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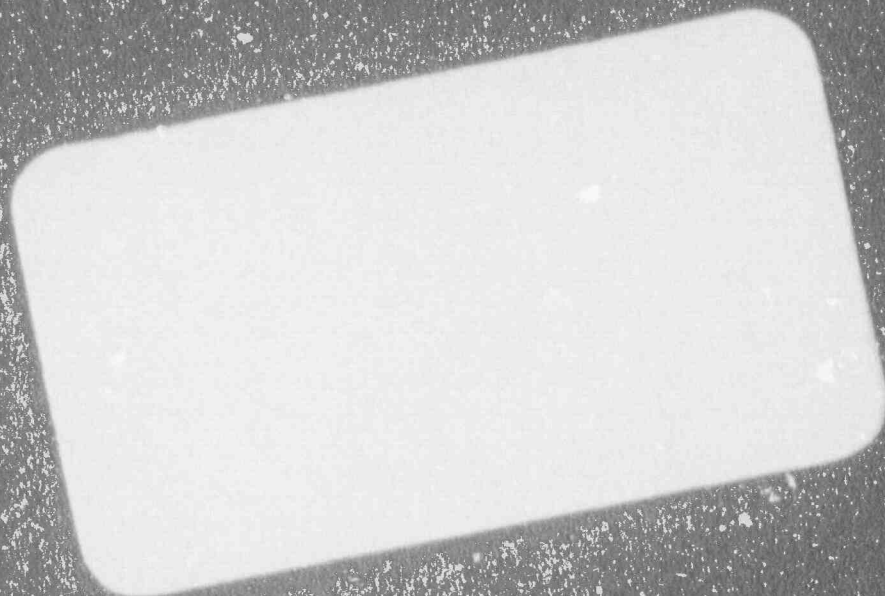


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WCAP-13516

ANALYSIS OF CAPSULE U FROM THE
DUKE POWER COMPANY
MCGUIRE UNIT 2 REACTOR VESSEL
RADIATION SURVEILLANCE PROGRAM

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Prepared by Westinghouse Electric Corporation
for the Duke Power Company

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T. A. Meyer, Manager
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PREFACE

This report has been technically reviewed and verified.

Reviewer:

Sections 1 through 5, 7, 8 and
Appendices A and C

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Section 6

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Appendix B

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TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
1.0	SUMMARY OF RESULTS	1-1
2.0	INTRODUCTION	2-1
3.0	BACKGROUND	3-1
4.0	DESCRIPTION OF PROGRAM	4-1
5.0	TESTING OF SPECIMENS FROM CAPSULE U	5-1
	5.1 Overview	5-1
	5.2 Charpy V-Notch Impact Test Results	5-4
	5.3 Tension Test Results	5-6
	5.4 Compact Tension Tests	5-7
6.0	RADIATION ANALYSIS AND NEUTRON DOSIMETRY	6-1
	6.1 Introduction	6-1
	6.2 Discrete Ordinates Analysis	6-2
	6.3 Neutron Dosimetry	6-7
7.0	SURVEILLANCE CAPSULE REMOVAL SCHEDULE	7-1
8.0	REFERENCES	8-1

APPENDIX A - LOAD-TIME RECORDS FOR CHARPY SPECIMEN TESTS

APPENDIX B - HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

APPENDIX C - UPPER SHELF ENERGY EVALUATION

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
4-1	Heat Treatment of the McGuire Unit 2 Reactor Vessel Surveillance Materials	4-4
4-2	Chemical Composition of the McGuire Unit 2 Reactor Vessel Surveillance Materials	4-5
4-3	Chemical Composition of Four McGuire Unit 2 Charpy Specimens Removed from Surveillance Capsule U	4-6
4-4	Chemistry Results from the NBS Certified Reference Standards	4-7
5-1	Charpy V-Notch Impact Data for the McGuire Unit 2 Intermediate Shell Forging 05 Irradiated at 550°F, Fluence 2.02×10^{19} n/cm ² (E > 1.0 MeV)	5-8
5-2	Charpy V-Notch Impact Data for the McGuire Unit 2 Reactor Vessel Weld Metal and HAZ Metal Irradiated at 550°F, Fluence 2.02×10^{19} n/cm ² (E > 1.0 MeV)	5-9
5-3	Instrumented Charpy Impact Test Results for the McGuire Unit 2 Intermediate Shell Forging 05 Irradiated at 550°F, Fluence 2.02×10^{19} n/cm ² (E > 1.0 MeV)	5-10
5-4	Instrumented Charpy Impact Test Results for the McGuire Unit 2 Weld Metal and Heat-Affected-Zone (HAZ) Metal, Irradiated at 550°F, Fluence 2.02×10^{19} n/cm ² (E > 1.0 MeV)	5-11
5-5	Effect of 550°F Irradiation to 2.02×10^{19} n/cm ² (E > 1.0 MeV) on the Notch Toughness Properties of the McGuire Unit 2 Reactor Vessel Surveillance Materials	5-12

LIST OF TABLES (Continued)

<u>Table</u>	<u>Title</u>	<u>Page</u>
5-6	Comparison of the McGuire Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99 Revision 2 Predictions	5-13
5-7	Tensile Properties for the McGuire Unit 2 Reactor Vessel Surveillance Materials Irradiated at 550°F to 2.02×10^{19} n/cm^2 ($E > 1.0$ MeV)	5-14
6-1	Calculated Fast Neutron Exposure Parameters at the Surveillance Capsule Center	6-15
6-2	Calculated Fast Neutron Exposure Parameters at the Pressure Vessel Clad/Base Metal Interface	6-16
6-3	Relative Radial Distributions of Neutron Flux ($E > 1.0$ MeV) within the Pressure Vessel Wall	6-18
6-4	Relative Radial Distributions of Neutron Flux ($E > 0.1$ MeV) within the Pressure Vessel Wall	6-19
6-5	Relative Radial Distributions of Iron Displacement Rate (dpa) within the Pressure Vessel Wall	6-20
6-6	Nuclear Parameters for Neutron Flux Monitors	6-21
6-7	Monthly Thermal Generation During the First Seven Fuel Cycles of the W. B. McGuire Unit 2 Reactor	6-22

LIST OF TABLES (Continued)

<u>Table</u>	<u>Title</u>	<u>Page</u>
6-8	Measured Sensor Activities and Reactions Rates Surveillance Capsule U	6-23
6-9	Measured Sensor Activities and Reactions Rates Surveillance Capsule X	6-25
6-10	Measured Sensor Activities and Reactions Rates Surveillance Capsule V	6-27
6-11	Summary of Neutron Dosimetry Results Surveillance Capsule U	6-29
6-12	Summary of Neutron Dosimetry Results Surveillance Capsule X	6-30
6-13	Summary of Neutron Dosimetry Results Surveillance Capsule V	6-31
6-14	Comparison of Measured and FERRET Calculated Reaction Rates at the Surveillance Capsule Center Surveillance Capsule U	6-32
6-15	Comparison of Measured and FERRET Calculated Reaction Rates at the Surveillance Capsule Center Surveillance Capsule X	6-33
6-16	Comparison of Measured and FERRET Calculated Reaction Rates at the Surveillance Capsule Center Surveillance Capsule V	6-34
6-17	Adjusted Neutron Energy Spectrum at the Center of Surveillance Capsule U	6-35
6-18	Adjusted Neutron Energy Spectrum at the Center of Surveillance Capsule X	6-36

LIST OF TABLES (Continued)

<u>Table</u>	<u>Title</u>	<u>Page</u>
6-19	Adjusted Neutron Energy Spectrum at the Center of Surveillance Capsule V	6-37
6-20	Comparison of Calculated and Measured Exposure Levels for W. B. McGuire Unit 2 Surveillance Capsules	6-38
6-21	Neutron Exposure Projections at Key Locations on the Pressure Vessel Clad/Base Metal Interface	6-40
6-22	Projected Fast Neutron Exposure Values	6-41
6-23	Updated Lead Factors for McGuire Unit 2 Surveillance Capsules	6-42

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
4-1	Arrangement of Surveillance Capsules in the McGuire Unit 2 Reactor Vessel	4-9
4-2	Capsule U Diagram Showing Location of Specimens, Thermal Monitors and Dosimeters	4-10
5-1	Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)	5-15
5-2	Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)	5-16
5-3	Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal	5-17
5-4	Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal	5-18
5-5	Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)	5-19
5-6	Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)	5-20

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5-7	Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal	5-21
5-8	Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal	5-22
5-9	Tensile Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)	5-23
5-10	Tensile Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)	5-24
5-11	Tensile Properties for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal	5-25
5-12	Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)	5-26
5-13	Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)	5-27
5-14	Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Surveillance Weld Metal	5-28
5-15	Engineering Stress-Strain Curves for Forging 05 Tensile Specimens DL1 and DL2 (Tangential Orientation)	5-29

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5-16	Engineering Stress-Strain Curve for Forging 05 Tensile Specimens DL3 (Tangential Orientation)	5-30
5-17	Engineering Stress-Strain Curves for Forging 05 Tensile Specimens DT1 and DT2 (Axial Orientation)	5-31
5-18	Engineering Stress-Strain Curve for Intermediate Shell Forging 05 Tensile Specimen DT3 (Axial Orientation)	5-32
5-19	Engineering Stress-Strain Curves for Weld Metal Tensile Specimens DW1 and DW2	5-33
5-20	Engineering Stress-Strain Curve for Weld Metal Tensile Specimen DW3	5-34
6-1	Plan View of a Dual Reactor Vessel Surveillance Capsule	6-13
6-2	Axial Distribution of Neutron Fluence ($E > 1.0$ MeV) Along the 45 Degree Azimuth	6-14

SECTION 1.0 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance capsule U, the third capsule to be removed from the Duke Power Company McGuire Unit 2 reactor pressure vessel, led to the following conclusions:

- o The capsule received an average fast neutron fluence ($E > 1.0$ MeV) of 2.02×10^{19} n/cm² after 6.05 EFPY of plant operation.
- o Irradiation of the reactor vessel intermediate shell forging 05 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major rolling direction (tangential orientation), to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) resulted in a 30 ft-lb transition temperature increase of 90°F and a 50 ft-lb transition temperature increase of 105°F. This results in a 30 ft-lb transition temperature of 15°F and a 50 ft-lb transition temperature of 45°F for tangentially oriented specimens.
- o Irradiation of the reactor vessel intermediate shell forging 05 Charpy specimens, oriented with the longitudinal axis of the specimen normal to the major rolling direction (axial orientation), to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) resulted in a 30 ft-lb transition temperature increase of 85°F and a 50 ft-lb transition temperature increase of 90°F. This results in a 30 ft-lb transition temperature of 60°F and a 50 ft-lb transition temperature of 115°F for axially oriented specimens.
- o The weld metal Charpy specimens irradiated to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) resulted in a 30 ft-lb transition temperature increase of 20°F and a 50 ft-lb transition temperature increase of 35°F. This results in a 30 ft-lb transition temperature of -30°F and a 50 ft-lb transition temperature of 10°F for the weld metal.

- o Irradiation of the reactor vessel weld Heat-Affected-Zone (HAZ) metal Charpy specimens to $2.02 \times 10^{19} \text{ n/cm}^2$ ($E > 1.0 \text{ MeV}$) resulted in a 30 ft-lb transition temperature increase of 85°F and a 50 ft-lb transition temperature increase of 85°F. This results in a 30 ft-lb transition temperature of -5°F and a 50 ft-lb transition temperature of 30°F for the weld HAZ metal.
- o Irradiation of intermediate shell forging 05 (tangential orientation) to $2.02 \times 10^{19} \text{ n/cm}^2$ ($E > 1.0 \text{ MeV}$) resulted in an average upper shelf energy decrease of 34 ft-lbs, resulting in an upper shelf energy of 122 ft-lbs.
- o Irradiation of intermediate shell forging 05 (axial orientation) to $2.02 \times 10^{19} \text{ n/cm}^2$ ($E > 1.0 \text{ MeV}$) resulted in an average upper shelf energy decrease of 10 ft-lb, resulting in an upper shelf energy of 84 ft-lbs.
- o The average upper shelf energy of the weld metal decreased 4 ft-lb after irradiation to $2.02 \times 10^{19} \text{ n/cm}^2$ ($E > 1.0 \text{ MeV}$). This results in an upper shelf energy of 129 ft-lb for the weld metal.
- o The average upper shelf energy of the weld HAZ metal decreased 11 ft-lb after irradiation to $2.02 \times 10^{19} \text{ n/cm}^2$ ($E > 1.0 \text{ MeV}$). This results in an upper shelf energy of 93 ft-lb for the weld HAZ metal.
- o The surveillance capsule U test results indicate that the intermediate shell forging 05 and the surveillance weld material 30 ft-lb transition temperature shift is less than the Regulatory Guide 1.99 Revision 2 predictions.
- o The surveillance capsule materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are expected to maintain an upper shelf energy of no less than 50 ft-lb throughout the life (32 EFPY) of the vessel as required by 10CFR50, Appendix G.

- o The calculated end-of-life (32 EFPY) maximum neutron fluence ($E > 1.0$ MeV) for the McGuire Unit 2 reactor vessel is as follows:

Vessel inner radius * - 2.04×10^{19} n/cm²

Vessel 1/4 thickness - 1.09×10^{19} n/cm²

Vessel 3/4 thickness - 2.20×10^{18} n/cm²

* Clad/base metal interface

SECTION 2.0 INTRODUCTION

This report presents the results of the examination of capsule U, the third capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Duke Power Company McGuire Unit 2 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the McGuire Unit 2 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials is presented in WCAP-9489, entitled "Duke Power Company William B. McGuire Unit No. 2 Reactor Vessel Radiation Surveillance Program" by Koyama and Davidson^[1]. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels". Westinghouse Power Systems personnel were contracted to aid in the preparation of procedures for removing capsule U from the reactor and its shipment to the Westinghouse Science and Technology Center 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 surveillance capsule U removed from the Duke Power Company McGuire Unit 2 reactor vessel and discusses the analysis of the data.

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 SA 508 Class 2 (base material of the McGuire Unit 2 reactor pressure vessel intermediate shell forging 05) 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.

A method for performing analyses to guard against fast fracture in reactor pressure vessels has been presented in "Protection Against Nonductile Failure," Appendix G to Section III of the ASME Boiler and Pressure Vessel Code^[4]. The method uses fracture mechanics concepts and is based on the reference nil-ductility temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208)^[5] or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens with the tangential axis oriented normal (axial orientation) to the major working direction of the forging. 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 to 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 surveillance program, such as the McGuire Unit 2 Reactor Vessel Radiation Surveillance Program^[1], in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens 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

Six surveillance capsules for monitoring the effects of neutron exposure on the McGuire Unit 2 reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant start-up. The six capsules were positioned in the reactor vessel between the neutron shielding pads and the vessel wall as shown in Figure 4-1. The vertical center of the capsules is opposite the vertical center of the core.

Capsule U was removed after 6.05 Effective Full Power Years (EFPY) of plant operation. This capsule contained Charpy V-notch, tensile, and 1/2 T compact tension (CT) specimens (Figure 4-2) from the submerged arc weld metal fabricated with the same weld wire and flux as used in the reactor vessel core region girth weld, and Charpy V-notch, tensile, CT, and bend bar specimens (Figure 4-2) from the intermediate shell forging 05. Capsule Y also contained Charpy V-notch specimens from weld Heat-Affected-Zone (HAZ) material. All heat-affected zone specimens were obtained from within the HAZ of forging 05.

Test specimens obtained from the intermediate shell forging (after thermal heat treatment and forming of the forging) were taken from at least one forging thickness from the quenched ends of the forging. Test specimens were machined from the 1/4 thickness location of the forging after performing a simulated postweld stress-relieving treatment on the material and also from weld and heat-affected-zone metal of a stress-relieved weldment joining the lower and intermediate shell forgings. All heat-affected-zone specimens were obtained from the weld heat-affected-zone of the intermediate shell forging.

Charpy V-notch specimens from the intermediate shell forging were machined in both tangential (longitudinal axis of the specimens parallel to the major working direction) and axial (longitudinal axis normal to the major working direction) orientations. The Charpy V-notch specimens from the weld and weld heat-affected-zone metal were machined perpendicular to the weld direction with the notch oriented in the direction of the weld.

Tensile specimens from the intermediate shell forging were machined with the tangential axis of the specimen both parallel and normal to the major working direction. Weld specimens were oriented normal to the weld direction.

Bend bar specimens were machined from the intermediate shell forging with the longitudinal axis of the specimen oriented parallel to the working direction of the forging such that the simulated crack would propagate normal to the working direction of the forging. All bend bar specimens were fatigue precracked according to ASTM E-399.

Compact tension test specimens from the intermediate shell forging were machined in both axial and tangential orientations. This was done to obtain toughness data both normal and parallel to the major working direction of the forging. Compact tension test specimens from the weld were machined normal to the weld direction with the notch oriented in the direction of the weld. All specimens were fatigue precracked according to ASTM E-399.

The chemical composition and heat treatment of the surveillance material is presented in Tables 4-1 and 4-2. The chemical analysis and heat treatment reported in Tables 4-1 and 4-2 were obtained from unirradiated material used in the surveillance program^[1]. In addition, a chemical analysis performed on four irradiated Charpy specimens from the weld and base metal forging 05 is reported in Table 4-3. The chemistry results from the NBS certified reference standards are given in Table 4-4.

Capsule U contained dosimeter wires of pure iron, copper, nickel, and aluminum-cobalt (cadmium-shielded and unshielded). In addition, cadmium-shielded dosimeters of Neptunium (Np^{237}) and Uranium (U^{238}) were contained in the capsule.

Thermal monitors made from two low-melting eutectic alloys and sealed in Pyrex tubes were included in the capsule and were located as shown in Figure 4-2.

The two eutectic alloys and their melting points are:

2.5% Ag, 97.5% Pb

Melting Point 579°F (304°C)

1.75% Ag, 0.75% Sn, 97.5% Pb

Melting Point 590°F (310°C)

The arrangement of the various mechanical test specimens, dosimeters and thermal monitors contained in Capsule U are shown in Figure 4-2.

TABLE 4-1

HEAT TREATMENT OF THE MCGUIRE UNIT 2
REACTOR VESSEL SURVEILLANCE MATERIALS^[1]

<u>Material</u>	<u>Temperature (°F)</u>	<u>Time (hr)</u>	<u>Coolant</u>
Intermediate Shell	1688-1697	3.5	Water quenched
Forging 05 (HT. 526840)	1229-1238	7.5	Air cooled
	1140 ± 25	22.0	Furnace cooled
Weld Metal	1140 ± 25	15.0	Furnace cooled

TABLE 4-2

CHEMICAL COMPOSITION OF
THE MCGUIRE UNIT 2 REACTOR VESSEL
SURVEILLANCE MATERIALS^[1]

<u>Element</u>	Forging 05	Weld Metal[a]
	(HT. 526840)	
	<u>(Wt. %)</u>	<u>(Wt. %)</u>
C	0.18	0.055
S	0.017	0.015
N	0.004	0.011
Co	0.019	0.007
Cu	0.16	0.031
Si	0.23	0.29
Mo	0.58	0.55
Ni	0.79	0.73
Mn	0.69	1.81
Cr	0.43	0.030
V	< 0.002	< 0.002
P	0.012	0.016
Sn	0.008	0.002
Ti	< 0.001	0.004
Pb	0.001	0.003
W	< 0.002	< 0.002
Zr	< 0.002	< 0.002
As	0.018	0.015
Al	0.016	0.015
B	< 0.003	< 0.003
Sb	< 0.002	< 0.002

- (a) Surveillance weld specimens were made of the same weld wire and flux as the girth seam weldment between forgings 04 and 05 (Weld Wire Heat No. 895075 and Grau L.O. Flux Lot No. P46)

TABLE 4-3
CHEMICAL COMPOSITION OF FOUR MCGUIRE UNIT 2 CHARPY SPECIMENS
REMOVED FROM SURVEILLANCE CAPSULE U

<u>Metals</u>	<u>Concentration in Weight Percent</u>			
	<u>DW-3 Weld</u>	<u>DW-11 Weld</u>	<u>DW-15 Weld</u>	<u>DL-8 Base Metal</u>
Fe	MATRIX	MATRIX	MATRIX	MATRIX
Co	0.014	0.014	0.014	0.014
Cr	0.045	0.031	0.051	0.410
Cu	0.039	0.036	0.045	0.151
Mn	1.875	1.963	1.882	0.730
Mo	0.590	0.597	0.613	0.628
Ni	0.765	0.747	0.776	0.820
P	0.014	0.015	0.014	0.014
Ti	0.002	0.005	0.005	<0.001
V	0.002	0.001	0.001	0.003

Al	0.011	0.012	0.012	0.007
As	0.014	0.014	0.014	0.027
B	0.004	0.004	0.002	0.003
Nb	0.011	0.002	0.002	0.001
Sn	<0.001	0.003	0.005	0.008
W	<0.001	0.001	0.001	0.004
Zr	<0.001	0.001	0.001	<0.001

C	0.063	0.053	0.061	0.167
S	0.007	0.005	0.005	0.012
Si	0.147	0.148	0.146	0.129

<u>Analyses</u>	<u>Method of Analysis</u>
Metals	ICPS, Inductively Coupled Plasma Spectrometry
Carbon	EC-12, LECO Carbon Analyzer
Sulfur	Combustion/titration
Silicon	Gravimetric

TABLE 4-4
CHEMISTRY RESULTS FROM THE NBS CERTIFIED REFERENCE STANDARDS

MATERIAL ID	Low Alloy Steel: NIST Control Standard			
	Concentration in Weight Percent			
	NBS 361		NBS 362	
Metals	Certified	Measured	Certified	Measured
Fe	95.6000	(Matrix)	95.3000	(Matrix)
Co	0.0320	0.031	0.3000	0.305
Cr	0.6940	0.678	0.3000	0.293
Cu	0.0420	0.043	0.5000	0.5
Mn	0.6600	0.668	1.0400	1.053
Mo	0.1900	0.195	0.0680	0.051
Ni	2.0000	2.034	0.5900	0.579
P	0.0140	0.018	0.0410	0.04
Ti	0.0200	0.013	0.0840	0.019
V	0.0110	0.011	0.0400	0.017
Al	0.0210	0.021	0.0950	0.081
As	0.0170	0.033	0.0920	0.134
B	0.0003	0.001	0.0025	0.004
Nb	0.0220	0.027	0.2900	0.022
Sn	0.0100	0.005	0.0160	0.016
W	0.0170	0.018	0.2000	0.180
Zr	0.0090	0.007	0.1900	0.196
C	0.3830	0.382	0.1600	0.158
S	0.0140	-----	0.0360	0.037
Si	0.2220	0.194	0.3900	0.407

=====

TABLE 4-4 continued
CHEMISTRY RESULTS FROM THE NBS CERTIFIED REFERENCE STANDARDS

MATERIAL ID	Low Alloy Steel: NIST Control Standard			
	<u>Concentration in Weight Percent</u>			
	NBS 363		NBS 364	
<u>Metals</u>	<u>Certified</u>	<u>Measured</u>	<u>Certified</u>	<u>Measured</u>
Fe	94.4000	(Matrix)	96.7000	(Matrix)
Co	0.0480	0.045	0.1500	0.154
Cr	1.3100	1.243	0.0630	0.066
Cu	0.1000	0.098	0.2490	0.249
Mn	1.5000	1.497	0.2550	0.251
Mo	0.0280	0.027	0.4900	0.494
Ni	0.3000	0.302	0.1440	0.139
P	0.0290	0.034	0.0100	0.010
Ti	0.0500	0.033	0.2400	0.211
V	0.3100	0.299	0.1050	0.105
Al	0.2400	0.242	0.0080	0.014
As	0.0100	0.018	0.0520	0.067
B	0.0008	0.002	0.0106	0.013
Nb	0.0490	0.059	0.1570	0.113
Sn	0.1040	0.106	0.0080	0.004
W	0.0460	<0.01	0.1000	0.047
Zr	0.0490	0.032	0.0680	0.062
C	0.6200	-----	0.8700	-----
S	0.0068	-----	0.0250	0.024
Si	0.7400	-----	0.0650	-----

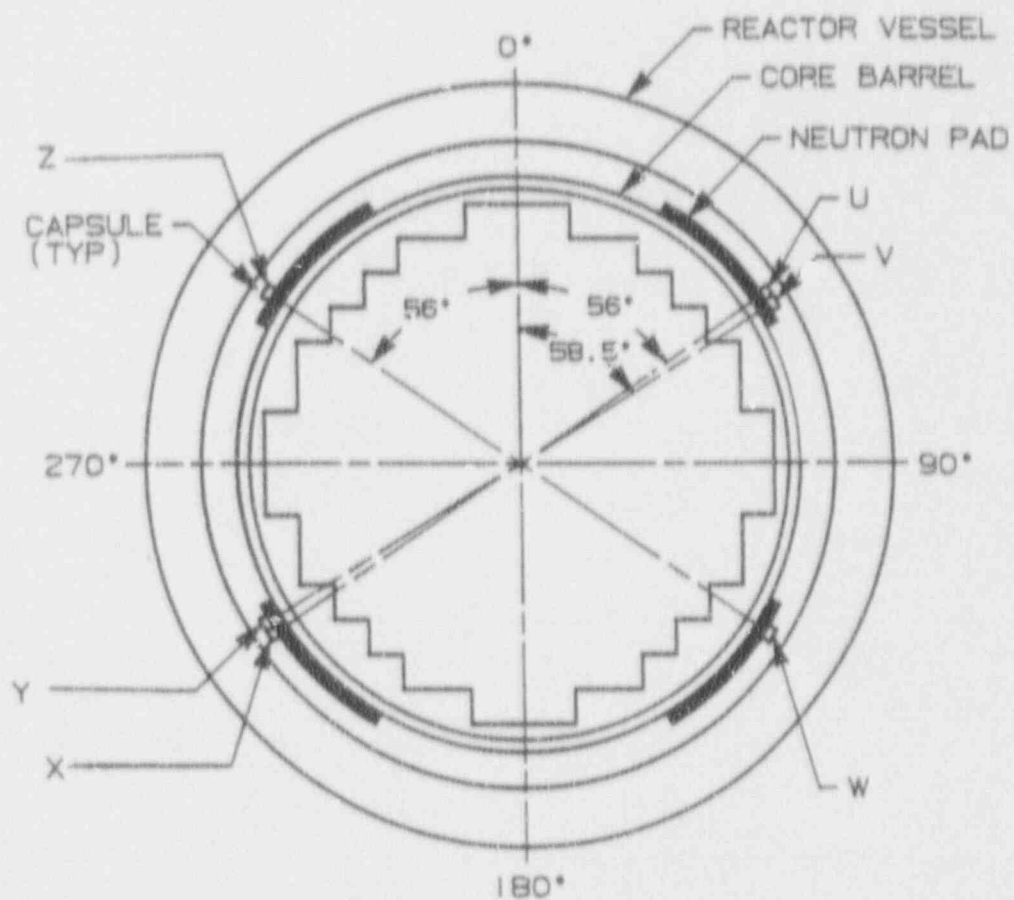
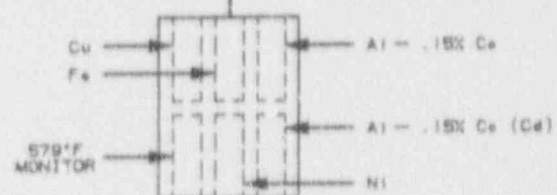
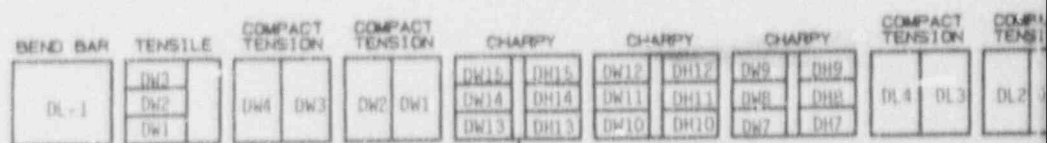


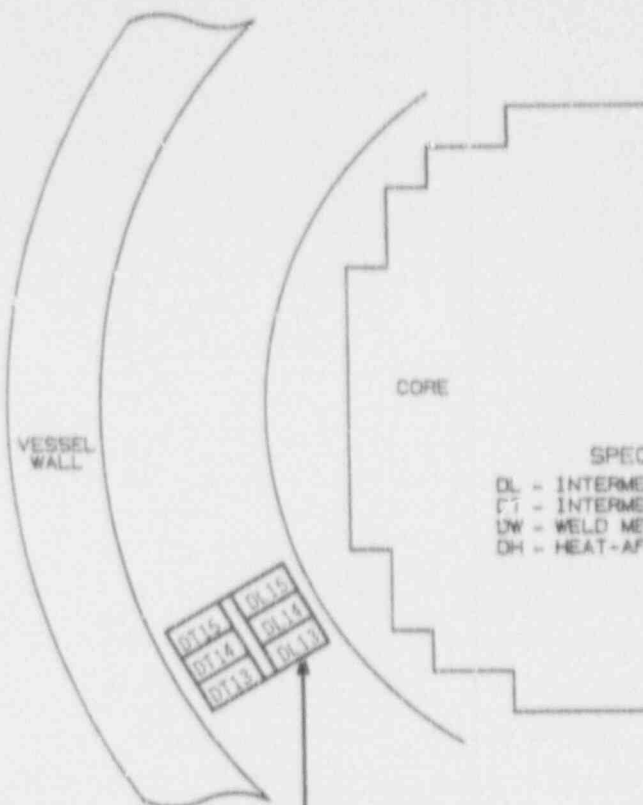
Figure 4-1. Arrangement of Surveillance Capsules in the McGuire Unit 2 Reactor Vessel



TO TOP OF VESSEL

SI APERTURE CARD

Also Available On
Aperture Card



SPECIMEN NUMBERING CODE:
DL - INTERMEDIATE SHELL FORGING OS (TANGENTIAL)
FI - INTERMEDIATE SHELL FORGING OS (AXIAL)
DW - WELD METAL
DH - HEAT-AFFECTED-ZONE MATERIAL

CAPSULE V

Np 237

u238

DOSIMETER

BLOCK

360

IN	CHARPY		CHARPY		DISTANCE BLOCK	TENSILE	CHARPY		CHARPY		CHARPY		CHARPY		CHARPY		COMPACT TENSION		COMPACT TENSION		TENSILE										
	DW6	DH6	DW3	DH3			DT15	DL15	DT12	DL12	DT9	DL9	DT6	DL6	DT3	DL3	DT5	DL5	DT2	DL2		DT4	DT3	DT2	DT1	DT3	DL3	DT2	DL2	DT1	DL1
1	DW6	DH6	DW3	DH3	360	DL3	DT15	DL15	DT12	DL12	DT9	DL9	DT6	DL6	DT3	DL3	DT5	DL5	DT2	DL2	DT4	DT3	DT2	DT1	DT3	DL3	DT2	DL2	DT1	DL1	
	DW5	DH5	DW2	DH2		DL2	DT14	DL14	DT11	DL11	DT8	DL8	DT5	DL5	DT2	DL2															
	DW4	DH4	DW1	DH1		DL1	DT13	DL13	DT10	DL10	DT7	DL7	DT4	DL4	DT1	DL1															



CENTER REGION OF VESSEL

TO BOTTOM OF VESSEL

Figure 4-2 Capsule U Diagram Showing
Location of Specimens,
Thermal Monitors and
Dosimeters

SECTION 5.0

TESTING OF SPECIMENS FROM CAPSULE U

5.1 Overview

The post-irradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Science and Technology Center hot cell with consultation by Westinghouse Power Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H^[2], ASTM Specification E185-82^[6], and Westinghouse Remote Metallographic Facility (RMF) Procedure RMF 8402, Revision 2 as modified by RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the hot cell laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-9488^[1]. No discrepancies were found.

Examination of the two low-melting point 579°F (304°C) and 590°F (310°C) eutectic alloys indicated no melting of either type of thermal monitor. Based on this examination, the maximum temperature to which the test specimens were exposed was less than 579°F (304°C).

The Charpy impact tests were performed per ASTM Specification E23-88^[7] and RMF Procedure 8103, Revision 1 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy machine is instrumented with a GRC 830I instrumentation system, feeding information into an IBM XT computer. 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 (Appendix A), 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 approximately 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) was calculated from the three-point bend formula having the following expression:

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

where L = distance between the specimen supports in the impact testing machine; B = the width of the specimen measured parallel to the notch; W = height of the specimen, measured perpendicularly to the notch; a = notch depth. 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 $\phi = 45^\circ$ and $\rho = 0.010"$, Equation 1 is valid with $C = 1.21$. Therefore (for $L = 4W$),

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

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

$$\sigma_Y = 33.3 \times P_{GY} \quad (3)$$

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 A370-89^[8]. 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 Model 1115, split-console test machine, per ASTM Specification E8-89b^[9] and E21-79 (1988)^[10], 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 axially 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^[12].

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 obtain desired specimen temperatures. Experiments indicated that this method is accurate to $\pm 2^\circ\text{F}$.

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.

5.2 Charpy V-Notch Impact Test Results

The results of the Charpy V-notch impact tests performed on the various materials contained in Capsule U, which was irradiated to 2.02×10^{19} n/cm² ($E > 1.0$ MeV), are presented in Tables 5-1 through 5-4 and are compared with unirradiated results^[1] as shown in Figures 5-1 through 5-4. The transition temperature increases and upper shelf energy decreases for the capsule U materials are summarized in Table 5-5.

Irradiation of the reactor vessel intermediate shell forging 05 Charpy specimens oriented with the longitudinal axis of the specimen parallel to the major rolling direction of the plate (tangential orientation) to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) at 550°F (Figure 5-1) resulted in a 30 ft-lb transition temperature increase of 90°F and in a 50 ft-lb transition temperature increase of 105°F. This resulted in the 30 ft-lb transition temperature of 15°F and a 50 ft-lb transition temperature of 45°F (tangential orientation).

The average Upper Shelf Energy (USE) of the intermediate shell forging 05 Charpy specimens (tangential orientation) resulted in a energy decrease of 34 ft-lb after irradiation to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) at 550°F. This results in an average USE of 122 ft-lb (Figure 5-1).

Irradiation of the reactor vessel intermediate shell forging 05 Charpy specimens oriented with the longitudinal axis of the specimen normal to the major rolling direction of the plate (axial orientation) to 2.02×10^{19} n/cm² ($E > 1.0$ MeV) at 550°F (Figure 5-2) resulted in a 30 ft-lb transition temperature increase of 85°F and in a 50 ft-lb transition temperature increase of 90°F. This resulted in the 30 ft-lb transition temperature of 60°F and a 50 ft-lb transition temperature of 115°F (axial orientation).

The average USE of the intermediate shell forging 05 Charpy specimens (axial orientation) resulted in an energy decrease of 10 ft-lb after irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F. This resulted in an average USE of 84 ft-lb (Figure 5-2).

Irradiation of the reactor vessel core region weld metal Charpy specimens to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F (Figure 5-3) resulted in a 20°F increase in 30 ft-lb transition temperature and a 50 ft-lb transition temperature increase of 35°F. This resulted in a 30 ft-lb transition temperature of -30°F and the 50 ft-lb transition temperature of 10°F.

The average USE of the reactor vessel core region weld metal resulted in an energy decrease of 4 ft-lb after irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F. This resulted in an average USE of 129 ft-lb (Figure 5-3).

Irradiation of the reactor vessel weld Heat-Affected-Zone (HAZ) metal specimens to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F (Figure 5-4) resulted in a 30 ft-lb transition temperature increase of 85°F and a 50 ft-lb transition temperature increase of 85°F. This resulted in a 30 ft-lb transition temperature of -5°F and the 50 ft-lb transition temperature of 30°F.

The average USE of the reactor vessel weld HAZ metal experienced an energy decrease of 11 ft-lb after irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F. This resulted in an average USE of 93 ft-lb (Figure 5-4).

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-5 through 5-8 and show an increasingly ductile or tougher appearance with increasing test temperature.

A comparison of the 30 ft-lb transition temperature increases and upper shelf energy decreases for the various McGuire Unit 2 surveillance materials with predicted values using the methods of NRC Regulatory Guide 1.99, Revision 2^[3] is presented in Table 5-6. This comparison indicates that the transition temperature increase and the upper shelf energy decrease of the intermediate shell forging 05 resulting from irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) are less than the Regulatory Guide predictions. This comparison also indicates that the transition temperature increase and the upper shelf energy decrease of the weld metal resulting from irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) is less than the Regulatory Guide predictions.

The load-time records for the individual instrumented Charpy specimen tests are shown in Appendix A.

5.3 Tension Test Results

The results of the tension tests performed on the various materials contained in capsule U irradiated to 2.02×10^{19} n/cm² (E > 1.0 MeV) are presented in Table 5-7 and are compared with unirradiated results^[1] as shown in Figures 5-9 through 5-11.

The results of the tension tests performed on the intermediate shell forging 05 (tangential orientation) indicated that irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F caused less than a 14 ksi increase in the 0.2 percent offset yield strength and less than a 9 ksi increase in the ultimate tensile strength when compared to unirradiated data^[1] (Figure 5-9).

The results of the tension tests performed on the intermediate shell forging 05 (axial orientation) indicated that irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F caused less than a 19 ksi increase in the 0.2 percent offset yield strength and less than a 15 ksi increase in the ultimate tensile strength when compared to unirradiated data^[1] (Figure 5-10).

The results of the tension tests performed on the reactor vessel core region weld metal indicated that irradiation to 2.02×10^{19} n/cm² (E > 1.0 MeV) at 550°F caused less than a 12 ksi increase in the 0.2 percent offset yield strength and less than a 8 ksi increase in the ultimate tensile strength when compared to unirradiated data^[1] (Figure 5-11).

The fractured tension specimens for the intermediate shell forging 05 material are shown in Figures 5-12 and 5-13, while the fractured specimens for the weld metal are shown in Figure 5-14.

The engineering stress-strain curves for the tension tests are shown in Figures 5-15 through 5-19.

5.4 Compact Tension Tests

Per the surveillance capsule testing program with the Duke Power Company, the 1/2-T compact tension fracture mechanics specimens will not be tested and will be stored at the Westinghouse Science and Technology Center.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE MCGUIRE UNIT 2
 INTERMEDIATE SHELL FORGING 05 IRRADIATED AT 550°F,
 FLUENCE 2.02×10^{19} n/cm² (E > 1.0 MeV)

Sample No.	Temperature (*F)	(*C)	Impact Energy (ft-lb)	(J)	Lateral Expansion (mils)	Expansion (mm)	Shear (%)
<u>Tangential Orientation</u>							
DL6	-100	(-73)	5	(7)	3	(0.08)	5
DL9	- 60	(-51)	10	(14)	11	(0.28)	10
DL1	- 25	(-32)	17	(23)	17	(0.43)	15
DL8	- 5	(-21)	9	(12)	3	(0.08)	10
DL3	0	(-18)	8	(11)	6	(0.15)	10
DL10	5	(-15)	25	(34)	14	(0.36)	15
DL12	25	(- 4)	47	(64)	30	(0.76)	30
DL13	50	(10)	67	(91)	45	(1.14)	50
DL2	75	(24)	97	(132)	58	(1.47)	70
DL4	125	(52)	80	(108)	55	(1.40)	70
DL14	130	(54)	85	(115)	59	(1.50)	85
DL5	150	(66)	98	(133)	67	(1.70)	90
DL11	200	(93)	127	(172)	79	(2.01)	100
DL15	240	(116)	121	(164)	79	(2.01)	100
DL7	275	(135)	118	(160)	73	(1.85)	100
<u>Axial Orientation</u>							
DT8	-75	(-59)	12	(16)	11	(0.28)	10
DT7	-50	(-46)	6	(8)	2	(0.05)	5
DT1	-25	(-32)	7	(9)	3	(0.08)	5
DT3	20	(- 7)	27	(37)	18	(0.46)	15
DT6	35	(2)	18	(24)	10	(0.25)	10
DT15	50	(10)	33	(45)	22	(0.56)	20
DT11	75	(24)	23	(31)	14	(0.36)	15
DT2	75	(24)	35	(47)	20	(0.51)	20
DT9	100	(38)	38	(52)	31	(0.79)	25
DT14	125	(52)	52	(71)	42	(1.07)	45
DT13	165	(74)	60	(81)	47	(1.19)	60
DT10	200	(93)	74	(100)	53	(1.35)	90
DT12	250	(121)	89	(121)	57	(1.45)	100
DT5	285	(141)	79	(107)	60	(1.52)	100
DT4	325	(163)	84	(114)	63	(1.60)	100

TABLE 5-2
 CHARPY V-NOTCH IMPACT DATA FOR THE MCGUIRE UNIT 2
 REACTOR VESSEL WELD METAL AND HAZ METAL IRRADIATED
 AT 550°F, FLUENCE 2.02×10^{19} n/cm² (E > 1.0 MeV).

<u>Sample No.</u>	<u>Temperature</u>		<u>Impact Energy</u>		<u>Lateral Expansion</u>		<u>Shear</u>
	<u>(°F)</u>	<u>(°C)</u>	<u>(ft-lb)</u>	<u>(J)</u>	<u>(mils)</u>	<u>(mm)</u>	
<u>Weld Metal</u>							
DW1	-100	(- 73)	8	(11)	5	(0.13)	10
DW5	- 75	(- 59)	10	(14)	12	(0.30)	29
DW7	- 50	(- 45)	28	(38)	22	(0.56)	45
DW15	- 40	(- 40)	27	(37)	20	(0.51)	45
DW12	- 25	(- 32)	36	(49)	26	(0.66)	60
DW2	0	(- 18)	40	(54)	31	(0.79)	65
DW4	5	(- 15)	48	(65)	37	(0.94)	70
DW13	25	(- 4)	60	(81)	45	(1.14)	75
DW8	75	(24)	89	(121)	58	(1.47)	80
DW11	110	(43)	105	(142)	67	(1.70)	90
DW9	150	(66)	120	(163)	84	(2.13)	95
DW10	175	(79)	116	(157)	82	(2.08)	100
DW3	215	(102)	118	(160)	82	(2.08)	100
DW14	250	(121)	141	(191)	90	(2.29)	100
DW6	300	(149)	141	(191)	83	(2.11)	100
<u>HAZ Metal</u>							
DH10	-100	(- 73)	17	(23)	9	(0.23)	15
DH9	- 80	(- 62)	27	(37)	11	(0.28)	20
DH13	- 60	(- 51)	19	(26)	13	(0.33)	25
DH5	- 40	(- 40)	20	(27)	17	(0.43)	45
DH3	- 25	(- 32)	28	(38)	19	(0.48)	50
DH11	- 5	(- 21)	28	(38)	22	(0.56)	65
DH7	0	(- 18)	24	(33)	20	(0.51)	60
DH6	15	(- 9)	53	(72)	34	(0.86)	75
DH2	40	(4)	57	(77)	42	(1.07)	80
DH12	70	(21)	65	(88)	44	(1.12)	85
DH4	100	(38)	74	(100)	52	(1.32)	90
DH14	140	(60)	91	(123)	60	(1.52)	100
DH8	175	(79)	92	(125)	61	(1.55)	100
DH15	225	(107)	98	(133)	62	(1.57)	100
DH1	275	(175)	92	(125)	61	(1.55)	100

TABLE 5-3

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE MCGUIRE UNIT 2 INTERMEDIATE
SHELL FORGING 05 IRRADIATED AT 550°F, FLUENCE 2.02×10^{19} n/cm² (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies			Yield Load (lbs)	Time to Yield (msec)	Maximum Load (lbs)	Time to Maximum (msec)	Fracture Load (lbs)	Arrest Load (lbs)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
(ft-lb/in ²)													
Tangential Orientation													
DL6	-100	5	40	32	8	3860	0.12	4073	0.13	4073	62	128	132
DL6	- 80	10	81	58	25	3931	0.14	4099	0.17	4099	66	131	133
DL1	- 25	17	137	120	17	3904	0.14	4708	0.43	4708	71	130	143
DL8	- 5	9	72	35	38	3658	0.13	3845	0.14	3760	-	121	126
DL3	0	8	84	31	33	3684	0.13	3734	0.14	3734	-	122	123
DL10	5	25	201	168	33	3690	0.13	4618	0.38	4618	74	123	138
DL12	25	47	378	343	38	3738	0.14	4864	0.70	4864	74	124	143
DL13	50	87	540	338	202	3627	0.14	4768	0.89	4478	112	120	139
DL2	75	97	781	339	442	3514	0.14	4830	0.68	3971	455	117	139
DL4	125	80	644	148	496	3148	0.13	4145	0.38	4056	1085	105	121
DL14	130	85	684	390	295	3244	0.13	4420	0.83	3126	971	108	127
DL5	150	98	789	318	471	3132	0.14	4549	0.69	2957	1508	104	128
DL11	200	127	1023	313	710	3169	0.13	4370	0.69	*	*	105	125
DL15	240	121	974	311	663	3168	0.15	4473	0.69	*	*	105	127
DL7	275	118	950	322	628	2908	0.13	4239	0.77	*	*	97	119
Axial Orientation													
DT8	- 75	12	97	75	22	4098	0.17	4386	0.21	4386	64	136	141
DT7	- 50	6	48	25	23	3032	0.12	3242	0.13	3242	125	101	104
DT1	- 25	7	56	40	17	3903	0.13	4043	0.15	4043	-	130	132
DT3	20	27	317	151	66	3739	0.14	4489	0.36	4352	-	124	136
DT6	35	18	145	122	23	3757	0.13	4271	0.30	4271	55	125	133
DT15	50	33	266	249	17	3676	0.14	4277	0.54	4577	67	122	137
DT11	75	23	185	106	80	3092	0.13	3730	0.31	3730	229	103	113
DT2	75	35	282	161	121	3484	0.13	4437	0.38	4437	366	116	132
DT9	100	38	306	241	65	3386	0.13	4433	0.54	4433	476	112	130
DT14	125	52	419	279	140	3450	0.15	4377	0.61	4140	740	115	130
DT13	165	30	483	292	191	3056	0.14	3957	0.54	3633	1631	101	116
DT10	200	74	598	300	295	3201	0.13	4363	0.69	3380	1753	106	126
DT12	250	89	717	229	488	3498	0.22	4266	0.54	*	*	116	129
DT5	285	79	636	215	421	3059	0.13	4171	0.52	*	*	102	120
DT4	325	84	676	209	467	2877	0.14	4023	0.52	*	*	96	115

*Fully ductile fracture; no brittle fracture load and no arrest load.

TABLE 5-4

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE MCGUIRE UNIT 2 WELD METAL AND
HEAT-AFFECTED-ZONE (HAZ) METAL, IRRADIATED AT 550°F, FLUENCE 2.02×10^{19} n/cm² (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies			Yield Load (lbs)	Time to Yield (msec)	Maximum Load (lbs)	Time to Maximum (msec)	Fracture Load (lbs)	Arrest Load (lbs)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
Weld Metal													
DW1	-100	8	64	32	32	3843	0.12	4010	0.13	4010	464	128	130
DW6	-75	10	81	37	43	3946	0.14	4039	0.15	4039	302	131	133
DW7	-50	28	226	169	56	3706	0.14	4648	0.38	4846	690	123	139
DW15	-40	27	217	176	50	3747	0.13	4507	0.38	4507	1601	124	137
DW12	-25	36	290	93	197	2208	0.10	2623	0.35	3718	984	73	80
DW2	0	40	322	237	85	3487	0.14	4507	0.52	4309	988	116	133
DW4	5	48	387	236	150	3689	0.14	4548	0.53	4548	1514	123	137
DW13	25	60	483	334	149	3458	0.13	4567	0.69	4567	2286	115	133
DW8	75	89	717	324	393	3337	0.13	4488	0.69	4056	2873	111	130
DW11	110	105	845	314	531	2775	0.12	3935	0.77	-	-	92	111
DW9	150	120	96A	359	607	3278	0.15	4527	0.77	-	-	109	126
DW10	175	116	934	309	626	2777	0.13	3729	0.77	-	-	92	108
DW3	215	118	950	284	665	2985	0.13	4148	0.69	-	-	98	118
DW14	250	141	1135	357	778	3174	0.15	4293	0.78	-	-	106	124
DW6	300	141	1125	422	714	3157	0.21	3901	1.02	-	-	106	117
HAZ Metal													
DH10	-100	17	137	111	26	4073	0.14	4793	0.27	4793	97	136	147
DH9	-80	27	217	81	136	4298	0.14	4551	0.21	4551	61	143	147
DH13	-60	19	153	127	28	4178	0.14	4751	0.30	4751	303	136	148
DH5	-40	20	161	71	90	4008	0.14	4297	0.21	4297	1606	153	138
DH3	-25	28	225	177	48	3936	0.14	4794	0.38	4794	530	131	145
DH11	-5	28	225	116	110	3451	0.13	4518	0.31	4518	1838	115	133
DH7	0	24	153	92	101	3674	0.15	4340	0.25	4340	1380	122	133
DH6	15	53	427	231	195	3591	0.13	4773	0.50	4394	1423	119	139
DH2	40	57	459	301	158	3911	0.13	4743	0.61	4743	2466	130	144
DH12	70	65	523	247	277	3641	0.14	4618	0.52	4352	2406	121	137
DH4	100	74	596	239	357	3585	0.14	4617	0.51	-	-	119	136
DH14	140	91	733	311	422	3301	0.13	4460	0.70	-	-	110	129
DH8	175	92	741	246	495	3464	0.13	4537	0.54	-	-	115	133
DH15	225	98	789	271	518	3250	0.14	4333	0.82	-	-	108	126
DH1	275	92	741	314	427	3145	0.22	4338	0.77	-	-	104	124

-Fully ductile fracture; no brittle fracture load and no arrest load.

TABLE 5-5
EFFECT OF 550°F IRRADIATION TO 2.02×10^{19} n/cm² (E > 1.0 MeV)
ON THE NOTCH TOUGHNESS PROPERTIES OF THE MCGUIRE UNIT 2 REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Average 30 ft-lb ⁽¹⁾			Average 35 mil ⁽¹⁾			Average 50 ft-lb ⁽¹⁾			Average Energy ⁽¹⁾		
	Transition			Lateral Expansion			Transition			Absorption at		
	Temperature (°F)			Temperature (°F)			Temperature (°F)			Full Shear (ft-lb)		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ(ft-lb)
Forging 05 (Axial)	- 25	60	85	25	110	85	25	115	90	94	84	- 10
Forging 05 (Tangential)	- 75	15	90	- 70	35	105	- 60	45	105	156	122	- 34
Weld Metal	- 50	- 30	20	- 45	0	45	- 25	10	35	133	129	- 4
HAZ Metal	- 90	- 5	85	- 70	30	100	- 55	30	85	104	93	- 11

(1) "AVERAGE" is defined as the value read from the curve fitted through the data points of the Charpy tests (Figures 5-1 through 5-4).

