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CGrimes
J. Higgins, I&E Reg. I
L. Sjostrum, I&E Region I
H. Wong, I&E Hdqtrs.

3/29/98

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14649

Gentlemen:

The Commission has issued the enclosed Amendment No. 17 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Station in response to your requests dated October 28, 1975 (transmitted by letter dated October 30), January 21, 1976, and July 18, 1977. These submittals were in response to our letter dated August 4, 1975, requesting re-evaluation of the containment leak testing program and Technical Specifications for compliance with the requirements of Appendix J to 10 CFR Part 50.

The amendment consists of changes to the Technical Specifications for the Integrated Leakage Rate Test to include definitions of containment pressure and leak rates, pretest requirements, venting inside containment, conduct of test, revised acceptance criteria, revised test frequency, and reporting requirements. Also, there are changes to the Technical Specifications for Local Leak Detection Tests involving acceptance criteria, corrective action and test frequency. Finally, the basis for containment testing is revised. All of these changes are for clarification and to satisfy the requirements of Appendix J to 10 CFR Part 50. The amendment also grants the following exemptions from Appendix J to 10 CFR Part 50 which have been included in the Technical Specification changes:

1. The maximum allowable leakage rate for reduced pressure tests is $Lt = La (Pt/Pa)^{1/2}$ rather than $Ltm/Lam < 0.7$ as specified by Appendix J to 10 CFR 50 item III.A.4.a.

2. Alternative use of makeup flow measurements with a rotometer instead of the methods specified in Appendix J of 10 CFR 50 item III.B.1.

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3. The third of three Type A containment tests in a 10 year period may be performed one year before or after the 10 year inservice inspection instead of simultaneously with the 10 year inservice inspection.

4. The airlock door seal tests may be performed within 48 hours after the first in a series of air lock openings instead of each opening.

A "one time" waiver from the requirements of Section III.A.1.d. of Appendix J for the containment integrated leakage rate test to be conducted during the March 1978 refueling outage is granted for selected penetrations specified in the staff safety evaluation report. The "one time" waiver relates to venting and draining requirements for specified containment pipe penetrations and isolation valves. Accordingly, proposed Technical Specification sections 4.4.1.2 c, d and e have been deleted. Additional justification for these changes and/or proposed modifications must be provided prior to the next containment leakage rate test following the test scheduled for 1978.

Based on our evaluation of the exemptions, contained in the enclosed Safety Evaluation, and pursuant to 10 CFR Section 50.12, we have determined that the granting of the specific exemptions and the waiver referred to above are authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

To meet our requirements, certain changes to the Technical Specifications which you proposed were necessary. These changes have been discussed with and agreed to by your staff.

A copy of our Notice of Issuance also is enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 17 to DPR-18
2. Safety Evaluation
3. Notice

SEE PREVIOUS YELLOW FOR PREVIOUS CONCURRENCE*

cc w/enclosures:
See next page

OFFICE➤	DOR:ORB #2	DOR:ORB #2	OELD RE/S	DOR:ORB #2	DOR:PSB/OT	
SURNAME➤	JJShea:ah	RMDiggs*	[illegible]	DLZiemann	WButler	
DATE➤	3/ /78	3/ /78	3/28/78	3/28/78	3/ /78*	

Docket No. 50-244

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ACRS (16)
TBAbernathy
JRbuchanan

C. Grimes
J. Higgins, I&E-Reg I
L. Sjostrum, I&E-Reg I
H. Wong, I&E HQ

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
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The amendment consists of changes to the Technical Specifications for the Integrated Leakage Rate Test to include definitions of containment pressure and leak rates, pretest requirements, venting inside containment, conduct of test, revised acceptance criteria, revised test frequency, and reporting requirements. Also, there are changes to the Technical Specifications for Local Leak Detection Tests involving acceptance criteria, corrective action and test frequency. Finally, the basis for containment testing is revised. All of these changes are for clarification and to satisfy the requirements of Appendix J to 10 CFR Part 50. The amendment also grants the following exemptions from Appendix J to 10 CFR Part 50 which have been included in the Technical Specification changes:

1. The maximum allowable leakage rate for reduced pressure tests is Lt = La (Pt/Pa) and not only when Ltm/Lam 0.7 as specified by Appendix J to 10 CFR 50 item III.A.4.a.

