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DUANE ARNOLD ENERGY CENTER REACTOR PRESSURE VESSEL SURVEILLANCE MATERIALS TESTING

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ABSTRACT

A surveillance capsule was removed from the Duane Arnold Energy Center reactor at the end of Fuel Cycle 7, after accumulating 5.9 effective full power years of operation. The capsule contained flux wires for neutron fluence measurement and Charpy and tensile test specimens for material property evaluation. A combination of flux wire testing and computer analysis was used to establish the vessel peak flux magnitude and end-of-life predicted fluence. Charpy V-Notch impact testing and uniaxial tensile testing were performed to establish the material properties of the irradiated vessel beltline. Unirradiated archive specimens were provided by Iowa Electric Light and Power to establish baseline Charpy and tensile data.

The base metal Charpy specimen testing results show a larger irradiation shift than predicted, while the weld metal showed no measurable shift. The predicted end-of-life conditions of reference temperature and upper shelf energy were calculated to be less severe than the limits requiring vessel thermal annealing.

ACKNOWLEDGMENTS

Flux wire testing was performed by G. C. Martin. The Charpy V-Notch testing was done by J. L. Bennett. S. B. Wisner and G. H. Henderson tested the tensile specimens. C. R. Judd performed the chemical composition testing.

1. INTRODUCTION

Part of the effort to assure reactor vessel integrity involves evaluation of the fracture toughness of the vessel ferritic materials. The key values which characterize a material's fracture toughness are the reference temperature of nil-ductility transition (RT_{NDT}) and the upper shelf energy (USE). These are defined in 10CFR50 Appendix G (Reference 1) and in Appendix G of the ASME Boiler and Pressure Vessel Code, Section III (Reference 2). These documents contain requirements that establish the pressure-temperature operating limits which must be met to avoid brittle fracture.

Appendix H of 10CFR50 (Reference 3) and ASTM E185-82 (Reference 4) establish the methods to be used for surveillance of the reactor vessel materials. In April, 1985 one of the surveillance specimen capsules required by Reference 3 was removed from the Duane Arnold Energy Center (DAEC) reactor after 7 fuel cycles of irradiation, or 5.9 effective full power years (EFPY) of operation. The surveillance capsule contained flux wires for neutron flux monitoring and Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated from the vessel plate and weld materials nearest the core (the beltline). The impact and tensile specimens were tested to establish material properties for the irradiated vessel materials. The results of surveillance specimen testing are presented in this report.

Unirradiated archive Charpy and tensile specimens of identical fabrication and material composition as those in the surveillance capsule were provided by Iowa Electric Light and Power (IEL&P) for testing to establish the unirradiated properties of the surveillance materials. The archive specimens were tested on the same equipment, by the same operators, as were the irradiated specimens. The results of unirradiated specimen testing are presented and comparisons to surveillance specimen test results are made.

Predictions of the RT_{NDT} and USE at end of reactor life (EOL) are made using Regulatory Guide 1.99, Revision 1 (Reference 5), incorporating results from the surveillance tests. The predictions are compared with allowable values in Reference 1. Operating limits curves are examined in light of the results of the surveillance testing. Technical Specification revisions are suggested, as are changes to the Updated Final Safety Analysis Report (UFSAR).

2. SUMMARY AND CONCLUSIONS

2.1 SUMMARY OF RESULTS

Surveillance capsule 1 was removed from the DAEC reactor at the end of Fuel Cycle 7 and shipped to the General Electric Vallecitos Nuclear Center. In addition, unirradiated archive Charpy and tensile specimens were sent by IEL&P for testing to establish baseline properties. The flux wires, Charpy V-Notch and tensile test specimens were tested according to ASTM E185-82 (Reference 4). The methods and results of the fracture toughness testing are presented in this report as follows:

a. Section 3: Surveillance Program Background

b. Section 4: Flux Wire Test Evaluation

c. Section 5: Charpy V-Notch Testing

d. Section 6: Tensile Testing

e. Section 7: Determination of Operating Limits and End-of-Life Conditions

Photographs of Charpy specimen fracture surfaces are in Appendices A and B for irradiated and unirradiated specimens, respectively. Suggested revisions to the Technical Specification are found in Appendix C, and UFSAR changes are recommended in Appendix D.

The significant results of the evaluation are summarized as follows:

a. Capsule 1 was removed from the 288° azimuth position of the reactor. The capsule contained 6 flux wires: 2 each of pure copper (Cu), nickel (Ni) and iron (Fe). There were 24 Charpy V-Notch specimens: 8 each of plate material, weld material and heat affected zone (HAZ) material. The 6 tensile specimens removed consisted of 2 plate, 2 weld and 2 HAZ metal specimens. All specimen materials were positively identified as being from the vessel beltline.

b. The chemical compositions of the beltline materials were identified through a combination of literature research and testing. The copper (Cu), phosphorus (P) and nickel (Ni) contents were determined for all heats of plate material and for the seam welds in the beltline. The chemistry for the limiting beltline plate is 0.13% Cu, 0.012% P and 0.47% Ni. For the limiting beltline weld, the chemistry is 0.03% Cu, 0.017% P and 1.02% Ni.

c. The flux wires were tested to determine the neutron flux at the surveillance capsule location. The fast flux (>1.0 MeV) measured was 2.6×10^{9} n/cm²-s. Based on the flux wire data, the surveillance specimens had received a nominal fluence of 4.9×10^{17} n/cm² at removal.

d. Lead factors for the DAEC vessel geometry and core power distribution were computed in Reference 6. The lead factors for the surveillance capsule are 0.78 to the peak vessel surface location and 0.99 to the peak 1/4 T depth location. The EOL fluence, based on 32 EFPY, was calculated with the 1/4 T lead factor and the flux wire results, including a 4% power uprate at 6 EFPY. The resulting EOL fluence is 3.6×10^{18} n/cm².

e. The irradiated and unirradiated surveillance Charpy V-Notch specimens were impact tested at temperatures selected to define the transition of the fracture toughness curves of the plate, weld, and HAZ materials. Measurements were taken of absorbed energy, lateral expansion and percent shear. Fracture surface photographs of each specimen are presented in Appendix A for irradiated specimens and in Appendix B for unirradiated specimens. From absorbed energy and lateral expansion results for the plate, weld and HAZ materials, the following values were extracted: index temperatures for 30 ft-lb, 50 ft-lb, and 35-mil lateral expansion (MLE) values, and USE.

f. The unirradiated Charpy specimen curves were compared to the irradiated specimen curves to establish the RT_{NDT} shift due to irradiation. The base metal test results indicate a 42°F shift, which is greater than the 32°F shift predicted by Reference 5. The weld metal test results indicate no measurable shift.

g. The irradiated tensile specimens were tested at room temperature (76°F) and reactor operating temperature (550°F). Unirradiated tensile specimens were tested at room temperature, reactor operating temperature and at the temperature corresponding to onset of upper shelf. The results tabulated for the unirradiated and irradiated specimens include yield and ultimate tensile strength (UTS), uniform and total elongation, and reduction of area. The plate, weld, and HAZ specimens behaved similarly, indicating decreasing properties with increasing temperature.

h. The irradiated material tensile test results are compared to unirradiated data from the reactor vessel fabrication test program. The results indicate yield strength and UTS have increased with irradiation, and the uniform and total elongation have decreased with irradiation. These trends are characteristic of the effects of irradiation.

i. The adjusted reference temperature for each beltline plate and weld was calculated for EOL conditions. A factor of 42/32 was used to increase the predicted shifts of the beltline plates, based on the surveillance test results. Plate 1-19 was identified as the limiting beltline material, because of its high initial RT_{NDT} of 40°F. The EOL reference temperature is 126°F.

j. The initial RT_{NDT} and predicted shift for plate 1-19 were compared with the corresponding values used to develop the operating limits curves in Reference 6. The shift calculation in this analysis incorporated a correction for the surveillance test results, which increases the shift for a given fluence. However, the EOL fluence calculated is lower than that used in Reference 6. The net result is that the EOL shift is identical to that reported in Reference 6.

k. The USE values for beltline plates 1-20 and 1-21 and for the beltline weld material were predicted using Reference 5 for the EOL condition. Comparison of the unirradiated and irradiated specimen USE values showed Reference 5 to be conservative. The plate values were corrected to 65% of the longitudinal USE to estimate the transverse USE, according to the recommendation in Branch Technical Position MTEB 5-2 (Reference 7). The lowest EOL values calculated were 70 ft-lb and 86 ft-lb for the plate and weld materials, respectively.

2.2 CONCLUSIONS

Surveillance testing is done to support or replace calculated information related to operating conditions and EOL vessel beltline conditions, as outlined in Reference 1. The results obtained from surveillance testing which impact these conditions are the predicted EOL fluence, the shift in RT_{NDT} for the beltline materials and the decrease in USE.

The flux wire testing, combined with the lead factors from the Reference 6 analysis, give an EOL fluence prediction which is about 20% lower than the value predicted in Reference 6. This fluence, based on a daily power history for seven fuel cycles, is considered more accurate than the fluence based on the flux wire test results from the end of Fuel Cycle 2.

Since the limiting beltline material initial RT_{NDT} and shift determined in this report are the same as were used in Reference 6, there is no need to change the operating limits curves developed in that report. The only changes needed in the Technical Specification and UFSAR relate to the completion of the testing of the first surveillance capsule. The suggested changes are in Appendices C and D.

The predicted EOL reference temperature and the USE results developed from the surveillance testing indicate that the DAEC vessel is in compliance with the fracture toughness requirements of 10CFR50 Appendix G.

3. SURVEILLANCE PROGRAM BACKGROUND

3.1 CAPSULE RECOVERY

The DAEC reactor was shut down in February 1985 for refueling and maintenance. The accumulated thermal power output was 3.45×10^6 MWd or 5.9 EFPY. The reactor pressure vessel (RPV) originally contained three surveillance capsules at 36°, 108°, and 288° azimuths at the core midplane. The specimen capsules are held against the RPV inside surface by a spring loaded specimen holder. Capsule 1 at 288° was removed by DAEC personnel on April 9, 1985. The capsule was cut from the holder assembly and shipped via a 200 Series cask to the General Electric Vallecitos Nuclear Center in Pleasanton, California.

Upon arrival at Vallecitos, the capsule was examined for identification. The reactor code of 35 from Reference 8 and the capsule number were confirmed on the capsule, as shown in Figure 3-1. The capsule contained two Charpy specimen packets and three tensile specimen tubes. Each Charpy packet contained 12 Charpy specimens and 3 flux wires. The three tensile specimen tubes contained a total of six specimens. The tensile specimen gage sections were protected by aluminum sleeves, and during removal of the sleeves, the threaded ends of the specimens were slightly damaged. The threads were later chased with a die-hex re-threading tool. The gage sections of the tensile specimens were not damaged during removal.

3.2 RPV MATERIALS AND FABRICATION BACKGROUND

3.2.1 Fabrication History

The DAEC RPV is a 183-inch diameter BWR/4. It was field fabricated by Chicago Bridge & Iron to Section III of the 1965 ASME Code with Addenda up to and including Winter 1967. The shell and head plates are ASME SA-533 Grade B, Class 1 low alloy steel (LAS). The nozzles and closure flanges are ASME SA-508 Class 2 LAS, and the closure flange bolting materials are ASTM A540 Grade B24 LAS. The fabrication process employed quench and temper heat treatment immediately after hot forming, then shielded metal arc welding (SMAW) and post-weld heat treatment in the field. The post-weld heat treatment was typically 10 hours at 1150°± 25°F. The arrangement of plates and welds relative to the core beltline and various nozzles is shown in Figure 3-2.

3.2.2 Material Properties of RPV at Fabrication

A search of General Electric Quality Assurance (QA) records was made to determine the chemical and mechanical properties of the plates and welds in the RPV beltline. The results were reported in Reference 6, and are summarized in Table 3-1, including the initial RT_{NDT} values calculated. The Reference 6 work was supplemented with more specific information from Chicago Bridge & Iron on the weld materials that went into the beltline longitudinal and girth welds. The limiting weld materials used in the beltline welds are shown in Table 3-1.

3.2.3 Specimen Chemical Composition

Samples were taken from base metal tensile specimens ETJ and ETK and from weld metal tensile specimens EU3 and EU6 after they were tested. Samples from the tensile specimens represent the Charpy specimens as well, since both were taken from the same plate and weld, as discussed in Subsection 3.3. A total of four samples were analyzed. Chemical analysis was performed using a plasma emission spectrometer. Each sample was decomposed and dissolved, and a portion prepared for evaluation by the spectrometer. The spectrometer was calibrated with a standard solution containing 700 ppm Fe, 8 ppm Mn, 2 ppm Cu, 5 ppm Ni, 5 ppm Mo, 5 ppm Cr, 1 ppm Si, 1 ppm Co, and levels of perchloric acid and lithium consistent with the test. The phosphorus calibration was done by analyzing a series of seven National Bureau of Standards (NBS) steels with known phosphorus contents. The chemical composition results are given in Table 3-2.

3.3 SPECIMEN DESCRIPTION

The surveillance capsule contained 24 Charpy specimens: base metal (8), weld metal (8), and HAZ (8). There were 6 tensile specimens: base metal (2), weld metal (2), and HAZ (2). The 6 flux wires recovered were iron (2), nickel (2) and copper (2). The chemistry and fabrication history for the Charpy and tensile specimens are described in this section.

3.3.1 Charpy Specimens

The fabrication of the Charpy specimens is described in Surveillance Test Drawings T4 through T12 (Reference 9). All materials used for specimens were beltline materials from the lower intermediate shell course.

The base metal specimens were cut from plate 1-21 (Heat B0673-1). The chemical analysis of this heat of low alloy steel is in Reference 6, as summarized in Table 3-1. The test plate was heat treated for 50 hours at 1150°F plus 25°F or minus 50°F to conservatively simulate the post-weld heat treatment of the vessel. The method used to machine the specimens from the test plate is shown in Figure 3-3. Specimens were machined from the 1/4 T and 3/4 T positions in the plate, in the longitudinal orientation (long axis parallel to the rolling direction). The identification of the base metal Charpy specimens recovered from the surveillance capsule is shown in Table 3-3.

The weld metal and HAZ Charpy specimens were fabricated from pieces of plate 1-21 that were welded together using "the same welding procedures as used to weld the core area shell plates together" (Reference 9). The chemical analysis of the surveillance weld material is shown in Table 3-2. The welded test plate for the weld and HAZ Charpy specimens received a heat treatment of 1150°F plus 25°F or minus 50°F for 50 hours to conservatively represent the heat treatment of the RPV. The weld specimens and HAZ specimens were fabricated as shown in Figures 3-4 and 3-5, respectively. The base metal orientation in the weld and HAZ specimens was longitudinal. Contained in Table 3-3 are the identifications of the weld metal and HAZ Charpy specimens from the surveillance capsule.

3.3.2 Tensile Specimens

Fabrication of the surveillance tensile specimens is described in the Reference 9 drawings. The chemical composition and heat treatment for the base, weld and HAZ tensiles are the same as those for the corresponding Charpy specimens. The identifications of the base, weld, and HAZ tensile specimens recovered from the surveillance capsule are given in Table 3-3. A summary of the fabrication methods is presented below.

The base metal specimens were machined from material at the 1/4 T and 3/4 T depth in plate 1-21. The specimens, oriented along the plate rolling direction, were machined to the dimensions shown in Figure 3-6. The gage section was tapered to a minimum diameter of 0.250 inch at the center. The weld metal tensile specimen material was cut from the welded test plate, as shown in Figure 3-7. The specimens were machined entirely from weld metal, scrapping material that might include base metal. The fabrication method for the HAZ tensile specimens is illustrated in Figure 3-8. The specimen blanks were cut from the welded test plate such that the gage section minimum diameter was machined at the HAZ centerline. The finished HAZ specimens are approximately half weld metal and half base metal, oriented along the plate rolling direction.

3.3.3 Unirradiated Archive Specimens

In order to develop meaningful results from the surveillance test data, unirradiated specimen test data must be available for comparison. After examining the Chicago Bridge & Iron (CB&I) fabrication test records, it was concluded that there had been no adequate unirradiated specimen data collected. The plate tests done as part of the fabrication test program by CB&I were on a different plate from that used for the surveillance specimens. There was no CB&I Charpy data located on the weld material used for the surveillance weld.Therefore, unirradiated archive specimens were retrieved from storage at IEL&P. The archive specimens are extra Charpy and tensile specimens fabricated for the surveillance program. They have the same geometries and material properties as the corresponding irradiated specimens taken from the surveillance capsule.

IEL&P provided 18 each of archive base, weld and HAZ metal Charpy specimens for testing. Since these archive specimens are irreplaceable, only those specimens needed to adequately define the Charpy toughness transition curves were tested. A total of 14 base, 12 weld and 12 HAZ specimens were tested. These specimen identifications are shown in Table 3-4.

IEL&P also provided 3 base metal and 3 weld metal tensile specimens (HAZ tensiles are no longer required by ASTM E185). These archive tensile specimens were tested at room temperature, reactor operating temperature (550°F) and at the temperature corresponding to each material's onset of upper shelf. The archive tensile specimen identifications are listed in Table 3-4.

Table 3-1

MATERIAL PROPERTIES OF VESSEL COMPONENTS

Identification		Heat/Lot No.	Chemistry (wt %) Cu	P	Ni	Charpy Test Temp. (°F)	Energy (ft-lb)	Dropweight Temp. (°F)	RT_{NDT}[a] (°F)
Lower Plates:									
(Shell 1)	1-18	C6439-2	0.09	0.012	0.51	40	36,48,43	40	40
	1-19	B0402-1	0.13	0.012	0.47	40	83,85,72	40	40
Lower-Intermed. Plates:									
(Shell 2)	1-20	B0436-2	0.15	0.008	0.64	40	57,54,62	-30	10
	1-21	B0673-1	0.15	0.011	0.61	40	99,104,121	-30	10
Longitudinal Welds		Ht. 432Z0471 Lot B003A27A	0.03	0.017	0.91	10	100,102,106	-	-50
Girth Weld		Ht. 07L669 Lot K004A27A	0.03	0.014	1.02	10	50,50,54	-	-50
Upper Plates:									
(Shell 4)	1-24	C6794-2				10	37,44,33	10	14
	1-25	C7090-1				10	61,67,79	10	10
Standby Liquid Control Nozzle		E20VW				40	38,60,26	40	58

[a] RT_{NDT} values are calculated to be equivalent to transverse RT_{NDT}.

Table 3-2

PLASMA EMISSION SPECTROMETRY CHEMICAL ANALYSIS OF RPV SURVEILLANCE PLATE AND WELD MATERIALS

Element	Base Metal Tensile ETJ	Base Metal Tensile ETK	Weld Metal Tensile EU3	Weld Metal Tensile EU6
Mn	1.4	1.3	1.3	1.2
P	0.006	0.006	0.011	0.010
Cu	0.15	0.15	0.02	0.02
Si [a]	0.07	0.06	0.32	0.33
Ni	0.70	0.69	1.00	0.90
Mo	0.63	0.62	0.49	0.49
Cr	0.14	0.14	0.04	0.03
Co	0.014	0.013	0.013	0.012

[a] Si results may be low, because of precipitation during dissolution heating.

Table 3-3

IDENTIFICATION OF CHARPY AND TENSILE SPECIMENS
REMOVED FROM SURVEILLANCE CAPSULE

Charpy Specimens

Base	Weld	HAZ
EBJ [a]	EE2	ELK
EBK	EE3	ELL
EBL	EE4	ELM
EBP	EE5	ELP
EBT	EJ1	ELT
EBU	EJ2	ELY
EBY	EJ3	EMT
EC1	EJ4	EMY

Tensile Specimens

Base	Weld	HAZ
ETJ [a]	EU3	EY1
ETK	EU6	EY2

[a] All specimen identifications include a double-dot over the middle symbol.

Table 3-4

IDENTIFICATION OF CHARPY AND TENSILE
ARCHIVE SPECIMENS

Charpy Specimens

Base	Weld	HAZ
ECK [a]	EED	EM1
ECL	EEE	EM2
ECM	EEK	EM3
ECP	EEL	EM4
ECT	EEM	EM5
ED1	EET	EM6
ED4	EEY	EMA
ED5	EK2	EP2
ED6	EK3	EP3
ED7	EK4	EP4
EDA	EK5	EP5
EDK	EK6	EP7
EDT		
EDY		

Tensile Specimens

Base	Weld
ET2 [a]	EUM
ET3	EUJ
ETA	EUP

[a] All specimen identifications include a double-dot over the middle symbol.