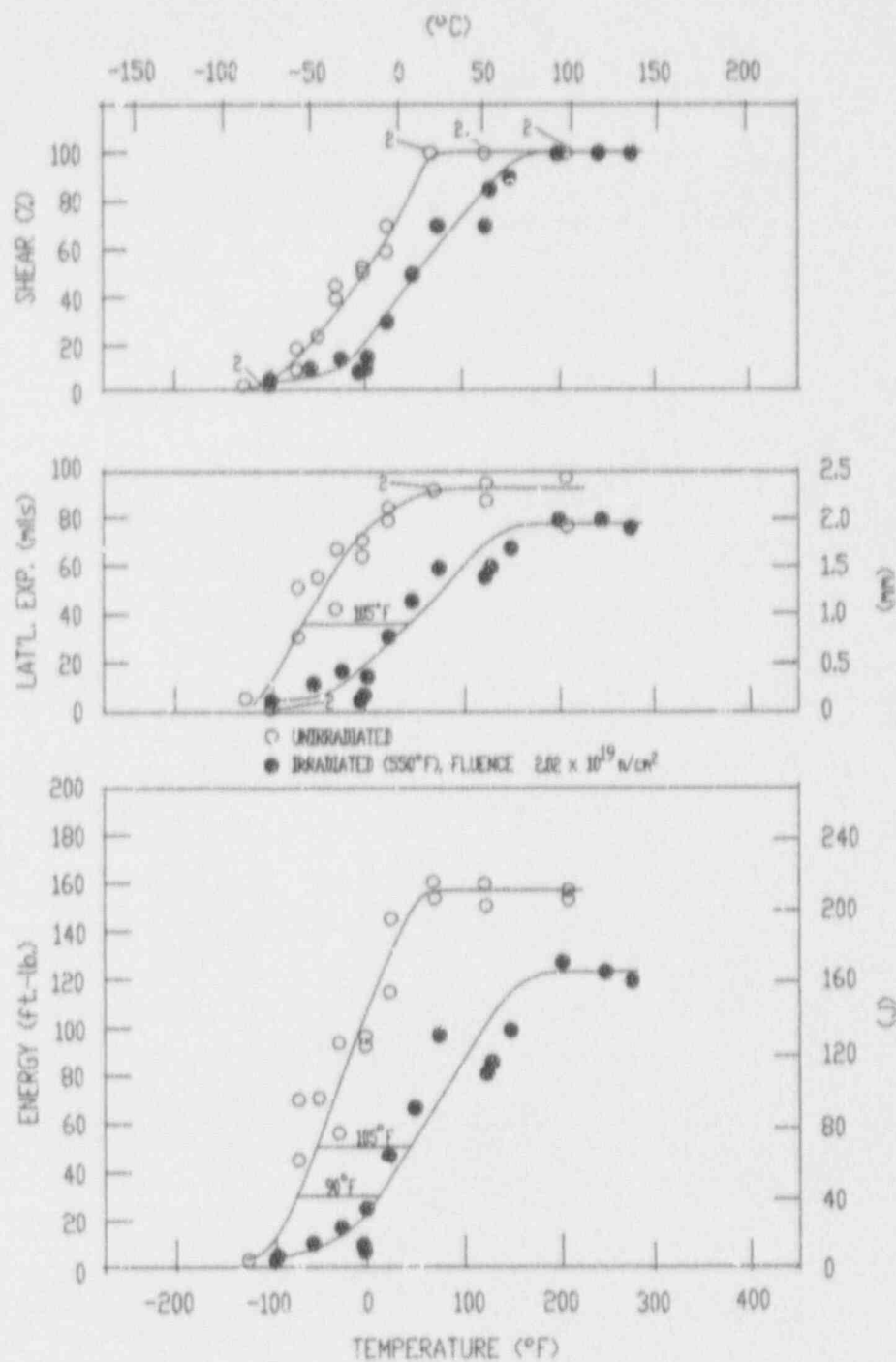
TABLE 5-6
COMPARISON OF THE MCGUIRE UNIT 2 SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE
SHIFTS AND UPPER SHELF ENERGY DECREASES WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

Material	Capsule	Fluence 10^{19} n/cm ²	30 ft-lb Transition Temp. Shift		Upper Shelf Energy Decrease	
			Predicted (a) (°F)	Measured (°F)	Predicted (a) (%)	Measured (%)
Forging 05 (Axial)	V	0.337	80	70	18	10
	X	1.45	126	105	26	18
	U	2.02	136	85	30	11
Forging 05 (Tangential)	V	0.337	80	65	18	14
	X	1.45	126	100	26	13
	U	2.02	136	90	30	22
Weld Metal	V	0.337	34	45	15	0
	X	1.45	54	35	21	0
	U	2.02	58	20	23	3
HAZ Metal	V	0.337	--	55	--	6
	X	1.45	--	75	--	0
	U	2.02	--	85	--	11

TABLE 5-7

TENSILE PROPERTIES FOR THE MCGUIRE UNIT 2 REACTOR VESSEL SURVEILLANCE
 MATERIALS IRRADIATED AT 550°F TO 2.02×10^{19} n/cm² (E > 1.0 MeV)

<u>Material</u>	<u>Sample Number</u>	<u>Test Temp. (°F)</u>	<u>0.2% Yield Strength (ksi)</u>	<u>Ultimate Strength (ksi)</u>	<u>Fracture Load (kip)</u>	<u>Fracture Stress (ksi)</u>	<u>Fracture Strength (ksi)</u>	<u>Uniform Elongation (%)</u>	<u>Total Elongation (%)</u>	<u>Reduction in Area (%)</u>
Forging 05, Heat No. 526840 (Tangential)	DL1	70	78.4	96.8	3.00	180.1	61.1	12.0	25.2	66
	DL2	180	73.8	92.7	2.90	184.9	59.1	12.0	24.1	68
	DL3	550	67.7	90.7	3.10	139.8	63.2	10.6	21.1	66
Forging 05 Heat No. 526840 (Axial)	DT1	100	80.2	98.6	3.45	165.2	70.3	12.0	22.6	57
	DT2	230	75.4	94.1	3.26	136.8	66.2	11.1	20.6	52
	DT3	550	70.3	92.7	3.70	145.4	75.4	9.0	17.3	48
Weld	DW1	40	75.4	92.7	2.75	178.6	56.0	13.6	28.4	69
	DW2	200	76.4	85.6	2.55	167.7	51.9	12.0	25.8	69
	DW3	550	72.1	85.6	2.80	164.0	57.4	10.6	20.4	63



TME
JEC0792A

Figure 5-1. Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

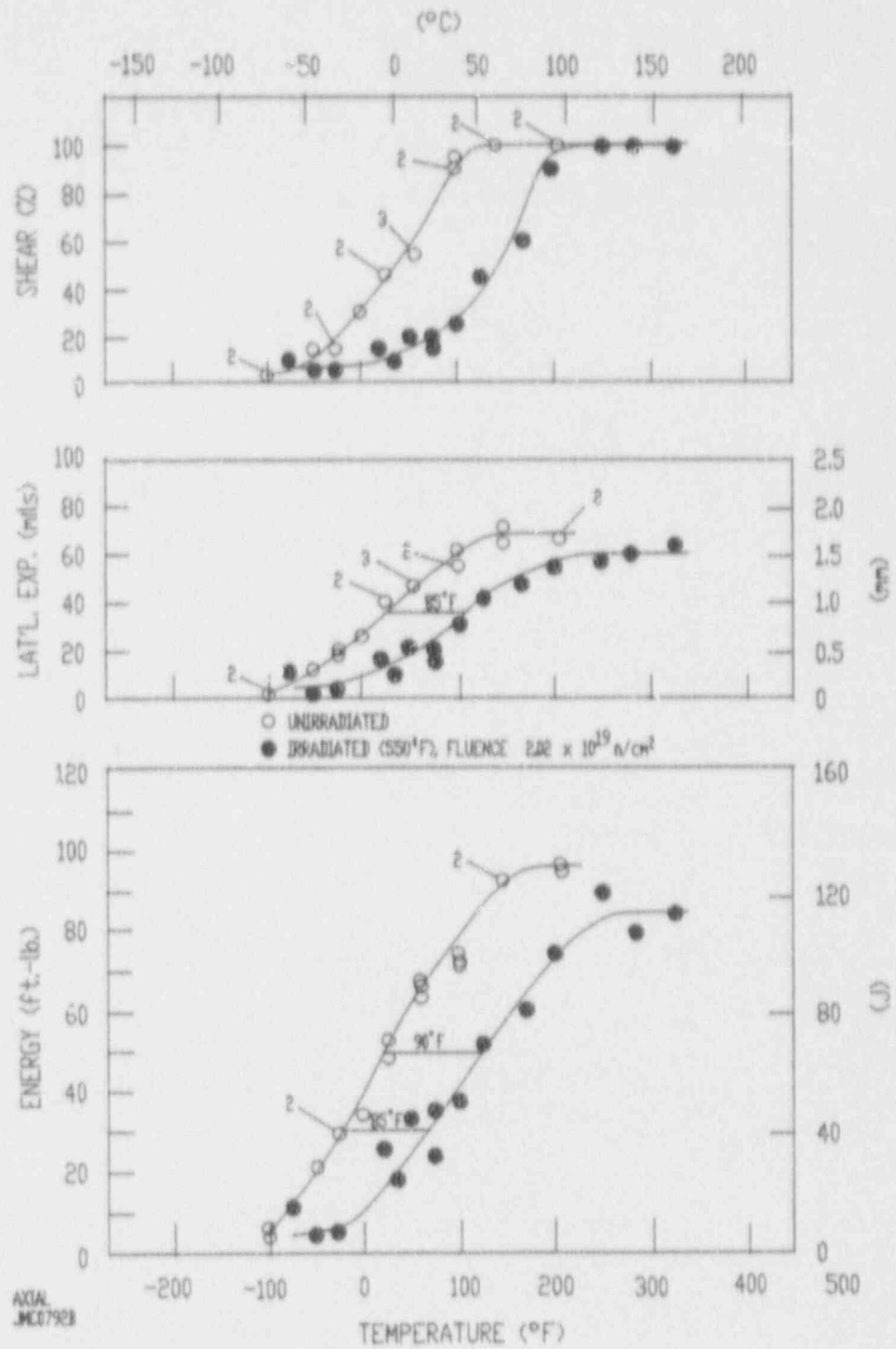


Figure 5-2. Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

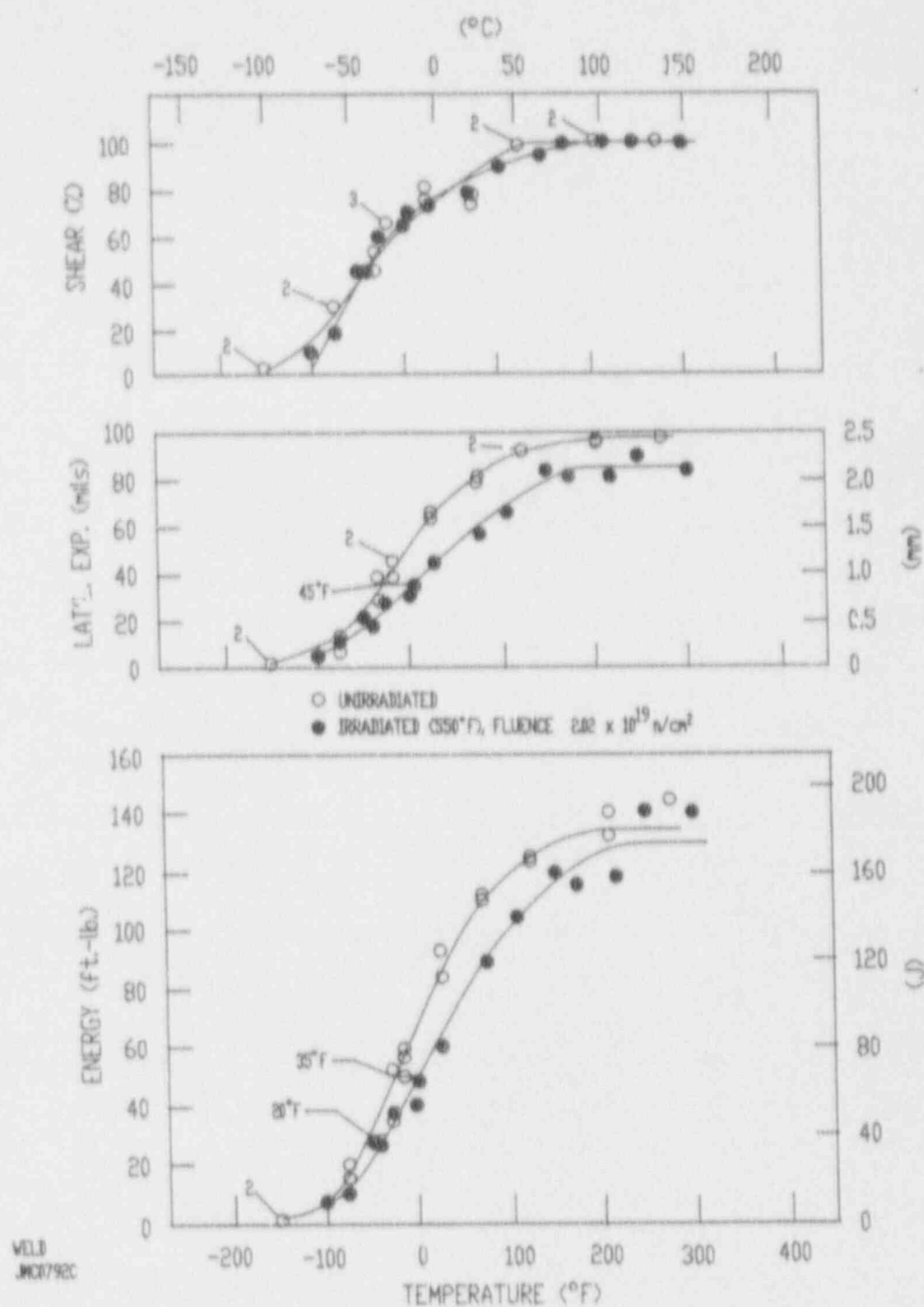


Figure 5-3. Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal

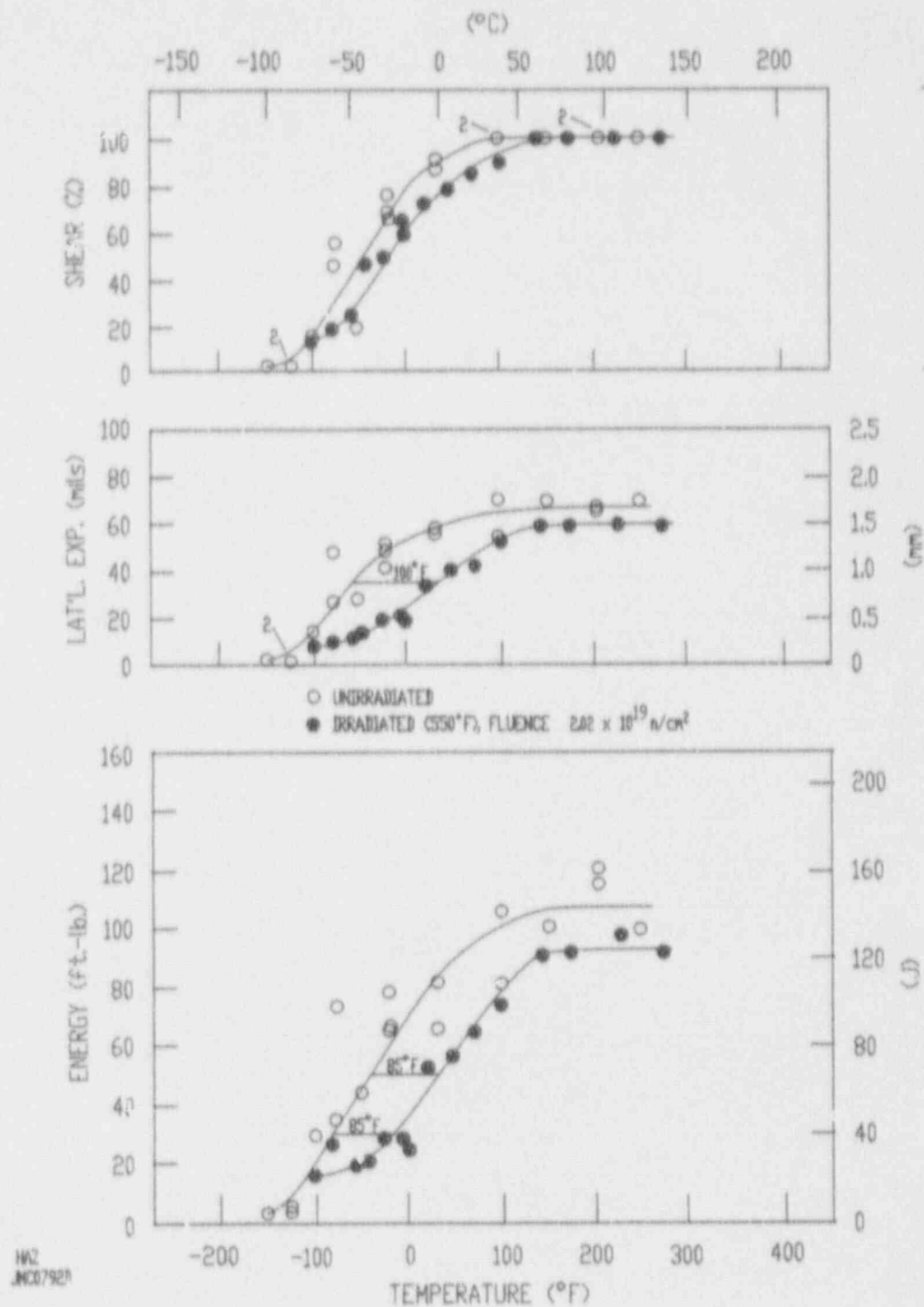


Figure 5-4. Charpy V-Notch Impact Properties for McGuire Unit 2 Reactor Vessel Weld Heat-Affected-Zinc Metal

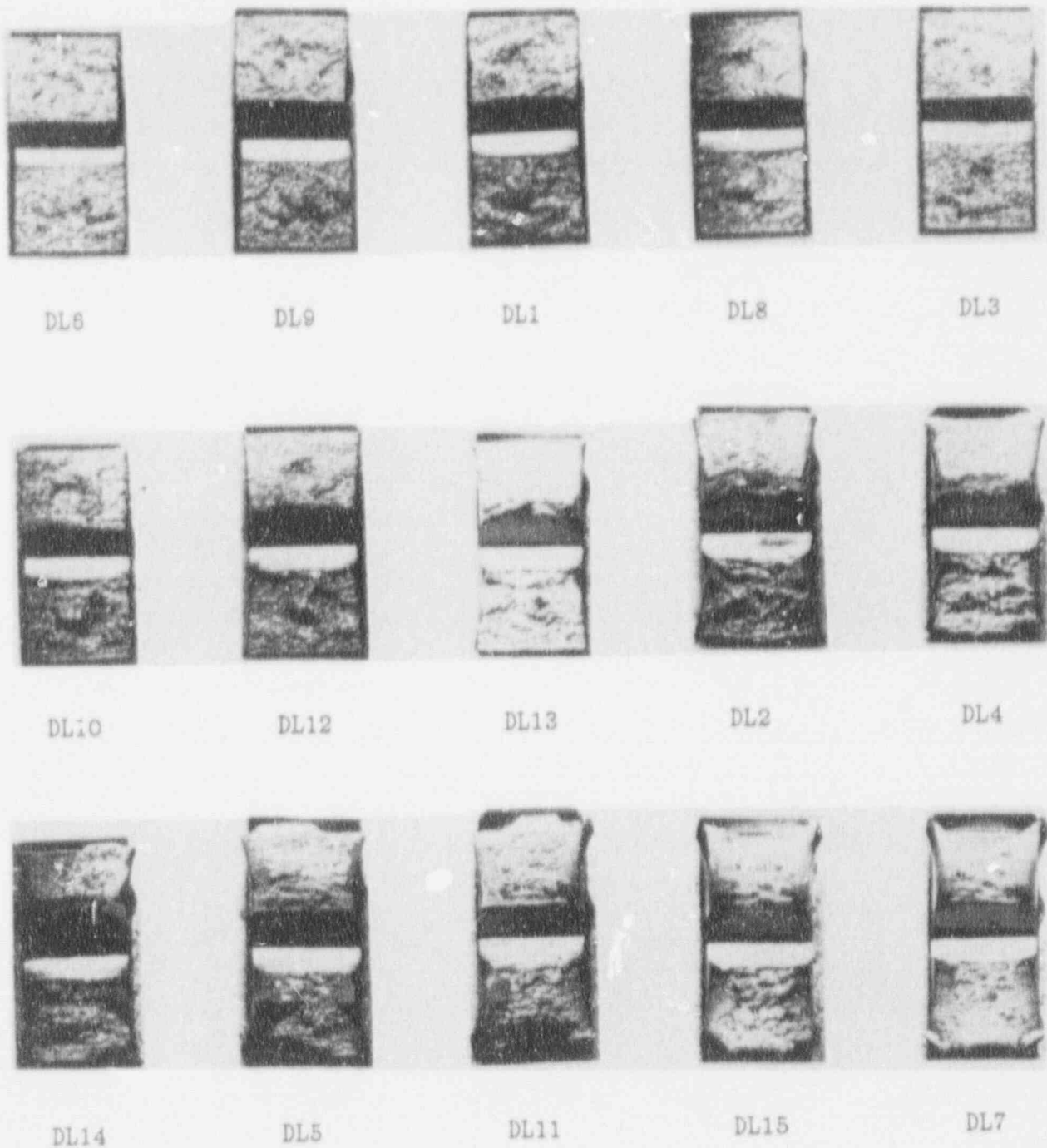


Figure 5-5. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

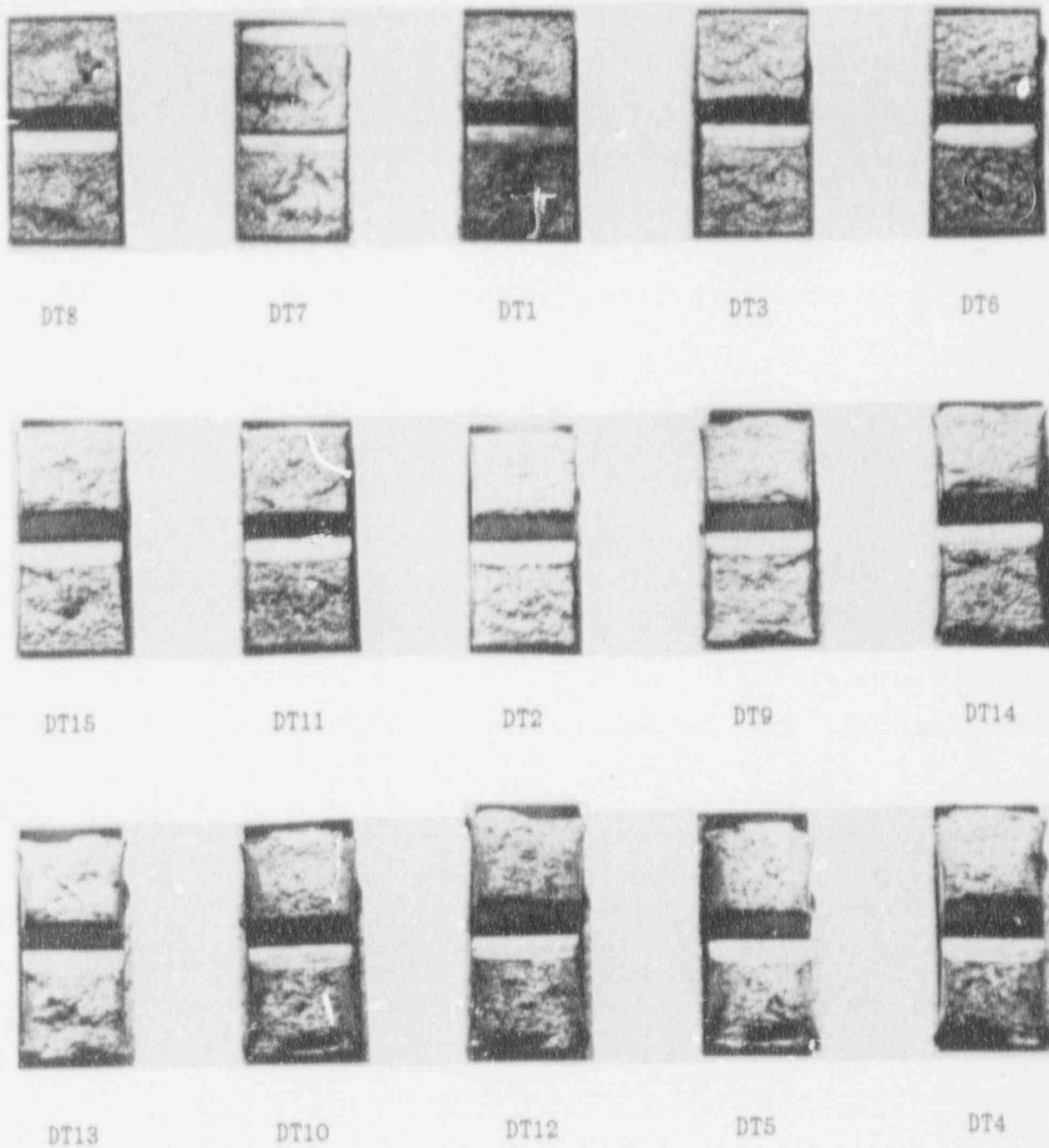
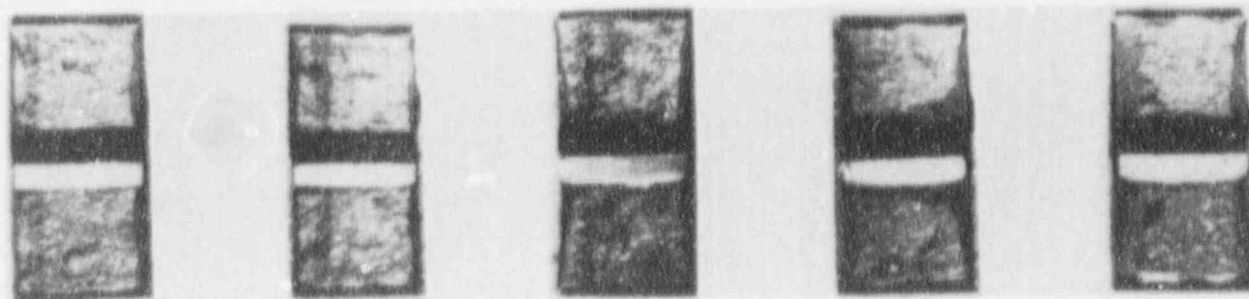


Figure 5-6. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)



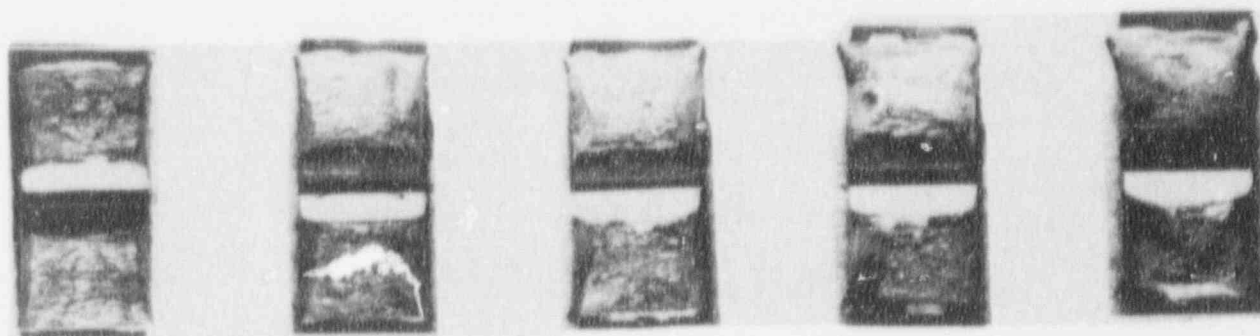
DW1

DW5

DW7

DW15

DW12



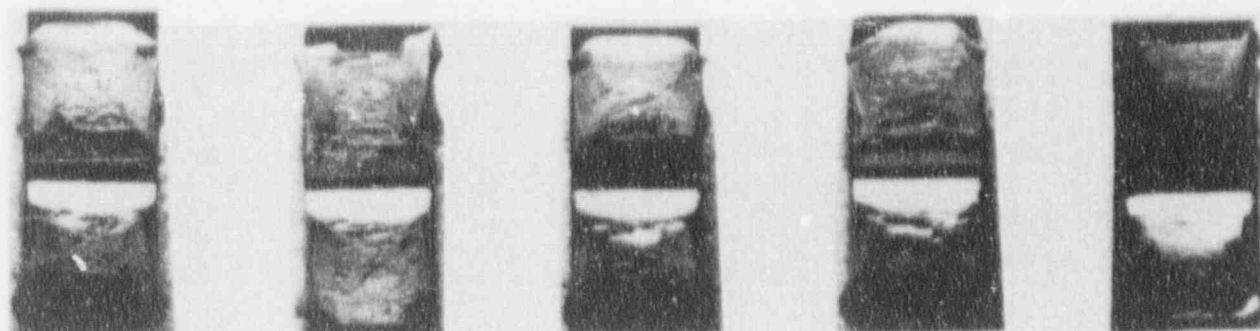
DW2

DW4

DW13

DW8

DW11



DW9

DW10

DW3

DW14

DW6

Figure 5-7. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2
Reactor Vessel Surveillance Weld Metal

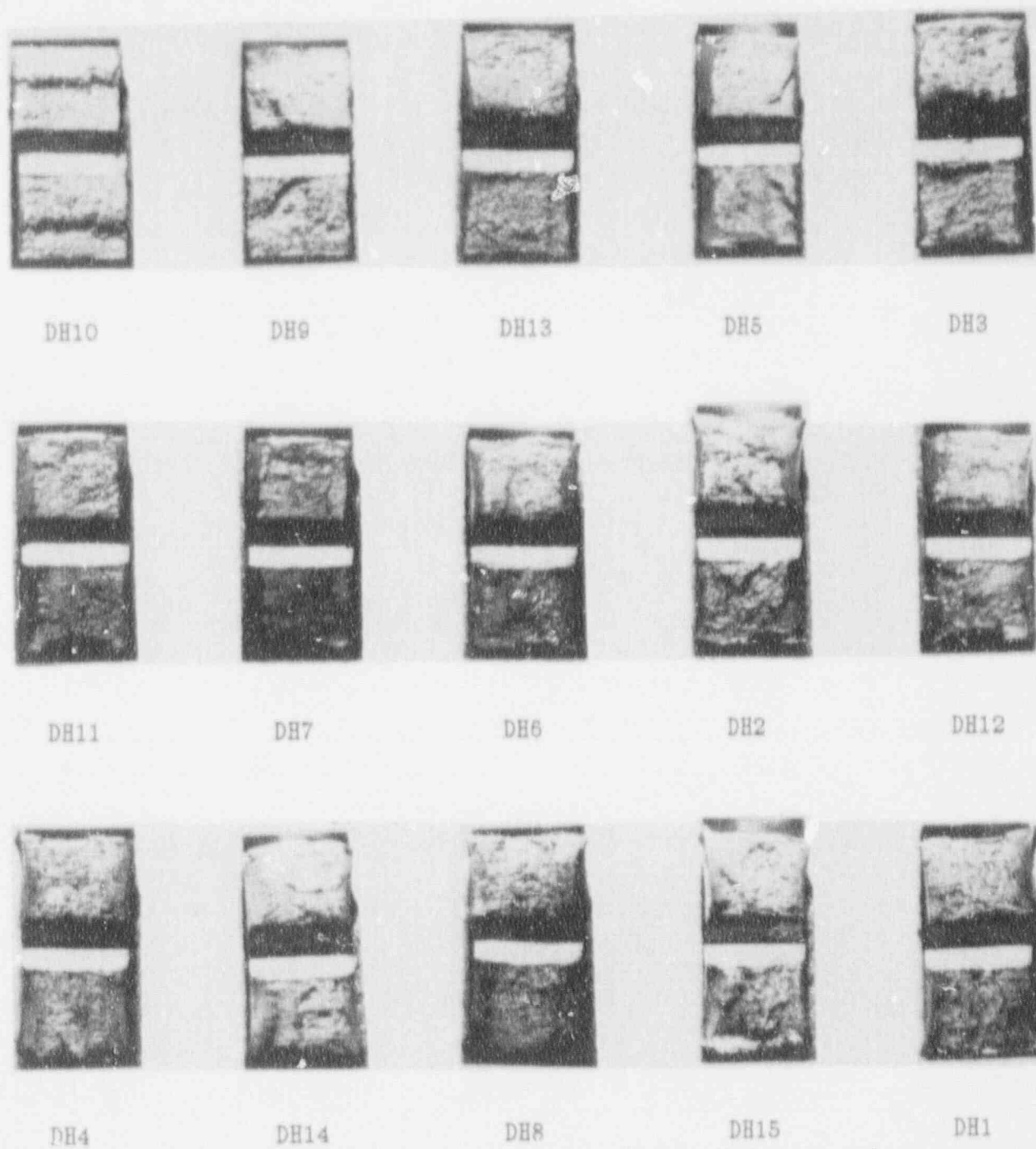


Figure 5-8. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal

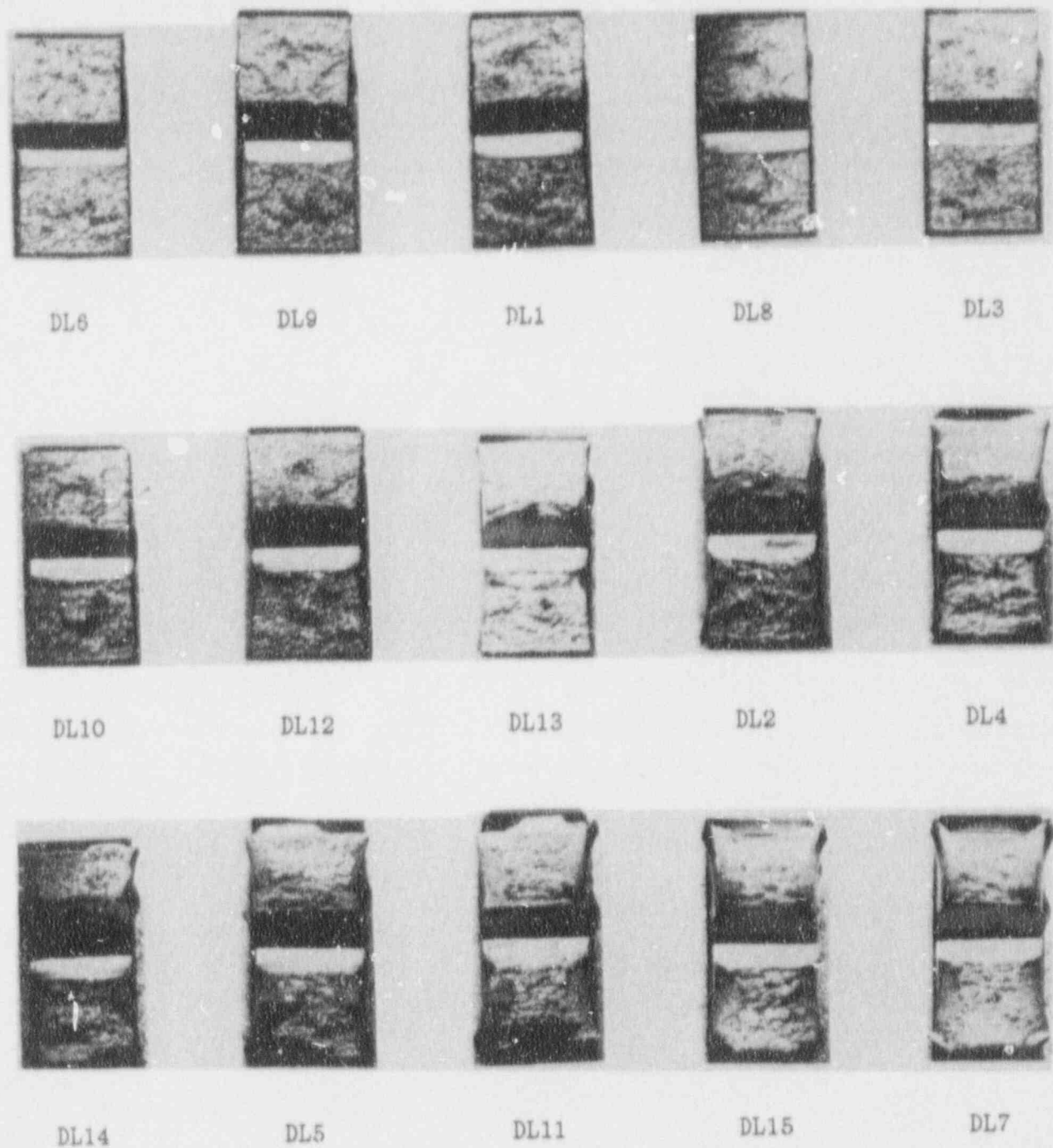
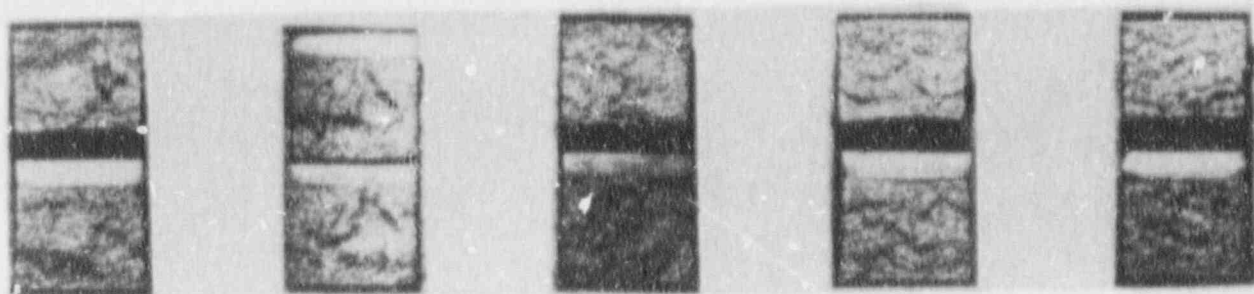


Figure 5-5. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)



DT8

DT7

DT1

DT3

DT6



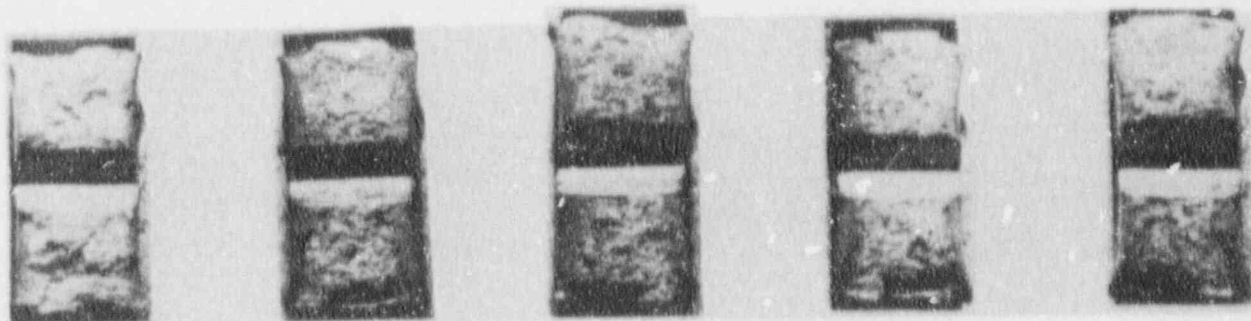
DT15

DT11

DT2

DT9

DT14



DT13

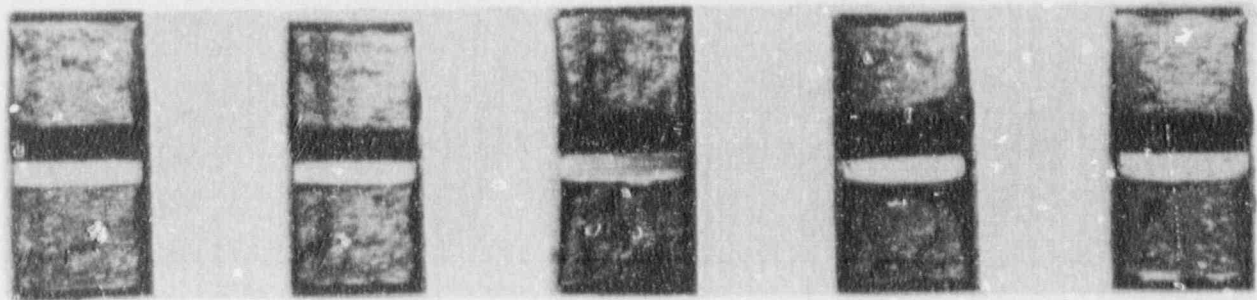
DT10

DT12

DT5

DT4

Figure 5-6. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2
Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)



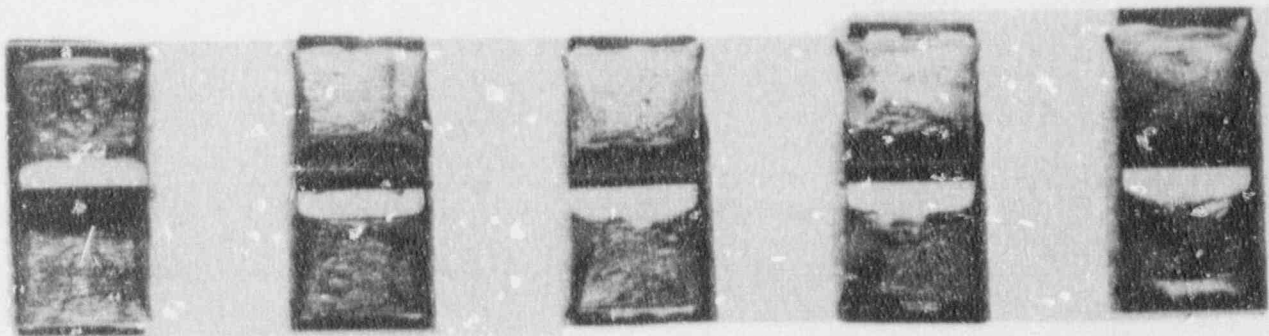
DW1

DW5

DW7

DW15

DW12



DW2

DW4

DW13

DW8

DW11



DW9

DW10

DW3

DW14

DW6

Figure 5-7. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal

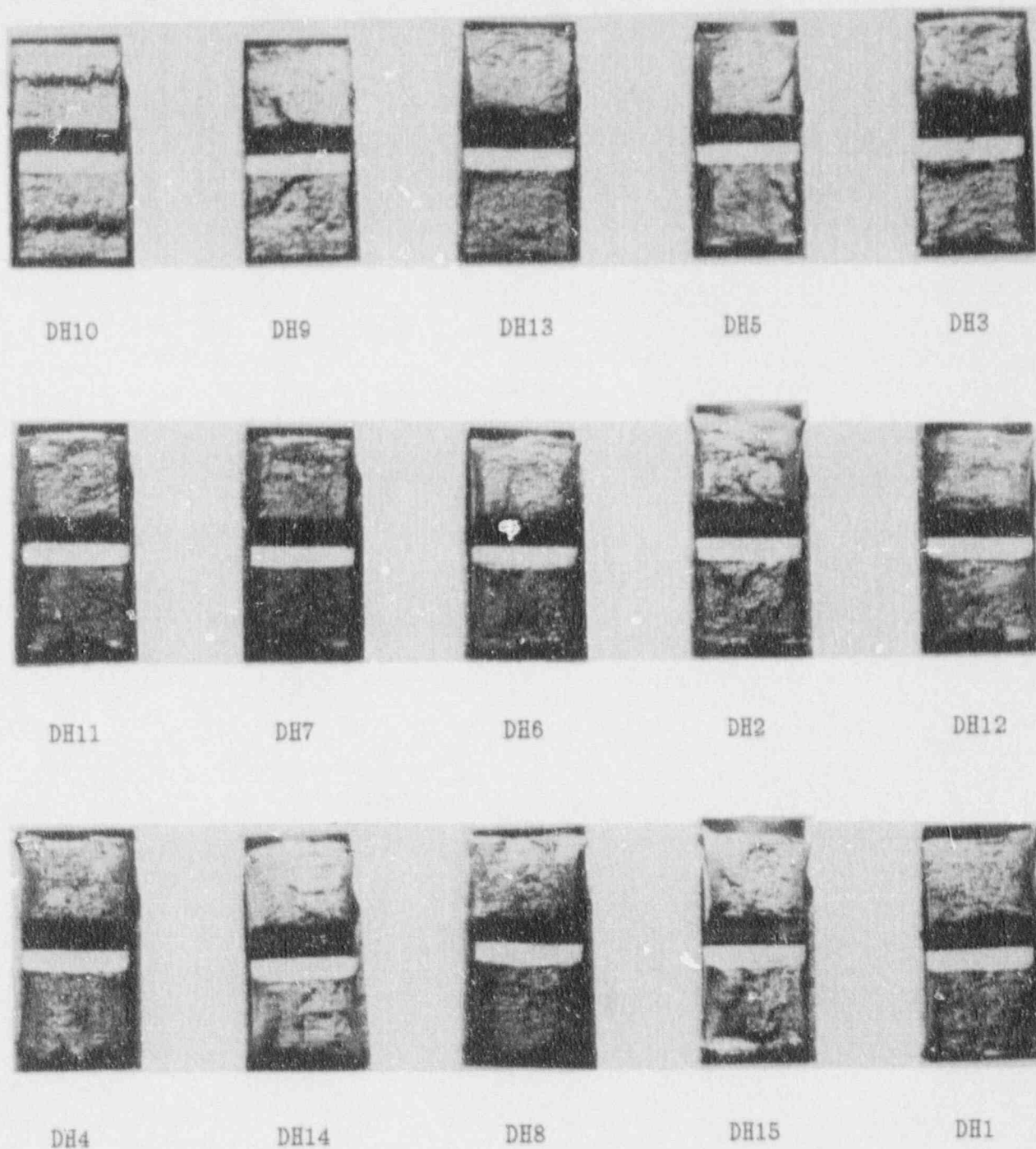


Figure 5-8. Charpy Impact Specimen Fracture Surfaces for McGuire Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal

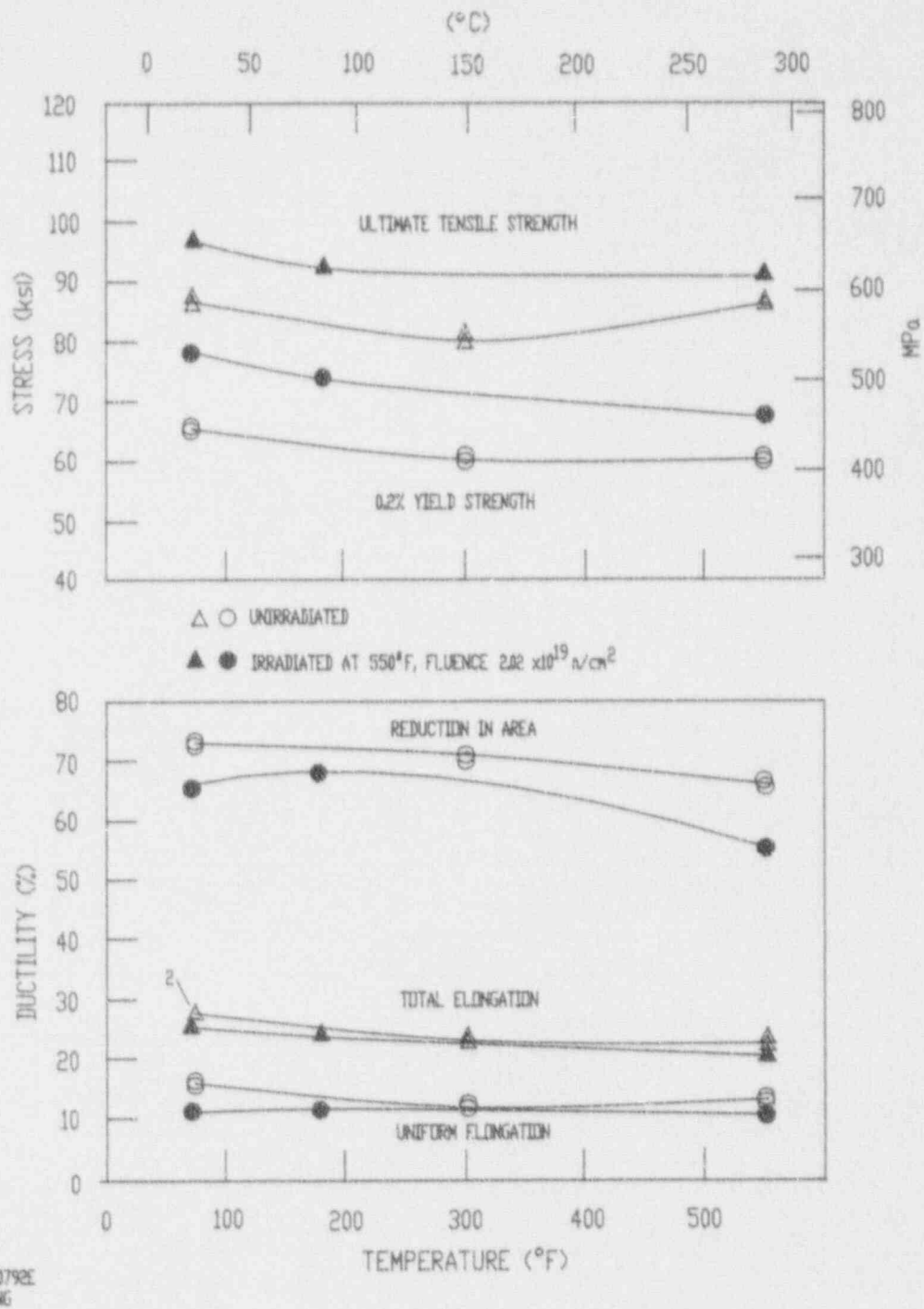


Figure 5-9. Tensile Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

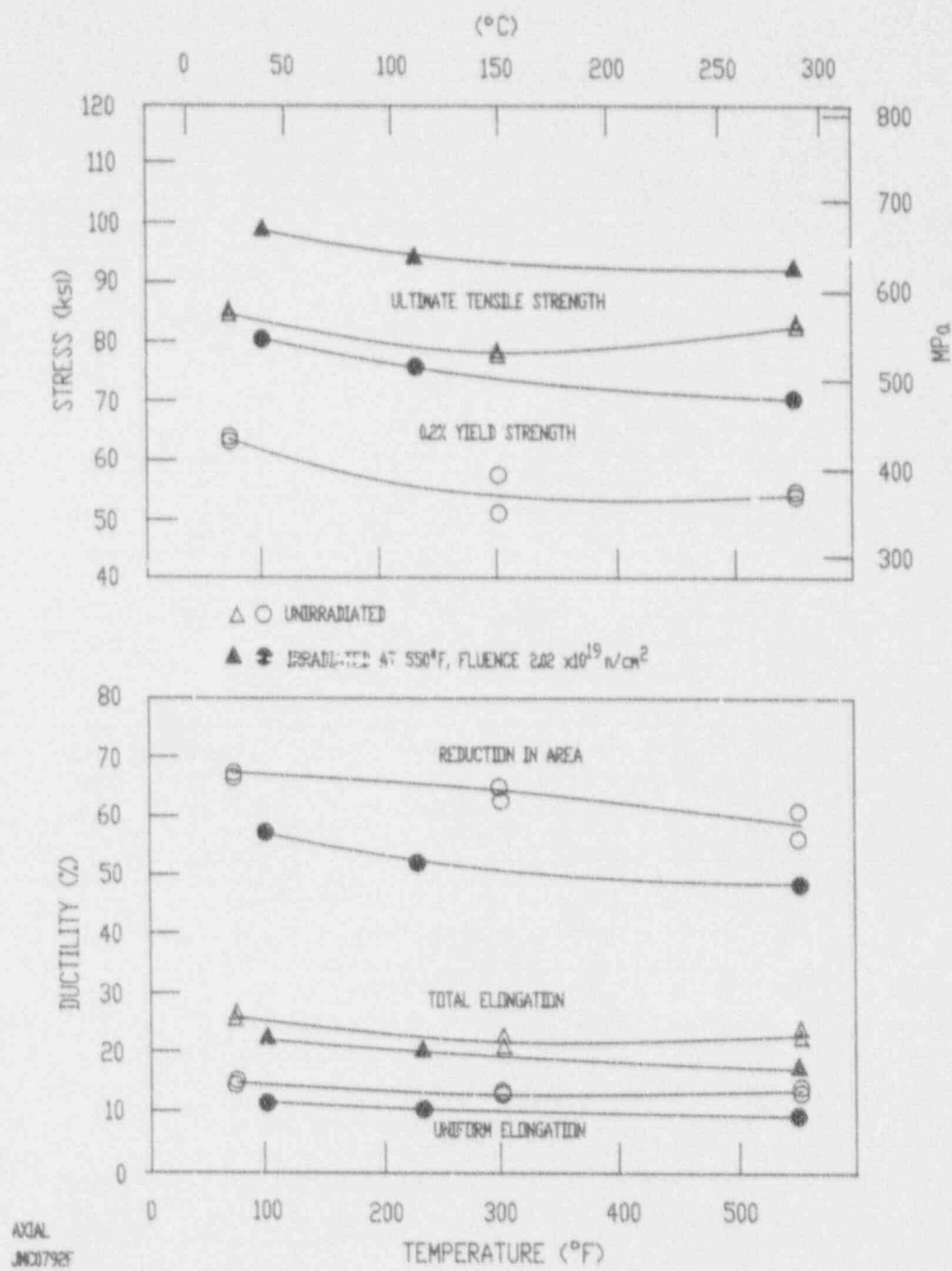


Figure 5-10. Tensile Properties for McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

