WESTINGHOUSE CLASS 3

Attachment 2

WCAP-8512

AMERICAN ELECTRIC POWER COMPANY
DONALD C. COOK UNIT NO. 2
REACTOR VESSEL RADIATION
SURVEILLANCE PROGRAM

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ABSTRACT

A pressure vessel steel surveillance program was developed for the American Electric Power Company, Donald C. Cook Unit No. 2, to obtain information on the effects of radiation on the reactor vessel material under operating conditions. The program comprises the evaluation of the radiation effects based on comparison preirradiation testing of a selected group of specimens to determine toughness properties of the reactor pressure vessel. Continuous monitoring of these specimens within the reactor pressure vessel provides data on the integrity of the vessel in terms of adequate toughness properties. A description of the surveillance capsules and preirradiation test results is also included.

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SECTION 1
PURPOSE AND SCOPE

The purpose of the American Electric Power Company, Donald C. Cook Unit No. 2, surveillance program is to obtain information on the effects of radiation on the reactor vessel materials of the reactor during normal operating conditions. Surveillance material is selected as the most limiting material based on surveillance selection procedures which are outlined in ASTM E185-73, Annex A1. Evaluation of the radiation effects is based on the preirradiation testing of Charpy V-notch, tensile, and dropweight specimens, and postirradiation testing of Charpy V-notch, tensile, and wedge-opening-loading (WOL) specimens.

Current reactor pressure vessel material test requirements and acceptance standards use the reference nil-ductility temperature, RT_{NDT}, as a basis. RT_{NDT} is determined from the dropweight nil-ductility transition temperature, NDTT, or the weak (transverse oriented) direction 50 ft lb Charpy V-notch impact temperature (which ever value is greater) as defined by the following equation:

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

or

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

where

RT_{NDT} = Reference nil-ductility temperature

NDTT = Nil-ductility transition temperature as per ASTM E208

$T_{50(35)}$ = 50 ft lb temperature from transverse oriented Charpy V-notch impact specimens (or the 35 mil-temperature, if it is greater[1])

1. In the case where at least 35 mils lateral expansion is not obtained at the 50 ft lb temperature, the temperature at which 35 mils lateral expansion occurs is used.

An empirical relationship between RT_{NDT} and fracture toughness for reactor vessel steels has been developed and is presented in appendix G, section III, of the ASME Boiler and Pressure Vessel Code (Protection Against Non-Ductile Failure). The relationship can be employed to set allowable pressure-temperature relationships, based on fracture mechanics concepts, for the normal operation of reactors. Appendix G of the ASME Boiler and Pressure Vessel Code defines an acceptable method for calculating these limitations.

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

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

1. L. F. Porter, "Radiation Effects in Steel," in *Materials in Nuclear Applications*, ASTM STP-276, pp. 147-195, American Society for Testing and Materials, Philadelphia, 1960.
2. L. E. Steele and J. R. Hawthorne, "New Information on Neutron Embrittlement and Embrittlement Relief of Reactor Pressure Vessel Steels," NRL-6160, August 1964.
3. U. Potapovs and J. R. Hawthorne, "The Effect of Residual Element on 550°F Irradiation Response of Selected Pressure Vessel Steels and Weldments," NRL-6803, September 1968.
4. L. E. Steele, "Structure and Composition Effects on Irradiation Sensitivity of Pressure Vessel Steels," in *Irradiation Effects on Structural Alloys for Nuclear Reactor Applications*, 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," in *The Effects of Radiation on Structural Metals*, ASTM STP-426, pp. 260-277, American Society for Testing and Materials, Philadelphia, 1967.
6. M. J. Manjoine, "Biaxial Brittle Fracture Tests," *Trans. Am. Soc. Mech. Engrs. 87*, Series D, 293-298 (1965).
7. L. Porse, "Reactor-Vessel Design Considering Radiation Effects," *Trans. Am. Soc. Mech. Engrs. 86*, Series D, 743-749 (1964).
8. R. E. Johnson, "Fracture Mechanics: A Basis for Brittle Fracture Prevention," WAPD-TM-505, 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, August 1965.
10. W. K. Wilson, "Analytic Determination of Stress Intensity Factors for the Manjoine Brittle Fracture Test Specimen," WERL-0029-3, August 1965.
11. R. E. Johnson and E. J. Pasierb, "Fracture Toughness of Irradiated A302-B Steel as Influenced by Microstructure," *Trans. Amer. Nucl. Soc. 9*, 390-392 (1966).

Postirradiation testing of the Charpy impact specimens provides a guide for determining pressure-temperature limits on the plant. A temperature shift in the reference temperature will occur in the irradiated Charpy impact specimen test data as a result of radiation exposure at plant temperatures. These data can then be reviewed to verify or establish new pressure-temperature limits of the vessel during start-up and cooldown. This allows a check of the predicted shift in the reference temperature. The postirradiation test results on the WOL specimens provide actual fracture toughness properties for the vessel material. These properties may be used for subsequent evaluation as per the methods outlined in the ASME Code, appendix G.

Eight material test capsules are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are located in guide tubes attached to the thermal shield. The capsules contain Charpy impact, WOL, and tensile specimens from the limiting core region plate. This plate is the reactor vessel intermediate shell plate adjacent to the core region and is 8 3/4 inches thick. Charpy impact, WOL, and tensile specimens obtained from the representative core region weld metal, and Charpy impact specimens from the weld material heat-affected zone (HAZ), are also located in the capsules. In addition, dosimeters to measure the integrated neutron flux and thermal monitors to measure temperature are located in each of the eight material test capsules.

The thermal history or heat treatment given to these specimens is similar to the thermal history of the reactor vessel material, except that the postweld heat treatment received by the specimens has been simulated (appendix A).

SECTION 2
SAMPLE PREPARATION

2-1. PRESSURE VESSEL MATERIAL

Reactor vessel material was supplied by The Chicago Bridge and Iron Company from the vessel intermediate shell plate C5521-2. A submerged arc weldment which joined sections of material from this plate and lower shell plate C5592-1 was also supplied by The Chicago Bridge and Iron Company. Data on the pressure vessel material are presented in appendix A.

2-2. MACHINING

Test material was obtained from the intermediate shell course plate when the thermal heat treatment was complete and the plate formed. All test specimens were machined from the 1/4-thickness section of the plate after a simulated postweld stress-relieving treatment on the test material was performed. The test specimens represent material taken at least one plate thickness (8 3/4 inches) from the quenched ends of the plate. Specimens were machined from weld and heat-affected zone (HAZ) material of a stress-relieved weldment which joined sections of the intermediate and lower shell plates. All HAZ specimens were obtained from the weld HAZ of intermediate shell plate C5521-2

2-3. Charpy V-Notch Impact Specimens (Figure 2-1)

Charpy V-notch impact specimens from intermediate shell plate C5521-2 were machined in both the longitudinal orientation (longitudinal axis of specimen parallel to major working direction) and transverse orientation (longitudinal axis of specimen perpendicular to major working direction). The core region weld Charpy impact specimens were machined from the weldment such that the long dimension of the Charpy was normal to the weld direction; the notch was machined such that the direction of crack propagation in the specimen was in the weld direction.

2-4. Tensile Specimens (Figure 2-2)

Tensile specimens were machined with the longitudinal axis of the specimen perpendicular to the major working direction of the plate.

