THREE MILE ISLAND NUCLEAR STATION UNIT 1

REACTOR CONTAINMENT BUILDING

INTEGRATED LEAK RATE TEST

APRIL 1978

METROPOLITAN EDISON COMPANY

SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

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1.0 SYNOPSIS

The Three Mile Island Nuclear Station Unit 1 reactor containment building was subjected to a periodic integrated leak rate test during the period from April 12, 1978 to April 15, 1978. The purpose of this test was to demonstrate the acceptability of the building leakage rate at an internal pressure 50.6 psig (P_a). Testing was performed in accordance with the requirements of 10 CFR 50, Appendix J and ANSI N45.4-1972.

The measured leakage rate based on the mass point method of analysis and using absolute values corrected for instrument error was found to be 0.061 percent by weight per day at 50.6 psig. The leakage rate at the upper bound of the 95 percent confidence interval is 0.064 percent by weight per day which is below the allowable leakage rate of 0.075 percent by weight per day at 50.6 psig.

Since the industrial cooler system was in operation during the integrated leak rate test, addition of the local leakage rate of the system isolation valves (RB-V2* and RB-V7) to the measured integrated leakage rate must be considered. In addition containment isolation valve IC-V4 could not be opened for draining. The combined local leakage rate of these isolation valves was 0.007 percent by weight per day. The addition of this value increases the total integrated leakage rate to 0.071 percent by weight per day.

The supplemental instrumentation verification at P_a was 3.0 percent, well within the 25 percent requirement of 10 CFR 50, Appendix J, Section III A.3.b.

All testing was performed by Metropolitan Edison Company with the technical assistance of Gilbert Associates, Inc. Procedural and calculational methods were witnessed by Nuclear Regulatory Commission personnel and audited by the Metropolitan Edison Company site Quality Control staff.

2.0 INTRODUCTION

The objective of the periodic integrated leak rate test was the verification of the overall leak tightness of the reactor containment building at the calculated design basis accident pressure of 50.6 psig. The allowable leakage is defined by the design basis accident applied in the safety analysis in accordance with site exposure guidelines specified by 10 CFR 100. For Three Mile Island Nuclear Station Unit 1, the maximum allowable integrated leakage rate at the design basis accident pressure of 50.6 psig (P_a) is 0.10 percent by weight per day (L_a).

Testing was performed in accordance with the procedural requirements as stated in Metropolitan Edison Company Three Mile Island Nuclear Station Unit 1 Surveillance Procedure 1303-6.1. This procedure was recommended for approval by the Three Mile Island Nuclear Station Unit 1 Plant Operations Review Committee and approved by the Unit Superintendent prior to the commencement of the test.

The combined local leakage rates from the reactor containment building isolation valves and penetrations required to be tested by 10 CFR 50, Appendix J, was less then 60 percent of the maximum allowable leakage rate (L_a) at 50.6 psig prior to the commencement of the integrated leak rate test (Refer to Appendix C).

Leakage rate testing was accomplished at the pressure level of 50.6 psig for a period of 44.5 hours. The 44.5 hour period was followed by an 8 hour supplemental test for a verification of test instrumentation.

3.0 ACCEPTANCE CRITERIA

Acceptance criteria established prior to the test and as specified by 10 CFR 50, Appendix J and ANSI N45.4-1972 are as follows:

a. The measured leakage rate (L_{am}) at the calculated design basis accident pressure of 50.6 psig (P_a) shall be less than 75 percent of the maximum allowable leakage rate (L_a), specified as 0.10 percent by weight of the building atmosphere per day. The acceptance criteria is determined as follows:

$$L_a = 0.10\%/\text{day}$$

$$0.75L_a = 0.075\%/\text{day}$$

b. The test instrumentation shall be verified by means of a supplemental test. Agreement between the containment leakage measured during the Type A test and the containment leakage determined during the supplemental test shall be within 25 percent of L_a.

4.0 TEST INSTRUMENTATION

4.1 SUMMARY OF INSTRUMENTS

The sensor locations were the same as those used for the preoperational ILRT in 1974. Test instruments employed are described, by system, in the following subsections.

4.1.1. Temperature Indicating System

Overall system accuracy: ± 0.19°F

Overall system repeatability: ± 0.19°F

Components:

a. Resistance Temperature Detectors

Quantity	20
Manufacturer	Rosemount
Type	Model 104 AAN, 100 ohm, plantinum
Range, °F	60-100
Accuracy, °F	± 0.1
Repeatability, °F	± 0.1

b. Bridge Cards

Quantity	20
Manufacturer	Rosemount
Type	Model 440-L3
Range, ^{o}F	60-110
Accuracy, ^{o}F	± 0.25% of span
Repeatability, ^{o}F	± 0.25% of span

c. Digital Indicator

Quantity	1
Manufacturer	Weston
Type	Model 1230 *
Range, ^{o}F	60-110
Accuracy, ^{o}F	± 0.1
Repeatability, ^{o}F	± 0.1

* Modified for direct digital temperature readout

4.1.2 <u>Dewpoint Indicating System</u>

Overall system accuracy: ± 1.12^{o}F

Overall system repeatability: ± 0.52^{o}F

Components:

a. Dewcell Elements

Quantity	10
Manufacturer	Foxboro
Type	Model 2711AG, 18 carat gold
Range, ^{o}F	0-100
Accuracy, ^{o}F	± 1.0
Repeatability, ^{o}F	± 0.5

b. Dewpoint Recorder

Quantity	1
Manufacturer	Foxboro
Type	Model Y/ERB12
Range, ^{o}F	0-100
Accuracy, ^{o}F	± 0.5% of span
Repeatability, ^{o}F	± 0.15% of span

4.1.3 Pressure Monitoring System

Overall system accuracy: ± 0.015% of indicated pressure

Overall system repeatability: ± 0.001 psia

Precision Pressure Gauges

Quantity	2
Manufacturer	Texas Instruments
Type	Model 145-01
Range, psia	0-100
Accuracy, psia	± 0.015% of indicated pressure
Repeatability, psia	± 0.001% of full scale

4.1.4 Supplemental Test Flow Monitoring System

Overall system accuracy: ± 1% of full scale

Flow meter

Quantity	1
Manufacturer	Brooks
Type	Model 1114-08
Range, scfh at 0 psig and 100°F	30.9 - 309
Accuracy, scfh	± 1% of full scale

4.2 CALIBRA[illegible] CHECKS

Temperature, dewpoint, pressure and flow measuring systems were checked for calibration before the test in accordance with Metropolitan Edison Company Procedure 1430-Y-23, as recommended by proposed ANS-56.8, N274, Draft No. 2, February 1, 1978. The results of the calibration checks are on file at Three Mile Island Nuclear Station Unit 1. The supplemental test at 50.6 psig confirmed the instrumentation acceptability.

