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"Request for Exemption"..with attachment "A" and "B", and Fig. 1...........Notarized 10-28-75
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LAW OFFICES OF
LEBOEUF, LAMB, LEIBY & MACRAE
1757 N STREET, N.W.
WASHINGTON, D.C. 20036

ARVIN E. UPTON
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140 BROADWAY
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WASHINGTON TELEPHONE
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October 30, 1975

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Secretary
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 Sir:

Pursuant to Section 50.12 of the regulations of the Nuclear Regulatory Commission, we hereby transmit on behalf of Rochester Gas and Electric Corporation an original of a document entitled "Request for Exemption" together with Attachments A and B. By this request, RG&E seeks relief from certain provisions of Appendix J to 10 C.F.R. Part 50. Two additional copies of this document are also transmitted for your convenience.

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

Enclosures

13503

BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

OCT 30 75

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 "Request for Exemption" by mailing copies thereof first class, postage prepaid, to each of the following persons this 30th day of October, 1975.

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

Mr. Michael Slade
1250 Crown Point Drive
Webster, New York 14580

Warren B. Rosenbaum, Esq.
One Main Street East
707 Wilder Building
Rochester, New York 14614

C. John Clemente, Esq.
New York State Department of Commerce
99 Washington Avenue
Albany, New York 12210

L. Dow Davis, IV, Esq.
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Thomas W. Reilly, Esq.
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dr. Franklin C. Daiber
Department of Biological
Sciences
University of Delaware
Newark, Delaware 19711

Dr. Emmeth A. Luebke
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dr. A. Dixon Callihan
Union Carbide Corporation
P. O. Box Y
Oak Ridge, Tennessee 37830

Mr. Robert N. Pinkney
Supervisor, Town of Ontario
107 Ridge Road West
Ontario, New York 14519

Hope M. Babcock

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

REQUEST FOR EXEMPTION

Pursuant to Section 50.12 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 it be exempted from certain provisions of Appendix J to 10 CFR Part 50. The specific exemptions requested are set forth in Attachment A to this application. A safety evaluation which demonstrates that the proposed exemptions will not endanger life and property or the common defense and security and are otherwise in the public interest is set forth in Attachment B. The proposed exemptions would 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.

WHEREFORE, Applicant respectfully requests that it be exempted from Appendix J to 10 CFR Part 50 as set forth in Attachment A.

ROCHESTER GAS AND ELECTRIC CORPORATION

By Leon D. White, Jr.

Leon D. White, Jr.
Vice President, Electric and Steam Production

Subscribed and sworn to before me this 28th day of October 1975.

Gary L. Reiss

Notary Public

GARY L. REISS
NOTARY PUBLIC, State of N. Y. Monroe Co.
My Commission Expires March 30, 19.77

ATTACHMENT A

In 1969 Rochester Gas and Electric Corporation performed type A preoperational containment leak rate testing at Ginna Station. The results of that testing at 60 psig and at a reduced pressure of 35 psig are shown on figure 1. Given that L_{tm} and L_{am} are the preoperational reduced pressure and full pressure test leakage rates respectively, and that L_a is the maximum allowable leakage rate, current 10 CFR 50 Appendix J regulations require that the acceptance criteria for subsequent reduced pressure testing be $L_a \left(\frac{L_{tm}}{L_{am}} \right)$; provided that $\frac{L_{tm}}{L_{am}} \leq 0.7$. In the event that $\frac{L_{tm}}{L_{am}} > 0.7$, the subsequent acceptance criteria is to be $L_a \left(\frac{P_t}{P_a} \right)^{1/2}$, or the maximum allowable leakage rate times the square root of the ratio of the test pressures.

As seen on figure 1, our 1969 reduced pressure test yielded a negative leakage value. The value is small and its error band includes positive values as expected under nearly all circumstances for a valid test. Literal interpretation of the regulations, however, would require all of our successive reduced pressure tests to show a negative leakage result. Since this is clearly impossible, a more realistic approach to determine an acceptance criterion is to reduce the maximum allowable leakage rate by a linear factor derived from the slope of the

line between the preoperational test data points with no regard to their absolute value (see figure 1). Positive values for successive tests would then be permissible. For RG&E's case, the resulting acceptance criteria would be approximately equal to that calculated using the ratio of the test pressures formula. Therefore, RG&E requests that an exemption from paragraph III. A. 4. (a) (1) (iii) of Appendix J to 10 CFR Part 50 be granted which will allow use of the ratio of the test pressures acceptance formula, $L_a \left(\frac{P_t}{P_a} \right)^{1/2}$, where L_a is 0.2 weight percent per day, P_a is 60 psig, and P_t is the gauge test pressure. This relationship will allow positive leakage rates for successive tests but still will maintain acceptable offsite accident doses as shown in prior safety analyses.

