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April 16, 1990
C311-90-2049

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Reactor Containment Building
Integrated Leak Rate Test
(10 CFR 50 Appendix J)

In accordance with Technical Specification Section 4.4.1.1.8 and 10 CFR 50 Appendix J, enclosed is the summary report entitled "Reactor Containment Building Integrated Leak Rate Test, 8R." This report addresses the Integrated Leak Rate Test (ILRT) conducted at the beginning of the 8R refueling outage during the period from January 9 - 13, 1990. The results of this test demonstrate the Reactor Building leakage to be well within the acceptance criteria.

Reactor Building Local Leak Rate Test (LLRT) Reports, included as Appendices F and G, contain a summary and analysis of the Type B and C tests performed to date since our last submittal dated February 9, 1987 which also demonstrate testing results well within the acceptance criteria.

GPUN believes that the results of the 8R ILRT satisfy the Technical Specification and Appendix J acceptance criteria and that no accelerated ILRT testing is required based upon these results. Therefore, the next ILRT should be performed at the Cycle 10 Refueling (10R) Outage.

Currently, 10 CFR 50 Appendix J requires an ILRT (which GPUN believes would be unnecessary) during the outage that coincides with the inservice inspection shutdown (9R for TMI-1). Therefore prior to the 9R Outage GPUN will need to request an exemption from Appendix J because of the connection between the ten year

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ISI schedule and the ten year ILRT schedule.[1] This exemption has been granted for other plants. Our exemption request is targeted for submittal in July, 1990 because we need to fix our schedules and plans for 9R as soon as practical.

The ILRT is a critical path item on the outage schedule. Also because of the need for contractor support, we will need the exemption approved no later than October, 1990 to properly prepare for the 9R Outage.

Sincerely,

H. D. Hukill
Vice President and Director, TMI-1

HDH/MRK

Attachment

cc: J. Stolz
R. Hernan
F. Young
T. Martin

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[1]The NRC's proposed rule on Containment Leak Rate Testing removes the connection between the ISI and ILRT schedules (see 51 FR 39538, dated October 29, 1986). Therefore, the need for an exemption for TMI-1 to perform the next ILRT during 10R would be eliminated if the rule goes into effect prior to Cycle 9 operation.

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

REACTOR CONTAINMENT BUILDING
INTEGRATED LEAK RATE TEST, 8R

JANUARY, 1990

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1.0 INTRODUCTION

The Three Mile Island Nuclear Station, Unit 1, (TMI-1) Reactor Containment Building was subjected to a periodic Integrated Leak Rate Test (ILRT) during the period from January 9, 1990, to January 13, 1990. The purpose of this test was to demonstrate the acceptability of the building leakage rate at the calculated Design Basis Accident pressure of 50.6 psig (P_a). The allowable leakage is defined by the Design Basis Accident, applied in the Safety Analysis, in accordance with the exposure guidelines specified by 10CFR100. For TMI-1, the maximum design basis integrated leakage rate at the design basis accident pressure, as defined in the FSAR, is 0.10 percent by weight per day (L_a). Testing was performed at the beginning of the 8R Outage prior to performing most of the Local Leak Rate Testing.

Integrated Leak Rate Testing (ILRT) was performed in accordance with the requirements of the TMI-1 Technical Specification (TS) 4.4.1.1, 10CFR50, Appendix J, ANSI N45.4-1972, ANSI/ANS 56.8-1987 (as applicable) and the procedural requirements specified in GPU Nuclear Corporation, TMI-1 Surveillance Procedure (SP) 1303-6.1. All testing was performed by GPU Nuclear Corporation with technical assistance provided by United Energy Service Corp., Volumetrics, and Gilbert/Commonwealth.

Local Leak Rate Testing (LLRT) performed since the previous ILRT is documented in Appendices F and G of this report. The testing for the 8R Outage is also documented herein, although most of the 8R testing was done after the ILRT.

The initial ILRT commenced at 1315 hours on January 10, 1990, following a satisfactory four (4) hour stabilization at test pressure. The leak rate was calculated using a computer and program provided by Gilbert/ Commonwealth.

Approximately eight (8) hours into the test, it became apparent that the Reactor Building leakage was stabilizing at about 0.11 w%/day using the Mass Point Method. An extensive search was made of all potential leak paths. No significant leakage was found on any of the containment isolation valves or other containment penetrations. The majority of the air leakage was attributed to in-leakage to the OTSG's, resulting in a drop in secondary side water level. The air in the Reactor Building had leaked through leaking body-to-bonnet gaskets on Hancock 5500W skin valves on the shell of the steam generator, into the Feedwater, Main Steam, and Emergency Feedwater piping, and out of the Reactor Building. The air then leaked through Feedwater Regulating Valve packing leaks and normally closed drain valves which had been opened to allow draining of the Secondary Plant for outage activities. The Reactor Building pressure at that time, was significantly higher than steam generator pressure. OTSG water levels were steadily dropping during the first test.

To correct this problem, the identified leakage paths on the Feedwater and Emergency Feedwater Systems were either isolated or shut. Since these were normally closed valves outside the scope of the ILRT valve lineup, this was permitted by procedure. To determine if the OTSG inleakage was the cause of the 0.11 w%/day leak rate, the N_2 overpressure on the OTSG's was also increased to 45 psig by Heise gauge indication. This process took about 4 to 5 hours while the test continued.

By 0300 hours, January 11, 1990, the Mass Point Leakage calculated began decreasing, indicating that the mass loss from the Reactor Building had either stopped or diminished significantly. By the completion of the initial 24 hour pressure drop test at 1315 hours, January 11, 1990, the calculated leakage by the Gilbert/Commonwealth computer was:

Mass Point Leakage:	0.0927 w%/day
95% Upper Confidence Level:	0.0959 w%/day

The TMI-1 design leakage limit of 0.1 w%/day was satisfied, but the test did not satisfy the Appendix J, 0.075 w%/day acceptance criteria to allow startup after the 8R Outage.

The TMI-1 FSAR, Section 5.2, and the TMI-1 TS 4.4.1.1.2, specify an allowable leakage of 0.1 w%/day at the Design Basis Accident pressure of 50.6 psig. Even with the excessive mass loss into the OTSG's, this criteria was satisfied. Appendix J, Section III A.5.(b) requires that this test criteria must be satisfied "... prior to any reactor operating period ...". To restart after 8R, therefore, the 0.075 w%/day, or 0.75 x 0.1 w%/day, criteria must be satisfied.

After reverification of low leakage through potential leakage paths and procedural compliance for the first test, a second test was begun at 2230 hours on January 11, 1990. Unfortunately, the local ambient weather conditions changed dramatically shortly after test start. These rapid temperature changes were reflected into the Reactor Building by the Industrial Cooling System. The test was aborted at 0800 hours, January 12, 1990 when it became apparent that the calculated leak rate was still not stabilizing after 10 hours of testing. The Industrial Cooling System was isolated from the Reactor Building at about 0915 hours, January 12, 1990, in an attempt to isolate the containment atmosphere from ambient changes and stabilize containment temperature.

Test conditions, at this time, were different in the following ways from the start of testing:

1. OTSG Secondary Side water level was stabilized. (The OTSG's were not returned to a Full Wet Layup condition).

2. N_2 overpressure on the OTSG's was increased from 30 psig to 45 psig. (This was still about 6 psig below containment pressure).

3. The Industrial Cooling System was isolated from the Reactor Building. (The Reactor Building ventilation fans were still operating to circulate the containment ai ass).

No other test conditions were changed.

The containment temperature was allowed to stabilize and the test was restarted at 1400 hours, January 12, 1990. This test went very smoothly and was terminated satisfactorily 24 hours later. The Mass Point "As Left" leakage as calculated by the Gilbert/Commonwealth computer was:

Mass Point Calculation:	0.0097 w%/day
95% Upper Confidence Level:	0.0126 w%/day

Since the Industrial Cooler System was not vented and drained during the Integrated Leak Rate Test, addition of the local leakage rate of the system isolation valves, RB-V2A and RB-V7, (46 sccm and 46 sccm respectively) to the measured integrated leakage rate was done. The combined local leakage rate of these isolation valves was 0.00005 percent by weight per day. Additionally, corrections for repair of AH-V1C are 0.0007 w%/day and tank level change corrections are -0.00009 w%/day. The addition of these values changed the calculated leak rate at the upper bound of the 95% confidence level to 0.01326 w%/day.

The supplemental Mass Point instrumentation verification difference at P_a was 0.0071 w%/day, which is within the 25% of L_a requirement of 10CFR50, Appendix J.

2.0 GENERAL AND TECHNICAL DATE

2.1 General Data

Owner:	General Public Utilities Nuclear Corporation (GPUNC)
Docket No.:	50-289
Location:	Three Mile Island, near the East Shore of the Susquehanna River in Dauphin County, Pennsylvania
Containment Description:	Reinforced concrete structure composed of cylindrical walls (prestressed with a post-tensioning tendon system in vertical and horizontal directions), with a flat foundation mat (conventional reinforcing) and a shallow dome roof (prestressed utilizing a three-way post tensioning tendon system). The inside surface is lined with a 3/8" thick carbon steel liner.
Date Test Completed:	January 13, 1990

2.2 Technical Data

Containment Net Free Volume:	2×10^6 cubic feet
Design Pressure:	55 psig
Design Temperature:	281 °F
Calculated Accident Peak Pressure:	50.6 psig
Calculated Accident Peak Temperature:	281 °F

3.0 ACCEPTANCE CRITERIA

Acceptance criteria were established as specified below. These criteria are specified in TMI-1 TS 4.4.1.1, 10CFR50, Appendix J. ANSI N45.4-1972.

a. The maximum allowable integrated leakage rate (L_a) from the Reactor Building at the calculated peak Reactor Building internal pressure of 50.6 psig (P_a) associated with the Design Basis Accident, shall not exceed 0.1 weight percent of the building atmosphere at that pressure per 24 hours (TS 4.4.1.1.2).

b. The calculated 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 0.1 percent by weight of the reactor building atmosphere per day at the upper bound of the 95% confidence prior to any reactor operating period. (Appendix J).

$$L_a = 0.10 \text{ weight percent/day}$$

$$0.75\ L_a = 0.075 \text{ weight percent/day}$$

c. The adequacy of the test instrumentation shall be verified by means of a supplemental test. The difference between the containment leakage calculated during the Type A test and the containment leakage determined during the supplemental test shall be within 25% of L_a (Appendix J).

4.0 TEST INSTRUMENTATION

4.1 Summary of Instruments

Test instruments employed are described in the following subsections. The instrumentation used satisfied the latest requirements for accuracy and sensitivity specified in Reference 9.6.

4.1.1 Temperature Indicating System

Resistance Temperature Detectors

Quantity	24
Manufacturer	Yellow Spring Instr.
Type	YSI Model, 4150-1/4-6-3-6-138-AW-G1/2-QR (platinum)
Range, °F	60-120
Accuracy, °F	± 0.1
Sensitivity, °F	± 0.01

4.1.2 Dewpoint Indicating System

Dewcell Elements

Quantity	10
Manufacturer	Foxboro
Type	BD154WB, Lithium Chloride
Range, °F	40-100
Accuracy, °F	± 1.5
Sensitivity, °F	± 0.1

4.1.3 Pressure Monitoring System

Precision Pressure Gauges Quantity	2
Manufacturer	Texas Instruments (modified by Volumetrics to interface with ILRT System)
Type	Model 145.02
Range, psia	40-100
Accuracy	± 0.015% of indicating pressure, ± 0.002% F.S.
Sensitivity, psi	± 0.001

NOTE: Since the pressure output was in terms of absolute pressure (psia), no instrumentation correction for barometric pressure was required.

4.1.4 Supplemental Test Flow Monitoring System

Flowmeter

Quantity	2
Manufacturer	Sierra
Type	Model 14636
Range, scfm	0.0 - 13.0
Accuracy	± 1 % full scale
Sensitivity	± 0.5 % of full scale

4.1.5 Outputs from the aforementioned sensors (with the exception of the output from the mass flowmeters) are forwarded to the Data Acquisition System (DAS) for conversion, display and forwarding to the data printer. The installed DAS unit was a Model A-100, manufactured by Volumetrics.

The DAS unit has the capability of monitoring over 100 channels. For the ILRT, the channels were utilized as follows:

a. Precision Pressure Gauge - 2 channels

b. RTDs - 24 channels (Channels 1 to 24)

c. Dewcells - 10 channels (Channels 30 to 39)

Output from the DAS went to a record printer, a utility printer, a local computer, and a remote computer (via modem).

4.1.6 Sensor input conditioning cards, precision pressure gauges, DAS unit, and other accessories were purchased as a rack mounted unit from Volumetrics by GPUN in March, 1984. Details of the unit and equipment specifications are available onsite for review.

4.1.7 After the DAS unit converted the signal input into the desired parameter of temperature or pressure, these values were transmitted to the computers and printers. The computer program performed the following functions:

a. Pressure - corrected the two inputs by the applicable calibration equation and averaged the two inputs. (One of the pressure inputs failed at about 0100 hours on January 11 during the first test. Thereafter, only one pressure input was used.)

b. Temperature - The data from the 23 RTD's was summed and the average was taken. No weighting factors were used. (One RTD was not scanned due to erroneously high readings. It was deleted from scan at about 1000 hours on January 10, 1990 prior to the start of the first test.)

c. Dewcells - The data from the 10 dewcells was summed and the average was taken. The average dewpoint temperature was then converted to partial pressure of water vapor using the Keenan and Keyes Steam Table Equation.

4.1.8 The accuracy of the DAS unit with respect to the different monitored parameters is given below:

a. Pressure - direct transmission of the number of counts from the precision pressure gauges to the computers and printers.

b. Dewpoint accuracy: ± 1.5 °F.

c. Temperature: ± 0.1 °F (60 °F to 120 °F range).

d. Repeatability - For automatic DAS:

Dry bulb temp: ± .01 °F

Dewpoint temp: ± .01 °F

Abs. press: ± .001 PSI

4.1.9 All operable RTDs and dewcells were assigned equal weighting factors. This is because:

a. There are very few cubicles inside the Reactor Building.

b. There is free communication between all levels of the building and also between the cubicles and the Reactor Building.

c. The air inside the Reactor Building is recirculated by the installed ventilation system.

d. Almost all of the equipment in the Reactor Building, with the exception of the aforementioned recirculating fans and required instrumentation, was deenergized during the test. This eliminated any heat producing equipment in the building which could cause local hot spots.

e. No stratification has ever been observed during an ILRT at TMI-1.

4.2 Schematic Arrangement

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

The arrangement of temperature sensors can be grouped into five levels as follows:

Level	Elevation	Sensors
1	287 feet	TE-655R TE-655S TE-655T TE-655U TE-655V
2	314 feet to 321 feet	TE-655M TE-655N TE-655O TE-655P TE-655Q
3	352 feet	TE-655A TE-655G TE-655I TE-655K

4	365 feet to 405 feet	TE-655D TE-655J TE-655L TE-655W TE-655X TE-655B
5	437 feet	TE-655C TE-655E TE-655H TE-655F

4.3 Calibration Checks

Temperature, dewpoint, pressure and flow measuring systems were checked for calibration before the test in accordance with GPU Nuclear Corporation Maintenance Procedure MP 1430-Y-23, as recommended by ANSI N56.8-1981. The results of the calibration checks are on file at Three Mile Island Nuclear Station, Unit 1. The supplemental test at 50.6 psig confirmed the instrumentation acceptability.

4.4 Instrumentation Performance

One of the pressure gauges failed at about 0100 hours, January 11, 1990 during the first test. Thereafter, its readings were not used. One of the RTD's read erroneously high and was not scanned during the period of testing. The remaining equipment performed satisfactorily throughout the integrated leakage rate test.

A third precision pressure gauge was installed to verify proper operation of the remaining unit. The output from this third gauge was recorded but not used in the calculation.

4.5 <u>Instrumentation Selection Guide Value</u>

Justification of instrumentation selection was accomplished, using manufacturer's sensitivity and repeatability tolerances stated in Section 4.1, by computing the instrumentation selection guide (ISG) value. Utilizing the methods, techniques, nomenclature and assumptions in Appendix G to ANSI N56.8-1981, the ISG was computed for the absolute method as follows:

a. Conditions

L_a = 0.1 w%/day

P = 65 psia

T = 81.25 °F = 540.94 °R dry bulb (typical)

Tdp = 67.2 °F dewpoint (typical)

t = 24 hours (test duration in hours)

b. Total Absolute Pressure: e_p

Sensor sensitivity error (E): ± 0.001% of full scale

Measurement system error (ϵ): ± 0.002% of full scale

$$e_p = \pm \frac{[(E_p)^2 + (\epsilon_p)^2]^{1/2}}{(no.\ of\ sensors)^{1/2}}$$

$$e_p = \pm \frac{[(0.001)^2 + (0.002)^2]^{1/2}}{(2)^{1/2}}$$

$$e_p = \pm .0016\ psia$$

c. Water Vapor Pressure: e_{pv}

Sensor sensitivity error (E): ± 0.1 °F

Measurement system error (ϵ), excluding sensor: ± 0.1 °F

At a dewpoint temperature of 67.2 °F, the equivalent water vapor pressure change (as determined from the steam tables) is 0.01149 psia/°F.

$$E_{pv} = \pm\, 0.1^{\circ}F\ (0.01149\ psia/^{\circ}F)$$

$$E_{pv} = \pm\, 0.001149\ psia$$

$$\epsilon_{pv} = \pm\, 0.1^{\circ}F\ (0.01149\ psia/^{\circ}F)$$

$$\epsilon_{pv} = \pm\, 0.001149\ psia$$

$$e_{pv} = \pm\, \frac{[(E_{pv})^2 + (\epsilon_{pv})^2]^{\frac{1}{2}}}{(no.\ of\ sensors)^{\frac{1}{2}}}$$

$$e_{pv} = \pm\, \frac{[(0.001149)^2 + (0.001149)^2]^{\frac{1}{2}}}{(10)^{\frac{1}{2}}}$$

$$e_{pv} = \pm\, 0.0005139\ psia$$

d. Temperature: e_T

No. of Sensors: 24

Sensor sensitivity Error (E_T): ± 0.01 °F = ± 0.01 °R

Measurement System Error (ϵ_T): ± 0.02 °F = ± 0.02 °R

$$e_T = \pm \frac{[(E_T)^2 + (\epsilon_T)^2]^{\frac{1}{2}}}{(no.\ of\ sensors)^{\frac{1}{2}}}$$

$$e_T = \pm \frac{[(0.01)^2 + (0.02)^2]^{\frac{1}{2}}}{(24)^{\frac{1}{2}}}$$

$$e_T = \pm 0.0045644\ ^\circ R$$

e. Instrumentation Selection Guide (ISG)

$$ISG = \pm \frac{2400}{t}\left[2\left(\frac{e_P}{P}\right)^2 + 2\left(\frac{eP_V}{P}\right)^2 + 2\left(\frac{e_T}{T}\right)^2\right]^{\frac{1}{2}}$$

$$ISG = \pm \frac{2400}{24\ hrs.}\left[2\left(\frac{0.0016}{65}\right)^2 + 2\left(\frac{0.000514}{65}\right)^2 + 2\left(\frac{0.00456}{540.94}\right)^2\right]^{\frac{1}{2}}$$

ISG = ± 0.00385 %/day

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

When the ISG is calculated for the actual system configuration (i.e., 1 precision pressure gauge and 1 RTD deleted) the revised ISG is ± 0.005138. GPUNC considers this acceptable because the third pressure gauge indicated

excellent correlation with the installed pressure gauge and the very small changes in Reactor Building pressure during the tests.

4.6 Supplemental Verification

4.6.1 Superimposed Test

In addition to the calibration checks described in Section 4.3, test instrumentation operation was verified by an approximately 6 hour flow 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 calculated leakage rate was

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

where,

L_c = calculated 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 calculated composite leakage rate.

