REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR INDIAN POINT UNIT NO. 2 ANALYSIS OF CAPSULE V

FINAL REPORT
SwRI Project No. 17-2108

Prepared for

Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

October 1988

SOUTHWEST RESEARCH INSTITUTE
SAN ANTONIO HOUSTON

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ABSTRACT

Capsule V, the fourth vessel material surveillance capsule removed from the Indian Point Unit No. 2 nuclear power plant, has been tested, and the results have been evaluated. The analysis of the data (1) confirms the decrease in fluence rate from the low leakage core vs cycles prior to cycle 6, and (2) indicates that the pressure vessel weld and plate materials will retain adequate shelf toughness throughout the 32 EFPY design life-time using either the new Regulatory Guide 1.99, Revision 2 or the original Revision 1 requirements.

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I. SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the fourth material surveillance capsule removed from the Indian Point Unit No. 2 reactor pressure vessel led to the following conclusions:

(1) Based on a calculated neutron spectral distribution, Capsule V received a fast fluence of 5.3×10^{18} n/cm^2 (E > 1 MeV) at its radial center line.

(2) The surveillance specimens of the core beltline plate materials experienced shifts in RT_{NDT} (50 ft-lb. values) over the range of 79°F (Plate B2002-2) to 239°F (Weld) as a result of fast neutron exposure up to the 1987 refueling outage.

(3) The core beltline weld exhibited the largest shift in RT_{NDT} and is projected to control the heatup and cooldown limitations throughout the design lifetime of the pressure vessel.

(4) From the previous capsule, Z, the estimated maximum neutron fluence of $3.33 \times 10^{18*}$ neutrons/cm^2 (E > 1 MeV) was received by the vessel wall in 5.17 effective full power years (EFPY) through Cycle 5, which is equal to a fluence rate of $6.44 \times 10^{17*}$ per EFPY. At the end of Cycle 8 (8.6 EFPY) the neutron fluence was 4.45×10^{18} n/cm^2 giving 3.26×10^{17} n/cm^2 per EFPY for Cycles 6 through 8. This calculated value for the decrease in fluence per EFPY agrees well with the experimental value for the decrease in fluence rate; i.e., 50.6% vs. 48.9%. The use of a low leakage core loading pattern beginning with Cycle 6 did significantly reduce the fluence rate on the pressure vessel wall.

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*Revised from Capsule Z report using the latest plant specific lead factors.

(5) The Indian Point Unit No. 2 vessel weld metal located in the core beltline region are projected to retain sufficient toughness to meet the current 50 ft-lb Charpy upper shelf requirements of 10CFR50 Appendix G throughout the design life of the pressure vessel using either Revision 2 or Revision 1 requirements of Regulatory Guide 1.99.

(6) Based on Regulatory Guide 1.99, Rev. 2, trend curves, the projected maximum RT_{NDT} for the Indian Point Unit No. 2 vessel core beltline materials at the 1/4T and 3/4T positions after 32 EFPY of operation are 266°F and 207°F, respectively. These values were used as the bases for computing heatup and cooldown limit curves to be used for up to 32 EFPY of operation.

II. BACKGROUND

The allowable loadings on nuclear pressure vessels are determined by applying the rules in Appendix G, "Fracture Toughness Requirements," of 10CFR50 (1). In the case of pressure-retaining components made of ferritic materials, the allowable loadings depend on the reference stress intensity factor (K_{IR}) curve indexed to the reference nil ductility temperature (RT_{NDT}) presented in Appendix G, "Protection Against Non-Ductile Failure," of Section III of the ASME Code (2). Further, the materials in the beltline region of the reactor vessel must be monitored for radiation-induced changes in RT_{NDT} per the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10CFR50.

The RT_{NDT} must be established for all materials, including weld metal and heat-affected zone (HAZ) material as well as base plates and forgings, which comprise the reactor coolant pressure boundary.

It is well established that ferritic materials undergo an increase in strength and hardness and a decrease in ductility and toughness when exposed to neutron fluences in excess of 10^{17} neutrons per cm^2 (E > 1 MeV) (4). Also, it has been established that tramp elements, particularly copper and nickel, affect the radiation embrittlement response of ferritic materials (5-7). The relationship between increase in RT_{NDT} and copper content is defined in Regulatory Guide 1.99, Rev. 1 and Rev. 2. Estimates of shifts in RT_{NDT} in this report are based on the May 1988 version of Revision 2 of Regulatory Guide 1.99 (8).

In general, the only ferritic pressure boundary materials in a nuclear plant which are expected to receive a fluence sufficient to affect RT_{NDT} are those materials which are located in the core beltline region of the reactor pressure vessel. Therefore, material surveillance programs include specimens machined from the plate or forging material and weldments which are located in the core beltline region of high neutron flux density to provide the data required to assess the degree of neutron embrittlement. ASTM E 185 (9) describes the recommended practice for monitoring and evaluating the radiation-induced changes occurring in the mechanical properties of pressure vessel beltline materials.

Westinghouse has provided such a surveillance program for the Indian Point Unit No. 2 nuclear power plant (10). The encapsulated C_V specimens are located on the O.D. surface of the thermal shield where the fast neutron flux density is 1.08 times that at the adjacent vessel wall surface (at 4° for Capsule V, see Table IV-2) (17). Therefore, the increases (shifts) in transition temperatures of the materials in the pressure vessel are slightly less than the corresponding shifts observed in the surveillance specimens. However, because of azimuthal variations in neutron flux density, capsule fluences may lead or lag the maximum vessel fluence in a corresponding exposure period. The capsules also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

The Indian Point Unit No. 2 material surveillance capsules also include tensile specimens as recommended by ASTM E 185. At the present time, irradiated tensile properties are used only to indicate that the materials tested continue to meet the requirements of the appropriate material specification. In addition, the material surveillance capsules contain wedge opening loading (WOL) fracture mechanics specimens. Current technology limits the testing of these specimens at temperatures well below the minimum service temperature to obtain valid fracture mechanics data per ASTM E 399 (11), "Standard Method of Test for Plane-Strain Fracture Toughness of Metallic Materials." Currently, the NRC suggests storing these specimens until an acceptable testing procedure has been defined for determining the J_{Ic} fracture toughness (12).

This report describes the results obtained from testing the contents of Capsule V. These data and those obtained previously from Capsules T, Y, and Z (13-15) are analyzed to estimate the radiation-induced changes in the mechanical properties of the pressure vessel at the end of Cycle 8 as well as predicting the changes expected to occur at selected times in the future operation of the Indian Point Unit No. 2 power plant. The future projections are based on the continued use of a low leakage core loading pattern, put in service at the start of Cycle 6, which involves placing burnt assemblies at the periphery and minimal fresh assemblies instead of all fresh assemblies at the periphery so that the peak vessel wall neutron flux is reduced by approximately 45 to 50 percent.

III. DESCRIPTION OF MATERIAL SURVEILLANCE PROGRAM

The Indian Point Unit No. 2 material surveillance program is described in detail in WCAP 7323 (10), dated May 1969. Eight materials surveillance capsules (five Type I and three Type II) were placed in the reactor vessel between the thermal shield and the vessel wall before startup (see Figures III-1 and III-2). The vertical center of each capsule is opposite the vertical center of the core. The neutron flux density at each 4° capsule location slightly exceeds 1.00 times the maximum flux density on the vessel I.D. (17). However, the peak vessel exposure rate has been significantly reduced since the introduction of a low leakage core loading pattern in Cycle 6.

Capsule V, a Type II capsule, was removed during the 1987 refueling outage. The Type II capsules each contain Charpy V-notch, tensile, and WOL specimens machined from the three SA533 Gr B, Cl 1 beltline shell plates plus Charpy V-notch specimens machined from a correlation monitor heat of steel. The chemistries and heat treatments of the vessel surveillance materials are summarized in Table III-1. All test specimens were machined from the test materials at the quarter-thickness (1/4T) location. The longitudinal base metal C_V specimens were oriented with their long axis parallel to the primary rolling direction and with V-notches perpendicular to the major plate surfaces. Tensile specimens were machined with the longitudinal axis parallel to the plate primary rolling direction. The WOL specimens were machined with the simulated crack perpendicular to the primary rolling direction and to the major plate surfaces. All mechanical test specimens (see Figure III-3) were taken at least one plate thickness from the quenched edges of the plate material.

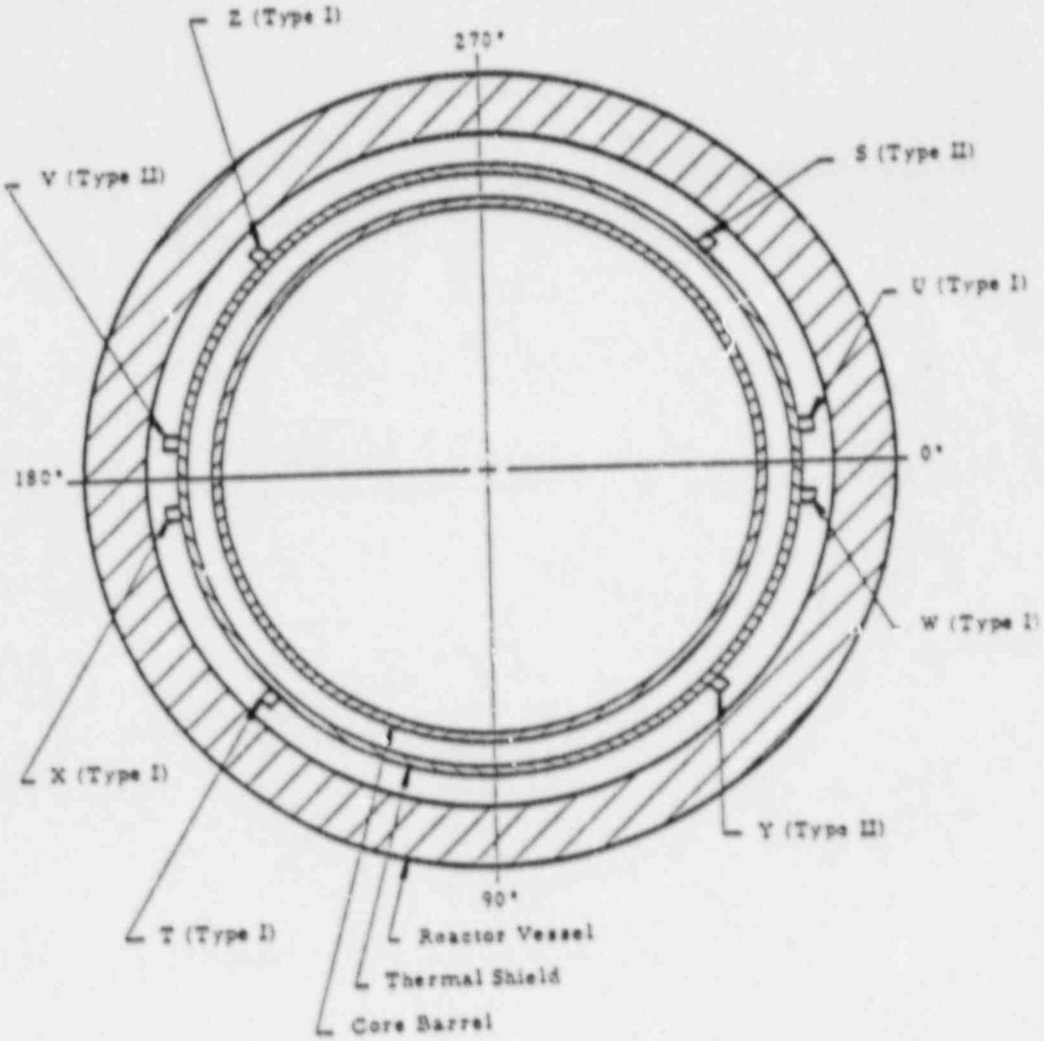


Figure III-1. Arrangement of surveillance capsules in the pressure vessel

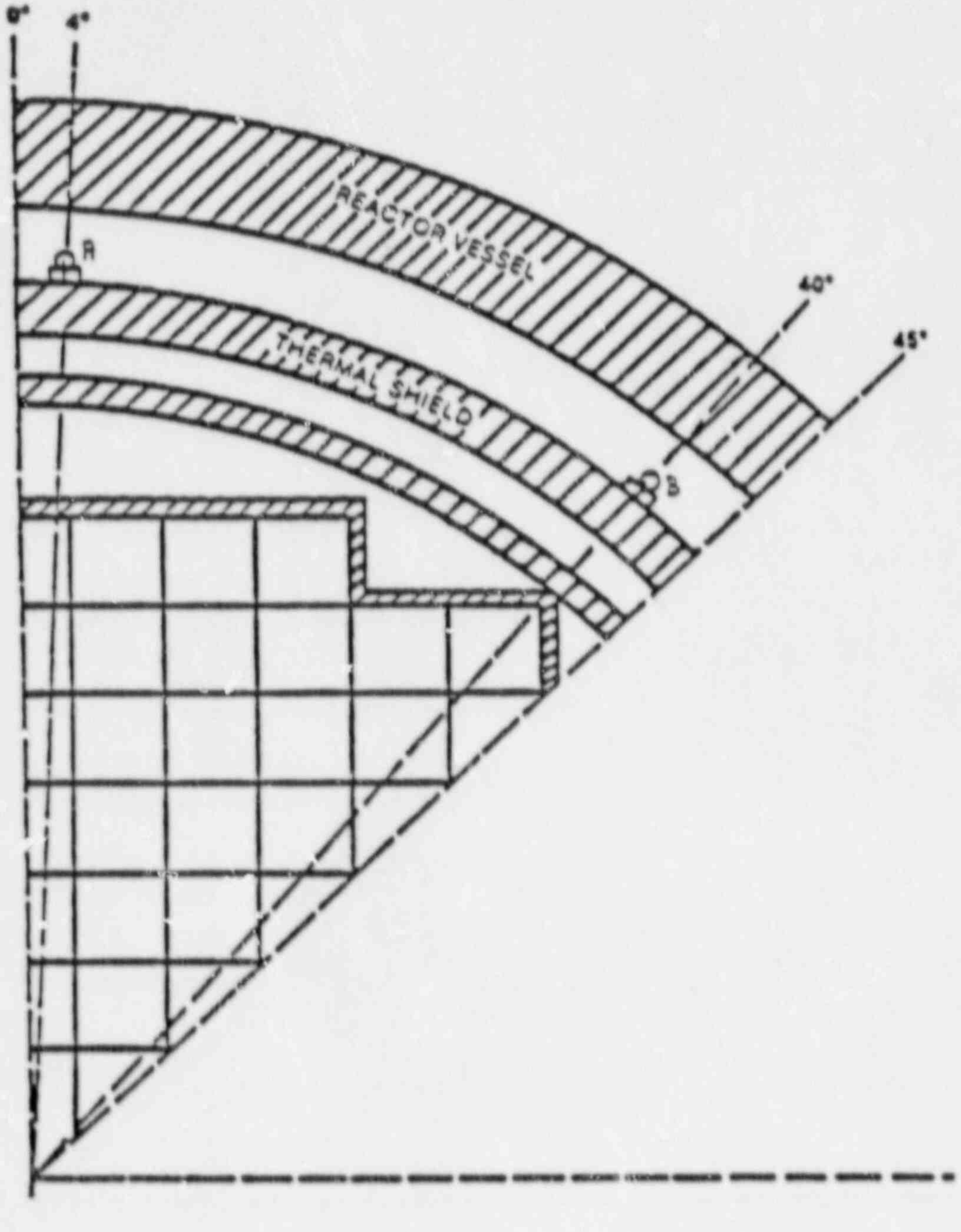


Figure III-2. Indian Point Unit 2 reactor geometry
(Reference ?)

Table III-1

INDIAN POINT UNIT NO. 2 REACTOR VESSEL SURVEILLANCE MATERIALS (10)

Heat Treatment History

Shell Plate Material:

Heated to 1550-1600°F for 4 hours, water quenched.
Tempered at 1225°F for 4 hours, air cooled.
Stress relieved at 1150°F for 40 hours, furnace cooled to 600°F

Weldment:

Stress relieved at 1150°F for 19.75 hours, furnace cooled to 600°F

Correlation Monitor:

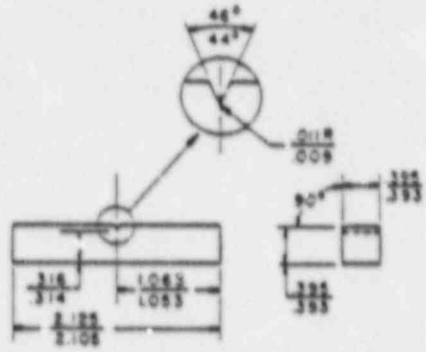
1650°F, 4 hours, water quenched to 300°F
1200°F, 6 hours, air cooled.

Chemical Composition (Percent)

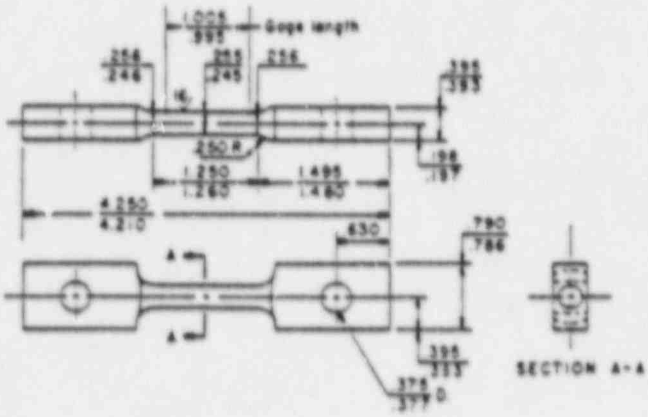
Material	C	Mn	P	S	Si	Ni	Mo	Cu
Plate B2002-1	0.20	1.28	0.010	0.019	0.25	0.58	0.46	0.25
Plate B2002-2	0.22	1.30	0.014	0.018	0.22	0.46	0.50	0.14
Plate B2002-3	0.22	1.29	0.011	0.020	0.25	0.57	0.46	0.14
Correlation Monitor	0.24	1.34	0.011	0.023	0.23	(a)	0.51	(a)
Weld Metal	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)

(a) Not reported in WCAP 7323 (10).

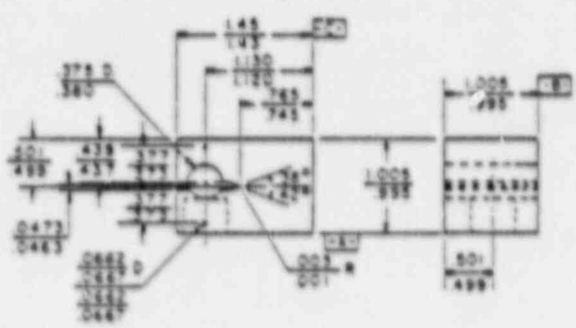
Capsule V contained 32 Charpy V-notched specimens, 4 tensile specimens (2 from weld metal and 2 from plate), and 4 base plate WOL specimens. The specimen numbering system and location within Capsule V is shown in Figures III-4 and III-5.