4.3 INSTRUMENTATION SELECTION

Justification of instrumentation selection was accomplished, using manufacturer's repeatability tolerances stated in Section 2.1, by computing the instrumentation selection guide (ISG) formula. Utilizing the methods, techniques and assumptions in Appendix G to proposed ANS-56.8, N274, Draft No. 2, dated February 1, 1978, the ISG was computed for the absolute method as follows:

a. Conditions

L_a = 0.1%/day

P = 65.3 psia

T = 74.5°F = 534.2°R dry bulb

Tdp = 61.8°F dewpoint

t = 24 hours (minimum expected test duration)

b. Total Absolute Pressure:

$$e_p = \pm \left[(0.001)^2\right]^{1/2} \Big/ \boxed{2}^{1/2}$$

e_p = ± 0.00071 psia

c. Water Vapor Pressure: e_{pv}

Sensor repeatability error (E): $\pm 0.5^{\circ}F$

Measurement system error (ε), excluding sensor: $\pm 0.15^{\circ}F$

$$E_{pv} = \pm 0.5^{\circ}F\ (0.0096\ psia/^{\circ}F)$$

$$E_{pv} = \pm 0.0048\ psia$$

$$\varepsilon_{pv} = \pm 0.15^{\circ}F\ (0.0096\ psia/^{\circ}F)$$

$$\varepsilon_{pv} = \pm 0.0014\ psia$$

$$e_{pv} = \pm \left[(E_{pv})^2 + (\varepsilon_{pv})^2\right]^{1/2} \Big/ \left[\text{no. of sensors}\right]^{1/2}$$

$$e_{pv} = \pm \left[(0.0048)^2 + (0.0014)^2\right]^{1/2} \Big/ \left[10\right]^{1/2}$$

$$e_{pv} = \pm 0.00158\ psia$$

d. Temperature: e_T

Sensor repeatability error (E): $\pm 0.1^{\circ}F = \pm 0.1^{\circ}R$

Measurement system error (ε), excluding sensor: $\pm 0.160^{\circ}F = \pm 0.160^{\circ}R$

$$e_T = \pm \left[(E_T^2) + (\varepsilon_T^2)\right]^{1/2} \Big/ \left[\text{no. of sensors}\right]^{1/2}$$

$$e_T = \pm \left[(0.1)^2 + (0.160)^2\right]^{1/2} \Big/ \left[20\right]^{1/2}$$

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$e_T = \pm\ 0.0422\ ^oR$

e. Instrumentation Selection Guide (ISG)

$$ISG = \pm \frac{2400}{t}\left[2\left(\frac{e_p}{P}\right)^2 + 2\left(\frac{e_{pv}}{P}\right)^2 + 2\left(\frac{e_T}{T}\right)^2\right]^{1/2}$$

$$ISG = \pm \frac{2400}{24}\left[2\left(\frac{0.00071}{65.3}\right)^2 + 2\left(\frac{0.00158}{65.3}\right)^2 + 2\left(\frac{0.0422}{534.2}\right)^2\right]^{1/2}$$

$$ISG = \pm\ 0.012\%/day$$

The ISG does not exceed 0.25 L_a (0.025%/day) and it is therefore concluded that the instrumentation selected was acceptable for use in determining the reactor containment integrated leakage rate.

4.4 SUPPLEMENTAL VERIFICATION

In addition to the calibration checks described in Section 4.2, test instrumentation operation was verified by a supplemental test subsequent to the completion of the 44.5 hour leakage rate test. This test consisted of imposing a known calibrated leakage rate on the reactor containment building. After the flow rate was established, it was not altered for the duration of the test.

During the supplemental test, the measured leakage rate was

$$L_c = L_{v'} + L_o$$

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where,

L_c = measured composite leakage rate consisting of the reactor building leakage rate plus the imposed leakage rate

L_o = imposed leakage rate

$L_{v'}$ = leakage rate of the reactor building during the supplemental test phase

Rearranging the above equation,

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$$L_{v'} = L_c - L_o$$

The reactor containment building leakage during the supplemental test can be calculated by subtracting the known superimposed leakage rate from the measured composite leakage rate.

The containment building leakage rate during the supplemental test ($L_{v'}$) was then compared to the measured reactor containment building leakage rate during the preceding 44.5 hour test (L_{am}) to determine instrumentation acceptability. Instrumentation is considered acceptable if the difference between the two building leakage rates is within 25 percent of the maximum allowable leakage rate (L_a).

4.5 INSTRUMENT ERROR CORRECTIONS

Subsequent to the 44.5 hour integrated leak rate test and the 8 hour supplemental leak rate test, calibration curves for the precision pressure gages, the RTD's (including readout) and the dewcells (including readout) were developed. These curves were developed using manufacturer's calibration data (for the precision pressure gages) and Metropolitan Edison Company Procedure, 1430-Y-23. Using these curves, each precision pressure gage reading, each RTD reading, and each dewcell reading was corrected for instrument error.

In addition, subsequent to testing, the Brooks flow meter was returned to the manufacturer for recalibration. Using this information, the flow meter readings taken during the supplemental leak rate test were corrected for instrument error.

5.0 TEST PROCEDURE

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5.1 PREREQUISITES

Prior to commencement of reactor containment building pressurization, the following basic prerequisites were satisfied:

a. Proper operation of all test instrumentation was verified.

b. All reactor containment building isolation valves were closed using the normal mode of operation. All associated system valves were placed in post-accident positions.

c. Equipment within the reactor containment building, subject to damage, was protected from external differential pressures.

d. Portions of fluid systems which, under post-accident conditions become extensions of the containment boundary, were drained and vented.

e. The penetration pressurization and fluid block systems were depressurized. Gauges were installed at penetration pressurization manifolds to provide means for detection of lekage into the system. These gauges were removed and the manifolds were vented prior to the start of the test.

f. Pressure gagues were installed on closed systems within containment to provide means for detection of leakage into such systems.

g. Local leakage rate testing of containment isolation valves and penetrations was concluded.

h. Potential pressure sources were removed or isolated from the containment.

i. All accessible liner weld channels (approximately 35 percent of the total) were vented to the containment atmosphere.

j. A general inspection of the accessible interior and exterior areas of the containment was completed.

5.2 GENERAL DISCUSSION

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Following the satisfaction of the prerequisites stated in Section 5.1, the reactor containment building pressurization was initiated at a rate of approximately 2.5 psi per hour. Building internal temperature was maintained at apprioximately 74°F. Building pressure and temperature were monitored hourly and the amperage required by the recirculation unit fans (AH-E-1A, 1B and 1C) was monitored every 5 psig. Leak rate testing was initiated at the 50.6 psig pressure level. Atmospheric pressure at time of leak rate test initiation was 14.37 psia. Forty-four and one half hours of data were collected at the 50.6 psig pressure level.

During the test the following occurred at half-hour intervals (See Appendix A):

a. Pressures indicated by each of the two precision gages were recorded and the average calculated.

b. The twenty RTD temperatures were recorded and the average calculated.

c. The ten dewpoint values were recorded. The average of the ten values was converted to vapor pressure using steam tables. This permitted correction of the total pressure to the partial pressure of air by subtracting the vapor pressure.

The use of vapor pressure (P_{wv}), average temperature (T) and the total pressure (P_T) is described in more detail in Section 6.1. All original data is on file at Three Mile Island Nuclear Station Unit 1. The plot of average temperature and weight of air was performed half hourly (See Appendix B).

When convenient, the available half-hourly values of P_{wv}, T and P_T were transmitted via on-site portable computer terminal to the Gilbert Associates, Inc. home office for analysis using the CLERCAL computer program. Computer program results, including a least squares fit of the data, were returned to the site via the terminal. A final computer run was made after data for a full 44.5 hour period was available.

Subsequent to the 44.5 hour leak test, a superimposed leakage rate was established for an additional 8 hour period. During this time, temperature, pressure and vapor pressure were monitored as described above.

5.3 TEST PERFORMANCE

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5.3.1 Pressurization Phase

Pressurization of the reactor building containment was started on April 12, 1978 at 1200. The pressurization rate was approximately 2.5 psi per hour. When containment internal pressure reached 12 psig, at 1807 on April 12, 1978, pressurization was secured. An inspection team entered containment to perform the 12 psig inspection. The 12 psig internal inspection was completely satisfactorily and pressurization was restarted at 1920 on April 12, 1978.

During pressurization to the 50.6 psig pressure level, the following observations were made:

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a. A slight buildup of pressure on several of the pressure gauges installed on penetration pressurization manifold indicated a small amount of leakage from the fuel transfer tube flanges, the personnel hatch, and emergency airlock door seals.

b. A decrease in the pressurizer level and an increase in the RB sump level was observed. This was investigated during the 12 psig inspection and was attributed to a ruptured tygon level tube which had been attached to the reactor coolant system cold leg. The tubing was then isolated which did not affect the integrated leak rate test.

c. The interspace between LR-V2 and the associated blind flange (on one of the two leak rate depressurization lines) was pressurized to containment pressure. No leakage was evident from the blind flange.

d. Pressure was slowly increasing in the interspace between LR-V3 and the associated blind flange (on the second leak rate depressurization line). To prevent leakage into this interspace, which would show up initially as containment leakage, LR-V3 was opened with the containment pressure at approximately 50 psig to equalize the interspace pressure. LR-V3 was then closed and no leakage was evident from the blind flange. Pressure was also slowly increasing in the purge exhaust interspace. This interspace was purposely equalized, using air from the purge exhaust air tank.