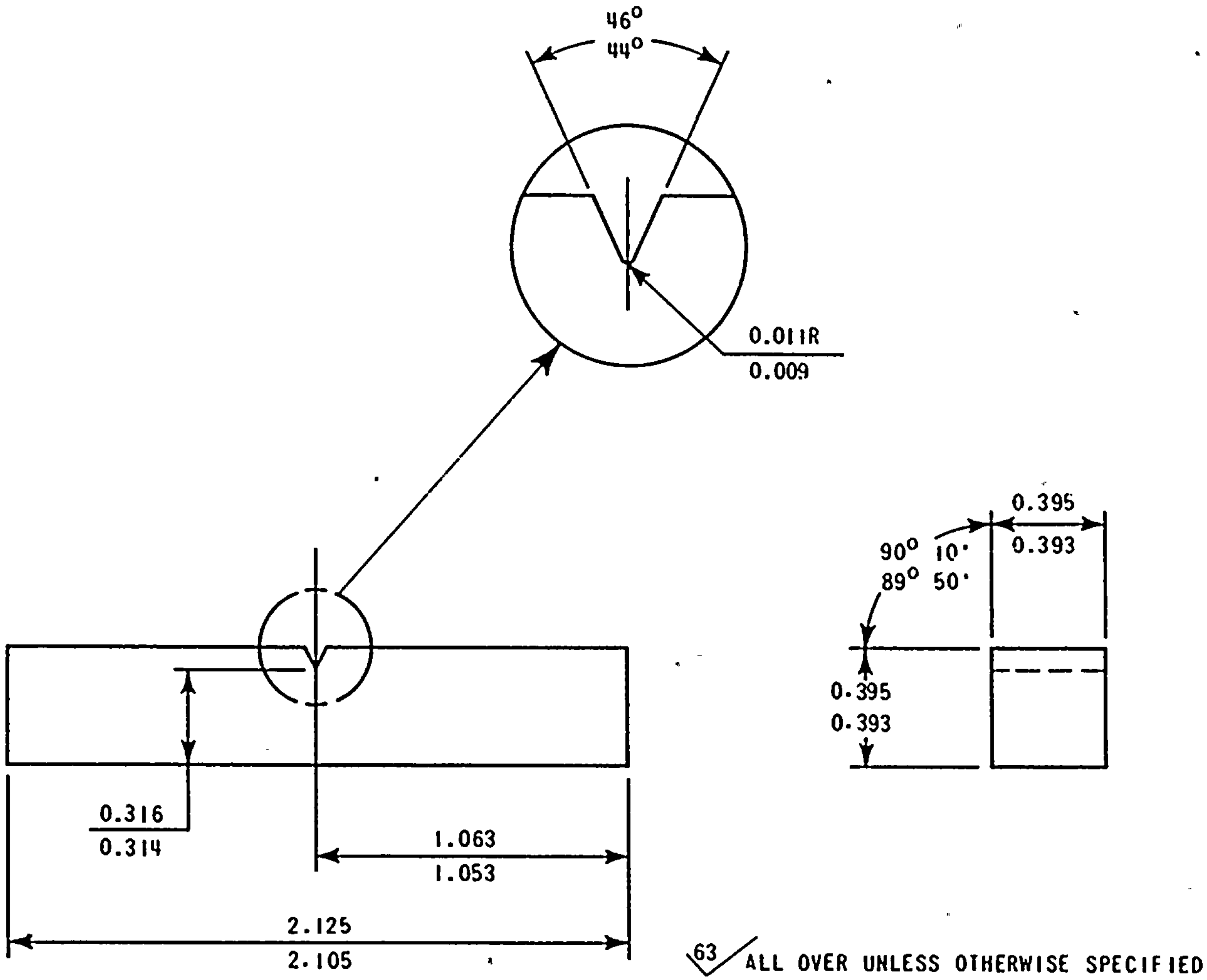


Figure 2-1. Charpy V-Notch Impact Specimen

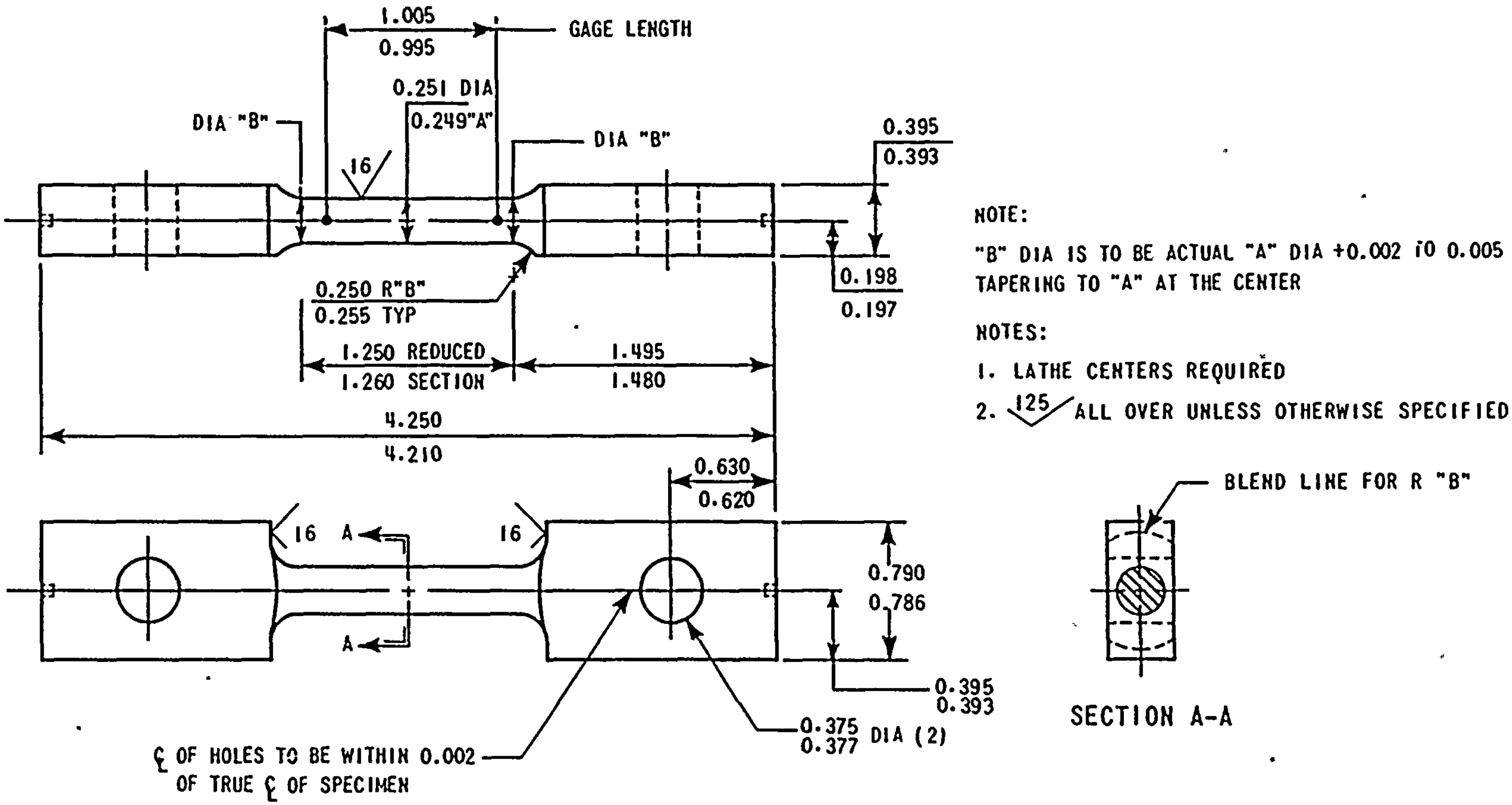


Figure 2-2. Tensile Specimen

2-5. Wedge Opening Loading Specimens (Figure 2-3)

Wedge opening loading (WOL) test specimens were machined along the transverse orientation so that the specimen would be loaded perpendicular to the major working direction of the plate and the simulated crack would propagate along the longitudinal direction. All specimens were fatigue precracked according to ASTM E399-70T.

2-6. MONITORS

2-7. Dosimeters

Eight capsules of the type shown in figure 2-4 contain dosimeters of copper, iron, nickel, and aluminum-cobalt wire (cadmium-shielded and unshielded), neptunium-237, and uranium-238. The dosimeters are used to measure the integrated flux at specific neutron energy levels.

2-8. Thermal Monitors

The capsules contain two low-melting-point eutectic alloys so that the maximum temperature attained by the test specimens during irradiation can be accurately determined. The thermal monitors are sealed in Pyrex tubes and then inserted in spacers (figure 2-4). The two eutectic alloys and their melting points are as follows:

