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Yellow
V2B

SEP 9 1974

Memo to File
Thru D. L. Caphton, Senior Reactor Inspector GHS for DLC

DN 50-219
OYSTER CREEK VACUUM BREAKERS

I attended a D. L. meeting held 9/6/74 with JCP&L and G. E. representatives to discuss proposed T. S. for Oyster Creek Vacuum breakers.

In essence, D. L. will issue new Technical Specifications, which among other things will permit operation with up to four of the fourteen suppression chamber - dry well breakers inoperable, provided that they are secured closed.

JCP&L will apparently not pursue proposed change No. 23 which was less restrictive (i.e. only six of the seven operable, while seven were routinely locked closed. Neither JCP&L or G. E. were able to refute the envelope of parameters tested at Bodega Bay. G. E. implied that a new internal model was available; however, no information has been furnished for staff review.

Informal discussion with D.L. indicated that the T. S. should be issued in about three weeks.

Ed Greenman

E. G. Greenman
Reactor Inspector

cc: Caphton
McCabe
Brunner
O'Reilly

B/69

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PDR FOIA
DEKOK95-258 PDR

Jersey Central Power and Light Company

Oyster Creek
Nuclear Generating Station
DPR - 16

Reactor Containment Building Integrated Leak Rate Test
June 1974

Performed by: R. M. Bright
C. S. Orogvany
E. V. Roscioli
E. S. Rosenfeld

Error Analysis by: E. Papscoe

8-28-74
Date of Approval

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K. O. E. Fickeissen, Jr.
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8-28-74
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9/10/74
Date of Approval

J. T. Carroll, Jr.
Station Superintendent

B/470

II. Acceptance Criteria

The maximum allowable leakage for the primary containment is 0.6267 weight percent of the contained air per 24 hours. This limit is 75 percent of $L_t(20)$ which is based upon either the measured leakage rate at 20 psig and 35 psig during the March, 1969 test or the test pressures of 20 psig and 35 psig. In equation form:

$$L_t(20) = 1.0\ L_m(20)/L_m(35) = 0.941 \text{ w/o per day} \qquad (1)$$

or

$$L_t(20) = 1.0 \left(\frac{P_t(20)}{P_t(35)}\right)^{1/2} = 0.8356 \text{ w/o per day} \qquad (2)$$

As per 10CFR50, Appendix J, Section III.A.4.a.1.iii, if equation (1) is greater than 0.7 w/o per day, equation (2) shall be the leakage limit used to determine the allowable leakage rate. 10CFR50, Appendix J, Section III.A.5.b.1 specifies that the maximum allowable leakage for the primary containment shall be 0.75 $L_t(20)$. This is less than the Technical Specification limit of the more restrictive of equations (1) and (2) and therefore more conservative. The limit for this test will then be 0.75 L_t (20) or 0.6267 w/o per day.

A general inspection of the accessible interior and exterior surfaces of the containment structures and components was performed prior to this Type A test to uncover any evidence of structural deterioration which may affect either the containment structure integrity or leak-tightness (10CFR50, Appendix J, Section V.A). No evidence of any significant structural deterioration was observed.