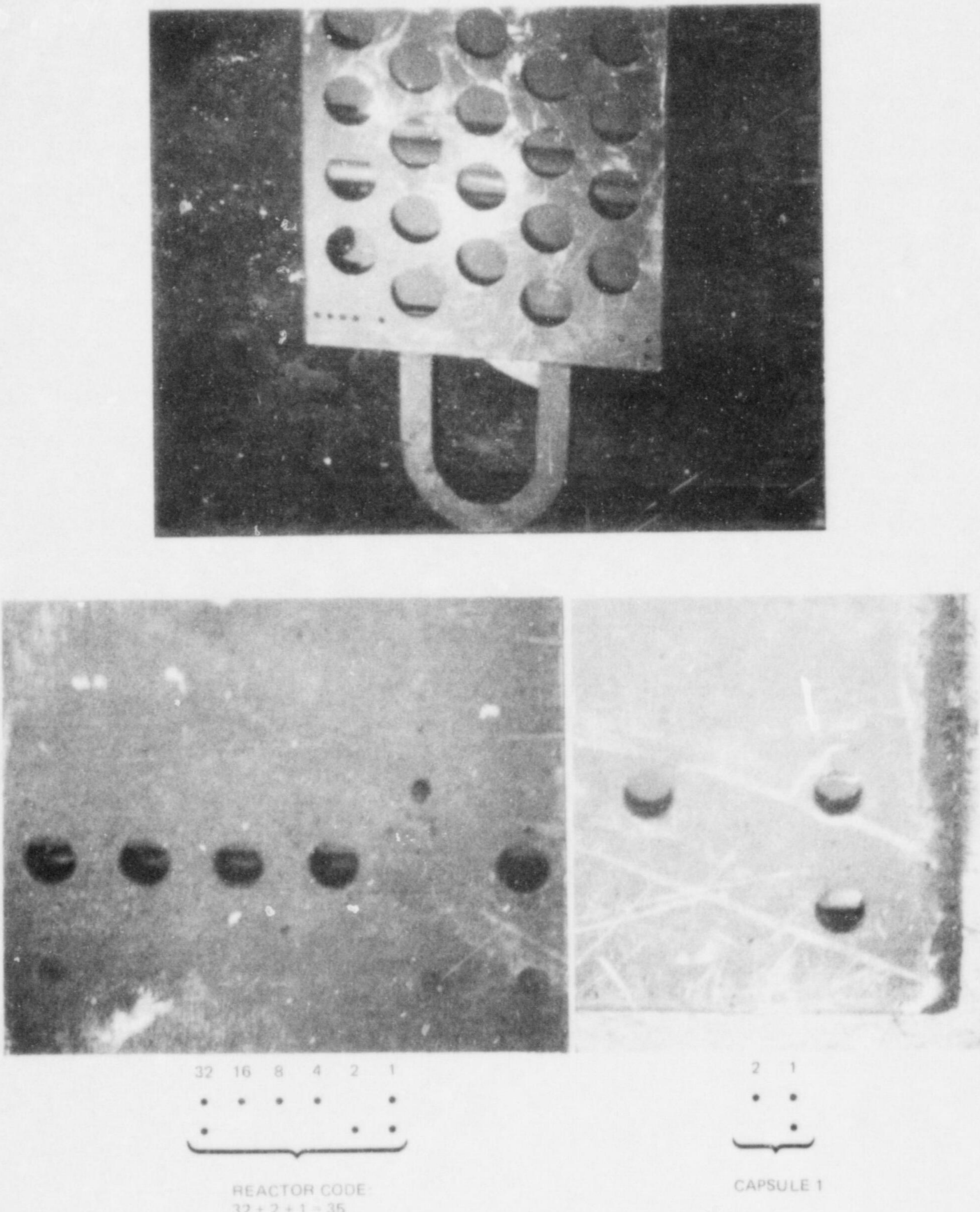


Figure 3-1. Surveillance Capsule Recovered from DAEC Reactor

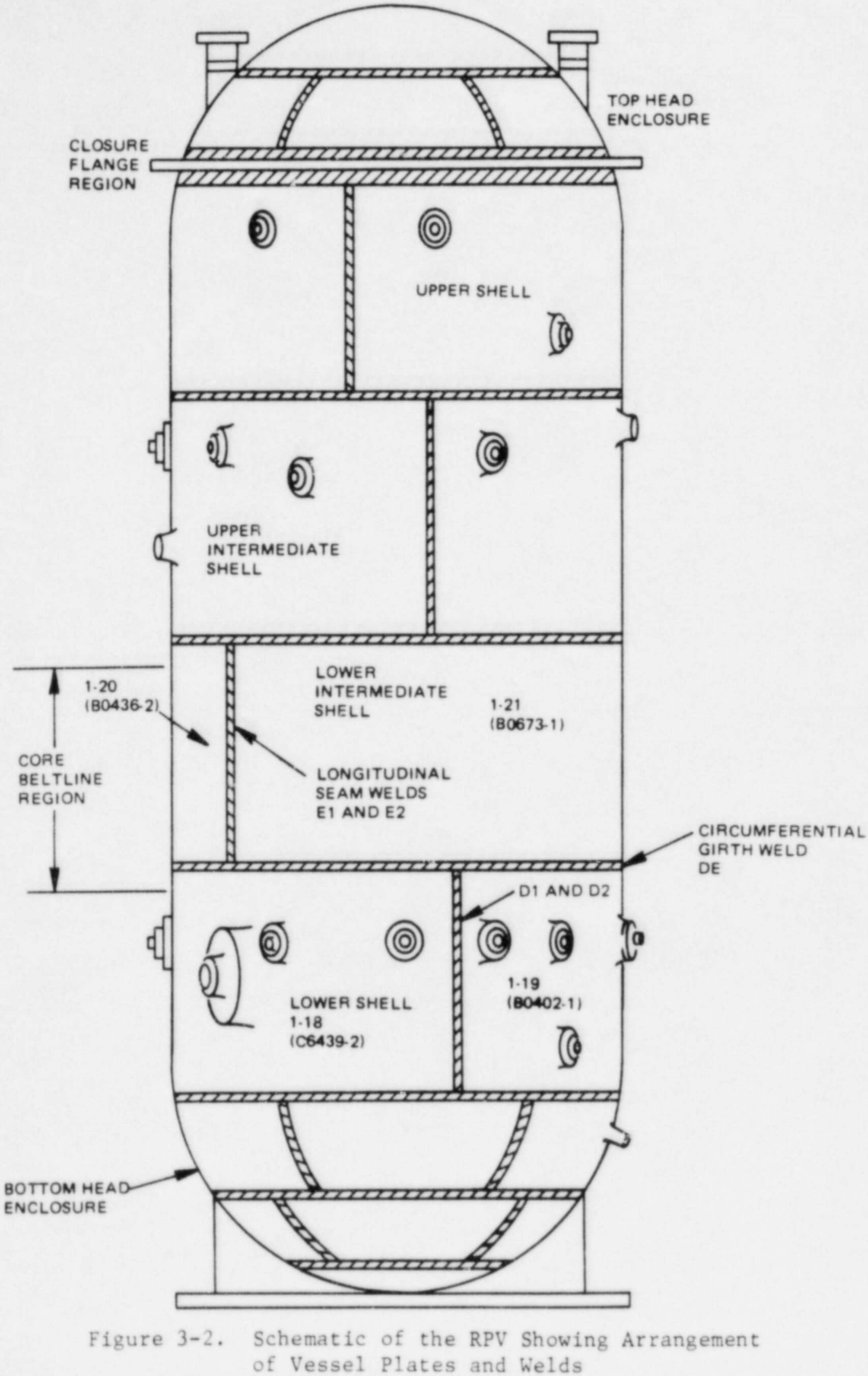


Figure 3-2. Schematic of the RPV Showing Arrangement of Vessel Plates and Welds

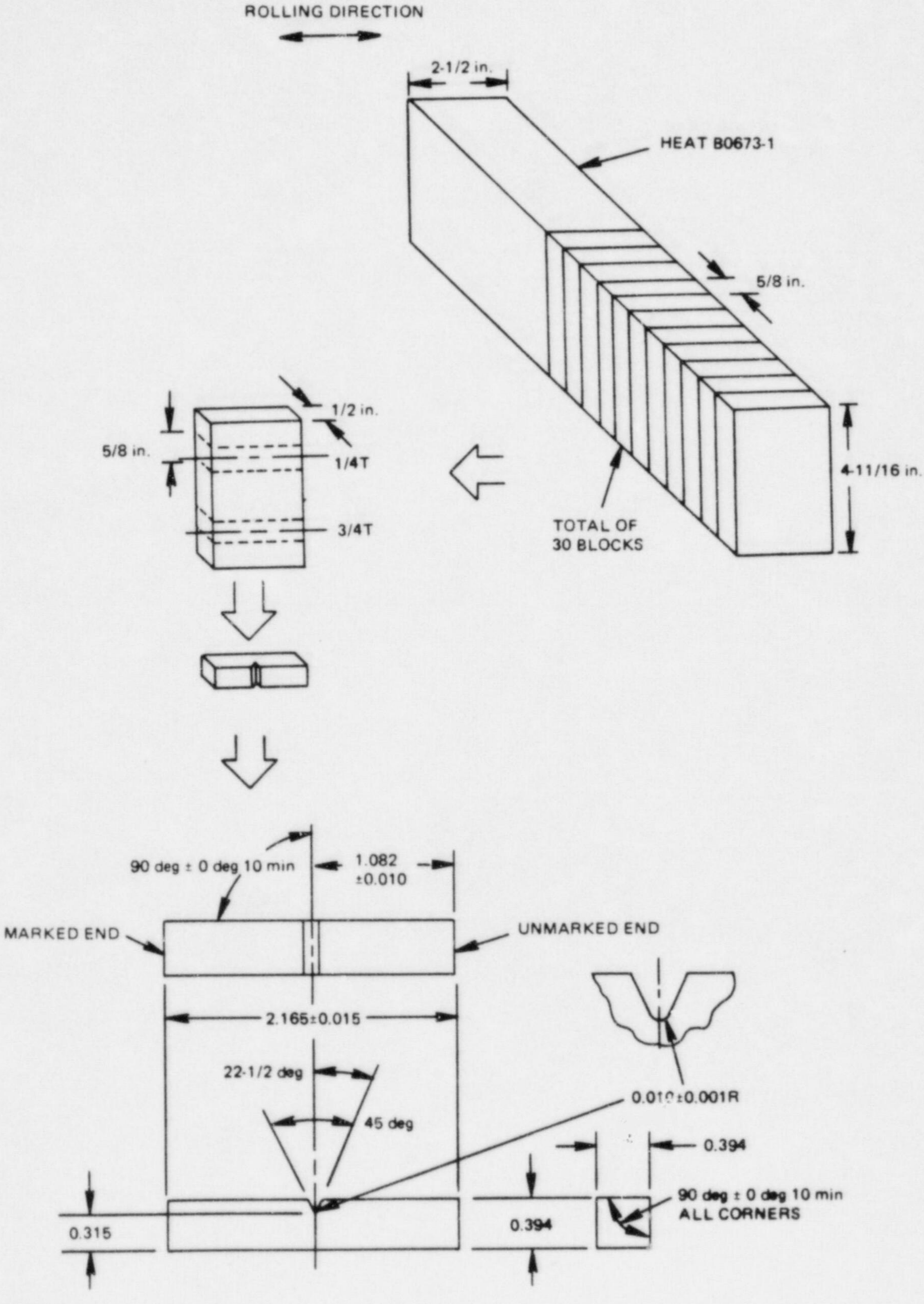


Figure 3-3. Fabrication Method for Base Metal Charpy Specimens

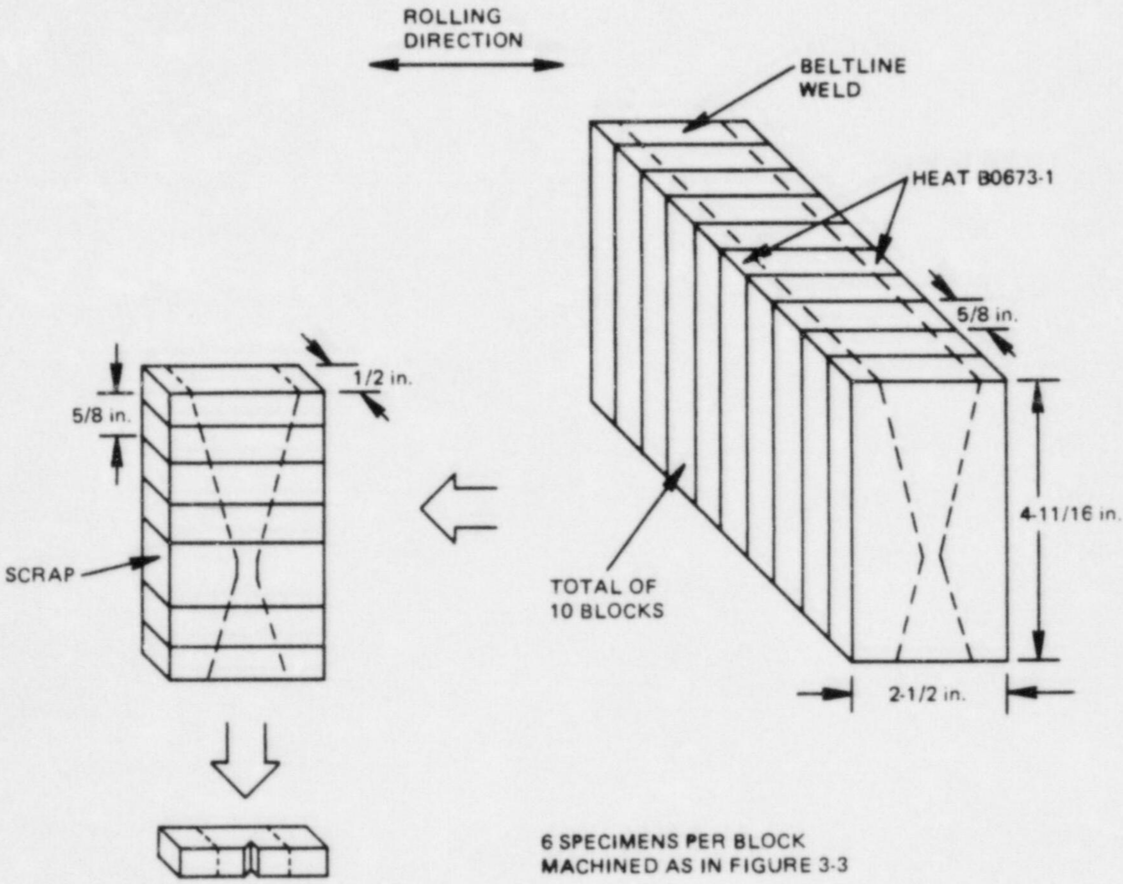


Figure 3-4. Fabrication Method for Weld Metal Charpy Specimens

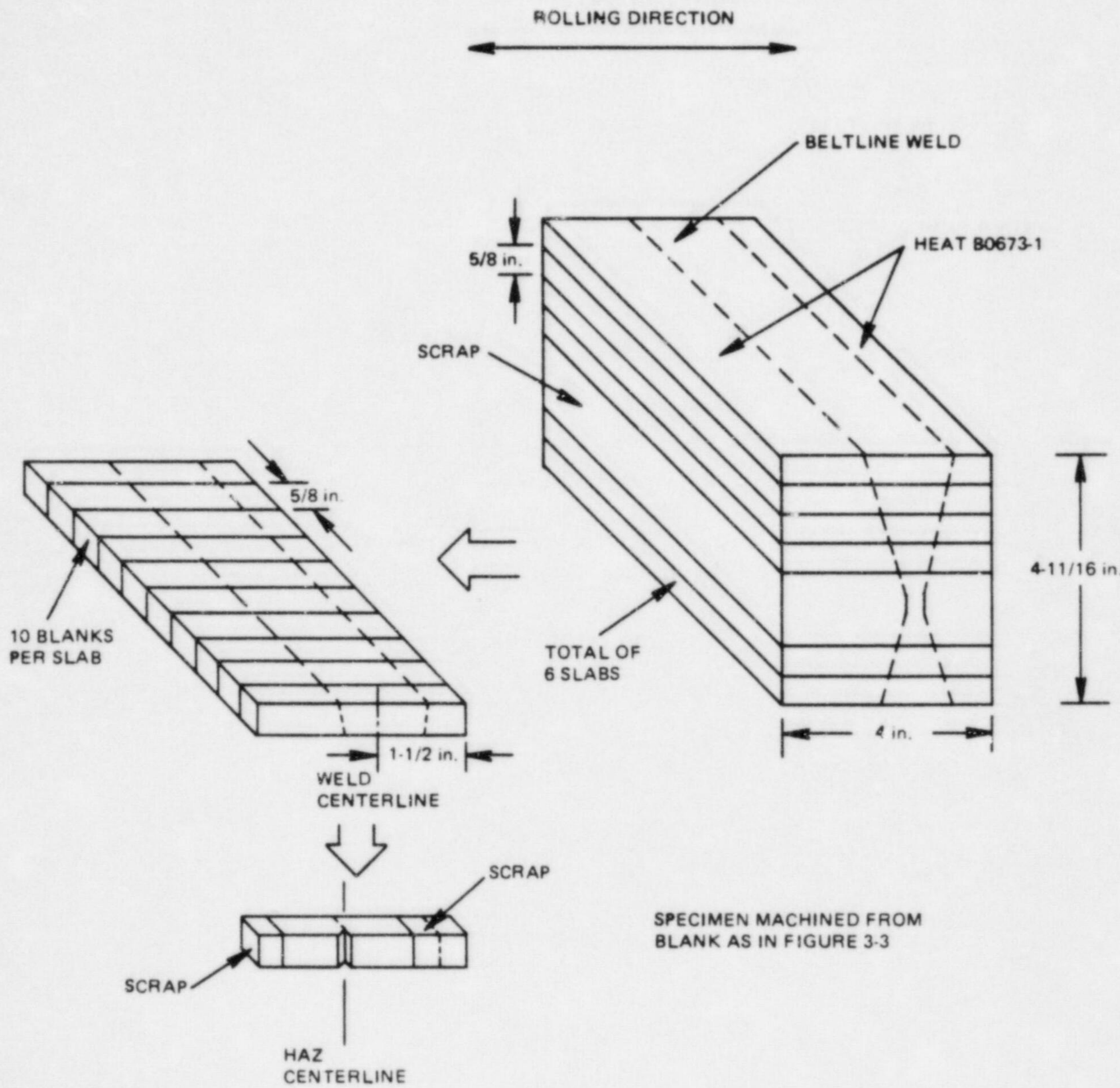


Figure 3-5. Fabrication Method for HAZ Charpy Specimens

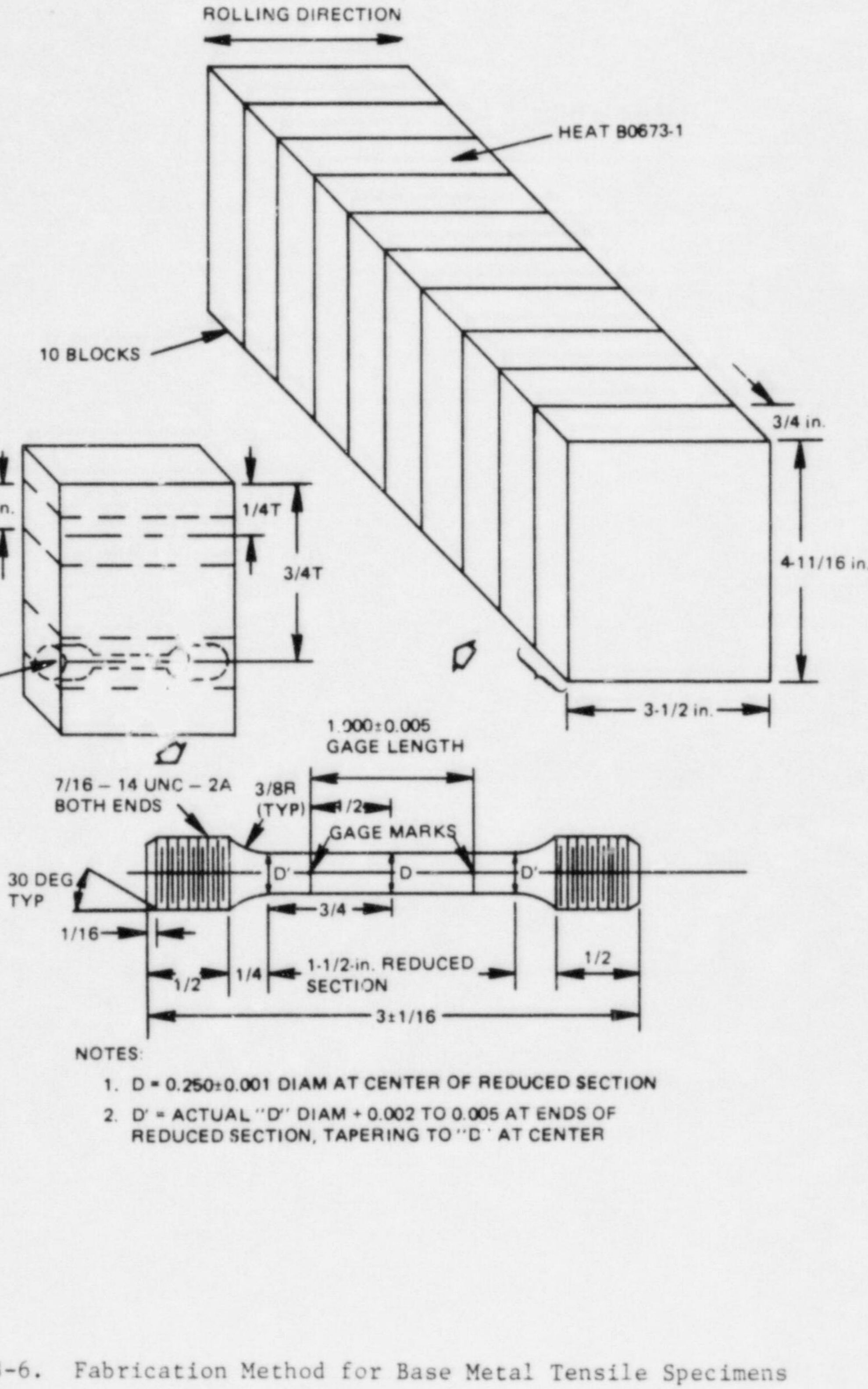


Figure 3-6. Fabrication Method for Base Metal Tensile Specimens

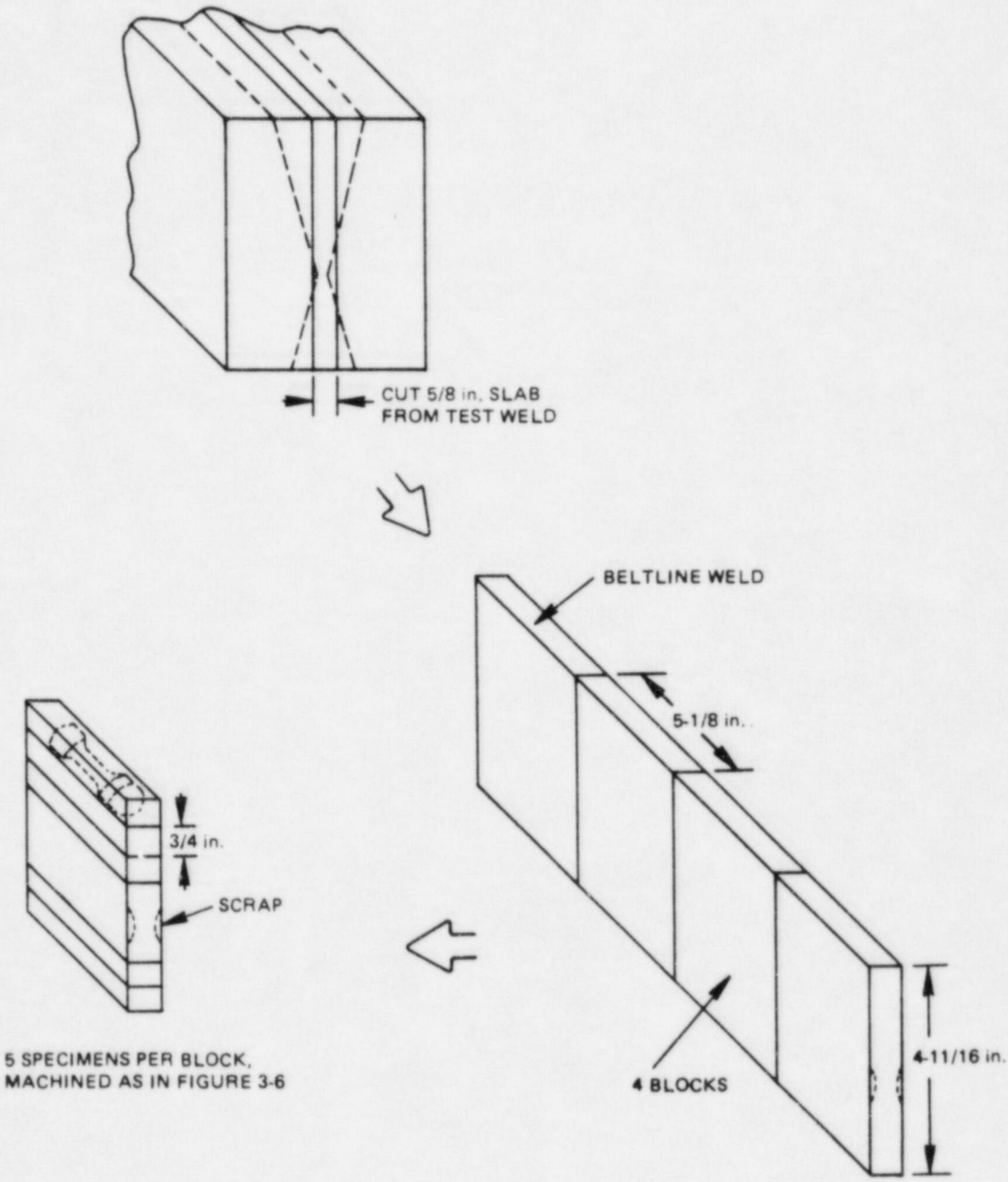


Figure 3-7. Fabrication Method for Weld Metal Tensile Specimens

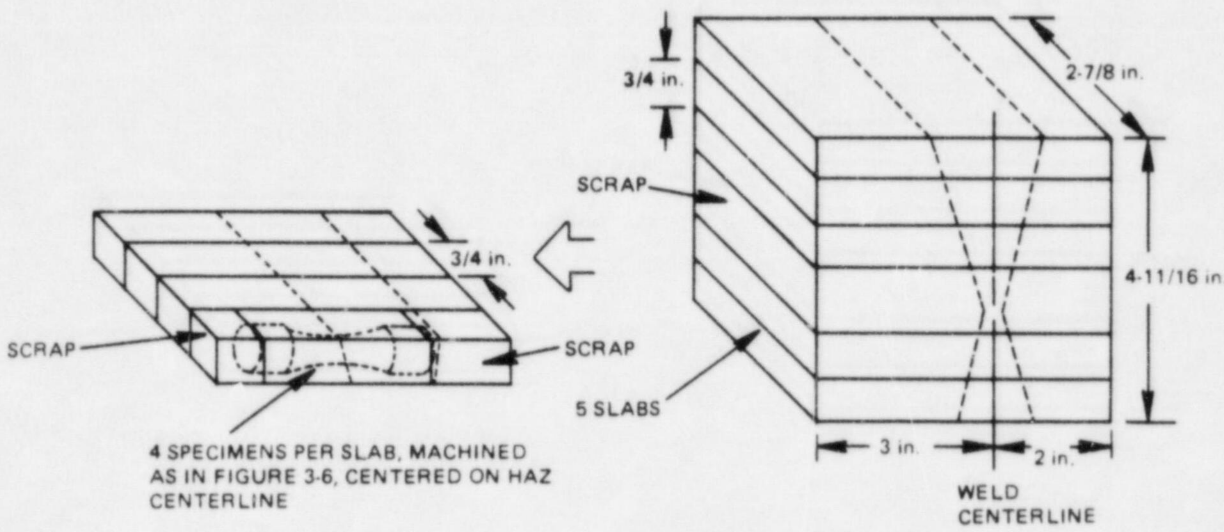


Figure 3-8. Fabrication Method for HAZ Tensile Specimens

4. FLUX WIRE TEST EVALUATION

Reference 6 includes a determination of the peak vessel flux and fluence, based on analytical results and results from a test performed on flux wires removed from the vessel after Fuel Cycle 2. The flux wires removed with the current surveillance capsule give more representative values of flux, as explained later, and provide a measurement of the fluence accumulated by the Charpy and tensile surveillance specimens. Updated values of peak vessel flux and fluence are computed using the latest flux wire results and the lead factors from Reference 6.

4.1 FLUX WIRE ANALYSIS

4.1.1 Procedure

The surveillance capsule contained six flux wires: two each of iron, nickel and copper. Each wire was removed from the capsule, cleaned with dilute acid, weighed, mounted on a counting card, and analyzed for its radioactivity content by gamma spectrometry. Each iron wire was analyzed for Mn-54 content, each nickel wire for Co-58 and each copper wire for Co-60 at a calibrated 4- or 10-cm source-to-detector distance with an 80-cc Ge(Li) detector system. The gamma spectrometer was calibrated using NBS material.

To properly predict the flux and fluence at the surveillance capsule from the activity of the flux wires, the periods of full and partial power irradiation and the zero power decay periods were considered. Operating days for each fuel cycle and the average reactor power fraction are shown in Table 4-1. Zero power days between fuel cycles are listed as well.

From the flux wire activity measurements and power history, reaction rates for Fe-54 (n,p) Mn-54, Ni-58 (n,p) Co-58 and Cu-63 (n,α) Co-60 were calculated. The >1 MeV fast flux reaction cross sections for the iron, copper, and nickel wires were estimated to be 0.157 barn, 0.0027 barn and 0.204 barn, respectively. These values were obtained from measured cross-section functions determined at Vallecitos from more than 65 spectral determinations for BWRs and for the General Electric Test Reactor using activation monitor and spectral unfolding techniques. Similarly, the >0.1 MeV cross sections for the iron, copper, and nickel wires were estimated from the measured 1-to-0.1 cross-section ratio of 1.6.

4.1.2 Results

The measured activity, reaction rate and determined full-power flux results for the surveillance capsule are given in Table 4-2. The >1 MeV and >0.1 MeV flux values of $2.6x10^9$ and $4.2x10^9$ n/cm^2-sec from the flux monitors were calculated by dividing the reaction rate measurement data by the appropriate cross sections. The corresponding fluence results, $4.9x10^{17}$ and $7.8x10^{17}$ n/cm^2 for >1 MeV and >0.1 MeV, respectively, were obtained by multiplying the full-power flux density values by the product of the total seconds irradiated ($2.79x10^8$ sec) and the full-power fraction (0.671).

Generally, for long-term irradiations, dosimetry results from copper flux wires are considered most accurate because of the length of the half-life of Co-60 (5.27 years) compared to those of iron's Mn-54 (312.5 days) and nickel's Co-58 (70.8 days). The iron and nickel flux wires, which are more sensitive to fluctuations in reactor power levels and to peripheral bundle power variations, showed reasonable agreement with the copper wires, varying by 10% and 20%, respectively. Given this, and the fact that the copper wire results were the most conservative, the copper flux wire results were used to predict capsule fluence and EOL vessel fluence.

The accuracies of the values in Table 4-2 for a 2σ deviation are estimated to be:

± 5% for dps/g (disintegrations per second per gram)

± 12% for dps/nucleus (saturated)

± 30% for flux and fluence >1 MeV

± 40% for flux and fluence >0.1 MeV

A set of flux wires from DAEC was evaluated by General Electric in 1977. The >1 MeV flux was 3.1×10^9 n/cm^2-sec. The flux from this study of 2.6×10^9 n/cm^2-sec is about 15% lower. Accounting for differences in lead factor between the two tests, the current flux is effectively 23% lower than expected, based on the first flux wire test result. This difference is probably due to the variations in operation typical in the first two fuel cycles. The most recent flux wires, which have been exposed to years of steady-state operation, provide a more reliable reading of the vessel steady-state flux. Therefore, the latest measured flux is used in Subsection 4.3 to predict EOL fluence.

4.2 FLUX DISTRIBUTION LEAD FACTORS

A lead factor is a calculated ratio between the flux at the surveillance capsule location and the flux at the location of peak flux in the beltline region (at the inside surface or at 1/4 T depth). The computer analyses required to establish the lead factors for the DAEC vessel were performed for Reference 6, and the results are discussed therein. The lead factors for the 288° capsule are 0.78 to the peak inside surface location and 0.99 to the peak 1/4 T location.

4.3 ESTIMATE OF END-OF-LIFE FLUENCE

The EOL fluence (f) is estimated using the upper bound (130%) of the measured flux from Table 4-2 with the lead factor to the 1/4 T location. As discussed in Reference 6, a power uprate from 1593 MW to 1658 MW is assumed at 6 EFPY. Flux is proportional to thermal power for a given core radial power shape, so the two fluxes are:

2.6×10^9 n/cm^2-s * 1.30/0.99 = $\underline{3.41 \times 10^9 \text{ n/cm}^2\text{-s}}$ for 6 EFPY, and

3.41×10^9 * (1658/1593) = $\underline{3.55 \times 10^9 \text{ n/cm}^2\text{-s}}$ for 7 through 32 EFPY.

The EOL condition of 32 EFPY represents 1.01×10^9 seconds of full power irradiation, or 3.16×10^7 seconds per EFPY. The EOL fluence is:

$f = 3.16 \times 10^7$ sec/EFPY(3.41×10^9 * 6 EFPY + 3.55×10^9 * 26 EFPY), or

$f = 3.6 \times 10^{18}$ n/cm^2.

This EOL fluence is significantly lower than the value used in the Reference 6 analysis of 4.4×10^{18} n/cm^2. The impact of the lower fluence value is discussed in Section 7.

Table 4-1

SUMMARY OF DAILY POWER HISTORY

Cycle	Cycle Dates	Operating Days	Percent of Full Power	Days Between Cycles
1a	5/14/74 - 6/6/75	389	0.525	
				42
1b	7/19/75 - 2/13/76	210	0.686	
				61
2	4/15/76 - 3/12/77	332	0.695	
				62
3	5/14/77 - 3/18/78	309	0.823	
				39
4a	4/27/78 - 6/17/78	52	0.700	
				265
4b	3/10/79 - 2/9/80	337	0.808	
				68
5	4/18/80 - 3/20/81	337	0.806	
				73
6	6/2/81 - 2/12/83	621	0.540	
				82
7	5/6/83 - 2/2/85	639	0.651	
		3226	0.671 (average)	

Table 4-2

SURVEILLANCE CAPSULE LOCATION FLUX AND FLUENCE

FOR IRRADIATION FROM 5/14/74 TO 2/2/85

Wire (Element)	Wire Weight (g)	dps/g Element (at end of Irradiation	Reaction Rate [dps/nucleus (saturated)]	Full Power Flux[a] (n/cm^2-s)		Fluence (n/cm^2)	
				>1 MeV	>0.1 MeV	>1 MeV	>0.1 MeV
Copper 64713	0.3611	1.99×10^{4}	7.15×10^{-18}				
Copper 64741	0.3588	1.87×10^{4}	6.75×10^{-18}				
		Average =	6.95×10^{-18}	2.6×10^{9}	4.2×10^{9}	4.9×10^{17}	7.8×10^{17}
Iron 64713	0.0694	1.39×10^{5}	3.67×10^{-16}				
Iron 64741	0.1663	1.37×10^{5}	3.62×10^{-16}				
		Average =	3.65×10^{-16}	2.3×10^{9}			
Nickel 64713	0.3234	1.84×10^{6}	4.04×10^{-16}				
Nickel 64741	0.3211	1.93×10^{6}	4.24×10^{-16}				
		Average =	4.14×10^{-16}	2.0×10^{9}			

[a] Full power of 1593 MW_t.

5. CHARPY V-NOTCH IMPACT TESTING

The 24 Charpy impact specimens recovered from the surveillance capsule and 38 of the unirradiated archive Charpy impact specimens provided by IEL&P were tested at temperatures selected to establish the toughness transition and USE of the RPV beltline materials. Testing was conducted in accordance with ASTM E23-82 (Reference 10).

5.1 PROCEDURE

The testing machine used was a Riehle Model PL-2 impact machine, serial number R-89916. The pendulum has a maximum velocity of 15.44 ft/sec and a maximum available hammer energy of 240 ft-lb. The test apparatus and operator were qualified using U.S. Army Watertown standard specimens. The standards are designed to fail at 74.1 ft-lb and 13.9 ft-lb at a test temperature of -40°F. According to Reference 10, the averaged test apparatus results must reproduce the Watertown standard values within an accuracy of ±5% or ±1.0 ft-lb, whichever is greater. The successful qualification of the Riehle machine and operator is summarized in Table 5-1.

Charpy V-Notch tests were conducted at temperatures between -100°F and 400°F. For tests between 32°F and 212°F, the temperature conditioning fluid was water. Dichloromethane was used at temperatures below 32°F. Above 212°F, a silicone oil was used. Cooling of the conditioning fluids was accomplished by circulating liquid nitrogen through an immersed coil of copper tubing, and heating by an immersion heater. The fluids were mechanically stirred to maintain uniform temperatures. The fluid temperature was measured by a chromel-alumel thermocouple and a copper-constantan thermocouple. These were calibrated with boiling water (212°F), and ice water (32°F). Once at test temperature, the specimens were manually transferred with centering tongs to the Riehle machine and impacted within 5 seconds.

For each Charpy V-Notch specimen tested, test temperature, energy absorbed, lateral expansion, and percent shear were evaluated. Lateral expansion and percent shear were measured according to Reference 10 methods. Percent shear was determined with Method 2 of Subsection 11.2.4.3 of Reference 10, which involves a comparison of the fracture surface appearance with the reference fracture surfaces in Figure 15 of Reference 10.

5.2 RESULTS

Eight irradiated Charpy V-Notch specimens each of base metal, weld metal and HAZ were tested at temperatures selected to define the toughness transition and USE portions of the fracture toughness curve. Twelve each of the unirradiated base, weld and HAZ specimens were similarly tested. Subsequently, two additional unirradiated base metal specimens were tested to better define the transition region. Absorbed energy, lateral expansion and percent shear data are listed in Table 5-2 for the irradiated specimens and in Table 5-3 for the unirradiated specimens. Plots of absorbed energy data for base, weld, and HAZ metal are presented in Figures 5-1, 5-2 and 5-3, respectively. Lateral expansion plots for base, weld and HAZ metal are given in Figures 5-4, 5-5 and 5-6, respectively. The plots show the unirradiated and irradiated data side-by-side to facilitate the determination of irradiation shifts.

The data sets were freehand-fit with best-estimate S-shaped curves, characteristic of fracture toughness transition curves. The curves of impact energy and lateral expansion were used to determine the index temperatures shown in Table 5-4. Longitudinal USE values were taken from the curves, and then corrected to equivalent transverse USE with the methods in Reference 7.

The HAZ data typically show the greatest scatter, because the HAZ has in effect been uniquely heat treated by the welding process, and because of the uncertainty of the specimen notch location relative to the weld fusion line. The HAZ results in this case are very consistent, probably because there is little difference in the Charpy curves for the base and weld metals.

An unirradiated base metal specimen was tested at 400°F, resulting in a no-break test. This was unexpected, since the material has an upper shelf of 160 ft-lb and the test machine capacity is 240 ft-lb. The specimen appeared to have been struck properly, but slight misalignment may have caused the no-break result. The result is not shown on Figure 5-1. A retest at 400°F was not considered necessary because the USE is well defined by the other data shown on Figure 5-1.

Photographs were taken of the Charpy specimen fracture surfaces. The fracture surface photographs were used to evaluate percent shear. The photographs and a summary of test results for each specimen are contained in Appendix A for irradiated specimens and in Appendix B for unirradiated specimens.

5.3 IRRADIATED VERSUS UNIRRADIATED CHARPY V-NOTCH PROPERTIES

The differences in index temperatures shown in Table 5-4 represent the change in fracture toughness properties due to the amount of irradiation experienced by the surveillance capsule. Reference 4 states that the change in RT_{NDT} of a material should be characterized by the change in the 30 ft-lb index temperature. Therefore, the experimental shift in RT_{NDT} for the base metal is 42°F, and for the weld metal is 0°F.