(a) Charpy V-notch Impact Specimen



(b) Tensile Specimen



(c) Wedge Opening Loading Specimen

Figure III-3. Vessel material surveillance specimens

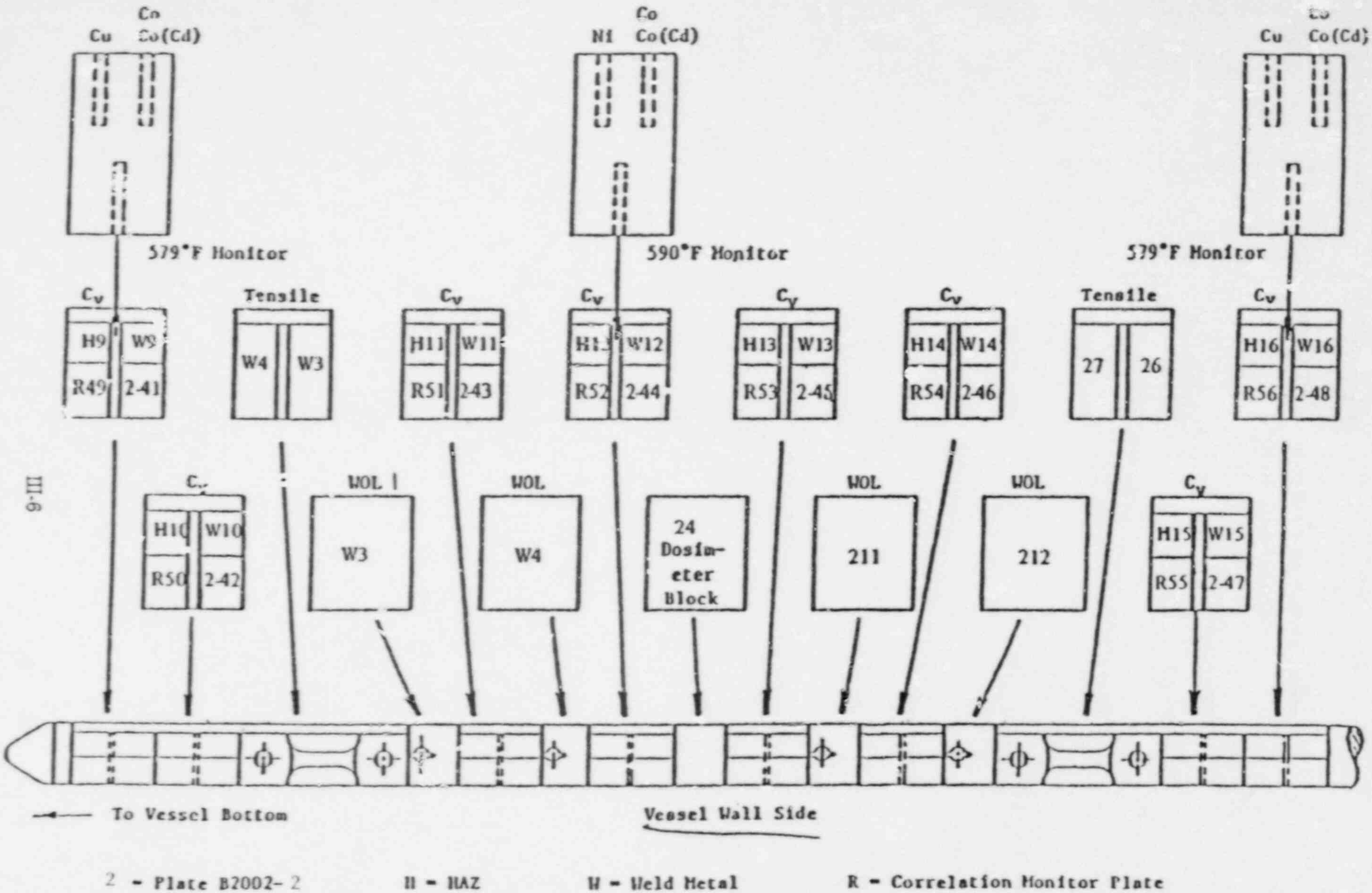


Figure III-4. Arrangement of specimens in Capsule V

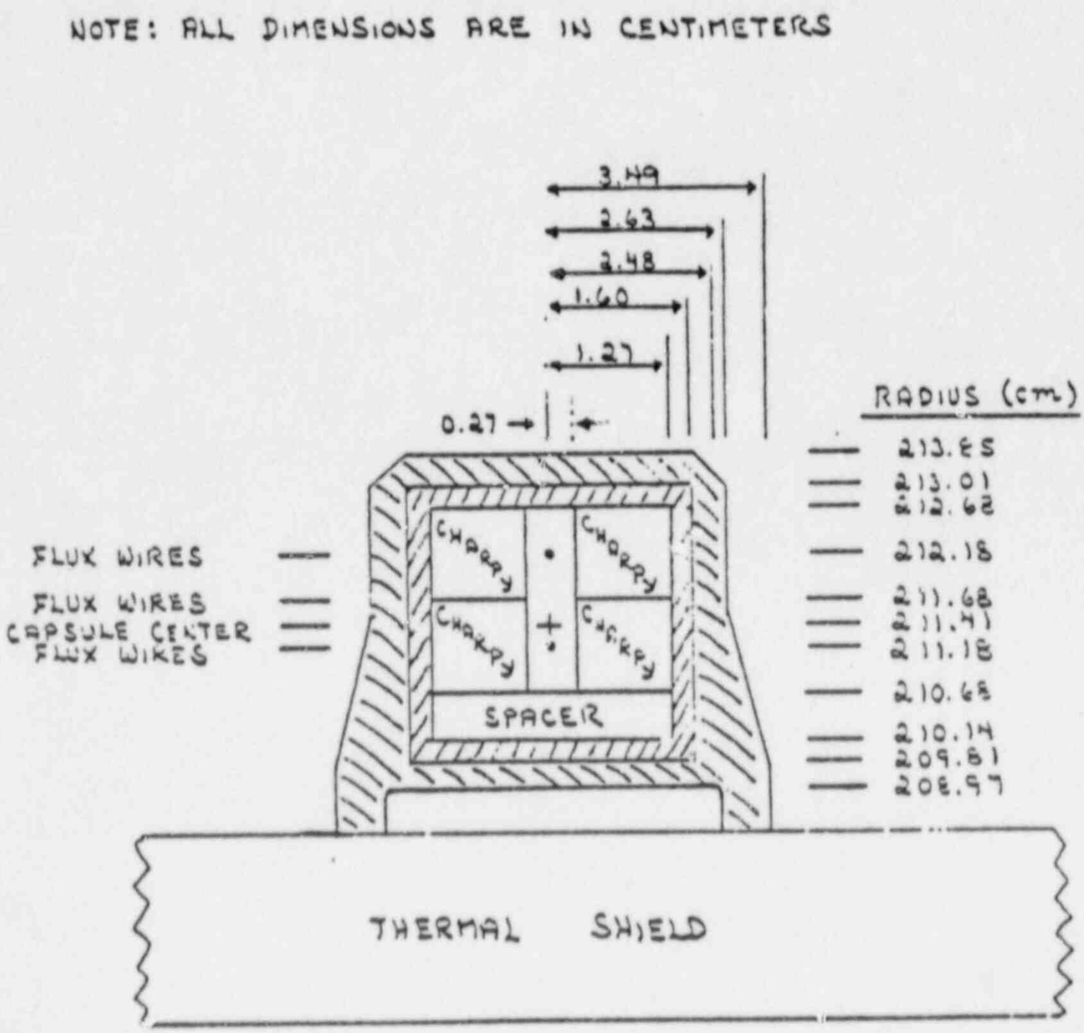


Figure III-5. Surveillance capsule geometry

(Reference 17)

Capsule V also contained the following dosimeters for determining the neutron flux density:

Table III-2

CAPSULE V NEUTRON FLUX DOSIMETERS

Target Element	Form	Quantity
Copper	Bare wire	[illegible]
Nickel	Bare wire	[illegible]
Cobalt (in aluminum)	Bare wire	[illegible]
Cobalt (in aluminum)	Cd shielded wire	3
Uranium	Oxide	1
Neptunium	Oxide	1

In addition, ends were cut from 10 tested Charpy specimens to serve as iron dosimeters.

Three eutectic alloy thermal monitors had been inserted in holes in the steel spacers in Capsule V. Two (located at the top and bottom) were 2.5% Ag and 97.5% Pb with a melting point of 579°F. The other (located at the center of the capsule) was 1.75% Ag, 0.75% Sn, and 97.5% Pb having a melting point of 590°F.

IV. TESTING OF SPECIMENS FROM CAPSULE V

The capsule shipment, capsule opening, specimen testing, and reporting of results were carried out in accordance with the Project Plan for Indian Point Unit No. 2 Reactor Vessel Irradiation Surveillance Program. The SwRI Nuclear Projects Operating Procedures called out in this plan include:

(1) XIII-MS-104-1, "Shipment of Westinghouse PWR Vessel Material Surveillance Capsule Using SwRI Cask and Equipment"

(2) XI-MS-101-1, "Determination of Specific Activity and Analysis of Radiation Detector Specimens"

(3) XI-MS-103-1, "Conducting Tension Tests on Metallic Specimens"

(4) XI-MS-104-1, "Charpy Impact Tests on Metallic Specimens"

(5) XIII-MS-103-1, "Opening Radiation Surveillance Capsules and Handling and Storing Specimens"

Copies of the above documents are on file at SwRI.

A. Shipment, Opening, and Inspection of Capsule

Southwest Research Institute utilized Nuclear Projects Operating Procedure XIII-MS-104-1, as incorporated in approved Consolidated Edison Co. procedures, for the shipment of Capsule V to the SwRI laboratories. On March 30, 1988, SwRI personnel severed the capsule from its extension tube, sectioned the extension tube into several lengths, supervised the loading of the capsule and extension tube materials into the shipping cask, and transported the cask to San Antonio, Texas. The capsule arrived at the SwRI Radiation Laboratory on April 5, 1988, and unloading of the capsule commenced the next day.

The capsule was opened and the contents identified and stored in accordance with Procedure XIII-MS-103-1. The long seam welds were milled off using a Bridgeport vertical milling machine. Before milling the long seam weld beads, transverse saw cuts were made to remove the capsule ends. After the long seam welds had been milled off, the top half of the capsule shell was removed. The specimens and spacer blocks were carefully removed and placed in indexed receptacles identifying each capsule location. After the disassembly had been completed, each specimen was carefully checked to insure agreement with the identification and location as listed in WCAP 7323 (10). The following discrepancies were found and corrected:

> Two Charpies were both marked R-55 on one end and R-56 on the other end. The Charpy that was in the R-55 position was remarked properly on the other end and the R-56 Charpy was also remarked by crossing out the R-55 and remarking the end as R-56.

The thermal monitors and neutron dosimeter wires were removed from the holes in the spacers. The thermal monitors, contained in quartz vials, were examined. No evidence of melting was observed, thus indicating that the maximum temperature during exposure of Capsule V did not exceed 579°F. All neutron dosimeters were in the positions called out in WCAP 7323 and were correctly accounted for. However, the Neptunium container had an appearance that had not been encountered before. The Uranium and Neptunium containers are n in Figure IV-1. The deformed condition of the Neptunium container caused the loss of st of the sample during opening.

B. Neutron Dosimetry

The dosimeter wires were weighed on a Mettler microbalance, and the Charpy slices were weighed on a Mettler digital balance. The gamma activities of the dosimeters were determined in accordance with Procedure XI-MS-101-1 using an IT-5400 multichannel analyzer and an intrinsic Ge coaxial detector system. The calibration of the equipment was accomplished with ^{54}Mn, ^{60}Co, and ^{137}Cs radioactivity standards obtained from the U.S. Department of Commerce National Bureau of Standards. All activities were corrected to the time-of-removal (TOR) at reactor shutdown.

Infinitely dilute saturated activities (A_{SAT}) were calculated for each of the dosimeters because A_{SAT} is directly related to the product of the energy-dependent microscopic activation cross section and the neutron flux density. The relationship between A_{TOR} and A_{SAT} is given by:

$$\frac{A_{TOR}}{A_{SAT}} = \sum_{m=1}^{m=n} P_m \left(1-e^{-\lambda T_m}\right) e^{-\lambda t_m}$$

where: λ = decay constant for the activation product, day $^{-1}$;

t_m = decay time after operating period m, days;

T_m = operating days;

P_m = average fraction of full power during operating period.

The values of T_m and P_m up to the 1987 refueling shutdown for Indian Point Unit No. 2 are presented in Table IV-1. The calculation of the neutronic factors is described below.

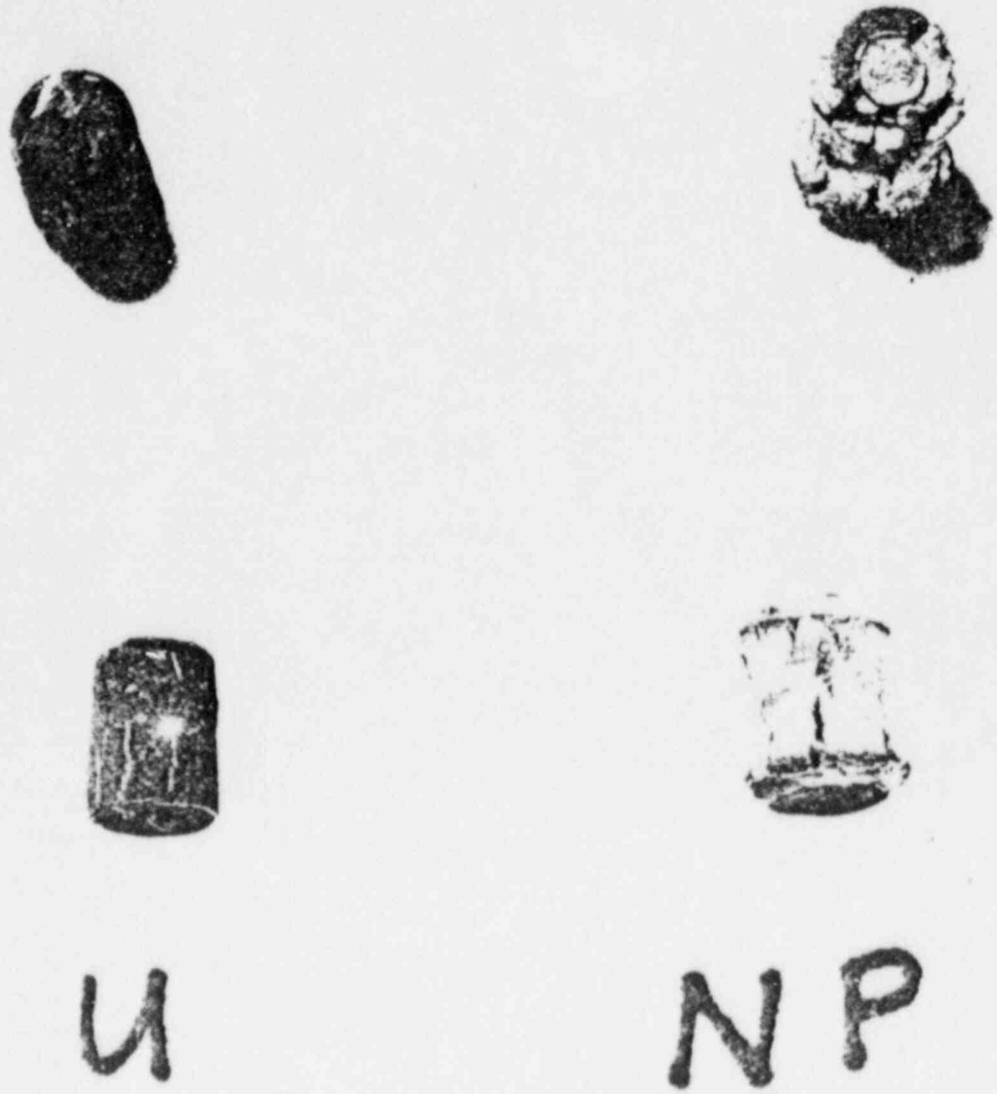


Figure IV-1. Uranium and Neptunium containers as removed from dosimeter block

Table IV-1

SUMMARY OF REACTOR OPERATIONS
INDIAN POINT UNIT NO. 2

Operating Period	Dates Start	Dates Stop	Operating Days (T_m)	Shutdown Days	Fraction of Full Power (P_m)
1	08/15/73	08/24/73	10	--	0.4377
	08/25/73	08/25/73	--	1	--
2	08/26/73	09/07/73	13	--	0.4532
	09/08/73	09/20/73	--	13	--
3	09/21/73	09/28/73	8	--	0.3161
	09/29/73	09/30/73	--	2	--
4	10/01/73	10/12/73	12	--	0.3088
	10/13/73	01/25/74	--	105	--
5	01/26/74	01/29/74	4	--	0.2412
	01/30/74	03/21/74	--	51	--
6	03/22/74	04/18/74	28	--	0.5438
	04/19/74	04/28/74	--	10	--
7	04/29/74	05/03/74	5	--	0.4962
	05/04/74	05/04/74	--	1	--
8	05/05/74	05/10/74	6	--	0.4743
	05/11/74	05/12/74	--	2	--
9	05/13/74	05/13/74	1	--	0.0730
	05/14/74	05/20/74	--	7	--
10	05/21/74	06/14/74	25	--	0.6653
	06/15/74	06/16/74	--	2	--
11	06/17/74	07/22/74	36	--	0.7691
	07/23/74	07/23/74	--	1	--
12	07/24/74	07/26/74	3	--	0.7593
	07/27/74	08/05/74	--	10	--
13	08/06/74	09/06/74	32	--	0.6653
	09/07/74	09/09/74	--	3	--
14	09/10/74	09/30/74	21	--	0.7429
	10/01/74	10/11/74	--	11	--
15	10/12/74	11/09/74	29	--	0.8637
	11/[illegible]/74	11/10/74	--	1	--
16	11/11/74	12/06/74	26	--	0.8306
	12/07/74	12/07/74	--	1	--
17	12/08/74	01/01/75	25	--	0.8495
	01/02/75	01/04/75	--	3	--
18	01/05/75	01/05/75	1	--	0.5450
	01/06/75	01/06/75	--	1	--
19	01/07/75	01/31/75	25	-	0.8810
	02/01/75	02/02/75	--	2	--
20	02/03/75	02/28/75	26	--	0.9408
	03/01/75	04/03/75	--	34	--
21	04/04/75	05/02/75	29	--	0.7632
	05/03/75	05/03/75	--	1	--
22	05/04/75	07/28/75	86	--	0.9114
	07/29/75	08/10/75	--	13	--
23	08/11/75	09/12/75	33	--	0.7108
	09/13/75	09/13/75	--	1	--
24	09/14/75	10/16/75	33	--	0.7962
	10/17/75	10/29/75	--	13	--
25	10/30/75	11/14/75	16	--	0.7407
	11/15/75	11/15/75	--	1	--
26	11/16/75	01/04/76	50	--	0.8427
	01/05/76	01/05/76	--	1	--
27	01/06/76	01/29/76	24	--	0.8703
	01/03/76	02/04/76	--	6	--
28	02/05/76	03/30/76	55	--	0.9122
	03/31/76	09/26/76	--	180	--
29	09/27/76	09/27/76	1	--	0.0680
	09/28/76	09/28/76	--	1	--
30	09/29/76	10/29/76	31	--	0.8423
	10/30/76	12/10/76	--	42	--

Table IV-1

SUMMARY OF REACTOR OPERATIONS
INDIAN POINT UNIT NO. 2 (CONT'D)

