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Docket No. 50-244

Rochester Gas & Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14604

Gentlemen:

The Commission has issudd the enclosed Amendment No. 14 to Provisional Operating License No. DPR-18. This amendment consists of changes to the Technical Specifications in response to your requests dated March 10, 1975, and February 1, 1977.

This amendment revises the Technical Specifications to clarify the surveillance specification for diesel-generator starting and breaker closing times under test conditions, to establish specifications for equipment designed to mitigate the consequences of flooding of safety-related equipment due to the failure of non-seismic piping and to establish specifications for safety-related shock suppressors (snubbers).

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

/S/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 14 to DPR-18
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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NRC FORM 318 (9-76) NRCM 0240

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

June 1, 1977

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ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14604

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See next page

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUOLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 14
License No. DPR-18

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

   A. The applications for amendment by Rochester Gas and Electric Corporation (the licensee) dated March 10, 1975, and February 1, 1977, comply 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 applications, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 14 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 1, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 14

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

Remove pages i, ii, 3.5-3, 3.5-4, 4.1-9, 4.6-1, and 4.6-2 and insert revised identically numbered pages.

Add pages: 3.5-4a
3.13-1
3.13-2
3.13-3
3.13-4
3.13-5
3.13-6
4.14-1
4.14-2
4.14-3
4.14-4
4.14-5

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		Page
4.0	SURVEILLANCE REQUIREMENTS	4.1-1

TABLE 3.5-1
INSTRUMENT OPERATION CONDITIONS

NO.	FUNCTIONAL UNIT	1 NO. of CHANNELS	2 NO. of CHANNELS TO TRIP***	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 5 CANNOT BE MET
1.	Manual	2	1	1	-*		Maintain hot shutdown
2.	Nuclear Flux Power Range					2 of 4 power range channels greater than 10% F.P. (low setting only)	Maintain hot shutdown
	low setting	4	2	3	2		
	high setting	4	2	3	2		
3.	Nuclear Flux Intermediate Range	2	1	1	-*	2 of 4 power range channels greater than 10% F.P.	Maintain hot shutdown. Note 1
4.	Nuclear Flux Source Range	2	1/	1	-*	1 of 2 intermediate range channels greater than 10^{-10} amps.	Maintain hot shutdown. Note 1
5.	Overtemperature Δ T	4	2	3	2		Maintain hot shutdown
6.	Overpower Δ T	4	2	3	2		Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2		Maintain hot shutdown
8.	Hi Pressurizer Pressure	3	2	2	1		Maintain hot shutdown
9.	Pressurizer-Hi Water Level	3	2	2	1		Maintain hot shutdown
10.	Low Flow in one loop (≥ 50% F.P.)	3/loop	2/loop (any loop)	2	1		Maintain hot shutdown
	Low Flow both Loops (10-50% F.P.)	3/loop	2/loop (any loop)	2	1		Maintain hot shutdown

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

NO. FUNCTIONAL UNIT	1 NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 5 CANNOT BE MET
17. Circulating Water Flood Protection a. Screenhouse	2	1	2^+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs. with two channels inoperable.
b. Condenser	2	1	2^+	_*		Power operation may be continued for a period of up to 7 days with 1 channel inoperable or for a period of 24 hrs. with two channels inoperable.

NOTE 1: When block condition exists, maintain normal operation.

F.P. = Full Power

* Not Applicable

** If both rod misalignment monitors (a and b) are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of rated power in addition to the increased surveillance noted.

*** If a functional unit is operating with the minimum operable channels, the number of channels to trip the reactor will be column 3 less column 4.

\+ A channel is considered operable with 1 out of 2 logic or 2 out of 3 logic.

TABLE 4.1-1 (Continued)

		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.	Secondary Coolant Samples	Radioiodine concentration(4)	Weekly (5)(6)	
19.	Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown	

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units 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 radioacticity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1.a or increases by

4.6 Emergency Power System Periodic Tests

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiated start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating. This test shall be conducted monthly on each diesel generator. Normal plant operation will not be affected.

b. Automatic start of each diesel generator and automatic restoration of particular vital equipment, initiated by an actual loss of all normal AC station service power supplies together with a simulated safety injection signal. This test shall be conducted during each refueling shutdown to assure that the diesel-generator will start and

and following maximum breaker closure times after the initial starting signal for trains A and B will not be exceeded.

