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GE-NE-B1301809-01

March 1997

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
WNP-2 RPV SURVEILLANCE MATERIALS
TESTING AND ANALYSIS

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ABSTRACT

The surveillance capsule at the 300° azimuthal location was removed at 7.2 EFPY (normalized full power of 3323 MWt) from the WNP2 reactor in Spring 1996 (the reactor was at uprated power level of 3486 MWt when WNP2 was shutdown). The capsule contained flux wires for neutron fluence measurement and Charpy and tensile test specimens for material property evaluations. The flux wires were evaluated to determine the fluence experienced by the test specimens. Charpy V-Notch impact testing and uniaxial tensile testing were performed to establish the properties of the irradiated surveillance materials.

The irradiated Charpy data for the base plate, weld and heat affected zone (HAZ) specimens were compared to the corresponding unirradiated specimen test data to determine the shift in Charpy curves due to irradiation. Both the irradiated and unirradiated Charpy base plate data are of longitudinal orientation. The shift results for the base plate and the weld materials were also compared with the predictions of the Regulatory Guide 1.99 Revision 2 (RG1.99) and were found to be within the predicted values.

The data from the irradiated materials tested at 70°F and 550°F were compared with those from the unirradiated data tested at the same temperatures to determine the effect of irradiation on the stress-strain relationship of the materials.

The flux wire results were used to calculate the 32 EFPY fluence. The resulting fluence is higher than those based on the neutron flux from the first cycle.

ACKNOWLEDGMENTS

The author gratefully acknowledges the efforts of other people towards completion of the contents of this report.

Cask shipment and capsule disassembly was performed by Jon Myers. Lead factor calculations was provided by D. Rogers and S. Wang. Charpy testing was completed by G. E. Dunning and B. D. Frew. Tensile specimen testing was done by S. B. Wisner and chemical composition analysis was performed by P. Wall. Flux wire testing and analysis was performed by L. Kessler, R. Kruger and R. Reager.

1. INTRODUCTION

An important part of the effort to assure reactor vessel integrity involves evaluation of the fracture toughness of the vessel ferritic materials. The key parameters which characterize a material's fracture toughness are the reference nil-ductility transition temperature (RT_{NDT}) and the upper shelf energy (USE). Both of these parameters, defined in 10CFR50 Appendix G[1] and in Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI [2], are required to be updated from surveillance sample testing and analyses at specified intervals of a reactor's effective full power years. The WNP2 FSAR[3] calls for the first surveillance capsule (at the 300° vessel azimuth location) to be withdrawn at 8 Effective Full Power Years (EFPYs).

Appendix H of 10CFR50[4] and ASTM E185-82[5] establish the methods to be used for surveillance testing of the WNP2 reactor vessel materials. It should be noted that the use of ASTM E185-82 is required by the Appendix H of 10CFR50 which specifies that, for each capsule withdrawal, the test procedures must meet the requirements of ASTM E185-82 to the extent practicable for the configuration of the specimens in the capsule. For the WNP2 surveillance specimen evaluation, however, there are no significant differences between ASTM E185-82 and its earlier version of E185-73 which was referenced in the WNP2 FSAR. During the scheduled outage of 1996, WNP2 completed the first surveillance capsule removal from the reactor at 7.2 EFPY (normalized full power of 3323 MWt, the reactor was at uprated power level of 3486 MWt when it was shutdown) and the irradiated samples were shipped in May, 1996 to the GE Vallecitos Nuclear Center (VNC) for testing.

The surveillance capsule contained flux wires for neutron flux monitoring and Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated from the same vessel materials as those located within the core beltline region. The impact and tensile specimens were tested to establish properties for the irradiated materials. Prior to this effort, GE had received from WNP2 the unirradiated Charpy impact and tensile baseline specimens, which were similarly tested. The test results from these unirradiated specimens constitute as the baseline for evaluating the irradiation effects on the material properties which include (i) the fracture toughness as measured in terms of Charpy impact (absorbed) energy, lateral expansion and percent shear area, and (ii) the stress-strain relationship of the vessel materials.

All the test code requirements and procedures are summarized in Table 1-1.

Table 1-1 Test Code Requirements

Test	Procedure
Tensile Test	Tensile tests are conducted in accordance with (i) ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels, and (ii) ASTM E8-95, "Standard Methods of Tension Testing of Metallic Materials", with the exception that specimen displacement is obtained from crosshead movement of the Instron load frame rather than an extensometer attached to the specimen.
Charpy Impact Test	Charpy tests are conducted according to (i) ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels, and (ii) ASTM E23-94b, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials".
Flux Wire Analysis	The analysis is conducted according to (i) ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels, and (ii) GE Report NEDE-11373, Method 10.1.6.0, Rev.4, "Determination of Neutron Fluence Rate and Fluence Using Neutron Dosimeters". The dosimetry method is adhered to ASTM E181, E261, E262, E263, E264, E523, E844 and E1005.
Chemical Composition Analysis	The analysis is conducted according to GE Report NEDE-11373, Method 10.2.1.3, Rev. 0, "Determination of Element Concentrations In Solutions by DCP Emission Spectroscopy CM & S Laboratory Manual"

2. SUMMARY AND CONCLUSIONS

2.1 SUMMARY OF RESULTS

The 300° azimuth position surveillance capsule was removed and shipped to VNC. The flux wires, Charpy V-Notch and tensile test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of the testing are presented in this report as follows:

Section 3: Surveillance Program Background
- RPV Materials and Fabrication
- Capsule Recovery
- Specimen Description

Section 4: Surveillance Specimen Chemical Composition

Section 5: Peak RPV Fluence Evaluation

Section 6: Charpy V-Notch Impact Testing

Section 7: Tensile Testing

Section 8: Adjusted Reference Temperature and Upper Shelf Energy

The test and analysis summary is provided as follows:

a. The 300° azimuth position capsule was removed from the reactor after 7.2 EFPY of operation. The capsule contained 6 flux wires: 2 each of copper (Cu), iron (Fe), and nickel (Ni). There were 24 Charpy V-Notch specimens in the capsule: 8 each of plate, weld, and heat affected zone (HAZ) materials. The 6 tensile specimens removed consisted of 2 plate, 2 weld and 2 HAZ metal specimens. (see Sections 3.2 and 3.3)

b. One box of unirradiated specimens was received with a total of 68 Charpy specimens (24 for base, 21 for weld and 23 for HAZ) and 12 tensile specimens (4 for base, 3 for weld and 5 for HAZ). Twelve Charpy specimens from each material were tested. Two tensile specimens from each material were tested at room temperature and at 550°F. The remaining samples will be returned to WNP2 for future evaluations.

c. The chemical composition of copper (Cu) and nickel (Ni) for the irradiated surveillance materials were determined from the chemical composition analyses. The average values for the surveillance base plate are 0.11% Cu and 0.49% Ni, and are 0.03% Cu and 0.89% Ni for the surveillance weld. (see Table 4-1)

d. A neutron transport computation had been performed based on Cycle 10 core data in representing the power shape and void distribution of the core. The lead factor was 0.95, relating the surveillance capsule flux to the peak inside surface flux. (see Section 5.2.3)

e. From the flux wire test data, the neutron flux (E >1 MeV) at the surveillance capsule location was determined to be 6.85×10^8 n/cm^2-s, which was larger than both the nominal and the upper bound flux values from the first cycle dosimetry data by 43% and 15% respectively. The flux wire measurement shows that the fluence (E >1 MeV) received by the surveillance specimens was 1.55×10^{17} n/cm^2 at removal. The resulting 32 EFPY RPV peak fluence prediction is 7.57×10^{17} n/cm^2 at the vessel inside diameter (I.D.) wall surface. At 1/4T from the I.D. surface , the 32 EFPY fluence prediction is 5.14×10^{17} n/cm^2. The 32 EFPY fluence was based on the sum of 6.57 EFPY at 3323 MWt for Cycles 1-10 and 25.43 EFPY at 3486 MWt from Cycle 11 to the end of life. (see Section 5.1.3, Section 5.3 and Table 5-4)

f. The surveillance Charpy V-Notch specimens were impact tested at temperatures selected to define the upper shelf energy (USE) and the transition of the Charpy impact (absorbed) energy curves for the plate, weld, and HAZ materials. Measurements were taken of the absorbed energy, lateral expansion and percentage shear. The absorbed energy and lateral expansion values were curve-fit with the hyperbolic tangent function. From these curves, the USE values and the index temperatures for 30 ft-lb, 50 ft-lb and 35 mils lateral expansion (MLE) were obtained (Table 6-4). Fracture surface photographs showing the shearing characteristics of each specimen are presented in Appendix A.

g. The curves of irradiated and unirradiated Charpy specimens established the 30 ft-lb shifts. The base material showed an -1.0°F shift (decrease) and 4.6 ft-lb (3%) increase in USE. The weld material showed an -4.9°F shift (decrease) and a 12.1 ft-lb (12%) increase in USE. (see Table 6-4)

h. The measured shifts (at 7.2 EFPY, End of Cycle 11) of -1.0°F for the base plate and -4.9°F for weld, at the fluence of 1.55×10^{17} n/cm^2, were well within the Reg. Guide 1.99, Rev.2 (RG1.99)[6] predicted range of -23 to 45°F and -50 to 62°F and 12.0°F for the plate and weld, respectively. (see Section 6.3)

i. Both the irradiated and unirradiated tensile specimens were tested at room temperature (70°F) and at reactor operating temperature (550°F). As expected, the results show that the irradiated data has an increase in strength and a decrease in ductility, which is a typical indication of irradiation embrittlement. (see Table 7-1)

j. The 32 EFPY adjusted reference temperature (ART = initial RT_{NDT} + ΔRT_{NDT} + Margin) was predicted for each beltline material, based on the RG1.99 methodology. The ART for the limiting plate (Heat No. C1272-1) at 32 EFPY is 83.8°F and is lower than the 200°F requirement of 10CFR50 Appendix G. (see Table 8-1)

k. An update of the beltline material USE values at 32 EFPY was performed using the RG1.99 methodology. The irradiated USE for all beltline materials will remain above 50 ft-lb through 32 EFPY as required in 10CFR50 Appendix G. (see Section 8.2)

2.2 CONCLUSIONS

The requirements of 10CFR50 Appendix G specifies the vessel design life conditions with limits of operation designed to prevent brittle fracture: From the evaluation of surveillance test results and the related analyses, the following conclusions are made:

a. **The 30 ft-lb shifts and changes in USE are well within the values predicted by RG1.99. From the surveillance test results it is clear that after 11 cycles of operation, the 30 ft-lb shifts and the USE of the base plate and the weld show little change after accumulating a peak irradiation fluence (E >1 MeV) of 1.55×10^{17} n/cm^2 at 7.2 EFPY (normalized full power of 3323 MWt, the reactor was at uprated power level of 3486 MWt for Cycle 11 from 6/9/95 to 3/2/96)**

b. **The values of ART and USE for the reactor vessel beltline materials are expected to remain within limits of 10CFR50 Appendix G (< 200°F and > 50 ft-lb., respectively) for at least 32 EFPY of reactor operation.**

3. SURVEILLANCE PROGRAM BACKGROUND

3.1 RPV MATERIALS AND FABRICATION HISTORY

Material certification records were retrieved from GE Quality Assurance (QA) records to determine chemical and mechanical properties of the vessel materials. The retrieved information is documented in the WNP2 FSAR. Table 3-1 shows the chemistry data for the beltline materials.

The WNP2 RPV is a 251 inch diameter BWR/5 design. Construction was performed by CBI Nuclear Company (CBIN) under the 1971 edition of the ASME Code through the 1971 Summer Addenda. The reactor pressure vessel was primarily constructed from high strength, low alloy (HSLA) steel plate and forging. Plates (for the shell and head plate) were ordered to ASME SA-533, Grade B, Class 1, and forging (for the nozzles and closure flanges) to ASME SA-508, Class 2, and the studs, nuts and washers for the main closure flange were ordered to ASME SA540, Grade B23 or Grade B24[3]. The fabrication process of the vessel plates included hot forming immediately followed by a quench and temper heat treatment. During the assembly process for the completed vessel, submerged arc welding and shielded metal arc welding of plates were applied and were followed by post-weld heat treatment at 1150°F. The identification of plates and welds in the beltline region is shown in Figure 3-1.

3.2 CAPSULE RECOVERY

The reactor pressure vessel (RPV) surveillance program consists of three surveillance capsules at 30°, 120°, and 300° azimuths at the core midplane. The specimen capsules are held against the RPV inside surface by a spring loaded specimen holder. Each capsule is expected to receive equal irradiation because of core symmetry. During the current 1996 Spring outage, a surveillance capsule was removed from the 300° azimuthal location. The capsule was cut from its holder assembly and shipped by cask to the GE Vallecitos Nuclear Center (VNC), where testing was performed.

Upon arrival at VNC, the capsule basket was examined for identification. The identification number stamped on the capsule basket (specimen holder) corresponded to basket number 131C7717G1 and reactor number 55, as specified by GE drawings, 131C7717 (specimen holder) and 105D4719G001 (Surveillance Program), for the WNP2 300° surveillance materials. The general condition of the basket as received is shown in Figure 3-2. The basket contained

two impact (Charpy) specimen containers (numbers stamped on the containers: 131C7716G4 & 131C7716G5, respectively shown in Figure 3-3(a) and three tensile specimen capsules (numbers stamped: G1, G2 and G3, respectively shown in Figure 3-3(b). During the removal of the Charpy impact specimens from the specimen holder, one specimen container was found to have leaked. The specimens were visually examined for features that could possibly affect test results. The specimens appeared somewhat darker in appearance than the other specimens. This uniform discoloration was most likely caused by the exposure of the specimen to the high temperature water environment. The surfaces of the discolored specimens were similar to the other specimens, i.e., no defects, pits, or detrimental corrosion was observed. This is expected since the reactor water trapped inside the capsule was expected to be stagnant in an enclosed space where the activities of corrosive agent in the trapped reactor water to be at minimum. Based on these observations, it is was concluded that the specimens were not affected by the exposure to water, and will give credible surveillance results.

3.3 SPECIMEN DESCRIPTION

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

3.3.1 Charpy Specimens

The fabrication of the Charpy specimens is described in the CBIN drawings[7] of the surveillance test program. The test plate used for the surveillance were cut from the same material of the same heat as one of the beltline plates[7,8,9] , which was Heat B5301-1. Sub-blocks were cut from the test plate for base, weld and HAZ test specimens. Two sub-blocks were welded together for the weld and HAZ specimens, using the weld electrode heat number of 3P4966[7] . The same weld electrode number was also used for one of the vertical welds in the lower No.1 shell (between slabs 1 and 2) during the vessel assembly process. Thus, the test plate had gone through the same fabrication process of a quench and temper heat treatment immediately after hot forming, then submerged arc welding. Finally, the base metal, weld metal and HAZ test plates or sub-blocks had gone through the post weld heat treatment for 50 hours at 1150°F ± 25°F followed by furnace cooling to below 600°F then air cooled, in a manner that will simulate the actual heat treatment performed on the core region shell plate of the completed vessel. The Charpy and the tensile specimens were then machined from these sub-blocks, as

described in the CBIN surveillance test specimen document [7] and the GE surveillance program document[8] . Charpy specimens were machined from the 1/4 T and 3/4 T positions in the plate, in the longitudinal orientation (long axis parallel to the rolling direction) and were stamped on one end with the fabrication codes[7,8] and on the other end the vessel code of 55 (WNP2). The base metal orientation in HAZ specimens was also in longitudinal direction.

3.3.2 Tensile Specimens

The surveillance tensile specimens were fabricated of the same materials of the Charpy specimens described in the surveillance specimen drawings[7,9] . The materials, and thus the chemical compositions and heat treatments for the base, weld, and HAZ Charpy and tensile specimens are identical.

TABLE 3-1(a) CHEMICAL COMPOSITION & INITIAL RT_{NDT} OF RPV BELTLINE PLATE MATERIALS FROM WNP2 FSAR RECORDS [1]

		Composition by Weight Percent								
Identification	Heat/Lot No.	Cu	Ni	C	Mn	P	S	Si	Mo	Initial RT_{NDT} (°F)
Lower Shell Plates:										
21-1-1	C1272-1	0.15	0.60	0.23	1.31	0.013	0.02	0.26	0.53	28
21-1-2	C1273-1	0.14	0.60	0.23	1.28	0.014	0.02	0.23	0.57	20
21-1-3	C1273-2	0.14	0.60	0.23	1.28	0.014	0.02	0.23	0.57	4
21-1-4	C1272-2	0.15	0.60	0.23	1.31	0.013	0.02	0.26	0.55	0
Lower-Intermediate Shell										
22-1-1	B5301-1	0.14	0.50	0.20	1.34	0.017	0.01	0.23	0.52	-20
22-1-2	C1336-1	0.13	0.50	0.21	1.36	0.017	0.01	0.22	0.49	-8
22-1-3	C1337-1	0.15	0.51	0.22	1.32	0.018	0.01	0.21	0.50	-20
22-1-4	C1337-2	0.15	0.51	0.22	1.32	0.018	0.01	0.21	0.50	-20

Note:
[1] Data from WNP2 Letter, Ref.[25].

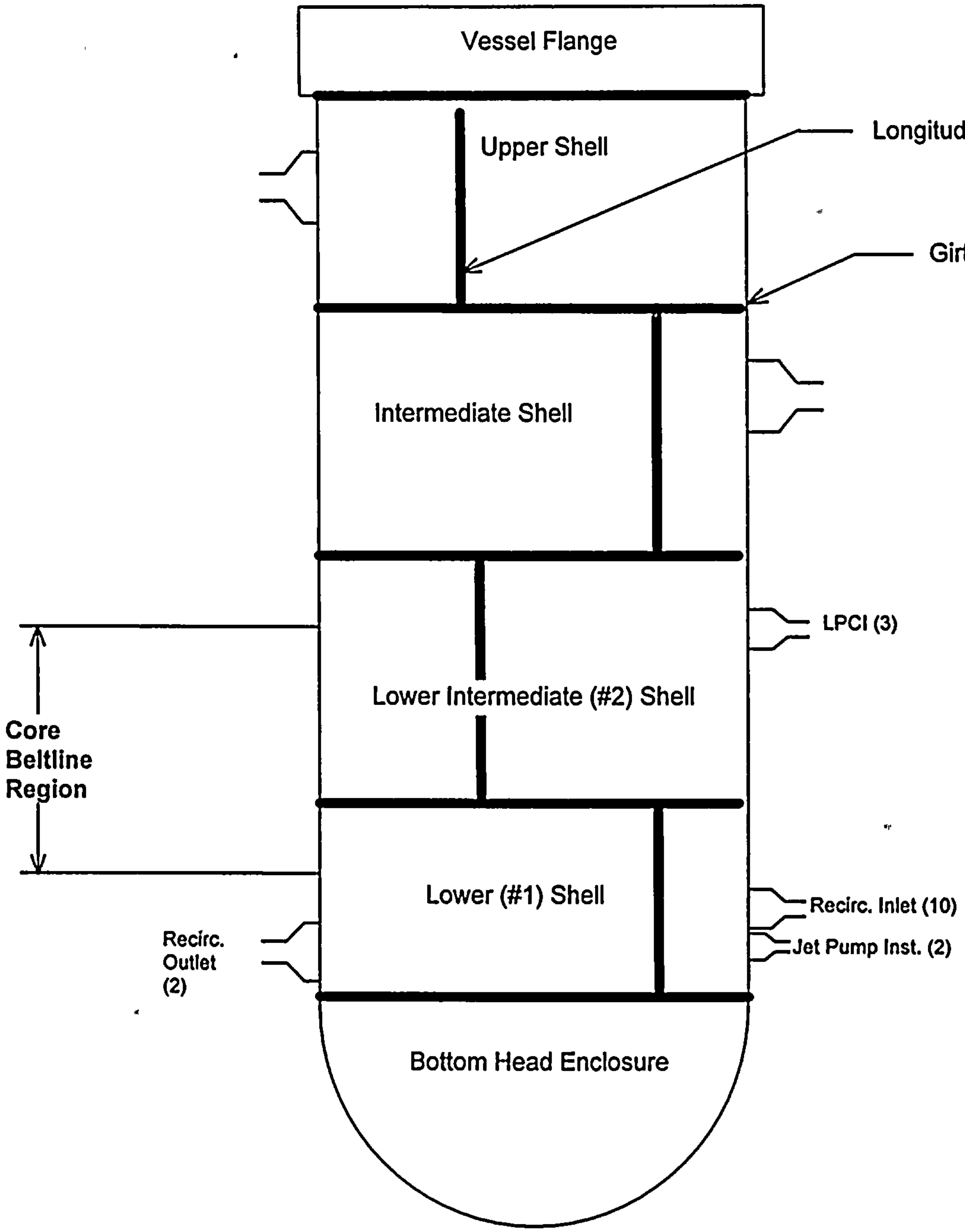
TABLE 3-1(b) CHEMICAL COMPOSITION & INITIAL RT_{NDT} OF RPV BELTLINE WELD MATERIALS FROM WNP2 FSAR RECORDS [1]

Identification[2]	Heat/Lot No.	Cu	Ni	C	Mn	P	S	Si	Mo	V	Initial RT_{NDT}[1] (°F)
		Composition by Weight Percent									
Lower - Long. BA, BB, BD	04P046	0.06	0.90	0.044	1.04	0.009	0.021	0.40	0.58	0.02	-48
Lower - Long. BA, BB	07L669	0.03	1.02	0.05	1.24	0.014	0.016	0.48	0.54	--	-50
Lower - Long. BA - BD	3P4966	0.02	0.80	0.059	1.35	0.013	0.013	0.38	0.50	0.005	-30
Lower - Long. BA - BD	3P4966	0.02	0.92	0.077	1.42	0.014	0.013	0.41	0.53	0.005	-48
Lower - Long. BB, BC, BD	C3L46C	0.02	0.87	0.063	0.96	0.019	0.017	0.32	0.53	--	-20
Lower - Long. BB	08M365	0.02	1.10	0.057	1.23	0.02	0.023	0.47	0.57	--	-48
Lower - Long. BC	09L853	0.03	0.86	0.052	1.23	0.018	0.023	0.46	0.51	--	-50
Lower - Lower Interm. BE-BH	3P4966	0.03	0.88	0.074	1.38	0.010	0.013	0.36	0.49	0.006	-26
Lower - Lower Interm. BE-BH	3P4966	0.03	0.90	0.067	1.39	0.011	0.014	0.38	0.53	0.008	-6
Long. BF, BH	04P046	0.06	0.90	0.044	1.04	0.009	0.021	0.40	0.58	0.02	-48
Long. BF	05P018	0.09	0.90	0.057	1.21	0.008	0.021	0.44	0.53	0.01	-38
Long. BG	624063	0.03	1.00	0.041	1.12	0.009	0.018	0.41	0.54	0.01	-50
Long. BH	624039	0.07	1.01	0.060	1.11	0.015	0.025	0.45	0.57	0.02	-50
Long. BG	624039	0.10	0.92	0.041	1.12	0.01	0.02	0.45	0.53	0.01	-36
Lower - Lower Interm. AB	492L4871	0.03	0.98	0.07	1.17	0.02	0.02	0.32	0.51	0.02	-50
Lower - Lower Interm. AB	5P6756	0.08	0.93	0.063	1.27	0.01	0.011	0.57	0.45	0.006	-50
Lower - Lower Interm. AB	5P6756	0.09	0.92	0.078	1.24	0.01	0.012	0.53	0.46	0.006	-50
Girth Weld AB	3P4955	0.025	0.90	0.035	1.33	0.016	0.011	0.56	0.52	0.006	-44
Girth Weld AB	3P4955	0.023	0.95	0.054	1.28	0.016	0.010	0.55	0.54	0.007	-16
Girth Weld AB	04T931	0.03	1.00	0.05	1.03	0.02	0.024	0.28	0.53	0.01	-50

Notes:

[1] Data from WNP2 Letter, Ref.[25]

[2] Weld Location Identification from WNP2 Power Uprate Report, Ref.[10]



