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REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR CAPSULE S - TURKEY POINT UNIT NO. 3 CAPSULE S - TURKEY POINT UNIT NO. 4

FINAL REPORT
SwRI Project No. 02-5131
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to

Florida Power & Light Company
P. O. Box 3100
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I. SUMMARY OF RESULTS AND CONCLUSIONS

The analyses of the reactor vessel material surveillance program capsules (coded "S") removed from the Florida Power and Light Company (FPL) Turkey Point Units 3 and 4 nuclear reactor vessels during the 1977-1978 refuelling outages led to the following conclusions:

(1) The intermediate and lower shell forging materials utilized in the reactor pressure vessels of Units 3 and 4 exhibited a low sensitivity to radiation embrittlement. The shelf energy reductions and the transition temperature shifts were equal to or below those predicted by the minimum response curves given in Regulatory Guide 1.99.

(2) Based on the surveillance program results to date and trend curves for low-copper materials, the irradiated properties of the intermediate and lower shell forging materials utilized in the pressure vessels for Units 3 and 4 will be adequate to meet the current requirements of 10CFR50, Appendix G, through the 40-year design lifetime.

(3) Capsule S from Unit No. 3 received a fast neutron fluence of 1.41×10^{19} cm^{-2}, E > 1 MeV. Based on a calculated lead factor of 1.76, the peak fast neutron exposure of the Unit No. 3 reactor vessel is projected to be 2.3×10^{19} cm^{-2} after 10 Effective Full Power Years (EFPY) of operation. The peak end-of-life (32 EFPY) fast neutron fluence is predicted to be 7.4×10^{19} cm^{-2}, in good agreement with the value of 6.65×10^{19} cm^{-2} projected earlier from the analysis of Capsule T.

(4) Capsule S from Unit No. 4 received a fast neutron fluence of 1.25×10^{19} cm^{-2}, E > 1 MeV. Based on a calculated lead factor of 1.76, the peak fast neutron exposure of the Unit No. 4 reactor vessel is

projected to be 2.1 x 10^{19} cm^{-2} after 10 EFPY of operation. The peak end-of-life fluence (32 EFPY) fast neutron fluence is predicted to be 6.7 x 10^{19} cm^{-2}, in excellent agreement with the value of 6.62 x 10^{19} cm^{-2} projected earlier from the analysis of Capsule T.

(5) Capsule V from each reactor vessel should be removed and tested after approximately 10 calendar years (~ 7 EFPY) of operation. The data obtained should provide the information needed to revise the heatup and cooldown limitations for operation beyond 10 EFPY.

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:

- Drop-weight Nil Ductility Temperature (DW-NDT) per ASTM E 208;[(3)]
- 60 deg F below the 50 ft-lb Charpy V-notch (C_V) temperature;
- 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 phosphorous, affect the radiation embrittlement response of ferritic materials.[(5-7)]

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* Superscript numbers refer to references at the end of the text.

There is some disagreement concerning the relationship between increase in RT_{NDT} and copper content. For example, Regulatory Guide 1.99[7] proposes an adjustment to RT_{NDT} proportional to the square root of the neutron fluence. Westinghouse Electric Corporation, in their comments on Regulatory Guide 1.99[8], feels that the proposed relationship overestimates the shift at high fluences (above 10^{19}) and underestimates the shift at low fluences (below 10^{19}). On the other hand, Combustion Engineering, in their comments on Regulatory Guide 1.99[9], suggests 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] It is important to resolve these questions because the analysis of reactor vessel material surveillance program data requires that estimations be made of shifts in RT_{NDT} and C_V upper shelf energy at fluences other than that received by the surveillance capsule.

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, reactor vessel material surveillance programs include specimens machined from the plate or forging, weld metal, and heat affected zone (HAZ) materials which are located in such a region of high neutron flux density. ASTM E 185[10] describes the current recommended practice for monitoring and evaluating the radiation-induced changes occurring in the mechanical properties of pressure vessel beltline materials.

Westinghouse has provided such a surveillance program for the two-unit Turkey Point nuclear power plant. The encapsulated C_V specimens are attached to the O.D. surface of each thermal shield where the fast neutron flux density is approximately twice that at the adjacent vessel wall surface. Therefore, the increases (shifts) in transition temperatures of the materials in the pressure vessel are generally less than the corresponding shifts observed in the surveillance specimens. However, because of azimuthal variations in neutron flux density, some capsule fluences may be less than the maximum vessel fluence in a corresponding exposure period. For example, the first capsules removed from Turkey Point Units 3 and 4 were reported to lead the maximum exposure point on the vessel I.D. by a factor of 2.48 while other capsules scheduled to be removed later are calculated to receive less than half of the fluence accumulated at the point of maximum vessel exposure. The capsules also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

The Turkey Point Units 3 and 4 material 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 assist in judging the credibility of the C_V data. In addition, the material surveillance capsules contain wedge opening loading (WOL) fracture mechanics specimens. Current technology limits the testing of these specimens at temperatures well below the minimum service temperature to obtain valid fracture mechanics data per ASTM E 399[11], "Standard Method of Test for Plane-Strain Fracture Toughness of

Metallic Materials." However, recent work reported by Mager and Witt[12] and Loss[13] may lead to methods for evaluating high-toughness materials with small fracture mechanics specimens. Currently, the NRC suggests storing these specimens until an acceptable testing procedure has been defined.

Capsule T was removed from Turkey Point Unit No. 3 during the 1974 refuelling outage, and the results have been reported.[14] The weld metal was found to be more sensitive to neutron radiation embrittlement than the forging or HAZ material contained in the capsule, and it was concluded that the weld metal is the most limiting beltline material in the Unit No. 3 vessel. Capsule T was removed from Turkey Point Unit No. 4 during the 1975 refuelling outage, and those results have also been reported.[15] The results indicated that the vessel beltline weld metal was more sensitive to radiation embrittlement than the lower shell forging and HAZ materials included in the capsule.

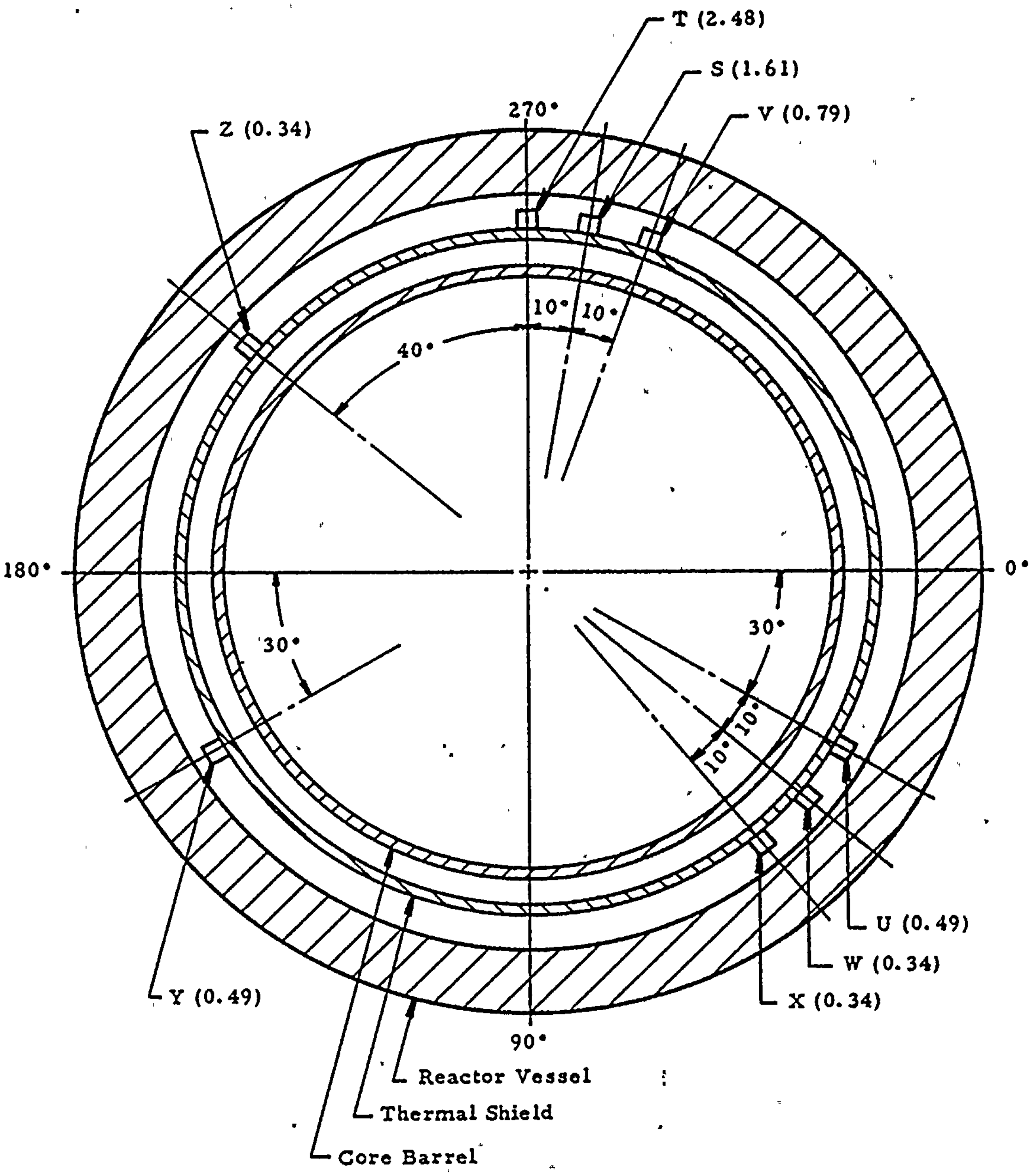
This report describes the results obtained from testing the contents of Capsule S removed from Unit No. 3 and Capsule S removed from Unit No. 4. These data are analyzed to estimate the radiation-induced changes in the mechanical properties of the respective pressure vessel forging materials at the time of the 1977-78 refuelling outage as well as predicting the changes expected to occur at selected times in the future operation of the Turkey Point nuclear power plant.

III. DESCRIPTION OF MATERIAL SURVEILLANCE PROGRAM

A. Introduction

The Turkey Point Units 3 and 4 material surveillance programs are described in detail in WCAP 7656[16] and WCAP 7660[17], respectively. Eight materials surveillance capsules (five Type I and three Type II) were placed in each reactor vessel between the thermal shield and the vessel wall prior to startup, see Figure 1. The vertical center of each capsule is opposite the vertical center of the core. The reported ratios of neutron flux density at the capsule location to the maximum flux density on the vessel I.D. are given by the factors in parentheses following the capsule identification letter in Figure 1. The Type I capsules each contain Charpy V-notch, tensile, and WOL specimens machined from the two vessel forgings located at the core beltline plus Charpy V-notch specimens machined from one of two reference heats of ferritic steel utilized in the Westinghouse surveillance programs. The Type II capsules include specimens machined from weld metal and HAZ material which were intended to be representative of those materials in the core beltline region of each vessel as well as one heat of forging material and the reference steels.

All test specimens were machined from the materials at the quarter-thickness (1/4T) location.[16,17] The base metal tensile and C_V specimens were oriented with their long axis parallel to the principal working direction; the C_V notches were perpendicular to the major forging surfaces. The WOL specimens were machined with the simulated crack perpendicular to the principal working direction and to the forging surfaces. All mechanical



Note: Numbers in parentheses indicate lead factors for the vessel I. D.

FIGURE 1. ARRANGEMENT OF SURVEILLANCE CAPSULES IN THE TURKEY POINT UNITS 3 and 4 PRESSURE VESSELS

test specimens, see Figure 2, were taken at least one plate thickness from the quenched edges of the forging material.

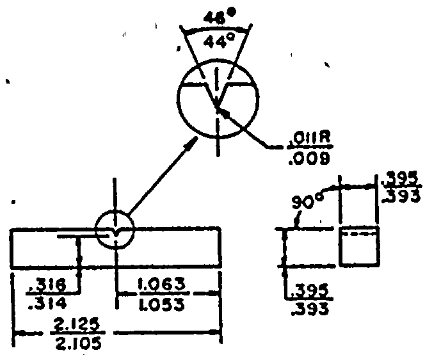
Capsule S (a Type I capsule) was removed from each of the Turkey Point nuclear reactor vessels during the 1977-1978 refuelling outages. The arrangement of specimens within these capsules is shown in Figure 3. Additional details concerning the contents of the capsules from each unit are discussed below.

B. Surveillance Capsule Materials - Unit No. 3, Capsule S

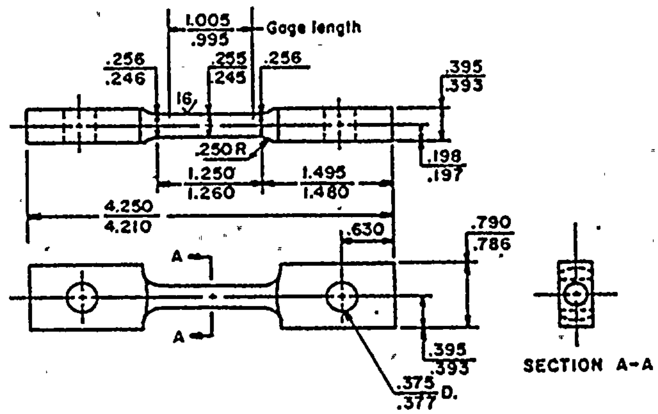
Babcock and Wilcox Company supplied prolongations from two 7-7/8-in. thick forged rings (Heat 123P461VA-1 and 123S266VA-1 produced by Bethlehem Steel Company) of SA 508, Class 2 steel used for the FPL Unit No. 3 reactor pressure vessel intermediate and lower shell course, respectively, and a weldment which joined sections of the two forgings. Correlation monitor material, supplied by U.S. Steel Corporation through Subcommittee II of ASTM Committee E 10, was obtained from a 6-in. thick A 302 Grade B plate which was melted using fine-grain practice and a transverse-to-longitudinal rolling ratio of one to one. The chemistries and heat treatments of the vessel surveillance materials contained in Capsule S from Unit No. 3 are summarized in Table I.

The capsule contained 28 Charpy V-notch specimens (10 from each of the two vessel forging materials plus 8 from the reference steel plate); 4 tensile specimens (2 from each forging material); and 6 WOL specimens (3 from each forging heat). The specimen numbering system and location within Capsule S for Unit No. 3 is shown in Table II.

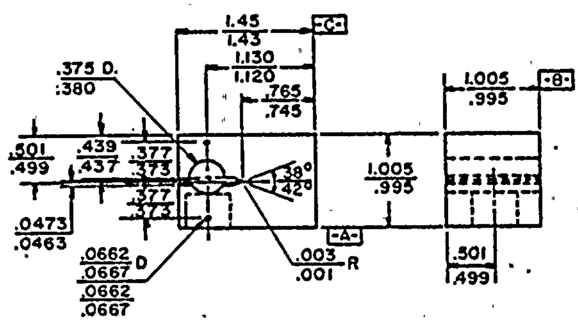
The capsule also was reported to contain the following dosimeters for determining the neutron flux density:[16]



(a) Charpy V-notch Impact Specimen

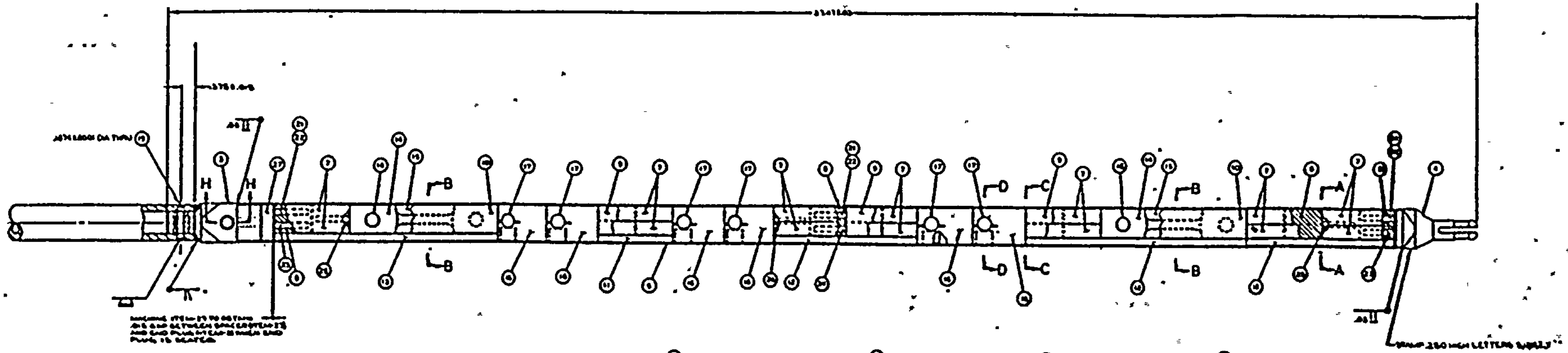


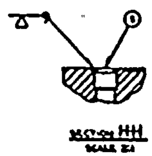
(b) Tensile Specimen



(c) Wedge Opening Loading Specimen

FIGURE 2. VESSEL MATERIAL SURVEILLANCE SPECIMENS





BILL OF MATERIAL

Item	Name	Drawing No.	Material	Qty
1	TOP SUPPORT PLUG	[illegible]		1
2	EXTENSION TUBE		SEE NOTE "A"	1
3	TOP END PLUG	[illegible]		1
4	BOTTOM END PLUG	[illegible]		1
5	PLUG	[illegible]		1
6	ENCLOSURE HALF	[illegible]		2
7	IMPACT SPECIMEN	[illegible]		28
8	SPACER	[illegible]		3
9	SPACER	[illegible]		4
10	SPACER	[illegible]		2
11	SPACER	[illegible]		1
12	SPACER	[illegible]		2
13	SPACER	[illegible]		2
14	TENSILE SPECIMEN	[illegible]		4
15	TENSILE COVER	[illegible]		8
16	WOL SPECIMEN	[illegible]		6
17	.375 x 1.000 LG. DOWEL PIN		CARBON STL.	8
18	.375 x .375 LG. DOWEL PIN		CARBON STL.	[illegible]
19	.187 x .875 LG. DOWEL PIN		COML. STL.	1
20	.500-13UNC x .375 LG. FLAT PT. SOC. SET SCR.		CARBON STL.	6
21	COBALT SPECIMEN		SEE NOTE "D"	6
22	CADMIUM SPECIMEN		CADMIUM	3
23	COPPER SPECIMEN		COPPER	2
24	NICKEL SPECIMEN		NICKEL	1
25	579°F THERMAL MONITOR		[illegible]	2
26	590°F THERMAL MONITOR		[illegible]	1
27	SPACER		TO BE SPEC. BY ENGR.	1

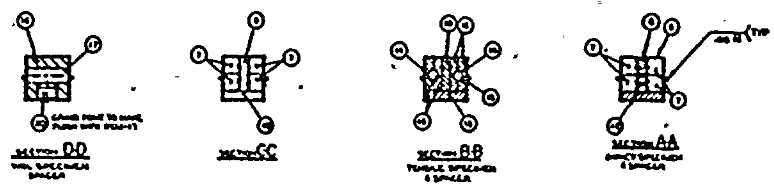


FIGURE 3. ARRANGEMENT OF SPECIMENS IN CAPSULE S, TURKEY POINT UNIT NOS. 3 AND 4

TABLE I

FPL TURKEY POINT UNIT NO. 3 REACTOR VESSEL SURVEILLANCE MATERIALS(14,16)

Heat Treatment History

Intermediate and Lower Shell Forgings	1550 F - 13 hours - water-quenched 1210 F - 18 hours - air-cooled 1125 F - 10-1/2 hours - furnace-cooled to 600 F
Weldment	1125 F - 10-1/4 hours - furnace-cooled to 600 F
Correlation Monitor	1650 F - 4 hours - water-quenched to 300 F 1200 F - 6 hours - furnace-cooled

Chemical Composition (wt-%)

Element	Lower Shell 123S266VA-1	Intermediate Shell 123P461VA-1	Correlation Monitor
C	0.19/0.21	0.20	0.24
Mn	0.61/0.62	0.64/0.64	1.34
P	0.010	0.010	0.011
S	0.008	0.010	0.023
Si	0.20/0.19	0.26	0.23
Ni	0.68/0.66	0.70	0.18
Cr	0.38	0.40/0.39	0.11
V	0.02	0.02	-
Mo	0.58/0.59	0.62	0.51
Co	0.015/0.016	0.011/0.010	-
Cu	0.079	0.058	0.20
Sn	0.008	0.010	-
Zn	0.001	0.001	-
Al	0.005	0.005	-
N_2	0.003	0.003	-
Ti	0.001*	0.001*	-
Sb	0.001*	0.001*	-
As	0.005*	0.005*	-
B	0.003*	0.003*	-
Zr	0.001*	0.001*	-

* Not detected. The number indicates the minimum limit of detection.

TABLE II

SPECIMEN IDENTIFICATION AND LOCATION IN TURKEY POINT VESSEL MATERIAL SURVEILLANCE CAPSULES(16,17)

Specimen Type		Capsule S
Charpy V	Top	S9, S10 P9, P10
Tensile	↑	P1, P2
WOL		P3
WOL		P2
Charpy V		S7, S8 P7, P8
WOL		P1
WOL		S3
Charpy V	Capsule	R7, R8 R5, R6
Charpy V		R3, R4 R1, R2
WOL		S2
WOL		S1
Charpy V		S5, S6 P5, P6
Tensile	↓	S1, S2
Charpy V	Bottom	S3, S4 P3, P4
Charpy V		S1, S2 P1, P2

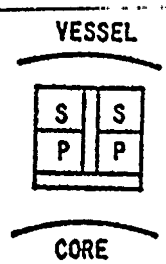
Specimen Code - Unit 3

S - Forging 123S266VA-1
P - Forging 123P461VA-1
R - ASTM Correlation Monitors

Specimen Code - Unit 4

S - Forging 122S180VA-1
P - Forging 123P481VA-1
R - ASTM Correlation Monitors

Specimen Orientation



Target Element	Form	Quantity
Copper	Bare wire	2
Nickel	Bare wire	1
Cobalt (in aluminum)	Bare wire	3
Cobalt (in aluminum)	Cd shielded wire	3

In addition, slices were taken from ten C_V specimens to serve as iron dosimeters.

Three eutectic alloy thermal monitors had been inserted in holes in the steel spacers in the capsule. Two (located top and bottom) were 2.5% Ag and 97.5% Pb with a melting point of 579 F. The third (located at the cneter of the capsule) was 1.75% Ag, 0.75% Sn, and 97.5% Pb having a melting point of 590 F.

C. Surveillance Capsule Materials - Unit No. 4, Capsule S

Babcock and Wilcox Company supplied prolongations from two 7-7/8 in. thick forged rings (Heat 123P481VA-1 and 122S180VA-1 produced by Bethlehem Steel Company) of SA 508, Class 2 steel used for the FPL Unit No. 4 reactor pressure vessel intermediate and lower shell course, respectively, and a weldment which joined sections of the two forgings. Correlation monitor material was supplied by the Oak Ridge National Laboratory from plate material used in the AEC-sponsored Heavy Section Steel Technology (HSST) Program. This material was obtained from a Lukens Steel Company 12-in. thick A533 Grade B, Class 1 plate (HSST Plate 02) which has been provided to Subcommittee II of ASTM Committee E10 on Radioisotopes and Radiation Effects to serve as correlation monitor material in reactor vessel surveillance programs. The chemistries and heat treatments of the vessel surveillance materials contained in Capsule S from Unit No. 4 are summarized in Table III.

TABLE III

FPL TURKEY POINT UNIT NO. 4 REACTOR VESSEL SURVEILLANCE MATERIALS[17]

	Heat Treatment History
Lower Shell (Heat 122S180VA-1)	1550 F - 10-1/4 hours - water-quenched 1210 F - 18 hours - air-cooled 1125 F - 10-1/2 hours - furnace-cooled to 600 F
Correlation Monitor	1675 ± 25 F - 4 hours - air-cooled 1600 ± 25 F - 4 hours - water-quenched 1225 ± 25 F - 4 hours - furnace-cooled 1150 ± 25 F - 40 hours - furnace-cooled to 600 F

Chemical Composition (wt-%)

Element	Lower Shell 122S180VA-1	Intermediate Shell 123P481VA-1	Correlation Monitor
C	0.21	0.22	0.22
Mn	0.67	0.67	1.48
P	0.011	0.010	0.012
S	0.009	0.009	0.018
Si	0.23	0.20	0.25
Ni	0.70	0.71	0.68
Cr	0.31	0.33	-
V	0.001	0.002	-
Mo	0.56	0.56	0.52
Co	0.015	0.017	-
Cu	0.056	0.054	0.14
Sn	0.008	0.008	-
Zn	0.001*	0.001*	-
Al	0.008	0.008	-
N_2	0.002	0.001	-
Ti	0.001*	0.001*	-
Pb	0.001*	0.001*	-
As	0.005	0.004	-
B	0.003*	0.003*	-
Zr	0.004	0.005	-
W	0.001*	0.001*	-
Nb	0.001	0.002	-
Ta	0.002	0.003	-

* Not detected. The number indicates the minimum limit of detection.

The capsule contained 28 Charpy V-notch specimens (10 from each of the two vessel forging materials plus 8 from the reference steel plate); 4 tensile specimens (2 from each forging material); and 6 WOL specimens (3 from each forging heat). The specimen numbering system and location within Capsule S for Unit No. 4 is given in Table II.

The neutron flux wires and thermal monitors contained in the capsule were reported to be the same type and location as those contained in Capsule S from Unit No. 3.[17] In addition, slices were taken from six C_V specimens to serve as iron dosimeters.

IV. TESTING OF SURVEILLANCE SPECIMENS

A. Introduction

The capsule shipment, capsule opening, specimen testing, and reporting of results were carried out under Quality Assurance Plans prepared by Southwest Research Institute (SwRI) and approved by Florida Power and Light Company (FP&L). These plans are on file at SwRI. Applicable SwRI Nuclear Project Operating Procedures which were called out in the Turkey Point Unit No. 3 project plan include:

- XI-MS-1, "Determination of Specific Activity of Neutron Radiation Detector Specimen"
- XI-MS-3, "Conducting Tension Tests on Metallic Materials"
- XI-MS-4, "Charpy Impact Tests on Metallic Materials"
- XIII-MS-1, "Opening Radiation Surveillance Capsules and Handling and Storing Specimens"
- XIII-MS-102, "Shipment of Westinghouse PWR Vessel Material Surveillance Capsule"

The applicable SwRI Nuclear Project Operating Procedures which were called out in the Turkey Point Unit No. 4 project plan include:

- XI-MS-101, "Determination of Specific Activity and Analysis of Neutron Radiation Detector Specimen"
- XI-MS-103, "Conducting Tension Tests on Metallic Specimens"
- XI-MS-104, "Conducting Impact Tests on Metallic Specimens"
- XIII-MS-103, "Opening Radiation Surveillance Capsules and Handling and Storing Specimens"
- XIII-MS-104, "Shipment of Westinghouse PWR Vessel Material Surveillance Capsule Using SwRI Cask and Equipment"

Copies of the above documents are on file at SwRI.

B. Opening of Surveillance Specimen Capsules and Recovery of Specimens

The capsule shell had been fabricated by making two long seam welds to join two half-shells together. The capsule ends were sawed off, then the long seam welds were milled away using a vertical milling machine. The top half of the capsule shell was removed, and the specimens and spacer blocks were carefully retrieved and placed in an indexed receptacle so that capsule location was identifiable.

