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SUBJECT:
SUBMITTAL OF REPORT ENTITLED "INTEGRATED LEAK RATE TEST OF THE REACTOR CONTAINMENT BUILDING" OPERATING LICENSE NO. DPR-23.

PLANT NAME: H B ROBINSON - UNIT 2

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Carolina Power & Light Company

July 12, 1978

FILE: NG 3514 (R)

SERIAL: GD-78-1925

Office of Nuclear Reactor Regulation
Division of Operating Reactors
ATTENTION: Mr. A. Schwencer, Chief
Operating Reactors Branch No. 1
United States Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
OPERATING LICENSE NO. DPR-23
REACTOR CONTAINMENT BUILDING INTEGRATED LEAK RATE TEST

Dear Mr. Schwencer:

Pursuant to 10CFR50, Appendix J, Paragraph V.B.1, the enclosed report titled "Integrated Leak Rate Test of the Reactor Containment Building" is transmitted for your use. This report presents the results of the H. B. Robinson Unit 2 Reactor Building Integrated Leak Rate Test performed in February, 1978. Completion of this test and submittal of this report is also in accordance with H. B. Robinson Unit 2 Technical Specifications 4.4.1.1 and 6.9.3.a.

The results presented indicated a leakage rate at the upper bound of the 95 percent confidence interval well below the allowable leakage rate of 0.0424 percent by weight per day at the 21.0 psig test pressure.

All containment isolation valves isolated during the test were locally tested consistent with the requirements of 10CFR50, Appendix J and H. B. Robinson Unit 2 Technical Specification 4.4.1.1.d. The combined leakage from these valves was determined to have a negligible impact on the measured integrated leakage rate.

The results included in the report indicate that the containment leakage rate is well within acceptable limits and can perform its designed function in the unlikely event of a major accident. In accordance with the results of this test, 10CFR50, Appendix J, and Technical Specification 4.4.1.1.g, the next integrated leak rate test will be performed at the end of the current ten (10) year in-service inspection interval.

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In addition to the Integrated Leak Rate Test, Enclosure 2 provides the results of the Local Leak Rate Tests performed during the recent outage on valves which receive seal water from the Isolation Valve Seal Water (IVSW) System. These results are being submitted as followup to RII:HLW 50-261/78-04 in which it was identified to Carolina Power & Light Company that Local Leak Tests would be necessary on all isolation valves receiving seal water from the IVSW System.

If additional information is needed, please contact us.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

MFP:CSB/mf(897-175)
Enclosures

cc: Mr. R. A. Hartfield

TEST PERFORMED
February 1978

GAI REPORT NO. 1976

CAROLINA POWER & LIGHT COMPANY

INTEGRATED LEAK RATE TEST OF THE REACTOR CONTAINMENT BUILDING

H. B. ROBINSON

STEAM ELECTRIC PLANT

UNIT NO. 2

ISSUE DATE

5/15/78

GILBERT ASSOCIATES, INC.

P.O. Box 1498

Reading, Pennsylvania 19603

TABLE OF CONTENTS

1.0 SYNOPSIS

The H.B. Robinson Steam Electric Plant Unit No. 2 reactor containment building was subjected to a periodic integrated leak rate test during the period from February 3, 1978 to February 7, 1978. The purpose of this test was to demonstrate the acceptability of the building leakage rate at an internal pressure of 21.0 psig (P_t). Testing was performed in conformance with the requirements of 10 CFR 50, Appendix J, ANSI N45.4-1972 and H.B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications.

The measured leakage rate based on the mass point method of analysis was found to be 0.026 percent by weight per day at 21.0 psig. The leakage rate at the upper bound of the 95 percent confidence interval is 0.035 percent by weight per day which is well below the allowable leakage rate of 0.0424 percent by weight per day at 21.0 psig.

During the stabilization phase of the test, the seat on the outside purge exhaust valve, V12-8, was manually adjusted and a nipple was installed at the outlet of the containment vacuum relief innerspace penetration pressurization system test tap valve, PP30A, to stop leaks. During the collection of test data the following three sample line drain valves and the associated IVSW isolation valve were closed:

Penetration	Drain Valve No.	Sample Line Description
P-31	989F	RCS Hot Leg Loops 2 and 3
P-30	953B	Pressurizer Liquid Space
P-29	951B	Pressurizer Steam Space

The addition of the local leakage rate of V12-8, PP30A and the isolation valves in the three sample lines (leakage rate of the isolation valve with the highest leakage rate for each sample line was used; see section 5.3.5, Local Testing) determined subsequent to the integrated leakage rate test, must be considered. The combined leakage rate of all five of these isolation valves, determined by local testing, is .00007 percent per day. The addition of this negligible value does not change the measured integrated leakage rate.

The supplemental instrumentation verification at P_t was 10.5 percent, well within the 25 percent requirement of 10 CFR 50, Appendix J, Section III A.3.b.

All testing was performed by Carolina Power and Light Company with the technical assistance of Gilbert Associates, Inc. Procedural and calculational methods were witnessed by Nuclear Regulatory Commission personnel and audited by the Carolina Power and Light 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 21.0 psig, one half the calculated design basis accident pressure. 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 H.B. Robinson Steam Electric Plant Unit No. 2, the maximum allowable integrated leak rate at one half the design basis accident pressure, 21.0 psig (P_t), is 0.057 percent by weight per day (L_t).

Testing was performed in accordance with the procedural requirements as stated in H.B. Robinson Steam Electric Plant Unit No. 2, Integrated Leak Rate Test Procedure, TP 3-2-78. This procedure was approved by the H.B. Robinson Plant Nuclear Safety Committee prior to the commencement of the test.

Leakage rate testing was accomplished at the pressure level of 21.0 psig for a period of 24 hours. The 24 hour period was followed by a 6 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, ANSI N45.4-1972 and the H.B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications are as follows:

a. The measured leakage rate (L_{tm}) at one half the calculated design basis accident pressure of 21.0 psig (P_t) shall be less than 75 percent of the maximum allowable leakage rate (L_t), specified as 0.057 percent by weight of the building atmosphere per day. The 1974 Integrated Leak Rate Test established the L_{tm}/L_{am} ratio was greater than 0.7 and therefore the acceptance criteria is determined as follows:

$$L_t = L_a \left(\frac{P_t}{P_a} \right)^{1/2}$$

where

L_a = 0.08%/day

P_a = 42.0 psig

P_t = 21.0 psig

Substituting the values for L_a, P_a, and P_t,

L_t = 0.057% per day

and .75 L_t = 0.0424% per day.

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

4.0 TEST INSTRUMENTATION

4.1 SUMMARY OF INSTRUMENTS

The sensor locations were essentially the same as those used for the ILRT in 1974. Test instruments employed are described, by system, in the following subsections. An Instrumentation Figure of Merit (FOM), as discussed in Section 4.5, was calculated to be ±0.011%/day.