#2 Shell Ring	
Plate:	Heat:
22-1-1	B5301
22-1-2	C1336
22-1-3	C1337
22-1-4	C1337

#1 Shell Ring	
Plate:	Heat:
21-1-1	C1272
21-1-2	C1273
21-1-3	C1273
21-1-4	C1272

FIGURE 3-1 RPV SCHEMATIC WITH BELTLINE PLATE IDENTIFICATION

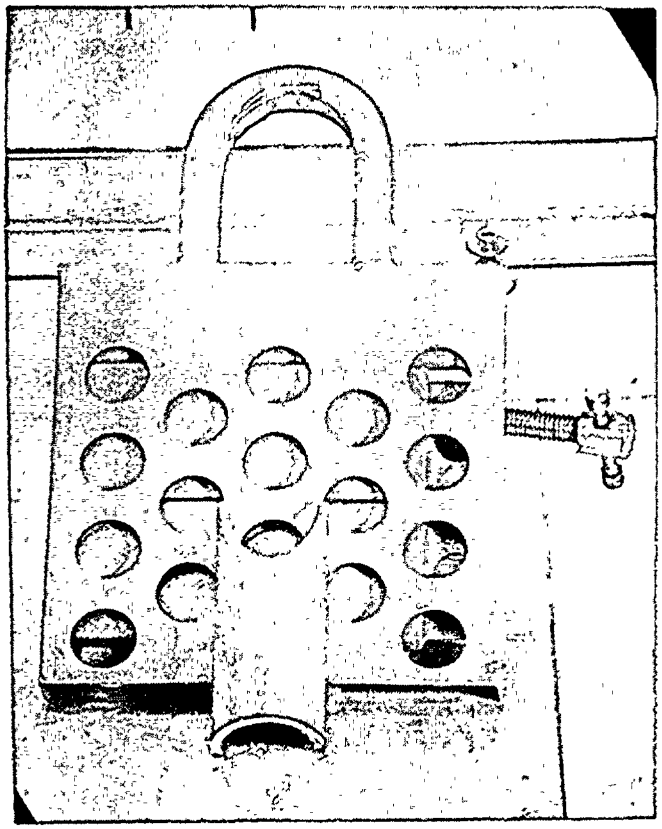


FIGURE 3-2 RECOVERED SURVEILLANCE CAPSULE BASKET

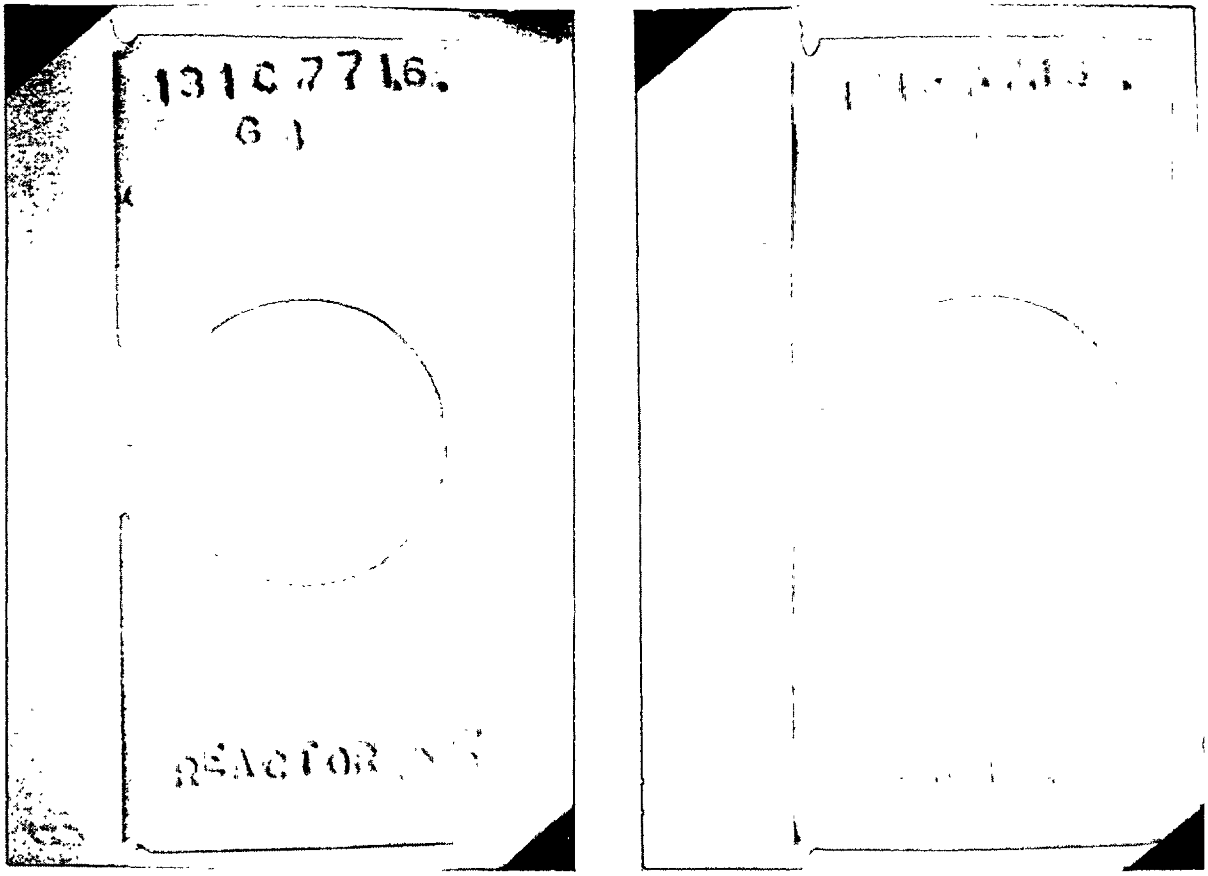


FIGURE 3-3(a) CHARPY SPECIMEN CONTAINER IDENTIFICATION

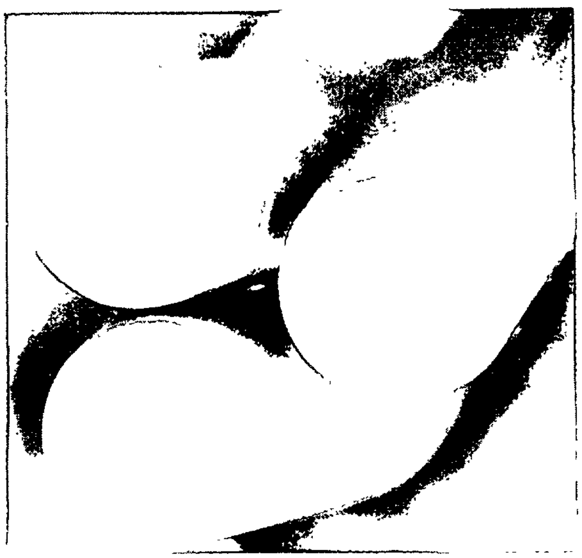


FIGURE 3-3(b) TENSILE SPECIMEN IDENTIFICATION

4. SURVEILLANCE SPECIMEN CHEMICAL COMPOSITION

Samples were taken from the irradiated base and weld Charpy specimens after they were tested. Chemical analyses were performed using a Spectraspan III plasma emission spectrometer. Each sample was dissolved in an acid solution to a concentration of 40 mg steel per ml solution. The spectrometer was calibrated for determination of Mn, Ni, Mo, Cr, Si and Cu by diluting National Institute of Standards and Technology (NIST) Spectrometric Standard Solutions. The phosphorus calibration involved analysis of five reference materials from NIST with known phosphorus levels. Analysis accuracies are ±0.005% (absolute) of reported value for phosphorus and ±5% (relative) of reported value for other elements. The chemical composition results are given in Table 4-1 for both irradiated and baseline surveillance plate and weld materials. The baseline data were taken from CBIN CMTR material certification records for the plate and weld surveillance specimens[3].

TABLE 4-1 CHEMICAL COMPOSITION OF WNP2 SURVEILLANCE SPECIMENS FROM GE CHEMICAL ANALYSIS

Specimen ID (Heat No.)	Metal Type	Cu (wt%) [1]	Ni (wt%) [1]	Mn (wt%)	Mo (wt%)	Si (wt%)	Cr (wt%)	P (wt%)
29141 (B5301)	Base	0.12	0.51	1.31	0.53	0.12	0.18	0.010
29143 (B5301)	Base	0.11	0.49	1.25	0.50	0.11	0.18	0.014
29146 (B5301)	Base	0.11	0.47	1.21	0.48	0.10	0.17	0.010
Baseline [2]	Base	0.14	0.50	1.34	0.52	0.23	NA	0.017
29148 (3P4966)	Weld	0.03	1.00	1.40	0.52	0.25	0.08	0.011
29125 (3P4966)	Weld	0.03	0.86	1.23	0.42	0.21	0.06	0.008
29153 (3P4966)	Weld	0.03	0.80	1.20	0.40	0.13	0.05	0.010
Baseline [2]	Weld	0.03	0.90	1.39	0.53	0.38	NA	0.011

Notes: [1] Average GE Compositions:

Base Cu: 0.11% Base Ni: 0.49%

Weld Cu: 0.03% Weld Ni: 0.89%

[2] From WNP2 FSAR, Ref.[3]

5. PEAK RPV FLUENCE EVALUATION

Flux wires removed from the 300° location capsule were analyzed, as described in Section 5.1, to determine flux and fluence received by the surveillance capsule. The lead factor, determined as described in Section 5.2, was used to establish the peak vessel fluence from the flux wire results. Section 5.3 includes 32 EFPY peak fluence estimates.

5.1 FLUX WIRE ANALYSIS

5.1.1 Procedure

The surveillance capsule contained two sets of flux wires: each set with one each of iron, copper and nickel, totaling 6 flux wires. Each wire was removed from the capsule, cleaned with dilute nitric acid followed by rinses with water and acetone, weighed, mounted on a counting card, and analyzed for its radioactivity content by gamma spectrometry. Each iron wire was analyzed for ^{54}Mn content, each nickel wire for ^{58}Co and each copper wire for ^{60}Co at a calibrated 4-cm, 10-cm and 20 cm source-to-detector distance with 35-cc and 100-cc Ge(Li) gamma spectrometers.

To properly predict the flux and fluence at the surveillance capsule from the activity of the flux wires, the periods of full and partial power irradiation and the zero power decay periods were considered. Operating days for each fuel cycle and the reactor average power fraction were derived from records provided by Washington Public Power Supply System and are shown in Tables 5-1 and 5-2 respectively.

From the flux wire activity measurements and power history, reaction rates for ^{54}Fe(n,p)^{54} Mn, ^{58}Ni(n,p)^{58}Co and ^{63}Cu(n,α)^{60}Co were calculated. The cross sections for the iron, nickel and copper wires are 0.213 barn, 0.273 barn and 0.00374 barn respectively. The values for Fe and Cu are based on the experimental data correlation which establishes the cross sections as a function of the water gap (between the RPV inside surface and the fuel radius). The cross section of ^{58}Ni was calculated assuming its ratio to that of ^{54}Fe is independent of the water gap, based on the observation that ^{54}Fe and ^{58}Ni are activated in the same energy range. In Browns Ferry Unit 3, which is also a 251 inch, 764 fuel bundle plant, the cross sections of ^{54}Fe and ^{58}Ni are 210 mb and 270 mb respectively[13]. Therefore, for WNP 2 the σ value of ^{58}Ni is ratioed according to 213*270/210 which is 273 mb. These σ values are consistent with the cross section data established at GE's Vallecitos Nuclear Center based on more than 65 spectral determinations for

BWRs and for the General Electric Test Reactor using activation monitors and spectrum unfolding techniques. The cross sections for > 0.1 MeV flux were determined from the measured 0.1 to 1 MeV cross section ratio of 1.6 [14].

5.1.2 Method of calculation:

The reaction rates can be calculated according to[15,16,17]

$$R = \left(\frac{dps}{g}\right)\frac{M}{A\alpha\sum_i p_i(1-e^{-\lambda t_{ei}})(e^{-\lambda t_{di}})} \qquad \text{(Eq 5-1)}$$

where dps/g is the measured disintegration rate of the sample (^{54}Mn, ^{58}Co and ^{60}Co respectively) at end-of irradiation, M is the atomic weight of the element, A is Avogadro's number, α is the isotopic abundance of the target nuclide, p_i is the full-power fraction for period i, using the information in Table 5-1, λ is the decay constant, t_{ei} is the exposure (irradiation duration) time of period i, and t_{di} is the decay (elapsed) time between the end of period i and the end of irradiation or counting. Data from the cycle 1-11 power generation histories were used (Table 5-1). A summary of these key parameters is provided in Table 5-3.

Using the reaction rate, the full power flux is calculated according to

$$\Phi_{fp} = R/\sigma \qquad \text{(Eq 5-2).}$$

where σ is the reaction cross sections shown in Table 5-3.

The E > 1 MeV fluence (total) is then calculated according to

$$\Phi_t = \Phi_{fp}\sum_i p_i t_i \qquad \text{(Eq 5-3).}$$

where, Φ_{fp} = full power flux value from Eq 5-2

t_i = operating time

p_i = full power fraction

where the summation equals 2626.2 days (with p_i normalized at 3323 MWt and the normalization affects the results of Eq. 5-1 and Eq. 5-2 but not Eq. 5-3) based on the power history data in Table 5-2.

5.1.3 Results

The fluence calculated according to Eq 5-3 is summarized in Table 5-4 which shows that the three wire fluences differ by less than 10%. For conservatism the maximum fluence (E > 1 MeV) value of 1.55×10^{17} n/cm^2 for Ni is used. The accuracies of the values in Tables 5-3 for a 2 σ deviation are influenced by the following sources of error:

Relative Error:	±3%
Counting Rates:	±1%
Power History:	±15%
Cross Section:	±10%

The overall 2σ error is $(3^2 + 1^2 + 15^2 + 10^2)^{0.5}$ % = 18%.

The 11 cycle results can be compared with those of the Cycle 1 analysis[18] , which included nominal and upper bound neutron fluxes (E > 1 MeV) of 4.8×10^8 and 6.0×10^8 n/cm^2-s, respectively. Comparing the flux result 6.85×10^8 n/cm^2-s of reported here, the difference may be accounted for by the following factors:

1. Cycle 1 fuel load may have included low enrichment of outer bundles.
2. The difference in capsule positions (Cycle 1 flux wire at vessel 30° azimuth and Cycle 11 wire at the 300° location) could account for some slight variation.
3. Cycle 1 operation could have had an unusual radial power distribution (averaged over the cycle)
4. Cycle 1 operation could have had an unusual axial power distribution (averaged over the cycle)

5.2 DETERMINATION OF LEAD FACTOR

The flux wires are used to detect neutron flux at the location of the surveillance capsule. The wires will reflect the power fluctuations associated with the operation of the plant. However, the flux wires are not necessarily at the location of peak vessel flux. A lead factor relating the flux at the flux wire to the peak flux at vessel surface has to be determined in order to assess the irradiation level at the reactor vessel wall. Vessel ID lead factor is defined as the ratio of flux at the surveillance capsule to the peak flux (φ_{id}) at the vessel inner surface, i.e. (lead factor) x φ_{id} = capsule flux. Lead factor is a function of core configuration as well as vessel geometry. It is also dependent on the power density and coolant density distributions in the core.

The lead factor for the WNP2 surveillance capsule was determined using Cycle 10 core data as basis in representing the power shape and void distribution of the core. WNP2 rated power was 3323 MWt until the end of Cycle 10. The two-dimensional transport code DORT[19] was used to calculate the spatial flux distribution produced by a fixed source of neutrons in the core region. DORT is a deterministic code using discrete-ordinates S_N method to solve the integro-differential form of the Boltzmann transport equation.

Two DORT calculations were performed to simulate R-θ and R-Z configurations of the reactor. The resultant flux distributions can be synthesized to obtain an equivalent R-θ-Z flux distribution. Lead factor, or the ratio of flux at the capsule location to that at the peak flux location, can be determined from this distribution.

5.2.1 R-θ Calculation

Flux distribution in the azimuthal direction was obtained by calculation in the R-θ geometry. The calculation model incorporates inner and peripheral core regions, the shroud, water regions inside and outside the shroud, and the reactor vessel wall. The core region material compositions and neutron source densities were typical of those at the core midplane. Core midplane is chosen for the R-θ calculation because it is representative for beltline flux evaluation, it is also the center elevation of the surveillance capsule specimen holder. Neutron cross-section data used in the calculation is a 26-group, spatial and composition dependent cross section set, which was condensed from the Los Alamos Scientific Laboratory (LASL) 80 group microscopic cross section data library. The spatial mesh of the calculation model consists of 155 radial intervals and 90 azimuthal intervals. Figure 5-1 is a schematic view of the R-θ model. Since WNP2 core has an eight-fold symmetry, only 1/8 of the core was modeled with reflective boundary conditions assumed at the 0° and 45° boundaries. A reflective boundary is also assumed at R = 167 cm to save computation time. Because the central core region with R < 167 cm has little effect on the flux level at reactor vessel, this portion of the core need not be modeled in detail. Output of this calculation is the flux distribution as functions of azimuthal angle and radial distance. The magnitude of neutron flux at the surveillance capsule center can be read from the output file at 30° and R = 319 cm. Figure 5-2 shows the angular (R-θ) flux variation at the core midplane.

5.2.2 R-Z Calculation

Neutron flux profile in the axial direction is determined by the R-Z calculation. The calculation model incorporates similar material regions as in the R-θ model. However, the full

length of the core was modeled with simulated data corresponding to those at core 30° azimuth, including material composition, radial dimension, coolant density, and power density distribution in the axial direction. Based on operating experiences, the axial profile of fast neutron flux near the core edge is not very sensitive to the radial distance nor the azimuthal direction of the axis. Therefore other azimuthal directions could be used as representative of the core for R-Z calculations, the results would be similar. In our analysis, azimuth 30° of a 45° model is chosen because it is equivalent to the surveillance capsule location. Figure 5-3 is a schematic view of the R-Z calculation model. There are 155 spatial meshes in the R-direction and 125 intervals in the Z-direction.

Output from the R-Z calculation provides flux variations as functions of elevation. A relative flux profile normalized to the peak vessel flux is given in Figure 5-4. The ratio of flux at core midplane to the peak flux can be readily estimated.

5.2.3 Results

The result of R-θ calculation demonstrated in Figure 5-2 shows that along the circumference of the reactor vessel, the flux peaks around 25° past quadrant reference. The flux level at core midplane, at the location of surveillance capsule is 7.34×10^8 n/cm^2-sec, which differs from the measured flux of 6.85×10^8 n/cm^2-sec by 7%. The maximum vessel surface flux at core midplane can be read from the same figure as approximately 7.23×10^8 n/cm^2-sec.

The axial flux profile shown in Figure 5-4 indicates that along the longitudinal direction of vessel surface, flux peaks approximately 100 inches above the bottom of active fuel (BAF). The peaking factor from core midplane to the peak elevation is 1/0.932, or 1.073. Therefore, peak vessel surface flux is quantified as 7.23×10^8 n/cm^2-sec x 1.073 = 7.76×10^8 n/cm^2-sec. Therefore the lead factor at the vessel surface is $7.34\times10^8 / 7.76\times10^8 = 0.95$. This value is very similar to the lead factor of 0.98 previously calculated for the first cycle flux wire measurement[18] .

A similar procedure is applied to the flux results at vessel 1/4 T. The peak flux at vessel 1/4 T is determined to be 5.16×10^8 n/cm^2-sec. Therefore the lead factor at vessel 1/4 T is $7.34\times10^8 / 5.16\times10^8 = 1.42$. This value is comparable to the lead factor of 1.51 previously calculated for the first cycle flux wire measurement[18].

5.3 EVALUATION OF 32 EFPY FLUENCE

Generally, the inside surface fluence (f_{surf}) at 32 EFPY is determined from the flux wire fluence at a particular EFPY and lead factor according to

$$f_{idmax} = (f_{cfc} * 32\ \text{EFPY})/(\text{LF} * \text{CEFPY}) \qquad \text{(Eq 5-5)}$$

where f_{idmax} = 32 EFPY fluence at the peak vessel inside surface
f_{cfc} = capsule fluence measured at the current CEFPY (WNP2: 1.55×10^{17} n/cm^2)
32 EFPY = end of life EFPY based on a 40-year operation at an 80% capacity factor
CEFPY = the current EFPY for the capsule (WNP2: 7.2 at 3323 MWt)
LF = lead factor (WNP2: 0.95)

Since the WNP2 was uprated for Cycle 11, the method of the fluence calculation of the above general equation has to be modified to include the uprated cycles. The value of CEFPY is the sum of 6.57 EFPY at 3323 rated thermal power (MWt) and 0.6 EFPY at 3486 MWt. Assuming that WNP2 will be operated at 3486 MWt for the remaining cycles, 32 EFPY is the sum of 6.57 EFPY at 3323 MWt and 25.43 EFPY at 3486 MWt. Therefore the fluence (f_{3323}) cumulated at 3323 MWt can be ratioed from the effective full power days (EFPD) provided in Table 5-2:

$$f_{3323} = f_{cfc} \times 2398.1 / (2398.1 + 228.3) = 1.42 \times 10^{17}\ \text{n/cm}^2$$

And the fluence (f_{3486}) cumulated at 3486 MWt is ratioed from f_{3323}:

$$f_{3486} = f_{3323} \times (25.43/6.57)(3486/3323) = 5.77 \times 10^{17}\ \text{n/cm}^2$$

Therefore the resulting 32 EFPY fluence value at the peak location of the vessel inside surface is:

$$f_{surf} = (f_{3323} + f_{3486})/0.95 = 7.57 \times 10^{17}\ \text{n/cm}^2$$

The above peak surface fluence is higher than both the nominal fluence of 4.9×10^{17} n/cm^2 and the upper bound fluence of 6.2×10^{17} n/cm^2 from the dosimetry measurement of the first cycle[18]. The probable causes for the difference in fluence from the first cycle and the 11th cycle measurements have already been discussed in Section 5.1.3.

Figure 5-4 shows that the fast neutron flux level drops off significantly near both ends (TAF and BAF) of the active fuel. The top of the WNP2 lower shell is at elevation 230 inches, or approximately 14 inches above BAF. From Figure 5-4, the flux level at this elevation is less than 80% of the peak flux. Since the lower shell has the most limiting RT_{NDT} value, the conventional method of calculating the ART based on the peak fluence is too restrictive. Therefore the fluence level at the lower shell is conservatively adjusted to be 80% of the vessel peak fluence. The ART calculation provided in Section 8 is based on the adjusted flux value to account for the axial location of the lower shell relative to the active fuel region.

The fracture toughness analysis is based on 1/4 T depth flaw in the beltline region, so the attenuation of the flux to that depth is considered. The attenuation is calculated according to the RG1.99 equation:

$$f = f_{surf}(e^{-0.24x}), \qquad \text{(Eq 5-6)}$$

where x is the distance, in inches, from the inner vessel surface to the 1/4 T depth. Based on the vessel beltline plate thickness of 7.5 inches for lower (No.1) shell and 6.44 inches for the lower-intermediate (No.2) shell[12] , the corresponding peak 1/4 T fluences at 32 EFPY are 3.86×10^{17} (adjusted) and 5.14×10^{17} n/cm^2, respectively.