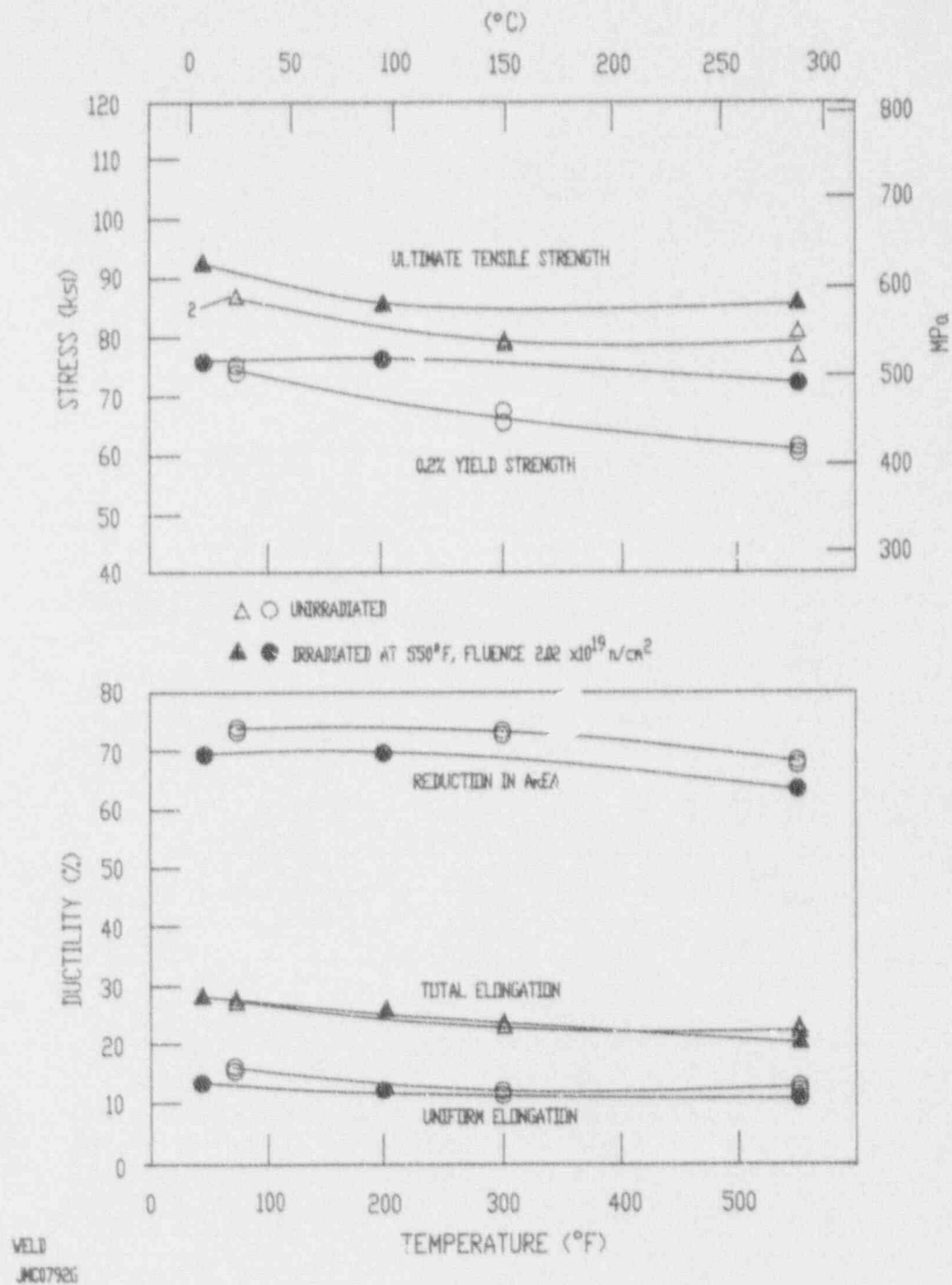
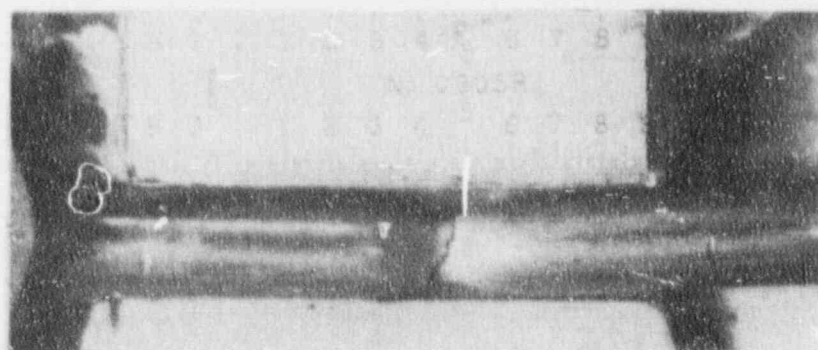


Figure 5-11. Tensile Properties for McGuire Unit 2 Reactor Vessel Surveillance Weld Metal



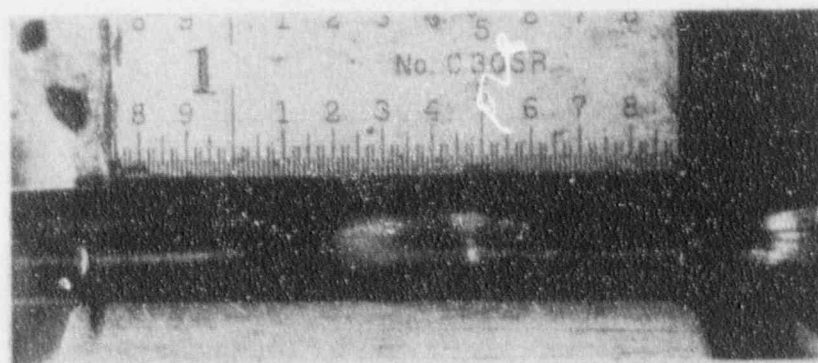
Specimen DL1

70°F



Specimen DL2

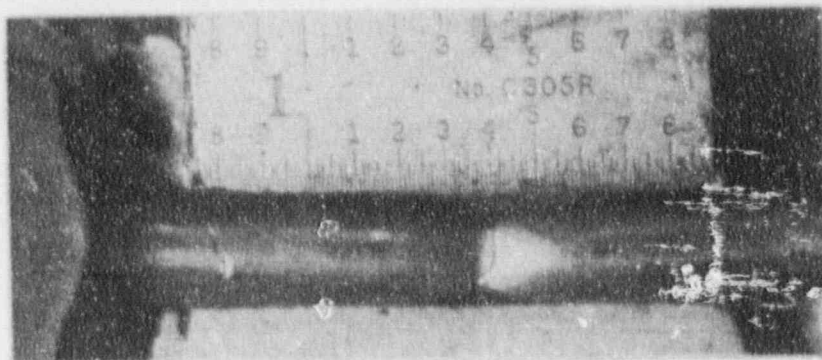
180°F



Specimen DL3

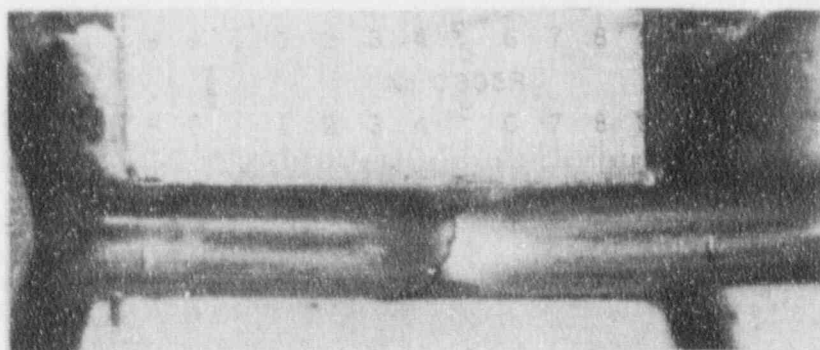
550°F

Figure 5-12. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)



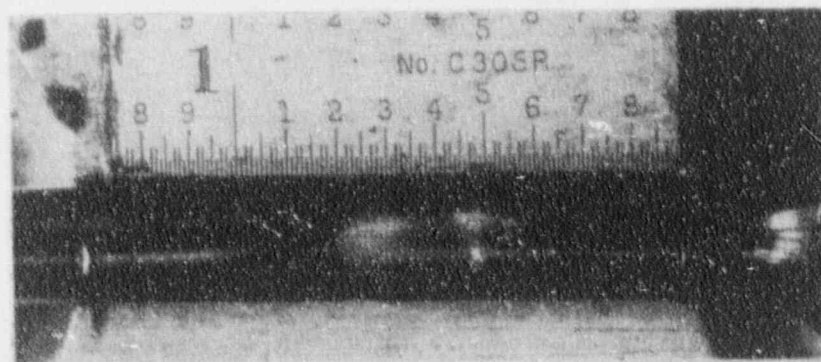
Specimen DL1

70°F



Specimen DL2

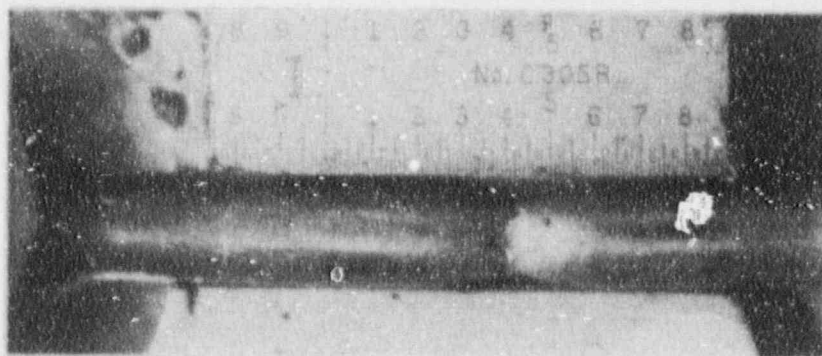
180°F



Specimen DL3

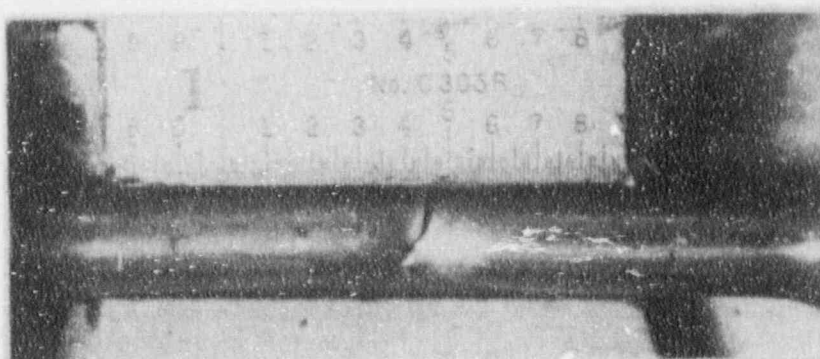
550°F

Figure 5-12. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)



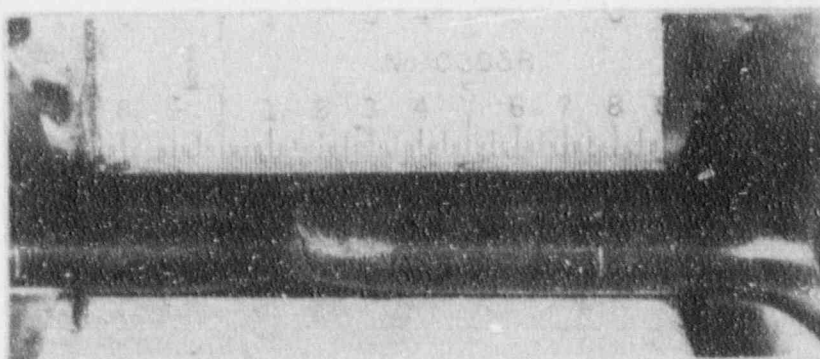
Specimen DT1

100°F



Specimen DT2

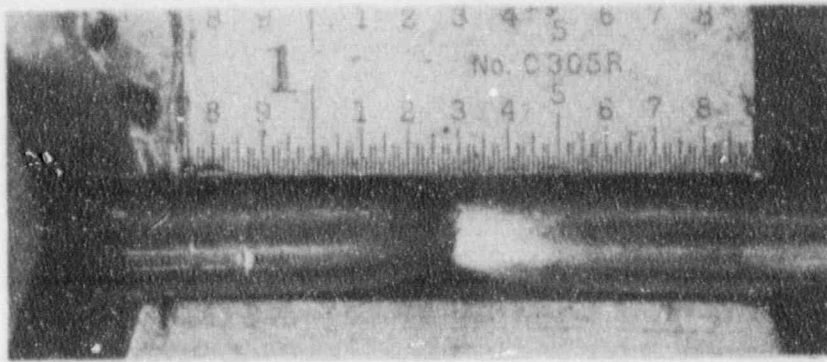
230°F



Specimen DT3

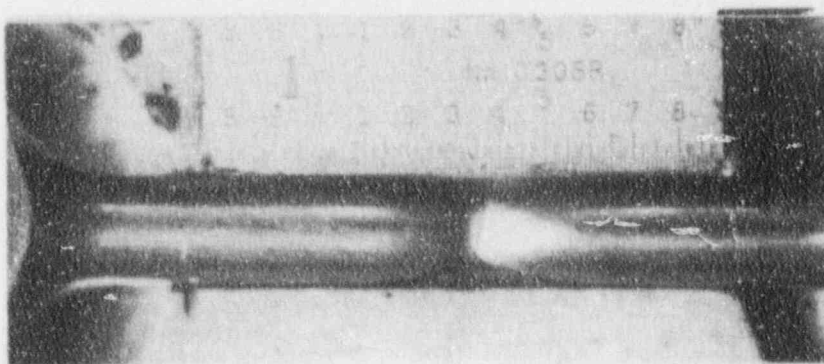
550°F

Figure 5-13. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)



Specimen DW1

40°F



Specimen DW2

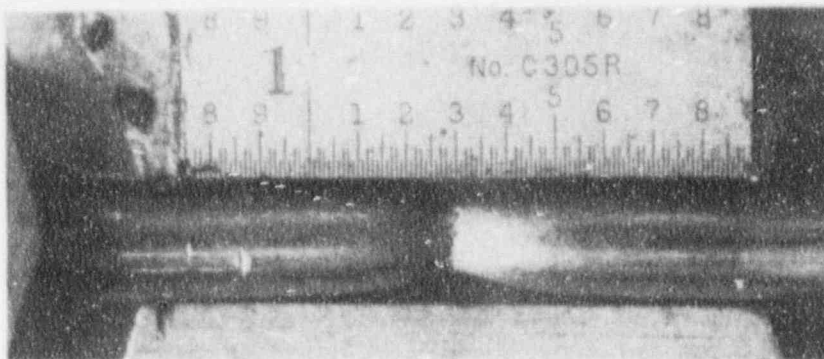
200°F



Specimen DW3

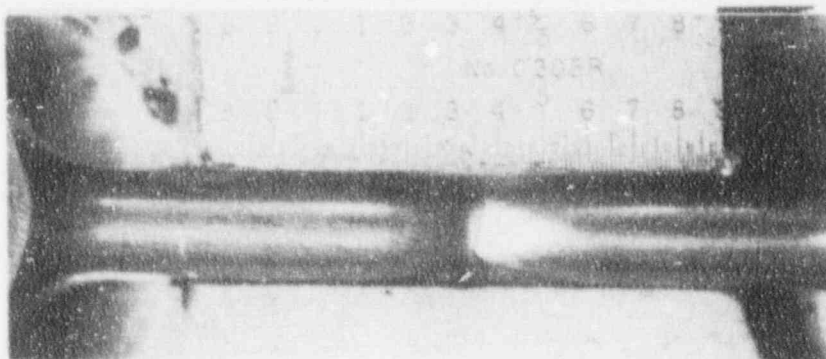
550°F

Figure 5-14. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Surveillance Weld Metal



Specimen DW1

40°F



Specimen DW2

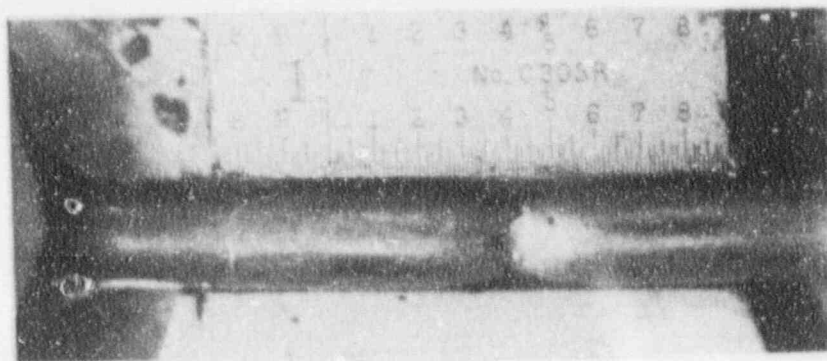
200°F



Specimen DW3

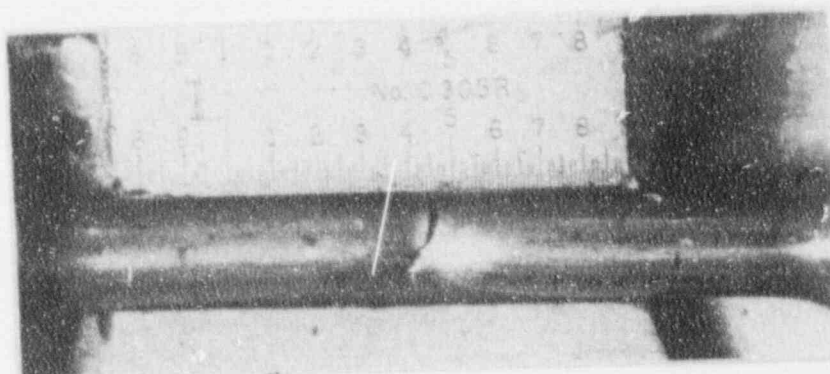
550°F

Figure 5-14. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Surveillance Weld Metal



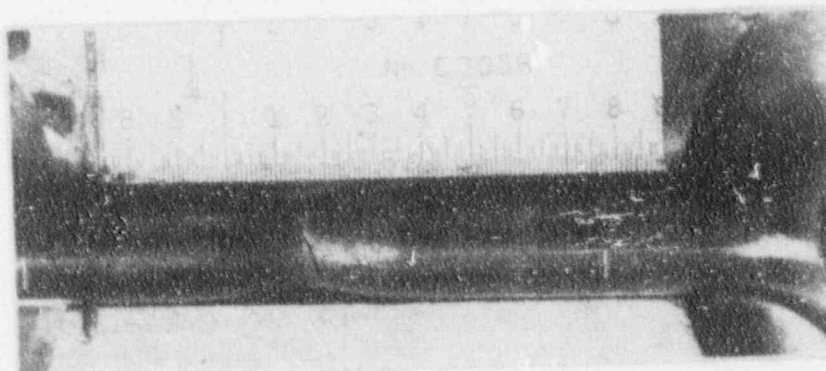
Specimen DT1

100°F



Specimen DT2

230°F



Specimen DT3

550°F

Figure 5-13. Fractured Tensile Specimens from McGuire Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

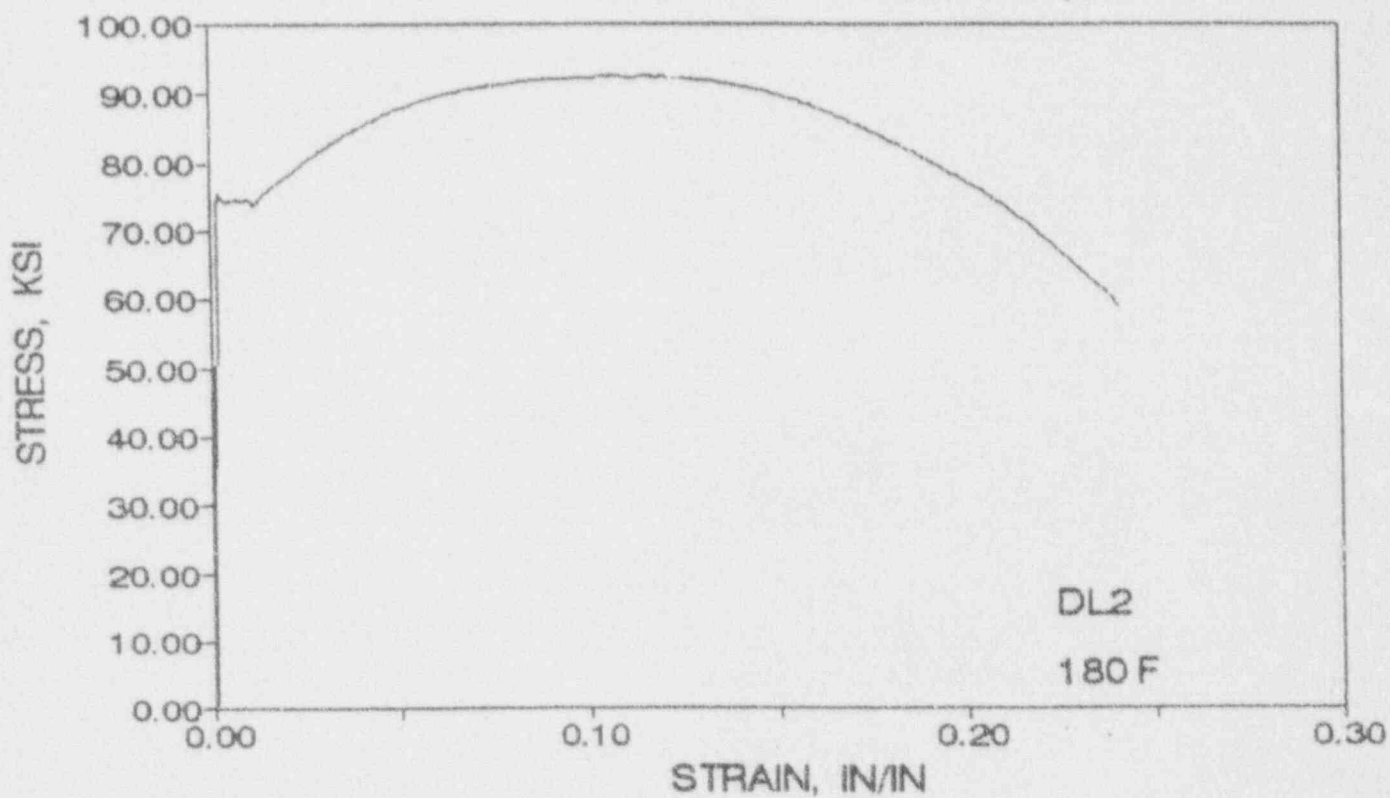
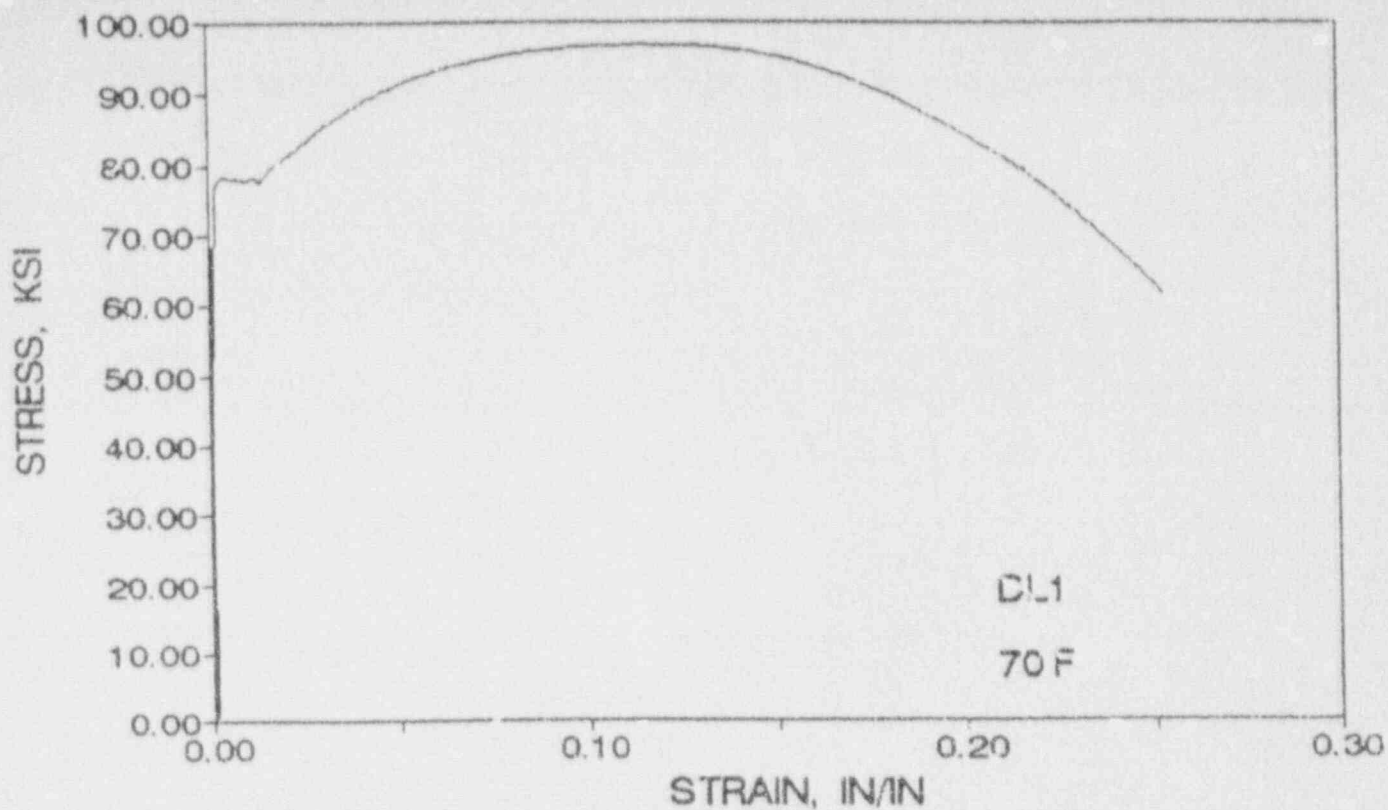


Figure 5-15. Engineering Stress-Strain Curves for Forging 05 Tensile Specimens DL1 and DL2 (Tangential Orientation)

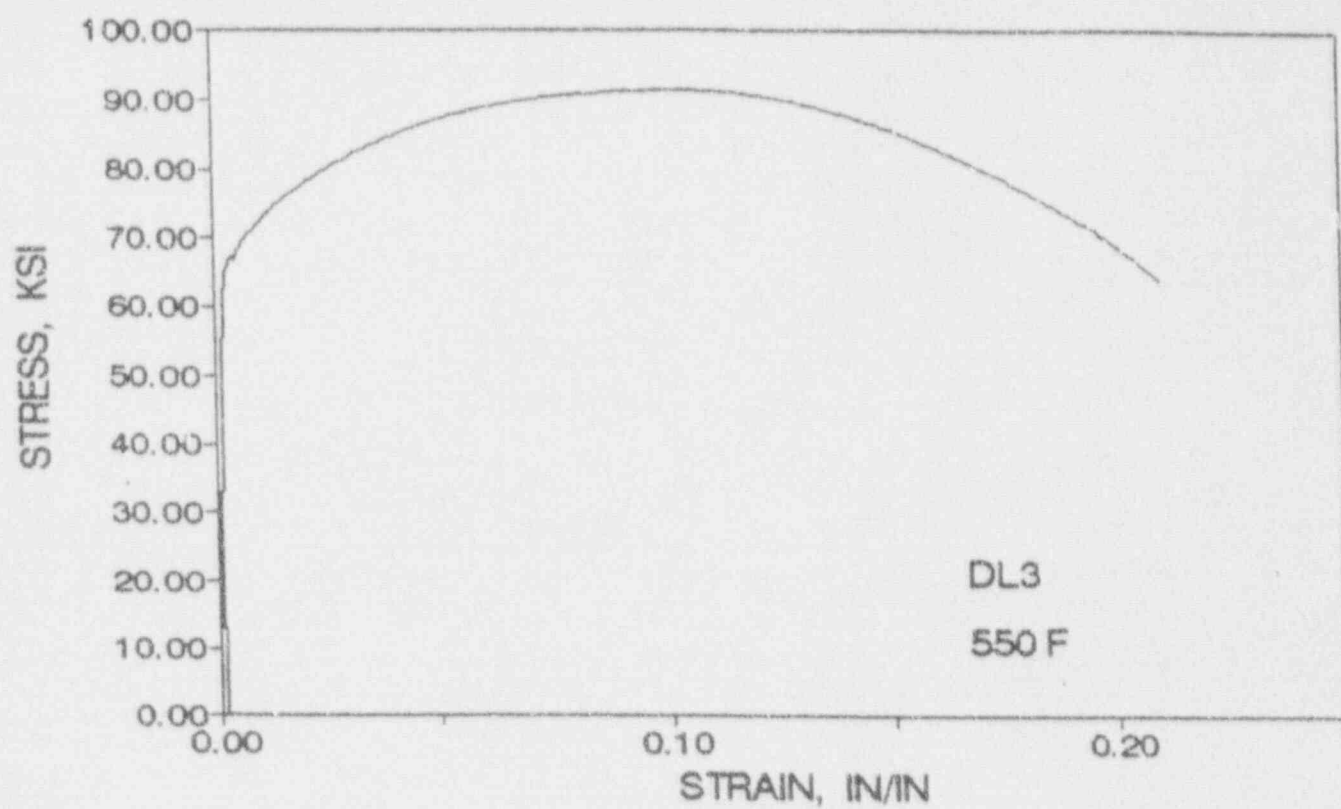


Figure 5-16. Engineering Stress-Strain Curve for Forging 05 Tensile Specimens DL3 (Tangential Orientation)

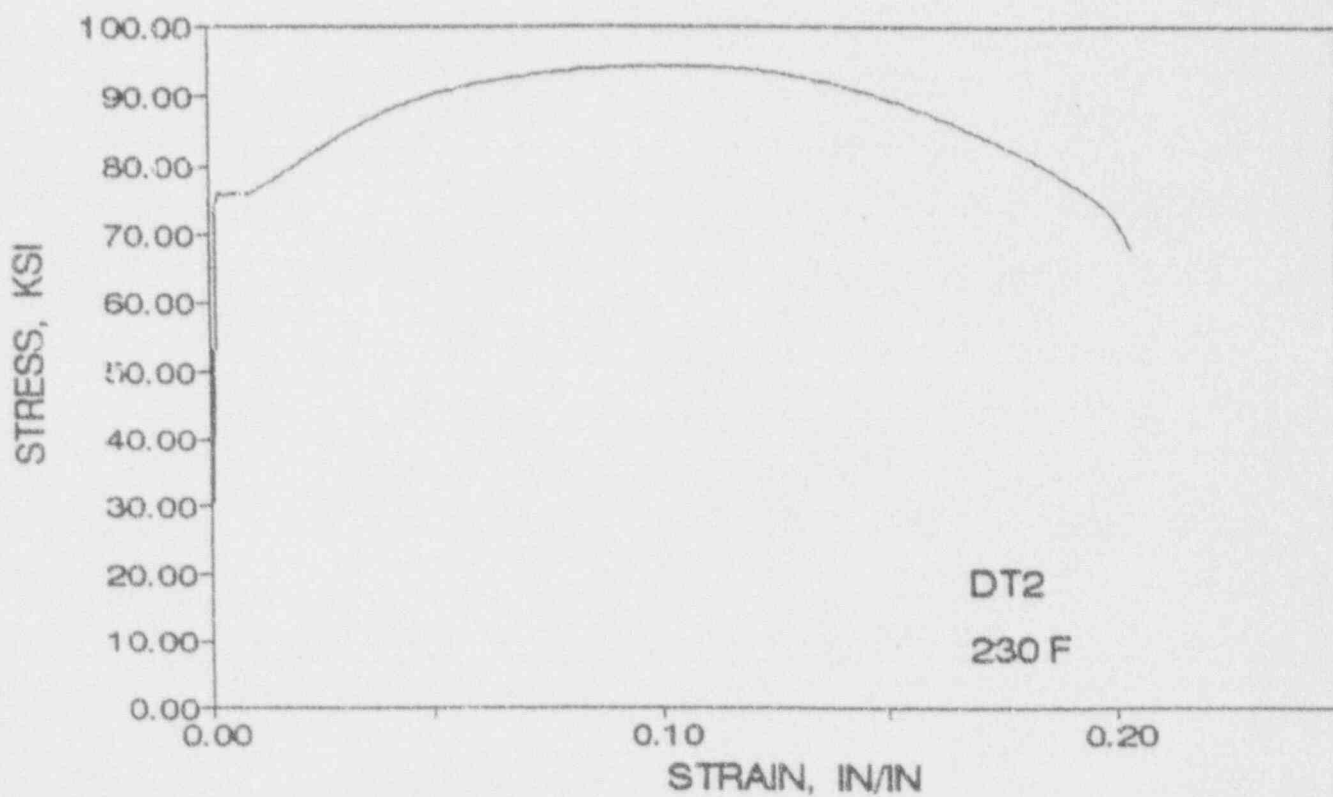
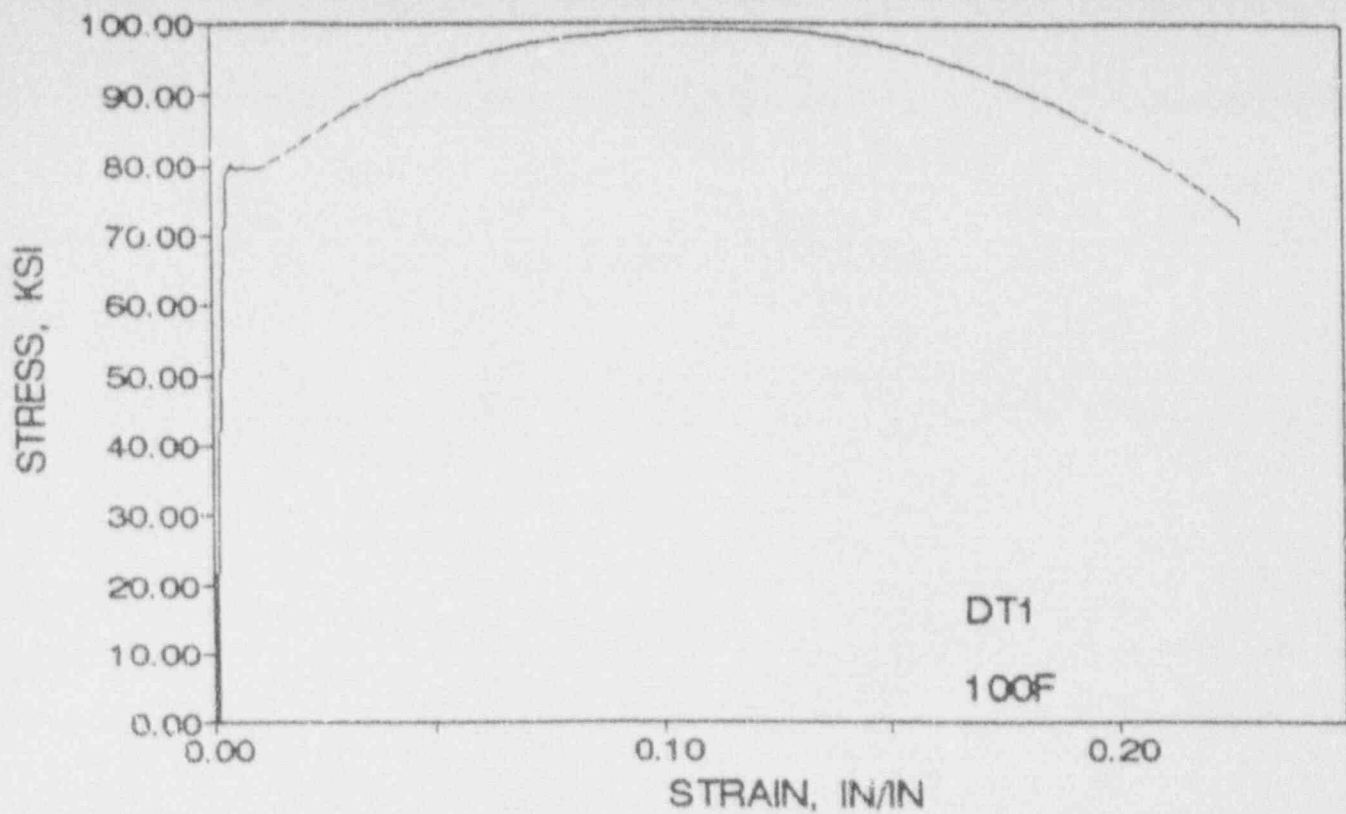


Figure 5-17. Engineering Stress-Strain Curves for Forging 05 Tensile Specimens DT1 and DT2 (Axial Orientation)

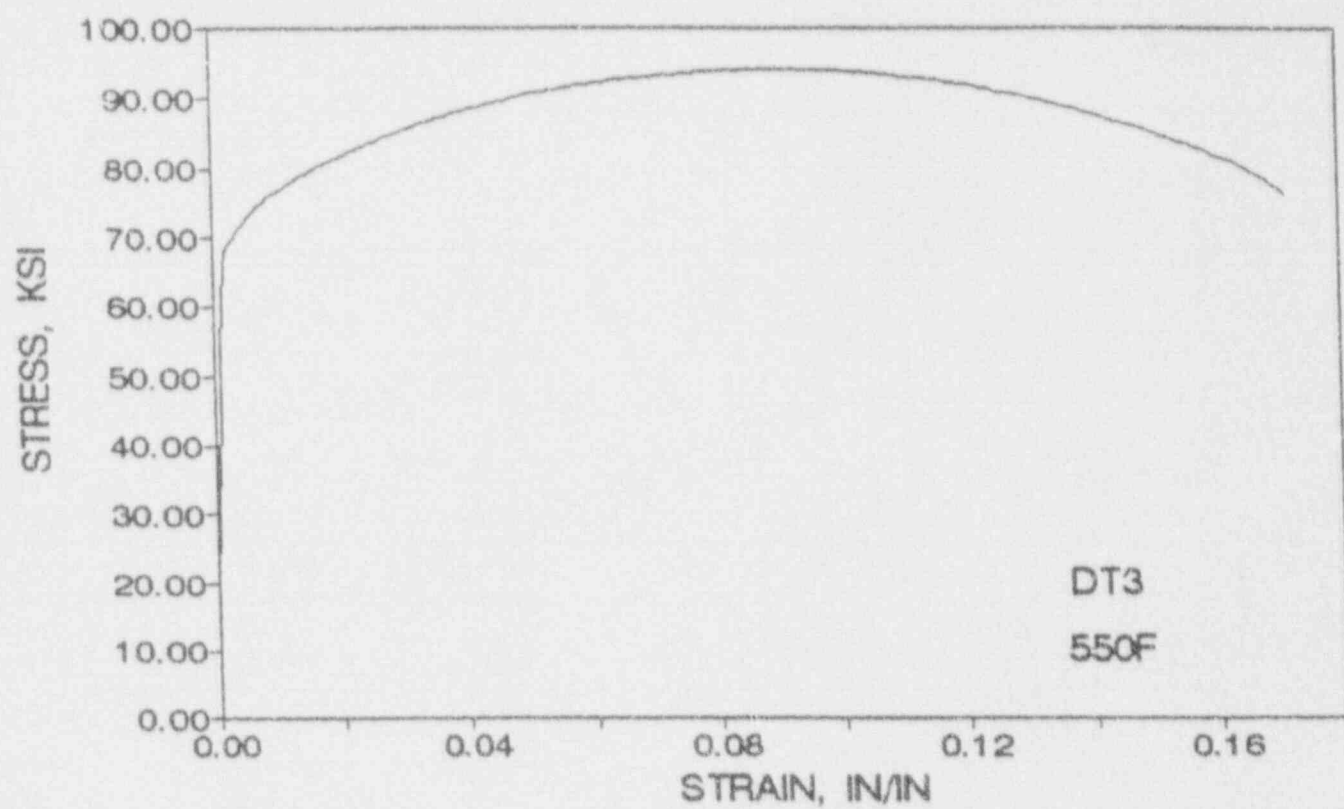


Figure 5-18. Engineering Stress-Strain Curve for Intermediate Shell Forging 05 Tensile Specimen DT3 (Axial Orientation)

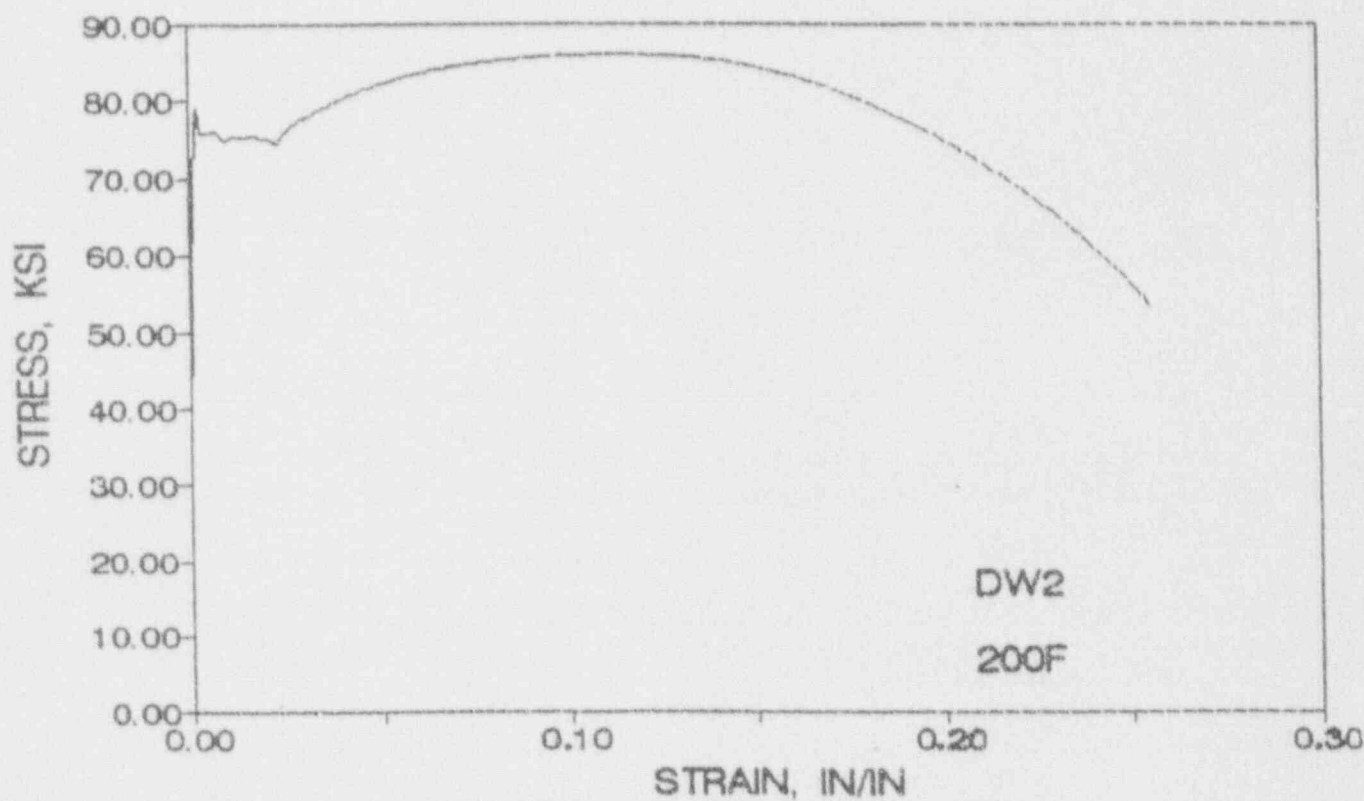
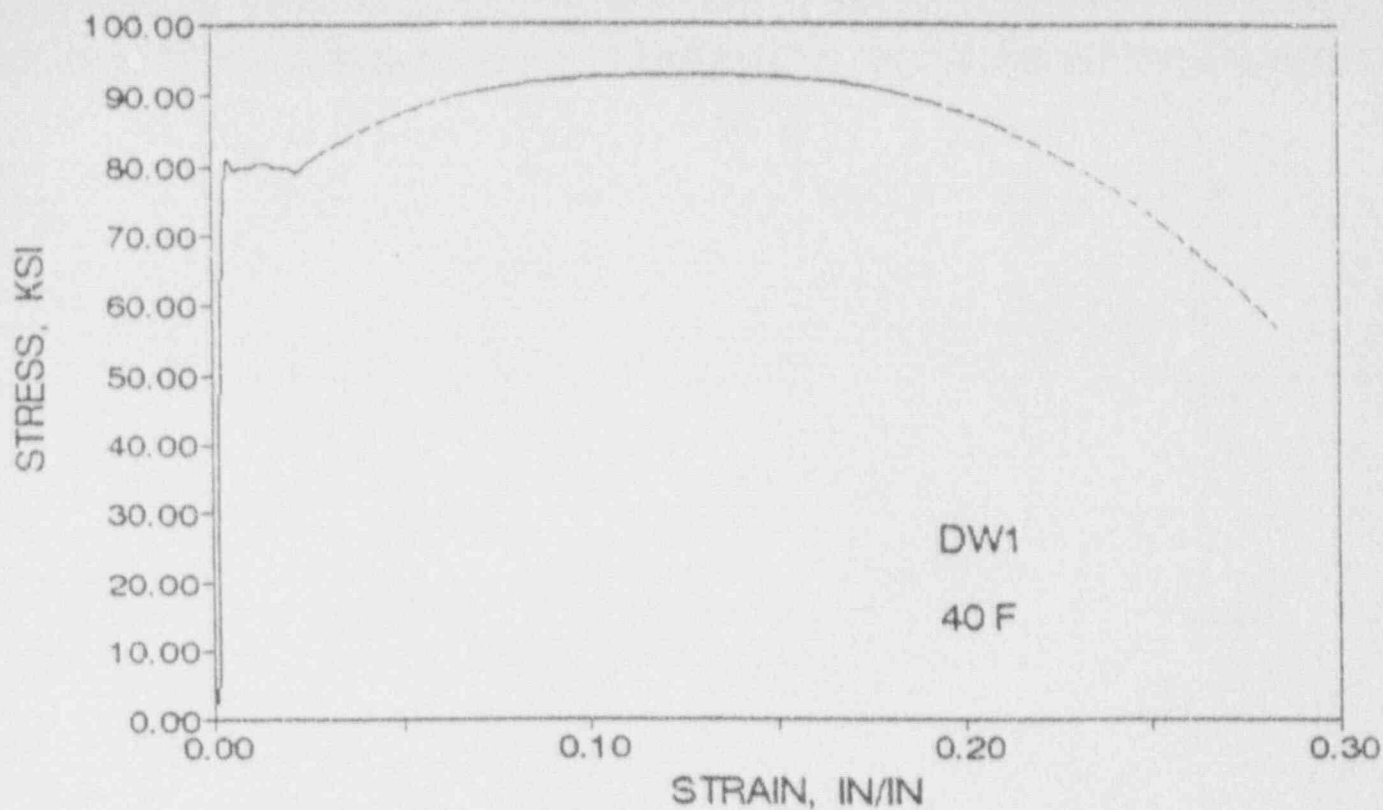


Figure 5-19. Engineering Stress-Strain Curves for Weld Metal Tensile Specimens DW1 and DW2

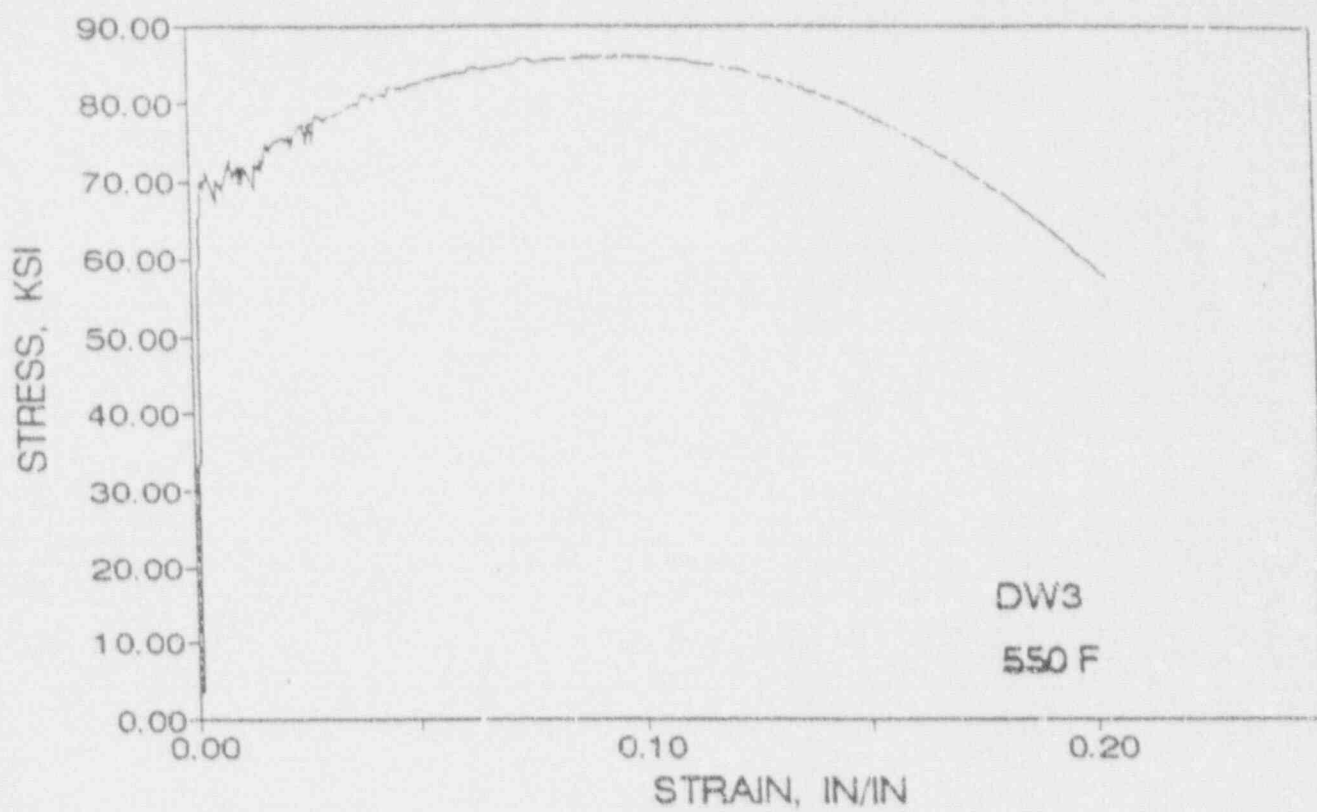


Figure 5-20. Engineering Stress-Strain Curve for Weld Metal Tensile Specimen DW3

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 surveillance capsule U, withdrawn at the end of the seventh fuel cycle. Also included are updates of dosimetry contained in capsule X, withdrawn at the end of Cycle 5, and capsule V, withdrawn at the conclusion of Cycle 1. These updates are based on current methodology and nuclear data; and together with the capsule U results provide a consistent up to date data base for use in evaluating materials properties of the W. B. McGuire Unit 2 reactor vessel.

