WESTINGHOUSE CLASS 3

WCAP-8810

SOUTHERN ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT
UNIT NO. 1
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

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ABSTRACT

A pressure-vessel-steel surveillance program was developed for the Southern Alabama Power Company Joseph M. Farley Unit No. 1 Nuclear Reactor to monitor the radiation effects on the reactor pressure vessel material under operating conditions.

A description of the program, including the materia. to be tested, specimen and capsule design, and preirradiation test results, is presented.

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SECTION 1
PURPOSE AND SCOPE

The purpose of this program is to monitor the radiation effects on the reactor vessel materials of the Southern Alabama Power Company Joseph M. Farley Unit No. 1 under actual operating conditions. Evaluation of the radiation effects is based on the pre-irradiation testing of Charpy V-notch, tensile, and dropweight specimens and post-irradiation testing of Charpy V-notch, tensile, precracked bend bar, and compact tension specimens.

Current reactor pressure vessel material test requirements and acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, as a basis. RT_{NDT} is determined from the drop-weight nil-ductility transition temperature (NDTT) per ASTM E208 and the "weak"[1] direction 50 ft lb Charpy V-notch temperature (or the 35-mil lateral expansion temperature if it is greater). RT_{NDT} is defined as the dropweight NDTT or the temperature 60°F less than the 50 ft lb (or 35-mil) Charpy V-notch temperature, whichever is higher.

Therefore,

$$RT_{NDT} = NDTT, \text{ if } NDTT \geq T_{50(35)} - 60°F$$

and

$$RT_{NDT} = T_{50(35)} - 60°F, \text{ if } T_{50(35)} - 60°F > NDTT$$

where

RT_{NDT} = Reference nil-ductility temperature
NDTT = Nil-ductility transition temperature per ASTM E208
$T_{50(35)}$ = 50 ft lb temperature from Charpy V-notch specimens oriented in the direction normal to the major rolling direction (or the 35 mil temperature if it is greater)

1. Normal to major rolling direction

An empirical relationship between RT_{NDT} and fracture toughness for reactor vessel steels has been developed in Appendix G, "Protection Against Non-ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. This relationship can be employed to set allowable pressure-temperature limitations for normal operation of reactors which are based on fracture mechanics concepts. Appendix G defines an acceptable method for calculating these limitations.

It is known that radiation can shift the Charpy V-notch impact energy curve to higher temperatures.[1,2] Thus, the 50 ft lb temperature and RT_{NDT} increase with radiation exposure. The extent of the shift in the impact energy curve, i.e., the radiation embrittlement is enhanced by certain chemical elements (such as copper) present in reactor vessel steels.[3,4]

The 50 ft lb temperature or RT_{NDT} increase with service can be monitored by a surveillance program which entails the periodic checking of irradiated reactor vessel surveillance specimens. The surveillance program is based on ASTM E185-73 (Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels). Compact Tension fracture mechanics specimens will be used in addition to the Charpy V-notch specimens and the precracked bend bar specimens to evaluate the radiation effects on the fracture toughness of the reactor vessel materials.[5,6,7,8,9,10,11]

Postirradiation testing of the Charpy impact specimens will provide a guide for determining pressure-temperature limits on the plant. Charpy impact test data will determine the shift

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[1] L. F. Porter, "Radiation Effects in Steel," in American Society for Testing Materials in Nuclear Applications, pp. 147-195, American Society for Testing Materials, Philadelphia, 1960.

[2] L. E. Steele and J. R. Hawthorne, "New Information on Neutron Embrittlement and Embrittlement Relief to Reactor Pressure Vessel Steels," NRL-6160, August, 1964.

[3] U. Potapovs and J. R. Hawthorne, "The Effect of Residual Elements on 550°F Irradiation Response of Selected Pressure Vessel Steels and Weldments," NRL 6803, September 9, 1968.

[4] L. E. Steele, "Structure and Composition Effects on Irradiation Sensitivity of Pressure Vessel Steels," ASTM-STP-484, pp. 164-175, American Society for Testing and Materials, Philadelphia, 1970.

[5] E. Landerman, S. E. Yanichko, and W. S. Hazelton, "An Evaluation of Radiation Damage to Reactor Vessel Steels Using Both Transition Temperature and Fracture Mechanics Approaches," American Society for Testing and Materials, STP 426, December, 1967.

[6] M. J. Manjoine, "Biaxial Brittle Fracture Tests," Trans. ASME, Ser. D, J. Basic Eng. 87, 293-98 (1965).

[7] L. Porse, "Reactor-Vessel Design Considering Radiation Effects," Trans. ASME, Ser. D, J. Basic Eng. 86, 743-49 (1964).

[8] R. E. Johnson, "Fracture Mechanics: A Basic for Brittle Fracture Prevention," WAPD-TM-505, Bettis Atomic Power Laboratory, November, 1965.

[9] E. T. Wessel and W. H. Pryle, "Investigation of the Applicability of the Biaxial Brittle Fracture Test for Determining Fracture Toughness," WERL-8844-11, Westinghouse Research Laboratories, August, 1965.

[10] W. K. Wilson, "Analytic Determination of Stress Intensity Factors for the Manjoine Brittle Fracture Test Specimen," WERL-0029-3, Westinghouse Research Laboratories, August, 1965.

[11] R. E. Johnson and E. J. Pasierb, "Fracture Toughness of Irradiated A302-B Steel as Influenced by Microstructure," Amer. Nucl. Soc. Trans. 9, 390-92 (1966).

of the reference temperature with radiation exposure at the plant temperatures. These data can then be reviewed to verify or revise pressure-temperature limits of the vessel during start-up and cooldown (the Charpy specimens are most nearly indicative of the radiation exposure experienced by the vessel). This will allow a check of the predicted shift in the reference temperature. The postirradiation test results of the compact tension specimens and precracked bend bar specimens will provide actual fracture toughness properties for Joseph M. Farley Unit No. 1. These properties may be utilized to establish allowable stress intensity factors for normal operation per ASME Code Appendix G methods.

Six material test capsules, located in the reactor between the neutron shielding pads and the vessel wall, are positioned opposite the center of the core. The test capsules are located in guide tubes attached to the neutron shielding pads. The capsules contain test specimens from a 9-inch-thick plate from the reactor vessel lower shell course adjacent to the core region, representative weld metal and heat-affected-zone (HAZ) metal. The thermal history or heat treatment given these specimens is similar to the thermal history of the reactor vessel material with the exception that the post-weld heat treatment received by the specimens has been simulated (see appendix A).

The six material test capsules contain Charpy V-notch impact specimens, precracked bend bar specimens (from the limiting core region lower shell course plate) tensile specimens, compact tension specimens (from the limiting core region lower shell course plate of the reactor vessel and associated weld metal) and Charpy V-notch impact specimens of HAZ metal. Dosimeters and thermal monitors to measure the integrated neutron flux and the temperature are also located in each of the six material test capsules.

SECTION 2
SAMPLE PREPARATION

2-1. PRESSURE VESSEL MATERIAL

Reactor vessel material from lower shell plate B6919-1 and a weldmen joining sections of material from this plate and an adjoining intermediate shell course plate were supplied by Combustion Engineering. Data on this material are presented in appendix A.

2-2. MACHINING

Test material obtained from the lower shell course plate (after thermal heat treatment and forming of the plate) was taken at least one plate thickness (9 inches) from the quenched ends of the plate. Test specimens were machined from the 1/4 thickness location of the plate after performing a simulated postweld stress-relieving treatment on the material. Specimens were also machined from weld and heat-affected zone metal of a stress-relieved weldment joining sections of the intermediate and lower shell plates. All heat-affected zone specimens were obtained from the weld-heat-affected zone of plate B6919-1.

2-3. Charpy V-Notch Impact Specimens

Charpy V-notch specimens from plate B6919-1 were machined with the longitudinal axis of the specimens both parallel and normal to the major rolling direction. Charpy V-notch specimens from the weld and weld heat-affected zone metal were machined perpendicular to the weld direction with the notch oriented in the direction of the weld (see figure 2-1).

2-4. Tensile Specimens

Tensile specimens from plate B6919-1 were machined with the longitudinal axis of the specimens both parallel and normal to the major rolling direction. Weld specimens were oriented normal to the weld direction (see figure 2-2).

2-5. Bend Bar Specimens

Bend Bar Specimens were machined from plate B6919-1 with the longitudinal axis of the specimen oriented normal to the rolling direction of the plate such that the simulated crack would propagate in the rolling direction of the plate. All bend bar specimens were fatigue precracked according to ASTM E399 (see figure 2-3).

