SwRI-7857

QUAD CITIES NUCLEAR POWER STATION UNIT 1 REACTOR VESSEL IRRADIATION SURVEILLANCE PROGRAM ANALYSIS OF CAPSULE NO. 8

by

E. B. Norris

FINAL REPORT

SwRI Project No. 06-7857

for

Commonwealth Edison Company
Nuclear Stations Division
P. O. Box 767, Room 1230 Edison
Chicago, Illinois 60690

August 1984

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Approved:

U. S. Lindholm, Director
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I. SUMMARY OF RESULTS

The analysis of the vessel material surveillance capsule basket No. 8 removed from the 215° position in the Quad Cities 1 pressure vessel during the 1982 refueling outage led to the following results.

(1) Based on a calculated neutron spectral distribution, the capsule received a fast fluence of 5.5×10^{16} n/cm^2, $E > 1$ MeV, in 6.64 Effective Full Power Years (EFPY) of operation.

(2) There were small positive or negative shifts in RT_{NDT} of the surveillance materials as a result of the above exposure. These data are consistent with the level of the exposure and may reflect data scatter rather than real changes in the RT_{NDT}.

(3) The Charpy upper shelf energy of the electroslag weld and heat-affected zone (HAZ) surveillance materials decreased, but that of the base plate material increased after exposure. This may also reflect data scatter.

(4) Based on a calculated neutron spectral and spatial distribution, the maximum fluence rate (flux) incident on the pressure vessel wall is 3.46×10^8 $n/cm^2/sec$, $E > 1$ MeV. Therefore, the projected maximum vessel wall neutron fluence after 32 EFPY of operation is 3.5×10^{17}, $E > 1$ MeV.

II. BACKGROUND

The allowable loadings on nuclear pressure vessels are determined by applying the rules in Appendix G, "Fracture Toughness Requirements," of 10CFR50 [1]. In the case of pressure-retaining components made of ferritic materials, the allowable loadings depend on the reference stress intensity factor (K_{IR}) curve indexed to the reference nil ductility temperature (RT_{NDT}) presented in Appendix G, "Protection Against Non-ductile Failure," of Section III of the ASME Code [2]. Further, the materials in the beltline region of the reactor vessel must be monitored for radiation-induced changes in RT_{NDT} per the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10CFR50.

The RT_{NDT} is defined in Paragraph NB-2331 of Section III of the ASME Code as the highest of the following temperatures:

(1) Drop-weight nil ductility temperature (DW-NDT) per ASTM E 208 [3]

(2) 60 deg F below the 50 ft-lb Charpy V-notch (C_V) temperature

(3) 60 deg F below the 35 mil C_V temperature.

The RT_{NDT} must be established for all materials, including weld metal and heat-affected zone (HAZ) material as well as base plates and forgings, which comprise the reactor coolant pressure boundary.

It is well established that ferritic materials undergo an increase in strength and hardness and a decrease in ductility and toughness when exposed to neutron fluences in excess of 10^{17} neutrons per cm^2 (E > 1 MeV) [4]. Also, it has been established that tramp elements, particularly copper, and nickel can affect the radiation embrittlement response of ferritic materials [5-7]. The relationship between increase in RT_{NDT} and copper content is defined in Regulatory Guide 1.99. Although this document is being revised by the NRC to reflect a more recent evaluation of neutron embrittlement data from reactor surveillance programs, estimates of shifts in RT_{NDT} in this report are based on the current Revision 1 of Regulatory Guide 1.99 [8].

In general, the only ferritic pressure boundary materials in a nuclear plant which are expected to receive a fluence sufficient to affect RT_{NDT} are those which are located in the core beltline region of the reactor pressure vessel. Therefore, material surveillance programs include specimens machined from or representative of the plate or forging material and weldments which are located in the core beltline region of high neutron flux density. ASTM E 185 [9] describes the recommended practice for monitoring and evaluating the radiation-induced changes occurring in the mechanical properties of pressure vessel beltline material.

General Electric has provided such a surveillance program for the Quad Cities 1 Nuclear Power Station. Six baskets of test specimens were located near the I.D. surface of the pressure vessel where the fast neutron flux density is slightly higher than that at the adjacent vessel wall surface. However, because of azimuthal variations in neutron flux density, these vessel wall basket fluences may lead or lag the maximum vessel fluence in a corresponding exposure period. Three sets of accelerated-exposure baskets were included to augment the surveillance program. These baskets, located near the core, would have much larger lead factors to provide early information on long-term behavior of the surveillance materials. The baskets also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

The Quad Cities 1 mechanical property surveillance capsules also include tensile specimens as recommended by ASTM E 185. At the present time, irradiated tensile properties are used to indicate that the materials tested continue to meet the requirements of the appropriate material specification and to judge credibility of the surveillance capsule Charpy data.

This report describes the results obtained from testing the contents of Basket No. 8 from the 215° position in Quad Cities Unit 1 (the second vessel wall surveillance capsule tested). The results were analyzed to determine the radiation-induced changes in the mechanical properties of the surveillance materials at the time of the refueling outage and the fluence rate at the vessel wall.

III. DESCRIPTION OF MATERIAL SURVEILLANCE PROGRAM

The G.E. vessel material surveillance program is described in detail in NEDO-10115 [10]. Six vessel wall baskets, each located opposite the vertical center of the core and containing encapsulated Charpy V-notch and tensile specimens, were placed in the Quad Cities 1 vessel at the pressure vessel wall prior to startup. In addition, three baskets were placed near the core. One vessel wall basket (No. 3) and two near-core baskets (Nos. 2 and 4) had been previously removed, tested, and the results reported [11,12].

The vessel wall Basket No. 8 , removed during the 1982 refueling outage from the 215° position, contained two impact capsules (each holding 12 Charpy V-notch specimens plus an iron, a nickel, and a copper flux wire) and three tensile capsules (each holding two tensile specimens). The capsules did not contain thermal monitors. Drawings of the impact and tensile specimens, and photographs of the impact and tensile capsules and the surveillance basket are shown in Appendix A.

According to NEDO-10115 [10], the Quad Cities 1 base metal specimens were made from flat slabs cut parallel to and one-quarter plate thickness from both of the plate surfaces, and were machined with their longitudinal axes parallel to the plate rolling direction. The electroslag weld (ESW) metal and ESW heat-affected zone (HAZ) specimens were cut from a test weld representing a vessel welded joint which had been fabricated from vessel base material, the weld and HAZ Charpy V-notch specimens being oriented with the long axis transverse to the weld direction and the weld HAZ tensile specimens with the long axis taken parallel to the weld direction. The notches of all Charpy V-notch specimens were perpendicular to the original plate surfaces.

The mechanical properties of unirradiated (baseline) surveillance specimens for the Quad Cities 1 vessel were determined and reported at the time of the testing and analysis of the specimens from the first baskets [13]. These data were used to determine the shifts in RT_{NDT} experienced by the materials in the 215° basket No. 8.

IV. TESTING OF IRRADIATED SPECIMENS

The capsule shipment, capsule opening, specimen testing, and reporting of results were carried out in accordance with the Project Plan for the Quad Cities 1 Nuclear Power Plant Reactor Vessel Irradiation Surveillance Program. Each of these activities is discussed below.

A. Shipment, Opening, and Inspection of Capsule

SwRI loaded the impact and tensile capsules into a radioactive material shipping cask and transported them to San Antonio, TX. After visually inspecting and photographing the surveillance capsule basket, SwRI personnel opened the basket and photographed the contents. There was no evidence of damage to the capsules, see Appendix A.

The capsules were opened and the contents identified and stored in accordance with SwRI Procedure XIII-MS-103-1. The end plugs were cut from each capsule with a band saw set up in the hot cell, then the test specimens and dosimeter wires were removed from the shell and placed in indexed receptacles.

Each mechanical test specimen was inspected for identification number, which was checked against the master list in NEDO-10115 [10], and no discrepancies were found. The neutron dosimeter wires were identified and placed in tagged containers.

B. Neutron Dosimetry

The gamma activities of the dosimeters were determined in accordance with SwRI Procedure XI-MS-101-1 using an IT-5400 multichannel analyzer and a Ge(Li) coaxial detector system. The calibration of the equipment was accomplished with ^{54}Mn, ^{60}Co, and ^{137}Cs radioactivity standards obtained from the U.S. Department of Commerce National Bureau of Standards. All activities were corrected to the time-of-removal (TOR) at reactor shutdown.

The dosimeter wires were weighed on a balance having a sensitivity of 1 microgram. Infinitely dilute saturated activities (A_{SAT}) were

calculated for each of the dosimeters because A_{SAT} is directly related to the product of the energy-dependent microscopic activation cross section and the neutron flux density. The relationship between A_{TOR} and A_{SAT} is given by:

$$\frac{A_{TOR}}{A_{SAT}} = \sum_{m=1}^{m=n} P_m\left(1 - e^{-\lambda T_m}\right)\left(e^{-\lambda t_m}\right)$$

where:
- P_m = fraction of full power (2511 MwTh) for the operating period m;
- λ = decay constant for the activation product, day^{-1};
- T_m = number of operating days for the period m;
- t_m = decay time after the operating period m, days.

The Quad Cities 1 operating history up to the 1982 refueling outage, which was used in the calculation of A_{TOR}, is presented in Table I.

The primary result desired from the dosimeter analysis is the total fast neutron fluence (> 1 MeV) which the surveillance specimens received. The average flux density at full power is given by:

$$\phi = \frac{A_{SAT}}{N_0 \bar{\sigma}}$$

where:
- ϕ = energy-dependent neutron flux density, n/cm^2/sec;
- A_{SAT} = saturated activity, dps/mg target element;
- $\bar{\sigma}$ = spectrum-averaged activation cross section, cm^2; and
- N_0 = number of target atoms per mg.

The total neutron fluence is then equal to the product of the average neutron flux density and the equivalent reactor operating time at full power.

A discrete ordinates Sn transport analysis for the Quad Cities 1 reactor vessel was performed using the DOT-IV code to determine the radial,

TABLE I

OPERATIONS SUMMARY - QUAD CITIES 1

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
04/72	30	2.62	0.0870	3781
05/72	31	8.16	0.2630	3750
06/72	30	5.95	0.1980	3720
07/72	31	10.08	0.3250	3689
08/72	31	14.74	0.4750	3658
09/72	30	17.65	0.5880	3628
10/72	31	21.12	0.6810	3597
11/72	30	22.11	0.7370	3567
12/72	31	23.66	0.7630	3536
01/73	31	23.93	0.7720	3505
02/73	28	20.34	0.7260	3477
03/73	31	23.40	0.7550	3446
04/73	30	15.36	0.5120	3416
05/73	31	14.10	0.4550	3385
06/73	30	25.81	0.8600	3355
07/73	31	26.30	0.8480	3324
08/73	31	21.23	0.6850	3293
09/73	30	18.67	0.6220	3263
10/73	31	21.02	0.6780	3232
11/73	30	20.86	0.6950	3202
12/73	31	25.88	0.8350	3171
01/74	31	18.73	0.6040	3140
02/74	28	23.58	0.8420	3112
03/74	31	24.54	0.7920	3081
04/74	30	-	0.0000	3051
05/74	31	-	0.0000	3020
06/74	30	-	0.0000	2990
07/74	31	3.54	0.1140	2959
08/74	31	25.45	0.8210	2928
09/74	30	26.31	0.8770	2898
10/74	31	15.24	0.4920	2867
11/74	30	23.22	0.7740	2837
12/74	31	26.17	0.8440	2806
01/75	31	7.70	0.2480	2775
02/75	28	3.11	0.1110	2747
03/75	31	27.74	0.8950	2716
04/75	30	28.00	0.9340	2686
05/75	31	19.11	0.6160	2655
06/75	30	24.32	0.8110	2625
07/75	31	23.17	0.7470	2594
08/75	31	20.68	0.6670	2563
09/75	30	23.23	0.7740	2533

TABLE I CONT.

OPERATIONS SUMMARY - QUAD CITIES 1

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
10/75	31	22.51	0.7260	2502
11/75	30	19.97	0.6660	2472
12/75	31	19.11	0.6160	2441
01/76	31	1.13	0.0360	2410
02/76	29	-	0.0000	2381
03/76	31	6.86	0.2580	2350
04/76	30	20.92	0.6970	2320
05/76	31	21.86	0.7050	2289
06/76	30	14.93	0.4980	2259
07/76	31	8.98	0.2900	2228
08/76	31	16.60	0.5350	2197
09/76	30	26.06	0.8690	2167
10/76	31	25.24	0.8140	2136
11/76	30	19.82	0.6610	2106
12/76	31	27.10	0.8740	2075
01/77	31	24.96	0.8050	2044
02/77	28	18.59	0.6640	2016
03/77	31	15.45	0.4980	1985
04/77	30	-	0.0000	1955
05/77	31	12.26	0.3950	1924
06/77	30	21.01	0.7000	1894
07/77	31	22.24	0.7170	1863
08/77	31	11.60	0.3740	1832
09/77	30	9.05	0.3020	1802
10/77	31	17.55	0.5660	1771
11/77	30	23.72	0.7900	1741
12/77	31	25.00	0.8070	1710
01/78	31	28.99	0.9350	1679
02/78	28	20.26	0.7240	1651
03/78	31	28.41	0.9160	1620
04/78	30	17.57	0.5860	1590
05/78	31	19.74	0.6370	1559
06/78	30	21.56	0.7190	1529
07/78	31	25.60	0.8260	1498
08/78	31	27.26	0.8790	1467
09/78	30	23.82	0.7940	1437
10/78	31	21.89	0.7060	1406
11/78	30	19.35	0.6450	1376
12/78	31	19.84	0.6400	1345
01/79	31	8.49	0.2740	1314
02/79	28	0.11	0.0040	1286
03/79	31	24.43	0.7880	1255

TABLE I CONT.

OPERATIONS SUMMARY - QUAD CITIES 1

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
04/79	30	26.85	0.8950	1225
05/79	31	21.13	0.6810	1194
06/79	30	28.33	0.9440	1164
07/79	31	29.13	0.9400	1133
08/79	31	28.40	0.9160	1102
09/79	30	17.02	0.5670	1072
10/79	31	29.55	0.9530	1041
11/79	30	26.31	0.8770	1011
12/79	31	21.36	0.6890	980
01/80	31	28.99	0.9350	949
02/80	29	25.43	0.8770	920
03/80	31	28.37	0.9150	889
04/80	30	27.43	0.9140	859
05/80	31	24.17	0.7800	828
06/80	30	19.79	0.6600	798
07/80	31	17.02	0.5490	767
08/80	31	17.88	0.5770	736
09/80	30	-	0.0000	706
10/80	31	-	0.0000	675
11/80	30	-	0.0000	645
12/80	31	5.03	0.1620	614
01/81	31	24.95	0.8050	583
02/81	28	26.20	0.9360	555
03/81	31	21.56	0.6950	524
04/81	30	29.30	0.9770	494
05/81	31	26.93	0.8690	463
06/81	30	28.05	0.9350	433
07/81	31	21.04	0.6790	402
08/81	31	26.20	0.8450	371
09/81	6	27.31	0.9100	341
10/81	31	29.35	0.9470	310
11/81	30	22.19	0.7400	280
12/81	31	29.23	0.9430	249
01/82	31	28.93	0.9330	218
02/82	28	26.17	0.9350	190
03/82	31	27.98	0.9030	159
04/82	30	23.66	0.7890	129
05/82	31	19.63	0.6330	98
06/82	30	19.26	0.6420	68
07/82	31	18.50	0.5970	37
08/82	31	16.04	0.5180	6
09/82	6	2.36	0.3930	0

Total irradiation time = 6.64 EFPY

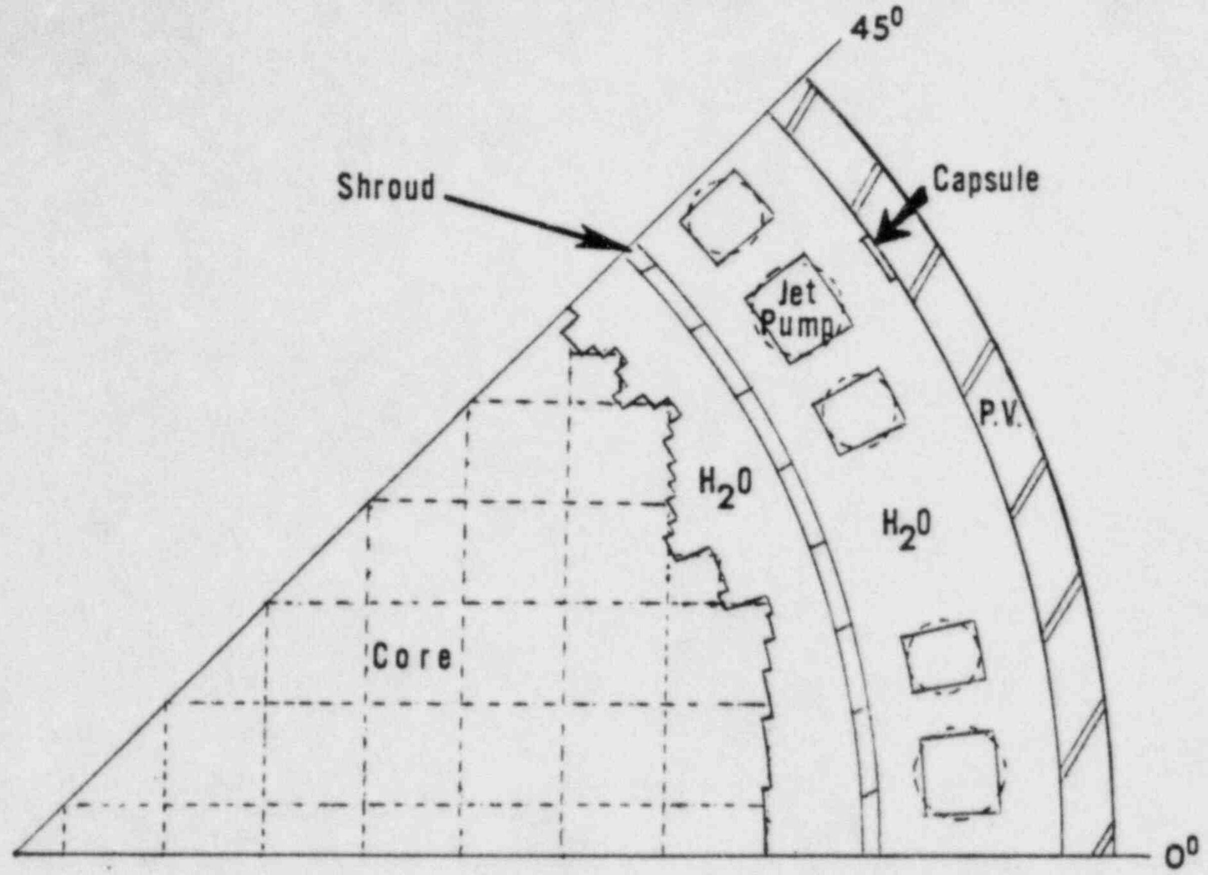
vertical, and azimuthal dependence of the fast neutron ($E > 1.0$ MeV) flux density and energy spectrum incident on the reactor vessel and surveillance basket capsules. The R-Θ and R-Z calculations were made using 15 of the 22 neutron group DLC-23/CASK cross section library, a P_1 expansion of the scattering matrix, and an S_8 order of angular quadrature. The R-Θ model utilized a one-eighth segment, and the R-Z model was represented as a one-half segment (see Figure 1) because of the symmetry involved. The various material zones (e.g., core, coolant, internals, pressure vessel, etc.) were described by homogenizing the major elements within each zone.

