May 22, 1982

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Docket No. 50-244
LS05-82-05-050

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

Dear Mr. Maier:

SUBJECT: CONTAINMENT ISOLATION

The Commission has issued the enclosed Amendment No. 50 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment responds to your application notarized April 16, 1982 (submitted by letter dated April 22, 1982).

The amendment approves provisions which update the containment isolation valve Table 3.6-1, to reflect plant system modifications.

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

Sincerely,

Original signed by

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

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DSU USE(07)

Enclosures:
1. Amendment No. 50 to License No. DPR-18
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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JPG - per recommended addition to SER

OELD recommendation included in SER Section 2.3

OFFICE	DL: ORB #5	DL: ORB #5	OELD	DL: ORB #5	DL: AD/SA		
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DATE	5/21/82	5/21/82	5/21/82	5/22/82	5/22/82		

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 50
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 April 16, 1982 (transmitted by letter dated April 22, 1982), 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.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 50, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 22, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 50

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines which indicate the area of changes.

REMOVE	INSERT
3.6-4 through 3.6-8	3.6-4 through 3.6-8
3.6-10	3.6-10
--	3.6-11

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)
29	Fuel transfer tube	flange	NA	(1)	NA
100	charging line to "B" loop	CV 370B	NA	(2)	NA
101	SI Pump 1B discharge	CV 889B CV 870B	NA NA	(5) (5)	NA NA
102	Alternate charging to "A" cold leg	CV 383B	NA	(2)	NA
103	Construction Fire Service Water	welded flange	NA	MV 5129	NA
105	Containment Spray Pump 1A	CV 862A	NA	(3)	NA
106	"A" Reactor Coolant Pump (RCP) seal water inlet	CV 304A	NA	(2)	NA
107	Sump A discharge to Waste Holdup Tank	AOV 1728	60	AOV 1723	60
108	RCP seal water out and excess letdown to VCT	MOV 313	60	(4)	NA
109	Containment Spray Pump 1B	CV 862B	NA	(3)	NA
110	"B" RCP seal water inlet	CV 304B	NA	(2)	NA
110	SI test line	MV 879	NA	(5)	NA
111	RHR to "B" cold leg	MOV 720(20)	NA	(6)	NA
112	letdown to Non-regen. Heat exchanger	AOV 371	60	MV 204A MV 820 (14)(17)	NA
113	SI Pump 1A discharge	CV 889A CV 870A	NA NA	(5) (5)	NA NA
120	Nitrogen to Accumulators	CV 8623	NA	AOV 846	60
120	Pressurizer Relief Tank (PRT) to Gas Analyzer (GA)	AOV 539	60	MV 546(7)	NA

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)
121	Nitrogen to PRT	CV 528	NA	MV 547(8)	NA
121	Reactor Makeup water to PRT	CV 529	NA	AOV 508	60
121	Cont. Press. transmitter PT-945 (10)	PT 945	NA	MV 1819A	NA
121	Cont. Press. transmitter PT-946 (10)	PT 946	NA	MV 1819B	NA
123	Reactor Coolant Drain Tank (RCDT) to GA	AOV 1789	60	MV 1655(7)	NA
124	Excess letdown supply and return to heat exchanger	AOV 745 CV 743	60 NA	(11) (11)	NA NA
124	Post Accident air sample "C" fan	MV 1569 MV 1572	NA NA	MV 1571 MV 1574	NA NA
125	Component Cooling Water (CCW) from 1B RCP	MOV 759B	NA	(12)	NA
126	CCW from 1A RCP	MOV 759A	NA	(12)	NA
127	CCW to 1A RCP	CV 750A	NA	MOV 749A	60
128	CCW to 1B RCP	CV 750B	NA	MOV 749B	60
129	RCDT & PRT to Vent Header	AOV 1787 CV 1713	60 NA	AOV 1786	60
130	CCW to reactor support cooling	MOV 813	60	(19)	NA
131	CCW to reactor support cooling	MOV 814	60	(19)	NA
132	Depressurization at power	AOV 7970	60	AOV 7971	60
140	RHR pump suction from "A" Hot leg	MOV 701(20)	NA	(6)	NA
141	RHR-#1 pump suction from Sump B	MOV 850A(13)	NA	MOV 851A(13)	NA
142	RHR-#2 pump suction from Sump B	MOV 850B(13)	NA	MOV 851B(13)	NA
143	RCDT pump suction	AOV 1721	60	AOV 1003A AOV 1003B	60 60
201	Reactor Compart. cooling Unit A & B	MV 4757(16) MV 4636(16)	NA NA	(11) (11)	NA NA
202	"B" Hydrogen recombiner (pilot & main)	MV 1076B MV 1084B	NA NA	SOV IV-3B SOV IV-5B	NA Normally Closed NA Normally Closed

