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Analysis of Capsule 97° from the Florida Power & Light Company
St. Lucie Unit 2 Reactor Vessel Radiation Surveillance Program

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Westinghouse

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Analysis of Capsule 97° from the Florida Power & Light Company St. Lucie Unit 2 Reactor Vessel Radiation Surveillance Program

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EXECUTIVE SUMMARY

The purpose of this report is to document the testing results of surveillance Capsule 97° from St. Lucie Unit 2. Capsule 97° was removed at 25.55 EFPY and post-irradiation mechanical tests of the Charpy V-notch and tensile specimens were performed. A fluence evaluation utilizing the neutron transport and dosimetry cross-section libraries was derived from the ENDF/B-VI database. Capsule 97° received a fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) after irradiation to 25.55 EFPY. The peak clad/base metal interface vessel fluence after 25.55 EFPY of plant operation was 1.73 x 10^{19} n/cm^2 (E > 1.0 MeV).

This evaluation led to the following conclusions: 1) The measured percent decreases in upper-shelf energy for the surveillance plate (longitudinal orientation) and weld materials contained in St. Lucie Unit 2 Capsule 97° are less than the Regulatory Guide 1.99, Revision 2 [Ref. 1] predictions. The measured decrease for the surveillance plate (transverse orientation) material is equivalent to the Regulatory Guide 1.99, Revision 2 [Ref. 1] prediction. 2) The St. Lucie Unit 2 surveillance plate, with consideration of all data or considering only the transverse orientation Charpy data points, and weld (Heat # 83637) data are judged to be credible. It is standard to use all surveillance plate data in subsequent reactor vessel integrity evaluations. However, a transverse orientation only credibility evaluation is presented in this report as an additional analysis of the data. This credibility evaluation can be found in Appendix D. 3) With consideration of surveillance data, all beltline materials exhibit adequate upper-shelf energy levels for continued safe plant operation and are predicted to maintain an upper-shelf energy greater than 50 ft-lb through end-of-license (55 EFPY) as required by 10 CFR 50, Appendix G [Ref. 2]. The upper-shelf energy evaluation is presented in Appendix E.

Lastly, a brief summary of the Charpy V-notch testing can be found in Section 1. All Charpy V-notch data was plotted using a symmetric hyperbolic tangent curve-fitting program.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance Capsule 97°, the third capsule removed and tested from the St. Lucie Unit 2 reactor pressure vessel, led to the following conclusions:

- Charpy V-notch test data were plotted using a symmetric hyperbolic tangent curve-fitting program. Appendix C presents the CVGRAPH, Version 6.0, Charpy V-notch plots for Capsule 97° and previous capsules, along with the program input data.

- Capsule 97° received an average fast neutron fluence (E > 1.0 MeV) of 2.25 x 10^{19} n/cm^2 after 25.55 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel Intermediate Shell Plate M-605-1 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 124.3°F and an irradiated 50 ft-lb transition temperature of 173.3°F. This results in a 30 ft-lb transition temperature increase of 132.7°F and a 50 ft-lb transition temperature increase of 140.1°F for the longitudinally oriented specimens.

- Irradiation of the reactor vessel Intermediate Shell Plate M-605-1 Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (transverse orientation), resulted in an irradiated 30 ft-lb transition temperature of 158.0°F and an irradiated 50 ft-lb transition temperature of 219.8°F. This results in a 30 ft-lb transition temperature increase of 127.6°F and a 50 ft-lb transition temperature increase of 148.3°F for the transversely oriented specimens.

- Irradiation of the Surveillance Program Weld Metal (Heat # 83637) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of -25.7°F and an irradiated 50 ft-lb transition temperature of 16.2°F. This results in a 30 ft-lb transition temperature increase of 24.8°F and a 50 ft-lb transition temperature increase of 28.7°F.

- Irradiation of the Heat-Affected Zone (HAZ) Material Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of -137.0°F and an irradiated 50 ft-lb transition temperature of 64.1°F. This results in a 30 ft-lb transition temperature increase of -103.9°F and a 50 ft-lb transition temperature increase of 46.2°F. It is noted that the scatter in the HAZ data was significant and the resulting CVGRAPH, Version 6.0 Charpy V-notch symmetric hyperbolic tangent curve-fit plots are not an ideal rendition of the data. However, this is inconsequential since HAZ material is not considered limiting as compared to the base and weld materials.

- The average upper-shelf energy of Intermediate Shell Plate M-605-1 (longitudinal orientation) resulted in an average energy decrease of 26 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 108 ft-lb for the longitudinally oriented specimens.

- The average upper-shelf energy of Intermediate Shell Plate M-605-1 (transverse orientation) resulted in an average energy decrease of 25 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 78 ft-lb for the transversely oriented specimens.

- The average upper-shelf energy of the Surveillance Program Weld Metal (Heat # 83637) Charpy specimens resulted in an average energy decrease of 20 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 95 ft-lb for the weld metal specimens.

- The average upper-shelf energy of the HAZ Material Charpy specimens resulted in an average energy decrease of 12 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 93 ft-lb for the HAZ Material.

- Comparisons of the measured 30 ft-lb shift in transition temperature values and upper-shelf energy decreases to those predicted by Regulatory Guide 1.99, Revision 2 [Ref. 1] for the St. Lucie Unit 2 reactor vessel surveillance materials are presented in Table 5-10.

  Standard Reference Material (SRM) HSST 01 Charpy specimens were not included in the St. Lucie Unit 2 Capsule 97°. However, the SRM HSST 01 Charpy specimens were reanalyzed in this report. The SRM HSST 01 material was contained in Capsule 263°, which was irradiated to a neutron fluence of 1.00 x 10^{19} n/cm^2 (E > 1.0 MeV). The results of the SRM HSST 01 reanalysis will be included in Table 5-10 and shown in Figures 5-13 through 5-15.

  - Irradiation of the SRM HSST 01 Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 157.1°F and an irradiated 50 ft-lb transition temperature of 200.9°F. This results in a 30 ft-lb transition temperature increase of 131.2°F and a 50 ft-lb transition temperature increase of 147.7°F.

  - The average upper-shelf energy of the SRM HSST 01 Charpy specimens resulted in an average energy decrease of 36 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 86 ft-lb.

- Based on the credibility evaluation presented in Appendix D, the St. Lucie Unit 2 surveillance plate, with consideration of all data or considering only the transverse orientation Charpy data points, and weld (Heat # 83637) data are both credible.

- Based on the upper-shelf energy evaluation in Appendix E, all beltline materials contained in the St. Lucie Unit 2 reactor vessel exhibit adequate upper-shelf energy levels for continued safe plant operation and are predicted to maintain an upper-shelf energy greater than 50 ft-lb through end-of-license (55 EFPY) as required by 10 CFR 50, Appendix G [Ref. 2].

- The maximum calculated 55 EFPY (end-of-license) neutron fluence (E > 1.0 MeV) for the St. Lucie Unit 2 reactor vessel beltline using the Regulatory Guide 1.99, Revision 2 attenuation formula (i.e., Equation #3 in the Guide) is as follows:

  Calculated (55 EFPY): Vessel clad/base metal interface fluence* = 4.53 x 10^{19} n/cm^2
  Vessel 1/4 thickness fluence = 2.700 x 10^{19} n/cm^2

  *This fluence value is documented in Table 6-2

2 INTRODUCTION

This report presents the results of the examination of Capsule 97°, the third capsule removed and tested in the continuing surveillance program, which monitors the effects of neutron irradiation on the Florida Power & Light Company St. Lucie Unit 2 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the St. Lucie Unit 2 reactor pressure vessel materials was designed and recommended by Combustion Engineering, Inc. A description of the surveillance program is contained in TR-L-MCM-001 [Ref. 3], "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of St. Lucie No. 2 Reactor Vessel Materials." The pre-irradiation mechanical properties of the reactor vessel materials are presented in BAW-1880 [Ref. 4]. The surveillance program was originally planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-73 [Ref. 5], "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." Capsule 97° was removed from the reactor after 25.55 EFPY of exposure and shipped to the Westinghouse Materials Center of Excellence Hot Cell Facility, where the post-irradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the testing and post-irradiation data obtained from surveillance Capsule 97° removed from the St. Lucie Unit 2 reactor vessel and discusses the analysis of the data.

3 BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low-alloy, ferritic pressure vessel steels such as SA533 Grade B Class 1 (base material of the St. Lucie Unit 2 reactor pressure vessel beltline) are well documented in the literature. Generally, low-alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness during high-energy irradiation.

A method for ensuring the integrity of reactor pressure vessels has been presented in "Fracture Toughness Criteria for Protection Against Failure," Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code [Ref. 6]. The method uses fracture mechanics concepts and is based on the reference nil-ductility transition temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop-weight nil-ductility transition temperature (NDTT per ASTM E208 [Ref. 7]) or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented perpendicular (transverse) to the major working direction of the plate. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{Ic} curve) which appears in Appendix G to Section XI of the ASME Code [Ref. 6]. The K_{Ic} curve is a lower bound of static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{Ic} 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 changes in mechanical properties of a given reactor pressure vessel steel, due to irradiation, can be monitored by a reactor vessel surveillance program, such as the St. Lucie Unit 2 reactor vessel radiation surveillance program, in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the initial RT_{NDT}, along with a margin (M) to cover uncertainties, to adjust the RT_{NDT} (ART) for radiation embrittlement. This ART (initial RT_{NDT} + M + ΔRT_{NDT}) is used to index the material to the K_{Ic} curve and, in turn, to set operating limits for the nuclear power plant that take into account the effects of irradiation on the reactor vessel materials.

4 DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the St. Lucie Unit 2 reactor pressure vessel core region (beltline) materials were inserted in the reactor vessel prior to initial plant startup. The six capsules were positioned in the reactor vessel, as shown in Figure 4-1, between the core barrel and the vessel wall, at various azimuthal locations. The vertical center of the capsules is opposite the vertical center of the core. The capsules contain specimens made from the following:

- Intermediate Shell Plate M-605-1 (longitudinal orientation)
- Intermediate Shell Plate M-605-1 (transverse orientation)
- Weld metal fabricated with weld wire Heat Number 83637, Linde Type 124 flux, Lot Number 0951, which is equivalent to the heat number used in the actual fabrication of the intermediate shell longitudinal weld seam repair and the lower shell longitudinal weld seams; however, these vessel welds used Linde Type 0091 flux in their fabrication
- Weld heat-affected zone (HAZ) material of Intermediate Shell Plate M-605-1
- Standard Reference Material (SRM) Heavy-Section Steel Technology (HSST)-01MY Plate

Test material obtained from the intermediate shell plate (after thermal heat treatment and forming of the plate) was taken at least one plate thickness from the quenched edges of the plate. All test specimens were machined from the ¼ thickness location of the plate after performing a simulated post-weld stress-relieving treatment on the test material. Test specimens were also removed from weld metal of a stress-relieved weldment joining Intermediate Shell Plate M-605-2 and adjacent Intermediate Shell Plate M-605-3. All heat-affected zone specimens were obtained from the weld heat-affected zone of Intermediate Shell Plate M-605-1.

Charpy V-notch impact specimens from Intermediate Shell Plate M-605-1 were machined in the longitudinal orientation (longitudinal axis of the specimen parallel to the major rolling direction) and also in the transverse orientation (longitudinal axis of the specimen perpendicular to the major rolling direction). The core-region weld Charpy impact specimens were machined from the weldment such that the long dimension of each Charpy specimen was perpendicular (normal) to the weld direction. The notch of the weld metal Charpy specimens was machined such that the direction of crack propagation in the specimen was in the welding direction.

Tensile specimens from Intermediate Shell Plate M-605-1 were machined in the transverse orientation only. Tensile specimens from the weld metal were oriented perpendicular to the welding direction.

Some of the St. Lucie Unit 2 capsules, specifically the previously tested Capsule 263° and also Capsule 104°, which is still in the reactor vessel, contain SRM, which was supplied by the Oak Ridge National Laboratory, from plate materials used in the HSST Program. The material for the St. Lucie Unit 2 Capsules was obtained from an A533, Grade B Class 1 plate labeled HSST 01. The plate was produced by the Lukens Steel Company and heat treated by Combustion Engineering, Inc.

All six capsules contain flux monitor assemblies that include sulfur pellets, iron wire, titanium wire, nickel wire *(cadmium-shielded)*, aluminum-cobalt wire *(cadmium-shielded and unshielded)*, copper wire *(cadmium-shielded)* and uranium foil *(cadmium-shielded and unshielded)*.

The capsules contain (12 total) thermal monitors made from four low-melting-point eutectic alloys, which were sealed in glass tubes. These thermal monitors were located in three different positions in the capsule. These thermal monitors are used to define the maximum temperature attained by the test specimens during irradiation. The composition of the four eutectic alloys and their melting points are as follows:

80.0% Au, 20.0% Sn	Melting Point: 536°F (280°C)
5.0% Ag, 5.0% Sn, 90.0% Pb	Melting Point: 558°F (292°C)
2.5% Ag, 97.5% Pb	Melting Point: 580°F (304°C)
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point: 590°F (310°C)

The chemical composition and the arrangement of the various mechanical specimens in Capsule 97° is presented in Tables 4-1 and 4-2, respectively. The data in Tables 4-1 and 4-2 was obtained from the original surveillance program report, TR-L-MCM-001 [Ref. 3], Tables III and XX.

Capsule 97° was removed after 25.55 effective full power years (EFPY) of plant operation. This capsule contained Charpy V-notch and tensile specimens, dosimeters, and thermal monitors. Figures 4-1 through 4-4 detail the arrangement of the surveillance capsules, an example of an original program surveillance capsule, a close-up of the Charpy impact specimen compartment, and the tensile and flux-monitor compartment assembly in the St. Lucie Unit 2 reactor vessel. Capsules 83°, 97°, 263° and 277° are radiologically equivalent to the 7° azimuth, while Capsules 104° and 284° are radiologically equivalent to the 14° azimuth.

Table 4-1 Chemical Composition (wt. %) of the St. Lucie Unit 2 Reactor Vessel Surveillance Materials (Unirradiated)

Element	Intermediate Shell Plate M-605-1[(a)]	Standard Reference Material HSST 01MY Plate[(b)]	Surveillance Weld Metal	
			Original CE Analysis[(c)]	Best-Estimate Analysis[(d)]
C	0.23	---	0.12	---
Mn	1.37	---	1.55	---
P	0.004	---	0.003	---
S	0.010	---	0.011	---
Si	0.23	---	0.38	---
Ni	**0.61**	**0.66**	**0.07**	**0.066**
Mo	0.57	---	0.59	---
Cr	0.07	---	0.04	---
Cu	**0.11**	**0.18**	**0.05**	**0.048**
Al	0.022	---	0.002	---
Co	0.010	---	0.006	---
Pb	<0.001	---	<0.001	---
W	<0.01	---	<0.01	---
Ti	<0.01	---	<0.01	---
Zr	<0.001	---	0.001	---
V	0.003	---	0.005	---
Sn	0.009	---	0.002	---
As	0.002	---	<0.001	---
Cb	<0.01	---	<0.01	---
Sb	0.0024	---	0.0030	---
N_2	0.009	---	0.003	---
B	<0.001	---	0.001	---

Notes:

(a) Data obtained from TR-L-MCM-001, Table III [Ref. 3]

(b) Data obtained from NUREG/CR-6413 [Ref. 8].

(c) Data obtained from TR-L-MCM-001, Table III [Ref. 3]. Weld Wire Heat Number 83637, Flux Type Linde 124, and Flux Lot Number 0951.

(d) Best-Estimate Cu and Ni wt. % values were taken from CE-NPSD-1039, Revision 2 [Ref. 9].

Table 4-2 Arrangement of Encapsulated Test Specimens within St. Lucie Unit 2 Capsule 97°

Compartment Position[(a)]	Compartment Number (Specimen Type and Material)[(a)]	Specimen Numbers[(a)]
1	K214 (Tensile HAZ Specimens)	4K5, 4J5, 4JK
2	K224 (Charpy Impact HAZ Specimens)	41P, 456, 42D, 46Y, 425, 47D, 42C, 47L, 43K, 473, 43D, 46A
3	K231 (Charpy Impact Longitudinal Plate Specimens)	124, 13K, 14M, 12B, 14D, 127, 117, 15D, 13C, 11D, 143, 15J
4	K242 (Tensile Transverse Plate Specimens)	2L5, 2KU, 2JM
5	K252 (Charpy Impact Transverse Plate Specimens)	21B, 25D, 23A, 263, 22K, 25M, 21D, 24M, 21U, 24P, 26E, 21Y
6	K263 (Charpy Impact Weld Specimens)	34U, 32A, 37Y, 31A, 36E, 32B, 355, 32L, 37P, 32E, 354, 311
7	K273 (Tensile Weld Specimens)	3J4, 3K7, 3L5

Note:
(a) Data obtained from TR-L-MCM-001, Table XIX and/or Table XX [Ref. 3].

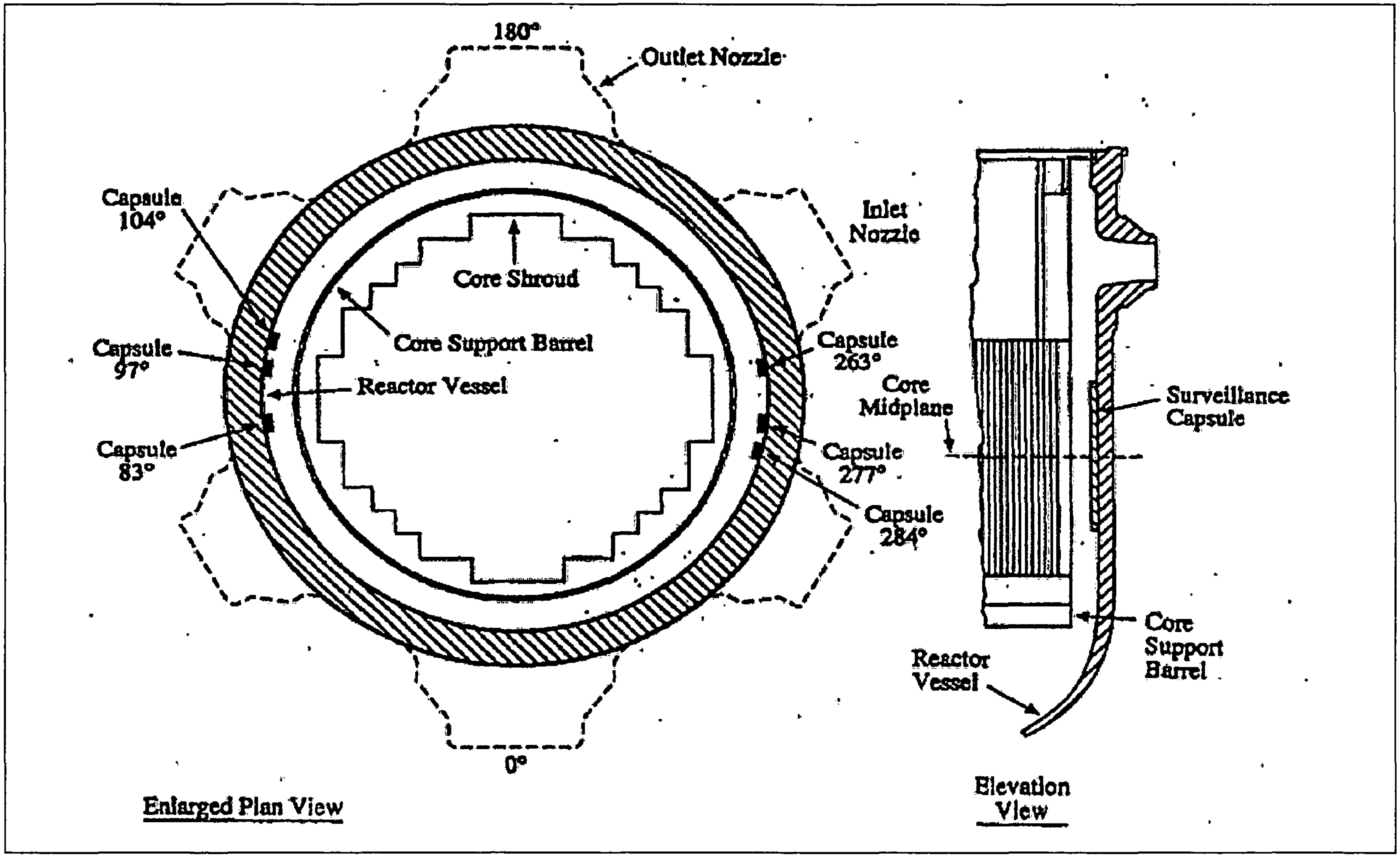


Figure 4-1 Arrangement of Surveillance Capsules in the St. Lucie Unit 2 Reactor Vessel

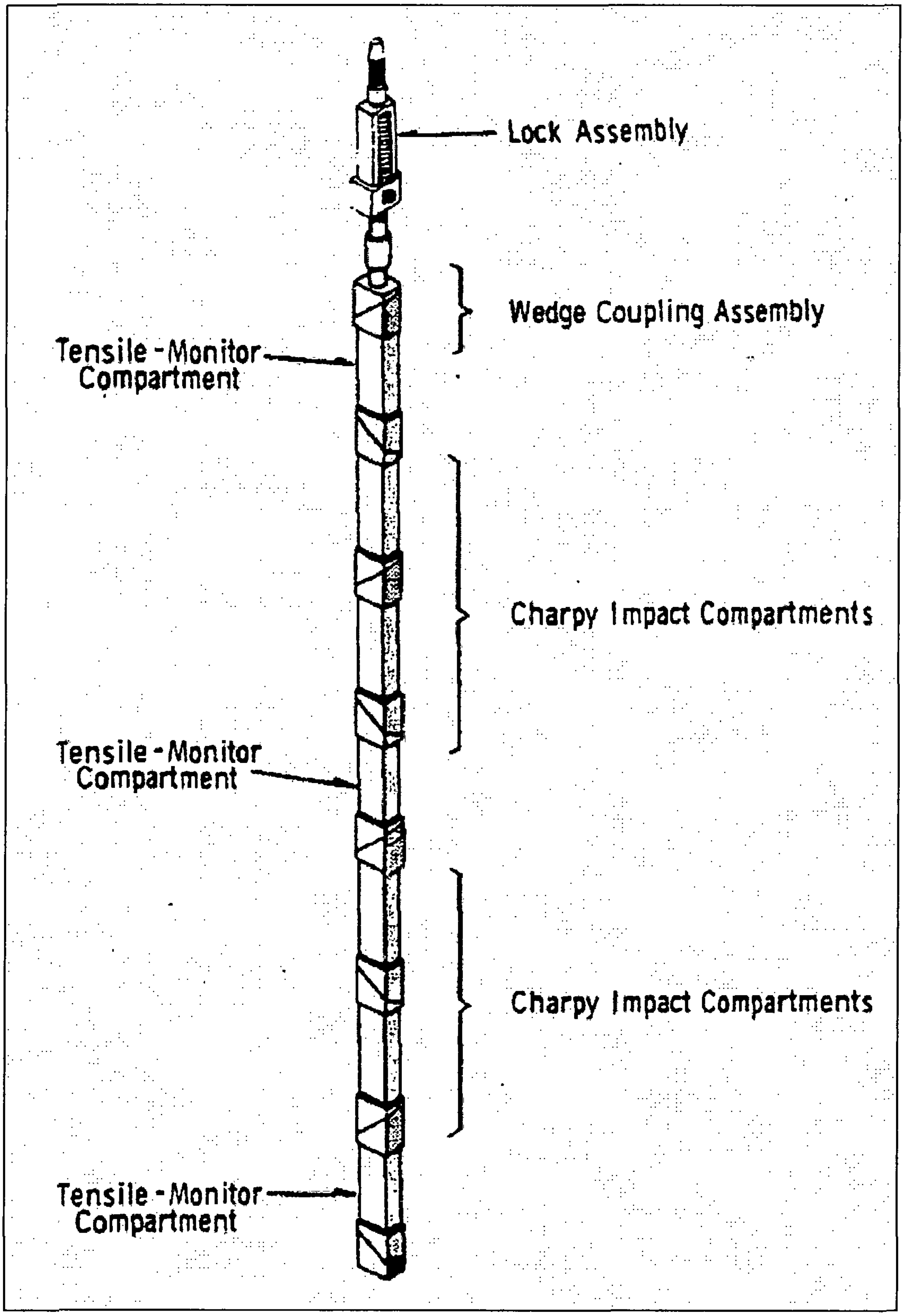


Figure 4-2 Original Surveillance Program Capsule in the St. Lucie Unit 2 Reactor Vessel

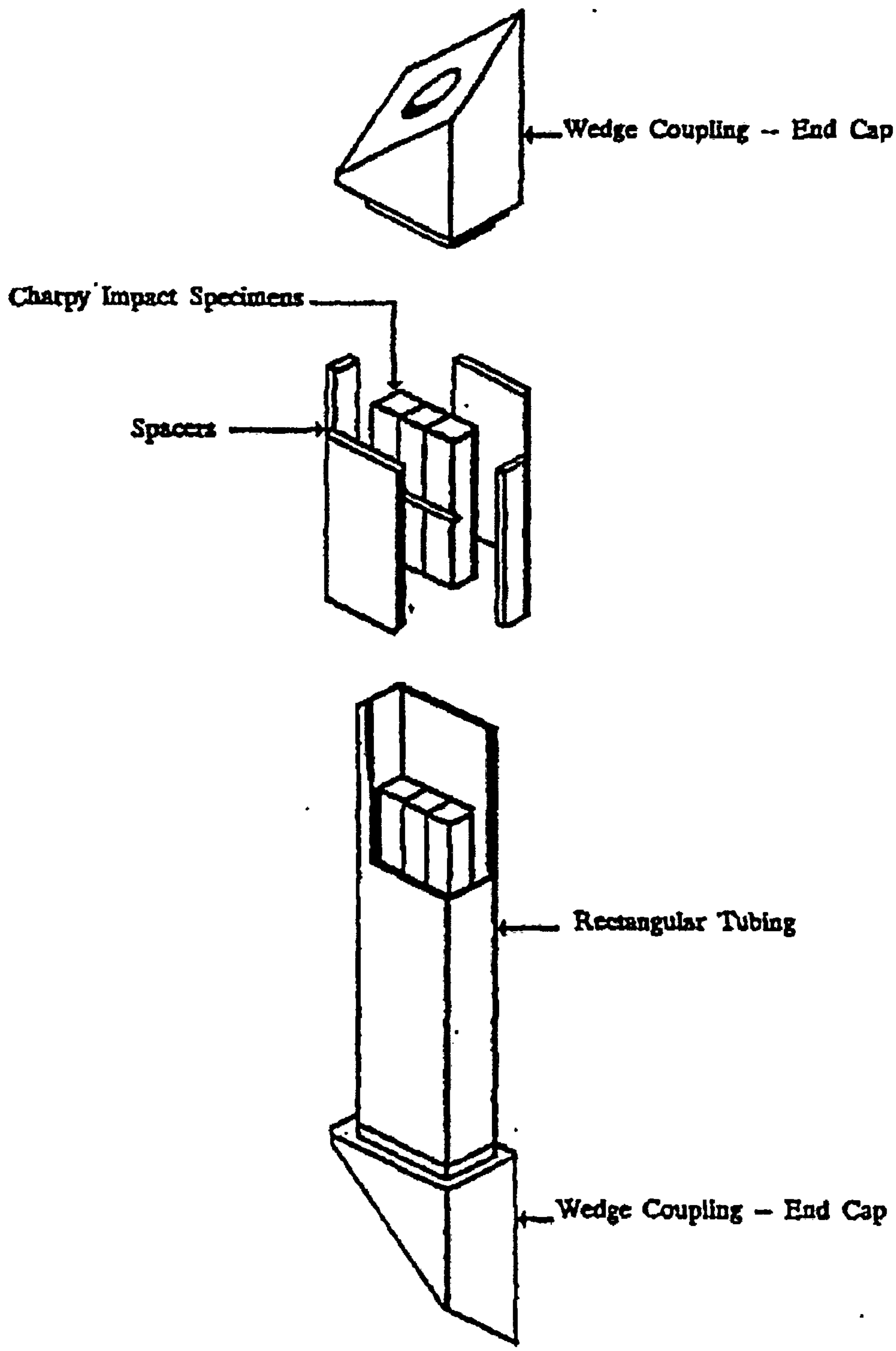


Figure 4-3 Surveillance Capsule Charpy Impact Specimen Compartment Assembly in the St. Lucie Unit 2 Reactor Vessel

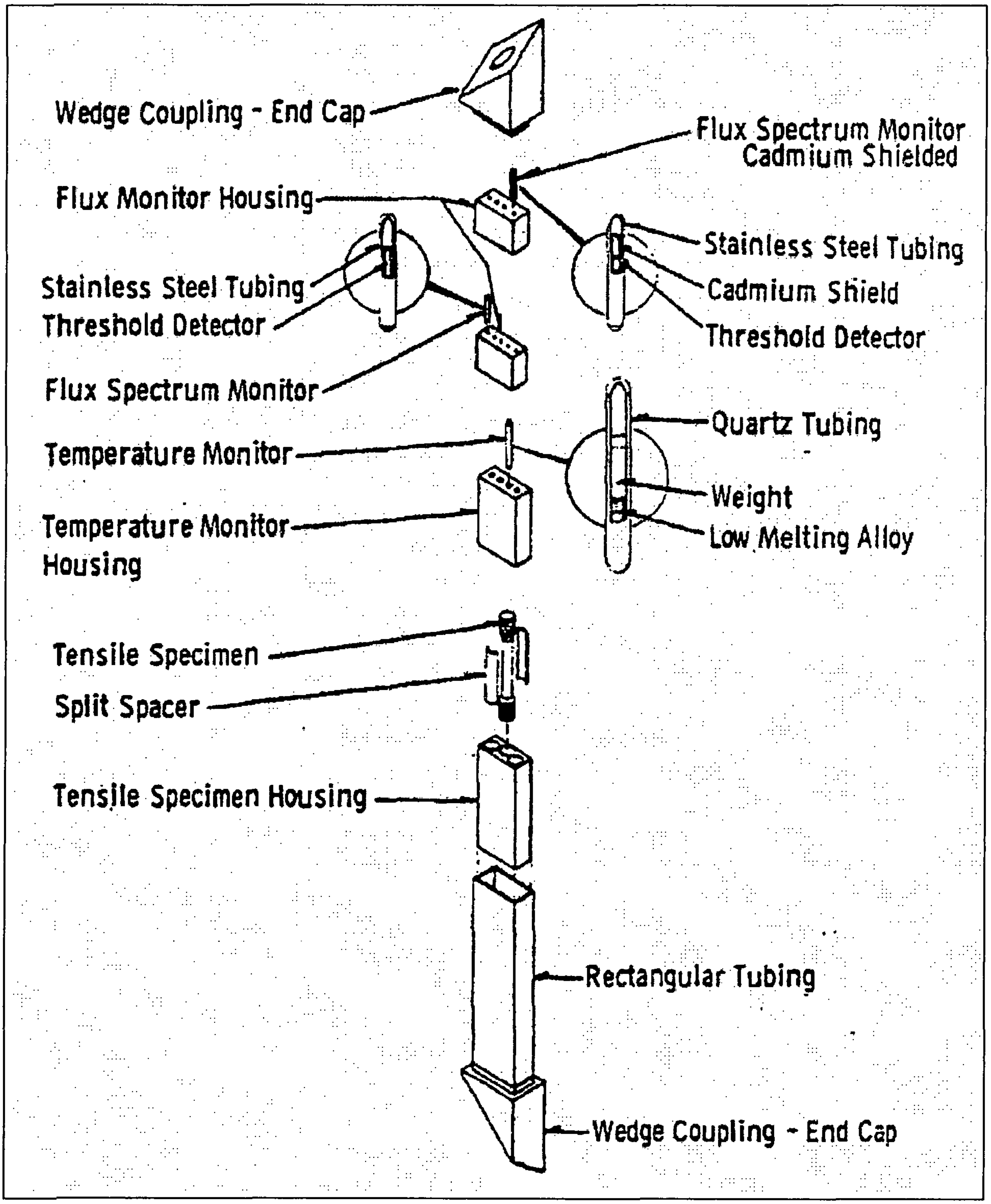


Figure 4-4 Surveillance Capsule Tensile and Flux-Monitor Compartment Assembly in the St. Lucie Unit 2 Reactor Vessel

5 TESTING OF SPECIMENS FROM CAPSULE 97°

5.1 OVERVIEW

The post-irradiation mechanical testing of the Charpy V-notch impact specimens and tensile specimens was performed at the Westinghouse Materials Center of Excellence Hot Cell Facility. Testing was performed in accordance with 10 CFR 50, Appendix H [Ref. 2] and ASTM Specification E185-82 [Ref. 10].