When containment internal pressure reached 50.7 to 50.8 psig at 1233 on April 13, 1978, pressurization was secured. All penetration pressurization system temporary manifold pressure gauges were removed.

5.3.2 <u>Integrated Leak Rate Testing Phase</u>

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After waiting 4 hours, leak rate testing was started at 1700 on April 13, 1978. From 1700 on April 13, 1978 until 1700 on April 14, 1978, a leakage rate at the upper bound of the 95 percent confidence interval of 0.069 percent per day was indicated by the data collected. With the addition of the local leakge rates for RB-V2*, RB-V7 and IC-V4, the total integrated leakage rate was 0.076 percent per day. Since this value exceeded the acceptance criteria of 0.075 percent per day and since the data had not been corrected for instrument error, the test was continued.

The pressurizer level was slowly decreasing and at 2025 on April 13, 1978, the casing drain on make-up pump 1C was found to be open. Water was flow from the drain at a rate of several gallons per minute to the auxiliary building sump. This water was leaking from the reactor coolant system through the make-up system check valves. These valves are not leak tight at low pressures. The loss of water from containment had a conservative effect on the indicated containment building integrated leakage rate.

Leak detection was initiated and potential leakage paths such as the outside purge exhaust valve were investiagted. No major source of leakage was discovered. At 0045 on April 15, 1978, during a valve lineup verification, it was discovered that control room indication for IC-V2 (inside containment isolation valve) was indicating open. IC-V2 was closed, however this had no effect on the containment leakage rate as the outside containment isolation valve, IC-V3, was holding.

At 1330 on April 15, 1978, the integrated leak rate test was concluded with an indicated containment integrated leakage rate of 0.064 percent per day based on 44.5 hours of data. The associated 95 percent confidence interval was 0.003 percent per day.

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5.3.3 Supplemental Leakage Rate Test Phase

After the 44.5 hour integrated leak rate test data was obtained and evaluated, and the leakage rate found to be acceptable, and a release permit had been obtained, a known leak rate was imposed on the reactor containment building through a calibrated flowmeter for a period of 8 hours.

5.3.4 Depressurization Phase

After all required data was obtained and evaluated, and the supplemental test results were found to be acceptable, and permission from the health physics department and unit superintendent was obtained, depressurization of the reactor containment building was started. A post test inspection of the building revealed no unusual findings.

5.3.5 Post-Test Leakage Repair

After the integrated leak rate test had been completed and during plant heatup, valve MS-V60A on the "A" steam generator was found to be blowing steam around its body/bonnet seal. The steam was blowing in a 360 degree arc to a distance of approximately eight feet. This secondary system leak path undoubtedly had a large effect on the observed containment building integrated leakage rate.

The valve had been replaced immediately prior to the integrated leak rate test and apparently its bonnet gasket was not installed. The leakage was corrected by injecting sealant into the bonnet joint.

6.0 METHODS OF ANALYSIS

6.1 GENERAL DISCUSSION

The absolute method of leakage rate determination was employed during testing at the 50.6 psig pressure level. The Gilbert Associates, Inc. CLERCAL computer code calculates the percent per day leakage rate using the mass point method of data analysis. The results presented are based on the mass point method.

The mass point method of computing leakage rates uses the following ideal gas law equation to calculate the weight of air inside containment for each half hour:

$$W = \frac{144\ PV}{RT} = \frac{KP}{T}$$

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where,

W = mass of air inside containment, lbm

$$K = 144\ V/R = 5.3983 \times 10^{6}\ \frac{\text{lbm} - {}^{o}\text{R} - \text{in.}^{2}}{\text{lbf}}$$

P = partial pressure of air, psia

T = average internal containment temperature, ^{o}R

$V = 2.0 \times 10^{6}\ ft^{3}$

The partial pressure of air, P, is calculated as follows:

$$P = \frac{P_{T1} + P_{T2}}{2} - P_{wv}$$

where,

P_{T1} = true corrected total pressure from PI-390, psia

P_{T2} = true corrected total pressure from PI-391, psia

P_{wv} = partial pressure of water vapor determined by averaging the ten dewpoint temperatures and converting to vapor pressure with the use of steam tables, psia

The average internal containment temperature, T, is calculated as follows:

$$T = \frac{\text{sum of 20 RTD's}}{20} + 459.69^{\circ}R$$

The weight of air is plotted versus time for the 44.5 hour test and for the 8 hour supplemental test. The Gilbert Associates, Inc. CLERCAL computer code fits the locus of these points to a straight line using a linear least squares fit. The equation of the linear least squares fit line is of the form $W = W_o + W_1 t$ where W_1 is the slope in lbm per hour and W_o is the weight at time zero. The least squares parameters are calculated as follows:

$$W_o = \frac{\Sigma t^2 \; \Sigma W_i - \Sigma t_i \; \Sigma t_i W_i}{S_{xx}}$$

$$W_1 = \frac{N \Sigma t_i W_i - \Sigma t_i \; \Sigma W_i}{S_{xx}}$$

where,

$$S_{xx} \quad N \Sigma t_i^2 - (\Sigma t_i)^2$$

The weight percent leakage per day can then be determined from the following equation:

$$\text{wt. \%/day} = \frac{-2400\ W_1}{W_o}$$

where the negative sign is used since W_1 is a negative slope to express the leakage rate as a positive quantity.

6.2 STATISTICAL EVALUATION

After performing the least squares fit, the CLERCAL computer code calculates the following statistical parameters:

a. Standard error of confidence for the curve fit (S_e).

b. Limits of the 95 percent confidence interval for the curve fit.

c. Limits of the 95 percent confidence interval for the leakage rate (C_L).

The significance of the measured leakage rate can then be evaluated in view of the number of data points exceeding the limits of the 95 percent confidence interval and by the magnitude of the upper bound of the 95 percent confidence interval for the leakage rate.

Standard error of confidence is defined as follows:

$$S_e = \left[\frac{\Sigma \left[W_i - (W_o + W_1 t_i) \right]^2}{N-2} \right]^{1/2}$$

where,

W_i = observed mass of air

$(W_o + W_1 t_i)$ = least squares calculated mass of air

N = number of data points

This parameter is an expression of the difference between an observed and a calculated (least squares) mass point. The 95 percent confidence interval of the fix is twice the standard error of confidence ($2S_e$). The "degree-of-fit" is evaluated by determining the number of data points, W_i, not falling in the interval $(W_o + W_1 t) \pm 2S_e$.

The 95 percent confidence limit for the mass leakage rate is calculated as follows:

$$C_L = t_{95} S_e \left[\frac{N}{S_{xx}} + \frac{S_{xx} + (\Sigma t_i)^2}{NS_{xx}} \right]^{1/2}$$

where,

t_{95} = Student's t distribution with N-2 degrees of freed

This parameter is an expression of the uncertainty in the measured leakage rate.

7.0 DISCUSSION OF RESULTS

7.1 RESULTS AT P_a

The method used in calculating the mass point leakage rate is defined in Section 6.0. The result of this calculation is a mass point leakage rate using absolute values corrected for instrument error of 0.061 %/day.

The 95 percent confidence limit associated with this leakage rate is 0.003 percent per day. Thus, the leakage rate at the upper bound of the 95 percent confidence interval becomes

$$L_{am} = 0.061 + 0.003 \text{ \%/day}$$

$$L_{am} = 0.064 \text{ \%/day}$$

The measured leakage rate at the upper bound of the 95 percent confidence level is below the acceptance criteria of 0.075 percent per day (0.75 L_a). A comparison of each of the observed weights with the weights calculated using the least squares line reveals only two of the ninety data points do not lie within the 95 percent confidence interval. Therefore, reactor containment building leakage at the calculated design basis accident pressure (P_a) of 50.6 psig is considered to be acceptable.