Table 5-1

QUALIFICATION TEST RESULTS USING
U.S. ARMY WATERTOWN SPECIMENS
(TESTED IN FEBRUARY 1986)

Qualification Test Specimen Identification	Test Temperature (°F)	Energy Absorbed Mechanical Gage (ft-lb)
EE3-0427	-40	72.0
EE3-0075	"	74.5
EE3-0233	"	80.0
EE3-0956	"	75.8
EE3-0365	"	74.2
Average		75.3
Allowable	-40	74.1 ± 3.7 Acceptable
DD5-0442	-40	15.0
DD5-0905	"	13.8
DD5-0486	"	14.0
DD5-0977	"	14.0
DD5-0831	"	14.0
Average		14.2
Allowable	-40	13.9 ± 1.0 Acceptable

Table 5-2

CHARPY V-NOTCH IMPACT TEST RESULTS
FOR IRRADIATED RPV MATERIALS

Specimen Identification	Test Temperature (°F)	Fracture Energy (ft-lb)	Lateral Expansion (mils)	Percent Shear (Method 2) (%)
Base:				
EBU	-60	4.0	4	0
EBP	-20	15.0	19	0
EBT	10	15.5	26	40
EC1	20	49.5	45	40
EBK	40	60.0	49	40
EBJ	120	101.5	70	70
EBL	200	144.5	95	90
EBY	400	160.0	98	90
Weld:				
EJ1	-100	6.5	11	0
EJ2	-60	15.3	24	10
EJ4	-40	22.0	26	10
EE5	-20	63.5	54	40
EE2	40	75.2	69	60
EE3	120	97.2	86	80
EE4	200	109.0	90	70
EJ3	400	101.0	91	100
HAZ:				
ELY	-60	8.0	15	0
ELP	-20	13.5	18	20
EMY	0	15.0	16	30
ELT	10	51.7	43	40
ELK	40	65.0	50	50
ELL	120	90.1	66	70
ELM	200	126.5	84	90
EMT	400	124.5	75	100

Table 5-3

CHARPY V-NOTCH IMPACT TEST RESULTS
FOR UNIRRADIATED RPV MATERIALS

Specimen Identification	Test Temperature (°F)	Fracture Energy (ft-lb)	Lateral Expansion (mils)	Percent Shear (Method 2) (%)
Base:				
ED4	-100	6.5	7	0
ED1	-80	4.2	5	0
ECT	-40	9.0	14	10
EDK	-40	24.0	30	5
ED7	-30	38.0	35	5
EDT	-30	49.0	30	5
ED6	-20	47.0	43	10
ED5	-10	43.0	43	20
ECP	0	56.5	49	25
ECM	40	98.5	73	40
ECL	120	134.5	73	80
ECK	200	158.5	93	85
EDY	300	163.5	81	90
EDA	400		no break	
Weld:				
EET	-100	6.5	11	10
EEK	-80	5.5	10	15
EK3	-60	16.8	19	40
EK4	-50	21.5	22	20
EEE	-40	49.0	45	20
EK2	-20	58.0	57	50
EEL	0	52.5	50	30
EEY	40	65.0	61	50
EED	120	102.0	88	80
EEM	200	94.0	65	70
EK5	300	88.2	93	60
EK6	400	113.7	96	70
HAZ:				
EM6	-100	5.8	12	5
EM5	-80	8.0	17	10
EM4	-40	23.8	25	20
EM3	0	28.0	28	20
EP2	20	37.0	36	30
EMA	40	65.0	62	40
EP3	80	71.5	67	60
EM1	120	90.0	77	80
EP4	160	109.0	90	60
EM2	200	128.0	89	85
EP5	300	113.0	89	60
EP7	400	107.0	94	85

Table 5-4

COMPARISON OF UNIRRADIATED AND IRRADIATED
CHARPY V-NOTCH DATA

Material	Index Temperature (°F) E=30 ft-lb	E=50 ft-lb	MLE=35 mil	Upper[a] Shelf Energy (ft-lb) L/T
Base:				
Unirradiated	-35°F	-22°F	-26°F	164/107
Irradiated	7°F	23°F	16°F	160/104
Difference	42°F	45°F	42°F	4/3 (2.5%)
Weld:				
Unirradiated	-33°F	-11°F	-34°F	101/101
Irradiated	-33°F	-11°F	-34°F	101/101
Difference	0°F	0°F	0°F	0/0 (0%)
HAZ:				
Unirradiated	-6°F	34°F	-2°F	112/73
Irradiated	3°F	38°F	40°F	126/82
Difference	9°F	4°F	42°F	-14/-9 (-11.1%)

[a] Longitudinal (L) USE is read directly from Figures 5-1, 5-2 and 5-3. Transverse (T) USE is taken as 65% of the longitudinal USE, according to Reference 7. L/T USE values are equal for weld metal, which has no orientation effect.

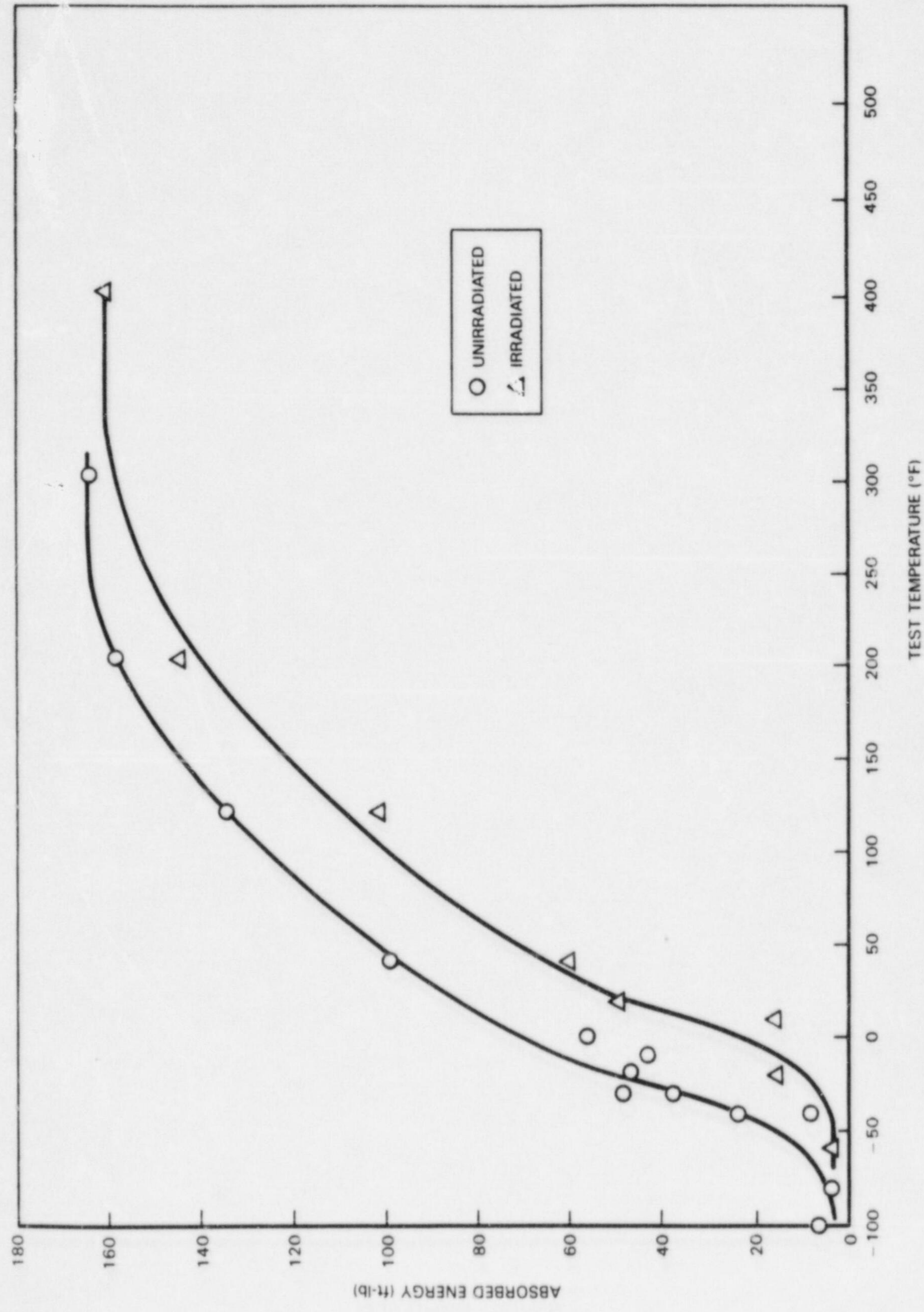


Figure 5-1. DAEC Base Metal Impact Energy

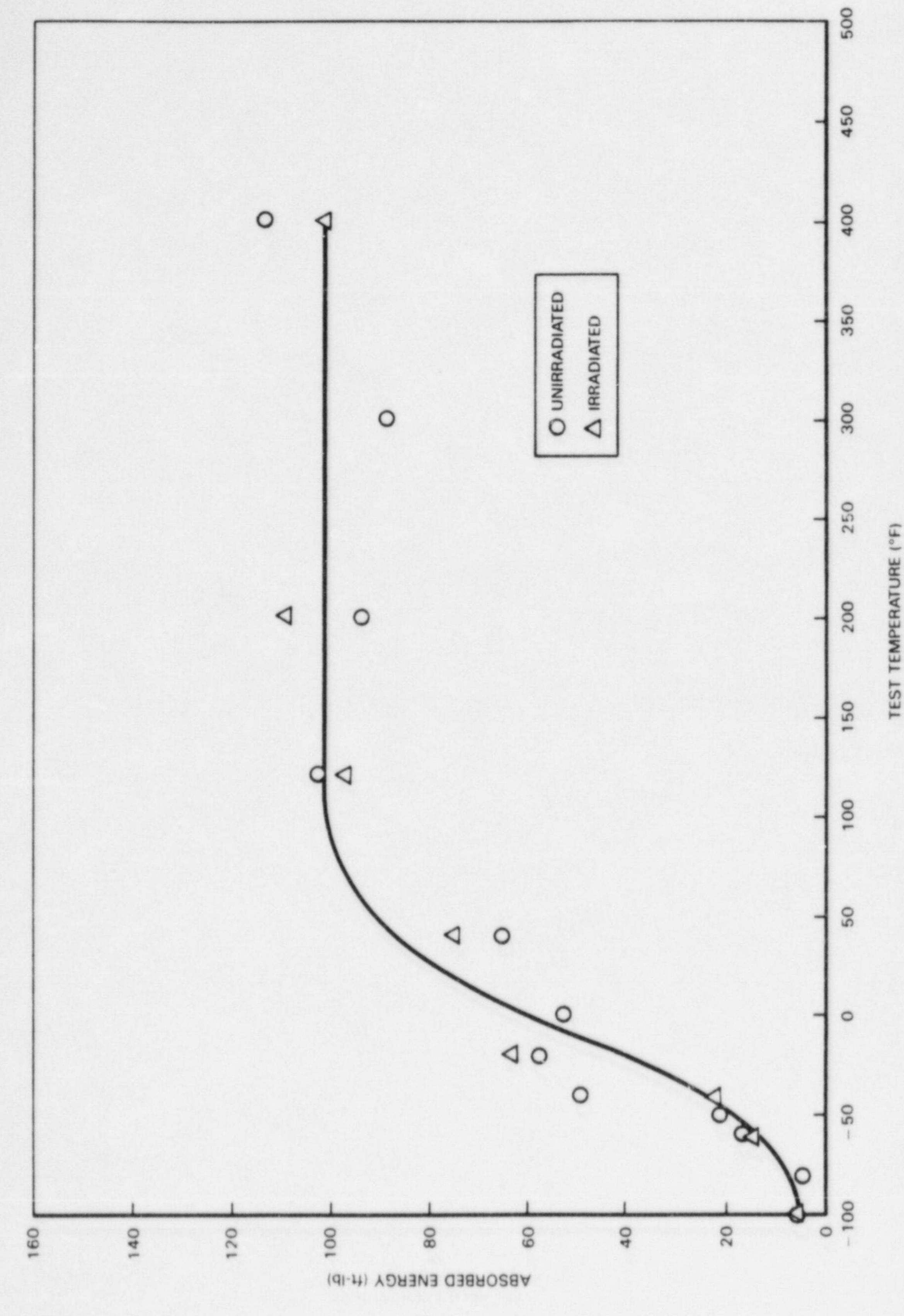


Figure 5-2. DAEC Weld Metal Impact Energy

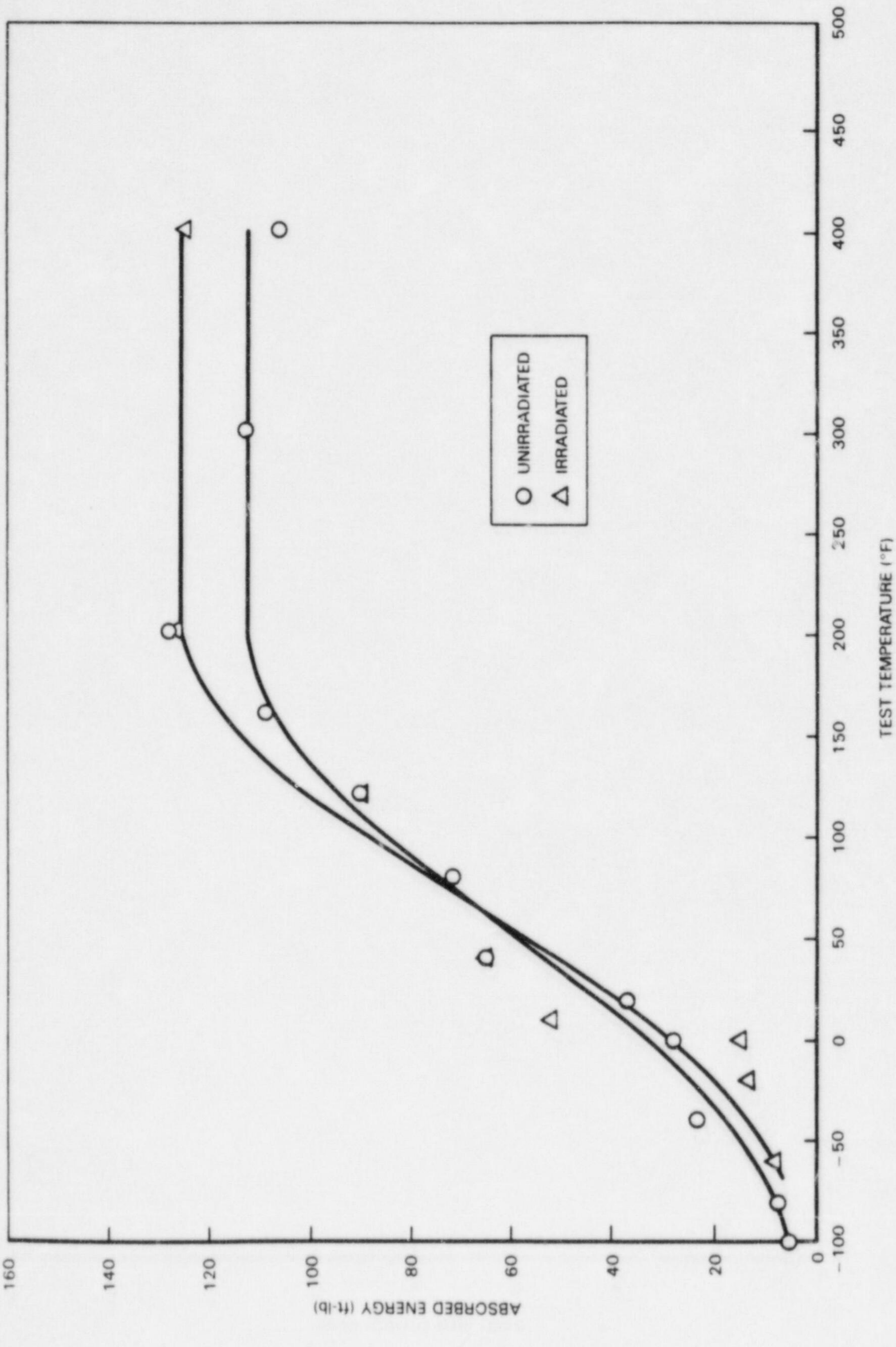


Figure 5-3. DAEC HAZ Metal Impact Energy

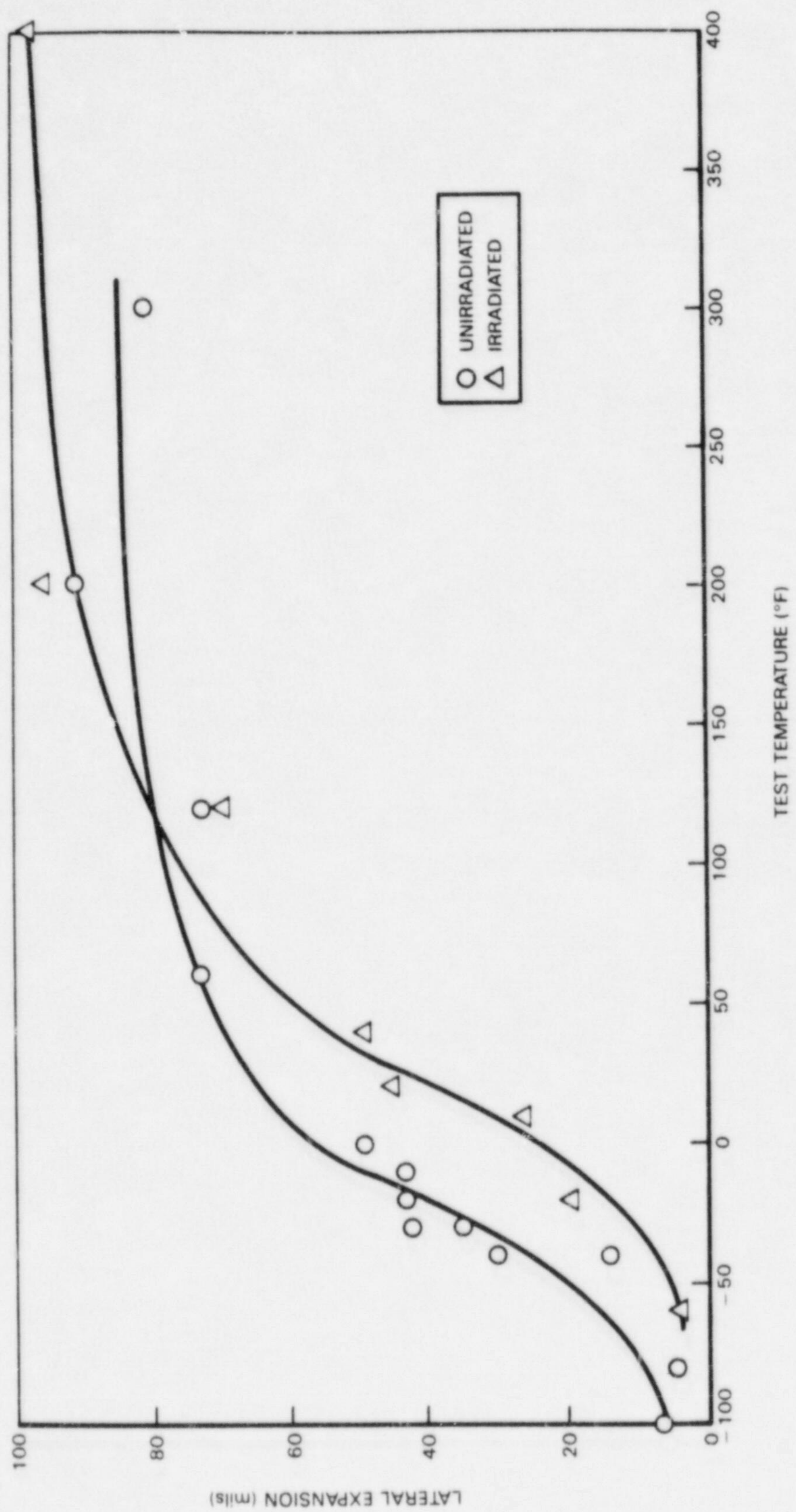


Figure 5-4. DAEC Base Metal Lateral Expansion

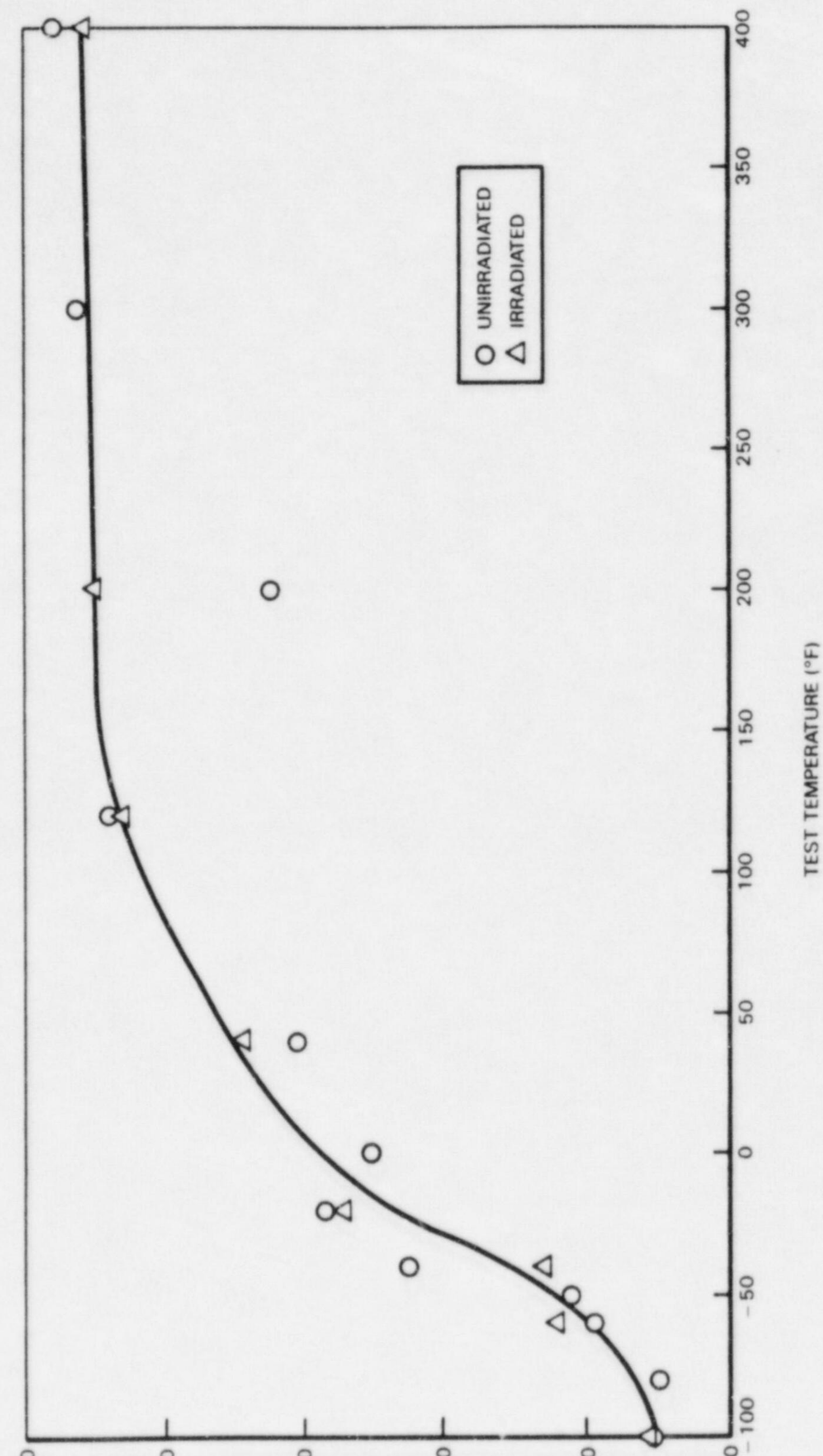


Figure 5-5. DAEC Weld Metal Lateral Expansion

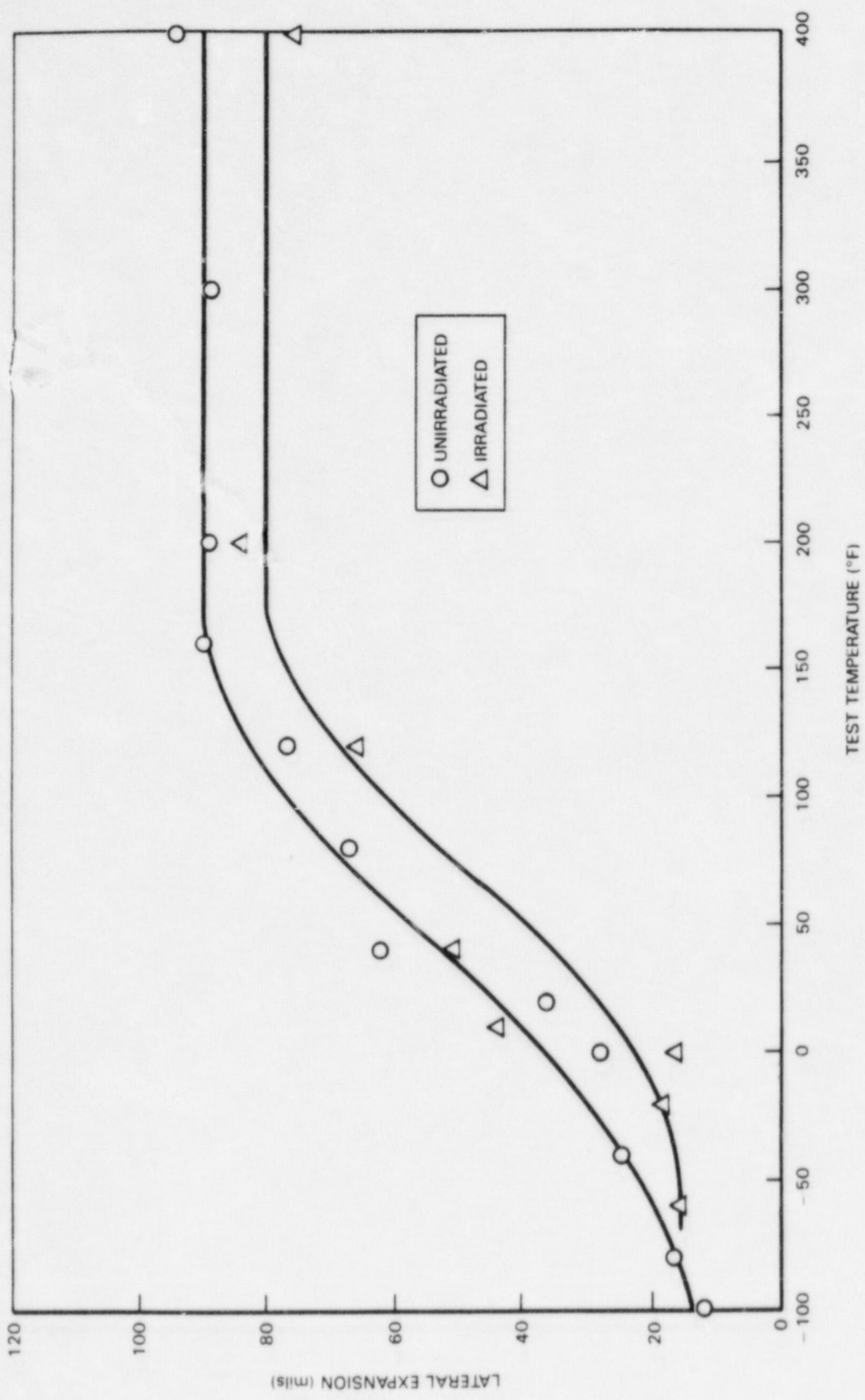


Figure 5-6. DAEC HAZ Metal Lateral Expansion

6. TENSILE TESTING

Uniaxial tensile tests were conducted on six irradiated tensile specimens recovered from the vessel surveillance capsule and on six unirradiated archive specimens provided by IEL&P. Tests were conducted in accordance with ASTM E8-81 (Reference 11) in air at room temperature, at onset to upper shelf temperature and at RPV operating temperature.

6.1 PROCEDURE

All tests were conducted using a screw-driven Instron test frame equipped with a 20-kip load cell and special pull bars and grips. Heating was done with a Satec resistance clamshell furnace centered around the specimen load train. Test temperature was monitored and controlled by a chromel-alumel thermocouple spot-welded to an Inconel clip that was friction-clipped to the surface of the specimen at its midline. Before the elevated temperature tests, a profile of the furnace was conducted at the test temperature of interest using an unirradiated steel specimen of the same geometry. Thermocouples were spot-welded to the top, middle, and bottom of a central 1-in. gage of this specimen. In addition, the clip-on thermocouple was attached to the midline of the specimen. When the target temperatures of the three thermocouples were within ±5°F of each other, the temperature of the clip-on thermocouple was noted and subsequently used as the target temperature for the irradiated specimens.

All tests were conducted at a calibrated crosshead speed of 0.005 in./min until well past yield, at which time the speed was increased to 0.05 in./min until fracture. A 1-in. span knife edge extensometer was attached directly to each specimen's central gage region and was used to monitor gage extension during the test.

The test specimens were machined with a minimum diameter of 0.250 inch at the center of the gage length. Irradiated specimens of base, weld and HAZ metal were tested at room temperature (RT = 76°F) and RPV operating temperature (550°F). Unirradiated base and weld metal specimens were tested at room temperature, at operating temperature and at the temperature corresponding to the onset of the material's upper shelf (185°F for base metal and 130°F for weld metal).

The yield strength (YS) and ultimate tensile strength (UTS) were calculated by dividing the nominal area (0.0491 in.^{2}) into the 0.2% offset load and into the maximum test load, respectively. The values listed for the uniform and total elongations were obtained from plots that recorded load versus specimen extension and are based on a 1-in. gage length. Reduction of area (RA) values were determined from post-test measurements of the necked specimen diameters using a calibrated blade micrometer and employing the formula:

$$RA(\%) = 100\% * (A_o - A_f)/A_o$$

After testing, each broken specimen was photographed end-on, showing the fracture surface, and lengthwise, showing fracture location and local necking behavior.

6.2 RESULTS

Tensile test properties of YS, UTS, RA, uniform elongation (UE) and total elongation (TE) are presented in Table 6-1 for irradiated and unirradiated specimens. Shown in Figure 6-1 is a stress-strain curve for a 550°F base metal irradiated specimen typical of the stress-strain characteristics of all the specimens tested. The results presented in Table 6-1 are shown graphically as follows:
YS in Figure 6-2, UTS in Figure 6-3, UE in Figure 6-4, TE in Figure 6-5 and RA in Figure 6-6. Photographs of fracture surfaces and necking behavior are given in Figures 6-7, 6-8 and 6-9 for irradiated

base, weld and HAZ specimens, respectively. The unirradiated specimen photographs are in Figure 6-10 for base metal and in Figure 6-11 for weld metal.

The base, weld, and HAZ materials follow the trend of decreasing properties with increasing temperature. The three materials behave very similarly, as seen in Figures 6-2 through 6-6. The third data value for the unirradiated tests shows that the relationship is non-linear. Straight lines have been drawn only to show the trends.

6.3 IRRADIATED VERSUS UNIRRADIATED TENSILE PROPERTIES

Unirradiated tensile test data are compared in Table 6-2 to the irradiated base metal and weld metal specimen data at 76°F and at 550°F to determine the degree of radiation strengthening. The percent difference is expressed relative to the irradiated value, as explained by the equation in the footnote to Table 6-2.

The changes in properties expected with irradiation embrittlement are an increase in YS and UTS with a corresponding decrease in UE, TE and RA. As shown in Table 6-2, the base metal values reflect such a trend, indicating that a measurable change has occurred as a result of irradiation. The weld metal properties show slight increases in YS and UTS, but the changes in UE, TE and RA are inconsistent with embrittlement. Any change in the weld metal properties as a result of irradiation would likely be small. These conclusions are consistent with the RT_{NDT} shifts reported in Section 5.

