UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 20, 1993

Docket No. 50-244

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Dr. Mecredy:

SUBJECT: ISSUANCE OF AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M77849)

The Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. The amendment is in partial response to your application dated October 15, 1990, as supplemented by letters of March 8, 1991, November 30, 1992, and April 5, 1993.

The amendment finds your proposed Technical Specification (TS) changes for leakage rate testing requirements, on certain containment isolation valves (CIVs) found in TS Table 3.6-1 (TS Table), acceptable. Steam generator blowdown and sample (SGBS) CIVs will be leakage rate tested to the original ASME, Section XI, Type C, leakage rate testing requirements of your approved Inservice Testing Program for Pumps and Valves (IST Program). Steam generator blowdown and sample CIVs do not require relief from the alternate method of leakage rate testing pursuant to 10 CFR Part 50, Appendix J, which is currently authorized for your IST Program.

Because SGBS CIVs are identified as the first isolation boundary outside containment and now use the steam generator tubes as the second isolation boundary, several SGBS manual valves (once considered as the second isolation boundary) are being removed from the TS Table. The SGBS manual valves therefore do not require leakage testing in accordance with 10 CFR Part 50, Appendix J.

Several reactor coolant system (RCS) manual valves (classified as CIVs) have been removed from the TS Table and replaced by other RCS air operated and relief valves. The TS Table changes resolve a discrepancy which exists between the TS Table and your approved IST Program. The RCS replacement CIVs perform the containment isolation functions required by General Design Criterion (GDC) 56 and are currently tested in accordance with 10 CFR Part 50, Appendix J.

All changes to the TS Table are consistent with your approved (IST) Program for Pumps and Valves, 1990-1999 Third 10-Year Interval.

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The remainder of the proposed amendment request as described above will be issued as a separate document at a later date.

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

Sincerely,

/S/

Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 52 to License No. DPR-18
2. Safety Evaluation

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Sincerely,

Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
License No. DPR-18

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

   A. The application for amendment filed by the Rochester Gas and Electric Corporation (the licensee) dated October 15, 1990, as supplemented March 5, 1991, November 30, 1992, and April 5, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 52 , 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 its date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.6-4	3.6-4
3.6-6	3.6-6
3.7-7A	3.6-7A
3.6-10	3.6-10

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	AOV 200A AOV 200B AOV 202 RV 203	60 60 60 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)
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
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	Shutdown Purge Supply Duct [Purge Supply Duct]	flange (22) [AOV 5870]	NA [5]	AOV 5869 (22)	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	(17)	NA
207	Przr. Steam space sample	AOV 966A	60	MV 956F	NA
207	"B" S/G sample	AOV 5736	60	(17)	NA
209	Reactor Compartment. 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

PENT. NO.	IDENTIFICATION/DESCRIPTION	ISOLATION BOUNDARY	MAXIMUM ISOLATION TIME *(SEC)	ISOLATION BOUNDARY	MAXIMUM/ ISOLATION TIME *(SEC)
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 (decomissioned)	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 4647 (16)	NA	(11)	NA
321	A S/G Blowdown	AOV 5738	60	(17)	NA
322	B S/G Blowdown	AOV 5737	60	(17)	NA
323	Service Water from "D" fan cooler	MV 4644 (16)	NA	(11)	NA
324	Demineralized water to Containment	CV 8419	NA	AOV 8418	NA
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

(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) The S/G tubes and secondary side provide a closed system inside containment.

(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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated October 15, 1990, as supplemented March 8, 1991, November 30, 1992, and April 5, 1993, the Rochester Gas and Electric Corporation (the licensee) submitted a request for changes to the R. E. Ginna Nuclear Power Plant, Technical Specifications (TS). The requested changes would revise the plant Technical Specifications (TS) to remove Containment Isolation Valve Table 3.6-1 and maintain the revised listing in their Updated Final Safety Analysis Report (UFSAR). The licensee also requested a number of other changes to the containment isolation valves (CIVs) particularly in the April 5, 1993, letter and requested staff approval for a limited number of specific changes to the TS Table 3.6-1. The licensee's request was in the integrated leak rate test (ILRT) to be conducted during the 1993 refueling outage. Specifically, the licensee proposed to revise CIVs for Penetration 112, letdown to non-regeneration heat exchanger, and for Penetrations 206B, 207B, 321 and 322, steam generator (SG) A/B sample and blowdown lines. For Penetration 112, the licensee proposed to delete valves 204A and 820 and add valves 200A, 200B, 202 and 203. For Penetration 206B, 207B, 321 and 322, the licensee proposed to delete normal valves 5733, 5734, 5701 and 5702 and revise note 17 by taking credit for a SG closed system. The licensee also proposed to eliminate 10 CFR Part 50, Appendix J, Type C testing related to four air-operated valves (AOVs) for Penetrations 206B, 207B, 321 and 322 since there are no requirements to perform this testing. The licensee proposed to hydrotest these valves to provide the necessary assurance for their isolation function.

