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ANALYSIS OF SEABROOK STATION UNIT 1 REACTOR VESSEL SURVEILLANCE CAPSULE U

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1.0 SUMMARY OF RESULTS

Capsule U, the first Seabrook Station, Unit 1, Reactor Vessel Surveillance Capsule was removed from the Seabrook vessel in August 1991, after 333 37 effective full power days (EFPD) of operation. The analyses of the capsule test specimens and neutron dosimetry led to the following conclusions:

- The capsule received an average neutron fluence (E>1Mev) of 3.11×10^{18} n/cm^2. This is equivalent to the fluence which will be received at the reactor vessel inner diameter after approximately four (4.0) effective full power years (EFPY) of operation.

- The reactor vessel lower shell plate material, R1808-3, was included in the surveillance capsule as the limiting plate material. For its Charpy specimens oriented in the longitudinal direction (LT), the 30 and 50 ft-lb transition temperatures increased by 36°F and 34°F, respectively. The plate's transversely (TL) oriented Charpy specimens experienced increases in the 30 and 50 ft-lb transition temperatures of 28°F and 20°F, respectively. The shift in the 35 mils-lateral-expansion (MLE) index temperature was 24.5°F for LT specimens and 15°F for the TL specimens.

- The weld metal irradiated to 3.11×10^{18} n/cm^2 experienced 30 ft-lb and 50 ft-lb transition temperature increases of 10°F and 15°F, respectively.

- The average upper shelf energy for transversely oriented specimens from lower shell plate R1808-3 decreased from 79 ft-lbs to 72 ft-lbs after irradiation to the fluence of 3.11×10^{18} n/cm^2. The weld metal, exposed to the same fluence as the plate material, experienced a decrease in upper shelf energy from 160 ft-lbs to 129 ft-lbs. The plate and weld materials exhibit upper shelf energies for continued safe plant operation. The upper shelf energy for these materials is expected to be maintained above 50 ft-lb throughout vessel life as required by 10CFR50, Appendix G.

- The adjusted RT_{NDT} values for the plate and weld material, based on the surveillance capsule data, are within the two standard deviations of Regulatory Guide 1.99, Revision 2, predictions.

2.0 INTRODUCTION

The Seabrook Station Unit No. 1 Reactor Vessel Radiation Surveillance Program is described in Westinghouse Report WCAP-10110, dated March 1983.[1] The program utilizes six surveillance capsules. Each capsule contains 60 Charpy V-notch specimens, 9 tensile specimens and 12-1/2T compact test specimens. The capsules contain vessel plate material R1808-3 with Charpy specimens oriented in the longitudinal and transverse direction, weld metal and HAZ material. Each capsule provides accelerated data relative to concurrent reactor vessel inner wall material condition, since the capsules are located in the reactor on the neutron shield pad between the core barrel and the reactor vessel wall, opposite the center of the core. The surveillance program meets the requirements of ASTM E-185-79, "Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels."

The first surveillance capsule in this program, designated Capsule U, was removed during the plant's first refueling outage in August of 1991. The capsule was irradiated for 333.37 effective-full-power-days (EFPD) of operation. The capsule specimens, with the exception of the compact specimens, were tested by B & W Nuclear Services Co.[2] The 1/2T compact test specimens are being saved for future test needs. The capsule data is attached as Appendix B to this report. The analysis of the specimen data and dosimetry was performed by the Yankee Atomic Electric Co. This analysis is the subject of this report.

1.Superscripts denote references loacted in Section 7.0 of this report.

3.0 SURVEILLANCE MATERIALS AND CHEMISTRY

Based on an evaluation of the vessel plate materials, considering initial RT_{NDT} values, chemistry and the irradiation prediction methods of Regulatory Guide 1.99, Revision 1, vessel lower shell plate R1808-3 was expected to have the highest end-of-life RT_{NDT}. This reactor vessel surveillance material was supplied by the vessel fabricator Combustion Engineering, Inc. Additionally, Combustion Engineering, Inc. supplied a weldment made up of sections of Lower Shell Plate R1808-3 and the adjacent Lower Shell Plate R1808-1. This weldment was made using Weld Wire Heat No. 4P6052 and Linde flux 0091, Lot No. 0145. The reactor vessel beltline weld, intermediate and lower shell longitudinal weld seams, and the intermediate to lower shell girth welds, were all fabricated using the above weld wire/flux combination. Therefore, the weld supplied for the surveillance program is the limiting weldment. The chemical analyses, heat treatment history, drop weight and RT_{NDT} values for the materials used in the beltline region of the Seabrook Station Unit No. 1 are provided in Appendix A to this report. The tables are reproduced from the Westinghouse description of the Surveillance Program, WCAP-10110[1].

4.0 PRE- AND POST-IRRADIATION TEST RESULTS

4.1 Baseline Data, WCAP 10110

The surveillance program materials were tested in the unirradiated or baseline condition by Westinghouse and these results are reported in Westinghouse report WCAP-10110.[1] The Charpy impact test results of the base line data reported in WCAP-10110 are shown in Table 4-1 as the zero fluence values. The tensile data is shown in Table 4-2 as the zero fluence values.

4.2 Irradiated Data, B&W Report BAW-2157

The Capsule U test specimens and dosimetry were tested by B & W, as mentioned previously. The test data is provided as Appendix B to this report. The Charpy impact results from the irradiated capsule is listed in Table 4-1 with the unirradiated data. The irradiated Charpy data were analyzed using the EPRI Tanh Curve Fitting Routine, Version 1.8 (see Section 4.3). The upper shelf energy values were calculated using the averaging technique described in ASTM E-185-82[3], and its current revision due for issuance in 1992. The tensile data from the surveillance capsule is listed in Table 4-2. The effects of irradiation on the tensile properties are shown graphically in Figures 4-1 through 4-3. All data were analyzed in accordance with the 1982 revision to ASTM E-185[3] as specified by 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4.3 Tanh Curve Fits

ASTM E-185-82[3] defines the Charpy V-notch impact test transition temperature as "the difference in the 30 ft-lbf (41J) index temperatures for the best fit (average) Charpy curve measured before and after irradiation." There are two methods employed in the industry for determining the best fit Charpy curve. The first is to "eye" the data and draw a best fit curve through it; the second is to use a hyperbolic tangent function (Tanh) and computer fit the data to determine the best curve shape. EPRI, in conjunction with industry experts, developed a computer routine for fitting Charpy test data with the Tanh function. The advantage to using this computer routine is that the curve fits are performed in a consistent manner. This reduces scatter in the reported shift data. To generate the irradiated Charpy results reported in Table 4-1, the EPRI Tanh Curve Fitting Routine Version 1.8 was used on the data reported in BAW-2157.

For the Seabrook Station Unit 1 Reactor Vessel Surveillance Program, the initial (unirradiated) Charpy values for the 30 ft-lb and 50 ft-lb fixes (T_{30} and T_{50}) and for the 35 mils-lateral-expansion (MLE) are documented in Westinghouse Report WCAP-10110. The irradiated values will be those generated by using the Tanh computer fit. Table 4-3 provides a comparison of the Westinghouse (baseline) reported values, B & W (irradiated) reported values, and the Tanh fit values.

4.4 Results

The tensile results are presented in Table 4-2 and graphically in Figures 4-1, 4-2, and 4-3. The irradiated yield and tensile strengths increased slightly over the unirradiated values. The properties which measure ductility, reduction in area and elongation, decreased with irradiation These changes are a result of irradiation induced microstructural changes and were expected. The tensile property changes will not effect reactor vessel operation.

The Charpy data, presented in Table 4-1, showed increases in the Charpy transition temperatures (T_{30}, T_{50}, MLE) and slight decreases in the Charpy upper shelf energies of the various materials. These results are shown graphically for the surveillance plate R1808-3 specimens in Figure 4-4 (longitudinal orientation) and Figure 4-5 (transverse orientation). The weld metal results are shown' graphically in Figure 4-6 and the heat-affected-zone (HAZ) material is shown graphically in Figure 4-7 . All graphs of Charpy data show the curve fit using the Tanh function for both the unirradiated and irradiated data. This was done in order to produce the graphs with computer software.

TABLE 4-1

Seabrook Surveillance Capsule Data Per
ASTM E-185 and Regulatory Guide 1.99.
Initial Values (Fluence = 0) Per WCAP-10110.

MATERIAL	FLUENCE E+19 n/cm^2	INITIAL RT_{NDT} °F	T_{30} °F	T_{30} SHIFT °F	ADJUSTED(b) RT_{NDT} °F	T_{50} °F	T_{50} SHIFT °F	USE(c) FT-LBS	% USE DROP	35 MLE °F	35 MLE SHIFT °F
Plate No. R1808-3 LT	0	NA	-25	--	--	0	--	125	--	0	--
(a)	0.3	--	11	36	--	34	34	118	5.6	24.5	24.5
Plate No. R1808-3 TL	0	40	10	--	--	60	--	79	--	50	--
(a)	0.3	--	38	28	96	80	20	72	8.8	65	15
WELD	0	-60	-60	--	--	-45	--	160	--	-50	--
(a)	0.3	--	-50	10	-40	-30	15	129	10	-32	18
HAZ	0	NA	-160	--	--	-120	--	129	--	-105	--
(a)	0.3	--	-104	56	--	-60.5	59.5	131	0	-57	48

NOTES:

(a) Except for USE (see Note (c)), data is derived from test data reported in BAW-2157, dated March 1992 using the EPRI Tanh Curve Fitting Routine Version 1.8.

(b) The adjusted RT_{NDT}s are determined using Regulatory Guide 1.99[4] where,

Adj. RT_{NDT} = Int. RT_{NDT} + T_{30} Shift + Margin, and

Margin = 2* $(\text{sigma}_i^2 + \text{sigma}_s^2)^{.5}$; sigma_s is 28°F for welds and 17°F for base metal except that sigma_s need not exceed 0.5 times the mean value of ΔRT_{NDT}.

For the plate (TL) and weld initial RT_{NDT}s, sigma_i is set to zero because the initial RT_{NDT}s were determined in accordance with ASME code, Section III, NB-2300 which is a conservative method based on drop weight and Charpy data. For the plate shift, the margin is 14°F which is 0.5 x T_{30} Shift. For the weld shift, the margin is 5°F which is 0.5*T_{30} Shift.

(c) Upper shelf energy is determined using the average of the values designated in BAW-2157 in accordance with ASTM E-185-82.

NA = Not Applicable

TABLE 4-2

Seabrook Station Unit 1 Surveillance Program Tensile Properties
Base Line Properties Data Per WCAP-10110
Irradiated Data Per BAW-2157

MATERIAL	FLUENCE E+19 n/cm^2	TEST TEMP. °F	UTS KSI	.2% YS KSI	RED. IN AREA %	TOTAL ELONG. %	UNIFORM ELONG. %
PLATE R1808-3 LT	0.00	75.00	92.00	74.00	65.00	24.00	14.50
	0.00	75.00	91.00	70.00	68.00	27.00	14.50
	0.00	300.00	85.00	64.00	68.00	24.00	13.00
	0.00	300.00	85.00	65.00	66.00	24.00	13.00
	0.00	550.00	89.00	63.00	61.00	24.00	14.00
	0.00	550.00	89.00	63.00	63.00	24.00	13.00
	0.30	70.00	94.20	73.60	63.50	23.70	9.80
	0.30	300.00	86.60	67.60	65.40	20.50	8.30
	0.30	550.00	91.10	67.30	62.10	19.40	7.80
PLATE R1808-3 TL	0.00	75.00	91.00	71.00	55.00	26.50	15.50
	0.00	75.00	91.00	71.00	55.00	26.50	15.50
	0.00	300.00	86.00	66.00	53.50	21.00	12.00
	0.00	300.00	86.00	66.00	55.00	22.00	12.00
	0.00	550.00	88.00	63.00	51.50	21.00	13.00
	0.00	550.00	88.00	64.00	47.50	24.00	15.50
	0.30	70.00	94.10	73.30	54.30	21.40	9.50
	0.30	300.00	85.70	67.20	55.90	17.80	7.80
	0.30	550.00	91.30	66.40	48.30	15.90	7.90
WELD MATERIAL	0.00	75.00	87.00	75.00	71.00	27.00	15.00
	0.00	75.00	88.00	74.00	75.00	28.00	14.00
	0.00	300.00	81.00	68.00	73.00	18.00	9.00
	0.00	300.00	81.00	67.00	73.00	23.00	10.00
	0.00	550.00	85.00	66.00	67.00	22.00	10.00
	0.00	550.00	84.00	65.00	71.00	22.00	11.00
	0.30	70.00	90.00	76.00	72.50	23.70	8.90
	0.30	300.00	83.70	70.50	71.70	21.20	7.50
	0.30	550.00	87.70	69.40	69.30	18.00	4.90

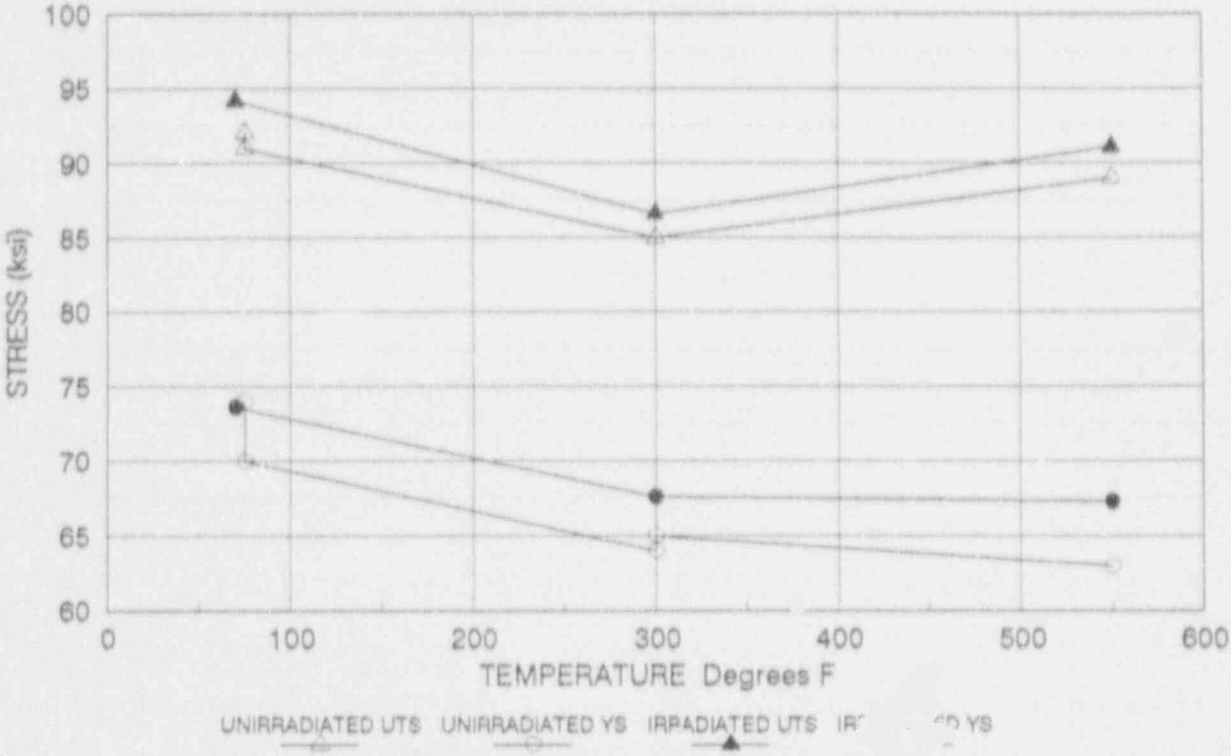
TABLE 4-3

A Comparison of Tanh Computed Charpy Values to Westinghouse and B&W Reported Values

MATERIAL	PARAMETER	Tanh	WESTINGHOUSE	B&W
		Unirradiated Data		
Plate	T_{30}	-28°F	-25°F	--
R1808-3	T_{50}	-4°F	0°F	--
LT	35MLE	-6°F	0°F	--
Plate	T_{30}	9°F	10°F	--
R1808-3	T_{50}	58°F	60°F	--
TL	35MLE	46°F	50°F	--
HAZ	T_{30}	-199°F	-160°F	--
(a)	T_{50}	-139°F	-120°F	--
	35MLE	-126°F	-105°F	--
WELD	T_{30}	-75°F	-60°F	--
	T_{50}	-50°F	-45°F	--
	35MLE	-47°F	-50°F	--
		Irradiated Data		
Plate	T_{30}	11°F	--	3°F
R1808-3	T_{50}	34°F	--	40°F
LT	35MLE	24°F	--	26°F
Plate	T_{30}	38°F	--	39°F
R1808-3	T_{50}	80°F	--	78°F
TL	35MLE	65°F	--	68°F
HAZ	T_{30}	-104°F	--	-85°F
(a)	T_{50}	-60°F	--	-41°F
	35MLE	-57°F	--	-47°F
WELD	T_{30}	-50°F	--	-56°F
	T_{50}	-30°F	--	-38°F
	35MLE	-32°F	--	-40°F

Note (a) HAZ data exhibit significant scatter which is typical of HAZ material.

Figure 4-1 TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL PLATE R1808-3 LONGITUDINAL PROP.
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA



TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL PLATE R1808-3 LONGITUDINAL PROP.
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA

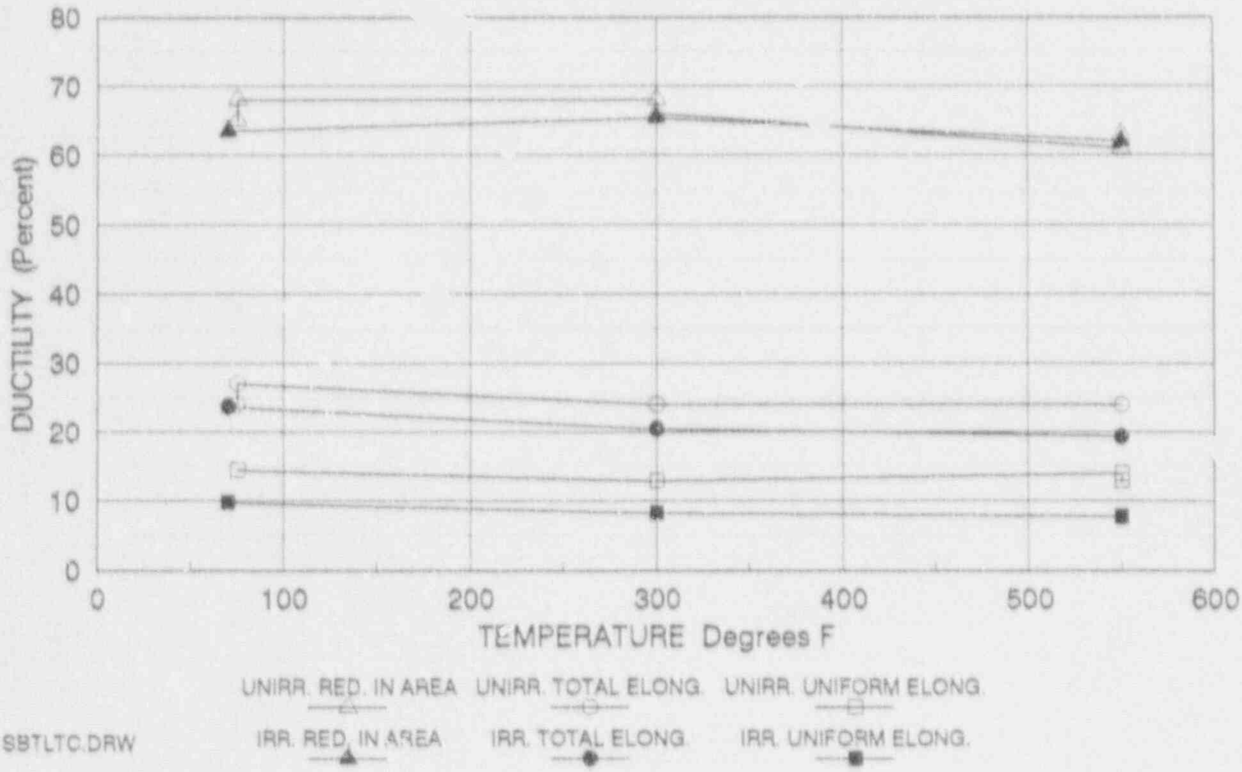


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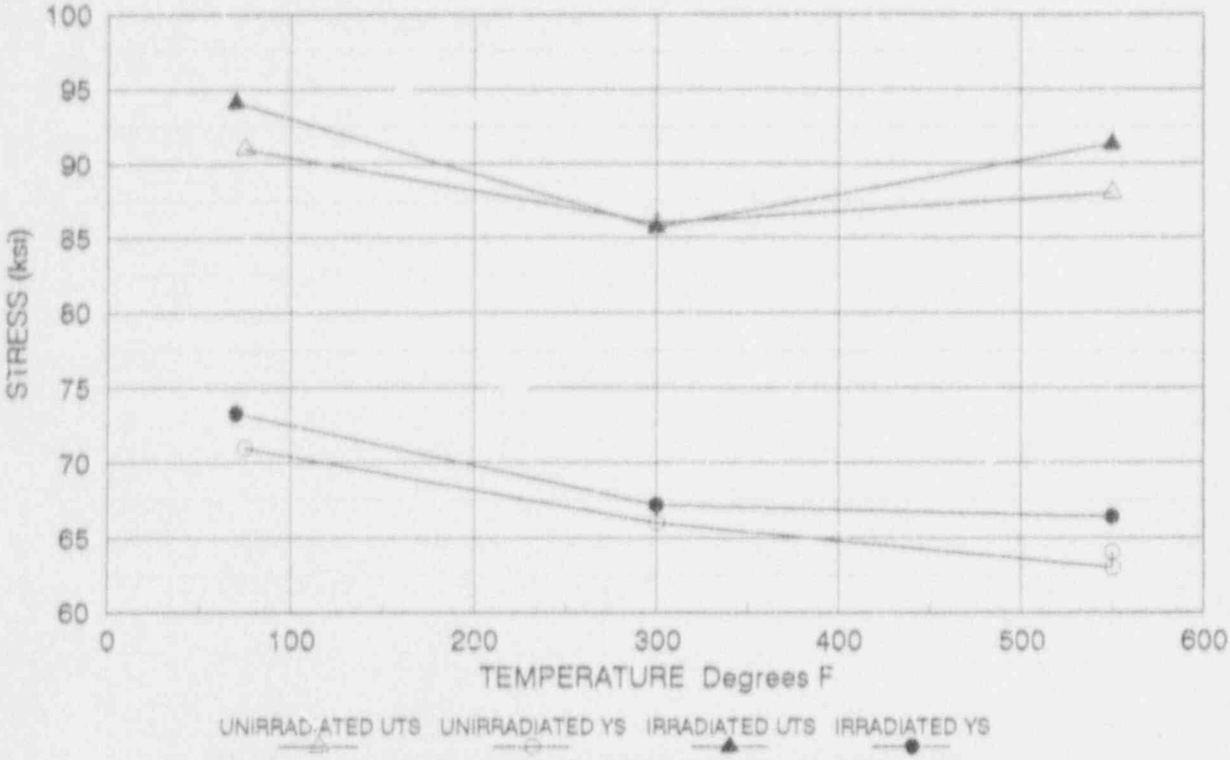
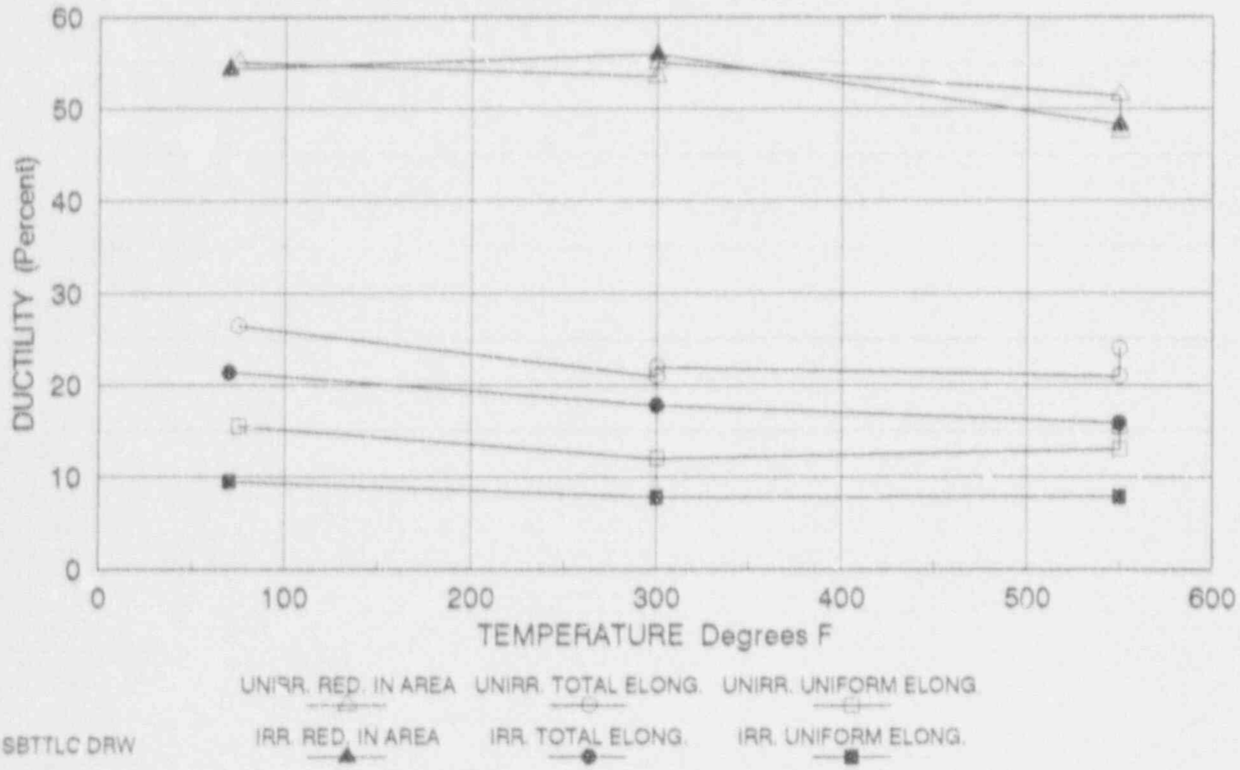
Figure 4-2

TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL PLATE R1808-3 TRANSVERSE PROP.
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA



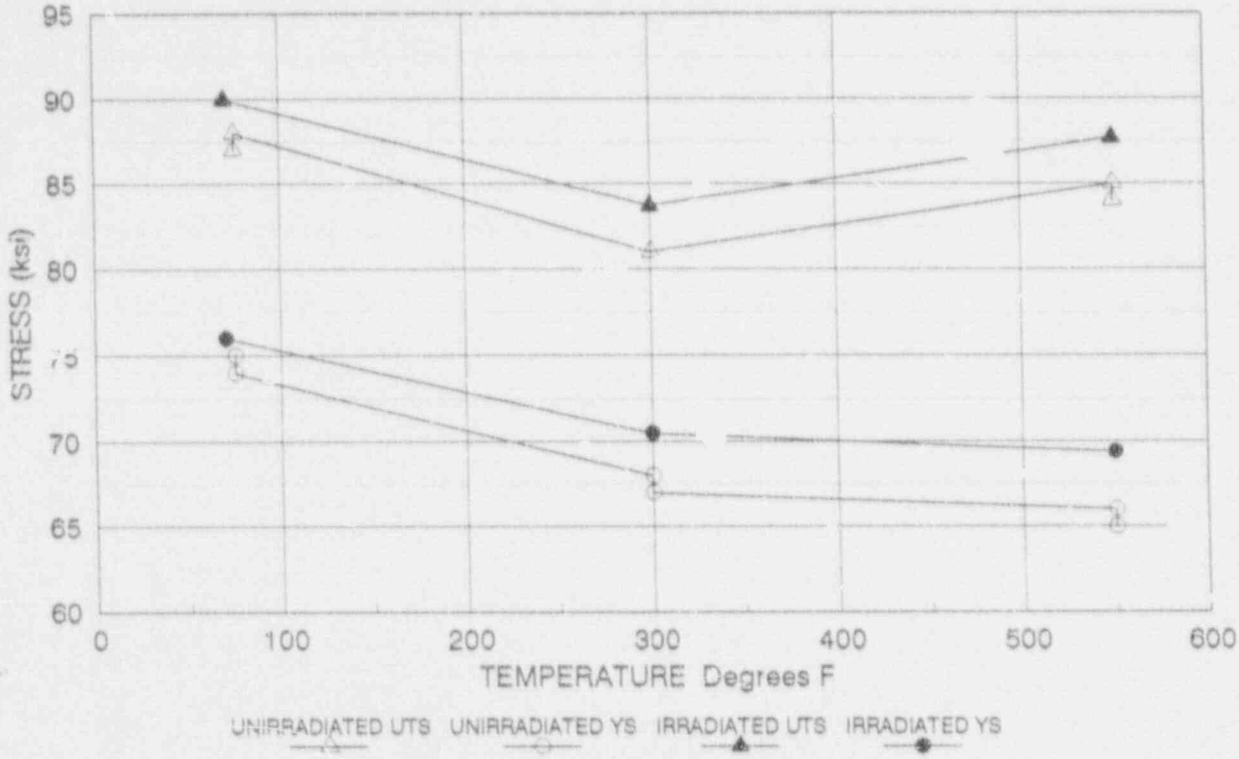
TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL PLATE R1808-3 TRANSVERSE PROP.
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA



SBTTLC DRW

Figure 4-3

TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL WELD METAL
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA



TENSILE PROPERTIES FOR SEABROOK STATION UNIT 1
REACTOR VESSEL WELD METAL
UNIRRADIATED AND SURVEILLANCE CAPSULE U DATA

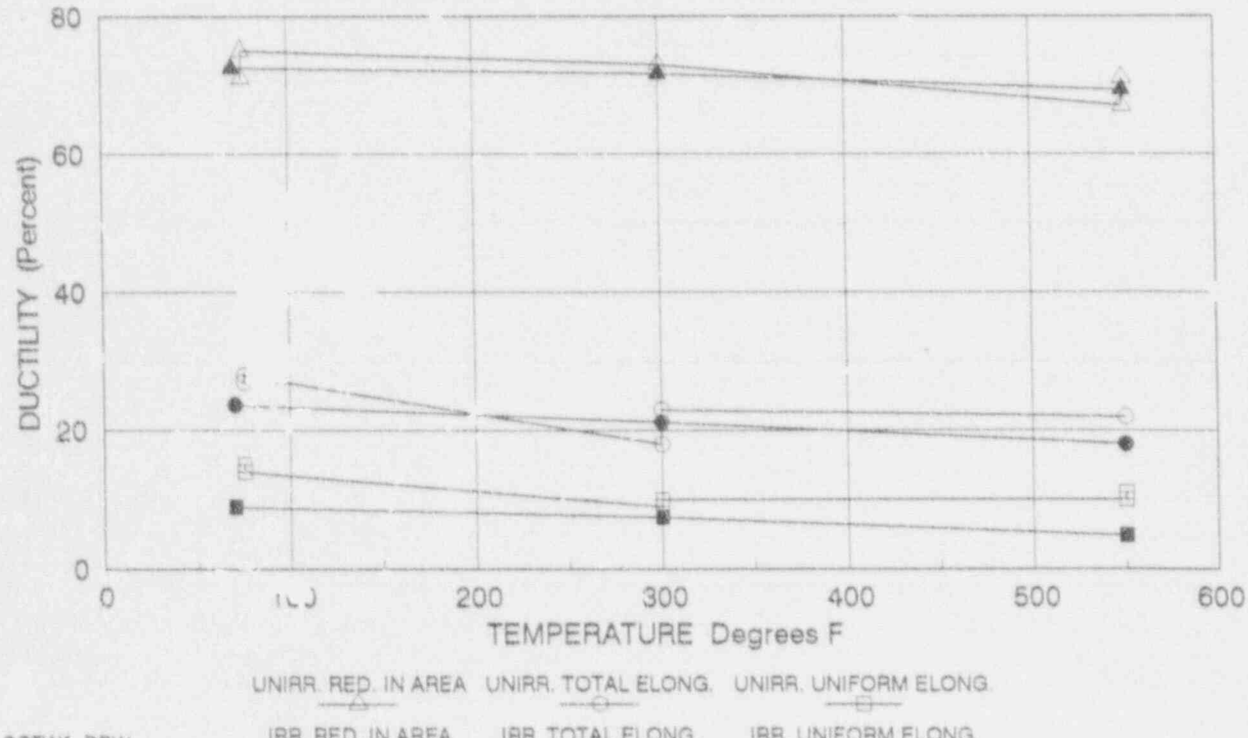


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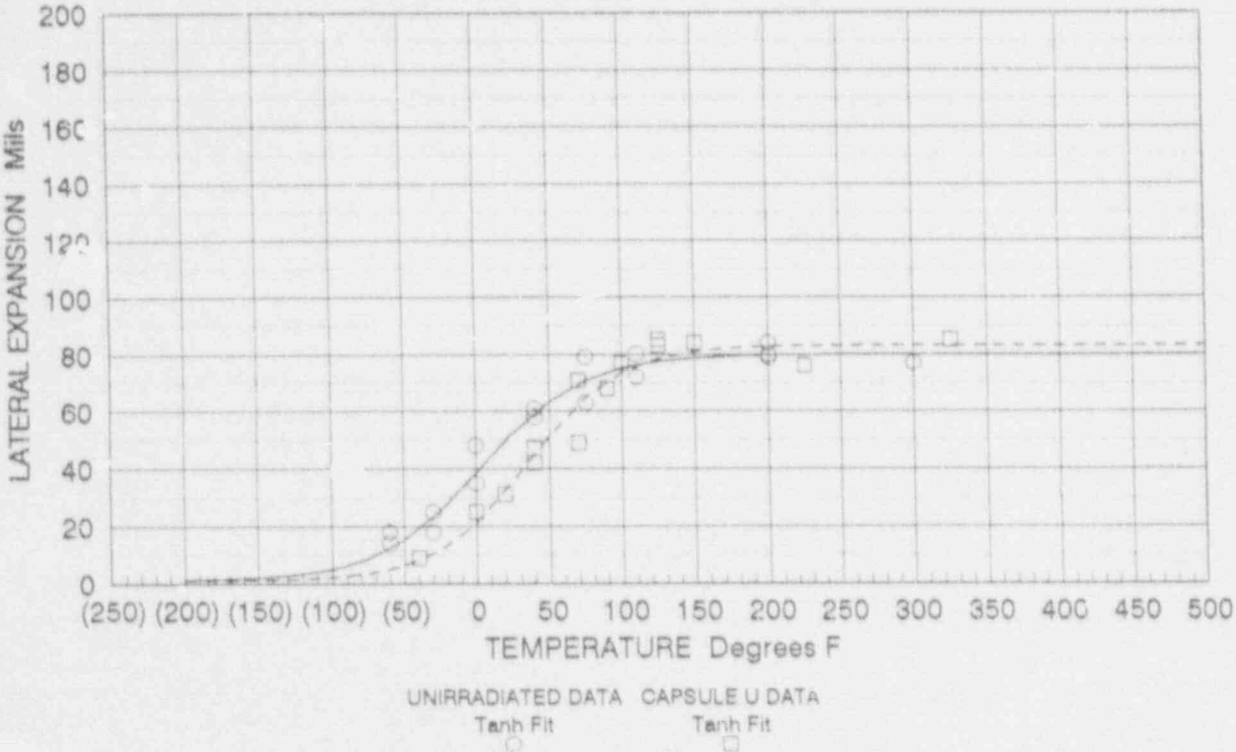
Figure 4-4

SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
LONGITUDINAL CHARPY IMPACT DATA - MILS LAT. EXP.



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CAPSULE U SURVEILLANCE RESULTS
LONGITUDINAL CHARPY IMPACT DATA

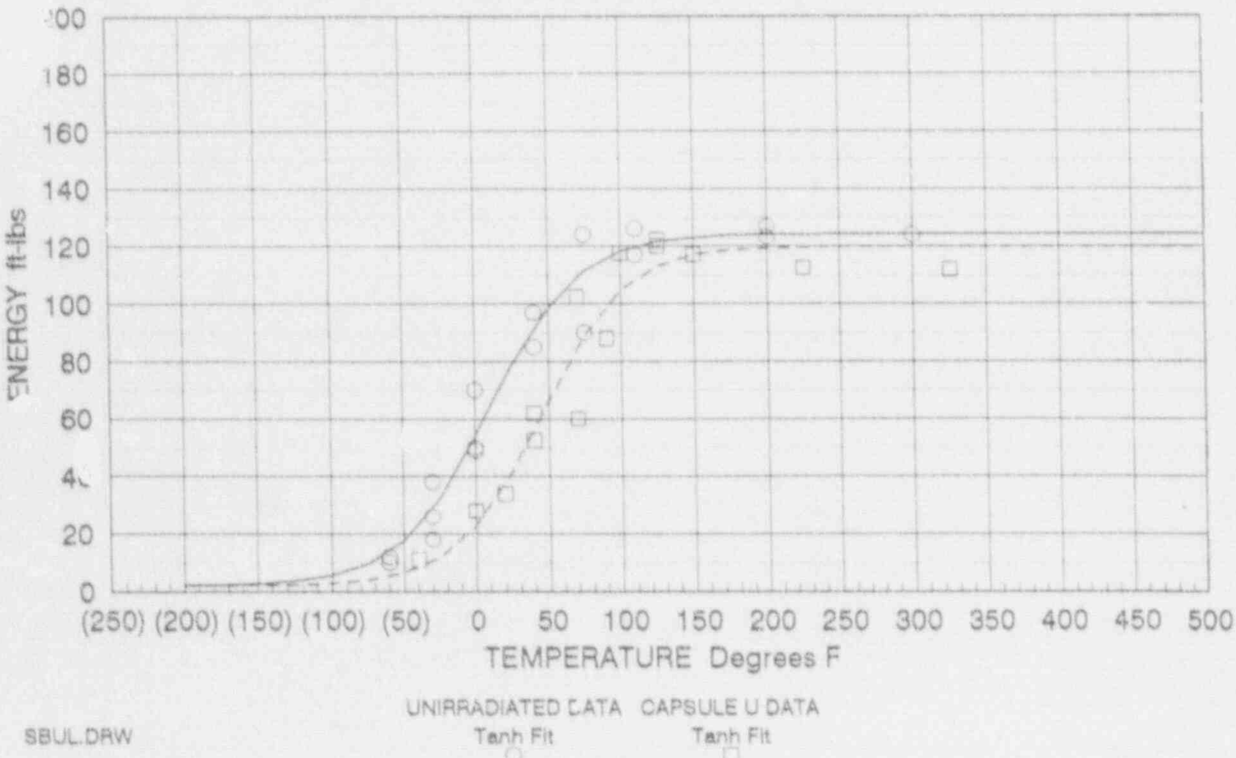

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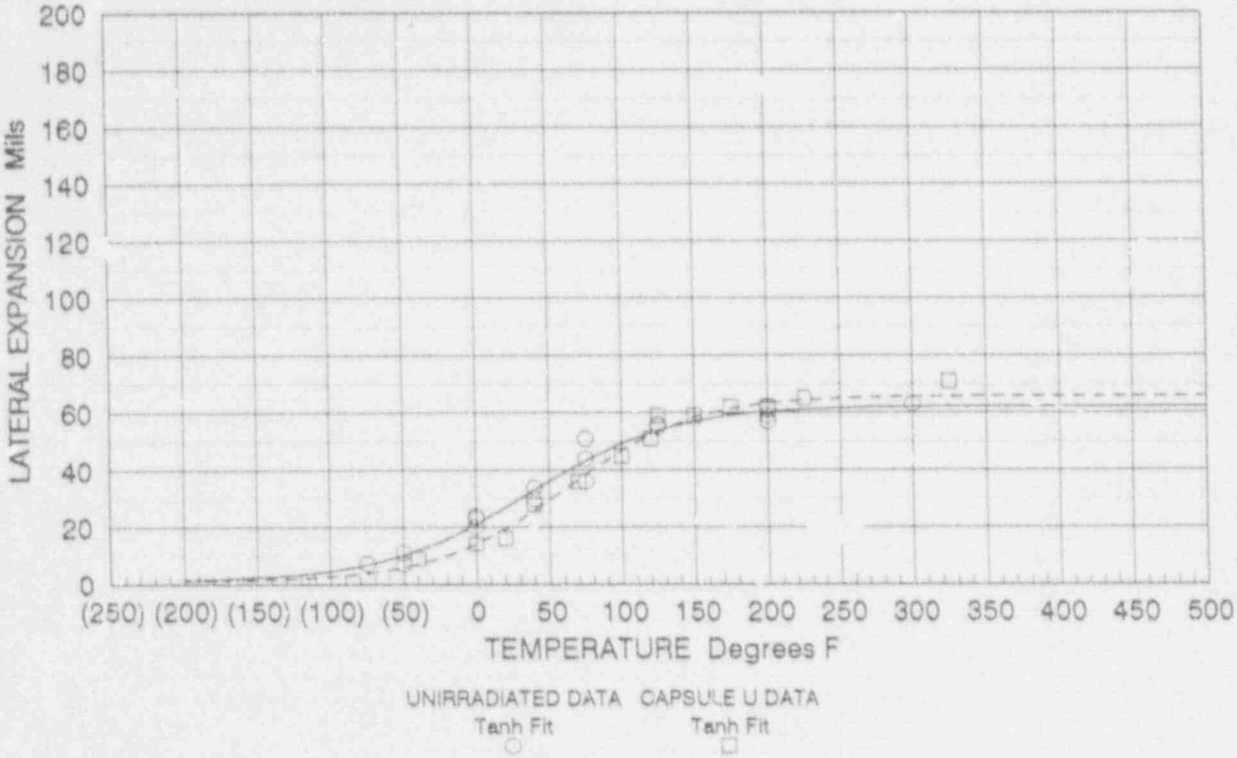
Figure 4-5

SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
TRANSVERSE CHARPY IMPACT DATA - MILS LAT. EXP.



SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
TRANSVERSE CHARPY IMPACT DATA

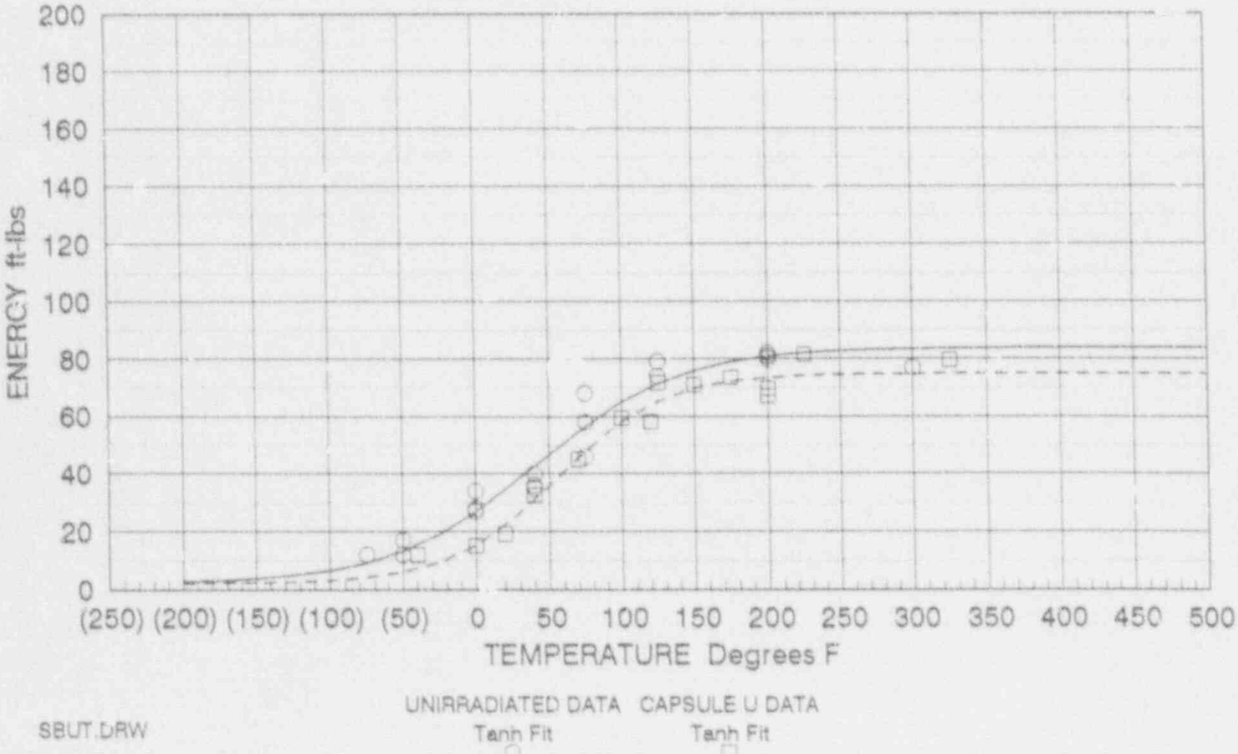


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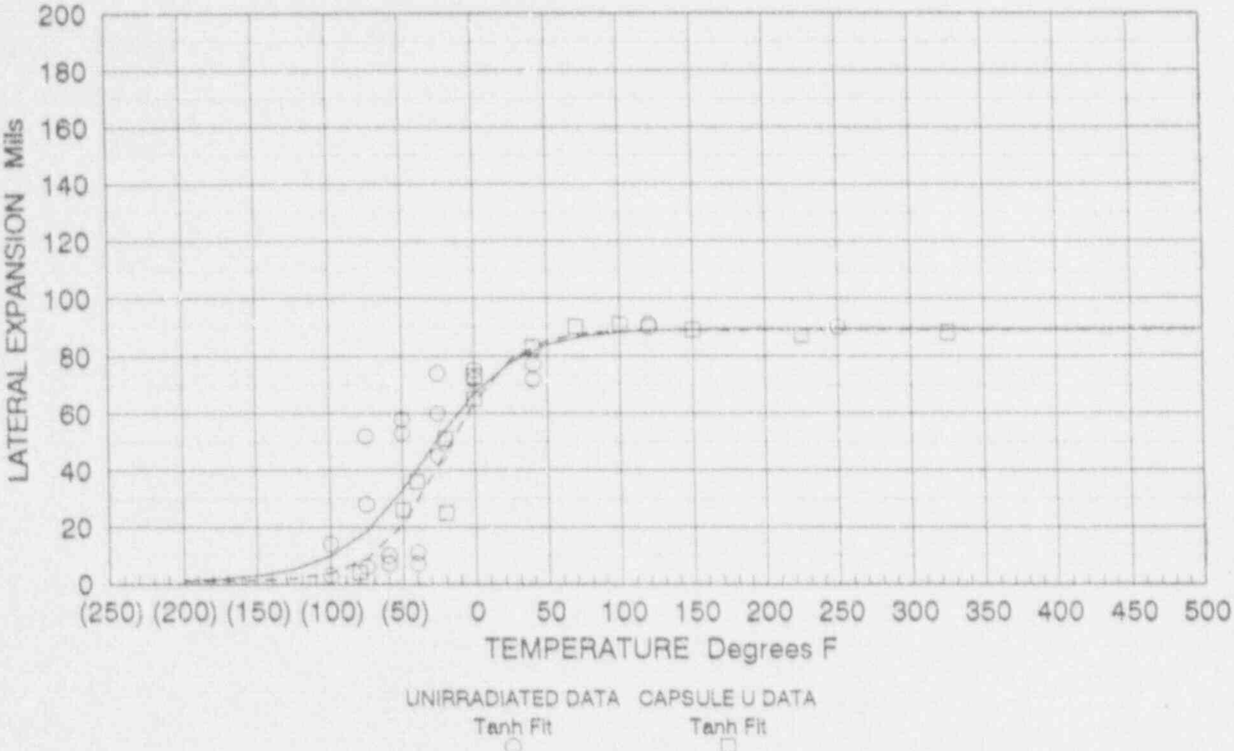
Figure 4-6

SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
WELD METAL CHARPY IMPACT DATA - MILS LAT. EXP.



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CAPSULE U SURVEILLANCE RESULTS
WELD METAL CHARPY IMPACT DATA

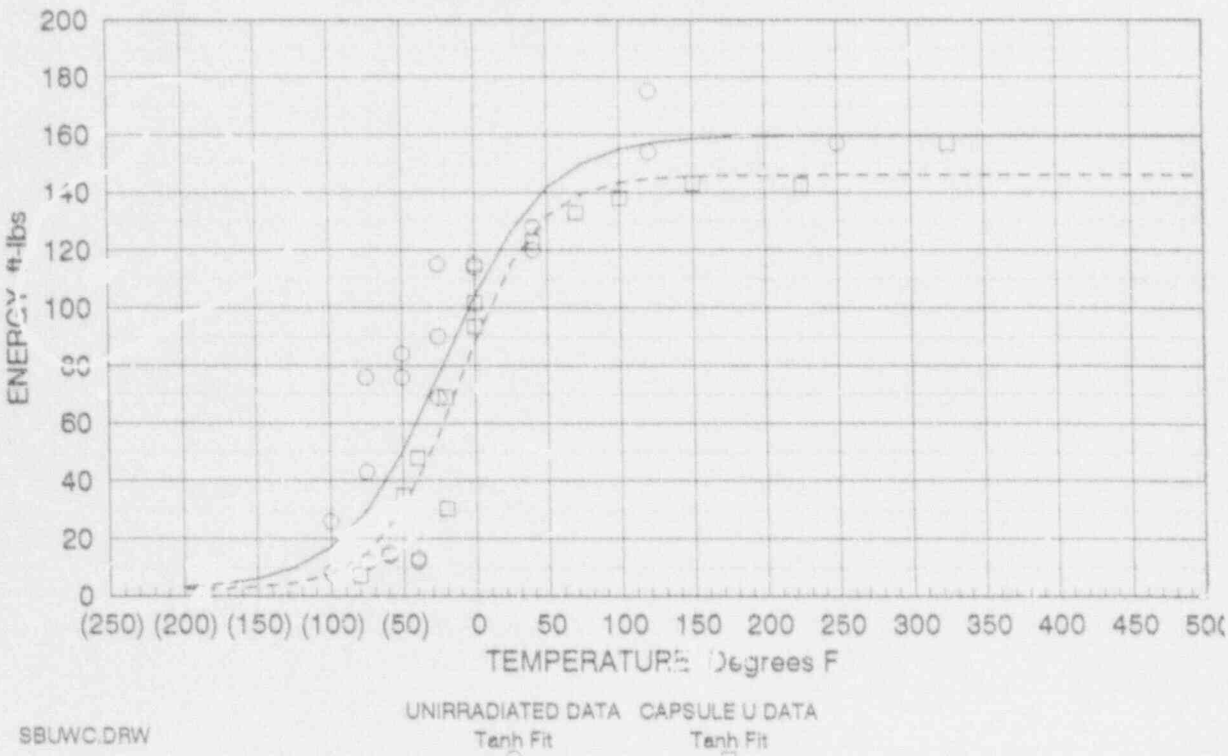

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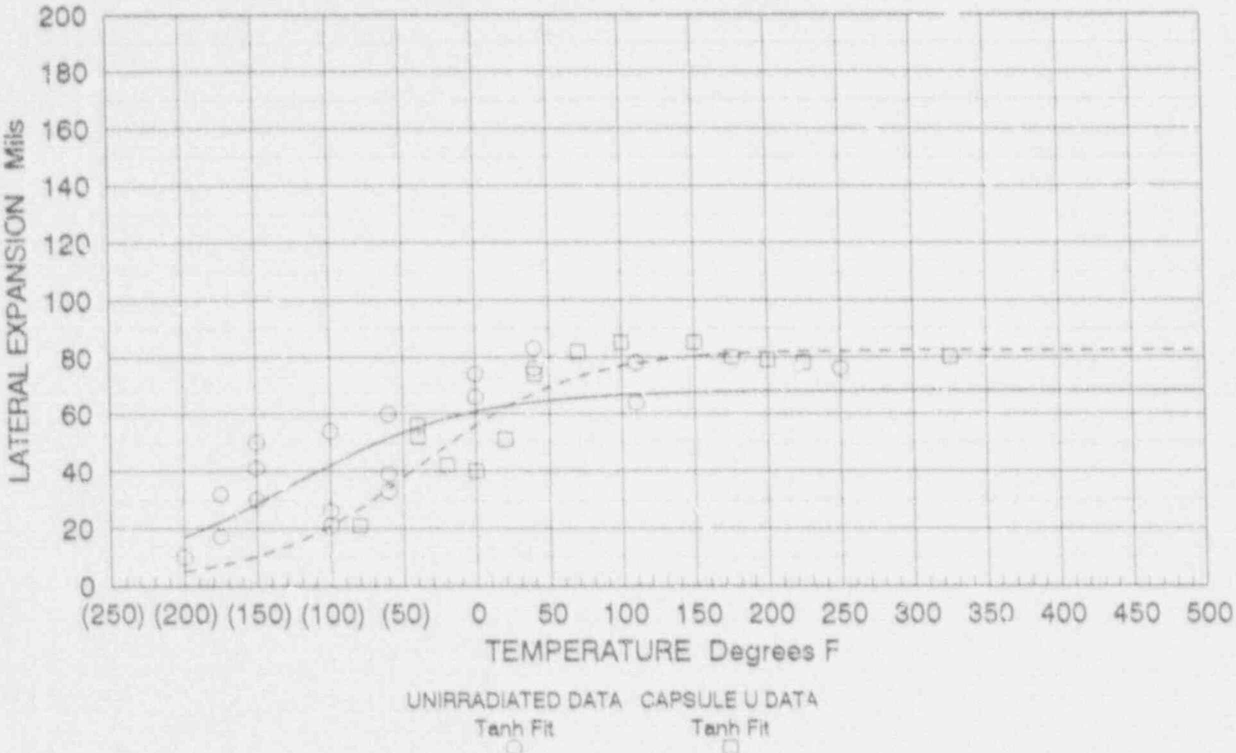
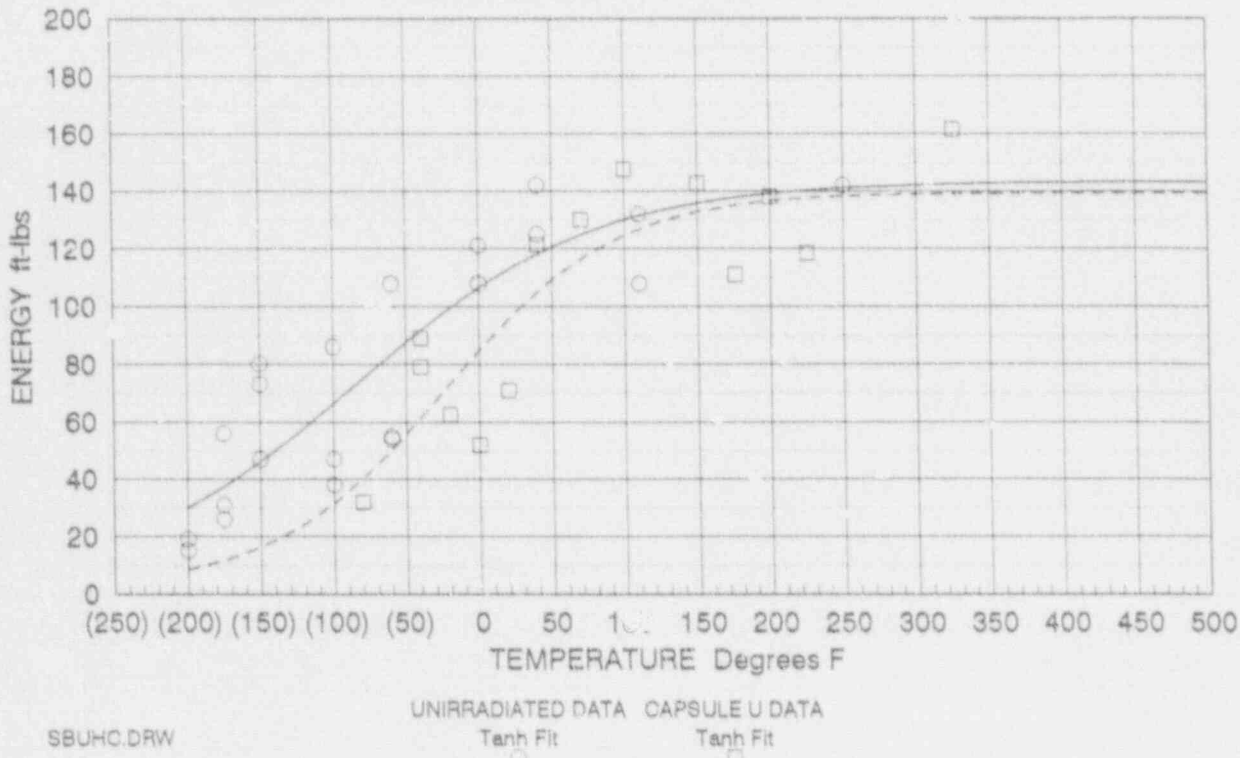
Figure 4-7

SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
HEAT-AFFECT-ZONE MAT'L Cv DATA - MILS LAT. EXP.



SEABROOK STATION UNIT NO. 1
CAPSULE U SURVEILLANCE RESULTS
HEAT AFFECTED ZONE MAT'L. CHARPY IMPACT DATA



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5.0 RT_{NDT} CALCULATION

5.1 Calculated RT_{NDT} Values

Adjusted RT_{NDT} values of 96°F for the R1808-3 surveillance plate and 40°F for the surveillance weld metal (see Table 4-1) were calculated from the surveillance capsule test results using the methodology of Regulatory Guide 1.99, Revision 2. Regulatory Guide 1.99, Revision 2,[4] requires that two or more credible surveillance data sets be available before actual surveillance data can be used for licensing purposes. Since this is the analysis of the first capsule (only one data set), these calculated RT_{NDT} values are presented for information only.

In the absence of two or more credible surveillance data sets, Section C.1 of Regulatory Guide 1.99, Revision 2 requires that the adjusted RT_{NDT} be based on a delta RT_{NDT} value calculated from fluence and derived chemistry factors for the material. To determine the adjusted RT_{NDT} numbers in Table 4-1, the delta RT_{NDT} value is the T_{30} shift value measured from Charpy specimen testing. This allows the surveillance data from this first capsule to be compared with Regulatory Guide 1.99 predictions (discussed in Paragraph 5.2). The adjusted RT_{NDT} calculation used to develop Table 4-1 values is as follows:

The adjusted RT_{NDT}s are determined using R. G. 1.99 equations, where,

Adjusted RT_{NDT} = Initial RT_{NDT} + T_{30} Shift + Margin

where, Margin = $2*(\text{sigma}_i^2 + \text{sigma}_s^2)^{.5}$

For the plate (TL) and weld initial RT_{NDT}s, the initial sigma margin (sigma_i) is set to zero because the initial RT_{NDT}s were determined in accordance with ASME Code, Section III, NB2300 which is a conservative (upper bound) method based on drop weight and Charpy data. Per Regulatory Guide 1.99, Revision 2, sigma_s is 28°F for welds and 17°F for base metal except that sigma_s need not exceed 0.5 times the mean value of T_{30} Shift. Therefore, sigma_s for plate is 14°F (28°F ÷ 2) and sigma_s for weld is 5°F (10°F ÷ 2).

Thus, the Table 4-1 values are:

Plate: Adj. RT_{NDT} = 40°F + 28°F + 2(14°F) = <u>96°F</u>

Weld: Adj. RT_{NDT} = -60°F + 10°F + 2(5°F) = <u>-40°F</u>

5.2 Comparison of Surveillance Capsule Results to Regulatory Guide 1.99, Revision 2 Predictions

As described in Section 5.0, the adjusted RT_{NDT} values reported in Table 4-1 were based on Charpy T_{30} shift results rather than Regulatory Guide 1.99 delta RT_{NDT} values calculated using material chemistry factors and the fluence function. To compare the capsule results to Reg. Guide 1.99 predictions, chemistry factors for the surveillance plate and weld were calculated using Tables 1 and 2 of Reg. Guide 1.99. Applying the chemistry factors to the Reg. Guide 1.99 fluence function, delta RT_{NDT} curves were generated to graphically show the Reg. Guide 1.99 predictions against the T_{30} shift data from the surveillance capsules.

Figure 5-1 depicts the T_{30} shifts for the plate L-T and T-L oriented Charpy specimens and weld specimens plotted against the Reg. Guide 1.99 prediction curve. The Reg. Guide curve has the form of:

delta RT_{NDT} (shift) = $(CF)f^{(0.28 - 0.10\log f)}$ where,

f = fluence (E+19), and

CF = chemistry factor.

For the plate, the CF is 37 (Cu = .06wt%, Ni = 57wt%). For the weld, the CF is 23.5 (Cu = .02wt%, Ni = .10wt%).

The surveillance capsule, transverse plate data (orientation used for RT_{NDT} determination) shown at the top of Figure 5-1 agrees well with the Reg. Guide 1.99 prediction line. The longitudinal data point falls above the prediction line. Both points fall below the two sigma margin limit specified by Regulatory Guide 1.99 indicating good agreement with Reg. Guide 1.99 predictions.