Table 6-1

TENSILE TEST RESULTS FOR RPV MATERIALS

Specimen Number	Material	Test Temp (°F)	Yield Strength (ksi)	Ultimate Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction of Area (%)
Irradiated:							
ETJ	Base	76	73.4	96.2	9.5	19.7	70.4
ETK	Base	550	67.7	91.7	7.6	15.9	65.9
EU3	Weld	76	79.4	92.9	8.2	17.5	70.0
EU6	Weld	550	65.7	83.5	7.1	15.1	64.0
EY1	HAZ	76	74.2	94.8	9.7	20.7	66.8
EY2	HAZ	550	67.7	87.6	6.6	14.4	59.0
Unirradiated:							
ETA	Base	76	66.5	90.2	10.2	20.8	71.1
ET2	Base	185	63.2	90.5	9.0	19.3	74.3
ET3	Base	550	61.2	86.7	8.2	16.9	70.0
EUP	Weld	76	75.4	89.7	10.0	21.6	71.3
EUJ	Weld	130	80.4	91.4	9.2	21.2	71.6
EUM	Weld	550	66.6	83.6	6.1	14.2	63.2

Table 6-2

COMPARISON OF UNIRRADIATED AND IRRADIATED TENSILE PROPERTIES

	Yield Strength (ksi)	Ultimate Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction of Area (%)
Base Metal at 76°F:					
Unirradiated	66.5	90.2	10.2	20.8	71.1
Irradiated	73.4	96.2	9.5	19.7	70.4
Difference [c]	9.4%	6.2%	-7.4%	-5.6%	-1.0%
Base Metal at 550°F:					
Unirradiated	61.2	86.7	8.2	16.9	70.0
Irradiated	67.7	91.7	7.6	15.9	65.9
Difference	9.6%	5.5%	-7.9%	-6.3%	-6.2%
Weld Metal at 76°F:					
Unirradiated	75.4	89.7	10.0	21.6	71.3
Irradiated	79.4	92.9	8.2	17.5	70.0
Difference	5.0%	3.4%	-22.0%	-23.4%	-1.9%
Weld Metal at 550°F:					
Unirradiated	66.6	83.6	6.1	14.2	63.2
Irradiated	65.7	83.5	7.1	15.1	64.0
Difference	-1.4%	-0.1%	14.0%	6.0%	1.3%

[c] Difference = [(Irradiated - Unirradiated)/Irradiated] * 100%

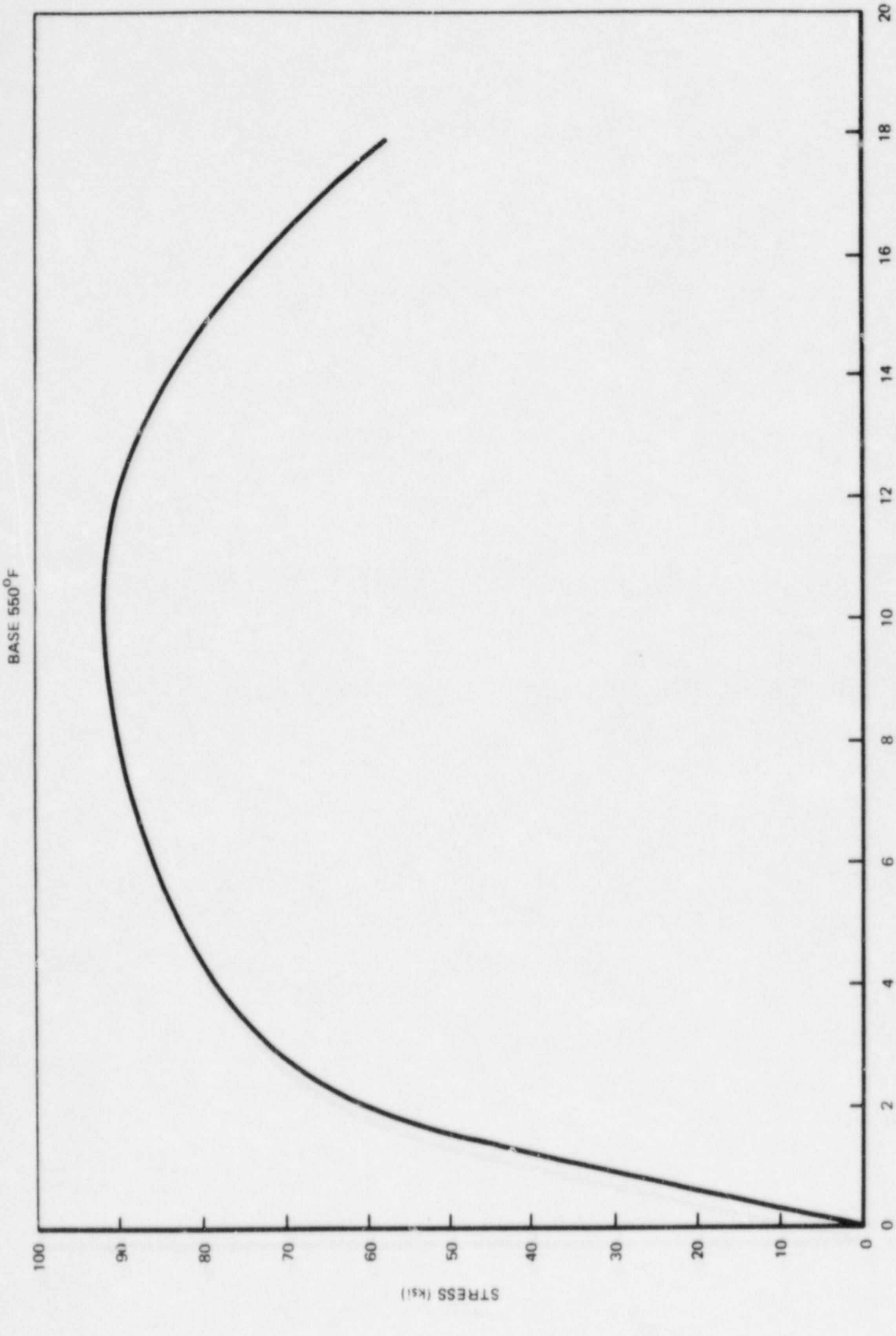


Figure 6-1. Typical Engineering Stress Versus Percent Strain for Irradiated RPV Materials

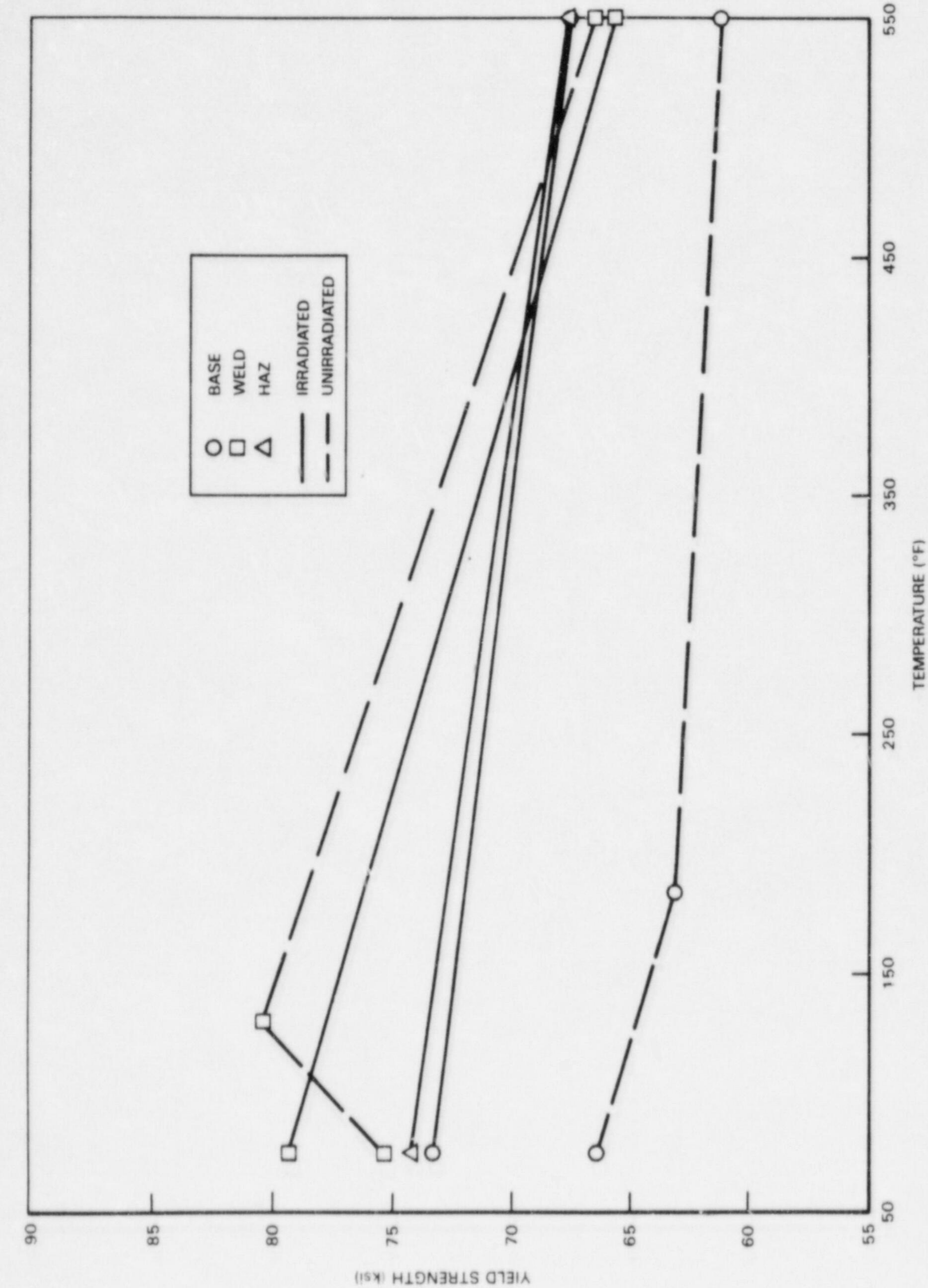


Figure 6-2. Tensile Specimen Yield Strength Versus Test Temperature

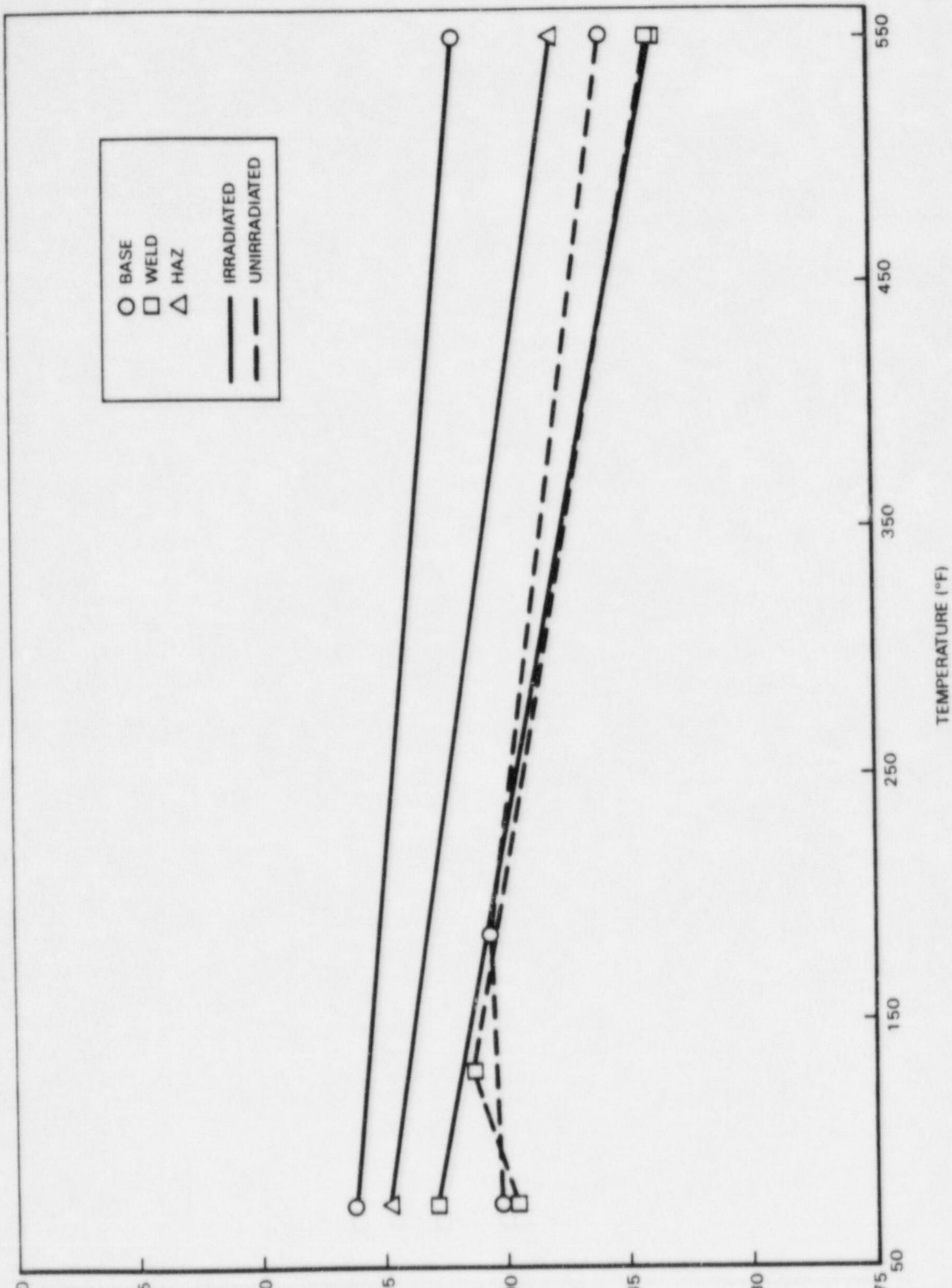


Figure 6-3. Tensile Specimen Ultimate Tensile Strength Versus Test Temperature

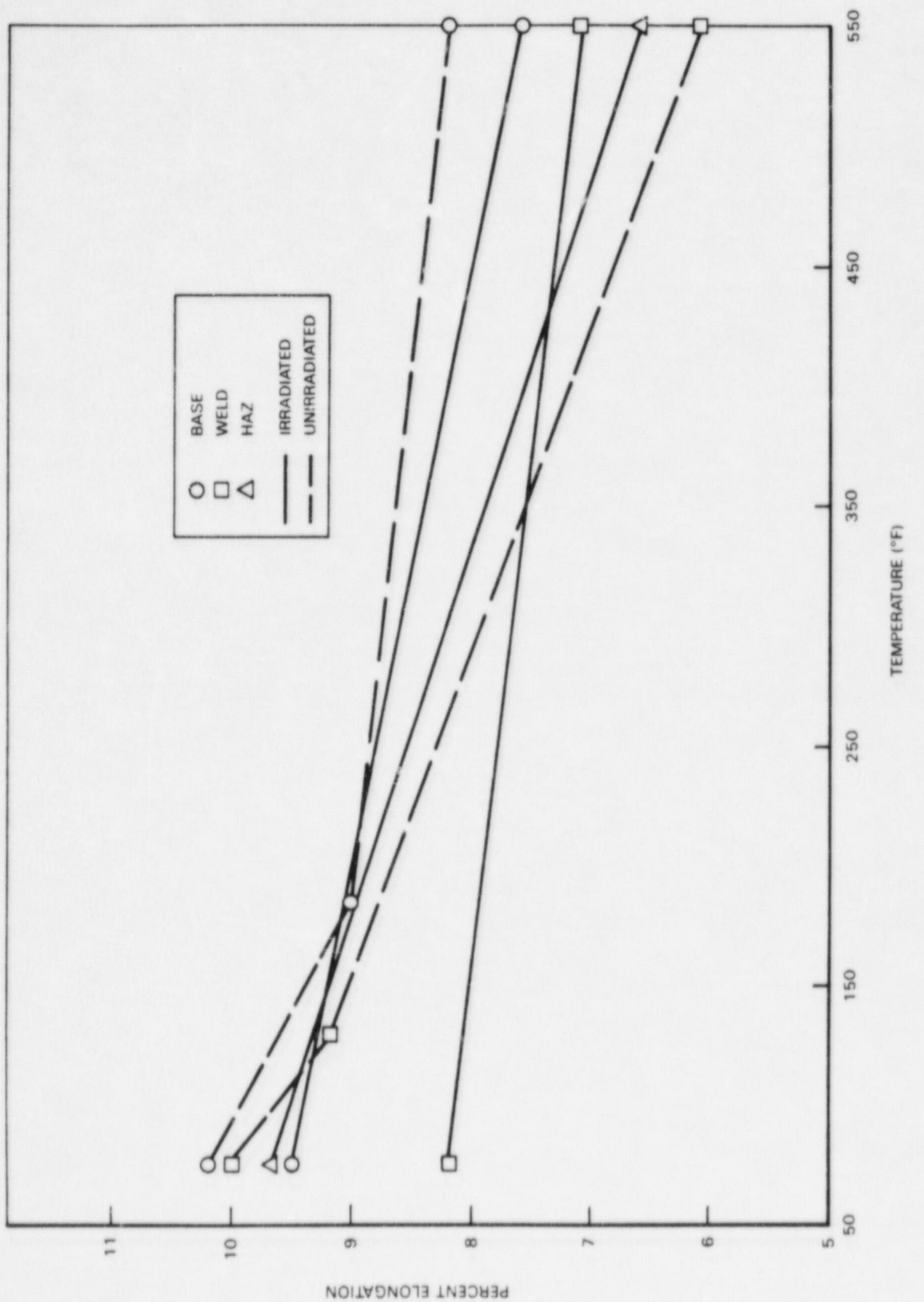


Figure 6-4. Tensile Specimen Uniform Elongation Versus Test Temperature

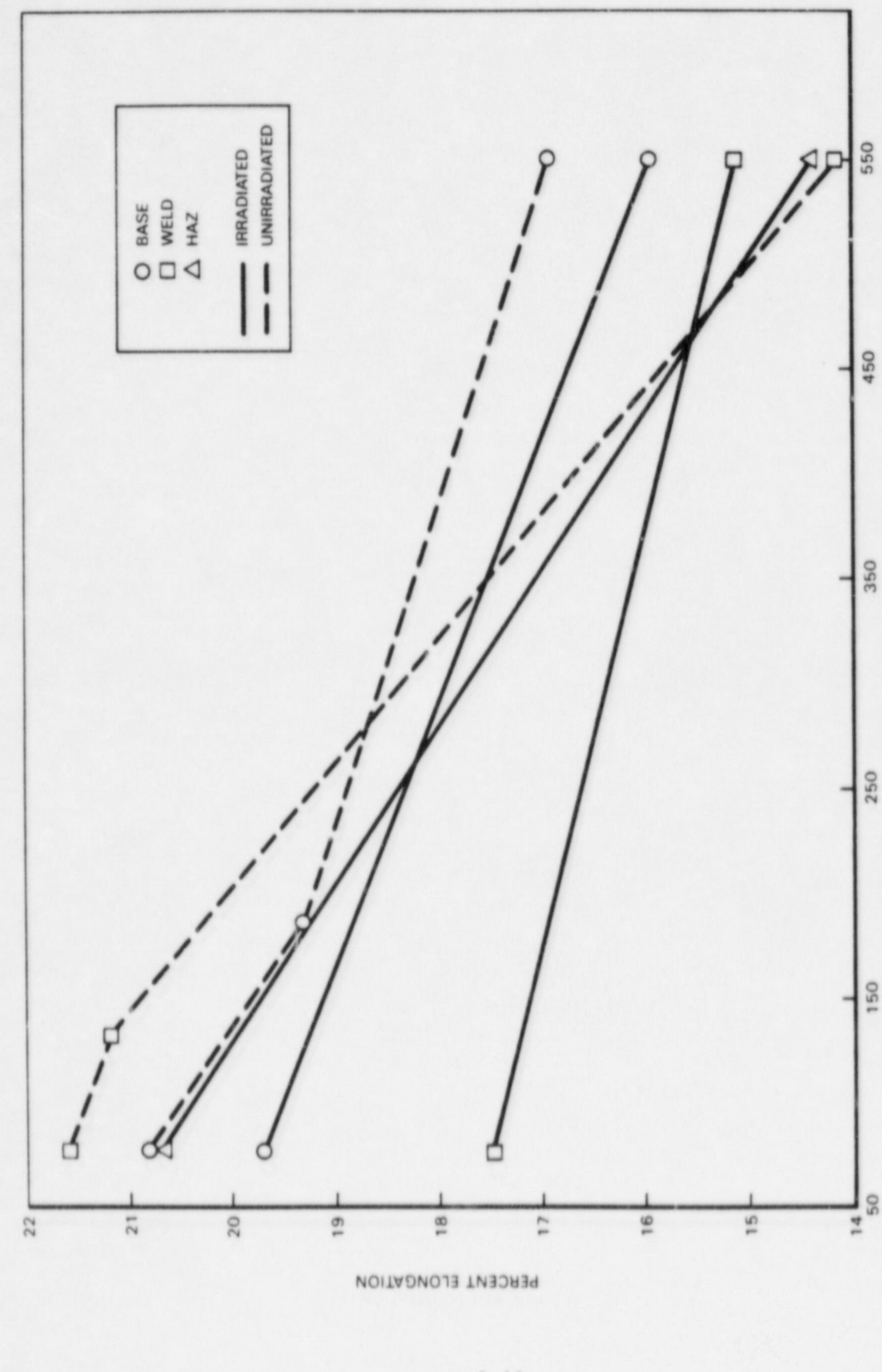


Figure 6-5. Tensile Specimen Total Elongation Versus Test Temperature

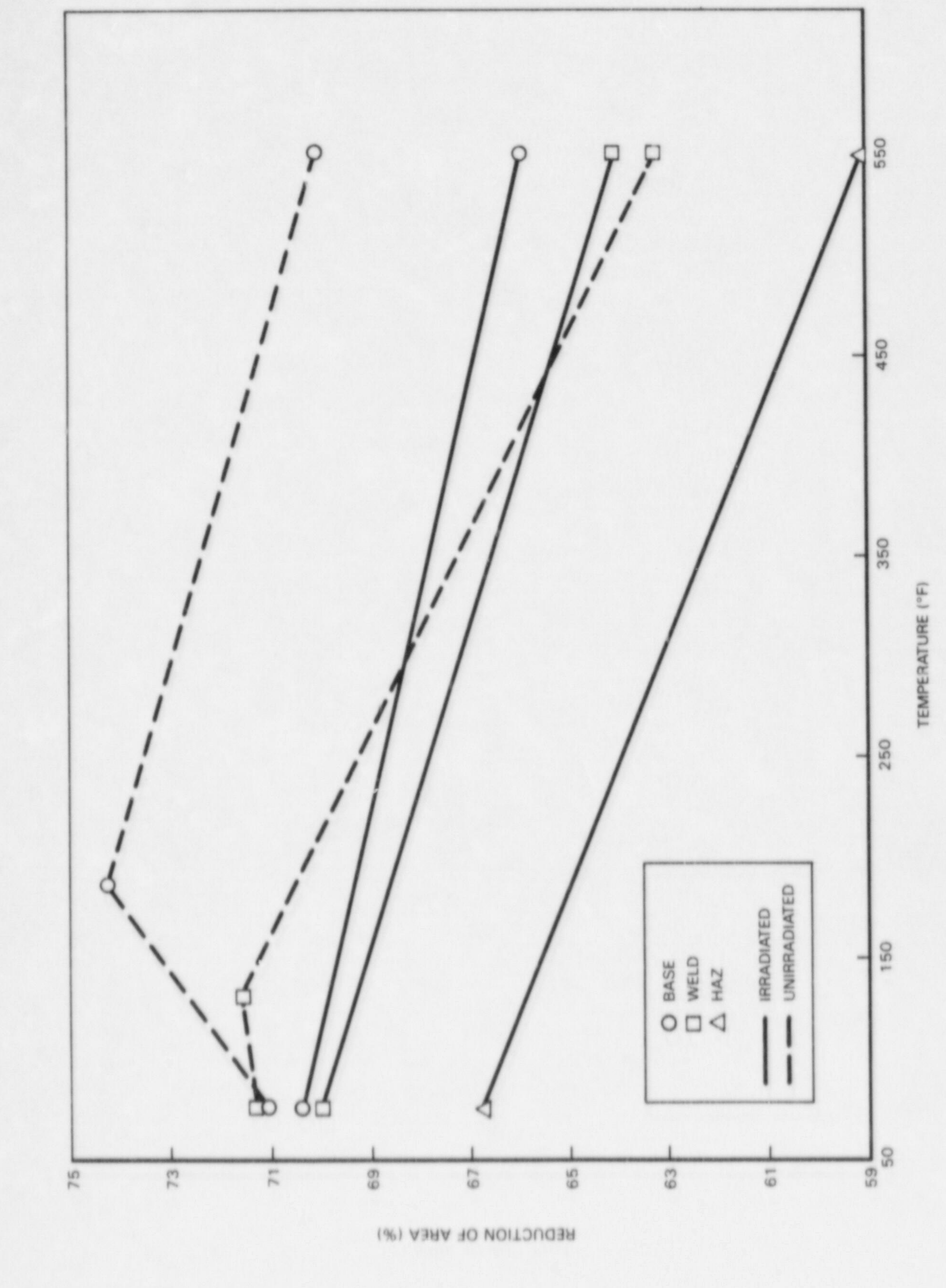
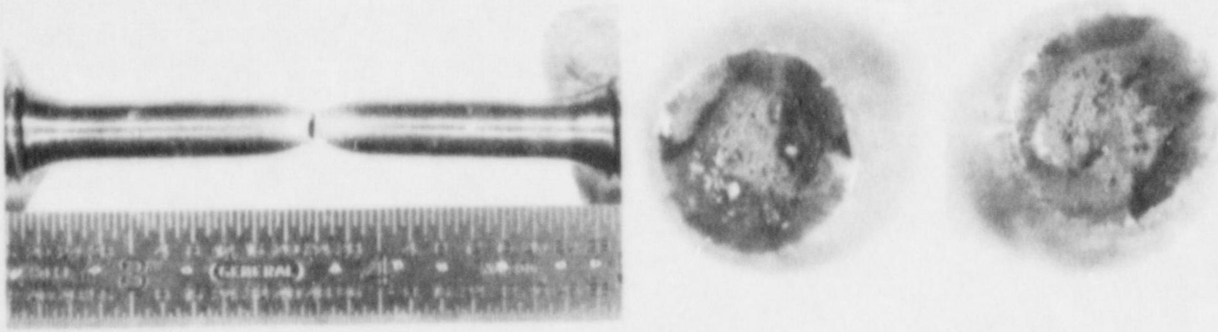
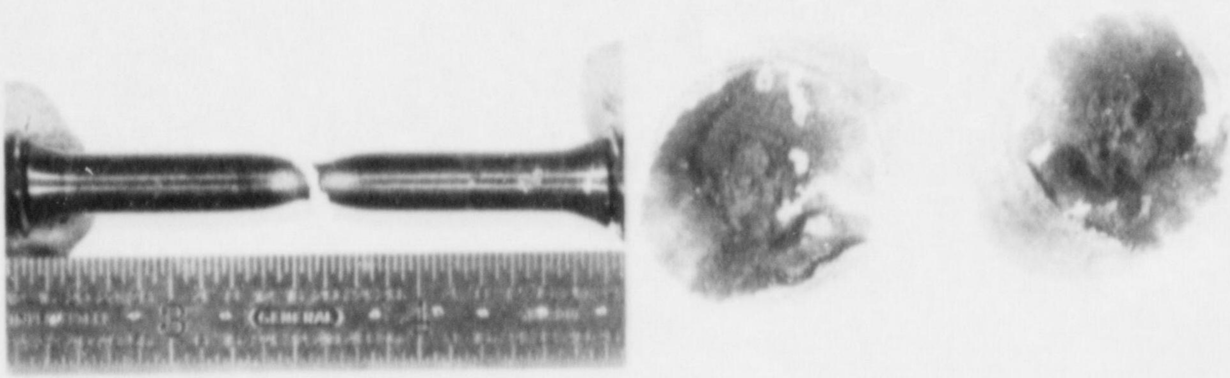


Figure 6-6. Tensile Specimen Reduction of Area Versus Test Temperature

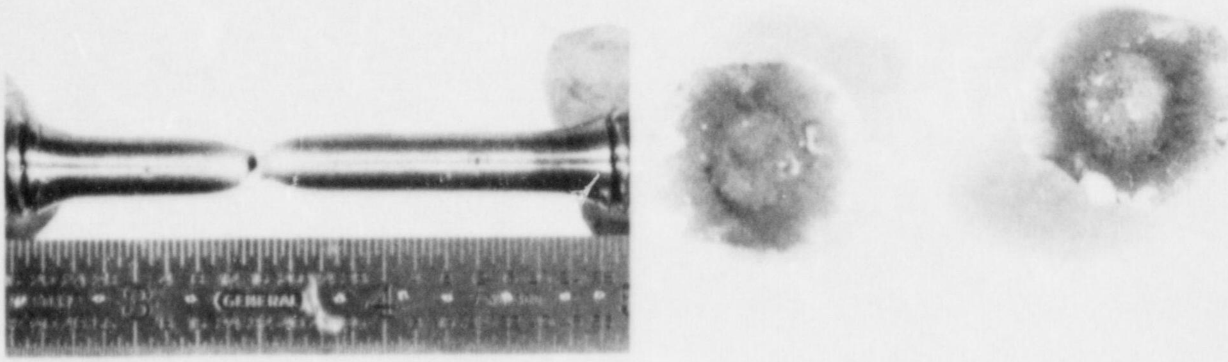


ETJ 76°F

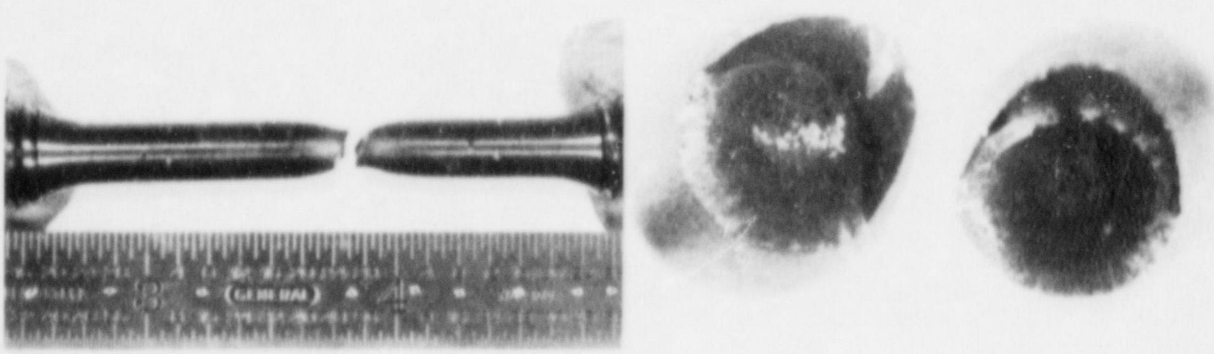


ETK 550°F

Figure 6-7. Fracture Location, Necking Behavior, and Fracture Appearance for Irradiated Base Metal Tensile Specimens