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

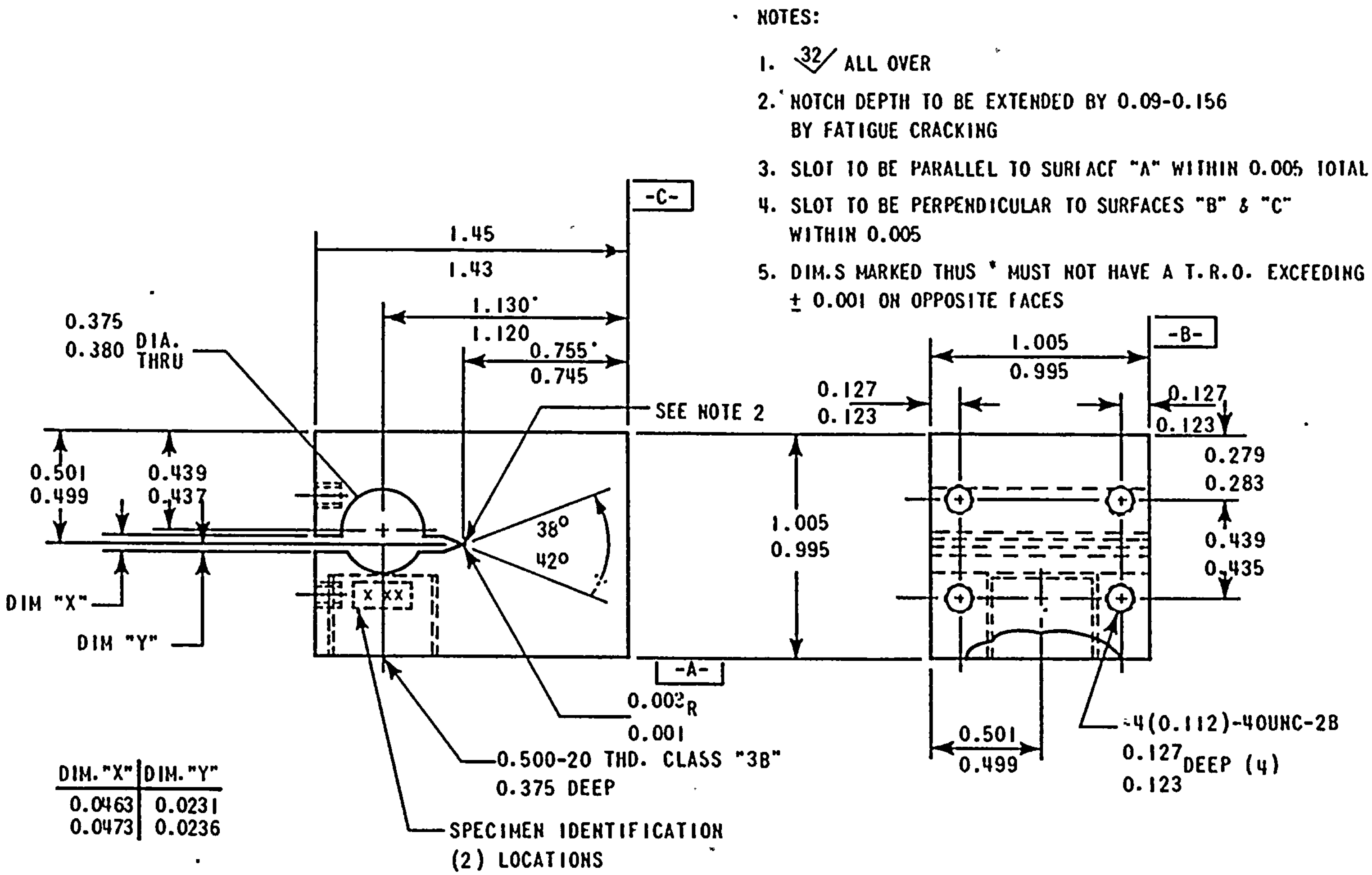
2-9. SURVEILLANCE CAPSULES

2-10. Capsule Preparation

The specimens were seal-welded into a square austenitic stainless steel capsule to prevent corrosion of specimen surfaces during irradiation. The capsules were then hydrostatically tested in demineralized water to collapse the capsule on the specimens so that optimum thermal conductivity between the specimens and the reactor coolant could be obtained. The capsules were helium-leak tested as a final inspection procedure. Finally, the capsules were coded S, T, U, V, W, X, Y, and Z. Fabrication details and testing procedures are listed in the notes in figure 2-4.

2-11. Capsule Loading

Upon receipt, the eight test capsules are positioned in the reactor between the thermal shield and the vessel wall at the locations shown in figure 2-4. Each capsule contains 44 Charpy V-notch specimens, 4 tensile specimens, and 4 WOL specimens.



DIM. "X"	DIM. "Y"
0.0463	0.0231
0.0473	0.0236

Figure 2-3. Wedge Opening Loading Specimen

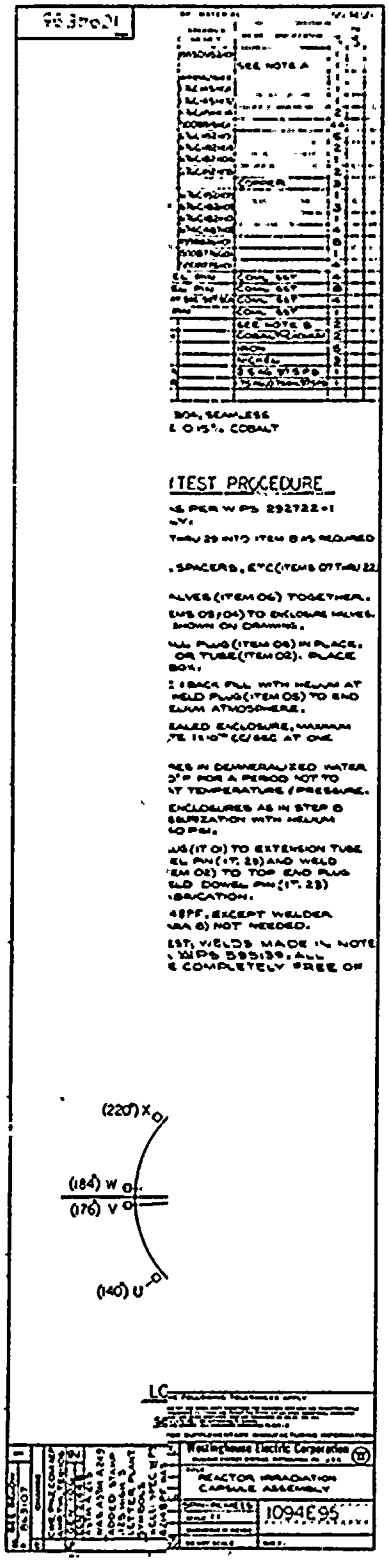


Figure 2-4. Irradiation Capsule Assembly

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

Dosimeters of pure iron, nickel, and copper, aluminum-0.15 percent cobalt, and cadmium-shielded aluminum-0.15 percent cobalt, wires are secured in holes drilled in spacers located in the capsule positions shown in figure 2-4. Each capsule also contains a dosimeter block (figure 2-5) which is located at the center of the capsule. Two cadmium-oxide shielded capsules, each containing isotopes of either U^{238} or Np^{237} (both 99.9 percent pure) 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. The amounts of each are presented in table 2-2. Both of them are held in a 3/8-inch long by 1/4-inch OD sealed brass tube and stainless steel tube, respectively. Each tube is placed in a 1/2-inch-diameter hole in the dosimeter block (one U^{238} and one Np^{237} tube per block), and the space around the tube filled with cadmium oxide. After placement of this material each hole is 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.

TABLE 2-1

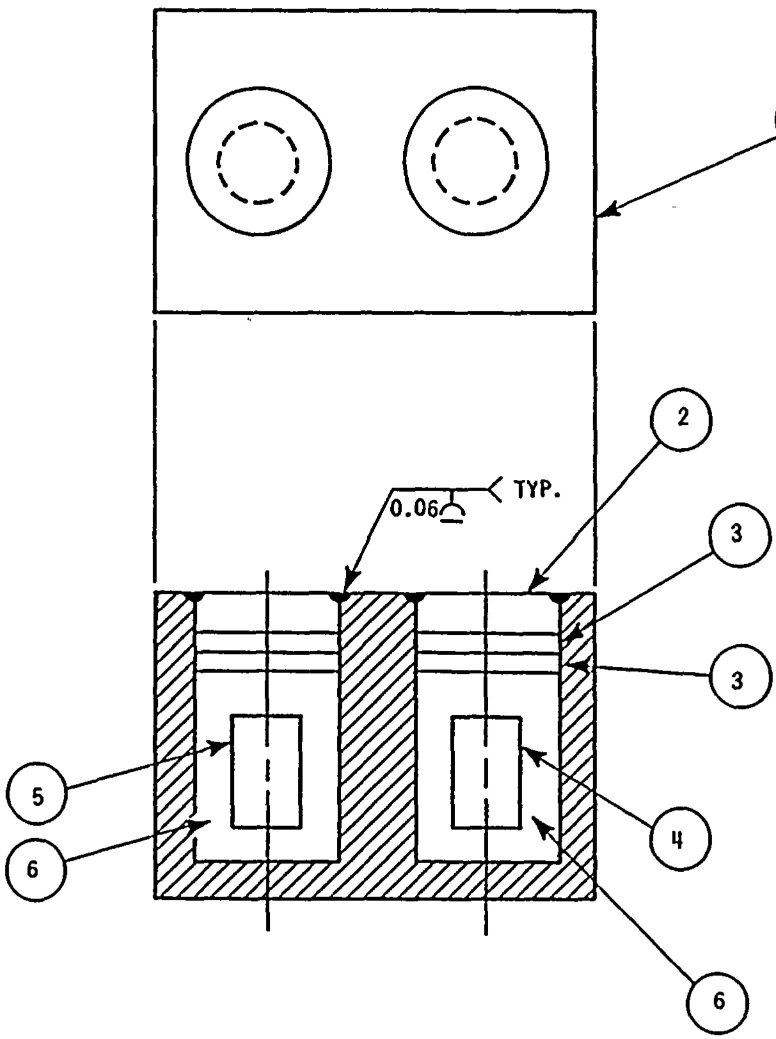
TYPE AND NUMBER OF SPECIMENS IN THE DONALD C. COOK UNIT NO. 2 SURVEILLANCE TEST CAPSULES