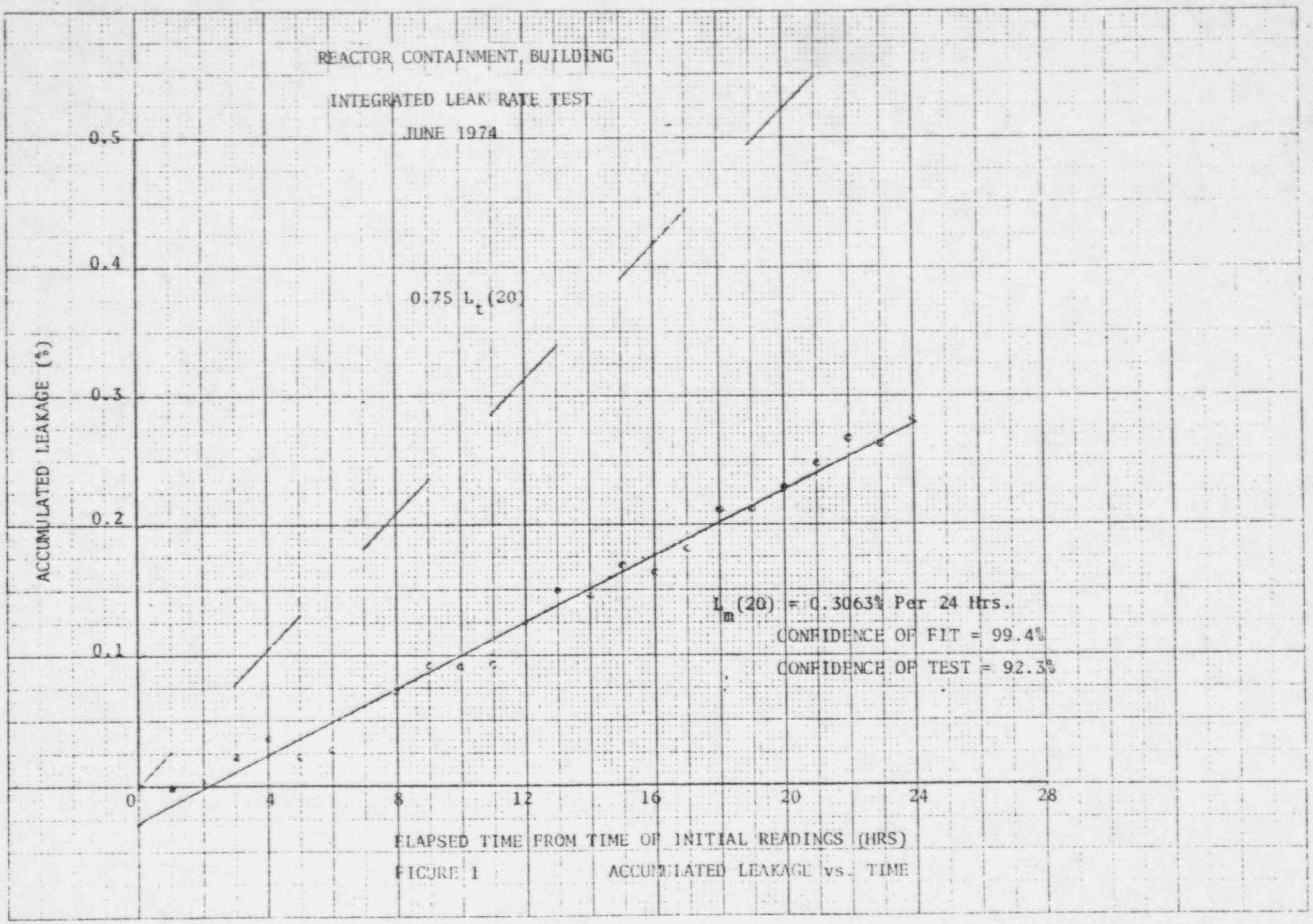

FIGURE 1 ACCUMULATED LEAKAGE vs. TIME

IV. Instrument Sensitivity Test

This test was performed by introducing a calibrated 50 SCFH leak on the containment system through a flow meter. Figure 2 illustrates the change in the observed accumulated leakage when the known leak was established. The observed leakage rate had been determined to be 0.3063 w/o per 24 hours, which corresponds to a leak of 90.2 SCFH. When the known leak was superimposed on this, the leak rate became 140.2 SCFH. The leakage measured during the instrument sensitivity test was 0.446 w/o per day or 131.2 SCFH.

This is well within the limit set by Section III.A.3.b of Appendix J which states that the difference between the sum of the test results plus the calibrated leak and the instrument sensitivity test results must be less than .25 L_t(20). The difference is 140.2 SCFH - 131.2 SCFH = 9 SCFH which is less than 61.5 SCFH (.25 L_t (20)).