Operating Period	Dates Start	Dates Stop	Operating Days (T_m)	Shutdown Days	Fraction of Full Power (P_m)
31	12/11/76	01/27/77	48	--	0.8396
	01/28/77	01/29/77	--	2	--
32	01/30/77	02/01/77	3	--	0.7250
	02/02/77	02/05/77	--	4	--
33	02/06/77	03/11/77	34	--	0.8825
	03/12/77	03/14/77	--	3	--
34	03/15/77	04/10/77	27	--	0.9242
	04/11/77	05/13/77	--	33	--
35	05/14/77	07/02/77	50	--	0.8936
	07/03/77	08/05/77	--	34	--
36	08/06/77	08/19/77	14	--	0.6372
	08/20/77	08/21/77	--	2	--
37	08/22/77	02/13/78	176	--	0.9022
	02/14/78	05/24/78	--	100	--
38	05/25/7ĸ	07/28/78	65	--	0.8960
	07/29/78	07/30/78	--	2	--
39	07/31/78	09/15/78	47	--	0.9820
	09/16/78	10/05/78	--	20	--
40	10/06/78	11/23/78	49	--	0.9360
	11/24/78	12/02/78	--	9	--
41	12/03/78	06/15/79	195	--	0.9690
	06/16/79	09/14/79	--	91	--
42	09/15/79	11/27/79	74	--	0.8120
	11/28/79	11/29/79	--	2	--
43	11/30/79	12/02/79	3	--	0.1840
	12/03/79	12/07/79	--	5	--
44	12/08/79	01/11/80	35	--	0.8710
	01/12/78	02/09/80	--	29	--
45	02/10/80	02/14/80	5	--	0.4200
	02/15/80	02/18/80	--	4	--
46	02/19/80	06/03/80	106	--	0.9310
	06/04/80	06/11/80	--	8	--
47	06/12/80	08/10/80	60	--	0.9310
	08/11/80	08/13/80	--	3	--
48	08/14/80	10/17/80	65	--	0.9400
	10/18/80	05/21/81	--	216	--
49	05/22/81	07/10/81	50	--	0.7120
	07/11/81	07/11/81	--	1	--
50	07/12/81	08/21/81	41	--	0.9640
	08/22/81	09/15/81	--	[illegible]5	--
51	09/16/81	10/05/81	20	--	0.9040
	10/06/81	10/15/81	--	10	--
52	10/16/81	11/11/81	27	--	0.9710
	12/12/81	11/22/81	--	11	--
53	11/23/81	04/02/82	131	--	0.9590
	04/03/82	04/03/82	--	1	--
54	04/04/82	05/17/82	44	--	0.9230
	05/18/82	05/23/82	--	6	--
55	05/24/82	08/12/82	81	--	0.9520
	08/13/82	08/14/82	--	2	--
56	08/15/82	09/02/82	19	--	0.7890
	09/03/82	09/07/82	--	5	--
57	09/08/82	09/17/82	10	--	0.7980
	09/18/82	01/01/83	--	106	--
58	01/02/83	01/05/83	4	--	0.3485
	01/06/83	01/06/83	--	1	--
59	01/07/83	01/08/83	2	--	0.0355
	01/09/83	01/10/83	--	2	--
60	01/11/83	01/31/83	21	--	0.7393
	02/01/83	02/11/83	--	11	--

Table IV-1

SUMMARY OF REACTOR OPERATIONS
INDIAN POINT UNIT NO. 2 (CONT'D)

Operating Period	Dates Start	Dates Stop	Operating Days (T_m)	Shutdown Days	Fraction of Full Power (P_m)
61	02/12/83	02/13/83	2	--	0.0090
	02/14/83	02/14/83	--	1	--
62	02/15/83	02/18/83	4	--	0.1025
	02/19/83	02/19/83	--	1	--
63	02/20/83	08/27/83	189	--	0.9619
	08/28/83	08/28/83	--	1	--
64	08/29/83	10/04/83	37	--	0.9572
	10/05/83	10/25/83	--	21	--
65	10/26/83	01/05/84	72	--	0.9248
	01/06/84	01/07/84	--	2	--
66	01/08/84	02/11/84	35	--	0.9228
	02/12/84	02/26/84	--	15	--
67	02/27/84	06/01/84	96	--	0.9100
	06/02/84	10/20/84	--	141	--
68	10/21/84	11/30/84	41	--	0.8706
	12/01/84	12/01/84	--	1	--
69	12/02/84	12/19/84	18	--	0.9147
	12/20/84	12/26/84	--	7	--
70	12/27/84	12/28/84	2	--	0.0060
	12/28/84	12/31/84	--	3	--
71	01/01/85	09/20/85	263	--	0.9509
	09/21/85	09/22/85	--	2	--
72	09/23/85	10/21/85	29	--	0.6813
	10/22/85	10/23/85	--	2	--
73	10/24/85	01/13/86	82	--	0.9298
	01/14/86	05/24/86	--	131	--
74	05/25/86	05/28/86	4	--	0.1688
	05/29/86	05/29/86	--	1	--
75	05/30/86	05/31/86	2	--	0.2885
	06/01/86	06/06/86	--	6	--
76	06/07/86	06/09/86	3	--	0.1020
	06/10/86	06/10/86	--	1	--
77	06/11/86	10/20/86	132	--	0.9339
	10/21/86	10/22/86	--	2	--
78	10/23/86	10/23/86	1	--	0.0710
	10/24/86	10/26/86	--	3	--
79	10/27/86	11/06/86	11	--	0.9146
	11/07/86	11/08/86	--	2	--
80	11/09/86	11/15/86	7	--	0.7864
	11/16/86	11/16/86	--	1	--
81	11/17/86	01/30/87	75	--	0.9393
	01/31/77	02/06/87	--	7	--
82	02/07/87	02/10/87	4	--	0.7058
	02/11/87	02/12/87	--	2	--
83	02/13/87	06/27/87	135	--	0.9804
	06/28/87	06/29/87	--	2	--
84	06/30/87	10/04/87	97	--	0.9810

Westinghouse performed a two-dimensional ordinates S_n transport analysis to determine the neutron fluxes and energy spectrum within the reactor vessel and surveillance capsule of Indian Point Unit 2. This analysis was undertaken to calculate the spectrum averaged cross sections for the threshold and the fission detectors, the lead factors for use in relative neutron exposure of the pressure vessel to that of the surveillance capsule and iron atom displacement (DPA).

Westinghouse undertook two distinct calculations for the Indian Point Unit 2 reactor pressure vessel. First was a single computation in the conventional forward mode to obtain relative neutron energy distributions throughout the reactor geometry as well as through the vessel wall. This transport calculation was carried out in R, θ geometry using the DOT two-dimensional discrete ordinates code and the SAILOR cross-section library. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In this calculation P_3 anisotropic scattering and S_8 order of angular quadrature was used. The reference forward calculations was normalized to a core mid-plane power density characteristic of operation at a thermal power level of 2758 MWt.

The second calculation consisted of a series of adjoint analysis relating the fast neutron flux ($E > 1.0$ MeV) at surveillance capsule positions and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. All adjoint analyses were also carried out using an S_8 order of angular quadrature and P_3 anisotropic scattering using the 47 group SAILOR Library as described above.

The core power distributions for each cycle used in fast neutron exposure evaluation were taken from Indian Point Unit 2 nuclear design reports.

The pertinent factors (i) calculated spectrum averaged reaction cross sections and (ii) calculated cycle dependent fluence lead factors obtained from these transport calculations are summarized in Table IV-2. The calculated spectrum averaged reaction cross sections are employed in the analysis of fast utron monitors activity data for the prediction of fast neutron flux/fluence (E > 1.0 MeV) at surveillance capsule location and the calculated lead factors for the prediction of reactor vessel flux/fluence (E > 1.0 MeV) from the surveillance. Neutron Cycle 5 lead factor results given in Table IV-2 are representative of a standard loading pattern cycle as Indian Point Unit 2 employed this loading pattern from Cycle 1 through Cycle 5. Cycle 8 results are for the low leakage loading pattern as the low leakage loading pattern was implemented at Indian Point Unit 2 starting from Cycle 6.

Table IV-2

RESULTS OF DISCRETE ORDINATES Sn TRANSPORT ANALYSIS (17)
INDIAN POINT UNIT NO. 2

A. **Calculated Spectrum-Averaged Reaction Cross Sections (σ_{eff}) for Analysis of Fast Neutron Monitors (E > 1.0 MeV)**

Reaction	(barns)	
	4°	40°
$^{54}Fe(n,p)^{54}Mn$	0.0887	0.067
$^{58}Ni(n,p)^{58}Co$	0.116	0.0914
$^{63}Cu(n,\alpha)^{60}Co$	0.00119	0.000694
$^{238}U(n,f)^{137}Cs$	0.372	0.343
$^{237}Np(n,f)^{137}Cs$	2.63	2.84

B. **Calculated Fluence Lead Factors[a] for Indian Point-2 Cycles 5 and 8**

Cycle	4°	40°
5	1.08	3.42
8	1.19	3.40

$$^{(a)}\text{L.F} = \frac{\text{EOC Fluence at Surveillance Location}}{\text{EOC Fluence at RPV O-T Location}}$$

The primary result desired from the dosimeter analysis is the total neutron fluence (E > 1 MeV) which the surveillance specim· ns and pressure vessel have received. The average flux a full power is given by:

$$\phi = A_{SAT}/N_o\bar{\sigma}$$

A_{SAT} = Saturated activity (rate of decay = rate of production) in disintegration/sec or Bq

where ϕ = energy dependent neutron flux, n/cm^2 sec

$\bar{\sigma}$ = spectrum-averaged activation cross section, cm^2; and

N_o = number of target atoms per mg.

The total neutron fluence is then equal to the product of the average neutron flux and the equivalent reactor operating time at full power.

In Capsule V, the Correlation Monitor and B2002-2 shell plate Charpy specimens were located in the specimen layer nearest to the vessel wall and the weld metal, heat-affected zone (HAZ) Charpy specimens were located in the specimen layer nearest to the core. Since there is a radial dependence of the fast neutron flux in the vessel, the neutron exposure received by the Correlation Monitor and B2002-2 shell plate Charpy specimens is expected to be lower than that received by the weld metal and HAZ Charpy specimens. The dosimetry program is capable of providing information on the radial dependence of the fast flux because the Charpy ends used for iron dosimetry were taken from both of the Charpy specimen layers (nearest to and farthest from the core).

Since Indian Point Unit No. 2 operated for 8.6 effective full power years (EFPY) up to the 1987 refueling outage, the calculated fluences for Capsule V and the vessel up to the 1987 refueling outage are as presented in Table IV-3. Thermal neutron flux (fluence rate) values from Capsule V are presented in Table IV-4.

Table IV-3

DOSIMETER ACTIVITIES AND MEASURED FLUENCE RATE IN CAPSULE V

Position	Dosimeter ID	A_{TOR} (Bq/Mg)	A_{SAT} (Bq/Mg)		Measured ϕ (> 1 MeV)[a] (n cm^{-2} sec^{-1})
R=211.18 (Core Side of Charpy Compartment):					
	Ni	1[illegible]5.4	16860.0		2 08E10
Bottom	Cu	76.8	138.6		1.76E10
Top	Cu	79.1	142.8		1.82E10
Bottom	Fe W-9	670.2	842.4		1.52E10
Bottom	Fe W-12	681.1	856.1		1.54E10
Bottom	Fe H-12	717.7	902.1		1.63E10
Bottom	Fe W-13	667.7	839.1		1.51E10
Top	Fe H-16	751.3	944.1		1.70E10
				Ave:	1.70E10±1.9E9
R=211.68:					
	238 U	239.1	1398.3		2.47E10
	237 Np	(9820)	(5740)		(1.31E11)

NOTE: Np Results are not reliable because an inadequate sample was recovered (see comments in text)

Position	Dosimeter ID	A_{TOR} (Bq/Mg)	A_{SAT} (Bq/Mg)		Measured ϕ (> 1 MeV)[a] (n cm^{-2} sec^{-1})
R=212.18 (Vessel Side of Charpy Compartment):					
Bottom	Fe 2-41	571.9	718.8		1.30E10
Bottom	Fe 2-44	582.0	731.5		1.32E10
Bottom	Fe R-52	615.6	773.8		1.39E10
Bottom	Fe 2-45	565.8	711.2		1.28E10
Top	Fe R-56	622.3	782.2		1.41E10
				Ave:	1.34E10±6.0E8

[a] $$\text{Measured } \phi\ (> 1\ \text{MeV}) = \frac{A_{SAT}}{N_o\,\sigma_{eff}} = \frac{(A_{TOR}/h)}{N_o\,\sigma_{eff}}$$

Table IV-3 (Cont'd)

DOSIMETER ACTIVITIES AND MEASURED FLUENCE RATE IN CAPSULE V

Determination of Fluence Rate at Centerline of
Surveillance Capsule V, Indian Point-2

Radial Position	Dosimeter ID	Dosimeter ϕ (> 1 MeV) n/cm^2 sec	Gradient Factor	Centerline ϕ (> 1 MeV) n/cm^2 sec
211.18	Ni	2.08E10	0.953	1.98E10
	Cu (bottom)	1.76E10	0.956	1.68E10
	Cu (Top)	1.82E10	0.956	1.74E10
	Fe W-9	1.52E10	0.951	1.45E10
	Fe W-12	1.54E10	0.951	1.46E10
	Fe H-12	1.63E10	0.951	1.55E10
	FeW-13	1.51E10	0.951	1.44E10
	Fe H-16	1.70E10	0.951	1.62E10
211.68	^{238}U (a)	2.47E10	1.050	2.60E10
	^{237}Np (a)	1.37E11	1.049	1.44E11
212.18	Fe 2-41	1.30E10	1.152	1.50E10
	Fe 2-44	1.32E10	1.152	1.52E10
	Fe R-52	1.39E10	1.152	1.60E10
	Fe 2-45	1.28E10	1.152	1.47E10
	Fe R-56	1.41E10	1.152	1.62E10

Average (a) Fluence Rate = 1.59E10±1.5E9 at Center of Capsule V

(a) ^{238}U and ^{237}Np results not included in average

(Cs-137 half life allows influence from high leakage cores in cycles 1 through 5)

± Value is 1σ from variation of individual values included in the average

Table IV-4

THERMAL NEUTRON FLUENCE RATE IN INDIAN POINT 2, CAPSULE V

	^{59}Co Bare		^{59}Co Cd Covered		
Axial Location	A_{TOR}, Bq/Mg	A_{SAT},[a] Bq/Mg	A_{TOR}, Bq/Mg	A_{SAT},[a] Bq/Mg	Thermal Flux n/cm^2-s
Top	3.22E6	5.81E6	1.37E6	2.47E6	8.81E9
Middle	3.10E6	5.60E6	1.39E6	2.51E6	8.15E9
Bottom	3.49E6	6.30E6	1.28E6	2.31E6	1.05E10
Average	3.27E6	5.90E6	1.35E6	2.43E6	9.15E9

(a) ^{60}Co saturation factor ≡ h = .554; $A_{SAT} = A_{TOR}/h$

The variations in the peak vessel flux values (±9.4% from variations in individual values) determined from the several dosimeter materials may be attributed to the uncertainties in measurements and calculations (in the calculated spectra and in the reaction cross sections). Uranium dosimeter values are higher than others because the Cs-137 product half-life is 30.1 yr and retains some activity from the earlier higher leakage cores.

Neptunium dosimeter values are not dependable because insufficient material was recovered from the capsule. The aluminum shell containing the Neptunium was brittle and cracked open on the lathe while being opened. Most of the Neptunium oxide was not recoverable.

Averaging the results obtained from the Capsule V iron, copper, and nickel neutron dosimeters, the peak neutron flux incident on the center of Capsule V is calculated from Table IV-3 to be 1.59 x 10^{10} n/cm^2 sec, (E > 1 MeV). This is equivalent to 3.42 x 10^{10} n/cm^2 sec (E > 1 MeV) (15).

C. Mechanical Property Tests

The irradiated Charpy V-notch specimens were tested on a calibrated** SATEC Model SI-1K 240 ft-lb, 16 ft/sec impact machine in accordance with Procedure XI-MS-104-1. The test temperatures, selected to develop the ductile-brittle transition and upper shelf regions, were obtained using a liquid conditioning bath monitored with a Fluke Model 2168A digital thermometer. The Charpy V-notch impact data obtained by SwRI on the specimens contained in Capsule V are presented in Tables IV-5 through IV-8. The shifts in the Charpy V-notch transition temperatures determined for the three vessel plates and the correlation monitor are shown in Figures IV-2 through IV-5. The Capsule T (14), Capsule Y (13), and Capsule Z (15) results, included in the figures for comparison, show that Capsule V is a low lead factor, low flux capsule, as expected.

A summary of the shifts in RT_{NDT} determined at the 46 ft-lb level as specified in NUREG-0800 (18) and Appendix G to 10CFR50 (1), and the reduction in C_V upper shelf energies for each material, is presented in Table IV-9.

**Inspected and calibrated using specimens and procedures obtained from the Army Materials and Mechanics Research Center.

TABLE IV-5

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

MATERIAL - (WELD)

Date June 2, 1988

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
W- 9	74°F	24.0	.019	0	
W-10	+130	26.5	.023	20	
W-11	+180	40.5	.035	40	
W-12	+220	53.0	.048	65	
W-13	+260	62.5	.054	95	
W-14	+300	76.0	.064	95	
W-16	+325	72.5	.065	95	
W-15	+350	76.0	.067	100	

TABLE IV-6

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

Project No. 17-2108-001
Date June 2, 1988

MATERIAL - B-2002-2

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
[illegible]	74°F	17.5	.016	5	
2-42	+120	50.0	.042	15	
2-48	+150	60.5	.046	20	
2-44	+180	93.0	.059	60	
2-43	+220	111.0	.080	90	
2-45	+260	109.5	.078	100	
2-46	+300	116.0	.075	100	
2-47	+330	106.0	.067	100	

TABLE IV-7

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

Project No. 17-2108-001

Date June 2, 1988

MATERIAL - (Reference)

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
R-49	74°F	13.5	.014	5	
R-50	+130	32.0	.041	20	
R-56	+150	32.5	.033	30	
R-51	+180	50.0	.044	75	
R-52	+230	62.0	.058	95	
R-53	+270	67.5	.059	100	
R-54	+320	70.5	.064	100	
R-55	+350	72.0	.062	100	

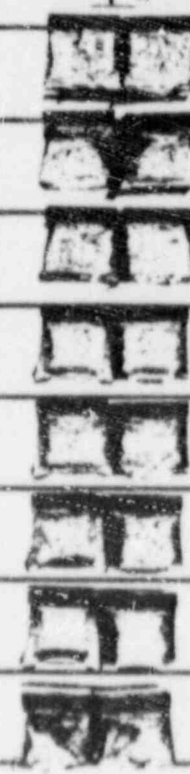
TABLE IV-8

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES

Project No. 17-2108-001
Date June 2, 1988

MATERIAL - (HAZ)

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
H-11	0	30.5	.023	25	
H-10	+30	65.0	.052	60	
H- 9	~~87~~ 74 mw	53.5	.040	50	
H-12	+110	53.5	.047	80	
H-13	+150	65.0	.053	80	
H-14	+220	93.5	.068	100	
H-16	+250	78.0	.067	40	
H-15	+280	122.5	.077	100	