2. Alternative use of makeup flow measurements with a rotometer instead of the methods specified in Appendix J of 10 CFR 50 item III.B.1.

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3. The third of three Type A containment tests in a 10 year period may be performed one year before or after the 10 year inservice inspection instead of simultaneously with the 10 year inservice inspection.

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Based on our evaluation of the exemptions, contained in the enclosed Safety Evaluation, and pursuant to Section 10 CFR Section 50.12, we have determined that the granting of the specific exemptions referred to above is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

To meet our requirements, certain changes to the Technical Specifications which you proposed were necessary. These changes have been discussed with and agreed to by your staff.

A copy of our Notice of Issuance also is enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. to DPR-18
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cc w/enclosures:
See next page

CG 3/24/78 corrections noted

OFFICE ➤	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:PSB	
SURNAME ➤	JJShea:ah	RMDiggs		DLZiemann	WButler	
DATE ➤	3/23/78	3/23/78	3/ /78	3/ /78	3/24/78	

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(w/cy of RG&E filings dtd.
10/30/75, 1/21/76 and
7/18/77)

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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. 17
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 October 28, 1975, as supplemented by filings dated January 21, 1976 and July 18, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Provisional 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. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 28, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 17

TO PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.4-1 thru 4.4-5	4.4-1 thru 4.4-5c
4.4-10 & 4.4-11	4.4-10 & 4.4-11

4.4 Containment Tests

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within specified values.

Specification

4.4.1 Integrated Leakage Rate Test

4.4.1.1 Definitions

Pa (psig) is the containment vessel design pressure of 60 psig.

Pt (psig) is the containment vessel reduced test pressure for periodic testing.

Lt (weight percent/24 hours) is the maximum allowable leakage rates of the containment vessel test atmosphere at pressure Pt.

La (weight percent/24 hours) is the maximum allowable leakage rate of the containment vessel test atmosphere at pressure Pa, 0.2%/24 hrs.

Lam and Ltm (weight percent/24 hours) are the total measured containment leakage rates of the containment vessel test atmosphere at pressures Pa and Pt respectively.

4.4.1.2 Pretest Requirements

a. A visual examination of the accessible interior and exterior surfaces of the containment structure shall be performed to uncover any evidence of structural deterioration which may affect either the containment structure integrity or leak-tightness. If there is evidence of structural deterioration, integrated leak rate testing shall not be performed until appropriate corrective action has been taken. Except for repairs to correct structural deterioration, however, no repairs or adjustments shall be made during the period between the initiation of the inspection and the performance of the test.

b. Closure of containment isolation valves shall be accomplished by normal operation and without any preliminary exercising or adjustments.

4.4.1.3 Conduct of Tests

a. All integrated leak rate tests shall be conducted in accordance with the provisions of American National Standard N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors, March 16, 1972.

b. The accuracy of each integrated leak rate test shall be verified by a supplemental test which confirms the accuracy of the test instrumentation and calculational methods by determining a leak rate which is within 0.25Lt of the test result. If results are not within 0.25Lt the reason shall be determined, corrective action taken and a successful supplemental test performed.

c. Integrated leak rate tests shall be conducted at an initial pressure (beginning of test) Pt $\geq$ 35 psig.

d. If during the test, including the supplemental test, potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the test not meeting the acceptance criteria, the test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and an integrated leak rate test performed.