	A	B
Diesel plus Safety Injection Pump plus RHR Pump	20 sec.	22 sec.
All breakers closed	40 sec.	42 sec.

c. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of stand-by service. The above tests will be considered satisfactory if all applicable equipment operates as designed.

d. Diesel generator electric loads shall not be increased beyond the long term rating of 1950 KW.

4.6.2 Diesel Fuel Tanks

A minimum oil storage of 10,000 gallons shall be at the station at all times.

4.6.3 Station Batteries

a. Every month the voltage of each cell (to the nearest 0.01 volt), the specific gravity and temperature of a pilot cell in each battery shall be measured and recorded.

b. Every 3 months the specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured and recorded.

3.13 **<u>Shock Suppressors (Snubbers)</u>**

Applicability:

Applies to the operability of all safety-related shock suppressors (snubbers) listed in Table 3.13-1.

Objective:

To specify the requirements for operability of shock suppressors (snubbers).

Specification:

3.13.1 The reactor shall not be made critical unless (1) all shock suppressors (snubbers) listed in Table 3.13-1 are operable.

3.13.2 Continued hot shutdown or power operation is permitted for a period up to 72 hours without the conditions of 3.13.1 being met. If the conditions of 3.13.1 are not met in that 72 hour period, then the reactor shall be in a cold shutdown condition within the next 36 hours.

Basis

Shock suppressors (snubbers) are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

Table 3.13-1

Safety Related Shock Suppressors (Snubbers)

Snubber Mark No.	Piping System	Accessible or Inaccessible (A or I)	High Radiation** Zone	Especially Difficult to Remove
SGA-1	"A" Steam Generator	I	Yes	Yes
SGA-2	"A" Steam Generator	I	Yes	Yes
SGA-3	"A" Steam Generator	I	Yes	Yes
SGA-4	"A" Steam Generator	I	Yes	Yes
SGA-5	"A" Steam Generator	I	Yes	Yes
SGA-6	"A" Steam Generator	I	Yes	Yes
SGA-7	"A" Steam Generator	I	Yes	Yes
SGA-8	"A" Steam Generator	I	Yes	Yes
SGB-1	"B" Steam Generator	I	Yes	Yes
SGB-2	"B" Steam Generator	I	Yes	Yes

Table 3.13-1 (Continued)

Safety Related Shock Suppressors (Snubbers)

Snubber Mark No.	Piping System	Accessible or Inaccessible (A or I)	High Radiation** Zone	Especially Difficult to Remove
SGB-3	"B" Steam Generator	I	Yes	Yes
SGB-4	"B" Steam Generator	I	Yes	Yes
SGB-5	"B" Steam Generator	I	Yes	Yes
SGB-6	"B" Steam Generator	I	Yes	Yes
SGB-7	"B" Steam Generator	I	Yes	Yes
SGB-8	"B" Steam Generator	I	Yes	Yes
FW-3	"A" Feedwater	A	No	No
FW-5	"A" Feedwater	I	No	Yes
FW-25	"A" Feedwater	A	No	No

Table 3.13-1 (Continued)

Safety Related Shock Suppressors (Snubbers)

Snubber Mark No.	Piping System	Accessible or Inaccessible (A or I)	High Radiation** Zone	Especially Difficult to Remove
AFW-25	"A" Aux. Feedwater	A	No	No
AFW-26	"A" Aux. Feedwater	A	No	No
AFW-27	"A" Aux. Feedwater	A	No	No
AFW-28	"A" Aux. Feedwater	A	No	No
AFW-29	"A" Aux. Feedwater	A	No	No
AFW-49	"A" Aux. Feedwater	A	No	No
FW-9	"B" Feedwater	I	No	Yes
FW-30	"B" Feedwater	A	No	No
FW-80	"B" Feedwater	A	No	No
FW-81	"B" Feedwater	A	No	No
FW-82	"B" Feedwater	A	No	No

Table 3.13-1 (Continued)

Safety Related Shock Suppressors (Snubbers)

