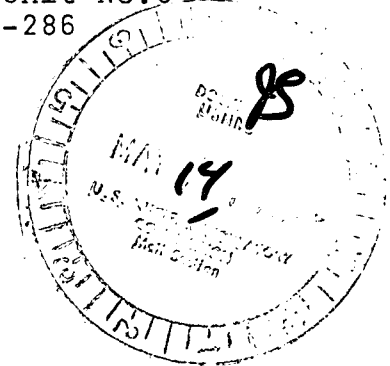**William J. Cahill, Jr.**
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

May 12, 1975

Re   Indian Point Unit No.3
     Docket No. 50-286

Mr. A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Giambusso

In a March 27, 1975 submittal to your office, we
transmitted forty (40) copies of the report entitled,
"Preoperational Integrated Leak Rate Test of the
Reactor Containment Building; Consolidated Edison
Corporation, Indian Point Unit 3", dated March 19, 1975.

Review of the report revealed a typographical error
and minor discrepancies in data reduction. The appro-
priate changes have been made to the report. Enclosed
are forty (40) copies of revised pages, together with
an instruction sheet outlining the method of incorpor-
ating the revised pages into the original report.

It should be noted that the reported leak rates have
changed only slightly. For the test at peak accident
pressure, the reported leak rate has changed from 0.023%/day
to 0.027%/day. For the reduced pressure test, the reported
leak rate has changed from 0.006%/day to 0.005%/day.

Very truly yours

William J. Cahill, Jr.
Vice President

Enc.
mrb

USNRC
MAY 1 4 1975
Office of the Secretary
Docketing & Service
Section
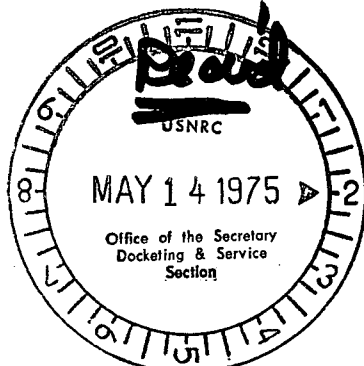
5348

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of   )
)
CONSOLIDATED EDISON COMPANY   )
   OF NEW YORK, INC.   )   Docket No. 50-286
(Indian Point Station,   )
   Unit No. 3)   )

CERTIFICATE OF SERVICE

I hereby certify that I have this 14th day of May,

1975, served the foregoing letter from William J. Cahill, Jr.

to Angelo Giambusso dated May 12, 1975 by mailing copies

thereof first class, postage prepaid, and properly addressed

to the following persons:

Samuel W. Jensch, Esq.   Max D. Paglin, Esq.
Chief Administrative Law   Atomic Safety and Licensing
   Judge   Board Panel
U.S. Nuclear Regulatory   U.S. Nuclear Regulatory
   Commission   Commission
Washington, D. C. 20555   Washington, D. C. 20555

Dr. Franklin C. Daiber   Mr. R. B. Briggs
College of Marine Studies   110 Evans Lane
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Newark, Delaware 19711
                                        Atomic Safety and Licensing
Mr. Ernest E. Hill   Board Panel
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J. Bruce MacDonald, Esq.
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  Counsel
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  Commerce
99 Washington Avenue
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Hon. George V. Begany
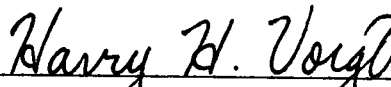Mayor, Village of Buchanan
Buchanan, New York 10511

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

Secretary (2 copies)
U.S. Nuclear Regulatory
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Washington, D. C. 20555
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Attorneys for Consolidated Edison
Company of New York, Inc.

1.0 SYNOPSIS

The Indian Point Nuclear Station Unit 3 reactor containment building was subjected to a preoperational integrated leak rate test during the period from January 15, 1975 to January 19, 1975. The purpose of this test was to demonstrate the acceptability of building leakage rates at internal pressures of 41 psig (P_a) and 21 psig (P_t). Testing was performed in conformance with the requirements of 10 CFR 50, Appendix J, ANSI N45.4-1972 and Indian Point Nuclear Station Unit 3 FSAR.

Leakage rates based on the point-to-point method of analysis were found to be 0.027 percent by weight per day at 41 psig and 0.005 percent by weight per day at 21 psig. These leakage rates are well below the acceptable test leakage rates of 0.075 percent per day at 41 psig and 0.014 percent per day at 21 psig.

