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REACTOR CONTAIN[illegible]T BUILDING
INTEGRATED LEA[illegible] [illegible]ATE TEST

APRIL 1977

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THREE MILE ISLAND NUCLEAR STATION UNIT I

REACTOR CONTAINMENT BUILDING

INTEGRATED LEAK RATE TEST

APRIL 1977

1412 153

METROPOLITAN EDISON COMPANY

SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

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TABLE OF CONTENTS

1412 154

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 16, 1977 to April 19, 1977. 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 was found to be 0.042 percent by weight per day at 50.6 psig. The leakage rate at the upper bound of the 95 percent confidence interval is 0.052 percent by weight per day which is well below the allowable leakage rate of 0.075 percent by weight per day at 50.6 psig.

The final leakage rate of 0.042 percent by weight per day was obtained after adjustments were made and the test was restarted. The initial building leakage rate indicated was in excess of 0.1 percent by weight per day. The adjustments made consisted of tightening mechanical joints and packings.

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. The combined local leakage rate of both these isolation valves was 0.007 percent by weight per day. The addition of this value increases the total integrated leakage rate to 0.049 percent by weight per day.

1412 155

The supplemental instrumentation verification at P_a was 1.0 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.

1412 156

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 than 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 D).

Leakage rate testing was accomplished at the pressure level of 50.6 psig for a period of 24 hours. The 24 hour period was followed by an 8 hour supplemental test for a verification of test instrumentation. During the 32 hour period of testing, the reactor containment building internal temperature was maintained at $72.0 \pm 0.3^{\circ}F$.

1412 157

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.

1412 158

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: $\pm$ 0.19°F

Overall system repeatability: $\pm$ 0.19°F

Components:

a. Resistance Temperature Detectors

Quantity	24
Manufacturer	Rosemount
Type	Model 104 AAN, 100 ohm, platinum
Range, °F	60-110
Accuracy, °F	$\pm$ 0.1
Repeatability, °F	$\pm$ 0.1

b. Bridge Cards

Quantity	24
Manufacturer	Rosemount
Type	Model 440-L3
Range, °F	60-110
Accuracy, °F	$\pm$ 0.25% of span
Repeatability, °F	$\pm$ 0.25% of span

1412 159

c. Digital Indicator

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

* Modified for direct digital temperature readout

4.1.2 Dewpoint Indicating System

Overall system accuracy: $\pm$ 1.12^{o}F

Overall system repeatability: $\pm$ 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	$\pm$ 1.0
Repeatability, ^{o}F	$\pm$ 0.5

b. Dewpoint Recorder

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

1412 160

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 CALIBRATION 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 ANSI N45.4-1972, Section 6.2 and 6.3. 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.

1412 161

4.3 INSTRUMENTATION PERFORMANCE

Prior to the start of the integrated leak rate test, one dewcell began indicating a dewpoint temperature approximately 30°F lower than the other 9 dewcells. This dewcell was eliminated from future readings. The remaining 9 dewcells performed well at all times and provided more than adequate coverage of the containment. The temperature, pressure, and flow measuring systems performed well throughout the test.

4.4 SYSTEMATIC ERROR ANALYSIS

Systematic error, in this test, is induced by the operation of the temperature indicating system, dewpoint indicating system and the pressure indicating system.

Justification of instrumentation selection was accomplished, using manufacturer's accuracy and repeatability tolerances stated in Section 4.1, by computing the figure of merit as follows.

The leakage rate, in weight percent per day (%/day), based on an interval of measurement of 24 hour duration is

$$L = 100 \left[1 - \frac{P_{24}\ T_o}{P_o\ T_{24}}\right] \text{ \%/day}$$

where:

$P_o = P_{To} - P_{wvo}$, psia = partial pressure of air at start

$P_{24} = P_{T24} - P_{wv24}$, psia = partial pressure of air at finish

T_o = building mean ambient internal temperature at start, °R

T_{24} = building mean ambient internal temperature at finish, °R

1412 162

The change, or uncertainty in L due to uncertainties in the systematic measured variables is given by

$$\delta L = 100\left[\left(\frac{\partial L}{\partial P_{24}}\ \tau P_{24}\right)^2 + \left(\frac{\partial L}{\partial P_o}\ \tau P_o\right)^2 + \left(\frac{\partial L}{\partial T_o}\ \tau T_o\right)^2 + \left(\frac{\partial L}{\partial T_{24}}\ \tau T_{24}\right)^2\right]^{1/2}$$

where τ is the systematic error for each variable. The error in L after differentiation is

$$e_L = 100\left[\left(\frac{T_o\ e_{P24}}{P_o\ T_{24}}\right)^2 + \left(\frac{P_{24}T_o\ e_{Po}}{P_o^2\ T_{24}}\right)^2 + \left(\frac{P_{24}\ e_{To}}{P_o\ T_{24}}\right)^2 + \left(\frac{P_{24}T_o\ e_{T24}}{P_o\ T_{24}^2}\right)^2\right]^{1/2}$$

where:

$$e_{Po} = \tau P_o$$

$$e_{P24} = \tau P_{24}$$

$$e_{To} = \tau T_o$$

$$e_{T24} = \tau T_{24}$$

Since the values of T_o and T_{24} are essentially the same, within $0.28^{o}F$, and P_o and P_{24} are essentially the same, within 0.002 psia, let $T_o = T_{24}$, $P_o = P_{24}$, $e_{Po} = e_{P24} = e_p$ and $e_{To} = e_{T24} = e_T$. The systematic error in L then reduces to

$$e_L = 141.4\left[\left(\frac{e_p}{P_o}\right)^2 + \left(\frac{e_T}{T_o}\right)^2\right]^{1/2} \qquad (1)$$

where the error in pressure (e_p) may be expressed as

$$e_P = (e_{P_a}^{\ 2} + e_{P_b}^{\ 2})^{1/2}$$

1412 163

and

e_{P_a} = error induced by the precision pressure gauges, or

$$e_{P_a} = \pm \frac{(0.00015)(65.340)}{(2)^{1/2}} \text{ psia}$$

$$e_{P_a} = \pm 0.0069 \text{ psia}$$

and

e_{P_b} = error induced by the dewcells, or

$$e_{P_b} = \pm \frac{1.12}{(9)^{1/2}} \ {}^{\circ}F$$

$$e_{P_b} = \pm 0.373^{\circ}F$$

From steam tables, at a dewpoint of $65^{\circ}F$, the pressure equivalent to $\pm 0.373^{\circ}F$ is

$$e_{P_b} = \pm 0.0039 \text{ psia}$$

Therefore,

$$e_P = [(0.0069)^2 + (0.0039)^2]^{1/2} \text{ psia}$$

$$e_P = \pm 0.0079 \text{ psia}$$

The error in temperature (e_T) may be expressed as

$$e_T = \pm \frac{0.19}{(24)^{1/2}} \ {}^{\circ}F$$

$$e_T = \pm 0.0388^{\circ}F$$

1412 164

Hence, for values at 50.6 psig,

P_o = 65.340 psia

T_o = 531.42°R

and substitution into equation (1) yields

$$e_L = 141.4 \left[\left(\frac{0.0029}{65.340}\right)^2 + \left(\frac{0.0388}{531.42}\right)^2\right]^{1/2}$$

$$e_L = \pm\ 0.020\%/\text{day}$$

The maximum expected systematic error (figure of merit) of the test instrumentation is e_L.

