PILGRIM NUCLEAR POWER STATION UNIT 1 REACTOR VESSEL IRRADIATION SURVEILLANCE PROGRAM

by

E. B. Norris

FINAL REPORT
SwRI Project No. 02-5951

for

Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

July 1981

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U. S. Lindholm, Director
Department of Materials Sciences

ABSTRACT

The first vessel material surveillance capsule removed from Pilgrim 1 has been tested and the results analyzed. The results project a maximum shift in the reference nil ductility temperature of 136°F after 32 effective full power years of operation. Also, the vessel beltline material toughness is projected to remain well above 10CFR50, Appendix G, requirements throughout the design lifetime of the unit.

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I. SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the vessel material surveillance capsule removed from the Pilgrim 1 pressure vessel during the 1980 refuelling outage led to the following results and conclusions.

(1) Based on a calculated neutron spectral distribution, the capsule received an average fast fluence of 2.3×10^{17} n/cm^2, E > 1 MeV.

(2) The surveillance weld metal experienced the largest shift in RT_{NDT} (55°F) and is projected to control the system RT_{NDT} to the end of the design lifetime of the pressure vessel.

(3) The shelf energies of the surveillance materials are projected to remain well above the 50 ft-lb minimum required by 10CFR50, Appendix G, throughout the design lifetime of the pressure vessel.

(4) The estimated maximum neutron fluence of 2.6×10^{17} n/cm^2, E > 1 MeV, incident on the I.D. of the pressure vessel wall accrued in 4.17 effective full power years (EFPY). Therefore, the projected maximum neutron fluence after 32 EFPY is 2.0×10^{18} n/cm^2, E > 1 MeV.

(5) Using Regulatory Guide 1.99 procedures, the projected maximum values of ΔRT_{NDT} for the Pilgrim 1 vessel core beltline materials at the 1/4T location are 82°F after 12 EFPY and 136°F after 32 EFPY of operation. In addition, a graph showing RT_{NDT} + 60°F as a function of plant operation was prepared.

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 phosphorus, affect the radiation embrittlement response of ferritic materials [5-7]. The relationship between increase in RT_{NDT} and copper content is not defined completely. For example, Regulatory Guide 1.99, originally issued in July 1975, and revised in April 1977 [7], proposes an adjustment to RT_{NDT} proportional to the square root of the neutron fluence. Westinghouse Electric Corporation, in their comments on the 1975 issue of Regulatory Guide 1.99 [8], believed that the proposed relationship overestimates the shift at fluences greater than 1.9×10^{19} and underestimates the shift at fluences less than 1.9×10^{19}. On the other hand, Combustion Engineering, in their comments on the 1975 issue of Regulatory Guide 1.99 [9], suggested that the proposed relationship is overly conservative at fluences below 10^{19} neutrons per cm^2 ($E > 1$ MeV). There is also disagreement concerning the prediction of C_v upper shelf response to exposure to neutron irradiation [7-9]. After reviewing the comments and evaluating additional surveillance program data, the NRC issued a revision to Regulatory Guide 1.99 which raised the upper limit of the transition temperature curve. Although Regulatory Guide 1.99 is currently being revised to reflect a more recent evaluation of neutron embrittlement data by the Metal Properties Council [10], in this report estimates of shifts in RT_{NDT} are based on Regulatory Guide 1.99, Revision 1 [7].

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 materials which are located in the core beltline region of the reactor pressure vessel. Therefore, material surveillance programs include specimens machined from the plate or forging material and weldments which are located in such a region of high neutron flux density. ASTM E 185 [11] describes the current recommended practice for monitoring and evaluating the radiation-induced changes occurring in the mechanical properties of pressure vessel beltline materials.

General Electric has provided such a surveillance program for the Pilgrim 1 Nuclear Power Station. The encapsulated C_v specimens are 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, the capsule fluences lag the maximum vessel fluence in a corresponding exposure period. The capsules also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

The Pilgrim 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 data.

This report describes the results obtained from testing the contents of the first mechanical property surveillance capsule from the Pilgrim 1. These data are analyzed to estimate the radiation-induced changes in the mechanical properties of the pressure vessel at the time of the refuelling outage as well as predicting the changes expected to occur at selected times in the future operation of the Pilgrim 1 Nuclear Power Station.

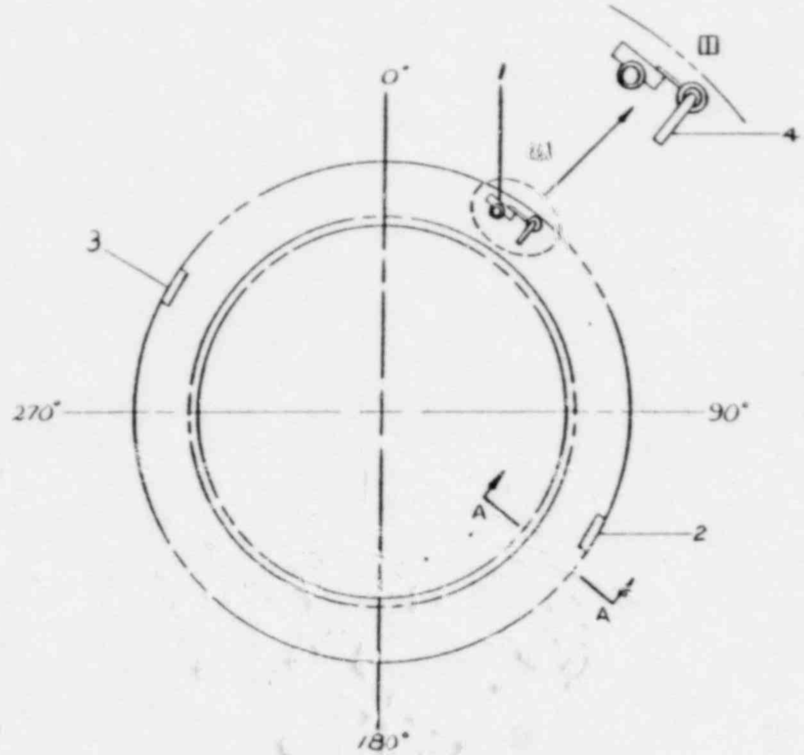
III. DESCRIPTION OF MATERIAL SURVEILLANCE PROGRAM

The Pilgrim 1 material surveillance program is described in detail in APED-5490 [12] and NEDO-10115 [13]. Three baskets containing encapsulated Charpy V-notch and tensile specimens were placed in the reactor vessel near the wall prior to startup, see Figure 1. Each capsule basket is located opposite the vertical center of the core.

The surveillance basket removed during the 1980 refuelling outage contained three impact capsules (each holding 12 Charpy V-notch specimens plus an iron, a nickel, and a copper flux wire) and four tensile capsules (each holding two tensile specimens). The capsules do not contain thermal monitors. Drawings of the impact and tensile specimens, the impact and tensile capsules, and the surveillance basket are shown in Appendix A.

The Pilgrim 1 base metal specimens were "made from flat slabs cut parallel to and one-quarter plate thickness from both of the plate surfaces ... with their longest dimension parallel to the plate rolling direction," and the weld metal and heat affected zone (HAZ) specimens were cut from a "test weld representing a vessel welded joint ... fabricated from vessel base material" [12]. The weld and HAZ Charpy V-notch specimens were oriented transverse to the weld direction, and the weld tensile specimens were taken parallel to the weld direction with the gage length made entirely of weld metal. The notches of all Charpy V-notch specimens were perpendicular to the original plate surfaces.

The mechanical and chemical properties of the unirradiated (baseline) surveillance specimens for the Pilgrim 1 vessel were determined



PART	BASKET CODE				LOCATION	AZIMUTH	REMOVAL PERIOD YEARS
	1	2	4	8			
1	● ○		○	○	WALL	30°	4
2	○	●	○	○	WALL	120°	12
3	● ○	●	○	○	WALL	300°	32
4	MANUF. SER. NO.				WALL	30°	1

FIGURE 1. LOCATIONS OF PILGRIM I VESSEL SURVEILLANCE CAPSULES

prior to the removal of the first capsule basket [14]. The chemical analyses reported for 20 of the broken unirradiated impact specimens are given in Table I. The tensile and Charpy V-notch properties determined from the unirradiated surveillance specimens are given in Appendix B.

TABLE I

CHEMICAL ANALYSIS OF BROKEN CHARPY SPECIMENS FROM PILGRIM BASELINE PROGRAM [14]

Specimen Identification	Material	Weight Percent of Indicated Elements									
		C	Mn	S	P	Si	Cu	Cr	Ni	Mo	V
Y1D	Base	0.23	1.44	0.013	0.014	0.19	0.13	0.14	0.62	0.58	0.014
Y1L	Base	0.24	1.42	0.013	0.014	0.16	0.14	0.12	0.64	0.60	0.017
Y12	Base	0.24	1.41	0.013	0.013	0.18	0.13	0.13	0.64	0.56	0.017
Y17	Base	0.24	1.41	0.014	0.015	0.16	0.14	0.14	0.65	0.60	0.017
Y2C	Base	0.24	1.26	0.014	0.017	0.15	0.13	0.13	0.64	0.59	0.017
Y2D	Base	0.24	1.27	0.013	0.015	0.19	0.14	0.13	0.62	0.61	0.017
Y22	Base	0.24	1.27	0.014	0.014	0.17	0.13	0.14	0.64	0.60	0.019
Y24	Base	0.24	1.27	0.015	0.013	0.18	0.14	0.14	0.65	0.61	0.019
Y25	Base	0.20	1.29	0.013	0.017	0.19	0.13	0.14	0.62	0.62	0.014
Y32	Base	0.20	1.33	0.014	0.013	0.18	0.13	0.14	0.62	0.59	0.014
Y4A	Weld	0.13	1.18	0.009	0.014	0.23	0.14	0.11	0.81	0.65	0.012
Y45	Weld	0.14	1.16	0.009	0.015	0.22	0.16	0.10	0.81	0.64	0.014
Y5M	Weld	0.13	1.16	0.010	0.014	0.25	0.17	0.10	0.78	0.60	0.014
Y6U	Weld	0.11	1.08	0.008	0.015	0.26	0.17	0.09	0.81	0.61	0.014
Y6Y	Weld	0.09	1.09	0.009	0.014	0.26	0.16	0.10	0.77	0.62	0.014
Y61	Weld	0.13	1.11	0.010	0.015	0.25	0.16	0.09	0.79	0.63	0.013
Y65	Weld	0.11	1.10	0.010	0.015	0.25	0.17	0.09	0.80	0.59	0.014
Y66	Weld	0.13	1.10	0.010	0.014	0.24	0.16	0.11	0.77	0.64	0.014
Y67	Weld	0.13	1.09	0.010	0.013	0.24	0.16	0.11	0.79	0.59	0.014
Y5T	Weld	0.12	1.15	0.010	0.015	0.25	0.16	0.09	0.81	0.61	0.014

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 Pilgrim 1 Nuclear Power Plant Reactor Vessel Irradiation Surveillance Program. The SwRI Nuclear Porjects Operating Procedures called out in this plan include:

(1) XI-MS-101-0, "Determination of Specific Activity and Analysis of Radiation Detector Specimens"

(2) XI-MS-103-0, "Conducting Tension Tests on Metallic Specimens"

(3) XI-MS-104-0, "Charpy Impact Tests on Metallic Specimens"

(4) XIII-MS-103-0, "Opening Radiation Surveillance Capsules and Handling and Storing Specimens"

Copies of the above documents are on file at SwRI.

A. Shipment, Opening, and Inspection of Capsule

After visually inspecting the basket and observing no damage, SwRI personnel assisted BeCo personnel in opening the basket and loading the impact and tensile capsules into the SwRI radioactive material shipping cask. SwRI personnel then transported the cask to San Antonio.

Upon arrival at SwRI, the individual capsules were inspected. Other than minor surface scuffing and discoloration, no physical damage was observed. Photographs showing the capsule identifications are included in Appendix C.

The capsules were opened and the contents identified and stored in accordance with Procedure XIII-MS-103-0. The end plugs were cut from each capsule with a band saw set up in the hot cell. 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 [13], 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 Procedure XI-MS-101-0 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 Mettler Type H6T balance. 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:

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

where: λ = decay constant for the activation product, day^{-1};

T_m = equivalent operating days at 1998 MwTh for operating period m; and

t_m = decay time after operating period m, days.

The Pilgrim 1 operating history up to the 1980 refuelling outage, which was used in the calculation of A_{TOR}, is presented in Table II.

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 Pilgrim 1 reactor vessel was performed to determine the radial and azimuthal dependence of the fast neutron (E > 1.0 MeV) flux density and energy spectrum within the reactor vessel and surveillance capsules. These results were used to calculate the spectrum-averaged cross sections for the threshold detectors and the lead factors for use in relating neutron exposure of the pressure vessel to that of the surveillance capsules. The pertinent factors obtained from this transport analysis are summarized in Table III.

The calculated cross section for the ^{54}Fe(n,p)^{54}Mn reaction is considerably higher than the 130 mb value assumed at the time of the vessel

TABLE II

OPERATIONS SUMMARY - PILGRIM NUCLEAR STATION

Operating Period, m	Dates		Shutdown Days	Operating Days	Reactor Power MWD_{th}	Equivalent[(a)] Operating Days, T_m	Decay Time in Days, t_m
	Start	Stop					
1	07/16/72	07/24/72	-	9	1,353	0.68	2716
	07/25/72	09/17/72	55	-	-	-	-
2	09/18/72	10/02/72	-	15	9,311	4.66	2646
	10/03/72	10/10/72	8	-	-	-	-
3	10/11/72	10/17/72	-	7	4,993	2.50	2631
	10/18/72	10/18/72	1	-	-	-	-
4	10/19/72	10/26/72	-	8	8,971	4.49	2622
	10/27/72	10/28/72	2	-	-	-	-
5	10/29/72	11/29/72	-	32	48,945	24.50	2588
	11/30/72	12/6/72	7	-	-	-	-
6	12/07/72	12/29/72	-	23	37,018	18.53	2558
	12/30/72	01/01/73	3	-	-	-	-
7	01/02/73	03/06/73	-	64	116,620	58.37	2491
	03/07/73	03/07/73	1	-	-	-	-
8	03/08/73	03/30/73	-	23	39,660	19.85	2467
	03/31/73	04/05/73	6	-	-	-	-
9	04/06/73	05/04/73	-	29	51,915	25.98	2432
	05/05/73	05/06/73	2	-	-	-	-
10	05/07/73	07/16/73	-	71	130,211	65.17	2359
	07/17/73	07/29/73	13	-	-	-	-
11	07/30/73	09/01/73	-	34	57,918	28.99	2312
	09/02/73	09/04/73	3	-	-	-	-
12	09/05/73	09/08/73	-	4	3,298	1.65	2305
	09/09/73	09/10/73	2	-	-	-	-
13	09/11/73	11/07/73	-	58	79,361	39.72	2245
	11/08/73	11/11/73	4	-	-	-	-
14	11/12/73	11/20/73	-	9	8,001	4.00	2232
	11/21/73	11/21/73	1	-	-	-	-
15	11/22/73	12/07/73	-	16	15,788	7.90	2215
	12/08/73	12/08/73	1	-	-	-	-
16	12/09/73	12/28/73	-	20	19,699	9.86	2194
	12/29/73	07/26/74	210	-	-	-	-
17	07/27/74	09/17/74	-	53	83,968	42.03	1931
	09/18/74	09/21/74	4	-	-	-	-
18	09/22/74	10/24/74	-	33	61,866	30.96	1894
	10/25/74	10/26/74	2	-	-	-	-
19	10/27/74	11/02/74	-	7	10,176	5.09	1885
	11/03/74	11/03/74	1	-	-	-	-
20	11/04/74	12/12/74	-	39	68,252	34.16	1845
	12/13/74	12/16/74	4	-	-	-	-
21	12/17/74	01/10/75	-	25	42,739	21.39	1816
	01/11/75	01/23/75	13	-	-	-	-
22	01/24/75	01/30/75	-	7	5,596	2.80	1796
	01/31/75	02/11/75	12	-	-	-	-
23	02/12/75	04/15/75	-	63	90,383	45.24	1721
	04/16/75	04/20/75	5	-	-	-	-
24	04/21/75	04/22/75	-	2	1,687	0.84	1714
	04/23/75	04/28/75	6	-	-	-	-
25	04/29/75	05/21/75	-	23	28,558	14.29	1685
	05/22/75	05/24/75	3	-	-	-	-
26	05/25/75	06/30/75	-	37	48,593	24.32	1645
	07/01/75	07/05/75	5	-	-	-	-
27	07/06/75	07/20/75	-	15	15,864	7.94	1625
	07/21/75	07/21/75	1	-	-	-	-

TABLE II (CONT.)

Operating Period, m	Dates		Shutdown Days	Operating Days	Reactor Power MWD_{th}	Equivalent[(a)] Operating Days, T_m	Decay Time in Days, t_m
	Start	Stop					
28	07/22/75	08/08/75	-	18	22,178	11.10	1606
	08/09/75	08/09/75	1	-	-	-	-
29	08/10/75	08/18/75	-	9	10,347	5.43	1596
	08/19/75	08/19/75	1	-	-	-	-
30	08/20/75	09/10/75	-	22	25,332	12.68	1573
	09/11/75	09/11/75	1	-	-	-	-
31	09/12/75	09/13/75	-	2	230	.12	1570
	09/14/75	10/27/75	44	-	-	-	-
32	10/28/75	01/05/76	-	70	75,815	37.95	1456
	01/06/76	01/18/76	13	-	-	-	-
33	01/19/76	01/27/76	-	9	7,452	3.73	1434
	01/28/76	05/31/76	125	-	-	-	-
34	06/01/76	01/28/77	-	242	350,129	175.24	1067
	01/29/77	02/02/77	5	-	-	-	-
35	02/03/77	03/22/77	-	48	80,232	40.16	1014
	03/23/77	03/23/77	1	-	-	-	-
36	03/24/77	05/01/77	-	39	61,553	30.81	974
	05/02/77	05/02/77	1	-	-	-	-
37	05/03/77	05/10/77	-	8	7,189	3.60	965
	05/11/77	05/11/77	1	-	-	-	-
38	05/12/77	08/06/77	-	87	138,608	69.37	877
	08/07/77	11/16/77	102	-	-	-	-
39	11/17/77	11/17/77	-	1	147	0.07	774
	11/18/77	11/21/77	4	-	-	-	-
40	11/22/77	11/22/77	-	1	9	0.01	769
	11/23/77	12/11/77	19	-	-	-	-
41	12/12/77	01/10/78	-	30	18,981	9.50	720
	01/11/78	02/04/78	25	-	-	-	-
42	02/05/78	02/06/78	-	2	252	0.13	693
	02/07/78	02/07/78	1	-	-	-	-
43	02/08/78	04/07/78	-	59	103,787	51.94	633
	04/08/78	04/10/78	3	-	-	-	-
44	04/11/78	04/28/78	-	18	27,400	13.71	612
	04/29/78	05/05/78	7	-	-	-	-
45	05/06/78	10/09/78	-	157	291,537	145.91	448
	10/10/78	10/27/78	18	-	-	-	-
46	10/28/78	03/25/79	-	149	286,368	143.33	281
	03/26/79	03/29/79	4	-	-	-	-
47	03/30/79	05/11/79	-	43	84,057	42.07	234
	05/12/79	05/29/79	18	-	-	-	-
48	05/30/79	06/15/79	-	17	31,975	16.00	199
	06/16/79	06/16/79	1	-	-	-	-
49	06/17/79	07/11/79	-	25	42,471	21.26	173
	07/12/79	07/16/79	5	-	-	-	-
50	07/17/79	07/27/79	-	11	14,879	7.45	157
	07/28/79	07/29/79	2	-	-	-	-
51	07/30/79	12/31/79	-	155	273,292	136.78	0
				Totals	3,045,464	1524.26[(b)]	

(a) At 1998 MW_t

(b) 1524.25 = 4.17 effective full power years

TABLE III

RESULTS OF DISCRETE ORDINATES Sn TRANSPORT ANALYSIS
PILGRIM 1

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

Reaction	$\bar{\sigma}$ (barns)
$^{54}Fe(n,p)^{54}Mn$	0.194
$^{58}Ni(n,p)^{58}Co$	0.237
$^{63}Cu(n,\alpha)^{60}Co$	0.00327

B. Calculated Capsule Lead Factors

Position[(a)]	Location within Vessel Wall	Lead Factor[(b)]
30°	I.D. Surface	0.87
30°	1/4T	1.3
30°	3/4T	3.8

(a) Azimuthal position from reference

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

wall dosimeter analysis [15]. It is believed that the values given in Table III are valid because SwRI-computed reaction cross sections for Browns Ferry Unit 3 [16] agree well with those measured by General Electric during the same core cycle [17].

The dosimetry results are presented in Table IV. A summary of the capsule and vessel I.D. fluxes calculated for full-power operation is as follows:

Dosimeter Type	Measured Capsule Flux $cm^{-2} \cdot sec^{-1}$, E > 1 MeV	Lead Factor	Peak Vessel Flux at I.D. $cm^{-2} \cdot sec^{-1}$, E > 1 MeV
Copper	1.93×10^9	0.87	2.2×10^9
Iron	1.55×10^9*	0.87	1.8×10^9
Nickel	1.73×10^9	0.87	2.0×10^9

* 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 3.1×10^9 $cm^{-2} \cdot sec^{-1}$. 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 and in the reaction cross sections. Other neutronic factors contributing to the estimated ± 16.5% uncertainty (1σ) in a calculated flux value are the determination of disintegration rates and the calculation of reaction rates (A_{SAT}/N_0).

Averaging the results obtained from all neutron dosimeters, the neutron flux at the surveillance capsule location during fuel cycle 1 is

TABLE IV

SUMMARY OF NEUTRON DOSIMETRY RESULTS
PILGRIM 1, FIRST SURVEILLANCE SPECIMEN CAPSULE

Dosimeter I.D.		Activation Reaction	Weight (mg)	A_{TOR} (dps/mg)	A_{SAT} (dps/mg)	Flux, E > 1 MeV(a) (cm^{-2}·sec^{-1})
Material	Capsule					
Fe	G1	^{54}Fe(n,p)^{54}Mn	156.3	1.569 x 10^2	1.84 x 10^2	1.52 x 10^9
	G2		151.8	1.567 x 10^2	1.84 x 10^2	1.52 x 10^9
	G3		168.5	1.674 x 10^2	1.97 x 10^2	1.62 x 10^9
					Average =	1.55 x 10^9
Cu	G1	^{63}Cu(n,α)^{60}Co	232.1	1.461 x 10^1	3.94 x 10^1	1.84 x 10^9
	G2		252.9	1.588 x 10^1	4.28 x 10^1	2.00 x 10^9
	G3		246.7	1.547 x 10^1	4.17 x 10^1	1.95 x 10^9
					Average =	1.93 x 10^9
Ni	G1	^{58}Ni(n,p)^{58}Co	328.6	2.443 x 10^3	2.71 x 10^3	1.63 x 10^9
	G2		376.0	2.627 x 10^3	2.91 x 10^3	1.76 x 10^9
	G3		384.7	2.703 x 10^3	3.00 x 10^3	1.81 x 10^9
					Average =	1.73 x 10^9

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

calculated to be 1.74×10^9 $cm^{-2} \cdot sec^{-1}$, $E > 1$ MeV. The peak full power neutron flux for the pressure vessel I.D. is given by:

$$1.74 \times 10^9 \div 0.87 = 2.0 \times 10^9 .$$

Since Pilgrim 1 operated for 4.17 effective full power years (EFPY) up to the October 1979 refuelling, the calculated capsule and vessel fluences to that time are as follows:

- Surveillance Capsule - 2.3×10^{17} n/cm^2
- Pressure Vessel I.D. Surface - 2.6×10^{17} n/cm^2
- Pressure Vessel 1/4T - 1.8×10^{17} n/cm^2
- Pressure Vessel 3/4T - 6.0×10^{16} n/cm^2

C. Mechanical Property Tests

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

The irradiated Charpy V-notch specimens were tested on a SATEC impact machine in accordance with Procedure XI-MS-104-0. The test temperatures were selected to develop the ductile-brittle transition and upper shelf regions. The Charpy V-notch impact data obtained on the irradiated materials are presented in Tables VI through VIII. The Charpy V-notch transition curves for the plate material, the weld metal, and the HAZ material are compared to the unirradiated data in Figures 2 through 4. The radiation-induced shift in transition temperatures are indicated at the 50 ft-lb, 30 ft-lb, and 35 mil lateral expansion levels. A summary of the shifts in RT_{NDT} and C_v upper shelf energies for each material are presented in Table IX.

