APR 2 3 1975

Docket No. 50-244

Rochester Gas & Electric Corporation
ATTN: Mr. Edward J. Nelson, President
89 East Avenue
Rochester, New York 14604

Gentlemen:

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The Commission has issued the enclosed Amendment No. 6 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant, Unit No. 1. This amendment includes Change No. 15 to the Technical Specifications and is in response to your request dated September 24, 1974.

This amendment clarifies the surveillance requirements during cold and refueling shutdown operating modes, removes some statements from the Technical Specifications that are no longer applicable, and corrects some references in the Technical Specifications.

The proposed change to the definition of a release of radioactive material that constitutes an abnormal occurrence has not been included in this amendment since Amendment No. 5 issued February 13, 1975, included a complete redefinition of abnormal occurrences.

Copies of the Safety Evaluation and the Federal Register Notice for Amendment No. 6 are also enclosed.

Sincerely,
Original signed by:
Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 6
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

CP
(1)

OFFICE	RL:OR#1	RL:OR#1	ELD	RL:OR		
SURNAME	TVWambach:lb	RAPurple	Ketchen	KRGoller		
DATE	3/18/75	3/11/75	4/21/75	3/ /75		

Form AEC-318 (Rev. 9-53) AECM 0240

☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 6
License No. DPR-18

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

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated September 24, 1974, 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; and

   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.

2. Accordingly, the license is amended by a change 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:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

"2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 15."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 15 to
Technical Specifications

Date of Issuance: APR 2 3 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 6

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

1. Replace pages 3.1-1, 3.1-20, 3.1-21, 3.10-10, 4.1-5, 4.1-8, 4.1-9, 4.1-10, 4.5-2, 4.5-3, 4.5-4, 4.5-5, 4.6-4, 4.7-1, 4.8-1, 4.12-3, 4.12-4, and 5.4-1 with the enclosed pages.

2. Delete the sentences in the Basis for Technical Specification 2.1 which read:

   "The fuel residence time for Cycle 3 is limited to 7×10^5 MWD to assure no additional fuel clad flattening without prior review by the Regulatory staff. Prior to 7×10^5 MWD, the licensee may provide the additional analyses required for operation beyond 7×10^5 MWD."

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Coolant System

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To specify those conditions of the Reactor Coolant System which must be met to assure safe reactor operation.

Specification:

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. At least one reactor coolant pump or the residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical and above 1% thermal power, at least one reactor coolant pump shall be in operation. 15

c. (i) Reactor power shall not be maintained above 130 MWT (8.5%) unless both reactor coolant pumps are in operation.

(ii) If either reactor coolant pump ceases operating, immediate power reduction shall be initiated under administrative control as necessary to reduce power

3.1.4 Maximum Coolant Activity

Specifications:

3.1.4.1 Whenever the reactor is critical or the reactor coolant temperature is greater than 500° F:

a. The total specific activity of the reactor coolant shall not exceed $84/\bar{E}$ μCi/gm, where $\bar{E}$ is the average beta and gamma energies per disintegration in Mev.

b. The I-131 equivalent of the iodine activity in the reactor coolant shall not exceed 3.0 μCi/gm.

c. The I-131 equivalent of the iodine activity on the secondary side of a steam generator shall not exceed 0.6 μCi/gm.

3.1.4.2 If any one of the activity limits in Specification 3.1.4.1,(a. and b.), are exceeded due to a power transient:

a. The activity shall be returned to within specification within 48 hours, or

b. The plant shall be brought to a hot shutdown condition within the next 4 hours. 15

Basis:

The total activity and iodine activity limits for the primary system correspond to operation with the plant design basis of 1% fuel defects. (1) Radiation shielding and the radioactive waste disposal systems were

designed for operation with 1% defects[2]. The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage (Specification 3.1.5.2). If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above 1% defects. Appropriate 15 action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the volume control tank gases to the gas decay tanks.

The specified activity limits provide protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the following analysis of a steam generator tube rupture accident.[3] The potential dose at the site boundary from this accident is larger and hence more limiting than the dose that would result from one year of operation with the maximum reactor coolant activity combined

control rod bank insertion is limited to the beginning-of-life limit with an inoperable rod, until the worth at greater insertion has been evaluated. | 15

References:

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

(2) FSAR, Section 7.3

(3) FSAR, Section 14.2.6

(4) Technical Supplement - Appendix A, Pg. 120

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 M*(3)	D(1) Q*(3)	B/W(2)	1) Heat balance calculation** 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 3) Upper & lower chambers for axial offset**
2.	Nuclear Intermediate Range	S(1)	N.A.	P (2)	1) Once/shift when in service 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 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R	M (1) (2)	1) Overtemperature-Delta T 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 2) Each six inches of rod motion when data logger 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