If equation (1) is solved using previously stated repeatability values, the figure of merit is calcuated to be

$$e_L = \pm\ 0.011\%/\text{day}$$

Containment leakage rate computations are a function of changes in temperature and pressure relative to each other, not absolute values. Therefore, the repeatability error analysis is more meaningful.

A conclusion reached from the above calculation was that the instrumentation selected yielded an error value five times less than the allowable leakage rate value of 0.10 percent per day and that the instrumentation combination was of sufficient sensitivity for this test. The e_L values are not based on a statistical analysis of leakage rate calculations and are used strictly for instrumentation selection.

1412 165

4.5 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 24 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

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

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,

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

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.

1412 166

The reactor containment building leakage rate during the supplemental test ($L_{v'}$) was then compared to the measured reactor containment building leakage rate during the preceding 24 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 lerkage rate (L_a).

1412 167

5.0 TEST PROCEDURE

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 leakage into the system. These gauges were removed and the manifolds were vented prior to the start of the test.

f. Pressure gauges 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.

1412 168

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

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 approximately 72°F. Building pressure and temperature were monitored half hourly and the amperage required by the recirculation unit fans (AH-E-1A, 1B and 1C) was monitored hourly. Leak rate testing was initiated at the 50.6 psig pressure level. Forty-three hours elapsed between reaching the 50.6 psig pressure level and the recording of official data. For the duration of the 24 hour leak test and the 8 hour supplemental test, the average internal containment temperature was maintained within a band of ± 0.3°F by varying the industrial cooler cooling water flow rate to the containment recirculation fan unit coolers.

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

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

1412 169

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

c. The nine dewpoint values were recorded. The average of the nine 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). Atmospheric weather conditions were clear from 1000 on April 16, 1977 to 1830 on April 18, 1977. From 1900 on April 18, 1977 to 1530 on April 19, 1977, the weather conditions were cloudy.

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 24 hour period was available.

Subsequent to the 24 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.

1412 170

5.3 TEST PERFORMANCE

5.3.1 Pressurization Phase

Pressurization of the reactor building containment was started on April 15, 1977 at 0500. The pressurization rate was approximately 2.5 psi per hour. When containment internal pressure reached 12 psig, at 1120 on April 15, 1977, pressurization was secured. An inspection team entered containment to perform the 12 psig inspection. During pressurization to the 12 psig pressure level, the Leak Rate Test System air dryer drain and the cyclone separator drain were not functioning properly. Pressurization was secured while temporary bypasses were installed. While at the 12 psig pressure level, these drains were repaired. The 12 psig internal inspection was completely satisfactorily and pressurization was restarted at 1336 on April 15, 1977.

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

a. Several penetration pressurization manifold isolation valves were suspected of leaking. The main header was then vented to ensure the penetration pressurization system would remain depressurized.

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

1412 171

c. A small amount of water leakage was noticed from Nuclear Services Closed Cycle Cooling Water valves NS-V4 and NS-V15.

d. One leak rate test dewcell began to indicate a dewpoint temperature approximately 30°F lower than the remaining nine dewcells. This dewcell was eliminated from data collection.

When containment internal pressure reached 50.7 to 50.8 psig, at 0600 on April 16, 1977, pressurization was secured. Temperature was controlled by throttling the industrial cooler pump discharge valve, RB-V18D, which supplies cooling water to the recirculation fan units cooling coils. All penetration pressurization system temporary manifold pressure gauges were removed.

5.3.2 Integrated Leak Rate Testing Phase

After waiting 4 hours, leak rate testing was started. Temperature had stabilized at approximately 72°F. From 1000 on April 16, 1977 until 0500 on April 18, 1977, an excessive leakage rate was indicated by the data collected. The weight of containment air and the average containment temperature versus time for this time period are presented in Appendix B, Exhibits 1 and 2. During this time, the following sequence of events took place:

a. At 1200 on April 16, 1977, the leakage rate, based on two hours of data, was 0.151 percent by weight per day. This established a baseline for the mass point versus time graph. Plant auxiliary operators were sent on routine leak detection. There was no cause for immediate concern since only a limited amount of data had been collected.

1412 172

b. Subsequent mass points were following approximately the same trend as previously reported. Pressure gauges were installed on manifolds "J", "N" and "O" of the penetration pressurization system for leakage detection. Plant auxiliary operators were again dispatched for leak detection.

c. At 1930 on April 16, 1977, the leakage rate, based on nine and one-half hours of data, was 0.199 percent by weight per day. The pressure gauge on manifold "O" (Fuel Transfer Tube Flanges) was replaced with a flow indicator.

d. The fluid block line to valve IC-V4 was isolated and vent valve FB-V122 was opened. Leakage through this path was evident.

e. The purge valves and the access lock doors were soap-checked and no leakage was indicated. A bonnet/packing leak on penetration pressurization system valve PP-V46 and reducer leaks on the reactor building pressure sensing lines near BS-V37C and BS-V37D were found and repaired.

f. An investigation revealed that several of the automatic fluid block initiation valves, specified to be open, were closed. All automatic fluid block initiation valves were opened.

g. Additional flow indicators were placed on the main steam lines from steam generator A (OTSG A) and steam generator B (OTSG B). At 2400 on April 16, 1977, the following leakages were indicated:

1412 173

Location	Leakage
Manifold "0"	160 sccm
OTSG A	850 sccm
OTSG B	0 sccm
WDG-V4	700 sccm

h. Since the amount of leakage found was insignificant compared to the leakage indicated by the data (250,000 sccm), leak detection continued. At 0245 on April 17, 1977, the reactor containment building was repressurized to between 50.7 and 50.8 psig.

i. Indicated leakage from OTSG A had increased to 1000 sccm. The fluid block line to IC-V4 was opened and no pressure buildup in manifolds "N" and "O" was observed.

j. As leak detection continued, the measured containment leakage rates were as follows:

Date	Time Interval	Leakage Rate	95% Confidence
4-17	0300-0700	0.120%/day	0.051%/day
4-17	0300-1400	0.126%/day	0.009%/day

k. A valve lineup verification was performed and no deviations were found. A systematic quadrant by quadrant check of penetrations and isolation valves failed to identify any significant leakage. The following adjustments were made on April 17, 1977:

1) Fittings and connections in the leak rate test panel were tightened.