The containment building leakage rate during the supplemental test ($L_{v'}$) was then compared to the calculated reactor containment building leakage rate during the preceding 24 hour test (L_{am}) to determine instrumentation acceptability. Instrumentation is considered acceptable if the agreement between the two building leakage rates is within 25 percent of the maximum allowable leakage rate (L_a).

5.0 TEST PROCEDURE

5.1 Prerequisites

Prior to commencement of the Reactor Building pressurization, the following basic prerequisites were satisfied:

a. Proper operation and calibration of all test instruments was verified.

b. All Reactor Building Containment Isolation Valves, with the exception of those in the Reactor Building Cooling System and Decay Heat System, were closed using the normal mode of operation. The Decay Heat System was in service to maintain the plant in a safe and stable condition during the test. The Reactor Building Cooling System was in operation during the first two tests to control the temperature during the tests. The system was secured during the final test, but the Reactor Building Recirculating Fans were kept running.

c. Equipment within the Reactor Building, subject to damage, was protected from external differential pressure.

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

e. The Penetration Pressurization System was depressurized. Manometers were installed at the penetration pressurization manifolds to provide means for detection of leaks into the system.

f. Pressure gauges or manometers were installed on the access hatches interspaces to provide indications of leakage into these areas.

g. Rotameters were installed on the interspaces between the inner and outer purge valves on both the Reactor Building Ventilation System inlet and exhaust. Flow through the rotameter was monitored during the pressurization and stabilization phase to determine whether leakage through the inner valve seats (i.e., the AH-V1B and V1C) was directionally sensitive. Testing was performed in accordance with STP 1-90-0005, AH-V1B/C Special Test During 8R Outage.

h. Local leak rate testing on selected containment penetrations "As Found" and "As Left" values were recorded. The ILRT results were adjusted, as required, to reflect any leakage corrected during these tests. Corrections were required for RB-V2A, RB-V7 and AH-V1C.

i. Potential pressure sources were removed or isolated from the building.

j. All accessible liner weld channels were vented to the containment atmosphere.

k. SP 1301-8.1, Reactor Building Annual Inspection, was performed. This procedure entails a visual inspection of the accessible interior and exterior areas of the containment. No significant problems were identified during this inspection.

l. The portions of the Fluid Block System still installed were placed in the following configuration:

   1) Manual valves were placed in the position required by the present normal valve lineup.

   2) Any automatic valves which were still operable were placed in the original post-accident condition.

   3) The non-seismic portions of the system still installed were vented to the extent practical based on existing system configuration.

   The Fluid Block System is presently disabled and is in the process of being dismantled.

5.2 Test Performance

5.2.1 Pressurization Phase

TMI-1 was shutdown for the 8R Outage on January 5, 1990. The ILRT was scheduled at the beginning of the outage to facilitate scheduling of work in the Reactor Building.

Pressurization of the Reactor Building began at about 1000 hours on January 9, 1990. A pressurization rate of approximately 2.5 psig/hour was maintained using rented diesel driven air compressors. Reactor Building temperature was monitored during the

pressurization. The recirculating fans were kept running to ensure adequate air mixing in the Reactor Building.

When containment pressure reached 12 psig at about 1500 hours on January 9, pressurization was secured to allow for an internal inspection of the Reactor Building. GPUNC supervisors entered the Reactor Building to inspect for any obvious damage due to the pressurization. The inspection was completed with no problems observed. Pressurization was restarted at 1600 hours.

No significant problems were identified during the pressurizations. The following observations were made:

1) Several of the installed manometers were found to be capped on the vent side. This cap prevented proper manometer performance. The caps were removed and the operators briefed on proper monitoring of the manometers.

2) Level in the secondary side of the OTSG was observed to be slowly decreasing.

3) To resolve concerns raised in IE Notice 88-73, the Reactor Building inside purge valves (AH-V1B and C) were leak tested in accordance with STP 1-90-0005, AH-V1B/1C Special Leak Test during 8R Outage. This STP concluded that pressurizing the inboard purge valve in the non-accident direction during LLRT provides equal test results. The inner seats remained basically leak tight as the Reactor Building was pressurized. This test was terminated and the associated STP test valves were placed in their ILRT required position prior to commencing the ILRT.

Pressurization of the Reactor Building was completed at 0900 hours on January 10, 1990, with the Reactor Building at about 51 psig pressure. The four (4) hour stabilization period was immediately started. The Reactor Building temperature changed by 1.69 °F (0.42 °F/hr.) during the four hour period and only 0.26 °F during the last one hour of this period. These parameters satisfied the procedural requirements and the ILRT was begun at 1315 hours, January 10, 1990.

5.2.2 Performance of Test

At 1315 hours on January 10, 1990, conditions in the Reactor Building were a pressure of 51.076 psig and a temperature of 82.75 °F. All prerequisites had been satisfied and the test was begun at this time. Data was collected automatically every 15 minutes by the computer. The following data was collected:

a. Reactor Building pressure by the precision pressure instruments.

b. The output from the 23 RTD's.

c. The output from 10 dewcells.

(One of the RTD's was deleted from scan prior to commencing the test due to consistent and erroneously high indications.)

The use of vapor pressure (P_{wv}), average temperature (T) and 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. Data for the average pressure, temperature and mass of air are maintained for each fifteen minute reading.

Periodically, the calculated leak rates were obtained. A graphic display of leak rates, temperature, pressure, etc., vs. time was also available. These displays were used in monitoring and trending the relevant parameters.

Reactor Building temperature was observed to decrease relatively quickly during the first four hours after starting the test. Overall Reactor Building temperature dropped by about 0.8 °F between 1315 hours and 1715 hours but stabilized, thereafter. This drop in temperature was probably due to ambient weather changes.

By 1900 hours, it became apparent that the calculated Reactor Building leakage was exceeding allowable values. The calculated Mass Point Leakage at 1900 hours was 0.0826 w%/day with a 95% UCL at 0.1134 w%/day. Both parameters were also showing an increasing trend. The Control Room was informed and a systematic search for the location of the leakage was begun.

The Control Room immediately dispatched on shift operators to examine the most obvious sources of leakage. The interspace volumes of the purge valves were checked for a pressure increase indicating leakage. Downstream vents from other containment penetrations were examined for air flow. SNOOP tests were performed on valve packing and instrument lines. The operators were instructed to locate leaks, but not to perform any corrective repairs, such as tightening valve packing. By 2100 hours, all obvious sources of leakage were checked, but the source of leakage could not be located.

Following this, other potential sources of leakage were examined. As noted earlier, the OTSG levels were dropping slowly due to inleakage from the pressurized containment combined with Feed/Steam System leaks outside of containment. No other evolutions involving the OTSG were in progress at this time. The inleakage was apparently through one or more OTSG skin valves. When the connections off of the secondary side of the OTSG's outside the Reactor Building were examined, several leaks were found. These included:

1. EF-V57 and EF-V58, which were found tagged open with the lines drained for work on the EFW Systems. N_2 and air was found leaking out of these drains.

2. Drains upstream of the Feedwater Regulating Valves which were opened to allow draining of the secondary plant. Water and air were found coming from these drains.

3. Leakage on MS-V70D, a steam line vent valve. This valve had an identified body-to-bonnet leak and was scheduled to be repaired after the ILRT. N_2 was coming out of this valve.

4. Gross water and air packing leaks were found on FW-V16A/B and FW-V17A/B.

Each source of water or gas leakage was identified and evaluated with respect to the test requirements and criteria prior to performing any repairs or isolation. Since these valves were out of their normal position and outside the test envelope, valves were shut or isolated and packing was tightened as required. These

activities were completed by about 2300 hours on January 10, 1990.

These activities appear to have been successful in eliminating the leakage. The calculated leakage reached a maximum of 0.1138 w%/day with a 95% UCL of 0.1300 w%/day at 2215 hours on January 10, 1990. Both parameters slowly dropped as the night went on.

Additionally, TCN 1-90-0013 was processed to allow increasing OTSG secondary side pressure to a pressure below the Reactor Building pressure. Control would be by a Heise, high accuracy pressure gauge. OTSG pressure was increased from 30 psig to 45 psig, by Heise gauge indications, to minimize the effects of the OTSG in-leakage from the Reactor Building. This pressurization also stabilized test conditions by filling the voids created by the water displacement. These voids would have been pressurized by air in-leakage to about Reactor Building pressure. Once pressurized, the N_2 source was isolated. The OTSG's had also been pressurized to 45 psig during the 6R ILRT.

The calculated leakage dropped slowly throughout the day of January 11, 1990. The test was run for a full 24 hour duration. The final calculated leakage at 1315 hours on January 11, 1990 was 0.0927 w%/day with a 95% UCL of 0.0959 w%/day. The test, at that time, was declared a failure. After further evaluation, it was determined that the leak rate was due to secondary plant conditions not related to the containment boundary and the test was declared an invalid test.

Further inspections for sources of leakage were performed on January 11, 1990, after the test completion. No significant leakage was identified. Minor leakage was documented but not repaired. Valve position, as required by the ILRT valve lineup, was not changed on any valve.

Since no sources of leakage could be located, it became apparent that the first test was not valid because of secondary side inleakage on the OTSG's due to leaking skin valves. The test requirements, valve lineup, and present plant conditions were reviewed and it was decided to start a second test at 2230 hours on

January 11, 1990. The OTSG's were at 45 psig by Heise gauge indications, which was about 6 psig below Reactor Building pressure. The previously identified water and gas leaks on the Feedwater, Main Steam and Emergency Feedwater Systems were isolated or repaired. It bears repeating again that no containment isolation valve or containment penetration was adjusted, tightened or repaired. Sources of leakage were identified, but not corrected.

The prerequisites were reverified, the temperature stabilization was verified, and a second test was begun. The four hour average was 0.13 °F/hr. with the last hour change 0.30 °F/hr. The second test began well, but during the night of January 11-12, 1990, a sudden change in ambient temperature caused an upset in Reactor Building temperature.

The Industrial Cooler, which was being used to control Reactor Building temperature, transferred this change in ambient conditions into the Reactor Building. As the temperature of the cooling water changed, the Reactor Building air moving past the cooling units also changed in temperature. The pressure change was instantaneous and indicated Building pressure changed slightly. The temperature response lagged this effect because it measured bulk air temperature. In order for the installed RTD's to respond, the bulk air temperature would have to change due to mixing from the air handling units. Since the temperature response lagged the pressure response by some time period, the computer used these non-synchronous inputs to generate a significant negative leakage.

The average temperature dropped by 1.5 °F from 2230 hours on January 11, 1990 to 0800 hours on January 12, 1990. The installed instruments and computer program could not accommodate these drastic temperature and pressure changes and calculated a significant long-term <u>negative</u> leakage. All containment penetrations were checked to ensure that no air sources had been inadvertently connected for some other activity. The calculated leakage was -0.0930 w%/day with a 95% UCL of -0.0768 w%/day at 0800 hours.

The test was terminated at about 0800 hours on January 12, 1990. TMI-1 considers this second

test to be invalid due to unstable temperature conditions. The results are not credible.

Due to continuing ambient temperature fluctuations, TCN 1-90-0015 was processed to isolate Industrial Cooling to the Reactor Building. This was done at 0915 hours. The stabilization for the third test began at 1000 hours, January 12, 1990. The stabilization was satisfactory, and the test began at 1400 hours, January 12, 1990. The four hour average change was 0.15 °F/hr. with the last hour change of 0.14 °F/hr.

At the start of the test, the Reactor Building pressure was 50.9 psig with a temperature of 81.74 °F. The Reactor Building pressure had dropped less than 0.3 psig since the start of the first test. No repressurization was required. OTSG secondary side pressure was 45 psig by Heise gauge indications.

In contrast to the previous testing, this test went very smooth with no significant problems. Reactor Building conditions were extremely stable during the test. Temperature increased slowly about 0.5 °F during the 24 hours. Reactor Building pressure changed by only 0.05 psig.

The final calculated Mass Point leakage at the end of the 24 hour test period, at 1400 hours January 13, 1990 was 0.0097 w%/day with a 95% UCL of 0.0126 w%/day. When the 95% UCL is corrected for Local Leak Rate Testing, it is increased to 0.0132 w%/day.

5.2.3 Supplemental Leakage Rate Test

After the 24 hour Integrated Leak Rate Test data was obtained and evaluated, the leak rate was found to be acceptable. A release permit was obtained and a known leak was imposed on the Reactor Building at 1400 hours on January 13, 1990, through a calibrated flowmeter for a period of approximately 6 hours. During this period, temperature, pressure and vapor pressure were monitored as described above. The average flow was 4.98 SCFM, which equates to 0.0825 w%/day. The computer calculated the leakage during this period to be 0.0851 w%/day.

When these values are analyzed, the difference is found to be 0.0071 w%/day as indicated below:

Calculated Leak Rate with Superimposed Flow	:	0.0851 w%/day
Calculated 24 Hr. Leak Rate	:	- 0.0097 w%/day
Superimposed Leakage	:	- 0.0825 w%/day
Difference	:	- 0.0071 w%/day

This value falls within the ± 25% of L_a (0.025) w%/day criteria and verifies that the installed instrumentation can quantify leakage from the Reactor Building.

5.2.4 Depressurization Phase

After all required data was obtained and evaluated and the supplemental test results were found to be acceptable, permission was obtained from Rad Con and the Control Room to depressurize the Reactor Building. The depressurization rate was about 5 psig/hr. Depressurization was completed on the morning of January 14, 1990. A post-test inspection identified no damage to any component due to the ILRT.

5.2.5 Pressurization of OTSG's During ILRT's

In order to ensure that air does not leak into OTSG's during ILRT's, the OTSG's are placed in a Full Wet Layup condition with a nitrogen overpressure applied to the Main Steam Lines. This in-leakage noted during the initial test has been traced to leaking skin valves on the OTSG's. Pressurizing the OTSG while in a Full Wet Layup condition will minimize the air inleakage, and therefore prevent deleterious oxygen assisted effects on secondary plant components. This is a standard practice at TMI-1. This is especially important during the ILRT's because sampling the secondary side to verify proper chemistry control, let alone correcting it, is not possible due to the valve lineup. This has been previously reported to the NRC in the 6R ILRT Report, Section 5.2.2. At that time, the OTSG's were pressurized to 45 psig using Plant Nitrogen.

SP 1303-6.1, Reactor Building ILRT, specifies the maximum allowable pressure and point of application in Section 3.1.3 for 8R. This allowable pressure was set at 35 psig to ensure that a wide margin (15 psig) was maintained between OTSG pressure and Reactor Building pressure. Actual pressure was 30 psig at the test start. It was believed at that time that 30 psig would reduce or completely eliminate air inleakage into the OTSG's. This assumption proved to be incorrect.

During Cycle 7 operations, several skin valve leaks were identified and repaired during a plant shutdown in November, 1989. Skin valves are generally defined as the first or second isolation valves off of the OTSG shell. These are instrument taps, sampling points, or drain isolation valves. They were temporarily repaired prior to the ILRT using leak sealing compound and the Furmanite process. These valves were then scheduled for replacement during 8R following the ILRT.

The Furmanite process, however, is sensitive to large temperature swings. A temperature change of ± 50 °F will generally break the temporary leak seal formed and require re-injection of more compound. These valves were resealed using Furmanite just prior to the ILRT at ambient conditions. Either this low temperature repair did not adequately seal the point of leakage or other valves had developed leaks, because OTSG levels were found to be decreasing during the ILRT. As noted, gas and water leakage was identified on the Feedwater, Main Steam, and Emergency Feedwater Systems outside the Reactor Building. No other plausible reason for the OTSG level drop could be identified.

When it became apparent that the size of this leakage was exceeding the allowable limits, these leakage paths were isolated or repaired. OTSG secondary side pressure was also increased to 45 psig in accordance with TCN 1-90-0013. A Heise high accuracy pressure gauge was installed to ensure that OTSG pressure was maintained below Reactor Building pressure.

This initial test was at first thought to be a failure of the ILRT. Further evaluation, however, did not support the original conclusion that the first test was a failure. Specifically:

1) None of the leakage paths were through Containment Isolation Valve (CIV's) or Containment Penetrations as they are generally defined for PWR containments.

2) The leakage path was abnormal in that the Feedwater and Emergency Feedwater Systems are normally pressurized full of water. Both of these systems were being drained for other outage activities. The open valves which allowed the unstable OTSG condition are normally closed while the Plant is operating.

3) No repairs or adjustments were made to any CIV or Reactor Building penetration.

4) The problem was eliminated once the OTSG's were pressurized and the leakage paths were eliminated.

5) Pressurizing the OTSG Secondary Side to 45 psig does not violate TMI-1 Tech. Specs. or Appendix J requirements for the ILRT, and is consistent with expected post-LOCA analysis conditions.

6) Regulatory guidance does not require leak rate testing of secondary plant valves for a PWR.

7) The OTSG's had been pressurized to 45 psig during the 6R ILRT.

This reasoning supports the premise that the first test is invalid and pressurizing the OTSG's, is an acceptable method of test performance.

GPU Nuclear was requested by the NRC to evaluate the practice of placing the OTSG's under a 45 psig nitrogen blanket. This analysis shows that in a post-accident condition, OTSG secondary side pressure quickly drops to 3 to 5 psig below Reactor Building pressure under all conditions analyzed. Since the OTSG's were pressurized to approximately 6 psig below the Reactor Building pressure, pressurizing the OTSG's to 45 psig

provides a more realistic determination of post-accident containment leakage than by depressurizing the OTSG's completely.

The first test, therefore, is not a failure of the ILRT when evaluated against the Type A leakage criteria but an invalid test. The problem was due to a previously unidentified leakage path outside the scope of the procedure. The ILRT procedure will be revised to correct this problem.

GPUN considers the 8R ILRT to be a valid "As Left" test and the Reactor Building to be operable for Cycle 8 operation. The next ILRT should be scheduled for the 10R Outage. (Note: an exemption from Appendix J will be required as discussed in the cover letter which transmits this report).

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 ILRT computer code calculates the weight percent per day leakage rate for the Mass Point method.

6.1.1 Mass Point Analysis

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 15 minute interval.

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

Where:

W = Mass of air inside containment, lbm

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

P = Partial pressure of air, psia.

T = Average internal containment temperature, °R.

$V = 2 \times 10^{6}\ \text{ft.}^3$

$R = 53.35\ \frac{\text{lbf} - \text{ft.}}{\text{lbm} - {}^{\circ}\text{R}}$

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

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

Where,

P_T = True corrected pressure by converting the pressure gauge readings and averaging, psia.

P_{wv} = Partial pressure of water vapor determined by averaging the dewpoint temperatures and converting to partial pressure of water vapor, psia.

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

T = Average of the operable RTD's + 459.69 °R

The weight of air is plotted versus time for the ILRT test and for the supplemental test. The ILRT 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 = At + B, where A is the slope in lbm per hour and B is the initial weight at time zero. The least squares parameters are calculated as follows from Ref. 9.7, App. B:

$$A = \frac{N\,(\Sigma t_i\, W_i) - (\Sigma t_i)(\Sigma W_i)}{S_{xx}}$$

$$B = \frac{(\Sigma t_i 2)(\Sigma W_i) - (\Sigma t_i)(\Sigma t_i\, W_i)}{S_{xx}}$$

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:

$$L_{am} = \frac{-2400\ A}{B}$$

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

6.2 Statistical Evaluation

6.2.1 General

After performing the least squares fit expressed in weight percent per day, the ILRT computer code calculates the limits of the 95% confidence interval for the mass point leakage rate (C_M).

These statistical parameters are then used to

determine that the measured leakage rate plus the 95% UCL meet the acceptance criteria.