In all cases, 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 4-1. Six irradiation capsules attached to the neutron pads are included in the reactor design to constitute the reactor vessel surveillance program. The capsules are located at azimuthal angles of 56.0, 58.5°, 124.0°, 236.0°, 238.5°, and 304.0° relative to the core cardinal axes as shown in Figure 4-1.

A plan view of a dual surveillance capsule holder attached to the neutron pad is shown in Figure 6-1. The stainless steel specimen containers are 1.182 by 1-inch and approximately 56 inches in height. The containers are positioned

axially such that the specimens are centered on the core midplane, thus spanning the central 5 feet of the 12-foot high reactor core.

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 carried out. The first, a single computation 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 ($\phi(E > 1.0 \text{ MeV})$, $\phi(E > 0.1 \text{ MeV})$, 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., $\text{dpa}/\phi(E > 1.0 \text{ MeV})$, 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 ($E > 1.0 \text{ MeV}$) 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 those 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 each cycle 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 Figures 4-1 and 6-1 was carried out in R, θ geometry using the DOT two-dimensional discrete ordinates code^[14] and the SAILOR cross-section library^[15]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotropic 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.

The reference core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 4-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was used. Since it is unlikely that a single reactor would have a power distribution at the nominal $+2\sigma$

level for a large number of fuel cycles, the use of this reference distribution is expected to yield somewhat conservative results.

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 each surveillance capsule. Again, these calculations were run in R, θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, $\phi (E > 1.0 \text{ MeV})$. 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: $R(r, \theta)$ = $\phi (E > 1.0 \text{ MeV})$ 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 analysis were based on a response function defined by the threshold neutron flux ($E > 1.0 \text{ MeV}$), prior calculations 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 $\text{dpa}/\phi (E > 1.0 \text{ MeV})$ is insensitive to changing core source distributions. In the application of these adjoint important functions to the Trojan reactor, therefore, the iron displacement rates (dpa) and the neutron flux ($E > 0.1 \text{ MeV}$) were computed on a cycle specific basis by using $\text{dpa}/\phi (E > 1.0 \text{ MeV})$ and $\phi (E > 0.1 \text{ MeV})/\phi (E > 1.0 \text{ MeV})$ ratios from the forward analysis in conjunction with the cycle specific $\phi (E > 1.0 \text{ MeV})$ solutions from the individual adjoint evaluations.

The reactor core power distribution used in the plant specific adjoint calculations was taken from the fuel cycle design reports for the first seven operating cycles of W. B. McGuire Unit 2 [16 through 22].

Selected results from the neutron transport analyses 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 the two surveillance capsule positions for both the design basis and the plant specific core power distributions. The plant specific data, based on the adjoint transport analysis, are meant to establish the absolute comparison of measurement with analysis. The design basis data derived from the forward calculation are provided as a point of reference against which plant specific fluence evaluations can be compared. Similar data is given in Table 6-2 for the pressure vessel inner radius. Again, the three pertinent exposure parameters are listed for both the design basis and the cycles 1 through 7 plant specific power distribution. 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 ($E > 1.0$ 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 45° azimuth is given by:

$$\phi_{1/4T}(45^\circ) = \phi(220.27, 45^\circ) F(225.75, 45^\circ)$$

where: $\phi_{1/4T}(45^\circ)$ = Projected neutron flux at the 1/4T position on the 45° azimuth

$\phi(220.27, 45^\circ)$ = Projected or calculated neutron flux at the vessel inner radius on the 45° azimuth.

$F(225.75, 45^\circ)$ = 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 W. B. McGuire Unit 2 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 4-2. The iron, nickel, copper, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within the capsules. The cadmium-shielded neptunium and uranium fission monitors were accommodated within the dosimeter block located near the center of the capsule.

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 [23 through 36]. Following sample preparation and weighing, the activity of each monitor was determined by means of a lithium-drifted germanium, Ge(Li), gamma spectrometer. The irradiation history of the W. B. McGuire Unit 2 reactor during cycles 1 through 7 was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable period.

The irradiation history applicable to capsules U, X, and V is given in Table 6-7. Measured and saturated reaction product specific activities as well as measured full power reaction rates are listed in Tables 6-8 through 6-10 for capsules U, X, and V, respectively. Reaction rate values were derived using the pertinent data from Tables 6-6 and 6-7.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [37]. The FERRET approach used the measured reaction rate data and the calculated neutron energy spectrum at 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 were 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 :

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

where i indexes the measured values belonging to a single data set s , g designates the energy group and α 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 log-normal 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 [38]. 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:

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

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

For the a priori calculated fluxes, a short-range correlation of $\gamma = 6$ 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^[39]. Maerker's results are closely duplicated when $\gamma = 6$. For the integral reaction rate covariances, simple normalization and random uncertainties were combined as deduced from experimental uncertainties.

Results of the FERRET evaluations of the capsules U, X, and V dosimetry are given in Table 6-11 through 6-13. The data summarized in these tables include fast neutron exposure evaluations in terms of fluence ($E > 1.0$ MeV), fluence ($E > 0.1$ MeV) and iron atom displacements (dpa). Summaries of the fit of the adjusted spectrum to the measurements for each of the three capsules are provided in Tables 6-14 through 6-16. In general, excellent results were achieved in the fits of the adjusted spectra to the individual experimental reaction rates. The adjusted spectra from the least squares evaluations are given in Tables 6-17 through 6-19 in the FERRET 53 energy group structure.

A summary of the measured and calculated neutron exposure of capsules U, X, and V is presented in Table 6-20. The agreement between calculation and measurement falls within $\pm 20\%$ for all fast neutron exposure parameters listed. The thermal neutron exposure calculated for the exposure period underpredicted the measured value by about 60 percent.

Neutron exposure projections at key locations on the pressure vessel inner radius are given in Table 6-21. Along with the current (6.05 EFPY) exposure derived from the set of capsule measurements, projections are also provided for an exposure period of 16 EFPY and to end of vessel design life (32 EFPY). Representations of the axial distribution of the maximum vessel exposure (45° azimuth) are shown in Figure 6.2 for all three exposure periods.

In computing these vessel exposures, the calculated values from Table 6-2 were scaled by the average measurement/calculation ratios observed from evaluations of dosimetry from capsules U, X, and V. The resultant best estimate exposure rates were then used to compute the integrated exposure of the vessel beltline. This procedure resulted in the following bias factors being applied to the analytical results:

Flux ($E > 1.0$ MeV)	Bias = 1.218
Flux ($E > 0.1$ MeV)	Bias = 1.208
dpa/sec	Bias = 1.168

Projections for future operation were based on the assumption that the time averaged exposure rates for the first seven cycles of operation would continue to be applicable throughout plant life.

In the calculation of exposure gradients for use in the development of heatup and cooldown curves for the W. B. McGuire Unit 2 reactor coolant system, exposure projections to 16 EFPY and 32 EFPY were also employed. Data based on both a fluence ($E > 1.0$ MeV) slope and a plant specific dpa slope through the vessel wall are provided in Table 6-21. In order to access RT_{NDT} vs. fluence trend curves, dpa equivalent fast neutron fluence levels for the 1/4T and 3/4T positions were defined by the relations

$$\phi' (1/4T) = \phi (\text{Surface}) \left(\frac{\text{dpa} (1/4T)}{\text{dpa} (\text{Surface})} \right)$$

$$\phi' (3/4T) = \phi (\text{Surface}) \left(\frac{\text{dpa} (3/4T)}{\text{dpa} (\text{Surface})} \right)$$

Using this approach results in the dpa equivalent fluence values listed in Table 6-22.

In Table 6-23 updated lead factors are listed for each of the W. B. McGuire Unit 2 surveillance capsules. These data may be used as a guide in establishing future withdrawal schedules for the remaining capsules.

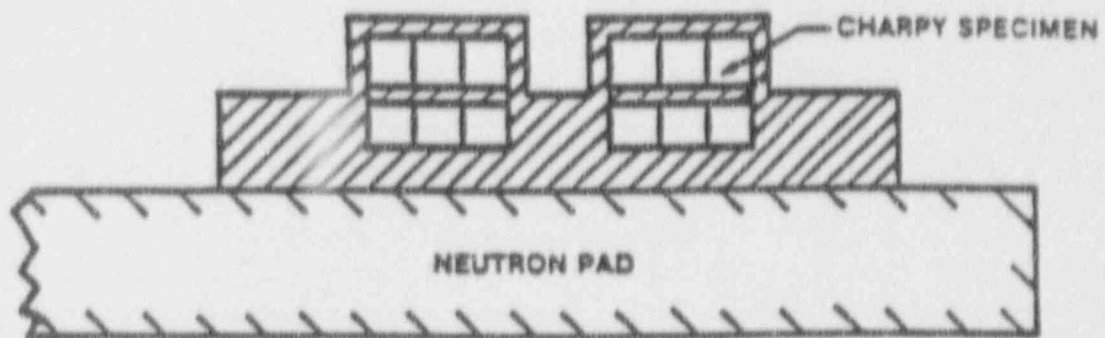


Figure 6-1. Plan View of a Dual Reactor Vessel Surveillance Capsule

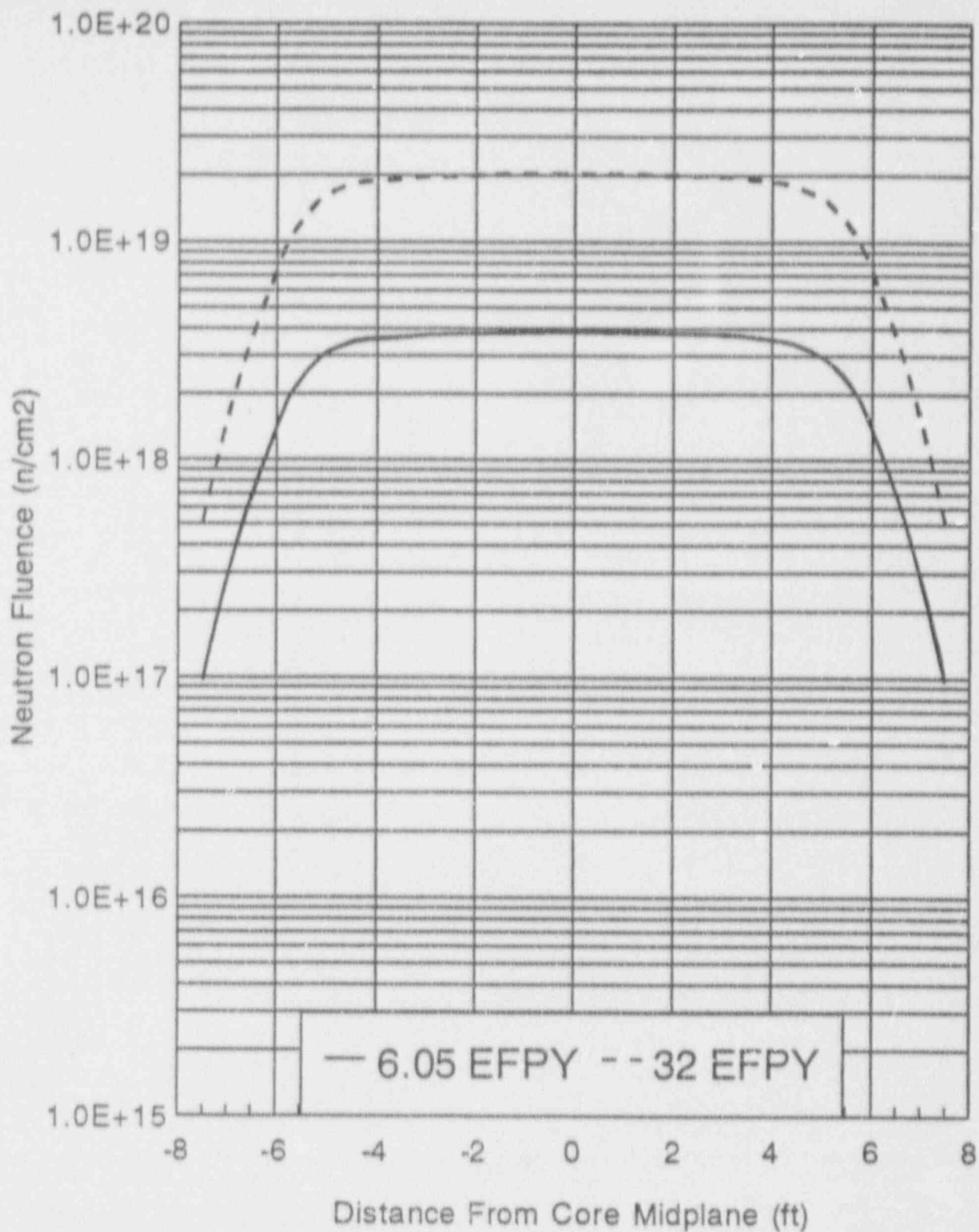


Figure 6-2 Axial Distribution of Neutron Fluence ($E > 1.0$ MeV)
Along the 45 Degree Azimuth

TABLE 6-1

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS
AT THE SURVEILLANCE CAPSULE CENTER

	$\phi(E > 1.0\text{MeV})$ [n/cm ² -sec]		$\phi(E > 0.1\text{MeV})$ [n/cm ² -sec]		Iron Displacement Rate [dpa/sec]	
	<u>31.5°</u>	<u>34.0°</u>	<u>31.5°</u>	<u>34.0°</u>	<u>31.5°</u>	<u>34.0°</u>
DESIGN BASIS	1.11×10^{11}	1.29×10^{11}	4.88×10^{11}	5.93×10^{11}	2.21×10^{-10}	2.62×10^{-10}
CYCLE 1	8.33×10^{10}	9.50×10^{10}	3.66×10^{11}	4.37×10^{11}	1.66×10^{-10}	1.93×10^{-10}
CYCLE 2	9.69×10^{10}	1.11×10^{11}	4.26×10^{11}	5.10×10^{11}	1.93×10^{-10}	2.25×10^{-10}
CYCLE 3	8.40×10^{10}	9.73×10^{10}	3.69×10^{11}	4.47×10^{11}	1.67×10^{-10}	1.97×10^{-10}
CYCLE 4	7.38×10^{10}	8.34×10^{10}	3.24×10^{11}	3.83×10^{11}	1.47×10^{-10}	1.69×10^{-10}
CYCLE 5	7.09×10^{10}	7.89×10^{10}	3.12×10^{11}	3.63×10^{11}	1.41×10^{-10}	1.60×10^{-10}
CYCLE 6	6.87×10^{10}	7.59×10^{10}	3.02×10^{11}	3.49×10^{11}	1.37×10^{-10}	1.54×10^{-10}
CYCLE 7	7.06×10^{10}	7.86×10^{10}	3.10×10^{11}	3.61×10^{11}	1.40×10^{-10}	1.59×10^{-10}

TABLE 6-2

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

DESIGN BASIS	$\phi(E > 1.0\text{MeV}) \text{ [n/cm}^2\text{-sec]}$			
	<u>0.0°</u>	<u>15.0°</u>	<u>30.0°</u>	<u>45.0°</u>
	1.45×10^{10}	2.21×10^{10}	1.69×10^{10}	2.44×10^{10}
CYCLE 1	1.08×10^{10}	1.62×10^{10}	1.27×10^{10}	1.80×10^{10}
CYCLE 2	1.27×10^{10}	1.92×10^{10}	1.48×10^{10}	2.11×10^{10}
CYCLE 3	1.03×10^{10}	1.58×10^{10}	1.28×10^{10}	1.86×10^{10}
CYCLE 4	1.06×10^{10}	1.57×10^{10}	1.14×10^{10}	1.57×10^{10}
CYCLE 5	1.02×10^{10}	1.54×10^{10}	1.10×10^{10}	1.47×10^{10}
CYCLE 6	1.11×10^{10}	1.62×10^{10}	1.08×10^{10}	1.42×10^{10}
CYCLE 7	1.04×10^{10}	1.56×10^{10}	1.10×10^{10}	1.47×10^{10}

DESIGN BASIS	$\phi(E > 0.1\text{MeV}) \text{ [n/cm}^2\text{-sec]}$			
	<u>0.0°</u>	<u>15.0°</u>	<u>30.0°</u>	<u>45.0°</u>
	3.02×10^{10}	4.66×10^{10}	4.25×10^{10}	6.11×10^{10}
CYCLE 1	2.25×10^{10}	3.42×10^{10}	3.19×10^{10}	4.51×10^{10}
CYCLE 2	2.65×10^{10}	4.05×10^{10}	3.72×10^{10}	5.28×10^{10}
CYCLE 3	2.15×10^{10}	3.33×10^{10}	3.22×10^{10}	4.66×10^{10}
CYCLE 4	2.21×10^{10}	3.31×10^{10}	2.87×10^{10}	3.93×10^{10}
CYCLE 5	2.12×10^{10}	3.25×10^{10}	2.77×10^{10}	3.68×10^{10}
CYCLE 6	2.31×10^{10}	3.42×10^{10}	2.72×10^{10}	3.56×10^{10}
CYCLE 7	2.17×10^{10}	3.29×10^{10}	2.77×10^{10}	3.68×10^{10}

TABLE 6-2 (continued)

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

DESIGN BASIS	<u>Iron Atom Displacement Rate [dpa/sec]</u>			
	<u>0.0°</u>	<u>15.0°</u>	<u>30.0°</u>	<u>45.0°</u>
	2.25×10^{-11}	3.41×10^{-11}	2.73×10^{-11}	3.88×10^{-11}
CYCLE 1	1.68×10^{-11}	2.50×10^{-11}	2.05×10^{-11}	2.86×10^{-11}
CYCLE 2	1.97×10^{-11}	2.96×10^{-11}	2.39×10^{-11}	3.35×10^{-11}
CYCLE 3	1.60×10^{-11}	2.44×10^{-11}	2.07×10^{-11}	2.96×10^{-11}
CYCLE 4	1.65×10^{-11}	2.42×10^{-11}	1.84×10^{-11}	2.50×10^{-11}
CYCLE 5	1.58×10^{-11}	2.38×10^{-11}	1.78×10^{-11}	2.34×10^{-11}
CYCLE 6	1.72×10^{-11}	2.50×10^{-11}	1.75×10^{-11}	2.26×10^{-11}
CYCLE 7	1.62×10^{-11}	2.41×10^{-11}	1.78×10^{-11}	2.34×10^{-11}

TABLE 6-3

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

Radius (cm)	0°	15°	35°	45°
220.27 ⁽¹⁾	1.00	1.00	1.00	1.00
220.64	0.979	0.979	0.980	0.979
221.66	0.891	0.891	0.893	0.889
222.99	0.771	0.769	0.773	0.766
224.31	0.655	0.652	0.658	0.648
225.63	0.552	0.549	0.555	0.543
226.95	0.463	0.459	0.467	0.452
228.28	0.387	0.383	0.390	0.376
229.60	0.322	0.318	0.326	0.311
230.92	0.268	0.263	0.271	0.257
232.25	0.222	0.218	0.225	0.211
233.57	0.183	0.180	0.187	0.174
234.89	0.151	0.148	0.155	0.142
236.22	0.125	0.121	0.128	0.116
237.54	0.102	0.0992	0.105	0.0945
238.86	0.0831	0.0807	0.0862	0.0762
240.19	0.0673	0.0650	0.0703	0.0608
241.51	0.0539	0.0512	0.0567	0.0472
242.17 ⁽²⁾	0.0508	0.0477	0.0536	0.0438

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)	0°	15°	35°	45°
220.27 ⁽¹⁾	1.00	1.00	1.00	1.00
220.64	1.00	1.00	1.00	1.00
221.66	1.00	1.00	1.00	0.995
222.99	0.974	0.966	0.982	0.956
224.31	0.928	0.915	0.938	0.902
225.63	0.875	0.859	0.886	0.843
226.95	0.819	0.802	0.832	0.782
228.28	0.762	0.743	0.777	0.722
229.60	0.705	0.686	0.721	0.663
230.92	0.649	0.529	0.665	0.605
232.25	0.594	0.575	0.611	0.549
233.57	0.540	0.522	0.558	0.495
234.89	0.488	0.470	0.506	0.443
236.22	0.436	0.421	0.455	0.392
237.54	0.386	0.373	0.406	0.343
238.86	0.337	0.326	0.358	0.296
240.19	0.290	0.280	0.310	0.248
241.51	0.244	0.232	0.261	0.201
242.17 ⁽²⁾	0.233	0.219	0.249	0.188

NOTES: 1) Base Metal Inner Radius

2) Base Metal Outer Radius

TABLE 6-5

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

Radius (cm)	0°	15°	35°	45°
220.27 ⁽¹⁾	1.00	1.00	1.00	1.00
220.64	0.982	0.982	0.986	0.984
221.66	0.911	0.910	0.923	0.915
222.99	0.813	0.812	0.837	0.821
224.31	0.721	0.718	0.751	0.730
225.63	0.637	0.633	0.673	0.646
226.95	0.562	0.558	0.602	0.572
228.28	0.496	0.491	0.539	0.505
229.60	0.438	0.433	0.481	0.447
230.92	0.387	0.381	0.430	0.394
232.25	0.341	0.335	0.383	0.347
233.57	0.300	0.295	0.341	0.305
234.89	0.263	0.258	0.302	0.266
236.22	0.230	0.225	0.267	0.231
237.54	0.199	0.195	0.234	0.199
238.86	0.171	0.168	0.203	0.169
240.19	0.145	0.142	0.174	0.140
241.51	0.121	0.117	0.146	0.113
242.17 ⁽²⁾	0.116	0.110	0.140	0.106

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	$\text{Cu}^{63}(\text{n}, \alpha)\text{Co}^{60}$	0.6917	$E > 4.7 \text{ MeV}$	5.272 yrs	
Iron	$\text{Fe}^{54}(\text{n}, \text{p})\text{Mn}^{54}$	0.0582	$E > 1.0 \text{ MeV}$	312.2 days	
Nickel	$\text{Ni}^{58}(\text{n}, \text{p})\text{Co}^{58}$	0.6830	$E > 1.0 \text{ MeV}$	70.90 days	
Uranium-238*	$\text{U}^{238}(\text{n}, \text{f})\text{Cs}^{137}$	1.0	$E > 0.4 \text{ MeV}$	30.12 yrs	5.99
Neptunium-237*	$\text{Np}^{237}(\text{n}, \text{f})\text{Cs}^{137}$	1.0	$E > 0.08 \text{ MeV}$	30.12 yrs	6.50
Cobalt-Aluminum*	$\text{Co}^{59}(\text{n}, \text{d})\text{Co}^{60}$	0.0015	$0.4 \text{ eV} > E > 0.015 \text{ MeV}$	5.272 yrs	
Cobalt-Aluminum	$\text{Co}^{59}(\text{n}, \text{d})\text{Co}^{60}$	0.0015	$E > 0.015 \text{ MeV}$	5.272 yrs	

*Denotes that monitor is cadmium shielded.

TABLE 6-7

MONTHLY THERMAL GENERATION DURING THE FIRST SEVEN FUEL CYCLES
OF THE W. B. MCGUIRE UNIT 2 REACTOR

THERMAL GENERATION		THERMAL GENERATION		THERMAL GENERATION		THERMAL GENERATION	
MONTH	(MW-hr)	MONTH	(MW-hr)	MONTH	(MW-hr)	MONTH	(MW-hr)
5/83	29804	8/85	1698280	11/87	2091145	2/90	2283648
6/83	281343	9/85	2451943	12/87	2449205	3/90	2469938
7/83	503	10/85	2342207	1/88	2375936	4/90	2446697
8/83	713998	11/85	2366518	2/88	2372171	5/90	2474847
9/83	1283803	12/85	1328243	3/88	2528325	6/90	2359097
10/83	1407029	1/86	2284248	4/88	2446326	7/90	2492907
11/83	1677739	2/86	2262594	5/88	1965065	8/90	2298172
12/83	1637288	3/86	1070655	6/88	0	9/90	5074
1/84	430628	4/86	0	7/88	182814	10/90	0
2/84	1908578	5/86	0	8/88	2312603	11/90	0
3/84	2306794	6/86	98860	9/88	2447331	12/90	55107
4/84	2301882	7/86	2419839	10/88	2518180	1/91	2425825
5/84	2131577	8/86	2295334	11/88	2395542	2/91	2231028
6/84	2382476	9/86	2455840	12/88	2528858	3/91	2538082
7/84	1558900	10/86	2268723	1/89	2475080	4/91	2444210
8/84	614384	11/86	834670	2/89	2280475	5/91	2523168
9/84	2272550	12/86	2539926	3/89	2248654	6/91	2372591
10/84	2260438	1/87	1942205	4/89	1929466	7/91	2191309
11/84	1822772	2/87	2094930	5/89	2259109	8/91	2528077
12/84	1719551	3/87	2537082	6/89	2365375	9/91	2315536
1/85	1948246	4/87	2391293	7/89	290705	10/91	2156131
2/85	0	5/87	20798	8/89	0	11/91	2185092
3/85	0	6/87	0	9/89	745948	12/91	2500590
4/85	0	7/87	1968987	10/89	2483669	1/92	628307
5/85	1613921	8/87	2304777	11/89	2400111		
6/85	2080510	9/87	2209664	12/89	2469574		
7/85	949838	10/87	2534464	1/90	2458034		

TABLE 6-8
MEASURED SENSOR ACTIVITIES AND REACTION RATES
SURVEILLANCE CAPSULE U

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top	1.54 x 10 ⁵	3.35 x 10 ⁵	5.23 x 10 ⁻¹⁷
Middle	1.63 x 10 ⁵	3.55 x 10 ⁵	
Bottom	1.56 x 10 ⁵	3.39 x 10 ⁵	
Average	1.58 x 10 ⁵	3.43 x 10 ⁵	
Fe-54(n,p) Mn-54			
Top	1.75 x 10 ⁶	3.19 x 10 ⁶	5.13 x 10 ⁻¹⁵
Middle	1.86 x 10 ⁶	3.39 x 10 ⁶	
Bottom	1.70 x 10 ⁶	3.10 x 10 ⁶	
Average	1.77 x 10 ⁶	3.22 x 10 ⁶	
Ni-58 (n,p) Co-58			
Top	1.16 x 10 ⁷	4.90 x 10 ⁷	7.09 x 10 ⁻¹⁵
Middle	1.19 x 10 ⁷	5.03 x 10 ⁷	
Bottom	1.18 x 10 ⁷	4.98 x 10 ⁷	
Average	1.18 x 10 ⁷	4.97 x 10 ⁷	
U-238 (n,f) Cs-137 (Cd)			
Middle	7.03 x 10 ⁵	5.58 x 10 ⁶	3.69 x 10 ⁻¹⁴

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

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	6.48×10^6	5.15×10^7	3.12×10^{-13}
Co-59 (n, β) Co-60			
Top	3.27×10^7	7.10×10^7	4.65×10^{-12}
Middle	3.27×10^7	7.10×10^7	
Bottom	3.10×10^7	7.19×10^7	
Average	3.28×10^7	7.13×10^7	
Co-59 (n, β) Co-60 (Cd)			
Top	2.11×10^7	4.59×10^7	2.89×10^{-12}
Middle	1.95×10^7	4.24×10^7	
Bottom	2.05×10^7	4.46×10^7	
Average	2.04×10^7	4.43×10^7	

TABLE 6-9
MEASURED SENSOR ACTIVITIES AND REACTION RATES
SURVEILLANCE CAPSULE X

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top	1.23×10^5	3.42×10^5	5.42×10^{-17}
Middle	1.32×10^5	3.67×10^5	
Bottom	1.28×10^5	3.56×10^5	
Average	1.28×10^5	3.55×10^5	
Fe-54(n,p) Mn-54			
Top	1.58×10^6	3.31×10^6	5.51×10^{-15}
Middle	1.71×10^6	3.58×10^6	
Bottom	1.67×10^6	3.49×10^6	
Average	1.65×10^6	3.46×10^6	
Ni-58 (n,p) Co-58			
Top	6.20×10^6	5.22×10^7	7.69×10^{-15}
Middle	6.55×10^6	5.52×10^7	
Bottom	6.44×10^6	5.43×10^7	
Average	6.40×10^6	5.39×10^7	
U-238 (n,f) Cs-137 (Cd)			
Middle	5.14×10^5	5.78×10^6	3.82×10^{-14}

TABLE 6-9
MEASURED SENSOR ACTIVITIES AND REACTION RATES - cont'd
SURVEILLANCE CAPSULE X

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	4.60×10^6	5.17×10^7	3.13×10^{-13}
Co-59 (n, θ) Co-60			
Top	3.10×10^7	8.61×10^7	5.44×10^{-12}
Middle	2.92×10^7	8.12×10^7	
Bottom	2.99×10^7	8.30×10^7	
Average	3.00×10^7	8.35×10^7	
Co-59 (n, θ) Co-60 (Cd)			
Top	1.78×10^7	4.95×10^7	3.11×10^{-12}
Middle	1.67×10^7	4.65×10^7	
Bottom	1.69×10^7	4.70×10^7	
Average	1.71×10^7	4.77×10^7	

TABLE 6-10
MEASURED SENSOR ACTIVITIES AND REACTION RATES
SURVEILLANCE CAPSULE V

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top	3.84×10^4	3.38×10^5	5.21×10^{-17}
Middle	4.03×10^4	3.55×10^5	
Bottom	3.76×10^4	3.31×10^5	
Average	3.88×10^4	3.42×10^5	
Fe-54(n,p) Mn-54			
Top	9.69×10^5	3.23×10^6	5.25×10^{-15}
Middle	1.01×10^6	3.37×10^6	
Bottom	9.87×10^5	3.29×10^6	
Average	9.89×10^5	3.30×10^6	
Ni-58 (n,p) Co-58			
Top	3.96×10^6	5.03×10^7	7.00×10^{-15}
Middle	3.58×10^6	4.55×10^7	
Bottom	4.04×10^6	5.13×10^7	
Average	3.86×10^6	4.90×10^7	
U-238 (n,f) Cs-137 (Cd)			
Middle	1.20×10^5	4.61×10^6	3.45×10^{-14}

TABLE 6-10
MEASURED SENSOR ACTIVITIES AND REACTION RATES - cont'd
SURVEILLANCE CAPSULE V

Monitor and Axial location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	1.20×10^6	5.24×10^7	3.17×10^{-13}
Co-59 (n, θ) Co-60			
Top	1.01×10^7	8.90×10^7	5.11×10^{-12}
Middle	8.33×10^6	7.34×10^7	
Bottom	8.86×10^6	7.81×10^7	
Average	8.89×10^6	7.83×10^7	
Co-59 (n, θ) Co-60 (Cd)			
Top	5.04×10^6	4.44×10^7	2.78×10^{-12}
Middle	4.72×10^6	4.16×10^7	
Bottom	4.74×10^6	4.16×10^7	
Average	4.83×10^6	4.26×10^7	

TABLE 6-11

SUMMARY OF NEUTRON DOSIMETRY RESULTS
SURVEILLANCE CAPSULE UTIME AVERAGED EXPOSURE RATES

ϕ (E > 1.0 MeV) (n/cm ² -sec)	1.06×10^{11}	$\pm 8\%$
ϕ (E > 0.1 MeV) (n/cm ² -sec)	4.84×10^{11}	$\pm 15\%$
dpa/sec	2.07×10^{-10}	$\pm 11\%$
ϕ (E < 0.414 eV) (n/cm ² -sec)	7.29×10^{10}	$\pm 25\%$

INTEGRATED CAPSULE EXPOSURE

Φ (E > 1.0 MeV) (n/cm ²)	2.02×10^{19}	$\pm 8\%$
Φ (E > 0.1 MeV) (n/cm ²)	9.24×10^{19}	$\pm 15\%$
dpa	3.95×10^{-2}	$\pm 11\%$
Φ (E < 0.414 eV) (n/cm ²)	1.39×10^{19}	$\pm 25\%$

NOTE: Total Irradiation Time = 6.05 EFPY

TABLE 6-12

SUMMARY OF NEUTRON DOSIMETRY RESULTS
SURVEILLANCE CAPSULE X

TIME AVERAGED EXPOSURE RATES

ϕ (E > 1.0 MeV) {n/cm ² -sec}	1.10×10^{11}	$\pm 8\%$
ϕ (E > 0.1 MeV) {n/cm ² -sec}	4.92×10^{11}	$\pm 15\%$
dpa/sec	2.13×10^{-10}	$\pm 11\%$
ϕ (E < 0.414 eV) {n/cm ² -sec}	9.55×10^{10}	$\pm 23\%$

INTEGRATED CAPSULE EXPOSURE

Φ (E > 1.0 MeV) {n/cm ² }	1.45×10^{19}	$\pm 8\%$
Φ (E > 0.1 MeV) {n/cm ² }	6.46×10^{19}	$\pm 15\%$
dpa	2.80×10^{-2}	$\pm 11\%$
Φ (E < 0.414 eV) {n/cm ² }	1.25×10^{19}	$\pm 23\%$

NOTE: Total Irradiation Time = 4.16 EFPY

TABLE 6-13

SUMMARY OF NEUTRON DOSIMETRY RESULTS
SURVEILLANCE CAPSULE V

<u>TIME AVERAGED EXPOSURE RATES</u>		
ϕ (E > 1.0 MeV) {n/cm ² -sec}	1.04×10^{11}	$\pm 8\%$
ϕ (E > 0.1 MeV) {n/cm ² -sec}	4.62×10^{11}	$\pm 15\%$
dpa/sec	2.06×10^{-10}	$\pm 11\%$
ϕ (E < 0.414 eV) {n/cm ² -sec}	9.63×10^{10}	$\pm 22\%$

<u>INTEGRATED CAPSULE EXPOSURE</u>		
Φ (E > 1.0 MeV) {n/cm ² }	3.37×10^{18}	$\pm 8\%$
Φ (E > 0.1 MeV) {n/cm ² }	1.50×10^{19}	$\pm 15\%$
dpa	6.48×10^{-3}	$\pm 11\%$
Φ (E < 0.414 eV) {n/cm ² }	3.12×10^{18}	$\pm 22\%$

NOTE: Total Irradiation Time = 1.03 EFPY

TABLE 6-14

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

<u>Reaction</u>	<u>Measured</u>	<u>Adjusted Calculation</u>	<u>C/M</u>
Cu-63 (n, α) Co-60	5.23×10^{-17}	5.20×10^{-17}	1.00
Fe-54 (n,p) Mn-54	5.13×10^{-15}	5.27×10^{-15}	1.03
Ni-58 (n,p) Co-58	7.09×10^{-15}	7.30×10^{-15}	1.03
U-238 (n,f) Cs-137 (Cd)	3.69×10^{-14}	3.23×10^{-14}	0.88
Np-237 (n,f) Cs-137 (Cd)	3.12×10^{-13}	3.28×10^{-13}	1.05
Co-59 (n, β) Co-60 (Cd)	2.89×10^{-12}	2.90×10^{-12}	0.99
Co-59 (n, β) Co-60	4.65×10^{-12}	4.62×10^{-12}	1.00

TABLE 6-15

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

<u>Reaction</u>	<u>Measured</u>	<u>Adjusted Calculation</u>	<u>C/M</u>
Cu-63 (n, α) Co-60	5.42×10^{-17}	5.42×10^{-17}	1.00
Fe-54 (n,p) Mn-54	5.51×10^{-15}	5.61×10^{-15}	1.02
Ni-58 (n,p) Co-58	7.69×10^{-15}	7.76×10^{-15}	1.01
U-238 (n,f) Cs-137 (Cd)	3.82×10^{-14}	3.38×10^{-14}	0.89
Np-237 (n,f) Cs-137 (Cd)	3.13×10^{-13}	3.34×10^{-13}	1.07
Co-59 (n, θ) Co-60 (Cd)	3.11×10^{-12}	3.12×10^{-12}	1.00
Co-59 (n, θ) Co-60	5.44×10^{-12}	5.40×10^{-12}	0.99

TABLE 6-16

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

<u>Reaction</u>	<u>Measured</u>	<u>Adjusted Calculation</u>	<u>C/M</u>
Cu-63 (n, α) Co-60	5.21×10^{-17}	5.29×10^{-17}	1.01
Fe-54 (n,p) Mn-54	5.25×10^{-15}	5.26×10^{-15}	1.00
Ni-58 (n,p) Co-58	7.00×10^{-15}	7.19×10^{-15}	1.03
U-238 (n,f) Cs-137 (Cd)	3.45×10^{-14}	3.17×10^{-14}	0.92
Np-237 (n,f) Cs-137 (Cd)	3.17×10^{-13}	3.26×10^{-13}	1.03
Co-59 (n, θ) Co-60 (Cd)	2.78×10^{-12}	2.79×10^{-12}	1.00
Co-59 (n, θ) Co-60	5.11×10^{-12}	5.07×10^{-12}	0.99

TABLE 6-17
ADJUSTED NEUTRON ENERGY SPECTRUM AT
CENTER OF SURVEILLANCE CAPSULE U

Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)	Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)
1	1.73x10 ¹	4.36x10 ⁶	28	9.12x10 ⁻³	2.18x10 ¹⁰
2	1.49x10 ¹	1.06x10 ⁷	29	5.53x10 ⁻³	2.82x10 ¹⁰
3	1.35x10 ¹	4.89x10 ⁷	30	3.36x10 ⁻³	8.89x10 ⁹
4	1.16x10 ¹	1.24x10 ⁸	31	2.84x10 ⁻³	8.54x10 ⁹
5	1.00x10 ¹	2.99x10 ⁸	32	2.40x10 ⁻³	8.27x10 ⁹
6	8.61x10 ⁰	5.40x10 ⁸	33	2.04x10 ⁻³	2.33x10 ¹⁰
7	7.41x10 ⁰	1.29x10 ⁹	34	1.23x10 ⁻³	2.13x10 ¹⁰
8	6.07x10 ⁰	1.87x10 ⁹	35	7.49x10 ⁻⁴	1.97x10 ¹⁰
9	4.97x10 ⁰	3.98x10 ⁹	36	4.54x10 ⁻⁴	1.88x10 ¹⁰
10	3.68x10 ⁰	5.36x10 ⁹	37	2.75x10 ⁻⁴	2.02x10 ¹⁰
11	2.87x10 ⁰	1.14x10 ¹⁰	38	1.67x10 ⁻⁴	2.19x10 ¹⁰
12	2.23x10 ⁰	1.60x10 ¹⁰	39	1.01x10 ⁻⁴	2.17x10 ¹⁰
13	1.74x10 ⁰	2.29x10 ¹⁰	40	6.14x10 ⁻⁵	2.14x10 ¹⁰
14	1.35x10 ⁰	2.57x10 ¹⁰	41	3.73x10 ⁻⁵	2.09x10 ¹⁰
15	1.11x10 ⁰	4.77x10 ¹⁰	42	2.26x10 ⁻⁵	2.01x10 ¹⁰
16	8.21x10 ⁻¹	5.52x10 ¹⁰	43	1.37x10 ⁻⁵	1.95x10 ¹⁰
17	6.39x10 ⁻¹	5.79x10 ¹⁰	44	8.32x10 ⁻⁶	1.84x10 ¹⁰
18	4.98x10 ⁻¹	4.22x10 ¹⁰	45	5.04x10 ⁻⁶	1.68x10 ¹⁰
19	3.88x10 ⁻¹	6.07x10 ¹⁰	46	3.06x10 ⁻⁶	1.55x10 ¹⁰
20	3.02x10 ⁻¹	6.03x10 ¹⁰	47	1.86x10 ⁻⁶	1.41x10 ¹⁰
21	1.83x10 ⁻¹	6.03x10 ¹⁰	48	1.13x10 ⁻⁶	1.06x10 ¹⁰
22	1.11x10 ⁻¹	4.81x10 ¹⁰	49	6.83x10 ⁻⁷	1.24x10 ¹⁰
23	6.74x10 ⁻²	3.30x10 ¹⁰	50	4.14x10 ⁻⁷	1.49x10 ¹⁰
24	4.09x10 ⁻²	1.86x10 ¹⁰	51	2.51x10 ⁻⁷	1.39x10 ¹⁰
25	2.55x10 ⁻²	2.53x10 ¹⁰	52	1.52x10 ⁻⁷	1.24x10 ¹⁰
26	1.99x10 ⁻²	1.19x10 ¹⁰	53	9.24x10 ⁻⁸	3.18x10 ¹⁰
27	1.50x10 ⁻²	1.49x10 ¹⁰			

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

TABLE 6-18
ADJUSTED NEUTRON ENERGY SPECTRUM AT
CENTER OF SURVEILLANCE CAPSULE X

Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)	Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)
1	1.73x10 ¹	4.50x10 ⁶	28	9.12x10 ⁻³	2.27x10 ¹⁰
2	1.49x10 ¹	1.01x10 ⁷	29	5.53x10 ⁻³	2.95x10 ¹⁰
3	1.35x10 ¹	5.07x10 ⁷	30	3.36x10 ⁻³	9.32x10 ⁹
4	1.16x10 ¹	1.30x10 ⁸	31	2.84x10 ⁻³	8.97x10 ⁹
5	1.00x10 ¹	3.13x10 ⁸	32	2.40x10 ⁻³	8.70x10 ⁹
6	8.61x10 ⁰	5.68x10 ⁸	33	2.04x10 ⁻³	2.46x10 ¹⁰
7	7.41x10 ⁰	1.37x10 ⁹	34	1.23x10 ⁻³	2.25x10 ¹⁰
8	6.07x10 ⁰	2.00x10 ⁹	35	7.49x10 ⁻⁴	2.09x10 ¹⁰
9	4.97x10 ⁰	4.26x10 ⁹	36	4.54x10 ⁻⁴	1.99x10 ¹⁰
10	3.68x10 ⁰	3.72x10 ⁹	37	2.75x10 ⁻⁴	2.14x10 ¹⁰
11	2.87x10 ⁰	1.21x10 ¹⁰	38	1.67x10 ⁻⁴	2.36x10 ¹⁰
12	2.23x10 ⁰	1.68x10 ¹⁰	39	1.01x10 ⁻⁴	2.31x10 ¹⁰
13	1.74x10 ⁰	2.37x10 ¹⁰	40	6.14x10 ⁻⁵	2.28x10 ¹⁰
14	1.35x10 ⁰	2.64x10 ¹⁰	41	3.73x10 ⁻⁵	2.22x10 ¹⁰
15	1.11x10 ⁰	4.87x10 ¹⁰	42	2.26x10 ⁻⁵	2.15x10 ¹⁰
16	8.21x10 ⁻¹	5.59x10 ¹⁰	43	1.37x10 ⁻⁵	2.08x10 ¹⁰
17	6.39x10 ⁻¹	5.84x10 ¹⁰	44	8.32x10 ⁻⁶	1.97x10 ¹⁰
18	4.98x10 ⁻¹	4.25x10 ¹⁰	45	5.04x10 ⁻⁶	1.80x10 ¹⁰
19	3.88x10 ⁻¹	6.10x10 ¹⁰	46	3.06x10 ⁻⁶	1.66x10 ¹⁰
20	3.02x10 ⁻¹	6.07x10 ¹⁰	47	1.86x10 ⁻⁶	1.51x10 ¹⁰
21	1.83x10 ⁻¹	6.09x10 ¹⁰	48	1.13x10 ⁻⁶	1.14x10 ¹⁰
22	1.11x10 ⁻¹	4.88x10 ¹⁰	49	6.83x10 ⁻⁷	1.39x10 ¹⁰
23	6.74x10 ⁻²	3.37x10 ¹⁰	50	4.14x10 ⁻⁷	1.74x10 ¹⁰
24	4.09x10 ⁻²	1.90x10 ¹⁰	51	2.51x10 ⁻⁷	1.70x10 ¹⁰
25	2.55x10 ⁻²	2.60x10 ¹⁰	52	1.52x10 ⁻⁷	1.58x10 ¹⁰
26	1.99x10 ⁻²	1.23x10 ¹⁰	53	9.24x10 ⁻⁸	4.54x10 ¹⁰
27	1.50x10 ⁻²	1.55x10 ¹⁰			

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

TABLE 6-19
ADJUSTED NEUTRON ENERGY SPECTRUM AT
CENTER OF SURVEILLANCE CAPSULE V

Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)	Group	Energy (Mev)	Adjusted Flux (n/cm ² -sec)
1	1.73×10^1	7.53×10^6	28	9.12×10^{-3}	2.03×10^{10}
2	1.49×10^1	1.69×10^7	29	5.53×10^{-3}	2.62×10^{10}
3	1.35×10^1	6.49×10^7	30	3.36×10^{-3}	8.20×10^9
4	1.16×10^1	1.44×10^8	31	2.84×10^{-3}	7.85×10^9
5	1.00×10^1	3.18×10^8	32	2.40×10^{-3}	7.58×10^9
6	8.61×10^0	5.45×10^8	33	2.04×10^{-3}	2.14×10^{10}
7	7.41×10^0	1.26×10^9	34	1.23×10^{-3}	1.98×10^{10}
8	6.07×10^0	1.81×10^9	35	7.49×10^{-4}	1.85×10^{10}
9	4.97×10^0	3.88×10^9	36	4.54×10^{-4}	1.77×10^{10}
10	3.68×10^0	5.23×10^9	37	2.75×10^{-4}	1.91×10^{10}
11	2.87×10^0	1.12×10^{10}	38	1.67×10^{-4}	2.10×10^{10}
12	2.23×10^0	1.59×10^{10}	39	1.01×10^{-4}	2.07×10^{10}
13	1.74×10^0	2.26×10^{10}	40	6.14×10^{-5}	2.05×10^{10}
14	1.35×10^0	2.54×10^{10}	41	3.73×10^{-5}	1.99×10^{10}
15	1.11×10^0	4.67×10^{10}	42	2.26×10^{-5}	1.92×10^{10}
16	8.21×10^{-1}	5.34×10^{10}	43	1.37×10^{-5}	1.86×10^{10}
17	6.39×10^{-1}	5.54×10^{10}	44	8.32×10^{-6}	1.77×10^{10}
18	4.98×10^{-1}	4.00×10^{10}	45	5.04×10^{-6}	1.62×10^{10}
19	3.88×10^{-1}	5.59×10^{10}	46	3.06×10^{-6}	1.51×10^{10}
20	3.02×10^{-1}	5.70×10^{10}	47	1.86×10^{-6}	1.39×10^{10}
21	1.83×10^{-1}	5.61×10^{10}	48	1.13×10^{-6}	1.03×10^{10}
22	1.11×10^{-1}	4.45×10^{10}	49	6.83×10^{-7}	1.31×10^{10}
23	6.74×10^{-2}	3.07×10^{10}	50	4.14×10^{-7}	1.72×10^{10}
24	4.09×10^{-2}	1.73×10^{10}	51	2.51×10^{-7}	1.69×10^{10}
25	2.55×10^{-2}	2.26×10^{10}	52	1.52×10^{-7}	1.60×10^{10}
26	1.99×10^{-2}	1.11×10^{10}	53	9.24×10^{-8}	4.63×10^{10}
27	1.50×10^{-2}	1.41×10^{10}			

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

TABLE 6-20

COMPARISON OF CALCULATED AND MEASURED EXPOSURE LEVELS
FOR W. B. MCGUIRE UNIT 2 SURVEILLANCE CAPSULES

	<u>CAPSULE U</u>		
	<u>Calculated</u>	<u>Measured</u>	<u>C/M</u>
$\Phi(E > 1.0 \text{ MeV}) \text{ (n/cm}^2\text{)}$	1.67×10^{19}	2.02×10^{19}	0.83
$\Phi(E > 0.1 \text{ MeV}) \text{ (n/cm}^2\text{)}$	7.67×10^{19}	9.24×10^{19}	0.83
dpa	3.39×10^{-2}	3.95×10^{-2}	0.86
$\Phi(E < 0.414 \text{ eV}) \text{ (n/cm}^2\text{)}$	6.54×10^{18}	1.39×10^{19}	0.47

	<u>CAPSULE X</u>		
	<u>Calculated</u>	<u>Measured</u>	<u>C/M</u>
$\Phi(E > 1.0 \text{ MeV}) \text{ (n/cm}^2\text{)}$	1.21×10^{19}	1.45×10^{19}	0.83
$\Phi(E > 0.1 \text{ MeV}) \text{ (n/cm}^2\text{)}$	5.56×10^{19}	6.46×10^{19}	0.86
dpa	2.46×10^{-2}	2.80×10^{-2}	0.88
$\Phi(E < 0.414 \text{ eV}) \text{ (n/cm}^2\text{)}$	4.74×10^{18}	1.25×10^{19}	0.38

TABLE 6-20 (Continued)

COMPARISON OF CALCULATED AND MEASURED EXPOSURE LEVELS
FOR W. B. MCGUIRE UNIT 2 SURVEILLANCE CAPSULES

	<u>CAPSULE V</u>		<u>C/M</u>
	<u>Calculated</u>	<u>Measured</u>	
$\Phi(E > 1.0 \text{ MeV}) \{n/cm^2\}$	2.70×10^{18}	3.37×10^{18}	0.80
$\Phi(E > 0.1 \text{ MeV}) \{n/cm^2\}$	1.19×10^{19}	1.50×10^{19}	0.79
dpa	5.39×10^{-3}	6.48×10^{-3}	0.83
$\Phi(E < 0.414 \text{ eV}) \{n/cm^2\}$	1.29×10^{18}	3.12×10^{18}	0.41

TABLE 6-21
NEUTRON EXPOSURE PROJECTIONS AT KEY LOCATIONS
ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE
AZIMUTHAL ANGLE

	<u>0°</u>	<u>15°</u>	<u>30°</u>	<u>45°</u>
<u>6.05 EFY</u>				
$\Phi(E>1.0 \text{ MeV})$ (n/cm ²)	2.52×10^{18}	3.76×10^{18}	2.79×10^{18}	3.85×10^{18}
$\Phi(E>0.1 \text{ MeV})$ (n/cm ²)	5.21×10^{18}	7.86×10^{18}	6.96×10^{18}	9.56×10^{18}
dpa	3.75×10^{-3}	5.56×10^{-3}	4.31×10^{-3}	5.86×10^{-3}
<u>16.0 EFY</u>				
$\Phi(E>1.0 \text{ MeV})$ (n/cm ²)	6.66×10^{18}	9.94×10^{18}	7.38×10^{18}	1.02×10^{19}
$\Phi(E>0.1 \text{ MeV})$ (n/cm ²)	1.38×10^{19}	2.08×10^{19}	1.84×10^{19}	2.53×10^{19}
dpa	9.92×10^{-3}	1.47×10^{-2}	1.14×10^{-2}	1.55×10^{-2}
<u>32.0 EFY</u>				
$\Phi(E>1.0 \text{ MeV})$ (n/cm ²)	1.33×10^{19}	1.99×10^{19}	1.48×10^{19}	2.04×10^{19}
$\Phi(E>0.1 \text{ MeV})$ (n/cm ²)	2.76×10^{19}	4.16×10^{19}	3.68×10^{19}	5.06×10^{19}
dpa	1.98×10^{-2}	2.94×10^{-2}	2.28×10^{-2}	3.10×10^{-2}

TABLE 6-22
PROJECTED FAST NEUTRON EXPOSURE VALUES

<u>16 EFY</u>						
	<u>NEUTRON FLUENCE (E > 1.0 MeV) SLOPE</u>			<u>dpa SLOPE</u>		
	(n/cm ²)			(equivalent n/cm ²)		
	<u>Surface</u>	<u>1/4 T</u>	<u>3/4 T</u>	<u>Surface</u>	<u>1/4 T</u>	<u>3/4 T</u>
0°	6.66 x 10 ¹⁸	3.62 x 10 ¹⁸	7.79 x 10 ¹⁷	6.66 x 10 ¹⁸	4.20 x 10 ¹⁸	1.46 x 10 ¹⁸
15°	9.94 x 10 ¹⁸	5.38 x 10 ¹⁸	1.12 x 10 ¹⁸	9.94 x 10 ¹⁸	6.22 x 10 ¹⁸	2.13 x 10 ¹⁸
30°	7.38 x 10 ¹⁸	4.04 x 10 ¹⁸	8.86 x 10 ¹⁷	7.38 x 10 ¹⁸	4.92 x 10 ¹⁸	1.88 x 10 ¹⁸
45° (a)	1.02 x 10 ¹⁹	5.46 x 10 ¹⁸	1.10 x 10 ¹⁸	1.02 x 10 ¹⁹	6.52 x 10 ¹⁸	2.23 x 10 ¹⁸

<u>32 EFY</u>						
	<u>NEUTRON FLUENCE (E > 1.0 MeV) SLOPE</u>			<u>dpa SLOPE</u>		
	(n/cm ²)			(equivalent n/cm ²)		
	<u>Surface</u>	<u>1/4 T</u>	<u>3/4 T</u>	<u>Surface</u>	<u>1/4 T</u>	<u>3/4 T</u>
0°	1.33 x 10 ¹⁹	7.24 x 10 ¹⁸	1.56 x 10 ¹⁸	1.33 x 10 ¹⁹	8.38 x 10 ¹⁸	2.91 x 10 ¹⁸
15°	1.99 x 10 ¹⁹	1.08 x 10 ¹⁹	2.25 x 10 ¹⁸	1.99 x 10 ¹⁹	1.25 x 10 ¹⁹	4.26 x 10 ¹⁸
30°	1.48 x 10 ¹⁹	8.10 x 10 ¹⁸	1.78 x 10 ¹⁸	1.48 x 10 ¹⁹	9.87 x 10 ¹⁸	3.77 x 10 ¹⁸
45° (a)	2.04 x 10 ¹⁹	1.09 x 10 ¹⁹	2.20 x 10 ¹⁸	2.04 x 10 ¹⁹	1.30 x 10 ¹⁹	4.47 x 10 ¹⁸

(a) Maximum point on the pressure vessel

TABLE 6-23

UPDATED LEAD FACTORS FOR MCGUIRE UNIT 2 SURVEILLANCE CAPSULES

<u>Capsule</u>	<u>Lead Factor</u>
U	5.28(a)
X	5.28(a)
W	5.28
Z	5.28
V	4.62(a)
Y	4.68

(a) Plant specific evaluation

SECTION 7.0
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following removal schedule meets ASTM E185-82 and is recommended for future capsules to be removed from the McGuire Unit 2 reactor vessel:

Capsule	Location (deg.)	Capsule Lead Factor	Removal Time (a)	Estimated Fluence (n/cm ²)
V	58.5	4.62	1.03 (Removed)	3.06 x 10 ¹⁸ (Actual)
X	236.0	5.28	4.16 (Removed)	1.45 x 10 ¹⁹ (Actual)
U	56.0	5.28	6.05 (Removed)	2.02 x 10 ¹⁹ (Actual)
W	124.0	5.28	9	3.03 x 10 ¹⁹
Z	304.0	5.28	Standby	---
Y	238.5	4.68	Standby	---

(a) Effective Full Power Years (EFPY) from plant startup.