The weld metal results are shown at the bottom of Figure 5-1. Here the weld Charpy shift result falls slightly below the prediction line. Again, the test data point falls below the two sigma margin indicating good agreement with Regulatory Guide 1.99 predictions. The +2 sigma margins are shown as the dashed lines in Figure 5-1. As explained in Section 5.1, the one sigma values for the analysis are 14°F for plate and 5°F for weld metal.

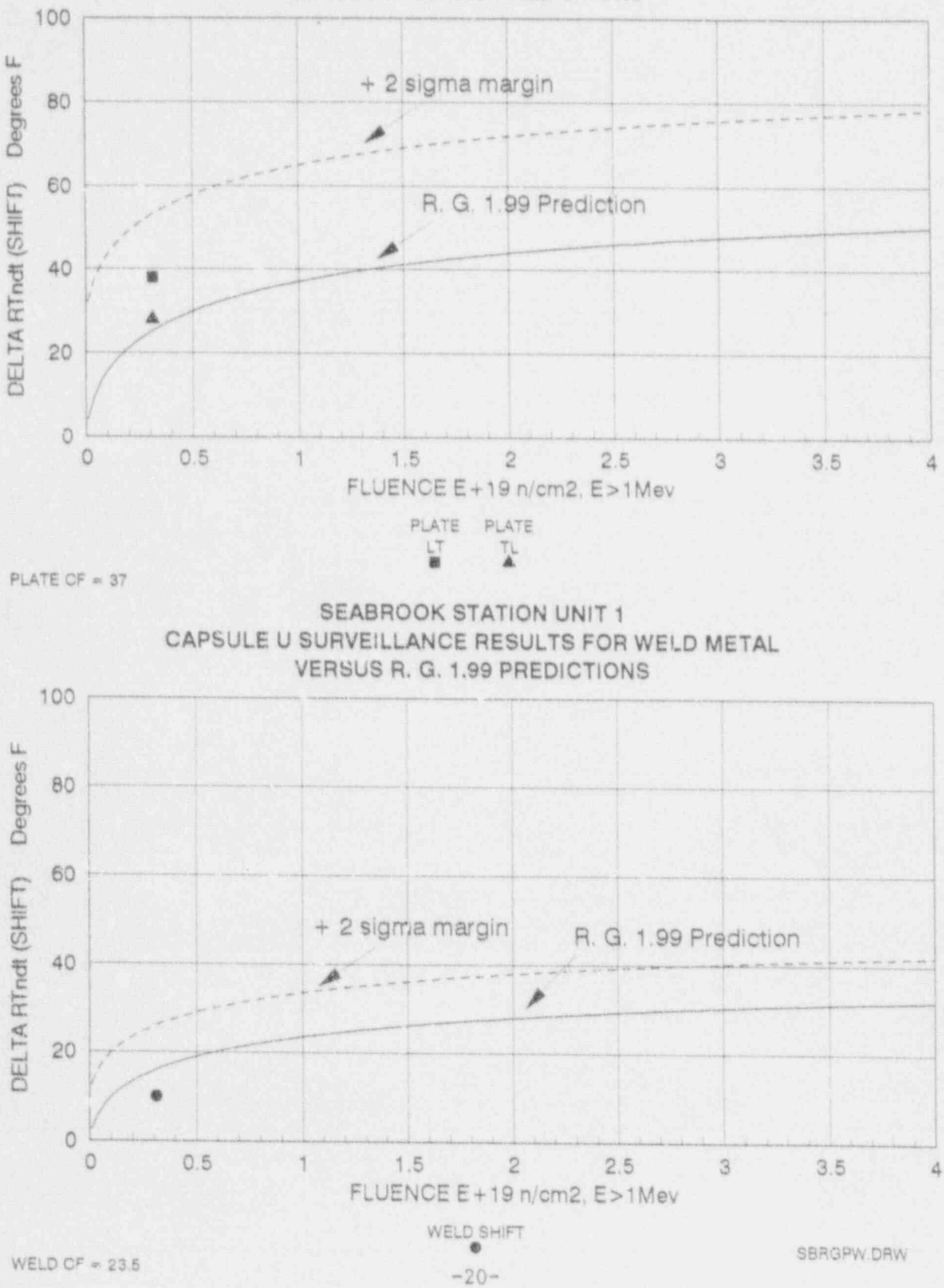
5.3 Effect of Surveillance Results on Plant Current Technical Specifications

The current Seabrook Technical Specifications contain heatup, cooldown, and LTOP curves which are based on the predicted RT_{NDT} shift in Regulatory

Guide 1.99, Revision 1, at 16 EFPY. A Technical Specification amendment which bases the RT_{NDT} prediction on Revision 2 of the Regulatory Guide will be submitted in the third quarter of 1992. The pressure-temperature curves in the amendment are identical to those in the current Technical Specifications, however, the applicability of the curves has been reduced to 11 EFPY due to the conservatism in the revised Regulatory Guide. These curves will be re-evaluated upon analysis of the second surveillance capsule which is scheduled for removal after approximately 5 EFPY of operation. At that time, two sets of irradiated material data will be available which is the minimum number of data sets required by Regulatory Guide 1.99, Revision 2, to establish credibility of the RT_{NDT} shift values.

SEABROOK STATION UNIT 1

Figure 5-1 CAPSULE U SURVEILLANCE RESULTS FOR PLATE VERSUS R. G. 1.99 PREDICTIONS



SBRGPW.DRW

6.0 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 Introduction

The characterization of the neutron environment with the reactor pressure vessel and surveillance capsule geometries is necessary to properly assess the effects of neutron irradiation on these components. The neutron energy spectrum, flux and fluence to which these components have been exposed are necessary elements in the analyses of neutron damage to the steels used in these components. In addition, a relationship must be determined which will relate the changes observed in the test specimens to present and future conditions of the reactor. To characterize the flux in the capsule requires a combination of analysis and measurement of the neutron flux monitors (dosimeters) contained in each surveillance capsule. The prediction of future conditions of the reactor is based on analysis.

The conversion of the measured dosimeter activity to flux for Seabrook Capsule U will be based on the use of spectrum averaged cross-sections derived from calculations performed on plants similar to Seabrook. This approach provides a good approximation to the flux experienced by the dosimeters. A Seabrook-specific two-dimensional discrete ordinate transport model is being developed. This Seabrook-specific model will be utilized to characterize the flux seen by Capsule U, to analyze future surveillance capsules, and to predict future conditions of the reactor vessel.

This section describes the analysis of the dosimeters and the development of the two-dimensional model.

6.2 Neutron Calculations

A Seabrook-specific two-dimensional model is being developed for future flux and fluence calculations. The calculation will be a forward transport calculation in r,θ geometry using the DORT two-dimensional discrete ordinates code[5] and the SAILOR[6] cross-section library. The SAILOR library is a coupled, self-shielded, 47 neutron, 20 gamma-ray, P_3, ENDF/B-IV based cross-section library for light water reactor applications. In the DORT analysis, anisotropic scattering will be treated with a P_3 expansion of the cross-sections and an S_8 angular quadrature. The reactor geometry being developed includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel, and water annuli) as well as the surveillance capsule and the appropriate reactor core fuel loading pattern and power distribution. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals will be accounted for in the analytical approach.

6.3 Neutron Dosimetry Analysis

In order to effect a correlation between fast neutron (E > 1.0 MeV) exposure and the radiation induced properties changes observed in the test specimens, a number of fast neutron flux monitors are included as an integral part of the reactor vessel surveillance program. In particular, the surveillance capsules contain dosimeters employing the following reactions.

Fe^{54} (n, P) Mn^{54}
Ni^{58} (n, P) Co^{58}
Cu^{63} (n, α) Co^{60}
Np^{237} (n, f) Cs^{137}
U^{238} (n, f) Cs^{137}

The activity of each dosimeter is determined using established ASTM procedures.[7-20] Measurements of the dosimeters activity is reported in BAW-2157.[2] Given the measured activity, the determination of the neutron flux proceeds with the calculation of the saturation activity from:

$$A=A_s\sum_{i=1}^{N}\left(\frac{P_i}{P}\right)(1-e^{-\lambda t_i})e^{-\lambda t_d}$$

where:

A	=	the measured activity corrected to end of power operation
A_s	=	saturation activity
P_i	=	core thermal power during irradiation period i
P	=	reference core thermal power or 3411 MWt
λ	=	decay constant for the radioactive nuclide
t_i	=	duration of the i^{th} period
t_d	=	decay time following i^{th} period

Using the saturation activity, the reaction rate, R, is determined from:

$$R = \frac{A_s}{N} Y$$

where Y is the yield and N is the number density and is calculated as:

$$N = \frac{N_o}{M} f$$

where:

N_o = Avogadro's number,

M = atomic weight of parent nuclide,

f = abundance of the parent nuclide.

As provided in Reference 12, the neutron fluence rate (flux) can be calculated as follows:

$$\phi = \frac{R}{\overline{\sigma}}$$

where $\overline{\sigma}$ is the spectrum averaged cross section. The spectrum averaged cross section for E > 1 Mev is calculated as:

$$\overline{\sigma} = \frac{\int_0^\infty \sigma(E)\phi(E)dE}{\int_{1Mev}^\infty \sigma(E)dE}$$

6.4 Results

Using the procedures in Section 6.3, the flux for E > 1 Mev can be calculated. The nuclear parameters for the dosimeters are provided in Table 6-1. The irradiation period, power ratio for the irradiation period and decay time are provided in Table 6-2. The decay time reflects the correction of the measured activity to reactor shutdown. Table 6-3 provides the measured activity, saturation activity and reaction rate for the dosimeters. Seabrook is the same class as a number of Westinghouse reactors.[21-25] In this case, class is defined by the location and dimensions of the core barrel, reactor vessel, and surveillance capsules. Spectrum average cross sections for this class of plants has been calculated by Westinghouse [21-25] and are provided in Table 6-4 for the dosimeters of interest. Using the measured reaction rate and the spectrum averaged cross section, the flux is calculated and provided in Table 6-5. With the flux and the Seabrook irradiation period of 333.37 effective-full-power-days or 0.91 effective-full-power-years the fluence of each dosimeter was calculated and is presented in Table 6-5. The average value of all dosimeters is 3.11×10^{18}. This value compares favorably to the other Westinghouse reactors [21-25] in the Seabrook class as presented in Table 6-6.

TABLE 6-1

Nuclear Parameters for Neutron Flux Monitors

Monitor Material	Reaction of Interest	Target Weight Fraction	Response Range	Product Half-Life	Fission Yield (%)
Copper	$Cu^{63}(n,\alpha)Co^{60}$	0.6917	E>4.7 MeV	5.272 years	
Iron	$Fe^{54}(n,p)Mn^{54}$	0.058	E>1.0 MeV	312.5 days	
Nickel	$Ni^{58}(n,p)Co^{58}$	0.6827	E>1.0 MeV	70.82 days	
Uranium-238*	$U^{238}(n,f)Cs^{137}$	1.0	E>0.4 MeV	30.17 years	6.0
Neptunium-237*	$Np^{237}(n,f)Cs^{137}$	1.0	E>0.08 MeV	30.17 years	6.5

*Denotes that monitor is cadmium shielded.

TABLE 6-2

Irradiation History of Neutron Sensors Contained in Capsule U

Irradiation Period	Pi	Pi/P	Irradiation Time (days)	Decay Time (days)
3/90	41	0.012	11	481
4/90	174	0.051	30	451
5/90	48	0.014	31	420
6/90	771	0.226	30	390
7/90	2060	0.604	31	359
8/90	2702	0.792	31	328
9/90	3217	0.943	30	298
10/90	2981	0.874	31	267
11/90	1569	0.460	30	237
12/90	3408	0.999	31	206
1/91	3408	0.999	31	175
2/91	2453	0.719	28	147
3/91	3138	0.920	31	116
4/91	2688	0.788	30	86
5/91	3408	0.999	31	55
6/91	2736	0.802	30	25
7/91	2159	0.633	25	0

P = 3411 MWt

TABLE 6-3

Measured Monitor Activities and Reaction Rates for Capsule U

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (reactions/sec-atom)
$Cu^{63}(n,\alpha)Co^{60}$			
Top	5.15×10^{4}	4.71×10^{5}	7.18×10^{-17}
Middle	4.67×10^{4}	4.27×10^{5}	6.51×10^{-17}
Bottom	4.60×10^{4}	4.20×10^{5}	6.40×10^{-17}
Average	4.81×10^{4}	4.39×10^{5}	6.69×10^{-17}
$Fe^{54}(n,p)Mn^{54}$			
Top	2.13×10^{6}	4.43×10^{6}	7.08×10^{-15}
Middle	1.92×10^{6}	3.99×10^{6}	6.37×10^{-15}
Bottom	1.87×10^{6}	3.88×10^{6}	6.20×10^{-15}
Average	1.97×10^{6}	4.10×10^{6}	6.55×10^{-15}
$Ni^{58}(n,p)Co^{58}$			
Top	5.39×10^{6}	6.80×10^{7}	9.71×10^{-15}
Middle	4.97×10^{6}	6.27×10^{7}	8.95×10^{-15}
Bottom	4.88×10^{6}	6.16×10^{7}	8.79×10^{-15}
Average	5.08×10^{6}	6.41×10^{7}	9.15×10^{-15}
$U^{238}(n,f)Cs^{137}$			
Middle	1.25×10^{5}	-	-
Corrected*	1.07×10^{5}	5.25×10^{6}	3.46×10^{-14}
$Np^{237}(n,f)Cs^{137}$			
Middle	-	-	-

*Corrected by 0.85 for U-235 fissions and Pu build-in.

TABLE 6-4

Spectrum Averaged Reaction Cross-Sections For Use In Fast Neutron Dosimetry Evaluations

Reaction	σ (barns)*
$Cu^{63}(n,\alpha)\ Co^{60}$	0.00070
$Fe^{54}(n,p)Mn^{54}$	0.0583
$Ni^{58}(n,p)Co^{58}$	0.0790
$U^{238}(n,f)Cs^{137}$	0.320
$Np^{237}(n,f)Cs^{137}$	3.30

* Values from References 21-25

TABLE 6-5

Results of Fast Neutron Dosimetry for Capsule U

Reaction	Measured Reaction Rate (reactions)	ϕ (E > 1.0 MeV) (n/cm^2·sec) Measured	ϕ (E > 1.0 MeV) (n/cm^2) Measured
$Cu^{63}(n,\alpha)Co^{60}$	6.69×10^{-17}	0.96×10^{11}	2.75×10^{18}
Fe^{54}(n, [illegible])Mn^{54}	6.55×10^{-15}	1.12×10^{11}	3.24×10^{18}
$Ni^{58}(n,p)Co^{58}$	9.15×10^{-15}	1.16×10^{11}	3.34×10^{18}
$U^{238}(n,f)Cs^{137}$	3.46×10^{-14}	1.08×10^{11}	3.11×10^{18}
$Np^{237}(n,f)Cs^{137}$	-	-	-
Average		1.23×10^{11}	3.11×10^{18}

TABLE 6-6

Comparison of Average Fluence for Capsule U to Other Westinghouse Reactors

U	IRRADIATION TIME (EFPY)	FLUENCE (n/cm^2)
Callaway Unit 1	1.05	3.27×10^{18}
Catawba Unit 1	0.79	3.08×10^{18}
Wolf Creek	1.08	3.39×10^{18}
Byron Unit 1	1.15	3.50×10^{18}
Braidwood Urit 1	1.10	3.79×10^{18}
Seabrook	0.91	3.11×10^{18}

7.0 REFERENCES

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14. ASTM Designation E263-88, Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Iron," in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

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APPENDIX A

Unirradiated Vessel

Plate and Weld

Data

TABLE A-1

Chemical Analysis of the Lower Shell Plates Used in the Core Region of the Seabrook Station Unit No. 1 Reactor Pressure Vessel

Element	Chemical Composition[a] (weight %)		
	Plate R1808-1	Plate R1808-2	Plate[b] R1808-3
C	.22	.22	.20
Mn	1.39	1.36	1.45
P	.005	.007	.007
S	.010	.012	.010
Si	.22	.21	.24
Ni	.58	.57	.57
Mo	.58	.56	.55
Cr	.04	.03	.06
Cu	.05	.05	.06
Al	.017	.021	.028
Co	.009	.009	.010
Pb	[c]	[c]	[c]
W	<.01	<.01	<.01
Ti	<.01	<.01	<.01
Zr	<.001	<.001	<.001
V	.004	.004	.003
Sn	.001	.002	.011
As	.002	.001	.006
Cb	<.01	<.01	<.01
N_2	.007	.008	.008
B	<.001	<.001	<.001

[a] Chemical Analysis by Combustion Engineering, Inc.
[b] Surveillance Program test plate.
[c] Not detected.

TABLE A-2

Chemical Analysis of the Intermediate Shell Plates
Used in the Core Region of the Seabrook Station
Unit No. 1 Reactor Pressure Vessel

Element	Chemical Composition[a] (weight %)		
	Plate R1806-1	Plate R1806-2	Plate R1806-3
C	.25	.22	.21
Mn	1.47	1.33	1.33
P	.012	.007	.007
S	.012	.009	.012
Si	.22	.21	.23
Ni	.64	.65	.65
Mo	.59	.59	.58
Cr	.08	.04	.03
Cu	.04	.05	.07
Al	.024	.018	.027
Co	.014	.012	.012
Pb	[b]	[b]	[b]
W	.02	.01	.02
Ti	<.01	<.01	<.01
Zr	.001	.001	.001
V	.006	.004	.004
Sn	.003	.004	.004
As	.006	.007	.07
Cb	<.01	<.01	<.01
N_2	.010	.008	.010
B	<.001	<.001	<.001

[a] Chemical Analysis by Combustion Engineering, Inc.
[b] Not detected.

TABLE A-3

Chemical Analysis of the Weld Metal Used in the Core Region of the Seabrook Station Unit No. 1 Reactor Vessel

	Intermediate Shell Longitudinal Seams (101-124 A, B, & C)[a]							Lower Shell Longitudinal Seams (101-142 A, B, & C)[a]							Intermediate To Lower Shell Girth Weld Seam (101-171)[a]						
	Chemical Composition (weight %)																				
Element	C	Mn	P	S	Si	Ni	Mo	Cr	Cu	Al	Co	Pb	W	Ti	Zr	V	Sn	As	Cb	N_2	B
Wire/Flux Test Weld Sample[b]	.13	1.24	.008	.009	.12	.02	.50	.01	.07	--	--	--	--	--	--	.004	--	--	--	.009	--
CE Weld Test Block "C"[b]	.15	1.35	.007	.008	.14	.10	.53	.03	.02	<.001	.008	<.001	<.01	<.01	<.001	.004	.003	.001	<.01	.01	<.001
W Weld Test Block "D"[c]	.14	1.21	.010	.005	.14	.05	.50	.025	.02	.004	.005	<.001	<.002	<.002	<.002	.002	.004	.006	<.002	.008	<.001

[a] Weld Wire Heat No. 4P6052, Linde 0091 flux, Lot No. 0145.
[b] Combustion Engineering, Inc. Certification Reports.
[c] Westinghouse Analysis.

TABLE A-4

Heat Treatment History of the Seabrook Station Unit No. 1 Reactor Pressure Vessel Core Region Shell Plates and Weld Seams

Material	Temperature (°F)	Time (hr)	Cooling
Lower Shell Plates R1808-1-2-3	Austenitizing: 1600 ± 25 Tempered: 1225 ± 25 Stress Relief: 1150 ± 50	4 4 16	Water-quenched Air-cooled Furnace-cooled
Intermediate Shell Plates R1806-1-2-3	Austenitizing: 1600 ± 25 Tempered: 1225 ± 25 Stress Relief: 1150 ± 50	4 4 16.5	Water-quenched Air-cooled Furnace-cooled
Lower Shell Plate Longitudinal Seam Welds	Stress Relief: 1150 ± 50	16	Furnace-cooled
Inter. Shell Plate Longitudinal Seam Welds	Stress Relief: 1150 ± 50	16.5	Furnace-cooled
Intermediate to Lower Shell Girth Seam Weld	Stress Relief: 1150 ± 50	12.50	Furnace-cooled
Surveillance Program Test Material			
Surveillance Program Test Plate R1808-3	Austenitizing: 1600 ± 25 Tempered: 1225 ± 25 Stress Relief:[a] 1150 ± 50	4 4 16.25	Water-quenched Air-cooled Furnace-cooled
Weldment	Stress Relief:[a] 1150 ± 50	17	Furnace-cooled

[a] The stress relief heat treatment received by the surveillance test plate and weldment have been simulated.