TABLE V

HARDNESS PROPERTIES OF SURVEILLANCE MATERIALS
PILGRIM 1

Test Material	Charpy Specimen No.		Hardness HRA	Hardness HRB(a)
Base Metal	Y14		55.0	
			55.2	
			54.8	
			54.8	
			55.7	
		Avg.	55.1	90
Weld Metal	Y41		54.2	
			54.2	
			54.2	
			53.9	
			54.1	
		Avg.	54.1	88
HAZ Material	Y73		56.8	
			57.2	
			55.8	
			55.9	
			56.2	
		Avg.	56.4	92

(a) Converted from HRA numbers per ASTM E 140-77.

TABLE VI

CHARPY V-NOTCH IMPACT PROPERTIES OF SURVEILLANCE MATERIALS
BASE METAL - PILGRIM 1

Specimen No.	Temp. (°F)	Energy (ft-lb)	Lateral Exp. (mils)	Fracture App. (% shear)
Y3P	-50	6.0	6	nil
Y2E	0	14.5	15	5
Y2B	25	17.0	18	10
Y3K	40	30.0	29	10
Y1U	50	41.0	37	20
Y35	60	55.0	47	20
Y2J	60	72.5	59	20
Y14	75	58.0	49	25
Y1P	110	90.0	68	60
Y15	150	112.5	82	100
Y16	210	118.0	89	100
Y1C	300	122.0	96	100

TABLE VII

CHARPY V-NOTCH IMPACT PROPERTIES OF SURVEILLANCE MATERIALS WELD METAL - PILGRIM 1

Specimen No.	Temp. (°F)	Energy (ft-lb)	Lateral Exp. (mils)	Fracture App. (% shear)
Y6J	-100	21.0	20	nil
Y6C	-75	40.0	33	5
Y6B	-50	52.0	45	10
Y5P	-25	37.5	42	10
Y5L	0	41.0	40	10
Y5A	25	57.0	53	20
Y53	50	61.0	59	30
Y41	75	69.5	64	60
Y4T	110	86.5	80	90
Y42	150	89.5	85	100
Y4E	210	93.0	88	100
Y4J	300	93.5	89	100

TABLE VIII

CHARPY V-NOTCH IMPACT PROPERTIES OF SURVEILLANCE MATERIALS
HAZ MATERIAL - PILGRIM 1

Specimen No.	Temp. (°F)	Energy (ft-lb)	Lateral Exp. (mils)	Fracture App. (% shear)
YBY	-100	9.0	8	nil
YBC	-100	40.0	28	5
YBM	-75	48.5	44	10
YBA	-75	74.5	55	10
YBS	-50	83.5	55	20
YB3	-25	100.0	68	50
YAD	0	70.0	51	20
YA1	25	105.0	70	60
Y73	75	129.5	80	100
Y75	150	91.0	67	100
Y7D	210	110.5	89	100
Y7Y	300	90.5	85	100

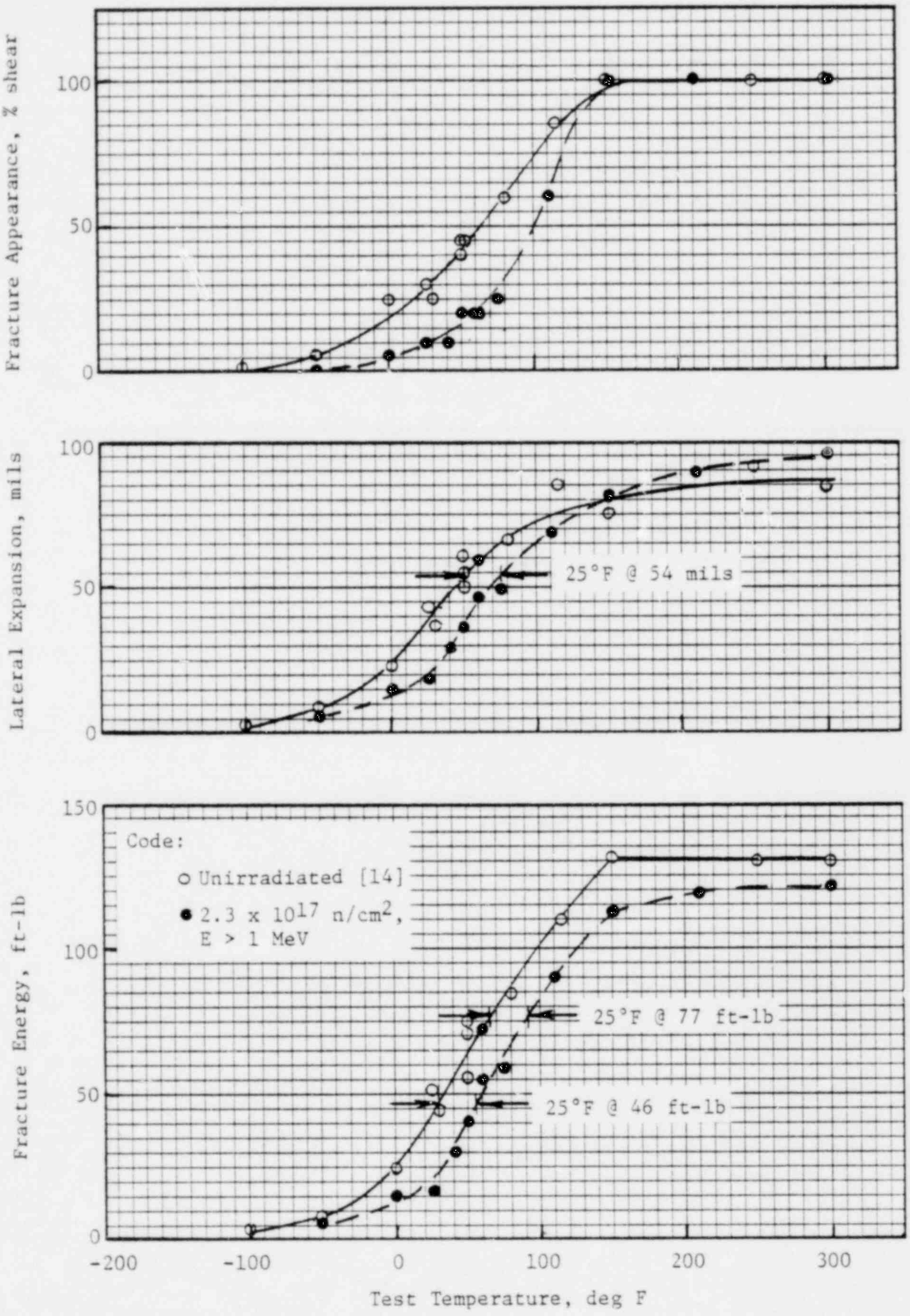


FIGURE 2. LONGITUDINAL BASE METAL CHARPY V-NOTCH IMPACT PROPERTIES, PILGRIM 1

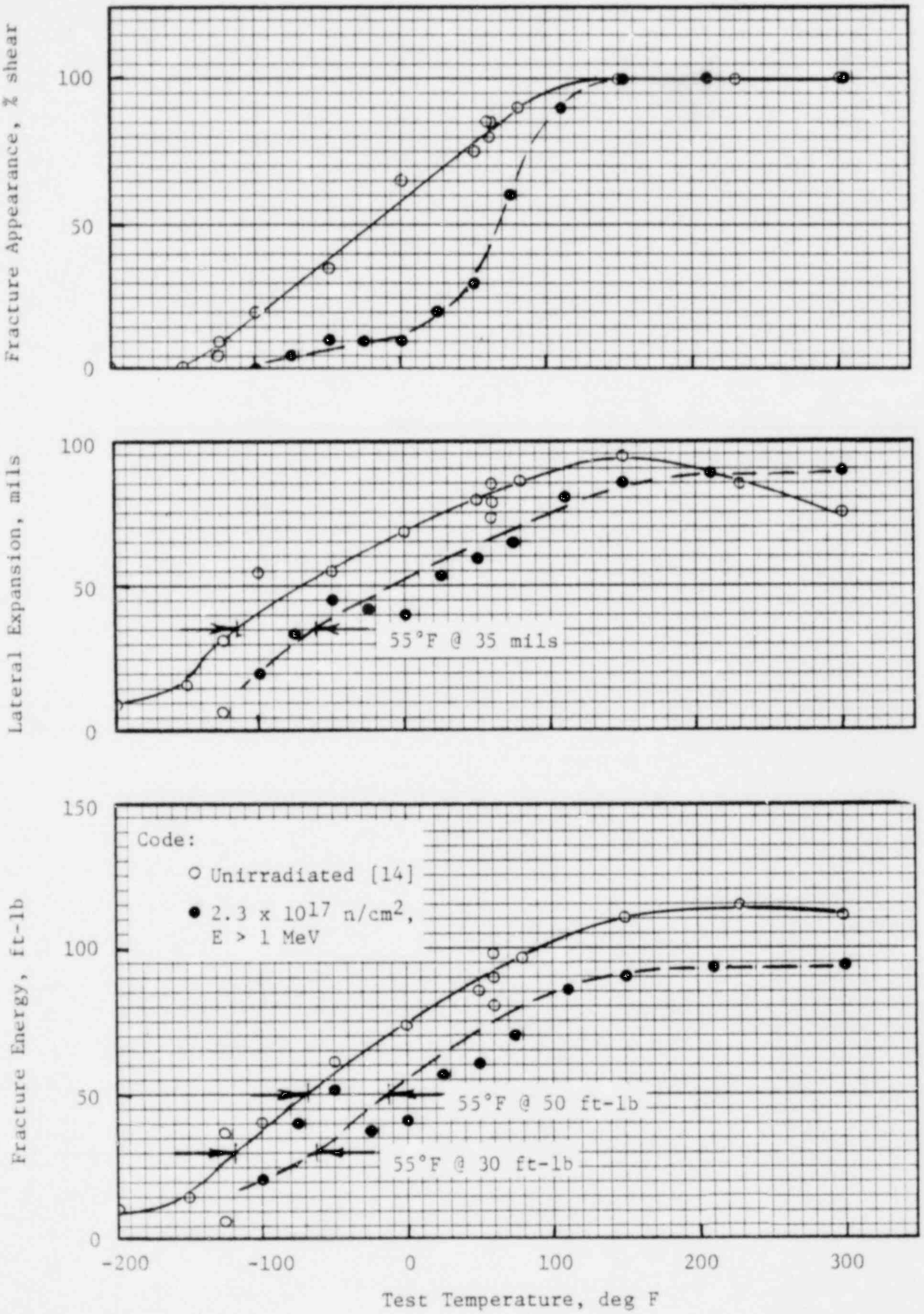


FIGURE 3. WELD METAL CHARPY V-NOTCH IMPACT PROPERTIES, PILGRIM 1

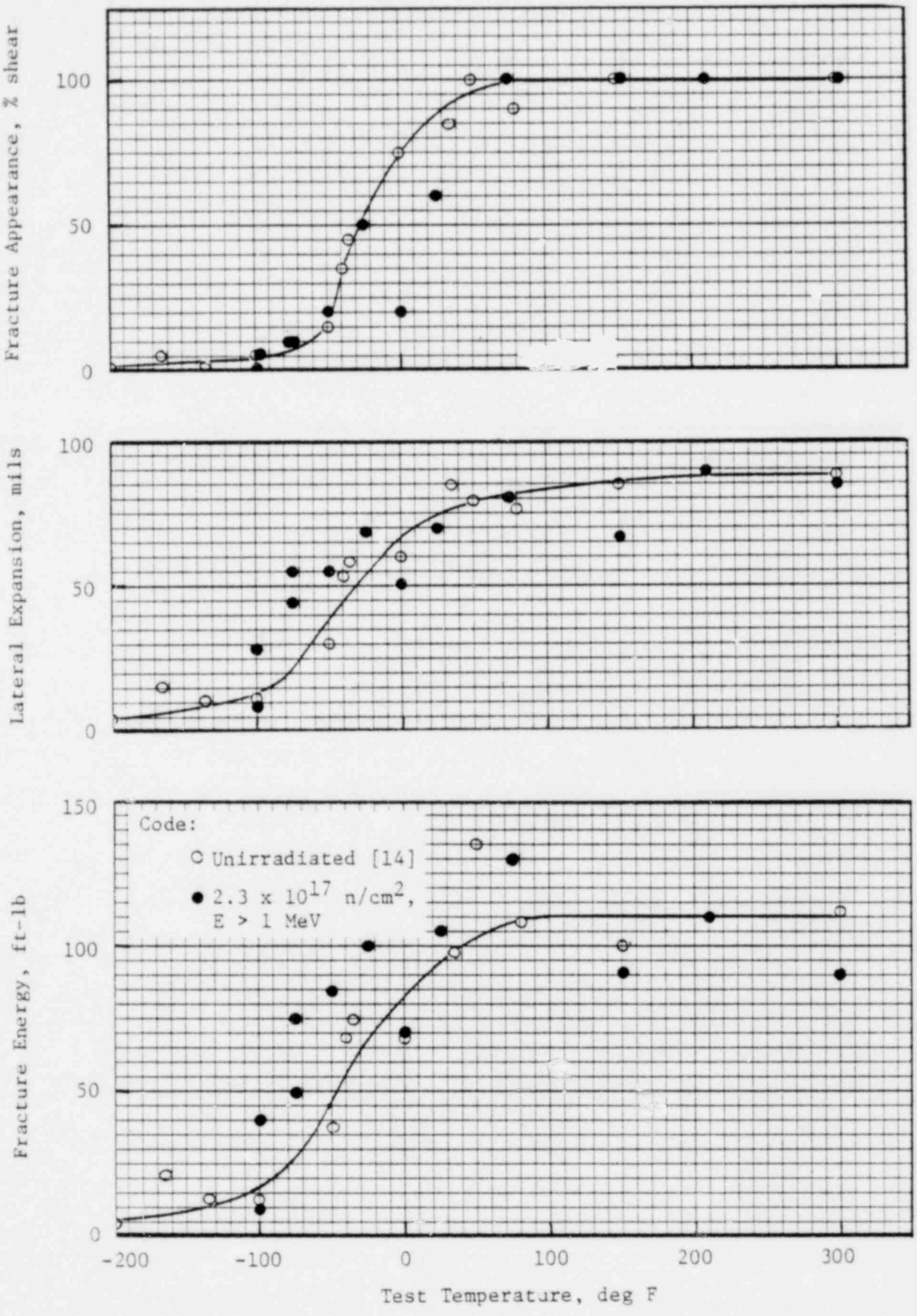


FIGURE 4. HAZ MATERIAL CHARPY V-NOTCH IMPACT PROPERTIES, PILGRIM 1

TABLE IX

EFFECT OF IRRADIATION ON VESSEL SURVEILLANCE MATERIALS
PILGRIM 1

Criterion(1)	Base Metal(2)	Weld Metal	HAZ Material
Transition Temperature Shift			
@ 50 ft-lb	25°F	55°F	nil
@ 30 ft-lb	25°F	55°F	nil
@ 35 mil	25°F	55°F	nil
ΔRT_{NDT}	25°F	55°F	nil
C_V Upper Shelf Drop	13 ft-lb (10%)	21 ft-lb (18%)	11 ft-lb (9%)

---

(1) Refer to Figures 2-4.

(2) Longitudinal orientation:
Use 77 ft-lb shift to estimate 50 ft-lb shift for transverse orientation.
Use 46 ft-lb shift to estimate 30 ft-lb shift for transverse orientation.
Use 54 mil shift to estimate 35 mil shift for transverse orientation.

Tensile tests were carried out on the irradiated materials in accordance with Procedure XI-MS-103-0 using a 50-kip capacity tester equipped with a strain gage extensometer, load cell, and autographic recording equipment. Tensile tests were run at room temperature, 150°F, and 550°F. The results are presented in Table X. The load-strain records and photographs of the tested specimens are included in Appendix D.

D. Check Analyses for Copper

One tested Charpy specimen from each material group was analyzed for copper content with an X-ray fluorescent technique per ASTM E 322-67 [19]. The results, summarized in Table XI, agree well with the analyses performed on the unirradiated specimens [14].

TABLE X

TENSILE PROPERTIES OF SURVEILLANCE MATERIALS
PILGRIM 1

Test Material	Spec. No.	Temp. (°F)	0.2% YS (ksi)	UTS (ksi)	Fracture Load (lb)	Fracture Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	R.A. (%)
Base Metal	YCM	72	71.2	93.0	2770	194	15.5	25.4	70.9
	YCA	150	67.9	89.4	2704	189	16.9	26.0	71.1
	YC1	550	61.5	85.2	2789	181	13.1	22.3	68.6
Weld Metal	YD7	150	69.7	87.7	2670	164	15.9	25.8	66.8
	YD5	550	60.3	81.1	2780	162	15.1	23.0	65.0
HAZ Material	YEB	72	71.7	89.8	2770	185	13.9	21.5	69.5
	YEL	72	70.1	88.0	2608	174	14.3	23.8	69.5
	YEK	550	55.9	79.5	2678	160	12.5	20.4	66.0

TABLE XI

CHECK COPPER ANALYSES

Material	Specimen No.	Copper (%)	Note
Base Metal	Y3P	0.14	(a)
Weld Metal	Y6J	0.15	(b)
HAZ Material	YBY	0.17	(c)

(a) Average copper content of ten unirradiated base metal specimens = 0.13 ± 0.01% [14].

(b) Average copper content of ten unirradiated weld metal specimens = 0.16 ± 0.01% [14].

(c) No data reported for HAZ specimens [14].

V. ANALYSIS OF RESULTS

The analysis of data obtained from surveillance program specimens has the following goals.

(1) Estimate the period of time over which the properties of the vessel beltline materials will meet the fracture toughness requirements of Appendix G of 10CFR50. This requires a projection of the measured reduction in C_v upper shelf energy to the vessel wall using knowledge of the energy and spatial distribution of the neutron flux and the dependence of C_v upper shelf energy on the neutron fluence.

(2) Determine the adjustment of the pressure vessel RT_{NDT} as a function of plant operations. This requires a projection of the measured shift in RT_{NDT} to the vessel wall using knowledge of the dependence of the shift in RT_{NDT} on the neutron fluence and the energy and spatial distribution of the neutron flux.

The energy and spatial distribution of the neutron flux for Pilgrim 1 was calculated with a discrete ordinates transport code. This analysis predicted that the lead factor (ratio of fast flux at the capsule basket location to the maximum pressure vessel flux) was 0.89. This analysis also predicted that the fast flux at the 1/4T and 3/4T positions in the 5.65-in. pressure vessel wall would be 67% and 23%, respectively, of that at the vessel I.D. The vessel wall fluence as a function of plant operation is shown in Figure 5.

A. RT_{NDT} Projections

A method for estimating the increase in RT_{NDT} as a function of neutron fluence and chemistry is given in Regulatory Guide 1.99, Revision 1 [7].

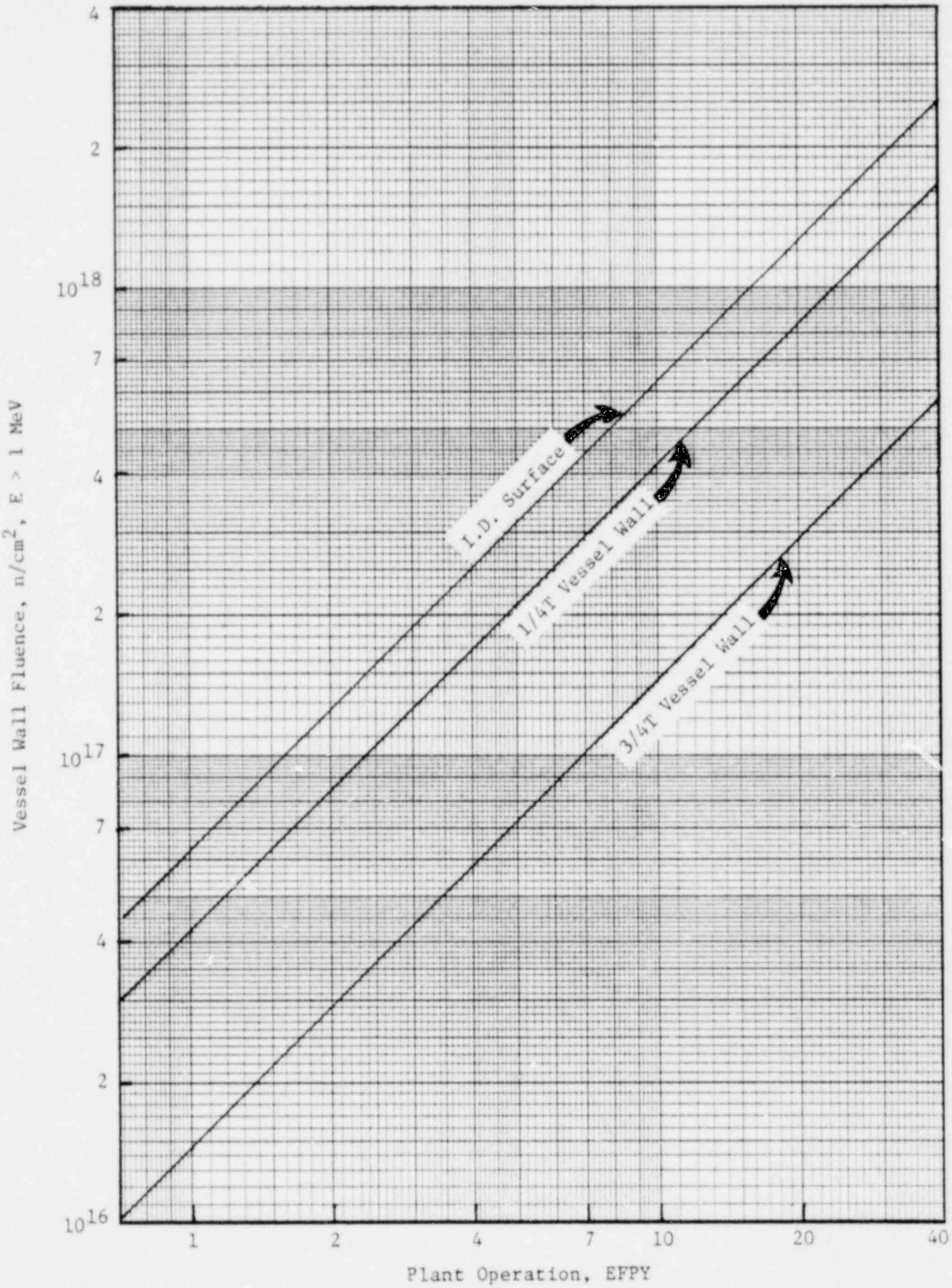


FIGURE 5. VESSEL WALL FLUENCE AS A FUNCTION OF OPERATION OF PILGRIM 1

However, the Guide also permits the extrapolation of credible surveillance data by constructing response curves through the data points and parallel to the Guide trend curves, as shown in Figure 6.