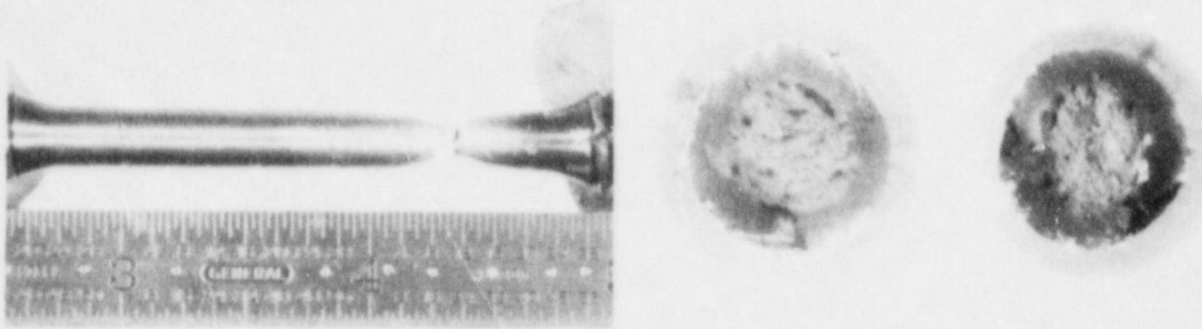


EU3 76°F

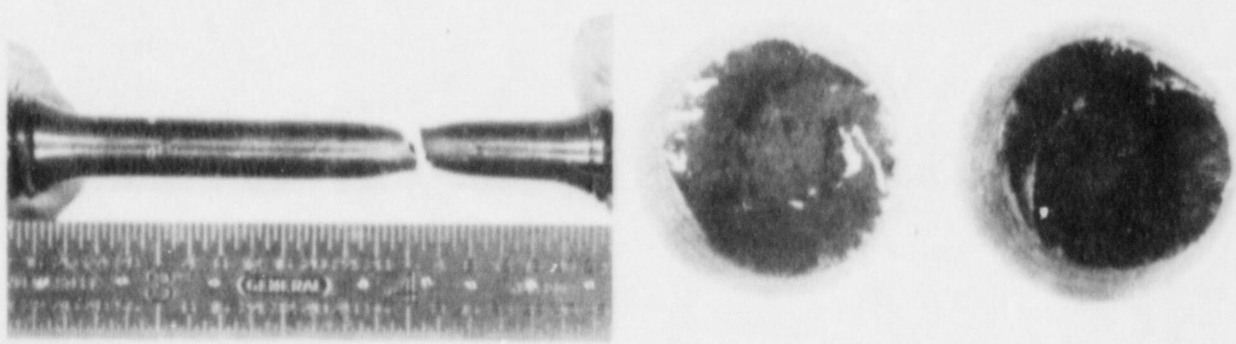


EU6 550°F

Figure 6-8. Fracture Location, Necking Behavior, and Fracture Appearance for Irradiated Weld Metal Tensile Specimens



EY1 76°F



EY2 550°F

Figure 6-9. Fracture Location, Necking Behavior, and Fracture Appearance for Irradiated HAZ Metal Tensile Specimens

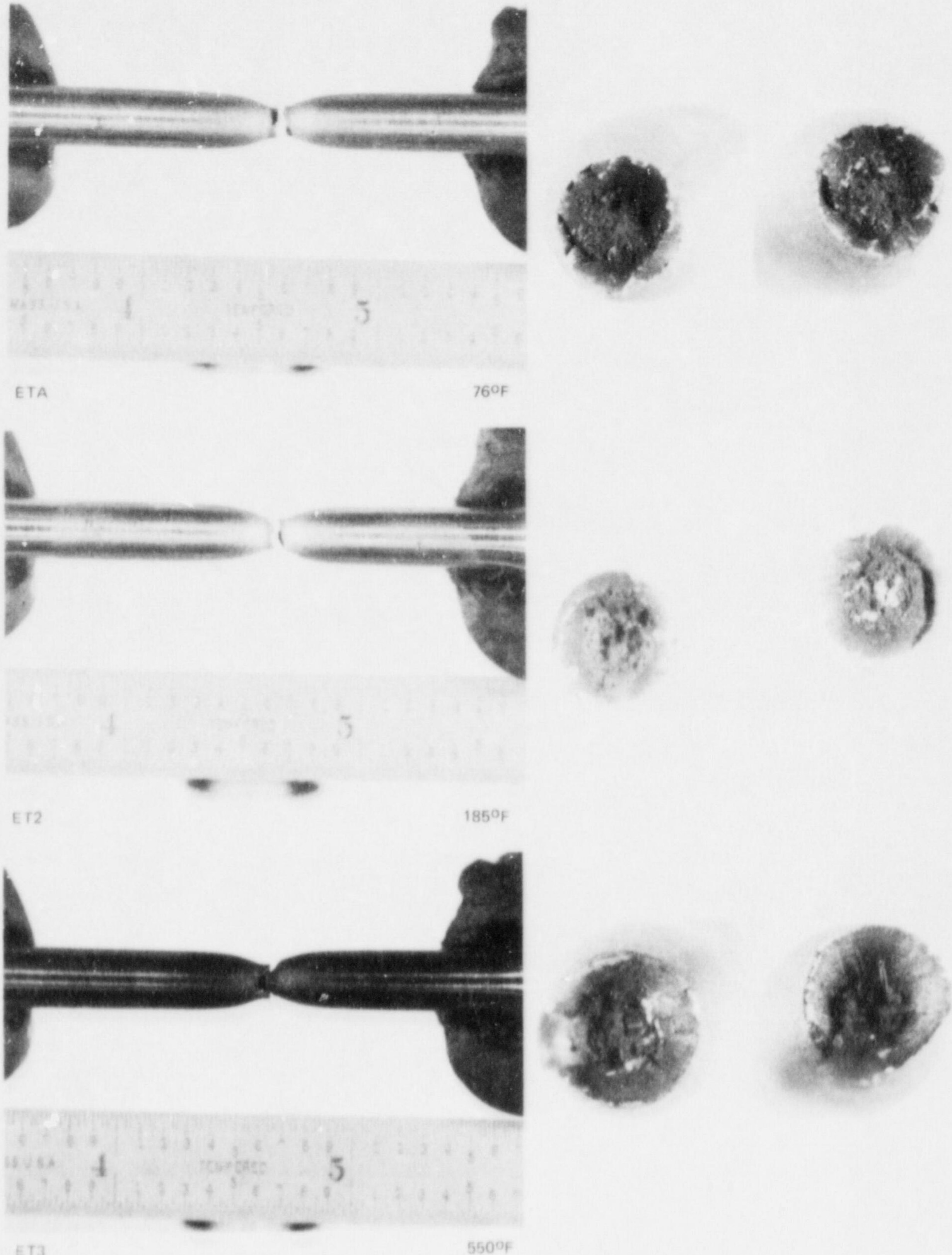


Figure 6-10. Fracture Location, Necking Behavior, and Fracture Appearance for Unirradiated Base Metal Tensile Specimens

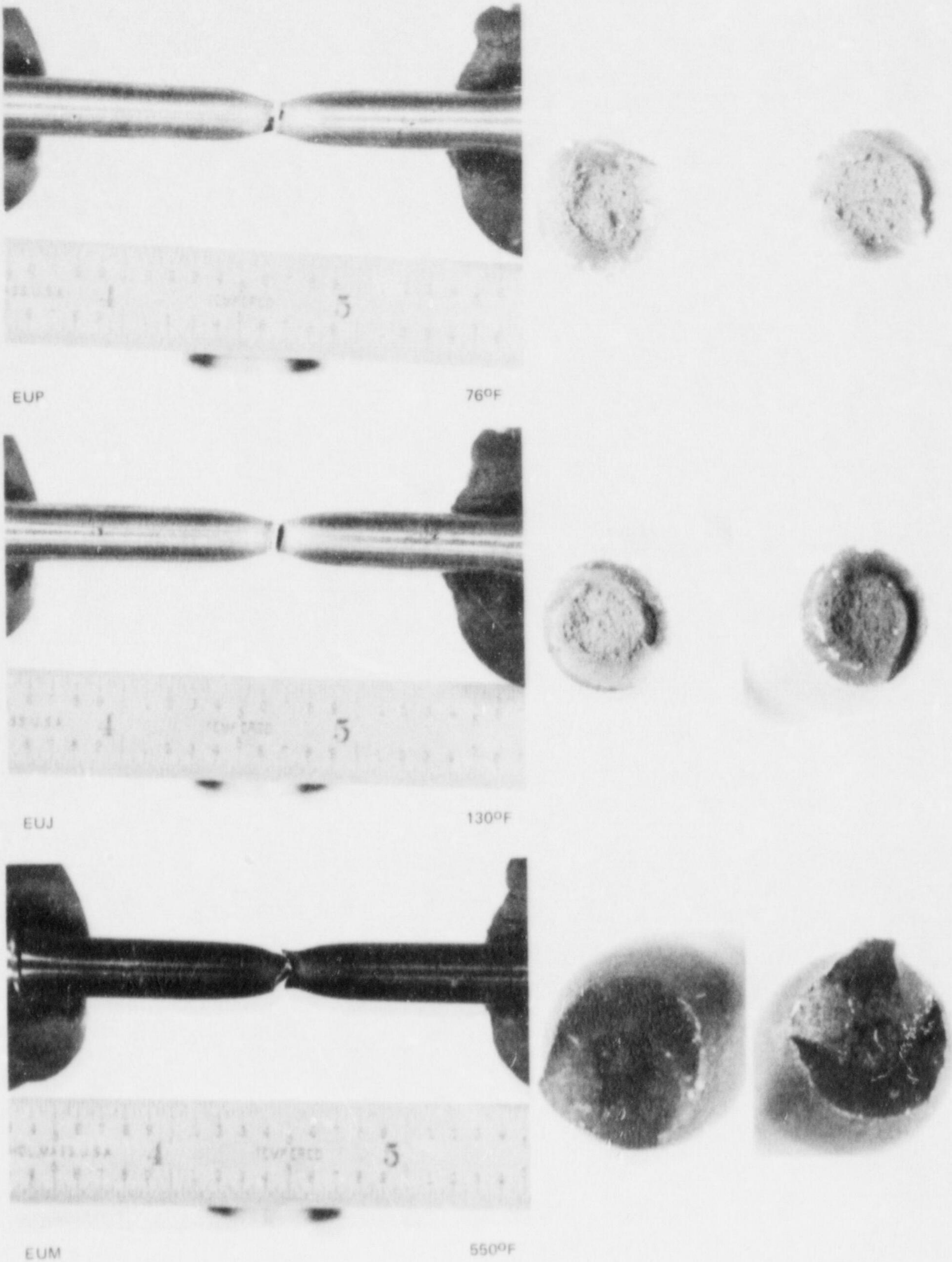


Figure 6-11. Fracture Location, Necking Behavior, and Fracture Appearance for Unirradiated Weld Metal Tensile Specimens

7. DETERMINATION OF OPERATING LIMITS AND END-OF-LIFE CONDITIONS

Reference 6 explains the procedure involved in developing operating limits curves for pressure testing, non-nuclear heatup/cooldown and core critical operation. The vessel is divided into three basic regions: the closure flange region, the beltline region and the non-beltline region. The only region affected by the surveillance test results is the beltline region. Therefore, the procedures discussed in Reference 6 for the other two regions are unchanged. The beltline operating limits are affected by the irradiation shift in RT_{NDT}, as measured by the surveillance testing. The methods used to incorporate the surveillance test results into the operating limits are discussed below.

7.1 EVALUATION OF RADIATION EFFECTS

The shift in fracture toughness properties in the beltline materials is a function of neutron fluence and the presence of certain elements, such as copper (Cu) and phosphorus (P). The specific relationship from Reference 5 is:

$$\text{SHIFT } (^\circ F) = [40 + 1000(\%Cu-0.08) + 5000(\%P-0.008)]*(f/10^{19})^{\frac{1}{2}} \qquad (7\text{-}1)$$

where:

$\%Cu$ = wt % of Cu present,

$\%P$ = wt % of P present,

f = fluence (n/cm^2) at selected EFPY.

The limiting beltline material, either plate or weld, is determined based on the Cu-P content and initial RT_{NDT} of the materials. These data are presented for the beltline materials in Table 3-1.

7.1.1 Measured Versus Predicted Surveillance Shift

Table 5-4 presents a measured shift for the base metal of 42°F and of 0°F for the weld metal. Obviously, the measured shift for the weld metal is less than predicted, but the measured shift for the surveillance plate exceeds the predicted value. The predicted shift of the surveillance plate, calculated according to Equation 7-1, assumes 0.15% Cu, 0.012% P and an upper bound fluence of:

$$f=(4.9\times10^{17}\ n/cm^2 \text{ for capsule})(1.30 \text{ uncertainty})=6.37\times10^{17}\ n/cm^2.$$

Upper bound fluence is used in order to be consistent with the assumptions used to establish the EOL fluence in Subsection 4.3. The predicted shift is 32°F, versus the measured shift of 42°F.

7.1.2 Modification of the Shift Relationship

Since the measured shift exceeds the predicted shift, using Reference 5 methods to predict the limiting beltline plate shift may be non-conservative. Therefore, the relationship in Equation 7-1 is modified for the plate materials. Equation 7-1 shows that the shift calculated is proportional to the material characteristics and to the square root of the fluence. Assuming that the fluence relationship is correct, the coefficient representing the materials in Equation 7-1 must be increased by the factor (42/32) or 1.31. Therefore, values of irradiation shift for the beltline plates will be multiplied by the correction factor of 1.31 to predict shifts for future EFPY and EOL conditions.

7.1.3 Limiting Beltline Material

The limiting beltline material must be determined in order to calculate the appropriate shift to be factored into the operating limits curves. The limiting material is found by evaluating each beltline plate and weld material at the EOL fluence of 3.6×10^{18} n/cm². Table 7-1 shows the four beltline plate chemistries and the worst case weld chemistries. Equation 7-1, including the 1.31 factor for the plates, was used for the listed chemistries and the EOL fluence to calculate the shifts in Table 7-1. These shifts, when added to the initial RT_{NDT} values, result in the adjusted reference temperature (ART) for each material. The highest ART is calculated for plate 1-19. Since 1-19 also has the highest initial RT_{NDT}, it is the limiting beltline material for all values of EFPY.

7.1.4 Operating Limits Curves

The ART for the limiting plate in Table 7-1 is 126°F. This consists of a 40°F initial RT_{NDT} and a shift of 86°F. These are the same values as were calculated in Reference 6 for the limiting material. In Reference 6, the highest Cu, P and initial RT_{NDT} values for the beltline plates were conservatively enveloped. In Table 7-1, the appropriate chemistry and RT_{NDT} values are considered for each plate. The other differences in Reference 6 are the higher value of fluence and the absence of a surveillance test correction factor on shift. As it happens, when these three differences are considered, the resulting ART for the limiting beltline material is the same as was reported in Reference 6. As a result, the operating limits curves generated in Reference 6 are still valid when the results of the surveillance testing are considered. The plot of irradiation shift versus EFPY (Figure 4-4 of Reference 6) for the limiting beltline material is shown in Figure 7-1. Operating limits curves, valid to 12 EFPY (Figure 5-4 of Reference 6) are shown in Figure 7-2.

7.2 PREDICTED END-OF-LIFE CONDITIONS

Reference 1 states several conditions which, if exceeded, require that vessel thermal annealing be performed. The conditions relate to RT_{NDT} and USE as follows:

a. The ART of the limiting beltline material must be less than 200°F at EOL.

b. The USE for all beltline materials must be greater than 50 ft-lb at EOL, accounting for reductions due to irradiation.

As described in the following subsections, these two conditions are met for the beltline materials where data were available.

7.2.1 Adjusted Reference Temperature

The ART of each beltline material is shown for EOL in Table 7-1. The highest value is 126°F for plate 1-19. This is well below the Reference 1 requirement of 200°F.

7.2.2 Upper Shelf Energy

The changes in USE due to irradiation are shown in Table 5-4 for the surveillance plate and weld materials. They are decreases of 2.5% and 0%, respectively. Predictions of decrease in USE can be made using Reference 5 as well. For the surveillance materials at a fluence of 4.9×10^{17} n/cm², decreases of 12% for the plate and 9% for the weld are predicted. The actual surveillance results show the Reference 5 values to be quite conservative.

USE predictions, based on Reference 5, are shown in Table 7-2. The unirradiated USE values for plate 1-21 and for the beltline weld were taken from Figures 5-1 and 5-2, respectively. The percentages of decrease in USE for these materials were based on Reference 5, rather than on actual surveillance results, for conservatism.

In addition to the surveillance specimen data, Charpy data were available for beltline plate 1-20 (Heat B0436-2). These data were generated by CB&I as part of the fabrication test program. The Charpy tests were performed up to temperatures of 200°F, which provides a reasonable and conservative estimate of the USE. The unirradiated USE measured for plate 1-20 is included in Table 7-2, along with the predicted EOL value of USE. USE data for beltline plates 1-18 and 1-19 were not available.

The USE values in Table 7-2 are based on longitudinal Charpy specimens. Reference 7 recommends a correction factor of 65% to estimate the transverse Charpy USE of plates. No correction is required for weld metal, which has no orientation effect. As seen in Table 7-2, the corrected transverse USE for the plate materials and the USE for the weld are above the minimum of 50 ft-lb at EOL.

Table 7-1

EOL ADJUSTED REFERENCE TEMPERATURES FOR BELTLINE MATERIALS

Material	Heat	% Cu	% P	Initial RT_{NDT}	RT_{NDT} Shift[a]	EOL ART
Beltline Plates:						
1-18	C6439-2	0.09	0.012	40°F	55°F	95°F
1-19	B0402-1	0.13	0.012	40°F	86°F	126°F
1-20	B0436-2	0.15	0.008	10°F	86°F	96°F
1-21	B0673-1	0.15	0.011	10°F	98°F	108°F
Beltline Welds:						
Longitudinal	432Z0471/ B003A27A	0.03	0.017	-50°F	51°F	1°F
Girth	07L669/ K004A27A	0.03	0.014	-50°F	42°F	-8°F

[a] RT_{NDT} shift for plates includes correction factor of 1.31 from surveillance test results.

Table 7-2

END-OF-LIFE UPPER SHELF ENERGY

Material	USE (ft-lb) for f = 0 (longitudinal)	USE (ft-lb) for f = 4.9×10^{17} (longitudinal)	USE (ft-lb) for f = 3.6×10^{18} (long./trans.)
Plate 1-21	164 [a]	144	134/87
Plate 1-20	134 [a]	118	108/70
Weld	101 [a]	92	86

[a] These are measured values. All others are calculated using methods from References 5 and 7.

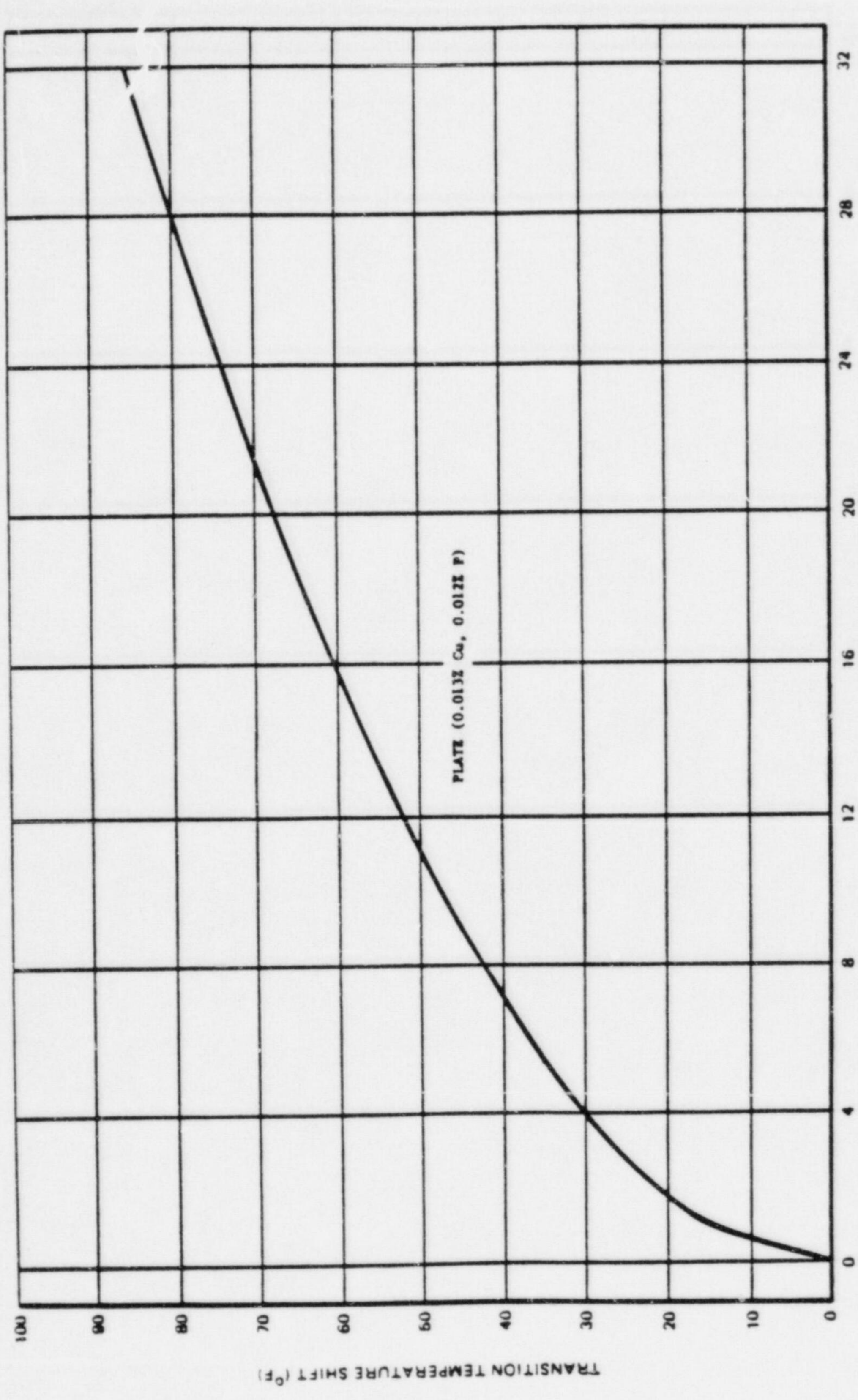


Figure 7-1. Shift in RT_{NDT} Versus Years of Operation, According to Regulatory Guide 1.99

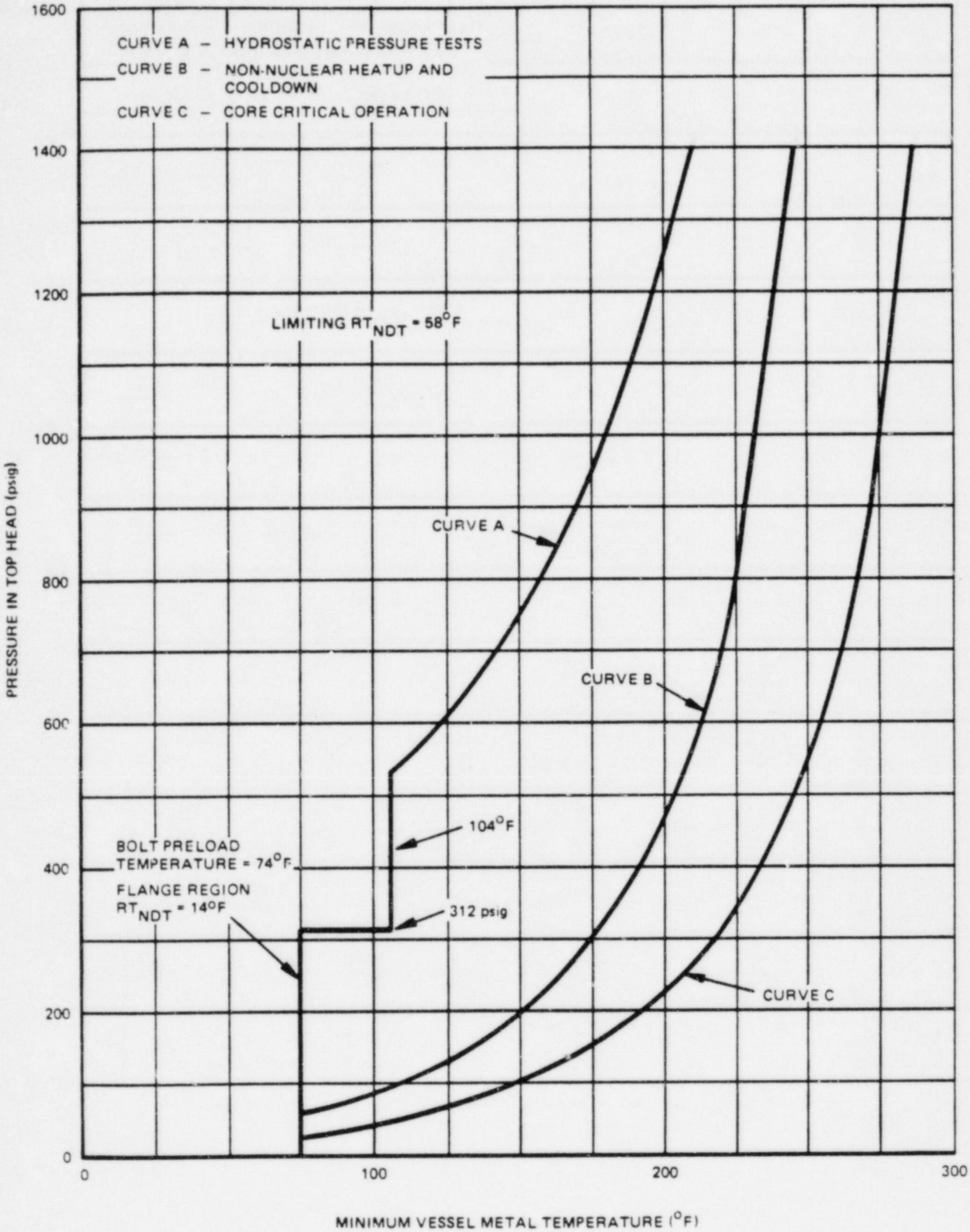


Figure 7-2. Pressure Versus Minimum Temperature, Valid to 12 EFPY

8. REFERENCES

1. "Fracture Toughness Requirements," Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, May 1983.

2. "Protection Against Non-Ductile Failure," Appendix G to Section III of the ASME Boiler & Pressure Vessel Code, Addenda to and including Winter 1985.

3. "Reactor Vessel Material Surveillance Program Requirements," Appendix H to Part 50 of Title 10 of the Code of Federal Regulations, May 1983.

4. "Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, E185-82, July 1982.

5. "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," USNRC Regulatory Guide 1.99, Revision 1, April 1977.

6. "DAEC RPV Fracture Toughness Analysis to 10CFR50 Appendix G, May 1983," General Electric Company, NEDC-30839, December 1984.

7. "Fracture Toughness Requirements," USNRC Branch Technical Position MTEB 5-2, Revision 1, July 1981.

8. "Drilled Hole Pattern," General Electric Drawing 117C4942, Revision 2, April 1971.

9. "Surveillance Test," Chicago Bridge & Iron Drawings T4 - T12.

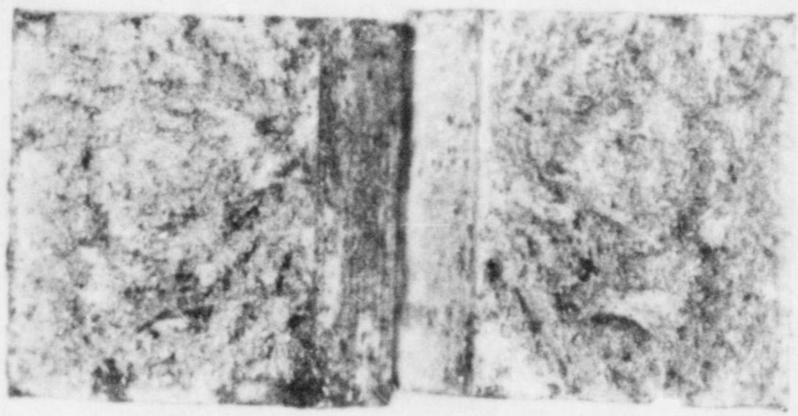
10. "Standard Methods for Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, E23-82, March 1982.

11. "Standard Methods of Tension Testing of Metallic Materials," Annual Book of ASTM Standards, E8-81.

APPENDIX A

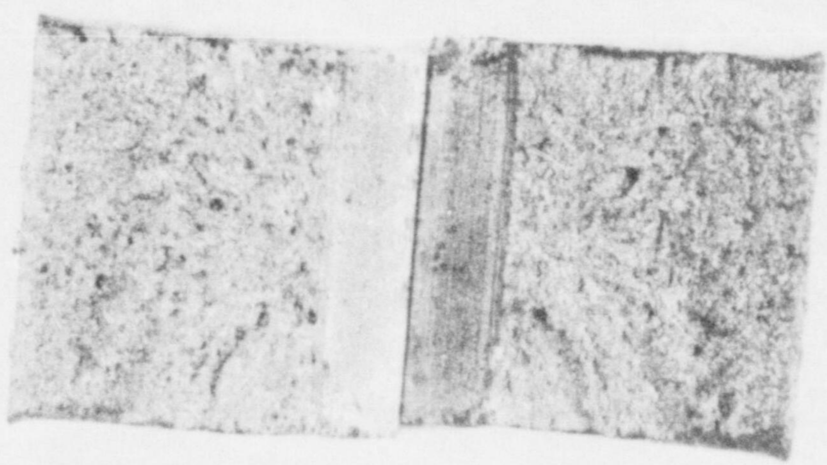
IRRADIATED CHARPY SPECIMEN FRACTURE SURFACE PHOTOGRAPHS

Photographs of the irradiated Charpy specimens' fracture surfaces were taken to facilitate the determination of percent shear, and to comply with the requirements of ASTM E185-82. The pages following show the fracture surface photographs along with a summary of the Charpy test results for each specimen. The pictures are arranged by increasing test temperature for each material, with the materials in the order of base, weld and HAZ.



