Westinghouse Energy Systems

WCAP-12819

ANALYSIS OF THE MAINE YANKEE REACTOR VESSEL SECOND WALL CAPSULE LOCATED AT 253°

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PREFACE

This report has been technically reviewed and verified.

Reviewer

Sections 1 through 5, 7 and 8   J. M. Chicots
Section 6   E. P. Lippincott

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SECTION 1.0
SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in the surveillance capsule located at 253°, the second wall capsule to be removed from the Maine Yankee reactor pressure vessel, led to the following conclusions:

o   The capsule received an average fast neutron fluence (E > 1.0 MeV) of 1.25×10^{19} n/cm^2 after 11.7 EFPY of plant operation.

o   Irradiation of the reactor vessel intermediate shell plate D-8406-1 Charpy specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F resulted in a 30 and 50 ft-lb transition temperature increase of 120 and 150°F, respectively, for specimens oriented parallel to the major working direction (longitudinal orientation).

o   The weld metal Charpy specimens irradiated to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F resulted in a 30 and 50 ft-lb transition temperature increase of 260°F. This results in a 30 ft-lb transition temperature of 230°F and a 50 ft-lb transition temperature of 270°F for the weld metal.

o   Irradiation of the reactor vessel weld Heat Affected Zone (HAZ) metal Charpy specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F resulted in a 30 ft-lb transition temperature increase of 90°F and a 50 ft-lb transition temperature increase of 110°F.

o   Irradiation of the reactor vessel Standard Reference Material (HSST, Plate 01) Charpy specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F resulted in a 30 ft-lb transition temperature increase of 160°F and a 50 ft-lb transition temperature increase of 180°F for specimens oriented parallel to the major working direction (longitudinal orientation).

o   The average upper shelf energy of intermediate shell plate D-8406-1
    showed a decrease of 35 ft-lbs after irradiation to 1.25×10^{19}
    n/cm^2 (E > 1.0 MeV) at 550°F for specimens oriented parallel to
    the majo working direction (longitudinal orientation).

o   The average upper shelf energy of the weld metal showed a decrease of
    39 ft-lbs after irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at
    550°F.

o   The calculated end-of-life (32 EFPY) maximum neutron fluence (E > 1.0
    MeV) for the Maine Yankee reactor vessel clad/base metal interface is
    2.22×10^{19} n/cm^2 assuming the cycle 8 core power distribution.

o   Based on an EOL (32 EFPY) clad/base metal interface fluence of $2.22 \times$
    10^{19} n/cm^2 (E>1.0 MeV) and Regulatory Guide 1.99 Revision 2 the
    intermediate shell plate D-8406-1 and weld metal, contained in the
    Maine Yankee Reactor Vessel Surveillance Program, are expected to
    exhibit an USE of at least 50 ft-lbs of energy through EOL (32 EFPY).

SECTION 2.0
INTRODUCTION

This report presents the results of the examination of the capsule located at 253°, the second wall capsule to be removed from the Maine Yankee reactor vessel in the continuing surveillance program which monitors the effects of neutron irradiation on the Maine Yankee reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Maine Yankee reactor pressure vessel materials was designed by Combustion Engineering, Inc to the requirements of ASTM 185-70. The design was advanced for its time and the program is comparable to the more recent program identified in ASTM E185-73. Per Maine Yankee a complete description of the surveillance program has been reported by Combustion Engineering, Inc in CENPD-37 "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Maine Yankee Reactor Vessel Materials".

Westinghouse Power Systems personnel were contracted to aid in the preparation of procedures for removing the capsule located at 253° from the reactor and its shipment to the Westinghouse Science and Technology Center Metallographic Hot Cell Facility where the postirradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the testing of and the postirradiation data obtained from the second wall surveillance capsule to be removed from the Maine Yankee reactor vessel and discusses the analysis of these data. The data are compared to the results of tests performed on unirradiated and irradiated material from the other surveillance capsules removed for the reactor vessel.[1,2,3,4,5]
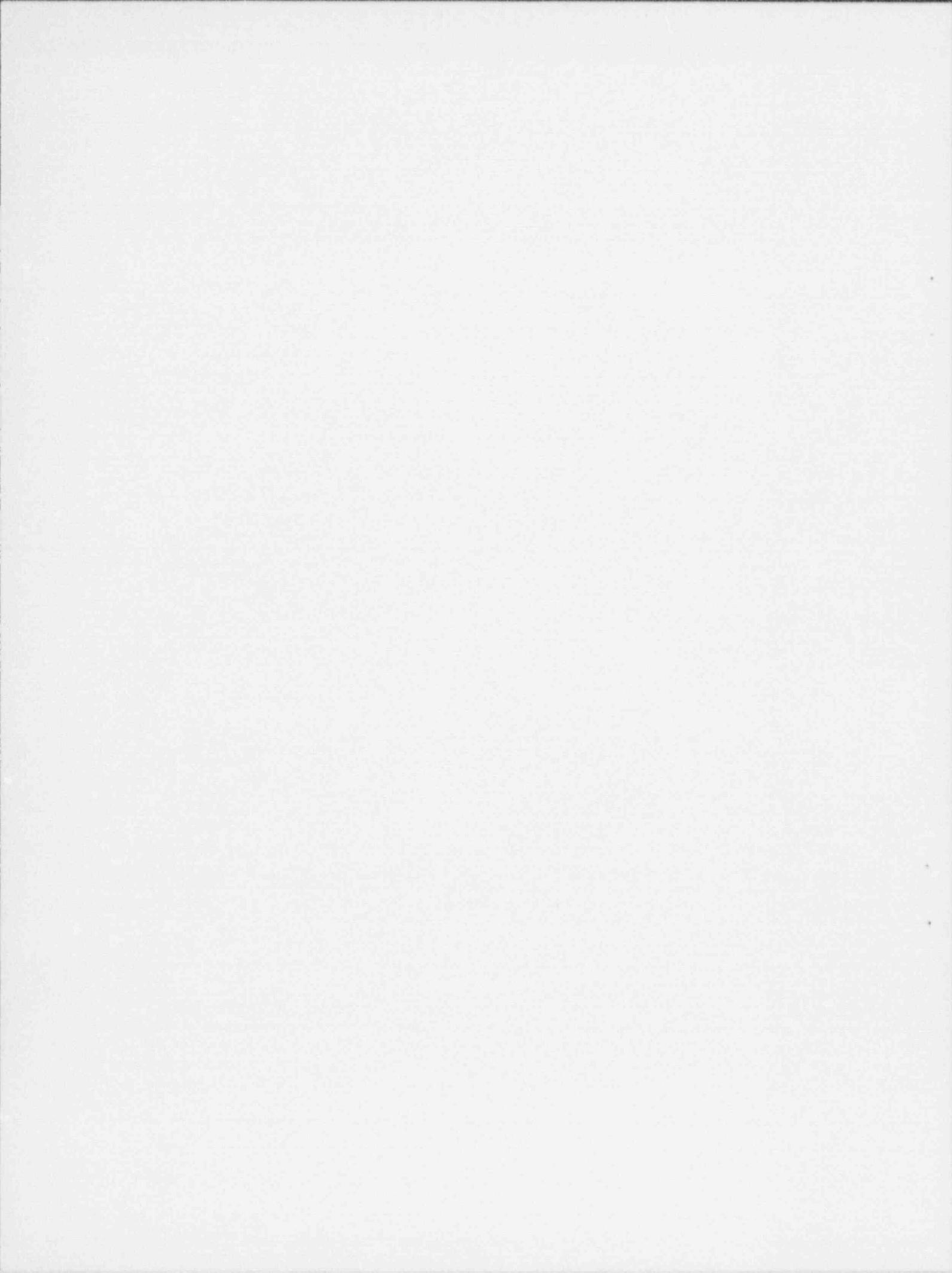
SECTION 3.0
BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels such as SA533 Grade B Class 1 (base material of the Maine Yankee reactor pressure vessel beltline) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

The method, recommended by 10CFR50 Appendix G[6], for performing analyses to guard against fast fracture in reactor pressure vessels is presented in "Protection Against Nonductile Failure," Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.[8] The method uses fracture mechanics concepts and is based on the reference nil-ductility temperature (RT_{NDT}).

The initial RT_{NDT} of the vessel material was established as follows:

Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2) At a temperature not greater than $T_{NDT} + 60°F$, each specimen of the C_V test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3) If the requirements of (2) above are not met, conduct additional C_V tests in groups of three specimens to determine the temperature T_{C_V} at which they are met. In this case the reference temperature $RT_{NDT} = T_{C_V} - 60°F$. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and $(T_{C_V} - 60°F)$.

4) If the C_v test has not been performed at $T_{NDT} + 60°F$, or when the C_v test at $T_{NDT} + 60°F$ does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed.

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 the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static 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.

RT_{NDT} and, in turn, the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor vessel surveillance program such as the Maine Yankee Reactor Vessel Radiation Surveillance Program, in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials.

SECTION 4.0
DESCRIPTION OF PROGRAM

Eight surveillance capsules for monitoring the effects of neutron radiation exposure on the Maine Yankee reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant start-up. A typical surveillance capsule assembly is shown in Figure 4-1. The eight irradiation capsule assemblies are located at radial positions about the core, with their axial positions bisected by the midplane of the core. The capsule assemblies are contained in capsule holders positioned circumferentially about the core at locations which include regions of maximum flux as shown in Figure 4-2. The two accelerated capsule assemblies were located near the outside wall of the core support barrel and so obtain the maximum vessel fluence in 1 or 2 years. The six initial vessel wall capsule assemblies were located near the inside wall of the reactor vessel.

Per Reference 39, a replacement wall capsule was installed in the vacated location at 263° during the Cycle 12 refueling outage. The replacement capsule was fabricated by Combustion Engineering (CE) and is similar in design to the surveillance capsule removed from that location. This replacement capsule had some minor design changes to facilitate installation. The contents of the capsule included:

CE supplied dosimeters similar to the original design except the Sulfur dosimeter was replaced by Neptunium 237 encapsulated in Vanadium.

Hanford Engineering Development Laboratory (HEDL) dosimeter sets.

Maine Yankee surveillance weld Charpy specimens.

The second wall capsule located at 253°F was removed after 11.7 effective full power years of plant operation. This capsule contained Charpy V-notch and tensile specimens from intermediate shell plate D-8406-1, weld metal and weld heat-affected zone (HAZ) material. This capsule also contained Charpy specimens made from standard reference material (SRM) HSST Plate 01.

The chemical composition of the Maine Yankee beltline materials is presented in Table 4-1[1]. The chemical analysis reported in Table 4-1 was obtained from unirradiated material used in the beltline region of the Maine Yankee Reactor Vessel.

The Maine Yankee reactor vessel was fabricated from 9-11/16-inch thick steel plate to ASME specification SA533 Grade B, with Class 1 mechanical properties. The Maine Yankee surveillance materials are made from the same material. The steel plates were purchased from Lukens Steel Company, and the heat treatment of the plates was as follows:

Austenitized: 1550° - 1650°F for 4 hours and water-quenched
Tempered: 1225°±25°F for 4 hours and air-cooled
Stress Relieved: 1150°±25°F for 40 hours and furnace-cooled to 600°
(plate & welds)

Because the lower and intermediate shell courses of the reactor vessel experience the highest fluences in the entire vessel, selection of candidate materials was restricted to the six plates used to fabricate the reactor beltline region. Plate D-8406-1, exhibiting the highest NDT as determined by the drop weight test, was selected as the source for the base and HAZ metal surveillance materials.

Weld metal test material was fabricated by submerged arc-welding lower shell plates D-8407-1 and D-8407-3. Weld HAZ test material was fabricated by welding together intermediate shell plates D-8406-1 and D-8406-2.

All test specimens were machined from the 1/4 thickness location of the base metal plate. Test specimens represent material taken at least one plate thickness from the quenched end. Longitudinal base metal specimens were oriented with the longitudinal axis of the specimen parallel to the major rolling direction of the plate whereas transverse base metal specimens were oriented with the longitudinal axis normal to the major rolling direction. Weld tension specimens were parallel to the weld direction. The axis of the notch of the Charpy specimens was machined perpendicular to the major surfaces of the plate.

The capsule contained fission threshold detectors (U-238) and threshold detectors of nickel (Ni), titanium (Ti), iron (Fe), sulfur (S), copper (Cu) with known cobalt (Co) content and Cobalt (Co) to monitor the thermal neutron exposure. The flux monitors were located in the three tensile-monitor assemblies as shown in Figure 4-3.

Thermal monitors made from low-melting eutectic alloys sealed in quartz tubes were included in the capsule and were located in the three tensile-monitor assemblies as shown in Figure 4-3. The composition of the alloys and their melting points are as follows:

	Melting Temperature	
Chemical Composition	(°C)	(°F)
80% Au, 20% Sn	280	536
90% Pb, 5% Sn, 5% Ag	292	558
97.5% Pb, 2.5% Ag	304	580
97.5% Pb, 0.75% Sn, 1.75% Ag	310	590

Contained in Figure 4-4 is a typical Charpy impact compartment assembly.

TABLE 4-1

Chemical Composition of the Unirradiated Maine Yankee Reactor Vessel Beltline Test Materials *

Matl	Code Numbers and Type							
	D8406-2 Plate	D8407-1 Plate	D8407-2 Plate	D8407-3 Plate	D8406-1 Plate	D8406-3 Plate	D8406-1/ D8406-2 Weld	D8407-1/ D8407-3 Weld
Si	.21	.23	.24	.24	.22	.24	.19	.22
S	.015	.011	.010	.010	.013	.013	.010	.012
P	.009	.008	.007	.007	.013	.010	.014	.015
Mn	1.29	1.40	1.38	1.38	1.27	1.38	1.08	1.38
C	.23	.25	.25	.24	.22	.24	.13	.14
Cr	.10	.11	.11	.13	.11	.06	.07	.07
Ni	.56	.62	.62	.65	.59	.62	.16	.78
Mo	.57	.59	.57	.60	.57	.57	.55	.55
B	.0008	.0003	.0003	.0003	.0004	.0004	.0002	.0002
Cb	< .01	< .01	< .01	< .01	< .01	< .01	< .01	< .01
V	.001	.002	.002	.002	< .001	< .001	.003	.003
Co	.011	.012	.012	.012	.010	.012	.005	.013
N	.007	.007	.007	.007	.006	.008	.006	.012
Cu	.17	.24	.23	.13	.15	.12	.27	.36
A1	.027	.020	.017	.023	.021	.022	.001	.004
Ti	< .01	< .01	< .01	< .01	< .01	< .01	< .01	< .01
W	.01	.01	.01	.01	.01	.01	.01	.01
As	< .01	< .01	< .01	.01	.01	.01	.01	< .01
Sn	.003	.008	.006	.001	.009	.007	.003	.001
Zr	.002	.002	.002	.002	.001	.001	.001	.002

* Chemical Composition as Reported in Reference 1.

Lock Assembly

Wedge Coupling Assembly

Tensile - Monitor
Compartment

Charpy Impact Compartments

Tensile - Monitor
Compartment

Charpy Impact Compartments

Tensile - Monitor
Compartment
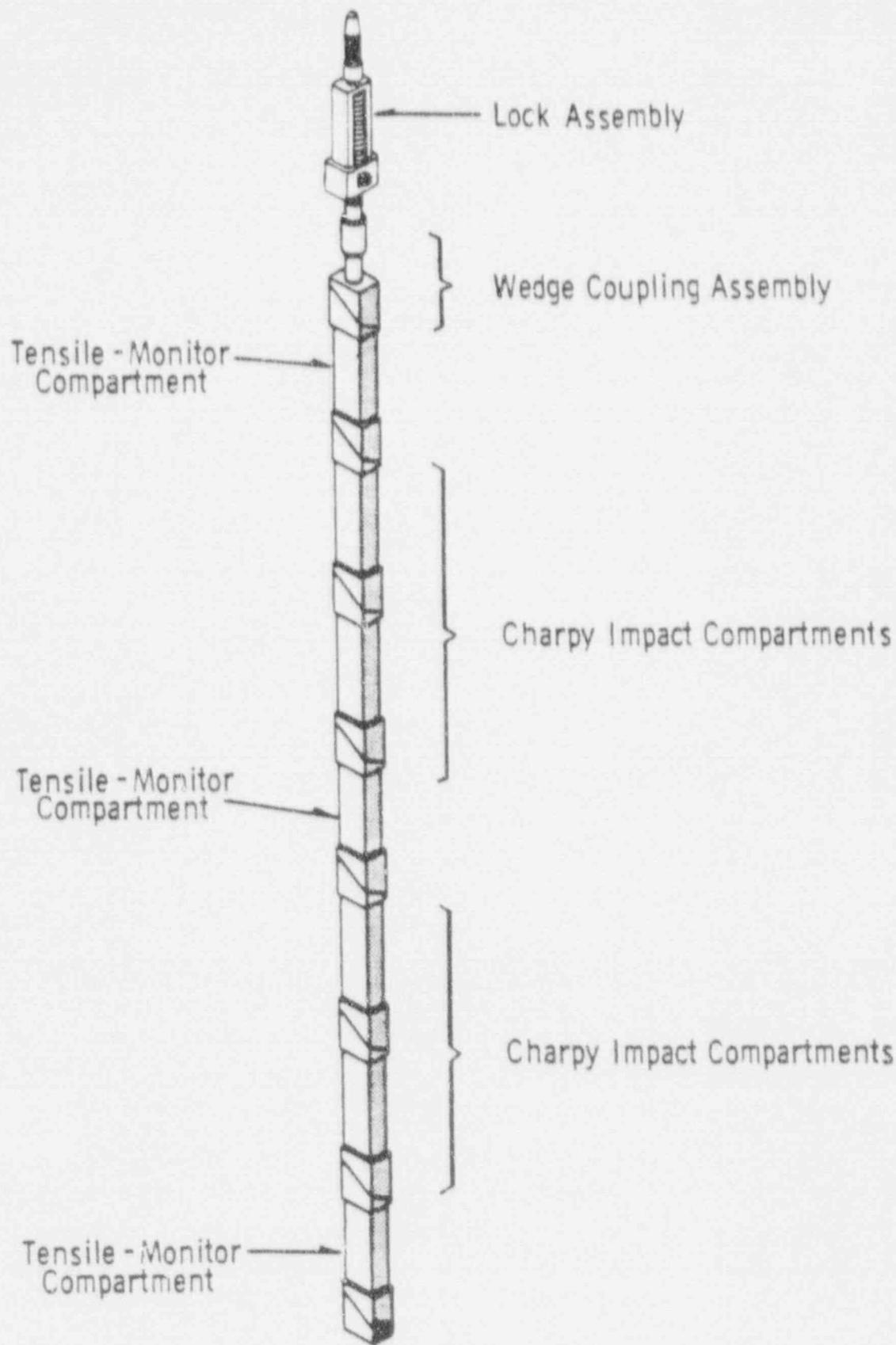
Figure 4-1.   Typical Surveillance Capsule Assembly
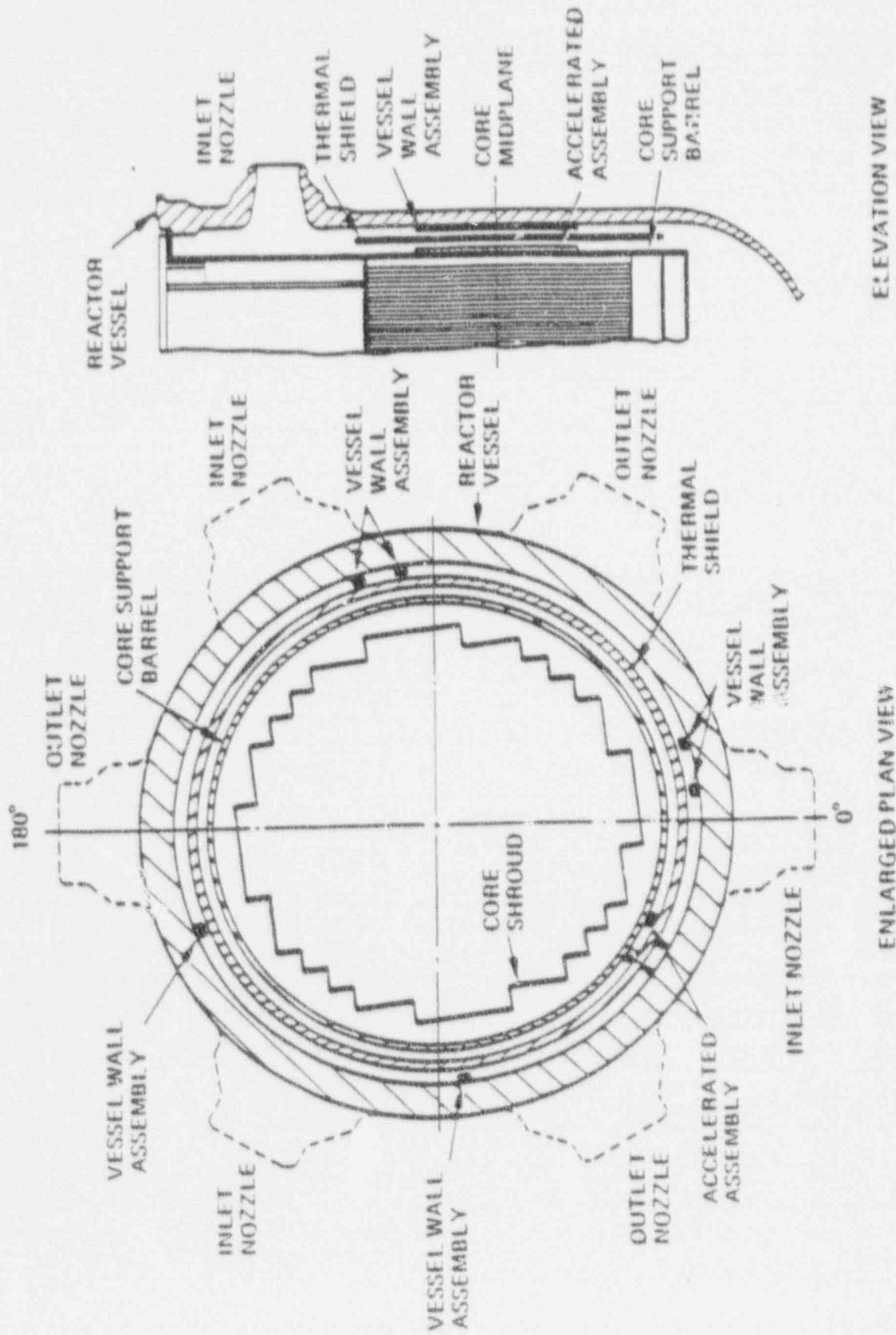
Figure 4-2. Typical Locations of Maine Yankee Surveillance Capsule Assemblies

Wedge Coupling - End Cap

Flux Spectrum Monitor
Cadmium Shielded

Flux Monitor Housing

Stainless Steel Tubing

Threshold Detector

Stainless Steel Tubing

Cadmium Shield

Threshold Detector

Flux Spectrum Monitor

Quartz Tubing

Temperature Monitor

Weight

Temperature Monitor
Housing

Low Melting Alloy

Tensile Specimen

Split Spacer

Tensile Specimen Housing

Rectangular Tubing

Wedge Coupling - End Cap

Figure 4-3. Typical Tensile - Monitor Compartment Assembly

4-7

Wedge Coupling - End Cap

Charpy Impact Specimens

Spacers

Rectangular Tubing

Wedge Coupling - End Cap
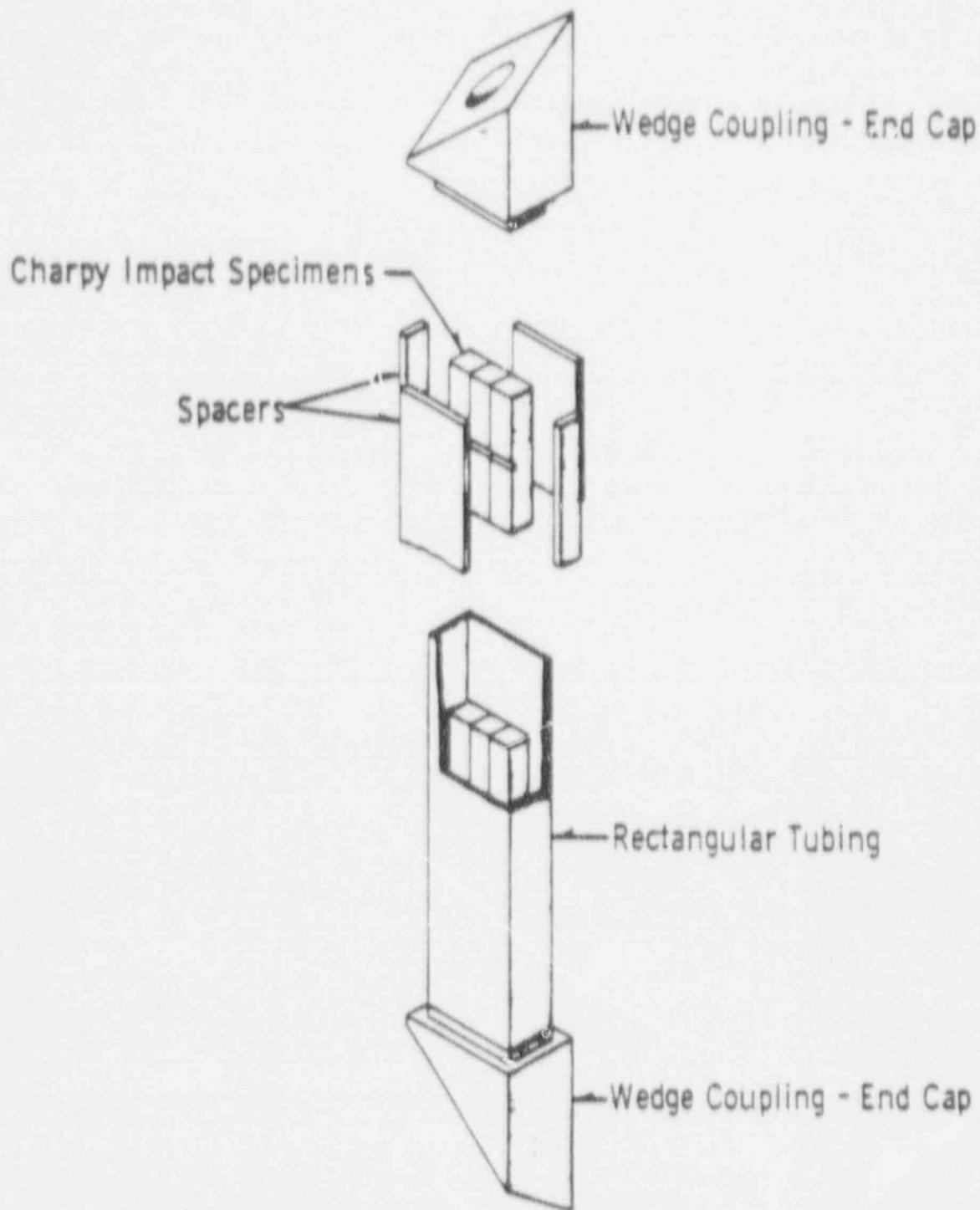
Figure 4-4. Typical Charpy Impact Compartment Assembly

SECTION 5.0
TESTING OF SPECIMENS FROM THE WALL CAPSULE LOCATED AT 253°

5.1 Overview

The post-irradiation mechanical testing of the Charpy V-notch and tensile specimens, contained in the surveillance capsule removed from the 253° location, was performed at the Westinghouse Science and Technology Center Hot Cell Laboratory with consultation by Westinghouse Power Systems personnel. Maine Yankee personnel participated in choosing test temperatures for the testing of the Charpy specimens contained in the surveillance capsule. Testing was performed in accordance with 10CFR50, Appendices G and H[6], ASTM Specification E185-82[10] and Westinghouse Procedure MHL 8402, Revision 1 as modified by RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list supplied by the Maine Yankee Atomic Power Co. No discrepancies were found.

The surveillance capsule had four low melting eutectic alloy thermal monitors which were removed from Blocks 3541, 3573, and 3514. The thermal monitors were encapsulated in quartz tubes of 1, 1.25, 1.50, and 1.75 inches long and had melting temperatures of 536°, 558°, 580° and 590°F, respectively. Examination of the thermal monitors in the various quartz tubes indicated melting of the material in the 1 inch tubes (536°F). Hence the maximum temperatures to which the test specimens were exposed was less than 558°F. Figure 5-1 shows the condition of the thermal monitors

The Charpy impact tests were performed per ASTM Specification E23-88[11] and RMF Procedure 8103, Revision 1 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy machine is instrumented with an Effects Technology Model 500 instrumentation system. With this system, load-time and energy-time signals can be recorded in addition to the standard measurement of Charpy

energy (E_D). From the load-time curve, the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined. Under some test conditions, a sharp drop in load indicative of fast fracture was observed. The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A).

The energy at maximum load (E_M) was determined by comparing the energy-time record and the load-time record. The energy at maximum load is roughly equivalent to the energy required to initiate a crack in the specimen. Therefore, the propagation energy for the crack (E_p) is the difference between the total energy to fracture (E_D) and the energy at maximum load.

The yield stress (σ_Y) is calculated from the three-point bend formula having the following expression:

$$\sigma_Y = P_{GY} * [L / (B * (W-a)^2 * C)] \tag{1}$$

where the constant C is dependent on the notch flank angle (ϕ), notch root radius (ρ), and the type of loading (i.e., pure bending or three-point bending). In three-point bending a Charpy specimen in which $\phi = 45°$ and $\rho = 0.010"$, Equation 1 is valid with $C = 1.21$. Therefore (for $L = 4W$),

$$\sigma_Y = P_{GY} * [L / (B * (W-a)^2 * 1.21)] = [3.3 P_{GY} W] / [B(W-a)^2] \tag{2}$$

For the Charpy specimens, $B = 0.394$ in., $W = 0.394$ in., and $a = 0.079$ in. Equation 2 then reduces to:

$$\sigma_Y = 33.3 \times P_{GY} \tag{3}$$

where σ_Y is in units of psi and P_{GY} is in units of lbs. The flow stress was calculated from the average of the yield and maximum loads, also using the three-point bend formula.

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Specification A370-88[12]. The lateral expansion was measured using a dial gage rig similar to that shown in the same specification.

Tension tests were performed on a 20,000-pound Instron, split-console test machine (Model 1115) per ASTM Specification E8-89b[13] and E21-79(1988)[14], and RMF Procedure 8102, Revision 1. All pull rods, grips, and pins were made of Inconel 718 hardened to HRC45. The upper pull rod was connected through a universal joint to improve axiality of loading. The tests were conducted at a constant crosshead speed of 0.05 inches per minute throughout the test.

Extension measurements were made with a linear variable displacement transducer (LVDT) extensometer. The extensometer knife edges were spring-loaded to the specimen and operated through specimen failure. The extensometer gage length is 1.00 inch. The extensometer is rated as Class B-2 per ASTM E83-85[15].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 9-inch hot zone. All tests were conducted in air.

Because of the difficulty in remotely attaching a thermocouple directly to the specimen, the following procedure was used to monitor specimen temperature: Chromel-alumel thermocouples were inserted in shallow holes in the center and each end of the gage section of a dummy specimen and in each grip. In the test configuration, with a slight load on the specimen, a plot of specimen temperature versus upper and lower grip and controller temperatures was developed over the range of room temperature to 550°F (288°C). The upper grip was used to control the furnace temperature. During the actual testing, the grip temperatures were used to obtained desired specimen temperatures. Experiments indicated that this method is accurate to ±2°F.

The yield load, ultimate load, fracture load, total elongation, and uniform elongation were determined directly from the load-extension curve. The yield strength, ultimate strength, and fracture strength were calculated using the original cross-sectional area. The final diameter and final gage length were determined from post-fracture photographs. The fracture area used to calculate the fracture stress (true stress at fracture) and percent reduction in area was computed using the final diameter measurement.

Rockwell B hardness values were obtained with a RAMS Rockford Hardness Tester Model No. 30-R.

5.2 Charpy V-Notch Impact Test Results

The results of Charpy V-notch impact tests performed on the various materials contained in the Maine Yankee wall capsule located at 253° irradiated at 550°F to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) are presented in Tables 5-1 through 5-6 and are compared with unirradiated results[1] as shown in Figures 5-2 through 5-5. The transition temperature increases and upper shelf energy decreases for the various materials contained in the Maine Yankee wall capsule located at 253° are summarized in Table 5-7.

Irradiation of the reactor vessel intermediate shell plate D-8406-1 Charpy specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550F (Figure 5-2) resulted in a 30 ft-lb transition temperature increase of 120°F and a 50 ft-lb transition temperature increase of 150°F for specimens oriented parallel to the major working direction (longitudinal orientation). This results in a 30 ft-lb transition temperature of 120°F and a 50 ft-lb transition temperature of 175°F.

The average upper shelf energy (USE) of the intermediate shell plate D-8406-1 Charpy specimens (longitudinal orientation) resulted in a decrease of 35 ft-lbs after irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This results in an USE of 105 ft-lbs (Figure 5-2).

Irradiation of the reactor vessel core region weld metal Charpy specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-3) resulted in a 30 and 50 ft-lb transition temperature increase of 260°F. This resulted in a 30 ft-lb transition temperature of 230°F and a 50 ft-lb transition temperature of 270°F.

The average upper shelf energy (USE) of the reactor vessel core region weld metal resulted in a energy decrease of 39 ft-lbs after irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This resulted in an USE of 66 ft-lb (Figure 5-3).

Irradiation of the reactor vessel weld metal Heat-Affected Zone (HAZ) specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-4) resulted in a 30 ft-lb transition temperature increase of 90°F and a 50 ft-lb transition temperature increase of 110°F. This results in a 30 ft-lb transition temperature of 30°F and a 50 ft-lb transition temperature of 65°F

The average upper shelf energy (USE) of the reactor vessel HAZ metal resulted in an decrease of 37 ft-lbs after irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This resulted in an USE of 98 ft-lb (Figure 5-4).

Irradiation of the Standard Reference Material (SRM) HSST - Plate 01 to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-5) resulted in a 30 ft-lb transition temperature increase of 160°F and a 50 ft-lb transition temperature increase of 180°F for specimens oriented parallel to the major working direction (longitudinal orientation). This results in a 30 ft-lb transition temperature of 175°F and a 50 ft-lb transition temperature of 220°F.

The average upper shelf energy (USE) of the Standard Reference Material (SRM) HSST - Plate 01 Charpy specimens (longitudinal orientation) resulted in a decrease of 34 ft-lbs after irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This results in an USE of 96 ft-lbs (Figure 5-5).

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-6 through 5-9 and show an increasingly ductile or tougher appearance with increasing test temperature.

A comparison of the 30 ft-lb transition temperature increases and the upper shelf energy decreases for the various Maine Yankee surveillance materials with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2[7] is presented in Table 5-8. This comparison indicates that the 30 ft-lb transition temperature increases and USE decreases resulting from irradiation to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) are in close agreement with Regulatory Guide 1.99, Revision 2 predictions.

Figure 5-10 is a plot of the measured 30 ft-lb transition temperature shift vs. fluence.

Figure 5-11 is a plot of the measured upper shelf energy decrease vs. fluence.

Figure 5-12 is a plot of the measured upper shelf energy percent decrease vs. fluence.

5.3 Tension Test Results

The results of tension tests performed on the reactor vessel intermediate shell plate D-8406-1 (longitudinal orientation), weld metal and weld HAZ metal irradiated to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F are shown in Table 5-9 and are compared with unirradiated results[1] as shown in Figures 5-13, 5-14 and 5-15.

Irradiation of the reactor vessel intermediate shell plate B-8406-1 tensile specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-13) resulted in an increase of 15 to 20 ksi in 0.2 percent offset yield strength and an increase of 10 to 17 ksi in ultimate tensile strength for specimens oriented with the longitudinal axis parallel to the major working direction of the plate (longitudinal orientation)

Irradiation of the reactor vessel weld metal tensile specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-14) resulted in an increase of 27 to 36 ksi for the 0.2 percent offset yield strength and an increase of 22 to 28 ksi for the ultimate tensile strength.

Irradiation of the reactor vessel weld HAZ metal tensile specimens to 1.25×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-15) resulted in an increase of 14 to 18 ksi for the 0.2 percent offset yield strength and an increase of 11 to 18 ksi for the ultimate tensile strength.

The fractured tension specimens for the reactor vessel intermediate shell plate B-8406-1 tensile specimens are shown in Figure 5-16.

The fractured tension specimens for the weld metal are shown in Figure 5-17.

The fractured tension specimens for the weld HAZ metal are shown in Figure 5-18.

Engineering stress-strain curves for the tension specimens are shown in Figures 5-19 through 5-23.

5.4 Hardness Test Results

The results of Rockwell B hardness tests are presented in Table 5-9.

5.5 Chemical Analysis Results

A chemical analysis using Inductively Coupled Plasma Spectrometry (ICPS) was performed on irradiated weld specimens 312 and 3A2 and SRM specimens 61D and 61E and is reported in Table 5-11. The chemistry results from the NBS certified reference standards are reported in Table 5-12.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE MAINE YANKEE REACTOR VESSEL INTERMEDIATE SHELL PLATE D-8406-1 IRRADIATED AT 550°F, FLUENCE 1.25×10^{19} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	(°C)	Impact Energy (ft-lb)	(J)	Lateral Expansion (mils)	(mm)	Shear (%)
			Longitudinal Orientation				
12T	25	(- 4)	7.0	(9.5)	7.0	(0.18)	5
13E	110	(43)	28.0	(38.0)	26.0	(0.66)	20
121	115	(46)	26.0	(35.5)	26.0	(0.66)	20
15M	125	(52)	27.0	(36.5)	26.0	(0.55)	20
124	135	(57)	42.0	(57.0)	36.0	(0.91)	30
12C	175	(79)	43.0	(58.5)	38.0	(0.97)	35
126	180	(82)	57.0	(77.5)	50.0	(1.27)	50
12M	185	(85)	57.0	(77.5)	49.0	(1.24)	50
12P	230	(110)	86.0	(116.5)	67.0	(1.70)	90
12D	250	(121)	99.0	(134.0)	78.0	(1.98)	100
12U	325	(163)	113.0	(153.0)	85.0	(2.16)	100
13K	400	(204)	104.0	(141.5)	86.0	(2.18)	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE MAINE YANKEE REACTOR VESSEL WELD METAL AND HAZ METAL IRRADIATED AT 550°F, FLUENCE 1.25×10^{19} n/cm² (E > 1.0 MeV)

Sample No.	Temperature (°F)	Temperature (°C)	Impact Energy (ft-lb)	Impact Energy (J)	Lateral Expansion (mils)	Lateral Expansion (mm)	Shear (%)
			Weld Metal				
37K	0	(-18)	3.0	(4.0)	4.0	(0.10)	5
31J	15	(- 9)	5.0	(7.0)	5.0	(0.13)	5
32L	180	(82)	13.0	(17.5)	13.0	(0.33)	20
32A	185	(85)	24.0	(32.5)	23.0	(0.58)	25
35D	220	(104)	26.0	(35.5)	26.0	(0.66)	60
32K	240	(116)	35.0	(47.5)	39.0	(0.99)	85
3A2	255	(124)	42.0	(57.0)	38.0	(0.97)	90
33M	260	(127)	50.0	(68.0)	44.0	(1.12)	95
37J	300	(149)	52.0	(70.5)	51.0	(1.30)	95
317	340	(171)	63.0	(85.5)	62.0	(1.57)	100
36C	400	(204)	67.0	(91.0)	56.0	(1.42)	100
312	425	(218)	68.0	(92.0)	59.0	(1.50)	100
			HAZ Metal				
47Y	- 30	(-34)	13.0	(17.5)	9.0	(0.23)	10
43J	0	(-18)	27.0	(36.5)	22.0	(0.56)	25
463	10	(-23)	BAD TEST (MACHINE MALFUNCTION)-				-
433	20	(- 7)	21.0	(28.5)	21.0	(0.53)	20
47C	20	(- 7)	46.0	(62.5)	33.0	(0.84)	50
47J	35	(2)	50.0	(68.0)	37.0	(0.94)	65
47T	75	(24)	42.0	(57.0)	30.0	(0.76)	75
43B	80	(27)	53.0	(72.0)	43.0	(1.09)	85
46Y	100	(38)	71.0	(96.5)	55.0	(1.40)	95
43K	125	(52)	108.0	(146.5)	80.0	(2.03)	100
46U	205	(96)	94.0	(127.5)	69.0	(1.75)	100
473	230	(110)	94.0	(127.5)	72.0	(1.83)	100

TABLE 5-3

CHARPY V-NOTCH IMPACT DATA FOR THE MAINE YANKEE STANDARD REFERENCE MATERIAL (HSST - PLATE 01) IRRADIATED AT 550°F, FLUENCE 1.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	Temperature (°C)	Impact Energy (ft-lb)	Impact Energy (J)	Lateral Expansion (mils)	Lateral Expansion (mm)	Shear (%)
61E	40	(4)	5.0	(7.0)	10.0	(0.25)	5
61A	115	(46)	17.0	(23.0)	16.0	(0.41)	15
61P	120	(49)	14.0	(19.0)	12.0	(0.30)	15
61C	155	(68)	18.0	(24.5)	16.0	(0.41)	20
61M	175	(79)	27.0	(36.5)	26.0	(0.66)	35
627	185	(85)	33.0	(44.5)	29.0	(0.74)	35
62A	210	(99)	38.0	(51.5)	35.0	(0.89)	45
61A	215	(102)	26.0	(35.5)	27.0	(0.69)	40
61D	230	(110)	72.0	(97.5)	57.0	(1.45)	95
61T	250	(121)	89.0	(120.5)	69.0	(1.75)	100
61U	300	(149)	93.0	(12605)	76.0	(1.93)	100
61J	400	(204)	107.0	(145.5)	90.0	(2.29)	100

TABLE 5-4

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE MAINE YANKEE INTERMEDIATE SHELL

PLATE D-8406-1 IRRADIATED AT 550°F, FLUENCE 1.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Charpy Ed/A (ft-lb/in²)	Maximum Em/A	Prop Ep/A	Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
						Longitudinal Orientation (RW)							
12T	25	7.0	58	32	24	1.86	200	3.15	245	3.15	0.30	81	83
13E	110	28.0	225	182	43	3.25	175	4.15	480	4.15	0.11	107	122
121	115	28.0	209	187	43	3.15	125	4.05	435	4.05	0.25	104	118
15W	125	27.0	217	166	52	3.55	135	4.80	400	4.50	0.35	117	134
124	135	42.0	238	206	132	3.05	95	4.20	480	4.05	0.40	101	130
12C	175	43.0	348	200	148	3.90	285	3.90	575	3.90	1.65	98	112
126	180	57.0	459	265	194	3.30	125	4.35	610	4.15	1.45	109	127
12W	185	57.0	459	333	126	3.60	145	4.80	890	4.75	1.75	120	140
12P	230	86.0	592	284	409	3.00	135	4.10	700	3.35	2.30	99	117
12D	260	99.0	797	237	580	3.00	135	4.35	605	-*	-*	99	115
12U	325	113.0	910	273	837	3.15	180	4.50	845	-*	-*	104	127
13K	400	104.0	837	243	595	2.45	95	3.70	830	-*	-*	81	102

*Fully ductile fracture, no arrest load.

TABLE 5-5

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE MAINE YANKEE
WELD METAL AND HAZ METAL IRRADIATED AT 550°F, FLUENCE 1.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Charpy Ed/A (ft-lb/in²)	Maximum Em/A (ft-lb/in²)	Prop Ep/A (ft-lb/in²)	Yield Load (kips)	Time to Yield (μsec)	Maximum Load (k'ps)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
Weld Metal													
37K	0	3.0	24	COMPUTER MALFUNCTION		-	-	-	-	-	-	-	-
31J	15	5.0	40	23	17	3.40	25	4.50	30	4.50	0.07	113	131
32L	180	13.0	105	45	59	3.55	125	3.80	1?0	3.55	0.80	117	121
32A	185	24.0	193	188	28	4.05	125	4.75	3?0	4.75	0.75	134	145
35D	220	28.0	209	116	93	3.35	170	4.35	3?5	4.35	2.20	111	127
32K	240	35.0	282	100	182	2.80	230	3.80	395	3.80	3.80	93	109
3A2	255	42.0	338	187	151	3.45	190	4.20	485	3.95	2.15	114	127
33M	260	50.0	403	137	266	3.00	145	3.95	387	3.85	3.50	100	116
37J	300	52.0	419	170	249	2.85	210	3.90	510	3.80	3.25	94	112
317	340	63.0	507	175	333	3.10	200	4.00	500	--	--	102	117
380	400	67.0	540	165	374	3.15	235	3.90	485	--	--	104	118
312	425	68.0	548	202	346	3.60	135	4.55	450	--	--	119	135
HAZ Metal													
47Y	-30	13.0	105	94	11	4.25	115	4.90	225	4.75	0.11	141	152
43J	0	27.0	217	183	34	4.60	165	5.20	405	5.15	1.05	153	152
403	10	BAD TEST MACHINE MALFUNCTION										-	-
433	20	21.0	169	122	47	3.70	135	4.40	310	4.30	0.60	122	133
470	20	48.0	370	266	104	4.10	200	5.15	550	5.05	2.05	136	154
47J	35	50.0	403	223	180	3.35	155	4.25	545	4.25	2.35	112	126
47T	75	42.0	338	169	169	3.15	165	4.00	470	4.00	2.95	103	118
43B	80	53.0	427	245	182	3.55	270	4.80	640	4.60	2.86	117	135
46Y	100	71.0	572	301	271	3.80	180	5.00	610	4.55	2.50	126	146
43K	125	108.0	870	294	578	3.70	135	4.85	625	--	--	122	141
46U	205	94.0	757	276	481	2.86	145	4.00	695	--	--	94	113
473	230	94.0	757	283	474	3.00	135	4.70	630	--	--	100	127

*Fully ductile fracture, no arrest load.