Several points in the regulations appear to be subject to interpretation. As a result, inconsistencies may exist in the regulations or between the regulations and Ginna Station's approved Technical Specifications. To resolve the following points, exemptions are requested from Appendix J to 10 CFR Part 50 if the Nuclear Regulatory Commission believes such exemptions are required:

(1) Paragraph II. K. of the regulations defines L_a as "... the maximum allowable leakage rate at pressure P_a, as specified for <u>preoperational</u> tests in the technical specifications or associated bases, <u>and</u> as specified for <u>periodic</u> tests in the operating license." (emphasis added) The value approved for periodic testing and appearing in Ginna Station's Technical Specifications is 0.20 weight percent at 60 psig. However, the acceptance

value used in preoperational testing was 0.1 weight percent at 60 psig. This was established with our supplier to ensure that the 0.20 percent leakage rate requirement would be conservatively met. Because our FSAR safety analyses assumed a 0.25 percent leakage, and the staff SER used a leakage rate of 0.20 percent with acceptable offsite doses resulting from the calculations and because Ginna Station Technical Specifications have used at least 0.20 percent for post operational tests, we intend to continue to use the 0.20 percent maximum allowable leakage rate appearing in Ginna Station Technical Specifications.

(2) Paragraph II. N. defines leakage rates that are "... obtained from testing the containment with components and systems in the state as close as practical to that which would exist under design basis accident conditions." Paragraph III. A. 1. (d), on the other hand, states that "[a]ll vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure ..." This paragraph also states that "[s]ystems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system, need not be vented."

In view of the differences in interpretation which may be

attached to these regulations, and with no technical specification covering these points, RG&E intends to pursue the following course of action.

(a) Venting Outside Containment

Lines which penetrate the containment and which are open to the containment atmosphere as in (b), 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.

(b) 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 will be vented to containment atmosphere. Lines which have never been postulated to rupture, consistent with the containment integrity analysis of section 14.3.4 of the FSAR, will not be vented. 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 will be left undisturbed.

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

(c) 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 will be made to vent to atmosphere from a point between the valves.

(3) Paragraph III. D. 2. of Appendix J to 10 CFR Part 50 states that "[a]ir locks shall be tested at 6 month intervals. However, air locks which are opened during such intervals, shall be tested after each opening." Ginna Station Technical Specifications, on the other hand, require that "... the personnel air lock seals shall be tested at 4 month intervals, except when the air locks are

not opened during the interval. In that case, the test is to be performed after each opening, except that no test interval is to exceed 12 months."

Because the regulations do not say specifically how the testing is to be performed, because extensive testing after each opening of the air lock when multiple openings may take place in short time spans is impractical, and because of the proven reliability of these air locks, RG&E intends to meet the intent of paragraph III. D. 2 of Appendix J to 10 CFR Part 50 by testing as follows...