6.2.2 Mass Point Confidence

The upper 95% confidence limit for the Mass Point leakage rate is calculated as follows:

$$C_M = 2400\ t_{95}\ (S_A/B)$$

Where:

C_M = Upper 95% confidence limit

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

S_A = Standard deviation of the slope of the least squares fit line

B = Intercept of the least squares fit line

The standard deviation of the slope of the least squares fit line (S_A) is calculated as follows:

$$S_A = \frac{S(N)^{\frac{1}{2}}}{\left[N(\Sigma t_i^2) - (\Sigma t_i)^2\right]^{\frac{1}{2}}}$$

Where:

S = Common standard deviation of the weights from the least squares fit line

N = Number of data points

t_i = Time interval of the ith data point

The common standard deviation (S) is defined by:

$$S = \left[\frac{\Sigma(W_i - W)^2}{N-2}\right]^{\frac{1}{2}}$$

Where:

W_i = Observed mass of air

W = Least squares calculated mass of air

The ILRT computer code calculates an upper 95% confidence leakage rate as follows:

$$UCL = L_{am} + 2400\ t_{95}\ (S_A/B)$$

This UCL value is then used to determine that the measured leakage rate at the upper 95% confidence limit meets the acceptance criteria.

7.0 DISCUSSION OF RESULTS

7.1 Test Results at P_a

The first two attempts at performing an ILRT were unsuccessful. The first test was declared invalid due to the OTSG skin valve leakage and procedural deficiencies noted. The second test was declared invalid due to a temperature upset condition. The third test produced acceptable results.

The method used to calculate the Mass Point Leakage Rate is detailed in Section 6. The result of this method provided a leak rate of 0.0097 w%/day. The leak rate at the 95% UCL is 0.0126 w%/day. When this is corrected for valve leakage, the 95% UCL is 0.0132 w%/day.

The measured leakage rate at the upper bound of the 95% UCL is well below the acceptance criteria of 0.075 w%/day (0.75 L_a). Therefore, Reactor Building leakage at the calculated Design Basis Accident Pressure (P_a) of 50.6 psig is considered to be acceptable. GPU Nuclear considered that the 1990 ILRT satisfies existing Tech Specs. and regulatory requirements. No accelerated ILRT testing is required.

With regard to the invalid tests, GPU Nuclear has implemented or intends to implement the following actions:

1. During 8R, all Hancock 5500W instrument root and drain/vent skin valves on the OTSG's were replaced with a different design valve which is much less prone to body-bonnet flange leakage.

2. The valves identified during 8R in the Main Steam, Feedwater and Emergency Feedwater Systems as affecting ILRT conditions will be added to the ILRT valve lineup.

3. The ILRT procedure will be revised to maintain 45 psig on the OTSG's while they are maintained in a Full Wet Layup condition.

4. The ILRT procedure will be revised to allow the Industrial Cooling Water flow to the Reactor Building coolers to be secured during future ILRT's once Reactor Building temperature stabilizes.

7.2 <u>Supplemental Test Results</u>

After completion of the 24 hour test at 50.6 psig, a mass flowmeter was placed in service and a flow rate of 4.98 scfm was established. This flow rate is equivalent to a leak rate of 0.0825 w%/day. After the flow was established, it was not altered for the duration of the supplemental test.

The measured composite leakage rate (L_c) using the absolute method during the supplemental test was calculated to be 0.0851 w%/day using the Mass Point Method.

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

$$L_{V'} = L - L_O$$

$$L_{V'} = 0.0851 - 0.0825\ w\%/day$$

$$L_{V'} = 0.0026\ w\%/day$$

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

$$\left| \frac{L_{am} - L_{V'}}{L_a} \right| = \left| \frac{0.0097 - 0.0026}{0.1} \right| = 0.071$$

The building leakage rates agree with 7.1% of L_a, which is within the ± 25% of L_a acceptance criteria.

Therefore, the acceptability of the test instruments is considered to have been verified.

8.0 <u>TYPE B AND C LEAKAGE RATE HISTORIES</u>

Refer to Appendices F and G for the report on Type B and C testing performed since the last ILRT report and following the 8R ILRT.

9.0 <u>REFERENCES</u>

1. SP 1303-6.1, "Reactor Building Integrated Leak Rate Test," Rev. 24.

2. ANSI N45.4 - 1972, "Leak Rate Testing of Containment Structures for Nuclear Reactors," American Nuclear Society, March, 1972.

3. MP 1430-Y-23, "Reactor Building Integrated Leak Rate test Instrument Calibration," Rev. 8.

4. 10CFR50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.

5. TMI-1 Technical Specification 4.4.1.1, Integrated Leakage Rate Tests.

6. ANSI/ANS 56.8, 1987; Containment System leakage Testing Requirements.

7. ANSI/ANS 56.8, 1981; Containment System Leakage Testing Requirements.

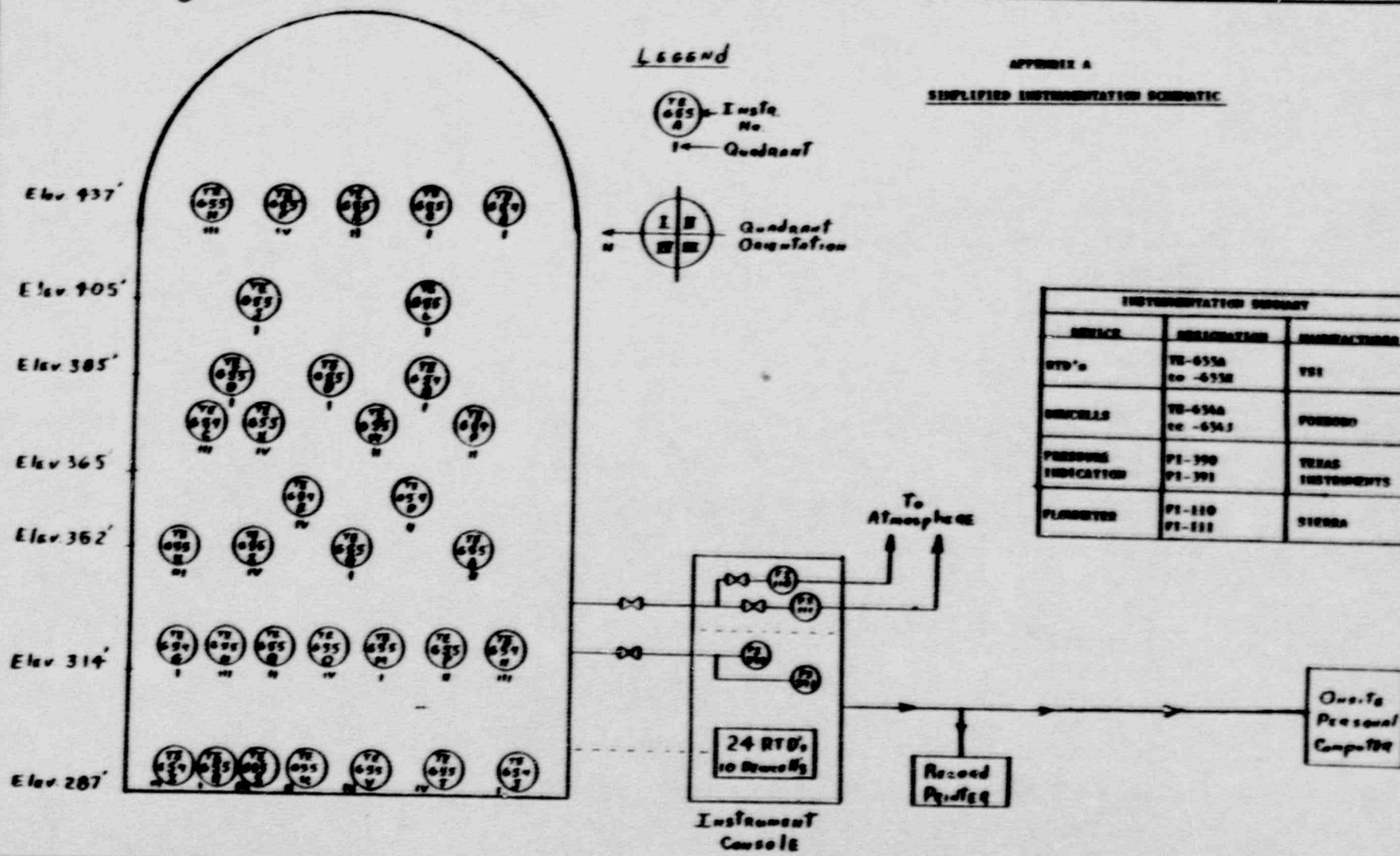
GPU Nuclear

Calculation Sheet

Subject: TMI-1 ILRT Schematic	Calc No. N/A	Sheet 8
Originator:	Reviewed by:	Date 2-2-87

APPENDIX A

SIMPLIFIED INSTRUMENTATION SCHEMATIC



INSTRUMENTATION SUMMARY		
DEVICE	DESIGNATION	MANUFACTURER
RTD's	TE-655A to -655W	YSI
DEWCELLS	TE-656A to -656J	FOXBORO
PRESSURE INDICATION	PI-390 PI-391	TEXAS INSTRUMENTS
FLOWMETER	FI-110 FI-111	SIERRA

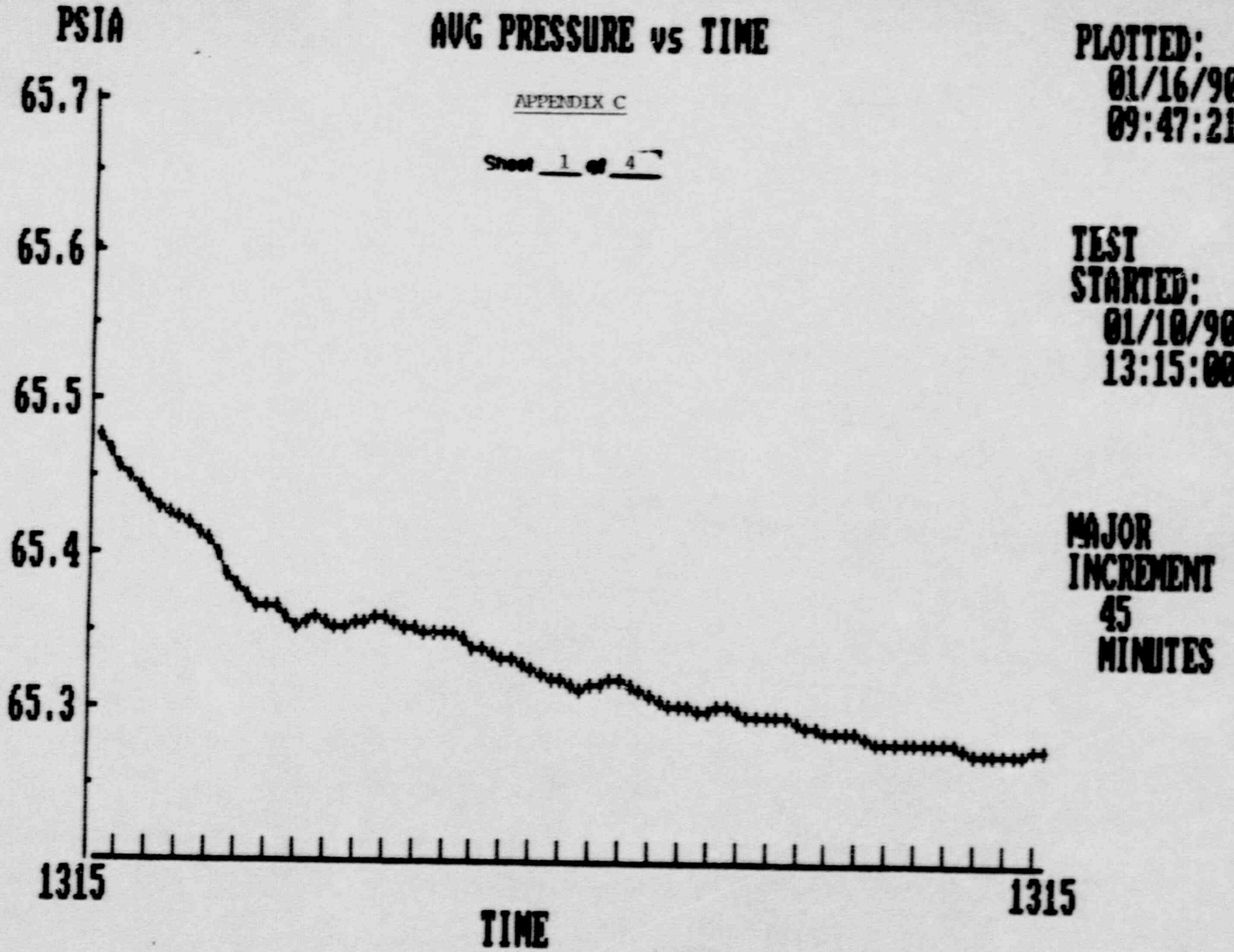
APPENDIX B: SUMMARY OF CHANGES AND DEFICIENCIES

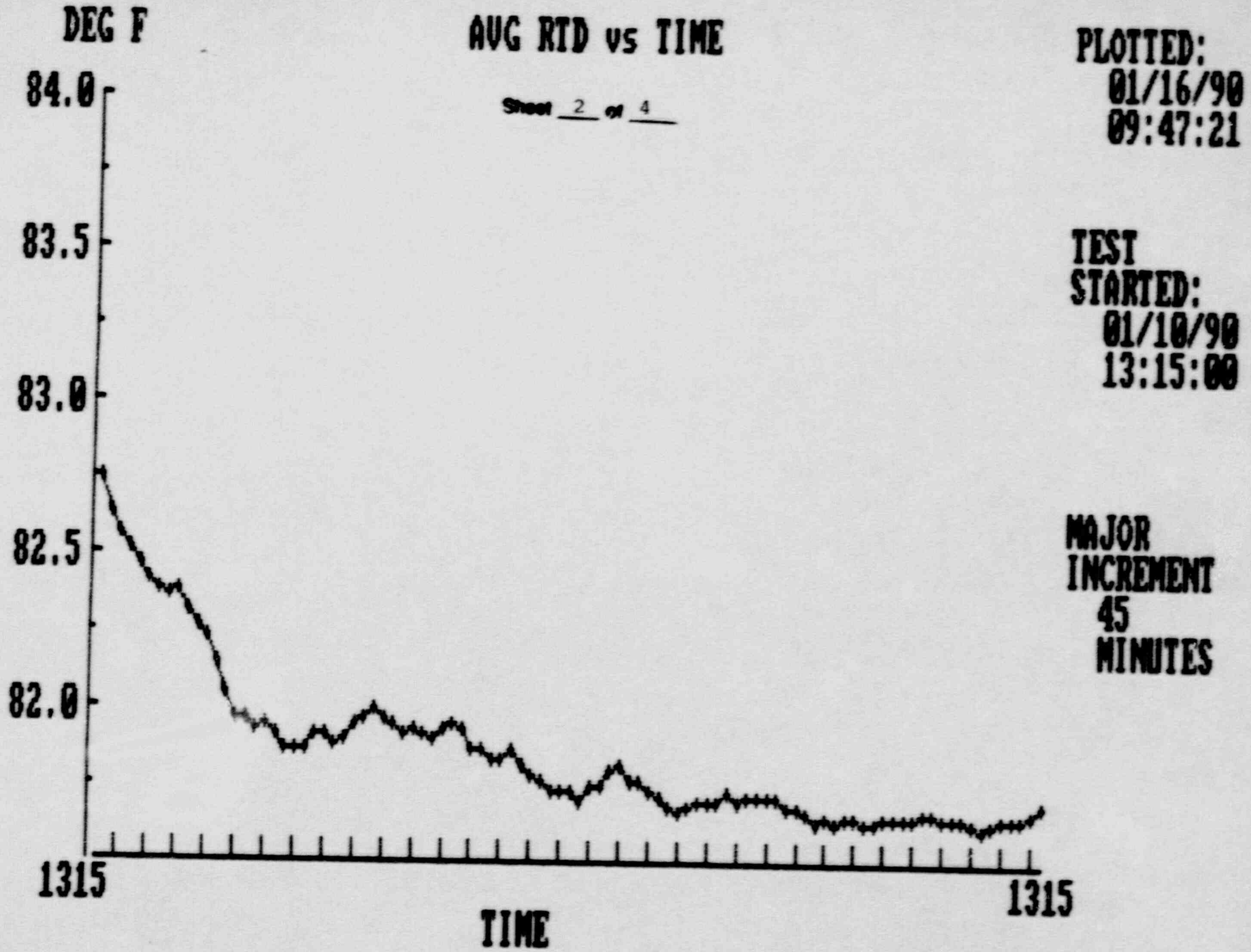
	DOCUMENT*	DATE	PURPOSE	COMMENTS
1.	TCN 1-90-0008	January 8, 1990	To clarify and correct procedural guidance based on actual plant conditions.	Provided guidance on valve lineup and RPS actuation components.
2.	TCN 1-90-0013	January 11, 1990	Increased allowable Secondary Side pressure. Actual pressure was set at 45 psig.	Also installed Heise Gauge to monitor and control pressure.
3.	SDR 1	January 10, 1990	Corrected pressure gauge number.	Incorrect number in SP 1303-6.1.
4.	SDR 2	January 11, 1990	Deleted manometer readings on Pen. Press. manifolds. Manometers were "Information Only" and were not installed.	Discrepancy between SP 1303-6.1 and MP 1430-Y-23.
5.	SDR 3	January 12, 1990	Identified Rx. Bldg. leakage exceeding 0.075 w%/day, but satisfying 0.1 w%/day design leakage.	Conservatively reported as a failure to NRC. Later evaluated as an invalid test result.
6.	SDR 4	January 12, 1990	Identified invalid ILRT results.	Test invalid due to temperature upset.

