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

FINAL REPORT
SwRI Project No. 17-2108
(Revised)

Prepared for

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

March 1990

SOUTHWEST RESEARCH INSTITUTE

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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 (October 1988) analysis of the data (1) confirmed the decrease in fluence rate from the low leakage core vs cycles prior to Cycle 6, and (2) indicated that the pressure vessel weld and plate materials will retain adequate shelf toughness throughout the 32 EFPY design life-time using the new Regulatory Guide 1.99, Revision 2. This revision of the original Final Report (October 1988) demonstrates that operation at "stretch power" may considerably reduce the benefits of the low leakage core by the end of 32 EFPY. However, the reactor pressure vessel should continue to meet Regulatory Guide 1.99, Revision 2 and PTS requirements through 32 EFPY.

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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 upon the analysis of dosimetry data at the end of Cycle 8, the fast neutron flux ($E > 1$ MeV) at Capsule V location was 1.59×10^{10} n/cm^{-2} sec^{-1}.

(2) The surveillance specimens of the core beltline plate materials experienced shifts in RT_{NDT} (from Charpy Impact curves) over the range of 80°F (46 ft-lb value for Plate B2002-2) to 239°F (50 ft-lb value for Weld) as a result of fast neutron exposure up to the 1987 refueling outage.

(3) 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 at the end of Cycle 8 operation in 8.6 EFPYs.

(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 at the vessel wall was 4.45×10^{18} n/cm^2. This gives 3.26×10^{17} n/cm^2 per EFPY for Cycles 6 through 8. The use of a low leakage core loading pattern beginning with Cycle 6 reduced the fluence rate on the pressure vessel wall by 50.6%, based upon data from surveillance capsules.

(5) The core beltline plate (B2002-3) exhibited the largest calculated adjusted RT_{NDT} (ART) change and is projected to control the heatup and cooldown limitations throughout the design lifetime of the pressure vessel.

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

(6) The Indian Point Unit No. 2 vessel plate (B2002-3) located in the core beltline region is the controlling material and is 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 Revision 2 requirements of Regulatory Guide 1.99.

(7) Based on Regulatory Guide 1.99, Rev. 2, trend curves, the projected maximum ART for the Indian Point Unit No. 2 vessel plate beltline materials at the 1/4T and 3/4T positions after 32 EFPY of operation are 240°F and 194°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. Estimated fluences for calculating 15, 20, and 32 EFPY values of ART are based upon assuming Indian Point Unit No. 2 operation at "stretch power" of 3071.4 MWL and vessel T_{avg} of 579.7°F starting from Cycle 10.

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) (3,4). 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 and nickel content is defined in Regulatory Guide 1.99, 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. Use of "stretch power" and higher vessel T_{avg} beginning with Cycle 10 increases the neutron flux by approximately 25 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. Westinghouse confirmed that the nozzle shell has three plates; the intermediate shell has three plates and the lower shell has two plates as provided in the capsule report. Plate numbers confirmed as B2003-1 and B2003-2, 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.

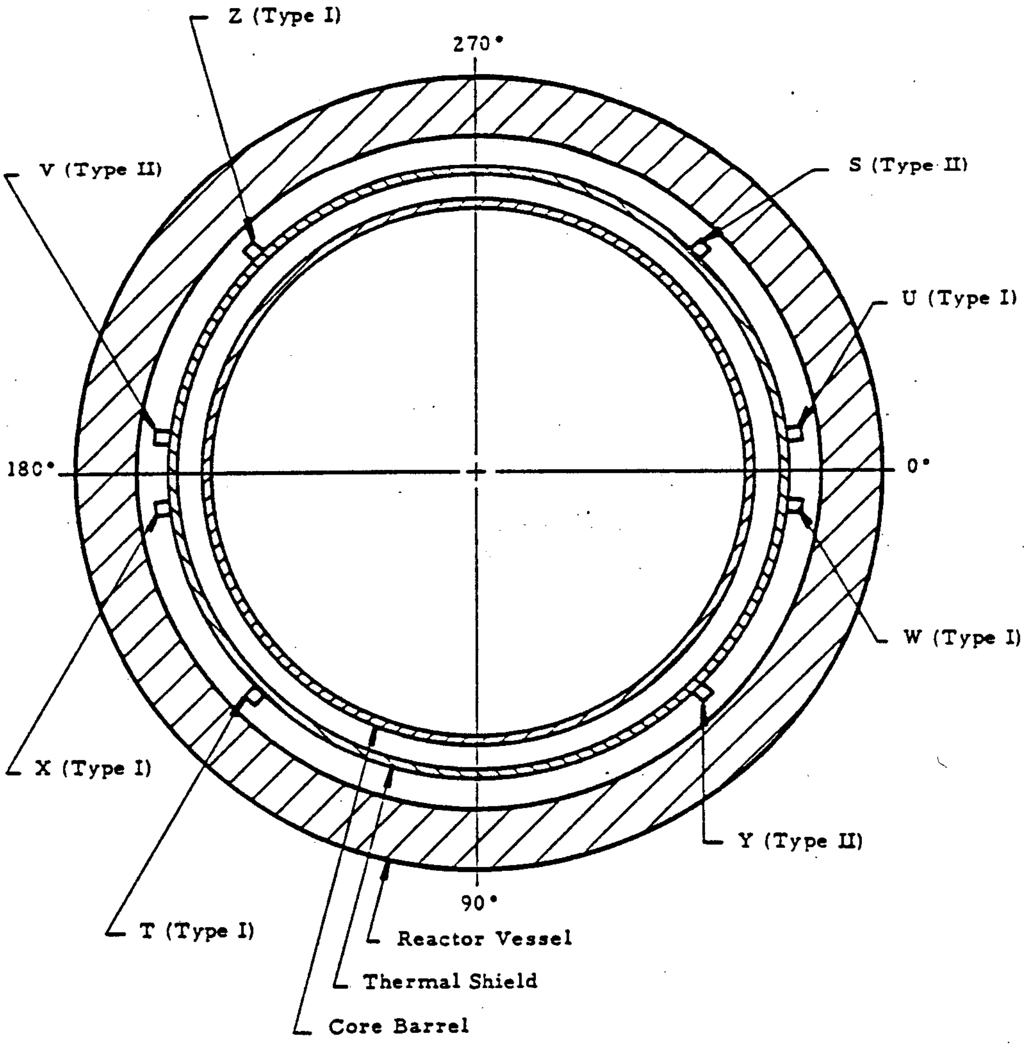


Figure III-1. Arrangement of surveillance capsules in the pressure vessel

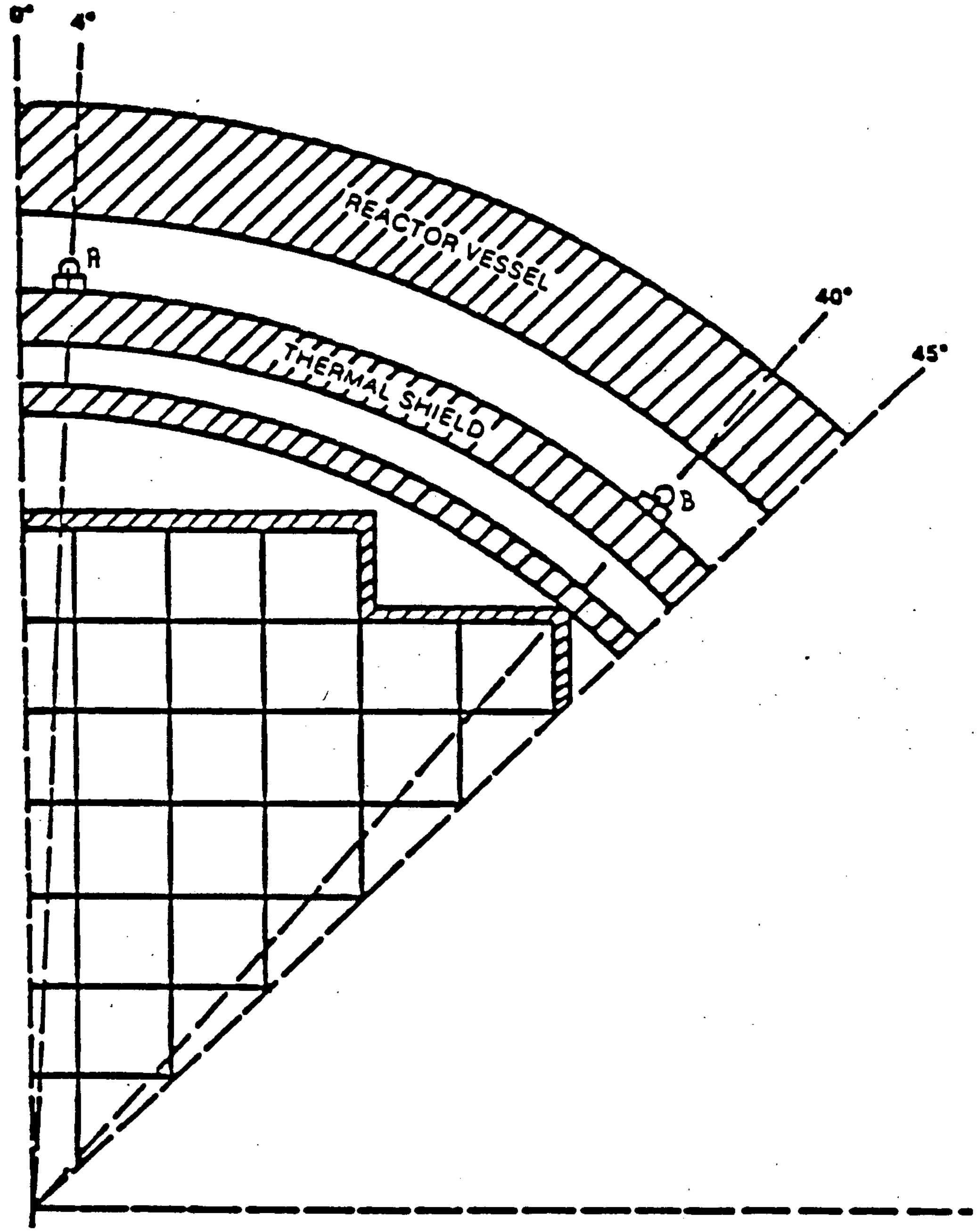


Figure III-2. Indian Point Unit 2 reactor geometry
(Reference 17)

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)

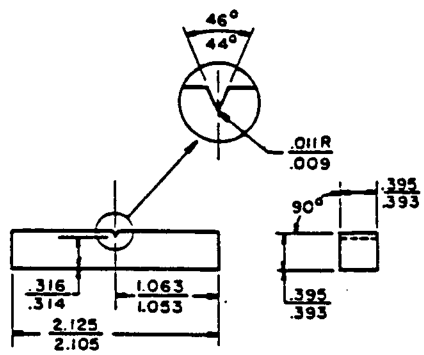
(a) Not reported in WCAP 7323 (10).

This additional information on the weld was obtained from Westinghouse in a telecon on February 2, 1990, in response to an NRC inquiry concerning the conditions under which the surveillance weld was made:

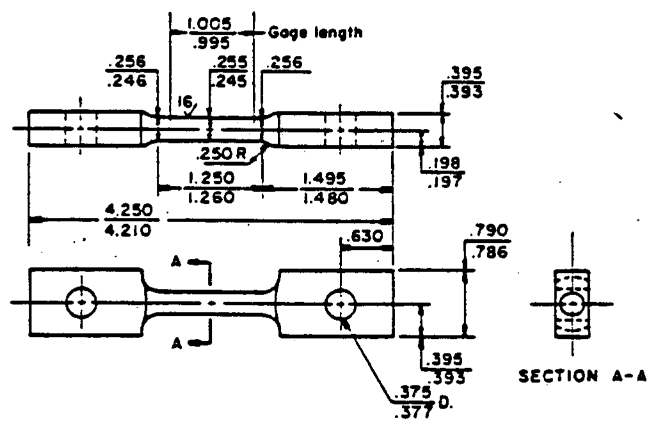
> The surveillance weld is part of the longitudinal reactor weld. The W5214 is a part of the heat number for the weld wire used in making the submerged arc weld. The complete heat number is W5214 N7048A. The weld wire type is RAC03+NI200. Cu, Ni, and Cr were not analyzed in the wire analysis. No chemistry was performed on the as-deposited weld metal. The flux used was Linde #92; lot number 3600.

In addition, the NRC requested a clarification on the number of plates used to form the lower shell section. Westinghouse confirmed that the nozzle shell has three plates; the intermediate shell has three plates; and the lower shell has two plates as provided in the capsule report. Plate numbers confirmed as B2003-1 and B2003-2.

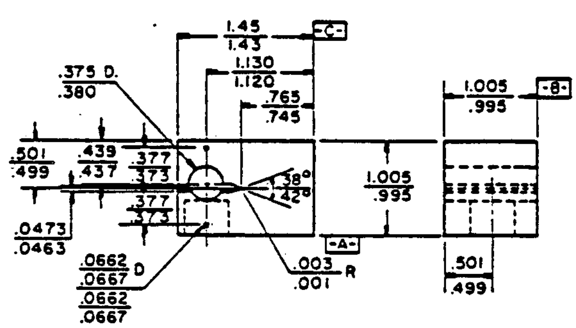
Capsule V contained 32 Charpy V-notched specimens, 4 tensile specimens (2 from weld metal and 2 from plate), and 4 base plate WOL specimens. The specimen numbering system and location within Capsule V is shown in Figures III-4 and III-5.



(a) Charpy V-notch Impact Specimen



(b) Tensile Specimen



(c) Wedge Opening Loading Specimen

Figure III-3. Vessel material surveillance specimens

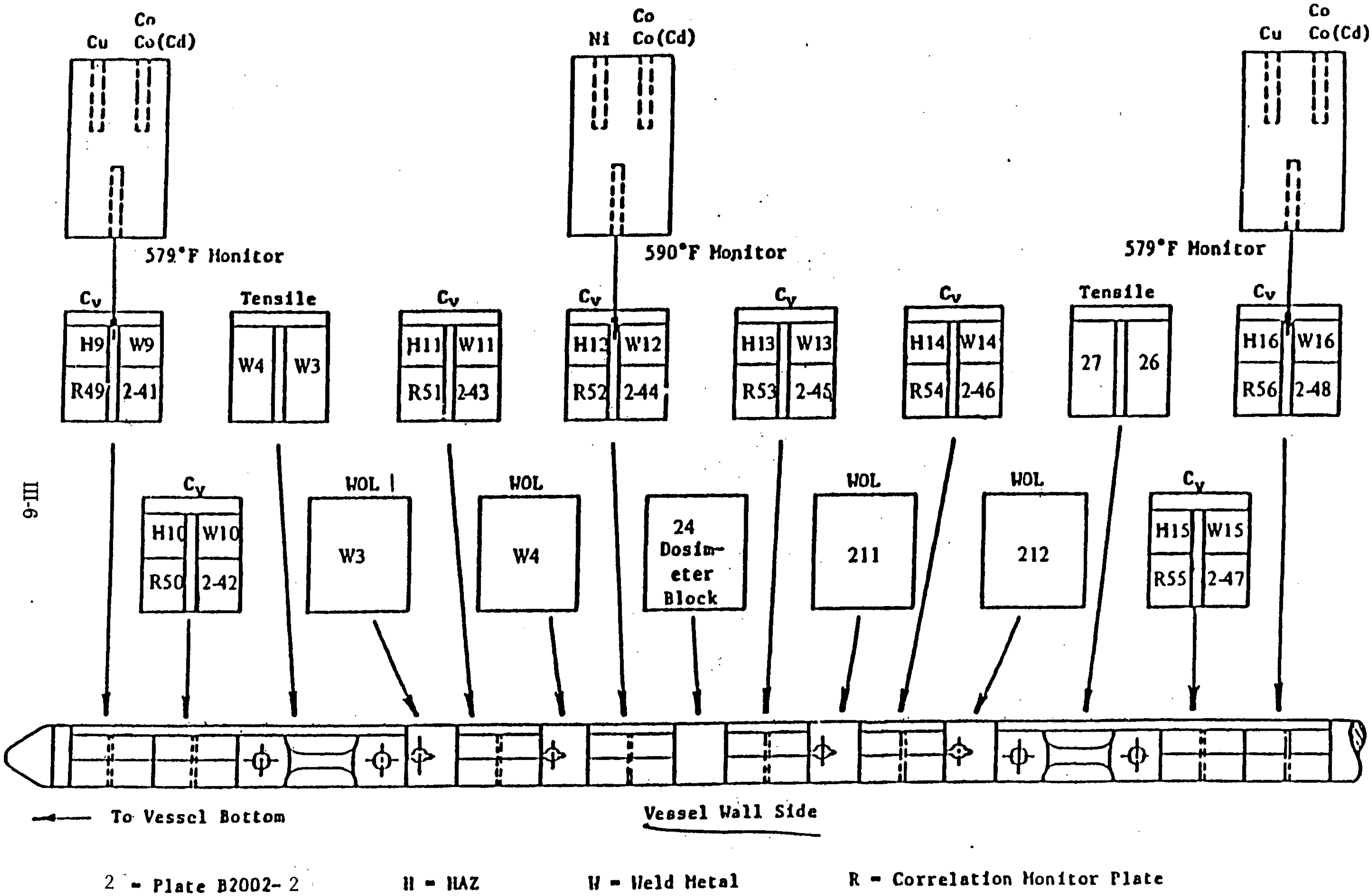


Figure III-4. Arrangement of specimens in Capsule V

Spec. No. MP 87-013

NOTE: ALL DIMENSIONS ARE IN CENTIMETERS

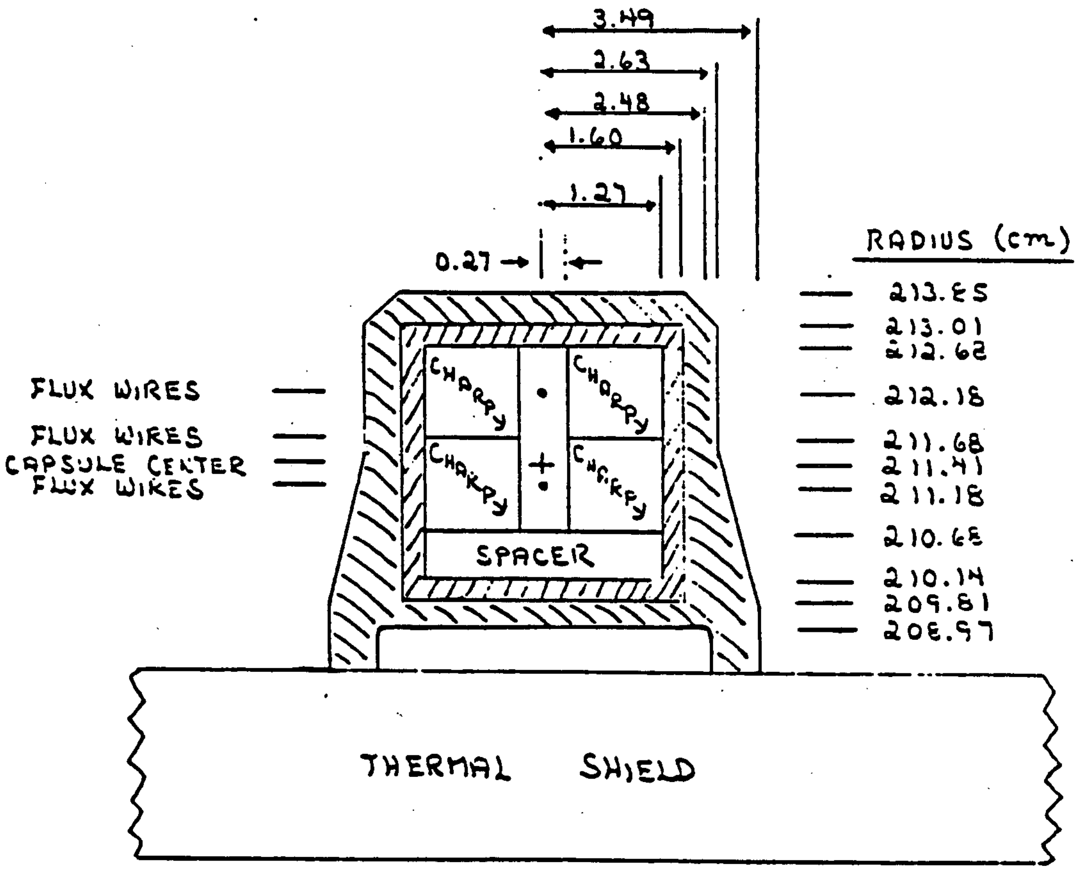


Figure III-5. Surveillance capsule geometry

(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	2
Nickel	Bare wire	1
Cobalt (in aluminum)	Bare wire	3
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 shown in Figure IV-1. The deformed condition of the Neptunium container caused the loss of most 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

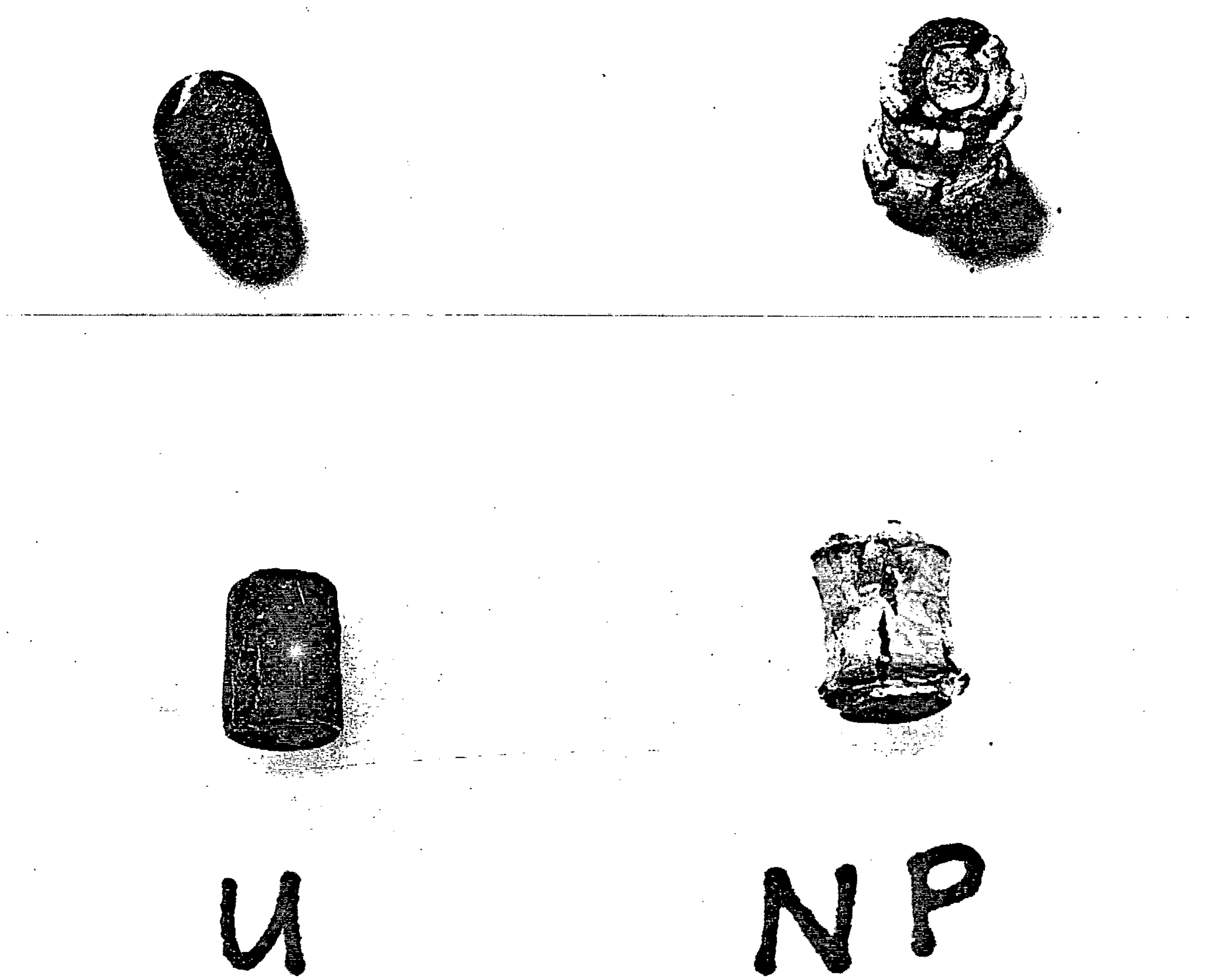


Figure IV-1. Uranium and Neptunium containers as removed from dosimeter block

^{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.

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

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
	11/10/74	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	--	0.7467
	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	05/25/78	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	--	25	--
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

and S_8 order of angular quadrature was used. The reference forward calculations were 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 neutron 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.

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

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

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

σ = 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.

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
$^{63}Cu(n,a)^{60}Co$	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

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

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 (EFPYs) up to the 1987 refueling outage, the calculated fluence rates for Capsule V from dosimetry measurements 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	16025.4	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)			
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) Measured ϕ (> 1 MeV) $= \dfrac{A_{SAT}}{N_o\,\sigma_{eff}} = \dfrac{(A_{TOR}/h)}{N_o\,\sigma_{eff}}$

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 1.59×10^{10} n/cm^2 sec, (E > 1 MeV). This is to be compared to 3.42×10^{10} n/cm^2 sec (E > 1 MeV) as reported in the "Analysis of Capsule Z,:" April 1984 (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

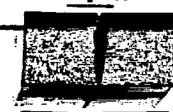
SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
W- 9	74°F	24.0	.019	0	
W-10	+130	26.5	.023	20	
W-11	+180	40.5	.035	40	
W-12	+220	53.0	.048	65	
W-13	+260	62.5	.054	95	
W-14	+300	76.0	.064	95	
W-16	+325	72.5	.065	95	
W-15	+350	76.0	.067	100	

Table IV-6

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES (CONT'D)

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

MATERIAL - B-2002-2

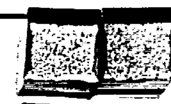
SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
2-41	74°F	17.5	.016	5	
2-42	+120	50.0	.042	15	
2-48	+150	60.5	.046	20	
2-44	+180	93.0	.059	60	
2-43	+220	111.0	.080	90	
2-45	+260	109.5	.078	100	
2-46	+300	116.0	.075	100	
2-47	+330	106.0	.067	100	

TABLE IV-7

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES (CONT'D)

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

MATERIAL - (Reference)

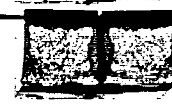
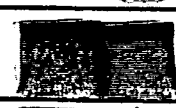
SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
R-49	74°F	13.5	.014	5	
R-50	+130	32.0	.041	20	
R-56	+150	32.5	.033	30	
R-51	+180	50.0	.044	75	
R-52	+230	62.0	.058	95	
R-53	+270	67.5	.059	100	
R-54	+320	70.5	.064	100	
R-55	+350	72.0	.062	100	

TABLE IV-8

CHARPY IMPACT DATA WITH PHOTOS OF FRACTURE FACES (CONT'D)

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

MATERIAL - (HAZ)

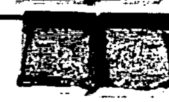
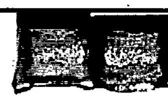
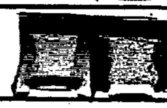
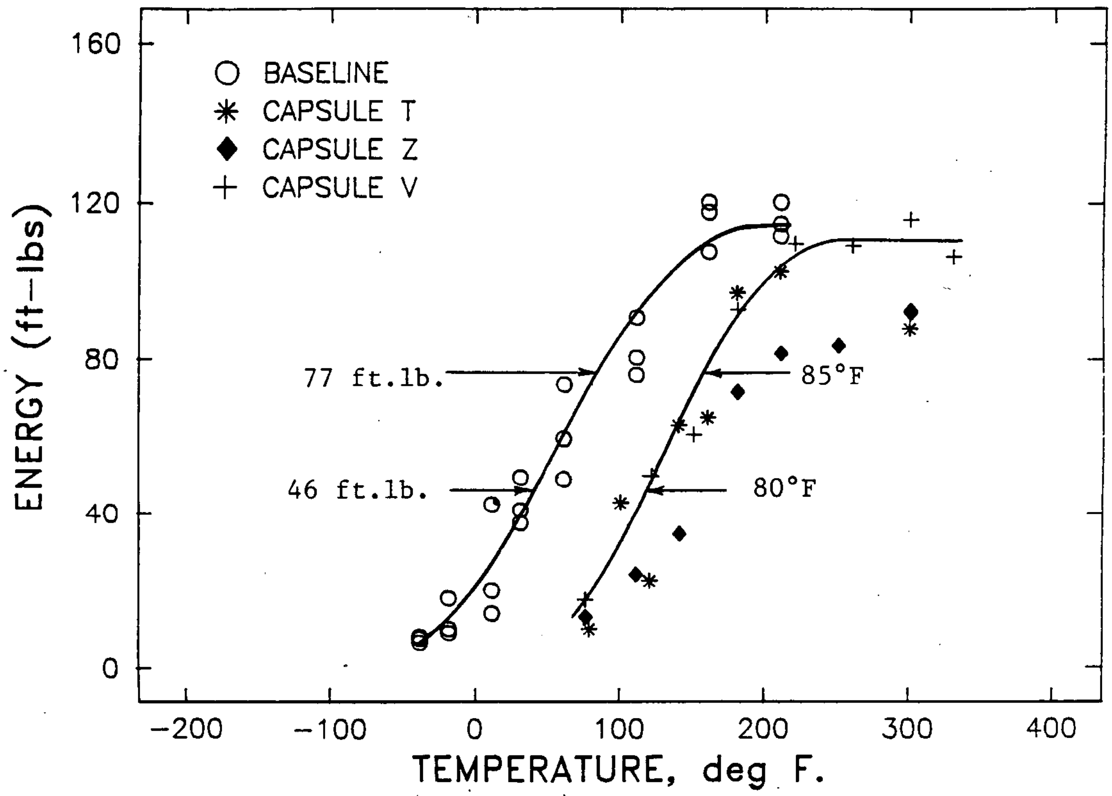
SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH 1 X
H-11	0	30.5	.023	25	
H-10	+30	85.0	.052	60	
H- 9	RT	53.5	.040	50	
H-12	+110	53.5	.047	80	
H-13	+150	65.0	.053	80	
H-14	+220	93.5	.068	100	
H-16	+250	78.0	.067	40	
H-15	+280	122.5	.077	100	