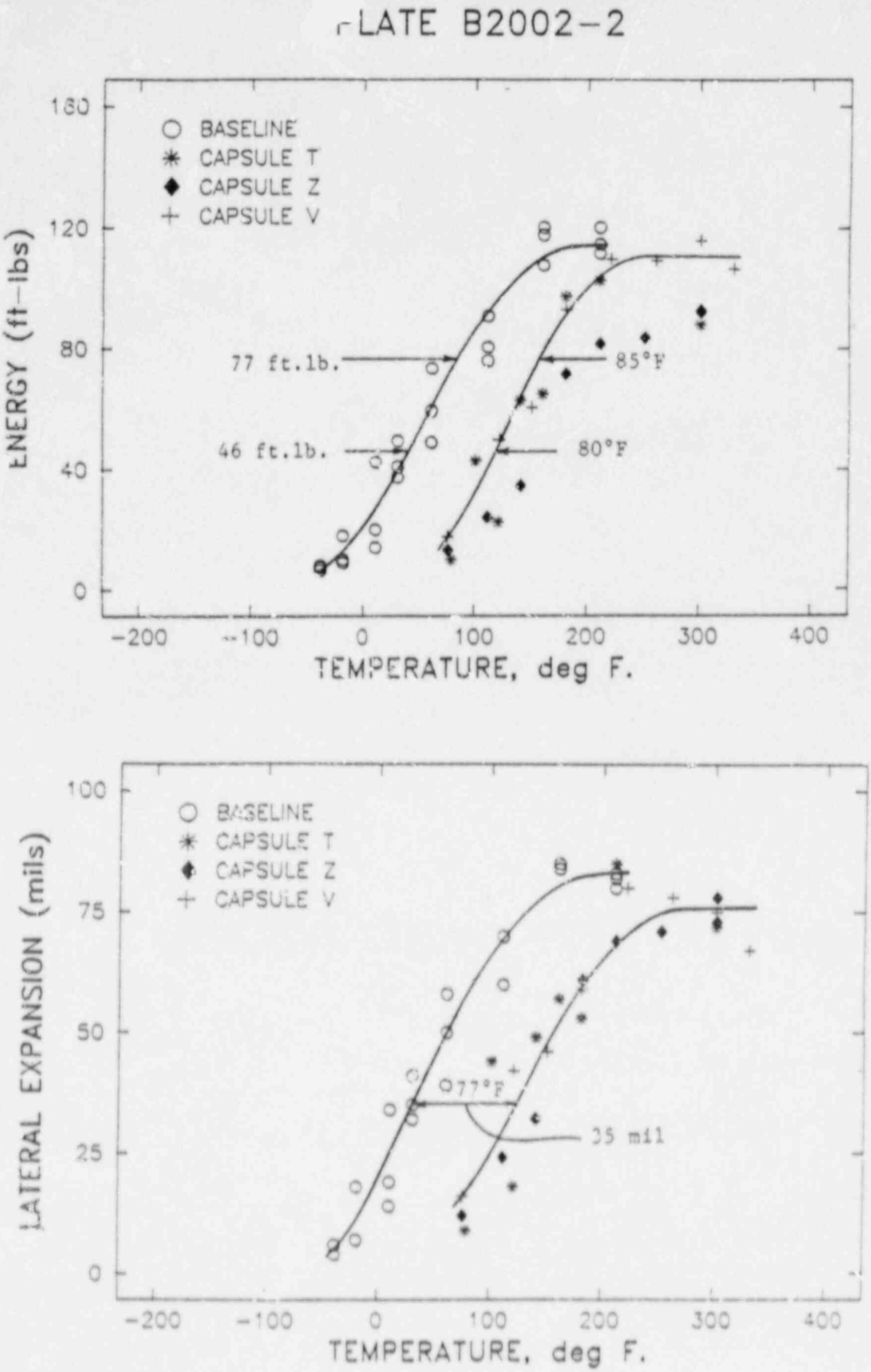


Figure IV-2. Radiation Response of Indian Point Unit 2 Shell Plate B2002-2

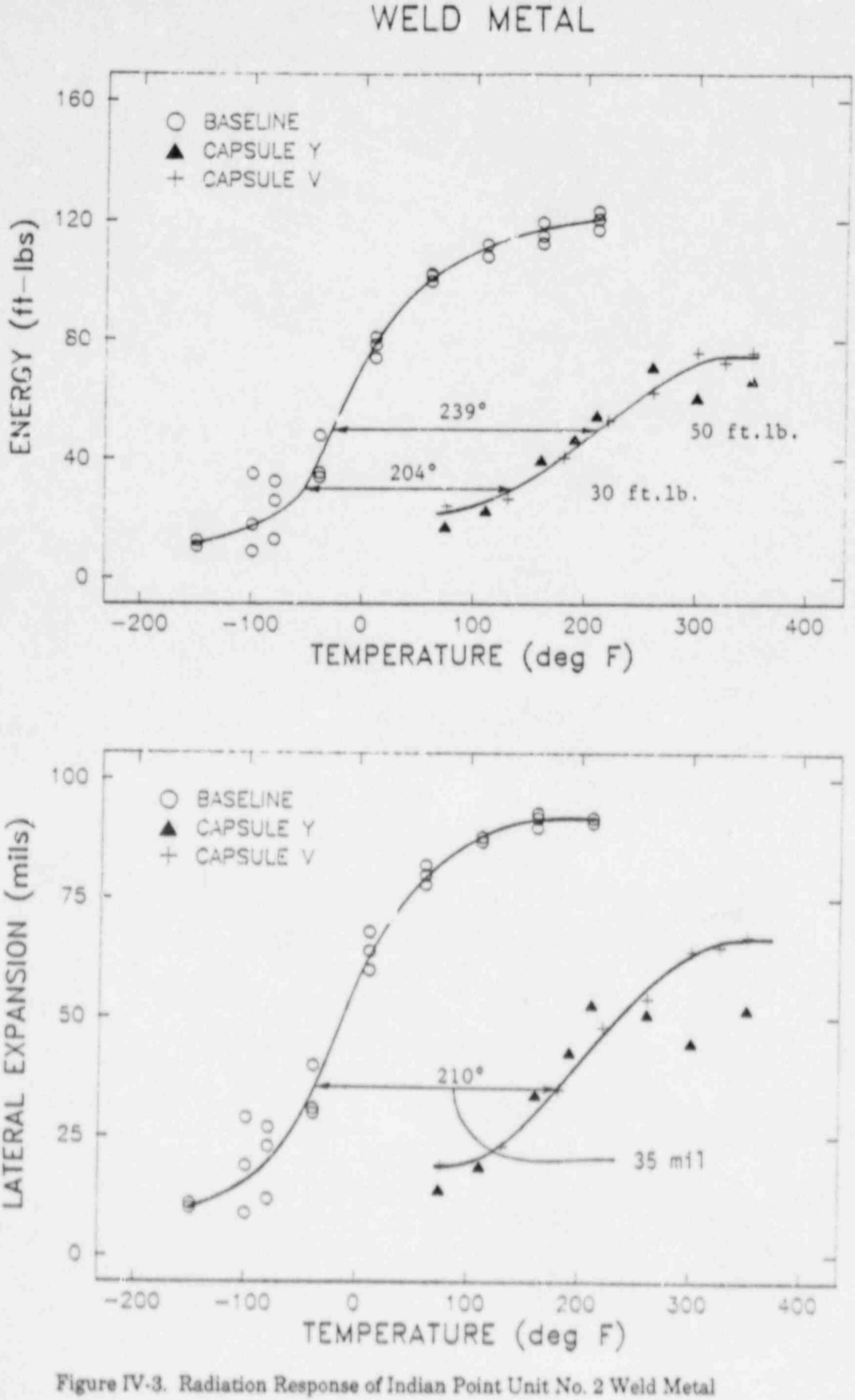


Figure IV-3. Radiation Response of Indian Point Unit No. 2 Weld Metal

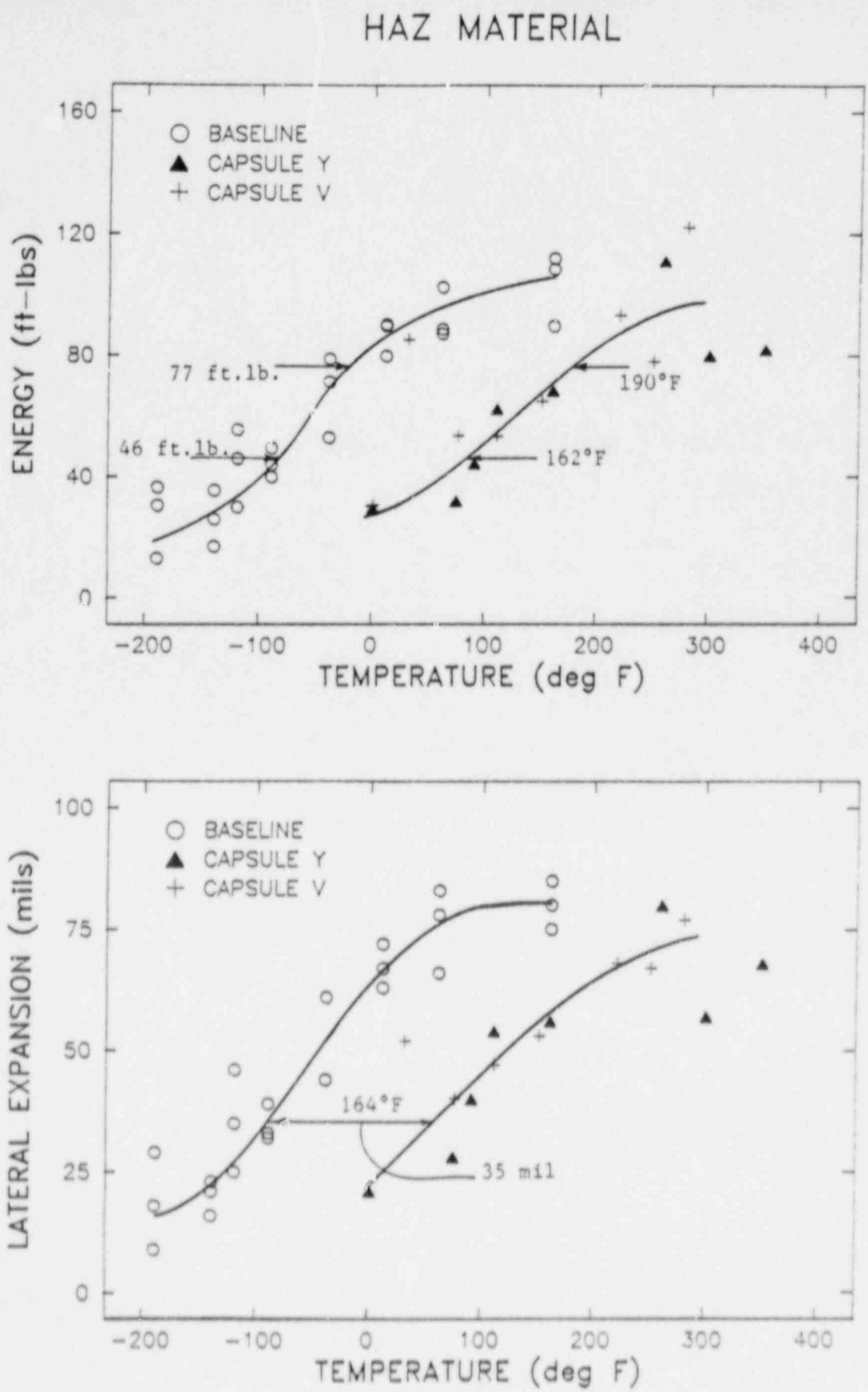


Figure IV-4. Radiation Response of Indian Point No. 2 Heat Affected Zone Material

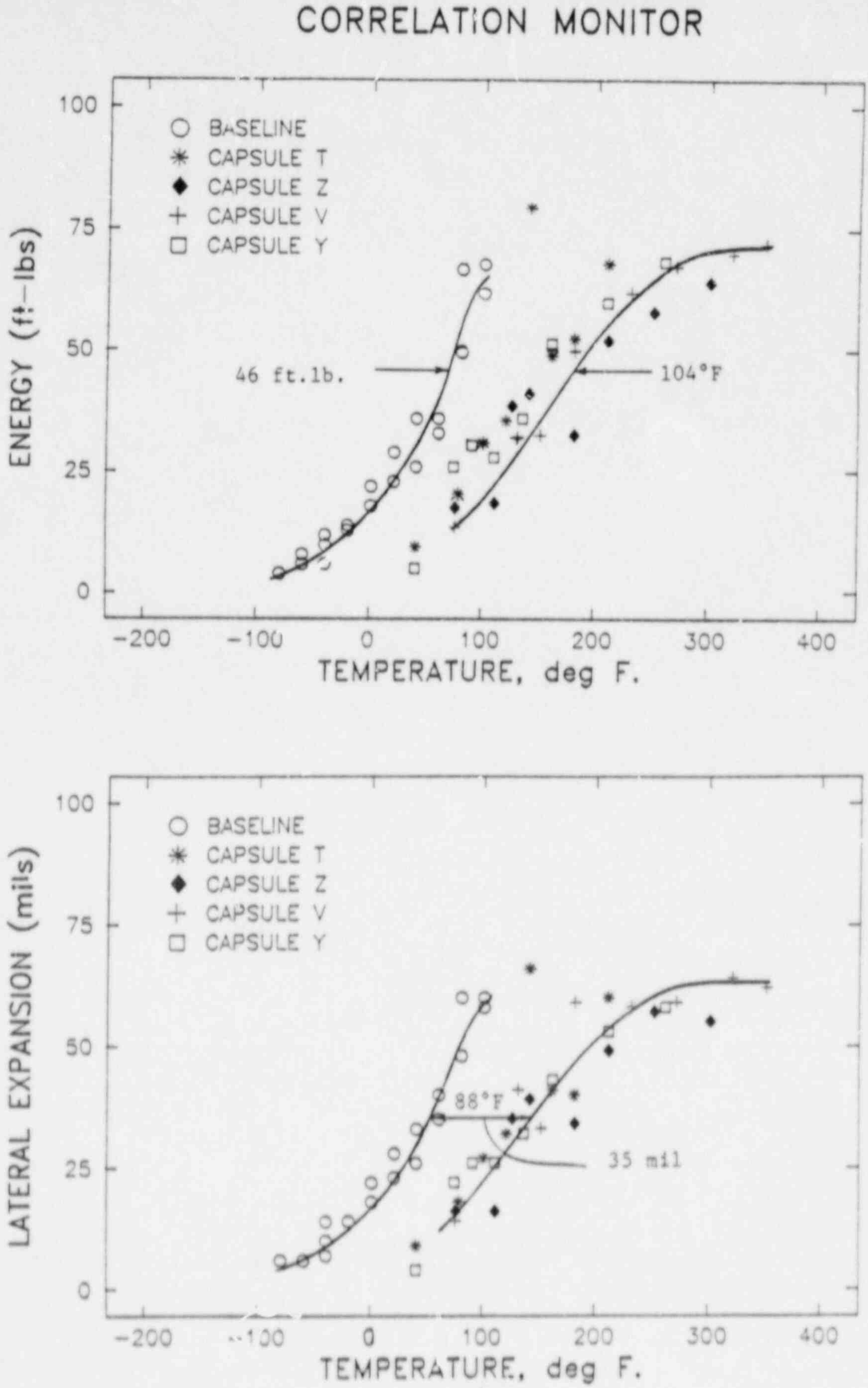


Figure IV-5. Radiation Response of Indian Point No. 2 Correlation Monitor Material

Table IV-9

SUMMARY OF RT_{NDT} SHIFTS AND UPPER SHELF ENERGY REDUCTION (C_V)
FOR MATERIALS IN CAPSULE V

A. **Summary of Fluence and Measured ΔRT_{NDT} Values for Test Specimens in Capsule V**

Type of Material	Fluence Neutron/cm^2	Measured ΔRT_{NDT} (°F) 50 Ft-Lbs	30 Ft-Lbs	35 mils*
Weld	5.59E18	239	204	230
		77 Ft-Lbs	40 Ft-Lbs	
Plate B2002-2	4.57E18	85	80	97
HAZ	5.59E18	190	162	184
Correlation Monitor	4.57E18	NA**	104	108

B. **Decrease in Upper Shelf Energy (C_V)**

Material	Initial Shelf Ft-lb	Capsule V*** Ft-lb	C_V Ft-lb	% Decrease
B2002-2	117	111	6	5
Weld Metal	118	75	43	36
HAZ	100	98	2 (nil)	2
Correlation Monitor	118	70	48	41

*35 mil + 20°F included in table.
**The upper shelf energy for this capsule was below 77 ft lbs.
*** Average of 3 Charpy measurements at ≈ 100% ductile failure.

Table IV-9 (Cont'd)

SUMMARY OF RT_{NDT} SHIFTS AND UPPER SHELF ENERGY REDUCTION (C_V)
FOR MATERIALS IN CAPSULE V

Charpy Impact Data for Decrease in Upper Shelf Energy

Shell Plate B2002-2			Weld Metal		
Sample	Ft-Lb	% Ductility*	Sample	Ft-Lb	% Ductility
2-45	109.5	100	W-14	76.0	95
2-46	116.0	100	W-16	72.5	95
2-47	106.5	100	W-15	76.0	100
Ave.**	111.0		Ave.**	75.0	

Heat-Affected Zone			Correlation Monitor		
Sample	Ft-Lb	% Ductility*	Sample	Ft-Lb	% Ductility
H-14	93.5	100	R-53	67.5	100
H-16	78.0	40	R-54	70.5	100
H-15	122.5	100	R-55	72.0	100
Ave.**	98.0		Ave.**	70.0	

*Fracture Appearance   Ave.** Average of 3 highest values with ≈ 100% ductility

Tensile tests were carried out in accordance with Procedure XI-MS-103-1 using a 22-kip capacity MTS Model 810 Material Test System equipped with an Instron Catalogue No. G-51-13A 2-inch strain gage extensometer and Hewlett Packard Model 7004B X-Y autographic recording equipment. Tensile tests on the plate material and the weld metal were run at room temperature at a strain rate of 0.005 in/in/min. through the 0.2% offset yield strength using servo-control and ramp generator. The results, along with the room temperature tensile data reported by Westinghouse on the unirradiated materials (10), are presented in Table IV-10. The load-strain records are included in Appendix B.

Table IV-10

TENSILE TEST DATA RECORDS
Capsule V DATA[(a)]

Test Material	Spc. No.	Temp (F)	0.2% YS (ksi)	UTS (ksi)	Fracture Load (lb)	Fracture Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Irradiated[(a)]									
Plate B2002-2	2-2	76	65.3	86.3	2940	157.9	24.6	25.5	62.4
	2-7	550	66.4	90.4	3170	250.4	17.9	17.4	74.0
Weld	W-3	76	92.7	106.9	3460	188.2	21.0	22.0	61.6
	W-4	550	82.5	100.2	3460	174.3	19.6	20.7	58.2
Unirradiated[(b)]									
Plate B2002-2	--	Room	62.4	83.8	(c)	(c)	(c)	27.1	70.0
	--	Room	66.8	90.5	(c)	(c)	(c)	28.2	69.6
	--	600	53.5	78.8	(c)	(c)	(c)	22.7	64.4
	--	600	54.7	81.4	(c)	(c)	(c)	24.7	64.4
Weld	--	Room	64.5	80.7	(c)	(c)	(c)	28.5	73.9
	--	Room	65.0	81.0	(c)	(c)	(c)	26.9	71.5
	--	600	56.6	79.8	(c)	(c)	(c)	24.4	62.0
	--	600	56.6	79.2	(c)	(c)	(c)	24.0	66.9

(a) Fluence = 5.59×10^{18} n/cm^2, E > 1 Mev

(b) WCAP 7323

(c) Data not reported in WCAP 7323

Testing of the WOL specimens was deferred at the request of Consolidated Edison Company. The specimens are in storage at the SwRI radiation laboratory.

D. Chemical Analysis Results

Check analyses for copper and nickel content of the ten broken Charpy V-notch specimens used for iron dosimetry and the three tested tensile specimens were run using ASTM Method E 322 (19). The results listed in Table IV-11 and IV-12 were obtained. For completeness, the list includes chemistry data from prior analyses of these and other surveillance samples of reactor vessel materials.