Capsule 97° was opened upon receipt at the hot cell laboratory. The specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in TR-L-MCM-001 [Ref. 3]. All of the items were in their proper locations.

Examination of the thermal monitors indicated that 4 of the 12 temperature monitors had melted, as described below:

- Capsule compartment K214, the 536°F (280°C) temperature monitor melted
- Capsule compartment K242, the 536°F (280°C) and 558°F (292°C) temperature monitors melted
- Capsule compartment K273, the 536°F (280°C) temperature monitor melted.

Based on this examination, the maximum temperature to which the specimens were exposed was less than 580°F (304°C), but greater than 558°F (292°C).

The Charpy impact tests were performed per ASTM Specification E185-82 [Ref. 10] and E23-12c [Ref. 11] on a Tinius-Olsen Model 74, 358J machine. The Charpy machine striker was instrumented with an Instron Impulse system. Instrumented testing and calibration were performed to ASTM E2298-13a [Ref. 12].

The instrumented striker load signal data acquisition rate was 819 kHz with data acquired for 10 ms. From the load-time curve, the load of general yielding (F_{gy}), the maximum load (F_m) and the time to maximum load were 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 brittle fracture load (F_{bf}). The termination load after the fast load drop is identified as the arrest load (F_a). F_{gy}, F_m, F_{bf}, and F_a were determined per the guidance in ASTM Standard E2298-13a [Ref. 12].

The energy at maximum load (W_m) was determined by integrating the load-time record to the maximum load point. 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 (W_P) is the difference between the total energy (W_t) and the energy at maximum load (W_m). W_t is compared to the dial energy (KV). W_t derived from the instrumented striker were all within 15% of the calibrated dial energy values as required in ASTM E2298-13a [Ref. 12].

Percent shear was determined from post-fracture photographs using the ratio-of-areas method in compliance with ASTM E23-12c [Ref. 11] and A370-13 [Ref. 13]. The lateral expansion was measured using a dial gage rig similar to that shown in the same specifications.

Tensile tests were performed on a 250 KN Instron screw driven tensile machine (Model 5985) per ASTM E185-82 [Ref. 10]. Testing met ASTM Specifications E8/E8M-13a [Ref. 14] or E21-09 [Ref. 15]. Load was applied through a threaded connection. The strain rate obtained met the requirements of ASTM E8/E8M-13a [Ref. 14] and ASTM E21-09 [Ref. 15].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 10-inch hot zone. Tensile specimens were soaked at temperature (±5°F) for a minimum of 20 minutes before testing. All tests were conducted in air.

The tensile specimens were 3.00 inches long with a 1.00 inch gage section and a reduced section of 1.50 inches long by 0.250 inch in diameter, as documented in Figure 6 (Drawing CND-B-3654 Rev 2) of TR-L-MCM-001 [Ref. 3]. The yield load, ultimate load, fracture load, uniform elongation and elongation at fracture were determined directly from the load-extension curve. The yield strength (0.2% offset method), ultimate tensile strength and fracture strength were calculated using the original cross-sectional area. Yield point elongation (YPE) was calculated as the difference in strain between the upper yield strength and the onset of uniform strain hardening using the methodology described in E8/E8M-13a [Ref. 14]. The final diameter and final gage length were determined from post-fracture photographs. This final diameter measurement was used to calculate the fracture stress (true stress at fracture) and the percent reduction in area. The final and original gage lengths were used to calculate total elongation after fracture.

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 97°, which received a fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) in 25.55 EFPY of operation, are presented in Tables 5-1 through 5-8 and are compared with the unirradiated and previously withdrawn capsule results as shown in Figures 5-1 through 5-12. The unirradiated and previously withdrawn capsule results were taken from BAW-1880 [Ref. 4] and WCAP-15040, Revision 1 [Ref. 16]. The previous capsules, along with the original program unirradiated material input data, were updated using CVGRAPH, Version 6.0 from the hand-drawn plots presented in the earliest reports. This accounts for the differences in measured values of 30 ft-lb and 50 ft-lb transition temperature between the results documented in this report and those shown in prior St. Lucie Unit 2 capsule reports.

The transition temperature increases and changes in upper-shelf energies for the Capsule 97° materials are summarized in Table 5-9 and led to the following results:

- Irradiation of the reactor vessel Intermediate Shell Plate M-605-1 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 124.3°F and an irradiated 50 ft-lb transition temperature of 173.3°F. This results in a 30 ft-lb transition temperature increase of 132.7°F and a 50 ft-lb transition temperature increase of 140.1°F for the longitudinally oriented specimens.

- Irradiation of the reactor vessel Intermediate Shell Plate M-605-1 Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (transverse orientation), resulted in an irradiated 30 ft-lb transition temperature of 158.0°F and an irradiated 50 ft-lb transition temperature of 219.8°F. This results in a 30 ft-lb transition temperature increase of 127.6°F and a 50 ft-lb transition temperature increase of 148.3°F for the transversely oriented specimens.

- Irradiation of the Surveillance Program Weld Metal (Heat # 83637) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of -25.7°F and an irradiated 50 ft-lb transition temperature of 16.2°F. This results in a 30 ft-lb transition temperature increase of 24.8°F and a 50 ft-lb transition temperature increase of 28.7°F.

- Irradiation of the Heat-Affected Zone (HAZ) Material Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of -137.0°F and an irradiated 50 ft-lb transition temperature of 64.1°F. This results in a 30 ft-lb transition temperature increase of -103.9°F and a 50 ft-lb transition temperature increase of 46.2°F. It is noted that the scatter in the HAZ data was significant and the resulting CVGRAPH, Version 6.0 Charpy V-notch symmetric hyperbolic tangent curve-fit plots are not an ideal rendition of the data. However, this is inconsequential since HAZ material is not considered limiting as compared to the base and weld materials.

- The irradiated upper-shelf energy of Intermediate Shell Plate M-605-1 (longitudinal orientation) resulted in an average energy decrease of 26 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 108 ft-lb for the longitudinally oriented specimens.

- The average upper-shelf energy of Intermediate Shell Plate M-605-1 (transverse orientation) resulted in an average energy decrease of 25 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 78 ft-lb for the transversely oriented specimens.

- The average upper-shelf energy of the Surveillance Program Weld Metal (Heat # 83637) Charpy specimens resulted in an average energy decrease of 20 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 95 ft-lb for the weld metal specimens.

- The average upper-shelf energy of the HAZ Material Charpy specimens resulted in an average energy decrease of 12 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 93 ft-lb for the HAZ Material.

- Comparisons of the measured 30 ft-lb shift in transition temperature values and upper-shelf energy decreases to those predicted by Regulatory Guide 1.99, Revision 2 [Ref. 1] for the St. Lucie Unit 2 reactor vessel surveillance materials are presented in Table 5-10.

  Standard Reference Material (SRM) HSST 01 Charpy specimens were not included in the St. Lucie Unit 2 Capsule 97°. However, the SRM HSST 01 Charpy specimens were reanalyzed in this report. The SRM HSST 01 material was contained in Capsule 263°, which was irradiated to a neutron fluence of 1.00 x 10^{19} n/cm^2 (E > 1.0 MeV). The results of the SRM HSST 01 reanalysis will be included in Table 5-10 and shown in Figures 5-13 through 5-15.

- Irradiation of the SRM HSST 01 Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 157.1°F and an irradiated 50 ft-lb transition temperature of 200.9°F. This results in a 30 ft-lb transition temperature increase of 131.2°F and a 50 ft-lb transition temperature increase of 147.7°F.

- The average upper-shelf energy of the SRM HSST 01 Charpy specimens resulted in an average energy decrease of 36 ft-lb after irradiation. This results in an irradiated average upper-shelf energy of 86 ft-lb.

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-16 through 5-19. The fractures show an increasingly ductile or tougher appearance with increasing test temperature. Load-time records for the individual instrumented Charpy specimens are contained in Appendix B.

With consideration of the surveillance data, all beltline materials exhibit adequate upper-shelf energy levels for continued safe plant operation and are predicted to maintain an upper-shelf energy greater than 50 ft-lb through end-of-license (55 EFPY) as required by 10 CFR 50, Appendix G [Ref. 2]. This evaluation can be found in Appendix E.

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in Capsule 97° irradiated to 2.25×10^{19} n/cm^2 (E > 1.0 MeV) are presented in Table 5-11 and are compared with unirradiated results as shown in Figures 5-20 through 5-22.

The results of the tensile tests performed on the Intermediate Shell Plate M-605-1 (transverse orientation) indicated that irradiation to 2.25×10^{19} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data [Ref. 4]. See Figure 5-20 and Table 5-11.

The results of the tensile tests performed on the Surveillance Program Weld Metal (Heat # 83637) indicated that irradiation to 2.25×10^{19} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data [Ref. 4]. See Figure 5-21 and Table 5-11.

The results of the tensile tests performed on the Heat Affected Zone Material indicated that irradiation to 2.25×10^{19} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data [Ref. 4]. See Figure 5-22 and Table 5-11.

The fractured tensile specimens for the Intermediate Shell Plate M-605-1 (transverse orientation) material are shown in Figure 5-23, the fractured tensile specimens for the Surveillance Program Weld Metal (Heat # 83637) are shown in Figure 5-24, and the fractured tensile specimens for the Heat Affected Zone Material are shown in Figure 5-25. The engineering stress-strain curves for the tensile tests are shown in Figures 5-26 through 5-31.

Table 5-1 Charpy V-notch Data for the St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) (Longitudinal Orientation)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
15D	70	21	19	26	15	0.38	20
14M	95	35	35	47	30	0.76	25
124	110	43	28	38	27	0.69	25
11D	120	49	43	58	36	0.91	30
14D	140	60	24	32	25	0.64	25
13K	150	66	28	38	29	0.74	30
117	170	77	67	91	59	1.50	60
13C	200	93	41	56	32	0.81	45
143	200	93	42	57	38	0.96	40
15J	260	127	111	150	89	2.26	100
12B	300	149	100	136	82	2.08	100
127	375	191	113	153	89	2.26	100

Table 5-2 Charpy V-notch Data for the St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) (Transverse Orientation)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
21B	70	21	18	24	14	0.36	15
22K	95	35	16	22	2	0.05	20
24M	120	49	27	37	24	0.61	25
25D	140	60	22	30	22	0.56	25
24P	150	66	37	50	34	0.86	45
26E	170	77	36	49	37	0.94	45
21U	200	93	25	34	23	0.58	35
21Y	200	93	29	39	23	0.58	45
25M	230	110	54	73	47	1.19	70
23A	270	132	80	108	60	1.52	100
21D	300	149	78	106	65	1.65	100
263	375	191	77	104	68	1.73	100

Table 5-3 Charpy V-notch Data for the St. Lucie Unit 2 Surveillance Program Weld Metal (Heat # 83637) Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
36E	-60	-51	17	23	14	0.36	15
34U	-40	-40	25	34	21	0.53	25
32E	-30	-34	22	30	15	0.38	20
31A	-25	-32	24	32	25	0.64	25
355	-20	-29	45	61	36	0.91	30
37P	0	-18	49	66	45	1.14	50
32A	20	-7	49	66	35	0.89	60
32B	70	21	61	83	60	1.52	80
354	120	49	93	126	84	2.13	95
37Y	170	77	113	153	84	2.13	98
311	250	121	82	111	72	1.83	100
32L	300	149	93	126	82	2.08	100

Table 5-4 Charpy V-notch Data for the St. Lucie Unit 2 Heat-Affected Zone (HAZ) Material Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
425	50	10	106	144	58	1.47	60
43D	70	21	17	23	18	0.46	30
41P	100	38	42	57	37	0.94	55
456	120	49	34	46	29	0.74	50
46Y	130	54	24	32	22	0.56	50
43K	140	60	120	163	82	2.08	95
47L	150	66	21	28	20	0.51	45
473	170	77	22	30	27	0.68	45
42C	180	82	86	117	53	1.35	85
42D	250	121	123	167	80	2.03	100
46A	300	149	59	80	63	1.60	100
47D	375	191	68	92	57	1.45	100

Table 5-5 Instrumented Charpy Impact Test Results for the St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) (Longitudinal Orientation)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, $(KV-W_t)/KV$ (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
15D	70	19	16.9	11.0	3.9	4600	0.13	3200	3700	N/A
14M	95	35	33.3	4.9	22.2	3900	0.43	3100	3800	800
124	110	28	26.9	3.9	21.6	3800	0.43	2900	3500	700
11D	120	43	41.0	4.6	32.0	4000	0.60	2800	3900	1100
14D	140	24	23.4	4.4	3.71	4100	0.13	2700	3600	400
13K	150	28	26.2	6.4	3.35	4100	0.16	2800	3900	900
117	170	67	61.5	8.2	3.6	4200	0.14	2500	3600	1800
13C	200	41	39.4	3.9	2.2	4100	0.10	3000	3800	1500
143	200	42	39.1	6.9	3.8	4100	0.14	2700	3400	1500
15J	260	111	106.8	3.8	29.9	3700	0.60	2500	N/A	N/A
12B	300	100	96.2	3.8	29.9	3800	0.60	2600	N/A	N/A
127	375	113	108.3	4.2	39.9	3900	0.79	2500	N/A	N/A

Table 5-6 Instrumented Charpy Impact Test Results for the St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) (Transverse Orientation)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, $(KV-W_t)/KV$ (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
21B	70	18	16.7	7.2	3.8	3500	0.13	3000	3400	500
22K	95	16	16.1	0.6	2.1	3600	0.37	2800	3600	400
24M	120	27	25.1	7.0	19.1	4200	0.36	3400	4100	900
25D	140	22	21.0	4.5	13.7	3400	0.30	2700	3200	1100
24P	150	37	35.8	3.2	21.0	3700	0.43	2800	3600	1900
26E	170	36	34.5	4.2	20.8	3600	0.43	2700	3400	1500
21U	200	25	24.6	1.6	2.1	3700	0.15	3200	3600	1100
21Y	200	29	27.3	5.9	4.4	4300	0.40	2800	3600	1200
25M	230	54	49.8	7.8	29.0	3600	0.60	2600	3500	2400
23A	270	80	75.9	5.2	28.0	4400	0.56	2800	N/A	N/A
21D	300	78	74.6	4.4	30.2	3600	0.63	2500	N/A	N/A
263	375	77	72.7	5.5	24.1	3700	0.56	2500	N/A	N/A

Table 5-7 Instrumented Charpy Impact Test Results for the St. Lucie Unit 2 Surveillance Program Weld Metal (Heat # 83637) Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, (KV-W_t)/KV (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
36E	-60	17	16.2	4.7	3.1	4100	0.09	3500	4000	200
34U	-40	25	24.4	2.4	3.2	4200	0.09	3500	3800	600
32E	-30	22	21.3	3.2	3.2	4100	0.09	3500	4000	600
31A	-25	24	23.0	4.2	3.3	4100	0.09	3200	4000	400
355	-20	45	41.9	6.7	2.8	4200	0.09	3300	3900	600
37P	0	49	45.2	7.8	33.6	4100	0.60	3100	3900	1300
32A	20	49	46.4	5.2	33.6	3900	0.60	3100	3700	1900
32B	70	61	57.3	6.1	31.8	3800	0.60	2900	3200	1800
354	120	93	89.6	3.6	30.8	3800	0.60	2700	2500	2100
37Y	170	113	108.3	4.2	33.7	4100	0.73	2600	N/A	N/A
311	250	82	79.4	3.2	29.5	3600	0.60	2600	N/A	N/A
32L	300	93	90.4	2.8	28.9	3600	0.60	2400	N/A	N/A

Table 5-8 Instrumented Charpy Impact Test Results for the St. Lucie Unit 2 Heat-Affected Zone (HAZ) Material Irradiated to a Fluence of 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, $(KV-W_t)/KV$ (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
425	50	106	101.0	4.7	47.4	4400	0.79	3200	3100	800
43D	70	17	16.1	5.3	2.1	4300	0.29	2600	3000	1500
41P	100	42	40.2	4.4	3.1	3900	0.09	2800	3800	1900
456	120	34	32.1	5.6	2.4	3900	0.09	3000	3700	2300
46Y	130	24	22.0	8.4	11.7	3500	0.26	2900	3400	1800
43K	140	120	115.3	3.9	44.6	4200	0.79	3000	3100	3000
47L	150	21	19.0	9.5	4.1	3900	0.15	2800	3400	1100
473	170	22	21.4	2.7	1.9	4600	0.24	2600	3400	1400
42C	180	86	81.6	5.0	43.3	4000	0.79	2900	3800	2700
42D	250	123	117.6	4.4	2.0	4600	0.12	2300	N/A	N/A
46A	300	59	56.0	5.1	28.7	3400	0.60	2400	N/A	N/A
47D	375	68	65.2	4.1	20.3	3700	0.47	2600	N/A	N/A

Table 5-9 Effect of Irradiation to 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV) on the Charpy V-Notch Toughness Properties of the St. Lucie Unit 2 Reactor Vessel Surveillance Capsule 97° Materials

Material	Average 30 ft-lb Transition Temperature[a] (°F)			Average 35 mil Lateral Expansion Temperature[a] (°F)			Average 50 ft-lb Transition Temperature[a] (°F)			Average Energy Absorption at Full Shear[a] (ft-lb)		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔE
Intermediate Shell Plate M-605-1 (Longitudinal)	-8.4	124.3	132.7	13.1	146.3	133.2	33.2	173.3	140.1	134	108	26
Intermediate Shell Plate M-605-1 (Transverse)	30.4	158.0	127.6	38.2	190.7	152.5	71.5	219.8	148.3	103	78	25
Surveillance Weld Material (Heat # 83637)	-50.5	-25.7	24.8	-27.1	-2.0	25.1	-12.5	16.2	28.7	115	95	20
Heat Affected Zone Material	-33.1	-137.0	0[b]	8.5	81.6	73.1	17.9	64.1	46.2	105	93	12

Notes:

(a) Average value is determined by CVGraph, Version 6.0 (see Appendix C).

(b) The St. Lucie Unit 2 Heat Affected Zone Material 30 ft-lb transition temperature shift was calculated to be a negative value. Physically, this should not occur; therefore, a conservative value of zero degrees F is shown in this table.

Table 5-10 Comparison of the St. Lucie 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	Capsule Fluence ($x10^{19}$ n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper-Shelf Energy Decrease	
			Predicted[(a)] (°F)	Measured[(b)] (°F)	Predicted[(a)] (%)	Measured[(b)] (%)
Intermediate Shell Plate M-605-1 (Longitudinal)	83°	0.140	36.2	45.1	12.5	11
	97°	2.250	90.4	132.7	24	19
Intermediate Shell Plate M-605-1 (Transverse)	83°	0.140	36.2	29.4	12.5	1
	263°	1.000	74.2	102.7	20	23
	97°	2.250	90.4	127.6	24	24
Surveillance Weld Material (Heat # 83637)	83°	0.140	16.6	15.8	12	13
	263°	1.000	34.1	26.5	19	9
	97°	2.250	41.5	24.8	23	17
Heat Affected Zone Material	83°	0.140	---	0.0[(c)]	---	0[(d)]
	263°	1.000	---	79.5	---	0[(d)]
	97°	2.250	---	0.0[(c)]	---	11
Standard Reference Material	263°	1.000	---	131.2	---	30

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the capsule fluence and mean weight percent values of copper and nickel of the surveillance material.
(b) Calculated by CVGraph Version 6.0 using measured Charpy data (See Appendix C).
(c) These ΔRT_{NDT} were calculated to be negative values. Physically, this should not occur; therefore, conservative values of zero degrees F are shown in this table.
(d) USE values were calculated to have increased. Physically, this should not occur; therefore, conservative values of zero percent are shown in this table.

Table 5-11 Tensile Properties of the St. Lucie Unit 2 Capsule 97° Reactor Vessel Surveillance Materials Irradiated to 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

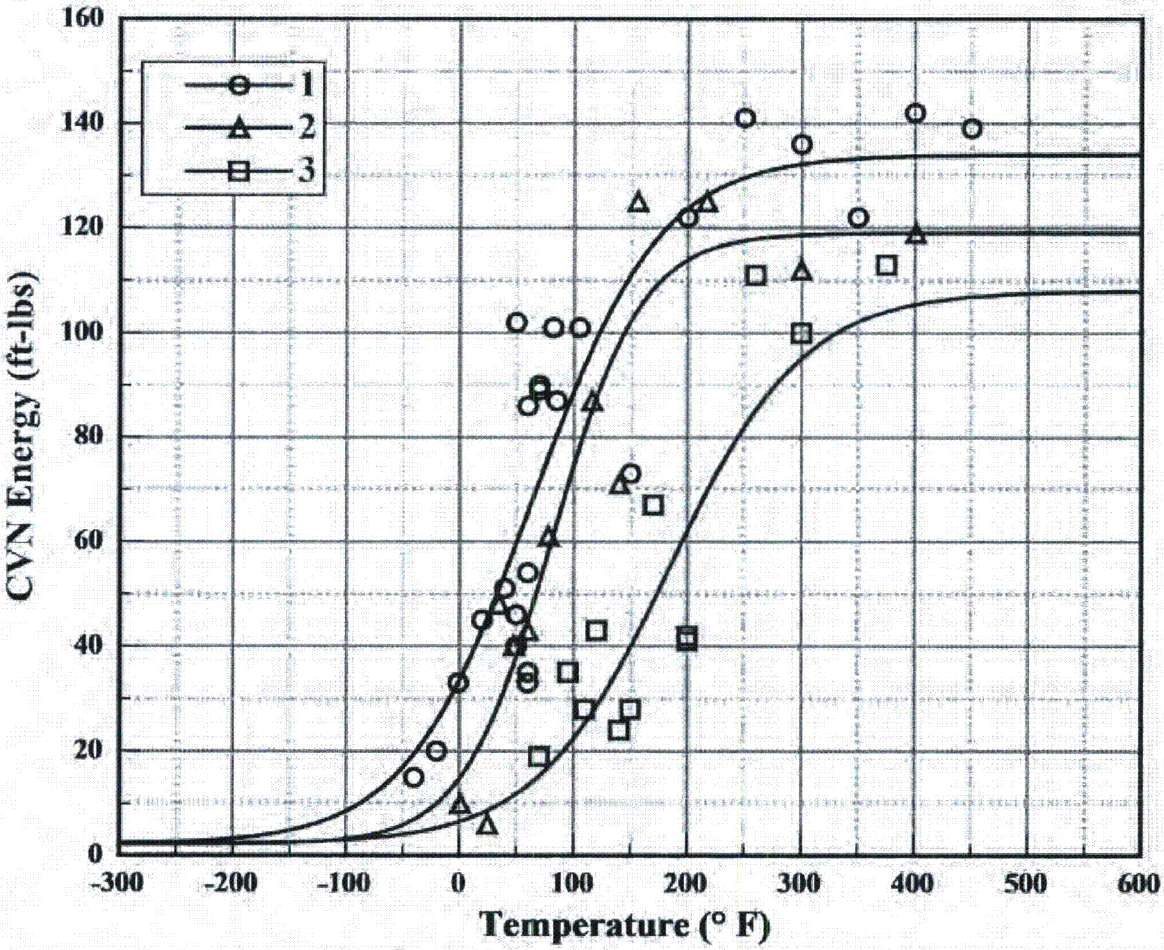
Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Strength (ksi)	Fracture True Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Intermediate Shell Plate M-605-1 (Transverse)	2L5	150	71.9	93.4	3.37	68.7	160	12.1	24.3	57
	2KU	280	68.0	89.0	3.43	69.9	162	11.7	21.4	57
	2JM	550	61.7	89.1	3.46	70.5	156	10.9	18.7	55
Surveillance Weld Material (Heat # 83637)	3J4	72	67.1	86.4	2.87	58.5	192	10.6	23.0	70
	3K7	150	68.9	85.0	2.73	55.5	152	10.4	25.0	64
	3L5	550	65.4	84.5	2.84	57.8	167	9.1	20.8	65
Heat Affected Zone Material	4K5	72	64.3	82.2	2.72	55.4	172	7.9*	8.9*	68
	4J5	240	65.7	83.2	2.59	52.8	168	9.4	22.6	69
	4JK	550	64.6	84.1	2.73	55.5	172	8.8	20.9	68

*Refer to Figure 5-25; specimen broke outside of gage section, so strain is not an accurate measurement. These values are omitted from Figure 5-22.