TABLE 5-6

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE MAINE YANKEE STANDARD REFERENCE

MATERIAL (HSST - PLATE 01) IRRADIATED AT 550°F, FLUENCE 1.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
				Longitudinal Orientation (RW)									
61E	40	5.0	40	30	10	2.55	105	4.05	145	4.05	0.01	84	109
614	115	17.0	137	107	30	2.85	115	3.90	310	3.90	0.04	95	112
61P	120	14.0	113	COMPUTER MALFUNCTION		-	-	-	-	-	-	-	-
61C	155	18.0	145	68	77	3.30	145	3.70	235	3.60	0.08	109	116
61M	175	27.0	217	140	78	2.75	125	3.75	400	3.85	1.50	91	107
627	185	33.0	288	198	70	3.75	185	4.65	480	4.65	0.90	124	139
62A	210	38.0	308	204	102	2.85	155	4.00	560	3.90	1.55	93	113
61A	215	28.0	209	131	78	2.95	125	3.80	370	3.80	0.90	97	112
61D	230	72.0	580	274	306	3.30	190	4.70	630	4.25	3.40	109	132
61T	250	89.0	717	292	424	3.60	155	4.85	625	†	†	118	140
61U	300	93.0	749	393	456	3.55	155	4.85	625	†	†	117	139
61J	400	107.0	882	272	590	3.25	155	4.60	630	†	†	108	130

†Fully ductile fracture, no arrest load.

5-13

TABLE 5-7

EFFECT OF 550°F IRRADIATION TO 1.25×10^{19} n/cm² (E > 1.0 MeV)

ON NOTCH TOUGHNESS PROPERTIES OF THE MAINE YANKEE REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Average 30 ft-lb [2] Temperature (°F)			Average 35 mil [2] Lateral Expansion Temperature (°F)			Average 50 ft-lb [2] Temperature (°F)			Average Energy [1,2] Absorption at Full Shear (ft-lb) [3]		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ(ft-lb)
Plate B8406-1 (Longitudinal Orientation)	0	120	120	10	150	140	25	175	150	140	105	35
Weld Metal	-30	230	260	-15	240	255	10	270	260	105	66	39
HAZ Metal	-60	30	90	-55	55	110	-45	65	110	135	98	37
SRM (HSST Plate 01) (Longitudinal Orientation)	15	175	160	25	200	175	40	220	180	130	96	34

(1) Unirradiated data presented here was taken from CR 75-269 Table 13 [1].

(2) The "AVERAGE" is defined as the value read from the curve fitted through the data points of the Charpy tests (Figures 5-2 through 5-5).

(3) Unirradiated values of average upper shelf energy presented here may not represent the average of data points with > 90% shear.

TABLE 5-8

MAINE YANKEE SURVEILLANCE MATERIAL 30 FT-LB TRANSITION
TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES

Material	Capsule	Fluence 10^{19} n/cm^2	Measured 30 ft-lb Temperature Shift (°F)	Measured Upper Shelf Energy Decrease[1] (ft-lb)	(%)
Plate D-8406-1	W-263	0.572	97.0	113.0	15.0
(Longitudinal	W-253	1.25	120.0	105.0	19.0
Orientation)	A-25	1.79	120.0	96.0	31.0
	A-35	6.53	185.0	85.0	39.0
Plate D-8406-1	W-263	0.572	93.0	96.0	16.0
(Transverse)	W-253	1.25	---	---	--
Orientation)	A-25	1.79	---	---	--
	A-35	6.53	195.0	71.0	38.0
Weld Metal	W-263	0.572	222.0	59.0	44.0
	W-253	1.25	260.0	66.0	37.0
	A-25	1.79	270.0	57.0	45.0
	A-35	6.53	345.0	50.0	53.0
HAZ Metal	W-263	0.572	91.0	103.0	22.0
	W-253	1.25	90.0	98.0	27.0
	A-25	1.79	85.0	103.0	23.0
	A-35	6.53	155.0	94.0	30.0
SRM (HSST -	W-263	0.572	---	---	--
Plate 01)	W-253	1.25	160.0	96.0	26.0
(Longitudinal	A-25	1.79	150.0	100.0	23.0
Orientation)	A-35	6.53	---	---	--

(1) The measured USE (% decrease) reported here for capsules A-25, A-35 and W-263 is from the capsule analysis reports [3,4,5]. The measured USE (% decrease) reported here for capsule W-253 is based on the data presented in Table 5-7.

TABLE 5-9

TENSILE PROPERTiES FOR MAINE YANKEE REACTOR VESSEL SURVEILLANCE AND REFERENCE MATERIAL
IRRADIATED AT 550°F TO 1.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Plate	1L4	74	80.5	101.4	3.30	183.3	67.2	13.5	26.1	63
D-8406-1	1KM	300	74.4	94.7	3.20	181.1	65.2	12.8	23.1	64
	1JL	560	68.8	93.7	3.35	200.9	68.2	10.5	21.3	66
Weld, Plates	3JM	74	107.0	115.1	4.00	152.2	81.5	12.7	24.8	46
D-8406-1 and	3J7	300	93.7	105.9	3.90	171.8	79.5	12.0	21.6	54
D-8407-3	3L4	560	87.6	105.9	4.20	174.6	85.6	10.5	17.9	51
HAZ, Plate	4L2	74	74.9	101.9	3.50	194.4	71.3	10.5	21.9	63
D-8406-1	4KK	300	73.3	95.7	3.20	181.1	65.2	9.8	19.5	64
	4K1	560	73.3	96.8	3.60	185.7	73.3	9.8	18.5	61

TABLE 5-10

ROOM TEMPERATURE ROCKWELL B HARDNESS VALUES FOR THE MAINE
YANKEE BASE METAL, WELD METAL AND REFERENCE MATERIAL
IRRADIATED AT 550°F, FLUENCE 1.25×10^{19} n/cm^2 (E > 1.0 MeV)

Specimen Identification	Rockwell B Hardness			Average
(1) Base Metal (Long.)				
12D	96	97	96	96
13E	94	94	96	95
12P	96	97	96	96
12M	97	96	97	97
121	95	96	98	96
13K	95	97	96	96
12T	93	95	93	94
12C	94	96	96	95
124	92	96	96	95
12U	94	98	98	97
126	95	96	97	96
15M	95	97	97	96
(2) Weld Metal				
37K	100	104	100	101
32A	100	101	100	100
35D	99	99	99	99
31J	100	100	100	100
32L	100	100	100	100
37J	100	100	99	100
32K	100	101	100	100
312	100	100	100	100
33M	100	100	100	100
36C	100	100	100	100
317	94	101	101	99
3A2	99	100	100	100
(3) Standard Reference Material				
614	95	98	96	96
61P	95	98	98	97
61C	95	97	98	97
61A	87	97	97	94
61M	98	98	98	98
61E	96	97	98	97
61D	97	97	97	97
61J	93	97	97	96
627	96	98	97	97
62A	97	97	98	97
61U	92	95	97	95
61T	93	94	94	94

TABLE 5-11

Chemical Compostion of Four Maine Yankee Charpy
Specimens contained in the Surveillance Capsule Located at 253°

| | Chemical Composition (wt.%) | | | |
| | Standard Reference Metal | | Weld Metal | |
Specimen No.	61E	61D	3A2	312
Fe	MATRIX ELEMENT: Remainder by Difference			
Co	0.0086	0.0082	0.0169	0.0161
Cu	0.173	0.170	0.432	0.356
Cr	0.080	0.080	0.071	0.069
Mn	1.524	1.487	1.369	1.334
Mo	0.577	0.556	0.504	0.487
Ni	0.665	0.654	0.745	0.728
P	0.0112	0.0104	0.0174	0.0149
V	<0.005	<0.005	<0.005	<0.005
Al	0.0244	0.0232	<0.0035	0.0128
As	<0.0030	0.0066	0.0034	<0.0030
B	0.0084	0.0079	0.0070	0.0064
Nb	<0.007	<0.007	<0.007	<0.007
Sn	0.0154	0.0145	0.0072	0.0064
Ti	0.0051	0.0045	0.0046	0.0036
W	0.0062	0.0069	<0.00504	<0.0050
Zr	<0.019	<0.019	<0.019	<0.019
C	0.221	0.215	0.130	0.126
S	0.0174	0.0174	0.0068	0.0177
Si	0.198	0.077	0.218	0.192
N	0.0079	0.0071	0.0141	0.0086

Analyses	Method of Analysis
Metals	ICPS, Inductively Coupled Plasma Spectrometry
Carbon	EC-12, LECO Carbon Analyzer
Sulfur	Combustion/titration
Silicon	Dissolution/gravimetric
Nitrogen	LECO NITROX

TABLE 5-12
Chemistry Results from the NBS
Certified Reference Standards

Material ID	Low Alloy Steel: NBS Certified Reference Standards			
	NBS 361		NBS 362	
	Certified	Measured	Certified	Measured
Metals	Concentration in Weight Percent			
Fe *	95.60	(matrix)	95.30	(matrix)
Co	0.032	0.0305	0.30	0.3205
Cu	0.042	0.041	0.50	0.506
Cr	0.694	0.692	0.30	0.302
Mn	0.66	0.656	1.04	1.046
Mo	0.19	0.192	0.068	0.065
Ni	2.00	1.992	0.59	0.577
P	0.014	0.0177	0.041	0.0406
V	0.011	0.009	0.040	0.041
Al	0.021	0.0191	0.095	0.0793
As	0.017	0.0177	0.092	0.0920
B	0.00037	0.00035	0.0025	0.00253
Nb	0.022	0.018	0.29	0.315
Sn	0.010	0.0100	0.016	0.0172
Ti	0.020	0.0235	0.084	0.0995
W	0.017	0.0187	0.20	0.2103
Zr	0.009	0.012	0.19	0.203
C	0.383	0.381	0.160	0.159
S	0.014	N.A.	0.036	0.0360
Si	0.222	0.219	0.39	N.A.

* Matrix element calculated as difference for material balance. ( )
Tentative value, certified ± 100% of value.   t
N.A. - Not analyzed

TABLE 5-12 (con't)
Chemistry Results from the NBS
Certified Reference Standards

Material ID	Low Alloy Steel: NBS Certified Reference Standards			
	NBS 363		NBS 364	
	Certified	Measured	Certified	Measured

Metals	Concentration in Weight Percent			
Fe *	94.4	(matrix)	96.7	(matrix)
Co	0.048	0.0477	0.15	0.1574
Cu	0.10	0.097	0.249	N.A.
Cr	1.31	1.338	0.063	N.A.
Mn	0.30	0.304	0.144	N.A.
Mo	1.50	1.505	0.255	N.A.
Ni	0.028	0.029	0.49	N.A.
P	0.029	0.0326	0.01	N.A.
V	0.31	0.317	0.105	N.A.
Al	0.24	0.2383	0.008 t	0.0081
As	0.010	0.0118	0.052	0.0520
B	0.00078	0.00071	0.0106	0.01074
Nb	0.049	0.037	0.157	0.157
Sn	0.104	0.1040	0.0008	<0.0066
Ti	0.050	0.0566	0.24	0.2392
W	0.046	0.0680	0.10	0.1367
Zr	0.049	0.050	0.068	0.071
C	0.62	N.A.	0.87	N.A.
S	0.0068	N.A.	0.0250	0.0251
Si	0.74	N.A.	0.065	N.A.

Nitrogen	LECO Std #656		LECO Std #657	
	0.0193	0.196	0.0102	0.0104
	±0.0008		±0.0004	

* Matrix element calculated as difference for material balance. ( )
  Tentative value, certified ± 100% of value.   t
  N.A. - Not analyzed

a) Block 3541
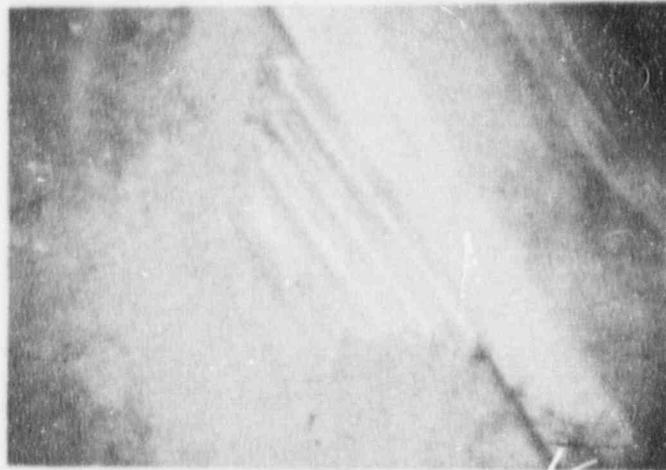
b) Block 3573

c) Block 3514

Figure 5-1.   Appearance of the Thermal Monitors in the Various Quartz Tubes.
Melting Occurred in the One Inch Long Tubes (536°F)
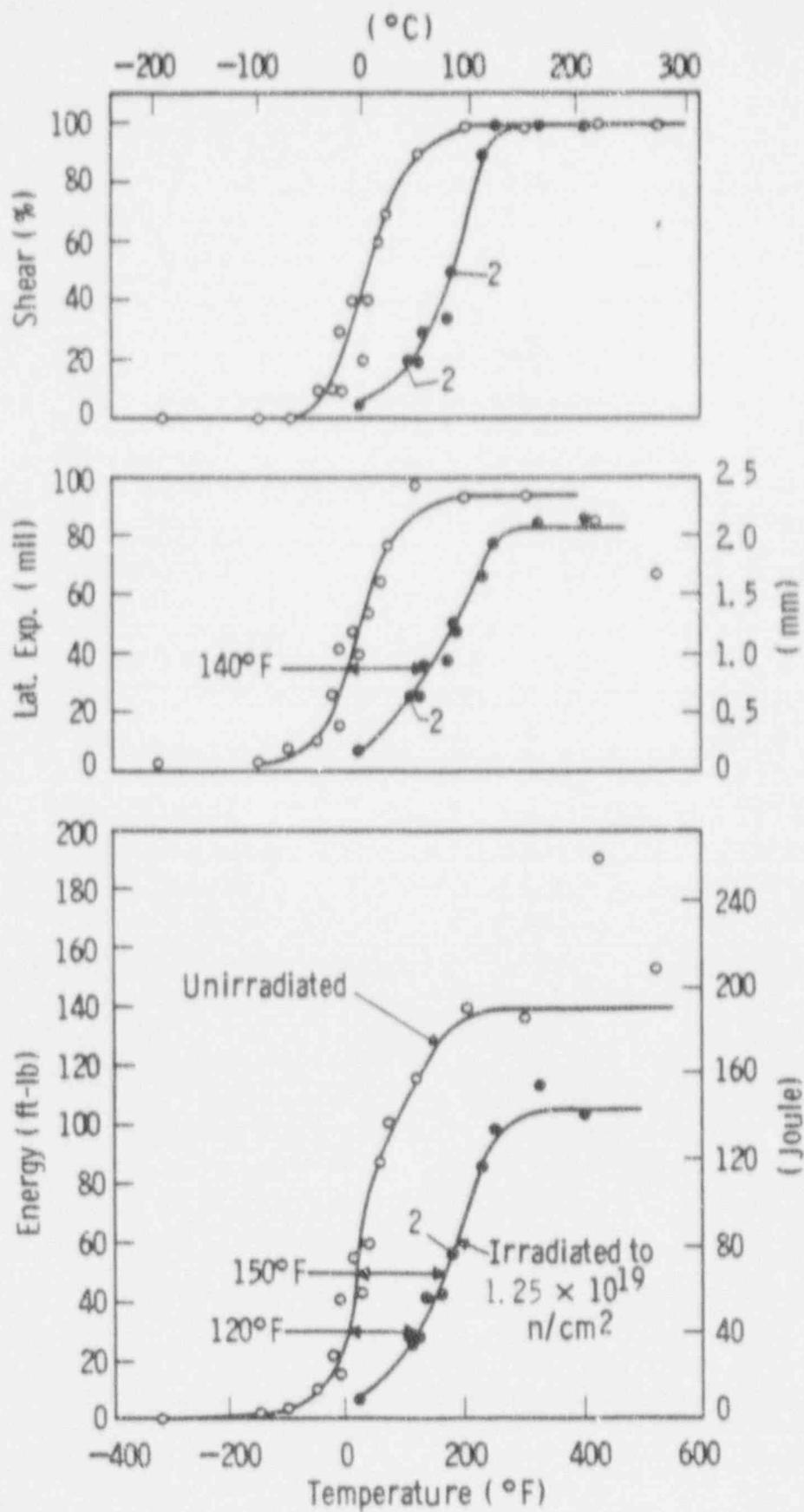
Figure 5-2. Charpy V-Notch Impact Properties for the Maine Yankee Reactor
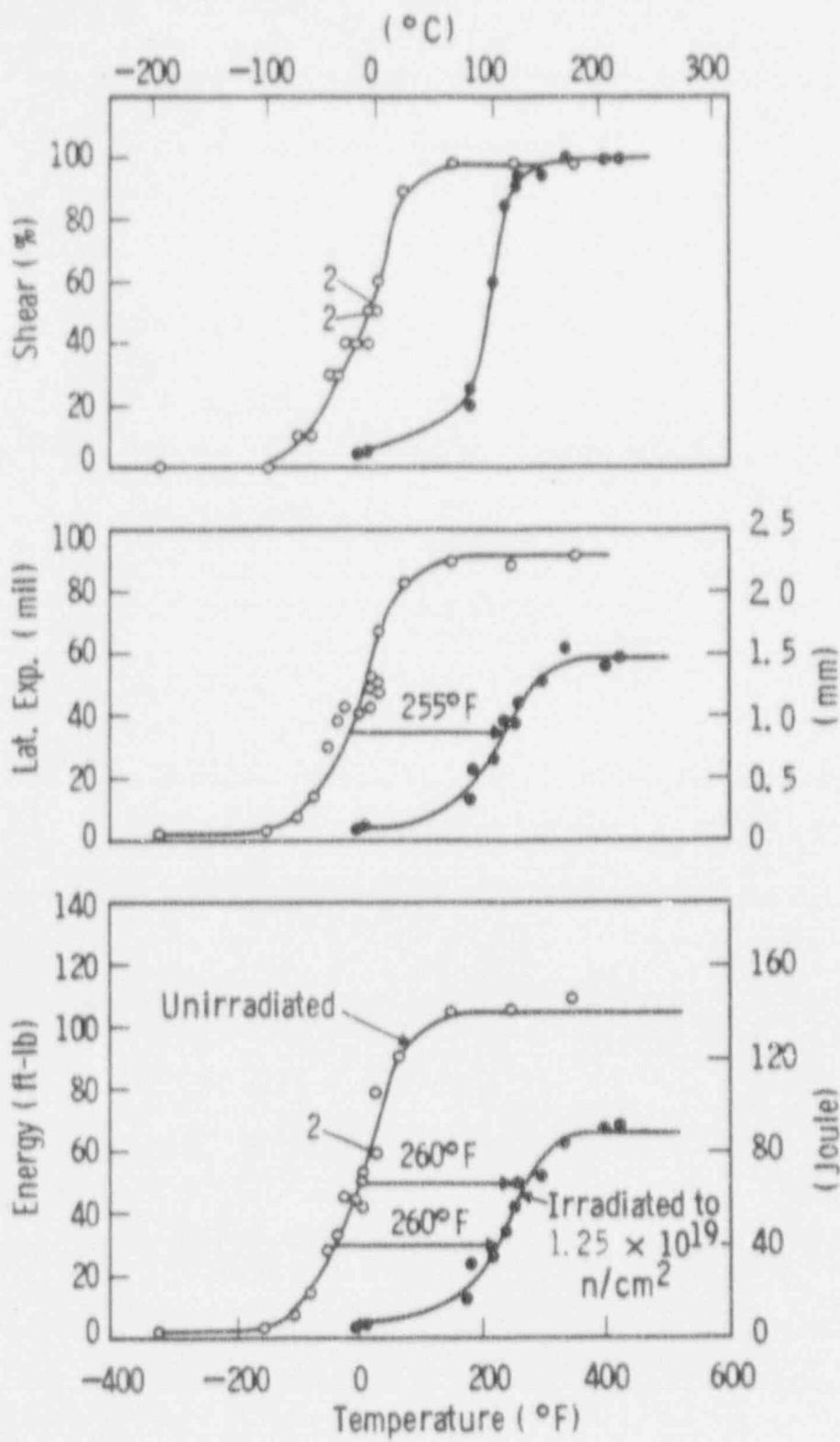Vessel Intermediate Shell Plate D-8406-1 (Longitudinal Orientation)

Figure 5-3.   Charpy V-Notch Impact Properties for Maine Yankee Reactor Vessel Weld Metal
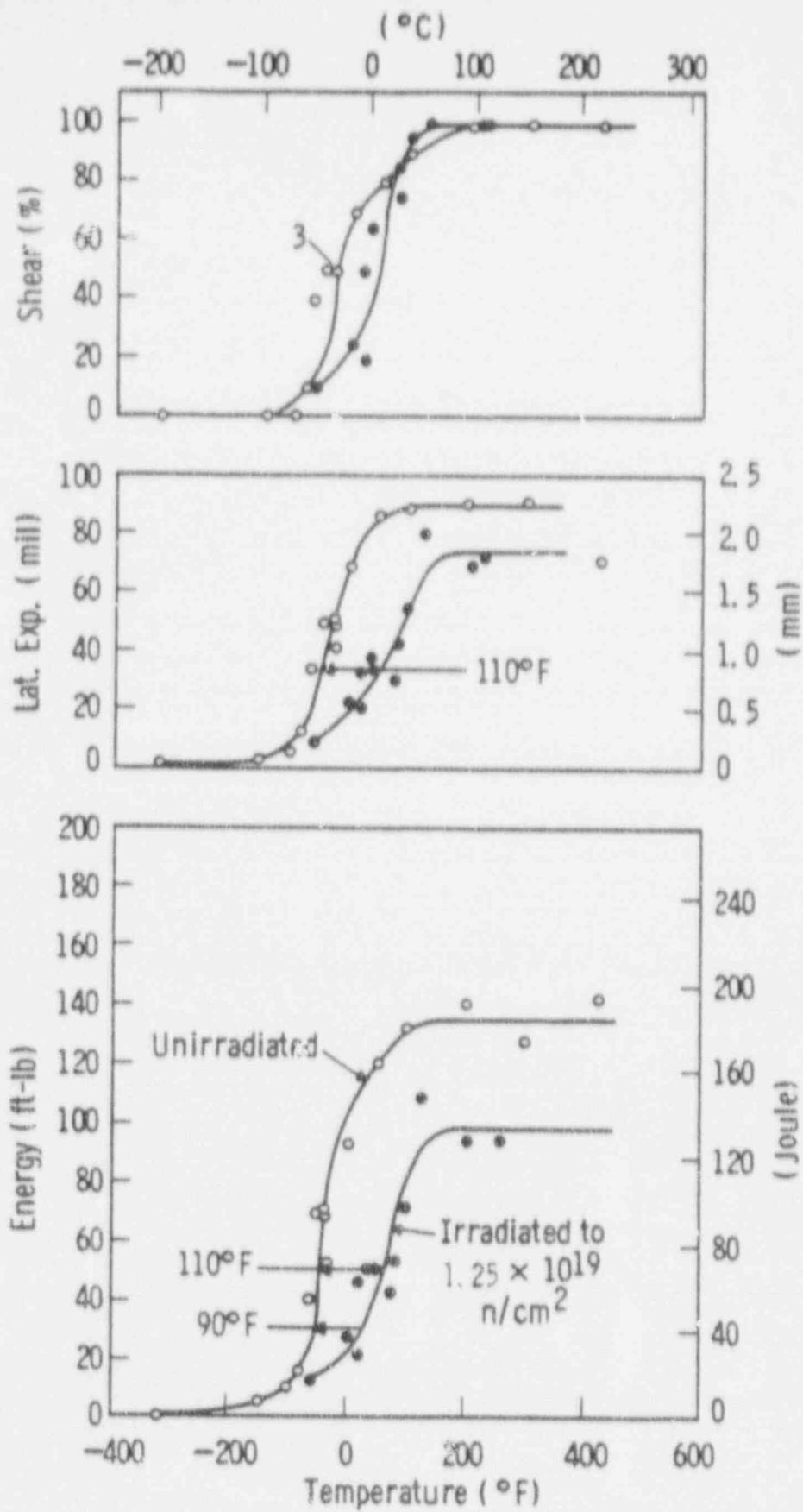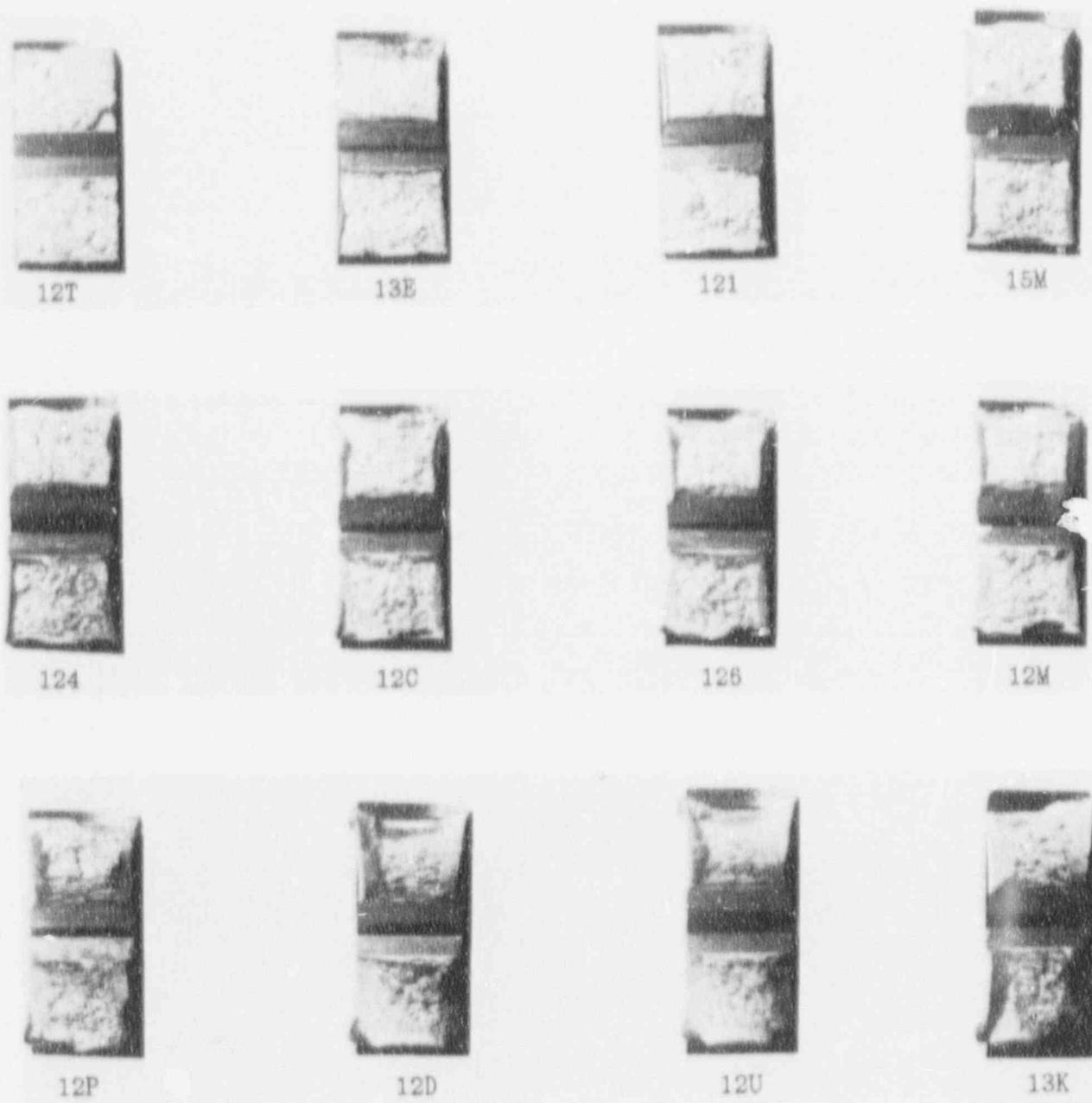
Figure 5-4.   Charpy V-Notch Impact Properties for Maine Yankee Reactor Weld
Heat Affected Zone Metal
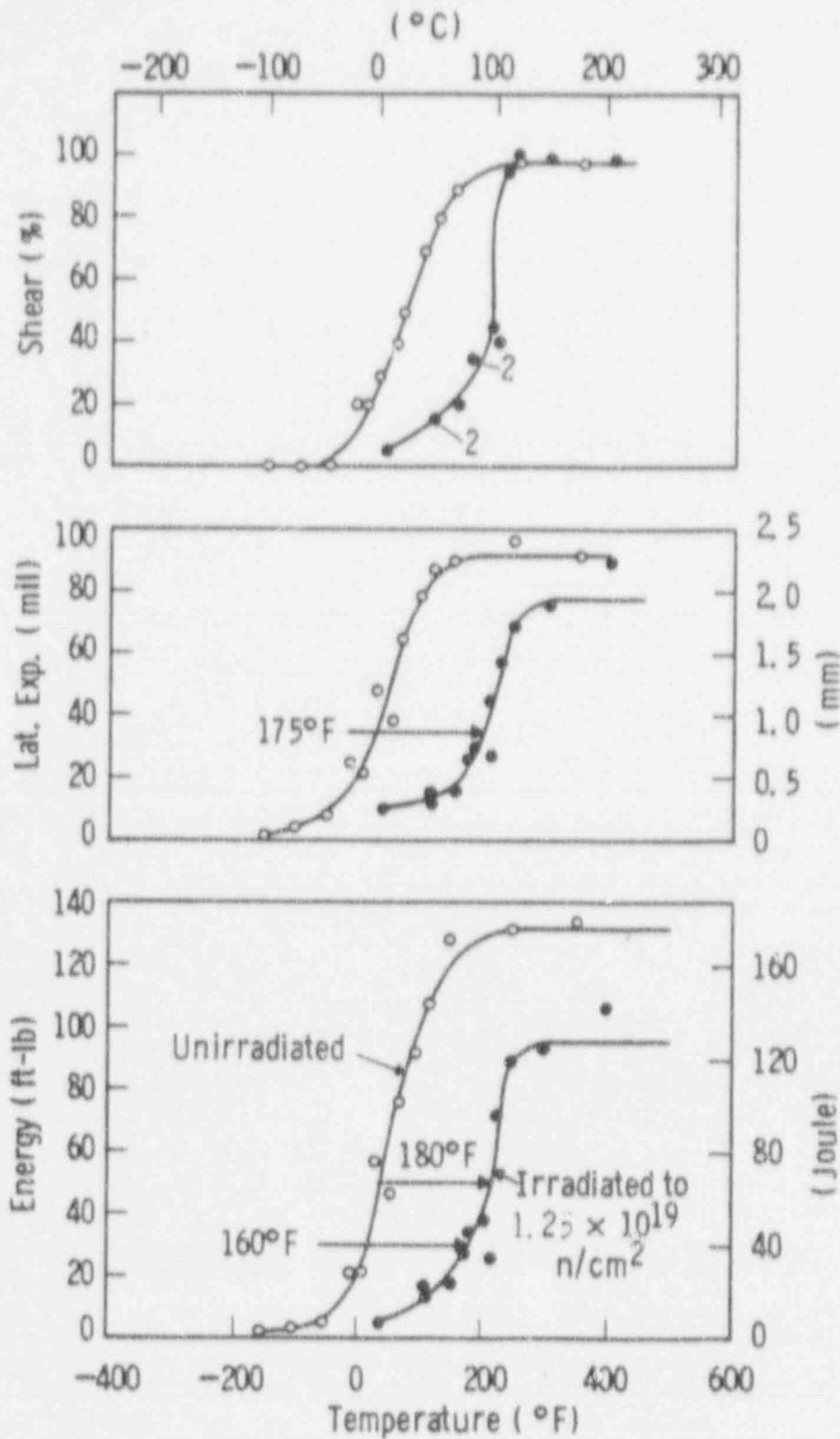
Figure 5-5.   Charpy V-Notch Impact Properties for the Maine Yankee Standard
Reference Material (HSST – Plate 01)

Figure 5-6. Charpy V-Notch Impact Specimen Fracture Surfaces for the Maine Yankee Reactor Vessel Intermediate Shell Plate D-8406-1 (Longitudinal Orientation)

37K   31J   32L   32A
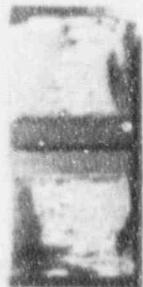
35D   32K   3A2   33M
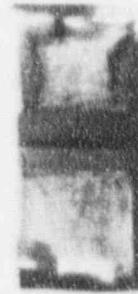
37J   317   36C   312

Figure 5-7.   Charpy Impact Specimen Fracture Surfaces for Maine Yankee
Reactor Vessel Weld Metal

Figure 5-8.   Charpy Impact Specimen Fracture Surfaces for Maine Yankee
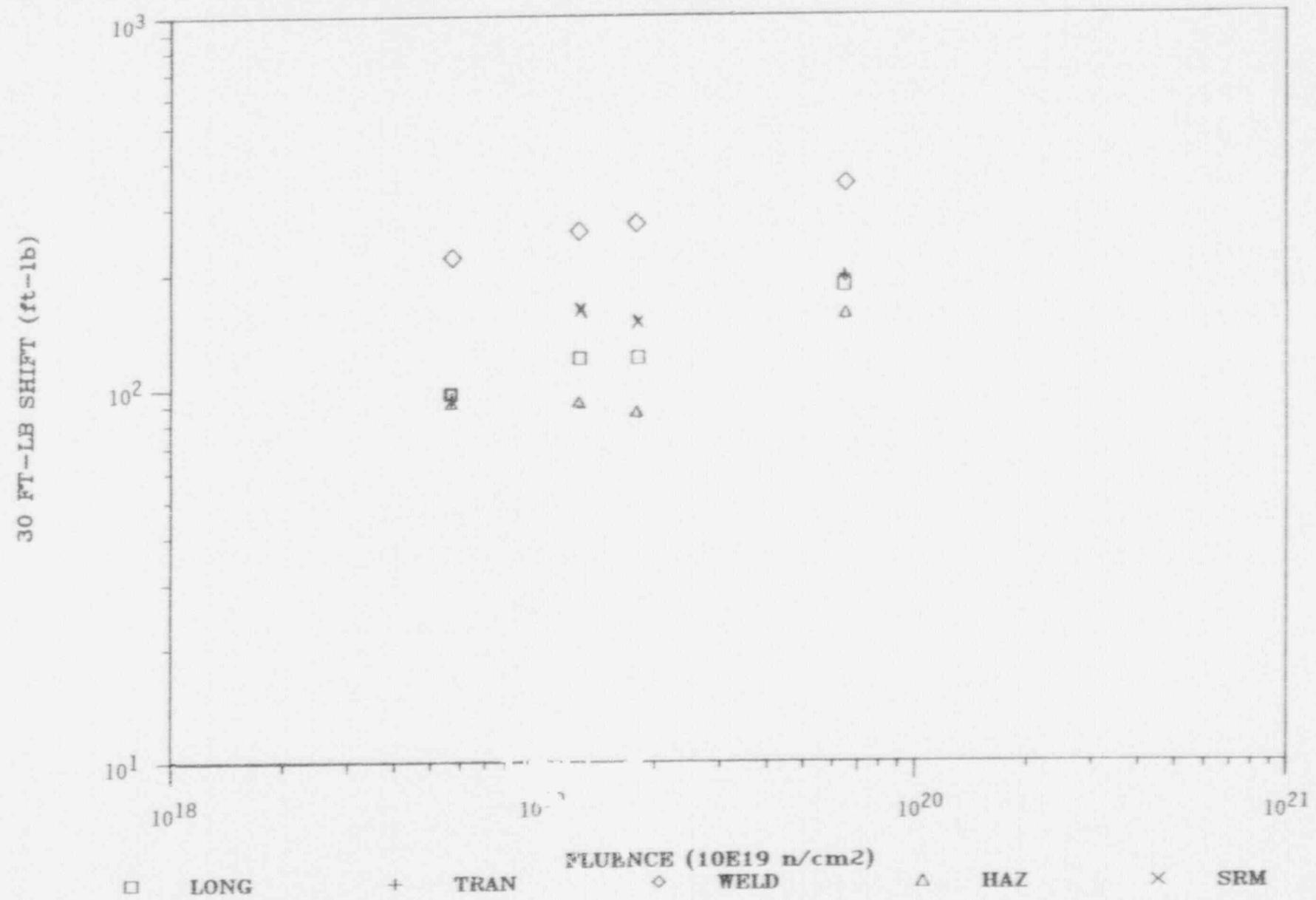   Reactor Vessel Weld Heat Affected Zone HAZ Metal
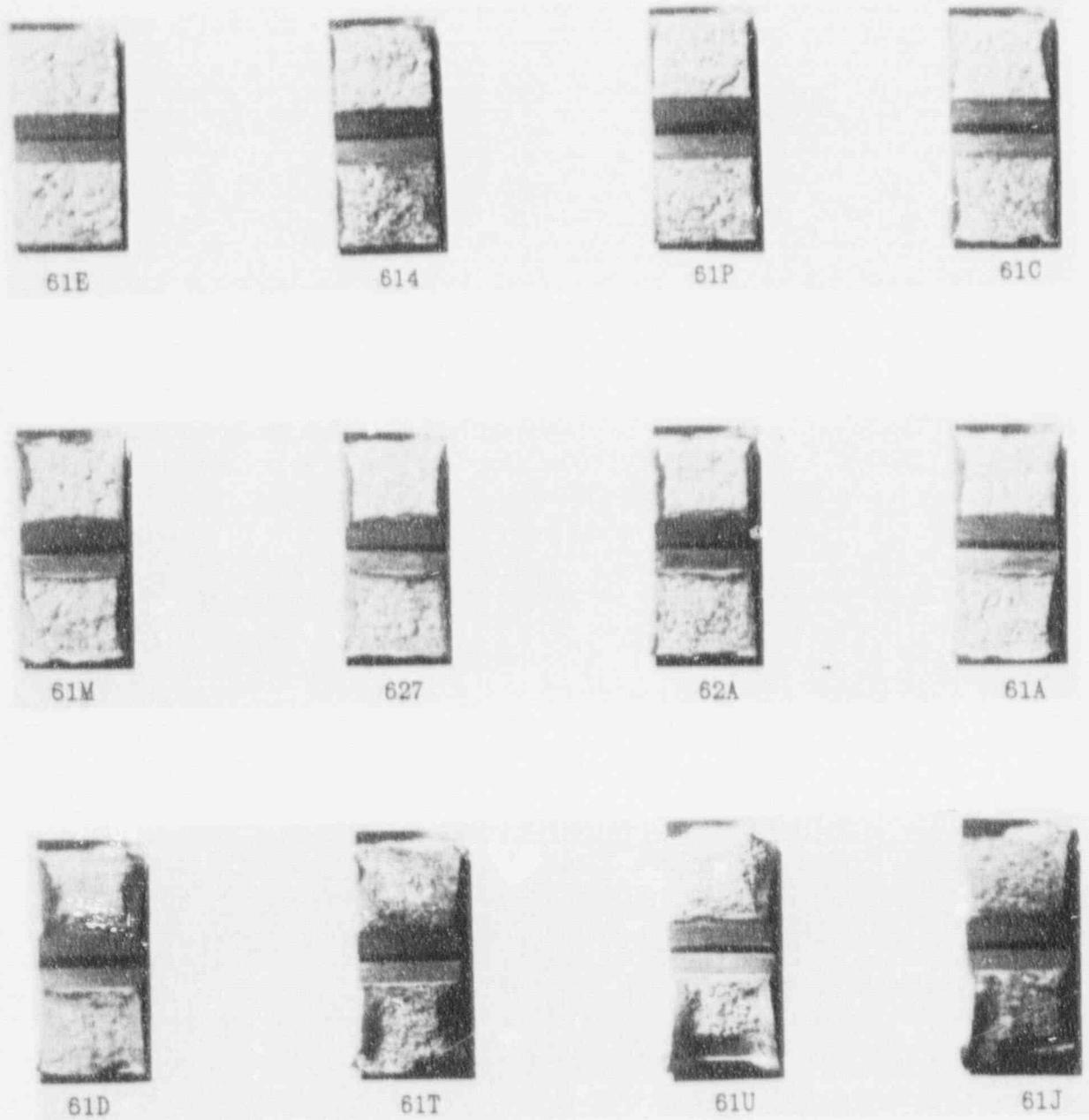
Figure 5-9. Charpy Impact Specimen Fracture Surfaces for Maine Yankee
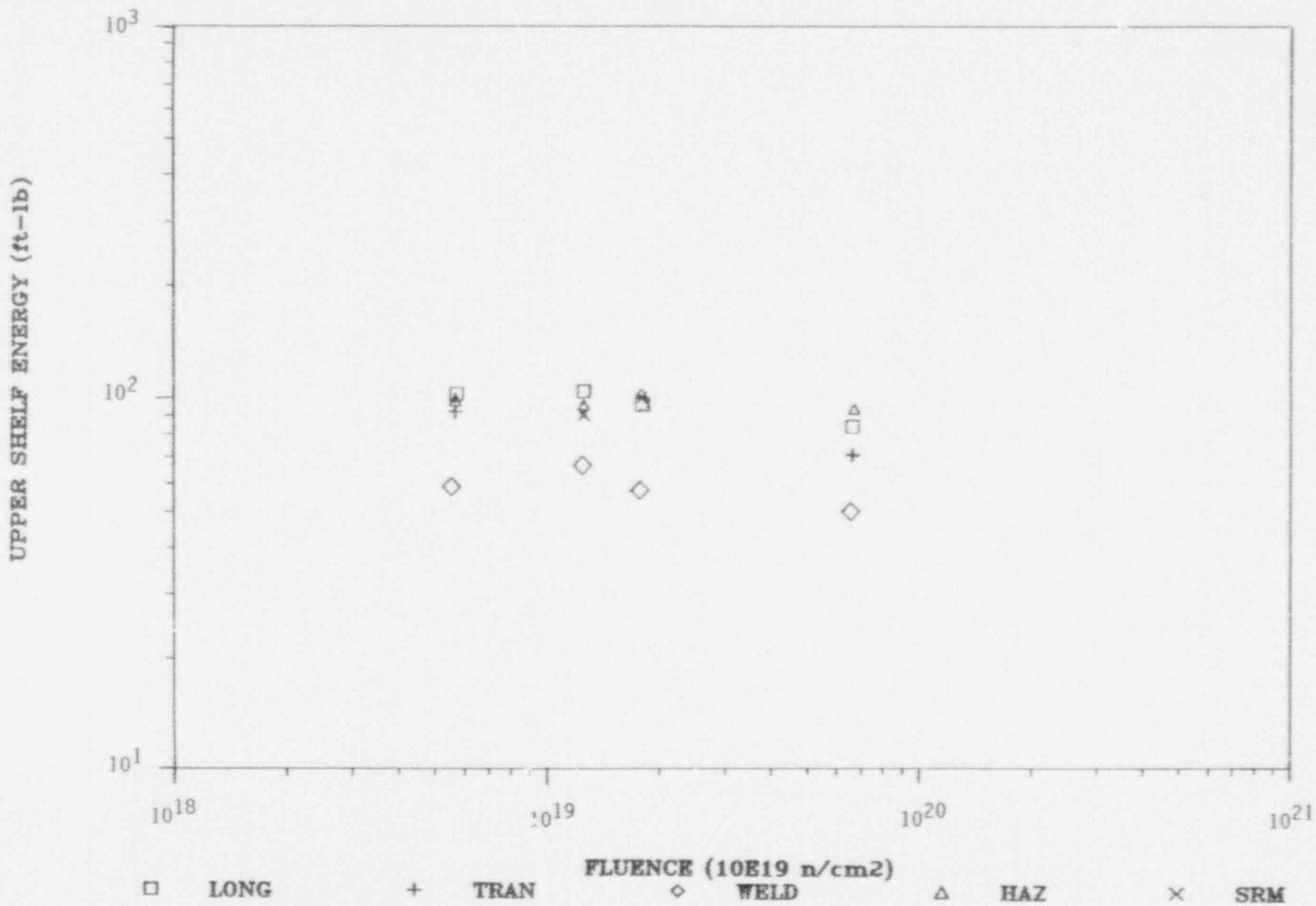Standard Reference Material (HSST - Plate 01)

RM-24668

FIGURE 5-10

MEASURED 30 FT-LB TRANSITION TEMP.

