference between the actual peaking factors and the maximum calculated peaking factors, for most plants, would offset the penalty resulting from the correction of the error. Second, for most reactors there are plant-specific parameters which bear upon aspects of the ECCS performance calculations. Utilities do not generally take credit for these plant-specific parameters, preferring to provide a simpler computation which conservatively disregards these individually small credits. Third, the error in the Westinghouse computations relates to the zirconium-water reaction heat source. This is an aspect of Appendix K, which is generally recognized to be very conservative. New experimental data indicate that the methods required by Appendix K appreciably over-estimate the heat source. Thus, while the error in fact entails a deviation from a specific requirement of Appendix K, it does not entail a matter of immediate safety significance.

Westinghouse continued to evaluate the impact of the error on previous plant specific LOCA analyses and performed scoping calculations, sensitivity studies and some plant specific reanalyses. In addition, Westinghouse investigated several modifications to the previously approved methods which if approved by the NRC staff would offset some of the immediate impact of the error on Technical Specifications limits and plant operating flexibility.

On March 29, 1978, Westinghouse and several of their customers met with members of the NRC staff in Bethesda. Westinghouse described in detail the origin of the error, explained how it affected the LOCA analyses, and how the error had been corrected and characterized its effect on current plant specific analyses. In order to avoid reduction in overall peaking factors (F_q), Westinghouse presented a description of three proposed ECCS-LOCA evaluation model modifications which would contribute a compensating reduction of PCT. They were characterized as follows:

1) Revised FLECHT 15 x 15 heat transfer correlation.

   This new reflood heat transfer correlation which had been recently developed and submitted by Westinghouse (Reference (1) was proposed as a replacement for the currently approved FLECHT correlation. To determine the benefit, the proposed correlation was incorporated into the LOCTA IV heatup code and was found to result in improved heat transfer during the reflood portion of the LOCA.

2) Revised Zircaloy Emissivity.

   Based on recent EPRI data (Reference 2), Westinghouse proposed to modify the presently approved equation for zircaloy cladding emissivity to a constant value of 0.9. The higher emissivity (previously below 0.8) provides increased radiative heat transfer from the hot fuel pin during the steam cooling period of reflood.

3) Post-CHF heat transfer.

   Westinghouse proposed to replace their present post-CHF transition boiling heat transfer correlation with the Dougall-Rohsenow film boiling correlation (Reference 3) which they stated was included in Appendix K to 10 CFR Part 50 as an acceptable post-CHF correlation.

These three model modifications were classified as generic, applicable to all plant analyses. Subsequently, as discussed below, these changes were rejected by the staff as providing generic benefit. However, a portion of the credit proposed by Westinghouse was approved by the staff to certain specific plants, which had provided specific calculations with the new 15 x 15 correlation. During the period March 29 to April 18, 1978, Westinghouse provided the staff with additional sensitivity analyses and plant specific analysis in which they evaluated the effects of some changes to plant-specific inputs in the LOCA analyses. These were as follows:

1. Assumed Plant Power Level

   A reduction of the plant power level assumed in the SATAN VI blowdown analyses from 102% of the Engineered Safeguards Design Rated Power (ESDR) level to 102% of rated power was proposed. Previously, analyses had been performed at approximately 4.5% over the rated power. This change was worth approximately 0.01 in F_Q, and is referred to as $\Delta FESDR$ in Table 1.

2. COCO Code Input

   A modification to the COCO code input (Reference 3) to more realistically model the painted containment walls was proposed. Since the paint on containment walls provides additional resistance to heat loss into the walls, the COCO code calculates an increase in containment back pressure, which results in a benefit to the calculated peak cladding temperature of 0 to 40°F, during the reflooding transient. The magnitude of the benefit is dependent on the type of plant and the heat transfer properties of the paint, and results in up to 0.03 benefit in F_Q, and is referred to as ΔF_{CP} in Table 1.

3. Initial Fuel Pellet Temperature

   A modification of the initial fuel pellet temperature from the design basis to the actual as-built pellet temperatures was proposed. In the present LOCA calculations, Westinghouse has assumed margins in the initial pellet temperature. The margin available in plant-specific ranges from 28°F to 55°F. Use of the actual pellet temperature rather than the assumed value results in a reduction in pellet temperature (stored energy) at the end of blowdown, as calculated by the SATAN code, of approximately 1/3 of the initial pellet temperature margin. Westinghouse has provided sensitivity analyses which indicate that a 37°F reduction in fuel pellet temperature at end of blowdown is worth approximately 0.1 in F_Q. This is referred to as ΔF_{PT} in Table 1.

4. Accumulator Water Volume Consideration

Westinghouse has evaluated the effect on ECCS performance of reducing the accumulator water volume, and has determined that for those plants for which the downcomer is refilled before the accumulators are emptied, there is a benefit in PCT. The sensitivity studies have indicated that this benefit in F_Q is plant-specific. This is referred to as ΔF_{ACV} in Table 1.