IS PLATE M-605-1 (LONGITUDINAL)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 7:16 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	A-8490-2
2	St. Lucie 2	83°	SA533B1	LT	A-8490-2
3	St. Lucie 2	97°	SA533B1	LT	A-8490-2



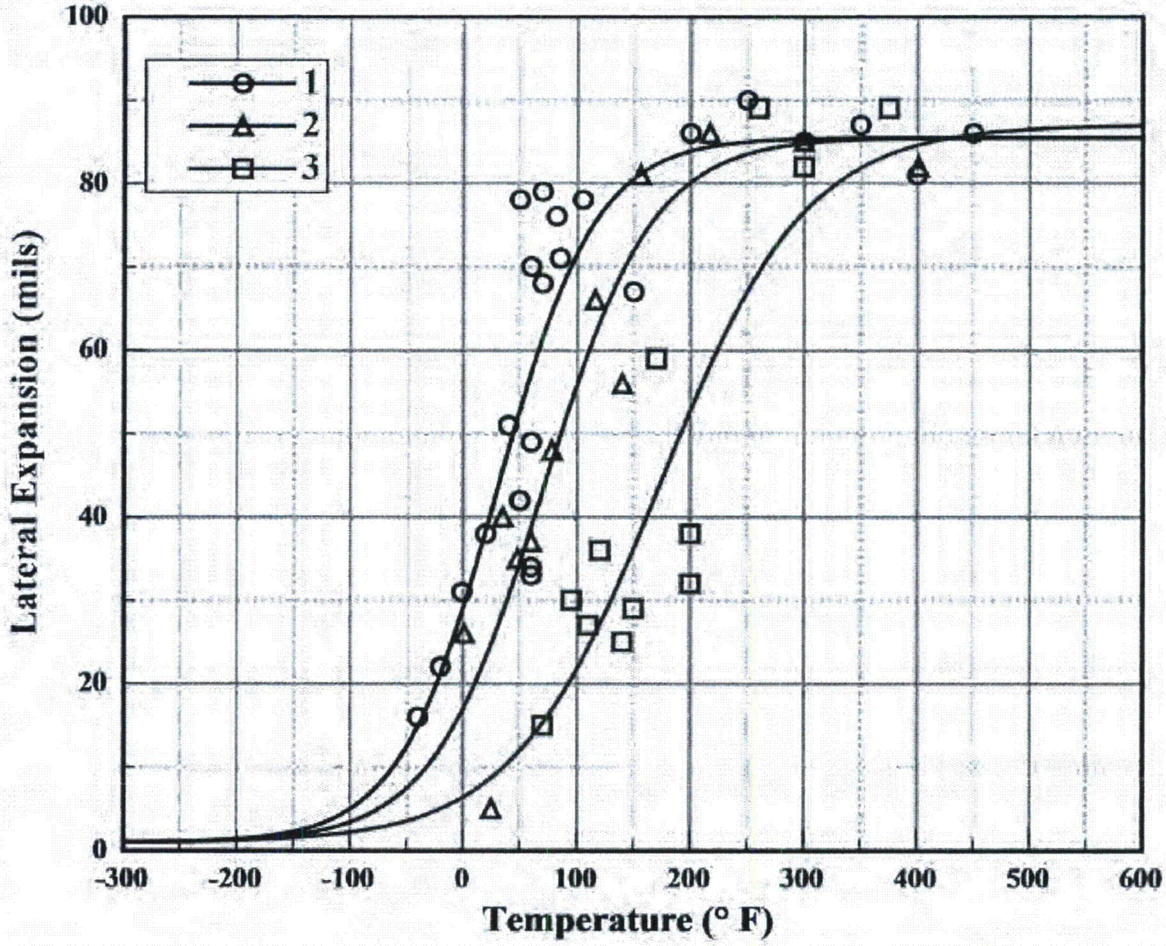
Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	134	0	-8.4	0	33.2	0
2		2.2	119	-15	36.7	45.1	67.3	34.1
3		2.2	108	-26	124.3	132.7	173.3	140.1

Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Longitudinal Orientation)

IS PLATE M-605-1 (LONGITUDINAL)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 4/10/2015 7:41 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	A-8490-2
2	St. Lucie 2	83°	SA533B1	LT	A-8490-2
3	St. Lucie 2	97°	SA533B1	LT	A-8490-2



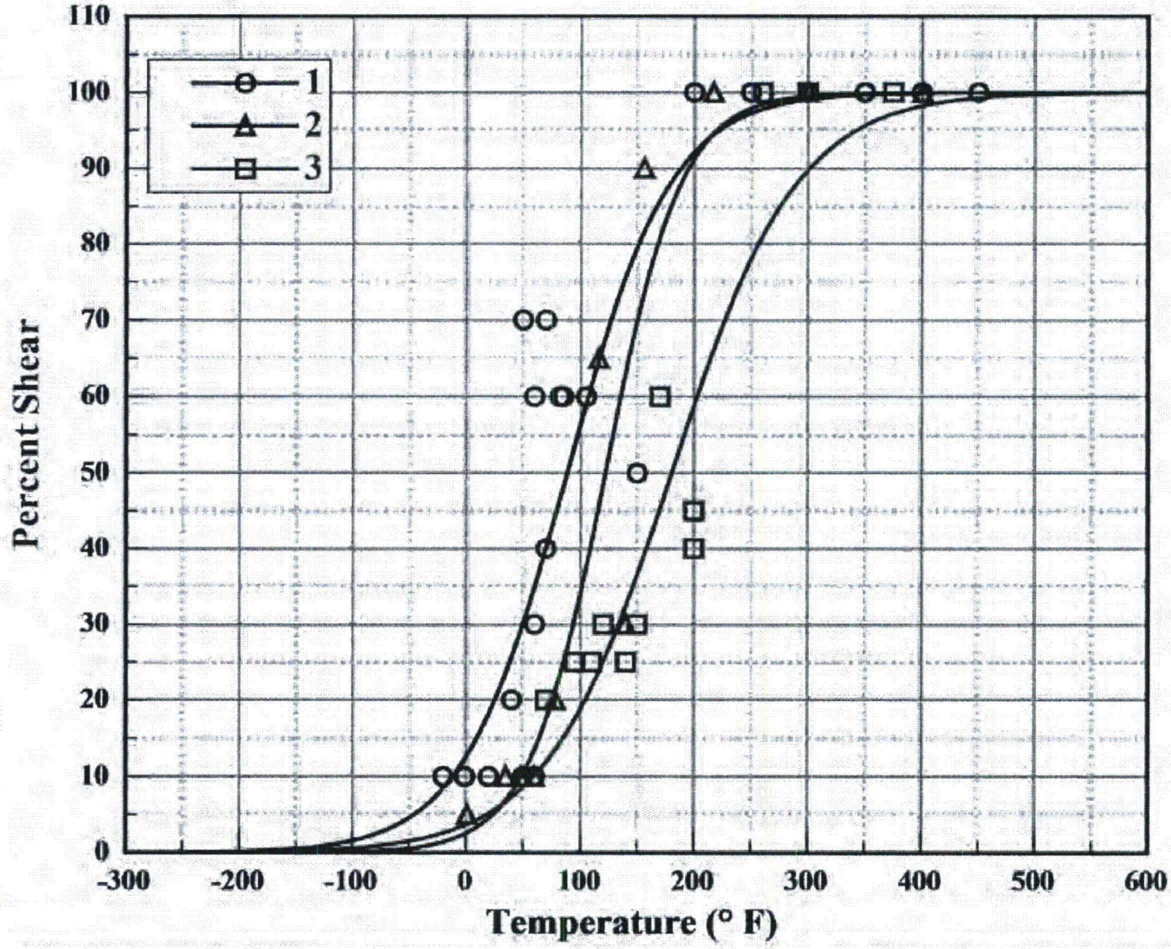
Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0	1	85.48	0	13.1	0
2	0	1	85.49	0.01	51.2	38.1
3	0	1	87	1.52	146.3	133.2

Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Longitudinal Orientation)

IS PLATE M-605-1 (LONGITUDINAL)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:08 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	A-8490-2
2	St. Lucie 2	83°	SA533B1	LT	A-8490-2
3	St. Lucie 2	97°	SA533B1	LT	A-8490-2



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		0	100	0	88.3	0
2		0	100	0	122	33.7
3		0	100	0	180.5	92.2

Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Longitudinal Orientation)

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:25 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	TL	A-8490-2
2	St. Lucie 2	83°	SA533B1	TL	A-8490-2
3	St. Lucie 2	263°	SA533B1	TL	A-8490-2
4	St. Lucie 2	97°	SA533B1	TL	A-8490-2

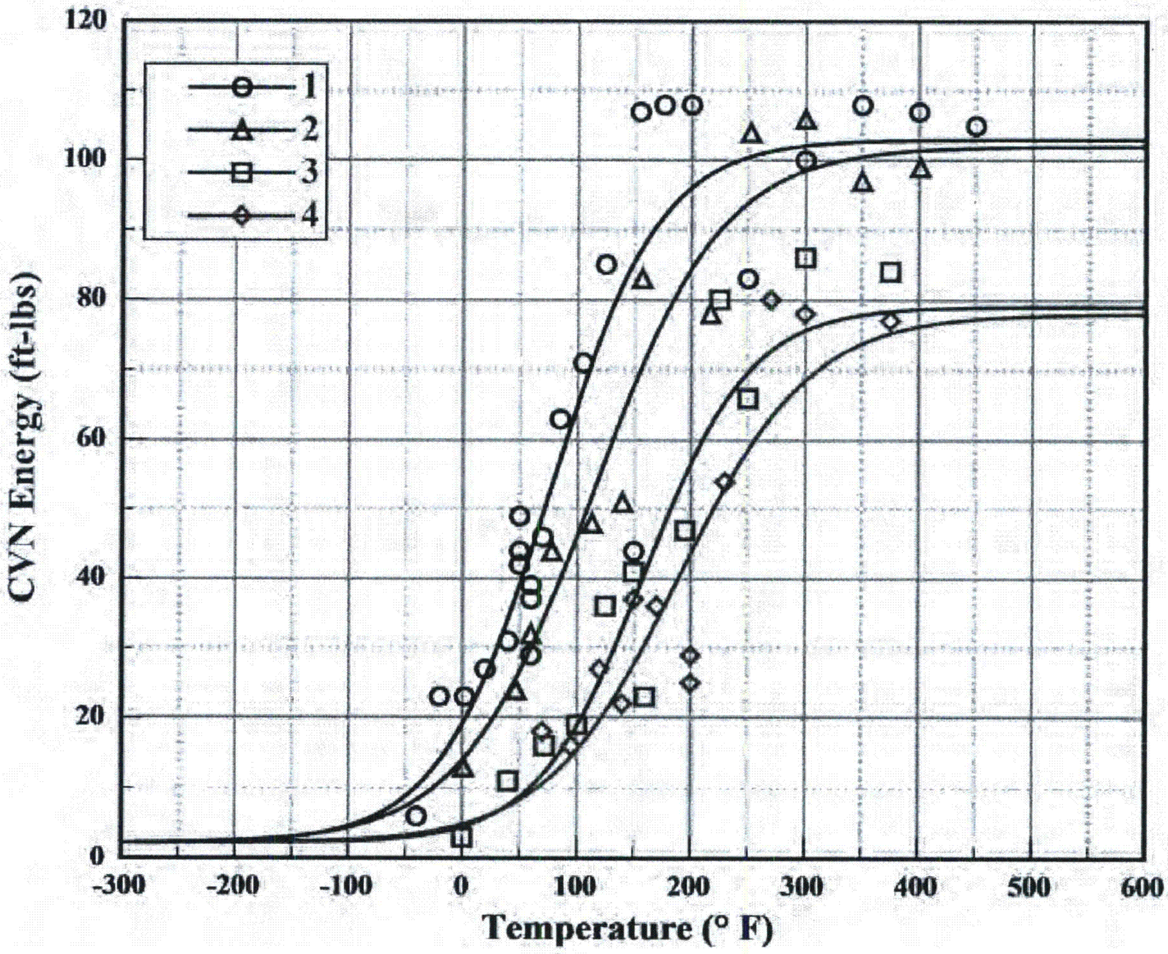


Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:25 AM

Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	103	0	30.4	0	71.5	0
2		2.2	102	-1	59.8	29.4	108.7	37.2
3		2.2	79	-24	133.1	102.7	182.1	110.6
4		2.2	78	-25	158	127.6	219.8	148.3

Figure 5-4(a) Charpy V-Notch Impact Energy vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation) – Continued

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 4/10/2015 7:47 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	TL	A-8490-2
2	St. Lucie 2	83°	SA533B1	TL	A-8490-2
3	St. Lucie 2	263°	SA533B1	TL	A-8490-2
4	St. Lucie 2	97°	SA533B1	TL	A-8490-2

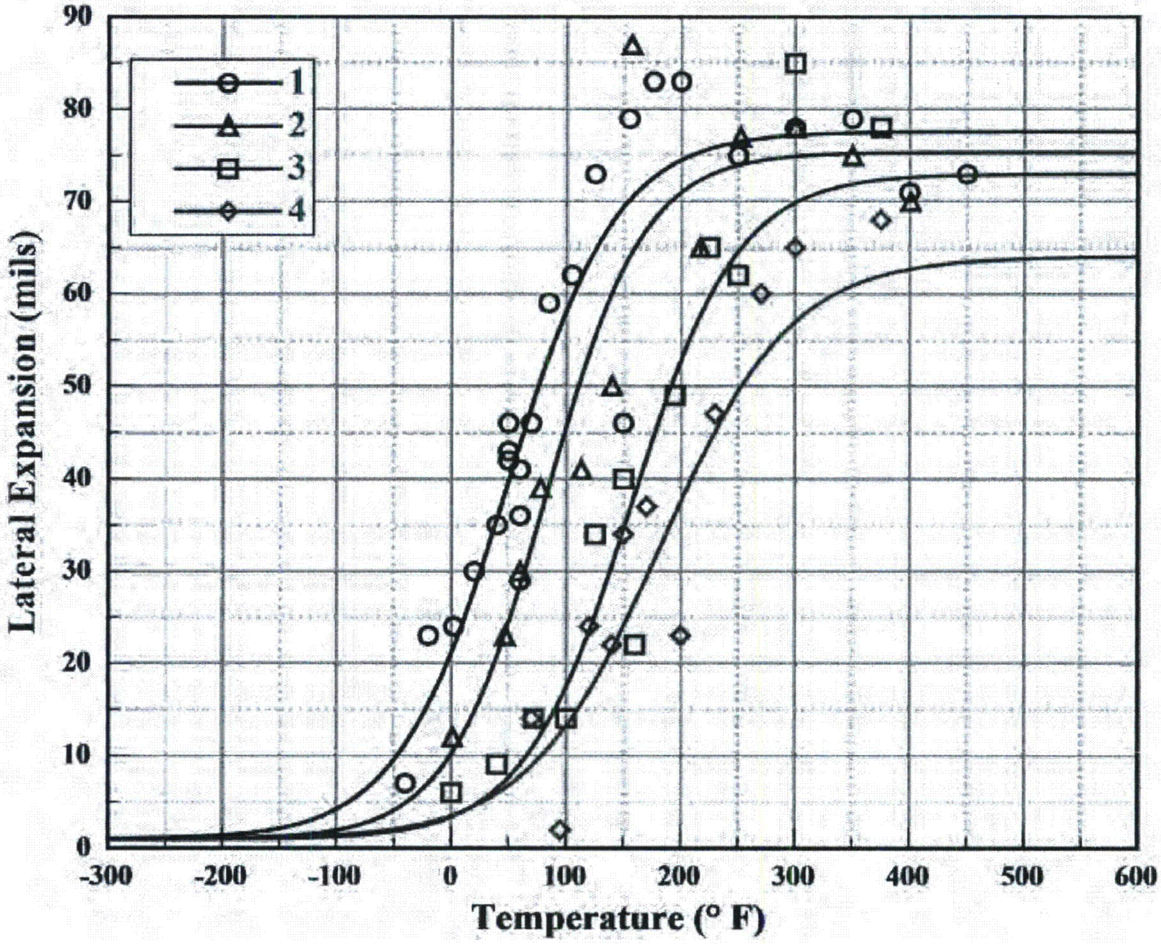


Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 4/10/2015 7:47 AM

Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0	1	77.6	0	38.2	0
2	0	1	75.37	-2.23	75.6	37.4
3	0	1	73	-4.60	151.3	113.1
4	0	1	64	-13.6	190.7	152.5

Figure 5-5(a) Charpy V-Notch Lateral Expansion vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation) – Continued

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:19 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	TL	A-8490-2
2	St. Lucie 2	83°	SA533B1	TL	A-8490-2
3	St. Lucie 2	263°	SA533B1	TL	A-8490-2
4	St. Lucie 2	97°	SA533B1	TL	A-8490-2

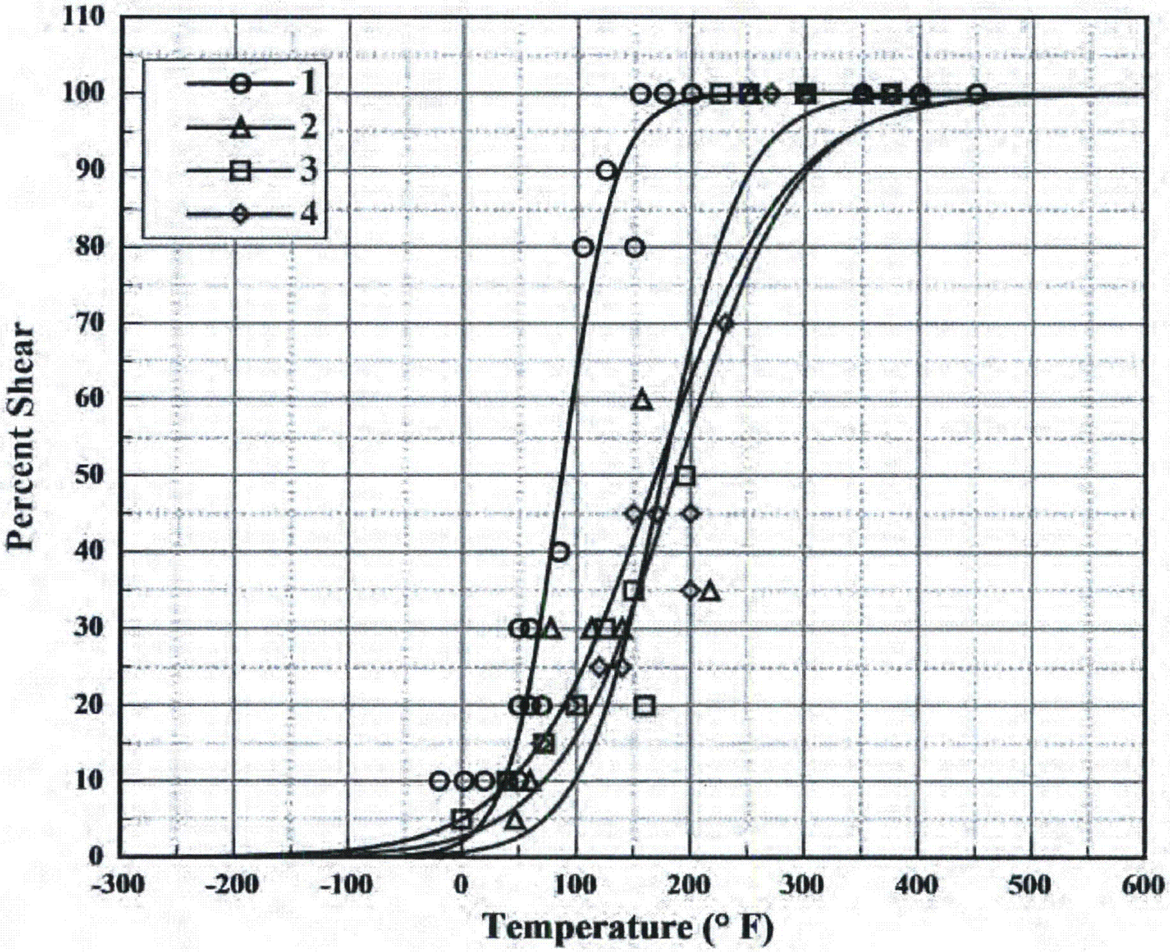


Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)

IS PLATE M-605-1 (TRANSVERSE)

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:19 AM

Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		0	100	0	87.1	0
2		0	100	0	169.6	82.5
3		0	100	0	175.2	88.1
4		0	100	0	187.3	100.2

Figure 5-6(a) Charpy V-Notch Percent Shear vs. Temperature for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation) – Continued

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:28 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SAW	NA	83637
2	St. Lucie 2	83°	SAW	NA	83637
3	St. Lucie 2	263°	SAW	NA	83637
4	St. Lucie 2	97°	SAW	NA	83637

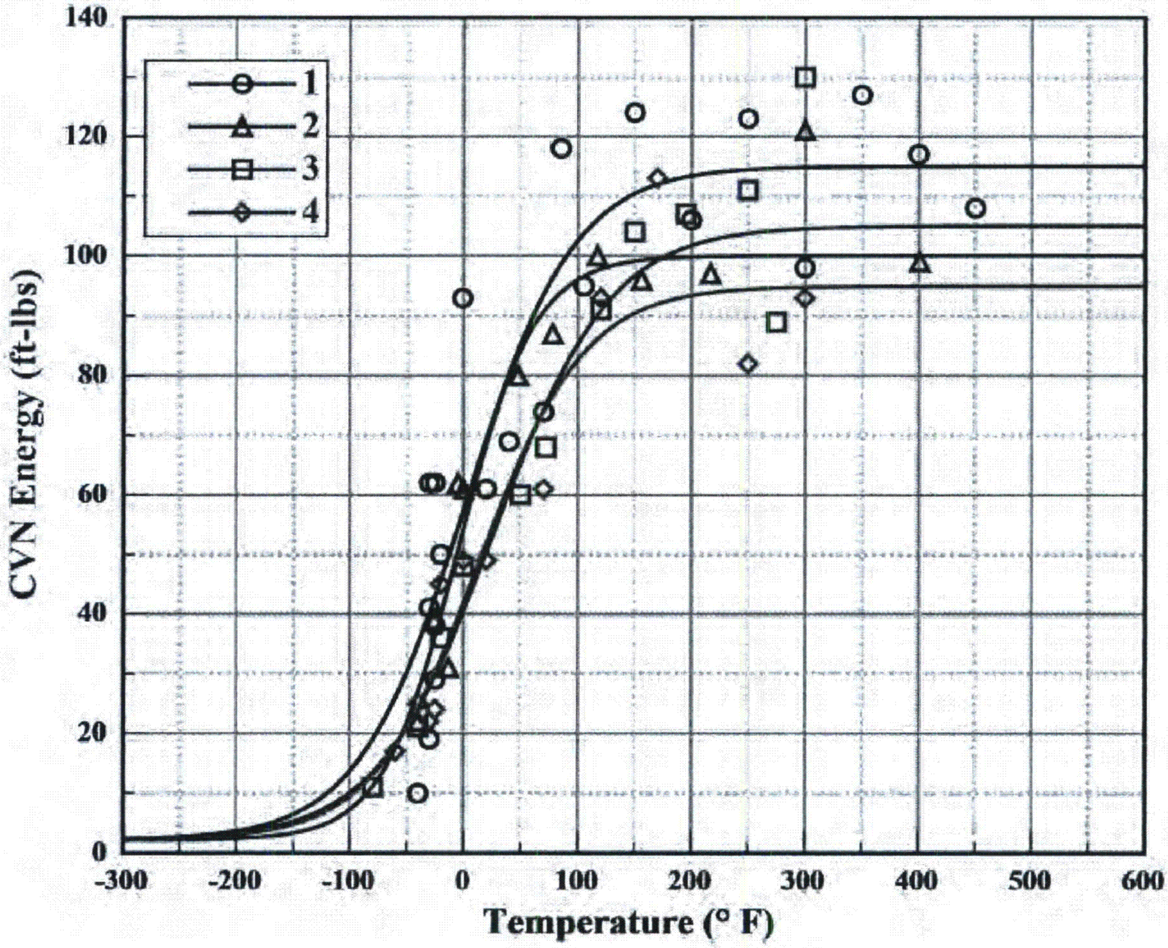


Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:28 AM

Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	115	0	-50.5	0	-12.5	0
2		2.2	100	-15	-34.7	15.8	-5.8	6.7
3		2.2	105	-10	-24	26.5	19.4	31.9
4		2.2	95	-20	-25.7	24.8	16.2	28.7

Figure 5-7(a) Charpy V-Notch Impact Energy vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637) – Continued

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:46 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SAW	NA	83637
2	St. Lucie 2	83°	SAW	NA	83637
3	St. Lucie 2	263°	SAW	NA	83637
4	St. Lucie 2	97°	SAW	NA	83637

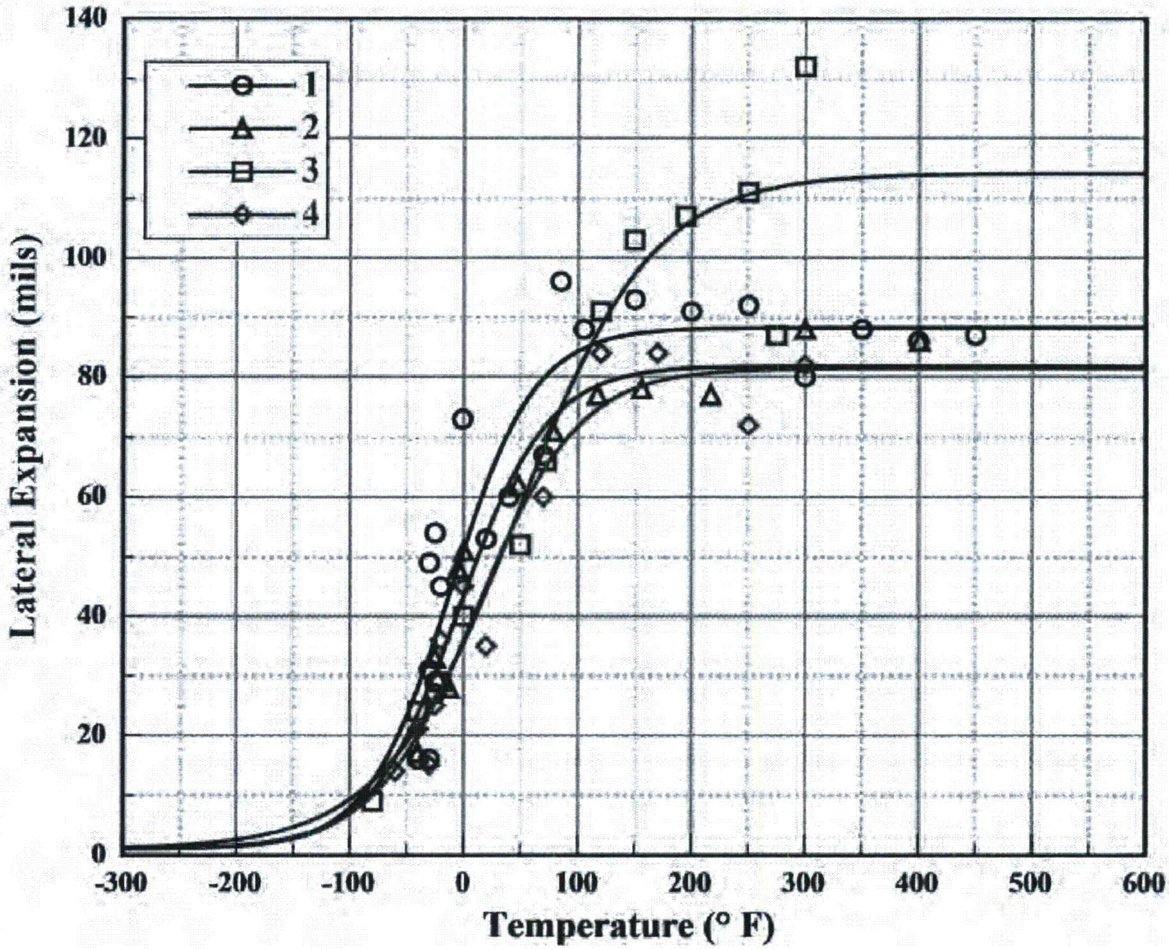


Figure 5-8 **Charpy V-Notch Lateral Expansion vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)**

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:46 AM

Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1		1	88.28	0	-27.1	0
2		1	81.86	-6.42	-16	11.1
3		1	114.16	25.88	-1.1	26
4		1	81.45	-6.83	-2	25.1

Figure 5-8(a) Charpy V-Notch Lateral Expansion vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637) – Continued