After the disassembly had been completed, the specimens were carefully checked for identification and location as listed in WCAP 7656[16] and WCAP 7660.[17] No discrepancies were found. The thermal monitors and dosimeter wires were removed from holes in the spacer blocks and placed in indexed receptacles.

C. Neutron Dosimetry

The specific activities of the dosimeters were determined with an NDC 2200 multichannel analyzer and an NaI(Th) 3 in. x 3 in. scintillation crystal. The calibration of the equipment was accomplished with appropriate standards and an interlaboratory cross check with two independent counting laboratories on ^{60}Co-, ^{54}Mn- and ^{58}Co-containing dosimeter wires. All activities were corrected to the time-of-removal (TOR) at reactor shutdown. Infinitely dilute saturated activities (A_{SAT}) were calculated for each of the dosimeters because A_{SAT} is directly related to the integral of the energy-dependent microscopic activation cross section and the neutron flux density. The relationship between A_{TOR} and A_{SAT} is given by:

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

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

T_m = equivalent operating days at 2200 MW_t for operating period m;

t_m = decay time after operating period m, days.

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:

$$\phi = \frac{A^i_{SAT}}{\bar{\sigma}_i} \qquad (2)$$

where: ϕ = energy-dependent neutron flux density, $cm^{-2} \cdot sec^{-1}$;

A^i_{SAT} = saturated activity of the i^{th} activation product at full power, dps/target nculeus;

$\bar{\sigma}_i$ = spectrum-averaged cross section for the i^{th} target nucleus, cm^2.

The neutron flux energy and spatial distribution were calculated for the Turkey Point pressure vessels with the DOT 3.5 two-dimensional discrete ordinates transport code, a 22-group neutron cross section library, a P_1 expansion of the scattering matrix and an S_8 order of angular quadrature. Using a one-eighth segment as being representative because of the symmetry involved, the core, core barrel, thermal shield, specimen capsules, and pressure vessel were described in R-θ coordinates for this computation, see Figure 4.

The calculations which produced the lead factors given in Figure 1 were based on the core power distribution given in the Turkey Point Units 3 and 4 FSAR. The DOT 3.5 calculations made in support of the Capsule S analyses, utilizing core power distributions based on plant records supplied by FPL, indicated that the Capsule S lead factors were higher than

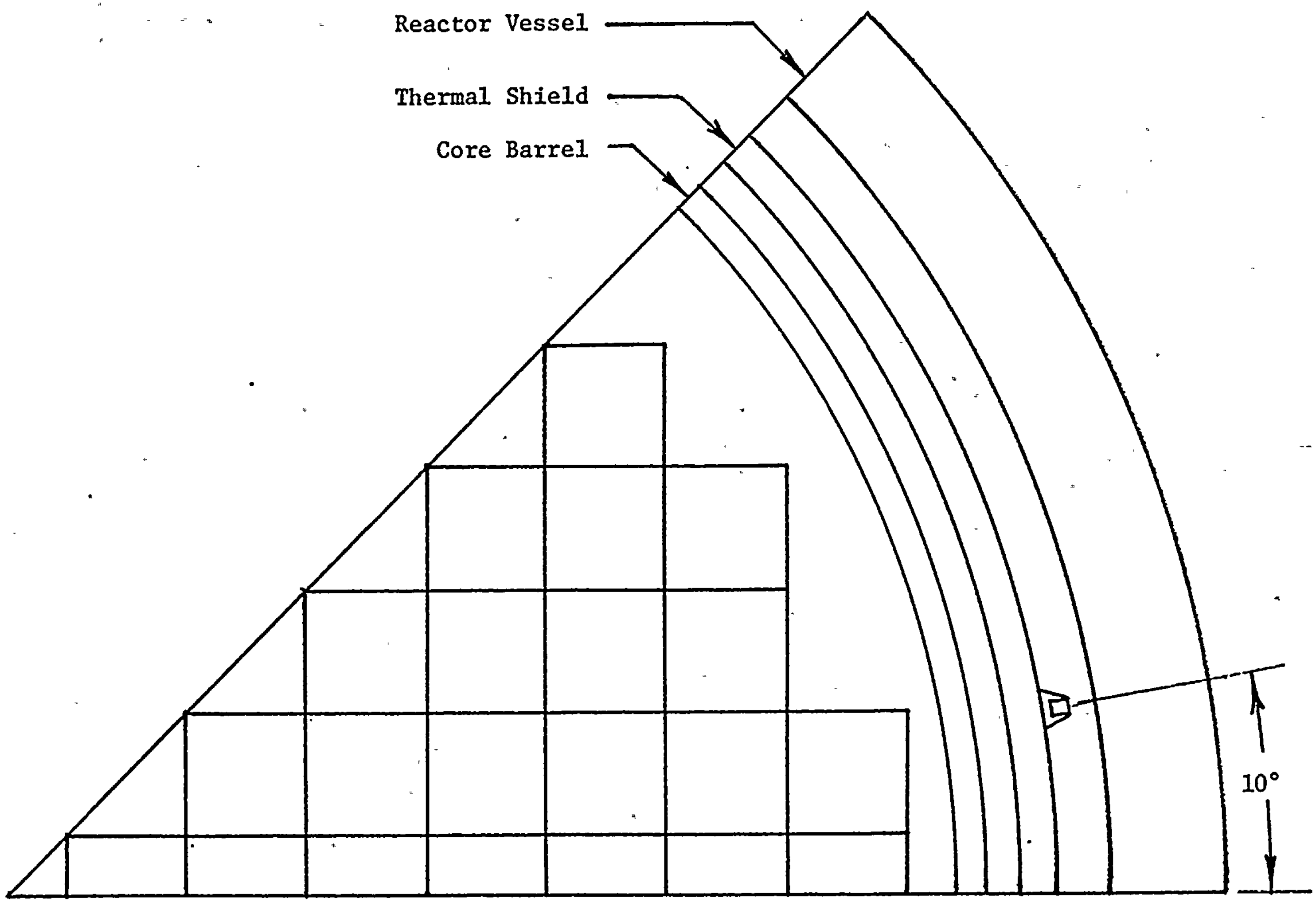


FIGURE 4. LOCATION OF CAPSULE S WITHIN ONE-EIGHTH SEGMENT USED AS MODEL FOR DISCRETE ORDINATE TRANSPORT CALCULATIONS

originally calculated. This is primarily caused by a reduction in the maximum calculated flux incident on the reactor vessel, resulting from the altered core power distribution, as shown below:

Source of Core Power Distribution	Calculated Flux, $cm^{-2} \cdot sec^{-1}$, E > 1 Mev: Capsule S	Calculated Flux, $cm^{-2} \cdot sec^{-1}$, E > 1 Mev: Vessel Maximum	Lead Factor
FSAR, Units 3 and 4	1.13×10^{11}	7.13×10^{10}	1.6
Plant Records, Unit 3	1.16×10^{11}	5.45×10^{10}	2.1
Plant Records, Unit 4	1.18×10^{11}	5.58×10^{10}	2.1

An additional contribution to the difference in these lead factors is the flux perturbation caused by the presence of a surveillance capsule. In the calculation based on the FSAR power distribution, the model did not include iron in the capsule position. However, in the calculation based on the actual core power distribution, the model did include iron in the capsule position. An analysis of the latter results indicates that the Capsule S lead factor is increased by 10% when the model includes iron in the capsule position. Therefore, the lead factor based on the FSAR power distribution should be 1.76 instead of 1.6. The extrapolation of capsule fluxes to the vessel wall described in Section VII of this report utilizes the perturbation-corrected lead factor of 1.76 since it is more conservative than the 2.1 factor. Also, the maximum vessel flux might be altered by future core loading schemes.

The spectrum-averaged cross sections needed for Equation (2) were calculated for the iron and copper reaction dosimeters as follows:

$$\bar{\sigma}_{cs}(> 1\text{ MeV}) = \frac{\sum_{0.11}^{12.5\text{ MeV}} \sigma_D(E)\phi(E)dE}{\sum_{1.00}^{12.5\text{ MeV}} \phi(E)dE} \quad . \qquad (3)$$

D. Mechanical Property Tests

The irradiated Charpy V-notch specimens were tested on an instrumented SATEC impact machine. The test temperatures were selected to develop the ductile-brittle transition and upper shelf regions for each material. Impact energy and lateral expansion transition curves were developed for each material using a tanh curve fitting program based on the following equation:

$$C_v \text{ parameter} = A + B \tanh \left[\frac{T-T_1}{T_2}\right]$$

where A, B, T_1, and T_2 are adjustable constants defined by employing a nonlinear least squares fitting routine. In this relationship, the tanh function varies from -1 at the low temperature extreme to +1 at the upper temperature extreme. Therefore, A - B is equal to the lower shelf parameter and A + B is equal to the upper shelf parameter.

Tensile tests were carried out in a Dillon 10,000-lb capacity tester equipped with a strain gage extensometer, load cell, and autographic recording equipment. The tensile specimens were tested at 250 F and 550 F.

Testing of the WOL specimens was deferred at the request of Florida Power & Light Company. The specimens are in storage at the SwRI radiation laboratory.

E. Check Chemical Analyses for Copper

Three tested Charpy specimens, representing the two forging heats and the correlation monitor material, were analyzed for copper content in accordance with ASTM Method E 322.

V. CAPSULE S TEST RESULTS - TURKEY POINT UNIT NO. 3

A. Neutron Dosimetry

Two discrepancies were noted when the gamma activities of the dosimeter wires were measured. The ^{60}Co activity of the cadmium-covered wire from the middle position was over four orders of magnitude below those of the top and bottom cadmium-covered dosimeters. Also, the nickel dosimeter did not exhibit a ^{58}Co peak. Using an energy dispersive x-ray diffraction technique, the middle nickel wire from the Unit No. 3 capsule was identified as being a Co-Al material, and the middle cadmium-covered wire from the Unit No. 3 capsule was identified as being a copper-base material.

The specific activities obtained from the remaining dosimeters in Capsule S are presented in Table IV. Two items are of interest: (1) the weights of the two dosimeters in question support the qualitative analyses described above; (2) the iron dosimetry results indicate a 20% decrease in neutron flux across the capsule in the radial direction, and that the capsule had been inserted in the vessel 180° out of phase. The saturated activities of the dosimeters, also given in Table IV, were based on the summary of plant operations given in Table V.

The neutron spectrum calculated for the Capsule S location is presented in Table VI along with the spectrum-averaged cross sections computed for the iron and copper dosimeters. The mean value of fast neutron flux density of the Capsule S location determined from the ten iron and two copper flux monitors was 1.29×10^{11} $cm^{-2} \cdot sec^{-1}$, $E > 1$ MeV. Since Unit No. 3 operated for an equivalent 1266.97 full power days up to the 1977-78 refuelling outage, the resulting value of neutron fluence for Capsule S is 1.41×10^{19} cm^{-2}, $E > 1$ MeV.

TABLE IV

SUMMARY OF NEUTRON DOSIMETRY RESULTS
CAPSULE S
TURKEY POINT UNIT NO. 3

Monitor Identification[a]	Radial Location in Capsule[14]	Activation Reaction	Dosimeter Weight (mg)	Measured Activity (dps/mg)	Saturated Activity (dps/mg)
Fe-S9 (Top)	Vessel Side	$^{54}Fe(n,p)^{54}Mn$	54.5	7.21×10^3	9.24×10^3
Fe-S7 (Top-Middle)			70.8	6.36×10^3	8.15×10^3
Fe-R8 (Middle)			45.6	5.78×10^3	7.40×10^3
Fe-S5 (Bottom-Middle)			59.3	6.63×10^3	8.49×10^3
Fe-S1 (Bottom)			62.3	6.44×10^3	8.26×10^3
Fe-P9 (Top)	Core Side		220.9	5.60×10^3	7.18×10^3
Fe-P7 (Top-Middle)			261.5	5.46×10^3	7.00×10^3
Fe-R6 (Middle)			167.4	5.16×10^3	6.61×10^3
Fe-P5 (Bottom-Middle)			233.0	5.69×10^3	7.29×10^3
Fe-P1 (Bottom)			270.8	5.29×10^3	6.77×10^3
Cu (Top)	Center	$^{63}Cu(n,\alpha)^{60}Co$	51.4	2.25×10^2	6.53×10^2
Cu (Bottom)			53.0	2.24×10^2	6.53×10^2
Ni (Middle)		$^{58}Ni(n,p)^{58}Co$	8.7	(b)	(b)
Co (Top)		$^{59}Co(n,\gamma)^{60}Co$	9.7	2.34×10^7	6.79×10^7
Co-Cd (Top)			9.4	1.11×10^7	3.23×10^7
Co (Middle)			9.5	2.07×10^7	6.02×10^7
Co-Cd (Middle)			30.7	(c)	(c)
Co (Bottom)			9.9	2.01×10^7	5.83×10^7
Co-Cd (Bottom)			7.9	9.40×10^6	2.73×10^7

(a) See Table II for location within capsule.
(b) Monitor material was identified as being bare Co in Al.
(c) Monitor material was identified as being a Cu-base alloy (purity undetermined).

TABLE V

SUMMARY OF PLANT OPERATIONS
TURKEY POINT UNIT NO. 3

Oper. Period	Dates Start	Dates Stop	Operating Days	Shutdown Days	Reactor Power Output (Mwd_t)	Equivalent Oper. Days, T_m	Decay Time after Period, t_m
1	11/2/72	11/4/72	2	-	14	0.01	1846
	11/4/72	11/5/72	-	1	-		
2	11/5/72	11/13/72	8	-	3,500	1.59	1837
	11/13/72	11/18/72	-	5	-		
3	11/18/72	11/21/72	3	-	577	0.26	1829
	11/21/72	12/4/72	-	13	-		
4	12/4/72	12/9/72	5	-	3,035	1.38	1811
	12/9/72	12/22/72	-	13	-		
5	12/22/72	1/16/73	25	-	21,225	9.65	1773
	1/16/73	1/23/73	-	7	-		
6	1/23/73	3/6/73	42	-	49,954	22.71	1724
	3/6/73	3/13/73	-	7	-		
7	3/13/73	4/6/73	24	-	30,845	14.02	1693
	4/6/73	4/16/73	-	10	-		
8	4/16/73	7/27/73	102	-	165,595	75.27	1581
	7/27/73	8/9/73	-	13	-		
9	8/9/73	10/21/73	73	-	112,547	51.16	1495
	10/21/73	10/29/73	-	8	-		
10	10/29/73	12/7/73	39	-	60,572	27.53	1448
	12/7/73	12/21/73	-	14	-		
11	12/21/73	12/27/73	6	-	9,122	4.15	1428
	12/27/73	12/30/73	-	3	-		
12	12/30/73	3/19/74	79	-	129,361	58.80	1346
	3/19/74	3/31/74	-	12	-		
13	3/31/74	5/12/74	42	-	83,442	37.93	1292
	5/12/74	5/13/74	-	1	-		
14	5/13/74	6/8/74	26	-	53,782	24.45	1265
	6/8/74	6/14/74	-	6	-		
15	6/14/74	9/14/74	92	-	180,930	82.24	1167
	9/14/74	9/23/74	-	9	-		
16	9/23/74	10/5/74	12	-	20,104	9.14	1146
	10/5/74	12/15/74	-	71	-		
17	12/15/74	3/4/75	79	-	152,829	69.47	996
	3/4/75	3/9/75	-	5	-		
18	3/9/75	7/16/75	129	-	270,408	122.91	862
	7/16/75	7/20/75	-	4	-		
19	7/20/75	10/26/75	98	-	200,868	91.30	760
	10/26/75	12/25/75	-	60	-		
20	12/25/75	1/2/76	8	-	13,035	5.92	692
	1/2/76	1/3/76	-	1	-		
21	1/3/76	2/22/76	50	-	107,391	48.81	641
	2/22/76	2/28/76	-	6	-		
22	2/28/76	3/7/76	8	-	16,907	7.68	627
	3/7/76	3/12/76	-	5	-		
23	3/12/76	5/1/76	50	-	106,529	48.42	572
	5/1/76	5/3/76	-	2	-		
24	5/3/76	5/28/76	25	-	53,240	24.20	545
	5/28/76	6/3/76	-	6	-		
25	6/3/76	6/20/76	17	-	35,010	15.91	522
	6/20/76	6/26/76	-	6	-		
26	6/26/76	8/13/76	48	-	103,731	47.15	468
	8/13/76	8/17/76	-	4	-		
27	8/17/76	8/25/76	8	-	16,556	7.53	456
	8/25/76	9/1/76	-	7	-		
28	9/1/76	11/14/76	74	-	159,357	72.44	375
	11/14/76	1/19/77	-	66	-		
29	1/19/77	4/24/77	95	-	196,939	89.52	214
	4/24/77	4/29/77	-	5	-		
30	4/29/77	7/20/77	82	-	173,054	78.66	127
	7/20/77	7/22/77	-	2	-		
31	7/22/77	11/24/77	125	-	256,866	116.76	0
				Total	2,787,325	1,266.97	

TABLE VI

FAST NEUTRON SPECTRUM AND FOIL ACTIVATION
CROSS SECTIONS FOR CAPSULE S
TURKEY POINT UNIT NO. 3

Energy Range (MeV)	DOT 3.5 Calculated Neutron Flux	$^{54}Fe(n,p)^{54}Mn$ Cross Section (barns)	$^{63}Cu(n,\gamma)^{60}Co$ Cross Section (barns)
10.00 - 12.5	3.04×10^{8}	0.521	4.78×10^{-2}
8.18 - 10.0	9.61×10^{8}	0.578	4.26×10^{-2}
6.36 - 8.18	2.57×10^{9}	0.578	1.53×10^{-2}
4.96 - 6.36	4.96×10^{9}	0.491	2.85×10^{-3}
4.06 - 4.96	5.00×10^{9}	0.352	2.78×10^{-4}
3.01 - 4.06	9.21×10^{9}	0.222	1.12×10^{-4}
2.35 - 3.01	1.52×10^{10}	0.098	5.96×10^{-5}
1.83 - 2.35	1.87×10^{10}	0.024	3.86×10^{-5}
1.11 - 1.83	5.03×10^{10}	0.0025	2.09×10^{-5}
1.00 - 1.11	0.92×10^{10}	-	1.20×10^{-5}
0.55 - 1.00	7.02×10^{10}	-	5.48×10^{-6}
0.11 - 0.55	1.24×10^{11}	-	8.46×10^{-7}

$\bar{\sigma}_{Fe}$ = 0.0902 barns (E > 1 MeV)

$\bar{\sigma}_{Cu}$ = 0.000985 barns (E > 1 MeV)

Much of the early work published on the radiation-induced embrittlement of ferritic steels correlated shifts in ductile-brittle transition temperature with neutron fluence calculated on the assumption that the neutron energies were distributed according to a fission neutron spectrum. To provide information for reference only, the Unit No. 3 Capsule S fast neutron flux density based on a fission-spectrum cross section of 98.26 mb (E > 1 MeV) for Fe[18] and 0.000606 mb (E > 1 MeV) for Cu[18], is calculated to be 1.31 x 10^{11} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. The Unit No. 3 Capsule S fast neutron flux (E > 1 MeV) computed with the DOT 3.5 code was 1.20 x 10^{11} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV.

B. Thermal Monitors

The thrmal monitors were examined and had not melted. This indicates that the capsule did not reach 579 F during the exposure period.

C. Mechanical Property Test Results

The Charpy V-notch impact results obtained from specimens contained in Capsule S are given in Table A-1 in Appendix A. The transition curves developed for each material using a tanh curve-fitting technique are also presented in Appendix A. A summary of the notch toughness properties of the Turkey Point Unit No. 3 surveillance materials contained in Capsule S are listed in Table VII. These results indicate that the lower shell forging (123S266VA-1) is slightly more susceptible to radiation embrittlement than the intermediate shell forging (123P461VA-1). This correlates with the reported copper contents of 0.079% and 0.058%, respectively.

The results of tensile tests on specimens representing the lower and intermediate shell forging materials contained in Capsule S are listed in Table A-2 in Appendix A. The stress-strain curves and tensile test data

TABLE VII

NOTCH TOUGHNESS PROPERTIES OF CAPSULE S SPECIMENS
TURKEY POINT UNIT NO. 3

	Forging 123S266VA-1	Forging 123P461VA-1	Correlation Monitor
50 ft-lb Cv Temp. (deg F)			
Irradiated, 1.41×10^{19} (a)	4	-6	204
Unirradiated	-41	-29	65
ΔT	45	23	139
35 mil Cv Temp. (deg F)			
Irradiated, 1.41×10^{19} (a)	-1	-10	182
Unirradiated	-53	-45	41
ΔT	52	35	141
Upper Shelf Energy (ft-lbs)			
Unirradiated	154	145	76
Irradiated, 1.41×10^{19} (a)	122	128	60
ΔE	32	17	16

(a) Neutron fluence, cm^{-2}, E > 1 MeV

sheets are also reproduced in Appendix A. The tensile strength and ductility data obtained on the forging materials are compared to the unirradiated properties[16] in Figures 5 and 6. These data also indicate that the higher copper forging material is slightly more sensitive to neutron radiation embrittlement than the lower copper forging material.

D. Check Chemical Analyses for Copper

Check chemical analyses for the copper content of four tested Charpy V-notch specimens gave the following results:

Specimen No.	Material Identification	Copper (%)
S 8	123S266VA-1	0.06
P 9	123P461VA-1	0.06
R 5	Correlation Monitor	0.16
R 6	Correlation Monitor	0.18

The results on the vessel surveillance materials confirm the copper contents indicated by WCAP 8631[14] and WCAP 7656.[16]

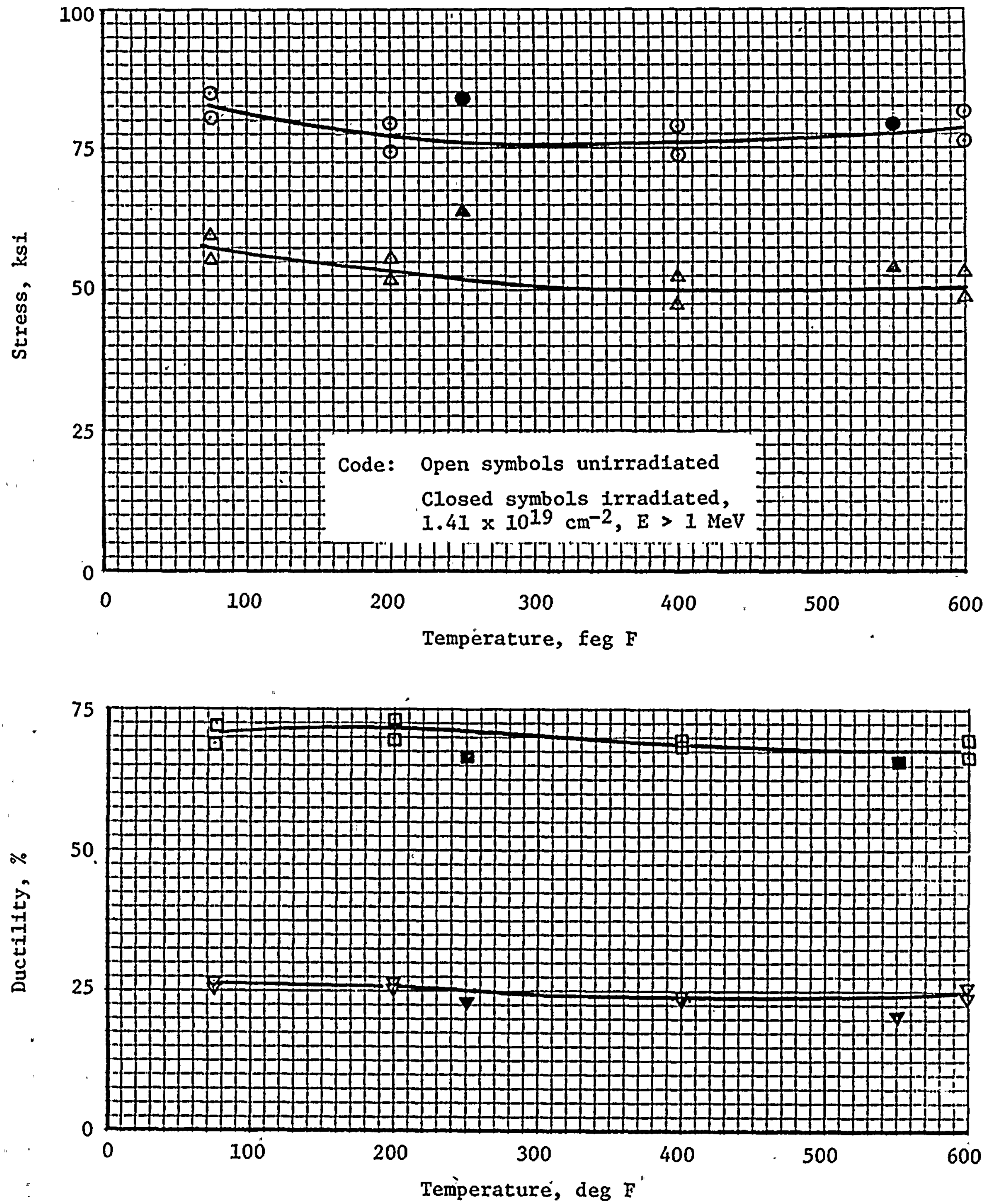
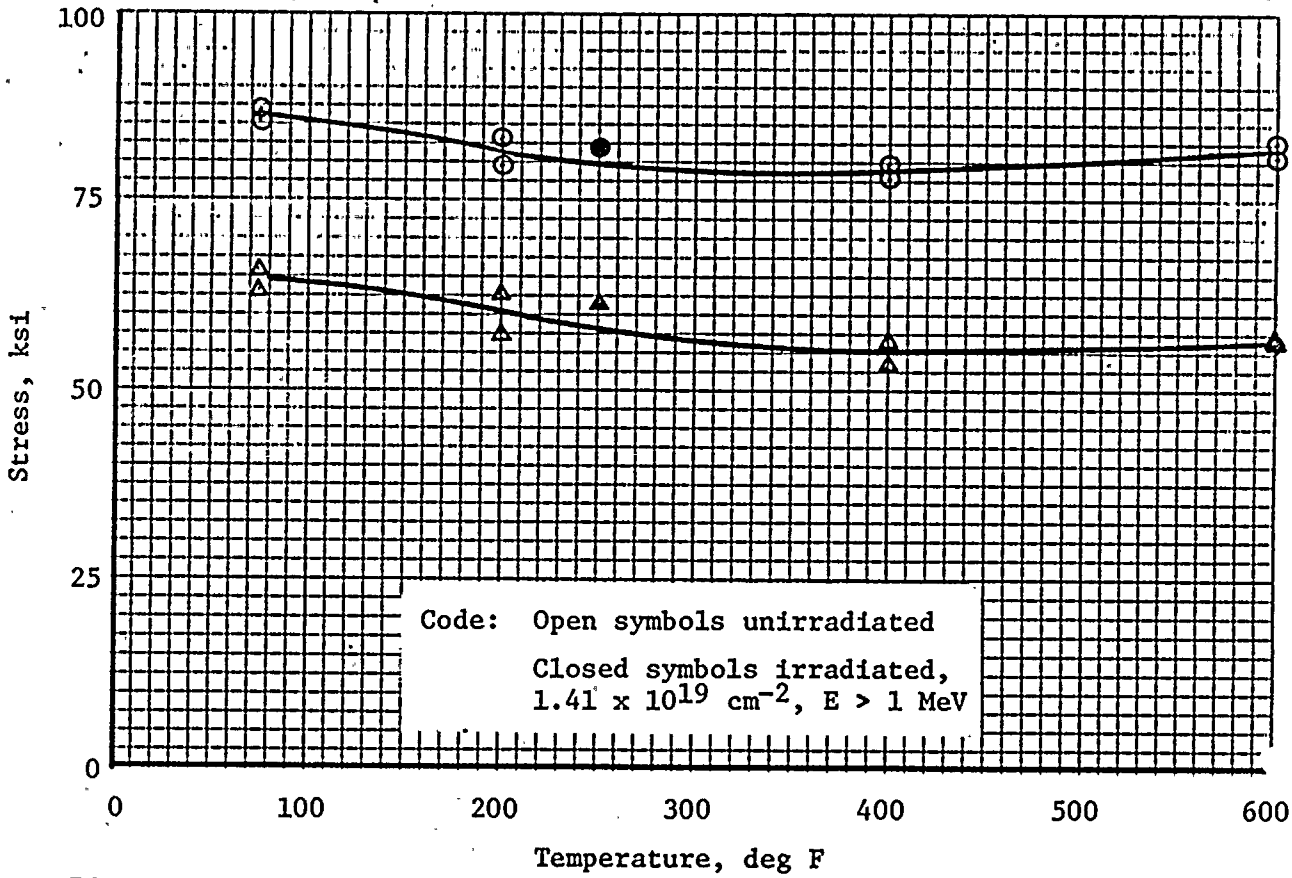