TABLE A-5

T_{NDT}, RT_{NDT}, and Upper Shelf Energy for the Seabrook Station Unit No. 1 Reactor Pressure Vessel Core Region Shell Plates and Weld Metal

	T_{NDT} [a]		RT_{NDT} [a]		Upper Shelf [a] Energy	
Material	(°C)	(°F)	(°C)	(°F)	(J)	(ft lb)
Lower Shell Plates:						
R1808-1	-34	-30	4	40	106	78
R1808-2	-29	-20	-12	10	104	77
R1808-3[b]	29	-20	4	40	106	78
Intermediate Shell Plates:						
R1806-1	-34	-30	4	40	111	82
R1806-2	-34	-30	-18	0	138	102
R1806-3	-40	-40	-12	10	156	115
Intermediate Shell and Lower Shell Longitudinal Seams-Weld Metal[c][d] and the Intermediate to Lower Shell Girth Seam-Weld Metal[c][d]	-51	-60	-51	-60	212	156

[a] Data by Combustion Engineering, Inc.

[b] Surveillance Program test plate.

[c] Weld Metal Heat No. 4P6052, Flux Type 0091, and Lot No. 0145.

[d] Combustion Engineering Surveillance Weld Test Plate "C `ertification Report.

APPENDIX B

BAW-2157

Test Results of Capsule U

R77\253

BAW-2157
May 1992

TEST RESULTS OF CAPSULE U
PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE
NEW HAMPSHIRE YANKEE DIVISION
SEABROOK STATION UNIT NO. 1

-- Reactor Vessel Material Surveillance Program --

by

A. L. Lowe, Jr., PE
R. E. Napolitano
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B&W Document No. 77-2157-00
(See Section 6 for document signatures)

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SUMMARY

This report describes the results of the testing of the specimens from the first capsule (Capsule U) of the Public Service Company of New Hampshire Seabrook Station Unit No. 1 reactor vessel surveillance program. The objective of the program is to monitor the effects of neutron irradiation on the tensile and fracture toughness properties of the reactor vessel materials by the testing and evaluation of tension and Charpy impact specimens. The program was designed in accordance with the requirements of 10CFR50, Appendix H, and ASTM Specification E185-79.

The results of the tension tests and the Charpy impact test results indicated that the materials exhibited normal behavior relative to the estimated neutron fluence exposure.

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1. INTRODUCTION

This report describes the results of the testing of the specimens from the first capsule (Capsule U) of the Public Service Company of New Hampshire Seabrook Station Unit No. 1 reactor vessel material surveillance program (RVSP). The capsule was removed and evaluated after being irradiated in the reactor as part of the Reactor Vessel Materials Surveillance Program. The objective of the program is to monitor the effects of neutron irradiation on the tensile and impact properties of reactor pressure vessel materials under actual operating conditions. The surveillance program for Seabrook Station Unit 1 was designed and furnished by Westinghouse Electric Corporation (W). The program was planned to monitor the effects of neutron irradiation on the reactor vessel materials for the 40-year design life of the reactor pressure vessel. The surveillance program for Seabrook Unit 1 was designed in accordance with E185-79[1] and thus is in compliance with 10CFR50, Appendixes G[2] and H[3].

2. BACKGROUND

The ability of the reactor pressure vessel to resist fracture is the primary factor in ensuring the safety of the primary system in light water-cooled reactors. The beltline region of the reactor vessel is the most critical region of the vessel because it is exposed to neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of such low-alloy ferritic steels as SA533, Grade B, used in the fabrication of the Seabrook Station Unit 1 reactor vessel, are well characterized and documented in the literature. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. The most significant mechanical property change in reactor pressure vessel steels is the increase in temperature for the transition from brittle to ductile fracture accompanied by a reduction in the Charpy upper-shelf energy value.

Appendix G to 10CFR50, "Fracture Toughness Requirements,"[2] specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors, and provides specific guidelines for determining the pressure-temperature limitations on operation of the RCPB. The toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Although the requirements of Appendix G to 10CFR50 became effective on August 13, 1973, the requirements are applicable to all boiling and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

Appendix H to 10CFR50, "Reactor Vessel Materials Surveillance Program Requirements,"[3] defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit determination of the conditions under which the vessel can be operated with adequate safety margins against fracture throughout its service life.

A method for guarding against brittle fracture in reactor pressure vessels is described in Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."[4] This method utilizes fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined as the greater of the drop weight nil-ductility transition temperature or the temperature that is 60F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve), which appears in Appendix G of ASME Section III. The K_{IR} curve is a lower bound of dynamic, static, and crack arrest fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined using these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits of a nuclear power plant, can be adjusted to account for the effects of radiation on the properties of the reactor vessel materials. The radiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which a surveillance capsule containing prepared specimens of the reactor vessel materials is periodically removed from the operating nuclear reactor and the specimens are tested. The increase in the Charpy V-notch 30 ft-lb temperature is added to the original RT_{NDT} to adjust it for radiation embrittlement. This adjusted RT_{NDT} is used to index the material to the K_{IR} curve which, in turn, is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

3. POST-IRRADIATION TESTING

3.1. Visual Examination and Inventory

All specimens were visually examined and no signs of abnormalities were found. The contents of the capsule were inventoried and found to be consistent with the surveillance program report inventory. There was no evidence of rust or of the penetration of reactor coolant into the capsule. The compact fracture toughness specimens and three-point bend bar were stored for future disposition.

3.2. Thermal Monitors

Surveillance Capsule U contained temperature monitor sets in each of three holder blocks. The holder blocks each contained one thermal monitor. The monitors located at the top and bottom of the capsule are designed to melt at 579F and the monitor located at the midpoint of the capsule is designed to melt at 590F. The holder blocks were radiographed for evaluation. None of the three sets of thermal monitors exhibited any signs of melting. From these data, it was concluded that the irradiated specimens had been exposed to a maximum temperature of less than 579F during the reactor vessel operating period. This is not significantly greater than the nominal inlet temperature of 558F, and is considered acceptable for inclusion of the data in the general pool of irradiated surveillance data. There appeared to be no significant signs of a temperature gradient along the capsule length.

3.3. Tension Test Results

The results of the postirradiation tension tests are presented in Table 3-1. Tests were performed on specimens at room temperature, 300, and 550F. They were tested on a computer controlled 55,000-lb load capacity MTS servohydraulic test machine at a crosshead speed of 0.005 inch per minute to yield point and thereafter 0.040 inch per minute. A 4-pole extension device with a strain gaged extensometer was used to determine the 0.2% yield point. Test conditions were in accordance with the applicable requirements of ASTM A370-77.[5] For each material type and/or condition, specimens were tested at room temperature, 300, and 550F to correspond to the unirradiated material test temperatures. The tension-compression load cell used had a certified accuracy of better than +0.5% of full scale (25,000 lb). Photographs of the tension test specimen fractured surfaces are presented in Figures 3-1 through 3-3.

In general, the ultimate strength and yield strength of the material increased with a corresponding slight decrease in ductility as compared to the unirradiated values; both effects were the result of neutron radiation. The type of behavior observed and the degree to which the material properties changed is within the range of changes to be expected for the radiation environment to which the specimens were exposed.

The stress-strain curves from the irradiated tension tests are presented in Figures 3-4 through 3-12.

3.4. Charpy V-Notch Impact Test Results

The test results from the irradiated Charpy V-notch specimens of the reactor vessel beltline material are presented in Tables 3-2 through 3-5 and Figures 3-13 through 3-16. Photographs of the Charpy specimen fracture surfaces are presented in Figures 3-17 through 3-20. The Charpy V-notch impact tests were conducted in accordance with the requirements of ASTM E23-88[6] on an Satec Sl-1K impact tester certified to meet Watertown standards.

The data show that the materials exhibited a sensitivity to irradiation within the values to be expected from their chemical composition and the fluence to which they were exposed.

Table 3-1. Tensile Properties Irradiated Base Metal and Weld Metal from Capsule U

Specimen No.	Test Temp, F	Strength, psi		Fracture Properties			Elongation, %		Reduction in Area, %
		Yield	Ultimate	Load, lbs	Stress, psi	Strength, psi	Uniform	Total	
Base Metal, R1808-3, Longitudinal									
KL2	70	73,600	94,200	2901	162,000	59,100	9.8	23.7	63.5
KL3	300	67,600	86,600	2877	169,500	58,600	8.3	20.5	65.4
KL1	550	67,300	91,100	3009	161,600	61,300	7.8	19.4	62.1
Base Metal, R1808-3, Transverse									
KT2	70	73,300	94,100	3374	150,400	68,700	9.5	21.4	54.3
KT1	300	67,200	85,700	2968	137,100	60,500	7.8	17.8	55.9
KT3	550	66,400	91,300	3643	144,800	74,800	7.9	15.9	48.3
Weld Metal, Transverse									
KW3	70	76,000	90,000	2556	189,600	52,100	8.9	23.7	72.5
KW1	300	70,500	83,700	2565	184,600	52,300	7.5	21.2	71.7
KW2	550	69,400	87,700	2645	175,600	53,900	4.9	18.0	69.3

Table 3-2. Charpy Impact Results for Irradiated Base Metal, Longitudinal (LT) Orientation, from Capsule U

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion Inch	Shear Fracture %
KL13	-40	11.0	0.009	0
KL2	0	28.0	0.025	25
KL15	20	34.0	0.031	25
KL6	40	62.0	0.047	50
KL14	40	52.5	0.042	30
KL1	70	102.5	0.071	85
KL4	70	60.0	0.049	50
KL7	90	88.0	0.068	80
KL12	100	117.5*	0.077	100
KL5	125	122.0*	0.085	100
KL9	125	120.0*	0.083	100
KL3	150	117.0*	0.084	100
KL11	225	112.5*	0.076	100
KL10	325	112.0	0.085	100
KL8	550	121.5	0.078	100

*Values used to determine upper-shelf energy value per ASTM E185.[7]

Table 3-3. Charpy Impact Results for Irradiated Base Metal, Transverse (TL) Orientation, from Capsule U

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion Inch	Shear Fracture %
KT1	-40	12.0	0.009	0
KT5	0	15.0	0.014	20
KT9	20	19.0	0.016	20
KT4	40	32.5	0.028	30
KT7	70	45.0	0.036	45
KT11	100	59.5	0.045	65
KT14	120	58.0	0.051	60
KT8	125	71.5*	0.059	100
KT6	150	71.0*	0.059	100
KT2	175	73.5*	0.062	100
KT12	200	69.5*	0.061	100
KT15	200	67.0*	0.062	100
KT13	225	81.5*	0.065	100
KT3	325	79.5	0.071	100
KT10	550	70.0	0.062	100

*Values used to determine upper-shelf energy value per ASTM E185.[7]

Table 3-4. Charpy Impact Results for Irradiated Heat-Affected Zone Metal, from Capsule U

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion Inch	Shear Fracture %
KH1	-80	32.0	0.021	45
KH4	-40	89.0	0.056	60
KH15	-40	79.0	0.052	60
KH7	-20	62.5	0.042	70
KH8	0	52.0	0.040	50
KH6	20	71.0	0.051	70
KH14	40	121.5*	0.074	100
KH12	70	130.0*	0.082	100
KH3	100	147.5*	0.085	100
KH9	150	142.5*	0.085	100
KH5	175	111.0*	0.080	100
KH10	200	138.0	0.079	100
KH13	225	118.5	0.078	100
KH11	325	161.5	0.080	100
KH2	550	132.0	0.082	100

*Values used to determine upper-shelf energy value per ASTM E185.[7]

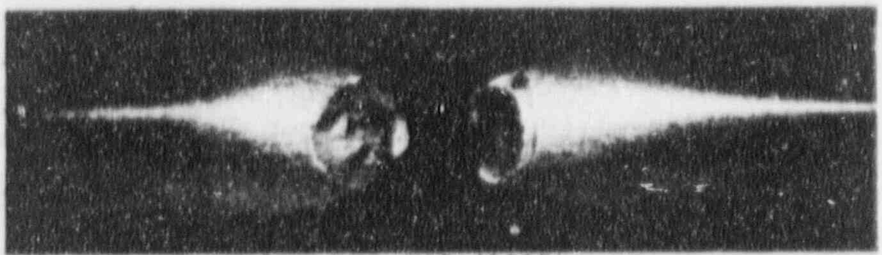
Table 3-5. Charpy Impact Test Results for Irradiated Weld Metal, Transverse (TL) Orientation, from Capsule U

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion Inch	Shear Fracture %
KW10	-80	7.0	0.004	10
KW11	-50	35.0	0.026	30
KW2	-40	48.0	0.036	40
KW5	-20	30.5	0.025	30
KW9	-20	69.0	0.051	55
KW1	0	101.5*	0.073	100
KW6	0	93.5	0.065	80
KW7	40	123.0	0.083	90
KW13	70	---	0.092	100
KW15	70	133.0*	0.090	100
KW12	100	138.0*	0.091	100
KW8	150	143.0*	0.089	100
KW14	225	142.5	0.087	100
KW3	325	157.0	0.088	100
KW4	550	153.0	0.085	100

*Values used to determine upper-shelf energy value per ASTM E185.[7]

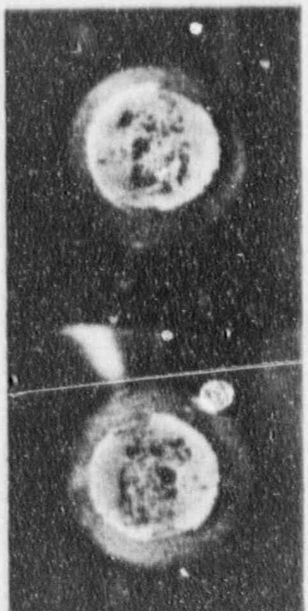
Figure 3-1. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Base Metal, Longitudinal Orientation

Specimen KL2 (70F)

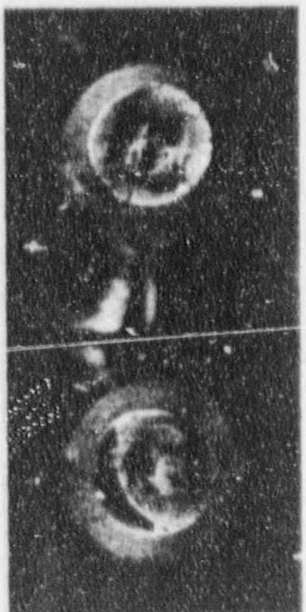


Specimen KL3 (300F)

Specimen KL1 (550F)



Specimen KL2 (70F)



Specimen KL3 (300F)

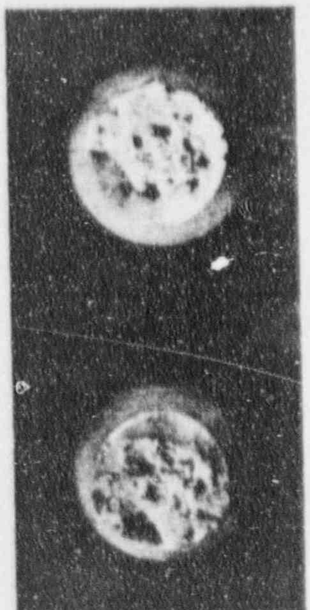
Specimen KL1 (550F)

Figure 3-2. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Base Metal, Transverse Orientation

Specimen KT2 (70F)

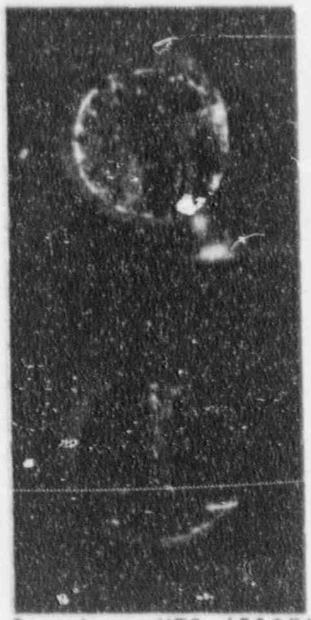
Specimen KT1 (300F)

Specimen KT3 (550F)



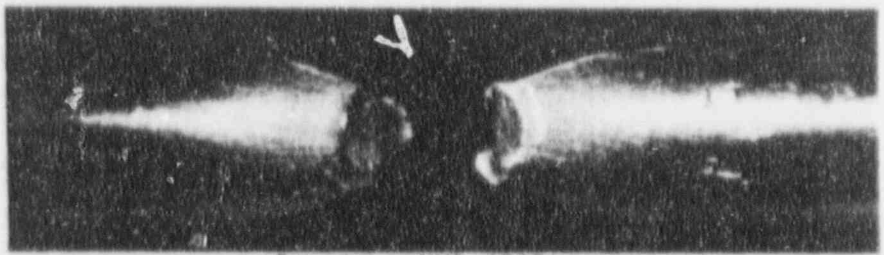
Specimen KT2 (70F)

Specimen KT1 (300F)

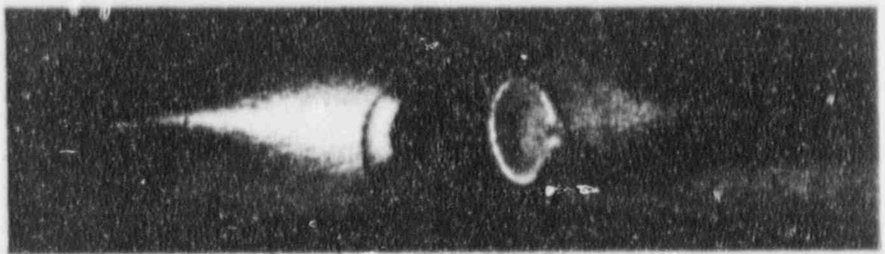


Specimen KT3 (550F)

Figure 3-3. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Weld Metal, Transverse Orientation

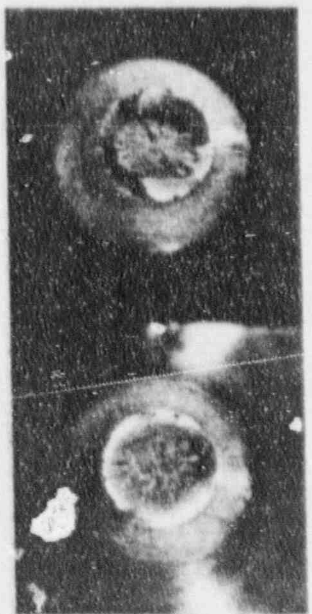


Specimen KW3 (70F)



Specimen KW1 (300F)

Specimen KW2 (550F)



Specimen KW3 (70F)

Specimen KW1 (300F)

Specimen KW2 (550F)

Figure 3-4. Tension Test Stress-Strain Curve for Base Metal Plate R1808-3, Specimen No. KL2, Tested at 70F

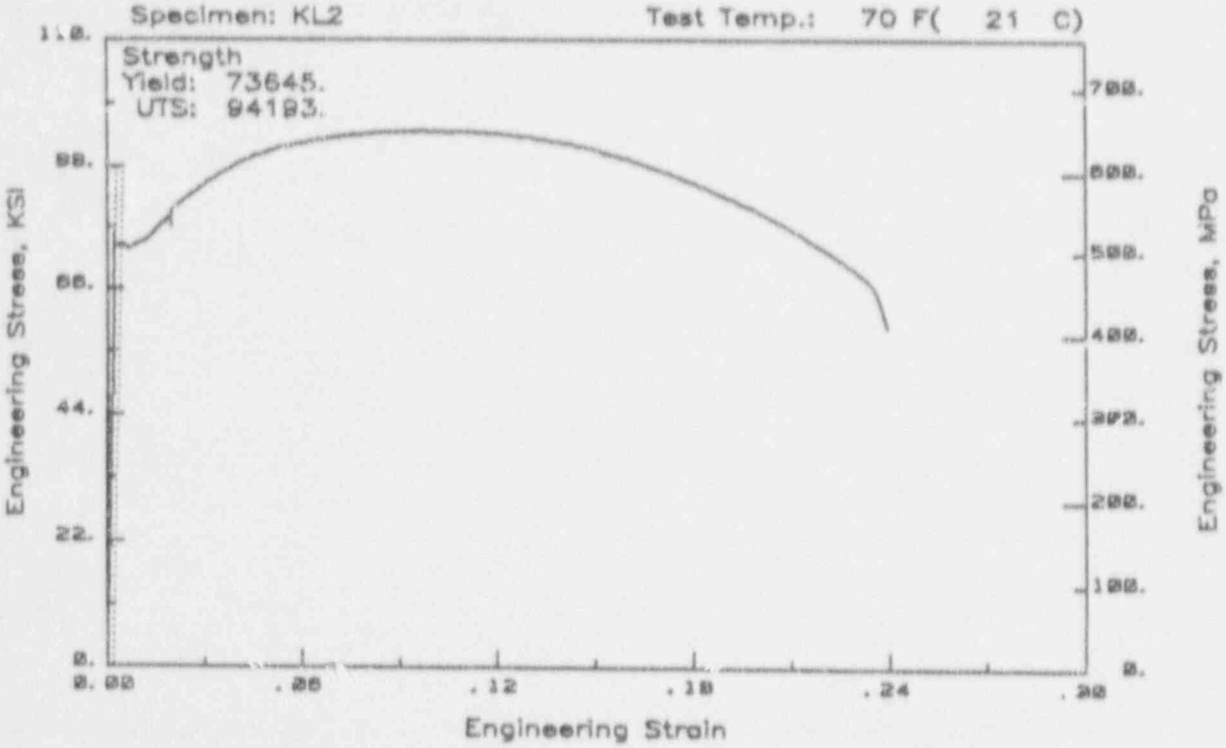


Figure 3-5. Tension Test Stress-Strain Curve for Base Metal Plate R1808-3, Specimen No. KL3, Tested at 300F