2) Flanges downstream of LR-V2 and LR-V3 were tightened.

1412 174

Minor amounts of leakage were evident at the following locations:

1) LR-V2 and LR-V3 packing

2) Purge supply interspace.

3) Personnel airlock

4) Purge exhaust interspace

5) OTSG A

l. At 2230 on April 17, pressurization of the secondary side of OTSG A was begun to determine if a change in the indicated reactor containment building leakage could be detected. With the OTSG A at 16 psig, it was decided to depressurize OTSG A since the data prior to 2230 had indicated an upward trend in the mass points. At 0210 on April 18, 1977, OTSG A was depressurized and it was decided to collect and evaluate a full 24 hours of data.

m. At 1030 on April 18, 1977, it was noted that approximately 5 psig pressure had built-up between the seals of the emergency airlock and the personnel airlock. Vents were opened, the pressure was bled off and the vents were left open to allow a leakage path to exist.

n. The containment leakage rate measured from 0300 to 1130 on April 18, 1977 was 0.097 percent by weight per day with an upper bound 95 percent confidence of 0.017 percent by weight per day.

1412 175

o. At 1320 on April 18, 1977, the concrete shield for the equipment hatch was put in place.

p. The measured containment leakage rate from 0300 to 1600 on April 18, 1977 was 0.103 percent by weight per day with an upper bound 95 percent confidence of 0.009 percent by weight per day.

q. Subsequent to 1600 on April 18, 1977 a shift in the trend of the containment mass points occurred.

r. An acceptable leakage rate of 0.042 percent by weight per day was obtained from 0500 on April 18, 1977 to 0500 on April 19, 1977.

Due to the lack of any local leakage rate determinations prior to the adjustments mentioned in Section 5.3.2.k., the initial unsatisfactory leakage rate indications must be assumed to constitute a failed test.

Nevertheless, since an extensive search failed to identify any significant sources of leakage, it is unlikely that the initial measured leakage rate values, which were in excess of 0.10 percent by weight per day, were true measurements of leakage from the reactor containment building to the outside atmosphere. Two possible explanations for the initial results are:

a. There was leakage into volumes internal to the containment building. The internal volumes may have been (1) the reactor coolant system, since a slow steady decrease in the pressurizer level was noted throughout the test with

no corresponding increase in reactor building sump level, and/or (2) the volume between isolation barriers. Additionally, there may have been air entrainment into the concrete and insulation material inside the containment. However, the length of time that the excessive leakage rate was present and the abrupt rather than gradual change in the leakage rate do not tend to support this explanation entirely.

b. The apparent leakage was the result of a diurnal effect. The heating of the containment during the day and the cooling of the containment during the night would cause a change in the containment internal pressure due to the expansion/contraction of the containment without a corresponding detectable change in the containment internal temperature. However, the data, as presented in Appendix B, Exhibits 1 and 2, does not appear to totally support an explanation based on diurnal effects.

5.3.3 Supplemental Leakage Rate Test Phase

After the 24 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.

1412 177

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.

1412 178

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:

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

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 nine dewpoint temperatures and converting to vapor pressure with the use of steam tables, psia

1412 179

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

$$T = \frac{\text{sum of 24 RTD's}}{24} + 459.69^{\circ}R$$

The weight of air is plotted versus time for the 24 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:

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

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

where,

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

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.

1412 180

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:

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

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 fit 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 $(W_o + W_1 t) \pm 2S_e$.

1412 181

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

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

where,

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

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

1412 182

7.0 DISCUSSION OF RESULTS

7.1 RESULTS AT P_a

Data obtained during the integrated leak rate test at P_a indicated the following maximum changes (highest reading to lowest reading) during the 24 hour test period:

Variable	Maximum Change
P_T	0.026 psia
P_{wv}	0.011 psia
T	0.47°F

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 of 0.042 %/day.

The 95 percent confidence limit associated with this leakage rate is 0.010 percent per day. Thus, the leakage rate at the upper bound of the 95 percent confidence interval becomes

$$L_{am} = 0.042 + 0.010$$
$$L_{am} = 0.052 \text{ \%/day}$$

The measured leakage rate and the measured leakage rate at the upper bound of the 95 percent confidence level are well 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 one of the forty-nine data points does 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 24 hour test at 50.6 psig, flowmeter FI-111 was placed in service and a flow rate of 207 SCFH was established. This flow rate is equivalent to a leakage rate of 0.056 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) during the supplemental test was calculated to be 0.099 percent per day using the mass point method of analysis. The 95 percent confidence interval associated with this leakage rate is 0.020 percent per day. None of the 25 data points is out of confidence.

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

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

$$L_{v'} = 0.099\%/\text{day} - 0.056\ \%/\text{day}$$

$$L_{v'} = 0.043\%/\text{day}$$

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

$$\frac{|L_{am} - L_{v'}|}{L_a} = \frac{|(0.042) - (0.043)|}{0.10} = 0.01$$

The building leakage rates agree within 1.0 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.

1412 184

8.0 TYPE B AND C LEAKAGE RATE HISTORIES

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

1412 185

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 Instrumemt Calibrations", Metropolitan Edison Company Procedure.

1412 186

APPENDICES

1412 187

APPENDIX A

REDUCED LEAKAGE DATA

1412 188

APPENDIX A

REDUCED TEST DATA

	Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (°R)	Weight of Containment Air (lbm)
4/18/77	0500	65.340	0.294	65.046	531.42	660,753.87
	0530	65.338	0.293	65.045	531.42	660,743.71
	0600	65.335	0.294	65.041	531.41	660,715.51
	0630	65.336	0.295	65.041	531.43	660,690.65
	0700	65.336	0.293	65.043	531.44	660,698.53
	0730	65.337	0.295	65.042	531.44	660,688.37
	0800	65.338	0.297	65.041	531.49	660,616.06
	0830	65.340	0.294	65.046	531.52	660,629.56
	0900	65.342	0.294	65.048	531.54	660,625.01
	0930	65.344	0.295	65.049	531.57	660,597.88
	1000	65.346	0.294	65.052	531.60	660,591.07
	1030	65.348	0.295	65.053	531.64	660,551.52
	1100	65.350	0.293	65.057	531.68	660,542.44
	1130	65.354	0.294	65.060	531.70	660,548.05
	1200	65.356	0.292	65.064	531.75	660,526.55
	1230	65.354	0.291	65.063	531.76	660,503.97
	1300	65.356	0.294	65.062	531.78	660,468.98
	1330	65.361	0.298	65.063	531.81	660,441.87
	1400	65.358	0.292	65.066	531.81	660,472.33
	1430	65.358	0.292	65.066	531.81	660,472.33
	1500	65.358	0.293	65.065	531.83	660,437.34
	1530	65.357	0.294	65.063	531.84	660,404.62
	1600	65.356	0.291	65.065	531.84	660,424.92
	1630	65.355	0.292	65.063	531.82	660,429.46
	1700	65.353	0.294	65.059	531.81	660,401.27
	1730	65.354	0.293	65.061	531.83	660,396.74
	1800	65.356	0.291	65.065	531.84	660,424.92
	1830	65.358	0.295	65.063	531.87	660,367.37
	1900	65.360	0.292	65.068	531.87	660,418.12