The resulting spectra were used to calculate the spectrum-averaged reaction cross sections for the threshold detectors and the lead factors needed to relate the neutron exposure of the pressure vessel to that of the surveillance basket capsules. The pertinent factors obtained from this transport analysis are summarized in Table II. The capsule flux is less than the peak fast flux incident on the vessel I.D. because of azimuthal and vertical flux variations related to the core geometry and power distribution. The calculated azimuthal and vertical flux maps are shown in Figure 2.

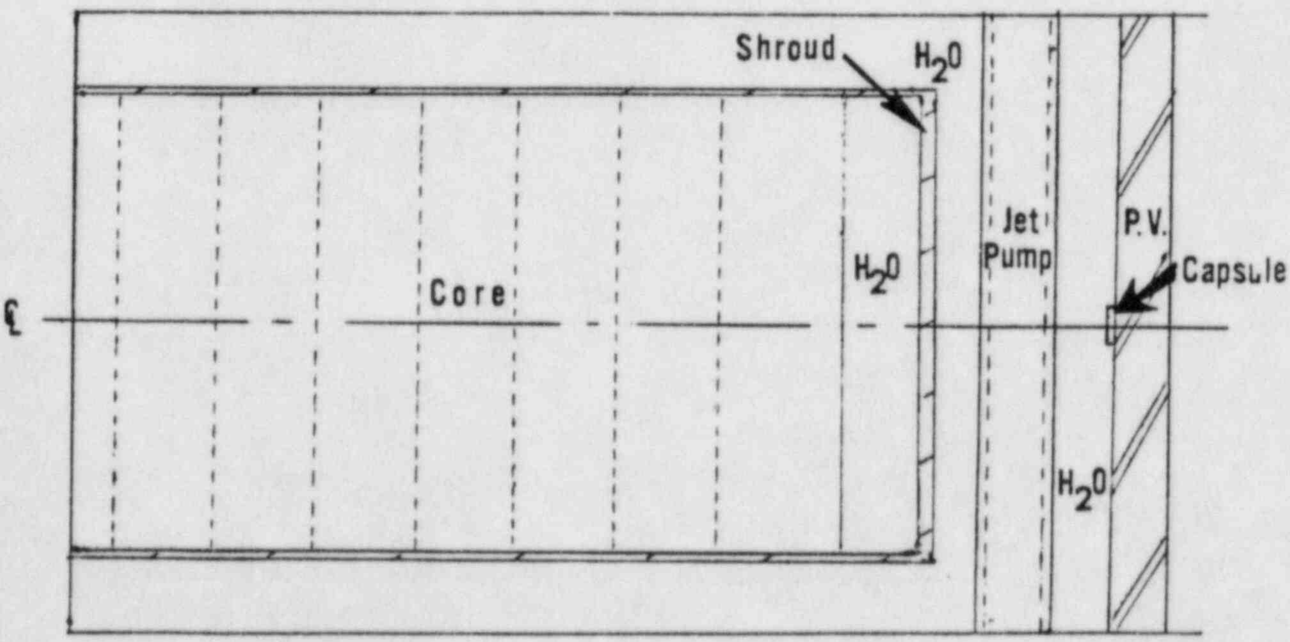
The calculated cross section for the $^{54}Fe(n,p)^{54}Mn$ reaction is somewhat less than the 236 millibarn (mb) value calculated for vessel wall Basket No. 3 [11]. It is believed that the values given in Table II are reasonable because SwRI-computed reaction cross sections for Browns Ferry Unit 3, which used the same methodology, agree well with those measured by General Electric during the same core cycle [14].

The dosimetry results obtained with the calculated spectra are presented in Table III. If a fission-spectrum energy distribution is assumed at the capsule location, the cross section for the $^{54}Fe(n,p)^{54}Mn$ reaction ($E > 1.0$ MeV) would be 98.26 mb [4], and the resulting value for fast flux at the capsule location would be 4.2×10^8 n/cm^2/sec. This value is reported for reference only and has not been used in the analysis of results.

The discrepancies in the peak vessel flux values determined from the several dosimeter materials are attributed primarily to the uncertainties in the calculated spectra, in the spatial distribution, and in the reaction



(a) R-Θ Model



(b) R-Z Model

FIGURE 1. DISCRETE ORDINATES CALCULATIONAL MODELS, QUAD CITIES UNIT 1

TABLE II

RESULTS OF DISCRETE ORDINATES Sn TRANSPORT ANALYSIS QUAD CITIES 1 AND 2

A. Calculated Reaction Cross Sections for Analysis of Fast Neutron Monitors ($E > 1.0$ MeV)

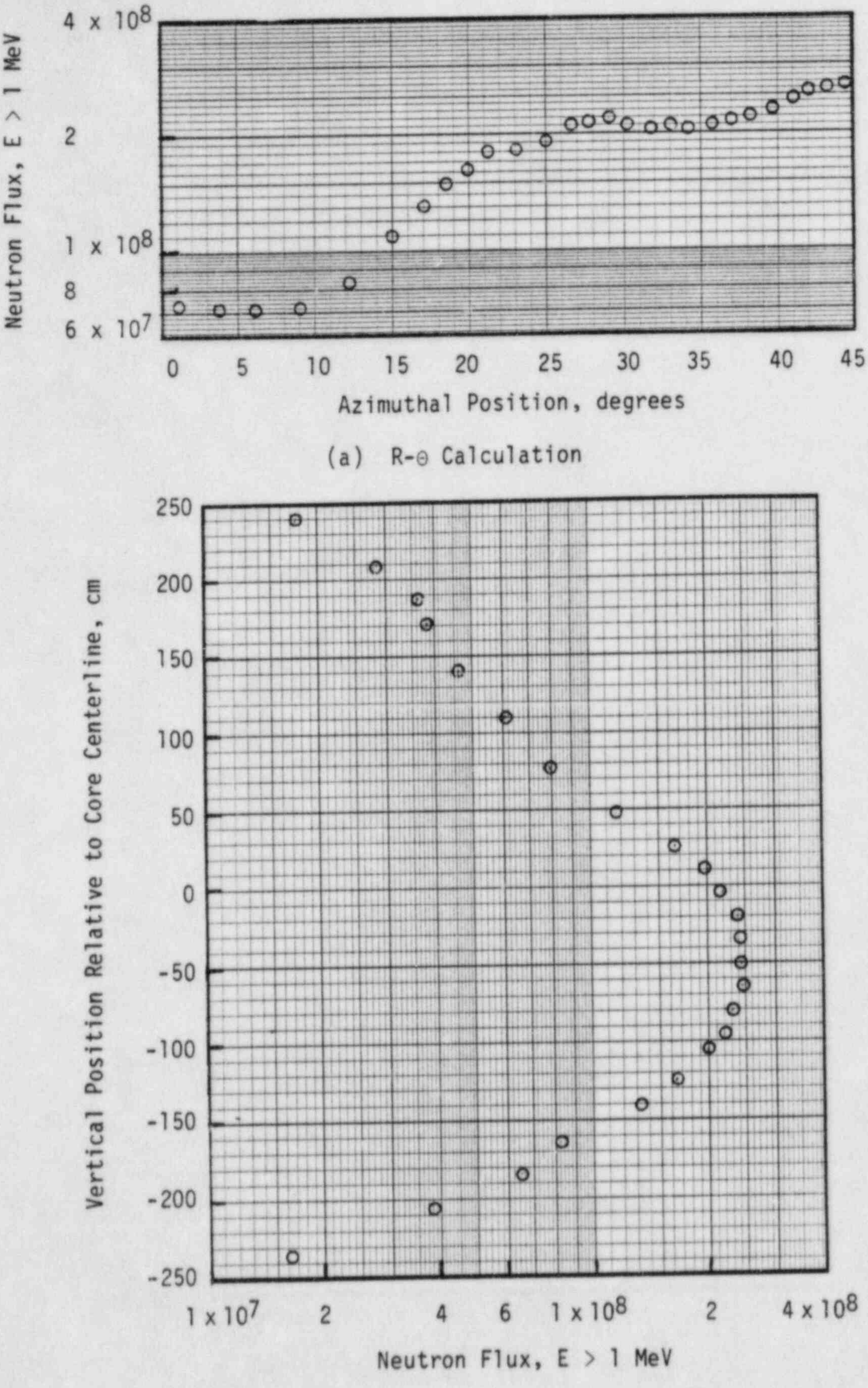
Reaction	$\bar{\sigma}$ (barns)
$^{54}Fe(n,p)^{54}Mn$	0.197
$^{58}Ni(n,p)^{58}Co$	0.237
$^{63}Cu(n,\alpha)^{60}Co$	0.00330

B. Calculated Basket Lead Factors ($E > 1$ MeV)

Position[a]	Location within Vessel Wall	Lead Factor[b]
215°	I.D. Surface	0.755
215°	1/4T	1.07
215°	3/4T	4.75

(a) Azimuthal position of surveillance Basket No. 8

(b) $\frac{\text{Capsule neutron flux density, } E > 1.0 \text{ MeV}}{\text{Maximum neutron flux density at vessel I.D., } E > 1.0 \text{ MeV}}$



(a) R-Θ Calculation

(b) R-Z Calculation

FIGURE 2. CALCULATED FAST FLUX (E > 1 MeV) DISTRIBUTION, QUAD CITIES UNIT 1

TABLE III

SUMMARY OF NEUTRON DOSIMETRY RESULTS
QUAD CITIES 1, VESSEL WALL SURVEILLANCE BASKET NO. 8 (215°)

Dosimeter I.D. Material	Dosimeter I.D. Capsule	Activation Reaction	Weight (mg)	A_{TOR} (dps/mg)	A_{SAT} (dps/mg)	Flux, E > 1 MeV(a) ($n/cm^2/sec$)
Fe	G1	$^{54}Fe(n,p)^{54}Mn$	112.8	1.813×10^1	2.518×10^1	2.08×10^8
↓	G2	↓	112.0	1.859×10^1	2.582×10^1	2.13×10^8
					Average =	2.10×10^8
Cu	G1	$^{63}Cu(n,\alpha)^{60}Co$	297.3	3.309×10^0	6.700×10^0	3.10×10^8
↓	G2	↓	290.9	3.351×10^0	6.785×10^0	3.14×10^8
					Average =	3.12×10^8
Ni	G1	$^{58}Ni(n,p)^{58}Co$	210.2	2.046×10^2	3.141×10^2	1.89×10^8
↓	G2	↓	213.0	2.091×10^2	3.210×10^2	1.93×10^8
					Average =	1.91×10^8

(a) Calculated flux values subject to a ± 16.5% uncertainty (1σ).

cross sections. Other neutronic factors contributing to the estimated ± 16.5% uncertainty (1σ) in the calculated fluxes are the determination of disintegration rates and the calculation of reaction rates (A_{SAT}/N_O). For example, the iron monitors effectively measured the capsule flux for the last three years of operation, and the copper monitors effectively measured the capsule flux for the entire exposure period.

Averaging the results obtained from the iron and copper threshold neutron dosimeters, the fast neutron flux at the surveillance basket location during full power operation would be 2.61×10^8 n/cm²/sec, E > 1 MeV, and the peak value incident on the pressure vessel I.D. would be 3.46×10^8. Since Quad Cities 1 operated for 6.64 effective full power years (EFPY) up to the September 1982 refueling outage, the calculated basket and vessel fluences to that time are as follows:

- Surveillance Basket - 5.47×10^{16} n/cm²
- Pressure Vessel I.D. Surface - 7.24×10^{16} n/cm²
- Pressure Vessel 1/4T - 5.11×10^{16} n/cm²
- Pressure Vessel 3/4T - 1.15×10^{16} n/cm²

The vessel wall fluence as a function of plant operation is shown in Figure 3.

C. Mechanical Property Tests

Hardness tests were run in accordance with ASTM Method E 18 [15] on one Charpy V-notch specimen selected from each material group. The results are given in Table IV.

The irradiated Charpy V-notch specimens were tested on a calibrated* 240-ft-lb, 16-ft/sec SATEC impact machine (Model SI-1K) in accordance with SwRI Procedure XI-MS-104-1. The test temperatures, selected to develop the ductile-brittle transition and upper shelf regions, were obtained using a liquid conditioning bath monitored with a Fluke Model 2168A

* Inspected and calibrated using specimens and procedures obtained from the Army Materials and Mechanics Research Center.