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)
203	Contain. Press. transmitter PT-947 & 948	PT 947 PT 948	NA NA	MV 1819C MV 1819D	NA NA
203	Post accident air sample to "B" fan	MV 1563 MV 1566	NA NA	MV 1565 MV 1568	NA NA
204	Purge Supply Duct	AOV 5870	5	AOV 5869	5
205	Hot leg loop sample	AOV 966C	60	MV 956D(14)	NA
206	Przr. liquid space sample	AOV 966B	60	MV 956E(14)	NA
206	"A" S/G sample	AOV 5735	60	MV 5733(7)	NA
207	Przr. Steam space sample	AOV 966A	60	MV 956F	NA
207	"B" S/G sample	AOV 5736	60	MV 5734(7)	NA
209	Reactor Compart. cooling Units A & B	MV 4758(16) MV 4635(16)	NA NA	(11) (11)	NA NA
210	Oxygen makeup to A & B recombiners	MV 1080A	NA	SOV IV-2A SOV IV-2B	NA Normally Closed NA Normally Closed
300	Purge Exhaust Duct	AOV 5878	5	AOV 5879	5
301	Aux. steam supply to containment	MV 6151	NA	MV 6165(15)	NA
303	Aux. steam condensate return	MV 6175	NA	MV 6152(15)	NA
304	"A" Hydrogen recombiner (pilot and main)	MV 1084A MV 1076A	NA NA	SOV IV-5A SOV IV-3A	NA Normally Closed NA Normally Closed
305	Radiation Monitors R-11, R-12 & R-10A Auto Inlet Isol.	AOV 1597	60	MV 1596	NA
305	R-11, R-12 & R-10A Outlet	CV 1599	NA	AOV 1598	60
305	Post Accident air sample (containment)	MV 1554 MV 1557 MV 1560	NA NA NA	MV 1556 MV 1559 MV 1562	NA NA NA
307	Fire Service Water	CV 9229	NA	AOV 9227	(18)

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)
308	Service Water to "A" fan cooler	MV 4627(16)	NA	(11)	NA
309	Leakage test depressurization	flange	NA	MOV 7445	NA Normally Closed
310	Service Air to Contain.	CV 7226	NA	MV7141	NA
310	Instrument Air to Contain.	CV 5393	NA	AOV 5392	60
311	Service Water from "B" fan cooler	MV 4630(16)	NA	(11)	NA
312	Service Water to "D" fan cooler	MV 4642(16)	NA	(11)	NA
313	Leakage test depressurization	flange	NA	MOV 7444	NA Normally Closed
315	Service Water from "C" fan cooler	MV 4643(16)	NA	(11)	NA
316	Service Water to "B" fan cooler	MV 4628(16)	NA	(11)	NA
317	Leakage test supply	flange	NA	MOV 7443	NA Normally Closed
318	Dead weight tester (decommissioned)	welded shut	NA	welded shut	NA
319	Service Water from "A" fan cooler	MV 4629(16)	NA	(11)	NA
320	Service water to "C" fan cooler	MV 4641(16)	NA	(11)	NA
321	A S/G Blowdown	AOV 5738	60	MV 5701(7)	NA
322	B S/G Blowdown	AOV 5737	60	MV 5702(7)	NA
323	Service Water from "D" fan cooler	MV 4644(16)	NA	(11)	NA
324	Demineralized water to Containment	CV 8419	NA	AOV 8418	60
332	Cont. Press. Trans. PT-944, 949 & 950	PT 944	NA	MV 1819G	NA
		PT 949	NA	MV 1819F	NA
		PT 950	NA	MV 1819E	NA
332	Leakage test and hydrogen monitor instrumentation lines	MV 7448	NA	cap	NA
		MV 7452	NA	cap	NA
		MV 7456	NA	cap	NA
		SOV 921	NA	(21)	NA
		SOV 922	NA	(21)	NA
		SOV 923	NA	(21)	NA
		SOV 924	NA	(21)	NA