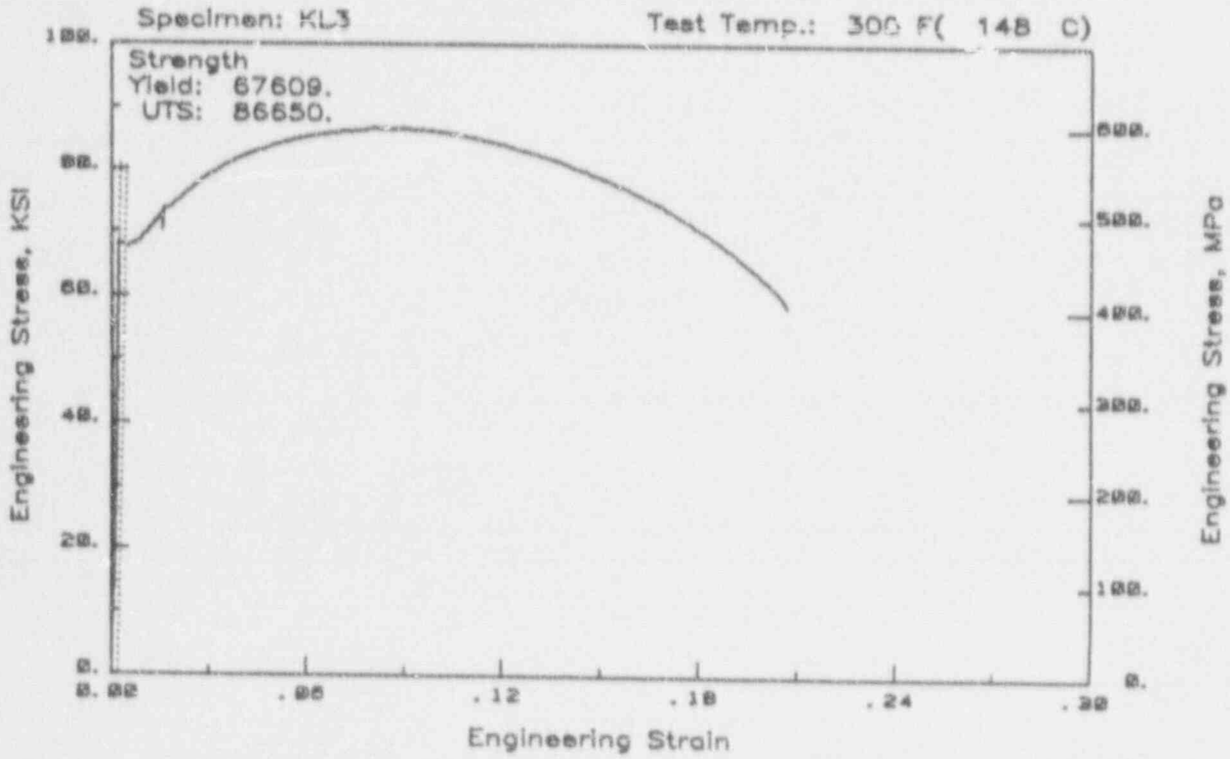


Figure 3-6. Tension Test Stress-Strain Curve for Base Metal Plate R18C8-3, Specimen No. KL1, Tested at 550F

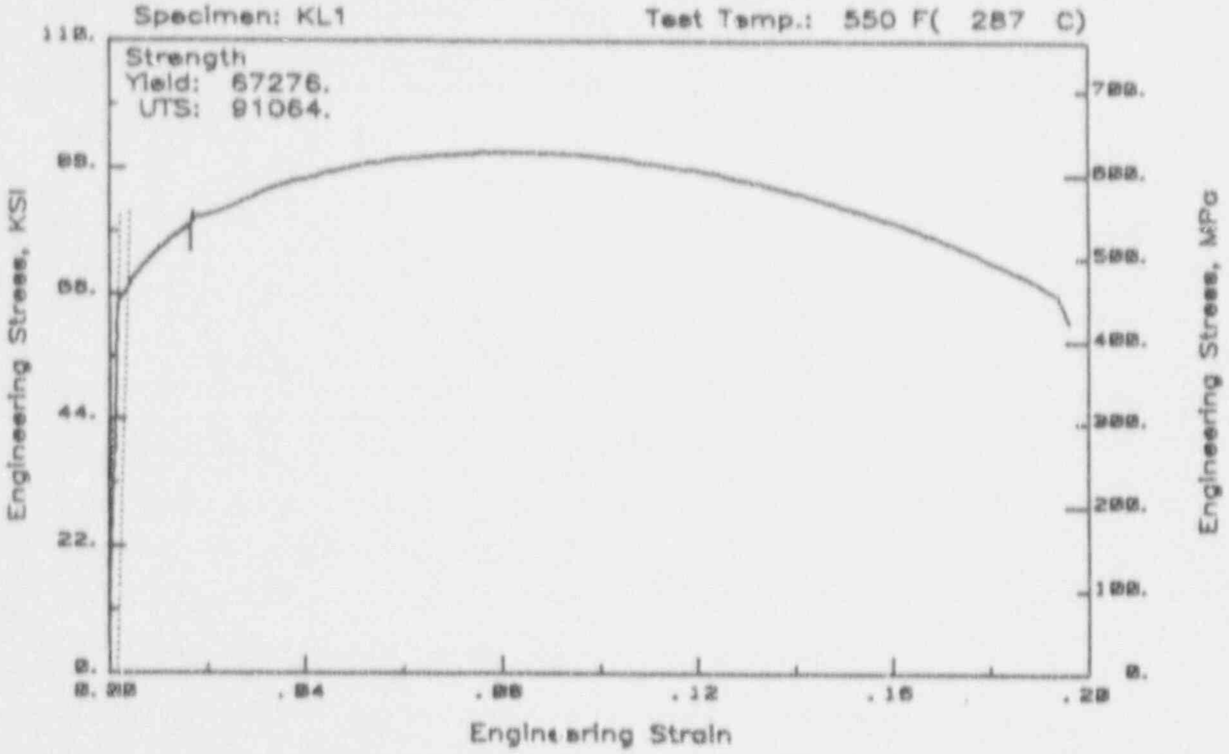


Figure 3-7. Tension Test Stress-Strain Curve for Base Metal Plate R1808-3, Specimen No. KT2, Tested at 70F

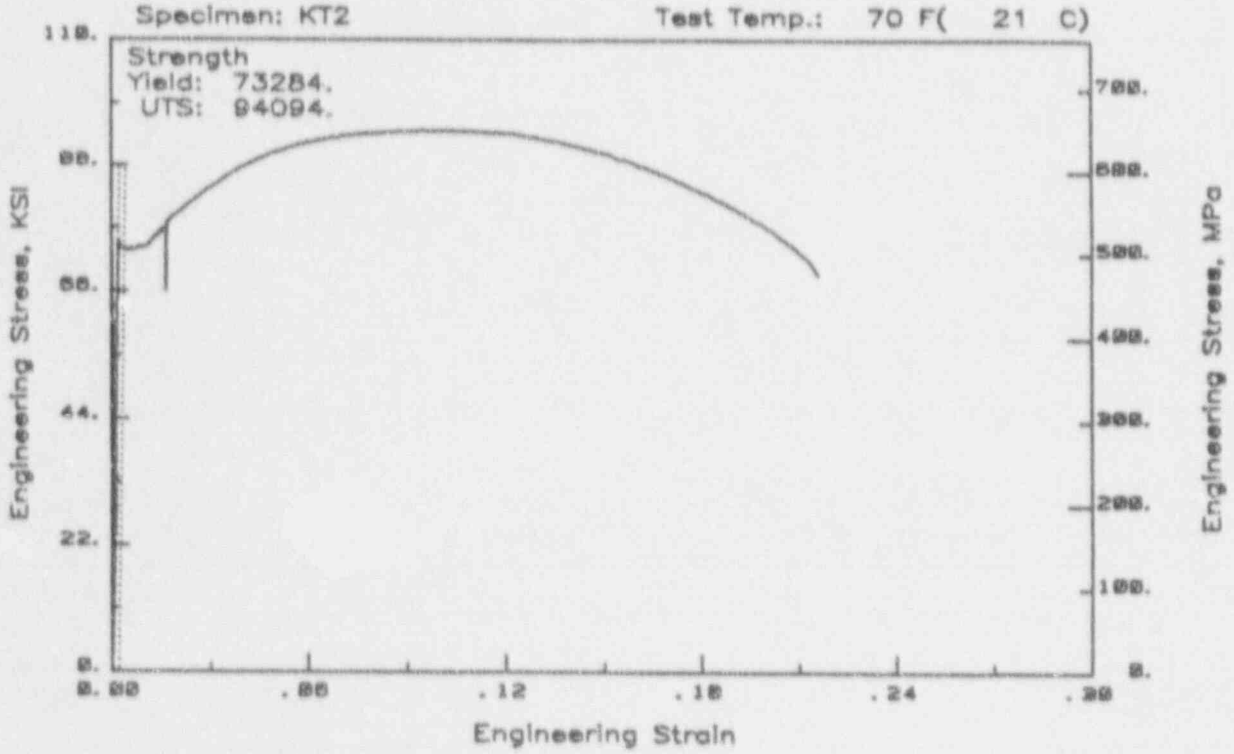


Figure 3-8. Tension Test Stress-Strain Curve for Base Metal Plate R1808-3, Specimen No. KT1, Tested at 300F

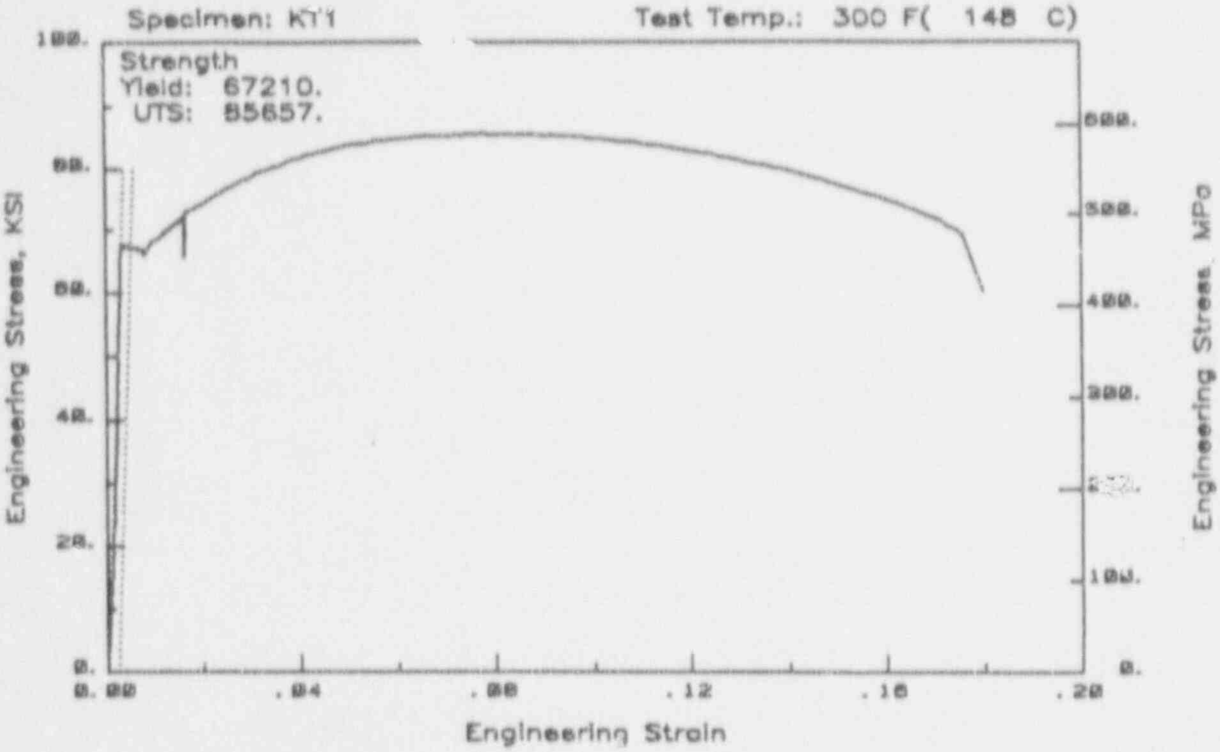


Figure 3-9. Tension Test Stress-Strain Curve for Base Metal Plate R1808-3, Specimen No. KT3, Tested at 550F

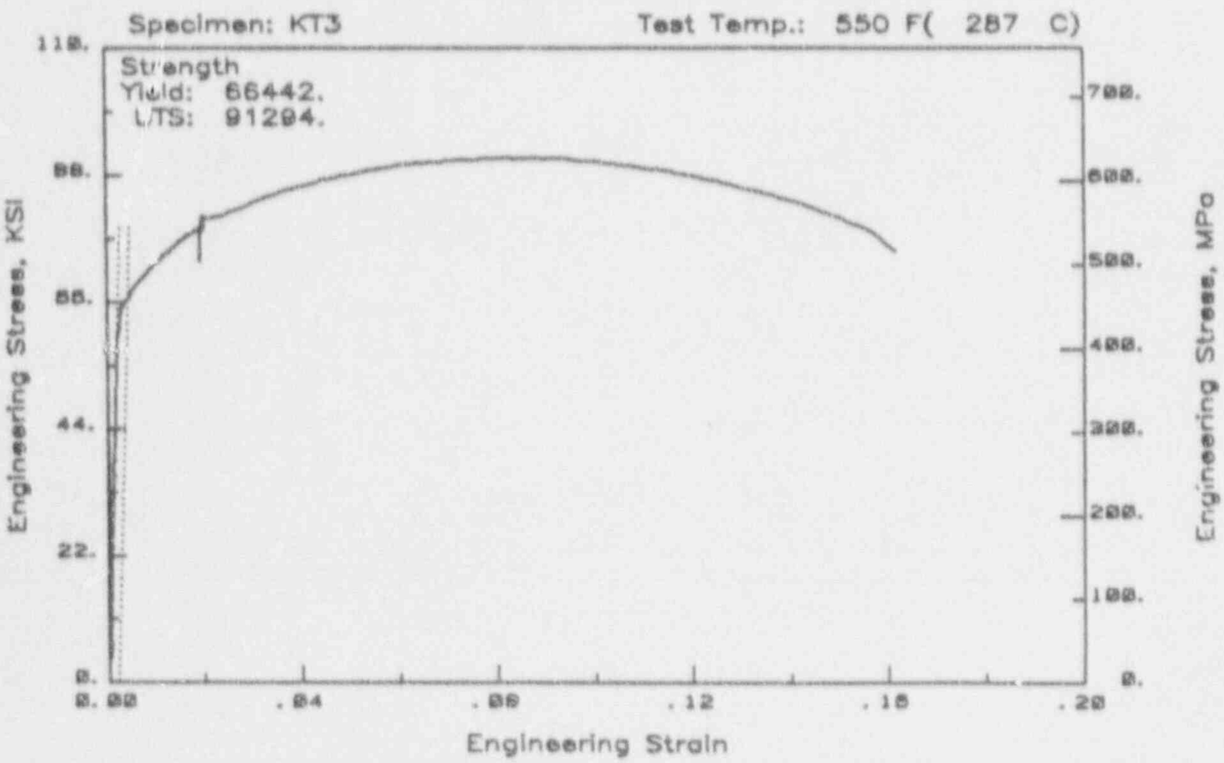


Figure 3-10. Tension Test Stress-Strain Curve for Weld Metal, Specimen No. KW3, Tested at 70F

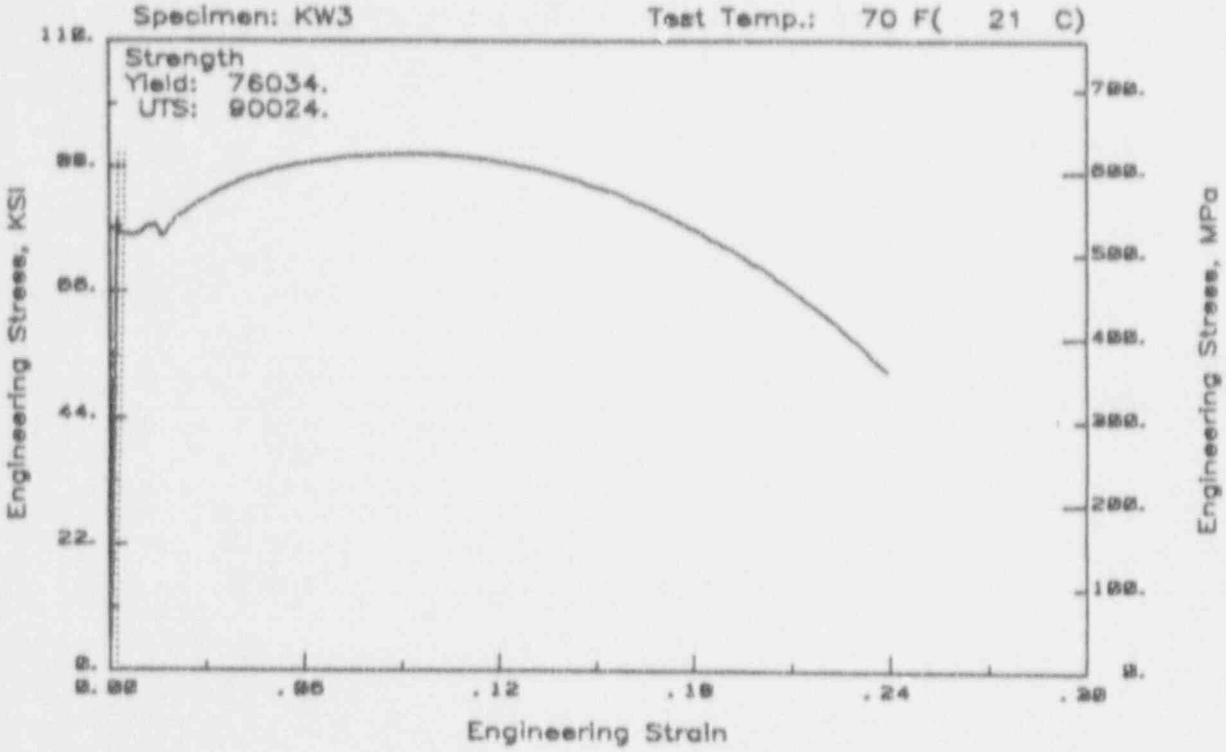


Figure 3-11. Tension Test Stress-Strain Curve for Weld Metal, Specimen No. KW1, Tested at 300F