SHIFT vs. FLUENCE

FIGURE 5-11

MEASURED UPPER SHELF ENERGY
DECREASE vs. FLUENCE

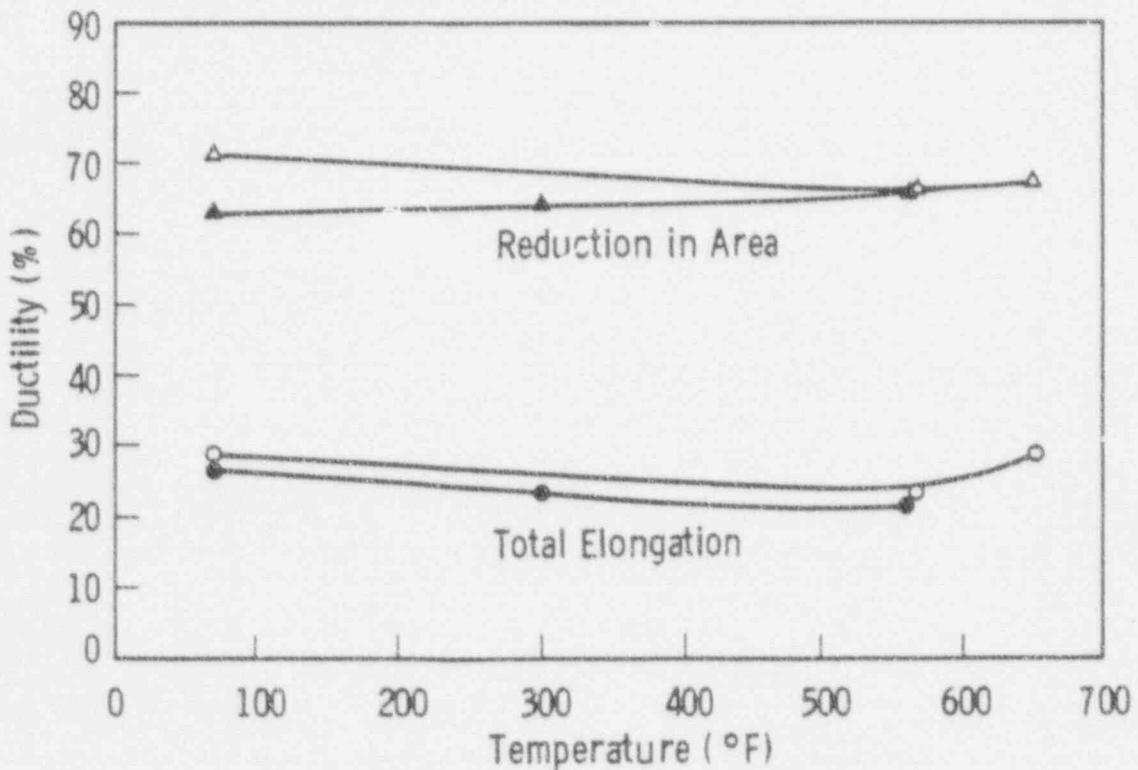FIGURE 5-12

MEASURED UPPER SHELF ENERGY
PERCENT DECREASE vs. FLUENCE
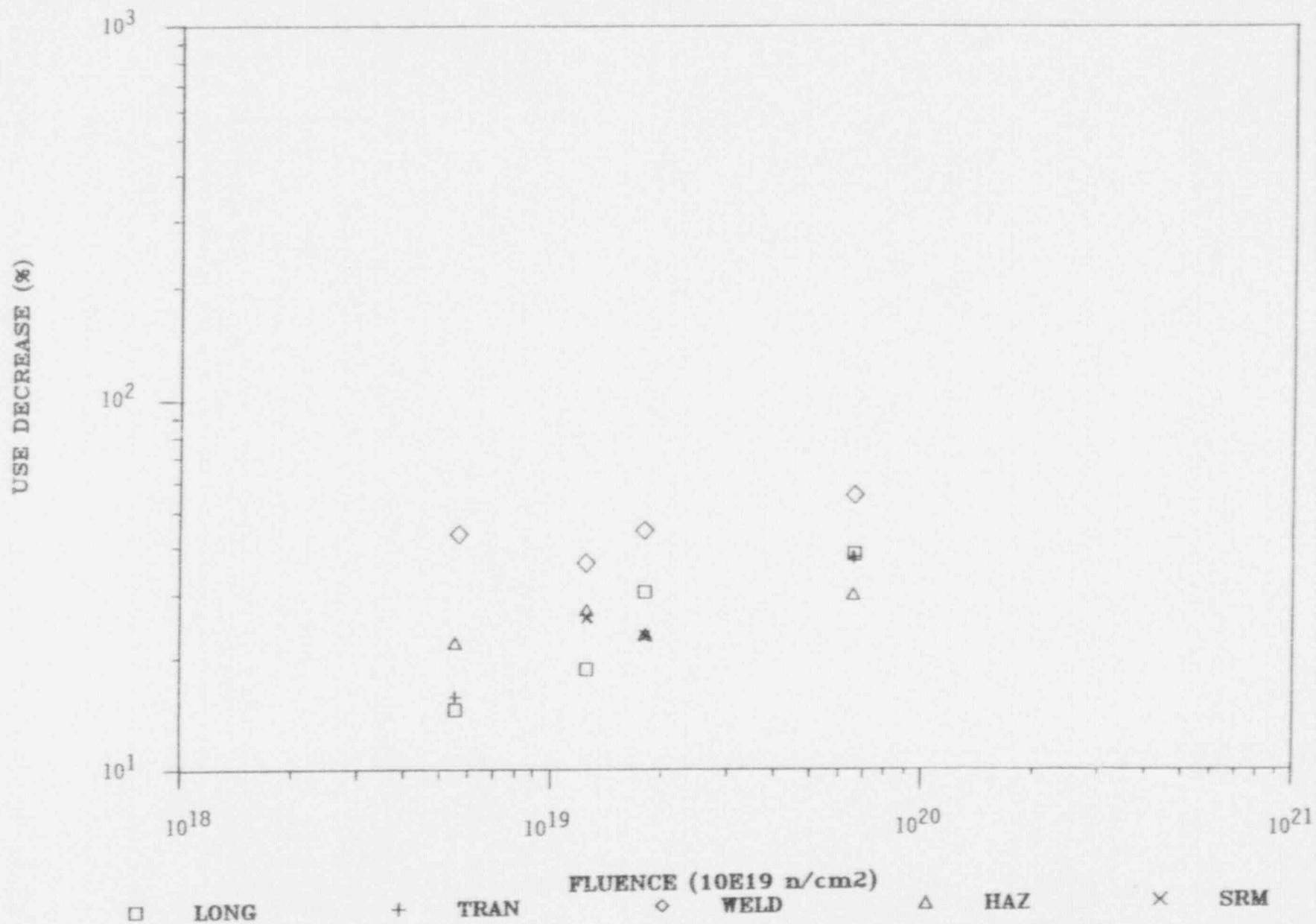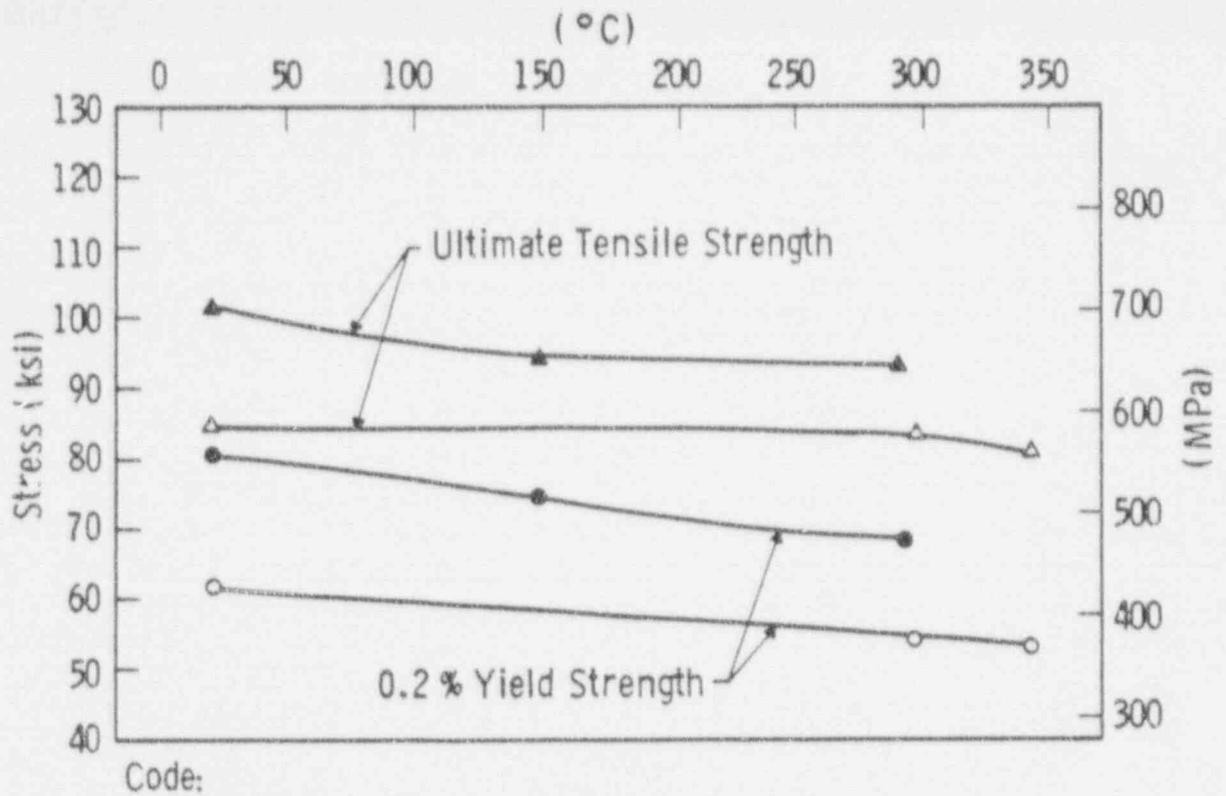
Figure 5-13.   Tensile Properties for the Maine Yankee Reactor Vessel
Intermediate Shell Plate D-8406-1 (Longitudinal Orientation)
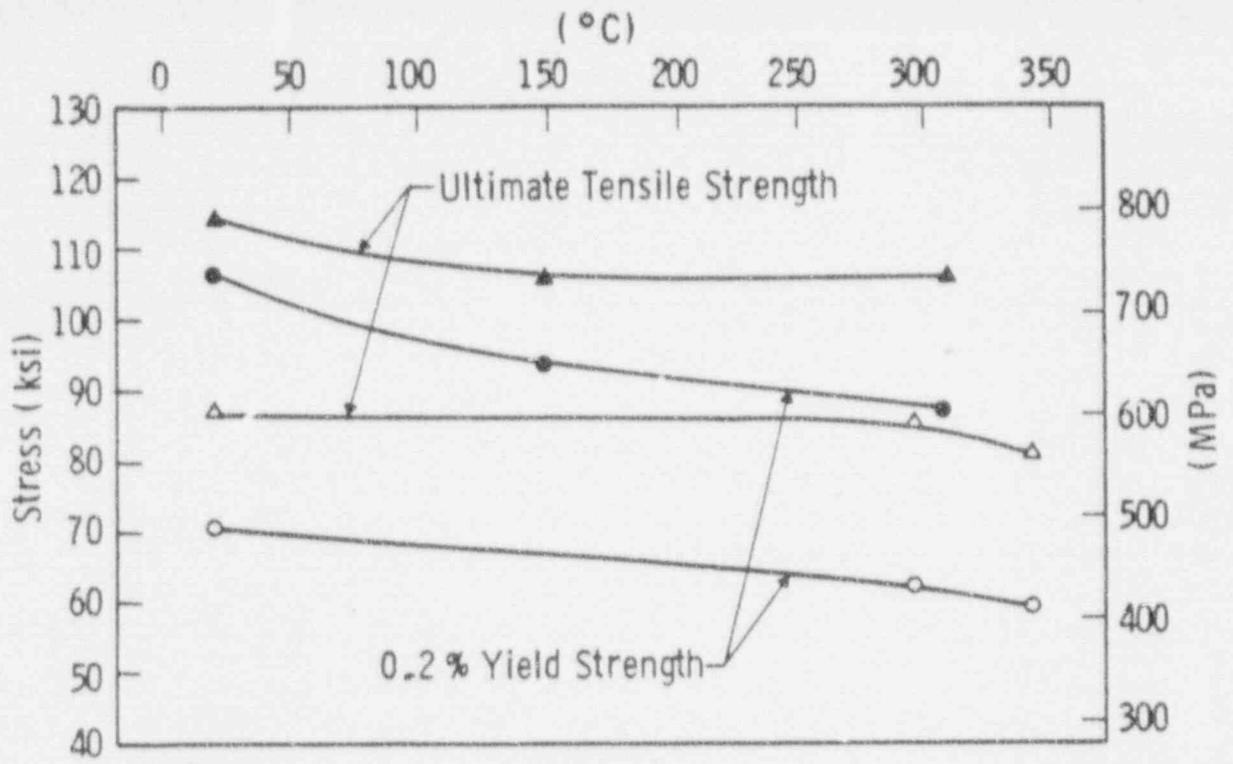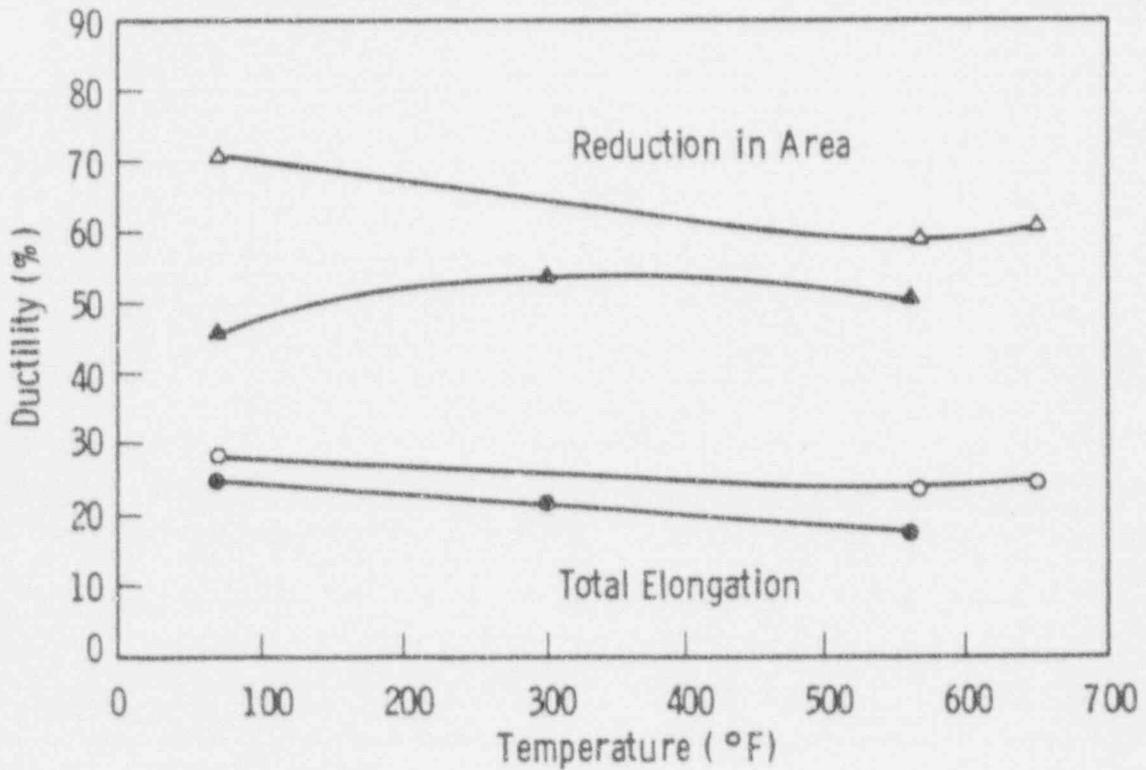
Figure 5-14.   Tensile Properties for Maine Yankee Reactor Vessel Weld Metal
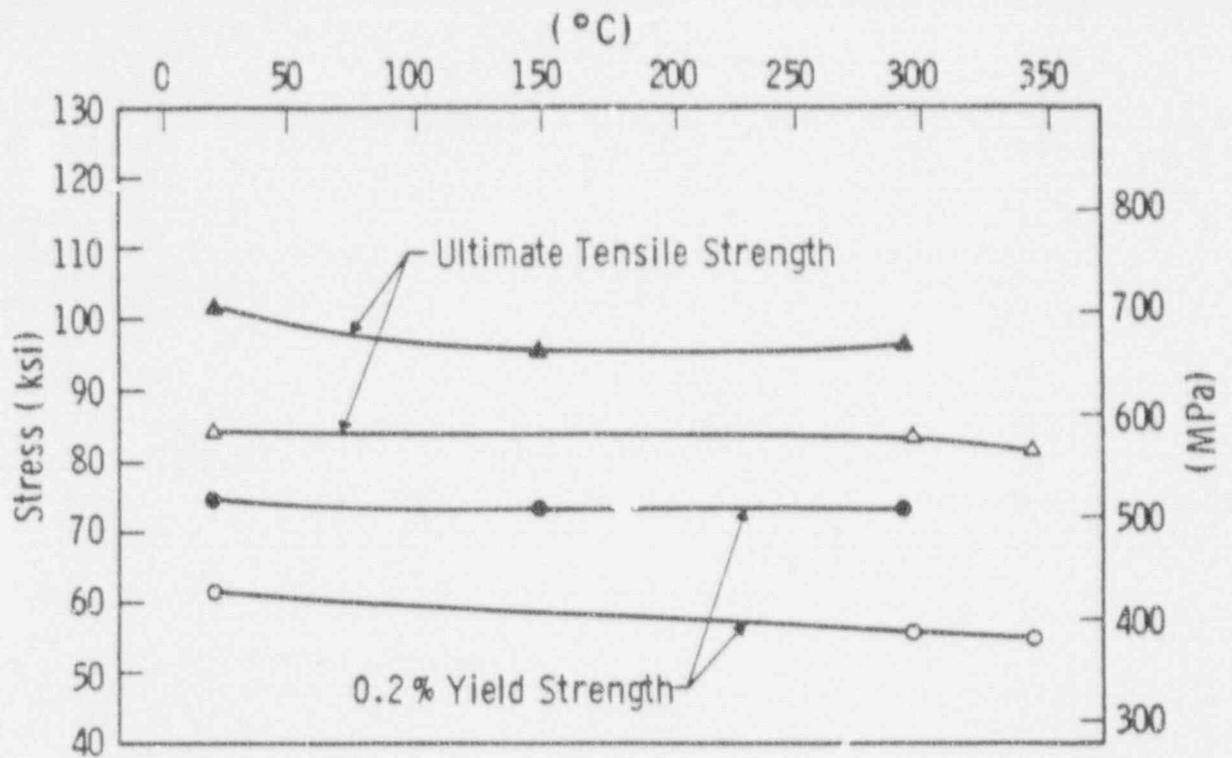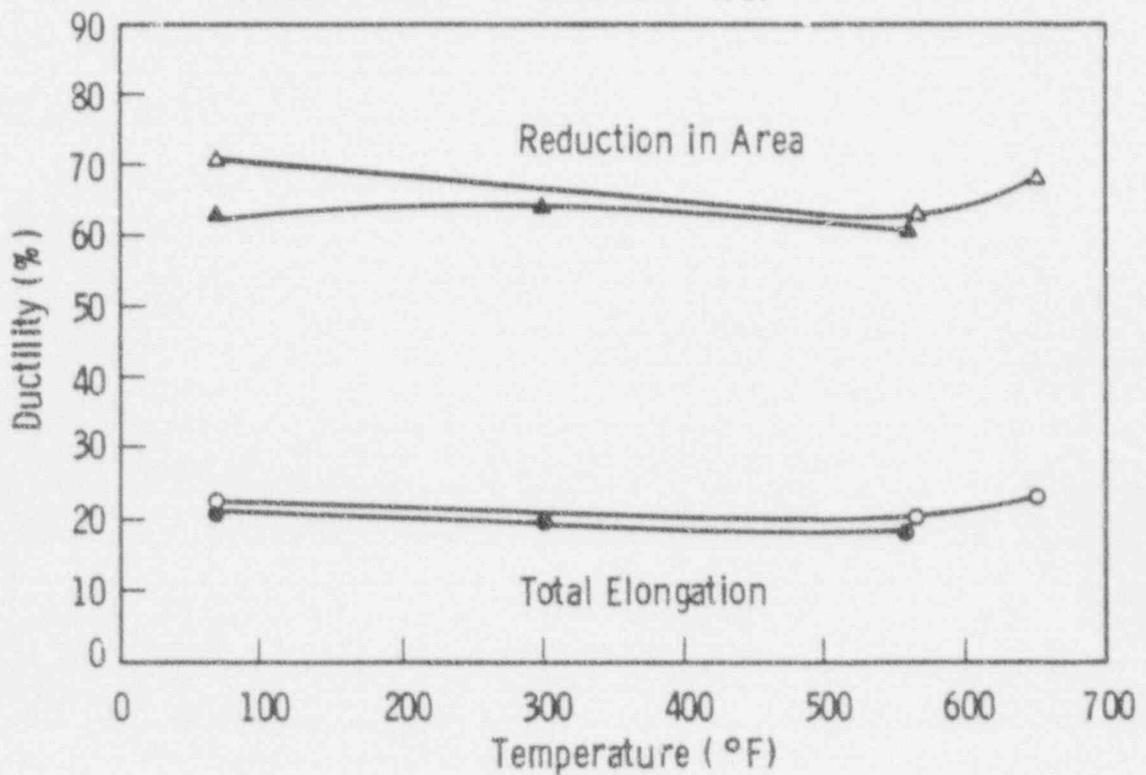
Figure 5-15. Tensile Properties for Maine Yankee Reactor Vessel Weld Heat
            Affected Zone (HAZ) Metal
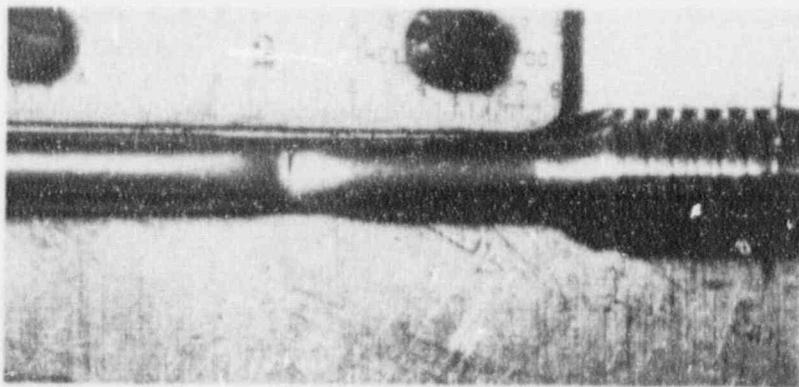
Specimen 1L4   74°F
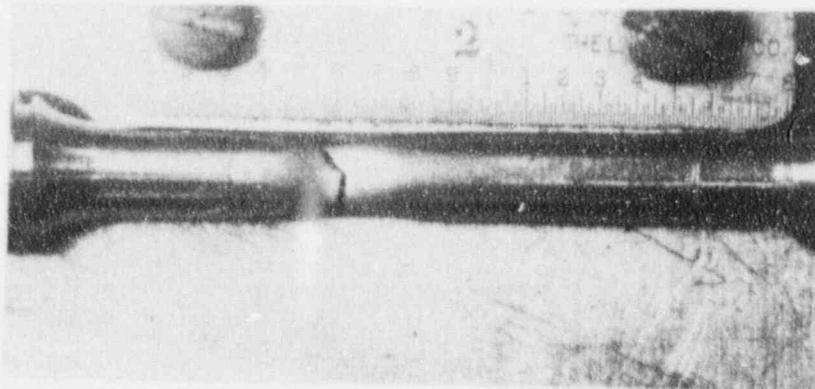
Specimen 1KM   300°F
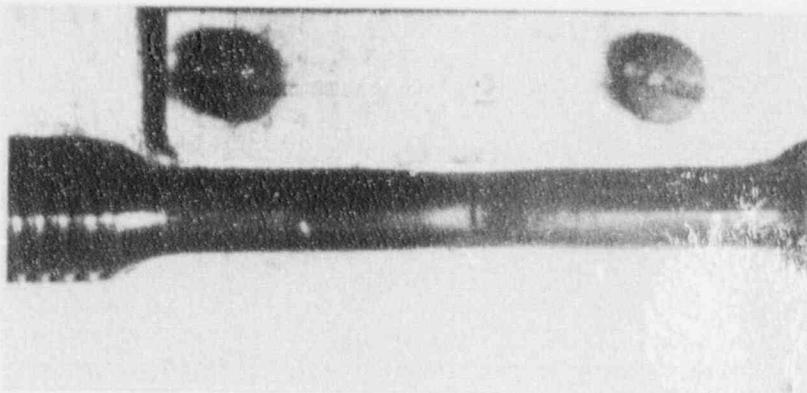
Specimen 1JL   550°

Figure 5-16.   Fractured Tensile Specimens from the Maine Yankee Reactor Vessel Intermediate Shell Plate D-8406-1 (Longitudinal Orientation)

RM-24669
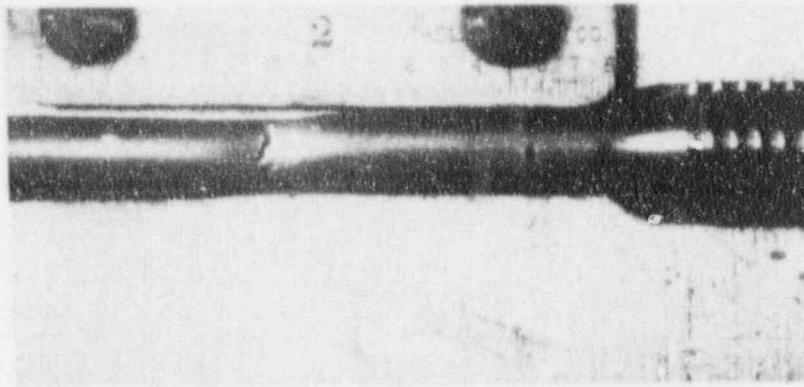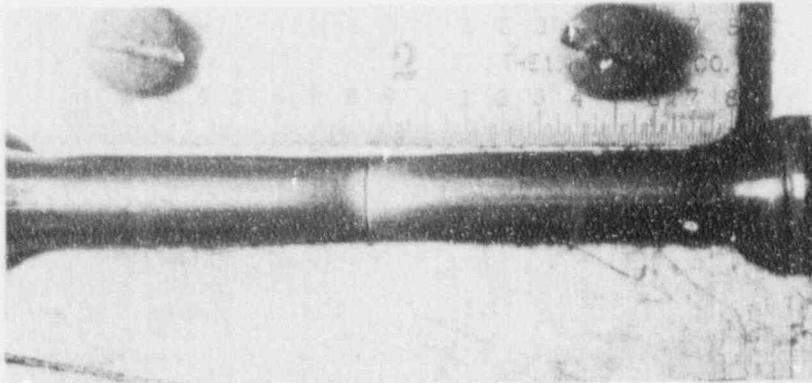
Specimen 3JM   74°F

Specimen 3J7   300°F
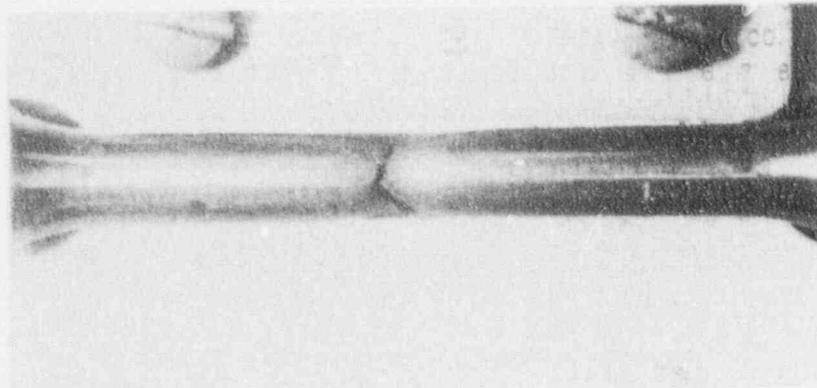
Specimen 3L4   550°F

Figure 5-17.   Fractured Tensile Specimens from Maine Yankee Reactor Vessel
Weld Metal

RM-24670
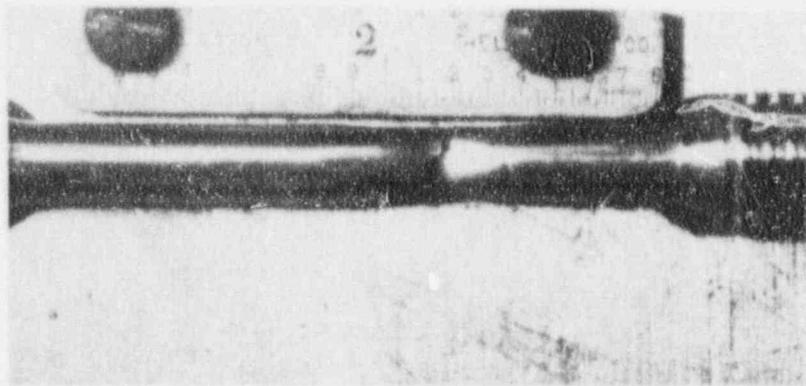
Specimen 4L2   74°F
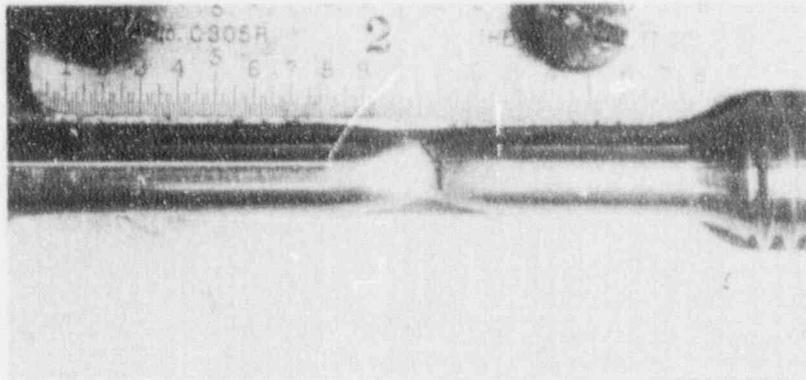
Specimen 4KK   300°F
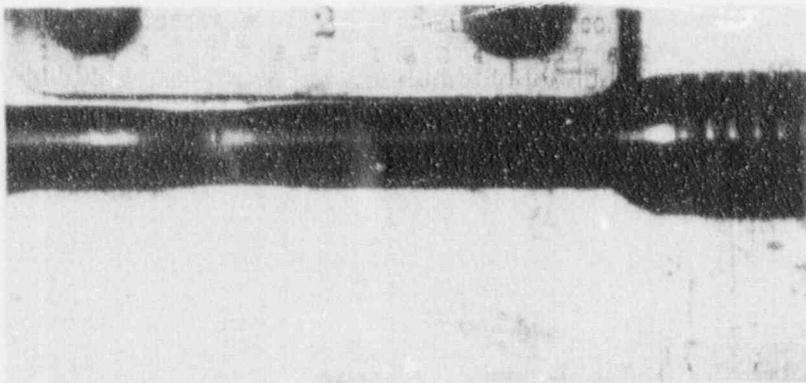
Specimen 4K1   550°F

Figure 5-18.   Fractured Tensile Specimens from the Maine Yankee Reactor Vessel
                 Weld Heat Affected Zone (HAZ) Metal
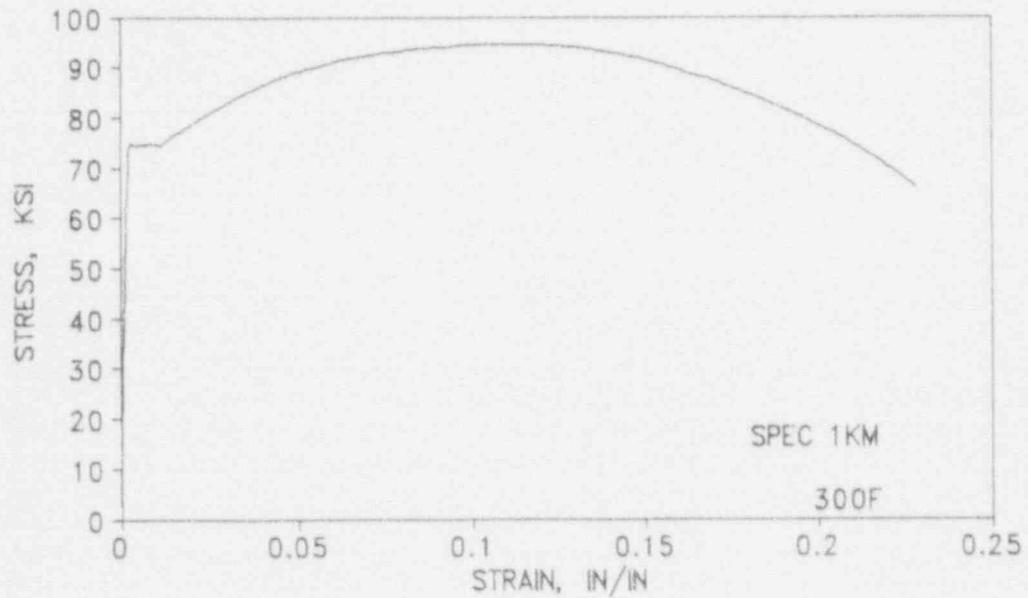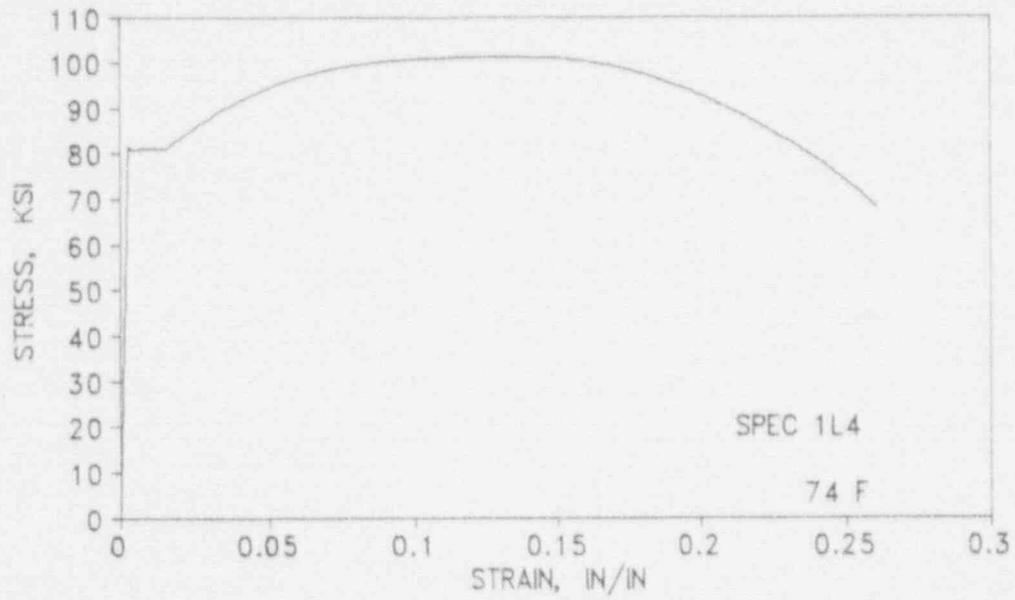
Figure 5-19. Engineering Stress-Strain Curves for the Maine Yankee Reactor Vessel Intermediate Shell Plate D-8406-1 Tension Specimens 1L4 and 1KM
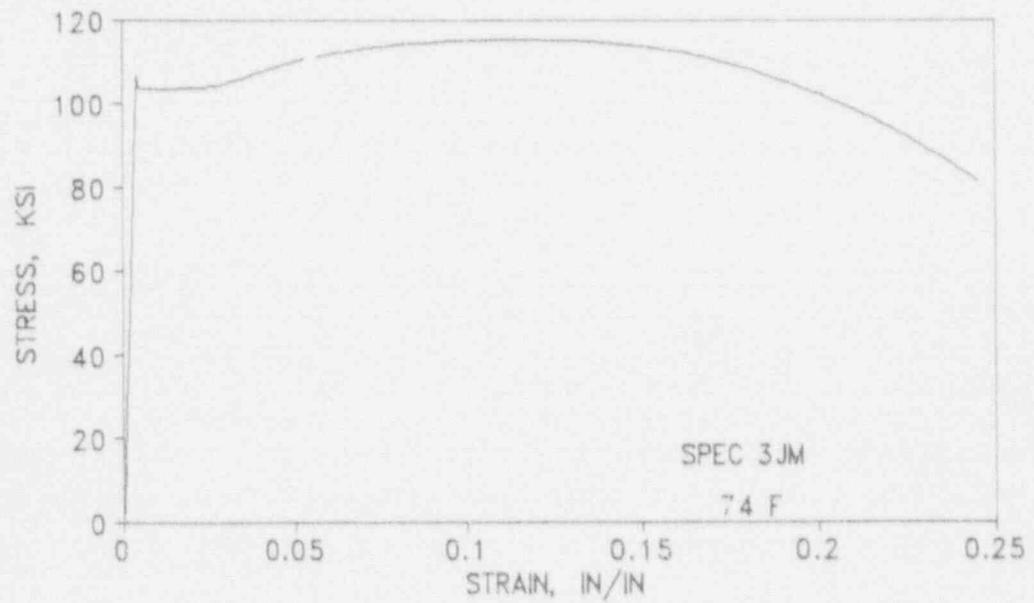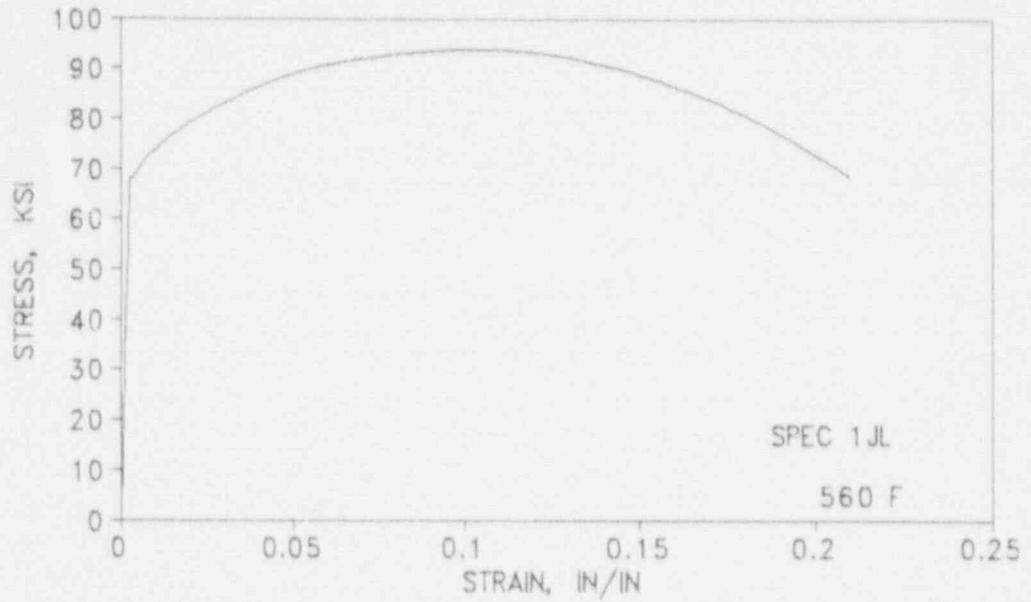
Figure 5-20.   Engineering Stress-Strain Curves for the Maine Yankee Reactor
                Vessel Intermediate Shell Plate D-8406-1 Tension Specimen 1JL
                and Weld Metal Tension Specimen 3JM

5-40
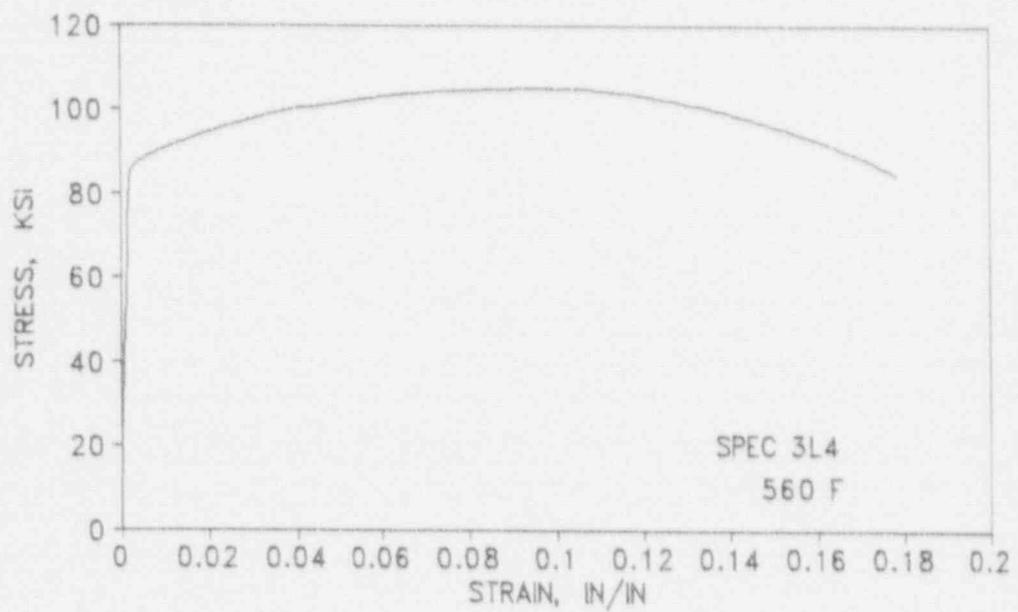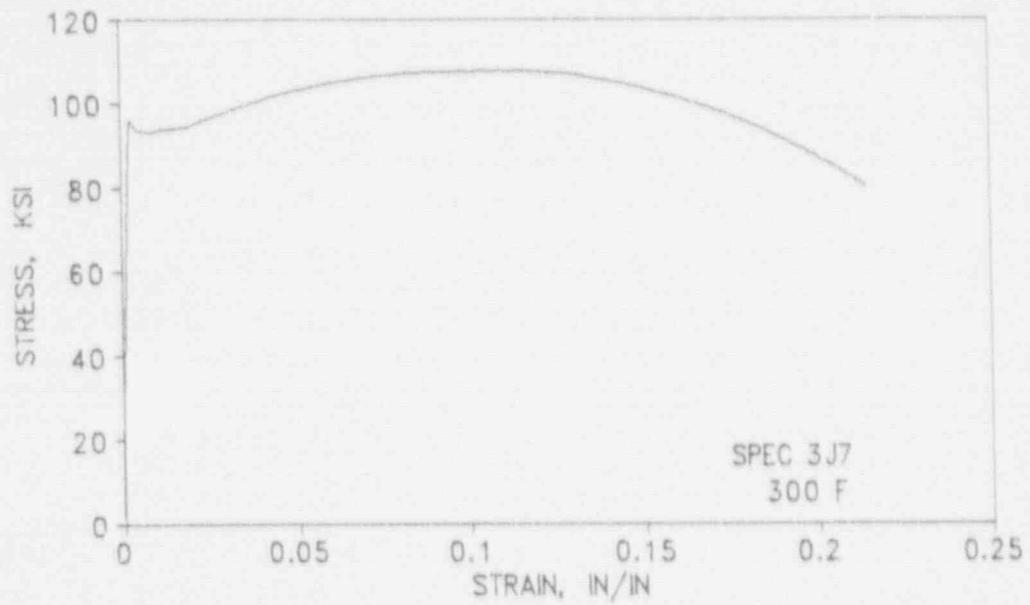
Figure 5-21.   Engineering Stress-Strain Curves for the Maine Yankee Reactor
Vessel Weld Metal Tension Specimens 3J7 and 3L4
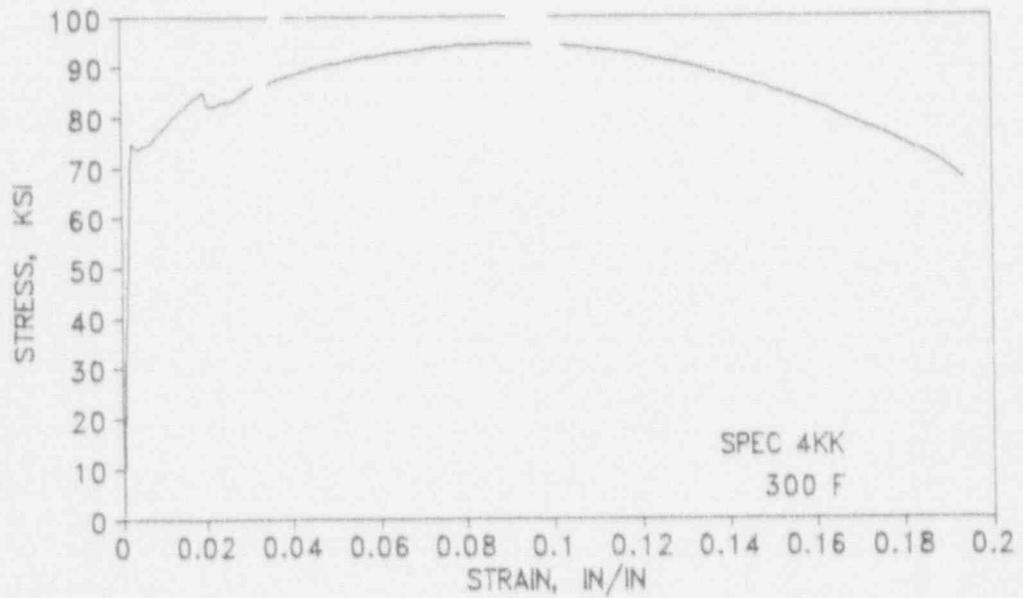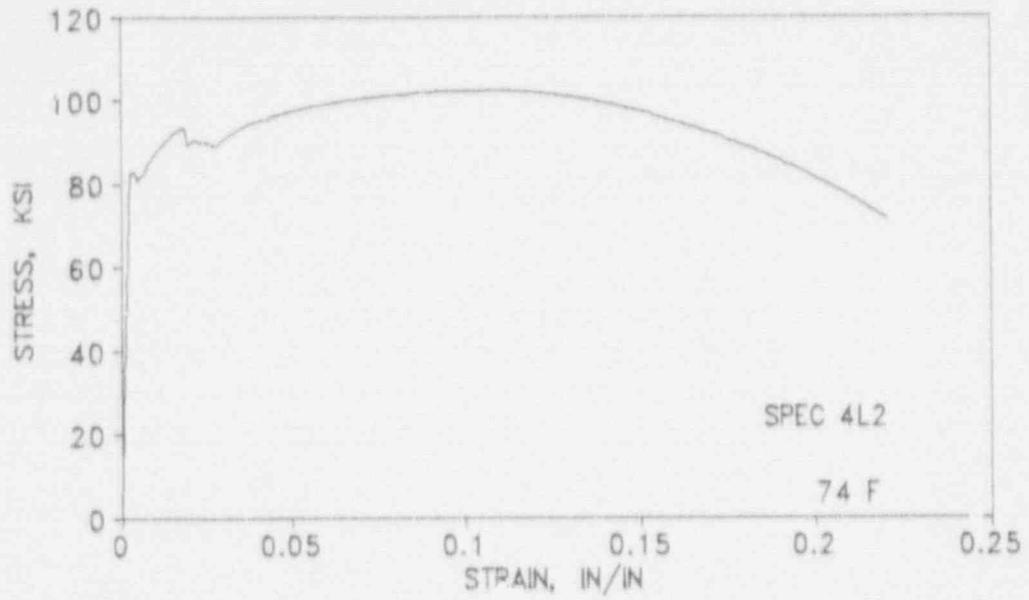
Figure 5-22. Engineering Stress-Strain Curves for the Maine Yankee Reactor Vessel Weld Heat Affected Zone (HAZ) Metal Tension Specimens 4L2 and 4KK
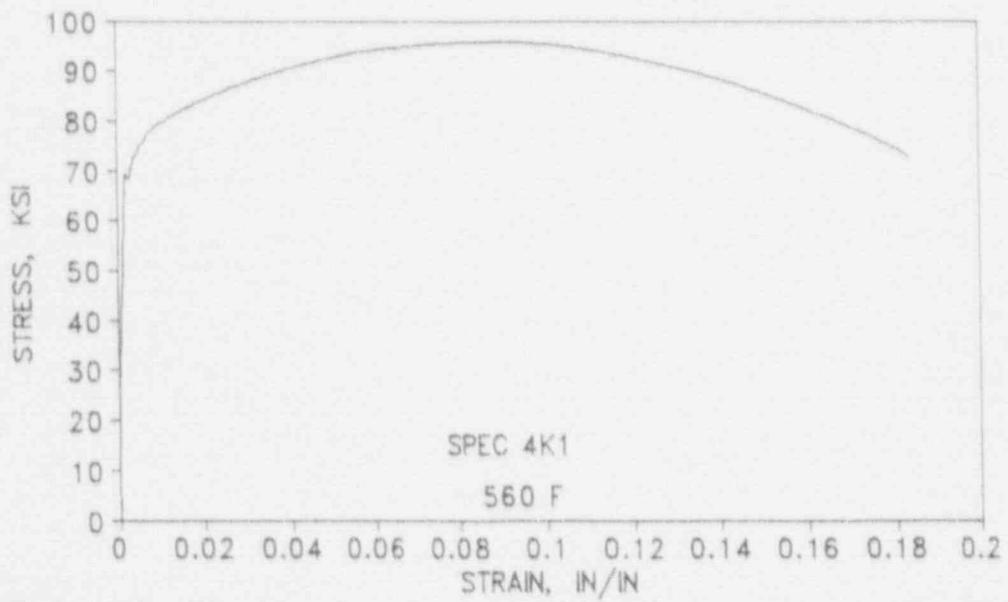
Figure 5-23. Engineering Stress-Strain Curve for the Maine Yankee Reactor
Vessel Weld Heat Affected Zone (HAZ) Metal Tension Specimen 4K1

SECTION 6.0
RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 Introduction

Knowledge of the neutron environment within the reactor pressure vessel
and surveillance capsule geometry is required as an integral part of LWR
reactor pressure vessel surveillance programs for two reasons. First, in
order to interpret the neutron radiation-induced material property changes
observed in the test specimens, the neutron environment (energy spectrum,
flux, fluence) to which the test specimens were exposed must be known.
Second, in order to relate the changes observed in the test specimens to
the present and future condition of the reactor vessel, a relationship
must be established between the neutron environment at various positions
within the reactor vessel and that experienced by the test specimens. The
former requirement is normally met by employing a combination of rigorous
analytical techniques and measurements obtained with passive neutron flux
monitors contained in each of the surveillance capsules. The latter
information is derived solely from analysis.