Therefore, the operational instrument error in measurement was:

$$\frac{140.2 - 131.2}{140.2} = 6.4\%$$

V. Comparison of Test Results with those of Previous Tests

Test Date	Test Results (w/o per day)
March, 1969	0.216
October, 1970	0.276
June, 1972	0.549
June, 1974	0.306

III. Results

Tables 1 through 6 summarize the results of these tests. Note also that all tabulated leakage rates are adjusted to a test pressure of 20 psig, according to the relationship:

$$Lm(20) = Lm(35) \times \left(\frac{Pt(20)}{Pt(35)}\right)^{\frac{1}{2}} = L_m(35) \times .8356$$

As noted on Tables 1, 2 and 3 all tested penetrations met the "Acceptance Criteria" after all repairs were performed. All leakage test results from Type B and C tests that failed to meet the acceptance criteria of 10 CFR 50, Appendix J, Sections III.B.3 and III.C.3, respectively, since the last reactor containment building integrated leak rate test (June, 1974), are reported in a separate accompanying summary report.

All local leak rate tests that were performed since June, 1972, were acceptable except as noted in the above mentioned accompanying report or in the following tables.

Table 2

Testable Penetrations and Isolation Valves (1973)

Item	P_t (psig)	Final Leakage Rate, SCFH
Drywell Access Air Lock	10.1	Non-Detectable
Drywell Sump Discharge		
V-22-28, 29	35.0	5.84×10^{-3}
DWEDT Discharge		
V-22-1, 2	35.0	2.03×10^{-1}
All Electrical Penetrations (39)	35.0	Non-Detectable
Main Steam Isolation Valves		
NSO3A	35.0	Non-Detectable
NSO3B*	35.0	Non-Detectable
NSO4A	35.0	5.68
NSO4B	35.0	Non-Detectable
Steam Dryer Penetrations (16)	35.0	Non-Detectable

Total Leakage = 5.89 SCFH

Acceptable = 55.36 SCFH

Acceptable for Any One Penetration = 9.23 SCFH

* After Repair

Table 4

Double Gasketed Seals (1974)

Item	P_t (psig)	*Initial Leakage Rate, SCFH	Final Leakage Rate, SCFH
Steam Dryer Penetration	35.0		3.07
Drywell Airlock	35.0		Non-Detectable
TIP Penetrations (4)	35.0		Non-Detectable
Torus Manhole Covers			
North	35.0		1.06×10^{-3}
South	35.0		Non-Detectable
Drywell Head Seal	35.0		2.1×10^{-2}
Drywell Head Manhole Cover	35.0		Non-Detectable
Biological Shield Inspection Covers (8)	35.0		1.57
Torus to Drywell Vacuum Breakers (for double gasket and 2 sets of "O" rings (14)	35.0		5.7×10^{-2}
Torus to Reactor Building Vacuum Breakers (for double gasket and 2 sets of "O" rings (2)	35.0		Non-Detectable

Total Leakage = 4.72 SCFH

Acceptable = 18.45 SCFH

*For repaired valves and penetration

SEP 17 1974

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK

NUCLEAR GENERATING STATION

DPR-16

Report of Failures of Type A, B, and C Tests

June 1972 thru June 1974

Jersey Central Power and Light Company

Oyster Creek

Nuclear Generating Station

DPR-16

Report of Failures of Type A, B, and C Tests

June, 1972 thru June, 1974

Prepared By: R. M. Bright
E. V. Roscioli

September 5, 1974
Date of Approval

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September 5, 1974
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September 10, 1974
Date of Approval

W. L. Reaves, Jr. for
J. T. Carroll, Jr.
Station Superintendent

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Requirements

It is required by Section V.3 of 10 CFR 50, Appendix J, that leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria of Section III.A.5(b), III.B.3 and III.C.3, respectively, since the last acceptable Type A test, shall be reported in a separate summary report accompanying each reactor containment building integrated leak rate test.

There were no Type A tests performed since the last acceptable Type A test in June 1972.

In general, the combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La (176.67 SCFH). Included in this report are those penetrations and valves which give a leakage rate, at the time they were tested, that made the combined leakage rate for all penetrations and valves greater than 0.60 La.

1. Torus to Reactor Building Vacuum Breaker Valves V-26-17 and V-26-18

On May 3, 1973 an attempt was made to leak test the torus to reactor building vacuum breaker valves V-26-17 and V-26-18. Due to excessive leakage, this penetration did not pressurize to the test pressure.