Snubber Mark No.	Piping System	Accessible or Inaccessible (A or I)	High Radiation** Zone	Especially Difficult to Remove
FW-83	"B" Feedwater	A	No	No
AFW-10	"B" Aux. Feedwater	A	No	No
AFW-13	"B" Aux. Feedwater	A	No	No
MS-2	"A" Main Steam	I	No	Yes
MS-3	"A" Main Steam	I	No	Yes
MS-22	"A" Main Steam	A	No	No
MS-7	"B" Main Steam	I	No	Yes
MS-8	"B" Main Steam	I	No	Yes
MS-146 (Top)	"B" Main Steam	A	No	No
MS-146 (Bottom)	"B" Main Steam	A	No	No

Table 3.13-1 (Continued)

Safety Related Shock Suppressors (Snubbers)

Snubber Mark No.	Piping System	Accessible or Inaccessible (A or I)	High Radiation** Zone	Especially Difficult to Remove
MS-147	"B" Main Steam	A	No	No
MS-148	"B" Main Steam	A	No	No
MS-159	"B" Main Steam	A	No	No
MS-160	"B" Main Steam	A	No	No
H-1	PRZR Relief	I	Yes	No
H-2	PRZR Relief	I	Yes	No
H-3	PRZR Relief	I	Yes	No
H-4	PRZR Relief	I	Yes	No
H-5	PRZR Relief	I	Yes	No
H-6	PRZR Relief	I	Yes	No
H-7	PRZR Relief	I	Yes	No
H-8	PRZR Relief	I	Yes	No

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.13-1 provided that a proposed revision to Table 3.13-1 is included with the next License Amendment request.

** Modifications to this table due to changes in high radiation areas (during shutdown) shall be submitted to the NRC as part of the next License Amendment request.

4.14 Shock Suppressors (Snubbers)

Applicability:

Applies to the operability of all shock suppressors (snubbers) listed in Table 3.13-1.

Objective:

To specify the requirements for inspection and functional testing of shock suppressors (snubbers).

Specifications:

Visual Inspections:

4.14.1 Each hydraulic shock suppressor (snubber) with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment, shall be determined operable by visual inspection in accordance with the inspection schedule in Table 4.14-1. Initiation of the Table 4.14-1 inspection schedule shall be made assuming the unit was previously at the 12 month inspection interval. The initial inspection shall be performed within 12 months of the effective date of this specification.

4.14.2 Each hydraulic shock suppressor (snubber) with seal material not fabricated from ethylene propylene or other material demonstrated compatible with the operating environment shall be determined operable at least once every 31 days by a visual inspection of the snubber.

4.14.3 The inspection of the hydraulic shock suppressors (snubbers) in 4.14.1 and 4.14.2 shall be a visual inspection and shall include inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to the piping or steam generators and anchors.

4.14.4 Shock suppressors (snubbers) in 4.14.1 may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon accessibility for inspection during reactor operation. These two groups may be inspected independently but in accordance with the applicable schedule.

4.14.5 Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

4.14.6 A hydraulic shock suppressor (snubber) is considered to be operable if the fluid reservoir level and the installation are in accordance with the manufacturer's instructions.

Functional Testing:

4.14.7 During each refueling outage, but at least once during each 18 month interval a representative sample of at least 10 hydraulic shock suppressors (snubbers) or 10%, whichever is less, of all shock suppressors (snubbers) listed in Table 3.13-1, shall be selected and functionally tested. These hydraulic shock suppressors (snubbers) shall be tested in accordance with the original manufacturer's test requirements. Hydraulic shock suppressors (snubbers) greater than 50,000 lbs. capacity may be excluded from functional testing requirements. Hydraulic shock suppressors (snubbers) selected for functional testing shall be selected on a rotating basis. Hydraulic shock suppressors (snubbers) identified as either being "especially difficult to remove" or in "high radiation zones" may be exempted from functional testing provided they were demonstrated operable during previous functional tests. Hydraulic shock suppressors(snubbers) found inoperable during functional testing shall be restored to operable status prior to being returned to service. For each hydraulic shock suppressor (snubber) found inoperable during these functional tests, an additional minimum of 10% of all hydraulic shock suppressors (snubbers) or 10, whichever is less, shall also be functionally tested until no more failures are found or all hydraulic shock suppressors (snubbers) have been functionally tested.