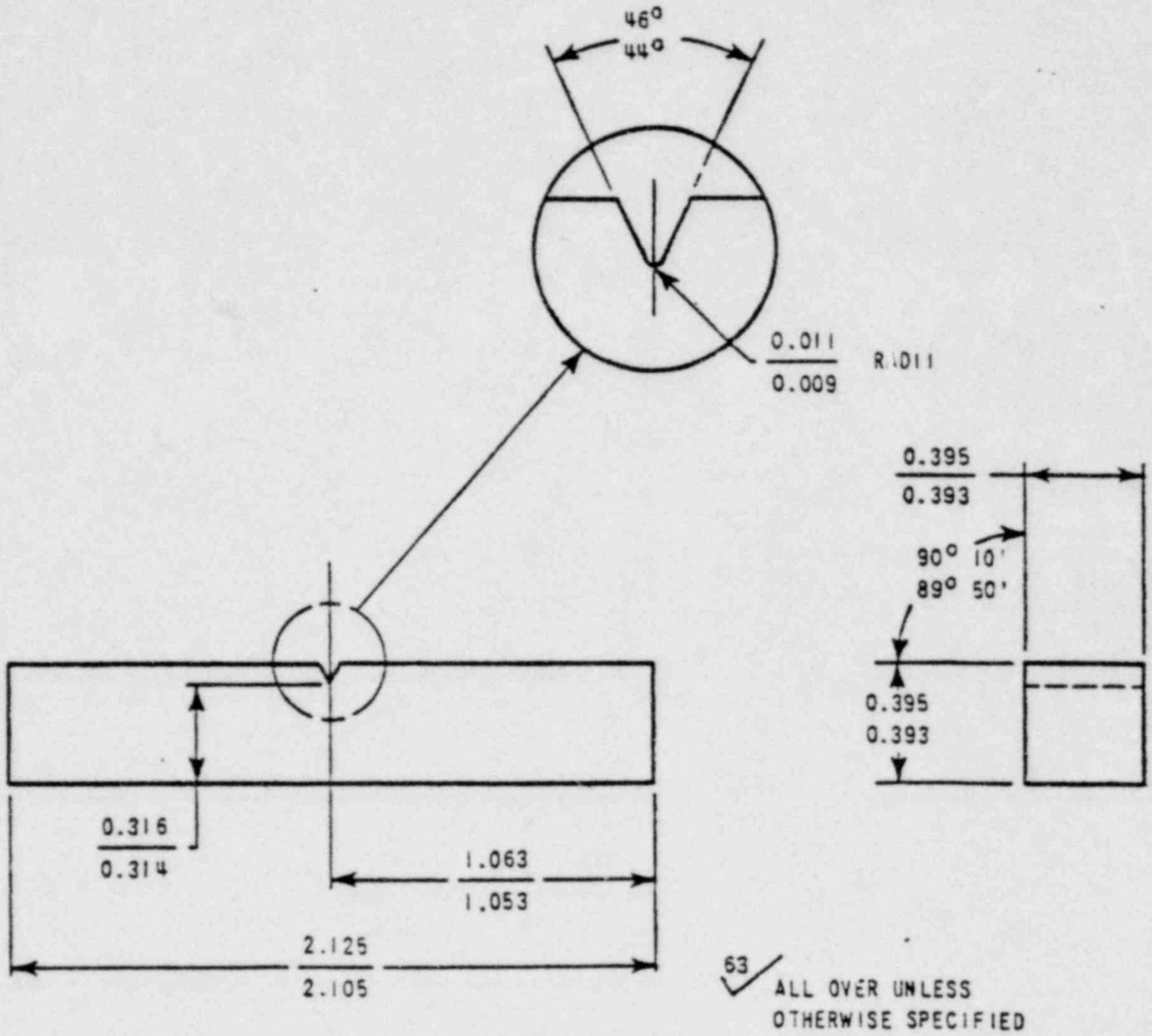

Figure 2-1. Charpy V-Notch Impact Specimen

NOTES:

1. LATHE CENTERS REQUIRED
2. 125 ALL OVER UNLESS OTHERWISE SPECIFIED
3. "B" DIA IS TO BE ACTUAL "A" DIA + 0.002 TO 0.005 TAPERING TO "A" AT THE CENTER

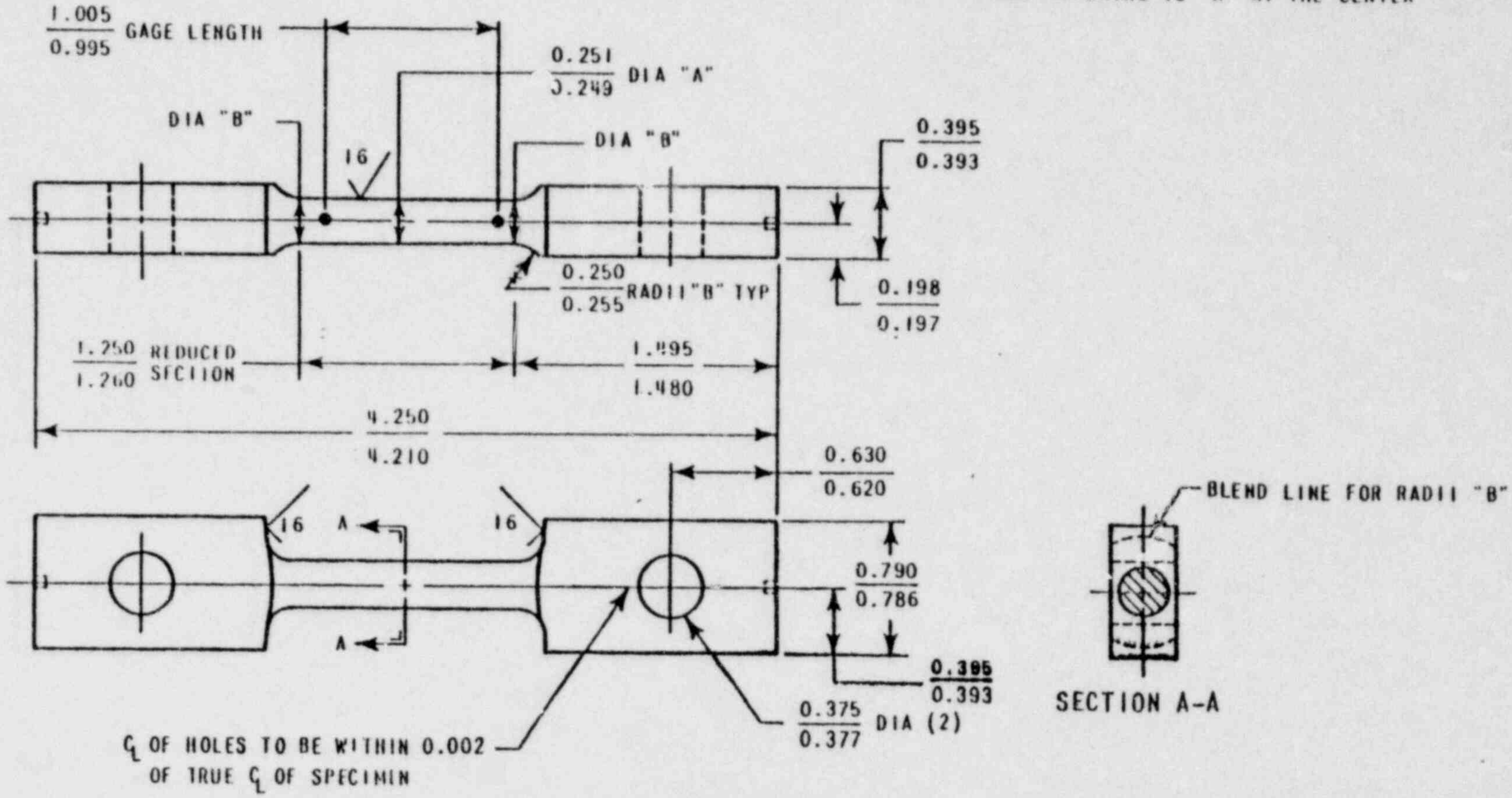

Figure 2-2. Tensile Specimen

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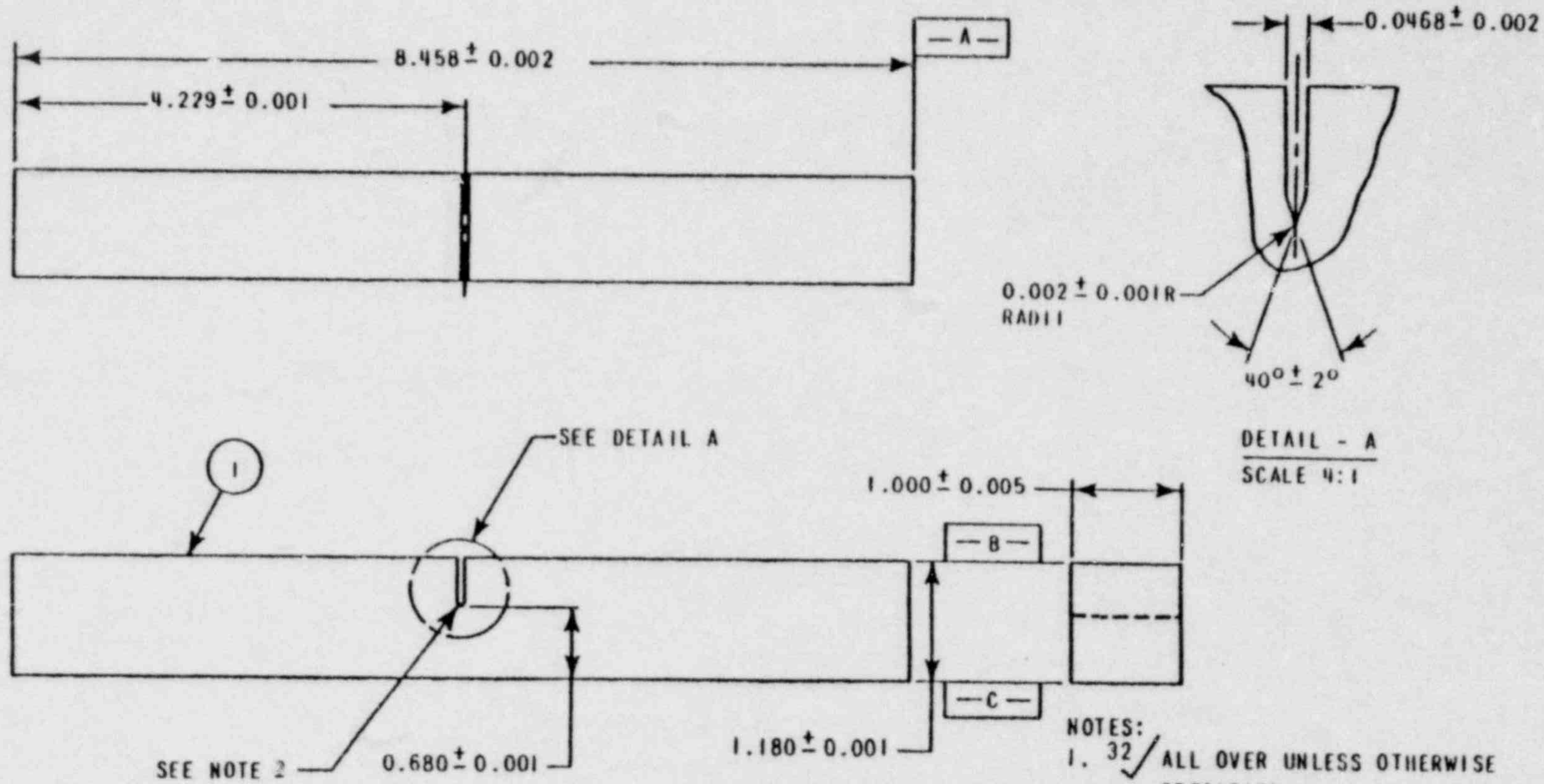