FIGURE 5. TENSILE PROPERTIES OF FORGING 123S266VA-1
TURKEY POINT UNIT NO. 3



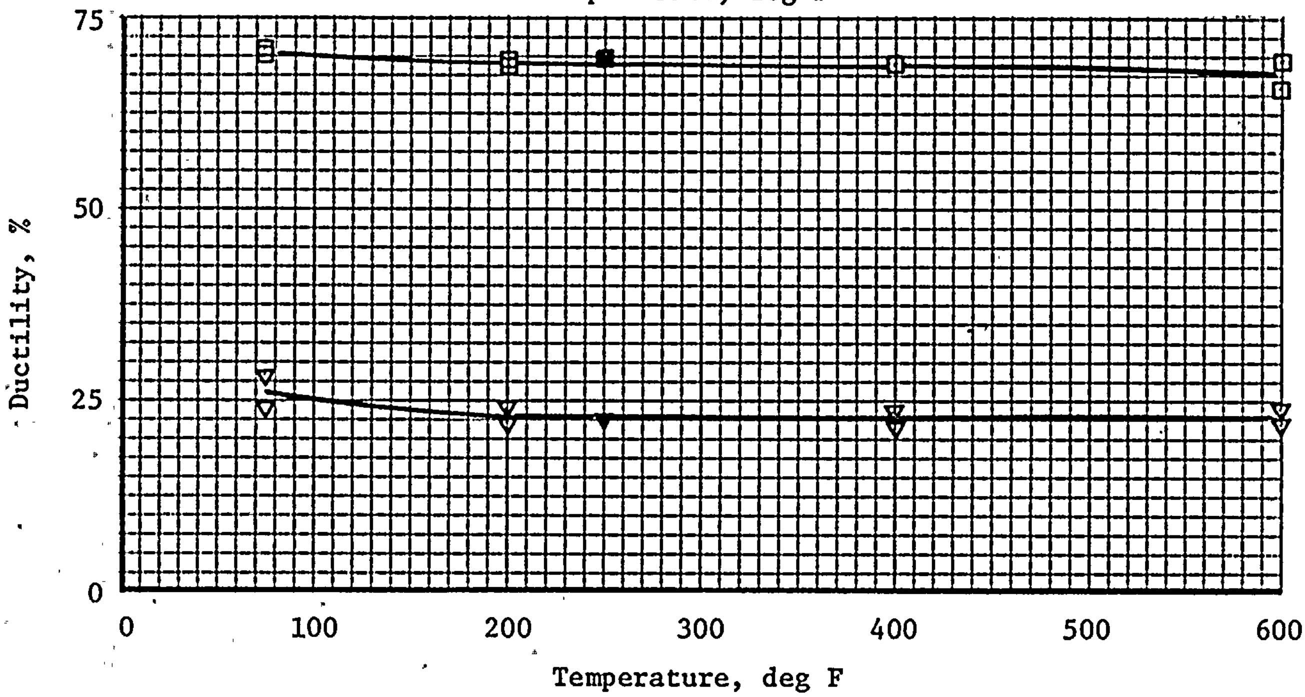


FIGURE 6. TENSILE PROPERTIES OF FORGING 123P461VA-1
TURKEY POINT UNIT NO. 3

VI. CAPSULE S TEST RESULTS - TURKEY POINT UNIT NO. 4

A. Neutron Dosimetry

The same anomalies concerning neutron dosimeter identification described for Unit No. 3 were noted for the Unit No. 4 Capsule S dosimeters, but these materials were not subjected to qualitative analysis identification check. The specific activities obtained from the remaining dosimeters in this capsule are given in Table VIII. The weights of the "middle cadmium-covered" and the "nickel" wires support the supposition that these dosimeters were made of a copper alloy and an aluminum-cobalt alloy, respectively. The iron results again show a 20% decrease in neutron flux across the capsule in the radial direction, and it appears that the capsule was placed in the vessel in the planned orientation. The saturated activities of the dosimeters, also given in Table VIII, were based on the summary of plant operations given in Table IX.

The neutron spectrum calculated for the Capsule S location is given in Table X along with the spectrum-averaged cross sections computed for the iron and copper dosimeters. The resulting mean value of fast neutron flux density at the Unit No. 4 Capsule S location was 1.16×10^{11} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. Since Unit No. 4 operated for an equivalent 1249.13 full power days of operation up to the 1978 refuelling outage, the calculated value of neutron fluence received by this capsule is 1.25×10^{19} cm^{-2}, E > 1 MeV.

To provide information for reference only, the Unit No. 4 Capsule S fast neutron flux density based on fission-spectrum cross sections(18) is calculated to be 1.25×10^{11} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. The fast neutron flux density at the Capsule S location computed with the DOT 3.5 code was 1.18×10^{11} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV.

TABLE VIII

SUMMARY OF NEUTRON DOSIMETRY RESULTS
CAPSULE S
TURKEY POINT UNIT NO. 4

Monitor Identification[a]	Radial Location in Capsule[15]	Activation Reaction	Dosimeter Weight (mg)	Measured Activity (dps/mg)	Saturated Activity (dps/mg)
Fe-P9 (Top)	Core Side	$^{54}Fe(n,p)^{54}Mn$	245.2	5.29×10^3	7.37×10^3
Fe-R5 (Middle)			193.1	5.13×10^3	7.15×10^3
Fe-P1 (Bottom)			174.7	5.25×10^3	7.31×10^3
Fe-S9 (Top)	Vessel Side		188.7	4.58×10^3	6.39×10^3
Fe-R7 (Middle)			265.2	4.20×10^3	5.85×10^3
Fe-S1 (Bottom)			206.0	4.48×10^3	6.25×10^3
Cu (Top)	Center	$^{63}Cu(n,\alpha)^{60}Co$	49.8	2.30×10^2	6.91×10^2
Cu (Bottom)			50.3	2.23×10^2	6.72×10^2
Ni (Middle)		$^{58}Ni(n,p)^{58}Co$	9.2	(b)	(b)
Co (Top)		$^{59}Co(n,\gamma)^{60}Co$	8.9	2.50×10^7	7.54×10^7
Co-Cd (Top)			8.1	9.36×10^6	2.82×10^7
Co (Middle)			9.5	1.91×10^7	5.75×10^7
Co-Cd (Middle)			18.0	(c)	(c)
Co (Bottom)			8.3	2.26×10^7	6.80×10^7
Co-Cd (Bottom)			9.6	1.02×10^7	3.08×10^7

(a) See Table II for location within capsule.
(b) Monitor material was identified as being bare Co in Al.
(c) Monitor material was identified as being a Cu-base alloy (purity undetermined).

TABLE IX

SUMMARY OF PLANT OPERATIONS
TURKEY POINT UNIT NO. 4

Oper. Period	Date Start	Date Stop	Shutdown Days	Operating Days	Reactor Power Output (Mwdt)	Equivalent Oper. Days, T_m	Decay Time after Period, t_m
1	06/19/73	07/01/73	-	12	3,269	1.49	1869
	07/01/73	07/02/73	1	-	-		
2	07/02/73	07/07/73	-	5	2,596	1.18	1863
	07/07/73	07/09/73	2	-	-		
3	07/09/73	07/14/73	-	5	3,036	1.38	1856
	07/14/73	07/18/73	4	-	-		
4	07/18/73	07/29/73	-	11	7,056	3.21	1841
	07/29/73	08/05/73	7	-	-		
5	08/05/73	09/01/73	-	27	21,408	9.73	1807
	09/01/73	09/02/73	1	-	-		
6	09/02/73	09/23/73	-	21	28,405	12.91	1785
	09/23/73	09/24/73	1	-	-		
7	09/24/73	11/04/73	-	41	61,074	27.76	1743
	11/04/73	11/16/73	12	-	-		
8	11/16/73	01/03/74	-	48	72,903	33.14	1683
	01/03/74	02/03/74	31	-	-		
9	02/03/74	04/03/74	-	59	103,061	46.85	1593
	04/03/74	04/05/74	2	-	-		
10	04/05/74	04/17/74	-	12	22,345	10.16	1579
	04/17/74	04/18/74	1	-	-		
11	04/18/74	05/25/74	-	37	75,126	34.15	1541
	05/25/74	05/31/74	6	-	-		
12	05/31/74	08/18/74	-	79	159,505	72.50	1456
	08/18/74	09/10/74	23	-	-		
13	09/10/74	10/27/74	-	47	98,345	44.70	1386
	10/27/74	11/02/74	6	-	-		
14	11/02/74	12/04/74	-	32	58,541	26.61	1348
	12/04/74	12/07/74	3	-	-		
15	12/07/74	01/06/75	-	30	63,850	29.02	1315
	01/06/75	01/10/75	4	-	-		
16	01/10/75	03/30/75	-	79	168,707	76.68	1232
	03/30/75	06/21/75	83	-	-		
17	06/21/75	08/03/75	-	43	86,227	39.19	1106
	08/03/75	08/09/75	6	-	-		
18	03/09/75	09/21/75	-	43	90,287	41.04	1057
	09/21/75	10/01/75	10	-	-		
19	10/01/75	10/12/75	-	11	22,450	10.20	1036
	10/12/75	10/13/75	1	-	-		
20	10/12/75	01/10/76	-	89	190,599	86.64	946
	01/10/76	01/17/76	7	-	-		
21	01/17/76	04/18/76	-	92	193,789	88.09	847
	04/18/76	06/10/76	53	-	-		
22	06/10/76	06/12/76	-	2	2,289	1.04	792
	06/12/76	06/16/76	4	-	-		
23	06/16/76	06/17/76	-	1	480	0.22	787
	06/17/76	06/19/76	2	-	-		
24	06/19/76	09/10/76	-	83	174,409	79.28	702
	09/10/76	09/16/76	6	-	-		
25	09/16/76	09/24/76	-	8	14,159	6.44	688
	09/24/76	09/29/76	5	-	-		
26	09/29/76	10/10/76	-	11	23,551	10.70	672
	10/10/76	10/14/76	4	-	-		
27	10/14/76	10/28/76	-	14	27,692	12.59	654
	10/28/76	12/03/76	36	-	-		
28	12/03/76	01/06/77	-	34	71,425	32.47	584
	01/06/77	01/11/77	5	-	-		
29	01/11/77	01/25/77	-	14	30,659	13.94	565
	01/25/77	01/30/77	5	-	-		
30	01/30/77	03/20/77	-	49	107,197	48.73	511
	03/20/77	03/26/77	6	-	-		
31	03/26/77	04/26/77	-	31	66,448	30.20	474
	04/26/77	05/04/77	8	-	-		
32	05/04/77	05/09/77	-	5	10,825	4.92	461
	05/09/77	08/03/77	86	-	-		
33	08/03/77	08/11/77	-	8	12,603	5.73	367
	08/11/77	08/15/77	4	-	-		
34	08/15/77	10/29/77	-	75	157,049	71.39	238
	10/29/77	11/11/77	13	-	-		
35	11/11/77	02/14/78	-	95	202,068	91.85	180
	02/14/78	03/09/78	23	-	-		
36	03/09/78	08/13/78	-	157	314,604	143.00	0
				Total	2,748,037	1249.13	

TABLE X

FAST NEUTRON SPECTRUM AND FOIL ACTIVATION CROSS SECTIONS FOR CAPSULE S TURKEY POINT UNIT NO. 4

Energy Range (MeV)	DOT 3.5 Calculated Neutron Flux	$^{54}Fe(n,p)^{54}Mn$ Cross Section (barns)	$^{63}Cu(n,\gamma)^{60}Co$ Cross Section (barns)
10.00 - 12.5	3.07×10^{8}	0.521	4.78×10^{-2}
8.18 - 10.0	9.71×10^{8}	0.578	4.26×10^{-2}
6.36 - 8.18	2.60×10^{9}	0.578	1.53×10^{-2}
4.96 - 6.36	5.03×10^{9}	0.491	2.85×10^{-3}
4.06 - 4.96	5.07×10^{9}	0.352	2.78×10^{-4}
3.01 - 4.06	9.35×10^{9}	0.222	1.12×10^{-4}
2.35 - 3.01	1.55×10^{10}	0.098	5.96×10^{-5}
1.83 - 2.35	1.90×10^{10}	0.024	3.86×10^{-5}
1.11 - 1.83	5.12×10^{10}	0.0025	2.09×10^{-5}
1.00 - 1.11	0.94×10^{10}	-	1.20×10^{-5}
0.55 - 1.00	7.14×10^{10}	-	5.48×10^{-6}
0.11 - 0.55	1.26×10^{11}	-	8.46×10^{-7}

$\bar{\sigma}_{Fe}$ = 0.0900 barns (E > 1 MeV)

$\bar{\sigma}_{Cu}$ = 0.000980 barns (E > 1 MeV)

B. Thermal Monitors

The thermal monitors were examined and had not melted. This indicates that the capsule did not reach 579 F during the exposure period.

C. Mechanical Property Test Results

The Charpy V-notch impact results obtained from specimens contained in Capsule S are given in Table B-1 in Appendix B. The transition curves developed for each material using a tanh curve-fitting technique are also presented in Appendix B. A summary of the notch toughness properties of the Turkey Point Unit No. 4 surveillance materials contained in Capsule S are listed in Table XI. These results indicate that forging heat 123P481VA-1 has more sensitivity to neutron radiation embrittlement than forging heat 122S180VA-1, even though their copper contents are almost identical (see Table III).

The results of tensile tests on specimens representing the lower and intermediate shell forging materials contained in Capsule S are listed in Table B-2 in Appendix B. Also included in this appendix are the stress-strain curves and tensile test data sheets. The tensile strength and ductility data obtained on these forging materials are compared to the unirradiated properties[17] in Figures 7 and 8. These results indicate that both forging heats have about the same low irradiation sensitivity as would be expected from the copper contents of these materials.

D. Check Chemical Analyses for Copper

Check chemical analyses for the copper content of three tested Charpy V-notch specimens, made with an x-ray fluorescence technique, gave the following results:

TABLE XI

NOTCH TOUGHNESS PROPERTIES OF CAPSULE S SPECIMENS
TURKEY POINT UNIT NO. 4

	Forging 122S180VA-1	Forging 123P481VA-1	Correlation Monitor
50 ft-lb C_V Temp. (deg F)			
Irradiated, 1.25×10^{19} (a)	3	60	195
Unirradiated	-8	25	80
ΔT	11	35	115
35 mil C_V Temp. (deg F)			
Irradiated, 1.25×10^{19} (a)	-4	46	174
Unirradiated	-15	-2	62
ΔT	11	48	112
Upper Shelf Energy (ft-lb)			
Unirradiated	132	135	122
Irradiated, 1.25×10^{19} (a)	122	123	88
ΔE	10	12	34

(a) Neutron Fluence, cm^{-2}, E > 1 MeV

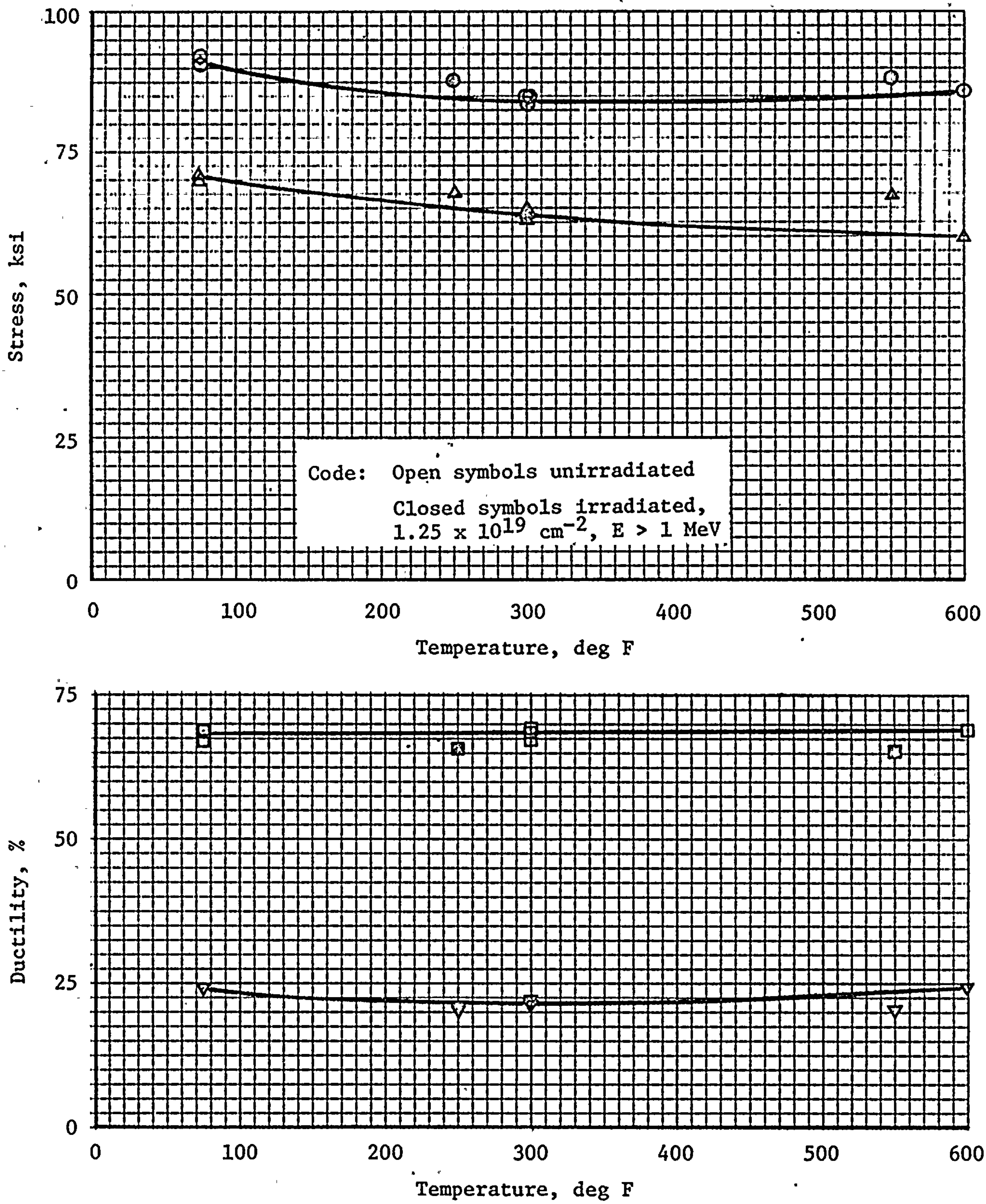


FIGURE 7. TENSILE PROPERTIES OF FORGING 122S180VA-1
TURKEY POINT UNIT NO. 4

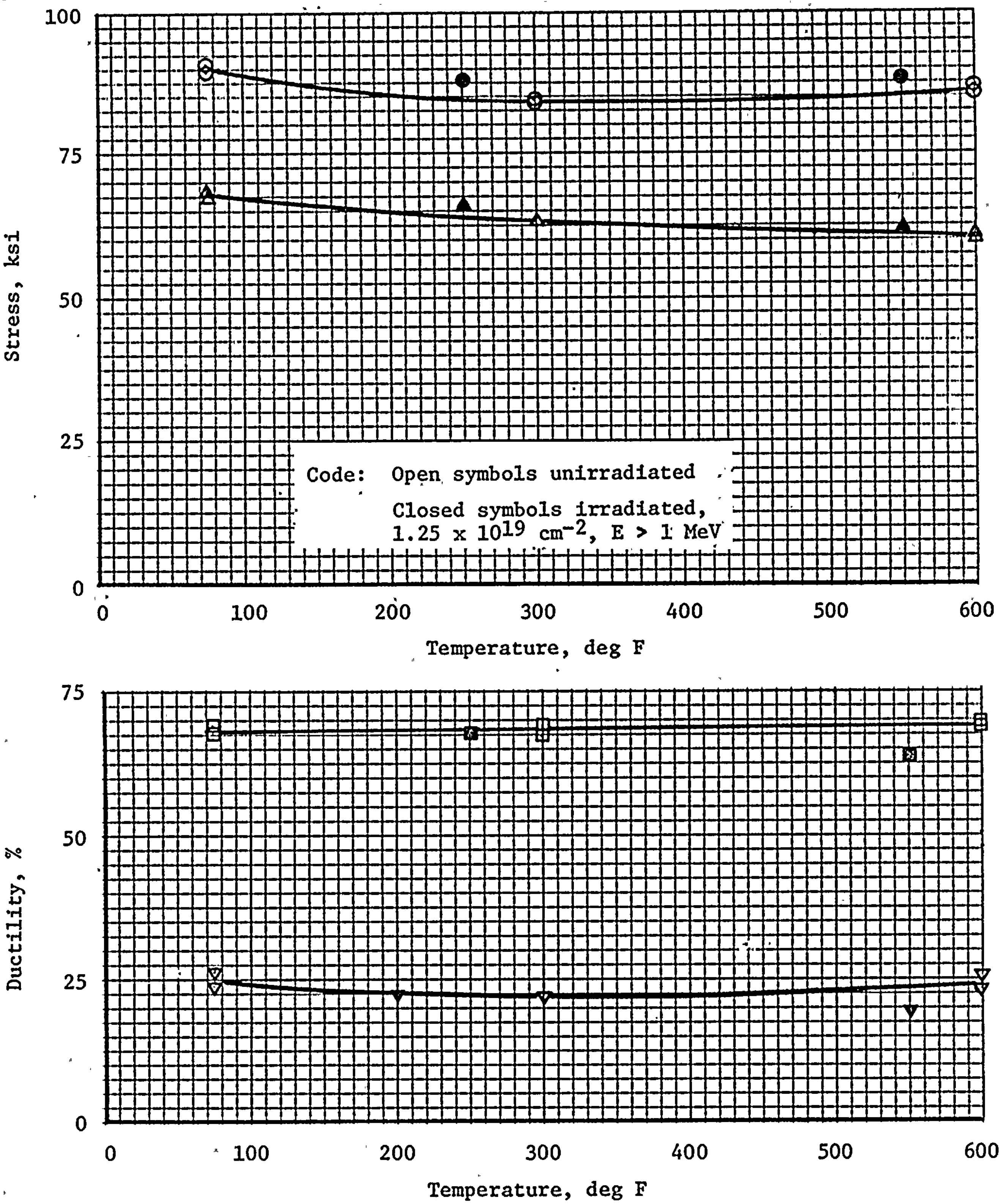


FIGURE 8. TENSILE PROPERTIES OF FORGING 123P481VA-1
TURKEY POINT UNIT NO. 4

Specimen No.	Material Identification	Copper (%)
S-1	122S180VA-1	nil
P-1	123P481VA-1	.02
R-1	Correlation Monitor	.08

These results are below those reported in WCAP 7660.[17] The background radiation resulting from the gamma activity of each irradiated specimen was nearly twice that observed for the Unit No. 3 chemical analysis samples. As a result, the background count was a much larger fraction of the total count in the copper peak, reducing the accuracy of the result.

VII. ANALYSIS OF RESULTS

A. Introduction

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 (trend curves).

(2) Develop heatup and cooldown curves to describe the operational limitations for selected periods of time. 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 (trend curves) and the energy and spatial distribution of the neutron flux.

The capsules removed from the Turkey Point Nuclear Power Plant pressure vessels during the 1977-78 refuelling outages contained specimens representing the intermediate and lower shell course beltline forging materials but did not contain any weld metal or HAZ specimens. Since the weld metal will control the RT_{NDT} for both units[14,15], the results of this analysis may not affect the current heatup and cooldown limits.

It is anticipated that the reliability of neutron embrittlement trend curves will be improved as more surveillance data become available and a better understanding of the factors affecting radiation embrittlement has been achieved. As an example of the latter, Mr. E. C. Biemiller of Combustion Engineering, in a paper[19] given at the 8th ASTM International Symposium on Effects of Radiation on Structural Materials held in St. Louis

in May 1976, indicated that a parameter of (% Ni + % Si) ÷ (% Mo + % Cr + % Mn) may explain the variation in radiation embrittlement observed in ferritic materials of nominally the same copper content. In addition, at the 9th ASTM International Symposium on Effects of Radiation on Structural Materials held in Richland, Washington, in July 1978, Mr. J. D. Varsik of Combustion Engineering presented a related paper entitled "An Empirical Evaluation of the Irradiation Sensitivity of Reactor Vessel Materials." At the same conference, Westinghouse presented information which indicates that neutron embrittlement may reach a limiting value when the irradiation is carried out for long times at approximately 550°F in lower neutron flux environments. Also, the Metal properties Council is developing new radiation damage curves that will be based on more data than those currently in use.

B. Pressure Vessel Fast Neutron Exposure

1. Turkey Point Unit No. 3

Based on the dosimetry results obtained from Capsule S, and using the conservative lead factor of 1.76 calculated for this capsule, the maximum fast flux incident on the Turkey Point Unit No. 3 pressure vessel is calculated to be 7.33×10^{10} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. The fast neutron flux is attenuated as it penetrates the pressure vessel wall. Conservative estimates of the ratio of fast flux at depths of 2 in. (1/4T) and 6 in. (3/4T) to that incident on the pressure vessel I.D. surface are 0.60 and 0.15, respectively.[20]

Utilizing these factors, the maximum fast flux at the 1/4T depth in the Turkey Point Unit No. 3 pressure vessel wall is estimated to be 4.40 x

10^{10} $cm^{-2} \cdot sec^{-1}$, and that at the 3/4T depth is estimated to be 1.10 x 10^{10} $cm^{-2} \cdot sec^{-1}$, approximately 12% higher than determined from the analysis of Capsule T.[14] The predicted neutron exposures for the Turkey Point Unit No. 3 pressure vessel at the I.D. surface, 1/4T and 3/4T positions after 5, 10, and 32 Effective Full Power Years (EFPY) of operation are summarized in Table XII.