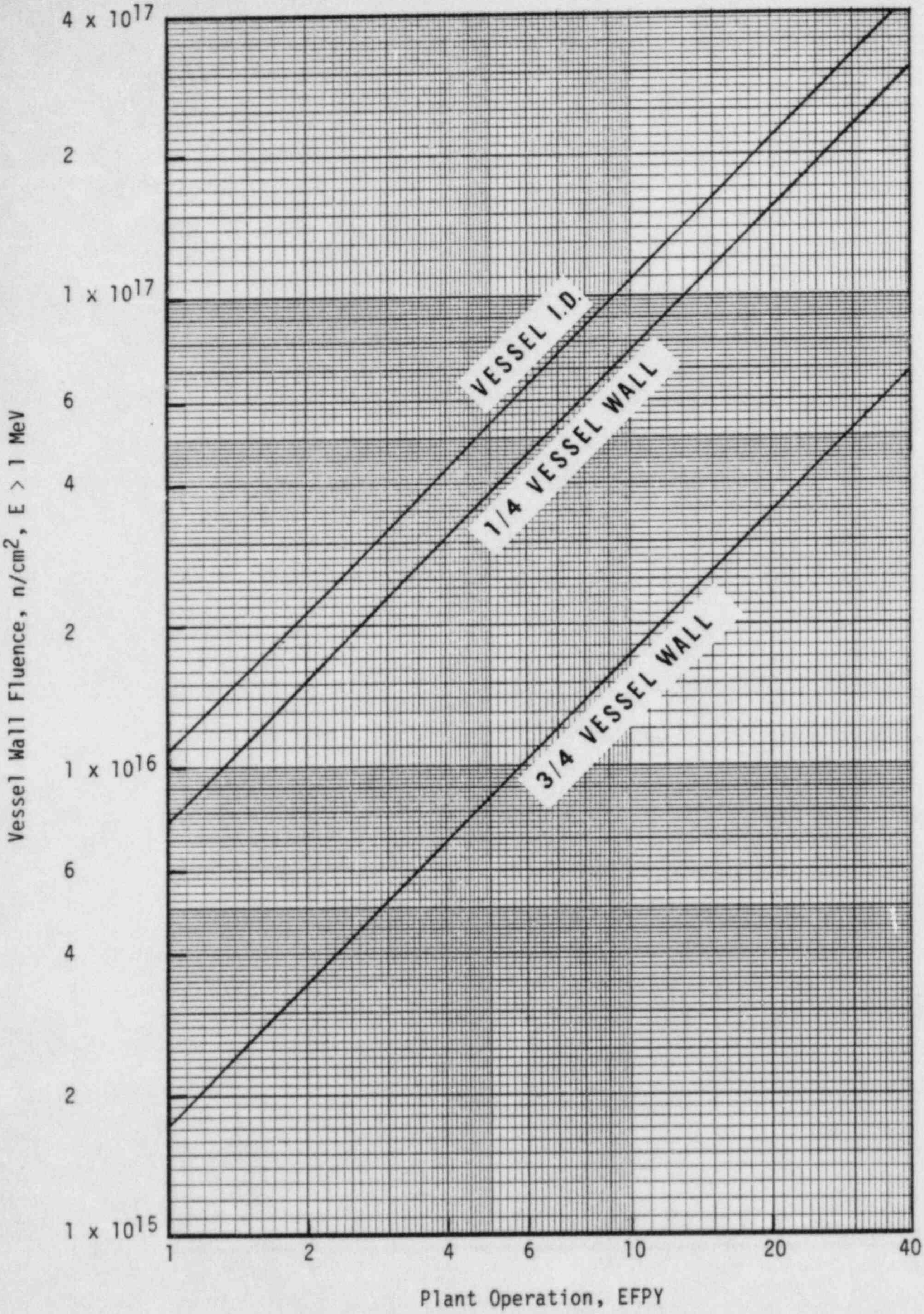


FIGURE 3. VESSEL WALL FLUENCE AS A FUNCTION OF OPERATION OF QUAD CITIES 1

TABLE IV

HARDNESS PROPERTIES OF SURVEILLANCE MATERIALS QUAD CITIES 1

(Fluence = 5.0×10^{16}, E > 1 MeV)

Test Material	Charpy Specimen No.[(a)]	Hardness (HRB)
Base	M16	94.0
		94.0
		94.5
		95.0
		94.5
		Average = 94.4
ESW	M7L	88.0
		90.0
		90.0
		90.5
		90.5
		Average = 89.8
ESHAZ	MPD	89.5
		89.5
		90.0
		91.0
		90.5
		Average = 90.1

(a) Specimen identification code given in Reference 10.

digital thermometer. The Charpy V-notch impact data obtained on the irradiated materials are presented in Table V, and photographs of the fracture faces are included in Appendix B. The Charpy V-notch transition curves for the irradiated plate material, weld metal, and HAZ material contained in the 215° capsules are compared to the unirradiated values in Figures 4 through 6. The shifts in the 50 ft-lb, 30 ft-lb, and 35-mil lateral expansion transition temperatures were small, see Table VI. The Charpy upper shelf energy behavior, also shown in Table VI, was not entirely consistent with the transition temperature changes. For example, the decrease in the transition temperature of the weld metal may result from data scatter rather than a real change in the material property. However, the small changes which were measured are consistent with the level of fluence received by capsule basket No. 8.

Tensile tests were carried out on the irradiated materials in accordance with SwRI Procedure XI-MS-103-1 using a 22-kip capacity MTS Model 810 tester equipped with an Instron Catalog No. G-51-13A 2-in. strain gage extensometer and Hewlett Packard Model 7004B X-Y autographic recording equipment. Tensile tests were run at room temperature and 550°F. The results are presented in Table VII. Each tensile load-strain record and photographs of the tested specimens are included in Appendix B.

D. Check Chemical Analyses

Check chemical analyses were run on samples cut from the fracture end of selected tested Charpy specimens. All of the weld metal specimens were tested for copper and nickel content at SwRI using an X-ray fluorescent technique. One base plate and two weld specimens were then sent to Westinghouse Advanced Reactors Division Analytical Laboratory for complete analyses using gravimetric (Si), combustion (C and S), and ICP Plasma (remainder of elements) methods of analyses. The results are summarized in Table VIII. The differences in the copper and nickel results can be attributed to at least three factors:

(1) The ICP Plasma method measures the chemical content of the full volume of sample material, while the X-ray fluorescent method looks only at the surface of the sample.

(2) The 1 square centimeter cross sectional area available from the Charpy samples is smaller than desired for detecting small amounts of residuals with an X-ray fluorescent technique.

(3) The gamma activity of the sample increases the difficulty in accurately measuring the fluorescent peak.

TABLE V

CHARPY V-NOTCH IMPACT DATA ON SURVEILLANCE SPECIMENS REMOVED FROM QUAD CITIES UNIT 1

(Fluence = 5.5×10^{16}, E > 1 MeV)

Material	Specimen No.[(a)]	Test Temperature (°F)	Impact Energy (ft-lb)	Lateral Expansion (mil)	Shear (%)
Base	M1B	-80	8.0	6	0
	M1D	-40	16.5	16	5
	M1J	-20	42.5	35	10
	M1L	0	44.0	34	15
	M16	72	92.0	71	50
	M13	120	104.0	79	100
	M1K	160	109.5	81	100
	M12	200	106.0	84	100
ESHAZ	MPA	-80	5.5	6	0
	ML6	-40	20.5	18	5
	MJB	-30	36.0	31	5
	MJU	-20	41.5	36	10
	MKB	0	40.0	37	10
	MPD	72	117.0	55	100
	MJ1	120	107.0	88	100
	MLU	160	112.5	89	100
ESW	MBU	-80	2.0	2	0
	MAD	-40	2.5	5	0
	MA4	-20	36.0	31	5
	MBY	0	55.0	43	10
	M7L	72	79.0	65	50
	M6D	120	83.0	63	65
	MDC	160	93.0	65	90
	MD4	200	105.0	75	100

(a) Specimen identification code given in Reference 10.

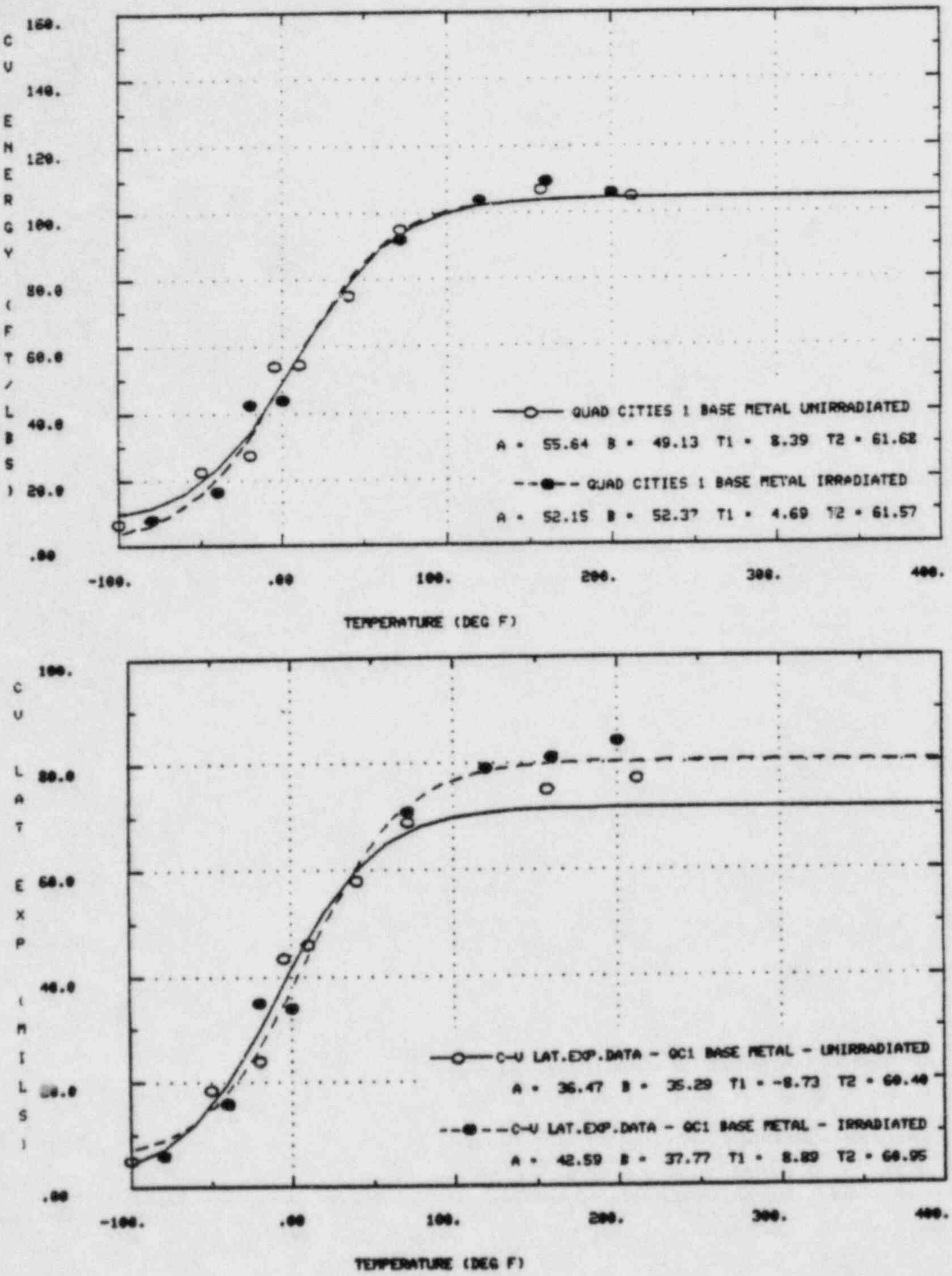


FIGURE 4. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES UNIT 1 SURVEILLANCE BASE PLATE

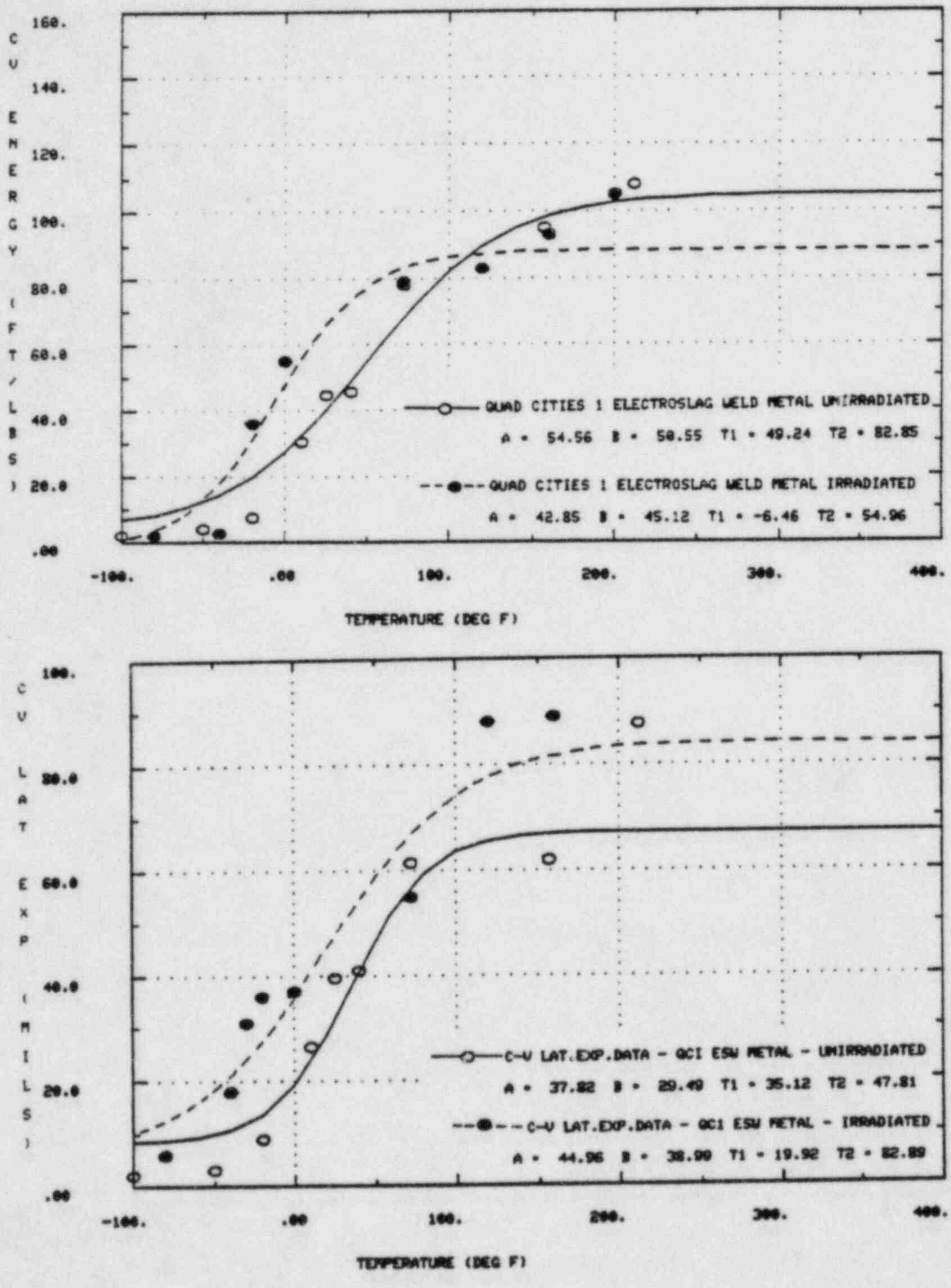
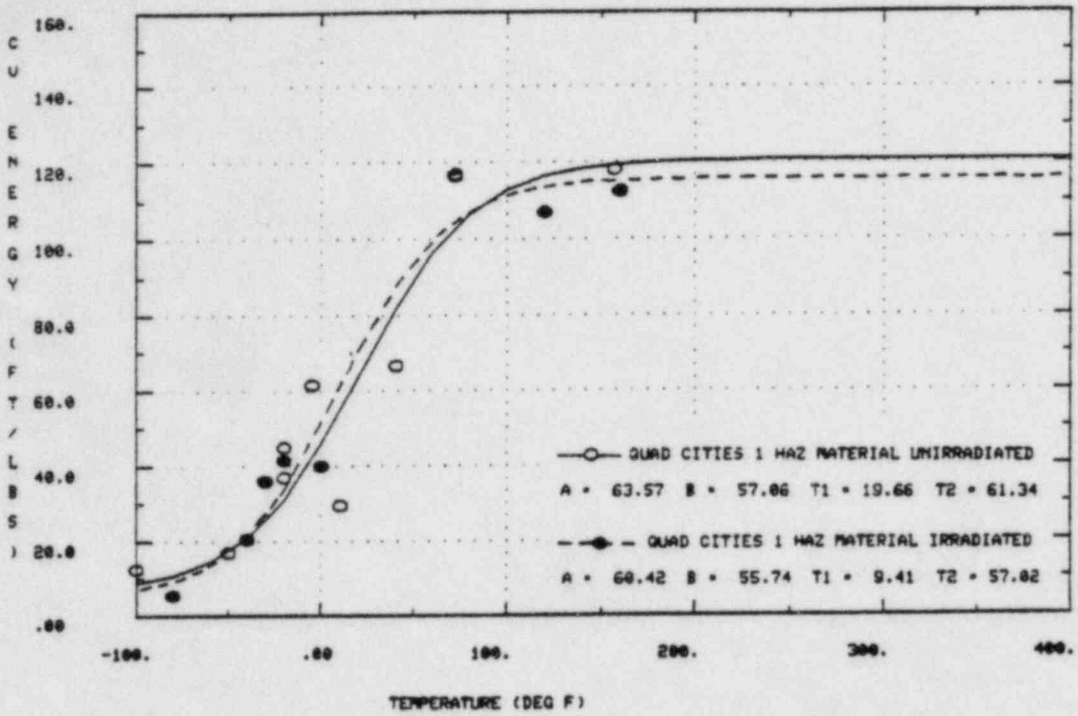


FIGURE 5. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES UNIT 1 SURVEILLANCE ELECTROSLAG WELD METAL



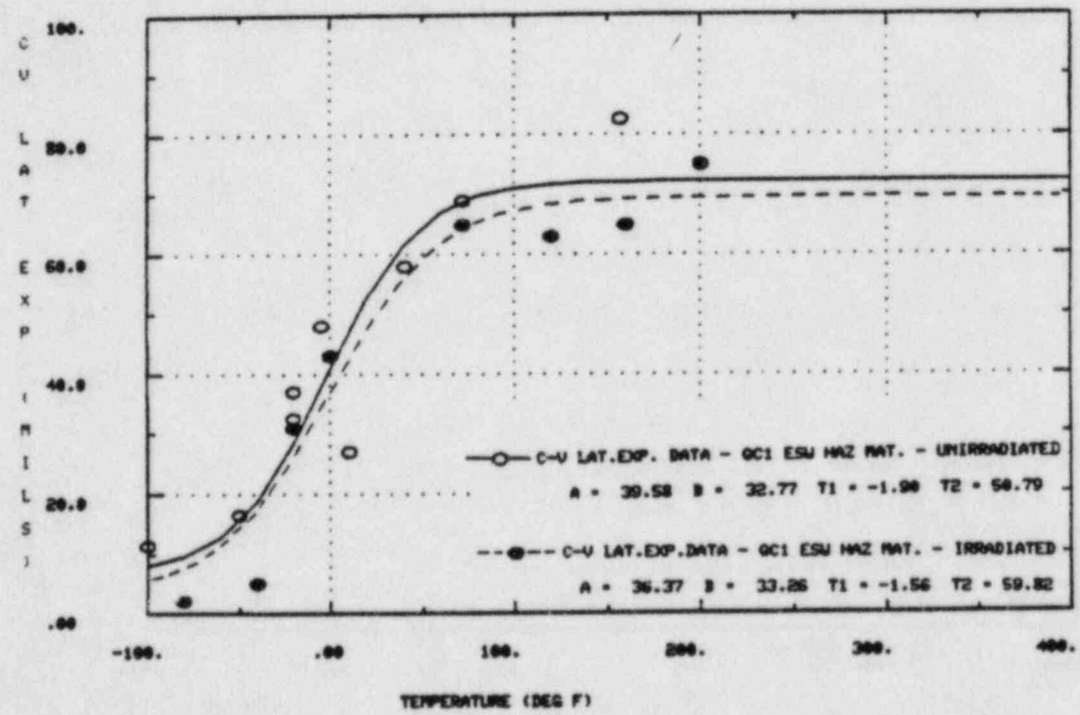


FIGURE 6. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES UNIT 1 SURVEILLANCE ELECTROSLAG WELD HEAT-AFFECTED ZONE MATERIAL

