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GEORGIA POWER COMPANY
ALVIN W. VOGTLE UNIT NO. 2
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

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Work Performed Under GBEJ-106

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PREFACE

This report has been technically reviewed and checked by S. E. Yanichko of Structural Materials and Reliability Technology.

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Date: April 14, 1987

ABSTRACT

A pressure vessel steel surveillance program per ASTM E-185-82 has been developed for the Georgia Power Company, Alvin W. Vogtle Unit No.2 to obtain information on the effects of radiation on reactor pressure vessel material under operating conditions. The radiation surveillance program for the Alvin W. Vogtle Unit No. 2 is designed to, and in compliance with, federal government regulations identified in appendix H to 10CFR, part 50 entitled "Reactor Vessel Material Surveillance Program Requirements."

Following is a description of the program, a description of the material involved, the specimen and capsule design and fabrication, and the preirradiation test results.

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SECTION 1

PURPOSE AND SCOPE

The purpose of this program is to monitor radiation effects under actual operating conditions of the core region reactor vessel materials in the Georgia Power Company, Alvin W. Vogtle Unit No. 2, a four-loop, nuclear power plant with a thermal output rating of 3425-megawatts. Evaluation of the radiation effects is based on preirradiation testing of Charpy V-notch, tensile, and dropweight specimens, and postirradiation testing of Charpy V-notch, tensile, and compact 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 dropweight nil-ductility transition temperature (T_{NDT}) 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 T_{NDT} or the temperature 60°F less than the 50 ft lb (or 35-mil) Charpy V-notch temperature, whichever is greater

Therefore

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

and

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

where

R_{NDT} = Reference nil-ductility temperature

T_{NDT} = Nil-ductility transition temperature per ASTM E208

$T_{50(35)}$ = 50 ft lb temperature from Charpy V-notch specimens oriented in the weak direction (or the 35-mil temperature if it is greater)

1. Longitudinal axis of the specimen oriented normal to the major working direction of the plate.

An emperical 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] and thus cause the RT_{NDT} to increase with radiation exposure. The extent of the shift in the impact energy curve, that is, radiation embrittlement, is enhanced by certain chemical elements (such as copper) present in reactor vessel steels.[3,4]

The adjustment in RT_{NDT} with service can be monitored by a surveillance program involving periodic checking of irradiated reactor vessel surveillance specimens. The surveillance program is based on ASTM E185-82 (Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels). Compact fracture mechanics specimens will be used in addition to Charpy V-notch specimens to evaluate the effects of radiation on the fracture toughness of reactor vessel materials.

Postirradiation testing of the Charpy V-notch impact specimens will provide a guide for determining pressure-temperature limits on the plant. Charpy impact test data will determine the shift of the reference temperature[a] with radiation exposure at plant temperatures.

a. The reference temperature as defined by 10CFR Part 50, Appendix G, Section II-E is as follows:
"Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects by adding to RT_{NDT} the temperature shift, measured at the 30 ft lb (41 J) level.

1. Porter, L. F., "Radiation Effects in Steel," in *Materials in Nuclear Applications*, ASTM-STP-276, pp. 147-195, American Society for Testing and Materials, Philadelphia, 1960.
2. Steele, L. E. and Hawthorne, J. R., "New Information on Neutron Embrittlement and Embrittlement Relief of Reactor Pressure Vessel Steels," NRL-6160, August 1964.
3. Potapovs, U. and Hawthorne, J. R., "The Effect of Residual Elements on 550°F Irradiation Response of Selected Pressure Vessel Steels and Weldments," NRL-6803, September 1968.
4. Steele, L. E., "Structure and Composition Effects on Irradiation Sensitivity of Pressure Vessel Steels," in *Irradiation Effects on Stuctural Alloys for Nuclear Reactor Applications*, ASTM-STP-484, pp. 164-175, American Society for Testing and Materials, Philadelphia, 1970.

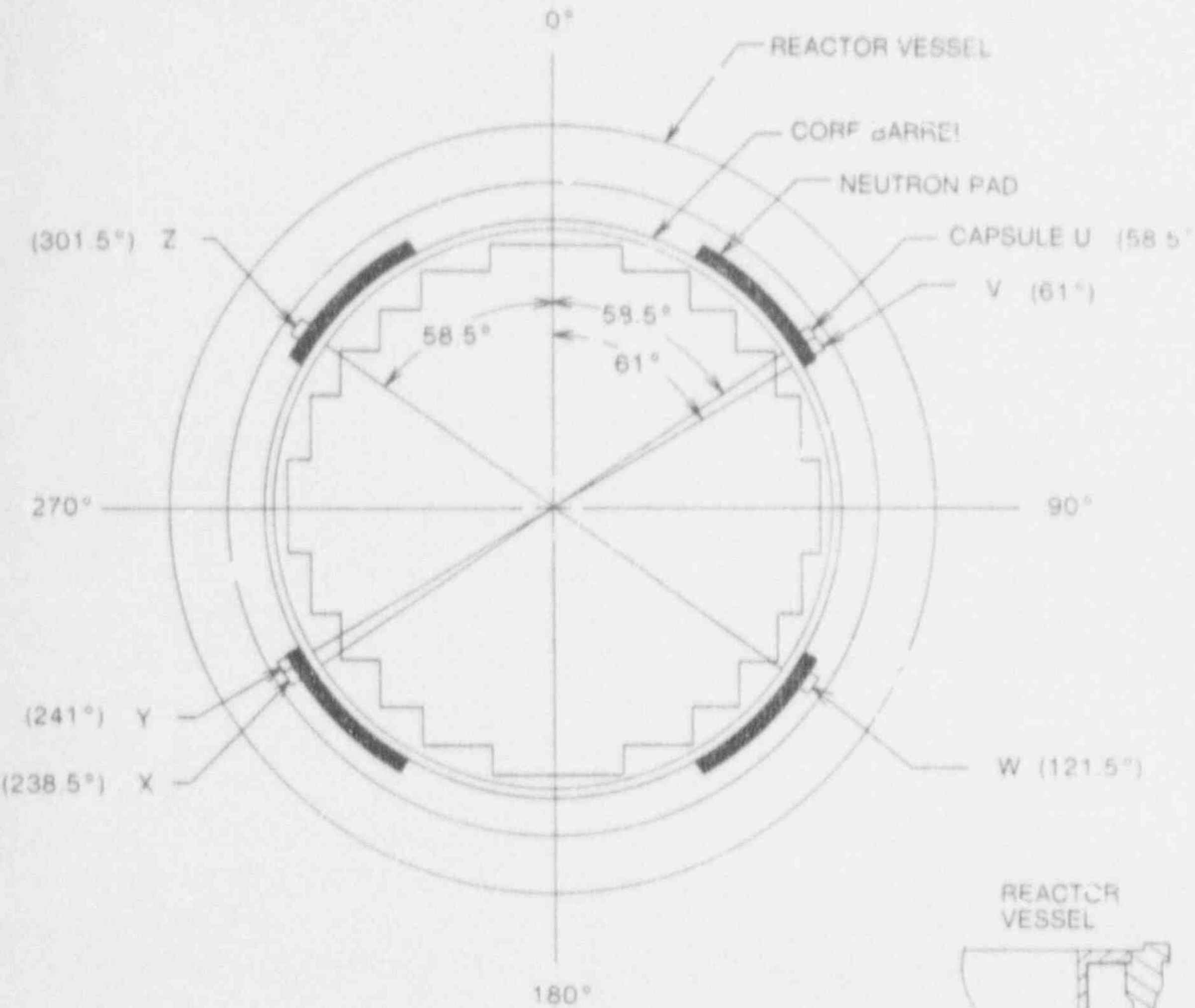
These data can then be reviewed to verify or revise pressure-temperature limits of the vessel during heatup and cooldown and will allow a check of the predicted shift in the reference temperature. The postirradiation test results of the compact specimens will provide actual fracture toughness properties of the vessel material. These properties may be used to establish allowable stress intensity factors for subsequent analyses.

Six material test capsules are fabricated containing specimens from the reactor vessel shell plate identified as being most likely to limit the operation of the reactor vessel.

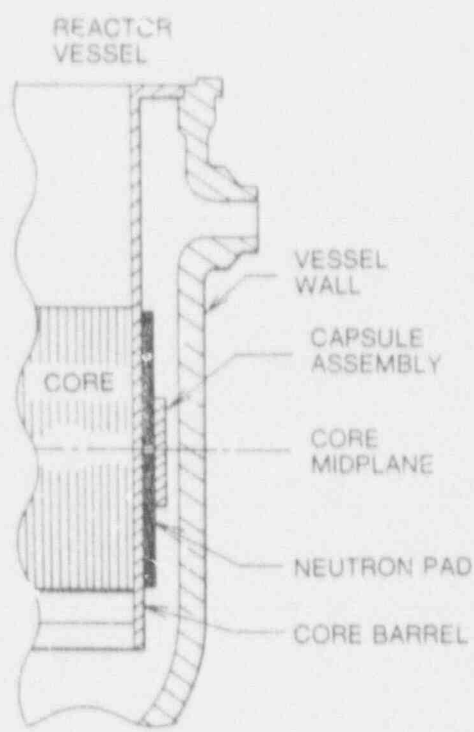
The specimens contained in the Alvin W. Vogtle Unit No. 2 test capsules are from the lower shell plate of the reactor vessel and representative weld metal and heat-affected-zone (HAZ) metal.

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 postweld heat treatment received by the specimens has been simulated (Appendix A).

The six material test capsules are then installed in the reactor in guide tubes attached to the neutron shield pads which are located in the reactor between the core barrel and the reactor vessel wall opposite the center of the core as shown in Figure 1-1.



PLAN VIEW



ELEVATION VIEW

FIGURE 1-1. LOCATION OF THE IRRADIATION TEST CAPSULES IN THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR VESSEL

SECTION 2
CAPSULE PREPARATION

2-1. PRESSURE VESSEL MATERIAL

Reactor vessel material was supplied by Combustion Engineering, Inc. from lower shell plate B8628-1, Heat No. C3500-2. Combustion Engineering, Inc., also supplied a weldment which joined sections of material of the lower shell plate B8628-1 and the adjacent lower shell plate B8825-1, Heat No. C3500-1. Data on the limiting core region plate (B8628-1), weld, and weld-heat-affected-zone material are provided in Appendix A.

2-2. MACHINING

Test material obtained from the lower shell plate (after the thermal heat treatment and forming of the plate) was taken at least one plate thickness from the quenched ends of the plate. All test specimens were machined from the ¼-thickness [a] location of the plate after performing a simulated postweld, stress-relieving treatment on the test material and also from weld and heat-affected-zone metal of a stress-relieved weldment joining lower shell plate B8628-1 and adjacent lower shell plate B8825-1. All heat-affected-zone specimens were obtained from the weld heat-affected-zone of lower shell plate B8628-1.

2.3 Charpy V-notch Impact Specimens

Charpy V-notch impact specimens corresponding to ASTM A370 Type A (Figure 2-1) were machined from lower shell plate B8628-1 in both the longitudinal orientation (longitudinal axis of specimen parallel to major rolling direction) and transverse orientation (longitudinal axis of specimen normal to major rolling direction). The core region weld Charpy impact specimens were machined from the weldment such that the long dimension of the Charpy specimen was normal to the weld direction. The notch was machined such that the direction of crack propagation in the specimen was in the welding direction.

a. The compact test specimens were obtained from the ¾T-thickness Location of the plate.