7.2 SUPPLEMENTAL TEST RESULTS

After conclusion of the 44.5 hour test at 50.6 psig, flowmeter FI-111 was placed in service and a flow rate, corrected for instrument error, of 193.9 SCFH was established. This flow rate is equivalent to a leakage rate of 0.053 percent per day. After the flow was established, it was not altered for the duration of the supplemental test.

The measured leakage rate (L_c) using absolute values corrected for instrument error during the supplemental test was calculated to be 0.111 percent per day using the mass point method of analysis. The 95 percent confidence interval associated with this leakage rate is 0.037 percent per day. None of the 17 data points is out of confidence.

The building leakage rate during the supplemental test is then determined as follows:

$$L_{v'} = L_c - L_o$$

$$L_{v'} = 0.111\ \%/\text{day} - 0.053\ \%/\text{day}$$

$$L_{v'} = 0.058\ \%/\text{day}$$

Comparing this leakage rate with the building leakage rate measured during the 24 hour test yields the following:

$$\frac{|L_{am} - L_{v'}|}{L_a} = \frac{|(0.061) - (0.058)|}{0.10} = 0.03$$

The building leakage rates agree within 3.0 percent of L_a which is well below the acceptance criteria of 25 percent of L_a. Therefore, the acceptability of the test instrumentation is considered to have been verified.

8.0 TYPE B AND C LEAKAGE RATE HISTORIES

Refer to Appendicies C and D for the report on Type B and C testing performed since the previous Type A test.

9.0 REFERENCES

1. SP 1303-6.1, "Reactor Building Integrated Leak Rate Test", Metropolitan Edison Company Surveillance Procedure.

2. Code of Federal Regulations, Title 10, Part 50, Appendix J, (1-1-75).

3. ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors", American Nuclear Society, (March 16, 1972).

4. Steam Tables, American Society of Mechanical Engineers, (1967).

5. CLERCAL, Computer Code, Gilbert Associates, Inc.

6. 1430-Y-23, "Reactor Building Integrated Leak Rate Test Instrument Calibrations", Metropolitan Edison Company Procedure.

7. ANS 56.8, N274, "Containment System Leakage Testing Requirements", American Nuclear Society, (Draft No. 2 - February 1, 1978).

APPENDICES

APPENDIX A

REDUCED LEAKAGE DATA

APPENDIX A

REDUCED TEST DATA

	Time	Containment Total Pressure (psia)	Partial Pressure Water Vapor (psia)	Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
4/13/78	1700	65.050	0.292	534.2	654405.94
	1730	65.046	0.294	534.2	654344.29
	1800	65.035	0.291	534.1	654386.96
	1830	65.032	0.291	534.1	654356.64
	1900	65.025	0.289	534.1	654306.10
	1930	65.027	0.291	534.1	654306.10
	2000	65.025	0.290	534.1	654295.99
	2030	65.018	0.290	534.1	654225.24
	2100	65.012	0.289	534.0	654297.21
	2130	65.007	0.288	534.0	654257.79
	2200	65.005	0.290	534.0	654216.34
	2230	65.004	0.288	534.0	654227.46
	2300	64.999	0.286	534.0	654197.13
	2330	64.994	0.285	533.9	654279.22
	2400	64.98[illegible]	0.285	533.9	654218.55
4/14/78	0030	64.984	0.286	533.8	654290.5[illegible]
	0100	64.974	0.285	533.8	654199.53
	0230	64.967	0.284	533.7	654261.42
	0200	64.962	0.283	533.7	654220.96
	0230	64.964	0.281	533.7	654261.42
	0300	64.961	0.282	533.7	654220.96
	0330	64.956	0.282	533.6	654292.98
	0400	64.956	0.281	533.6	6543[illegible].10
	0430	64.947	0.279	533.6	654231.27

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APPENDIX A (Cont'd)

REDUCED TEST DATA

Time	Containment Total Pressure (psia)	Partial Pressure Water Vapor (psia)	Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
0500	64.947	0.281	533.6	654212.05
0530	64.941	0.275	533.5	654332.65
0600	64.933	0.276	533.5	654242.59
0630	64.927	0.283	533.5	654112.06
0700	64.924	0.281	533.4	654224.57
0730	64.919	0.281	533.4	654173.97
0800	64.912	0.279	533.3	654245.01
0830	64.908	0.279	533.3	654204.52
0900	64.902	0.283	533.3	654104.31
0930	64.898	0.274	533.3	654152.90
1000	64.892	0.277	533.2	654185.47
1030	64.889	0.275	533.2	654174.34
1100	64.884	0.278	533.2	654094.35
1130	64.886	0.278	533.2	654114.60
1200	64.887	0.275	533.3	654031.43
1230	64.884	0.277	533.2	654104.48
1300	64.882	0.278	533.2	654074.11
1330	64.886	0.278	533.3	653991.95
1400	64.890	0.275	533.3	654061.79
1430	64.892	0.274	533.4	653969.53
1500	64.891	0.274	533.4	653959.41
1530	64.893	0.274	533.4	653979.65
1600	64.892	0.275	533.4	653959.41
1630	64.894	0.276	533.4	653970.55

APPENDIX A (Cont'd)

REDUCED TEST DATA

	Time	Containment Total Pressure (psia)	Partial Pressure Water Vapor (psia)	Containment Temperature (°R)	Weight of Containment Air (lbm)
	1700	64.897	0.273	533.5	653907.66
	1730	64.898	0.276	533.5	653888.44
	1800	64.896	0.277	533.5	653858.08
	1830	64.893	0.273	533.5	653867.19
	1900	64.889	0.272	533.4	653958.40
	1930	64.884	0.270	533.4	653927.03
	2000	64.883	0.270	533.4	653916.91
	2030	64.882	0.272	533.4	653887.56
	2100	64.883	0.273	533.4	653888.57
	2130	64.881	0.272	533.4	653877.44
	2200	64.881	0.271	533.4	653887.56
	2230	64.881	0.273	533.4	653868.33
	2300	64.881	0.273	533.4	653868.33
	2330	64.880	0.273	533.4	653858.21
	2400	64.879	0.273	533.4	653848.09
4/15/78	0030	64.875	0.273	533.4	653807.61
	0100	64.871	0.275	533.4	653746.88
	0130	64.867	0.271	533.3	653868.45
	0200	64.863	0.273	533.3	653808.73
	0230	64.861	0.273	533.3	653788.49
	0300	64.859	0.275	533.3	653748.00
	0330	64.855	0.274	533.2	653840.23
	0400	64.852	0.274	533.2	653809.86
	0430	64.846	0.271	533.2	653778.47

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APPENDIX A (Cont'd)

REDUCED TEST DATA

	Time	Containment Total Pressure (psia)	Partial Pressure Water Vapor (psia)	Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
	0500	64.844	0.271	533.2	653758.23
	0530	64.843	0.272	533.2	653737.98
	0600	64.840	0.271	533.1	653840.35
	0630	64.834	0.271	533.1	653779.60
	0700	64.830	0.272	533.1	653728.97
	0730	64.826	0.269	533.1	653717.83
	0800	64.825	0.269	533.1	653707.70
	0830	64.820	0.266	533.0	653808.07
	0900	64.813	0.269	533.0	653708.81
	0930	64.803	0.269	532.9	653730.18
	1000	64.794	0.266	532.8	653790.06
	1030	64.790	0.266	532.8	653749.53
	1100	64.776	0.266	532.7	653730.38
	1130	64.772	0.268	532.7	653670.59
	1200	64.767	0.267	532.7	653630.06
	1230	64.765	0.266	532.7	653618.91
	1300	64.757	0.265	532.6	653670.68
	1330	64.758	0.269	532.6	653642.30
			SUPERIMPOSED TEST		
4/15/78	2100	64.708	0.265	532.4	653419.40
	2130	64.708	0.267	532.4	653400.13
	2200	64.705	0.265	532.4	653378.84
	2230	64.701	0.265	532.3	653480.29
	2300	64.694	0.266	532.3	653390.03
	2330	64.690	0.266	532.3	653349.46