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:51 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SAW	NA	83637
2	St. Lucie 2	83°	SAW	NA	83637
3	St. Lucie 2	263°	SAW	NA	83637
4	St. Lucie 2	97°	SAW	NA	83637

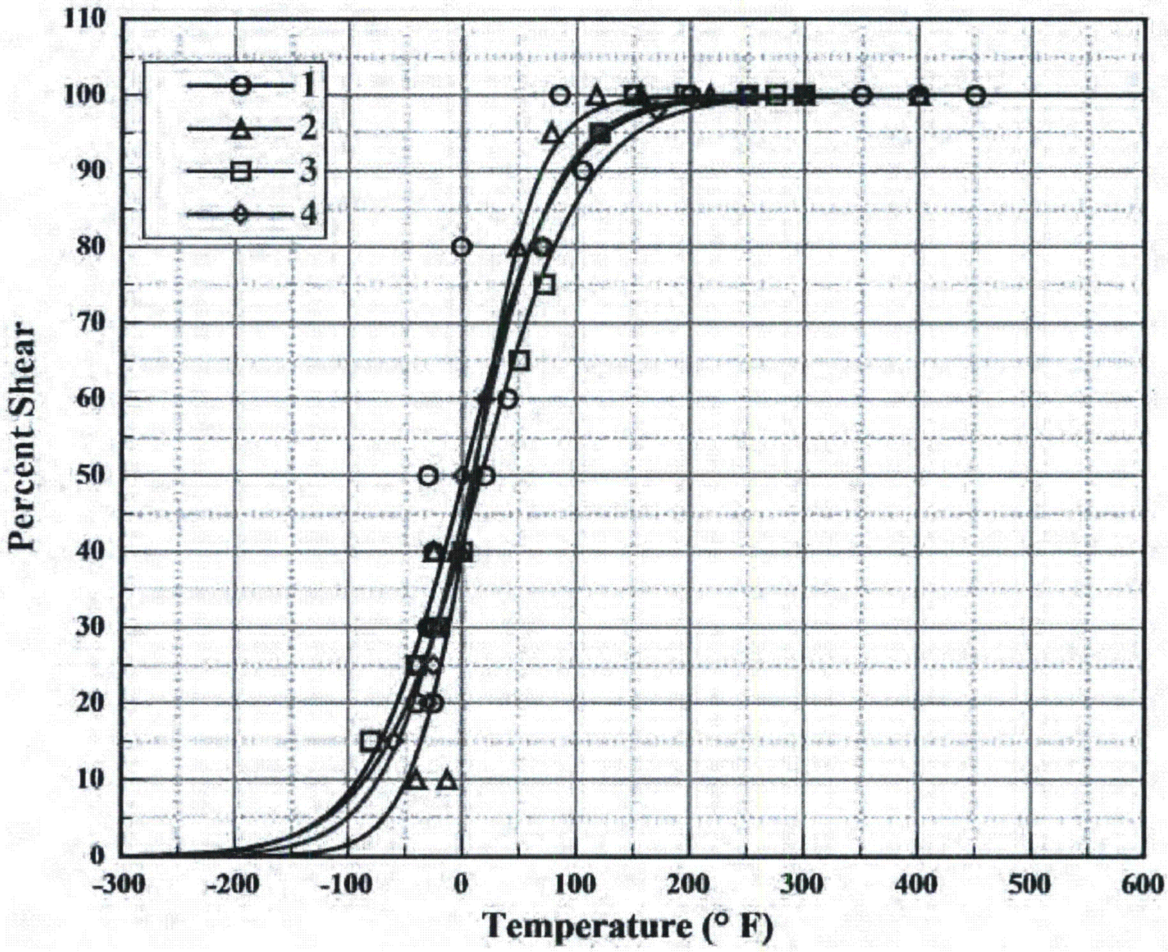


Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)

SURVEILLANCE PROGRAM WELD METAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 8:51 AM

Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		0	100	0	1.3	0
2		0	100	0	10.8	9.5
3		0	100	0	16.4	15.1
4		0	100	0	8.8	7.5

Figure 5-9(a) Charpy V-Notch Percent Shear vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637) – Continued

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 7:21 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	NA	A-8490-2
2	St. Lucie 2	83°	SA533B1	NA	A-8490-2
3	St. Lucie 2	263°	SA533B1	NA	A-8490-2
4	St. Lucie 2	97°	SA533B1	NA	A-8490-2

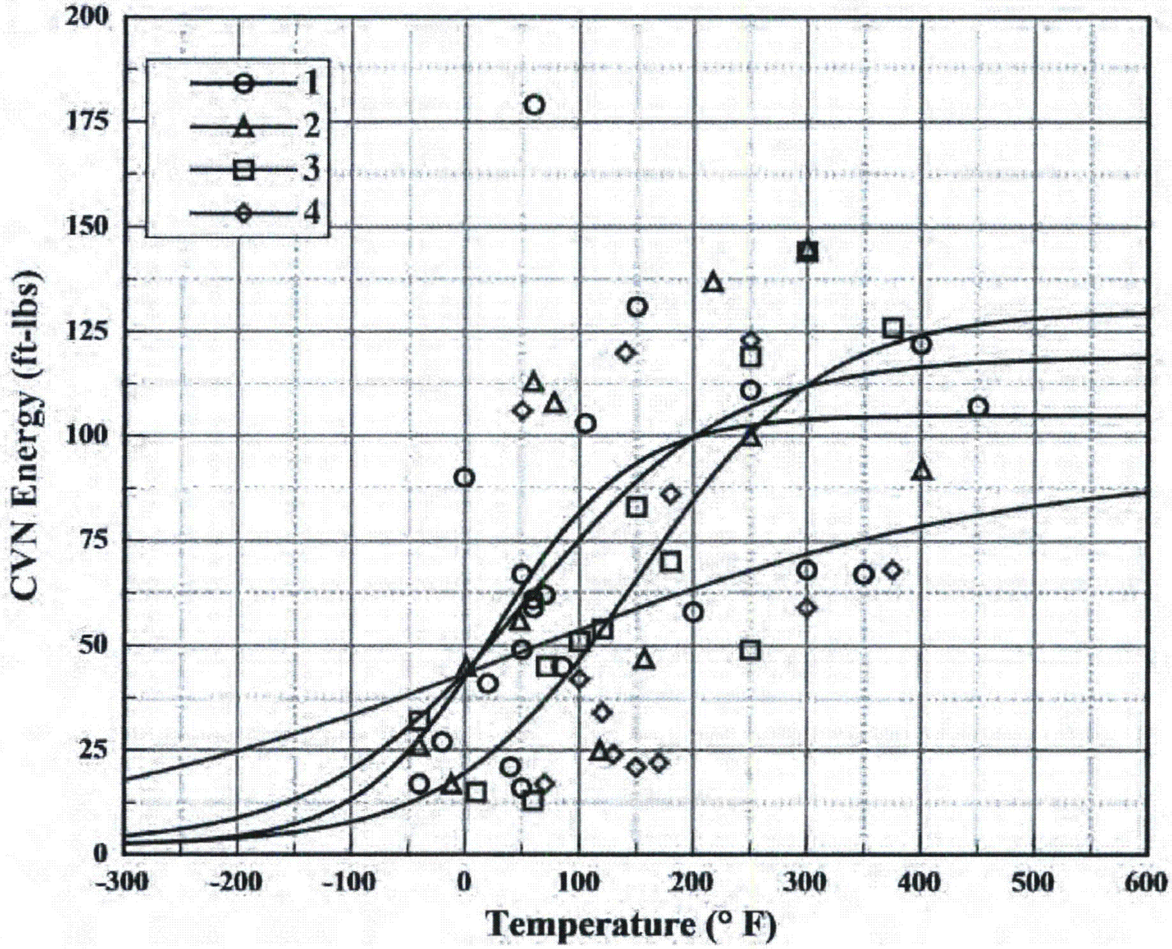


Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 7:21 AM

Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	105	0	-33.1	0	17.9	0
2		2.2	119	14	-51.5	-18.4	20.5	2.6
3		2.2	130	25	46.4	79.5	109.1	91.2
4		2.2	93	-12	-137	-103.9	64.1	46.2

Figure 5-10(a) Charpy V-Notch Impact Energy vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material – Continued

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 7:29 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	NA	A-8490-2
2	St. Lucie 2	83°	SA533B1	NA	A-8490-2
3	St. Lucie 2	263°	SA533B1	NA	A-8490-2
4	St. Lucie 2	97°	SA533B1	NA	A-8490-2

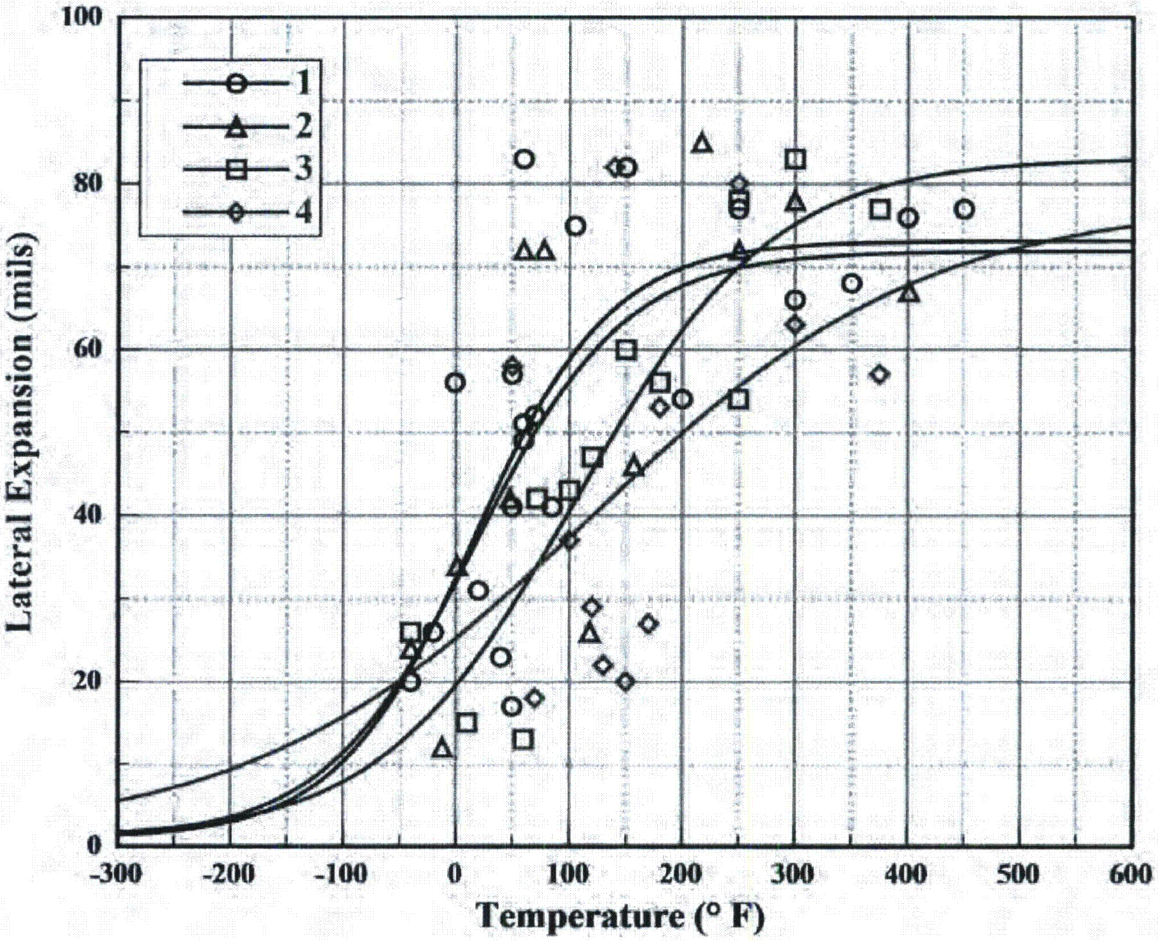


Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 7:29 AM

Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1		1	73.16	0	8.5	0
2		1	71.91	-1.25	11.3	2.8
3		1	83.1	9.94	76.6	68.1
4		1	77.97	4.81	81.6	73.1

Figure 5-11(a) Charpy V-Notch Lateral Expansion vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material – Continued

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 8:21 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	NA	A-8490-2
2	St. Lucie 2	83°	SA533B1	NA	A-8490-2
3	St. Lucie 2	263°	SA533B1	NA	A-8490-2
4	St. Lucie 2	97°	SA533B1	NA	A-8490-2

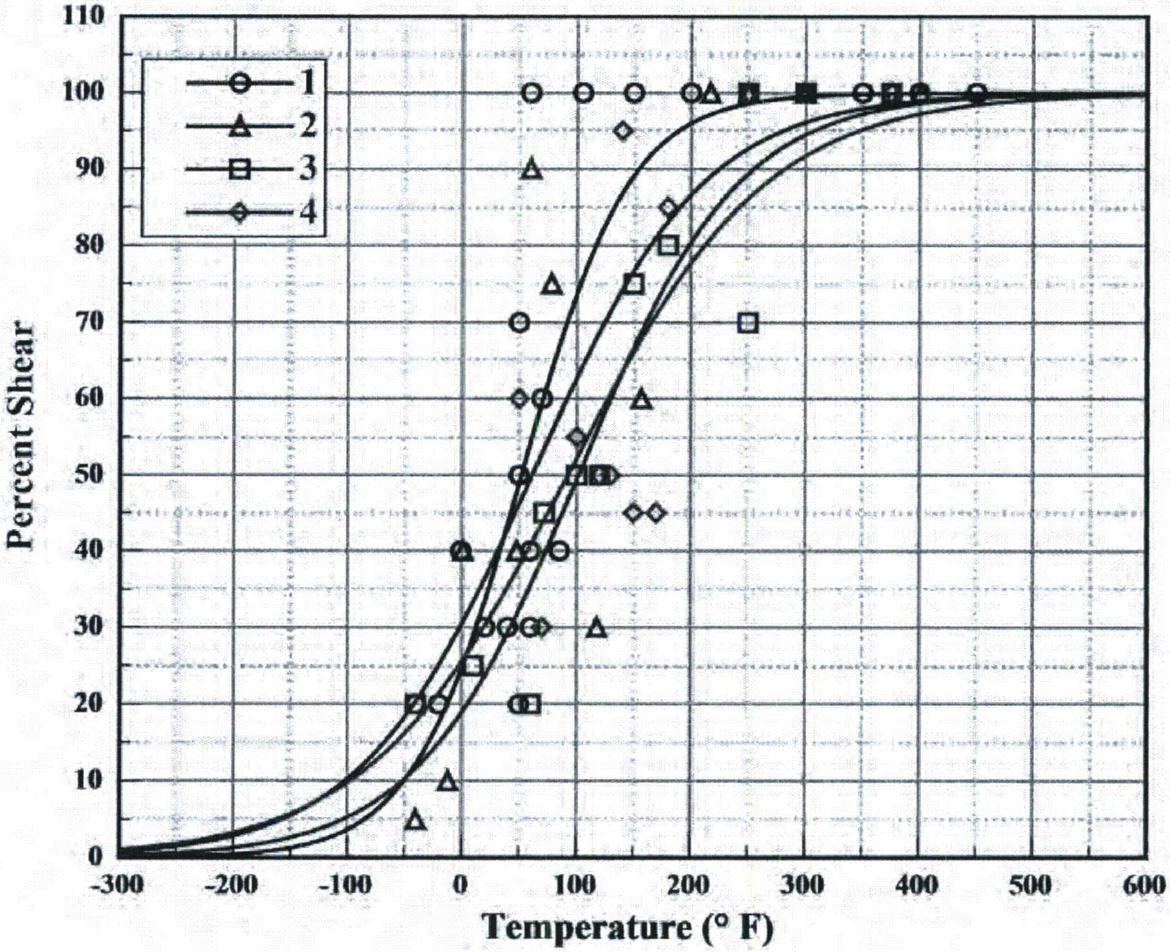


Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material

HEAT AFFECTED ZONE

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 1/9/2015 8:21 AM

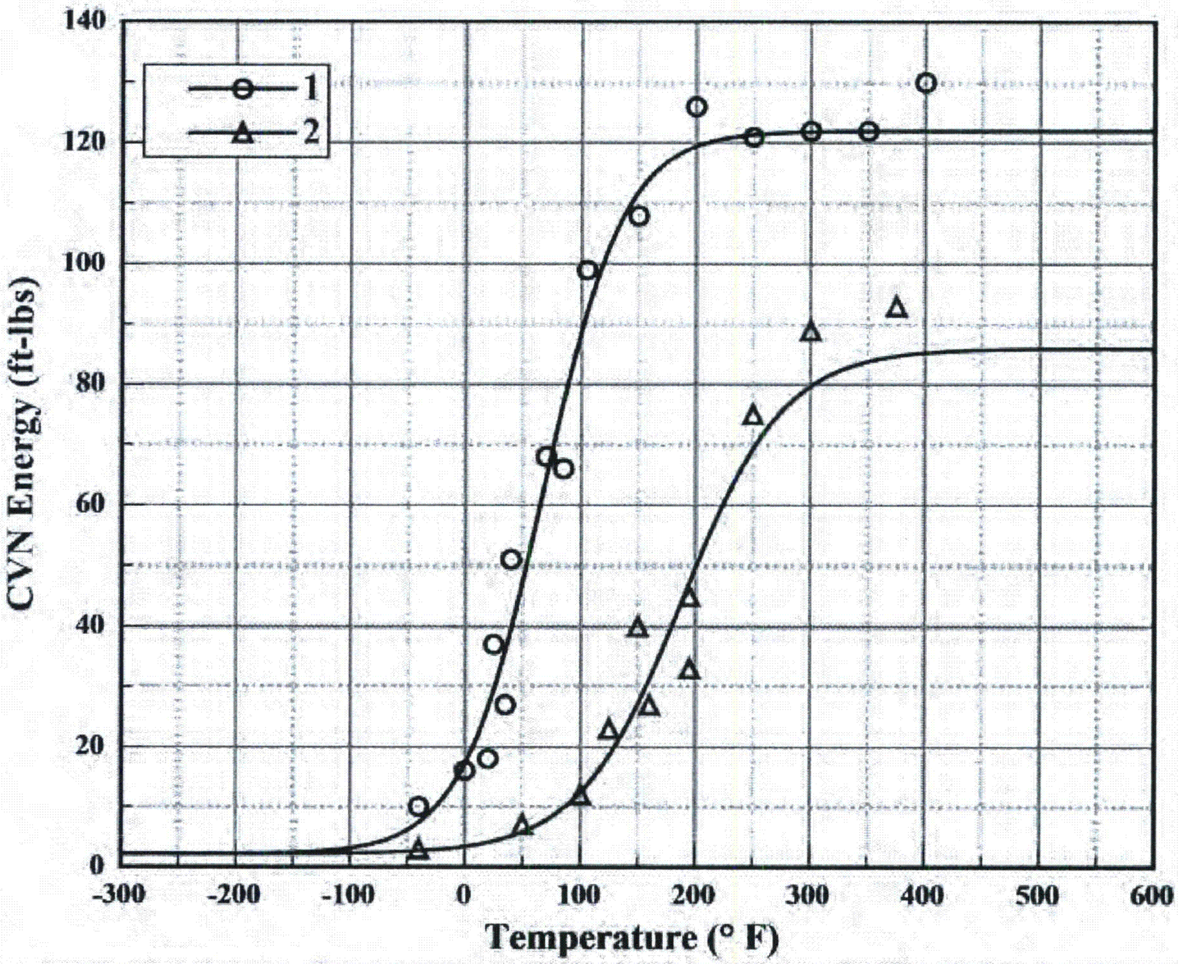
Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		0	100	0	52.1	0
2		0	100	0	61.5	9.4
3		0	100	0	101.7	49.6
4		0	100	0	92.4	40.3

Figure 5-12(a) Charpy V-Notch Percent Shear vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material – Continued

STANDARD REFERENCE MATERIAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 9:09 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	HSST-01MY
2	St. Lucie 2	263°	SA533B1	LT	HSST-01MY



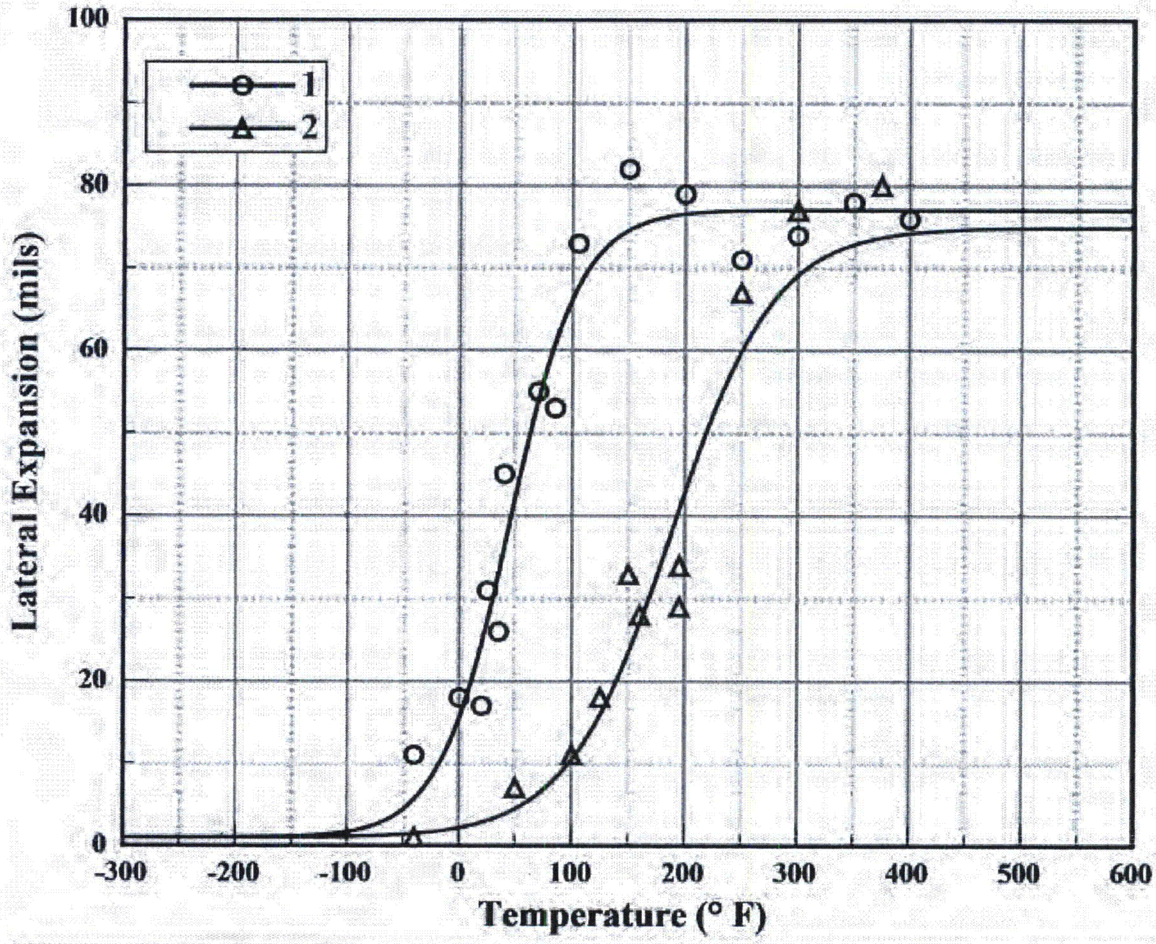
Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1		2.2	122	0	25.9	0	53.2	0
2		2.2	86	-36	157.1	131.2	200.9	147.7

Figure 5-13 Charpy V-Notch Impact Energy vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Standard Reference Material

STANDARD REFERENCE MATERIAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 4/10/2015 7:50 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	HSST-01MY
2	St. Lucie 2	263°	SA533B1	LT	HSST-01MY



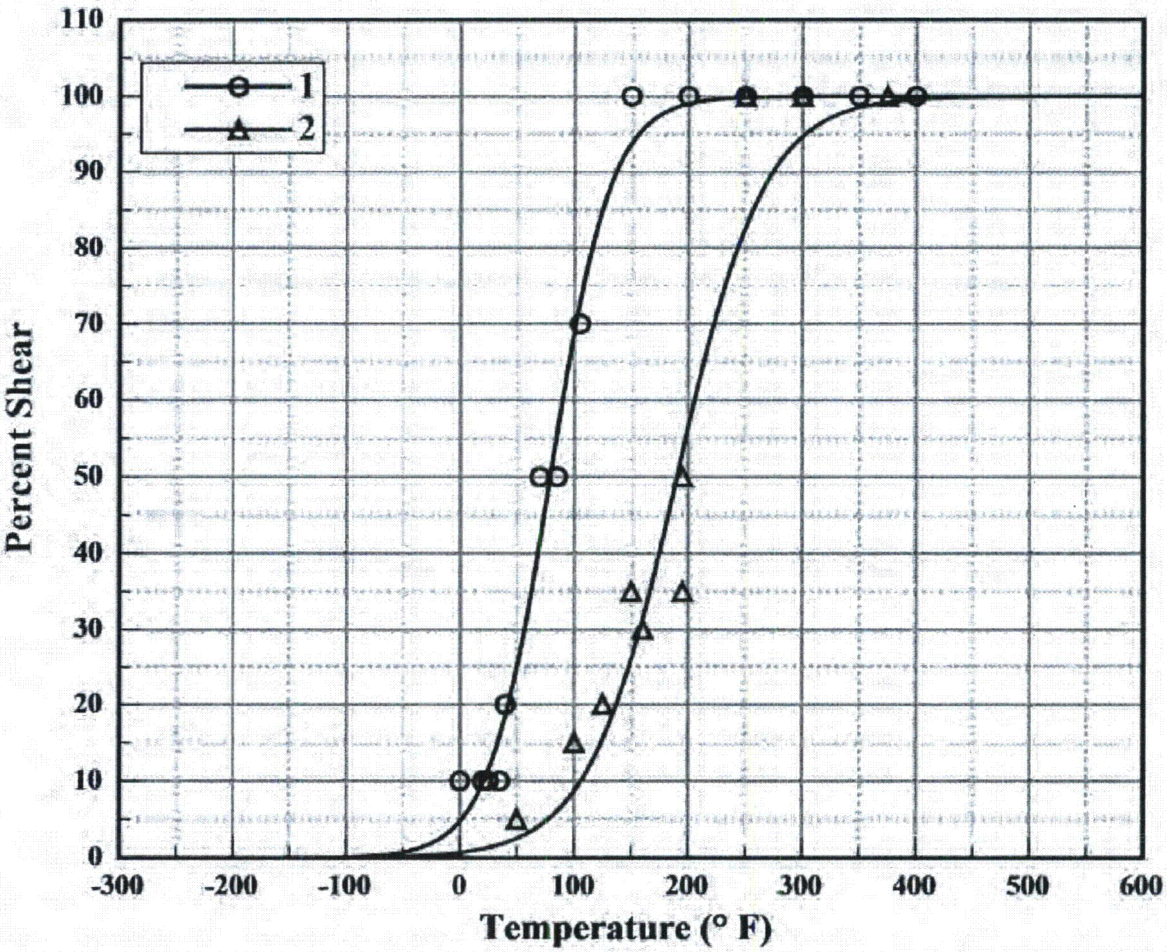
Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0	1	77.11	0	38	0
2	0	1	75	-2.11	180.7	142.7

Figure 5-14 Charpy V-Notch Lateral Expansion vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Standard Reference Material

STANDARD REFERENCE MATERIAL

CVGraph 6.0: Hyperbolic Tangent Curve Printed on 12/9/2014 9:14 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	St. Lucie 2	Unirrad	SA533B1	LT	HSST-01MY
2	St. Lucie 2	263°	SA533B1	LT	HSST-01MY



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1		0	100	0	79	0
2		0	100	0	190.1	111.1

Figure 5-15 Charpy V-Notch Percent Shear vs. Temperature for the St. Lucie Unit 2 Reactor Vessel Standard Reference Material