2. Turkey Point Unit No. 4

Using the Capsule S dosimetry results, and the conservative lead factor of 1.76 calculated for this capsule, the maximum fast flux incident on the Turkey Point Unit No. 4 pressure vessel is calculated to be 6.59 x 10^{10} $cm^{-2} \cdot sec^{-1}$, E > 1 MeV. The maximum fast flux values calculated for the 1/4T and 3/4T positions within the vessel wall are 3.95 x 10^{10} and 0.99 x 10^{10} $cm^{-2} \cdot sec^{-1}$, respectively. These values are within 1% of those determined from the analysis of Capsule T.[15] The predicted neutron exposures for the Turkey Point Unit No. 4 pressure vessel at the I.D. surface, 1/4T and 3/4T positions after 5, 10, and 32 EFPY are presented in Table XIII.

C. Vessel Material Notch Toughness

A method for estimating the reduction in C_v upper shelf energy as a function of neutron fluence is given in Regulatory Guide 1.99.[7] The results obtained to date on the vessel beltline forging materials and the reference steels contained in Capsules S and T are compared to a portion of Figure 2 of Regulatory Guide 1.99 in Figure 9. The shelf energy response of each vessel beltline forging material from Turkey Point Unit Nos. 3 and 4 was equal to or less than the minimum base metal response curve (0.10% Cu) given in Figure 2 of Regulatory Guide 1.99. The shelf

TABLE XII

PROJECTED MAXIMUM PRESSURE VESSEL EXPOSURES[a]
TURKEY POINT UNIT NO. 3

Location in Vessel Wall	Neutron Flux E > 1 MeV	Neutron Fluence, E > 1 MeV		
		5 EFPY	10 EFPY	32 EFPY
I.D. Surface	7.33×10^{10} cm$^{-2}\cdot$sec^{-1}	1.16×10^{19} cm^{-2}	2.3×10^{19} cm^{-2}	7.4×10^{19} cm^{-2}
1/4T Depth	4.40×10^{10} cm$^{-2}\cdot$sec^{-1}	6.94×10^{18} cm^{-2}	1.4×10^{19} cm^{-2}	4.4×10^{19} cm^{-2}
3/4T Depth	1.10×10^{10} cm$^{-2}\cdot$sec^{-1}	1.73×10^{18} cm^{-2}	3.5×10^{18} cm^{-2}	1.1×10^{19} cm^{-2}

(a) Based on results from Capsule S

TABLE XIII

PROJECTED MAXIMUM PRESSURE VESSEL EXPOSURES[a]
TURKEY POINT UNIT NO. 4

Location in Vessel Wall	Neutron Flux E > 1 MeV	Neutron Fluence, E > 1 MeV		
		5 EFPY	10 EFPY	32 EFPY
I.D. Surface	6.59×10^{10} $cm^{-2} \cdot sec^{-1}$	1.04×10^{19} cm^{-2}	2.1×10^{19} cm^{-2}	6.7×10^{19} cm^{-2}
1/4T Depth	3.95×10^{10} $cm^{-2} \cdot sec^{-1}$	6.24×10^{18} cm^{-2}	1.2×10^{19} cm^{-2}	4.0×10^{19} cm^{-2}
3/4T Depth	0.99×10^{10} $cm^{-2} \cdot sec^{-1}$	1.56×10^{18} cm^{-2}	3.1×10^{18} cm^{-2}	1.0×10^{19} cm^{-2}

(a) Based on results from Capsule S

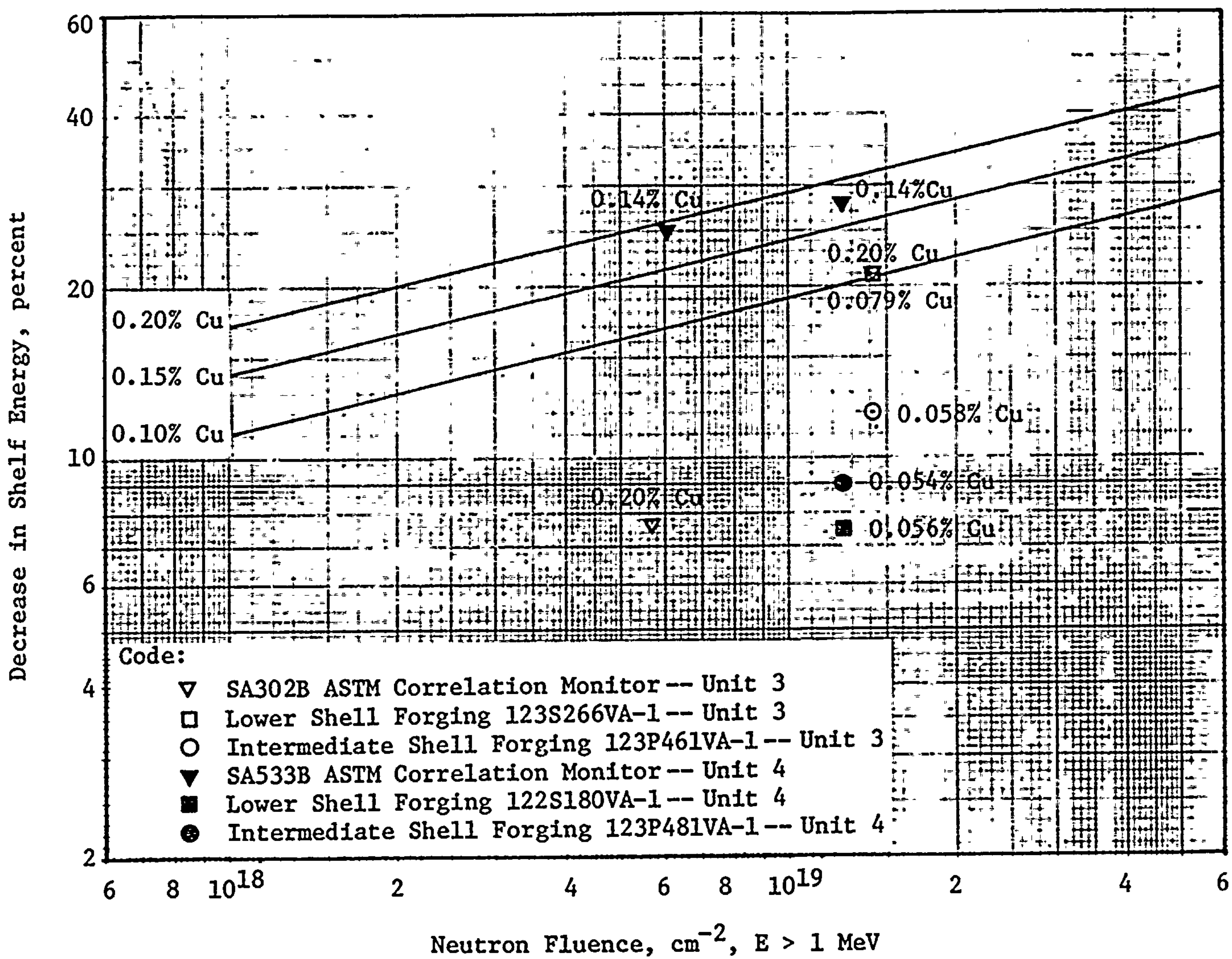


FIGURE 9. COMPARISON OF DECREASE IN SHELF ENERGIES OF TURKEY POINT UNIT NOS. 3 AND 4 VESSEL FORGING MATERIALS AND SURVEILLANCE REFERENCE STEELS TO REGULATORY GUIDE 1.99 TREND CURVES

energy response of the 0.20% Cu A302B reference steel (Unit 3) was also less than the appropriate (0.20% Cu) trend curve, but the shelf energy response of the 0.14% Cu A533 reference material (Unit 4) was above the applicable (0.15% Cu) trend curve.

D. Adjusted Reference Temperature

A similar approach can be taken to estimate the increase in RT_{NDT} as a function of fast neutron fluence. Figure 10, which compares the Turkey Point Unit Nos. 3 and 4 vessel forging material and reference steel results to the appropriate radiation damage trend curves developed by Westinghouse(15), indicates that the responses of the forging materials are well below the 0.10% Cu trend curve and the responses of the reference steels are in good agreement with the appropriate trend curves.

The same data are compared to the trend curves of Regulatory Guide 1.99(7) in Figure 11. This shows that the responses of the vessel forging materials are below the 0.08% Cu trend curve, the response of the A533B reference steel is in good agreement with the 0.14% Cu trend curve, and the response of the A302B reference steel is below the 0.20% Cu trend curve.

Although there is considerable scatter in the data, the transition temperature shifts determined for the vessel surveillance materials are in reasonable agreement with both sets of trend curves. However, the vessel forging material data appear to follow the slope of the Regulatory Guide 1.99 trend curves (Figure 11) and the correlation monitor material data appear to follow the slope of the Westinghouse trend curves (Figure 10).

E. Heatup and Cooldown Limits for Normal Operation

Heatup and cooldown limit curves were developed for 0-5 and 5-10 EFPY of operation for the Turkey Point Unit Nos. 3 and 4 nuclear power plants after the removal of the first surveillance capsule (T) from each

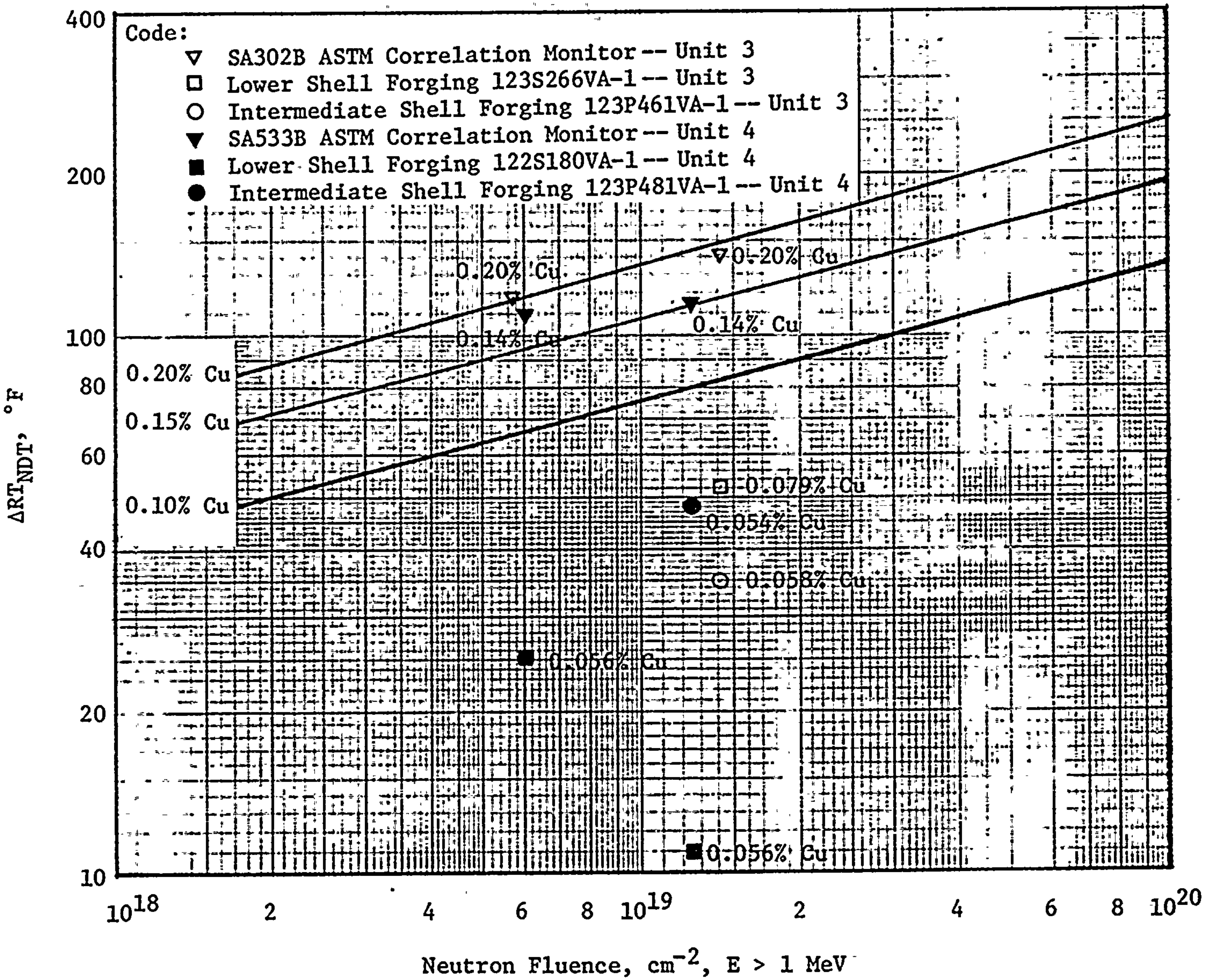


FIGURE 10. COMPARISON OF INCREASE IN REFERENCE TEMPERATURES OF TURKEY POINT UNIT NOS. 3 AND 4 VESSEL FORGING MATERIALS AND SURVEILLANCE REFERENCE STEELS TO WESTINGHOUSE TREND CURVES(14)

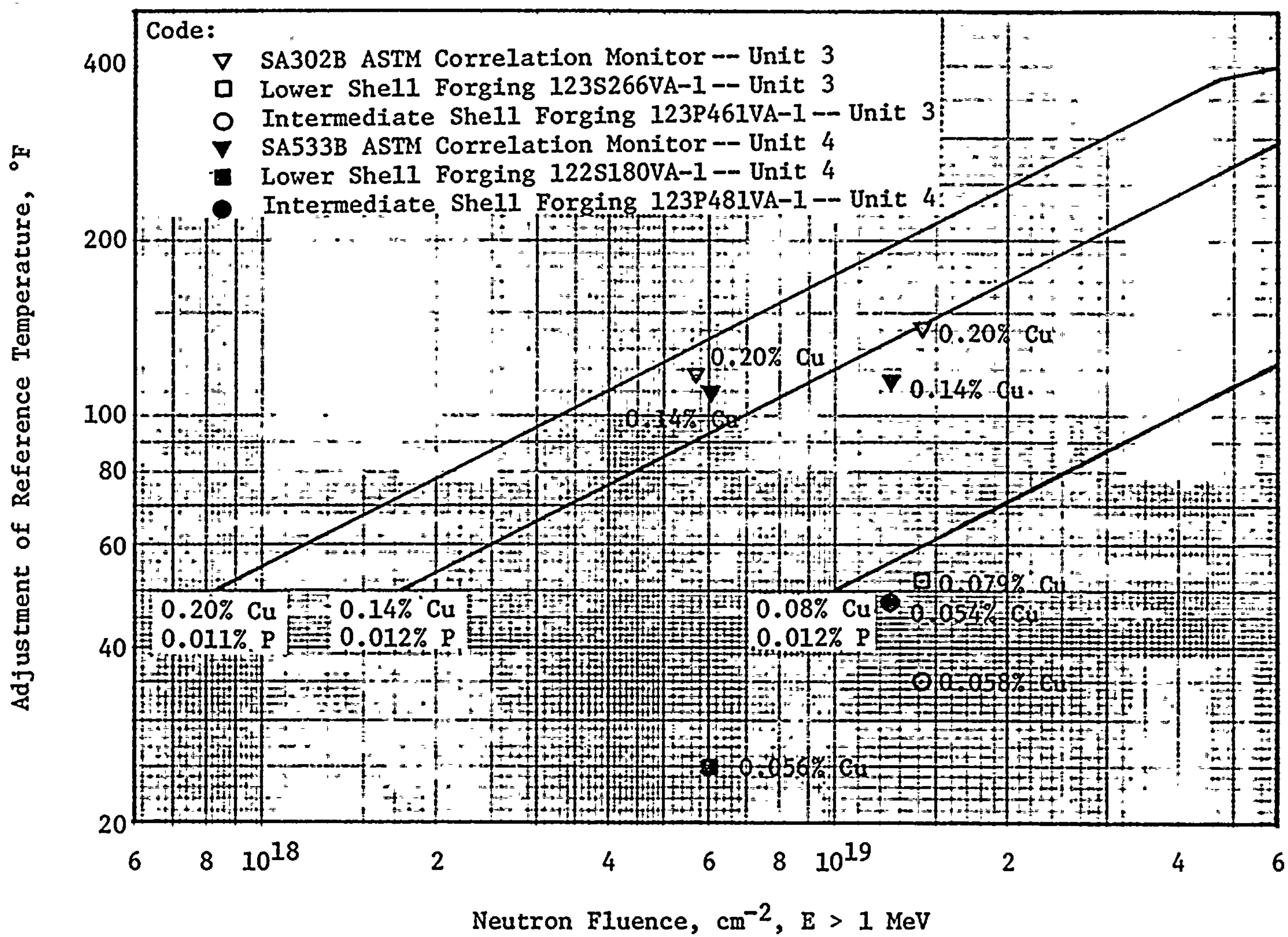


FIGURE 11. COMPARISON OF INCREASE IN REFERENCE TEMPERATURES OF TURKEY POINT UNIT NOS. 3 AND 4 VESSEL FORGING MATERIALS AND SURVEILLANCE REFERENCE STEELS TO REGULATORY GUIDE 1.99 TREND CURVES

reactor vessel.[21] The projected fast neutron exposures resulting from the analyses of the second surveillance capsule (S) from each unit are in good agreement with those reported earlier.[14,15] Also, since the S capsules did not contain specimens representing the controlling (weld metal) beltline material, there is no basis for revising the projected values of RT_{NDT} used to develop the current set of heatup and cooldown limit curves.

F. Capsule Removal Schedule

A third capsule is scheduled for removal from each reactor vessel after 10 calendar years of operation. Based on the past operating histories of the Turkey Point nuclear power plants, 10 calendar years of operation should correspond to approximately 7 EFPY of operation. It is recommended that Capsule V, a Type II capsule containing weld metal specimens, be removed from each vessel at that time. The projected fast neutron fluence for the V capsules after 7 EFPY is 1.3×10^{19} cm^{-2} ($E > 1$ MeV), approximately twice the fluence received by the T capsules.[14,15] The data obtained from the V capsules should provide the information necessary to revise the heatup and cooldown limitations for operation beyond 10 EFPY of operation.

VII. REFERENCES

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APPENDIX A

Charpy V-Notch and Tensile Test Data
Turkey Point Unit No. 3
Capsule S

TABLE A-1

CHARPY IMPACT DATA - CAPSULE S
TURKEY POINT UNIT NO. 3
(Neutron Fluence = 1.41×10^{19} cm^{-2}, E > 1 MeV)

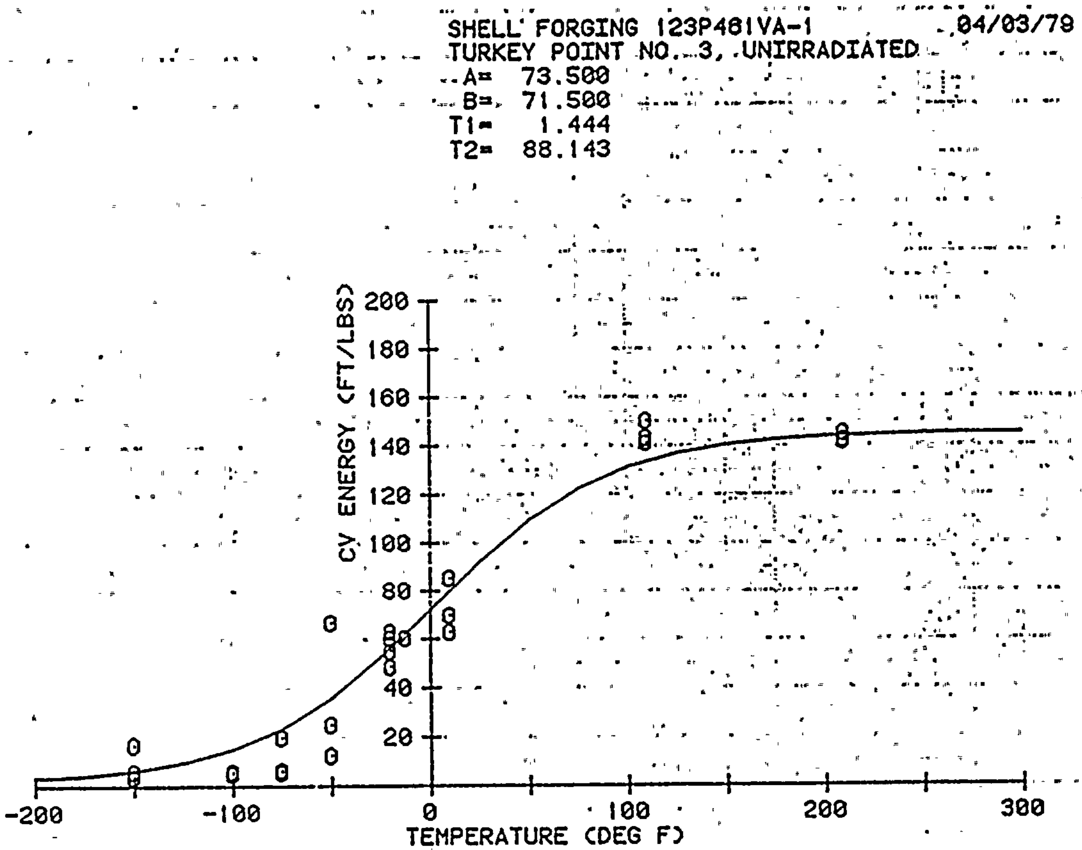
Material	Spec. No.	Temp. (deg F)	Energy (ft-lb)	Lat. Exp. (mils)	Fract. App. (% Shear)
Forging 123P461VA-1	P-9	-50	3.5	5	nil
	P-7	-20	33.0	24	5
	P-10	-10	70.0	49	75
	P-8	0	72.0	53	50
	P-6	10	59.5	45	80
	P-5	30	79.5	58	90
	P-1	72	77.0	59	10
	P-2	140	127.5	92	100
	P-3	210	132.0	89	100
	P-4	300	125.5	94	100
Forging 123S266VA-1	S-9	-50	13.5	9	nil
	S-7	-20	31.0	27	5
	S-8	0	46.0	37	5
	S-6	10	48.0	34	5
	S-10	20	67.0	51	75
	S-5	30	86.0	74	100
	S-1	72	117.5	81	50
	S-2	140	121.5	90	100
	S-3	210	124.0	95	100
	S-4	300	122.0	91	100
Correlation Monitor Steel	R-1	72	15.0	13	5
	R-2	140	21.0	19	15
	R-5	160	49.5	40	40
	R-6	180	33.0	31	50
	R-8	200	51.5	50	95
	R-3	210	53.0	47	90
	R-7	250	35.5	34	60
	R-4	300	60.0	51	100

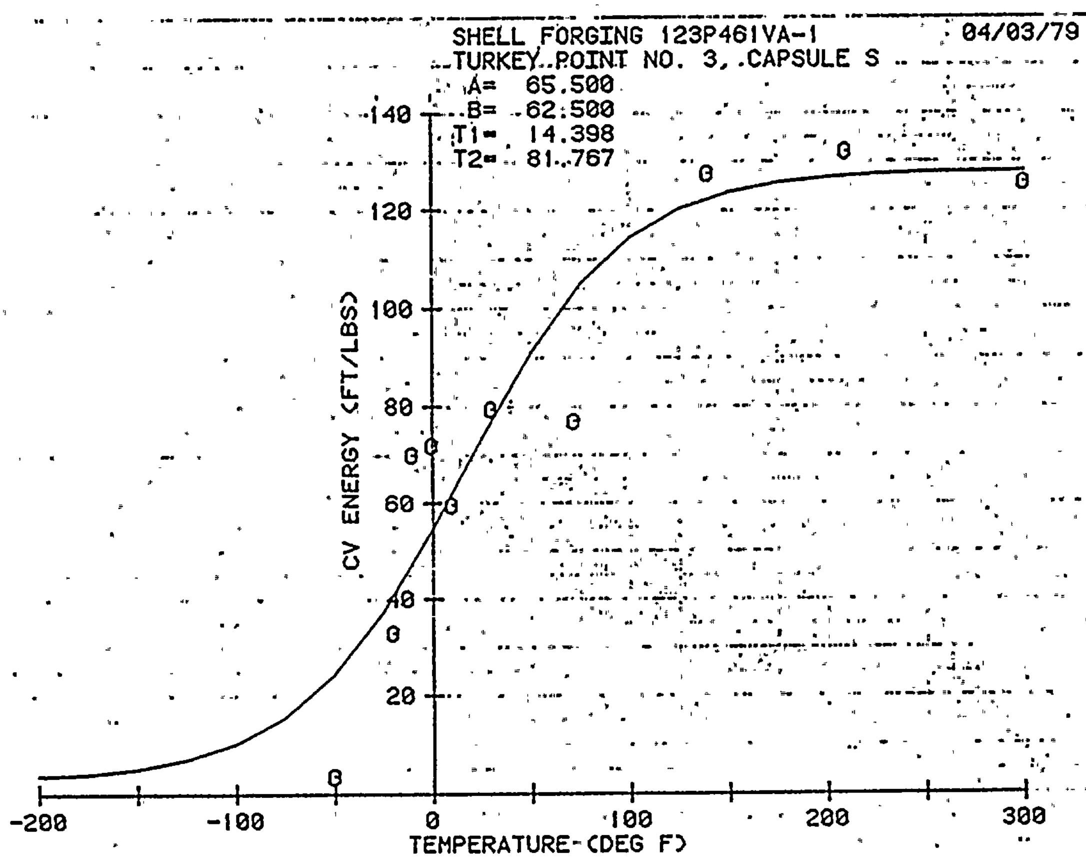
TABLE A-2

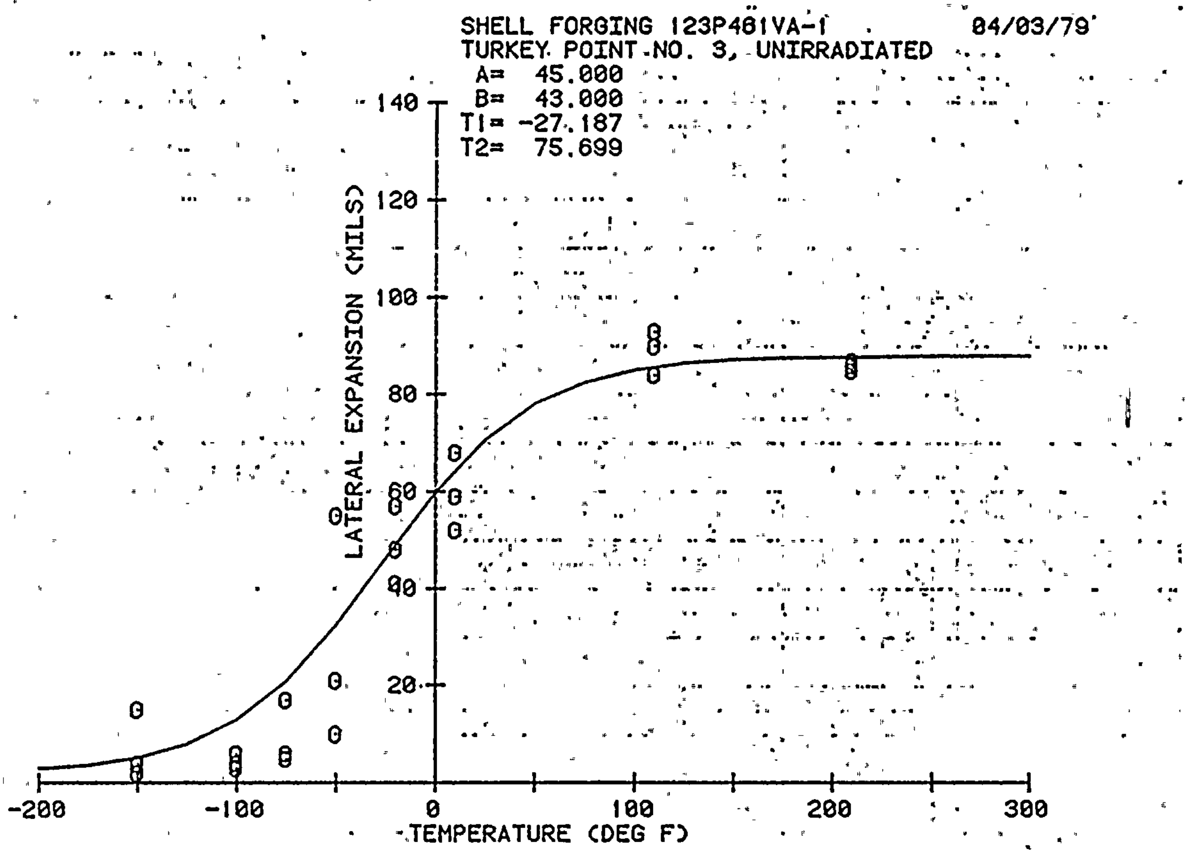
TENSILE PROPERTIES OF SURVEILLANCE MATERIALS
CAPSULE S - TURKEY POINT UNIT NO. 3
(Neutron Fluence = 1.41 x 10^{19} cm^{-2}, E > 1 MeV)