NOTES:
1. 32/ ALL OVER UNLESS OTHERWISE SPECIFIED
2. NOTCH DEPTH TO BE EXTENDED BY 0.080 ± 0.020 BY FATIGUE CRACKING
3. SLOT TO BE PARALLEL TO SURFACE "A" WITHIN 0.005 TOTAL
4. SLOT TO BE PERPENDICULAR TO SURFACES B&C WITHIN 0.005
5. DIM. MARKED THUS ' MUST NOT HAVE A T.R.O. EXCEEDING ± 0.001 ON OPPOSITE FACE

10.062

2-6. 1/2 Compact Tension Specimens

Compact tension test specimens from plate B6919-1 were machined in both the transverse and longitudinal orientations. This was done to obtain fracture toughness data both normal and parallel to the rolling direction of the plate, and to initiate propagation of the simulated crack in both orientations. Compact tension test specimens from the weld metal were machined normal to the weld direction with the notch oriented in the direction of the weld. All specimens were fatigue precracked according to ASTM E399 (see figure 2-4).

2-7. DOSIMETERS

Six capsules of the type shown in figure 2-5 contain dosimeters of pure copper, iron, nickel and aluminum 0.15 wt percent cobalt wire (cadmium-shielded and unshielded) and Cd-shielded Np^{237} and U^{238} which will measure the integrated flux at specific neutron energy levels.

2-8. THERMAL MONITORS

The capsules contain two low-melting-point eutectic alloys to define more accurately the maximum temperature attained by the test specimens during irradiation. The thermal monitors will be sealed in Pyrex tubes and then inserted in spacers located as shown in figure 2-5. The two eutectic alloys and their melting points are the following:

2.5% Ag, 97.5% Pb	Melting point 579°F
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting point 590°F

2-9. CAPSULE LOADING

The six test capsules coded U, V, W, X, Y, and Z are positioned in the reactor between the neutron shielding pads and the vessel wall at the locations shown in figure 2-5. Each capsule contains 60 Charpy V-notch specimens, nine tensile specimens, twelve 1/2T compact tension specimens and one bend bar.

The relationship of the test material to the type and number of specimens in each capsule is shown in table 2-1.

Dosimeters of aluminum 0.15 percent cobalt, cadmium-shielded aluminum 0.15 percent cobalt, pure copper, iron and nickel wires are secured in holes drilled in spacers located at capsule positions shown in figure 2-5. Each capsule also contains a dosimeter block (figure 2-6) which will be located at the center of the capsule. Two cadmium-oxide-shielded capsules, each containing isotopes of either U^{238} or Np^{237}, are located in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the U^{238} and Np^{237} and their activation products. Each dosimeter block contains approximately

12 milligrams of U^{238} and 17 milligrams of Np^{237} held in a 3/8-inch-long by 1/4-inch-OD sealed brass tube and stainless steel tube, respectively. Each tube was placed in a 1/2-inch-diameter hole in the dosimeter block (one U^{238} and one Np^{237} tube per block), and the around the tube was filled with cadmium oxide. After placement of this material, each ho was blocked with two 1/16-inch-thick aluminum spacer discs and an outer 1/8-inch-thick s cover disc welded in place.

The numbering system for the capsule specimens and their locations are shown in figure 2

2-10. SPECIMEN CAPSULE

The specimens are seal-welded into a square capsule of austenitic stainless to prevent corro of specimen surfaces during irradiation. The capsules were hydrostatically tested in demine ized water to collapse the capsule on the specimens optimizing thermal conductivity betw the specimens and the reactor coolant. The capsules were helium leak tested as a final ins procedure. Fabrication details and testing procedures are listed in figure 2-5.