* TCN = Temporary Change Notice
SDR = Surveillance Deficiency Report

APPENDIX C

Sheet 1 of 4

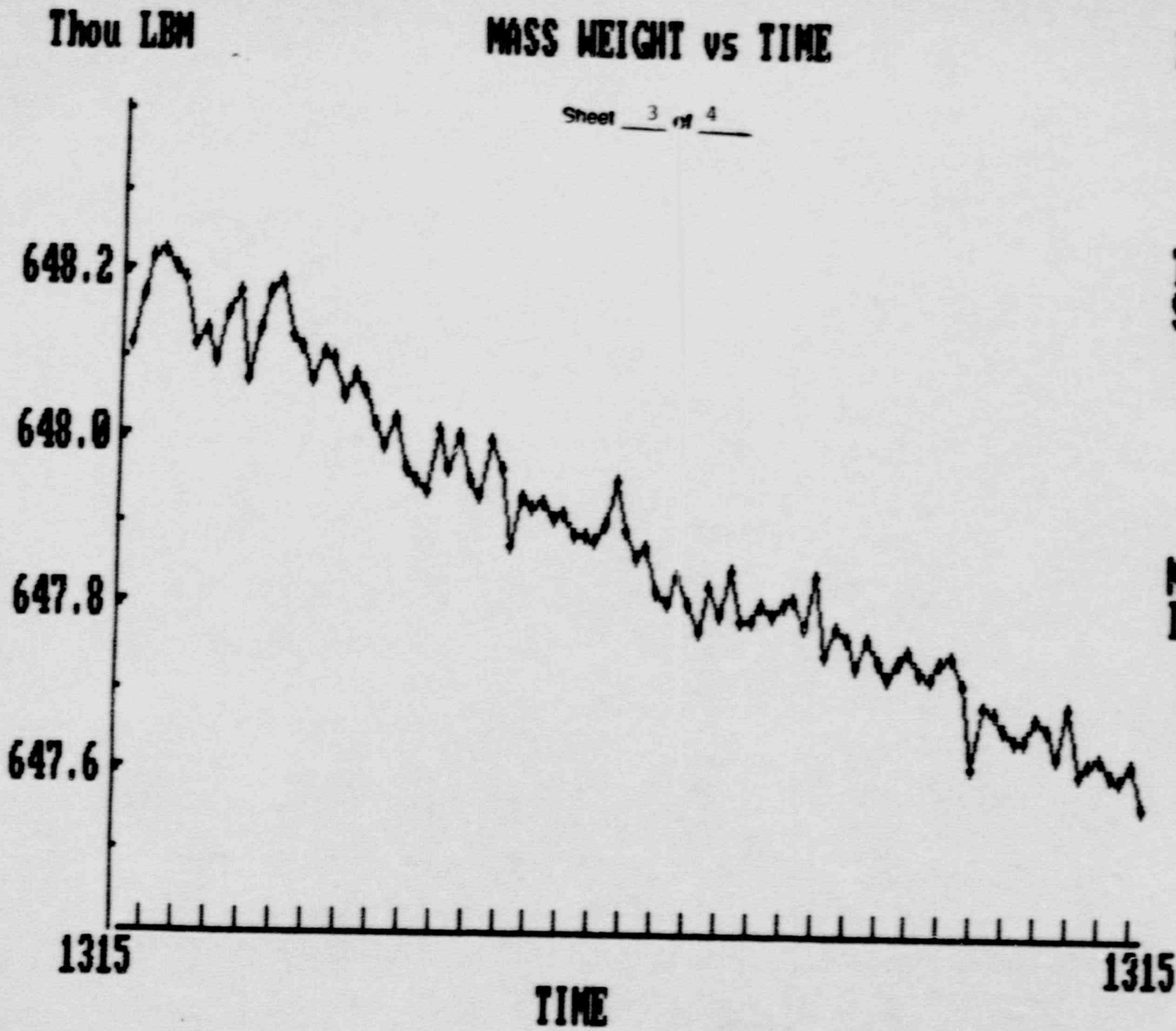




DEG F
AVG RTD vs TIME
Sheet 2 of 4
84.0
83.5
83.0
82.5
82.0
1315
1315
TIME
PLOTTED:
01/16/90
09:47:21
TEST
STARTED:
01/10/90
13:15:00
MAJOR
INCREMENT
45
MINUTES

MASS WEIGHT vs TIME

Sheet 3 of 4



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TEST
STARTED:
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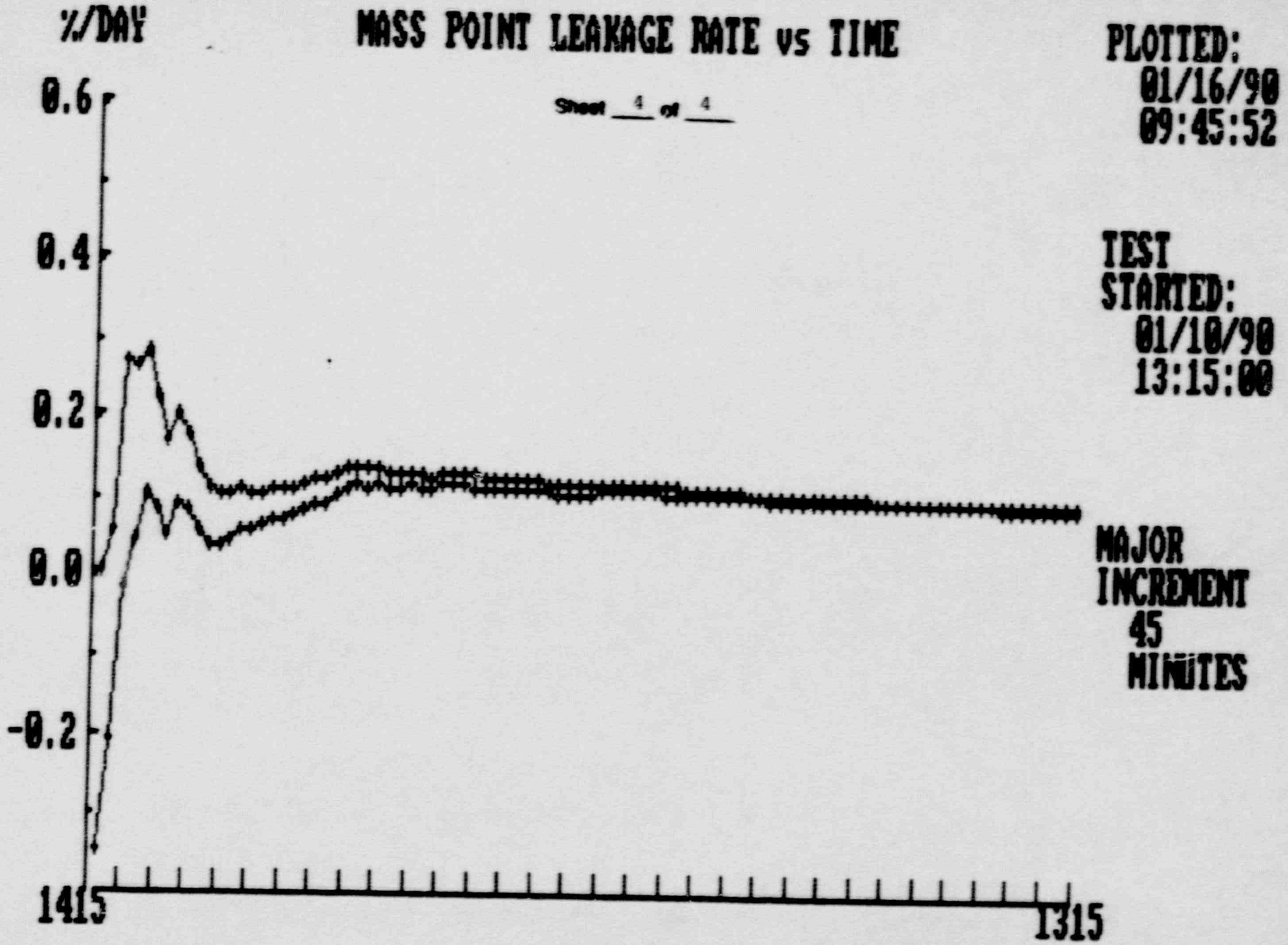
13:15:00

MAJOR
INCREMENT
45
MINUTES



%/DAY
MASS POINT LEAKAGE RATE vs TIME
Sheet 4 of 4
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STARTED:
01/10/90
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INCREMENT
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MINUTES
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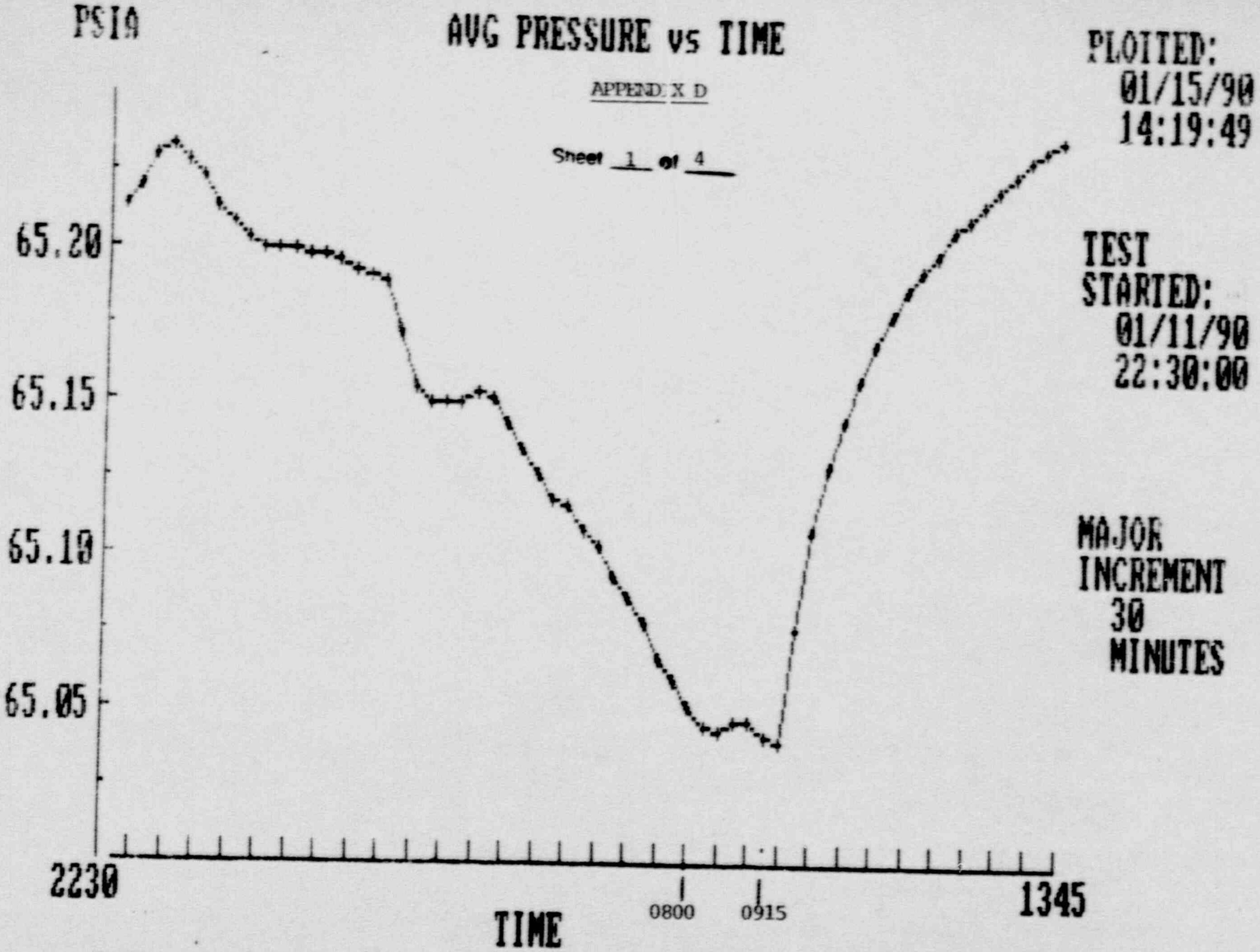
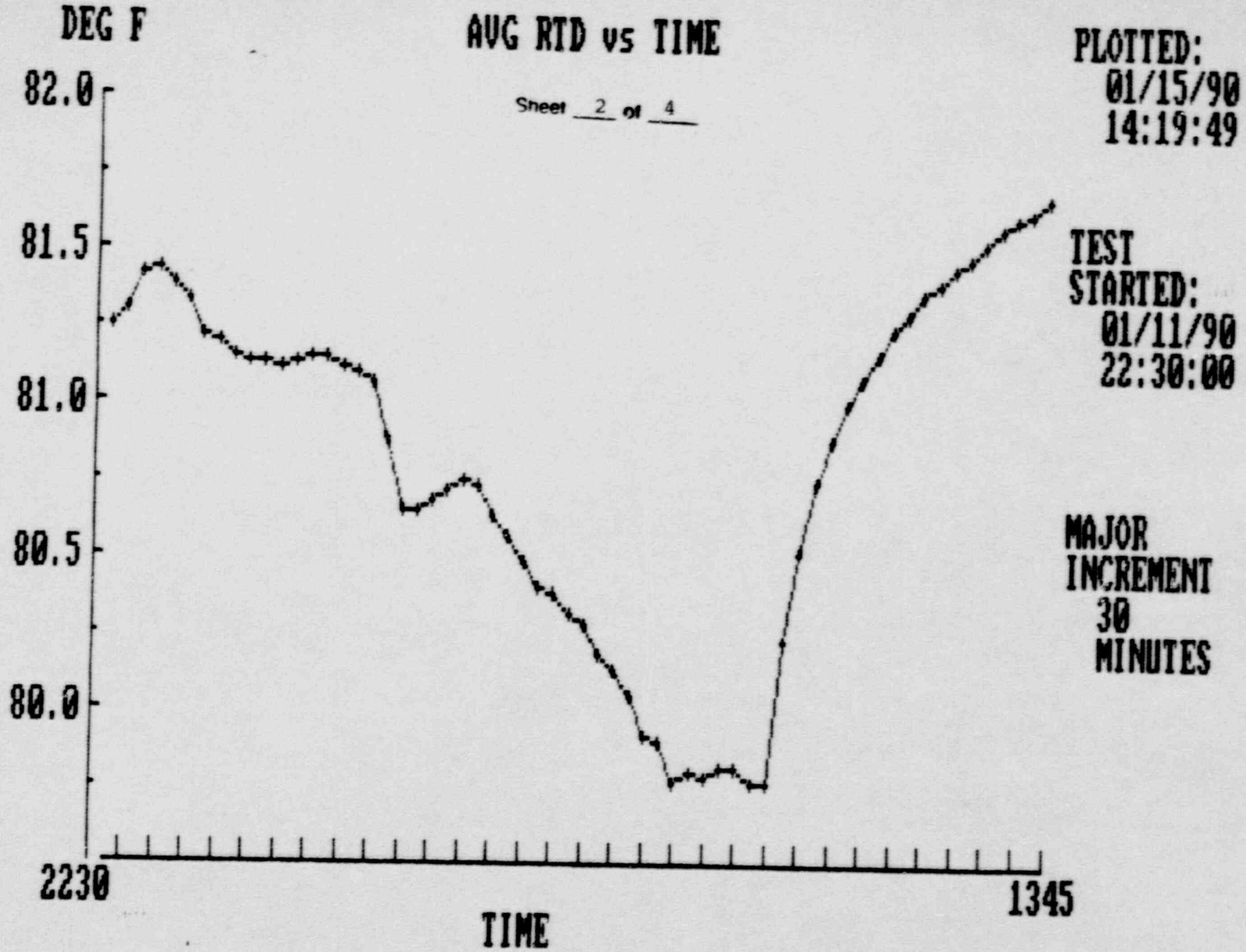
0.2
0.0
-0.2
1415
1315



Test was terminated at 0800. Cooling water to the Reactor Building was secured at 0915. Test data is presented to show the dramatic effect of securing cooling water. Third test began at 1400, Jan., 12, 1990.


DEG F
AVG RTD vs TIME
Sheet 2 of 4
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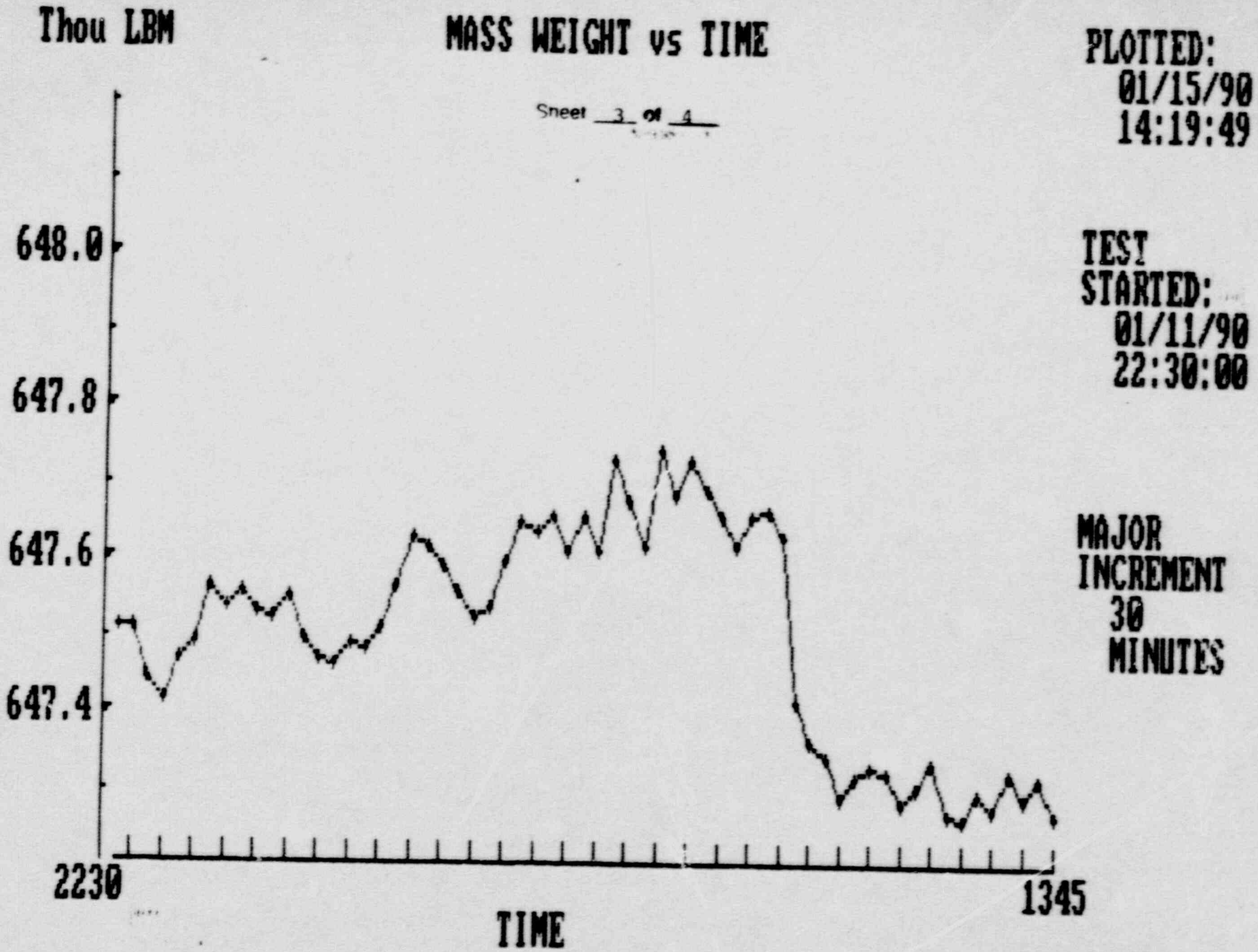
82.0
81.5
81.0
80.5
80.0
TEST
STARTED:
01/11/90
22:30:00
MAJOR
INCREMENT
30
MINUTES
2230
1345
TIME