4.1.1 Temperature Indicating System

Components:

a. Resistance Temperature Detectors (sensors)

Quantity	21
Manufacturer	Rosemount
Type	Model 77, 100 ohm platinum (17)
	Model 104 MC (3)
	Model 78 (1)
Range, ^{o}F	-100 to +500

Range, ^oF (Calibrated)	32 to 212
Accuracy, ^oF (Interchangeability)	±0.9 (Model 77)
	±0.5 (Models 104 MC, 78)
Repeatability, ^oF	±0.09

b. Digital Temperature Indicator (Including Bridge Cards)

Quantity	1 Indicator (with 24 input channels)
Manufacturer	Rosemount
Type	Model 2501-4-A10-SF20 with 2506 Switching Units
Range, ^oF	0-200 (32^o to 122^oF maximum accuracy)
Accuracy, ^oF	±0.25
Repeatability, ^oF	±0.10

4.1.2 Dewpoint Indicating System

Components:

a. Dewcell Elements

Quantity	4

Manufacturer	Foxboro
Type	Model 2701RPG
Range, ^{o}F (Dewcell Element)	0-200
Accuracy, ^{o}F	±1.0
Repeatability, ^{o}F	±0.5

b. Dewpoint Recorder

Quantity	1
Manufacturer	Foxboro
Type	Model ERB
Range, ^{o}F (Dewcell Element)	0-200
Accuracy, ^{o}F	±1.0
Repeatability, ^{o}F	±0.4

4.1.3 Pressure Monitoring System

Precision Pressure Gauges

Quantity	1
Manufacturer	Texas Instruments
Type	Model 145 (with direct readout)

Range, psia	0-75
Accuracy, psia	0.010% of indication plus 0.002% of full scale
Repeatability, psia	±0.001% of full scale

4.1.4 Supplemental Test Flow Monitoring System

Flowmeter

Quantity	1
Manufacturer	Wallace & Tiernan
Type	Model 5210601108
Range scfm at 1 psig and $80^{\circ}F$	0-6
Accuracy, scfm	±1% of full scale

4.2 SCHEMATIC ARRANGEMENT

The arrangement of the four measuring systems summarized in Section 4.1 is depicted in Appendix A.

Temperature sensors were placed throughout the reactor containment building volume to permit monitoring of internal temperature variations at 21 locations. A temperature survey was performed after the sensors were installed which verified there were no large areas of temperature variation. Dewcells were placed in 4 locations. Placement of temperature sensors and dewcells was as follows:

Elev./Coordinates	No. of Temperature Sensors	No. of Dewcells
238'/S	1	0
238'/N	1	0
253'/SE	0	1
263'/NW	1	0
263'/NE	1	0
263'/S	1	0
263'/E	1	0
268'/NE	1	0
304'/W	1	1
304'/E	1	1
304'/N	1	0
335'/SW	1	0
315'/S	1	0
335'/NE	1	0
315'/N	1	0

Elev./Coordinates	No. of Temperature Sensors	No. of Dewcells
364'/S	1	0
364'/W	1	0
364'/E	1	0
364'/N	1	0
364'/NE	0	1
380'/W	1	0
400'/S	1	0
400'/N	1	0

4.3 CALIBRATION CHECKS

Temperature, dewpoint, pressure and flow measuring systems were checked for calibration before the test in accordance with H.B. Robinson Steam Electric Plant Unit No. 2, ILRT Instrumentation Pre-Test Calibration, TP 2-2-78, as recommended by ANSI N45.4-1972, Section 6.2 and 6.3. The results of the calibration checks are on file at H.B. Robinson Steam Electric Plant Unit No. 2. The supplemental test at 21.0 psig confirmed the instrumentation acceptability.

4.4. INSTRUMENTATION PERFORMANCE

During the calibration check, one temperature sensor and one dewcell element were broken. Two temperature sensors, one precision pressure gauge and one dewcell element did not pass the calibration check and were not used for the test. The remaining 21 temperature sensors, four dewcells, one precision pressure guage, and two flow meters performed well throughout the test and provided more than adequate coverage of the containment.

4.5 SYSTEMATIC ERROR ANALYSIS

Systematic error, in this test, is induced by the operation of 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 (FOM).

Utilizing the methods, techniques and assumptions in Appendix G to proposed ANSI N274, ANS-56.8, Draft 1, Revision 3, dated June 25, 1976, the FOM was computed for the absolute method as follows:

a. Conditions

L_t = 0.057%/day

P = 35.7 psia

T = 63°F = 522.69°R drybulb

Tdp = 46°F dewpoint

t = 24 hours

b. Total Absolute Pressure: e_p

No. of Sensors: 1

Range 0–75 psia

Measurement Sensor repeatability error (E_p): ±0.001% of full scale

$$E_p = 0.00001\ (75\ \text{psia})$$

$$E_p = 0.00075\ \text{psia}$$

$$e_p = \pm \left[(E_p)^2 + (\varepsilon_p)^2\right]^{1/2} / \left[\text{no. of sensors}\right]^{1/2}$$

$$e_p = \left[(.00075)^2\right]^{1/2} / \left[1\right]^{1/2}$$

$$e_p = \pm\ 0.00075\ \text{psia}$$

c. Water Vapor Pressure: e_{pv}

No. of sensors: 4

Sensor repeatability error (E): ±0.5°F

Measurement system error (ε), excluding sensor: ±0.4°F

$$E_{pv} = \pm 0.5^{\circ}\text{F}\ (0.0058\ \text{psia}/^{\circ}\text{F})$$

$$E_{pv} = \pm 0.0029\ \text{psia}$$

$$\varepsilon_{pv} = \pm 0.4^{\circ}\text{F}\ (0.0058\ \text{psia}/^{\circ}\text{F})$$

$$\varepsilon_{pv} = \pm 0.00232\ \text{psia}$$

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

$$e_{pv} = \pm \left[(0.0029)^2 + (0.00232)^2\right]^{\frac{1}{2}} / \left[4\right]^{\frac{1}{2}}$$

$$e_{pv} = \pm\ 0.00187 \text{ psia}$$

d. Temperature

No. of sensors: 21

Sensors repeatability error (E) = $\pm 0.09^{o}F = \pm 0.09^{o}R$

Measurement system error (ε), exluding sensor: $\pm 0.1^{o}F = \pm 0.1^{o}R$

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

$$e_T = \pm \left[(0.09)^2 + (0.1)^2\right]^{\frac{1}{2}} / \left[21\right]^{\frac{1}{2}}$$

$$e_T = \pm 0.0294\ ^{o}R$$

e. Figure of Merit (FOM)

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

$$FOM = \pm \frac{2400}{24} \left[2\left(\frac{0.00075}{35.7}\right)^2 + 2\left(\frac{0.00187}{35.7}\right)^2 + 2\left(\frac{0.0294}{522.69}\right)^2\right]^{\frac{1}{2}}$$

$$FOM = \pm\ 100 \left[8.827 \times 10^{-10} + 5.487 \times 10^{-9} + 6.328 \times 10^{-9}\right]^{\frac{1}{2}}$$

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

The FOM does not exceed 0.25 L_t (0.014%/day) and it is therefore concluded that the instrumentation selected was acceptable for use in determining the reactor containment integrated leakage rate.

4.6 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.

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_{tm}) 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_t).

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 and vented for the test.

f. Pressure gauges were installed on the following systems to provide a means of detection for leakage into these systems:

1. Purge Supply

2. Purge Exhaust

3. Personnel Access Lock Doors

4. Equipment Access Hatch

g. Containment recirculation fans were operational and orifices were installed in three of the four fans to prevent motor overload at test pressure.