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured
materials properties changes to the neutron exposure of the material for
light water reactor applications has traditionally been accepted for
development of damage trend curves as well as for the implementation of
trend curve data to assess vessel condition. In recent years, however, it
has been suggested that an exposure model that accounts for differences in
neutron energy spectra between surveillance capsule locations and
positions within the vessel wall could lead to an improvement in the
uncertainties associated with damage trend curves as well as to a more
accurate evaluation of damage gradients through the pressure vessel wall.

Because of this potential shift away from a threshold fluence toward an
energy dependent damage function for data correlation, ASTM Standard
Practice E853, "Analysis and Interpretation of Light Water Reactor
Surveillance Results," recommends reporting displacements per iron atom

(dpa) along with fluence (E > 1.0 MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom." The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the pressure vessel wall has already been promulgated in Revision 2 to the Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials."

This section provides the results of the neutron dosimetry evaluations performed in conjunction with the analysis of test specimens contained in Capsule W-253. Fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV), fast neutron fluence (E > 0.1 Mev), and iron atom displacements (dpa) are established for the capsule irradiation history. The analytical formalism relating the measured capsule exposure to the exposure of the vessel wall is described and used to project the integrated exposure of the vessel itself. Also uncertainties associated with the derived exposure parameters at the surveillance capsule and with the projected exposure of the pressure vessel are provided.

6.2 Discrete Ordinates Analysis

A plan view of the reactor geometry at the core midplane is shown in Figure 4-2. A plan view of Capsule W-253 is shown in Figure 6-1. In regard to the azimuthal locations of the surveillance capsules and vessel structures, it should be noted that positions; i.e. 253 degrees, are referenced to "pressure vessel 0.0" as indicated in the vessel design drawings. In performing neutron transport analyses, however, it is common practice to reference azimuthal 0.0 to an axis extending radially through the core flats. In the case of Maine Yankee, these two coordinate systems are displaced by 10 degrees. Thus, Capsule W-253 is located at 7 degrees relative to core 0.0. Likewise, vessel inner radius positions of 260, 275, 290, and 305 degrees relative to vessel 0.0 translate to 0, 15, 30, and 45 degrees in the core coordinate system.

From a neutron transport standpoint, the surveillance capsule structures are significant. They have a marked effect on both the distribution of neutron flux and the neutron energy spectrum in the water annulus between the neutron pad and the reactor vessel. In order to properly determine the neutron environment at the test specimen locations, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, two distinct sets of transport calculations were available from prior analyses[16]. The first, consisting of two computations in the conventional forward mode, was used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters (ϕ(E > 1.0 Mev), ϕ(E > 0.1 Mev), and dpa) through the vessel wall. The neutron spectral information was required for the interpretation of neutron dosimetry withdrawn from the surveillance capsule as well as for the determination of exposure parameter ratios; i.e., dpa/ϕ(E > 1.0 MeV), within the pressure vessel geometry. The relative radial gradient information was required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consisted of a series of adjoint analyses relating the fast neutron flux (E > 1.C MeV) at surveillance capsule positions, and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provided the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for the first 11 cycles of irradiation; and established the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses included not only spatial variations of fission rates within the reactor core; but, also accounted for the effects

of varying neutron yield per fission and fission spectrum introduced by the build-up of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra and radial distribution information from the forward calculation provided the means to:

1. Evaluate neutron dosimetry obtained from surveillance capsule locations.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

3. Enable a direct comparison of analytical prediction with measurement.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

The forward transport calculation for the reactor model summarized in Fi 4-2 and 6-1 was carried out in R, θ geometry using the DOT two-dimensio discrete ordinates code[17] and the SAILOR cross-section library[18]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotopic scattering was treated with a P_3 expansion of the cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature.

In regard to the geometric model used in the transport analysis, it should be noted that in performing the current capsule evaluation an explicit model of Capsule W-253 was not included in the geometric model. However, in the prior evaluations documented in reference 16, a model of Capsule W-263 was included. Since the forward transport results at the capsule location are used only to establish a relative energy distribution for use as an a priori guess in the spectrum adjustment procedure described later in this section, it was judged that the W-263 degree spectrum would be sufficient and that expenditure of additional resources to update the forward calculation was not warranted. Note that relative to the core 0.00 azimuthal position the two capsules are only 4

degrees apart; i. e., they are positioned at 3 and 7 degrees. Maine Yankee Atomic Power Company concurred with this judgement. Due to the proximity of the two capsules, the assumption of a similar spectrum is valid and is supported by the measured reaction rate ratios from the respective capsule dosimetry sets[37].

In the analyses documented in reference 16, forward calculations were carried out for core designs representative of conventional out-in and low leakage fuel management. Plant specific core power distributions utilized in the forward analyses were based on a burnup weighted average of the cycles 1-6 core design for the first computation and on the cycle 8 fuel management for the second. The cycle 1-6 analysis was representative of reactor operation prior to the implementation of low leakage fuel management, while the cycle 8 calculation was typical of low leakage operation. Results of these two evaluations showed that the relative neutron spectra in the capsules and vessel wall are not significantly impacted by the implementation of low leakage core designs

All adjoint analyses were also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. Adjoint source locations were chosen at several azimuthal locations along the pressure vessel inner radius as well as the geometric center of the surveillance capsule. Again, these calculations were run in R, θ geometry to pro.ide neutron source distribution importance functions for the exposure parameter of interest; in this case, ϕ (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$R (r, \theta) = \int_r \int_\theta \int_E I(r, \theta, E) \, S (r, \theta, E) \, r \, dr \, d\theta \, dE$$

where: $R (r, \theta)$ = ϕ (E > 1.0 MeV) at radius r and azimuthal angle θ

$I (r, \theta, E)$ = Adjoint importance function at radius, r, azimuthal angle θ, and neutron source energy E.

$S (r, \theta, E)$ = Neutron source strength at core location r, θ and energy E.

Although the adjoint importance functions used in the Maine Yankee analysis were based on a response function defined by the threshold neutron flux ($E > 1.0$ MeV), prior calculations[16] have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and the spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the ratio of dpa/ϕ ($E > 1.0$ MeV) is insensitive to changing core source distributions. In the application of these adjoint importance functions to the Maine Yankee reactor, therefore, the iron displacement rates (dpa) and the neutron flux ($E > 0.1$ MeV) were computed on a cycle specific basis by using dpa/ϕ ($E > 1.0$ MeV) and ϕ ($E > 0.1$ MeV)/ϕ ($E > 1.0$ MeV) ratios from the forward analysis in conjunction with the cycle specific ϕ ($E > 1.0$ MeV) solutions from the individual adjoint evaluations.

Again, with the forward analysis, an explicit adjoint library was not generated for the Capsule W-253 location. Rather, the Capsule W-263 library was used in conjunction with an azimuthal translation factor applied to the problem normalization to estimate the exposure at the center of Capsule W-253. This approach assumes that the relative neutron energy spectra at the center of the two capsules is the same and that the capsules are in close enough proximity that the ratio of the exposure rates at the capsule center is independent of core power distribution. In this case, considering core symmetry, the capsules are only 4 degrees apart in an area of the vessel that does not exhibit extreme azimuthal gradients in the observed exposure rates. Therefore, it was judged that using this method of calculation does not introduce significant uncertainty in the overall evaluation.

The reactor core power distribution and fission fractions by isotope used in the plant specific adjoint calculations were supplied by Maine Yankee Atomic Power Company[36].

Selected results from the neutron transport analyses performed for the Maine Yankee reactor are provided in Tables 6-1 through 6-5. The data listed in

these tables establish the means for absolute comparisons of analysis and measurement for the capsule irradiation period and provide the means to correlate dosimetry results with the corresponding neutron exposure of the pressure vessel wall.

In Table 6-1, the calculated exposure parameters [ϕ (E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa] are given at the geometric center of Capsule W-253. These plant specific data, based on the adjoint transport analysis, are meant to establish the absolute comparison of measurement with analysis. Similar data are given in Table 6-2 for the pressure vessel inner radius. Again, the three pertinent exposure parameters are listed for the cycle 1 through 11 plant specific power distributions. It is important to note that the data for the vessel inner radius were taken at the clad/base metal interface; and, thus, represent the maximum exposure levels of the vessel wall itself.

Radial gradient information for neutron flux (E > 1.C MeV), neutron flux (E > 0.1 MeV), and iron atom displacement rate is given in Tables 6-3, 6-4, and 6-5, respectively. The data, obtained from the forward neutron transport calculation, are presented on a relative basis for each exposure parameter at several azimuthal locations. Exposure parameter distributions within the wall may be obtained by normalizing the calculated or projected exposure at the vessel inner radius to the gradient data given in Tables 6-3 through 6-5.

For example, the neutron flux (E > 1.0 MeV) at the 1/4T position on the 0° azimuth is given by:

$$\phi_{1/4T}(0°) = \phi(221.\text{F} \quad \text{°}) \text{ F } (227.15, 0°)$$

where: $\phi_{1/4T}(0°)$ = Projected neutron flux at the 1/4T position on the 0° azimuth

ϕ (221.67,0°)   = Projected or calculated neutron flux at the vessel inner radius on the 0° azimuth.

F (227.15, 0°) = Relative radial distribution function from Table 6-3.

Similar expressions apply for exposure parameters in terms of ϕ (E > 0.1 MeV) and dpa/sec.

6.3 Neutron Dosimetry

The passive neutron sensors included in the Maine Yankee surveillance program are listed in Table 6-6. Also given in Table 6-6 are the primary nuclear reactions and associated nuclear constants that were used in the evaluation of the neutron energy spectrum within the capsule and the subsequent determination of the various exposure parameters of interest [ϕ (E > 1.0 Mev), ϕ (E > 0.1 MeV), dpa].

The relative locations of the neutron sensors within the capsules are shown in Figure 6-1. The individual sensors were placed in holes drilled in the dosimeter blocks at several axial levels within the capsules. All sensors were positioned at the radial center of the dosimetry blocks.

The use of passive monitors such as those listed in Table 6-6 does not yield a direct measure of the energy dependent flux level at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

o   The specific activity of each monitor.
o   The operating history of the reactor.
o   The energy response of the monitor.
o   The neutron energy spectrum at the monitor location.
o   The physical characteristics of the monitor.

The specific activity of each of the neutron monitors was determined using established ASTM procedures [19 through 32]. Following sample preparation and weighing, the activity of each monitor was determined by means of a lithium-drifted germanium, Ge(Li), gamma spectrometer.

In regard to the condition of the sensors removed from Capsule W-253, it is of interest to note that all of the uranium foils were oxidized and reduced to a powder form requiring dissolution and uranium separation to obtain accurate counting results. In addition, rapid oxidation of the titanium wires was observed upon removal from the capsule. In this case, the oxidation did not impair the ability to obtain measured specific activities. Also, in the case of Compartment 3541 the copper wire was fused with its cadmium jacket and could not be be recovered for analysis.

The irradiation history of the Maine Yankee reactor during cycles 1 through 11 was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable period and is given in Table 6-7. Measured and saturated reaction product specific activities as well as measured full power reaction rates are listed in Table 6-8. Reaction rate values were derived using the pertinent data from Tables 6-6 and 6-7. Due to its extremely short product half-life, no attempt was made to include reaction rates from the sulfur monitors in this evaluation.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [33]. The FERRET approach used the measured reaction rate data and the calculated neutron energy spectrum at the the center of the surveillance capsule as input and proceeded to adjust a priori (calculated) group fluxes to produce a best fit (in a least squares sense) to the reaction rate data. The exposure parameters along with associated uncertainties where then obtained from the adjusted spectra.

In the FERRET evaluations, a log normal least-squares algorithm weights both the a priori values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values f are linearly

related to the flux ϕ by some response matrix A:

$$f_i^{(s,\alpha)} = \sum_g A_{ig}^{(s)} \phi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group and a delineates spectra that may be simultaneously adjusted. For example,

$$R_i = \sum_g \sigma_{ig} \phi_g$$

relates a set of measured reaction rates R_i to a single spectrum ϕ_g by the multigroup cross section σ_{ig}. (In this case, FERRET also adjusts the cross-sections.) The lognormal approach automatically accounts for the physical constraint of positive fluxes, even with the large assigned uncertainties.

In the FERRET analysis of the dosimetry data, the continuous quantities (i.e., fluxes and cross-sections) were approximated in 53 groups. The calculated fluxes from the discrete ordinates analysis were expanded into the FERRET group structure using the SAND-II code [34]. This procedure was carried out by first expanding the a priori spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure for interpolation in regions where group boundaries do not coincide. The 620-point spectrum was then easily collapsed to the group scheme used in FERRET.

The cross-sections were also collapsed into the 53 energy-group structure using SAND II with calculated spectra (as expanded to 620 groups) as weighting functions. The cross sections were taken from the ENDF/B-V dosimetry file. Uncertainty estimates and 53 x 53 covariance matrices were constructed for each cross section. Correlations between cross sections were neglected due to data and code limitations, but are expected to be unimportant.

For each set of data or a priori values, the inverse of the corresponding relative covariance matrix M is used as a statistical weight. In some cases, as for the cross sections, a multigroup covariance matrix is used. More often, a simple parameterized form is used:

$$M_{gg'} = R_N^2 + R_g R_{g'} P_{gg'}$$

where R_N specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the corresponding set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix:

$$P_{gg'} = (1 - \theta) \delta_{gg'} + \theta \exp \left[\frac{-(g-g')^2}{2\partial^2} \right]$$

The first term specifies purely random uncertainties while the second term describes short-range correlations over a range ∂ (θ specifies the strength of the latter term).

For the a priori calculated fluxes, a short-range correlation of $\partial = 6$ groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long-range correlations (or anticorrelations) were justified based on information presented by R.E. Maerker[35]. Maerker's results are closely duplicated when $\partial = 6$. For the integral reaction rate covariances, simple normalization and random uncertainties were combined as deduced from experimental uncertainties.

Results of the FERRET evaluation of Capsule W-253 dosimetry are given in Table 6-9. The data summarized in Table 6-9 indicated that the capsule received an integrated exposure of 1.25×10^{19} n/cm^2 (E > 1.0 MeV) with an associated 1 σ uncertainty of $\pm$ 7% in the derived fluence. Also reported are capsule exposures in terms of fluence (E > 0.1 MeV) and iron atom displacements (dpa). Summaries of the fit of the adjusted spectrum are provided in Table 6-10. In general, excellent results were achieved in the fits of the adjusted spectrum to the individual experimental reaction rates. The adjusted spectrum itself is tabulated in Table 6-11 for the FERRET 53 energy group structure.

A summary of the measured and calculated neutron exposure of Capsule W-253 is presented in Table 6-12. The agreement between calculation and measurement falls within ± 20% for all fast neutron exposure parameters listed with the calculation underpredicting measurement in all cases.

Neutron exposure projections at the maximum fluence locations on the pressure vessel inner radius are given in Table 6-13. Projections are provided based on the cycle 8 core power distribution as well as for the cycle 10 fuel management design. The cycle 8 data represents a bounding low leakage pattern committed to by Maine Yankee Atomic Power Company while the cycle 10 results are representative of typical low leakage operation. Projected data are provided for periods up to 32 effective full power years of operation.

In developing these projections, the dosimetry from Capsule W-253 was included in an overall Maine Yankee measurement data base to provide a best estimate bias factor to be used in generating pressure vessel exposures. The development of this data base is described in detail in reference 37 and includes the use of surveillance capsule and reactor cavity dosimetry. Based on the use of this bias factor approach maximum fluence levels were computed by applying a factor of 1.22 to the analytical results.

In Table 6-14 estimated lead factors are listed for each of the remaining Maine Yankee surveillance capsules. For comparison purposes, values are provided for the cycles 1-6 non-low leakage core power distribution as well as for both the cycle 8 and cycle 10 low leakage core designs. The lead factors are defined as the ratio of the neutron flux at the center of the respective surveillance capsules to the maximum flux at the pressure vessel inner radius. These data may be used as a guide in establishing future withdrawal schedules for these capsules.

Relative to both the projected maximum vessel fluence and the capsule lead factors, it should be noted that the capsule results are representative of the core axial midplane; whereas, the maximum vessel exposure occurs at an axial elevation approximately 22 inches below core midplane. Adjustments for this axial effect were included for the vessel projections and the lead factors.
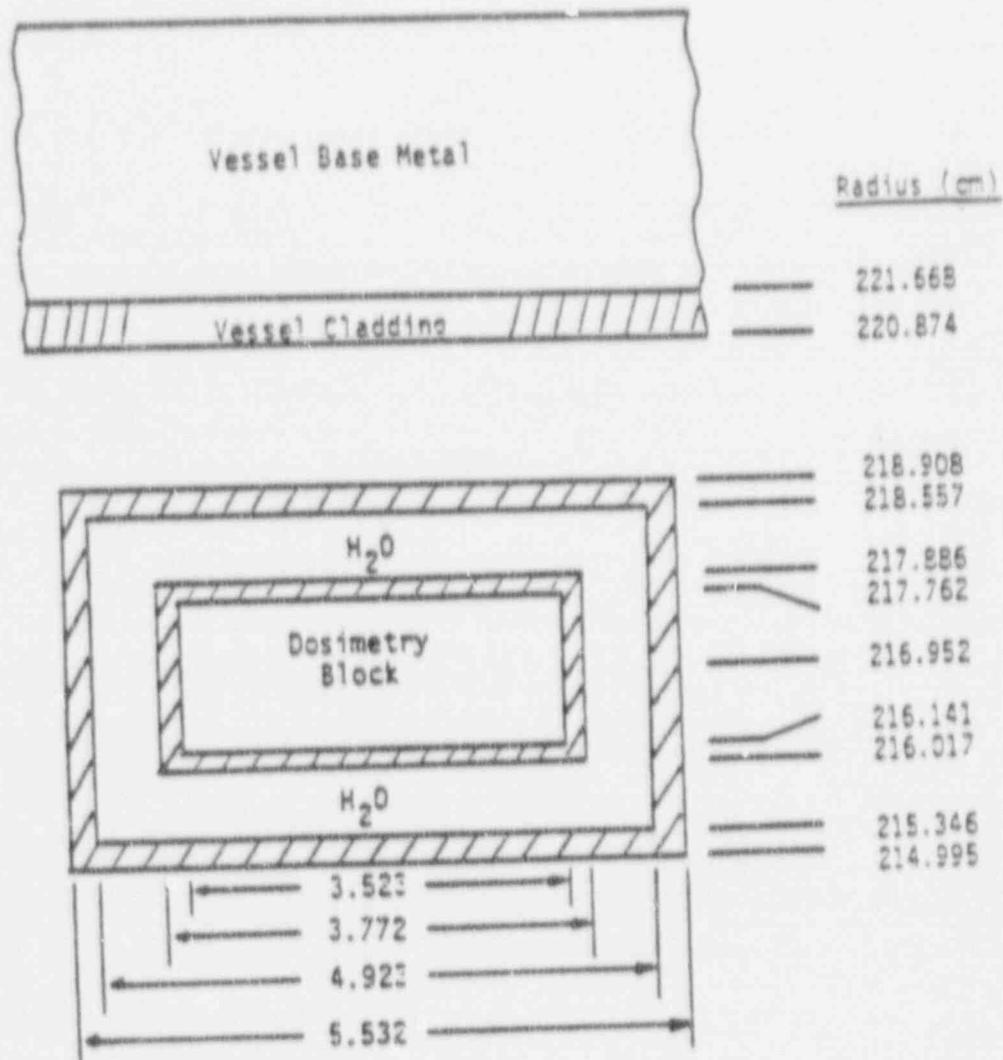
Figure 6-1. Plan View of a Wall Surveillance Capsule

TABLE 6-1

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS
AT THE CENTER OF CAPSULE W-253

ϕ (E > 1.0 MeV) (n/cm^2-sec)

CYCLES 1-6	3.23×10^{10}
CYCLE 7	2.86×10^{10}
CYCLE 8	2.19×10^{10}
CYCLE 9	2.15×10^{10}
CYCLE 10	1.69×10^{10}
CYCLE 11	1.85×10^{10}

ϕ (E > 0.1 MeV) (n/cm^2-sec)

CYCLES 1-6	6.59×10^{10}
CYCLE 7	5.84×10^{10}
CYCLE 8	4.47×10^{10}
CYCLE 9	4.39×10^{10}
CYCLE 10	3.45×10^{10}
CYCLE 11	3.78×10^{10}

IRON ATOM DISPLACEMENT RATE (dpa/sec)

CYCLES 1-6	4.94×10^{-11}
CYCLE 7	4.38×10^{-11}
CYCLE 8	3.35×10^{-11}
CYCLE 9	3.29×10^{-11}
CYCLE 10	2.59×10^{-11}
CYCLE 11	2.83×10^{-11}

TABLE 6-2

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS
AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

ϕ (E > 1.0 MeV) (n/cm^2-sec)

	260°	270°
CYCLES 1-6	2.41×10^{10}	1.97×10^{10}
CYCLE 7	2.16×10^{10}	1.63×10^{10}
CYCLE 8	1.64×10^{10}	1.37×10^{10}
CYCLE 9	1.66×10^{10}	1.32×10^{10}
CYCLE 10	1.26×10^{10}	1.15×10^{10}
CYCLE 11	1.38×10^{10}	1.23×10^{10}

ϕ (E > 0.1 MeV) (n/cm^2-sec)

	260°	270°
CYCLES 1-6	6.12×10^{10}	5.00×10^{10}
CYCLE 7	5.48×10^{10}	4.14×10^{10}
CYCLE 8	4.16×10^{10}	3.48×10^{10}
CYCLE 9	4.21×10^{10}	3.35×10^{10}
CYCLE 10	3.20×10^{10}	2.92×10^{10}
CYCLE 11	3.50×10^{10}	3.12×10^{10}

IRON ATOM DISPLACEMENT RATE (dpa/sec)

	260°	270°
CYCLES 1-6	3.86×10^{-11}	3.15×10^{-11}
CYCLE 7	3.46×10^{-11}	2.61×10^{-11}
CYCLE 8	2.62×10^{-11}	2.19×10^{-11}
CYCLE 9	2.63×10^{-11}	2.11×10^{-11}
CYCLE 10	2.02×10^{-11}	1.84×10^{-11}
CYCLE 11	2.21×10^{-11}	1.97×10^{-11}

TABLE 6-3

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 1.0 MeV) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	Depth Into Vessel (in)	260°	270°	280°
221.67[1]	0.000	1.000	1.000	1.000
221.94	0.106	0.981	0.984	0.982
223.65	0.780	0.816	0.822	0.823
225.57	1.535	0.639	0.646	0.650
227.49	2.291	0.491	0.498	0.504
229.41	3.047	0.375	0.382	0.387
231.33	3.803	0.284	0.291	0.296
233.26	4.563	0.215	0.221	0.225
235.18	5.319	0.161	0.167	0.171
237.10	6.075	0.121	0.126	0.129
239.02	6.841	0.0899	0.0938	0.0965
240.94	7.587	0.0665	0.0695	0.0719
242.74	8.295	0.0485	0.0513	0.0537
243.58[2]	8.625	0.0406	0.0437	0.0473

NOTES: 1) Base Metal Inner Radius

2) Base Metal Outer Radius

TABLE 6-4

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 0.1 MeV)
WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	Depth Into Vessel (in)	260°	270°	280°
221.67[1]	0.000	1.000	1.000	1.000
221.94	0.106	0.981	0.984	0.982
223.65	0.780	0.816	0.822	0.823
225.57	1.535	0.639	0.646	0.650
227.49	2.291	0.491	0.498	0.504
229.41	3.047	0.375	0.382	0.387
231.33	3.803	0.284	0.291	0.296
233.26	4.563	0.215	0.221	0.225
235.18	5.319	0.161	0.167	0.171
237.10	6.075	0.121	0.126	0.129
239.02	6.841	0.0899	0.0938	0.0965
240.94	7.587	0.0665	0.0695	0.0719
242.74	8.295	0.0485	0.0513	0.0537
243.58[2]	8.625	0.0406	0.0437	0.0473

NOTES: 1) Base Metal Inner Radius

2) Base Metal Outer Radius

TABLE 6-5

RELATIVE RADIAL DISTRIBUTIONS OF IRON ATOM DISPLACEMENT RATE (dpa/sec) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	Depth Into Vessel (in)	260°	270°	280°
221.67[1]	0.000	1.000	1.000	1.000
221.94	0.106	0.988	0.985	0.985
223.65	0.780	0.865	0.869	0.870
225.57	1.535	0.726	0.733	0.736
227.49	2.291	0.611	0.617	0.620
229.41	3.047	0.512	0.519	0.525
231.33	3.803	0.429	0.436	0.442
233.26	4.563	0.357	0.367	0.373
235.18	5.319	0.298	0.307	0.314
237.10	6.075	0.246	0.255	0.263
239.02	6.841	0.201	0.210	0.217
240.94	7.587	0.160	0.169	0.177
242.74	8.295	0.125	0.134	0.142
243.58[2]	8.625	0.106	0.117	0.128

NOTES: 1) Base Metal Inner Radius

2) Base Metal Outer Radius

TABLE 6-6

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 yrs	
Iron	$Fe^{54}(n,p)Mn^{54}$	0.0582	E > 1.0 MeV	312.2 days	
Nickel	$Ni^{58}(n,p)Co^{58}$	0.6830	E > 1.0 MeV	70.90 days	
Uranium-238*	$U^{238}(n,f)Cs^{137}$	1.0	E > 0.4 MeV	30.12 yrs	5.99
Neptunium-237*	$Np^{237}(n,f)Cs^{137}$	1.0	E > 0.08 MeV	30.12 yrs	6.50
Cobalt-Aluminum*	$Co^{59}(n,\partial)Co^{60}$	0.0015	0.4ev>E> 0.015 MeV	5.272 yrs	
Cobalt-Aluminum	$Co^{59}(n,\partial)Co^{60}$	0.0015	E > 0.015 MeV	5.272 yrs	

*Denotes that monitor is cadmium shielded.

TABLE 6-7

IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE W-253

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
11/72	728	.270	23	6468
12/72	1341	.497	31	6437
1/73	1608	.595	31	6406
2/73	929	.344	28	6378
3/73	1023	.379	31	6347
4/73	1813	.672	30	6317
5/73	1597	.591	31	6286
6/73	1273	.472	30	6256
7/73	0	.000	31	6225
8/73	757	.280	31	6194
9/73	665	.246	30	6164
10/73	1490	.552	31	6133
11/73	1923	.712	30	6103
12/73	1727	.640	31	6072
1/74	1939	.718	31	6041
2/74	1818	.674	28	6013
3/74	1826	.676	31	5982
4/74	1851	.686	30	5952
5/74	1909	.707	31	5921
6/74	1772	.656	30	5891
7/74	0	.000	31	5860
8/74	0	.000	31	5829
9/74	0	.000	30	5799
10/74	1113	.412	31	5768
11/74	1717	.636	30	5738
12/74	1782	.660	31	5707
1/75	1414	.524	31	5676

TABLE 6-7 (continued)
IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE W-253

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
2/75	1870	.692	28	5648
3/75	2048	.759	31	5617
4/75	2049	.759	30	5587
5/75	117	.044	31	5556
6/75	29	.011	30	5526
7/75	1426	.528	31	5495
8/75	2269	.840	31	5464
9/75	2341	.867	30	5434
10/75	2414	.894	31	5403
11/75	1858	.688	30	5373
12/75	2308	.855	31	5342
1/76	2339	.867	31	5311
2/76	2230	.826	29	5282
3/76	2225	.824	31	5251
4/76	2335	.865	30	5221
5/76	2255	.835	31	5190
6/76	2200	.815	30	5160
7/76	2238	.829	31	5129
8/76	2385	.883	31	5098
9/76	1274	.472	30	5068
10/76	2287	.847	31	5037
11/76	2414	.894	30	5007
12/76	2401	.889	31	4976
1/77	2298	.851	31	4945
2/77	2318	.859	28	4917
3/77	1973	.731	31	4886
4/77	382	.142	30	4856

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
5/77	0	.000	31	4825
6/77	1237	.458	30	4795
7/77	2405	.691	31	4764
8/77	2393	.886	31	4733
9/77	2414	.894	30	4703
10/77	2425	.898	31	4672
11/77	2412	.893	30	4642
12/77	2373	.879	31	4611
1/78	2415	.894	31	4580
2/78	2517	.932	28	4552
3/78	2395	.887	31	4521
4/78	2349	.870	30	4491
5/78	2442	.905	31	4460
6/78	2155	.798	30	4430
7/78	643	.238	31	4399
8/78	112	.042	31	4368
9/78	1042	.386	30	4338
10/78	2178	.807	31	4307
11/78	2467	.914	30	4277
12/78	2466	.913	31	4246
1/79	2389	.885	31	4215
2/79	2428	.899	28	4187
3/79	1019	.376	31	4156
4/79	0	.000	30	4126
5/79	0	.000	31	4095
6/79	1943	.719	30	4065
7/79	2522	.934	31	4034

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
8/79	2498	.925	31	4003
9/79	21	.008	30	3973
10/79	2070	.767	31	3942
11/79	2444	.905	30	3912
12/79	2018	.748	31	3881
1/80	595	.220	31	3850
2/80	0	.000	29	3821
3/80	771	.286	31	3790
4/80	2416	.895	30	3760
5/80	1949	.722	31	3729
6/80	1454	.538	30	3699
7/80	2427	.899	31	3668
8/80	1690	.626	31	3637
9/80	1694	.627	30	3607
10/80	2535	.939	31	3576
11/80	2495	.924	30	3546
12/80	1263	.468	31	3515
1/81	2550	.944	31	3484
2/81	2541	.941	28	3456
3/81	2553	.946	31	3425
4/81	2551	.945	30	3395
5/81	619	.229	31	3364
6/81	0	.000	30	3334
7/81	482	.178	31	3303
8/81	2378	.881	31	3272
9/81	2402	.890	30	3242
10/81	2118	.784	31	3211

TABLE 6-7 (continued)
IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE W-253

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
11/81	2392	.886	30	3181
12/81	2365	.876	31	3150
1/82	1712	.634	31	3119
2/82	2304	.853	28	3091
3/82	456	.169	31	3060
4/82	2490	.922	30	3030
5/82	2416	.895	31	2999
6/82	2439	.903	30	2969
7/82	2524	.935	31	2938
8/82	2264	.838	31	2907
9/82	1753	.649	30	2877
10/82	0	.000	31	2846
11/82	0	.000	30	2816
12/82	1078	.399	31	2785
1/83	1801	.667	31	2754
2/83	123	.046	28	2726
3/83	2025	.750	31	2695
4/83	2591	.960	30	2665
5/83	252	.093	31	2634
6/83	2595	.961	30	2604
7/83	2586	.958	31	2573
8/83	2502	.927	31	2542
9/83	2530	.937	30	2512
10/83	2591	.960	31	2481
11/83	1311	.485	30	2451
12/83	1631	.604	31	2420
1/84	2369	.877	31	2389

TABLE 6-7 (continued)
IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE W-253

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
2/84	2715	1.01	29	2360
3/84	2176	.806	31	2329
4/84	0	.000	30	2299
5/84	0	.000	31	2268
6/84	181	.067	30	2238
7/84	2503	.927	31	2207
8/84	2603	.964	31	2176
9/84	2600	.963	30	2146
10/84	2488	.921	31	2115
11/84	2024	.749	30	2085
12/84	2583	.957	31	2054
1/85	2422	.897	31	2023
2/85	2415	.895	28	1995
3/85	2443	.905	31	1964
4/85	2565	.950	30	1934
5/85	2276	.843	31	1903
6/85	2558	.948	30	1873
7/85	2221	.823	31	1842
8/85	1034	.383	31	1811
9/85	0	.000	30	1781
10/85	264	.098	31	1750
11/85	2161	.801	30	1720
12/85	2545	.943	31	1689
1/86	2418	.896	31	1658
2/86	1843	.683	28	1630
3/86	1577	.584	31	1599
4/86	2258	.836	30	1569

TABLE 6-7 (continued)
IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE W-253

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
5/86	2504	.927	31	1538
6/86	2151	.797	30	1508
7/86	973	.361	31	1477
8/86	2465	.913	31	1446
9/86	2608	.966	30	1416
10/86	2518	.933	31	1385
11/86	2469	.915	30	1355
12/86	2582	.956	31	1324
1/87	2598	.962	31	1293
2/87	2360	.874	28	1265
3/87	1766	.654	31	1234
4/87	0	.000	30	1204
5/87	0	.000	31	1173
6/87	352	.130	30	1143
7/87	23	.009	31	1112
8/87	528	.196	31	1081
9/87	2585	.957	30	1051
10/87	2610	.966	31	1020
11/87	2608	.966	30	990
12/87	2592	.960	31	959
1/88	2515	.932	31	928
2/88	2589	.959	29	899
3/88	2628	.973	31	868
4/88	2620	.971	30	838
5/88	2613	.968	31	807
6/88	2598	.962	30	777
7/88	2620	.971	31	746

Irradiation Period	P_j (MW$_t$)	$\dfrac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
8/88	1516	.562	31	715
9/88	1956	.724	30	685
10/88	905	.335	31	654
11/88	0	. 0	30	624
12/88	13	.005	31	593
1/89	2442	.905	31	562
2/89	1795	.665	28	534
3/89	2628	.973	31	503
4/89	2548	.944	30	473
5/89	2629	.974	31	442
6/89	2611	.967	30	412
7/89	2608	.966	31	381
8/89	2614	.968	31	350
9/89	2554	.946	30	320
10/89	2134	.790	31	289
11/89	1713	.635	30	259
12/89	2615	.969	31	228
1/90	2585	.957	31	197
2/90	2565	.950	28	169
3/90	2548	.944	31	138
4/90	2023	.749	7	131

NOTE: Reference Power = 2700 MW$_t$

TABLE 6-8
MEASURED SENSOR ACTIVITIES AND REACTION RATES

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
COMP 3514	1.49×10^5	2.78×10^5	
COMP 3541	Sensor was fused with Cd cover		
COMP 3573	1.39×10^5	2.60×10^5	
Average	1.44×10^5	2.69×10^5	4.10×10^{-17}
Ti-46 (n,p) Sc-46			
COMP 3514	1.42×10^5	6.55×10^5	
COMP 3541	1.25×10^5	5.76×10^5	
COMP 3573	1.30×10^5	5.99×10^5	
Average	1.32×10^5	6.10×10^5	5.88×10^{-16}
Fe-54 (n,p) Mn-54			
COMP 3514	9.54×10^5	2.20×10^6	
COMP 3541	8.28×10^5	1.92×10^6	
COMP 3573	8.55×10^5	1.98×10^6	
Average	8.79×10^5	2.03×10^6	3.24×10^{-15}
Ni-58 (n,p) Co-58			
COMP 3514	5.98×10^6	3.34×10^7	
COMP 3541	5.02×10^6	2.80×10^7	
COMP 3573	5.38×10^6	3.00×10^7	
Average	5.46×10^6	3.05×10^7	4.35×10^{-15}

TABLE 6-8
MEASURED SENSOR ACTIVITIES AND REACTION RATES - cont'd

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
U-238 (n,f) Cs-137 (Cd)			
COMP 3514	5.88×10^5	2.71×10^6	
COMP 3541	5.59×10^5	2.58×10^6	
COMP 3573	5.73×10^5	2.64×10^6	
Average	5.73×10^5	2.64×10^6	1.76×10^{-14}
Co-59 (n,∂) Co-60			
COMP 3514	3.59×10^7	6.71×10^7	
COMP 3541	4.03×10^7	7.53×10^7	
COMP 3573	2.70×10^7	5.04×10^7	
Average	3.44×10^7	6.42×10^7	4.19×10^{-12}
Co-59 (n,∂) Co-60 (Cd)			
COMP 3514	5.12×10^6	9.56×10^6	
COMP 3541	5.34×10^6	9.97×10^6	
COMP 3573	4.97×10^6	9.28×10^6	
Average	5.14×10^6	9.61×10^7	6.27×10^{-12}

TABLE 6-9

SUMMARY OF NEUTRON DOSIMETRY RESULTS

<u>TIME AVERAGED EXPOSURE RATES</u>

ϕ (E > 1.0 MeV) (n/cm^2-sec)	3.41 x 10^{10}	± 7%
ϕ (E > 0.1 MeV) (n/cm^2-sec)	7.09 x 10^{10}	± 15%
dpa/sec	4.96 x 10^{-11}	± 8%
ϕ (E < 0.414 eV) (n/cm^2-sec)	1.43 x 10^{11}	± 13%

<u>INTEGRATED CAPSULE EXPOSURE</u>

ϕ (E > 1.0 MeV) (n/cm^2)	1.25 x 10^{19}	± 7%
ϕ (E > 0.1 MeV) (n/cm^2)	2.61 x 10^{19}	± 15%
dpa	1.82 x 10^{-2}	± 8%
ϕ (E < 0.414 eV) (n/cm^2)	5.26 x 10^{19}	± 13%

NOTE: Total Irradiation Time = 11.7 EFPY

TABLE 6-10

COMPARISON OF MEASURED AND FERRET CALCULATED
REACTION RATES AT THE SURVEILLANCE CAPSULE CENTER

Reaction	Measured	Adjusted Calculation	C/M
Cu-63 (n,α) Co-60	4.10×10^{-17}	4.10×10^{-17}	1.00
Ti-46 (n,p) Sc-46	5.88×10^{-16}	5.86×10^{-16}	1.00
Fe-54 (n,p) Mn-54	3.24×10^{-15}	3.27×10^{-15}	1.01
Ni-58 (n,p) Co-58	4.35×10^{-15}	4.34×10^{-15}	1.00
U-238 (n,f) Cs-137 (Cd)	1.47×10^{-14}	1.47×10^{-14}	1.00
Co-59 (n,∂) Co-60 (Cd)	4.19×10^{-12}	4.18×10^{-12}	1.00
Co-59 (n,∂) Co-60	6.27×10^{-13}	6.26×10^{-13}	1.00

TABLE 6-11
ADJUSTED NEUTRON ENERGY SPECTRUM AT
THE SURVEILLANCE CAPSULE CENTER

Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)	Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)
1	1.73×10^1	4.29×10^6	28	9.12×10^{-3}	3.55×10^9
2	1.49×10^1	1.02×10^7	29	5.53×10^{-3}	3.68×10^9
3	1.35×10^1	4.41×10^7	30	3.36×10^{-3}	1.22×10^9
4	1.16×10^1	1.06×10^8	31	2.84×10^{-3}	1.22×10^9
5	1.00×10^1	2.46×10^8	32	2.40×10^{-3}	1.23×10^9
6	8.61×10^0	4.23×10^8	33	2.04×10^{-3}	3.72×10^9
7	7.41×10^0	9.91×10^8	34	1.23×10^{-3}	3.76×10^9
8	6.07×10^0	1.36×10^9	35	7.49×10^{-4}	3.83×10^9
9	4.97×10^0	2.47×10^9	36	4.54×10^{-4}	3.90×10^9
10	3.68×10^0	2.58×10^9	37	2.75×10^{-4}	4.01×10^9
11	2.87×10^0	4.61×10^9	38	1.67×10^{-4}	4.48×10^9
12	2.23×10^0	1.02×10^{10}	39	1.01×10^{-4}	4.25×10^9
13	1.74×10^0	5.00×10^9	40	6.14×10^{-5}	4.25×10^9
14	1.35×10^0	4.01×10^9	41	3.73×10^{-5}	4.25×10^9
15	1.11×10^0	5.91×10^9	42	2.26×10^{-5}	4.23×10^9
16	8.21×10^{-1}	5.76×10^9	43	1.37×10^{-5}	4.20×10^9
17	6.39×10^{-1}	5.32×10^9	44	8.32×10^{-6}	4.20×10^9
18	4.98×10^{-1}	3.82×10^9	45	5.04×10^{-6}	4.20×10^9
19	3.88×10^{-1}	4.35×10^9	46	3.06×10^{-6}	4.20×10^9
20	3.02×10^{-1}	6.82×10^9	47	1.86×10^{-6}	4.14×10^9
21	1.83×10^{-1}	5.81×10^9	48	1.13×10^{-6}	4.06×10^9
22	1.11×10^{-1}	4.92×10^9	49	6.83×10^{-7}	4.37×10^9
23	6.74×10^{-2}	4.24×10^9	50	4.14×10^{-7}	5.20×10^9
24	4.09×10^{-2}	3.42×10^9	51	2.51×10^{-7}	1.71×10^{10}
25	2.55×10^{-2}	2.08×10^9	52	1.52×10^{-7}	3.22×10^{10}
26	1.99×10^{-2}	1.98×10^9	53	9.24×10^{-8}	8.82×10^{10}
27	1.50×10^{-2}	3.42×10^9			

NOTE: Tabulated energy levels represent the upper energy of each group.