PLATE B2002-2



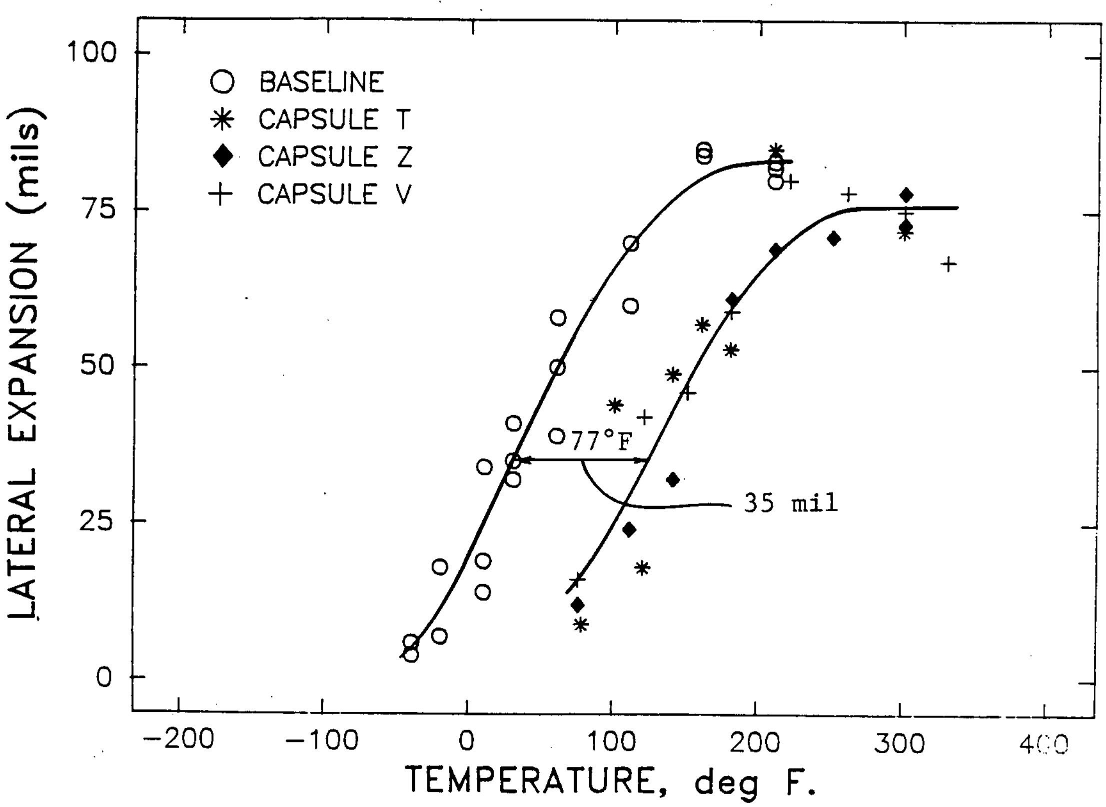


Figure IV-2. Radiation Response of Indian Point Unit 2 Shell Plate B2002-2

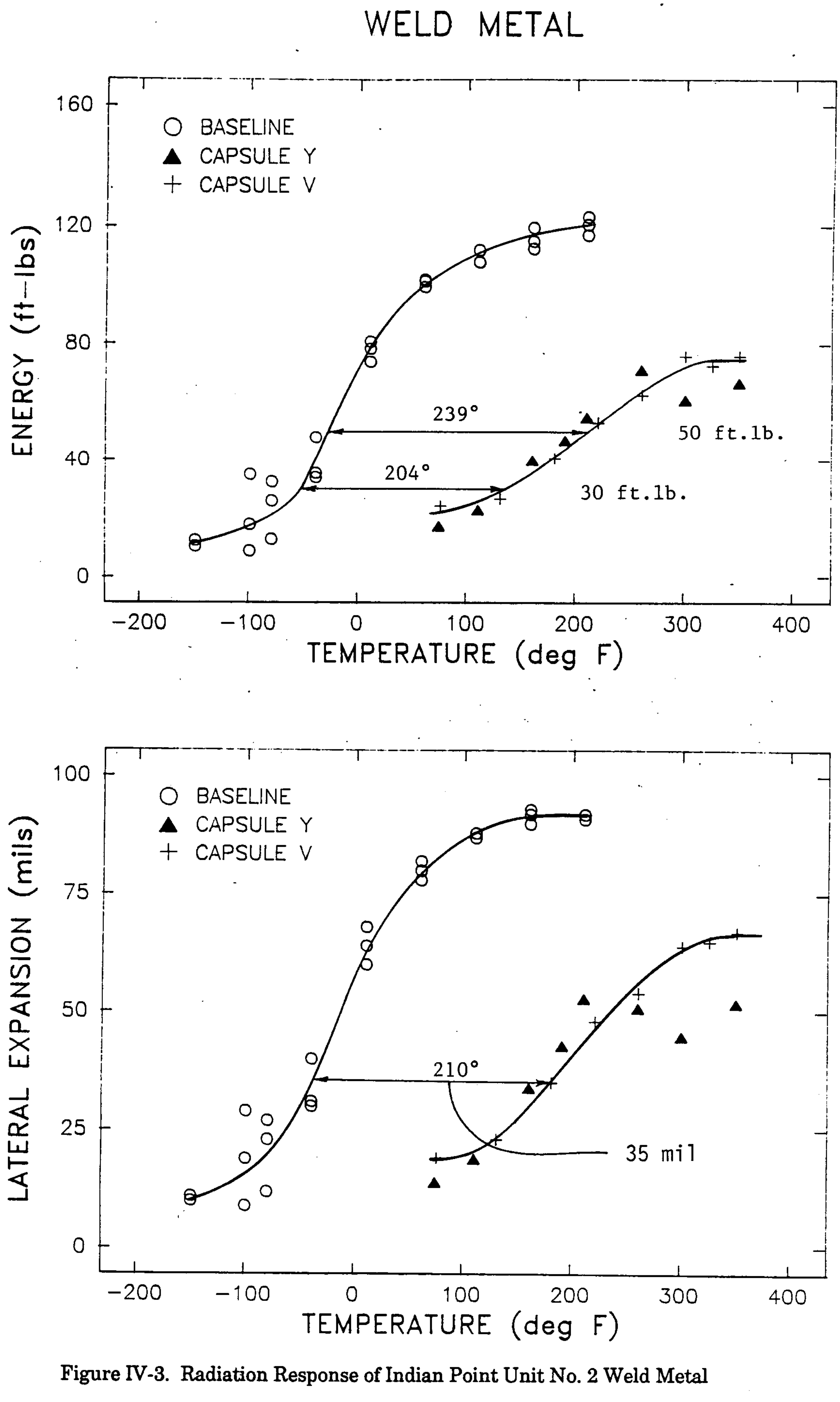


Figure IV-3. Radiation Response of Indian Point Unit No. 2 Weld Metal

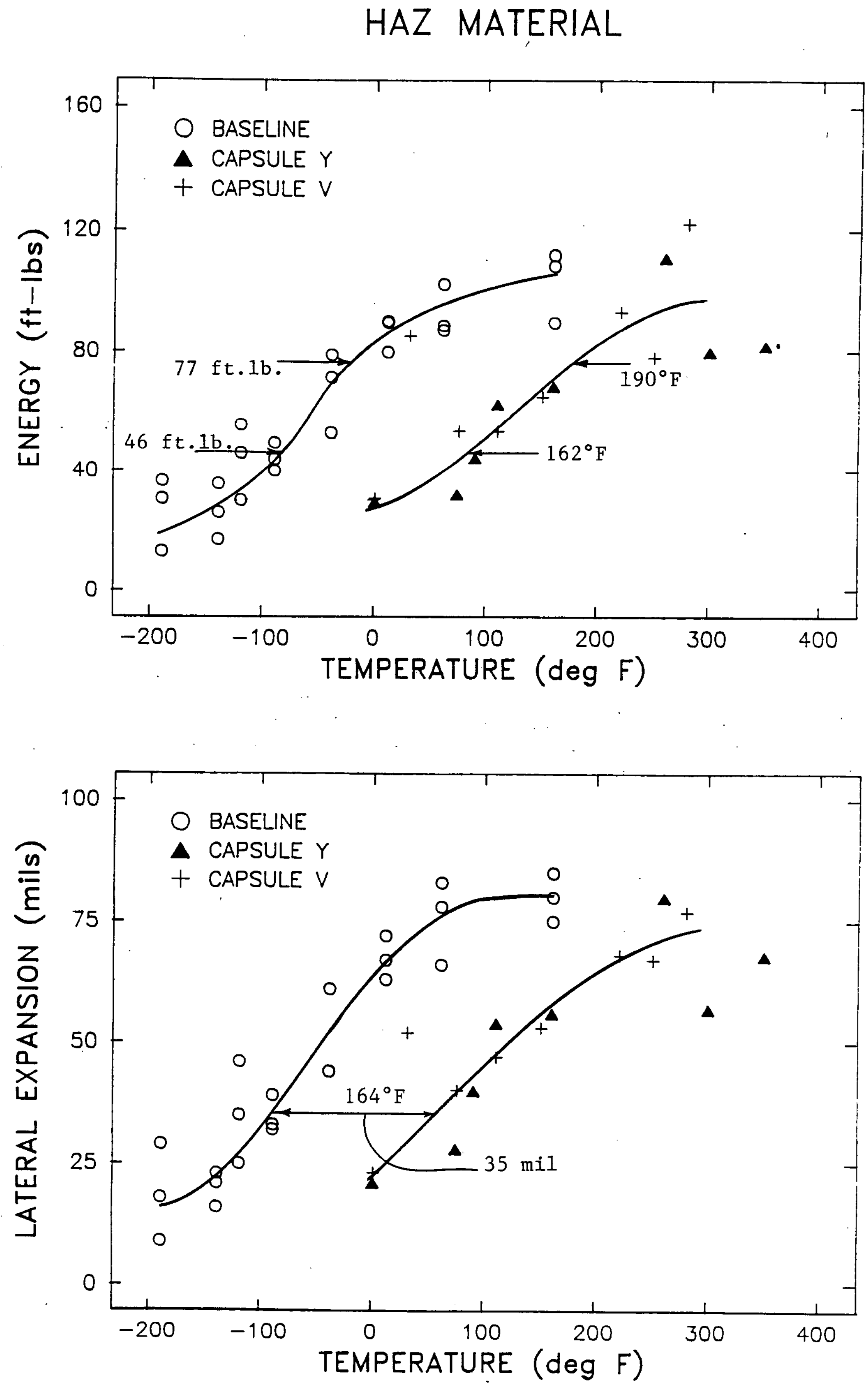


Figure IV-4. Radiation Response of Indian Point No. 2 Heat Affected Zone Material

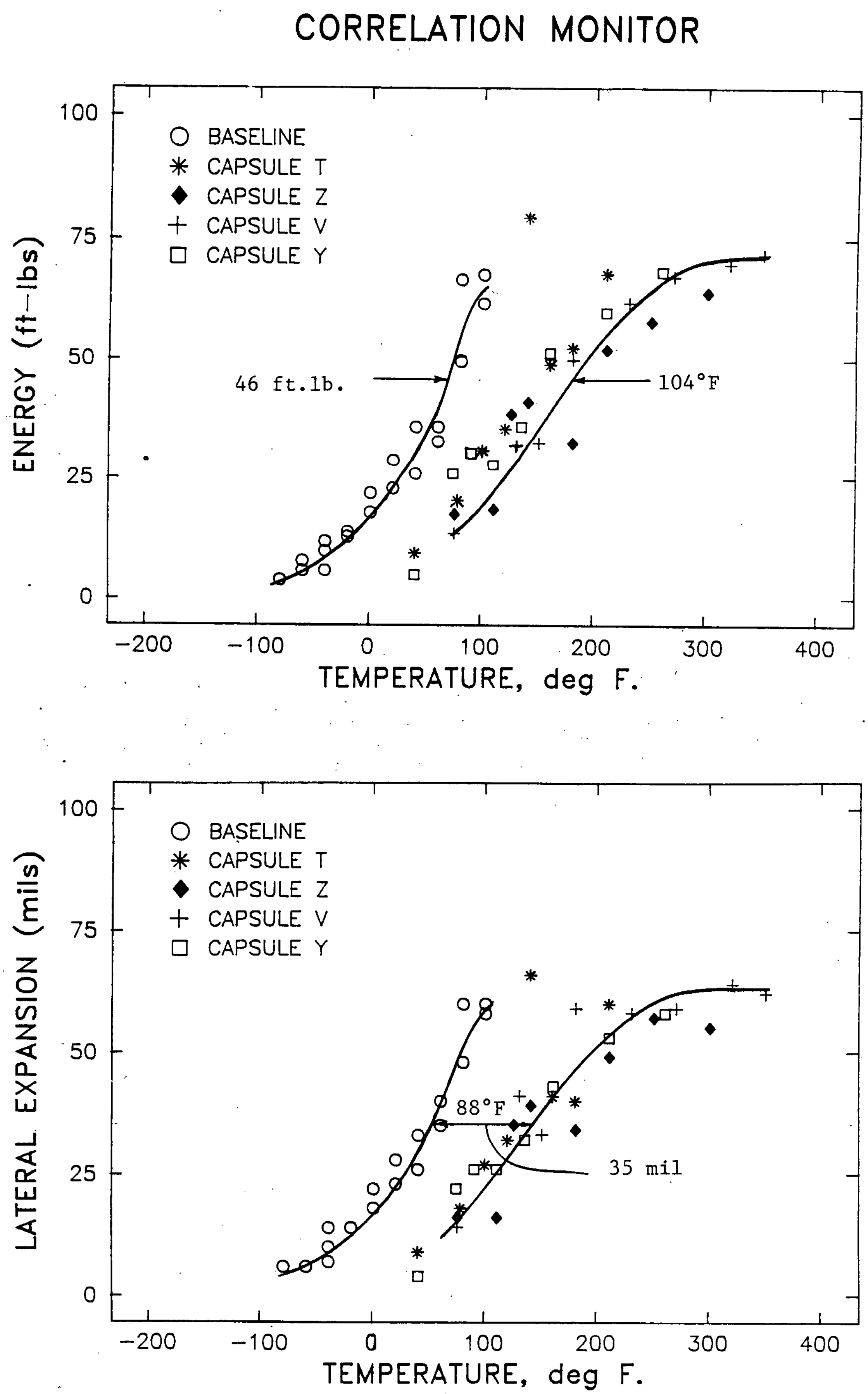


Figure IV-5. Radiation Response of Indian Point No. 2 Correlation Monitor Material

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 RT_{NDT} Values for Test Specimens in Capsule V**

Type of Material	Fluence Neutron/cm^2	Measured RT_{NDT} (°F) 50 Ft-Lbs	30 Ft-Lbs	35 mils*
Weld	5.59E18	239	204	230
		77 Ft-Lbs	46 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	Spe. 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.

The calculation of fluence up to Cycle 9 assumes a fluence rate of $6.44 \times 10^{17} n/cm^2$ per EFPY through Cycles 1 to 5 in 5.17 EFPYs and a fluence rate of $3.26 \times 10^{17} n/cm^2$ per EFPY through cycles 6 to 9 in 4.46 EFPYS. Up to Cycle 5 Indian Point 2 used a standard loading pattern and the fluence rate is based upon Capsule Z measurements (15) and starting from Cycle 6 Indian Point Unit 2 has been using a low leakage loading pattern and the fluence rate is based upon Capsule V measurements.

The projected fluence starting from Cycle 10 assumes Indian Point Unit 2 operation at 3071.4 MWt instead of 2758 MWt power level and at vessel T_{avg} of 579.7°F instead of 549°F. For the calculation of flux with an increase in T_{avg}, it was assumed that a 1°F increase in vessel T_{avg} would increase vessel flux by 0.5%.

A method for estimating the increase in RT_{NDT} as a function of neutron fluence and chemistry is given in Regulatory Guide 1.99, Revision 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. The low flux leakage core loading produced a 48.9% reduction in fast neutronflux (E > 1 MeV) for Cycles 6 through 8 as compared to the first 5 cycles. Revision 2 results from Capsule V are included in this section.

The B2002-3 plate continues to be the controlling material 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). Table V-2 is a comparison of measured and calculated RT_{NDT} values. This revision of the original Final Report (October 1988) demonstrates that operation at "stretch power" may considerably reduce the benefits of the low leakage core by the end of 32 EFPY. However, the reactor pressure vessel should continue to meet Regulatory Guide 1.99, Revision 2, and PTS requirements through 32 EFPY.

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 2 (8). 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 2, for nominal weld chemistries of 0.20% Cu and 1.03% Ni and plate chemistries of 0.25% Cu and 0.60% Ni. Extrapolation to 1.39×10^{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.

Results are obtained using Revision 2 of Regulatory Guide 1.99 for Capsule V materials in Figure V-1. Extrapolation to 32 EFPY fluence of 1.39 x 10^{19} n/cm^2 on Figure V-1 gives predicted values below upper limit for weld metal and plate controlling 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 with 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 for the same case agrees within the measurement uncertainties as shown in Table V-4.

Table V-1

ADJUSTED RT_{NDT} VALUES FOR INDIAN POINT-2

Time	Material	Location	Initial RT_{NDT}	Fluence[(a)] (>1 MeV) DPA	RT_{NDT} Rev. 2	ART (Adjusted RT_{NDT}) Rev. 2	Margin	RT_{PTS}
EOC8 [8.6 EFPY]	B2002-3 (Plate)	OT	21°F	4.45E18	136	205*		
		1/4T	21	2.9E18	116	185		
		3/4T	21	1.1E18	77	146		
	HAZ	OT	0°F	4.5E18	67	115		
		1/4T	0	2.9E18	57	105		
		3/4T	0	1.1E18	38	86		
	Weld[(b)]	OT	-56°F	4.5E18	175	185		
		1/4T	-56	2.9E18	149	159		
		3/4T	-56	1.1E18	99	109		
15 EFPY	B2002-3 (Plate)	OT	21°F	6.99E18	158	227*		
		1/4T	21	4.54E18	137	206		
		3/4T	21	1.75E18	95	164		
	HAZ	OT	0°F	6.99E18	77	125		
		1/4T	0	4.54E18	67	115		
		3/4T	0	1.75E18	46	94		
	Weld[(b)]	OT	-56°F	6.99E18	203	214		
		1/4T	-56	4.54E18	176	187		
		3/4T	-56	1.75E18	122	132		
20 EFPY	B2002-3 (Plate)	OT	21°F	9.03E18	171	240*		
		1/4T	21	5.87E18	150	219		
		3/4T	21	2.26E18	105	174		
	HAZ	OT	0°F	9.03E18	84	132		
		1/4T	0	5.87E18	73	121		
		3/4T	0	2.26E18	52	100		
	Weld[(b)]	OT	-56°F	9.03E18	220	230		
		1/4T	-56	5.87E18	192	203		
		3/4T	-56	2.26E18	136	146		
32 EFPY	B2002-3 (Plate)	OT	21°F	1.39E19	192	261*	48	244
		1/4T	21	9.04E18	171	240	48	225
		3/4T	21	3.48E18	125	194	48	189
	HAZ	OT	0°F	1.39E19	94	142		
		1/4T	0	9.04E18	83	131		
		3/4T	0	3.48E18	61	109		
	Weld[(b)]	OT	-56°F	1.39E19	247	257	66	181
		1/4T	-56	9.04E18	220	230	66	162
		3/4T	-56	3.48E18	160	170	66	127

(a) The actual 3/4T and 1/4T 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 3/4T and 1/4T 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, chemistry factor is 226 F, for Rev. 2 analysis.

* Plate is controlling material, 0.25 percent Cu and 0.6 percent Ni and chemistry factor is 176 for Rev. 2 analysis.

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

*Flux at each position from transport calculations normalized to inner vessel wall (flux from transport calc./flux at inner vessel wall)

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. 2	Reg. Guide 1.99 Rev. 2 + Margin
Plate B2002-2	80	73	121
Weld	204	175	241
HAZ	162	125[b] 67[b]	191[b] 115[b]
Correlation Monitor	104	104[c]	152

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

(b) Based on Weld and Base Plate Calculations, respectively

(c) Based on averaged values from plate (B2002-2 and B2002-3) since chemical values not reported in WCAP 7323.

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	1.51E19	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 3071.4 Mwt pressurized water reactor operated by Consolidated Edison 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) and continue to indicate that the plate 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 240°F at the 1/4T and 194°F at the 3/4T vessel wall locations, as controlled by plate 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 0°F, 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, VI-3, VI-4, VI-5, and VI-6.

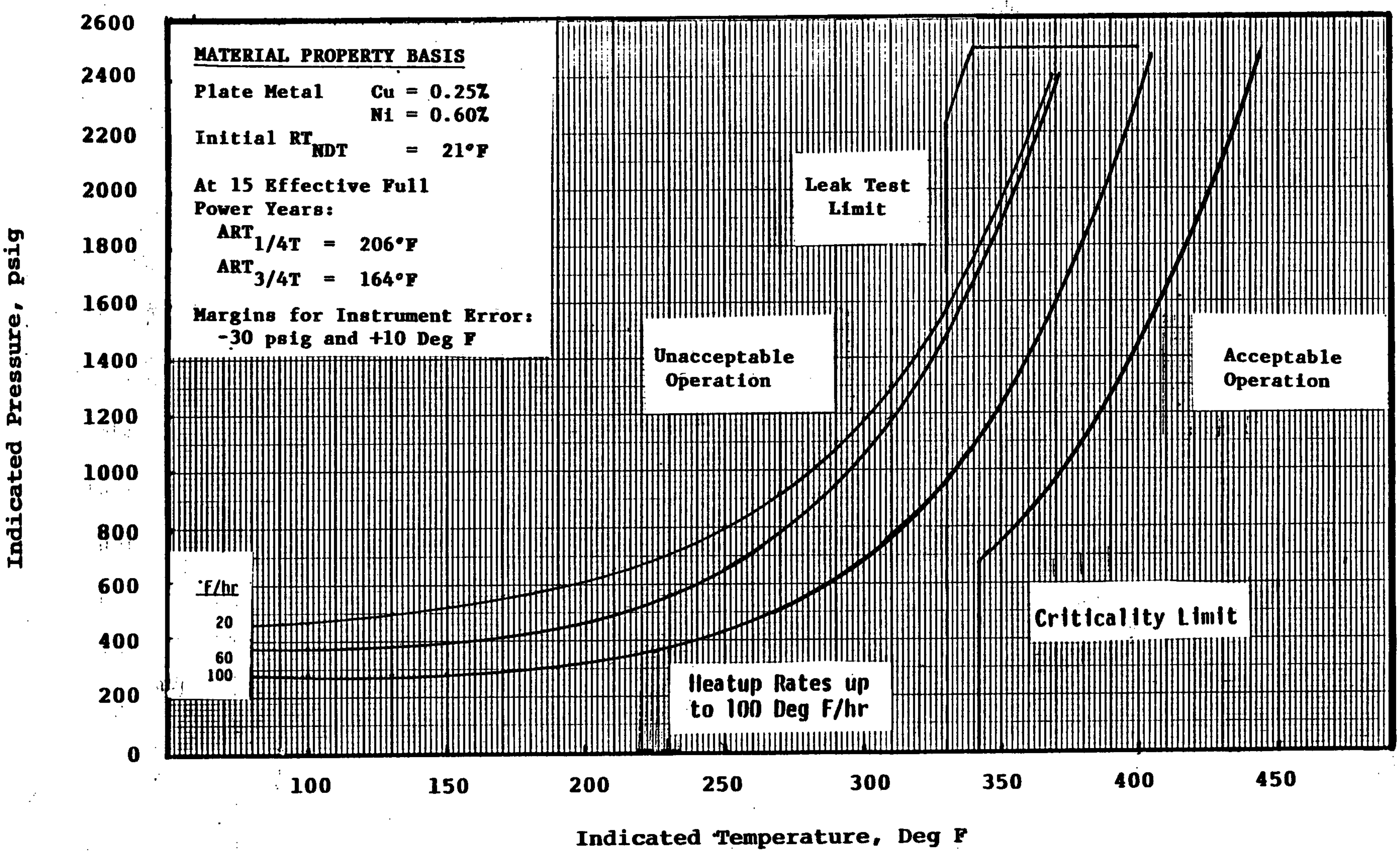


Figure VI-1. Indian Point Unit No. 2 reactor coolant heatup limitations applicable for periods up to 15 effective full power years (with criticality limit)

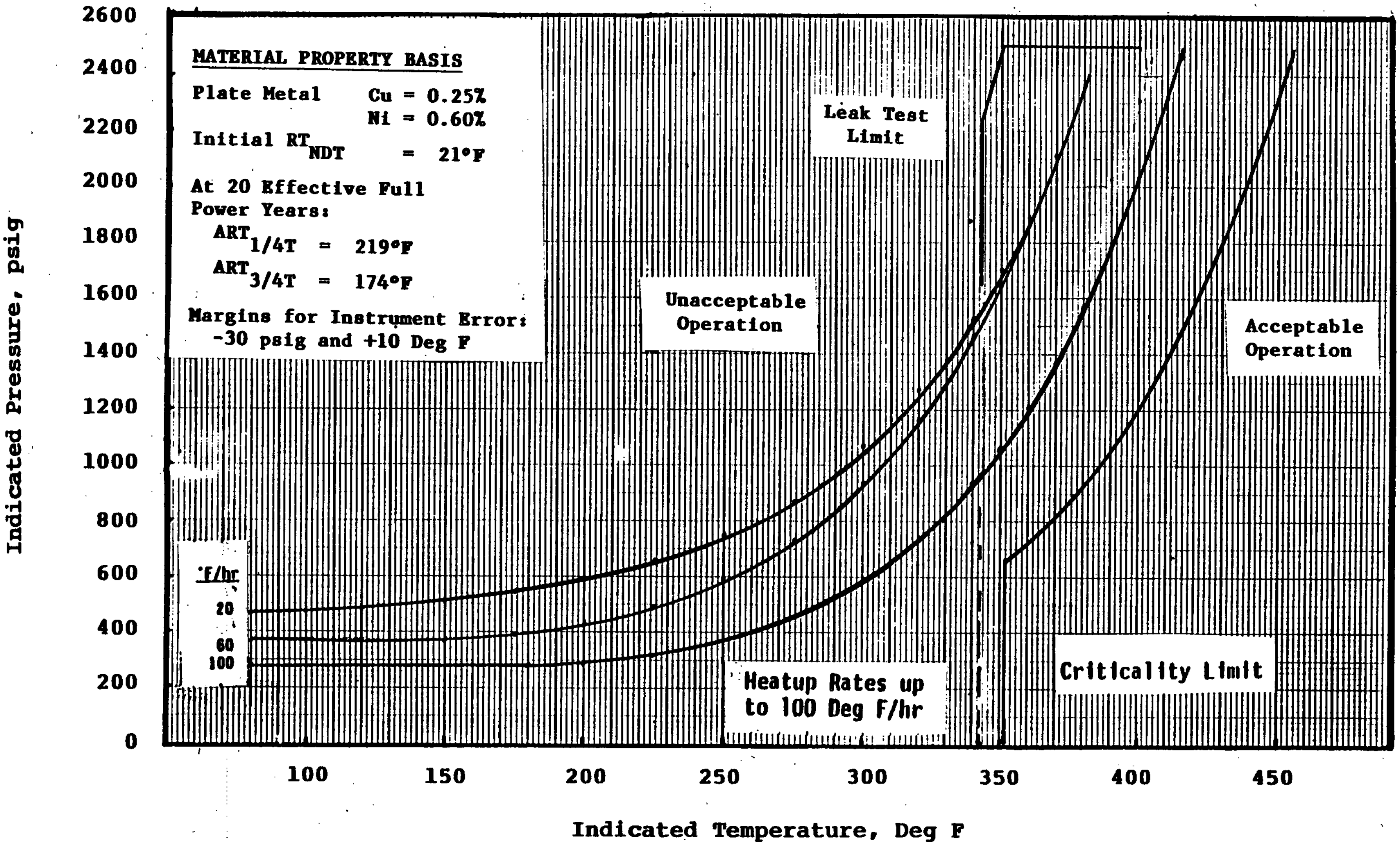


Figure VI-2. Indian Point Unit No. 2 reactor coolant heatup limitations applicable for periods up to 20 effective full power years (with criticality limit)

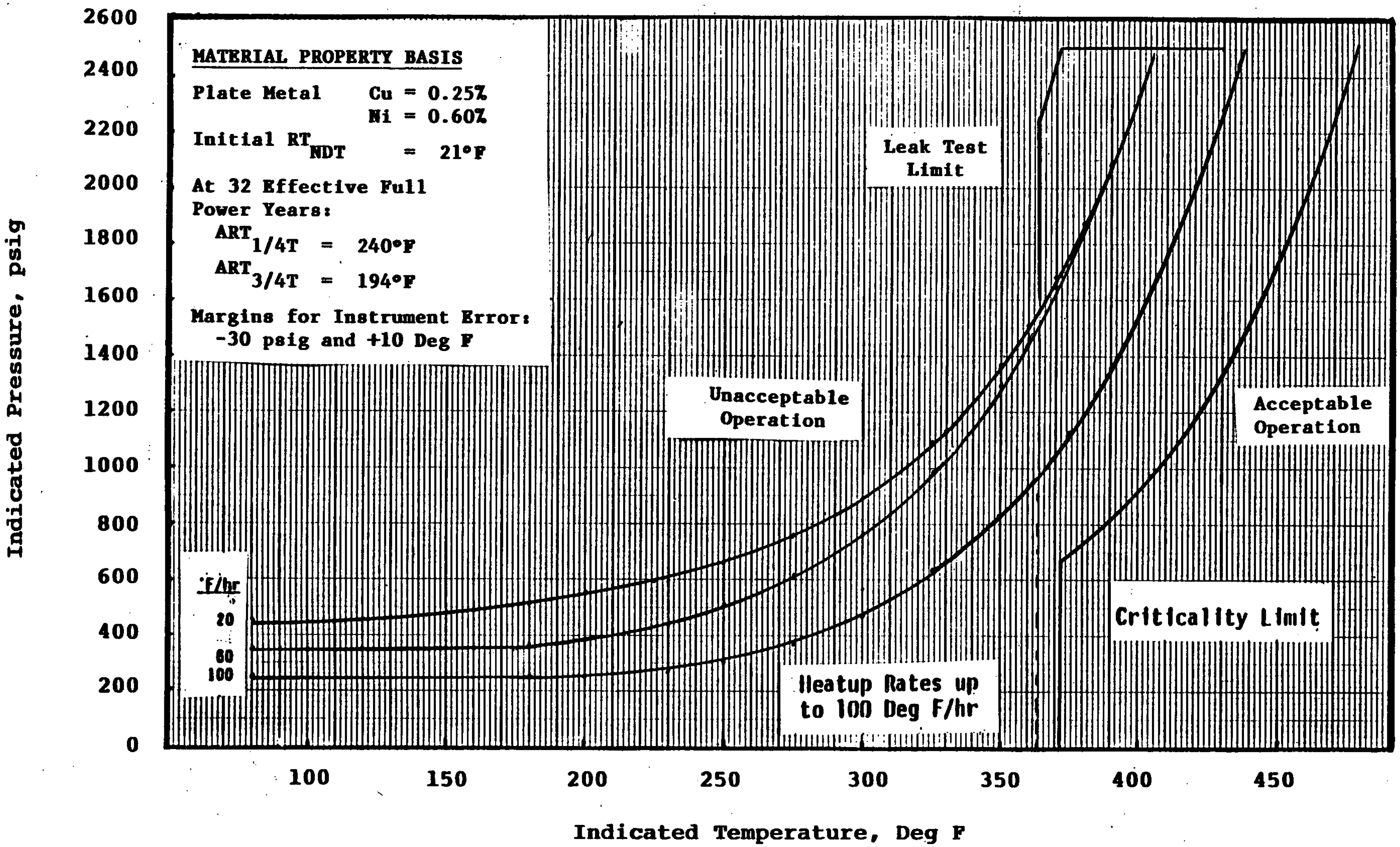


Figure VI-3. Indian Point Unit No. 2 reactor coolant heatup limitations applicable for periods up to 32 effective full power years (with criticality limit)