Basis:

The hydraulic shock suppressors (snubbers) are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety systems is maintained during and following a seismic or other event initiating dynamic loads. The only hydraulic shock suppressors (snubbers) excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

The inspection frequency applicable to hydraulic shock suppressors (snubbers) containing seals fabricated from materials which have been demonstrated compatible with their operating environment is based upon maintaining a constant level of protection. Therefore, the required inspection interval varies inversely with the observed failures. The number of inoperable hydraulic shock suppressors (snubbers) found during an inspection determines the time interval for the next required inspection. To provide further assurance of hydraulic shock suppressor (snubber) reliability, a representative sample of the installed hydraulic shock suppressors (snubbers) will be functionally tested during each refueling shutdown but at least every 18 months. Observed failures of these samples will require functional testing of additional units. To minimize personnel exposures, hydraulic shock suppressors (snubbers) installed in high radiation zones or in especially difficult to remove locations may be exempted from functional testing requirements provided the operability of hydraulic shock suppressors (snubbers) was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

Table 4.14-1

Shock Suppressor (Snubber) Inspection Schedule

Number of shock suppressors (snubbers) found inoperable during inspection or during inspection interval	Next required inspection interval
0	18 months $\pm$ 25%
1	12 months $\pm$ 25%
2	6 months $\pm$ 25%
3 or 4	124 days $\pm$ 25%
5, 6, or 7	62 days $\pm$ 25%
$\geq$ 8	31 days $\pm$ 25%

Note: The above required inspection interval shall not be lengthened more than one step at a time.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA ATOMIC POWER STATION

DOCKET NO. 50-244

Introduction

By letters dated March 10, 1975, and February 1, 1977, Rochester Gas and Electric Corporation (the licensee) requested amendments to License No. DPR-18 to revise the R. E. Ginna Plant (the facility) Technical Specifications. These amendment requests propose changes to clarify the surveillance specification for diesel-generator starting and breaker closing times under test conditions, to establish Technical Specifications for equipment designed to mitigate the consequences of flooding of safety-related equipment due to the failure of non-seismic piping and to establish Technical Specifications for hydraulic shock suppressors (snubbers).

Discussion

Diesel-Generator Testing

The March 10, 1975 request for license amendment proposed changes to the Technical Specifications to clarify the required start and breaker closure times for diesel-generator trains A and B under a simulated safety injection signal and proposed a specification for maximum closure times for "all breakers closed."

Flood Protection Equipment Testing

The licensee's request for license amendment dated March 10, 1975, also proposes Technical Specification surveillance requirements for the flood protection related circulating water pump trip equipment.

Shock Suppressors (Snubbers)

The licensee's request for license amendment dated February 1, 1977, proposes Technical Specifications for Hydraulic Snubbers based on NRC Model Technical Specifications.

Evaluation

Diesel-Generator Testing

The Engineered Safety Features Actuation Sequence specified in Table 8.2-4 of the Final Safety Analysis Report (FSAR) provides the design basis for the delay time for residual heat removal (RHR) pumps and for safety injection (SI) pumps reaching full rated flow when powered by the emergency diesel-generators. A 28 second total time delay in reaching rated flow for these pumps was assumed in the Emergency Core Cooling System Analysis. Section 6.2.3 of the FSAR provides the event sequence which constitutes this 28 second delay. Other required loads automatically start later in the loading sequence for the diesel-generator. Those other loads include the service water pumps (total delay of 33 seconds), the containment fans (43 seconds) and the Auxiliary Feedwater Pumps (48 seconds).

The current Technical Specifications require that "the diesel-generator start and assume the required load in less than 30 seconds after the initial starting signal." The proposed change would specify the allowable time delay for each emergency diesel-generator train. The licensee included a period of 5 seconds from breaker closure time to reaching rated flow for each of these pumps. The 28 second total delay time assumed in the ECCS analysis for the RHR and SI pumps to reach rated flow also includesa 1 second time delay to initiate a Safety Injection Signal (SIS) and to account for instrument time delay. Subtracting the 1 second and 5 second portions noted above, leaves 22 seconds from actuation of a SIS until the train B RHR pump breaker must close. (Train A RHR Pump breaker must close in 20 seconds). The licensee has therefore proposed diesel start and SI and RHR pump breaker closure times of 20 seconds and 22 seconds for Train A and Train B respectively. Since these times are those assumed in the previous acceptable safety analysis, we find this Technical Specification change to be acceptable.