The Pilgrim 1 surveillance weld metal is more sensitive than the other core beltline surveillance materials to irradiation embrittlement. Also, since the RT_{NDT} of the unirradiated weld metal was higher than that of the unirradiated base material [14], the weld metal will control the adjusted value of RT_{NDT} throughout the 40-year design life of the plant. A summary of the projected values of RT_{NDT} for 12 and 32 EFPY is presented in Table XII.

The peak value of RT_{NDT} + 60°F as a function of neutron fluence and plant operation (in terms of EFPY) is shown in Figure 7. The horizontal line at $RT_{NDT} + 60°F = 100°F$ is based on the initial system RT_{NDT} [20]. After 3 EFPY, the value of RT_{NDT} + 60°F is controlled by the beltline weld metal.

B. Shelf Energy Projections

Regulatory Guide 1.99 also gives a method for estimating the reduction in C_v upper shelf energy as a function of neutron fluence [7]. The Pilgrim 1 surveillance results are compared to applicable Regulatory Guide 1.99 trend curves in Figure 8. The weld metal is more sensitive to neutron embrittlement than the base plate and HAZ materials, and also is more sensitive than predicted by the Regulatory Guide 1.99 trend curve for 0.15% Cu weld metal.

Using the extrapolated curve through the weld metal surveillance data point, a shelf energy reduction of approximately 30% is predicted

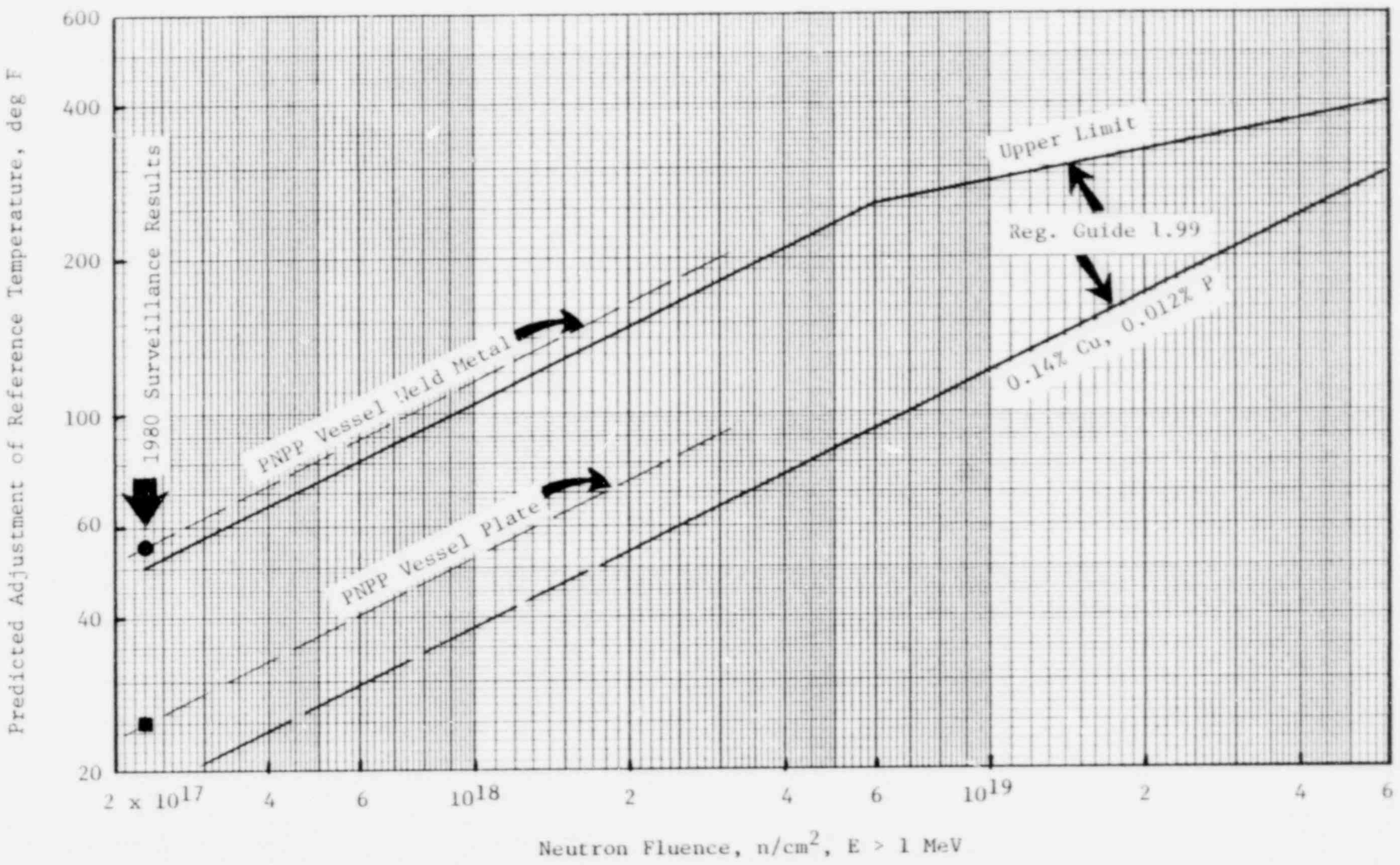


FIGURE 6. EFFECT OF NEUTRON FLUENCE ON RT_{NDT} SHIFT, PILGRIM 1

TABLE XII

PROJECTED VALUES OF RT_{NDT} FOR PILGRIM 1

Material Property Basis

Weld Metal Copper = 0.16%
Initial RT_{NDT} = 0°F

EFPY	Location	Fluence	ΔRT_{NDT}	Adj. RT_{NDT}
12	I.D.	7.5×10^{17}	100	100
	1/4T	5.1×10^{17}	82	82
	3/4T	1.7×10^{17}	47	47
32	I.D.	2.0×10^{18}	162	162
	1/4T	1.4×10^{18}	136	136
	3/4T	4.6×10^{17}	78	78

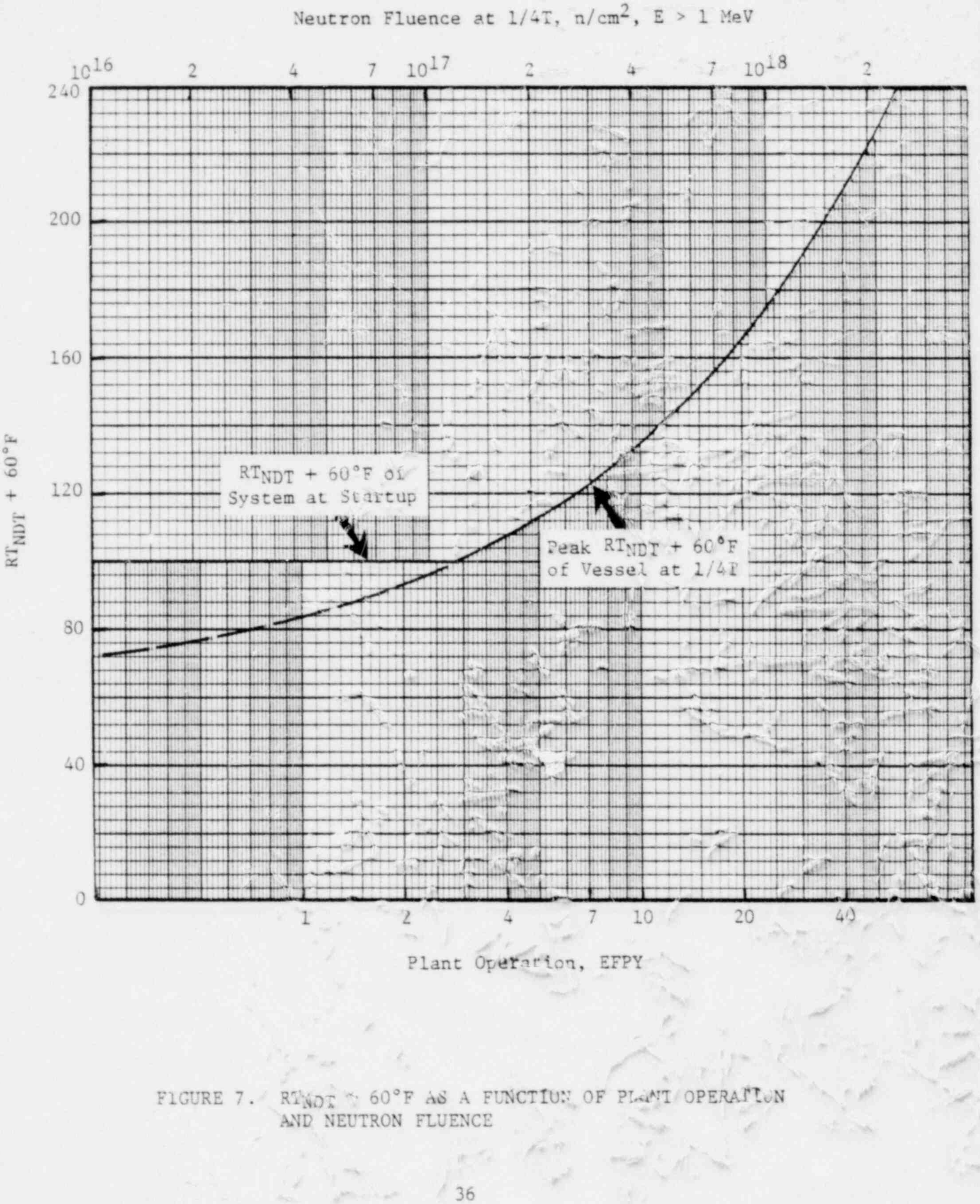


FIGURE 7. RT_{NDT} + 60°F AS A FUNCTION OF PLANT OPERATION AND NEUTRON FLUENCE

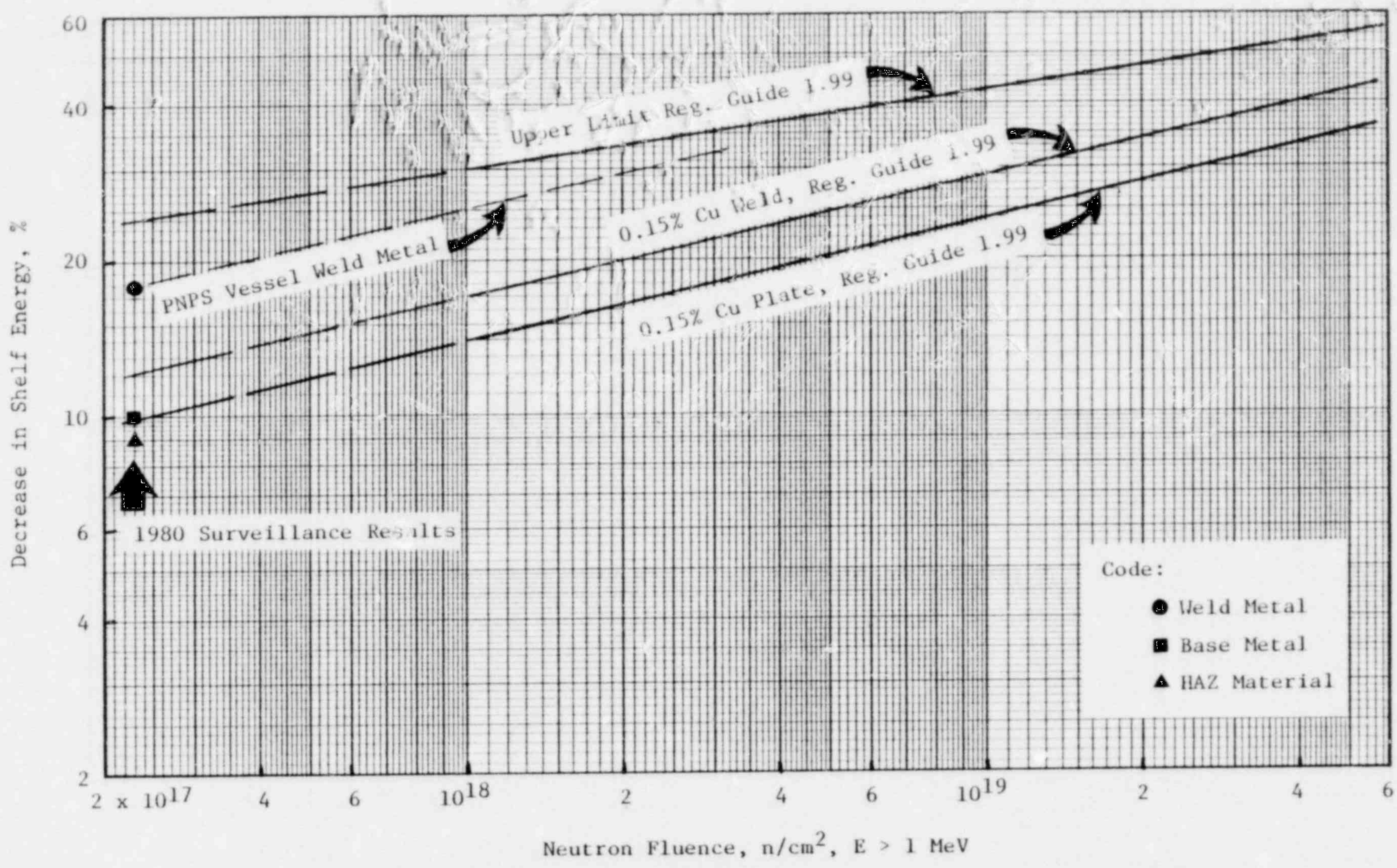


FIGURE 8. DEPENDENCE OF C_v UPPER SHELF ENERGY ON NEUTRON FLUENCE, PILGRIM 1

at the 32 EFPY vessel fluence of 2.0×10^{18} n/cm^2, E > 1 MeV. Since the initial value of weld metal toughness was 113 ft-lb, the projected weld metal shelf energy after 32 EFPY of operation is 79 ft-lb, well above the 50 ft-lb limit required by 10CFR50, Appendix 5 [1].

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," 1974 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. Steele, L. E., and Serpan, C. Z., Jr., "Analysis of Reactor Vessel Radiation Effects Surveillance Programs," ASTM STP 481, December 1970.

5. Steele, L. E., "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," 1974 Edition.

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

8. Comments on Regulatory Guide 1.99, Westinghouse Electric Corporation, obtained from NRC Public Document Room, Washington, D.C.

9. Position on Regulatory Guide 1.99, Combustion Engineering Power Systems, obtained from NRC Public Document Room, Washington, D.C.

10. "Prediction of Shift in the Ductile-Brittle Transition Temperature of LWR Pressure Vessel Materials," prepared by the Metal Properties Council, Subcommittee 6 on Nuclear Materials, July 1, 1980.

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

12. Higgins, J. P., "Modified Surveillance Program for General Electric BWR Pressure Vessel Steels," AEPD-5490, April 1967.

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

14. Perrin, J. S., et al, "Evaluation of Mechanical Properties of Unirradiated Specimens from Pilgrim Unit No. 1," October 25, 1979.

15. Norris, E. B., "Vessel Material Surveillance Program for Pilgrim Nuclear Power Station," Topical Report No. 1, Vessel Wall Dosimeter Analysis, SwRI Project 02-3812-001, August 15, 1974.

16. Norris, E. B., "Analysis of Vessel Wall Neutron Dosimeter from Browns Ferry Unit 3 Pressure Vessel," Final Report, SwRI Project 02-4884-003, September 1979.

17. Martin, G. C., "Browns Ferry Unit 3 In-Vessel Neutron Spectral Analysis," NEDO-24793, August 1980.

18. ASTM E 18-81, "Standard Methods of Test for Rockwell Hardness and Rockwell Superficial Hardness of Metallic Materials," 1981 Annual Book of ASTM Standards.

19. ASTM E 322-67, "Standard Method for Spectrochemical Analysis of Low-Alloy Steels and Cast Irons Using an X-ray Fluorescence Spectrometer," 1974 Annual Book of ASTM Standards.

20. Final Safety Analysis Report, Pilgrim Nuclear Power Station, p. 4.2-9, Revision June 11, 1970.

APPENDIX A

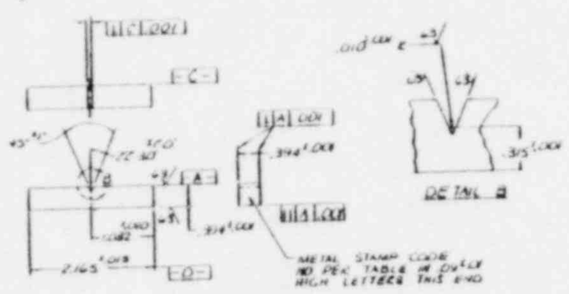
SURVEILLANCE SPECIMEN AND CAPSULE DRAWINGS [12]

GENERAL ELECTRIC

921D277

CHARPY IMPACT SPECIMEN

SURVEILLANCE PROGRAM



PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO RUN SERIALLY	PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO RUN SERIALLY	PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO RUN SERIALLY
1	BASE	21A1106	1İ1 TO 1ḂY	27	BASE	21A1112	D11 TO D3Y	54	BASE	21A1115	J81 TO JÖY
2	WELD		1Ċ1 TO 1ŸY	28	WELD		D41 TO D6Y	55	WELD		JĖ1 TO JĊY
3	HEAT AFFECTED ZONE (HAZ)		1D̈1 TO 2ŸY	29	HAZ		D71 TO D8Y	56	HAZ		JĹ1 TO JÄY
4	BASE	21A1109	5İ1 TO 5ṠY	30	BASE	21A1114	3K1 TO 3[illegible]Y	57	BASE	21A1100AJ	EP1 TO EÜY
5	WELD (SUB ARC)		6İ1 TO 6ṠY	31	WELD		3Ÿ1 TO 4ÏY	58	WELD		EŸ1 TO JŻY
6	WELD (ELECTROSLAG)		5Ġ1 TO 5ĖY	32	HAZ		421 TO 4ŠY	59	HAZ		J31 TO JŠY
7	HAZ (SUB ARC)		6Ġ1 TO 6ĊY	33	BASE	21A1111	756 TO 7ŸY	60	BASE	21A1100AF	5Ÿ1 TO 5ŠY
8	HAZ (ELECTROSLAG)		5J1 TO 5ÜY	34	WELD		7Ä1 TO 7ĊL	61	WELD		5Ä1 TO 5ĊY
9	BASE	21A1109	Kİ1 TO KṠY	35	HAZ		7ĊM TO 7ŸC	62	HAZ		5Ÿ1 TO 5B̈Y
10	WELD (SUB ARC)		Lİ1 TO LṠY	36	BASE	21A1111	7ĊM TO 7ŸC	63	BASE	21A1112	JĊP TO JËE
11	WELD (ELECTROSLAG)		KĠ1 TO KĖY	37	WELD		7ŸD TO AÏS	64	WELD		JËJ TO JK̈A
12	HAZ (SUB ARC)		LĠ1 TO LĊY	38	HAZ		AÏ6 TO A3Y	65	HAZ		JK̈B TO JḦ4
13	HAZ (ELECTROSLAG)		KJ1 TO LÜY	39	BASE	21A1111	DJ1 TO DĹY				
14	BASE	21A1113	Mİ1 TO MṠY	40	WELD		DM1 TO DTY				
15	WELD (SUB ARC)		Pİ1 TO PṠY	41	HAZ		DÜ1 TO EİY				
16	WELD (ELECTROSLAG)		MĠ1 TO MĖY	42	BASE	21A1111	ES1 TO EİY				
17	HAZ (SUB-ARC)		PĠ1 TO PĊY	43	WELD		EÄ1 TO EĊY				
18	HAZ (ELECTROSLAG)		MJ1 TO MÜY	44	HAZ		EÖ1 TO EJY				
19	BASE	21A1113	Tİ1 TO TṠY	45	BASE	21A1111	6Ä5 TO 6ĠY				
20	WELD (SUB ARC)		Uİ1 TO UṠY	46	WELD		6Ÿ1 TO 6BL				
21	WELD (ELECTROSLAG)		TĠ1 TO TĖY	47	HAZ		6B̈M TO 6ËC				
22	HAZ (SUB ARC)		UĠ1 TO UĊY	48	BASE	21A1100AK	AĊ6 TO AÄY				
23	HAZ (ELECTROSLAG)		TJ1 TO TÜY	49	WELD		AB̈1 TO AÖL				
24	BASE	21A1110	Bİ1 TO BĊY	50	HAZ		AÖM TO AĊC				
25	WELD		BÖ1 TO BŸY	51	BASE	21A1110	Yİ1 TO Y3Y				
26	HEAT AFFECTED ZONE (HAZ)		C11 TO CĊY	52	WELD		Y41 TO YĊY				
				53	HAZ		Y71 TO Y8Y				

P60 TO P62 - 117C4351
P63 TO P65 117C3513

PC# P1 TO P3 - 117C3572
P4 TO P13 - 117C3728
P14 TO P23 - 117C3570
P24 TO P26 - 117C3769
P27 TO P29 - 117C3913
P30 TO P32 - 117C3911
P33 TO P38 - 117C4047
P39 TO P47 - 117C4053
P48 TO P50 - 117C4066
P51 TO P53 - 117C4073
P54 TO P56 - 117C4083
P57 TO P59 - 117C4191

NOTE:

1. 63√ SURFACES SHALL BE GROUND OR MACHINED WITHOUT NOTICABLE TOOL MARKS

ADDED PT 57 TO P59

ADDED PT 51 TO 53

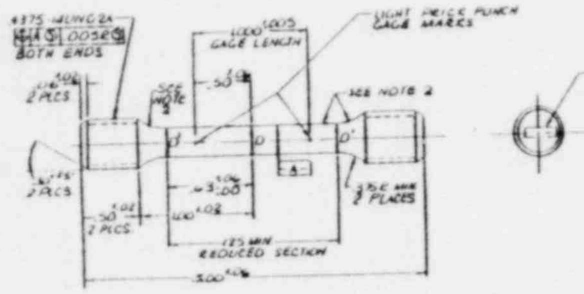
ADDED PT 37 TO 47

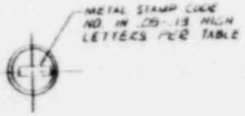
921D277

GENERAL ELECTRIC

921D276

TENSILE TEST SPECIMEN SURVEILLANCE PROGRAM





FOR P1 to P3 - 117C3572
P4 to P13 - 117C3724
P14 to P23 - 117C3731
P24 to P26 - 117C3433
P27 to P29 - 117C3312
P30 to P32 - 117C3370
P33 to P35 - 117C4046
P39 to P47 - 117C4057
P48 to P50 - 117C4065
P51 to P53 - 117C4072
P54 to P56 - 117C4032
P57 to P59 - 117C4152
P60 to P62 - 117C4350
P63 to P65 - 117C3312

NOTES:

1. D = .250 ±.001 DIA AT CENTER OF REDUCED SECTION.
D' = ACTUAL D DIA +.002 TO .005 AT ENDS OF REDUCED SECTION TAPERING TO D AT CENTER.
2. GRIND REDUCED SECTION & RADII TO 32√. RADII TO BE TANGENT TO REDUCED SECTION WITH NO CIRCULAR TOOL MARKS AT POINT OF TANGENCY OR WITHIN REDUCED SECTION. POINT OF TANGENCY SHALL NOT LIE WITHIN REDUCED SECTION.

PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO BE STAMPED SERIALLY	PART	MATERIAL TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO BE STAMPED SERIALLY	PART	MAT'L TYPE	MATERIAL SPECIFICATION	FAB CODE NO PER 158B7275 NO TO BE STAMPED SERIALLY
1	BASE	21A1106	2Ä1 TO 2B̈Y	27	BASE	21A1112	DĊ1 TO DĊY	54	BASE	21A1115	JṪ1 TO JṪY
2	WELD		2C̈1 TO 2D̈Y	28	WELD		DḊ1 TO DḊY	55	WELD		JU̇1 TO JU̇Y
3	HEAT AFFECTED ZONE (HAZ)		2Ë1 TO 2J̈Y	29	HAZ		DĖ1 TO DĖY	56	HAZ		JẎ1 TO JẎY
4	BASE	21A1109	6Ḋ1 TO 6ĖY	30	BASE	21A1114	4G̈1 TO 4G̈Y	57	BASE	21A1100AJ	JĊ1 TO JĊY
5	WELD (SUB ARC)		6Ṗ1 TO 6İY	31	WELD		47̈1 TO 47̈Y	58	WELD		J7̇1 TO J7̇Y
6	WELD (ELECTROSLAG)		6J1 TO 6K̇Y	32	HAZ		4Ä1 TO 4ÄY	59	HAZ		JȦ1 TO JȦY
7	HAZ (SUB ARC)		6U̇1 TO 6ẎY	33	BASE	21A1111	7J̈D TO 7K̈S	60	BASE	21A1100AF	5C̈1 TO 5C̈Y
8	HAZ (ELECTROSLAG)		6L̇1 TO 6ṀY	34	WELD		7K̈6 TO 7K̈Y	61	WELD		5D̈1 TO 5D̈Y
9	BASE	21A1109	LḊ1 TO LĖY	35	HAZ		7L̈1 TO 7L̈L	62	HAZ		5Ë1 TO 5ËY
10	WELD (SUB ARC)		LṖ1 TO LİY	36	BASE	21A1111	A4̈1 TO A4̈L	63	BASE	21A1112	JB̈1 TO JB̈E
11	WELD (ELECTROSLAG)		LJ1 TO LK̇Y	37	WELD		A4̈M TO AS̈C	64	WELD		JB̈J TO JC̈4
12	HAZ (SUB ARC)		LU̇1 TO LẎY	38	HAZ		AS̈D TO AG̈S	65	HAZ		JC̈5 TO JC̈M
13	HAZ (ELECTROSLAG)		LL̇1 TO LṀY	39	BASE	21A1111	E2̇1 TO E2̇Y				
14	BASE	21A1113	PḊ1 TO PĖY	40	WELD		E3̇1 TO E3̇Y				
15	WELD (SUB ARC)		PṖ1 TO PİY	41	HAZ		E4̇1 TO E4̇Y				
16	WELD (ELECTROSLAG)		PJ̇1 TO PK̇Y	42	BASE	21A1111	EK̇1 TO EK̇Y				
17	HAZ (SUB ARC)		PU̇1 TO PẎY	43	WELD		EL̇1 TO EL̇Y				
18	HAZ (ELECTROSLAG)		PL̇1 TO PṀY	44	HAZ		EṀ1 TO EṀY				
19	BASE	21A1113	UḊ1 TO UĖY	45	BASE	21A1111	6ËD TO 6J̈S				
20	WELD (SUB ARC)		UṖ1 TO UİY	46	WELD		6J̈C TO 6ŸY				
21	WELD (ELECTROSLAG)		UJ̇1 TO UK̇Y	47	HAZ		6K̈1 TO 6K̈L				
22	HAZ (SUB ARC)		UU̇1 TO UẎY	48	BASE	21A1100AK	AK̈D TO AL̈S				
23	HAZ (ELECTROSLAG)		UL̇1 TO UṀY	49	WELD		AL̈6 TO AL̈Y				
24	BASE	21A1110	CḊ1 TO CJ̇Y	50	HAZ		AM̈1 TO AM̈L				
25	WELD		CK̇1 TO CṀY	51	BASE	21A1110	YĊ1 TO YĊY				
26	HEAT AFFECTED ZONE (HAZ)		CȦ1 TO CU̇Y	52	WELD		YḊ1 TO YḊY				
				53	HAZ		YĖ1 TO YĖY				

921D276

GENERAL ELECTRIC

117C3913

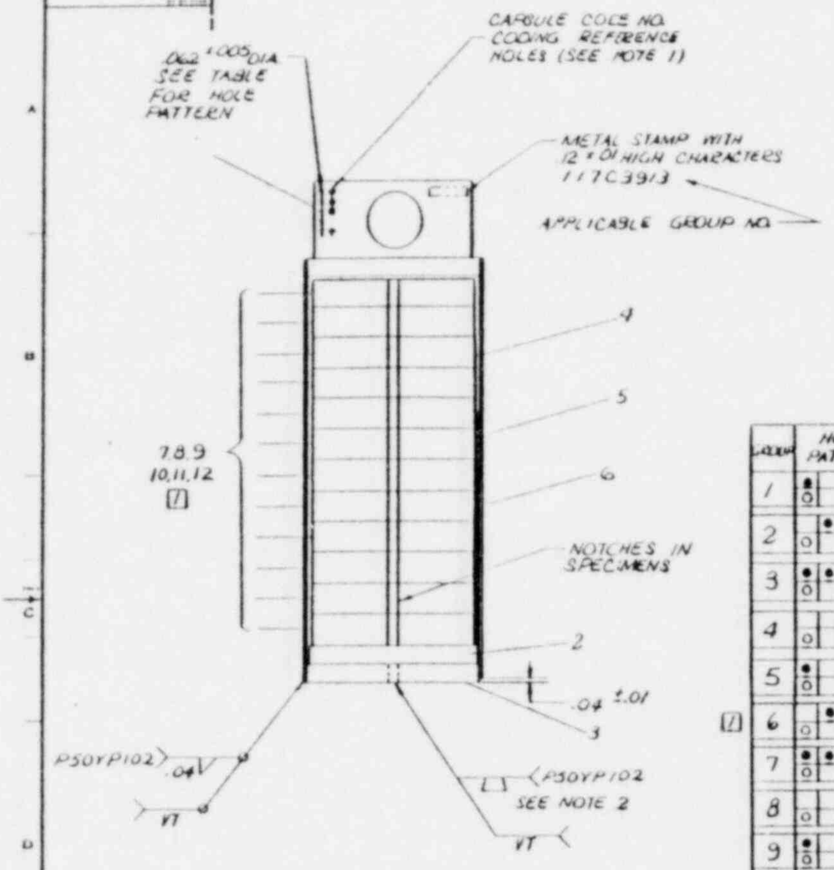
145A5451	✓				

IMPACT SPECIMEN CAPSULE

SURVEILLANCE PROGRAM

PL ISSUED

FCF: G1-G5-117C3911
G6-G10-117C4370



	GROUP	HOLE PATTERN				
	1	● ○		○	○	○
	2	○	●	○	○	○
	3	● ○	●	○	○	○
	4	○		● ○	○	○
	5	● ○		● ○	○	○
1	6	○	●	● ○	○	○
	7	● ○	●	● ○	○	○
	8	○		○	● ○	○
	9	● ○		○	● ○	○
1	10	○	●	○	● ○	○

NOTES:

1. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW.
   1 2 4 8 16
   ● ● ● ● ●
   ● ● ● ● ● - CODING REFERENCE HOLES
2. EXHAUST AIR TO REDUCE CAPSULE INTERNAL PRESSURE TO ONE (1) TORR, BACKFILL WITH WELDING GRADE HELIUM AT ATMOSPHERIC PRESSURE AND SEAL WELD HOLE.
3. MASS SPECTROMETER LEAK TEST COMPLETED ASSEMBLY. ALLOWABLE LEAKAGE 1×10^{-6} CC/MIN.

117C3913

1

REVISIONS

1 NE-22234

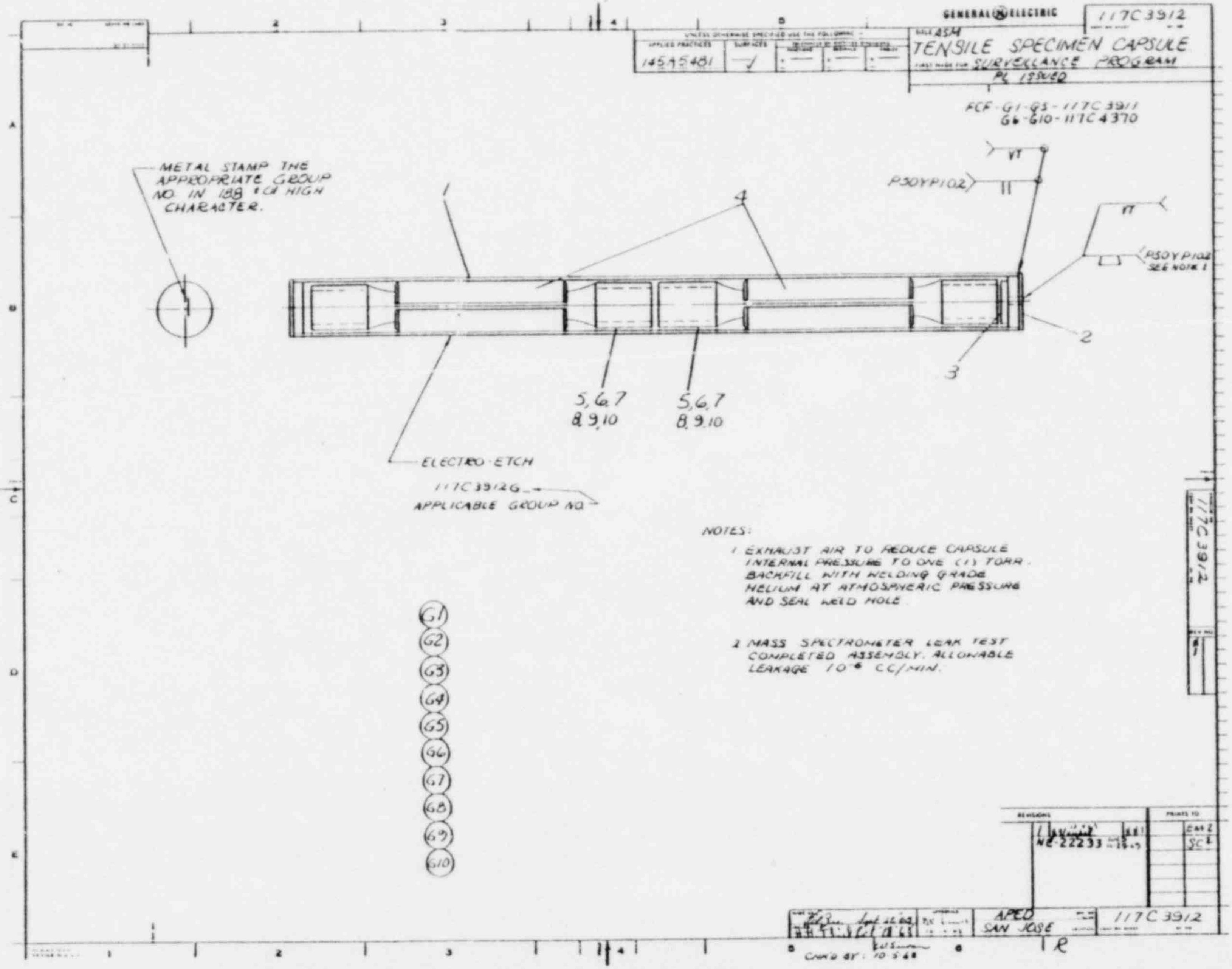
PRINTS TO

EM 2
SC 2

DR
CHK'D BY W. Swan 10-3-68

APED
SAN JOSE

117C3913



GENERAL ELECTRIC

117C3912

ASM
TENSILE SPECIMEN CAPSULE
SURVEILLANCE PROGRAM
PL ISSUED

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING

APPLIED PRACTICES	SURFACES
145A5481	√

FCF-G1-G5-117C3911
G6-G10-117C4370

NOTES:

1 EXHAUST AIR TO REDUCE CAPSULE INTERNAL PRESSURE TO ONE (1) TORR. BACKFILL WITH WELDING GRADE HELIUM AT ATMOSPHERIC PRESSURE AND SEAL WELD HOLE.

2 MASS SPECTROMETER LEAK TEST COMPLETED ASSEMBLY. ALLOWABLE LEAKAGE 10^{-6} CC/MIN.

REVISIONS

1 NE-22233

PRINTS TO

EAH Z
SC 1

APED
SAN JOSE

117C3912

CHK'D BY: 10-5-68

1 R

MC-10.2

GENERAL ELECTRIC

117C4074

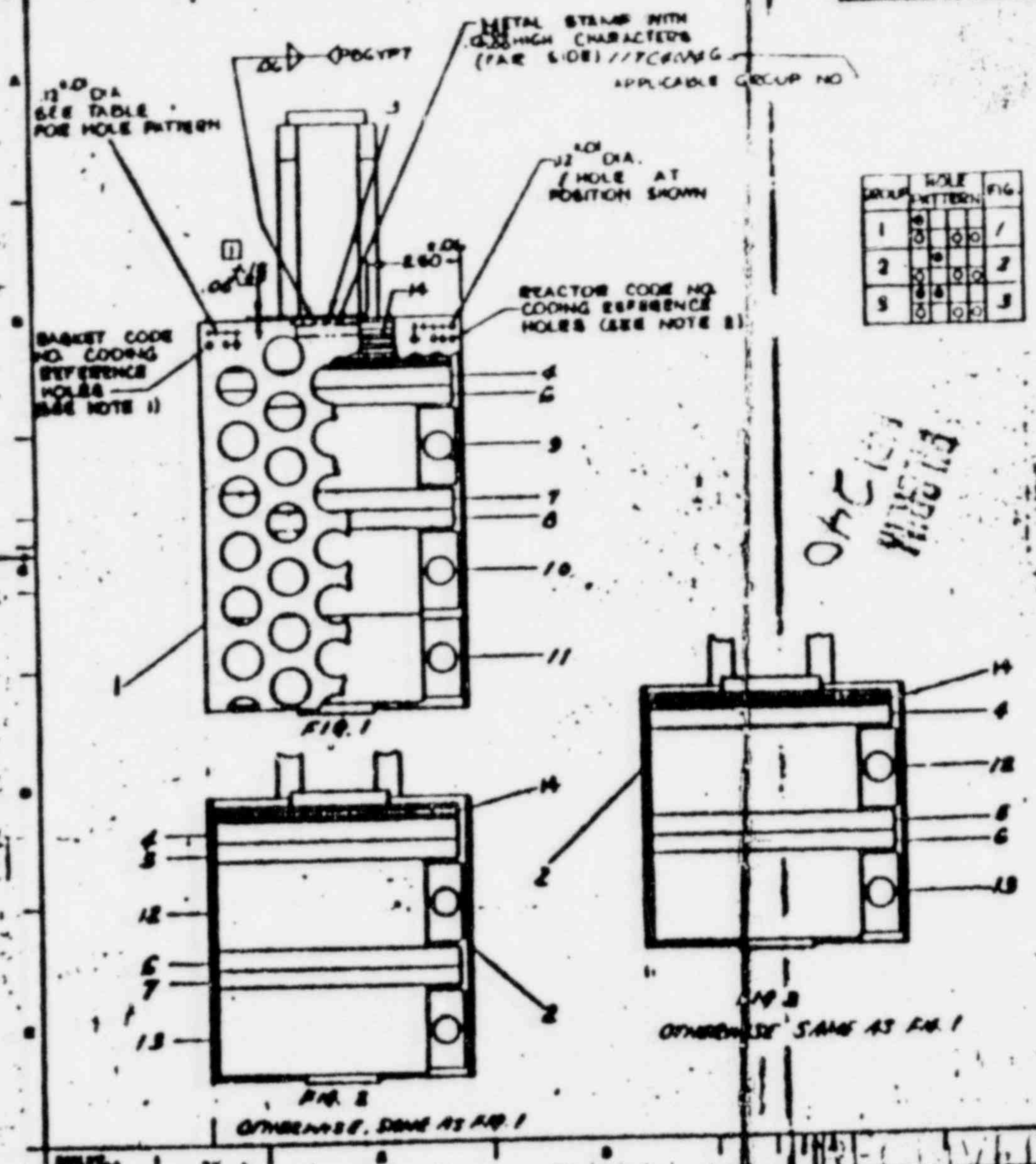
CAPSULE BASKET

SURVEILLANCE PROGRAM

PL 155U10

203A4270 | 1

FCF: 01 T093-117C4075



GROUP	HOLE PATTERN				FIG.
1	● ○		○	○	1
2	○	●	○	○	2
3	● ○	●	○	○	3

NOTES:

1. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW: CODING REFERENCE HOLES
2. THE HOLE PATTERN IS A BINARY NUMBER MADE BY ADDING THE VALUES OF THE HOLES AS INDICATED BELOW: CODING REFERENCE HOLES
3. WELDING PER 203A4230
4. ARRANGEMENT OF CAPSULES WITHIN BASKET IS OPTIONAL
5. CLEAN WITH ACETONE WASH
6. CLEAN PER P50YP107
7. INSTALL PLUG P3 FIRMLY AGAINST STACK OF CAPSULES AND SPACERS TO TAKE UP ALL END PLAY OF PARTS AND WELD.

PILGRIM

240

THIS IS A
REDUCED PRINT

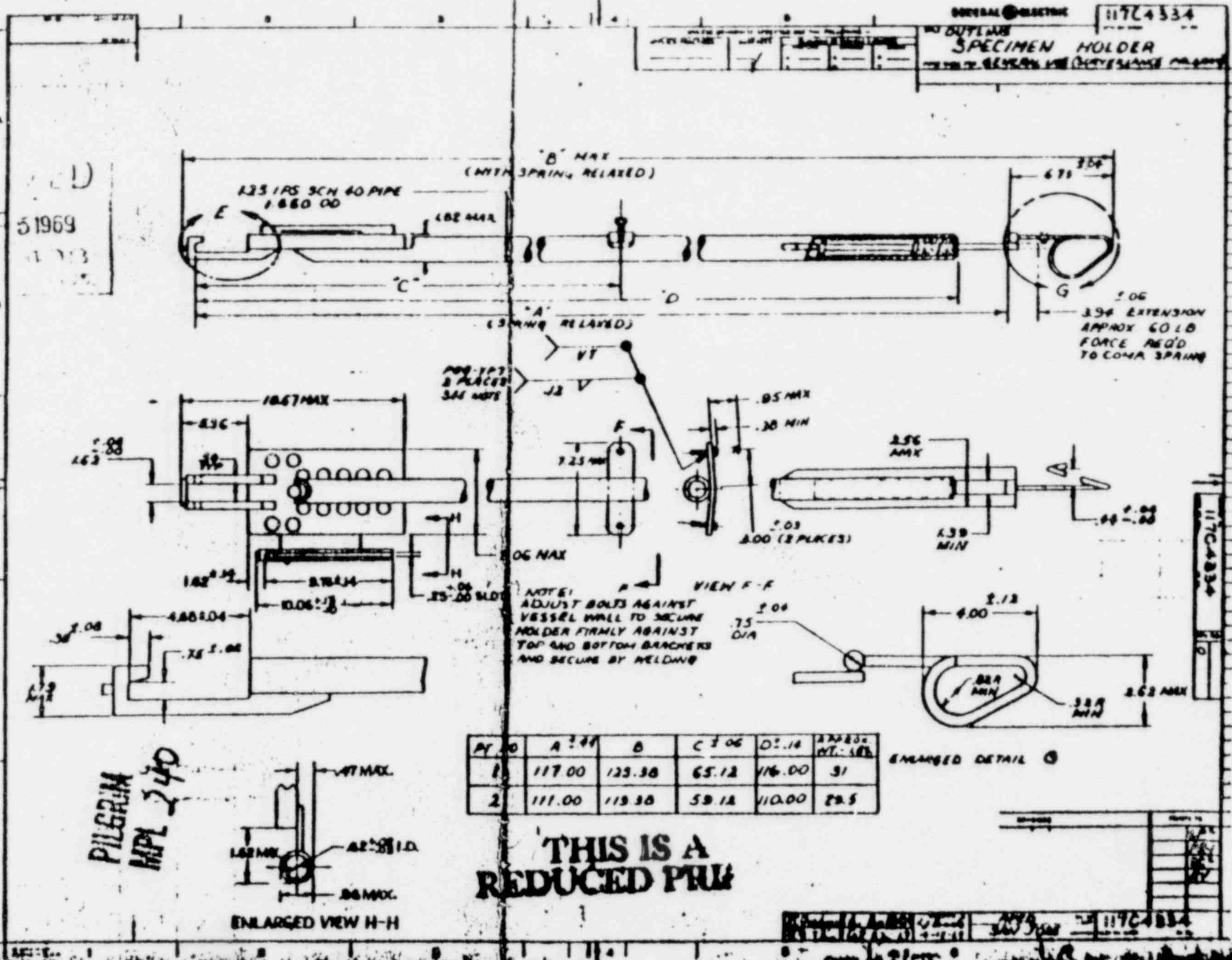
117C4074

117C4074

DEC - 5 1969

MIC 39.1

RECEIVED
DEC 5 1969



NOTE:
ADJUST BOLTS AGAINST
VESSEL WALL TO SECURE
HOLDER FIRMLY AGAINST
TOP AND BOTTOM BRACKETS
AND SECURE BY WELDING

PT NO	A ±.44	B	C ±.06	D ±.14	APPROX WT. LBS
1	117.00	125.38	65.12	116.00	31
2	111.00	119.38	59.12	110.00	29.5

THIS IS A
REDUCED PRI

PILGRIM
MPL 240

ASSY

117C4334

APPENDIX B

TENSILE AND CHARPY V-NOTCH PROPERTIES OF UNIRRADIATED PILGRIM 1 SURVEILLANCE MATERIALS [13]

TABLE 2. CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM BASE METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
Y2/C	-100	3.1	2.4	1
Y1/L	-50	8.3	9.0	5
Y2/4	0	24.0	23.2	25
Y3/2	25	52.0	43.2	30
Y2/D	30	44.0	37.4	25
Y1/2	50	74.9	54.2	45
Y3/Y	50	71.3	61.2	40
Y2/1	50	56.4	50.0	45
Y1/D	80	83.8	67.0	60
Y2/2	115	110.0	85.2	85
Y2/5	150	131.9	75.0	100
Y3/M	250	130.3	92.0	100
Y1/7	300	130.2	84.2	100

TABLE 3. CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM WELD METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
Y4/B	-200	11.0	9.4	0
Y5/M	-150	14.5	17.0	1
Y6/U	-125	6.8	6.8	5
Y6/5	-125	37.0	31.6	10
Y6/Y	-100	41.0	54.0	20
Y4/5	-50	62.0	54.8	35
Y6/7	0	73.4	68.2	65
Y5/T	50	85.8	79.0	75
Y4/D	60	81.0	72.8	80
Y5/U	60	98.0	83.8	85
Y4/7	60	89.0	77.8	85
Y6/1	80	96.9	86.6	90
Y4/A	150	111.1	93.8	100
Y4/3	230	114.8	85.4	100
Y6/6	300	112.2	75.0	100