SECTION 8.0

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APPENDIX A

Load-Time Records for Charpy Specimen Tests

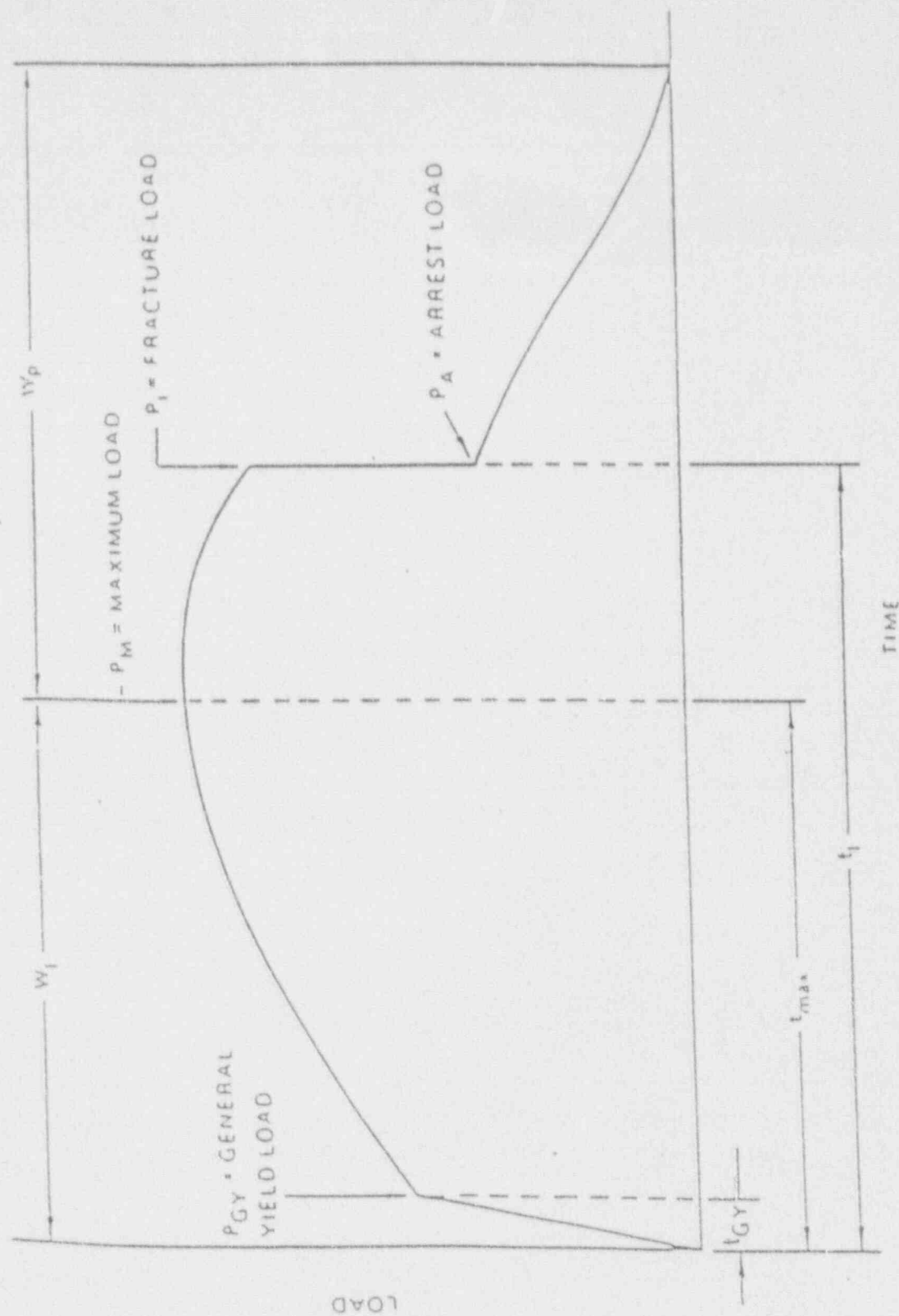


Figure A-1. Idealized load-time record

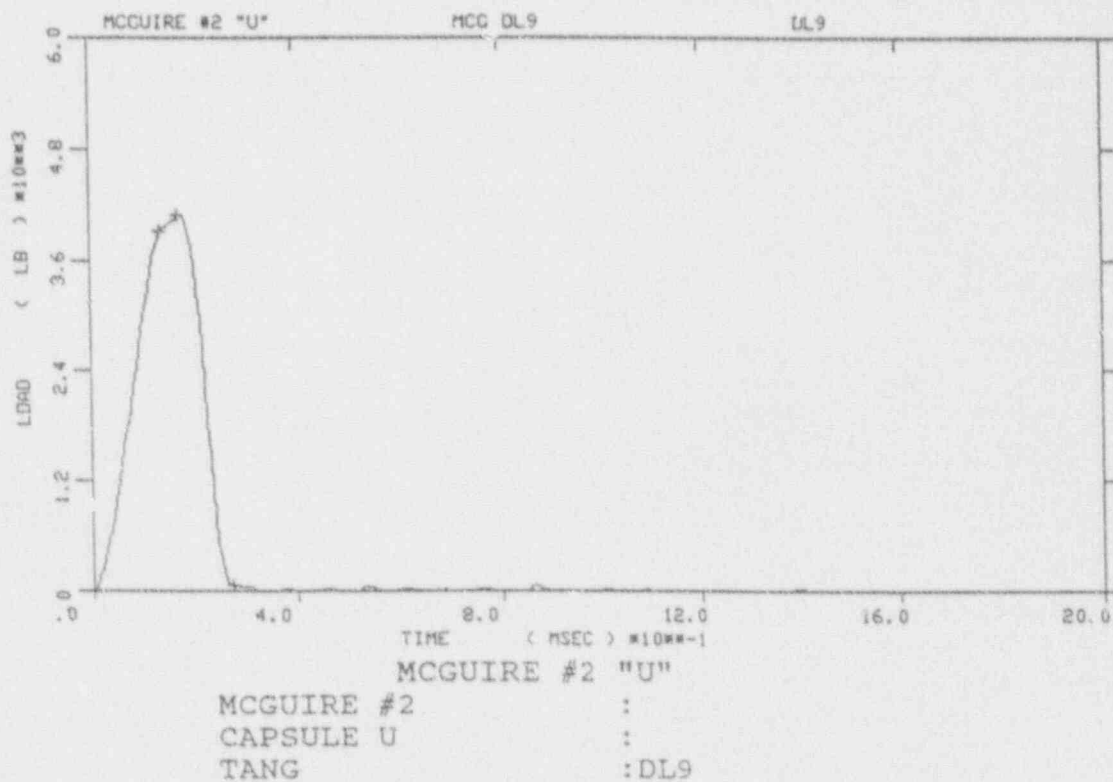
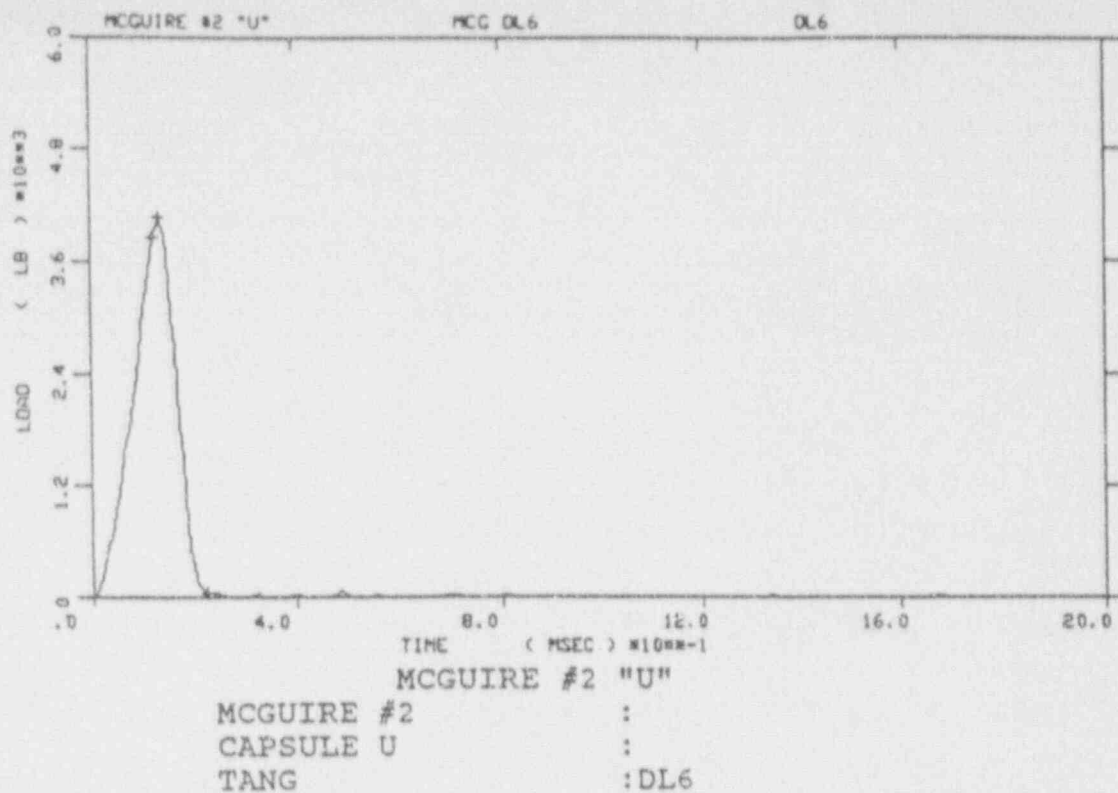


Figure A-7. Load-time records for Specimens DL6 and DL9

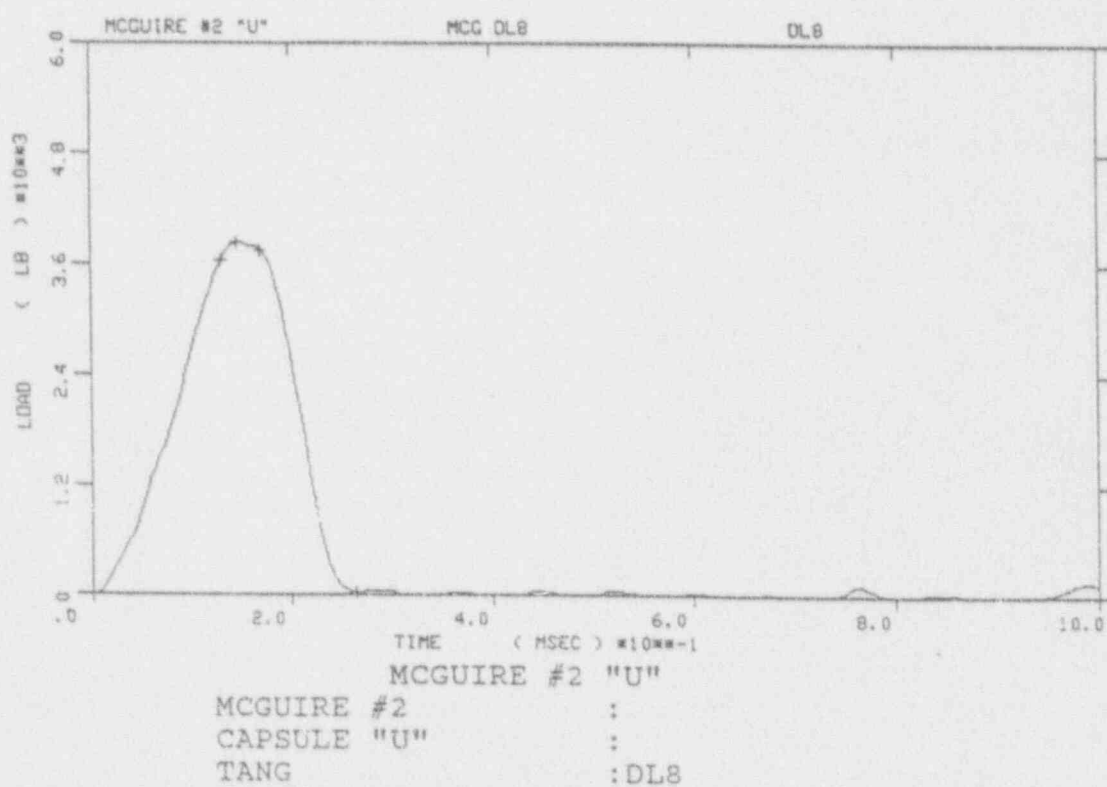
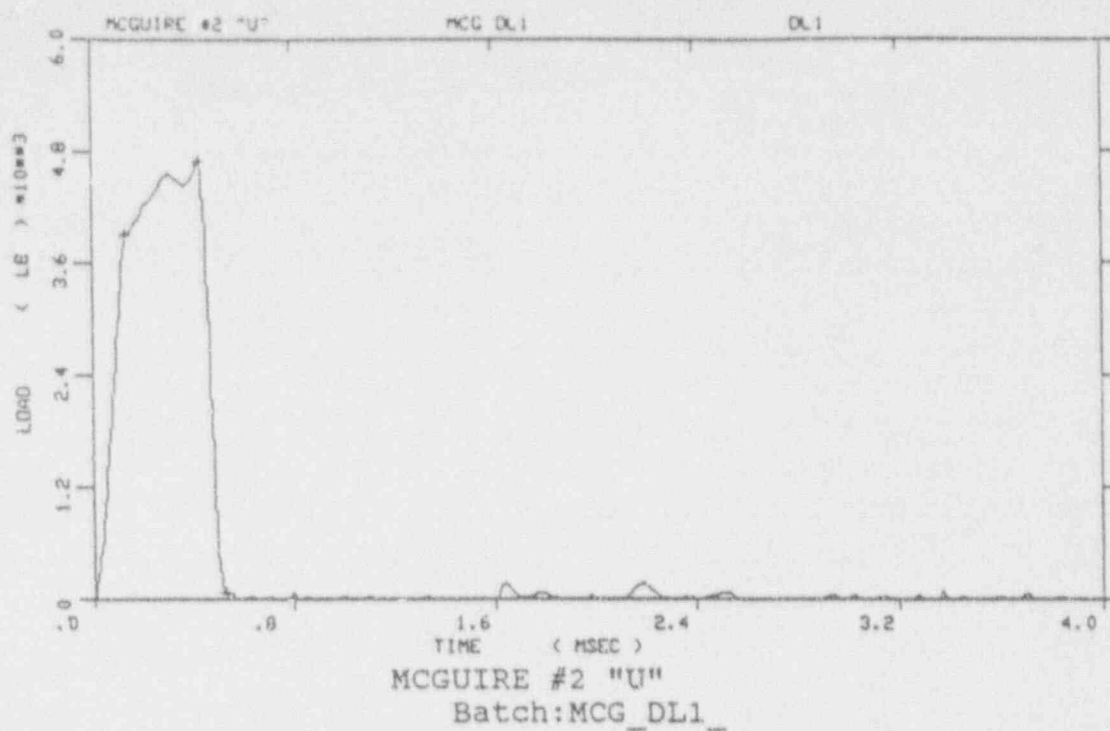


Figure A-3. Load-time records for Specimens DL1 and DL8

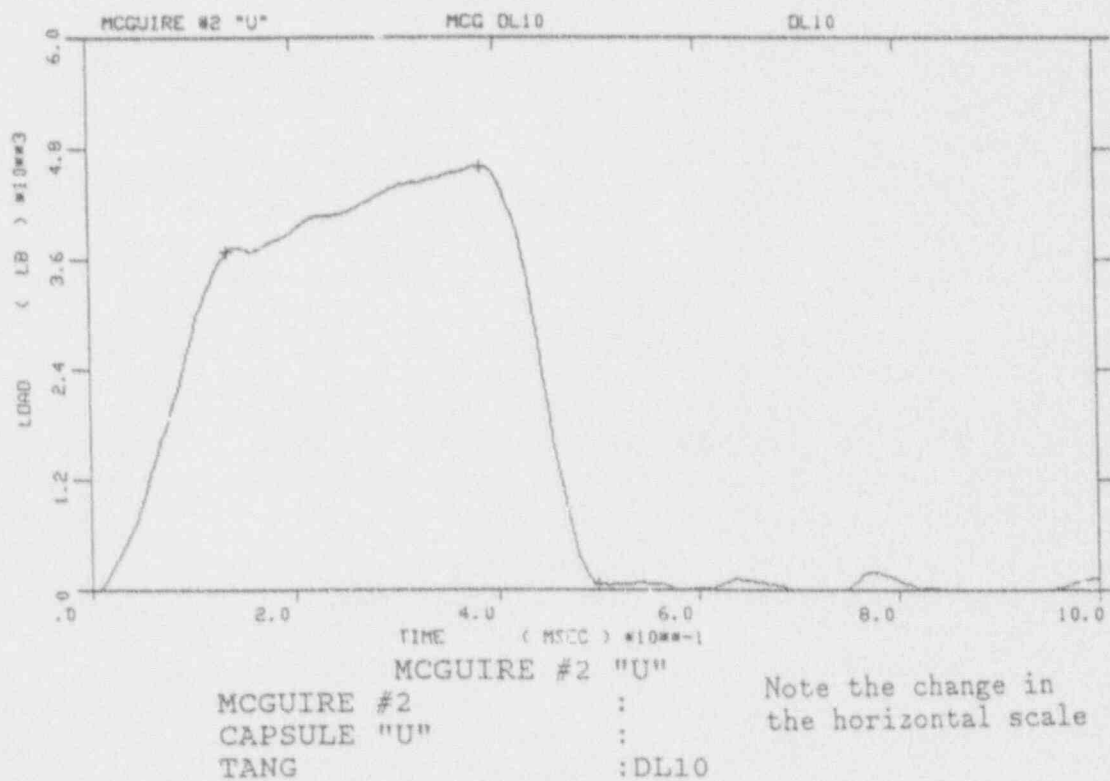
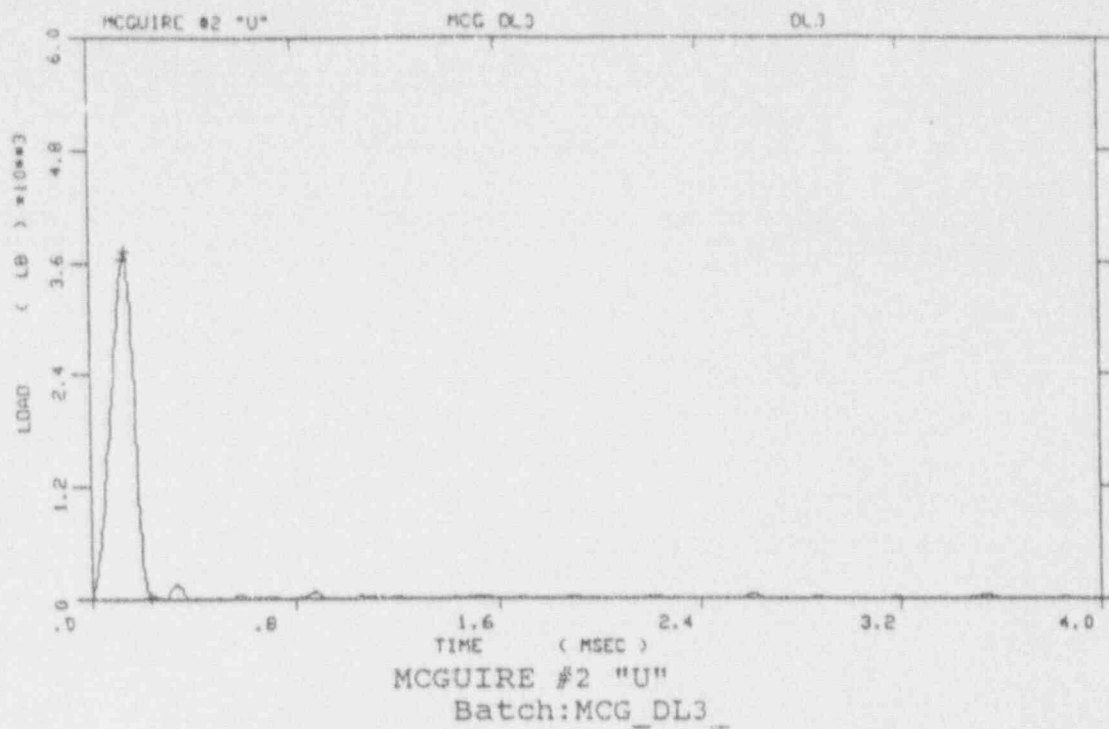


Figure A-4. Load-time records for Specimens DL3 and DL10

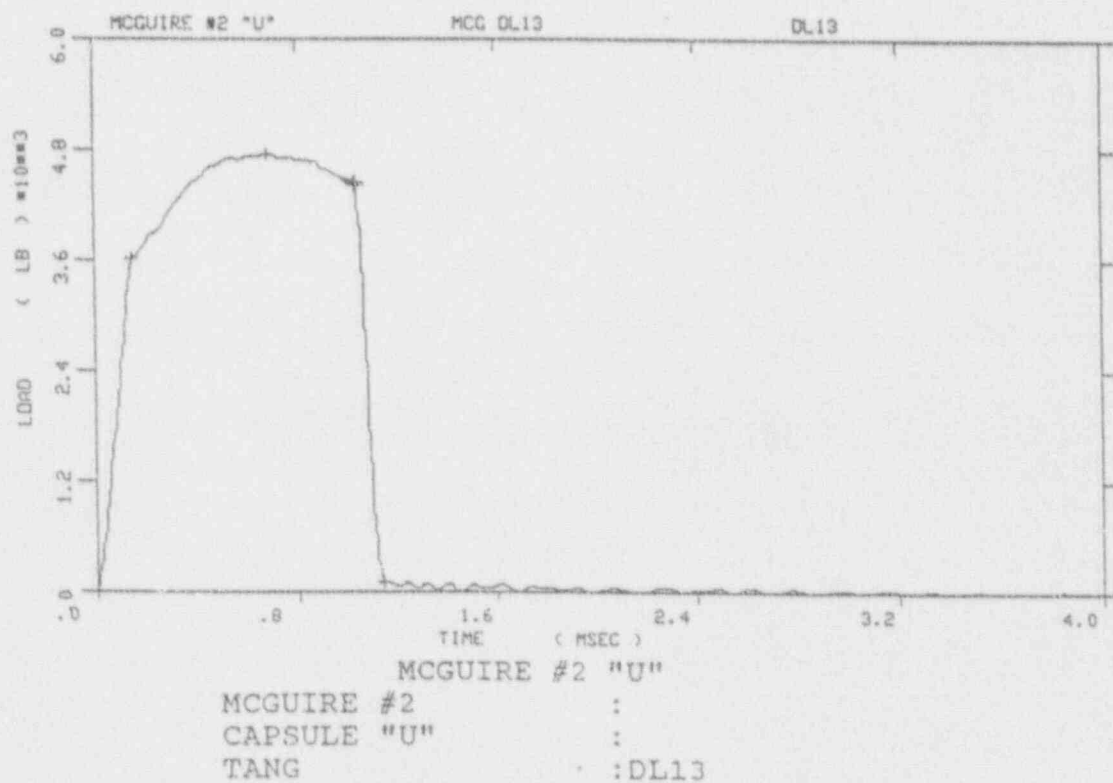
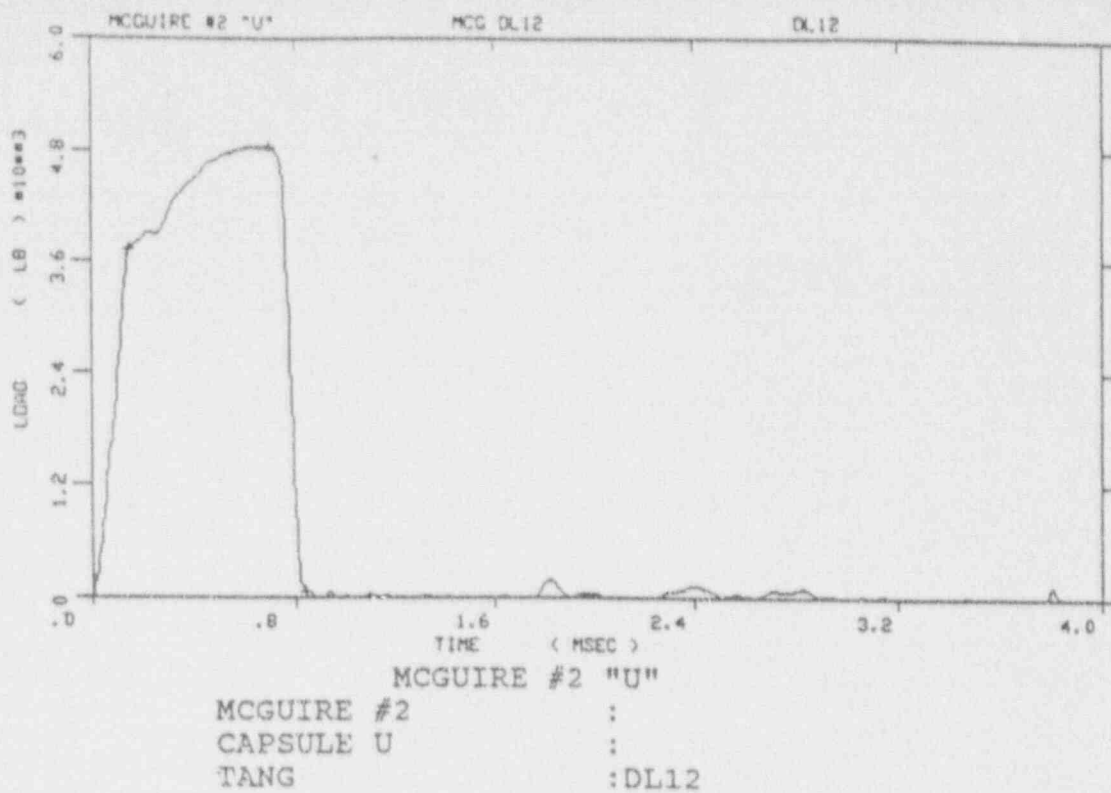


Figure A-5. Load-time records for Specimens DL12 and DL13

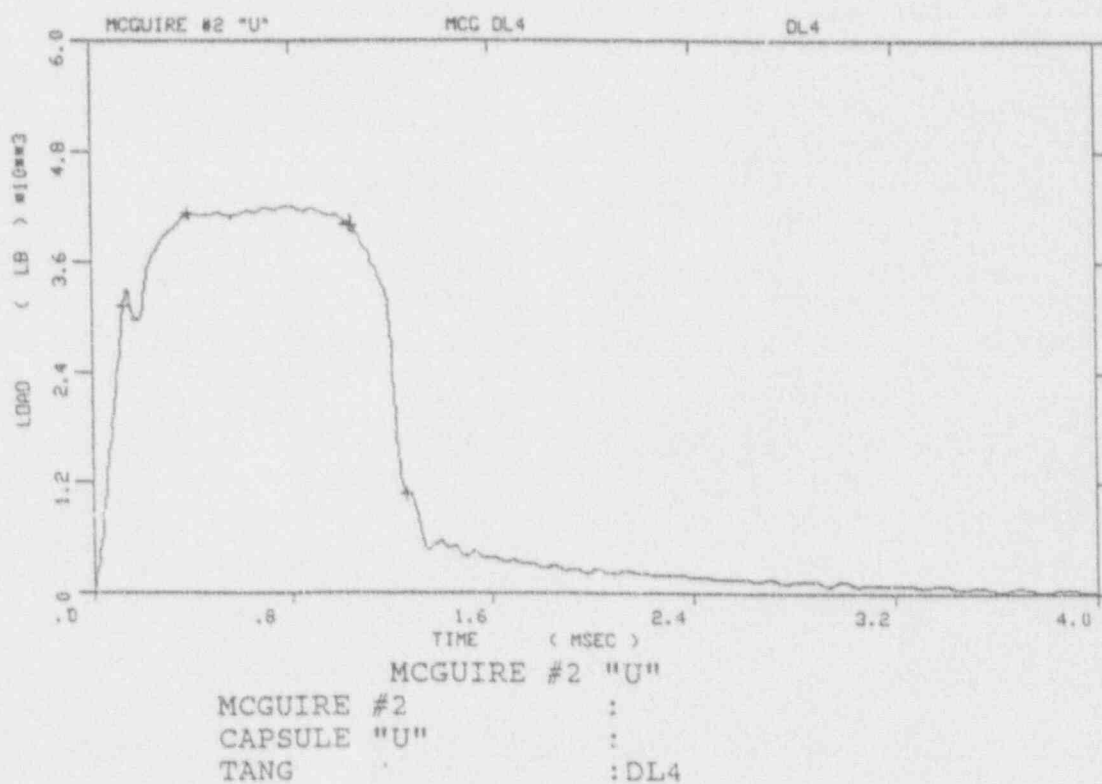
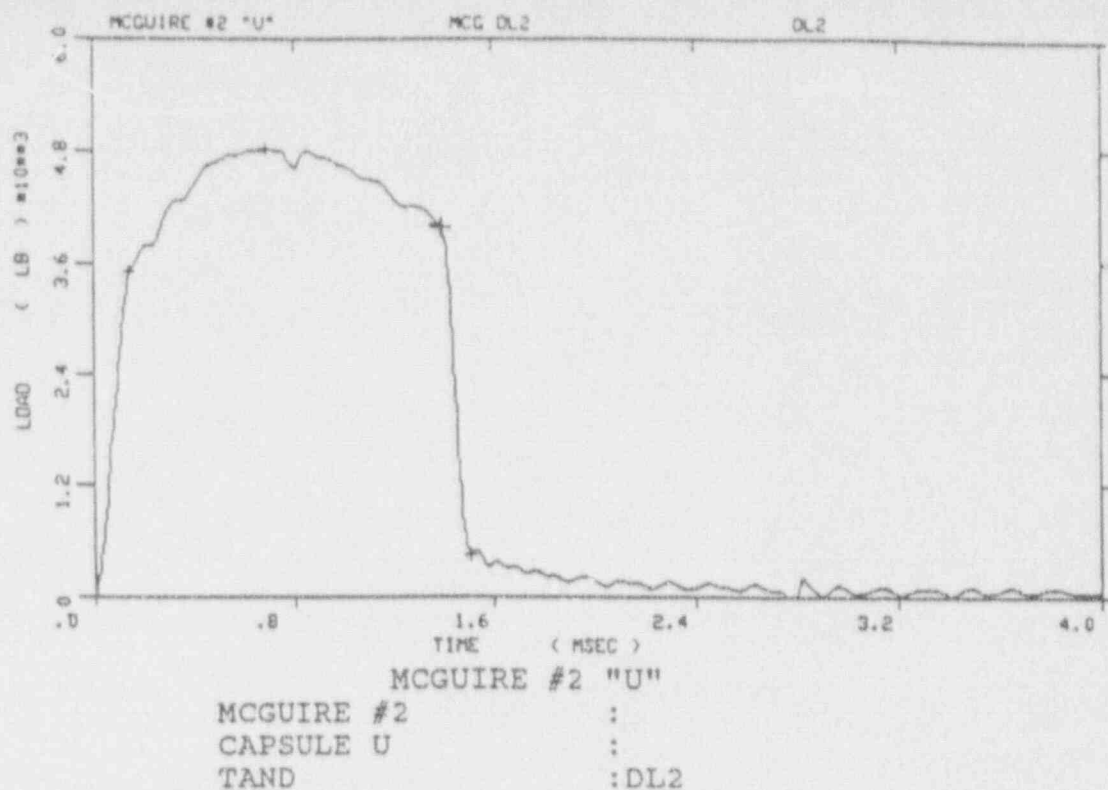


Figure A-6. Load-time records for Specimens DL2 and DL4

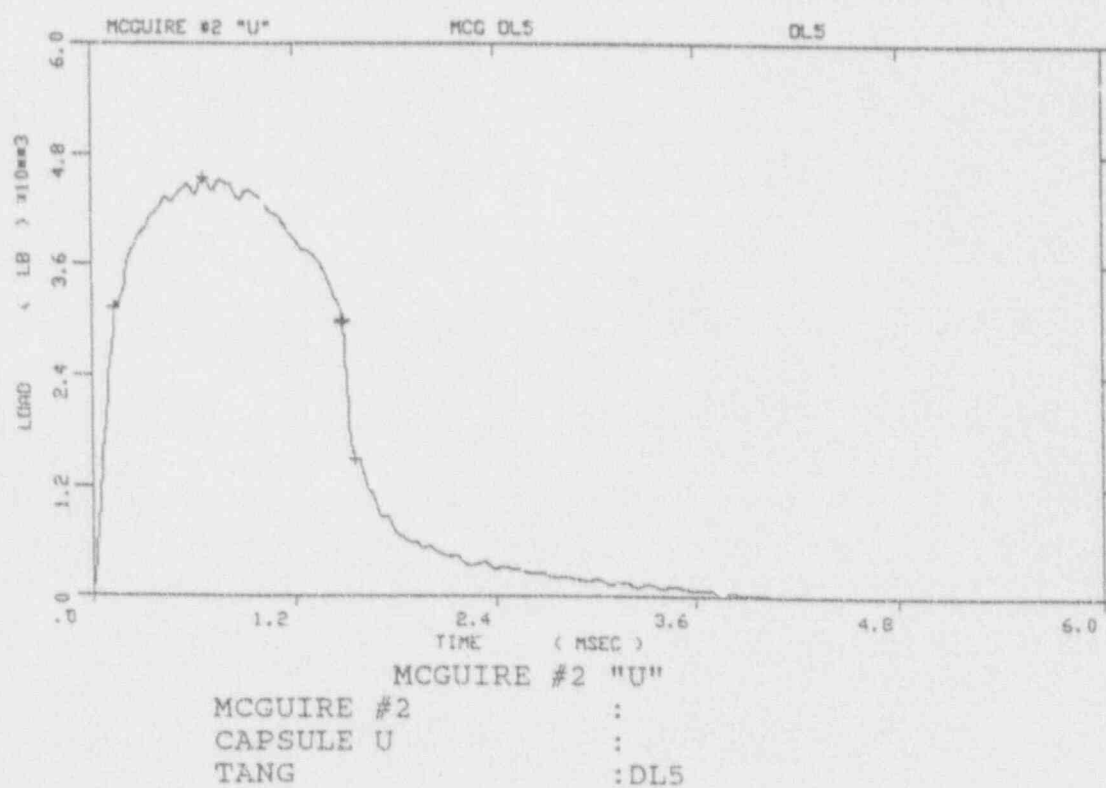
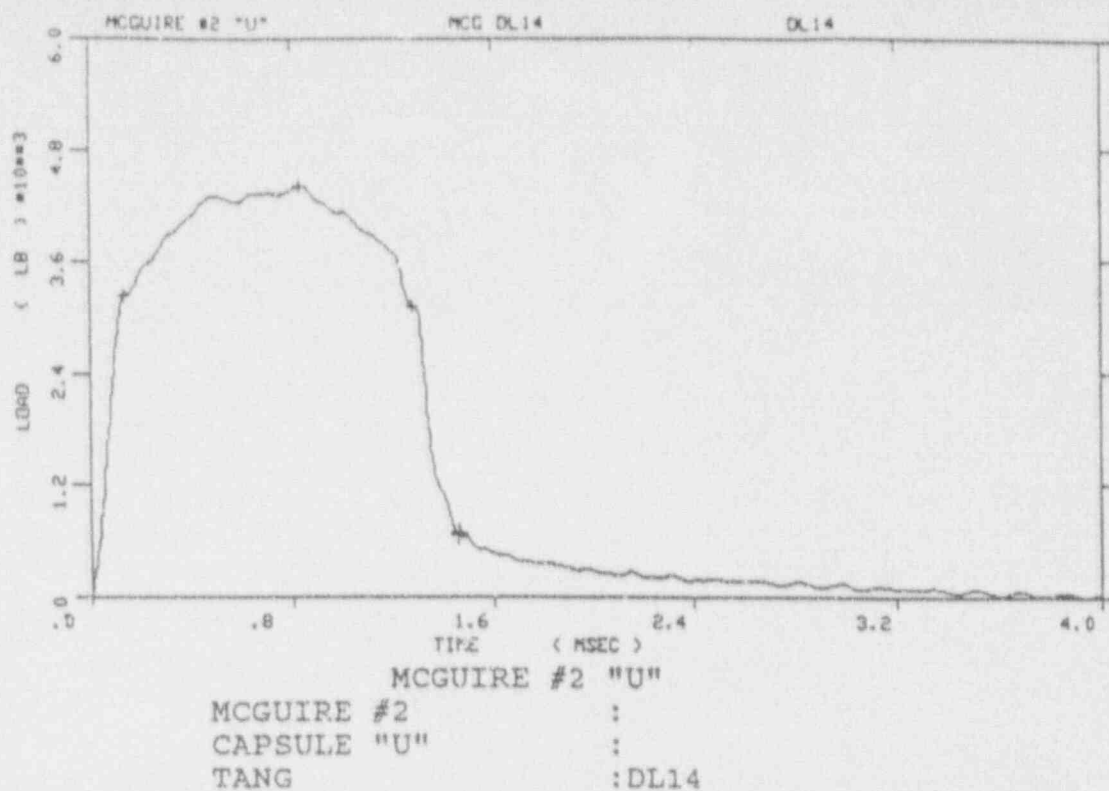
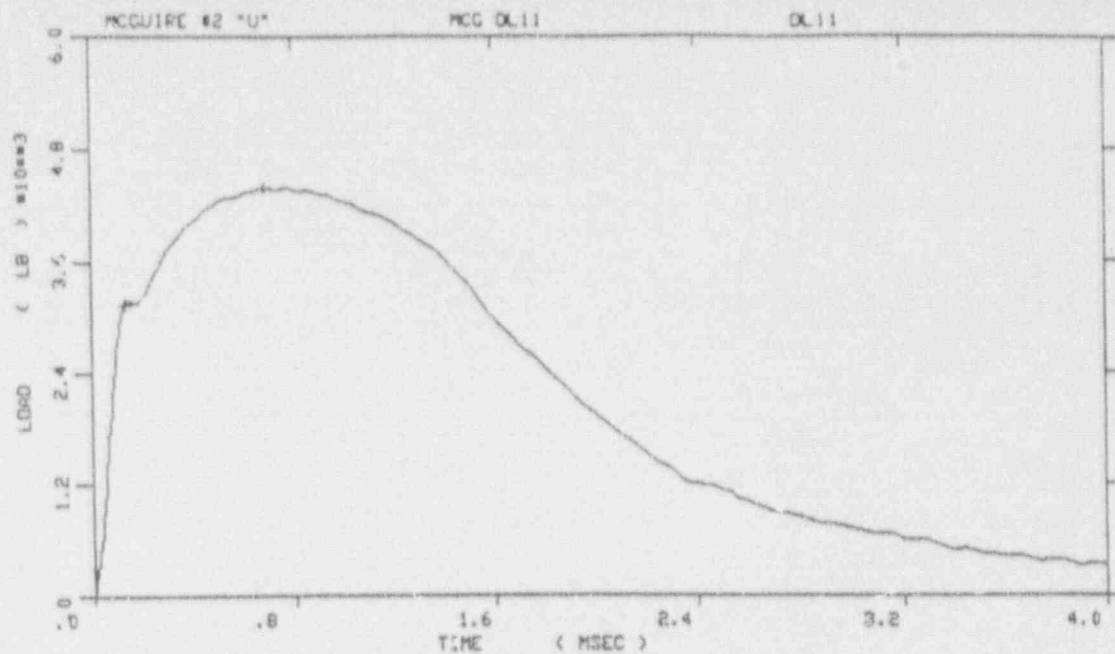
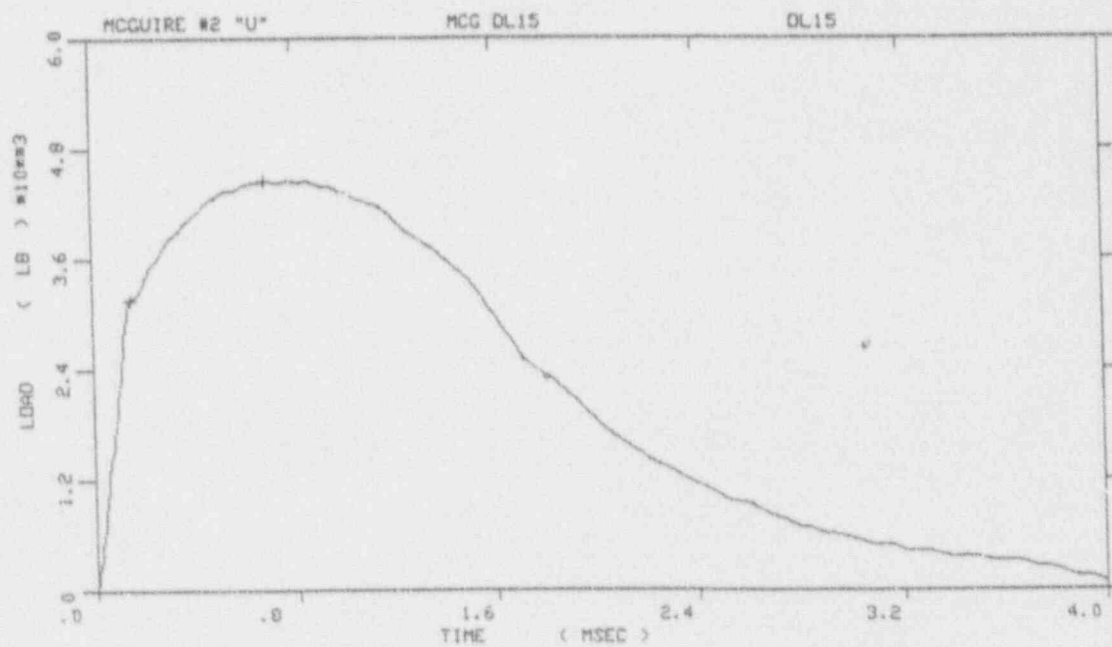


Figure A-7. Load-time records for Specimens DL14 and DL5



McGUIRE #2 "U"
 McGUIRE #2 :
 CAPSULE U :
 TANG :DL11



McGUIRE #2 "U"
 McGUIRE #2 :
 CAPSULE "U" :
 TANG :DL15

Figure A-8. Load-time records for Specimens DL11 and DL15

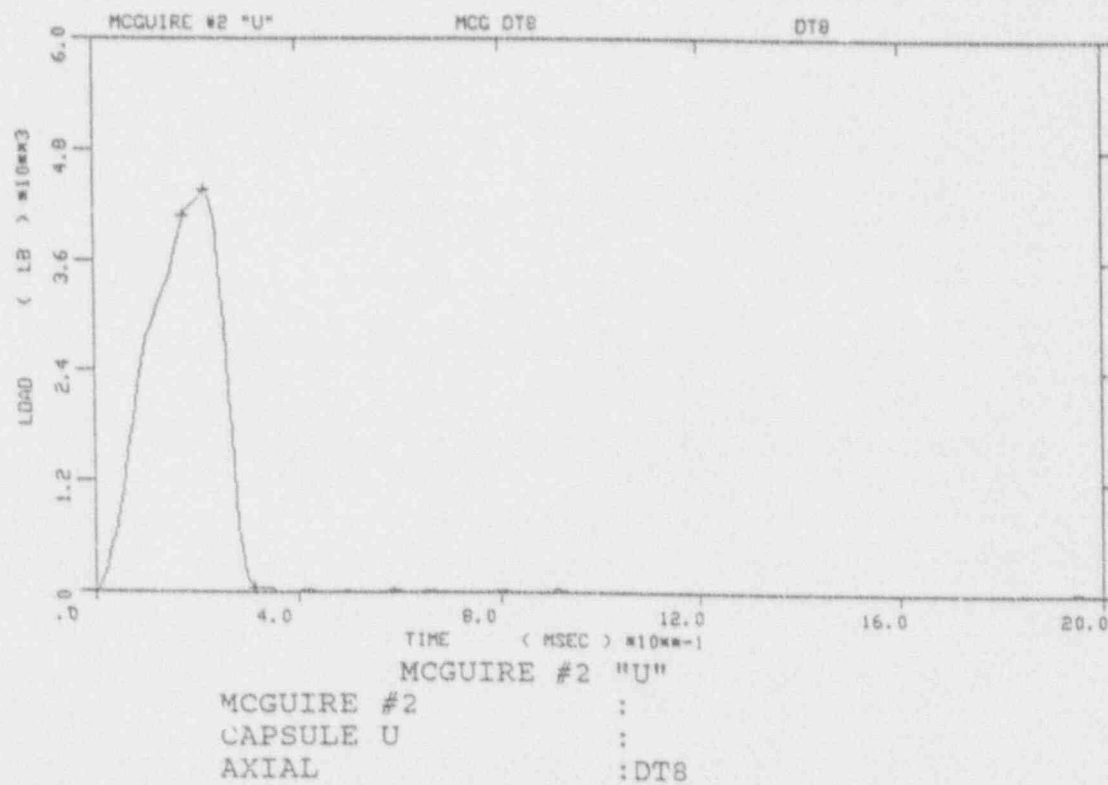
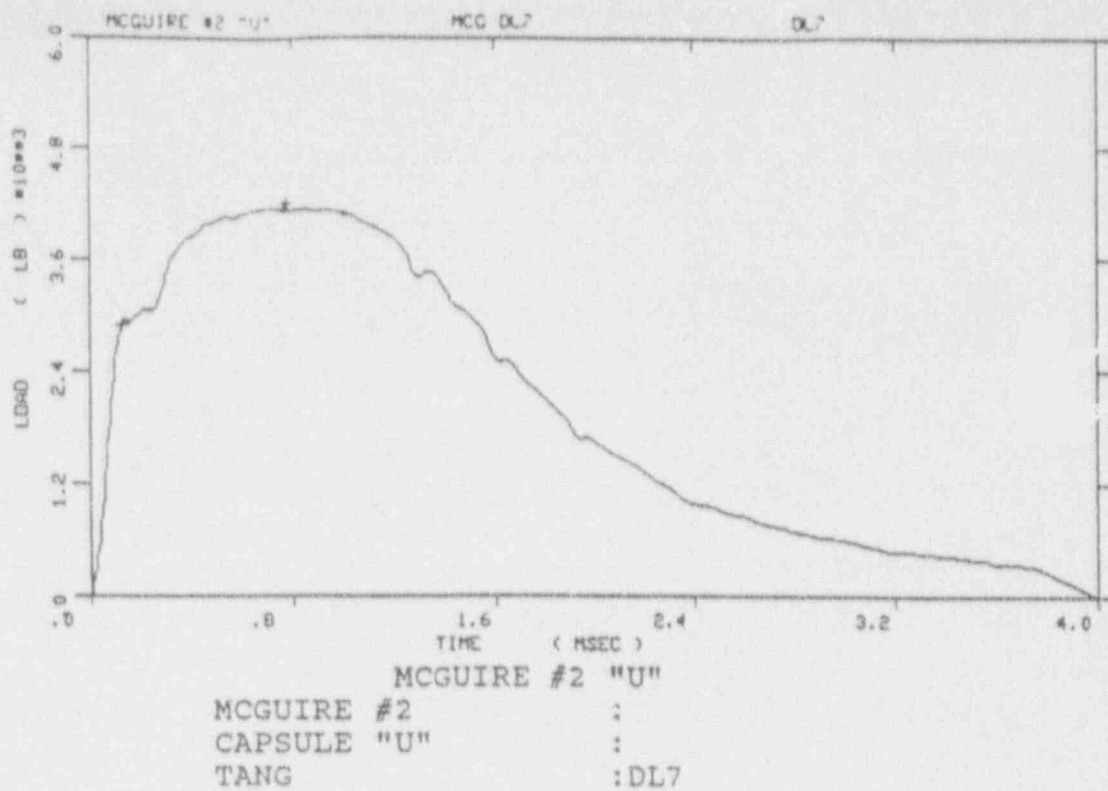