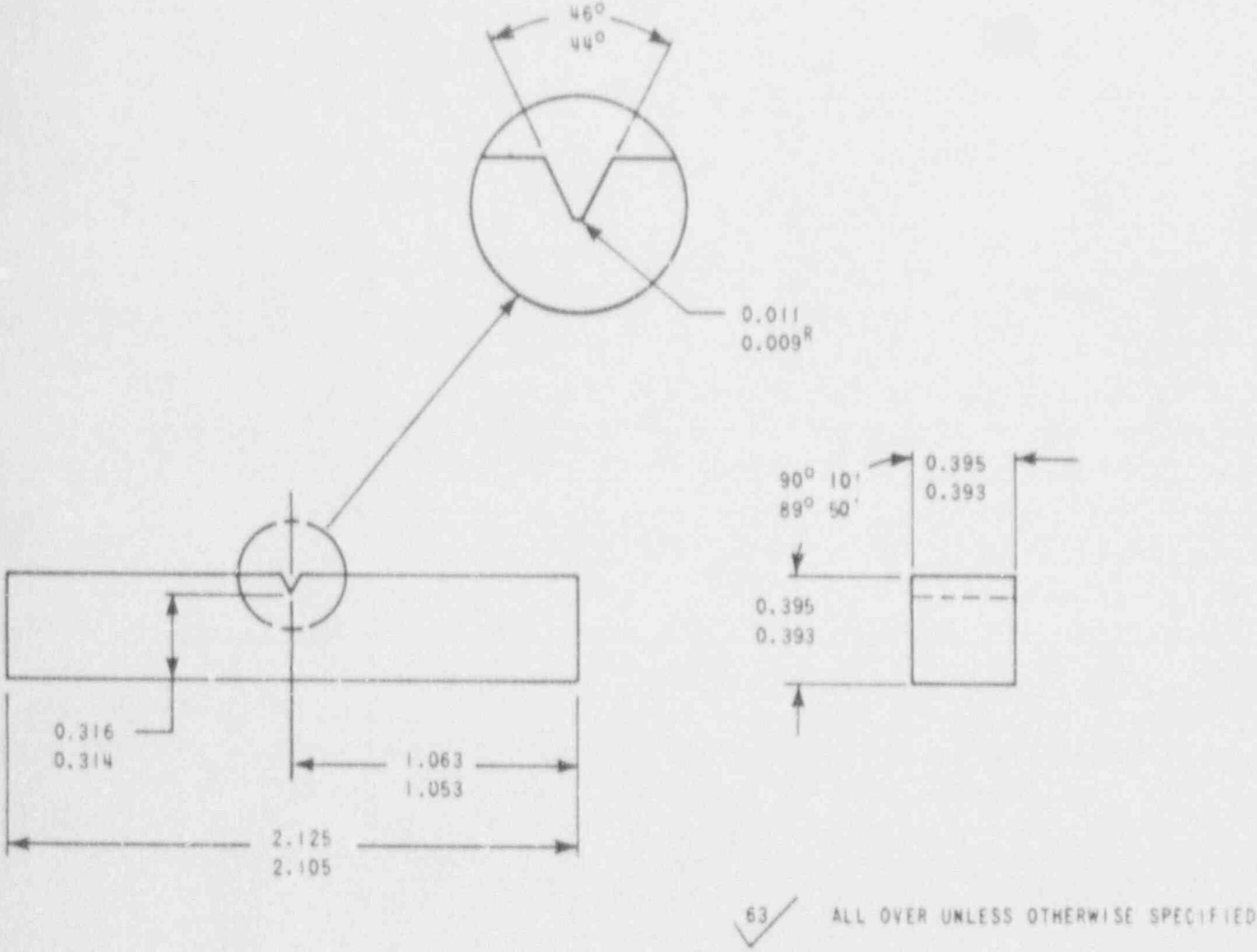


Figure 2-1. Charpy V-notch Impact Specimen

2-4. Tensile Specimens

Tensile specimens (Figure 2-2) from shell plate B8628-1 were machined in both the longitudinal and transverse orientation. Tensile specimens from the weld were oriented normal to the welding direction.

2-5. 1/2T Compact Specimens

Compact test specimens (Figure 2-3) from shell plate B8628-1 were machined in both the longitudinal and transverse orientations. Compact test specimens from the weld metal were machined with the notch oriented in the direction of welding. All specimens were fatigue precracked according to ASTM E399.

2-6. DOSIMETERS

Each of the six test capsules of the type shown in Figure 2-4 contain dosimeters of copper, iron, nickel and aluminum 0.15 weight percent cobalt wire (cadmium-shielded and unshielded) and cadmium-shielded Np^{237} and U^{238} which will measure the integrated flux at specific neutron energy levels.

2-7. THERMAL MONITORS

The capsules contain two low-melting-point eutectic alloys to more accurately define the maximum temperature attained by test specimens during irradiation. The thermal monitors are sealed in Pyrex tubes and then inserted in spacers located as shown in Figure 2-4. The two eutectic alloys and their melting points are the following:

2.5 percent Ag, 97.5 percent Pb	Melting point: 304°C (579°F)
1.5 percent Ag, 1.0 percent Sn, 97.5 percent Pb	Melting point: 310°C (590°F)

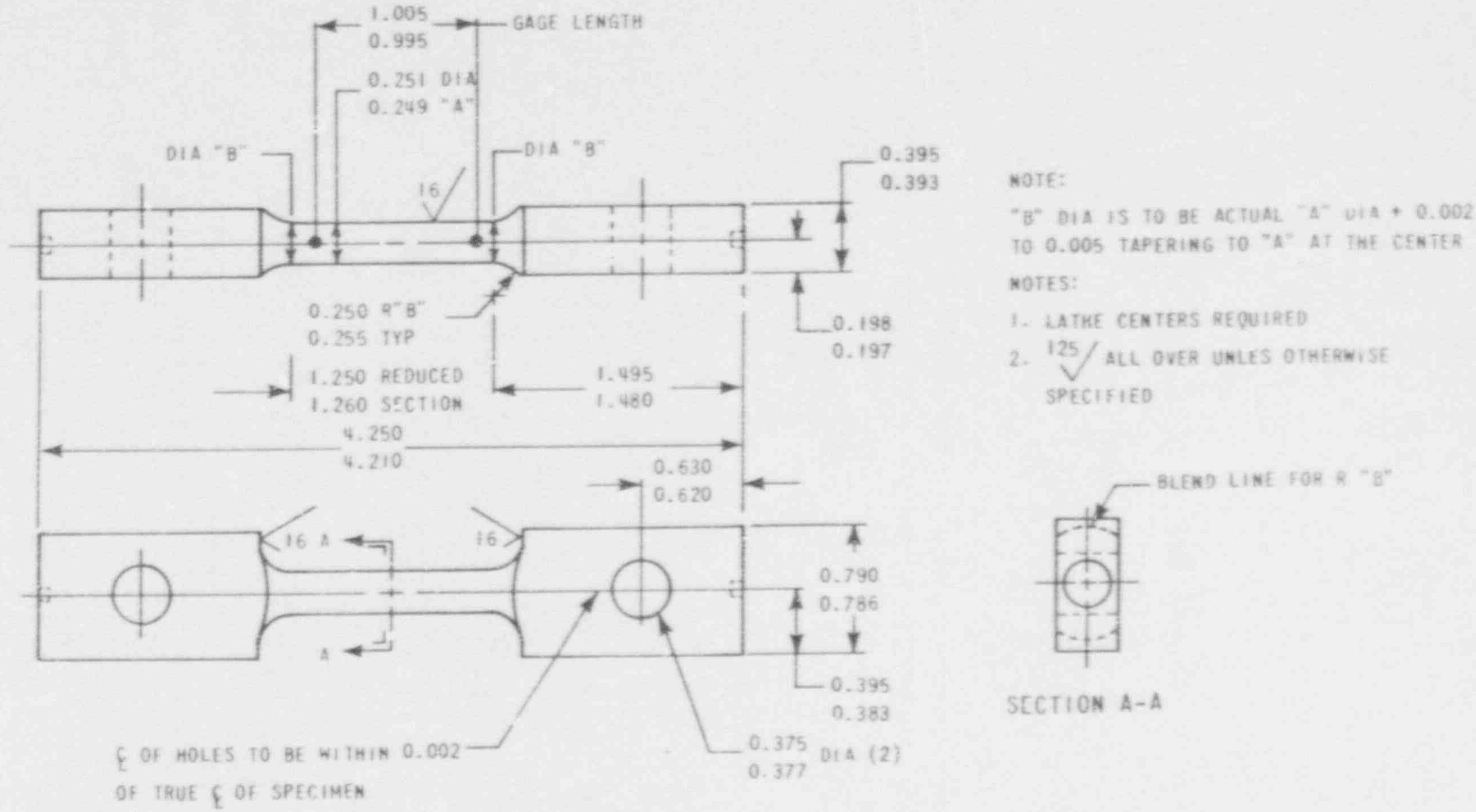


Figure 2-2. Tensile Specimen

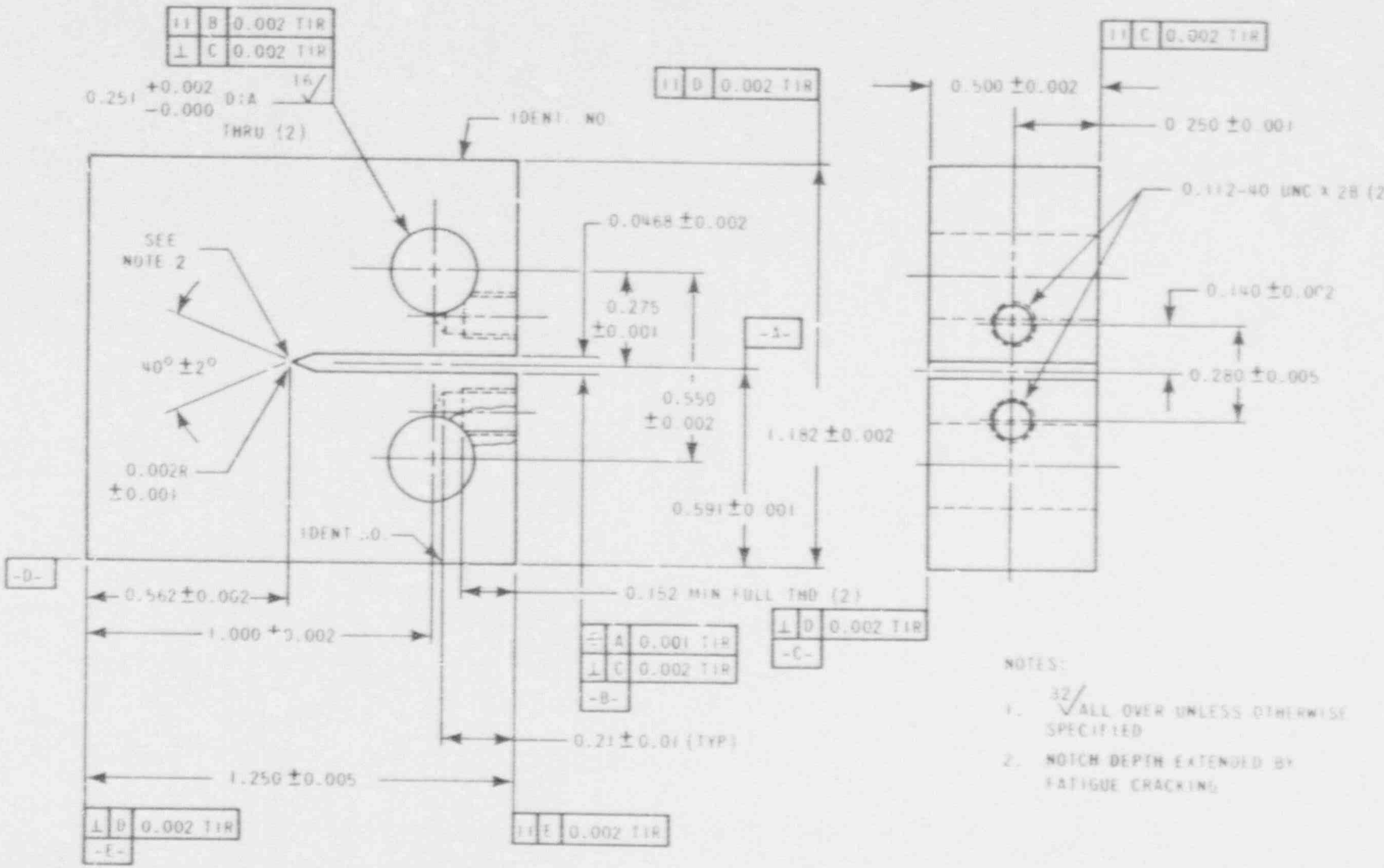
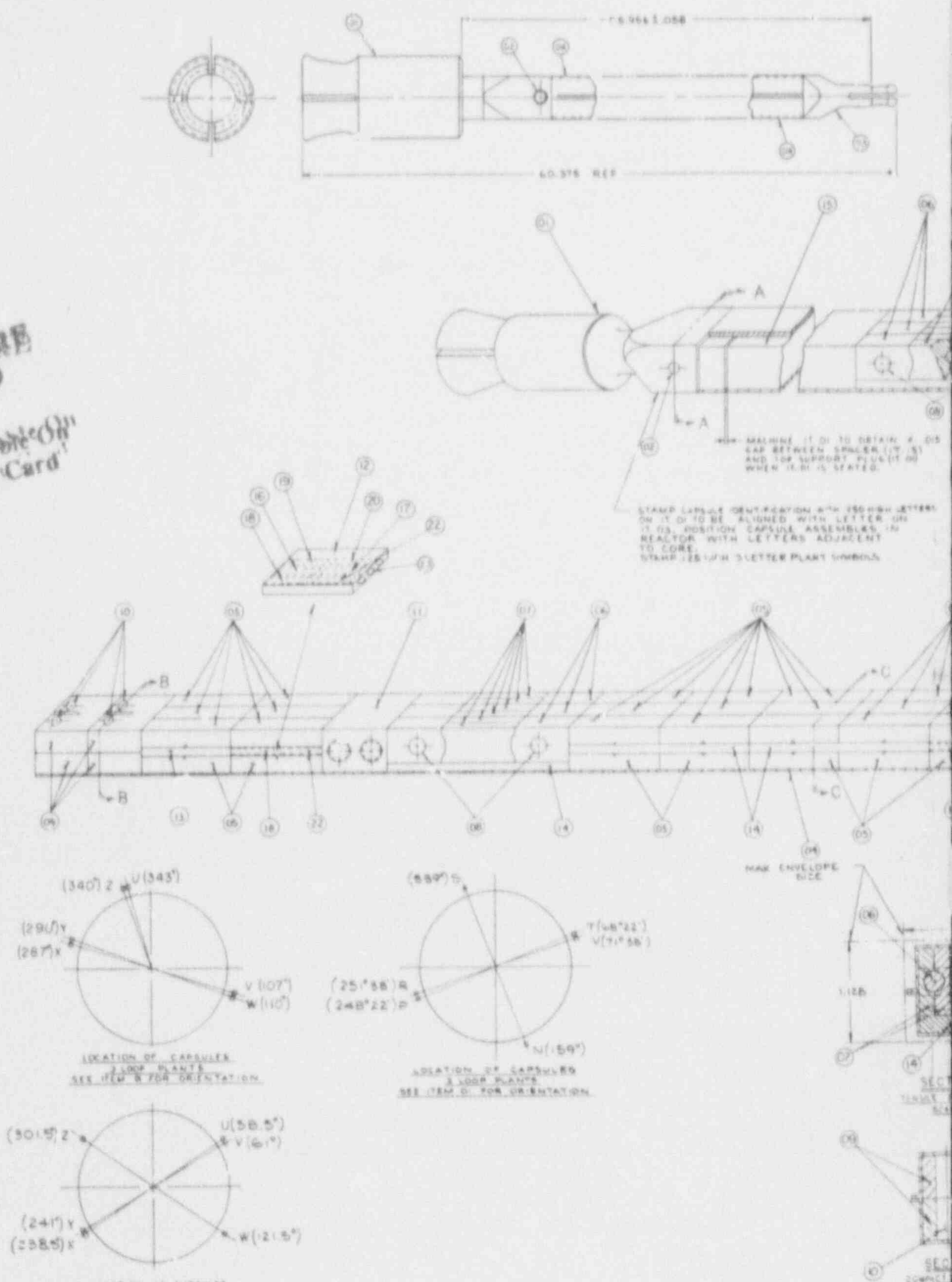