	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)		
	$\bar{E}$ Determination (2)	Monthly (6)		15
	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)		15
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 a full length rods	Each refueling shutdown	7	
6. Full Length Control Rod	Partial movement of all rods	Every 2 weeks (6)	7	15
7. Pressurizer Safety Valves	Set point	Each Refueling shutdown	4	
8. Main Steam Safety Valves	Set point	Each Refueling shutdown	10	
9. Containment Isolation Trip	Functioning	Each Refueling shutdown	5	
10. Refueling System Interlocks	Functioning	Prior to refueling operations	9.4.5	

TABLE 4.1-2 (Continued)

	Test	Frequency	FSAR Section Reference
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 concentration	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. Secondard Coolant Samples	Radioiodine concentration (4)	Weekly (5)(6)	15

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of 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 $\bar{E}$.. A radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10μCi/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 μCi/gm from the previous measured level.

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

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.

(6) Not required during a cold or refueling shutdown

(7) During a cold or refueling shutdown, primary coolant Gross Radioactivity will be determined weekly. 15

b. The system test will be considered satisfactory if control board indication and visual observations indicate that all valves have received the Safety Injection signal and have completed their travel. The proper sequence and timing of the rotating components are to be verified in conjunction with Section 4.6.1 b.

4.5.1.2 Containment Spray System

a. System tests shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves, in the spray supply lines, at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. The spray nozzles shall be checked for proper functioning at least every five years.

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.5.2 Component Tests

4.5.2.1 Pumps

a. Except during cold or refueling shutdowns the safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started at intervals not to exceed one month. The pumps shall be tested prior to startup if the time since the last test exceeds 1 month. 15

b. Acceptable levels of performance for the pumps shall be that the pumps start, operate, and develop the minimum discharge pressure listed in the table below:

PUMPS	RECYCLE FLOW RATE	DISCHARGE PRESSURE
Containment Spray Pumps	35 gpm	240 psig
Residual Heat Removal Pumps	200 gpm	140 psig
Safety Injection Pumps	50 gpm	1420 psig

4.5.2.2 Valves

a. The refueling water storage tank outlet valves shall be tested at intervals not to exceed one month.

b. Except during cold or refueling shutdowns the spray additive valves shall be tested at intervals not to exceed one month. With the pumps shut down and the refueling water 15

storage tank outlet valves closed, each valve will be opened and closed by operator action. This test shall be performed prior to startup if the time since the last test exceeds 1 month. | 15

c. The accumulator check valves shall be checked for operability during each refueling shutdown.

4.5.2.3 Air Filtration System

a. Measurement of the pressure drop shall be performed at the frequency specified for the integrated leak tests. Any observable leak paths or other significant difference in appearance or deviation of the pressure drop from initial conditions shall be corrected. In addition, in-place Freon testing shall be performed under ambient conditions with a filter efficiency not less than 99%.

b. The iodine removal efficiency of at least one charcoal filter cell shall be measured at the frequency specified for the integrated leakage rate tests. The filter cell to be tested shall be selected randomly from those cells with the longest in-bank residence time. The acceptance criteria for filter efficiency rate is 99.9% for removal of molecular iodine. If the acceptance criteria are not met, the activated charcoal in the installed filter units shall

c. Except during cold or refueling shutdowns the charcoal filter isolation valves shall be tested at intervals not greater than one month to verify operability and proper orientation. The valves shall be tested prior to startup if the time since the last test exceeds 1 month. 15

d. The HEPA filter banks shall be tested with thermally generated DOP at each refueling shutdown and indications of abnormal leakage corrected. The acceptance for filter efficiency is 99.0%.

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a Safety Injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The annual systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action

place the diesel generators themselves on test.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capability of the battery. As a check upon the effectiveness of the equalizing charge, the battery should be loaded rather heavily and the voltage monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively indicating replacement or maintenance.

The minimum permissible on-site fuel oil inventory, 10,000 gallons, is sufficient for operation under loss-of-coolant accident conditions of two engineered safety features trains for 48 hours, or for one train for 80 hours, or for operation under hot standby non-accident conditions for 111 hours.[2] | 15

References:

(1) FSAR, Section 8.2

(2) FSAR, Section 8.2.3

| 15

4.7 Main Steam Isolation Valves

Applicability

Applies to periodic testing of the main steam isolation valves.

Objective

To verify the ability of the main steam isolation valves to close upon signal.

Specification

The main steam isolation valves shall be tested at each refueling interval. Closure time of five seconds or less shall be verified. The valves are tested under no flow and at no load conditions.

Basis

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.