1412 189

Gilbert/Commonwealth

APPENDIX A (Cont'd)

REDUCED TEST DATA

	Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
	1930	65.356	0.292	65.064	531.85	660,402.35
	2000	65.359	0.294	65.065	531.86	660,400.09
	2030	65.358	0.294	65.064	531.86	660,389.94
	2100	65.360	0.294	65.066	531.88	660,385.40
	2130	65.361	0.293	65.068	531.88	660,405.70
	2200	65.360	0.290	65.070	531.87	660,438.42
	2230	65.357	0.294	65.063	531.84	660,404.62
	2300	65.353	0.294	65.059	531.82	660,388.85
	2330	65.352	0.294	65.058	531.79	660,415.96
	2410	65.349	0.294	65.055	531.77	660,410.34
4/19/77	0030	65.350	0.293	65.057	531.74	660,467.90
	0100	65.346	0.292	65.054	531.73	660,449.87
	0130	65.348	0.292	65.056	531.73	660,470.17
	0200	65.346	0.291	65.055	531.72	660,472.44
	0230	65.346	0.291	65.055	531.74	660,447.60
	0300	65.346	0.292	65.054	531.73	660,449.87
	0330	65.348	0.294	65.054	531.75	660,425.03
	0400	65.348	0.292	65.056	531.77	660,420.49
	0430	65.344	0.292	65.052	531.71	660,454.40
	0500	65.342	0.287	65.055	531.70	660,497.29

SUPERIMPOSED TEST

	Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
	0700	65.330	0.294	65.036	531.68	660,329.22
	0800	65.333	0.289	65.044	531.73	660,348.34
	0830	65.334	0.291	65.043	531.76	660,300.94
	0900	65.333	0.292	65.041	531.76	660,280.63
	0930	65.330	0.288	65.042	531.73	660,328.04
	1000	65.328	0.291	65.037	531.73	660,277.28
	1030	65.327	0.291	65.036	531.72	660,279.54

Gilbert / Commonwealth

1412 190

APPENDIX A (Cont'd)

REDUCED TEST DATA

Time	Average Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^{o}R)	Weight of Containment Air (lbm)
1100	65.327	0.291	65.036	531.73	660,267.13
1130	65.330	0.290	65.040	531.74	660,295.32
1200	65.331	0.292	65.039	531.77	660,247.91
1230	65.327	0.289	65.038	531.77	660,237.76
1300	65.322	0.294	65.028	531.71	660,210.74
1330	65.320	0.293	65.027	531.70	660,213.00
1400	65.324	0.291	65.033	531.77	660,187.01
1430	65.328	0.293	65.035	531.84	660,120.41
1500	65.332	0.290	65.042	531.88	660,141.82
1530	65.332	0.291	65.041	531.90	660,106.84