Figure 2-3. Compact Specimen


60.375 REF
A
B
C
MAX ENVELOPE SIZE
1.188
(340°) Z
U (343°)
(291°) Y
(287°) X
V (107°)
W (110°)
LOCATION OF CAPSULES
2 LOOP PLANTS
(S59°) S
T (48°22')
V (71°38')
(251°38') R
(248°22') P
N (159°)
LOCATION OF CAPSULES
3 LOOP PLANTS
SEE ITEM 01 FOR ORIENTATION
(301.5°) Z
U (58.5°)
V (61°)
(241°) Y
(238.5) X
W (121.5°)
LOCATION OF CAPSULES
4 LOOP PLANTS
SEE ITEM 01 FOR ORIENTATION

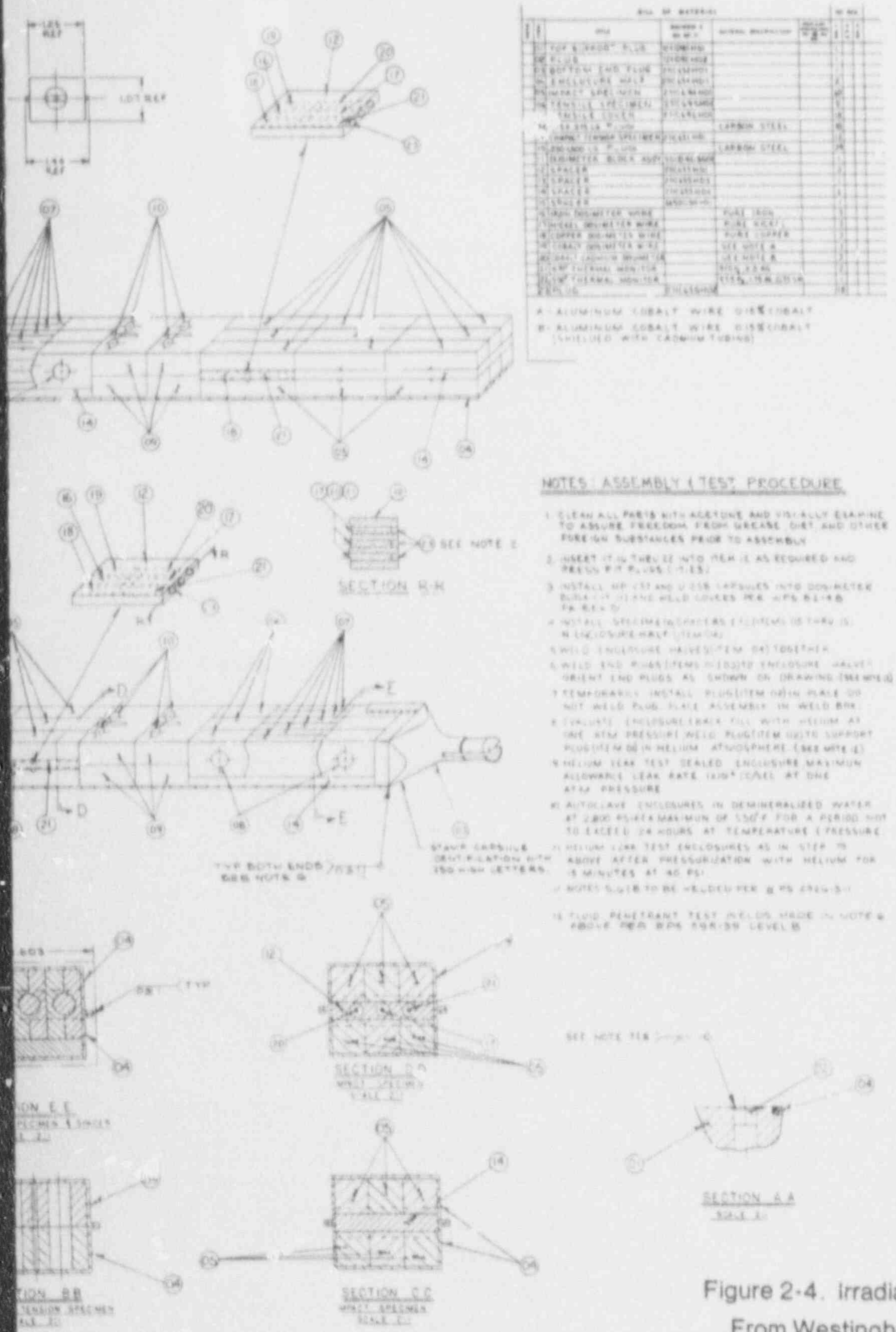


Figure 2-4. Irradiation Capsule Assembly
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2-8. CAPSULE LOADING

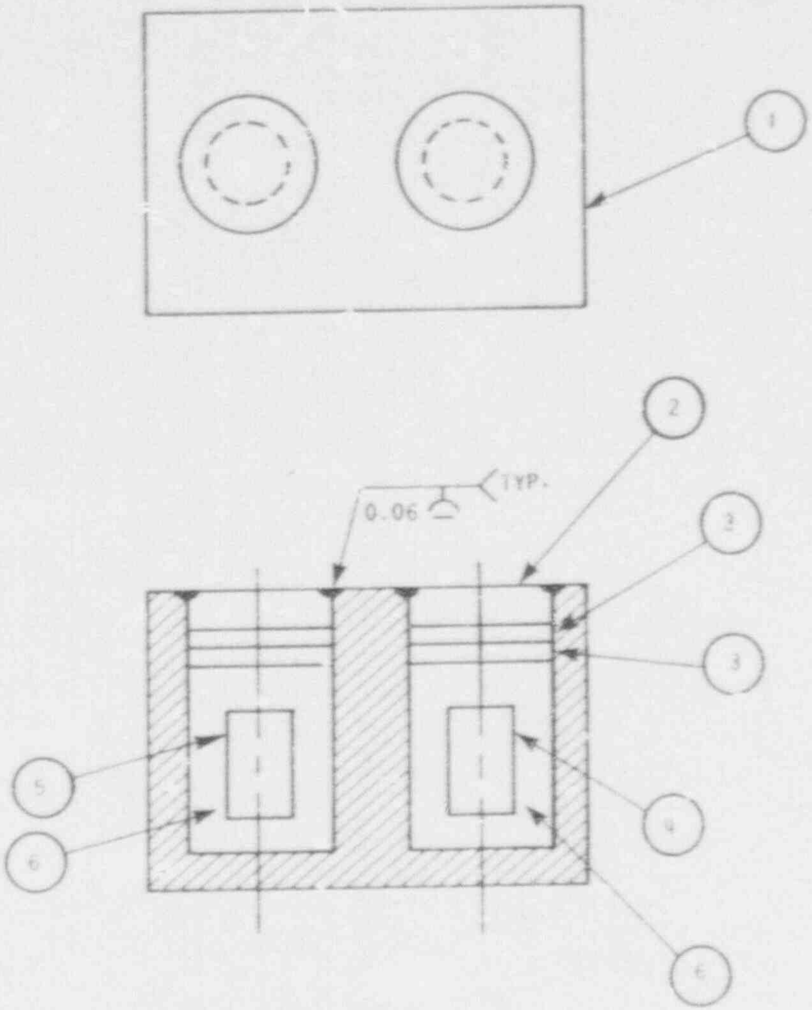
The six test capsules coded U, V, W, X, Y, and Z are positioned in the reactor between the neutron shielding pads and vessel wall at the locations shown in Figure 2-4. Each capsule contains 60 Charpy V-notch specimens, 9 tensile specimens and 12 compact specimens. The relationship of the test material to the type and number of specimens in each capsule is shown in Table 2-1.

TABLE 2-1

TYPE AND NUMBER OF SPECIMENS IN THE ALVIN W. VOGTLE UNIT NO. 2 SURVEILLANCE TEST CAPSULES

Material	Capsules U, V, W, X, Y, and Z		
	Charpy	**Tensile**	**Compact**
Plate B8628-1 Longitudinal	15 (Specimens Each Capsule)	3	4
Transverse	15	3	4
Weld Metal	15	3	4
HAZ	15	—	—

Dosimeters of copper, iron, nickel, aluminum 0.15 weight percent cobalt, and cadmium-shielded aluminum-cobalt wires are secured in holes drilled in spacers located at capsule positions shown in Figure 2-4. Each capsule also contains a dosimeter block (Figure 2-5) located at the center of the capsule. Two cadmium-oxide-shielded tubes, one containing an isotope of U^{238} and the other an isotope of 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}(Table 2-2) held in a $^{3}/_{8}$-inch-long by ¼-inch outside diameter sealed stainless steel tube, respectively. Each tube was placed in a ½-inch-diameter hole in the dosimeter block (one U^{238} and one Np^{237} tube per block), and the space around the tube was



ITEM	TITLE	MATERIAL SPECIFICATION	NO REQ'D
1	BLOCK	CARBON STEEL	1
2	COVER	CARBON STEEL	2
3	SPACER	ALUMINUM	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)	STAINLESS STEEL	1
6	CADMIUM OXIDE		AS REQ'D

Figure 2-5. Dosimeter Block Assembly

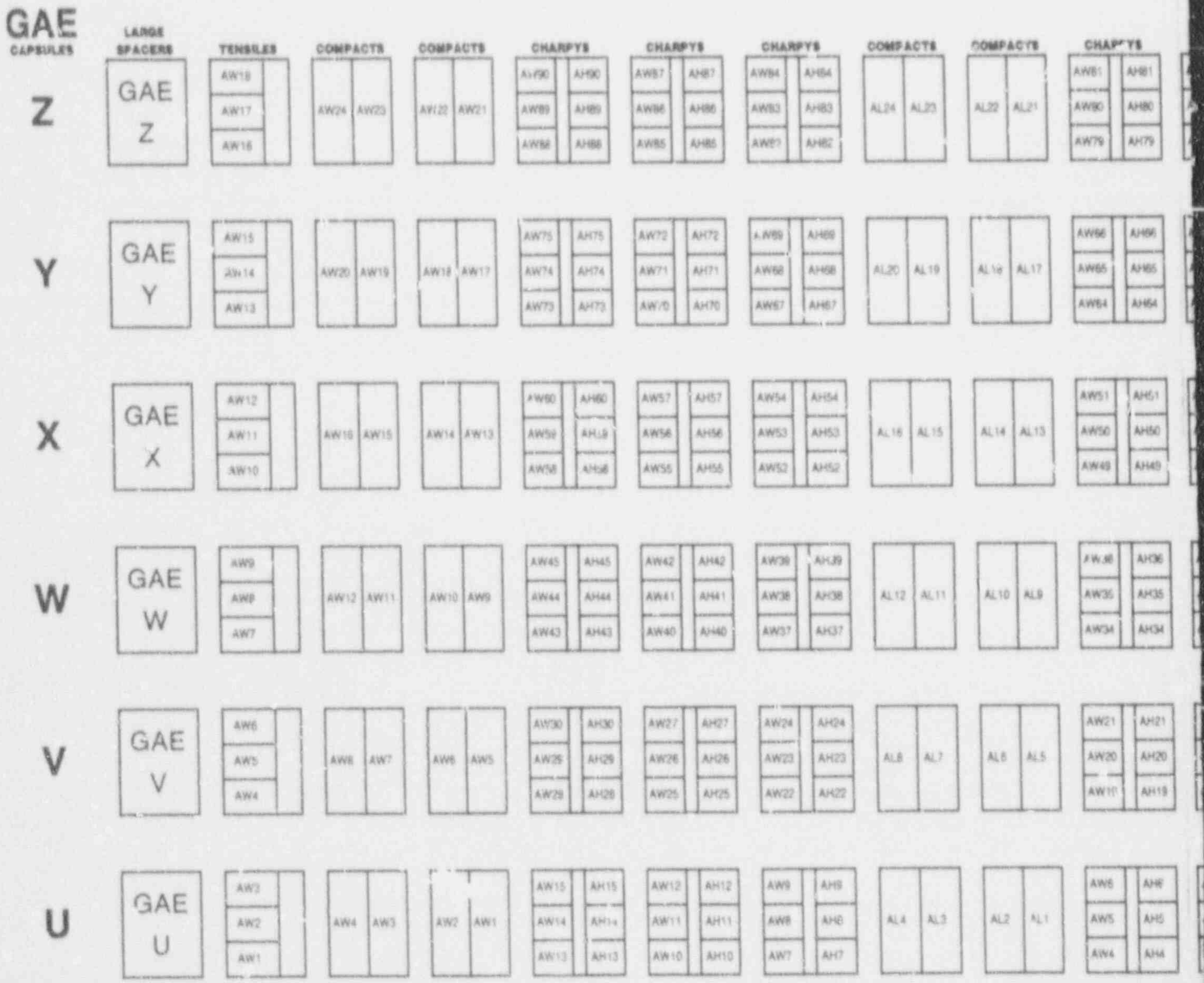
filled with cadmium oxide. After placement of this material, each hole was blocked with two $^1/_{16}$-inch-thick aluminum spacer discs and an outer $^1/_8$-inch-thick steel cover disc welded in place.