TABLE 4. CHARPY V-NOTCH IMPACT RESULTS FOR PILGRIM HAZ METAL

Specimen Identification	Test Temperature, F	Impact Energy, ft-lb	Lateral Expansion, mils	Fracture Appearance, Percent Shear
YA/J	-200	4.5	3.8	0
Y7/P	-165	21.5	16.0	5
Y7/A	-135	13.0	10.8	1
YB/1	-100	13.0	12.4	5
Y7/U	-50	37.5	30.0	15
YA/3	-40	68.6	53.4	35
YA/P	-35	74.5	58.4	45
YA/T	0	68.0	60.4	75
Y7/2	35	97.5	84.0	85
YB/2	50	135.7	79.4	100
YB/6	81	107.9	77.2	90
Y7/7	150	99.9	84.8	100
YB/E	300	112.0	87.8	100

TABLE 6. TENSILE TEST RESULTS FOR PILGRIM UNIRRADIATED BASELINE SPECIMENS

Material	Specimen Number	Test Temp, F	0.2 Percent Offset Yield Strength, ksi	Ultimate Tensile Strength, ksi	Fracture			Reduction in Area, percent	Elongation, %	
					Load, lb	Strength, ksi	Stress, ksi		Uniform	Total
Base	YC-5	72	64.5	86.4	2800	55.7	185	69.90	12.4	25.5
Base	YC-6	550	57.7	82.8	2925	58.2	166	65.00	10.6	21.4
Weld	YD-T	72	68.8	78.4	2450	49.6	178	72.02	14.5	28.6
Weld	YD-K	550	54.6	78.3	2725	54.7	148	63.04	12.0	23.0
HAZ	YE-D	72	61.7	79.6	2475	49.7	178	72.17	8.5	22.9
HAZ	YE-E	550	57.2	78.1	2410	48.4	136	64.48	7.9	18.9

APPENDIX C

PHOTOGRAPHS OF IMPACT AND TENSILE CAPSULES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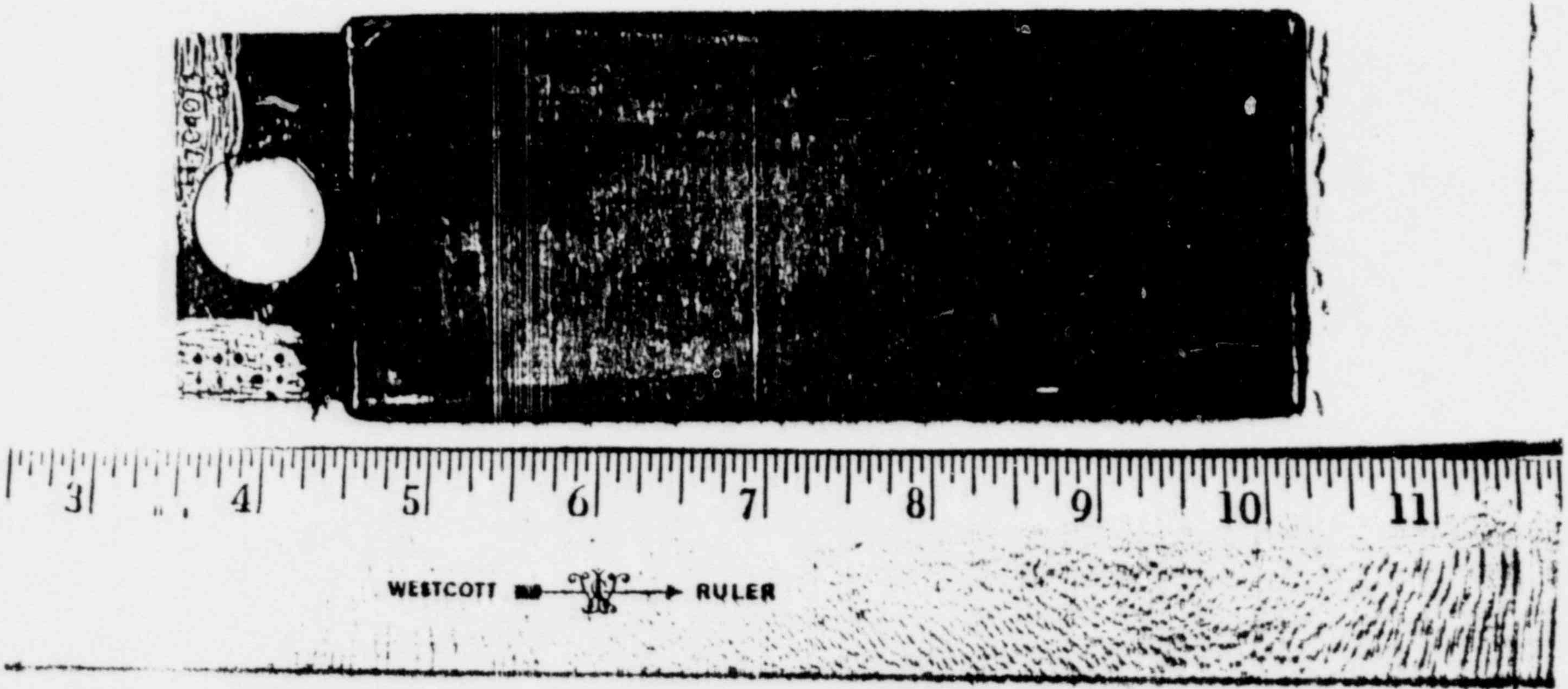
WESTCOTT
RULER



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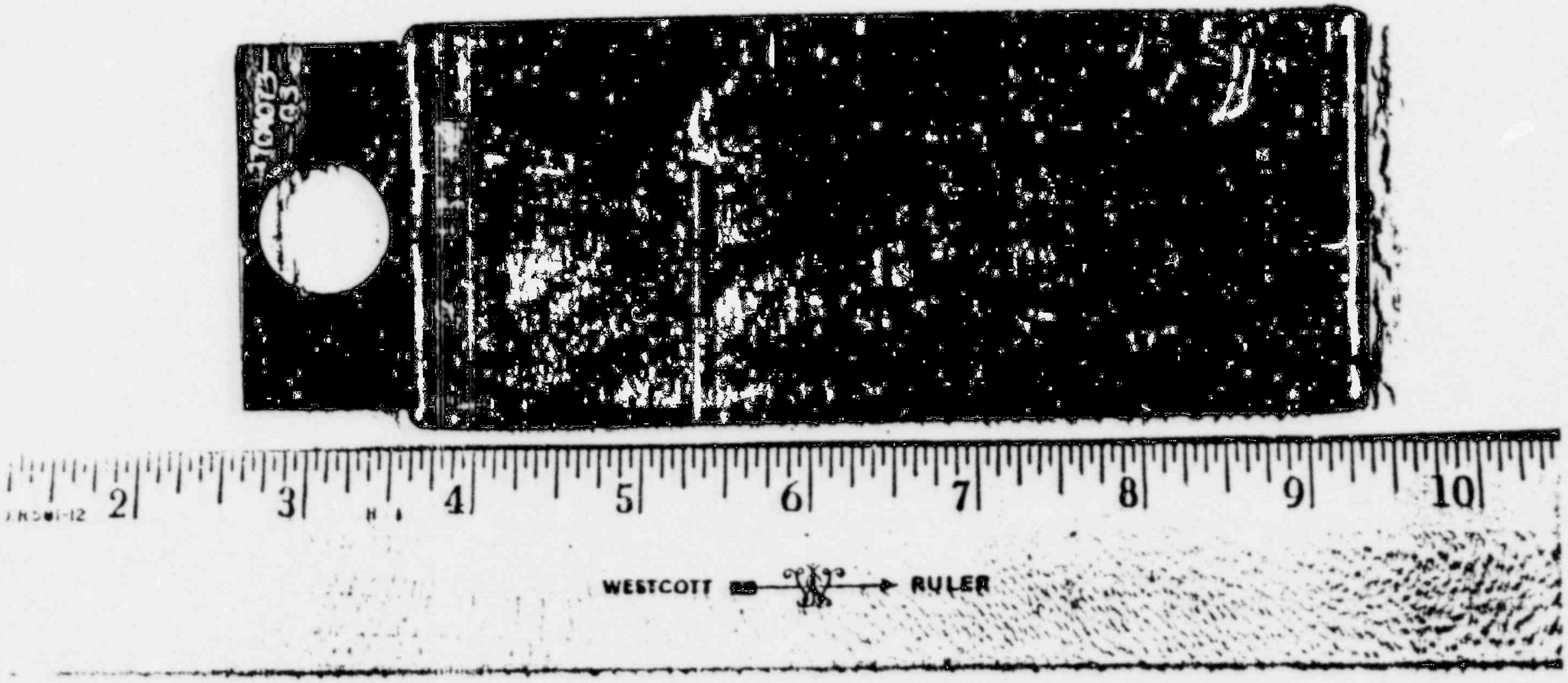
WESTCOTT RULER



1704073
G3
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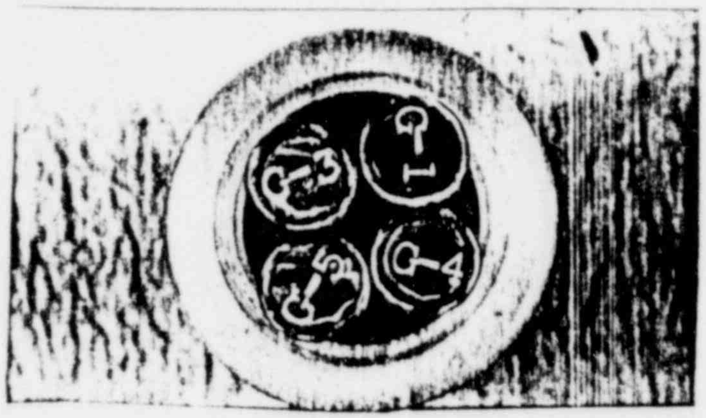
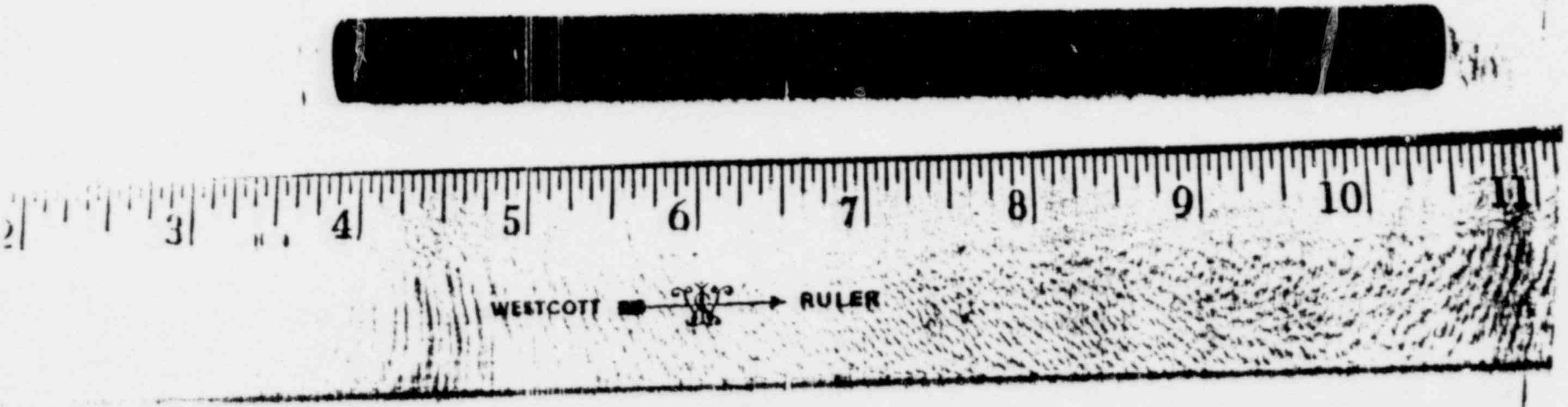
WESTCOTT RULER

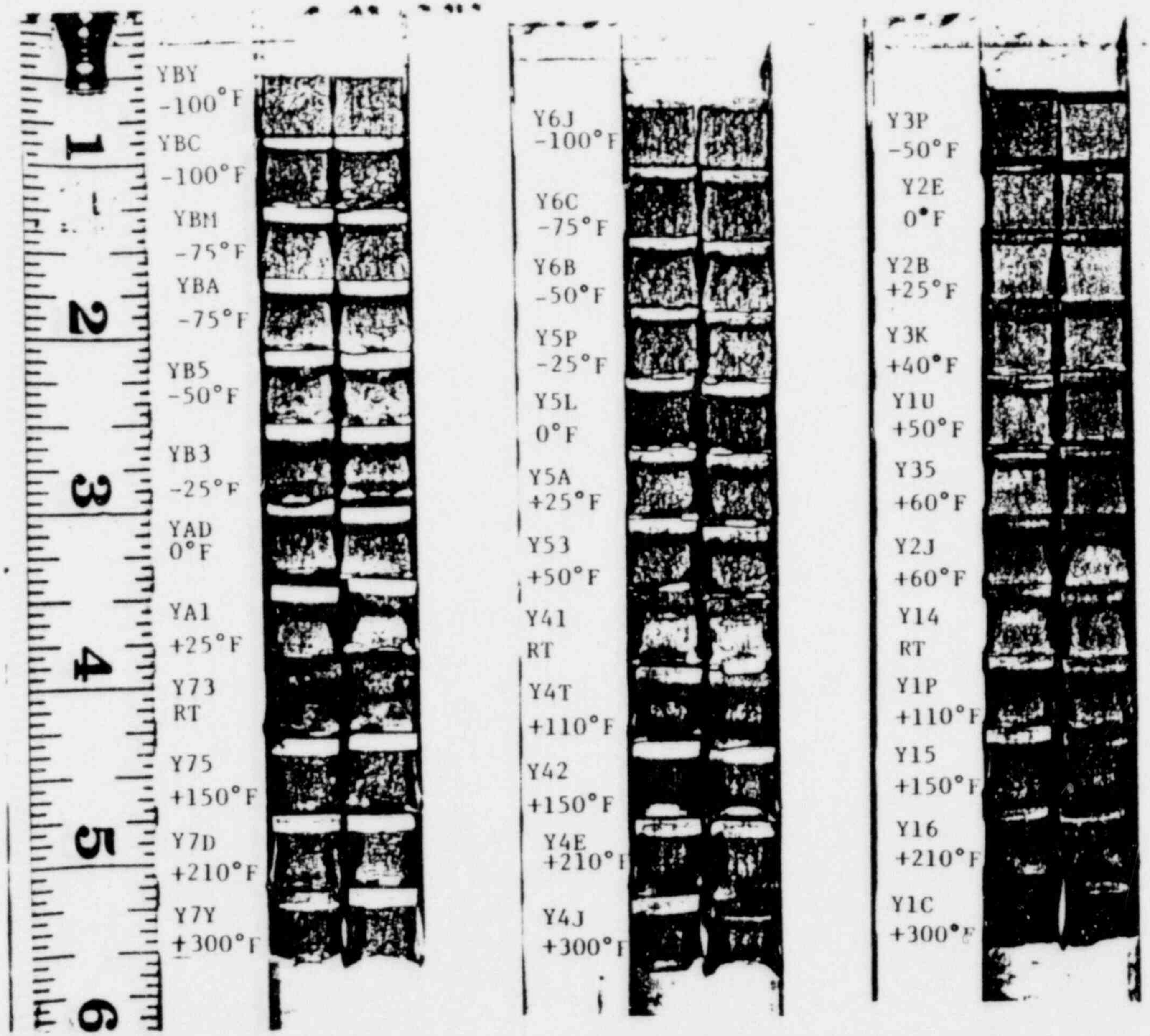


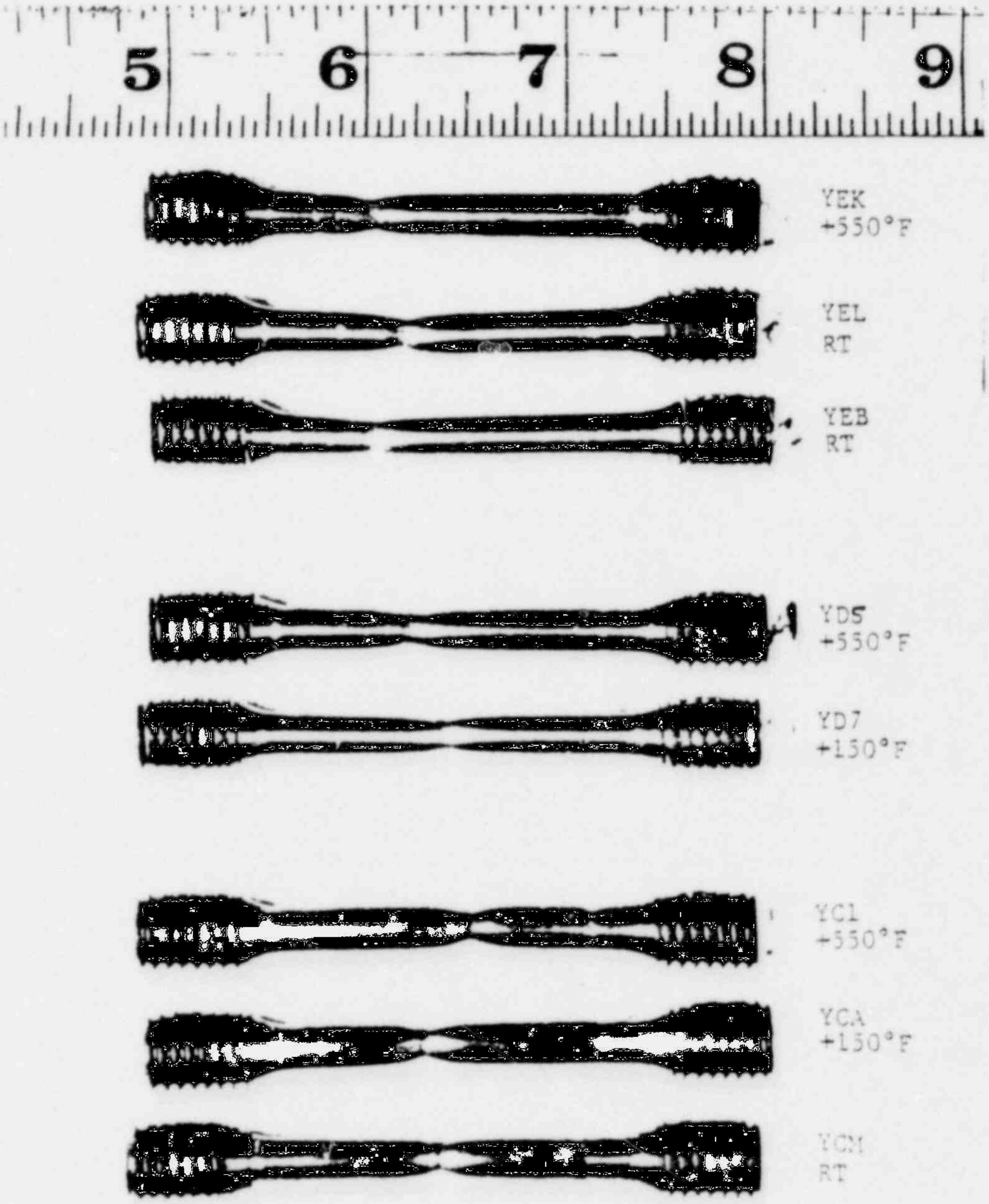
G-4



APPENDIX D

PHOTOGRAPHS OF TESTED SPECIMENS AND TENSILE TEST RECORDS


1
2
3
4
5
6
YBY
-100°F
YBC
-100°F
YBM
-75°F
YBA
-75°F
YB5
-50°F
YB3
-25°F
YAD
0°F
YA1
+25°F
Y73
RT
Y75
+150°F
Y7D
+210°F
Y7Y
+300°F
Y6J
-100°F
Y6C
-75°F
Y6B
-50°F
Y5P
-25°F
Y5L
0°F
Y5A
+25°F
Y53
+50°F
Y41
RT
Y4T
+110°F
Y42
+150°F
Y4E
+210°F
Y4J
+300°F
Y3P
-50°F
Y2E
0°F
Y2B
+25°F
Y3K
+40°F
Y1U
+50°F
Y35
+60°F
Y2J
+60°F
Y14
RT
Y1P
+110°F
Y15
+150°F
Y16
+210°F
Y1C
+300°F



YEK
+550°F

YEL
RT

YEB
RT

YD5
+550°F

YD7
+150°F

YC1
+550°F

YCA
+150°F

YCM
RT

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-________

Spec. No. YE-L

Temperature 72 °F

Strain Rate .005 in/in/min

Est. U.T.S. ________ psi

Initial G. L. 1.013 in.

Initial Dia. .250 in.

Initial Thickness ______ in.

Initial Width ________ in.

Project No. 02-5951-00

Machine No. 22K,P

Date 28 Aug 1980

Initial Area .0491 in²

Top Temperature ________ °F

Bottom Temperature ________ °F

Final Gage Length 1.254 in.

Final Diameter .138 in.