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

i. 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 3.0 psi per hour. Building internal temperature was maintained at approximately 62°F. Building pressure and temperature were monitored half hourly and the amperage required by the recirculation unit fans (HVH-1,2,3 and 4) was monitored hourly. Leak rate testing was initiated at the 21.0 psig pressure level. Sixty-eight hours elapsed between reaching the 21.0 psig pressure level and the recording of official data. For the duration of the 24 hour leak test and the 6 hour supplemental test, the average internal containment temperature was maintained within a band of $\pm 0.3^{\circ}$F by varying the service water flow rate to the containment recirculation fan unit coolers.

During the test the following occurred at half-hour intervals (See Appendix C-Reduced Leakage Data):

a. Pressures indicated by the precision gauge was recorded.

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

c. The four dewpoint values were recorded. The average of the four 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}), temperature (T) and the total pressure (P_T) is described in more detail in Section 6.1. All original data is on file at H.B. Robinson Steam Electric Plant Unit No. 2.

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 24 hour period was available.

Immediately following the 24 hour leak test, a superimposed leakage rate was established for an additional 6 hour period. During this time, temperature, pressure and vapor pressure were monitored as described above.

5.3 TEST PERFORMANCE

5.3.1 Pressurization Phase

Pressurization of the reactor building containment was started on February 3, 1978 at 0220. The pressurization rate was approximately 3.0 psi per hour. During pressurization of the reactor containment building, a buildup of pressure between the purge exhaust valves V12-8 and V12-9 was noted. The exterior purge exhaust valve was checked and found to be leak tight and pressurization continued.

When containment internal pressure reached 21.1 to 21.2 psig, at 0900 on February 3, 1978, pressurization was secured. At 0920 valves RMS-1 and RMS-2 were discovered to be open which caused the containment internal pressure to drop below 21.1 psig. RMS-1 and RMS-2 were closed and it was decided to increase containment internal pressure to 21.2 psig. Pressurization was reinitiated at 1001 on February 3, 1978 and terminated at 1010 on February 3, 1978.

5.3.2 Integrated Leak Rate Testing Phase

The first set of data was recorded at 1015 on February 3, 1978. Containment internal temperature was controlled throughout the test by throttling the RB recirculation unit cooling coil water outlet valves, V6-34A,B,C and D. During the stabilization period the following was observed:

a. Pressure had built up between the purge exhaust valves, V12-8 and V12-9 and was equal to the containment internal pressure. At this time the outside purge exhaust valve was discovered to be leaking passed the seat. The valve was manually adjusted to tighten the seating surface and the seat leakage stopped. Subsequent to the integrated leakage rate test, the outside purge exhaust valve V12-8 was repaired and locally tested.

b. A leak was discovered at the containment vacuum relief innerspace penetration pressurization system test valve PP30A. A cap was installed on the nipple at the outlet of PP30A and the leak stopped. Subsequent to the integrated leakage rate test, valve PP30A was repaired and locally tested.

After waiting 4 hours, leak rate testing was started. Temperature had stabilized at approximately 62°F. From 1430 on February 3, 1978 until 0600 on February 6, 1978, an excessive leakage rate was indicated by the data collected. During this time, the following sequence of events took place:

a. At 1750 on February 3, 1978, the leakage rate, based on three hours of data, was 0.271 percent by weight per day. This established a baseline for the mass point versus time graph. Plant operators were sent on routine leak detection. There was no cause for immediate concern since only a limited amount of data had been collected.

b. At 0420 on February 4, 1978, the leakage rate, based on seven hours of data, was 0.231, percent by weight per day.

c. A small leak was found at a fitting in the containment pressure sensing line at the instrumentation panel. The leak was stopped by tightening the fitting.

d. At 0930 on February 4, 1978, the leakage rate based on two and one-half hours of data, was 0.247 percent by weight per day.

e. A valve lineup verification was performed on all systems but the penetration pressurization system and no deviations were found.

f. As leak detection continued, an excessive leakage rate was continuing and the measured containment leakage rates were as follows:

Date	Time Interval	Leakage Rate	95% Confidence Interval
2-3/4	2100-1600	0.034%/day	0.019%/day
2-4/5	1500-0330	0.085%/day	0.021%/day
2-4/5	2030-0330	0.058%/day	0.048%/day

g. Several leakage paths were identified and the following adjustments were made at 0415 on February 5, 1978.

1) The RCS Hot Leg Loops 2 and 3 sample line drain valve, 989F, and its associated IVSW isolation valve were closed.

2) The Pressurizer Liquid Space sample line drain valve, 953B, and its associated IVSW isolation valve were closed.

3) The Pressurizer Steam Space sample line drain valve, 951B, and its associated IVSW isolation valve were closed.

h. Following the adjustments made in item g. above it appeared that the measured reactor containment leakage rate was below the acceptance criteria value of 0.75 L_t.

The leakage rate as of 2300 on February 5, 1978, based on 10.5 hours of data was 0.021%/day. The associated 95 percent confidence interval was 0.012%/day. Data points plotted from 2300 until 0400 on February 6, 1978 indicated that the leakage rate was holding constant.

i. At 0400 on February 6, 1978, a sharp decrease in weight was indicated. On investigation two potential causes of the decrease were discovered:

1) Workmen had been performing a calibration check on the containment vessel wide range pressure transmitter, which affected the pressure sensing line for the ILRT precision pressure gauge. The workmen were instructed to discontinue the calibration check.

2) A 55 gallon drum had been placed on the ILRT precision pressure gauge line, pinching the poly tubing. The drum was removed from the tubing.

j. Following restoration of the conditions mentioned in item i above, an acceptable leakage rate of 0.026%/day with an associated 95 percent confidence interval of 0.009 percent by weight per day was obtained from 0600 on February 6, 1978 to 0600 on February 7, 1978.

5.3.3 Supplemental Leakage Rate Test Phase

After the 24 hour integrated leak rate test data was obtained and evaluated, the leakage rate found to be acceptable, and a release permit had been obtained, a leak rate was imposed on the reactor containment building through a calibrated flowmeter for a period of 6 hours.

5.3.4 Depressurization Phase

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

5.3.5 Local Testing

Subsequent to the ILRT, the following valves were locally leak rate tested:

Penetration	Valve	As Found Leakage (sccm)	As Left Leakage (sccm)	Total for Penetration (sccm)
P-38	V12-8 & 9 combination	60,900 ± 1700	0.2 ± 0.2	0.2 ± 0.2
P-42	PP30A	7530 ± 1700	0.2 ± 0.2	0.2 ± 0.2
P-29	956A	1.6 ± 1.6	1.6 ± 1.6	1.6 ± 1.6
	956B	1.6 ± 1.6	1.6 ± 1.6	
P-30	956C	32,900 ± 1700	31 ± 12	57 ± 12
	956D	3120 ± 1700	57 ± 12	
P-31	956E	7.2 ± 1.6	1.6 ± 1.6	1.6 ± 1.6
	956F	341 ± 20.0	1.6 ± 1.6	

For the five penetrations involved, the total as left leakage of 61 ± 12 sccm, which is equivalent to .00007%/day must be added to the ILRT results. The addition of this negligible value does not change the measured integrated leakage rate.