L_{tm}/L_{am} is therefore established at 0.185. In accordance with 10 CFR 50, Appendix J and the Indian Point Nuclear Station Unit 3 FSAR, Section 15.4.4, Revision 10 (Technical Specifications), subsequent integrated leakage rate tests may be performed at P_t with a maximum allowable leakage value of 0.019%/day based on an L_t value of L_a (L_{tm}/L_{am}), since L_{tm}/L_{am} is less than 0.7. Therefore, the acceptable leakage (L_{tm}) for subsequent test should be less than the 0.75 L_t value which is 0.014 percent per day.

Since several penetrations were being used to conduct the leakage rate tests, the addition of the local leakage rate of penetrations YY, XX and RR to the measured values of L_{am} and L_{tm} was warranted. However, subsequent to the leak rate test the combined local leakage rate of these penetrations was measured and found to be zero.

The supplemental instrumentation verification at P_a and P_t was 16 percent and 15.8 percent, respectively; well within the 25 percent requirement of 10 CFR 50, Appendix J, Section III A.3.b.

All testing was performed by Wedco Corporation for Consolidated Edison Corporation with the technical assistance of Gilbert Associates, Inc. and Energy Incorporated. Calculations were checked by Gilbert Associates, Inc.

2) $\quad L_t = L_a (P_t/P_a)^{\frac{1}{2}}$ if $L_{tm}/L_{am} > .7$

Based on the test results (Section 3.2), L_t was determined using criterion 1) above, as follows:

$$L_t = L_a (L_{tm}/L_{am})$$

$$L_t = 0.1 (.005/.027)$$

$$L_t = 0.019\%/day$$

The acceptance criterion for the leakage rate at P_t was then determined, as follows:

$$L_t = 0.019\%/day$$

$$0.75L_t = 0.014\%/day$$

c. The acceptance criterion that the test instrumentation be verified by means of a supplemental test within 25 percent L_a (or L_t) was established in accordance with 10 CFR 50, Appendix J.

3.2 Conclusions

a. The measured leakage rate (L_{tm}) at a containment internal pressure of 21 psig (P_t) was 0.005 percent per day. This value is well below the above stated acceptance criterion of 0.014 percent per day. Therefore, reactor containment building leakage at reduced pressure (P_t) of 21 psig is considered to be acceptable.

b. The measured leakage rate (L_{am}) at a containment internal pressure of 40.6 psig (P_a) was 0.027 percent per day. This value is well below the above stated acceptance criterion of 0.075 percent per day. Therefore, reactor containment building leakage at design basis accident pressure (P_a) of 40.6 psig is considered to be acceptable.

c. Verification of test accuracy at P_t and P_a was accomplished by means of a supplemental test in each case, during which a superimposed, controlled leakage rate from the containment was instituted. Appendix J of 10 CFR 50 requires that the difference between the supplemental test results and type A test results be within 25 percent of L_a at peak pressure (P_a) and within 25 percent of L_t at reduced pressure (P_t).

The following summary indicates values for these tests:

	24 Hour Leakage Rate (%/day)	Supplemental Test Leakage Rate (%/day)	Difference (%/day)
P_a	0.027	0.011	0.016
P_t	0.005	0.002	0.003

A comparison of these results yields the following:

$$\text{At } P_a: \quad \frac{\left|L_{am} - L'_{am}\right|}{L_a} = \frac{\left|0.027 - 0.011\right|}{0.10\%/\text{day}} =$$

$$0.16, \text{ or } 16\% \text{ of } L_a$$

At P_t: $\dfrac{\left| L_{tm} - L'_{tm} \right|}{L_t}$ = $\dfrac{\left| 0.005 - 0.002 \right|}{0.019\%/\text{day}}$ =

0.158, or 15.8% of L_t

These comparisons are both well below the 25 percent limit specified by Appendix J of 10 CFR 50. Therefore, the supplemental tests are considered to have satisfactorily verified the acceptability of the test instrumentation.

d. In accordance with 10 CFR 50, Appendix J, the following conclusion was reached concerning the value of L_t to be used for subsequent reactor containment building integrated leak rate tests:

$$L_t = L_a \ (L_{tm}/L_{am})$$

$$L_t = 0.1 \ (0.005/0.027)$$

$$L_t = 0.019\%/\text{day}$$

This determination of L_t was used since (L_{tm}/L_{am}) was less than 0.7 (i.e., L_{tm}/L_{am} = 0.19).