Final Area ~~.0149~~ .0150 in.²

Maximum Load 4320 lb

0.2% Offset Load 3440 lb

0.02% Offset Load ______ lb

Upper Yield Point ______ lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = ~~88.2~~ 88.0 ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = ~~70.2~~ 70.1 ~~psi~~ Ksi

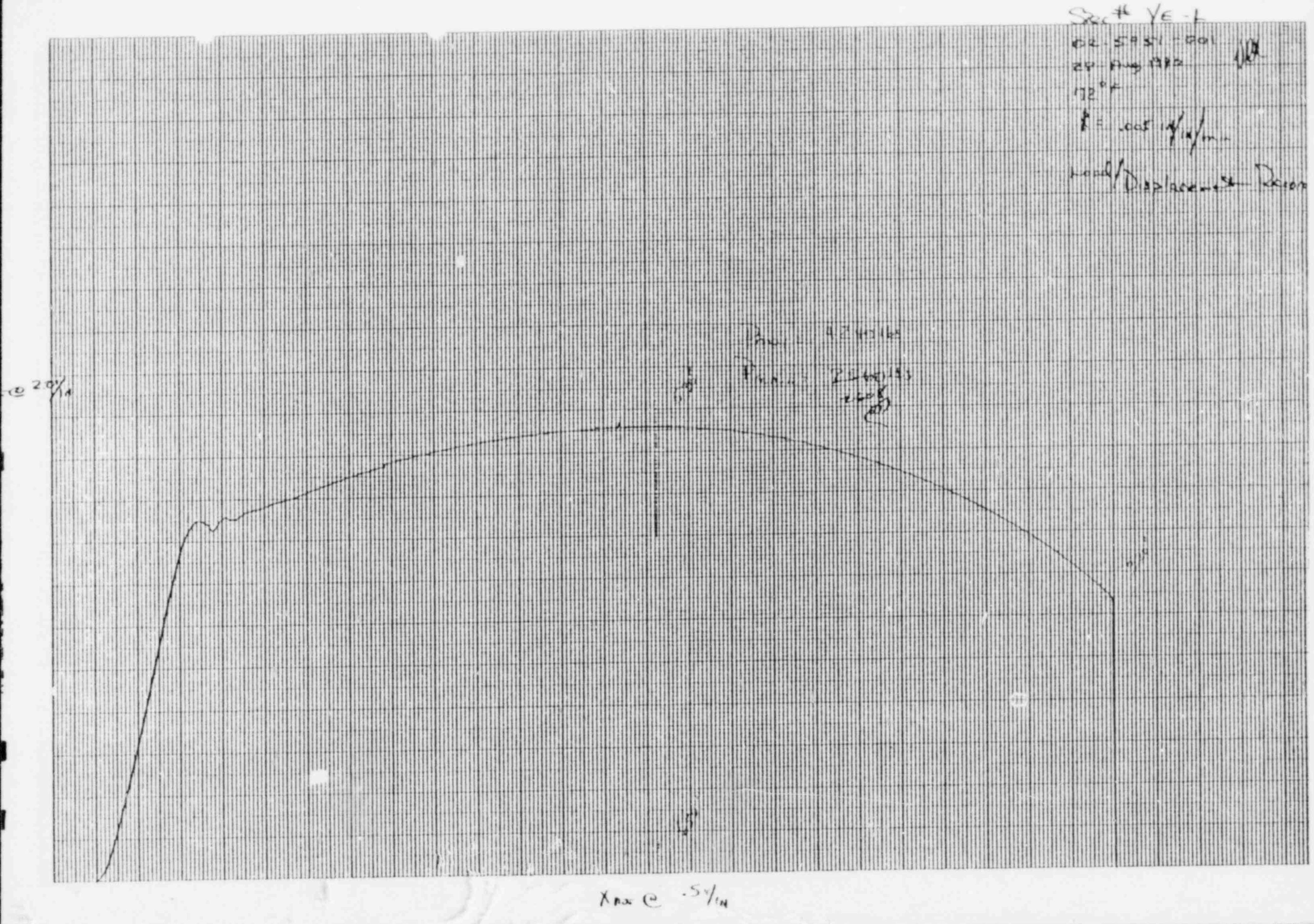
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ________ psi

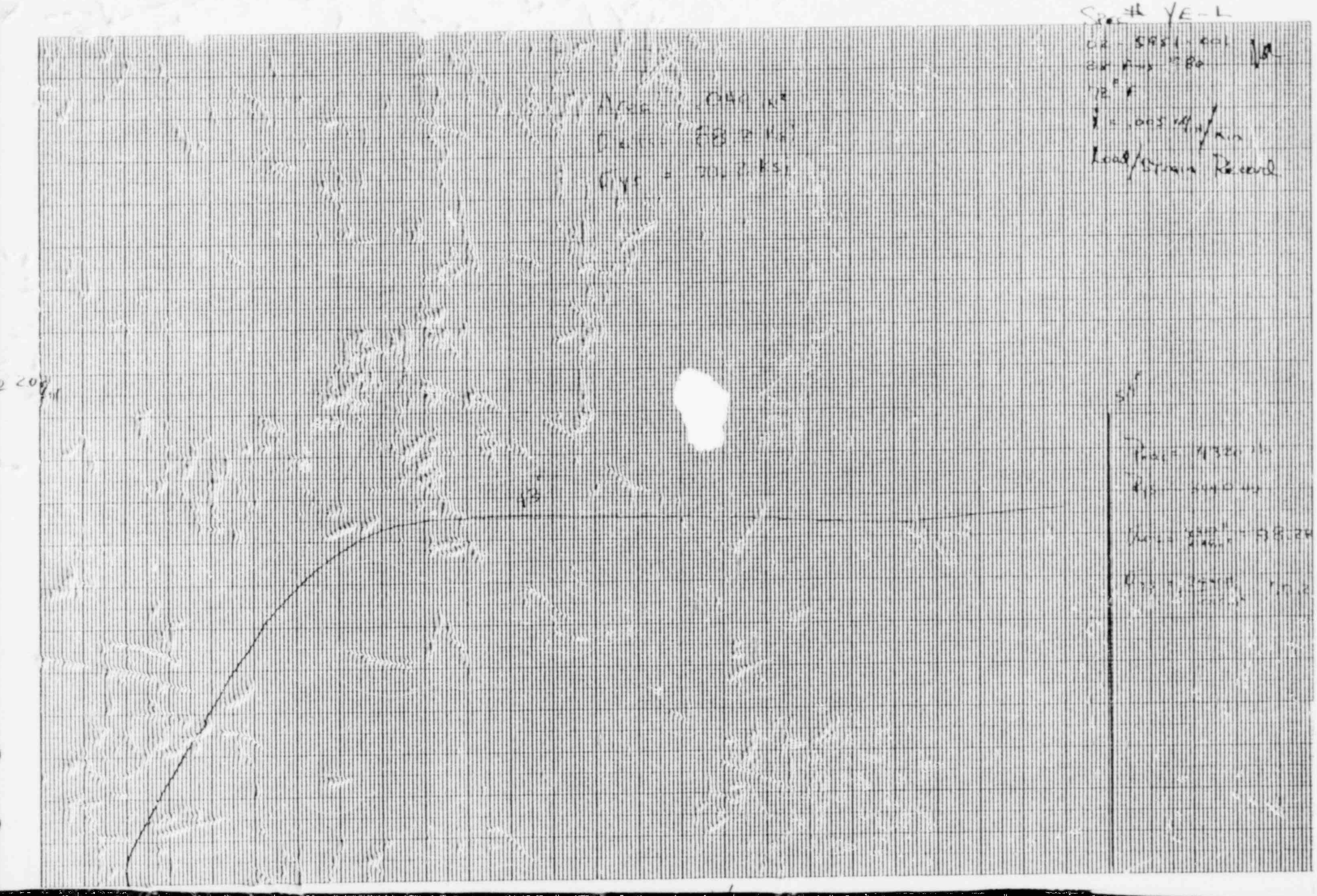
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ________ psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 23.8 ✓ %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 69.5 ✓ %

Signature: Vic Aaron


Spec # YE-1
02-5951-001
28 Aug 1982
72 °F
ε̇ = .005 in/in/min
Load/Displacement Record
Pmax = 4240 lbs
Pfinal = 2240 lbs
2208
@ 20%/in
Xax @ .5"/in



Spec # YE-L
CR - 5951 - 001 VA
8 Aug '80
72° F
$\dot{\varepsilon}$ = .005 in/in/min
Load/Strain Record

Area = .044 in²
Dia = .238 in
σ_{ys} = 201.0 ksi

P_{max} = 9320 lb
P_{ys} = 8840 lb
Red A = [illegible] = 38.2%
σ_{ts} = [illegible] [illegible]

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-________

Spec. No. YE-B

Temperature 72 °F

Strain Rate .005 IN/IN/min

Est. U.T.S. ________ psi

Initial G. L. 1.013 in.

Initial Dia. .250 in.

Initial Thickness ______ in.

Initial Width ______ in.

Project No. 02-5951-001

Machine No. 22KIP

Date 28 Aug 1980

Initial Area .0491 IN^2

Top Temperature ________ °F

Bottom Temperature ________ °F

Final Gage Length 1.231 in.

Final Diameter .138 in.

Final Area ~~.0149~~ .0150 in.^{2}

Maximum Load 4408 lb

0.2% Offset Load 3520 lb

0.02% Offset Load ______ lb

Upper Yield Point ______ lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 89.8 ~~89.9~~ ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 71.7 ~~71.8~~ ~~psi~~ Ksi

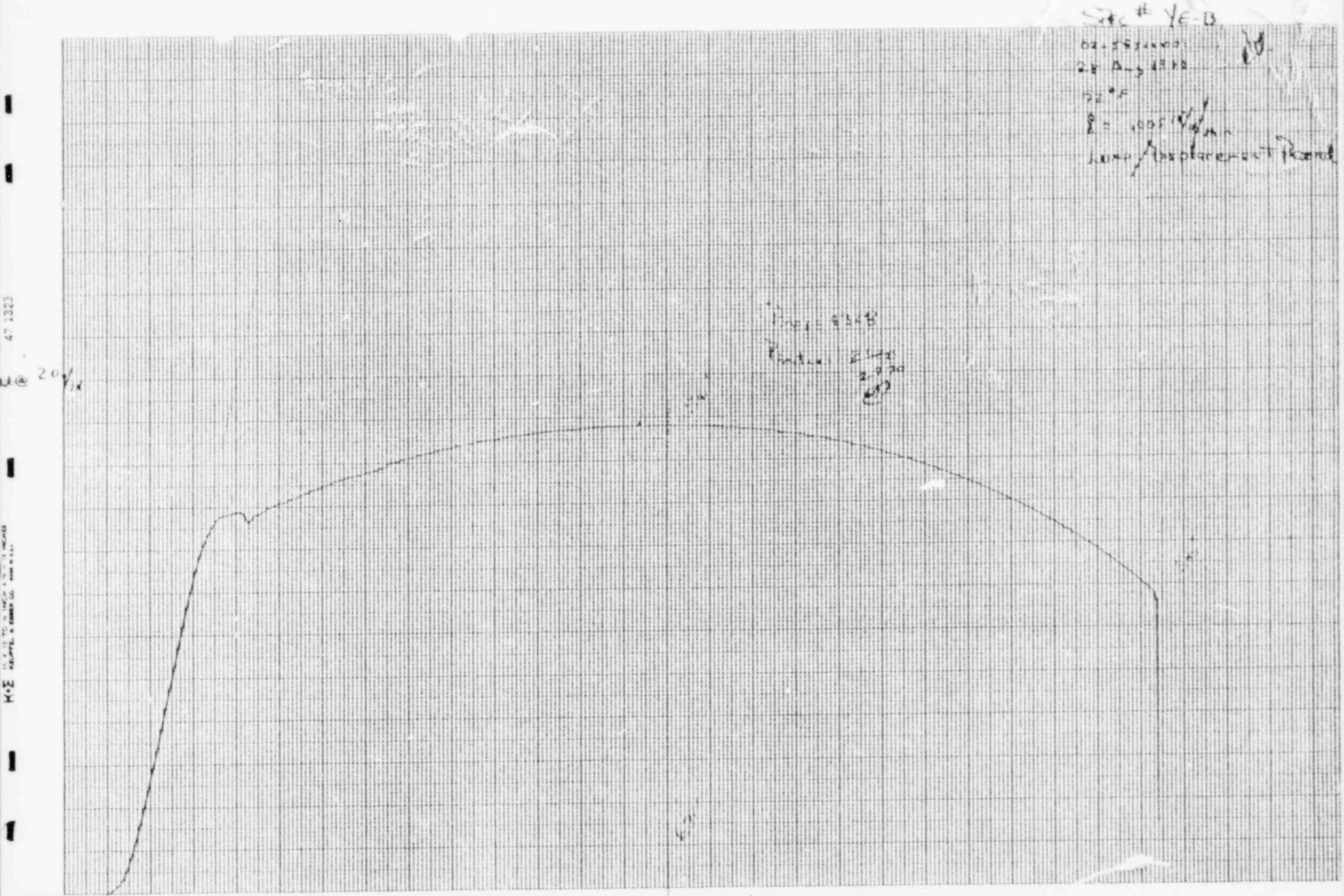
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ________ psi

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

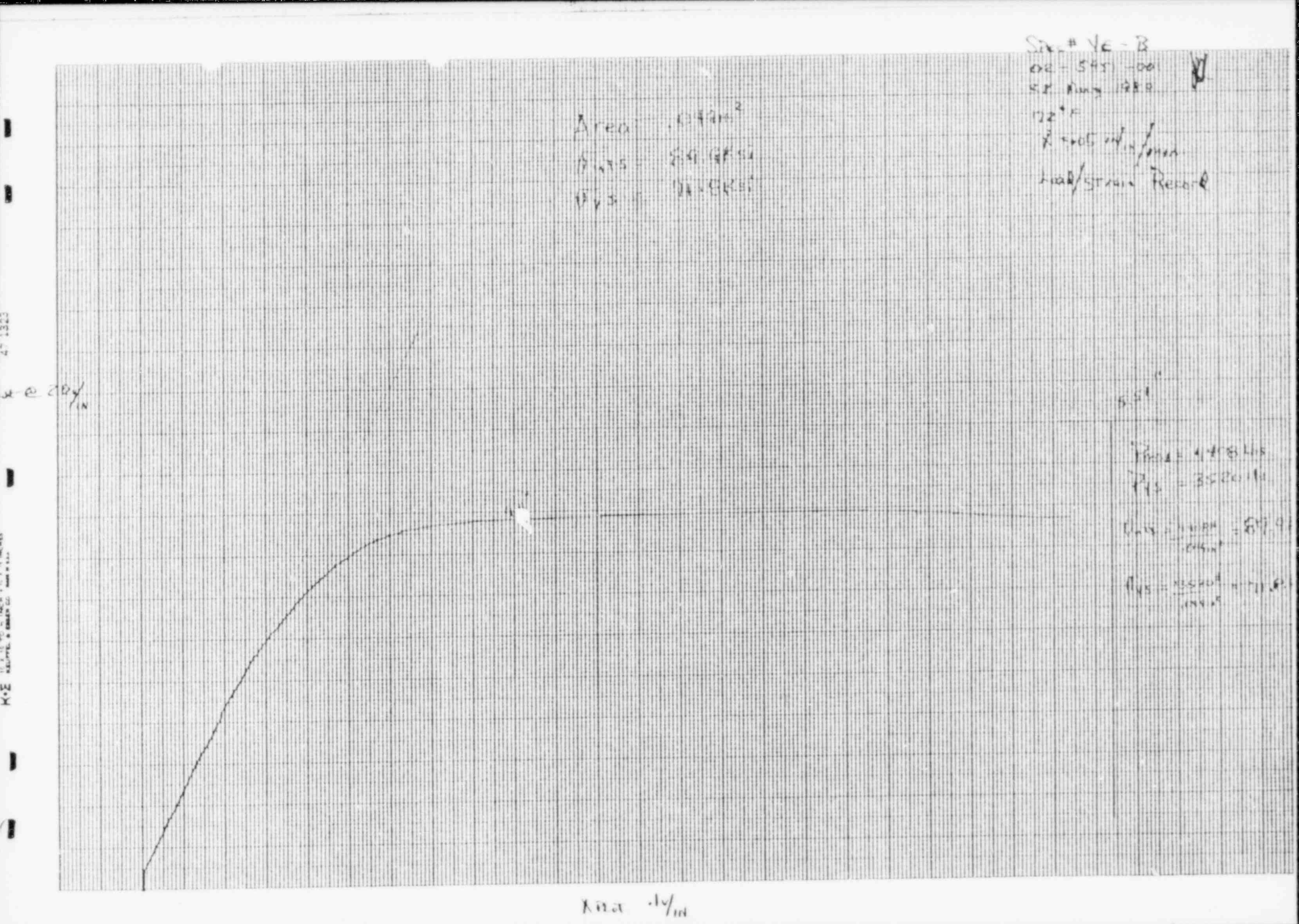
% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 21.5 ✓ %

% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 69.5 ✓ %

Signature: Vic Ovan



Spec # YE-B
72 °F
Load/Displacement Record
Max = 6348
Fracture 2.42
2.930
Load @ 20/1x
K+E 10 X 10 TO ½ INCH
KEUFFEL & ESSER CO. MADE IN U.S.A.
47 1323


Spec # VE-B
02-5451-001
28 Aug 1980
72°F
$\dot{x}$ .005 in/in/min
Load/Strain Record
Area .0494in²
σu.s 89.9KSI
σy.s 71.6KSI
6.5
Pmax = 4448 LBS
Py.s = 3540 lb
σu.s = 4448#/.0494in² = 89.9
σy.s = 3540#/.0494in² = 71.6
Load @ 200#/IN
X-hd .1"/IN
K+E
47 1323

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-

Spec. No. YC-M

Temperature 72 °F

Strain Rate .005 in/in/min

Est. U.T.S. psi

Initial G. L. 1.013 in.

Initial Dia. .250 in.

Initial Thickness in.

Initial Width in.

Project No. 02-5951-001

Machine No. 22KIP

Date 28 Aug 1980

Initial Area .0491 in²

Top Temperature °F

Bottom Temperature °F

Final Gage Length 1.270 in.

Final Diameter .135 in.

Final Area .0143 in.^{2}

Maximum Load 4568 lb

0.2% Offset Load 3496 lb

0.02% Offset Load lb

Upper Yield Point lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = ~~93.2~~ 93.0 ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = ~~71.4~~ 71.2 ~~psi~~ Ksi

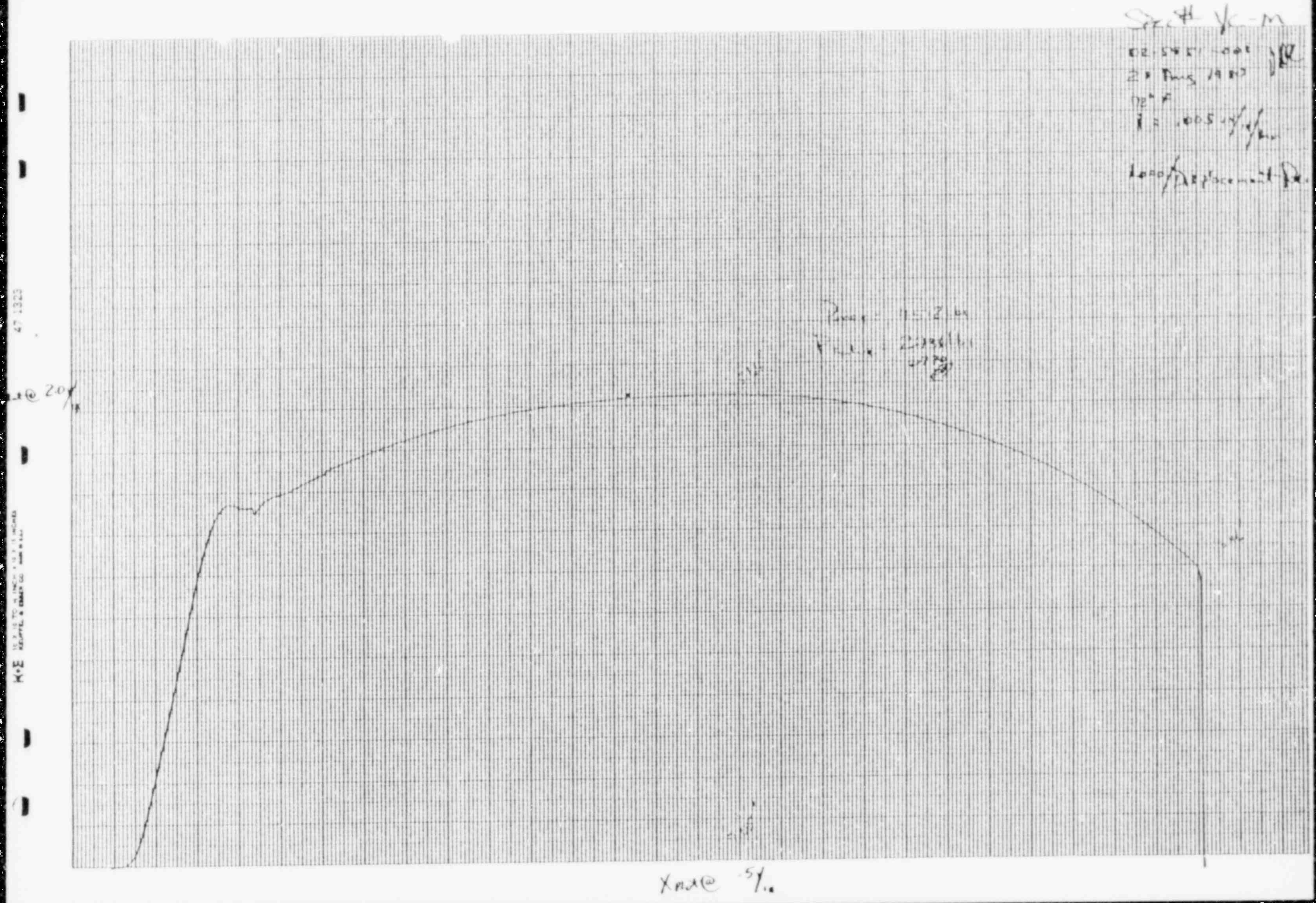
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = psi

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 = 25.4 %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = ~~70.8~~ 70.9 %

Signature:



Spec # VC-M

02-5451-001

2 Aug 1983

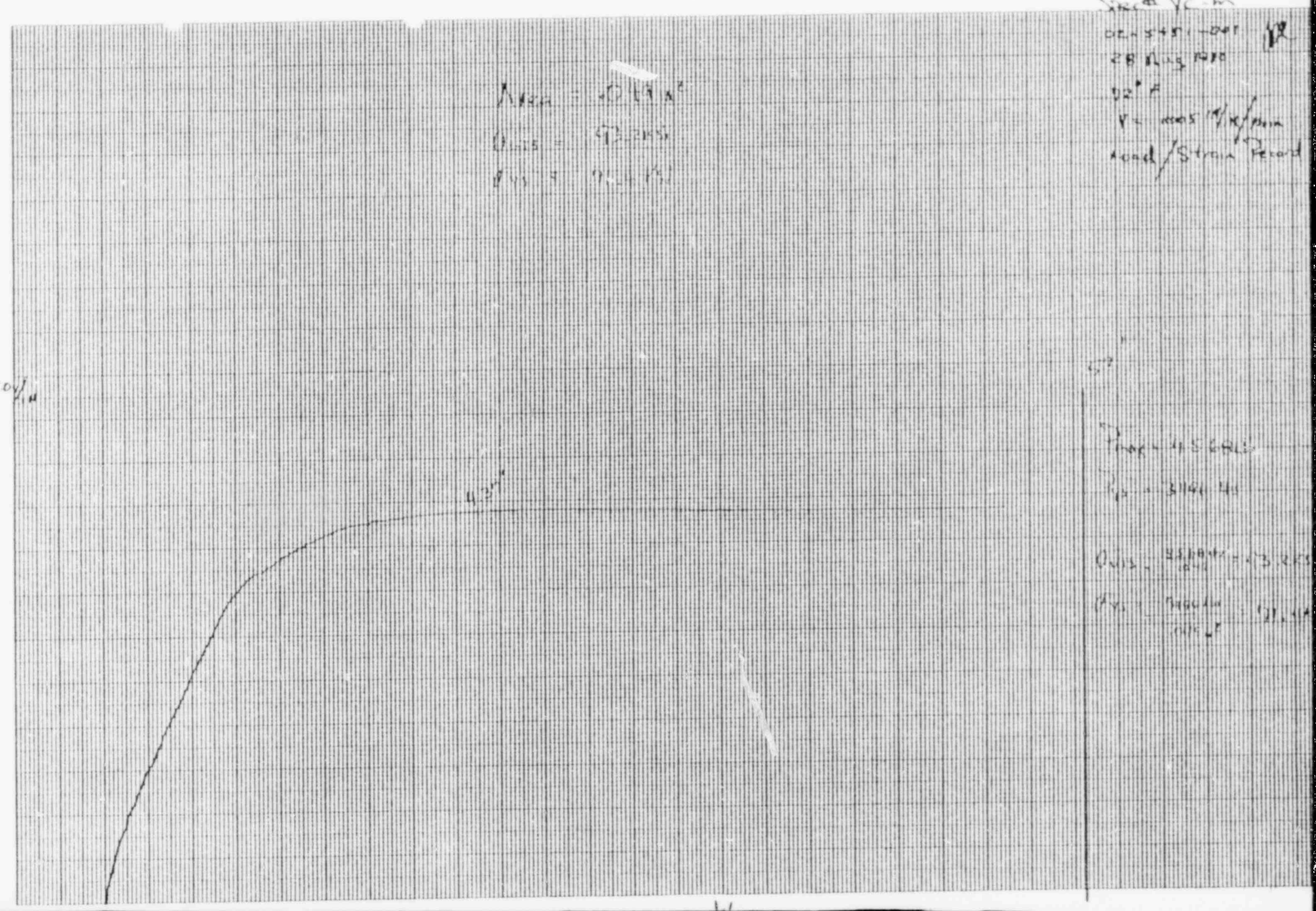
72° F

$\dot{\varepsilon}$ = .005 in/in/min

1000 / Displacement Pa.