Gilbert/Commonwealth

1412 191

APPENDIX B

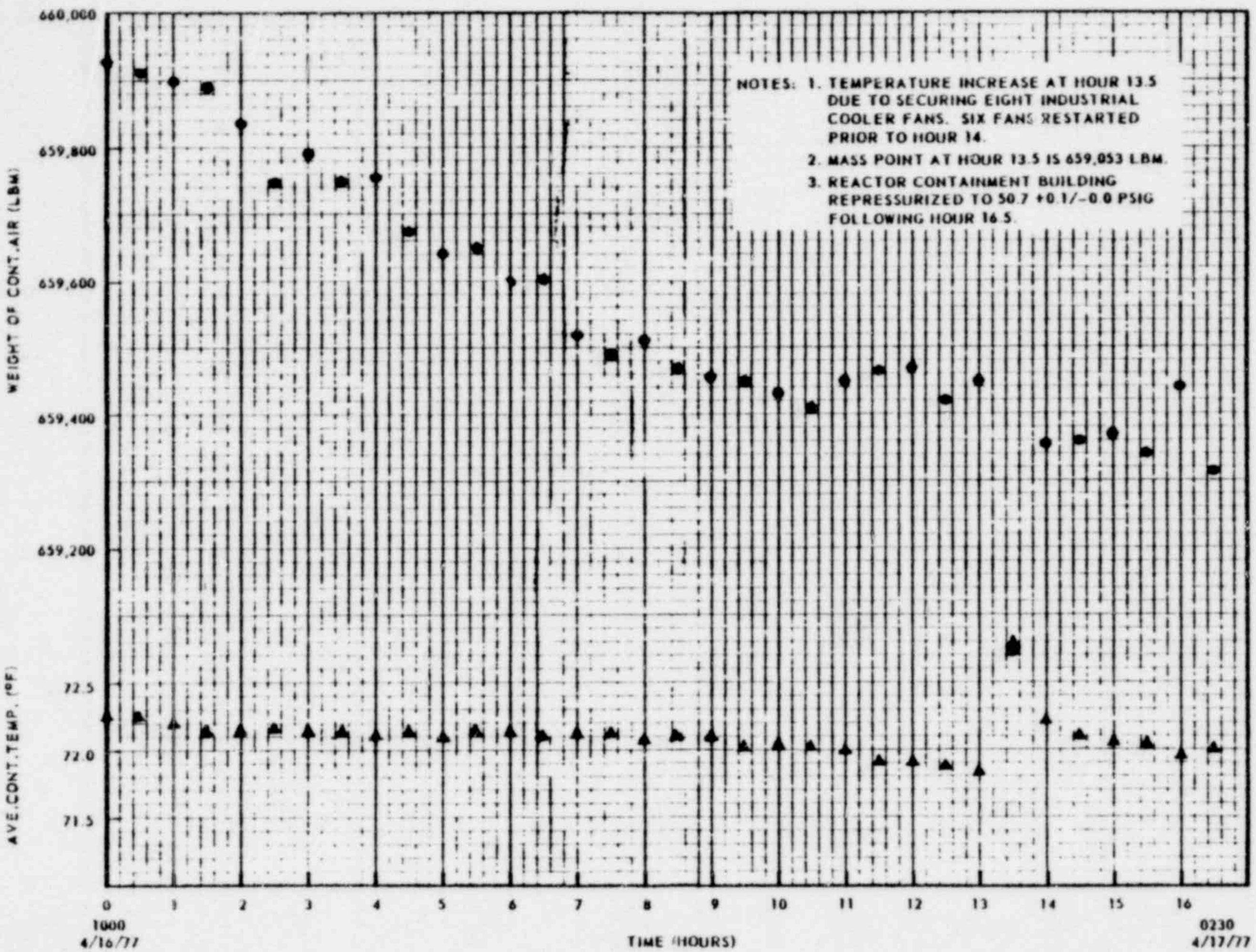
WEIGHT OF CONTAINMENT AIR AND

AVERAGE CONTAINMENT TEMPERATURE

1412 192

TABLE OF CONTENTS

APPENDIX B - EXHIBIT 1
WEIGHT OF CONTAINMENT AIR AND
AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME

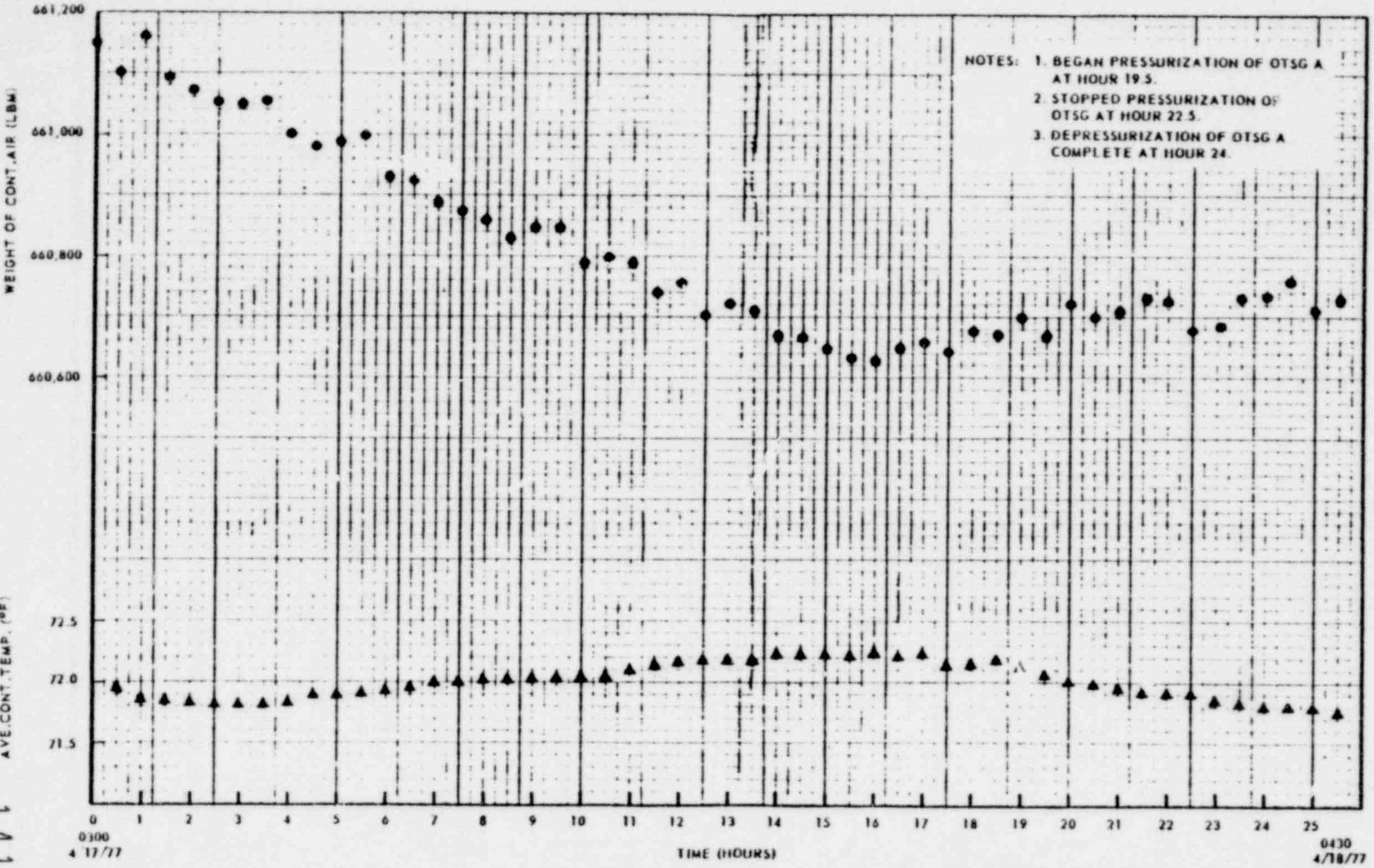


POOR ORIGINAL

THREE MILE ISLAND NUCLEAR STATION
UNIT 1
INTEGRATED LEAK RATE TEST
APPENDIX B - EXHIBIT 1

1412 194

APPENDIX B – EXHIBIT 2
WEIGHT OF CONTAINMENT AIR AND
AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME

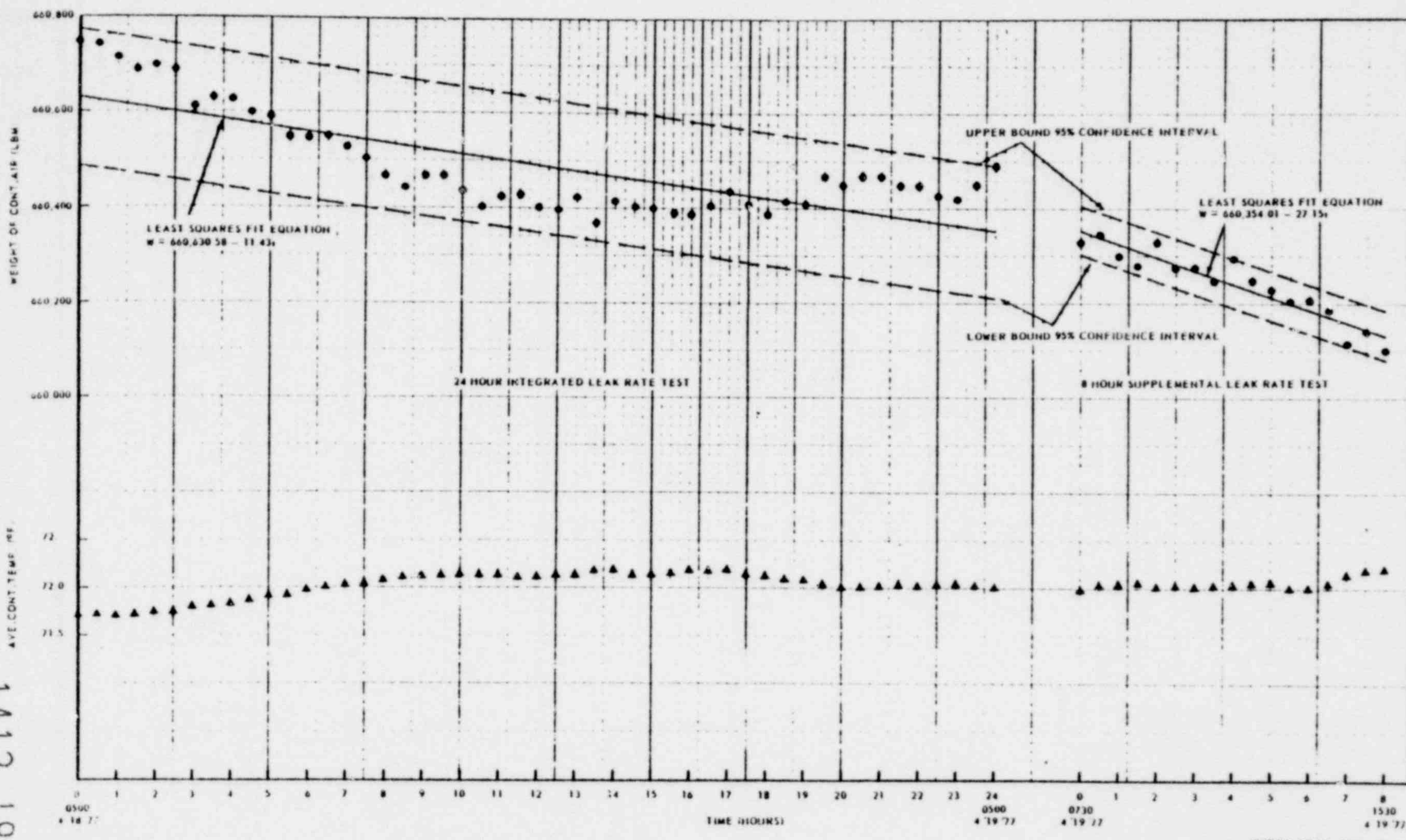


POOR ORIGINAL

THREE MILE ISLAND NUCLEAR STATION
UNIT 1
INTEGRATED LEAK RATE TEST
APPENDIX B - EXHIBIT 2

1412 195

APPENDIX B - EXHIBIT 3
WEIGHT OF CONTAINMENT AIR AND
AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME



POOR ORIGINAL

THREE MILE ISLAND NUCLEAR STATION
UNIT 1
INTEGRATED LEAK RATE TEST
APPENDIX B - EXHIBIT 3

1412 196

APPENDIX C

THREE MILE ISLAND UNIT 1

1976 REFUELING

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

SP 1303-11.18

1412 197

INDEX

APPENDICES

I. PURPOSE

A. To provide a summary analysis to the Nuclear Regulatory Commission on the first periodic Type B and Type C leakage tests performed since the last Type A test. 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 requires the contents of this summary report to become part of the next Type A test report to be submitted to the NRC.

B. To summarize the violation of the TMI-1 Technical Specifications, paragraph 4.4.1.2 (Local Leakage Rate Tests) in that the Reactor Building local leak rate surveillance testing, performed during the first regueling period, resulted in a combined leakage above the acceptance criteria of $0.6L_a$ (Maximum allowable leakage rate at Pa). This event was considered to be a Reportable Occurrence as defined in the Technical Specifications, paragraph 6.9.2.B(2).

SECTION B

SUMMARY

SECTION B - SUMMARY OF WORK ACCOMPLISHED

1. TESTING

Reactor building refueling frequency 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 Metropolitan Edison Company's September 17, 1975 letter to the NRC (GQL 1515).

For valves that are fluid blockable, the leak tests were performed both by nitrogen pressure in the process line and by water pressure in the valve bonnet. The nitrogen tests results were used for acceptance.

A total of approximately one hundred thirteen (113) leak tests were performed, man as retests after repairs. Fourteen (14) of the fifty four (54) containment isolation valves had higher indicated as-found leakage rates than the cognizant engineer could accept and repairs were performed.

AH-V1A/1B/1C/1D	CM-V1	MU-V116