Figure A-9. Load-time records for Specimens DL7 and DT8

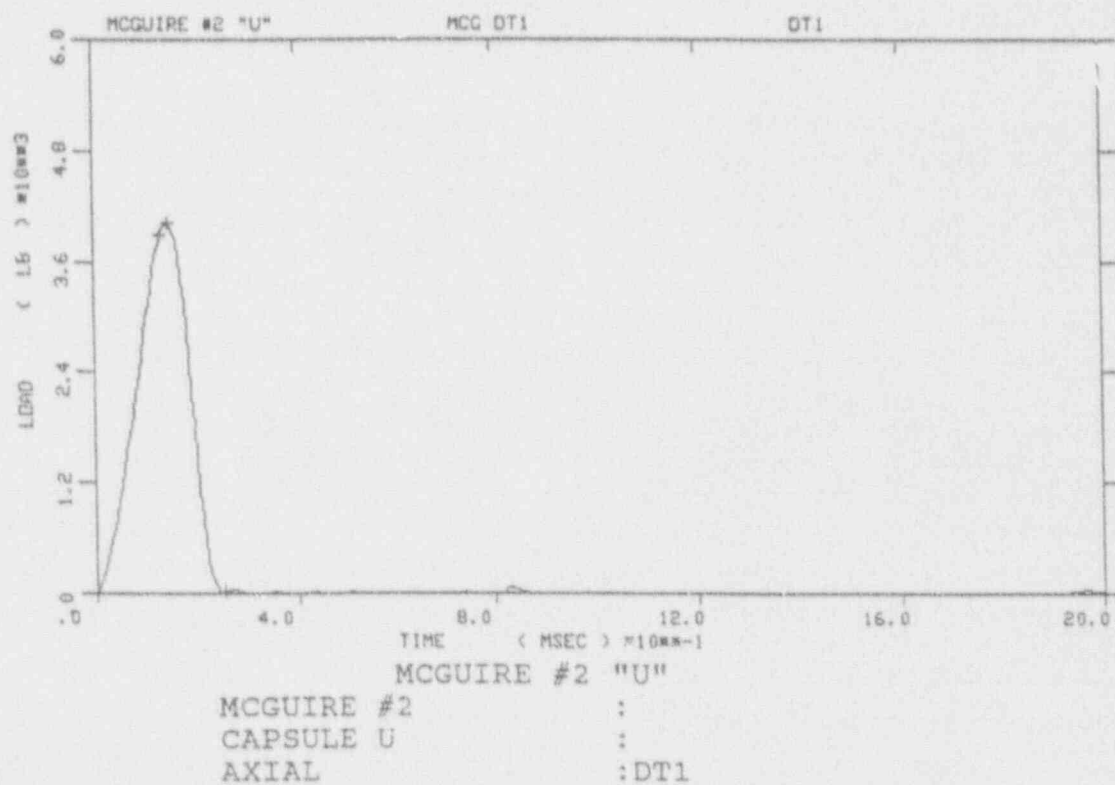
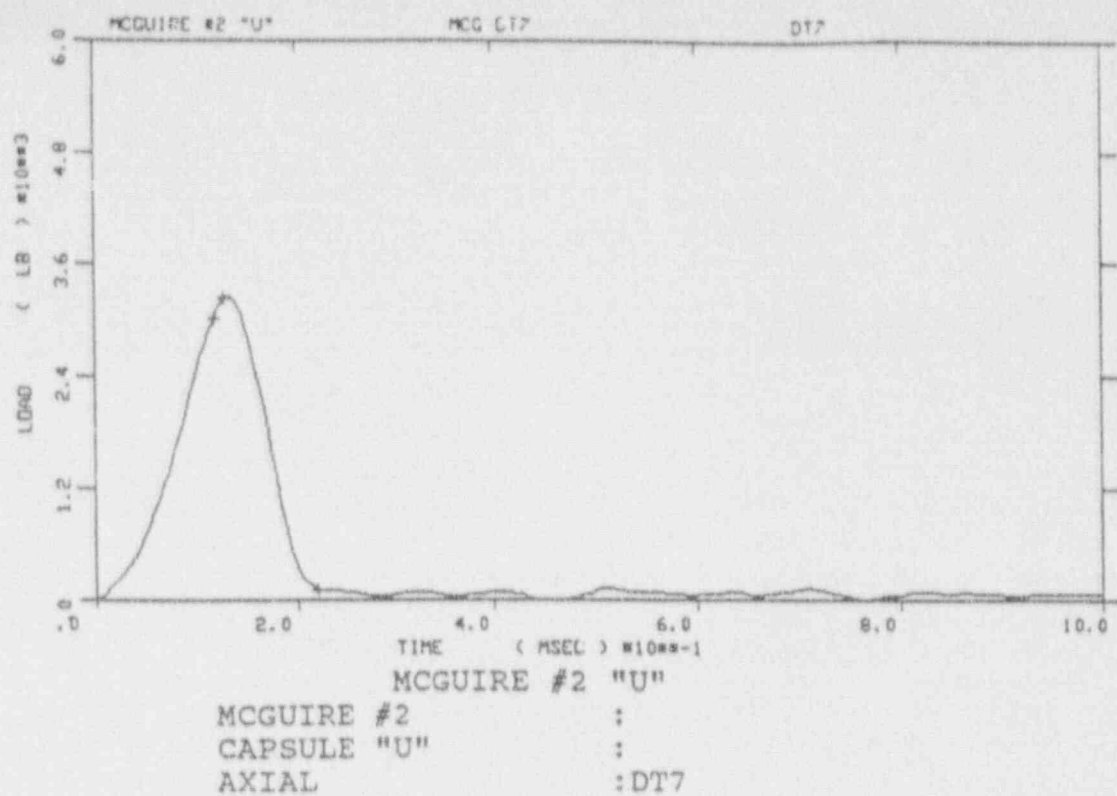
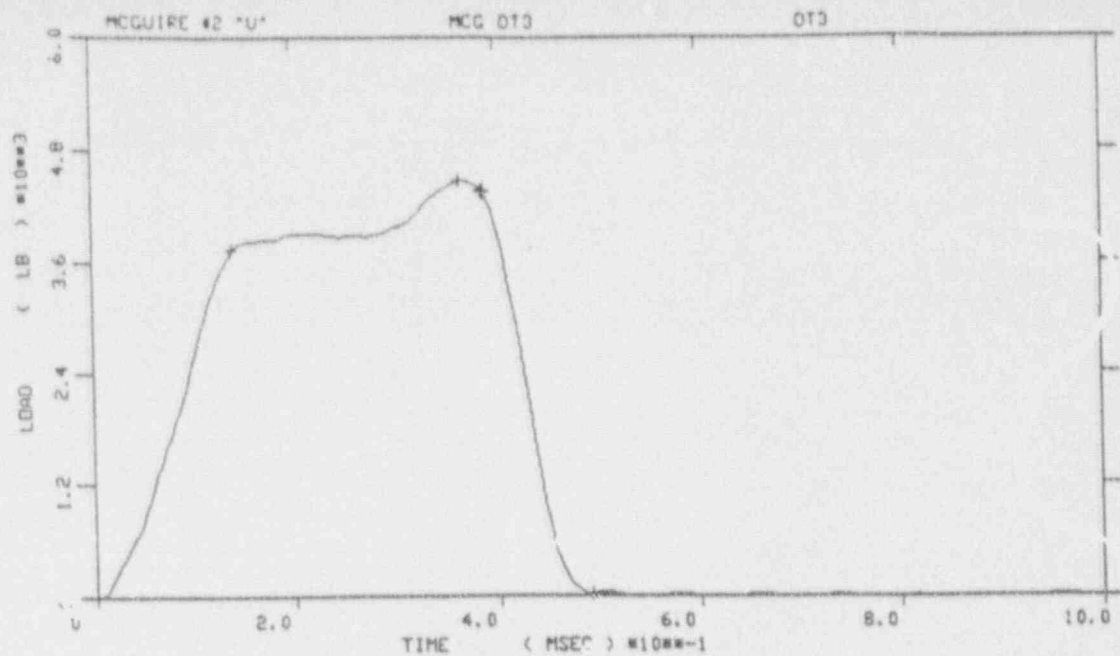
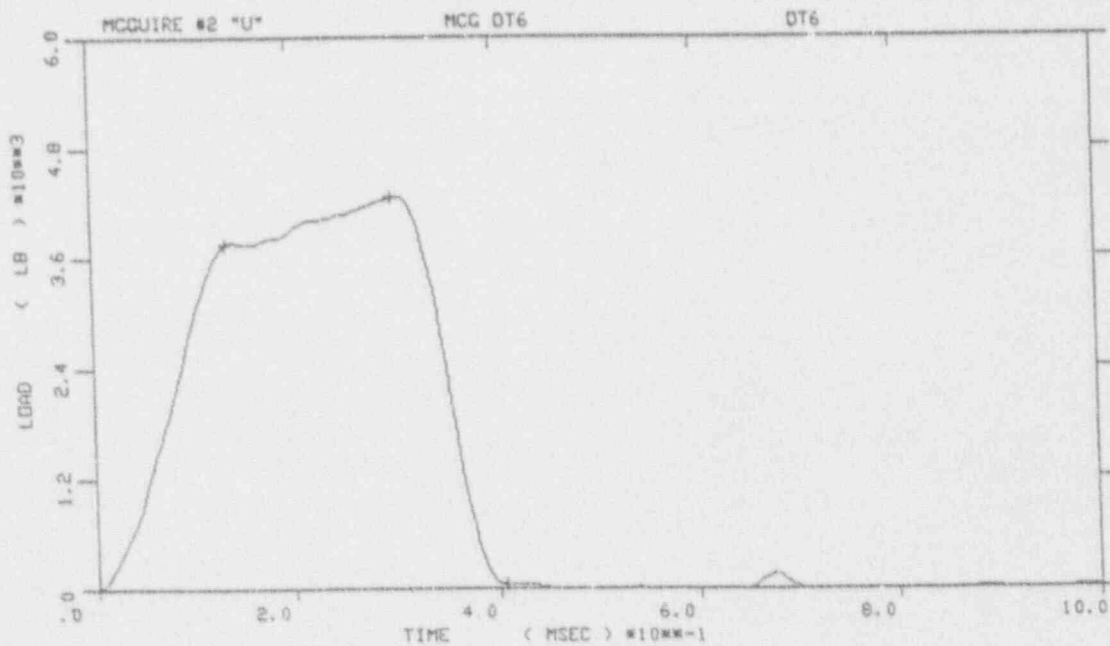


Figure A-10. Load-time records for Specimens DT7 and DT1

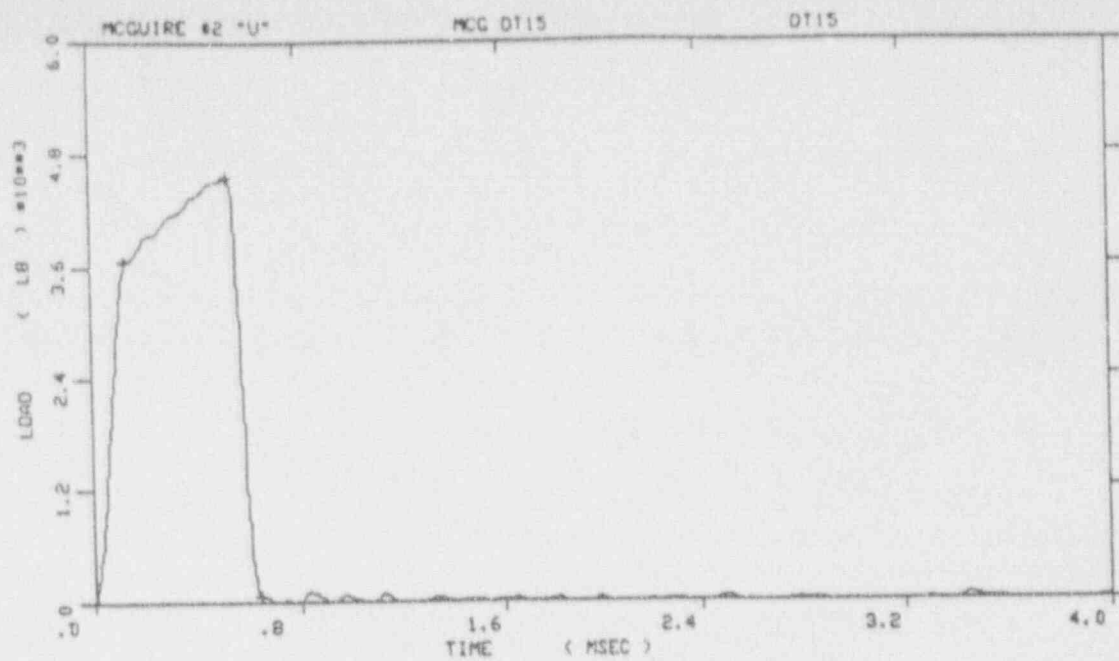


MCGUIRE #2 :
 CAPSULE U :
 AXIAL :DT3

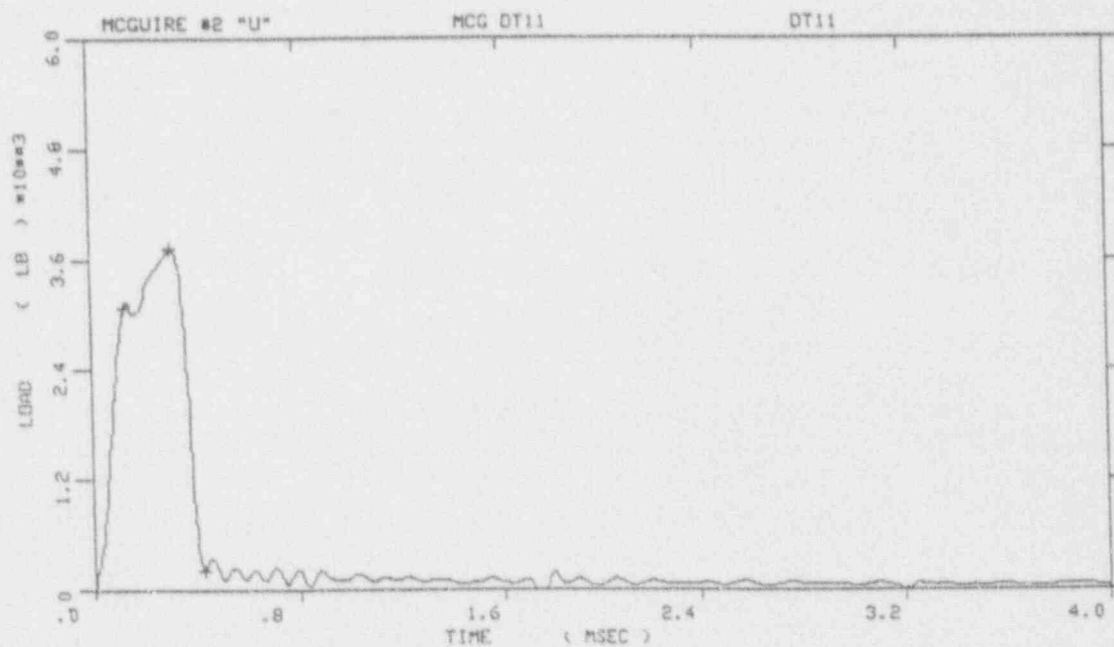


MCGUIRE #2 :
 CAPSULE "U" :
 AXIAL :DT6

Figure A-11. Load-time records for Specimens DT3 and DT6



MCGUIRE #2 :
 CAPSULE U :
 AXIAL :DT15



MCGUIRE #2 :
 CAPSULE "U" :
 AXIAL :DT11

Figure A-12. Load-time records for Specimens DT15 and DT11

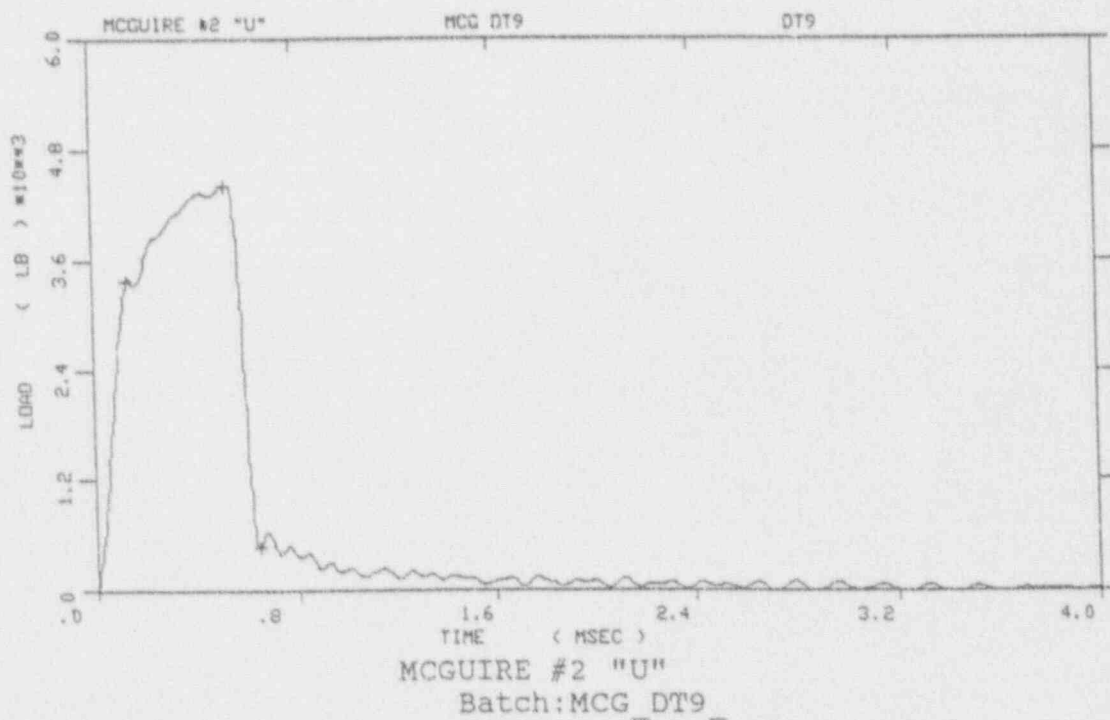
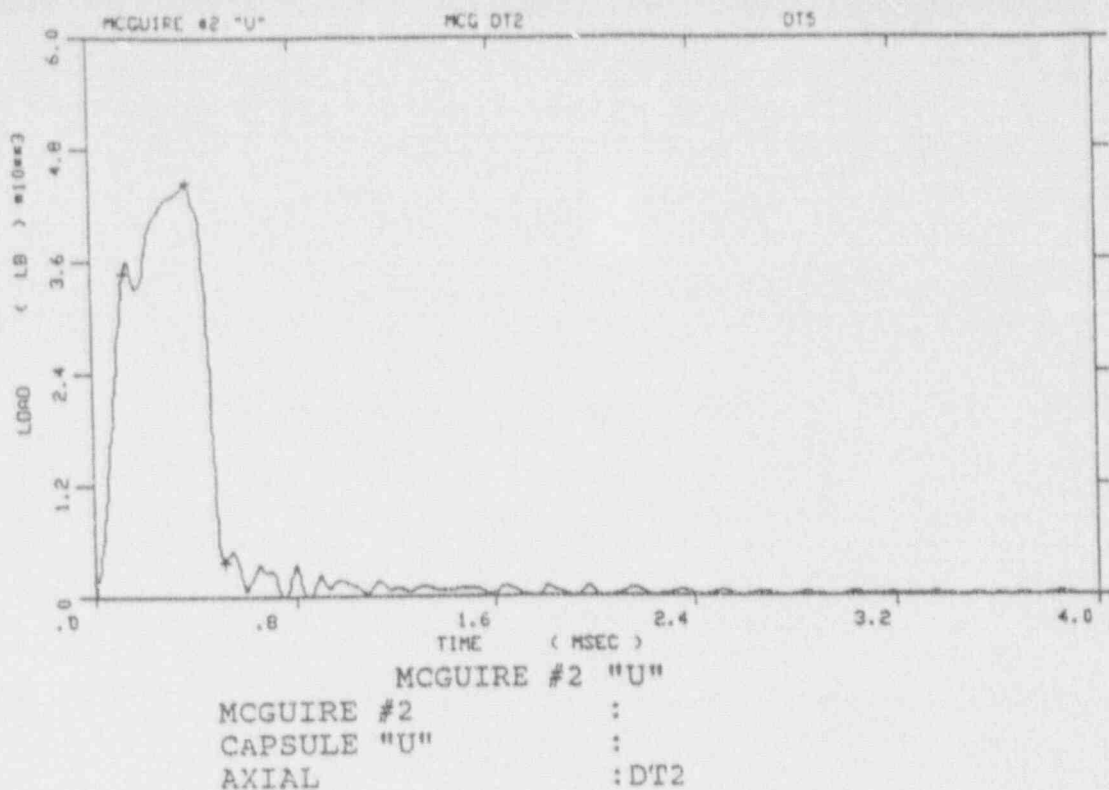


Figure A-13. Load-time records for Specimens DT2 and DT9

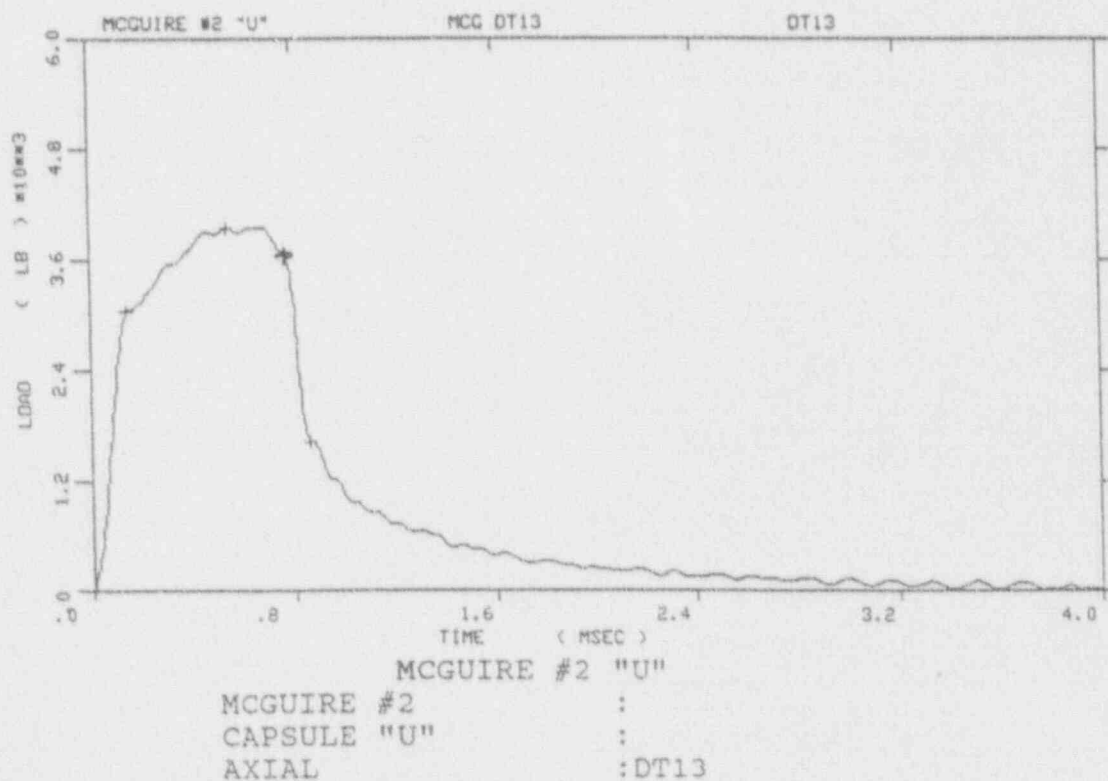
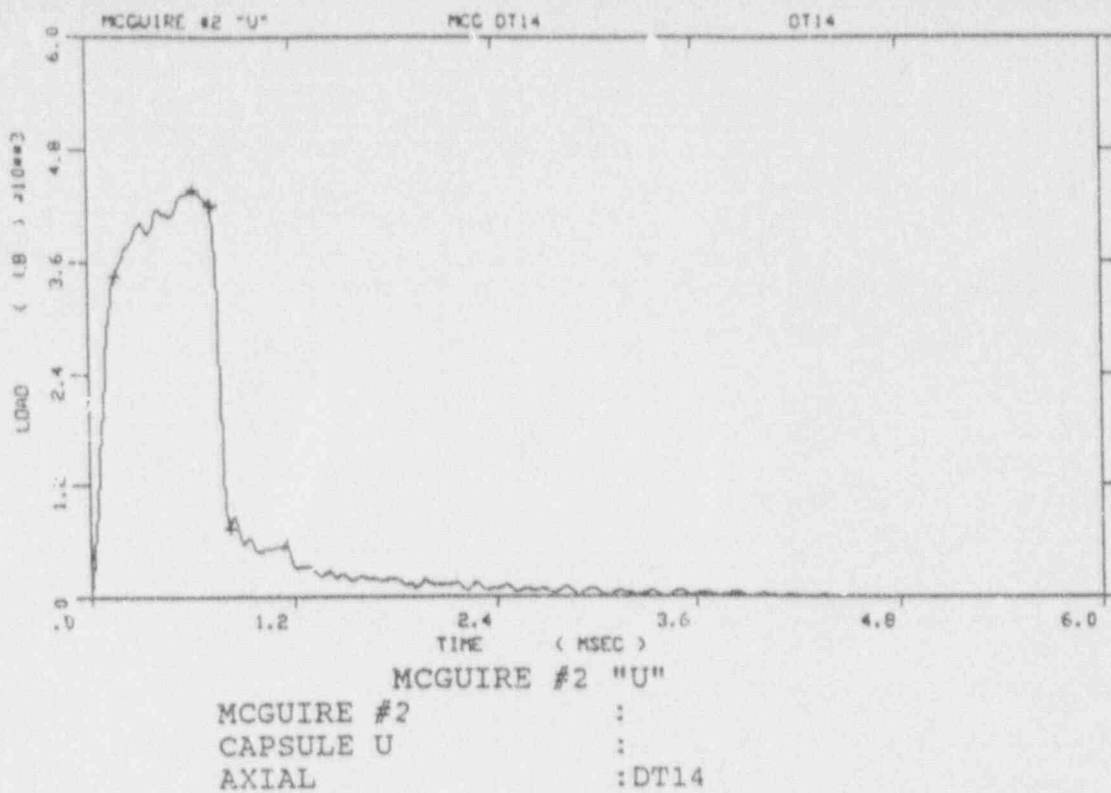


Figure A-14. Load-time records for Specimens DT14 and DT13

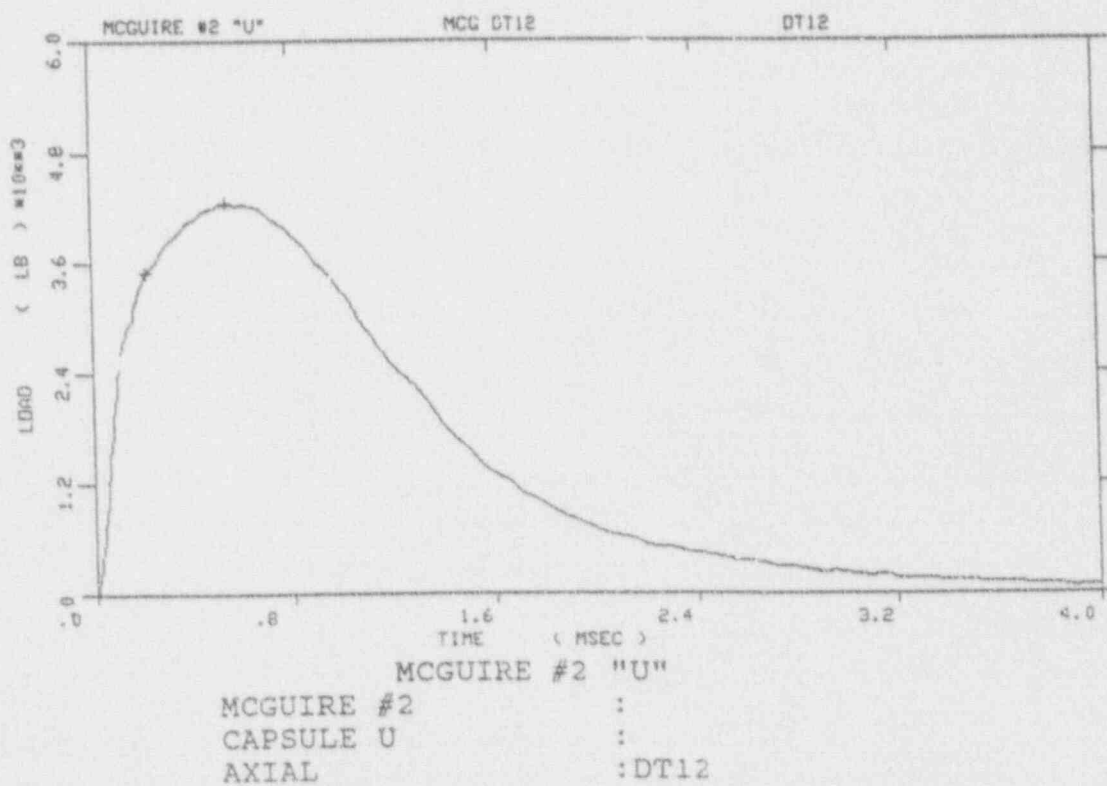
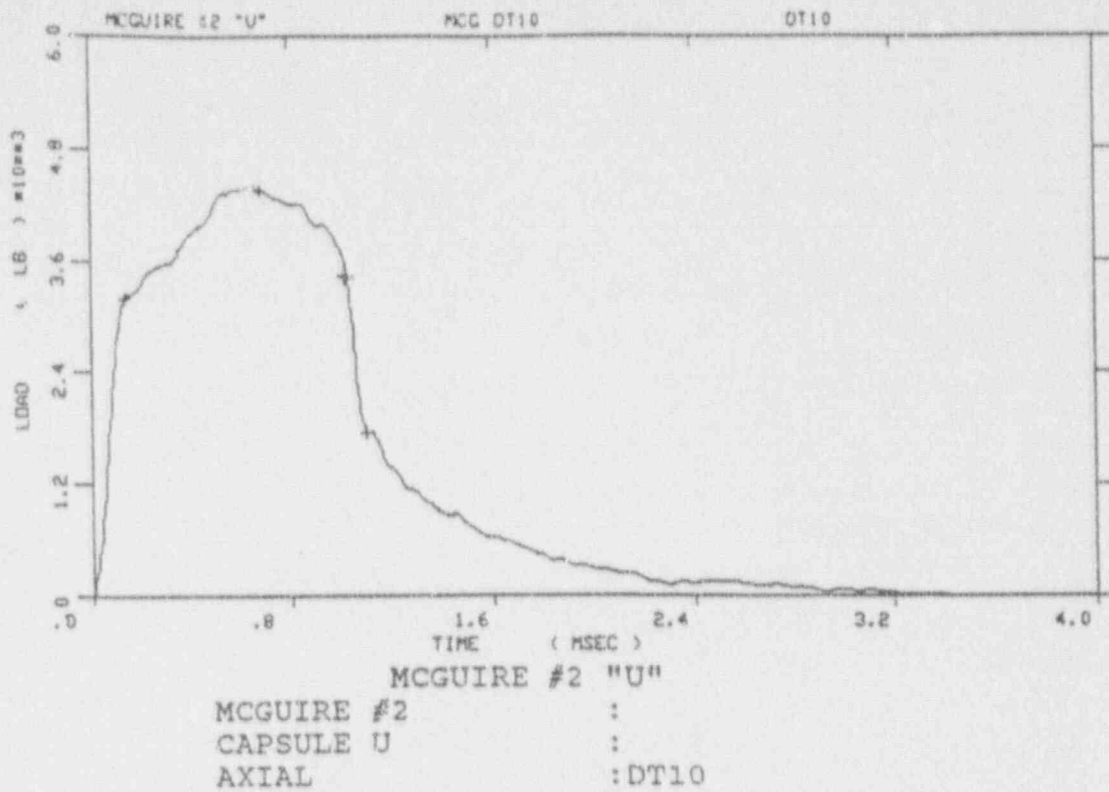


Figure A-15. Load-time records for Specimens DT10 and DT12

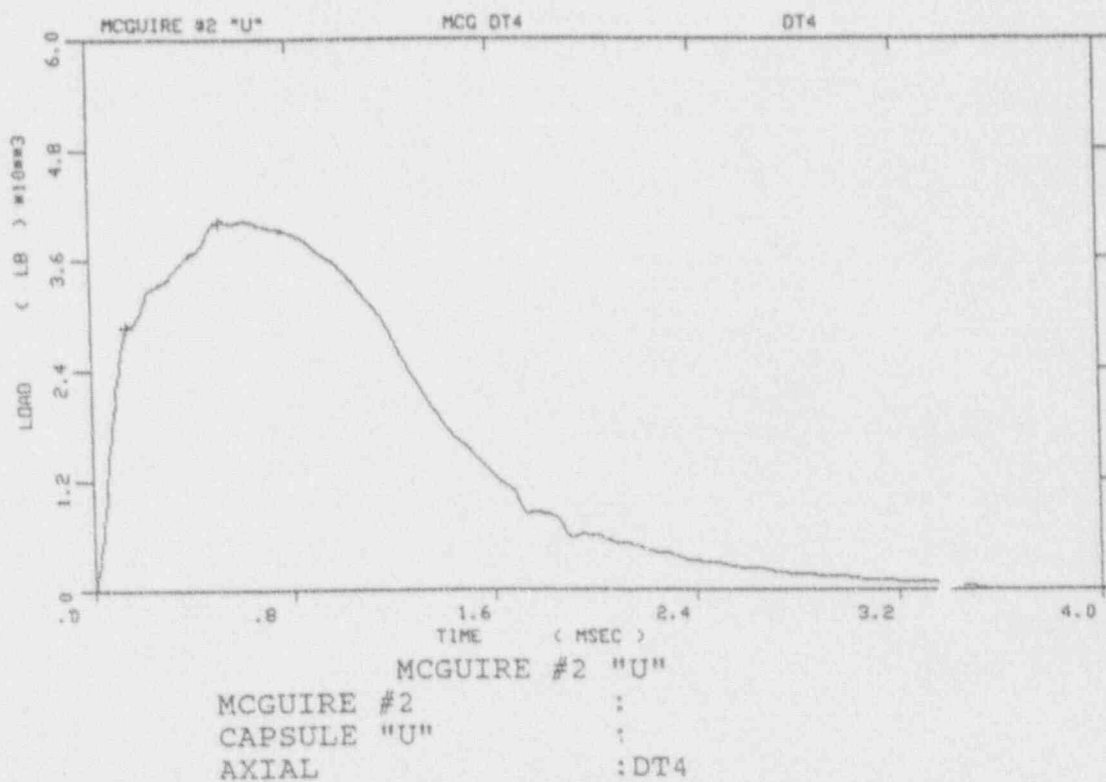
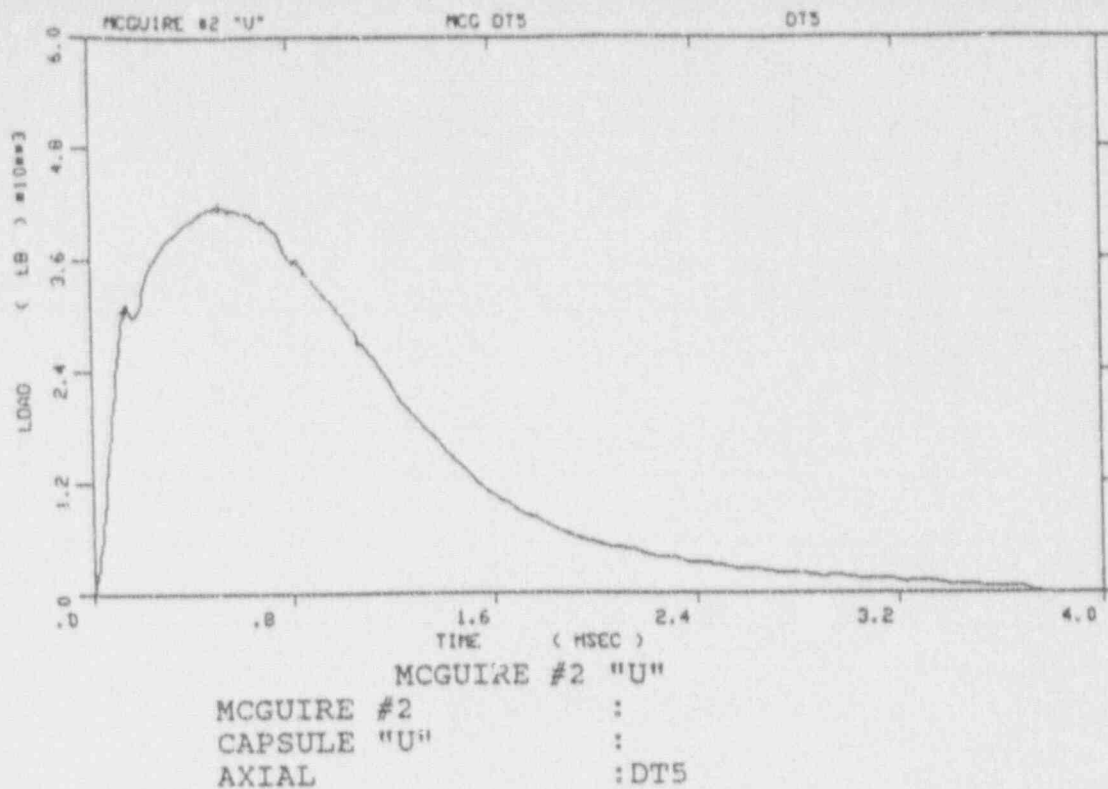


Figure A-16. Load-time records for Specimens DT5 and DT4

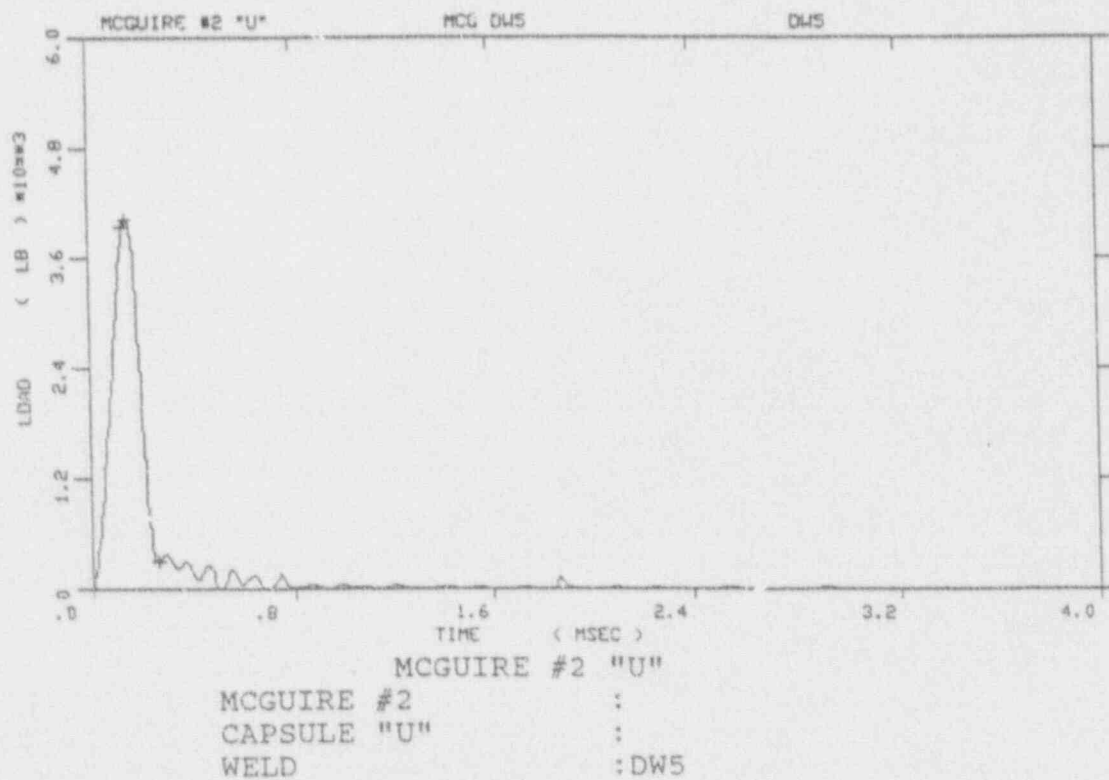
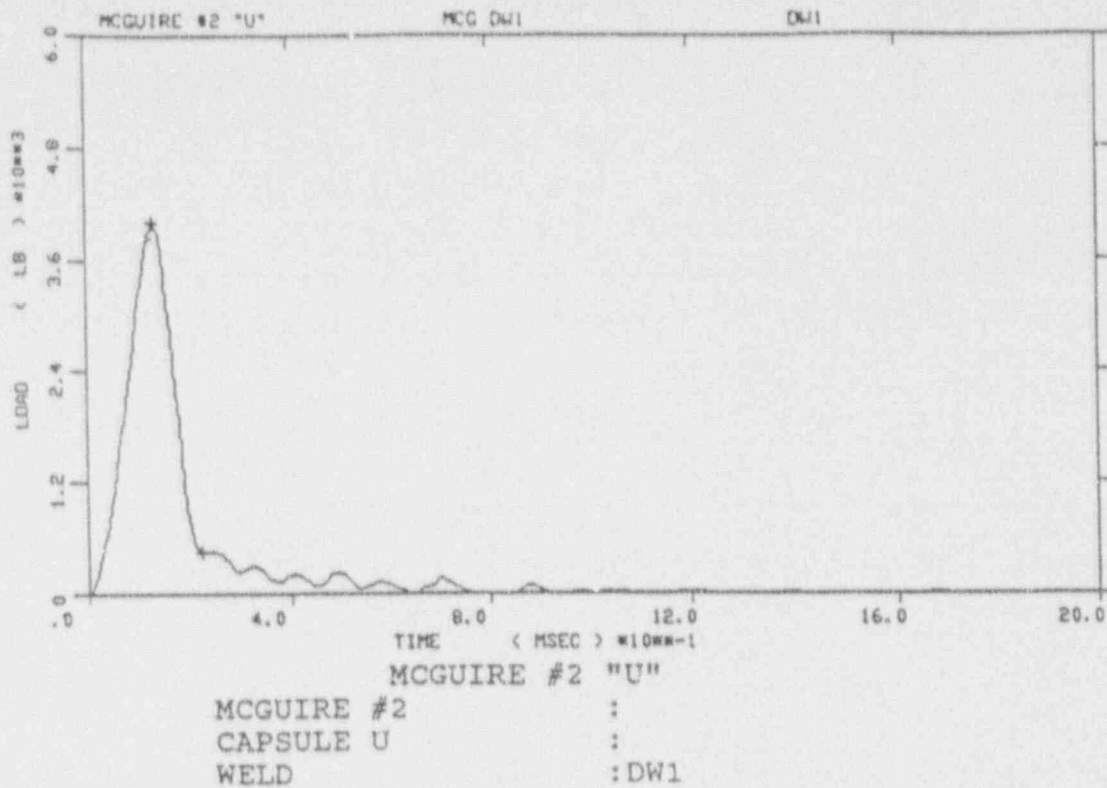


Figure A-17. Load-time records for Specimens DW1 and DW5

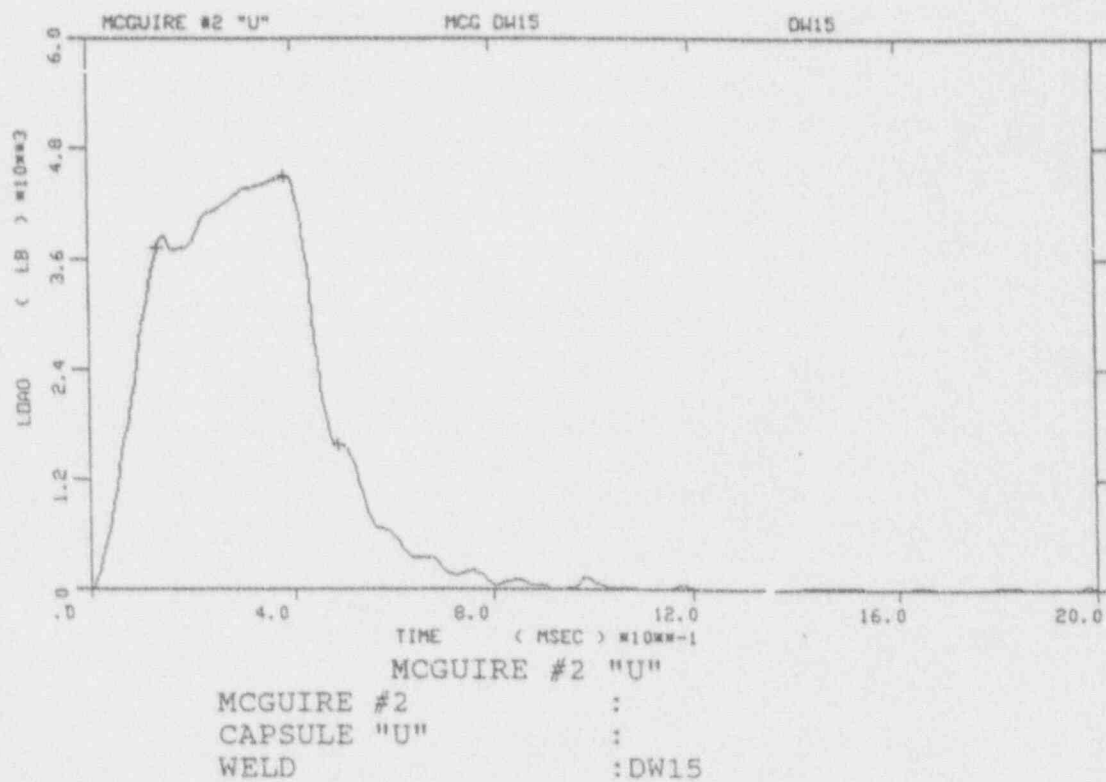
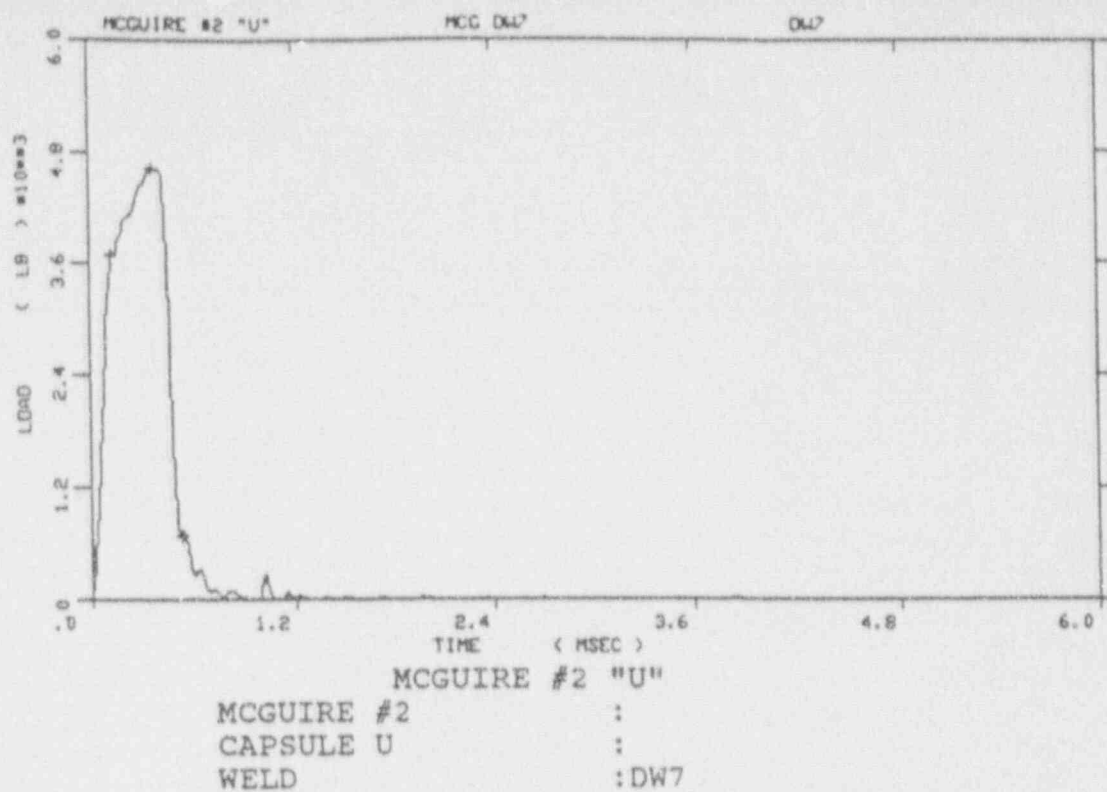