6.0 METHODS OF ANALYSIS

6.1 GENERAL DISCUSSION

The absolute method of leakage rate determination was employed during testing at the 21.0 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

$$K = 144\ V/R = 5.66782 \times 10^6\ \frac{\text{lbm} - {}^{o}\text{R} - \text{in.}^2}{\text{lbf}}$$

P = partial pressure of air, psia

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

$V = 2.1 \times 10^6\ ft^3$

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

$$P = P_{T1} - P_{wv}$$

where,

P_{T1} = true corrected total pressure from PI-201A, psia

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

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

$$T = \frac{\text{sum of 21 RTD's}}{21} + 459.69^{o}R$$

The weight of air is plotted versus time for the 24 hour test and for the 6 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 squared 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.

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

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

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.

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 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 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 + $W_1 t_i$) $\pm 2S_e$.

The 95 percent confidence interval 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 freedom

This parameter is an expression of the uncertainty in the measured leakage rate. The values of t_{95} used by the CLERCAL program establish a band about the calculated leakage such that there is a 95 percent chance the actual leakage rate is inside the band. There is only a 2.5 percent chance the actual leakage rate exceeds the upper limit.

7.0 DISCUSSION OF RESULTS

7.1 RESULTS AT P_t

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

Variable	Maximum Change
P_T	0.052 psia
P_{wv}	0.008 psia
T	0.57 ^oF

The method used in calculating the mass point leakage rate is defined in Section 6.0. The results of this calculation is a mass point leakage rate of 0.026%/day.

The 95 percent confidence interval associated with this leakage rate is 0.009 percent per day. Thus, the leakage rate at the upper bound of the 95 percent confidence interval becomes

$$L_{tm} = 0.026 + 0.009$$

$$L_{tm} = 0.035\%/\text{day}$$

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 0.0424 percent per day (0.75 L_t). 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 one half the calculated design basis accident pressure (P_t) of 21.0 psig is considered to be acceptable.

The accuracy of the one point that does not lie within the 95 percent confidence (the 16.5 hour data point) is questionable in that it is more than double the confidence interval. The leakage rate was calculated, excluding this point, and was determined to have no affect on the actual leakage rate at the upper bound of the 95 percent confidence interval of 0.035 percent per day.

7.2. SUPPLEMENTAL TEST RESULTS

After conclusion of the 24 hour test at 21.0 psig, flowmeter FI-1 was placed in service and a flow rate of 1.5 SCFM was established. This flow rate is equivalent to a leakage rate of 0.0424 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 0.062 percent per day using the mass point method of analysis. The 95 percent confidence interval associated with this leakage rate is 0.039 percent day. None of the 13 data points is out of confidence.

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

$$L_v' = L_c - L_o$$

$$L_v' = 0.062\%/\text{day} - 0.042\%/\text{day}$$

$$L_v' = 0.020\%/\text{day}$$

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

$$\frac{\left| L_{tm} - L_v' \right|}{L_t} = \frac{\left| (0.026) - (0.020) \right|}{0.057} = 0.105$$

The building leakage rates agree within 10.5 percent of L_t which is well below the acceptance criteria of 25 percent of L_t. Therefore, the acceptability of the test instrumentation is considered to have been verified.

8.0 TYPE B AND C LEAKAGE RATE HISTORIES

8.1 Discussion of Type B Tests

Since the Integrated Leak Rate Tests performed during the 1974 refueling outage, leakage measurements were obtained through the component leakage surveillance system at a pressure of 46.0 psig in lieu of Type B tests in accordance with 10 CFR 50 Appendix J, Section III B. The dates and measured leakages are reported below. It should be noted that leakage measured includes any leakage through the containment isolation valves which also are pressurized by this system.

Date	Leakage
June, 1974	0.102 SCFM
December, 1975	0.540 SCFM
October, 1976	0.671 SCFM

8.2 Discussion of Type C Tests

During the refuelings since the Integrated Leak Rate Test of 1974, leakage measurements were made on all isolation valves subject to Type C testing. The leakage reported excludes leakage from containment isolation valves that are sealed with fluid from a seal system in accordance with 10 CFR 50, Appendix J, Section III C.3. The leakage from these valves does not exceed that specified in the Technical Specifications and the isolation valve seal water system is sufficient to ensure the sealing function for at least 30 days at a pressure of 1.10 P_a. The only leakage measurement not subject to the above is the containment pressure manometer line isolation valve leakage measurement. The measured leakages for each individual measurement is as follows:

Date	Leakage
June, 1974	0
December, 1975	0
October, 1976	0

8.3 Discussion of Combined Leakage

The combined leakages measured since the last ILRT are listed below:

Date	Leakage
June, 1974	0.102 SCFM
December, 1975	0.540 SCFM
October, 1976	0.671 SCFM

All measurements were well below the acceptance criteria value ($0.60L_a$) of 2.78 SCFM.

9.0 REFERENCES

1. TP 3-2-78, "Integrated Leak Rate Test 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. ANS-56.8 N274, "Containment System Leakage Testing Requirements", American Nuclear Society, Draft No. 1, Revision 3 (June 25, 1976).

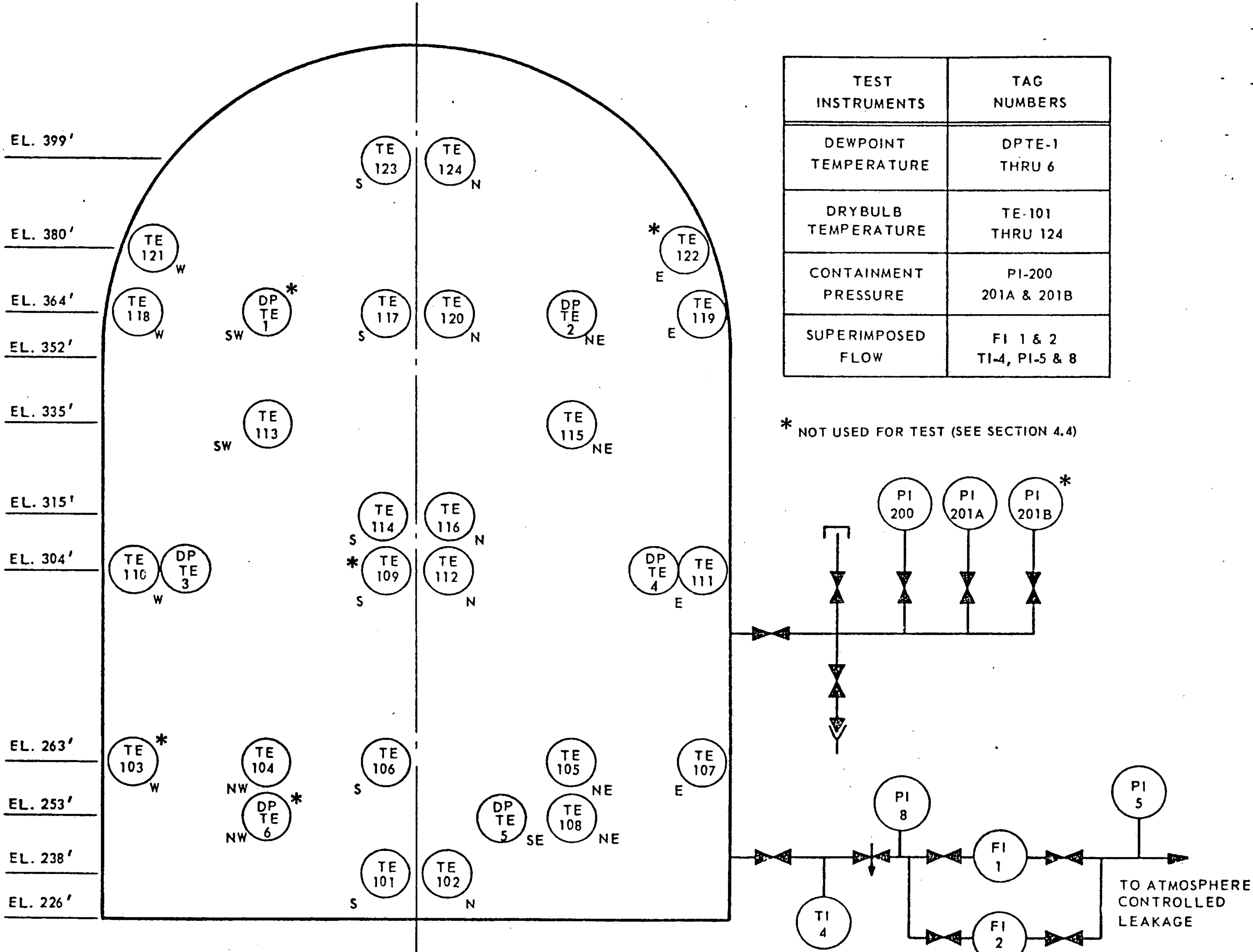
7. "H.B. Robinson Steam Electric Plant Unit No. 2 Reactor Containment Building Integrated Leak Rate Test and Structural Integrity Retest", Carolina Power & Light Company, (August 30, 1974).