The licensee has proposed to include all other safety load circuit breakers which must automatically close during the diesel-generator loading sequence following an SIS. FSAR Table 8.2-4, the diesel-generator loading sequence, identifies the auxiliary feedwater pump for each train as the last load required to be automatically started. Considering the 5 second delay from breaker closure to reaching full rated flow, the times for 1A and 1B Auxiliary Feed Pumps breaker closure are 40 seconds and 42 seconds respectively. Since these times are the same as those specified in the FSAR, we find this revision of this specification acceptable also.

Flood Protection Equipment

The licensee has made modifications to prevent the loss of function of engineering safety features (safeguards) equipment due to flooding that could be caused by a circulating water pipe or expansion joint failure. Redundant water level instrumentation channels have been installed in the condenser pit and in the screen-house pit to automatically trip the circulating water (CW) pumps. Each redundant instrument channel consists of three float switch level detectors arranged in a two-out-of-three logic to actuate the CW pump trip circuitry. The circulating water pumps will be tripped if any two of three level switches at any of the four locations sense a water level two feet above their respective pit floor elevations.

The licensee's letter dated March 10, 1975, proposed surveillance Technical Specifications for circulating water flood protection equipment. The proposed surveillance frequency is each refueling shutdown. The inherent high reliability of the redundant float switch level detectors and the fail-safe character of the pump trip relays result in a highly reliable circulating water pump trip circuit. Additionally, independent trip circuits are provided for each potential flood location. Surveillance testing at each refueling shutdown, provides additional assurance of the continued operability of this flood protection equipment and is therefore acceptable.

The licensee initially proposed to perform a functional test on each circulating water pump trip channel. The licensee subsequently agreed to revise the proposal to a calibration of each channel. This would insure that the float switches properly actuate at 2 feet above their respective pit floors and that each channel is functional.

The proposed change to the Technical Specification for circulating water flood protection equipment did not include limiting conditions for operation. During discussions with the staff, the licensee also agreed to limiting conditions for operation (LCO's) for the circulating water flood protection equipment in Technical Specification Table 3.5.1. The LCO's would require redundant circulating water pump trip channels in the screen house pit and in the condenser pit to be operable. Power operation for a period of up to seven days with one redundant channel inoperable or for a period of up to 24 hours with two channels inoperable would be allowed. Additionally a channel would be considered operable if it functions on either a one-out-of-two logic or a two-out-of-three logic. The licensee's proposed LCOs for circulating water flood protection equipment will assure that (1) the circulating water pumps will be tripped as a result of flooding in either the condenser pits or the screen well house, (2) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (3) that the specified coincidence logic is maintained. Continued power operation for a period of up to seven days with one redundant channel inoperable or for a period of 24 hours with two redundant channels inoperable, provides a brief period for maintenance prior to requiring plant shutdown and is acceptable because of low probability of a circulating water pipe failure during the allowable maintenance period. Additionally, the operator has the capability to manually trip the circulating water pumps from the control room upon receipt of a flood alarm. In the event of a failure of one of three level detectors (float switches) in a single channel, the channel is considered operable if it will perform its safety function with a one-out-of-two logic using the remaining two operable level detectors (float switches). Sufficient detector redundancy is maintained using a one-out-of-two logic to insure that the channel is capable of performing its safety function considering a failure of the redundant channel. Therefore, channel operation on either a two-out-of-three or one-out-of-two tripping logic is acceptable.

Shock Suppressors (Snubbers)

Shock suppressors (snubbers) are installed at strategic locations to assist in maintaining the structural integrity of the reactor coolant system and other safety related systems against postulated seismic events or other events that could initiate an abnormal dynamic load. It is, therefore, necessary, that shock suppressors installed to protect such safety system piping and components be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective hydraulic shock suppressors at reactor facilities has shown that the high incidence of failures observed during the summer of 1973 were caused by severe degradation of seal materials and subsequent leakage of hydraulic fluid. The basic seal materials used in Bergen Paterson hydraulic shock suppressors were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations at operating reactor facilities. The molded polyurethane exhibited greater resistance to these conditions, however, it also may be unsuitable for application in higher temperature environments. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which give satisfactory performance under the most severe conditions expected in reactor installations.

An extensive seal replacement program has been carried out at many reactor facilities. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incidence of failures, some failures continue to occur. These failures have severally been attributed to faulty assembly and installation, loose fittings and connections and excessive pipe vibration. The failures have been observed in both PWRs and BWRs and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of hydraulic shock suppressor failures, we have concluded that operability and surveillance requirements for hydrualic shock suppressors should be included in the Technical Specifications in all reactor facilities, regardless of manufacturer.