TABLE IV-11

SUMMARY OF CHEMISTRY VALUES FOR INDIAN POINT UNIT NO. 2 MATERIALS

Material	Source of Data	Cu W%	Ni W%
Plate B2002-1	WCAP 7323	(.25)*	(.58)*
	Capsule-Z: C_V Specimen 1-33	.22	.62
	Capsule-Z: C_V Specimen 1-38	.19	.71
	Capsule-Z: Tensile Specimen 1-5	(.29)*	.61
	Capsule-T: C_V Specimen 1-2	.17	--
	Capsule-T: C_V Specimen 1-3	.15	--
	Capsule-T: Tensile Specimen 1-1	21	--
	Average	.19	.65
Plate B2002-2	WCAP 7323	(.14)*	(.46)*
	Capsule-V: C_V Specimen 2-44	.17	.46
	Capsule-V: C_V Specimen 2-44	.15	.41
	Capsule-V: Tensile Specimen 2-6	(.06)*	(.27)*
	Capsule-V: Tensile Specimen 2-7	(.08)*	.42
	Capsule-Z: C_V Specimen 2-33	.19	.47
	Capsule-Z: C_V Specimen 2-36	.17	.46
	Capsule-Z: C_V Specimen 2-40	.20	.50
	Capsule-Z: Tensile Specimen 2-5	.15	.52
	Capsule-T: C_V Specimen 2-2	.18	--
	Capsule-T: C_V Specimen 2-3	.17	--
	Capsule-T: Tensile Specimen 2-1	.13	--
	Average	.17	.46

TABLE IV-11(Cont'd)

SUMMARY OF CHEMISTRY VALUES FOR INDIAN POINT UNIT NO. 2 MATERIALS

Material	Source of Data	Cu W%	Ni W%
Plate B2002-3	WCAP 7328	(.14)*	(.57)*
	Capsule-Z: C_V Specimen 3-33	.30	.64
	Capsule-Z: C_V Specimen 3-38	.27	.59
	Capsule-Z: Tensile Specimen 3-5	.23	.58
	Capsule-Y: C_V Specimen 3-41	.21	--
	Capsule-Y: C_V Specimen 3-45	.22	--
	Capsule-Y: Tensile Specimen 3-6	(.11)*	--
	Capsule-Y: Tensile Specimen 3-7	(.10)*	--
	Capsule-T: C_V Specimen 3-2	.27	--
	Capsule-T: C_V Specimen 3-3	.23	--
	Capsule-T: Tensile Specimen 3-1	(.09)*	--
	Average	.25	.60
HAZ	Capsule-V: C_V Specimen H-16	.08	1.2
	Capsule-V: C_V Specimen H-12	.06	1.2
	Capsule-Y: C_V Specimen H-21	.15	--
	Capsule-Y: C_V Specimen H-23	.20	--
	Average	.12	1.2
Weld	Capsule-V: C_V Specimen W-13	.23	1.02
	Capsule-V: C_V Specimen W-12	.20	1.06
	Capsule-V: Tensile Specimen W-3	.20	(.69)*
	Capsule-V: C_V Tensile Specimen W-4	(.12)*	1.00
	Capsule-Y: C_V Specimen W-17	.19	--
	Capsule-Y: C_V Specimen W-19	.22	--
	Capsule-Y: Tensile Specimen W-5	.18	--
	Capsule-Y: Tensile Specimen W-6	.20	--
	Average	.20	1.03
Correlation Monitor	Capsule-V: C_V Specimen R-56	.20	.18
	Capsule-V: C_V Specimen R-52	.18	.27
	Capsule-Z: C_V Specimen R-33	.35	.28
	Capsule-Z: C_V Specimen R-36	.31	.27
	Capsule-Z: C_V Specimen R-40	.21	.21
	Capsule-Y: C_V Specimen R-60	.17	--
	Capsule-Y: C_V Specimen R-62	.19	--
	Capsule-T: C_V Specimen R-2	.25	--
	Average	.23	.24

*Values in parentheses discarded because of excessive deviation or were WCAP values. Surveillance specimen WCAP values not used since chemical analyses were available.

Table IV-12

CHEMISTRY FACTORS FOR INDIAN POINT-2 MATERIALS
BASED ON REG. GUIDE 1.99, REV. 2

Material	W% Cu	W% Ni	Reg. Guide 1.99, Rev. 2 Chemistry Factor (°F)
Plate B2002-1	.19	.65	151
Plate B2002-2	.17	.46	115
Plate B2002-3	.25	.60	176
Surveillance HAZ	.12	1.2	86
Surveillance Weld Mat.	.20	1.03	226
Correlation Monitor	.23	.24	130

V. RESULTS OF ANALYSIS

The analysis of data obtained from surveillance program specimens has the following goals:

(1) Estimate the period of time over which the properties of the vessel beltline materials will meet the fracture toughness requirements of Appendix G of 10CFR50. This requires a projection of the measured reduction in C_V upper shelf energy to the vessel wall using knowledge of the energy and spatial distribution of the neutron flux and the dependence of C_V upper shelf energy on the neutron fluence.

(2) Develop heatup and cooldown curves to describe the operational limitations for selected periods of time. This requires a projection of the measured shift in RT_{NDT} to the vessel wall using knowledge of the dependence of the shift in RT_{NDT} on the neutron fluence and the energy and spatial distribution of the neutron flux.

The energy and spatial distribution of the neutron flux for Indian Point Unit No. 2 was calculated for Capsule V with a discrete ordinates transport by the Power Systems Division of Westinghouse Electric Corporation (17). Results from this analysis establish the means for the interpretation of surveillance capsule dosimetry and for the subsequent projection of neutron exposure results to the pressure vessel wall. Furthermore, the results of the evaluations are appropriate for absolute comparison with measurement.

A method for estimating the increase in RT_{NDT} as a function of neutron fluence and chemistry is given in Regulatory Guide 1.99, Revisions 1 and 2 (8). However, the Guide also permits interpolation between credible surveillance data and chemistry factors and extrapolation by extending the response curves parallel to the guide trend curves. These surveillance capsule results agree well with the Regulatory Guide 1.99, Revision 1, trend curves (see Figure V-1).

Revision 1 information is provided for comparison with earlier capsule data to show the effect of the low flux leakage core loading which produced a 48.9% reduction in fast neutron flux (E > 1 MeV) for cycles 6 through 8 as compared to the flux for the first 5 cycles. Revision 2 results from Capsule V are also included in this section.

The weld metal has now become the controlling material in place of the B2002-3 plate as can be seen in Table V-1. A long-term projection of vessel RT_{NDT} has been made from Cycle 8 and beyond using a low leakage core loading pattern which significantly reduces the pressure vessel fluence rate from that produced by the Design Basic Core (17) (see Table V-1).

A method for estimating the adjusted RT_{NDT} and the reduction in C_V upper shelf energy as a function of neutron fluence is also given in Regulatory Guide 1.99, Revision 1 (8). The results from Capsules T, Y, Z, and V are compared in Figures V-1 and V-2 which are adapted from the Regulatory Guide 1.99, Revision 1, Figures 1 and 2. The shelf energy responses of the pressure vessel surveillance materials from all four capsules are reasonably consistent and fall below the predictive trend curves of Regulatory Guide 1.99, Revision 1, for nominal weld chemistries of 0.20% Cu and 1.03% Ni. A long-term projection of the degradation in upper shelf energy has been made using the end-of-life fluences is given in Table V-2. The projected 32 EFPY 1/4T fluence is less than that received by Capsule Z and the shelf energy decreases of the Capsule V vessel specimens were all lower than earlier capsule specimens as shown in Figure V-2. Extrapolation to 1.2×10^{19} n/cm^2 for 32 EFPY predicts that all Indian Point Unit 2 materials will be below upper limit values for either RT_{NDT} or decrease in shelf energy.

Similar results are obtained using Revision 2 of Regulatory Guide 1.99 for Capsule V materials in Figure V-3. Extrapolation to 32 EFPY fluence of 1.2×10^{19} n/cm^2 on Figure V-3 gives predicted values of greater than 50 ft. lb. for shelf energies for weld metal (controlling materials) as well as plate and HAZ materials.

The current Indian Point Unit No. 2 reactor vessel surveillance program removal schedule conforms to ASTM E 185-79 (9) and is summarized in Table V-3. There are four capsules remaining in the vessel, of which three are standbys.

Table V-4 provides a comparison of End of Cycle 8 (EOC8) fluence values from transport calculations w h Capsule V dosimetry analysis and a comparison of projected fluence rates with transport calculations for Cycle 9. These comparisons, comparisons calculated with experimental values, show excellent agreement. EOC8 values differ by only two percent and the fluence rates for Cycle 9 differ by only about 10 percent.

The flux derived from Capsule V, 1.59E10±1.5E9 compared with the transport calculation 'or the same case agrees within the measurement uncertainties as shown in Table V-4.

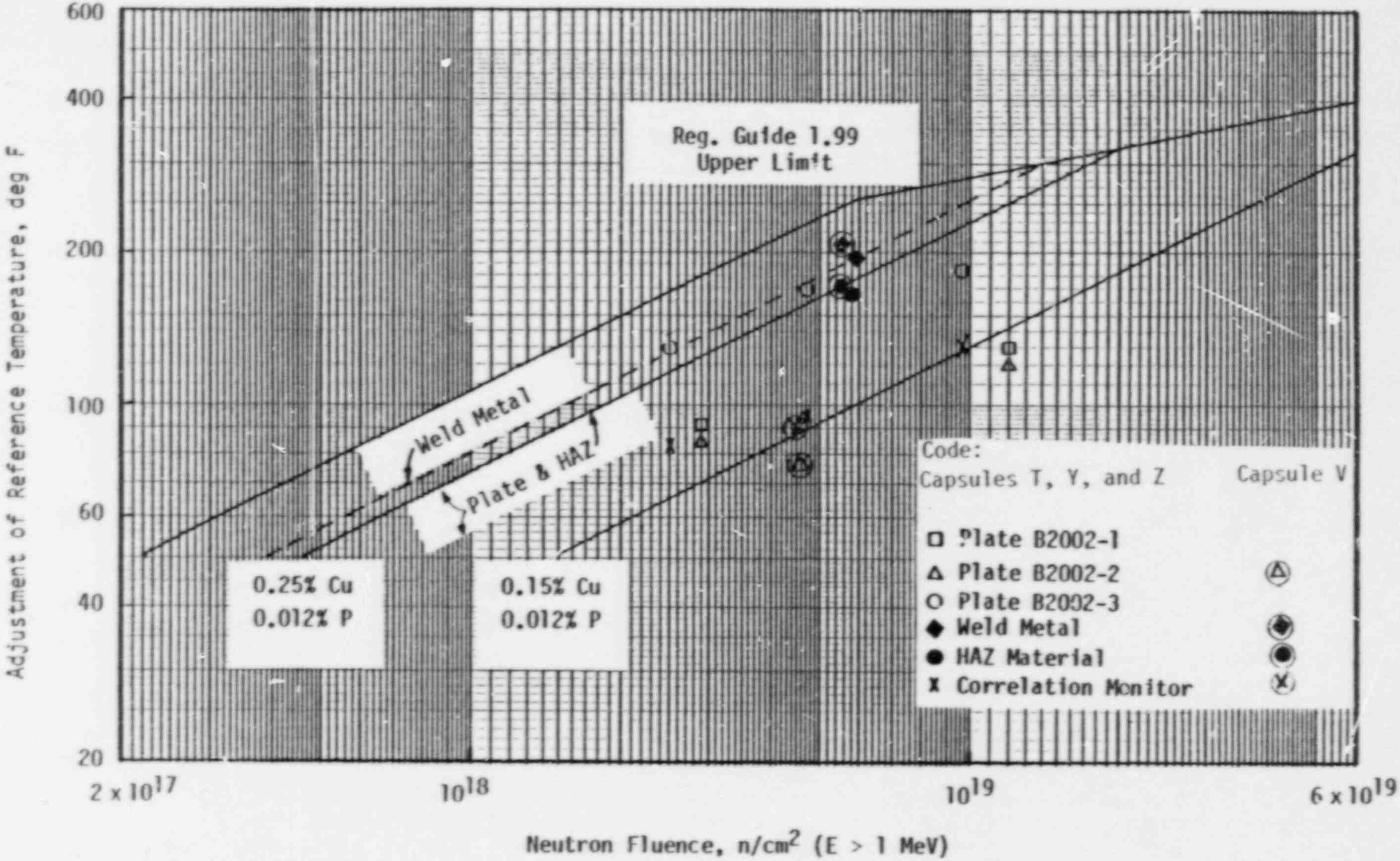


Figure V-1. Effect of neutron fluence on RT_{NDT} shift, Indian Point Unit No. 2
(Regulatory Guide 1.99, Revision 1)

Table V-1

ADJUSTED RT_{NDT} VALUES FOR INDIAN POINT-2

Time	Material	Location	Initial RT_{NDT}	Fluence[a] (>1 MeV)	DPA	ΔRT_{NDT} Rev. 1	ΔRT_{NDT} Rev. 2	ART (Adjusted RT_{NDT}) Rev. 1	ART (Adjusted RT_{NDT}) Rev. 2	PTS[c]
EOC8 [8.6 EFPY]	B2002-3 (Plate)	OT	34°F	4.5E18	4.5E18	154	137	188	205	
		3/4T	34	2.3E18	2.9E18	110	117	144	185	
		3/4T	34	4.7E17	1.1E18	50	77	84	145	
	HAZ	OT	0°F	4.5E18	4.5E18	154	125	154	181	
		1/4T	0	2.3E18	2.9E18	110	107	110	161	
		3/4T	0	4.7E17	1.1E18	50	71	50	107	
	Weld[b]	OT	0°F	4.5E18	4.5E18	172	176	172	232	
		1/4T	0	2.3E18	2.9E18	123	150	123	206	
		3/4T	0	4.7E17	1.1E18	56	99	56	149	
15 EFPY	B2002-3 (Plate)	OT	34°F	6.5E18	6.5E18	185	155	219	223	
		1/4T	34	3.4E18	4.2E18	134	134	168	202	
		3/4T	34	6.9E17	1.6E18	60	92	94	160	
	HAZ	OT	0°F	6.5E18	6.5E18	185	142	185	198	
		1/4T	0	3.4E18	4.2E18	134	122	134	178	
		3/4T	0	6.9E17	1.6E18	60	82	60	123	
	Weld[b]	OT	0°F	6.5E18	6.5E18	207	199	207	255	
		1/4T	0	3.4E18	4.2E18	150	172	150	228	
		3/4T	0	6.9E17	1.6E18	67	117	67	173	
20 EFPY	B2002-3 (Plate)	OT	34°F	8.2E18	8.2E18	208	166	242	234	
		1/4T	34	4.3E18	5.3E18	151	145	185	213	
		3/4T	34	8.5E17	2.0E18	67	102	101	170	
	HAZ	OT	0°F	8.2E18	8.2E18	208	152	208	208	
		1/4T	0	4.3E18	5.3E18	151	133	151	189	
		3/4T	0	8.5E17	2.0E18	67	94	67	141	
	Weld[b]	OT	0°F	8.2E18	8.2E18	232	213	232	269	
		1/4T	0	4.3E18	5.3E18	168	186	168	242	
		3/4T	0	8.5E17	2.0E18	75	131	75	187	
32 EFPY	B2002-3 (Plate)	OT	34°F	1.2E19	1.2E19	253	185	287	253	250
		1/4T	34	6.4E18	7.8E18	183	164	217	232	224
		3/4T	34	1.3E18	3.0E18	82	118	116	186	174
	HAZ	OT	0°F	1.2E19	1.2E19	253	169	253	225	
		1/4T	0	6.4E18	7.8E18	183	150	183	206	
		3/4T	0	1.3E18	3.0E18	82	109	82	164	
	Weld[b]	OT	0°F	1.2E19	1.2E19	282	237	282	293	212
		1/4T	0	6.4E18	7.8E18	204	210	204	266	186
		3/4T	0	1.3E18	3.0E18	91	151	91	207	138

(a) The fluence values shown at 1/4T and 3/4T locations are based on attenuation factors for Ø(>1 MeV) and used for Rev. 1 calculations only. *The actual 1/4T and 3/4T fluence used in Rev. 2 results were based on DPA attenuations*, conservatively estimated to be 0.65 and 0.25, respectively (see Table V-2). Thus based on this approach the fluence at 1/4T and 3/4T locations is equal to the 0-T fluence multiplied by DPA attenuation factors.

(b) Composition of weld No. 9-042 assumed to correspond to the surveillance data 0.20 percent Cu and 1.03 percent Ni, for Rev. 2 analysis.

(c) RT_{PTS} values based on PTS rule in 10CFR50.61

Table V-1 (Cont'd)

RELATIVE RADIAL VARIATION OF DISPLACEMENT PER ATOM (DPA) AND FLUX (E > 1 MeV) ATTENUATION WITHIN RPV, AT LOCATION OF MAXIMUM INCIDENT FLUX

Radius (cm)	Relative Flux Attenuation	Relative DPA Attenuation
220.27[(1)]	1.00	1.00
220.64	0.977	0.983
221.66	0.885	0.915
222.99	0.756	0.820
224.31	0.637	0.730
225.63	0.534	0.647
225.75[(a)]	0.526	0.640
226.95	0.443	0.573
228.28	0.367	0.507
229.60	0.303	0.449
230.92	0.250	0.397
232.25	0.206	0.349
233.57	0.169	0.307
234.89	0.138	0.269
236.22	0.113	0.233
236.70[(b)]	0.105	0.221
237.54	0.0912	0.201
238.86	0.0736	0.170
240.19	0.0584	0.141
241.51	0.0454	0.113
242.17[(2)]	0.0422	0.106

NOTES: (1) Base Metal Inner Radius
(2) Base Metal Outer Radius
(a) 1/4T Location
(b) 3/4T Location

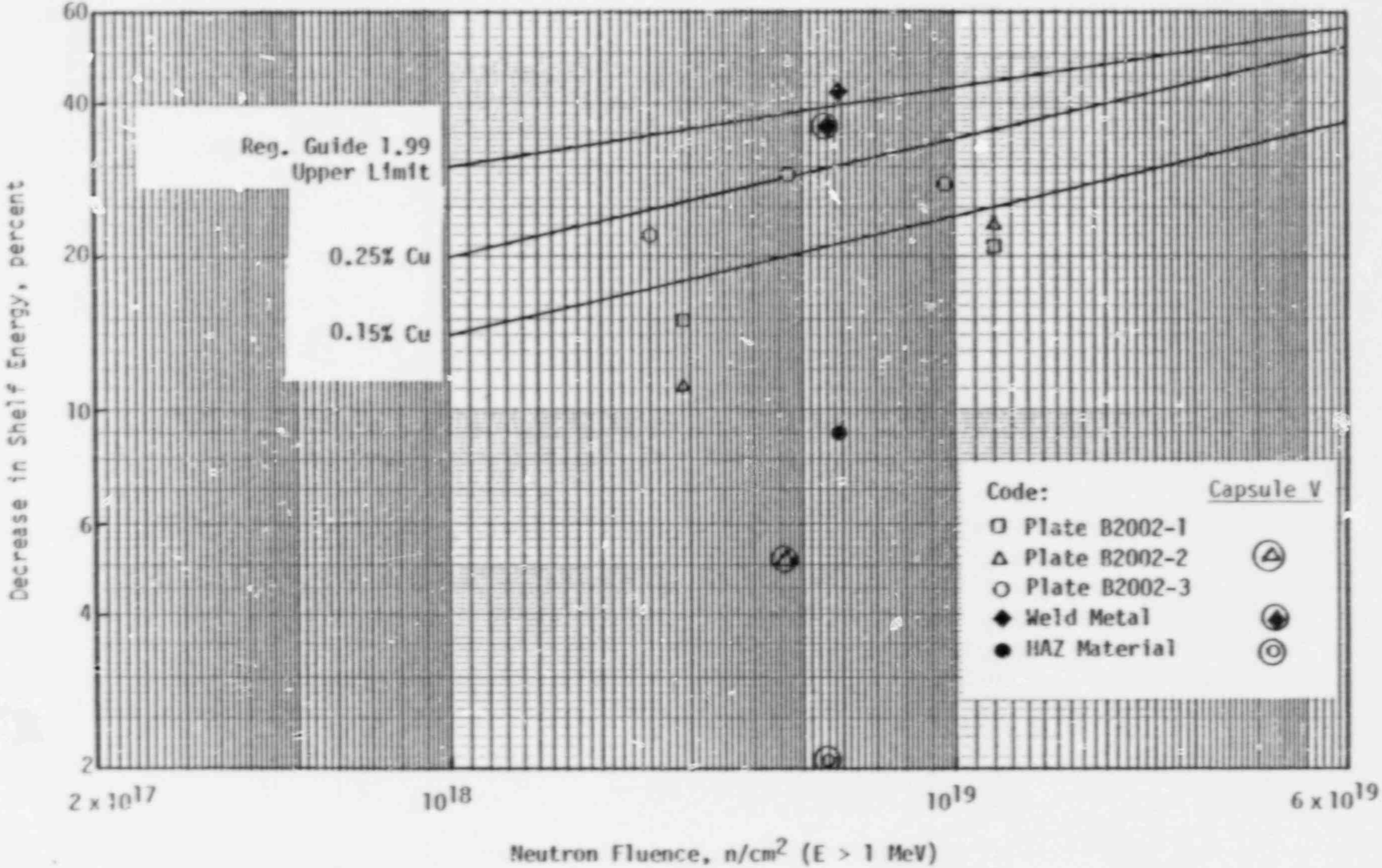


Figure V-2. Dependence of C_V shelf energy on neutron fluence, Indian Point Unit No. 2
(Regulatory Guide 1.99, Revision 1)

Table V-2

COMPARISON OF MEASURED AND CALCULATED RT_{NDT} VALUES FOR INDIAN POINT-2 CAPSULE V MATERIALS

Material	Measured[a]	Reg. Guide 1.99 Rev. 1[b]	Rev. 2	Rev. 2 + Margin
Plate B2002-2	78	90	90	124
Weld	204	195	189	245
HAZ	166	170	135 72 (c)	191 106(c)
Correlation Monitor	90	90	102	136

(a) 30 Ft-Lbs or 46 Ft-Lbs Value, as appropriate, Figures IV-2, 3, 4, and 5; Table IV-9

(b) Obtained from Fig. V-1 at 5.59×10^{18} n/cm^2 for weld and HAZ or 4.57×10^{18} n/cm^2 for Plate 2002-2 and Correlation Monitor.