APPENDIX A (Cont'd)

REDUCED TEST DATA

	Time	Containment Total Pressure (psia)	Partial Pressure Water Vapor (psia)	Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
	2400	64.685	0.266	532.2	653421.51
4/16/78	0030	64.679	0.265	532.2	653370.79
	0100	64.672	0.268	532.2	653270.38
	0130	64.671	0.266	532.1	653402.28
	0200	64.666	0.265	532.1	653361.70
	0230	64.665	0.264	532.2	653248.06
	0300	64.662	0.267	532.1	653301.84
	0330	64.659	0.268	532.1	653261.26
	0400	64.654	0.266	532.1	653229.81
	0430	64.650	0.265	532.1	653208.50
	0500	64.645	0.265	532.1	653157.78

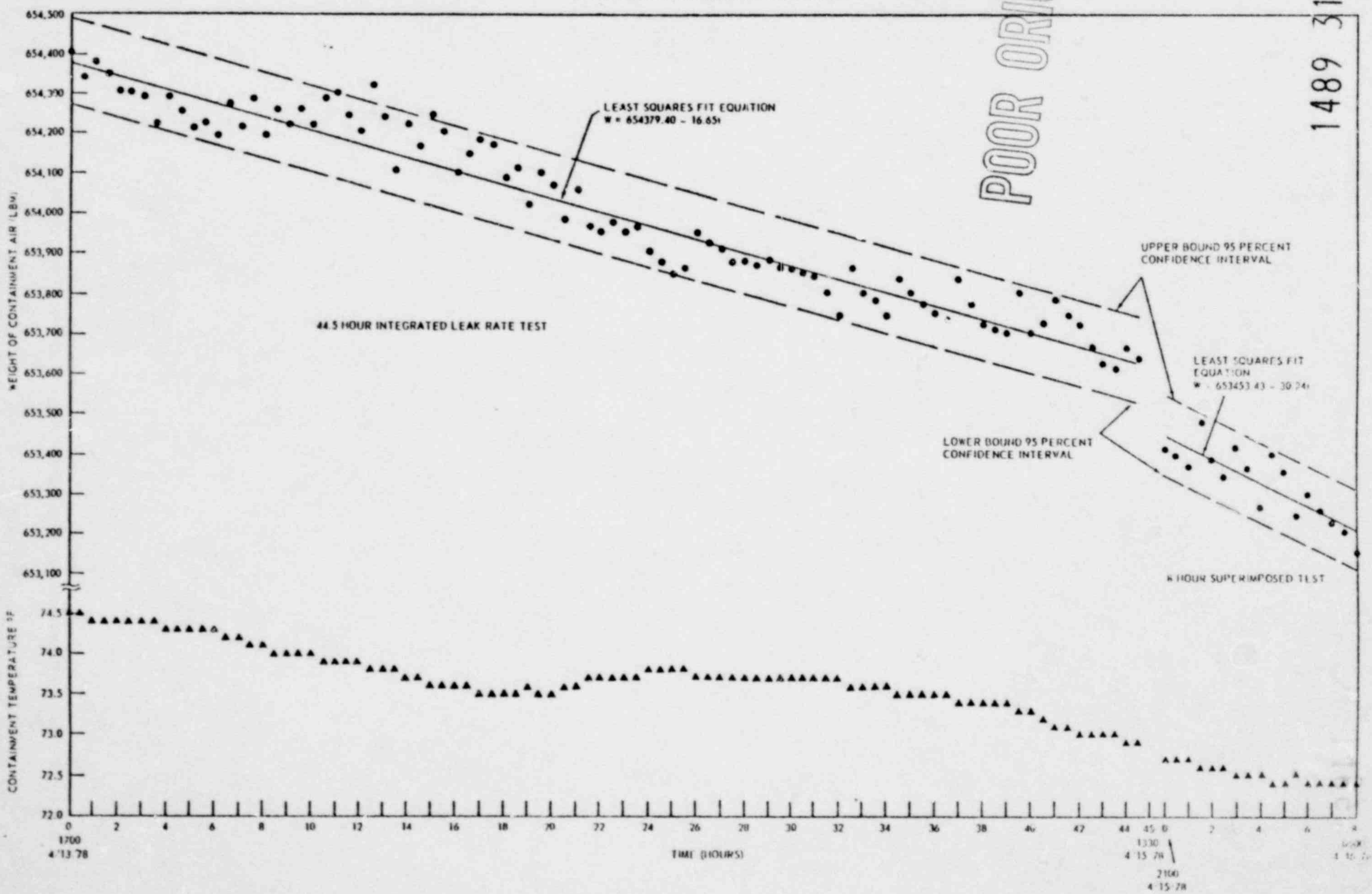
APPENDIX B

WEIGHT OF CONTAINMENT AIR AND

AVERAGE CONTAINMENT TEMPERATURE

APPENDIX B

WEIGHT OF CONTAINMENT AIR AND AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME



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APPENDIX C

THREE MILE ISLAND UNIT 1

1978 REFUELING

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

SP 1303-11.18

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APPENDIX C

THREE MILE ISLAND UNIT 1

1978 REFUELING

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

SP 1303-11.18

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INDEX

ATTACHMENTS

REACTOR BUILDING LOCAL LEAK RATE TESTING NRC REPORT

1978 REFUELING

POOR ORIGINAL

1. PURPOSE

1.1. To provide analysis to the Nuclear Regulatory Commission on the third periodic type B and type C leakage tests performed along with the second periodic integrated leak rate test of Three Mile Island Unit 1 reactor building.

This is in accordance with "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," Appendix J, Part 50, Title 10 Code of Federal Regulations which required the contents of this summary report to become part of the type A test report along with the details of any other type B and type C testing performed since the previous type A test. (Also required per technical specification 4.4.1.1.8)

2. SUMMARY OF WORK ACCOMPLISHED

2.1. Testing

Reactor building refueling frequency local leak rate testing was performed on the containment isolation valves and penetrations listed in the technical specifications and those additionally committed to be tested per Reference 2. Twelve (12) valves (IA-V6/20, SA-V2/3, LR-V1/2/3/4/5/6/49) and four other devices (equipment access flange, Penetration 241, Fuel Transfer Tube Flanges) which were previously tested by quarterly penetration pressurization system flow meter readings were tested this year by Type C (Appendix J) test methods. A total of approximately eighty-one (81) seat and/or packing leak tests were performed, nine (9) as retests after repairs. Three (3) of the containment isolation valves had higher seat and/or packing leakage than the cognizant engineer could accept and repairs were performed.

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2.2. VALVE REPAIRS

Two (2) gate valves (IC-V4, LR-V2) required refinishing of seating surfaces. The seats in one ball valve (CM-V2) and one butterfly valve (AH-V1B) were replaced. AH-V1B had satisfactory leakage as-found but the rubber seats had slight cracking. The packing was replaced in seven (7) valves (LR-V1, 2, 3, 4, 5, 6, 49)

3. METHODS OF TESTING

Testing was performed by use of TMI Unit 1 surveillance procedure SP 1303-11.18 Reactor Building Local Leak Rate Testing. This procedure gives detailed guidance on the test equipment and methods to be used for each penetration/valve. The following general philosophy is contained in the surveillance procedure.