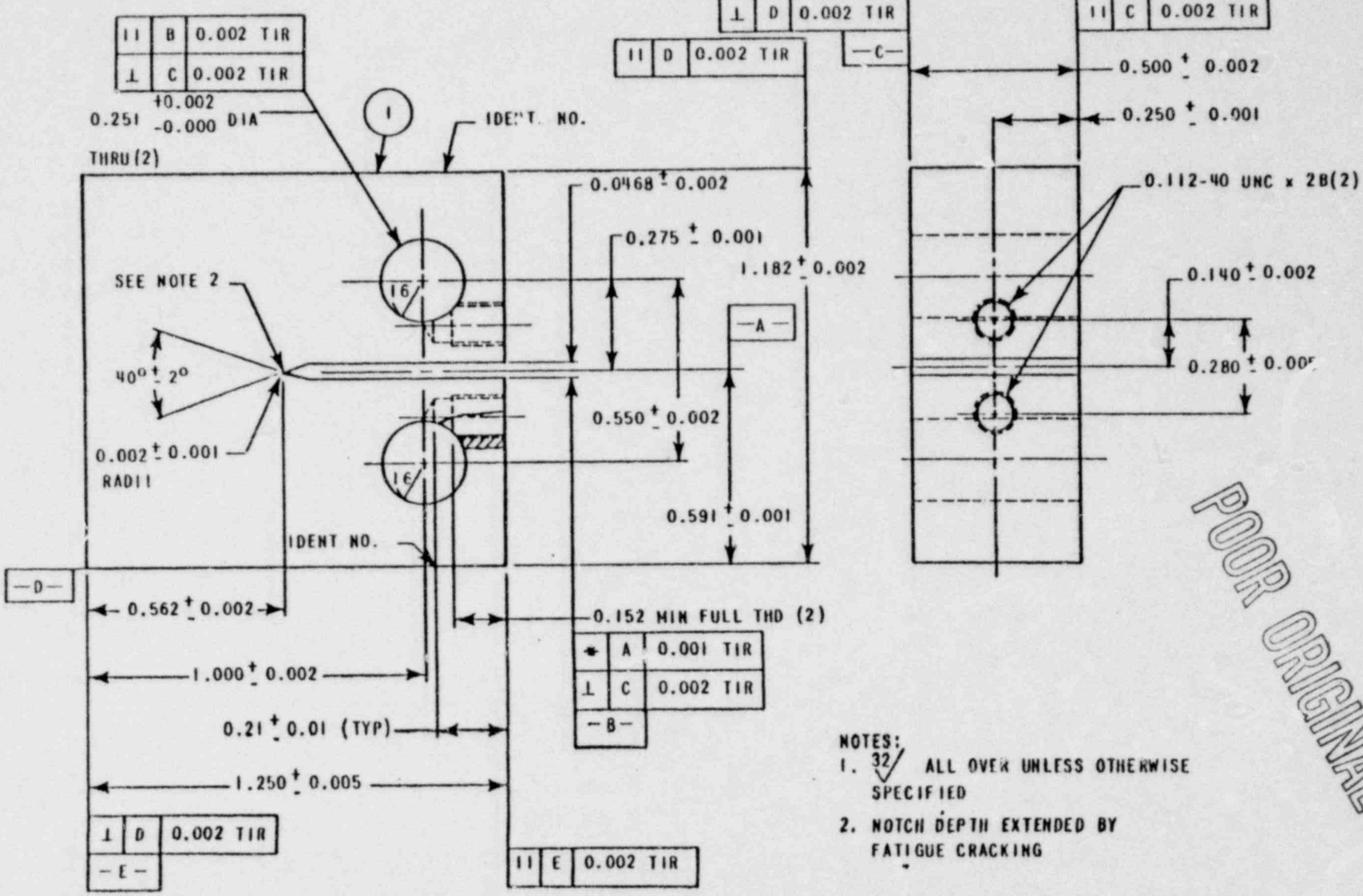

Figure 2-4. Compact Tension Specimen

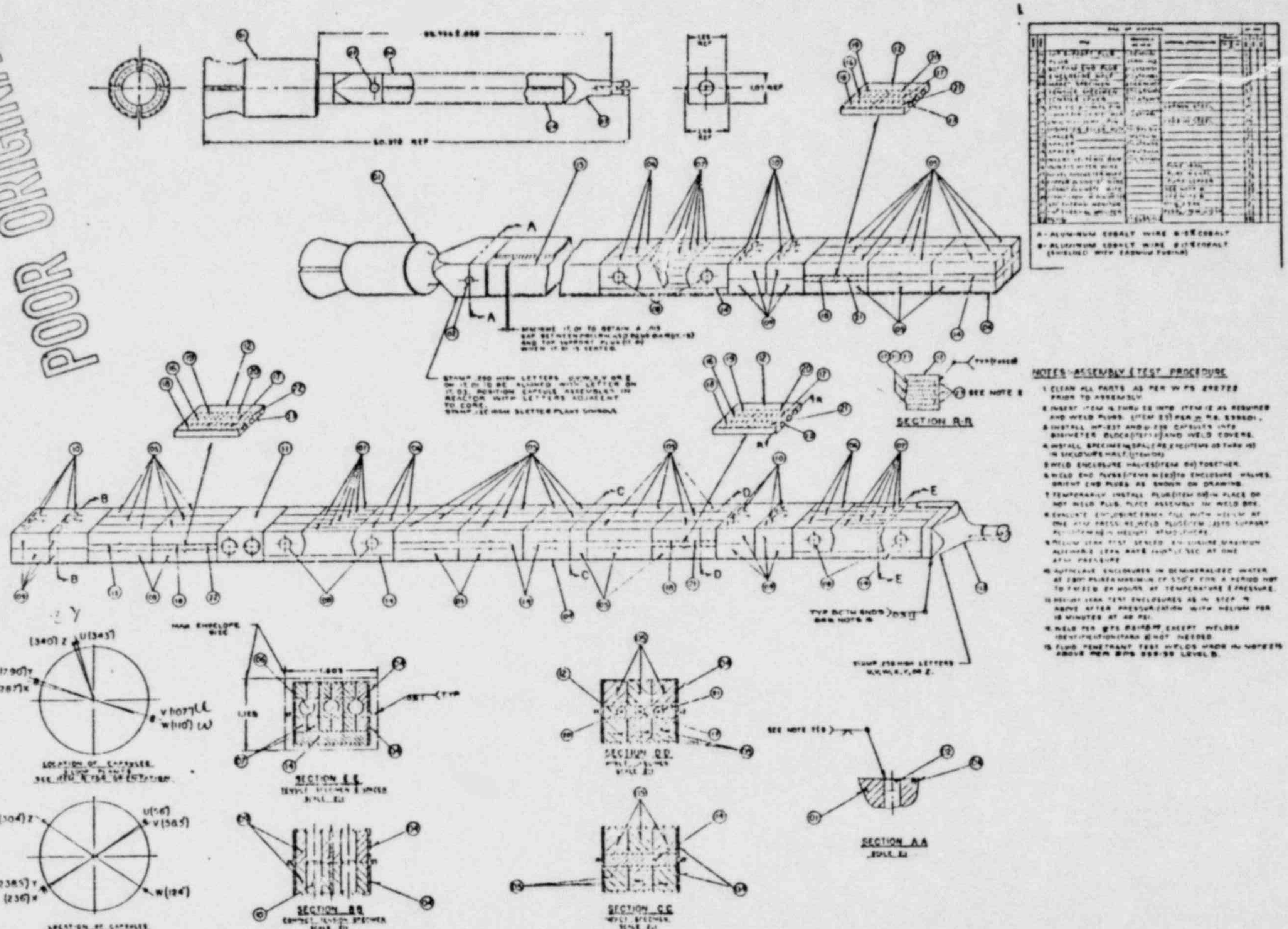

Figure 2-5. Irradiation Capsule Assembly

ITEM	TITLE	MATERIAL SPECIFICATION	NO REQ'D
1	BLOCK		1
2	COVER		2
3	SPACER		4
4	NEPTUNIUM ^{237} SEALED CAPSULE (0.250 OD x 0.375 LG)	STAINLESS STEEL	1
5	URANIUM ^{238} SEALED CAPSULE (0.250 OD x 0.375 LG)	BRASS	1
6	CADMIUM OXIDE		AS REQ'D

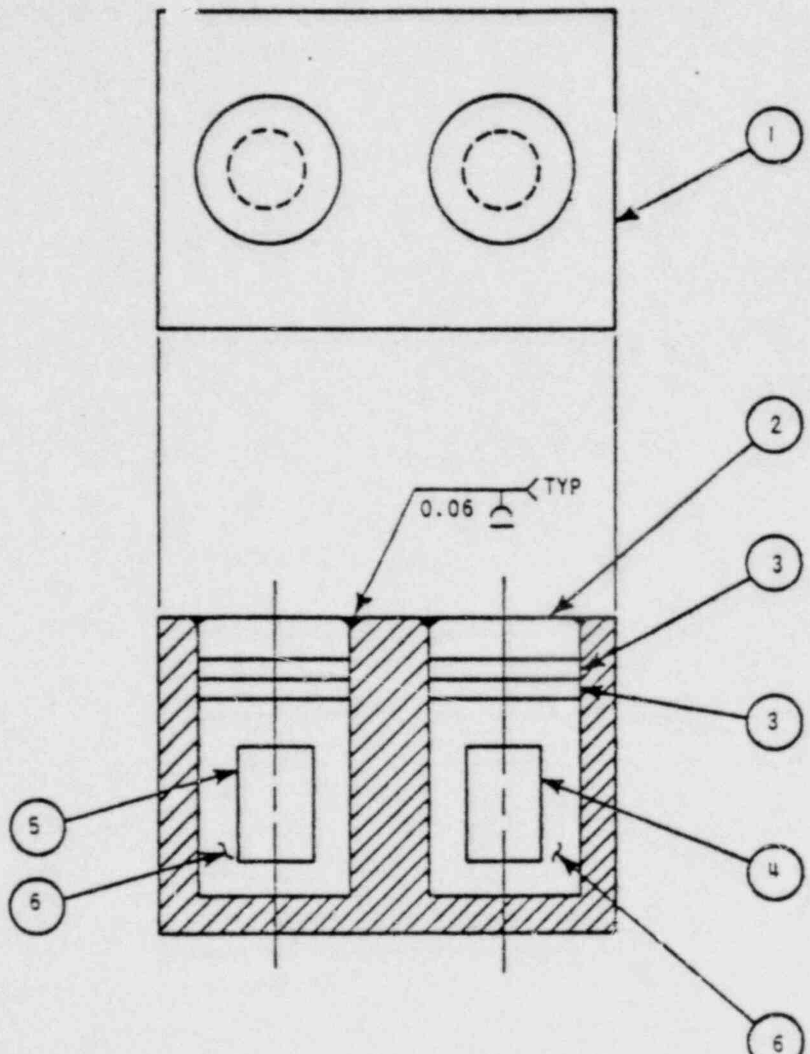

Figure 2-6. Dosimeter Block Assembly

NOTES:

AT - PLATE B6919-1 TRANSVERSE
AL - PLATE B6919-1 LONGITUDINAL
AW - WELD METAL
AH - HEAT AFFECTED ZONE

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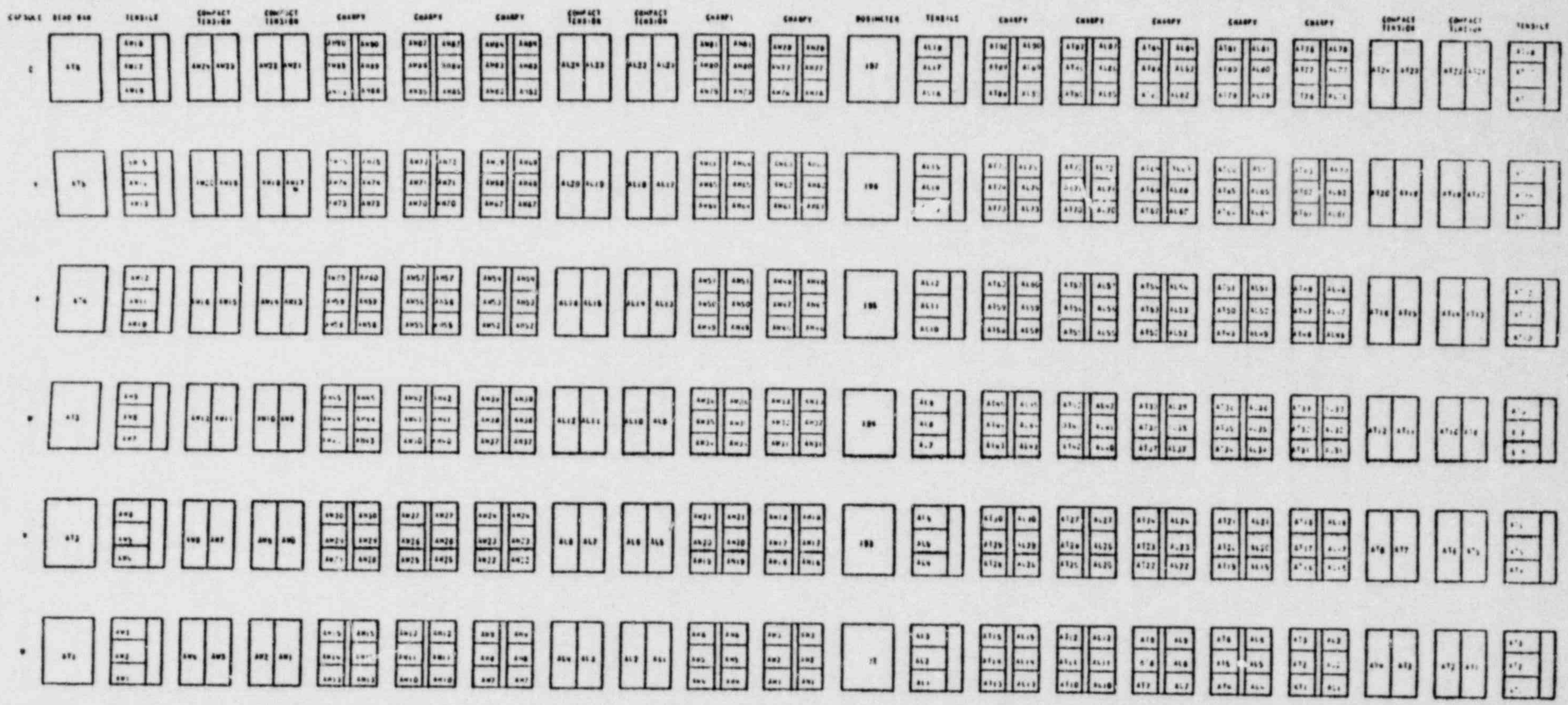

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Figure 2-7. Location of Specimens In The Reactor Surveillance Test Capsule

TABLE 2-1

TYPE AND NUMBER OF SPECIMENS IN THE JOSEPH M. FARLEY UNIT NO. 1 SURVEILLANCE TEST CAPSULES

Material	Capsules U, V, W, X, Y, and Z			
	Charpy	Tensile	CT	Bend Bar
Plate B6919-1 (Longitudinal)	15	3	4	–
(Transverse)	15	3	4	1
Weld Metal	15	3	4	–
HAZ	15	–	–	–

SECTION 3
PREIRRADIATION TESTING

3-1. CHARPY V-NOTCH TESTS

Charpy V-notch impact tests were performed on material from the vessel lower shell plate B6919-1 at various test temperatures from -50 to 210°F to obtain a Charpy V-notch transition curve in both the longitudinal and transverse orientations (tables 3-1 and 3-2 and figures 3-1 and 3-2). Tests were also performed on weld metal and HAZ metal at various temperatures from -150 to 210°F. The results are reported in tables 3-3 and 3-4 and figures 3-3 and 3-4.