(c) Based on Weld and Base Plate Correlation, respectively

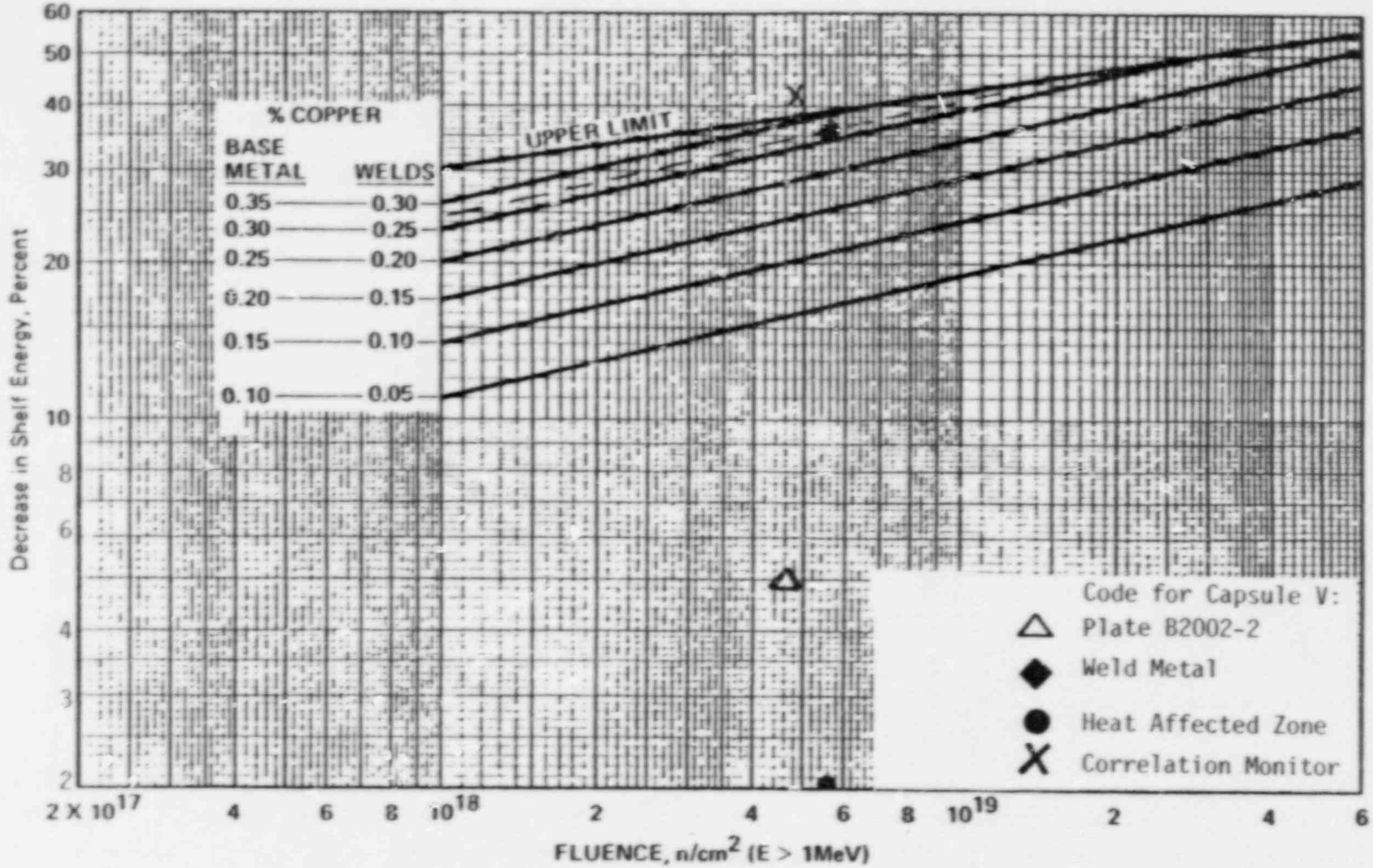


Figure V-3. Predicted decrease in shelf energy as a function of copper content and fluence
(Adapted from Regulatory Guide 1.99, Revision 2; Data from Table IV-9)

Table V-3

REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE (21)
INDIAN POINT UNIT NO. 2

Capsule Ident. No.	Code	WOL Material	Removal Time	Equivalent Vessel Fluence
1	T	Three Plates	1.08 EFPY[(a)]	3.4 EFPY at I.D.
2	Y	Weld & B2002-3	2.34 EFPY[(b)]	11 EFPY at I.D.
3	Z	Three Plates	5.17 EFPY[(c)]	29 EFPY at I.D.
4	V	Weld & B2002-2	8.6 EFPY[(d)]	8.92 EFPY at I.D.

(a) Removed after core cycle 1.

(b) Removed after core cycle 3.

(c) Removed after core cycle 5.

(d) Removed after core cycle 8.

Note: Fifth capsule is scheduled for removal at the end of Cycle 16.

The remaining capsules within the reactor vessel are:

Code	WOL Material
S	Weld & B2002-1
U	Three Plates
W	Three Plates
X	Three Plates

Table V-4

COMPARISON OF END OF CYCLE 8 FLUENCE VALUES FROM TRANSPORT CALCULATIONS AND CAPSULE V DOSIMETRY ANALYSIS

Location	Transport Calculation (n/cm^2)	Dosimetry Results (n/cm^2)	C/E*
4° S. C.	5.19E18	5.30E18	0.98
40° S. C.	1.48E19	1.51E18	0.98
RPV O-T	4.35E18	4.45E18	0.98

COMPARISON OF PROJECTED FLUENCE RATES WITH TRANSPORT CALCULATIONS FOR CYCLE 9

Location	Transport Calculation (n/cm^2 sec)	Dosimetry** Results (n/cm^2 sec)	C/E*
4° S. C.	1.75E10	1.57E10	1.11
40° S. C.	3.77E10	3.42E10	1.10
RPV O-T	1.13E10	1.03E10	1.10

*C/E is calculated/experimental.

**Capsule V values used as the "projected" dosimetry results.

VI. HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION OF INDIAN POINT UNIT NO. 2

Indian Point Unit No. 2 is a 2758 Mwt pressurized water reactor operated by Consolidated Edisor Company. The unit has been provided with a reactor vessel material surveillance program as required by 10CFR50, Appendix H.

The fourth surveillance capsule (Capsule V) was removed during the 1987 refueling outage. This capsule was tested by Southwest Research Institute, the results being described in the earlier sections of this report. In summary, these results show a marked decrease in fluence as compared to three capsules (Capsules T, Y, and Z) but now indicate that the weld material will control the value of RT_{NDT} over the plant design lifetime.

The adjusted RT_{NDT} (Regulatory Guide 1.99, Rev. 2, May 1988) after 32 effective full power years (EFPY) of operation is predicted to be 266°F at the 1/4T and 207°F at the 3/4T vessel wall locations, as controlled by weld material. The Unit No. 2 heatup and cooldown limit curves for up to 32 EFPY of operation have been computed on the basis of the above values of adjusted RT_{NDT} using Code procedures (2) and the following pressure vessel constants:

Vessel Inner Radius, r_i	= 86.50 in.
Vessel Outer Radius, r_o	= 95.28 in.
Operating Pressure, P_o	= 2235 psig
Initial Temperature, T_o	= 70°F
Final Temperature, T_f	= 550°F
Effective Coolant Flow Rate, Q	= 136.3×10^6 lb_m/hr
Effective Flow Area, A	= 26.719 ft^2
Effective Hydraulic Diameter, D	= 15.051 in.

Heatup curves were computed for heatup rates of 20°F/hr. 40°F/hr, 60°F/hr and 100°F/hr.

The Unit No. 2 heatup, cooldown, and leak test curves for up to 32 EFPY are given in Figures VI-1, VI-2, and VI-3.

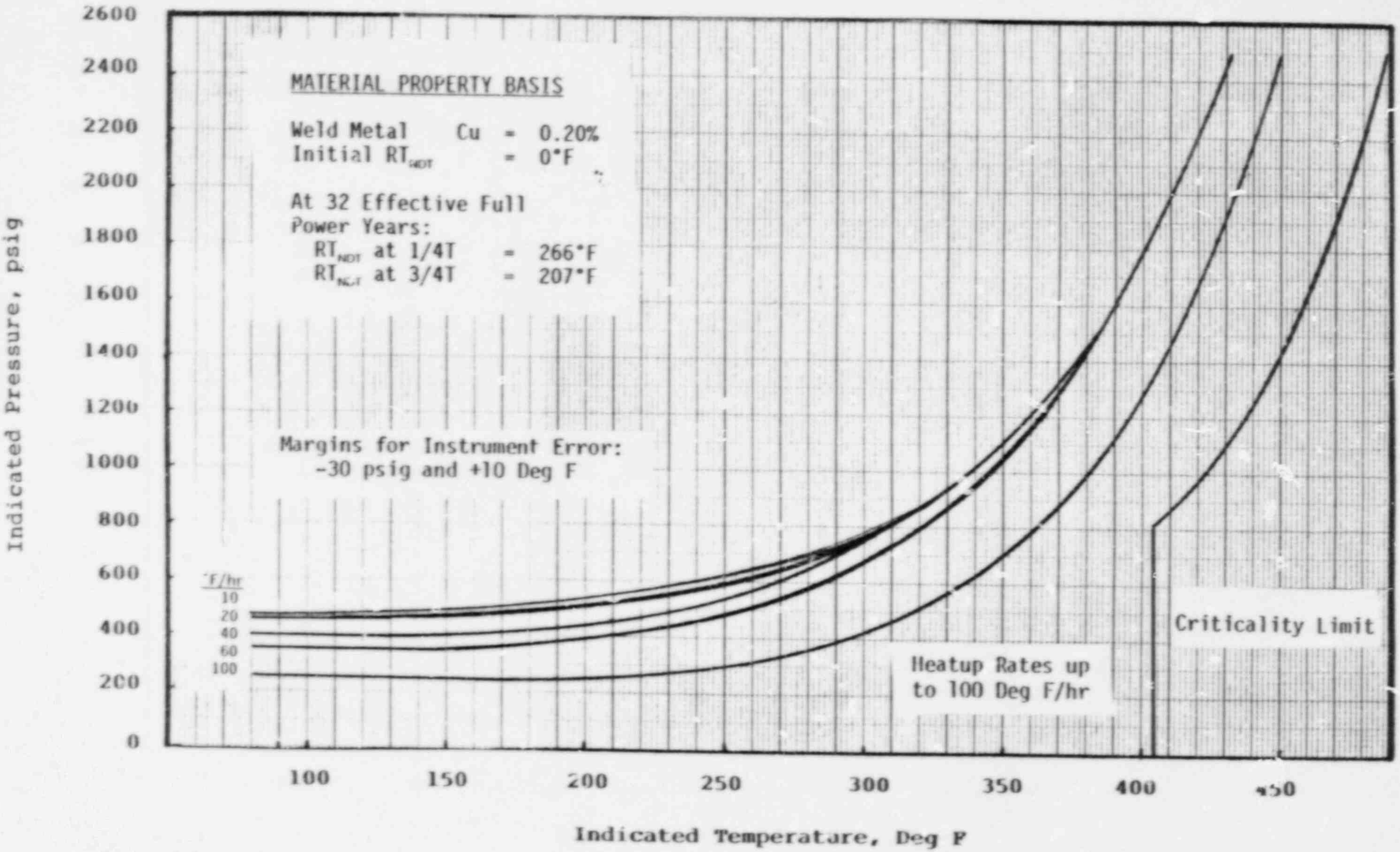


Figure VI-1. Indian Point Unit No. 2 Reactor Coolant Heatup Limitations Applicable for Periods Up to 32 Effective Full Power Years (With Criticality Limit)

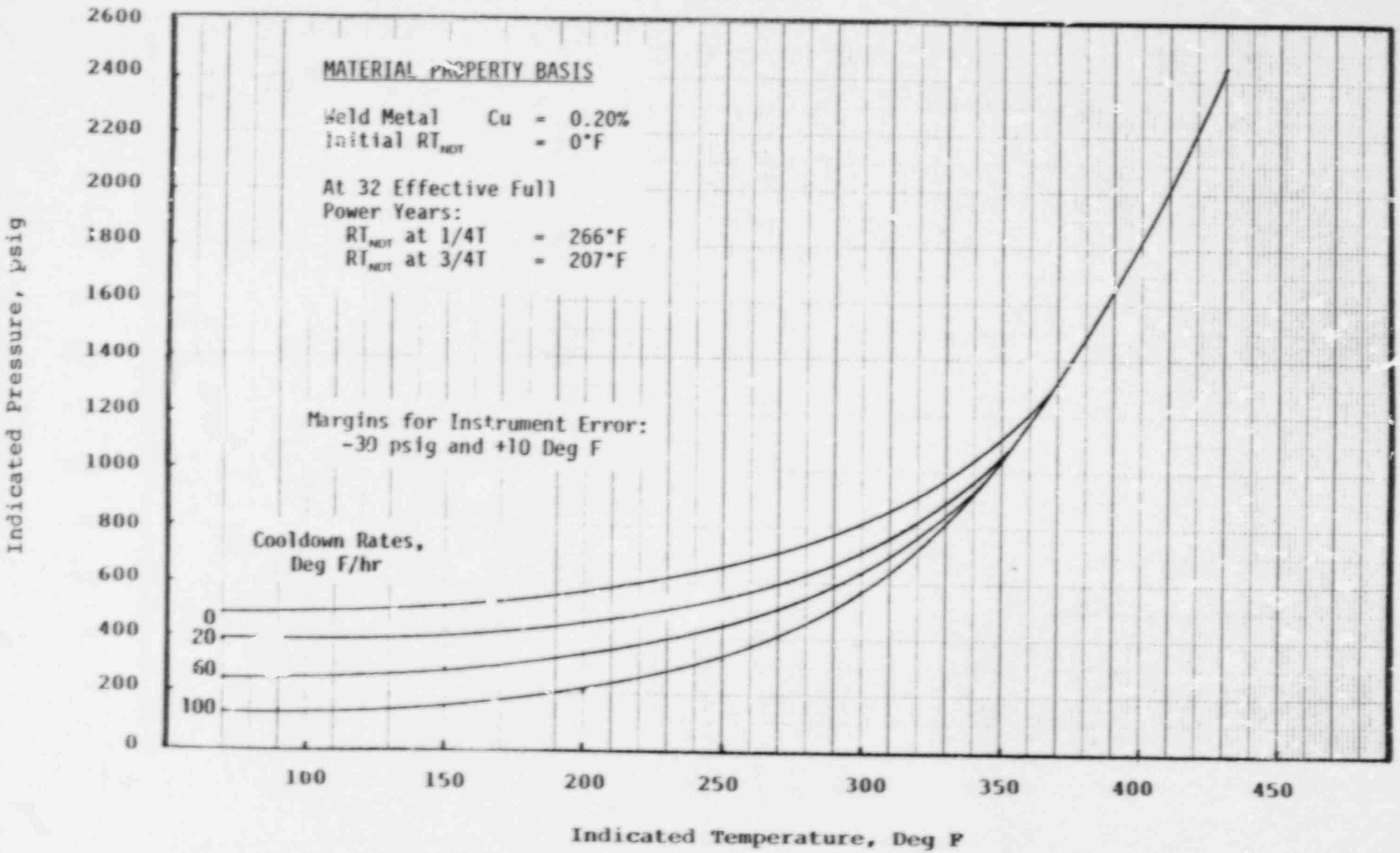


Figure VI-2. Indian Point Unit No. 2 Coolant Cooldown Limitations Applicable for Periods Up to 32 Effective Full Power Years

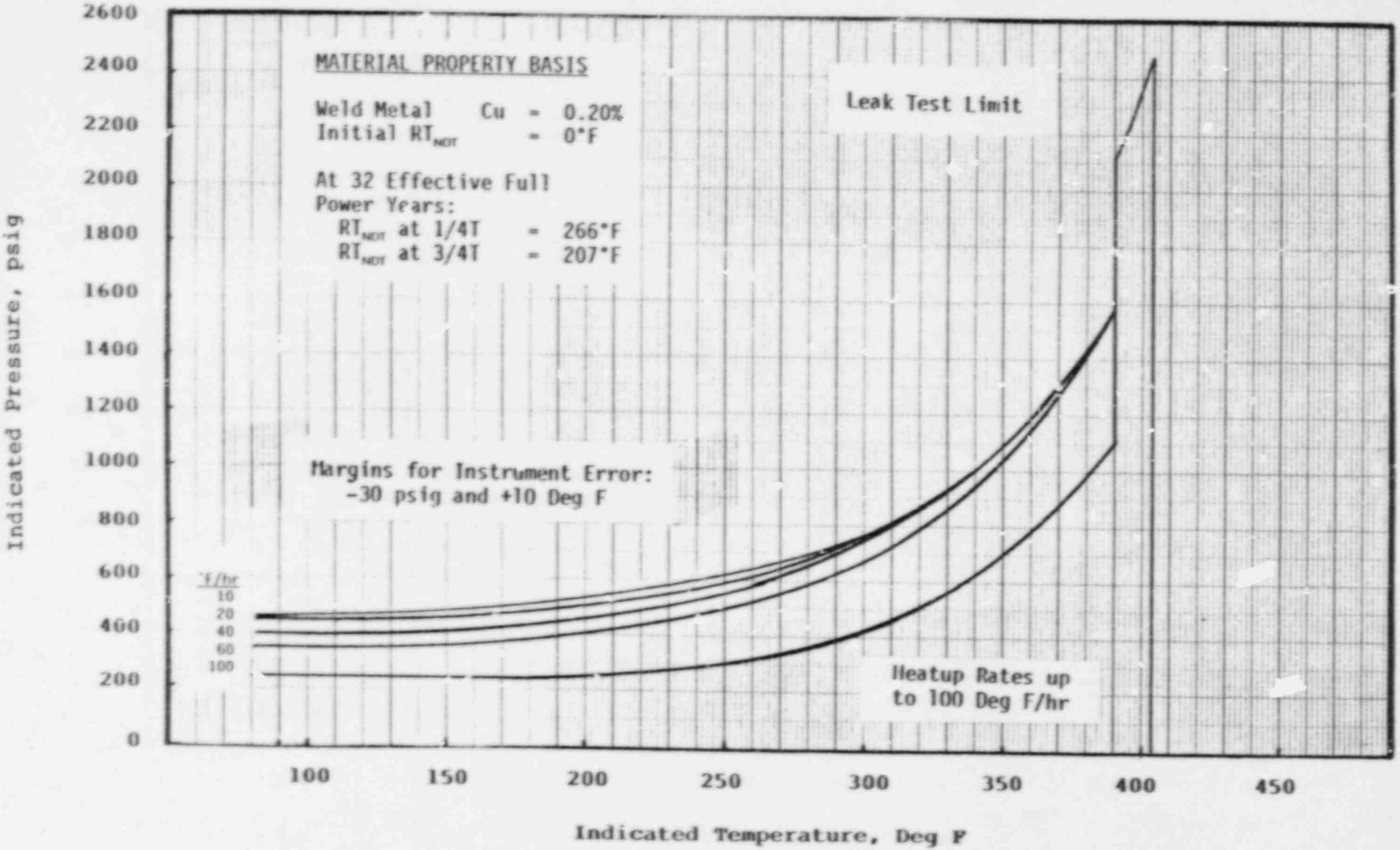


Figure VI-3. Indian Point Unit No. 2 Reactor Coolant Heatup Limitations Applicable for Periods Up to 32 Effective Full Power Years (With Leak Test Limit)

VII. REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

3. ASTM E 208-81, "Standard Method for Conducting Drop-Weight Test to Determine Ni-Ductility Transition Temperature of Ferritic Steels," 1982 Annual Book of ASTM Standards.