TABLE 5-1 SUMMARY OF DAILY POWER HISTORY

Cycle	Begin	End	Duration (t_i)	MWd	Cycle	Begin	End	Duration (t_i)	MWd
1	5/27/84	3/16/86	659	1091606	11	6/9/95	6/13/95	5	370
2	6/4/86	4/10/87	311	723417		6/13/95	6/14/95	1	802
3	6/19/87	4/30/88	317	763293		6/14/95	7/3/95	19	0
4	6/19/88	4/29/89	315	793865		7/3/95	7/5/95	2	1875
5	6/25/89	4/21/90	301	844042		7/5/95	7/9/95	4	7521
6	8/4/90	4/13/91	253	736045		7/9/95	7/13/95	4	13605
7	9/26/91	4/18/92	206	496456		7/13/95	8/13/95	31	108172
8	7/5/92	5/1/93	301	764290		8/13/95	8/20/95	7	16850
9	6/19/93	6/22/93	4	673		8/20/95	11/3/95	75	258193
	6/22/93	6/24/93	2	2270		11/3/95	11/6/95	3	8555
	6/24/93	6/26/93	2	2020		11/6/95	11/26/95	20	66250
	6/26/93	6/27/93	1	3907		11/26/95	11/30/95	4	14340
	6/27/93	6/28/93	1	3313		11/30/95	12/2/95	2	5591
	6/28/93	7/16/93	18	59136		12/2/95	12/27/95	25	58252
	7/16/93	7/17/93	1	3023		12/27/95	1/11/96	15	52195
	7/17/93	8/3/93	17	54661		1/11/96	1/16/96	5	13291
	8/3/93	8/16/93	13	6851		1/16/96	1/19/96	3	10367
	8/16/93	1/14/94	151	502247		1/19/96	1/22/96	3	7429
	1/14/94	1/16/94	2	4686		1/22/96	2/3/96	12	41712
	1/16/94	2/27/94	42	137755		2/3/96	2/7/96	4	10043
	2/27/94	3/7/94	8	26228		2/7/96	2/11/96	4	12670
	3/7/94	3/8/94	1	647		2/11/96	2/14/96	3	9655
	3/8/94	3/13/94	5	17332		2/14/96	2/25/96	11	25185
	3/13/94	3/14/94	1	3445		2/25/96	2/29/96	4	12981
	3/14/94	3/22/94	8	24974		2/29/96	3/1/96	1	233
	3/22/94	3/26/94	4	12434		3/1/96	3/2/96	1	2550
	3/26/94	3/27/94	1	3023					
	3/27/94	3/30/94	3	8474					
	3/30/94	4/26/94	27	62146					
10	7/25/94	8/3/94	10	7136					
	8/3/94	2/1/95	182	603115					
	2/1/95	2/5/95	4	10822					
	2/5/95	2/22/95	17	43170					
	2/22/95	2/25/95	3	2536					
	2/25/95	2/26/95	1	5293					
	2/26/95	3/4/95	6	5620					
	3/4/95	4/5/95	32	106638					
	4/5/95	4/13/95	8	1686					
	4/13/95	4/22/95	9	29290					

Note: Full power of Cycles 1 - 10 = 3323 MWt
Full power of Cycle 11 = 3486 Mwt

TABLE 5-2 SUMMARY OF WNP2 IRRADIATION PERIODS

Cycle	On	Off	Duration	MWt-d	Full power days	Full Power Factor	Rated Thermal Power (MW)
1	5/27/84	3/16/86	659	1091606	328.5	0.498	3323
2	6/4/86	4/10/87	311	723417	217.7	0.700	3323
3	6/19/87	4/30/88	317	763293	229.7	0.725	3323
4	6/19/88	4/29/89	315	793865	238.9	0.758	3323
5	6/25/89	4/21/90	301	844042	254.0	0.844	3323
6	8/4/90	4/13/91	253	736045	221.5	0.875	3323
7	9/26/91	4/18/92	206	496456	149.4	0.725	3323
8	7/5/92	5/1/93	301	764290	230.0	0.764	3323
9	6/19/93	4/26/94	312	940409	283.0	0.907	3323
10	7/25/94	4/22/95	272	815464	245.4	0.902	3323
11	6/9/95	3/2/96	268	758554	217.6 [1]	0.812	3486

Total EFPD = 2615.7
Total EFPY = 7.2

Note: [1] Equivalent full power day for Cycle 11 normalized at 3323 MWt = 217.6 x 3486 /3323 = 228.3.
Total EFPD for Cycles 1-11 normalized at 3323 Mwt = 2626.4, EFPY = 7.2

TABLE 5-3 DOSIMETER NUCLEAR PARAMETERS

Dosimeter	Target nucleus	Isotopic abundance (%)	Radionuclide	Half-life	λ (d)	> 1 MeV cross section (mb), ($\pm$ 2s)
Fe	^{54}Fe	5.8	^{54}Mn	312.3 d	2.2195 E-3	213 $\pm$ 21
Ni	^{58}Ni	68.3	^{58}Co	70.82 d	9.7874 E-3	273 $\pm$ 27
Cu	^{63}Cu	69.2	^{60}Co	5.271 y	3.6003 E-4	3.74 $\pm$ 0.4

TABLE 5-4 SURVEILLANCE CAPSULE FLUX AND FLUENCE FOR IRRADIATION FROM START-UP TO 3/2/96 (300° Azimuth Capsule at 7.2 EFPY at full power of 3323 MWt)

Wire (Element)	Average dps/g Element (at end of irradiation)	Average Reaction Rate [1] [dps/nucleus (saturated)]	Full Power Flux [2] (n/cm^2-s) E≥1 MeV	Full Power Flux [3] (n/cm^2-s) E≥0.1 MeV	Fluence (n/cm^2) E≥1 MeV	Fluence [3] (n/cm^2) E≥0.1 MeV
Fe	6.85E+04	1.45E-16	6.83E+08	1.09E+09	1.55E+17	2.48E+17
Ni	1.13E+06	1.87E-16	6.85E+08	1.10E+09	1.55E+17	2.48E+17
Cu	8.28E+03	2.49E-18	6.65E+08	1.06E+09	1.51E+17	2.42E+17

Value used fluence[4] : 1.55E+17

Notes:
[1] Rounded off to two decimal points
[2] Full power flux, based on thermal power of 3323 MWt
[3] 1.6 times the E >1 MeV result
[4] Maximum of the three flux wire fluences

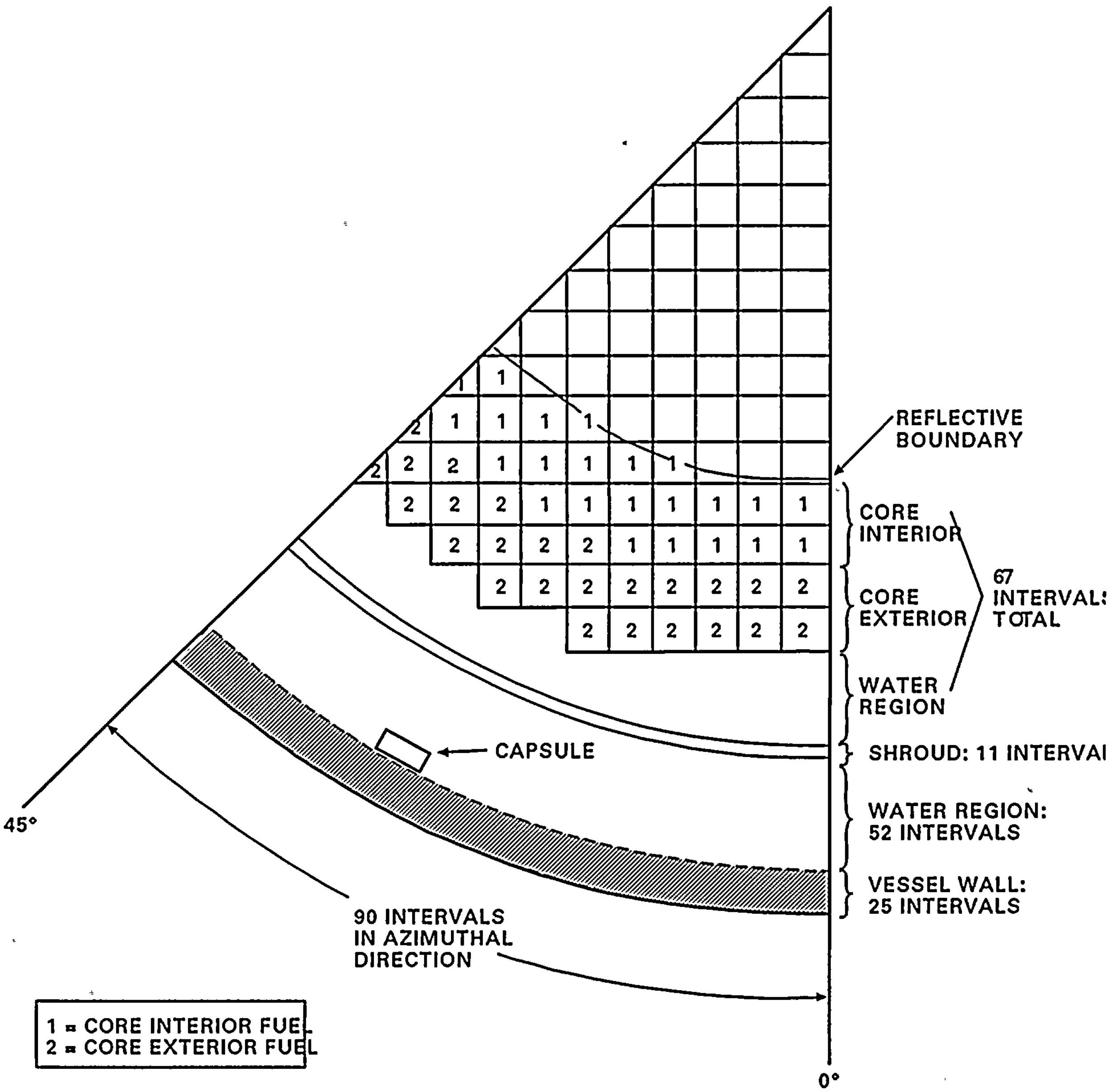


FIGURE 5-1 AZIMUTHAL FLUX DISTRIBUTION ANALYSIS MODEL

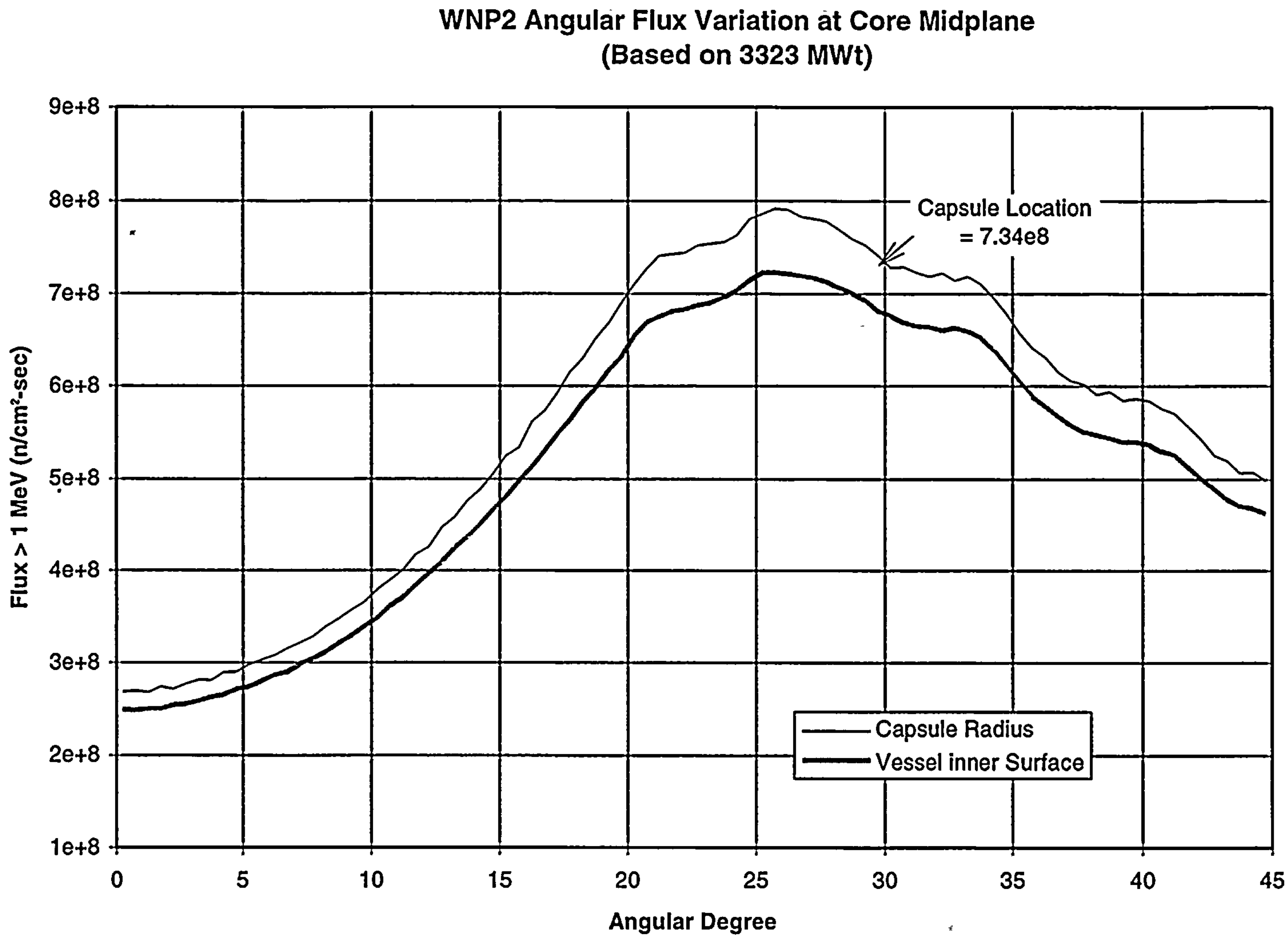


FIGURE 5-2 ANGULAR FLUX VARIATION AT CORE MIDPLANE

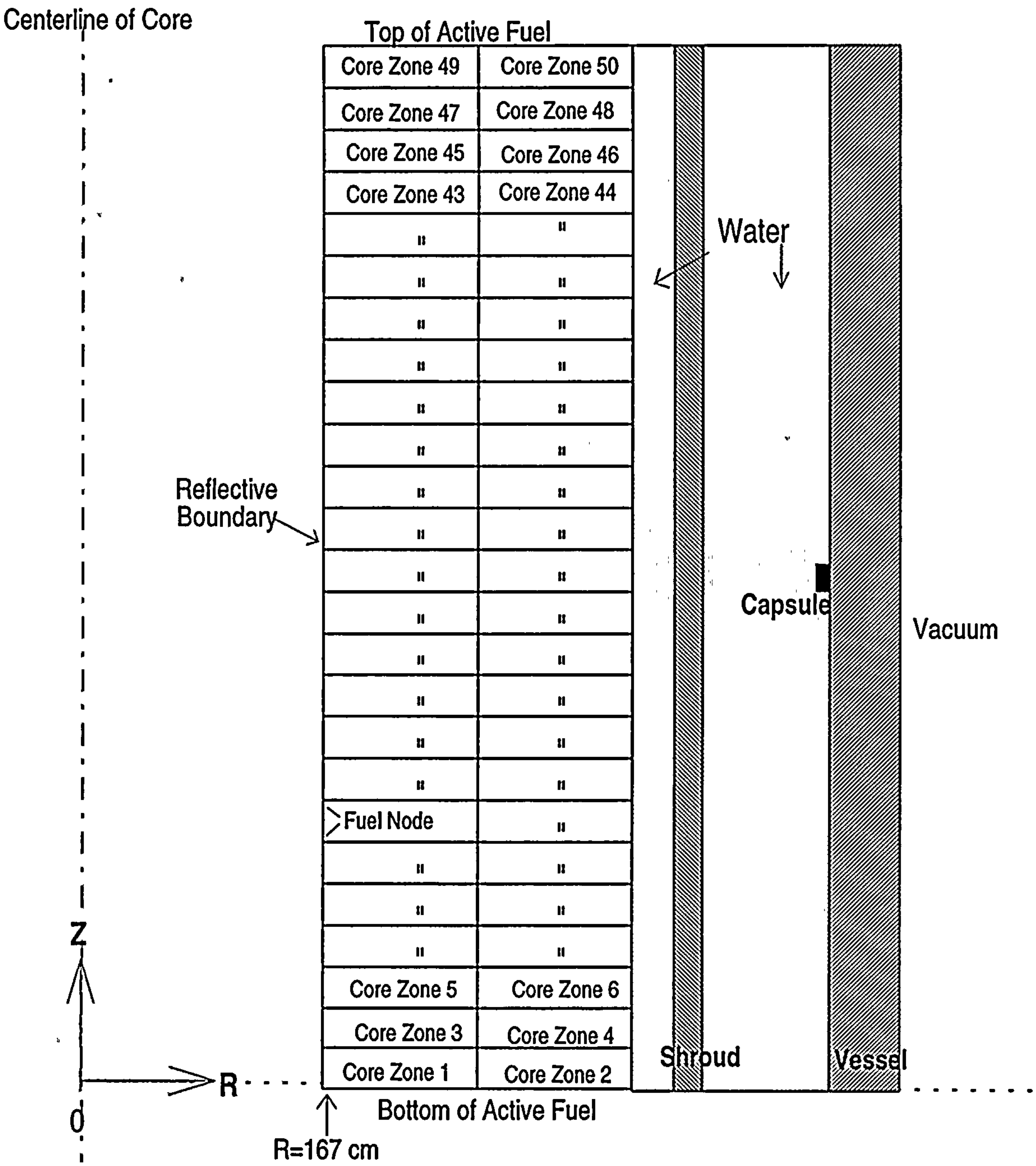


FIGURE 5-3 R-Z CALCULATION MODEL

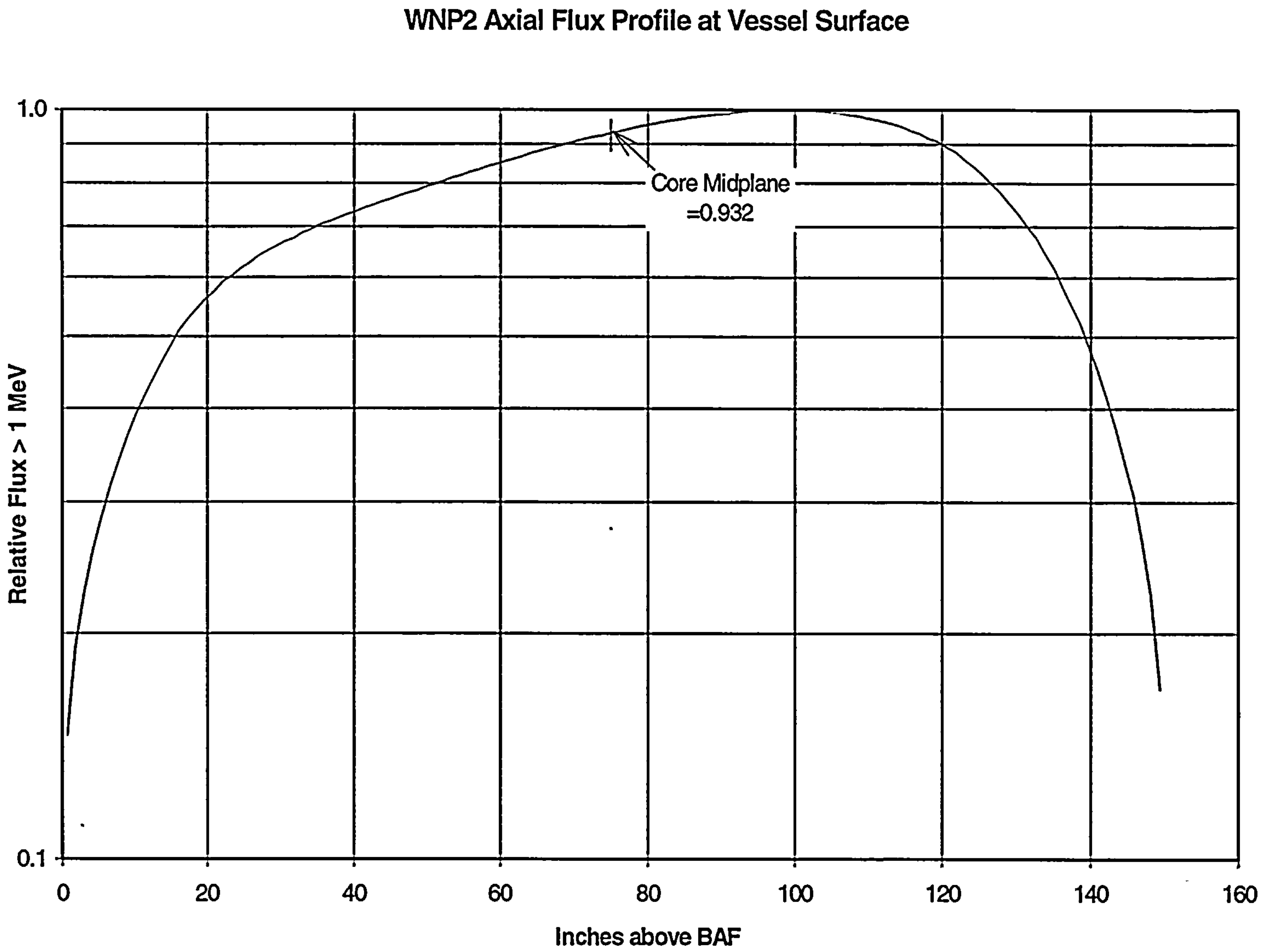


FIGURE 5-4 AXIAL FLUX PROFILE AT VESSEL I.D. SURFACE

6. CHARPY V-NOTCH IMPACT TESTING

The 24 Charpy specimens recovered from the surveillance capsule as well as the 36 unirradiated Charpy specimens were impact tested at temperatures selected to establish the toughness transition and upper shelf of the irradiated RPV materials. Testing was conducted in accordance with ASTM E23-94b [20].

6.1 IMPACT TEST PROCEDURE

The Vallecitos testing machine used for irradiated specimens was a Riehle Model P1-2 impact machine, serial number R-89916. The maximum energy capacity of the machine is 240 ft-lb, which produces a test velocity at impact of 15.44 ft/sec. The test apparatus and operator were qualified using NIST standard reference material specimens. The Standard Reference Materials (SRMs) consist of three sets of specimens which cover the energy range of the apparatus. Each set has a designated failure energy and a standard test temperature. According to ASTM E23-94b, the test apparatus averaged results must reproduce the NIST standard values within an accuracy of ±5% or ±1.0 ft-lb, whichever is greater. The qualification of the Riehle machine and operator is summarized in Table 6-1.

Charpy V-Notch tests were conducted at temperatures between -100°F and 300°F. The cooling fluid used for irradiated specimens tested at 50°F or below was ethyl alcohol. At temperatures between 50°F and 200°F, water was used as the temperature conditioning fluid. The specimens were heated in silicon oil for test temperatures above 200°F. Cooling of the conditioning fluids was done by heat exchange with liquid nitrogen; heating was done by an immersion heater. The bath of fluid was mechanically stirred to maintain uniform temperatures. The fluid temperature was measured with a calibrated thermocouple. After equilibration at the test temperature for at least 5 minutes, the specimens were manually transferred with centering tongs to the Charpy test machine and impacted in less than 5 seconds.

For each Charpy V-Notch specimen the test temperature, energy absorbed, lateral expansion, and percent shear were determined. In addition, for the fractured specimens, photographs were taken of fracture surfaces. Lateral expansion and percent shear were measured. Percent shear was established by first measuring the length and width of the cleavage surface in inches and then determined from Table 2 of ASTM E23-94b.

6.2 IMPACT TEST RESULTS

Twelve Charpy V-Notch specimens each of unirradiated base, weld, and HAZ materials were used to define the baseline toughness transition and upper shelf portions of the fracture toughness curves. They were tested at temperature ranges of -100°F to 300°F. The absorbed energy, lateral expansion, and percent shear data are listed for each material in Table 6-2. Eight Charpy V-Notch specimens each of irradiated base, weld, and HAZ material were similarly tested in the temperatures range of -60°F to 300°F. During the test of an irradiated weld specimen at 50°F, the Charpy energy was inadvertently not recorded and the test was repeated at the same temperature. As shown later, the resulting shift in ΔRT_{NDT} of the weld metal was not affected by the loss of one data point. The absorbed energy, lateral expansion, and percent shear data are listed for each irradiated material in Table 6-3. The key parameters of the index temperatures at Charpy Energy of 30 ft-lb , 50 ft-lb , lateral expansion at 35 mils and the USE of each material are tabulated in Table 6-4. The fracture surface photographs revealing shear area characteristics of each specimen as well as the individual test data are contained in Appendix A.

For each irradiated and unirradiated base plate, weld and HAZ material, the test data of absorbed Charpy energy and the lateral expansion are respectively fit with the hyperbolic tangent function developed by Oldfield for the EPRI Irradiated Steel Handbook[21]

$$Y = A + B * \mathrm{TANH}\,[(T - T_0)/C] \qquad \text{(Eq 6-1)}$$

where Y = impact energy or lateral expansion
T = test temperature, and
A, B, T_0 and C are regression curve fit parameters.

The TANH function is one of the few continuous functions with a shape characteristic of low alloy steel fracture toughness transition curves. The usefulness of this particular relationship is apparent in the physical significance of curve fit parameters: A-B = lower shelf, A+B = upper shelf, T_0 = mid-transition temperature and B/C = slope of the transition temperature region. The resulting transition temperature curves are presented in Figures 6-1 through 6-6.