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME.*(SEC)	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)
401	Main steam from A S/G	NA**	NA	NA	NA
402	Main steam from B S/G	NA**	NA	NA	NA
403	Feedwater line to A S/G	NA**	NA	NA	NA
404	Feedwater line to B S/G	NA**	NA	NA	NA
1000	Personnel Hatch	NA	NA	NA	NA
2000	Equipment Hatch	NA	NA	NA	NA

*The maximum isolation time does not include diesel start time.
**The MSIVs and feedwater isolation valves are not considered to be containment isolation valves. The containment boundary is the steam generator secondary side and tubes.

MV - Manual Valve
MOV - Motor Operated Valve
AOV - Air Operated Valve
CV - Check Valve
SOV - Solenoid Operated Valve

(10) The pressure transmitter provides a boundary.

(11) Normally operating incoming and outgoing lines which are connected to closed systems inside containment and protected against missiles throughout their length, are provided with at least one manual isolation valve outside containment. (FSAR 5.2.2 pg. 5.2.2-2).

(12) The single remotely controlled containment isolation valve is normally open and motor operated. The cooling water return line is not directly connected to the reactor coolant system and, should remain open while the coolant pump is running. A second automatic isolation barrier is provided by the component cooling water loop, a closed system. (FSAR 5.2.2 pg. 5.2.2-1a)

(13) See FSAR Table 5.2.2-1 and Figure 5.2.2-2. Sump lines are in operation and filled with fluid following an accident. Containment leakage testing is not required. The valves are subjected to RHR system hydrostatic test.

(14) Normally operating outgoing lines connected to the Reactor Coolant System are provided with at least one automatically operated trip valve and one manual isolation valve in series located outside the containment. In addition to the isolation valves, each line connected to the Reactor Coolant System is provided with a remote operated root valve located near its connection to the Reactor Coolant System. (FSAR 5.2.2 pg. 5.2.2-1)

(15) See FSAR Table 5.2.2-1 and Figure 5.2.2-17.

(16) The Service Water system operates at a pressure higher than the containment accident pressure and is missile protected inside containment. Therefore, these valves are used for flow control only and need not be leak tested.

(17) A manual valve outside containment in series with an automatic valve is provided for normally operating outgoing RCS lines (FSAR pg. 5.2.2-1).

(18) Fire Service Water will be used only to fight fires inside containment. AOV 9227 is closed during power operation. A containment isolation signal to automatically close this valve is not required because a spurious signal during a fire may be hazardous to personnel and may impede fire suppression activities.

(19) See FSAR Table 5.2.2-1 and Figure 5.2.2-16.

(20) Containment leakage testing is not required per L. D. White, Jr. letter to Dennis L. Ziemann, USNRC dated September 21, 1978.

(21) Acceptable isolation capability is provided for instrument lines by two isolation boundaries outside containment. One of the boundaries outside containment may be a Seismic Class 1 closed system which is subjected to Type C leak rate testing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 Background

On April 22, 1982 the licensee* submitted proposed changes to the Ginna Technical Specifications (Reference 1). These changes involved modifications to Technical Specification Table 3.6-1 which identifies containment isolation valves.

Containment isolation provisions for containment piping penetrations are found in General Design Criteria 54 through 57 of 10 CFR 50, Appendix A. Although the GDC provides examples of acceptable isolation provisions for containment piping penetrations, flexibility is permitted if acceptability can be found on "some other defined basis." The licensee's submittal proposes other defined bases for three piping penetrations into containment.

2.0 Evaluation

2.1 Penetration 318, Deadweight Tester

The deadweight tester is no longer used to calibrate pressurizer instrumentation. The 1/8" tube penetration has been decommissioned and is currently bounded by a screw-type cap inside containment and a closed manual valve outside containment. Local leak testing is currently performed on the valve outside containment.

The licensee has proposed removing the manual valve outside containment and replacing it with a screw type cap similar to the barrier found inside containment. The licensee's basis for this modification is that (1) a capped barrier is a less likely leak path than a manual valve and (2) capping both ends of the penetration eliminates the need for local leak testing.

The staff, however, is concerned over the use of screw type caps and the elimination of local leak testing. Elimination of the local leak tests prevents periodic verification that the pipe caps remain secure and intact.