TABLE 6-12

COMPARISON OF CALCULATED AND MEASURED
EXPOSURE LEVELS FOR CAPSULE W-253

	Calculated	Measured	C/M
$\phi(E > 1.0 \text{ MeV})$ (n/cm^2)	1.01×10^{19}	1.25×10^{19}	0.81
$\phi(E > 0.1 \text{ MeV})$ (n/cm^2)	2.06×10^{19}	2.61×10^{19}	0.79
dpa	1.54×10^{-2}	1.82×10^{-2}	0.85

TABLE 6-13

NEUTRON EXPOSURE PROJECTIONS AT KEY LOCATIONS
ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE FOR MAINE YANKEE

$\phi(E>1.0 \text{ MeV})$ (n/cm^2)

OPERATING TIME (EFPY	CYCLE 8 PROJECTION 260 DEG	270 DEG	CYCLE 10 PROJECTION 260 DEG	270 DEG
11.7	9.20×10^{18}	7.57×10^{18}	9.20×10^{18}	7.57×10^{18}
15.0	1.13×10^{19}	9.31×10^{18}	1.08×10^{19}	9.03×10^{18}
20.0	1.45×10^{19}	1.19×10^{19}	1.32×10^{19}	1.12×10^{19}
25.0	1.77×10^{19}	1.45×10^{19}	1.56×10^{19}	1.34×10^{19}
30.0	2.09×10^{19}	1.71×10^{19}	1.80×10^{19}	1.56×10^{19}
32.0	2.22×10^{19}	1.82×10^{19}	1.90×10^{19}	1.65×10^{19}

TABLE 6-14

UPDATED LEAD FACTORS FOR MAINE YANKEE
WALL SURVEILLANCE CAPSULES

Lead Factor

CAPSULE	CYCLES 1-11	CYCLES 1-6	CYCLE 8	CYCLE 10
W-83	1.451	1.454	1.446	1.456
W-263		Previously Withdrawn		
W-353	1.451	1.454	1.446	1.456
W-253	1.341	Current Evaluation		
W-343	1.341	1.343	1.336	1.345
W-160	1.220	1.223	1.216	1.224

SECTION 7.0
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following removal schedule meets ASTM E185-82 and is suggested for future capsules to be removed from the Maine Yankee reactor vessel:

Capsule Location (deg.)	Capsule Lead Factor	Capsule Removal EFPY	Estimated Fluence (n/cm^2)
A-25	Withdrawn	-----	1.76×10^{19} (a)
A-35	Withdrawn	-----	6.53×10^{19}
W-263	Withdrawn	4.58	0.572×10^{19}
W-253 (b)	1.34	11.65	1.25×10^{19} (c)
W-353	1.45 (d)	22.0	2.2×10^{19} (e)
W-343	1.34 (d)	32.0 (EOL)	2.98×10^{19} (f)
W-83	1.45 (d)	Standby	- - - -
W-160	1.22 (d)	Standby	- - - -
W-263 (replacement capsule)		Standby	- - - -

(a) Approximate 15 EFPY fluence

(b) Current evaluation

(c) Approximate fluence at 1/4 thickness reactor vessel wall at end of life (32 EFPY)

(d) Estimate based on azimuthal distribution using cycle 8 radial power distribution

(e) Approximate fluence at reactor vessel clad/base metal interface at end of life (32 EFPY)

(f) Fluence not less than once or greater than twice the peak end of life (32 EFPY) clad/base metal interface fluence

SECTION 8.0
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APPENDIX A

Load-Time Records for Charpy Specimen Tests
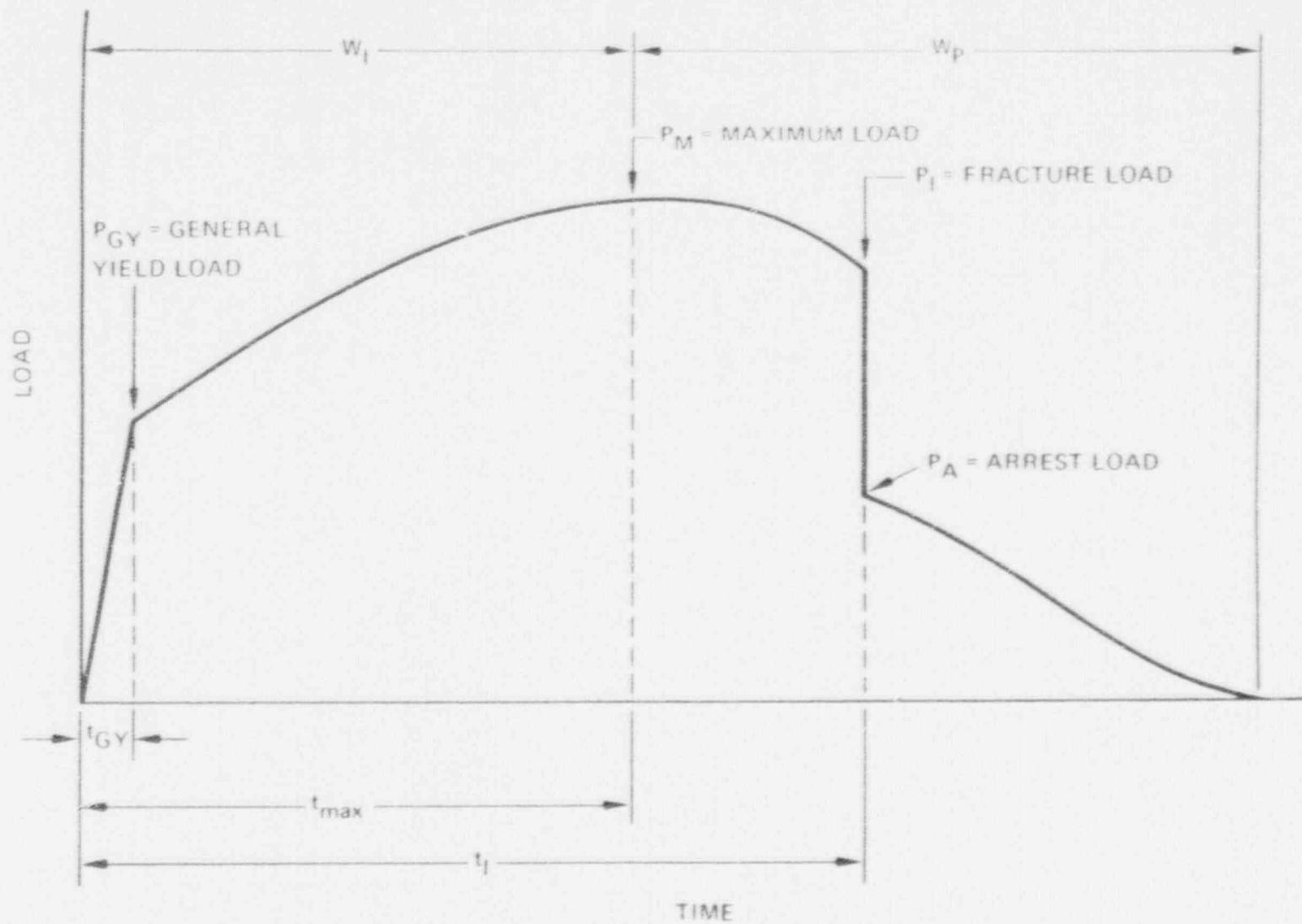
Figure A-1. Idealized Load-Time Record

A-1
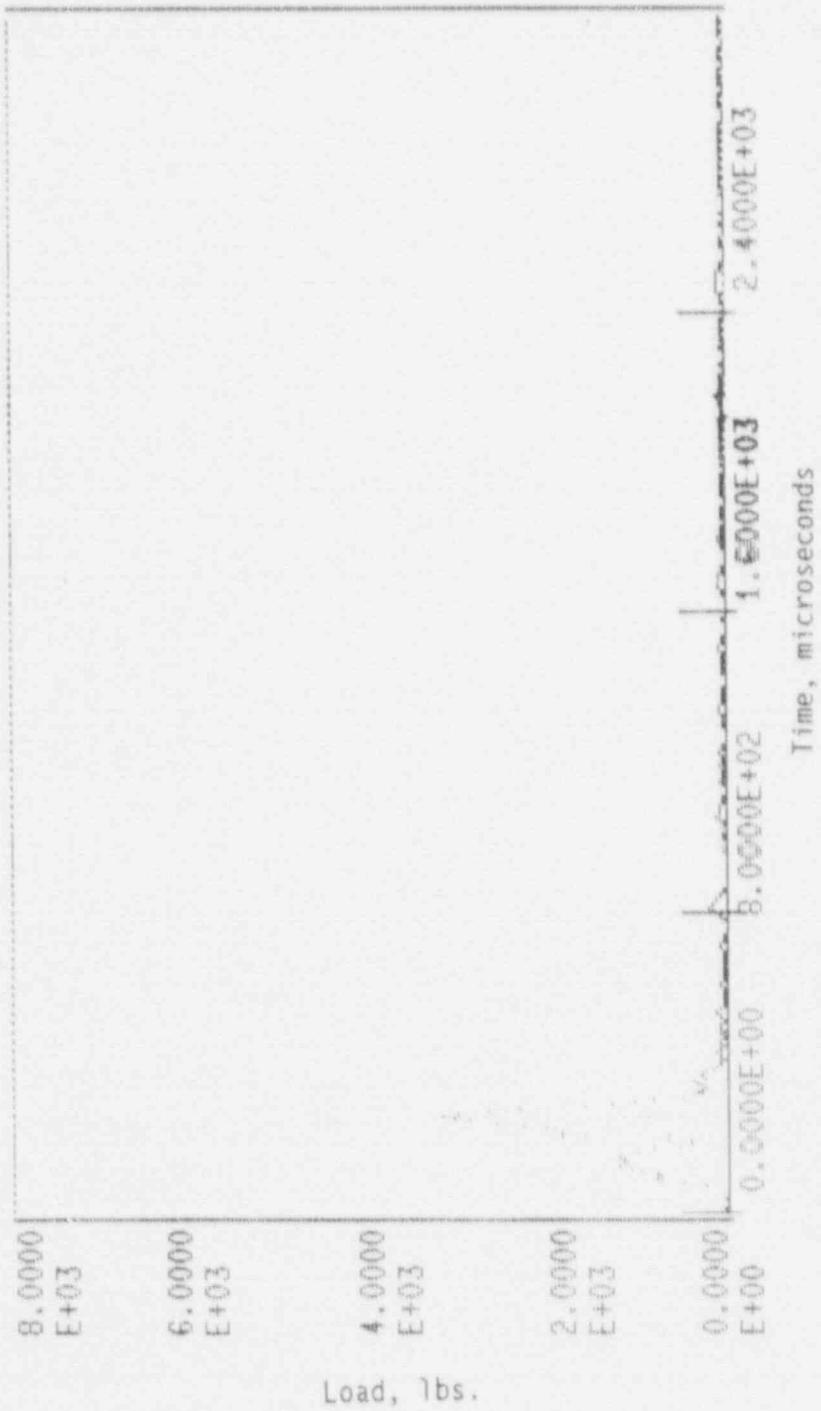
Figure A-2. Load-time record for Specimen 12T

Load, lbs.