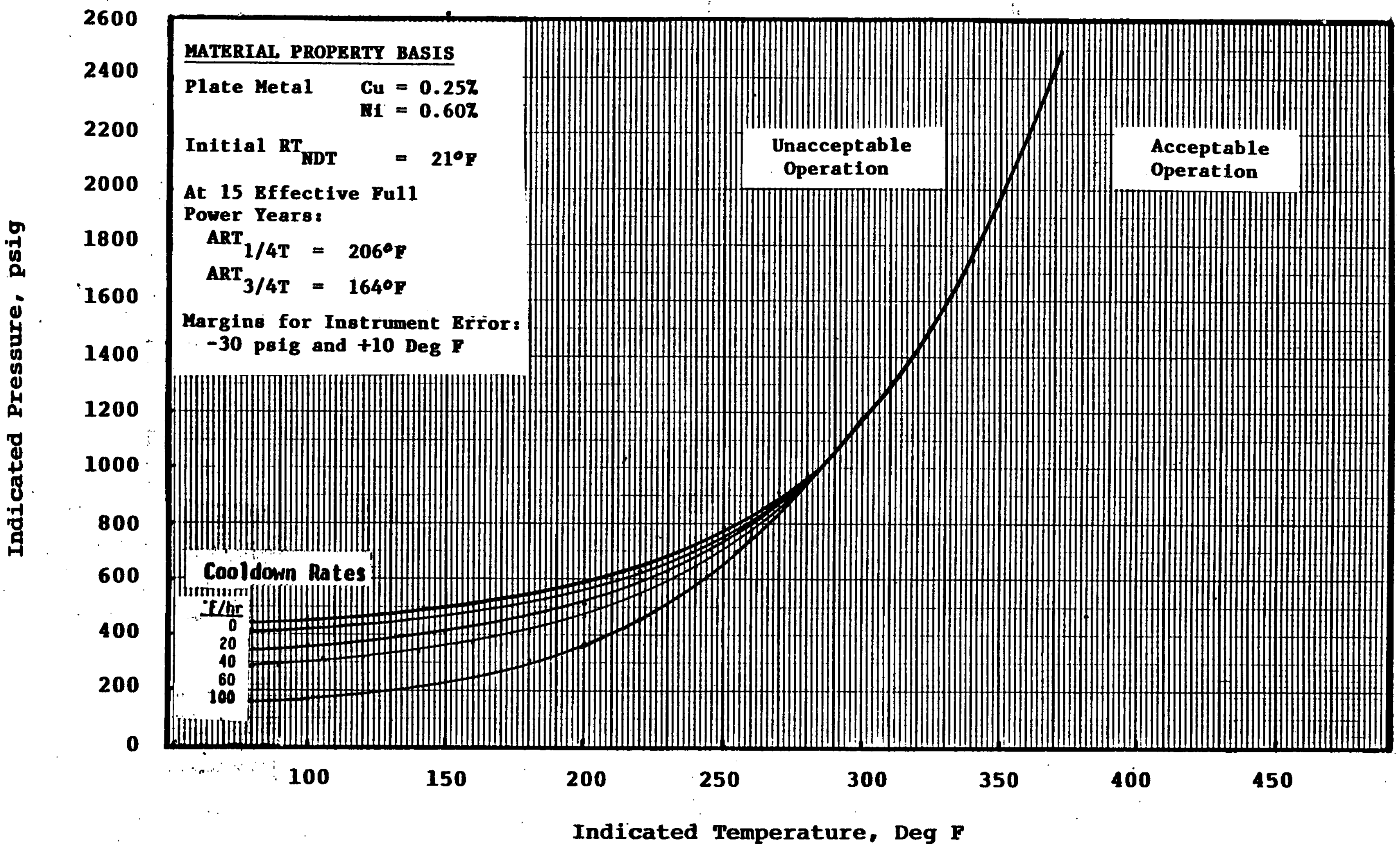


Figure VI-4. Indian Point Unit No. 2 reactor coolant cooldown limitations applicable for periods up to 15 effective full power years (with criticality limit)

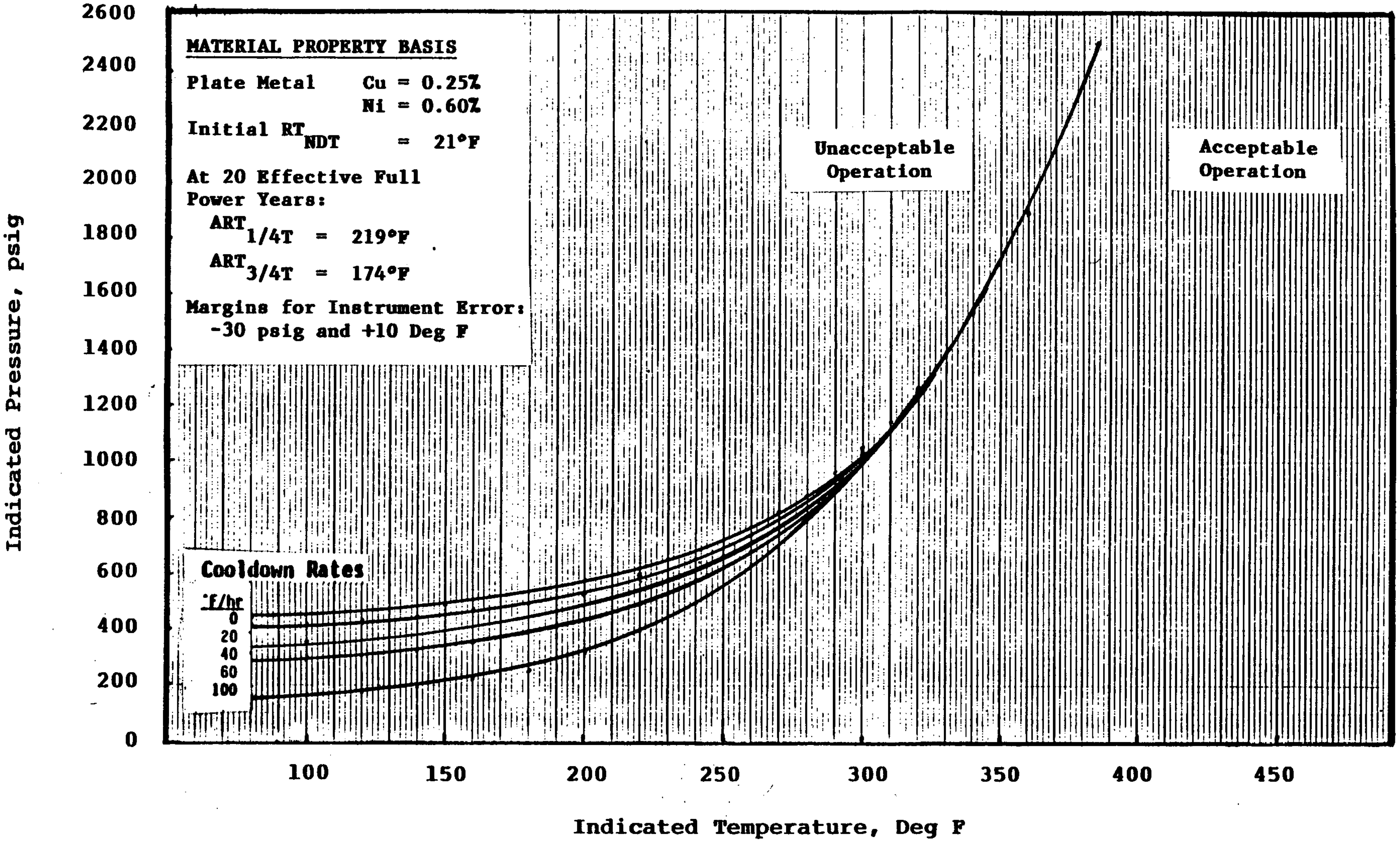


Figure VI-5. Indian Point Unit No. 2 reactor coolant cooldown limitations applicable for periods up to 20 effective full power years (with criticality limit)

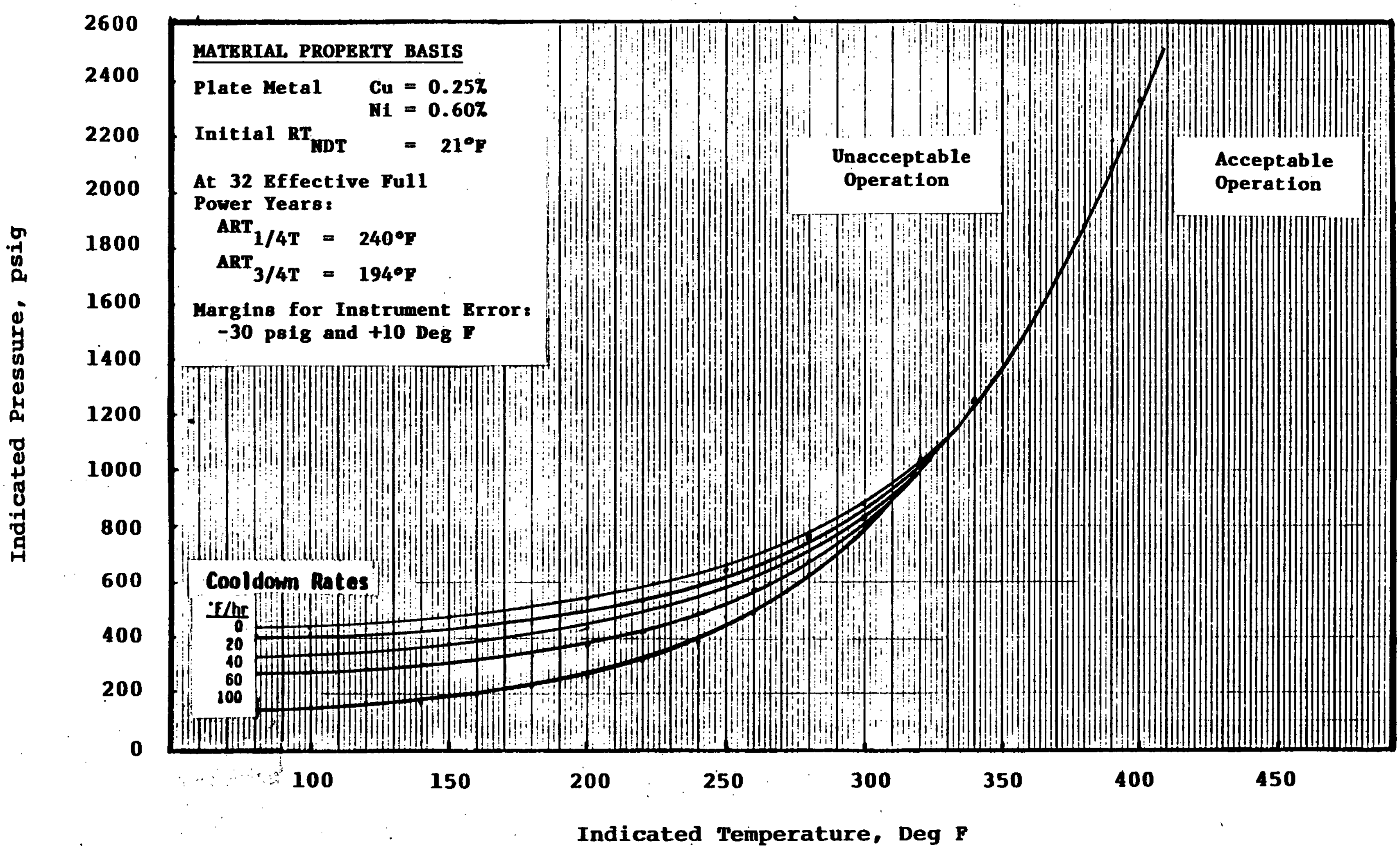


Figure VI-6. Indian Point Unit No. 2 reactor coolant cooldown limitations applicable for periods up to 32 effective full power years (with criticality limit)

VII. REFERENCES

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

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

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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. Office of Standards Development, U.S. Nuclear Regulatory Commission, 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.

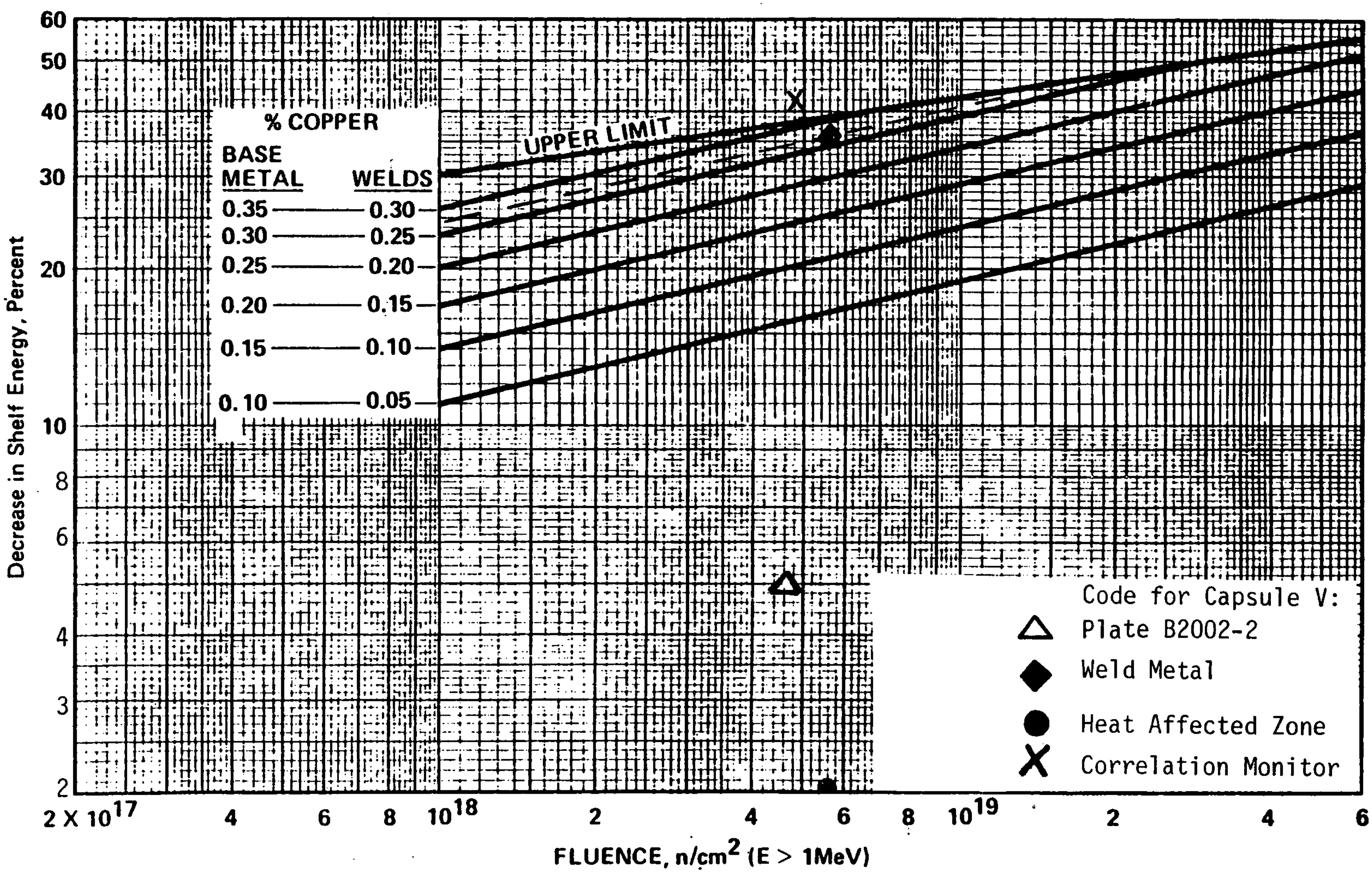


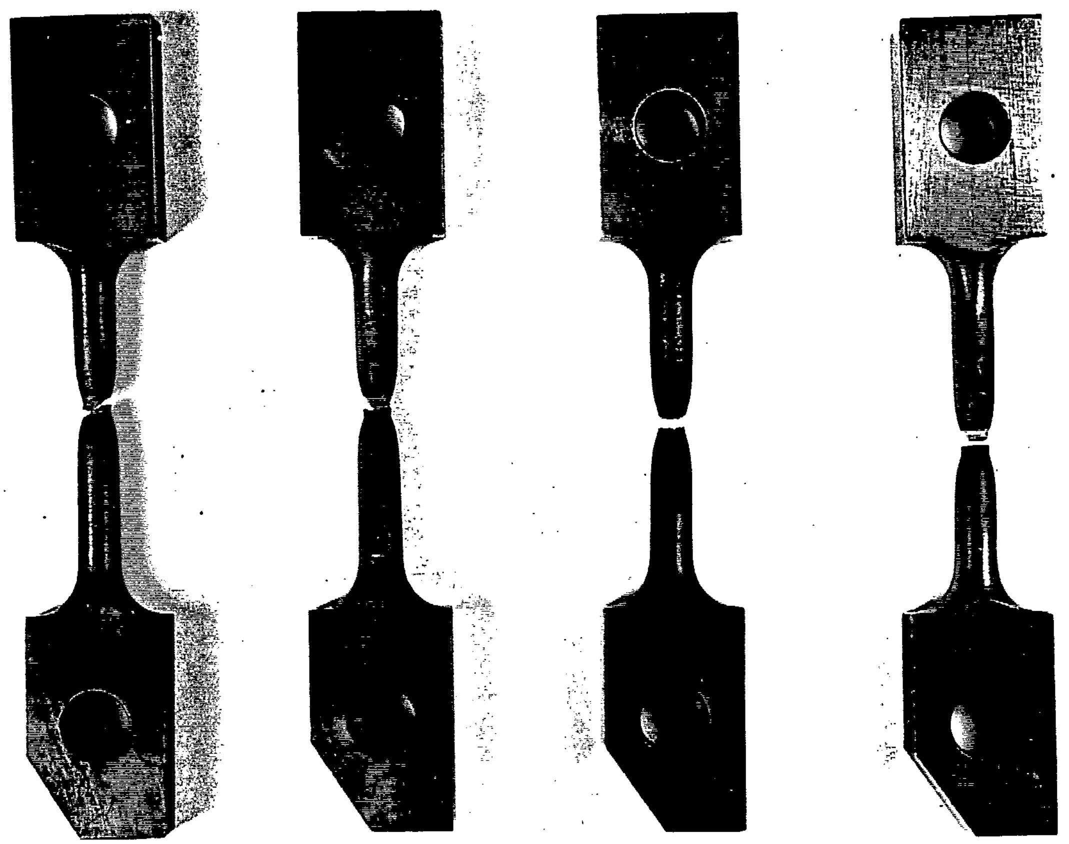
Figure V-1. Predicted decrease in shelf energy as a function of copper content and fluence
(Adapted from Regulatory Guide 1.99, Revision 2; Data from Table IV-9)

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. W3

Project No. 17-2108

Test Temperature RT

Machine Ident. 4

Strain Rate .005"/IN/MIN

Date of Test 6/15/88

Initial Diameter	.247	Final Diameter	.153
Initial Area	.04789	Final Area	.01838
Initial Gage Length	1.0	Final Gage Length	1.220
Specimen Temperature:		Maximum Load	5120
Top T.C.		0.2% Offset Load	4440
Middle T.C.	75°F	Fracture Load	3460
Bottom T.C.		Elong. to Max. Load	.2099

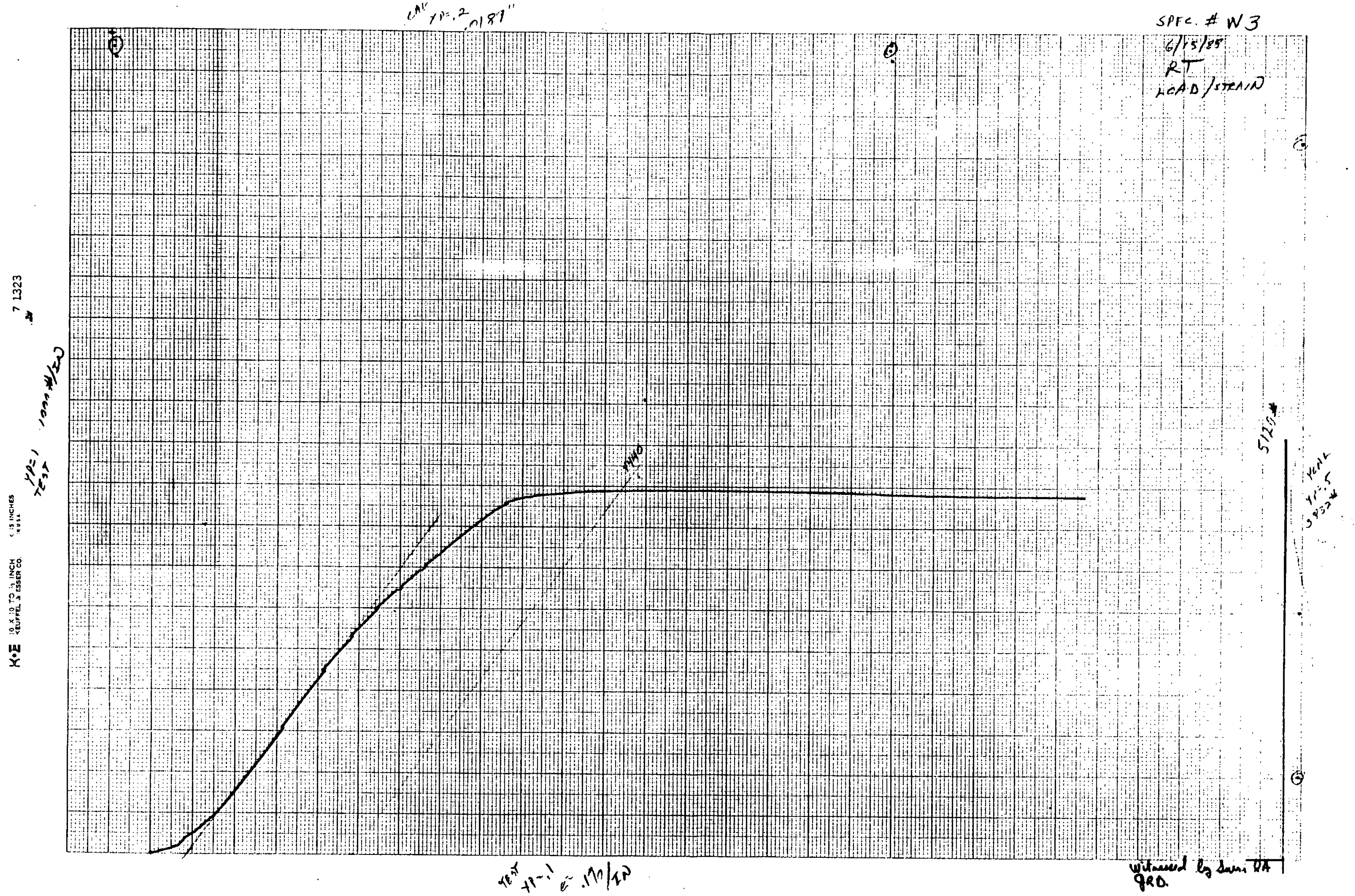
Witnessed by Sun Q.A JLD

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

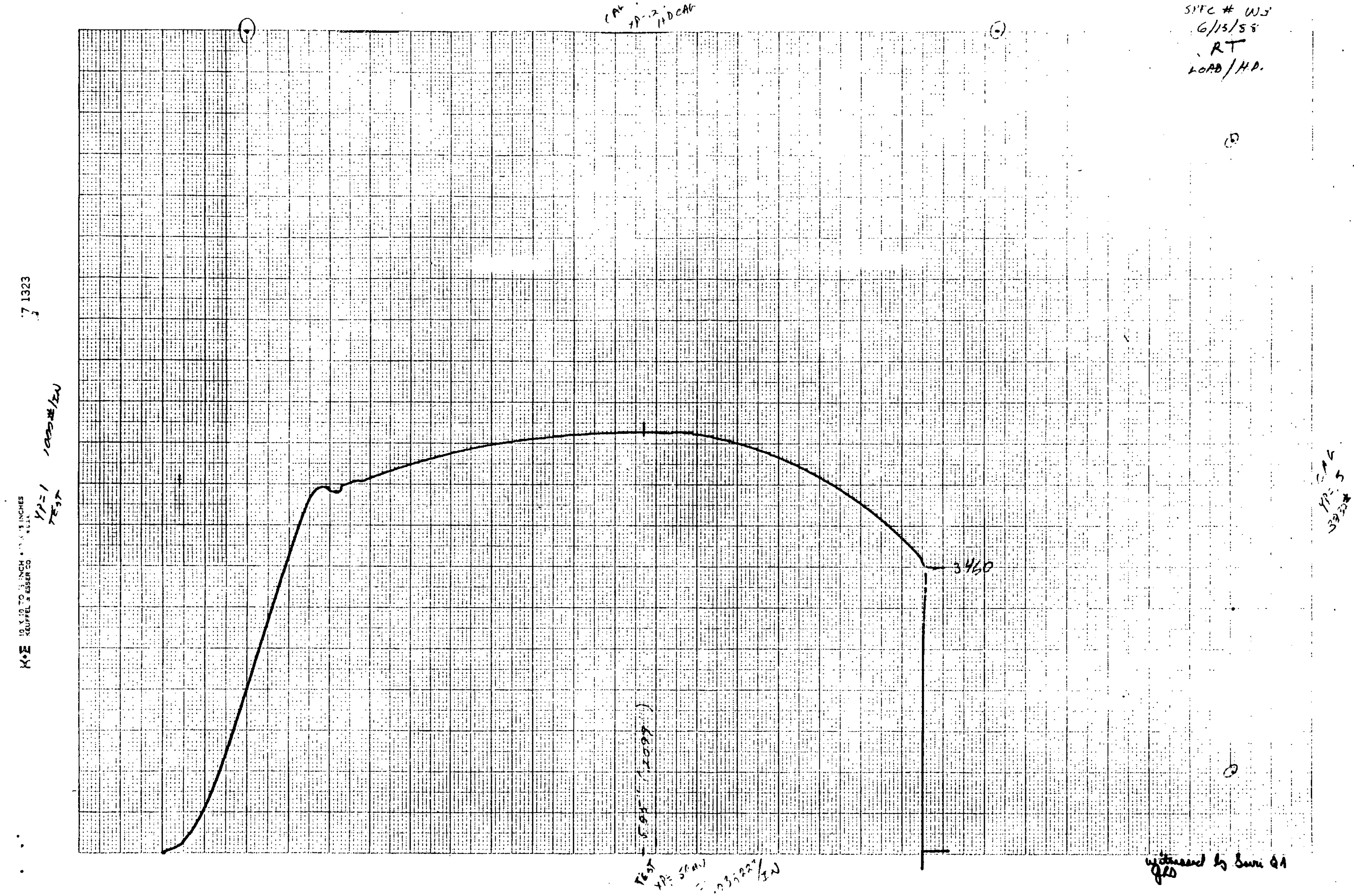
Test Performed by: T. MASDEN / R. ATIYEH

Calculations Performed by: Tom Masden (Date) 6/17/88

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



SPEC. # W3
6/15/85
RT
LOAD/STRAIN
5125#
1440
TEST
K+E 10 X 10 TO 1/2 INCH KEUFFEL & ESSER CO.


SPEC # WJ
6/13/58
RT
LOAD/H.D.
1000#/IN
YP=1
TEST
3460
TEST
K+E 10 X 10 TO 1/2 INCH
KEUFFEL & ESSER CO.

Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. W4

Test Temperature 550°F

Strain Rate .005"/IN/MIN

Project No. 17-2108

Machine Ident. 4

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

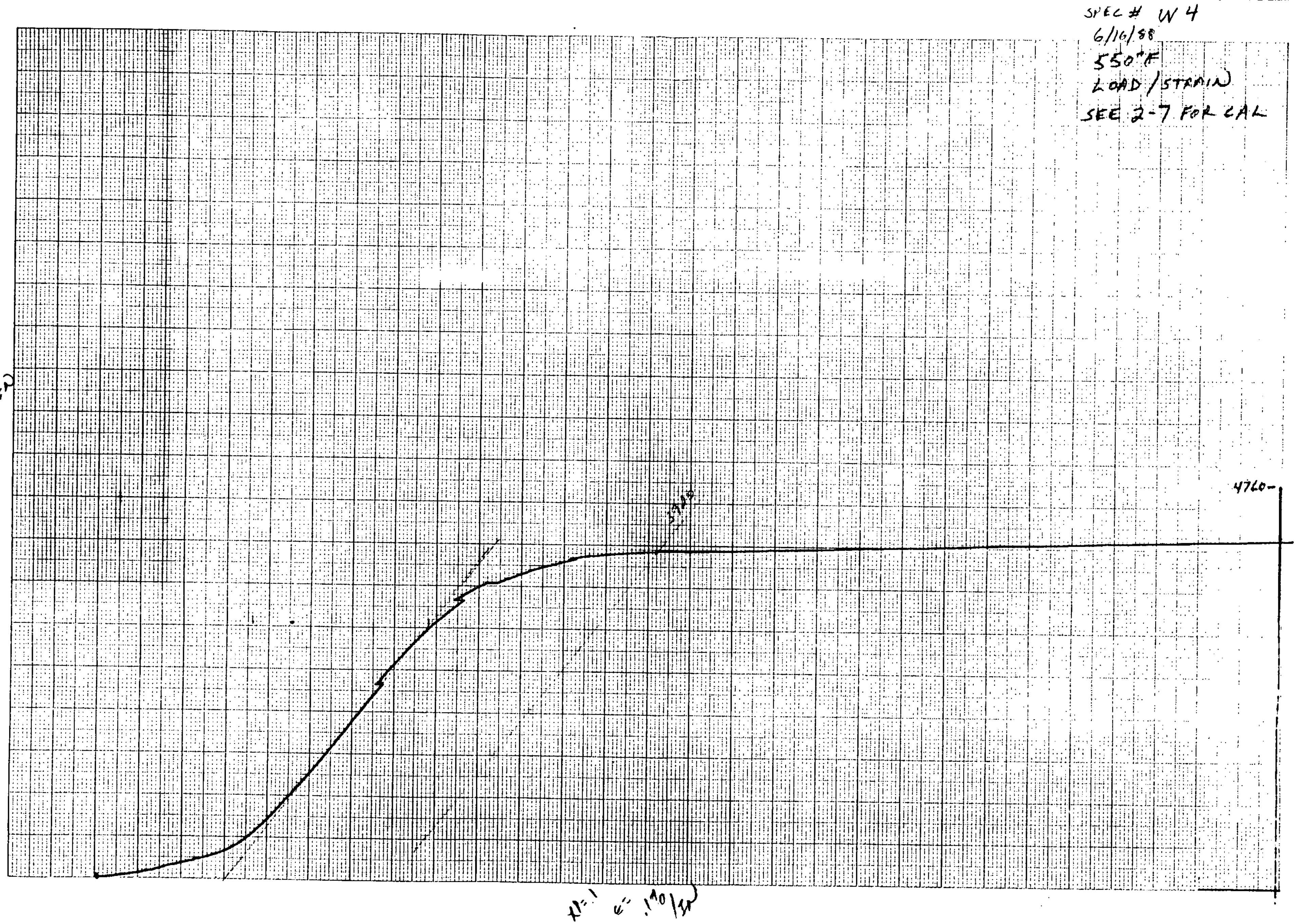
Test Performed by: T MASDEN / R. ATIYEH

Calculations Performed by: [signature] (Date) 6/19/88

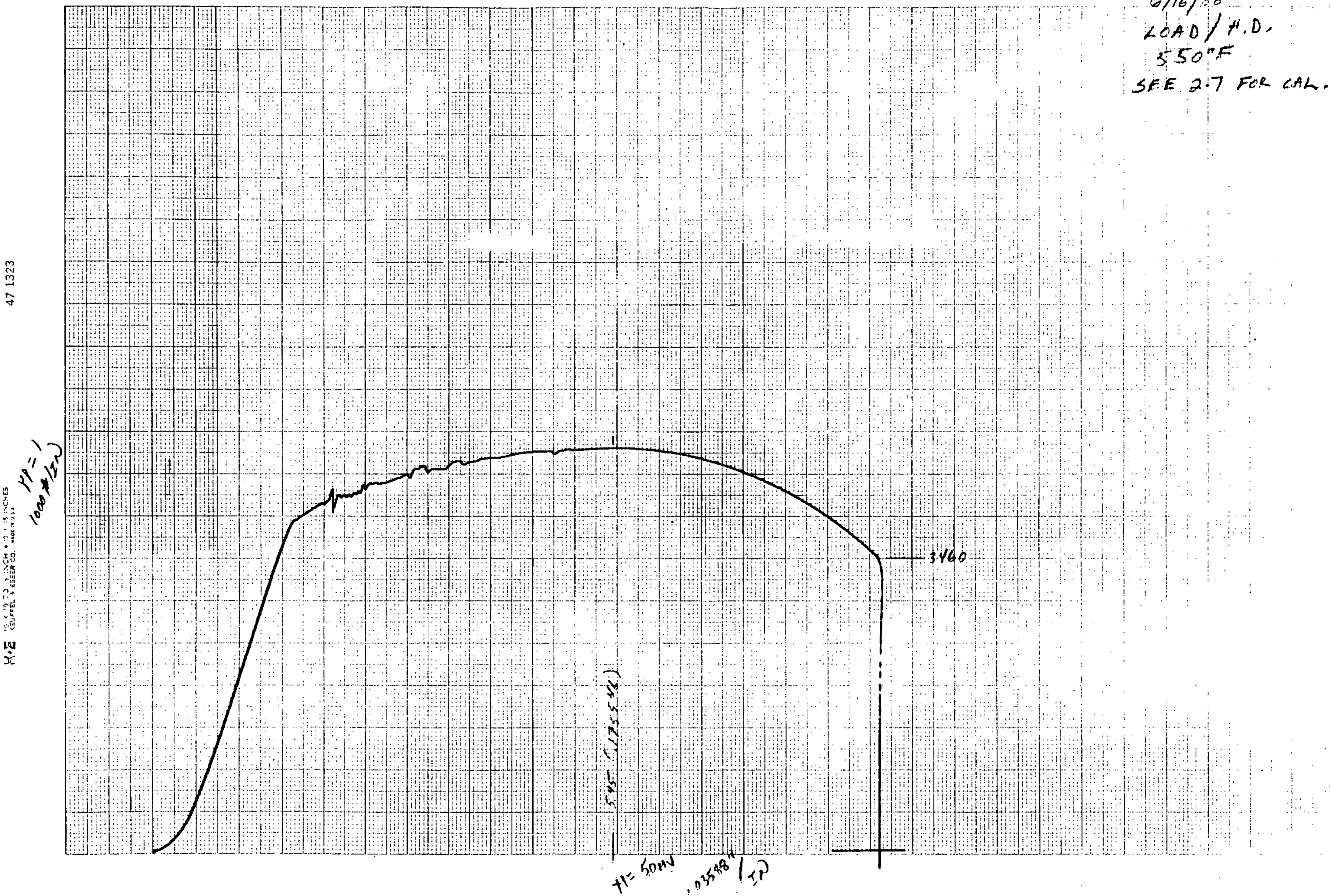
Calculations Checked by: [signature] (Date) 6/21/88

47 1323

K+E 10 X 10 TO 1/2 INCH KEUFFEL & ESSER CO. MADE IN U.S.A.



SPEC # W4
6/16/58
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

Project No. 17-2108

Test Temperature RT

Machine Ident. 4

Strain Rate .005"/IN/MIN

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 [illegible] 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

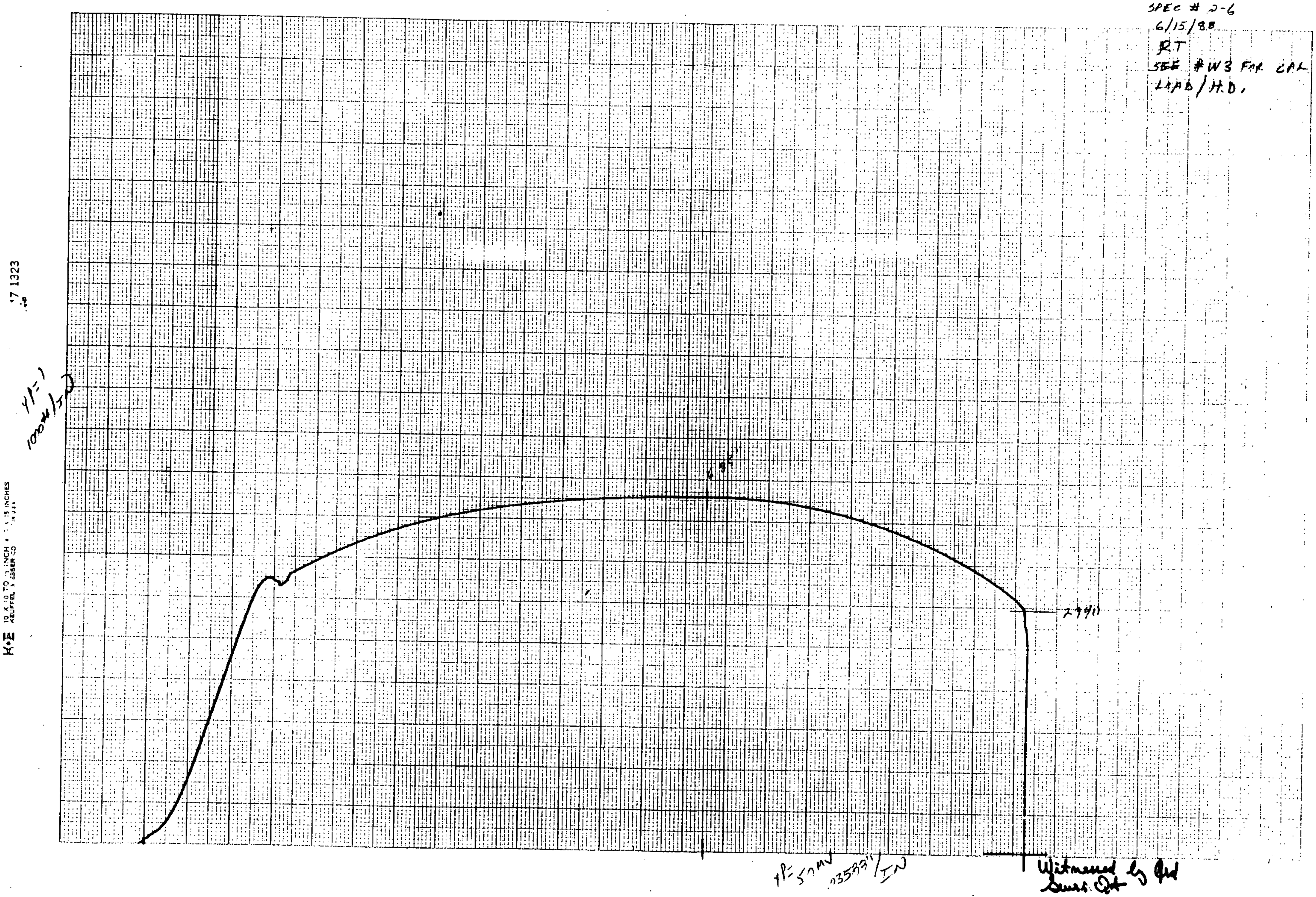
Test Performed by: T. MASDEN / R. ATIYEH

Calculations Performed by: [illegible] (Date) 6/19/88

Calculations Checked by: David [illegible] Jr. (Date) 6/21/88


SPEC # 2-6
6/15/88
RT
SEE # W.3 FOR CAL
LOAD/STRAIN
4270
2230
1000#/IN
Witnessed by
47 1323
K+E 10 X 10 TO 1/2 INCH
KEUFFEL & ESSER CO

SPEC # 2-6
6/15/88
RT
SEE #W3 FOR CAL
LOAD/H.D.



Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. 27

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. 549°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

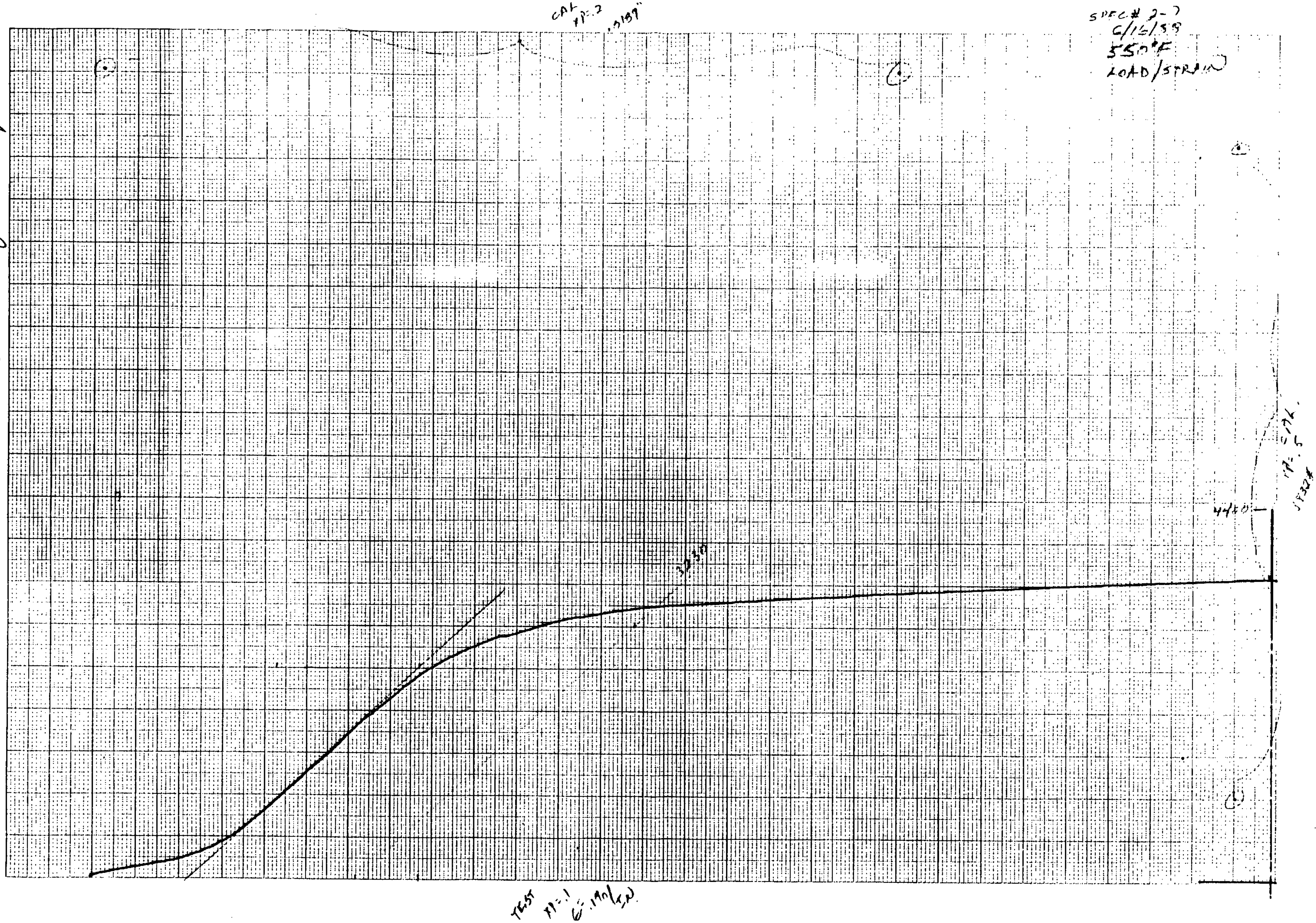
Witnessed by Suri Dut. QLD

U.T.S. = Maximum Load/Initial Area	=	90405
0.2% Y.S. = 0.2% Offset Load/Initial Area	=	66365
Frature Stress = Fracture Load/Final Area	=	250395
% 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

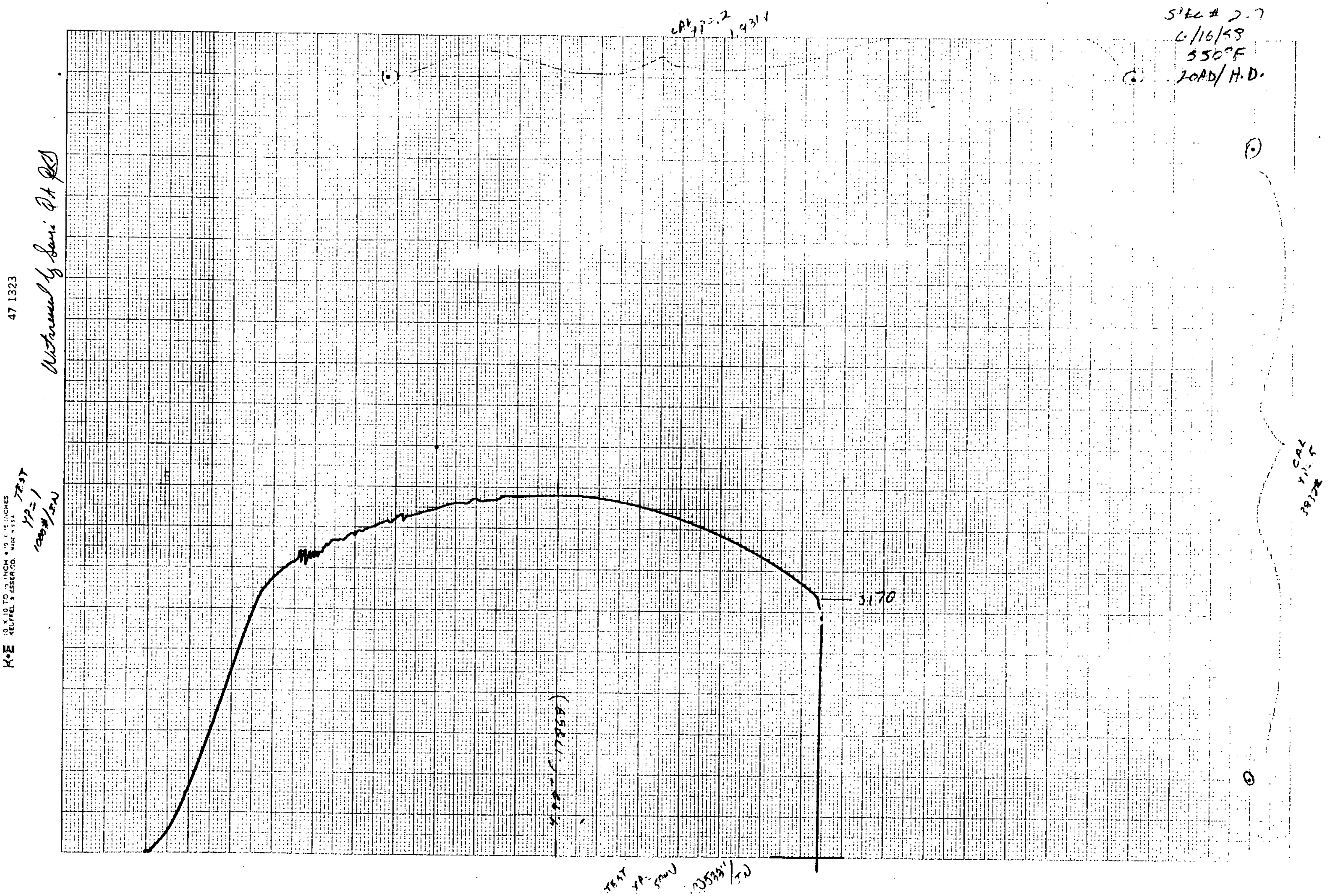
Calculations Performed by: Tim Masden (Date) 6/19/88

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



47 1323

K+E 10 X 10 TO ½ INCH
KEUFFEL & ESSER CO.



Spec # 2-7
6/16/58
550°F
LOAD/H.D.
CAL YP=.2
1.9314
47 1323
Witnessed by
K&E 10 X 10 TO 1/2 INCH
KEUFFEL & ESSER CO.
TEST YP=1
1000#/IN
3170
TEST YP=5mv
.03533"/IN
CAL YP=.5

TO: NRC
FEB-05-'01 MON 09:15 ID:NS&L TEL NO: #167 P02

FROM: CBI TECH SVCS BHAM TO: FEB 2, 1990 3:02PM #730 P.02

CBI

Chicago Bridge & Iron
Technical Services Company
One Perimeter Park South
Suite 400-S
Birmingham, Alabama 35243

Phone 205 969 9200
FACS 205 969 9205

February 2, 1990

Consolidated Edison Company of New York
Indian Point Unit 2 Nuclear Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

Attention: Melissa Driscoll
Reactor Engineer

REFERENCE: SPENT FUEL STORAGE RACKS
INDIAN POINT UNIT 2
HOLTEC P.O. 81000-302
CBI CONTRACT 881161

SUBJECT: CONTRACT DRAWING REPRODUCTION

Dear Melissa:

CBI Contract Drawings 14 and 15 may be reproduced for the purpose of review of license amendment request.

Yours very truly,

Richard L. Bentley

Richard L. Bentley
Engineering Supervisor

RLB/af

Enclosure

cc: Dr. Kris Singh, President
Holtec International
139A Gaither Drive
Mt. Laurel, NJ 08054

ATTACHMENT I

ADDITIONAL INFORMATION REGARDING
STRUCTURAL SEISMIC ASPECTS OF INDIAN POINT
INCREASE IN SPENT FUEL POOL STORAGE CAPACITY

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
JANUARY, 1990

I. SPENT FUEL POOL ANALYSIS

1. Provide sketches and/or drawings of the pool showing elevations, basemat and pool wall thicknesses, water levels, and safety related components (such as piping in the pool, and their clearances from the racks.

RESPONSE

The following sketches and drawings are provided in response to the above request:

Sketches:

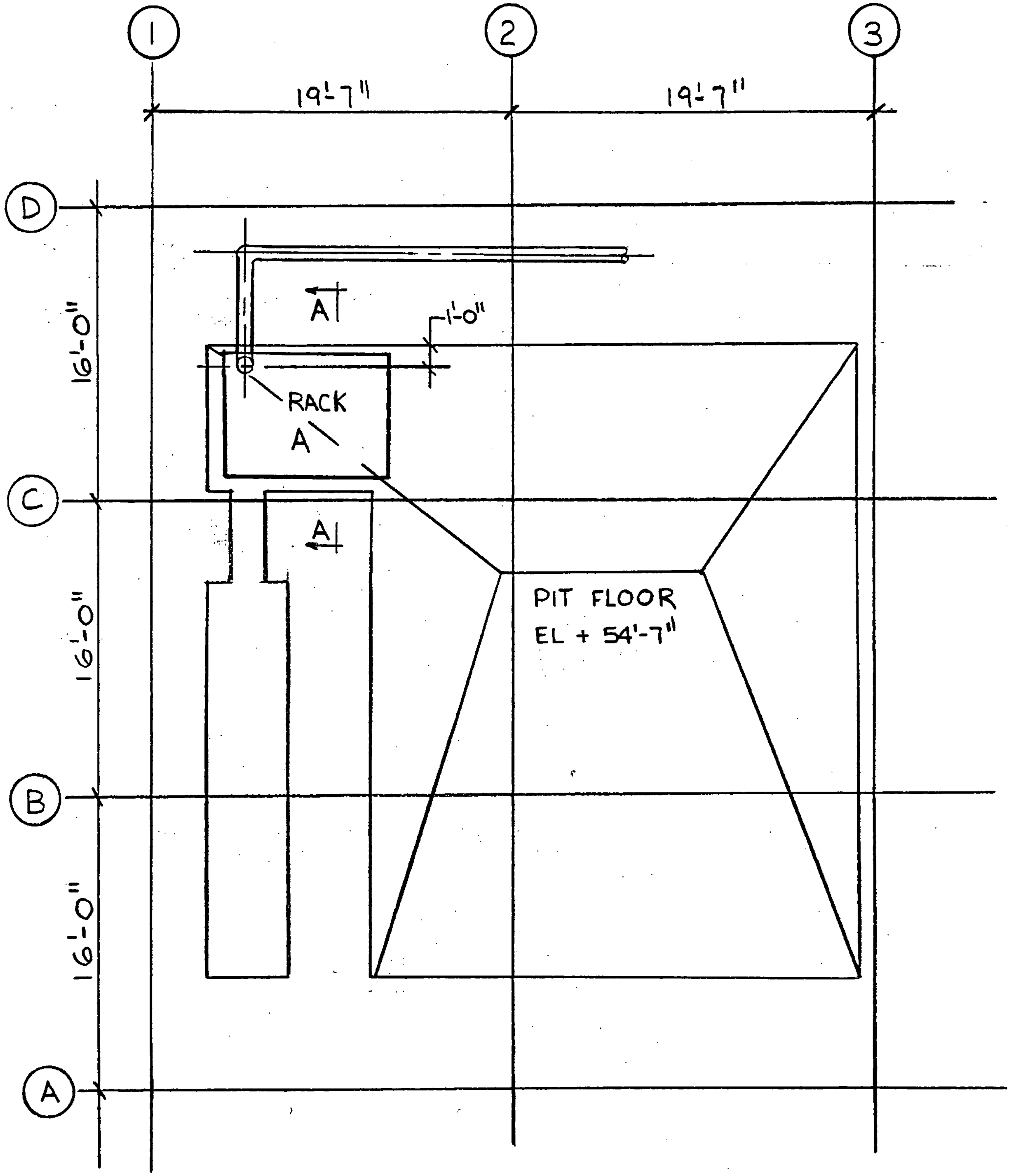
1 Spent Fuel Pool showing location of spent fuel pool cooling piping in pool.

2 Details of portion of spent fuel pool cooling piping in spent fuel pool with clearance to racks.

Drawings:

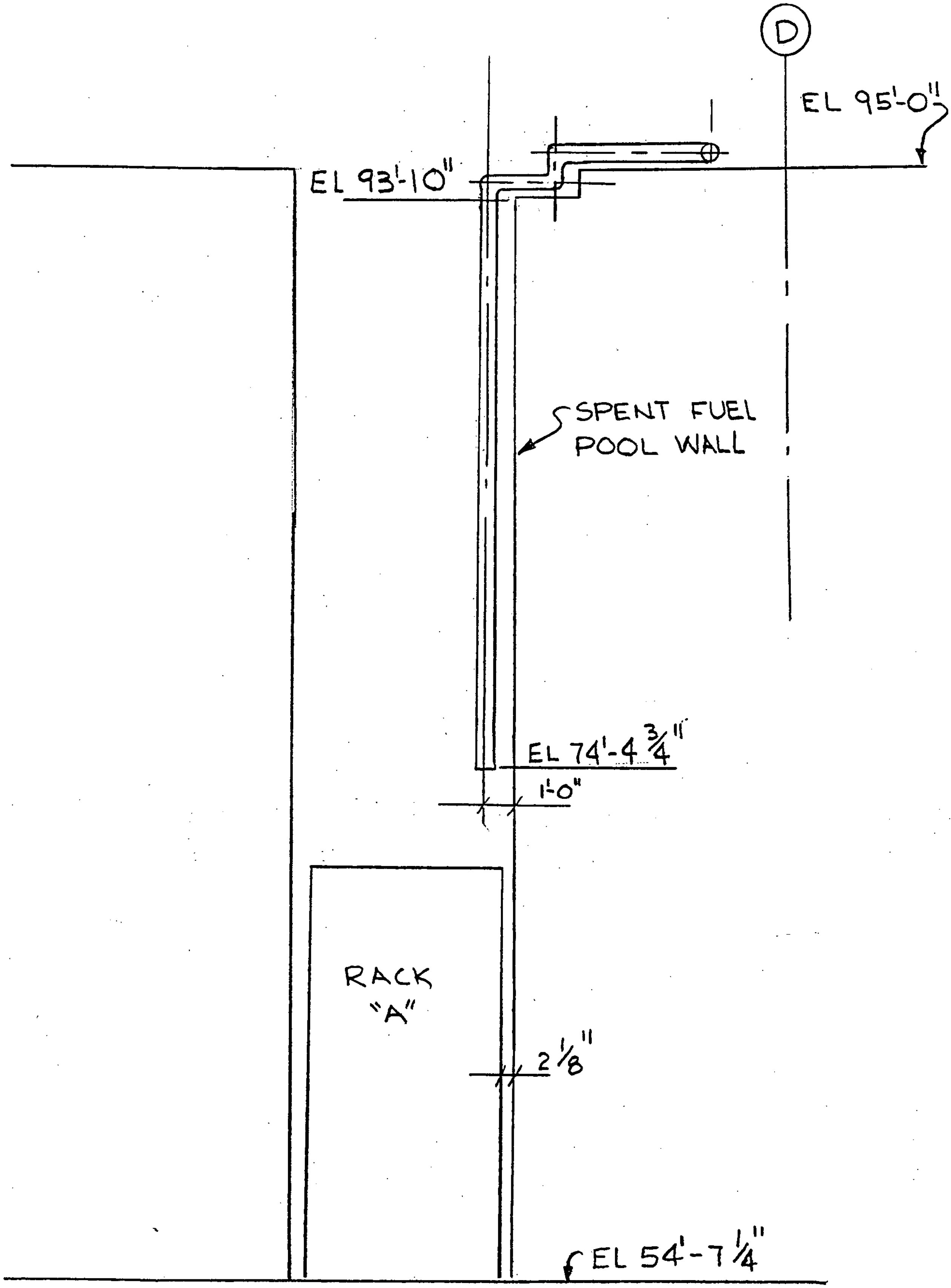
9321-F-2514	Fuel Storage Building General Arrangement Plans and Elevations.
9321-F-1196	Fuel Storage Building Concrete Details - Sheet 1.
9321-F-1197	Fuel Storage Building Concrete Details - Sheet 2.
9321-F-1198	Fuel Storage Building Concrete Details - Sheet 3.
9321-F-1199	Fuel Storage Building Concrete Details - Sheet 4.
9321-F-1200	Fuel Storage Building Concrete Details - Sheet 5.
9321-F-1301	Fuel Storage Building Tank Liner Plates - Sheet 1.
9321-F-1302	Fuel Storage Building Tank Liner Plates - Sheet 2.

The drawings listed above show the pool elevations, basemat and pool wall thicknesses. Drawing 9321-F-2514 shows the pool water level in the section of the drawing labeled 'Elevation @ Section "A-A"'. The water level given is the normal water level, 93'8", and can vary during operation by ±6". Sketch 1 provides an overview of the spent fuel pool indicating the location of the portion of the spent fuel pool cooling piping located in the pool. Sketch 2 provides the details of the clearance between this pipe and the rack below it. This section of pipe is the only safety-related equipment, except for the storage racks, in the spent fuel pool.



PIT PLAN SCALE 1/8=1'-0"

SKETCH 1



SECTION A-A SCALE 3/16" = 1'-0"

SKETCH 2

I. SPENT FUEL POOL ANALYSIS

2. Provide information on how the additional weight of high density racks (HDRs) and impacts on floor and walls under the postulated seismic events are incorporated in the design of the pool structure. Provide information related to pool structure seismic responses (including hydrodynamic loads) due to the proposed reracking, controlling load combinations and stresses at critical structural sections.

RESPONSE

The Spent Fuel Pool is designed as a Seismic Category I structure. This structure was reanalyzed, with the new racks assumed to be installed, to determine compliance with ACI-318(77), and SRP 3.8 of NUREG-0800. The details of the pool structure, applicable loadings, and summarized results are given in the following.

The IP-2 Spent Fuel Pool is a reinforced concrete structure built on a rock foundation. The pool slab is 45 feet by 42 feet in plan and three feet in thickness. Referring to Figure 1, the pool floor is at elevation 54′-7". The load bearing (external) walls are 48" thick for the bottom 16′-2" above the pool slab, and increase to 75" thickness over a 2′-5" height. The thickness of the walls remains uniform (6′-3") for the remainder of the 20′ of the top portion of the pool. The bottom 24′-5" (up to elevation 79′-0") of the pool walls and slab are below grade. Thus, from a structural standpoint, the pool slab and bottom 24′-5" of the pool walls are supported by a semi-infinite elastic continuum. The pool is filled with borated water up to the height of 39′-1". The size and location of reinforcement bars parallel to the plane of section are shown in Figure 2 for Section A-A and Figure 4 for Section B-B. It is noted that the top of the racks (which extend for approximately 178" from the pool liner) is well below the grade level.

The pool liner is 1/4" thick and is made from SA240-304 austenitic stainless material.

The foundation bedrock consists of hard limestone capable of supporting loads up to 50 tons per square foot. The foundation boring logs indicate limestone with unconfined compressive strength of 7810 psi in the vicinity of the spent fuel pool.

The structural analysis is carried out using a finite element model of representative sections of the pool. The floor and walls are modeled using shell elements, and the foundation modeled using 3-D brick elements. Two sections were analyzed, denoted as Section A-A and Section B-B, respectively, in Figure 1. Figure 3 shows a 2-D slice of the fuel pit for Section A-A and the surrounding rock foundation. Section A-A is a sectional view parallel to the widest and weakest section. This section is assumed fully populated with the heaviest racks for structural analysis purposes.

The effective depth of rock substructure is assumed as 10 feet and the centerline of the opposite walls (N/S) is assumed to be a 40′ span (greater than the actual inside span of the pool at the location). The weight of concrete plus reinforcement is assumed to be such that the combined weight density is 180 lb./cu.ft. The following properties are used in the analysis:

Reinforcement strength	f_y	= 60000 psi
Concrete strength	f_c	= 3000 psi
Young's Modulus of Rock	E	= 8400000 psi

Section B-B is chosen for analysis because it contains an internal pool wall (left wall in Figure 1) which does not have the lateral foundation support. Figure 4 shows the 2-D cut away section.