Thou LBM
MASS WEIGHT vs TIME
Sheet 3 of 4
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INCREMENT
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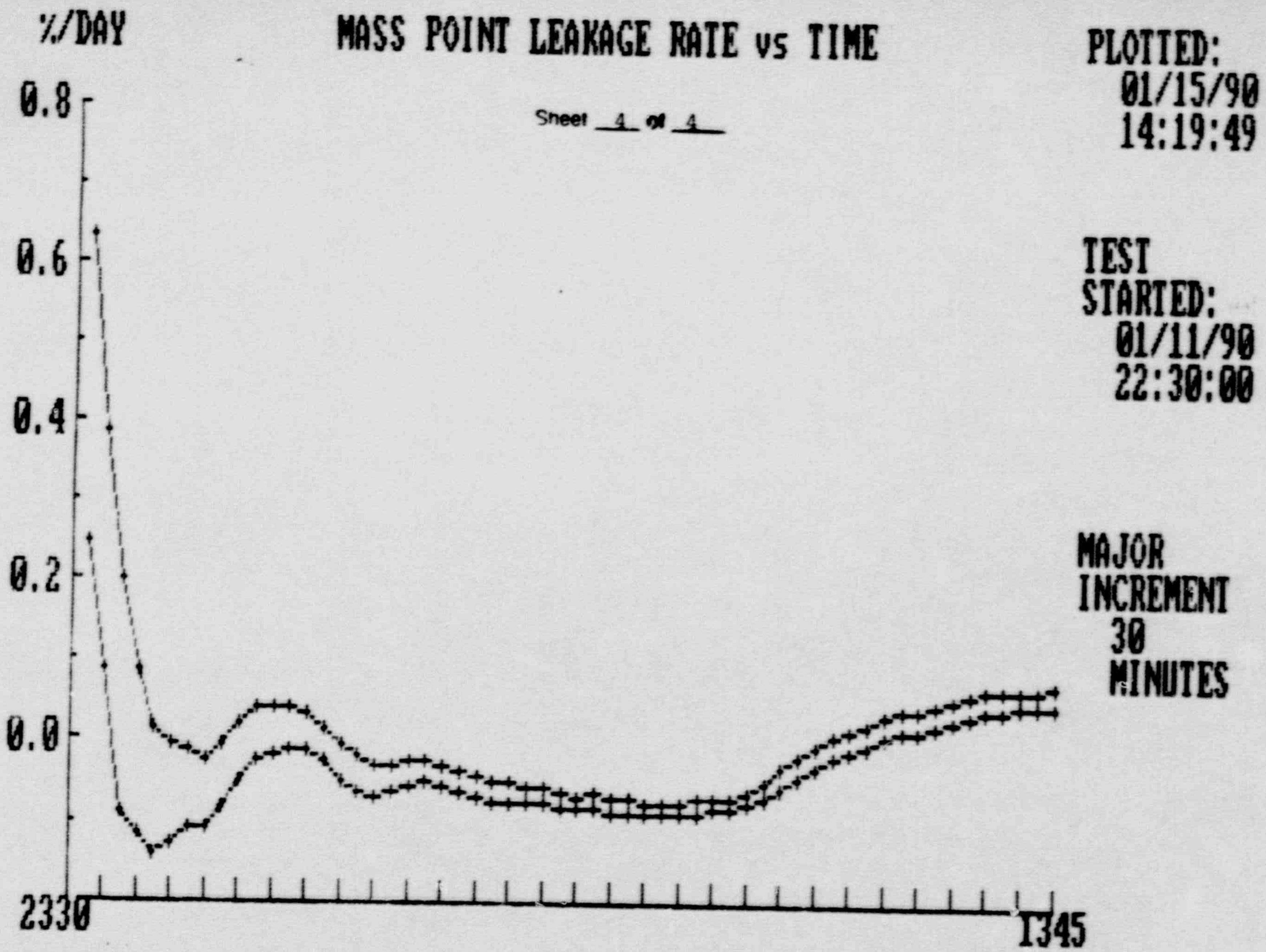
MINUTES
2230
1345
TIME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MASS POINT LEAKAGE RATE vs TIME
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0.4
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0.0
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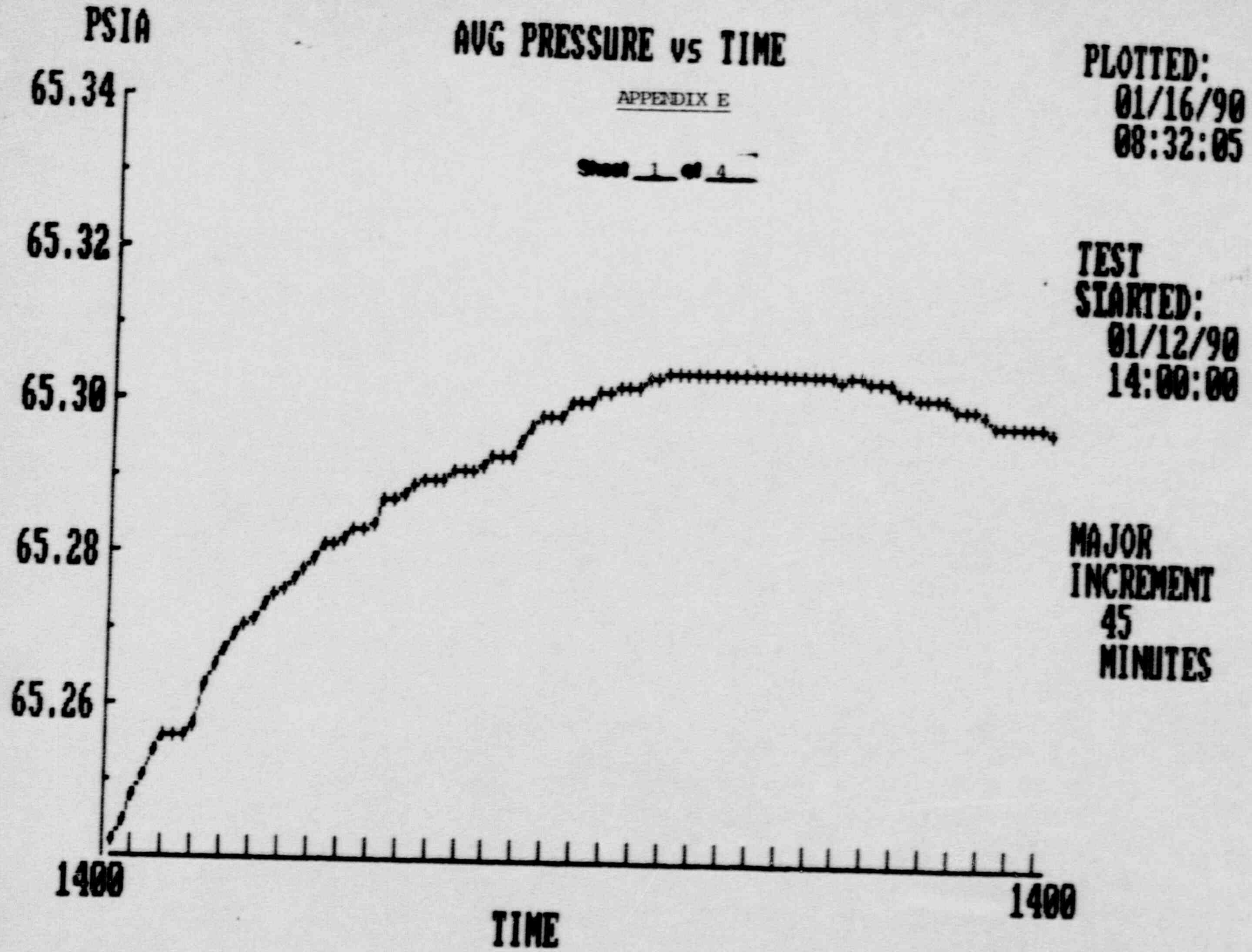
INCREMENT
30
MINUTES


PSIA
AVG PRESSURE vs TIME
APPENDIX E
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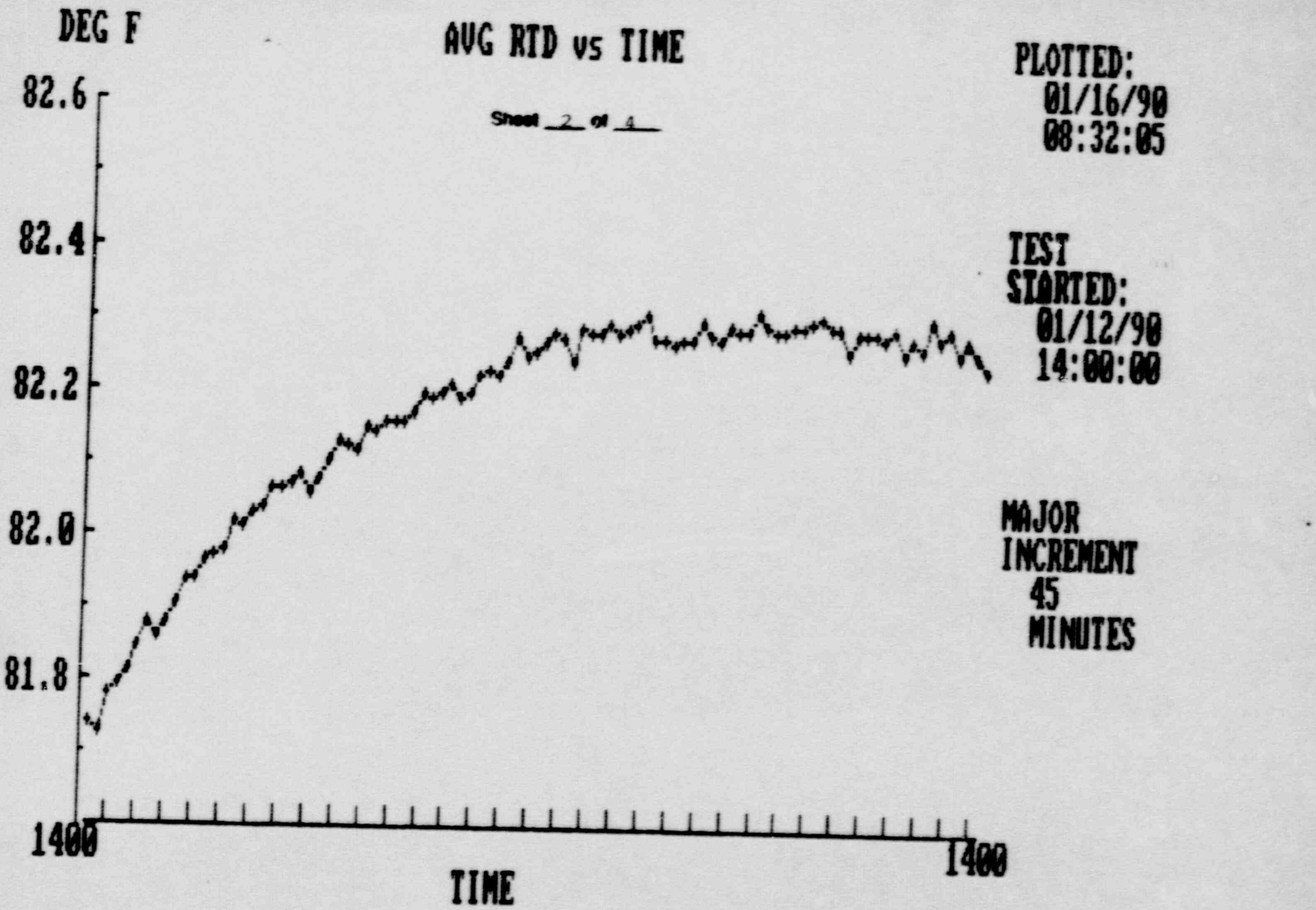
MINUTES
1400
1400
TIME


DEG F
AVG RTD vs TIME
Sheet 2 of 4
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82.4
82.2
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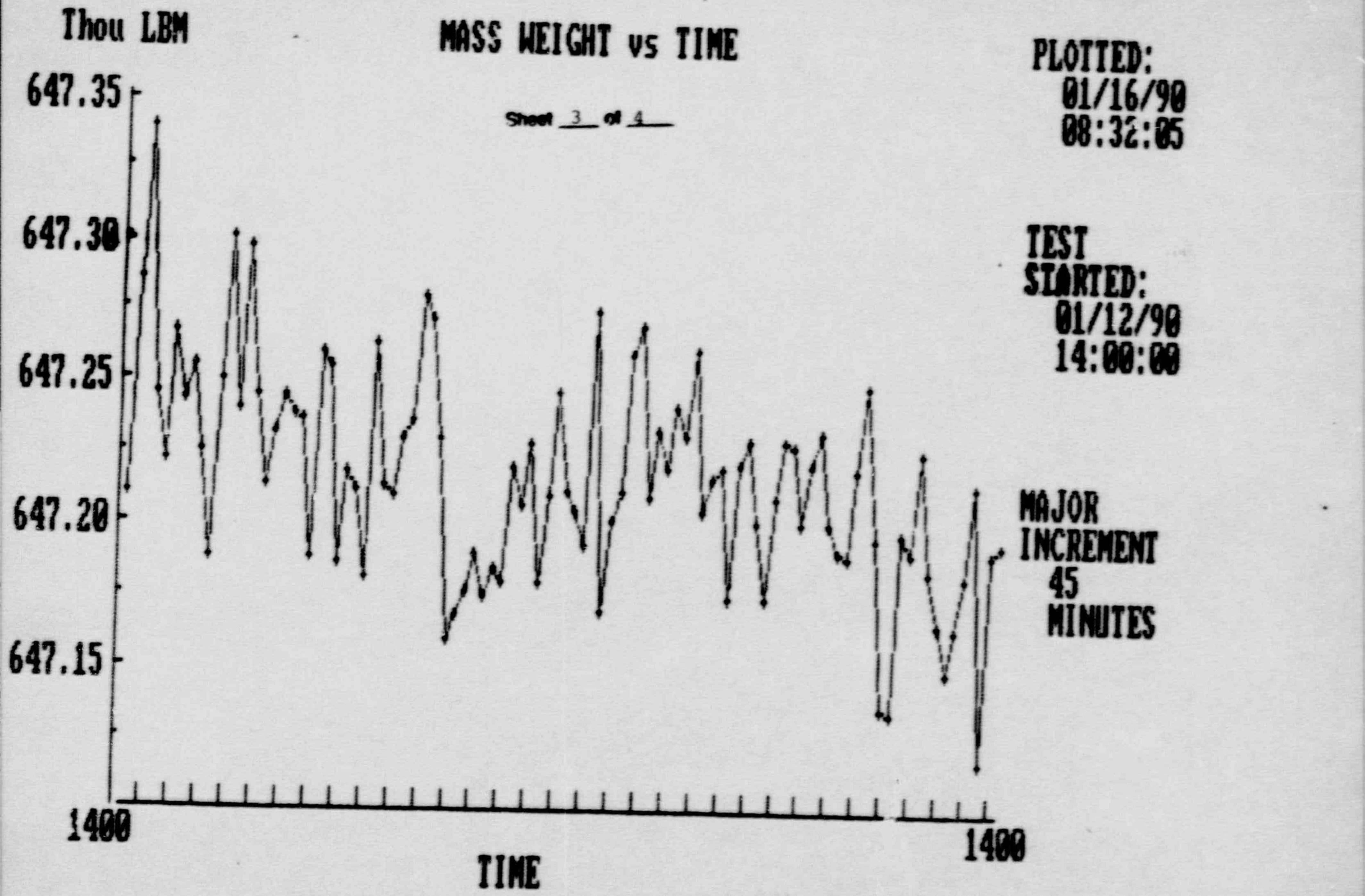
MINUTES
1400
1400
TIME


Thou LBM
MASS WEIGHT vs TIME
Sheet 3 of 4
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STARTED:
01/12/90
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INCREMENT
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MINUTES
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647.25
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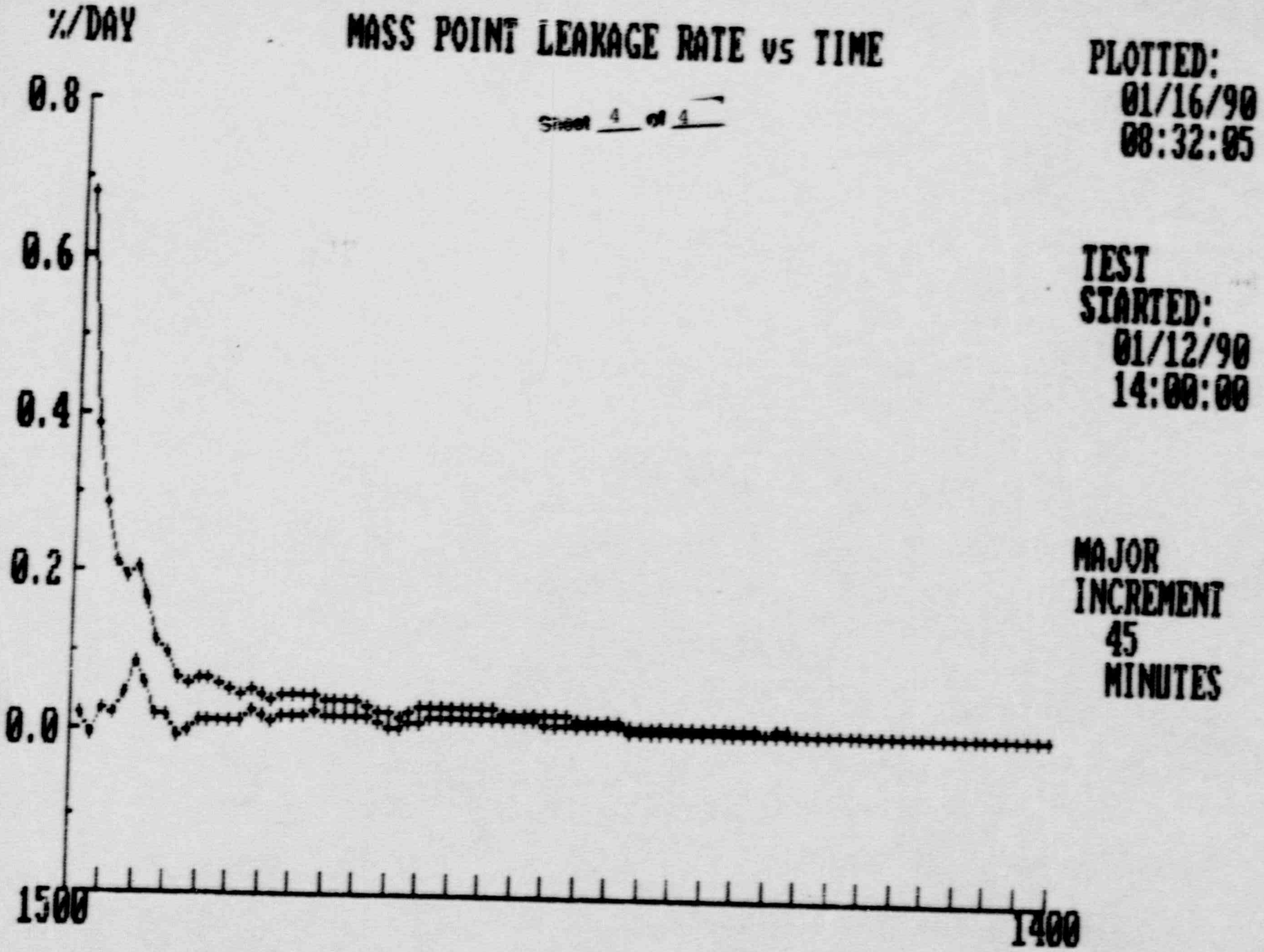
647.15
1400
1400
TIME


%/DAY
MASS POINT LEAKAGE RATE vs TIME
Sheet 4 of 4
0.8
0.6
0.4
0.2
0.0
1500
1400
PLOTTED:
01/16/90
08:32:05
TEST
STARTED:
01/12/90
14:00:00
MAJOR
INCREMENT
45
MINUTES

APPENDIX F

THREE MILE ISLAND UNIT 1

1988 REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT
(Includes 7R Refueling Outage and
6R Operating Cycle Test Data)

SP 1303-11.18

INDEX - 1988 REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

APPENDF

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

1988 REFUELING FREQUENCY

1. PURPOSE

1.1 To provide analysis to the Nuclear Regulatory Commission on the Eleventh Periodic Type B and Type C leakage tests performed on the Three Mile Island, Unit 1, Reactor Building.

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

A majority of the local leak rate testing was performed when the plant was shutdown for the 7R Refueling Outage. Testing began on June 27, 1988 and was completed August 8, 1988. This also includes 1987 test data for access hatches, purge valves, penetration pressurization system and other CIV's not previously reported in GPUN's 6R ILRT.

2. SUMMARY OF WORK ACCOMPLISHED

2.1 Valve Testing/Repairs/Modifications

Appendix J, Type B and C leak tests were performed on the components as listed in TMI, Unit 1, FSAR, Update Number 8, Table 5.7-2 and 5.7-3, respectively (Technical Specification Amendment 151 dated August 31, 1989 removed valve list from Technical Specification Section 4.4.1.2.1). In addition, the following components were leak tested though not yet listed in the FSAR.

1. HM-V1A/B, 2A/B, 3A/B, 4A/B

2. NI-V26.

3. IR-V1/10/49 (As Found data only, modified piping configuration).

4. PP-V101/102/133/134 (As Found data only, check valves converted to normally closed globe valves).

Repairs or modifications were initiated on the following components due to higher than desirable leakage during 7R or during previous outages.