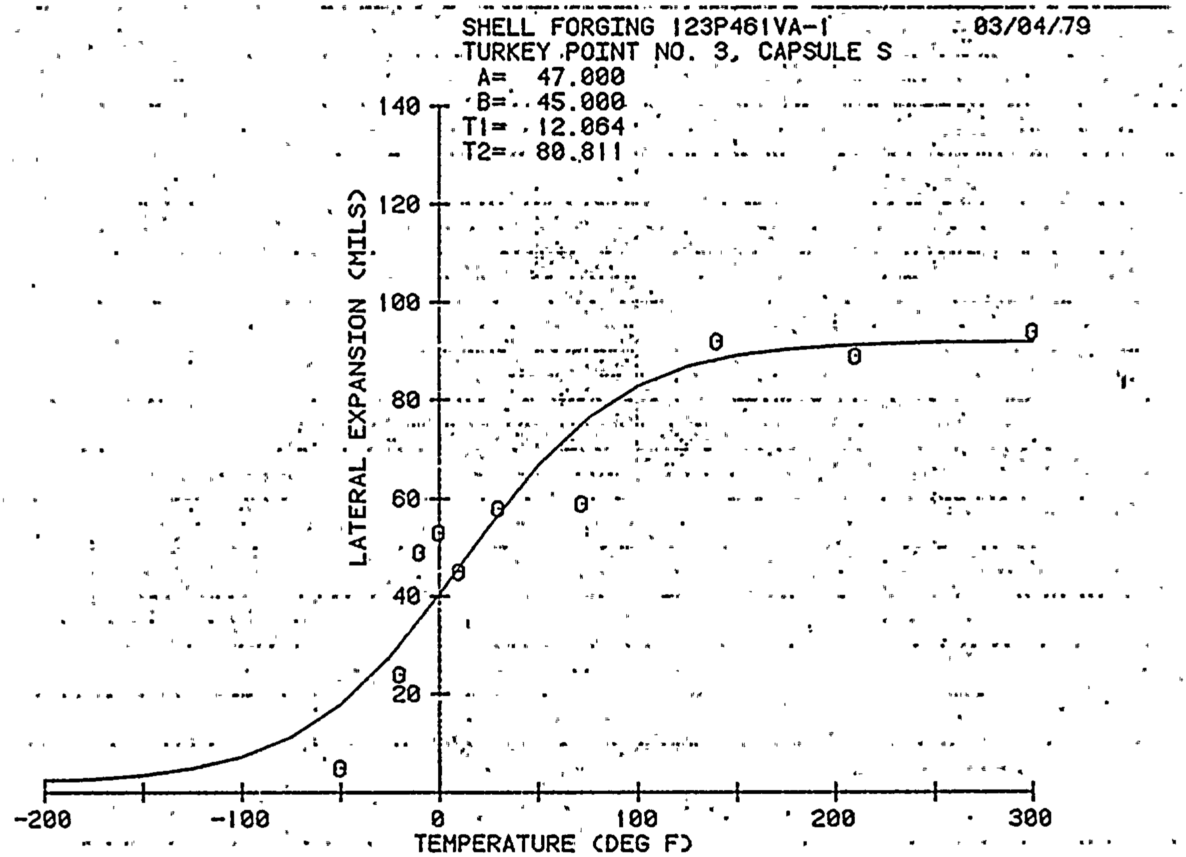
Spec. No.[a]	Test Temp. (°F)	0.2% Yield Strength (psi)	Tensile Strength (psi)	Fracture Strength (psi)	Fracture Stress (psi)	Uniform Elongation[b] (%)	Total Elongation (%)	Reduction in Area (%)
S-2	250	64,200	84,000	52,100	157,100	10.3	23.4	66.8
S-1	550	54,400	78,900	50,900	149,700	7.6	20.9	66.0
P-2	250	61,300	81,200	50,900	167,100	10.2	22.5	69.5
P-1	550	(c)	(c)	(c)	(c)	(c)	(c)	(c)

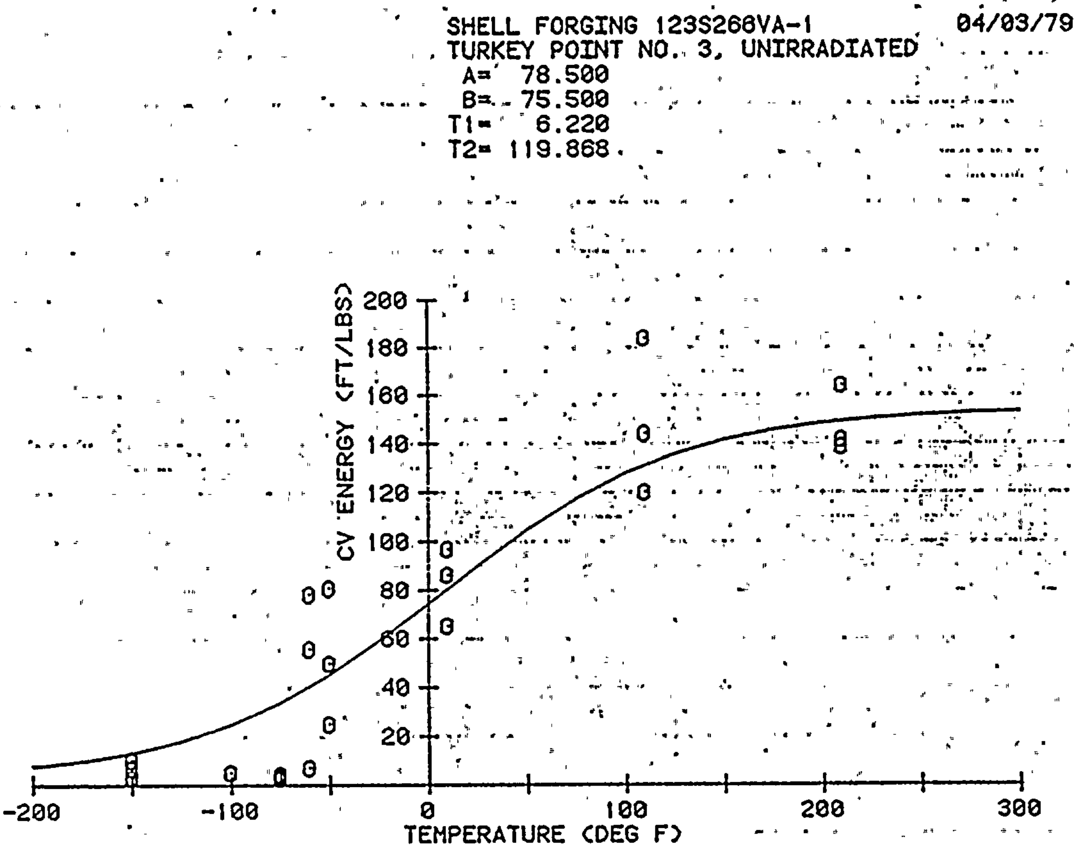
(a) Material Code: S = 123S266VA-1; P = 123P461VA-1
(b) Using change of cross-sectional area in unnecked portion of specimen per ASTM E 184-62.
(c) Specimen not tested. Temperature controller malfunction caused overheating of specimen.

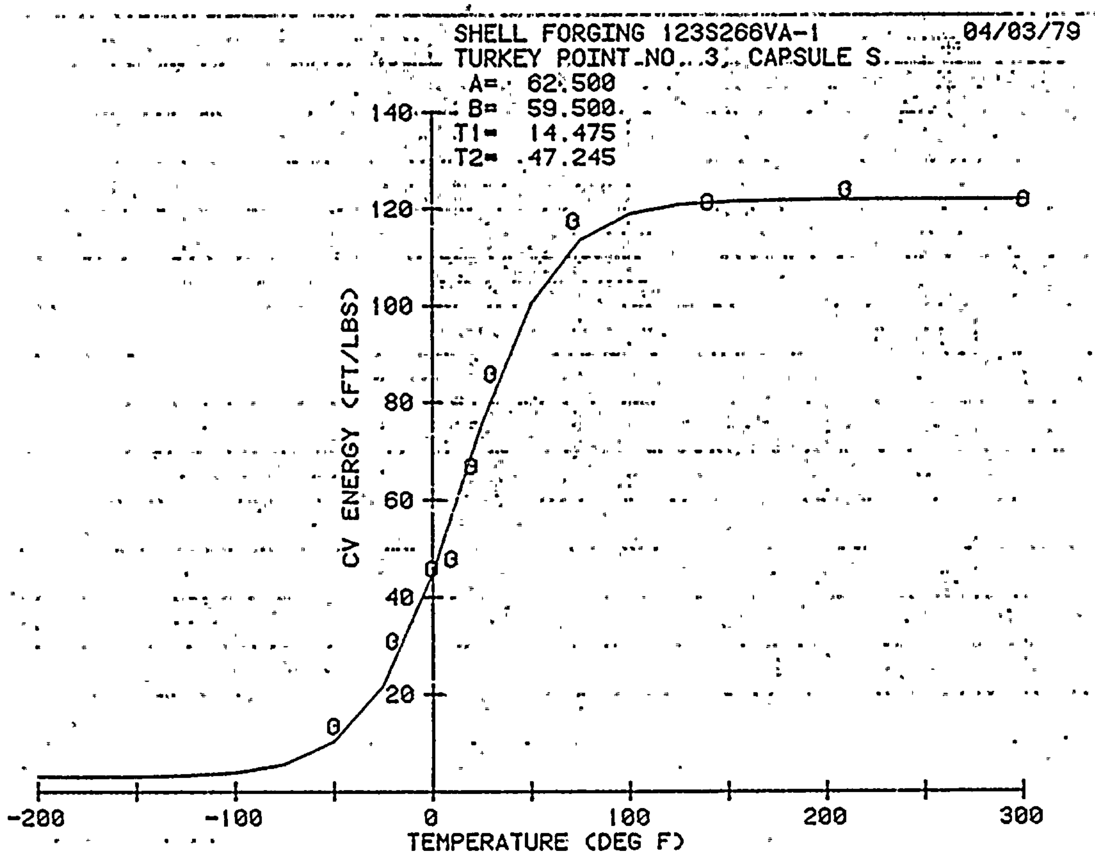

SHELL FORGING 123P461VA-1
04/03/79
TURKEY POINT NO. 3, UNIRRADIATED
A= 73.500
B= 71.500
T1= 1.444
T2= 88.143
CV ENERGY (FT/LBS)
200
180
160
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

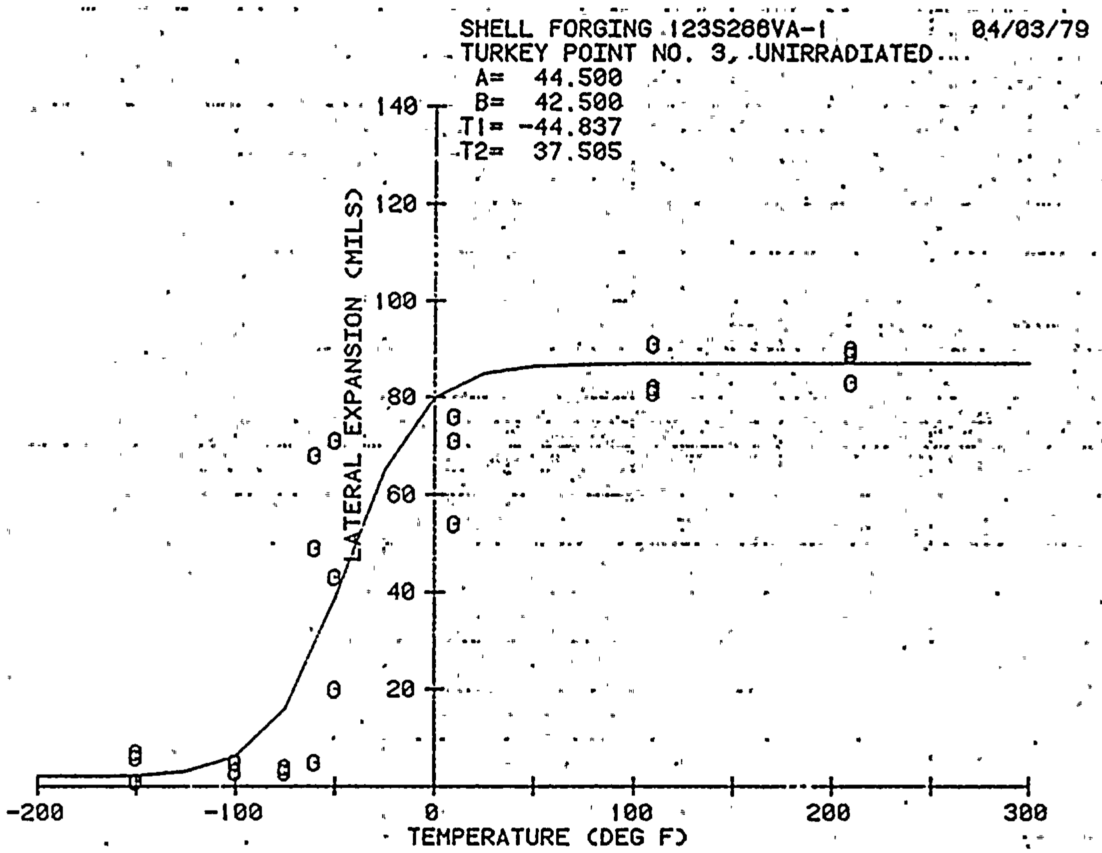

SHELL FORGING 123P461VA-1
04/03/79
TURKEY POINT NO. 3, CAPSULE S
A= 65.500
B= 62.500
T1= 14.398
T2= 81.767
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

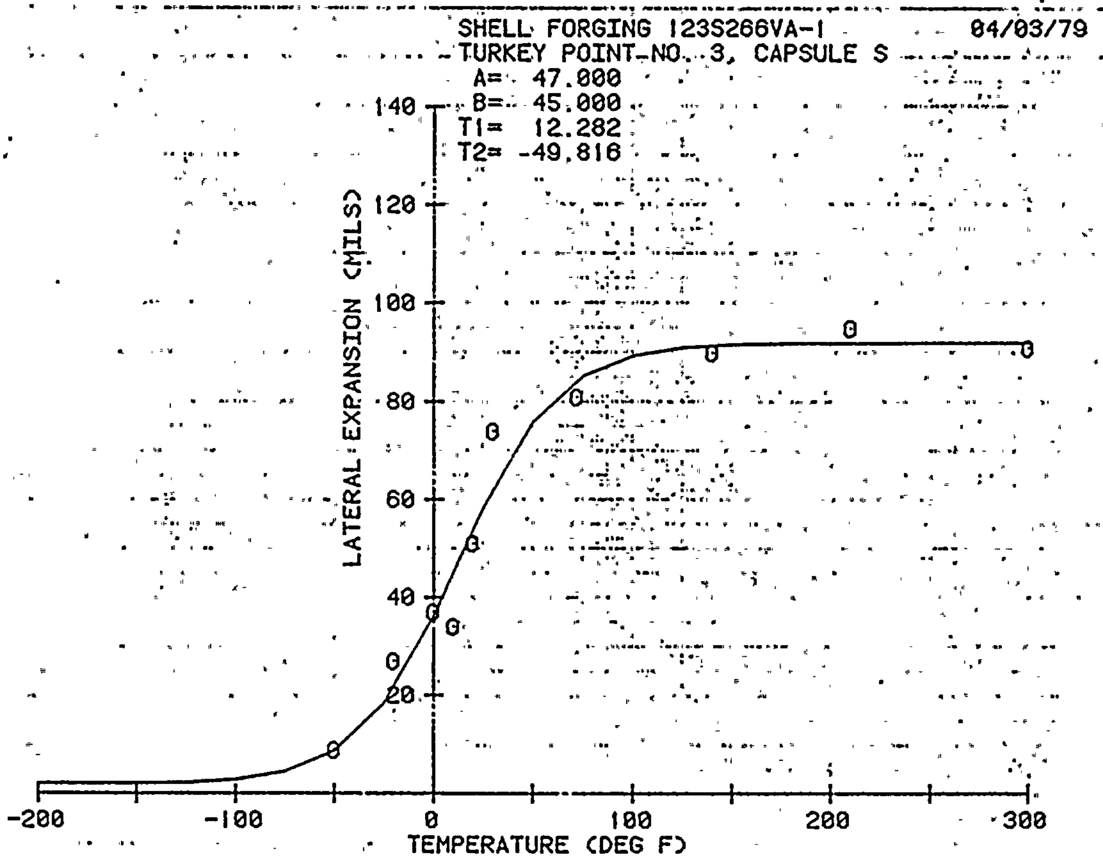

SHELL FORGING 123P461VA-1
04/03/79
TURKEY POINT NO. 3, UNIRRADIATED
A= 45.000
B= 43.000
T1= -27.187
T2= 75.699
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)

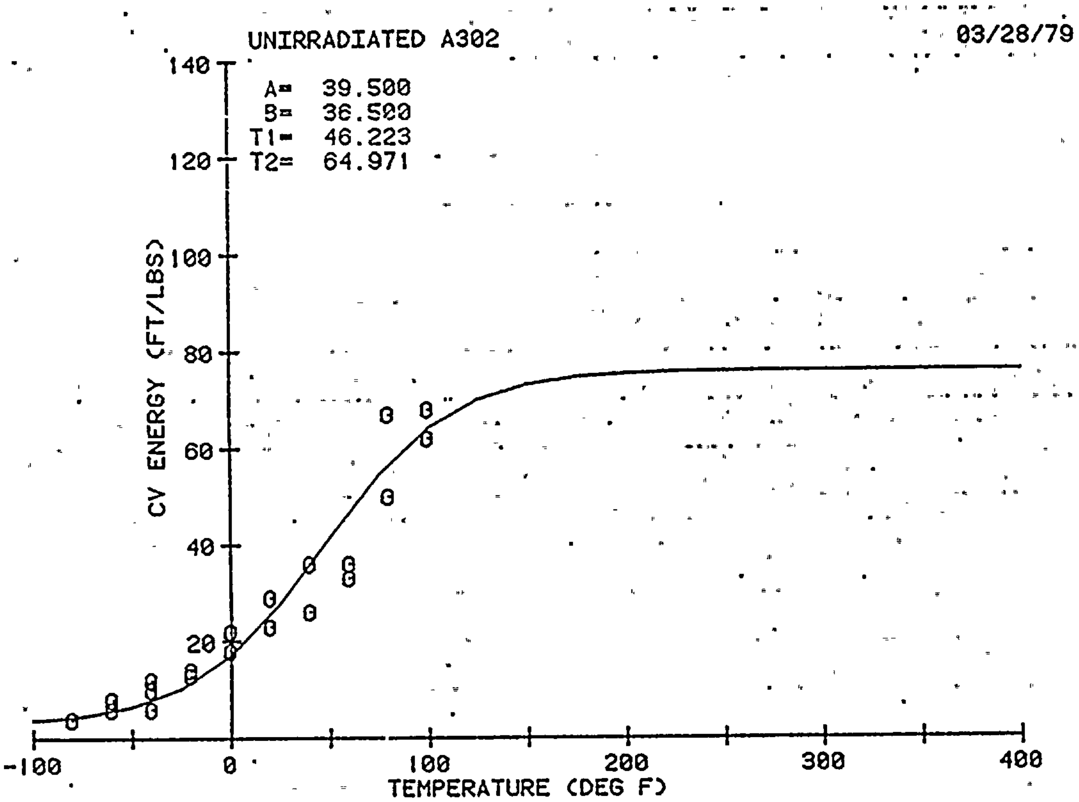

SHELL FORGING 123P461VA-1
03/04/79
TURKEY POINT NO. 3, CAPSULE S
A= 47.000
B= 45.000
T1= 12.064
T2= 80.811
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)

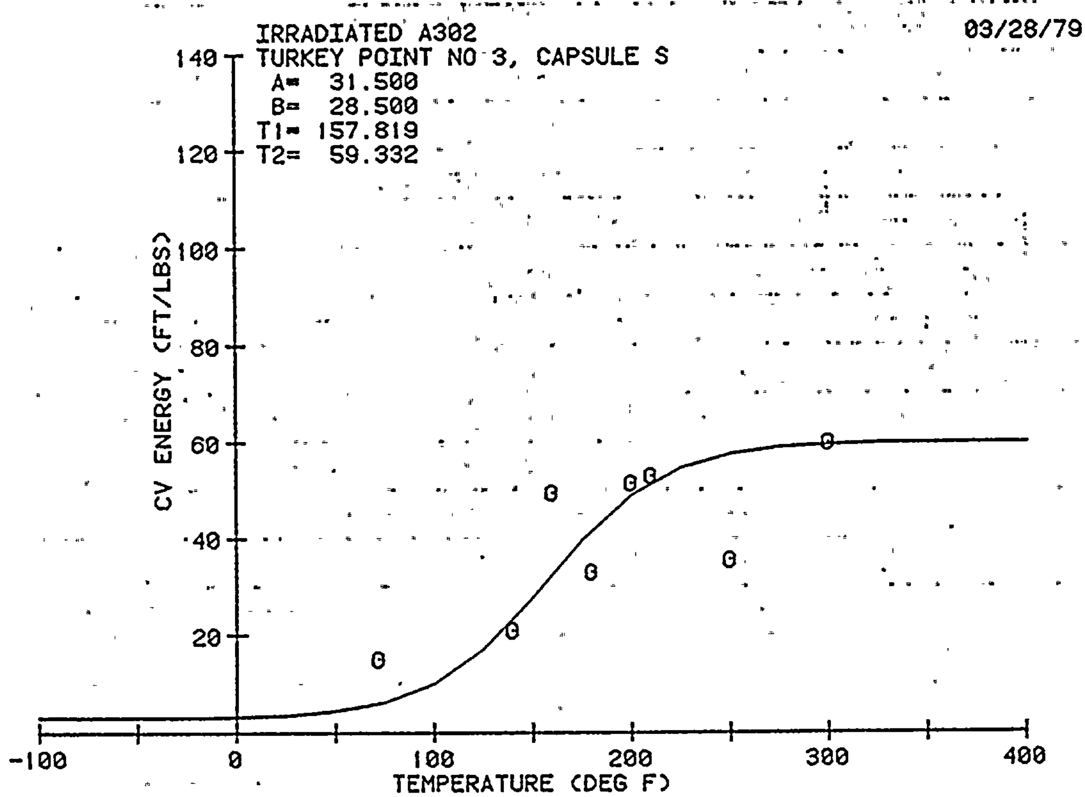

SHELL FORGING 123S266VA-1
04/03/79
TURKEY POINT NO. 3, UNIRRADIATED
A= 78.500
B= 75.500
T1= 6.220
T2= 119.868
CV ENERGY (FT/LBS)
200
180
160
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

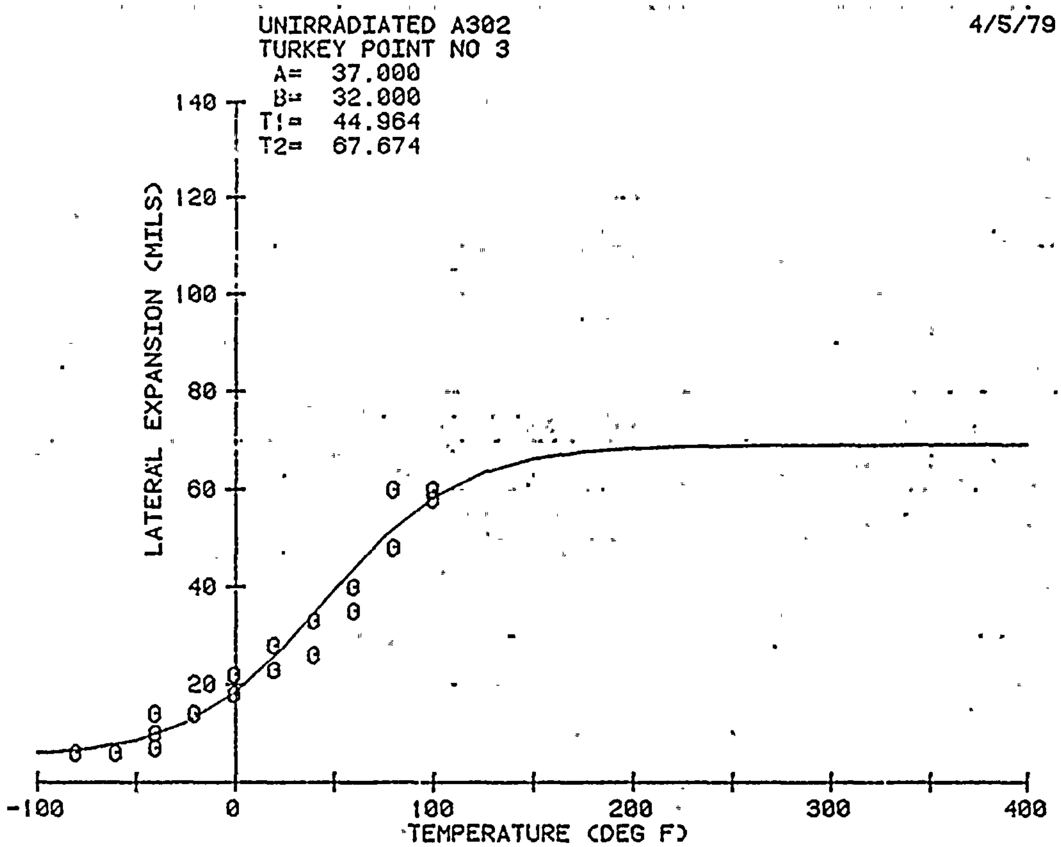

SHELL FORGING 123S266VA-1
04/03/79
TURKEY POINT NO. 3, CAPSULE S
A= 62.500
B= 59.500
T1= 14.475
T2= 47.245
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)


SHELL FORGING 123S266VA-1 04/03/79
TURKEY POINT NO. 3, UNIRRADIATED
A= 44.500
B= 42.500
T1= -44.837
T2= 37.505
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)
140
120
100
80
60
40
20
-200
-100
0
100
200
300