6.3 COMPARISON OF THE MEASURED AND PREDICTED IRRADIATION SHIFTS

The measured transition temperature shift for the base plate and weld material was compared to the predictions calculated according to RG1.99. The inputs and the calculated shifts are as follows:

	BASE	WELD
Copper	0.11%	0.03%
Nickel	0.49%	0.89%
CF	73	41
Measured Fluence, f (at 7.2 EFPY)	1.55×10^{17} n/cm^2	1.55×10^{17} n/cm^2
Calculated Fluence Factor: $f^{(0.28 - 0.10 \log f)}$	0.147	0.147
Reg. Guide 1.99 ΔRT_{NDT}	11°F	6°F
Reg. Guide 1.99 $\Delta RT_{NDT} + 2\sigma_\Delta$	-23 to 45 °F	-50 to 62 °F
Measured shift at 30 ft-lb	**-1.0 °F**	**-4.9 °F**

The weight percents of Cu and Ni are based on the average value of GE chemical composition measurements shown in Table 4-1. The CF shown above is the chemistry factor from RG1.99 Tables 1 and 2. The fluence factor is provided by RG1.99 as:

$$\text{fluence factor} = f^{(0.28 - 0.10 \log f)} \qquad \text{(Eq 6-2)}$$

which can either be calculated or read from Figure 1 of RG1.99. The predicted 30 ft-lb temperature shift (ΔRT_{ndt}) was also calculated according to the following RG1.99 equation

$$\Delta RT_{NDT} = (CF)\, f^{(0.28 - 0.10 \log f)} \qquad \text{(Eq 6-3)}$$

The measured 30 ft-lb temperature shifts of -1.0°F for the base plate and -4.9°F for the weld (Table 6-4) are well within the bounds of the RG prediction with the uncertainty of $\pm 2\sigma$. (σ is the square root of the sum of $\sigma_I^2 + \sigma_\Delta^2$ where σ_I = Standard deviation on initial RT_{NDT}, which is zero and σ_Δ = Standard deviation on ΔRT_{NDT}, which is 28°F for welds and 17°F for base material.

6.4 CHANGE IN USE

Based on the copper content and the fluence data provided in the previous sections, RG1.99 predicts decreases in the irradiated Charpy USE of approximately 8% for the base, and 6% for the weld material at the fluence of 1.55×10^{17} n/cm^2. The measured USE values for these materials however show increases of 3% and 12%, respectively for the base and weld material.

Upper Shelf Energy is expected to decrease due to irradiation. The amount of expected decrease is related to both copper content and fluence, which is relatively low (about an order of magnitude less than in PWRs) in BWRs. Both the copper content in the WNP2 vessel (0.11%) and the fluence (7.57×10^{17} n/cm^2) are relatively low, and therefore, materials may not experience significant decreases in USE. Experience has shown that, at relatively low fluence and considering typical scatter in Charpy data, BWR vessel material USE test results may show an increase. Given the typical scatter in Charpy data and the low fluence of the irradiated specimens, the increase in WNP2 plate and weld material is not unexpected.

TABLE 6-1 VALLECITOS QUALIFICATION TEST RESULTS USING NIST STANDARD REFERENCE SPECIMENS

	Specimen Identification	Bath Medium	Test Temperature (°F)	Energy Absorbed (ft-lb)	Acceptable Range (ft-lb)
Vallecitos Riehle Machine (tested 8/95)	HH-46 1	Ethyl Alcohol	-40	74	
	HH-46 2	Ethyl Alcohol	-40	72.5	
	HH-46 3	Ethyl Alcohol	-40	75.5	
	HH-46 4	Ethyl Alcohol	-40	73	
	HH-46 5	Ethyl Alcohol	-40	77	
			Average	74.4	74.3 ± 3.7 pass
	LL-45 1	Ethyl Alcohol	-40	13	
	LL-45 2	Ethyl Alcohol	-40	13	
	LL-45 3	Ethyl Alcohol	-40	13	
	LL-45 4	Ethyl Alcohol	-40	13	
	LL-45 5	Ethyl Alcohol	-40	13	
			Average	13	12.8 ± 1.0 pass
	SH-5 1	Ethyl Alcohol	70	170	
	SH-5 2	Ethyl Alcohol	70	170	
	SH-5 3	Ethyl Alcohol	70	162.5	
	SH-5 4	Ethyl Alcohol	70	161.5	
	SH-5 5	Ethyl Alcohol	70	156	
			Average	164.0	164.1± 8.2 pass

TABLE 6-2 UNIRRADIATED CHARPY V-NOTCH IMPACT TEST RESULTS

	Specimen Identification	Bath Medium	Test Temperature (°F)	Fracture Energy (ft-lb)	Lateral Expansion (mils)	Percent Shear (%)
Base: Heat B5301-1, Longitudinal,	B1	ETOH	-60	5.0	11	1
	B2	ETOH	-20	10.5	12	20
	B3	ETOH	10	17.0	20	34
	B4	ETOH	20	38.0	35	34
	B5	ETOH	30	27.0	26	37
	B6	ETOH	40	41.0	37	38
	B7	ETOH	60	77.5	63	45
	B8	WATER	80	99.0	81	63
	B9	WATER	108	113.0	86	69
	B10	WATER	120	130.5	83	80
	B11	WATER	200	148.0	90	100
	B12	OIL	300	153.5	91	100
Weld: Heat 3P4966 Lot 1214 Linde 124 Flux	W1	ETOH	-100	7.0	9	12
	W2	ETOH	-60	9.0	14	14
	W3	ETOH	-20	25.5	30	42
	W4	ETOH	-10	52.0	48	48
	W5	ETOH	0	42.0	40	37
	W6	ETOH	10	36.0	43	55
	W7	ETOH	20	53.0	52	69
	W8	ETOH	60	74.5	70	87
	W9	WATER	108	89.0	76	97
	W10	WATER	120	98.5	76	100
	W11	WATER	200	107.0	67	100
	W12	OIL	300	97.0	91	100
HAZ: Heat B5301-1, Longitudinal and Weld heat 3P4966 Lot 1214 Linde 124 Flux	HAZ1	ETOH	-60	28.0	17	20
	HAZ2	ETOH	-50	16.0	16	18
	HAZ3	ETOH	-20	48.0	37	37
	HAZ4	ETOH	-10	34.0	35	41
	HAZ5	ETOH	0	44.5	44	70
	HAZ6	ETOH	20	83.0	62	65
	HAZ7	ETOH	60	103.5	81	88
	HAZ8	WATER	80	74.5	73	90
	HAZ9	WATER	108	110.5	86	88
	HAZ10	WATER	120	118.0	84	100
	HAZ11	WATER	200	121.5	76	100
	HAZ12	OIL	300	118.0	89	100

TABLE 6-3 IRRADIATED CHARPY V-NOTCH IMPACT TEST RESULTS (for 300° Azimuth Surveillance Capsule at 7.2 EFPY)

	Specimen Identification	Bath Medium	Test Tempera-ture (°F)	Fracture Energy (ft-lb)	Lateral Expansion (mils)	Percent Shear (%)
Base: Heat B5301-1, Longitudinal,	29146	ETOH	0	16.0	16	22
	29141	ETOH	30	30.0	26	30
	29143	ETOH	39	56.5	36	30
	29145	ETOH	67	66.0	50	35
	29140	WATER	100	109.0	73	56
	29144	WATER	150	139.5	87	70
	29139	WATER	200	154.0	85	96
	29142	OIL	300	152.5	91	100
Weld: Heat 3P4966 Lot 1214 Linde 124 Flux	29148	ETOH	-20	43.5	36	31
	29153	ETOH	-10	35.0	36	34
	29152	ETOH	22	50.0	48	54
	29147	ETOH	50	N/A	71	96
	29157	ETOH	50	72.0	66	67
	29151	WATER	85	102.5	87	94
	29149	WATER	100	104.0	76	91
	29150	WATER	200	108.0	76	100
HAZ: Heat B5301-1, Longitudinal and Weld heat 3P4966 Lot 1214 Linde 124 Flux	29157	ETOH	-60	7.0	8	1
	29159	ETOH	-40	28.0	26	35
	29158	ETOH	-20	58.5	43	39
	29162	ETOH	8	62.0	45	43
	29156	ETOH	67	88.0	61	76
	29160	WATER	105	105.5	77	91
	29155	WATER	130	127.0	82	100
	29161	WATER	200	127.0	81	100

TABLE 6-4 KEY PARAMETER SUMMARY OF IRRADIATED AND UNIRRADIATED CHARPY V-NOTCH IMPACT DATA (for 300° Azimuth Surveillance Capsule at 7.2 EFPY)

Material	Index Temp (°F) E=30 ft-lb	Index Temp (°F) E=50 ft-lb	Index Temp (°F) MLE=35 mil	USE (ft-lb)
PLATE (B5301-1):				
Unirradiated	22.2	43.7	31.4	150.2
Irradiated	21.2	45.8	41.6	154.8
Difference	-1.0 (↓)	2.1 (↑)	10.2 (↑)	4.6 (+3%↑)
RG1.99, ΔRT_{NDT} = 11 °F				
RG1.99, ($\Delta RT_{NDT} \pm 2\sigma$) = -23 °F to 45 °F				
RG1.99, Decrease in USE = 8%				
WELD (3P4966) :				
Unirradiated	-19.6	15.8	-14.8	102.5
Irradiated	-24.5	9.4	-13.1	114.6
Difference	- 4.9 (↓)	- 6.4 (↓)	1.7 (↑)	12.1 (+12%↑)
RG1.99, ΔRT_{NDT} = 6 °F				
RG1.99, ($\Delta RT_{NDT} \pm 2\sigma$) = -50 °F to 62 °F				
RG1.99, Decrease in USE = 6%				
HAZ:				
Unirradiated	-37.5	-3.8	-16.9	119.8
Irradiated	-43.0	-6.0	-15.3	129.9
Difference	- 5.5 (↓)	- 2.2 (↓)	1.6 (↑)	10.1 (+8%↑)

FIGURE 6-1 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 Base Plate Impact Energy

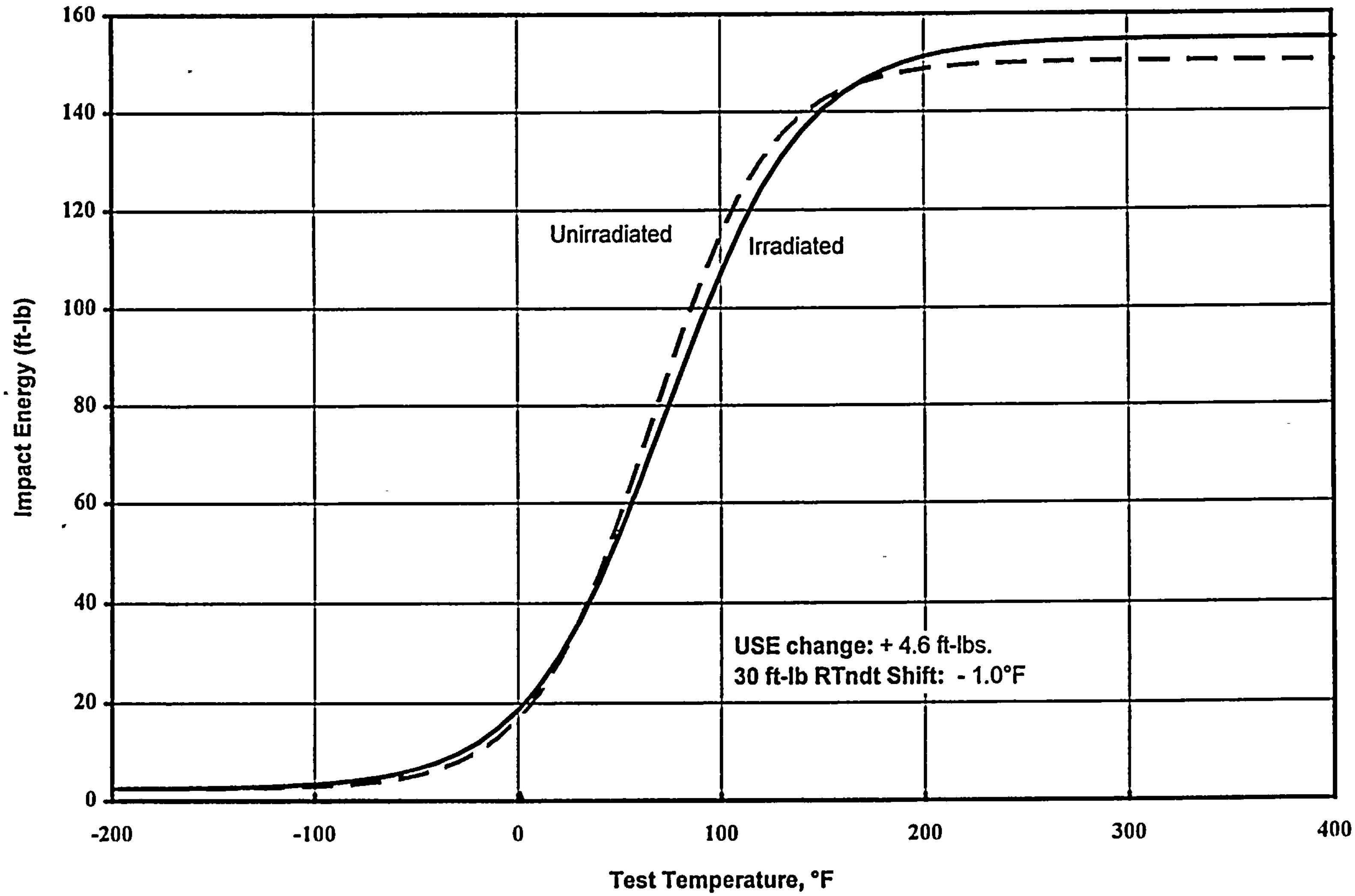


FIGURE 6-2 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 Base Plate Lateral Expansion

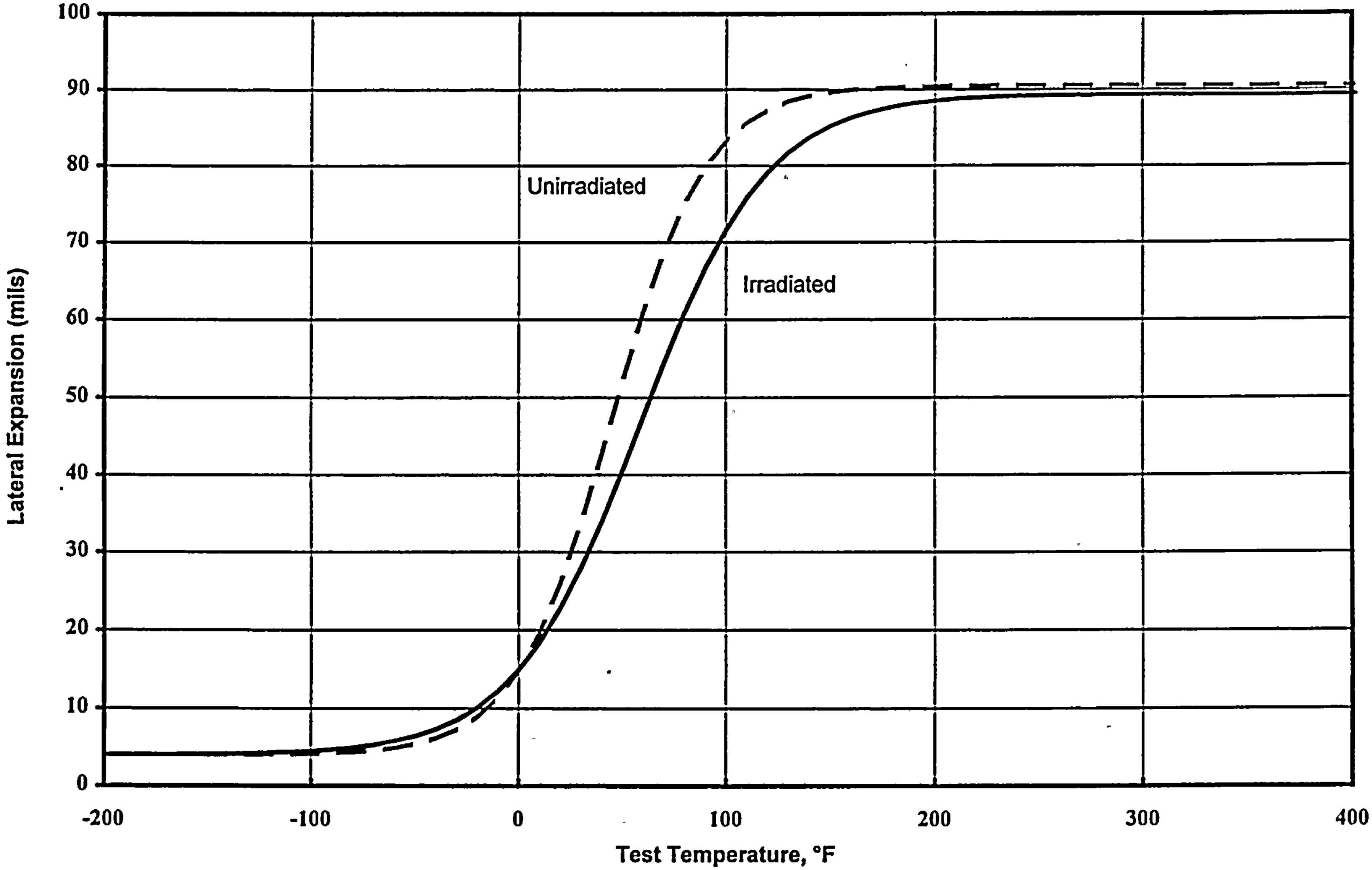


FIGURE 6-3 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 Weld Impact Energy

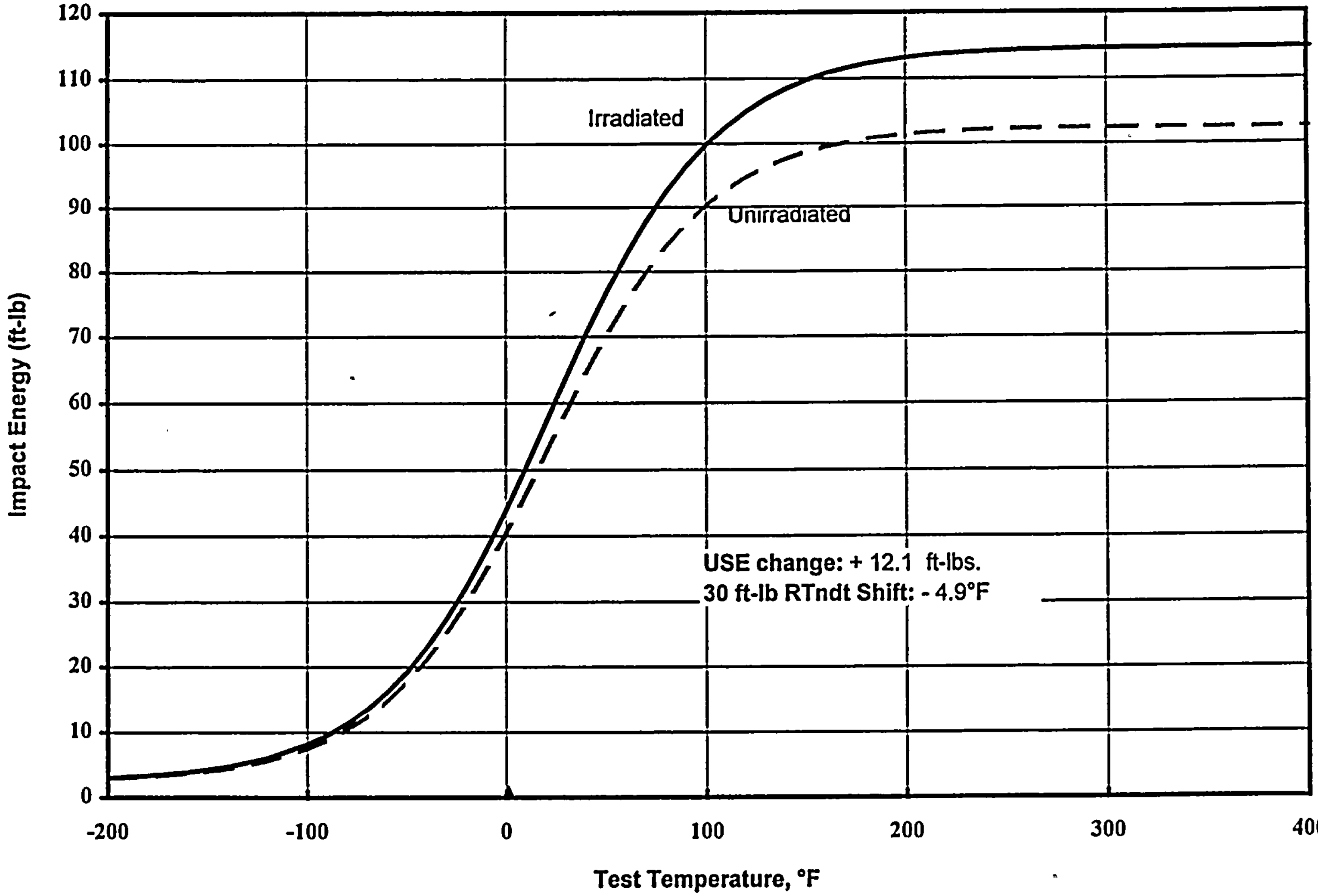


FIGURE 6-4 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 Weld Lateral Expansion

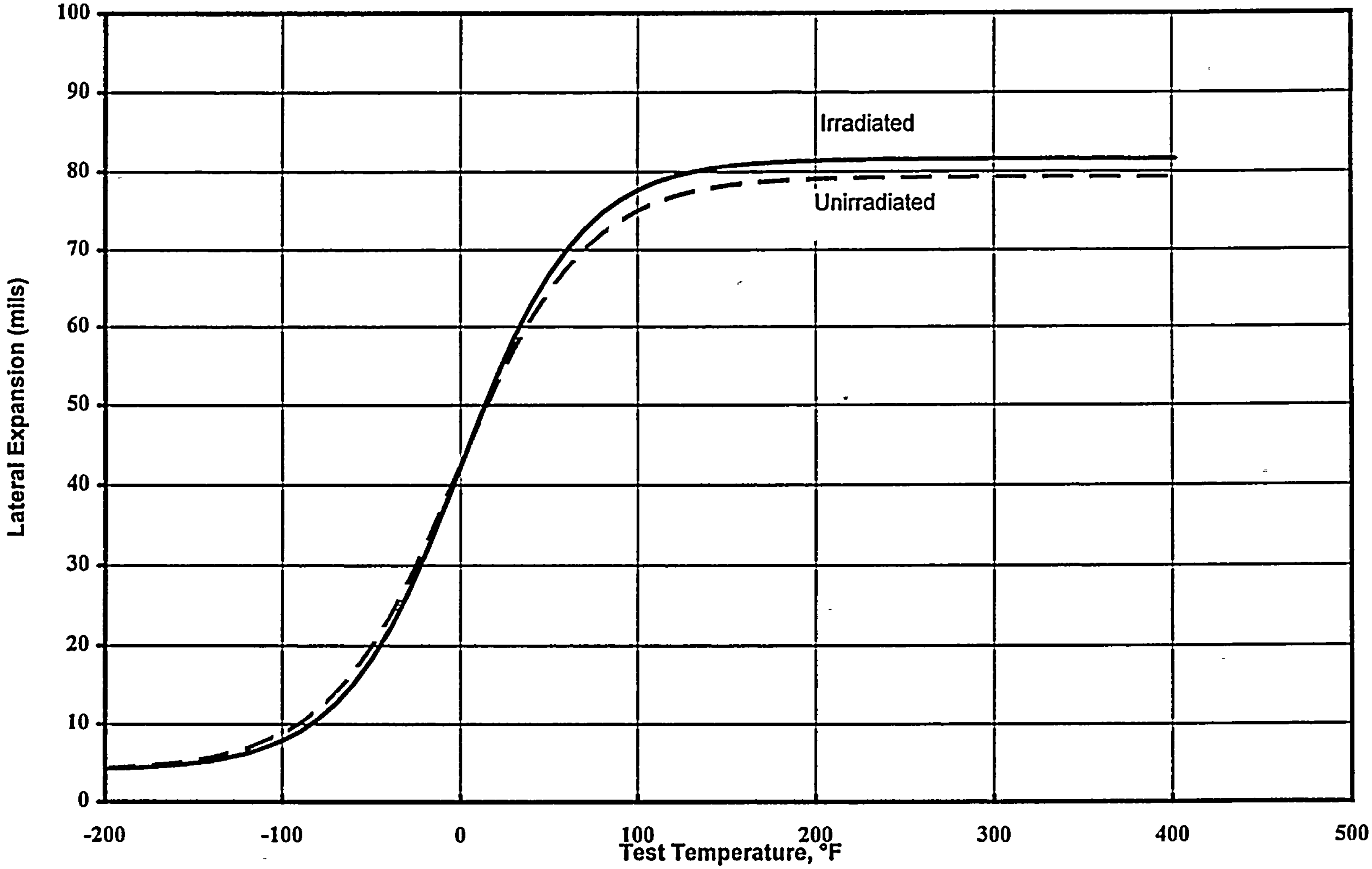


FIGURE 6-5 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 HAZ Impact Energy