Figure A-18. Load-time reco. s for Specimens DW7 and DW15

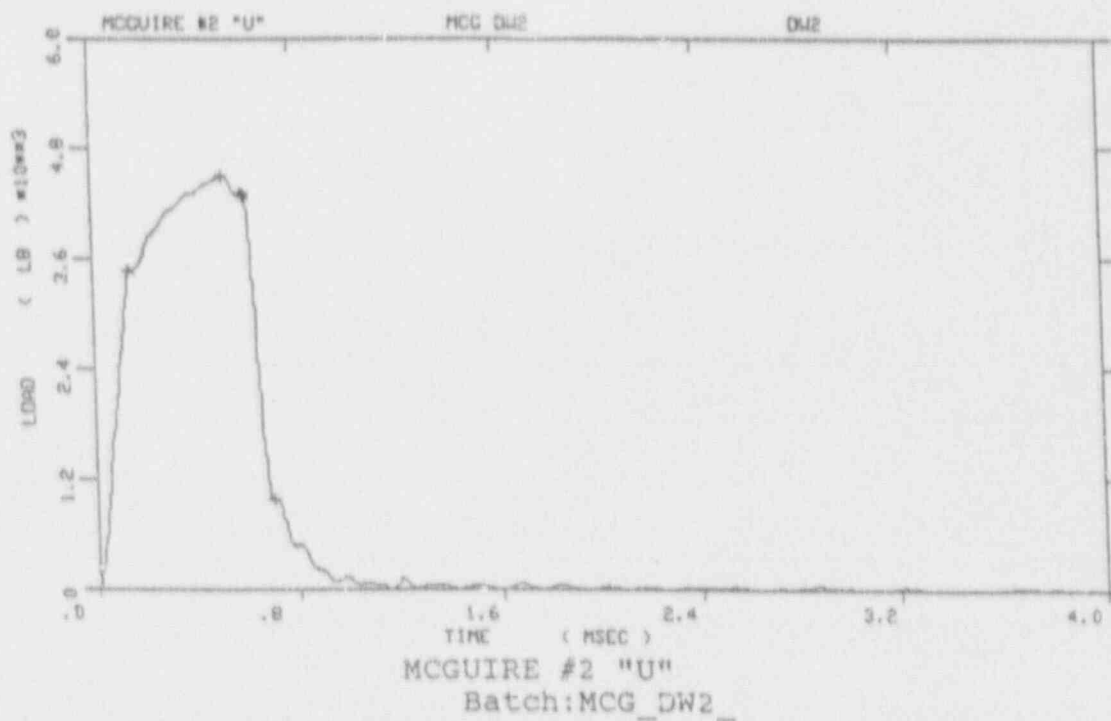
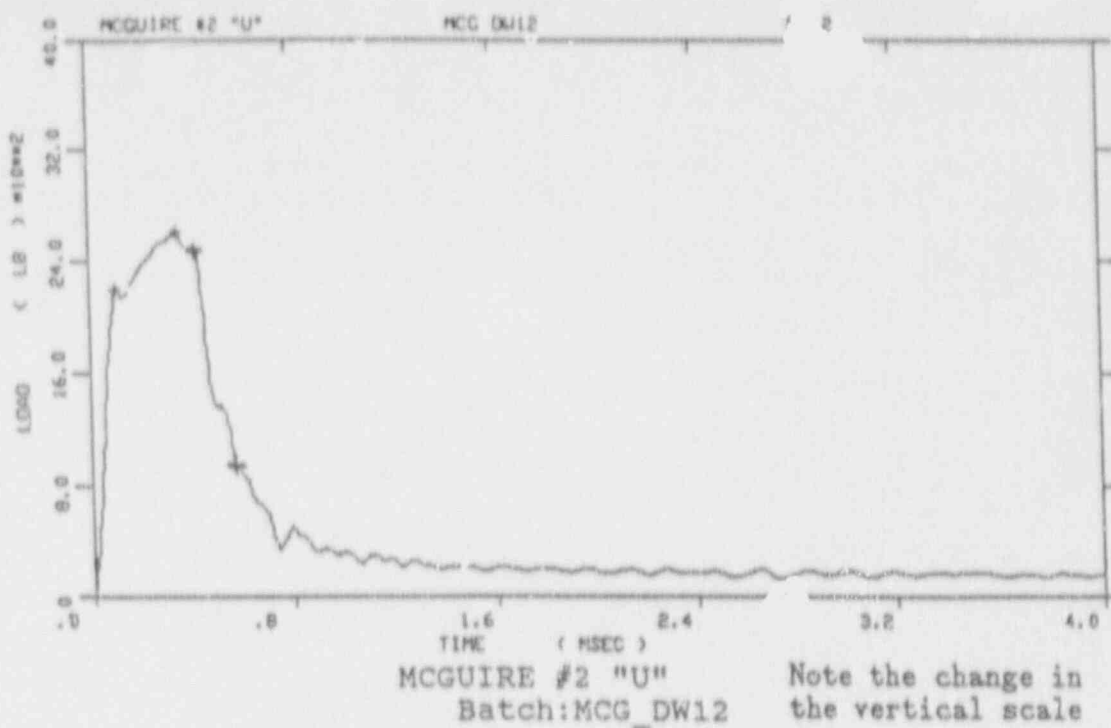


Figure A-19. Load-time records for Specimens DW12 and DW2

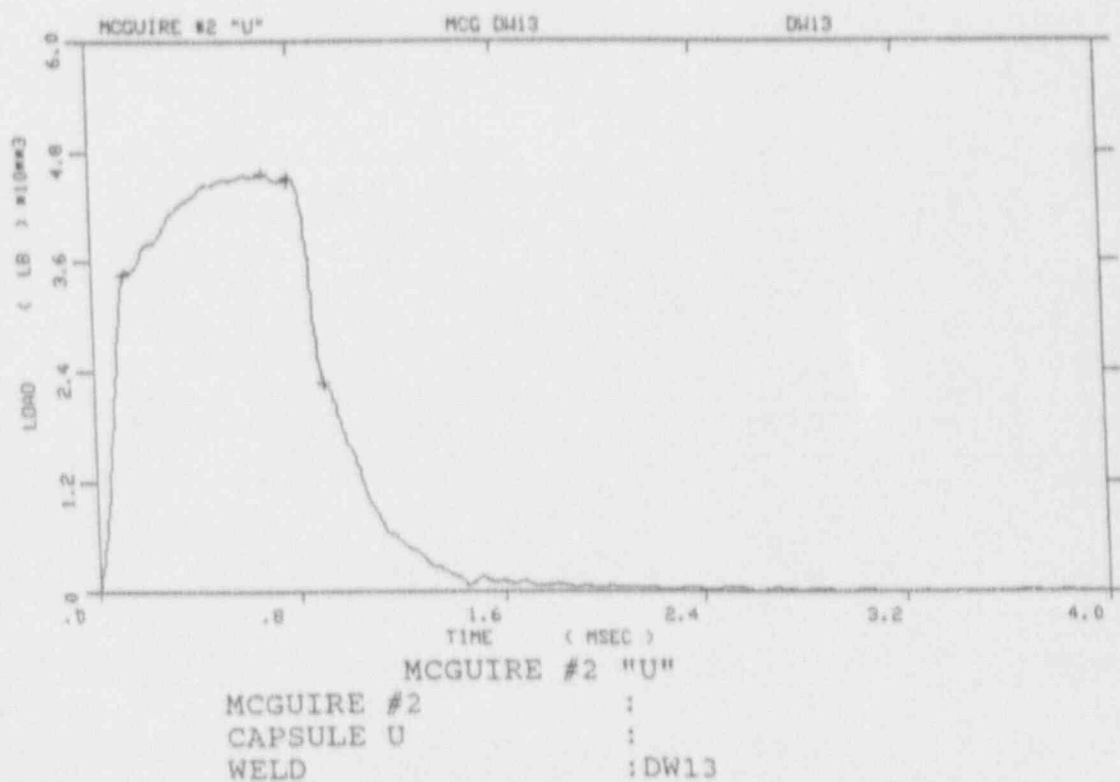
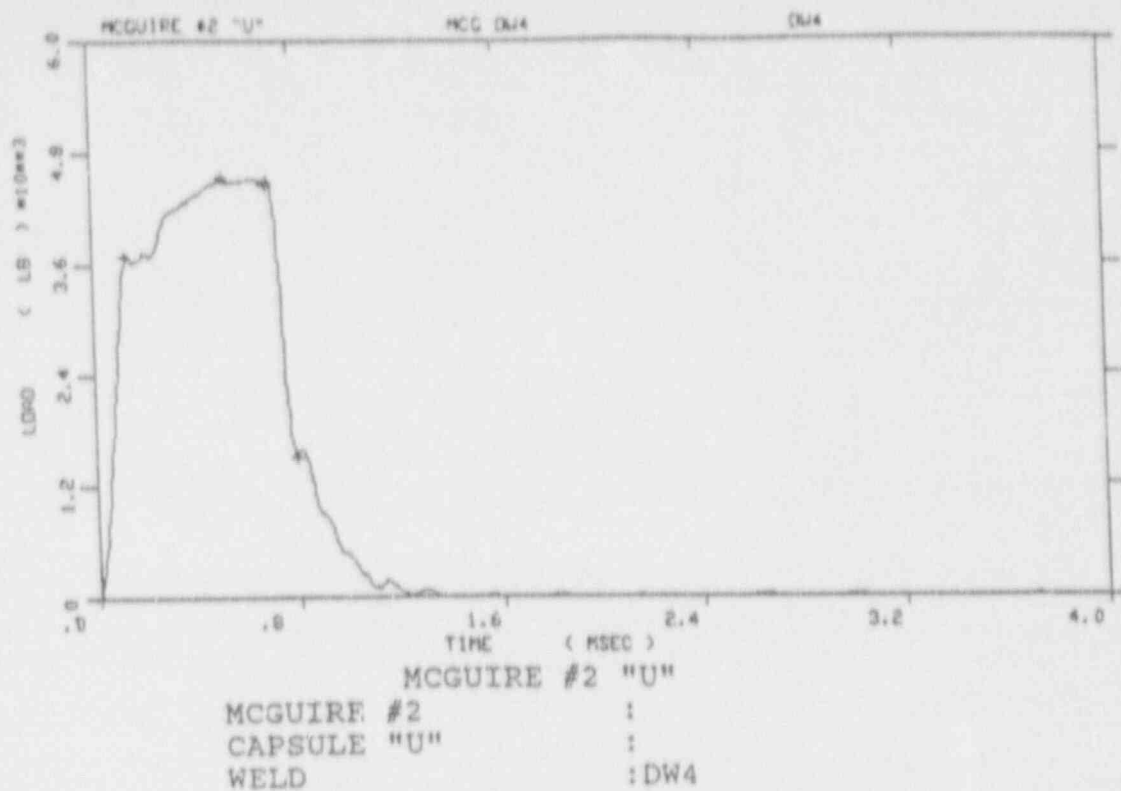


Figure A-20. Load-time records for Specimens DW4 and DW13

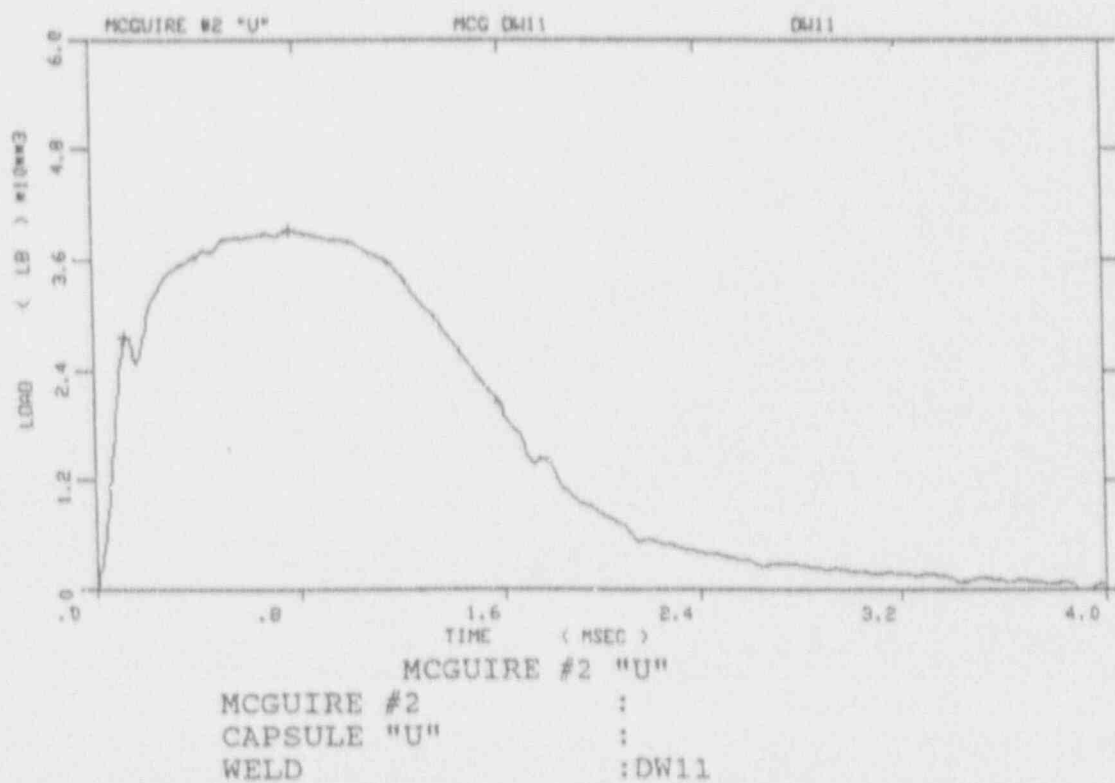
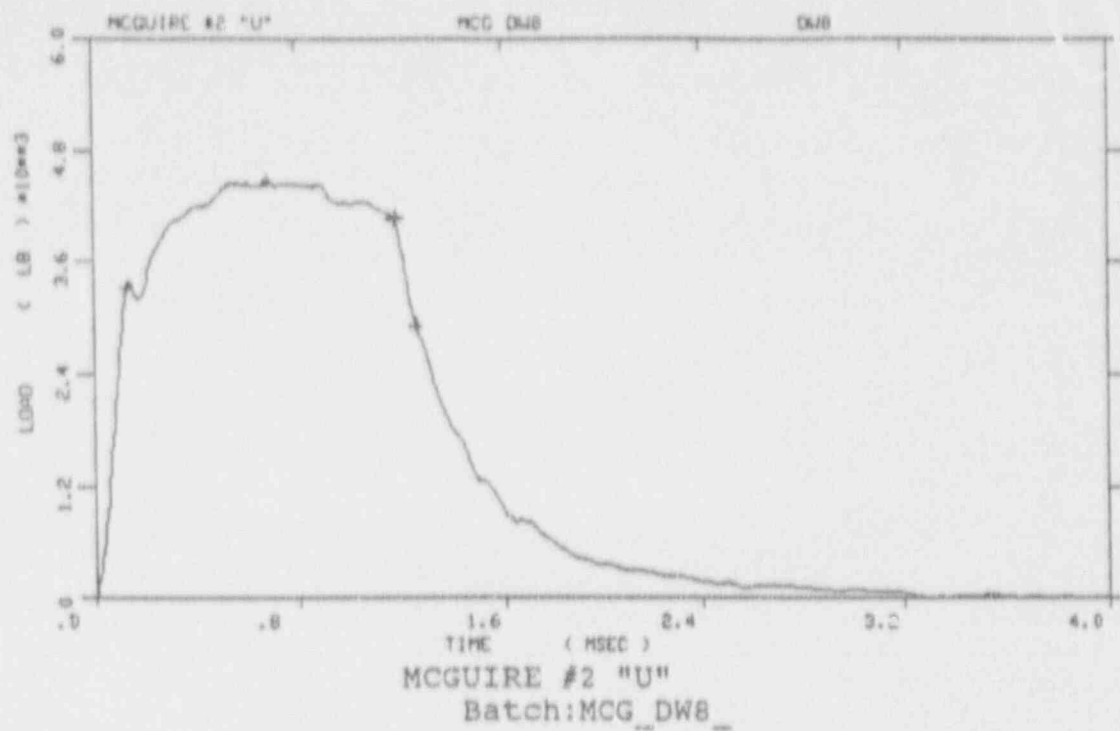


Figure A-21. Load-time records for Specimens DW8 and DW11

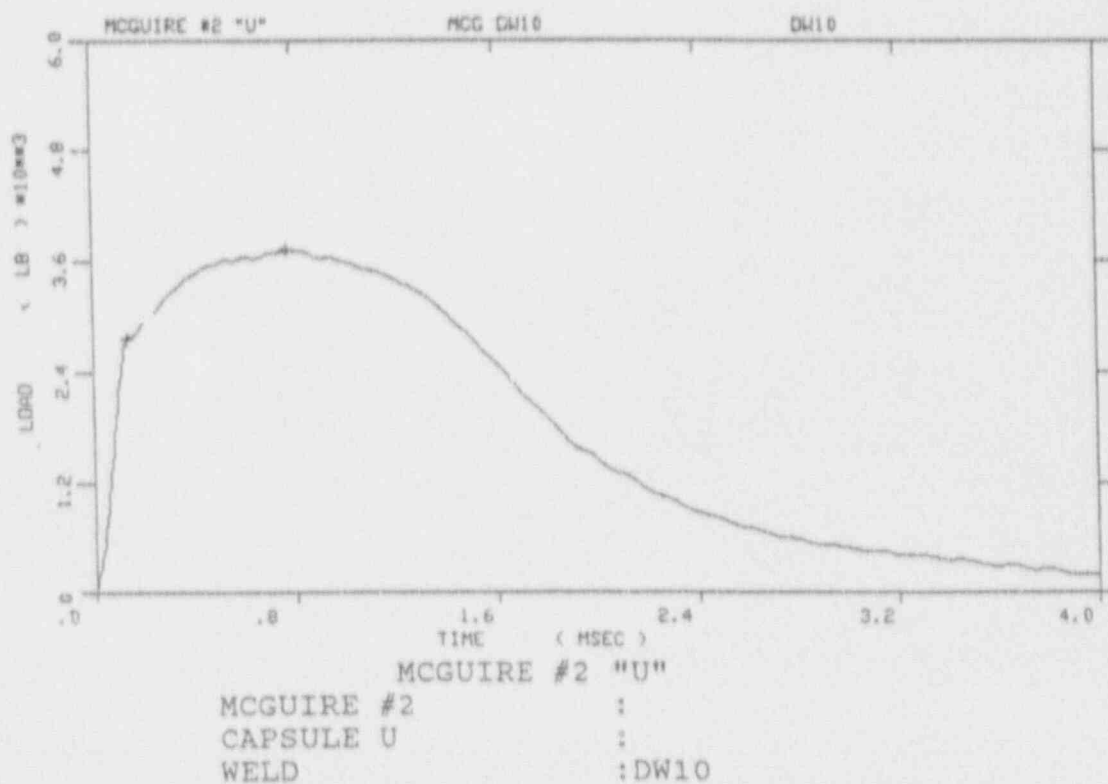
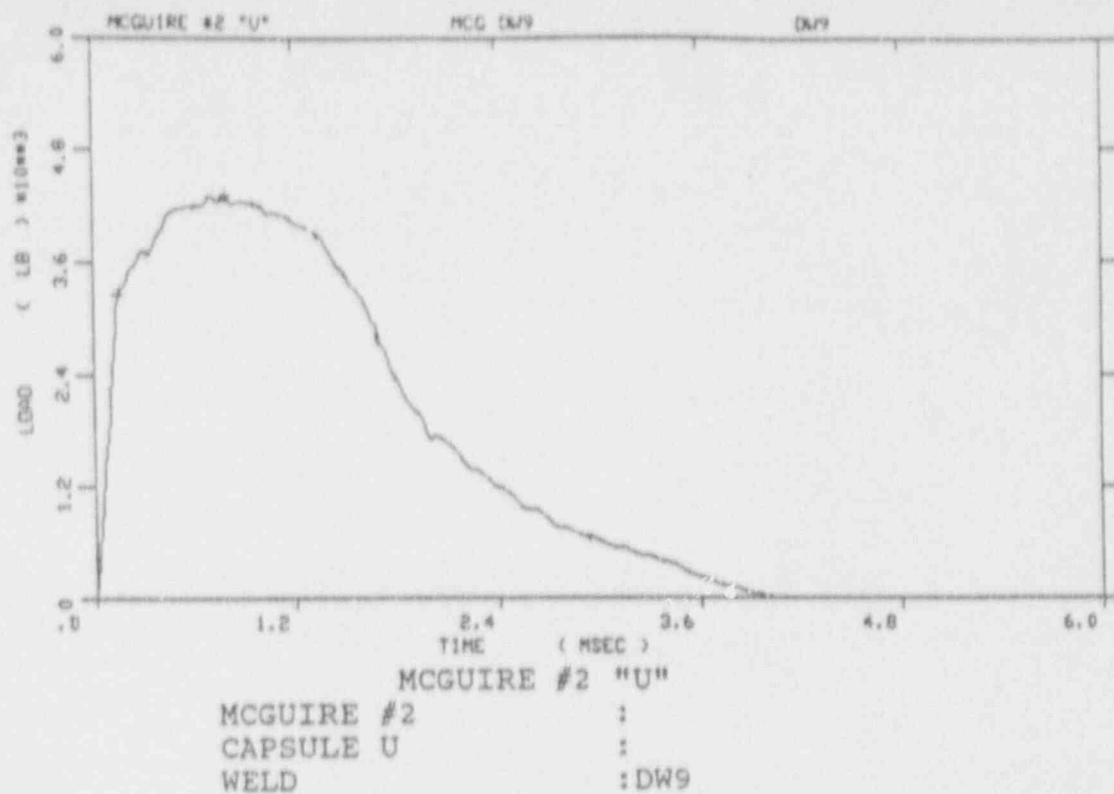


Figure A-22. Load-time records for Specimens DW9 and DW10

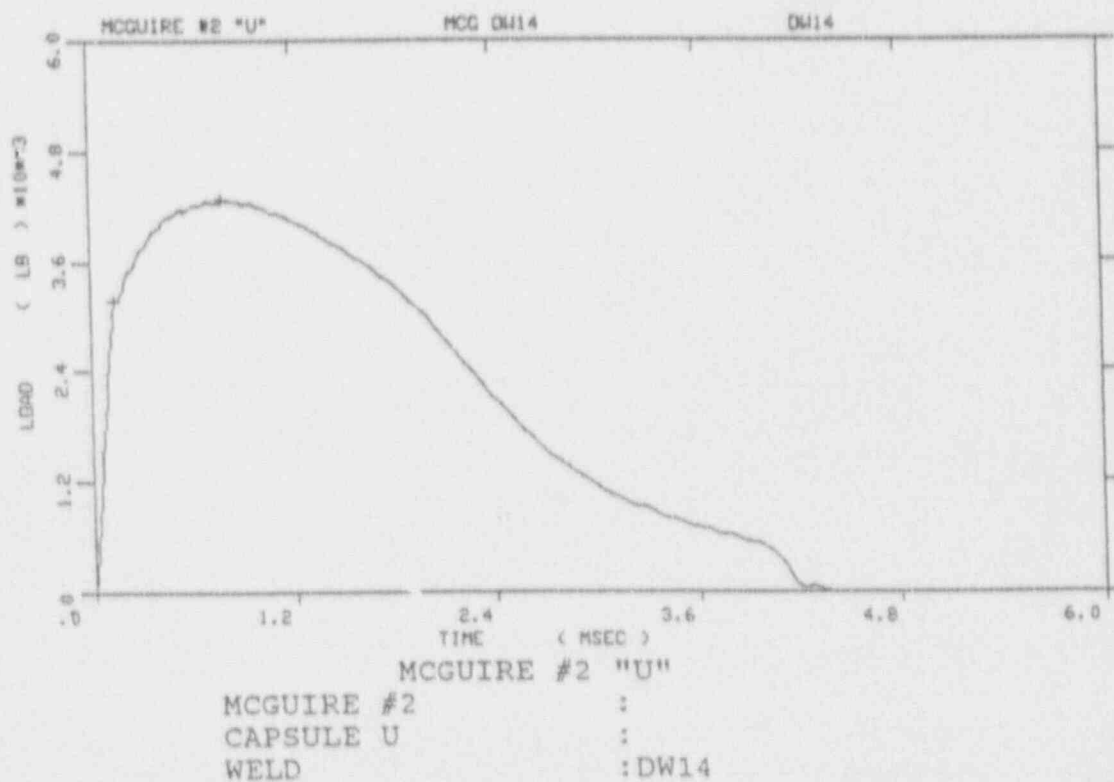
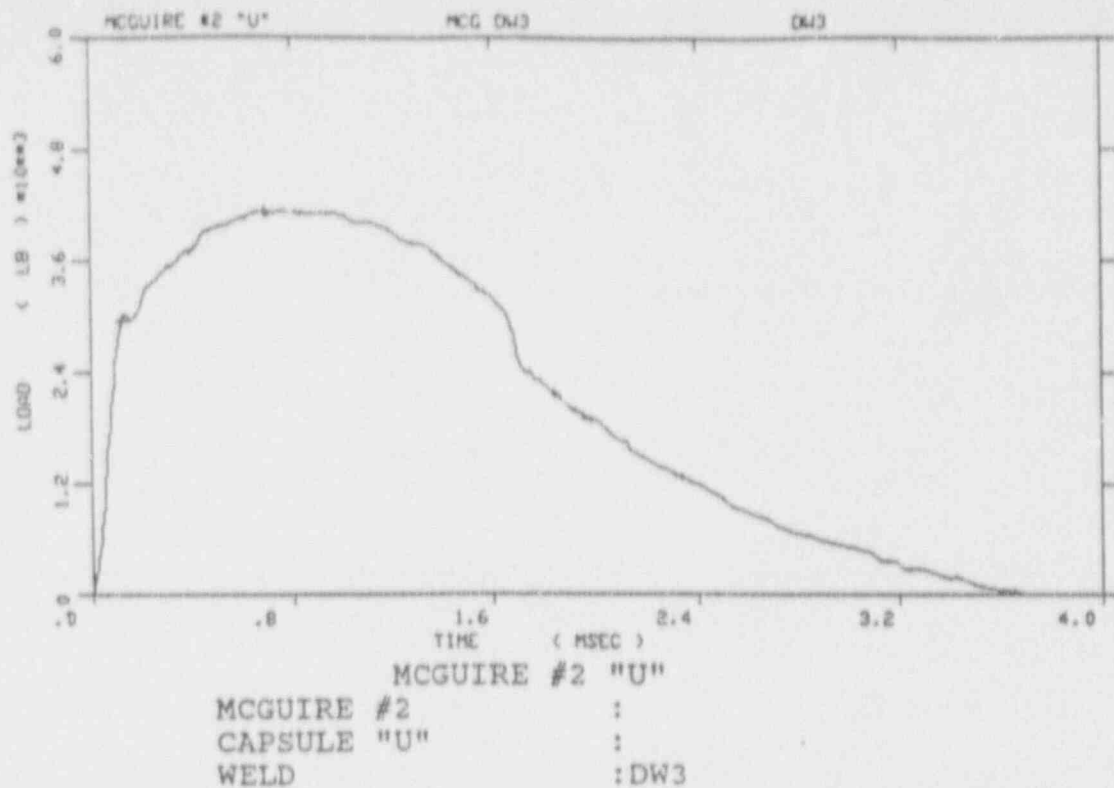


Figure A-23. Load-time records for Specimens DW3 and DW14

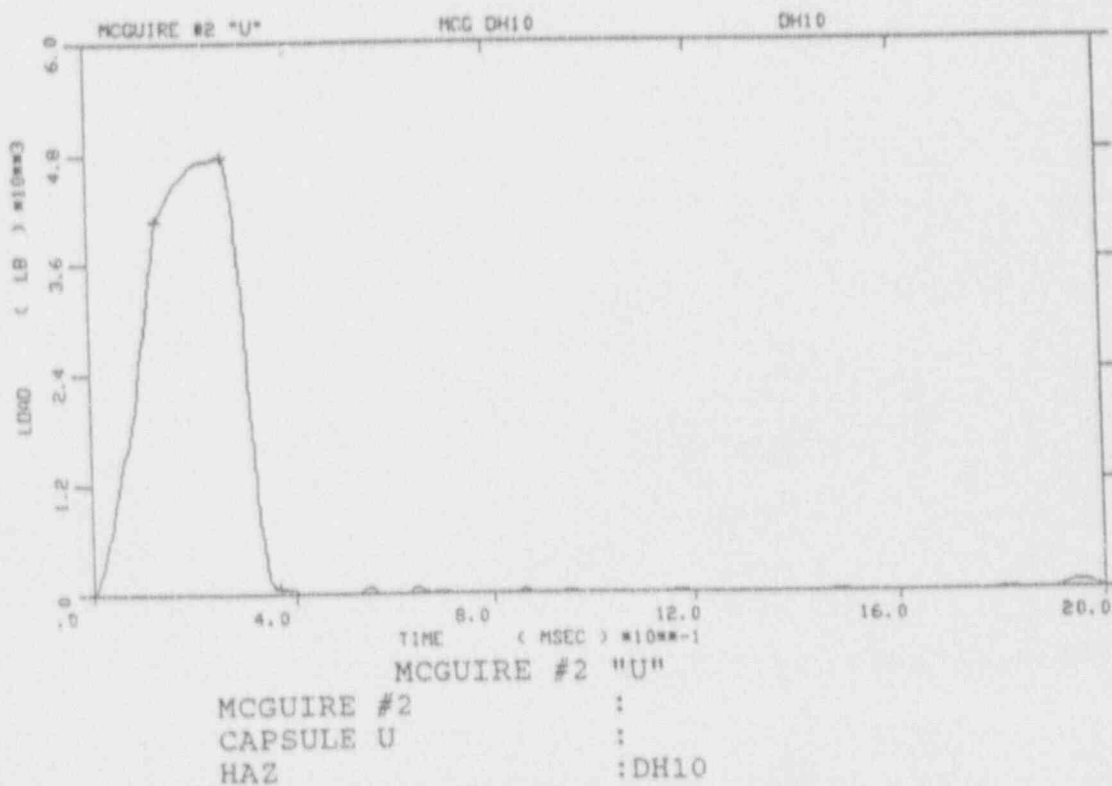
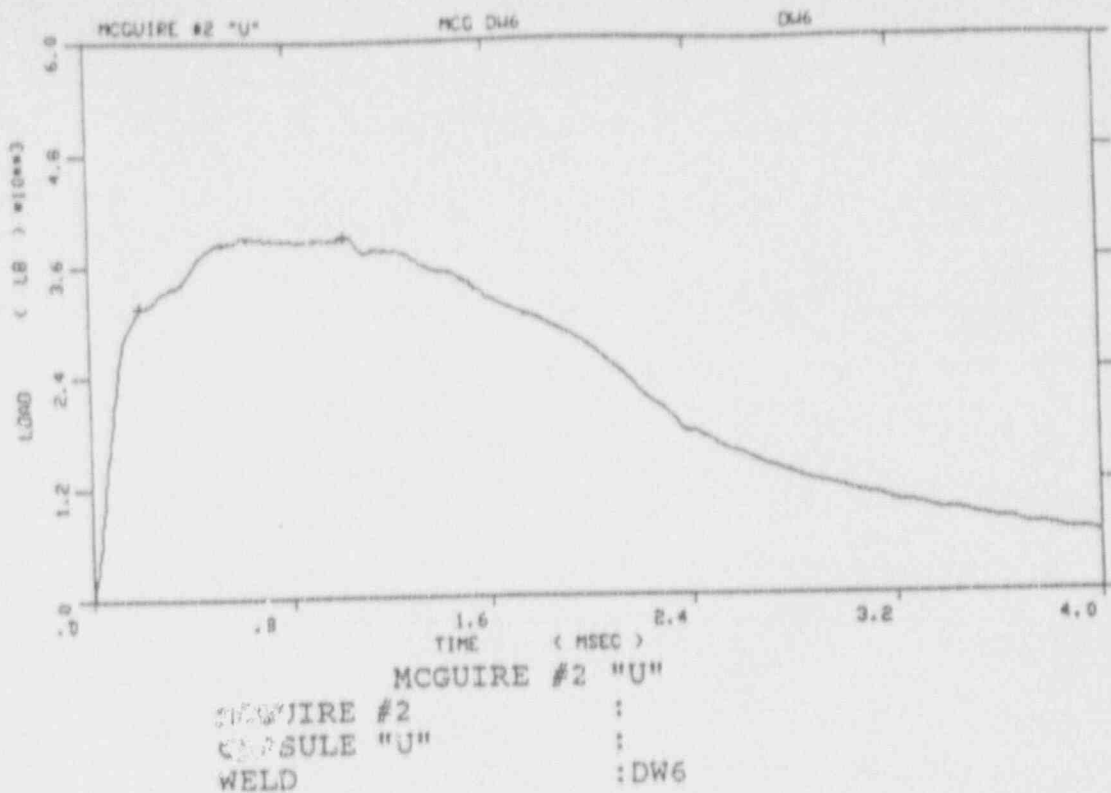


Figure A-24. Load-time records for Specimens DW6 and DH10

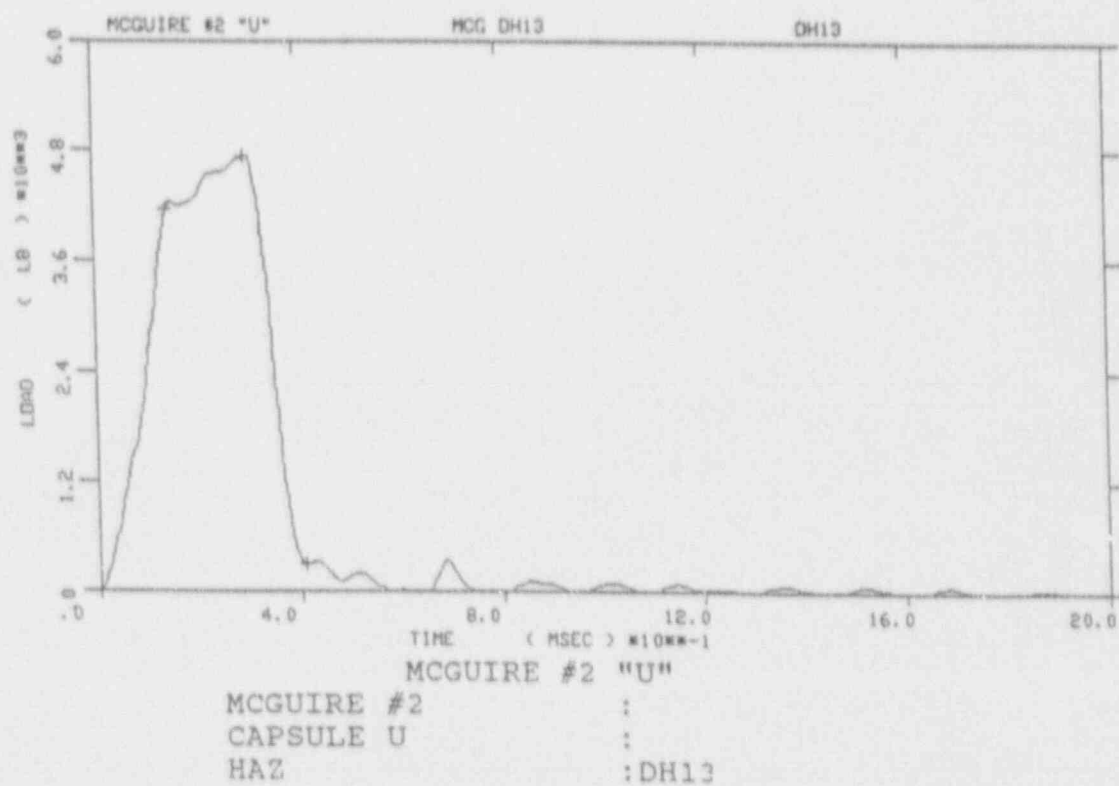
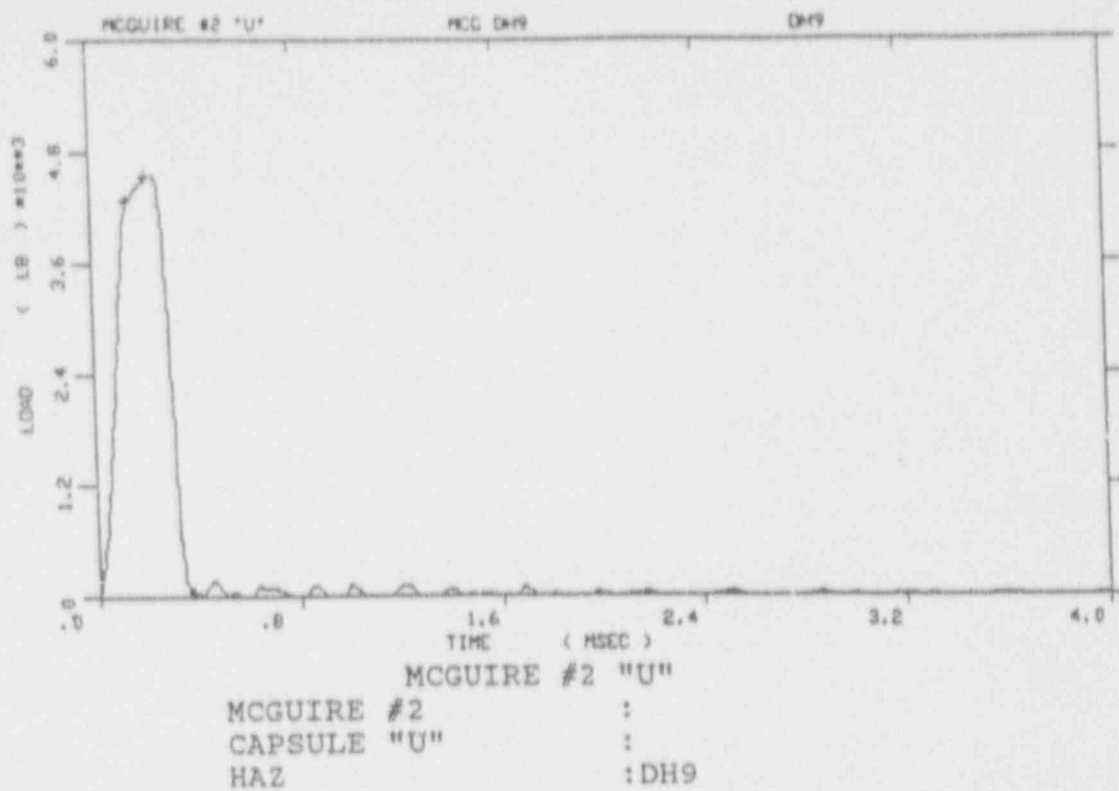


Figure A-25. Load-time records for Specimens DH9 and DH13

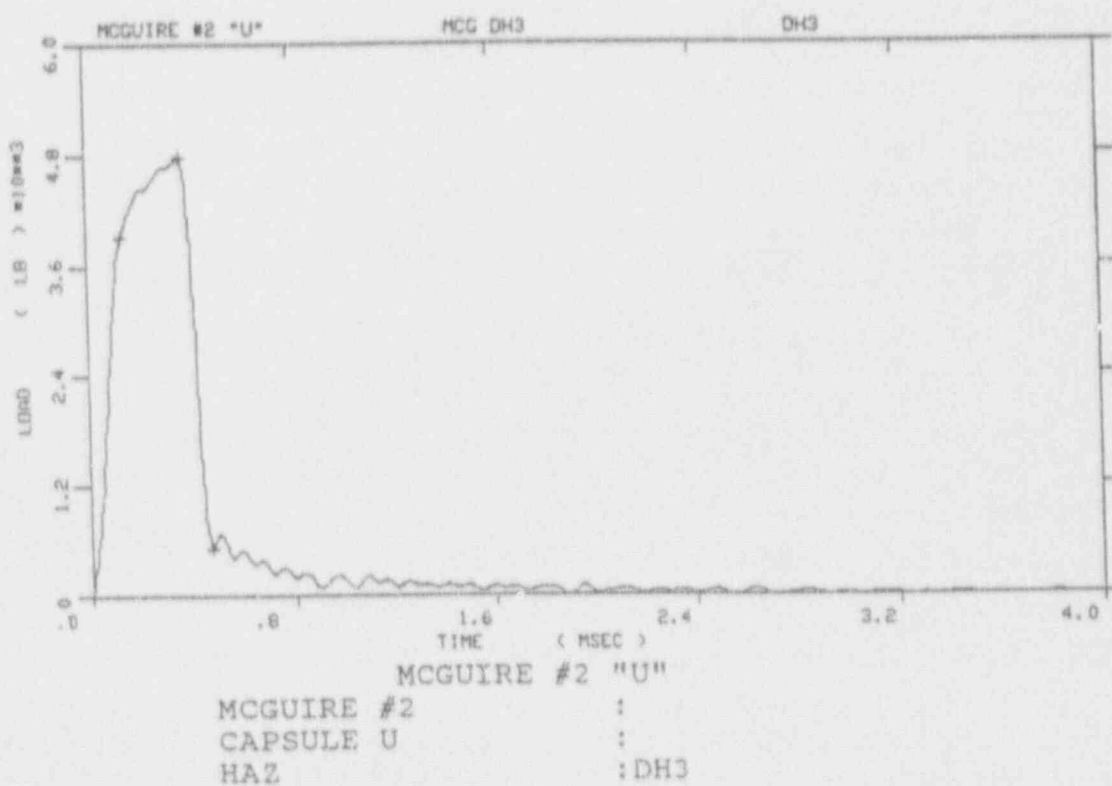
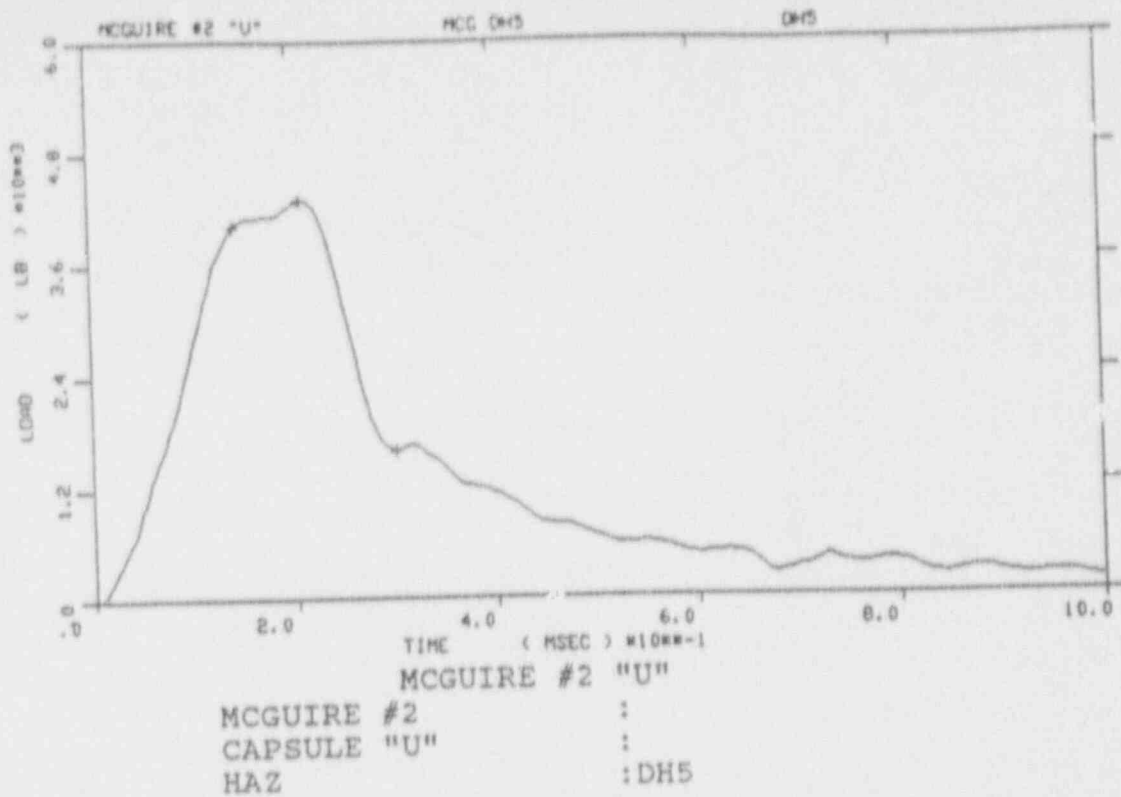


Figure A-26. Load-time records for Specimens DH5 and DH3

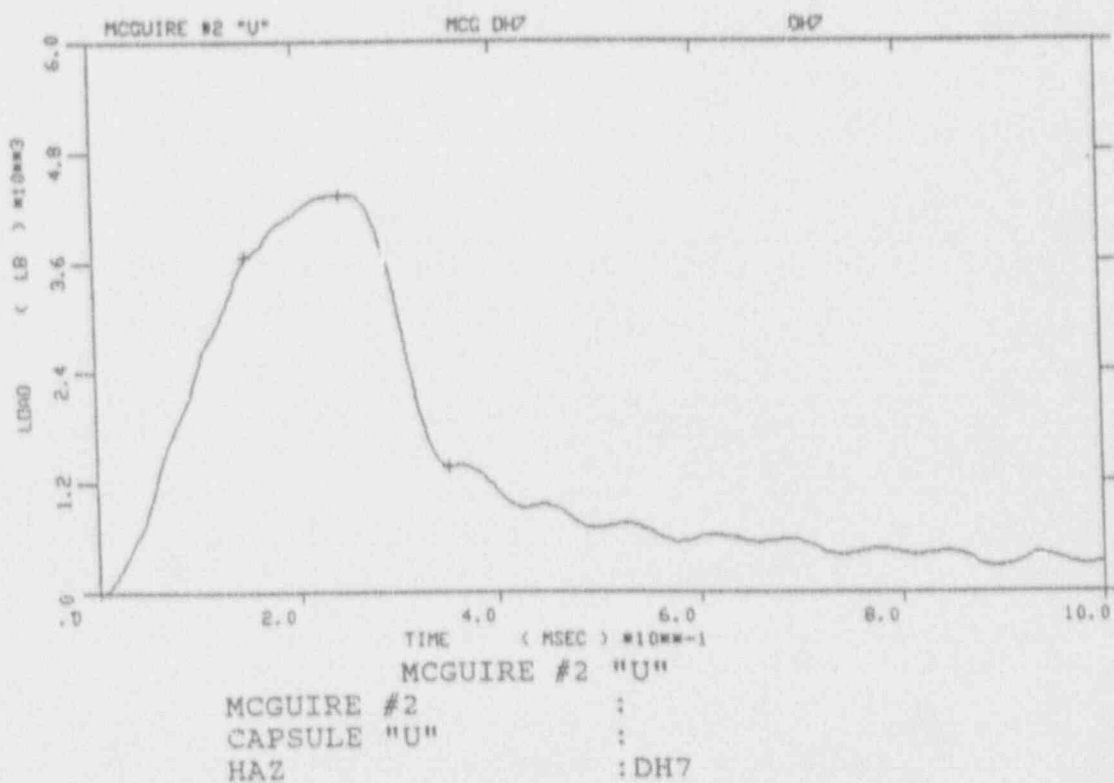
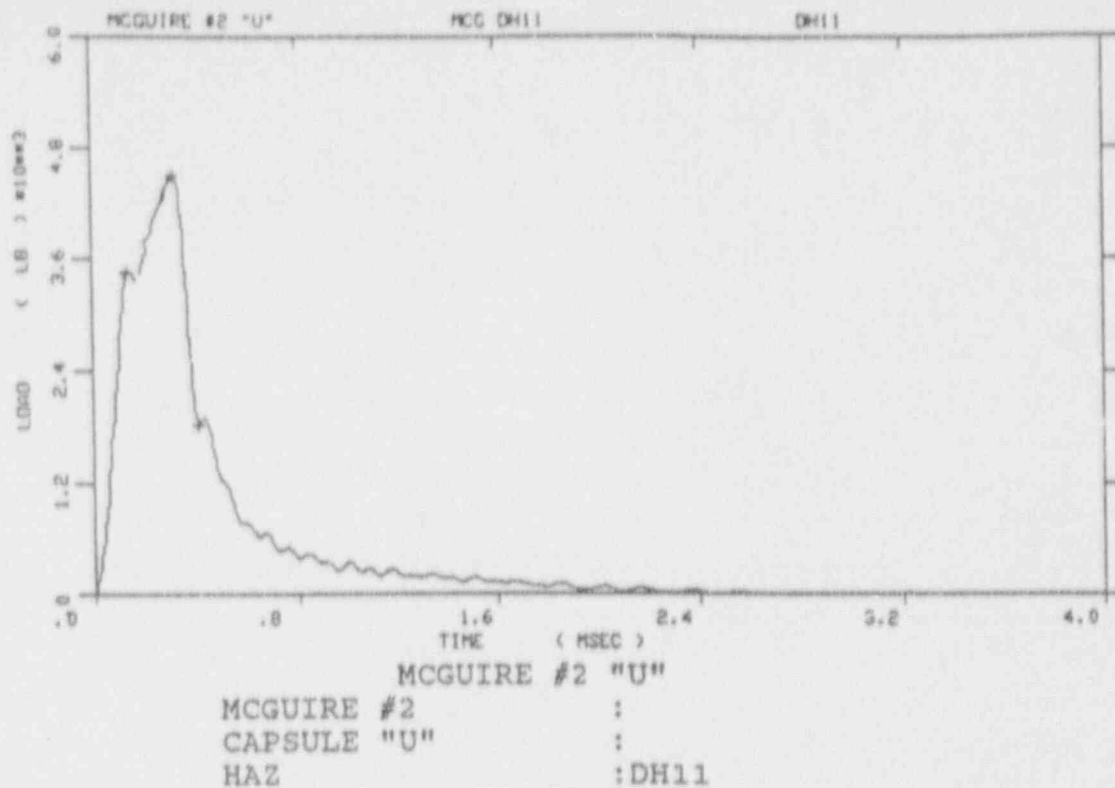