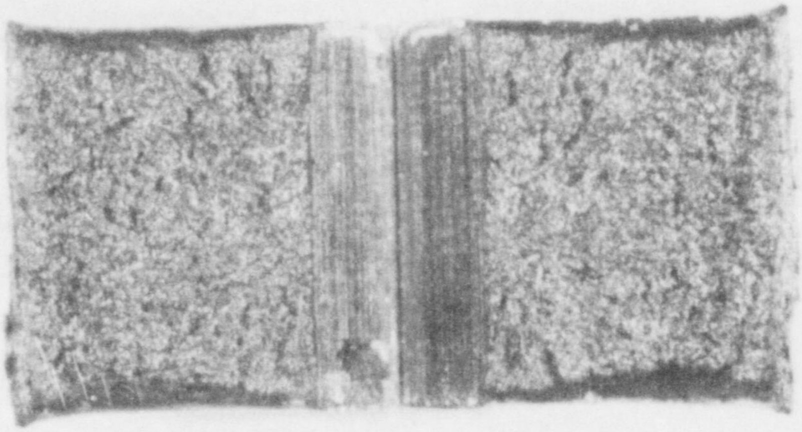
BASE: EPU
TEMP: $-60^{o}F$
ENERGY: 4.0 ft-lb

MLE: 4
% SHEAR: 0



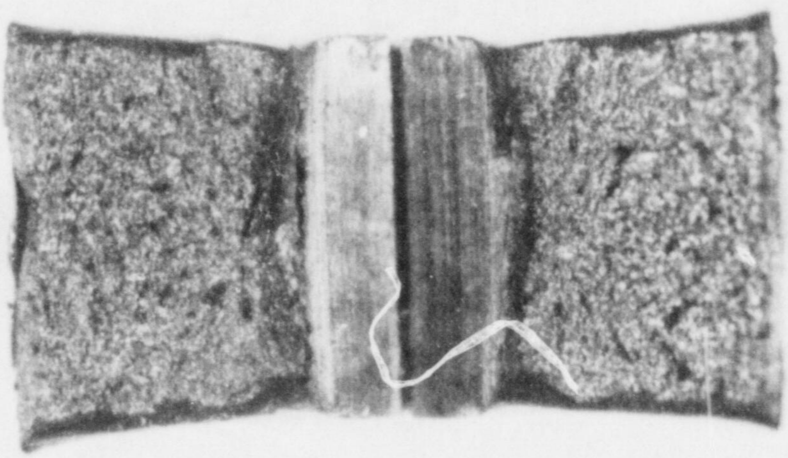
BASE: EBP
TEMP: $-20^{o}F$
ENERGY: 15.0 ft-lb

MLE: 19
% SHEAR: 0



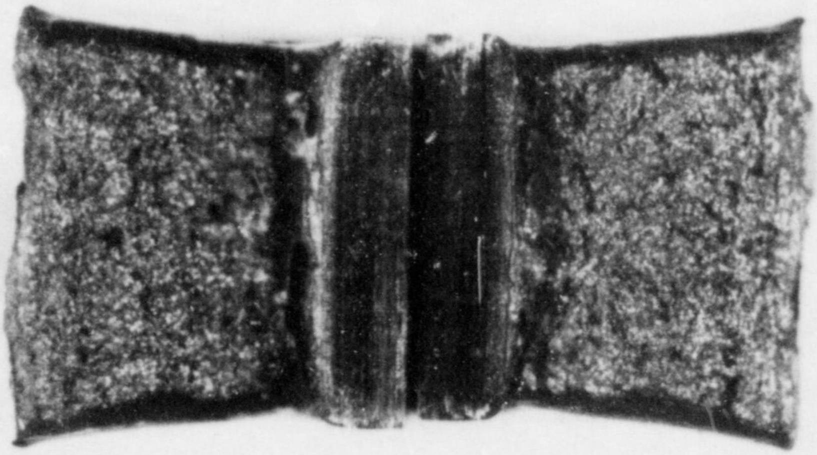
BASE: EBT
TEMP: $10^{o}F$
ENERGY: 15.5 ft-lb

MLE: 26
% SHEAR: 40

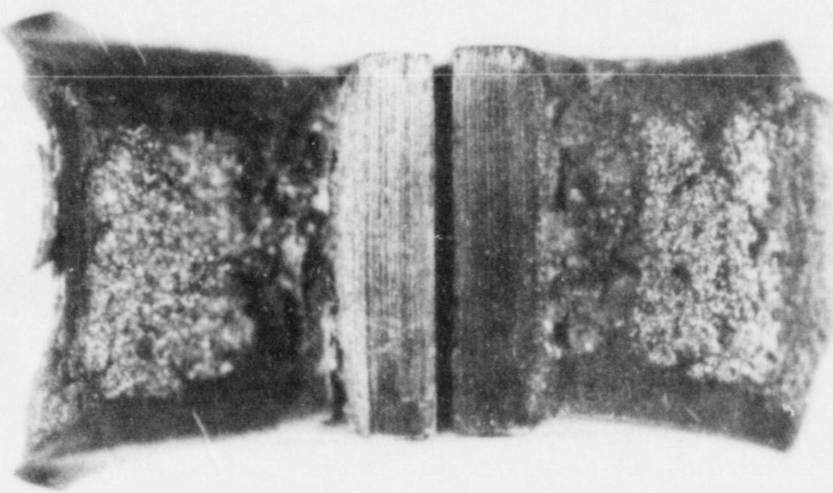


BASE: ECI
TEMP: $20^{o}F$
ENERGY: 49.5 ft-lb

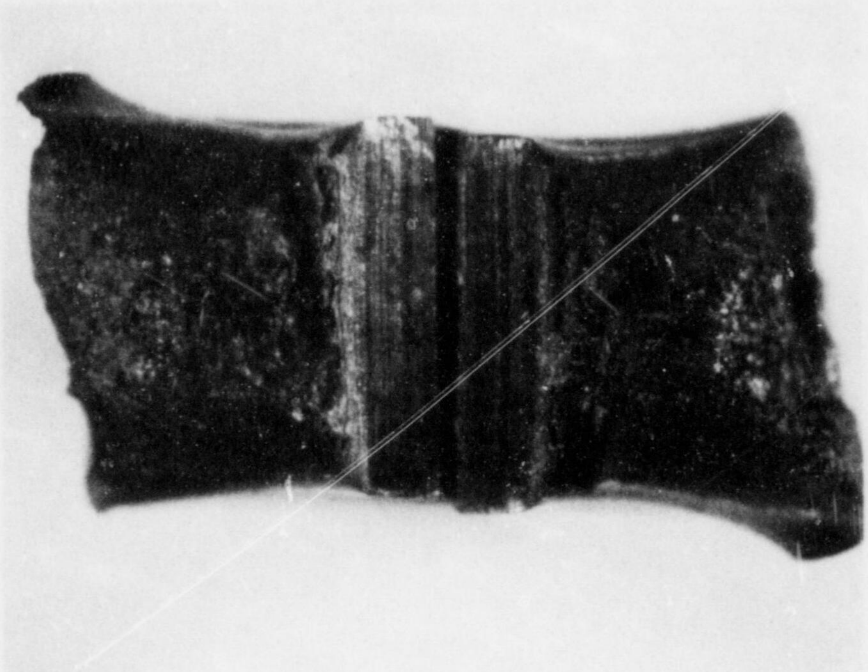
MLE: 45
% SHEAR: 40



BASE: EBK
TEMP: 40°F
ENERGY: 60.0 ft-lb
MLE: 49
% SHEAR: 40

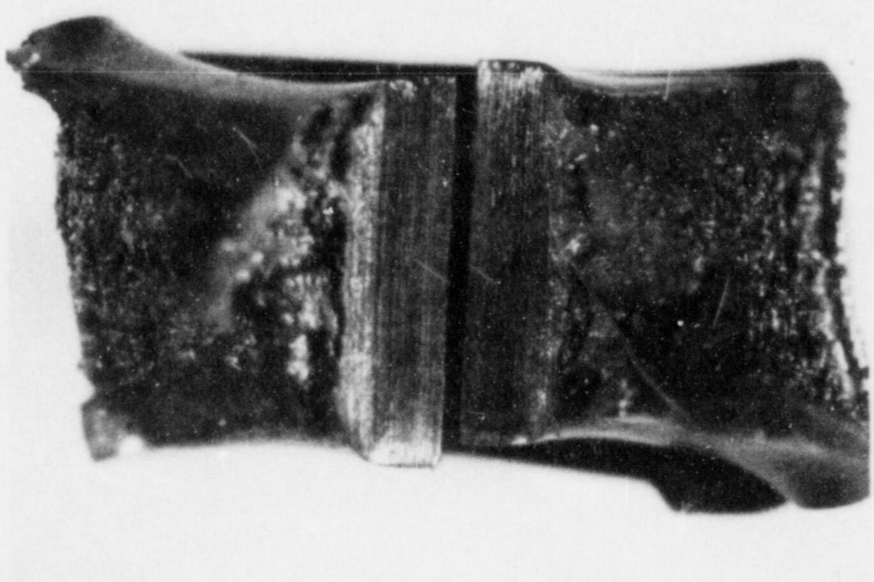


BASE: EBJ
TEMP: 120°F
ENERGY: 101.5 ft-lb
MLE: 70
% SHEAR: 70



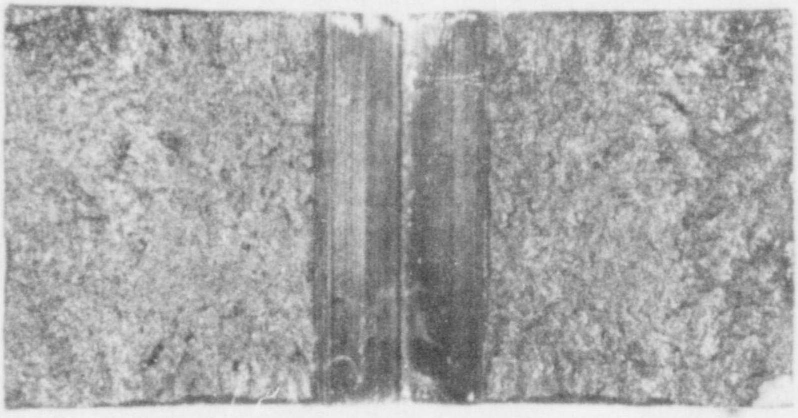
BASE: EBL
TEMP: 200°F
ENERGY: 144.5 ft-lb

MLE: 95
% SHEAR 90

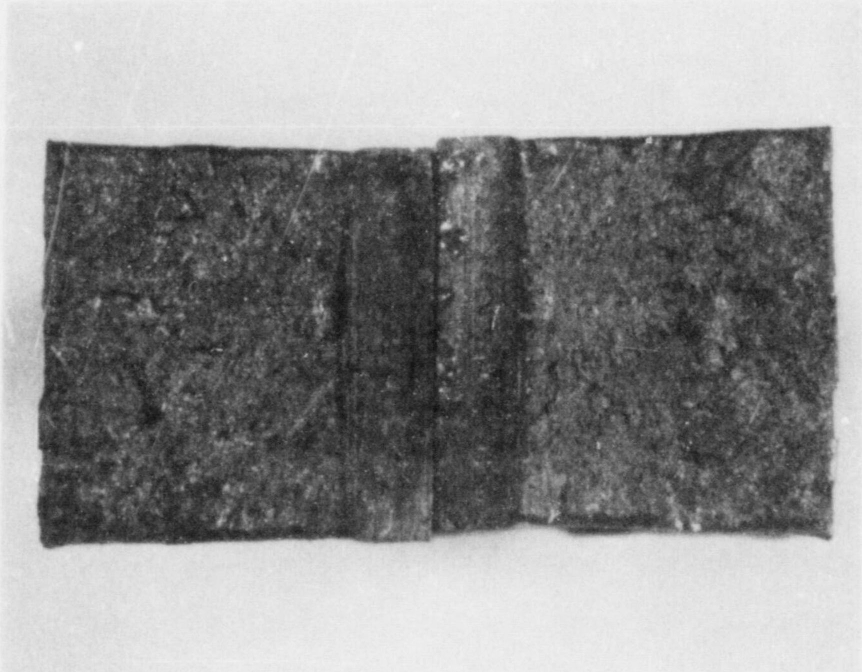


BASE: EBY
TEMP: 400°F
ENERGY: 160 ft-lb

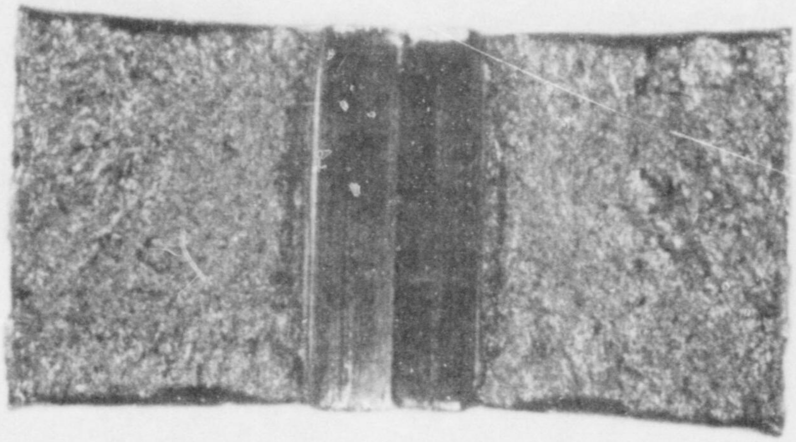
MLE: 98
% SHEAR 90



WELD: EJ1
TEMP: $-100^{o}F$
ENERGY: 6.5 ft-lb
MLE: 11
% SHEAR: 0

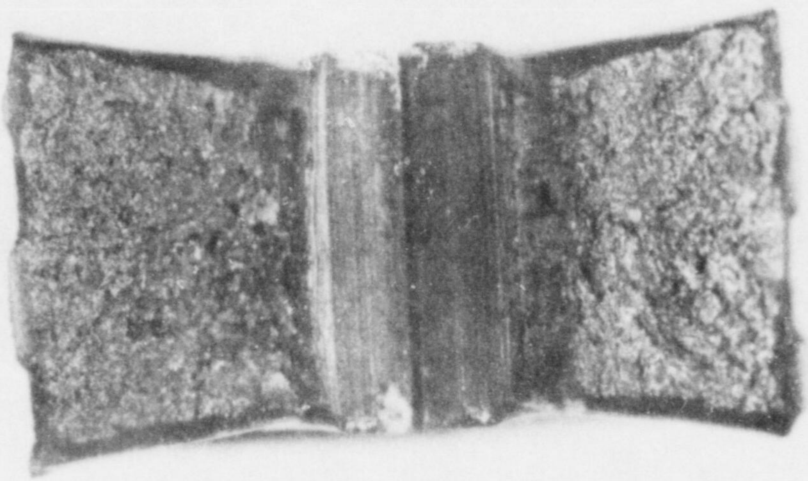


WELD: EJ2
TEMP: $-60^{o}F$
ENERGY: 15.3 ft-lb
MLE: 24
% SHEAR: 10



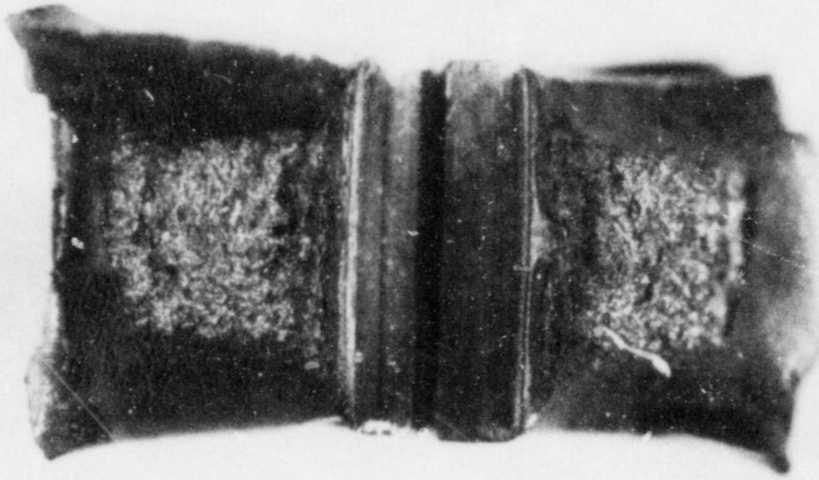
WELD: EJ4
TEMP: $-40^{O}F$
ENERGY: 22.0 ft-lb

MLE: 26
% SHEAR: 10

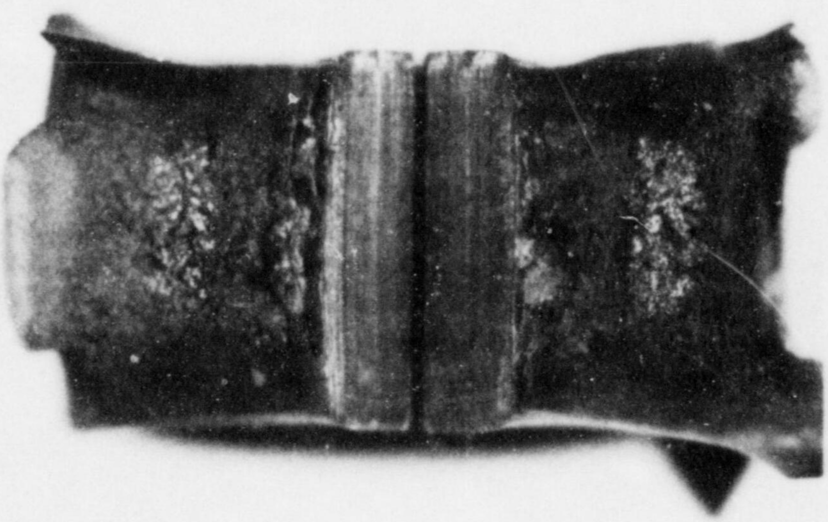


WELD: EE5
TEMP: $-20^{O}F$
ENERGY: 63.5 ft-lb

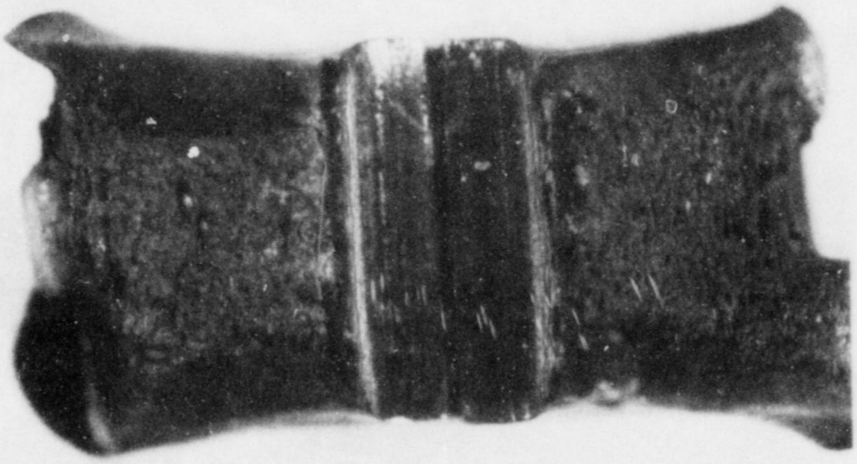
MLE: 54
% SHEAR: 40



WELD: EE2
TEMP: 40^{O}F
ENERGY: 75.2 ft-lb
MLE: 69
% SHEAR: 60



WELD: EE3
TEMP: 120^{O}F
ENERGY: 97.2 ft-lb
MLE: 86
% SHEAR: 80

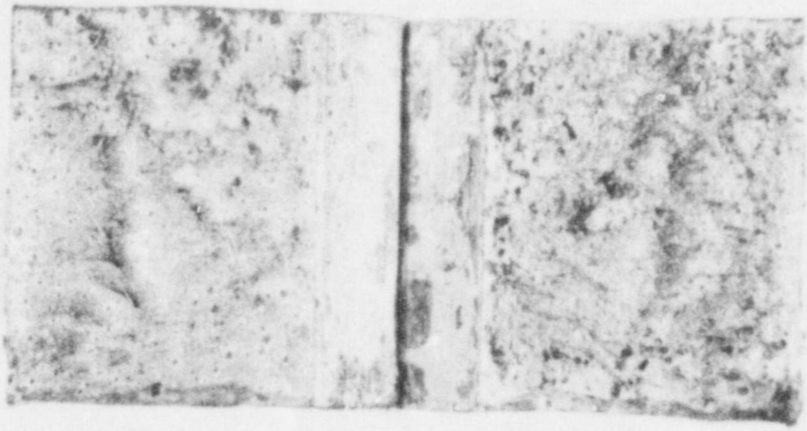


WELD: EE4
TEMP: 200°F
ENERGY: 109.0 ft-lb

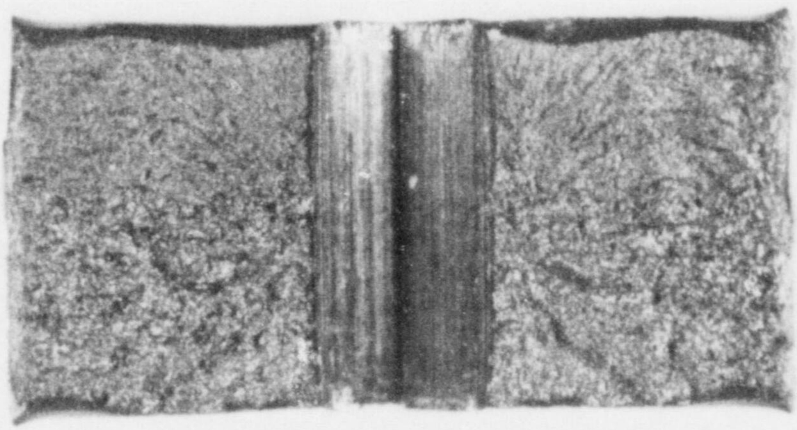
MLE: 90
% SHEAR: 70

WELD: EJ3
TEMP: 400°F
ENERGY: 101.0 ft-lb

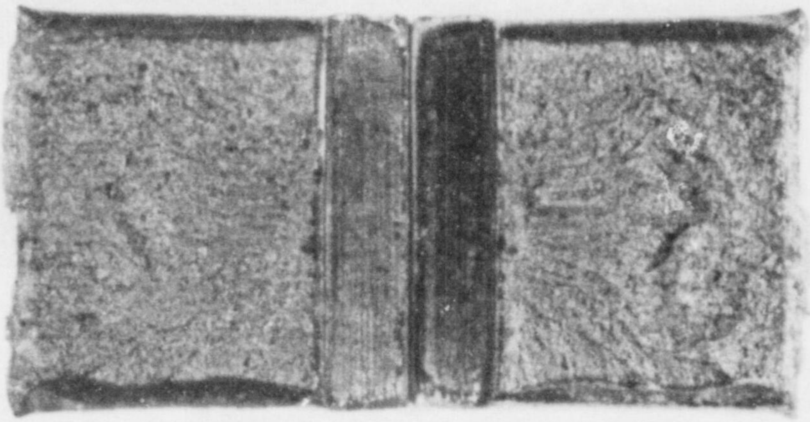
MLE: 91
% SHEAR: 100



HAZ: ELY
TEMP: $-60^{o}F$
ENERGY: 8.0 ft-lb
MLE: 15
% SHEAR: 0

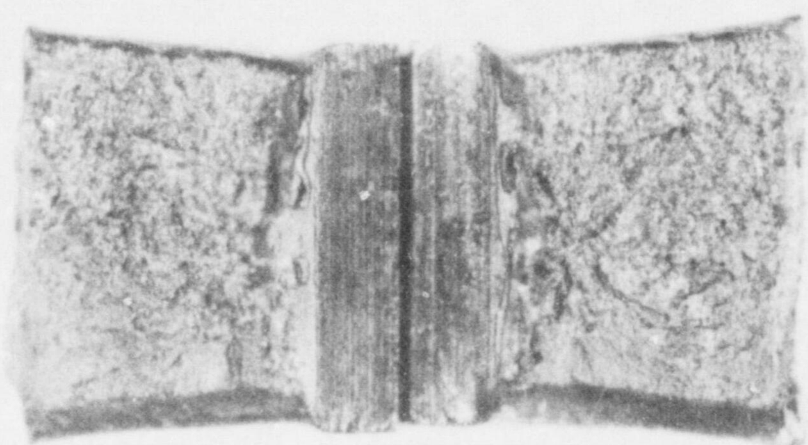


HAZ: ELP
TEMP: $-20^{o}F$
ENERGY: 13.5 ft-lb
MLE: 18
% SHEAR: 20



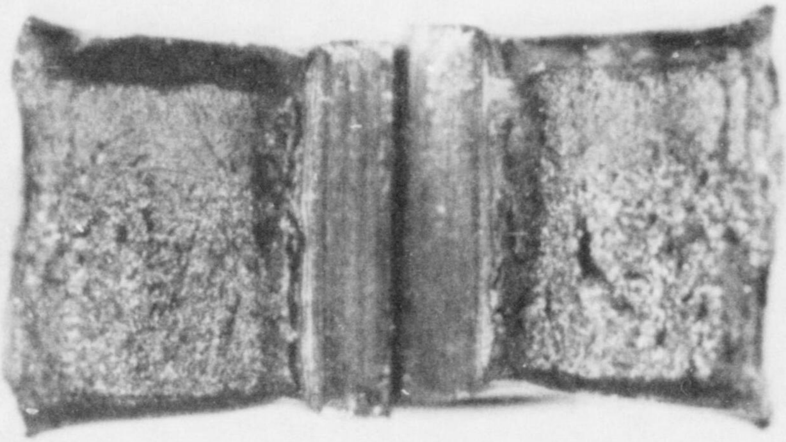
HAZ: EMY
TEMP: $0^{o}F$
ENERGY: 15.0 ft-lb

MLE: 16
% SHEAR: 30



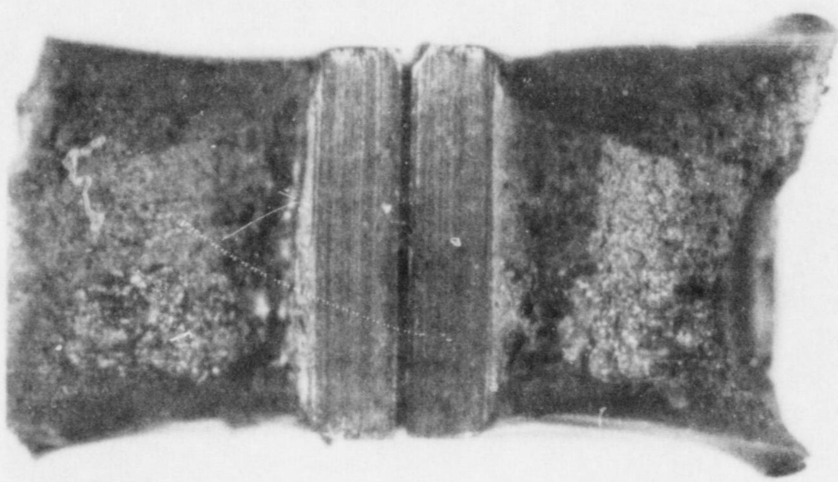
HAZ: ELT
TEMP: $10^{o}F$
ENERGY: 51.7 ft-lb

MLE: 43
% SHEAR: 40



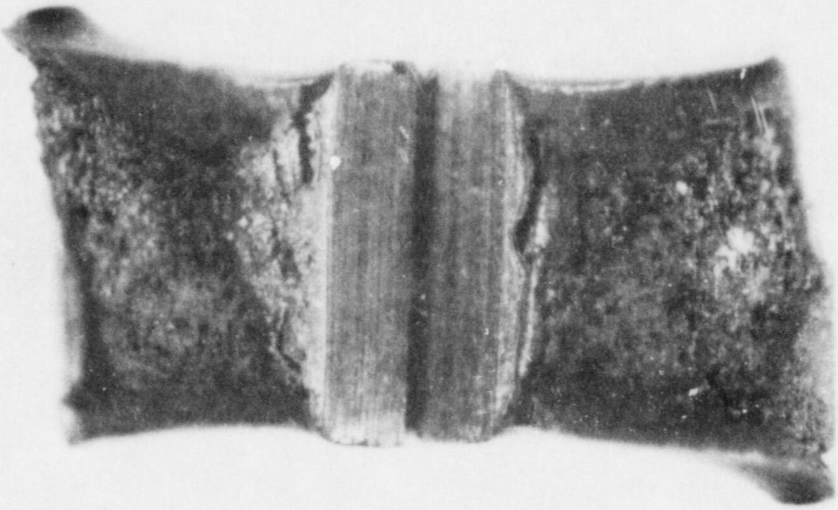
HAZ: ELK
TEMP: $40^{o}F$
ENERGY: 65.0 ft-lb

MLE: 50
% SHEAR: 50

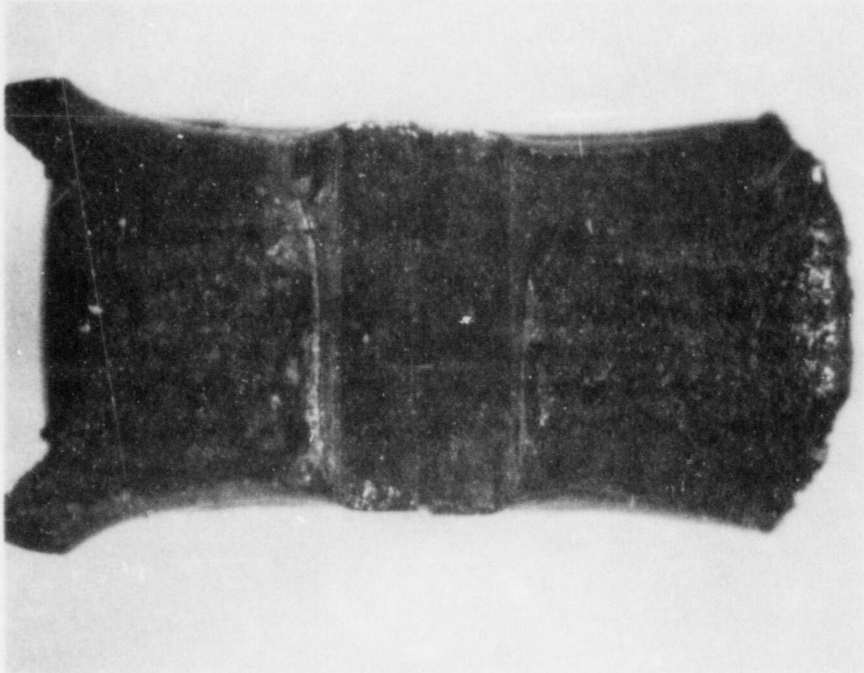


HAZ: ELL
TEMP: $120^{o}F$
ENERGY: 90.1 ft-lb

MLE: 66
% SHEAR: 70



HAZ: ELM
TEMP: 200°F
ENERGY: 126.5 ft-lb
MLE: 84
% SHEAR: 90

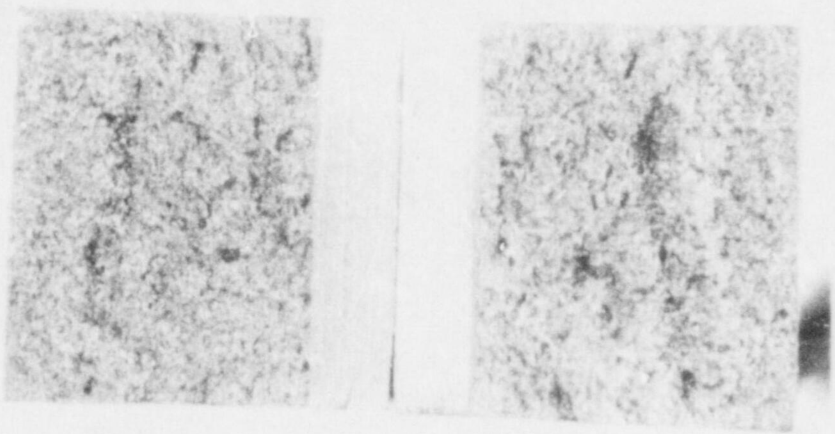


HAZ: EMT
TEMP: 400°F
ENERGY: 124.5 ft-lb
MLE: 75
% SHEAR: 100

APPENDIX B

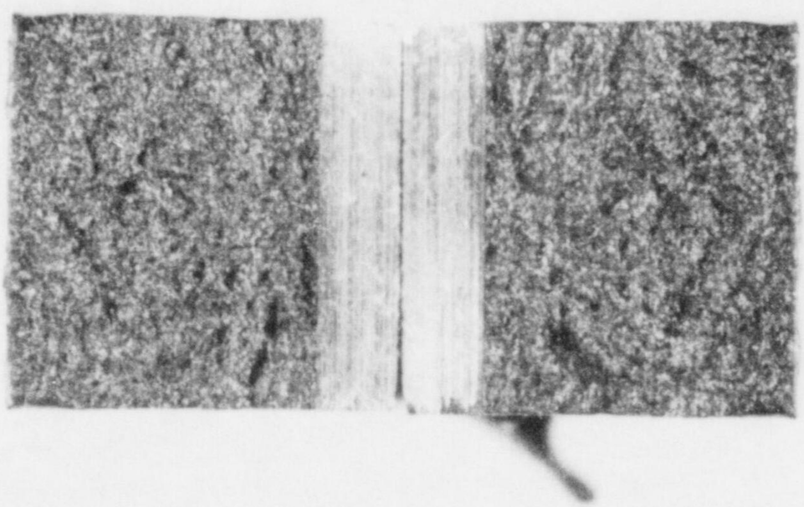
UNIRRADIATED CHARPY SPECIMEN FRACTURE SURFACE PHOTOGRAPHS

Photographs of the unirradiated Charpy specimens' fracture surfaces were taken to facilitate the determination of percent shear, and to comply with the requirements of ASTM E185-82. The pages following show the fracture surface photographs along with a summary of the Charpy test results for each specimen. The pictures are arranged by increasing test temperature for each material, with the materials in the order of base, weld and HAZ.