Time, microseconds
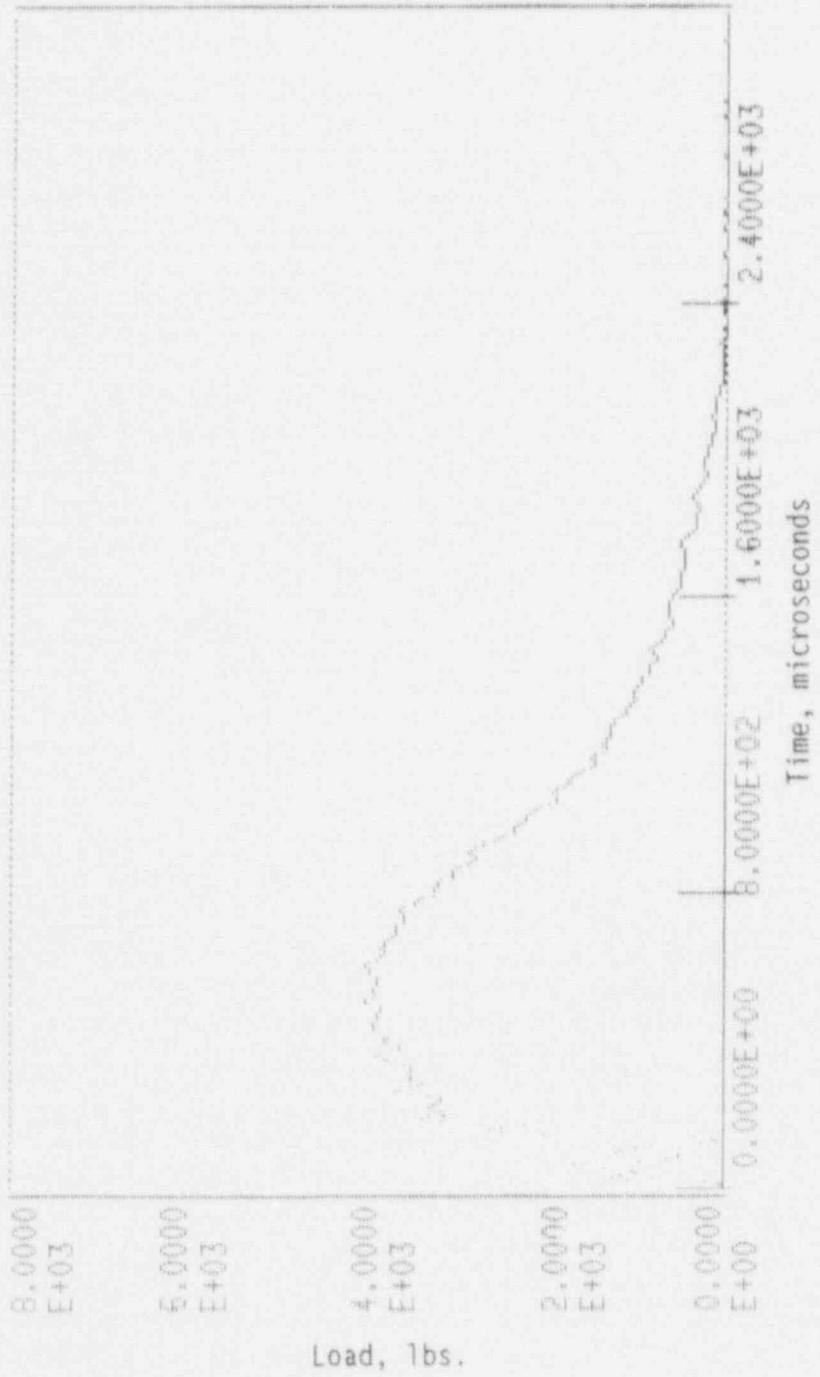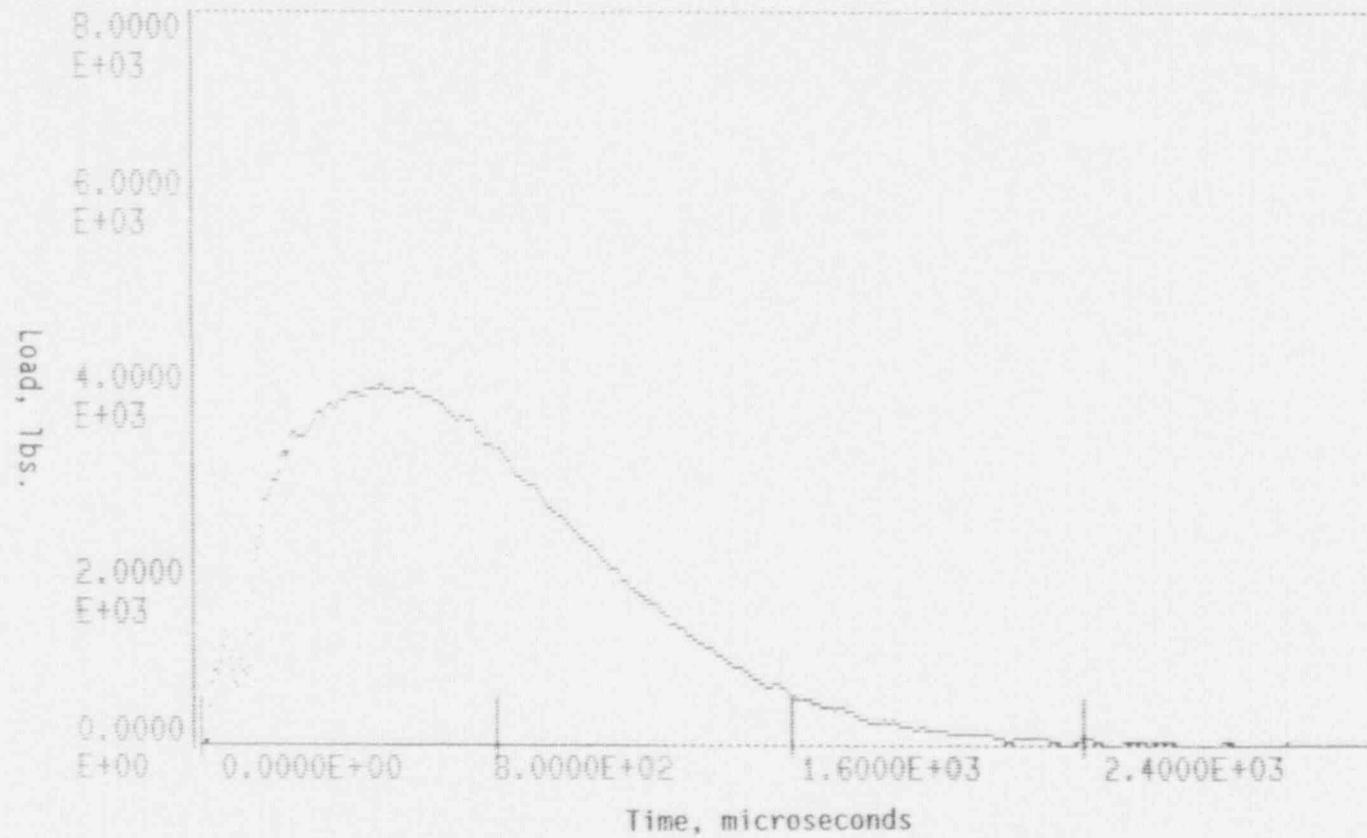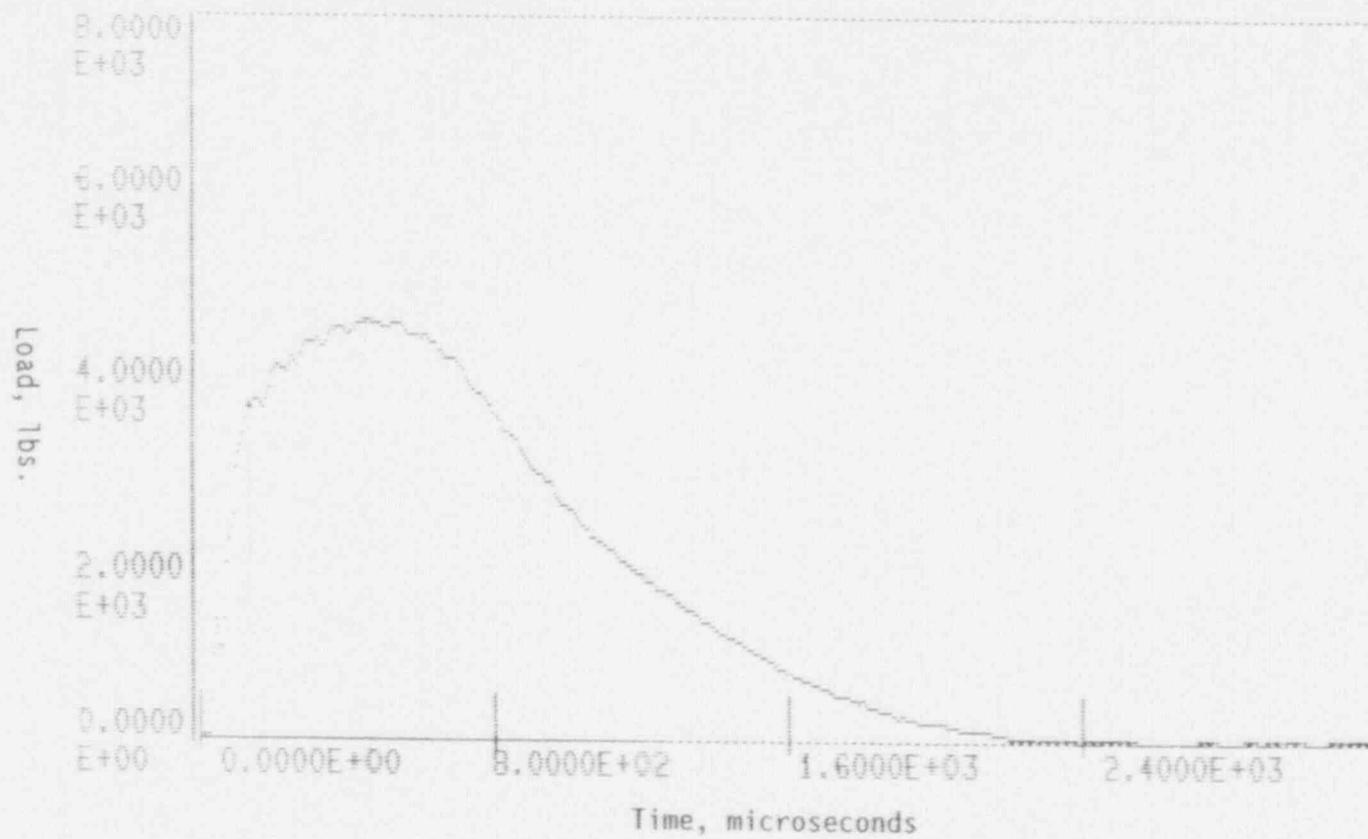
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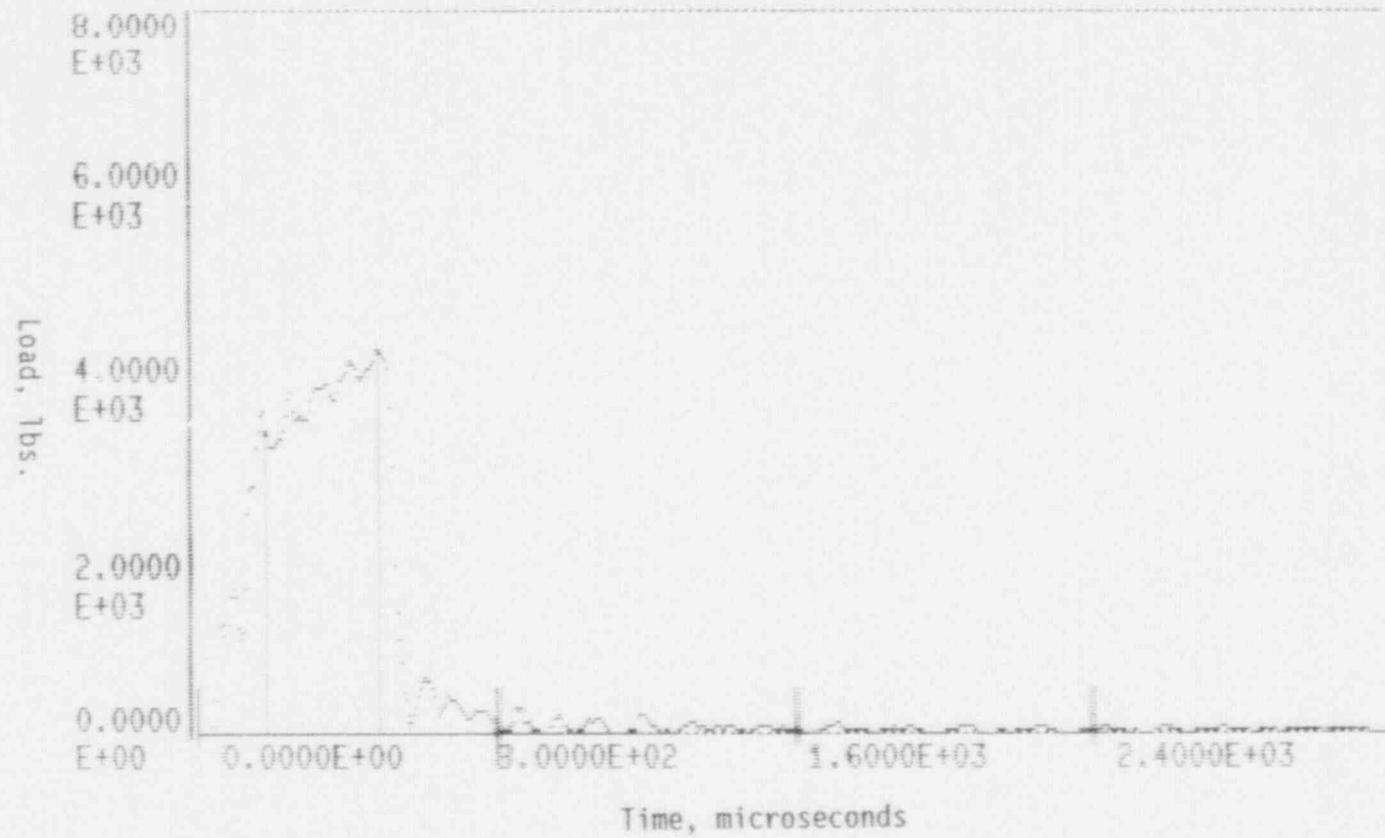
Figure A-3. Load-time record for Specimen 13E
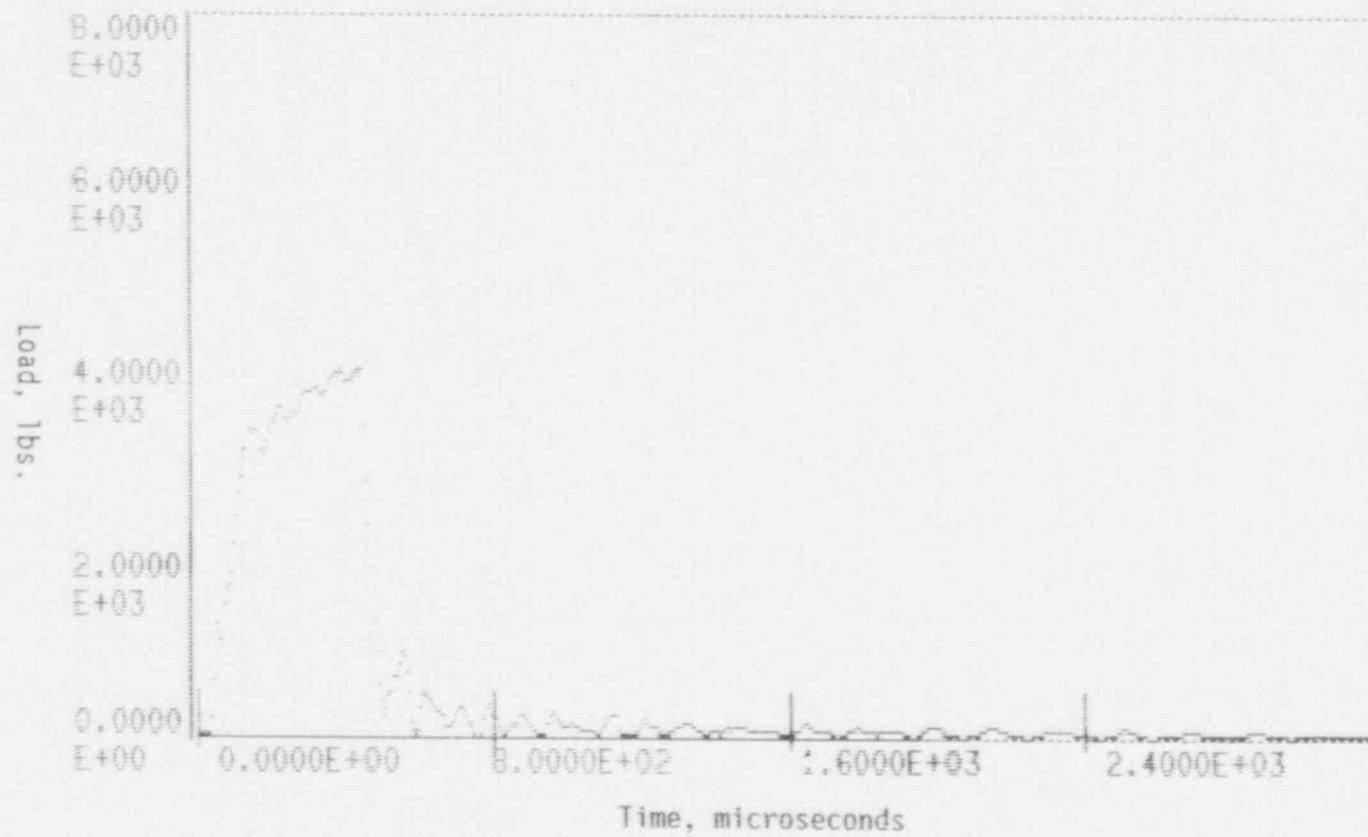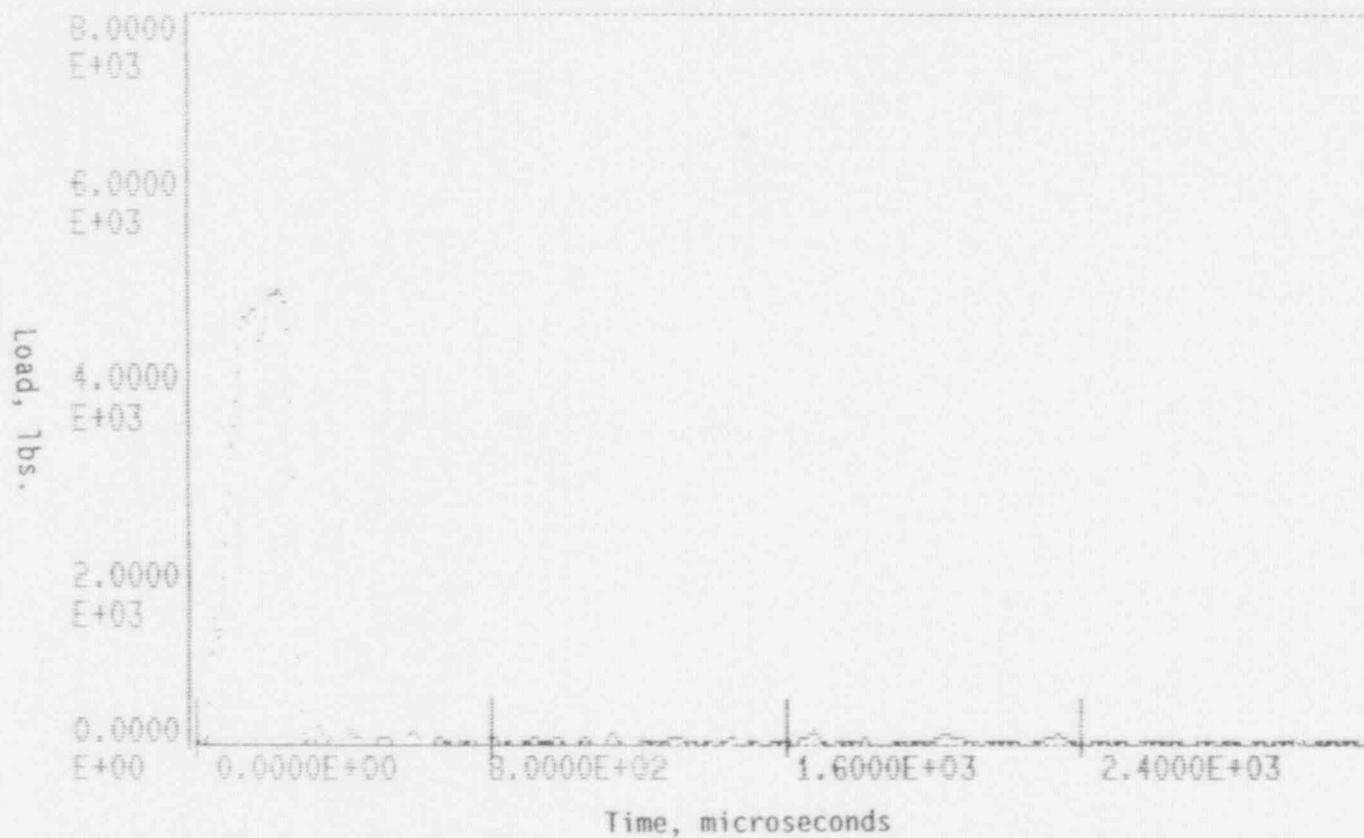
Figure A-4. Load-time record for Specimen 121
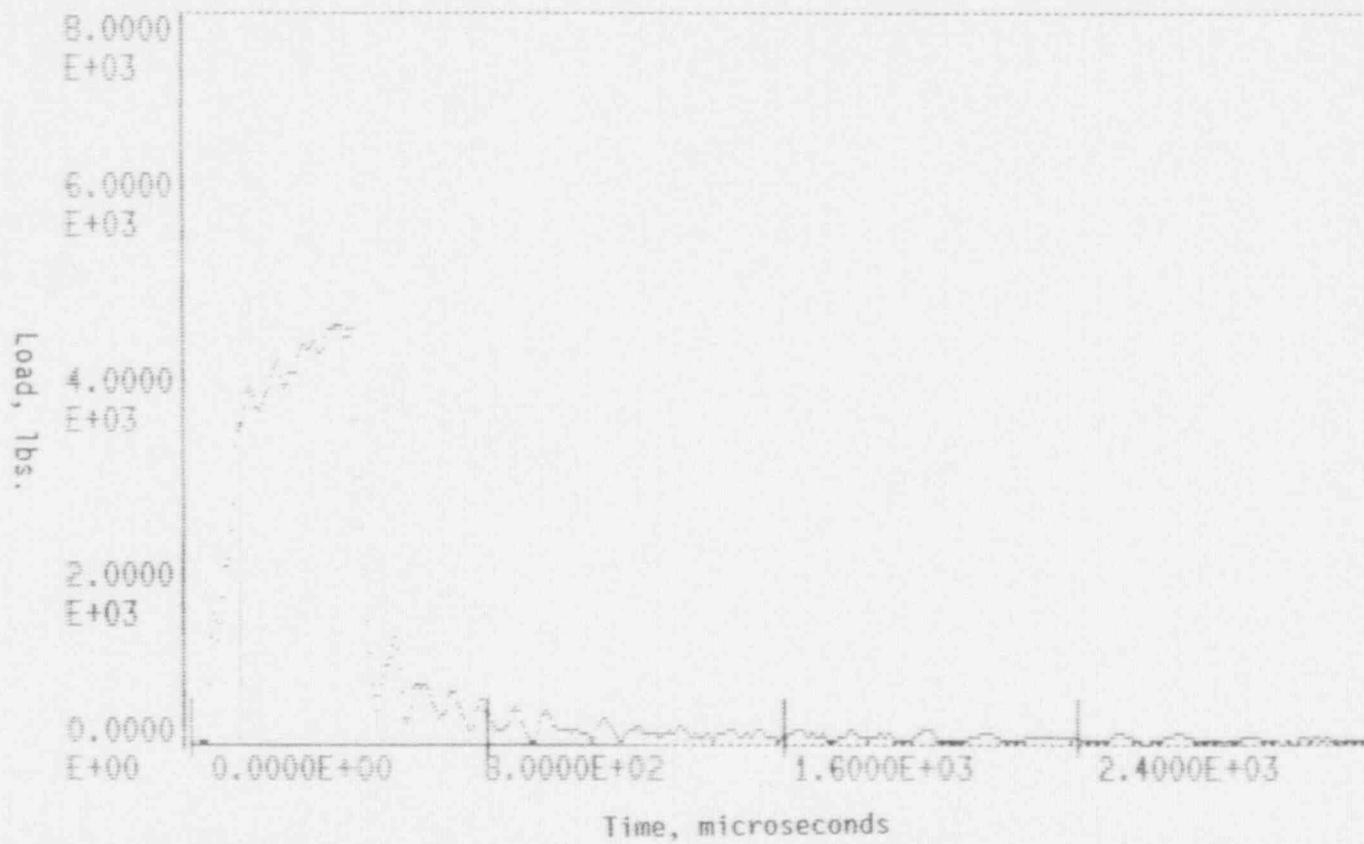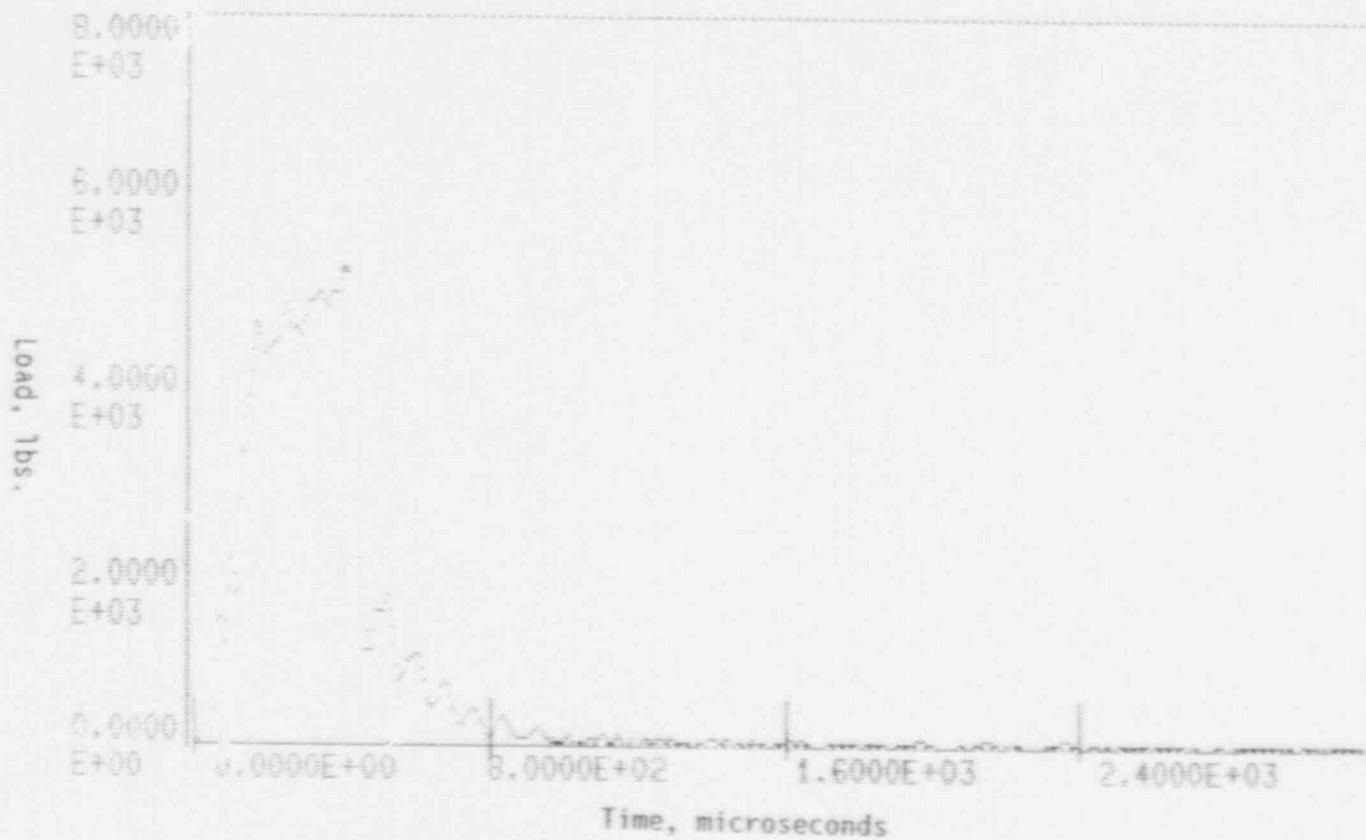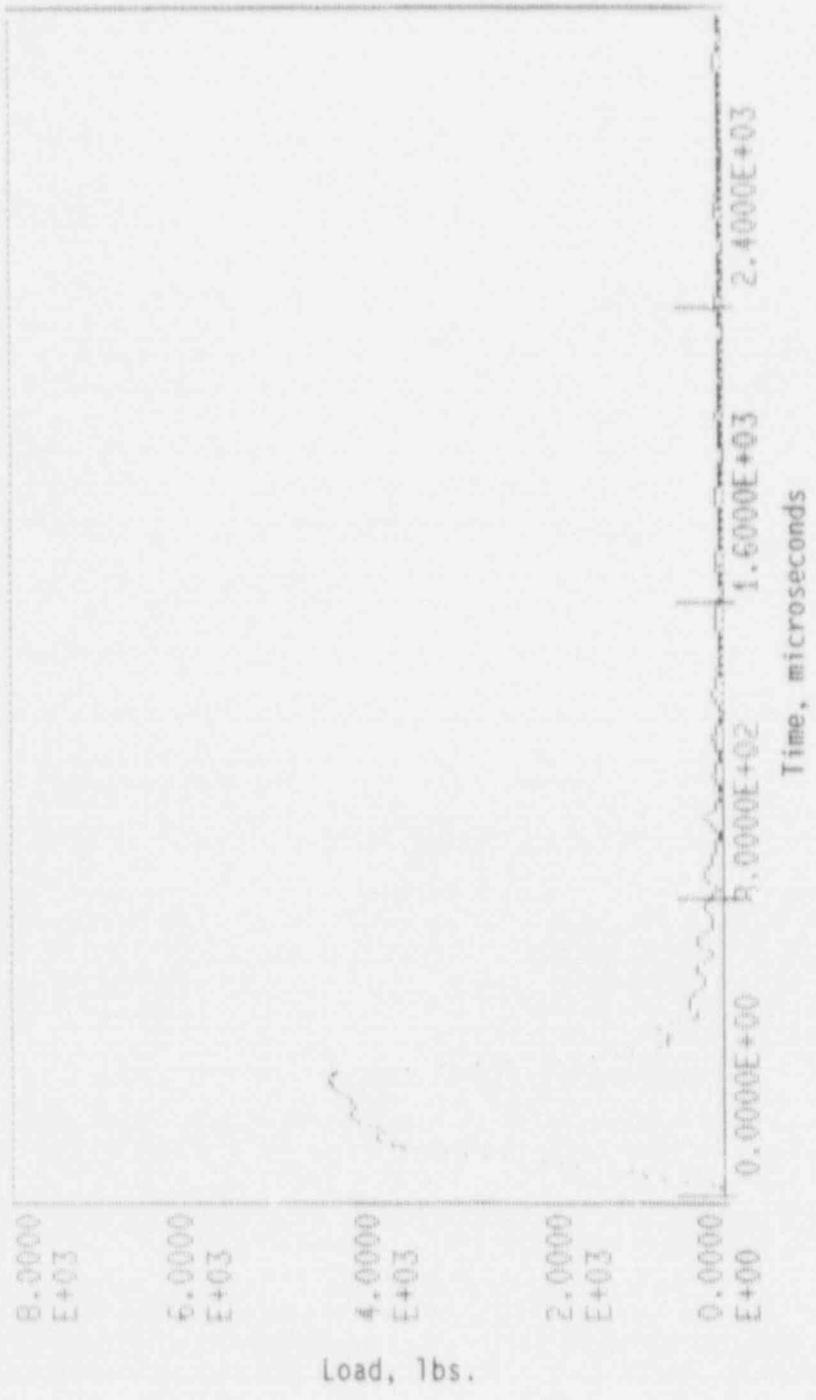
Figure A-5. Load-time record for Specimen 15M
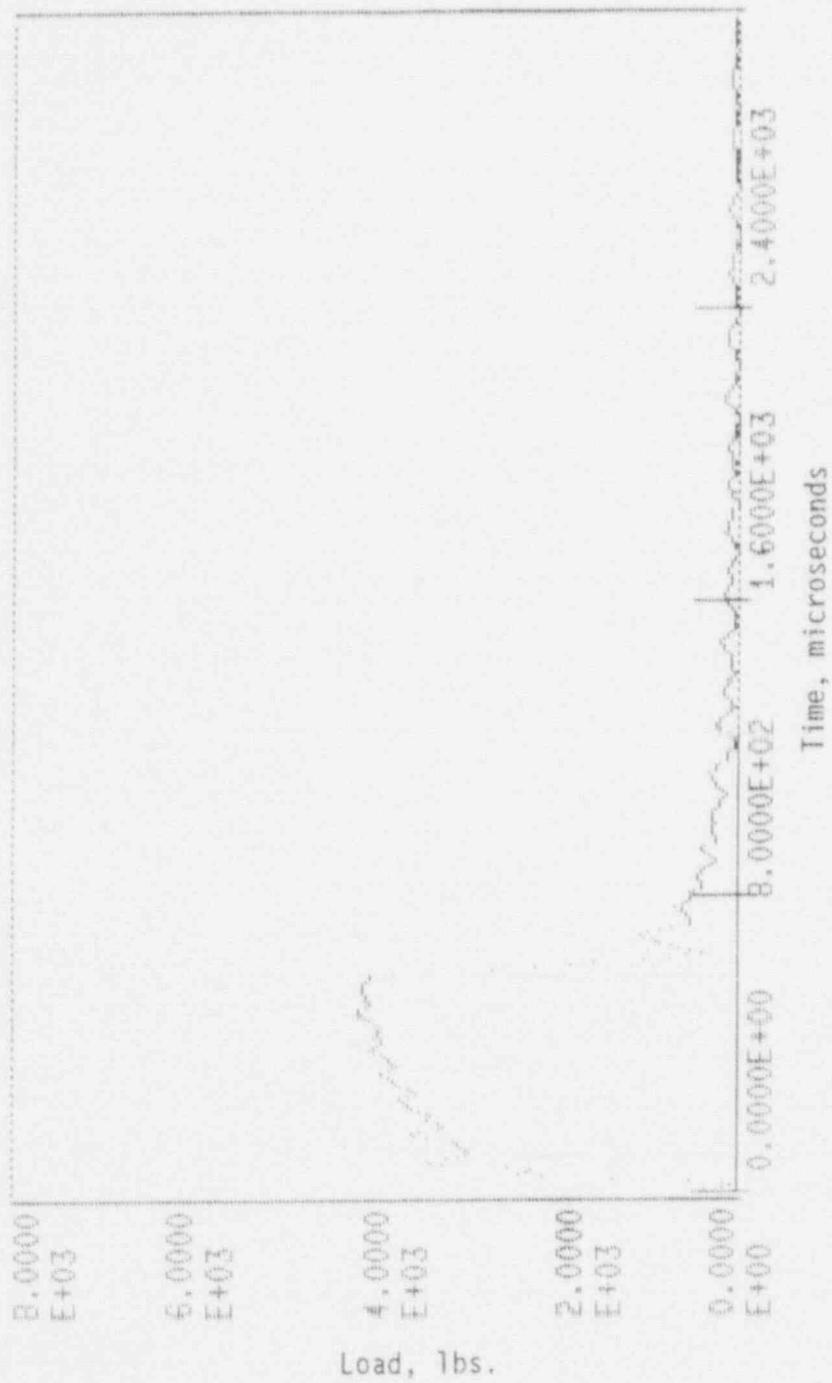
Figure A-6. Load-time record for Specimen 124
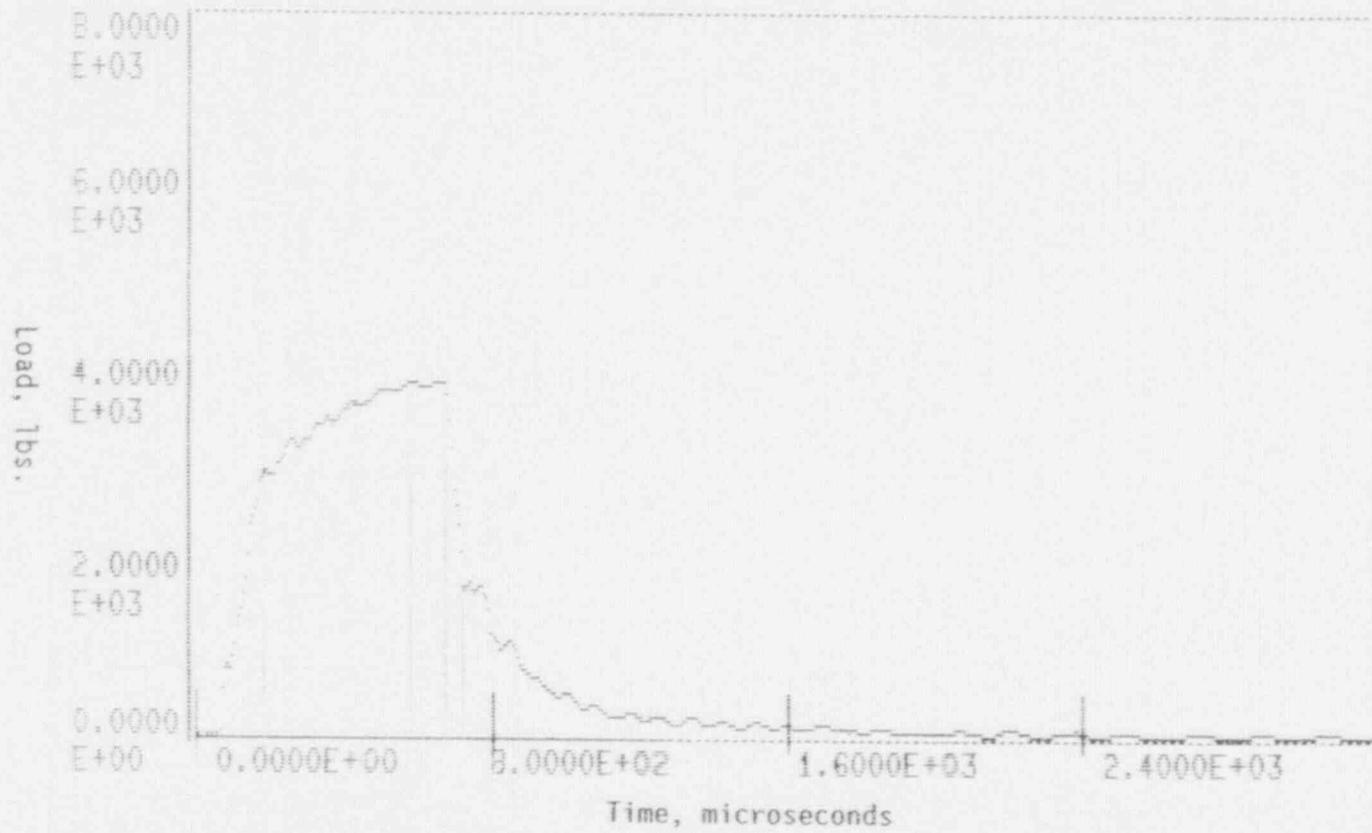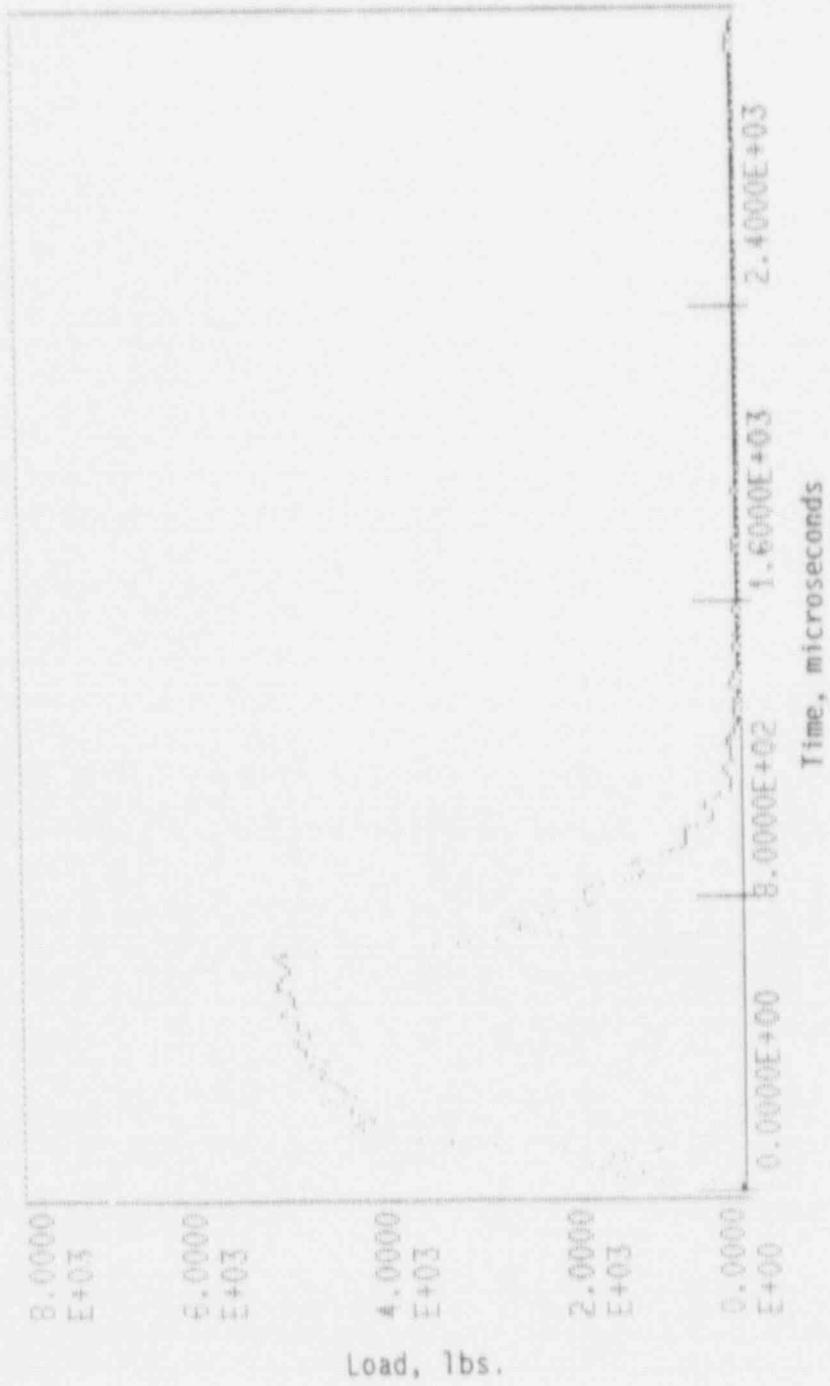
Figure A-7. Load-time record for Specimen 12C
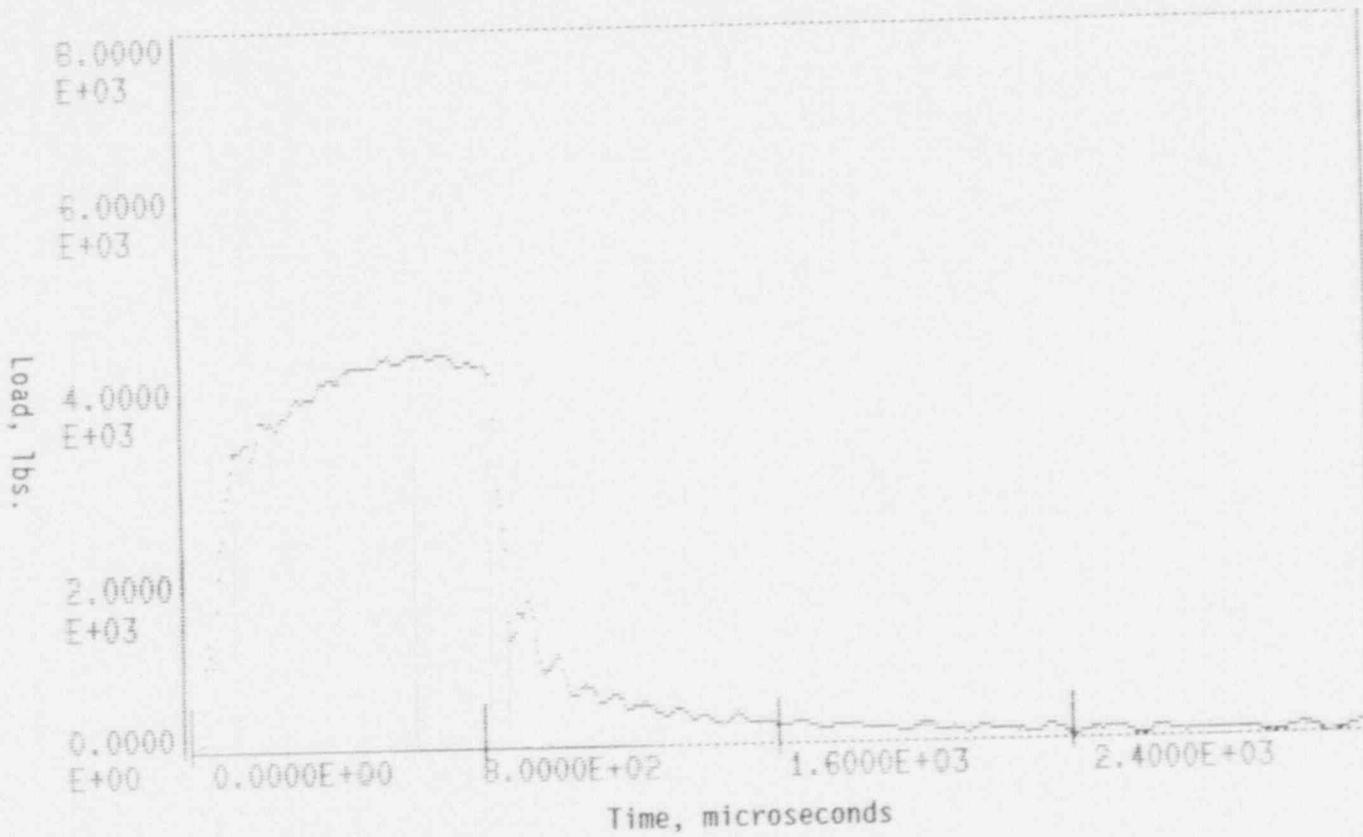
Figure A-8. Load-time record for Specimen 126
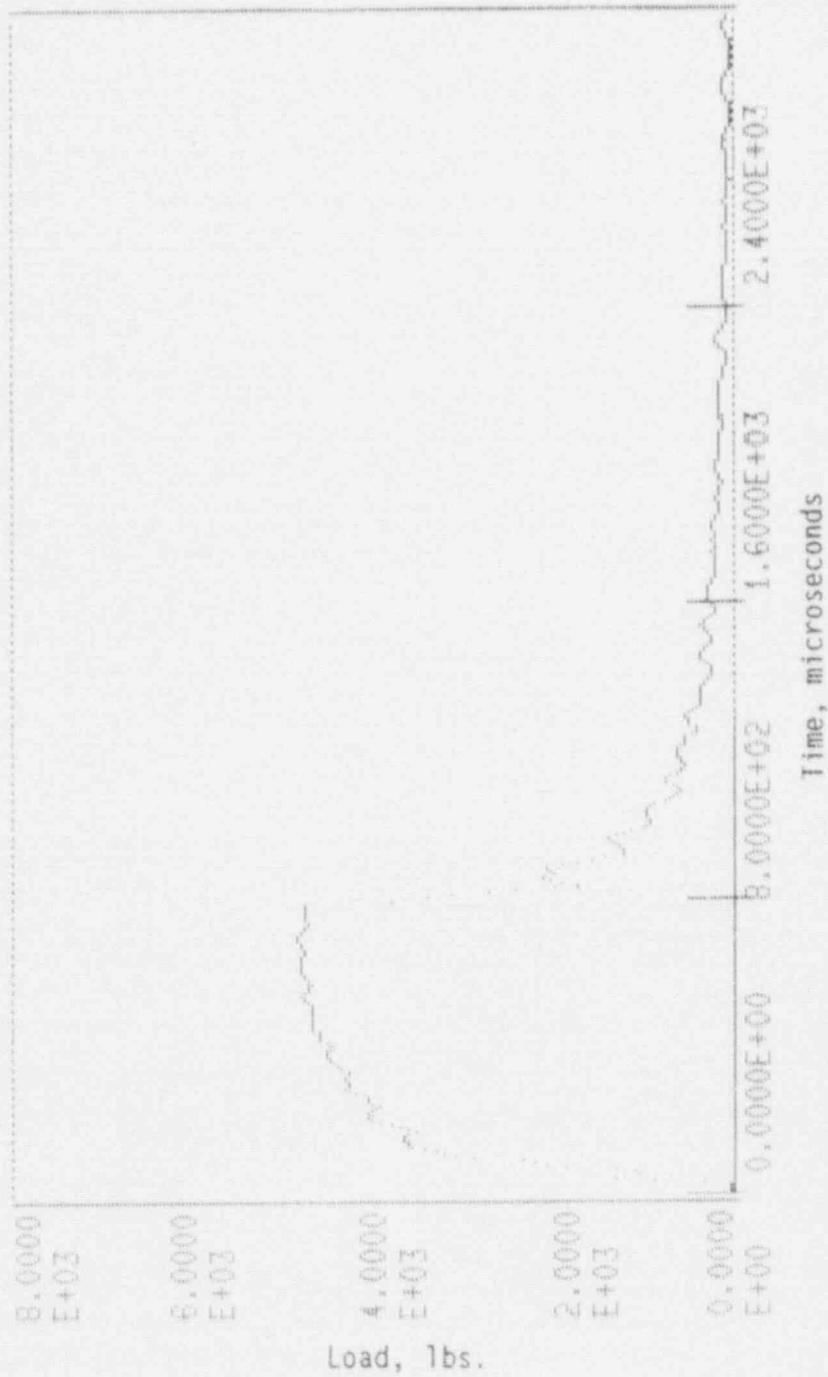
Figure A-9. Load-time record for Specimen 12M
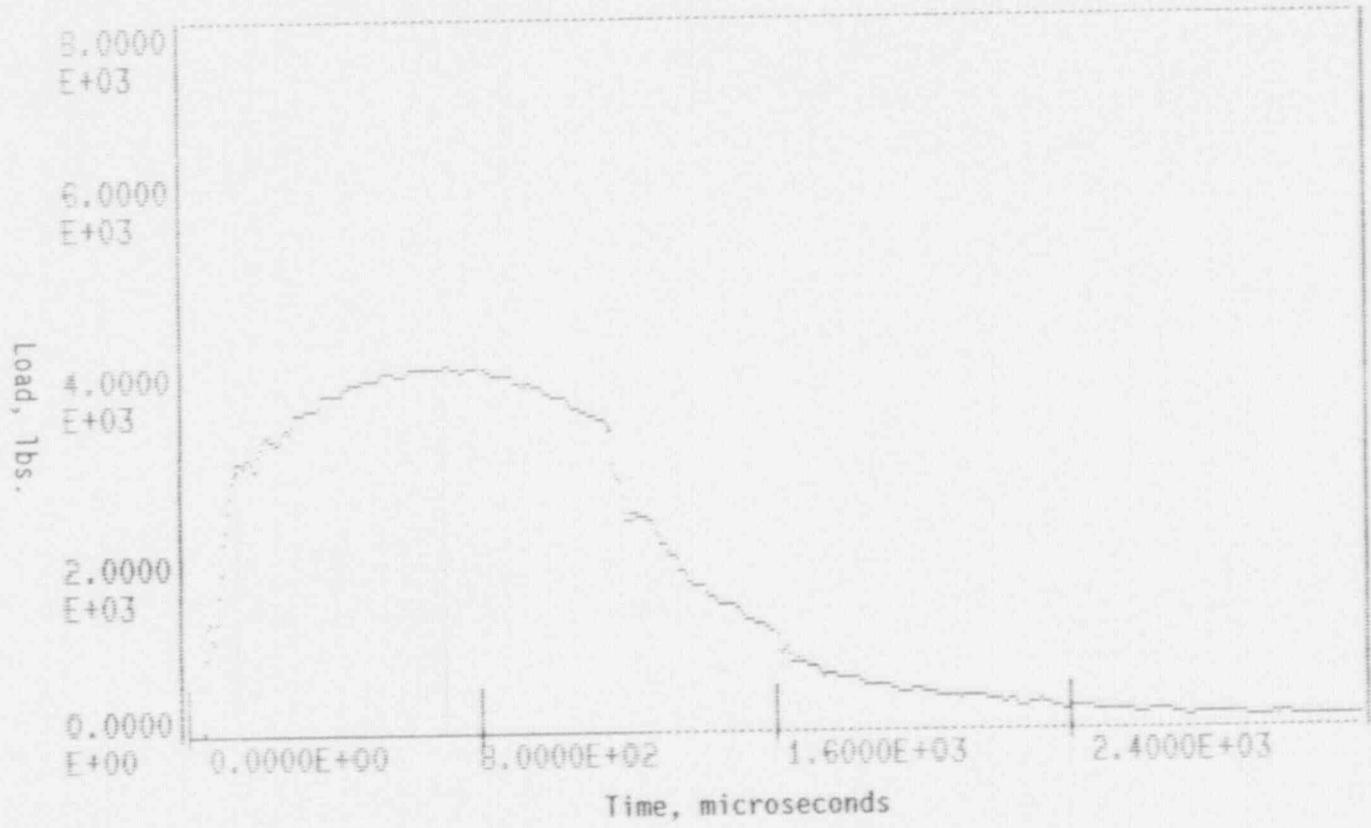
Figure A-10. Load-time record for Specimen 12P
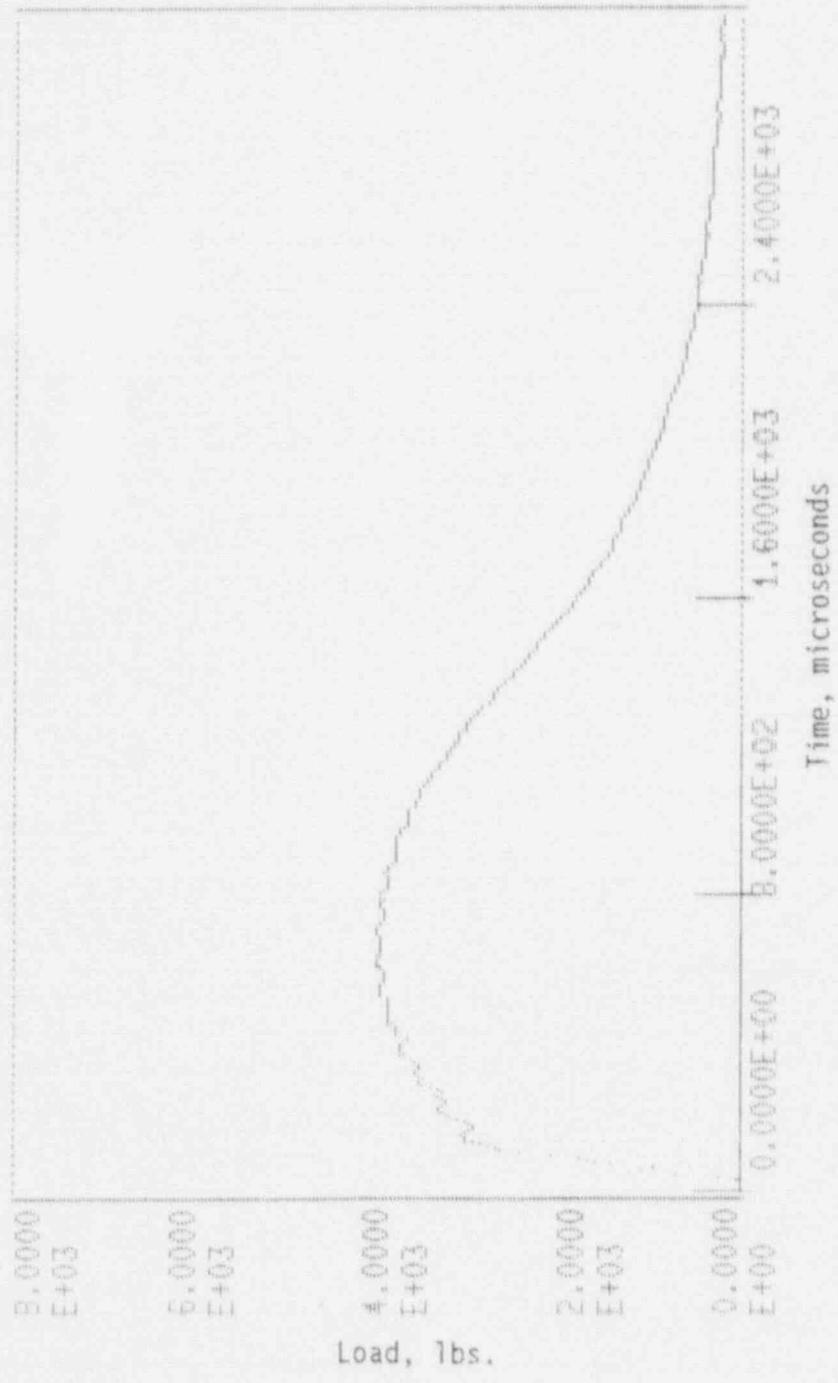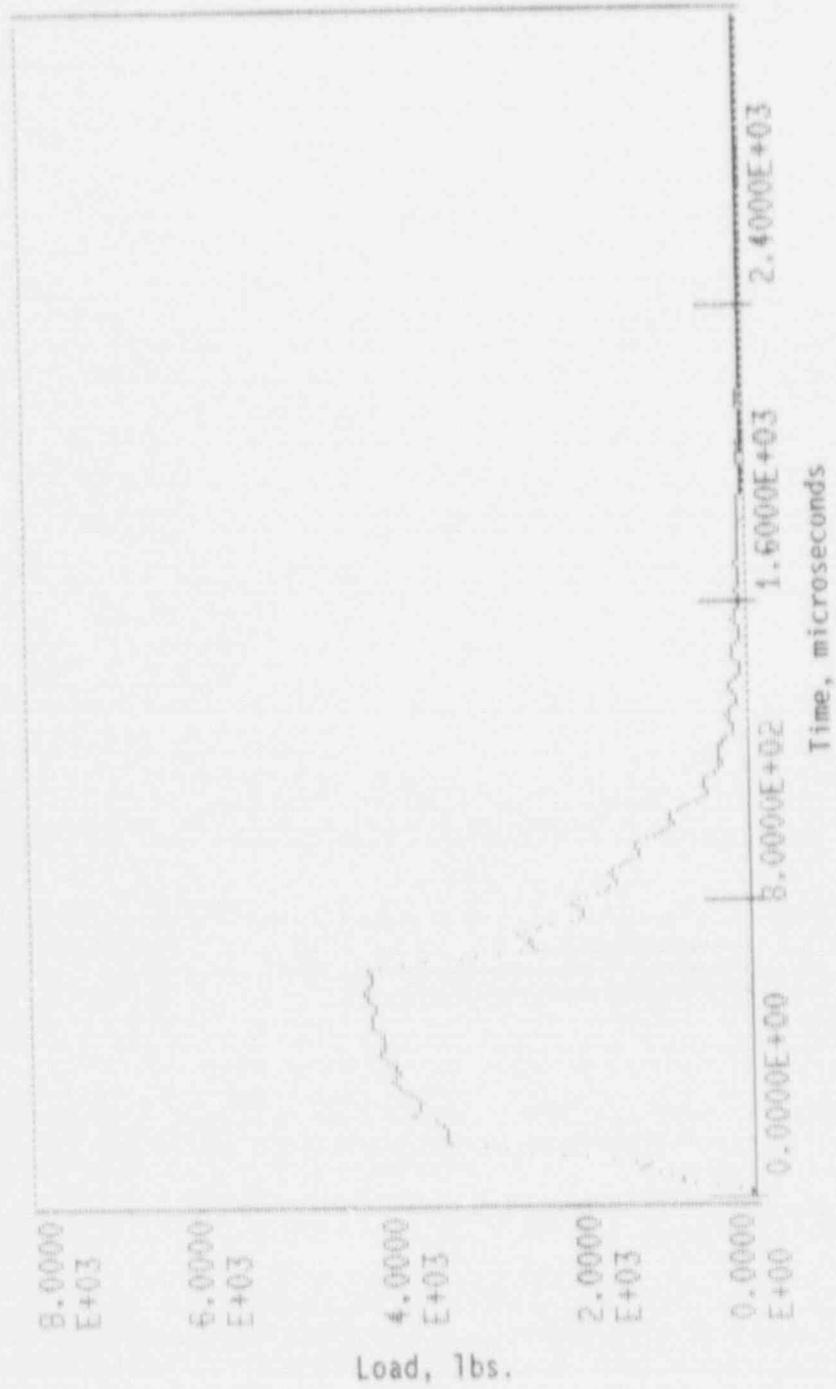
Figure A-11. Load-time record for Specimen 12D
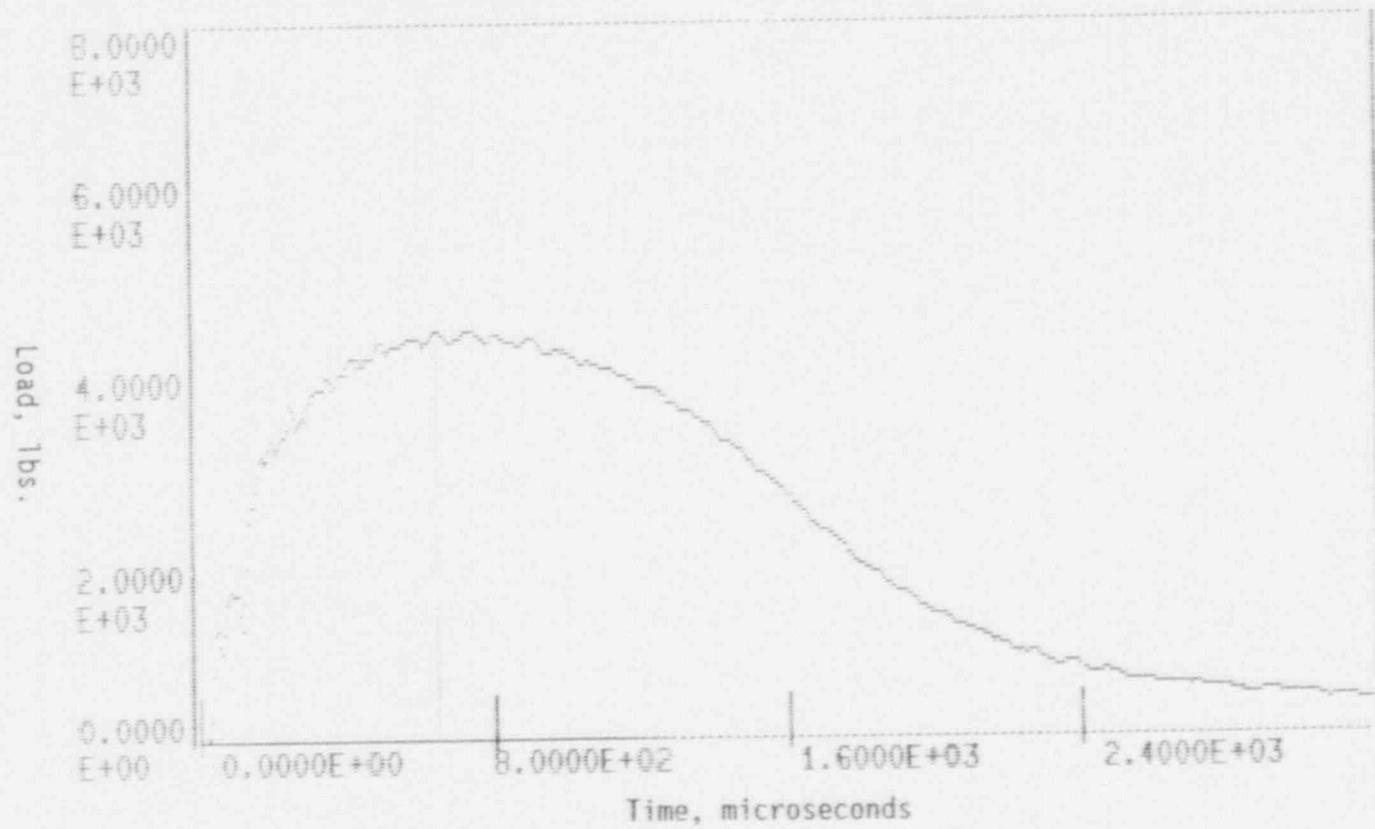
Figure A-12. Load-time record for Specimen 12U
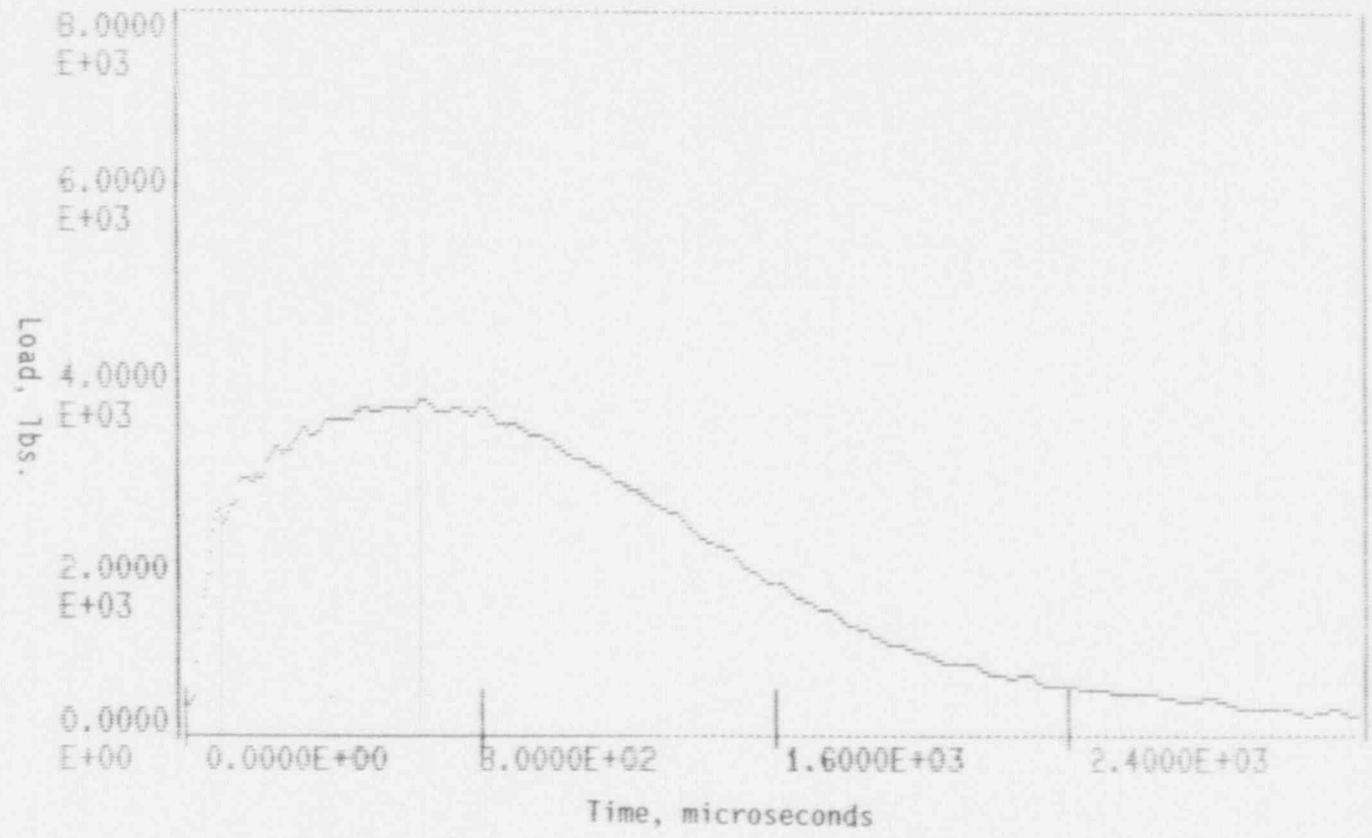
Figure A-13. Load-time record for Specimen 13K
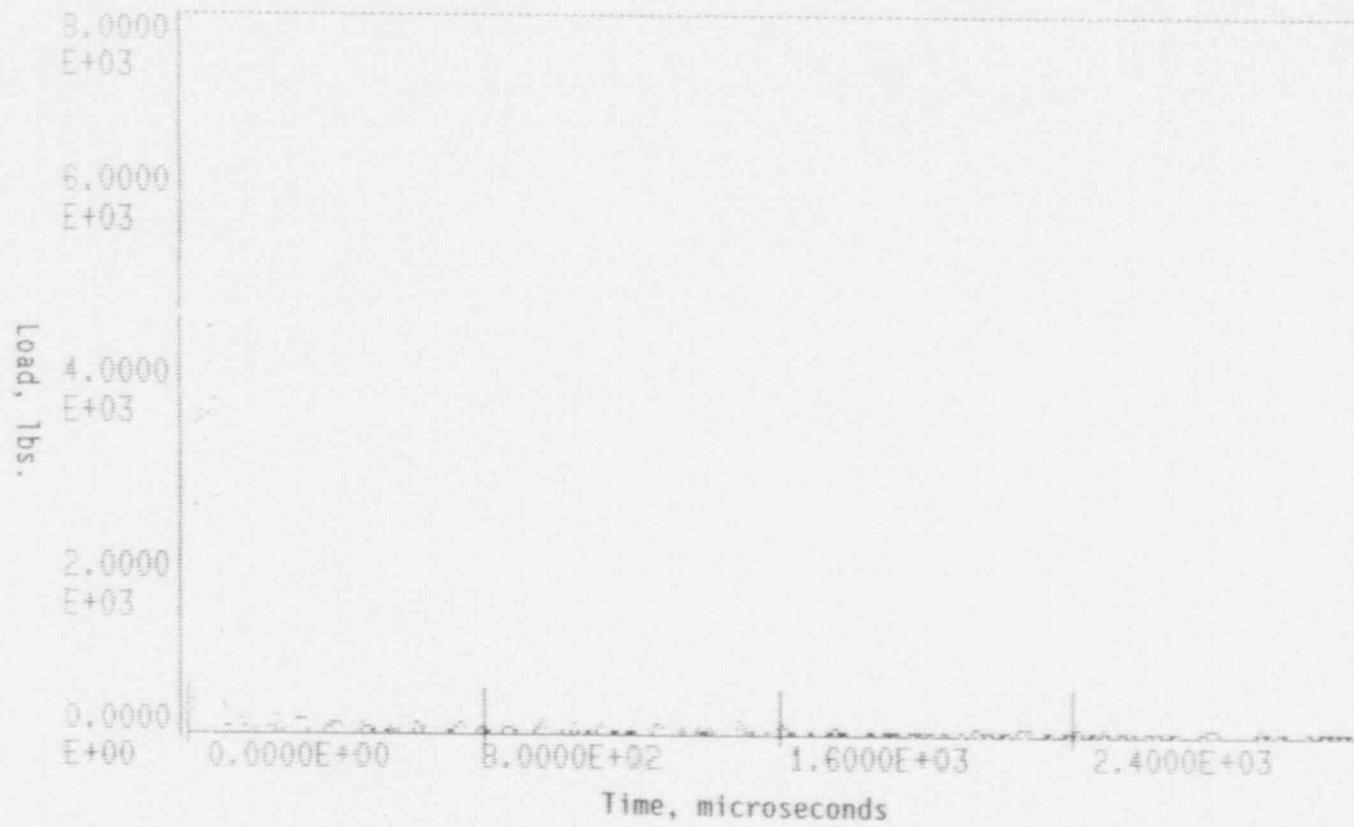
Figure A-14. Load-time record for Specimen 31J
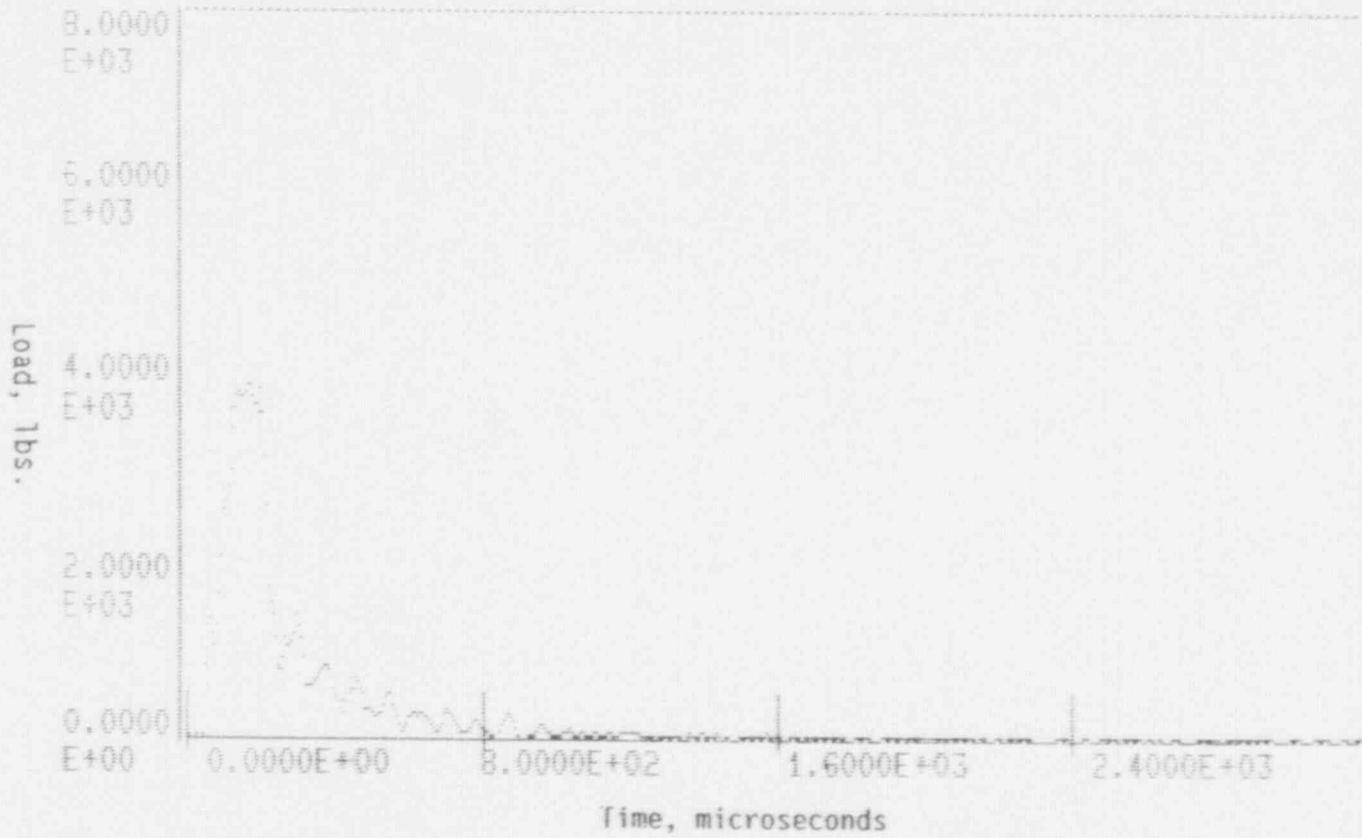
Figure A-15. Load-time record for Specimen 32L

