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DESCRIPTION: Ltr. notarized 1-16-76... Letter trans the following.....

ENCLOSURES:

Proposed Amdt. to OL/Change to Tech. Specs: Consisting of revisions to Tech. Specs. with regards to changing the surveillance requirements for intergrated leakage rate from the facility's containment.....
(1 Cy. Received)
Cert. of Service showing service upon local officials.....

PLANT NAME: RE Ginna # 1

FOR ACTION/INFORMATION

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BRANCH CHIEF Purple W/6 Cys

PROJECT MANAGER _____

LIC. ASST. S. Sheppard W/16 CYS ACRS

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Regulatory File Cy

January 21, 1976

Mr. Ben C. Rusche
Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Rochester Gas and Electric Corporation
R. E. Ginna Nuclear Power Plant Unit No. 1
Docket No. 50-244

Dear Mr. Rusche:

On October 28, 1975, Rochester Gas and Electric Corporation filed a request for exemption from certain provisions of Appendix J to 10 C.F.R. Part 50. The request for exemption serves as a basis for this request for change to technical specifications.

More specifically, this letter transmits forty (40) copies of a proposed change to Technical Specifications 4.4.1 and 4.4.2 to change the surveillance requirements for integrated leakage rate from the facility's containment. In addition, three (3) signed originals and nineteen (19) copies of a document entitled "Application for Amendment to Operating License" together with a Certificate of Service showing service of these documents upon the persons listed therein is also enclosed.

Very truly yours,

LeBoeuf, Lamb, Leiby & MacRae

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Rochester Gas
and Electric Corporation

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	Docket No. 50-244
ROCHESTER GAS AND ELECTRIC CORPORATION	)	
(R.E. GINNA Nuclear Power Plant, Unit No. 1)	)	

APPLICATION FOR AMENDMENT

TO

OPERATING LICENSE

Pursuant to section 50.90 of the regulations of the Nuclear Regulatory Commission, Rochester Gas and Electric Corporation ("RG&E"), holder of Provisional Operating License No. DPR-18, hereby requests that Technical Specifications 4.4.1 and 4.4.2 set forth in Appendix A to that license be amended as set forth in Attachment A to this application.

This amendment effects the surveillance requirements for certain plant equipment but does not authorize any change in the types or any increase in the amounts of normal plant effluents or any change in the authorized power level of the facility. A safety evaluation is set forth in Attachment B.

WHEREFORE, Applicant respectfully requests that Appendix A to Provisional Operating License No. DPR-18 be amended in the form attached hereto as Attachment A.

ROCHESTER GAS AND ELECTRIC CORPORATION

by Keith W. Amish

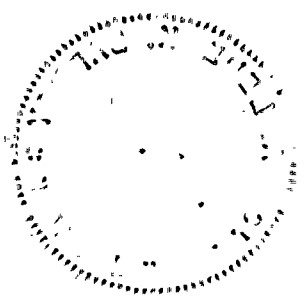
Keith W. Amish
Executive Vice President

Subscribed and sworn to before
me on this 16th day of January, 1976

Arlene K. Barney

Notary Public

ARLENE K. BARNEY
NOTARY PUBLIC, State of N.Y., Monroe County
My Commission Expires March 30, 1976



ATTACHMENT A

Replace 4.4.1 with the following:

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.

c. Venting Outside Containment

Lines which penetrate the containment and which are open to the containment atmosphere as in (d), will be vented to the atmosphere outside of the containment. Where piping configurations outside containment exist such that the fluid in fluid carrying lines does not drain to expose the isolation valves to the atmosphere by opening existing vents and drains, the fluid will be left in the lines.

d. Venting Inside Containment

Portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment will be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and that also pass inside the primary shield wall near the broken leg, and which are postulated to rupture as a result of a loss of coolant accident consistent with the containment integrity analysis of section 14.3.4 of the FSAR, will be vented to containment atmosphere. Where check valves or piping configurations exist between the primary shield wall and the containment penetration or in places where damage to the piping system is not postulated to occur as a result of a LOCA such that fluid seals are formed as a result of normal operation and containment isolation, the fluid may be left undisturbed.