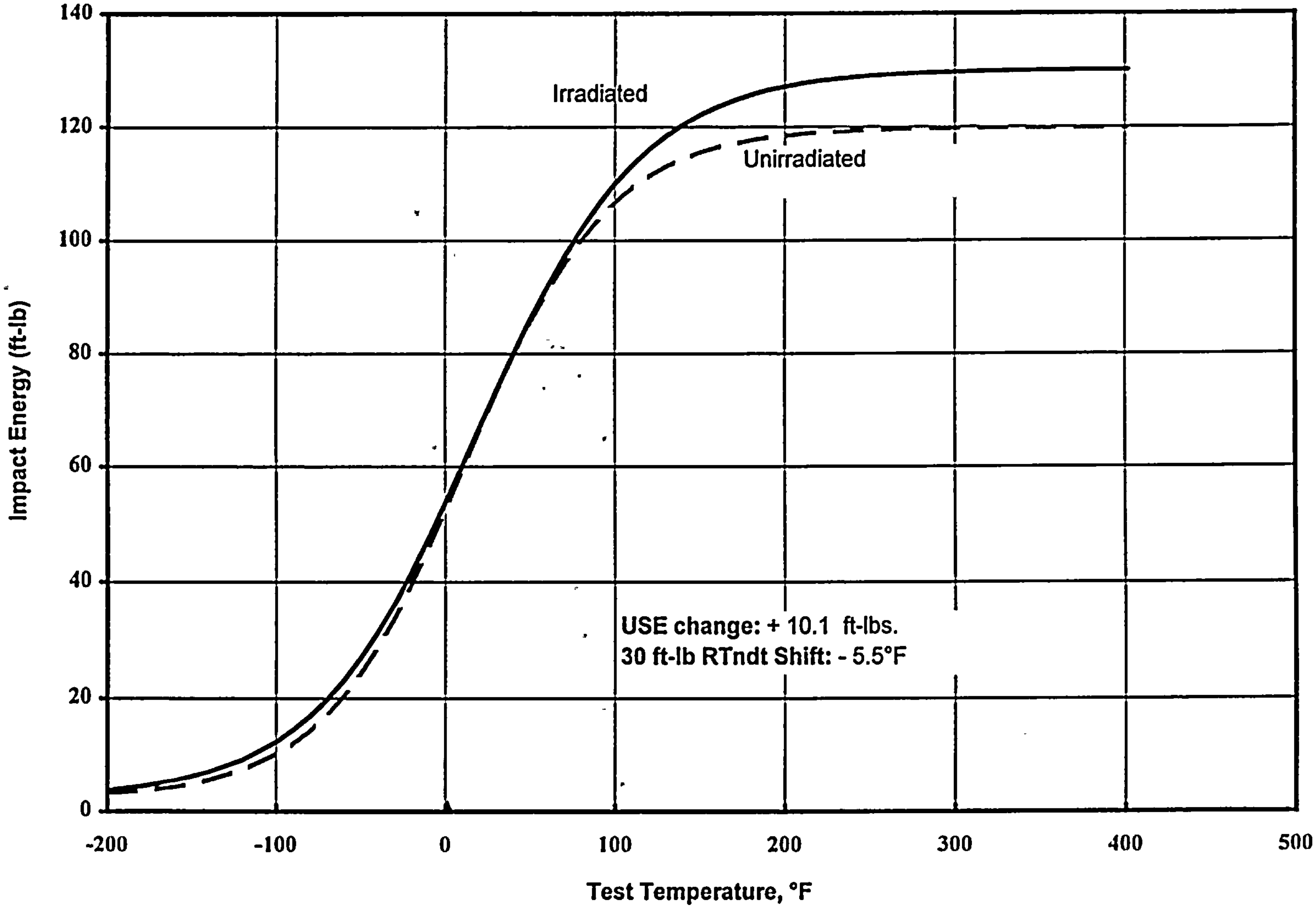
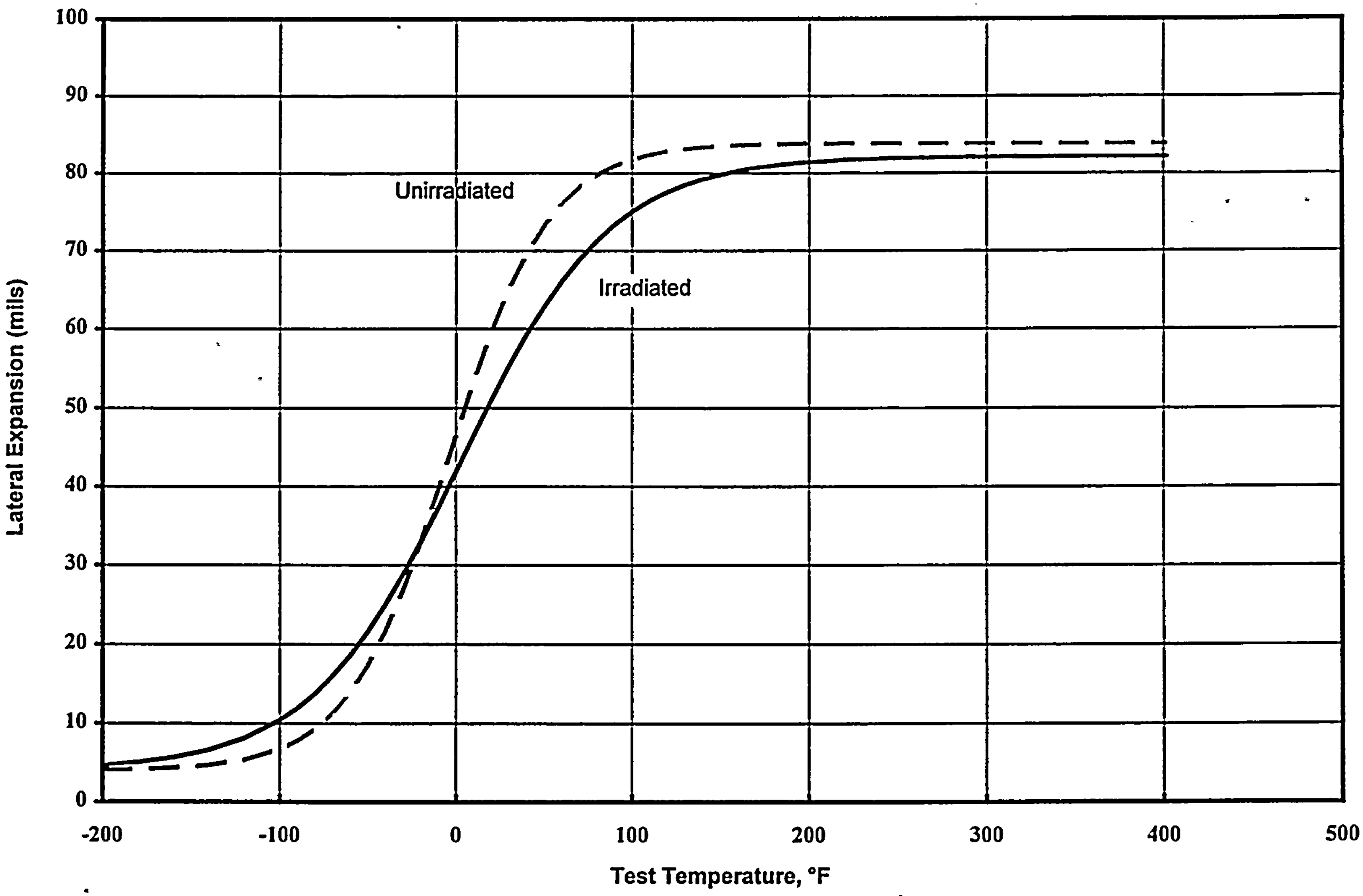


FIGURE 6-6 IRRADIATED AND UNIRRADIATED CHARPY TESTS
WNP2 HAZ Lateral Expansion



7. TENSILE TESTING

Eight round bar tensile specimens were recovered from the surveillance capsule. Uniaxial tensile tests were conducted in air at room temperature (70°F) at RPV operating temperature (550°F) and at an intermediate temperature of 150°F for the two additional base and weld specimens. The tests were conducted in accordance with ASTM E8-95[22].

7.1 PROCEDURE

All tests were conducted using a screw-driven Instron test frame equipped with a 20-kip load cell and special pull bars and grips. Heating was done with a Satec resistance clamshell furnace centered around the specimen load train. The test temperature was monitored by a chromel-alumel thermocouple spot-welded to an Inconel clip that was friction-clipped to the surface of the specimen at its midline.

All tests were conducted at a calibrated crosshead speed of 0.005 in/min until well past yield, at which time the speed was increased to 0.05 inch/min until fracture. Crosshead displacement was used to monitor specimen extension during the test.

The test specimens were machined with a minimum nominal diameter of 0.250 inch at the center of the gage length. The yield strength (YS) and ultimate tensile strength (UTS) were calculated by dividing the measured area into the 0.2% offset load and into the maximum test load, respectively. The values listed for the uniform and total elongation were obtained from plots that recorded load versus specimen extension and are based on a 1.5 inch nominal gage length. Reduction of area (RA) values were determined from post-test measurements of the necked specimen diameters using a calibrated blade micrometer and employing the following formula:

$$RA = 100\% * (A_0 - A_f)/A_0 \qquad \text{(Eq 7-1)}$$

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

7.2 RESULTS

Irradiated tensile test properties of Yield Strength (YS), Ultimate Tensile Strength (UTS), Reduction of Area (RA), Uniform Elongation (UE), and Total Elongation (TE) are presented in Table 7-1. A stress-strain curve for a 550°F base metal irradiated specimen is shown in Figure 7-1. This curve is typical of the stress-strain characteristics of all the tested specimens.

The surveillance materials generally follow the trend of decreasing properties with increasing temperature. Photographs of the necking behavior and the fracture surfaces are provided in Appendix B. As expected the data shows the general trend of increase in YS and UTS and decrease in TE with irradiation.

TABLE 7-1 TENSILE TEST RESULTS FOR IRRADIATED AND UNIRRADIATED RPV MATERIALS
(for 300° Azimuth Surveillance Capsule at 7.2 EFPY)

	Specimen Number	Test Temp. (°F)	Yield[a] Strength (ksi)	Ultimate Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction of Area (%)
Irradiated Base:	P1-A	70	59.6	84.3	12.5	21.6	72.7
	P1-B	550	55.7	79.7	10.9	18.4	70.4
Irradiated Weld:	P2-A	70	67.3	83.9	12.8	22.3	68.4
	P2-B	550	59.0	78.0	12.4	18.6	63.8
Irradiated HAZ:	P3-A	70	56.8	81.3	12.2	21.1	71.5
	P3-B	550	60.3	80.7	11.5	18.1	65.9
Unirradiated Base:	P1-A	70	56.5	80.4	15.2	26.1	73.8
	P1-B	550	53.4	77.6	12.2	19.7	71.2
Unirradiated Weld:	P2-A	70	61.4	80.2	13.2	22.1	68.2
	P2-B	550	58.3	77.0	12.7	21.1	65.3
Unirradiated HAZ:	P3-A	70	53.4	78.2	12.5	20.8	72.9
	P3-B	550	58.1	78.2	10.7	18.3	66.1

[a] Yield Strength is determined by 0.2% offset.

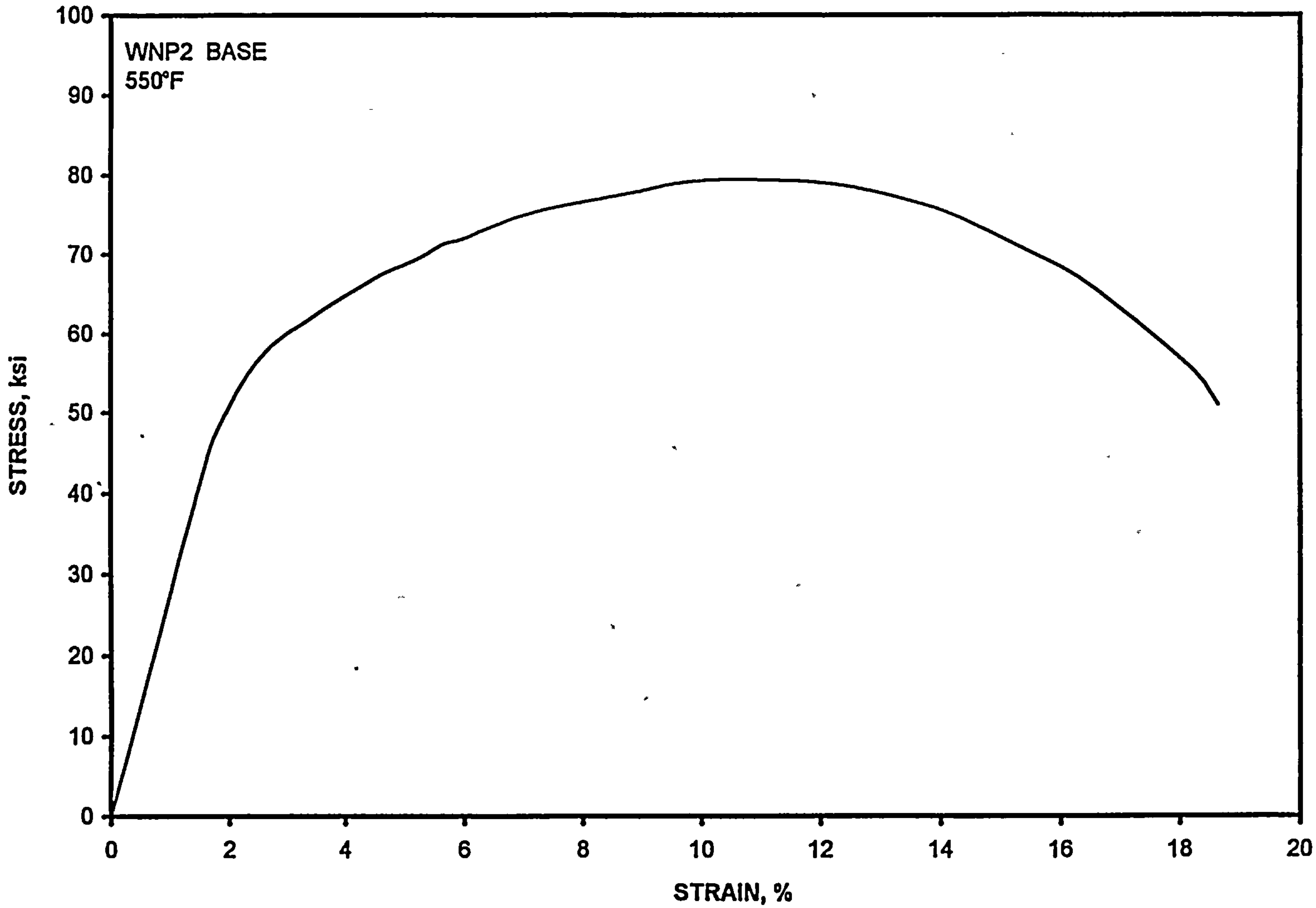


FIGURE 7-1 TYPICAL ENGINEERING STRESS-STRAIN FOR IRRADIATED RPV MATERIALS

8. ADJUSTED REFERENCE TEMPERATURE AND UPPER SHELF ENERGY

The 32 EFPY peak fluence value of 7.57×10^{17} n/cm^2 in Section 5.3 is used to calculate the 32 EFPY 1/4 T fluence values of 3.86×10^{17} n/cm^2 and 5.14×10^{17} n/cm^2 for the lower shell and lower intermediate shell, respectively. It should be noted that the value of 3.86×10^{17} n/cm^2 for the lower shell represents 80% of the vessel peak fluence as discussed in Section 5.3. Based on these fluence values, the adjusted reference temperatures (ARTs) and upper shelf energy (USE) for the beltline materials are calculated.

8.1 ADJUSTED REFERENCE TEMPERATURE AT 32 EFPY

The effect on adjusted reference temperature (ART) due to irradiation in the beltline materials is determined according to the methods in RG1.99, as a function of neutron fluence and the elemental compositions of copper (Cu) and nickel (Ni). The RG1.99 ART equation is:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad \text{(Eq 8-1)}$$

where $$\Delta RT_{NDT} = CF \times f^{(0.28-0.10 \log f)} \qquad \text{(Eq 8-2)}$$

$$\text{Margin} = 2\,(\sigma_I^2 + \sigma_\Delta^2)^{0.5} \qquad \text{(Eq 8-3)}$$

CF = Chemistry factor from Tables 1 or 2 of RG1.99,
f = 1/4 T fluence (n/cm^2) divided by 10^{19}
σ_I = Standard deviation on initial RT_{NDT} and is equal to zero in this case.
σ_Δ = Standard deviation on ΔRT_{NDT}, is 28°F for welds and 17°F for base material, except that σ_Δ need not exceed 0.50 times the ΔRT_{NDT} value.

Since the chemical composition measured by GE as described in Section 4 can be considered as an addition to the existing data, the chemistry used to calculate the ART is based on the average of the GE data and those reported in the FSAR. Therefore the chemistry for plate B5301-1 is 0.13%Cu and 0.49%Ni, which has a chemistry factor of 88. The chemistry for weld 3P4966 is 0.03%Cu and 0.90%Ni, which has a corresponding chemistry factor of 41. The composition for the remaining materials remains unchanged, same as those provided in Section 4.

Each beltline plate and weld ΔRT_{NDT} value is determined by multiplying the CF from RG1.99 determined for the Cu-Ni content of the material, by the fluence factor for the EFPY being

evaluated. The Initial RT_{NDT}, Δ RT_{NDT} and Margin are added to get the ART of the material. The 32 EFPY ART values for the beltline plate and weld are shown in Tables 8-1 and 8-2 respectively. It should be noted that the WNP2 FSAR requires that, in order to satisfy the 10CFR50 Appendix G, initial RT_{NDT} is either NDT or transverse Charpy V-notch (CVN) 50 ft-lb temperature minus 60°F. Since all the tests performed were from specimens with longitudinal orientation, the transverse CVN 50 ft-lb transition temperature has to be calculated from the existing data of longitudinal orientation in the following manner. The lowest longitudinal CVN energy, if below 50 ft-lb, is adjusted to derive a longitudinal CVN 50 ft-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equals or exceeds 50 ft-lb, the test temperature is used. Once the longitudinal 50 ft-lb temperature is derived, an additional 30°F is added to account for orientation changed from longitudinal 50 ft-lb to transverse 50 ft-lb. For the vessel weld metal, the 30°F addition is omitted, since there is no principal working direction in weld metal. The initial RT_{NDT} for plate materials shown in Tables 8-1 are adjusted for the transverse orientation as required by the ASME Code after Summer 1972.

8.2 UPPER SHELF ENERGY AT 32 EFPY

Paragraph IV.B of 10CFR50 Appendix G sets limits on the upper shelf energy of the beltline materials. The USE must be above 50 ft-lb at all times during plant operation; assumed here to be up to 32 EFPY. However, the WNP2 vessel materials did not have USE data taken at fabrication, the calculation of 32 EFPY USE provided in Table 8-3 is for the beltline materials represented by the surveillance specimens only. The equivalent transverse USE of the plate material shown in the table is taken as 65% of the longitudinal USE, according to USNRC MTEB 5-2[23]. Unlike the plate, the weld metal USE has no transverse/longitudinal correction because weld metal has no orientation effect.

Since the USE evaluation in Table 8-3 is limited to the surveillance materials and in order to show that the remaining beltline materials also satisfy the 10CFR50 Appendix G 50 ft-lb limit, WNP2 has been evaluated against the BWR Owners' Group Equivalent Margin Analyses. From Tables 3-1(a) & (b), the limiting copper content of the beltline plate and weld materials are 0.15% and 0.1%, respectively. Based on this, the RG1.99 Figure 2 predicts a decrease in the USE approximately by 12% for both the plate and the weld at the 32 EFPY 1/4 T fluence of 5.14×10^{17} n/cm^2. The Equivalent Margin Analysis shows that the USE decreases are bounded by the allowed limits of 21% and 34%, respectively for the plate and weld materials. Thus, the analysis demonstrates that the 10 CFR 50, Appendix G safety requirements are satisfactorily met

for WNP2. The Owners' Group Program Report[24] was submitted to the NRC in December 1993 and approved by SER on December 8, 1993.

TABLE 8-1 BELTLINE BASE PLATE ART FOR 32 EFPY AT 1/4 T

Lower (No.1) Shell
Plate Thickness = 7.5 inches

Lower Intermediate (No.2) Shell
Plate Thickness = 6.44 inches

Girth Weld
Weld Thickness= 6.44 inches

Lower (No.1) Shell
32 EFPY with Peak I.D. fluence = 7.57E+17 n/cm^2
32 EFPY with Adjusted 1/4 T fluence = 3.86E+17 n/cm^2 [1]

Lower Intermediate (No.2) Shell
32 EFPY with Peak I.D. fluence = 7.57E+17 n/cm^2
32 EFPY with Peak 1/4 T fluence = 5.14E+17 n/cm^2

Girth Weld
32 EFPY Peak 1/4 T Weld fluence = 5.14E+17 n/cm^2

COMPONENT	HEAT No.	%Cu	%Ni	CF	Fluence Factor	Initial RTndt °F	32 EFPY ΔRTndt °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
Lower Shell with Adjusted Fluence [1]												
21-1-1	C1272-1	0.15	0.60	110	0.25	28	27.9	0	14.0	27.9	55.8	83.8
21-1-2	C1273-1	0.14	0.60	100	0.25	20	25.4	0	12.7	25.4	50.8	70.8
21-1-3	C1273-2	0.14	0.60	100	0.25	4	25.4	0	12.7	25.4	50.8	54.8
21-1-4	C1272-2	0.15	0.60	110	0.25	0	27.9	0	14.0	27.9	55.8	55.8
Lower-Interm. Shell Peak Fluence												
21-1-1	B5301-1 [2]	0.13	0.49	88	0.30	-20	26.2	0	13.1	26.2	52.3	32.3
22-1-2	C1336-1	0.13	0.5	88	0.30	-8	26.2	0	13.1	26.2	52.3	44.3
22-1-3	C1337-1	0.15	0.51	105	0.30	-20	31.2	0	15.6	31.2	62.5	42.5
22-1-4	C1337-2	0.15	0.51	105	0.30	-20	31.2	0	15.6	31.2	62.5	42.5

Notes:
[1] Lower shell fluence adjusted to 80% of the peak value to consider axial flux distribution (see Figure 5-4).
[2] Wt% of Cu and Ni is the average of the data measured by GE and those listed in TABLE 3-1(a)

TABLE 8-2 BELTLINE WELD ART FOR 32 EFPY AT 1/4 T

Lower (No.1) Shell
Plate Thickness = 7.5 inches

Lower Intermediate (No.2) Shell
Plate Thickness = 6.44 inches

Girth Weld
Weld Thickness= 6.44 inches

Lower (No.1) Shell
32 EFPY with Peak I.D. fluence = 7.57E+17 n/cm^2
32 EFPY with Peak 1/4 T fluence = 3.86E+17 n/cm^2 [1]

Lower Intermediate (No.2) Shell
32 EFPY with Peak I.D. fluence = 7.57E+17 n/cm^2
32 EFPY with Peak 1/4 T fluence = 5.14E+17 n/cm^2

Girth Weld
32 EFPY Peak 1/4 T Weld fluence = 5.14E+17 n/cm^2

COMPONENT	HEAT No.	%Cu	%Ni	CF	Fluence Factor	Initial RTndt °F	32 EFPY ΔRTndt °F	σ_I	σ_Δ	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
Lower-Long. BA,BB,BD	04P046	0.06	0.90	82	0.25	-48	20.8	0	10.4	20.8	41.6	-6.4
Lower-Long. BA, BB	07L669	0.03	1.02	41	0.25	-50	10.4	0	5.2	10.4	20.8	-29.2
Lower-Long. BA-BD	3P4966 [2]	0.025	0.895	34	0.25	-30	8.6	0	4.3	8.6	17.3	-12.7
Lower-Long. BA-BD	3P4966 [2]	0.025	0.895	34	0.25	-48	8.6	0	4.3	8.6	17.3	-30.7
Lower-Long. BB,BC,BD	C3L46C	0.02	0.87	27	0.25	-20	6.9	0	3.4	6.9	13.7	-6.3
Lower-Long. BB	08M365	0.02	1.10	27	0.25	-48	6.9	0	3.4	6.9	13.7	-34.3
Lower-Long. BC	09L853	0.03	0.86	41	0.25	-50	10.4	0	5.2	10.4	20.8	-29.2
Lower-Lower Int. BE-BH	3P4966 [2]	0.025	0.885	34	0.30	-26	10.1	0	5.1	10.1	20.2	-5.8
Lower-Lower Int. BE-BH	3P4966 [2]	0.025	0.895	34	0.30	-6	10.1	0	5.1	10.1	20.2	14.2
Long. BF,BH	04P046	0.06	0.90	82	0.30	-48	24.4	0	12.2	24.4	48.7	0.7
Long. BF	05P018	0.09	0.90	122	0.30	-38	36.3	0	18.1	36.3	72.5	34.5
Long. BG	624063	0.03	1.00	41	0.30	-50	12.2	0	6.1	12.2	24.4	-25.6
Long. BH	624039	0.07	1.01	95	0.30	-50	28.2	0	14.1	28.2	56.5	6.5
Long. BG	624039	0.10	0.92	135	0.30	-36	40.1	0	20.1	40.1	80.2	44.2
Lower - Lower Int. AB	492L4871	0.03	0.98	41	0.30	-50	12.2	0	6.1	12.2	24.4	-25.6
Lower - Lower Int. AB	5P6756	0.08	0.93	122	0.30	-50	36.3	0	18.1	36.3	72.5	22.5
Lower - Lower Int. AB	5P6756	0.09	0.92	122	0.30	-50	36.3	0	18.1	36.3	72.5	22.5
Girth Weld AB	3P4955	0.025	0.90	34	0.30	-44	10.1	0	5.1	10.1	20.2	-23.8
Girth Weld AB	3P4955	0.023	0.95	34	0.30	-16	10.1	0	5.1	10.1	20.2	4.2
Girth Weld AB	04T931	0.03	1.00	41	0.30	-50	12.2	0	6.1	12.2	24.4	-25.6

Notes:

[1] Lower shell fluence adjusted to 80% of the peak value to consider axial flux distribution (see Figure 5-4)

[2] The data provided by GE shown in Table 4-1 are considered here as additional data points for the chemical composition provided in Table 3-1(b). Therefore for 3P4966 weld, the average value of GE and Table 3-1(b) data is used.