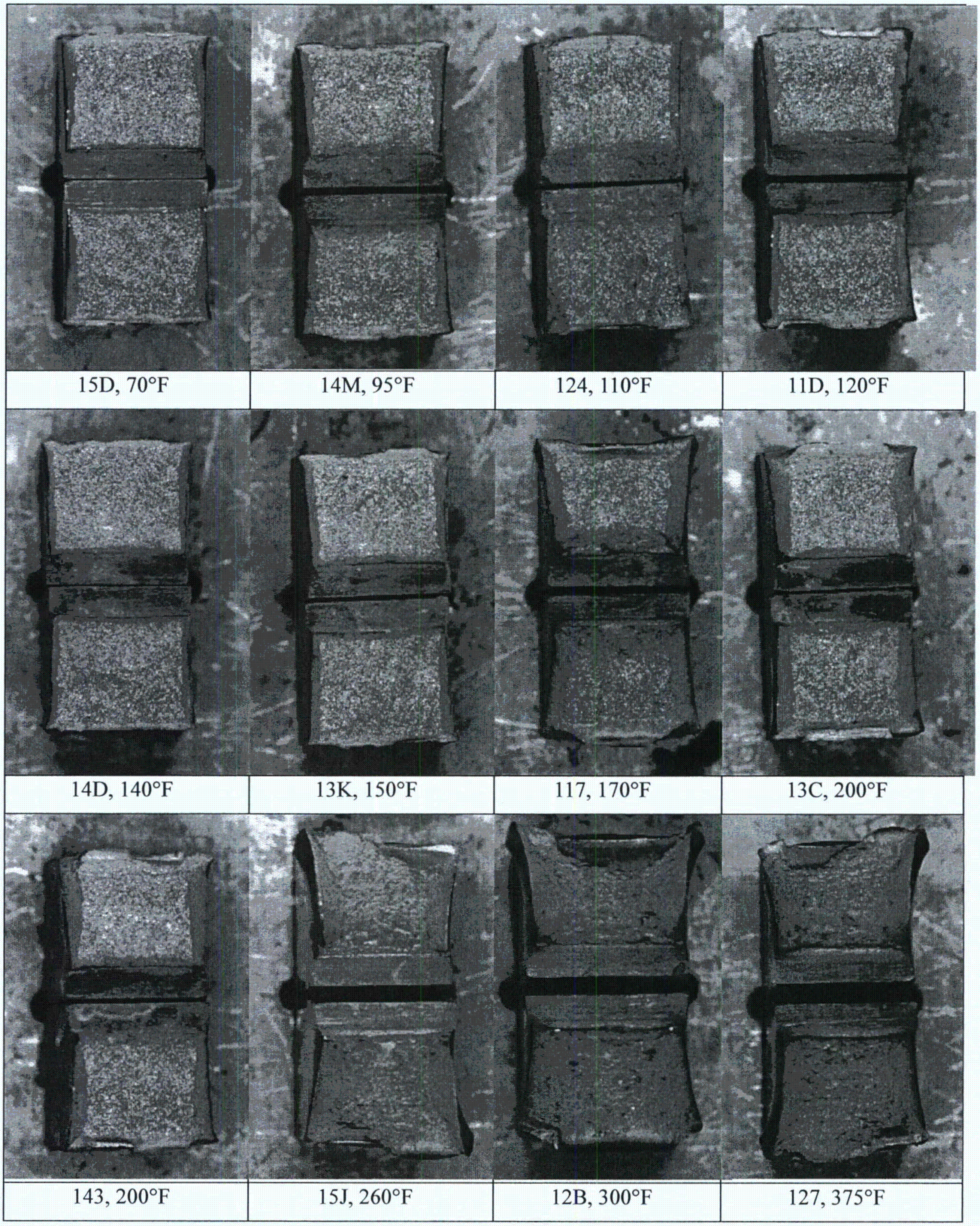


Figure 5-16 Charpy Impact Specimen Fracture Surfaces for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Longitudinal Orientation)

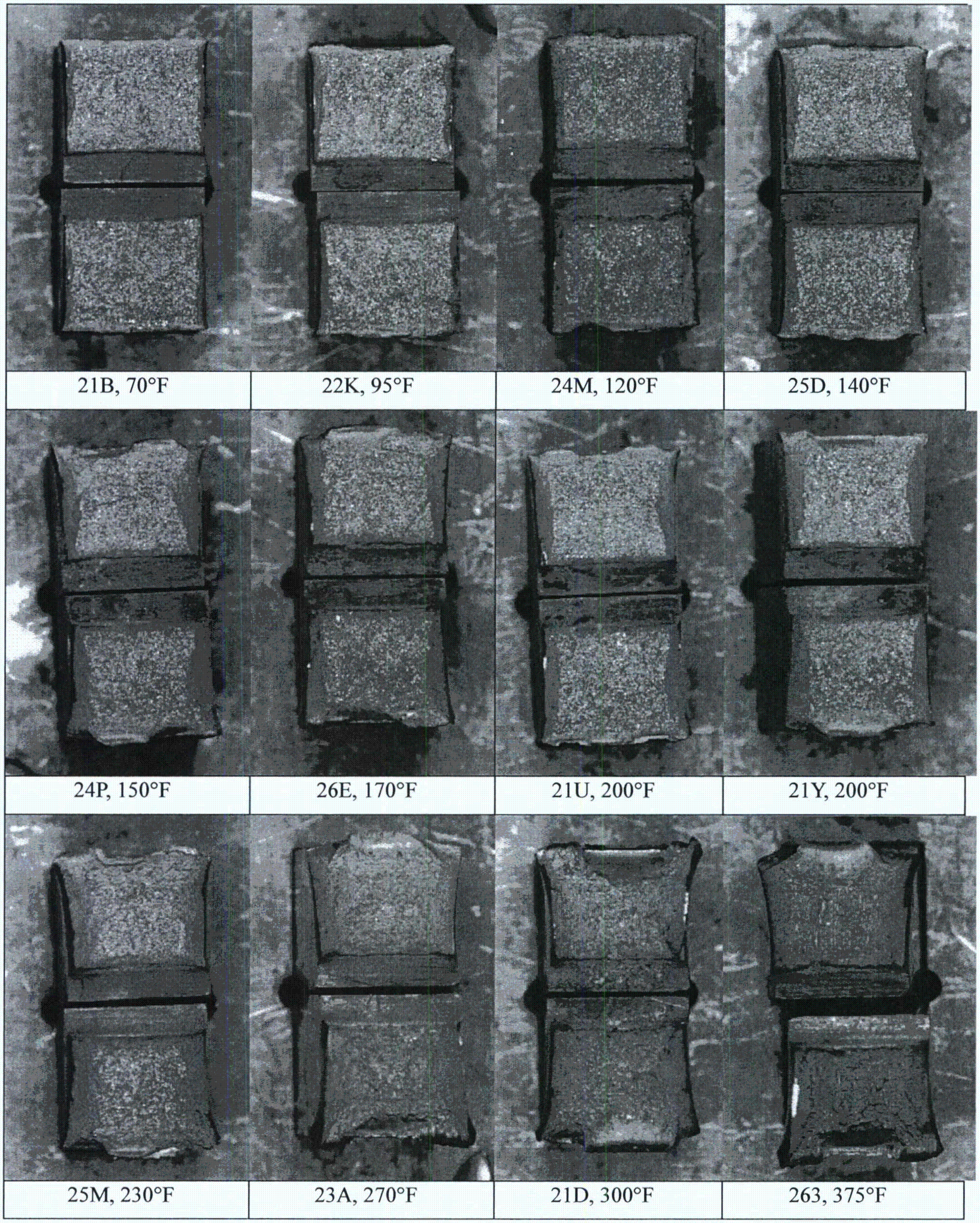


Figure 5-17 Charpy Impact Specimen Fracture Surfaces for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)

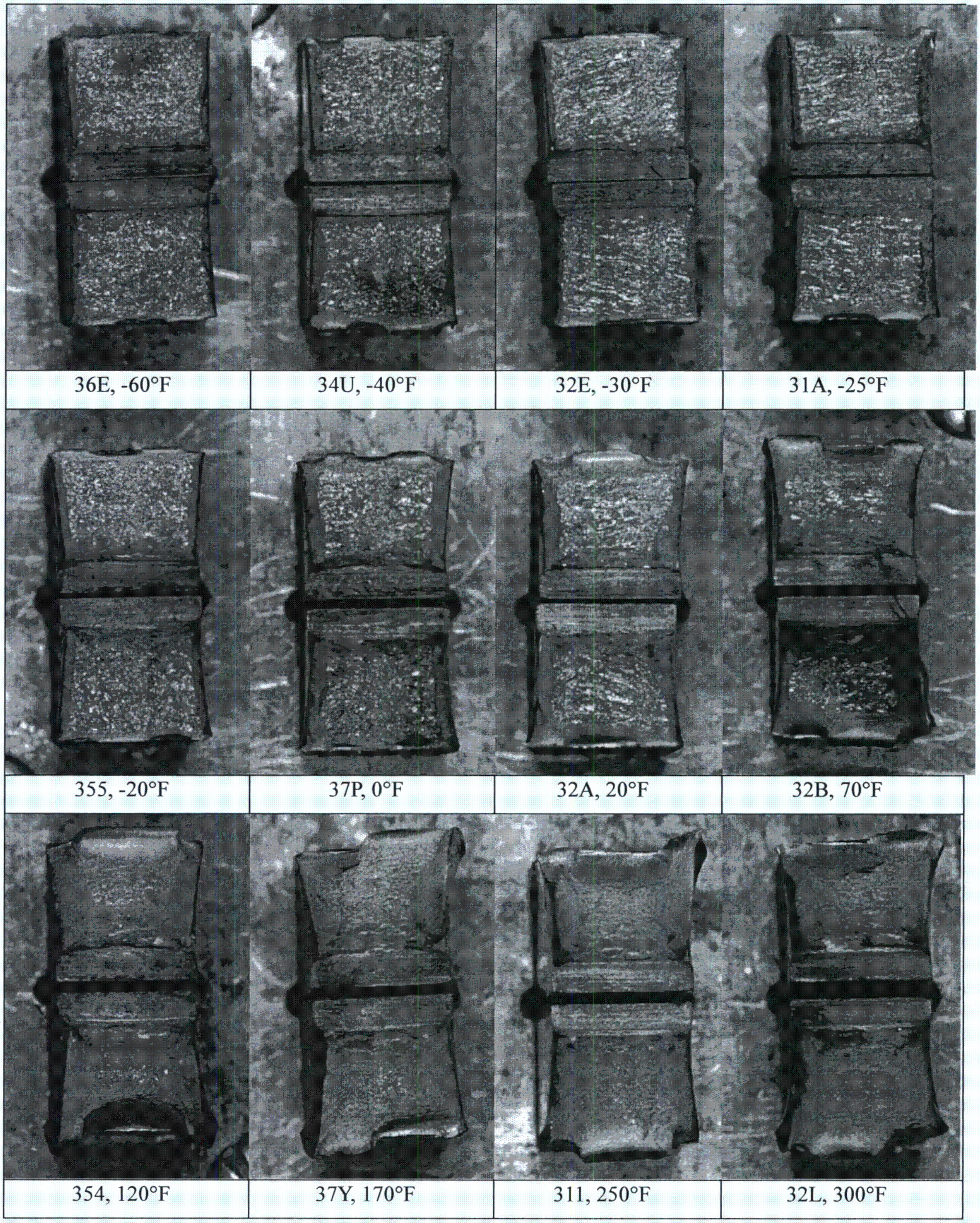
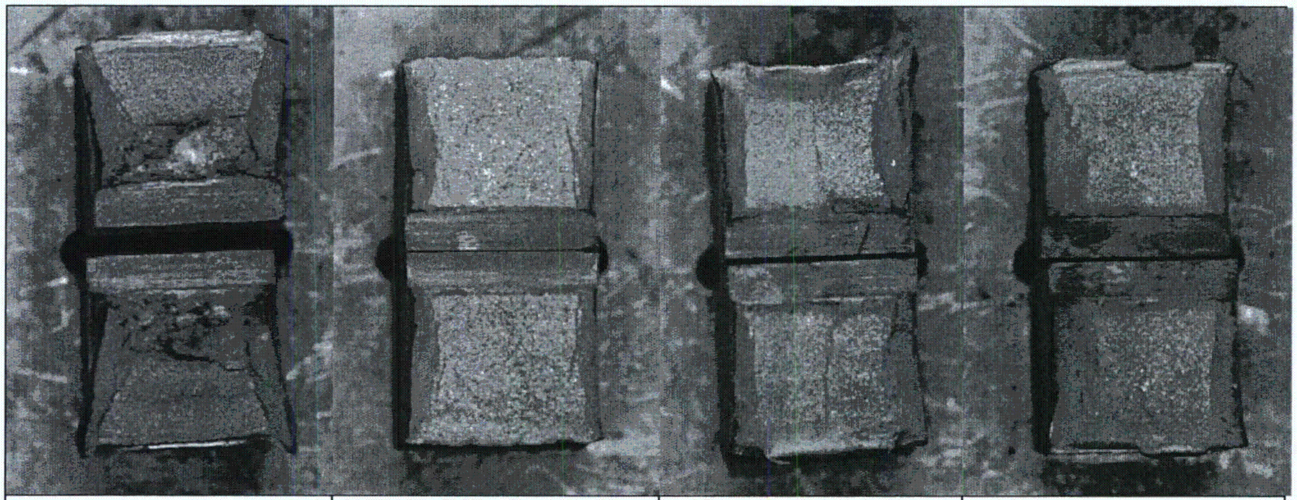
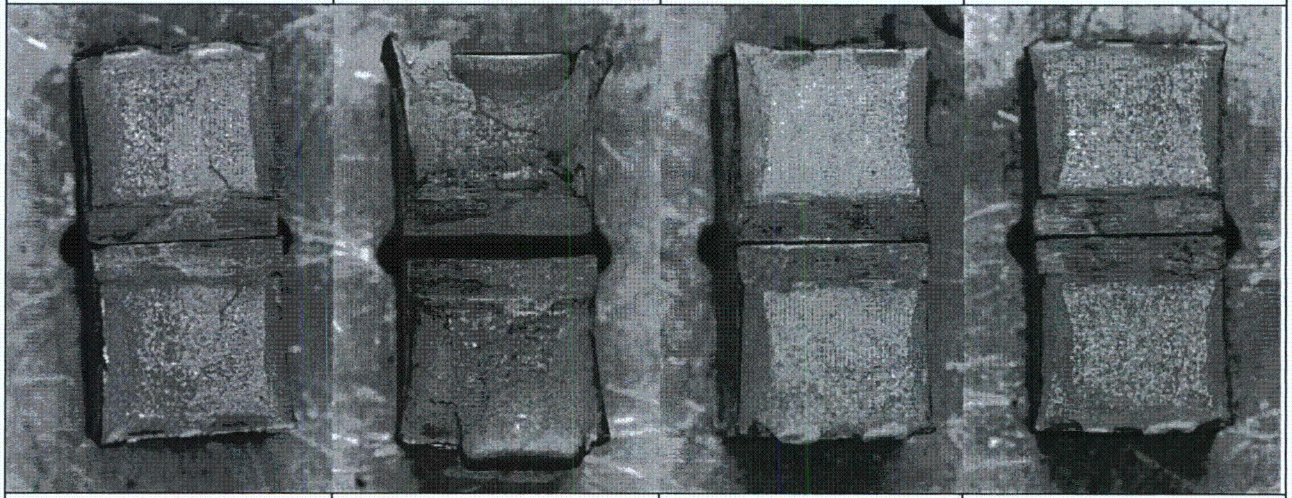


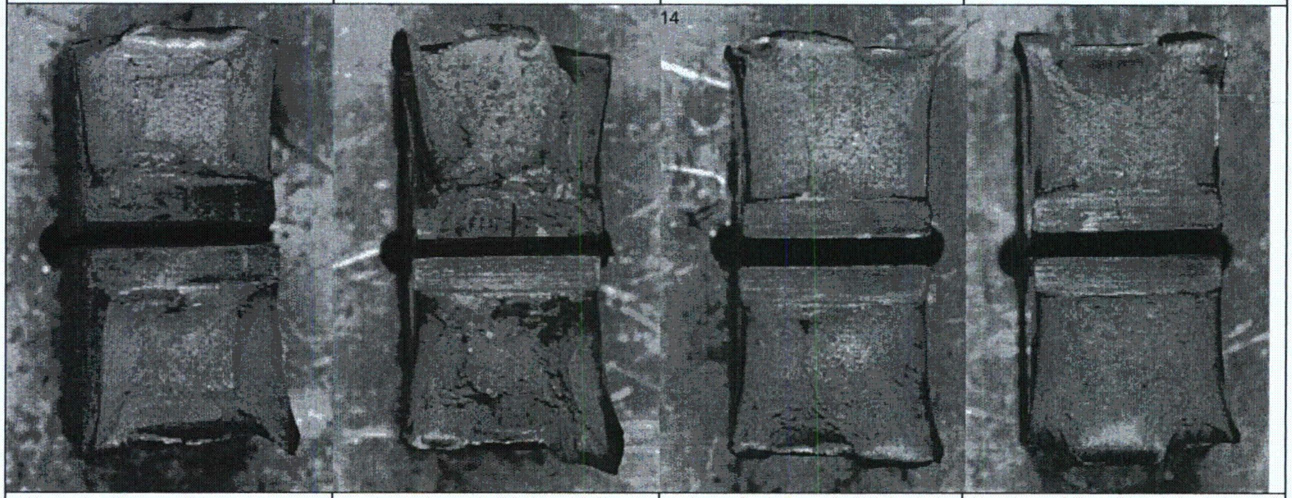
Figure 5-18 Charpy Impact Specimen Fracture Surfaces for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)



425, 50°F	43D, 70°F	41P, 100°F	456, 120°F

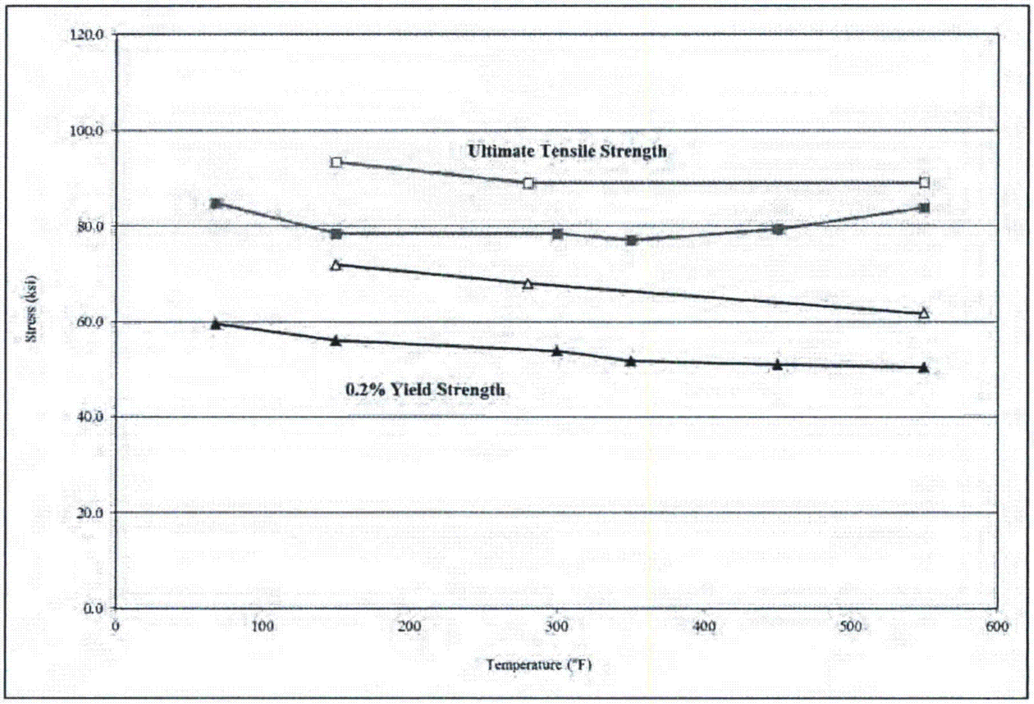


46Y, 130°F	43K, 140°F	47L, 150°F	473, 170°F



42C, 180°F	42D, 250°F	46A, 300°F	47D, 375°F

Figure 5-19 Charpy Impact Specimen Fracture Surfaces for the St. Lucie Unit 2 Reactor Vessel Heat-Affected Zone Material



Legend: ▲and ● and ■ are unirradiated
Δ and ○ and □ are irradiated to 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

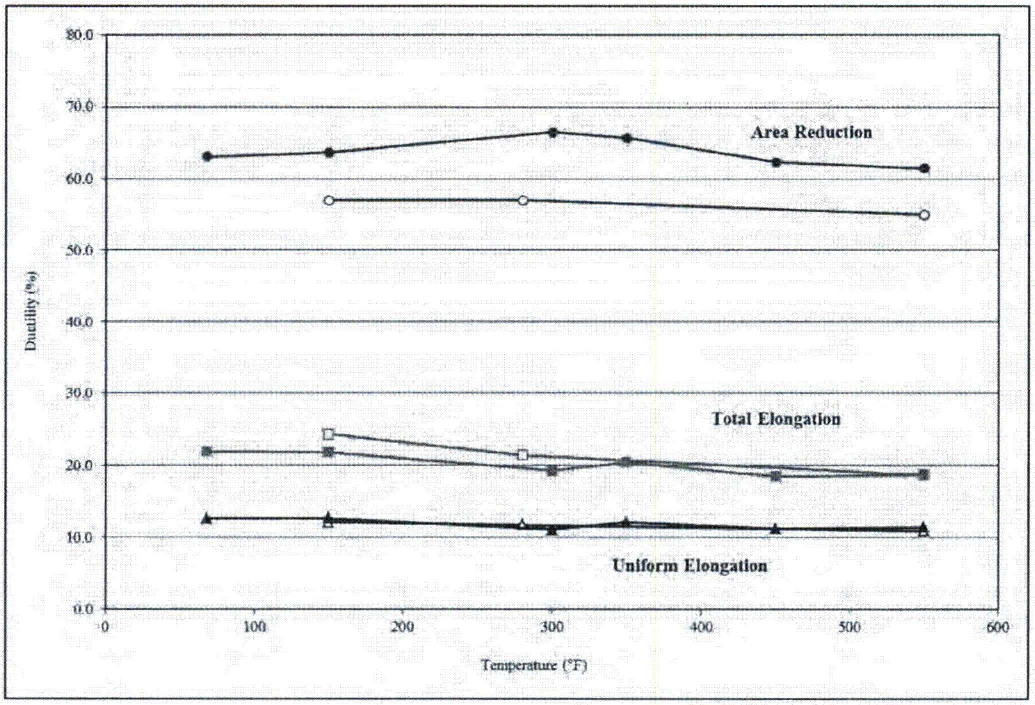
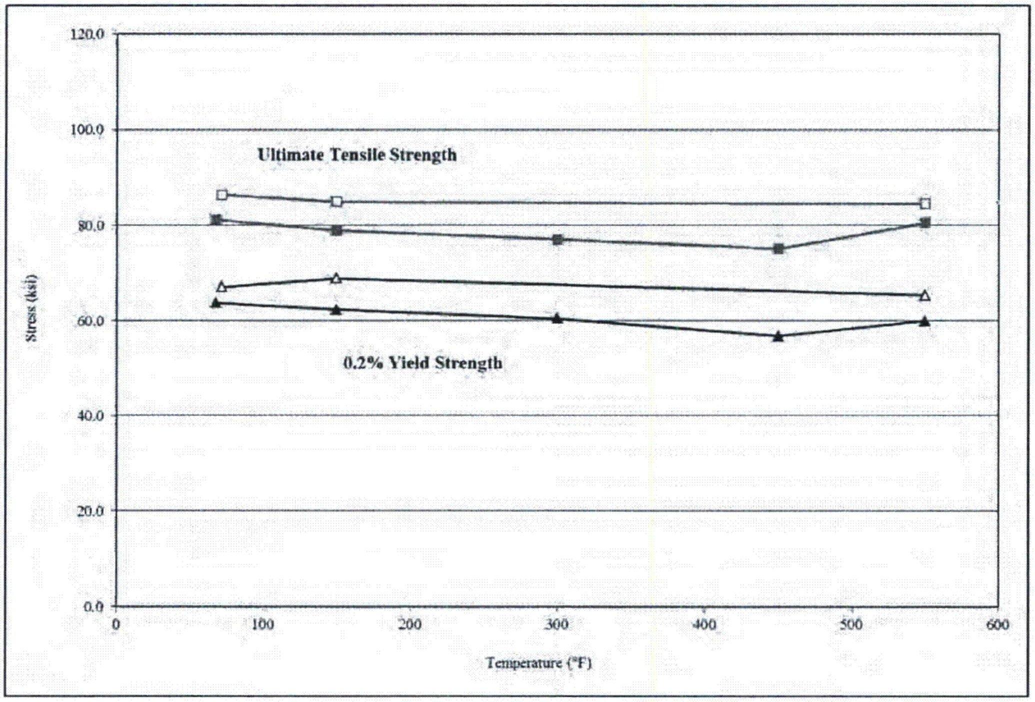


Figure 5-20 Tensile Properties for St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)



Legend: ▲and ● and ■ are unirradiated
Δ and ○ and □ are irradiated to 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

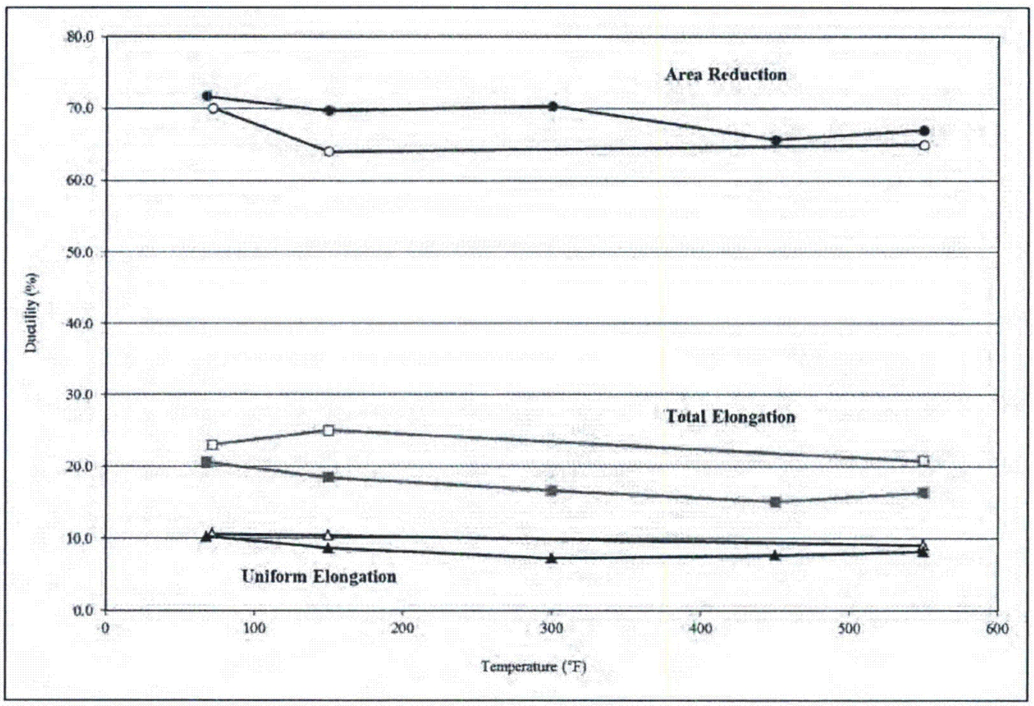
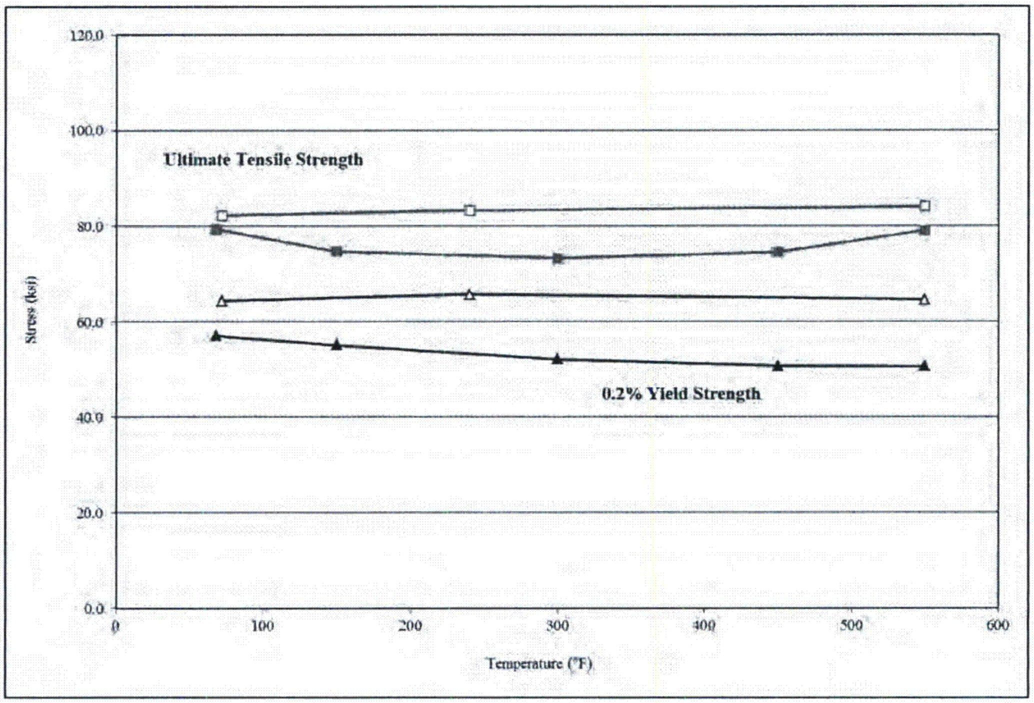