The Torus to Reactor Building Vacuum Breaker Valves have a volume of 16.05 ft^3 and are pressurized through a 1/8" opening. Therefore large amounts of air cannot be supplied to this penetration quickly in order to perform a pressure decay test. At leakage rates above 70 SCFH, the penetration is extremely difficult to pressurize if, indeed, it can be pressurized at all.

Inspection of V-26-18 revealed that the valve disc was 0.010 inches off its seat, indicating that the linkage of the valve arm required adjustment. The boot seat and butterfly disc were cleaned and the valve linkage was adjusted to position the valve disc properly on the seat. The isolation valves were then retested for leakage and it was found to be approximately 0.49 SCFH.

Reference: Letter to Mr. A. Giambusso from Mr. D. A. Ross dated May 15, 1973.
Semi-Annual Report No. 8, January-June, 1973, pp. III-7 and VIII-3.

2. Drywell Head Manhole Cover Double Gasket

On May 27, 1973, the drywell head manhole cover double gasket was leakage tested and could not be pressurized to the test pressure due to excessive leakage. Visual inspection of the gaskets revealed that the outer gasket was brittle and cracked when pulled, however, the inner gasket was still pliable. Both gaskets were subsequently replaced and the new gaskets were leakage tested at 35 psig. The leakage was 2.95×10^{-3} SCFH.

Reference: Letter to Mr. A. Giambusso from Mr. D. A. Ross dated June 5, 1973.
Semi-Annual Report No. 8, January-June, 1973, pp. III-5 and VIII-3.

3. Main Steam Isolation Valve: NS03B

On September 9, 1973, MSIV NS03B was leakage tested and found to have a leak rate of approximately 200 SCFH. This leakage could only be estimated as the test was performed using a pressure buildup method instead of the normal flowmeter method. The cause of the leakage was traced to the valve disc which is not self-centering and therefore was not fully seated during closure. The valve was repaired by building up the poppet guides with stellite 21 and then machining the guides to the required tolerances. The leakage through the valve was again tested and could not be detected using an 0-11 SCFH flowmeter.

The data on the first leakage rate test could not be recovered.

Reference: Semi-Annual Report No. 9, July - December, 1973, pp. III-5 and VIII-3.

4. Main Steam Isolation Valve: NS04A

On September 27, 1973, MSIV NS04A was leakage tested and found to have a leak rate of approximately 96 SCFH. The cause of the leakage was traced to the main stem packing which was leaking. The packing was replaced and the leakage through the valve was again tested. There was no detectable leakage through the valve during the retest. The data on the leakage tests, which were pressure decay tests, could not be recovered.

Reference: Semi-Annual Report No. 9, July - December 1973, pp. III-6 and VIII-3.

5. Torus to Reactor Building Vacuum Breaker Valves V-26-15 and V-26-16

On April 4, 1974 an attempt was made to leak test the torus to reactor building vacuum breaker valves V-26-15 and V-26-16. This penetration could not be pressurized to the test pressure due to excessive leakage as commented upon in Section 1. Upon disassembly of valve V-26-16, it was found that a plug in the valve disc was leaking. The valve was cleaned, the plug sealed and V-26-16 reassembled. The valves were again leak tested and found to have leakage of approximately 11.8 SCFH.

Reference: Semi-Annual Report No. 10, January-June, 1974, pp. III-7 and VIII-22.

6. Torus Vent Valves V-28-17, V-28-18 and V-28-47

On May 7, 1974, the torus vent valves V-28-17, V-28-18 and V-28-47 were leak tested and found to leak at a rate of approximately 400 SCFH at the 35 psig test ressure. Because of the small volume of this penetration (1.41 ft^3) the pressure ecay readings could only be approximated. Therefore, a least square fit and an error analysis were not performed. Disassembly of the valves revealed that the seats of valves V-28-17 and V-28-18 had become brittle and non-pliable. Since no spare parts or replacement valves were available at the plant or from the manufacturer, attempts were made to repair the valves by fabricating new seats. These attempts proved unsuccessful as demonstrated by visual inspection and leak rate testing. As a result, replacement valves of a different manufacture were purchased, qualified and installed. Leak rate testing after installation revealed no detectable leakage.