The numbering system for the capsule specimens and their locations is shown in Figure 2-6. The specimens are seal-welded into a square capsule of austenitic stainless steel to prevent corrosion of specimen surfaces during irradiation. The capsules are hydrostatically compressed in demineralized water to collapse the capsule on the specimens so that optimum thermal conductivity between the specimens and the reactor coolant is obtained. The capsules are then leak-tested with helium after pressurization and then dye penetrant tested as a final inspection procedure. Fabrication details and testing procedures are listed in Figure 2-4.

TABLE 2-2

QUANTITY OF ISOTOPES CONTAINED IN THE DOSIMETER BLOCKS

Isotope	Weight (mg)	Compound	Weight (mg)
Np^{237}	17 ± 1	NpO_2	20 ± 1
U^{238}	12.0	U_3O_8	14.25



LEGEND: AL - INTERMEDIATE SHELL PLATE B8805-3 (LONGITUDINAL)
AT - INTERMEDIATE SHELL PLATE B8805-3 (TRANSVERSE)
AW - WELD METAL
AH - HEAT-AFFECTED-ZONE MATERIAL

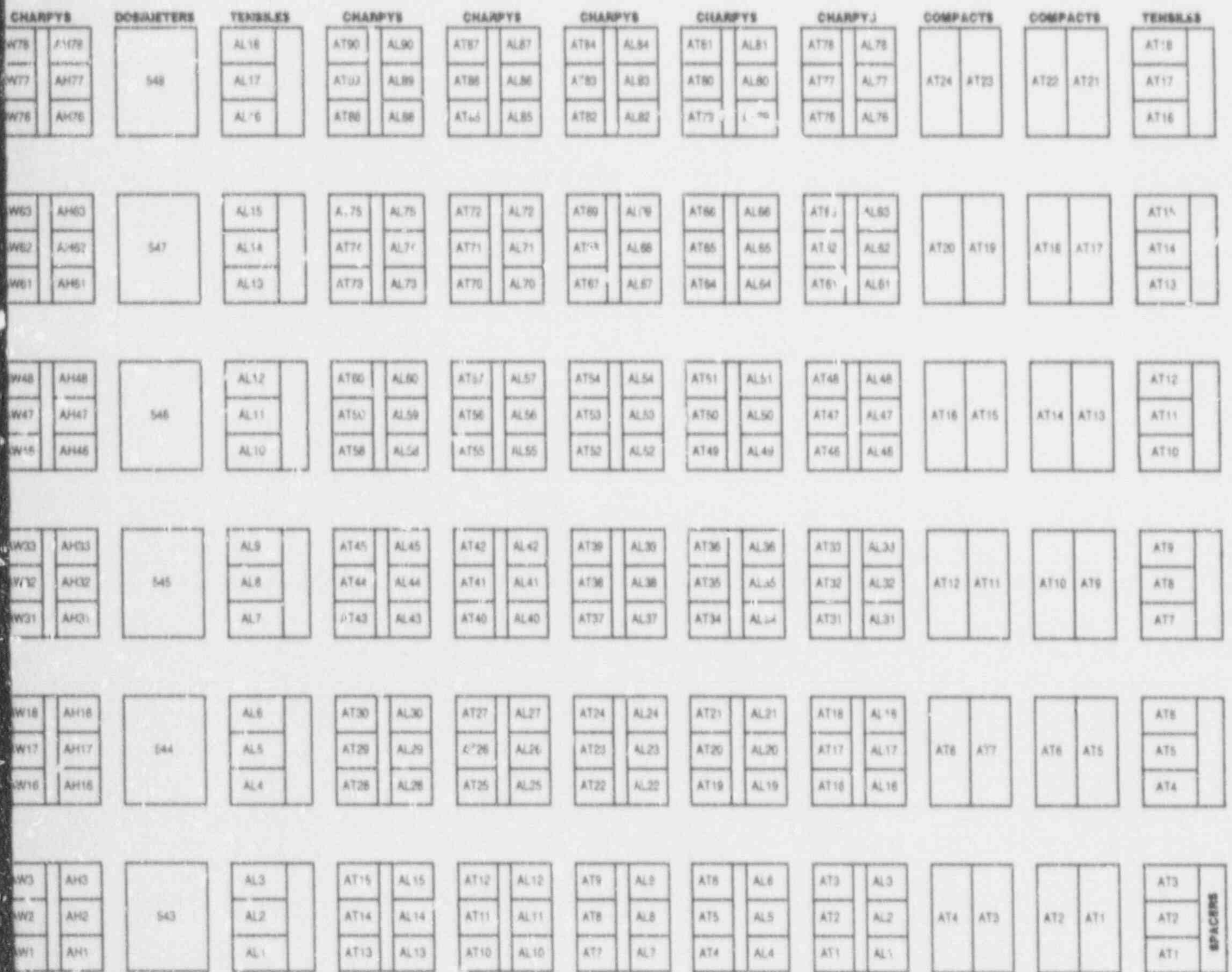


Figure 2-6. Specimen Location in the Alvin W. Vogtle Unit No. 1 Reactor Surveillance Test Capsules

SECTION 3
PREIRRADIATION TESTING

3-1. CHARPY V-NOTCH TESTS

Charpy V-notch impact tests were performed according to ASTM E23 with specimens from the vessel lower shell plate B8628-1. Specimens of both longitudinal and transverse orientations were tested at various test temperatures in the range from −62°C to 149°C (−80°F to 300°F), yeilding a full Charpy V-notch transition temperature curve in both orientations (Tables 3-1 and 3-2 and Figures 3-1 and 3-2). Tests were also performed on the weld metal and HAZ metal at various temperatures from −118°C to 149°C (−180°F to 300°F) and are shown in Tables 3-3 and 3-4 and Figures 3-3 and 3-4.

Charpy V-notch impact tests by Westinghouse on the surveillance plate B8628-1 in the transverse direction resulted in an upper shelf energy of 70 ft.-lbs. as shown in Figure 3-2. Material qualifacation tests performed by Combustion Engineering Inc., per Section 3 of the ASME Boiler and Pressure Vessel Code resulted in a somewhat higher shelf energy (85 ft. lb.) and therefore the plate is in compliance with the 10 CFR 50 Appendix G requirement "that reactor vessel beltline materials have a Charpy upper shelf energy of no less than 75 ft. lb. initially." The difference in upper shelf energy between the two tests is attributed to 1.) variation in sulfur content due to segregation during ingot solidifacation which then can result in significant sulfur variation in the final rolled plate, 2.) difference in post weld stress relief time which can produce significant difference in toughness

A summary of the Charpy V-notch impact tests results including upper shelf energy (USE), 41-joule (30 ft lb), 68-joule (50 ft lb), and 35 mils (0.89mm) lateral expansion index temperatures are presented in Table 3-5.

The specimens were tested on a SONNTAG UNIVERSAL impact machine, Model Number SI-1 with a hammer energy capacity of 240 foot pounds and a striking velocity of 17.02 feet per second. The machine is calibrated every 6 months using Charpy V-notch impact specimens of known energy values supplied by Watertown Arsenal. Specimen conditioning for high temperature testing is maintained using a Fisher chest type ceramic furnace with a Newport temperature controller with direct digital temperature readout. For all low temperature specimen conditioning liquid Nitrogen is used. The specimen temperatures are monitored by the use of Chromel Aluminal thermal couples at high temperatures and by the use of Copper Constantan thermal couples at low temperature testing.

3-2. TENSILE TESTS

Table 3-6 and Figures 3-5, 3-6, and 3-7 show the results of tensile tests (per ASTM E8 and E-21 test criteria) from vessel lower shell plate B8628-1 and from the weld metal. Specimens from plate B8628-1 and the weldment were tested at 24°C (75°F), 149°C (300°F) and 288°C (550°F) in both the longitudinal and transverse directions.

A SATEC UNIVERSAL tensile testing machine Model 30WBN, was used with a SATEC 30,000 lb. load cell as an integral part of the testing macnine. The testing machine is calibrated daily and verified annually to the National Bureau of Standards. The gripping mechanism utilizes threaded adapters to pull rods attached to the cross head/load cell and frame. The recording device utilizes a Hewlett Packard X-Y recorder and s chart in console, serial number 7047A calibrated to a dual range high temperature extensometer, serial number BDRE-1. The extensometer is calibrated by test equipment which has been certified by the National Bureau of Standards. The measurement and control of speeds in the tests conform to ASTM A370-77 (Mechanical Testing of Steel Products). A typical stress-strain curve is shown in Figure 3-8. For high temperature tests an Applied Systems 3-zone type furnace was used with independent zone control. Temperatures were controlled by a Athena Controls, Type K $^T/_c$, Model Number 4000-T-302F temperature controller utilizing type "K" thermal couples with direct digital temperature readout.

3-3. DROPWEIGHT TESTS

The nil ductility transition temperature (T_{NDT}) was determined for plate B8628-1 and the core region weld metal by dropweight tests (ASTM E-208) performed at Combustion Engineering, Inc. From this test data the RT_{NDT} was calculated using the methods as described in Section 1. The T_{NDT} and RT_{NDT} for lower shell plate B8628-1, weld metal are as follows:

Note: T_{NDT} and RT_{NDT} for all the beltline shell plates is given in Appendix A.

Material	T_{NDT} (°F)	RT_{NDT} (°F)
Plate B8628-1	– 20[a]	50
Weld Metal (Longitudinal Seams	– 10[b]	– 10
Weld Metal (Girth Seam)	– 50[c]	– 30

a. Combustion Engineering Inc. Materials Certification Report
b. Combustion Engineering Welding Material Qualification Test H-32255
c. Combustion Engineering Welding Material Qualification Test P-32255

TABLE 3-1

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (LONGITUDINAL ORIENTATION)

Temperature		Impact Energy		Lateral Expansion		Shear
(°C)	(°F)	(J)	(ft lb)	(mm)	(mils)	(%)
− 62	− 80	5.0	4.0	0.03	1.0	0
− 62	− 80	9.5	7.0	0.08	3.0	0
− 34	− 30	16.0	12.0	0.25	10.0	5
− 34	− 30	27.0	20.0	0.36	14.0	5
− 34	− 30	35.0	26.0	0.46	18.0	5
− 18	− 0	32.5	24.0	0.53	21.0	10
− 18	0	37.0	27.0	0.53	21.0	10
− 18	0	46.0	34.0	0.63	25.0	10
− 1	30	41.0	30.0	0.61	24.0	20
1	30	43.0	32.0	0.66	26.0	20
1	30	69.0	51.0	1.02	40.0	15
16	60	64.0	47.0	0.91	36.0	30
16	60	73.0	54.0	1.09	43.0	35
16	60	79.0	58.0	1.14	45.0	45
27	80	92.0	68.0	1.42	56.0	75
27	80	96.0	71.0	1.37	54.0	75
27	80	100.0	70.0	1.40	55.0	65
49	120	110.0	81.0	1.80	71.0	100
49	120	118.0	87.0	1.78	70.0	100
49	120	119.0	88.0	1.80	71.0	100
71	160	118.0	87.0	1.78	70.0	100
71	160	125.0	92.0	1.93	76.0	100
116	240	122.0	90.0	1.96	77.0	100
116	240	122.0	90.0	1.88	74.0	100
116	240	130.0	96.0	1.75	69.0	100
149	300	119.0	88.0	1.88	74.0	100
149	300	122.0	90.0	1.88	74.0	100

TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (TRANSVERSE ORIENTATION)

Temperature		Impact Energy		Lateral Expansion		Shear
(°C)	(°F)	(J)	(ft lb)	(mm)	(mils)	(%)
– 62	– 80	7.0	5.0	0.05	2.0	0
– 62	– 80	7.0	5.0	0.08	3.0	0
– 34	– 30	22.0	16.0	0.36	14.0	5
– 34	– 30	24.5	18.0	0.36	14.0	5
– 18	0	26.0	19.0	0.48	18.0	10
– 18	0	30.0	22.0	0.51	20.0	10
– 18	0	32.5	24.0	0.51	20.0	10
– 1	30	28.5	21.0	0.53	21.0	25
– 1	30	37.0	27.0	0.66	26.0	10
– 1	30	45.0	33.0	.79	31.0	10
16	60	51.5	38.0	.89	35.0	35
16	60	51.5	38.0	.97	38.0	40
16	60	56.0	41.0	.97	38.0	35
38	100	76.0	56.0	1.32	52.0	80
38	100	87.0	64.0	1.42	56.0	95
38	100	99.0	73.0	1.63	64.0	95
49	120	89.5	66.0	1.52	60.0	95
49	120	100.0	74.0	1.73	68.0	100
49	120	107.0	79.0	1.80	71.0	100
71	160	91.0	67.0	1.47	58.0	100
71	160	92.0	68.0	1.63	64.0	100
71	160	92.0	68.0	1.63	64.0	100
116	240	85.5	63.0	1.52	60.0	100
116	240	93.5	69.0	1.70	67.0	100
149	300	89.5	66.0	1.73	68.0	100
149	300	95.0	70.0	1.78	70.0	100
177	350	84.0	62.0	1.50	59.0	100
177	350	88.0	65.0	1.57	62.0	100
204	400	98.0	72.0	1.65	65.0	100
204	400	102.0	75.0	1.52	60.0	100
232	450	96.0	71.0	1.60	63.0	100
232	450	103.0	76.0	1.68	66.0	100

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR
PRESSURE VESSEL CORE REGION
WELD METAL

Temperature		Impact Energy		Lateral Expansion		Shear
(°C)	(°F)	(J)	(ft lb)	(mm)	(mils)	(%)
− 73	−100	7.0	5.0	0.03	1.0	0
− 73	−100	9.5	7.0	0.08	3.0	0
− 73	−100	9.5	7.0	0.08	3.0	0
− 51	− 60	15.0	11.0	0.23	9.0	5
− 51	− 60	15.0	11.0	0.18	7.0	5
− 51	− 60	37.0	27.0	0.56	22.0	10
− 34	− 30	11.0	8.0	0.23	9.0	10
− 34	− 30	12.0	9.0	0.25	10.0	15
− 34	− 30	27.0	20.0	0.41	16.0	10
− 29	− 20	20.0	15.0	0.28	11.0	15
− 29	− 20	51.5	38.0	0.86	34.0	20
− 29	− 20	66.0	49.0	0.99	39.0	25
− 18	0	47.5	35.0	0.76	30.0	25
− 18	0	60.0	44.0	0.97	38.0	20
− 18	0	77.0	57.0	1.14	45.0	35
− 1	30	73.0	54.0	1.17	46.0	35
− 1	30	88.0	65.0	1.45	57.0	55
− 1	30	102.0	75.0	1.57	62.0	65
27	80	102.0	75.0	1.63	64.0	80
27	80	107.0	79.0	1.73	68.0	85
27	80	114.0	84.0	1.83	72.0	90
49	120	117.0	86.0	1.88	74.0	95
49	120	119.0	88.0	1.90	75.0	95
49	120	121.0	89.0	1.98	78.0	95
71	160	122.0	90.0	2.06	81.0	100
71	160	122.0	90.0	1.98	78.0	100
71	160	127.5	94.0	2.08	82.0	100
116	240	125.0	92.0	2.06	81.0	100
116	240	125.0	92.0	2.11	83.0	100
149	300	125.0	92.0	2.08	82.0	100
149	300	125.0	92.0	2.11	83.0	100

TABLE 3-4

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION WELD HEAT-AFFECTED-ZONE MATERIAL

Temperature		Impact Energy		Lateral Expansion		Shear
(°C)	(°F)	(J)	(ft lb)	(mm)	(mils)	(%)
−118	−180	9.5	7.0	0.10	4.0	0
−118	−180	12.0	9.0	0.08	3.0	0
−84	−120	18.0	13.0	0.13	5.0	5
−84	−120	22.0	16.0	0.23	9.0	5
−84	−120	26.0	19.0	0.28	11.0	5
−73	−100	19.0	14.0	0.13	5.0	5
−73	−100	41.0	30.0	0.46	18.0	5
−73	−100	47.5	35.0	0.56	22.0	10
−62	−80	38.0	28.0	0.46	18.0	10
−62	−80	58.0	43.0	0.63	25.0	20
−62	−80	70.5	52.0	0.81	32.0	25
−51	−60	35.0	26.0	0.48	19.0	20
−51	−60	46.0	34.0	0.53	21.0	25
−51	−60	54.0	40.0	0.74	29.0	35
−34	−30	70.5	52.0	0.94	37.0	60
−34	−30	81.0	60.0	1.09	43.0	45
−34	−30	91.0	67.0	1.12	44.0	60
−18	0	108.5	80.0	1.32	52.0	70
−18	0	115.0	85.0	1.27	50.0	60
−18	0	131.5	97.0	1.68	66.0	90
−1	30	130.0	96.0	1.74	68.0	100
−1	30	134.0	99.0	1.68	66.0	100
−1	30	148.0	109.0	1.85	73.0	100
27	80	130.0	96.0	1.83	72.0	100
27	80	138.0	102.0	1.65	65.0	100
27	80	154.5	114.0	1.83	72.0	100
49	120	136.0	100.0	1.63	64.0	100
49	120	138.0	102.0	1.63	64.0	100
49	120	165.0	122.0	1.98	78.0	100
71	160	130.0	96.0	1.85	73.0	100
71	160	149.0	110.0	1.80	71.0	100
71	160	161.0	119.0	1.83	72.0	100
99	210	130.0	96.0	1.83	72.0	100
99	210	168.0	124.0	1.85	73.0	100

TABLE 3-5

SUMMARY OF ALVIN W. VOGTLE UNIT NO. 2
REACTOR PRESSURE VESSEL IMPACT TEST RESULTS FOR
LOWER SHELL PLATE B8628-1 AND
CORE REGION WELD AND HEAT-AFFECTED-ZONE MATERIAL

Material	Upper Shelf Energy (USE)		41-J (30-ft lb) Index Temp		68-J (50-ft lb) Index Temp		0.89 mm (35 mils) Index Temp	
	(J)	(ft lb)	(°C)	(°F)	(°C)	(°F)	(°C)	(°F)
Plate B8628-1 (Longitudinal Orientation)	121	89	−12	−10	− 7	45	− 2	35
Plate B8628-1 (Transverse Orientation)	95.0	70	− 1	30	24	75	4	40
Weld Metal	125	92	−26	−15	−15	5	−21	− 5
Heat Affected Zone	144	106	−62	−80	−43	−45	−47	−50

TABLE 3-6
PREIRRADIATION TENSILE PROPERTIES FOR THE
ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE
VESSEL LOWER SHELL PLATE B8628-1
AND CORE REGION WELD METAL

Material	Test Temperature		0.2% Yield Strength		Ultimate Tensile Strength		Fracture Load		Fracture Stress		Fracture Strength		Uniform Enlongation (%)	Total Enlongation (%)	Reduction in Area (%)
	°C	°F	(ksi)	(MPa)	(ksi)	(MPa)	(kip)	(N)	(ksi)	(MPa)	(ksi)	(MPa)			
Plate B8628-1 (Longitudinal Orientation)	24	75	69.0	476.0	90.0	621.0	3.0	13,344.0	170.0	1,172.0	61.0	421.0	13.0	27.0	64.0
	24	75	69.0	476.0	90.0	621.0	3.0	13,344.0	165.0	1,138.0	62.0	428.0	13.0	28.0	62.0
	149	300	63.0	434.0	83.0	572.0	2.8	12,454.0	154.0	1,062.0	56.0	386.0	11.0	23.0	64.0
	149	300	62.0	428.0	83.0	572.0	2.8	12,454.0	144.0	993.0	57.0	393.0	12.0	23.0	60.0
	288	550	61.0	421.0	87.0	600.0	3.0	13,344.0	149.0	1,027.0	60.0	414.0	12.0	22.0	60.0
	288	550	61.0	421.0	87.0	600.0	3.0	13,344.0	155.0	1,069.0	62.0	428.0	13.0	24.0	60.0
Plate B8628-1 (Transverse Orientation)	24	75	69.0	476.0	89.0	614.0	3.2	14,234.0	136.0	938.0	64.0	441.0	12.0	24.0	53.0
	24	75	69.0	476.0	90.0	621.0	3.0	13,344.0	160.0	1,103.0	60.0	414.0	12.0	25.0	63.0
	149	300	62.0	428.0	83.0	572.0	3.0	13,344.0	129.0	889.0	58.0	400.0	11.0	22.0	55.0
	149	300	62.0	428.0	83.0	572.0	3.1	13,789.0	125.0	862.0	63.0	434.0	11.0	21.0	49.0
	288	550	61.0	421.0	88.0	607.0	3.4	15,123.0	140.0	965.0	69.0	476.0	12.0	22.0	51.0
	288	550	61.0	421.0	88.0	607.0	3.3	14,678.0	154.0	1,062.0	67.0	462.0	13.0	24.0	57.0
Weld Metal	24	75	67.0	462.0	83.0	572.0	2.8	12,454.0	158.0	1,089.0	56.0	386.0	13.0	25.0	64.0
	24	75	62.0	428.0	84.0	579.0	3.0	13,344.0	173.0	1,193.0	59.0	407.0	13.0	25.0	66.0
	149	300	61.0	421.0	77.0	531.0	2.8	12,454.0	155.0	1,069.0	57.0	393.0	10.0	21.0	64.0
	149	300	63.0	434.0	79.0	545.0	2.6	11,565.0	164.0	1,131.0	54.0	372.0	11.0	22.0	67.0
	288	550	63.0	434.0	83.0	572.0	3.0	13,344.0	158.0	1,089.0	59.0	407.0	11.0	22.0	63.0
	288	550	63.0	434.0	83.0	572.0	3.6	13,344.0	161.0	1,110.0	59.0	407.0	11.0	21.0	63.0

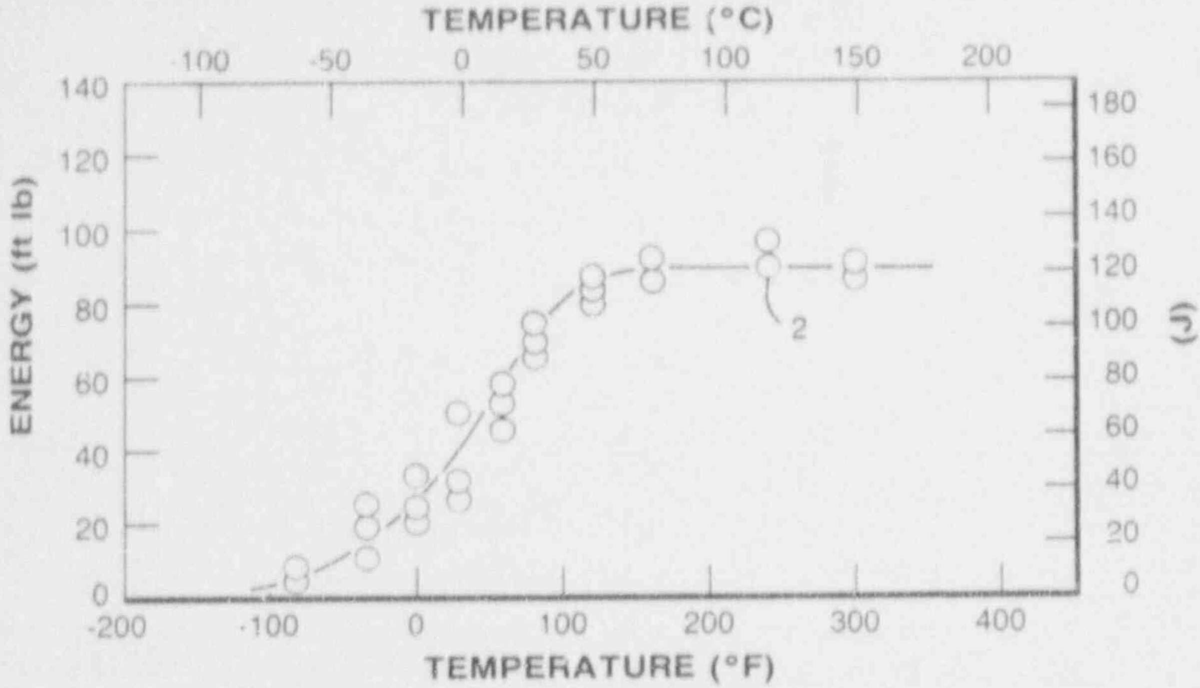


FIGURE 3-1. PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE ALVIN VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (LONGITUDINAL ORIENTATION)