Figure 5-21 Tensile Properties for the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)



Legend: ▲ and ● and ■ are unirradiated
Δ and ○ and □ are irradiated to 2.25 x 10^{19} n/cm^2 (E > 1.0 MeV)

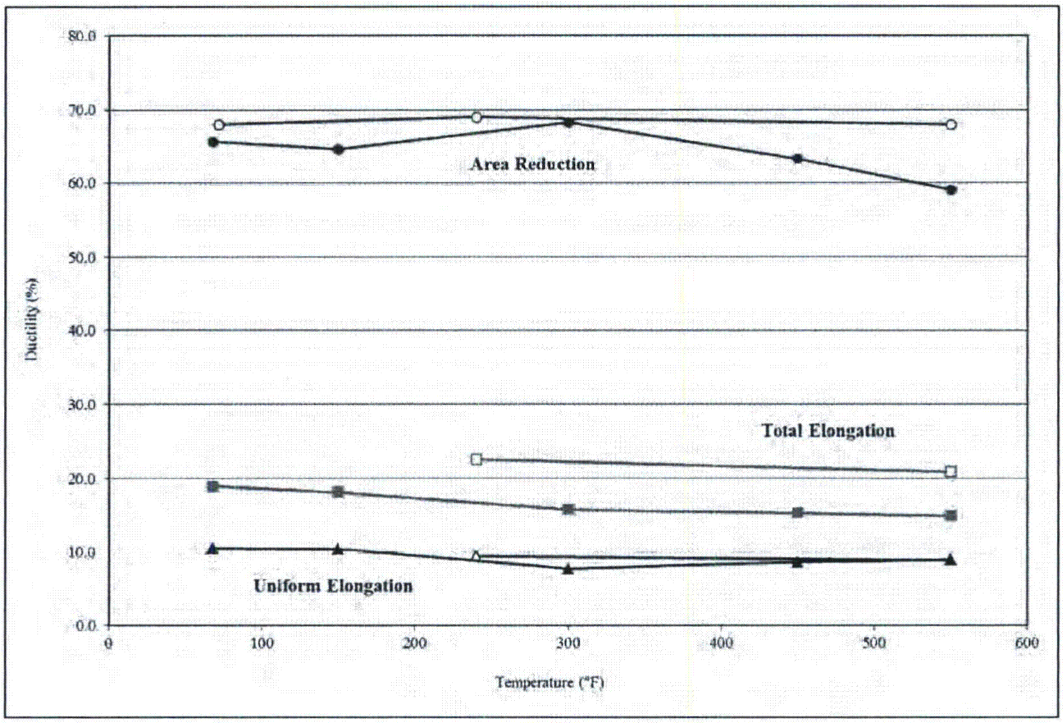
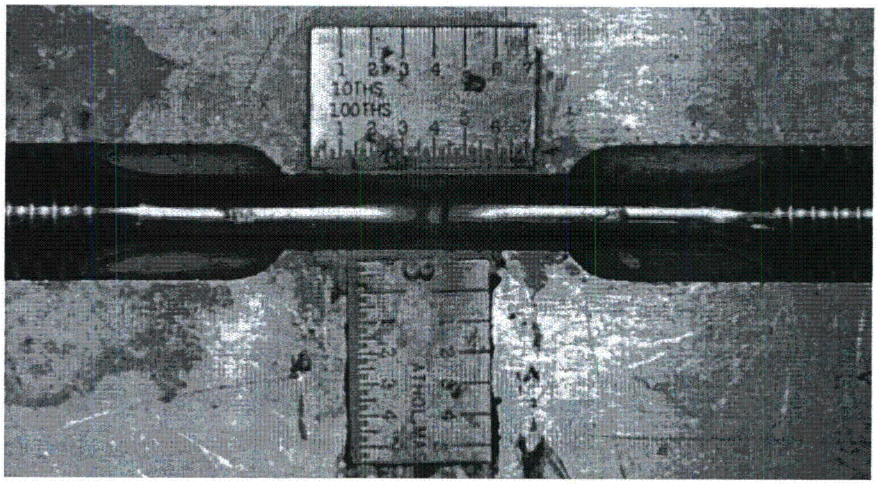
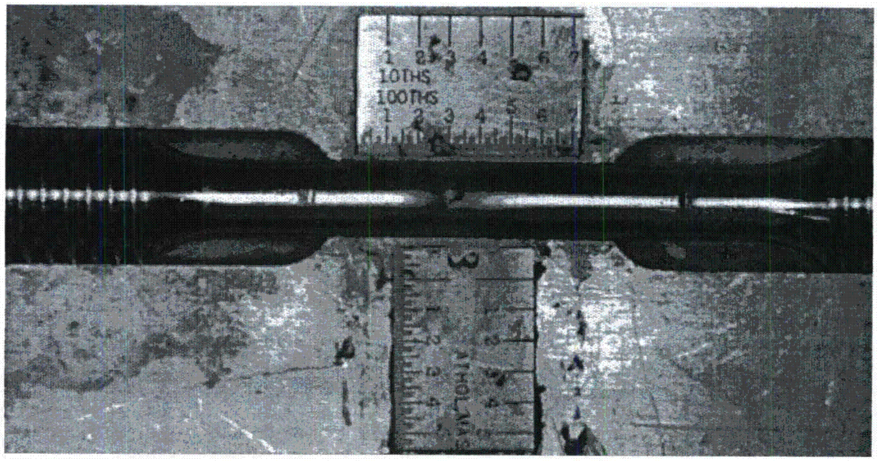


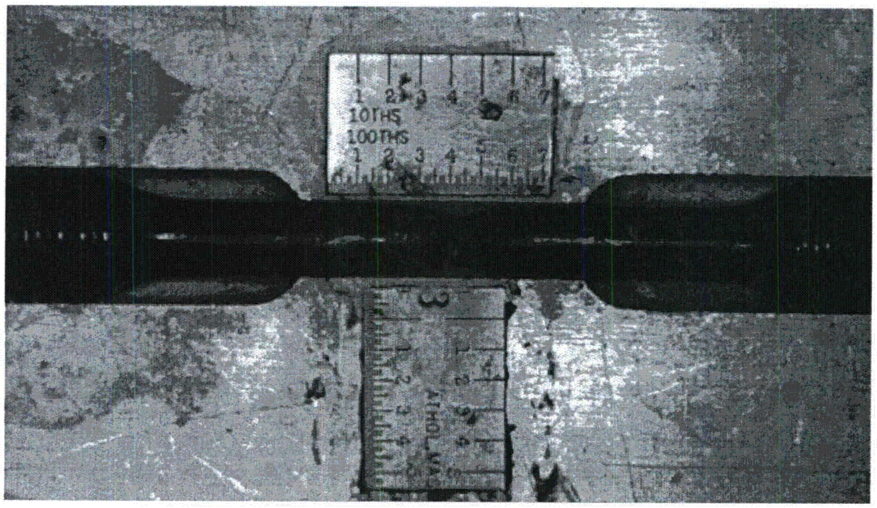
Figure 5-22 Tensile Properties for the St. Lucie Unit 2 Reactor Vessel Heat Affected Zone Material



2L5 – Tested at 150°F

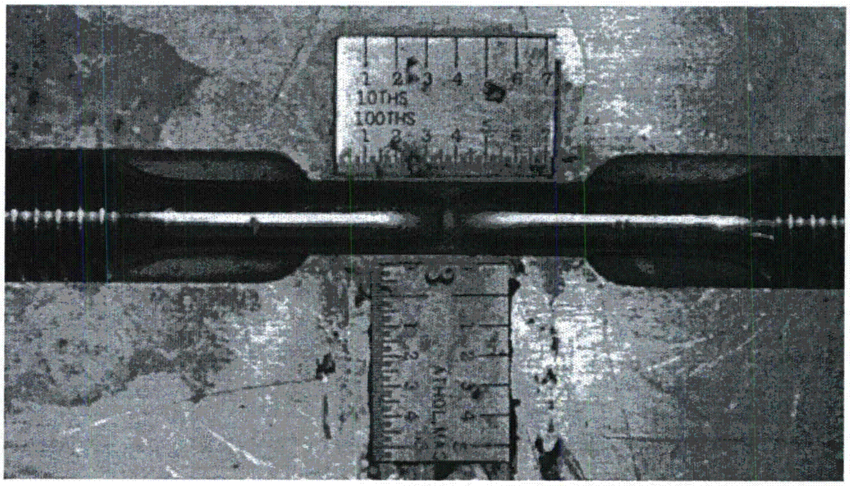


2KU – Tested at 280°F

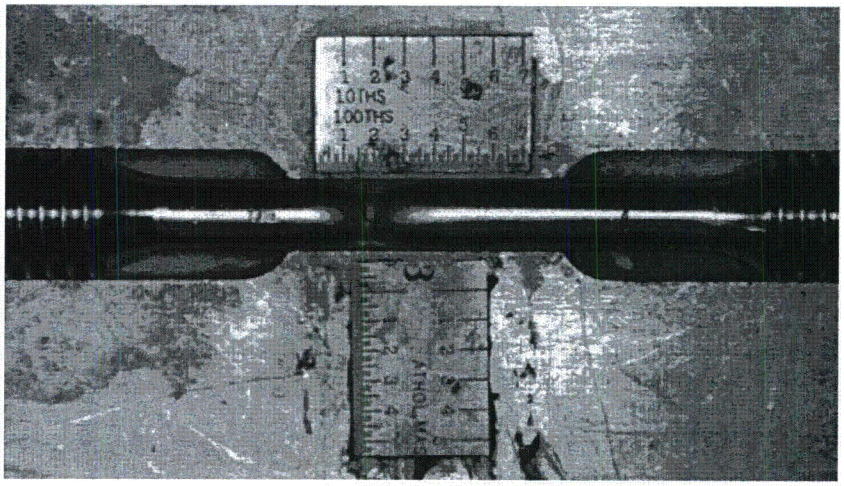


2JM – Tested at 550°F

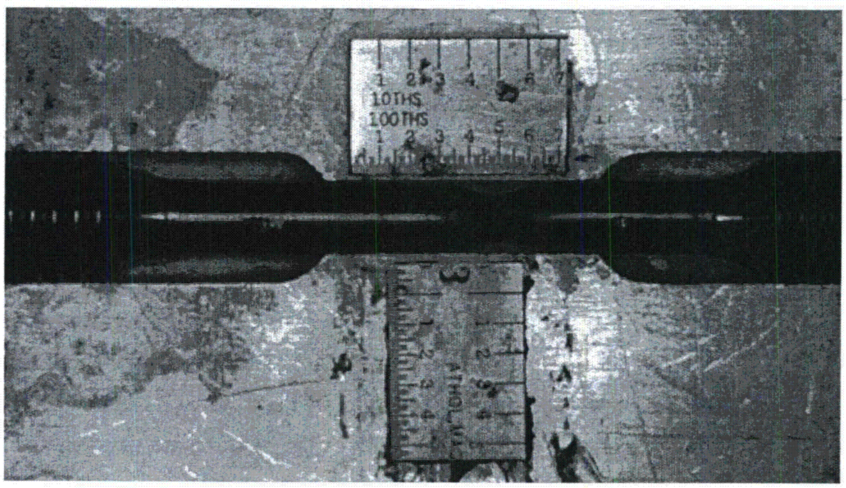
Figure 5-23 Fractured Tensile Specimens from St. Lucie Unit 2 Reactor Vessel Intermediate Shell Plate M-605-1 (Transverse Orientation)



3J4 – Tested at 72°F

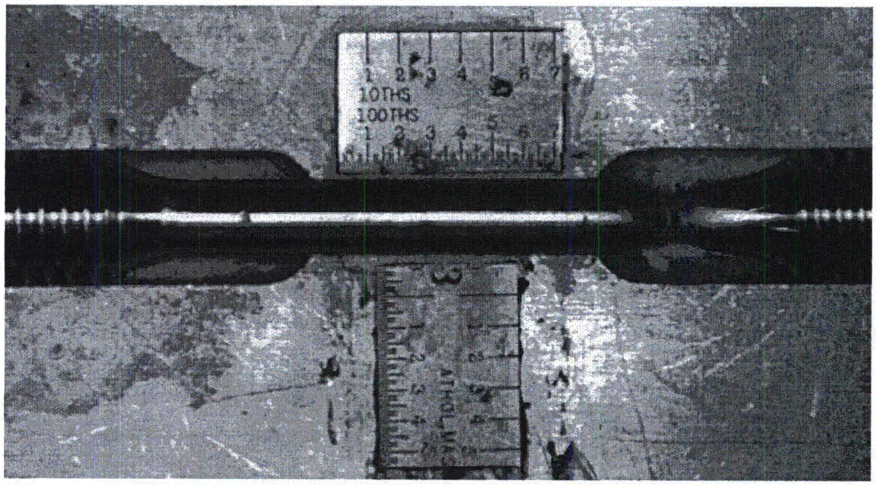


3K7 – Tested at 150°F

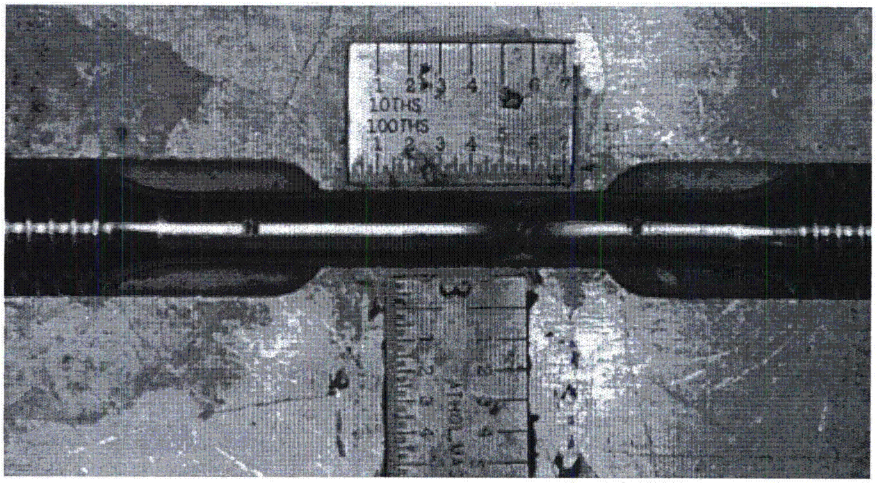


3L5 – Tested at 550°F

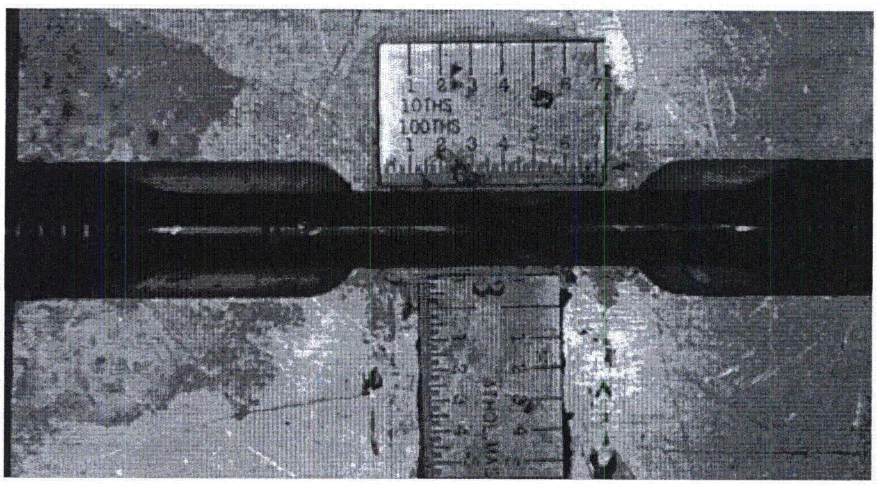
Figure 5-24 Fractured Tensile Specimens from the St. Lucie Unit 2 Reactor Vessel Surveillance Program Weld Metal (Heat # 83637)



4K5 – Tested at 72°F

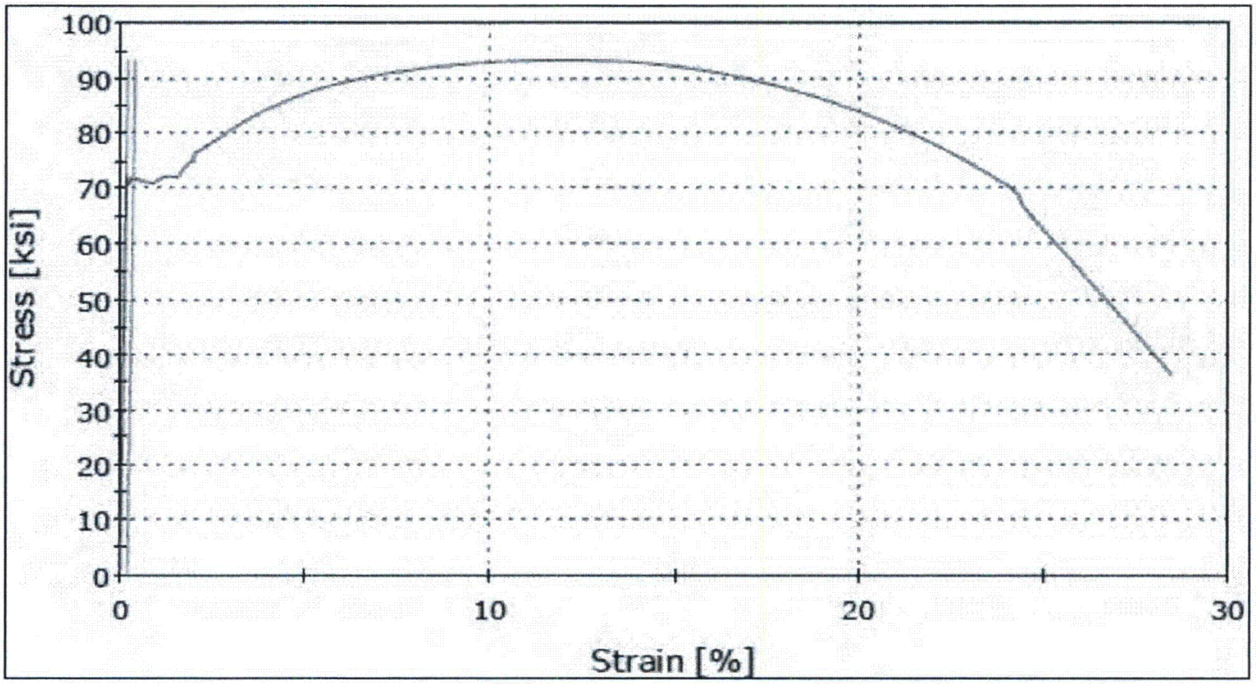


4J5 – Tested at 240°F

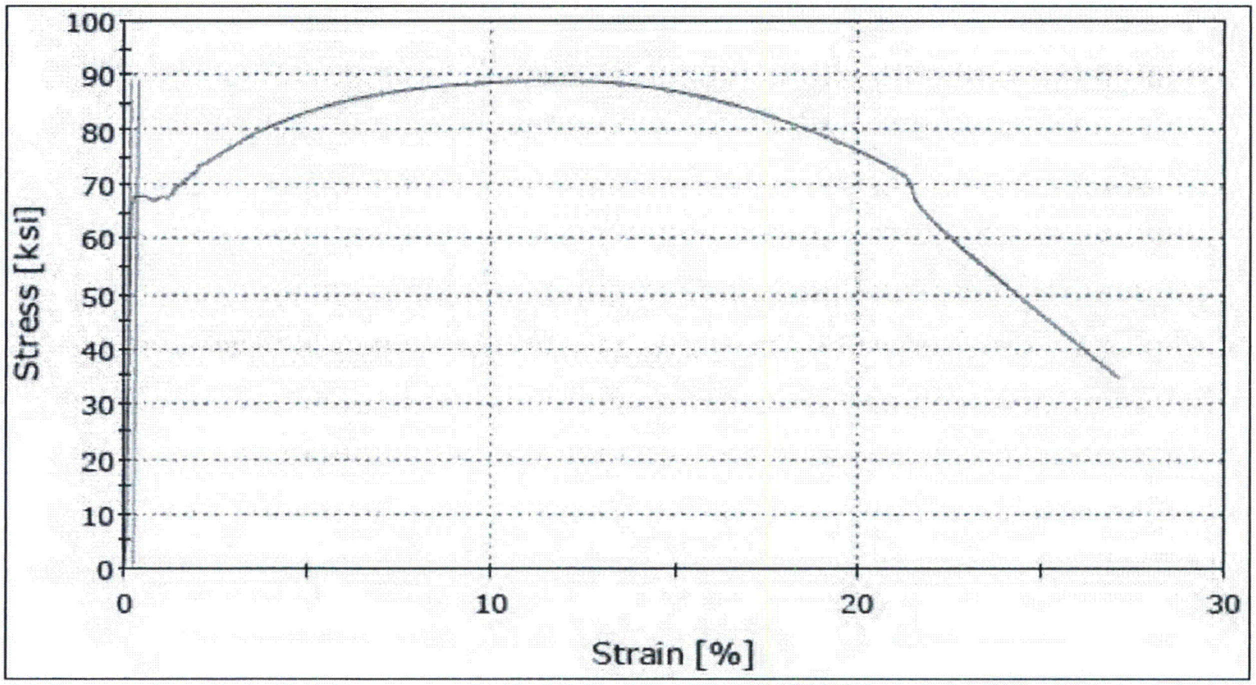


4JK – Tested at 550°F

Figure 5-25 Fractured Tensile Specimens from the St. Lucie Unit 2 Reactor Vessel Heat Affected Zone Material

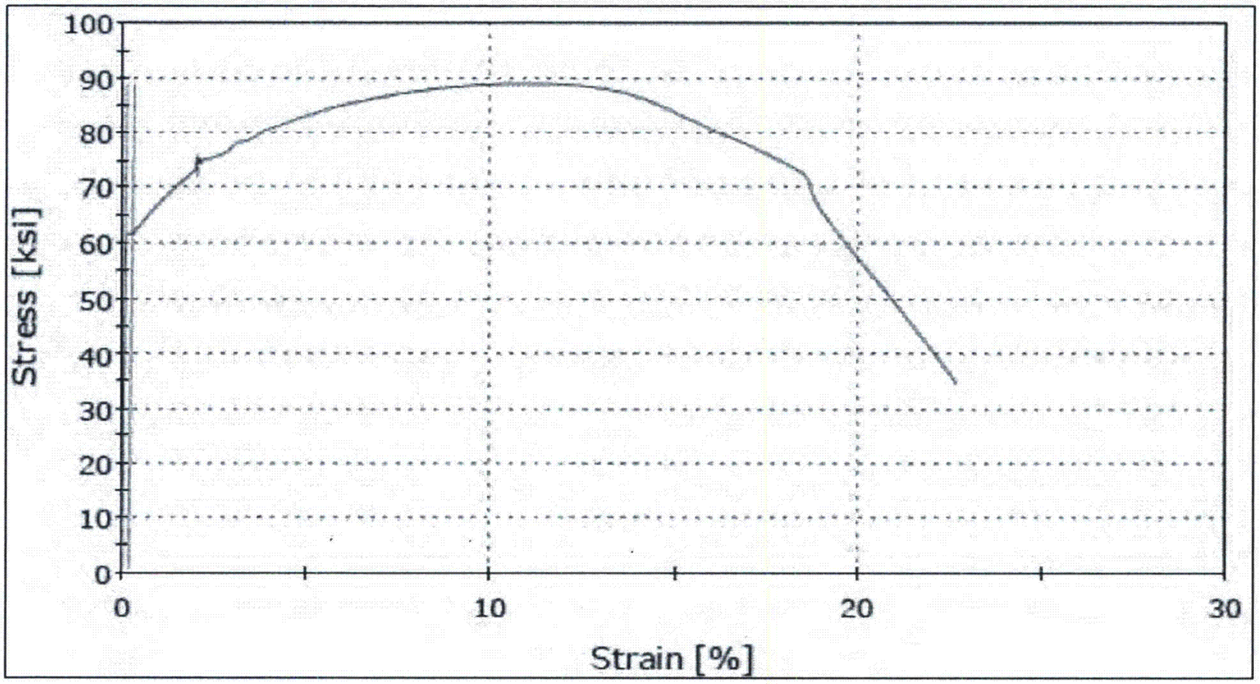


Tensile Specimen 2L5 Tested at 150°F



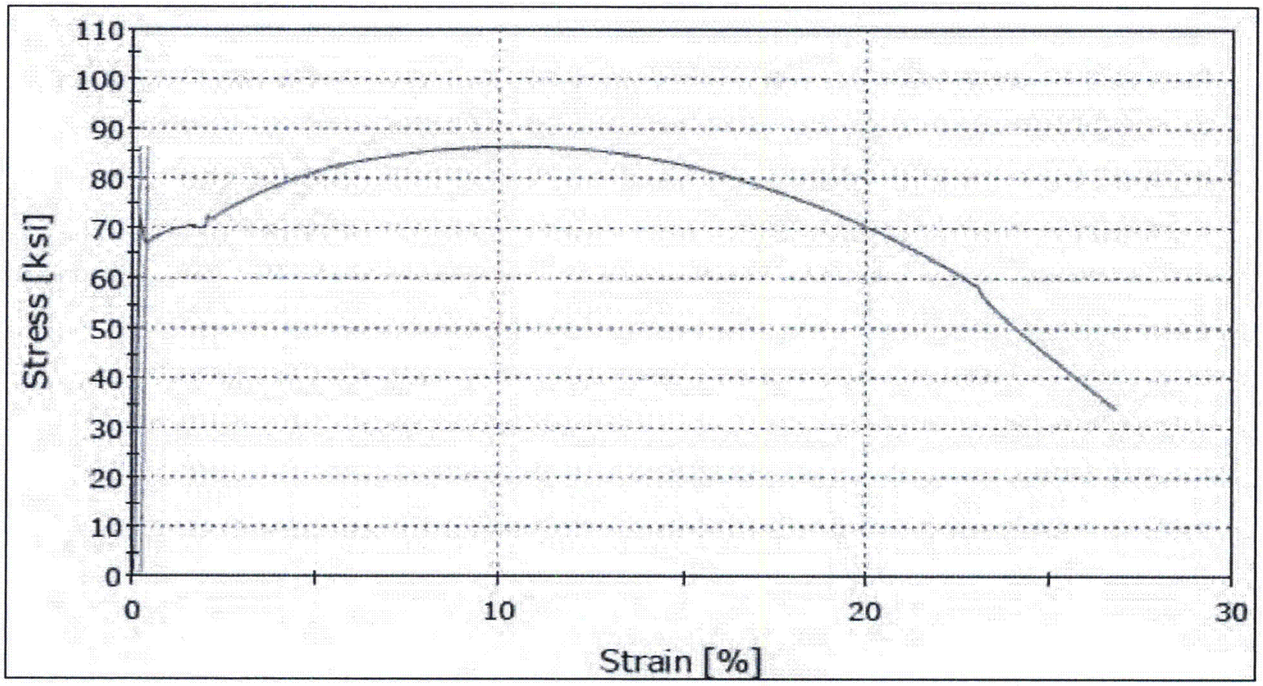
Tensile Specimen 2KU Tested at 280°F

Figure 5-26 Engineering Stress-Strain Curves for St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Tensile Specimens 2L5 and 2KU (Transverse Orientation)