In addition to the mechanical loadings, the pool structure was also subjected to the temperature induced loadings. For this purpose, the thermal boundary conditions were conservatively specified as 180°F pool water temperature and 0°F outside ambient. The thermal moments computed by the finite element analyses were combined with those due to mechanical loads as described below.

Structural Loadings on the Pool Slab and Walls

The following loadings are considered:

(i) Dead weight of slab and walls (D_1)
(ii) Dead weight of rack modules (D_2)
(iii) Dead weight of stored fuel assemblies (D_3)
(iv) Dead weight of 39′-1" water in the pool (D_4)
(v) Hydrostatic pressure on pool walls (D_5)
(vi) Hydrodynamic pressure on the pool walls during seismic event (D_6)
(vii) Impact loads due to response of racks during seismic event (D_7)
(viii) Thermal Moment (temperature gradient loading)(D_8)

Table 1 gives information concerning these loads.

As noted previously, in order to obtain a conservative assessment of the stresses in the pool structure, the most controlling sections in the pool were analyzed using a Finite Element Model. The Finite Element Model consists of 270 elements and 544 nodes. Figure 4 shows the concrete sections modeled by shell elements. The contribution of the surrounding rack continuum is modeled using three dimensional solid elements. The following load combinations are per SRP 3.8.4.

Consider:

$1.4D + 1.9E$

$D + E'$

$0.75\ [1.4D + 1.9E + 1.7T_o]$

where, referring to Table 1,

$D = D_1 + D_2 + D_3 + D_4$ (on pool slab)
$E' = D_6$ on pool walls $+ D_7$ on slab

For added conservatism, we combine the two governing loading combinations into a "bounding loading condition" as

$$1.4D + 1.9E' + 1.275T_o$$

The section moments and shears at critical locations are provided in Table 2 and 3 for Section A-A and B-B, respectively, and compared to their respective Design Strengths.

In addition to the conservative load combination, several other assumptions in the pool structural analysis produce inherent margins of safety in the computed values. The key assumptions are synopsized below:

a) A lowered bound value of foundation modulus of the equivalent elastic foundation representing the subgrade surrounding the outside pool walls is utilized in the analysis.

b) The lateral support provided by the "plate" effect of the wall is incorporated in the 2-D model in a conservative manner.

c) In the temperature profile analysis of the pool walls, and the elastic continuum surrounding it, a lower bound value of the thermal conductivity is used so as to produce a most adverse temperature gradient.

It is noted from the results presented in Tables 2 and 3 that despite these conservative assumptions, there are large margins between the factored loads and corresponding design strengths.

Table 1

GROSS LOADINGS

		Value in KIPS unless otherwise stated
(i)	Reinforced concrete and water dead weight ($D_1 + D_4$)	5.39 KSF
(ii)	Dead weight of rack modules (empty) (D_2) (per Table 2.2 of Licensing Report)	217.1
(iii)	Dead weight of 137 stored spent fuel assemblies (1453 lb. each rounded off to 1500 lbs) (D_3)	2061.
(iv)	Maximum hydrostatic pressure of water (triangular profile from top to bottom)	16.94. psi
(v)	Hydrodynamic pressure on walls due to seismic motion of water in pool (D_6)	7.8 psi (on Section A-A) 5.63 psi (on Section B-B)
(vi)	Maximum hydrodynamic† pressure on pool walls (constant for the bottom 178" height of the walls) due to gaps between rack and wall (D_6)	2 psi
(vii)	Pool slab impact loading due to SSE (per spindle) (D_7)	.52 x dead load per spindle
(viii)	Thermal gradient loading, D_8	As defined in the preceding test

† Obtained from DYNARACK simulations, assuming 1% damping for the SSE condition.

Table 2

CRITICAL REGIONS OF SECTION A-A
(Results given in absolute value)

Location	Calculated Factored Moment (KIP in./in.)	Limit Factored Moment (KIP in./in.)
Pool Wall (6′-3″ section)	164.9	255.8
Pool Wall transition section	41.0	207.8
Pool Wall (4′ section)	24.8	159.8
Pool Slab (center section)	1.9	117.1
Pool Slab (outer section)	3.8	285.1
Foundation pressure under slab	308.9 psi	694. psi
Foundation pressure on North Wall	72.2 psi	694. psi

Table 3

CRITICAL REGIONS OF SECTION B-B
(Results given in absolute value)

Location	Calculated Factored Moment (KIP in./in.) or Pressure (psi)	Limit Factored Moment (KIP in./in.) or Pressure (psi)
Pool Wall Top Section	92.0	316.9
Pool Wall Bottom Section	119.5	386.1
Pool Slab Center Section	1.9	117.1
Pool Slab Adjacent to Pedestals	6.4	117.1
Foundation Pressure Under Slab	302.2 psi	694

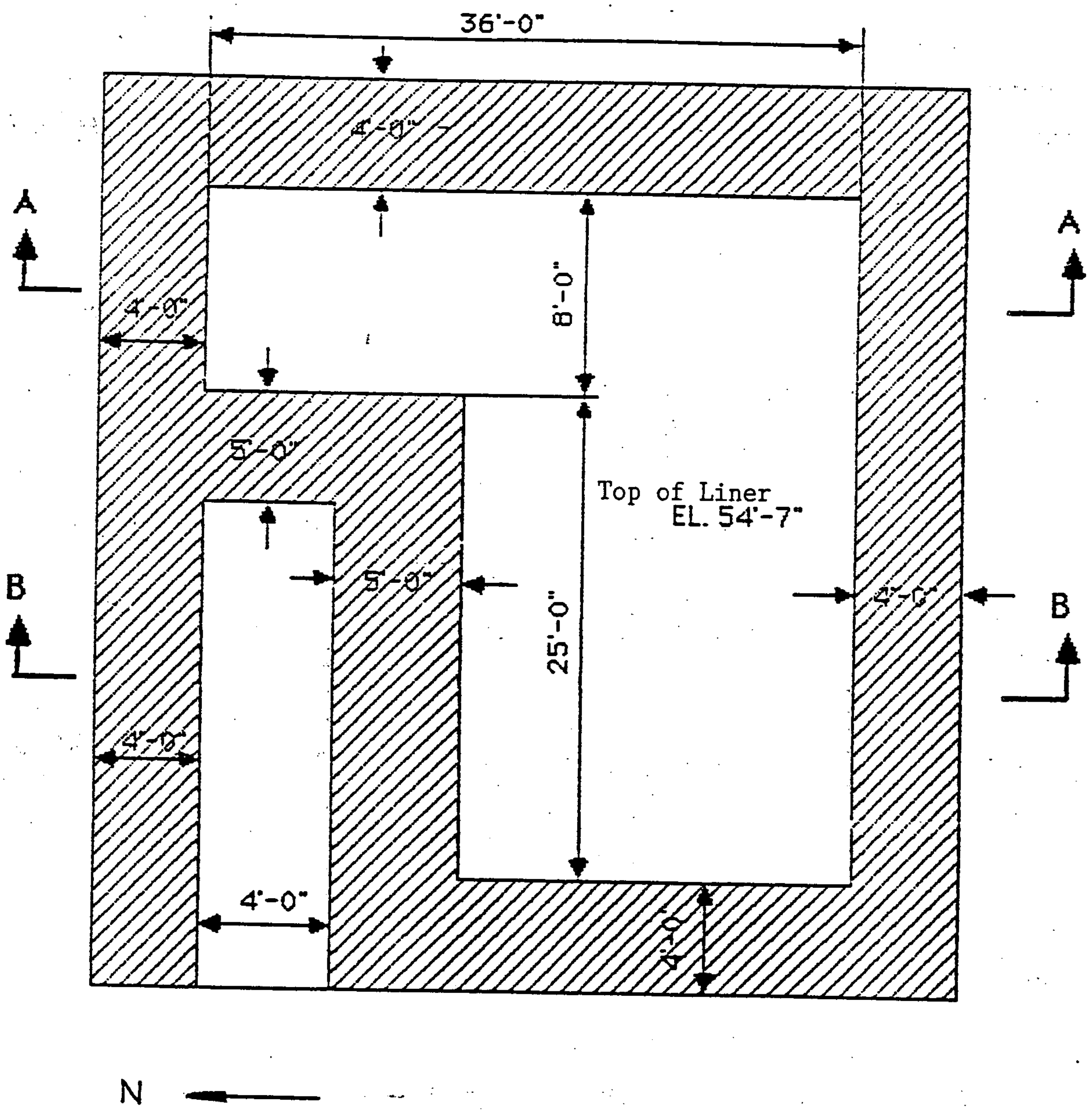


FIGURE 1 PLAN VIEW OF IP-2 POOL

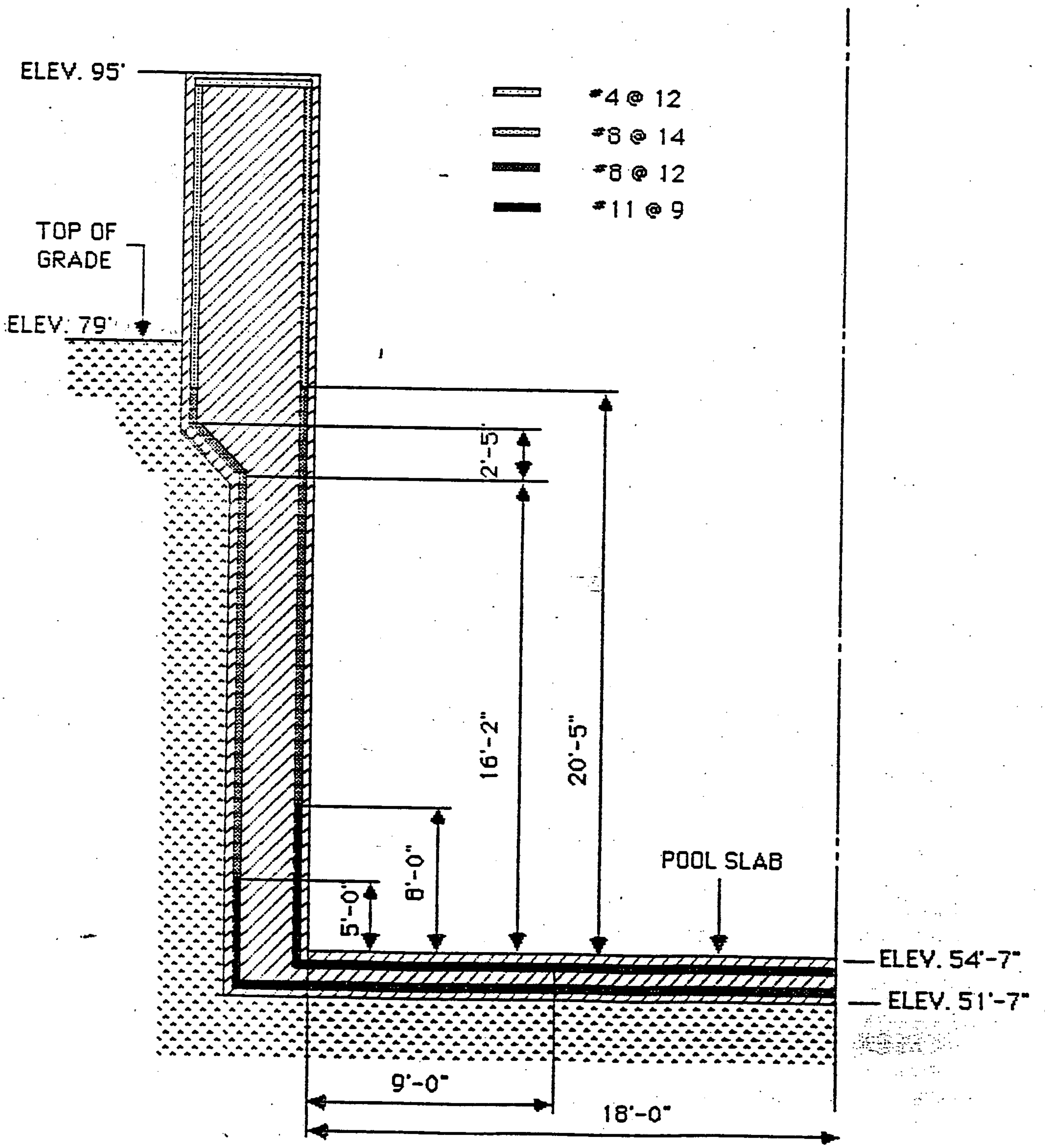


FIGURE 2 HALF SECTION (VIEW A-A ON FIGURE 1)
SHOWING REINFORCEMENT

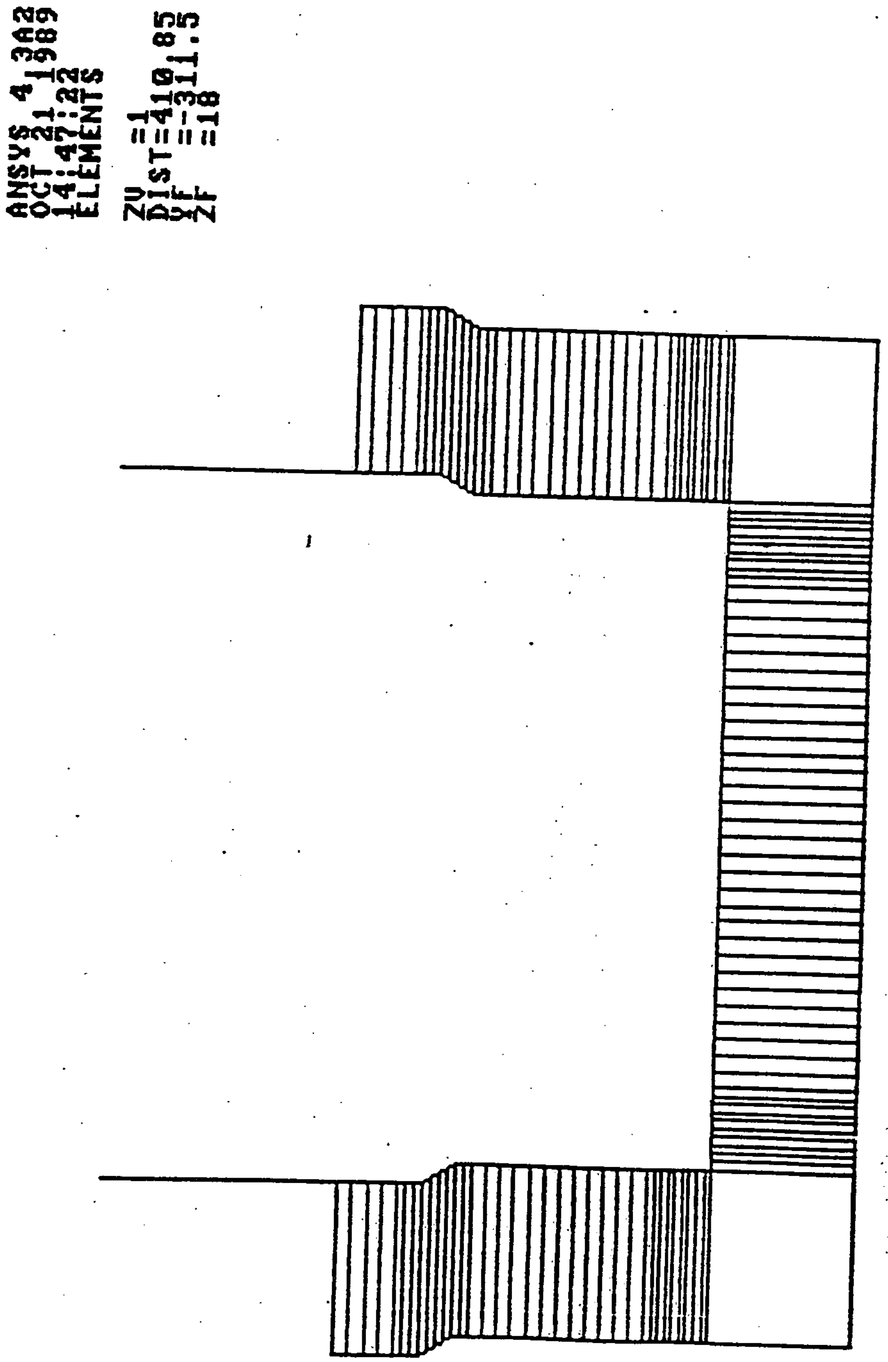


FIGURE 3 2D FINITE ELEMENT MODEL FOR IP-2 SLAB
(Section A-A)

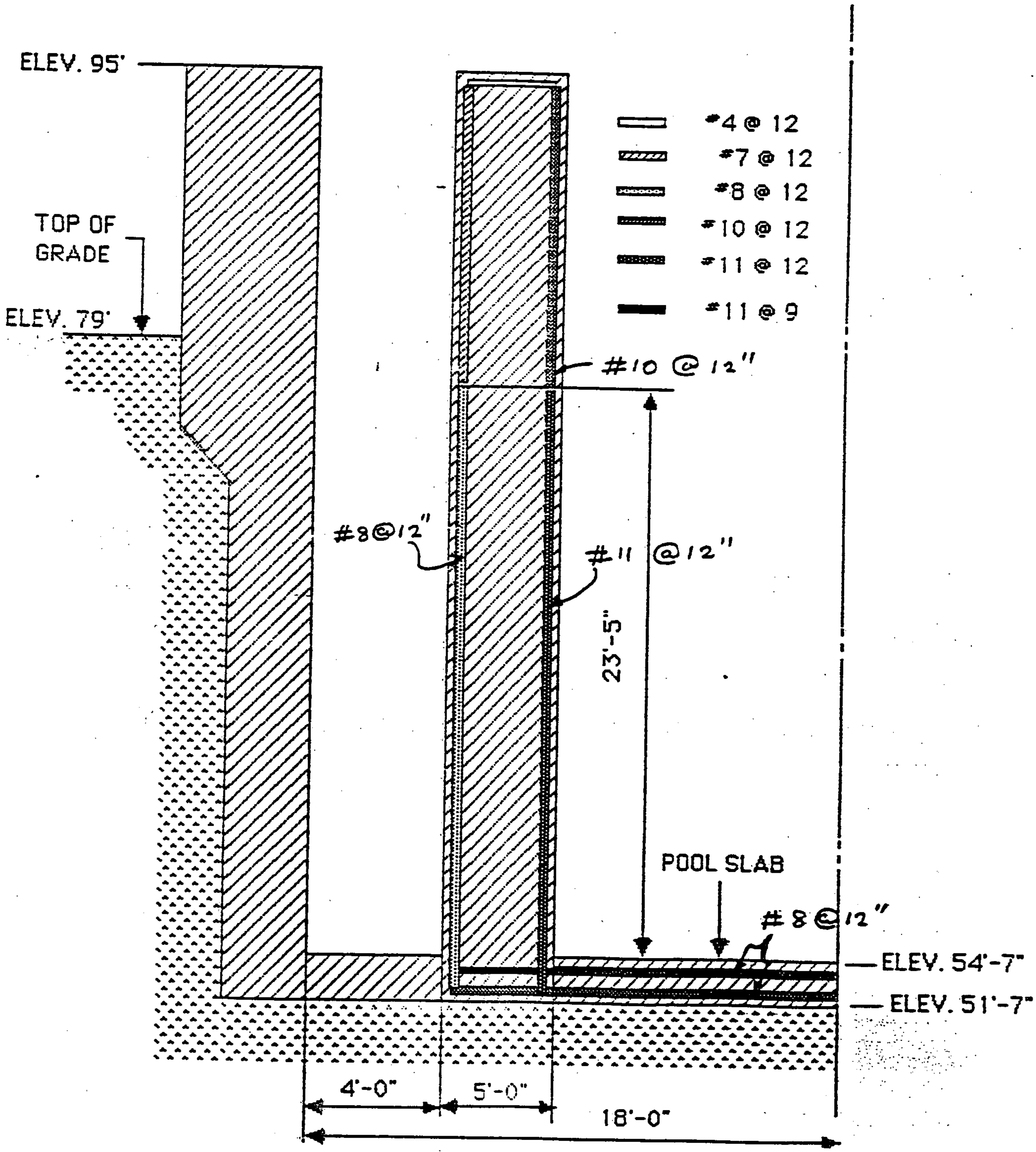


FIGURE 4 HALF SECTION (B-B on FIGURE 1)
SHOWING REINFORCEMENT

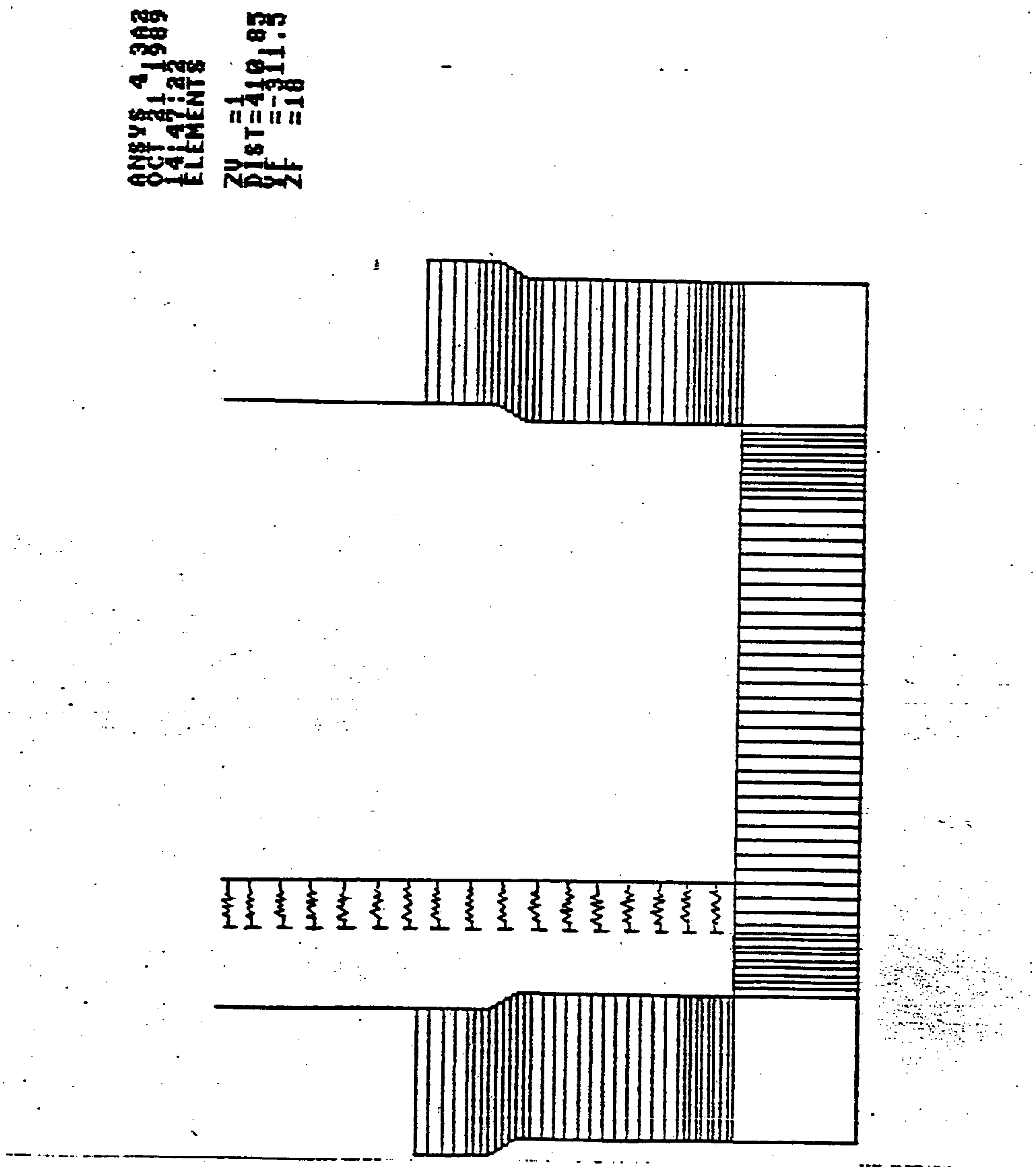


FIGURE 5 2D FINITE ELEMENT MODEL FOR IP-2 SLAB (SECTION B-B)

I. SPENT FUEL POOL ANALYSIS

3. Provide information on the locations of the rack pedestals with respect to the leak-chases and other embedments.

RESPONSE

In order to provide a complete description of the rack pedestals with respect to embedments in the spent fuel pool, Drawing 531 entitled "Support ID & Bearing Pads" is being provided. This drawing shows the support pads that the rack pedestals will be placed on. The various embedments in the pool floor are shown as well as the pool liner weld seams (dashed lines running N-S and E-W). It should be noted that the Indian Point 2 spent fuel pool was built and licensed without a leak chase, since the pool structure rests on bedrock.

II. SEISMIC INPUT MOTION

1. The plant FSAR (Table 1.11-1) requires that 1% damping be used for steel welded structures such as the rack. Provide justification for using 2% (LAR Section 6.2.4) damping for the rack analyses.

RESPONSE

All governing loading cases reported in Section 6 of the Licensing Report have been re-run with 1% structural damping. The responses, as expected, have increased slightly. The results are presented in Tables II.1 and II.2. There is no effect on the rack structural integrity conclusions presented in the licensing submittal.

Table II.1

STRESS FACTORS AND RACK TO FUEL IMPACT LOAD (1% DAMPING)

Run I.D.	Remarks	Rack/Fuel Impact Load (lb.) (Per Cell)		STRESS FACTORS R_1	R_2	R_3
D0b	Rack D Cof = .8, SSE Filled with Regular Fuel	252.9	* **	.013 ---- .183	.014 ---- .031	.152 ---- .122
D0d	Rack D Cof = .2, SSE Full load Regular Fuel	252.7		.013 ---- .182	.014 ---- .030	.152 ---- .124
G2a	Rack G2 (11x12) Cof = .8, SSE Full load Regular Fuel	330.8		.013 ---- .147	.014 ---- .015	.097 ---- .042
B02	Rack B (9x12) Cof = .8, SSE Full load Regular Fuel	328.0	* **	.008 ---- .167	.010 ---- .034	.069 ---- .097
B03	Rack B Cof = .2, SSE Full load Regular Fuel	328.0		.008 ---- .166	.010 ---- .032	.068 ---- .106

* Upper values are for rack cell just above baseplate.

** Lower values are for support foot cross section (upper part).
See last page of this table for stress factors R_4-R_7.

Table II.1
(continued)

Run I.D.	Stress Factors R_4	R_5	R_6	R_7
DOb	$\frac{.095}{.070}$	$\frac{.181}{.292}$	$\frac{.212}{.312}$	$\frac{.024}{.051}$
DOd	$\frac{.095}{.070}$	$\frac{.180}{.290}$	$\frac{.210}{.309}$	$\frac{.023}{.050}$
G2a	$\frac{.083}{.034}$	$\frac{.139}{.183}$	$\frac{.162}{.190}$	$\frac{.016}{.018}$
BO2	$\frac{.073}{.058}$	$\frac{.097}{.246}$	$\frac{.114}{.259}$	$\frac{.013}{.040}$
BO3	$\frac{.074}{.066}$	$\frac{.095}{.247}$	$\frac{.111}{.263}$	$\frac{.012}{.040}$

Table II.2

RACK DISPLACEMENTS AND SUPPORT LOADS (1% DAMPING)
(all loads are in lbs.)

Run I.D.	Floor Load (sum of all support feet)	Maximum Support Load		Vertical Load*	Shear Load**	DX*** (in.)	DY (in.)
D0b	2.465×10^5	1	115000.	114960.	13973.	.1801	.1854
		2	105900.	112052.	23368.	.0007	.0009
		3	100700.				
		4	113900.				
D0d	2.465×10^5	1	114400.	114398.	17408.	.1803	.1853
		2	105900.	113165.	22633.	.0011	.0038
		3	100200.				
		4	113400.				
G2a	2.292×10^5	1	87790.	92343.	7890.	.1330	.1285
		2	91740.	42684.	8241.	.0006	.0006
		3	90720.				
		4	92410.				
B02	1.953×10^5	1	105000.	104955.	16371.	.1767	.0884
		2	96960.	49529.	17541.	.0013	.0009
		3	81130.				
		4	98520.				
B03	1.953×10^5	1	104700.	104667.	20935.	.1761	.0884
		2	93860.	104667.	20935.	.0016	.0026
		3	80660.				
		4	98130.				

* The first line in any set of data is near the maximum vertical load and the second line reported is the vertical load when the net horizontal shear at the liner is maximum.

** The first line is the net horizontal liner shear when the vertical load is near the maximum; the second line is the maximum value of the net horizontal shear on any single support foot.

*** The first line reports results at the top of the rack; the second line reports results at the baseplate. The times at which these maximums occur may be different.

II. SEISMIC INPUT MOTION

2. Provide information on how the statistical independence (LAR Section 6.1) of the three components of earthquake was established.

RESPONSE

The statistical independence of the three components of synthetic time histories was established by computing the normalized cross covariance of each pair of time histories (a total of three pairs). An effective technique to obtain the desired level of non-correlation between the time histories involves changing the random seed number, and the enveloping function for the time history profile. The time history generation techniques permit the use of different envelope functions. Trapezoidal, exponential decay, and sinusoidal envelopes are some of the commonly used bounding functions. It is found that using a different genre bounding function for two time histories results in a lower level of covariance between them. This statistical correlation function was found to be less than 0.1 in all the cases.

The synthetic time histories in the N-S, E-W and vertical directions may be labeled as $a_i(\tau)$; i = 1,2,3 respectively (τ is time coordinate). If γ_{ij} represents the normalized statistical correlation function between a_i and a_j, then the computed values of γ_{ij} are as follows:

γ_{12} = .02933
γ_{13} = .02155
γ_{23} = .01550

III. ANALYSIS AND DESIGN OF HDRs

A. RACK ANALYSIS

1. Provide justification for the use of five rattling masses (to represent fuel assemblies) instead of rattling masses at every grid locations. How is the impact on fuel grid computed? (LAR Section 6.2.1a)

RESPONSE

The grid straps are only on the order of a few mils thick, and therefore cannot be postulated as definitively designated impact locations. The low flexural stiffness of the fuel assembly and fluid force contribution of water further ensure that the assembly will undergo various curved contours, and the rattling impacts will occur at non-grid strap locations. Our model, therefore, discretizes the assembly into five discrete masses, which are equispaced along the assembly length. This is in contrast to seven grid strap locations. Therefore, the number of lumped masses used in our analysis is less than the number of grid strap locations. Consequently, each lumped mass in our model is bigger than the discretized mass if the lumped masses were provided at each grid strap locations. A larger lumped mass implies a greater impact load due to rattling of the mass in the storage cell. Consequently, the impact force at each of the five mass locations in our model bounds the value that one would obtain from the model employing a lumped mass at each grid strap location. However, to be conservative, the maximum impact load obtained from the dynamic analysis is assumed to be applicable to the grid strap, as well. In summary, the impact force computed at a mass node point in our analysis would exceed that calculated for each grid strap location. Therefore, our analysis is conservative. The maximum values of the fuel assembly-to-cell wall impact load are given in Table II.1 (see response to Question II.1), and impact capacities of the fuel assembly are provided in the response to Question III.A.2.