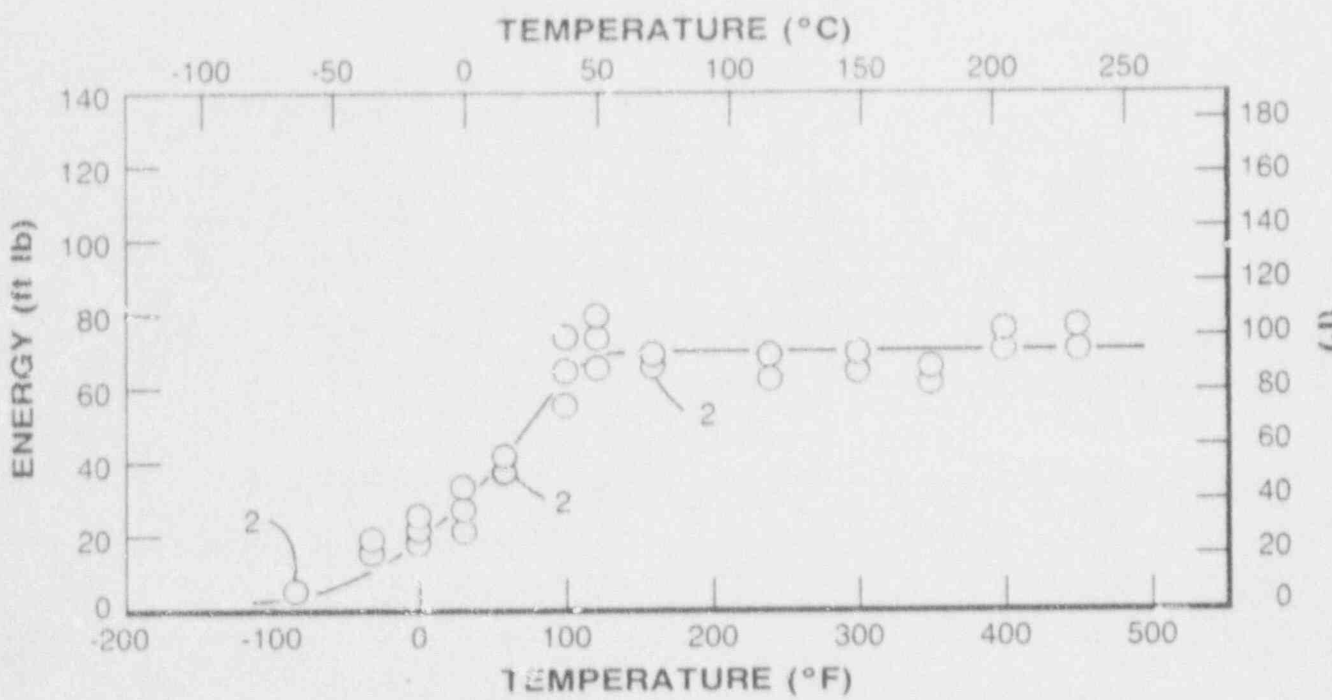


FIGURE 3-2. PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (TRANSVERSE ORIENTATION)

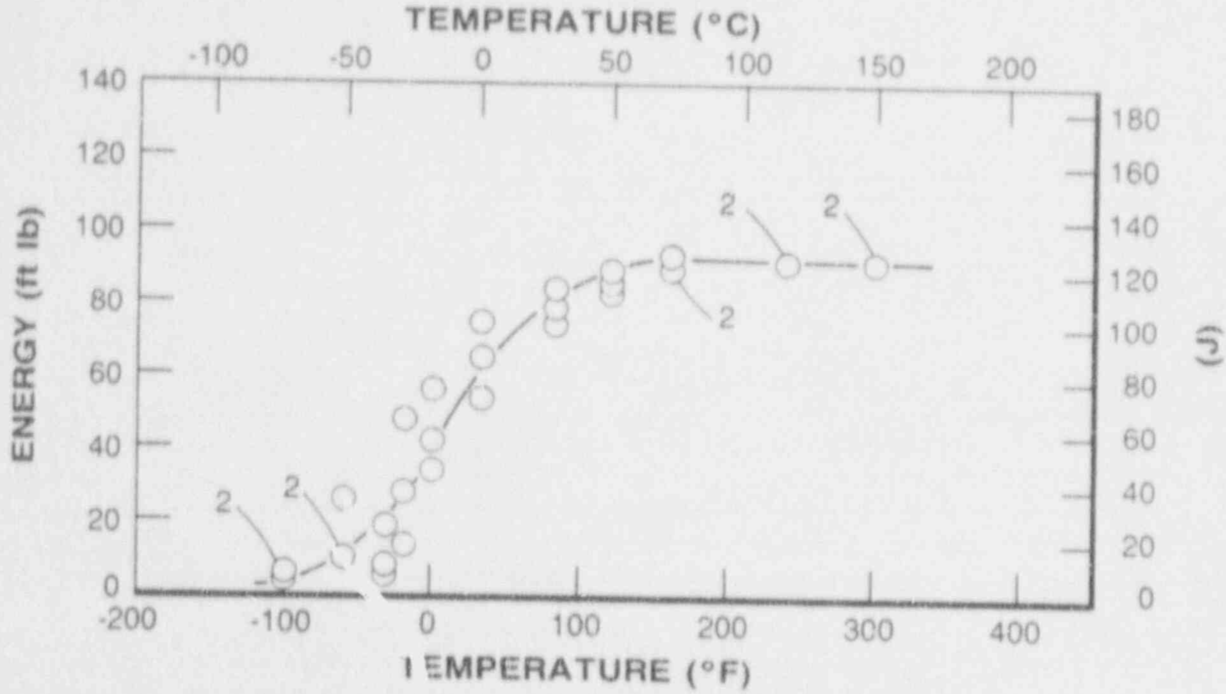


FIGURE 3-3. PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

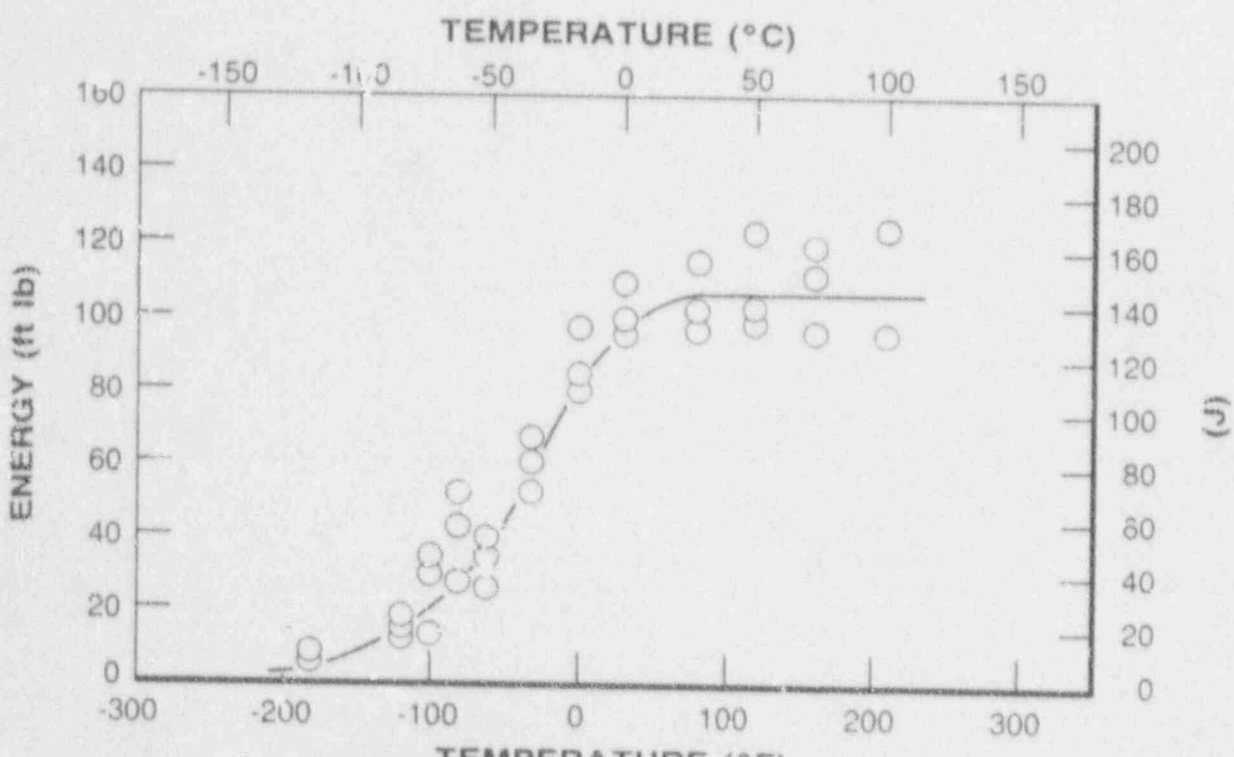


FIGURE 3-4. PREIRRADIATION CHARPY V-NOTCH IMPACT ENERGY FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL WELD HEAT-AFFECTED-ZONE MATERIAL

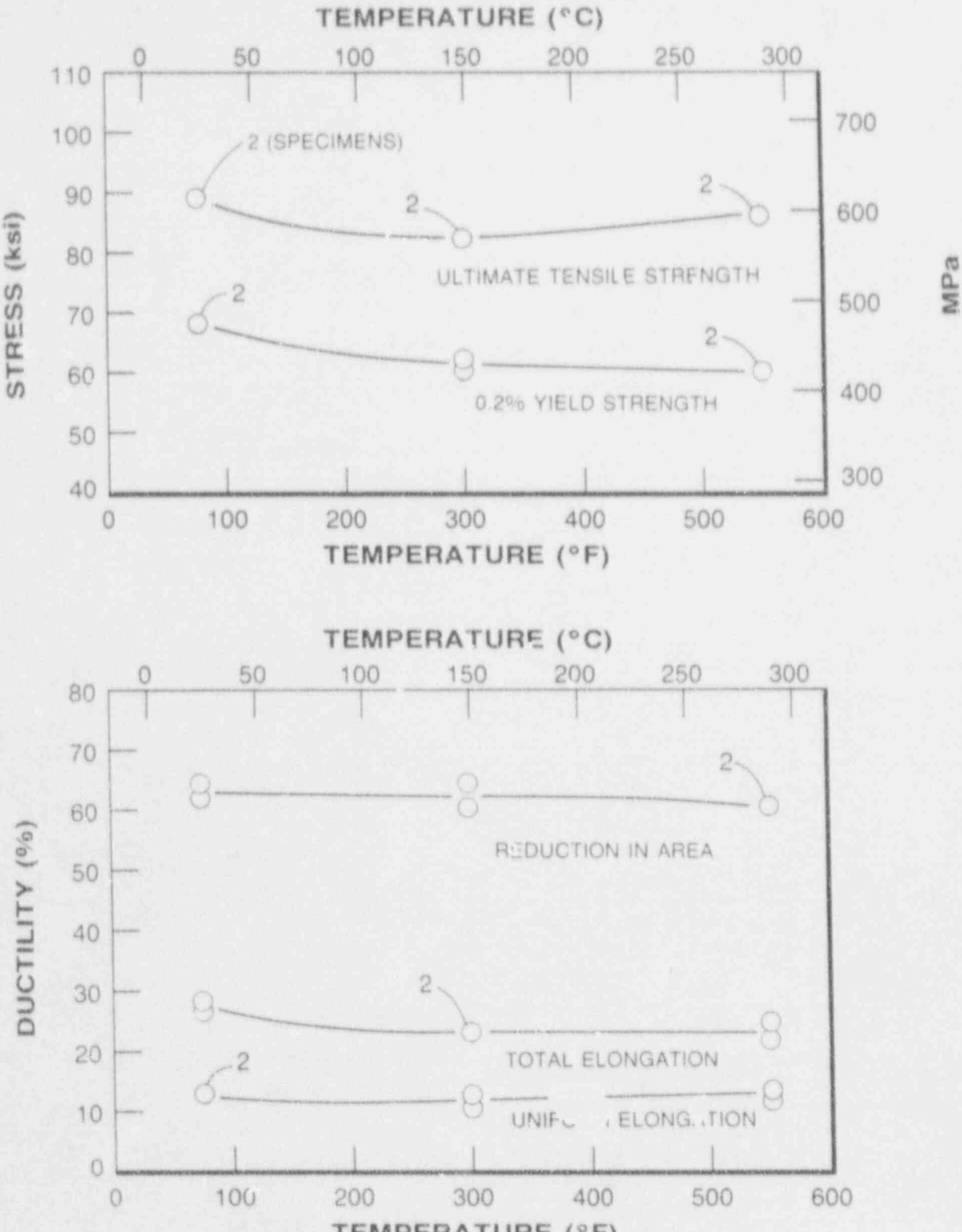


FIGURE 3-5. PREIRRADIATION TENSILE PROPERTIES FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (LONGITUDINAL ORIENTATION)