Material	Capsule S,V,W, and X			Capsule T,U,Y, and Z		
	Charpy	Tensile	WOL	Charpy	Tensile	WOL
Plate C5521-2 (longitudinal)	8	--	--	8	--	--
Plate C5521-2 (transverse)	12	2	4	12	2	--
Weld Metal	12	2	--	12	2	4
HAZ	12	--	--	12	--	--

TABLE 2-2

QUANTITY OF ISOTOPES CONTAINED IN THE DOSIMETER BLOCKS

Isotope	Weight (mg)	Compound	Weight (mg)
Np^{237}	12 ± 1	NpO_2	20 ± 1
U^{238}	12	U_3O_8	14.25



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

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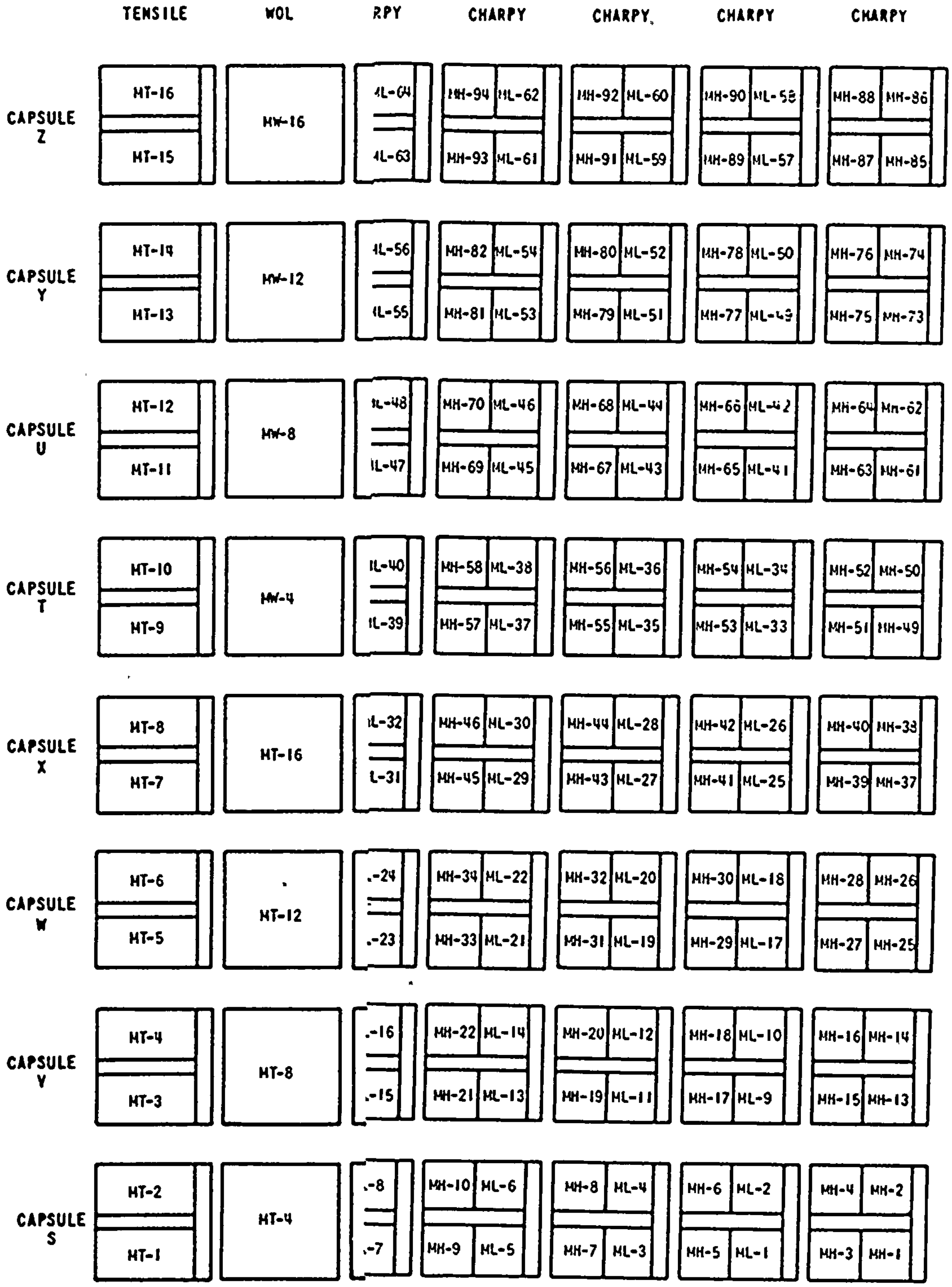
Figure 2-5. Dosimeter Block Assembly



SPECIMEN CODE: MT - PLAT
ML - PLAT
MW - WELD
MH - WELD

Figure 2-6. Location of Specimens in the Reactor Surveillance Test Capsules

SECTION 3
PREIRRADIATION TESTING

3-1. CHARPY V-NOTCH IMPACT TESTS

Charpy V-notch impact tests were performed on the vessel intermediate shell plate C5521-2, at various temperatures from -50° to 210°F to obtain a full 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). Charpy impact tests were performed on weld metal and HAZ material at various temperatures from -100° to 300°F. The results are reported in tables 3-3 and 3-4 and figures 3-3 and 3-4, respectively.

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

3-2. TENSILE TESTS

Tensile tests were performed on the vessel intermediate shell plate C5521-2 (in the transverse orientation) and the weld metal at room temperature, 300°F, and 550°F. The results are shown in table 3-5 and figures 3-5 and 3-6.

Tensile tests for the intermediate shell plate and weld metal were performed on an Instron TT-C tensile testing machine using the standard Instron gripping devices. A full stress-strain curve was obtained for each specimen using a Baldwin-Lima-Hamilton Class B-1 extensometer and chart recorder, the latter calibrated to the extensometer. The method of measuring and controlling speeds for tensile tests on the Instron TT-C are governed by ASTM A370-68 (Mechanical Testing of Steel Products).

The Instron TT-C tensile testing machine and the Baldwin-Lima-Hamilton extensometer are calibrated by test equipment which has been certified by the National Bureau of Standards. A typical stress-strain curve is shown in figure 3-7.

TABLE 3-1

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE DONALD C. COOK NO. 2 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE C5521-2 (LONGITUDINAL ORIENTATION)

Test Temp (°F)	Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
0	15	18	12
0	18	19	13
0	15	18	11
25	26	30	20
25	31	25	23
25	43	25	29
50	52	35	38
50	47	35	37
50	46	35	34
70	65	42	47
70	65	42	49
70	76	55	54
100	91	65	66
100	98	70	76
100	90	62	67
125	126	85	78
125	114	77	79
125	103	75	70
210	122	100	83
210	132	100	86
210	128	100	84

TABLE 3-2

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA FOR THE DONALD C. COOK NO. 2 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE C5521-2 (TRANSVERSE ORIENTATION)

Test Temp (°F)	Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
–50	5.5	5	0
–50	6.0	5	1
–50	6.0	5	0
10	39.0	29	27
10	29.0	25	17
10	25.0	30	18
70	43.0	40	32
70	42.0	43	33
70	39.0	43	28
100	66.0	60	47
100	71.5	63	53
100	68.0	65	49
120	67.5	58	56
120	76.0	65	60
120	75.0	72	59
210	81.0	100	64
210	88.0	100	63
210	90.0	100	66