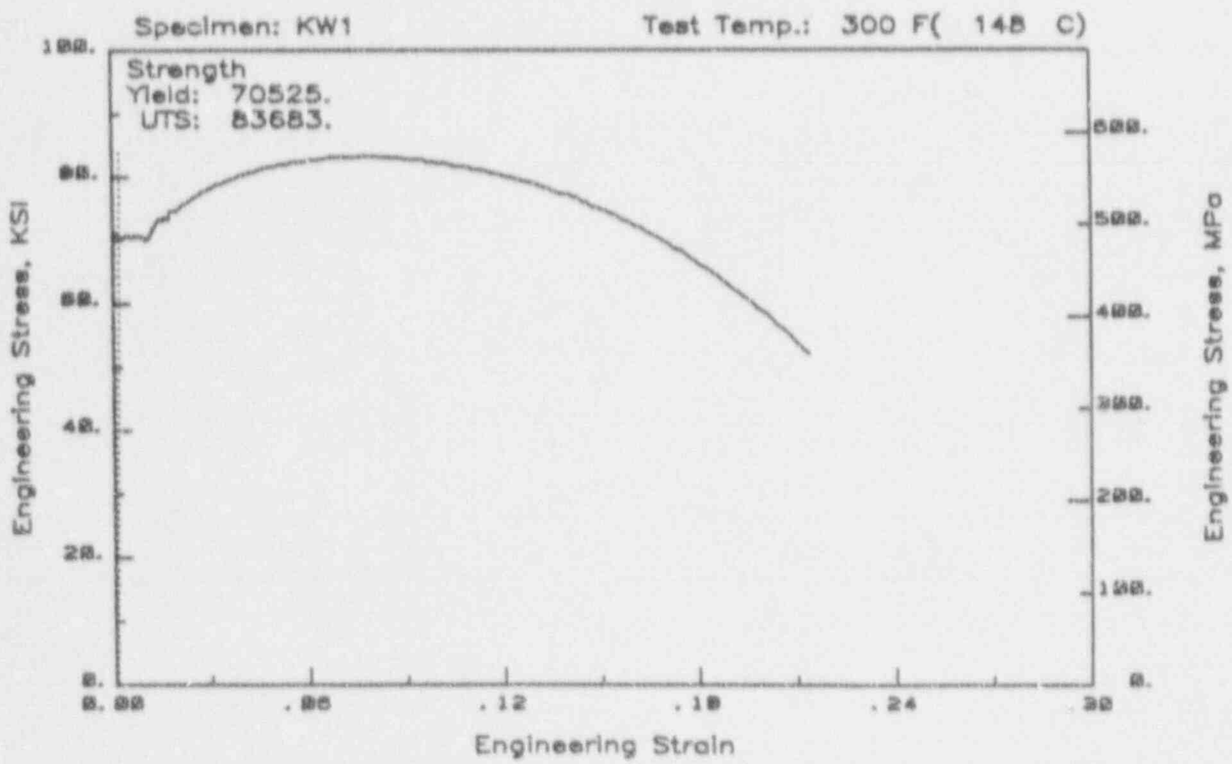


Figure 3-12. Tension Test Stress-Strain Curve for Weld Metal, Specimen No. Kw2, Tested at 550F

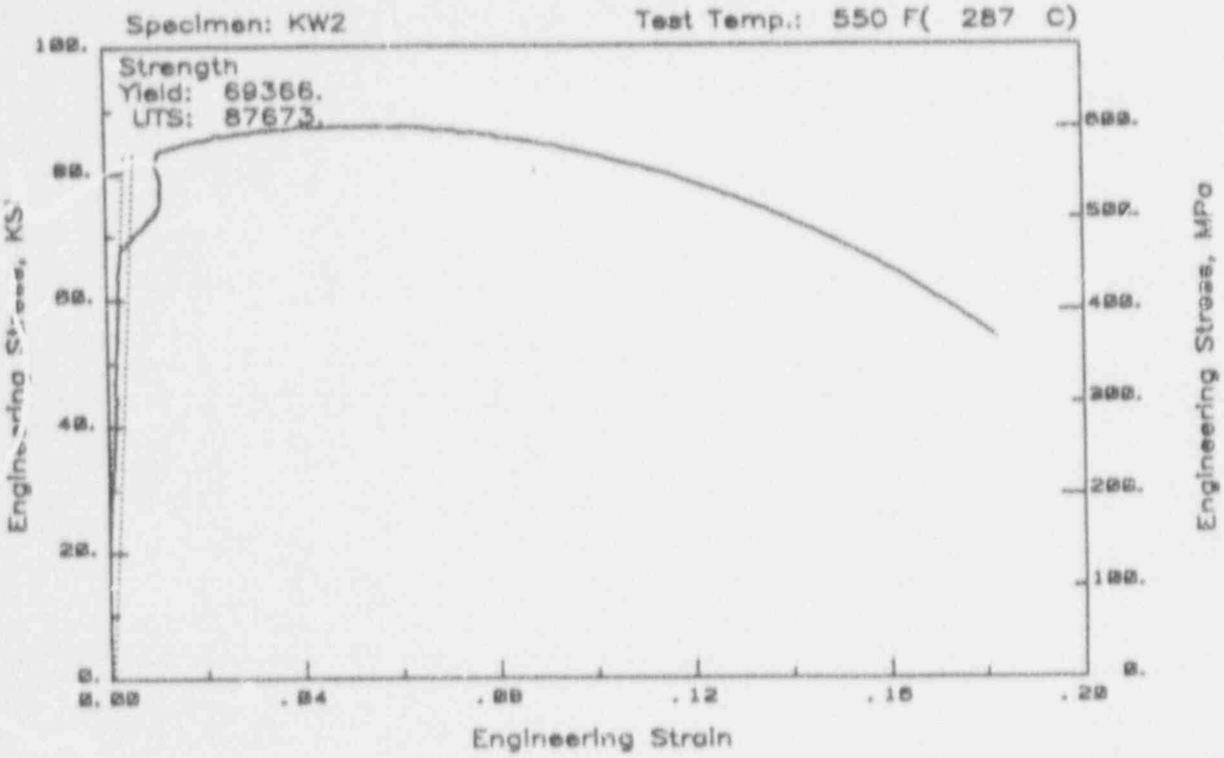


Figure 3-13. Charpy Impact Data for Irradiated Plate Material, R1808-3, Longitudinal Orientation

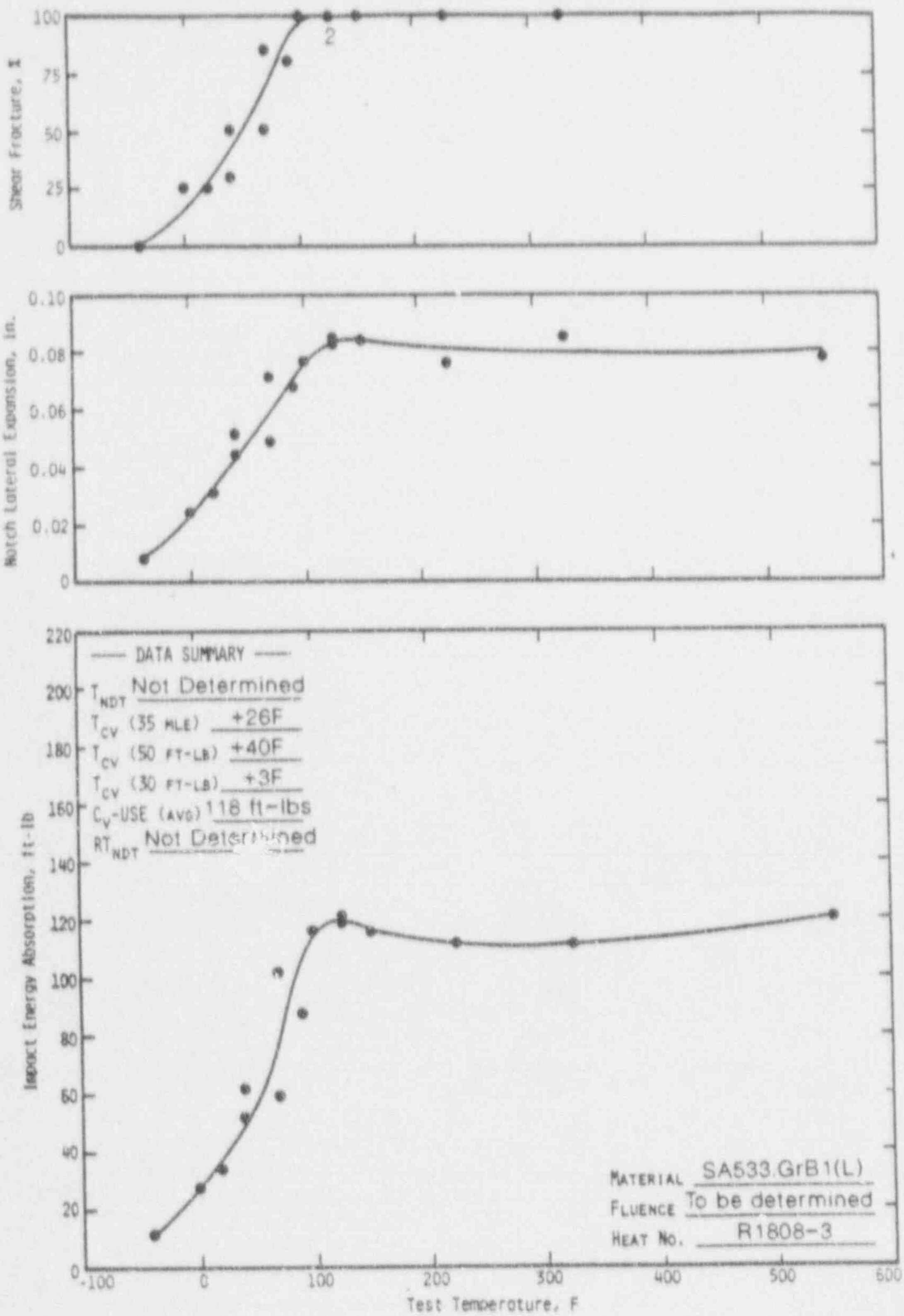


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Figure 3-14. Charpy Impact Data for Irradiated Plate Material, R1808-3, Transverse Orientation

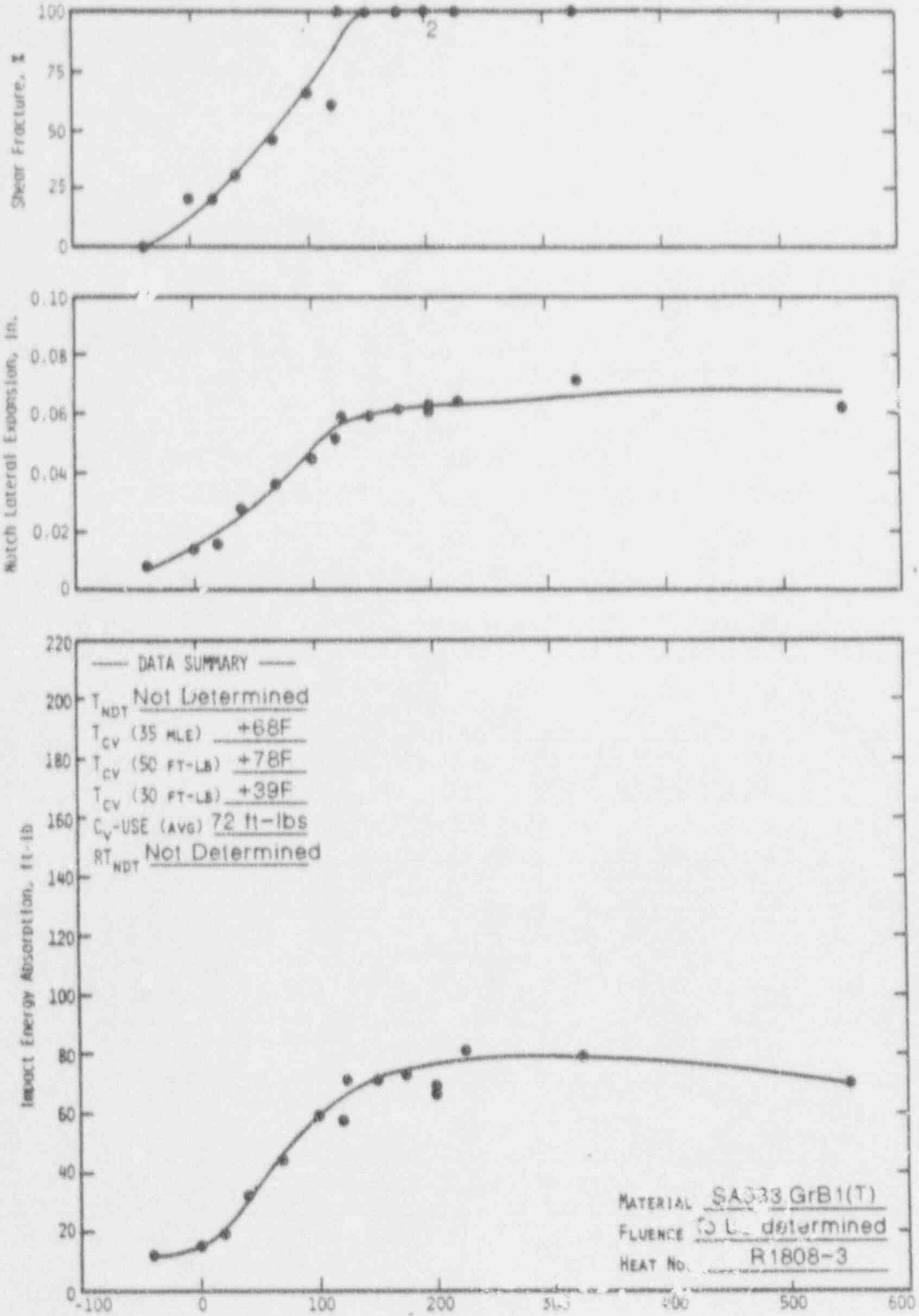


Figure 3-15. Charpy Impact Data for Irradiated Plate Material, R1808-3, Heat-Affected Zone