Reference: Semi-Annual Report No. 10, Page VII-3.

7. Main Steam Isolation Valve Bypass Valves: V-1-106 and V-1-107

On June 11, 1974, during the performance of the Reactor Containment Building Integrated Leak Rate Test, the MSIV Bypass Valves, V-1-106 and V-1-107, were found to be leaking. The outer valves of this penetration (V-1-110 and V-1-111) were closed successfully and the test was continued. When the test was completed, the cause of the leakage was found to be non-mating sealing surfaces of these gate valves. The seating surface of V-1-106 was built up and then lapped to mate. The seating surface of V-1-107 was lapped to mate. The penetration was again leakage tested with a flowmeter and no leakage was detectable through the valves.

To find the leakage through the valves, the data for the six (6) hours prior to closing V-1-110 and V-1-111 from the RCBILRT was analyzed and subtracted from the leakage during the RCBILRT which is reported in the Reactor Containment Building Integrated Leak Rate Test, June 1974. The least squares fit of the leakage during the RCBILRT and the error analysis of the intrumentation used for the data collection are also reported there.

The data is:

	Date and Time	Accumulate w/o per hr leakage
6-10-74	2300	.128
	2400	.191
6-11-74	0100	.267
	0200	.307
	0300	.358
	0400	.419
	0500	.483

A least square fit of these data points results in a 1.3817 w/o per day leakage for the Reactor Containment Building plus V-1-106 and V-1-107 with a correlation co-efficient of .9976. When the leakage for the Reactor Containment Building (0.3063 w/o per day) is subtracted from this, the leakage through valves V-1-106 and V-1-107 was 1.0754 w/o per day or 315.64 SCFH.

Reference: Semi-Annual Report No. 10, January-June, 1974, pp. III-10 and VIII-1.

8. Drywell Access Airlock

On October 3, 1973, the drywell access airlock was leakage tested as required by 10 CFR 50, Appendix J, Section III.D.2 and was found to leak at a rate of 16.46 SCFH. Even though this leak rate was greater than that allowed for a single penetration, primary containment integrity was not violated. This test is being included in this report as an item of interest.

Upon inspection the air was found to leak out of the airlock through the electrical wires between the conductor and the insulation. The wires were cut and the electrical penetrations to the airlock sealed. The airlock was again leakage tested and found to have no detectable leakage. The penetrations are being supplied with airtight electrical connectors for the wires.

Reference: Semi-Annual Report No. 9, July-December, 1973, pp. III-8 and VIII-3.