APPENDICES

APPENDIX A

SCHEMATIC ARRANGEMENT OF TEST INSTRUMENTATION

APPENDIX A
SCHEMATIC ARRANGEMENT OF TEST INSTRUMENTATION



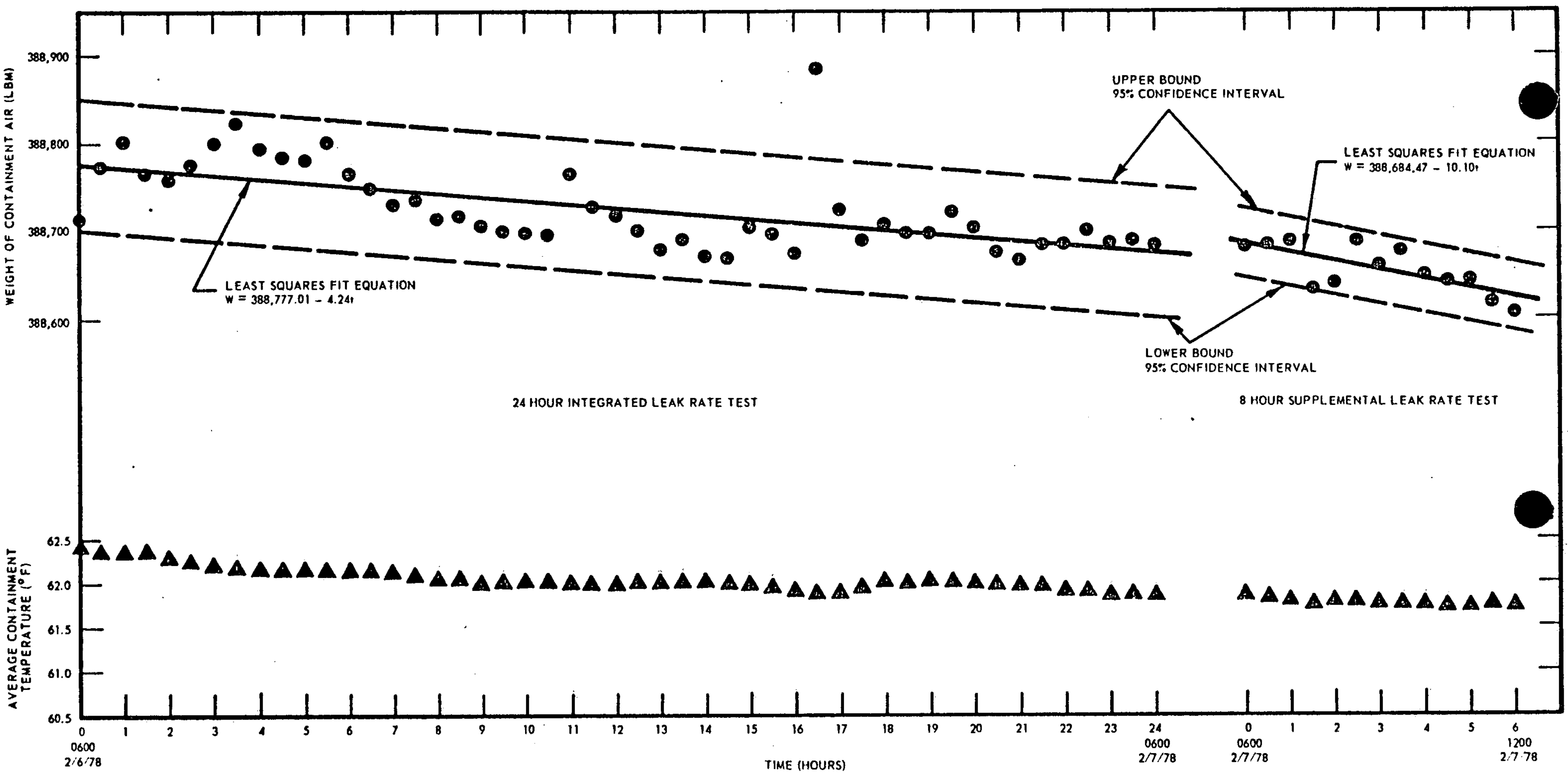
TEST INSTRUMENTS	TAG NUMBERS
DEWPOINT TEMPERATURE	DPTE-1 THRU 6
DRYBULB TEMPERATURE	TE-101 THRU 124
CONTAINMENT PRESSURE	PI-200 201A & 201B
SUPERIMPOSED FLOW	FI 1 & 2 TI-4, PI-5 & 8

* NOT USED FOR TEST (SEE SECTION 4.4)

APPENDIX B

WEIGHT OF CONTAINMENT AIR AND AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME

APPENDIX B
WEIGHT OF CONTAINMENT AIR AND
AVERAGE CONTAINMENT TEMPERATURE VERSUS TIME



APPENDIX C

REDUCED LEAKAGE DATA

APPENDIX C

REDUCED TEST DATA

	Time	Containment Pressure (psia)	Partial Pressure of Containment Water Vapor (psia)	Partial Pressure of Containment Air (psia)	Average Containment Temperature (^oR)	Weight of Containment Air (lbm)
2/6/78	0600	35.9632	0.1562	35.807	522.10	388,714[illegible]0
	0630	35.9632	0.1542	35.809	522.05	388,773.04
	0700	35.9652	0.1542	35.811	522.05	388,794.76
	0730	35.9622	0.1558	35.806	522.02	388,762.81
	0800	35.9582	0.1560	35.802	521.97	388,756.62
	0830	35.9552	0.1542	35.801	521.93	388,775.55
	0900	35.9542	0.1537	35.800	521.89	388,794.49
	0930	35.9522	0.1510	35.801	521.87	388,820.25
	1000	35.9502	0.1530	35.797	521.85	388,791.71
	1030	35.9483	0.1520	35.796	521.85	388,780.85
	1100	35.9472	0.1526	35.795	521.83	388,784.89
	1130	35.9492	0.1527	35.796	521.83	388,795.75
	1200	35.9453	0.1526	35.793	521.82	388,770.61
	1230	35.9433	0.1517	35.792	521.83	388,752.30
	1300	35.9413	0.1527	35.789	521.81	388,734.62
	1330	35.9383	0.1521	35.786	521.77	388,731.83
	1400	35.9353	0.1537	35.782	521.73	388,718[illegible]8
	1430	35.9333	0.1520	35.781	521.72	388,714.77
	1500	35.9313	0.1537	35.778	521.68	388,711.98
	1530	35.9303	0.1520	35.778	521.70	388,697.07
	1600	35.9303	0.1521	35.778	521.70	388,697.07
	1630	35.9293	0.1521	35.777	521.69	388,693.66
	1700	35.9363	0.1533	35.783	521.68	388,766.30
	1730	35.9303	0.1515	35.778	521.67	388,719.43
	1800	35.9293	0.1527	35.777	521.65	388,723.47
	1830	35.9303	0.1527	35.778	521.69	388,704.52

APPENDIX C (Cont'd)

REDUCED TEST DATA

	Time	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)
	1900	35.9293	0.1536	35.776	521.69	388,682.80
	1930	35.9293	0.1526	35.777	521.69	388,693.[illegible]6
	2000	35.9293	0.1544	35.775	521.69	388,671.93
	2030	35.9273	0.1539	35.773	521.67	388,665.10
	2100	35.9273	0.1521	35.775	521.65	388,701.74
	2130	35.9253	0.1527	35.773	521.62	388,702.36
	2200	35.9233	0.1543	35.769	521.60	388,673.80
	2230	35.9383	0.1527	35.786	521.56	388,888.35
	2300	35.9233	0.1511	35.772	521.58	388,721.30
	2330	35.9233	0.1515	35.772	521.62	388,691.49
	2400	35.9293	0.1511	35.778	521.69	388,704.53
2/7/78	0030	35.9293	0.1520	35.777	521.69	388,693.66
	0100	35.9293	0.1520	35.777	521.69	388,693.66
	0130	35.9273	0.1485	35.779	521.68	388,722.84
	0200	35.9263	0.1489	35.777	521.68	388,701.11
	0230	35.9243	0.1511	35.773	521.66	388,672.56
	0300	35.9223	0.1504	35.772	521.65	388,669.14
	0330	35.9213	0.1494	35.772	521.63	388,684.[illegible]4
	0400	35.9193	0.1494	35.770	521.60	388,684.67
	0430	35.9183	0.1489	35.769	521.57	388,696.16
	0500	35.9153	0.1494	35.766	521.54	388,685.91
	0530	35.9143	0.1480	35.766	521.54	388,685.91
	0600	35.9134	0.1485	35.748	521.53	388,497.75

SUPERIMPOSED TEST

Time	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)
0600	35.9134	0.1485	35.748	521.53	388,497.75
0630	35.9114	0.1485	35.763	521.50	388,683.12
0700	35.9094	0.1480	35.761	521.47	388,683.74
0730	35.9064	0.1515	35.755	521.45	388,633.43

APPENDIX C (Cont'd)

REDUCED TEST DATA

Time	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)
0800	35.9064	0.1494	35.757	521.47	388,640.27
0830	35.9074	0.1475	35.760	521.45	388,687.78
0900	35.9054	0.1480	35.757	521.45	388,655.17
0930	35.9044	0.1469	35.758	521.43	388,680.95
1000	35.9034	0.1490	35.754	521.41	388,652.38
1030	35.9024	0.1500	35.752	521.40	388,638.09
1100	35.9024	0.1494	35.753	521.41	388,641.51
1130	35.9044	0.1521	35.752	521.43	388,615.73
1200	35.9024	0.1527	35.750	521.41	388,608.90

APPENDIX D

COMPUTER PRINTOUT

APPENDIX D

COMPUTER PRINTOUT

CP&L H. B. ROBINSON UNIT 2 ILRT - START TIME 0600 ON 2/6/78

LEAST SQUARES RESULTS BASED ON 49 DATA POINTS

TIME PERIOD	OBSERVED WEIGHT (LB)	95% CONFIDENCE LIMIT MINUS	CALCULATED	PLUS	OBS. MINUS CALC. (LB)
0.0	388714.14	388702.11	388777.01	388851.90	-62.86
0.5	388773.09	388699.99	388774.88	388849.78	-1.80
1.0	388802.25	388697.87	388772.76	388847.65	29.48
1.5	388767.20	388695.75	388770.64	388845.53	-3.44
2.0	388758.83	388693.63	388768.52	388843.41	-9.69
2.5	388775.60	388691.50	388766.40	388841.29	9.20
3.0	388799.96	388689.38	388764.27	388839.17	35.69
3.5	388822.47	388687.26	388762.15	388837.04	60.31
4.0	388793.92	388685.14	388760.03	388834.92	33.89
4.5	388784.15	388683.02	388757.91	388832.80	26.24
5.0	388780.58	388680.89	388755.79	388830.68	24.80
5.5	388801.22	388678.77	388753.66	388828.56	47.56
6.0	388767.40	388676.65	388751.54	388826.43	15.86
6.5	388748.00	388674.53	388749.42	388824.31	-1.42
7.0	388730.31	388672.41	388747.30	388822.19	-16.98
7.5	388734.05	388670.28	388745.18	388820.07	-11.13
8.0	388713.88	388668.16	388743.05	388817.95	-29.18
8.5	388718.07	388666.04	388740.93	388815.82	-22.86
9.0	388707.67	388663.92	388738.81	388813.70	-31.14
9.5	388700.38	388661.80	388736.69	388811.58	-36.31
10.0	388699.29	388659.68	388734.57	388809.46	-35.28
10.5	388695.88	388657.55	388732.44	388807.34	-36.57
11.0	388766.34	388655.43	388730.32	388805.21	36.02
11.5	388728.16	388653.31	388728.20	388803.09	-0.04
12.0	388719.16	388651.19	388726.08	388800.97	-6.92
12.5	388700.22	388649.07	388723.96	388798.85	-23.73
13.0	388679.58	388646.94	388721.84	388796.73	-42.25
13.5	388690.45	388644.82	388719.71	388794.60	-29.27
14.0	388670.89	388642.70	388717.59	388792.48	-46.70
14.5	388669.49	388640.58	388715.47	388790.36	-45.98
15.0	388703.95	388638.46	388713.35	388788.24	-9.39
15.5	388698.06	388636.33	388711.23	388786.12	-13.17
16.0	388673.84	388634.21	388709.10	388784.00	-35.26

APPENDIX D (Cont'd)