4.4.1.4 Acceptance Criteria

a. The leakage rate Ltm shall be less than 0.75 Lt.

b. Lt shall be determined as $Lt=La\left(\frac{Pt}{Pa}\right)^{1/2}$

4.4.1.5 Test Frequency

a. A set of the three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted in the final year of the 10-year service period or one year before or after the final year of the 10-year service period provided:

i. the interval between any two Type A tests does not exceed four years,

ii. following one in service inspection, the containment airlocks and equipment hatch are leak tested prior to returning the plant to operation, and

iii. any repair, replacement, or modification of a containment barrier resulting from the inservice inspections shall be followed by the appropriate leakage rate test.

b. If any test fails to meet the acceptance criteria of 4.4.1.4.a the test schedule for subsequent regularly scheduled inservice tests shall be submitted to the Commission for review and approval.

c. If two consecutive tests fail to meet the acceptance criteria of 4.4.1.4.a, a retest shall be performed at each refueling shutdown or approximately every 18 months, whichever comes first, until two consecutive tests meet the acceptance criteria of 4.4.1.4.a, after which time the retest schedule of 4.4.1.5.a may be resumed.

4.4.1.6 Additional Requirements

a. A summary technical report shall be submitted to the Commission after the conduct of each integrated leak rate test. Information on any valve closure malfunction or valve leakage that requires corrective action before the test shall be included in the report.

4.4.2 Local Leak Detection Tests

4.4.2.1 Test

a. Local leakage rate tests shall be performed at intervals specified in 4.4.2.4 below and at a pressure of not less than 60 psig.

b. The local leakage rate shall be measured for each of the following components:

i. Containment penetrations that employ resilient seal gaskets or sealant compounds.

ii. Air lock and equipment door seals.

iii. Fuel transfer tube.

iv. Isolation valves on the testable fluid systems lines penetrating the containment.

v. Other containment components, which require leak repair in order to meet the acceptance criterion for any integrated leakage rate test.

4.4.2.2 Acceptance Criterion

The total leakage from all penetrations and isolation valves shall not exceed 0.60La.

4.4.2.3 Corrective Action

a. If at any time it is determined that the total leakage from all penetrations and isolation valves exceeds 0.60La, repairs shall be initiated immediately.

b. If repairs are not completed and conformance to the acceptance criterion of 4.4.2.2 is not demonstrated within 48 hours, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets this acceptance criterion.

4.4.2.4 Test Frequency

a. Except as specified in b. and c. below, individual penetrations and containment isolation valves shall be tested during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years.

b. The containment equipment hatch and fuel transfer tube shall be tested at each refueling shutdown or after each use, if that be sooner.

c. The containment air locks shall be tested at intervals of no more than six months by pressurizing the space between the air lock doors. In addition, following opening of the air lock door during the interval, a test shall be performed by pressurizing between the dual seals of each door opened, within 48 hours of the opening, unless the reactor was in the cold shutdown condition at the time of the opening or has been subsequently brought to the cold shutdown condition. A test shall also be performed by pressurizing between the dual seals of each door within 48 hours of leaving the cold shutdown condition, unless the doors have not been opened since the last test performed either by pressurizing the space between the air lock doors or by pressurizing between the dual door seals.

4.4.3 Recirculation Heat Removal Systems

4.4.3.1 Test

a. the portion of the residual heat removal system that is outside the containment shall either be tested by use in normal operation or hydrostatically tested at 350 psig at the interval specified in 4.4.3.4.

b. Suction piping from containment sump B to the reactor coolant drain tank pump and the discharge piping from the pumps to the residual heat removal system shall be hydrostatically tested at no less than 100 psig at the interval specified in 4.4.3.4.

The Specification also allows for possible deterioration of the leakage rate between tests, by requiring that the total measured leakage rate be only 75% of the maximum allowable leakage rate.

The duration and methods for the integrated leakage rate test established by ANSI N45.4-1972 provide a minimum level of accuracy and allow for daily cyclic variation in temperature and thermal radiation. The frequency of the integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns. Refueling shutdowns are scheduled at approximately one year intervals.