1. CA-V2: Upgraded seat design. Two piece rather than solid wedge.

2. CA-V5A: Upgraded seat design. Two piece rather than solid wedge.

3. CA-V5B: Upgraded seat design. Two piece rather than solid wedge.

4. CA-V189: Disassembled and cleaned

5. PP-V101: Check valve upgraded to normally closed globe valve, renumbered to PP-V210.

6. PP-V102: Check valve upgraded to globe valve, renumbered to PP-V212.

7. PP-V133: Check valve upgraded to globe valve, renumbered to PP-V213.

8. PP-V134: Check valve upgraded to globe valve, renumbered to PP-V211.

9. WDL-V304: Packing Leakage.

2.2 <u>Access Hatch Testing/Repairs</u>

2.2.1 Access Hatch Door Seals, SP 1303-11.25 (Reference 8.6)

Access Hatch Door seal leak tests were performed as required by Technical Specification 4.4.1.2.5.

2.2.2 Overall Hatch Test SP 1303-11.18 (Reference 8.2)

Semi-annual integrated type leak tests were performed as required by Technical Specification 4.4.1.2.5.

2.3 <u>Penetration Pressurization, SP 1303-11.24</u> (Reference 8.5)

Quarterly readings were recorded from the flow rotameters which supply air pressure or nitrogen pressure to Reactor Building mechanical and electrical penetrations as was required by Technical Specification 4.4.1.2.5.e.

APPENDF

3. METHODS OF TESTING

3.1 Valve Test Methods

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

3.1.1 Use air or nitrogen to establish a pressure differential across the valve greater than P_a (50.6 psig - calculated accident pressure). 55 psig nitrogen was normally used.

3.1.2 Assure that the pressure is exerted in the accident test direction unless it can be demonstrated that pressurizing in the non-accident direction provides equal or conservative leak rate data. Butterfly valves AH-V1B/1C, and globe valves WDG-V4, DH-V64, SA-V3, and IA-V20 were tested in the reverse direction.

3.1.3 Assure that the test volume is drained of liquid so that air or nitrogen test pressure is against valve seats.

3.1.4 Assure that the test verifies valve packing integrity in those cases where the packing would be a Reactor Building leakage boundary.

3.1.5 Assure adequate time period for stabilization of test conditions.

3.1.6 Assure test equipment is calibrated and used in a manner consistent with the data accuracy desired (weekly meter standardization was performed during the test program to verify meters accurate within $\pm$ 4% full scale (Reference 8.1).

3.1.7 Assure valves to be tested are closed by the normal method prior to testing.

3.1.8 Document As-Found conditions (prior to adjustments/repairs) and As-Left conditions.

3.1.9 Record test instrument scale readings prior to doing any data corrections.

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

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

APPENDF

3.2 Access Hatch Test Methods

3.2.1 Access Hatch Seal Leak Tests-Method

Access Hatch Door seal leak tests were performed in accordance with procedure SP 1303-11.25 (Reference 8.6). This procedure provides detailed guidance on the test equipment and methods to be used.

The Access Hatch Door seal tests are performed by pressurizing the interspace between the double seals on each Access Hatch with metered air at the manufacturers recommended test pressure of 10 psig. After stabilization, the air rotameter indicates the rate of air input required to maintain the test pressure.

3.2.2 Overall Access Hatch Leak Test - Semi-annual overall hatch leak testing was performed in accordance with procedure SP 1303-11.18, Reactor Building Local Leak Rate Testing. This procedure provides detailed guidance on the test equipment and methods to be used. The overall integrated leak test verified the integrity of all of the following barriers:

1. Hatch shell/welds
2. Rubber door seals
3. Teflon operating shaft packing
4. Bulkhead electrical penetrations
5. Penetration pressurization check valves
6. Emergency air flange and associated "O" rings on outer bulkhead
7. Bulkhead equalizing ball valves and associated mounting flanges/"O" rings

The overall leak test was performed by pressurizing the hatch to greater than calculated accident pressure and observing the rate of pressure drop on a high accuracy (Heise) pressure gage.

Pressure corrections were made by reference to a barometer. Minimum test duration was 4 hours after a 1 hour stabilization period.

APPENDF

3.3 Penetration Pressurization - Method

Quarterly readings were taken on the flow rotameters which are permanently installed in the Penetration Pressurization System. These readings represent the air/nitrogen makeup rate required to maintain approximately 60 psig in mechanical penetrations and 30 psig in electrical penetrations. High meter readings have occasionally occurred and were attributed to leaks in the compression fittings in the penetration pressurization system or to malfunctioning (stuck) rotameters which were promptly corrected even though the leakage had no bearing on the integrity of the containment penetration. Testing was performed in accordance with SP 1303-11.24 (Reference 8.5).

APPENDF

4. TEST EQUIPMENT USED

4.1 Valve Test Equipment (See Attachment 1)

a. Rotameters - Sets of 3

Mfgr. - Brooks Inst. Co.
Model - 1114 Full View

Ranges:

Float Mat'l.	Tube No.	Range
Pyrex	R-2-15D	8-1,120 SCCM
Sapphire	R-2-15C	100-12,200 SCCM
Carboloy	R-6-15B	1,000-142,000 SCCM

Accuracy ± 2% full scale industrial accuracy

b. Temperature Indicators (as follows or similar)

Mfgr. - Ashcroft
Model - EH or AH / 3" or 5" Dial
Range - 30° to 130°F
Accuracy - ± 2°F

c. Pressure Indicators (as follows or similar)

Mfgr. - Ashcroft
Model - 1279 - 4-1/2" Dial
Range - 0 to 60 or 0 to 100 psig
Accuracy - ± 2 psig

d. Pressure Regulator (as follows or similar)

Mfgr. - Union Carbide Corp.
Model - UPG 3-75-580
Range - 0 to 100 psi output / 0 to 3000 psi input

e. Calibration Rotameters (Set of 2)

Mfgr. - Brooks Inst. Co.
Models - 1110-05K2B1Z49, 1110-08K2B1Z06
Ranges - 20 to 16,000 SCCM, 3,600 to 234,000 SCCM
Repeatability - ± 1/4% of instantaneous
Accuracy - ± 1% instantaneous

APPENDF

f. Flow rate Calibrator
Mfgr. - Brooks Inst. Co.
Model - 1056A
Range 0 to 2,400 SCCM
Accuracy - ± 0.2% of indicated volume

4.2 Access Hatch Test Equipment

a. Precision Pressure Gage (as follows or similar)

Mfgr. - [illegible]
Model - [illegible]
Range - 0 to [illegible]0 psig
Resolution - 0.25 psig
Accuracy - 0.1% of instrument span

b. Barometer (as follows or similar)

Mfgr. - Pennwalt
Model - FA185260A
Range - 10.8 to 15.5 psia
Resolution - 0.005 psia
Accuracy - 0.1% of instrument span

4.3 Penetration Pressurization Test Equipment

a. Flow Rotameters - (Permanent System Equipment)

Mfgr. - Brooks Inst. Co.
Model - 1114
Range - 0 to 10 SCFH at 60 psia air
Accuracy - ± 2% industrial accuracy

APPENDF

5. SUMMARY AND INTERPRETATION OF DATA

5.1 Valve Test Results

As-Found/As-Left Leakage to this date - Also see tabulation of individual results in Attachment #2.

	Total Leakage	Tech. Spec. Limit	% Tech. Spec. Limit
As-Found MAXPATH	27,485 SCCM	104,846 SCCM	<26.3%
As-Left MAXPATH	15,889 SCCM	104,846 SCCM	<15.2%

NOTE: The total shown above is "Maximum Pathway" leakage. Only the highest valve leakage on each penetration is counted. This number is labeled as "MAXPATH" on the tabulation of results in Attachment 2.

EXAMPLE: Penetration XYZ has three containment isolation valves inside the Reactor Building in parallel and one outside. The leakage from the three inside totals 500 SCCM and the outside valve is 1000 SCCM. The penetration "MAXPATH" leakage is counted as 1,000 SCCM not 1,500 SCCM. "Maxpath" leakage assumes the higher leaking Reactor Building penetration pressure boundary is the only boundary left intact during a DBA.

1987 REACTOR BUILDING PURGE VALVE DATA

FREQUENCY	DATE	AH-V1A/1B LEAKAGE RESULTS (SCCM)	DATE	AH-V1C/1D LEAKAGE RESULTS (SCCM)
FIRST QUARTER	03-16-87	546	03-16-87	2750
SECOND QUARTER	06-25-87	410	06-24-87	1580
THIRD QUARTER	09-14-87	429	09-13-87	1404
FOURTH QUARTER	12-14-87 12-15-87	*AS FOUND 68020 AS LEFT 234	12-14-87 12-14-87	**AS FOUND 7098 AS LEFT 2184

* Repaired seat leakage on AH-V1B.
**Packing leak repaired on AH-V1D.

Other leak rate test data not submitted during previous 1987 Reactor Building Local Leak Rate Testing Report as follows:

VALVE	AS-FOUND/DATE (SCCM)	AS-LEFT/DATE (SCCM)
CA-V5A	----	243/03-10-87
CA-V5B	----	250/03-10-87
MU-V2A	----	65/03-07-87
MU-V3	66/03-06-87	66/03-06-87
PP-V133/134	3120/02-25-87	1755/03-10-87
PP-V101/102	4134/02-24-87	6142/03-10-87
SA-V2/3	88/03-17-87	66/06-24-87

APPENDF

5.2 Access Hatch Test Results

5.2.1 Overall semi-annual access hatch leakage test results in accordance with SP 1303-11.18 (Reference 8.2):

COMPONENT DESCRIPTION	FIRST HALF TEST		SECOND HALF TEST	
	DATE	LEAKAGE RESULTS	DATE	LEAKAGE RESULTS
PERSONNEL ACCESS HATCH	05-29-88	594	11-24-88	1057
EQUIPMENT ACCESS HATCH	05-28-88	1708	11-26-88	3862
PERSONNEL ACCESS HATCH	05-23-87	660	11-26-87	1453
EQUIPMENT ACCESS HATCH	05-17-87	3639	11-27-87	*AS-FOUND 5941 AS-LEFT 4159

The leakage from these tests were within the established target criteria of less than 2500 SCCM with the exception of the Equipment Access Hatch. However, Plant Engineering accepted this leakage since total LLRT leakage is substantially less than 104,846 SCCM as required by Technical Specifications.

*PP-V113 fitting and valve stem leakage repaired.

5.2.2 Door Seal Leakage Test in accordance with SP 1303-11.25 (Reference 8.6)

The Personnel and Equipment Hatch Door seals had been satisfactory leak tested though-out 1987 and 1988 resulting in leakage from each door seal to be less than 3 SCFH.

5.3 Penetration Pressurization System Quarterly Leakage Test Results In accordance with SP 1303-11.24 (Reference 8.5):

1987 LEAKAGE RESULTS

FREQUENCY	DATE	MECHANICAL LEAKAGE (SCFH)	ELECTRICAL LEAKAGE (SCFH)
FIRST QUARTER	03-14-87	8.5	0
SECOND QUARTER	06-12-87	37	0
THIRD QUARTER	09-10-87	10	0
FOURTH QUARTER	12-12-87	10.5	0

APPENDF

1988 LEAKAGE RESULTS

FREQUENCY	DATE	MECHANICAL LEAKAGE (SCFH)	ELECTRICAL LEAKAGE (SCFH)
FIRST QUARTER	03-13-88	14	0
SECOND QUARTER	06-12-88	0	0
THIRD QUARTER	09-11-88	15	0.4
FOURTH QUARTER	12-12-88	27	0

There is no Technical Specification limit on Penetration Pressurization System Leakage. The system leakage was maintained as low as practical.

APPENDF

6. ERROR ANALYSIS

6.1 Valve Testing Errors (For purge valves, see Section 6.2)

The flow meters used in the field have normal industrial accuracies of ± 2% full scale in the 10-100% (15-150 mm) scale range. Prior to use, mm versus sccm graphs were developed for the meters by 10 point calibrations using high accuracy (± 1% instantaneous) lab rotameters. During the leak test program, weekly 3 point standardizations were performed on the field rotameters to verify continued accuracy. The acceptance criteria for these standardizations was a variance of no more than 4% from the calibration graphs. If meters were repaired or the 3 point standardization exceeded the inaccuracy limit, a new 10 point calibration was performed. Scale readings on the leak rate procedure (SP 1303-11.18) data sheets were evaluated and corrected using the methods in Attachment 3. Conservative bias was introduced into the results by assuming 15 mm (10% of scale) as the minimum scale. Approximately half of the test results actually showed a minimum scale reading. More involved error corrections were not considered meaningful based on the acceptable total leakage As-Found and the low total leakage As-Left.

6.2 Access Hatch and Purge Valve Testing Errors

The measured pressure drops were corrected by adding the minimum scale increment of the gage used for both the Heise gage and the barometer. This conservatively corrected for the resolution and repeatability errors. Gages used were recently calibrated. A minimum one hour temperature/pressure stabilization period was used prior to each pressure drop test. The access hatches and purge valves are not instrumented to allow temperature corrections.

6.3 Penetration Pressurization Testing Errors

These test results were used for information only and do not count toward the total leakage limit for Technical Specification conformance. The meters, installed permanently in the system, have ± 2% full scale industrial accuracy.

7.0 LESSONS LEARNED/IMPROVEMENTS/DEGRADATION

7.1 The leak tightness of TMI-1 Reactor Building penetrations was found to be excellent. As-Found MAXPATH leakage was less than one-third the Technical Specification limit of 104,846 SCCM. The As-Left MAXPATH leakage was about one-fifth the Technical Specification limit. This improvement was due to repairs and modifications on containment isolation valves as summarized in Section 2.1.

APPENDF

7.2 Due to the tack welding installation of seat rings in gate valves CA-V2/5A/5B, the seat rings were slightly distorted. During the 8R Outage the valves will be re-leak tested and the seat rings may be pinned into place on a case-by-case basis.

7.3 CA-V189 was disassembled, cleaned, and seat ring measurements were obtained.

7.4 PP-V101/102/133/134 check valves were successfully converted to normally closed globe valves. Also, test connections were installed outboard of the PP valves. These changes significantly improved the leak tightness of the PP system containment isolation valves. It also made testing much more convenient for AH-V1A/1B/1C/1D.

7.5 Retested WDL-V303 after a packing leak and supply side fitting was tightened.

7.6 Retested WDL-V304 after packing leak was repaired.

7.7 CF-V2B, MU-V2B, and WDL-V303 were retested due to MOVATS adjustments.

8. REFERENCES

8.1 MP 1430-Y-22, Standardization of Flow Rotameters

8.2 SP 1303-11.18, Reactor Building Local Leak Rate Testing

8.3 Three Mile Island, Unit 1, Technical Specification 4.4.1

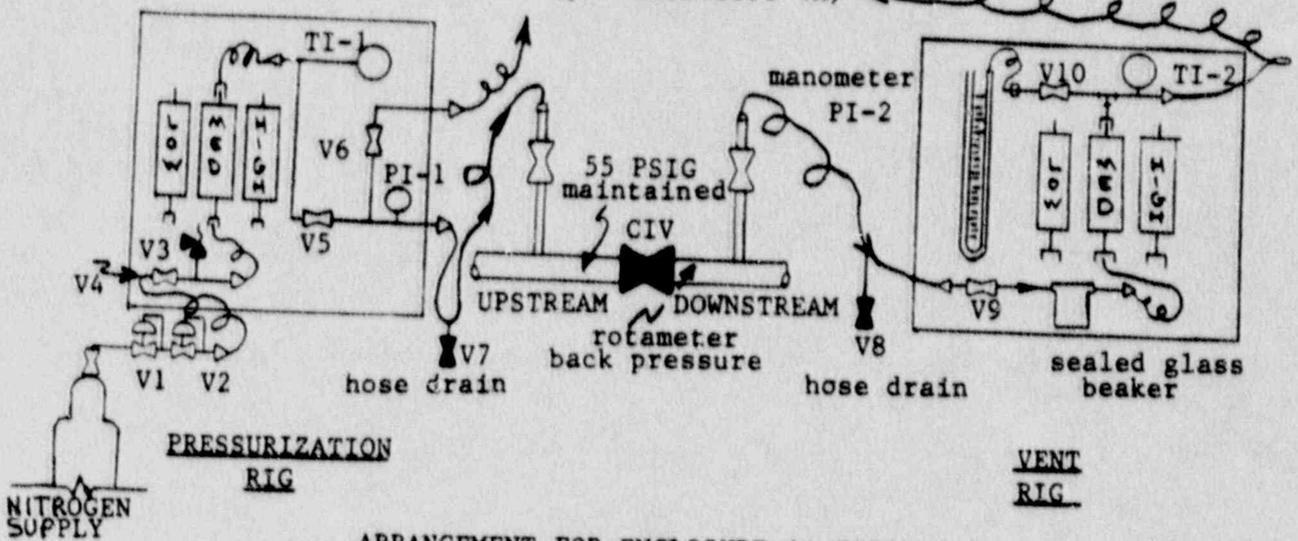
8.4 TMI Surveillance File (Records stored in CARIRS, Data Base AA60 REC.TYPE 018-12)

8.5 SP 1303-11.24, Reactor Building Local Leakage Penetration Pressurization

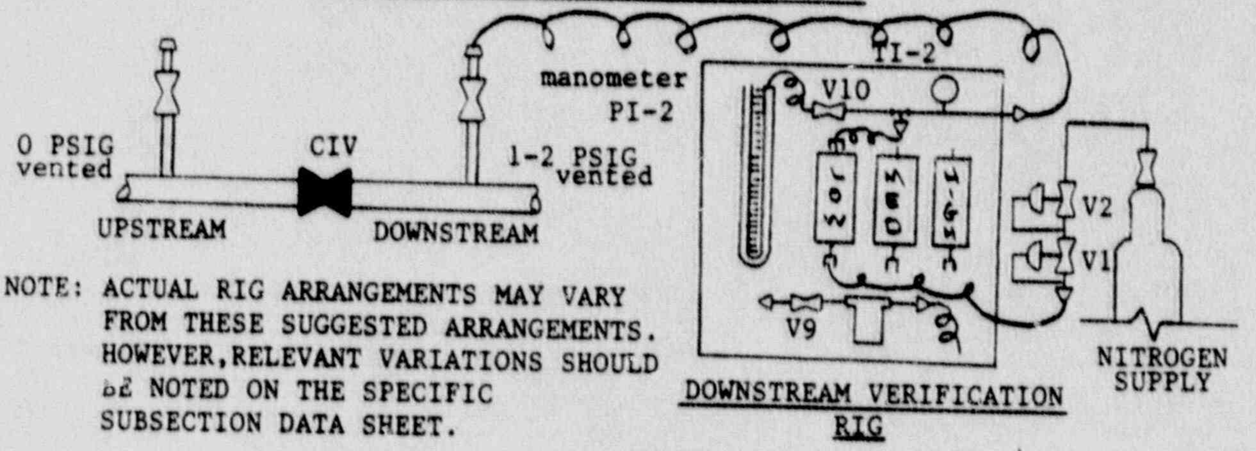
8.6 SP 1303-11.25, Reactor Building Local Leakage Access Hatch Door Seals

APPENDF

Attachment 1



ARRANGEMENT FOR ENCLOSURE 4A TEST



NOTE: ACTUAL RIG ARRANGEMENTS MAY VARY FROM THESE SUGGESTED ARRANGEMENTS. HOWEVER, RELEVANT VARIATIONS SHOULD BE NOTED ON THE SPECIFIC SUBSECTION DATA SHEET.