The specimens were tested on a Sontag SI-1 impact machine which is inspected and calibrated every 12 months. Charpy V-notch impact specimens of known energy values, supplied by the Watertown Arsenal, are used for the calibration.

3-2. TENSILE TESTS

Table 3-5 and figures 3-5, 3-6, and 3-7 give results of tensile tests performed on material from the vessel lower shell plate B6919-1 and from the weld metal. Specimens from the shell plate were tested at room temperature, 300°F and 550°F in both longitudinal and transverse directions.

The Instron TT-C tensile testing machine was set up with the standard Instron gripping devices. A Baldwin-Lima-Hamilton Class B-1 extensometer and chart recorder provided a stress-strain curve for each specimen. The chart recorder was calibrated to the Class B-1 extensometer. The measurement and control of speeds in the tests conformed to ASTM A370-68 (Mechanical Testing of Steel Products). The Instron TT-C and the Baldwin-Lima-Hamilton extensometer are certified as traceable to the National Bureau of Standards. A typical stress-strain curve is shown in figure 3-8.

3-3. DROPWEIGHT TESTS

The nil-ductility transition temperature (NDTT) was determined for plate B6919-1, the core region weld metal and the heat-affected zone by dropweight tests (ASTM E-208) performed at Combustion Engineering. The following results were obtained:

Material	NDTT (^{o}F)
Plate B6919-1	-20
Weld Metal	-60
HAZ	-10

TABLE 3-1

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B6919-1 (LONGITUDINAL DIRECTION)

Test Temp (°F)	Impact Energy (ft lb)	Shear%	Lateral Expansion (mils)
-50	26	14	18
-50	12	10	8
-50	11	14	8
0	59	30	44
0	48	27	37
0	56	27	38
40	80	50	62
40	52	35	44
40	68	42	53
80	107	80	70
80	100	80	73
80	106	80	71
130	135	100	80
130	140	100	90
130	145	100	88
210	131	100	83
210	129	100	85
210	142	100	85

TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B6919-1 (TRANSVERSE DIRECTION)

Test Temp (°F)	Impact Energy (ft lb)	Shear%	Lateral Expansion (mils)
-40	12	14	7
-40	25	14	17
-40	29	20	18
0	30	25	21
0	27	25	24
0	35	25	26
40	26	37	26
40	37	29	30
40	44	52	37
RT	60	55	50
RT	53	64	45
RT	53	50	46
110	75	77	56
110	69	79	53
110	80	90	52
210	92	100	71
210	91	100	72
210	89	100	68

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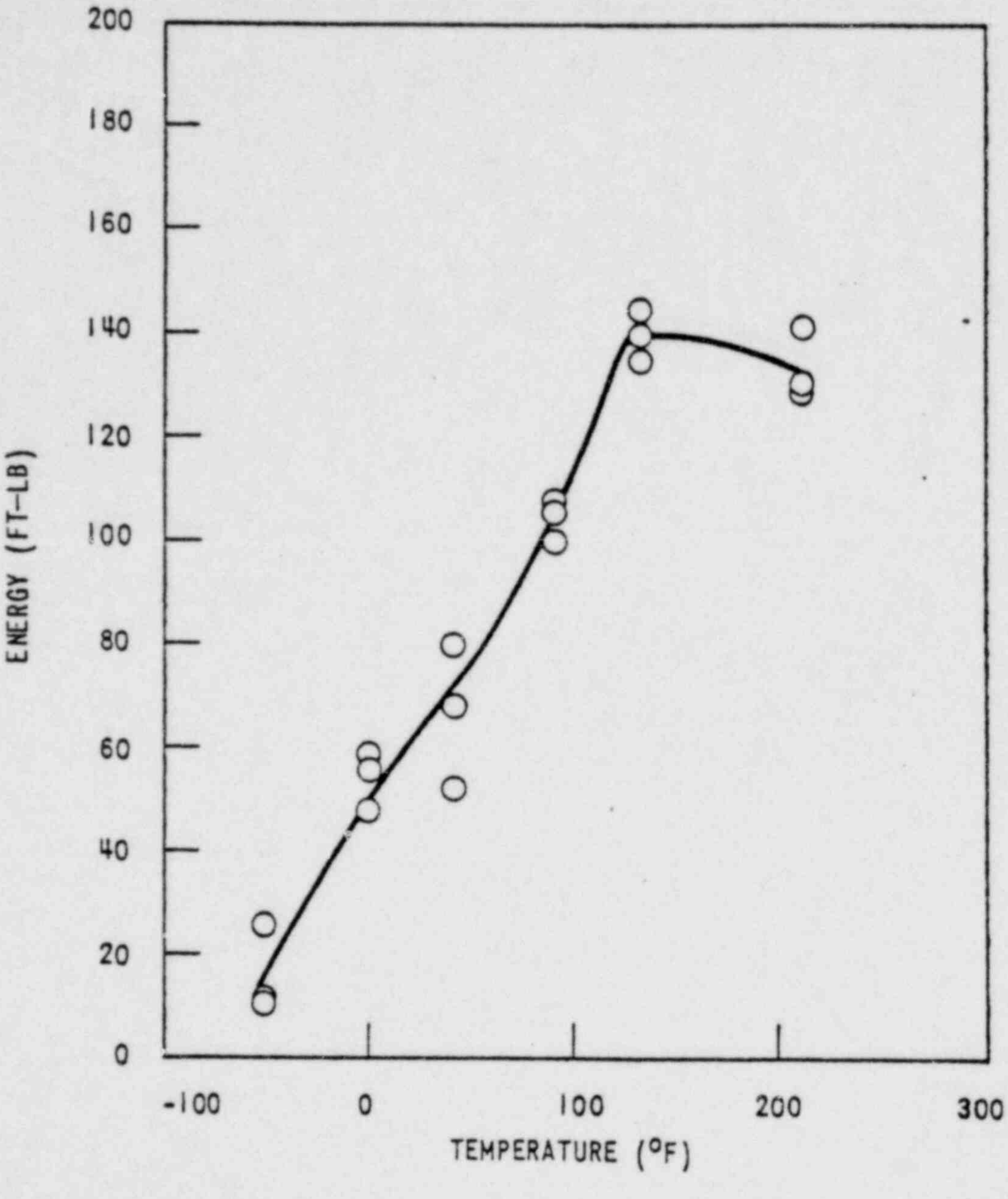

Figure 3-1. Preirradiation Charpy V-Notch Impact Energy For The Joseph M. Farley Unit No. 1 Reactor Pressure Lower Shell Plate B6919-1. (Longitudinal Orientation)

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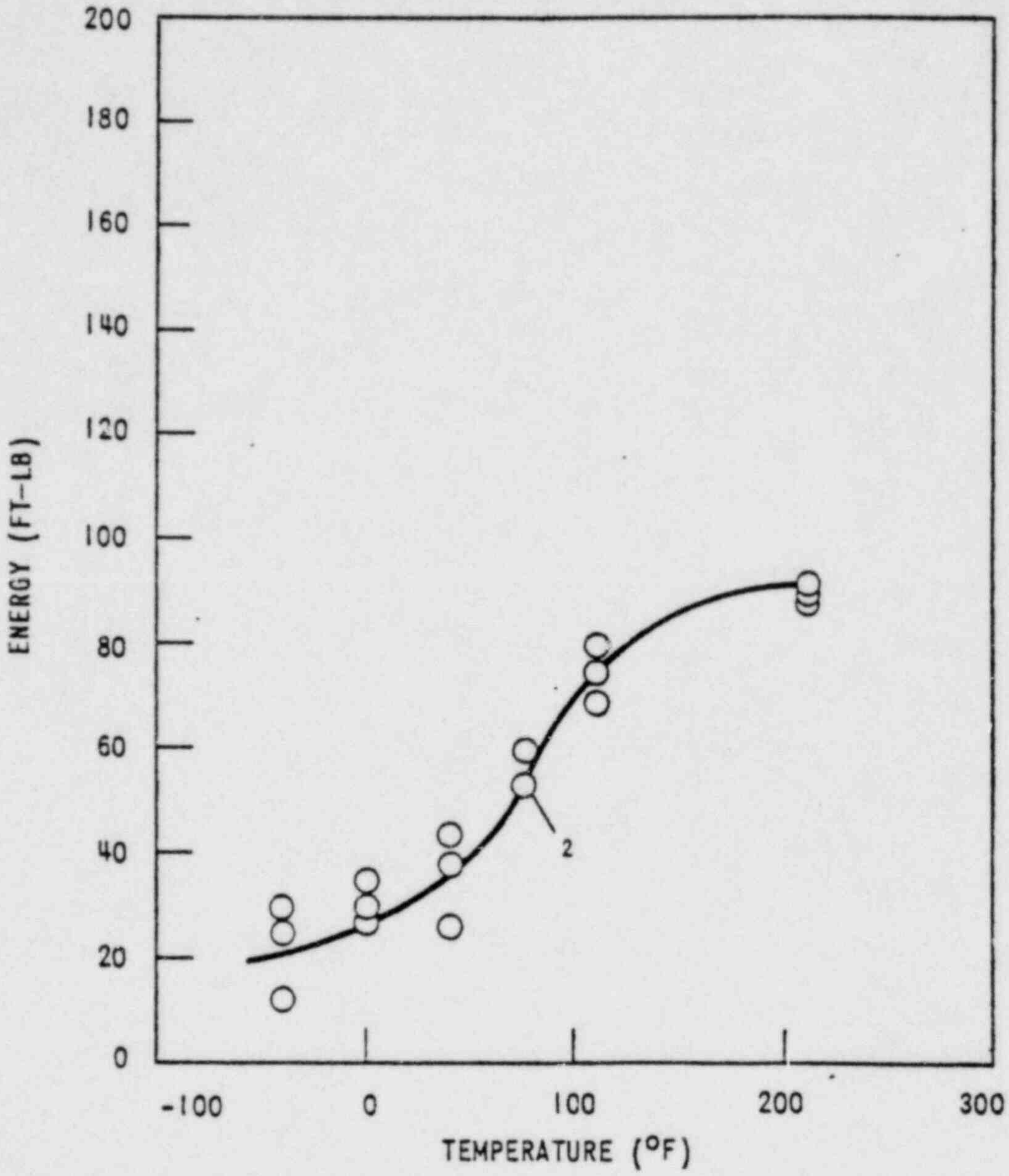