The specified frequency of integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the use of weld channels to test the leaktightness of the welds during erection, (b) conformance of the complete containment to a 0.1% per day leak rate at 60 psig during preoperational testing, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60 La) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program, which provides assurance that an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage of 0.60 La from penetrations and isolation valves is that only a portion of the allowable integrated leakage rate should be from those sources in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. Because most leakage during an integrated leak rate test occurs through penetrations and isolation valves, and because for most penetrations and isolation valves a smaller leakage rate would result from an integrated leak test than from a local test, adequate assurance of maintaining the integrated leakage rate within the specified limits is provided. The limiting leakage rates from the Recirculation Heat Removal Systems are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

Appendix J to 10 CFR Part 50 "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" was published on February 14, 1973. Many operating nuclear plants had either received an operating license or were in advanced stages of design or construction at that time. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which provided assurance that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the Ginna licensee and the results are reflected in the following evaluation.

In our letter, dated August 4, 1975, we requested that the Rochester Gas and Electric Corporation (RG&E) review the Ginna Plant in terms of the current containment leak testing program, and the associated Technical Specifications, for compliance with the requirements of Appendix J to 10 CFR Part 50. As part of this request, RG&E was to determine the planned actions and the associated schedule for attaining conformance with the above cited regulation.

RG&E submitted its responses to our request for information on October 30, 1975, January 21, 1976, and July 18, 1977. In these responses, RG&E proposed changes to the Technical Specifications for the R. E. Ginna Nuclear Power Plant and requested several exemptions from and clarifications of the requirements of Appendix J to 10 CFR 50.

On March 8, 1978, we met with members of the licensee's staff to discuss the results of our evaluation of the containment leak testing program for the Ginna facility. In that meeting, we discussed the testing provisions which would be required for the containment integrated leakage rate test to be conducted in late March 1978. The staff evaluation of the Ginna plant identified the testing provisions and requested exemptions which have been found acceptable and others which require additional justification before we can complete our evaluation.

Evaluation

Section II.K of Appendix J defines La as the maximum allowable leakage rate for the containment at a pressure Pa, as specified for the preoperational test in the technical specifications or associated bases, and as specified in the operating license for periodic tests. The licensee indicated that for the preoperational leakage rate test, a maximum allowable leakage rate of 0.1 weight percent per day of the containment atmosphere was conservatively assumed. Subsequently, in the licensee's Final Safety Analysis Report the potential consequences of a design basis accident were determined assuming a containment leakage rate of 0.2 weight percent per day. In the staff's evaluation for the Ginna provisional operating license, dated January 20, 1972, we concluded that a containment leakage rate of 0.2 weight percent per day would result in offsite doses that are less than the 10 CFR Part 100 guidelines. On this basis, we conclude that the specification of La as 0.2 weight percent per day is consistent with Section II.K and is, therefore, acceptable.

Section III.A.1.(d) of Appendix J requires that fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions, and those portions of closed systems inside containment that penetrate the containment and rupture as a result of a loss of coolant accident, shall be vented and drained to the extent necessary to assure exposure of the containment isolation valves to the test pressure. Section II.N of Appendix J indicates that the containment leakage rate is measured with the plant systems in a state as close as practical to that which would exist under design basis accident conditions.

The licensee has stated that in applying these sections of Appendix J, a determination is made whether systems which are postulated to rupture will completely drain. Where piping configurations are such as to indicate that a water leg will exist, fluid would remain in the lines. In addition, the length of line between redundant containment isolation valves is not vented. The licensee references Section 14.3.4 of the FSAR with regard to the criteria for identifying which systems are postulated to rupture. However, Section 14.3.4 only addresses the analysis techniques for the containment response, and does not discuss the criteria for postulating line breaks.

We find that the licensee has not provided sufficient justification to support its practices for venting and draining. It is not clear that water legs will exist in the systems which are postulated to rupture, when considering the potential for fluid flashing and the uncertainty associated with the behavior of the water legs given a smaller pressure differential associated with a reduced pressure test.