ARRANGEMENT FOR DOWNSTREAM VERIFICATION

EQUIPMENT DESCRIPTIONS

VALVES: V1/V2-Regulator, V3-65# Relief, V4/V5/V7/V8/V9/V10-Ball, V6- toggle/globe

PRESSURE REGULATORS: PI-1,0-60 PSIG usage range with 2 PSI increments; PI-2, manometer with 36 inch long scale

TEMPERATURE INDICATORS: TI-1/TI-2, 0-200 F usage range with 2°F increments

ROTAMETERS:

ROTAMETER	TUBE #	FLOAT	AT 0 PSIG (SCCM)	AT 55 PSIG(SCCM)
LOW	R-2-15-D	BLACK GLASS	0-380	45-1120
MED	R-2-15-C	SYN SAPHIRE	250-5300	800-12500
HIGH	R-6-15-B	CARBOLOY	4750-62000	12000-144000

Attachment 2

ITEM	TAG	ASFOUND	ASLEFT	ASLTDATE	COMMENTS
****	********	********	********	********	********
1	AH-V1A/B	371	371	3/15/88	LOW
2	2ND	429	429	6/11/88	LOW
3	3RD	468	468	8/4/88	LOW
4	4TH	1794	1794	12/14/88	OK
5	AH-V1C/D	2691	2691	3/14/88	OK
6	2ND	1404	1404	6/12/88	OK
7	3RD	6474	6474	8/6/88	HIGH
8	4TH	6767	6767	10/30/88	HIGH
9					
10					
11	CA-V1	74	74	6/27/88	OK
12	CA-V2	1866	1729	7/31/88	HIGH
13	CA-V3	74	74	6/27/88	OK
14	CA-V4A	74	74	6/28/88	OK
15	CA-V4B	74	74	6/28/88	OK
16	CA-V5A	551	256	7/30/88	A/D MOD
17	CA-V5B	2150	38	7/30/88	A/D MOD
18	CA-V13	74	74	6/27/88	OK
19	CA-V189	964	561	7/13/88	OK
20	CA-V192	135	135	7/11/88	OK
21					
22					
23	CF-V2A	75	75	6/30/88	OK
24	CF-V2B	75	71	7/29/88	OK
25	CF-V12A	287	287	7/ 1/88	OK
26	CF-V12B	87	87	7/ 1/88	OK
27	CF-V19A	843	843	7/ 1/88	OK
28	CF-V19B	74	74	7/ 1/88	OK
29	CF-V20A	192	192	6/30/88	OK
30	CF-V20B	74	74	6/30/88	OK
31	CM-V1	87	87	7/ 7/88	OK
32	CM-V2	87	87	7/ 7/88	OK
33	CM-V3	87	87	7/ 7/88	OK
34	CM-V4	87	87	7/ 7/88	OK
35	DH-V64	75	75	7/19/88	OK
36	DH=V69	84	120	7/25/88	NEW VALV
37					
38					
39	FTTEAST	183	45	8/7/88	OK
40	FTTWEST	85	440	8/5/88	OK
41	HM=V1A	87	87	7/ 6/88	OK
42	HM=V1B	87	87	7/ 6/88	OK
43	HM-V2A	87	87	7/ 6/88	OK
44	HM-V2B	87	87	7/ 6/88	OK
45	HM-V3A	87	87	7/ 6/88	OK
46	HM-V3B	87	87	7/ 6/88	OK
47	HM-V4A	87	87	7/ 6/88	OK
48	HM-V4B	87	87	7/ 6/88	OK
49	HP-V1	75	75	7/ 7/88	LOW
50	HP-V6	95	95	7/ 7/88	LOW
51	HR-V2A/B	65	74	8/5/88	OK
52	HR-V4A/B	65	230	8/5/88	OK
53	HRV22A/B	65	65	6/21/88	OK
54	HR-V23A	65	65	6/21/88	OK
55	HR-V23B	65	65	6/21/88	OK
56					
57					
58	IA-V6/20	45	45	8/ 2/88	OK

59	IC-V2	87	87	7/ 4/88	LOW
60	IC-V3	87	87	7/ 4/88	OK
61	IC-V4	87	87	7/ 3/88	OK
62	IC-V6	87	87	7/ 4/88	OK
63	IC-V16	476	476	7/ 4/88	OK
64	IC-V18	87	87	7/ 3/88	LOW
65	LR-V1/10	66	.01	6/20/88	ADD FLG
66	LR-V4	45	45	6/21/88	OK
67	LR-V5	157	45	6/21/88	OK
68	LR-V6	45	45	6/21/88	OK
69	LR-V49	66	.01	6/20/88	ADD FLG
70					
71					
72	MU-V2A	74	74	6/24/88	OK
73	MU-V2B	74	71	8/1/88	OK
74	MU-V3	74	71	8/1/88	OK
75	MU-V18	268	268	7/ 1/88	OK
76	MU-V20	286	286	6/23/88	OK
77	MU-V25	306	306	6/26/88	OK
78	MU-V26	45	45	6/26/88	OK
79	MU-V116	266	266	6/23/88	OK
80					
81	NI-V26	74	74	8/ 3/88	OK
82	NI-V27	74	74	8/ 3/88	OK
83	NS-V4	74	74	6/29/88	LOW
84	NS-V11	2859	2859	6/29/88	HIGH
85	NS-V15	30	30	6/29/88	LOW
86	NS-V35	190	190	6/29/88	LOW
87					
88	PENET417	.01	75	8/9/88	OK
89	PENET104	45	45	8/6/88	OK
90	PENET105	45	45	8/ 3/88	OK
91	PENET106	45	45	8/3/88	OK
92	PENET210	45	45	8/6/88	OK
93	PENET211	45	75	8/9/88	OK
94	PENET241	84	84	7/ 5/88	OK
95	PP101/02	3120	.01	6/11/88	REMOVED
96	PP133/34	4291	.01	7/27/88	REMOVED
97	PP-V210	.01	17	8/4/88	NEW VALV
98	PP-V211	.01	19	7/31/88	NEW VALV
99	PP-V212	.01	17	8/4/88	NEW VALV
100	PP-V213	.01	19	7/31/88	NEW VALV
101	RB-V2A	74	74	7/2/88	LOW
102	RB-V7	74	74	7/2/88	LOW
103	SA-V2/3	75	75	8/9/88	OK
104	SF-V23	214	214	7/8/88	OK
105	WDG-V3/4	74	74	7/8/88	OK
106	WDL-V303	999	75	7/26/88	HIGH
107	WDL-V304	1806	322	7/27/88	HIGH
108	WDL-V534	74	74	7/9/88	LOW
109	WDL-V535	74	74	7/9/88	LOW
110					
111					
112	EQPFLG	208	208	7/5/88	OK
113	PERACCES	594	594	5/29/88	LOW
114	2ND	1057	1057	11/24/88	OK
115	EMEACCES	1708	1708	5/28/88	OK
116	2ND	3862	3862	11/26/88	OK
117					
118					
119	MINPATH	17182	8939		LOW
120	MAXPATH	27485	15889		LOW

GPU Nuclear	TMI-1 Surveillance Procedure	Number 1303-11.18
Title RB Local Leak Rate Testing		Revision No. 44

Attachment 3

Leakage Results Evaluation

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

FOR USE OF SUPPLY ROTAMETER DATA: Procedure:	FOR USE OF VENT ROTAMETER DATA: Procedure
a. Record supply meter reading in (1) below*. Also identify the meter used by tube No. in (8) below and the metering pressure in (9). b. Convert meter units to SCCM units using latest lab meter calibrate curve. Enter in (3) below. c. Correct results for temperature Enter supply temperature in (4) below. Calculate and enter in (7) below. * If meter scale reading was less than 15mm (minimum scale) use 15mm in calculations.	a. Record vent meter reading in (1) below*. b. Record downstream verification meter reading in (2) below. Also identify the respective meters used in (8) below and the metering pres'ures in (9). c. Convert meter units to SCCM units using latest lab meter calibration curve. Enter in (3) below. d. Correct results for temperature. Enter vent temperature (°F) in (4) below. then Calculate and enter in (5) below. e. If measurements of any other significant leakage paths (fluid block check valve, packing) are being claimed enter corrected flow (SCCM) in (6) below.

(mm) (SCCM)

(_____ + _____) convert (_____ + _____) X $\dfrac{530}{\_\_\_\_ + 460}$ = _____SCCM
(1) (2) (3) (4) (5)

\+ _____SCCM
(6)

_____ _____
(8) (Identify meters used)
at _____ _____
(9) (Meter Pressures)

= CIV Leakage _____SCCM
(7)

1081c, 1109:

APPENDF

LOCAL LEAK RATE TEST RESULTS

THREE MILE ISLAND UNIT 1 REACTOR BUILDING

The following terminology abbreviations were used in the LLRT computer data sheets:

1) .1 - (Alone) or any other number other than zero in the first decimal place means test scheduled.

2) .01 - (Alone) means no data available for the year or that the test was delayed. (e.g. valve not installed yet or not in previous testing scope.)

3) .001 - (Or any number other than zero in the third decimal place) after a leak rate (i.e. 59500.001) means actual leak rate was greater than measured/recorded value.

4) AsFound - leak rate (SCCM) in the As-Found condition before any repairs or adjustments.

5) AsLeft - The leak rate (SCCM) after any adjustments/repairs.

6) Dates - Date of the last acceptable test results for the item.

7) Desc - Description of the valve or penetration.

8) Oper - Type of valve operator (actuator).

9) Notest - The Tech Spec scope did not require this valve to be tested during the respective year.

10) Novalve - This valve was installed during a later refueling outage.

11) Comments - Cognizant Engineer subjective comments about the results:

    A) Failed - Exceeded the plant established leakage rate limit from SP 1303-11.18, Enclosure 7, which made repair/adjustment necessary.
    B) High - Based on leak rate history of component. The leak rate is much greater than Target Criteria but is still less than the established maximum leak rate limit. Repair <u>might</u> not be worth the effort.
    C) Low (Excellent) - Based on leak rate history of component. The leak rate is much less than would be predicted by the normal method of assigning Target Criteria.
    D) OK - No problems with leakage.
    E) Other - E.G. newvalve, novalve, notest, repacked seatwork, stembent, etc. (self-explanatory). "A/D MOD" means Anchor Darling Split Seat Modification, "REMOVED" means valve was eliminated from plant design. "ADD FLG" means a newly installed flange is the containment isolation boundary.

12) Size - The nominal pipe size for the leakage barrier.

APPENDF

APPENDIX G

THREE MILE ISLAND UNIT 1

1990 REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT
(Includes 8R Refueling Outage Test Data
and Miscellaneous 1989 Test Data)

SP 1303-11.18

APPENDG

INDEX - 1990 REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

APPENDG

REACTOR BUILDING LOCAL LEAK RATE TESTING REPORT

1990 REFUELING FREQUENCY

1. PURPOSE

1.1 To provide analysis to the Nuclear Regulatory Commission on the Twelfth Periodic Type B and Type C leakage tests performed on the Three Mile Island, Unit 1, Reactor Building.

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

A majority of the local leak rate testing was performed when the plant was shutdown for the 8R Refueling Outage. Testing began on January 5, 1990 and was completed February 26, 1990.

2. SUMMARY OF WORK ACCOMPLISHED

2.1 Valve Testing/Repairs/Modifications

Appendix J, Type B and C leak tests were performed on the components as listed in TMI, Unit 1, FSAR, Update Number 8, Table 5.7-2 and 5.7-3, respectively. In addition the following components were leak tested though not yet listed in the FSAR (will be added during Update Number 9).

1. HM-V1A/B, 2A/B, 3A/B, 4A/B

2. NI-V26

Repairs or modifications were initiated on the following components due to higher than desirable leakage during 8R or during previous outages. Also listed are valves which were retested due to MOVATS testing of motor operators.

1. CA-V2: Upgrade seat design by pinning seat rings to valve body rather than welding.

2. CA-V5A: Upgraded seat design by pinning seat rings to valve body rather than welding.

3. NS-V11: Cleaned seats.

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4. IC-V16: Cleaned seats.

5. WDG-V3: Replaced stem and disc.

6. NS-V35: Cleaned seats, MOVATS testing/adjustments.

7. Fuel Transfer Tube Flanges were converted from Flexitallic type gaskets to rubber "O" rings with remachined flanges.

8. All Fluid Block System connections to containment isolation valves were disconnected.

9. MOVATS Testing:

   CA-V1
   CA-V3
   CA-V4A
   CA-V4B
   CF-V2A
   CF-V2B
   IC-V2
   NS-V4
   NS-V15
   NS-V35

2.2 Access Hatch Testing/Repairs

2.2.1 Access Hatch Door Seals, SP 1303-11.25 (Reference 8.6)

Access Hatch Door seal leak tests were performed as required by Technical Specification 4.4.1.2.5.

2.2.2 Overall Hatch Test SP 1303-11.18 (Reference 8.2)

Semi-annual integrated type leak tests were performed as required by Technical Specification 4.4.1.2.5.

2.3 Penetration Pressurization SP 1303-11.24 (Reference 8.5)

Quarterly readings were recorded from the flow rotameters which supply air pressure or nitrogen pressure to Reactor Building mechanical and electrical penetrations as was required by Technical Specification 4.4.1.2.5.e.

Technical Specification Change Request No. 191 dated June 13, 1989 deleted requirement to perform quarterly penetration pressurization rotameter readings. The change was made effective on August 31, 1989 per Technical Specification Amendment Number 151.

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3. METHODS OF TESTING

3.1 Valve Test Methods

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

3.1.1 Use air or nitrogen to establish a pressure differential across the valve greater than P_a (50.6 psig - calculated accident pressure). 55 psig nitrogen was normally used.

3.1.2 Assure that the pressure is exerted in the accident test direction unless it can be demonstrated that pressurizing in the non-accident direction provides equal or conservative leak rate data. Butterfly valves AH-V1B/1C, and globe valves WDG-V4, DH-V64, SA-V3, and IA-V20 were tested in the reverse direction.

3.1.3 Assure that the test volume is drained of liquid so that air or nitrogen test pressure is against valve seats.

3.1.4 Assure that the test verifies valve packing integrity in those cases where the packing would be a Reactor Building leakage boundary.

3.1.5 Assure adequate time period for stabilization of test conditions.

3.1.6 Assure test equipment is calibrated and used in a manner consistent with the data accuracy desired (weekly meter standardization was performed during the test program to verify meters accurate within ± 4% full scale (Reference 8.1).

3.1.7 Assure valves to be tested are closed by the normal method prior to testing.

3.1.8 Document As-Found conditions (prior to adjustments/repairs) and As-Left conditions.

3.1.9 Record test instrument <u>scale</u> readings prior to doing any data corrections.

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

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

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3.2 Access Hatch Test Methods

3.2.1 Access Hatch Seal Leak Tests-Method

Access Hatch Door seal leak tests were performed in accordance with SP 1303-11.25 (Reference 8.6). This procedure provides detailed guidance on the test equipment and methods to be used.

The Access Hatch Door seal tests are performed by pressurizing the interspace between the double seals on each Access Hatch with metered air at the manufacturers recommended test pressure of 10 psig. After stabilization, the air rotameter indicates the rate of air input required to maintain the test pressure.

3.2.2 Overall Access Hatch Leak Test - Semi-annual overall hatch leak testing was performed in accordance with SP 1303-11.18, Reactor Building Local Leak Rate Testing. This procedure provides detailed guidance on the test equipment and methods to be used. The overall integrated leak test verified the integrity of all of the following barriers:

1. Hatch shell/welds
2. Rubber door seals
3. Teflon operating shaft packing
4. Bulkhead electrical penetrations
5. Penetration pressurization check valves
6. Emergency air flange and associated "O" rings on outer bulkhead
7. Bulkhead equalizing ball valves and associated mounting flanges/"O" rings

The overall leak test was performed by pressurizing the hatch to greater than calculated accident pressure and observing the rate of pressure drop on a high accuracy (Heise) pressure gage.

Pressure corrections were made by reference to a barometer. Minimum test duration was 4 hours after a 1 hour stabilization period.

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3.3 Penetration Pressurization - Method

Quarterly readings were taken on the flow rotameters which are permanently installed in the Penetration Pressurization System. These readings represent the air/nitrogen makeup rate required to maintain approximately 60 psig in mechanical penetrations and 30 psig in electrical penetrations. High meter readings have occasionally occurred and were attributed to leaks in the compression fittings in the penetration pressurization system or to malfunctioning (stuck) rotameters which were promptly corrected even though the leakage had no bearing on the integrity of the containment penetration. Testing was performed in accordance with SP 1303-11.24 (Reference 8.5).

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4. TEST EQUIPMENT USED

4.1 Valve Test Equipment (See Attachment 1)

a. Rotameters - Sets of 3

Mfgr. - Brooks Inst. Co.
Model - 1114 Full View

Ranges:

Float Mat'l.	Tube No.	Range
Pyrex	R-2-15D	8-1,120 SCCM
Sapphire	R-2-15C	100-12,200 SCCM
Carboloy	R-6-15B	1,000-142,000 SCCM

Accuracy ± 2% full scale industrial accuracy

b. Temperature Indicators (as follows or similar)

Mfgr. - Ashcroft
Model - EH or AH / 3" or 5" Dial
Range - 30° to 130°F
Accuracy - ± 2°F

c. Pressure Indicators (as follows or similar)

Mfgr. - Ashcroft
Model - 1279 - 4-1/2" Dial
Range - 0 to 60 or 0 to 100 psig
Accuracy - ± 2 psig

d. Pressure Regulator (as follows or similar)

Mfgr. - Union Carbide Corp.
Model - UPG 3-75-580
Range - 0 to 100 psi output / 0 to 3000 psi input

e. Calibration Rotameters (Set of 2)

Mfgr. - Brooks Inst. Co.
Models - 1110-05K2B1Z49, 1110-08K2B1Z06
Ranges - 20 to 16,000 SCCM, 3,600 to 234,000 SCCM
Repeatability - ± 1/4% of instantaneous
Accuracy - ± 1% instantaneous

f. Flow rate Calibrator
Mfgr. - Brooks Inst. Co.
Model - 1056A
Range 0 to 2,400 SCCM
Accuracy - ± 0.2% of indicated volume

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4.2 Access Hatch Test Equipment

a. Precision Pressure Gage (as follows or similar)

Mfgr. - Heise
Model - CM
Range - 0 to 60 psig
Resolution - 0.25 psig
Accuracy - 0.1% of instrument span

b. Barometer (as follows or similar)

Mfgr. - Pennwalt
Model - FA185260A
Range - 10.8 to 15.5 psia
Resolution - 0.005 psia
Accuracy - 0.1% of instrument span

4.3 Penetration Pressurization Test Equipment

a. Flow Rotameters - (Permanent System Equipment)

Mfgr. - Brooks Inst. Co.
Model - 1114
Range - 0 to 10 SCFH at 60 psia air
Accuracy - ± 2% industrial accuracy

APPENDG

5. SUMMARY AND INTERPRETATION OF DATA

5.1 Valve Test Results

As-Found/As-Left Leakage to this date - Also see tabulation of individual results in Attachment #2.

	Total Leakage	Tech. Spec. Limit	% Tech. Spec. Limit
As-Found MAXPATH	62,687 SCCM	104,846 SCCM	<59.8%
As-Left MAXPATH	22,153 SCCM	104,846 SCCM	<21.1%

NOTE: The total shown above is "MAXIMUM PATHWAY" leakage. Only the highest valve leakage on each penetration is counted. This number is labeled as "MAXPATH" on the tabulation of results in Attachment 2.

EXAMPLE: Penetration XYZ has three containment isolation valves inside the Reactor Building in parallel and one outside. The leakage from the three inside totals 500 SCCM and the outside valve is 1000 SCCM. The penetration "MAXPATH" leakage is counted as 1,000 SCCM not 1,500 SCCM. "MAXPATH" leakage assumes the higher leaking Reactor Building penetration pressure boundary is the only boundary left intact during a DBA.