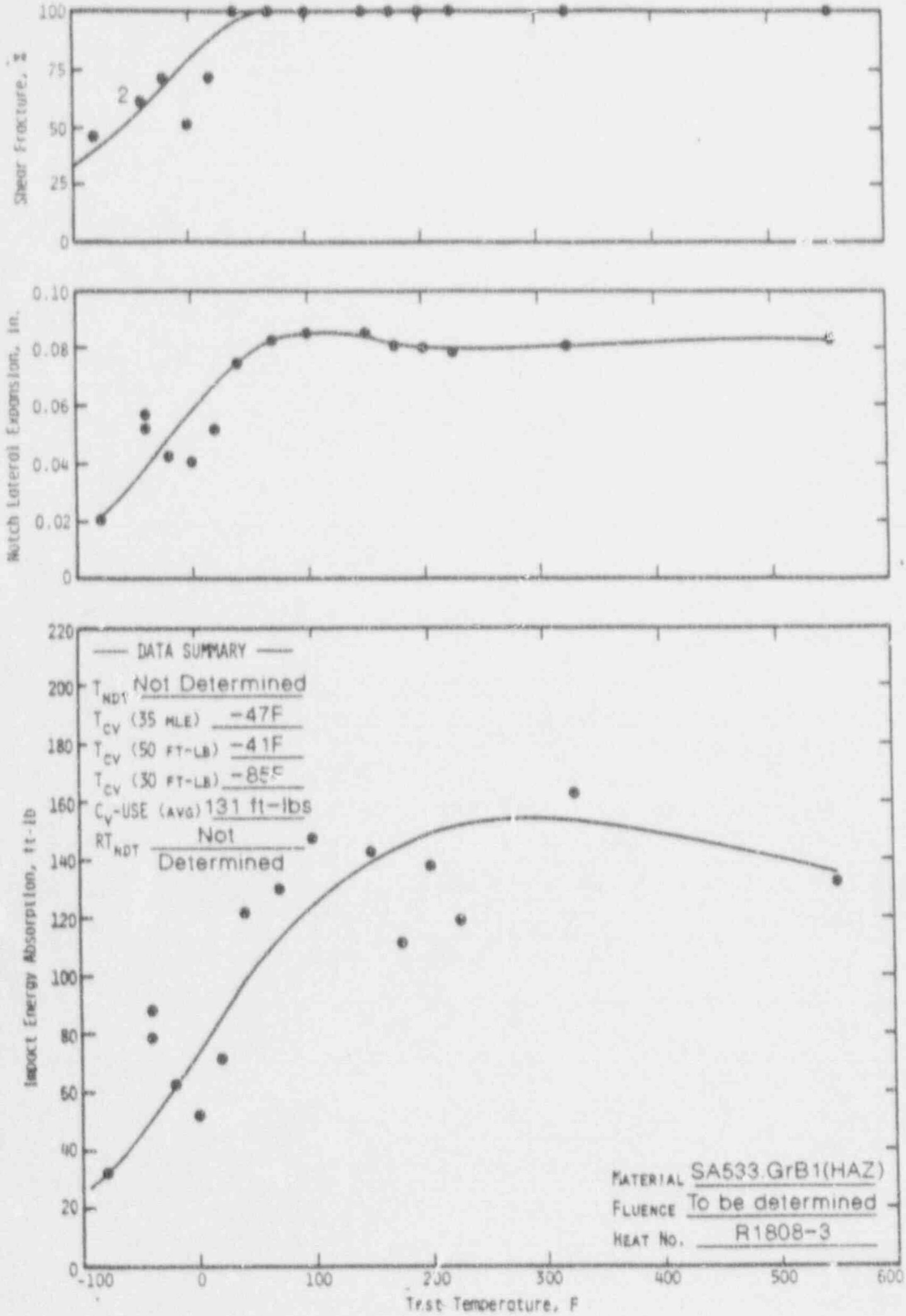


Figure 3-16. Charpy Impact Data for Irradiated Weld Metal

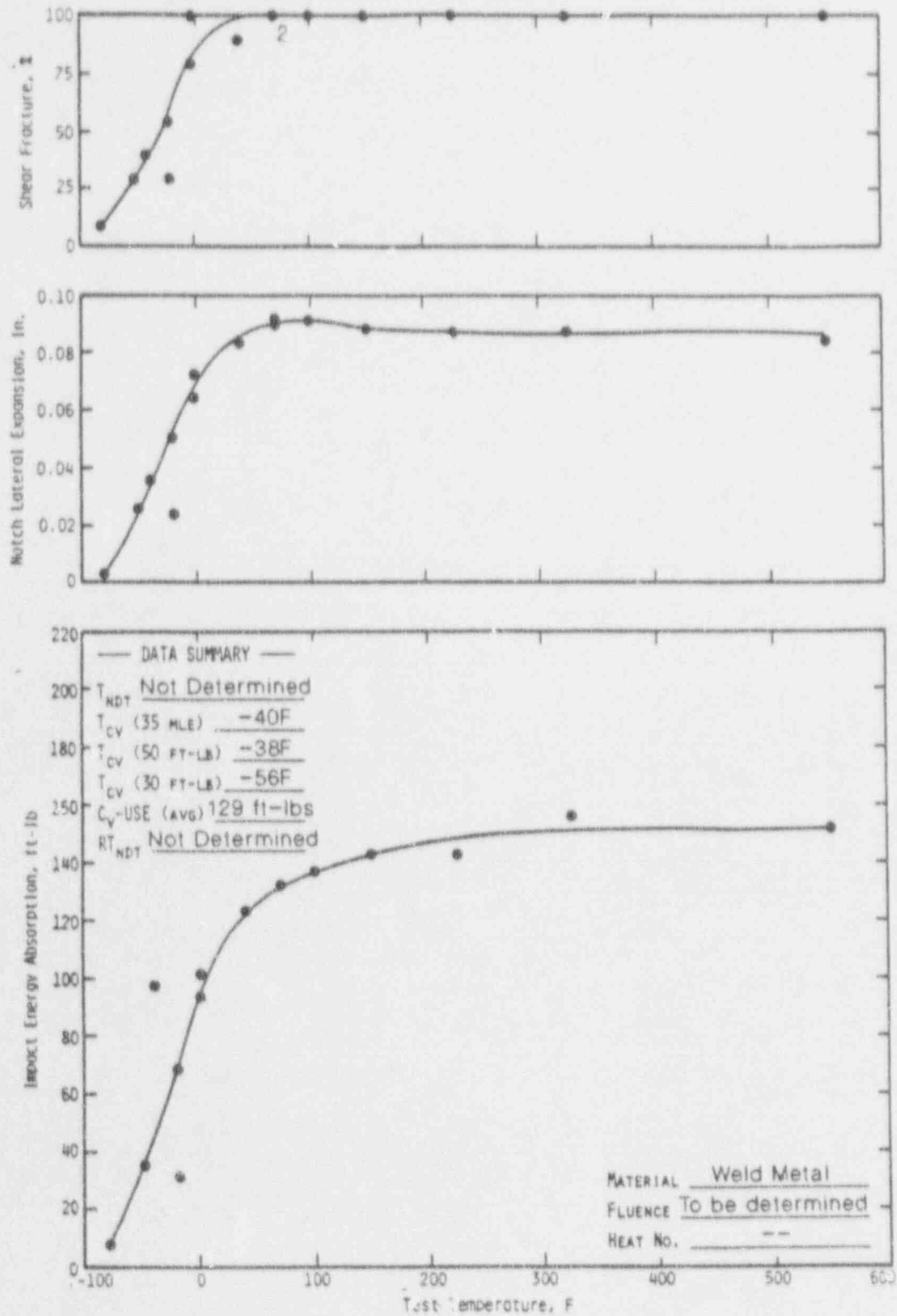
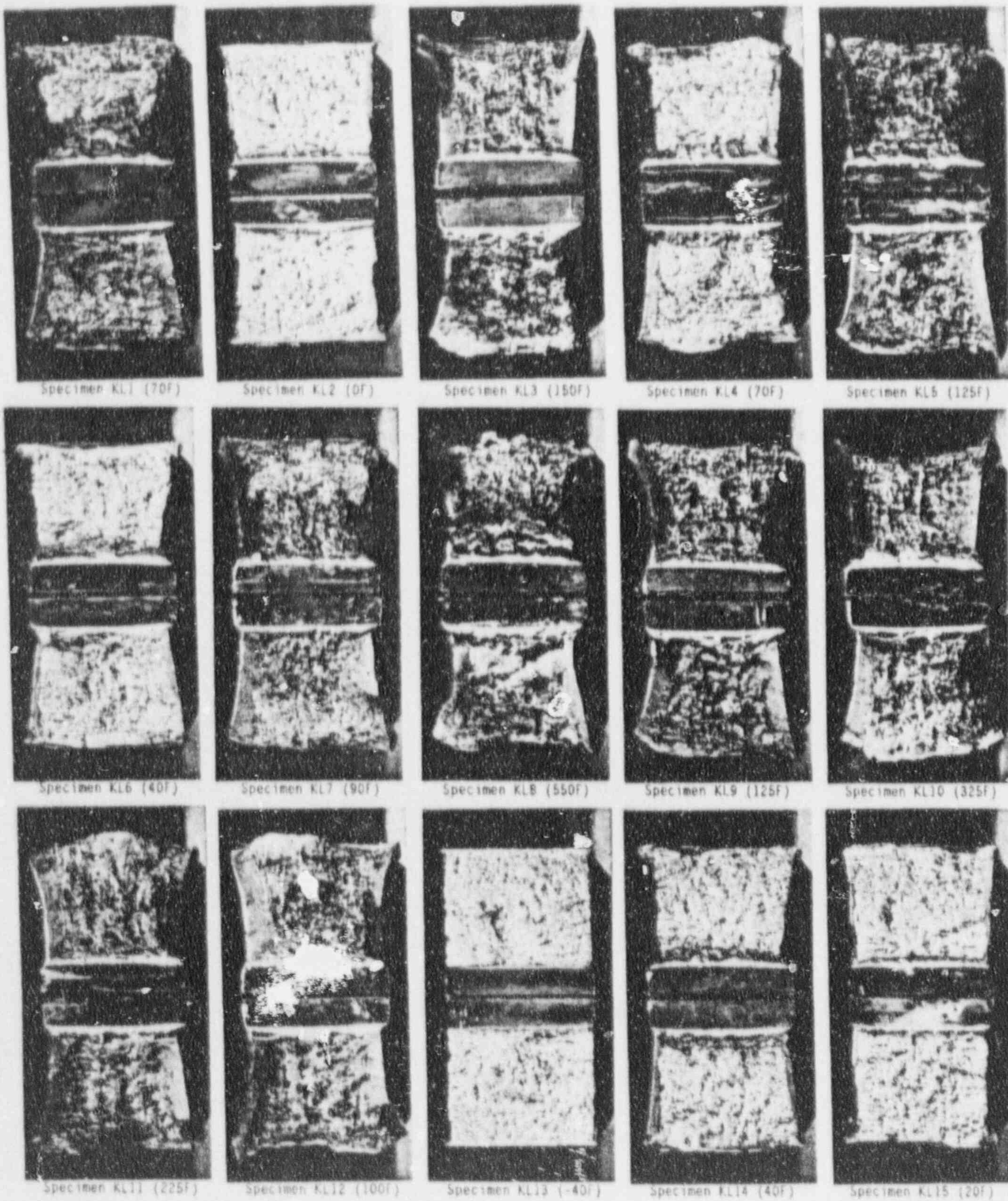


Figure 3-17. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material Longitudinal Orientation

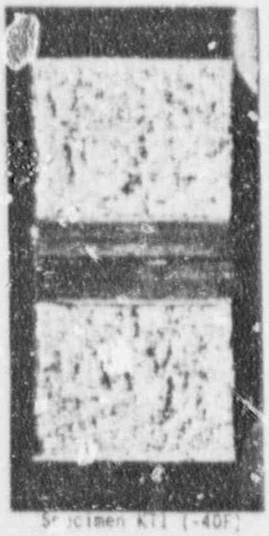


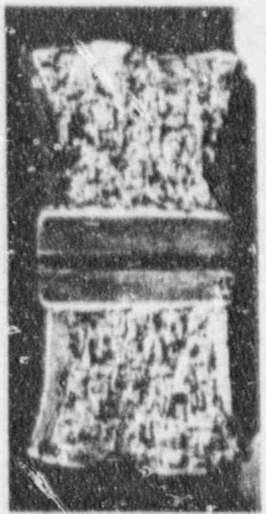
Specimen KL1 (70F) | Specimen KL2 (0F) | Specimen KL3 (150F) | Specimen KL4 (70F) | Specimen KL5 (125F)

Specimen KL6 (40F) | Specimen KL7 (90F) | Specimen KL8 (550F) | Specimen KL9 (125F) | Specimen KL10 (325F)

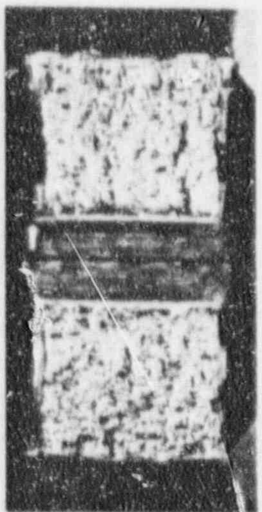
Specimen KL11 (225F) | Specimen KL12 (100F) | Specimen KL13 (-40F) | Specimen KL14 (40F) | Specimen KL15 (20F)

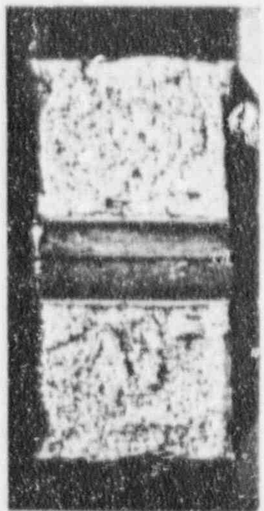
Figure 3-18. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material Transverse Orientation

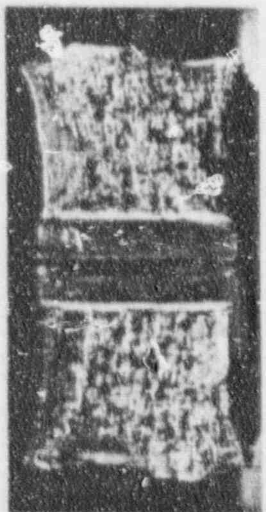

Specimen KT1 (-40F)


Specimen KT2 (175F)


Specimen KT3 (320F)

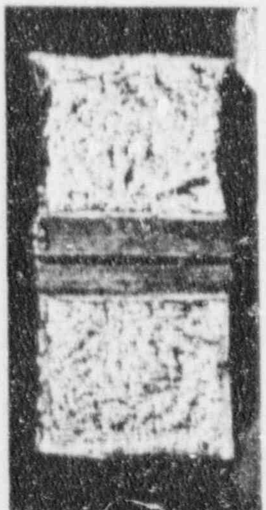

Specimen KT4 (40F)

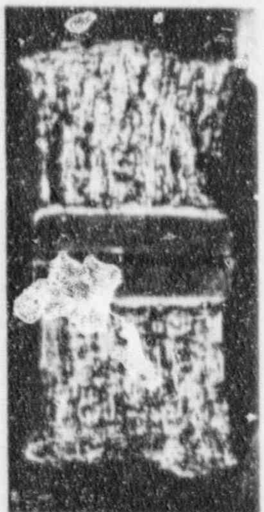

Specimen KT5 (0F)


Specimen KT6 (150F)


Specimen KT7 (70F)

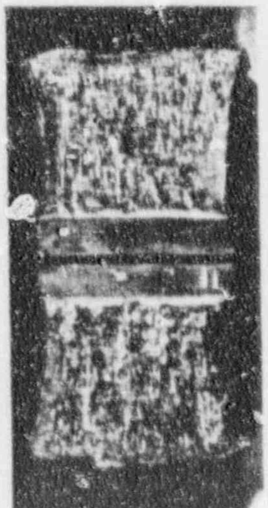

Specimen KT8 (125F)

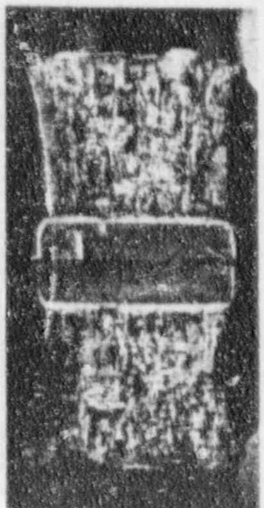

Specimen KT9 (20F)


Specimen KT10 (550F)


Specimen KT11 (100F)


Specimen KT12 (200F)


Specimen KT13 (225F)


Specimen KT14 (120F)

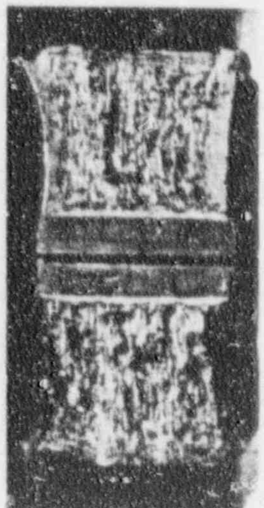

Specimen KT15 (200F)

Figure 3-19. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material, Heat-Affected Zone

Figure 3-20. Photographs of Charpy Impact Specimen Fracture Surfaces - Weld Metal

Specimen KW1 (0F) | Specimen KW2 (-40F) | Specimen KW3 (325F) | Specimen KW4 (550F) | Specimen KW5 (-20F)

Specimen KW6 (0F) | Specimen KW7 (40F) | Specimen KW8 (150F) | Specimen KW9 (-20F) | Specimen KW10 (-80F)

Specimen KW11 (-50F) | Specimen KW12 (100F) | Specimen KW13 (70F) | Specimen KW14 (225F) | Specimen KW15 (70F)

4. DOSIMETER MEASUREMENTS

4.1. Introduction

One set of dosimetry from the reactor vessel surveillance capsule (RVSP) denoted by the letter "U" was delivered to the Nuclea.· Environmental Laboratories by the Failure Analysis and Facility Operations Group of same affiliation (NES).

The set consisted of seventeen dosimeters made up of shielded and unshielded Co/Al wires, unshielded Cu, Ni, and Fe wires, and unshielded ^{238}U and ^{237}Np fission powders. Cadmium sleeves were used to shield the Co/Al wires. Each dosimeter was contained in one of three stainless steel holder blocks that were installed in various positions in the assembly.

The dosimeters were delivered in vials identified by labels consisting of the position of the holder block in the assembly, and the position of the dosimeter item in the holder (see part 4.2 for explanation).

4.2. Dosimeter Preparation

Vials were prepared for the dosimeters by labeling them with identifications that indicated their positions in the holder blocks. For example, the first wire in the top block was labeled Sbl, 01-U TOP 1. When the nuclides to be analyzed were determined by gamma scanning, the identifications were appended accordingly. For example, Sbl, 01-U TOP 1 Cu. This identification code stands for Seabrook Unit 1, Cycle 1 Capsule U, Top holder block, first wire, Copper.

The stainless steel fission powder capsules were clamped in a metal-working vise which was mounted by a suction cup in a hood. A flat mill-bastard file was used to file the capsules open.

The cadmium-covered wires had been crimped at the ends so that the wires had to be removed by "nibbling" through the shield with diagonal cutters and removing the wires.

The dosimeter wires were cleaned by washing in reagent grade acetone, blotting dry with a laboratory towel. Each dosimeter wire diameter was measured with a certified micrometer caliper, and weighed on a certified analytical balance. Each was then mounted in the center of a PetriSlide™ with double-sided tape. Wires over 1/2" in length were bent in a "∩" before mounting.

The exact oxide compositions of the uranium and neptunium dosimeters were uncertain. It was not possible to correct for self-absorption of the powders, so it was necessary to dissolve them and put them into geometries for which our gamma spectrometer was calibrated. This was the 20 cc liquid scintillation vial geometry. The uranium dosimeters were dissolved in 8N HNO_3 and diluted to ca. 20 mL with the same acid in a pre-weighed 20 cc scintillation vial. The neptunium dosimeters were digested in 6N HCl/16N HF with addition of H_2O_2 in increments until dissolved. These were also diluted up to ca. 20 mL in a pre-weighed 20 cc scintillation vials.

4.3. Quantitative Gamma Spectrometry

Each of the dosimeters, in the PetriSlide™ (point source) or 20 cc vial geometry, was given a 300 second preliminary count on the 31% PGT gamma spectrometer. This provided information with which to judge the best distance at which to count the dosimeter to get a minimum of 10,000 counts in the photopeak of interest while keeping the counter dead time below 15%. It also provided qualitative identification of the dosimeters. This identification was made from the presence of the gamma rays in the table below in the spectra.

Dosimeter	Analyte
Cobalt	^{60}Co @ 1332 keV from ^{59}Co, very high activity
Iron	^{54}Mn @ 834 keV from ^{54}Fe
Nickel	^{58}Co @ 811 keV from ^{58}Ni
Copper	^{60}Co @ 1332 keV from ^{59}Co, very low activity compared to Co wires, wire has coppery color
Titanium	^{46}Sc @ 889 keV from ^{46}Ti
^{238}U	^{137}Cs @ 662 keV, ^{234}Pa @ 1001 keV
^{237}Np	^{137}Cs @ 662 keV, ^{233}Pa @ 312 keV

The spectra confirmed the idcntifies of the dosimeters.

The spectra were then measured quantitatively at the appropriate counting positions and for the appropriate count times determined from the preliminary counts.

Table 4-1. Copper Dosimetry Measurements from Capsule U Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Target Abundance	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sb1, 01-U Top 1	Copper	Wire	Cu-63	Co-60	0.692	No	0.08022
2	Sb1, 01-U Mid 1	Copper	Wire	Cu-63	Co-60	0.692	No	0.07936
3	Sb1, 01-U Bot 1	Copper	Wire	Cu-63	Co-60	0.692	No	0.07954

Item No.	Attenuation Coefficient μ	Distance (cm.)	Co-60 (μCi)	% Error Co-60	Geometry Offset Factor	Corrected Activity (μCi/Gram)	μCi/Gram Target
1	4.544E-01	17.536	1.097E-01	1.01	0.9959	1.393E+00	2.012E+00
2	4.544E-01	17.536	9.841E-02	0.97	0.9959	1.263E+00	1.825E+00
3	4.544E-01	17.536	9.706E-02	0.99	0.9959	1.242E+00	1.795E+00

Table 4-2. Iron Dosimetry Measurements from Capsule U
Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Target Abundance	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sb1, 01-U Top 2	Iron	Wire	Fe-54	Mn-54	0.058	No	0.08056
2	Sb1, 01-U Mid 2	Iron	Wire	Fe-54	Mn-54	0.058	No	0.08135
3	Sb1, 01-U Bot 2	Iron	Wire	Fe-54	Mn-54	0.058	No	0.07965

Item No.	Attenuation Coefficient μ	Distance (cm.)	Co-60 (μCi)	% Error Co-60	Geometry Offset Factor	Corrected Activity (μCi/Gram)	μCi/Gram Target
1	5.800E-02	17.536	4.613E+00	0.56	0.9957	5.762E+01	9.934E+02
2	5.800E-02	17.536	4.193E+00	0.59	0.9956	5.187E+01	8.943E+02
3	5.800E-02	17.536	3.993E+00	0.59	0.9956	5.045E+01	8.698E+02

Table 4-3. Cobalt Dosimetry Measurements from Capsule U
Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Target Abundance	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sbl, 01-U Top 3	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	No	0.01057
2	Sbl, 01-U Mid 3	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	No	0.01072
3	Sbl, 01-U Bot 3	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	No	0.00998

Item No.	Attenuation Coefficient μ	Distance (cm.)	Co-60 (μCi)	% Error Co-60	Geometry Offset Factor	Corrected Activity (μCi/Gram)	μCi/Gram Target
1	4.532E-01	17.536	2.995E+00	0.68	0.9973	2.850E+02	1.900E+05
2	4.532E-01	17.536	3.172E+00	0.68	0.9971	2.977E+02	1.984E+05
3	4.532E-01	17.536	3.173E+00	0.67	0.9971	3.198E+02	2.132E+05

Table 4-4. Cadmium Shielded Cobalt Dosimetry Measurements from Capsule U Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Target Abundance	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sbl, 01-U Top 4	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	Yes	0.00862
2	Sbl, 01-U Mid 5	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	Yes	0.01110
3	Sbl, 01-U Bot 4	Cobalt	Wire Alloy	Co-59	Co-60	0.0015	Yes	0.00964

Item No.	Attenuation Coefficient μ	Distance (cm.)	Co-60 (μCi)	% Error Co-60	Geometry Offset Factor	Corrected Activity (μCi/Gram)	μCi/Gram Target
1	4.532E-01	17.536	1.309E+00	0.91	0.9972	1.528E+02	1.018E+05
2	4.532E-01	17.536	1.630E+00	0.61	0.9971	1.477E+02	9.849E+04
3	4.532E-01	17.536	1.537E+00	0.96	0.9971	1.604E+02	1.069E+05

Table 4-5. Nickel Dosimetry Measurements from Capsule U
Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Target Abundance	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sbl, 01-U Top 5	Nickel	Wire	Ni-58	Co-58	0.683	No	0.07773
2	Sbl, 01-U Mid 4	Nickel	Wire	Ni-58	Co-58	0.683	No	0.08024
3	Sbl, 01-U Bot 5	Nickel	Wire	Ni-58	Co-58	0.683	No	0.08105

Item No.	Attenuation Coefficient μ	Distance (cm.)	Co-60 (μCi)	% Error Co-60	Geometry Offset Factor	Corrected Activity (μCi/Gram)	μCi/Gram Target
1	0.6092	28.943	1.107E+02	0.33	0.9974	1.456E+03	2.132E+03
2	0.6092	28.943	1.054E+02	0.34	0.9974	1.343E+03	1.966E+03
3	0.6092	28.943	1.045E+02	0.33	0.9974	1.318E+03	1.930E+03

Table 4-6. Uranium-238 Dosimetry Measurements from Capsule U Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sbl, 01-U FIS 1	U-238	Powder	U-238	Cs-137	No	0.0105

Item No.	Distance (cm.)	Cs-137 (μCi)	% Error Cs-137	μCi/Gram Target
1	0.650	3.560E-02	1.10	3.387E+00

Table 4-7. Neptunium-237 Dosimetry Measurements from Capsule U Seabrook Station Unit No. 1

Item No.	Dosimeter No./ Location	Material	Type	Target Nuclide	Analyte Nuclide	Shielded (Yes/No)	Post-Irrad. Weight (Grams)
1	Sbl, 01-U FIS 2	Np-237	Powder	Np-237	Cs-137	No	0.0022

Item No.	Distance (cm.)	Cs-137 (μCi)	% Error Cs-137	μCi/Gram Target
1	7.387	6.010E-02	1.30	2.689E+01

5. REFERENCES

1. ASTM Designation E185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.
2. Code of Federal Regulation, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix G, Fracture Toughness Requirements.
3. Code of Federal Regulation, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix H, Reactor Vessel Material Surveillance Program Requirements.
4. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure (G-2000).
5. ASTM Designation A370-77, "Methods and Definitions for Mechanical Testing of Steel Products," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.
6. ASTM Designation E23-88, "Methods for Notched Bar Impact Testing of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.
7. ASTM Designation E185-XX (to be released), Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

6. CERTIFICATION

The specimens were tested, and the data obtained from Public Service Company of New Hampshire Seabrook Station Unit No. 1, reactor vessel surveillance Capsule U were evaluated using accepted techniques and established standard methods and procedures in accordance with the requirements of 10CFR50, Appendixes G and H.

P.E. 5/28/92

A. L. Lowe, Jr., P.E. — Date
Project Technical Manager

This report has been reviewed for technical content and accuracy.

6/2/92

M. J. Devan (Material Analysis) — Date
M&SA Unit

Verification of independent review.

William R. Mayhew / KEM — 2 June 92

K. E. Moore, Manager — Date
M&SA Unit

This report is approved for release.

T. L. Baldwin — 6/2/92

T. L. Baldwin — Date
Program Manager