Figure 3-2. Preirradiation Charpy V-Notch Impact Energy For The Joseph M. Farley Unit No. 1 Reactor Pressure Vessel Lower Shell Plate B6919-1. (Transverse Orientation)

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
-100	5	15	1
-100	14	25	11
-100	18	20	11
- 40	53	43	43
- 40	60	32	44
- 40	75	50	55
10	79	65	54
10	86	73	63
10	80	65	58
72	117	100	82
72	123	100	80
72	113	100	79
150	151	100	90
150	144	100	89
150	118	100	81
210	138	100	88
210	159	100	85
210	151	100	85

TABLE 3-4

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL CORE REGION WELD-HEAT-AFFECTED ZONE MATERIAL

Test Temp (°F)	Impact Energy (ft-lb)	Shear (%)	Lateral Expansion (mils)
-150	15	18	7
-150	11	18	6
-150	58	45	29
-100	103	55	54
-100	33	32	19
-100	67	45	40
- 75	101	65	62
- 75	110	65	57
- 20	122	80	72
- 20	120	80	71
- 20	84	65	52
50	141	100	83
50	142	100	85
75	150	100	74
210	132	100	85
210	170	100	83
210	163	100	85

10306-5

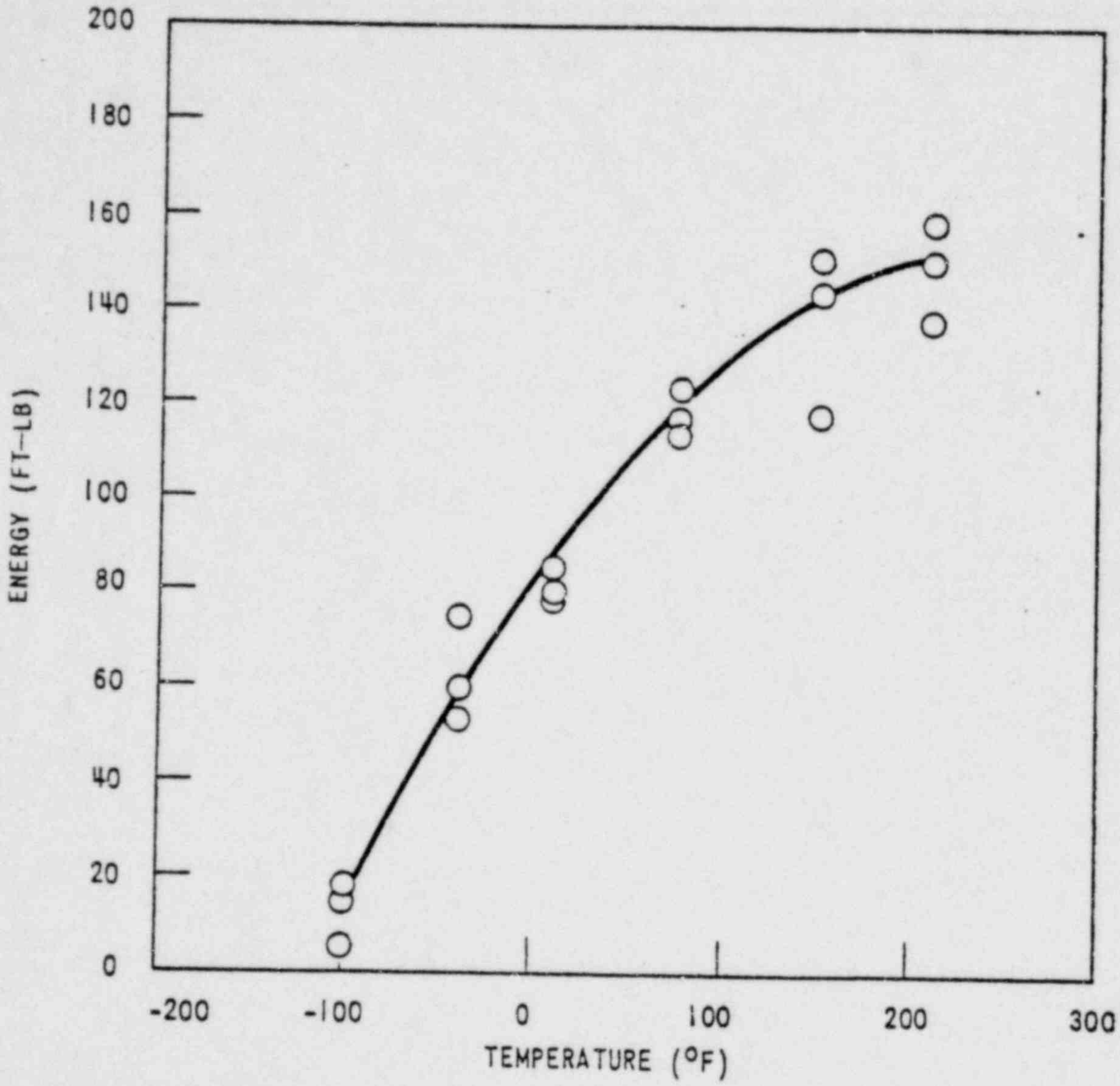

Figure 3-3. Preirradiation Charpy V-Notch Impact Energy For The Joseph M. Farley Unit ! o. 1 Reactor Pressure Vessel Core Region Weld Metal

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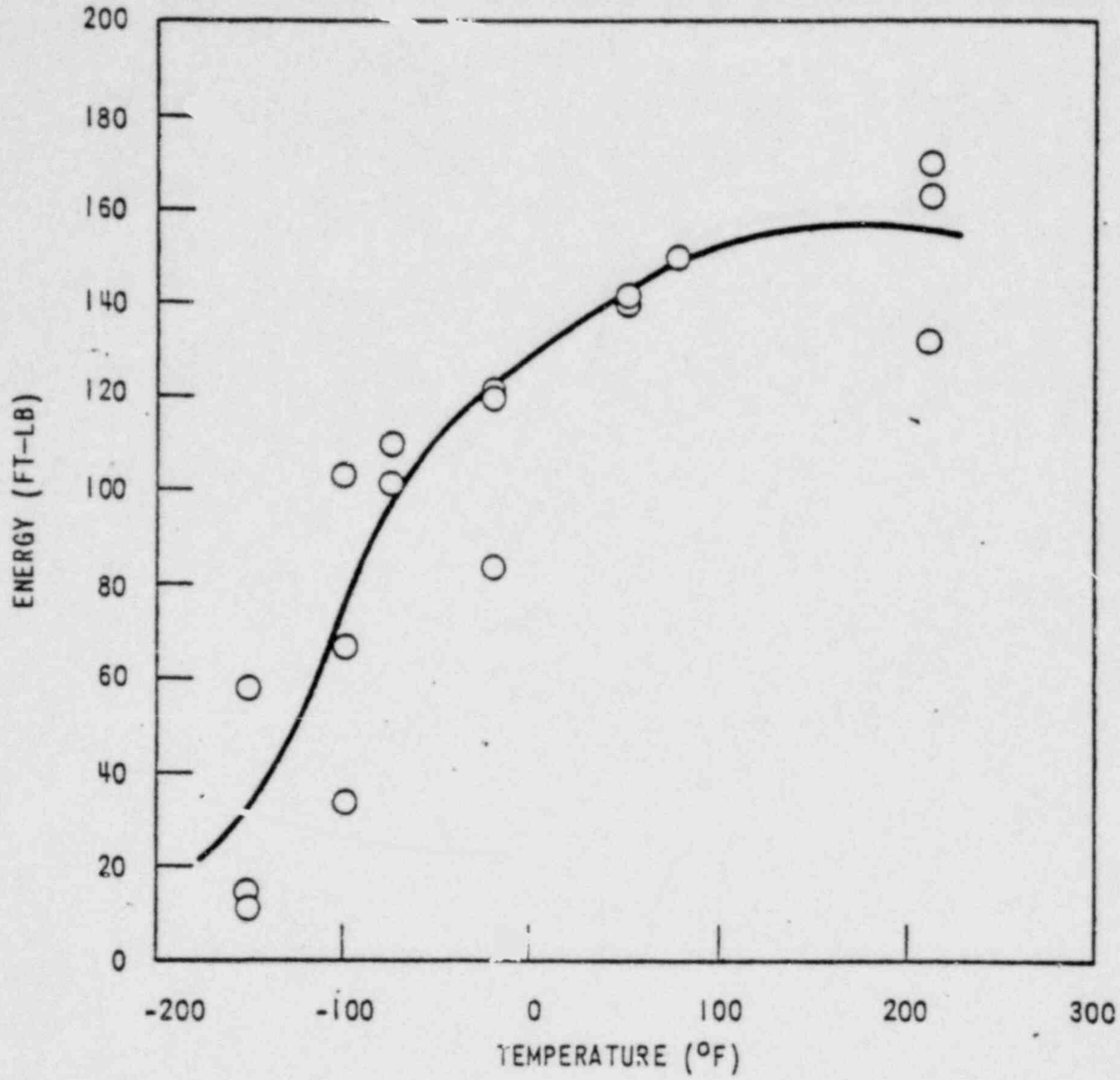