CA-V2, 5A, 5B	IC-V2,3,4,6	RB-V7

Five (5) of these valves were accepted after one(1) repair/retest. four (4) after two (2) repairs/retests and the remaining five (5) after numerous additional repairs and retests.

Five (5) of the valve leakages remained higher than desirable though further repairs were not considered worthwhile or parts were not available.

CA-V2,5A	CM-V2	MU-V18, 116

2. REPAIRS

Refer to Section 9 of NRC Reportable Occurrence Report No. ER 76-19/3L (Appendix I attached) for a summary of the valve repairs which were performed on containment isolation valves.

POOR ORIGINAL

SECTION C METHODS OF TESTING

Section C

POOR ORIGINAL

METHOD USED FOR TESTING

Type B and C tests, or options, as defined in 10 CFR 50 Appendix J, were performed. The procedures utilized acceptable test methods per 10 CFR 50 Appendix J.

1. Nitrogen/Air (Type B/C)Testing

Unit 1 Surveillance Procedure 1303-11.18 (Reactor Building Local Leakage-Air/Nitrogen Type Tests) was used for nitrogen.air testing of Type B&C penetrations.

Testing of the valves/penetrations was by one fo three below listed methods. Valves/penetrations were designated as to which of the below methods by which it would be tested.

1.1 Pressurization of a test volume with nitrogen or air to _Pa with subsequent measurement of the rate of pressure loss of the test chamber.

1.2 Local pressurization to _ Pa with leakage measurements taken by an upstream variable area flowmeter.

1.3 Local pressurization to _ Pa with leakage measurements taken by a downstream variable area flowmeter.

2. Water-Fluid Block Type Testing

Unit 1 Surveillance Procedure 1303-11.23 (Reactor Building local leakage-fluid block system) was used for leak testing ~~contain~~ the fifteen (15) containment isolation valves which are pressurized by the fluid block system.

Testing of the fluid blocked valves was performed by isolating the containment isolation valve from the fluid block system (by closing the fluid block line maintenance isolation valve), opening upstream

and downstream (from CIV) vents/drains, and pressurizing the bonnet of the closed test valve to _ 1.10 Pa through a fluid block test connection valve. Leakage is calculated by observing the level decrease of a graduated cylinder (reservoir) on the pressurization pump suction.

SECTION D

TEST EQUIPMENT

1. For Nitrogen Testing

Two nitrogen pressurization rigs and two vent rigs were fabricated on portable boards with piping arrangements as shown in Figure 1. Two sets of three Brooks rotameters (ranges shown on Figure 1) were provided for field use on the test rigs. This equipment was used as directed in the Surveillance Procedure SP 1303-11.18.