Our October 2, 1973 letter required the licensee to initiate a monthly inspection of the R.E. Ginna hydraulic shock suppressors. No hydraulic shock suppressors were found inoperable during these monthly inspections.

The licensee's proposed Technical Specifications provide additional assurance of satisfactory shock suppressor operation and reliability. The proposed specifications require that shock suppressors be operable during reactor operation and prior to start up. Additionally, because protection is only required during low probability events, a period of 72 hours is allowed for repair or replacement of defective units before reactor shutdown must be initiated. The licensee also proposed that the Technical Specifications allow continued reactor operation beyond 72 hours after showing by analysis that the integrity of the system with one or more shock suppressors inoperable can be maintained under design loading conditions. This option for continued operation beyond 72 hours without specific NRC review and prior approval is not acceptable.

The licensee therefore concurred in deleting this provision for continued operation from the proposed Technical Specification.

The licensee's proposed surveillance provides assurance that safety-related shock suppressors remain operable. The inspection frequency is based on maintaining high reliability. Thus, the inspection interval varies inversely with observed failures. The longest inspection interval, after a record of no failures, is nominally 18 months. Experience at the R. E. Ginna facility and other operating facilities has shown that the proposed surveillance program will provide an acceptable level of hydraulic shock suppressor performance provided that the seal materials are compatible with the operating environment. Hydraulic shock suppressors containing seal material which has not been demonstrated to be compatible with the operating environment will continue to be inspected each 31 days. To further increase assurance of reliable performance, the licensee has proposed to functionally test 10 percent or 10, whichever is less, hydraulic shock suppressors at each refueling shutdown, not to exceed 18 months between tests.

The proposed Technical Specifications for hydraulic shock suppressors are consistent with our model Technical Specifications dated December 15, 1975, and are consistent with our Standard Technical Specifications for Westinghouse Pressurized Water Reactors dated May 15, 1976. With minor changes which were discussed with and agreed to by the licensee and for the reasons stated above, we have determined that the proposed Technical Specifications for hydraulic shock suppressors provide assurance of satisfactory performance and reliability and are therefore acceptable.

For the purpose of initiating the surveillance required by Technical Specification 4.14.1, the inspection schedule will be established assuming the unit was previously at the 12 month interval. The initial inspection will be performed within twelve months of the effective date of the license amendment implementing shock suppressor surveillance. This initial inspection interval is longer than the six month interval proposed in the licensee's February 1, 1977 letter which was based on our Model Technical Specifications. This longer initial inspection interval is acceptable based on over three years of monthly inspections of hydraulic shock suppressors without a failure. Additionally, the seal materials used in the hydraulic shock suppressors installed in the R. E. Ginna facility are considered to be compatible with the operating environment, based upon satisfactory performance during prior inspections.

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) 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: June 1, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 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 revises the Technical Specifications to clarify the surveillance specification for diesel-generator starting and breaker closing times under test conditions, to establish specifications for equipment designed to mitigate the consequences of flooding of safety-related equipment due to the failure of non-seismic piping and to establish specifications for safety-related shock suppressors (snubbers).

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 10, 1975 and February 1, 1977, (2) Amendment No. 14 to Provisional License No. DPR-18 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 1st day of June 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

PRELIMINARY DETERMINATI

NOTICING OF PROPOSED LICENSING AMENDMENT

Licensee: Rochester Gas and Electric Corporation - Ginna

Request for: Tech Spec change to clarify the loading time for the diesel generator test and to add surveillance requirements for the flooding level switches in the turbine building

Request Date: March 10, 1975

Proposed Action:
( ) Pre-notice Recommended
(X) Post-notice Recommended
( ) Determination delayed pending completion of Safety Evaluation

Basis for Decision: The loading times given in the proposed Technical Specification are consistent with what is presented in the FSAR. By putting them in the Tech Specs, it removes any confusion or misinterpretation of the testing requirements. The addition of surveillance requirements for the flooding level switches results from the addition of the level switches to the plant as a result of modifications previously approved by NRC.

CONCURRENCES:		DATE:
1.	T. V. Wambach TVW	3/31/75
2.	R. A. Purple	3/31/75
3.	K. R. Coller KRG	3/31
4.	Office of Executive Legal Director R. Culp	4/8/75