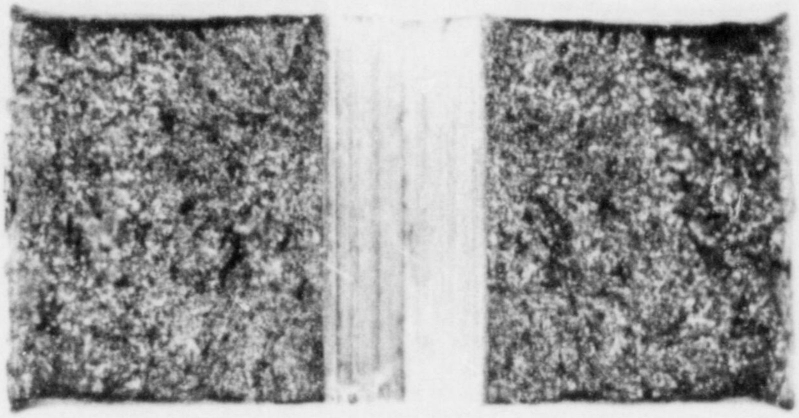
BASE: ED4
TEMP: $-100^{o}F$
ENERGY: 6.5 ft-lb

MLE: 7
% SHEAR: 0

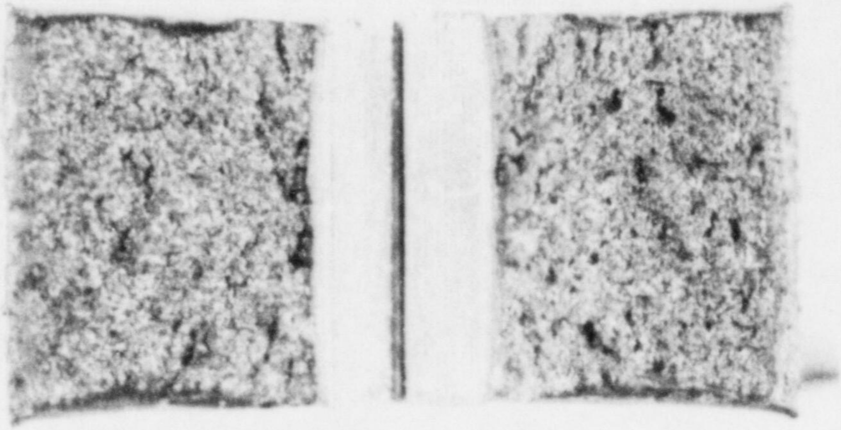


BASE: ED1
TEMP: $-80^{o}F$
ENERGY: 4.2 ft-lb

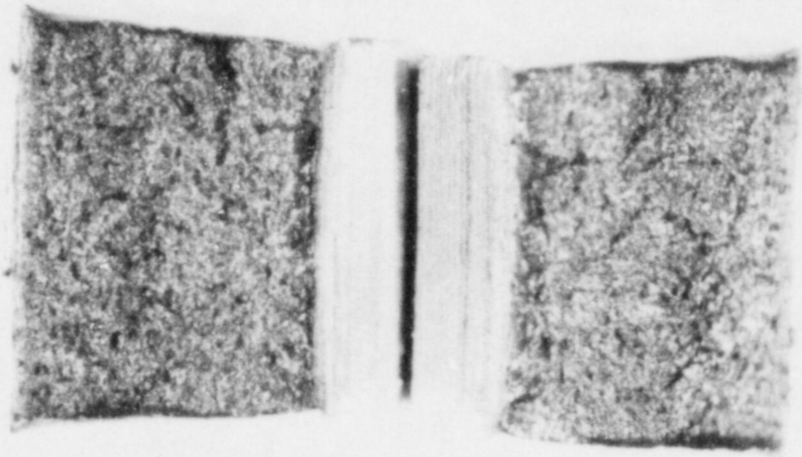
MLE: 5
% SHEAR: 0



BASE: ECT
TEMP: −40°F
ENERGY: 9.0 ft-lb
MLE: 30
% SHEAR: 10

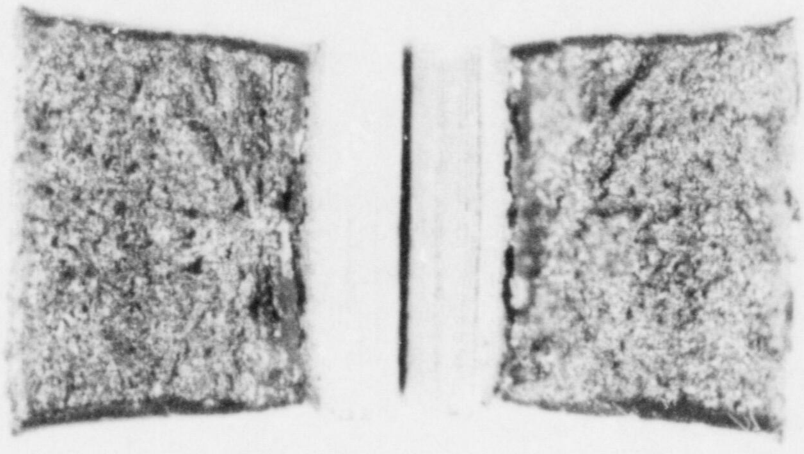


BASE: EDK
TEMP: −40°F
ENERGY: 24.0 ft-lb
MLE: 30
% SHEAR: 5



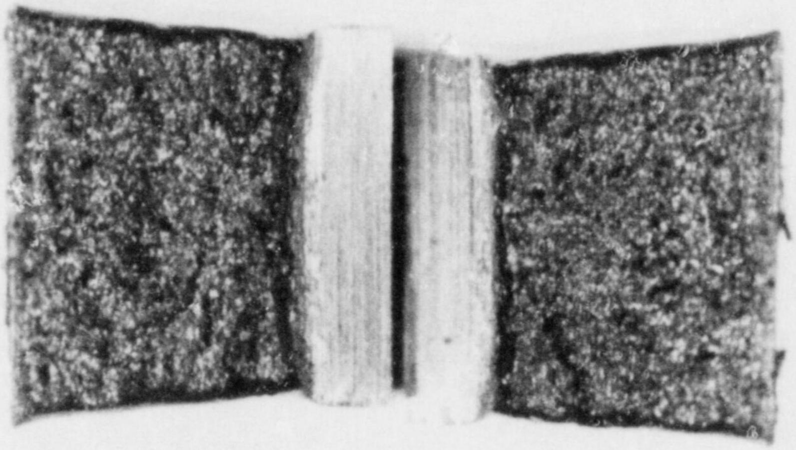
BASE: ED7
TEMP: −30°F
ENERGY: 38.0 ft-lb

MLE: 30
% SHEAR: 5



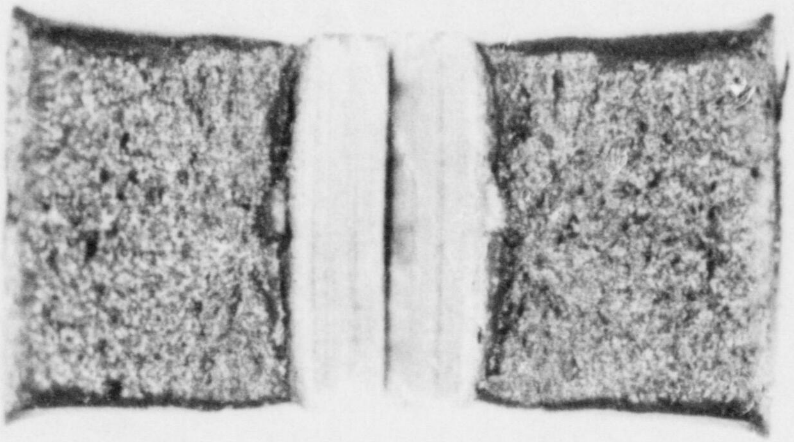
BASE: EDT
TEMP: −30°F
ENERGY: 49.0 ft-lb

MLE: 30
% SHEAR: 5



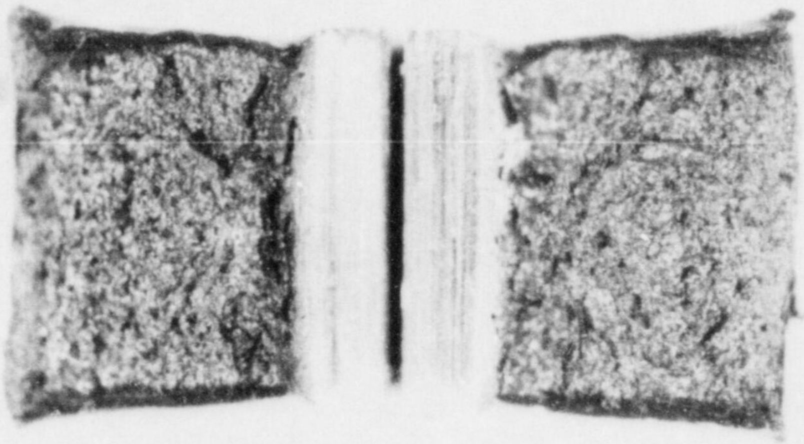
BASE: ED6
TEMP: −20°F
ENERGY: 47.0 ft-lb

MLE: 43
% SHEAR: 20



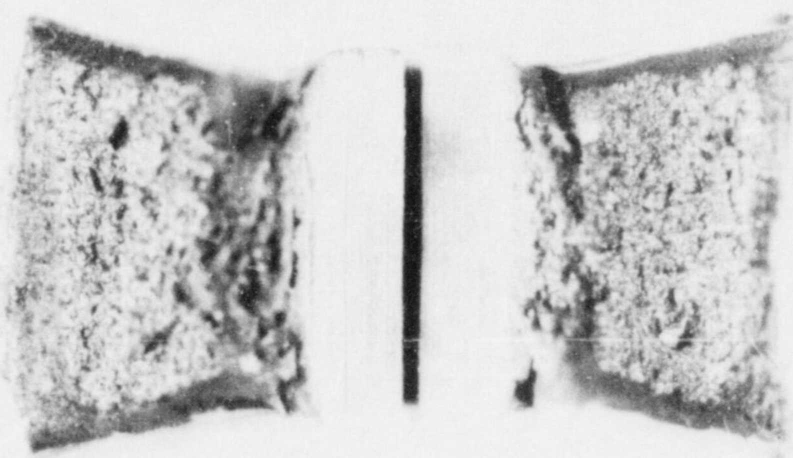
BASE: ED5
TEMP: −10°F
ENERGY: 43.0 ft-lb

MLE: 43
% SHEAR: 20



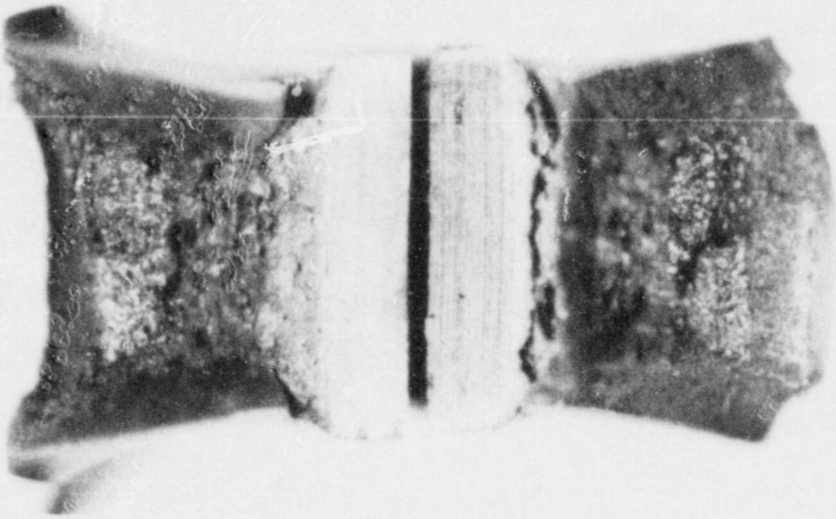
BASE: ECP
TEMP: 0°F
ENERGY: 56.5 ft-lb

MLE: 49
% SHEAR: 25



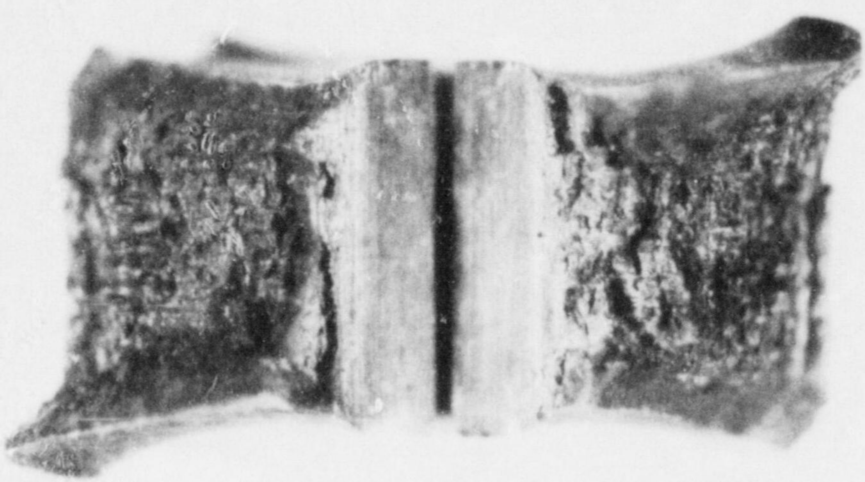
BASE: ECM
TEMP: 40°F
ENERGY: 98.5 ft-lb

MLE: 73
% SHEAR: 40



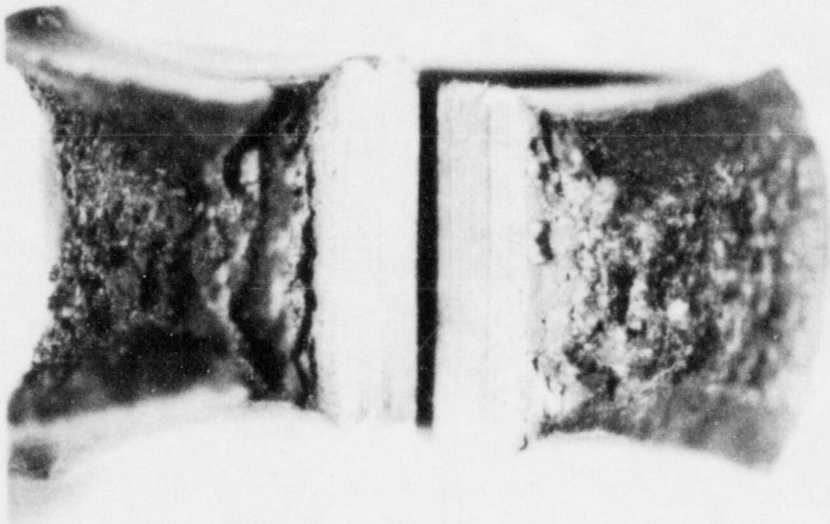
BASE: ECL
TEMP: 120°F
ENERGY: 134.5 ft-lb

MLE: 73
% SHEAR: 80

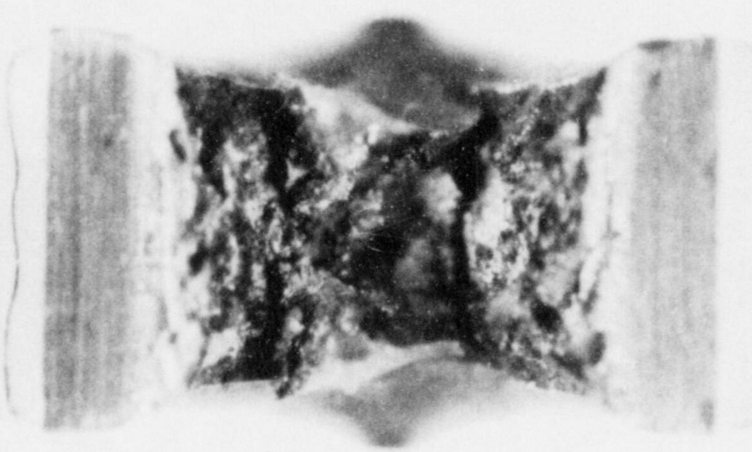


BASE: ECK
TEMP: 200°F
ENERGY: 158.5 ft-lb

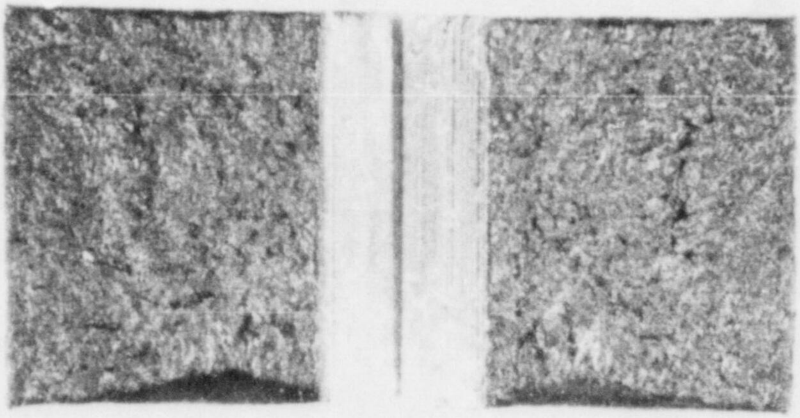
MLE: 93
% SHEAR: 85



BASE: EDY
TEMP: 300°F
ENERGY: 163.5 ft-lb
MLE: 81
% SHEAR: 90

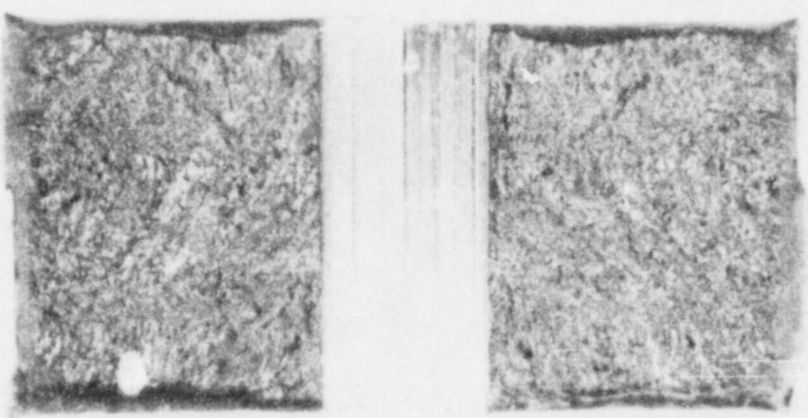


BASE: EDA
TEMP: 400°F
ENERGY: NO BREAK
MLE: N/A
% SHEAR: N/A



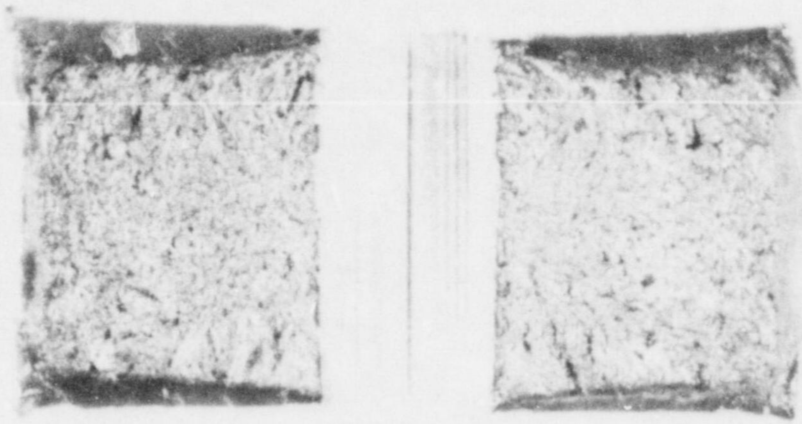
WELD: EET
TEMP: −100°F
ENERGY: 6.5 ft-lb

MLE: 11
% SHEAR: 10



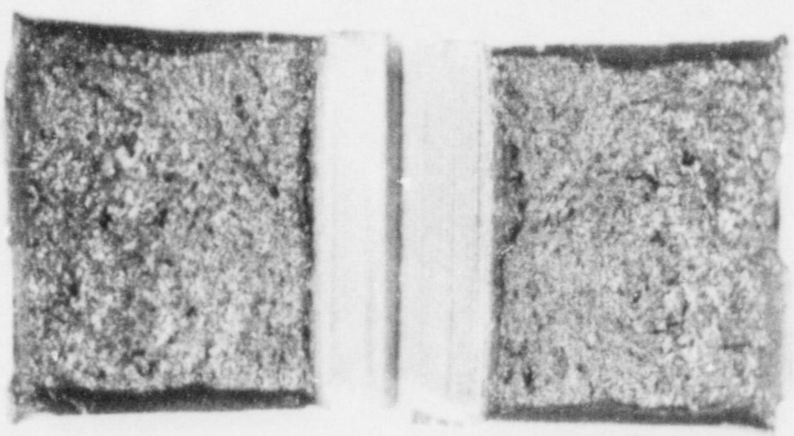
WELD: EEK
TEMP: −80°F
ENERGY: 5.5 ft-lb

MLE: 10
% SHEAR: 15



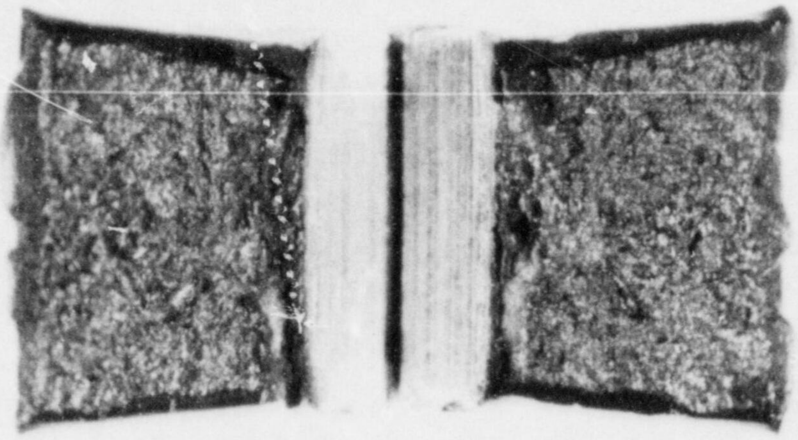
WELD: EK3
TEMP: −60°F
ENERGY: 16.8 ft-lb

MLE: 19
% SHEAR: 40



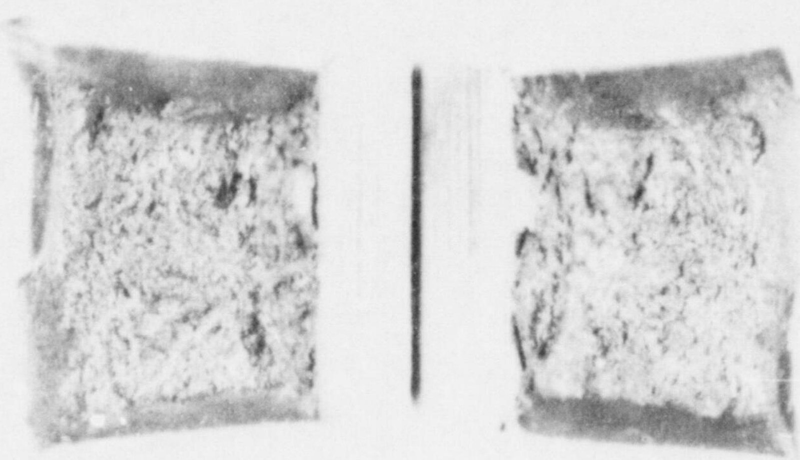
WELD: EK4
TEMP: −50°F
ENERGY: 21.5 ft-lb

MLE: 22
% SHEAR: 20



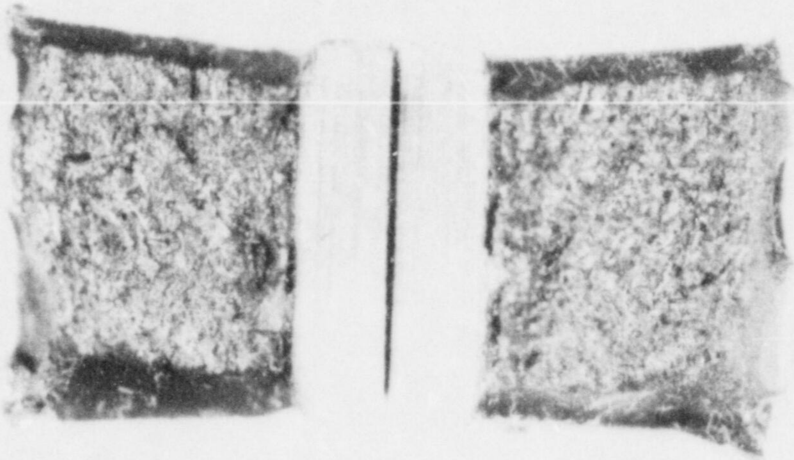
WELD: EEE
TEMP: −40°F
ENERGY: 49.0 ft-lb

MLE: 45
% SHEAR: 20

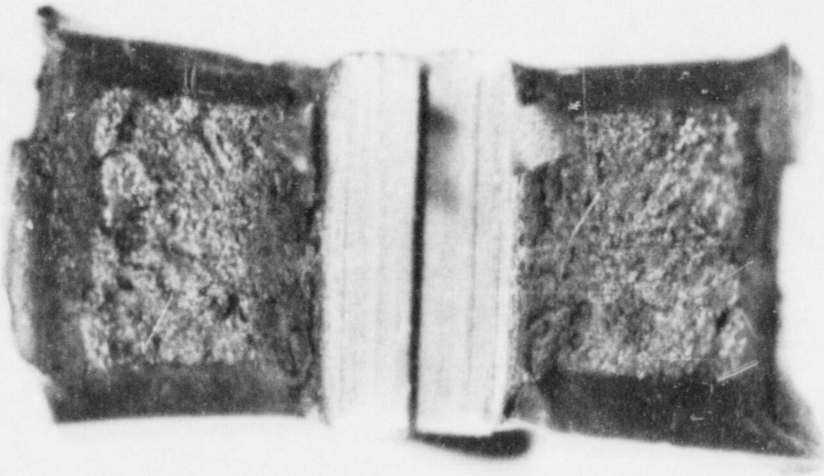


WELD: EK2
TEMP: −20°F
ENERGY: 58.0 ft-lb

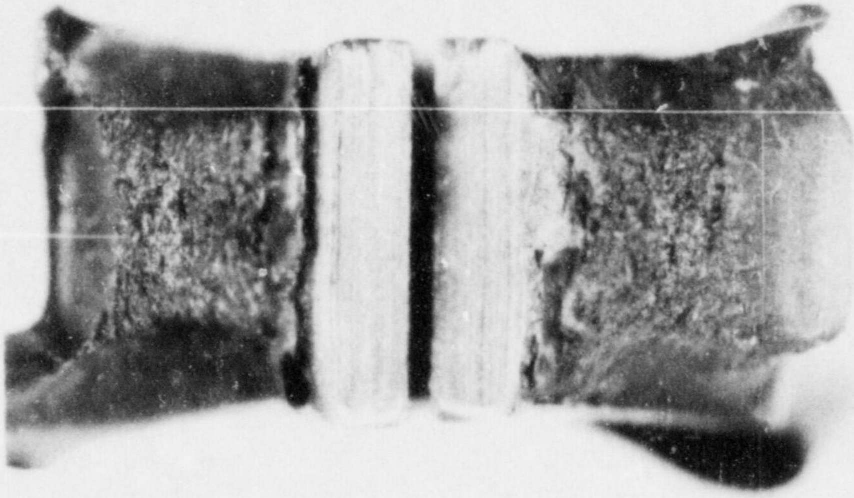
MLE: 57
% SHEAR: 50



WELD: EEL
TEMP: 0°F
ENERGY: 52.5 ft-lb
MLE: 50
% SHEAR: 30

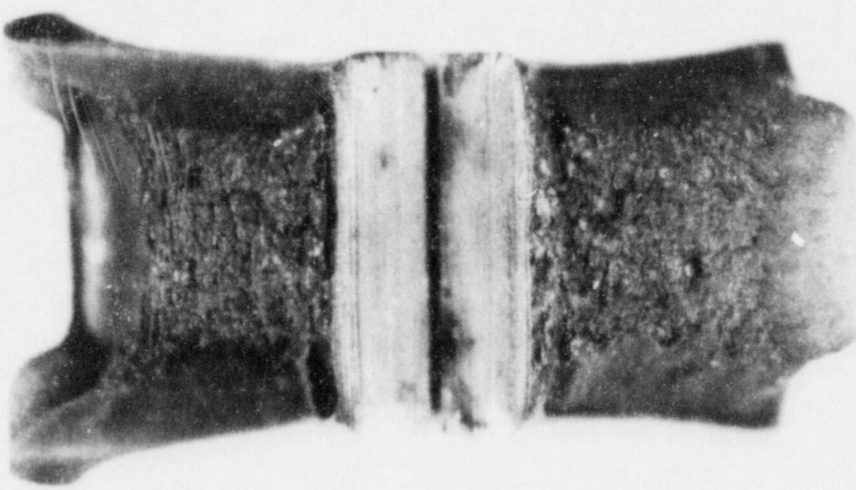


WELD: EEY
TEMP: 40°F
ENERGY: 65.0 ft-lb
MLE: 61
% SHEAR: 50



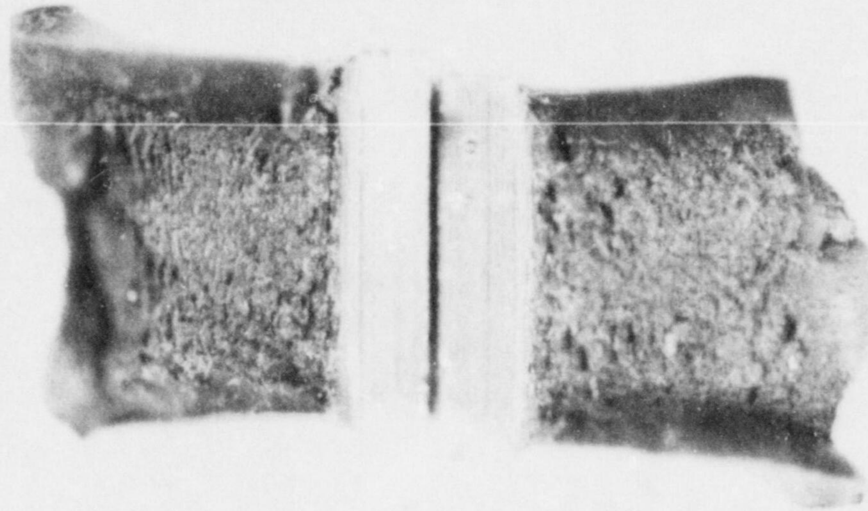
WELD: EED
TEMP: 120°F
ENERGY: 102.0 ft-lb