Pmax = 11.2 [illegible]

Ymax = 2[illegible]



Spec# VC-M
02-545-001
28 Aug 1980
92° F
V = .005 in/in/min
Load/Strain Record

Area = .119 in²
σuts = 43.2 ksi
σys = 29.4 ksi

Pmax = [illegible] kips
Py = 3494 lb

σuts = 5140#/.119 = 43.2 ksi
σys = 3494 lb/.119 in² = 29.4 ksi

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-________

Spec. No. YC-A

Temperature 150 °F

Strain Rate .005 in/in/min

Est. U.T.S. ________ psi

Initial G. L. 1.013 in.

Initial Dia. .251 in.

Initial Thickness ______ in.

Initial Width ________ in.

Project No. 02-5951-001

Machine No. 22KIP

Date 28 Aug 1980

Initial Area .04948 in²

Top Temperature ________ °F

Bottom Temperature ________ °F

Final Gage Length 1.276 in.

Final Diameter .135 in.

Final Area .01431 in.²

Maximum Load 4424 lbs

0.2% Offset Load 3360 lb

0.02% Offset Load ______ lb

Upper Yield Point ______ lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 89.4 ✓ ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 67.9 ✓ ~~psi~~ Ksi

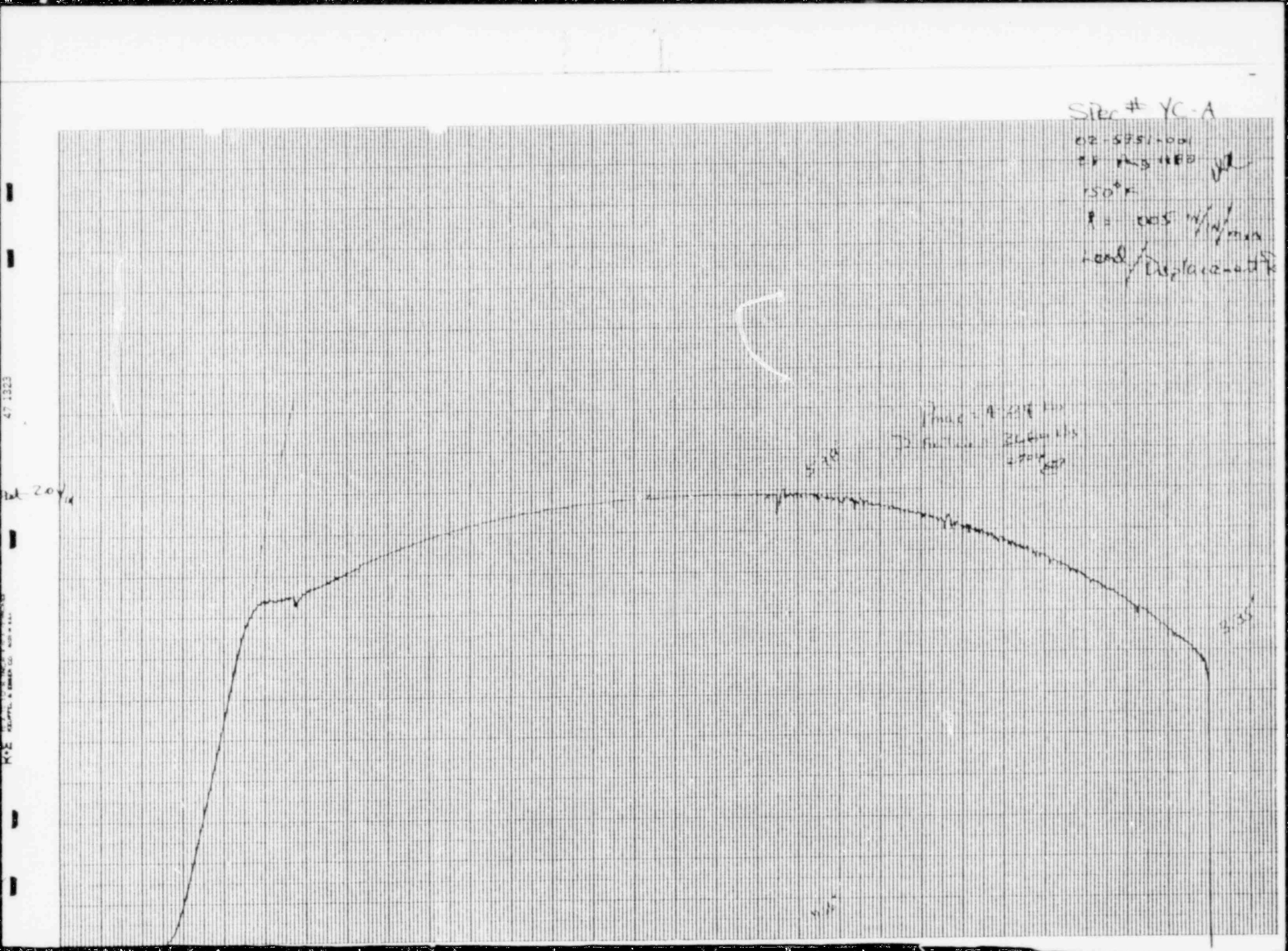
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ________ psi

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 = 26.0 ✓ %

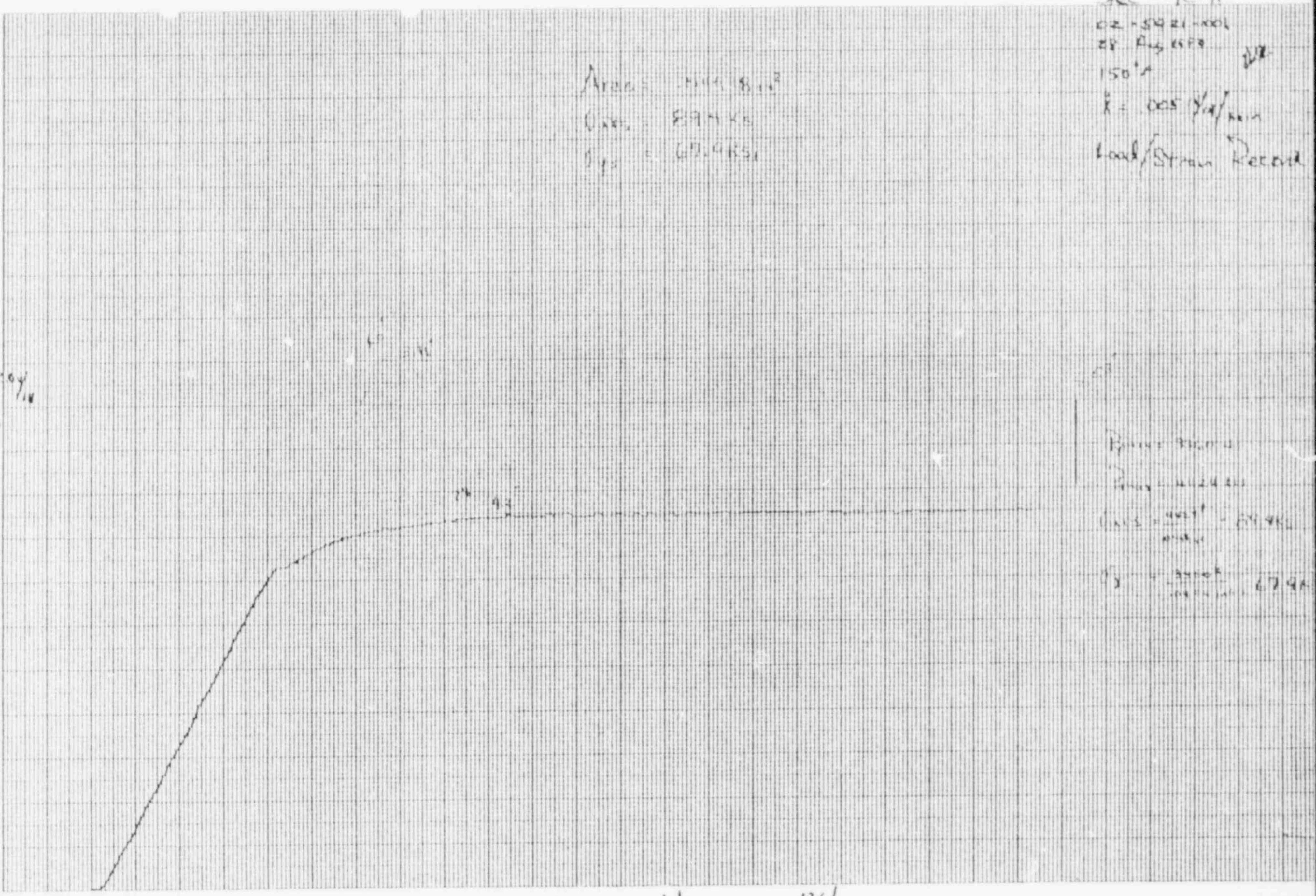
% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 71.1 ✓ %

Signature: ____________


Spec # YC-A
02-5751-001
[illegible]
150°F
ε = .005 in/in/min
Load/Displacement
Pmax = 4,[illegible] lbs
P failure = [illegible] lbs
5.18
5.35
[illegible]-20#/in
47 1323
K+E KEUFFEL & ESSER CO.

Spec # YC-A
02-5421-001
28 Aug 1979
150°F
$\dot{\varepsilon}$ = .005 in/in/min
Load/Strain Record

Area = .1448 in²
σ_{ult} = 8.94 Kb
σ_{ys} = 69.4 KSI



P [illegible]
P [illegible]
σ_{ult} = [illegible] = 69.4 KSI
σ_y = [illegible] = 67.9 KSI

47 1323

K+E 10 X 10 TO ½ INCH KEUFFEL & ESSER CO.

Southwest Research Institu
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- ~~YD-3~~
Spec. No. VD-7
Temperature 150 °F
Strain Rate .005 in/in/min.

Est. U.T.S. ______ psi
Initial G. L. 1.013 in.
Initial Dia. .250 in.
Initial Thickness ______ in.
Initial Width ______ in.

Project No. 02-5951-00
Machine No. 22 KIP
Date 28 Aug 1980
Initial Area .0491 in²

Top Temperature ______ °F
Bottom Temperature ______ °F
Final Gage Length 1.274 in.
Final Diameter .144 in.
Final Area .0163 ✓ in.²

Maximum Load 4208 lb
0.2% Offset Load 3424 lb
0.02% Offset Load ______ lb
Upper Yield Point ______ lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = ~~85.9~~ 85.7 ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = ~~69.9~~ 69.7 ~~psi~~ Ksi

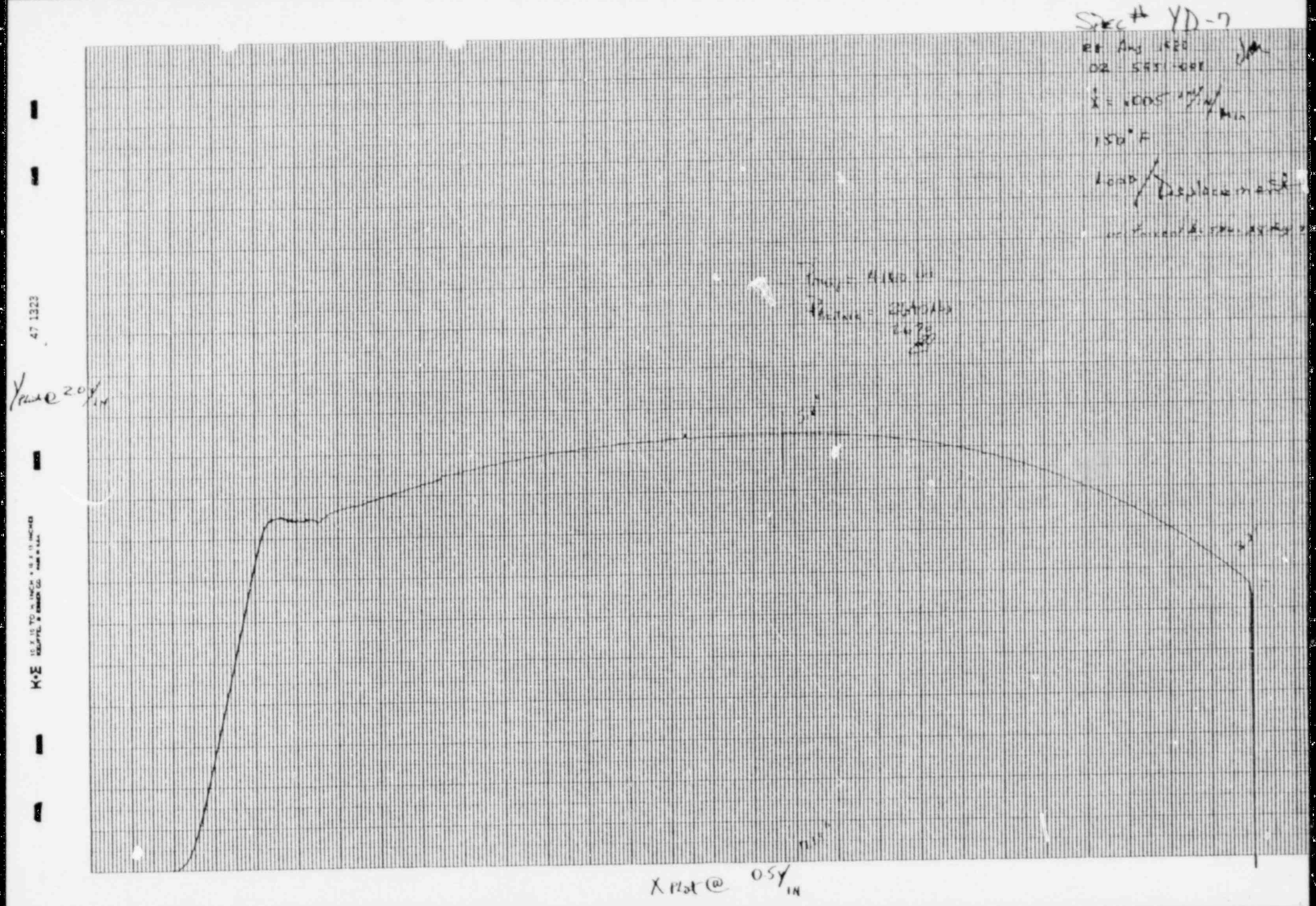
0.02% Y.S. = $\frac{\text{0.02\% Offset Load}}{\text{Initial Area}}$ = ______ psi

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 = 25.8 ✓ %

% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = ~~66.7~~ 66.8 %

Signature: Vic Aaron
witnessed by TAM 28 Aug 80



Spec # YD-7

28 Aug 1980

02 5551-001

$\dot{x}$ = .005 in/in/min

150° F

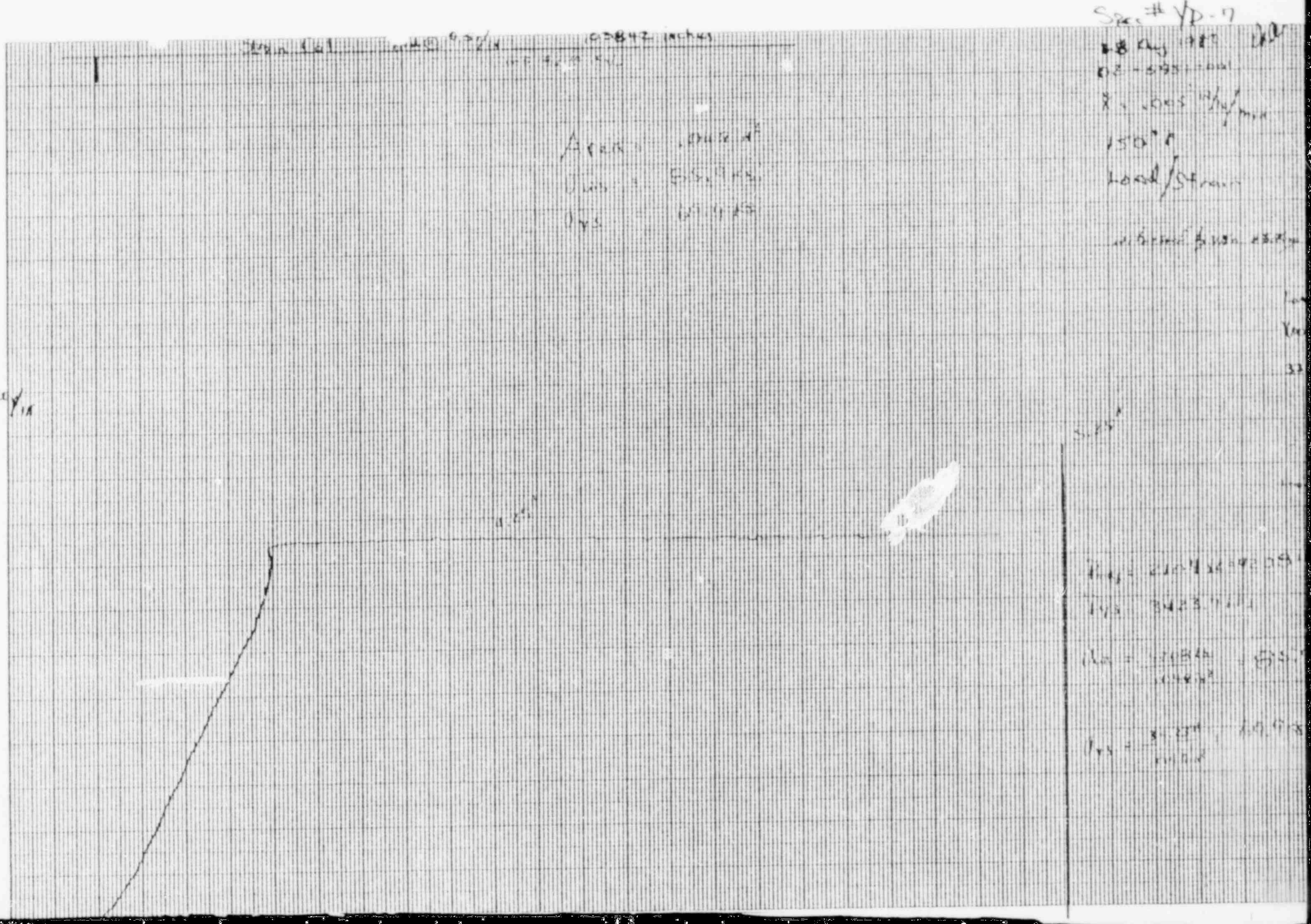
Load / Displacement

[illegible]

P_{max} = 4180 lb

$P_{fracture}$ = 2640 lbs

2428



47 1323

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

Southwest Research Institut
Department of Materials Sciences

TENSILE TEST DATA SHEET

est No. T-YE-K

pec. No. ______

emperature 550 °F

train Rate .005 in/in/min

Est. U.T.S. ______ psi

Initial G. L. 1.0130 in.

Initial Dia. .250 in.

Initial Thickness ______ in.

Initial Width ______ in.

Project No. 02-5951-001

Machine No. 22Kip

Date 27 Aug 1980

Initial Area .0491 in^2

Top Temperature ______ °F

Bottom Temperature ______ °F

Final Gage Length 1.220 in.

Final Diameter .146 in.

Final Area .0167 ✓ in.^{2}

Maximum Load 3904 lb

0.2% Offset Load 2744 lb

0.02% Offset Load ______ lb

Upper Yield Point ______ lb

$$\text{U.T.S.} = \frac{\text{Maximum Load}}{\text{Initial Area}} = $$ 79.5 ~~79.7~~ ~~psi~~ Ksi

$$\text{0.2\% Y.S.} = \frac{\text{0.2\% Offset Load}}{\text{Initial Area}} = $$ 55.9 ~~56.0~~ ~~psi~~ Ksi

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

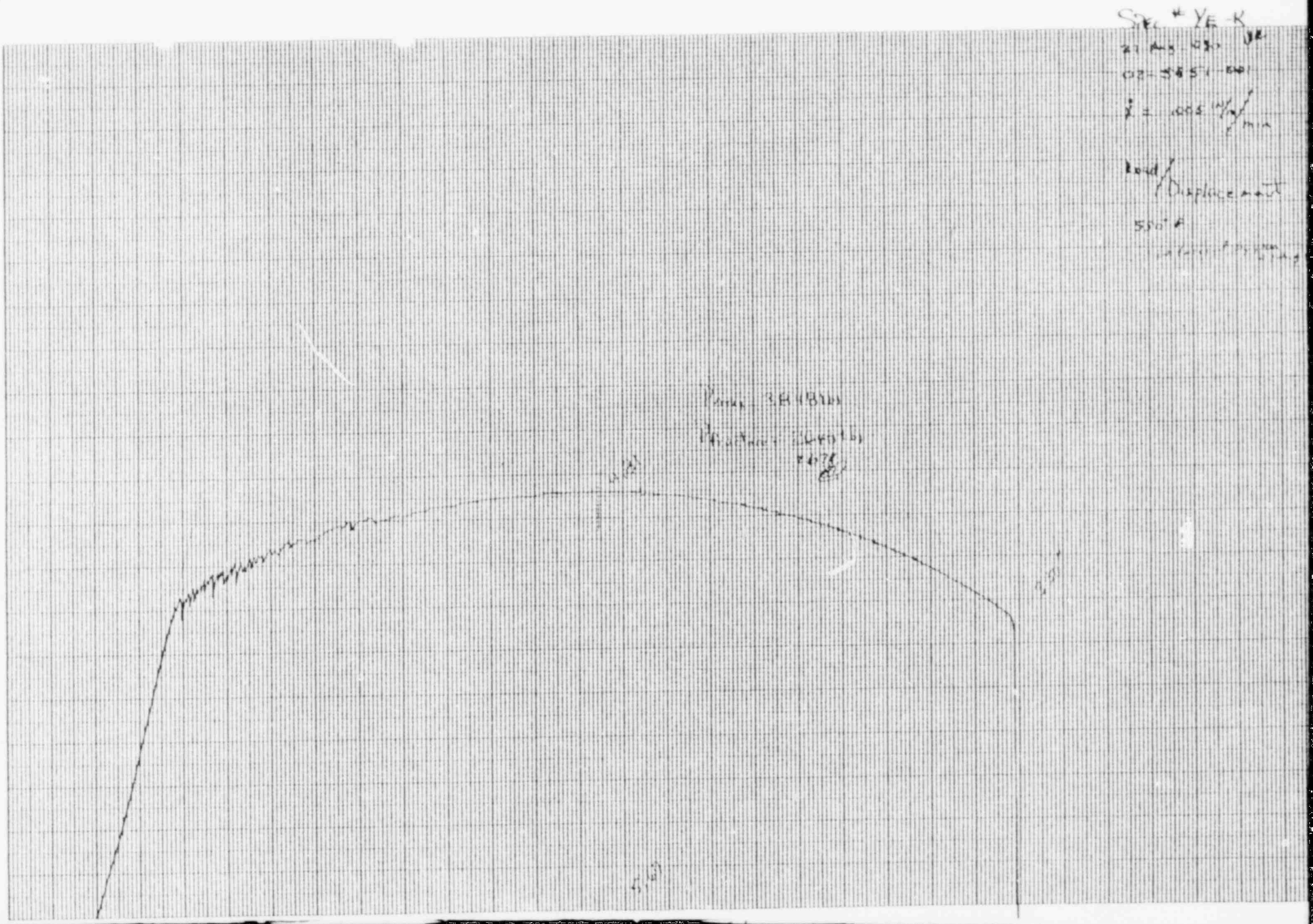
$$\text{Upper Y.S.} = \frac{\text{Upper Yield Point}}{\text{Initial Area}} = $$ ______ psi