That is, those portions of systems not postulated to rupture as the result of a LOCA need not be drained unless they drain unaided to the postulated breaks in the systems.

e. Isolation Valves

Where two isolation valves exist in a single line which are either check valves, or valves capable of automatic closure, or a combination thereof, no attempt need be made to vent to atmosphere from a point between the valves.

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) $P_t \geq 35$ 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 3 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 when the plant is shut down for the 10-year plant inservice inspections.

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 4.4.1.4a, 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 4.4.1.4a, after which time the retest schedule of 4.4.1.5a 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.

Delete 4.4.2.1.b

Renumber 4.4.2.1.c as 4.4.2.1.b

Replace 4.4.2.2 with the following:

4.4.2.4 Acceptance Criterion

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

Replace 4.4.2.3.a with the following:

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.

Replace 4.4.2.4 with the following:

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 6 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.

Replace pages 4.4.10 and 4.4.11 of the bases for specification 4.4 with the following:

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 1 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.

ATTACHMENT B

Safety Evaluation

The proposed changes to the technical specifications provide a conservative method of assuring that the leakage rate from the containment under accident conditions will remain within the specified limits for the following reasons.

Those lines postulated to be broken by a loss of coolant accident, as given in the containment integrity analysis, will be vented both inside and outside containment. Thus the full differential pressure will be applied across the isolation valve penetration for all these lines even though all of these lines may not be open to containment atmosphere under accident conditions or may be part of closed systems outside containment.

Even though no vent need be established between containment isolation valves during the test, this condition duplicates the post accident condition. The possibility that an isolation valve may fail to close is considered by performing the integrated leak test after closing the isolation valves by normal operation and without any preliminary exercising or adjustment. Periodic local leakage testing of the isolation valves assures that the valves are operable.

Conduct of the integrated leak rate test in accordance with ANSI N45.4-1972 will assure that the test results are valid.

An appropriate factor is to be applied to reduce the acceptance criterion for reduced pressure testing. Although it is difficult to establish the relationship between leakage rates at different test pressures for a specific containment, mass flow through orifices will generally behave as a function of the square root of the differential pressure. Thus, in the absence of extensive test data, the square root relationship is believed to be valid and will reduce the maximum allowable leakage rate by an appropriate amount for reduced pressure testing.

Testing personnel air lock doors by pressurizing the space between the doors at least once during each 6 month interval will preclude structural deterioration or wear from causing excessive leakage. The proposed method of testing personnel air lock doors within the six month intervals after an opening of a door will adequately ensure the integrity of the doors by detecting damage to the seals which may have resulted during the opening of the air lock. Testing the air locks by pressurizing between the seals will require approximately 15 minutes whereas testing by pressurizing the entire access hatch will require approximately 24 hours. By not inhibiting entry and inspections inside the containment, if they are required, the procedure tends to augment the safe operation of the plant.

The total leakage from penetrations and isolation valves during local leakage tests are limited to 0.60 La. This will provide adequate assurance of maintaining the integrated leakage rate within the specified limits 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.

The maximum allowable leakage rate at 60 psig has previously been shown to result in offsite doses, under postulated accident conditions, well within the requirements of 10 CFR Part 100.

Thus the test methods and procedures to be applied to meet the requirements of Appendix J to 10 CFR Part 50 will result in a valid test to determine the integrity of the reactor containment and will provide assurance that under accident conditions the leakage rate from the containment will remain within the specified limits.

BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION (R. E. Ginna Nuclear Power Plant, Unit No. 1)	)	Docket No. 50-244

CERTIFICATE OF SERVICE

I hereby certify that I have served a document entitled "Application for Amendment to Operating License" by mailing copies thereof first class, postage prepaid, to each of the following persons this 21st day of January, 1976.

Chairman, Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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