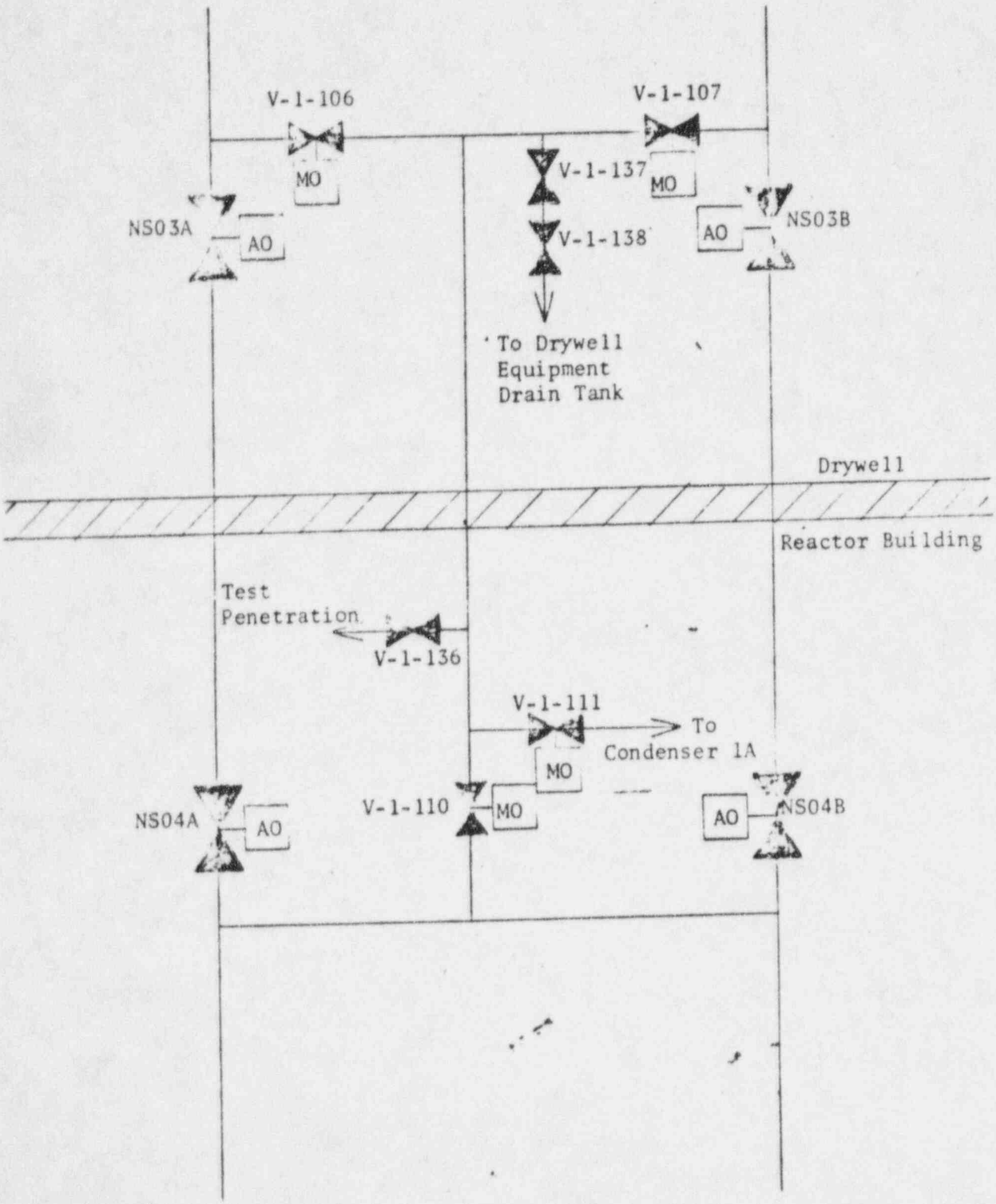
V-1-106
V-1-107
V-1-137
V-1-138
MO
MO
NS03A
AO
AO
NS03B
To Drywell
Equipment
Drain Tank
Drywell
Reactor Building
Test
Penetration
V-1-136
V-1-111
To
Condenser 1A
MO
V-1-110
MO
NS04A
AO
AO
NS04B

Figure B-1

INSTRUMENT ERROR ANALYSIS*

The measured leak rate in weight percent per day is computed using the reference vessel method by the formula:

$$M = (100)\ \frac{(24)}{(H)} \left[\frac{T_1\ \Delta P_2}{T_2\ P_1} - \frac{\Delta P_1}{P_1} \right] \qquad (1)$$

$$= (100)\ \frac{(24)}{(H)} \left[\frac{T_1\ P'_2}{T_2\ P_1} - \frac{T_1\ P_2}{T_2\ P_1} - \frac{P'_1}{P_1} + 1 \right]$$

where:

$P_1 = P_1 - P_{v1}$ = total containment atmosphere absolute pressure, in psia, at the start of the test, corrected for water vapor pressure.

$P_2 = P_2 - P_{v2}$ = total containment atmosphere absolute pressure, in psia, at the end of the test, corrected for water vapor pressure.

P'_1 , P'_2 = reference vessel pressure at the start and end of the test, in psia.

T_1 , T_2 = containment mean atmospheric temperature in ^{o}R, at the start and at the end respectively.

H = test interval in hours.

R = gas constant (assumed to be a constant for the entire range of pressure and temperature).