The acceptance criterion for subsequent integrated leak rate tests then becomes 0.014 percent per day. This value was determined as follows:

$$L_t = 0.019\%/\text{day}$$

$$0.75 \ L_t = 0.014\%/\text{day}$$

7.0   DISCUSSION OF RESULTS

7.1   Results at P_t

Data obtained during the leak rate test at P_t indicated the following changes during the 24 hour test period:

Variable	Maximum Change
P_T	0.013 psia
P_{wv}	0.008 psia
T	$0.220^{\circ}F$

The methods used in calculating the leakage rate are as defined in Section 6.0. The results of the calculations are as follows:

Method	Leakage Rate (%/Day)	Corrected Leakage Rate (%/Day)
Point-to-Point	0.004	0.005

In accordance with Indian Point Nuclear Station Unit 3 FSAR, Section 15.4.4, Revision 10, leakage rates have been corrected from test conditions to design basis accident conditions. Therefore, these values are more conservative than normally required.

Based upon the point-to-point method of calculation, the leakage rate (L_{tm}) was 0.005%/day. (see Appendix D)

The confidence limit associated with this leakage rate derived from a least squares fit of the data is 0.041 percent per day. Correcting for systematic error (e_{L_t} = 0.011%/day), this value reduces to 0.030 percent per day. The random error introduced is

three times less than the maximum allowable leakage rate value of 0.10 percent per day and therefore it may be concluded that random error was not of any major significance.

7.2   Results at P_a

Data obtained during the leak rate test of P_a indicated the following changes during the 24 hour test period:

Variable	Maximum Change
P_T	0.025 psia
P_{wv}	0.009 psia
T	$0.290^{\circ}F$

The methods used in calculating the leakage rate are defined in Section 6.0. Results of these calculations are as follows:

Method	Leakage Rate (%/Day)	Corrected Leakage Rate (%/Day)
Point-to-Point	0.023	0.027

In accordance with Indian Point Nuclear Station Unit 3 FSAR, Section 15.44, Revision 10, leakage rates have been corrected from test conditions to design basis accident conditions. Therefore, these values are more conservative than normally required.

Based upon the point-to-point method of calculation, the leakage rate (L_{am}) was 0.027 %/day. (see Appendix D)

The confidence limit associated with this leakage rate derived from a least squares fit of the data is 0.046 percent per day. Correcting for systematic error ($e_{L_a} = 0.010\%/\text{day}$), this value reduces to 0.036 percent per day. The random error introduced is approximately two and one-half to three times less than the maximum allowable leakage rate value of 0.10 percent per day and therefore it may be concluded that random error was not of any major significance.

7.3 Supplemental Test Results

The results of the supplemental test at pressure P_t are as follows:

$$L'_{tm} = L_c - L_o$$
$$L'_{tm} = 0.036 - 0.034$$
$$L'_{tm} = 0.002\%/\text{day}$$

This value compares favorably with the measured leakage rate L_{tm} of 0.005 percent per day. This agreement is 15.8 percent of L_t, well below the 25 percent of L_t which is allowable.

The results of the supplemental test at pressure P_a are as follows:

$$L'_{am} = L_c - L_o$$
$$L'_{am} = 0.063 - 0.052$$
$$L'_{am} = 0.011\%/\text{day}$$

This value compares favorably with the measured leakage rate

of 0.027 percent per day. This agreement is 16 percent of L_a,

well below the 25 percent of L_a which is allowable.

This verification, through supplemental tests, clearly

established the acceptability of the test instrumentation.

The two measured leakage rates values (L_c), mentioned above,

are L_{mean} as determined by the point-to-point method.