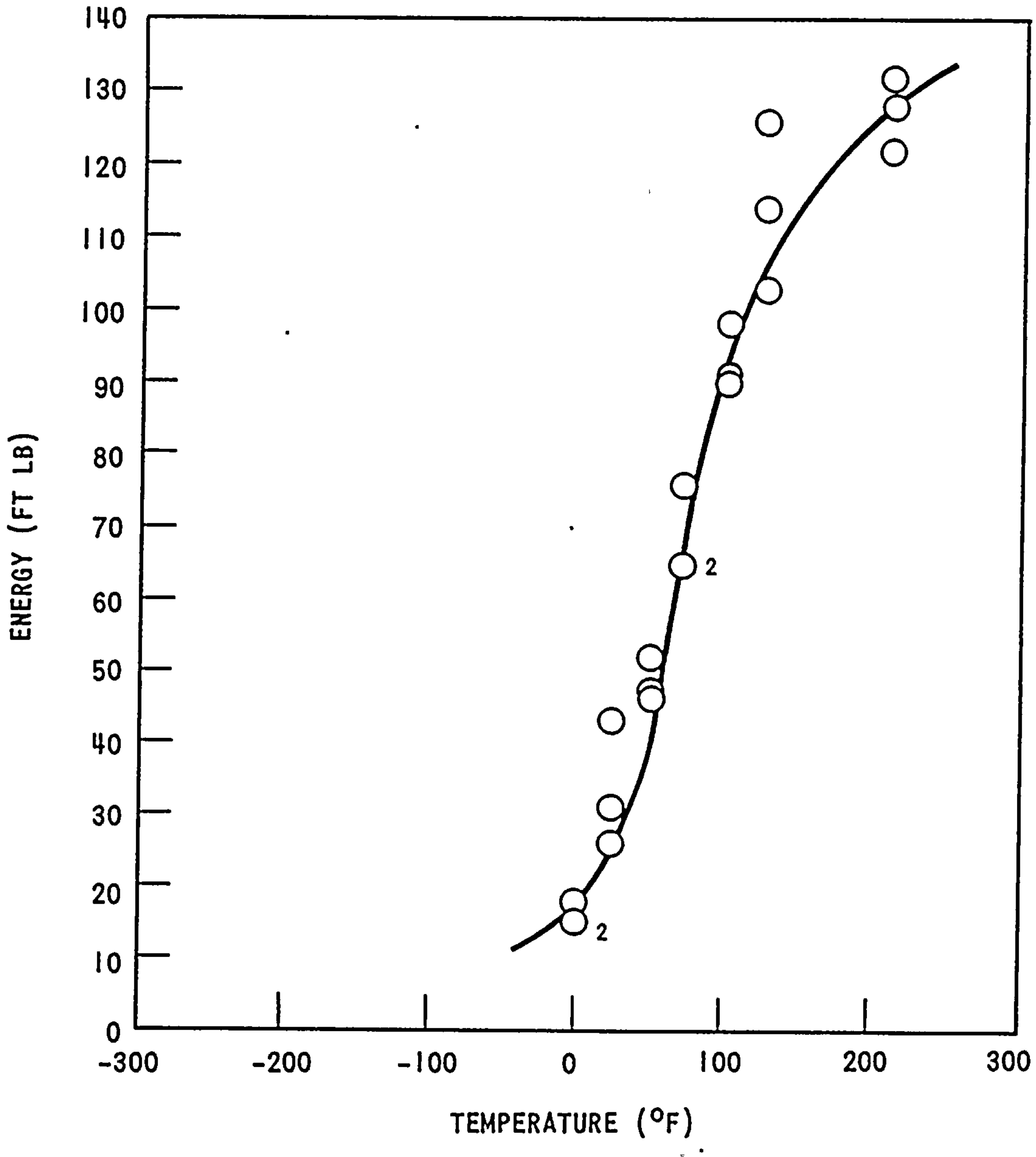


Figure 3-1. Preirradiation Charpy V-Notch Impact Energy Curve for the Donald C. Cook Unit No. 2 Reactor Pressure Vessel Intermediate Shell Plate C5521-2 (Longitudinal Orientation)

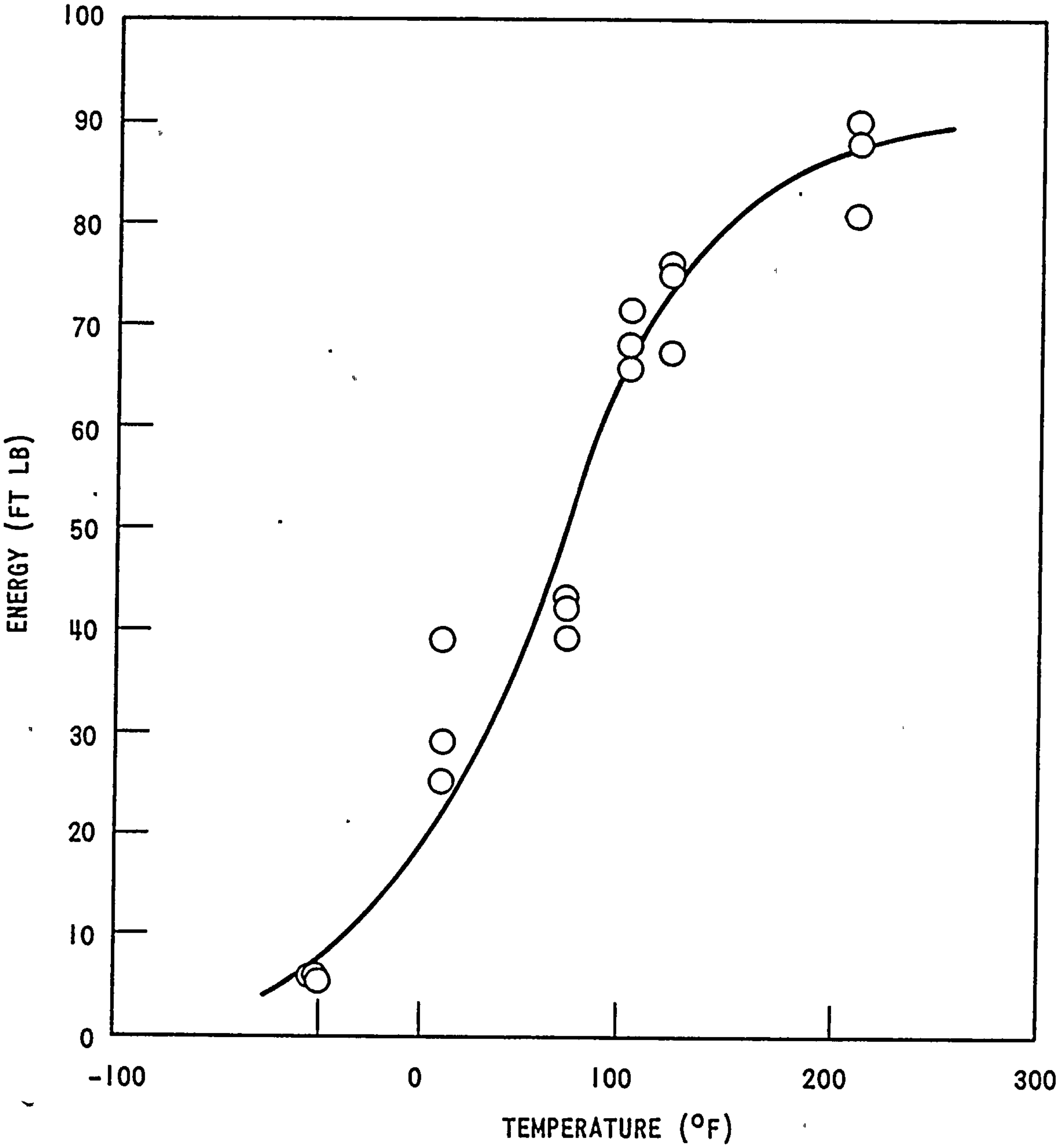


Figure 3-2. Preirradiation Charpy V-Notch Impact Energy Curve for the Donald C. Cook Reactor Pressure Vessel Intermediate Shell Plate C5521-2 (Transverse Orientation)

TABLE 3-3

PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE DONALD C. COOK NO. 2 REACTOR PRESSURE
VESSEL CORE REGION WELD METAL

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
–25	20.0	48	20
–25	22.0	30	17
–25	31.5	40	25
20	32.0	38	24
20	35.0	47	28
20	33.0	50	27
60	58.0	74	48
60	47.0	65	37
60	39.0	50	29
100	74.0	95	63
100	56.0	85	47
100	65.0	95	53
210	72.0	100	68
210	70.0	100	63
210	77.0	100	64
300	72.0	98	66
300	79.0	100	71
300	81.0	100	70

TABLE 3-4
PREIRRADIATION CHARPY V-NOTCH IMPACT DATA
FOR THE DONALD C. COOK NO. 2 REACTOR PRESSURE
VESSEL CORE REGION WELD HEAT-AFFECTED ZONE MATERIAL