TABLE VI

EFFECT OF IRRADIATION ON THE CHARPY V-NOTCH
UPPER SHELF ENERGIES OF QUAD CITIES UNIT NO.1
VESSEL SURVEILLANCE MATERIALS BASKET NO. 8 (215°)
(Fluence = 5.5×10^{16}, E > 1 MeV)

Criterion(1)	Base Plate	ESW Weld Metal	ESW HAZ Material
Transition Temperature Shift(2)			
@ 50 ft-lb	1°F	-40°F	-6°F
@ 30 ft-lb	4°F	-28°F	-4°F
@ 35 mil	7°F	-32°F	5°F
ΔRT_{NDT}(3)	4°F	(4)	(4)
C_V Upper Shelf Drop	nil	3 ft-lb (3%)	6 ft-lb (5%)

(1) Refer to Figures 4-6

(2) C_V parameter determined by hyperbolic tangent fit where

$$C_V \text{ parameter} = A + B \tanh((T - T_1)/T_2)$$

(3) Transition temperature shift @ 30 ft-lb[1]

(4) Apparent negative shift in RT_{NDT}

TABLE VII

TENSILE PROPERTIES OF SURVEILLANCE MATERIALS
BASKET NO. 8 (215°), QUAD CITIES 1

(Fluence = 5.5×10^{16}, E > 1 MeV)

Test Material	Spec. No.[a]	Temp (°F)	0.2% YS (ksi)	UTS (ksi)	Fracture Load (lb)	Fracture Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	R.A. (%)
Base	PDE	72	71.3	93.5	2860	194	14.8	23.7	69.9
	PDC	550	58.3	82.7	2650	160	12.6	21.5	66.4
ESHAZ	PL5	72	65.2	84.3	2550	178	11.7	19.8	70.4
	PLJ	550	59.8	76.8	2530	160	10.2	17.1	67.2
ESW	PJ5	72	65.6	85.0	2670	178	14.1	24.9	69.5
	PJ7	550	58.0	78.7	2780	153	12.0	18.1	62.4

(a) Specimen identification code given in Reference 10.

TABLE VIII

CHECK CHEMICAL ANALYSIS RESULTS
BASKET NO. 8, QUAD CITIES 1

Specimen Identification[a]	Source[b]	Weight Percent of Element								
		C	S	P	Si	Cu	Ni	Cr	Mo	V
M16	W	.210	.010	.010	.16	.182	.48	.092	.45	<.001
	S					.25	.48			
MA4	W	.158	.006	.012	.05	.136	.25	.063	.45	<.001
	S					.19	.26			
MAD	W	.236	.004	.012	.08	.138	.26	.063	.45	<.001
	S					.19	.38			
M7L	S	-	-	-	-	.20	.30	-	-	-
MBU	S	-	-	-	-	.24	.33	-	-	-
M6D	S	-	-	-	-	.16	.39	-	-	-
MDC	S	-	-	-	-	.21	.40	-	-	-
MD4	S	-	-	-	-	.19	.33	-	-	-
MBY	S	-	-	-	-	.21	.40	-	-	-
MPD	S	-	-	-	-	.27	.56	-	-	-

(a) Cut from tested C_V specimen, see Table V for material I.D.
(b) W = Westinghouse Atomic Power Division; S = Southwest Research Institute.

V. HEATUP AND COOLDOWN LIMIT CURVES FOR OPERATION OF QUAD CITIES UNIT NO. 1

Unit No. 1 is one of four similar 251-in. I.D. boiling water reactors (Dresden Units 2 & 3 and Quad Cities Units 1 & 2) operated by Commonwealth Edison Company. Each of the similar units has been provided with a reactor vessel material surveillance program as required by 10CFR 50, Appendix H [1].

Based on the capsule analyses, heatup and cooldown limit curves for Level A and B service and for up to 10 EFPY of operation have been computed. The limit curves developed are a generic set for all four vessels listed above. The curves, given in Figure 7, are based on the worst case for vessel fluence rate, copper/nickel content, and initial RT_{NDT}. They were computed in accordance with Regulatory Guide 1.99, Revision 1 [8] and NUREG-75/087 [15] using the following bases:

1/4T RT_{NDT}	= 60°F
3/4T RT_{NDT}	= 49°F
Cooling/heating rate	= 100°F/hr.
Vessel inner radius	= 125.5 in.
Vessel outer radius	= 131.8 in.
Operating pressure	= 1030 psig
Initial temperature	= 60°F
Final temperature	= 550°F
Effective coolant flow rate	= 1.08×10^8 lbm/hr.
Effective flow area	= 96.0 ft^2
Effective hydraulic diameter	= 24.0 in.

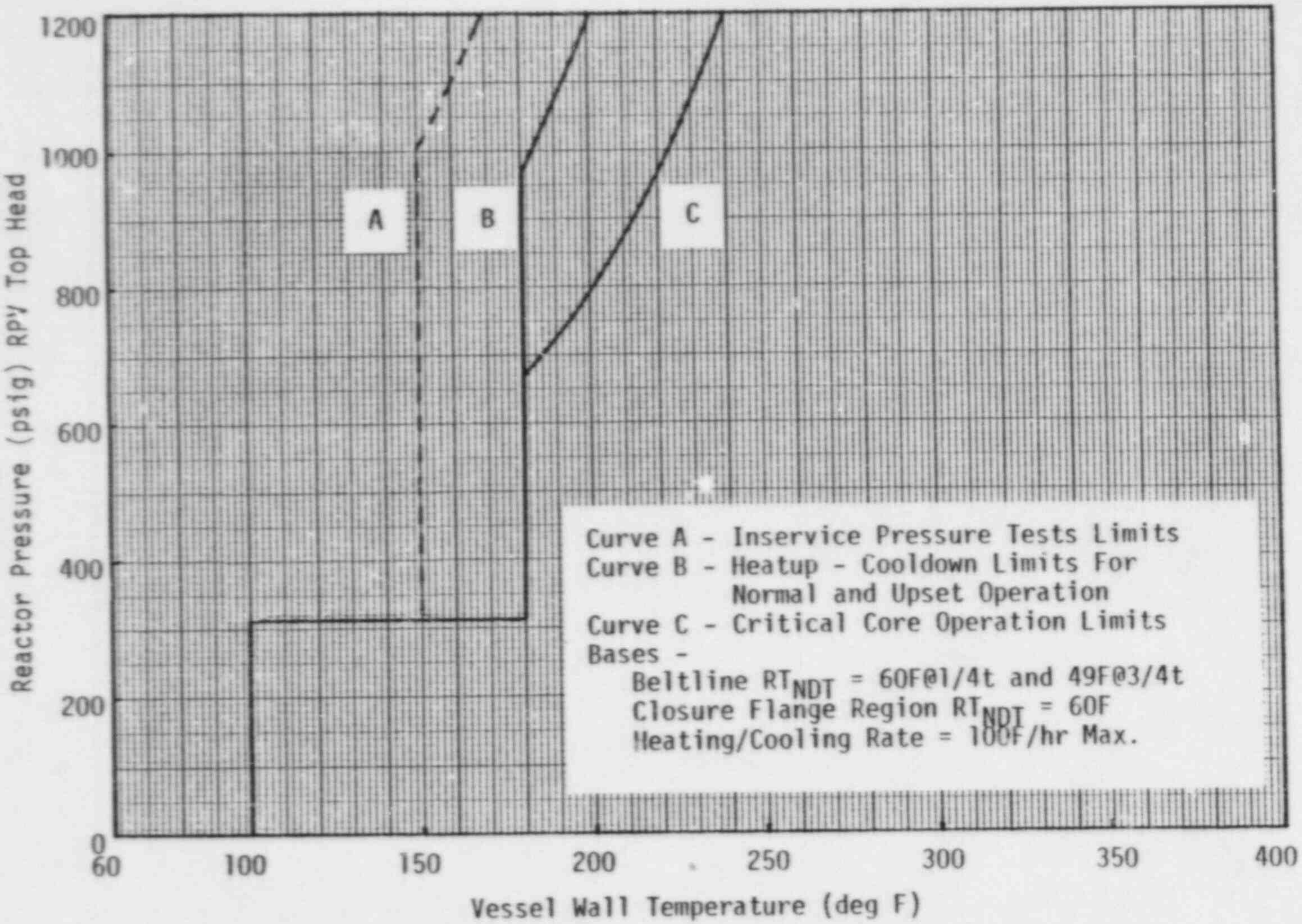


FIGURE 7. PRESSURE-TEMPERATURE LIMITS FOR LEVELS A & B SERVICE FOR UP TO 10 EFPY

VI. REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1980 Edition.

3. ASTM E 208-69, "Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," 1975 Annual Book of ASTM Standards.

4. L. E. Steel and C. Z. Serpan, Jr., "Analysis of Reactor Vessel Radiation Effects Surveillance Programs," ASTM STP 481, December 1970.

5. L. E. Steele, "Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels," International Atomic Energy Agency, Technical Reports Series No. 163, 1975.

6. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1980 Edition.

7. J. F. Perrin, R. A. Wullaert, G. R. Odette, and P. M. Lombrozo, "Physically Based Regression Correlations of Embritt;ement Data From Reactor Pressure Vessel Surveillance Programs", EPRI NP-3319, January 1984.

8. Regulatory Guide 1.99, Revision 1, Office of Standards Development, U.S. Nuclear Regulatory Commission, April 1977.

9. ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," 1975 Annual Book of ASTM Standards.

10. J. P. Higgins and F. A. Brandt, "Mechanical Property Surveillance of General Electric BWR Vessels," NEDO-10115, July 1969.

11. J. S. Perrin et al, "Quad Cities Nuclear Plant Unit No. 1 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 2 and Capsule Basket No. 3 ," Battelle Columbus Laboratories Report, March 1, 1975.

12. T. R. Mager et al, "Analysis of the Third Capsule from the Commonwealth Edison Company Quad Cities Unit 1 Nuclear Plant Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation WCAP- 9920, Electric Power Research Institute Research Project 1021-3 Topical Report, August 1981.

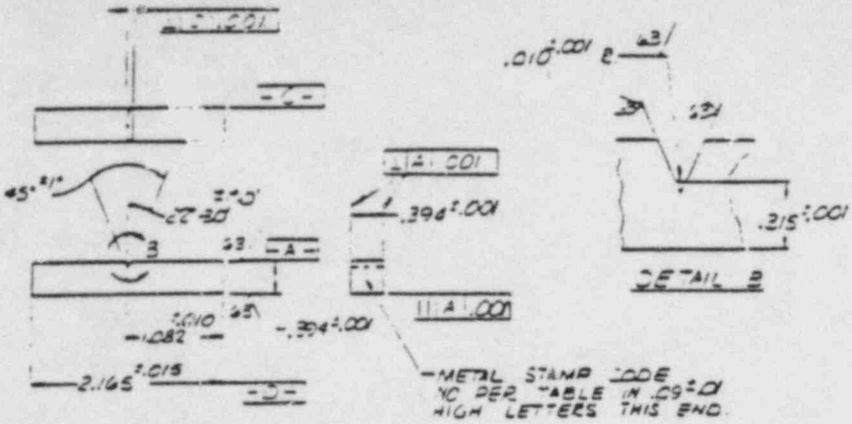
13. J. S. Perrin and L. M. Lowry, "Quad Cities Nuclear Plant Unit No. 1 and Unit No. 2 Reactor Pressure Vessel Surveillance Programs: Unirradiated Mechanical Properties," Battelle Columbus Laboratories Final Report to Commonwealth Edison Comapny, February 15, 1975.

14. E. B. Norris, "Spectral Analysis of a BWR Vessel," Proceedings of the Fourth ASTM-EURATOM Symposium on Reactor Dosimetry," NUREG/CP-0029, Vol. 2, pp. 1043-1050.

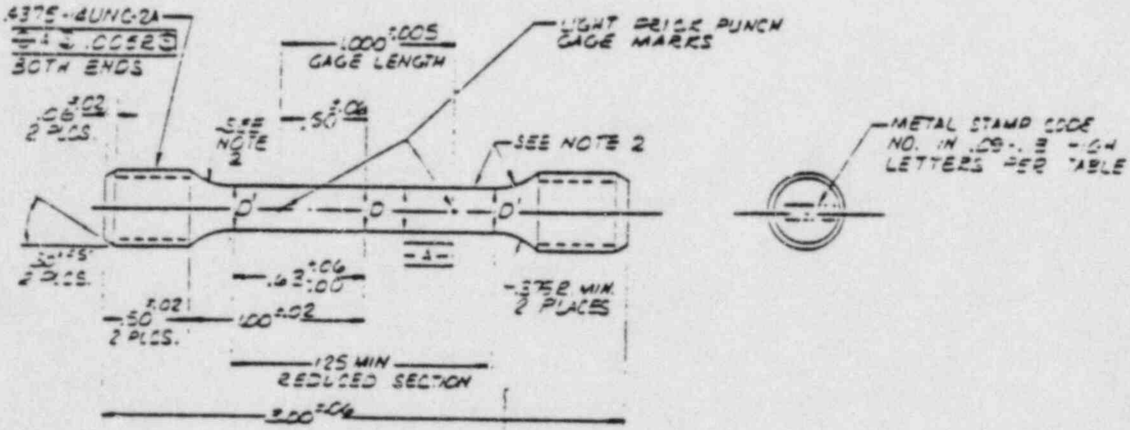
15. US NRC Standard Review Plan, NUREG-75/087, Section 5.3.2, Pressure-Temperature Limits, November 24, 1975.