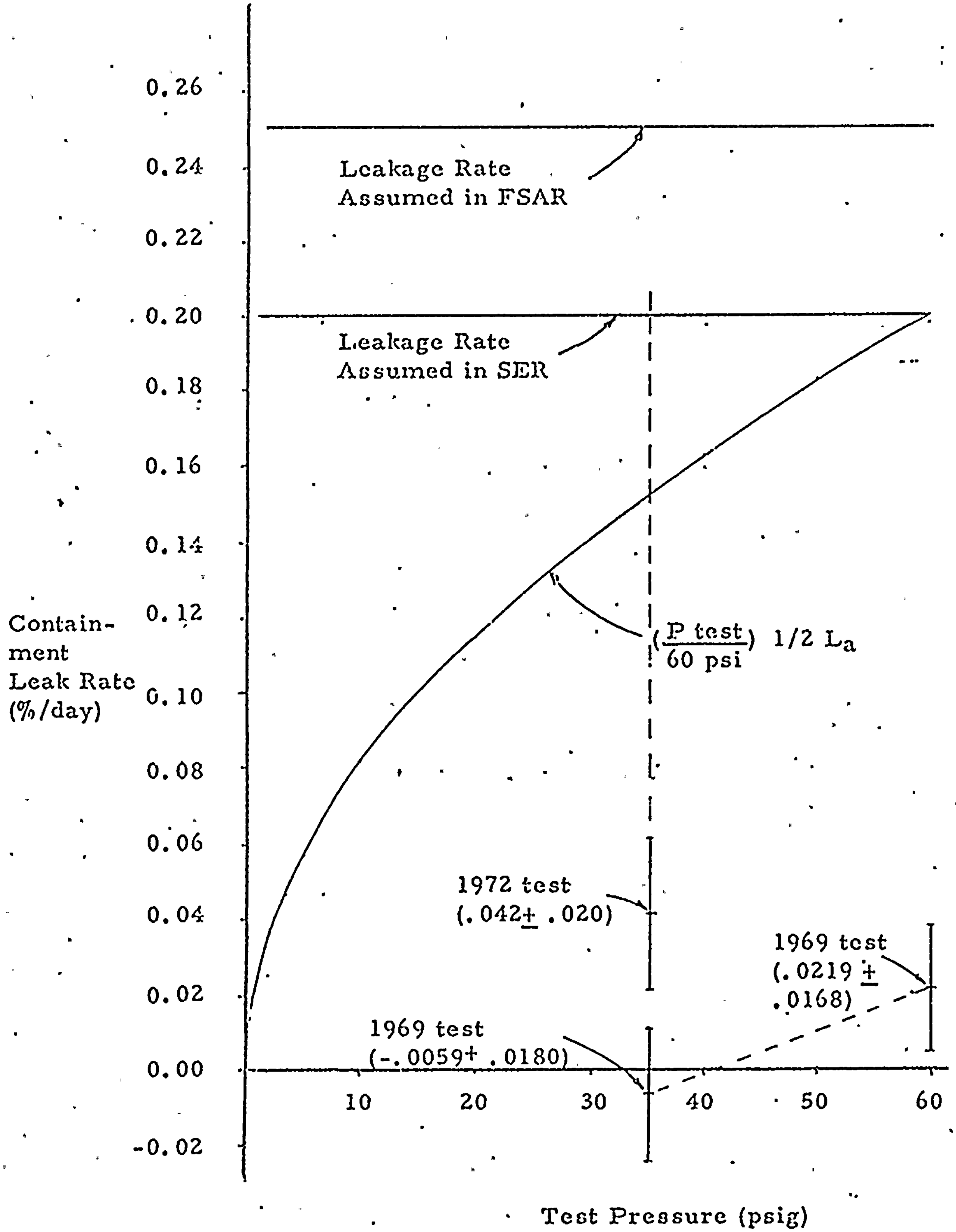
The containment air locks are to be tested at intervals of no more than 6 months by pressurizing the space between the air lock doors. (An application for an appropriate amendment to Ginna Station Technical Specifications will be submitted at a later date.) In addition, following opening of the air lock door during the interval, a test will 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. Thus, in the event that several openings occur within a short period of time, one test within 48 hours of the first opening will satisfy the requirements for leak testing.

(4) Paragraph III.B.1. gives those test methods which are acceptable means of performing periodic Type B tests. One of the methods is to measure "the rate of pressure loss of the test chamber of the containment penetration..." The method of testing preferred by RG&E is a measurement by means of a rotometer of the air flow to the test chamber which is required to maintain the chamber at the test pressure. We believe that this method produces accurate results and meets the intent of the regulations.

FIGURE 1

42-33

ENG. DEPT.	STATION: Ginna	DATE: 7/7/75	PAGE 1 OF 1
JOB: Containment Leak Rate Test		MADE BY: P. Wilkens	CK:



ATTACHMENT B

Safety Evaluation

The requested exemption will present no significant hazard to the health and safety of the public for the following reasons:

The proposed change in reduced pressure test acceptance criterion does not alter the maximum acceptable leakage rate. The leakage rates used in the Staff's Safety Evaluation Report and the FSAR were 0.20 and 0.25 percent per day respectively. Offsite dose calculations using these leakage rates demonstrated acceptable public exposures well below 10 CFR Part 100 values.

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 assumed in the Staff's Safety Evaluation Report by an appropriate amount for reduced pressure testing.

Therefore, the overall effect of the requested exemption is to provide an acceptance criterion which has already been found acceptable in accident analyses. The limit on containment leakage is such that there is no undue risk to the health and safety of the public.

The proposed method of testing personnel air lock doors within the six month intervals after an opening of a door will adequately insure 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 alternate procedure tends to augment the safe operation of the plant.

The regulations specify that lines which rupture as the result of a loss of coolant accident should be vented to the containment atmosphere prior to testing. The lines which will be vented are selected based upon assumptions made in the containment integrity analysis.

The proposed method of venting and draining systems penetrating the containment is consistent, we believe, with the intent of the regulations.

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. The acceptance standards for this test have been shown to result in offsite doses, under postulated accident conditions, well within the requirements of 10 CFR Part 100.