TABLE 8-3 UPPER SHELF ENERGY ANALYSIS FOR WNP2 BELTLINE MATERIALS REPRESENTED BY THE SURVEILLANCE SPECIMENS

Location (Heat)	Initial Long. USE (ft-lb)	Initial Trans. USE (ft-lb)	%Cu	% Decrease in USE [1]	32 EFPY Trans. USE
Base Plate 22-1-1 (B5301-1)	150	98	0.11%	10%	88
Weld Lower-Int BE-BH (3P4966)	--	103	0.03%	9.5%	93

Note: [1] Based on the peak 1/4T fluence of 5.14×10^{17} n/cm^2. The percentages were estimated from Figure 2 of RG1.99. The curve for 0.05%Cu was used for the weld.

9. REFERENCES

[1] Fracture Toughness Requirements, Appendix G to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.

[2] Protection Against Non-Ductile Failure, Appendix G to Section XI of the 1989 ASME Boiler & Pressure Vessel Code.

[3] WNP2 Final Safety Analysis Report, with amendments, June 1983

[4] Reactor Vessel Material Surveillance Program Requirements, Appendix H to Part 50 of Title 10 of the Code of Federal Regulations, December 1995.

[5] Surveillance Test for Nuclear Reactor Vessels, Annual Book of ASTM Standards, E185-82, March 1982.

[6] Radiation Embrittlement of Reactor Vessel Materials, USNRC Regulatory Guide 1.99, Revision 2, May 1988.

[7] CBIN, Surveillance test, VPF 3133-446 to VPF 3133-453, Fabrication Report P.O. #205-AE023, Contract: #72-2647

[8] Surveillance Samples for Reactor Pressure Vessel, GE Document 21A8731, Rev.1

[9] Hanford 2 Fabrication Report - Manufactures Data reports and Vessel Certifications, P.O. No. 205-AE023, Contract: #72-2647 (December 1974 & July 1981)

[10] WNP2 Power Uprate Project NSSS Engineering Report, GE-NE-208-17-0993, Rev.1

[11] T. A. Caine to D.M. Kelly, WNP-2 109.2% Power Uprate Impact On Vessel Fracture Toughness, March 31, 1993

[12] T. A. Caine to D.M. Kelly, WNP-2 109.2% Power Uprate Impact On Vessel Fracture Toughness, May 5, 1993.

[13] G.C. Martin, Fast Neutron Cross Section Determination for BWRs using Neutron Dosimeters, November 11, 1993 (FMT Transmittal 93-212-0045)

[14] Martin, G.C., Browns Ferry Unit 3 In-Vessel Neutron Spectral Analysis, GENE, San Jose, CA, August 1980, (GE Report NEDO-24793).

[15] Standard Test Methods for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron, Annual Book of ASTM Standards, E263-93.

[16] Standard Test Methods for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel, Annual Book of ASTM Standards, E264-93.

[17] Standard Test Methods for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper, Annual Book of ASTM Standards, E523-93.

[18] T. A. Caine, Flux Wire Dosimeter Evaluation for Washington Public Power Supply System - Nuclear Plant 2 September 1986, (MDE-98-0986).

[19] GE DORT Code - a part of Code Package CCC-543: TORT: Three Dimensional Discrete Ordinates Transport maintained by Oak Ridge National Laboratory Radiation Shielding Information Center.

[20] Standard Methods for Notched Bar Impact Testing of Metallic Materials, Annual Book of ASTM Standards, E23-94b.

[21] Nuclear Plant Irradiated Steel Handbook, EPRI Report NP-4797, September 1986.

[22] Standard Methods of Tension Testing of Metallic Materials, Annual Book of ASTM Standards, E8-95.

[23] Fracture Toughness Requirements, USNRC Branch Technical Position MTEB 5-2, Revision 1, July 1981.

[24] Letter from James T. Wiggins to Mr. Lesley A. England, "Acceptance for Referencing of Topical Report NEDO-32205, Revision 1, '10CFR50 Appendix G Equivalent Margin

Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels," USNRC, Washington, D.C., December 8, 1993.

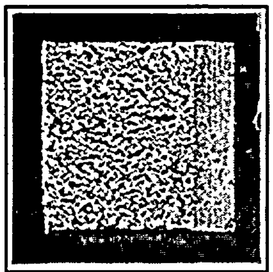
[25] Letter No. G02-95-235 (Docket No. 50-397) from J.V. Parrish to USNRC, "WNP2, Operating License NPF-21 Final Response to Revision 1, Supplement 1, of Generic Lettter 92-01, Reactor Vessel Structural Integrity", November 2, 1995.

APPENDIX A

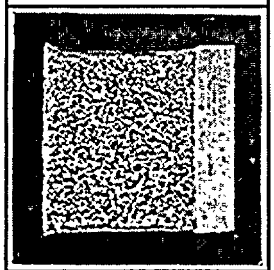
UNIRRADIATED AND IRRADIATED CHARPY SPECIMEN FRACTURE SURFACE PHOTOGRAPHS

Photographs of each Charpy specimen fracture surface were taken per the requirements of ASTM E185-82. The fracture surface photographs along with a summary of the Charpy test results for each unirradiated and irradiated specimen are provided in this Appendix. The pictures are arranged in the order of base, weld, and HAZ materials.

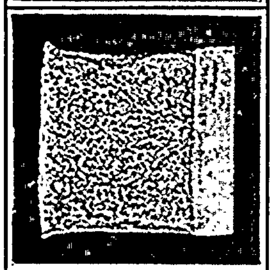
APPENDIX A - UNIRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



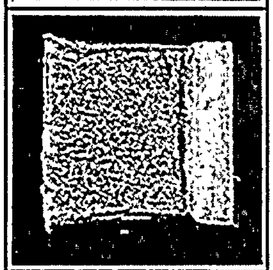
BASE: B1
Test Temp: -60°F
Energy: 5.0 ft-lbs
Lateral Exp: 11 mils
Shear: 1%



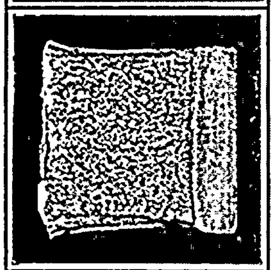
BASE: B2
Test Temp: -20°F
Energy: 10.5 ft-lbs
Lateral Exp: 12 mils
Shear: 20%



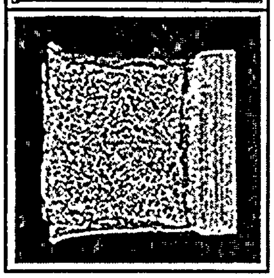
BASE: B3
Test Temp: 10°F
Energy: 17.0 ft-lbs
Lateral Exp: 20 mils
Shear: 34%



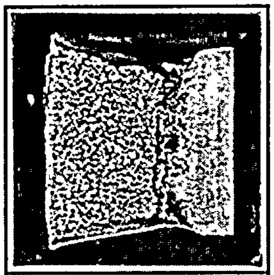
BASE: B4
Test Temp: 20°F
Energy: 38.0 ft-lbs
Lateral Exp: 35 mils
Shear: 34%



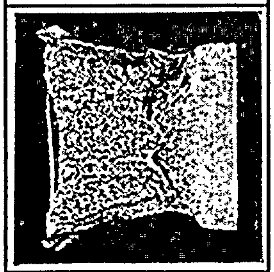
BASE: B5
Test Temp: 30°F
Energy: 27.0 ft-lbs
Lateral Exp: 26 mils
Shear: 37%



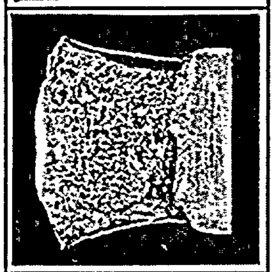
BASE: B6
Test Temp: 40°F
Energy: 41.0 ft-lbs
Lateral Exp: 37 mils
Shear: 38%



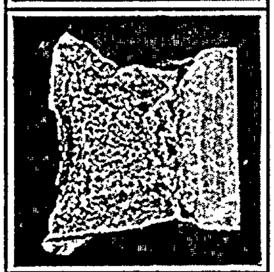
BASE: B7
Test Temp: 60°F
Energy: 77.5 ft-lbs
Lateral Exp: 63 mils
Shear: 45%



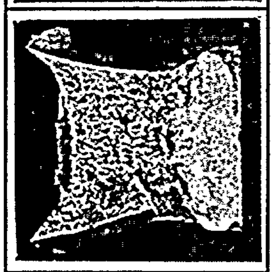
BASE: B8
Test Temp: 80°F
Energy: 99.0 ft-lbs
Lateral Exp: 81 mils
Shear: 63%



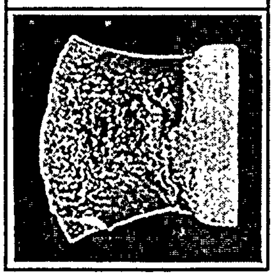
BASE: B9
Test Temp: 108°F
Energy: 113.0 ft-lbs
Lateral Exp: 86 mils
Shear: 69%



BASE: B10
Test Temp: 120°F
Energy: 130.5 ft-lbs
Lateral Exp: 83 mils
Shear: 80%

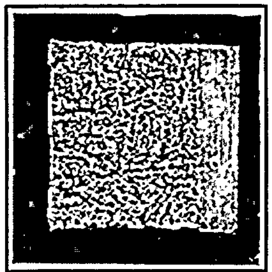


BASE: B11
Test Temp: 200°F
Energy: 148.0 ft-lbs
Lateral Exp: 90 mils
Shear: 100%

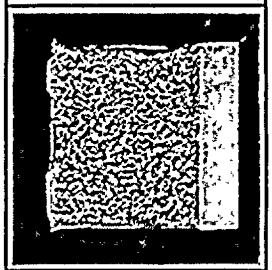


BASE: B12
Test Temp: 300°F
Energy: 153.5 ft-lbs
Lateral Exp: 91 mils
Shear: 100%

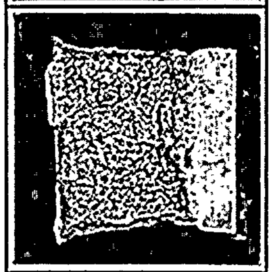
APPENDIX A - UNIRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



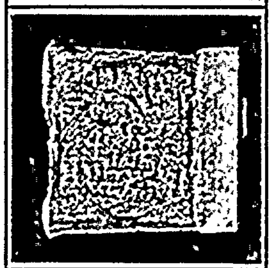
WELD: W1
Test Temp: -100°F
Energy: 7.0 ft-lbs
Lateral Exp: 9 mils
Shear: 12%



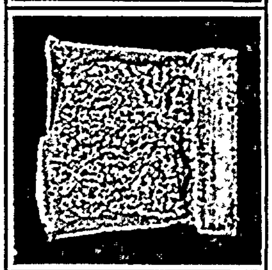
WELD: W2
Test Temp: -60°F
Energy: 9.0 ft-lbs
Lateral Exp: 14 mils
Shear: 14%



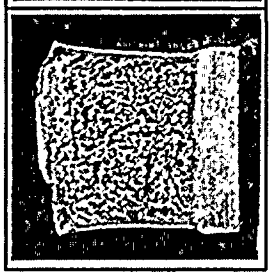
WELD: W3
Test Temp: -20°F
Energy: 25.5 ft-lbs
Lateral Exp: 30 mils
Shear: 42%



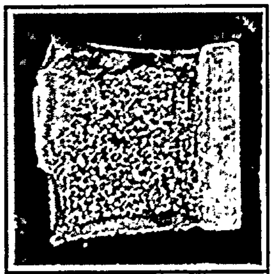
WELD: W4
Test Temp: -10°F
Energy: 52.0 ft-lbs
Lateral Exp: 48 mils
Shear: 48%



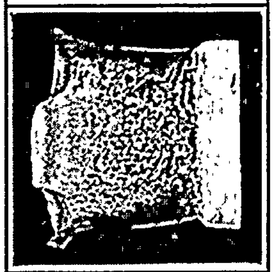
WELD: W5
Test Temp: 0°F
Energy: 42.0 ft-lbs
Lateral Exp: 40 mils
Shear: 37%



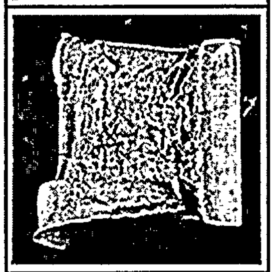
WELD: W6
Test Temp: 10°F
Energy: 36.0 ft-lbs
Lateral Exp: 43 mils
Shear: 55%



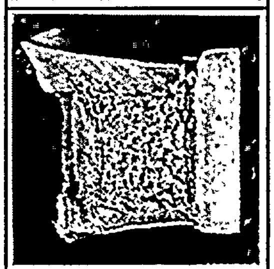
WELD: W7
Test Temp: 20°F
Energy: 53.0 ft-lbs
Lateral Exp: 52 mils
Shear: 69%



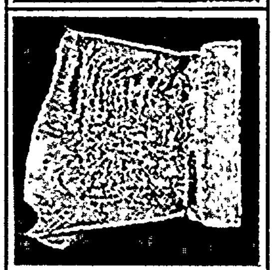
WELD: W8
Test Temp: 60°F
Energy: 74.5 ft-lbs
Lateral Exp: 70 mils
Shear: 87%



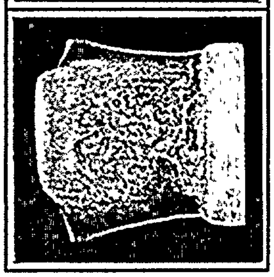
WELD: W9
Test Temp: 108°F
Energy: 89.0 ft-lbs
Lateral Exp: 76 mils
Shear: 97%



WELD: W10
Test Temp: 120°F
Energy: 98.5 ft-lbs
Lateral Exp: 76 mils
Shear: 100%

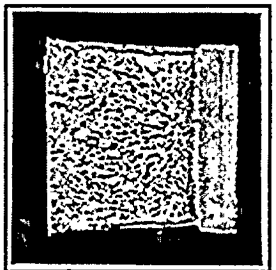


WELD: W11
Test Temp: 200°F
Energy: 107.0 ft-lbs
Lateral Exp: 67 mils
Shear: 100%

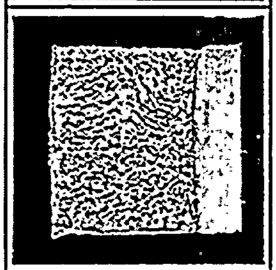


WELD: W12
Test Temp: 300°F
Energy: 97.0 ft-lbs
Lateral Exp: 91 mils
Shear: 100%

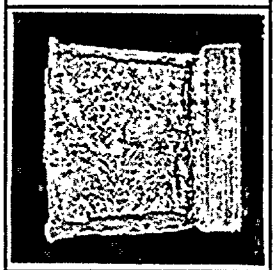
APPENDIX A - UNIRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



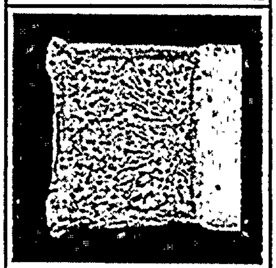
HAZ: HAZ1
Test Temp: -60°F
Energy: 28.0 ft-lbs
Lateral Exp: 17 mils
Shear: 20%



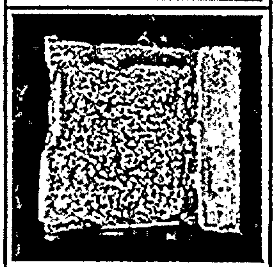
HAZ: HAZ2
Test Temp: -50°F
Energy: 16.0 ft-lbs
Lateral Exp: 16 mils
Shear: 18%



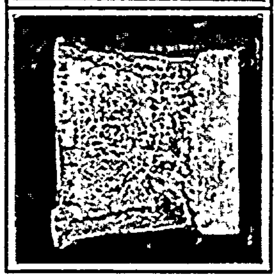
HAZ: HAZ3
Test Temp: -20°F
Energy: 48.0 ft-lbs
Lateral Exp: 37 mils
Shear: 37%



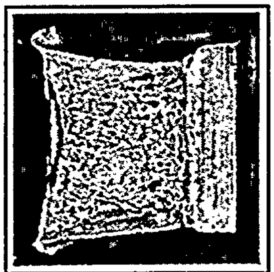
HAZ: HAZ4
Test Temp: -10°F
Energy: 34.0 ft-lbs
Lateral Exp: 35 mils
Shear: 41%



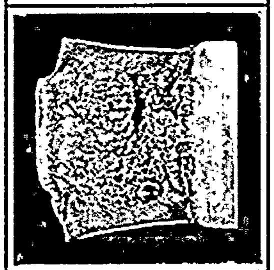
HAZ: HAZ5
Test Temp: 0°F
Energy: 44.5 ft-lbs
Lateral Exp: 44 mils
Shear: 70%



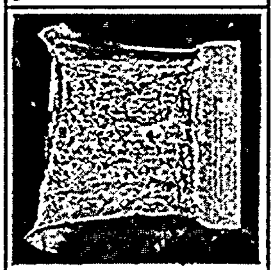
HAZ: HAZ6
Test Temp: 20°F
Energy: 83.0 ft-lbs
Lateral Exp: 62 mils
Shear: 65%



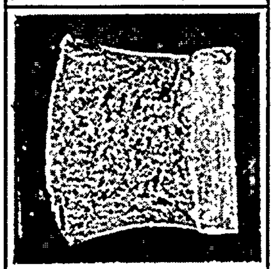
HAZ: HAZ7
Test Temp: 60°F
Energy: 103.5 ft-lbs
Lateral Exp: 81 mils
Shear: 88%



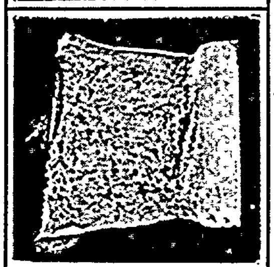
HAZ: HAZ8
Test Temp: 80°F
Energy: 74.5 ft-lbs
Lateral Exp: 73 mils
Shear: 90%



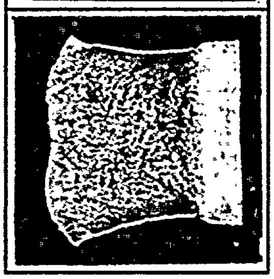
HAZ: HAZ9
Test Temp: 108°F
Energy: 110.5 ft-lbs
Lateral Exp: 86 mils
Shear: 88%



HAZ: HAZ10
Test Temp: 120°F
Energy: 118.0 ft-lbs
Lateral Exp: 84 mils
Shear: 100%

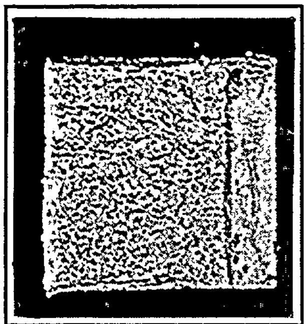


HAZ: HAZ11
Test Temp: 200°F
Energy: 121.5 ft-lbs
Lateral Exp: 76 mils
Shear: 100%

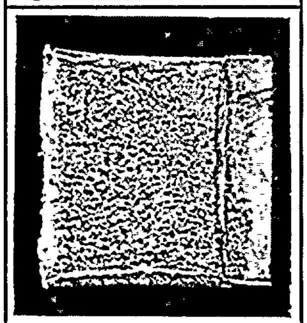


HAZ: HAZ12
Test Temp: 300°F
Energy: 118.0 ft-lbs
Lateral Exp: 89 mils
Shear: 100%

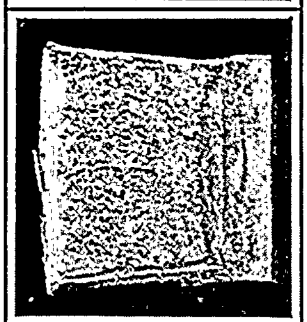
APPENDIX A - IRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



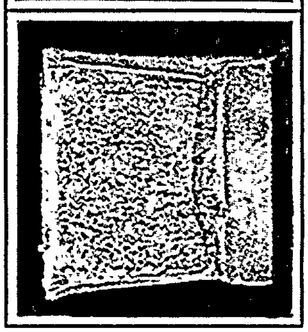
BASE: 29146
Test Temp: 0°F
Energy: 16.0 ft-lbs
Lateral Exp: 16 mils
Shear: 22%



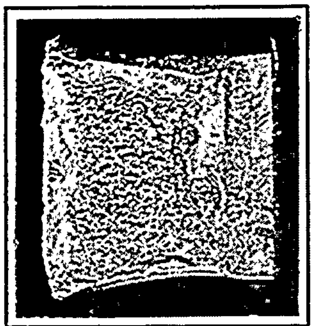
BASE: 29141
Test Temp: 30°F
Energy: 30.0 ft-lbs
Lateral Exp: 26 mils
Shear: 30%



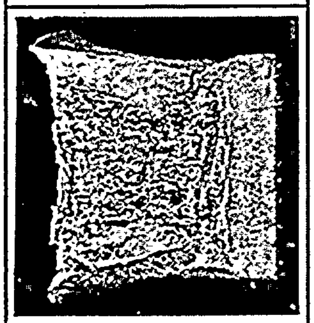
BASE: 29143
Test Temp: 39°F
Energy: 56.5 ft-lbs
Lateral Exp: 36 mils
Shear: 30%