MLE: 88
% SHEAR: 80



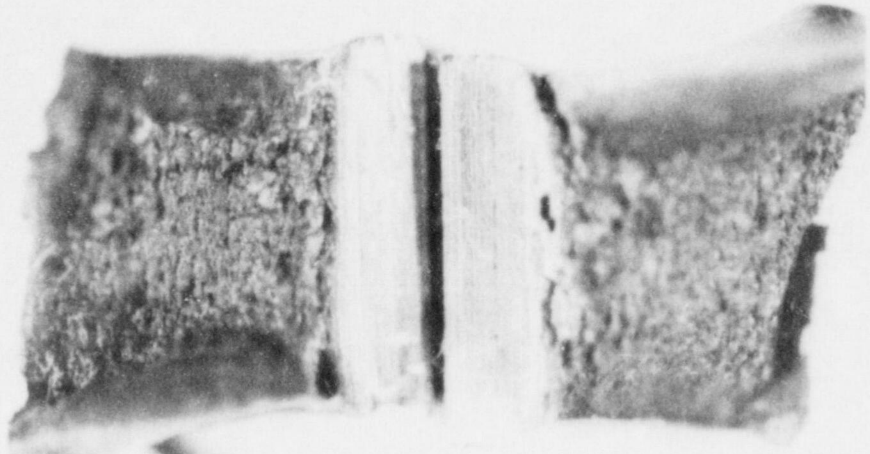
WELD: EEM
TEMP: 200°F
ENERGY: 94.0 ft-lb

MLE: 65
% SHEAR: 70



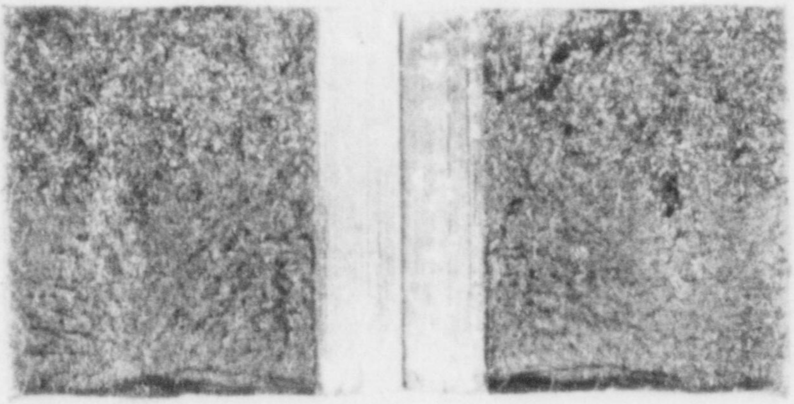
WELD: EK5
TEMP: 300°F
ENERGY: 88.2 ft-lb

MLE: 93
% SHEAR: 60



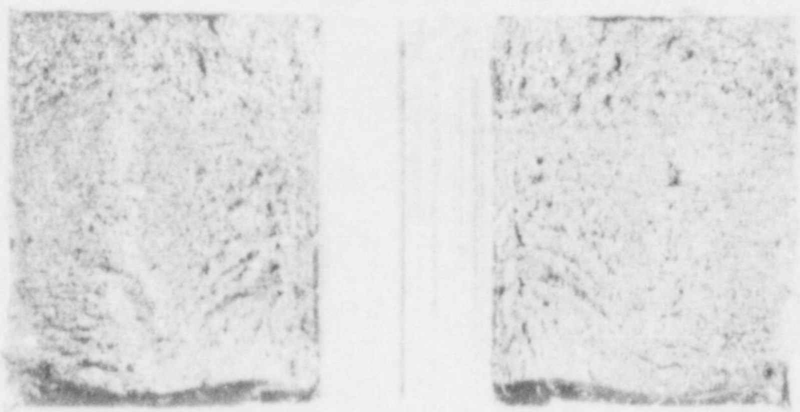
WELD: EK6
TEMP: 400°F
ENERGY: 113.7 ft-lb

MLE: 96
% SHEAR: 70



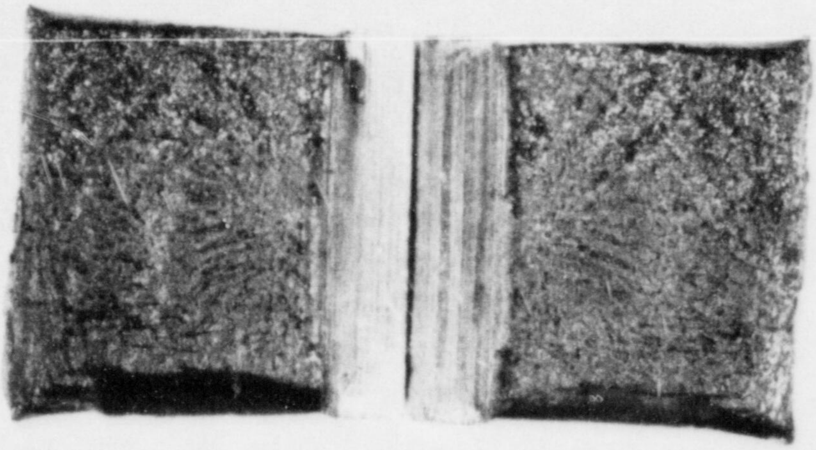
HAZ: EM6
TEMP: $-100^{o}F$
ENERGY: 5.8 ft-lb

MLE: 12
% SHEAR: 5



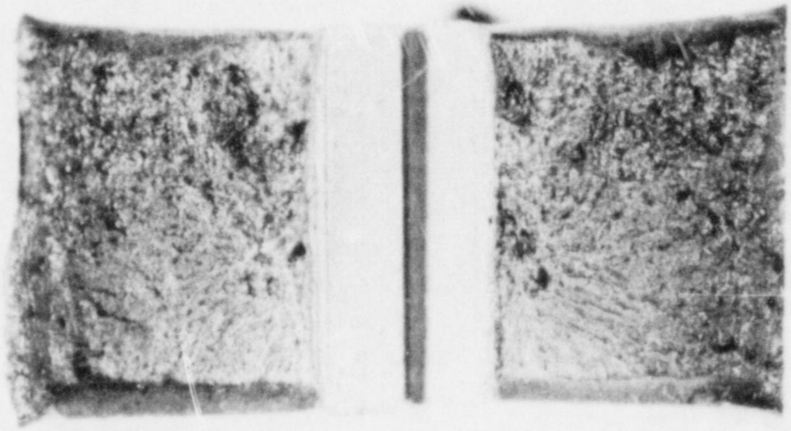
HAZ: EM5
TEMP: $-80^{o}F$
ENERGY: 8.0 ft-lb

MLE: 17
% SHEAR: 10



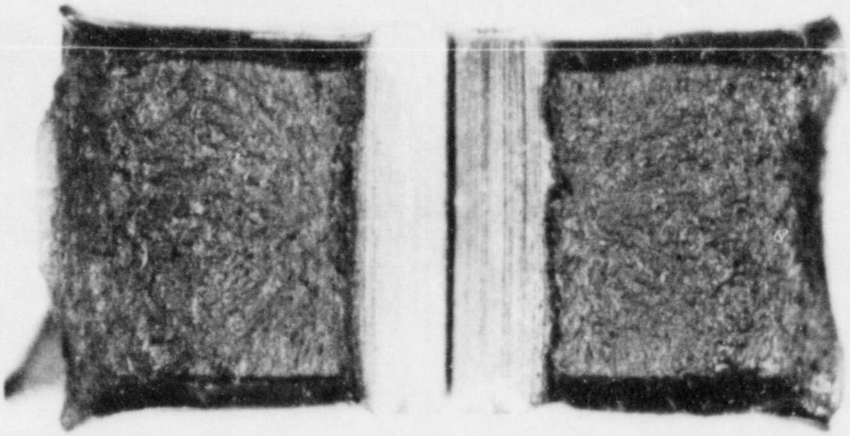
HAZ: EM4
TEMP: $-40^{o}F$
ENERGY: 23.8 ft-lb

MLE: 25
% SHEAR: 20

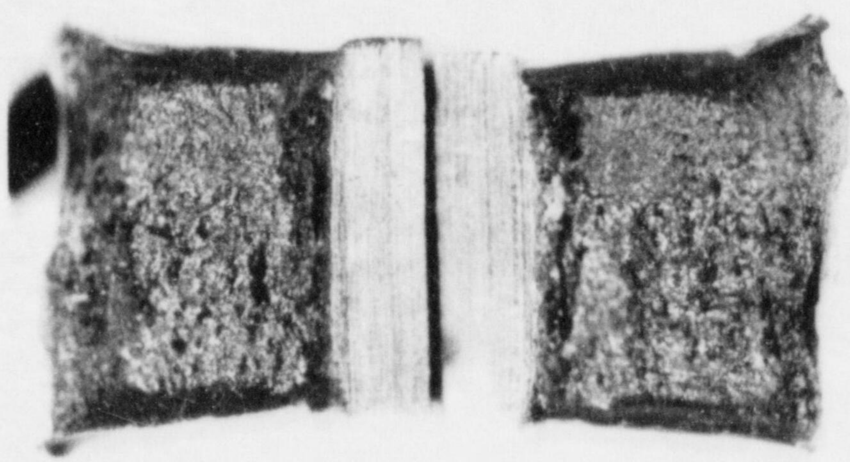


HAZ: EM3
TEMP: $0^{o}F$
ENERGY: 28.0 ft-lb

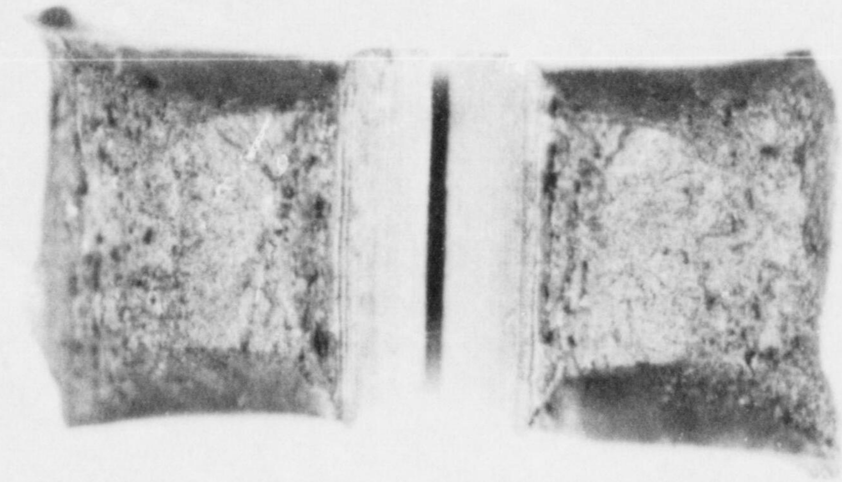
MLE: 28
% SHEAR: 20



HAZ: EP2
TEMP: 20^oF
ENERGY: 37.0 ft-lb
MLE: 36
% SHEAR: 30

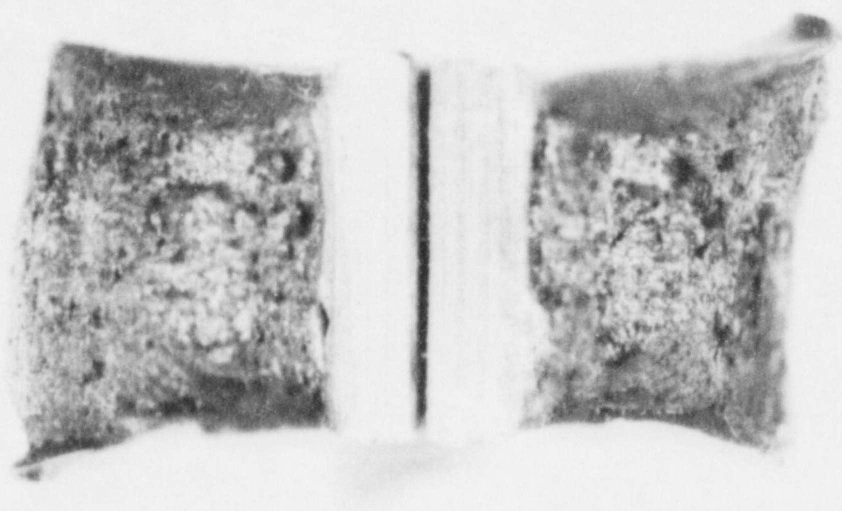


HAZ: EMA
TEMP: 40^oF
ENERGY: 65.0 ft-lb
MLE: 62
% SHEAR: 40



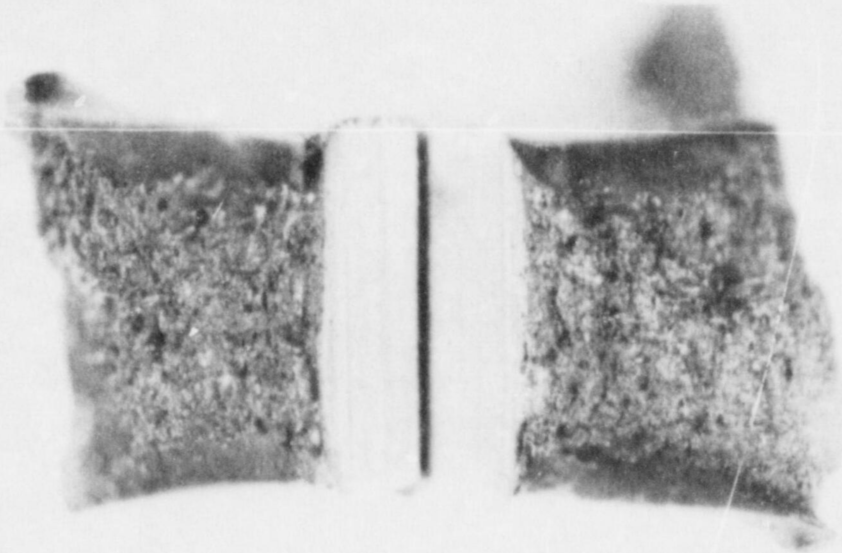
HAZ: EP3
TEMP: 80°F
ENERGY: 71.5 ft-lb

MLE: 67
% SHEAR: 60



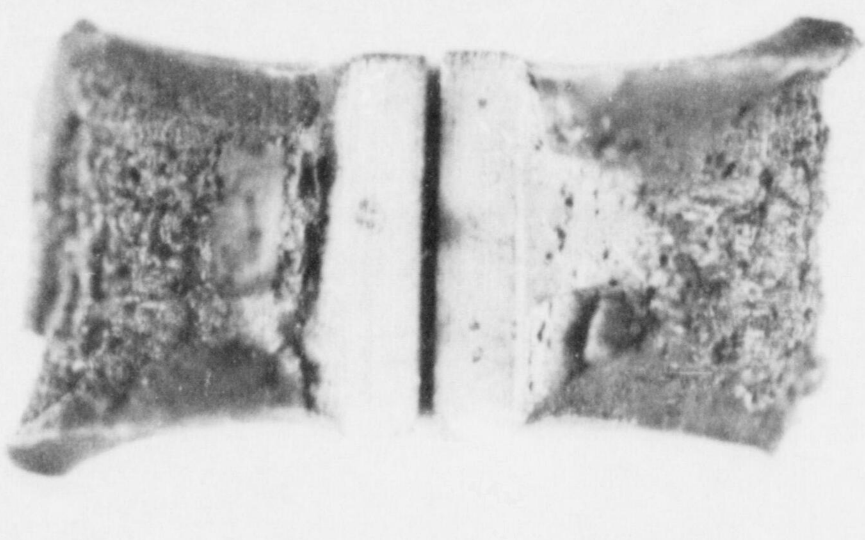
HAZ: EMI
TEMP: 120°F
ENERGY: 90.0 ft-lb

MLE: 77
% SHEAR: 80



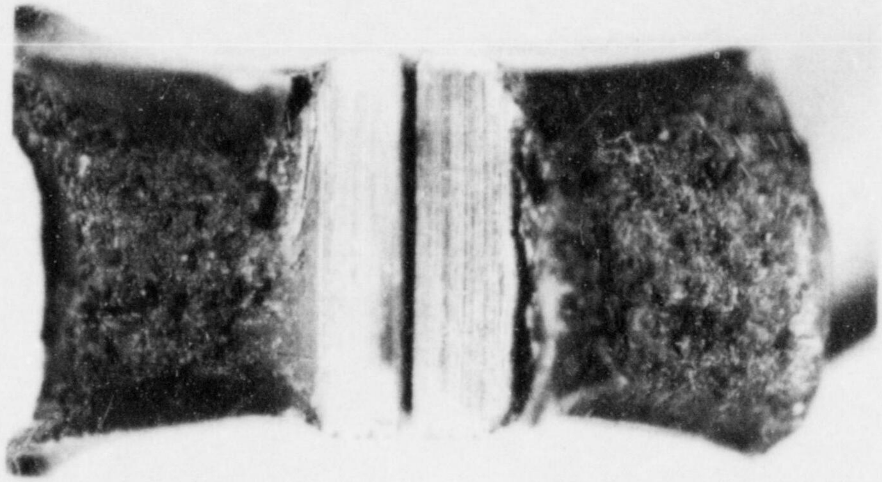
HAZ: EP4
TEMP: 160°F
ENERGY: 109.0 ft-lb

MLE: 90
% SHEAR: 60

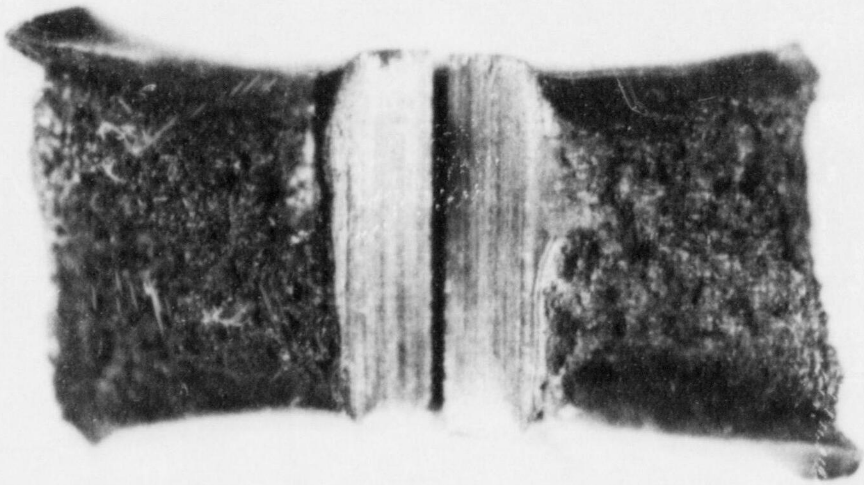


HAZ: EM2
TEMP: 200°F
ENERGY: 128.0 ft-lb

MLE: 89
% SHEAR: 85



HAZ: EP5
TEMP: 300^{o}F
ENERGY: 113.0 ft-lb
MLE: 89
% SHEAR: 60



HAZ: EP7
TEMP: 400^{o}F
ENERGY: 107.0 ft-lb
MLE: 94
% SHEAR: 85

APPENDIX C

REVISIONS TO THE TECHNICAL SPECIFICATIONS

Appendix C contains suggested revisions to sections 3.6.A and 4.6.A of the DAEC Technical Specifications. Suggested revisions to these sections were recently made in GE report NEDC-30839. The changes listed here are based on sections 3.6.A and 4.6.A as they were revised in Appendix A of NEDC-30839. The changes do not affect the operating limits. They simply report that the first capsule has been tested, and any significant results related to that testing.

C.1 SPECIFICATION 4.6.A

The only change needed in the specifications is in the surveillance requirements section 4.6.A.2, the last paragraph. The paragraph should be changed to read as follows:

> Samples from surveillance capsule 1 at 288° were withdrawn at 6 effective full power years and tested in accordance with 10CFR50, Appendix H. Neutron flux wires installed in the surveillance capsule were tested to experimentally determine the flux and fluence at one-fourth of the beltline shell thickness, used to determine the NDTT shift. Results of the surveillance capsule testing, including fluence, material chemistry results and actual material irradiation shift, are reflected in the curves in Figure 3.6.1. The next surveillance capsule shall be withdrawn at 15 effective full power years.

C.2 BASES FOR 3.6.A AND 4.6.A

The bases for 3.6.A and 4.6.A are essentially unchanged. The tenth paragraph, which discusses the surveillance program should be revised to read as follows:

> Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The first capsule was removed after fuel cycle 7, at 6 effective full power years. The neutron flux wires tested were used to determine the end-of-life fluence at the 1/4 T depth in the vessel wall of 3.6×10^{18} n/cm^2. Irradiated and unirradiated Charpy specimens were tested, showing that the plate shift in NDTT was 31% higher than predicted by Regulatory Guide 1.99, Revision 1. The weld showed no measurable shift. The results of the surveillance testing, including a correction factor of 1.31 for the plate materials, have been incorporated into the curves on Figure 3.6.1.

APPENDIX D

REVISIONS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

Recommendations are made for changes to the Updated Final Safety Analysis Report (UFSAR) related to reporting the results of the surveillance capsule testing. The UFSAR sections affected are 5.3.1.5.3 on initial RT_{NDT} values, 5.3.1.6 on material surveillance, 5.3.2.1 on irradiation effects on core beltline, and 5.3.3.1 on reactor vessel design.

D.1 SECTION 5.3.1.5.3

The paragraph on initial RT_{NDT} values for the vessel shows an initial RT_{NDT} for the beltline weld of -6°F. This was based on an examination of all weld material in the vessel shell welds. More specific information from Chicago Bridge & Iron on the beltline welds allows the RT_{NDT} of -6°F to be changed to -50°F. This change should be incorporated in section 5.3.1.5.3.

D.2 SECTION 5.3.1.6

The beginning of this section of the UFSAR discusses the compliance of the surveillance program with ASTM E185. The only change in this portion, which addresses the physical components of the program, concerns the number of specimens, which is revised in Table 5.3-2. The text portion which should be revised begins with the discussion of the withdrawal schedule. That paragraph and the remainder of 5.3.1.6 should be replaced with the following text:

The contemplated withdrawal schedule for the samples is:

Withdrawal Period (Full Power Years)	Integrated Flux (10^{18} nvt >1 MeV)
6	0.7
15	1.7
32	3.6

There are 37 spare impact specimens and 12 tensile out-of-reactor spare specimens.

The surplus base metal is approximately 12 by 21 by 4-11/16 in. The surplus weld sample plate is approximately 6 by 33 by 4-11/16 in. The surplus plates, if it becomes necessary, can be made into specimens with the following dimensions:

Charpy V-notch specimen - 2.1 by 0.39 by 0.39 in.
Tensile specimen - 0.25 in. in diameter by 3 in. long

The first surveillance capsule was withdrawn after Fuel Cycle 7 for testing. It contained 24 Charpy specimens, 6 tensile specimens and six flux wires. The test results are presented in General Electric report NEDC-31166, Revision 1, and are summarized below.

Gamma spectrometry was used to determine the radioactivity content of iron, copper and nickel flux wires. The fast flux reaction cross sections of these metals, based on over 65 spectral determinations for BWRs and the GE Test Reactor, were used along with the power history, shown in Table 5.3-3, to determine the full power flux at the surveillance capsule location. The flux wire test results are shown in Table 5.3-4.

Table 5.3-2

NUMBER OF SPECIMENS BY SOURCE

Specimen	Base	Weld	Heat-Affected Zone	Suggested Period (EFPY)	Actual Specimen (EFPY)
Tested Unirradiated Baseline					
C [a]	14	12	12	--	--
T [b]	3	3	3	--	--
In-reactor					
C	12	12	12	15	--
T	2	2	2	15	--
C	8	8	8	6	5.9
T	2	2	2	6	5.9
C	8	8	8	32	--
T	2	2	2	32	--
Out of Reactor Spares					
C	11	13	13	--	--
T	3	3	6	--	--

[a] C is standard Charpy V-notch impact bar.

[b] T is 1/4-in. gauge diameter tensile specimen.

Table 5.3-3

SUMMARY OF DAILY POWER HISTORY

Cycle	Cycle Dates	Operating Days	Percent of Full Power	Days Between Cycles
1a	5/14/74 - 6/6/75	389	0.525	
				42
1b	7/19/75 - 2/13/76	210	0.686	
				61
2	4/15/76 - 3/12/77	332	0.695	
				62
3	5/14/77 - 3/18/78	309	0.823	
				39
4a	4/27/78 - 6/17/78	52	0.700	
				265
4b	3/10/79 - 2/9/80	337	0.808	
				68
5	4/18/80 - 3/20/81	337	0.806	
				73
6	6/2/81 - 2/12/83	621	0.540	
				82
7	5/6/83 - 2/2/85	639	0.651	
		3226	0.671 (average)	

Table 5.3-4

SURVEILLANCE CAPSULE LOCATION FLUX AND FLUENCE
FOR IRRADIATION FROM 5/14/74 TO 2/2/85

Wire (Element)	Wire Weight (g)	dps/g Element (at end of Irradiation	Reaction Rate [dps/nucleus (saturated)]	Full Power Flux [a] (n/cm^2-s) >1 MeV	Full Power Flux [a] (n/cm^2-s) >0.1 MeV	Fluence (n/cm^2) >1 MeV	Fluence (n/cm^2) >0.1 MeV
Copper 64713	0.3611	1.99×10^{4}	7.15×10^{-18}				
Copper 64741	0.3588	1.87×10^{4}	6.75×10^{-18}				
		Average =	6.95×10^{-18}	2.6×10^{9}	4.2×10^{9}	4.9×10^{17}	7.8×10^{17}
Iron 64713	0.0694	1.39×10^{5}	3.67×10^{-16}				
Iron 64741	0.1663	1.37×10^{5}	3.62×10^{-16}				
		Average =	3.65×10^{-16}	2.3×10^{9}			
Nickel 64713	0.3234	1.84×10^{6}	4.04×10^{-16}				
Nickel 64741	0.3211	1.93×10^{6}	4.24×10^{-16}				
		Average =	4.14×10^{-16}	2.0×10^{9}			

[a] Full power of 1593 MW_t.

The full power flux calculated was $2.6x10^9$ n/cm^2-s. The integrated fluence (>1 MeV) for the surveillance capsule was determined to be $4.9x10^{17}$ n/cm^2. These values have a 2σ accuracy of

± 5% for dps/g (disintegrations per second per gram)
± 12% for dps/nucleus (saturated)
± 30% for flux and fluence >1 MeV
± 40% for flux and fluence >0.1 MeV

Unirradiated and irradiated Charpy specimens were tested at temperatures chosen to establish the fracture toughness transition curves for the base, weld and HAZ metals. The RT_{NDT} shift for each material was determined from the change in the 30 ft-lb energy index temperature. Experimental shifts of 42°F for base metal and 9°F for HAZ metal were measured. The weld metal test results indicated no measurable shift.

D.3 SECTION 5.3.2.1

The wording to section 5.3.2.1 in NEDC-30839 should be revised to read as follows:

Estimated maximum changes in RT_{NDT} as a function of the end-of-life (EOL) fluence at the 1/4 T depth of the vessel beltline materials are listed below. The updated predicted peak EOL fluence at the 1/4 T depth of the beltline is $3.6x10^{18}$ n/cm^2 after 40 years of service (32 full power years). The EOL fluence prediction is based on a combination of experimental and analytical results. Flux at the 288° surveillance capsule location in the RPV was evaluated by testing flux wires removed with the surveillance capsule after Fuel Cycle 7. The relationship between the capsule location and the peak flux location at the 1/4 T depth was determined by a combination of two-dimensional and one-dimensional flux distribution computer analysis.

The transition temperature shift due to irradiation was calculated with Regulatory Guide 1.99, Revision 1, taking into account the data from the surveillance testing. Specifically, the plate shift of 42°F exceeds the predicted shift of 32°F by 31%. Therefore, a factor of 1.31 was applied to the predicted shift of the limiting beltline plate. The surveillance test results for the weld showed no measurable shift, so the predicted shift was not modified. The results for the core beltline materials are tabulated below:

	Plate	Weld
Limiting Material Chemistry:	0.13% Cu, 0.012% P	0.03% Cu, 0.017% P
EOL Transition Temperature Shift:	86°F	51°F
Initial Reference Temperature:	40°F	-50°F
EOL Adjusted Reference Temperature:	126°F	1°F

Since the predicted EOL adjusted reference temperatures are below 200°F, provisions to permit thermal annealing of the RPV in accordance with Paragraph IV.B of 10CFR50 Appendix G are not required.

D.4 SECTION 5.3.3.1

The third paragraph of section 5.3.3.1 refers to the EOL fluence. This reference should be revised to state the latest calculated EOL fluence of 3.6×10^{18}.