5. Steam Generator Tube Plugging Consideration

In previous analyses, Westinghouse has assumed values of steam generator tube plugging which were greater than the actual plant-specific degree of plugging. Sensitivity analyses submitted in Reference 4 were used to evaluate the benefit available by realistically representing the plant-specific data. For the plants affected, the benefit in PCT ranged from 7 to 66°F which was conservatively worth from 0.007 to .066 F_Q. This is referred to as ΔF_{SG} in Table 1.

Safety Evaluation

The information provided by Westinghouse was separated into two categories; the generic evaluation model modifications and the plant specific sensitivity studies and reanalyses. The NRC staff reviewed the peaking factor limits proposed by Westinghouse to verify their conservatism.

The metal-water reaction heat generation error in the Westinghouse ECCS evaluation model was evaluated by the staff to determine an appropriate interim penalty. Westinghouse provided two preliminary separate effects calculations which indicated that a maximum penalty of from 0.14 to 0.17 was appropriate to compensate for the model error. As indicated in Reference 5, the staff conservatively rounded up this penalty to 0.20.

As is noted above, Westinghouse had proposed several compensating generic changes in their evaluation model to offset any necessary reductions in peaking factor due to the error. These changes were assessed by the staff and as noted in Reference 5.

1) No credit was given at this time, for the changes in the post-CHF heat transfer correlation and new zircaloy emissivity data.

2) Partial credit (70%) would be given at this time for the use of the new 15 x 15 FLECHT correlation only for plants which had provided a specific calculation demonstrating that such credit was appropriate.

Based on this review the staff developed recommended interim peaking factor limits for all the operating plants and recommended that any other plant specific interim factors (benefits) not related to the generic review be considered separately. In addition, the staff reviewed plant specific reanalyses for DC Cook, Units 1 & 2, Zion, Units 1 & 2, and Turkey Point, Unit 3 which had corrected the error in metal water reaction. In these analyses the Dougall-Rohsenow and zircaloy emissivity credits were not considered, while the new 15 x 15 FLECHT correlation was included. The staff concluded that these reanalyses could serve as a basis for conservatively determining interim peaking factor limits for these plants.

For most of the operating plants the staff's generic review resulted in a lower allowable peaking factor than Westinghouse had proposed. However, in one case, Westinghouse had proposed more limiting peaking factors in order to prevent clad temperatures at the rupture node from exceeding 2200°F. The staff concluded that it would be properly conservative to use the minimum of these values.

Based on plant specific sensitivity studies, performed by Westinghouse, the licensees submitted requests for interim plant specific benefits. The staff reviewed these sensitivity studies and recommended that appropriate credits be accepted. The results of these analyses are shown in Table 1.

We informed each licensee by telephone on April 3, 1978, that he should administratively reduce his peaking factor limit from the limit contained in his Technical Specifications to the interim peaking factor limit contained in the right hand column of Table 1. In those cases where the limit in Table 1 is 2.32, this represents no change from the Technical Specifications limit. The peaking factor limit of 2.32 is generically supported and approved for Westinghouse reactors employing constant axial offset control operating procedures.[6]

For the reactor having an interim peaking factor limit of 2.31, we requested no further justification of the limit. This is because the generic analysis supporting the limit of 2.32 approaches the limit only at beginning of the first cycle. Since the affected reactors have operated past this point, it is clear that the maximum attainable peaking factor will be less than 2.32. While this margin has not been quantified, the staff is convinced it is substantially greater than the 0.01 for which we are requiring no additional justification from the plants with an interim limit of 2.31.

For the reactors with an interim limit less than 2.31, we requested that the licensee furnish administratively imposed procedures to replace Technical Specifications either:

1. To provide a plant specific constant axial offset control analysis of 18 cases of load following which would ensure that the interim limit would not be exceeded in normal operation of the power plant, or, at his option, if such analysis were unobtainable, inappropriate or insufficient,

2. To institute procedures for axial power distribution monitoring of the interim limit using a system designed for this purpose or manual procedures as indicated in Standard Technical Specifications 3/4 2.6 and ancillary Specifications.

We requested the licensees to provide indication that they have adopted the above interim LOCA analyses, interim peaking factor limits and administrative procedures by April 10, 1978, if their reactors were operating, and by April 17, 1978, if the reactors were not operating.

Conclusion

We conclude that when final revised calculations for the facility are submitted using the revised and corrected model, they will demonstrate that with the peaking factor set forth herein, operation will conform to the criteria of 10 CFR 50.46(b). Such revised calculations fully conforming to 10 CFR 50.46(b) are to be provided for the facility as soon as possible.

As discussed herein, the peaking factor limit specified in Table 1, in combination with any necessary operating surveillance requirements, will assure that the ECCS will conform to the performance requirements of 10 CFR 50.46(b). Accordingly, limits on calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling provide reasonable assurance that the public health and safety will not be endangered.