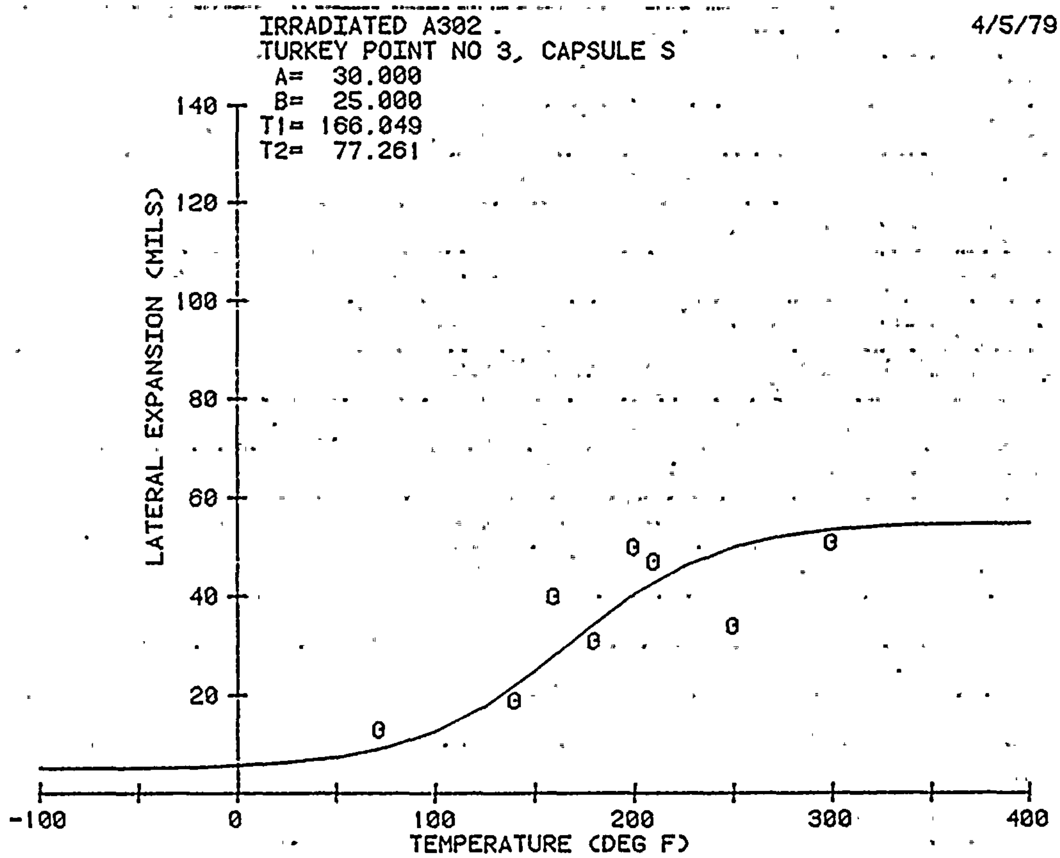

SHELL FORGING 123S266VA-1 04/03/79
TURKEY POINT NO. 3, CAPSULE S
A= 47.000
B= 45.000
T1= 12.282
T2= -49.816
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)
140
120
100
80
60
40
20
-200
-100
0
100
200
300


UNIRRADIATED A302
03/28/79
A= 39.500
B= 36.500
T1= 46.223
T2= 64.971
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-100
0
100
200
300
400
TEMPERATURE (DEG F)


IRRADIATED A302
TURKEY POINT NO 3, CAPSULE S
03/28/79
A= 31.500
B= 28.500
T1= 157.819
T2= 59.332
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-100
0
100
200
300
400
TEMPERATURE (DEG F)



UNIRRADIATED A302
TURKEY POINT NO 3
A= 37.000
B= 32.000
T1= 44.964
T2= 67.674
4/5/79
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)



IRRADIATED A302
TURKEY POINT NO 3, CAPSULE S
A= 30.000
B= 25.000
T1= 166.049
T2= 77.261
4/5/79
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 1 | Est. U.T.S. ______ psi | Project No. 02-5131-001

Spec. No. 5-2 | Initial G. L. 1.000 in. | Machine No. DILLON

Temperature 250 °F | Initial Dia. .250 in. | Date 6-30-78

Strain Rate .01"/MIN | Initial Thickness ______ in. | Initial Area 0.0491

Initial Width ______ in.

Top Temperature 254° °F | Maximum Load 4125 lb

Bottom Temperature 253° °F | 0.2% Offset Load 3150 lb

Final Gage Length 1.234 in. | 0.02% Offset Load ______ lb

Final Diameter .144 in. | Upper Yield Point ______ lb

Final Area 0.0163 in.^{2} | FINAL DIA. (UN-NECKED PORTION) .238 IN.

$$\text{U.T.S.} = \frac{\text{Maximum Load}}{\text{Initial Area}} = 84{,}010 \text{ psi}$$

$$0.2\%\ \text{Y.S.} = \frac{0.2\%\ \text{Offset Load}}{\text{Initial Area}} = 64{,}150 \text{ psi}$$

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

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

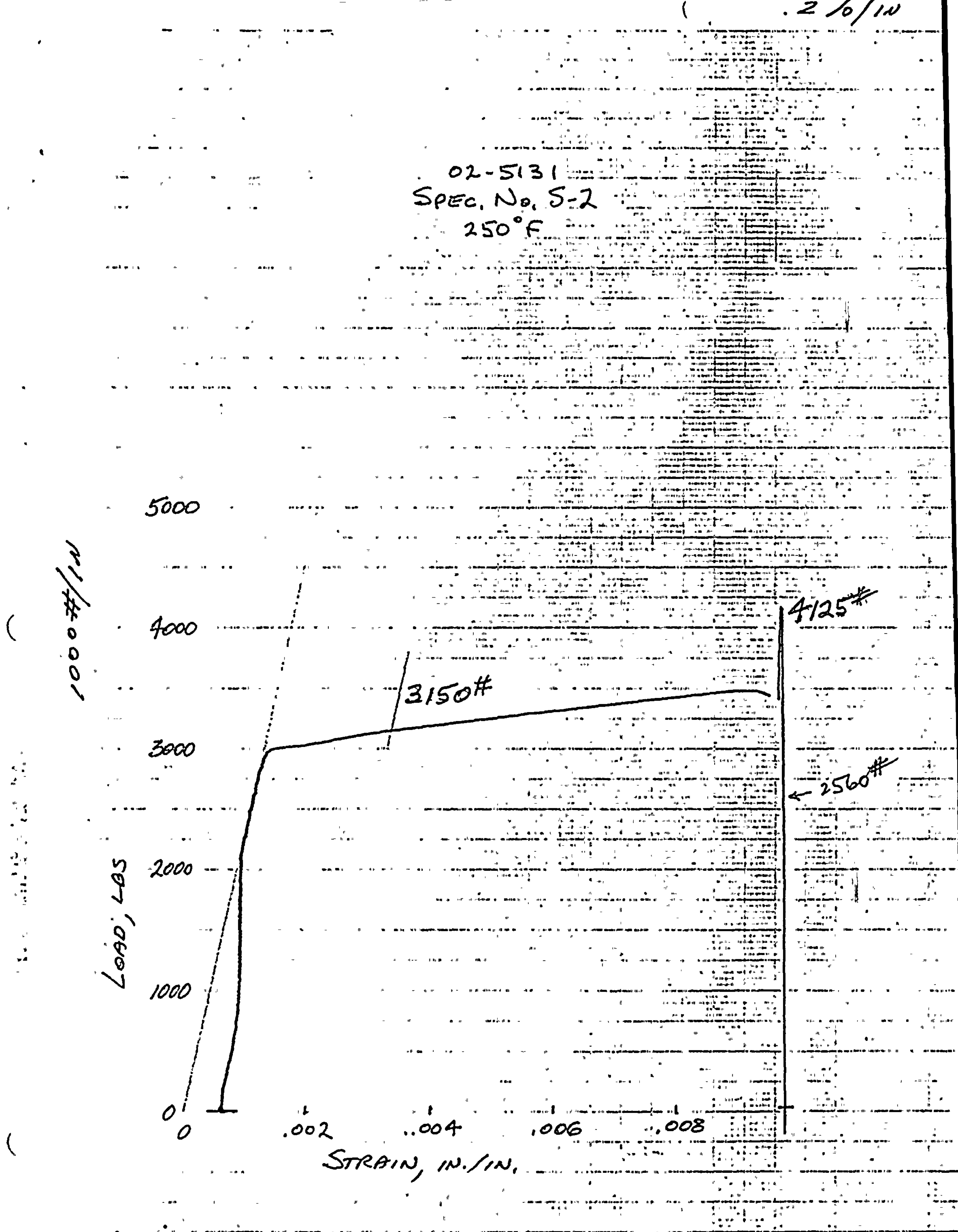
$$\%\ \text{Elongation} = \frac{\text{Final G.L.} - \text{Initial G.L.}}{\text{Initial G.L.}} \times 100 = 23.4\ \%$$

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

$$\text{UNIFORM ELONG.} = \left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. (UN-NECKED)}}\right)^2 - 1\right] \times 100 = 10.3\ \%$$

Signature: [signature]

[signature]


.2 %/IN
02-5131
SPEC. NO. S-2
250°F
1000#/IN
5000
4000
3000
2000
1000
0
LOAD, LBS
3150#
4125#
2560#
0
.002
.004
.006
.008
STRAIN, IN./IN.

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 4 | Est. U.T.S. ______ psi | Project No. 02-5131-001

Spec. No. S-1 | Initial G. L. 1.000 in. | Machine No. DILLON

Temperature 550 °F | Initial Dia. .250 in. | Date 8-14-78

Strain Rate .01"/min | Initial Thickness ______ in. | Initial Area 0.0491

Initial Width ______ in.

Top Temperature ______ °F | Maximum Load 3875 lb

Bottom Temperature ______ °F | 0.2% Offset Load 2670 lb

Final Gage Length 1.209 in. | 0.02% Offset Load ______ lb

Final Diameter .146 in. | Upper Yield Point ______ lb

Final Area 0.0167 in.^{2} | FINAL DIA. (UN-NECKED PORTION .241 IN.

U.T.S. $= \frac{\text{Maximum Load}}{\text{Initial Area}} =$ 78,920 psi

0.2% Y.S. $= \frac{\text{0.2\% Offset Load}}{\text{Initial Area}} =$ 54,380 psi

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

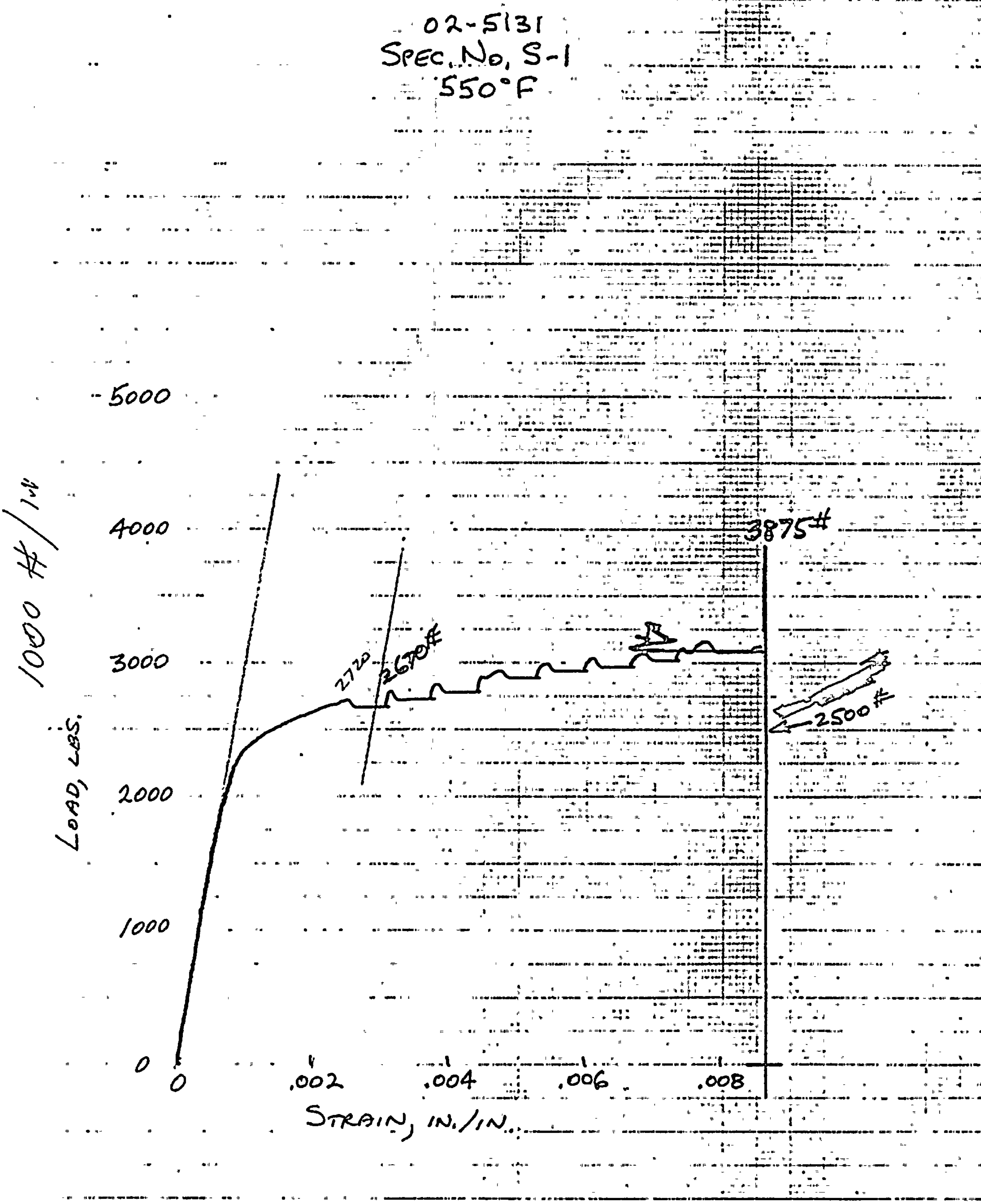
Upper Y.S. $= \frac{\text{Upper Yield Point}}{\text{Initial Area}} =$ ______ psi

% Elongation $= \frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}} \times 100 =$ 20.9 %

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

UNIFORM ELONG. $= \left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. (UN-NECKED)}}\right)^2 - 1\right] \times 100 =$ 7.6 %

Signature: Richard C. [signature]
E. B. Jones [signature]



02-5131
SPEC. No. S-1
550°F
5000
4000
3000
2000
1000
0
1000 #/in
LOAD, LBS.
2720
2670#
3875#
2500#
0
.002
.004
.006
.008
STRAIN, IN./IN.

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 2
Est. U.T.S. ______ psi
Project No. 02-5131-001

Spec. No. P-2
Initial G. L. 1.000 in.
Machine No. DILLON

Temperature 250 °F
Initial Dia. .252 in.
Date 6-30-78

Strain Rate .01"/min
Initial Thickness ______ in.
Initial Area 0.0499

Initial Width ______ in.

Top Temperature 253 °F
Maximum Load 4050 lb

Bottom Temperature 251 °F
0.2% Offset Load 3060 lb

Final Gage Length 1.225 in.
0.02% Offset Load ______ lb

Final Diameter .139 in.
Upper Yield Point ______ lb

Final Area .0152 $in.^2$
FINAL DIAMETER (UN-NECKED) .240 IN.

$$\text{U.T.S.} = \frac{\text{Maximum Load}}{\text{Initial Area}} = 81{,}160 \text{ psi}$$

$$0.2\%\ \text{Y.S.} = \frac{0.2\%\ \text{Offset Load}}{\text{Initial Area}} = 61{,}320 \text{ psi}$$

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

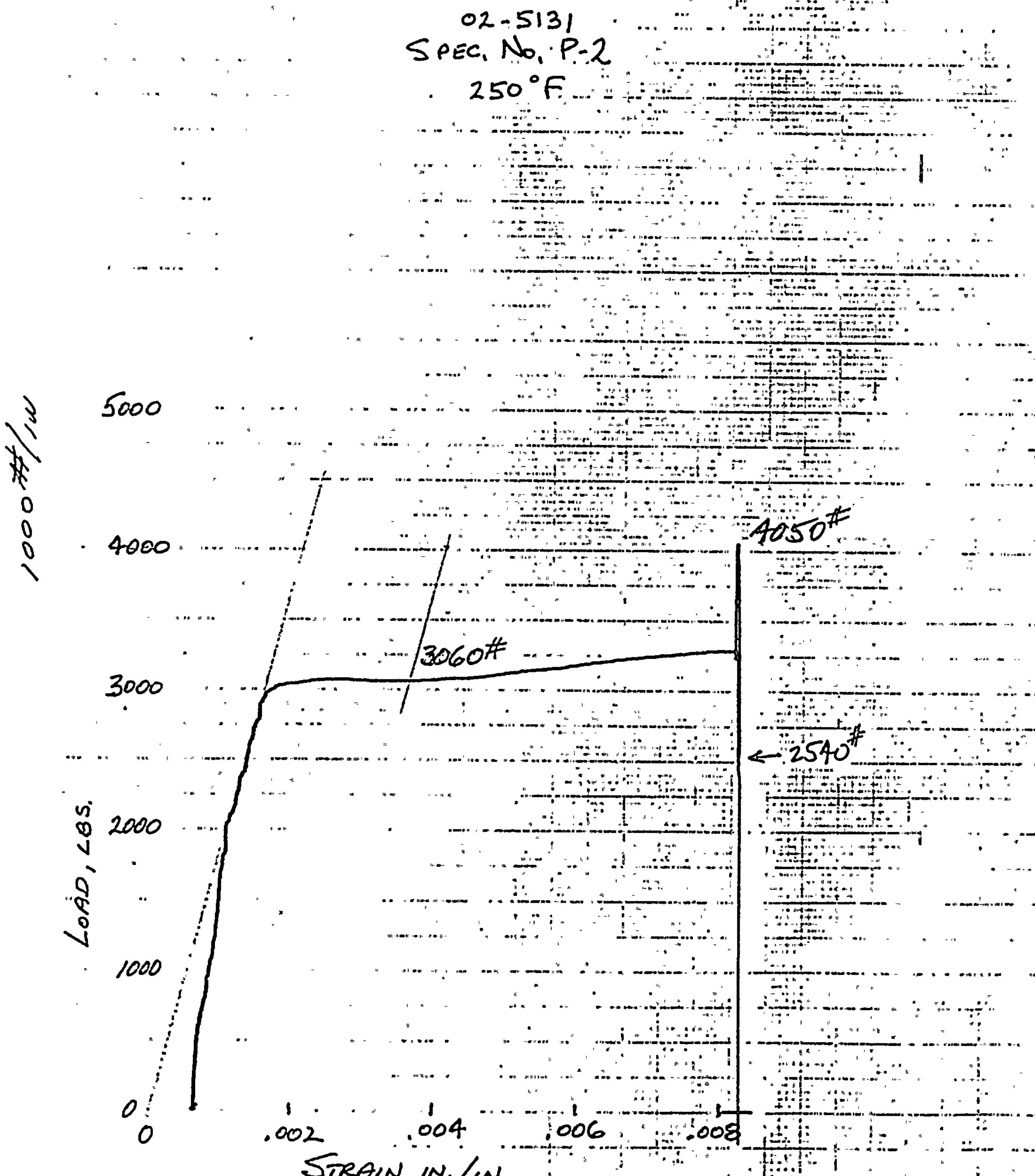
$$\text{Upper Y.S.} = \frac{\text{Upper Yield Point}}{\text{Initial Area}} = \_\_\_\_\_\_ \text{ psi}$$

$$\%\ \text{Elongation} = \frac{\text{Final G. L.} - \text{Initial G. L.}}{\text{Initial G. L.}} \times 100 = 22.5\ \%$$

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

$$\text{UNIFORM ELONG.} = \left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. (UN-NECKED)}}\right)^2 - 1\right] \times 100 = 10.2\%$$

Signature: [signature]
E. B. Jones



02-5131
SPEC. No. P-2
250°F
1000#/IN
LOAD, LBS.
5000
4000
3000
2000
1000
0
0
.002
.004
.006
.008
STRAIN, IN./IN.
4050#
3060#
2540#

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 3 | Est. U.T.S. ______ psi | Project No. 02-5131-001

Spec. No. P1 | Initial G. L. 1.0 in. | Machine No. DILLON

Temperature 550° °F | Initial Dia. .250 in. | Date 7-5-78

Strain Rate .01"/MIN | Initial Thickness ______ in. | Initial Area ______

Initial Width ______ in.

TEST NOT RUN - SPECIMEN OVERHEATED

Top Temperature ______ °F | Maximum Load ______ lb

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

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

Final Diameter ______ in. | Upper Yield Point ______ lb

Final Area ______ in.^{2}

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

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

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

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

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

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

Signature: R. Fitzgerald

APPENDIX B

CHARPY V-NOTCH AND TENSILE TEST DATA
TURKEY POINT UNIT NO. 4
CAPSULE S

TABLE B-1

CHARPY IMPACT DATA - CAPSULE S(a)
TURKEY POINT UNIT NO. 4
(Neutron Fluence = 1.25×10^{19} cm^{-2}, E > 1 MeV)

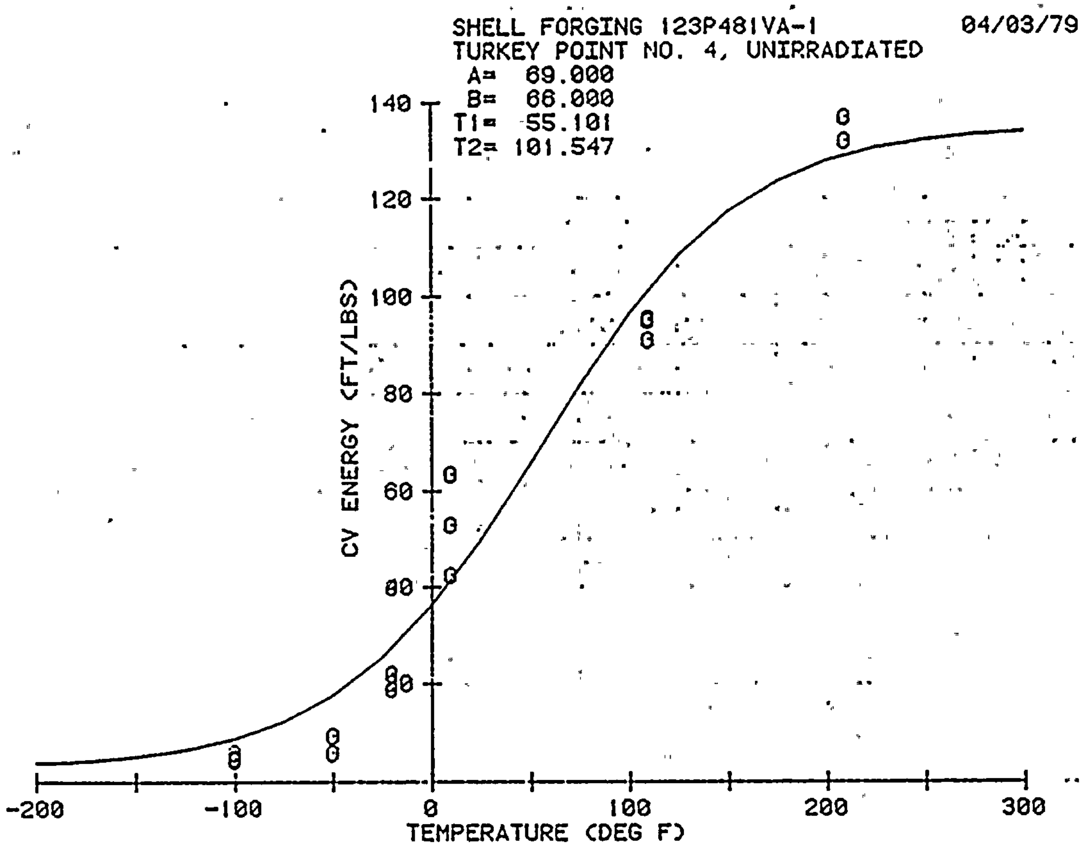
Material	Spec. No.	Temp. (deg F)	Energy (ft-lb)	Lat. Exp. (mils)	Fract. App. (% Shear)
Forging 123P481VA-1	P-10	0	14.2	11	nil
	P-9	20	7.8	7	nil
	P-8	40	51.2	41	nil
	P-7	60	44.6	36	10
	P-1	80	76.1	65	15
	P-2	110	67.5	57	25
	P-3	160	98.3	75	95
	P-4	210	124.5	94	100
	P-5	235	116.7	89	100
	P-6	260	122.0	94	100
Forging 122S180VA-1	S-10	-30	49.2	41	10
	S-9	0	35.6	27	nil
	S-8	20	59.2	47	5
	S-7	40	91.1	69	30
	S-6	60	102.1	81	70
	S-1	80	85.9	70	25
	S-2	110	89.7	72	50
	S-3	160	123.6	93	100
	S-4	210	121.1	93	100
	S-5	235	121.7	87	100
Correlation Monitor Steel	R-1	80	14.4	13	nil
	R-2	110	20.9	17	10
	R-8	135	19.1	16	15
	R-3	160	34.1	29	20
	R-7	185	45.3	40	25
	R-4	210	55.4	48	60
	R-5	260	89.7	75	95
	R-6	310	87.2	68	100