Furthermore, the licensee has not provided the specific criteria used for postulating system ruptures. However, during our meeting on March 8, 1978, we were advised by RG&E representatives that certain systems cannot be completely drained without physical modifications. Based on the information supplied, we have concluded that a "one-time" waiver from the requirements of Section III.A.1.d of Appendix J for the containment integrated leakage rate test to be conducted during the March 1978 outage, is acceptable for the following containment penetrations:

1. Makeup to Pressurizer Relief Tank
2. RCS Charging
3. "A" RCP Seal Injection
4. "B" RCP Seal Injection
5. Alternate Charging
6. Demineralized Water
7. Pressurizer Liquid and Gas Sample
8. Containment Sump

These systems should be vented and drained to the extent physically practicable. Although the systems identified above cannot be drained to to the extent required by Appendix J, evidence of excessive leakage will be detectable should the fluid in these systems be leaking. The Type A test is conducted using air as the leaking medium, while following a postulated loss-of-coolant accident in a steam-water-air mixture will exist. The actual post-accident environment has a higher viscosity and will, therefore, leak at a reduced rate. In addition, the Type A measured leakage rate includes a statistical margin for uncertainty. We find that these conservatisms will adequately account for differences incurred by the licensee's present venting and draining practices. Therefore, there is adequate assurance that the results of the test will provide a conservative estimate of the containment integrated leakage rate. On this basis, we conclude that a waiver of the requirements of Section III.A.1.d. of Appendix J is acceptable. However, additional justification and or modifications will be required for future tests.

If the venting and draining of certain systems will interfere with maintaining the plant in a safe-shutdown condition, then these systems need not be vented and drained for the containment integrated leakage rate test. However, the local leakage rates measured for the containment isolation valves in such systems shall be added to the result of the containment integrated leakage rate, before determining the acceptability of the test results. Accordingly, the Auxiliary Steam and Condensate Return system need not be vented inside containment but shall be drained to the extent practicable for the test, and the ILRT Depressurization, Air Supply Header, and Instrumentation lines Type C leakage rates for these penetrations shall be added to the Type A test results.

Section III.A.4.(a) of Appendix J requires that the acceptance criteria for reduced pressure tests be determined from the preoperational measured leakage rates at reduced pressure (Ltm) and peak pressure (Lam). Specifically, the allowable leakage rate at reduced pressure (Lt) is determined from the allowable leakage rate at peak pressure (La) by use of the following relationships:

a) If $Ltm/Lam < 0.7$, $Lt = La\ (Ltm/Lam)$; and
b) If $Ltm/Lam > 0.7$, $Lt = La\ (Pt/Pa)^{1/2}$

The results of the preoperational reduced pressure leakage rate test for Ginna indicated a negative leakage rate; i.e., the leakage was so small that it was contained within the error band about a negative leakage rate. By a strict application of the requirements of Appendix J, all future reduced pressure tests must also result in a negative leakage rate.

The licensee has requested an exemption from the requirements of Section III.A.4.(a) of Appendix J to permit the specification of the allowable leakage rate at reduced pressure (Lt) from the relationship $Lt = La\ (Pt/Pa)^{1/2}$ even though the ratio of the measured preoperational leakage rates is less than 0.7. The objective of this requirement in Appendix J is to provide a measure of the plant specific variation of leakage with pressure. By the application of the requirements cited above, the licensee is being penalized for measuring an extremely low leakage rate. Based on our review of the derivation of leakage extrapolation factors, we find that the square root of the pressure ratio provides a more accurate representation of the variation of containment leakage with pressure, than that obtained by the ratio of the preoperational test measured leakage rates. On this basis, we conclude that the licensee's proposed exemption for the determination of the allowable reduced pressure leakage rate is acceptable.

Section III.B.1 of Appendix J specifies the acceptable techniques for performing local (Type B) leakage rate tests. RG&E has requested an exemption from Section III.B.1 to permit the alternative use of makeup flow measurements with a rotometer. By this technique, the leakage rate from the test volume is measured by monitoring the flow required to maintain a constant pressure in the test volume. If performed properly, this technique will provide results with an accuracy equivalent to those methods currently listed in Section III.B.1 of Appendix J. On this basis, we find the licensee's proposed exemption acceptable.