APPENDIX A

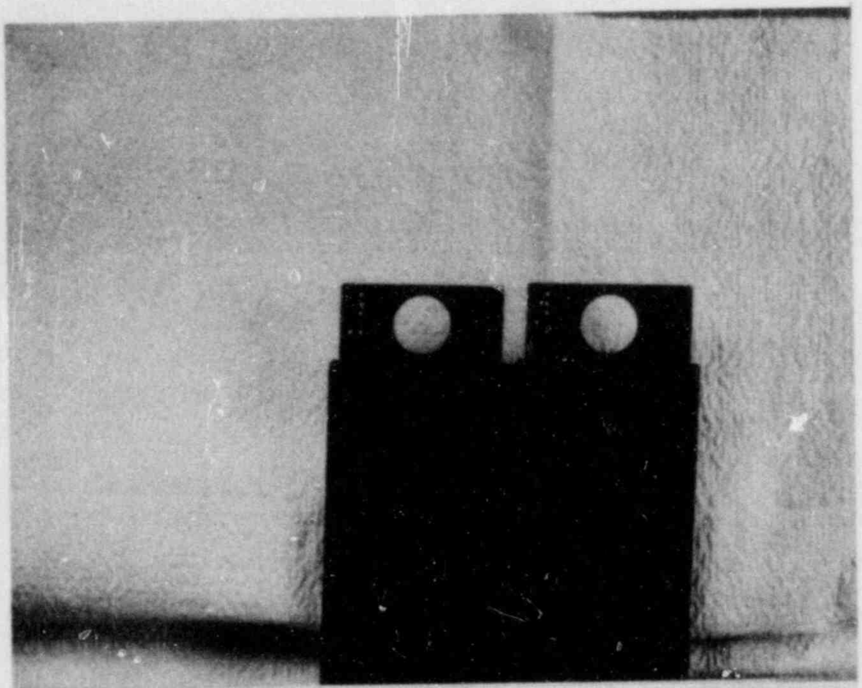
SURVEILLANCE SPECIMEN DRAWINGS AND CAPSULE PHOTOGRAPHS

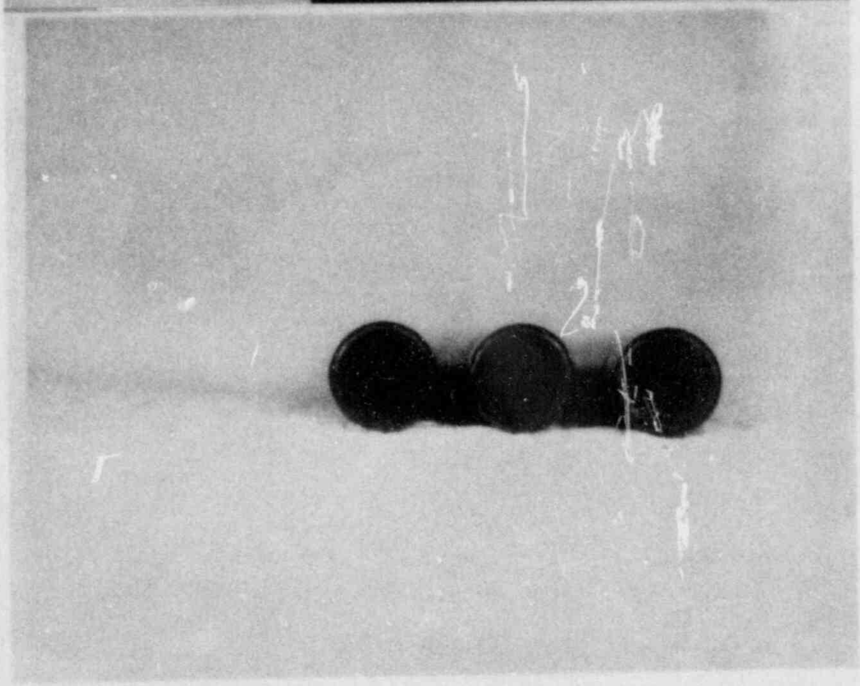


PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 15837275. NO TO RUN SERIALLY
4	BASE	21A1109	5İ1 TO 5ŚY
5	WELD (SUB-ARC)		6İ1 TO 6ŚY
6	WELD (ELECTROSLAG)		5Ġ1 TO 5ĖY
7	HAZ (SUB-ARC)		6Ġ1 TO 6ĊY
8	HAZ (ELECTROSLAG)		5J1 TO 5ÜY



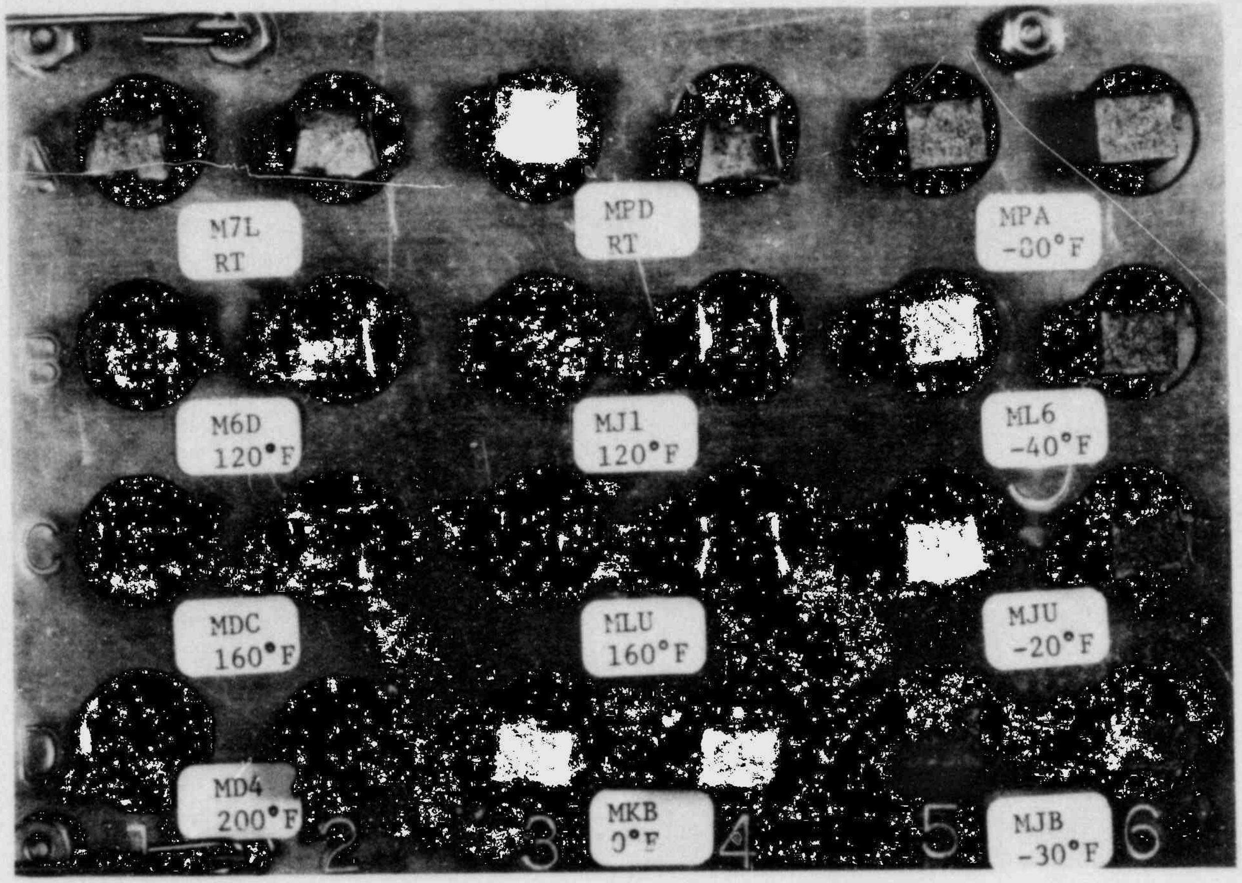
PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO. PER 5837275. NO. TO RUN SERIALLY
4	BASE	21A1109	6Ċ1 TO 6ĖY
5	WELD (SUB-ARC)		6Ṗ1 TO 6Ṫ1
[illegible]	WELD (ELECTROSLAG)		6J1 TO 6ĶY
[illegible]	HAZ (SUB-ARC)		6Ü1 TO 6ṪY
8	HAZ (ELECTROSLAG)		6L1 TO 6MY

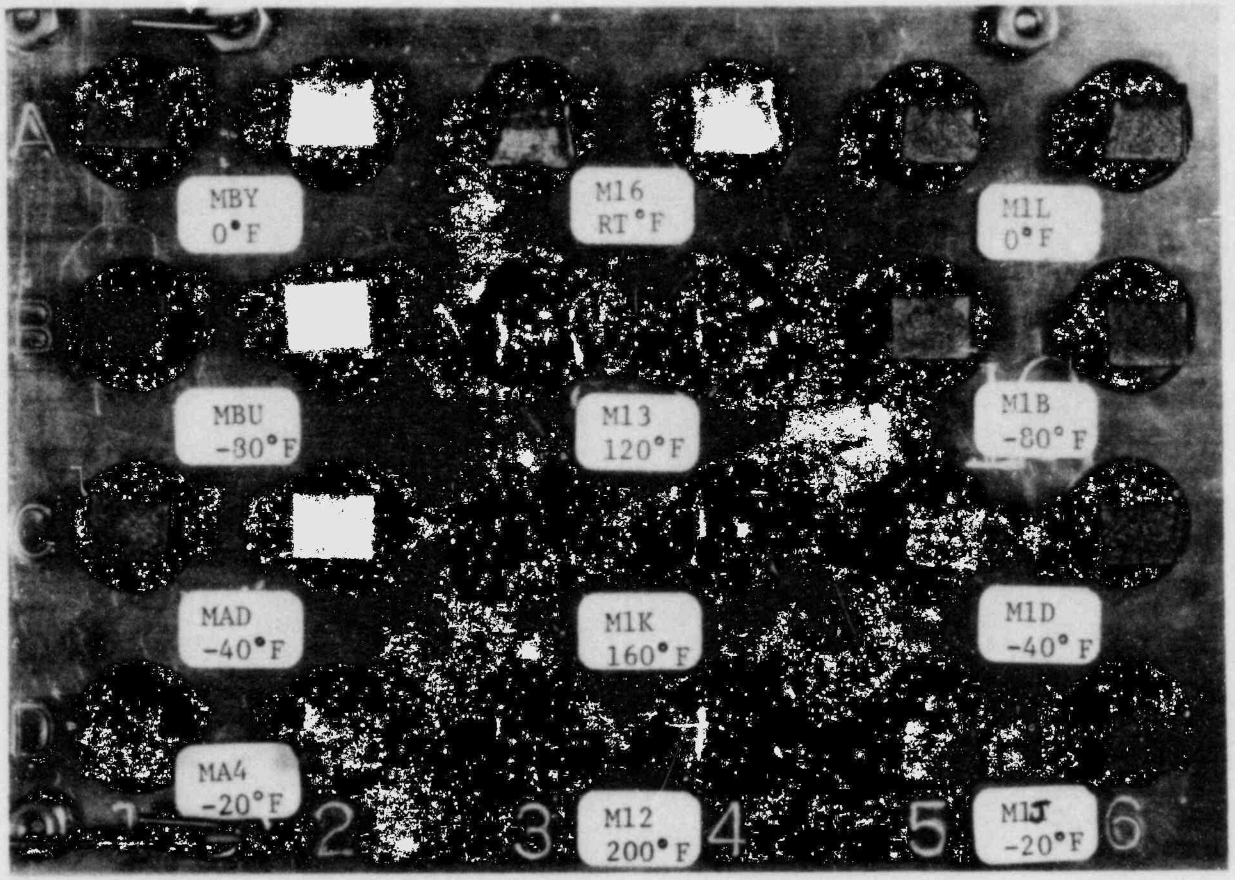




APPENDIX B

PHOTOGRAPHS OF TESTED SPECIMENS AND TENSILE TEST RECORDS


M7L
RT
MPD
RT
MPA
-80°F
M6D
120°F
MJ1
120°F
ML6
-40°F
MDC
160°F
MLU
160°F
MJU
-20°F
MD4
200°F
MKB
0°F
MJB
-30°F


A
B
C
D
MBY 0°F
M16 RT°F
M1L 0°F
MBU -80°F
M13 120°F
M1B -80°F
MAD -40°F
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M12 200°F
M1J -20°F
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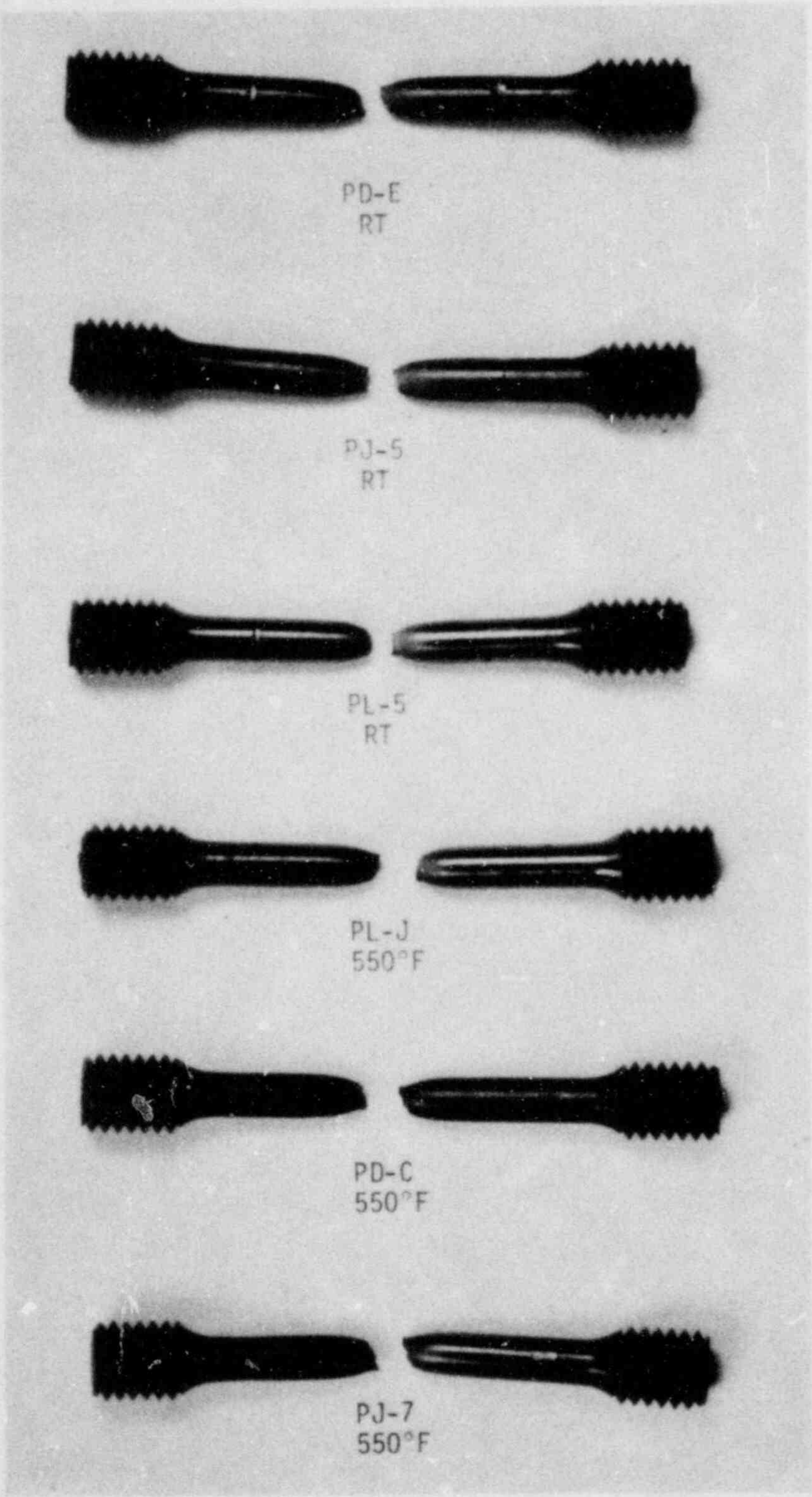
2
3
4
5
6


PD-E
RT
PJ-5
RT
PL-5
RT
PL-J
550°F
PD-C
550°F
PJ-7
550°F

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 1 Est. U.T.S. ______ psi Project No. 06-7857-001

Spec. No. PDE Initial G.L. 1.022 in. Machine No. 22KIP MTS

Temperature 72 °F Initial Dia. .250 in. Date 5-18-84 ✓

Strain Rate .005 in/in/min Initial Thickness ______ in. Initial Area .04908 in^2

Initial Width ______ in.

Top Temperature ______ °F Maximum Load 4590 lb

Bottom Temperature ______ °F 0.2% Offset Load 3500 lb

Final Gage Length 1.264 in. 0.02% Offset Load ______ lb

Final Diameter 0.137 in. Upper Yield Point ______ lb

Final Area 0.01474 ✓ in.^{2}

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 93,507 ✓ psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 71,301 ✓ psi

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ______ psi

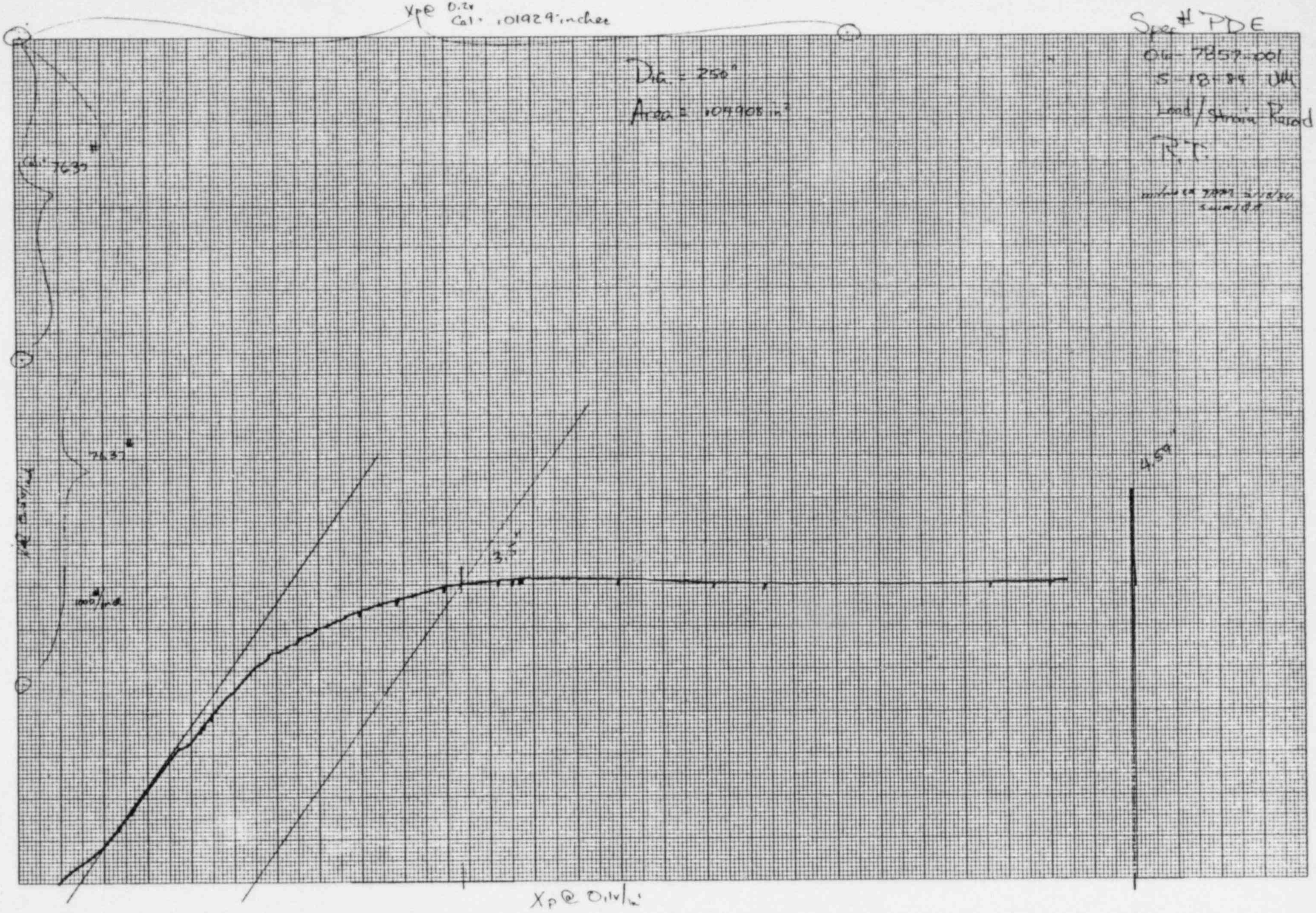
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