1989 Reactor Building Purge Valve Data				
FREQUENCY	DATE	AH-V1A/1B LEAKAGE RESULTS (SCCM)	DATE	AH-V1C/1D LEAKAGE RESULTS (SCCM)
FIRST QUARTER	03-26-89	1209	03-24-89	7566
SECOND QUARTER	06-26-89	1229	06-24-89	3783
THIRD QUARTER	09-26-89	897	09-24-89	10335
FOURTH QUARTER	12-27-89	995	12-25-89	*13924

* Thrust plate flange gaskets were found to be leaking. Repaired valve and retested on January 8, 1990. Leakage was reduced to 1209 SCCM.

5.2 Access Hatch Test Results

5.2.1 Overall semi-annual access hatch leakage test results in accordance with SP 1303-11.18 (Reference 8.2):

COMPONENT	FIRST HALF TEST		SECOND HALF TEST	
DESCRIPTION	DATE	LEAKAGE RESULTS	DATE	LEAKAGE RESULTS
PERSONNEL ACCESS HATCH	05-28-89	1750	11-24-89	859
EQUIPMENT ACCESS HATCH	05-28-89	1782	11-25-89	1782

The leakage from these tests were within the established target criteria of less than 2500 SCCM.

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5.2.2 Door Seal Leakage Test in accordance with SP 1303-11.25 (Reference 8.6).

The Personnel and Equipment Hatch Door seals had been satisfactory leak tested throughout 1989 resulting in leakage from each door seal to be less than 3 SCFH.

5.3 Penetration Pressurization System Quarterly Leakage Test Results in accordance with SP 1303-11.24 (Reference 8.5):

FREQUENCY	DATE	MECHANICAL LEAKAGE (SCFH)	ELECTRICAL LEAKAGE (SCFH)
FIRST QUARTER	03-13-89	35	0.6
SECOND QUARTER	06-13-89	1.0	0
THIRD QUARTER	No longer required by Technical Specifications		
FOURTH QUARTER	As of August 13, 1989		

There is no Technical Specification limit on Penetration Pressurization System Leakage. The system leakage was maintained as low as practical.

APPENDG

6. ERROR ANALYSIS

6.1 Valve Testing Errors (For purge valves see Section 6.2)

The flow meters used in the field have normal industrial accuracies of ± 2% full scale in the 10-100% (15-150 mm) scale range. Prior to use, mm versus sccm graphs were developed for the meters by 10 point calibrations using high accuracy (± 1% instantaneous) lab rotameters. During the leak test program, weekly 3 point standardizations were performed on the field rotameters to verify continued accuracy. The acceptance criteria for these standardizations was a variance of no more than 4% from the calibration graphs. If meters were repaired or the 3 point standardization exceeded the inaccuracy limit, a new 10 point calibration was performed. Scale readings on the leak rate procedure (SP 1303-11.18) data sheets were evaluated and corrected using the methods in Attachment 3. Conservative bias was introduced into the results by assuming 15 mm (10% of scale) as the minimum scale. Approximately half of the test results actually showed a minimum scale reading. More involved error corrections were not considered meaningful based on the acceptable total leakage As-Found and the low total leakage As-Left.

6.2 Access Hatch and Purge Valve Testing Errors

The measured pressure drops were corrected by adding the minimum scale increment of the gage used for both the Heise gage and the barometer. This conservatively corrected for the resolution and repeatability errors. Gages used were recently calibrated. A minimum one hour temperature/pressure stabilization period was used prior to each pressure drop test. The access hatches and purge valves are not instrumented to allow temperature corrections.

6.3 Penetration Pressurization Testing Errors

These test results were used for information only and do not count toward the total leakage limit for Technical Specification conformance. The meters, installed permanently in the system, have ± 2% full scale industrial accuracy.

7. LESSONS LEARNED/IMPROVEMENTS/DEGRADATION

7.1 The As-Found MAXPATH leakage was less than two-thirds the Technical Specification limit of 104,846 SCCM. Of this total 31,527 SCCM was attributed to CA-V2 seat leakage. The As-Left MAXPATH leakage was about one-fifth the Technical Specification limit. This improvement was due to repairs and modifications on containment isolation valves as summarized in Section 2.1.

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7.2 The seat rings in CA-V2/5A were replaced and pinned into place rather than welded. The Hylomar thread sealant was upgraded to GTS graphite thread sealant. Both of these changes significantly reduced the leakage past the seat rings of the valve. CA-V5B leakage was acceptable therefore no upgrade was performed.

7.3 A special test (associated with NRC IEN 88-73) on AH-V1B/1C was performed during the integrated leak rate test which demonstrated that testing these valves in the non-accident direction yields equivalent or more conservative leak test results.

7.4 CA-V189 seat leakage was considered high but acceptable. There are no plans to modify seat internals with split discs unless the valve exhibits significantly higher leak rates.

7.5 The cycling of valves unnecessarily will be minimized where practical during future LLRT. The existing surveillance procedure is being revised to provide guidance.

8. REFERENCES

8.1 1430-Y-22, Standardization of Flow Rotameters

8.2 SP 1303-11.18, Reactor Building Local Leak Rate Testing

8.3 Three Mile Island, Unit 1, Technical Specification 4.4.1

8.4 TMI Surveillance File (Records stored in CARIRS, Data Base AA60 REC.TYPE 018-12)

8.5 SP 1303-11.24, Reactor Building Local Leakage Penetration Pressurization

8.6 SP 1303-11.25, Reactor Building Local Leakage Access Hatch Door Seals

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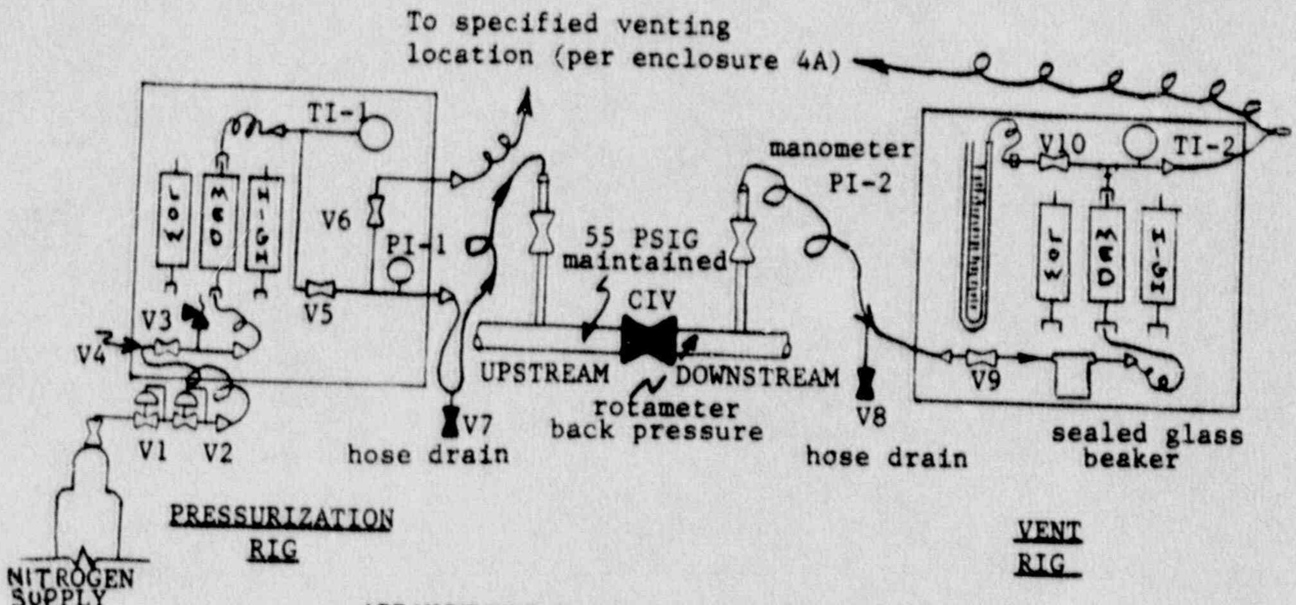
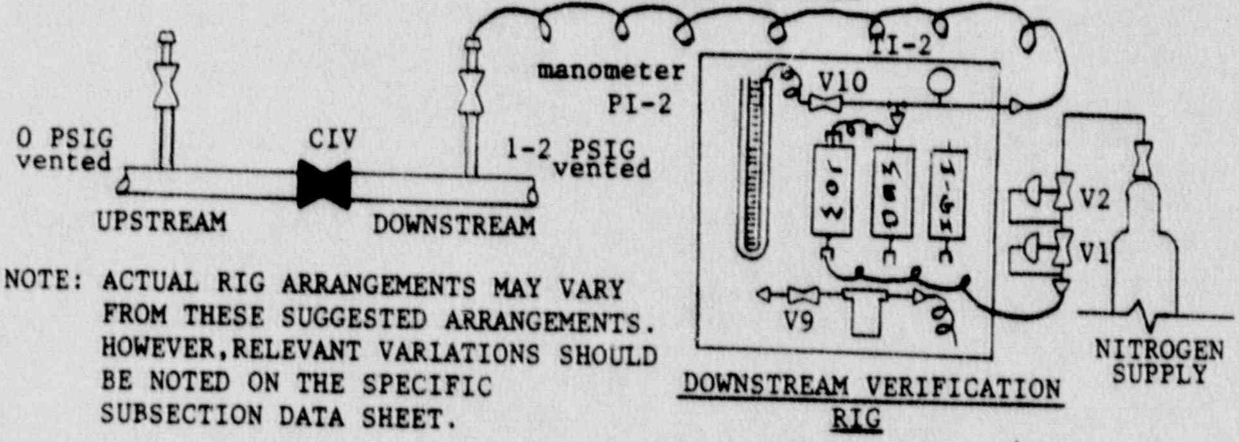
Attachment 1



ARRANGEMENT FOR ENCLOSURE 4A TEST



NOTE: ACTUAL RIG ARRANGEMENTS MAY VARY FROM THESE SUGGESTED ARRANGEMENTS. HOWEVER, RELEVANT VARIATIONS SHOULD BE NOTED ON THE SPECIFIC SUBSECTION DATA SHEET.

ARRANGEMENT FOR DOWNSTREAM VERIFICATION

EQUIPMENT DESCRIPTIONS

VALVES: V1/V2-Regulator, V3-65# Relief, V4/V5/V7/V8/V9/V10-Ball, V6- toggle/globe

PRESSURE REGULATORS: PI-1,0-60 PSIG usage range with 2 PSI increments; PI-2, manometer with 36 inch long scale

TEMPERATURE INDICATORS: TI-1/TI-2, 0-200°F usage range with 2°F increments

ROTAMETERS:

ROTAMETER	TUBE #	FLOAT	AT 0 PSIG (SCCM)	AT 55 PSIG(SCCM)
LOW	R-2-15-D	BLACK GLASS	0-380	45-1120
MED	R-2-15-C	SYN SAPHIRE	250-5300	800-12500
HIGH	R-6-15-B	CARBOLOY	4750-62000	12000-144000

Attachment 2

ITEM	TAG	ASFOUND	ASLEFT	ASLTDATE	COMMENTS
****	********	********	*******	********	********
1	AH-VIA/B	1794	1794	1/9/90	LOW
2	2ND	.01	.01		
3	3RD	.01	.01		
4	4TH	.01	.01		
5	AH-V1C/D	1209	1209	1/8/90	LOW
6	2ND	.01	.01		
7	3RD	.01	.01		
8	4TH	.01	.01		
9					
10					
11	CA-V1	71	57	2/10/90	OK
12	CA-V2	31527	57	2/10/90	FAILED
13	CA-V3	69	46	2/10/90	OK
14	CA-V4A	15	44	2/15/90	OK
15	CA-V4B	60	44	2/15/90	OK
16	CA-V5A	4473	44	2/15/90	HIGH
17	CA-V5B	1378	2420	2/15/90	HIGH
18	CA-V13	69	46	2/10/90	OK
19	CA-V189	1237	1237	1/21/90	HIGH
20	CA-V192	71	71	1/21/90	LOW
21					
22					
23	CF-V2A	71	80	2/20/90	OK
24	CF-V2B	70	45	2/15/90	OK
25	CF-V12A	307	307	1/17/90	OK
26	CF-V12B	70	70	1/17/90	OK
27	CF-V19A	626	626	1/16/90	OK
28	CF-V19B	13	13	1/16/90	OK
29	CF-V20A	1072	1072	1/16/90	HIGH
30	CF-V20B	70	70	1/16/90	OK
31	CM-V1	70	70	1/15/90	OK
32	CM-V2	87	87	1/15/90	OK
33	CM-V3	70	70	1/15/90	OK
34	CM-V4	70	70	1/15/90	OK
35	DH-V64	86	86	1/28/90	OK
36	DH=V69	44	44	1/27/90	OK
37					
38					
39	FTTEAST	45	84	2/21/90	OK
40	FTTWEST	374	840	2/21/90	HIGH
41	HM=V1A	71	71	1/15/90	OK
42	HM=V1B	70	70	1/15/90	OK
43	HM-V2A	71	71	1/15/90	OK
44	HM-V2B	70	70	1/15/90	OK
45	HM-V3A	71	71	1/15/90	OK
46	HM-V3B	70	70	1/15/90	OK
47	HM-V4A	71	71	1/15/90	OK
48	HM-V4B	70	70	1/15/90	OK
49	HP-V1	71	71	1/17/90	LOW
50	HP-V6	71	71	1/17/90	LOW
51	HR-V2A/B	60	47	2/24/90	OK
52	HR-V4A/B	60	47	2/24/90	OK
53	HRV22A/B	60	60	1/8/90	OK
54	HR-V23A	60	60	1/8/90	OK
55	HR-V23B	60	60	1/8/90	OK
56					
57					
58	IA-V6/20	85	85	12/24/90	OK

59	IC-V2	70	171	2/11/90	LOW
60	IC-V3	70	70	1/20/90	LOW
61	IC-V4	69	69	1/20/90	LOW
62	IC-V6	44	44	1/25/90	LOW
63	IC-V16	4756	858	2/1/90	HIGH
64	IC-V18	70	70	1/20/90	LOW
65					
66	LR-V4	46	.01	1/14/[illegible]	REMOVED
67	LR-V5	46	.01	1/14/90	REMOVED
68	LR-V6	46	.01	1/14/90	REMOVED
69					
70					
71					
72	MU-V2A	44	44	1/22/90	OK
73	MU-V2B	44	44	1/23/90	OK
74	MU-V3	44	44	1/22/90	LOW
75	MU-V18	385	385	1/30/90	OK
76	MU-V20	156	156	1/24/90	OK
77	MU-V25	170	170	1/23/90	OK
78	MU-V26	45	45	1/23/90	OK
79	MU-V116	707	707	1/23/90	OK
80					
81	NI-V26	46	46	2/23/90	OK
82	NI-V27	46	46	2/23/90	OK
83	NS-V4	137	1634	2/9/90	LOW
84	NS-V11	2840	1802	1/27/90	HIGH
85	NS-V15	327	833	1/31/90	OK
86	NS-V35	575	2662	2/8/90	HIGH
87	PENET414	.01	85	2/24/90	OK
88	PENET417	85	85	1/17/90	OK
89	PENET104	85	85	2/25/90	OK
90	PENET105	86	86	2/24/90	OK
91	PENET106	86	86	2/24/90	OK
92	PENET210	85	85	2/24/90	OK
93	PENET211	85	85	2/24/90	OK
94	PENET241	83	83	2/24/90	OK
95					
96					
97	PP-V210	285	285	2/26/90	OK
98	PP-V211	19	19	2/26/90	OK
99	PP-V212	448	448	2/25/90	OK
100	PP-V213	19	19	2/25/90	OK
101	RB-V2A	46	71	2/9/90	LOW
102	RB-V7	46	71	2/9/90	LOW
103	SA-V2/3	86	86	2/25/90	OK
104	SF-V23	45	45	1/29/90	OK
105	WDG-V3/4	3608	85	2/18/90	HIGH
106	WDL-V303	48	48	1/26/90	OK
107	WDL-V304	44	44	1/26/90	OK
108	WDL-V534	85	85	2/4/90	LOW
109	WDL-V535	85	85	2/4/90	LOW
110					
111					
112	EQPFLG	135	135	2/11/90	OK
113	PERACCES	859	859	11/24/89	LOW
114	2ND	.01	.01		
115	EMEACCES	1782	1782	11/25/89	OK
116	2ND	.01	.01		
117					
118					
119	MINPATH	14543	13344		OK
120	MAXPATH	62687	22153		OK

GPU Nuclear	TMI-1 Surveillance Procedure	Number 1303-11.18
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Attachment 3

Leakage Results Evaluation

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

FOR USE OF SUPPLY ROTAMETER DATA: Procedure:	FOR USE OF VENT ROTAMETER DATA: Procedure
a. Record supply meter reading in (1) below*. Also identify the meter used by tube No. in (8) below and the metering pressure in (9). b. Convert meter units to SCCM units using latest lab meter calibrate curve. Enter in (3) below. c. Correct results for temperature Enter supply temperature in (4) below. Calculate and enter in (7) below. * If meter scale reading was less than 15mm (minimum scale) use 15mm in calculations.	a. Record vent meter reading in (1) below*. b. Record downstream verification meter reading in (2) below. Also identify the respective meters used in (8) below and the metering pressures in (9). c. Convert meter units to SCCM units using latest lab meter calibration curve. Enter in (3) below. d. Correct results for temperature. Enter vent temperature (°F) in (4) below. then Calculate and enter in (5) below. e. If measurements of any other significant leakage paths (fluid block check valve, packing) are being claimed enter corrected flow (SCCM) in (6) below.

(mm) (_____ (1) + _____ (2)) convert (SCCM) (_____ + (3) _____) X 530 / (_____ (4) + 460) = _____ (5) SCCM

\+ _____ (6) SCCM

_____ _____ (8) (Identify meters used)

at _____ _____ (9) (Meter Pressures)

= CIV Leakage _____ (7) SCCM

LOCAL LEAK RATE TEST RESULTS

THREE MILE ISLAND UNIT 1 REACTOR BUILDING

The following terminology abbreviations were used in the LLRT computer data sheets:

1) .1 - (Alone) or any other number other than zero in the first decimal place means test scheduled.

2) .01 - (Alone) means no data available for the year or that the test was delayed. (e.g. valve not installed yet or not in previous testing scope.)

3) .001 - (Or any number other than zero in the third decimal place) after a leak rate (i.e. 59500.001) means actual leak rate was greater than measured/recorded value.

4) AsFound - leak rate (SCCM) in the As-Found condition before any repairs or adjustments.

5) AsLeft - The leak rate (SCCM) after any adjustments/repairs.

6) Dates - Date of the last acceptable test results for the item.

7) Desc - Description of the valve or penetration.

8) Oper - Type of valve operator (actuator).

9) Notest - The Tech Spec scope did not require this valve to be tested during the respective year.

10) Novalve - This valve was installed during a later refueling outage.

11) Comments - Cognizant Engineer comments about the results:

    A) Failed - Exceeded the plant established leakage rate limit from SP 1303-11.18 Enclosure 7 which made repair/adjustment necessary.
    B) High - Based on leakage history of component. The leak rate is much greater than Target but is still less than the established maximum leak rate limit. Repair <u>might</u> not be worth the effort.
    C) Low (Excellent) - Based on leakage history of component. The leak rate is much less than would be predicted by the normal method of assigning Target Criteria.
    D) OK - No problems with leakage.
    E) Other - E.G. newvalve, novalve, notest, repacked seatwork, stembent, etc. (self-explanatory), "REMOVED" means valve was eliminated from plant design.

12) Size - The nominal pipe size for the leakage barrier.