COMPUTER PRINTOUT

TIME PERIOD		OBSERVED WEIGHT (LB)	95% CONFIDENCE LIMIT MINUS	CALCULATED	PLUS	OBS. MINUS CALC. (LB)
16.5	*	388884.04	388632.09	388706.98	388781.87	177.06
17.0		388723.52	388629.97	388704.86	388779.75	18.66
17.5		388689.36	388627.85	388702.74	388777.63	-13.37
18.0		388706.74	388625.72	388700.62	388775.51	6.13
18.5		388696.96	388623.60	388698.49	388773.39	-1.53
19.0		388696.96	388621.48	388696.37	388771.26	0.59
19,5		388720.71	388619.36	388694.25	388769.14	26.46
20.0		388705.50	388617.24	388692.13	388767.02	13.37
20.5		388674.77	388615.11	388690.01	388764.90	-15.23
21.0		388668.10	388612.99	388687.88	388762.78	-19.79
21.5		388683.00	388610.87	388685.76	388760.65	-2.76
22.0		388683.62	388608.75	388683.64	388758.53	-0.02
22.5		388700.55	388606.63	388681.52	388756.41	19.03
23.0		388684.87	388604.50	388679.40	388754.29	5.47
23.5		388689.22	388602.38	388677.27	388752.17	11.94
24.0		388681.45	388600.26	388675.15	388750.04	6.30

* - INDICATES VALUE OUTSIDE OF 95% CONFIDENCE

THE LEAST SQUARES EQUATION IS W = WO + W1 * T

WO = 388777.01 LB
W1 = -4.24 LB/HR
95% CONFIDENCE LIMIT FOR WO = 20.65 LB
95% CONFIDENCE LIMIT FOR W1 = 1.48 LB/HR
STANDARD ERROR OF CONFIDENCE = 37.45 LB
95% CONFIDENCE LEAKAGE RATE = 0.009% PER DAY
LEAKAGE RATE = 0.026% PER DAY

COMPUTER PRINTOUT

CP&L H. B. ROBINSON UNIT 2 SLRT - START TIME 0600 ON 2/7/78

LEAST SQUARES RESULTS BASED ON 13 DATA POINTS

TIME PERIOD	OBSERVED WEIGHT (LB)	95% CONFIDENCE LIMIT MINUS	CALCUALTED	PLUS	OBS. MINUS CALC. (LB)
0.0	388681.45	388645.61	388684.47	388723.34	-3.02
0.5	388682.08	388640.56	388679.42	388718.29	2.65
1.0	388688.13	388635.51	388674.38	388713.24	13.76
1.5	388632.39	388630.46	388669.33	388708.19	-36.93
2.0	388640.31	388625.41	388664.28	388703.14	-23.97
2.5	388686.74	388620.36	388659.23	388698.09	27.51
3.0	388659.56	388615.31	388654.18	388693.04	5.39
3.5	388675.56	388610.26	388649.13	388687.99	26.43
4.0	388649.32	388605.21	388644.08	388682.94	5.24
4.5	388642.48	388600.16	388639.03	388677.89	3.46
5.0	388641.55	388595.11	388633.98	388672.84	7.57
5.5	388619.04	388590.06	388628.93	388667.79	-9.89
6.0	388605.68	388585.01	388623.88	388662.74	-18.20

THE LEAST SQUARES EQUATION IS W = WO + W1 * T

WO = 388684.47 LB
W1 = -10.10 LB/HR
95% CONFIDENCE LIMIT FOR WO = 22.42 LB
95% CONFIDENCE LIMIT FOR W1 = 6.34 LB/HR
STANDARD ERROR OF CONFIDENCE = 19.43 LB
95% CONFIDENCE LEAKAGE RATE = 0.039% PER DAY
LEAKAGE RATE = 0.062% PER DAY

ENCLOSURE 2

Carolina Power and Light Company

H. B. Robinson Steam Electric Plant

Unit No. 2

LOCAL LEAK RATE TEST RESULTS
OF THE
CONTAINMENT ISOLATION VALVES
SERVICED BY THE
ISOLATION VALVE SEAL WATER SYSTEM

Report Date: June, 1978

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LOCAL LEAK RATE TEST RESULTS
OF THE
CONTAINMENT ISOLATION VALVES
SERVICED BY THE
ISOLATION VALVE SEAL WATER SYSTEM

By letter of March 9, 1978, the NRC formally notified Carolina Power and Light Company of a change in the interpretation of 10CFR50, Appendix J, as it relates to "Type C" local leak testing at H. B. Robinson. This change in interpretation required "Type C" local tests be performed on all containment isolation valves which receive seal water during containment isolation from the Isolation Valve Seal Water (IVSW) System.

Previous to this interpretation, pursuant to and consistent with Appendix J, subparagraph III. C.3 and subsection 4.4.2 of the H. B. Robinson Technical Specifications, the subject isolation valves have been leak tested during performance of the refueling interval periodic test of the IVSW system. The capability of the IVSW System to provide seal water and the total leakage from all the isolation valves receiving seal water, regardless of direction, at the seal water pressure of 46 psig is checked during this test. Pursuant to the exception noted in subparagraph III. C.3 of 10CFR50, Appendix J relating to seal systems, no individual local leak tests were previously required and none were performed.

Consistent with the requirements of the March 9, 1978 letter, "Type C" local leak tests were performed on the isolation valves which receive seal water from the IVSW System. The tests were completed and results of the tests are presented in the attached table.

Tests were performed using instrument air at 42 psig as the test medium. Test methods involved the use of the "in-leakage" and "out-leakage" measurements. Using the "in-leakage" method, the interspace between valves in series or between the seats of double disk valves was pressurized with air to the test pressure and the makeup air to the interspace volume was measured as leakage from the valves under test. Using the "out-leakage" method, a constant test pressure was maintained upstream of the valve tested and leakage was measured through a down-stream connection. Valves were tested in a conservative direction or in the direction of accident flow. Test procedures were developed using the guidelines of 10CFR50, Appendix J and proposed standard ANS-56.8 Draft 1, Revision 3, "Containment System Leakage Testing Requirements".

Results of the leak tests are presented both by valve and by penetration. The total leakage presented is the total from all pene-trations and is consistent with the reporting methods of Draft 1 of ANS-56.8. Valve acceptance criteria was based on a total leakage of 150 SCC/min per inch of valve diameter. This limit was taken from Section XI of the 1977 Edition of the ASME Boiler and Pressure Vessel Code. Final acceptance criteria was based on the total leakage of all valves not exceeding the sum of the individual acceptance values and no single valve exceeding 0.05 L_a.* Certain valves had leakage which exceeded the individual valve acceptance criteria;

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however, the total leakage from all valves was well within the final acceptance criteria and total leakage from all penetrations was well below the 10CFR50, Appendix J, (0.6 L_a^*) limit.