APPENDIX C

REDUCED TEST DATA

LEAKAGE RATE AT 21 PSIG

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^{o}R)	Superimposed Flow Rate (1bm/hr)
1/16/75 1300	35.775	.171	35.604	535.95	-
1400	35.777	.173	35.604	535.98	-
1500	35.775	.174	35.601	535.92	-
1600	35.773	.175	35.597	535.90	-
1700	35.770	.176	35.594	535.83	-
1800	35.766	.175	35.591	535.78	-
1900	35.765	.176	35.589	535.76	-
2000	35.765	.175	35.590	535.79	-
2100	35.767	.176	35.591	535.78	-
2200	35.768	.176	35.592	535.81	-
2300	35.770	.176	35.594	535.83	-
2400	35.772	.176	35.596	535.83	-
1/17/75 0100	35.773	.176	35.597	535.87	-
0200	35.777	.176	35.601	535.93	-
0300	35.777	.177	35.601	535.91	-
0400	35.777	.177	35.600	535.91	-
0500	35.777	.177	35.600	535.92	-
0600	35.776	.177	35.599	535.89	-
0700	35.776	.176	35.600	535.90	-
0800	35.773	.177	35.596	535.84	-

APPENDIX C

REDUCED TEST DATA

LEAKAGE RATE AT 21 PSIG

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^{o}R)	Superimposed Flow Rate (1bm/hr)
0900	35.772	.177	35.595	535.83	-
1000	35.772	.177	35.595	535.80	-
1100	35.773	.179	35.594	535.81	-
1200	35.773	.177	35.596	535.83	-
1300	35.772	.179	35.593	535.81	-
1400	35.776	.180	35.596	535.86	-
1500	35.779	.180	35.599	535.91	5.75
1600	35.781	.179	35.602	535.94	5.79
1700	35.778	.180	35.598	535.89	5.79
1800	35.773	.178	35.595	535.84	5.76
1900	35.772	.179	35.593	535.83	5.76
2000	35.774	.181	35.593	535.85	5.76
2100	35.772	.181	35.591	535.84	5.76
2200	35.770	.181	35.589	535.81	5.75

APPENDIX C

REDUCED TEST DATA

LEAKAGE RATE AT 41 PSIG

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (°R)	Superimposed Flow Rate (lbm/hr)
1/18/75 1400	55.785	.208	55.577	538.00	-
1500	55.788	.206	55.582	538.03	-
1600	55.790	.204	55.586	538.05	-
1700	55.788	.205	55.583	538.05	-
1800	55.773	.206	55.567	537.88	-
1900	55.769	.205	55.564	537.88	-
2000	55.773	.206	55.567	537.94	-
2100	55.779	.207	55.572	538.02	-
2200	55.778	.206	55.572	538.00	-
2300	55.778	.205	55.573	538.00	-
2400	55.778	.205	55.573	538.03	-
1/19/75 0100	55.778	.203	55.575	538.04	-
0200	55.779	.206	55.573	538.03	-
0300	55.780	.204	55.576	538.07	-
0400	55.786	.203	55.583	538.13	-
0500	55.788	.201	55.587	538.17	-
0600	55.778	.203	55.575	538.06	-
0700	55.769	.203	55.566	538.00	-
0800	55.770	.200	55.570	538.00	-
0900	55.769	.200	55.569	538.01	-