Figure 3-4. Preirradiation Charpy V-Notch Impact Energy For The Joseph M. Farley Unit No. 1 Reactor Pressure Vessel Core Region Weld Heat-Affected Zone Material

TABLE 3-5

PREIRRADIATION TENSILE PROPERTIES FOR THE JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B6919-1 AND CORE REGION WELD MATERIAL

Vessel Mat.	Test Temp(°F)	0.2% Yield Strength(psi)	UTS(psi)	Fracture Load (lb)	Fracture Stress (psi)	Uniform Elon. (%)	Total Elon. (%)	Red. in Area (%)
Plate B6919-1	(75°)	64,600	84,700	2550	180,850	15.8	27.5	71.5
(Longitudinal Direction)	(75°)	64,100	84,800	2550	178,200	15.8	28.8	70.9
	300	58,300	78,000	2500	167,200	13.6	24.5	69.6
	300	59,000	77,850	2400	170,200	13.0	24.3	71.5
	550	54,600	80,450	2600	159,700	11.5	23.6	67.1
	550	54,800	80,500	2600	165,600	13.7	24.1	68.6
Plate B6919-1	(75°)	65,150	86,000	2850	157,100	15.7	25.0	63.1
(Transverse Direction)	(75°)	68,350	90,700	3000	172,100	13.2	22.5	64.8
	300	58,600	78,300	3000	143,800	12.7	20.9	57.8
	300	59,350	78,800	2750	153,600	12.4	21.3	63.7
	550	58,850	85,200	2975	157,750	13.0	21.2	61.7
	550	60,800	87,450	3150	158,700	12.3	20.7	59.9
Weld Metal	(75°)	78,250	90,450	2720	205,000	12.8	25.3	73.0
	(75°)	77,800	90,250	2710	198,000	11.1	21.9	72.1
	300	69,900	83,400	2625	183,450	10.3	20.8	71.1
	300	70,900	83,650	2825	186,200	12.0	20.8	71.4
	550	67,400	86,200	2775	168,100	11.2	21.1	66.6
	550	69,100	88,200	3050	177,300	10.6	20.2	55.0

10306-3

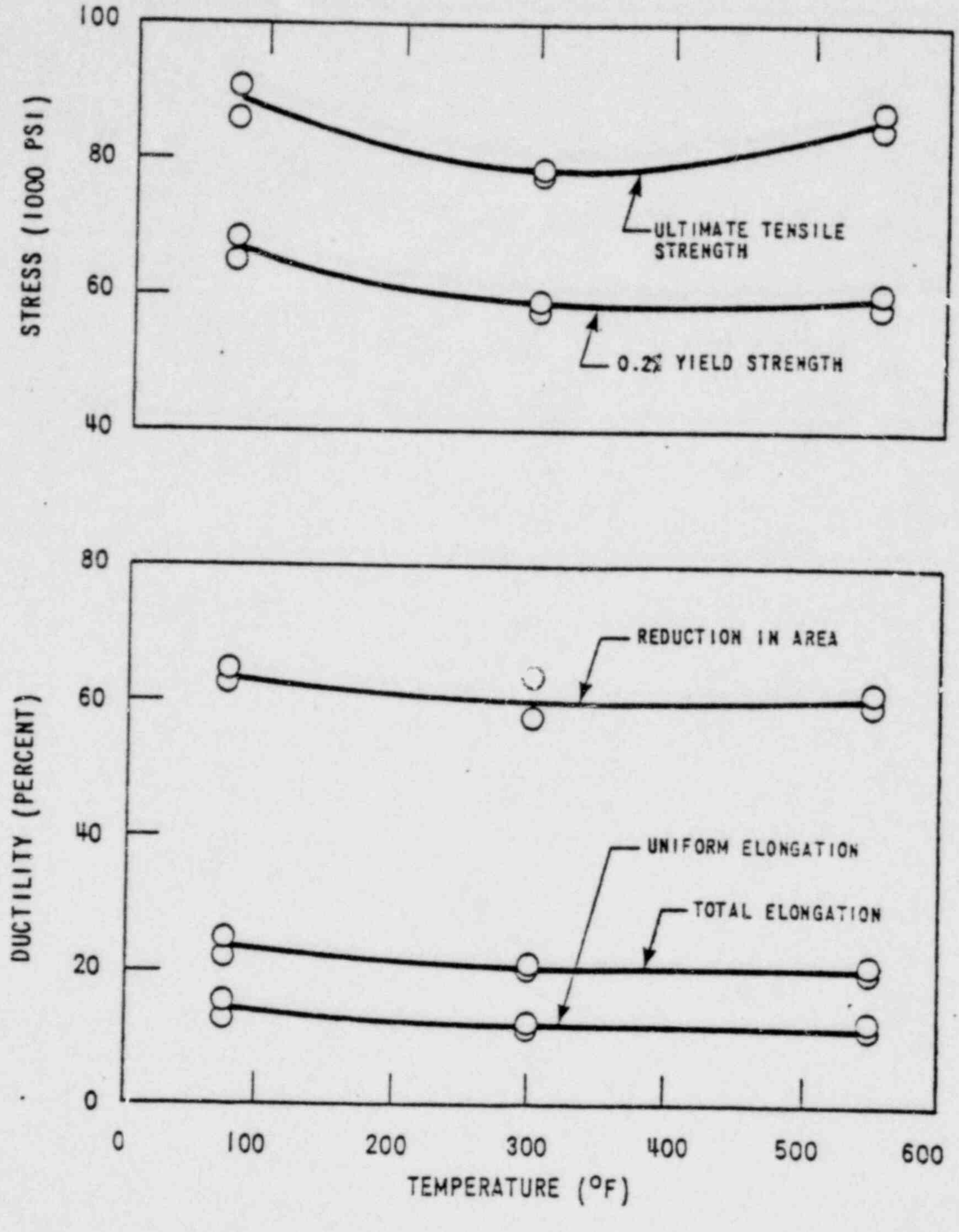

Figure 3-5. Preirradiation Tensile Properties For The Joseph M. Farley Unit No. 1 Reactor Pressure Vessel Lower Shell Plate B6919-1 (Transverse Direction)

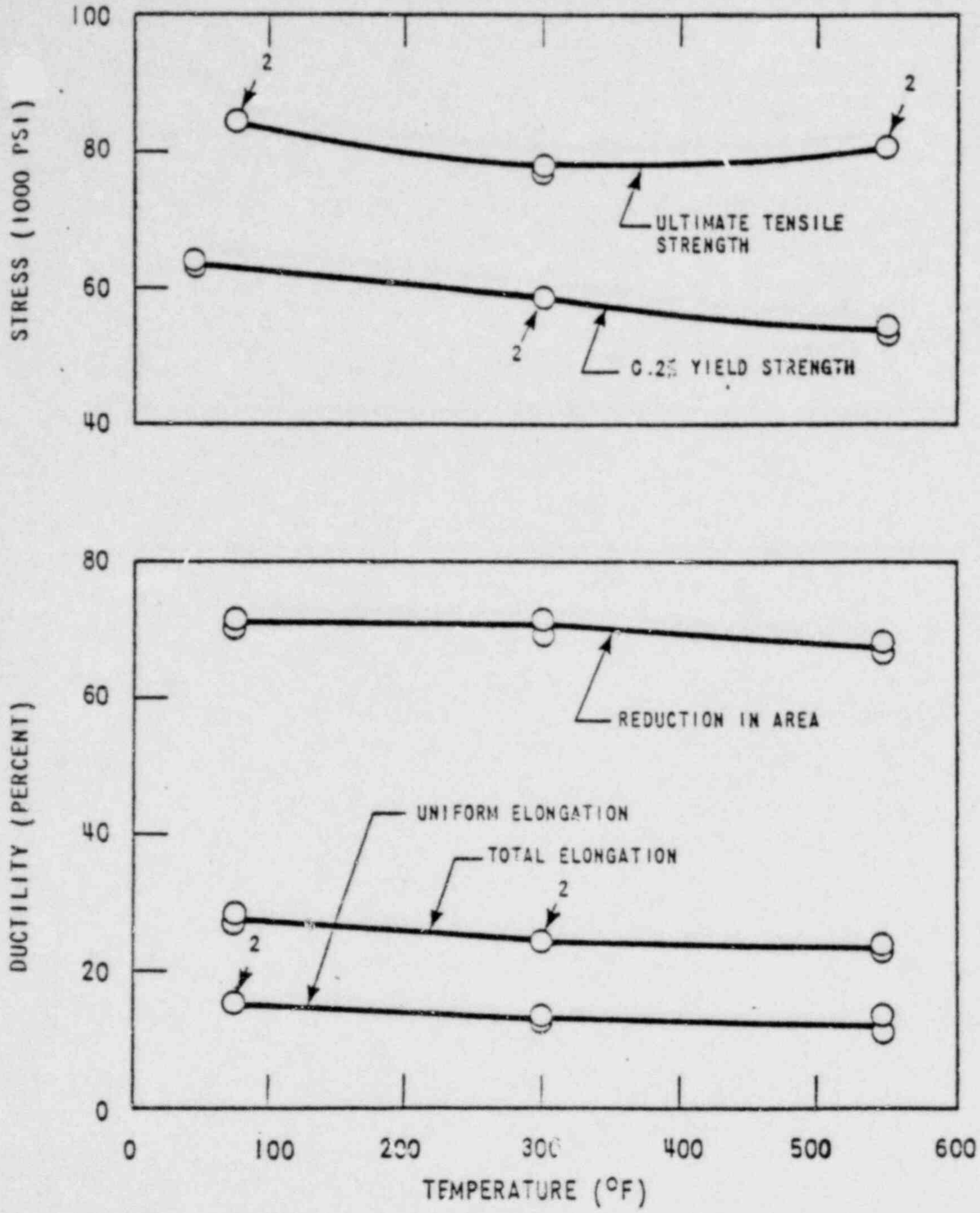

Figure 3-6. Preirradiation Tensile Properties For The Joseph M. Farley Unit No. 1 Reactor Pressure Vessel Lower Shell Plate B6919-1. (Longitudinal Direction)