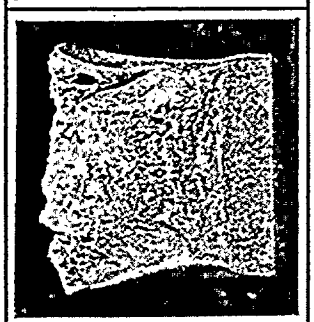
BASE: 29145
Test Temp: 67°F
Energy: 66.0 ft-lbs
Lateral Exp: 50 mils
Shear: 35%



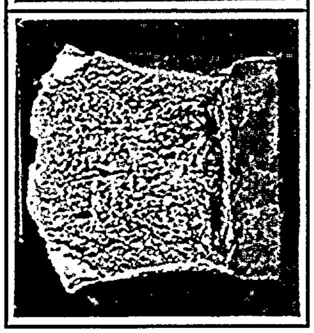
BASE: 29140
Test Temp: 100°F
Energy: 109.0 ft-lbs
Lateral Exp: 73 mils
Shear: 56%



BASE: 29144
Test Temp: 150°F
Energy: 139.5 ft-lbs
Lateral Exp: 87 mils
Shear: 70%

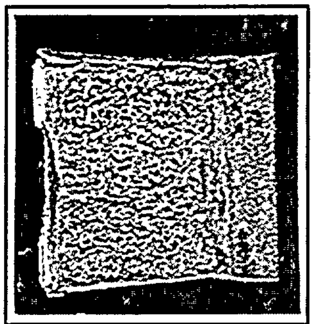


BASE: 29139
Test Temp: 200°F
Energy: 154.0 ft-lbs
Lateral Exp: 85 mils
Shear: 96%

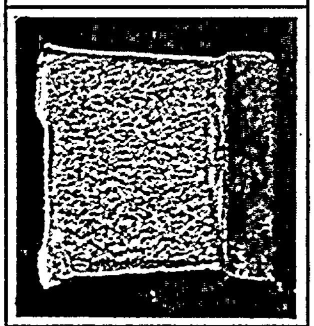


BASE: 29142
Test Temp: 300°F
Energy: 152.5 ft-lbs
Lateral Exp: 91 mils
Shear: 100%

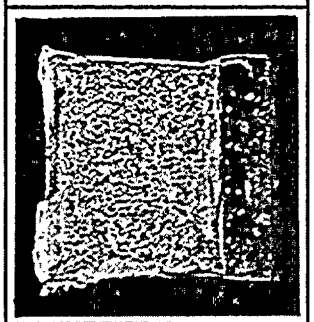
APPENDIX A - IRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



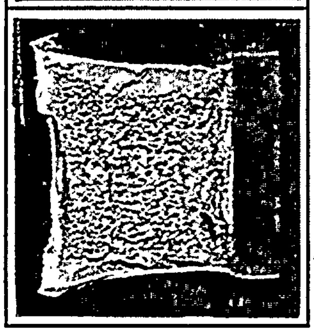
WELD: 29148
Test Temp: -20°F
Energy: [illegible] ft-lbs
Lateral [illegible] 36 mils
Shear: 31%



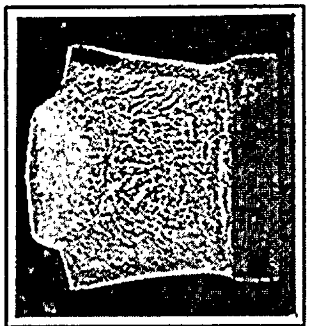
WELD: 29153
Test Temp: -10°F
Energy: 35.0 ft-lbs
Lateral Exp: 36 mils
Shear: 34%



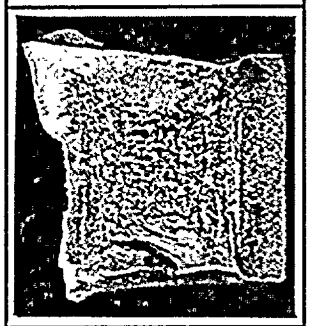
WELD: 29152
Test Temp: 22°F
Energy: 50.0 ft-lbs
Lateral Exp: 48 mils
Shear: 54%



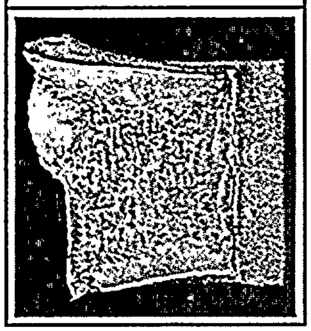
WELD: 29154
Test Temp: 50°F
Energy: 72.0 ft-lbs
Lateral Exp: 66 mils
Shear: 67%



WELD: 29151
Test Temp: 85°F
Energy: 102.5 ft-lbs
Lateral Exp: 87 mils
Shear: 94%

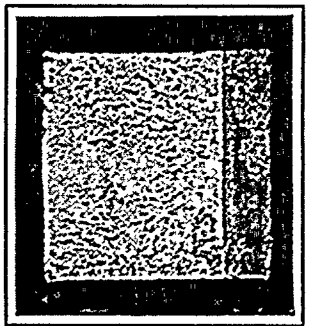


WELD: 29149
Test Temp: 100°F
Energy: 104.0 ft-lbs
Lateral Exp: 76 mils
Shear: 91%

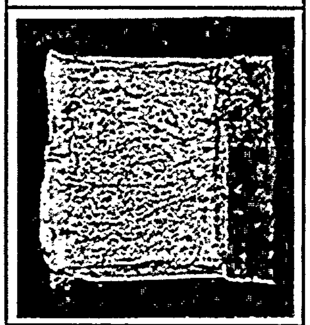


WELD: 29150
Test Temp: 200°F
Energy: 108.0 ft-lbs
Lateral Exp: 76 mils
Shear: 100%

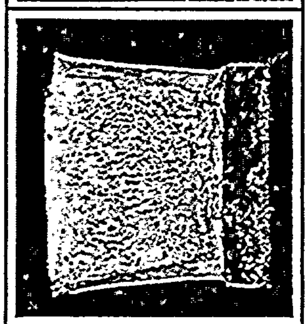
APPENDIX A - IRRADIATED SPECIMEN FRACTURE SURFACE APPEARANCE



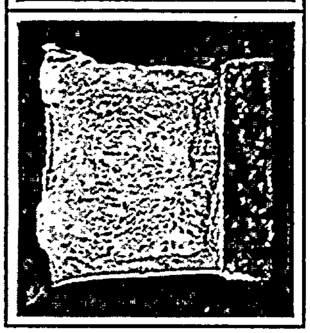
HAZ: 29157
Test Temp: -60°F
Energy: 7.0 ft-lbs
Lateral Exp: 8 mils
Shear: 1%



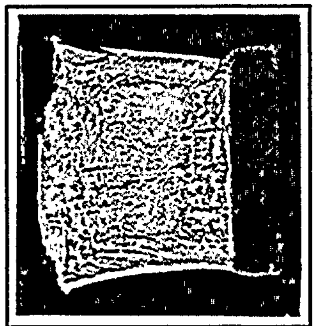
HAZ: 29159
Test Temp: -40°F
Energy: 28.0 ft-lbs
Lateral Exp: 26 mils
Shear: 35%



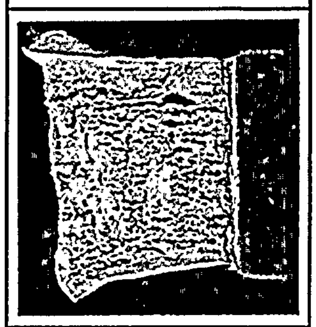
HAZ: 29158
Test Temp: -20°F
Energy: 58.5 ft-lbs
Lateral Exp: 43 mils
Shear: 39%



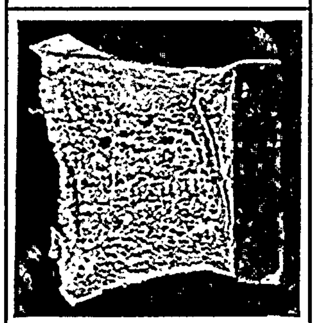
HAZ: 29162
Test Temp: 8°F
Energy: 62.0 ft-lbs
Lateral Exp: 45 mils
Shear: 43%



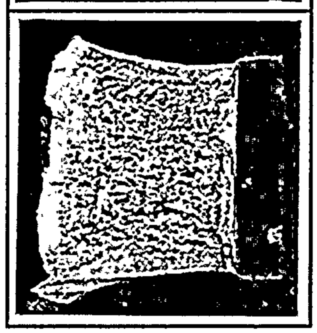
HAZ: 29156
Test Temp: 67°F
Energy: 88.0 ft-lbs
Lateral Exp: 61 mils
Shear: 76%



HAZ: 29160
Test Temp: 105°F
Energy: 105.5 ft-lbs
Lateral Exp: 77 mils
Shear: 91%



HAZ: 29155
Test Temp: 130°F
Energy: 127.0 ft-lbs
Lateral Exp: 82 mils
Shear: 100%



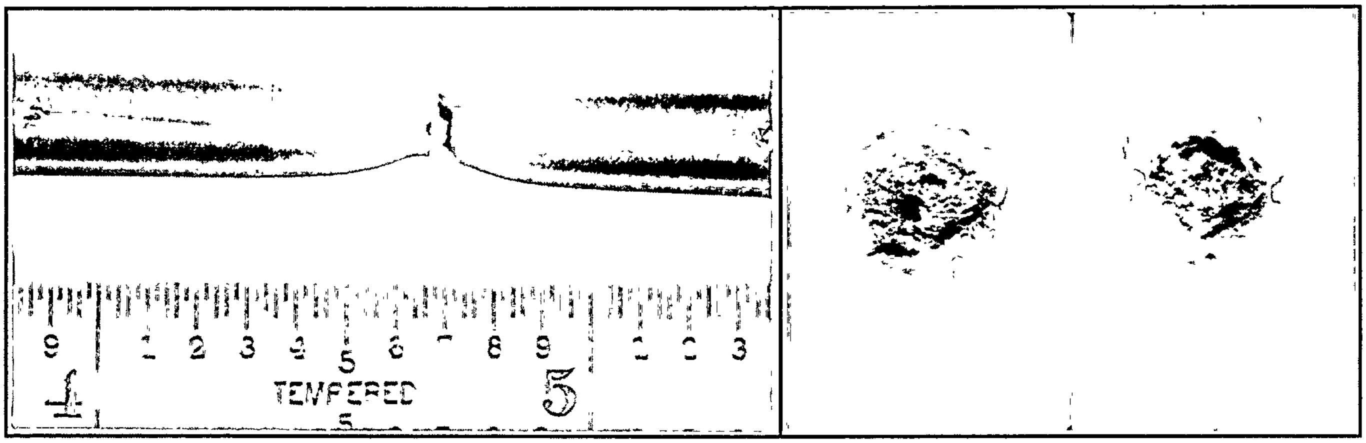
HAZ: 29161
Test Temp: 200°F
Energy: 127.0 ft-lbs
Lateral Exp: 81 mils
Shear: 100%

APPENDIX B

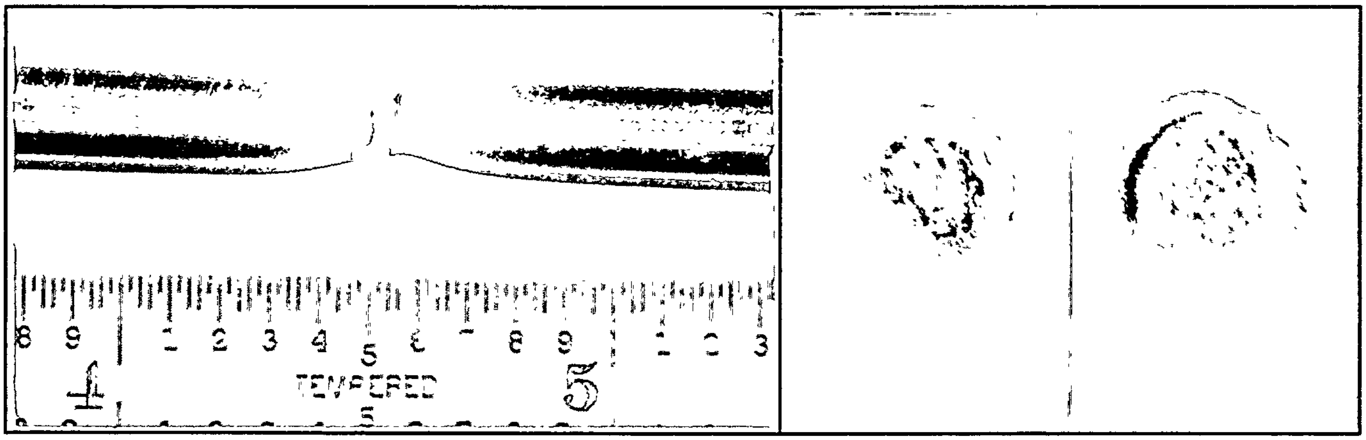
UNIRRADIATED AND IRRADIATED TENSILE SPECIMEN FRACTURE APPEARANCE

The necking behavior and the fracture appearance of the tensile test specimens are provided in this Appendix. The pictures are arranged in the order of Base, Weld and HAZ materials at the test temperatures of 70° F and 550°F, respectively.