III. ANALYSIS AND DESIGN OF HDRs

A. RACK ANALYSIS

2. Provide calculations showing how the impact capacity (LAR Section 6.9.1) of cell-walls are estimated. Are the concurrent longitudinal stresses considered in combination with the stresses due to impact? What is the impact capacity of fuel assemblies?

RESPONSE

The maximum fuel assembly-cell wall impact loads are calculated by DYNARACK and compared with the limit capacity of the section. Since these impact loads are localized, the only criteria is that collapse of the section does not occur. A beam section having length equal to the unsupported cell width and subject to two concentrated loads applied where the corners of the assembly would impact is analyzed for the limit state. The thickness of the beam section is .075". The actual impact load is compared to the limit load (with a safety factor of 2 built into the limit calculation). Figure 6 shows the configuration used for the impact load calculation.

The worst impact load on a cell is obtained from the DYNARACK computer code simulations as 424 lbs. Limit analysis applied to the configuration of Figure 6 yields

$$Q_L = \sigma_y \frac{L}{C} t^2 \times \frac{1}{SF} \qquad (1)$$

where σ_y = 25000 psi, L = 21.125" (1/8 of rack height)

t = .075" (cell wall thickness)

If we know the inside cell dimension (8.75") and the outside dimension of the impacting assembly (taken as 8.3"), then for calculation purposes

$$c = \frac{8.75 - 8.3}{2} = .225"$$

Assuming a factor of safety SF = 2 on the bending limit load yields

Q_L = 6602 lbs. per cell

Assuming a failure in shear of the cell wall over a length L, and a yield stress in shear equal to $\sigma_y/2$, the corresponding limit load for pure shear failure of the cell wall (with a safety factor of 2.0) is

$$Q_s = \sigma_y \frac{t}{2} (a + L) = 2.759 \times 10^4 \text{ lbs.} \quad (a = 8.3", \; L = 21.125")$$

It is noted that the actual maximum impact load is a small fraction of the cell capacity.

Concurrent longitudinal stresses are not considered in combination with impact load since these longitudinal primary stresses decrease with distance above the baseplate and are small in the region where maximum impacts occur.

The impact capacity of the fuel assemblies is approximately 5000 lbs. at each grid location, and an order of magnitude greater at other locations.

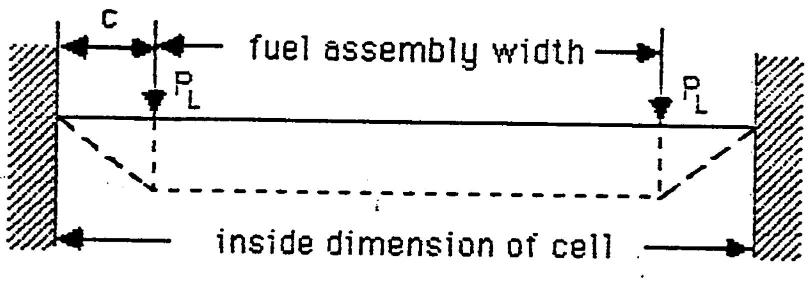


FIGURE 6 CELL-WALL IMPACT CAPACITY
(UNIT DEPTH)

III. ANALYSIS AND DESIGN OF HDRs

A. RACK ANALYSIS

3. It is not clear (LAR Section 6.2.1b) whether the entire fuel mass is modeled to vibrate in phase under the seismic event or a portion of it. If it is the later, provide justification for such assumption.

RESPONSE

The entire fuel mass is assumed to vibrate in phase under the seismic event.

III. ANALYSIS AND DESIGN OF HDRs

A. RACK ANALYSIS

4. With respect to the cross-coupling effects (LAR Section 6.2.1m), provide the following information:

(a) What is the nominal gap-multiplier for IP-2?

(b) How much is the cross-coupling consideration contributed to the resistance to the rack movement under the SSE?

RESPONSE

(a) Nominal gaps of 50% of water-rack spacing and 100% of rack-wall spacing are used. Each rack is assumed to move out of phase with any adjacent rack so as to maximize impact potential. Hydrodynamic flow around each rack is assumed to occur from the alternate squeezing and opening of channels transverse to local seismic wall motion which forces the fluid along the sides of the rack to the opposing channel. There is no "nominal gap multiplier" in single rack 3-D analysis. This term is meaningful only in the context of a 2-D multi-rack analysis. Paragraph 6.2.1 (m) of our licensing report is intended to explain how the physical effect of fluid coupling is mathematically simulated in the context of fuel rack movements.

(b) The DYNARACK output does not permit separation of the effects of different components of the hydrodynamic effect. Therefore, we cannot quantify the "cross coupling component".

III. B. RACK DESIGN

1. Provide rack drawings (or sketches) showing the details of inter-box welding and separation elements for Region I and Region II racks.

RESPONSE

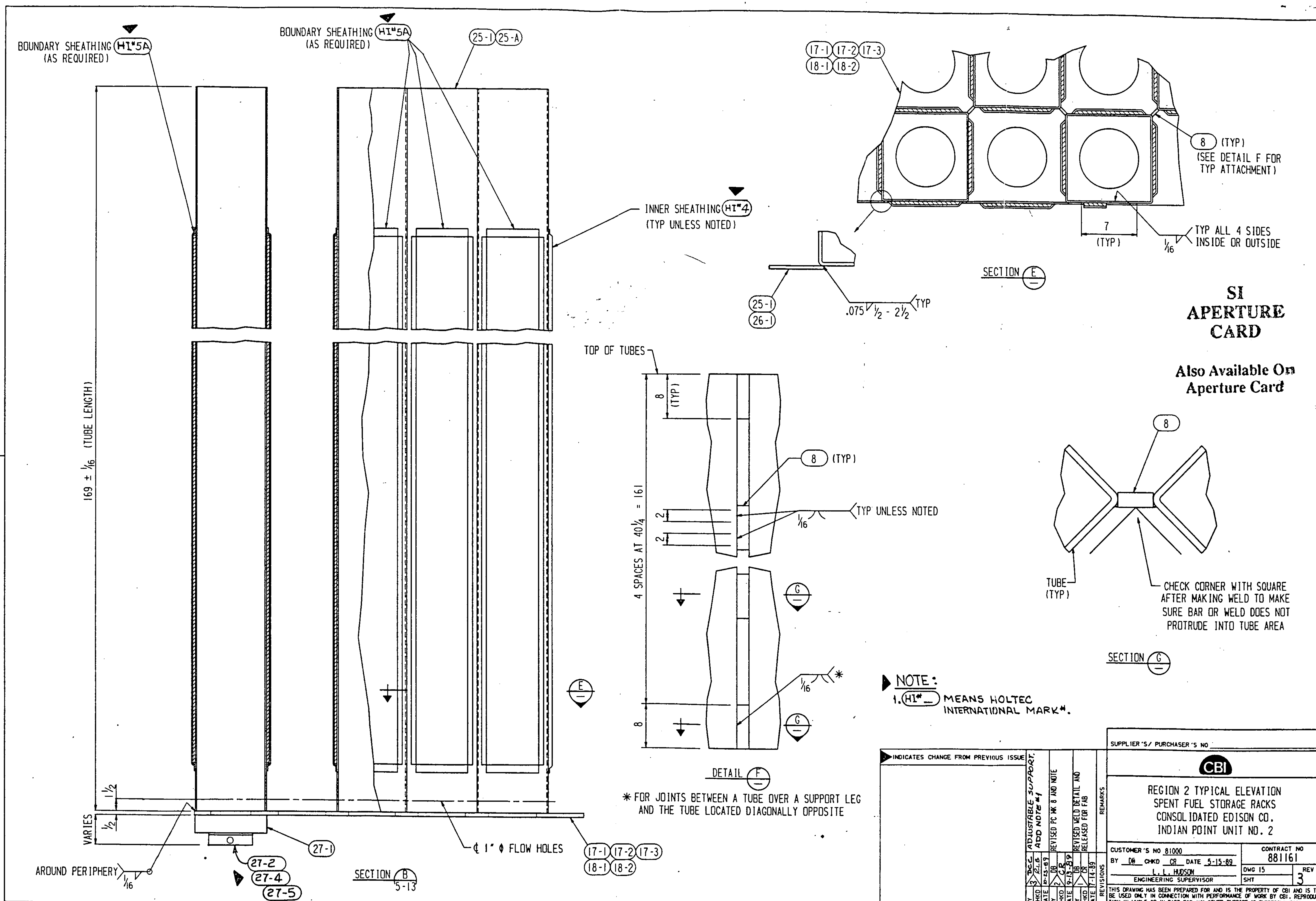
The following drawings and information are provided in response to the above request:

Drawings

14	Region 1 Typical Elevation Spent Fuel Storage Racks
15	Region 2 Typical Elevation Spent Fuel Storage Racks
1-A 1-C 1-D	Region 1 Rack A Spread Sheet Detailing QA Check List Requirements
1-A 1-D	Region 2 Rack D Spread Sheet Detailing QA Check List Requirements

Information

Region 1 Shop Check Lists
Region 2 Shop Check Lists
Use of the Shop Check List System Procedure



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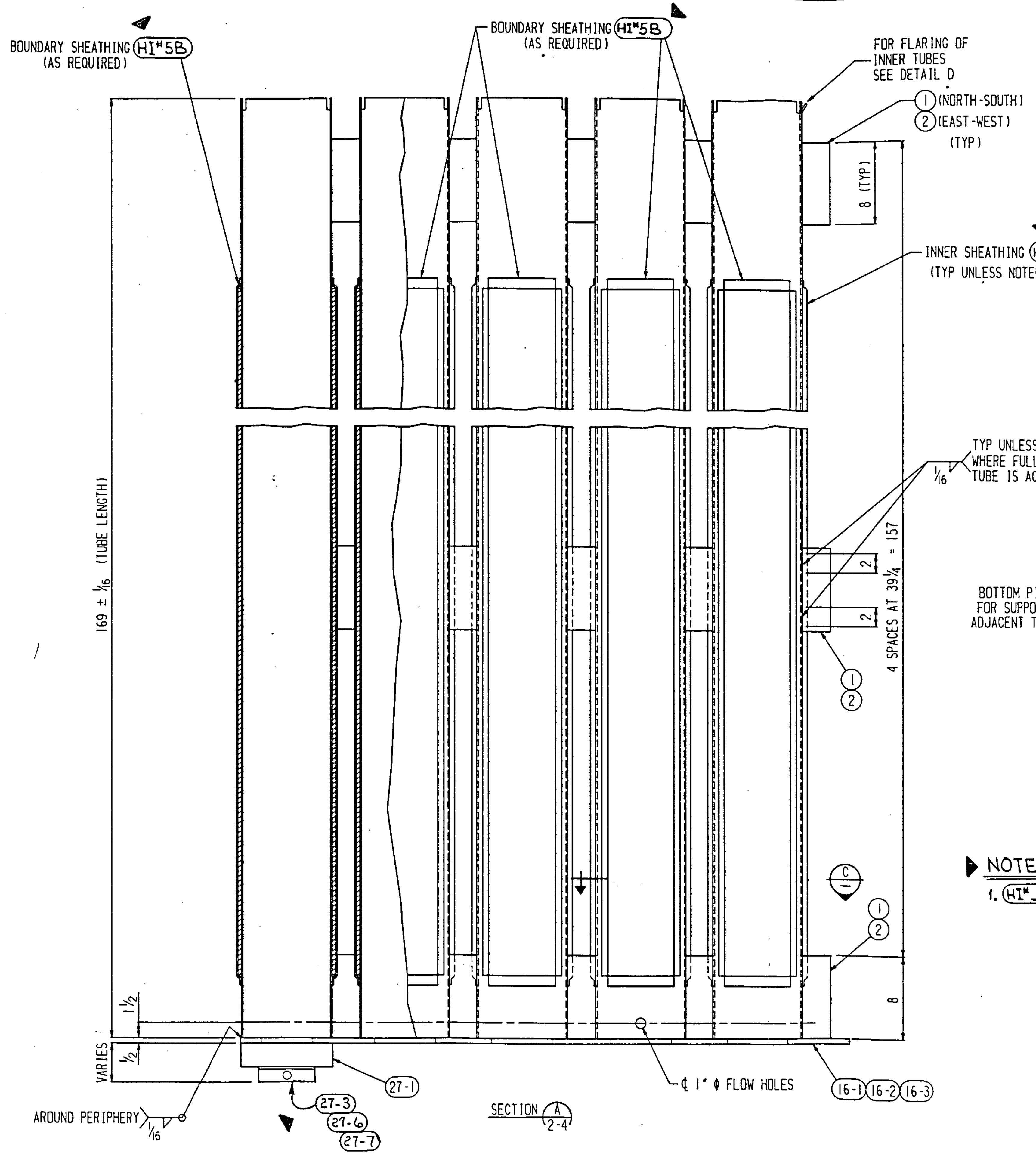
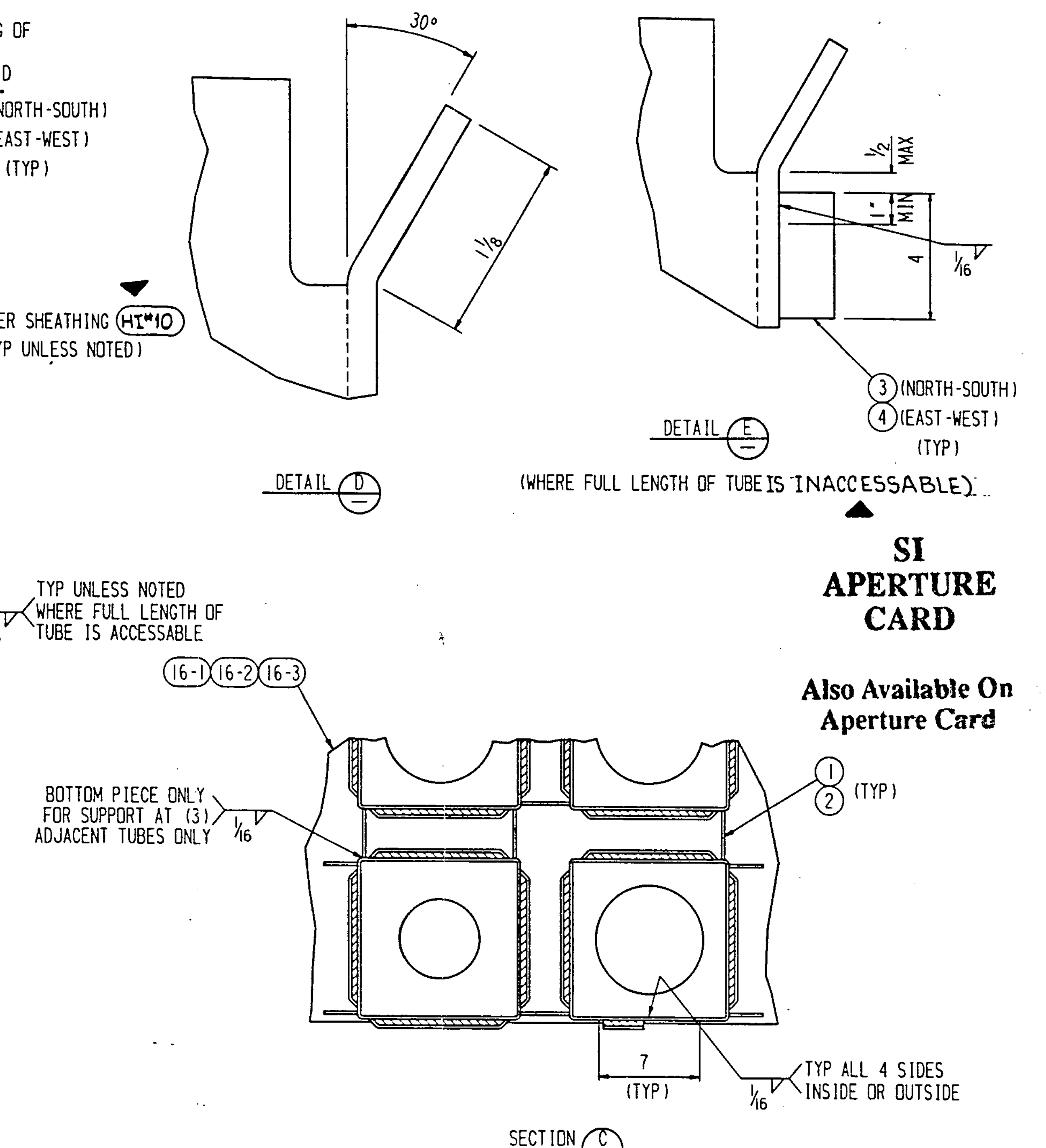




SI APERTURE CARD

Also Available On Aperture Card

NOTES:

1. HI#_ MEANS HOLTEC INTERNATIONAL MARK#.

BY	CHKD	DATE	REMARKS
DCC 3	ELB	10-23-89	ADJUSTABLE SUPPORTS ADD NOTE #1 CORRECT SPELLING REVISED MAX TO MIN
DB 2	CR	9-13-89	REVISED DETAIL E AND RELEASED FOR FAB
DB 1	CR	7-14-89	

REVISIONS

INDICATES CHANGE FROM PREVIOUS ISSUE

SUPPLIER'S/ PURCHASER'S NO

CBI

REGION 1 TYPICAL ELEVATION
SPENT FUEL STORAGE RACKS
CONSOLIDATED EDISON CO.
INDIAN POINT UNIT NO. 2

CUSTOMER'S NO 81000
BY DB CHKD CR DATE 5-15-89
L. L. HUDSON
ENGINEERING SUPERVISOR

CONTRACT NO 881161
DWG 14 SHT
REV 3

9002080340-02

Region 1

Shop Check Lists

CBI

SPECIAL CHECK LIST - SPENT FUEL RACK

QA INSPECTOR

THE LISTED OPERATIONS, EXAMS & CHECKS WERE PERFORMED

NO.	OPERATIONS, INSPECTIONS & CHECKS TO BE PERFORMED	INITIALS	DATE
1	INSPECT BASEPLATE SIZE & STENCILING AT CM 150		
2	INSPECT BASEPLATE HOLE SIZE, HOLE SPACING & GRID LAYOUT AT HBM		
3	INSPECT BASEPLATE ALIGNMENT ON FIXTURE		
4	INSPECT BASEPLATE FLATNESS ON FIXTURE		
5	SURVEILLANCE OF CELL PLACEMENT ON BASEPLATE TO TEMPLATE BEAM		
6	SCRIBE BENCH MARKS AT FOUR CORNERS OF RACK		
7	INSPECT LOCATION OF SUPPORT FEET & I.D.'S		
8	RANDOM INSPECT LEAD IN ANGLE AT TOP OF CELLS (REGION 1 RACKS ONLY)		
9	RE-LEVEL RACK IN GAGING STATION		
10	GAGE RACK PER IGT1N -RECORD RESULTS ON IGTR		
11	RECORD OVERALL DIMENSIONS OF BOUNDARY CELLS ON RACK/ PRISMATIC ENVELOPE OF RACK		
12	PERFORM CHECK OF TRAVEL RANGE ON SUPPORT FEET		
13	INSPECT RACK STENCILS (POST ASSEMBLY)		
14	INSPECT RACK CLEANLINESS		
15	ALL REQUIRED RECORDS RELATED TO THIS FUEL RACK ARE COMPLETE AND ARE ON FILE		
16			
17			
18			

Rack A	MADE BY	APP'D BY	REVISIONS	MADE				CONTRACT NO. 881161	SHEET
	DATE	DATE		APP'D					
				DATE					

CCM #14

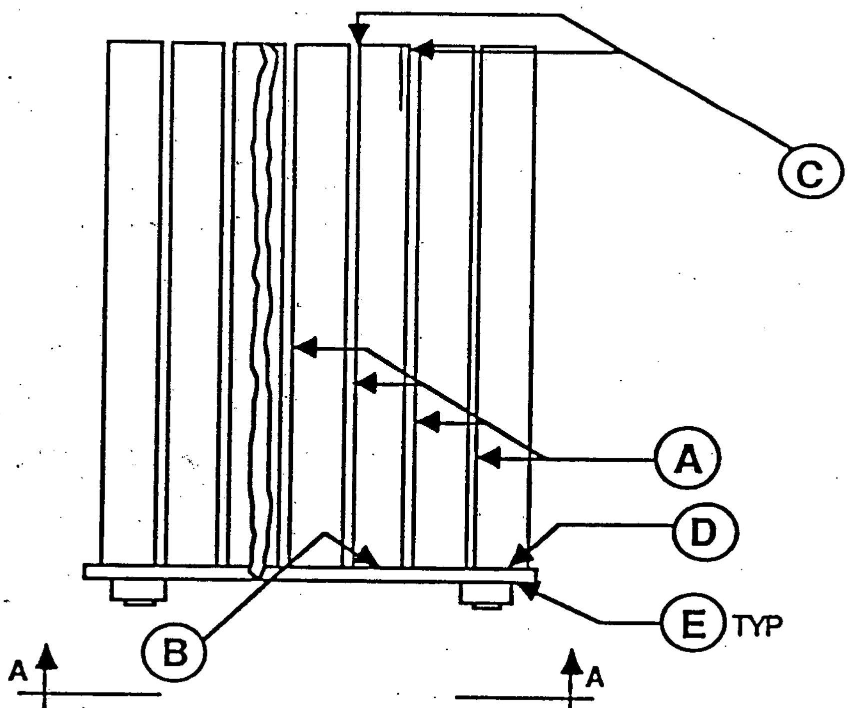
CBI SHOP CHECK LIST

APPLICABLE WELD PROCEDURES

NO.	PROCEDURE	REV.
1	WPS-E308L	0
2	WPS-ER308L-GM2	0
3	WPS-ER308L-GT1	0
4	WPS-SPOT-1	1
5	WPS-GT-3	0
6	WPS-ER308L-GM1	0

DESCRIPTION OF WELD SEAMS

A. TUBES TO TUBES
B. TUBES TO BASE PL (PERIPHERAL)
C. TUBES TO TUBES (TOP)
D. TUBES TO BASE PL (INSIDE)
E. SUPPORTS TO BASE PL



SEE SHEET 3 FOR SECTION A-A

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION
1 A	QA REVIEW CHECKLIST	2D	FINISH JOINT CHECK	2J	FINISHED JOINTS CHECKED
1B	FOREMAN REVIEW CHECKLIST	2E	VISUAL EXAM FINAL JOINT	2K	VT ALL JOINTS
2A	CHECK FIT UP SEAM A&D	2 F	CHECK FIT UP SEAMS B&C	3	QA FINAL CHECKLIST REVIEW
2B	CHECK WELD PROCED. A&D	2G	CHECK WELD PROCEDURES B&C		
2C	RECORD WELDER I.D. A&D	2H	RECORD WELDER I.D. B&C		

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:

INSPECTOR ____________________ DATE__________

Reviewed with ANI before use:
N/A ANI   N/A Date

Reviewed by QA Manager
________ Name   ________ Date

CONTRACT NO. 881161

NO. RACK - A
SHT. 1 OF 3

Made By	Chkd By	REV	By			
Date	Date		APP'D			
			Date			

Reviewed By: FOREMAN

CCM GE 516 REV Jun 87

CBI **SHOP CHECK LIST**

These examinations & operations were performed, results evaluated and accepted to applicable procedures.

REF. MARK	1 VI4X FIT UP CHECKED REV.1	2 MATERIAL ID RECORDED (TUBE)	3 WELD PROCED. AND REPAIR PROCED.	4 RECORD WELDERS ID ON SPREAD SHEET	5 VI4X WELDING CHECKED REV.1	6 VT5X REV.1	7	8	9	10	11	Witness Hold (W/H) ANI	COL. NO. CUSTOMER	See Non-conformance Control List No.
A	☒	☒ *	☒ 2 5 3	RECORD ON SHEET 1 A	☒	☒						N/A N/A		
B	☒		☒ 2 5 3	RECORD ON SHEET 1D	☒	☒						N/A N/A		
C	☒		☒ 2 5 3	RECORD ON SHEET 1C	☒	☒						N/A N/A		
D	☒		☒ 2 6 3	RECORD ON SHEET 1D	☒	☒						N/A N/A		
												N/A N/A		
												N/A N/A		
RACK A												N/A N/A N/A		
												N/A N/A		
												N/A N/A		
												N/A N/A		
												N/A N/A		

Made By	Chkd By	REV	
Date	Date	By / APP'D / Date	

* RECORD ON SPREAD SHEET 1 A

CONTRACT NO. 881161

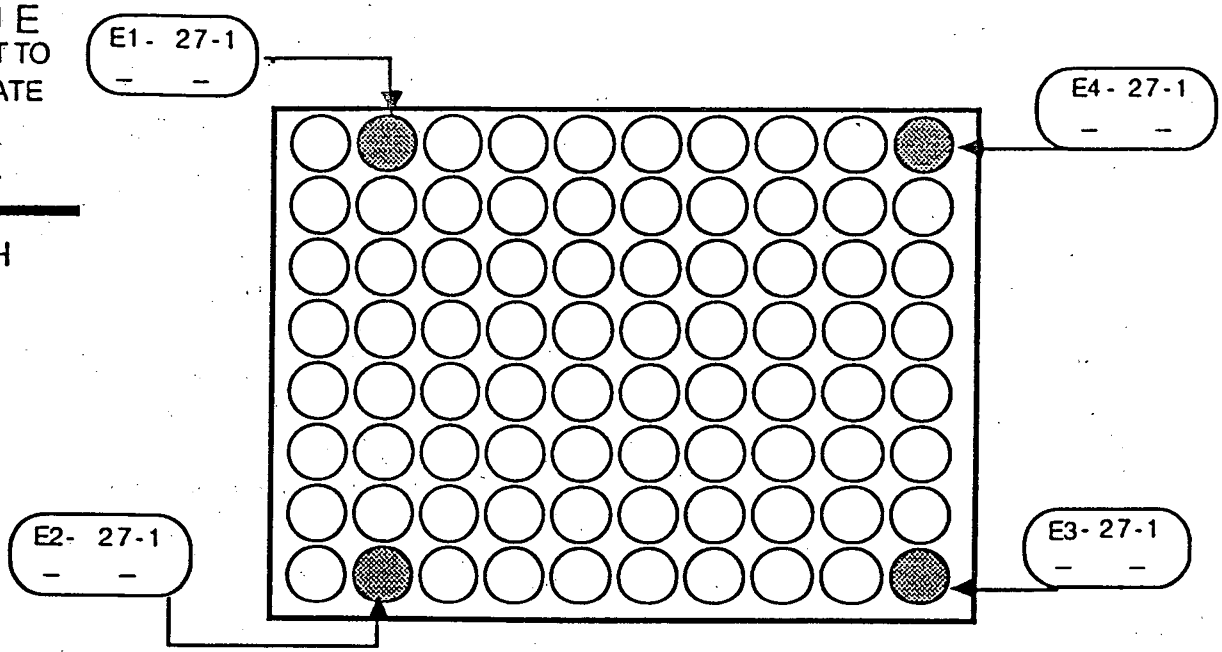
NO. RACK -A

SHT. 2 OF 3

CBI SHOP CHECK LIST

SEAM E
SUPPORT TO
BASE PLATE

POOL
NORTH



SECTION A - A   RACK - A

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION
1A	QA REVIEW CHECKLIST	2D	INSPECT FINAL SEAM		
1B	FOREMAN REVIEW CHECKLIST	2E	VISUAL EXAM SEAM		
2A	CHECK FIT-UP SEAM E	2F	PT EXAM SEAM		
2B	CHECK WELD PROCEDURE				
2C	RECORD WELDER I.D.				

These examinations & operations were performed, results evaluated and accepted to applicable procedures.

REF. MARK	1 Fit Up Checked	2 Matl ID Record	3 Weld Proced. Spec.and Repair Proced.	4 Record Welder's ID	5 VI4X Welding Checked REV. 1	6 VT5X REV.1	7 PT5X REV.1	8	9	Witness Hold (W / H) ANI	COL NO. CUSTOM.	See Non-conform-ance Control List No.
E 1	X	X	X ① ③	X	X	X	X			N/A N/A		
E 2	X	X	X ① ③	X	X	X	X			N/A N/A		
E 3	X	X	X ① ③	X	X	X	X			N/A N/A		
E 4	X	X	X ① ③	X	X	X	X			N/A N/A		

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:

INSPECTOR ______________ DATE ______

Reviewed with ANI before use:
N/A (ANI)   N/A (Date)

Reviewed by QA Manager
Name ______ Date ______

CONTRACT NO. 881161

NO. RACK - A
SHT. 3 OF 3

Made By	Chkd By	REV	By			
Date	Date		APP'D			
			Date			

Reviewed By: FOREMAN

Region 2

Shop Check Lists

CBI

SPECIAL CHECK LIST - SPENT FUEL RACK

NO.	OPERATIONS, INSPECTIONS & CHECKS TO BE PERFORMED	QA INSPECTOR THE LISTED OPERATIONS, EXAMS & CHECKS WERE PERFORMED INITIALS	DATE
1	INSPECT BASEPLATE SIZE & STENCILING AT CM 150		
2	INSPECT BASEPLATE HOLE SIZE , HOLE SPACING & GRID LAYOUT AT HBM		
3	INSPECT BASEPLATE ALIGNMENT ON FIXTURE		
4	INSPECT BASEPLATE FLATNESS ON FIXTURE		
5	SURVEILLANCE OF CELL PLACEMENT ON BASEPLATE TO TEMPLATE BEAM		
6	SCRIBE BENCH MARKS AT FOUR CORNERS OF RACK		
7	INSPECT LOCATION OF SUPPORT FEET & I.D.'S		
8	RANDOM INSPECT LEAD IN ANGLE AT TOP OF CELLS (REGION 1 RACKS ONLY)		
9	RE-LEVEL RACK IN GAGING STATION		
10	GAGE RACK PER IGT1N -RECORD RESULTS ON IGTR		
11	RECORD OVERALL DIMENSIONS OF BOUNDARY CELLS ON RACK/ PRISMATIC ENVELOPE OF RACK		
12	PERFORM CHECK OF TRAVEL RANGE ON SUPPORT FEET		
13	INSPECT RACK STENCILS (POST ASSEMBLY)		
14	INSPECT RACK CLEANLINESS		
15	ALL REQUIRED RECORDS RELATED TO THIS FUEL RACK ARE COMPLETE AND ARE ON FILE		
16			
17			
18			

RACK D	MADE BY	APP'D BY	REVISIONS	MADE				CONTRACT NO. 881161	SHEET
	DATE	DATE		APP'D					
				DATE					

CCM #14

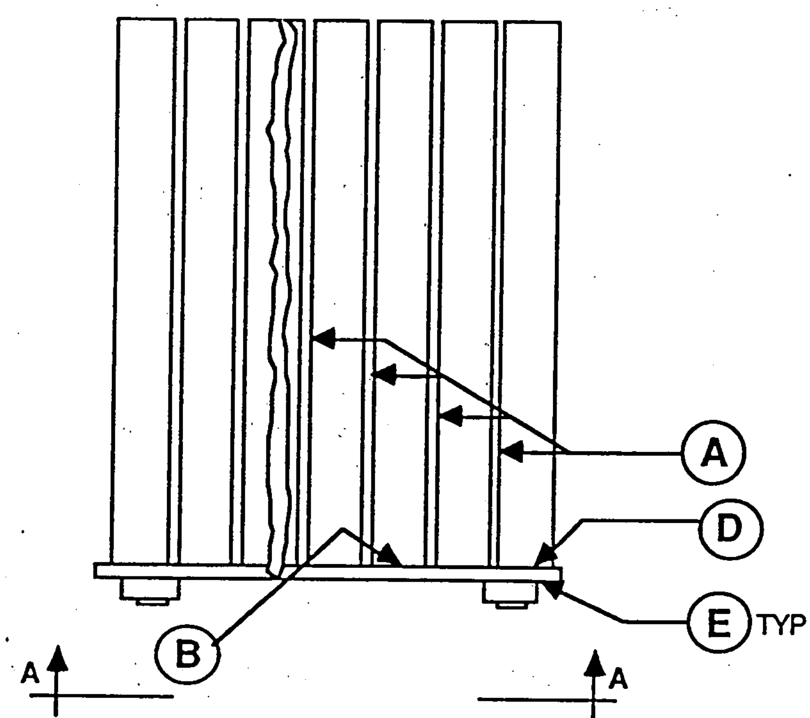
CBI SHOP CHECK LIST

APPLICABLE WELD PROCEDURES

NO	PROCEDURE	REV
1	WPS-E308L	0
2	WPS-ER308L-GM2	0
3	WPS-ER308L-GT1	0
4	WPS-SPOT-1	1
5	WPS-GT-3	0
6	WPS-ER308L-GM1	0

DESCRIPTION OF WELD SEAMS

A. TUBES TO TUBES
B. TUBES TO BASE PL (PERIPHERAL)
D. TUBES TO BASE PL (INSIDE)
E. SUPPORTS TO BASE PL



SEE SHEET 3 FOR SECTION A-A

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION
1 A	QA REVIEW CHECKLIST	2D	FINISH JOINT CHECK	2J	FINISHED JOINTS CHECKED
1B	FOREMAN REVIEW CHECKLIST	2E	VISUAL EXAM FINAL JOINT	2K	VT ALL JOINTS
2A	CHECK FIT UP SEAM A&D	2 F	CHECK FIT UP SEAMS B	3	QA FINAL CHECKLIST REVIEW
2B	CHECK WELD PROCED. A&D	2G	CHECK WELD PROCEDURES B		
2C	RECORD WELDER I.D. A&D	2H	RECORD WELDER I.D. B		

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:

INSPECTOR ____________ DATE ________

Reviewed with ANI before use: N/A (ANI) N/A (Date)

Reviewed by QA Manager
______ Name ______ Date

CONTRACT NO. 881161

NO. RACK - D
SHT. 1 OF 3

Made By	Chkd By	R E V	By			
			APP'D			
Date	Date		Date			

Reviewed By: FOREMAN

CBI SHOP CHECKLIST

These examinations & operations were performed, results evaluated and accepted to applicable procedures.