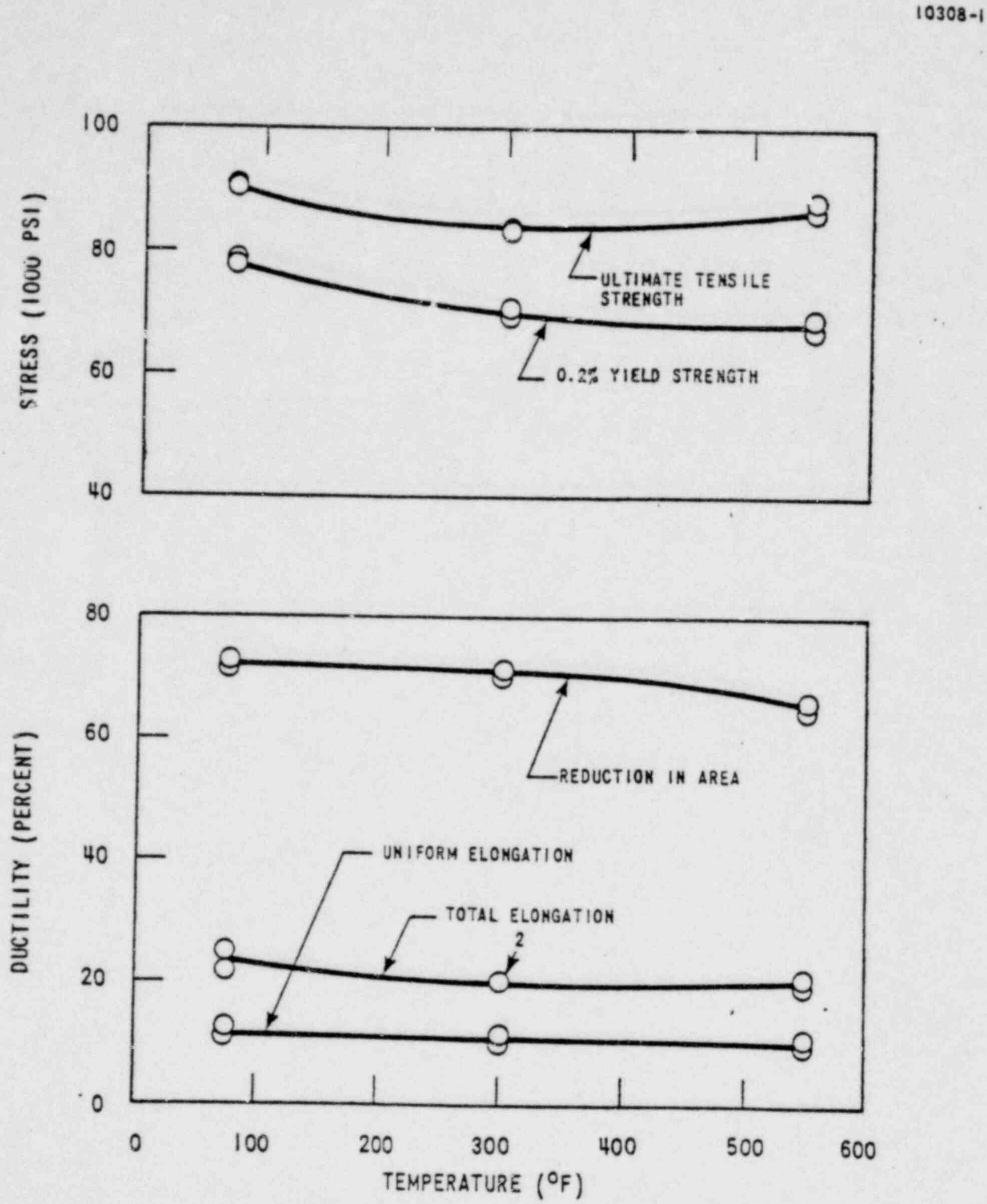

Figure 3-7. Preirradiation Tensile Properties For The Joseph M. Farley Unit No. 1 Reactor Pressure Vessel Core Region Weld Metal

Figure 3-8. Typical Stress/Strain Curve of Tensile Test

SECTION 4
POSTIRRADIATION TESTING

4-1. CAPSULE REMOVAL

Specimen capsules will be removed from the reactor only during normal refueling periods. The recommended capsule removal schedule is as follows:

Capsule Identification	Multiplying Factor By Which the Capsule Leads Vessel Maximum Exposure	Removal Time*
U	2.6	End of First Core Cycle
W	2.0	10 Years
Y	2.0	20 Years
Z	2.0	30 Years
V	2.6	Standby
X	2.6	Standby

Each specimen capsule, removed after radiation exposure, will be transferred to a post-irradiation test facility for capsule disassembly and testing of all specimens.

4-2. CHARPY V-NOTCH IMPACT TESTS

The testing of the Charpy impact specimens from the lower shell plate, the weld metal, and HAZ metal in each capsule can be done singly at approximately five different temperatures. The extra specimens can be used to run duplicate tests at temperatures of interest.

The initial Charpy specimen from the first capsule should be tested at room temperature. The test value for this temperature should be compared with preirradiation test data. The temperatures for the remaining specimens should then be appropriately higher or lower. For succeeding tests after longer irradiation periods, the test temperature in each case should be chosen in the light of results from the previous capsule.

4-3. TENSILE TESTS

The tensile and fracture toughness specimens for each of the irradiated materials should be tested at room temperature, 300°F, and 550°F.

4-4. FRACTURE TOUGHNESS TESTS ON 1/2T COMPACT TENSION SPECIMENS AND BEND BAR SPECIMENS

In light of current requirements of 10 CFR, Part 50, ASME Code, appendix G, 1/2T compact tension (CT) specimens should be tested dynamically to adequately characterize the fracture toughness properties of the reactor vessel up to the initiation of the fracture toughness upper shelf. The CT specimens for each of the irradiated materials should be tested in accordance with ASMT E399-74 with appropriate modifications necessary for dynamic tests. Testing dynamically in the fracture toughness ductile-to-brittle transition region and at upper shelf initiation temperatures results not only in lower bound data but also provides an opportunity for obtaining valid [1] fracture toughness data up to the onset of upper shelf. This results from nonlinear cleavage behavior which occurs only in dynamic testing at these temperatures. The load-displacement curve exhibits a clear drop in load at the onset of crack initiation, thereby eliminating any possible doubt as to the start of crack initiation, as is the case in static loading conditions at these temperatures. Recommended test temperatures are equal to or lower than those characteristic of the upper fracture toughness shelf initiation temperature.

Analysis should be performed using the J-Integral or Equivalent Energy Concept [1,2]. Testing at temperatures characteristic of the fracture toughness upper shelf is not suggested due to the uncertainty of the point of crack initiation even when dynamic testing is performed. At these temperatures, static J_{Ic} testing appears to be most indicative of conservative upper shelf fracture toughness properties. Research in this area is currently being conducted by Westinghouse Research and Development Laboratory, ASTM E24, NRC, and others. Use of this technique will be further evaluated as it applies to surveillance specimen testing. The precracked bend bars will be used to obtain additional toughness data at a temperature indicated by the toughness results of the compact tension testing.

4-5. POSTIRRADIATION TEST EQUIPMENT

The following minimum equipment is required for the postirradiation testing operations:

- Milling machine or special cutoff wheel for opening capsules, dosimeter blocks and spacers

1. "Fracture Toughness," American Society for Testing and Materials, STP 514, September, 1972.
2. T. R. Mager, "Experimental Verification of Lower Bound K_{Ic} Values Utilizing the Equivalent Energy Concept," HSST Semi-Annual Information Meeting, Paper No. 23, April, 1972.

- Hot-cell tensile testing machine with:

  1) pin-type adapter for testing tensile specimens
  2) three-point loading assembly for testing the bend bar specimens

- Hot-cell dynamic CT testing machine with clevis and appropriate measuring equipment associated with dynamic testing

- Hot-cell Charpy impact testing machine

- Sodium-iodide scintillation detector and pulse-height analyzer for gamma counting of the specific activities of the dosimeters

APPENDIX A

JOSEPH M. FARLEY UNIT NO. 1 REACTOR PRESSURE VESSEL SURVEILLANCE MATERIAL

For the Reactor Vessel Radiation Surveillance Program, Combustion Engineering supplied Westinghouse with sections of SA533 Grade B Class 1 plate used in the core region of the Joseph M. Farley Unit No. 1 Reactor Pressure Vessel, specifically, from the 9-inch lower shell plate B6919-1. Also supplied was a weldment made from sections of plate B6919-1 and adjoining intermediate shell plate B6903-2, using weld wire representative of that used in the original fabrication. The plates were produced by Lukens Steel Company. The heat treatment history and chemical analysis of the pressure vessel surveillance material is as follows:

	Heat Treatment
Lower Shell Plate B6919-1	1550°/1650°F - 4 hr - Water Quenched 1225°F ± 25°F 4 hr 1150°F ± 25°F 40 hr - Furnace Cooled to 600°F
Weldment	1150°F ± 25°F 16 hr - Furnace Cooled

TABLE A-1

CHEMICAL COMPOSITION (WT-%)

Element*	Plate B6919-1		Weld Metal
	Combustion Engineering Analysis	Westinghouse Analysis	Westinghouse Analysis
C	0.20	–	0.13
S	0.015	0.013	0.009
N_2	–	0.003	0.005
Co	0.008	0.016	0.018
Cu	0.14	0.10	0.14
Si	0.18	0.28	0.27
Mo	0.56	0.51	0.50
Ni	0.55	0.56	0.19
Mn	1.39	1.40	1.06
Cr	–	0.13	0.063
V	–	<0.001	0.003
P	0.015	0.015	0.016
Sn	–	0.008	0.005
Al	0.025	–	0.009

*All elements not listed are less than 0.010 weight %.