The change or uncertainty interval in M due to uncertainties in the measured variables is given by:

$$\delta M = \frac{2400}{H} \left[\left(\frac{dM}{dP_2} \cdot \delta P_2 \right)^2 + \left(\frac{dM}{dP_1} \cdot \delta P_1 \right)^2 + \left(\frac{dM}{dP'_2} \cdot \delta P'_2 \right)^2 + \left(\frac{dM}{dP'_1} \cdot \delta P'_1 \right)^2 + \left(\frac{dM}{dT_1} \cdot \delta T_1 \right)^2 + \left(\frac{dM}{dT_2} \cdot \delta T_2 \right)^2 \right]^{\frac{1}{2}} \qquad (2)$$

where δ is the standard error for each variable.

* Based on method used in Bechtel Topical Report BN-TOP-1

6. Containment mean dewpoint temperature = $80^{o}F$ based on 6 volume weighted dewpoint temperature sensors.

Equation (3) becomes:

$$e_M = 100 \left[2\left(\frac{e_P}{P}\right)^2 + 2\left(\frac{e_{P'}}{P}\right)^2 + 2\left(\frac{\Delta P \; e_T}{TP}\right)^2 \right]^{1/2} \qquad (4)$$

where:

e_P = the error in pressure which accounts for the error in the total containment pressure measurement system, both total absolute pressure and water vapor pressure.

$$e_P = \left[(e_{P_t})^2 + (e_{P_V})^2 \right]^{1/2}$$

e_{P_T} = RMS value of instrument error = error in total absolute pressure in psia.

e_{P_V} = RMS value of instrument error = error in water vapor pressure (dewpoint) indicator in psia at $80^{o}F$.

e_T = RMS value of instrument error = error in temperature, ^{o}R.

$e_{P'}$ = RMS value of instrument error = error in reference vessel pressure measurement, psia.

Based on the above derivation and formulae, and using the attached accuracy and repeatability data, two analyses are performed; the first using the accuracy values and the second using the repeatability values.

Using the attached accuracy values:

$$e_{P_t} = \sqrt{(.004)^2 + (.012)^2} = \pm .0126 \text{ psia.}$$

$$e_{P_V} = \sqrt{(.5/\sqrt{6})^2 + (.75/\sqrt{1})^2} = \pm .78^{\circ}\text{F.}$$

From steam tables at a dewpoint of 80°F.

$$e_{P_V} = \pm .019 \text{ psia}$$

$$e_P = \pm \sqrt{(.0126)^2 + (.019)^2} = \pm .023 \text{ psia.}$$

$$e_t = \pm \sqrt{(.5/\sqrt{10})^2 + (.75/\sqrt{1})^2} = \pm .766^{\circ}\text{F.}$$

$$e_{P'} = \pm \sqrt{(.023)^2 + (.0002)^2} = \pm .023 \text{ psia.}$$

$$e_m = \pm 100 \left[2\left(\frac{.023}{35}\right)^2 + 2\left(\frac{.023}{35}\right)^2 + 2\left(\frac{(.361)\ (.766)}{(549.7)\ (35)}\right)^2 \right]^{1/2}$$

$$e_m = \pm .131 \text{ wt percent/day}$$

Repeatability Error Analysis

Using the attached repeatability values:

$$e_{P_t} = \sqrt{(.00004)^2 + (.00012)^2} = \pm .000126 \text{ psia}$$

$$e_{P_V} = \sqrt{(.1/\sqrt{6})^2 + (.75/\sqrt{1})^2} = \pm .751^{\circ}F$$

From the steam tables at a dewpoint of 80°F

$$e_{P_V} = .0125 \text{ psia}$$

$$e_P = \sqrt{(.0125)^2 + (.000126)^2} = .0125 \text{ psia}$$

$$e_{P'} = \sqrt{(.0125)^2 + (.000002)^2} = .0125 \text{ psia}$$

$$e_t = \sqrt{(.1/\sqrt{10})^2 + (.75/\sqrt{1})^2} = .751^{\circ}F$$

$$e_m = 100 \left[2 \left(\frac{.0125}{35} \right)^2 + 2 \left(\frac{.0125}{35} \right)^2 + 2 \left(\frac{.361\ (.751)}{549.7\ (35)} \right)^2 \right]^{\frac{1}{2}}$$

$$e_m = \pm .071 \text{ wt percent/day}$$

It should be noted that since the test values change less than 1% during the test duration, the repeatability error analysis is more applicable.