REF. MARK	1 VI4X FIT UP CHECKED REV.1	2 MATERIAL ID RECORDED (TUBE)	3 WELD PROCED. AND REPAIR PROCED.	4 RECORD WELDERS ID ON SPREAD SHEET	5 VI4X WELDING CHECKED REV.1	6 VT5X REV.1	7	8	9	10	11	Witness Hold (W / H) ANI	COL. NO. CUSTOMER	See Non-conformance Control List No.
A	X	X *	X 2 5 3	RECORD ON SHEET 1 A	X	X						N/A N/A		
B	X		X 2 5 3	RECORD ON SHEET 1 D	X	X						N/A N/A		
				RECORD ON SHEET								N/A N/A		
D	X		X 2 6 3	RECORD ON SHEET 1 D	X	X						N/A N/A		
												N/A N/A		
												N/A N/A		
ACK												N/A N/A		
												N/A N/A		
												N/A N/A		
												N/A N/A		
												N/A N/A		

de By	Chkd By	REV By			
ate	Date	APP'D			
		Date			

* RECORD ON SPREAD SHEET 1 A

CONTRACT NO. 881161

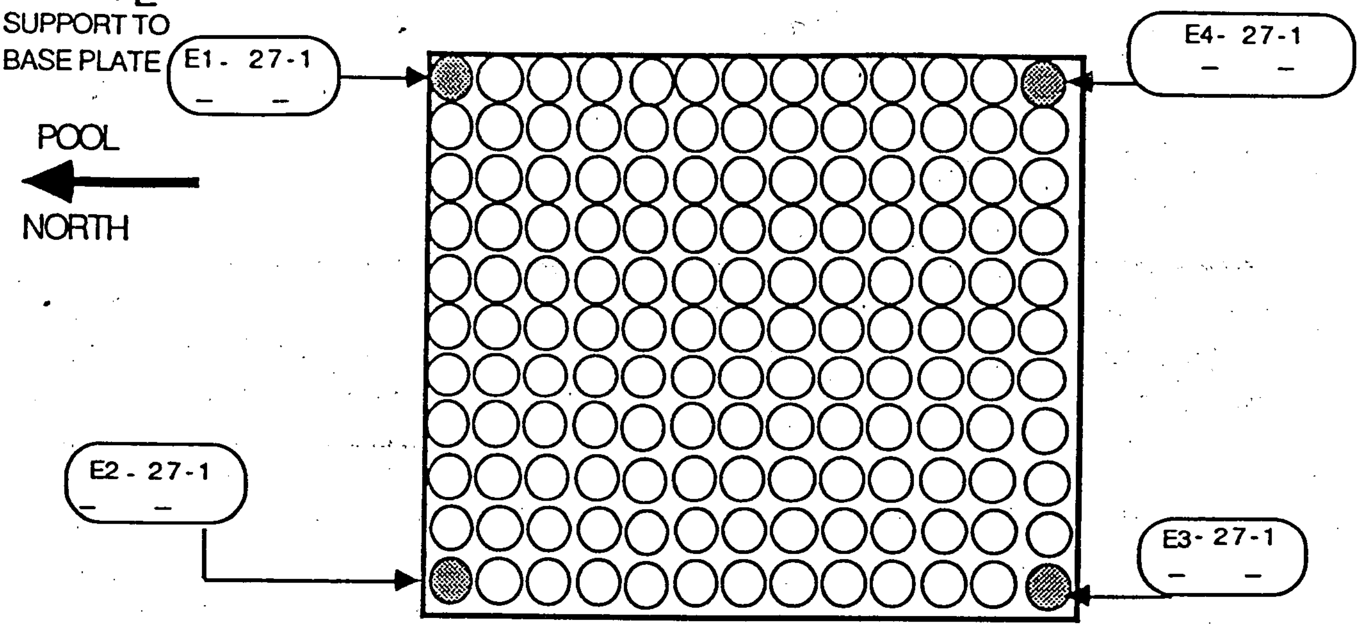
NO. RACK - D

SHT. 2 OF 3

SHOP CHECK LIST

CBI

SEAM E
SUPPORT TO
BASE PLATE



SECTION A - A RACK - D

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION
1A	QA REVIEW CHECKLIST	2D	INSPECT FINAL SEAM		
1B	FOREMAN REVIEW CHECKLIST	2E	VISUAL EXAM SEAM		
2A	CHECK FIT-UP SEAM E	2F	PT EXAM SEAM		
2B	CHECK WELD PROCEDURE				
2C	RECORD WELDER I.D.				

These examinations & operations were performed, results evaluated and accepted to applicable procedures.

REF. MARK	1 Fit Up Checked	2 Matl ID Record	3 Weld Proced. Spec.and Repair Proced.	4 Record Welder's ID	5 VI4X Welding Checked REV. 1	6 VT5X REV.1	7 PT5X REV.1	8	9	Witness Hold (W / H) ANI	COL NO. CUSTOM.	See Non-conformance Control List No.
E 1	X	X	1 3	X	X	X	X			N/A N/A		
E 2	X	X	1 3	X	X	X	X			N/A N/A		
E 3	X	X	1 3	X	X	X	X			N/A N/A		
E 4	X	X	1 3	X	X	X	X			N/A N/A		

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:

INSPECTOR ______________ DATE ________

Reviewed with ANI before use:
N/A (ANI) N/A (Date)

Reviewed by QA Manager
Name ______ Date ______

CONTRACT NO. 881161
NO. RACK - D
SHT. 3 OF 3

Made By	Chkd By	REV	By			
Date	Date		APP'D			
			Date			

Reviewed By: FOREMAN

APPROVED	Engr	Corp Weld	Corp QA	Const	Mfg		BY	DATE
		LRS	REK		RRW	PREPARED	PTC	7-7-87
						REVISED	RGL	7-11-89
						AUTHORIZED	RRK	8-7-89
						REFERENCE STANDARD		REV. NO.

1.0 SCOPE

This procedure describes the check list system in a shop for process control. When using the Shop Check List System, control per this procedure is mandatory for Type A material and welds thereto (including repairs) on all classes of work and for the following additional areas on specific classes:

Class 1 - repairs to Type B and C material.

Class MC - welds of Type B material together (including repair of such welds) within 16t of Type A material.

Class 2 & 3 Tanks - welds of bottom plates together and nozzle-to-bottom plates. For roofs made from Type B material - welds of roof plates together and nozzle to roof plates. Includes repairs of these welds.

2.0 REFERENCES

2.1 AP 2-8, Classification of Materials
AP 9-15, Welder I.D. Requirements
AP 11-1, Handling of Nonconformances
AP 14-1, General Procedure for Quality Assurance Records

2.2 Reference to the above procedures includes equivalent FAP's, AP Addenda or contract procedures.

3.0 RESPONSIBILITIES

3.1 The Production Superintendent shall prepare the Shop Check Lists and distribute them to the Production Foremen.

3.2 The QA Coordinator, who reports to the QA Manager, shall review the Shop Check Lists before use, assign QA witness and/or hold points, present the Shop Check Lists to the ANI, and review and file completed process control documents.

3.3 The Production Foreman, who reports to the Production Superintendent, shall review the Shop Check Lists prior to performing any operations and obtain the required signoffs on the Shop Check Lists.

4.0 DEFINITIONS

None

5.0 SHOP CHECK LIST SYSTEM

5.1 This system is designed for use on items for which sequencing of operations is not important.

5.2 This system includes the following:

5.2.1 Check List (Attachments 1, 2 and 3)

5.2.2 Control List (Attachment 4)

5.2.3 Repair Check List (Attachment 5)

6.0 CHECK LISTS

6.1 Process control for welding, heat treating, NDE and forming operations requiring procedures shall be outlined on the check lists. Contract drawings and the Contract QA Handbook are used for information. The Production Superintendent is responsible for preparation of the check lists. (See paragraph 6.6 for contents of check lists.)

6.2 Prior to use, the check lists shall be reviewed by the QA Coordinator for inclusion of QA requirements, using the contract drawings and Contract QA Handbook. QA hold points shall be indicated on the check lists by the QA Coordinator. His approval of the check lists shall be documented by signoff on the check lists.

6.3 The QA Coordinator shall present the check lists and associated contract drawings to the ANI for his review. The ANI may place witness and/or hold points on the check lists for the listed operations. The ANI's review shall be documented by the ANI's initials and date on the check lists.

6.4 After review by the ANI, the check lists are returned to the Production Superintendent for use. Working copies of the check lists may be used to control operations. When working copies are used, the Production Superintendent shall complete the official check lists (check lists containing the original ANI and QA approval) from the working copies.

6.5 Following distribution by the Production Superintendent, the Production Foreman shall initial and date the check lists before starting any operations, signifying that he has reviewed and understands the listed requirements.

6.6 Entries on the check lists shall include:

6.6.1 Identification with a contract number and check list number.

6.6.2 A listing of required operations. The "Seq. Operation", "Weld Procedure Spec and Repair Procedure", and "Proc & Rev." columns shall be used for this purpose. Sequence numbers need not be assigned provided that witness and/or hold points are not bypassed. An "X" shall be placed in the small box within the signoff square to indicate each required operation. For temporary attachments and plate cleanup, documentation may be by groups and columns for "Fit Up Checked" and "Material ID Recorded" are not applicable.

6.6.2.1 Per AP 9-15, specify the requirements for and provide a place to record welder I.D.

6.6.2.2 When required in AP 2-8 for Type A material, provide a place for the Production Superintendent to record item location by identification information (piece mark, or piece mark and heat serial code, or piece mark and serial number) from the item.

6.6.3 A sketch, when needed for clarity. Sketch ID shall be entered in the "Ref. Mark" column to identify specific items such as weld seams or piece surfaces.

6.6.4 "Hold" and "Witness" points, where required by the customer's inspector, ANI, or QA Manager. Also, when required, the customer inspector's signoff on the completed form. "Hold" and "Witness" points shall be controlled in accordance with paragraph 10.0.

6.6.5 Signoffs by QA Inspectors under "Fit-Up Checked" certifying that fit-up was checked prior to welding.

6.6.6 For Type A material, when required by AP 2-8, provide for signoffs by QA Inspectors under "Material ID Recorded" certifying that material identification was recorded.

6.6.7 Signoffs by QA Inspectors under "Welding Checked" certifying that requirements of the referenced WPS were met, welders were qualified and, except for temporary attachments, surface and configuration of the completed weld meet applicable requirements.

6.6.8 Signoffs under "Proc & Rev" by NDE personnel certifying that:

A. Examinations were completed
B. Reports were made and are traceable from the check list
C. Nonconformities have been corrected.

6.6.9 Signoffs under "Proc & Rev" by individuals responsible for performance of other required operations (e.g., PWHT and dimensional checks requiring procedures), certifying that the operation was performed per requirements.

6.6.10 References by the QA Coordinator under "Nonconformance Control List No." for applicable nonconformities (see AP 11-1).

6.6.11 Unit acceptance for the assembly by the QA Inspector, whose signoff on a completed check list is his certification that:

A. All items were identified on the check list and serialization or heat coding was accomplished when required.

B. Assemblies were properly marked.

C. Fabrication workmanship meets Code and customer requirements.

6.6.12 A final review by the QA Coordinator, and signoff on the completed check list, certifying that all operations have been completed and signed off, that repairs have been completed in accordance with referenced repair procedures, and that related required records are on file.

6.7 The Production Foreman is responsible to obtain the required signoffs for all operations, including final inspection, and maintaining custody of the check lists until they are completed. The "official" check list (containing the original signoffs) shall be kept in the office of the Production Foreman responsible for the current operations or in another location designated by the Production Superintendent. It may be removed to other offices as required; however, it shall be the responsibility of the Production Foreman to know the location of check lists removed.

6.8 The Production Foreman shall be responsible to notify the QA Coordinator of upcoming witness and/or hold points in a timely manner so the QA Coordinator can signoff his witness and/or hold points and can give the ANI reasonable notice in advance of his witness and/or hold points.

6.9 Completed check lists are sent to the QA Coordinator for final review and acceptance. The check lists are then given to the ANI for his final review and acceptance which is documented on the check list.

6.10 Revisions to check lists shall be prepared, reviewed and approved in the same manner as the original check lists.

7.0 <u>CONTROL LISTS</u>

7.1 Control Lists (Attachment 4) are maintained by the QA Coordinator as a summary of the check lists that have been issued. They provide space for recording the dates that the check lists were issued and completed.

8.0 <u>REPAIR CHECK LIST</u>

8.1 Repair Check Lists (Attachment 5) shall be used to control the repair of nonconformities (see AP 11-1). Repair Check Lists shall be initiated and maintained by the Production Superintendent.

8.2 Initial entries shall include:

8.2.1 The nonconformance number

8.2.2 A description of the nonconformity (size, depth and location of nonconformity for base metal defects and as may be necessary for control) unless included in a repair procedure.

8.2.3 Reference to applicable repair procedure and revision.

8.2.4 Reference to other applicable procedure and revision numbers or, if there is no written procedure, a complete description using as many lines or spaces as necessary to fully describe repair steps.

8.2.5 Under "Hold or Witness", designation of the proper releasing authority for established "hold" and "witness" points.

8.2.6 The ANI's initials in the ANI column to indicate that the foregoing entries were reviewed with him prior to repair.

8.3 Additional entries during the progress of work shall include:

8.3.1 Under "NDE Rept. ID", the identification of NDE reports (not required when traceability to the report is accomplished by process control document and sequence number).

8.3.2 Under "Welder ID", the identification of welders performing work per AP 9-15.

8.3.3 At the completion of each listed operation, signoffs by the applicable QA Inspector (Welding, NDE or other) under "CBI QA" certifying that:

A. Welders were qualified, requirements of the WPS were met and surface and configuration of the completed weld meet applicable requirements.

B. Required NDE was completed, and reports were made and are traceable from the check list.

C. The operation (other than welding or NDE) was performed per requirements.

8.3.4 The ANI may initial under "ANI" to indicate operations or examinations witnessed. (See paragraph 10.0 for "hold and witness points").

8.3.5 The last column may be used by the customer's inspector or others to indicate operations or examinations witnessed.

9.0 WELDED CORRECTIONS

9.1 Welded corrections made to welds during the course of deposition (prior to submittal for NDE acceptance examination) are handled as part of the welding operation.

9.2 Correction of welds found unacceptable due to visual inspection after final acceptance (welding checked signed off and/or NDE signed off) and prior to PWHT shall be performed to a correction procedure and documented in the same manner as required for repairs. Typical unacceptability is due to improperly sized butt welds, improper length or size of fillet welds or undercut.

9.3 Each repair (or group of repairs) to be controlled on the Shop Check List shall be entered on a separate line of the applicable check list. The following shall be documented:

9.3.1 Identification of repair procedure, welding procedure and welder I.D. per AP 9-15.

9.3.2 Signoffs for NDE of repair cavity (if required), checking of repair weld and NDE of completed repair.

9.3.3 Signoffs by the ANI for repairs he has witnessed.

10.0 <u>"HOLD" AND "WITNESS" POINT CONTROL</u>

10.1 Work shall not proceed beyond a designated "hold" point until the "hold" point is signed off by the authority who placed it or he has had it voided.

10.2 The authority voiding a "hold" point must initial and date such action on the Check List.

10.3 Inspectors placing "hold" points shall be given timely notification (per local arrangement) of the anticipated reaching of the "hold" point.

10.4 Work may proceed past a "witness" point, provided the individual placing it has been given timely notification (per local arrangement) of the anticipated reaching of the "witness" point.

11.0 <u>RECORDS</u>

11.1 The following records completed by this procedure are Quality Assurance records and shall be handled per AP 14-1:

11.1.1 Shop Check Lists

11.1.2 Repair Check Lists

11.1.3 Control Lists

12.0 ATTACHMENTS

12.1 Attachment 1 - Shop Check List, Form GE515

12.2 Attachment 2 - Shop Check List, Form GE516

12.3 Attachment 3 - Shop Check List, Form G01258

12.4 Attachment 4 - Control List, Form GE518

12.5 Attachment 5 - Repair Check List, Form G01002

ATTACHMENT 1

CBI SHOP CHECK LIST

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION

These operations and examinations were performed, results evaluated and accepted to applicable procedures.

REF. MARK	1 Fit Up Checked	2 Mat'l ID Recorded	3 Weld Procedure Spec. and Repair Procedure	4 Record Welders ID	5 Welding Checked	6 Proc & Rev	7 Proc & Rev	8 Proc & Rev	9 Proc & Rev	Witness or Hold Points: Witness W / Hold H — COL. NO. — ANI	See Nonconformance Control List No.

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:
Inspector ____________ Date ______

Reviewed with ANI before use:
____________ ANI ______ Date

Reviewed by QA Manager
____________ Name ______ Date

Reviewed By:
____

CONTRACT NO.

NO. ______

Sht ____ of ______

Made By	App'd By	REV	By			
Date	Date		App'd			
			Date			

GE 515 REV JUN 89

ATTACHMENT 2

CBI SHOP CHECK LIST

SEQ	OPERATION	SEQ	OPERATION	SEQ	OPERATION

CBI
ASSEMBLY INSPECTED & ACCEPTED BY:

Inspector________________ Date ________

Reviewed with ANI before use:

ANI ________ Date ________

Reviewed by QA Manager

Name ________ Date ________

Reviewed By:

CONTRACT NO.

NO. ________

Sht ____ of ____

Made By	App'd By	REV	By			
			App'd			
Date	Date		Date			

ATTACHMENT 3

CBI SHOP CHECK LIST

These operations and examinations were performed, results evaluated and accepted to applicable procedures.

Reference Mark	1 Fit Up Checked	2 Mat'l ID Recorded	3 Weld Procedure Spec. and Repair Procedure	4 Record Welders ID	5 Welding Checked	6 Proc & Rev	7 Proc & Rev	8 Proc & Rev	9 Proc & Rev	10 Proc & Rev	11 Proc & Rev	12 Proc & Rev	13 Proc & Rev	Witness or Hold Points: Witness W___ Hold H___ Cat. No.; ANI	See Non-conformance Control List No.

Made By / Date	App'd By / Date	REV: By / App'd / Date				CONTRACT NO.	NO.____ Sheet ___ of ___

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ATTACHMENT 4

CBI CONTROL LIST
SHOP CHECK LIST

CONT. NO. ______
SHEET ______ OF ______
REV. ______ DATE ______

CHECK LIST NUMBER	CHECK LIST LTR.	ASSEMBLY MARK	DESCRIPTION	DATE ISSUED	DATE COMPLETED

Printed in USA

ATTACHMENT 5

CBI REPAIR CHECK LIST

NOTE 1: enter H or W and:
ANI - Authorized Nuclear Inspector
C - Customer
Q - QA or CQA

Defect Description / Repair Procedure No.	Entered By and Date	Procedure or Operation as Required		NDE Rept ID	Hold or Witness See Note 1 Enter as Req'd	Welder ID	CBI QA Listed Operations Exams & Checks were performed Results Evaluated and Accepted By: Initials	CBI QA Date	ANI Witnessed listed Operations or Examinations Initials	ANI Date	Witnessed listed Operations or Examinations Initials	Date
		Reviewed with ANI before use										
		Exam Cavity or Fit-up										
		Weld										
		Re-exam										
		Reviewed with ANI before use										
		Exam Cavity or Fit-up										
		Weld										
		Re-exam										
		Reviewed with ANI before use										
		Exam Cavity or Fit-up										
		Weld										
		Re-exam										

Location ____________

Contract ________ Page ____ of ____

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III. B. RACK DESIGN

2. Explain the sentence (LAR Section 3.1.4): "The extent of welding is selected to 'detune' the racks from the ground motion (OBE and SSE)."

RESPONSE

The extent of cell-to-cell welding determines the "beam mode" stiffness of the rack. Although the response of a rack to seismic loadings is extremely non-linear, the maximum displacements (including sliding, tilting, twist, etc.) are found to be sensitive to the beam mode stiffness of the module. At the rack module design stage, some parametric studies of module response for the specified seismic loadings helps establish a rack design which is not apt to experience large kinematic response under the postulated loadings.

The volume of material associated with these parametric studies is quite large (8-10,000 pages). If further details of this methodology and it's application to the proposed Indian Point 2 racks is desired by the NRC, a technical audit of this material at Con Edison's rack designer's office can be provided.

III. B. RACK DESIGN

3. For weld stresses between the baseplate and support leg, justify the use of limit analysis when there is a partial penetration grove weld joining the components. Provide information on how two directional bending and shear at the junction are considered in constructing the interaction diagram. Also, provide 'R' factor if only elastic analysis (instead of limit analysis) were used.

RESPONSE

The weld joint between the baseplate and the internally threaded member of the support leg assembly is a partial penetration groove weld reinforced by a covering fillet weld. The weld wire is also of austenitic stainless steel stock (ER308). Even though the material yield strength of the weld wire is considerably greater than that of the base material, its yield strength is conservatively taken equal to that of the base material.

The governing code for the stress analysis of the weld structure is Section III subsection NF of the ASME Code which, at the present time, contains no stress limits for welds section under Level D loadings (which corresponds to the SSE condition). Even for normal and upset conditions the Code prescribes stress limits for equivalent static loads. The dynamic analysis of the rack provides the peak values of reactions produced by the interaction of inertia and fluid forces. In the interest of conservatism, these peak values, rather than equivalent static loads, are used for computing the weld section stresses.

In the absence of a uniquely prescribed stress limit in the Code, the stress analysis of the weld section has followed the practice of strength evaluation of reinforced concrete section. SRP 3.8.4 of NUREG 0800 provides procedures for calculating "Design Strength" of reinforced concrete structures. Following the same design approach, the "Design Strength" of the weld section is calculated assuming that the stress distribution is fully plastic across the cross-section. Recalling that the yield strength of 304 stainless steel is only 35.2% of its ultimate strength, it is concluded that the computed "Design Strength" of the weld cross-section has an inherent factor of safety against failure equal to 2.84. In other words, if the applied loads are found to reach the limit of Design Strength of the weld section based on the rectangular stresses distribution assumption, then the inherent factor of safety against failure at that point is equal to 2.84. This is totally consistent with definition of Level D condition which postulates that permanent deformation is acceptable but total structure collapse is not.

A comparison of the factor of safety corresponding to the weld Design Strength approach, and that used in base materials points up the added conservatism in this method. Section F.1332 seeks to limit the base metal stress for Level D condition to 0.7Su (Su = ultimate strength), which implies a factor of safety of 1.428 against failure. As stated above, the inherent factor of safety in the weld analysis using the Design Strength approach is much greater (= 2.84).

The governing codes, mentioned above, do not require combination of shear loads with two bending moments. However, the interaction analysis is performed using the vectorial resultant of the two moments and direct thrust on the support pedestal-baseplate interface.

In the following, we present the results assuming a linear elastic stress distribution in the welds.

The results for 1% damping (case DOB) in Table II.1, II.2, provided in response to question II.1, give stress factors R_1 = .183 and R_6 = .312 for the spindle cross-section. Since R_6 is the sum of direct and bending effects for the spindle, and R_1 is the stress factor for direct compression, we can calculate the stress factor for bending as:

$$R_b = R_6 - R_1 = .129$$

Since the allowable stress is $.6\sigma_y$ = 15000 psi, the actual direct stress on the spindle is

$$\sigma_1 = 15000\ R_1 = 2745 \text{ psi}$$

The bending stress at the extreme fiber of the spindle is

$$\sigma_B = 15000\ R_B = 1935 \text{ psi}$$

Because of the relatively low value of bending moment, there is no tension acting on any cross-section of the spindle at the baseplate spindle interface. Thus, when the additional fillet weld area is accounted for, the maximum normal stress at the extreme fiber will be less than 4680 psi which translates to a throat shear stress of 6619 psi. Note that this result occurs for the SSE seismic event.

III. B. RACK DESIGN

4. The stress factors (Ri) only addresses stresses in the support feet and base plate. Provide stresses in the cell walls under an SSE considering the longitudinal (overall rack behavior), transverse compressive (due to hydrodynamic load between the racks) and impact loads from the fuel assemblies.

RESPONSE

The governing code for rack structural design as mandated by the OT Position Paper (USNRC c' 1978) is ASME Section III Subsection NF for Class 3 structures. This Code places strict limits on all "primary stresses" which are subdivided into seven categories. These are reported in Table 6.5 of the SAR as dimensionless factors (R_i, i = 1,2....7). The contribution of the hydrodynamic loads and fuel assembly impact on the overall rack behavior is included in the above stress factors. The Code prescribes no limit on the local stresses which develop in the baseplate or the cell. These stresses are defined as "local bending" or "secondary" stresses in the Code. The governing ASME Code (Section III NF Class 3) places no limit on the "local" stresses.

The stresses reported in the top line of Table 6.5 for each run are the stresses in the cell walls considering overall rack behavior. The transverse compressive stresses due to hydrodynamic loads and impact loads due to fuel assemblies do not have a prescribed Code stress limit, and therefore are not required to be evaluated and combined with the primary stresses.

IV. OTHER ACCIDENT CONSIDERATIONS

1. Provide calculation which demonstrates the assertions in the submittal (LAR Sections 7.1.1 a and b) that the structural integrity of the rack and subcriticality of the stored fuel is assured.

RESPONSE

The LAR contains statements on results of certain accident scenarios which are postulated. Here we enclose results of bounding calculations which demonstrate that the postulated accident conditions do not cause unacceptable conditions in the fuel racks.

Detailed calculations are provided here for

1. Dropping of a fuel assembly

   a. from a height of 36" above the top of the rack and have it hit the top plate

   b. from a height of 36" above the top of the rack and have it hit the baseplate

In the case of accident 1a, permanent deformation would be confined to the top region of the rack above the active fuel region. This is an area where no other postulated conditions would result in a continuing high stress. We therefore do not consider any other loading acting in concert with the above postulated accident.

For the case of accident conditions 1b, the concern would be to maintain the integrity of the pool floor liner plate and to maintain the center-to-center distance between adjacent storage locations.

The center-to-center distance between adjacent storage cells is not dependent on the presence or absence of support from the baseplate. The purpose of the above calculation is only to show that there would be no danger to the liner. In the event of a dropped fuel assembly, it is correct to say that the baseplate may separate from the tube in the immediate vicinity of the affected tube. While this would result in baseplate plastic bending, it would not affect center-to-center spacing since there would be no effect on the welds between adjacent tubes nor on the baseplate-to-tube welds away from the immediate vicinity of the dropped assembly.

Accident #1a - A Mass (assumed weight = 2000lb) drops 36" in water and hits the top of the rack.

When a rigid body moves with velocity V and strikes the edge of an elastic rod or plate, it may be shown (Timoshenko and Goodier, 1951, pp. 441-442) that an impact stress develops of magnitude

$$= EV/C; \quad C^2 = E/\quad; \quad = \text{mass density}$$

Based on a drop velocity of 135.22 in/sec at impact at the top of the rack, we show that the wave propagation stress at the point of impact is below yield:

$$\sigma = (E\rho)^{1/2} V = 19330 \text{ psi}$$

We also examine the depth of propagation down the cell should local bucking occur causing the cell wall to have to support the impact load by shear alone.

Let the impact be spread over the width W of one fuel assembly. Let d be the depth (toward the active fuel region) that is capable of carrying shear and resisting the impact. Then, if σ is the impact stress, we have

$$W\sigma t = 2d\tau_y t$$

Where τ_y is the shear yield stress. Since $\tau_y = .577\sigma_y$ the depth of cell required to support σ is

$$d = \frac{\sigma W}{\sigma_y (1.154)} = 5.127''$$

(Note: σ_y increased by 15% for dynamic loading)

That is, in the worst case, yielding may occur to a depth of 5.127" below the top of the rack. This is above the active fuel region so there is no safety concern for this condition.

Accident 1b - A Fuel Assembly drops to the baseplate

As noted, the major concern is with the integrity of the pool liner. In the dry condition, damage to the liner is not a safety concern since there is no water to contend with. The design analysis simply shows that while welds may break, the baseplate structure has sufficient strength to prevent the dropped fuel assembly from hitting the liner. We examine only the wet condition where there is a potential safety concern.