1. Use air or nitrogen at a pressure differential across the valve greater than P_a (Calculated accident pressure)
2. Assure that the pressure is exerted in the accident test direction unless it can be demonstrated that pressurizing in the opposite direction is as conservative.
3. Assure that the test volume is drained of liquid so that air or nitrogen test pressure is against valve seats.
4. Assure that the test verifies valve packing integrity.
5. Assure adequate time period for stabilization of test conditions.
6. Assure test equipment is calibrated and used in a manner consistent with the data accuracy desired. (Weekly meter standardization was performed to verify meters accurate within ± 5% full scale. MP 1430-Y-23)
7. Assure that the fluid blocking system is drained and vented during tests on the associated containment isolation valves to prevent any effects it might have on the test results. (The majority of the F. B. system is seismic 3)
8. Assure valves to be tested are closed by the normal method prior to testing.
9. Document as-found conditions (prior to adjustments/repairs) and as-left conditions.
10. Record test instrument <u>scale</u> readings prior to doing any data corrections.

11. Perform test rig bypass valve tests weekly.

12. Assure that system drains and vents which could serve as containment isolation valves, are closed and capped and tagged after completion of the test program.

A training program prior to the refueling outage was also performed to help assure that the above philosophy was understood by the personnel involved in the testing.

4. TEST EQUIPMENT (See Figure 1)

Brooks Model 1114 01F'A1A rotometers were used to measure the supply and/or vent flow rate for each valve and penetration (except for the purge valves which were tested by pressure drop methods). These flow meters are fitted with 0 - 150 mm scales and have quick-disconnect couplings to allow switching meters for proper scale. The range of the meters for both zero and fifty five psig metering conditions is given on Figure #1, which also shows the valving, tubing and other controls for the testing apparatus. The flow rotometers were standardized once a week against identical lab meters which had been factory calibrated prior to the outage. (See Reference 1)

The testing apparatus also included calibrated pressure gages for regulation of proper test pressure and thermometers to allow correction of readings for significant variations from calibration conditions.

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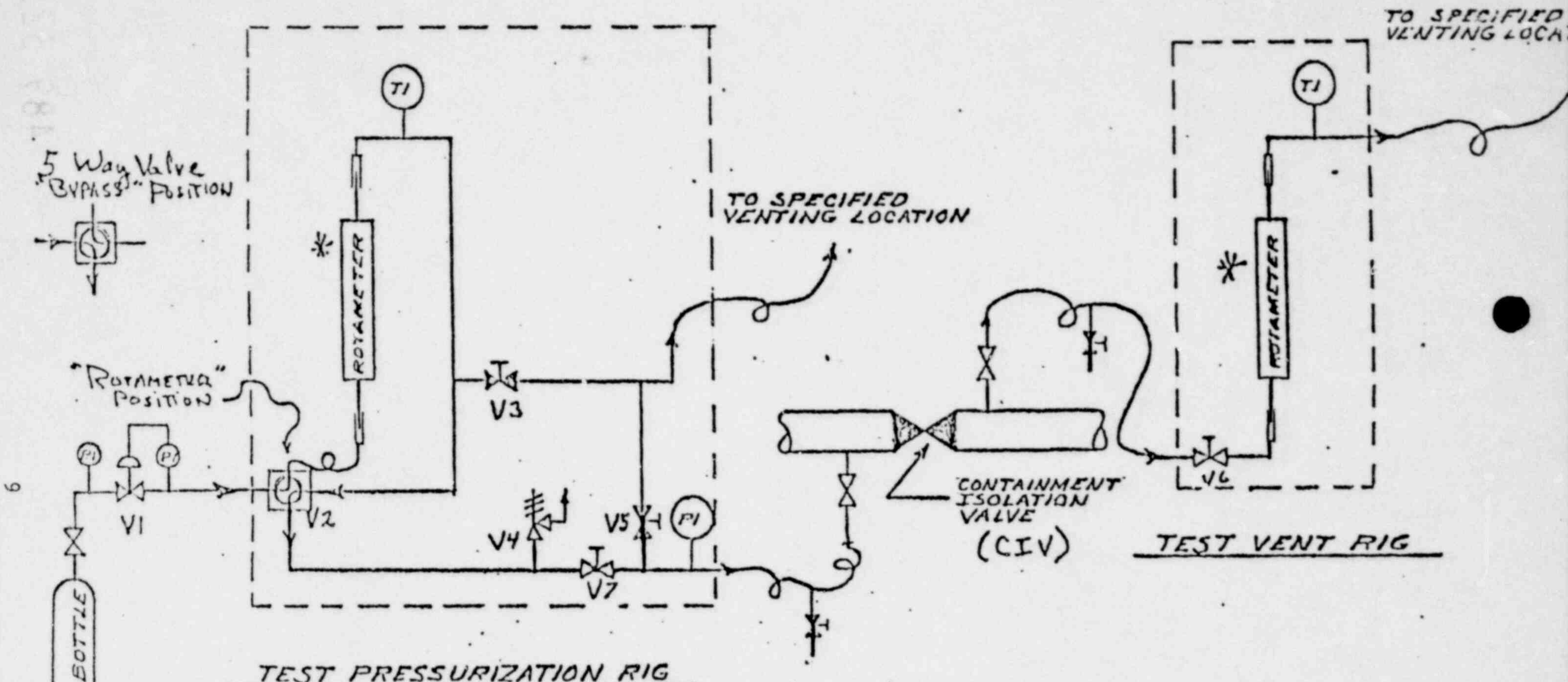

TEST PRESSURIZATION RIG

Equipment

Reactor plant grade N_2 supply

Pressure gage 0-100 psi ± 1 psi accuracy

Temperature Indicators 25-125°F ± 2°F accuracy

* Test Rotameters (Note: these rotameters are fitted with quick-connect fittings at inlet and outlet)

Brooks	@ 0 psi	@ 55 psi
R-2-15-AA / Tantalum	10-480 SCCM	140-1470 SCCM
R-2-15-C / Saphire	100-5300 SCCM	200-13,100 SCCM
R-6-15-B / Carboloy	1000-61000 SCCM	2000-182000 SCCM
Fisher & Porter 035-1/16-SC	10-80 SCCM	
118-1/8-SS	150-1500 SCCM	
334-1/4-SS	1000-31000 SCCM	

ISOLATION VALVE TEST RIG

3/24/76

N_2 TEST RIG

FIGURE # 1

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5. ANALYSIS OF RESULTS AS-FOUND/AS-LEFT (See Attachment II for data)

"As-found" leakage data were recorded on an individual data sheet for each valve/penetration tested. The data sheet was signed by the Test Foreman and a Cognizant Engineer.

Retesting was performed for those valves which were repaired.

5.1. Interpretation of Data

5.1.1 As-found leakage Results (Also see Attachment II)

The "as-found" total Reactor Building local leakage is shown in the below table along with a comparison to Technical Specifications criteria.

AS-FOUND TOTAL REACTOR BUILDING LOCAL LEAKAGE

Type Test	Total Leakage	Tech. Spec/ FSAR Limit	Percent Tech. Spec./FSAR Limit	Remarks
N2/Air	69,179 sccm	104,846 sccm	66%	

NOTE: The totals shown are cumulative by penetration and not the total of all valves, i.e., highest valve(s) on penetrations added.

Example: Penetration XYZ has one containment isolation valve inside the reactor building and one outside the reactor building. One valve leaks 500 sccm and the other leaks 1,000 sccm. The leakage for the penetration is 1,000 sccm not 1,500 sccm. The maximum leakage which can be forced through the worst valves at a pressure of Pa is still 1,000 sccm.

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5.1.2 As-left Leakage Results (Also see Attachment II)

(Subsequent to repair/maintenance) The existing combined reactor building local leakage is shown below. Comparison to FSAR limit is also given.

AS-LEFT REACTOR BUILDING LOCAL LEAKAGE

Type Test	Total Leakage	Tech. Spec. Limit	Percent Tech. Spec. Limit	Remarks
N2/Air	50,163 sccm	104,846 sccm	47.8%	

The total shown is cumulative by penetration and not the total of all valves tested. (See discussion note of section 1.1)

5.2. Error Analysis

The flowmeters used in the field have normal industrial accuracies of ± 2% full scale in the 10-100% scale range. However, weekly comparisons of these meters with lab meters were done to verify better than ± 5% full scale accuracy. The lab meters were certified as ± 1% full scale accuracy from 10-100% F.S. by the manufacturer. See Ref. 6.1. for the meter Standardization Procedure.