Prior to receiving official notification that local leak rate tests of the IVSW supplied isolation valves were required, the refueling interval test on the IVSW System had been performed. The system had performed satisfactorily although leakage greater than the acceptance criteria leak rate was detected through some of the valves. Some adjustments had been made for some of the valves and maintenance was scheduled for the remaining leaking valves. Prior to this maintenance, the individual "Type C" tests were performed on all valves serviced by the IVSW System including those requiring maintenance. During the "Type C" tests, all the valves which had passed the IVSW test showed leakage rates well within the individual valve acceptance criteria established for the local tests. Correspondingly, the only valves which indicated significant leakage had previously been identified by the routine refueling interval test. After concluding the "Type C" testing, including required maintenance, the leakage from the IVSW System was again checked and all header leakages were well within the limits established based on valve design.

Based on the resulting agreement between the refueling interval periodic test results and those of the "Type C" tests, Carolina Power and Light intends to seek credit, in the Safety Analysis, for the IVSW System and request relief from future "Type C" tests on isolation valves supplied by this system.

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*L_a is the allowable leakage as defined by Technical Specification 4.4.1.1.f (1) as L_p.

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PIPE LINE IDENTIFICATION	VALVE NO.	ACCEPTANCE CRITERIA* (SCC/min)	AS FOUND LEAKAGE (SCC/min)	AS LEFT LEAKAGE (SCC/min)	PENETRATION NO.	LEAKAGE FOR PENETRATION	TEST METHOD
Pressurizer Relief Tank to Gas Analyzer	RC-516 RC-553	56 56	1.6 $\pm$ 1.6 156 $\pm$ 1.6	1.6 $\pm$ 1.6 156 $\pm$ 1.6	1	156 $\pm$ 1.6	Out Leakage In Leakage
Primary Water Make-up to Pressurizer Relief Tank	RC-519A RC-519B	900	758 $\pm$ 20	758 $\pm$ 20	3	758 $\pm$ 20	In-Leakage
Reactor Coolant Drain Tank to Vent Header	WD-1786 WD-1787	300	0.2 $\pm$ 0.2	0.2 $\pm$ 0.2	4	0.2 $\pm$ 0.2	In-Leakage
Reactor Coolant Drain Tank to Gas Analyzer	WD-1794 WD-1789	224	12 $\pm$ 12	12 $\pm$ 12	5	12 $\pm$ 12	In-Leakage
Reactor Coolant Drain Pump Discharge	WD-1721 WD-1722	900	1.6 $\pm$ 1.6	1.6 $\pm$ 1.6	6	1.6 $\pm$ 1.6	In-Leakage
Component Cooling Water to Reactor Coolant Pumps	CC-716B	900	12 $\pm$ 12	12 $\pm$ 12	18	12 $\pm$ 12	Out-Leakage
Component Cooling Water from Reactor Coolant Pump Motors	CC-730	900	6.8 $\pm$ 0.2	6.8 $\pm$ 0.2	19	6.8 $\pm$ 0.2.	In-Leakage
Component Cooling Water from Reactor Coolant Pump Thermal Barrier	CC-FCV-626 CC-735	450 450	1.6 $\pm$ 1.6 1.6 $\pm$ 1.6	1.6 $\pm$ 1.6 1.6 $\pm$ 1.6	20	1.6 $\pm$ 1.6	Out-Leakage Out-Leakage
Reactor Coolant Letdown Line	CVC-204A CVC-204B	300 300	1.6 $\pm$ 1.6 90 $\pm$ 2.0	1.6 $\pm$ 1.6 90 $\pm$ 2.0	23	90 $\pm$ 2.0	Out-Leakage In-Leakage
Charging Line	CVC-282 CVC-202A CVC-309A	450 450 450	360 $\pm$ 20 1.6 $\pm$ 1.6 15.6 $\pm$ 1.6	360 $\pm$ 20 1.6 $\pm$ 1.6 15.6 $\pm$ 1.6	24	360 $\pm$ 20	In-Leakage Out-Leakage Out-Leakage
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PIPE LINE IDENTIFICATION	VALVE NO.	ACCEPTANCE CRITERIA* (SCC/min)	AS FOUND LEAKAGE (SCC/min)	AS LEFT LEAKAGE (SCC/min)	PENETRATION NO.	LEAKAGE FOR PENETRATION	TEST METHOD
Seal Water To	CVC-297A CVC-297B CVC-297C CVC-293A CVC-293C CVC-292A CVC-295 CVC-295A	1950	310 ± 20	310 ± 20	25 26 27	310 ± 20	In-Leakage
Reactor Coolant Pumps to RCP Seal Water Hx	CVC-381	450	9.2 ± 0.2	9.2 ± 0.2	28	9.2 ± 0.2	In-Leakage
Pressurizer Steam Space to Sample Hx	SS-956A SS-956B	56 56	1.6 ± 1.6 1.6 ± 1.6	1.6 ± 1.6 1.6 ± 1.6	29	1.6 ± 1.6	Out-Leakage
Pressurizer Liquid Space to Sample Hx	SS-956C SS-956D	56 56	46,337 ± 1700 4863 ± 1700	72.9 ± 12 130 ± 12	30	130 ± 12	Out-Leakage
Reactor Coolant Sample Line	SS-956E SS-956F	56 56	337 ± 20 546 ± 20	4.8 ± 1.6 1.6 ± 1.6	31	4.8 ± 1.6	Out-Leakage
Safety Injection to Hot Legs	SI-869	450	99,120 ± 1700	2.25 ± 0.2	43	2.25 ± 0.2	In-Leakage
Containment Spray System Header	SI-891A	900	532 ± 20	532 ± 20	44	532 ± 20	In-Leakage
Containment Spray System Header	SI-891B	900	25.6 ± 1.6	25.6 ± 1.6	45	25.6 ± 1.6	In-Leakage

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PIPE LINE IDENTIFICATION	VALVE NO.	ACCEPTANCE CRITERIA* (SCC/min)	AS FOUND LEAKAGE (SCC/min)	AS LEFT LEAKAGE (SCC/min)	PENETRATION NO.	LEAKAGE FOR PENETRATION	TEST METHOD
Safety Injection System Test Line	SI-895V	112	1.6 ± 1.6	1.6 ± 1.6	48	1.6 ± 1.6	Out-Leakage
	SI-898W	112	1.6 ± 1.6	1.6 ± 1.6			
Accumulator Sample Line	SS-956G	56	546 ± 20	19 ± 12	60	19 ± 12	Out-Leakage
	SS-956H	56	382 ± 20	1.6 ± 1.6			
Sump Pump Discharge	WD-1723 WD-1728	600	0.2 ± 0.2	0.2 ± 0.2	61	0.2 ± 0.2	In-Leakage
Safety Injection to Cold Legs and Recirc.	SI-870A SI-870B SI-883 SI-883L	1100	2.4 ± 0.2	2.4 ± 0.2	62 63 64	2.4 ± 0.2	In-Leakage
TOTAL	-	14,058	-	2538 ± 49	-	2437 ± 47	-

* Acceptance criteria is as follows: Individual values listed are comparative guideline criteria (not final criteria) based on Section XI of the 1977 Edition of the ASME Boiler and Pressure Vessel Code. Final acceptance criteria for satisfactory completion of the test was based on a maximum single penetration leakage of 0.05 La (6561 SCC/MIN) and a total leakage of the sum of the individual leakages (14,058 SCC/MIN) which is well below the 10 CFR 50 Appendix J limit of 0.6 La (78,730 SCC/MIN). The acceptance criteria for leakage is consistent with guidelines included in draft standard ANS-56.8.

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