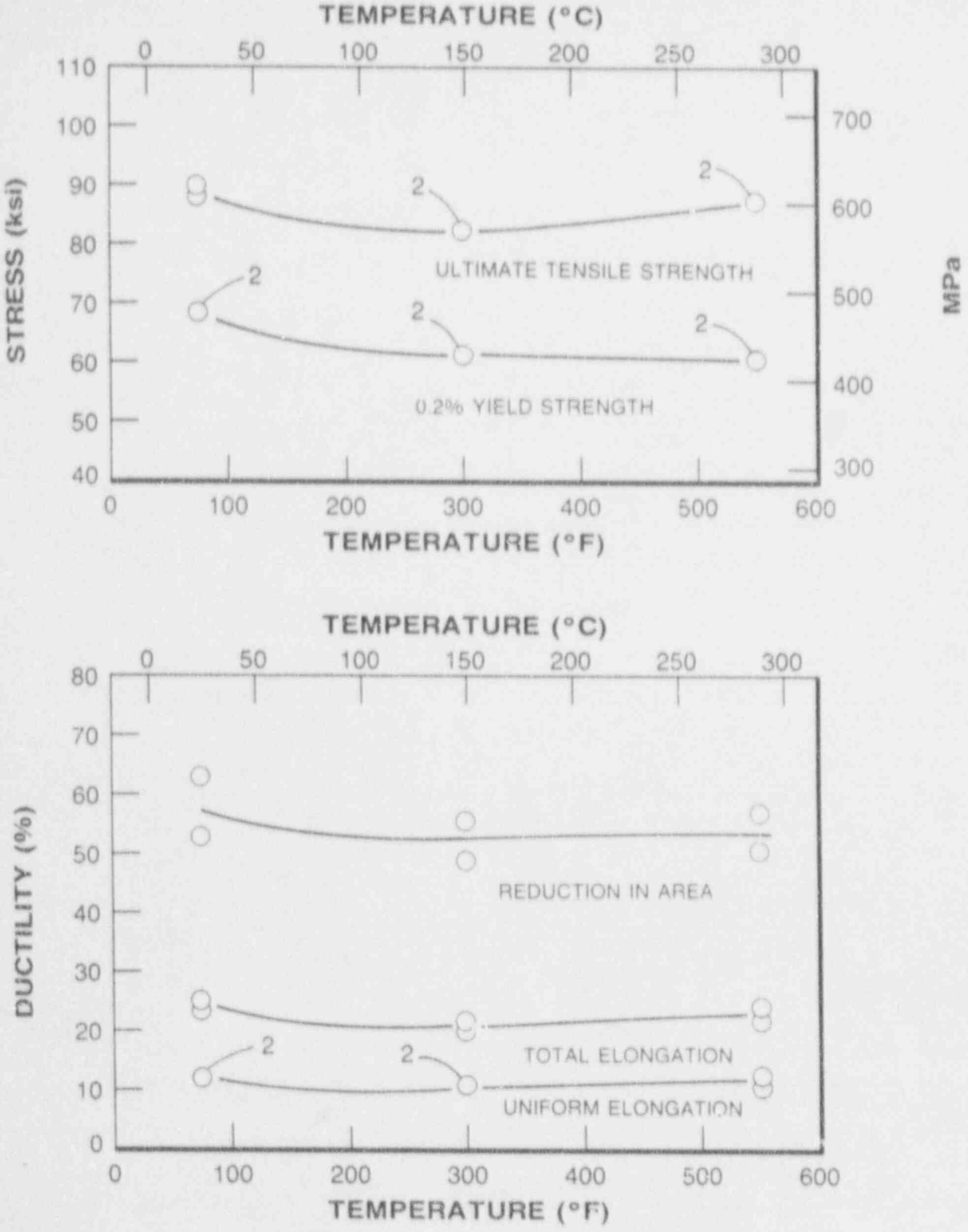


FIGURE 3-6. PREIRRADIATION TENSILE PROPERTIES FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL LOWER SHELL PLATE B8628-1 (TRANSVERSE ORIENTATION)

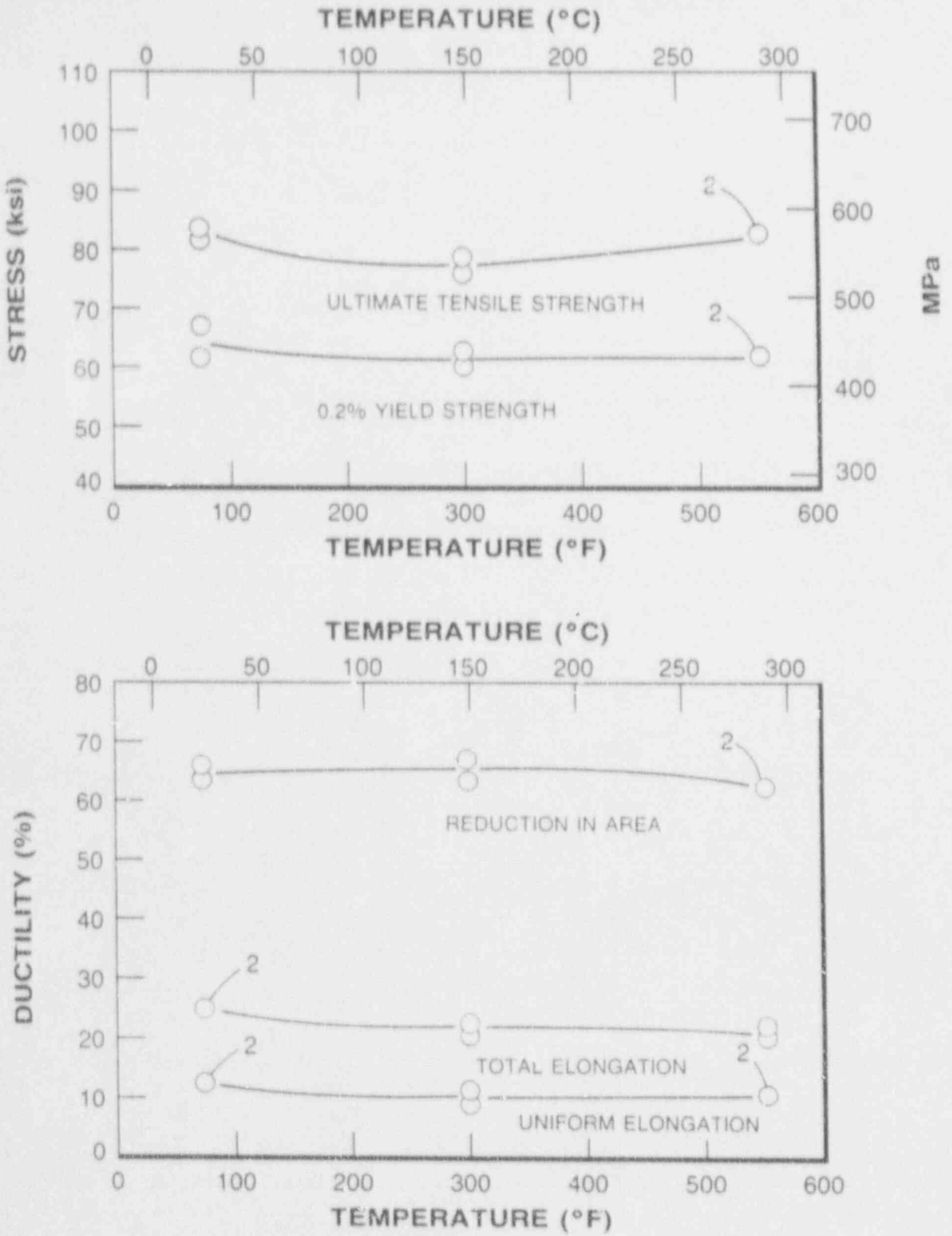


FIGURE 3-7. PREIRRADIATION TENSILE PROPERTIES FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION WELD METAL

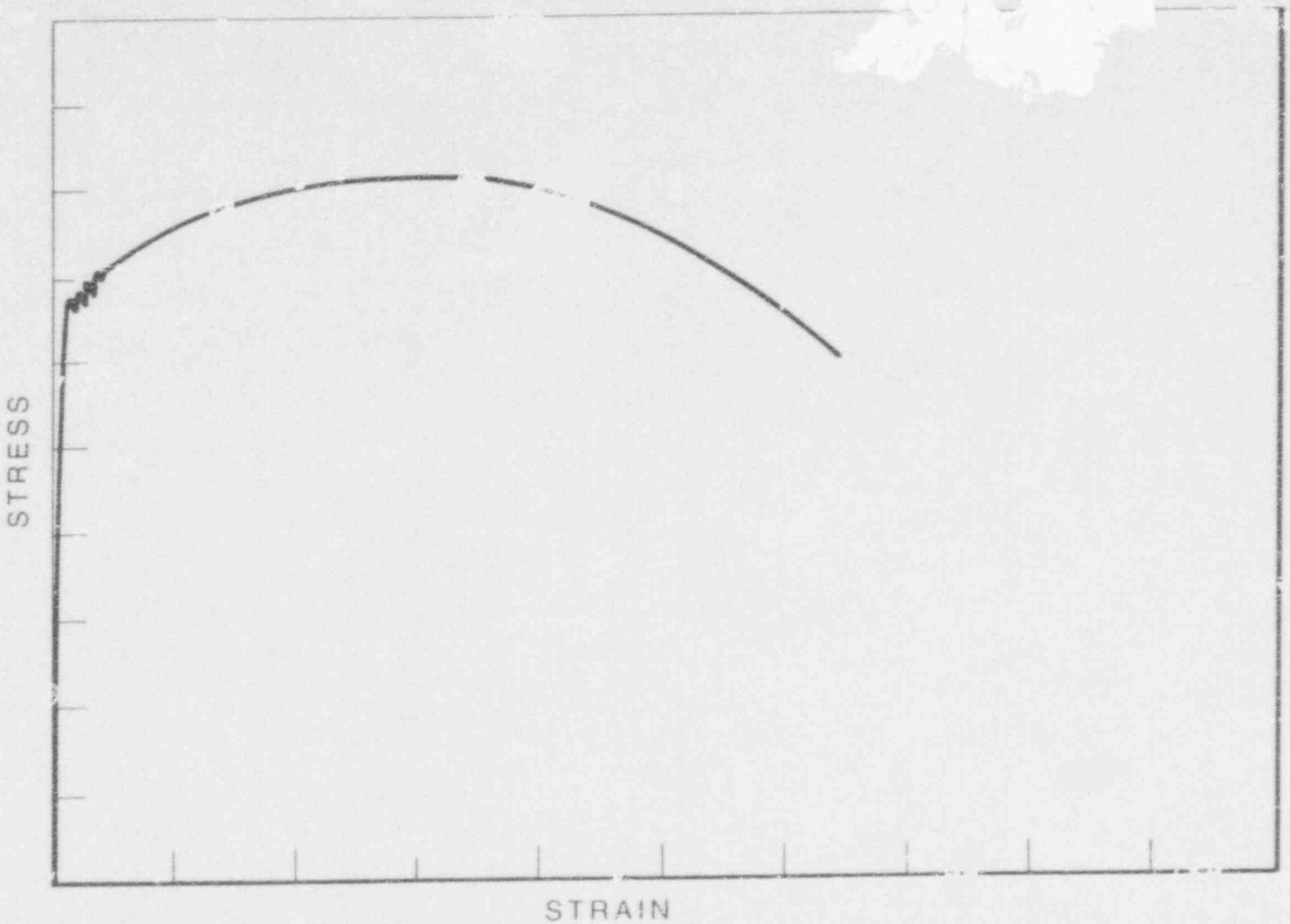


Figure 3-8. Typical Stress-Strain Curve for Tensile Test

SECTION 4
POSTIRRADIATION TESTING

4-1. CAPSULE REMOVAL

The first capsule (Capsule U) should be removed at the end of the first core cycle (1st refueling) as shown in Table 4-1. Subsequent capsules should be removed at 5, 9, and 15 EFPY (Effective Full Power Years) as indicated. Each specimen capsule, removed after exposure, will be transferred to a postirradiation test facility for disassembly and testing of all the specimens.

TABLE 4-1
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Identification	Orientation of Capsules[a]	Lead Factor[b]	Removal Time	Expected Capsule Fluence (n/cm^2)
U	58.5°	4.00	1st Refueling	4.84×10^{18}
Y	241°	3.69	5 EFPY	1.64×10^{19}[c]
V	61°	3.69	9 EFPY	3.21×10^{19}[d]
X	238.5°	4.00	15 EFPY	5.80×10^{19}
W	121.5°	4.00	Stand-By	———
Z	301.5°	4.00	Stand-By	———

a. Reference Irradiation Capsule Assembly Drawing, Figure 2-4.

b. The factor by which the capsule fluence leads the vessels maximum inner wall fluence.

c. Approximate Fluence at ¼-wall thickness at End-of-Life.

d. Approximate Fluence at vessel inner wall at End-of-Life.