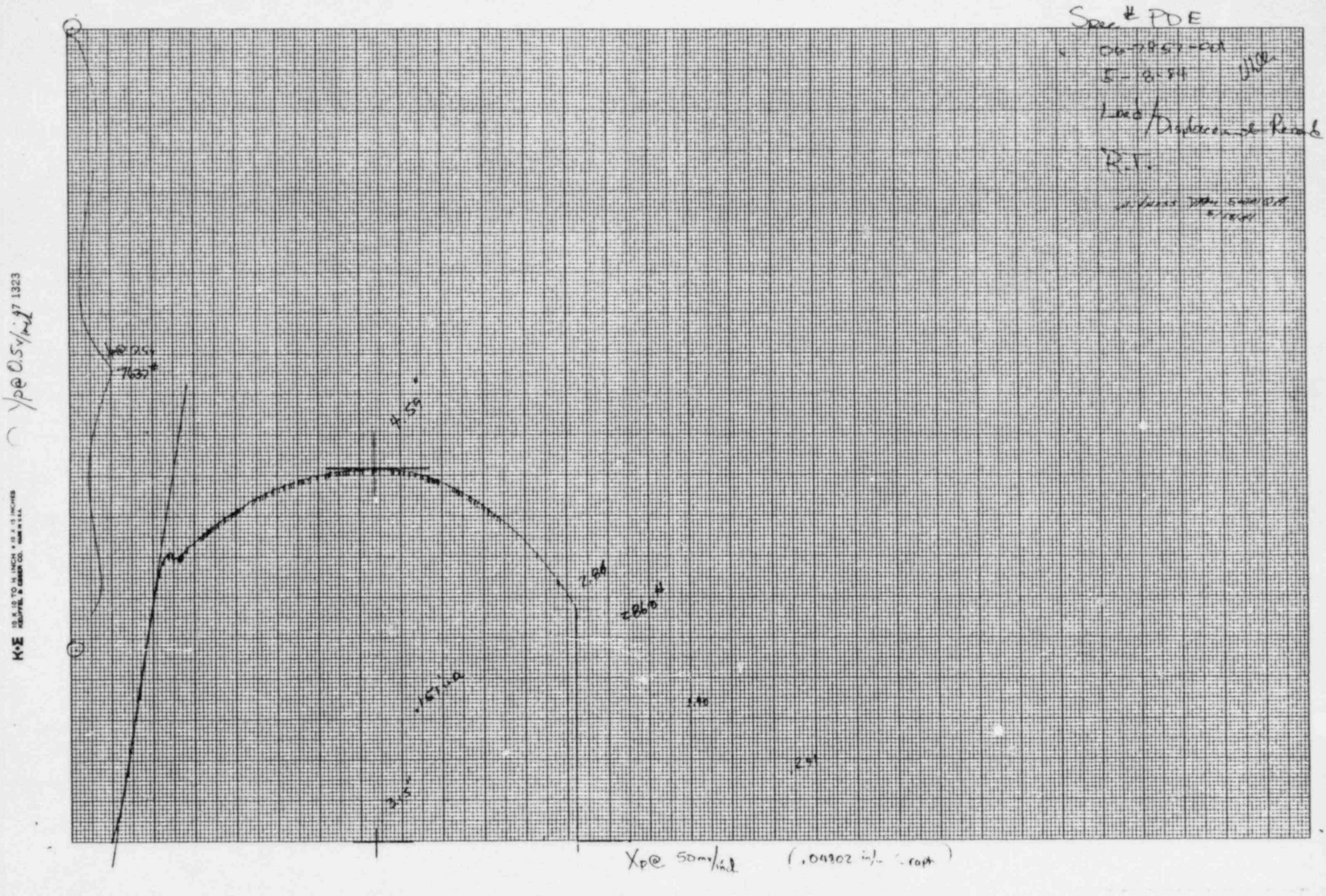
% Elongation = $\frac{\text{Final G.L. - Initial G.L.}}{\text{Initial G.L.}}$ x 100 = 23.7 ✓ %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 69.9 ✓ %

Signature: V. Aaron

Checked [signature]


Spec # PDE
04-7857-001
5-18-84 UM
Load/Strain Rated
R.T.
Dia. = .250"
Area = .04908 in²
Xp @ 0.2%
Cal = .01929 inches
Cal = 7637#
7637#
1000#/in
3.3#
4.59#
Xp @ 0.1v/in
Vp @ 1.0v/in



Spec # PDE
5-8-84
Load/Displacement Record
R.T.
7637#
4.59
2.86
2860#
3.5
.167 in.
Xp@ 50mv/inch
(.04902 in/in graph)
Yp@ 0.5v/inch
K+E 10 X 10 TO ½ INCH • 10 X 15 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A. 47 1323

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 2 | Est. U.T.S. ______ psi | Project No. 06-7857-001

Spec. No. PJ5 | Initial G.L. 1.020 in. | Machine No. 22 KIP MTS

Temperature 72 °F | Initial Dia. .250 in. | Date 5-18-84

Strain Rate .005 in/in/min | Initial Thickness ______ in. | Initial Area .04908 in²

Initial Width ______ in.

Top Temperature ______ °F | Maximum Load 4170 lb

Bottom Temperature ______ °F | 0.2% Offset Load 3220 lb

Final Gage Length 1.274 in. | 0.02% Offset Load ______ lb

Final Diameter .138 in. | Upper Yield Point ______ lb

Final Area .014957 in.²

U.T.S. = Maximum Load / Initial Area = 84,963 psi

0.2% Y.S. = 0.2% Offset Load / Initial Area = 65,607 psi

0.02% Y.S. = 0.02% Offset Load / Initial Area = ______ psi

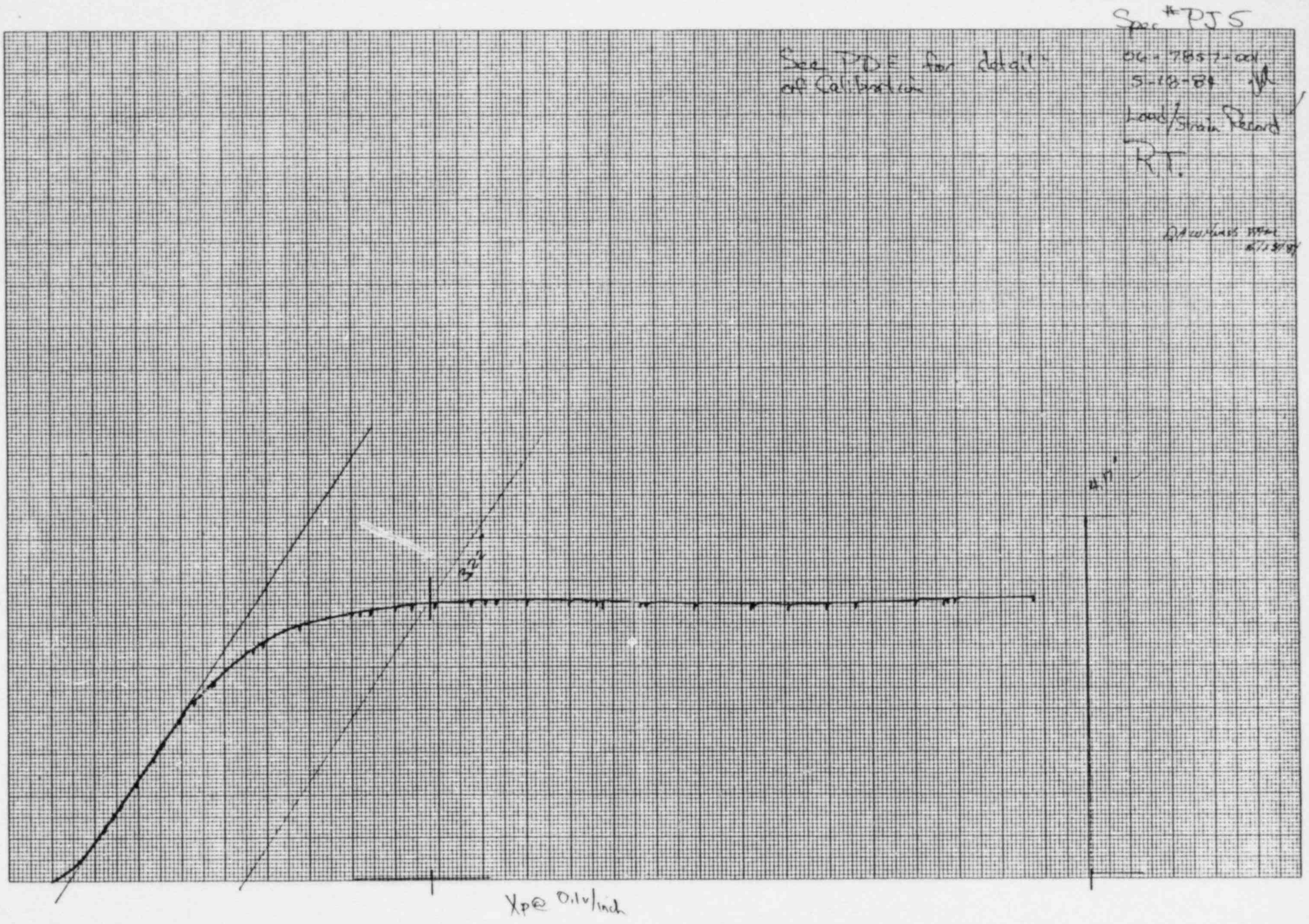
Upper Y.S. = Upper Yield Point / Initial Area = ______ psi

% Elongation = (Final G.L. - Initial G.L.) / Initial G.L. x 100 = 24.9 %

% R.A. = (Initial Area - Final Area) / Initial Area x 100 = 69.5 %

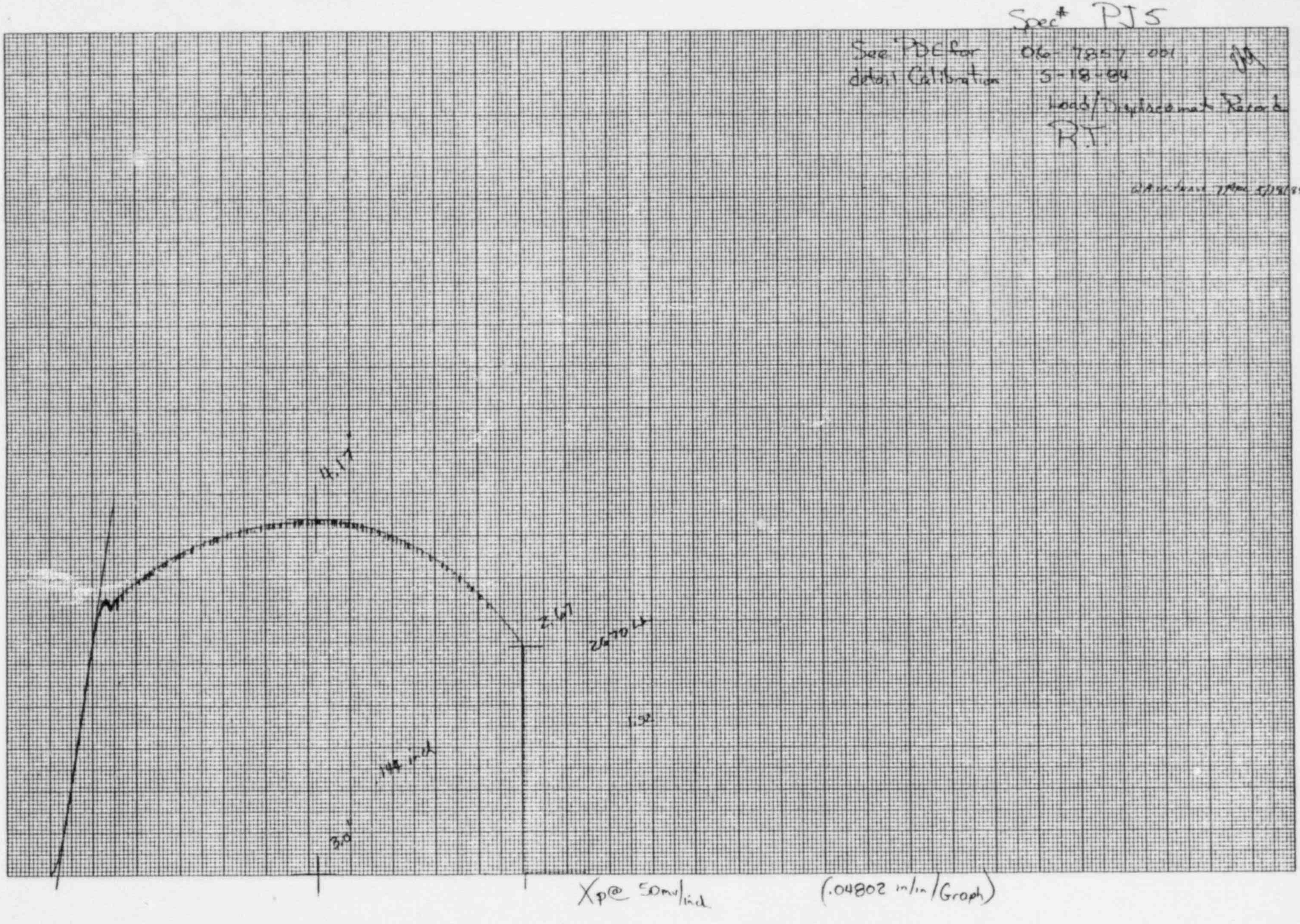
Signature: D. Aaron

Checked [signature]



Yp@ 0.5v

K+E 10 X 10 TO ½ INCH • 10 X 15 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A. 47 1323


Spec# PJ5
See PDC for
detail Calibration
06-7857-001
5-18-84
Load/Displacement Record
R.T.
4.17
2.67
1.92
3.0
Xp@ 50mv/inch
(.04802 in/in/Graph)
Yp@ 0.5v

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 3 Est. U.T.S. ________ psi Project No. 06-7857-001

Spec. No. PL5 Initial G.L. 1.019 in. Machine No. 22 Kip MTS

Temperature 72 °F Initial Dia. .248 in. Date 5-18-84

Strain Rate .005 in/in/min Initial Thickness ______ in. Initial Area .048305 in²

Initial Width ________ in.

Top Temperature ________ °F Maximum Load 4070 lb

Bottom Temperature ________ °F 0.2% Offset Load 3150 lb

Final Gage Length 1.221 in. 0.02% Offset Load ________ lb

Final Diameter .135 in. Upper Yield Point ________ lb

Final Area .014314 in.²

U.T.S. = Maximum Load / Initial Area = 84,259 psi

0.2% Y.S. = 0.2% Offset Load / Initial Area = 65,211 psi

0.02% Y.S. = 0.02% Offset Load / Initial Area = ________ psi

Upper Y.S. = Upper Yield Point / Initial Area = ________ psi

% Elongation = (Final G.L. - Initial G.L.) / Initial G.L. x 100 = 19.8 %

% R.A. = (Initial Area - Final Area) / Initial Area x 100 = 70.4 %

Signature: D. Aaron

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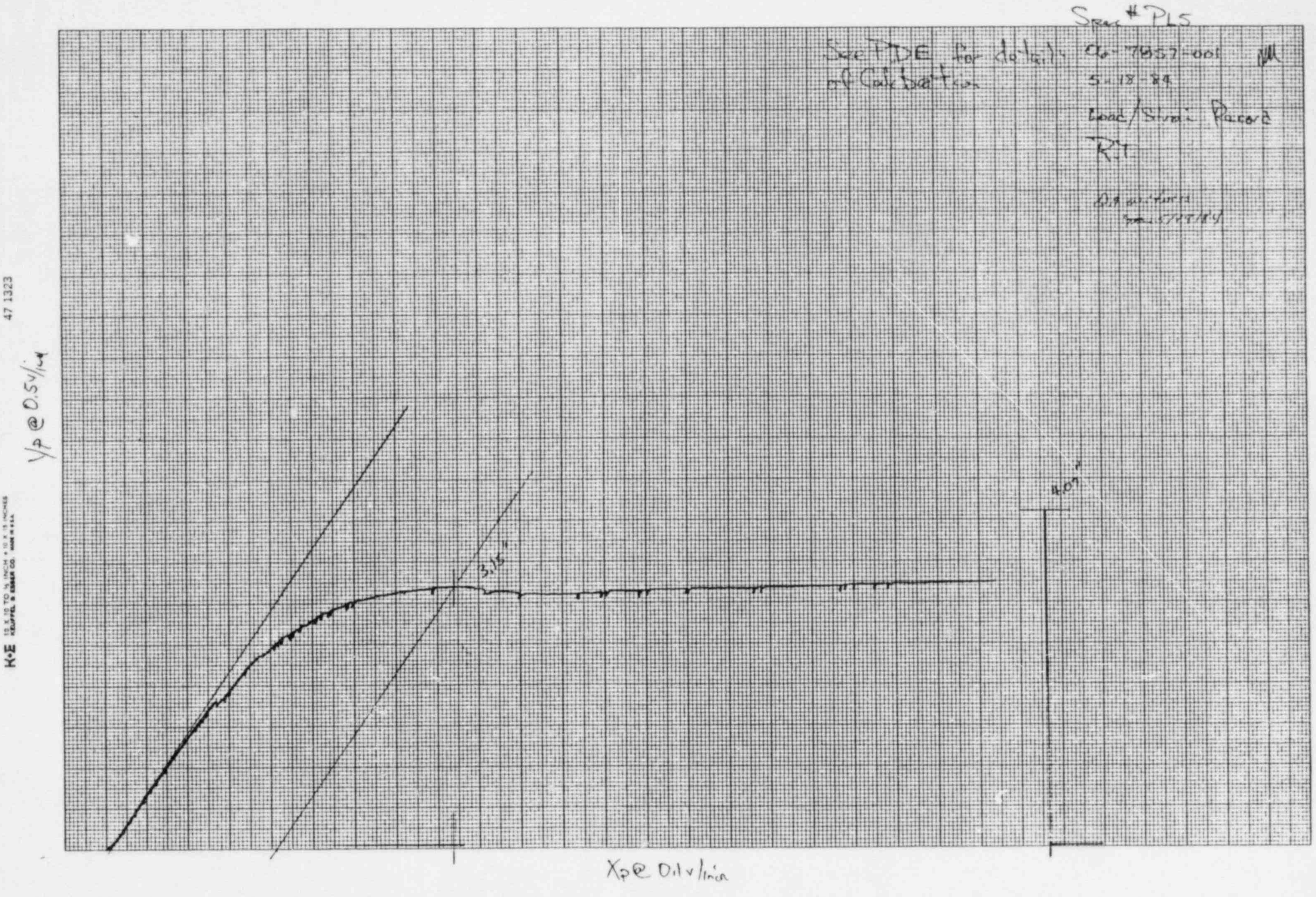
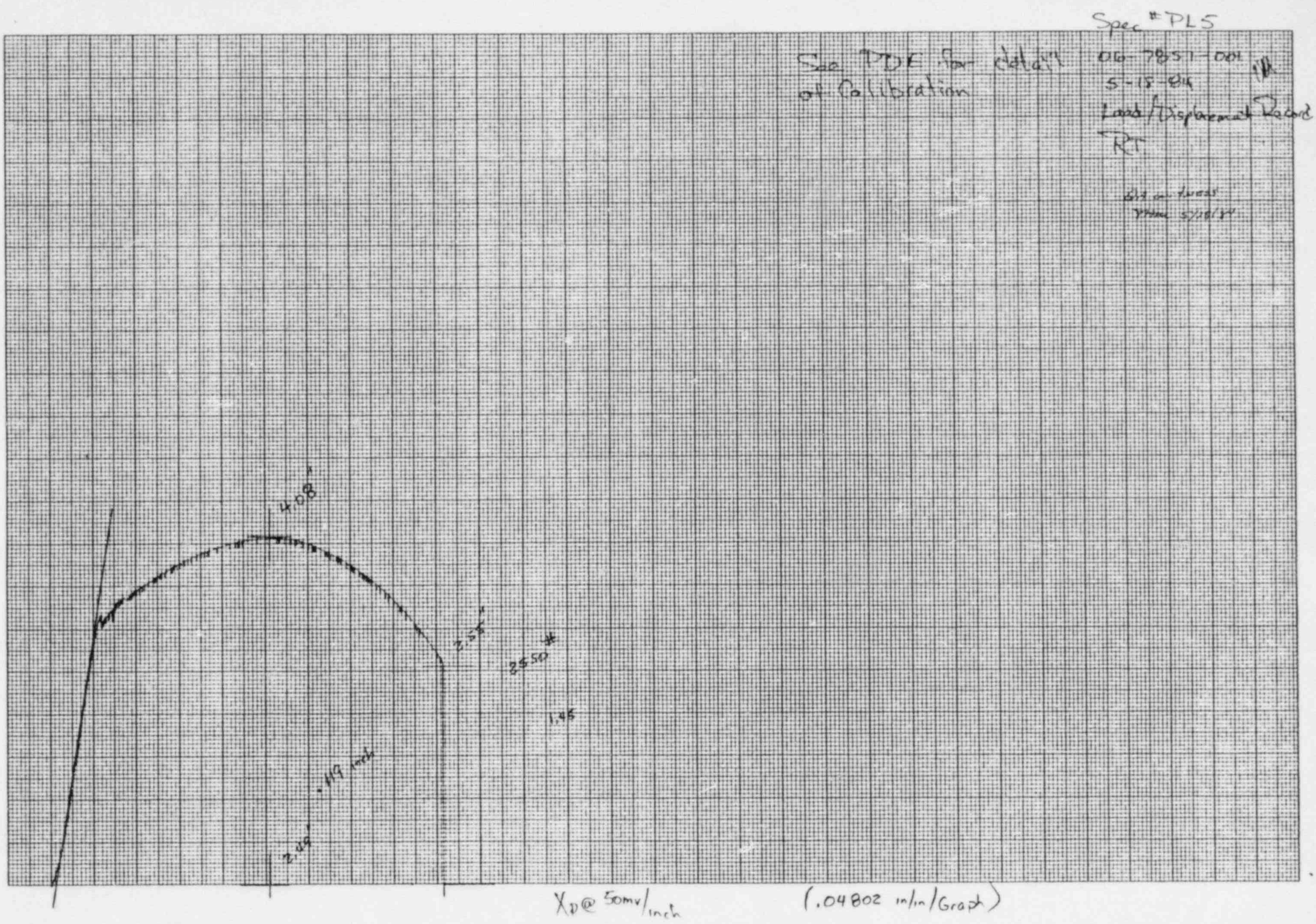


K+E 10 X 10 TO ½ INCH • 10 X 15 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

47 1323


Spec #PL5
06-7851-001
5-15-84
Load/Displacement Record
RT
See PDE for detail of Calibration
4.08
3.55
3550 #
1.45
.119 inch
2.09
Y_P @ 0.5/inch
X_D @ 50mv/inch
(.04802 in/in/Graph)

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 4
Spec. No. PLJ
Temperature 550 °F
Strain Rate .005 in/in/min

Est. U.T.S. ______ psi
Initial G. L. 1.025 in.
Initial Dia. .248 in.
Initial Thickness ______ in.
Initial Width ______ in.

Project No. 06-7857-001
Machine No. 22 KIP MTS
Date 5-19-84
Initial Area .048305 in²

Top Temperature ______ °F
Bottom Temperature ______ °F
Final Gage Length 1.200 in.
Final Diameter .142 in.
Final Area .015837 in.²

Maximum Load 3,710 lb
0.2% Offset Load 2890 lb
0.02% Offset Load ______ lb
Upper Yield Point ______ lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 76,804 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 59,828 psi

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ______ psi

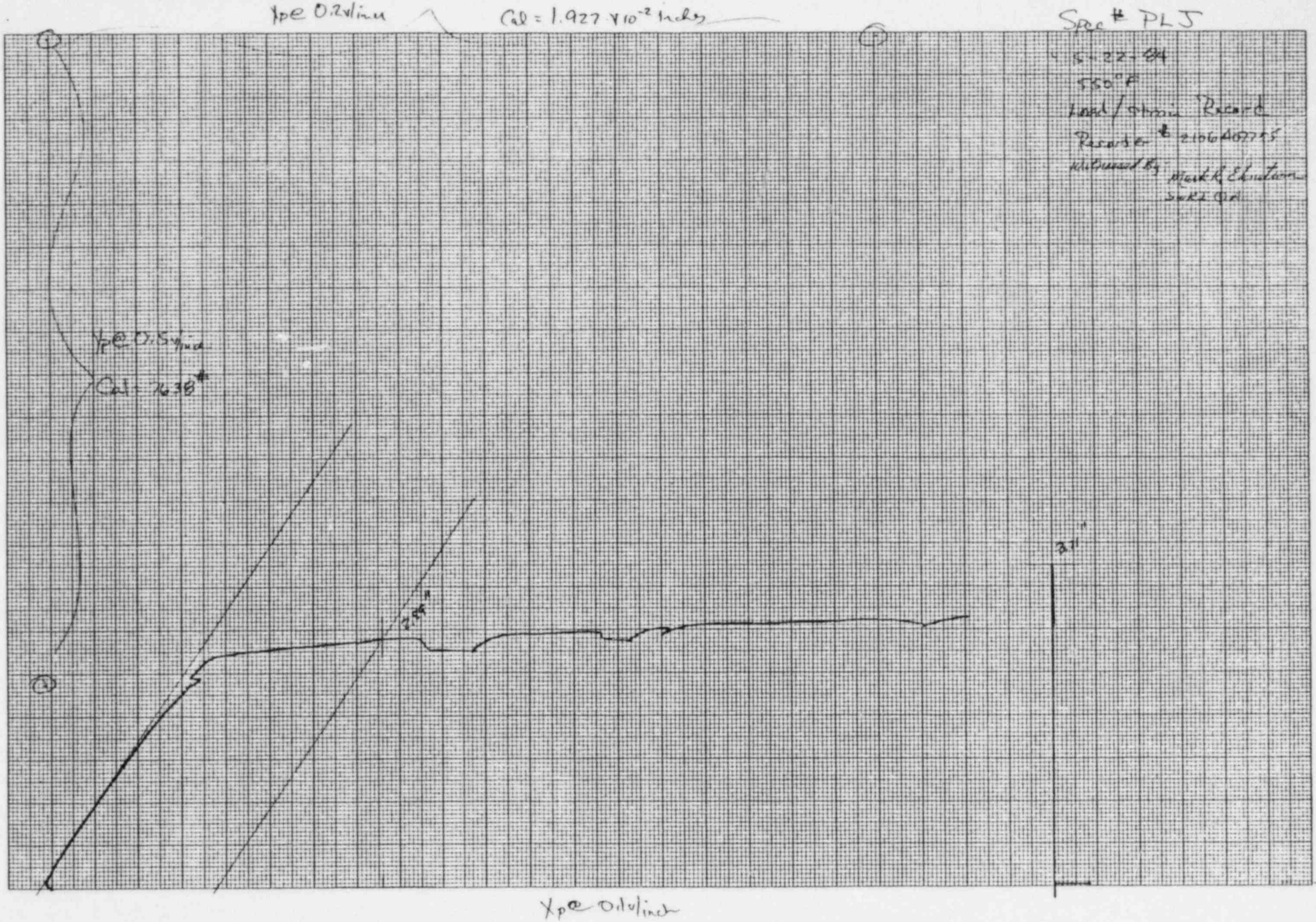
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 17.1 %

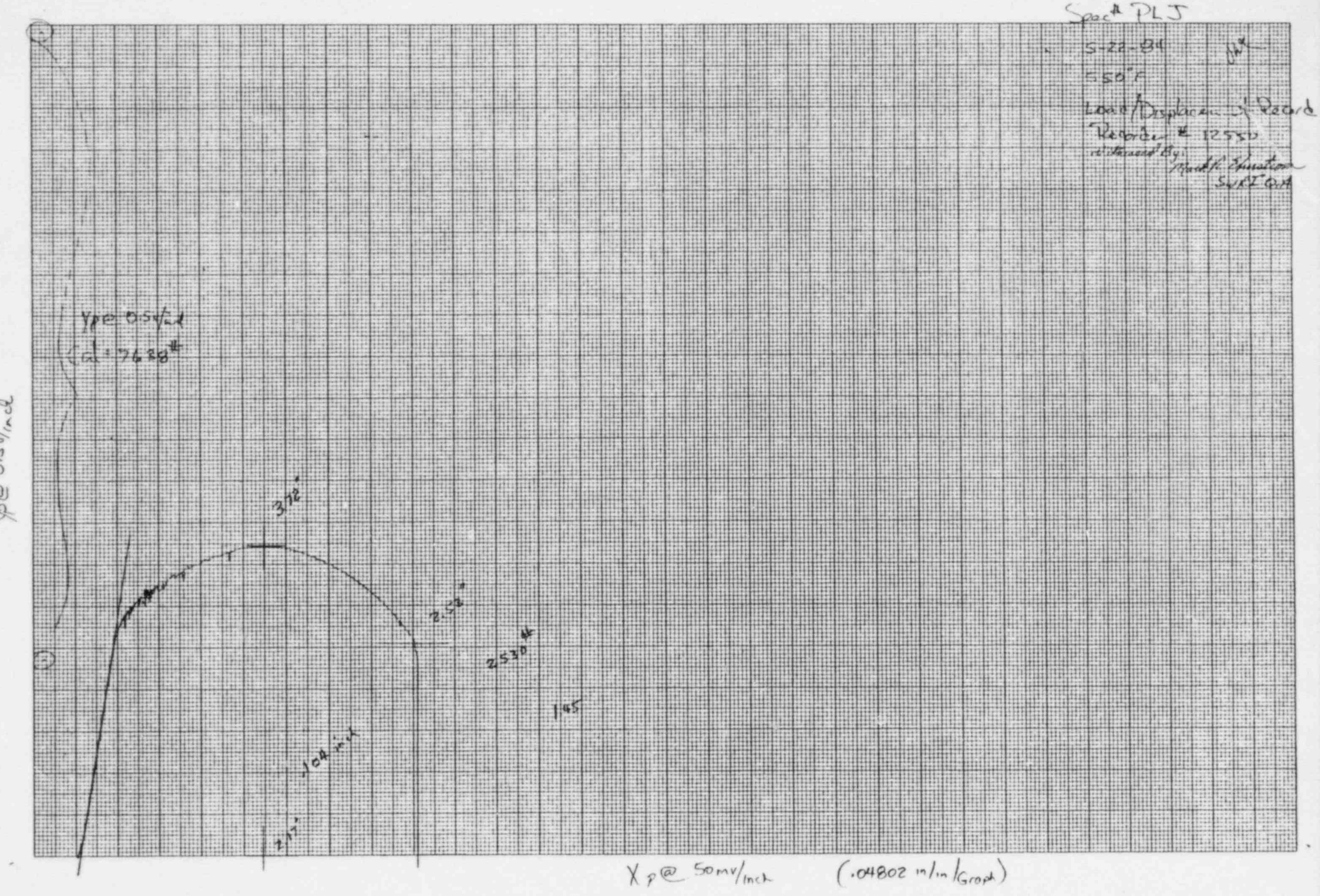
% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 67.2 %

Signature: W. Aaron

Checked [signature]



Xp@ 0.2v/inch
Cal = 1.927 x10^-2 inches
Spec # PLJ
5-22-84
550°F
Load/Strain Record
Recorder # 2106A07755
Witnessed By
Mark R. Ehnstrom
SwRI QA
Yp@ 0.5v/inch
Cal = 7638#
2.88#
3.7"
Yp@ 0.5v/inch
Xp@ 0.1v/inch
47 1323
K+E


Spec# PLJ
5-22-84
550°F
Load/Displacement Record
Recorder # 12550
Witnessed By:
SwRI QA
Yp @ 0.5v/inch
Cal = 7638#
3.72"
2.52"
2530#
1.45
.104 inch
2.17"
Xp @ 50mv/inch
(.04802 in/in/Graph)
Yp @ 0.5v/inch

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 5 | Est. U.T.S. ______ psi | Project No. 06-7857-001

Spec. No. PDC | Initial G. L. 1.020 in. | Machine No. 22 KIP MTS

Temperature 550 °F | Initial Dia. .250 in. | Date 5-19-84

Strain Rate .005 in/in/min | Initial Thickness ______ in. | Initial Area .049087 in.^{2}

Initial Width ______ in.