References:

FSAR - Section 10.4

FSAR - Section 14.2.5

4.8 Auxiliary Feedwater System

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specification

4.8.1 Except during cold or refueling shutdowns each motor driven auxiliary feedwater pump will be started at intervals not to exceed one month and a flow rate of 200 gpm established. | 15

4.8.2 Except during cold or refueling shutdowns the steam turbine driven auxiliary feedwater pump will be started at intervals not to exceed one month and a flow rate of 400 gpm established. | 15

4.8.3 Except during cold or refueling shutdowns the auxiliary feedwater pumps discharge valves shall be exercised at intervals not to exceed one month. | 15

4.8.4 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. These tests shall be performed prior to exceeding 5% power during a startup if the time since the last test exceeds one month. | 15

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet design. The flow rates will be measured at a simulated steam generator pressure

release rate of gaseous activity to be determined on an hourly basis by evaluation of the recorded data from the vent gas monitors to assure that the specified rates are not exceeded.

b. Plant records of vent releases of iodines and particulates with half-lives greater than eight days shall be maintained on the basis of all filter cartridges counted.

c. Gaseous release of tritium shall be calculated and recorded on a monthly basis from measured data.

4.12.2.2 Prior to all batch releases of gaseous wastes, samples shall be obtained and an isotopic analysis performed. Such batch releases should be controlled to minimize the radiological effects on the environs.

4.12.2.3 a. Samples from the air ejector shall be taken at least quarterly and an isotopic analysis performed. Samples are not required during a cold or refueling shutdown. 15

b. If the air ejector discharge monitor is inoperable, samples shall be taken and analyzed for gross activity and the ratio of long-lived (greater than 8 days half life) to short-lived activity determined at least 5 days per week and whenever the primary coolant gross activity or the unidentified

leak rate increases by a factor of 2. Samples are not required during a cold or refueling shutdown. | 15

c. When the secondary iodine concentration is greater than 25% of the limit in Specification 3.1.4.1.c, samples from the air ejector shall be taken at least weekly and whenever the gross activity release rate increases by 50% over the previous day's maximum hourly average release rate. An isotopic analysis shall be performed on each sample.

4.12.2.4 a. When the release rate of iodines and particulates with half-lives greater than eight days is less than 10% of the maximum release rate given in Specification 3.9.2.2, the filter cartridges shall be analyzed to determine the release rate at least weekly, otherwise the cartridges shall be removed for analysis daily until a steady release level has been established.

b. When the gross activity release rate exceeds 10% of the maximum release rate given in Specification 3.9.2.1 and the average daily gross activity release rate increases by 50% over the previous day, the cartridges shall be analyzed to determine the release rate increase for iodines and particulates.

5.4 Fuel Storage

Specification

5.4.1 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water.

5.4.2 The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The fuel is stored vertically in an array with the sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.90$ even if unborated water were used to fill the pit.

5.4.3 The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit. | 15

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

(CHANGE NO. 15 TO TECHNICAL SPECIFICATIONS)

ROCHESTER GAS AND ELECTRIC CORPORATION

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By letter dated September 24, 1974, Rochester Gas and Electric Corporation requested changes to the Technical Specifications appended to Provisional Operating License DPR-18, for the R. E. Ginna Nuclear Power Plant. The proposed amendment includes changes to some surveillance test requirements to clarify when they are applicable, deletions of some statements and phrases which are no longer applicable, a change in the limiting condition for operation for primary coolant activity, and some minor corrections of terminology.

Discussion

Rochester Gas and Electric Corporation has requested changes to surveillance requirements that provide clarification as to the mode of reactor operation during which they are applicable. They propose that the following tests be specifically identified as not being required during cold and refueling shutdowns:

(1) primary coolant radio-chemical determination;

(2) primary coolant gross radio-iodine determination;

(3) primary coolant $\bar{E}$ determination;

(4) primary coolant Tritium determination;

(5) full length control rod partial movement;

(6) secondary coolant radioiodine concentration;

(7) main condenser air ejector effluent samples;

(8) starting tests of safety injection pumps, residual heat removal pumps and containment spray pumps;

(9) tests of the containment spray additive valves and charcoal filter isolation valves;

(10) testing of the auxiliary feedwater system.

They propose sampling the primary coolant once per week during cold or refueling shutdown for gross radioactivity concentration determination. They also propose that a specific exemption from the requirement of heat balance calibrations of the nuclear power range channels and upper and lower chamber axial offset calibrations during hot, cold, or refueling shutdowns be stated in the Technical Specifications. They also propose that the limiting condition for operation for primary coolant activity be changed to require that any transient activity level associated with a power level change be within the steady-state limits within 48 hours or the reactor be placed in hot shutdown.