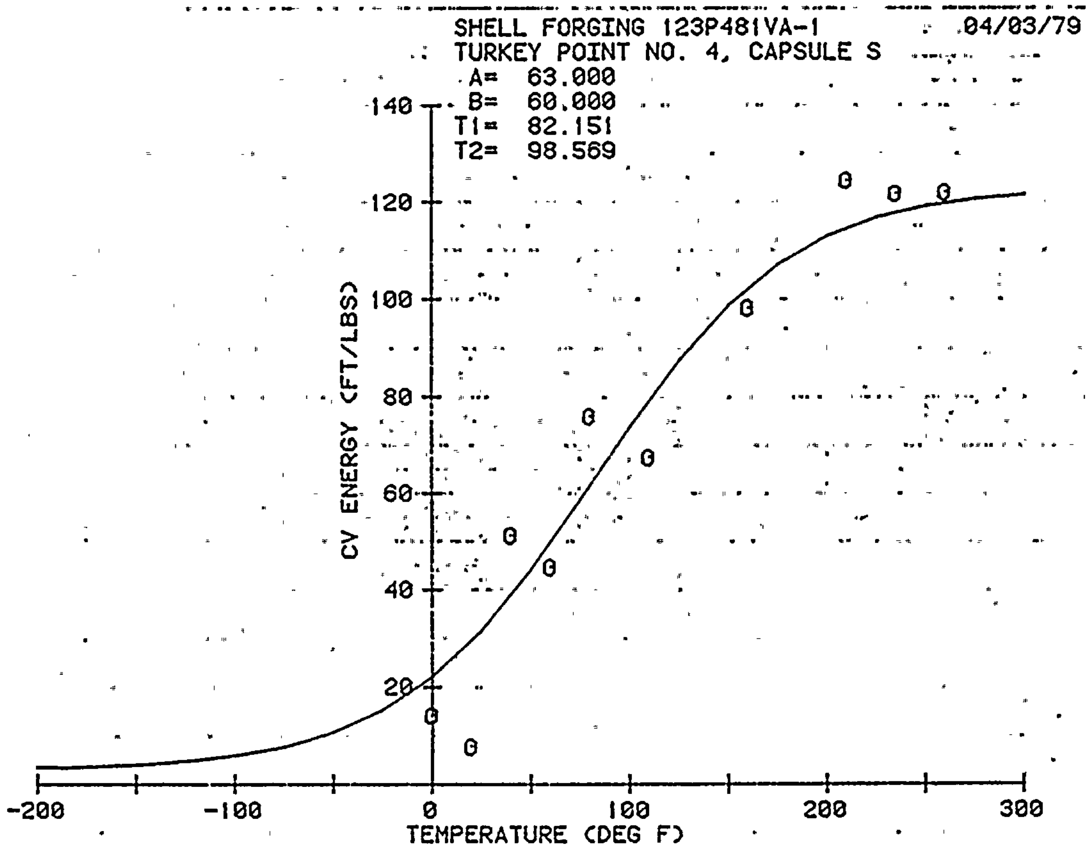
TABLE B-2

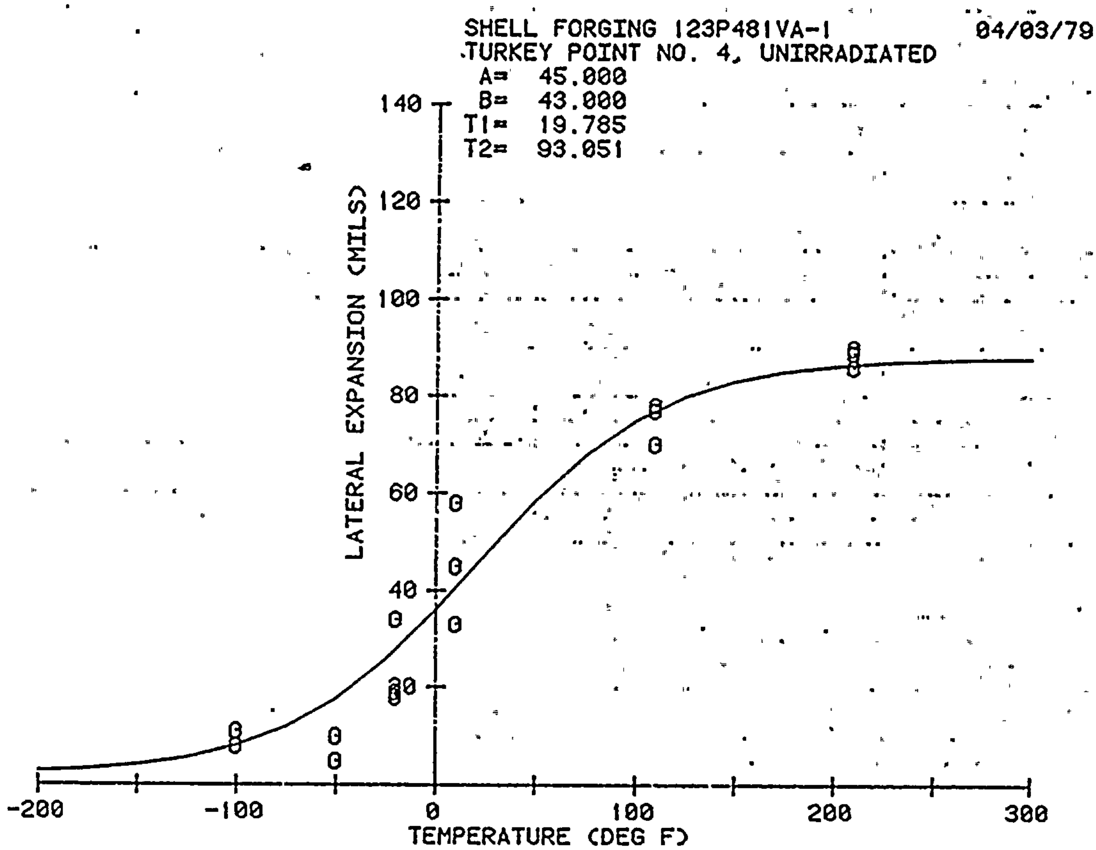
TENSILE PROPERTIES OF SURVEILLANCE MATERIALS
CAPSULE S - TURKEY POINT UNIT NO. 4
(Neutron Fluence = 1.25 x 10^{19} cm^{-2}, $E > 1$ MeV)

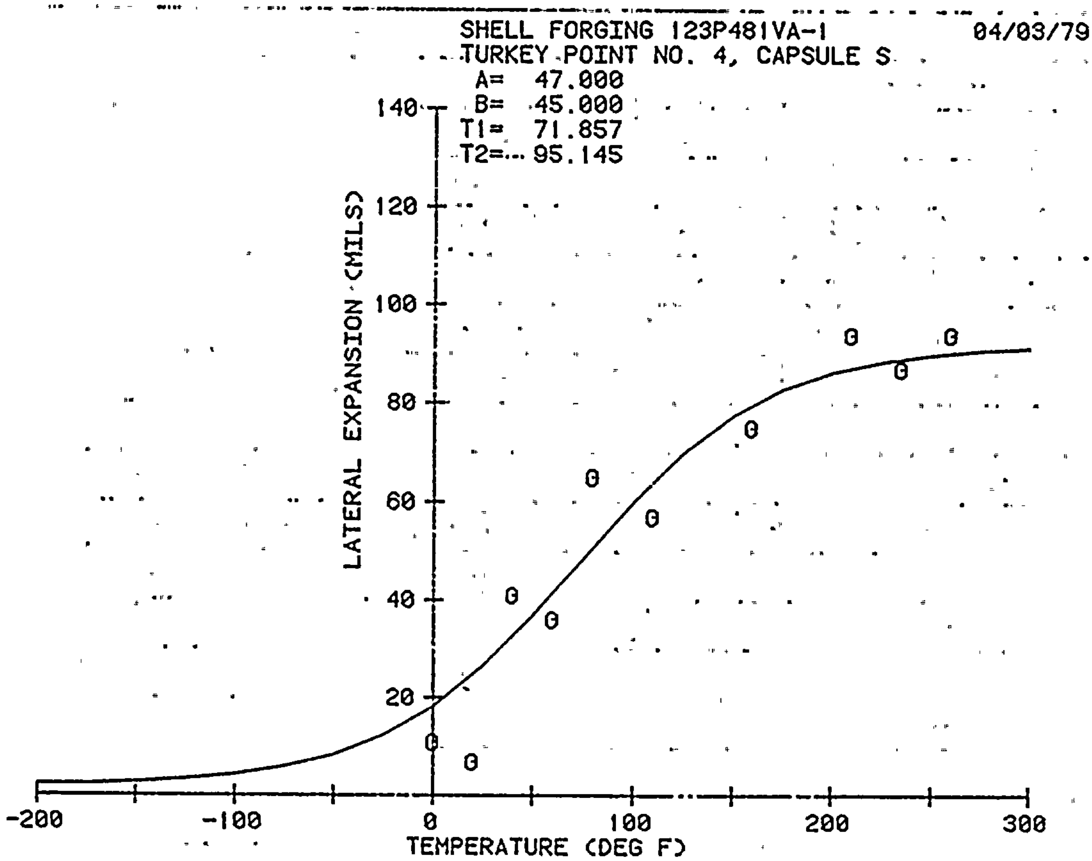
Spec. No.[a]	Test Temp. (°F)	0.2% Yield Strength (psi)	Tensile Strength (psi)	Fracture Strength (psi)	Fracture Stress (psi)	Uniform Elongation[b] (%)	Total Elongation (%)	Reduction in Area (%)
S-1	250	67,800	88,700	58,000	170,200	6.7	20.7	65.9
S-2	550	67,500	91,100	57,400	168,200	5.9	20.9	65.8
P-1	250	66,200	87,800	56,700	176,900	7.6	22.5	68.0
P-2	550	62,300	88,100	61,300	170,300	7.6	19.3	64.0

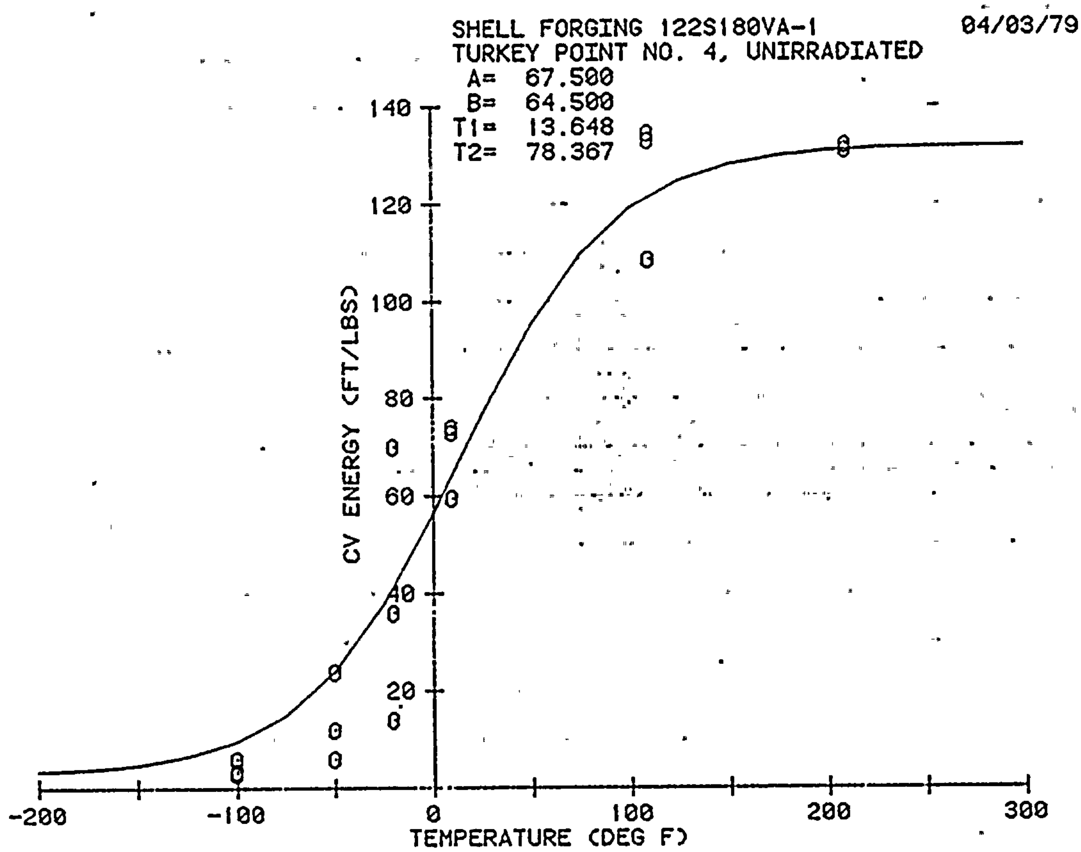
(a) Material Code: S = 122S180VA-1; P = 123P481VA-1
(b) Using change of cross-sectional area in unnecked portion of specimen per ASTM E 184-62.

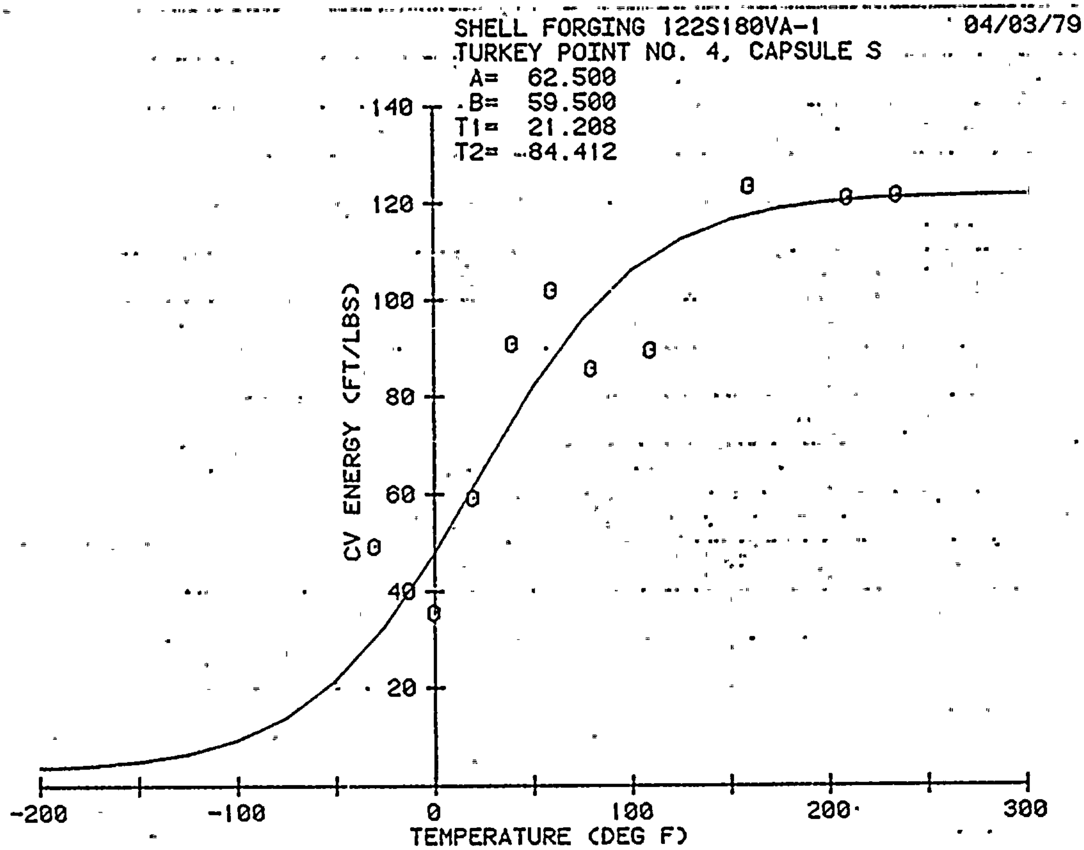

SHELL FORGING 123P481VA-1
04/03/79
TURKEY POINT NO. 4, UNIRRADIATED
A= 69.000
B= 66.000
T1= 55.101
T2= 101.547
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

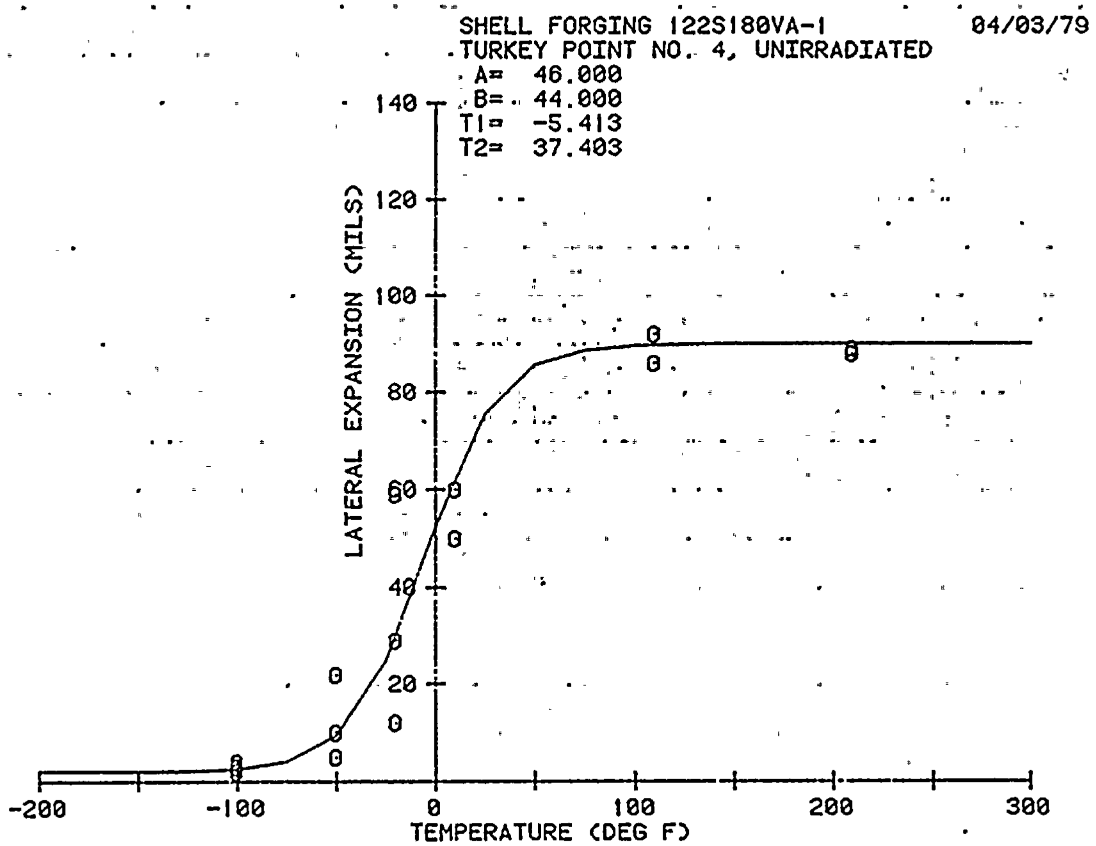

SHELL FORGING 123P481VA-1
04/03/79
TURKEY POINT NO. 4, CAPSULE S
A= 63.000
B= 60.000
T1= 82.151
T2= 98.569
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

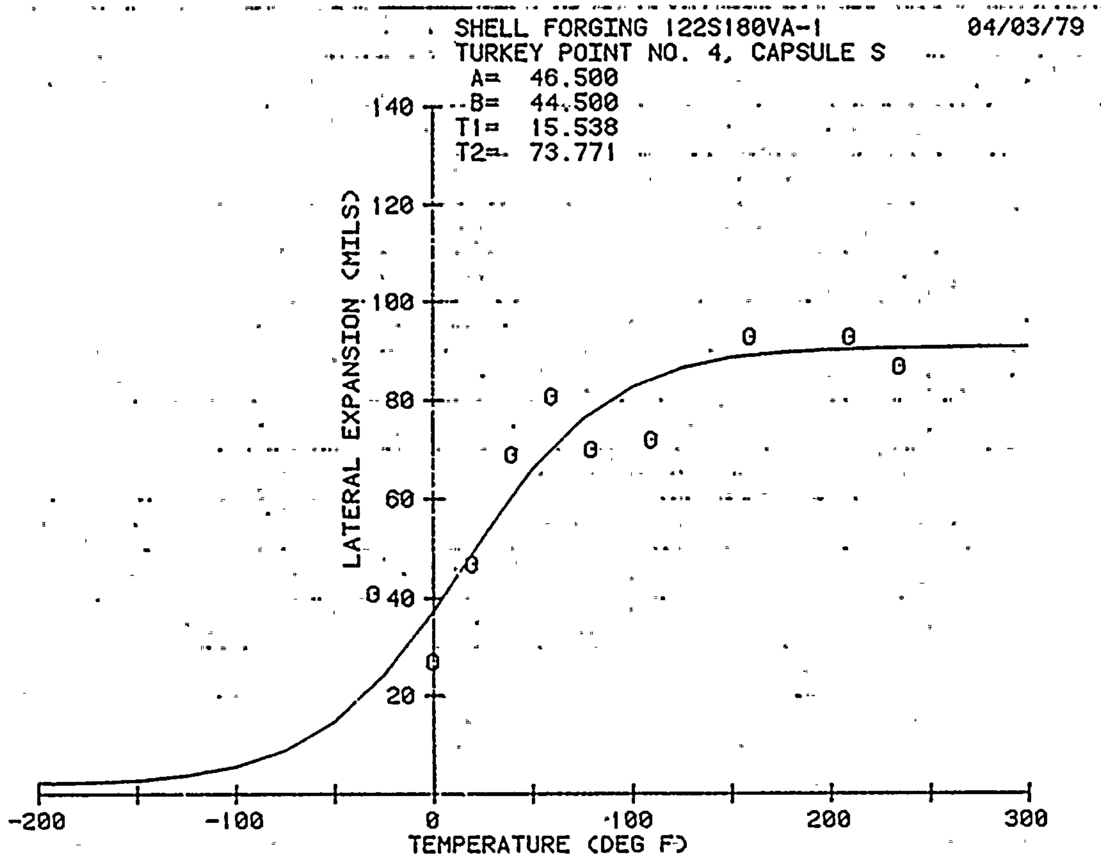

SHELL FORGING 123P481VA-1
04/03/79
TURKEY POINT NO. 4, UNIRRADIATED
A= 45.000
B= 43.000
T1= 19.785
T2= 93.051
LATERAL EXPANSION (MILS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

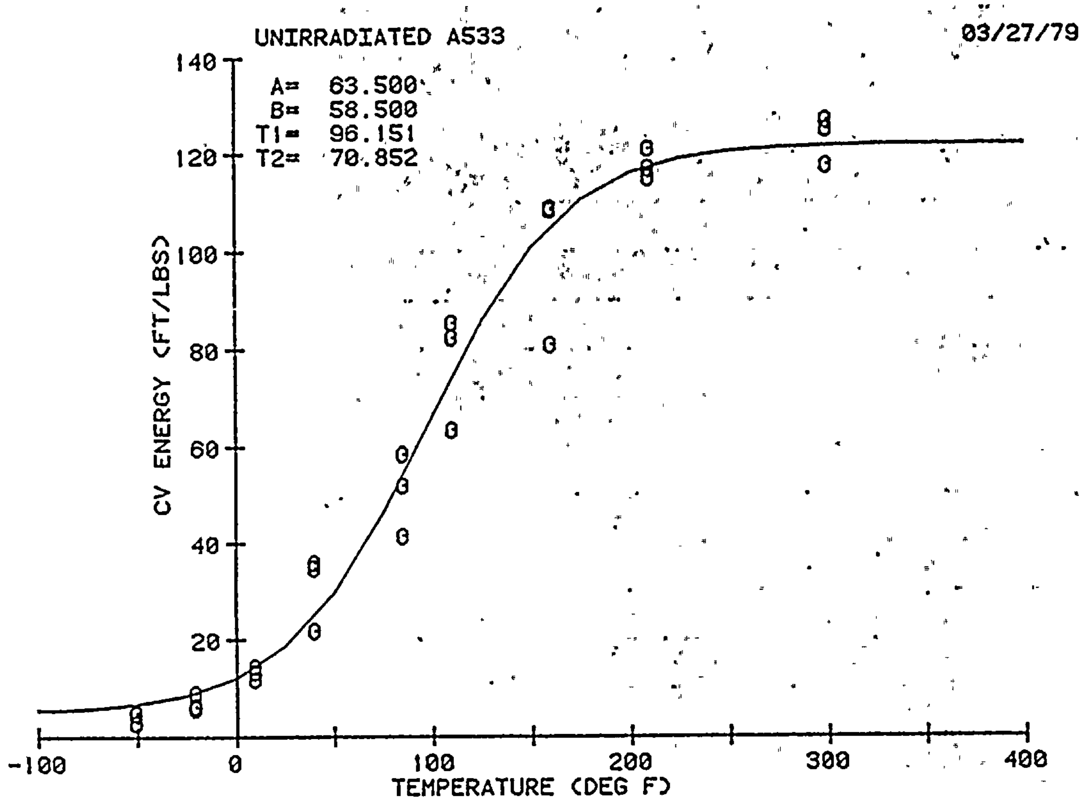

SHELL FORGING 123P481VA-1
04/03/79
TURKEY POINT NO. 4, CAPSULE S
A= 47.000
B= 45.000
T1= 71.857
T2= 95.145
LATERAL EXPANSION (MILS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

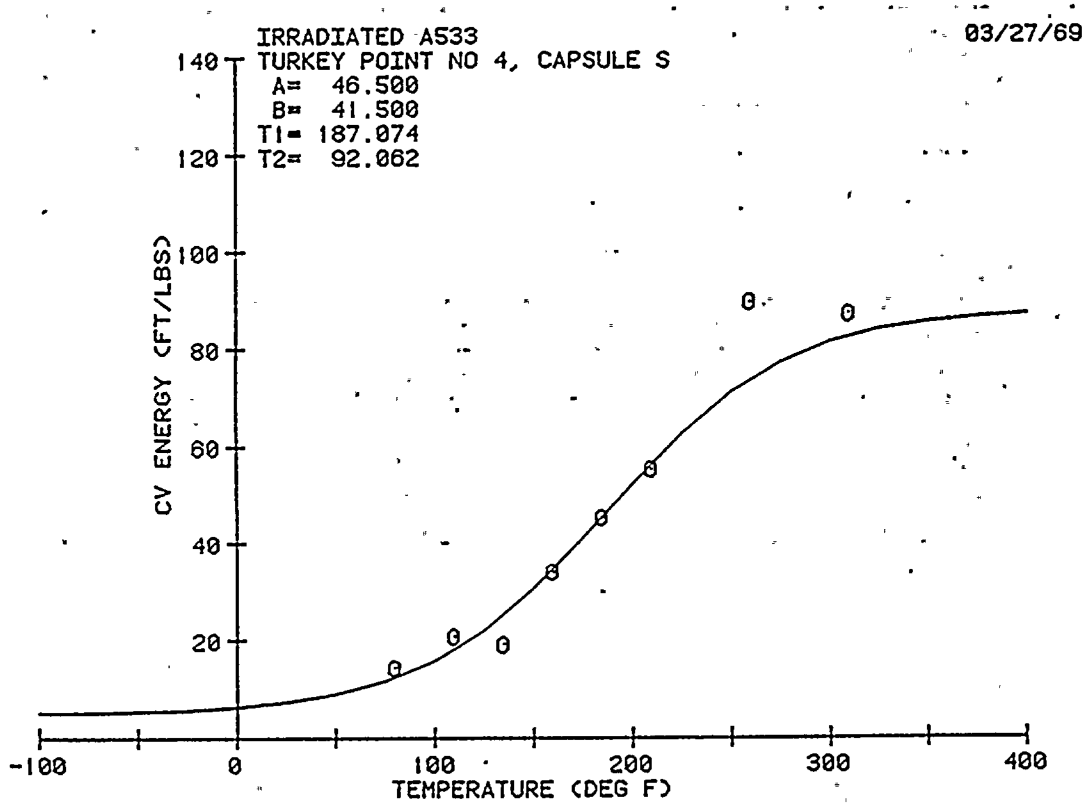

SHELL FORGING 122S180VA-1
04/03/79
TURKEY POINT NO. 4, UNIRRADIATED
A= 67.500
B= 64.500
T1= 13.648
T2= 78.367
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