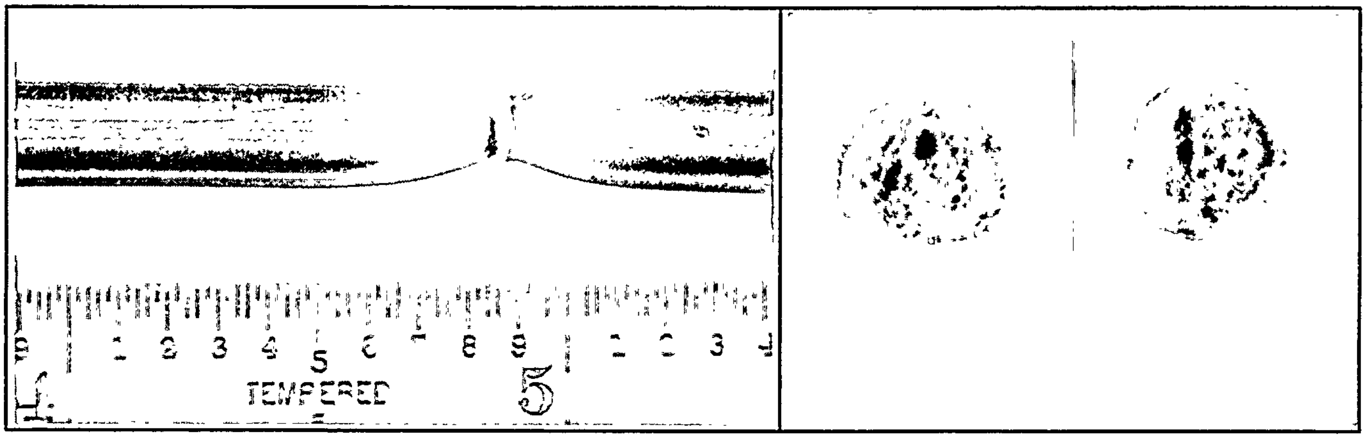
APPENDIX B - UNIRRADIATED TENSILE SPECIMEN FRACTURE APPEARANCE



Unirradiated Base at 70°F

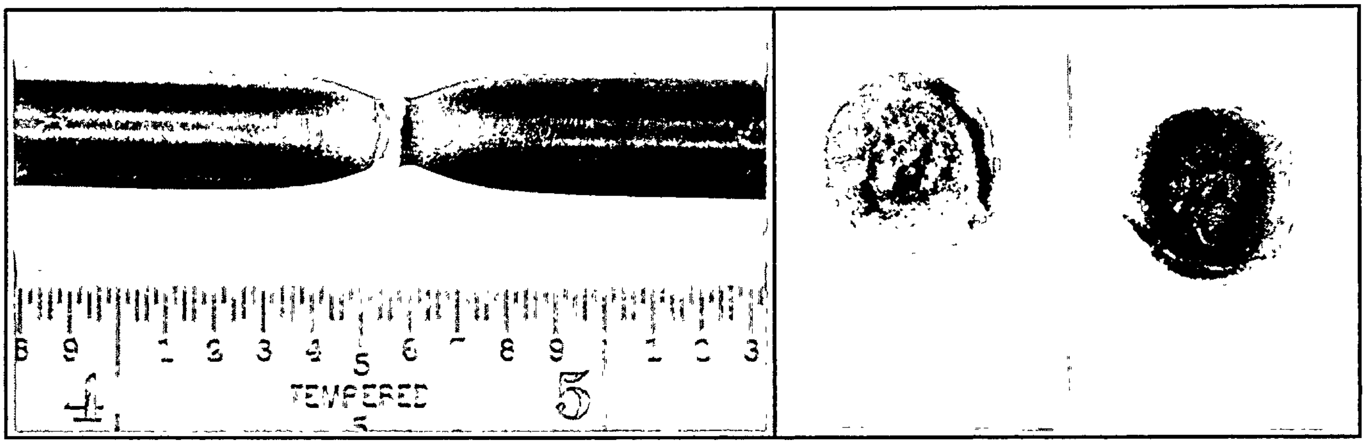


Unirradiated Weld at 70°F

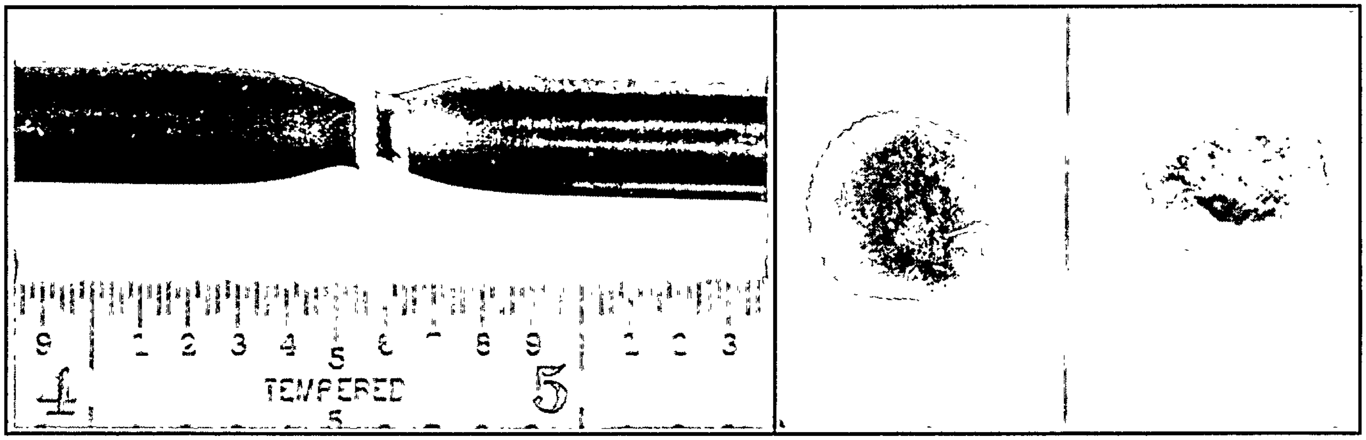


Unirradiated HAZ at 70°F

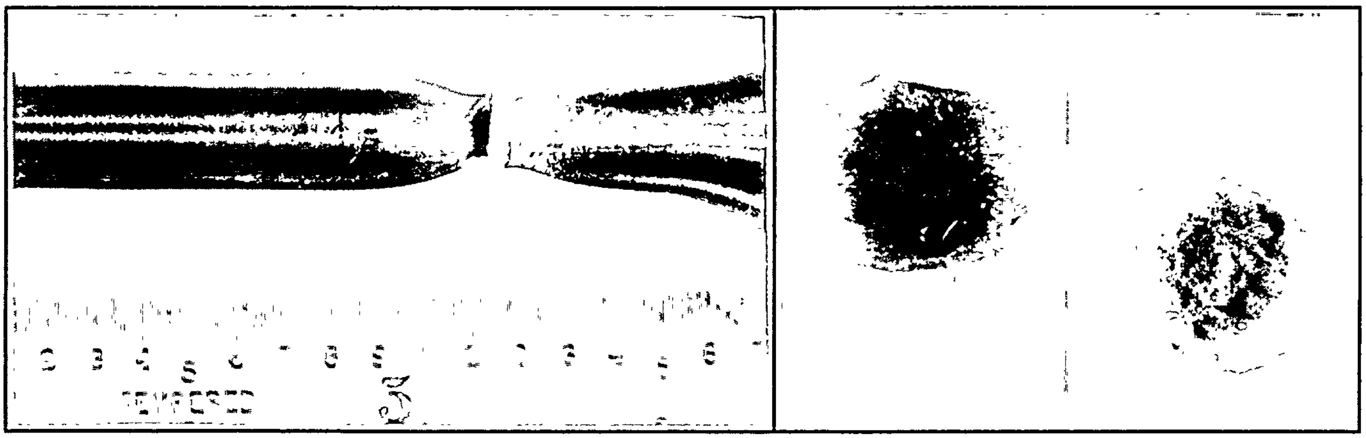
APPENDIX B - UNIRRADIATED TENSILE SPECIMEN FRACTURE APPEARANCE



Unirradiated Base at 550°F

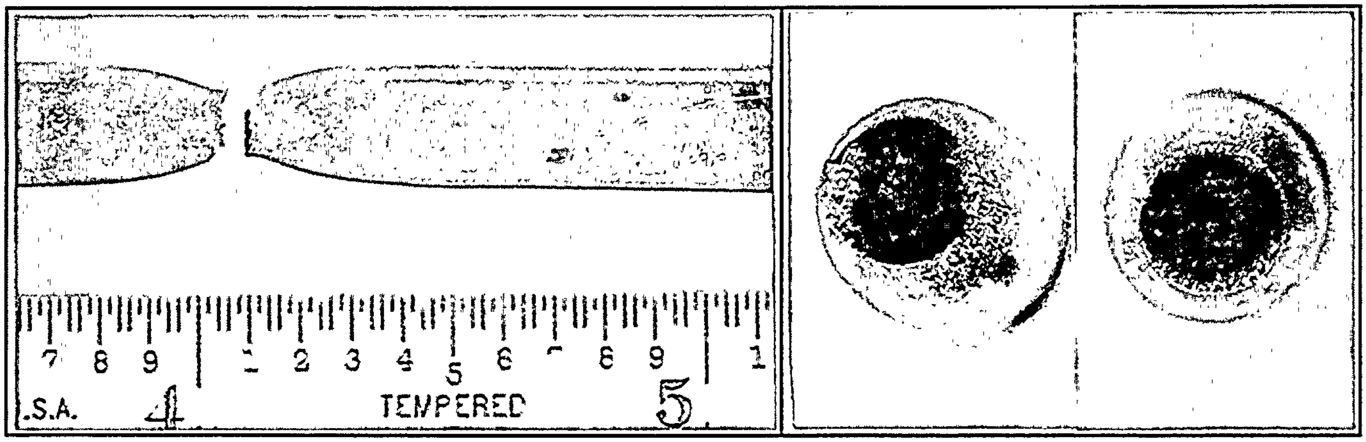


Unirradiated Weld at 550°F

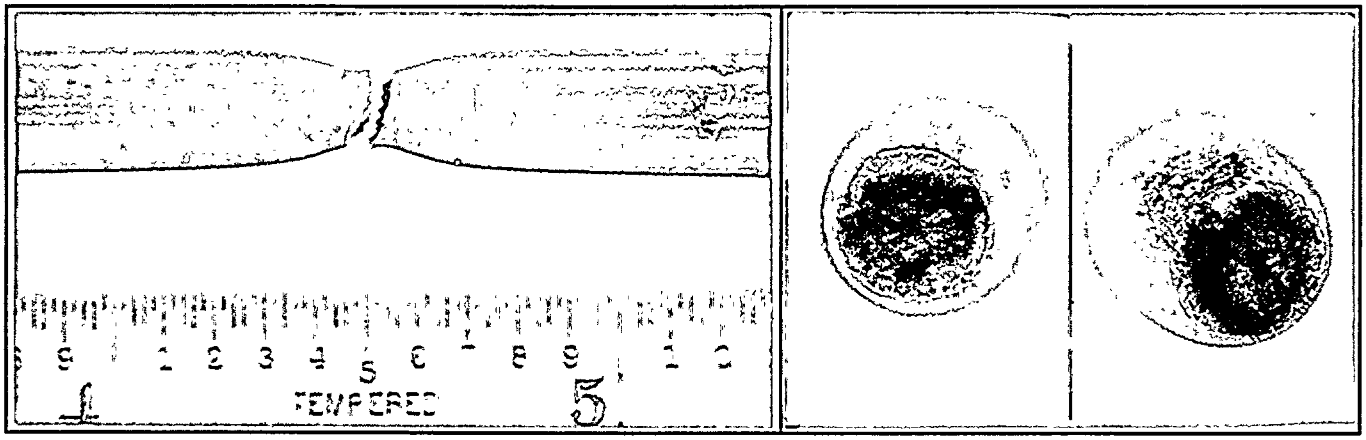


Unirradiated HAZ at 550°F

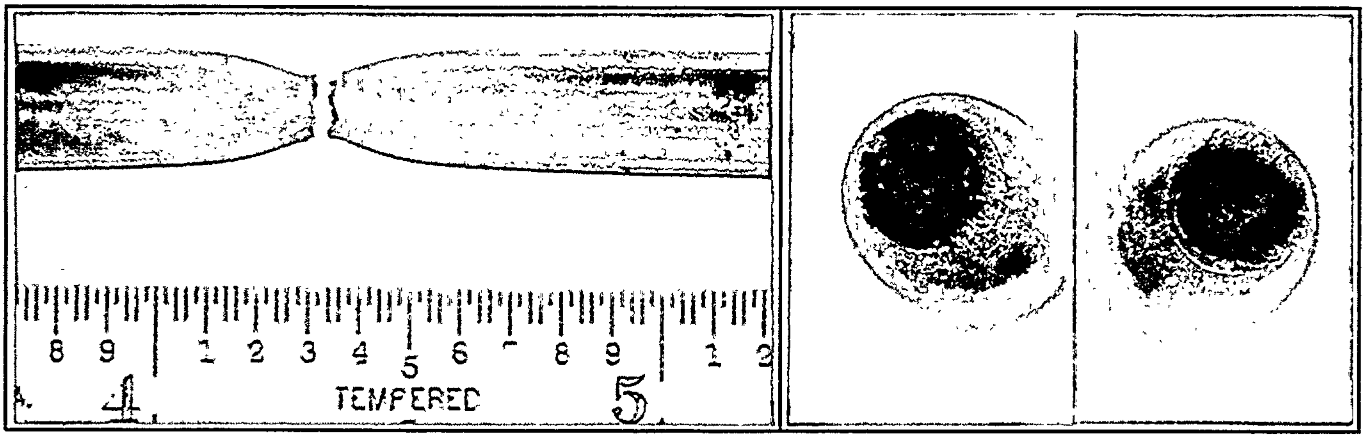
APPENDIX B - IRRADIATED TENSILE SPECIMEN FRACTURE APPEARANCE



Irradiated Base at 70°F

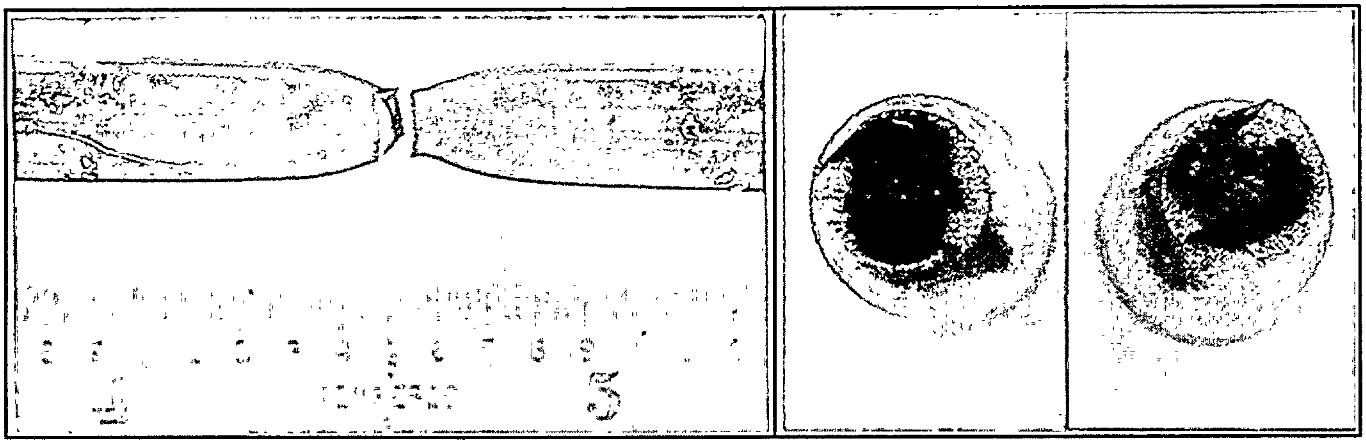


Irradiated Weld at 70°F

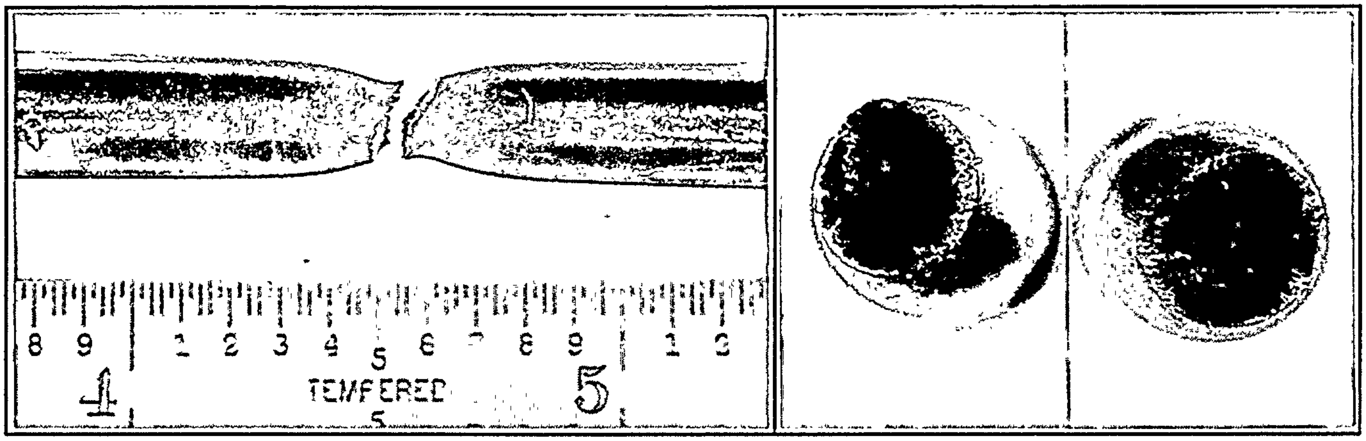


Irradiated HAZ at 70°F

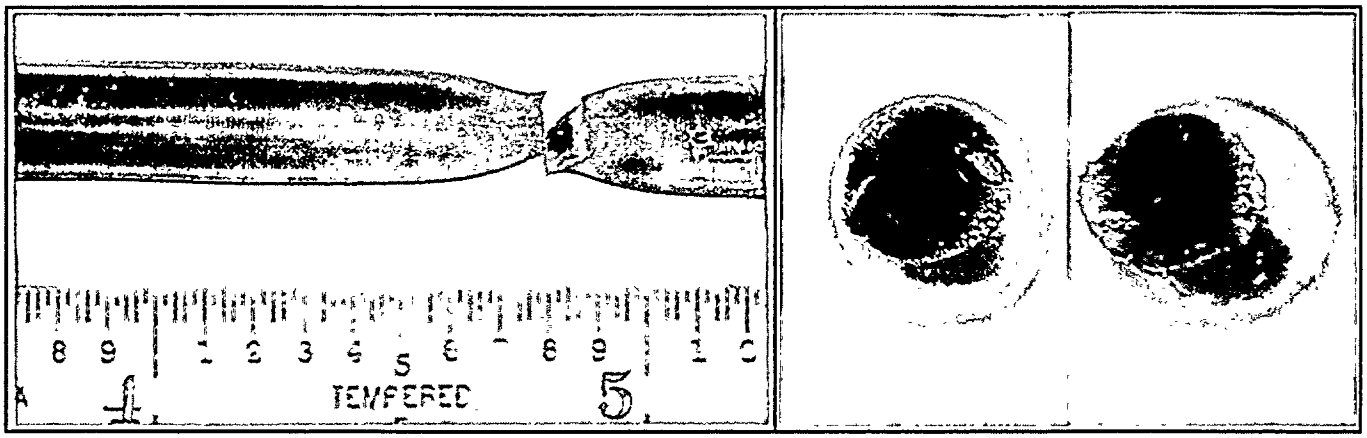
APPENDIX B - IRRADIATED TENSILE SPECIMEN FRACTURE APPEARANCE



Irradiated Base at 550°F



Irradiated Weld at 550°F



Irradiated HAZ at 550°F

OPERATIONAL QUALITY ASSURANCE PROGRAM DESCRIPTION PROPOSED REVISION 26, DESCRIPTION OF CHANGES AND JUSTIFICATION

Section 1.3.3.1 is being changed to correct the reference from the WNP-2 Technical Specifications to Appendix III of the Operational Quality Assurance Program Description (OQAPD) as the location for the requirements for the Plant Operations Committee (POC). The revision to Section 1 - ORGANIZATION is an editorial change and the section will remain at Revision 25.

Table 2-1 of Section 2 - QUALITY ASSURANCE PROGRAM is changed to list new Site Wide Procedures that implement the Quality Assurance Program. This is an editorial change.

APPENDIX I- QUALIFICATION REQUIREMENTS

APPENDIX 1, Revision 12, defines the minimum qualification requirements for the Manager, Quality and Supervisor, Quality Services as:

Manager, Quality
a. Education: Bachelor Degree or equivalent* in Engineering or a related science.
b. Experience: Ten (10) years experience in the field of quality assurance, or equivalent number of years of nuclear industry experience in a management position or a combination of the two. The requirement that the manager have at least two years of experience in the administration of and adherence to the Quality Assurance Program in a significant management role directly involving nuclear power plants is being deleted.

Supervisor, Quality Services
a. Education: Bachelor Degree or equivalent* in Engineering or a related science.
b. Experience: Four (4) years in the field of quality assurance, or equivalent number of years of nuclear plant experience in a supervisory position, preferably at an operating nuclear plant, or a combination of the two. At least one (1) of these four (4) years of experience shall be nuclear power plant experience in the implementation of the quality assurance program.

*Equivalency will be determined based upon an evaluation of the following factors:
1. High School diploma or GED.
2. Sixty (60) semester hours of related technical education taught at the college level (900 classroom or instructor conducted hours).
3. Qualified as an NRC senior operator at the assigned plant.
4. Four (4) years of additional experience in his area of responsibility.
5. Four (4) years of supervisory or management experience.
6. Demonstrated ability to communicate clearly (verbally and in writing).
7. Certification of academic ability and knowledge by corporate management.
8. Successful completion of the Engineer-In-Training examination.
9. Professional Engineer License.
10. Associated degree in Engineering or a related science.

The Supply System proposes to modify the Qualification Requirement as follows:

"The Manager, Quality or the Supervisor, Quality Services fulfills the position described in ANSI/ANS-3.1-1978, Section 4.4.5, Quality Assurance. The qualifications of this position are:

a. Education: Bachelor Degree or equivalent* in Engineering or a related science.
b. Experience: Six (6) years experience in the field of quality assurance, or equivalent number of years of nuclear industry experience in a supervisory/management position or a combination of the two. At least two (2) years of these six years experience shall be nuclear power plant experience in the overall implementation of the quality assurance program. (This experience shall be obtained within the quality assurance organization.)

This proposed revision clearly specifies the Qualification Requirements for these two positions. This revision would bring the qualification requirements into alignment with NUREG-0800, Standard Review Plan, Section 17.2, Quality Assurance During The Operations Phase, which states, "The qualifications of the QA Manager should be at least equivalent to those described in Section 4.4.5 of ANSI/ANS-3.1-1978, Selection and Training of Nuclear Power Plant Personnel, as endorsed by the regulatory positions in Regulatory Guide 1.8."

Allowing either position to meet the requirements of ANSI/ANS-3.1-1978 provides Supply System management the flexibility to use one of the positions for management rotation, to expand on the individual's knowledge base. This will assure the Supply System continues to strengthen the knowledge and experience of individuals moving into top management positions. There will always be at least one individual, the Quality Manager or the Supervisor, Quality Services with the required qualifications and the necessary knowledge for the position.

CONCLUSION: The Supply System has concluded that this revision to Qualification Requirements is a reduction of a current commitment. However, the change will continue to meet the requirements of 10 CFR 50, Appendix B and ANSI/ANS-3.1-1978, Section 4.4.5.

Wolf Creek Nuclear Operating Corporation has a similar qualification statement in their FSAR Section 13.1.2.4, which states in part, "..The Supervisor Quality Evaluation or the Manager Performance Improvement and Assessment fulfills the position described in ANSI/ANS 3.1-1978, 4.4.5, Quality Assurance..."

APPENDIX II - POSITION STATEMENTS

An editorial change is being made to page 1 to correct the reference to the Supply System Quality Department.

APPENDIX III - ADDITIONAL QUALITY PROGRAM REQUIREMENTS

APPENDIX III, Revision 0, identifies additional quality program requirements that were formerly located in the WNP-2 Technical Specifications, Section 6.0, Administrative Controls.

PROPOSAL 1. REVIEW AND APPROVAL OF PROGRAMS AND PROCEDURES

The Supply System proposes to modify the POC procedure review responsibilities for nuclear safety related procedures and procedure changes and assign the responsibility to the line organizations. POC will continue to perform safety reviews associated with procedures that are of safety significance.

The proposed changes to the OQAPD, Appendix III reduce the administrative burden on POC by establishing a procedure review and approval process which shifts more responsibility for the review of nuclear safety related procedures and procedure changes from POC to the line organizations. Currently, all Technical Specification required procedures and changes thereto must be reviewed by POC. Instead, these items will be reviewed and approved through a new procedure review and approval process.

The new process for procedure review and approval shall be controlled by administrative procedures. The process requires that all nuclear safety related procedures and procedure changes be reviewed by two designated technical reviewers, qualified licensing basis impact determination (LBID), including 10 CFR 50.59, preparer and reviewer and approved by a responsible procedure owner. The designated technical reviewers verify the technical accuracy and usability of procedures and revisions including human factor considerations. A qualified preparer performs the 10 CFR 50.59 screening and, as necessary, safety evaluation associated with the procedure or procedure change which is then independently reviewed by a qualified reviewer.

The following changes reflect the implementation of the procedure review and approval process:

a. The proposed change adds the use of "Qualified Procedure Reviewer" in the procedure review process. LBID/technical reviewers will perform reviews of procedures and procedure changes prior to approval by the responsible procedure owner.

b. A new proposed Section (4.0) which sets forth the principal provisions of the review and approval of programs and procedures. The review and approval of programs and procedures is outlined below and shall be controlled by an administrative procedure as required by Appendix A of Regulatory Guide 1.33. OQAPD Appendix III, Section 4.1 requires that an administrative procedure be maintained and established covering procedure review and approval activities.

1. Each program and procedure required by Technical Specification 5.4 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a Qualified Procedure Reviewer who is knowledgeable in the functional area affected, but is not the individual who prepared the procedure or procedure change. The Qualified Procedure Reviewer shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971, for applicable positions with the exclusion of the positions identified in Sections 4.3.2 and 4.5. Individuals whose positions are described in Sections 4.3.2 and 4.5 may qualify as a Qualified Procedure Reviewer provided they meet the qualification described in other positions of Section 4. All required cross-disciplinary reviews of new procedures, procedure revisions or changes thereto shall be completed prior to approval.

2. Each program and procedure required by Technical Specification 5.4 and other procedures that affect nuclear safety, and changes thereto, will be reviewed by a qualified LBID preparer and reviewer. The qualified LBID preparer and reviewer evaluate the procedure from a safety perspective. This reviewer completes the 10 CFR 50.59 safety screening and evaluation.

3. Each new procedure or proposed procedure change required by Technical Specification 5.4 shall be reviewed by the procedure sponsor. The procedure sponsor is to ensure that the proposed activity has been prepared, documented, and reviewed in accordance with the administrative procedure governing the procedure review and approval process. The procedure sponsor is responsible for the procedures technical accuracy and usability including human factor considerations.

4. If a safety evaluation is not required, the new procedure or procedure change shall receive final approval by the procedure sponsor prior to implementation. POC review is not required.

5. If a safety evaluation is required, the procedure or procedure change shall be reviewed by POC. The new procedure or procedure change shall receive final approval by the appropriate member(s) of management, as determined by the Plant General Manager.

6. Proposed changes to the Process Control Program and the Offsite Dose Calculation Manual must be reviewed by POC and accepted by the Plant General Manager prior to implementation whether or not a safety evaluation is necessary.

CONCLUSION: The proposed change to use a Qualified Procedure Reviewer is considered to be an improvement to the process. The approval of procedures and programs by methods other than by POC and the Plant General Manager is considered a reduction in commitments.

The proposed change in OQAPD, Appendix III, Section 4.5 allows for an appropriate responsible member of management to approve procedures rather than the Plant General

[illegible]

[illegible]

Manager. This section is changed to allow for the implementation of the proposed procedure review and approval process. Nuclear safety related procedures and procedure changes will be reviewed and approved, prior to implementation, by the appropriate member(s) of management, as determined by the Plant General Manager. If the procedure requires a safety evaluation, then the evaluation shall be reviewed by POC. POC will recommend approval or disapproval of the evaluation to the Plant General Manager.

The proposed change in OQAPD, Appendix III, Sections 3.1 and 3.2 allows for temporary procedure changes to be reviewed per the procedure review and approval process and approved by the procedure sponsor within 14 days. POC review of the temporary procedure change may or may not be required depending on whether or not the procedure change has a safety evaluation associated with it. The procedure sponsor is responsible for ensuring that the intent of the procedure is not changed. The wording in Section 3.2.b. is also being changed to clarify that the approval of a temporary change must be approved by the supervisor in charge of the shift. This clarification is to align the process with the requirements of ANSI N 18.7-1976, Section 5.2.2 Procedure Adherence. The current OQAPD, Appendix III, Section 3.2, required each temporary procedure change to be reviewed by POC and approved by the Plant General Manager.

CONCLUSION: The proposed change for approval of temporary procedure changes is considered a reduction in commitments.

The proposed procedure review and approval process requires that POC review safety evaluations associated with procedures and procedure changes. These changes alter the scope of POC's review function by eliminating reviews of items that do not impact the environment or nuclear safety and, therefore, enhance POC's function by improving its efficiency and effectiveness. POC can then focus its attention on matters which could affect nuclear safety.

The proposed process establishes flow and documentation requirements for the review and approval of all nuclear safety related procedures and procedure changes. The program builds on a safety evaluation process which is implemented through an administrative procedure which meets the requirements of 10 CFR 50.59.

Each procedure or procedure change will be reviewed by a qualified reviewer. The "review" is a two step process comprised of:—1) a Licensing Basis Impact Determination and 2) a 10 CFR 50.59 Safety Evaluation. The Licensing Basis Impact Determination screen consists of a series of questions whose responses determine whether or not a Safety Evaluation is required. The results of the screening process and Safety Evaluation will be documented in writing. If the review for a new procedure or procedure change concludes that a safety evaluation is required, then the evaluation(s) shall be prepared and then reviewed by POC. Personnel completing the screening process and safety evaluation will be appropriately qualified and trained to perform this function.

The completed procedure or procedure change package is reviewed by a Qualified Procedure Reviewer. The Qualified Procedure Reviewer shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable position with the exclusion of the positions identified in Sections 4.3.2 and 4.5. Individuals whose positions are described in Sections 4.3.2 and 4.5 may qualify as a Qualified Procedure Reviewer provided they meet the qualifications described in other portions of Section 4. They can be the same individual as the qualified LBID preparer/or reviewer if qualified to perform both functions. The Qualified Procedure Reviewers are knowledgeable in the functional/technical subject matter related to the proposed activity and are designated by the Department Managers. Qualified Procedure Reviewers shall be responsible for reviewing the procedure or procedure change for adequacy, completeness, and accuracy. They shall also be responsible for identifying whether or not cross disciplinary reviews are required.

After all the necessary reviews (LBID and procedure) have been completed, the procedure or procedure change package is reviewed by the procedure sponsor. The procedure sponsor is responsible for ensuring that the proposed activity has been prepared, documented and reviewed in accordance with the administrative procedure that governs the procedure review and approval process. The procedure sponsor is responsible for the technical accuracy and usability of procedures and revisions, including human factor considerations.

If a safety evaluation is not required, the new procedure or procedure change receives final approval from the procedure sponsor. POC review is not required. If however, a safety evaluation is required, POC is responsible for reviewing the safety evaluation and recommending to the Plant General Manager approval or disapproval of it. The procedure change still receives approval from the procedure sponsor.

The Supply System implemented the bases of this procedure review program in January 1996, with the exception that POC has continued review all procedures and procedure changes. All requirements currently specified for POC and procedure review and approval remained in place.

Changes similar to those proposed for the Review and Approval of Programs and Procedures in the OQAPD, Appendix III, Section 2.1.1 were found acceptable by the NRC and approved for Indian Point Nuclear Generating Unit No. 3, for Wolf Creek Generating Station, Unit No. 1, and Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, & 3.

PROPOSAL 2. PLANT OPERATIONS COMMITTEE (POC) COMPOSITION

OQAPD, Appendix III, Sections 2.1.1 and 2.1.2 will be modified to specify the POC composition in terms of the following plant functional areas rather than by designating members by organizations:

Operations	Radiation Protection
Maintenance	Technical Services
Engineering	Chemistry
Quality	Planning/Scheduling/Outage
Administrative Services	

The proposed change would require the Plant General Manager appoint, in writing, a Vice Chairman and individual members experienced in the designated functional areas. The qualifications of all members shall meet the minimum requirements of ANSI/ANS-3.1-1981, Section 4.7, and cumulatively have expertise in the designated functional areas noted above.

The OQAPD, Appendix III, Section 2.1.1 currently defines the POC composition and designates POC members by specific organizational titles. The proposed changes will delete the position titles and replace them with the functional areas to be represented in POC. POC serves as a multi-disciplinary review and advisory organization to the Plant General Manager on matters related to nuclear and radiological safety. The proposed method of designating POC composition will provide the Plant General Manager with the flexibility to appoint qualified individuals from disciplines within a functional area. The POC composition change will also alleviate the need to process OQAPD changes for future organizational changes involving organizational position titles, while maintaining the requirement for a multi-disciplinary POC membership.

The proposed change to OQAPD, Appendix III, Section 2.1.2 establishes that the Plant General Manager will appoint, in writing, the POC Vice Chairman, and individual members from each of the designated functional areas. POC members and alternate qualifications specified as Section 4.7 of ANSI/ANS-3.1-1981 assures that the education and experience level of each member meets or exceeds the current qualification requirements specified in Technical Specification 5.3.1 for those individuals who fill organizational positions considered part of the unit staff in Technical Specification 5.3.1 . This will ensure POC will continue to be staffed by qualified personnel having a variety of expertise and experience.
CONCLUSION: This change to POC composition is not considered to be a reduction of commitments.

Changes similar to those proposed for POC composition in OQAPD, Appendix III, Sections 2.1.1 and 2.1.2 were found acceptable by the NRC and approved for the Kewaunee Station, for Fermi 2 Station, and for St. Lucie Plants 1 and 2.

PROPOSAL 3. CNSRB MEMBERSHIP AND RESPONSIBILITIES

There are two changes to the Corporate Nuclear Safety Review Board (CNSRB) responsibilities.

The proposed change to OQAPD, Appendix III, Section 2.2.2 would clarify the number of members required for the CNSRB. The present number of nine would be modified to read, "The CNSRB shall be composed of at least nine and no more that twelve members appointed..." Section 2.2.6 would also be modified to add, "The quorum shall consist of not less than the majority of the members, or duly appointed alternates." This means for:

9 members - a quorum is 5
10 members - a quorum is 5
11 members - a quorum is 6
12 members - a quorum is 6

Appendix III, Sections 2.2.9.a. & 2.2.9.b. are being clarified to allow the distribution of CNSRB records within 15 working days, rather than 14 days. This is considered an administrative change to the process and it does not have an impact on the review results.

These changes will continue to meet the requirements of ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, Section 4.3.2. Standing Committees Functioning as Independent Review Bodies.

CONCLUSION: These changes are not considered a reduction of commitments.