4. Steele, L. E., and Serpan, C. Z., Jr., "Analysis of Reactor Vessel Radiation Effects Surveillance Programs," ASTM STP 481, December 1970.

5. Steele, L. E., "Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels," International Atomic Energy Agency, Technical Reports Series No. 163, 1975.

6. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974 Edition.

7. Randall, P. N., "NRC Perspective of Safety and Licensing Issues Regarding Reactor Vessel Steel Embrittlement - Criteria for Trend Curve Development," presented at the American Nuclear Society Annual Meeting, Detroit, Michigan, June 14, 1983.

8. Regulatory Guide 1.99, Revision 1, Office of Standards Development, U.S. Nuclear Regulatory Commission, April 1977 and Regulatory Guide 1.99, Revision 2, May 1988.

9. ASTM E 185-79, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," 1981 Annual Book of ASTM Standards.

10. "Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-7323, May 1969.

11. ASTM E 399-81, "Standard Method of Test for Plane-Strain Fracture Toughness of Metallic Materials," 1982 Annual Book of ASTM Standards.

12. ASTM E 813-81, "Standard Test Method for J_{Ic}, a Measure of Fracture Toughness," 1982 Annual Book of ASTM Standards.

13. Norris, E. B., "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2; Analysis of Capsule Y," SwRI Report 02-5212, November 16, 1980.

14. Norris, E. B., "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2; Analysis of Capsule T," Final Report 02-4531, June 30, 1977, and Supplement to Final Report 02-4531, December 1980.

15. Norris, E. B., "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2; Analysis of Capsule Z," SwRI Report 06-7379, April 1984.

16. Anderson, S. L., "Analysis of Neutron Flux Levels and Surveillance Capsule Lead Factors for the Indian Point Unit No. 2 Reactor Using a Low Leakage Core," July 1982.

17. Anderson, S. L., "Plant Specific Fast Neutron Exposure Evaluation of the Indian Point Unit 2 Reactor Pressure Vessel and Surveillance Capsules Fuel Cycles 1 through 9," Westinghouse Electric Corp., Power Systems Division, July 1988.

18. U.S. NRC Standard Review Plan, NUREG-0800, Section 5.3.2, Pressure-Temperature Limits, Revision 1, July 1981.

19. ASTM E 322, "Standard Method for Spectrochemical Analysis of Low Alloy Steels and Cast Irons Using an X-ray Fluorescence Spectrometer."

20. Letter dated March 29, 1978, from W. J. Cahill, Jr. (Consolidated Edison) to R. W. Reid (NRC), "Indian Point Unit No. 2 Reactor Vessel Material Surveillance Program."

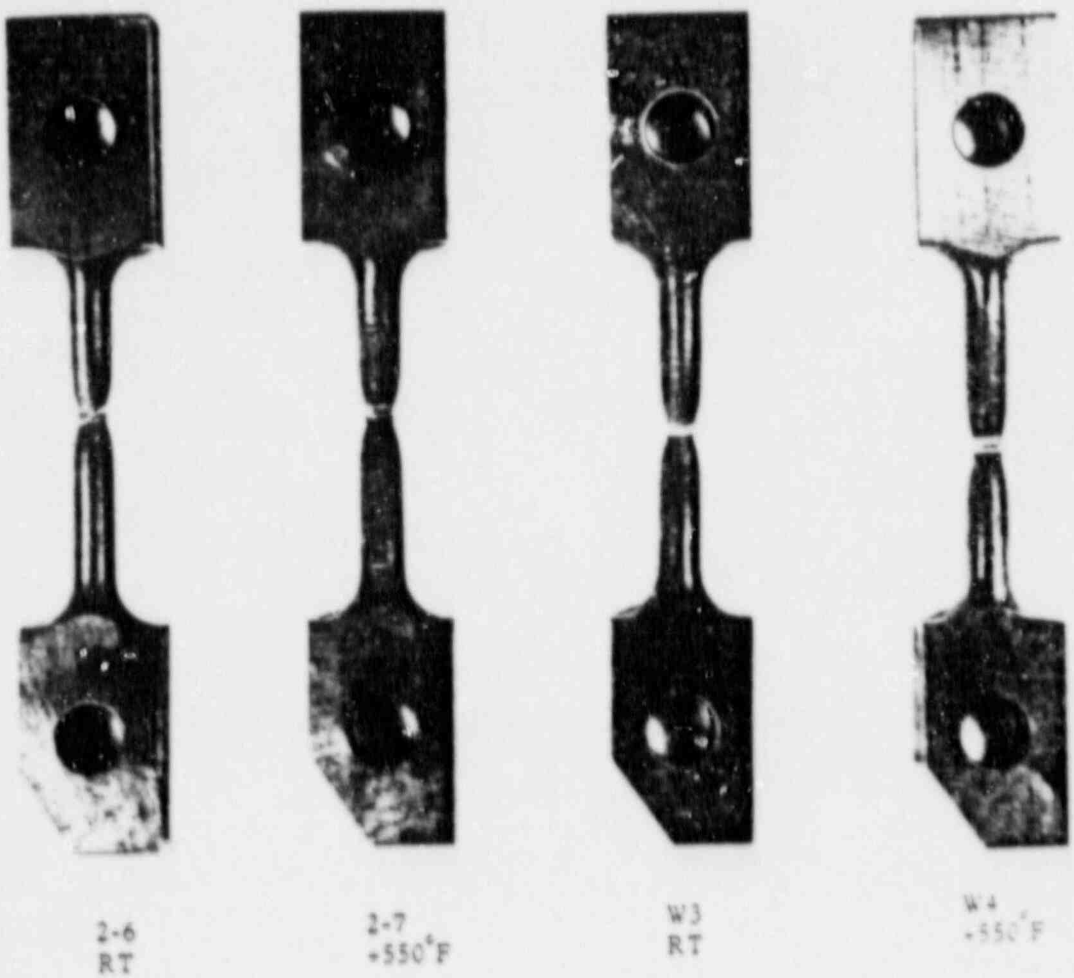
21. Indian Point Unit No. 2 Technical Specifications.

APPENDIX A

TENSILE TEST DATA RECORDS

Photograph of Specimens After Testing

Specimens: W-3
W-4
2-6
2-7



Photograph of tensile specimens after testing

Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. N3

Project No. 17-2108

Test Temperature RT

Machine Ident. 4

Strain Rate .005"/IN/MIN

Date of Test 6/15/88

Initial Diameter .247
Initial Area .04789
Initial Gage Length 1.0
Specimen Temperature:
Top T.C.
Middle T.C. 75°F
Bottom T.C.

Witnessed by Sun Q.A. JLD

Final Diameter .153
Final Area .01838
Final Gage Length 1.220
Maximum Load 5120
0.2% Offset Load 4440
Fracture Load 3460
Elong. to Max. Load .2099

U.T.S. = Maximum Load/Initial Area = 106,912

0.2% Y.S. = 0.2% Offset Load/Initial Area = 92,712

Frature Stress = Fracture Load/Final Area = 188,248

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 61.62

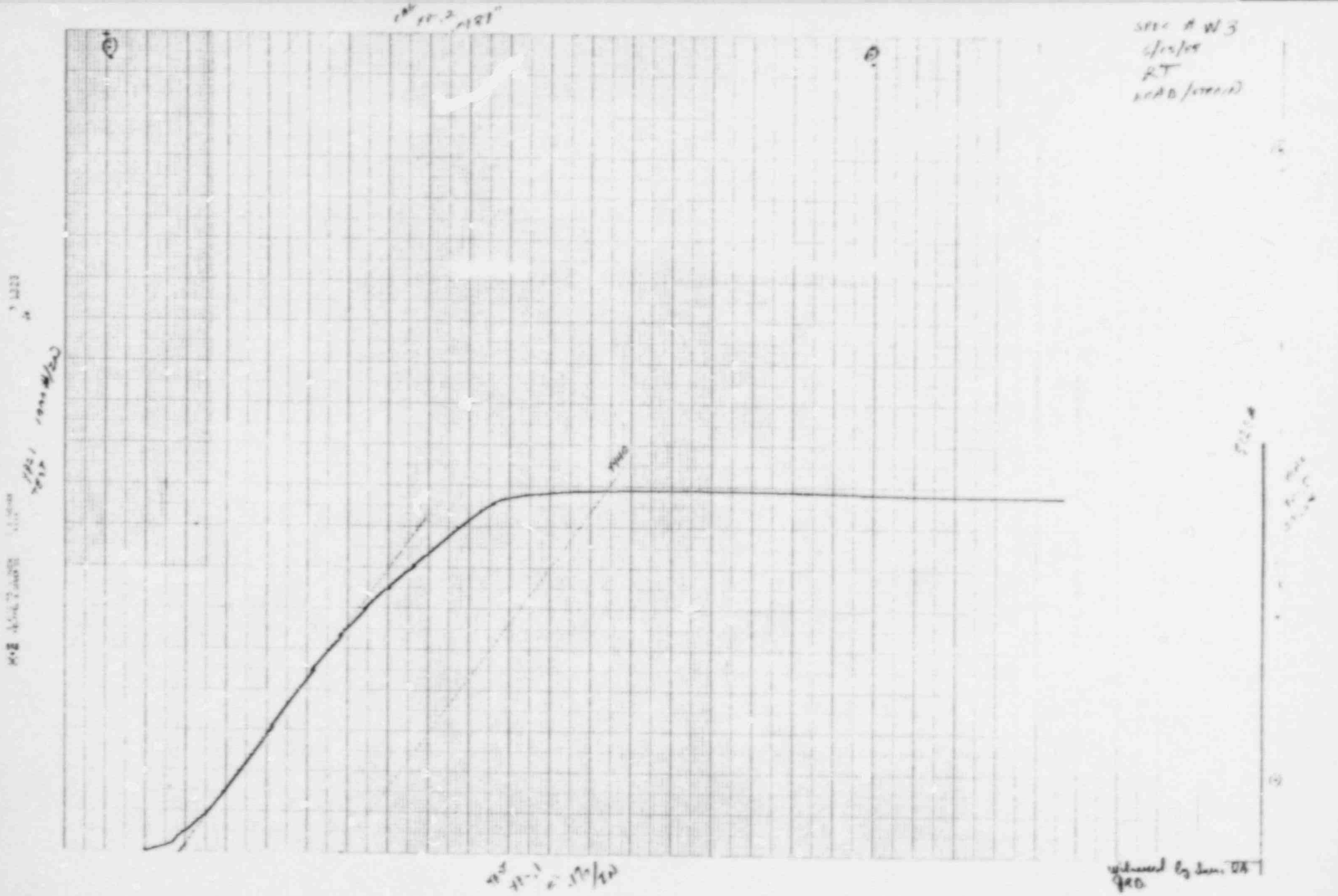
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 22.0

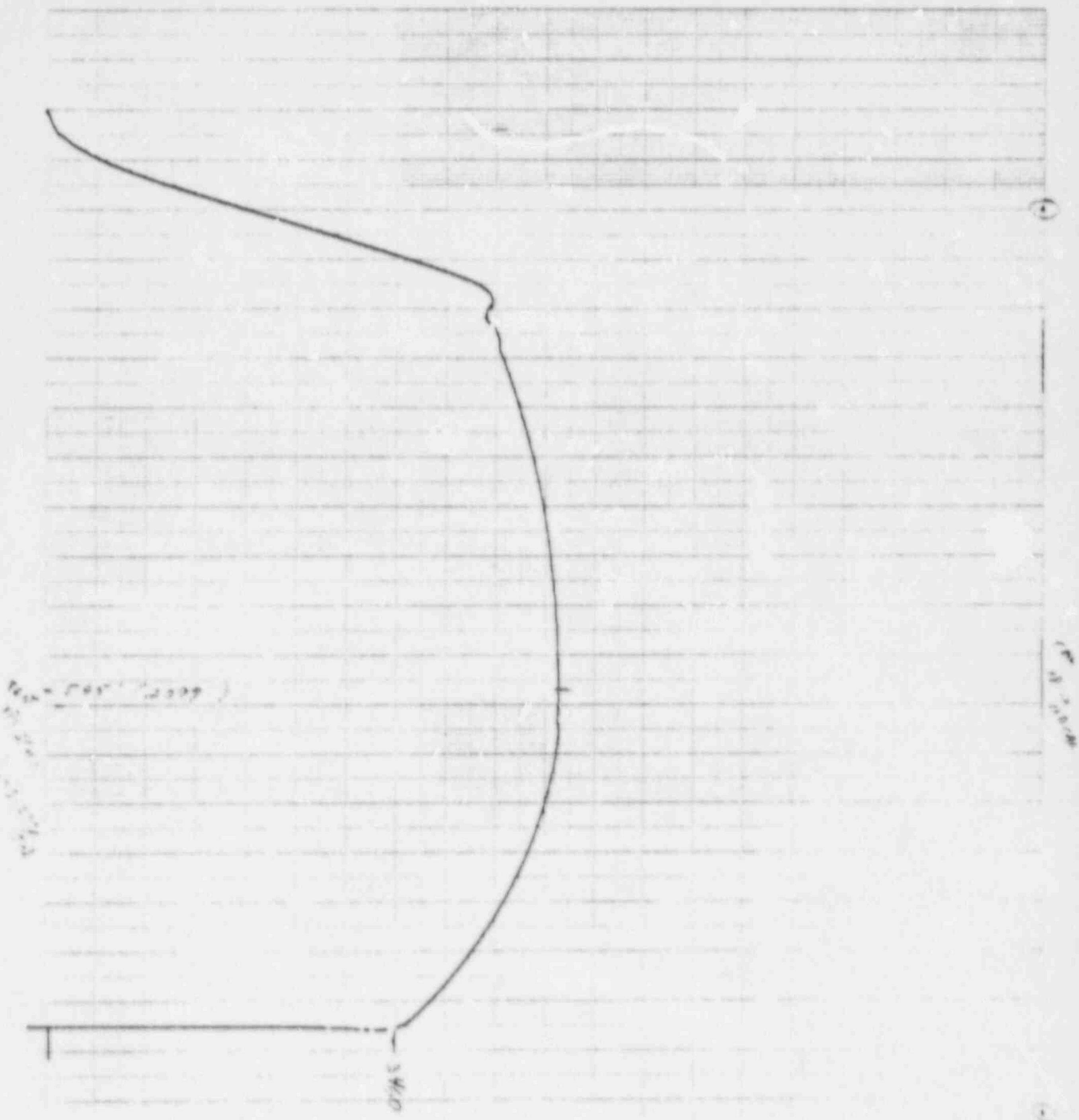
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 20.99

Test Performed by: T. MASNEN / R. ATIYEH

Calculations Performed by: Tom Masden (Date) 6/19/88

Calculations Checked by: David G. Cochran (Date) 6/21/88


SPEC # W3
RT



Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. W4

Project No. 17-2108

Test Temperature 550°F

Machine Ident. 4

Strain Rate .005"/in/min

Date of Test 6/16/88

Initial Diameter .246
Initial Area .047505
Initial Gage Length 1.0
Specimen Temperature:
Top T.C. 553°F
Middle T.C. N/A
Bottom T.C. 553°F

Final Diameter .159
Final Area .019846
Final Gage Length 1.207
Maximum Load 4760
0.2% Offset Load 3920
Fracture Load 3460
Elong. to Max. Load .195546

U.T.S. = Maximum Load/Initial Area = 100,200

0.2% Y.S. = 0.2% Offset Load/Initial Area = 82,518

Frature Stress = Fracture Load/Final Area = 174,342

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 58.22

% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 20.70

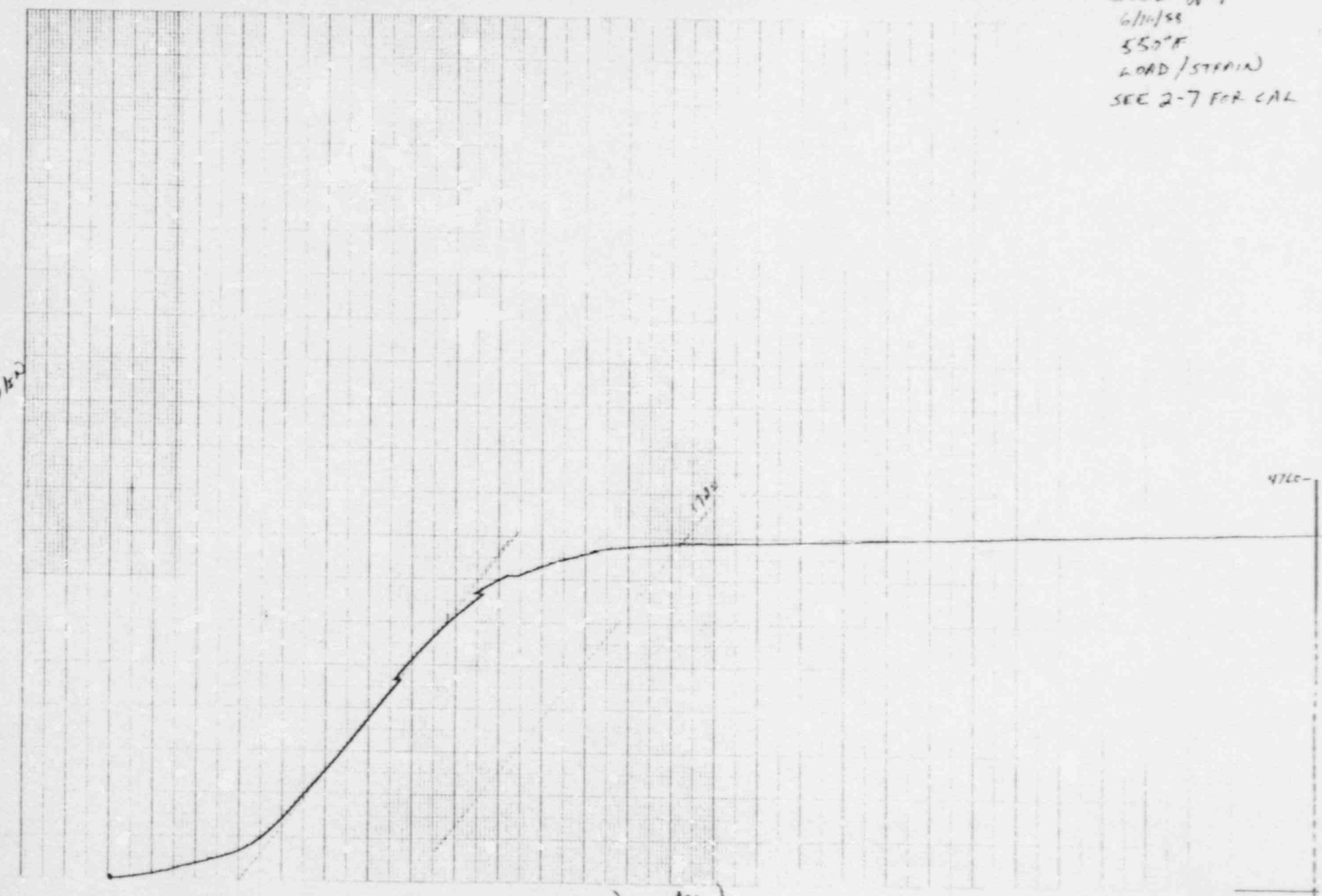
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 19.55