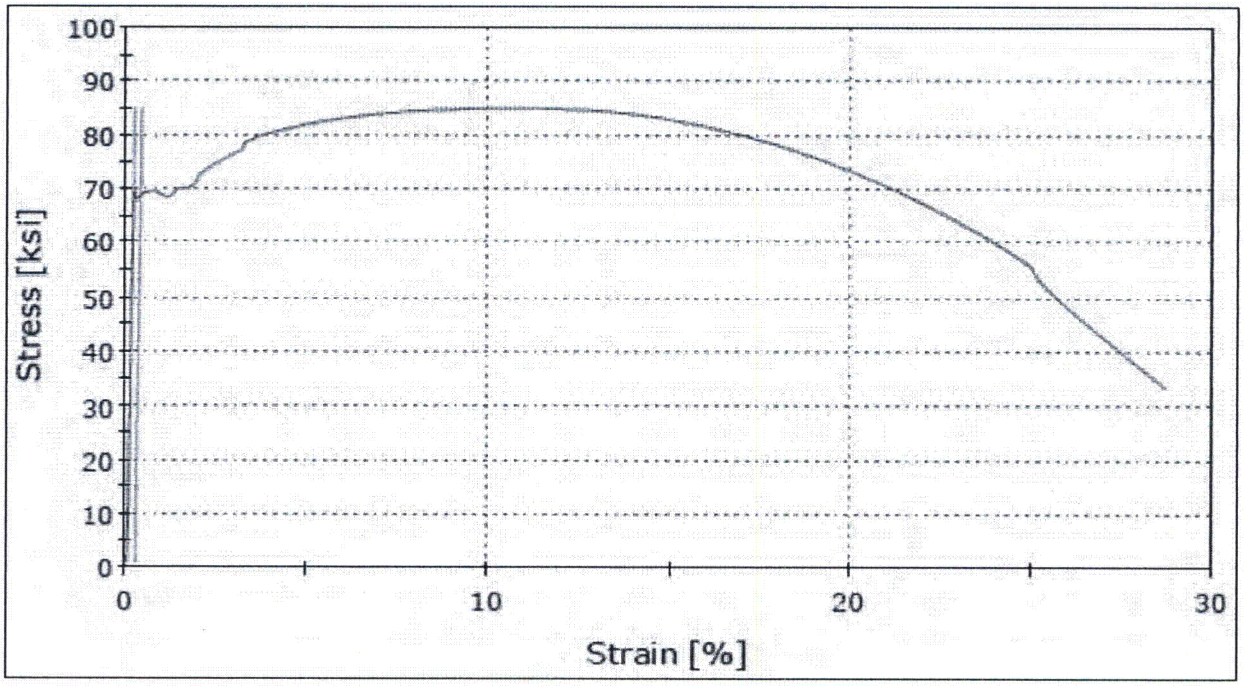


Tensile Specimen 2JM Tested at 550°F

Figure 5-27 Engineering Stress-Strain Curve for St. Lucie Unit 2 Intermediate Shell Plate M-605-1 Tensile Specimen 2JM (Transverse Orientation)

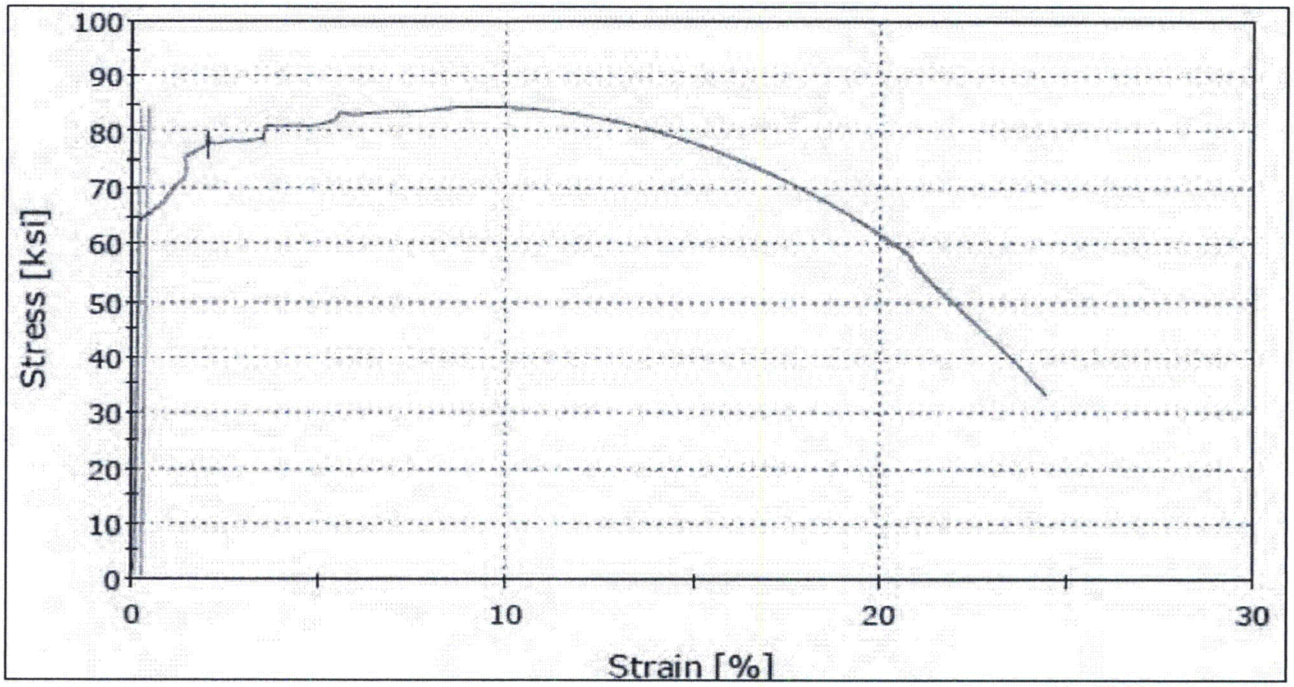


Tensile Specimen 3J4 Tested at 72°F



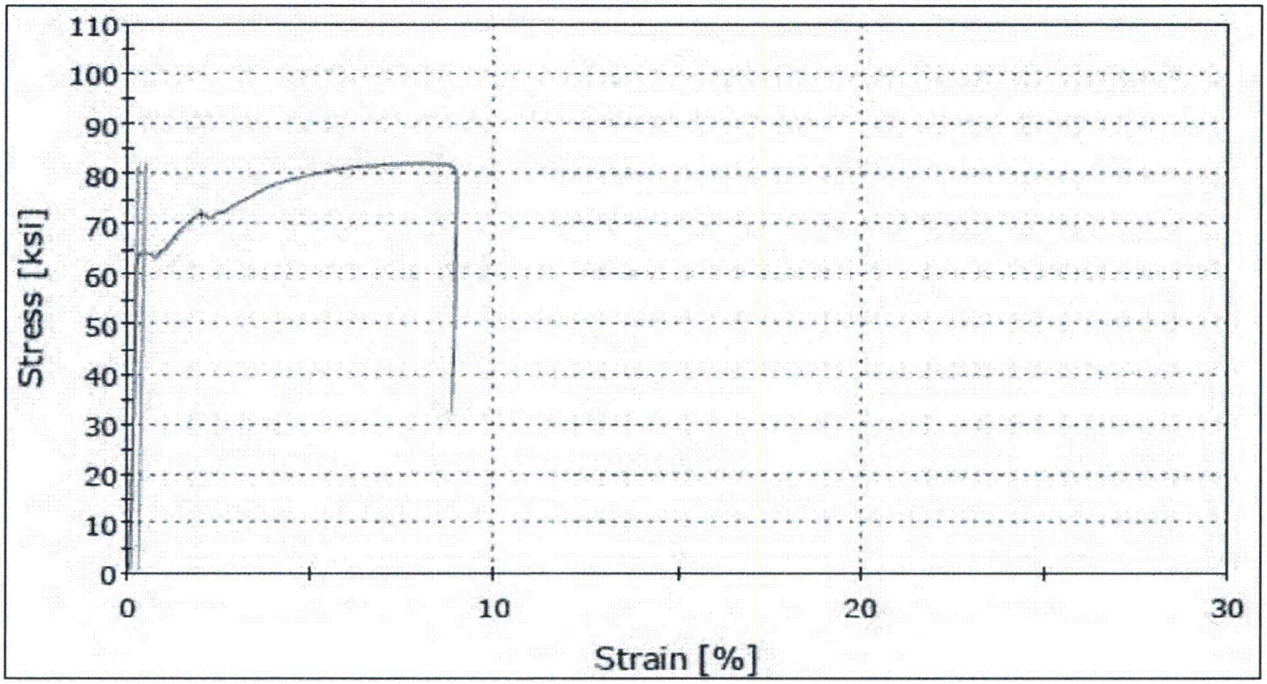
Tensile Specimen 3K7 Tested at 150°F

Figure 5-28 Engineering Stress-Strain Curves for St. Lucie Unit 2 Surveillance Program Weld Metal (Heat # 83637) Tensile Specimens 3J4 and 3K7



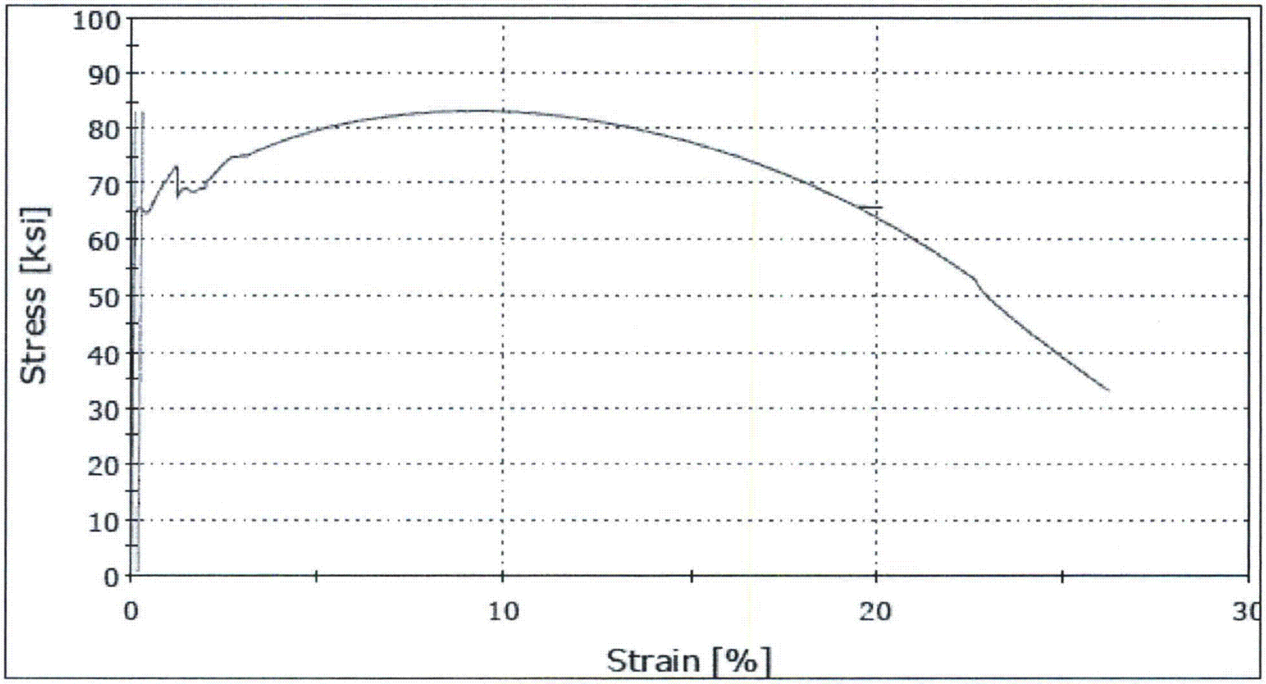
Tensile Specimen 3L5 Tested at 550°F

Figure 5-29 Engineering Stress-Strain Curve for St. Lucie Unit 2 Surveillance Program Weld Metal (Heat # 83637) Tensile Specimen 3L5



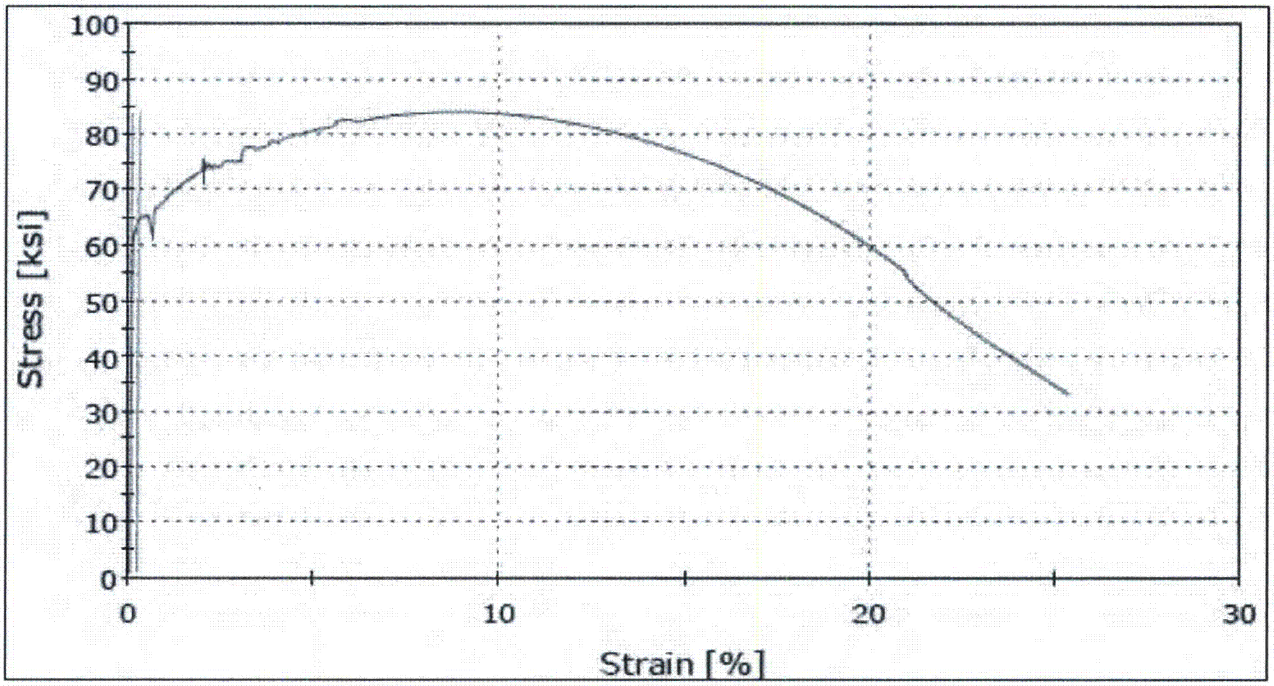
Tensile Specimen 4K5 Tested at 72°F

(Note: Specimen broke outside of gage section, so strain is not an accurate measurement)



Tensile Specimen 4J5 Tested at 240°F

Figure 5-30 Engineering Stress-Strain Curves for St. Lucie Unit 2 Heat Affected Zone Material Tensile Specimens 4K5 and 4J5



Tensile Specimen 4JK Tested at 550°F

Figure 5-31 Engineering Stress-Strain Curve for St. Lucie Unit 2 Heat Affected Zone Material Tensile Specimen 4JK

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

This section describes a discrete ordinates S_n transport analysis performed for the St. Lucie Unit 2 reactor to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this analysis, fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV) and iron atom displacements (dpa) were established on a plant- and fuel-cycle-specific basis. An evaluation of the most recent dosimetry sensor set from Capsule 97°, withdrawn at the end of the twentieth plant operating cycle, is provided. In addition, the sensor sets from the previously withdrawn capsules (83° and 263°) are presented. Comparisons of the results from these dosimetry evaluations with the analytical predictions served to validate the plant-specific neutron transport calculations. These validated calculations subsequently formed the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 60 effective full-power years (EFPY) at 3020 MWt.

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured material property changes to the neutron exposure of the material has traditionally been accepted for the development of damage trend curves as well as for the implementation of trend curve data to assess the condition of the vessel. However, in recent years, 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 and improved accuracy in the evaluation of damage gradients through the reactor vessel wall.

Because of this potential shift away from a threshold fluence toward an energy-dependent damage function for data correlation, ASTM Standard Practice E853-13, "Analysis and Interpretation of Light-Water Reactor Surveillance Results," [Reference 17] recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a database for future reference. The energy-dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693-94, "Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom" [Reference 18]. The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Reference 1].

All of the calculations and dosimetry evaluations described in this section and in Appendix A were based on nuclear cross-section data derived from ENDF/B-VI and using the latest available calculational tools. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Reference 19]. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004 [Reference 20]. As an improvement, instead of the fluence rate synthesis technique, three-dimensional transport calculations were performed.

6.2 DISCRETE ORDINATES ANALYSIS

The arrangement of the surveillance capsules in the St. Lucie Unit 2 reactor vessel is shown in Figure 4-1. Six irradiation capsules attached to the pressure vessel inside wall are included in the reactor design that constitutes the reactor vessel surveillance program. The capsules are located at azimuthal angles of 83°, 97°, 104°, 263°, 277°, and 284° as shown in Figure 4-1. These full-core positions correspond to the following octant symmetric locations represented in Figure 6-1: 7° from the core cardinal axes (for the 83°, 97°, 263° and 277° capsules) and 14° from the core cardinal axes (for the 104° and 284° capsules). The stainless steel specimen containers are 1.5-inch by 0.75-inch and are approximately 98 inches in height. The containers are positioned axially such that the test specimens are centered on the core midplane, thus spanning the approximate central eight feet of the 11.4-foot-high reactor core.

From a neutronic standpoint, the surveillance capsules and capsule holders are significant. The presence of these materials has a significant effect on both the spatial distribution of neutron fluence rate and the neutron spectrum in the vicinity of the capsules. However, the capsules are far enough apart that they do not interfere with one another. In order to determine the neutron environment at the test specimen location, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the St. Lucie Unit 2 reactor vessel and surveillance capsules, a series of fuel-cycle-specific forward transport calculations were carried out using a three-dimensional geometrical reactor model. For the St. Lucie Unit 2 transport calculations, the r,θ,z models depicted (given as r,θ section view) in Figures 6-1 and 6-2 were utilized since, with the exception of the capsules, the reactor is octant symmetric. The r,z section view depicted in Figure 6-3 shows the model having an axial span from an elevation one foot below the bottom of the active fuel and one foot above the top of the active fuel. These r,θ,z models include the core, the reactor internals, the surveillance capsules, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. These models formed the basis for the calculated results and enabled making comparisons to the surveillance capsule dosimetry evaluations. In developing these analytical models, nominal design dimensions were employed for the various structural components. For the reactor pressure vessel, the vessel averaged inner radius was used. Likewise, water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. The coolant densities were treated on a fuel-cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, et cetera. The geometric mesh description of the r,θ,z reactor models consisted of 151 radial by 119 azimuthal by 135 axial intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration fluence rate convergence criterion utilized in the r,θ,z calculations was set at a value of 0.001.

The core power distributions used in the plant-specific transport analysis were provided by Florida Power & Light Company for each of the first 21 fuel cycles at St. Lucie Unit 2. Specifically, the data utilized included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel-cycle-averaged neutron fluence rate, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core

source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations supporting this analysis were carried out using the RAPTOR-M3G discrete ordinates code [Reference 21] and the BUGLE-96 cross-section library [Reference 22]. The BUGLE-96 library provides a coupled 47-neutron, 20-gamma-group cross-section data set produced specifically for light-water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion, and angular discretization was modeled with an S_8 order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

Selected results from the neutron transport analyses are provided in Tables 6-1 through 6-4. In Table 6-1, the calculated exposure rates and integrated exposures, expressed in terms of both neutron fluence ($E > 1.0$ MeV) and dpa, are given at the radial and azimuthal center of the octant symmetric surveillance capsule positions, i.e., for the 7° capsule and 14° capsule. These results, representative of the average axial exposure of the material specimens, establish the calculated exposure of the surveillance capsules withdrawn to date as well as projected into the future.

Similar information, in terms of both calculated fluence ($E > 1.0$ MeV) and dpa data are provided in Table 6-2, for the reactor vessel inner radius at four azimuthal locations. The vessel data given in Table 6-2 were taken at the clad/base metal interface and thus represent maximum calculated exposure levels on the vessel. From the data provided in Table 6-2, it is noted that the peak clad/base metal interface vessel fluence ($E > 1.0$ MeV) at the end of the 20^{th} fuel cycle (i.e., after 25.55 EFPY at 3020 MWt of plant operation) was 1.73E+19 n/cm^2.

These data tabulations include both plant- and fuel-cycle-specific calculated neutron exposures at the end of the 20^{th} fuel cycle, as well as future projections to 32, 36, 40, 48, 55, and 60 EFPY at 3020 MWt. The calculations account for uprates from 2560 MWt to 2700 MWt that occurred at the end of Cycle 1, and from 2700 MWt to 3020 MWt that occurred at the end of Cycle 19. The projections were based on the assumption that the core power distributions and associated plant operating characteristics from Cycle 21 were representative of future plant operation. The future projections are also based on the current reactor power level of 3020 MWt.

The calculated fast neutron exposures for the three surveillance capsules withdrawn from the St. Lucie Unit 2 reactor are provided in Table 6-3. These assigned neutron exposure levels are based on the plant- and fuel-cycle-specific neutron transport calculations performed for the St. Lucie Unit 2 reactor. From the data provided in Table 6-3, Capsule 97° received a fluence ($E > 1.0$ MeV) of 2.25E+19 n/cm^2 after exposure through the end of the 20^{th} fuel cycle (i.e., after 25.55 EFPY at 3020 MWt of plant operation).

Updated lead factors for the St. Lucie Unit 2 surveillance capsules are provided in Table 6-4. The capsule lead factor is defined as the ratio of the calculated axial average fluence ($E > 1.0$ MeV) at the geometric radial and azimuthal center of the surveillance capsule to the corresponding maximum calculated fluence at the pressure vessel clad/base metal interface. In Table 6-4, the lead factors for capsules that have been withdrawn from the reactor (83°, 263°, and 97°) were based on the calculated fluence values for the

irradiation period corresponding to the time of withdrawal for the individual capsules. For the capsules remaining in the reactor (104°, 277°, and 284°), the lead factor corresponds to the calculated fluence values at the end of Cycle 20, the last completed fuel cycle for St. Lucie Unit 2.

6.3 NEUTRON DOSIMETRY

The validity of the calculated neutron exposures previously reported in Section 6.2 is demonstrated by a direct comparison against the measured sensor reaction rates and via a least-squares evaluation performed for each of the capsule dosimetry sets. However, since the neutron dosimetry measurement data merely serve to validate the calculated results, only the direct comparison of measured-to-calculated results for the most recent surveillance capsule removed from service is provided in this section of the report. For completeness, the assessment of all measured dosimetry removed to date, based on both direct and least-squares evaluation comparisons, is documented in Appendix A.

The direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule 97°, which was withdrawn from St. Lucie Unit 2 at the end of the twentieth fuel cycle, is summarized below.

Reaction	Reaction Rate (rps/atom)		M/C
	Measured (M)	Calculated (C)	
Ti-46(n,p)Sc-46	7.12E-16	6.43E-16	1.11
Fe-54(n,p)Mn-54	3.77E-15	3.65E-15	1.03
Ni-58(n,p)Co-58	4.98E-15	4.77E-15	1.05
U-238(n,f)Cs-137	9.79E-15	1.25E-14	0.78
Average			0.99
% standard deviation			14.7

The measured-to-calculated (M/C) reaction rate ratios for the Capsule 97° threshold reactions range from 0.78 to 1.11, and the average M/C ratio is 0.99 ± 14.7% (1σ). This direct comparison falls within the ± 20% criterion specified in Regulatory Guide 1.190. These comparisons validate the current analytical results described in Section 6.2; therefore, the calculations are deemed applicable for St. Lucie Unit 2.

6.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the St. Lucie Unit 2 surveillance capsule and reactor pressure vessel is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

1. Comparison of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL).

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment.

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the neutron exposure assessments.

4. Comparisons of the plant-specific calculations with all available dosimetry results from the St. Lucie Unit 2 surveillance program.

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross sections. This phase, however, did not test the accuracy of commercial core neutron source calculations nor did it address uncertainties in operational or geometric variables that impact power reactor calculations. The second phase of the qualification (H. B. Robinson comparisons) addressed uncertainties in these additional areas that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant-specific input parameters. The overall calculational uncertainty applicable to the St. Lucie Unit 2 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with St. Lucie Unit 2 measurements) was used solely to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used in any way to modify the calculated surveillance capsule and pressure vessel neutron exposures previously described in Section 6.2. As such, the validation of the St. Lucie Unit 2 analytical model based on the measured plant dosimetry is completely described in Appendix A.

The following summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Reference 20.

Description	**Capsule and Vessel IR**
PCA Comparisons	3%
H. B. Robinson Comparisons	3%
Analytical Sensitivity Studies	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%
Net Calculational Uncertainty	13%

The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random, and no systematic bias was applied to the analytical results.

The plant-specific measurement comparisons described in Appendix A support these uncertainty assessments for St. Lucie Unit 2.