The usable scale range for the field meters and the lab meters was 15-150 m millimeters.

The relationship used to determine meter accuracy from standardization data was as follows:

$$\%\ \text{Field Meter Accuracy} = \sqrt{(\text{Lab meter accuracy})^2 + (\text{Largest deviation})^2 \text{ or } (\text{Industrial Accuracy})^2 \text{ whichever is largest}}$$

In cases where this calculated value exceeded 5%, (it was normally approximatel; 3%) or where the meter float did not move freely when the meter was turned alternately upside down and then right side up, the meter was dissasembled, cleaned, repaired, and then reassembled and retested.

The scale readings on the data sheets were evaluated using SP1303-11.18 Enclosure 9 (See Attachment I).

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6. REFERENCES

6.1. 1430-Y-23 Standardization of Flow Rotometers

6.2. Met-Ed to NRC Licensing Letter 9/17/75 - Comparison of TMI 1 Tech. Spec. with Appendix J - 10 CFR 50

6.3. SP 1303-11.18 Reactor Building Local Leak Rate Testing

6.4. Three Mile Island Unit 1 Technical Specification 4.4.1

6.5. TMI Surveillance File (for Data Sheets)

ATTACHMENTS

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ATTACHMENT I

RESULTS EVALUATION PROCEDURE

(SP1303-11.18 Enclosure 9)

RESULTS EVALUATION

The vent rotameter reading will be used if it can be demonstrated by the test data that all significant CIV leakage is being accounted for. If CIV packing, fluid block check valve, or gasket leakage was evident the supply rotameter results will be used unless this non-seat leakage was measured reliably and documented.

POOR ORIGINAL

FOR USE OF SUPPLY ROTAMETER DATA: Procedure:	FOR USE OF VENT ROTAMETER DATA: Procedure:
a) Record supply meter reading in (1) below*. Also identify the meter used by tube # in (8) below and the metering pressure in (9). b) Convert meter units to SCCM units using latest lab meter calibration curve. Enter in (3) below. c) Correct results for temperature. Enter supply temperature in (4) below. Calculate and enter in (7) below. * If meter scale reading was less than 15mm (minimum scale) use 15mm in calculations.	a) Record vent meter reading in (1) below*. b) Record downstream verification meter reading in (2) below. Also identify the respective meters used in (8) below and the metering pressures in (9). c) Convert meter units to SCCM units using latest lab meter calibration curve. Enter in (3) below. d) Correct results for temperature. Enter vent temperature (°F) in (4) below. then Calculate and enter in (5) below. e) If measurements of any other significant leakage paths (fluid block check valve, packing) are being claimed enter corrected flow (SCCM) in (6) below.

$$\left(\underset{(1)}{\underline{\qquad}} \overset{(mm)}{+} \underset{(2)}{\underline{\qquad}}\right) \xrightarrow{\text{convert}} \left(\underset{(3)}{\underline{\qquad}} \overset{(SCCM)}{+} \underline{\qquad}\right) \times \sqrt{\frac{530}{\underset{(4)}{\underline{\qquad}} + 460}} = \underset{(5)}{\underline{\qquad}}\ \text{SCCM}$$

(8)(Identify meters used)

$+ \underset{(6)}{\underline{\qquad}}$ SCCM

@ ______ (9)(Meter Pressures)

= CIV Leakage $\underset{(7)}{\underline{\qquad}}$ SCCM

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ATTACHMENT II

DATA 1978 REFUELING

REACTOR BUILDING LEAK RATE TESTING

1489 330

REACTOR BUILDING LOCAL LEAK RATE TESTING
SP-1303-11.18
1978 REFUELING RE[illegible]TS

May 11, 1978
9:10 AM

TAG	TARGET	ASFD78	COMMENTS	RETEST1	RETEST2	ASLT78	ASLTDATE
********	******	******	********	*******	*******	******	********
AH-V1A/B	1500	1693	NEWSEATB			2613	2/78
AH-V1C/D	1500	3393				3393	3/78
CA-V1	150	170				170	3/25/78
CA-V2	600	101				101	3/31/78
CA-V3	150	170				170	3/25/78
CA-V4A	150	129				129	3/27/78
CA-V4B	150	109				109	3/27/78
CA-V5A	600	238				238	3/27/78
CA-V5B	600	238				238	3/28/78
CA-V13	150	102	REPAIRED	128		128	3/31/78
CA-V189	600	1497	HIGH			1497	3/19/78
CF-V2A	75	102				102	3/22/78
CF-V2B	75	102				102	3/22/78
CF-V12A	225	102				102	3/22/78
CF-V12B	225	102				102	3/22/78
CF-V19A	300	2288	HIGH			2288	3/22/78
CF-V19B	300.	350				350	3/22/78
CF-V20A	300	102				102	3/22/78
CF-V20B	300	102				102	3/22/78
CM-V1	100	188				188	3/24/78
CM-V2	100	2964	REPAIRED	129		129	3/29/78
CM-V3	100	193				193	3/24/78
CM-V4	100	168				168	3/24/78
DH-V64	75	50				50	4/8/78
DH-V69	225	280				280	4/8/78
HP-V1	400	229				229	3/19/78
HP-V6	400	102				102	3/19/78
IA-V6/20	100	50				50	4/10/78
IC-V2	800	1544	HIGH			1544	3/21/78
IC-V3	800	131				131	3/21/78
IC-V4	800	19256	REPAIRED	30170	30000	1191	4/11/78
IC-V6	400	48				48	3/26/78
LR-V1	500	50				50	3/28/78
LR-V2	500	7686	REPAIRED	55400	8565	8565	4/7/78
LR-V3	500	1005				1005	3/28/78
LR-V4	50	452	HIGH			452	3/28/78
LR-V5	50	492	HIGH			492	3/28/78
LR-V6	50	95				95	3/28/78
LR-V49	500	50				50	3/28/78
MU-V2A	225	159				159	3/19/78
MU-V2B	225	102				102	3/20/78
MU-V3	900	102				102	3/19/78
MU-V18	900	408				408	3/16/78
MU-V20	600	102				102	3/19/78
MU-V25	150	102				102	3/21/78
MU-V26	600	102				102	3/21/78
MU-V116	225	301				301	3/20/78
NS-V4	800	294				294	3/31/78
NS-V15	800	69				69	3/26/78
NS-V35	800	294				294	3/31/78
RB--V2A	800	296				296	3/31/78
RB-V7	800	10160	HIGH			10160	3/25/78
SA-V2/3	100	50				50	4/10/78

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Attachment II (continued)

TAG	TARGET	ASFD78	COMMENTS	RETEST1	RETEST2	ASLT78	ASLTDATE
********	******	*******	********	*******	*******	*******	********
SF-V23	600	102				102	3/18/78
WDG-V3/4	400	5886	HIGH			5886	3/25/78
WDL-V303	600	130				130	3/28/78
WDL-V304	600	130				130	3/28/78
WDL-V534	1600	130				130	4/2/78
WDL-V535	1600	130				130	4/2/78
PENET104	0	0				0	4/11/78
PENET105	0	0				0	4/7/78
PENET106	0	0				0	4/7/78
PENET210	0	0				0	4/7/78
PENET211	0	0				0	4/7/78
PENET241	0	0				0	4/4/78
FTTEAST	0	95				95	4/7/78
FTTWEST	0	95				95	4/7/78
EQPFLG	0	51				51	4/4/78
PERACCES	1400	1849				1849	6/25/77
EMEACCES	1400	4163				4163	5/17/77
TOTAL	30625	71425				52350	
PENTOTAL		69179				50163	
ACC CRIT		104846				104846	

LRTERMS - TERMINOLOGY USED IN COMPUTER PROGRAM FOR LOCAL LEAK RATE TESTING RESULTS.