Top Temperature ______ °F | Maximum Load 4060 lb

Bottom Temperature ______ °F | 0.2% Offset Load 2860 lb

Final Gage Length 1.239 in. | 0.02% Offset Load ______ lb

Final Diameter .145 in. | Upper Yield Point ______ lb

Final Area .016513 in.^{2}

U.T.S. = Maximum Load / Initial Area = 82,710 psi

0.2% Y.S. = 0.2% Offset Load / Initial Area = 58,264 psi

0.02% Y.S. = 0.02% Offset Load / Initial Area = ______ psi

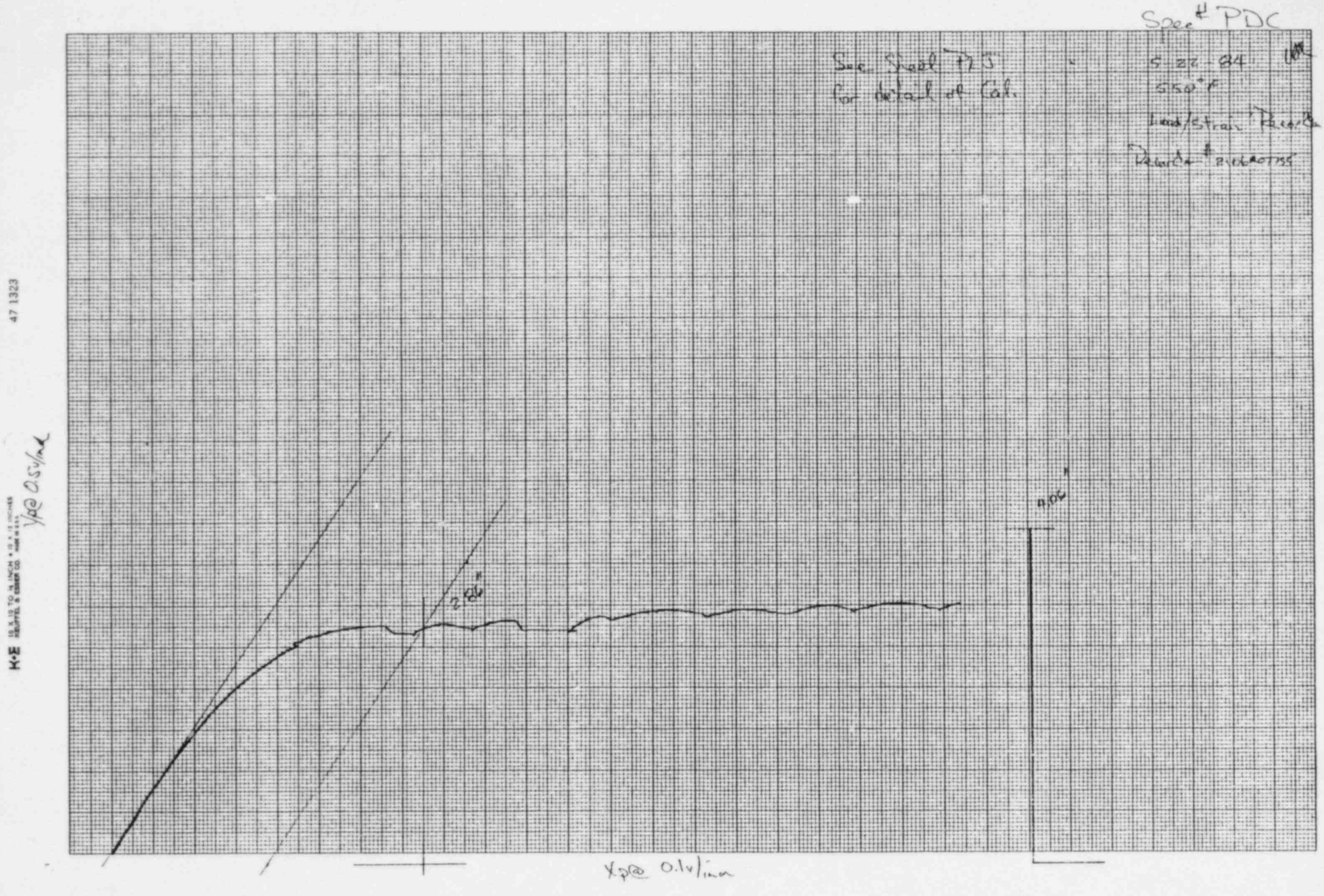
Upper Y.S. = Upper Yield Point / Initial Area = ______ psi

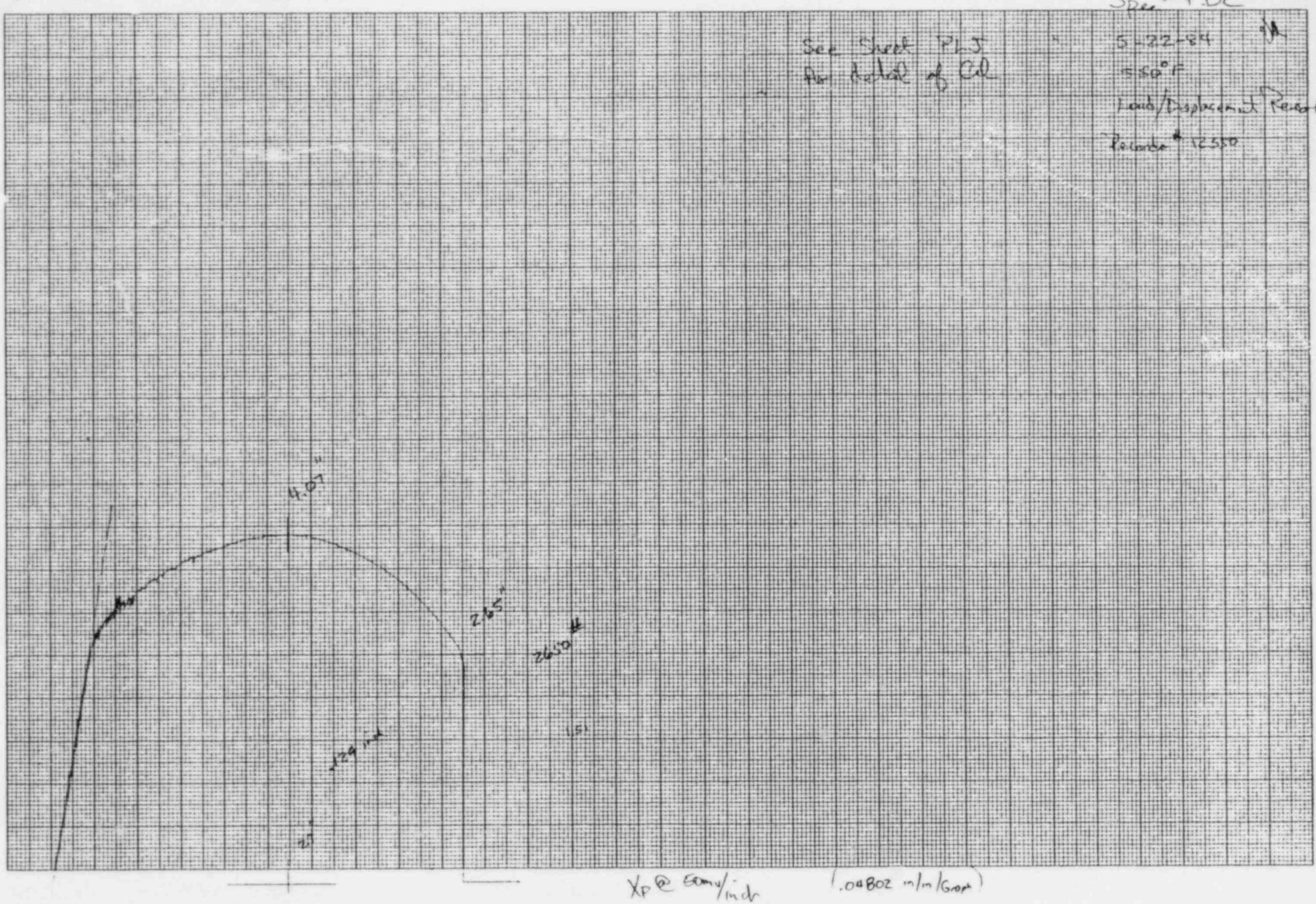
% Elongation = (Final G.L. - Initial G.L.) / Initial G.L. x 100 = 21.5 %

% R.A. = (Initial Area - Final Area) / Initial Area x 100 = 66.4 %

Signature: S. Aaron

Checked E. Di [illegible]


Spec # PDC
See Sheet P7 J
for detail of Cal.
5-22-84
550°F
Load/Strain Recorder
2.86"
4.06"
X@ 0.1v/inch
Y@ 0.5v/inch
K+E 10 X 10 TO ½ INCH
KEUFFEL & ESSER CO.
47 1323



Spec # PDC
5-22-84
550°F
Load/Displacement Record
Recorder # 12550
See Sheet PLJ
for detail of CL
4.07"
2.65"
2650#
.124 ind
.151
.25"
Xp @ 50mv/inch
(.04802 in/in/Graph)
Yp @ 0.5v/inch

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 6 Est. U.T.S. ______ psi Project No. 06-7857-001

Spec. No. PJ7 Initial G. L. 1.025 in. Machine No. 22KIP MTS

Temperature 550 °F Initial Dia. .248 in. Date 5-19-84

Strain Rate .005 in/in/min Initial Thickness ______ in. Initial Area .048305 in²

Initial Width ______ in.

Top Temperature ______ °F Maximum Load 3800 lb

Bottom Temperature ______ °F 0.2% Offset Load 2800 lb

Final Gage Length 1.210 in. 0.02% Offset Load ______ lb

Final Diameter .152 in. Upper Yield Point ______ lb

Final Area .018146 in.²

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 78,667 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 57,965 psi

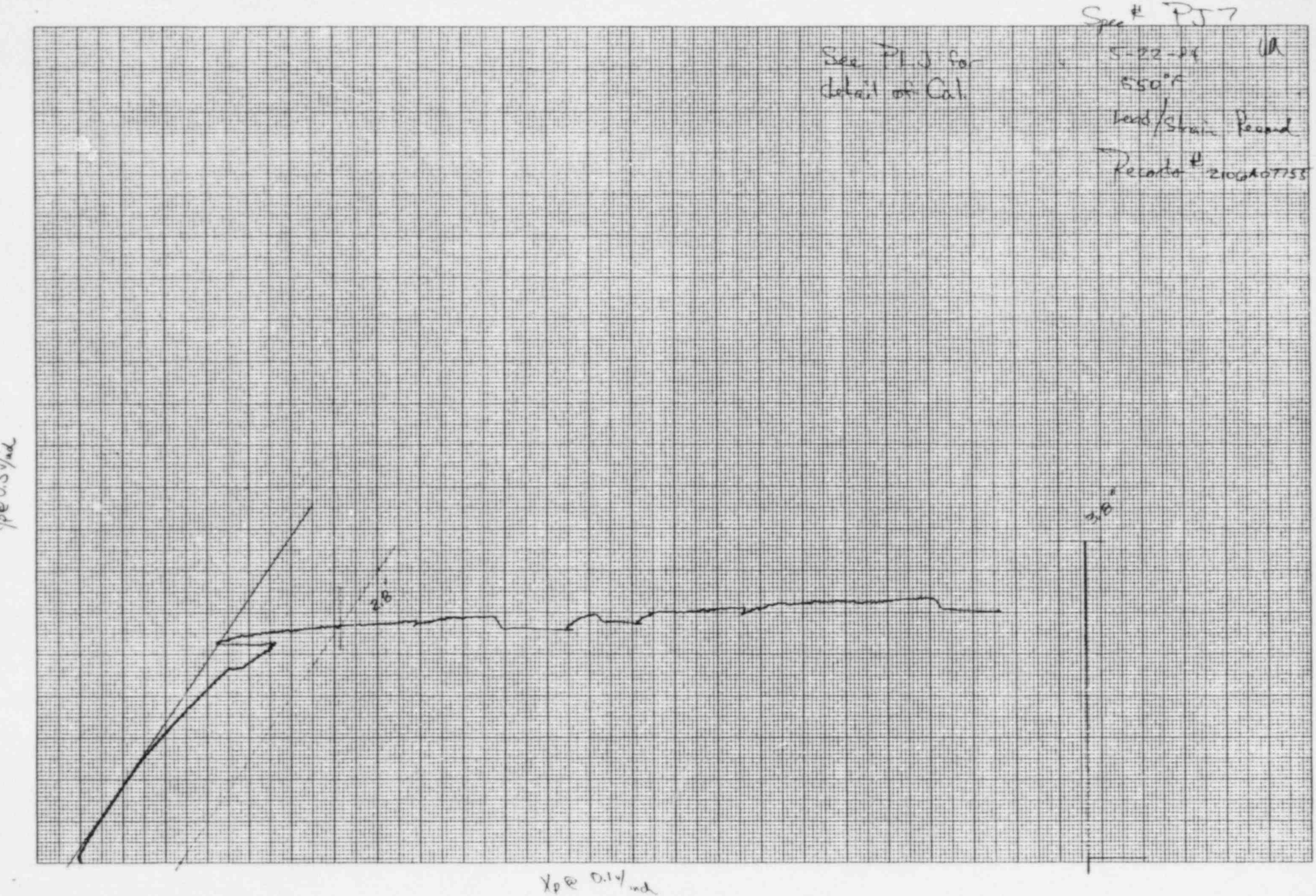
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ______ psi

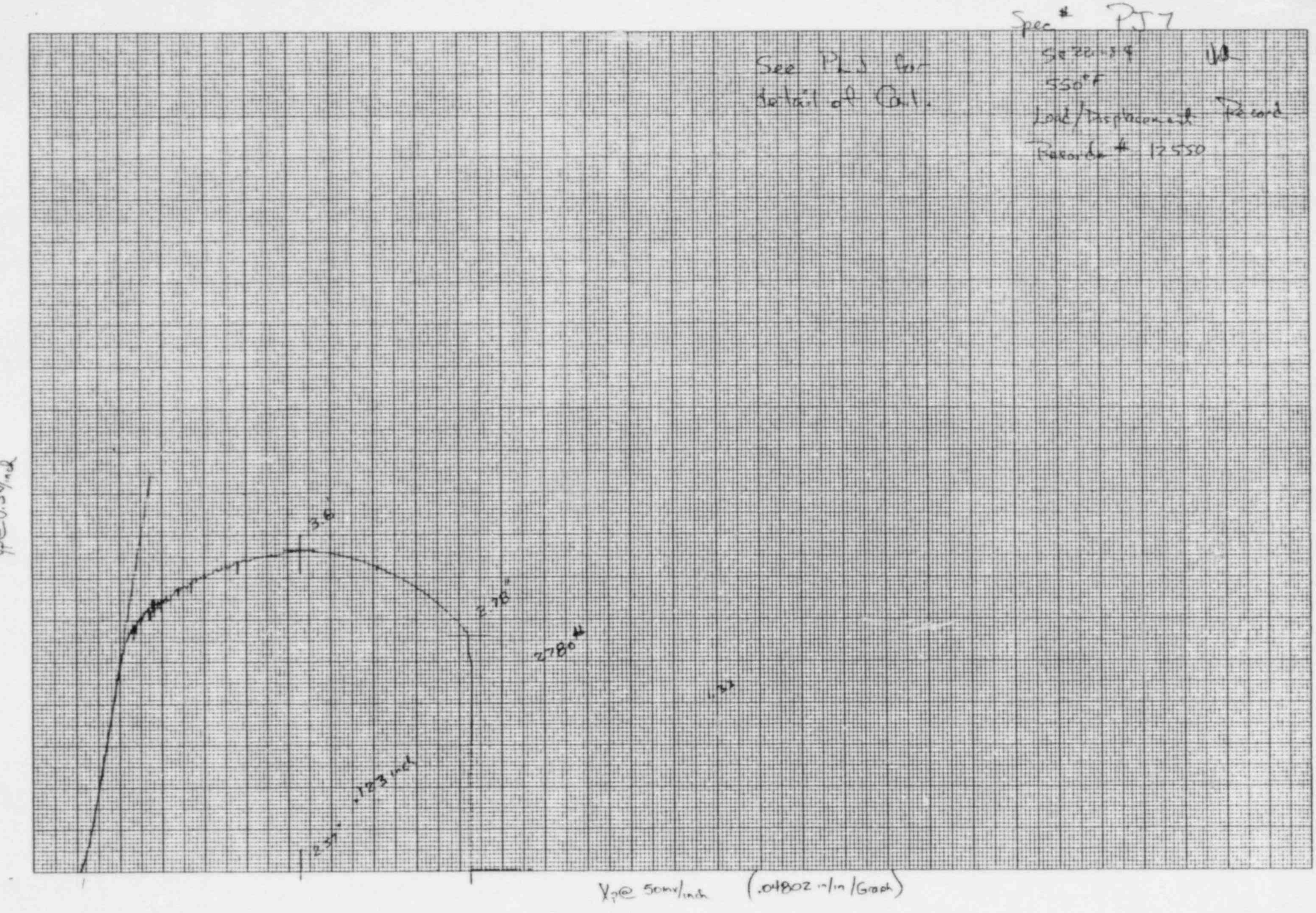
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 18.1 %

% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 62.4 %

Signature: S. Garon Checked E. J. Jones


Spec # PJ 7
5-22-86
550°F
Load/Strain Record
Recorder # 2106A07755
See PL.J for
detail of Cal.
2.8
3/8"
Xp @ 0.1v/inch
Yp @ 0.5v/inch
K+E
47 1323


Spec # PJ7
Se 22-3 8
550°F
Load/Displacement Record
Recorder # 12550
See PLJ for
detail of Cal.
3.6
2.78"
2780#
1.33
.183 inch
2.37"
Yp@ 0.5v/inch
Xp@ 50mv/inch
(.04802 in/in/Graph)