Test Temp (°F)	Impact Energy (ft lb)	Shear (%)	Lateral Expansion (mils)
–100	21.0	30	12
–100	5.0	12	1
–100	14.0	29	8
–50	34.0	35	16
–50	23.0	27	21
–50	70.5	53	39
–25	89.0	65	52
–25	70.0	60	43
–25	90.0	60	52
0	95.0	70	59
0	76.0	65	52
0	130.0	100	75
50	84.0	90	55
50	67.0	85	48
50	136.0	100	76
125	95.0	95	66
125	104.0	99	75
125	82.0	90	71
210	147.0	100	77
210	113.0	100	80
210	86.0	100	71

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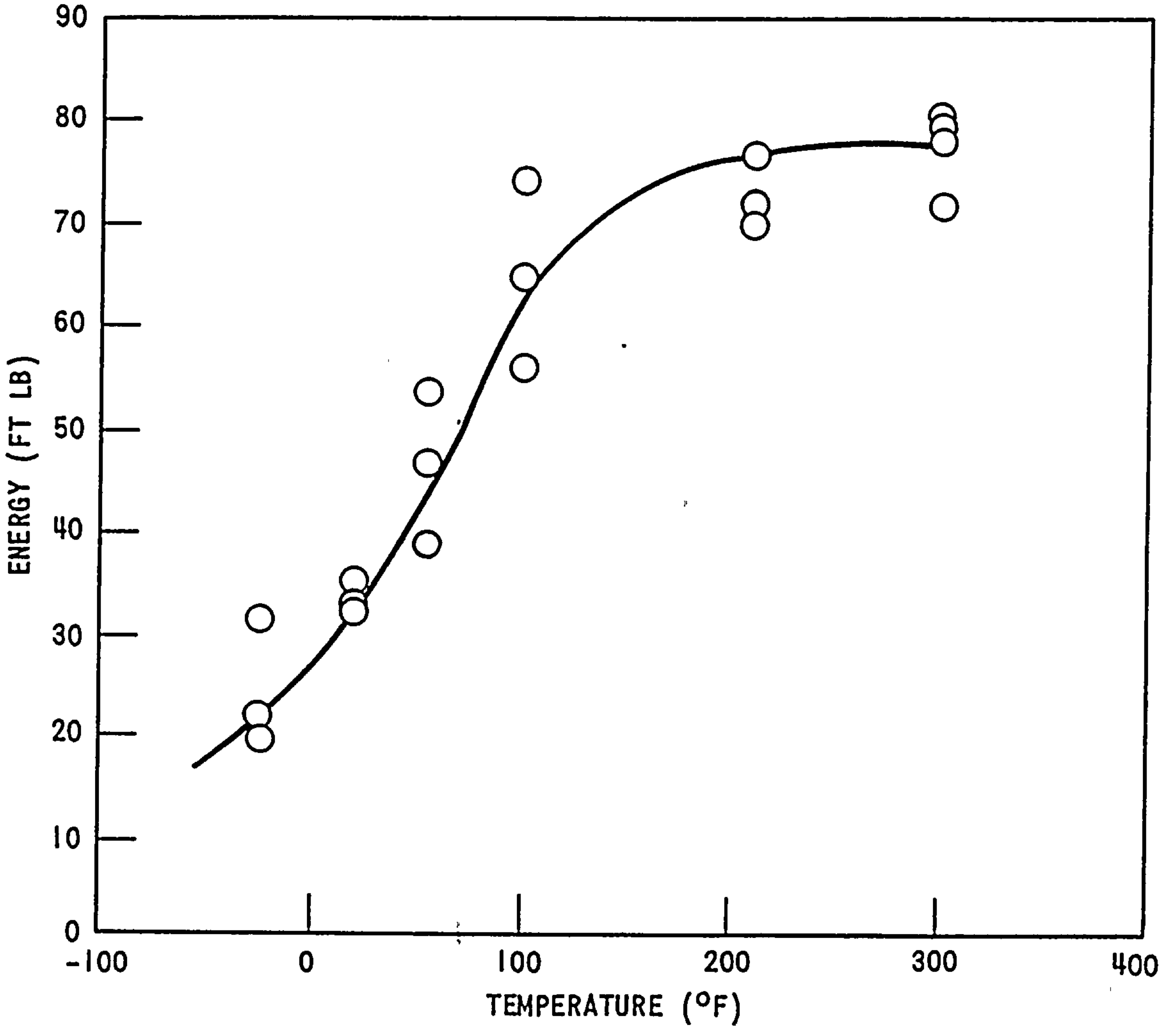

Figure 3-3. Preirradiation Charpy V-Notch Impact Energy Curve for the Donald C. Cook Unit No. 2 Reactor Pressure Vessel Core Region Weld Metal

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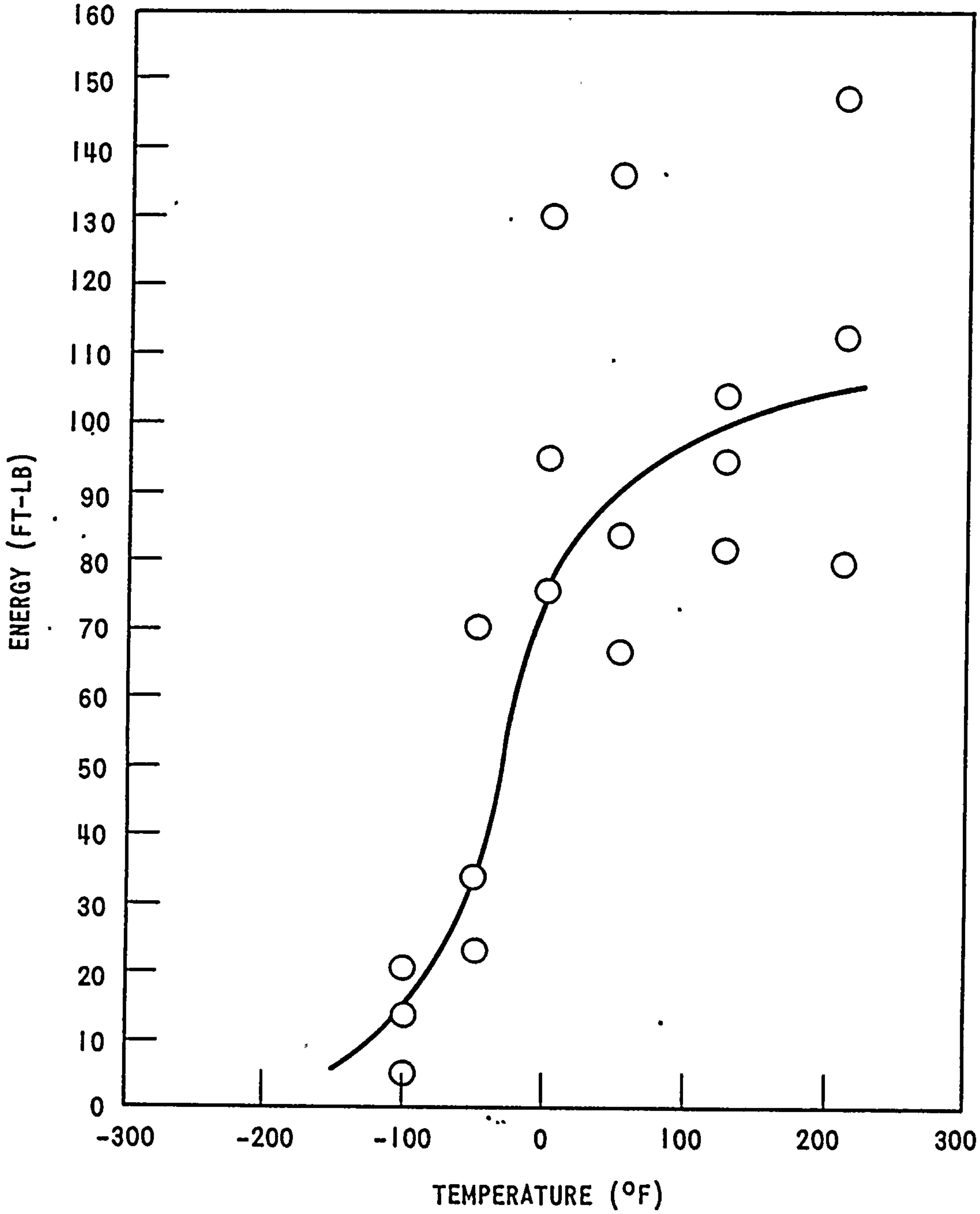

Figure 3-4. Preirradiation Charpy V-Notch Impact Energy Curve for the Donald C. Cook Unit No. 2 Reactor Pressure Vessel Core Region Weld Heat-Affected Zone Material

TABLE 3-5

PREIRRADIATION TENSILE PROPERTIES FOR THE DONALD C. COOK UNIT NO. 2 REACTOR PRESSURE VESSEL INTERMEDIATE SHELL PLATE C5521-2 AND CORE REGION WELD METAL