GILBERT ASSOCIATES, INC.
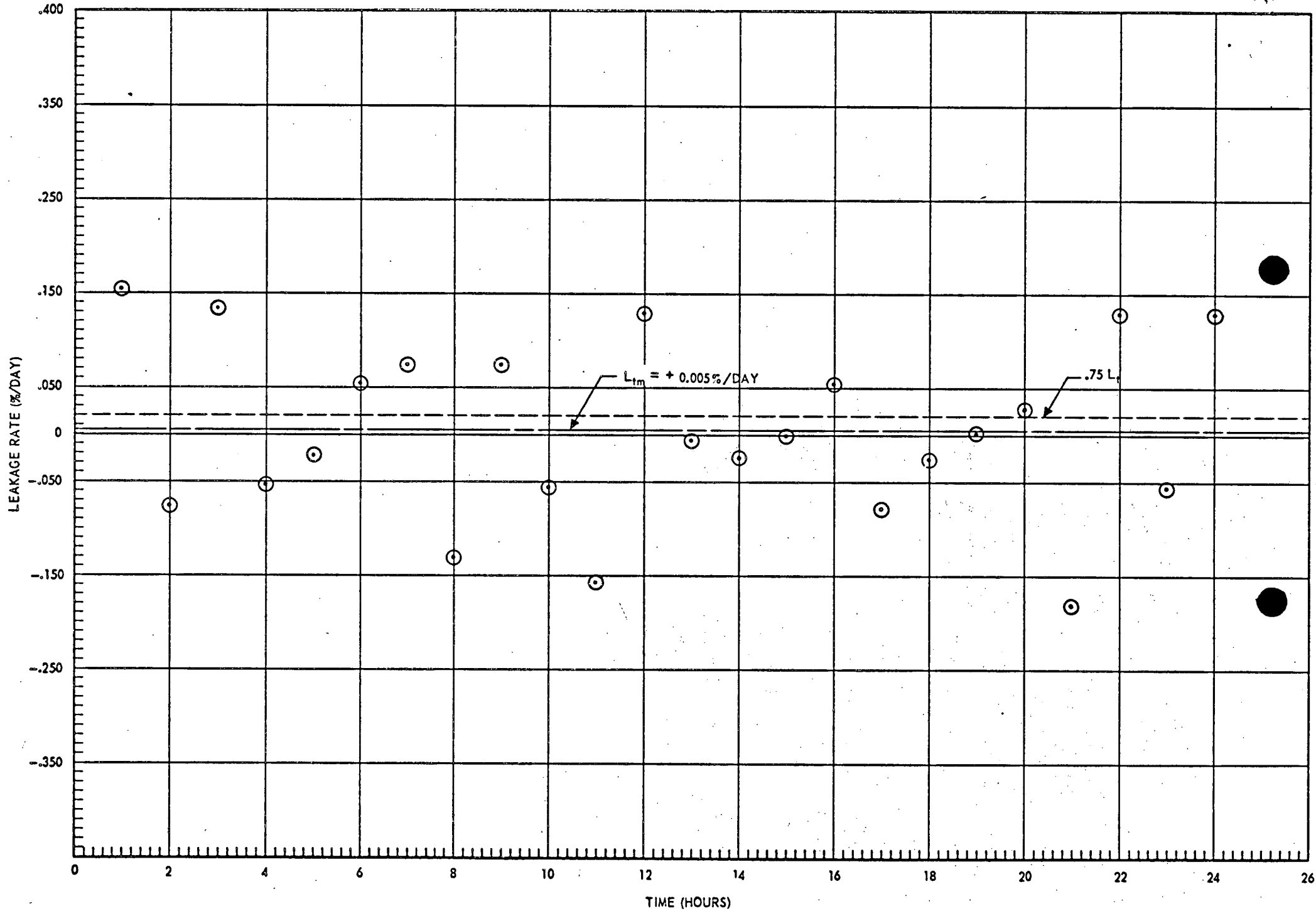
APPENDIX C

REDUCED TEST DATA

LEAKAGE RATE AT 41 PSIG

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature ($^\circ$R)	Superimposed Flow Rate (lbm/hr)
1000	55.768	.201	55.567	538.00	–
1100	55.766	.199	55.567	537.98	–
1200	55.765	.199	55.566	537.98	–
1300	55.767	.202	55.565	537.99	–
1400	55.767	.201	55.566	538.02	–
1500	55.769	.203	55.566	538.09	13.59
1600	55.777	.200	55.577	538.13	13.55
1700	55.779	.202	55.577	538.17	13.58
1800	55.771	.199	55.572	538.08	13.55
1900	55.765	.199	55.566	538.06	13.55
2000	55.763	.198	55.565	538.04	13.53
2100	55.762	.203	55.559	538.06	13.54
2200	55.760	.200	55.560	538.06	13.54