Other corrections and clarifications proposed included:

(1) deletion of statements involving the limit for fuel residence time in Cycle 3;

(2) deletion of phrases involving pre-operational conditions;

(3) correction of the basis for control rod insertion limit to make it consistent with the specification that was issued with Change No. 9 to the Technical Specifications;

(4) clarification of the basis for diesel oil storage requirements to make it conform with the FSAR;

(5) changing the name of the main steam stop valves to main steam isolation valves in the Technical Specifications for consistency with the terminology used in the FSAR and at the plant.

Evaluation

During cold and refueling shutdowns, detailed analyses of the primary coolant radioactive concentrations are not required since the generation of new contamination of the primary coolant from the fission process is greatly reduced and the primary coolant is depressurized thereby removing the driving forces that would be needed to transport any primary coolant release to the public. Therefore, the probability and consequences of a release of primary coolant under these conditions are both extremely reduced, and there is no decrease in a margin of safety. The weekly sample for the determination of gross radioactive concentrations of the coolant is sufficient to monitor any changes that may result from clad damage to deterioration during shutdown. There is also continuous monitoring by installed instrumentation that will sound an alarm upon detection of high primary coolant activity.

The only source of radioactive contamination of the secondary coolant is leakage through the steam generators. During cold or refueling shutdown, the primary side of the steam generators is depressurized and no leakage would occur. Therefore, no sampling of the secondary coolant is required during cold or refueling shutdowns. The air ejector, which exhausts non-condensibles from the secondary coolant system, is not operating during cold and refueling shutdowns and sampling is not required.

Periodic partial movement of the full length control rods assures that the rods are mechanically free and operable for insertion into the core if required. During cold and refueling shtudown, the full length rods are fully inserted in the core and therefore this test is unnecessary.

The safety injection system and the containment spray system are not required to be operable during cold or refueling shutdowns. These systems are tested during shutdown because it is not possible to perform the tests while the plant is operating. During plant power operation individual component tests are performed to provide the necessary assurance of reliability. These component tests, (pump starting and valve stroking) are not required during cold or refueling shutdowns. We have modified the licensee's proposal to the extent that, if the pumps and valves have not been operated within the last month prior to a start-up, the technical specifications, as changed, require them to be operated prior to criticality. This change as modified ensures that the probability and consequences of the accidents previously analyzed are not increased, and there is no significant decrease in a margin of safety.

The auxiliary feedwater system is not required to be operable during cold or refueling shutdowns since the secondary system is not operating and cooling is being performed by the Residual Heat Removal System. Therefore, tests of the auxiliary feedwater system are not required in the cold or refueling shutdown modes. We have modified the licensee's proposal to include the requirements that, if the auxiliary feedwater system has not been operated within the last month prior to resumption of power operation, the technical specifications, as changed, require the tests to be performed prior to exceeding 5% of rated power. Less than 5% power is sufficient to provide the steam to operate the steam-driven auxiliary feedwater pump. This change, as modified, ensures that the probability and consequences of accidents previously analyzed are not increased, and there is no significant decrease in a margin of safety.

The nuclear power channels cannot be calibrated when the reactor is in the cold refueling, or hot shutdown modes because there is insufficient neutron flux to provide a signal in these channels. In addition, because these channels are not required for reactor protection during these modes, the surveillance tests are not required. Accordingly, there is no increase in the probability or consequences of an accident and no decrease in a margin of safety.

They are also consistent with our interpretation of the R. E. Ginna Technical Specifications in the past. These changes are being made to specifically state the allowed exclusions and thereby avoid any misinterpretation of the Technical Specification requirements.

The proposed change to the primary coolant activity limiting condition for operation is more restrictive than the previous technical specification. Whereas the previous specification required that within 48 hours a determination had to be made that the activity was returning to within the steady-state limit, the proposed specification requires that within 48 hours the activity must be within the steady-state limit. Also the proposed specification requires putting the plant in the hot shutdown mode if the limits are not met whereas the previous specification gave allowance for other corrective measures to be attempted. We have modified the proposed change by placing a 4 hour time limit on reaching the hot shutdown mode.

The remaining changes are editorial in nature and do not have any significance with regard to safety.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: APR 2 3 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 6 to Provisional Operating License No. DPR-18 issued to Rochester Gas and Electric Corporation which revised Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant located in Wayne County, New York. The amendment is effective as of its date of issuance.

This amendment clarifies the surveillance requirements during cold and refueling shutdown operating modes, removes some statements from the Technical Specifications that are no longer applicable, and corrects some references in the Technical Specifications.

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 is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated September 24, 1974, (2) Amendment No. 6 to License No. DPR-18, with Change No. 15, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection

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at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the Lyons Public Library, 67 Canal Street, Lyons, New York 14489 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 Reactor Licensing.

Dated at Bethesda, Maryland, this APR 2 3 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

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