The staff evaluation to the proposed changes is given below:

The March 8, 1991, November 30, 1992, and April 5, 1993, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Penetration 112 letdown to Non-regeneration Heat Exchanger

The licensee proposed to replace manual valves 204A and 820 listed as CIVs for Penetration 112 since their pressure and containment isolation functions are performed by valves AOV-200A, 200B and 202. The AOVs 200A, 200B and 202 were

identified as CIVs as part of the Third 10-Year Inservice Testing (IST) Program. The licensee also proposed to add relief valve 203 to Table 3.6-1 since this valve is located between AOVs 200A, 200B, 202 and second isolation valve 371. All four valves 200A, 200B, 202 and 203 are currently in the 10 CFR Part 50, Appendix J testing program. The licensee also proposed to delete present note 17 from Table 3.6-1 seeing it is no longer needed. The staff has reviewed the licensee's submittal for Penetration 112 and finds the proposed changes acceptable and meet the explicit double isolation requirements of General Design Criterion (GDC) 56.

Penetrations 206B, 207AB, 321 and 322 - SG Sample and Blowdown Lines

The licensee indicated that the present TS Table 3.6-1 lists a manual valve in series with an AOV as the CIVs for penetration 206B, 207B, 321 and 322. Since the SG tubes and secondary side provide one containment barrier the licensee proposes to delete manual valves 5733, 5734, 5701 and 5702 as CIVs from Table 3.6-1.

The SG blowdown lines transfer secondary water to the SG blowdown system for cleanup during plant operation. The SG blowdown lines are neither a part of the reactor coolant system (RCS) pressure boundary nor do they open directly to the containment atmosphere under post loss-of-coolant accident (LOCA) conditions. In accordance with General Design Criterion (GDC) 57, the SG shell and the lines attached to it, such as the main steam line, feedwater line and the SG blowdown lines, constitute a closed system inside containment, and it is one of the containment barriers for each associated containment penetration. The second redundant barrier at each penetration is a valve, such as the main steam or feedwater isolation valves and the SG blowdown valves. Based on the above, the staff finds the licensee's request to delete the manual valves 5733, 5734, 5701 and 5702 as CIVs from Table 3.6-1, acceptable. The licensee has confirmed that the SG blowdown lines inside containment are designed to seismic category I, and Safety Class 2, and are not susceptible to any high energy line sources including the affects of pipe whip and jet-impingement. New note 17 provides information with respect to SG closed systems. The main steam and main feedwater isolation valves are not required to be locally leakage rate tested (type C tests) in accordance with Section II.4.4 of Appendix J to 10 CFR Part 50. As the SG blowdown valves are analogous to the main steam and feedwater lines, the licensee proposed to delete the local leakage rate testing of isolation valves AOV-5735, 5736, 5737, and 5738 for these lines. The integrity of the closed systems inside containment are typically tested during Type A (integrated leakage rate) testing. The licensee has committed to leakage rate testing of these valves as part of the IST program using water as the test medium in accordance with ASME Section XI. The staff considers leakage rate testing of these valves to be a necessary part of periodic verification of the valve's capability to fulfill their safety function. Therefore, the testing proposed by the licensee is acceptable. Based on the above, the staff finds the licensee's proposal to discontinue Type C testing of the SG blowdown valves acceptable.

Based on the above evaluation, the staff finds acceptable the licensee's proposal to revise Table 3.6-1; Penetration 112 - to delete valves 204A and 820, and add valves 200A, 200B, 202 and 203; Penetration 206B, 207B, 321 and 322 - to replace manual isolation valves with the SG closed system, and 10 CFR Part 50, Appendix J, type C testing of valves 5735, 5736, 5737, and 5738, and substitute hydro testing as part of the IST Program per ASME Section XI.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State Official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (55 FR 51186 and 51187, dated December 12, 1990). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Raj Goel

Date: April 20, 1993