4-2. CHARPY V-NOTCH IMPACT TESTS

The testing of the Charpy impact specimens from the lower shell plate B8628-1 weld metal, and HAZ metal in each capsule can be done singly at approximately ten different temperatures. The extra specimens should be used to run duplicate tests at temperatures of interest to develop the complete Charpy impact energy transition curve.

The initial Charpy specimen from the first capsule removed should be tested at room temperature. The test value of this temperature should be compared with preirradiation test data. The test temperature for the remaining specimens should then be adjusted higher or lower so as to develop a complete transition curve. 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

A tensile test specimen from each of the selected irradiated materials shall be tested at a temperature representative of the upper end of the Charpy energy transition region. The remaining tensile specimens from each material shall be tested at the service temperature (550°F) and the midtransition temperature.

4.4 FRACTURE TOUGHNESS TESTS ON 1/2 COMPACT SPECIMENS

In light of current requirements of 10CFR, Part 50, Appendix G and applications of ASME Secition III, Apendix G and Section XI, Appendix A, the ½-inch thick compact specimens should be tested in such a manner as to determine both static, crack initiation, and propagation parameters throughout the temperature range of interest with emphasis on the sharp fracture toughness transition and upper shelf regions consistent with specimen availability. The specimens should thus be staticaily tested in accordance with ASTM E399-81 procedures modified to account for the size of the specimens available.[1] Specific test procedures should include unloading compliance and data interpretation should utilize the Equivalent Energy and J-Integral concepts.[2,3,4]

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1. Witt, F. J., "Fracture Toughness Parameters Obtained from Single Small Specimen Tests", WCAP-9397, October 1978.
2. Buchalet, C. and Mager, T. R., "Experimental Verification of Lower Bound K_{Ic} Values Utilizing the Equivalent Energy Concept", in *Progress in Flaw Growth and Fracture Toughness Testing*, ASTM-STP-536, pp. 281-296, American Society for Testing and Materials, Philadelphia, 1973.
3. Landes, J. D. and Begley, J. A., "Recent Developments in J_{Ic} Testing", in *Developments in Fracture Mechanics Test Methods Standardization*, ASTM-STP-632, pp. 57-81, American Society for Testing and Materials, Philadelphia, 1977.
4. McCabe, D. E., "Determination of R-Curves for Structural Materials Using Nonlinear Mechanics Methods", in *Flaw Growth and Fracture*, ASTM-STP-631, pp. 245-226, American Society for Testing and Materials, Philadelphia, 1977.

Fracture toughness data so obtained will be K_{Ic}, J_{Ic} and dJ/da or engineering estimates thereof. Advantages should be taken of the Charpy impact and tensile data in the selection of initial test temperatures. Test procedures actually performed on the specimens will reflect state-of-the-art at the time of testing.

4.5 POSTIRRADIATION TEST EQUIPMENT

Required minimum equipment for the postirradiation testing operations is as follows:

- Milling machine or special cutoff wheel for opening capsules, dosimeter blocks and spacers.
- Hot cell tensile testing machine with pin-type adapter for testing tensile specimens.
- Hot cell static CT testing machine with clevis and appropriate measuring equipment modified to account for the size of the specimens.
- 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
DESCRIPTION AND CHARACTERIZATION OF THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR VESSEL BELTLINE AND SURVEILLANCE MATERIALS

Based on the initial RT_{NDT}, chemical compostion (copper and phosphorus) and the end-of-life neutron fluence, the reactor vessel lower shell plate B8628-1 is expected to have the highest end-of-life ΔRT_{NDT} using the prediction methods of Regulatory Guide 1.99 Revision 1, and latest ASTM revisions. This material is therefore considered to be the limiting vessel beltine region material and has been used in the reactor vessel surveillance program.

For the surveillance program Combustion Engineering, Inc., supplied Westinghouse with sections of the A533 Grade B Class 1 Steel plate produced by Lukens Steel Company. This steel was used in the fabrication of the Alvin W. Vogtle Unit No. 2 reactor pressure vessel, specifically, from the 9⅝-inch lower shell plate B8628-1. Also supplied was a submerged arc weldment made from sections of lower shell plate B8628-1 and adjacent lower shell plate B8825-1. This test weldment was fabricated using $\frac{3}{16}$ inch Mil B-4 weld filler wire, heat number 87005 and Linde 124 flux, lot number 1061 and is identical to that used by Combustion Engineering, Inc. in the Alvin W. Vogtle Unit No. 2 reactor vessel fabrication process specifically the closing girth seam between the intermediate and lower shell plates.

The chemical analyses, T_{NDT}, RT_{NDT}, upper shelf energy and heat treatment history of all the core region pressure vessel shell plates used in the fabrication of the Alvin W. Vogtle Unit No. 2 reactor pressure vessel are summarized in Tables A-1 thru A-6 respectively. This data is as reported in the vessel fabricators (Combustion Engineering, Inc.) certification reports or from subsequent Westinghouse analyses of similar materials used for the Alvin W. Vogtle Unit No. 2 surveillance program. Weld material identical to that used in the fabrication of all the core region beltline welds[a] has been correlated with the Westinghouse surveillance program test weldment "D" test results and available Combustion Engineering, Inc. weld certification reports and their surveillance program test weldment "C". This data is also reported in Tables A-3 thru A-6 of this Appendix.

a. The beltline welds are considered to include the intermediate and lower shell plate longitudinal seams and the closing intermediate to lower shell girth seam.

TABLE A-1

CHEMICAL ANALYSIS OF THE INTERMEDIATE SHELL PLATES USED IN THE CORE REGION OF THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL

Element	Chemical Compositon[a] (weight %)		
	Plate R-4-1	Plate R-4-2	Plate R-4-3
C	.21	.20	.25
Mn	1.28	1.25	1.37
P	.009	.009	.009
S	.010	.009	.012
Si	.23	.22	.20
Ni	.64	.62	.59
Mo	.57	.55	.56
Cr	.03	.03	.03
Cu	.06	.05	.05
Al	.026	.024	.017
Co	.012	.011	.008
Pb	not detected	not detected	not detected
W	< .01	< .01	< .01
Ti	< .01	< .01	< .01
Zr	< .001	< .001	< .001
V	.005	.004	.005
Sn	.005	.005	.004
As	.007	.007	.008
Cb	< .01	< .01	< .01
N_2	.009	.008	.008
B	< .001	< .001	< .001

a. Chemical Analysis by Combustion Engineering, Inc.

TABLE A-2

CHEMICAL ANALYSIS OF THE LOWER SHELL PLATES USED IN THE CORE REGION OF THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL

Element	Chemical Compositon (weight %)			
	Plate[b]		Plate	
	B8825-1	R-8-1	B8628-1[a] [b]	B8628-1[a] [c]
C	.23	.21	.24	.23
Mn	1.31	1.34	1.34	1.30
P	.006	.007	.007	.007
S	.014	.012	.016	.014
Si	.25	.25	.25	.23
Ni	.59	.62	.59	.59
Mo	.59	.62	.59	.50
Cr	.02	.02	.02	.07
Cu	.05	.06	.05	.05
Al	.031	.019	.029	.034
Co	.004	.011	.004	.008
Pb	not detected	not detected	not detected	< .07
W	< .01	< .01	< .01	< .05
Ti	< .01	< .01	< .01	.005
Zr	< .001	< .001	< .001	< .03
V	.004	.004	.004	< .005
Sn	.004	.004	.017	.007
As	.006	.008	.007	.008
Cb	< .01	< .01	< .01	< .05
N_2	.009	.011	.008	.007
B	< .001	< .001	< .001	.008

a. Surveillance program test plate.

b. Chemical Analysis by Combustion Engineering, Inc.

c. Chemical Analysis by Westinghouse.

TABLE A-3

CHEMICAL ANALYSIS OF THE WELD METAL USED FOR THE INTERMEDIATE AND LOWER SHELL PLATES LONGITUDINAL SEAMS FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL

Element	Chemical Compositon (weight %)
	Wire Flux Test Weld Sample[a] (Weld Wire Heat No. 87005 Linde 0091 Flux, Lot No. 0145)
C	.15
Mn	1.34
P	.007
S	.011
Si	.13
Ni	.13
Mo	.55
Cr	———
Cu	.07
Al	———
Co	———
Pb	———
W	———
Ti	———
Zr	———
V	.005
Sn	———
As	———
Cb	———
N_2	———
B	———

a. Chemical Analysis by Combustion Engineering, Inc.

TABLE A-4

CHEMICAL ANALYSIS OF THE WELD METAL USED FOR THE INTERMEDIATE TO LOWER SHELL CLOSING GIRTH SEAM OF THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL

Element	Chemical Compositon (weight %)	
	Wire Flux Test Weld Sample[a] (Weld Wire Heat No. 87005 Linde 124 Flux Lot No. 1061)	Surveillance Weldment Test Plate D[b]
C	.075	.099
Mn	1.27	1.25
P	.007	.008
S	.010	.013
Si	.50	.43
Ni	.12	.17
Mo	.52	.47
Cr	.07	.061
Cu	.06	.040
Al	----	.015
Co	----	.002
Pb	----	< .01
W	----	< .01
Ti	----	< .001
Zr	----	< .01
V	.004	< .004
Sn	----	< .001
As	----	.003
Cb	----	< .002
N_2	----	.002
B	----	.009

a. Chemical Analysis by Combustion Engineering, Inc.

b. Chemical Analysis by Westinghouse of the Surveillance Program test plate "D" representative of the closing girth seam. Weld wire Heat No. 87005, Linde 124 Flux Lot No. 1061.

TABLE A-5

T_{NDT}, RT_{NDT} AND UPPER SHELF ENERGY FOR THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION SHELL PLATES AND WELD METAL

Material	T_{NDT} [a] [b]		RT_{NDT}		Upper Shelf [a] [c] Energy	
	(°C)	(°F)	(°C)	(°F)	(J)	(ft lb)
Intermediate Shell Plates:						
R-4-1	−29	−20	−23	10	129	95
R-4-2	−23	−10	−23	10	141	104
R-4-3	−18	0	−1	30	114	84
Lower Shell Plates:						
B8825-1	−29	−20	4	40	113	83
R-8-1	−29	−20	4	40	118	87
B8628-1	−29	−20	10	50	115	85

a. Data obtained from Combustion Engineering, Inc. Reactor Vessel Material Certification Reports

b. Drop weight data obtained from the transverse material properties (normal to the major working direction)

c. From impact data obtained from the transverse material properties (normal to the major working direction)

Material	T_{NDT} [d]		RT_{NDT}		Upper Shelf [d] Energy	
	(°C)	(°F)	(°C)	(°F)	(J)	(ft lb)
Intermediate and Lower Shell Longitudinal Weld Seams (Weld Wire Heat No. 87005, Linde 0091 Lot No. 0145)	−23	−10	−23	−10	206	152
Closing Girth Weld Seam Joining the Intermediate to Lower Shell (Weld Wire Heat No. 87005, Linde 124 Flux, Lot No. 1061)	−46	−50	−34	−30	122	90

d. Data obtained from Combustion Engineering, Inc. Wire/Flux Weld Deposit Material Certification Tests.

TABLE A-6
HEAT TREATMENT HISTORY OF THE ALVIN W. VOGTLE UNIT NO. 2 REACTOR PRESSURE VESSEL CORE REGION SHELL PLATES AND WELD SEAMS

Material	Temperature (°F)	Time (hr)	Cooling
Intermediate Shell Plates R-4-1 R-4-2 R-4-3	Austenitizing: 1600 ± 25 (871°C)	4 [a]	Water-quenched
	Tempered: 1225 ± 25 (663°C)	4 [a]	Air-cooled
	Stress Relief: 1150 ± 50 (621°C)	16.5[b]	Furnace-cooled
Lower Shell Plates B8825-1 R-8-1 B8628-1	Austenitizing: 1600 ± 25 (871°C)	4 [a]	Water-quenched
	Tempered: 1225 ± 25 (663°C)	4 [a]	Air-cooled
	Stress Relief: 1150 ± 50 (621°C)	12.0[b]	Furnace-cooled
Intermediate Shell Longitudinal Seam Welds	Stress Relief: 1150 ± 50 (621°C)	16.5[b]	Furnace-cooled
Lower Shell Longitudinal Seam Welds		12.0[b]	Furnace-cooled
Intermediate to Lower Shell Girth Seam Weld	Local Stress Relief 1150 ± 50 (621°C)	5.0	Furnace-cooled
Surveillance Program Test Material			
Surveillance Program Weldment Test Plate "D" (Representative of closing Girth Seam)	Post Weld Stress Relief: 1150 ± 50 (621°C)	6.0 [c]	Furnace-cooled

a. Lukens Steel Company, Combustion Engineering, Inc. Certification Reports

b. Stress Relief includes the Intermediate to Lower Shell Closing Girth Seam Post Weld Heat Treatment.

c. The Stress Relief Heat Treatment received by the Surveillance Test Weldment has been simulated.