Vessel Material	Test Temp (°F)	0.2% Yield Strength (psi)	Ultimate Tensile Strength (psi)	Fracture Load (lb)	Fracture Stress (psi)	Uniform Elongation (%)	Total Elongation (%)	Reduction In Area (%)
Plate C5521-2 (Transverse Orientation)	ROOM	67400	87350	3200	161200	13.4	23.4	59.6
	ROOM	65450	85900	2950	156400	15.0	27.1	61.7
	300	58800	78600	2650	146100	13.0	22.6	63.1
	300	60500	79500	2675	157600	10.6	19.8	65.4
	550	57500	83000	3225	142150	11.5	19.0	53.8
	550	58950	83150	3150	145600	12.7	20.5	56.0
Weld Metal	ROOM	75750	93250	2850	173400	13.9	25.7	66.8
	ROOM	76900	91300	2950	178800	12.2	22.6	66.6
	300	70750	88000	2900	171000	10.7	20.7	66.0
	300	71000	85350	2875	179000	10.3	21.2	67.5
	550	70000	87250	3160	157200	10.1	19.2	59.6
	550	68200	87800	3050	166000	9.3	20.2	62.8

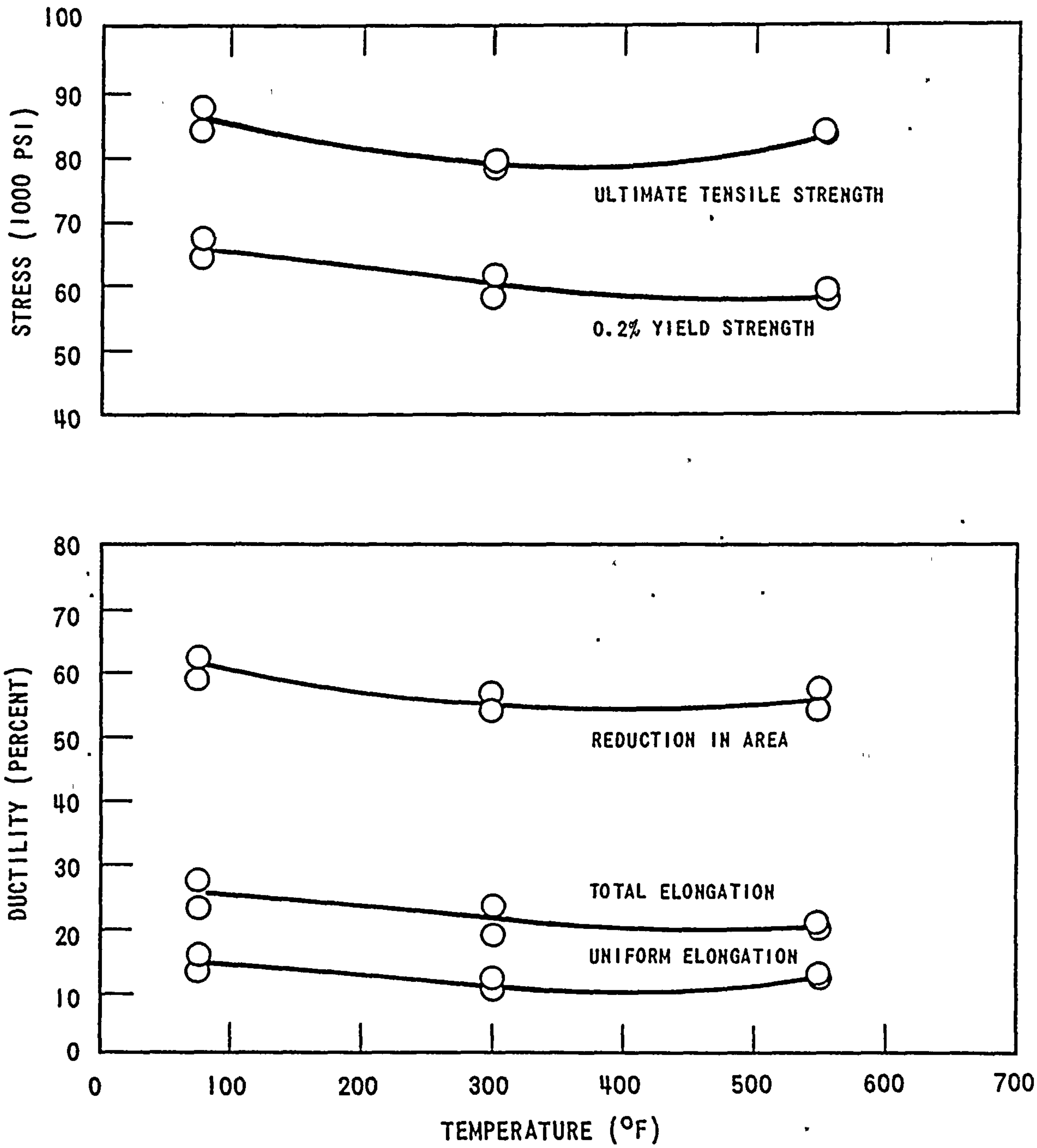

Figure 3-5. Preirradiation Tensile Properties for the Donald C. Cook Unit No. 2 Reactor Pressure Vessel Intermediate Shell Course Plate C5521-2 (Transverse Orientation)

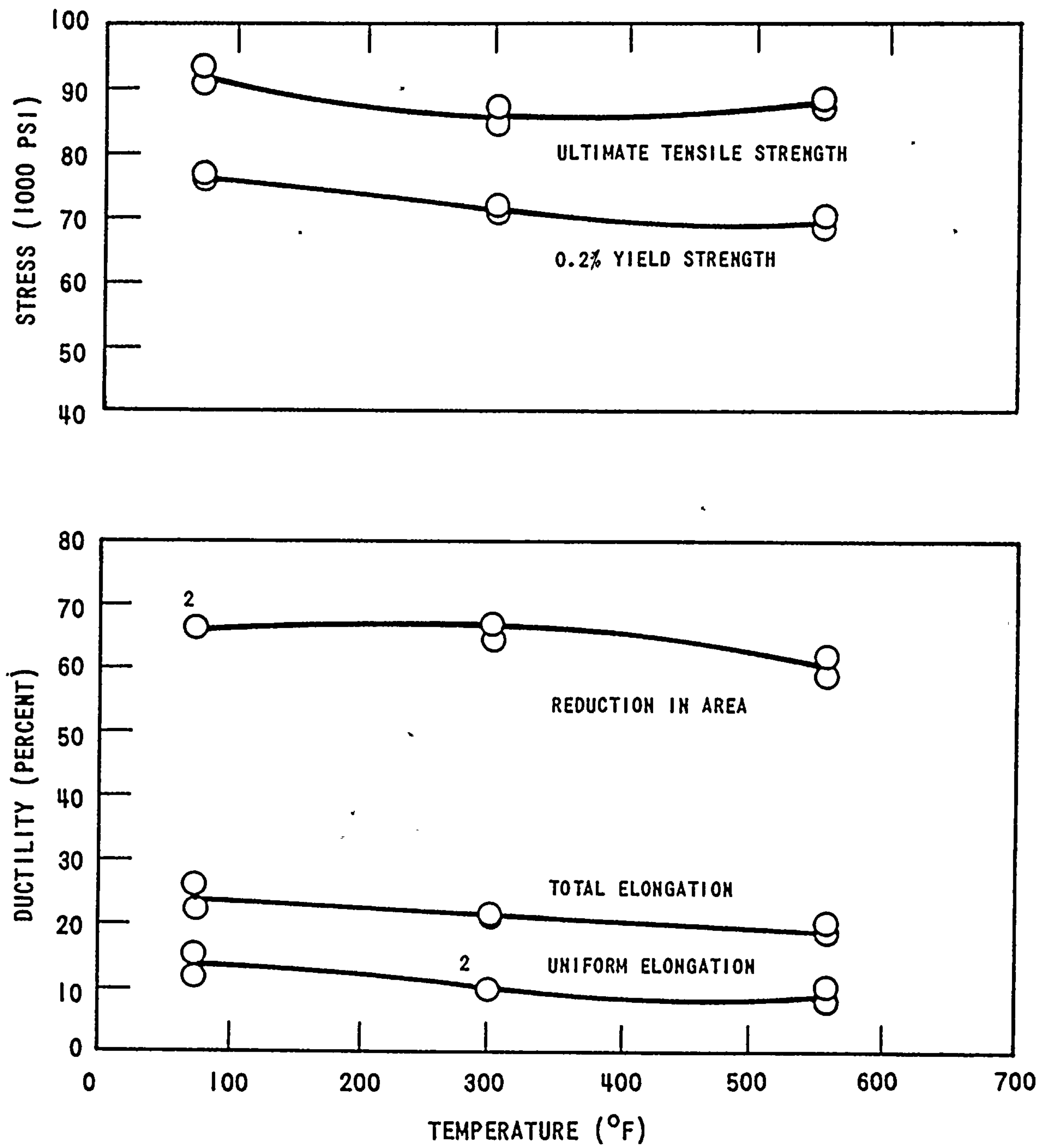

Figure 3-6. Preirradiation Tensile Properties for the Donald C. Cook Unit No. 2 Reactor Pressure Vessel Core Region Weld Metal