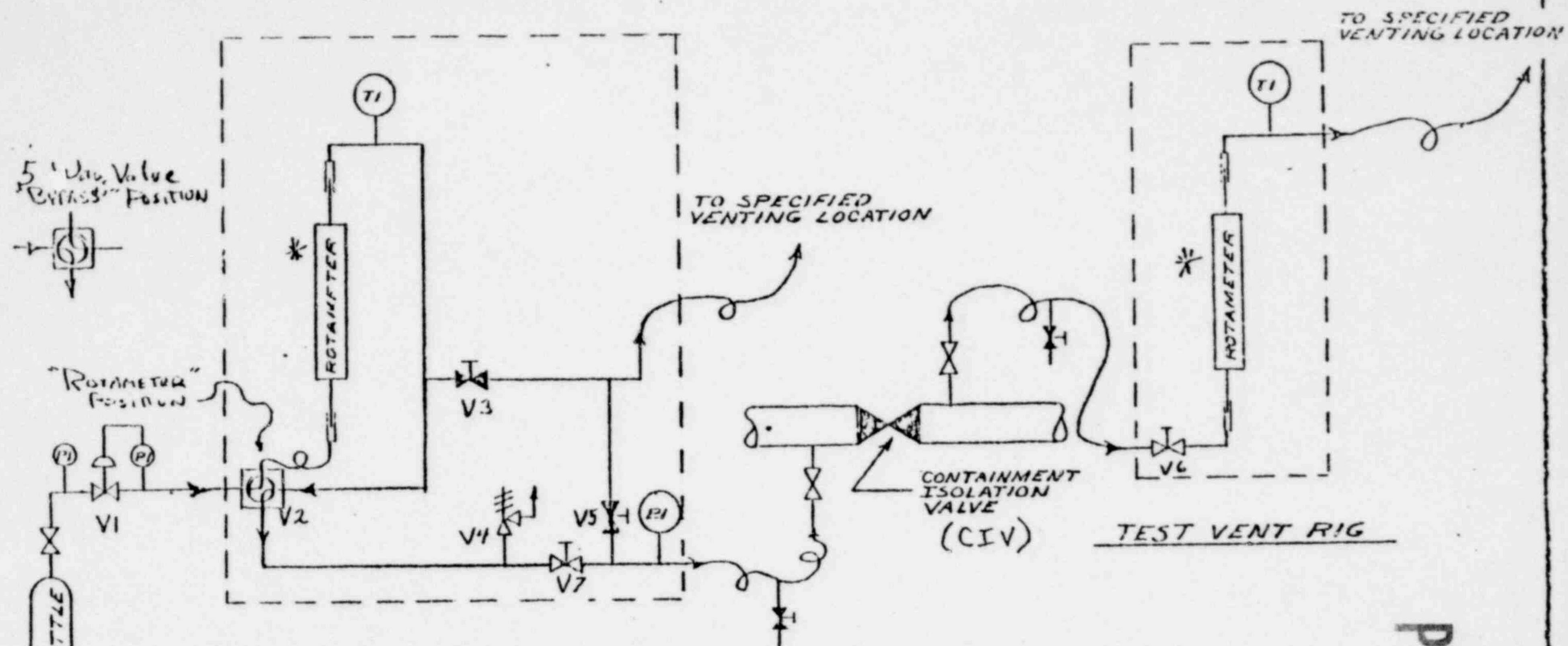
One rotameter standardization rig was fabricated. (See sketch in Appendix IV). A set of three Brooks rotameters (each calibrated to ± 1% FSA) is mounted on this board along with necessary range switching valves and quick-connects for temporary mounting of the field test meters. This standardization rig was used in the hot instrument shop to do weekly re-standardizations of the field test meters. A comparison of flow readings was made for each meter tested at three selected scale readings, with the intention of verifying better than ± 5% Full Scale Accuracy. See Appendix IV for the Standardization Procedure.

2. For Fluid Block Testing

A Sprague deionized water hydro test pump with piping arrangements as shown in Figure 3 was used. The procedure used is Surveillance Procedure 1303.11.23

1412 205

FIGURE ONE



TEST PRESSURIZATION RIG

Equipment

Reactor plant grade N_2 supply

Pressure gage 0-100 psi ± 1 psi accuracy

Temperature Indicators 25-125 °F ± 1 °F accuracy

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

Brooks	@ 0 psi	@ 35 psi
R-2-15-AA / Tantalum	10-480 SCCM	140-1420 SCCM
R-2-15-C / Saphire	100-5300 SCCM	200-12,700 SCCM
R-6-15-B / Carboloy	1000-61000 SCCM	2000-143000 SCCM
Fisher & Porter		
035-1/16-SS	10-80 SCCM	
118-1/8-SS	150-1500 SCCM	
374-1/4-SS	1000-32600 SCCM	

ISOLATION VALVE TEST RIG

N_2 Test Rig

Figure #1

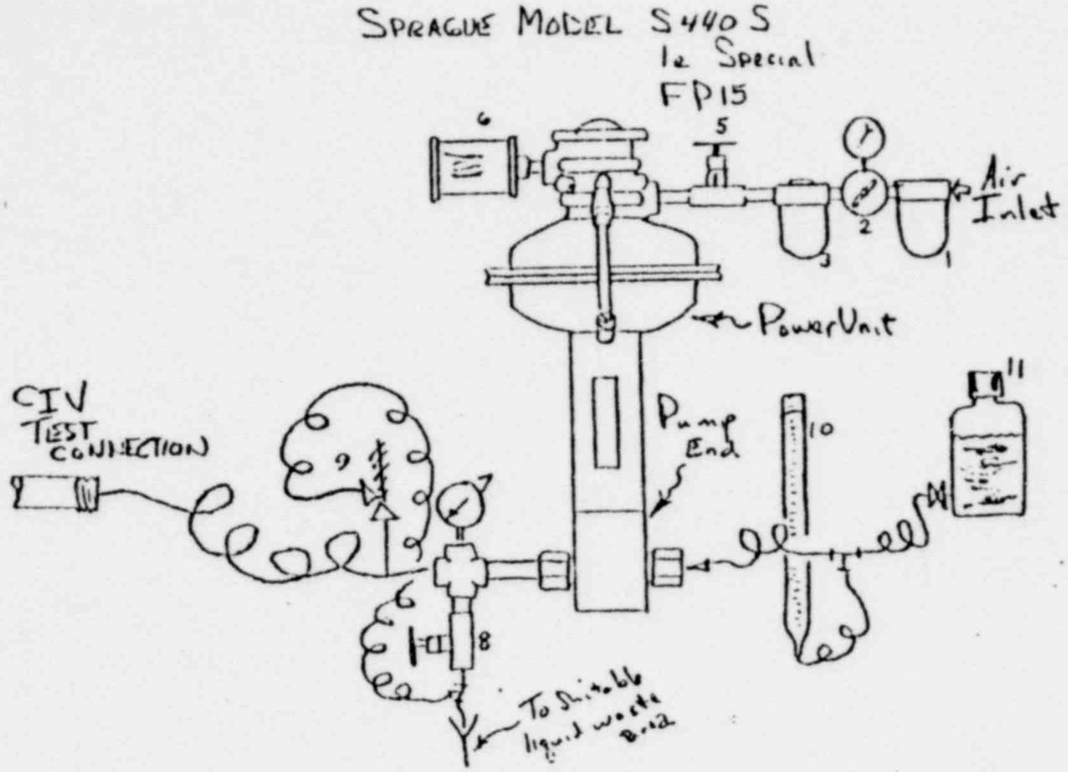
1412 206

POOR ORIGINAL

3/24/76

V2

FIGURE TI.