1) .01 - (ALONE) MEANS NO DATA AVAILABLE.
2) .01 - (OR ANY DECIMAL VALUE) AFTER A LEAK RATE (I.E.59500.01) MEANS ACTUAL LEAK RATE GREATER THAN MEASURED/RECORDED VALUE.
3) TARGET- ADMINISTRATIVE LEAKAGE LIMIT BASED ON TESTING EXPERIENCE. COMPLETE EXPLANATION GIVEN IN SP1303-11.18.
4) ASFD__ - LEAK RATE (SCCM) IN THE AS-FOUND VALVE CONDITION, BEFORE ANY REPAIRS OR ADJUSTMENTS. FOR THE DESIGNATED YEAR.
5) ASLT__ - LEAK RATE (SCCM) ATTAINED AFTER ANY ADJUSTMENTS/REPAIRS.
6) DESC - DESCRIPTION OF VALVE OR PENETRATION.
7) SIZES - PIPE DIAMETER (INCHES) FOR VALVE/PENETRATION.
8) RUNTOTAL- RUNNING TOTAL. THIS IS THE LIST OF LEAKAGES WHICH IS USED FOR DETERMINATION OF REPORTABILITY. A NEW ASFD LEAKAGE REPLACES THE PREVIOUS YEARS ASLT LEAKAGE. RETEST RESULTS ARE NOT INCLUCED UNTIL AFTER DETERMINING REPORTABILITY

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APPENDIX D

THREE MILE ISLAND UNIT 1

MISCELLANEOUS LEAK TESTING SINCE PREVIOUS

TYPE A TEST (APRIL 1977 TO APRIL 1978)

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

1489 333

APPENDIX D

INDEX

1. PURPOSE
2. SUMMARY
3. METHODS
4. TEST EQUIPMENT
5. ANALYSIS OF RESULTS - AS FOUND/AS LEFT
6. DATA

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1. PURPOSE

To provide analysis to the Nuclear Regulatory Commission on the various non refueling frequency Type B and Type C leakage tests performed on Three Mile Island Unit 1 reactor building since the previous Type A test.

1489 335

2. SUMMARY OF TESTING PERFORMED AND REPAIRS

B. 1. Penetration Pressurization System quarterly flow meter readings SP 1303-11.24

TESTING

Quarterly readings of the installed system flow rotometers were taken and compared to the acceptance criteria specified in the procedure.

The acceptance criteria deals with:

a. Limits on individual manifold flows

b. On the total system flow and

c. On the correspondance between the sum of individual manifold flow indicators and the indication on the supply to the system.

Test dates and results are shown in section 6.1.

REPAIRS

The repairs performed on the system were in every case repairs to the penetration pressurization system and not to the penetration seals. The repairs normally consisted of tightening or replacing tubing fittings.

1489 336

2.2. Access Hatches (Air Locks) Door Seal Tests

TESTING

Door seal tests were performed routinely after door usage as required by Technical Specification 4.4.1.2.5b. These tests were performed per SP 1303-11.25 which requires pressurization to P_a (calculated accident test pressure) and the reading of a supply rotometer.

The supply flow meter readings were compared against an arbitrary 3 SCFH target criteria to determine the need for repairs. Reportability was based on indication of leakage through one of the doors of greater than 60 SCFH (The range of the installed flow indication.

REPAIRS

Failures of the door seals were generally followed by cleaning, lubrication and inspection of seals. If that did not eliminate the leakage problem, door adjustments were made, or seals were replaced. Due to a test pressure requirement which exceeds the manufacturer's recommended pressure, the door seals quite likely have failed leak tests at times when they would have been very capable of sealing in the accident pressure direction. The periodic door seal tests are not a realistic mock-up of accident conditions but are much more demanding on the equipment.

2.3. Access Hatches-Integrated Leakage (all seals)

TESTING

Integrated hatch tests, by pressure drop, were performed semi-annually as required by Technical Specification 4.4.1.2.5b. These tests were performed per SP 1303-11.18C which requires pressurization to Pa (calculated accident test pressure) and a four hour pressure drop test.

The dates of testing and results are shown in report section 6.3

The calculated leak rates were used to update the total for Reactor Building Local Leakage and the updated total was the basis for reportability. A target criteria of 1400 SCCM was established to give some basis for requiring repairs.

REPAIRS

Failures of the hatch integrated tests (as determined by the cognizant engineer) were always followed by tubing fitting adjustments and door seal cleaning and lubrication. Tightening of shaft seals was performed on the basis of local leak checks.

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3. METHODS OF TESTING

3.1. Penetration Pressurization System

Installed system flow rotameters are read once per quarter and the readings are compared to the procedure acceptance criteria.

3.2. Access Hatches - Door Seals

Door seal tests are performed by pressurizing between the concentric rubber seals with 55 psig metered air. The supply flow rate is read, recorded, and evaluated. High leakage could be indicative of door seal problems, or problems with numerous other hatch penetrations which are pressurized from the same supply.

3.3. Access hatches - Integrated

Integrated hatch tests are performed by pressurizing the hatch interior to P_a (calculated accident pressure) or greater and observing the pressure drop over a four hour period.

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4. TEST EQUIPMENT

4.1. Penetration Pressurization System

Flow Rotometers

Manufacturer - Brooks Inst. Co.

Model #1114

Range 0 - 9 SCFH (individual manifolds)

0 - 60 SCFH (Supply to system)

4.2. Access Hatches - Door Seals

Flow Rotometers

Manufacturer - Brooks Inst. Co.

Model #1114

Range 0 -9 SCFH

Accuracy ± 2% Full Scale

Pressure Regulator

Manufacturer - Fisher Control Co.

4.3. Access Hatches - Integrated Leakage

Pressure Gauge (Temporary for Integrated Test)

Accuracy ± 0.25% or better

Range 0 - 60 psig with 0.01 psi scale divisions

Barometer (Temporary for Integrated Test)

Accuracy ± 0.05 in Hg

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5. ANALYSIS OF RESULTS

5.1. Penetration Pressurization System--Analysis of Results

There were no instances of excessive leakage on any containment boundary constantly pressurized by the penetration pressurization system. Where high flow meter readings were noted, it was always due to leakage in system piping, usually requiring tightening of fittings.

5.2. Access Hatches - Door Seals - Analysis of Results

The number of tests and failures for each access hatch are shown in the following table:

TESTS/FAILURES

Personnel 170/2	Emergency 29/0

There were no instances of concurrent excessive leakage on both doors of either hatch thereby providing assurance of one of the two series leakage barriers. When excessive leakage was found, the doors were tested independently and the one with a good seal was locked closed pending repairs/retesting of the other door.

5.3. Access Hatches - Integrated - Analysis of Results

None of the hatch integrated tests yielded results which would cause the updated total of local leakage to exceed the 0.6 L_a criteria of Tech. Spec. 4.4.1.

Where the test results were significantly above the target criteria, repairs were promptly performed and the hatch was retested.

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6. DATA - MISCELLANOUS LEAK TESTS (4/20/77 - 4/12/78)

1. Penetration Pressurization Quarterly Flow Meter Readings

Test Date	(SCFH) As-Found	As-Left
6/8/77	less than 60*	Satisfactory*
9/10/77	0	0
12/14/77	18.5	20
2/26/78	19	19

* Data sheet lost

6.2. Access Hatch Door Seal-Meter Readings-Periodic

NOTE: Only tests which exceed the target criteria (3 SCFH) are listed here.

Hatch	Test Date	As-Found	As-Left
Personnel	6/27/77	4	less than 3
Personnel	4/4/78	4.1	less than 3

6.3. Access Hatch Integrated-Semiannual Pressure Drop

	Test Date	As-Found	As-Left
Personnel	6/21/77	38000	761
Emergency	5/17/77	3700	3700
Personnel	12/12/77	1849	1849
Emergency	12/7/77	8183	4163