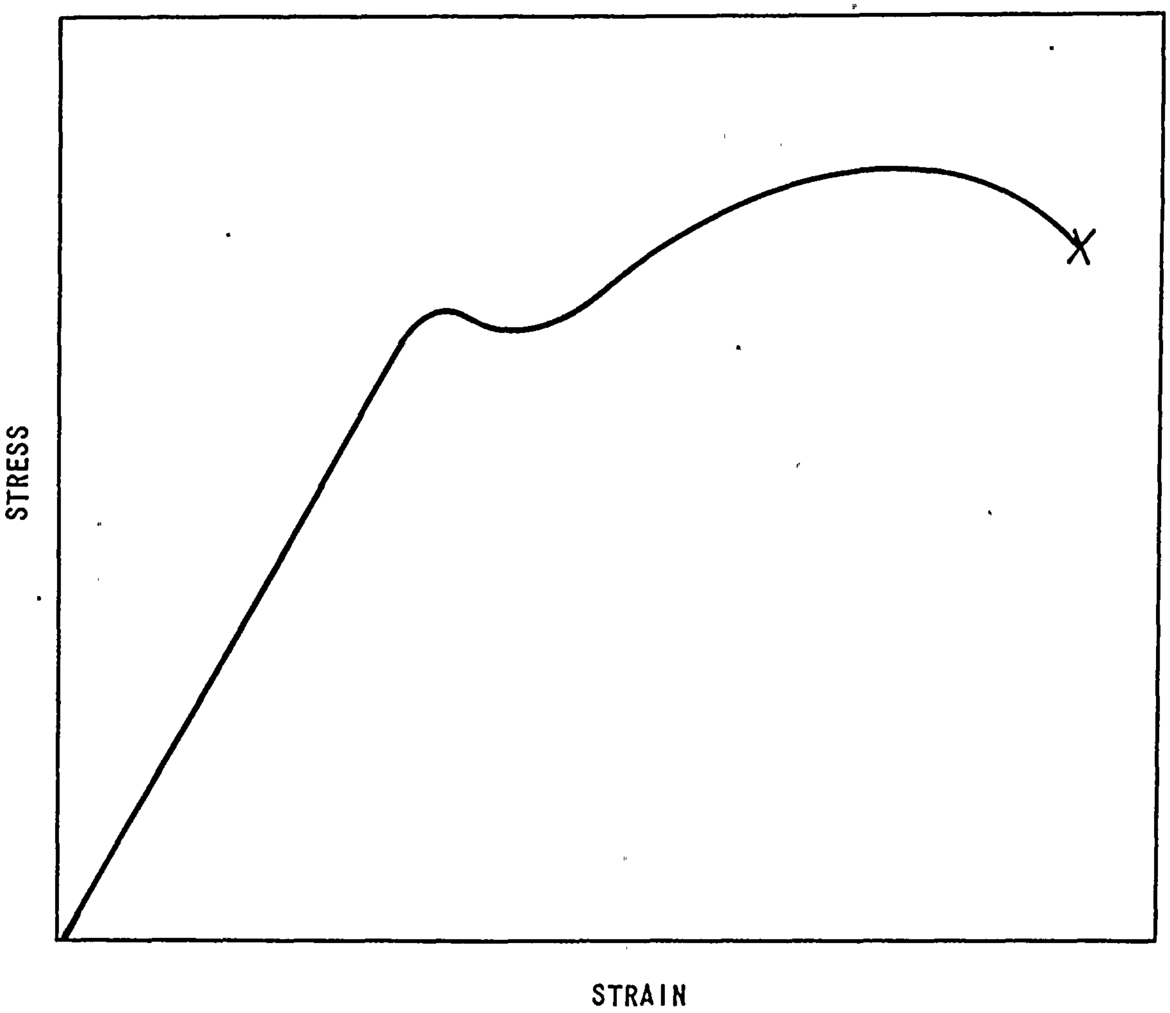

Figure 3-7. Typical Tensile Test Stress-Strain Curve

3-3. DROPWEIGHT TESTS

The NDTT was determined for plate C5521-2, the core region weld metal, and HAZ material by dropweight tests (ASTM E-208) performed at The Chicago Bridge and Iron Company. The following results were obtained:

Material	NDTT (°F)
Plate C5521-2	+10
Weld Metal	−40
HAZ	−10

SECTION 4
POSTIRRADIATION

4-1. CAPSULE REMOVAL

Specimen capsules are removed from the reactor only during normal refueling periods. The recommended schedule for removal of capsules is presented in table 4-1.

TABLE 4-1

SCHEDULE FOR REMOVAL OF SPECIMEN CAPSULES

Capsule Identification	Multiplying Factor By Which the Capsule Leads Vessel Maximum Exposure	Removal Time
T	2.9	End of first core cycle
U	2.9	9 years
X	2.9	18 years
Y	2.9	30 years
S	1.0	Standby
V	1.0	Standby
W	1.0	Standby
Z	1.0	Standby

Each specimen capsule is removed after radiation exposure and transfered to a post-irradiation test facility for disassembly of the capsule and testing of all specimens within that capsule.

4-2. CHARPY V-NOTCH IMPACT TESTS

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

The initial Charpy specimen from the first capsule removed should be tested at room temperature. The impact energy value for this temperature should be compared with the preirradiation test data; the testing temperatures for the remaining specimens should then be raised and lowered as needed. The test temperatures of specimens from capsules exposed to longer irradiation periods should be determined by the test results for the previous capsule.

4-3. TENSILE TESTS

The tensile specimens for each of the irradiated materials should be tested at test temperatures identical to the WOL fracture toughness test temperatures of the material.

4-4. WEDGE OPENING LOADING K_{Id} FRACTURE TOUGHNESS TESTS

In light of current requirements of 10CFR, Part 50, ASME Code, appendix G, the WOL specimens should be tested dynamically to adequately characterize the fracture toughness properties of the reactor vessel. The WOL specimens for each of the irradiated materials should be tested in accordance with ASTM E399-70T with appropriate modifications necessary for dynamic tests. Test temperatures which are recommended are the irradiated 50 ft lb temperature, 212°F, and temperatures representative of the irradiated Charpy V-notch upper shelf region if the 212°F test temperature occurs in the transition region. When the material fracture toughness at these temperatures is too high to be valid according to ASTM E399-70T, test data can then be interpreted by either the J Integral Concept[1] or the Equivalent Energy Concept[2].

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, and dosimeter blocks and spacers

1. *Fracture Toughness*, ASTM STP-514, American Society for Testing and Materials, Philadelphia, 1972.

2. T. R. Mager and C. Buchalet, "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.

- Hot cell tensile testing machine with pin-type adapter for testing tensile specimens
- Hot cell dynamic WOL testing machine with clevis and appropriate displacement 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

DONALD C. COOK UNIT NO. 2 REACTOR PRESSURE VESSEL SURVEILLANCE MATERIAL

The Chicago Bridge and Iron Company supplied the Westinghouse Electric Corporation with sections of A533 Grade B, Class 1 plate used in the core region of the Donald C. Cook Unit No. 2 reactor pressure vessel for the Reactor Vessel Radiation Surveillance Program. The sections of material were removed from the 8 3/4-inch intermediate shell plate C5521-2 of the pressure vessel. The Chicago Bridge and Iron Company also supplied a weldment made from sections of plate C5521-2 and adjoining lower shell plate C5592-1 using weld wire representative of that used in the original fabrication. The heat treatment history and quantitative chemical analysis of the pressure vessel surveillance material are presented in tables A-1 and A-2, respectively.

TABLE A-1

HEAT TREATMENT HISTORY

Material	Temperature (°F)	Time (hrs)	Coolant
Intermediate Shell (Plate C5521-2)	1650/1750	4 1/2	Water quenched
	1550/1650	5	Water quenched
	1200/1300	4 1/2	Air cool
	1150 ± 25	51 1/2	Furnace cooled
Weld	1140 ± 25	9	Furnace cooled