1. Air Filter
2. Air Regulator
3. Lubricator
4. Operating Air Gage
5. Operating Air Shutoff
6. Muffler
7. Output Pressure Gage
8. Fluid Pressure Bypass
9. 65 psig Relief
10. Graduated Cylinder (100 ml Burette)
11. Dispensing Cylinder

1412 207

POOR ORIGINAL

VI 3

FLUID BLOCK TEST RIG
FIGURE # 2

Section E ANALYSIS OF RESULTS AS-FOUND/AS-LEFT

Section E

ANALYSIS OF RESULTS AS-FOUND/AS LEFT (See Appendix III 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, a predesignated test witness and a Cognizant Engineer.

The safety analysis for the excessive leakage as-found is included in the NRC reportable Occurrence Report (See Appendix I).

1. Interpretation of Data

   1.1 As-Found Leakage Results

   The "as-found" combined Reactor Building local leakage for both nitrogen/air and fluid block testing is shown in the below table. Comparison to Technical Specification and FSAR limits are also shown in the table on Page 2. Comparison to Technical Specification and FSAR limits are also shown.

1.2 As-left Leakage Results

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

As-Left) REACTOR BUILDING LOCAL LEAKAGE

Type Test	Total Leakage	Tech. Spec. Limit	Percent Tech. Spec. Limit	Remarks
N2/Air	59,171	111,899 sccm	52.9%	See notes 1,2,&3

NOTES 1. The cumulative total does not have the error analysis factor included.

2. The total shown is cumulative by penetration and not the total of all valves tested. (See discussion note 3 of Section V1.A).

3. The total includes fluid blocked valves.

The existing leakages on containment isolation valves/ penetrations are listed in Appendix II.

If the error analysis is included in the data, the as-left leakage becomes 75,140 sccm or 67.1% of the Tech. Spec. limit. (See Section E2 "Error Analysis" for discussion).

POOR ORIGINAL

AS-FOUND TOTAL REACTOR BUILDING LOCAL LEAKAGE				
Type Test	Total Leakage	Tech. Spec./ FSAR Limit	Percent Tec. Spec./FSAR Limit	Remarks
N_2/Air	202,478 sccm	111,899 sccm	180.9%	See Notes 1&3
Fluid Block Tank FB-T1A	>9.79 cc/min.	13.40 cc/min	>73.2%	See Note 2
Fluid Block Tank FB-T1B	>33.62 cc/min.	13.13 cc/min.	>256.5%	See Note 2

NOTES: 1. The Cumulative is taken from raw data, i.e., error analysis not included.

The total does not include nitrogen/air leakage through fluid blocked CIVS. Official initial leak testing on the 15 fluid block valves was by fluid block testing methods.

2. Cummulative taken from raw data.

3. The totals shown are cummulative 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.

The exact "as-found" leakage was not ascertainable in that five (5) of the nitrogen/air tested valves exceeded the flow capabilities of the pressurization equipment and thus could not be pressurized to the required test pressure. A similar situation existed for three (3) valves associated with fluid block tank FB-T1A and two (2) valves on FB-T1B. Therefore, the exact "as-found" leakages are unknown but are above the valves given. Individual valve/penetration "as-found" leakages are listed in Appendix I.

2. Error Analysis

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

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

In all cases temperature effects on the test results were considered to be insignificant.

To determine the leakage, corrected for meter accuracy on each individual leak test, 5% was added to the recorded data sheet scale readings. If this corrected value still did not exceed the minimum usable (10%) value for meter reading the flow rate corresponding to 10% full scale (15mm) was used as the corrected leakage value.

Example:

For DH-V63
Data sheet recorded scale reading = 3mm
Add 5% F.S. 3+(0.05x150) = 10.5mm
Which is less than 10% F.S. therefore the corrected leakage value was taken as the flow rate corresponding to 15 mm.

For pressure drop tests the value for pressure gage resolution was used to make error corrections. i.e., the gage resolution was added to the intial pressure and subtracted from the final reading.

SECTION F - REFERENCES

1. SP 1303-11.18 Reactor Building Local Rate Testing

2. Met-Ed to NRC Licensing Letter 9/17/75--Comparison of TMI-1 Tech. Spec. with Appendix J--10CFR50.

3. SP 1303-11.23 Reactor Building Local Leakage - Fluid Block System

4. Appendix J - 10 CFR 50 Primary Reactor Containment Leakage Rate Testing for Water Cooled Power Reactors

5. Three Mile Island Unit 1 Technical Specifications 4.4.1.

6. TMI Surveillance File (for Data Sheets)

APPENDICES

APPENDIX I

REPORTABLE OCCURRENCE

NO. 76-19/3L - COPY

APPENDIX II

DESCRIPTION OF EQUIPMENT TESTED

POOR ORIGINAL

APPENDIX II

DESCRIPTION OF EQUIPMENT TESTED

SUBJECT ______ DATE ______ SHEET ______

LOCATION ______

ENGINEER ______

VALVE TYPE	Number Tested	SYSTEM	TAG #	Size	MFGR. Valve	FIG. #	MFGR Operator	OPERATOR TYPE
4 BALL	4	CM	1,2,3,4	1"	Crane Flo-Matic	N138	Bettis	PO
4 BUTTERFLY	2	AH	1A, 1D	48"	PRATT	RIA	Limitorque	MO
	2	AH	1C, 1D	48"	PRATT	RIA	Limitorque	MO
4 CHECK	2	CF	12A, 12B	1"	HANCOCK	5500W	NA	Lift Check
	1	DH	69	1½"	HANCOCK	7150W	NA	Stop Check
	1	MU	116	1½"	VELAN	W7-334P-13WS	NA	Piston Check
29 GATE	3	CA	2, 5A, 5B	1"	VELAN	W5-354B-13MS	Kieley Mueller	PO
	1	CA	189	2"	HANCOCK	950W		PO
	4	CF	19A, 19B, 20A, 20B	1"	HANCOCK	950W		PO
	2	HP	1, 6	6"	WALWORTH	5202WE	NA	HW
	1	IC	2	6"	WALWORTH	5202WE	MILLER	PO
	2	IC	3, 4	6"	WALWORTH	5202WE	MILLER	PO
	1	IC	6	3"	Aloyco	N216ACC-SP	MILLER	PO
	3	NS	4, 15, 35	8"	WALWORTH	5202WE	Limitorque	MO
	1	MU	3	2½"	Aloyco	NG226ACC-SP	Flick/Reddy	PO
	1	MU	18	2½"	POWELL	11303WE	PATHON	PO

1412 217