References

(1) R. S. Dougall, W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities", MIT Report 9079-26, September 1963.

(2) EPRI Report NP-525, "High Temperature Properties of Zircaloy-Oxygen Alloy", March 1977.

(3) WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version", February 1978.

(4) WCAP 8986 "Perturbation Technique for Calculating ECCS Cooling Performance", February, 1978.

(5) DSS SER "Metal-Water Reaction Heat Generation Error in Westinghouse ECCS Evaluation Model Computer Programs", Z. R. Rosztoczy to D. F. Ross/D. G. Eisenhut, 4/7/78.

(6) T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September, 1974.

TABLE 1
F_Q Analysis

	PCT °F	F_Q OLD	ΔF_T	ΔF_{ZrO_2}	ΔF_{FLECHT}	F_{PCT}	F_{SE}	$F_{Q,MIN}$	ΔF_{ESDR}	ΔF_{CP}	ΔF_{PT}	ΔF_{SG}	ΔF_{ACV}	F_Q LIMIT
2 Loop														
Pt. Beach 1	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.029	-	2.32
Pt. Beach 2	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.066	-	2.32
Ginna	1972	2.32	.26	-.2	-	2.32	2.32	2.32	-	-	-	.053	-	2.32
Kewaunee	2172	2.25	.03	-.2	.05	2.13	2.25	2.13	.01	.02	-	-	-	2.16 -
Prairie Island 1/2	2187	2.32	.01	-.2	.05	2.18	2.26	2.18	.01	.02	-	-	.03	2.24(+)
3 Loop														
North Anna	2181	2.32	.02	-.2	-	2.14	2.32	2.14	-	-	-	-	-	2.14
Beaver Valley	2041	2.32	.15	-.2	-	2.27	2.32	2.27	-	-	.036	-	-	2.31
Farley	1991	2.32	.24	-.2	-	2.32	2.32	2.32	.01	.005	-	-	-	2.32
Surry 1	2177	1.85	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Surry 2	2177	1.85	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Turkey Point 3	2019*	1.90	.14	0	-.03	2.01	2.05	2.01	-	-	-	.020	-	2.03
Turkey Point 4	2195	2.05	.00	-.2	.05	1.90	1.91	1.90	-	-	-	.01	-	1.91
4 Loop														
Indian Point 2	2086	2.32	.11	-.2	-	2.23	2.23	2.23	.01	-	-	-	-	2.24
Indian Point 3	2125	2.32	.07	-.2	.06	2.25	2.19	2.19	.01	-	.03	-	-	2.23
Trojan	1975	2.32	.26	-.2	-	2.32	2.32	2.32	.01	-	.037	-	-	2.32
Salem 1	2135	2.32	.06	-.2	-	2.18	2.32	2.18	.01	-	.024	-	-	2.21
Zion 1/2	2189**	2.07	-	0	-.03	2.04	-	2.04	-	-	-	-	-	2.04(+)
Cook 1	2161*	1.90	.03	0	-.03	1.90	1.98	1.90	-	-	-	-	-	1.90
Cook 2	2190*	2.10	.01	0	0	2.11	-	2.11	0	0	0	0	0	2.11

ΔF_T - Credit in F_Q for PCT margin to 2200°F limit.

ΔF_{ZrO_2} - Metal Water Reaction penalty on F_Q.

ΔF_{FLECHT} - Credit in F_Q for improvements to 15x15 FLECHT Correlation.

F_{PCT} - Staff estimated F_Q based on 2200°F PCT limit.

F_{SE} - Westinghouse proposed F_Q based on stored energy sensitivity studies.

*Denotes reanalysis at F_Q old value error corrected.

**Denotes reanalyses at F_Q old value, error corrected, accumulator Vol. Change of 100 ft^3, accumulator pressure of 650 psia

(+) These limits are applicable assuming licensee modifies accumulator conditions as appropriate. If not, Prairie Island 1/2 F_Q=2.21, Zion 1/2 F_Q=1.9

7590-01

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. 19 to Provisional Operating License No. DPR-18, issued 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. The amendment is effective as of its date of issuance.

The amendment changes the Appendix A Technical Specifications to support operation in Cycle 8 with reload fuel by Exxon Nuclear Company (ENC). This fuel has been designed by ENC to be compatible with the fuel supplied previously by Westinghouse. In addition, the amendment allows Technical Specification changes that are required for startup tests.

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.

Notice of proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 21, 1978 (43 FR 7275). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 Commission's Order for Modification of License dated August 27, 1976, (2) the application for amendment dated January 6, 1978, and supplements thereto dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, (3) Amendment No. 19 to License No. DPR-18, (4) the Commission's related Safety Evaluation, and (5) the Exemption related to the requirements of 10 CFR 50.46(a)(1) and the Safety Evaluation dated April 18, 1978, attached thereto. 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 (1), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, 1st day of May, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors