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QUAD CITIES NUCLEAR POWER STATION UNIT 2 REACTOR VESSEL IRRADIATION SURVEILLANCE PROGRAM ANALYSIS OF CAPSULE NO. 18

by
E. B. Norris

FINAL REPORT
SwRI Project No. 06-7484-002

for

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

March 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. 18 removed from the 215° position in the Quad Cities 2 pressure vessel during the 1981 refueling outage led to the following results.

(1) Based on a calculated neutron spectral distribution, the capsule received a fast fluence of 6.6×10^{16} n/cm^2, E > 1 MeV, in 5.63 Effective Full Power Years (EFPY) of operation.

(2) The shifts in RT_{NDT} were small, from -2°F for the base plate to +43°F for the weld metal, 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 RT_{NDT}.

(3) The Charpy upper shelf energy of the base plate and the electroslag heat-affected zone (HAZ) surveillance materials increased, but that of the electroslag weld metal decreased 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 4.89×10^8 $n/cm^2/sec$, E > 1 MeV. Therefore, the projected maximum vessel wall neutron fluence after 32 EFPY of operation is 4.9×10^{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 2 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 2 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. 18 from the 215° position in Quad Cities Unit 2 (the second vessel wall surveillance capsule tested). The results were analvzed 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 2 vessel at the pressure vessel wall prior to startup. In addition, three baskets were placed near the core. One vessel wall basket (No. 13) and two near-core baskets (Nos. 12 and 14) had been previously removed, tested, and the results reported [11,12].

The vessel wall Basket No. 18, removed during the 1981 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 2 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 2 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.18.

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 2 Nuclear Power Plant Reactor Vessel Irradiation Surveillance Program. Each of these activities is discussed below.

A. Shipment, Opening, and Inspection of Capsule

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. SwRI loaded the impact and tensile capsules into a radioactive material shipping cask and transported them to San Antonio, Texas.

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 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 2 operating history up to the 1981 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 2 reactor vessel was performed using the DOT-IV code to determine the radial,

TABLE I

OPERATIONS SUMMARY - QUAD CITIES 2

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
6/72	30	4.08	.136	3355
7/72	31	9.76	.315	3324
8/72	31	12.34	.398	3293
9/72	30	-	.000	3263
10/72	31	9.89	.319	3232
11/72	30	22.23	.741	3202
12/72	31	21.98	.709	3171
1/73	31	21.61	.657	3140
2/73	28	22.43	.801	3112
3/73	31	29.36	.947	3081
4/73	30	12.57	.419	3051
5/73	31	20.12	.649	3020
6/73	30	24.39	.813	2990
7/73	31	24.64	.795	2959
8/73	31	23.84	.769	2928
9/73	30	21.30	.710	2898
10/73	31	21.76	.702	2867
11/73	30	24.84	.828	2837
12/73	31	27.40	.884	2806
1/74	31	23.59	.761	2775
2/74	28	24.86	.888	2747
3/74	31	20.27	.654	2716
4/74	30	22.86	.762	2686
5/74	31	25.14	.811	2655
6/74	30	11.13	.371	2625
7/74	31	22.66	.731	2594
8/74	31	24.27	.783	2563
9/74	30	5.31	.177	2533
10/74	31	21.76	.702	2502
11/74	30	20.58	.686	2472
12/74	31	13.55	.437	2441
1/75	31	-	.000	2410
2/75	28	-	.000	2382
3/75	31	-	.000	2351
4/75	30	0.03	.010	2321
5/75	31	20.96	.676	2290
6/75	30	20.34	.678	2260
7/75	31	18.66	.602	2229

TABLE I CONT.

OPERATIONS SUMMARY - QUAD CITIES 2

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
8/75	31	10.70	.345	2198
9/75	30	17.85	.595	2168
10/75	31	2.17	.070	2137
11/75	30	17.55	.585	2107
12/75	31	27.68	.893	2076
1/76	31	29.70	.958	2045
2/76	29	19.37	.668	2016
3/76	31	23.31	.752	1985
4/76	30	26.76	.892	1955
5/76	31	23.56	.760	1924
6/76	30	20.25	.675	1894
7/76	31	21.39	.690	1863
8/76	31	20.37	.657	1832
9/76	30	6.42	.214	1802
10/76	31	-	.000	1771
11/76	30	20.13	.671	1741
12/76	31	22.66	.731	1710
1/77	31	29.02	.936	1679
2/77	28	17.81	.636	1651
3/77	31	8.52	.275	1620
4/77	30	27.78	.926	1590
5/77	31	23.56	.760	1559
6/77	30	22.02	.734	1529
7/77	31	18.44	.595	1498
8/77	31	12.74	.411	1467
9/77	30	13.65	.455	1437
10/77	31	20.58	.664	1406
11/77	30	17.85	.595	1376
12/77	31	22.48	.725	1345
1/78	31	8.99	.290	1314
2/78	28	-	.000	1286
3/78	31	14.10	.455	1255
4/78	30	25.95	.865	1225
5/78	31	17.08	.551	1194
6/78	30	15.15	.505	1164
7/78	31	24.92	.804	1133
8/78	31	27.87	.899	1102
9/78	30	25.17	.839	1072

TABLE I CONT.

OPERATIONS SUMMARY - QUAD CITIES 2

Operating Period	No. of Days (T_m)	Effective Full Power Days	Fraction Full Power (P_m)	Decay Days (t_m)
10/78	31	23.84	.769	1041
11/78	30	23.40	.780	1011
12/78	31	26.72	.862	980
1/79	31	26.69	.861	949
2/79	28	21.39	.764	921
3/79	31	23.44	.756	890
4/79	30	20.49	.683	860
5/79	31	23.47	.757	829
6/79	30	21.93	.731	799
7/79	31	21.24	.685	768
8/79	31	19.13	.617	737
9/79	30	15.24	.508	707
10/79	31	15.31	.494	676
11/79	30	10.71	.357	646
12/79	31	-	.000	615
1/80	31	-	.000	584
2/80	29	-	.000	555
3/80	31	-	.000	524
4/80	30	2.79	.093	494
5/80	31	24.02	.775	463
6/80	30	27.42	.914	433
7/80	31	25.36	.818	402
8/80	31	27.56	.889	371
9/80	30	28.62	.954	341
10/80	31	22.69	.732	310
11/80	30	21.66	.722	280
12/80	31	22.10	.713	249
1/81	31	27.53	.888	218
2/81	28	24.25	.866	190
3/81	31	29.05	.937	159
4/81	30	25.98	.866	129
5/81	31	28.24	.911	98
6/81	30	25.20	.840	68
7/81	31	22.72	.733	37
8/81	31	21.48	.693	6
9/81	6	3.92	.653	0

Total irradiation time = 5.63 EFPY.

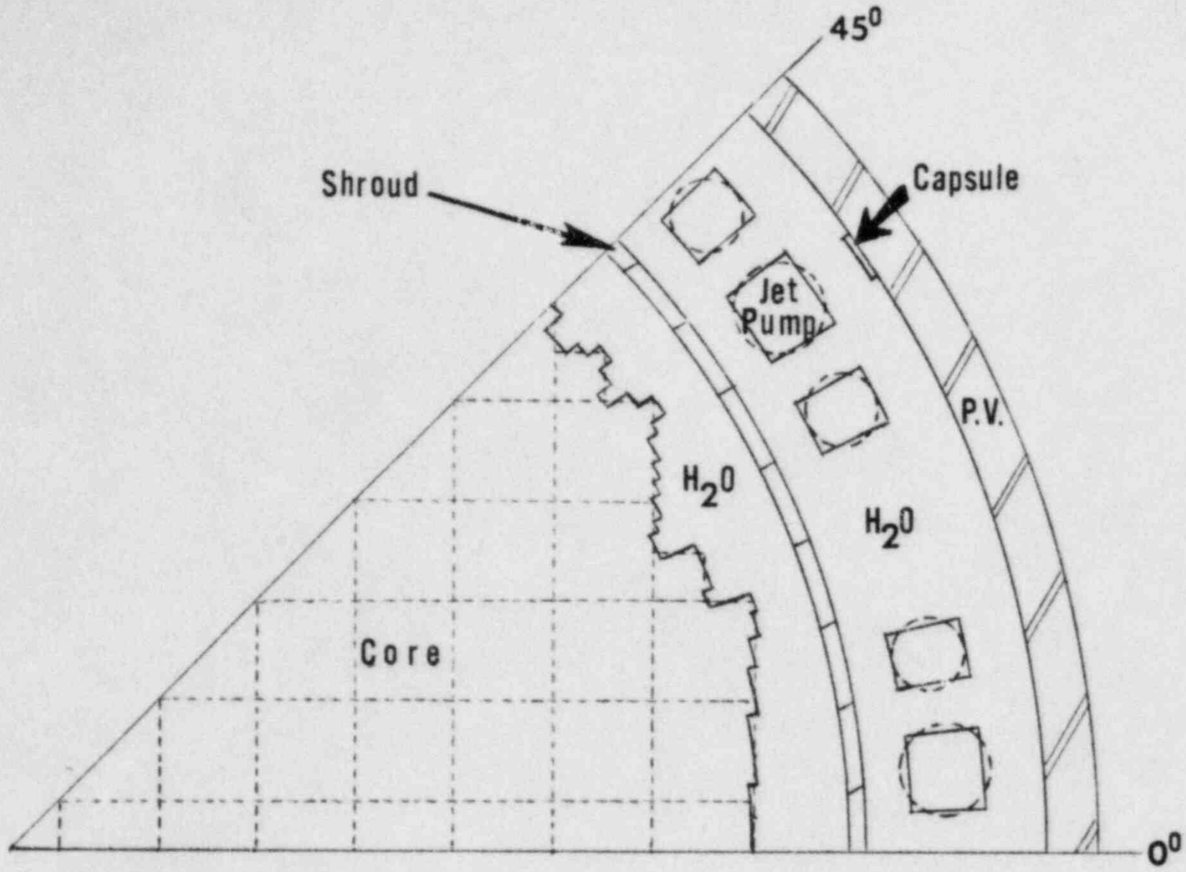
vertical, and azimuthal dependence of the fast neutron (E > 1.0 MeV) flux density and energy spectrum incident on the reactor vessel and surveillance basket capsules. The R-Θ and R-Z calculations were made using 15 of the 22 neutron group DLC-23/CASK cross section library, a P_1 expansion of the scattering 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.

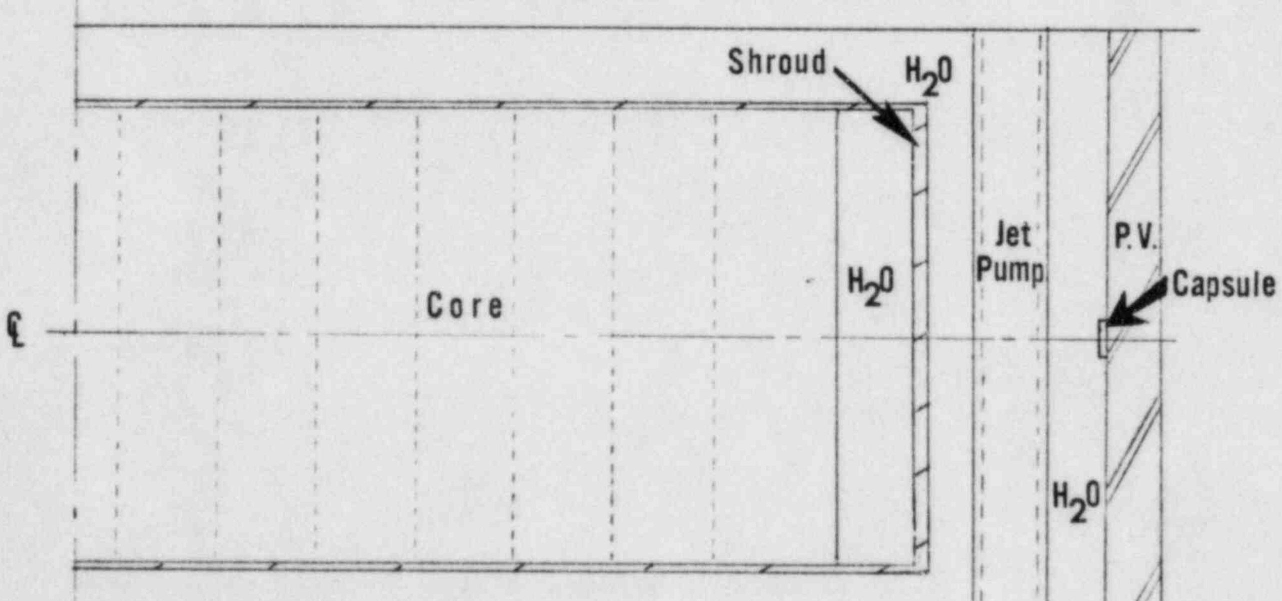
The calculated cross section for the $^{54}Fe(n,p)^{54}Mn$ reaction is somewhat less than the 236 millibarn (mb) value calculated for vessel wall Basket No. 13 [11]. It is believed that the values given in Table II are reasonable because SwRI-computed reaction cross sections for Browns Ferry Unit 3, 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 4.9×10^8 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 2

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)

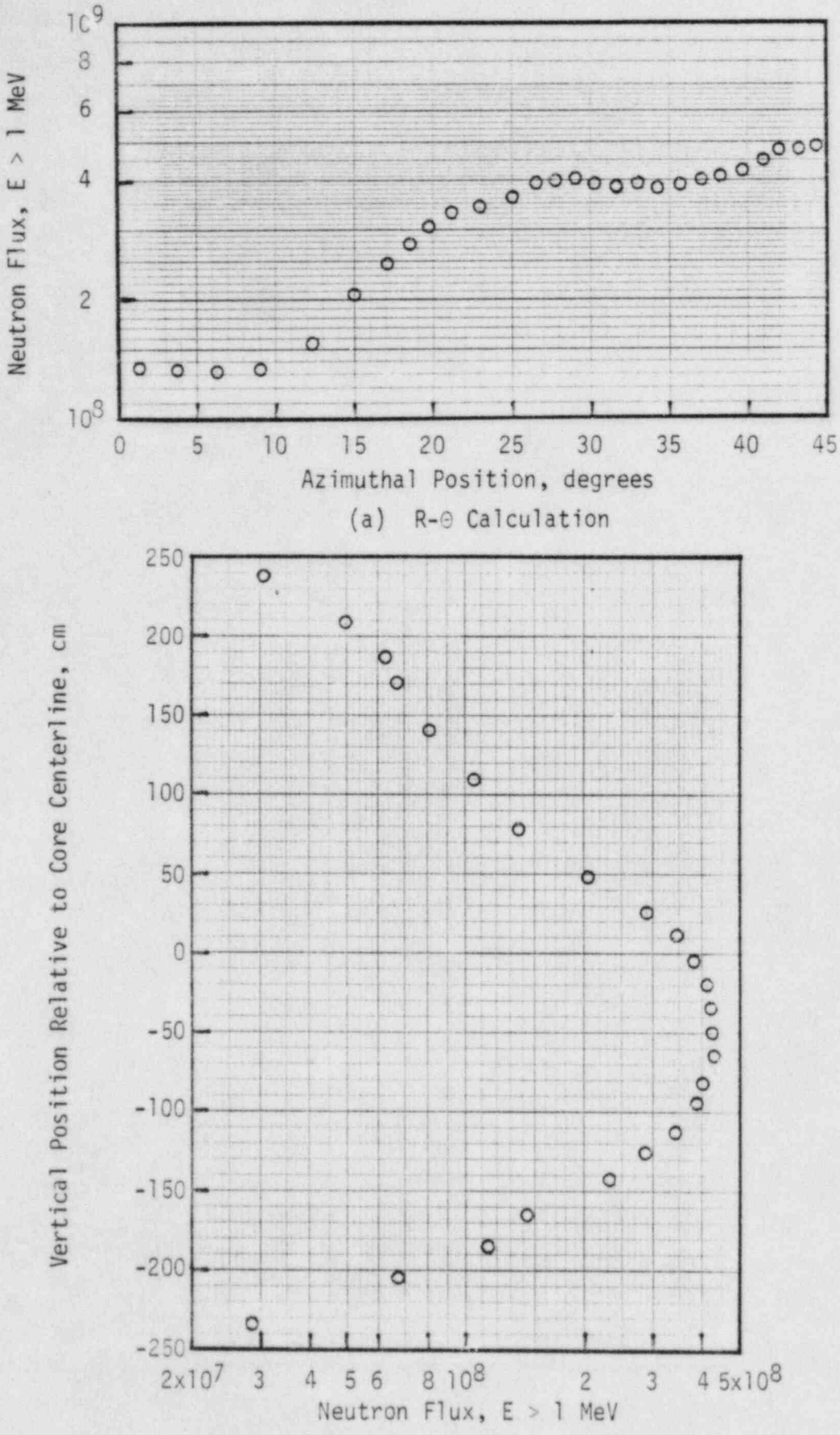
Reaction	$\bar{\sigma}$ (barns)
$^{54}Fe(n,p)^{54}Mn$	0.197
$^{58}Ni(n,p)^{58}Co$	0.237
$^{63}Cu(n,\alpha)^{60}Co$	0.00330

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. 18

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



(a) R-Θ Calculation

(b) R-Z Calculation

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

TABLE III

SUMMARY OF NEUTRON DOSIMETRY RESULTS
QUAD CITIES 2, VESSEL WALL SURVEILLANCE BASKET NO. 18 (215°)

Dosimeter I.D. Material	Dosimeter I.D. Capsule	Activation Reaction	Weight (mg)	A_{TOR} (dps/mg)	A_{SAT} (dps/mg)	Flux, E $\geq$ 1 MeV[a] ($n/cm^2/sec$)
Fe	G10	$^{54}Fe(n,p)^{54}Mn$	155.2	2.093×10^1	2.994×10^1	2.47×10^8
	G11		138.3	2.136×10^1	3.057×10^1	2.52×10^8
					Average =	2.49×10^8
Cu	G10	$^{63}Cu(n,\alpha)^{60}Co$	360.2	5.015×10^0	1.144×10^1	5.29×10^8
	G11		331.8	4.258×10^0	9.717×10^0	4.49×10^8
					Average =	4.89×10^8
Ni	G10	$^{58}Ni(n,p)^{58}Co$	290.2	(b)	-	-
	G11		247.2	(b)	-	-

(a) Calculated flux values subject to a ± 16.5% uncertainty (1σ).
(b) ^{58}Co had decayed away at time of measurement.

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 neutron dosimeters (the Co-58 in the nickel dosimeters could not be detected because of the long decay period), the fast neutron flux at the surveillance basket location during full power operation would be 3.69×10^8 n/cm^2/sec, E > 1 MeV, and the peak value incident on the pressure vessel I.D. would be 4.89×10^8. Since Quad Cities 2 operated for 5.63 effective full power years (EFPY) up to the September 1981 refueling outage, the calculated basket and vessel fluences to that time are as follows:

- Surveillance Basket - 6.56×10^{16} n/cm^2
- Pressure Vessel I.D. Surface - 8.68×10^{16} n/cm^2
- Pressure Vessel 1/4T - 6.13×10^{16} n/cm^2
- Pressure Vessel 3/4T - 1.38×10^{16} n/cm^2

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

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

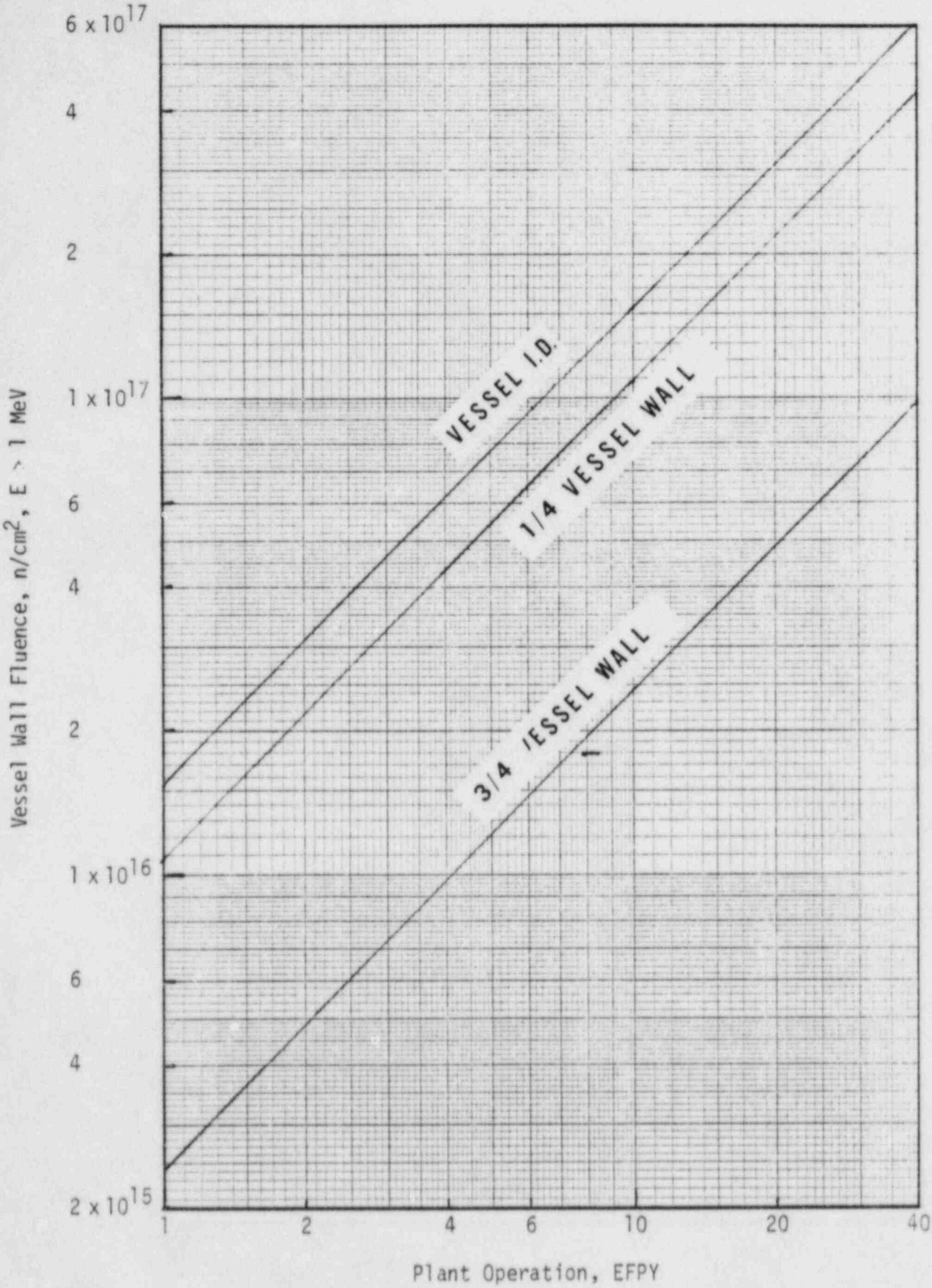


FIGURE 3. VESSEL WALL FLUENCE AS A FUNCTION OF OPERATION OF QUAD CITIES 2

TABLE IV

HARDNESS PROPERTIES OF SURVEILLANCE MATERIALS
QUAD CITIES 2

(Fluence = 6.6×10^{16}, E > 1 MeV)

Test Material	Charpy Specimen No.[(a)]	Hardness (HRB)
Base	T4D	92.0
		92.0
		91.0
		Average = 97.7
ESW	TBP	88.0
		90.0
		90.0
		Average = 89.3
ESHAZ	TML	90.0
		91.0
		91.0
		Average = 90.7

(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, an increase in the upper shelf energy of the base metal and the HAZ material may result from data scatter rather than a real change in the property of the material. However, the small changes which were measured are consistent with the level of fluence received by capsule basket No. 18.

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. Two each base plate and 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 2

(Fluence = 6.6×10^{16}, E > 1 MeV)

Material	Specimen No.[a]	Test Temperature (°F)	Impact Energy (ft-lb)	Lateral Expansion (mil)	Shear (%)
Base	T2D	-50	9.0	9	2
	T27	-25	22.5	20	5
	T2E	0	42.0	34	10
	T3P	40	74.0	59	20
	T4D	75	117.5	79	75
	T3M	120	139.0	94	100
	T37	160	150.5	80	100
	T2B	210	145.5	95	100
ESHAZ	TL5	-50	16.5	17	5
	TLK	-25	40.0	37	10
	TJ2	0	72.5	57	15
	TJU	40	40.0	38	15
	TML	75	76.5	66	45
	TL7	120	145.5	88	100
	TM1	160	149.5	97	100
	TMA	210	153.0	78	100
ESW	TAE	-25	10.5	11	2
	TAT	0	31.5	27	5
	TAP	20	33.0	29	10
	T72	40	74.0	60	20
	TBP	75	52.0	50	25
	TBM	120	86.0	65	90
	TB1	160	103.0	86	100
	T6K	210	78.0	63	100

(a) Specimen identification code given in Reference 10.

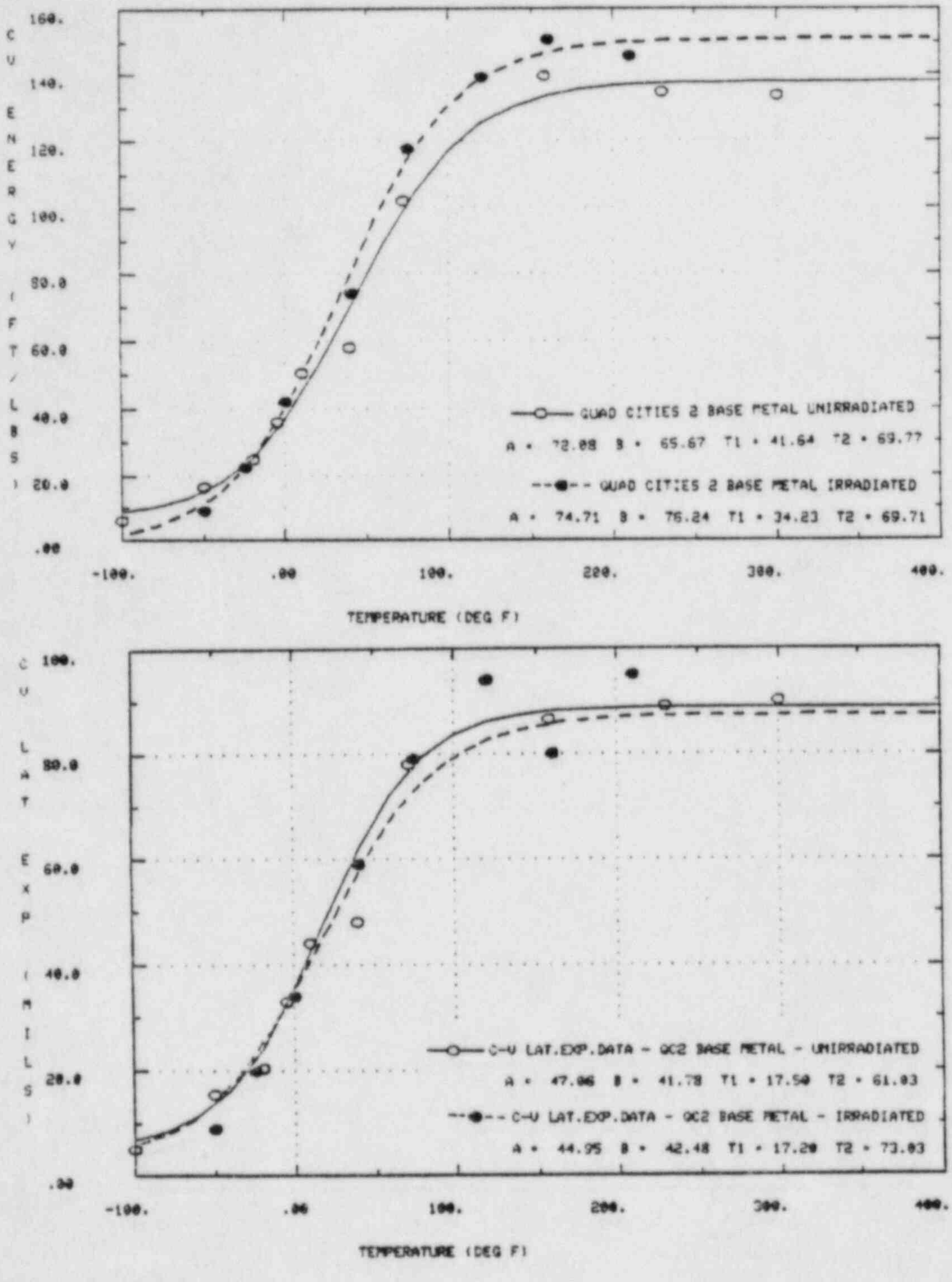


FIGURE 4. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES UNIT 2 SURVEILLANCE PLATE

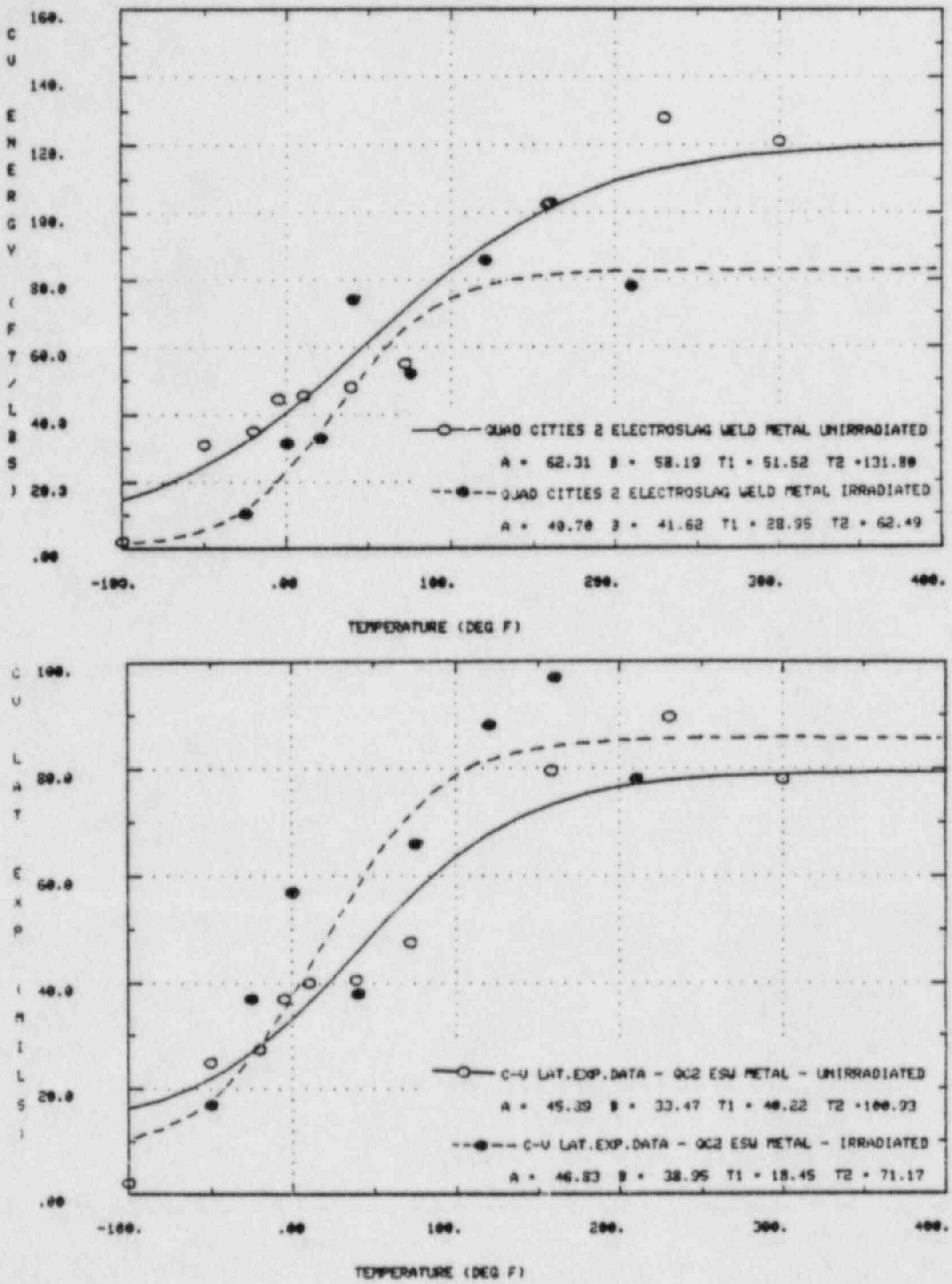


FIGURE 5. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES UNIT 2 SURVEILLANCE ELECTROSLAG WELD METAL

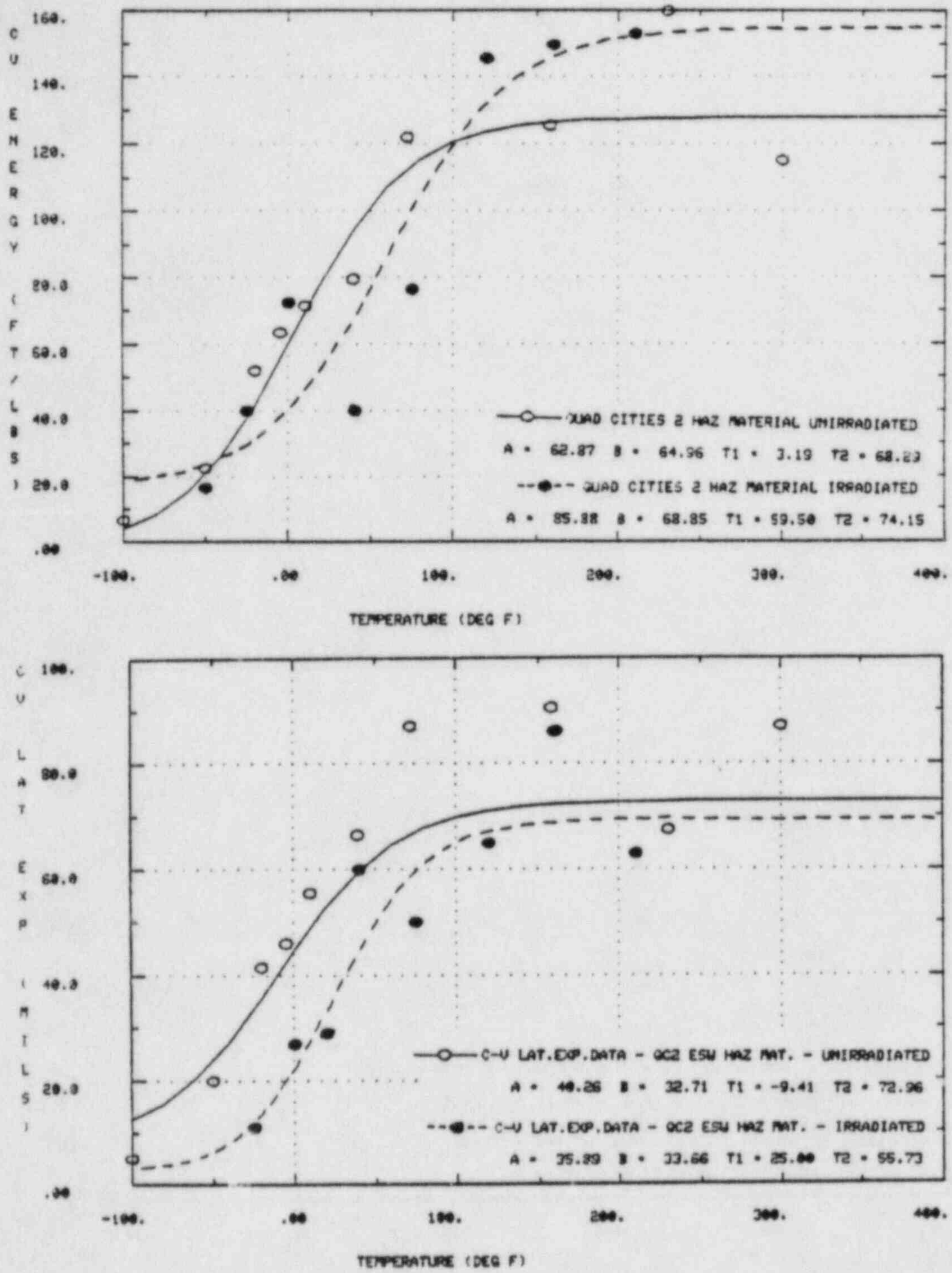


FIGURE 6. CHARPY V-NOTCH IMPACT PROPERTIES OF QUAD CITIES SURVEILLANCE ELECTROSLAG WELD HEAT-AFFECTED ZONE MATERIAL

EFFECT OF IRRADIATION ON THE CHARPY V-NOTCH PROPERTIES OF
THE QUAD CITIES UNIT 2 VESSEL SURVEILLANCE MATERIALS
BASKET NO. 18 (215°)

(Fluence = 6.6×10^{16}, E > 1 MeV)

Criterion[1]	Base Plate	ESW Weld Metal	ESW HAZ Material
Transition Temperature Shift[2]			
@ 50 ft-lb	-6°F	20°F	27°F
@ 30 ft-lb	-2°F	43°F	9°F
@ 35 mil	nil	-10°F	45°F
ΔRT_{NDT}[3]	(4)	43°F	9°F
C_v Upper Shelf Drop	-9 ft-lb (-7%)	34 ft-lb (27%)	-29 ft-lb (-24%)

(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. 18 (215°), QUAD CITIES 2
(Fluence = 6.6×10^{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	UDA	73	71.0	92.3	2669	171	17.5	20.1	68.2
	UDT	550	61.2	87.0	2639	171	15.7	22.6	68.6
ESHAZ	UL1	73	63.9	84.4	2539	185	16.1	21.6	72.1
	ULK	550	56.6	78.3	2547	161	13.0	19.7	67.7
ESW	UJ1	73	64.3	86.2	2663	165	18.9	26.4	67.3
	UJE	550	60.0	80.0	2639	143	15.0	20.5	62.5

(a) Specimen identification code given in Reference 10.

TABLE VIII

CHECK CHEMICAL ANALYSIS RESULTS
BASKET NO.18, QUAD CITIES 2

Specimen Identification[a]	Source[b]	Weight Percent of Element								
		C	S	P	Si	Cu	Ni	Cr	Mo	V
T27	W	.265	.019	.008	.199	.072	.494	.127	.469	.009
	S					.12	.60			
T2D	W	.287	.019	.008	.233	.078	.525	.132	.488	.008
	S					.10	.45			
TAE	W	.209	.017	.008	.115	.129	.313	.079	.503	.008
	S					.14	.32			
TAT	W	.216	.018	.008	.126	.122	.359	.091	.522	.009
	S					.16	.33			
TBP	S	-	-	-	-	.16	.34	-	-	-
TB1	S	-	-	-	-	.17	.37	-	-	-
T6K	S	-	-	-	-	.17	.39	-	-	-
TBM	S	-	-	-	-	.17	.37	-	-	-
T72	S	-	-	-	-	.20	.41	-	-	-
TAP	S	-	-	-	-	.12	.32	-	-	-

(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. 2

Unit No. 2 is one of four similar 251-in. I.D. boiling water reactors (Dresden Units 2 & 3 amd 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 10CFR50, 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 the vessel fluence rate, copper/nickel content, and initial RT_{NDT}. They were computed in accordance with Regulatory Guide 1.99, Revision 1 [8] amd 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 hydraulis diameter	= 24.0 in.

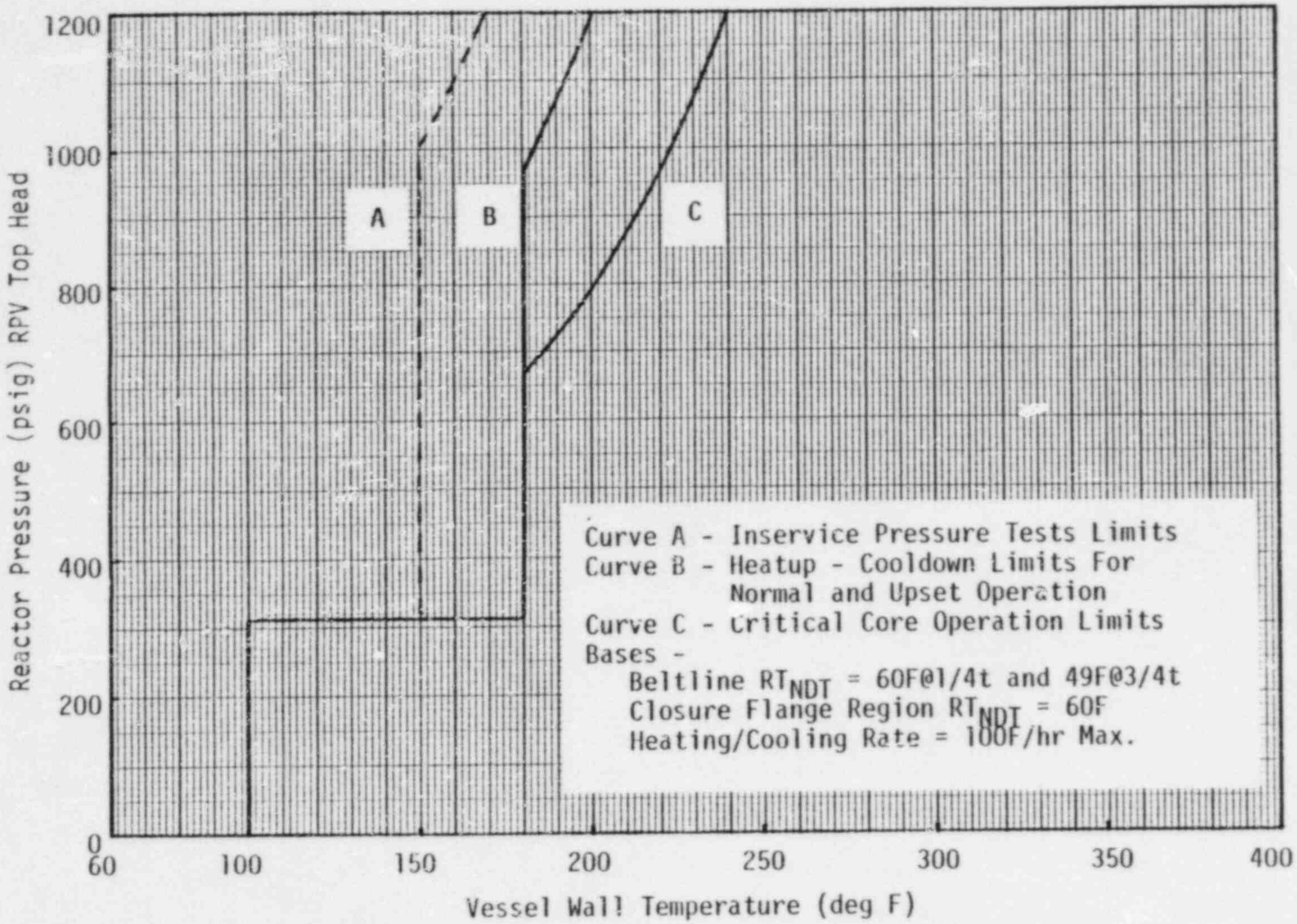


FIGURE 7. PRESSURE-TEMPERATURE LIMITS FOR LEVELS A & B SERVICE FOR UP TO 10 EFPY

V. 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 Embrittlement 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. 2 Reactor Pressure Vessel Surveillance Program: Capsule Basket No. 12 and Capsule Basket No. 13," Battelle Columbus Laboratories Report, September 19, 1975.

12. T. R. Mager et al, "Analysis of the Third Capsule from the Commonwealth Edison Company Quad Cities Unit 2 Nuclear Plant Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation WCAP-10064, Electric Power Research Institute Research Project 1021-3 Topical Report, April 1982.

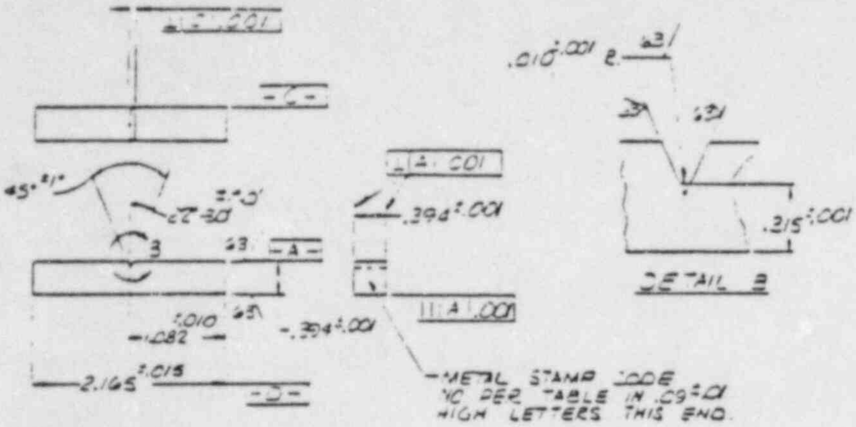
13. J. S. Perrin and L. M. Lowry, "Quad Cities Nuclear Plant Unit No. 1 and Unit No. 2 Reactor Pressure Vessel Surveillance Programs: Unirradiated Mechanical Properties," Battelle Columbus Laboratories Final Report to Commonwealth Edison Comapny, February 15, 1975.

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.

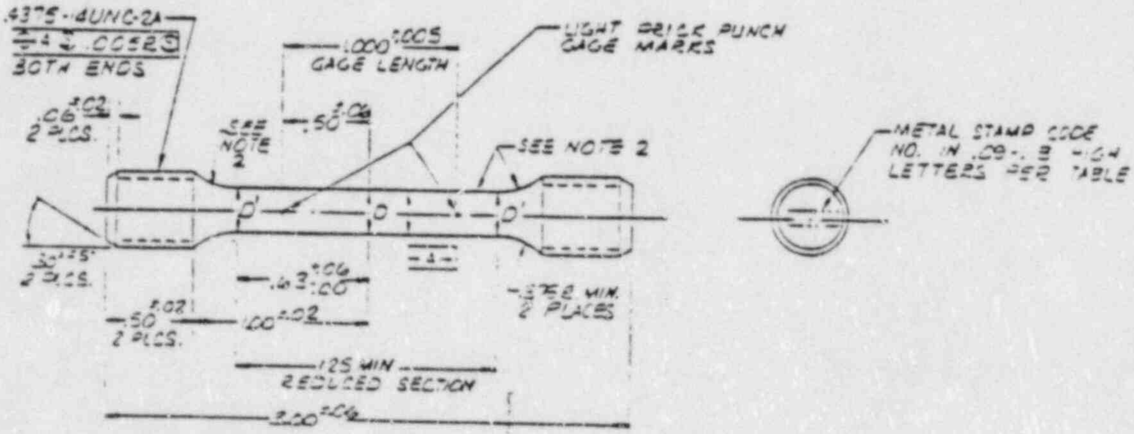
15. US NRC Standard Review Plan, NUREG-75/087, Section 5.3.2, Pressure-Temperature Limits, November 24, 1975.

APPENDIX A

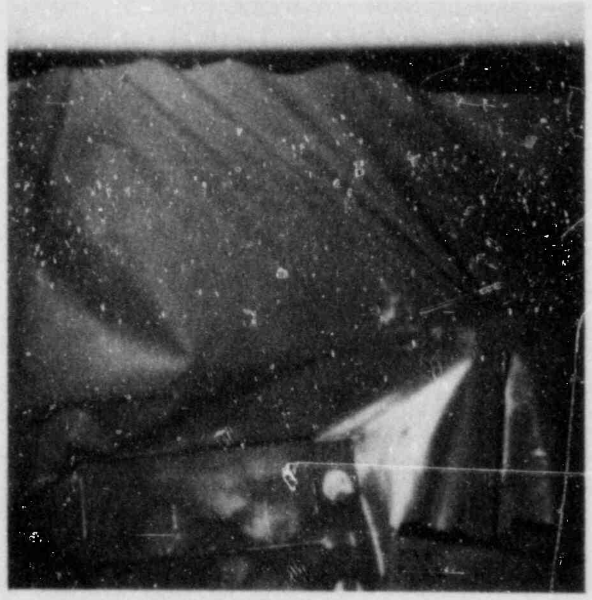
SURVEILLANCE SPECIMEN DRAWINGS AND CAPSULE PHOTOGRAPHS



PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 5837275. NO. TO RUN SERIALLY
4	BASE	21A1109	5İ1 TO 5ṠY
5	WELD (SUB-ARC)		6İ1 TO 6ṠY
6	WELD (ELECTROSLAG)		5Ġ1 TO 5ĖY
7	HAZ (SUB-ARC)		6Ġ1 TO 6ĊY
8	HAZ (ELECTROSLAG)		5J̇1 TO 5U̇Y

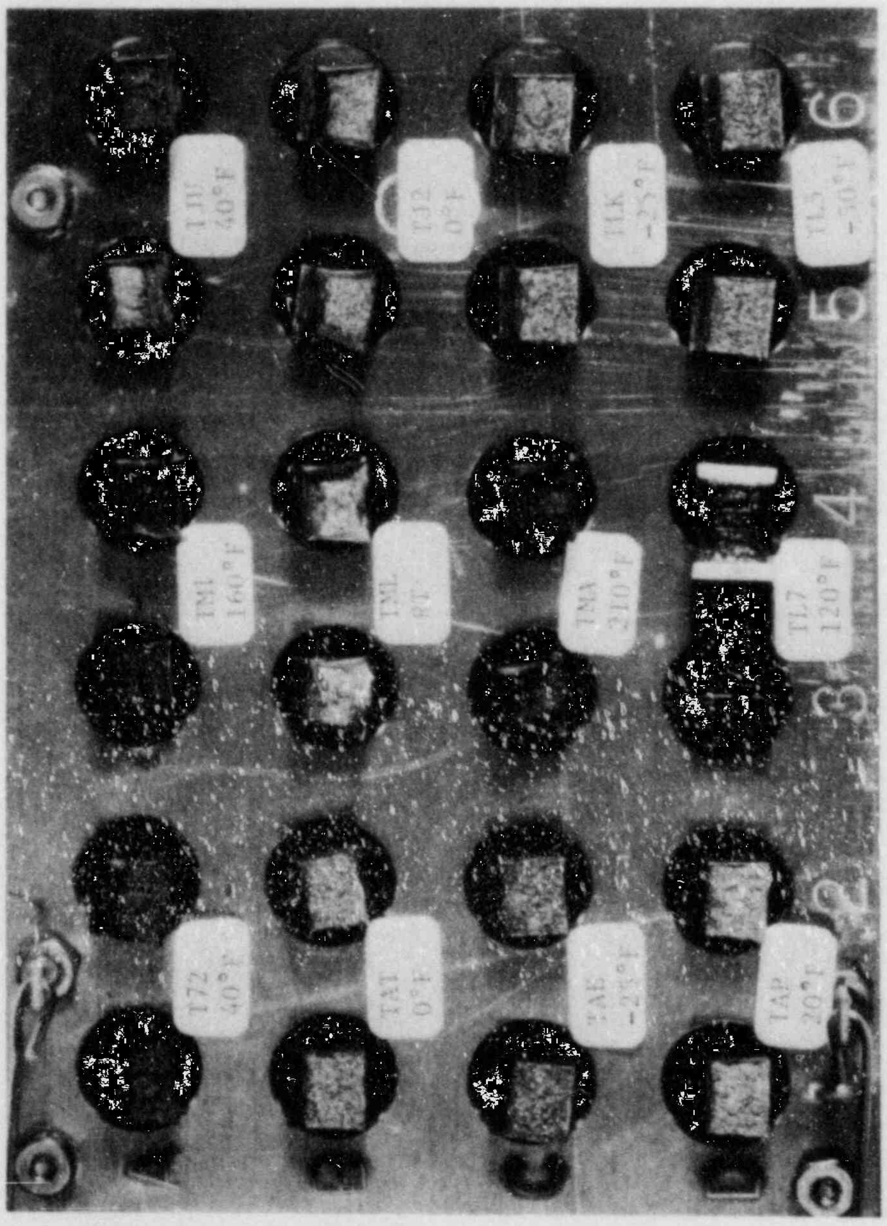


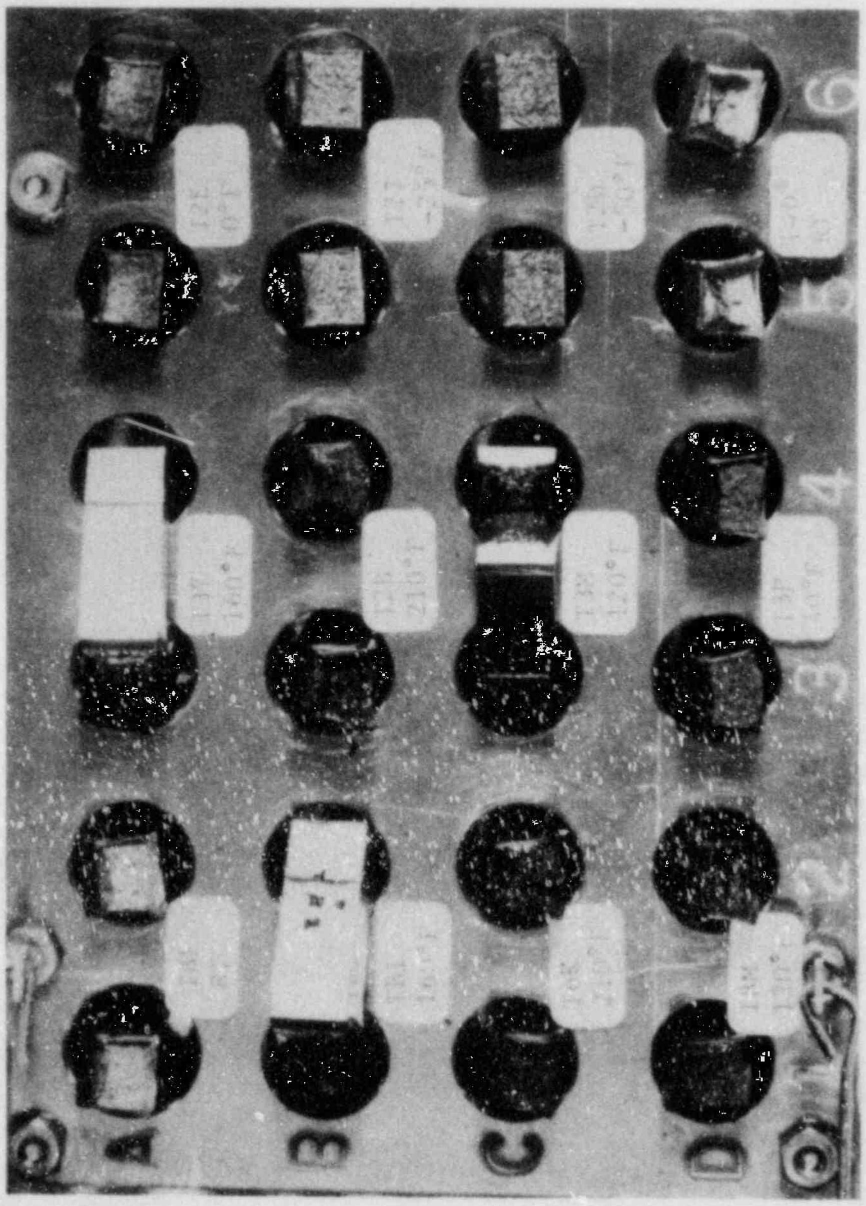
PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO. PER 5837275. NO. TO RUN SERIALLY
4	BASE	21A1109	6Ċ1 TO 6ĖY
5	WELD (SUB-ARC)		6Ṗ1 TO 6ṪY
[illegible]	WELD (ELECTROSLAG)		6J̇1 TO 6K̇Y
[illegible]	HAZ (SUB-ARC)		6U̇1 TO 6ẎY
8	HAZ (ELECTROSLAG)		6L̇1 TO 6ṀY

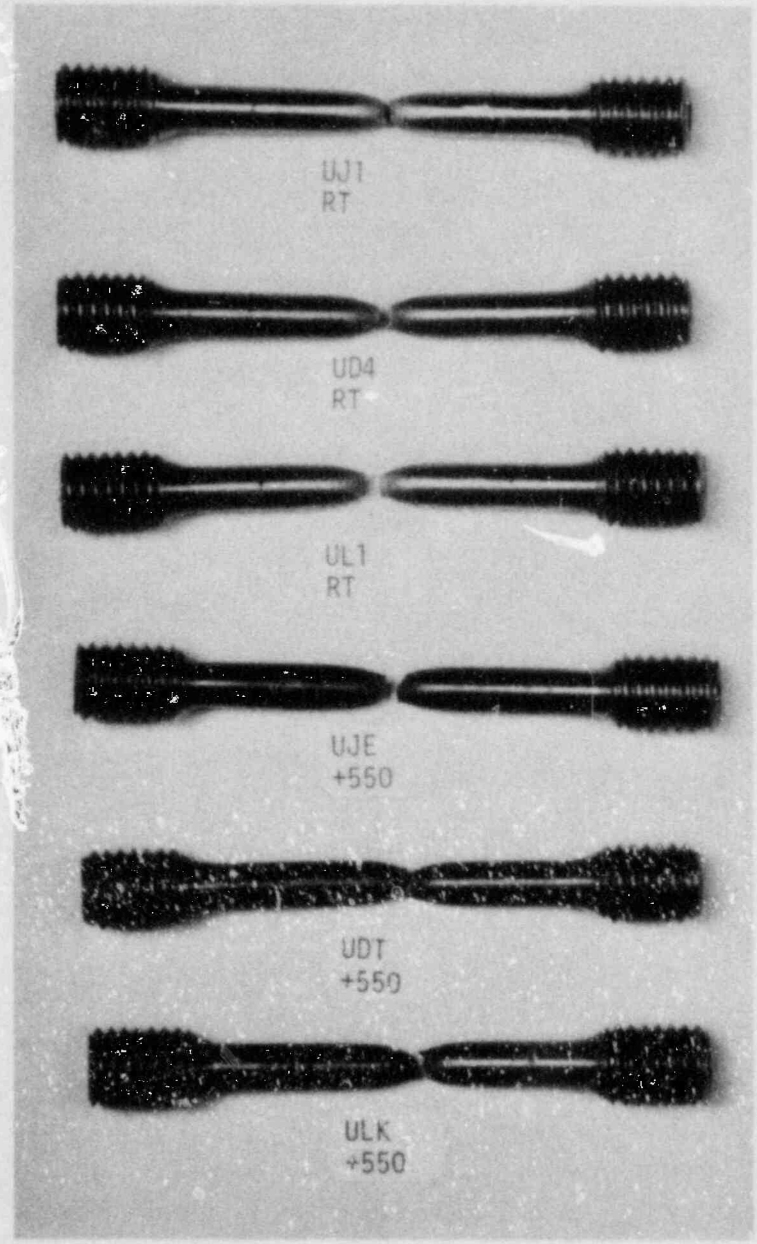


APPENDIX B

PHOTOGRAPHS OF TESTED SPECIMENS AND TENSILE TEST RECORDS


T72
40°F
TAP
20°F
TML
160°F
TMA
210°F
TL7
120°F
2
3
4
5
6




UJ1
RT
UD4
RT
UL1
RT
UJE
+550
UDT
+550
ULK
+550

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TENSILE TEST DATA SHEET

Test No. T- 2 | Est. U.T.S. ______ psi | Project No. 06-7484-002

Spec. No. UD4 | Initial G. L. 1.024 in. | Machine No. 11 KIP

Temperature 73 °F | Initial Dia. .250 in. | Date 10-21-83

Strain Rate ______ | Initial Thickness ______ in. | Initial Area 0.04909 in.^{2}

Initial Width ______ in.

Top Temperature RT °F

Bottom Temperature RT °F

Final Gage Length 1.281 in.

Final Diameter 0.141 in.

Final Area 0.0156 in.^{2}

Maximum Load 4533 lb

0.2% Offset Load 3485 lb

0.02% Offset Load 3445 lb

Upper Yield Point 3476 lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 92341 psi ✓

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 70992 psi ✓

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = 70177 psi

Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = 70809 psi

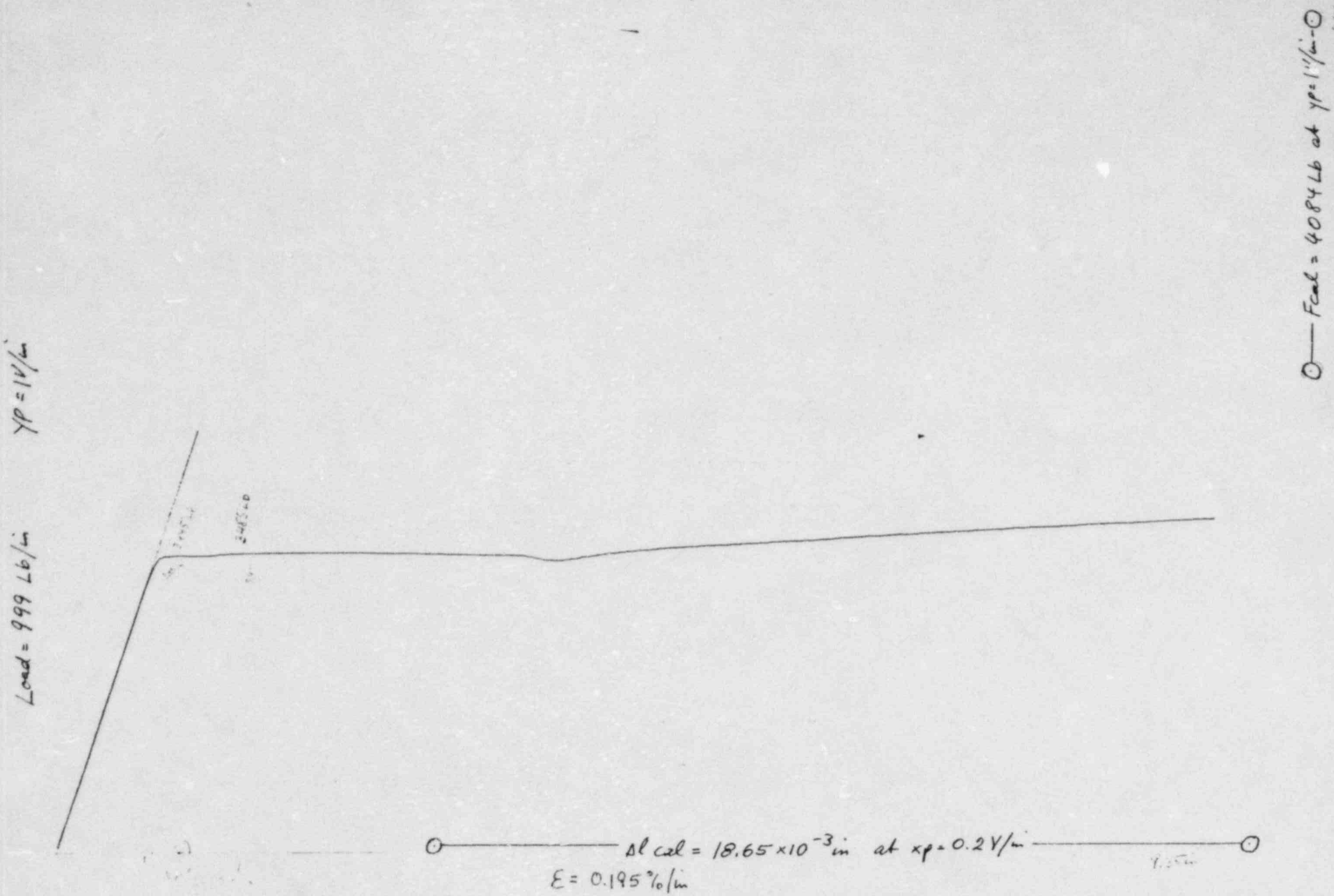
% Elongation = $\frac{\text{Final G.L. - Initial G.L.}}{\text{Initial G.L.}}$ x 100 = 20.1 %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 68.2 %

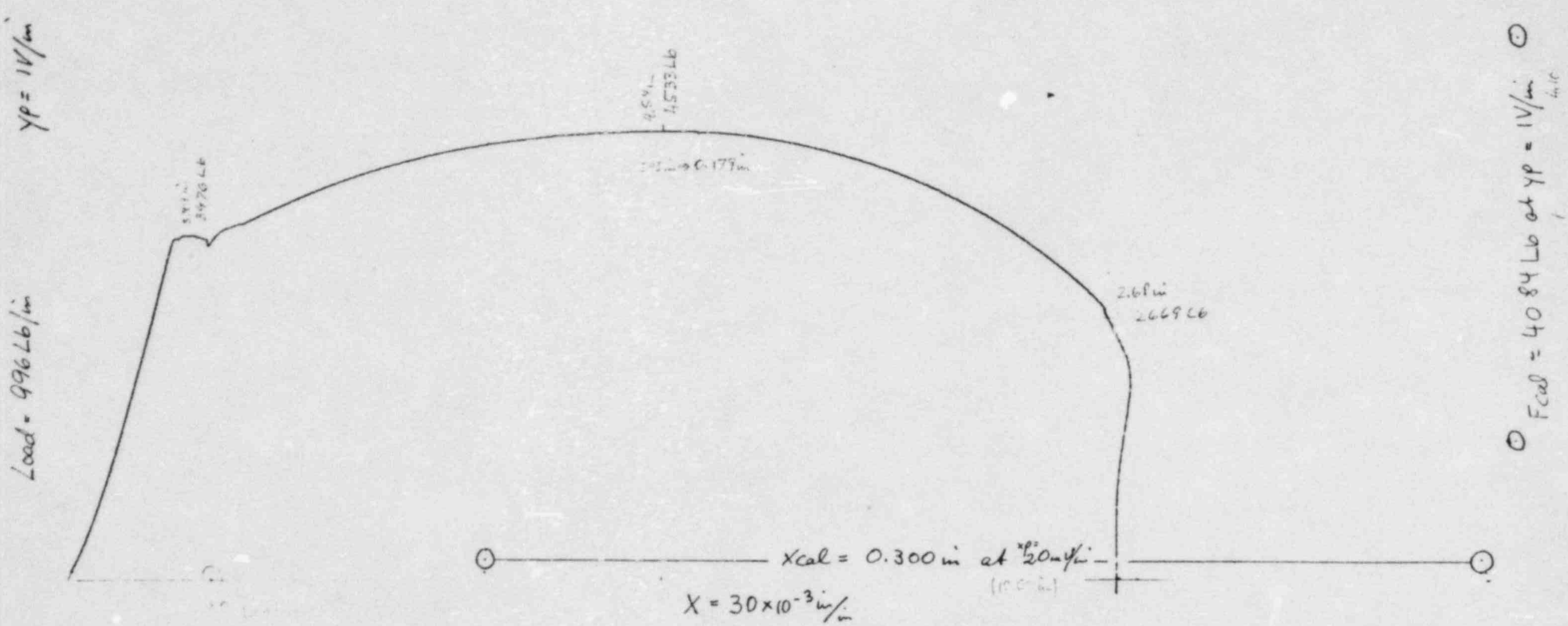
Signature: E B [signature]

Test # 2 Load vs. Extensiometer. Sp# UD4 T = 73°F 21 Oct 83

06-7484-002



Test #2 Load vs. Headdisplacement. Sp # UD4 T = 73°F 21 Oct 83 06-7484-002



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TENSILE TEST DATA SHEET

Test No. T- 5 | Est. U.T.S. ______ psi | Project No. 06-7484-002
Spec. No. UDT | Initial G. L. 1.025 in. | Machine No. 11KIP
Temperature 550 °F | Initial Dia. 0.250 in. | Date 10/24/83
Strain Rate ______ | Initial Thickness ______ in. | Initial Area 0.04909 in.^{2}
Initial Width ______ in.

Top Temperature 550 °F | Maximum Load 4273 lb
Bottom Temperature 551 °F | 0.2% Offset Load 3006 lb
Final Gage Length 1.257 in. | 0.02% Offset Load 2766 lb
Final Diameter 0.140 in. | Upper Yield Point NA lb
Final Area 0.0154 in.^{2}

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 87044 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 61234 psi

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = 56345 psi

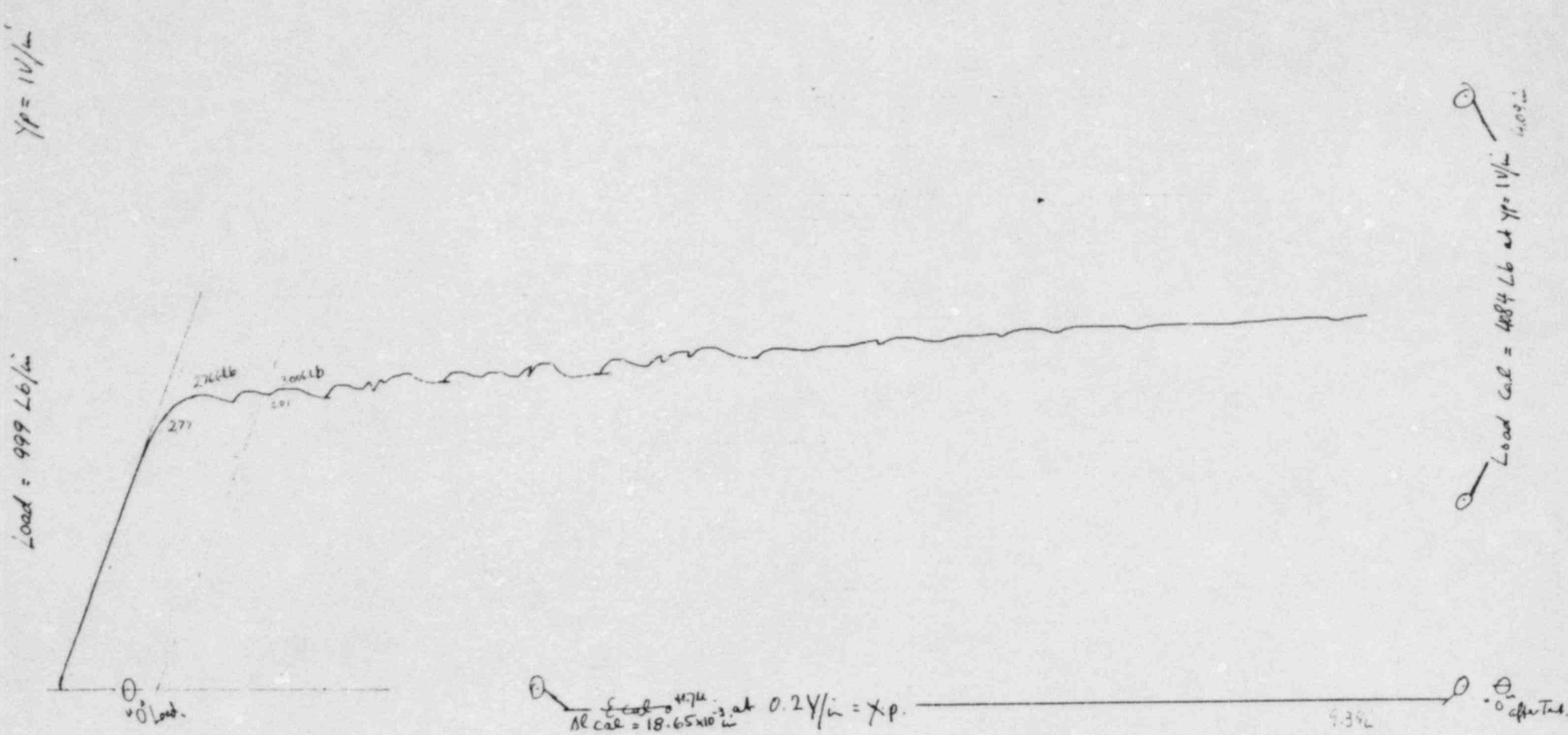
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 22.6 %

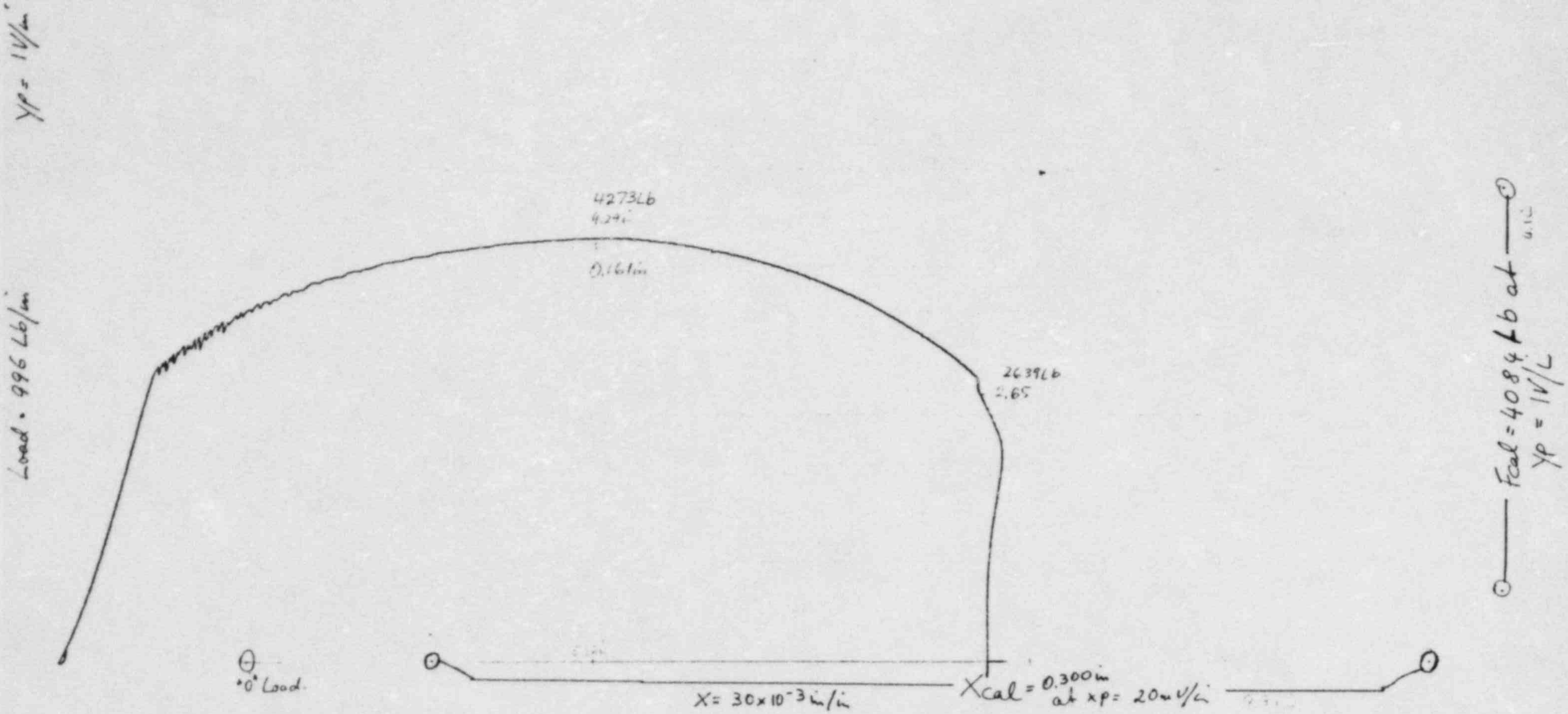
% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 68.6 %

Signature: E B Norris

Test #5 Load vs. Extensiometer. Sp # UDT 24 Oct 83 T = 550 °F 06-7484-002



Test# 5 Load vers. Headdisplacement. Sp# UDT 24 Oct 83 T = 550°F 06-7484002



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TENSILE TEST DATA SHEET

Test No. T- 3 | Est. U.T.S. ______ psi | Project No. 06-7484-002
Spec. No. UL1 | Initial G. L. 1.020 in. | Machine No. 11 KIP
Temperature 73 °F | Initial Dia. .250 in. | Date 10-21-83
Strain Rate ______ | Initial Thickness ______ in. | Initial Area 0.04909 in²
Initial Width ______ in.

Top Temperature ______ °F | Maximum Load 4143 lb
Bottom Temperature ______ °F | 0.2% Offset Load 3137 lb
Final Gage Length 1.240 in. | 0.02% Offset Load 2997 lb
Final Diameter 0.132 ~~1.240~~ in. | Upper Yield Point 3127 lb
Final Area 0.0137 ~~0.132~~ in.²

U.T.S. = Maximum Load / Initial Area = 84396 psi

0.2% Y.S. = 0.2% Offset Load / Initial Area = 63903 psi

0.02% Y.S. = 0.02% Offset Load / Initial Area = 61051 psi

Upper Y.S. = Upper Yield Point / Initial Area = 63699 psi

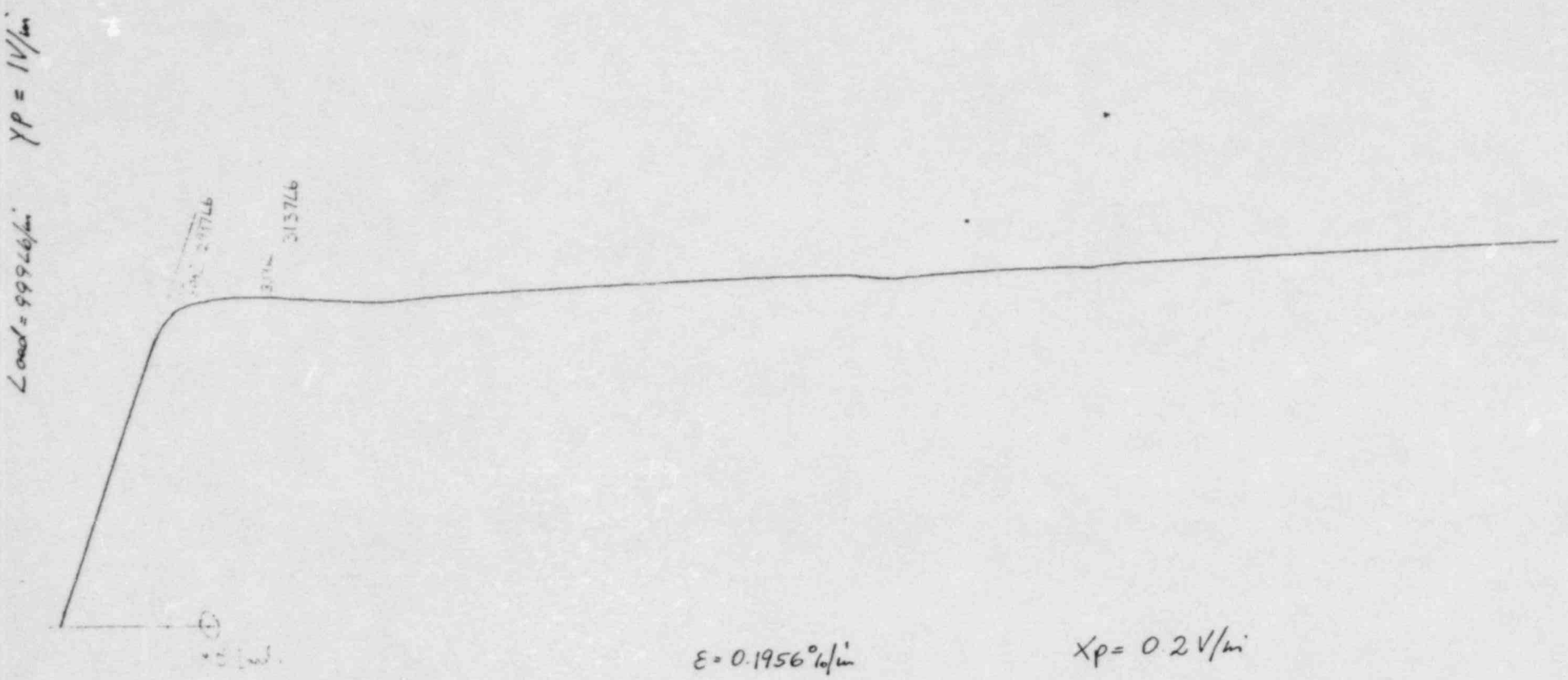
% Elongation = (Final G. L. - Initial G. L.) / Initial G. L. x 100 = 21.6 %

% R.A. = (Initial Area - Final Area) / Initial Area x 100 = 72.1 %

Signature: E B [signature]

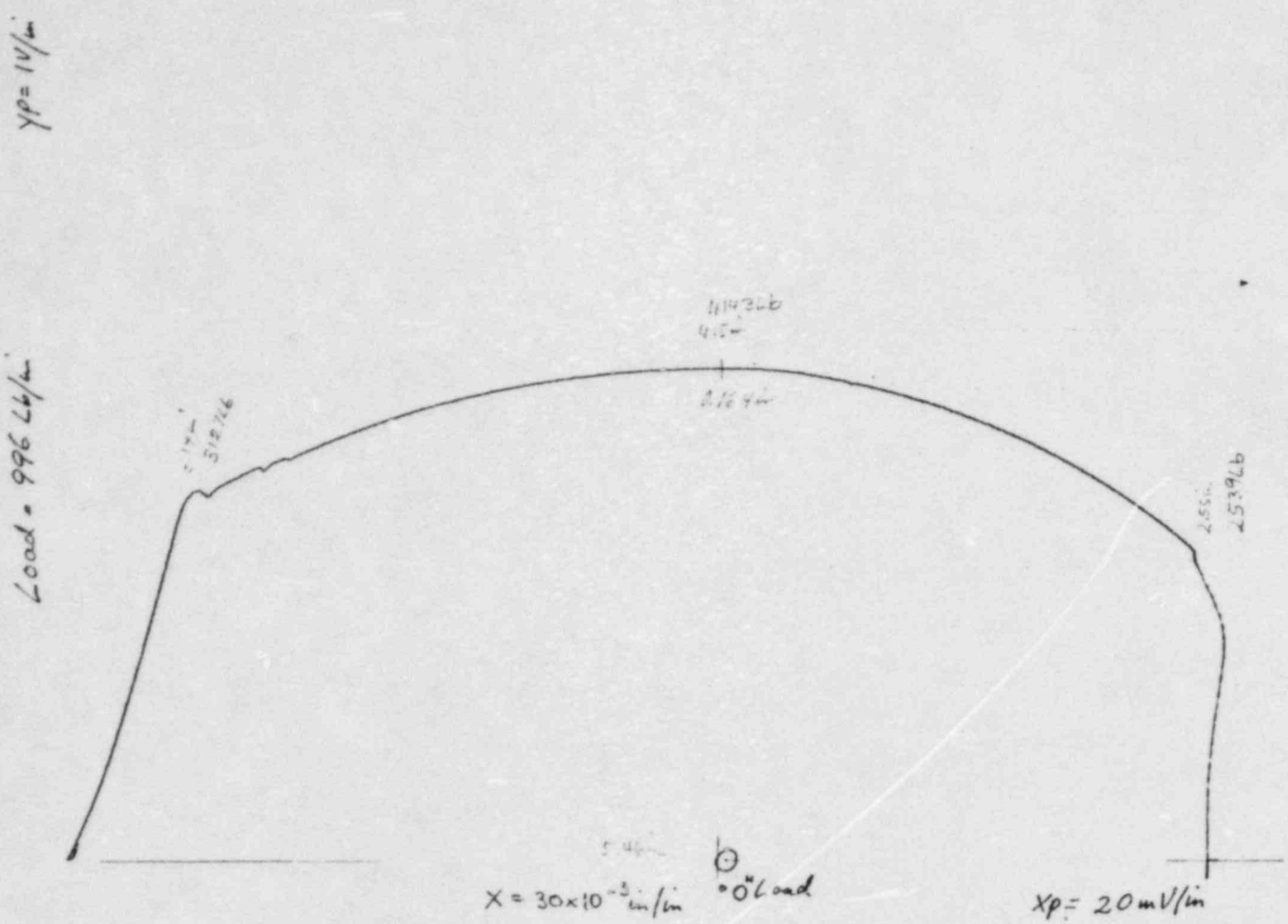
Test# 3 Load vers. Extensiometer. Sp# UL1 Temp. 73°F 21 Oct 83 06-87484-002

for Cal see test # 2
Sp # UD4



Test #3 Load vs. Head displacement Sp# UL1 Temp. 73°F 21 Oct 83 H. J. L. 06-7484-002

for Cal see test #2 UD4 Sp# UD4



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TENSILE TEST DATA SHEET

Test No. T- 6 | Est. U.T.S. ______ psi | Project No. 06-7484-002
Spec. No. ULK | Initial G. L. 1.025 in. | Machine No. 11 K IP
Temperature 550 °F | Initial Dia. 0.250 in. | Date 10/24/83
Strain Rate ______ | Initial Thickness ______ in. | Initial Area 0.04909 in.^{2}
Initial Width ______ in.

Top Temperature 550 °F
Bottom Temperature 553 °F
Final Gage Length 1.227 in.
Final Diameter 0.142 in.
Final Area 0.0158 in.^{2}

Maximum Load 3846 lb
0.2% Offset Load 2777 lb
0.02% Offset Load 2466 lb
Upper Yield Point NA lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 78346 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 56570 psi

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = 50234 psi

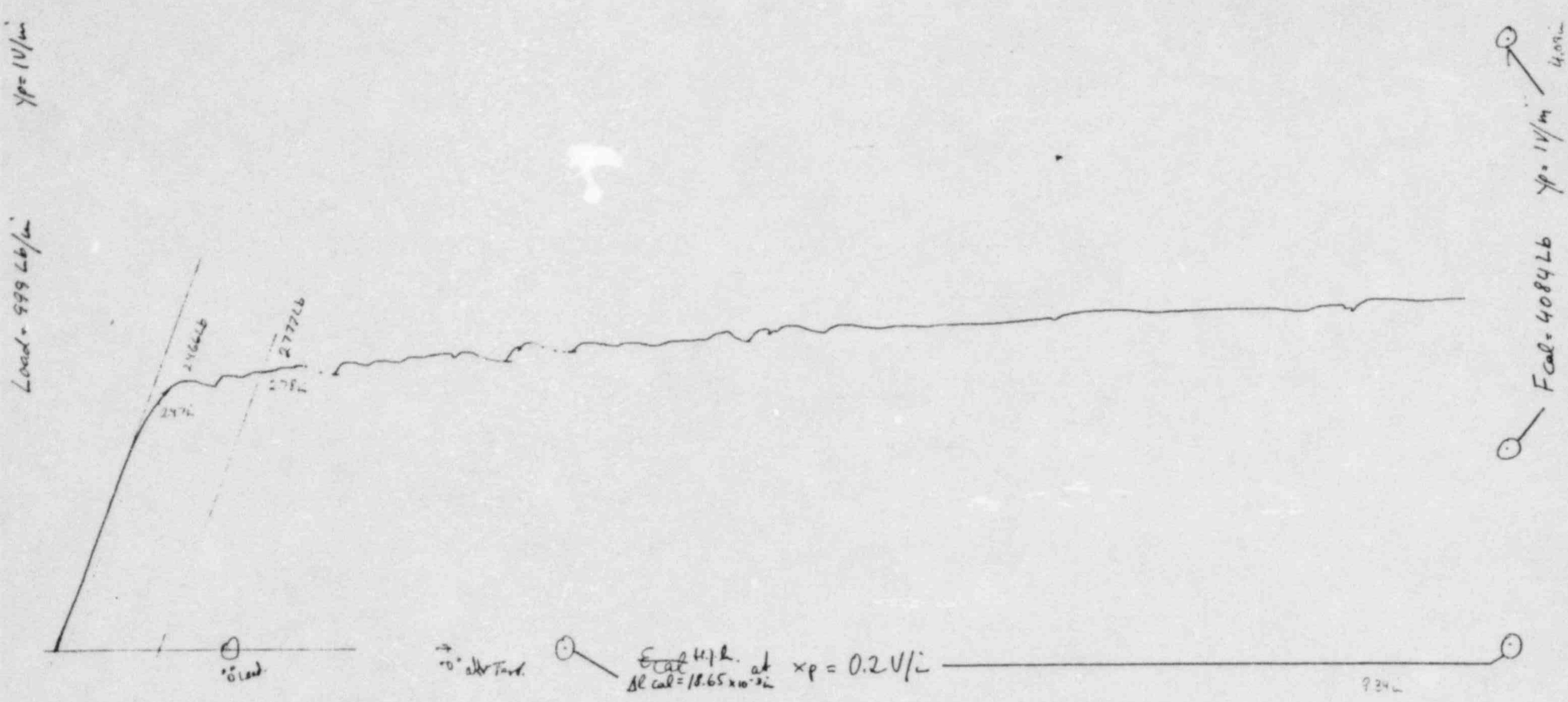
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 19.7 %

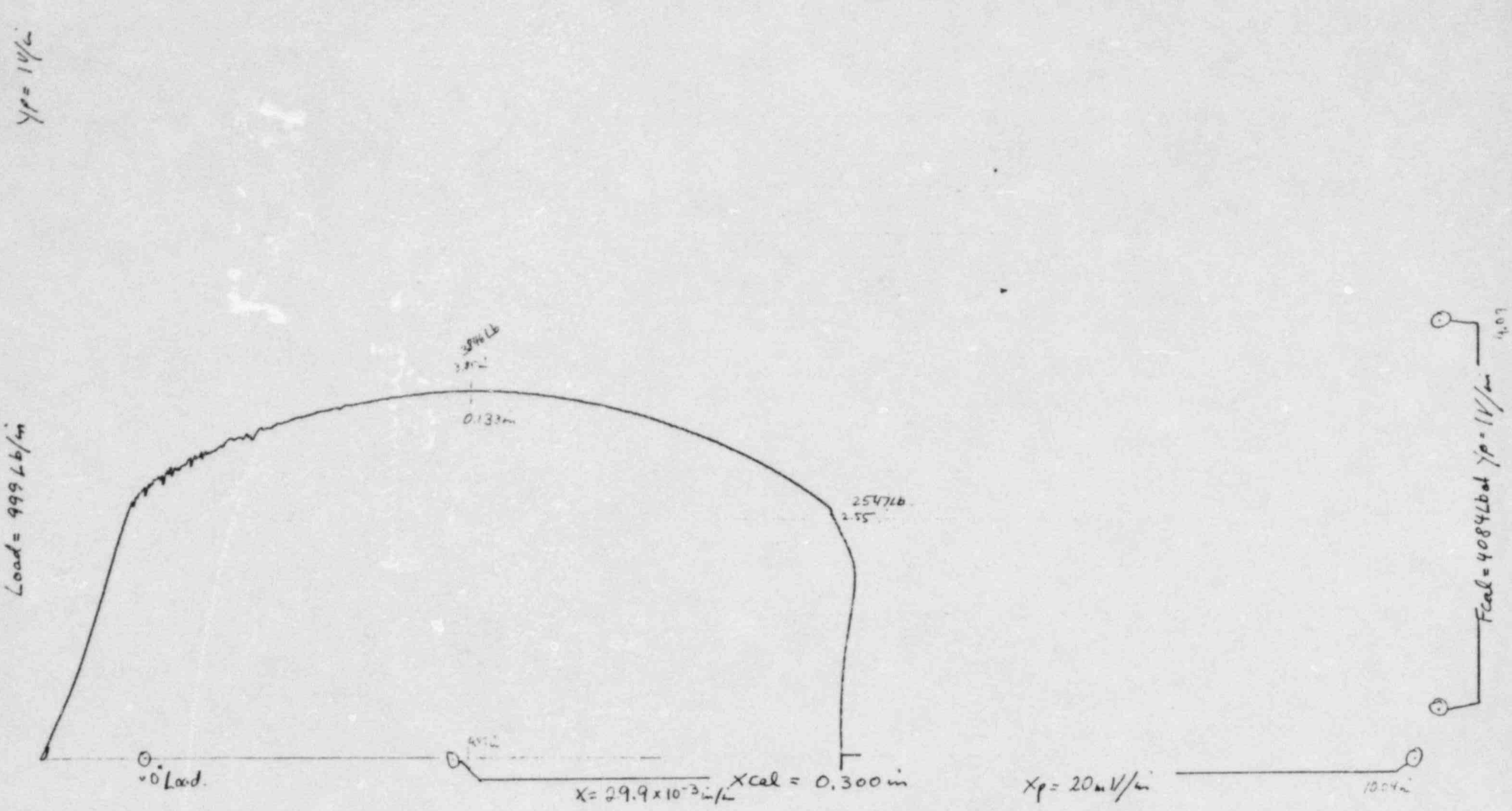
% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 67.7 %

Signature: E B [signature]

Test #6 Load vs. Extensiometer. Sp# ULK 24 Oct 83 Temp. 551 °F 06-7484-002



Test # 6   Load vers. Head displacement.   Sp # ULK   24 Oct 83   Temp. 551 °F   06-7484-002



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TENSILE TEST DATA SHEET

Test No. T- 1
Spec. No. UJ1
Temperature 73 °F
Strain Rate ______

Est. U.T.S. ______ psi
Initial G. L. 1.000 in.
Initial Dia. .250 in
Initial Thickness ______ in.
Initial Width ______ in.