Figure A-16. Load-time record for Specimen 32A
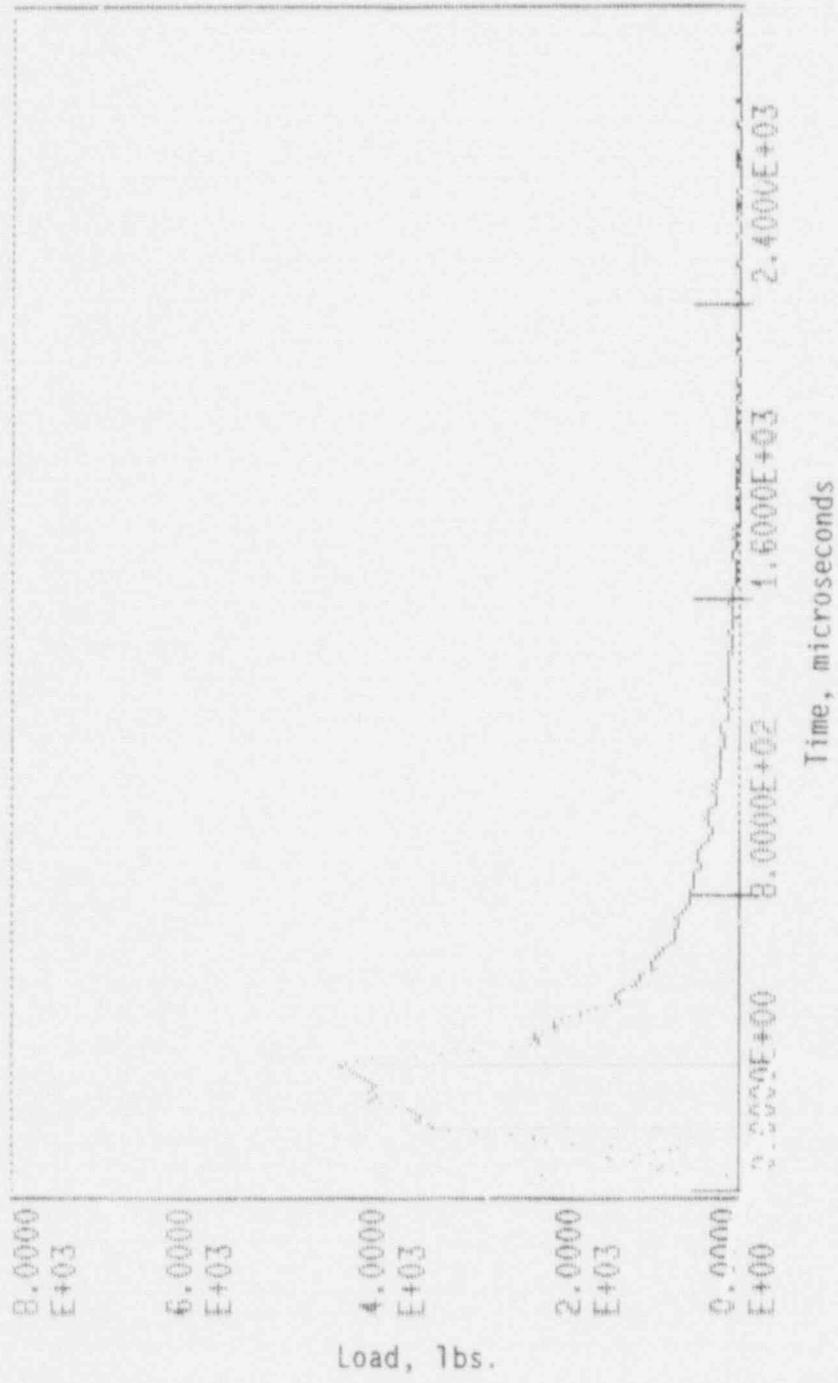
Figure A-17. Load-time record for Specimen 350
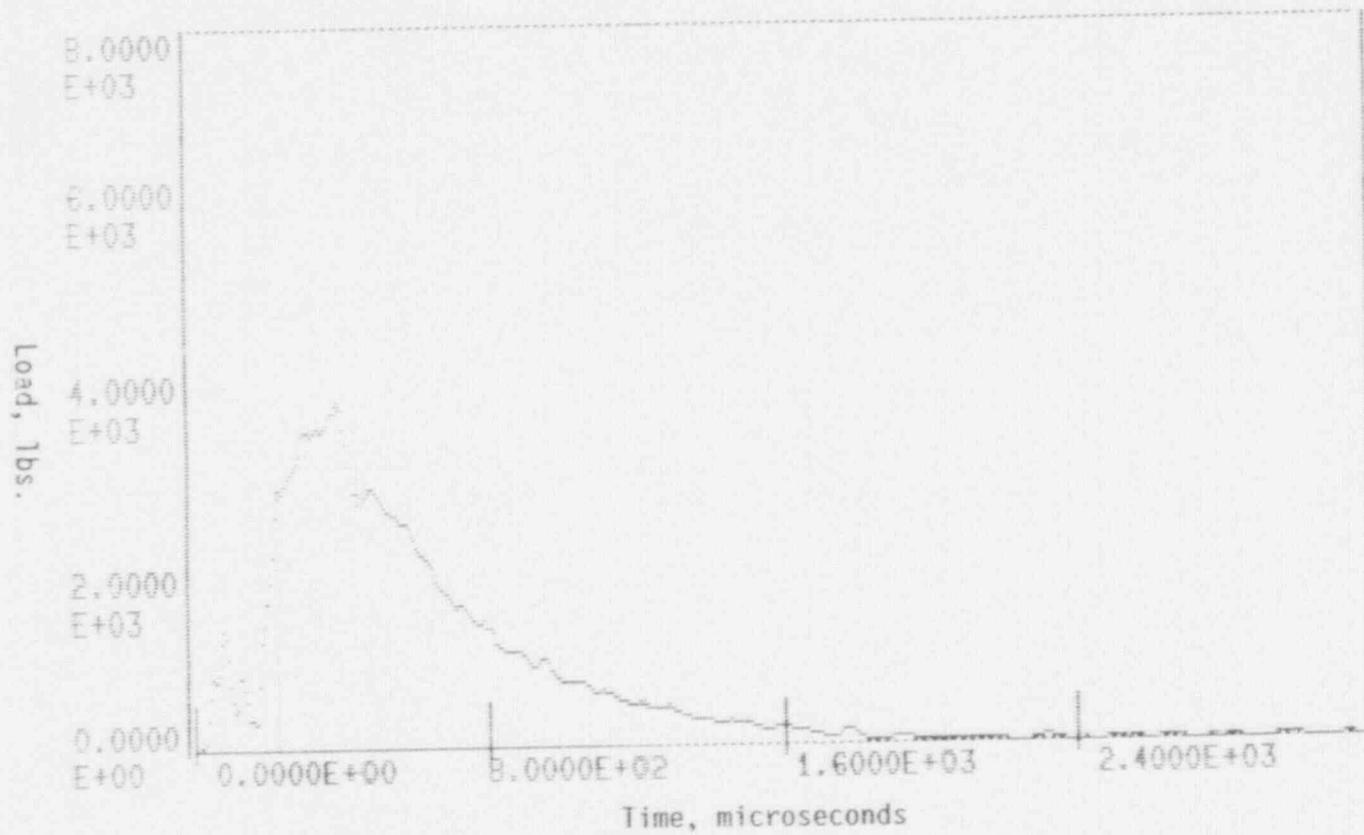
Figure A-18. Load-time record for Specimen 32K
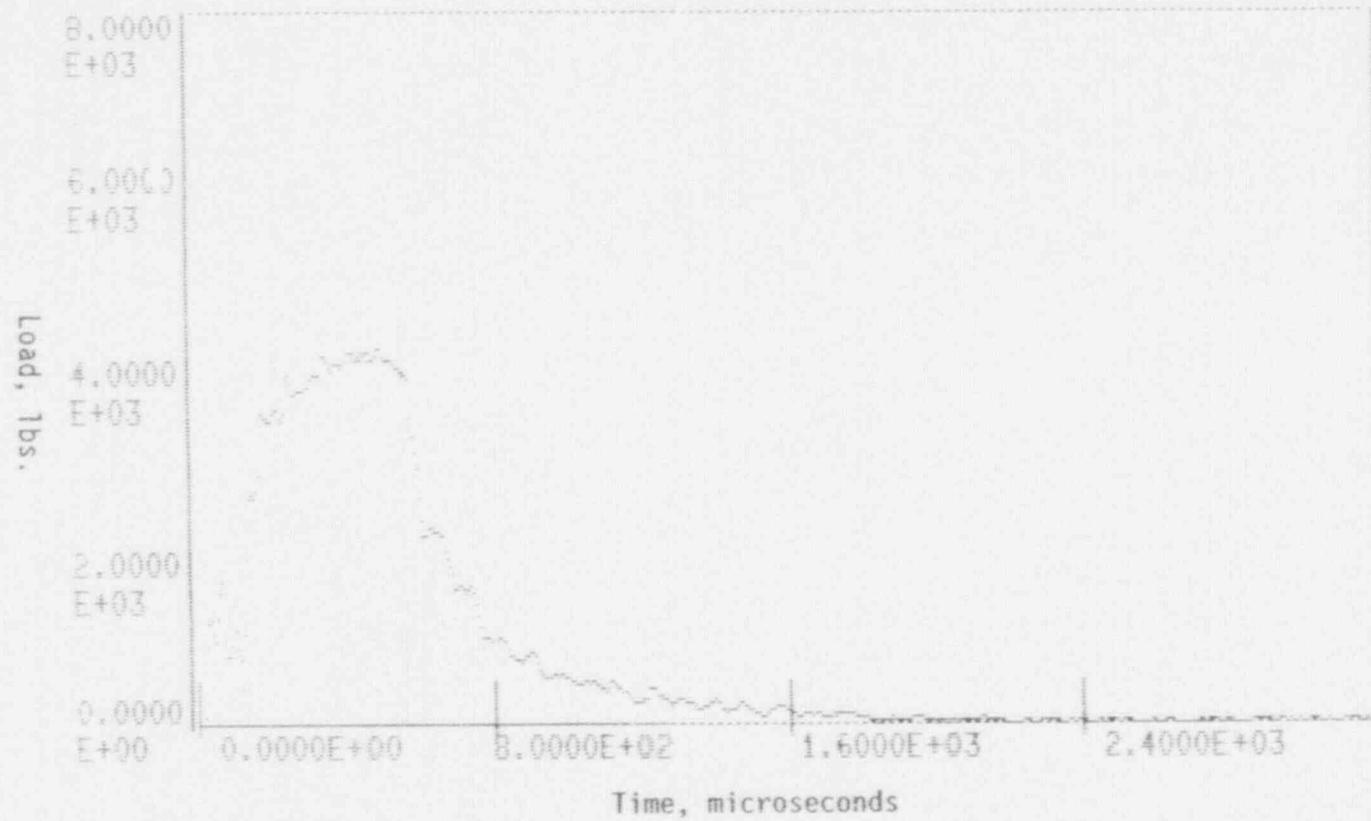
Figure A-19. Load-time record for Specimen 3A2
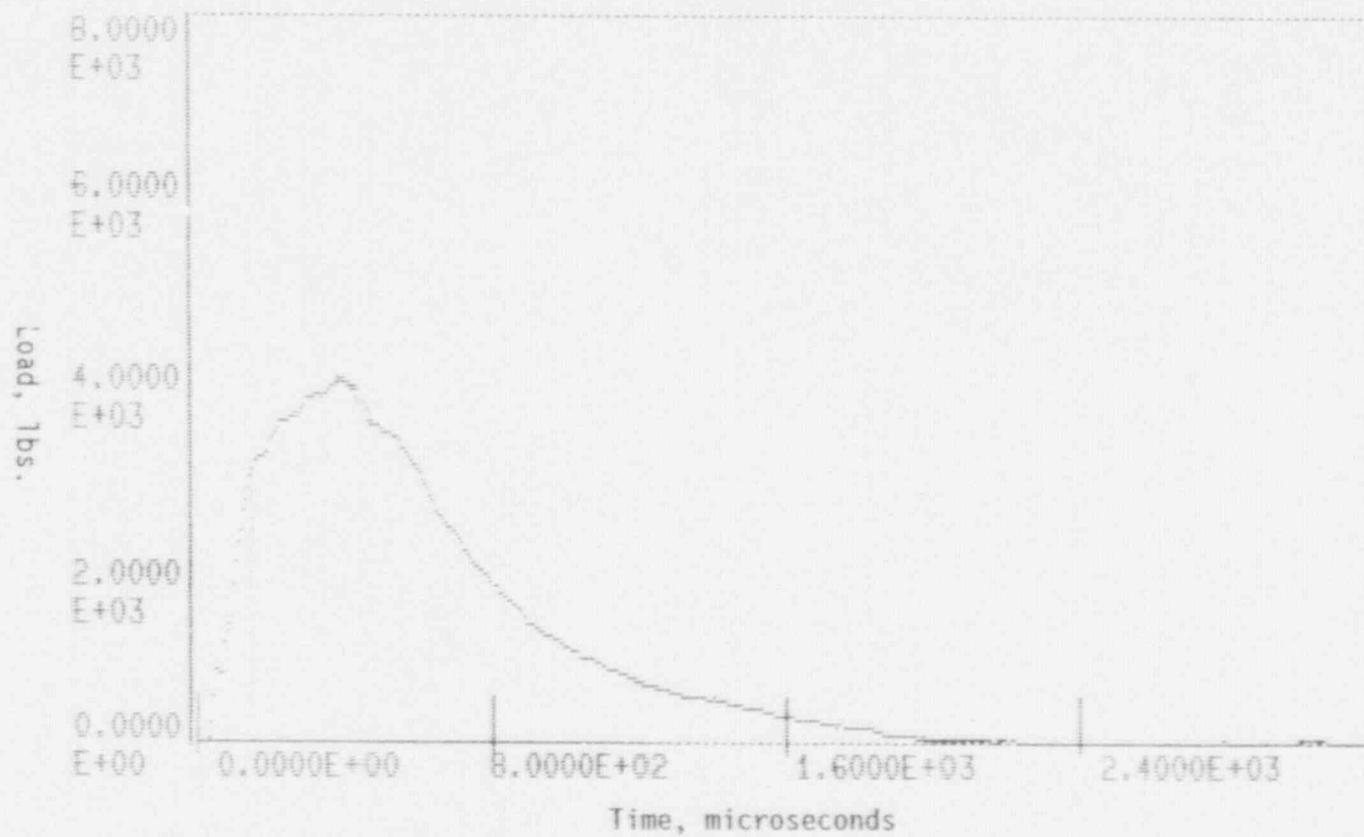
Figure A-20. Load-time record for Specimen 33M
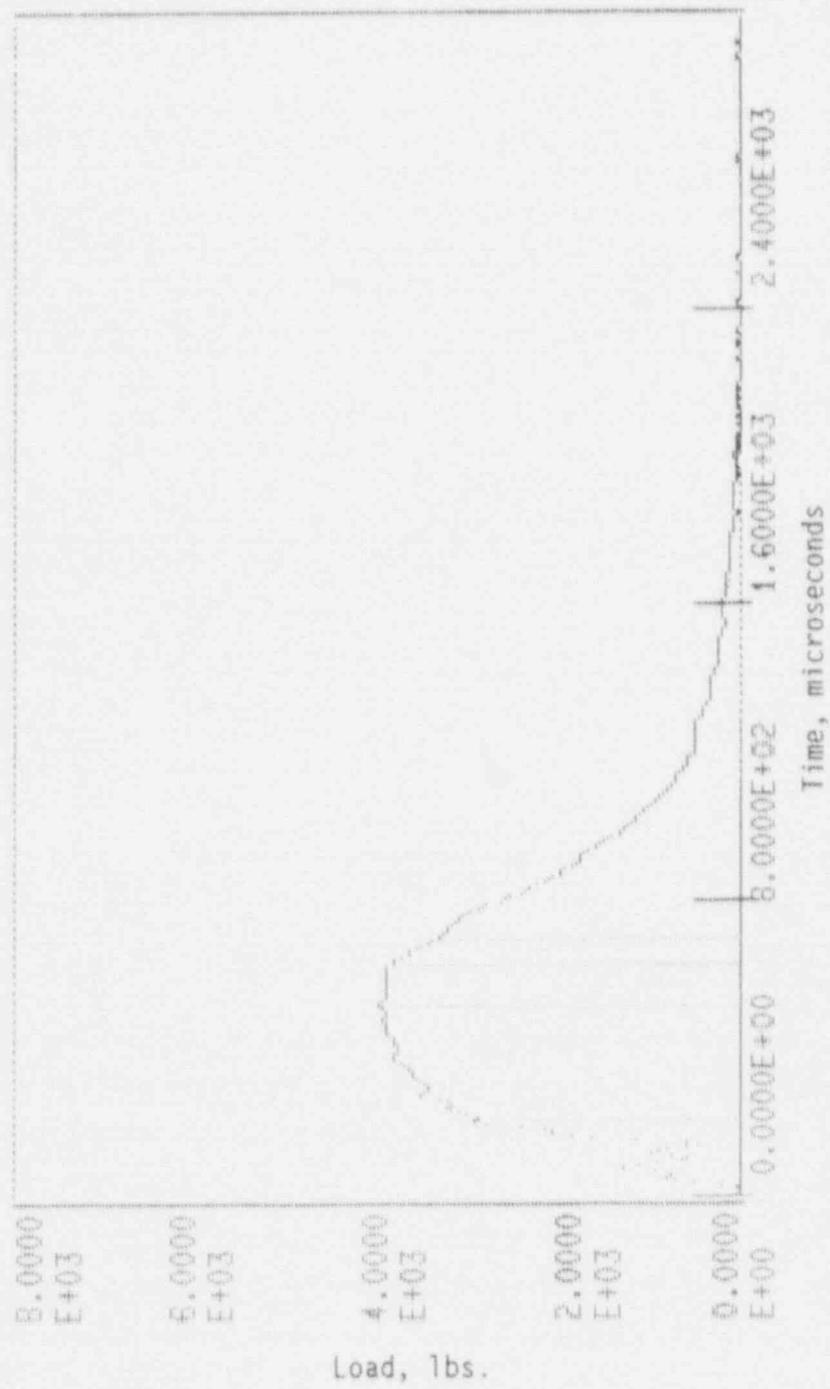
Figure A-21. Load-time record for Specimen 37J

Figure A-22. Load-time record for Specimen 317

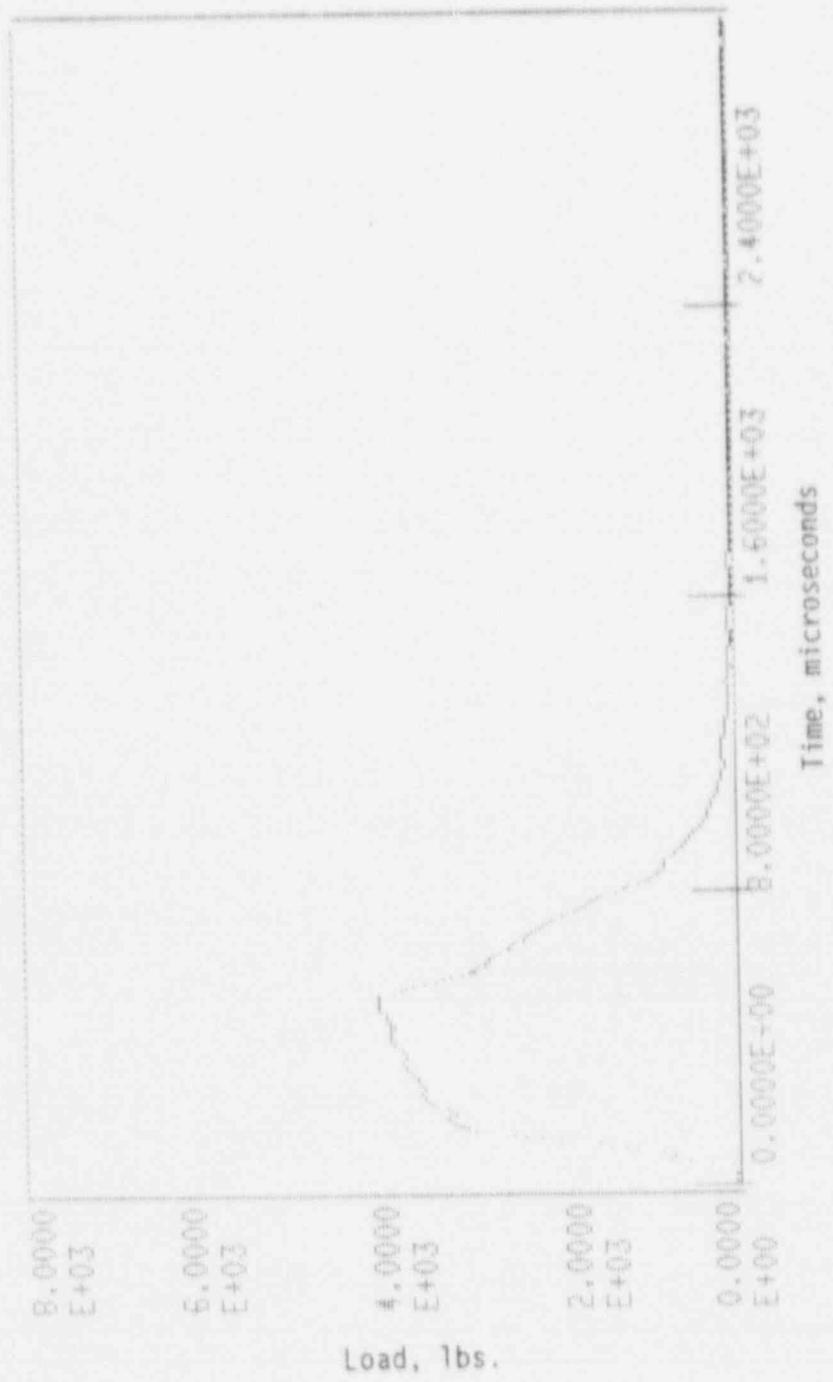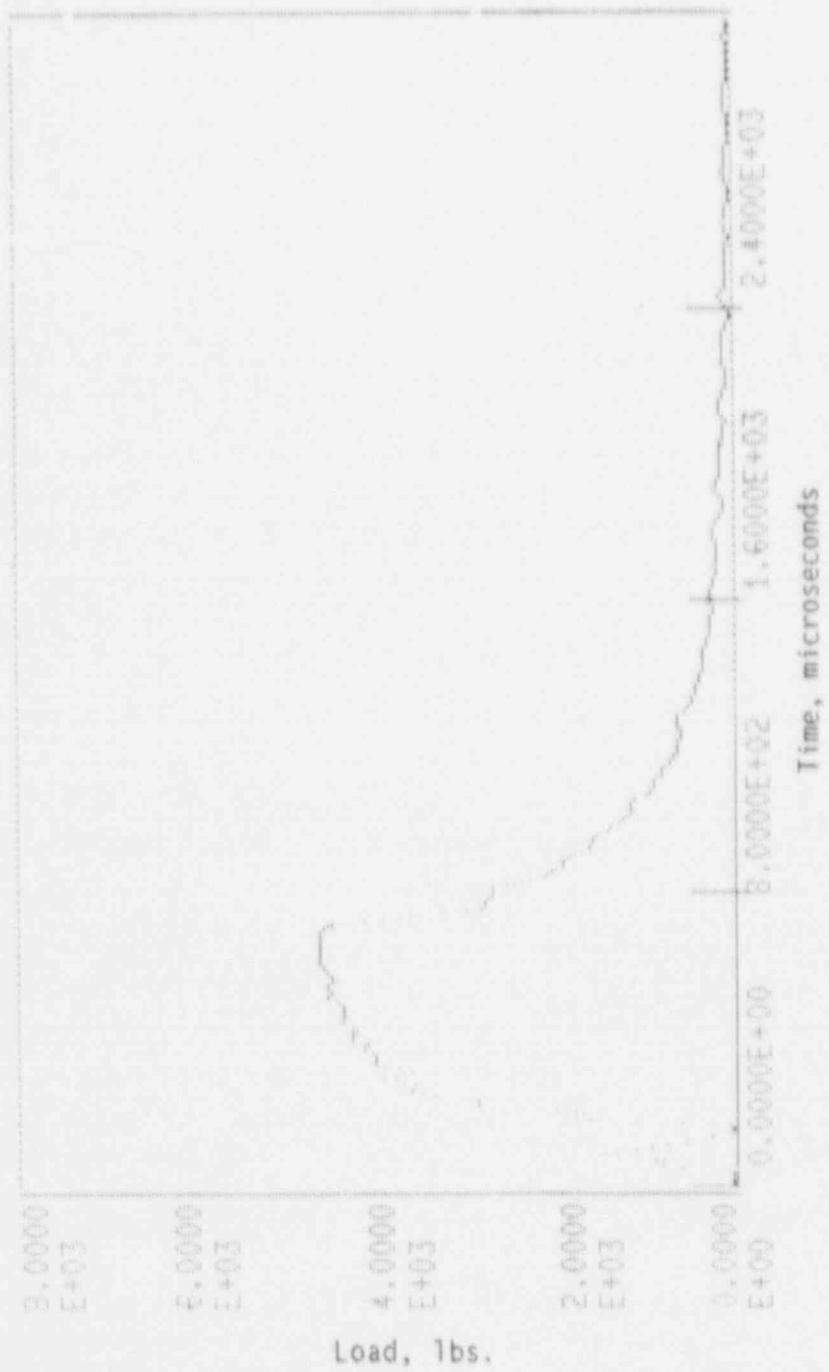Figure A-23. Load-time record for Specimen 36C

Figure A-24. Load-time record for Specimen 312
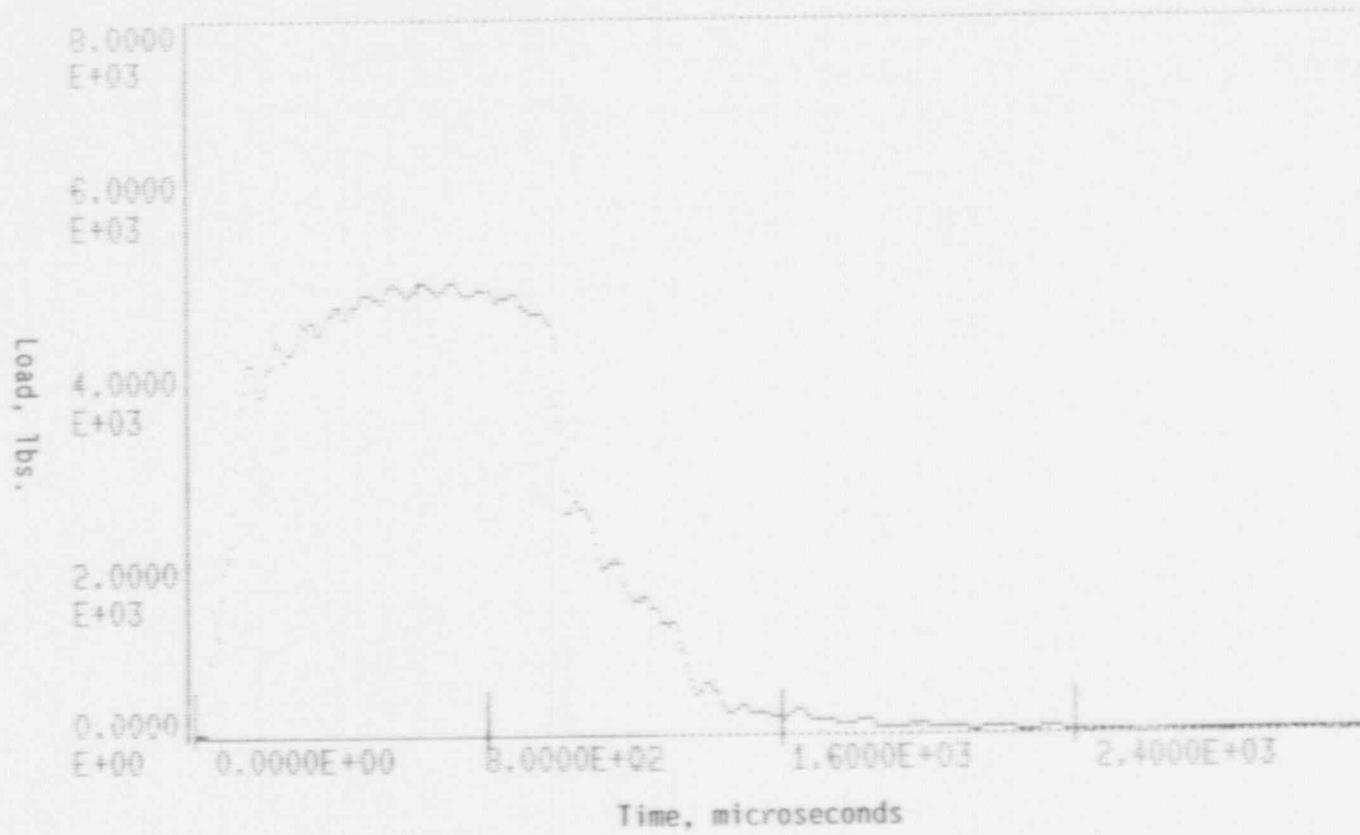
A-24

Figure A-25. Load-time record for Specimen 47Y

Figure A-26. Load-time record for Specimen 43J

Figure A-27. Load-time record for Specimen 433

Figure A-28. Load-time record for Specimen 47C

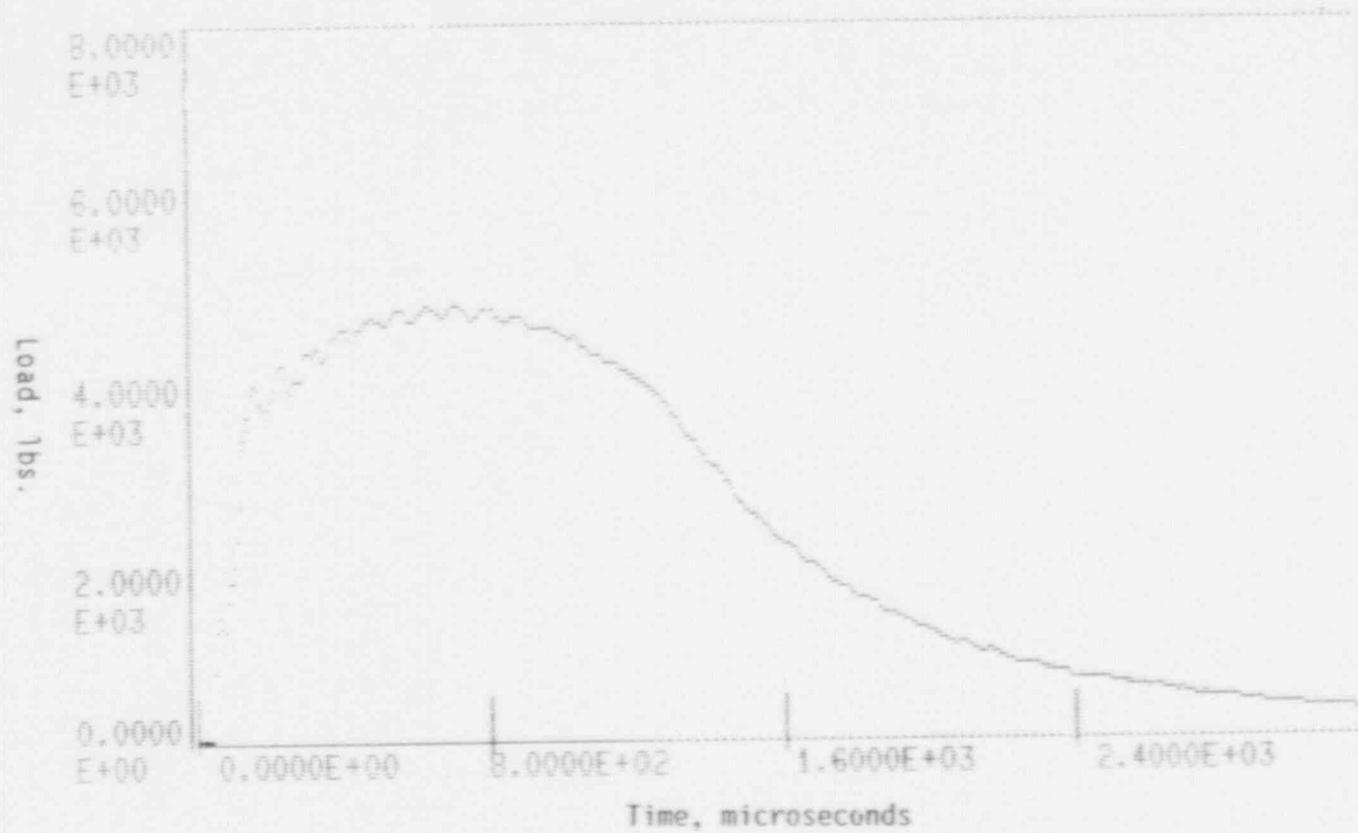Load, lbs.

Time, microseconds

Figure A-29. Load-time record for Specimen 47J

Figure A-30. Load-time record for Specimen 47T

Load, lbs.

Time, microseconds

8.0000 E+03
6.0000 E+03
4.0000 E+03
2.0000 E+03
0.0000 E+00

0.0000E+00
8.0000E+02
1.6000E+03
2.4000E+03

Figure A-31. Load-time record for Specimen 438

Load, lbs.

Time, microseconds

Figure A-32. Load-time record for Specimen 46Y
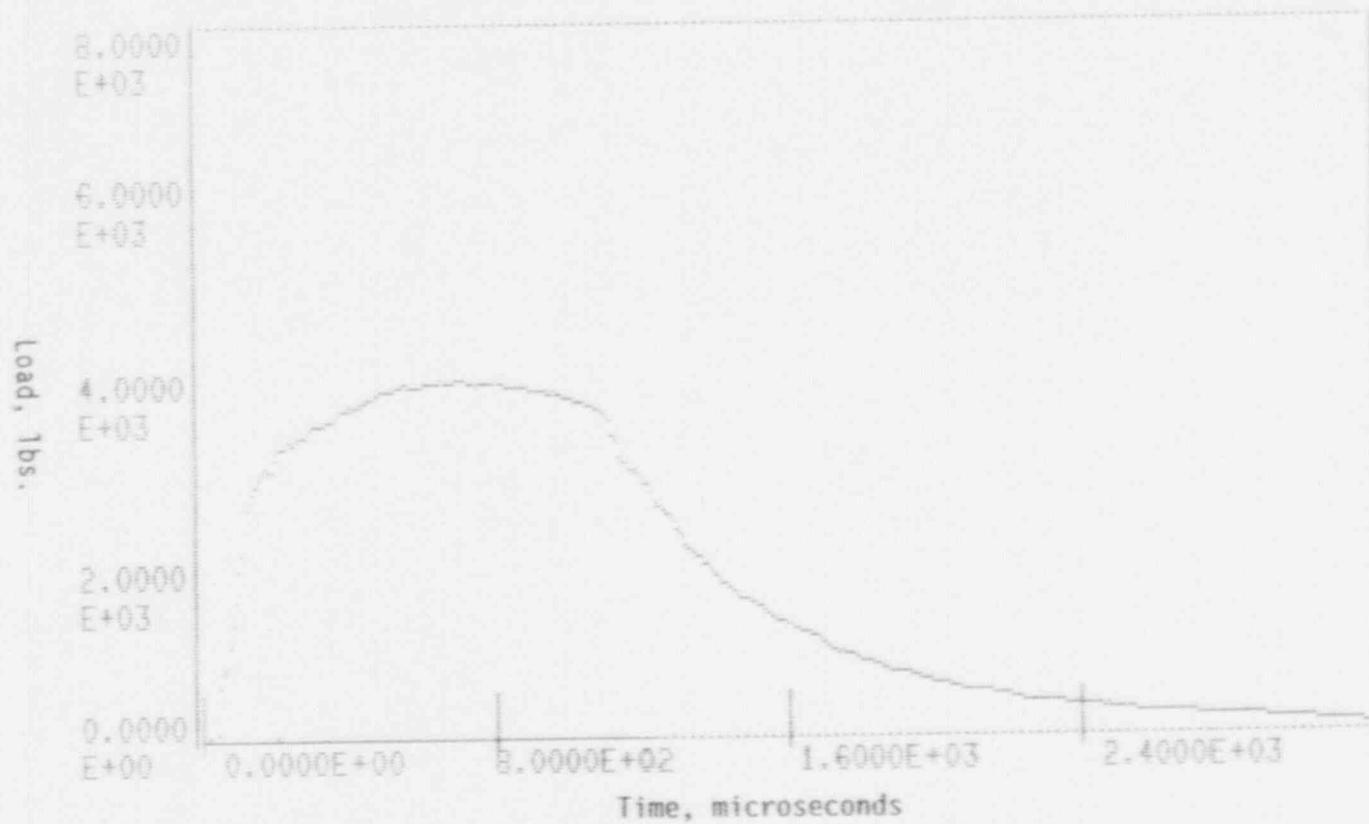
Figure A-33. Load-time record for Specimen 43K
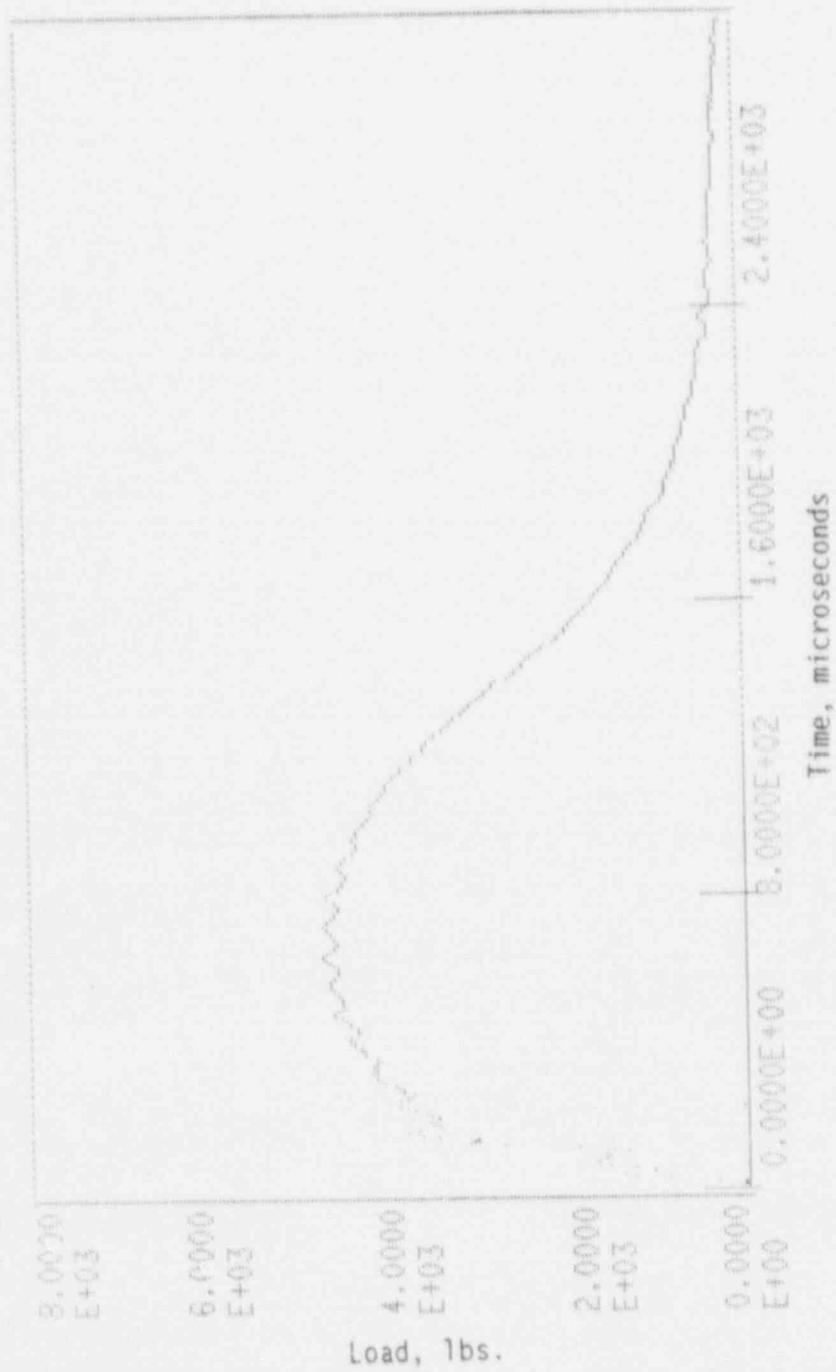
Figure A-34. Load-time record for Specimen 46U
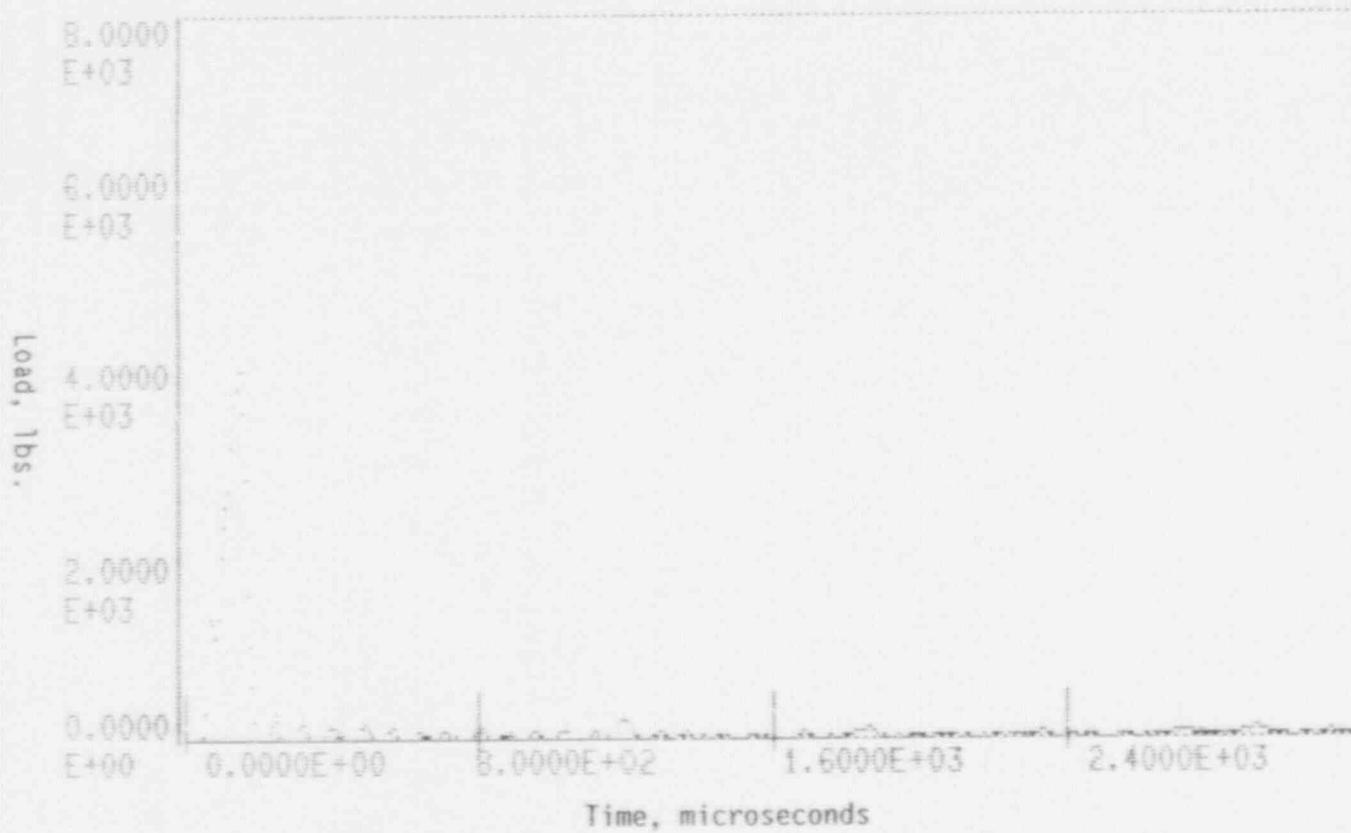
Figure A-35. Load-time record for Specimen 473

Figure A-36. Load-time record for Specimen 61E

Figure A-37. Load-time record for Specimen 614

Load, lbs.

Time, microseconds

Figure A-38. Load-time record for Specimen 61C

Load, 1bs.

8.0000 E+03

6.0000 E+03

4.0000 E+03

2.0000 E+03

0.0000 E+00

0.0000E+00   8.0000E+02   1.6000E+03   2.4000E+03

Time, microseconds

Figure A-39. Load-time record for Specimen 61M

Load, lbs.

Time, microseconds

Figure A-40. Load-time record for Specimen 627

Load, lbs.