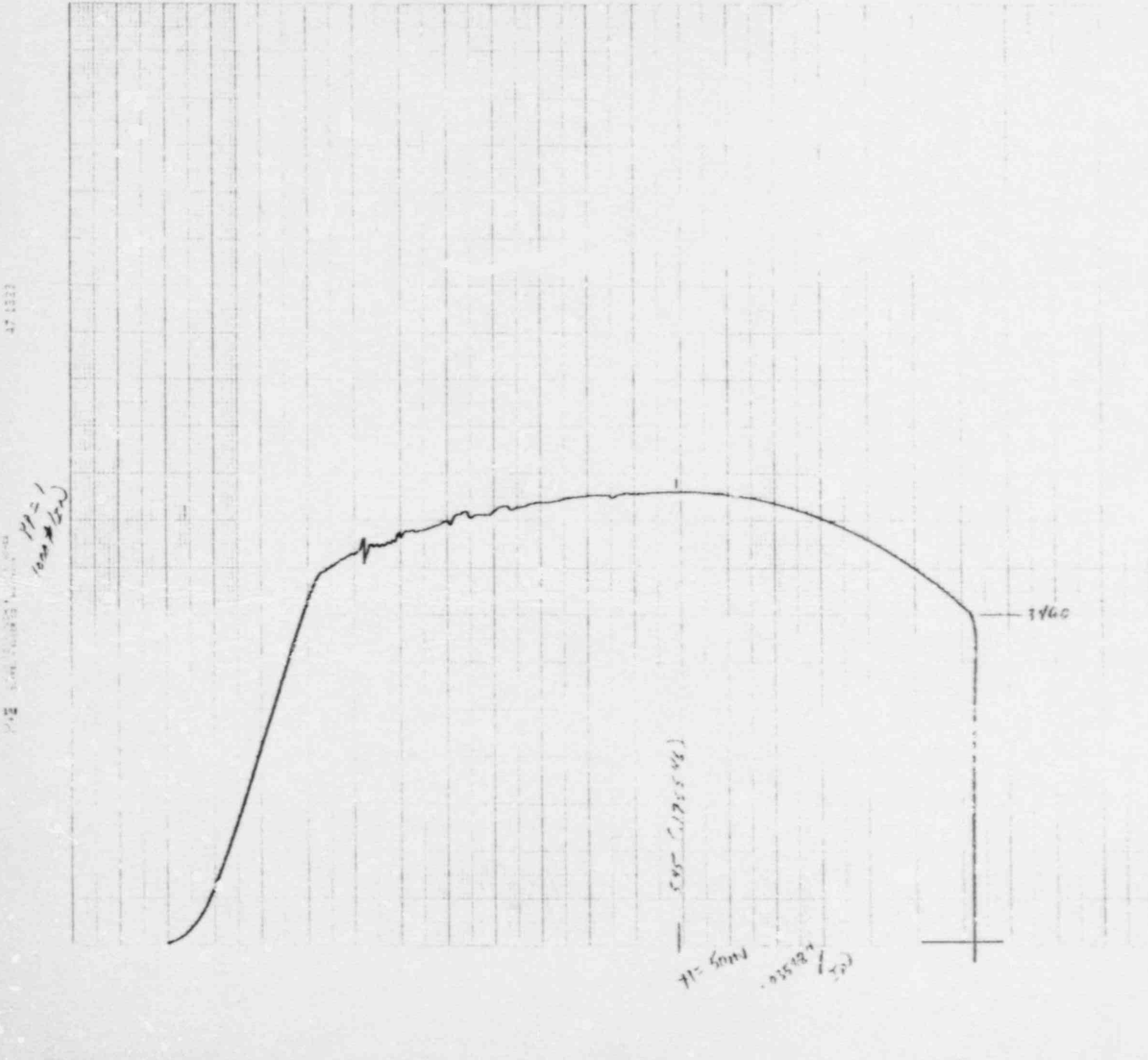
Test Performed by: T MASDEN/R ANTHEH

Calculations Performed by: [signature] (Date) 6/17

Calculations Checked by: [signature] (Date) 6/21/88

SPEC # W4
6/11/58
550°F
LOAD / STRAIN
SEE 2-7 FOR CAL





SPEC # W4
6/10/84
LOAD / H.D.
550°F
SEE 2.7 FOR CAL.

Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. 2-6

Test Temperature RT

Strain Rate .005"/IN/MIN

Project No. 17-2108

Machine Ident. 4

Date of Test 6/15/88

Initial Diameter .251
Initial Area .04946
Initial Gage Length 1.0
Specimen Temperature:
Top T.C.
Middle T.C. RT 76°F
Bottom T.C.

Final Diameter .154
Final Area .01862
Final Gage Length 1.255
Maximum Load 4270
0.2% Offset Load 3230
Fracture Load 2940
Elong. to Max. Load .24578

Witnessed by Sur Hopud Sr.

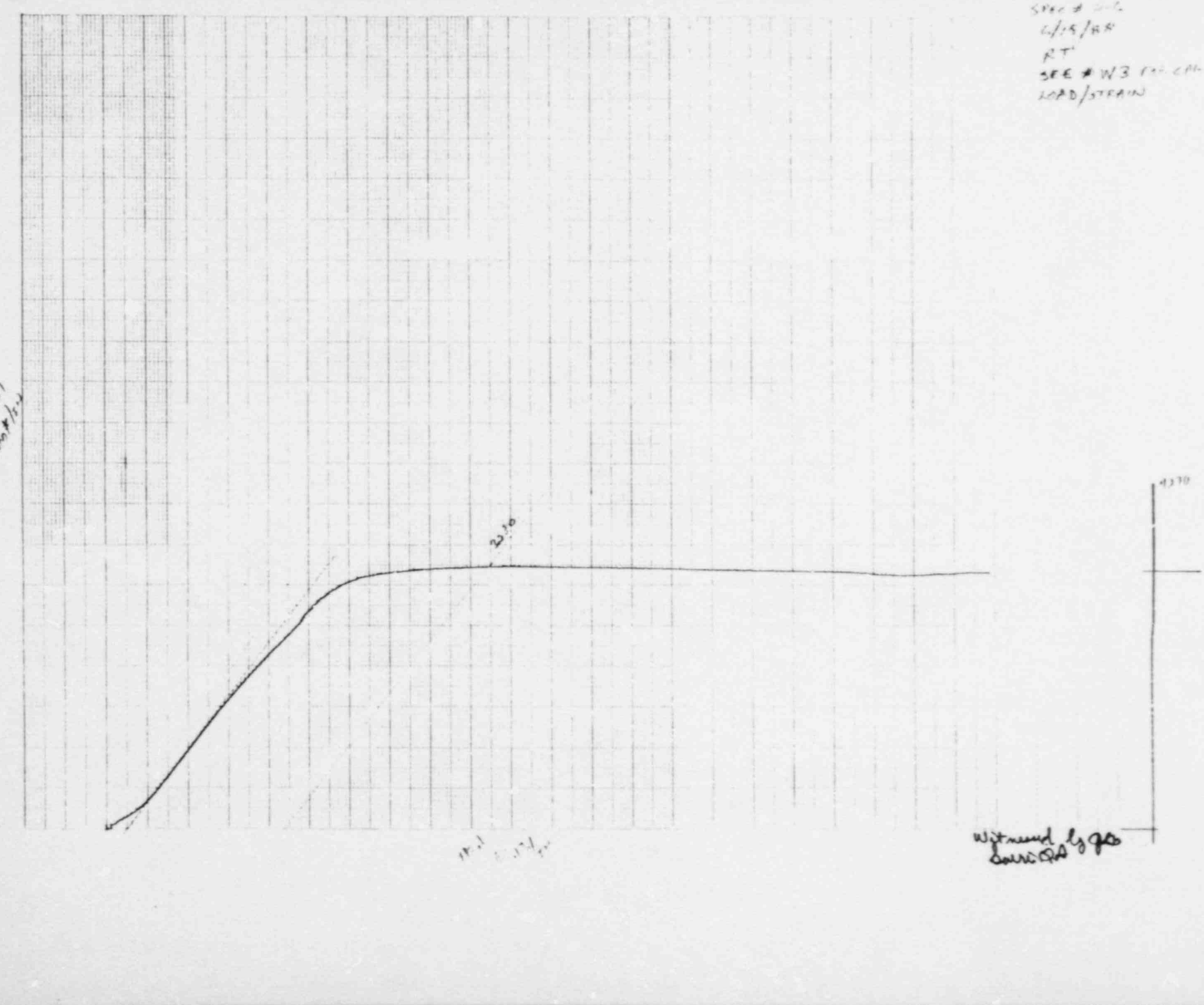
U.T.S. = Maximum Load/Initial Area	=	86,332
0.2% Y.S. = 0.2% Offset Load/Initial Area	=	65,305
Frature Stress = Fracture Load/Final Area	=	157,895
% R.A. = 100 (Init. Area-Final Area)/Init. Area	=	62.353
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L.	=	25.5
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L.	=	24.578

Test Performed by: T MADSEN / R ATIYEH

Calculations Performed by: Tom Madsen (Date) 6/19/88

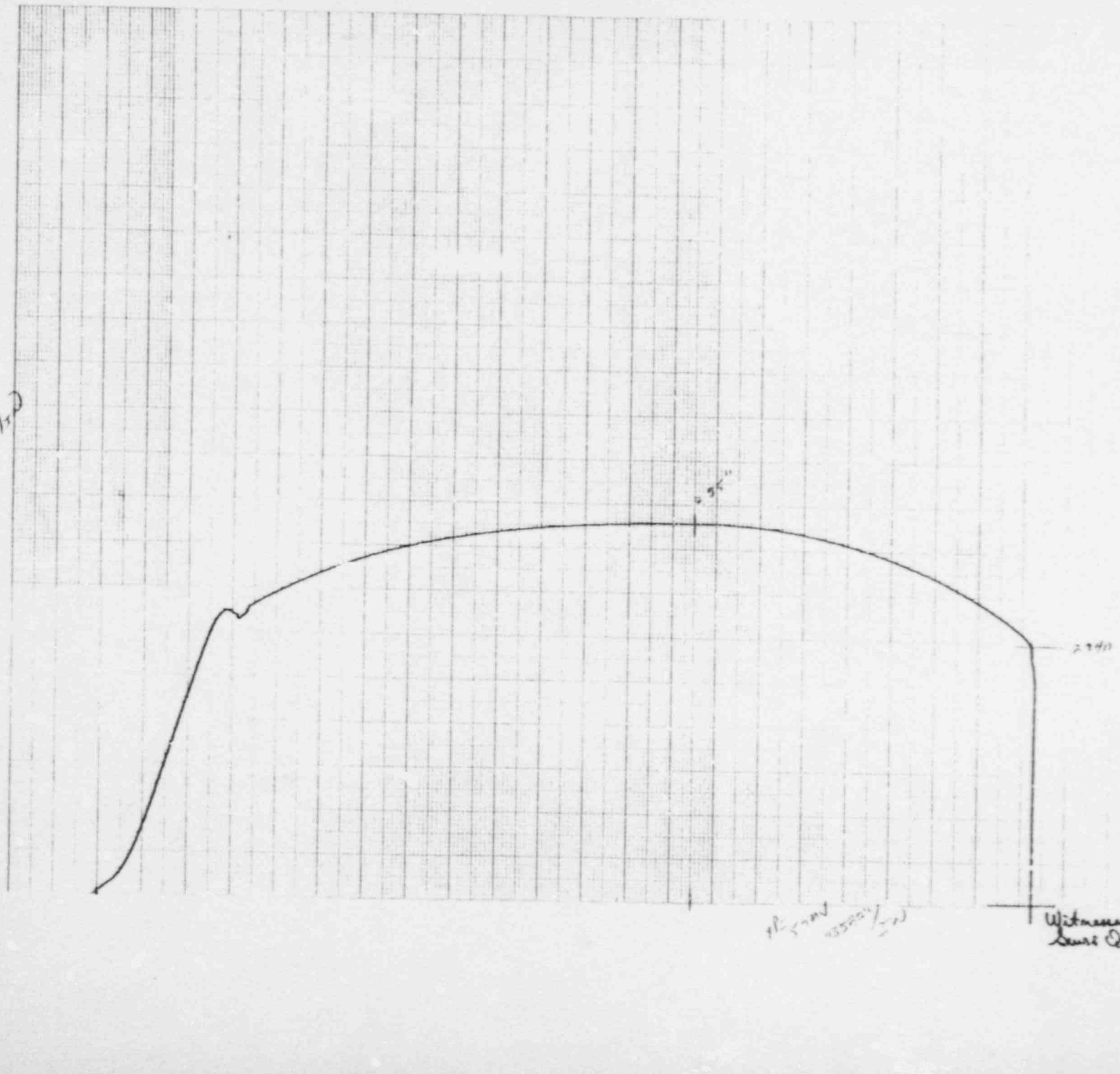
Calculations Checked by: David G Cahua Jr (Date) 6/21/88

SPEC # 2-?
6/15/88
RT
SEE # W3 FOR CAL
LOAD/STRAIN



Witnessed by [illegible]
[illegible] QA

SPEC # 2-6
6/15/80
RT
SEE #W3 FOR CAL
2:PO/H.D.



Witnessed by

Southwest Research Ins .tute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. 2-7

Project No. 17-2108

Test Temperature 550°F

Machine Ident. 4

Strain Rate .005"/IN/MIN

Date of Test 6/16/88

Initial Diameter .249
Initial Area .04867
Initial Gage Length 1.0
Specimen Temperature:
Top T.C. 551°F
Middle T.C. N/A
Bottom T.C. 548°F

Final Diameter .127
Final Area .01266
Final Gage Length 1.174
Maximum Load 4400
0.2% Offset Load 3230
Fracture Load 3170
Elong. to Max. Load .17868

Witnessed by Suri Dat. QLD

U.T.S. = Maximum Load/Initial Area = 90,405

0.2% Y.S. = 0.2% Offset Load/Initial Area = 66,365

Frature Stress = Fracture Load/Final Area = 250,395

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 73.98

% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 17.40

% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 17.868

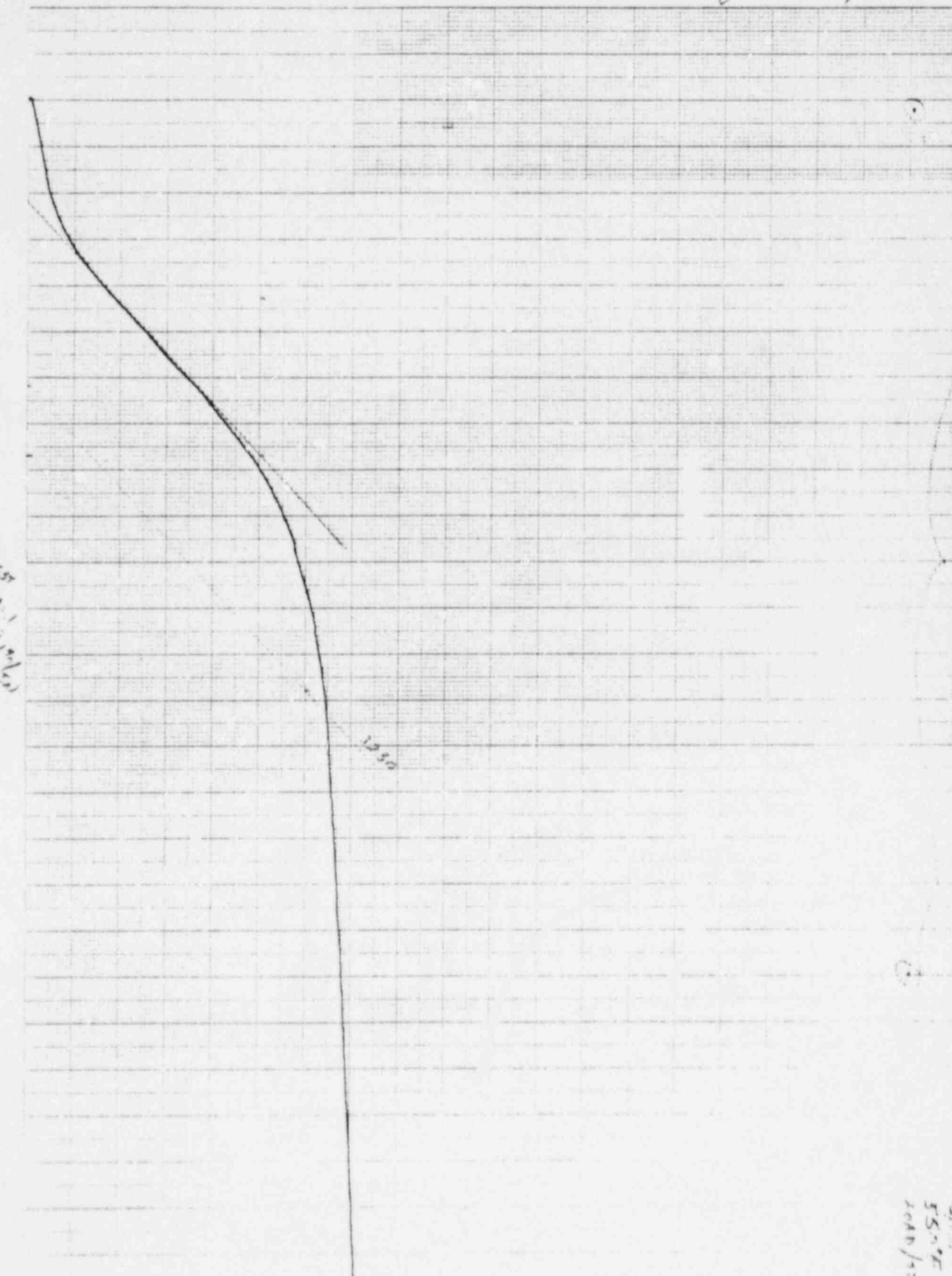
Test Performed by: T MASDEN / R. ATIYEH

Calculations Performed by: [signature] (Date) 6/17/88

Calculations Checked by: David G. Cahoon (Date) 6/21/88

47 1223

Witnessed by



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