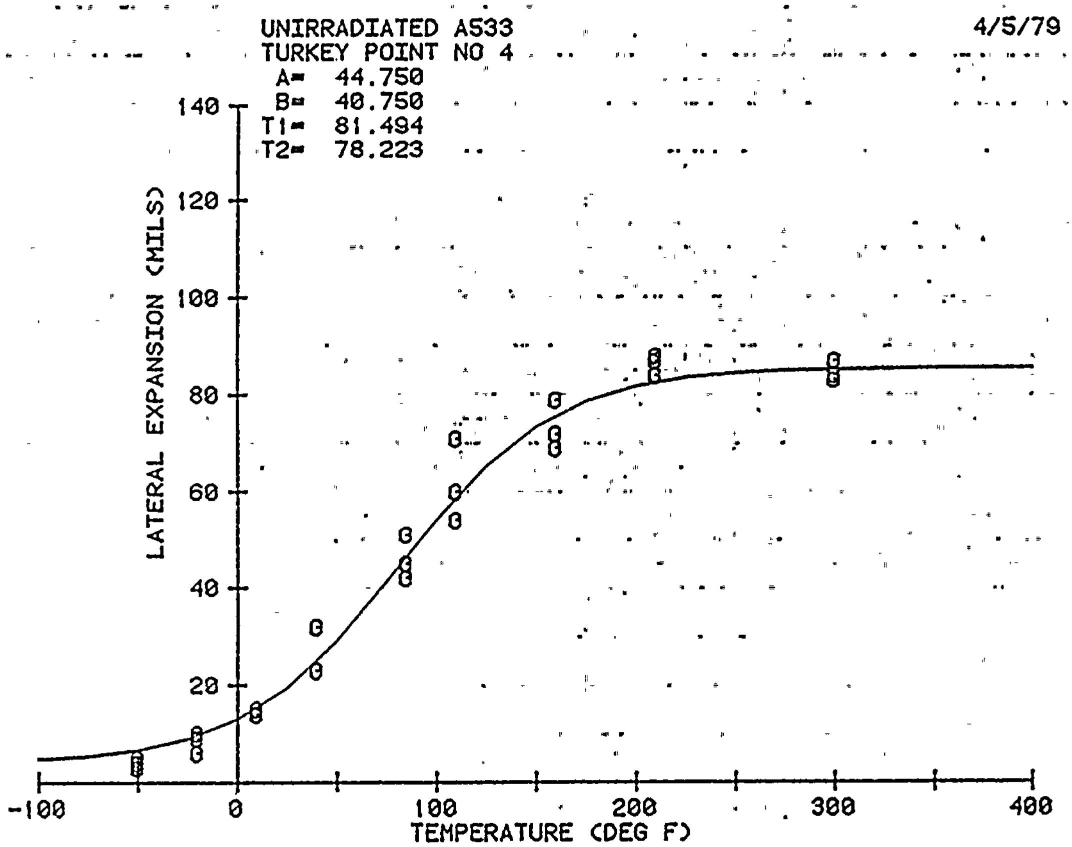

SHELL FORGING 122S180VA-1
04/03/79
TURKEY POINT NO. 4, CAPSULE S
A= 62.500
B= 59.500
T1= 21.208
T2= 84.412
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)

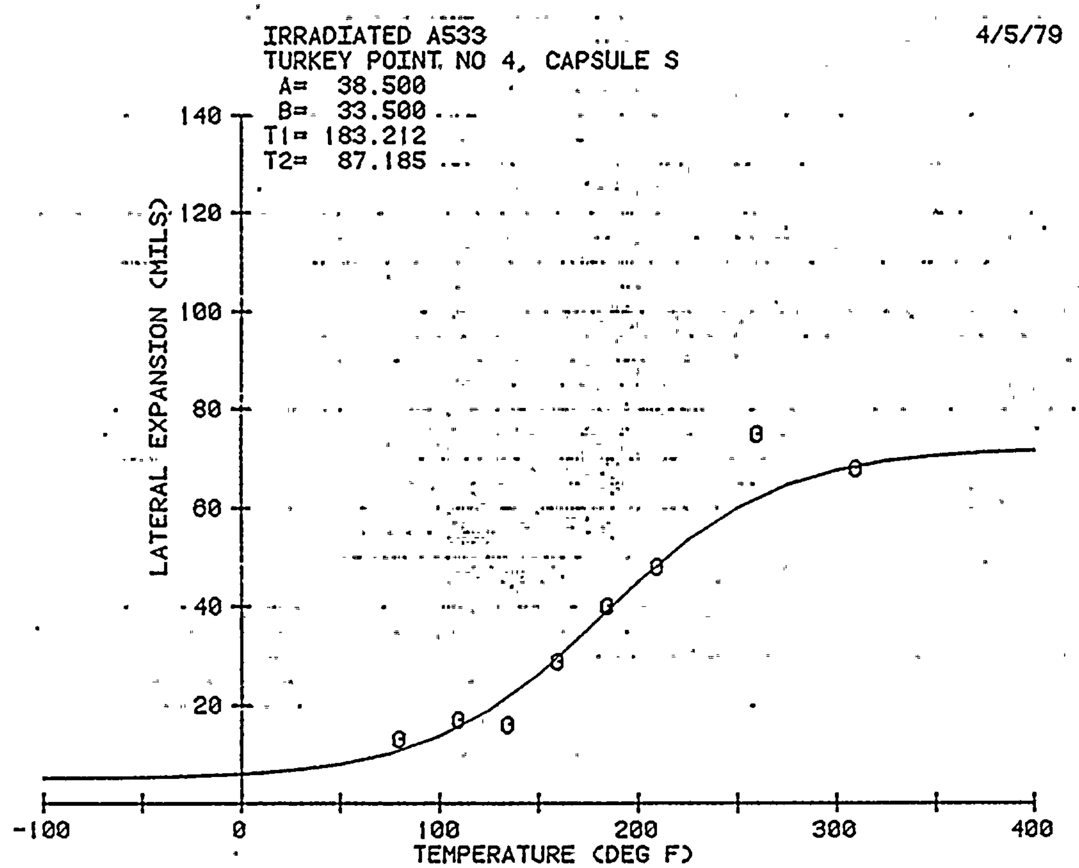

SHELL FORGING 122S180VA-1
04/03/79
TURKEY POINT NO. 4, UNIRRADIATED
A= 46.000
B= 44.000
T1= -5.413
T2= 37.403
LATERAL EXPANSION (MILS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)


SHELL FORGING 122S180VA-1
04/03/79
TURKEY POINT NO. 4, CAPSULE S
A= 46.500
B= 44.500
T1= 15.538
T2= 73.771
LATERAL EXPANSION (MILS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)


UNIRRADIATED A533
03/27/79
A= 63.500
B= 58.500
T1= 96.151
T2= 70.852
CV ENERGY (FT/LBS)
TEMPERATURE (DEG F)
140
120
100
80
60
40
20
-100
0
100
200
300
400


IRRADIATED A533
TURKEY POINT NO 4, CAPSULE S
03/27/69
A= 46.500
B= 41.500
T1= 187.074
T2= 92.062
CV ENERGY (FT/LBS)
TEMPERATURE (DEG F)
140
120
100
80
60
40
20
-100
0
100
200
300
400


UNIRRADIATED A533
TURKEY POINT NO 4
A= 44.750
B= 40.750
T1= 81.494
T2= 78.223
4/5/79
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)


IRRADIATED A533
TURKEY POINT NO 4, CAPSULE S
A= 38.500
B= 33.500
T1= 183.212
T2= 87.185
4/5/79
LATERAL EXPANSION (MILS)
TEMPERATURE (DEG F)

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 2 | Est. U.T.S. ____ psi | Project No. 02-5380-001

Spec. No. S-1 | Initial G. L. 1.000 in. | Machine No. DILLON

Temperature +250 °F | Initial Dia. .250 in. | Date 12-5-78

Strain Rate .01"/min | Initial Thickness ____ in. | Initial Area .0490

Initial Width ____ in.

Top Temperature +250 °F | Maximum Load 4355 lb

Bottom Temperature +249 °F | 0.2% Offset Load 3330 lb

Final Gage Length 1.207 in. | 0.02% Offset Load ____ lb

Final Diameter .146 in. | Upper Yield Point ____ lb

Final Area .01674 in.^{2} | FINAL DIA. (UN-NECKED PORTION) .242 IN.

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 88,897 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 67,821 psi

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

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

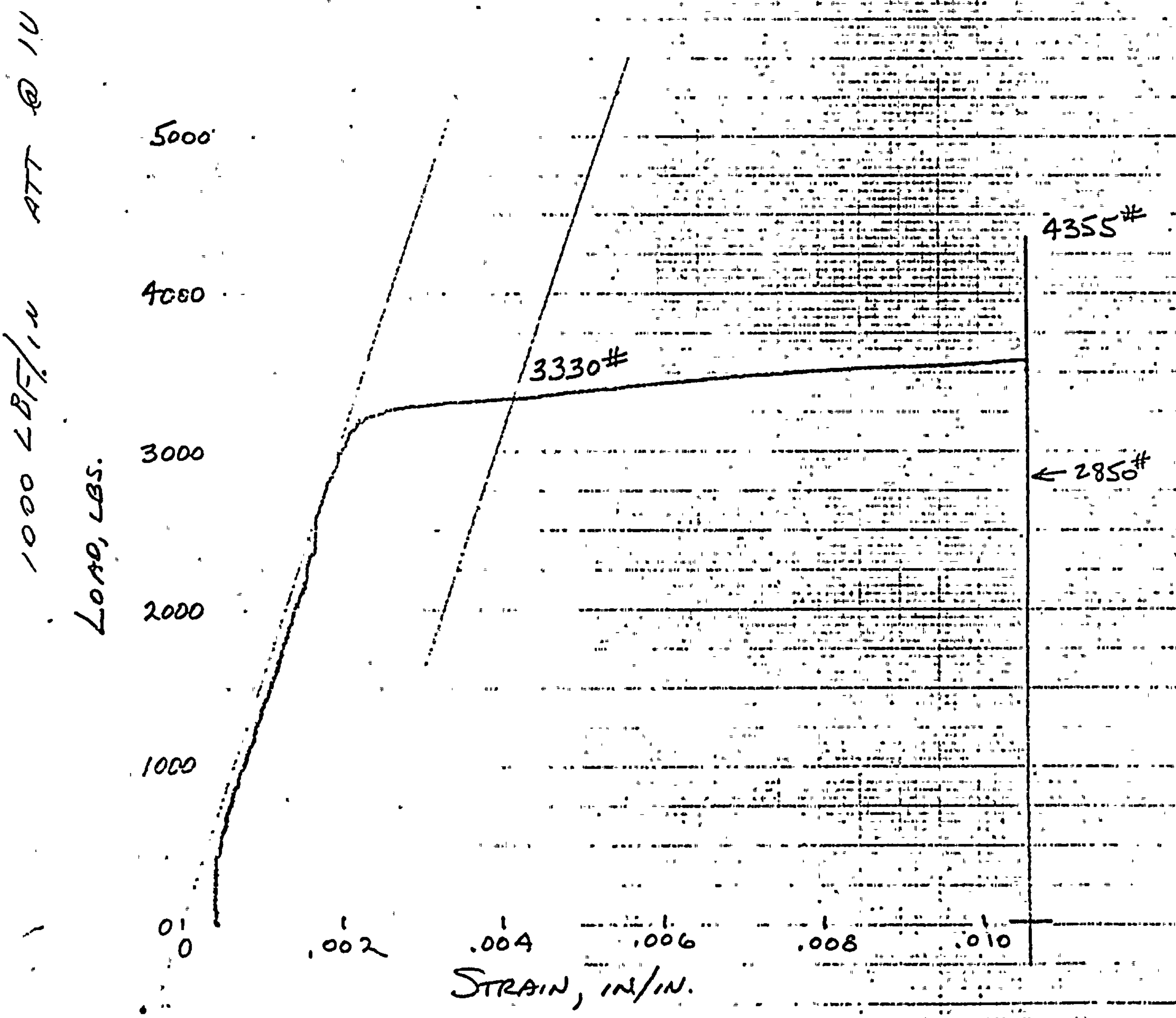
% Elongation = $\frac{\text{Final G. L. - Initial G. L.}}{\text{Initial G. L.}}$ x 100 = 20.7 %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 65.80 %

UNIFORM ELONG. = $\left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. (UN-NECKED)}}\right)^2 - 1\right]$ x100 = 6.7 %

Signature: [signature]

02-5380
SPEC. NO. S-1
250°F



2.70/IN att @ .

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TENSILE TEST DATA SHEET

Test No. T- 4
Spec. No. S-2
Temperature +550 °F
Strain Rate .01"/MIN

Est. U.T.S. ______ psi
Initial G. L. 1.000 in.
Initial Dia. .248 in.
Initial Thickness ______ in.
Initial Width ______ in.

Project No. 02-5380-001
Machine No. DILLON
Date 12-6-78
Initial Area .0483

Top Temperature +551 °F
Bottom Temperature +549 °F
Final Gage Length 1.209 in.
Final Diameter .145 in.
Final Area .0165 in.^{2}

Maximum Load 4400 lb
0.2% Offset Load 3260 lb
0.02% Offset Load ______ lb
Upper Yield Point ______ lb
FINAL DIA. (UN-NECKED PORTION) .241 IN.

$$\text{U.T.S.} = \frac{\text{Maximum Load}}{\text{Initial Area}} = 91{,}097 \text{ psi}$$

$$0.2\% \text{ Y.S.} = \frac{0.2\% \text{ Offset Load}}{\text{Initial Area}} = 67{,}495 \text{ psi}$$

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

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

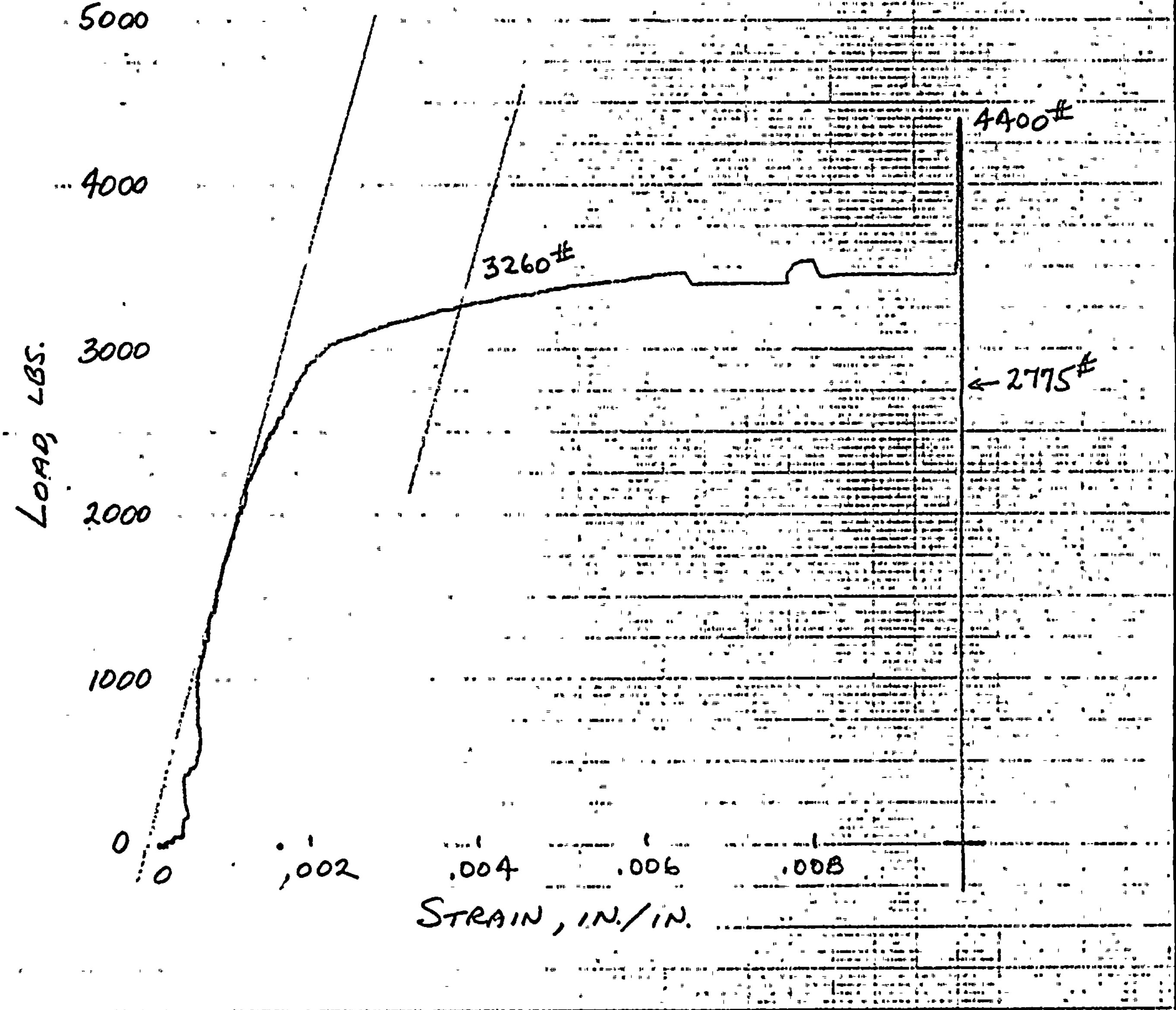
$$\%\text{ Elongation} = \frac{\text{Final G.L.} - \text{Initial G.L.}}{\text{Initial G.L.}} \times 100 = 20.9\,\%$$

$$\%\text{ R.A.} = \frac{\text{Initial Area} - \text{Final Area}}{\text{Initial Area}} \times 100 = 65.8\cancel{4}\,\%$$

$$\text{UNIFORM ELONG.} = \left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. (UN-NECKED)}}\right)^2 - 1\right] \times 100 = 5.9\,\%$$

Signature: [signature]

02-5380
Spec. No. S-2
550°F



.2 %/in

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 1
Spec. No. P-1
Temperature 250 °F
Strain Rate .01 "/MIN

Est. U.T.S. ______ psi
Initial G. L. 1.000 in.
Initial Dia. .249 in.
Initial Thickness ______ in.
Initial Width ______ in.

Project No. 02-5380-001
Machine No. DILLON
Date 12-5-78
Initial Area .0487

Top Temperature +251 °F
Bottom Temperature +249 °F
Final Gage Length 1.225 in.
Final Diameter .141 in.
Final Area .0156 $in.^2$

Maximum Load 4275 lb
0.2% Offset Load 3225 lb
0.02% Offset Load ______ lb
Upper Yield Point ______ lb
FINAL DIA. (UN-NECKED PORTION .240 IN.

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 87,782 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 66,222 psi

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

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

% Elongation = $\frac{\text{Final G.L. - Initial G.L.}}{\text{Initial G.L.}}$ x 100 = 22.5 %

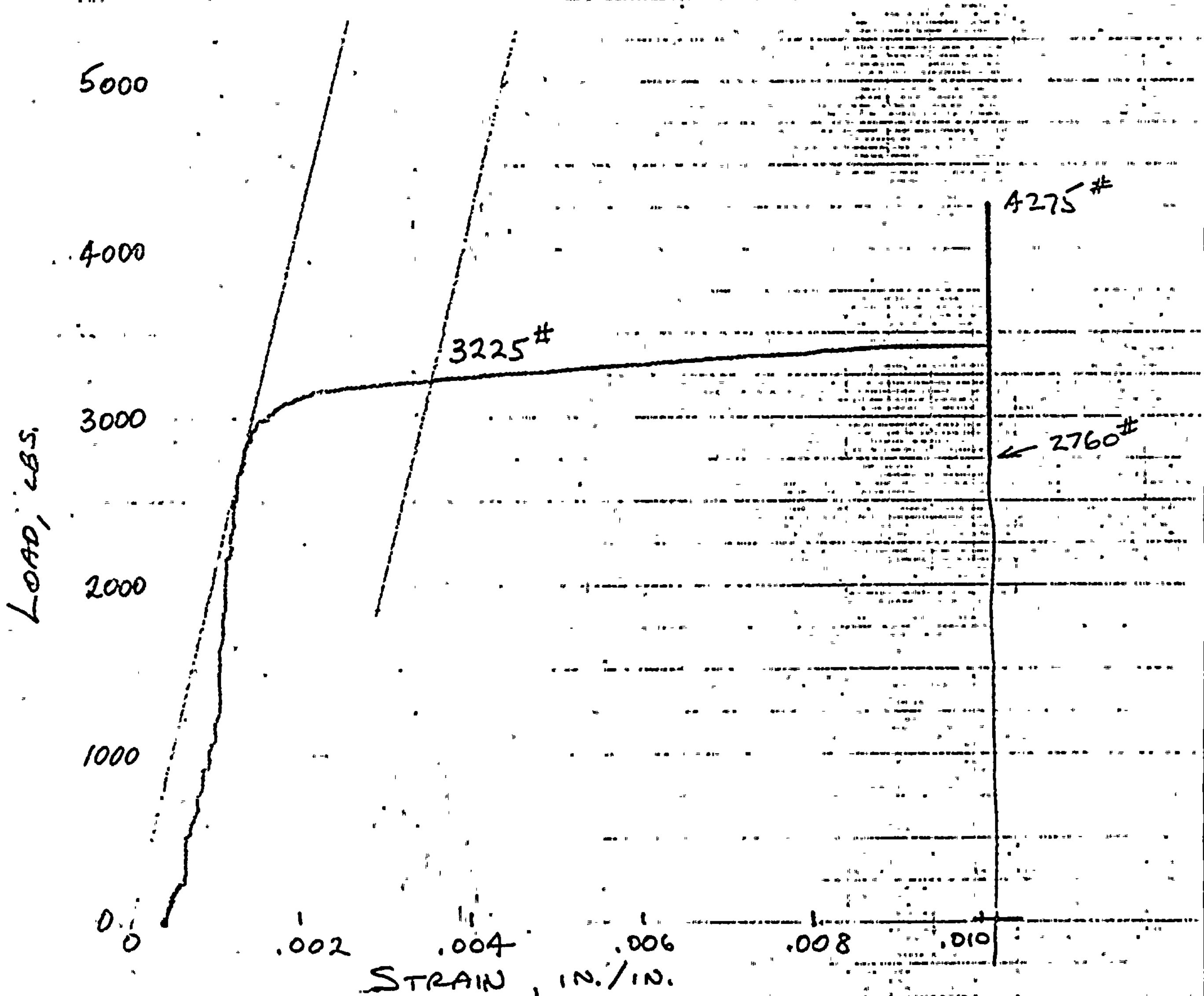
% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 67.97 %

UNIFORM ELONG. = $\left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA. - UNNECKED}}\right)^2 - 1\right] \times 100$ = 7.6 %

Signature: Richard S. Fitzgerald

02-5380
Spec. No. P-1
250°F

1000 LBF/IN ATT @ 1IN



.2 %/IN

Southwest Research Institute
Department of Materials Sciences

TENSILE TEST DATA SHEET

Test No. T- 3
Est. U.T.S. ______ psi
Project No. 02-5380-001

Spec. No. P-2
Initial G. L. 1.000 in.
Machine No. Dillon

Temperature 550 °F
Initial Dia. .250 in.
Date 12/7/78

Strain Rate .01 "/min.
Initial Thickness ______ in.
Initial Area .0491

Initial Width ______ in.

Top Temperature +551 °F
Maximum Load 4325 lb

Bottom Temperature +550 °F
0.2% Offset Load 3060 lb

Final Gage Length 1.193 in.
0.02% Offset Load ______ lb

Final Diameter 0.150 in.
Upper Yield Point ______ lb

Final Area .01767 in.^{2}
FINAL DIA. (UN-NECKED PORTION) .241 IN.

U.T.S. = $\frac{\text{Maximum Load}}{\text{Initial Area}}$ = 88,085 psi

0.2% Y.S. = $\frac{\text{0.2\% Offset Load}}{\text{Initial Area}}$ = 62,322 psi

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

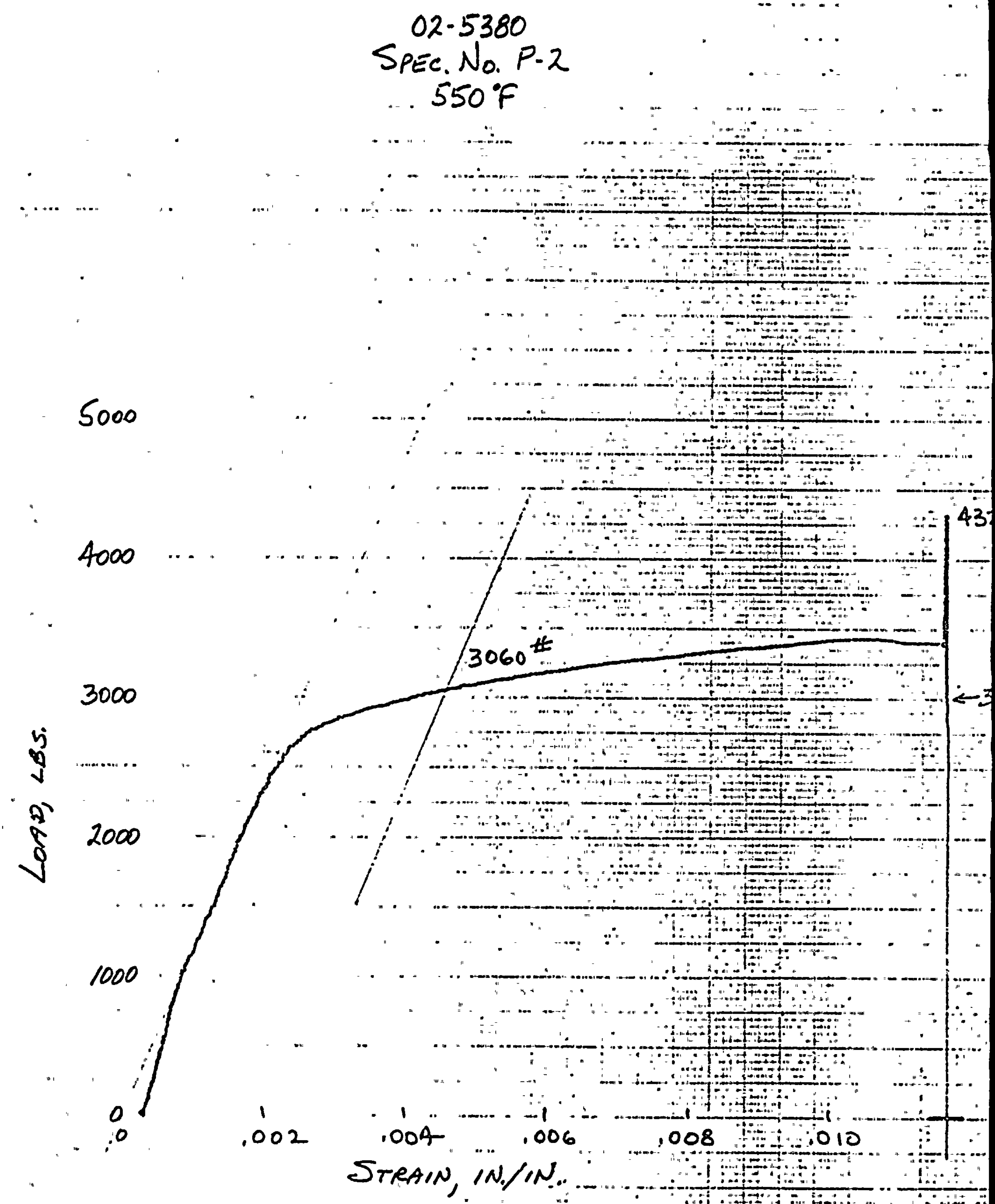
Upper Y.S. = $\frac{\text{Upper Yield Point}}{\text{Initial Area}}$ = ______ psi

% Elongation = $\frac{\text{Final G.L. - Initial G.L.}}{\text{Initial G.L.}}$ x 100 = 19.3 %

% R.A. = $\frac{\text{Initial Area - Final Area}}{\text{Initial Area}}$ x 100 = 64.00 %

UNIFORM ELONG. = $\left[\left(\frac{\text{INITIAL DIA.}}{\text{FINAL DIA (UN-NECKED)}}\right)^2 - 1\right] \times 100$ = 7.6 %

Signature: ______


02-5380
SPEC. No. P-2
550°F
5000
4000
3000
2000
1000
0
LOAD, LBS.
3060#
43
.0
.002
.004
.006
.008
.010
STRAIN, IN./IN.