Due to this concern the licensee has proposed cutting the 1/8" tube off at both ends of the containment penetration and then seal weld over the penetration can. This will essentially eliminate the piping penetration and provide a permanent leak tight barrier. We find this proposal acceptable.

* Rochester Gas and Electric Corporation

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2.2 Penetration 332, Leakage Test Instrumentation Lines

The three leakage test instrumentation lines that enter penetration 332 are only used for the integrated leak rate tests required by Appendix J to 10 CFR 50. During normal operation they are bounded by screw-type caps inside containment and normally closed manual valves outside containment.

The licensee has proposed modifying these lines to include installation of containment hydrogen monitors as required by NUREG-0737. Hydrogen monitor lines will branch off the leakage test instrumentation lines between the containment wall and the existing manual valves. The two barriers for the hydrogen monitor lines will consist of a normally closed remote manual isolation valve and the hydrogen monitor. The hydrogen monitors, which act as closed systems, are designed and installed to meet Seismic Category 1 criteria and will be leak tested to Appendix J Type C standards.

The modified configuration requires removal of the pipe caps inside containment. Caps will be placed on each of the leakage test instrumentation lines just outside of the manual valves. Caps will serve as the second barrier for the leakage test instrumentation lines.

The two isolation barriers for both the leakage test instrumentation lines and the hydrogen monitor lines are located outside containment. Plant safety is enhanced by placing both of the isolation barriers of the hydrogen monitors outside containment. Similarly both isolation barriers of the leakage test instrumentation lines are located outside containment so as not to interfere with operation of the safety related hydrogen monitor lines.

Since both of these lines have two containment isolation barriers and a basis exists to locate the isolation barriers outside containment, we find the licensee's proposal acceptable.

2.3 Penetration 307, Fire Service Water

The fire service water penetration currently has a check valve inside containment and an air operated valve outside containment that receives an automatic containment isolation signal. The outside valve has position indication and tamper indication in the control room and is remote manually controlled from the control room.

The licensee has proposed removing the automatic containment isolation signal from the normally closed outside valve. The valve is only open when firefighters are putting out fires inside containment. The licensee has noted that a spurious isolation signal generated due to a fire in containment could cause a loss of-fire water to the firefighters. The potential hazard to the firefighters is an unacceptable personnel risk and is a greater risk than that posed by a potential leakage path through two normally closed valves following a LOCA.

We concur with the licensee that an undue risk exists with respect to plant personnel. Based on our evalaution, we have determined that the control room position indication and tamper indication for the outside air operating valve along with the valve alignment checks provided to verify valve position, provide an acceptable defined basis for assuring containment isolation, in accordance with GDC 56, without the need for automatic containment isolation. On the basis, and with the control room valve position and tamper indication and the valve alignment checks, we find the licensee's proposal to remove automatic containment isolation for the fire service water line to be acceptable.

3.0 Summary

General Design Criteria 54 through 57 of 10 CFR 50, Appendix A describes the containment isolation provisions required of piping penetrations entering containment. Alternatives to that found in the GDC can be permissible if it can be found acceptable on some other defined bases.

We conclude that the licensee has provided acceptable bases for the proposed containment isolation modifications. Therefore we find the proposed modifications to the plant's Technical Specifications to be acceptable.

4.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types, increase in total amounts of effluents, or an increase in power level, and will not result in any significant environmental impact. Having made this determination, we have concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 Conclusion

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

6.0 Acknowledgments

This safety evaluation was prepared by Doug Pickett

7.0 References

1. Letter from LeBoeuf, Lamb, Leiby & MacRae (counsel for Rochester Gas & Electric Corporation) to U.S.N.R.C. (F. Frank Lyman to Harold R. Denton) dated April 22, 1982.

2. Robert E. Ginna Technical Specifications

3. General Design Criteria 54-57, 10 CFR 50, Appendix A

Date: May 22, 1982

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. 50 to Provisional Operating License No. DPR-18, to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant (facility) located in Wayne County, New York. This amendment is effective as of its date of issuance.

The amendment approves provisions which update the containment isolation valve Table 3.6-1, to reflect plant system modifications.

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

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

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For further details with respect to this action, see (1) the application for amendment notarized April 16, 1982 (transmitted by letter dated April 22, 1982, (2) Amendment No. 50 to License No. DPR-18, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 22nd day of May, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

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