Figure A-27. Load-time records for Specimens DH11 and DH7

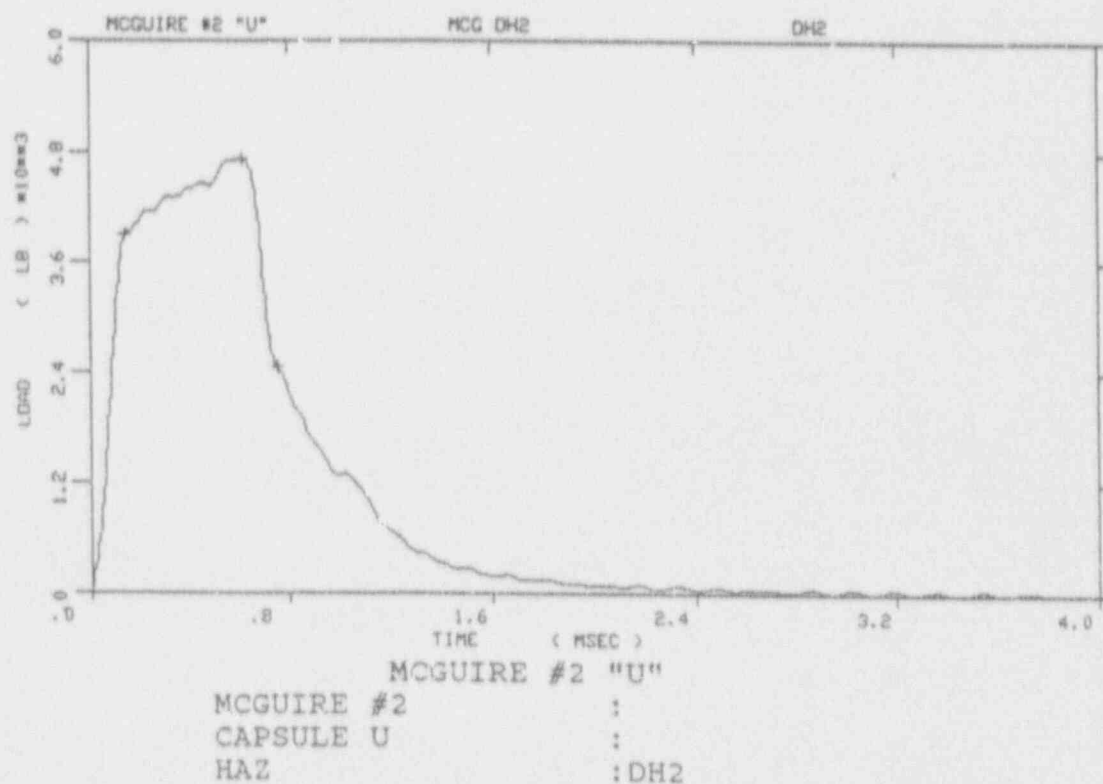
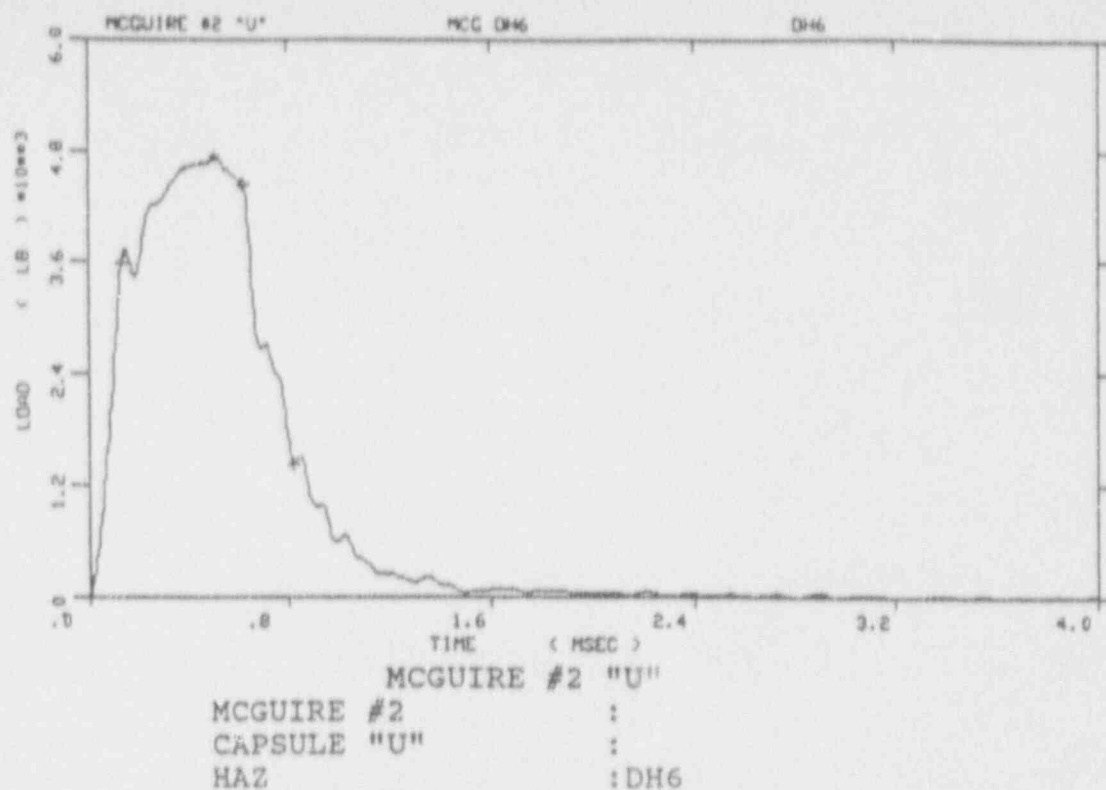


Figure A-28. Load-time records for Specimens DH6 and DH2

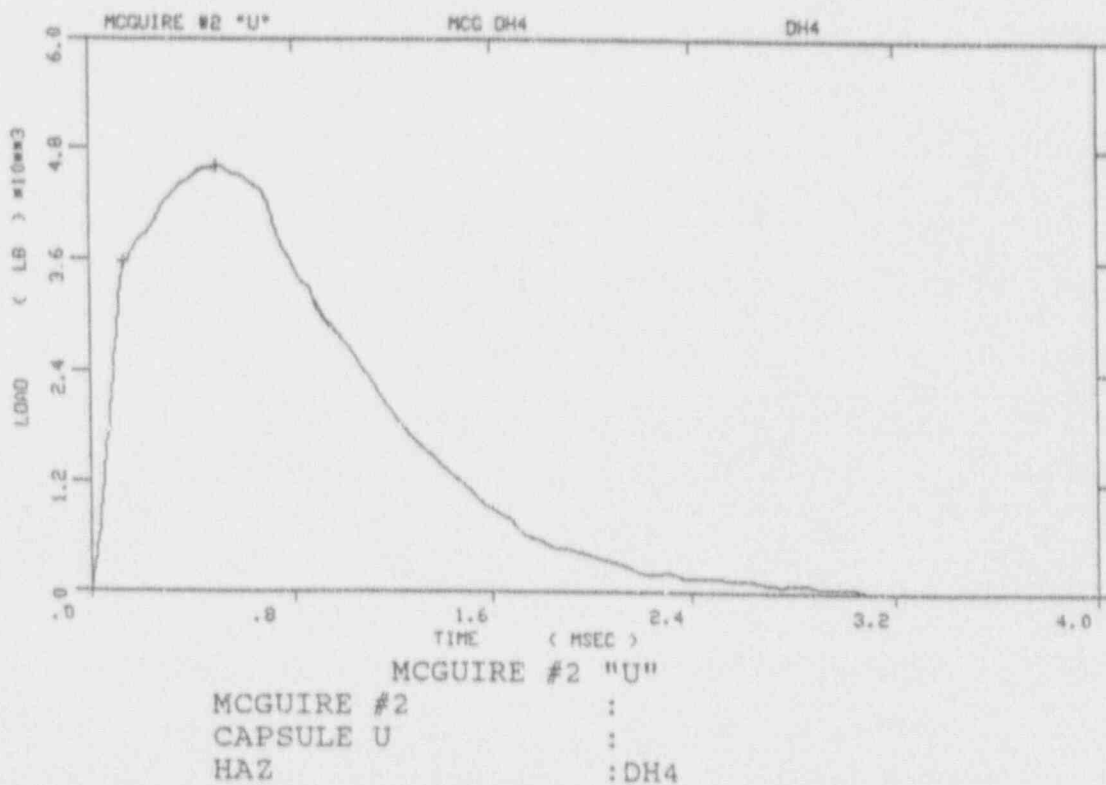
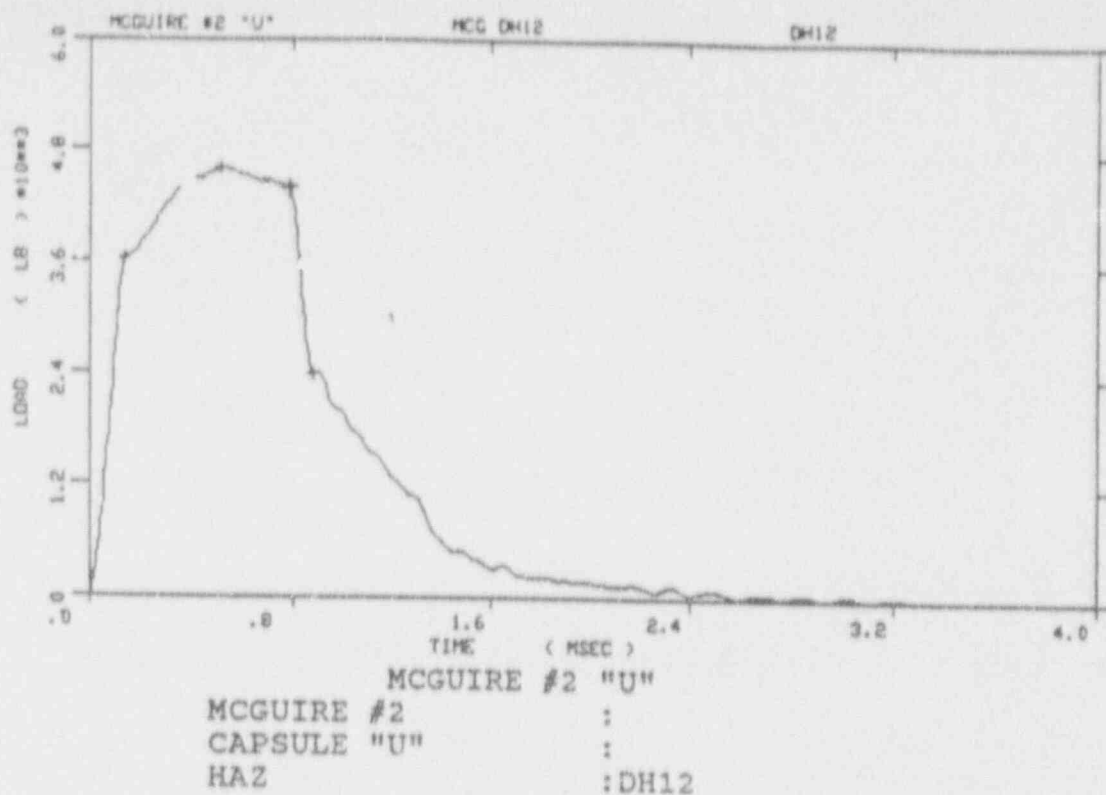
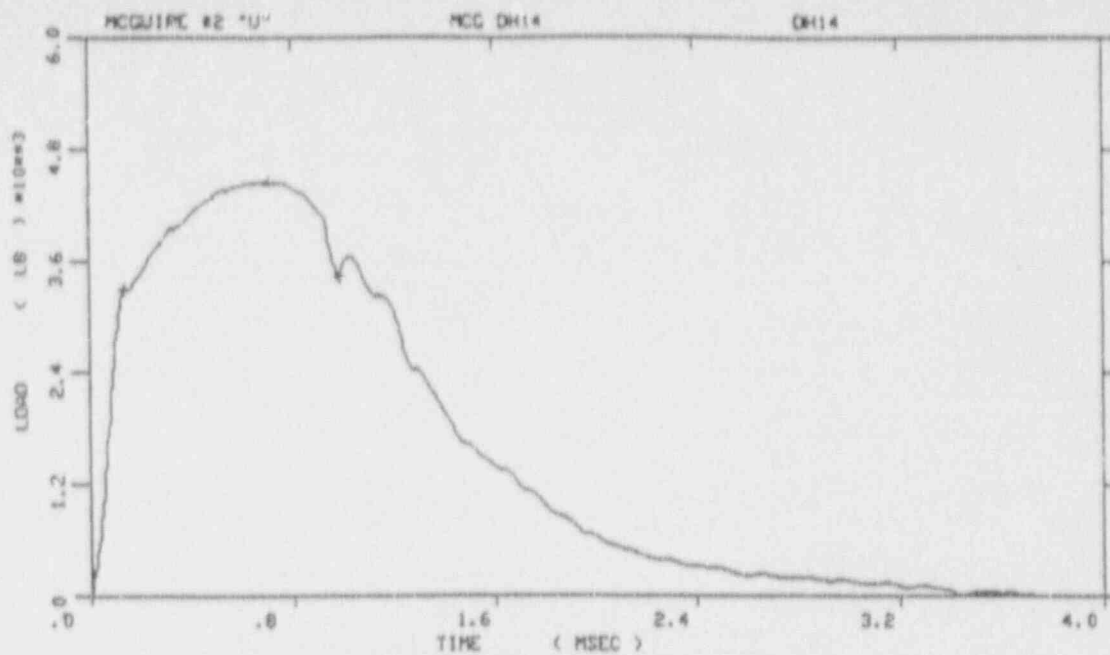
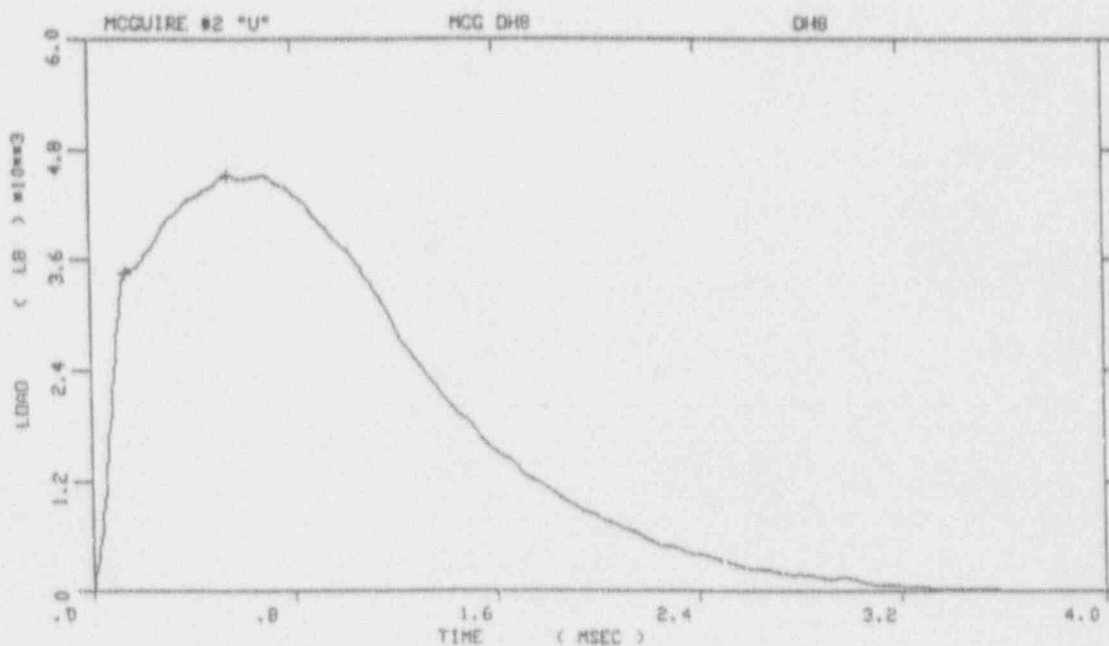


Figure A-29. Load-time records for Specimens DH12 and DH4



McGUIRE #2 "U"
 CAPSULE "U"
 HAZ :DH14



McGUIRE #2 "U"
 CAPSULE U
 HAZ :DH8

Figure A-30. Load-time records for Specimens DH14 and DH8

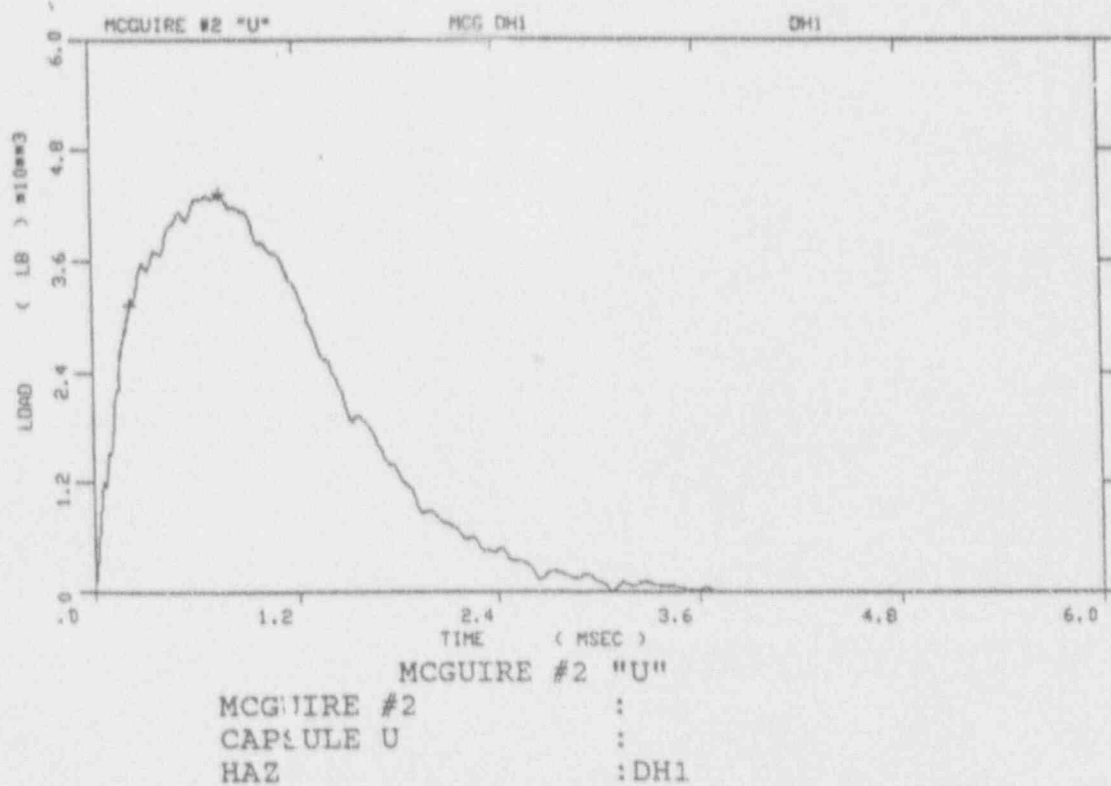
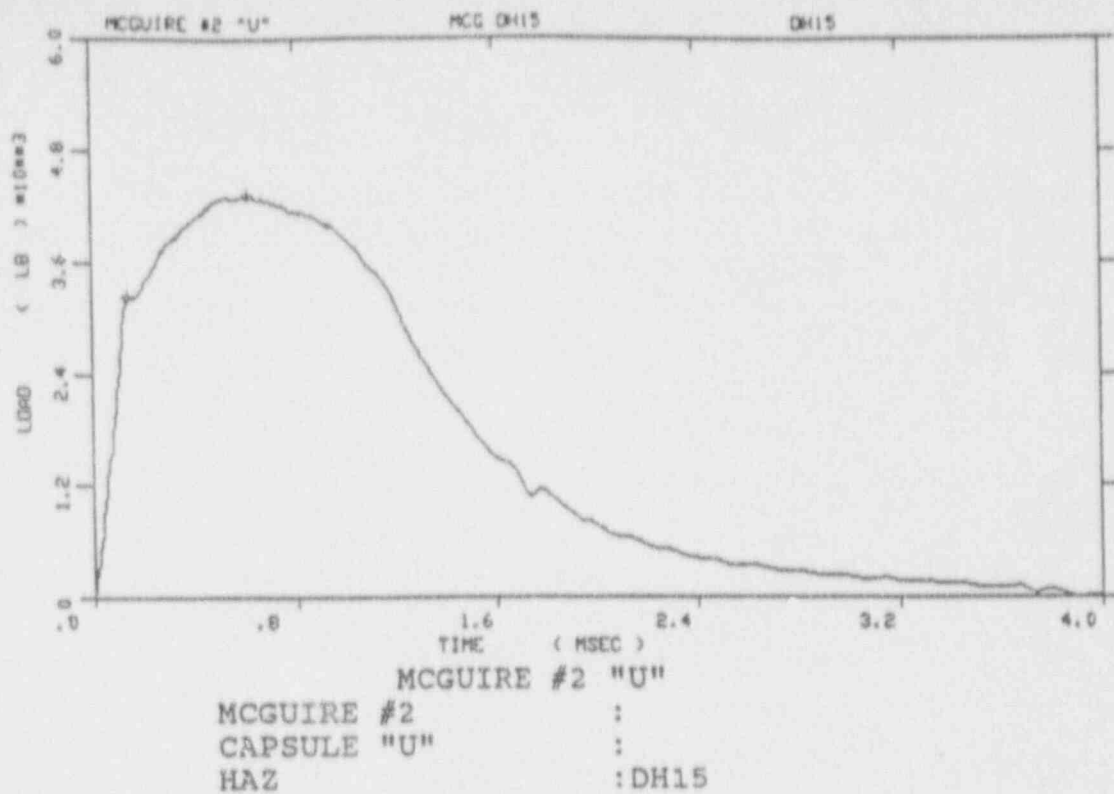


Figure A-31. Load-time records for Specimens DH15 and DH1

APPENDIX B

HEATUP AND COOLDOWN LIMIT CURVES
FOR NORMAL OPERATION FOR
MCGUIRE UNIT 2

TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
	LIST OF ILLUSTRATIONS	B-2
	LIST OF TABLES	B-2
B-1	INTRODUCTION	B-3
B-2	FRACTURE TOUGHNESS PROPERTIES	B-3
B-3	CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS	B-4
B-4	HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES	B-7
B-5	CALCULATION OF ADJUSTED REFERENCE TEMPERATURE	B-9
B-6	REFERENCES	B-15
	ATTACHMENT B1: DATA POINTS FOR HEATUP AND COOLDOWN CURVES WITH MARGINS OF 12°F AND 70 PSIG FOR INSTRUMENTATION ERRORS	B-16
	ATTACHMENT B2: DATA POINTS FOR HEATUP AND COOLDOWN CURVES WITHOUT MARGINS FOR INSTRUMENTATION ERRORS	B-24

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
B-1	McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heat up rates up to 60°F/hr) Applicable for the First 16 EFPY (With Margins of 12°F and 70 psig For Instrumentation Errors)	B-13
B-2	McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 16 EFPY (With Margins of 12°F and 70 psig For Instrumentation Errors)	B-14

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
B-1	McGuire Unit 2 Reactor Vessel Toughness Table (Unirradiated)	B-10
B-2	Summary of Adjusted Reference Temperatures (ART's) at 1/4T and 3/4T Locations for 16 EFPY	B-11
B-3	Calculation of Adjusted Reference Temperatures at 16 EFPY for the Limiting McGuire Unit 2 Reactor Vessel Material - Lower Shell Forging 04	B-12

B-1. INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT} . RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT} . The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)^[B1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values at 1/4T and 3/4T locations (T is the thickness of the vessel at the beltline region).

B-2. FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan^[B2]. The pre-irradiation fracture-toughness properties of the McGuire Unit 2 reactor vessel are presented in Table B-1.

B-3. CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} , for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code^[B3]. The K_{IR} is given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T - RT_{NDT} + 160)] \quad (1)$$

where

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code^[B3] as follows:

$$C * K_{IM} + K_{IT} \leq K_{IR} \quad (2)$$

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT} , and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT} , for the reference flaw are computed. From equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curves in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw.

During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} exceeds K_{IT} , the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various

intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a $1/4 T$ defect at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the $1/4 T$ crack during heatup is lower than the K_{IR} for the $1/4 T$ crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR} 's do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the $1/4 T$ flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a $1/4 T$ deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the

steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curve under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 1983 Amendment to 10CFR50^[B4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for McGuire Unit 2).

Table B-1 indicates that the limiting unirradiated RT_{NDT} of 1°F occurs in the closure head flange of McGuire Unit 2, so the minimum allowable temperature of this region is 121°F. These limits plus additional margins of 12°F and 70 psig are shown in Figures B-1 and B-2 whenever applicable.

B-4. HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor pressure vessel have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section B-3. If pressure readings are measured at locations other than the limiting beltline region, the pressure differences between the pressure transmitter and the limiting beltline region must be accounted for when using the pressure-temperature limit curves herein.

Figure B-1 presents the heatup curves using heatup rates up to 60°F/hr applicable for the first 16 EFPY. Figure B-2 presents the cooldown curves using cooldown rates up to 100°F/hr applicable for the first 16 EFPY. Margins of 12°F and 70 psig for possible instrumentation errors are included in the development of heatup and cooldown curves.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures B-1 and B-2. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in Figure B-1 represents minimum temperature requirements at the leak test pressure specified by applicable codes [B2,B3]. The leak test limit curve was determined by methods of References B2 and B4.

The criticality limit curve shown in Figure B-1 specify pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference B4. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section B-3. The minimum temperature for the inservice hydrostatic leak test for the McGuire Unit 2 reactor vessel at 16 EFPY is 251°F. A vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures B-1 and B-2 define limits for ensuring prevention of nonductile failure for the McGuire Unit 2 reactor vessel.

The data points used to develop the heatup and cooldown pressure-temperature limit curves shown in Figures B-1 and B-2 are presented in Attachment B1. Attachment B2 contains the data points for the 16 EFPY heatup and cooldown pressure-temperature limit curves generated without margins for instrumentation errors.

5. CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2[B1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$\text{ART} = \text{Initial RT}_{\text{NDT}} + \Delta\text{RT}_{\text{NDT}} + \text{Margin} \quad (3)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

$\Delta\text{RT}_{\text{NDT}}$ is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta\text{RT}_{\text{NDT}} = [\text{CF}]f^{(0.28-0.10 \log f)} \quad (4)$$

To calculate $\Delta\text{RT}_{\text{NDT}}$ at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f(\text{depth } x) = f_{\text{surface}}(e^{-.24x}) \quad (5)$$

where x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (4) to calculate $\Delta\text{RT}_{\text{NDT}}$ at the specific depth.

CF ($^{\circ}\text{F}$) is the chemistry factor, obtained from Tables in Reference B1, using the mean values of the copper and nickel content as reported in Table B-1. All materials in the beltline region of McGuire Unit 2 were considered in determining the limiting material. The results of the ART's at 1/4T and 3/4T are summarized in Table B-2. From Table B-2, it can be seen that the limiting material is the lower shell forging 04, for heatup and cooldown curves applicable up to 16 EFPY. Sample calculations to determine the ART values for the lower shell forging at 16 EFPY are shown in Table B-3.

TABLE B-1

MCGUIRE UNIT 2 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Closure Head Flange	--	--	1 (a)
Vessel Flange	--	--	- 4 (a)
Intermediate Shell Forging 05*	0.153	0.793	- 4 (b)
Lower Shell Forging 04*	0.15	0.88	-30 (b)
Circumferential Weld	0.039	0.73	-68 (b)

(a) Initial RT_{NDT} values were estimated per U.S. NRC Standard Review Plan [B2]. These values are used for considering flange requirements for the heatup/cool-down curves [B4].

(b) The initial RT_{NDT} values for the plates and welds are measured values.

* The mean values of copper and nickel content determined as indicated below:

Material	Data Source	Copper (wt. %)	Nickel (wt. %)
Forging 05	Original Mill Test Report	0.16	0.85
	Surveillance Program [B5]	0.16	0.79
	Capsule V Report [B6]	0.14	0.71
	See Table 4-3, Section 4.0	0.151	0.82
	Mean Value	0.153	0.793
Forging 04	Original Mill Test Report	0.15	0.88
Circ. Weld	Original Mill Test Report	0.05	0.70
	Surveillance Program [B5]	0.031	0.73
	Capsule V Report [B6]	0.03	0.66
	See Table 4-3, Section 4.0	0.039	0.765
	See Table 4-3, Section 4.0	0.036	0.747
	See Table 4-3, Section 4.0	0.045	0.776
	Mean Value	0.039	0.730

TABLE B-2
SUMMARY OF ADJUSTED REFERENCE TEMPERATURES (ART's)
IN °F AT 1/4T and 3/4T LOCATIONS FOR 16 EFPY

Material Description	1/4-T	3/4-T
Intermediate Shell Forging 05	131 (86)	100 (63)
Lower Shell Forging 04	104*	73*
Circumferential Weld	33 (-14)	19 (-22)

ART numbers within () are based on chemistry factors calculated using surveillance capsule data.

- * These ART numbers were used to generate heatup and cooldown curves.

TABLE B-3
CALCULATION OF ADJUSTED REFERENCE TEMPERATURES AT 16 EFPY FOR THE LIMITING
MCGUIRE UNIT 2 REACTOR VESSEL MATERIAL - LOWER SHELL FORGING 04

Parameter	Regulatory Guide 1.99 - Revision 2	
	16 EFPY	
	1/4 T	3/4 T
Chemistry Factor, CF ($^{\circ}F$)	115.8	115.8
Fluence, f (10^{19} n/cm ²) (a)	.614	.222
Fluence Factor, ff	.863	.595

$\Delta RT_{NDT} = CF \times ff$ ($^{\circ}F$)	100	69
Initial RT_{NDT} , I ($^{\circ}F$)	-30	-30
Margin, M ($^{\circ}F$) (b)	34	34

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature,	104	73
$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$		

(a) Fluence, f , is based upon f_{surf} (10^{19} n/cm², $E > 1$ Mev) = 1.02 at 16 EFPY. The McGuire Unit 2 reactor vessel wall thickness is 8.465 inches at the beltline region.

(b) Margin is calculated as, $M = 2 [\sigma_I^2 + \sigma_{\Delta}^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term, σ_I , is assumed to be $0^{\circ}F$ since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} term, σ_{Δ} , is $17^{\circ}F$ for the plate, except that σ_{Δ} need not exceed 0.5 times the mean value of ΔRT_{NDT} . σ_{Δ} is $8.5^{\circ}F$ for the plate (half the value) when surveillance data is used.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART AT 16 EFY: 1/4T, 104°F

3/4T, 73°F

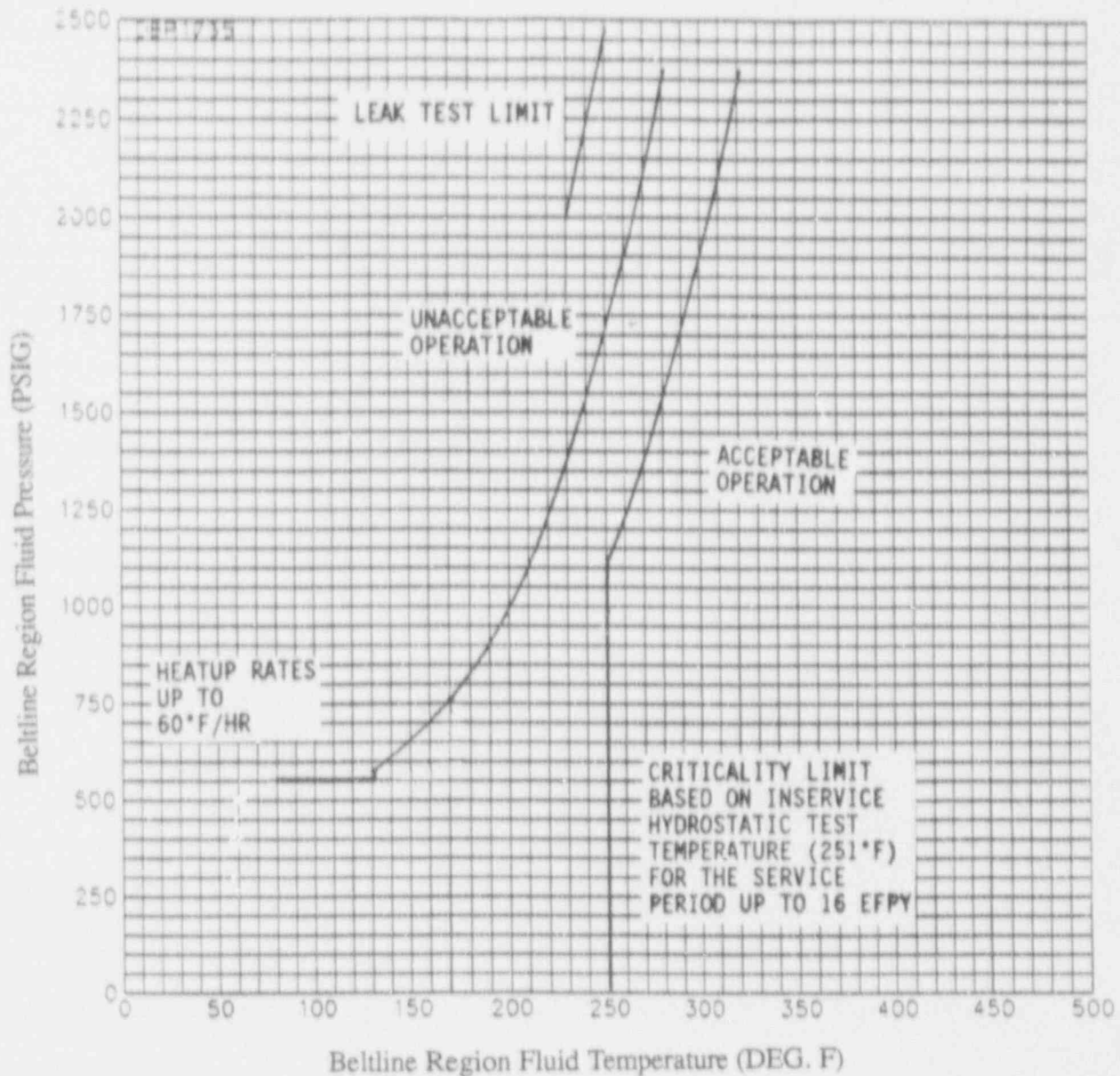


Figure B-1 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heat up rates up to 60°F/hr) Applicable for the First 16 EFY (With Margins of 12°F and 70 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04

LIMITING ART AT 16 EFPY: 1/4T, 104°F

3/4T, 73°F

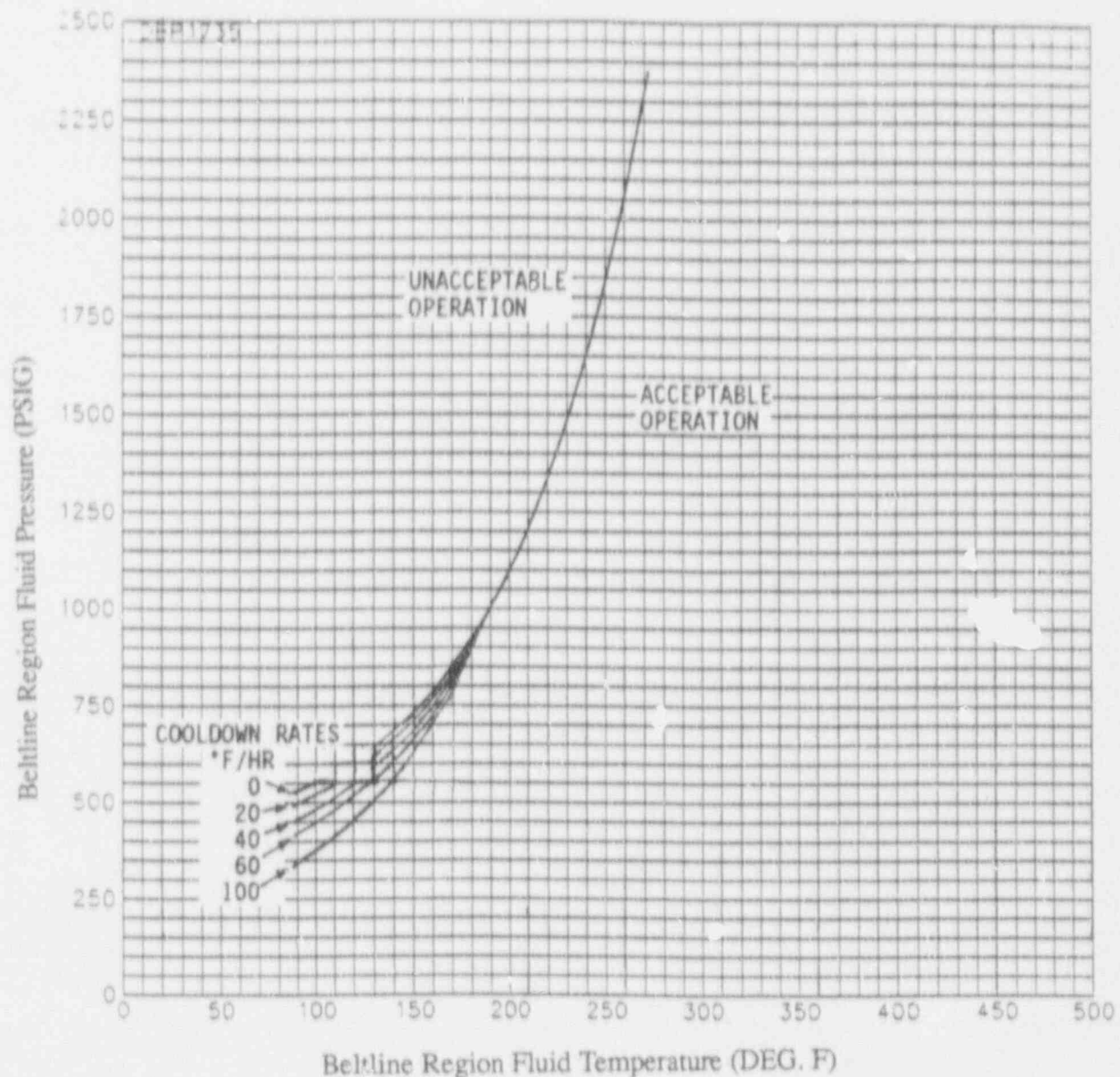


Figure B-2 McGuire Unit 2 Reactor Coolant System Cooldown Limitations
(Cooldown rates up to 100°F/hr) Applicable for the First 16
EFPY (With Margins of 12°F and 70 psig For Instrumentation
Errors)

B-6. REFERENCES

- [B1] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.
- [B2] "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.
- [B3] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure", pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.
- [B4] Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements", U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.
- [B5] K. Koyama and J. A. Davidson, "Duke Power Company William B. McGuire Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-9489, May 1979.
- [B6] S. E. Yanichko, et al., "Analysis of Capsule V from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program", WCAP-11029, January 1986.

ATTACHMENT B1
DATA POINTS FOR HEATUP AND COOLDOWN CURVES
(With Margins of 12°F and 70 psig for Instrumentation Errors)

09/29/92

THE FOLLOWING DATA WERE CALCULATED FOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (16,000 EPY)

PRESSURE (PSI)	TEMPERATURE (DEG. F)
2000	231
2485	251

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ. RT. IN.)
2000	21853	92160
2485	26973	114772

09/29/92

OBP HEATUP CURVES REG. GUIDE 1.99, REV. 2 WITH MARGIN

HEATUP RATE(S) (DEG. F/HR) = 60.0

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2

IRRADIATION PERIOD = 16.000 EFP YEARS

FLAW DEPTH = (1-AWIN)I

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	87.000	557.66	15	157.000	28	222.000
2	92.000	558.18	16	162.000	29	227.000
3	97.000	559.48	17	167.000	30	232.000
4	102.000	560.15	18	172.000	31	237.000
5	107.000	548.54	19	177.000	32	242.000
6	112.000	550.27	20	182.000	33	247.000
7	117.000	558.20	21	187.000	34	251.000
8	122.000	562.84	22	192.000	35	257.000
9	127.000	578.07	23	197.000	36	262.000
10	132.000	585.56	24	202.000	37	267.000
11	137.000	600.57	25	207.000	38	272.000
12	142.000	617.73	26	212.000	39	277.000
13	147.000	637.09	27	217.000	40	282.000
14	152.000	658.76				

* Flange requirement 551 psi

DRP COOLDOWN CURVES REG. GUIDE 1.99, REV. 2 WITH MARGIN

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 16,000 EFP YEARS

FLAW DEPTH = 0.01 IN

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		
1	87.000	331.86	8	122.000	458.53	15	157.000	575.68
2	92.000	345.93	9	127.000	482.89	16	162.000	717.30
3	97.000	361.26	10	132.000	509.28	17	167.000	762.07
4	102.000	377.74	11	137.000	537.69	18	172.000	810.28
5	107.000	395.68	12	142.000	568.44	19	177.000	862.25
6	112.000	415.04	13	147.000	601.54	20	182.000	918.04
7	117.000	435.91	14	152.000	637.19	21	187.000	978.39

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16,000 EFP YEARS

FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	87.000	411.51	9	127.000	543.10	16	162.000	744.74
2	92.000	423.93	10	132.000	555.98 *	17	167.000	783.03
3	97.000	437.28	11	137.000	590.41	18	172.000	824.23
4	102.000	451.76	12	142.000	616.92	19	177.000	868.62
5	107.000	467.42	13	147.000	645.38	20	182.000	916.38
6	112.000	484.18	14	152.000	675.99	21	187.000	967.82
7	117.000	502.40	15	157.000	703.18	22	192.000	1023.16
8	122.000	522.02						

* Flange requirement 551 psi

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN 1

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	87.000	450.68	9	127.000	574.00 }	16	162.000	761.81
2	92.000	462.38	10	132.000	595.01 }	17	167.000	797.43
3	97.000	475.03	11	137.000	618.26	18	172.000	835.65
4	102.000	488.53	12	142.000	642.85	19	177.000	876.81
5	107.000	503.22	13	147.000	669.44	20	182.000	921.07
6	112.000	519.04	14	152.000	697.97	21	187.000	968.71
7	117.000	535.98	15	157.000	728.65	22	192.000	1019.94
8	122.000	554.94 *						

* Flange requirement 551 psi

DBP COOLDOWN CURVES REG. GUIDE 1.99, REV. 2 WITH MARGIN

08/29/92

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	
1	87.000	489.40	9	127.000	665.47	* }	17	167.000	813.80
2	92.000	500.46	10	132.000	628.25		18	172.000	849.39
3	97.000	512.40	11	137.000	646.74		19	177.000	887.70
4	102.000	525.10	12	142.000	669.79		20	182.000	928.80
5	107.000	538.93	13	147.000	694.54		21	187.000	973.02
6	112.000	553.89	14	152.000	721.06		22	192.000	1020.54
7	117.000	569.81	15	157.000	749.87		23	197.000	1071.67
8	122.000	586.86	16	162.000	780.58				

* Flange requirement 551 psi

DBP COOLDOWN CURVES REG. GUIDE 1.99 REV. 2 WITH MARGIN

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 16,000 EFP YEARS
FLAW DEPTH = ADMIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	
1	87.000	527.68	14	152.000	745.52	217.000	1299.37
2	92.000	538.18	15	157.000	772.17	222.000	1367.18
3	97.000	549.46	16	162.000	801.07	227.000	1439.53
4	102.000	561.50	17	167.000	831.90	232.000	1517.55
5	107.000	574.50	18	172.000	865.01	237.000	1600.92
6	112.000	588.50	19	177.000	900.83	242.000	1690.28
7	117.000	603.61	20	182.000	939.16	247.000	1786.04
8	122.000	618.76	21	187.000	980.31	252.000	1888.47
9	127.000	633.89	22	192.000	1024.50	257.000	1998.13
10	132.000	649.08	23	197.000	1071.98	262.000	2115.31
11	137.000	665.43	24	202.000	1123.01	267.000	2240.65
12	142.000	682.43	25	207.000	1177.72	272.000	2374.19
13	147.000	720.49	26	212.000	1236.50		

* Flange requirement 551 psi

ATTACHMENT B2
DATA POINTS FOR HEATUP AND COOLDOWN CURVES
(Without Margins for Instrumentation Errors)

THE FOLLOWING DATA WERE CALCULATED FOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (16,000 EFY)

PRESSURE (PSI)	TEMPERATURE (DEG. F)
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2000	215
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2485	236
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PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ. RT. IN.)
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2000	21114	88944
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2485	26234	111470
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09/30/92

DBP HEATUP CURVES, REG. GUIDE 1.99, REV. 2 WITHOUT MARGIN

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG. F/HR) = 60.0

IRRADIATION PERIOD = 16,000 EFP YEARS
FLAW DEPTH * (1-AOWIN)

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	518.46	14	150.000	27	215.000
2	90.000	518.46	15	155.000	28	220.000
3	95.000	618.54	16	160.000	29	225.000
4	100.000	620.27	17	165.000	30	230.000
5	105.000	625.59	18	170.000	31	235.000
6	110.000	632.84	19	175.000	32	240.000
7	115.000	642.87	20	180.000	33	245.000
8	120.000	655.56	21	185.000	34	250.000
9	125.000	670.57	22	190.000	35	255.000
10	130.000	687.73	23	195.000	36	260.000
11	135.000	707.09	24	200.000	37	265.000
12	140.000	728.76	25	205.000	38	270.000
13	145.000	752.60	26	210.000		

* Flange requirement 621 psi

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 16.000 EFF YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	619.46	13	145.000	842.17	25	205.000	1369.37
2	90.000	634.60	14	150.000	871.07	26	210.000	1437.18
3	95.000	644.60	15	155.000	901.90	27	215.000	1509.53
4	100.000	658.59	16	160.000	935.01	28	220.000	1587.55
5	105.000	673.61	17	165.000	970.83	29	225.000	1670.92
6	110.000	689.78	18	170.000	1009.16	30	230.000	1760.28
7	115.000	707.09	19	175.000	1050.31	31	235.000	1856.04
8	120.000	725.83	20	180.000	1094.50	32	240.000	1958.47
9	125.000	745.78	21	185.000	1141.98	33	245.000	2068.13
10	130.000	767.43	22	190.000	1193.01	34	250.000	2185.31
11	135.000	790.49	23	195.000	1247.72	35	255.000	2310.65
12	140.000	815.52	24	200.000	1306.50	36	260.000	2444.19

* Flange requirement 621 psi

09/30/92

DBP COOLDOWN CURVES, REG. GUIDE 1.99, REV. 2 WITHOUT MARGIN

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG. F / HR COOLDOWN)

IRRADIATION PERIOD = 16,000 EFP YEARS

FLAW DEPTH = ADMIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	582.40	8	120.000	695.25 *	15	155.000	883.80
2	90.000	595.10	9	125.000	716.74	16	160.000	919.39
3	95.000	608.93	10	130.000	739.79	17	165.000	957.70
4	100.000	623.80	11	135.000	764.54	18	170.000	998.80
5	105.000	639.84 *	12	140.000	791.06	19	175.000	1043.02
6	110.000	656.00 *	13	145.000	819.87	20	180.000	1090.54
7	115.000	678.47 *	14	150.000	850.58	21	185.000	1141.67

* Flange requirement 621 psi

09/30/92

DBP COOLDOWN CURVES, REG. GUIDE 1.99, REV. 2 WITHOUT MARGIN

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16,000 EFP YEARS
FLAW DEPTH = ADMIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	545.03	8	120.000	15	155.000
2	90.000	558.53	9	125.000	16	160.000
3	95.000	573.22	10	130.000	17	165.000
4	100.000	589.04	11	135.000	18	170.000
5	105.000	605.98	12	140.000	19	175.000
6	110.000	624.04	13	145.000	20	180.000
7	115.000	644.00	14	150.000		

* Flange requirement 621 psi

09/30/92

DBP COOLDOWN CURVES, REG. GUIDE 1.99, REV. 2 WITHOUT MARGIN

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16 000 EFP YEARS
FLAW DEPTH = ADMIN T

	INDICATED TEMPERATURE (DEG F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG F)	INDICATED PRESSURE (PSI)	INDICATED TEMPERATURE (DEG F)	INDICATED PRESSURE (PSI)
1	85.000	507.28	8	120.000	15	853.03
2	90.000	521.76	9	125.000	16	894.23
3	95.000	537.42	10	130.000	17	938.62
4	100.000	554.18	11	135.000	18	986.38
5	105.000	572.40	12	140.000	19	1037.82
6	110.000	592.02	13	145.000	20	1093.16
7	115.000	613.10	14	150.000		

* Flange requirement 621 psi.

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = ADMIN T

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSI)
1	85.000	431.26	8	120.000	579.28	14	150.000	787.30
2	90.000	447.74	9	125.000	607.65	15	155.000	832.07
3	95.000	465.68	10	130.000	638.44	16	160.000	880.28
4	100.000	485.04	11	135.000	671.54	17	165.000	932.25
5	105.000	505.91	12	140.000	707.19	18	170.000	988.04
6	110.000	528.53	13	145.000	745.68	19	175.000	1048.39
7	115.000	552.89						

APPENDIX C

Upper Shelf Energy Evaluation

The low upper shelf Charpy energy concern is associated with the determination of acceptable reactor vessel toughness when the vessel is operating at normal temperatures of 525°F or higher (the "upper shelf" of measured fracture toughness).

In 1973, in an effort to improve the quality of reactor pressure vessel integrity and to base the assessment of vessel integrity on a theoretical rather than an empirical basis, the concept of fracture mechanics techniques was implemented through ASME activities and NRC regulation. These requirements are included in 10CFR50, Appendix G, "Fracture Toughness Requirements". 10CFR50 Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the upper shelf energy of any of the reactor vessel materials would be predicted to drop below 50 ft-lb as measured by Charpy V-notch specimen testing. One source of the irradiated materials that can be used to make this prediction on a plant specific basis is the reactor vessel surveillance capsule program. The surveillance capsules contain material identical to or representative of the critical reactor vessel materials that would be irradiated during plant operation. The surveillance capsules are attached inside the reactor vessel at locations designed to provide a higher irradiation rate than the reactor vessel itself, thus providing an irradiation "lead" factor that allows for prediction of future vessel irradiation damage.

There are three different methods that can be applied to estimate the upper shelf energy:

- o Use of Revision 2 of Regulatory Criteria 1.99;
- o Use of Plant Specific Surveillance Capsule data; and
- o Use of alternative industry trend curves.

In the event that the 50 ft-lb requirements cannot be satisfied as stated in 10CFR50, Appendix G, or by alternative procedures acceptable to the NRC, a reactor may continue to operate provided all the following requirements of 10CFR50, Appendix G, paragraph V.C are satisfied:

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B of Appendix G is made, and any flaws are characterized according to Section XI of the ASME Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation.
2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation must be obtained from results of supplemental fracture toughness tests.
3. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

The issue of low upper shelf toughness has been a subject of active concern with the NRC for several years. In 1981, NUREG 0744 was issued. This document suggested an analysis procedure for determining the margins of safety present in reactor vessels. However, no definite criteria for acceptance margins were given. In 1982 a formal request was issued to the ASME Code Section XI committee to develop more specific criteria. In 1989 the ASME completed development of alternative criteria to define acceptable reactor vessel integrity when the Charpy upper shelf energy drops below 50 ft-lb. These criteria were transmitted to the NRC for their evaluation and incorporation into the regulations governing reactor vessel integrity.

Upper shelf energy values for each of the beltline region materials in the McGuire Unit 2 reactor vessel were calculated for 32 and 48 effective full power years (EFPY) of operation. The results show that all of the beltline region materials will remain above the 50 ft-lb screening criteria at 32 and 48 EFPY as indicated in Table C-1.

TABLE C-1
MCGUIRE UNIT 2 UPPER SHELF ENERGY VALUES (ft-lbs)

Material Description	Initial USE	32 EFPY	48 EFPY
Lower Shell Forging 04	141	106	103
Inter. Shell Forging 05	94	71	69
Circumferential Weld	133	106	104