Time, microseconds
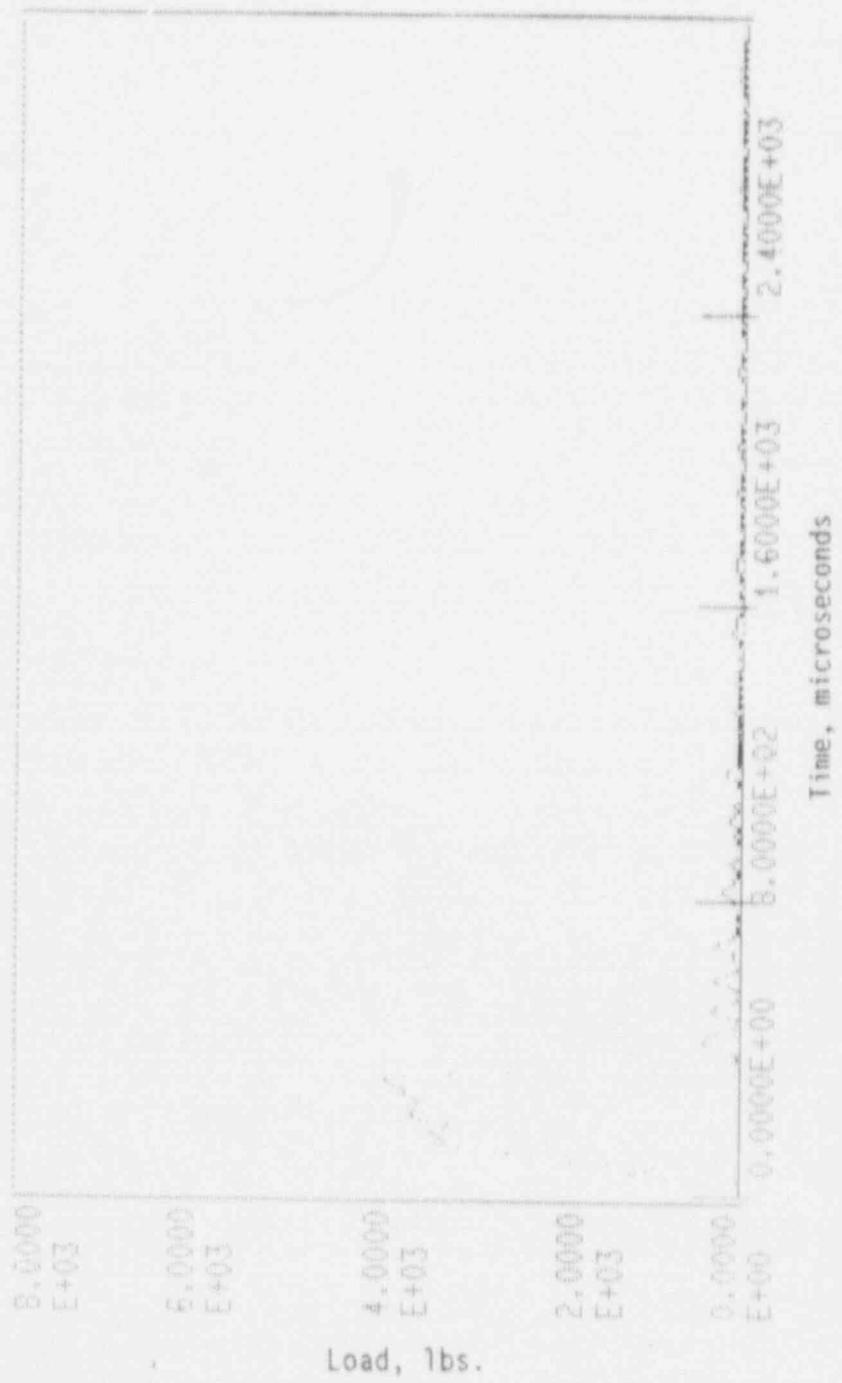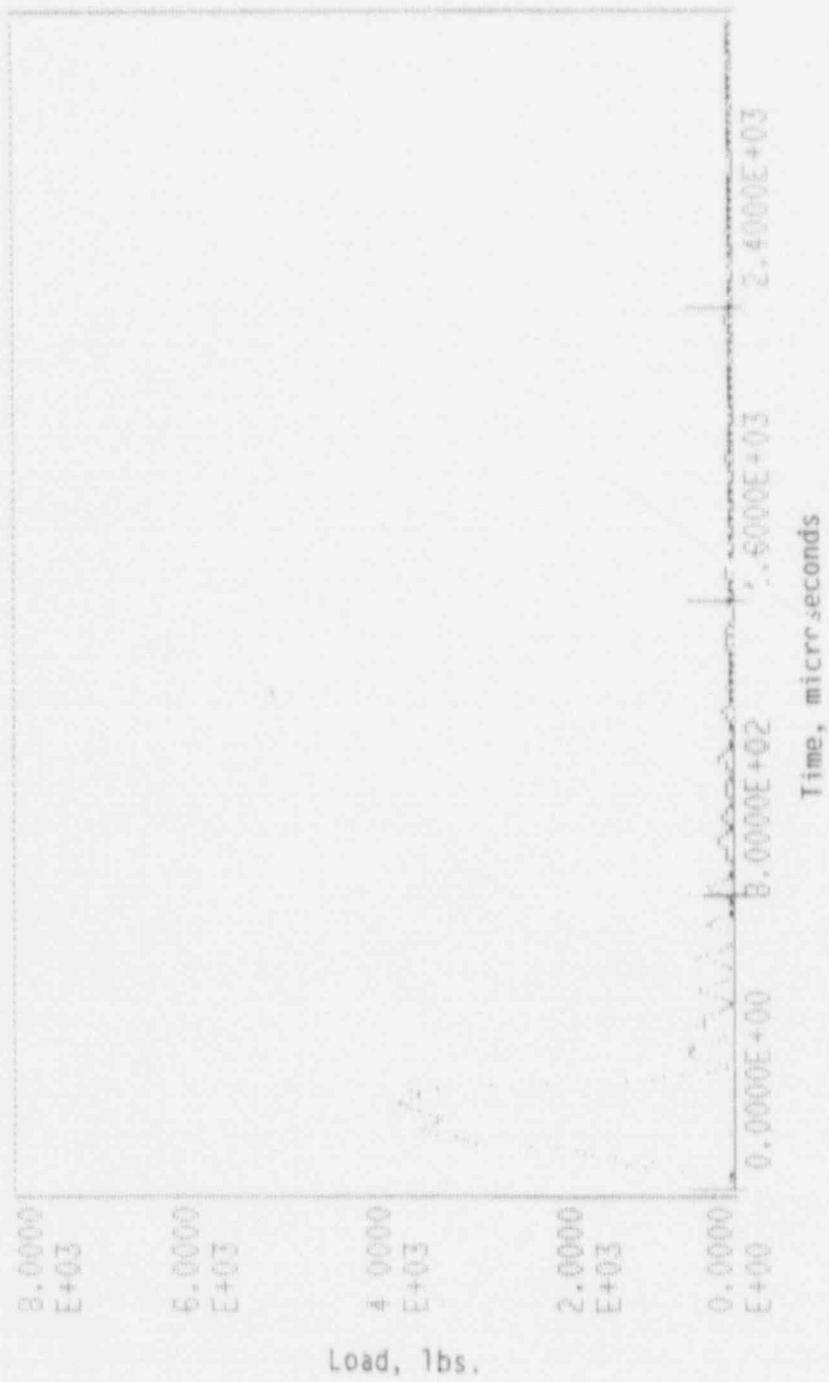
Figure A-41. Load-time record for Specimen 62A
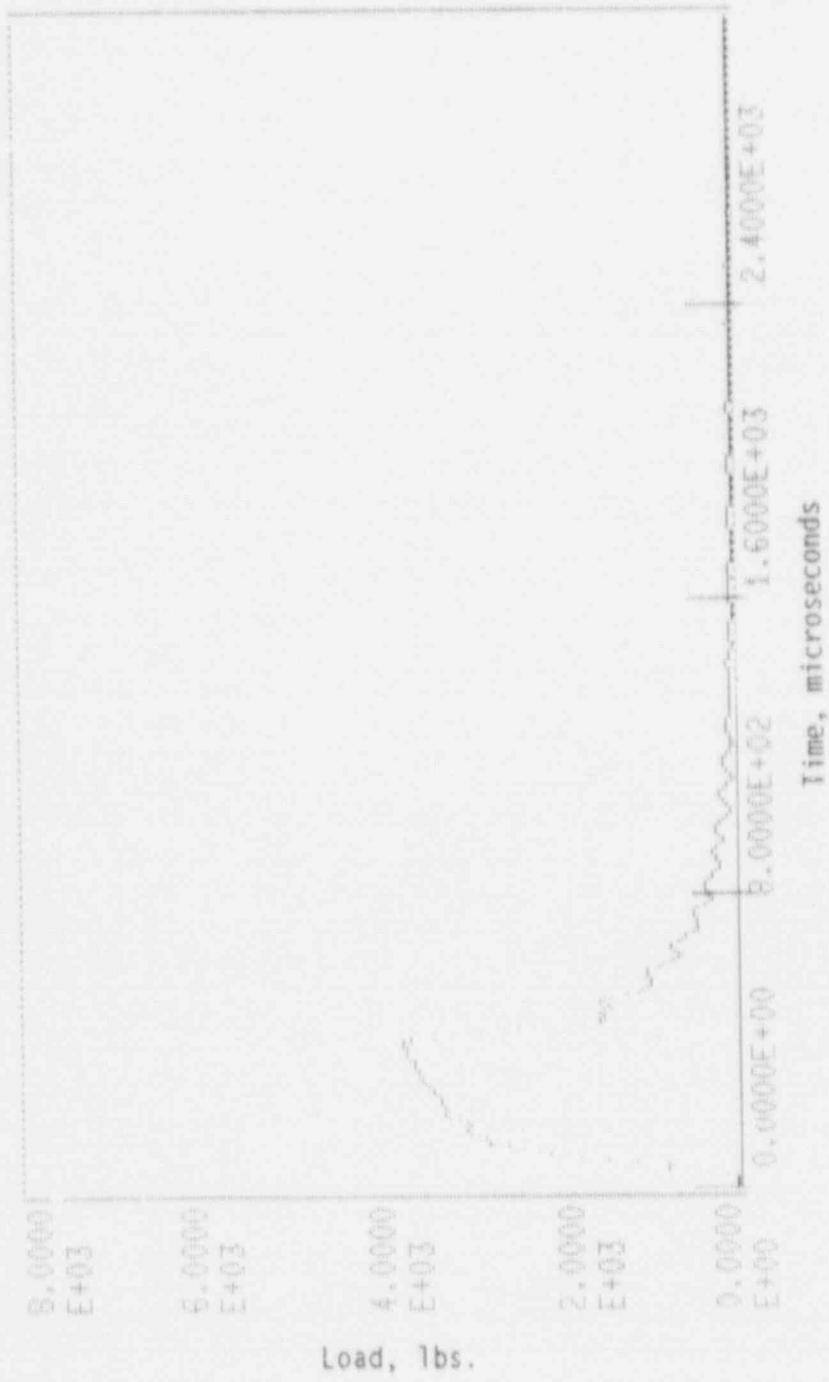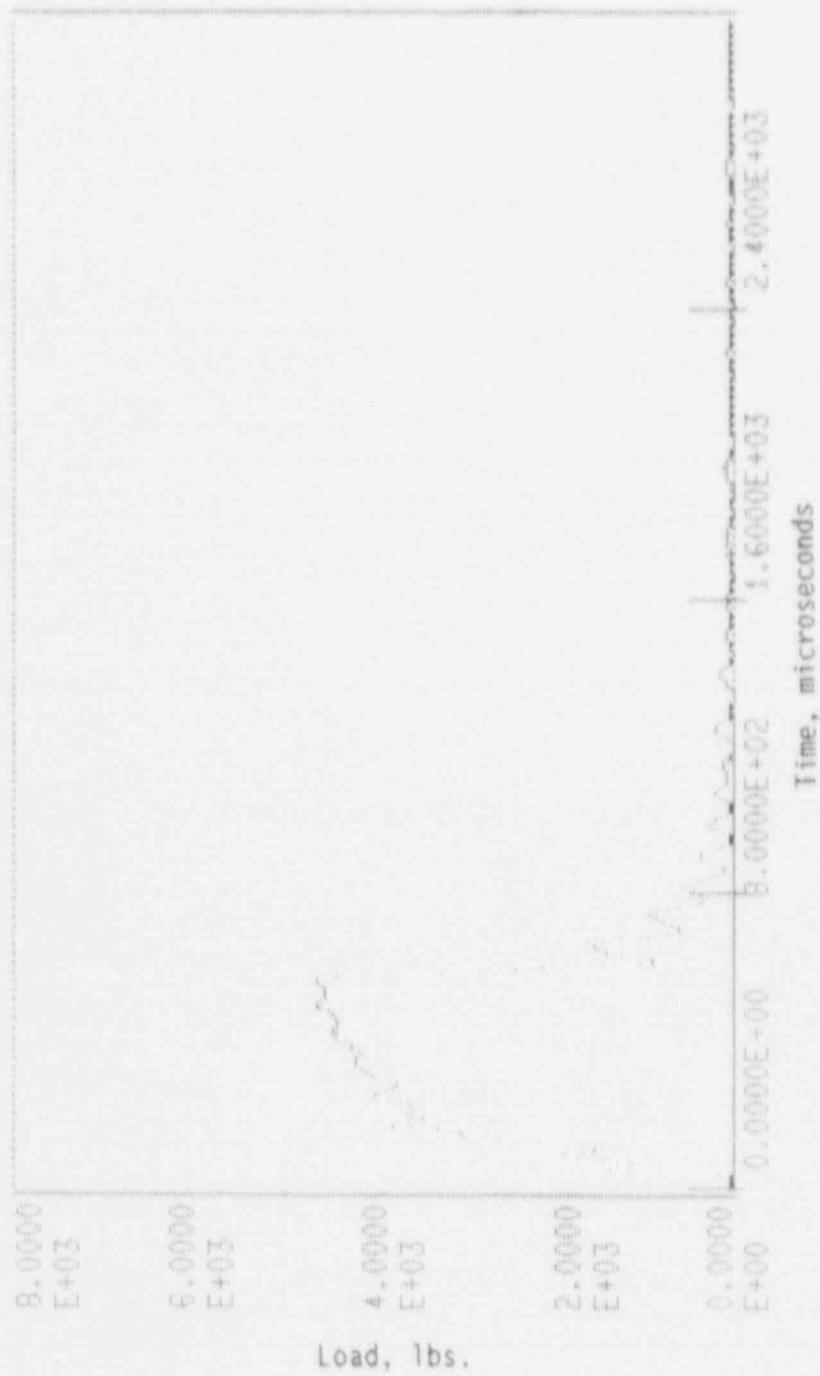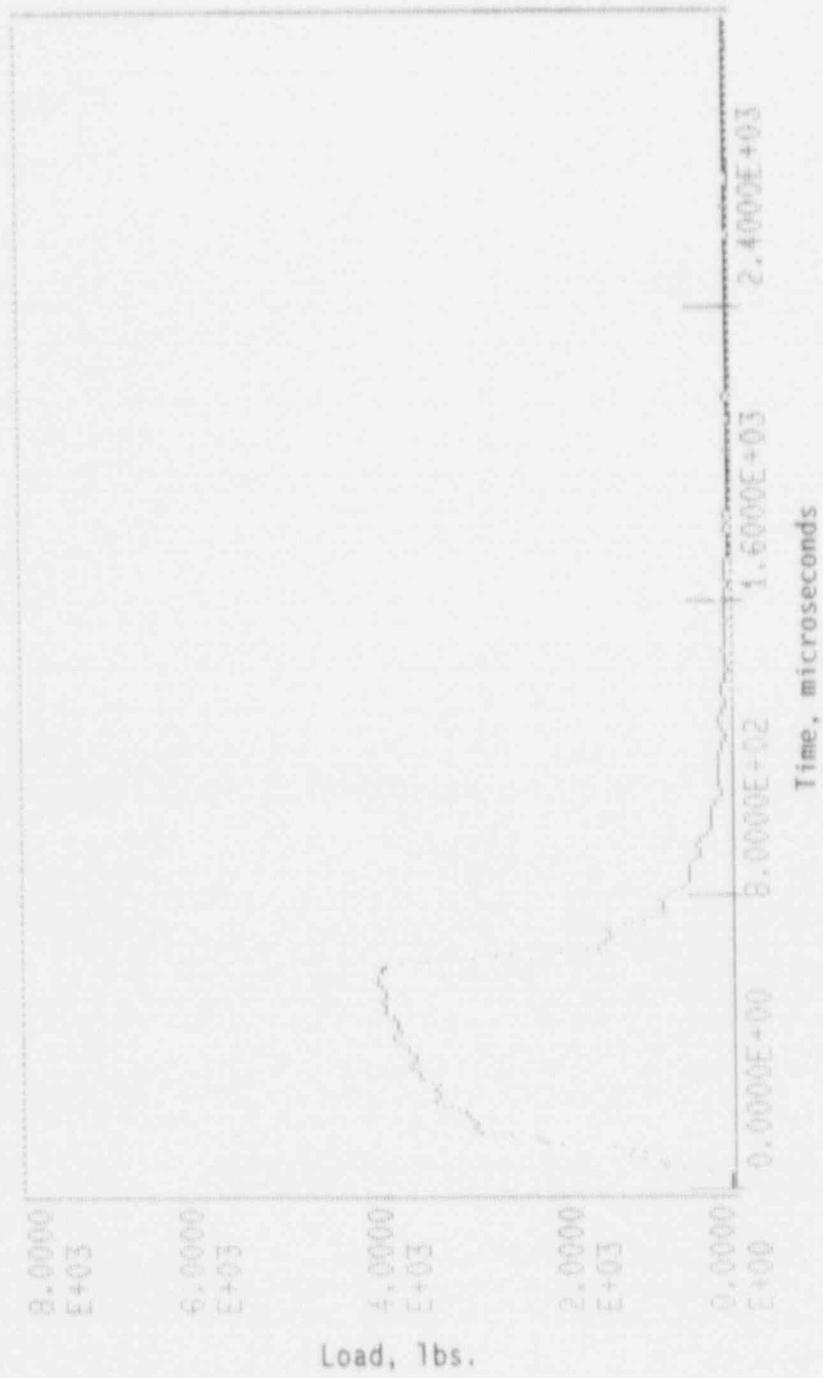
A-41

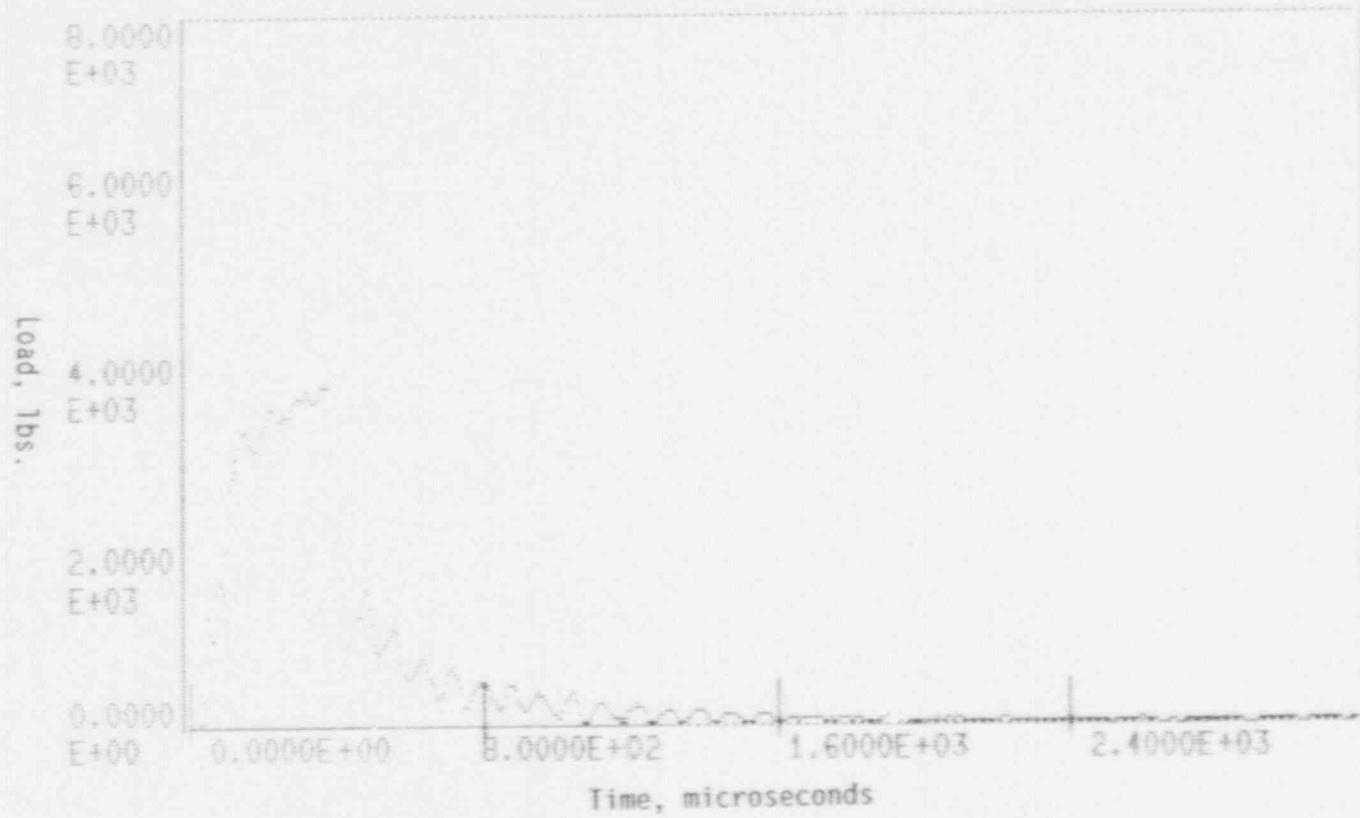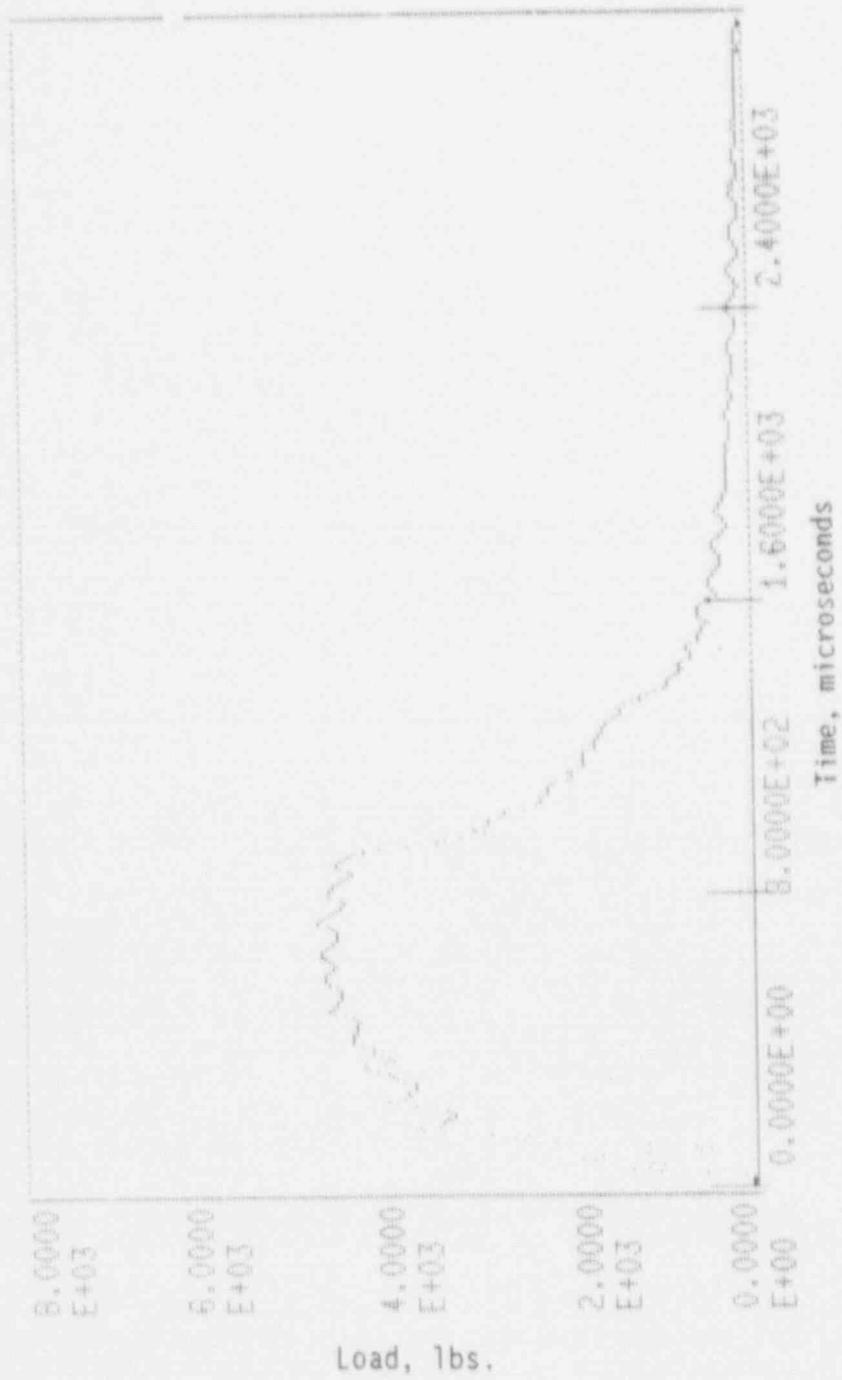Figure A-42. Load-time record for Specimen 61A
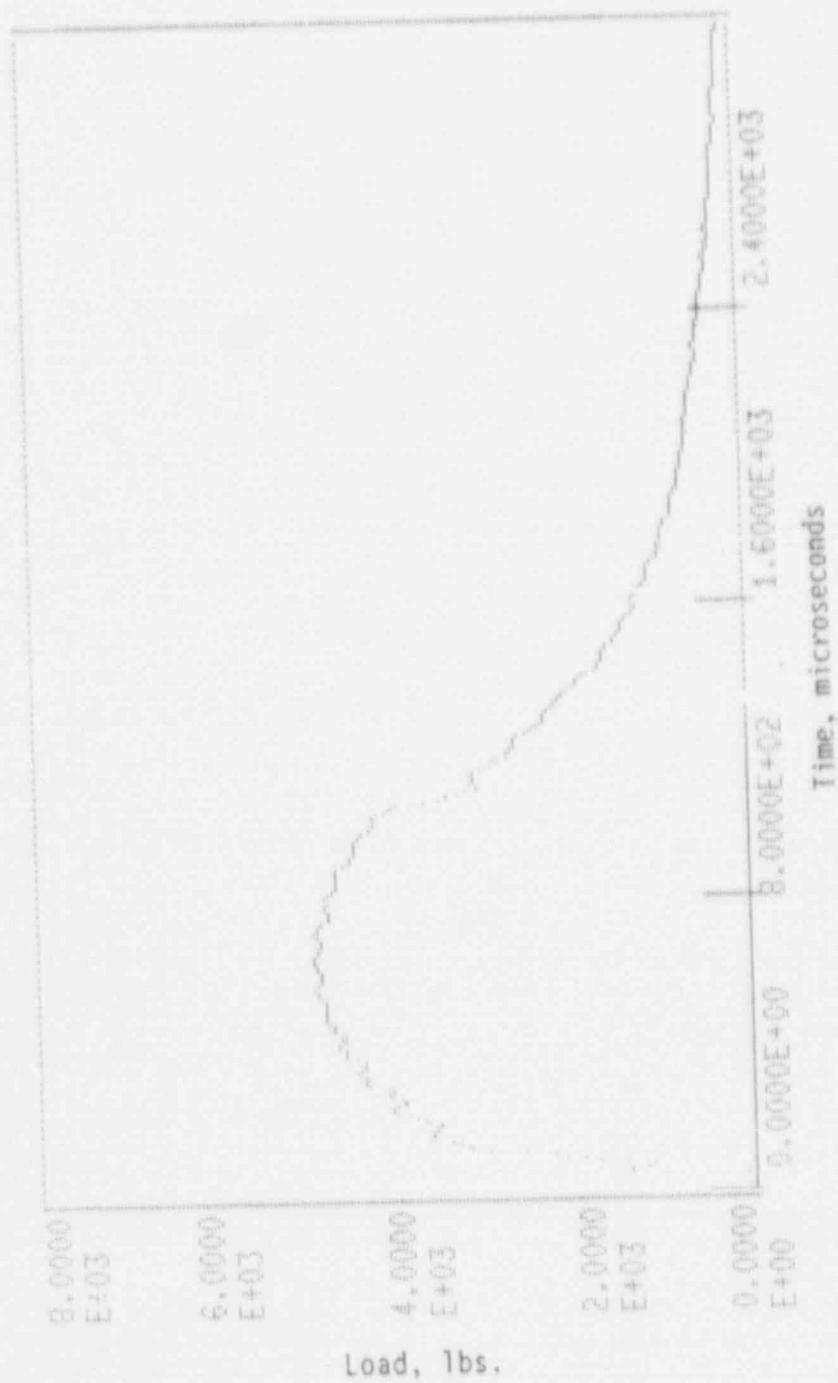
Figure A-43. Load-time record for Specimen 61D
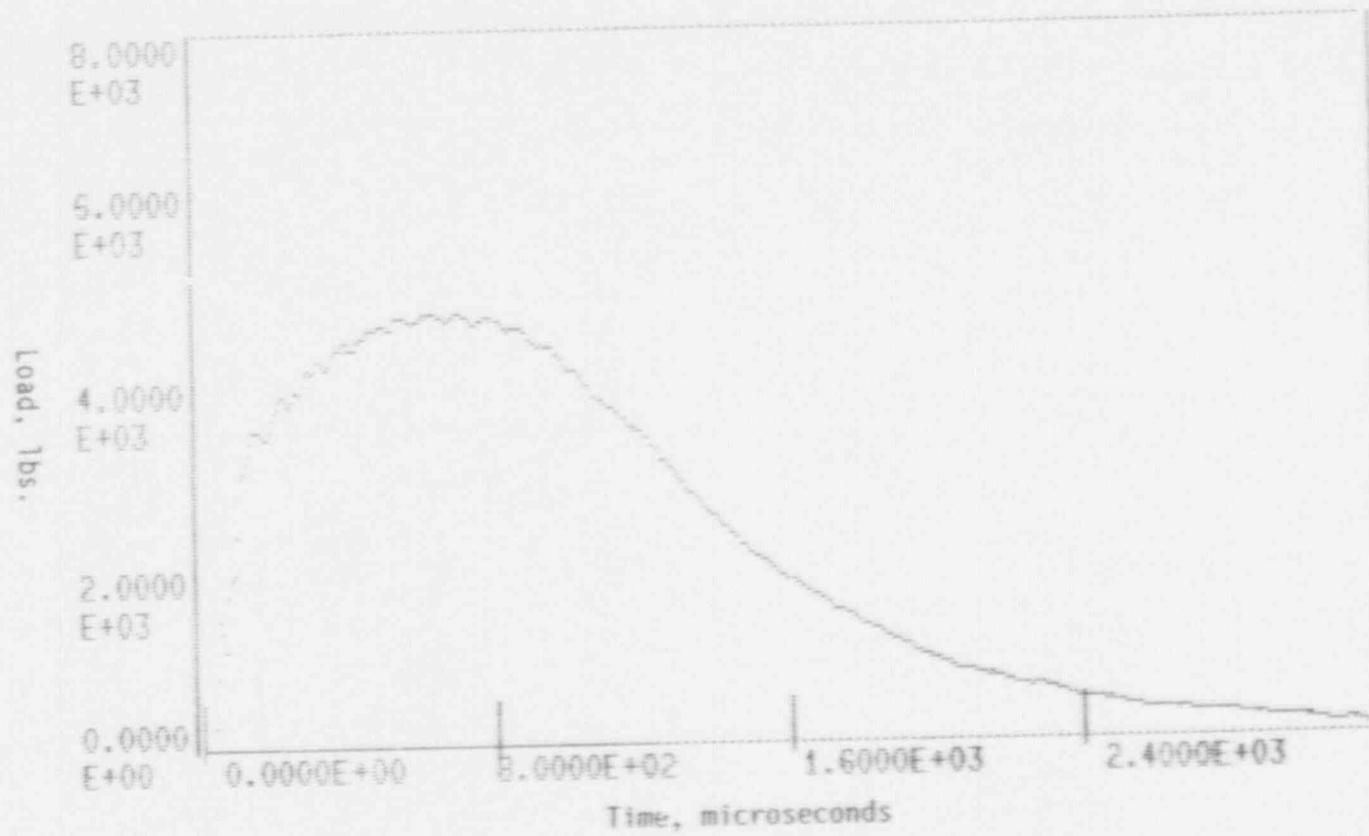
Figure A-44. Load-time record for Specimen 6'T

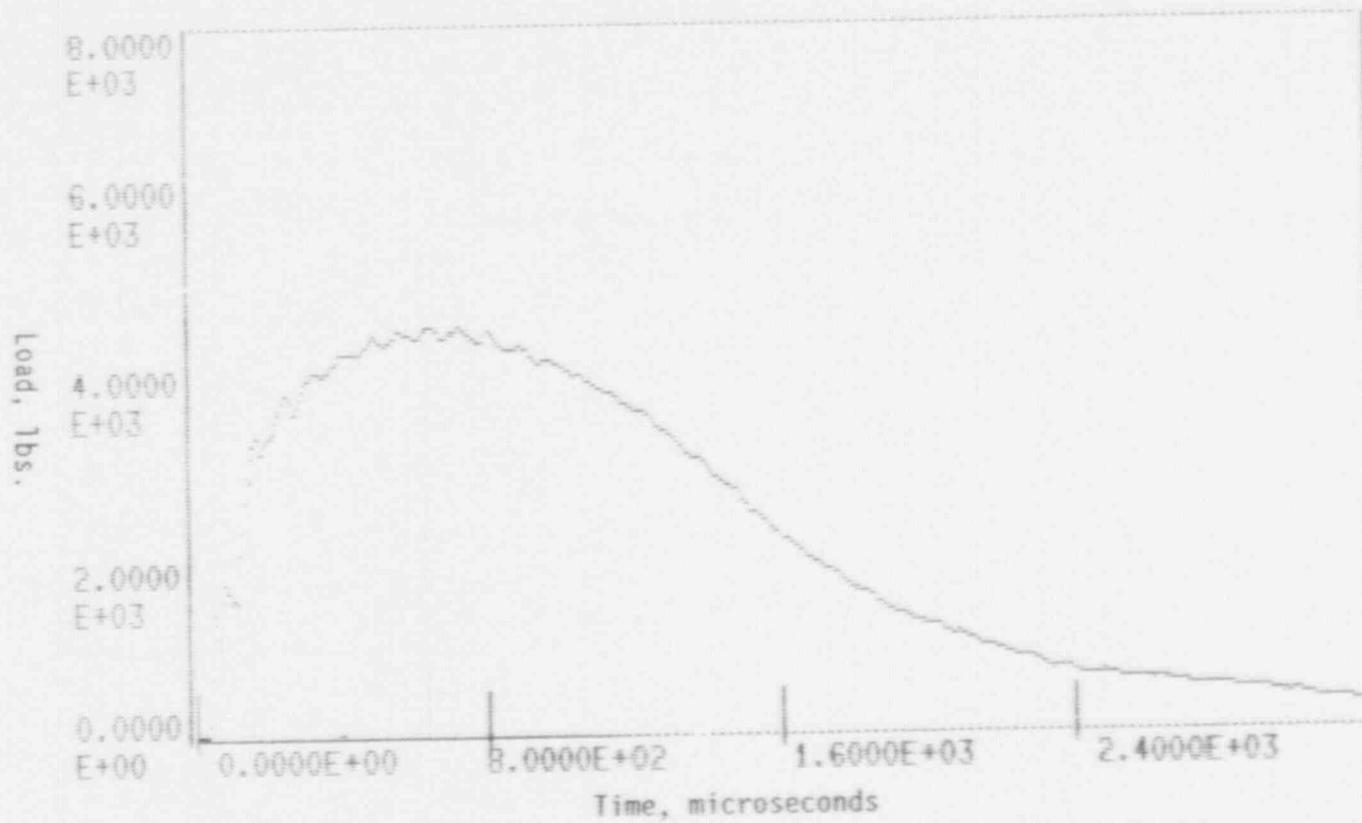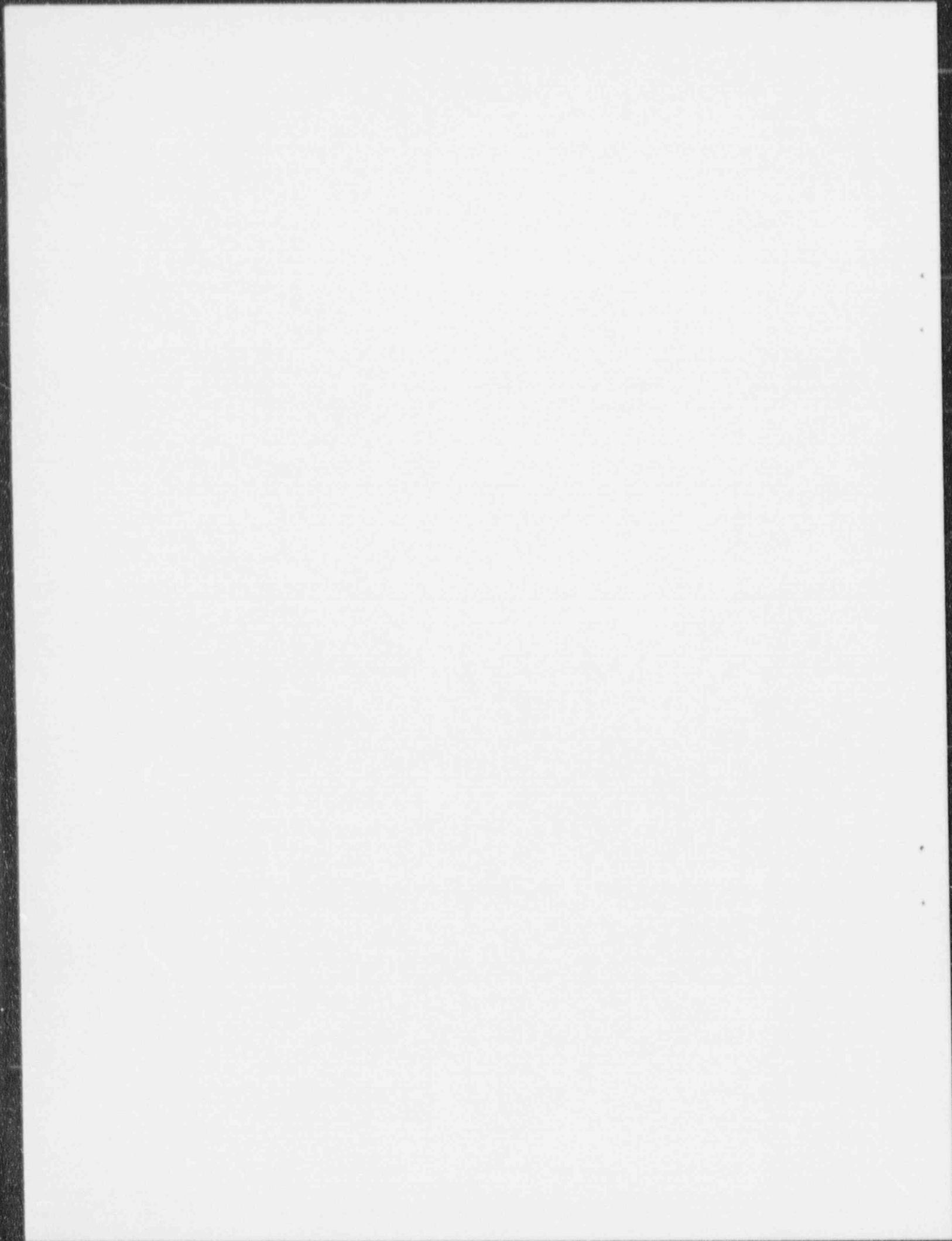Figure A-45. Load-time record for Specimen 61U

Figure A-46. Load-time record for Specimen 61J

APPENDIX B

Equipment Calibration Data

Report & Certificate of Verification

C&M COLLINS CALIBRATION SERVICE

THIS IS TO CERTIFY that the following described testing machine has been calibrated by C&M CALIBRATION and the loading range(s) shown below found to be within a maximum tolerance of 1 %.

MACHINE _INSTRON 3481_ TYPE _LOAD cell TENSION_

LOCATION _WESTINGHOUSE ELECTRIC CORP._ CAPACITY _20000 lbs_

1310 BEULAH RD PITTSBURGH PA SERIAL NO. _166_

DATE OF VERIFICATION _MAY 9 1990_

Method of verification and below recorded data is in accordance with A.S.T.M. E-4 _83A_. The testing device(s) used for this calibration have certifications traceable to the National Institute of Standards Technology.

Machine Readings ☒ Pounds ☐ Newtons ☐ Kilograms   Readings temperature corrected for _73.4_ °F.

MACHINE READING	CALIBRATION DEVICE READING	MACHINE ERROR lbs	%	C.D NO	MACHINE READING	CALIBRATION DEVICE READING	MACHINE ERROR lbs	%	C.D NO
50	50.15	-.15	.30	1	200	200.29	-.29	.14	2
100	100.30	-.30	.30	1	400	400.43	-.43	.11	2
200	200.11	-.11	.06	1	800	802.96	-2.96	.37	2
300	300.32	-.32	.11	2	1200	1205.2	-5.2	.43	2
400	399.98	+.02	.01	2	1600	1606.1	-6.1	.38	2
500	499.87	+.13	.03	2	2000	2009.2	-9.2	.46	3
100	100.59	-.59	.59	1	500	503.43	-3.43	.69	2
200	201.11	-1.11	.55	1	1000	1006.7	-6.7	.67	2
400	402.70	-2.70	.67	2	2000	2014.7	-14.7	.73	3
600	603.39	-3.39	.56	2	3000	3020.9	-20.9	.70	3
800	804.52	-4.52	.56	2	4000	4027.1	-27.1	.68	3
1000	1006.2	-6.2	.62	2	5000	5033.2	-33.2	.66	3

CALIBRATION APPARATUS - Morehouse Proving Rings, Stainsense load cells & Troemner Dead Weights. Verifications traceable to the National Institute of Standards Technology, in accordance with A.S.T.M. latest specifications.

C.D. CODE	SERIAL NO.	LOADING RANGE CLASS A VALUE	CALIB. DATE	NATIONAL INSTITUTE OF STANDARDS TECHNOLOGY LAB NO.	TEMP. OF C.D.
1	4758	25-300 lbs	5-27-88	66/156007	73
2	4757	240-2K lbs	5-27-88	737/229759	73
3	4755	1200-20K lbs	7-21-89	737.229759 73207/241080	73
4					
5					

MACHINE RANGE	LOADING RANGE
0 - 500 lbs	50 - 500 lbs
0 - 1000 lbs	100 - 1000 lbs
0 - 2000 lbs	200 - 2000 lbs
0 - 5000 lbs	500 - 5000 lbs

Primary Load Indicating Device
INSTRON CHART RECORDER

C & M COLLINS CALIBRATION SERVICE
230 Haymont Dr., Box 149, Gibsonia, PA 15044
(412) 443-7631

By: _R. Burt Collins_
(SERVICE ENGINEER)

Witnessed By: _____

Report & Certificate of Verification

C&M COLLINS CALIBRATION SERVICE

THIS IS TO CERTIFY that the following described testing machine has been calibrated by C&M CALI-
BRATION and the loading range(s) shown below found to be within a maximum tolerance of
_1_ %.

MACHINE __INSTRON 3481__ TYPE _LOAD CELL TENSION_

LOCATION __WESTINGHOUSE ELECTRIC CORP.__ CAPACITY __20000 lbs__

__1310 BUFFALT Rd   PITTSBURGH PA.__ SERIAL NO. __166__

DATE OF VERIFICATION __MAY 9 1990__

Method of verification and below recorded data is in accordance with A.S.T.M. E-4 _33A_. The testing device(s) used for
this calibration have certifications traceable to the National Institute of Standards Technology.

Machine Readings ☒ Pounds ☐ Newtons ☐ Kilograms   Readings temperature corrected for _73.4_ °F.

MACHINE READING	CALIBRATION DEVICE READING	MACHINE ERROR lbs	%	C.D. NO.	MACHINE READING	CALIBRATION DEVICE READING	MACHINE ERROR	%	C.D. NO.
1000	1007.2	-7.2	.72	1					
2000	2015.2	-15.2	76	2					
4000	4029.7	-29.7	74	2					
6000	6036.1	-36.1	60	2					
8000	8049.1	-49.1	.61	2					
10000	10059	-55	.59	2					
2000	1997.2	+2.8	.14	2					
4000	3999.7	+.3	.01	2					
8000	8047.0	-47.0	.59	2					
12000	12047.	-47.	.39	2					
14000	14096.	-96.	.68	2					
16000	16117.	-117.	.73	2					

CALIBRATION APPARATUS - Morehouse Proving Rings, Stainsense load cells & Troemner Dead Weights. Verifications
traceable to the National Institute of Standards Technology, in accordance with A.S.T.M. latest specifications.

O.D. CODE	SERIAL NO.	LOADING RANGE CLASS A VALUE	CALIB. DATE	NATIONAL INSTITUTE OF STANDARDS TECHNOLOGY LAB NO.		TEMP OF C.D.
1	4757	240-2K lbs	5-27-88	737/229759		73
2	4755	1808-20K lbs	3-21-89	737 229759	732.07/241080	73
3						
4						
5						

MACHINE RANGE	LOADING RANGE
0 - 10000 lbs	1000 - 10000 lbs
0 - 20000 lbs	2000 - 16000 lbs

Primary Load Indicating Device
INSTRON CHART RECORDER

C & M COLLINS CALIBRATION SERVICE
230 Haymont Dr., Box 149, Gibsonia, PA 15044
(412) 443-7631

By: _R. Bert Collins_
(SERVICE ENGINEER)

Witnessed By _____

B-3

DATE: 6/1/90

INSTRUMENT BEING CALIBRATED: Tinius-Olsen Mod. 74, Ser. # 123159 Impact machine .

LOCATION: Bldg. 302A low level cell

OPERATOR: L. M. Thomas

CALIBRATION METHOD: NIST standards per ASTM E23-86
 Replace Watertown Standards
RESULTS:

| Series | Average Value | | Expected Energy |
	Dial	Computer	Range
LL22	11.6	11.8	11-15 ft-lb
SRM 2092			
MM17	70.6	71.9	66-78 ft-lb
SRM 2096			

CALIBRATION   ACCEPTABLE   X   NOT ACCEPTABLE

COMMENTS:

APPROVAL: _____   6/1/90
        R. P. Shogan   Date
        Supervisor, Nuclear Services and Component Analysis

National Institute of Standards & Technology

Certificate

Standard Reference Materials 2092 & 2096

Certified Low-Energy and High-Energy Samples for Charpy V-Notch Testing Machines

Standard Reference Materials (SRM's) 2092 and 2096 are intended primarily for the certification of Charpy V-Notch testing machines in accordance with ASTM Standard E-23. Each SRM consists of five 10 x 10 mm bars of 4340 alloy steel. The bars were fabricated from vacuum melted, vacuum arc remelted steel. The bars were cut to finished lengths, machined, and heat treated in SRM "lots" of 1200 bars.

A random sampling of bars was taken from each SRM lot and tested by a number of Charpy V-Notch testing machines at NIST's Boulder, Colorado laboratory. These machines were chosen to represent the wide variety of Charpy V-Notch testing machines available in today's market. The data were statistically evaluated to assure the quality of the lot and a certified energy value for the lot was derived. Below is a table of hardness and energy ranges for SRM 2092 (Low Energy) and SRM 2096 (High Energy).

SRM No.	Hardness Range (HRC)	Energy Range (ft-lbs)	(J)	(m-Kg)
2092	45 ± 1	11-15	15-20	1.5-2.1
2096	32 ± 1	66-78	90-106	9.1-10.8

Note: Each bar is identified by a Series number (two letters followed by two digits) identifying the lot from which the bar came and an identification number (three or four digits) representing the location of the bar within the lot. These numbers are stamped on one end of each bar.

Use:

Prior to testing a Charpy V-Notch testing machine, the machine should be checked to assure compliance with sections 4 and 5 of ASTM Standard E-23. Specimens are to be tested at -40°F(-40°C) and in accordance with the testing procedures of Sections 11.2.1 and 11.2.3.1 of E-23. An accurate machine will produce values within 1.0 ft-lb or 5 percent of the certified energy value, whichever is greater.

Notice to User:

The five specimens in both SRM 2092 and SRM 2096 (a total of ten specimens) must be tested for certification in accordance with ASTM Standard E-23.

Each SRM must consist of samples with identical Series numbers.

Because the source(s) of erroneous energy values at one energy level may not be the same at different energy levels, calibration or correction curves should not be used.

Gaithersburg, MD 20899
March 26, 1990
(Revision of certificates
dated 8-31-89 & 12-20-89)

William P. Reed, Acting Chief
Standard Reference Materials Program

(over)

<u>Certification of Machine:</u>

The National Institute of Standards and Technology (NIST) will issue a report of its findings to your facility upon receipt of the fractured specimens and completed questionnaire. If your machine produces acceptable values, this report will certify the conformance of your machine. If your machine produces values outside the allowable tolerance of the certified energy values, this report will suggest changes in machine design, repair or replacement of machine parts, changes in testing techniques, etc. Fractured specimens and completed questionnaires should be returned to Charpy Program Coordinator, NIST Code 430, 325 Broadway, Boulder CO 80303. A plastic, self-locking bag is provided for the return of broken specimens. Tape broken pieces together as described in the wrapping instructions (last page of the questionnaire).

The certification you will receive is valid for one year from the date of the test.

If your machine is moved or undergoes any major repairs or adjustments, the current certification becomes invalid and the machine must be recertified. Removal of the pendulum, replacement of the anvils or adjustment of the height of drop are examples of such major repairs or adjustments. If a specimen requires over 80 percent of the machine capacity to fracture, the machine should be checked to assure that the pendulum is straight, the anvils or striker have not been damaged and that all bolts are still tight.

<u>General Information:</u>

Shipping charges for the return of broken specimens are the responsibility of the customer. The mailing label which is provided must be used to expedite shipping and, for overseas shipments, clearance by U.S. Customs.

Note to International Customers: Overseas shipments should be sent by airmail so that after the packages have been cleared by U.S. Customs, they will be forwarded directly to NIST-Boulder. If more rapid shipment is necessary, choose an overnight delivery service that will handle U.S. Customs clearance and will deliver directly to NIST-Boulder. Unless you assure delivery, airfreight packages may be returned to the customer by U.S. Customs instead of being forwarded to NIST-Boulder.

This SRM is anticipated to have an indefinite shelf life under normal storage conditions. The SRM is coated with oil, wrapped in a corrosion inhibiting paper, and sealed in a plastic envelope. We recommend that the SRM be retained in this package to protect it from moisture until it is used. The protective oil coating should be wiped from the SRM just prior to testing.

The overall direction and coordination of the technical measurements leading to certification of test specimens and machines, evaluation of test results and issuance of the report on machine conformance are under the direction of D. A. Shepherd, Materials Reliability Division, NIST. Questions concerning any of the above should be directed to D.A. Shepherd, Telephone: (303) 497-3211, Telefax: (303) 497-5030.

Technical and support aspects concerning the preparation, certification of the material and issuance of these Standard Reference Materials were coordinated through the Standard Reference Materials Program by J. Falco and R.L. McKenzie.

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dynatup
General Research Corp.
a GRC International Company

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SANTA BARBARA, CA 93117-3093
TELEPHONE (805) 685-2772
TELEFAX (805) 685-9638

ALUMINUM CHARPY CALIBRATION SPECIMENS

These calibration specimens allow the dynamic calibration of the tup and data system to be checked. To perform a check, compare the average measured maximum load for a group of at least five calibration specimens to the reference maximum load listed below. Variations from the reference value that exceed the listed tolerance indicate a calibration problem. Note that energy values will vary significantly. These specimens cannot be used for energy calibration. For equipment requiring energy calibration, Watertown Arsenal (AMMRC) specimens must be used.

These standard dimension Charpy specimens (ASTM Method E23) are prepared from aluminum alloy bar. Each specimen is identified with a scribed number. The material is slightly strain rate sensitive. For this reason, the average maximum load value for a minimum of five specimens will vary with impact velocity.

Maximum Load lb (kn)	Tolerance Percent	Velocity ft/s (m/s)
1845 (8.206)	4.0%	11.0 (3.353)
1846 (8.211)	4.0%	17.0 (5.182)

The instrumentation response time (TR) should be approximately 80 micro-seconds. Smaller TR values will result in higher maximum load values. When using the DYNATUP Model 730-I or 600 series data systems, the test time, filtering and smoothing must be set as follows to achieve the correct TR value.

Model Number	Test Time	Filter No.	Smoothing (2m+1 data pts)
730-I or 830-I			
1024 data pts (Normal)	10 ms	1	m = 4
2048 data pts	10 ms	1	m = 8
600 Series			
All Systems	10 ms	1	m = 8

Note:
While all standard scientific precautions are followed in the production of these specimens, they are not manufactured in accordance with specifications written by the NRC, ASTM, etc. For applications requiring equipment to be calibrated to any particular specification, use of these specimens may not be accepted.

These ___100 ea.___ specimens are from Lot No. ___B 6/90___

Shipping Date: ___6-26-90___

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