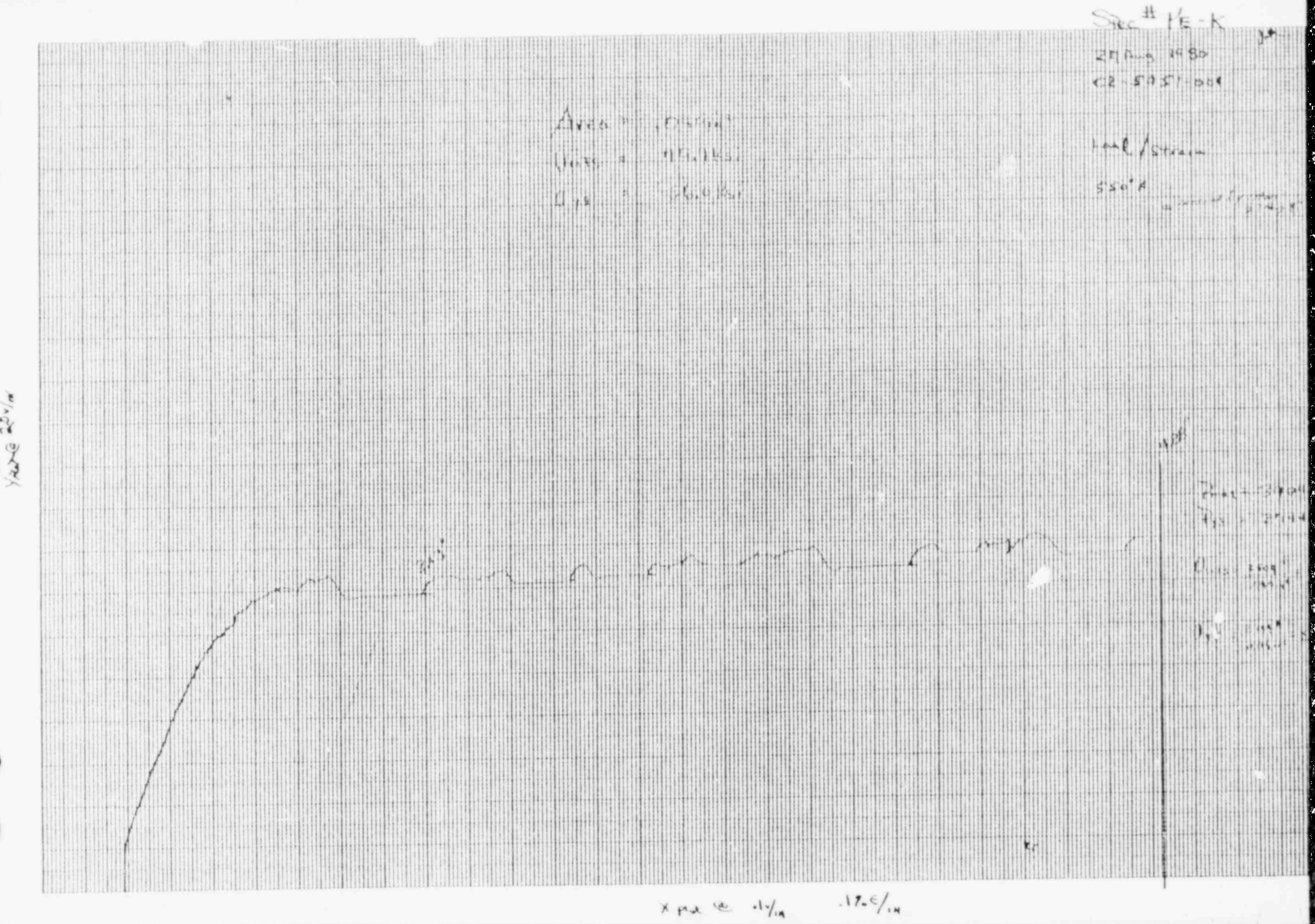
$$\text{\% Elongation} = \frac{\text{Final G.L. - Initial G.L.}}{\text{Initial G.L.}} \times 100 = $$ 20.4 ✓ %

$$\text{\% R.A.} = \frac{\text{Initial Area - Final Area}}{\text{Initial Area}} \times 100 = $$ 66.0 ~~65.9~~ ✓ %

ignature: Vic Oberon

witnessed by TKm 27 Aug 80


Spec. # YE-K
21 Aug 1980
02-5851-001
$\dot{x}$ = .005 in/in/min
Load/Displacement
550° F
Pmax = 3848 lbs
Pfracture = 2640 lbs
Base @ 2.0 %/in


Spec # 1E-K
28 Aug 1980
C2-5951-004
Load/Strain
550°F
Area = .0509
σ_ULT = 95.1 ksi
σ_YS = 76.0 ksi
Y axis @ 20v/in
X axis @ .1v/in .17 €/in
47 1323

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-

Spec. No. YC- 1

Temperature 550 °F

Strain Rate .005 in/in/min.

Est. U.T.S. psi

Initial G.L. 1.013 in.

Initial Dia. .250 in.

Initial Thickness in.

Initial Width in.

Project No. 02-5951-001

Machine No. 22KIP

Date 27 Aug 1980

Initial Area .0491 in²

Top Temperature °F

Bottom Temperature °F

Final Gage Length 1.239 in.

Final Diameter .140 in.

Final Area .0154 ✓ in.²

Maximum Load 4184 lb

0.2% Offset Load 3020 lb

0.02% Offset Load lb

Upper Yield Point lb

$$\text{U.T.S.} = \frac{\text{Maximum Load}}{\text{Initial Area}} = $$ ~~85.4~~ 85.2~~00~~ ~~psi~~ Ksi

$$\text{0.2\% Y.S.} = \frac{\text{0.2\% Offset Load}}{\text{Initial Area}} = $$ ~~61.6~~ 61.5~~00~~ ~~psi~~ Ksi

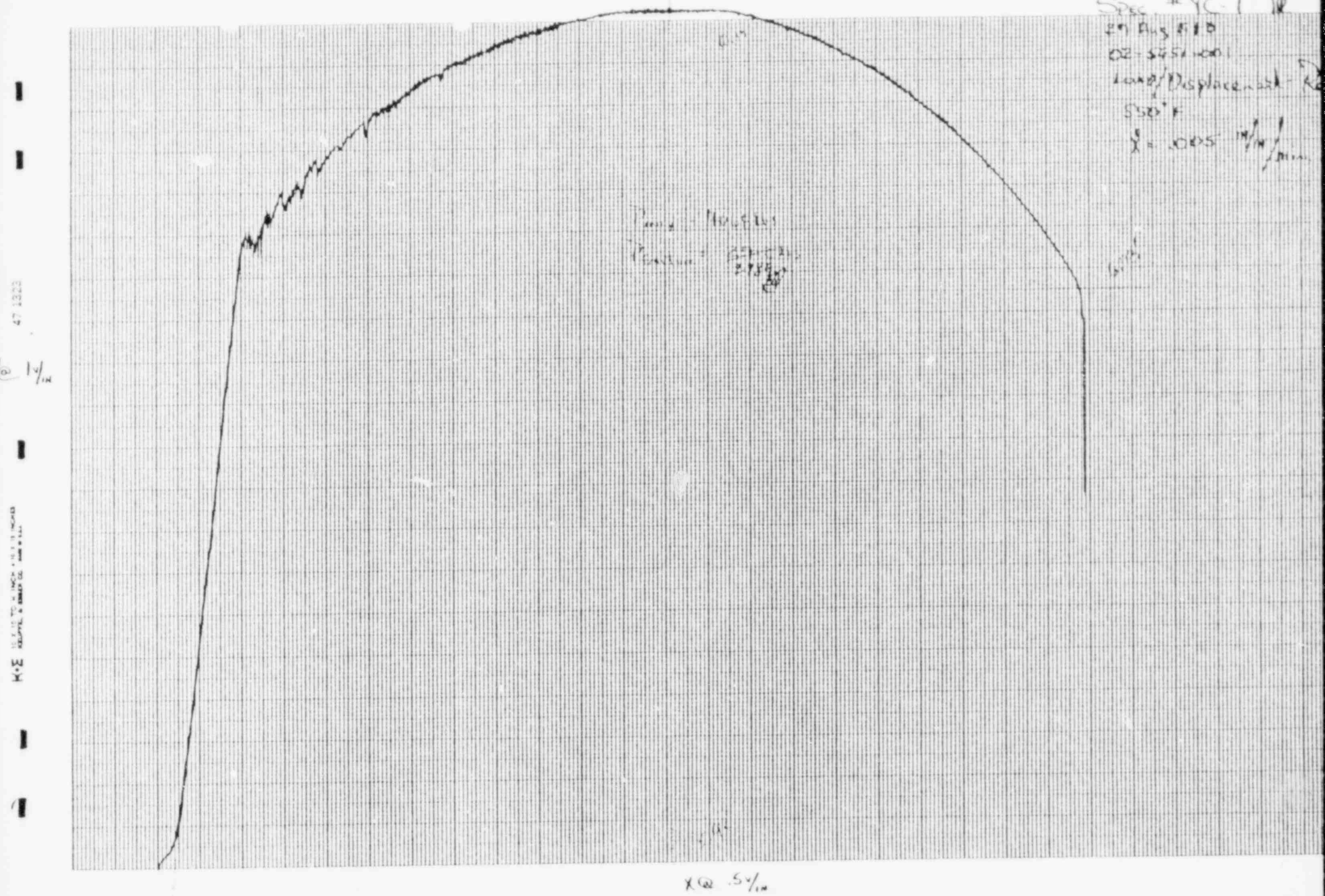
$$\text{0.02\% Y.S.} = \frac{\text{0.02\% Offset Load}}{\text{Initial Area}} = $$ psi

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

$$\text{\% Elongation} = \frac{\text{Final G.L.} - \text{Initial G.L.}}{\text{Initial G.L.}} \times 100 = $$ 22.3 ✓ %

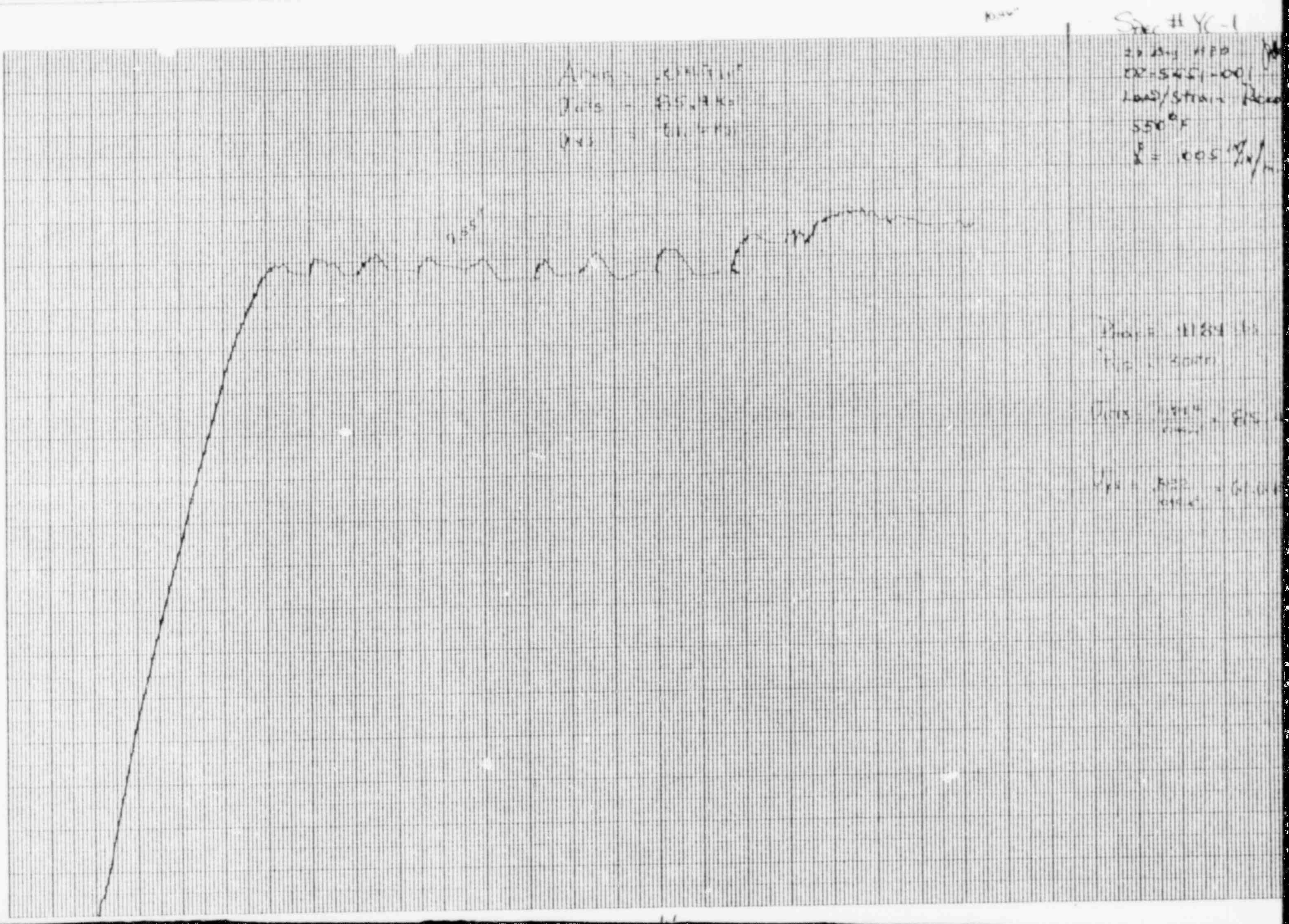
$$\text{\% R.A.} = \frac{\text{Initial Area} - \text{Final Area}}{\text{Initial Area}} \times 100 = $$ 68.6 ✓ %

Signature: Vic Aaron



Spec. #YC-1 18
29 Aug 80
02-5451-001
Load/Displacement - R
550° F
$\dot{\epsilon}$ = .005 in/in/min

[illegible]
[illegible]


Spec # YC-1
27 Day HPD
CR-5651-001
Load/Strain Reco
550°F
ε̇ = .005 %/m
Area = .0[illegible] in²
σUTS = 85.4 Ksi
σYS = 61.6 Ksi
.55"
Pmax = 1184 lb
Py = [illegible]
σUTS = 1184/[illegible] = 85.[illegible]
σYS = 852/[illegible] = 61.6
Ye 1.04/in
47 1323
K+E 10 X 10 TO ½ INCH
KEUFFEL & ESSER CO. MADE IN U.S.A.

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T-

Spec. No. YD-5

Temperature 550 °F

Strain Rate .005 in/in/min

Est. U.T.S. psi

Initial G. L. 1.013 in.

Initial Dia. .250 in.

Initial Thickness in.

Initial Width in.

Project No. 02-5951-001

Machine No. 22 KIP

Date 28 Aug 1980

Initial Area .0491 in²

Top Temperature °F

Bottom Temperature °F

Final Gage Length 1.246 in.

Final Diameter .148 in.

Final Area .0172 in.²

Maximum Load 3984 ~~4000~~ lb

0.2% Offset Load 2960 lb

0.02% Offset Load lb

Upper Yield Point lb

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 81.1 ✓ ~~81.6~~ ~~psi~~ Ksi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 60.3 ✓ ~~60.4~~ ~~psi~~ Ksi

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

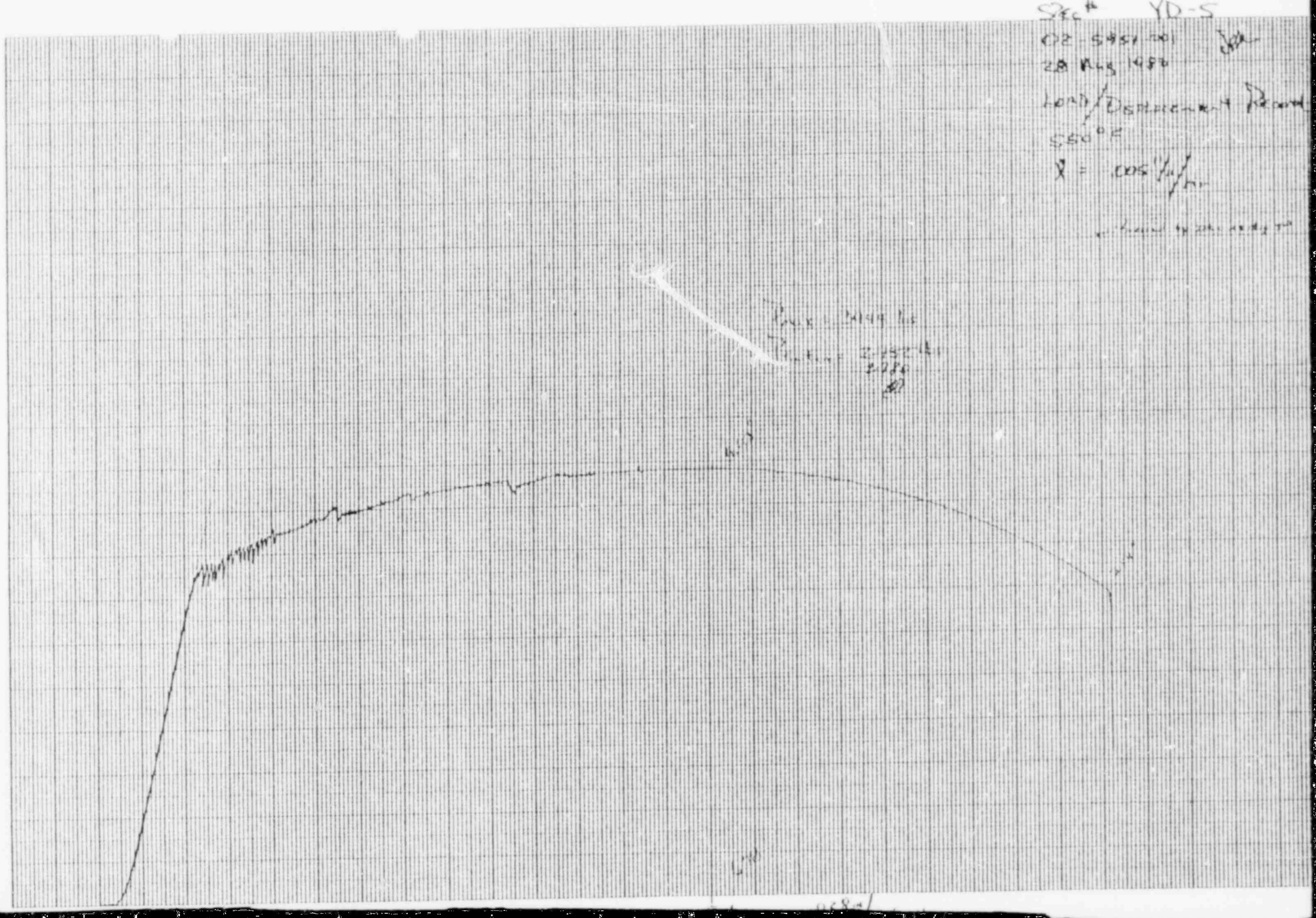
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = psi

% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 23.0 ✓ %

% R. A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 65.0 ~~64.9~~ ✓ %

Signature: Vic

witnessed by TAM 28 Aug 80


Spec # YD-5

02-5451-001

28 Aug 1986

Load/Displacement Record

550 °F

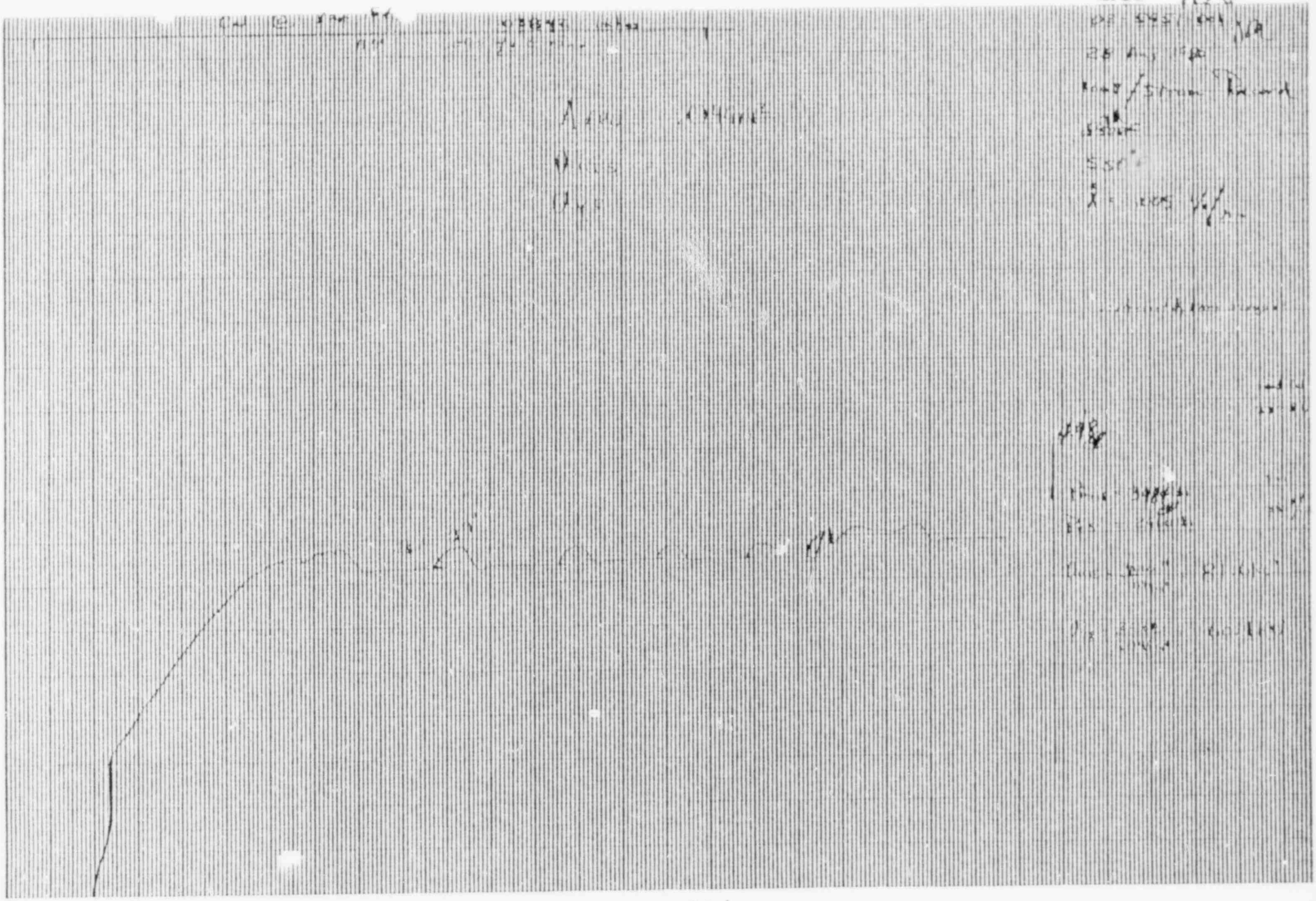
$\dot{X}$ = .005 in/in/hr

P_{max} = 2449 lbs

P_{limit} = 2252 lbs

47 1323

Spec # YD-5



X @ 0.1"/in