Table 6-1 Calculated Neutron Exposure Rates and Integrated Exposures at the Surveillance Capsule Center

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)		Fluence (n/cm^2)	
			7-Degree	14-Degree	7-Degree	14-Degree
1	1.11	1.11	3.99E+10	2.86E+10	1.40E+18	1.00E+18
2	1.12	2.23	3.94E+10	2.72E+10	2.79E+18	1.96E+18
3	1.22	3.45	3.55E+10	2.48E+10	4.16E+18	2.92E+18
4	1.16	4.61	2.62E+10	2.16E+10	5.12E+18	3.71E+18
5	1.3	5.91	2.58E+10	2.16E+10	6.17E+18	4.60E+18
6	1.35	7.26	2.54E+10	1.85E+10	7.25E+18	5.39E+18
7	1.21	8.47	2.74E+10	1.96E+10	8.30E+18	6.14E+18
8	1.38	9.85	1.69E+10	1.40E+10	9.04E+18	6.75E+18
9	1.22	11.07	2.57E+10	2.16E+10	1.00E+19	7.58E+18
10	1.44	12.51	2.73E+10	2.21E+10	1.13E+19	8.59E+18
11	1.32	13.83	2.45E+10	1.95E+10	1.23E+19	9.39E+18
12	1.51	15.34	2.39E+10	1.75E+10	1.34E+19	1.02E+19
13	1.29	16.63	2.65E+10	1.96E+10	1.45E+19	1.10E+19
14	1.43	18.06	2.33E+10	1.75E+10	1.56E+19	1.18E+19
15	1.15	19.21	2.72E+10	1.98E+10	1.65E+19	1.25E+19
16	1.25	20.46	2.80E+10	2.06E+10	1.77E+19	1.33E+19
17	1.25	21.71	2.65E+10	1.97E+10	1.87E+19	1.41E+19
18	1.42	23.13	2.59E+10	1.98E+10	1.99E+19	1.50E+19
19	1.19	24.32	3.24E+10	2.44E+10	2.11E+19	1.59E+19
20	1.23	25.55	3.72E+10	2.72E+10	2.25E+19	1.70E+19
21	1.38	26.93	3.56E+10	2.63E+10	2.41E+19	1.81E+19
Future(1)	5.07	32.00	3.92E+10	2.90E+10	3.03E+19	2.28E+19
Future	4.00	36.00	3.92E+10	2.90E+10	3.53E+19	2.64E+19
Future	4.00	40.00	3.92E+10	2.90E+10	4.02E+19	3.01E+19
Future	8.00	48.00	3.92E+10	2.90E+10	5.01E+19	3.74E+19
Future	7.00	55.00	3.92E+10	2.90E+10	5.88E+19	4.38E+19
Future	5.00	60.00	3.92E+10	2.90E+10	6.49E+19	4.84E+19

Notes:

1. Fluence rate (and fluence) projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-1 (Continued) Calculated Neutron Exposure Rates and Integrated Exposures at the Surveillance Capsule Center

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	dpa/s		dpa	
			7-Degree	14-Degree	7-Degree	14-Degree
1	1.11	1.11	5.81E-11	4.18E-11	2.04E-03	1.47E-03
2	1.12	2.23	5.74E-11	3.99E-11	4.07E-03	2.87E-03
3	1.22	3.45	5.18E-11	3.64E-11	6.06E-03	4.28E-03
4	1.16	4.61	3.82E-11	3.17E-11	7.46E-03	5.44E-03
5	1.3	5.91	3.77E-11	3.17E-11	9.00E-03	6.73E-03
6	1.35	7.26	3.70E-11	2.72E-11	1.06E-02	7.89E-03
7	1.21	8.47	4.00E-11	2.87E-11	1.21E-02	8.99E-03
8	1.38	9.85	2.47E-11	2.06E-11	1.32E-02	9.89E-03
9	1.22	11.07	3.76E-11	3.16E-11	1.46E-02	1.11E-02
10	1.44	12.51	3.99E-11	3.23E-11	1.64E-02	1.26E-02
11	1.32	13.83	3.57E-11	2.85E-11	1.79E-02	1.38E-02
12	1.51	15.34	3.49E-11	2.56E-11	1.96E-02	1.50E-02
13	1.29	16.63	3.87E-11	2.88E-11	2.12E-02	1.62E-02
14	1.43	18.06	3.40E-11	2.57E-11	2.27E-02	1.73E-02
15	1.15	19.21	3.97E-11	2.90E-11	2.41E-02	1.84E-02
16	1.25	20.46	4.09E-11	3.01E-11	2.58E-02	1.96E-02
17	1.25	21.71	3.87E-11	2.89E-11	2.73E-02	2.07E-02
18	1.42	23.13	3.78E-11	2.90E-11	2.90E-02	2.20E-02
19	1.19	24.32	4.73E-11	3.58E-11	3.08E-02	2.33E-02
20	1.23	25.55	5.42E-11	3.99E-11	3.29E-02	2.49E-02
21	1.38	26.93	5.19E-11	3.86E-11	3.51E-02	2.66E-02
Future(1)	5.07	32.00	5.71E-11	4.24E-11	4.43E-02	3.34E-02
Future	4.00	36.00	5.71E-11	4.24E-11	5.15E-02	3.87E-02
Future	4.00	40.00	5.71E-11	4.24E-11	5.87E-02	4.41E-02
Future	8.00	48.00	5.71E-11	4.24E-11	7.31E-02	5.48E-02
Future	7.00	55.00	5.71E-11	4.24E-11	8.57E-02	6.42E-02
Future	5.00	60.00	5.71E-11	4.24E-11	9.47E-02	7.09E-02

Notes:

1. dpa/s (and dpa) projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-2 Calculated Azimuthal Variation of Maximum Exposure Rates and Integrated Exposures at the Reactor Vessel Clad/Base Metal Interface

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)				
			0-Degree	15-Degree	30-Degree	45-Degree	Maximum
1	1.11	1.11	3.08E+10	2.00E+10	1.62E+10	1.26E+10	3.08E+10
2	1.12	2.23	3.22E+10	1.94E+10	1.61E+10	1.27E+10	3.22E+10
3	1.22	3.45	2.89E+10	1.76E+10	1.39E+10	1.06E+10	2.89E+10
4	1.16	4.61	1.97E+10	1.59E+10	1.48E+10	1.17E+10	1.97E+10
5	1.3	5.91	1.92E+10	1.58E+10	1.47E+10	1.13E+10	1.92E+10
6	1.35	7.26	2.07E+10	1.35E+10	1.16E+10	9.79E+09	2.07E+10
7	1.21	8.47	2.22E+10	1.41E+10	1.15E+10	1.01E+10	2.22E+10
8	1.38	9.85	1.36E+10	1.09E+10	1.24E+10	1.01E+10	1.36E+10
9	1.22	11.07	1.84E+10	1.55E+10	1.31E+10	9.87E+09	1.89E+10
10	1.44	12.51	1.99E+10	1.58E+10	1.31E+10	1.06E+10	2.01E+10
11	1.32	13.83	1.85E+10	1.40E+10	1.36E+10	1.15E+10	1.85E+10
12	1.51	15.34	1.88E+10	1.24E+10	1.18E+10	1.07E+10	1.88E+10
13	1.29	16.63	2.06E+10	1.39E+10	1.24E+10	1.13E+10	2.06E+10
14	1.43	18.06	1.80E+10	1.25E+10	1.17E+10	1.06E+10	1.80E+10
15	1.15	19.21	2.10E+10	1.39E+10	1.14E+10	9.72E+09	2.10E+10
16	1.25	20.46	2.14E+10	1.44E+10	1.17E+10	9.87E+09	2.14E+10
17	1.25	21.71	2.01E+10	1.38E+10	1.12E+10	9.65E+09	2.01E+10
18	1.42	23.13	1.93E+10	1.40E+10	1.15E+10	1.02E+10	1.93E+10
19	1.19	24.32	2.46E+10	1.73E+10	1.57E+10	1.29E+10	2.46E+10
20	1.23	25.55	2.88E+10	1.93E+10	1.76E+10	1.53E+10	2.88E+10
21	1.38	26.93	2.75E+10	1.87E+10	1.74E+10	1.49E+10	2.75E+10
Future(1)	5.07	32.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10
Future	4.00	36.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10
Future	4.00	40.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10
Future	8.00	48.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10
Future	7.00	55.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10
Future	5.00	60.00	3.03E+10	2.06E+10	1.91E+10	1.64E+10	3.03E+10

Notes:

1. Fluence rate projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-2 (Continued) Calculated Azimuthal Variation of Maximum Exposure Rates and Integrated Exposures at the Reactor Vessel Clad/Base Metal Interface

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)				
			0-Degree	15-Degree	30-Degree	45-Degree	Maximum
1	1.11	1.11	1.08E+18	7.01E+17	5.67E+17	4.42E+17	1.08E+18
2	1.12	2.23	2.21E+18	1.38E+18	1.13E+18	8.86E+17	2.21E+18
3	1.22	3.45	3.31E+18	2.05E+18	1.67E+18	1.29E+18	3.31E+18
4	1.16	4.61	4.03E+18	2.64E+18	2.21E+18	1.72E+18	4.03E+18
5	1.3	5.91	4.82E+18	3.28E+18	2.81E+18	2.19E+18	4.82E+18
6	1.35	7.26	5.68E+18	3.85E+18	3.30E+18	2.59E+18	5.68E+18
7	1.21	8.47	6.52E+18	4.38E+18	3.73E+18	2.97E+18	6.52E+18
8	1.38	9.85	7.11E+18	4.85E+18	4.26E+18	3.41E+18	7.11E+18
9	1.22	11.07	7.78E+18	5.42E+18	4.75E+18	3.77E+18	7.78E+18
10	1.44	12.51	8.69E+18	6.13E+18	5.34E+18	4.25E+18	8.69E+18
11	1.32	13.83	9.46E+18	6.72E+18	5.91E+18	4.73E+18	9.46E+18
12	1.51	15.34	1.04E+19	7.31E+18	6.47E+18	5.24E+18	1.04E+19
13	1.29	16.63	1.12E+19	7.87E+18	6.97E+18	5.70E+18	1.12E+19
14	1.43	18.06	1.20E+19	8.44E+18	7.50E+18	6.18E+18	1.20E+19
15	1.15	19.21	1.28E+19	8.94E+18	7.91E+18	6.53E+18	1.28E+19
16	1.25	20.46	1.36E+19	9.51E+18	8.37E+18	6.92E+18	1.36E+19
17	1.25	21.71	1.44E+19	1.01E+19	8.81E+18	7.30E+18	1.44E+19
18	1.42	23.13	1.53E+19	1.07E+19	9.33E+18	7.76E+18	1.53E+19
19	1.19	24.32	1.62E+19	1.13E+19	9.92E+18	8.24E+18	1.62E+19
20	1.23	25.55	1.73E+19	1.21E+19	1.06E+19	8.84E+18	1.73E+19
21	1.38	26.93	1.85E+19	1.29E+19	1.14E+19	9.49E+18	1.85E+19
Future(1)	5.07	32.00	2.33E+19	1.62E+19	1.44E+19	1.21E+19	2.33E+19
Future	4.00	36.00	2.72E+19	1.88E+19	1.68E+19	1.42E+19	2.72E+19
Future	4.00	40.00	3.10E+19	2.14E+19	1.93E+19	1.63E+19	3.10E+19
Future	8.00	48.00	3.86E+19	2.66E+19	2.41E+19	2.04E+19	3.86E+19
Future	7.00	55.00	4.53E+19	3.11E+19	2.83E+19	2.41E+19	4.53E+19
Future	5.00	60.00	5.01E+19	3.44E+19	3.13E+19	2.66E+19	5.01E+19

Notes:

1. Fluence rate (and fluence) projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-2 (Continued) Calculated Azimuthal Variation of Maximum Exposure Rates and Integrated Exposures at the Reactor Vessel Clad/Base Metal Interface

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	dpa/s				
			0-Degree	15-Degree	30-Degree	45-Degree	Maximum
1	1.11	1.11	4.68E-11	3.06E-11	2.47E-11	1.94E-11	4.68E-11
2	1.12	2.23	4.89E-11	2.97E-11	2.46E-11	1.95E-11	4.89E-11
3	1.22	3.45	4.39E-11	2.70E-11	2.13E-11	1.63E-11	4.39E-11
4	1.16	4.61	3.01E-11	2.44E-11	2.27E-11	1.81E-11	3.01E-11
5	1.3	5.91	2.93E-11	2.43E-11	2.25E-11	1.74E-11	2.93E-11
6	1.35	7.26	3.15E-11	2.08E-11	1.78E-11	1.50E-11	3.15E-11
7	1.21	8.47	3.38E-11	2.16E-11	1.76E-11	1.55E-11	3.38E-11
8	1.38	9.85	2.07E-11	1.67E-11	1.90E-11	1.56E-11	2.07E-11
9	1.22	11.07	2.81E-11	2.38E-11	2.01E-11	1.52E-11	2.88E-11
10	1.44	12.51	3.03E-11	2.41E-11	2.00E-11	1.63E-11	3.06E-11
11	1.32	13.83	2.83E-11	2.15E-11	2.08E-11	1.77E-11	2.83E-11
12	1.51	15.34	2.86E-11	1.91E-11	1.80E-11	1.65E-11	2.86E-11
13	1.29	16.63	3.14E-11	2.13E-11	1.90E-11	1.74E-11	3.14E-11
14	1.43	18.06	2.74E-11	1.92E-11	1.79E-11	1.62E-11	2.74E-11
15	1.15	19.21	3.21E-11	2.13E-11	1.75E-11	1.49E-11	3.21E-11
16	1.25	20.46	3.26E-11	2.21E-11	1.79E-11	1.52E-11	3.26E-11
17	1.25	21.71	3.06E-11	2.12E-11	1.71E-11	1.48E-11	3.06E-11
18	1.42	23.13	2.95E-11	2.14E-11	1.76E-11	1.57E-11	2.95E-11
19	1.19	24.32	3.74E-11	2.65E-11	2.39E-11	1.98E-11	3.74E-11
20	1.23	25.55	4.39E-11	2.95E-11	2.70E-11	2.35E-11	4.39E-11
21	1.38	26.93	4.19E-11	2.87E-11	2.66E-11	2.29E-11	4.19E-11
Future(1)	5.07	32.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11
Future	4.00	36.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11
Future	4.00	40.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11
Future	8.00	48.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11
Future	7.00	55.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11
Future	5.00	60.00	4.61E-11	3.15E-11	2.92E-11	2.52E-11	4.61E-11

Notes:

1. dpa/s projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-2 (Continued) Calculated Azimuthal Variation of Maximum Exposure Rates and Integrated Exposures at the Reactor Vessel Clad/Base Metal Interface

Cycle ID	Cycle Length (EFPY)	Total Time (EFPY)	dpa				
			0-Degree	15-Degree	30-Degree	45-Degree	Maximum
1	1.11	1.11	1.64E-03	1.07E-03	8.66E-04	6.79E-04	1.64E-03
2	1.12	2.23	3.35E-03	2.11E-03	1.73E-03	1.36E-03	3.35E-03
3	1.22	3.45	5.04E-03	3.15E-03	2.55E-03	1.99E-03	5.04E-03
4	1.16	4.61	6.14E-03	4.04E-03	3.38E-03	2.65E-03	6.14E-03
5	1.3	5.91	7.34E-03	5.04E-03	4.30E-03	3.36E-03	7.34E-03
6	1.35	7.26	8.65E-03	5.90E-03	5.04E-03	3.99E-03	8.65E-03
7	1.21	8.47	9.92E-03	6.71E-03	5.70E-03	4.57E-03	9.92E-03
8	1.38	9.85	1.08E-02	7.43E-03	6.52E-03	5.23E-03	1.08E-02
9	1.22	11.07	1.18E-02	8.31E-03	7.25E-03	5.79E-03	1.18E-02
10	1.44	12.51	1.32E-02	9.41E-03	8.16E-03	6.53E-03	1.32E-02
11	1.32	13.83	1.44E-02	1.03E-02	9.03E-03	7.27E-03	1.44E-02
12	1.51	15.34	1.58E-02	1.12E-02	9.89E-03	8.05E-03	1.58E-02
13	1.29	16.63	1.70E-02	1.21E-02	1.07E-02	8.76E-03	1.70E-02
14	1.43	18.06	1.83E-02	1.29E-02	1.15E-02	9.50E-03	1.83E-02
15	1.15	19.21	1.94E-02	1.37E-02	1.21E-02	1.00E-02	1.94E-02
16	1.25	20.46	2.07E-02	1.46E-02	1.28E-02	1.06E-02	2.07E-02
17	1.25	21.71	2.19E-02	1.54E-02	1.35E-02	1.12E-02	2.19E-02
18	1.42	23.13	2.32E-02	1.64E-02	1.43E-02	1.19E-02	2.32E-02
19	1.19	24.32	2.47E-02	1.74E-02	1.52E-02	1.27E-02	2.47E-02
20	1.23	25.55	2.64E-02	1.85E-02	1.62E-02	1.36E-02	2.64E-02
21	1.38	26.93	2.82E-02	1.98E-02	1.74E-02	1.46E-02	2.82E-02
Future(1)	5.07	32.00	3.56E-02	2.48E-02	2.20E-02	1.86E-02	3.56E-02
Future	4.00	36.00	4.14E-02	2.88E-02	2.57E-02	2.18E-02	4.14E-02
Future	4.00	40.00	4.72E-02	3.28E-02	2.94E-02	2.50E-02	4.72E-02
Future	8.00	48.00	5.88E-02	4.07E-02	3.68E-02	3.13E-02	5.88E-02
Future	7.00	55.00	6.90E-02	4.77E-02	4.33E-02	3.69E-02	6.90E-02
Future	5.00	60.00	7.63E-02	5.27E-02	4.79E-02	4.09E-02	7.63E-02

Notes:

1. dpa/s (and dpa) projections were increased by a factor of 10% to allow for variations in future core power distributions.

Table 6-3 Calculated Fast Neutron Exposure of Surveillance Capsules Withdrawn from St. Lucie Unit 2

Capsule	Irradiation Cycle(s)	Irradiation Time [EFPY]	Fluence (E > 1.0 MeV) [n/cm^2]	Iron Atom Displacements [dpa]
83°	1	1.11	1.40E+18	2.04E-03
263°	1-9	11.07	1.00E+19	1.46E-02
97°	1-20	25.55	2.25E+19	3.29E-02

Table 6-4 Calculated Surveillance Capsule Lead Factors

Capsule Location	Status	Lead Factor
83°	Withdrawn EOC 1	1.30
263°	Withdrawn EOC 9	1.29
97°	Withdrawn EOC 20	1.30
104°	In Reactor(1)	0.98
277°	In Reactor(1)	1.30
284°	In Reactor(1)	0.98

Notes:
1. Lead factors are based on the cumulative exposures from Cycles 1 through 20.

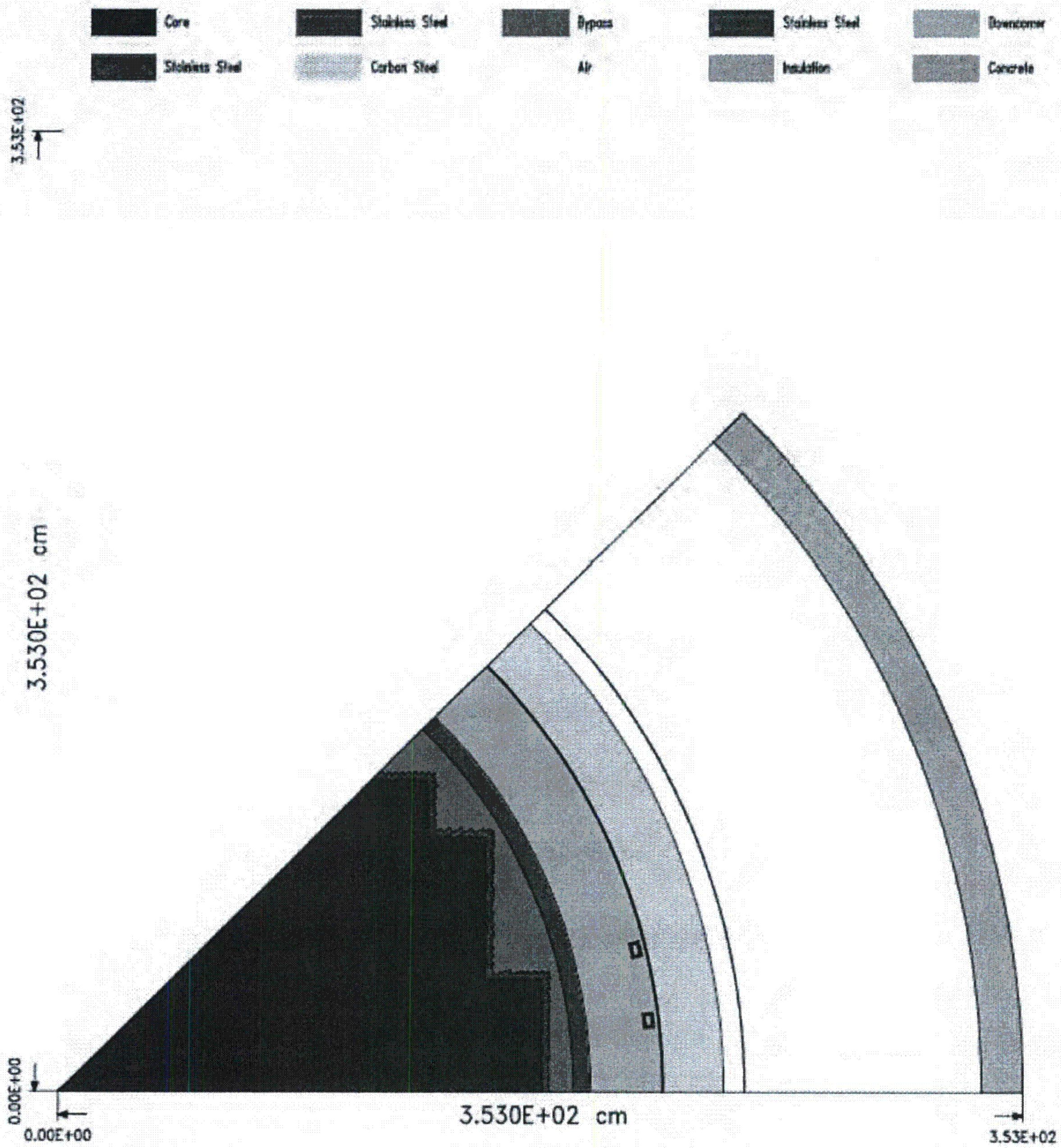


Figure 6-1 St. Lucie Unit 2 r,θ,z Reactor Geometry r,θ Plan View without Surveillance Capsules

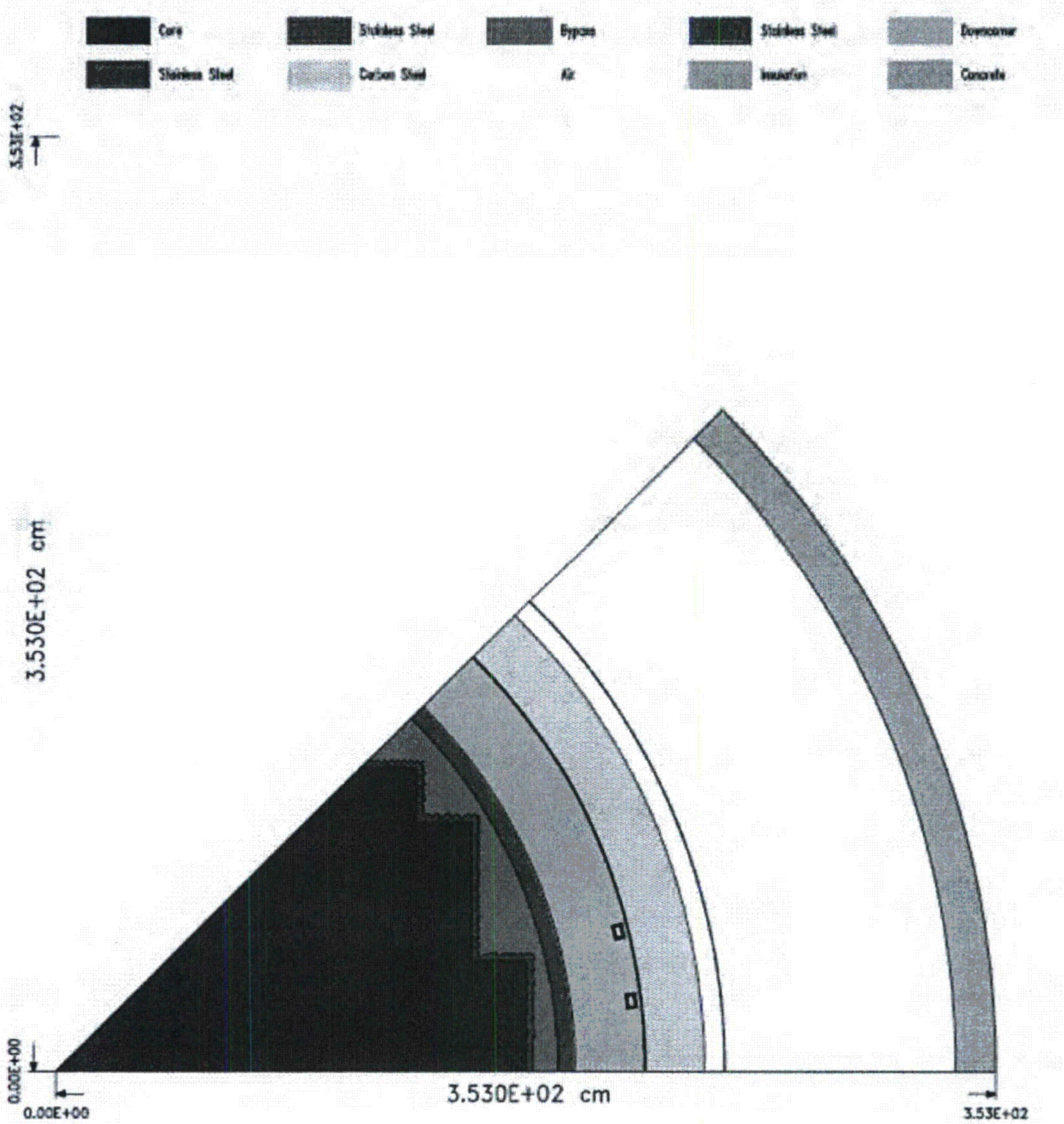


Figure 6-2 **St. Lucie Unit 2 r,θ,z Reactor Geometry r,θ Plan View with 7° and 14° Surveillance Capsules**

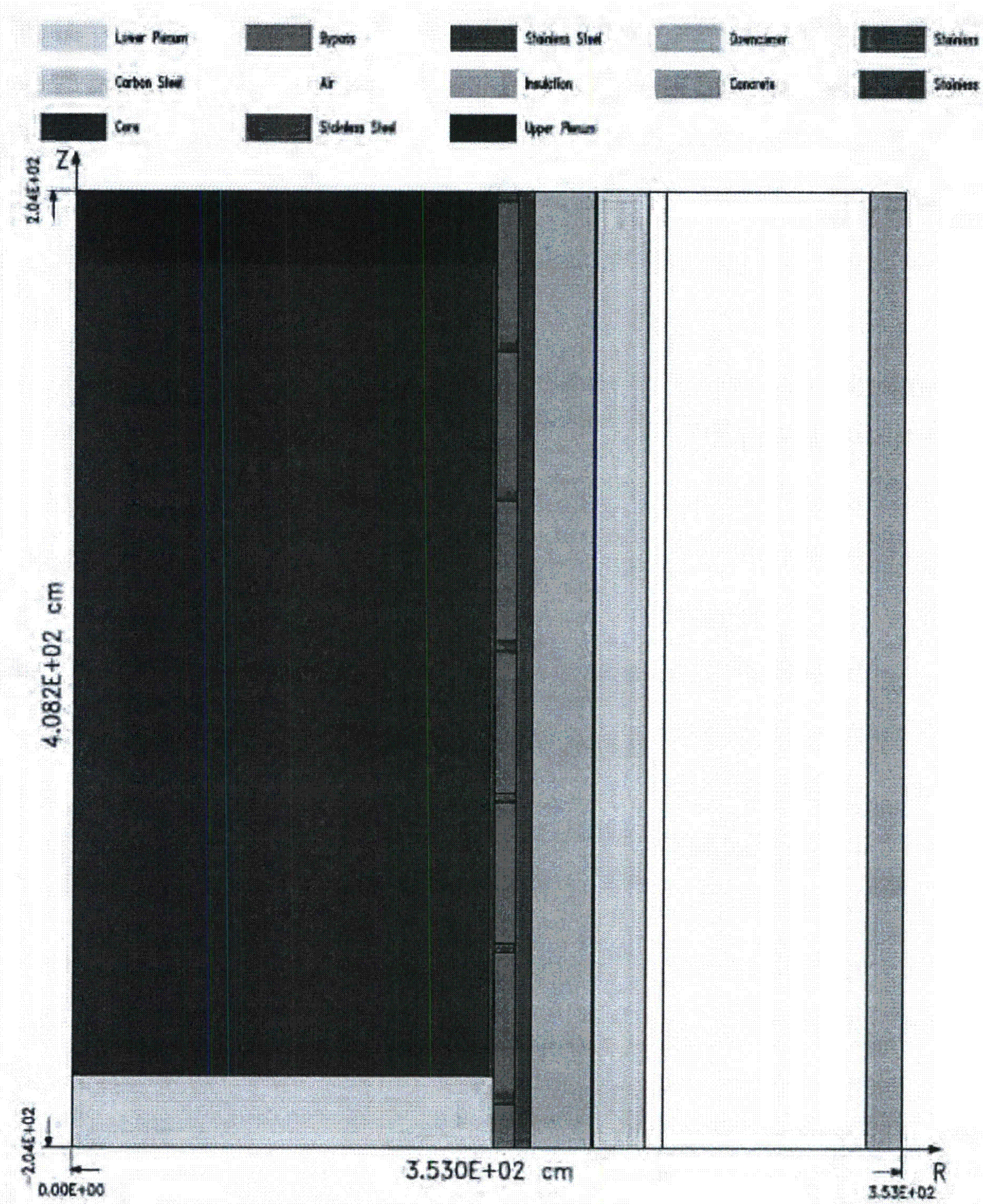


Figure 6-3 St. Lucie Unit 2 r,θ,z Reactor Geometry r,z Axial View

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule (Table 7-1) meets the requirements of ASTM E185-82 [Ref. 10]. Note that it is recommended for future capsule(s) to be removed from the St. Lucie Unit 2 reactor vessel.

Table 7-1 Surveillance Capsule Withdrawal Schedule

Capsule ID and Location	Status[(a)]	Capsule Lead Factor[(a)]	Withdrawal EFPY[(b, c)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)[(c)]
83°	Withdrawn (EOC 1)	1.30	1.11	1.40E+18
263°	Withdrawn (EOC 9)	1.29	11.07	1.00E+19
97°	Withdrawn (EOC 20)	1.30	25.55	2.25E+19
277°	In Reactor	1.30	44.1[(d)]	4.53E+19[(d)]
104°	In Reactor	0.98	(e)	(e)
284°	In Reactor	0.98	(e)	(e)

Notes:

(a) Updated in Capsule 97° dosimetry analysis; see Table 6-4.

(b) EFPY from plant startup.

(c) Updated in Capsule 97° dosimetry analysis; see Table 6-3.

(d) Capsule 277° should be withdrawn at the next vessel refueling outage after 44.1 EFPY of plant operation, which is when the fluence on the capsule would equal the projected 60-year (55 EFPY) peak vessel fluence.

(e) Capsules 104° and 284° currently have a lead factor slightly less than one. If additional metallurgical data is needed for St. Lucie Unit 2, such as in support of a second license renewal to 80 total years of operation (estimated at 74 EFPY), relocation of one or both of these capsules to a higher lead factor location will be required. Since it is not known when or if St. Lucie will apply for a second license extension, and given that many cycles of irradiation will be required for Capsules 104° and 284° to accumulate fluence greater than the vessel wall fluence at 74 EFPY, it is suggested that a potential relocation decision be implemented prior to 40 total years of operation.

8 REFERENCES

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7. ASTM E208, *Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels*, ASTM.

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