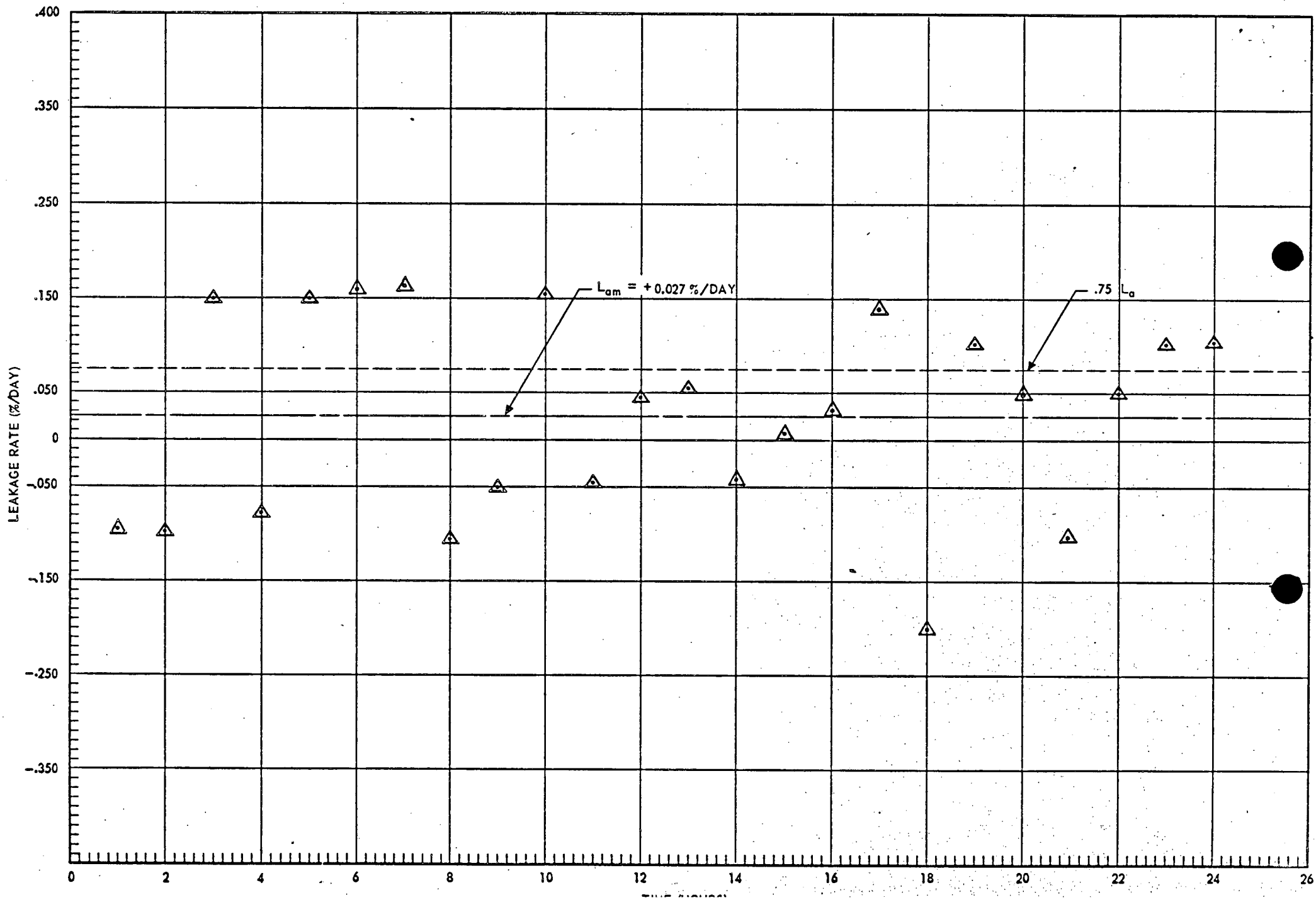## 21 PSIG POINT-TO-POINT LEAKAGE RATE VERSUS TIME

41 PSIG POINT–TO–POINT LEAKAGE RATE VERSUS TIME

INSTRUCTION SHEET

Revised Pages for "Preoperational Leak Rate Test of the Reactor Containment Building; Consolidated Edison Corporation, Indian Point Unit 3"

The attached revised pages contain certain changes made to the report subsequent to its submittal to the Nuclear Regulatory Commission on March 27, 1975.

The revised pages should be added to the report as noted below. This instruction sheet should be inserted in the report immediately following the title page.

Remove Old Page	Insert Revised Page
1	1
2	2
6	6
7	7
8	8
34	34
35	35
36	36
37	37
App. C Sheet 1 of 2 (21 psig)	App. C Sheet 1 of 2 (21 psig)
App. C Sheet 2 of 2 (21 psig)	App. C Sheet 2 of 2 (21 psig)
App. C Sheet 1 of 2 (41 psig)	App. C Sheet 1 of 2 (41 psig)
App. C Sheet 2 of 2 (41 psig)	App. C Sheet 2 of 2 (41 psig)
App. D (21 psig)	App. D (21 psig)
App. D (41 psig)	App. D (41 psig)

Date of Revision: May 12, 1975