Project No. 06-7484-002
Machine No. 11KIP
Date 10-20-83
Initial Area 0.04909 in^2

Top Temperature RT °F
Bottom Temperature RT °F
Final Gage Length 1.264 in.
Final Diameter 0.143 in.
Final Area 0.0161 in.^{2}
Uniform Elong. = 0.189"

Maximum Load 4234 lb
0.2% Offset Load 3155 lb
0.02% Offset Load 2357 lb
Upper Yield Point 3183 lb
Fracture Load = 2663 lb.

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 86250 psi ✓

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 64270 psi ✓

0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = 48014 psi

Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = 64840 psi

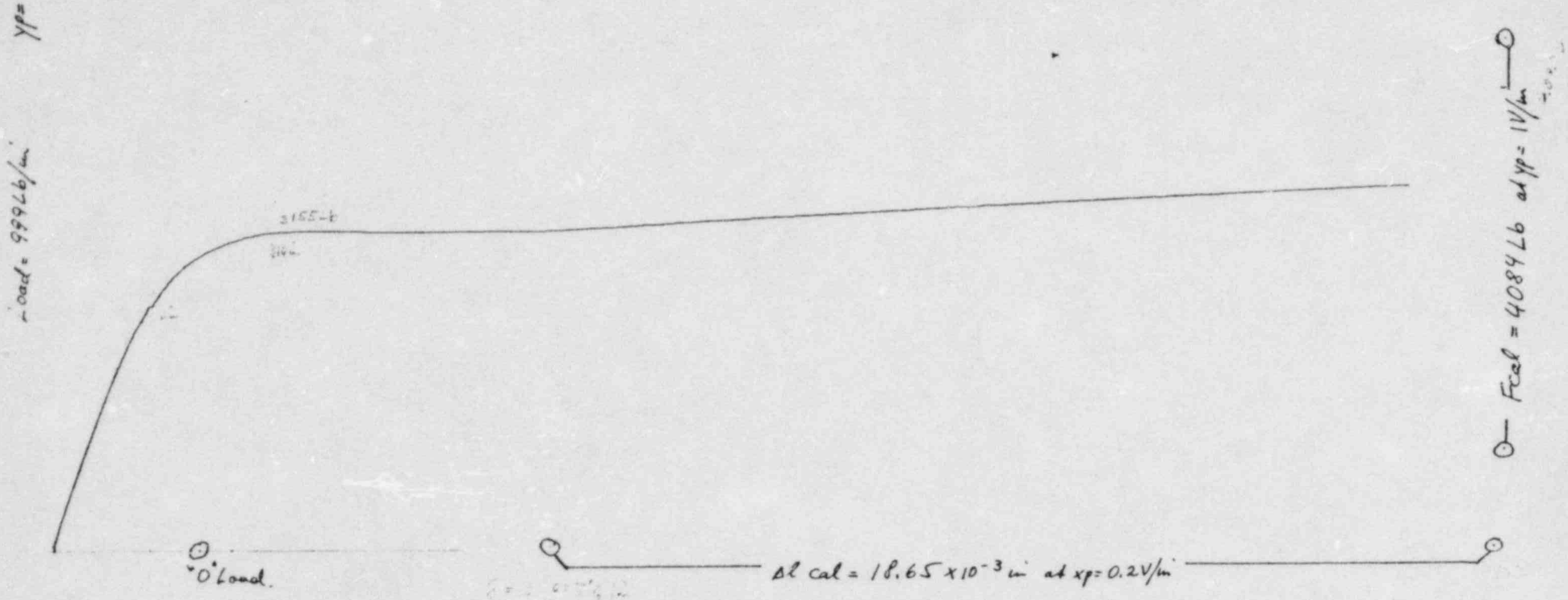
% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 26.4 %

% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 67.3 %

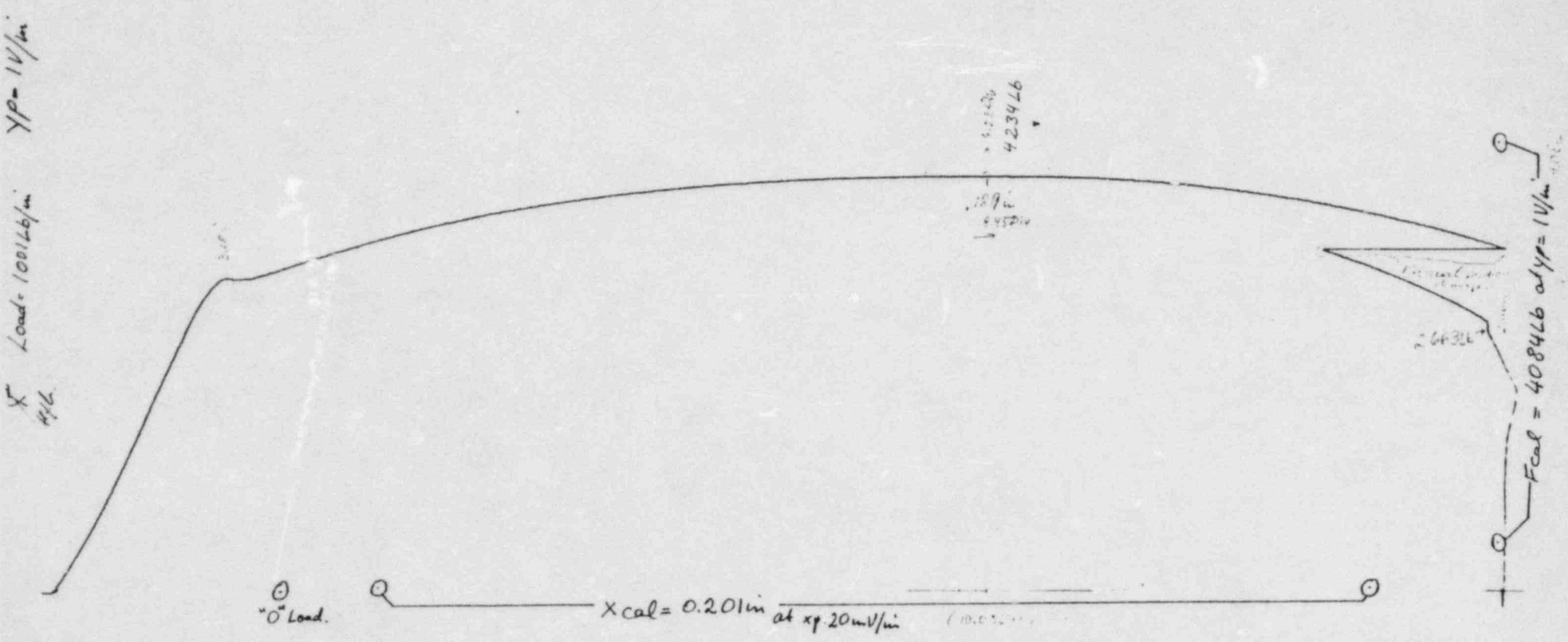
Uniform Elong. = $\frac{0.189(100)}{1.000}$ = 18.9%
Fract. Strength = 54,250 psi
Fract. Stress = 165,400 psi

Signature: [signature]

Test# 1 Load vers Extensiometer Sp# UJ#1 20 Oct 83 H.J.L. Temp = 73°F 06-7484-002



Test #1 Load vers. Head displacement. Sp #UJ#1 20 Oct 83 H.T.L. Temp. 73°F 02-7484-002



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TENSILE TEST DATA SHEET

Test No. T- 4

Spec. No. UJE

Temperature 550 °F

Strain Rate ______

Est. U.T.S. ______ psi

Initial G.L. 1.023 in.

Initial Dia. .250 in.

Initial Thickness ______ in.

Initial Width ______ in.

Project No. 06-7484-002

Machine No. 11 KIP

Date 10-21-83

Initial Area 0.04909 in.^{2}

Top Temperature 550 °F

Bottom Temperature 547 °F

Final Gage Length 1.233 in.

Final Diameter 0.153 ~~1.233~~ in.

Final Area 0.0184 in.^{2}

Maximum Load 3925 lb

0.2% Offset Load 2946 lb

0.02% Offset Load 2706 lb

Upper Yield Point NA lb

U.T.S. = Maximum Load / Initial Area = 79955 psi

0.2% Y.S. = 0.2% Offset Load / Initial Area = 60012 psi

0.02% Y.S. = 0.02% Offset Load / Initial Area = 55123 psi

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

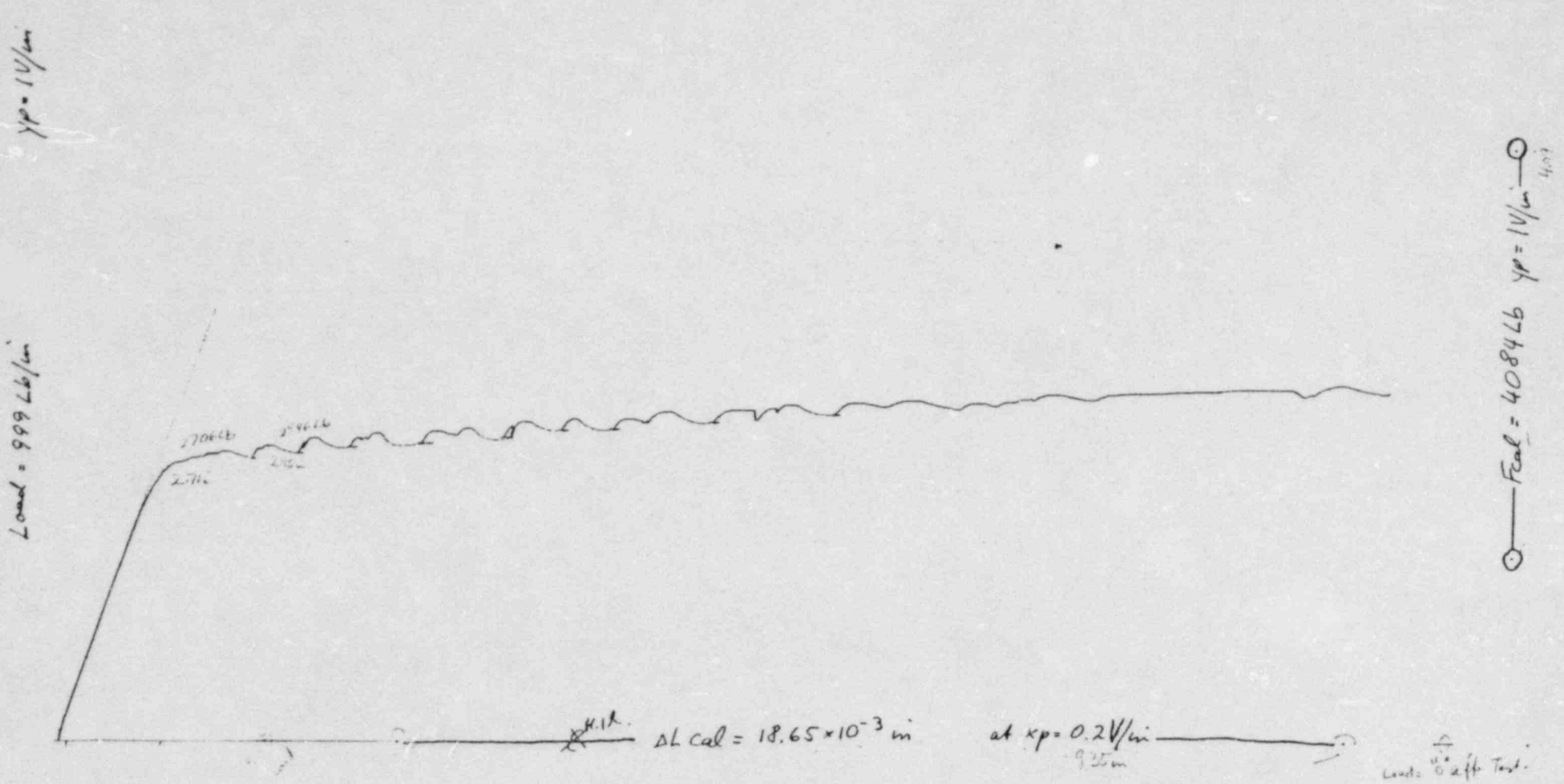
% Elongation = (Final G.L. - Initial G.L.) / Initial G.L. x 100 = 20.5 %

% R.A. = (Initial Area - Final Area) / Initial Area x 100 = 62.5 %

Signature: [signature]

Test # 4 Load vs. Extensiometer. Sp# UJE 21 Oct 83 T = 550 °F 06-7484-002

Tested by H/ Luettenhaupt



Test#4 Load vs. Head displacement.

Sp# UJE

21 Oct 83

Temp. 550°F 06-7484002

Tested by H. J. [illegible]