Section III.D.1 of Appendix J requires that containment integrated leakage rate tests (Type A) be performed three times, at approximately equal intervals, during each ten-year service period. The third test shall be conducted when the plant is shutdown for the ten-year inservice inspection. RG&E has requested an exemption from this requirement to permit the third Type A test to be performed within one year before or after the ten-year inservice inspection, on the basis that there is no physical reason why the Type A test and the inservice inspection should coincide and that the $\pm$ 1 year interval is consistent with the requirements of ASME Section XI. The licensee requested this exemption because conducting the Type A test in conjunction with inservice inspection results in an excessively long outage period.

The objective in performing the third Type A test in conjunction with the ten-year inservice inspection outage is to verify the containment integrity prior to returning the plant to operation and to provide a means for establishing the appropriate intervals between Type A tests. However, the scope of the inservice inspection program is directed primarily toward the reactor coolant pressure boundary. The only components of the containment boundary that will be directly involved in the inservice inspection are the containment isolation valves. In addition, the containment airlocks and equipment hatch may be subjected to extensive use during conduct of the inservice inspection program. For the Ginna plant, all of these components can be locally leak tested. Moreover, Section V.A of Appendix J requires that any major modification or replacement of a component that is part of the containment boundary be followed by a local leakage rate test. Accordingly, the provisions of Appendix J, with regard to the schedule for Type A test, can be accomplished by alternative means.

We, therefore, find the licensee's proposal to conduct the third Type A leakage test within one year prior to or following the ten-year inservice inspection acceptable, providing: (1) the application of this relaxation period does not result in an interval between Type A tests which exceeds four years; (2) following the inservice inspection, the containment airlocks and equipment hatch will be leak tested prior to returning the plant to operation; and (3) any repair, replacement, or modification of a containment barrier resulting from the inservice inspection and/or structural proof test shall be followed by the appropriate local leakage rate test. We conclude that these provisions will provide a demonstration of the containment integrity, equivalent to that obtained by air testing requirements of Section III.D.1 of Appendix J, prior to returning the plant to operation following an inservice inspection outage.

Section III.D.2 of Appendix J requires that containment airlocks be leak tested at six month intervals; however, airlocks which are opened during such intervals shall be tested after each opening. RG&E has requested an exemption from this requirement to permit leak testing of only the airlock door seals within 48 hours following the first in a series of airlock openings.

Based on plant operating experience, we have concluded that the requirement to leak test airlocks after each opening is impractical, especially when frequent airlock use is required over a short period of time. Since the entire airlock assembly is leak-tested at six month intervals, the requirement for tests following each opening is directed toward assuring the integrity of the door seals. Testing the airlock door seals within a limited time period following an initial opening is more practical and will provide sufficient safety. The limited time period allowed for testing the airlock door seals has been established considering the frequency of airlock use, the probabilities of a door seal failure coincident with a postulated accident of sufficient magnitude to require containment integrity, and the double barrier protection afforded by the airlock. Further, at no time in plant operating experience has the simultaneous failure of both airlock door seals been observed. We conclude that testing the airlock door seals within 48 hours following the first in a series of airlock door openings will provide adequate assurance of the continued leak tightness of the airlock. On this basis, we find the licensee's proposed exemption acceptable.

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: March 28, 1978

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

The amendment incorporated Technical Specification changes for purposes of clarification to satisfy the requirements of Appendix J to 10 CFR Part 50, and to incorporate exemptions from 10 CFR Part 50, Appendix J, related to permissible test leakage rate, use of a rotometer to measure leak rate, frequency of Type A containment leak rate tests, frequency of airlock door seal tests, and containment pipe penetration draining and venting requirements.

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. The Commission also has found that the exemption to specific provisions of Appendix J to 10 CFR Part 50 is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. 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 application for exemption and amendment dated October 28, 1975 (transmitted by letter dated October 30), as supplemented by filings dated January 21, 1976 and July 18, 1977, (2) Amendment No. 17 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 Operating Reactors.

Dated at Bethesda, Maryland, this 28th day of March, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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