Considerations of a fuel assembly dropping through a narrow channel filled with water lead to the result that the impact velocity at the rack base of a 2000 lbs assembly is

$$VF = 257 \text{ in./sec}$$

To check maximum baseplate deformation after local weld damage, we treat the baseplate deforming section as a circular plate and wish to show that maximum baseplate deformation h is less than the minimum distance from the baseplate to the liner. The energy to be absorbed is

$$U = \frac{1}{2} \; \frac{W}{g} \; VF^2$$

We consider the plate absorbing this energy by stretching as a membrane. Thus

$$\frac{1}{2} \left(\left(\sigma_r \epsilon_r + \sigma_\theta \epsilon_\theta \right) \pi R^2 \, T \right) = U$$

R = radius, T = thickness of circular plate

For the simple case considered here, $\epsilon_r = \epsilon_\theta = \frac{\delta}{R}$

where δ is the stretch of the plate.

Assuming that the failure stress is Y_F, then $\sigma_r \simeq \sigma_\theta = Y_F$ yields

$$\delta = \frac{U}{Y_F \, \pi \, R \, T}$$

Since

$$\delta = [R^2 + h^2]^{1/2} - R = \frac{h^2}{2R}$$

an estimate of h is

$$h^2 = \frac{2\,U}{Y_F \, \pi \, T}$$

In terms of VF

$$h^2 = \frac{W}{g} \; \frac{VF^2}{Y_F \, \pi \, T}$$

Using the conservative estimate leads to the conclusion that the baseplate will contain the drop with the possibility of some local baseplate to cell weld damage occurring adjacent to the cell in question. This does not affect the ability of the rack to withstand any concurrent seismic loadings.

$h = 2.752"$ if $Y_F = \sigma_Y$

IV. OTHER ACCIDENT CONSIDERATIONS

2. Provide information on the procedures for removing the existing racks and installing new racks including the possibility of rack drop on the pool floor or a wall.

RESPONSE

The removal of old racks and installation of new racks will be carried out using written procedures which will be reviewed and approved in accordance with the Indian Point 2 review process before use. A list of activities that will be covered by procedures is provided below along with brief explanatory notes, where necessary, for clarification.

(i) Receipt Inspection Procedure:

Includes receipt inspection of transit damage, dummy gage test and dimensional overchecks.

(ii) Horizontal Lift and Upending of Racks.

(iii) Vertical Lift and Preliminary Leveling of Racks.

(iv) Purpose and Scope of Removal of Existing Racks and Installation of New Racks.

(v) New Racks Installation

This procedure will contain the following information:

- o Materials and equipment
- o Safe rigging practice
- o NUREG 0612 requirements
- o Load travel path well defined
  Note: The path specified will preclude movement of racks over fuel assemblies at any time.
- o Sketches of lifting fixtures
- o Sketches of remote tooling
- o QA hold points
- o Fuel shuffles

(vi) New Rack Leveling

(vii) Underwater Diving (if necessary)

(viii) Vacuum Box Testing for Leak Detection Procedure

(ix) Underwater Vacuum Cleaning

(x) Site Free Path Gauge Test

(xi) Cell Rework

The detailed procedures for the above activities are currently under preparation and review by Con Edison. The removal and installation of the racks involves a carefully planned sequence of fuel assembly relocation followed by old rack removal and new rack placement. Figures 1 through 6 show the sequence of rack regions occupied by fuel, racks removed and racks installed in the pool. Old racks are indicated by numerals 1 through 12. New racks are designated by alphanumeric identifications used in the Licensing Report. The dimension x (with subscripts where necessary) indicates the shortest distance between the fuel and the new racks at each reracking stage. These figures provide proposed fuel shuffles which meet the objective of maintaining a minimum distance of four feet between stored fuel and a rack being installed. The final reshuffling plan may differ somewhat from Figure 1 through 6, but the objective of maintaining the minimum distance of four feet will be met.

In the unlikely event that a rack was dropped, a thorough inspection of the affected area including but not limited to the pool liner, potentially damaged racks, and the spent fuel cooling system would be done with remote inspection equipment such as underwater cameras. If necessary, divers would be utilized to augment and/or verify the remote inspection. Based on the inspection results an action plan and associated procedures would be developed to correct any deficiencies identified.

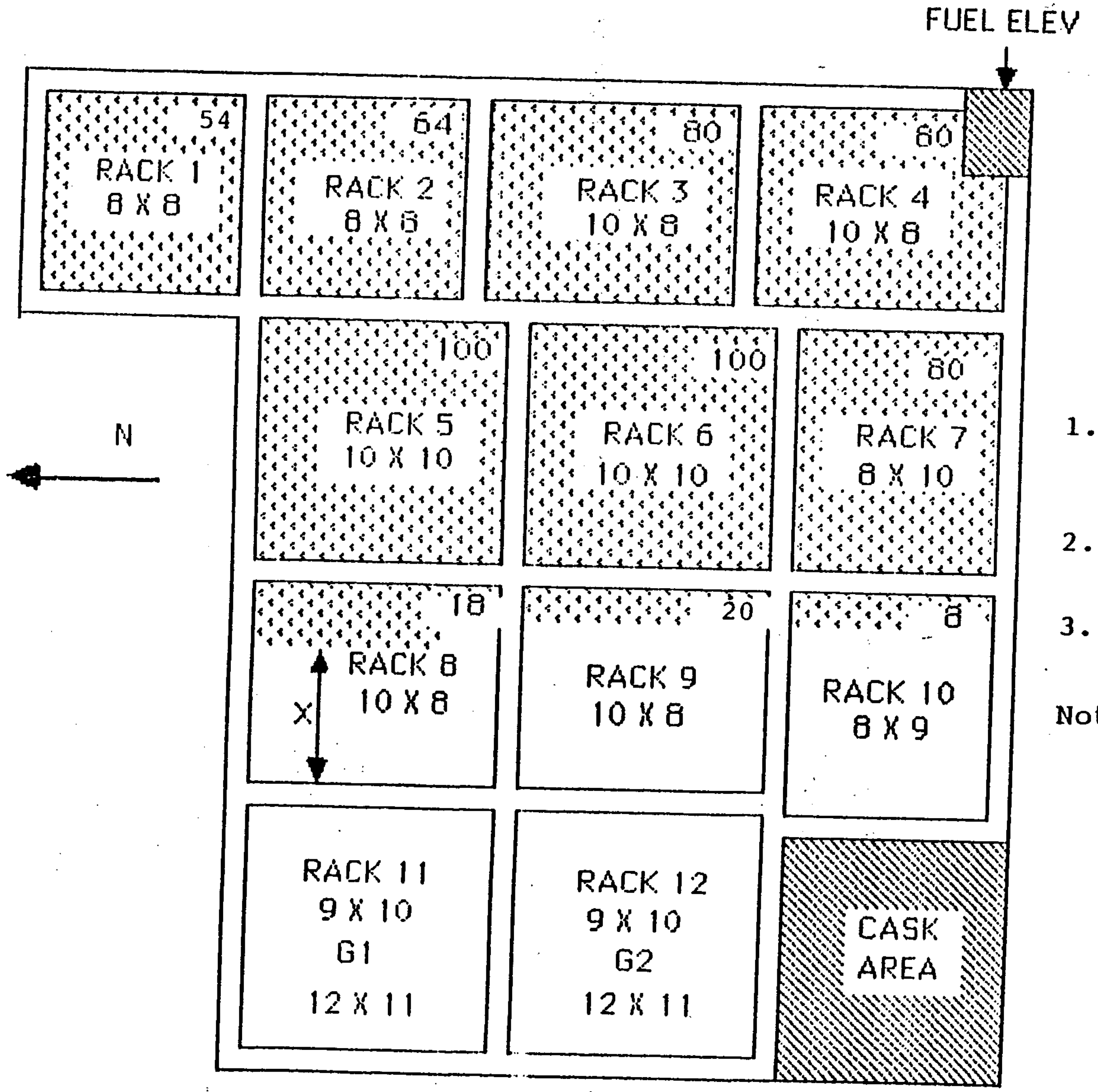

FIGURE 1

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle all fuel assemblies to locate them as shown in this Figure.

2. Remove Racks 11 and 12 for disposal.

3. Install Racks G1 and G2.

Note: 5' < x '

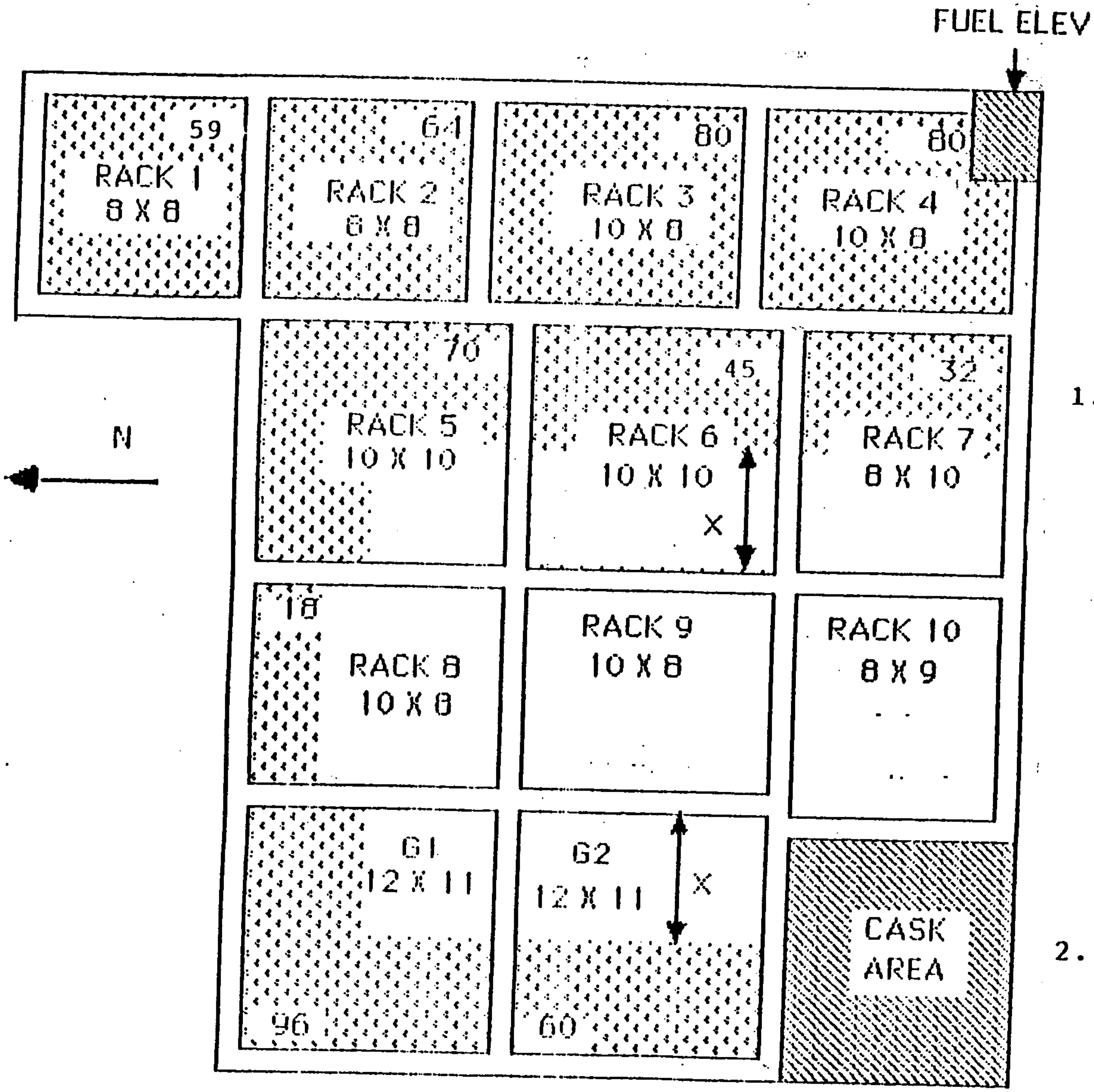

FIGURE 2

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle fuel to obtain the configuration shown in this Figure. The following net movements from racks are involved.

Rack ID	Assemblies Added (+) or Removed (-)
1	+5
5	-30
6	-55
7	-48
9	-20
10	- 8
G1	+96
G2	+60

2. Remove Racks 9 and 10.

Note: $4' < x$

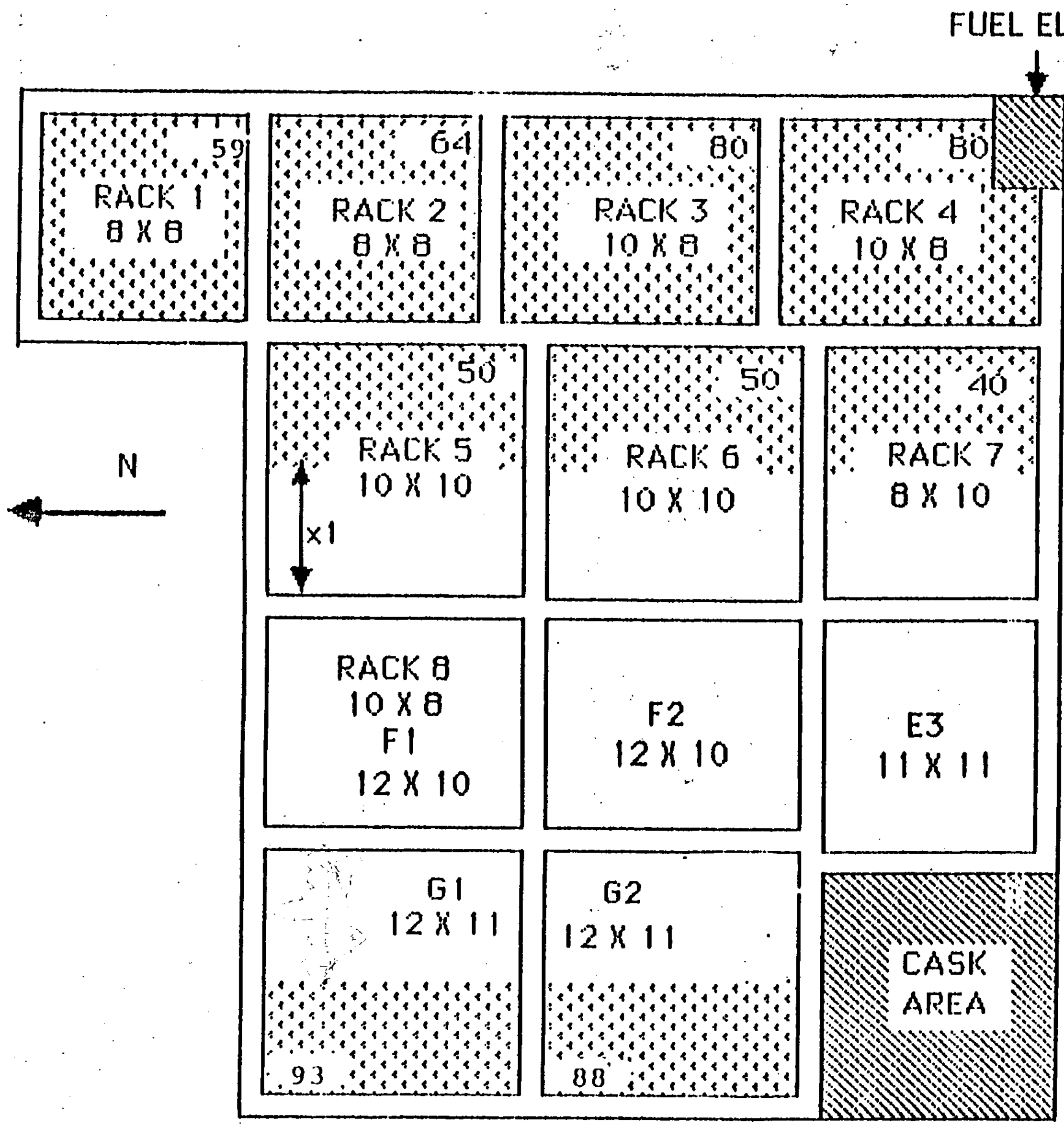

FIGURE 3

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle and move fuel to obtain the storage configuration shown in this Figure. The following net movements from racks are involved.

Rack ID	No. of Assemblies Added (+) or Removed (-)
5	-20
6	+5
7	+8
8	-18
G1	-3
G2	+28

2. Remove Rack 8 for disposal.

3. Install Rack F1

4. Install Racks F2 and E3.

Note: 4' < x1

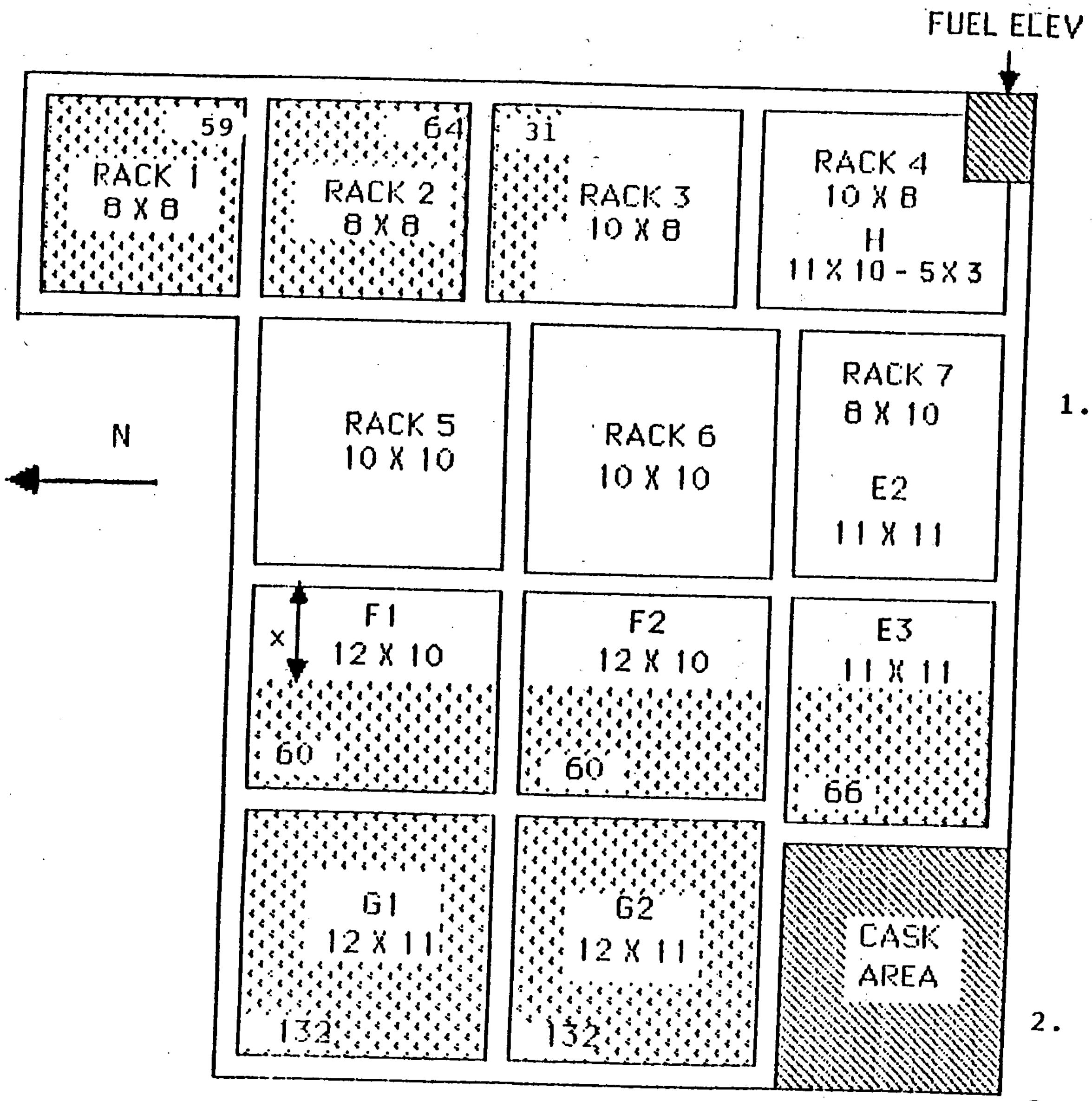

FIGURE 4

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle fuel to obtain the storage configuration shown in this Figure. The following net movements from racks are involved.

Rack ID	No. of Assemblies Added (+) or Removed (-)
3	-49
4	-80
5	-50
6	-50
7	-40
F1	+60
F2	+60
E3	+66
G1	+39
G2	+44

2. Remove Racks 4 and 7 for disposal.

3. Install Racks E2 and H.

Note: 4' < x

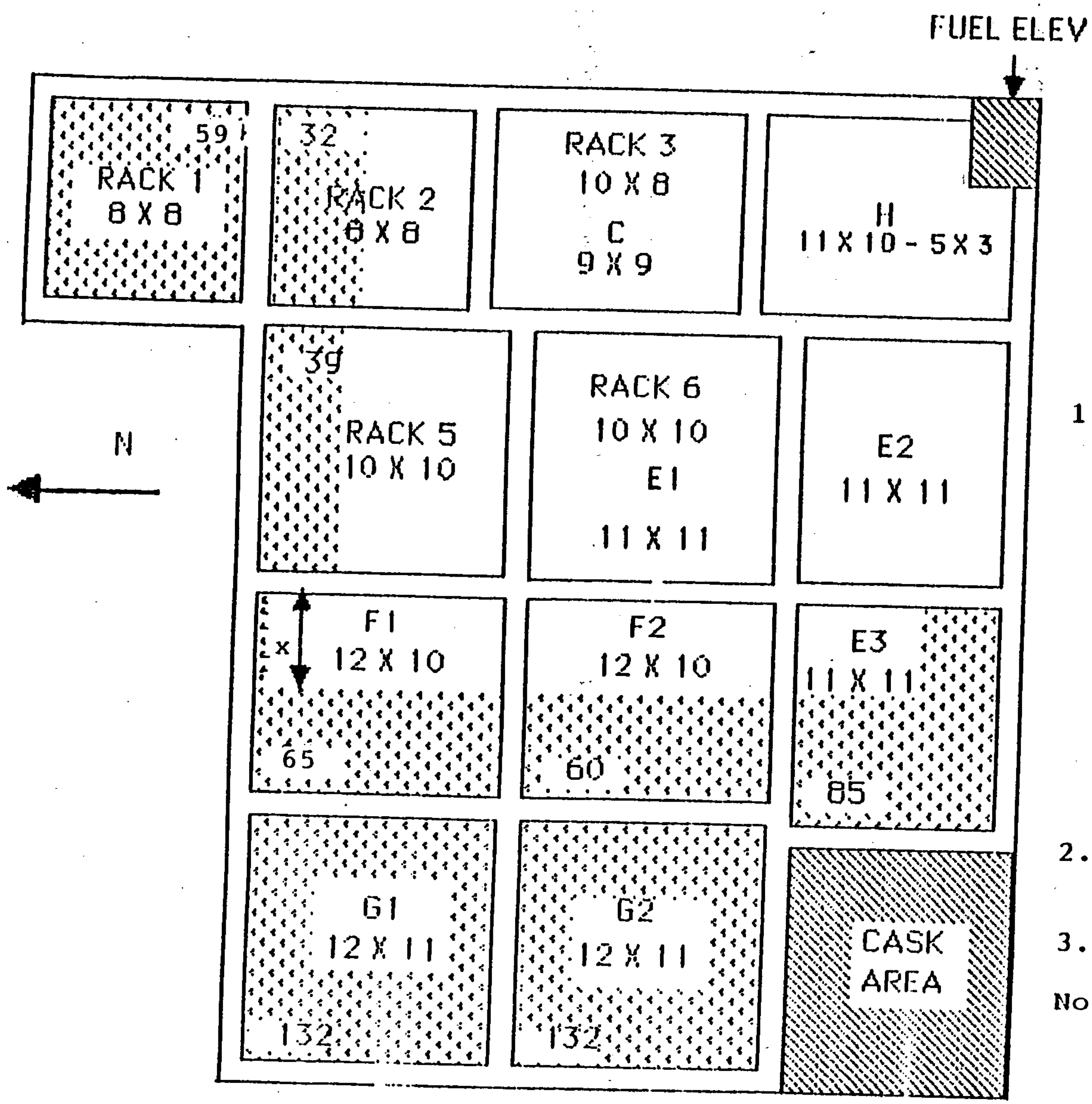

FIGURE 5

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle fuel to arrange them in the configuration shown in this Figure. Required assembly movements tabulated below.

Rack ID	No. of Assemblies Added (+) or Removed (-)
2	-32
3	-31
5	+39
E3	+19
F1	+5

2. Remove Racks 6 and 3 for disposal.

3. Install Racks E1 and C.

Note: 4′ < x

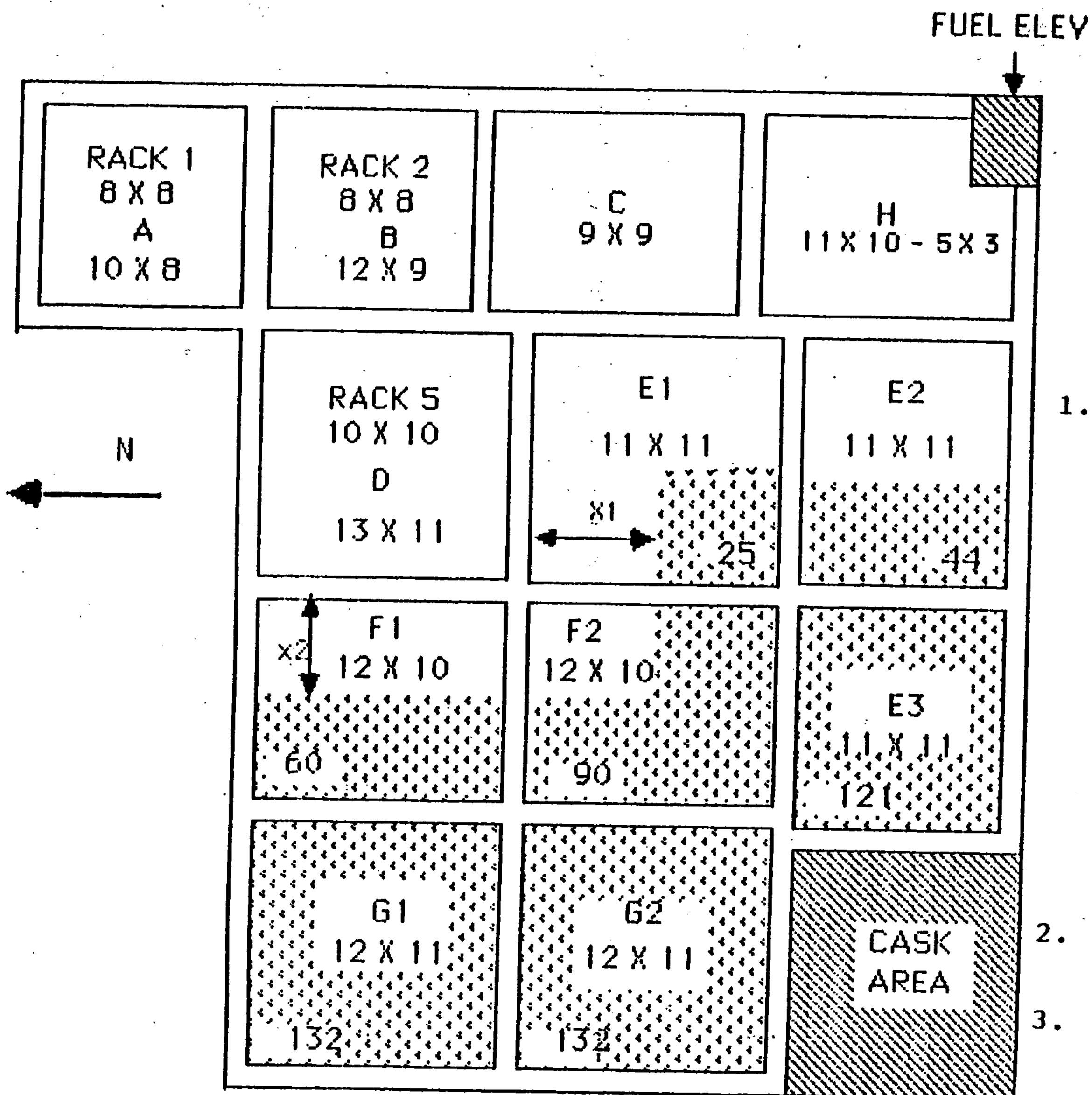

FIGURE 6

INDIAN POINT UNIT II

INSTALLATION AND REMOVAL SEQUENCE AND PROPOSED FUEL SHUFFLE STEPS:

1. Shuffle fuel to arrange them in the configuration shown in this Figure. Required assembly movements tabulated below.

Rack ID	No. of Assemblies Added (+) or Removed (-)
1	-59
2	-32
5	-39
E1	+25
E2	+44
F2	+30
E3	+36
F1	-5

2. Remove Racks 5 and 2 and Rack 1 for disposal.

3. Install Racks D, B and A.

Note: 4' < x1
4' < x2

V. MISCELLANEOUS ITEMS

1. Provide details of the proposed installation procedures indicating how the elevations of the racks and designated gaps between the racks will be maintained and monitored.

RESPONSE

A brief description of the procedure indicating how control of rack elevations and inter-rack gaps will be realized during installation is given below:

a. Equipment Required:

- o 50′ transit pole
- o Optical level
- o Shim plates
- o Hydraulic jacks
- o Stainless steel shims

b. Floor Elevation Readings

- o Using 50′ transit pole and optical level record the elevation readings of the locations of the four-corner shim plates where the rack is to be installed.

c. Rack Installation and Leveling in Spent Fuel Pool

- o Prior to lowering rack into pool, adjust four-corner feet to account for the differences in elevations of the four-corner shim plates.

- o Set the new rack in its designated location. Using leveling tool, ensure all four-corner feet are in contact with shim plates.

- o Record rack height elevations using optical level and 35′ transit pole in each of the four corners.

- o If the differences in elevation are within $\pm$1/16" then the rack is acceptable.

- o If the differences are not within $\pm$1/16" using hydraulic jacks, raise rack just enough to make the proper amount of turns for the necessary adjustments to bring elevation differences to within the acceptable tolerance $\pm$1/16".

- o Check all four-corner feet with leveling tool to ensure they are still in contact with the shim plates.

d. Rack Position

- The rack is placed on the pool bearing pad locations as illustrated in Drawing 531 provided in response to question I.3. Minor adjustment of the rack location may be required to satisfy the inter-rack gap and rack-to-wall gap requirements. For this purpose, the "go-no-go" gage blocks are used to determine whether the rack location criterion is satisfied. If necessary, hydraulic jacks along with the crane are used to nudge the bearing pad or the rack to its final designated location.

NOTE: Rack leveling readings and measurements between racks will be verified by Contractor's Q.C. using remote camera and an optical level.

V. MISCELLANEOUS ITEMS

2. Provide a summary of plant safety procedure for the following cases:

   (a) Fuel drop (or rack drop) accident.

   (b) A seismic event.

   (c) Loss-of-water from the pool detected by leak chases.

RESPONSE

All of the procedures referenced below are available at Indian Point 2 for review.

a) In the event of a fuel drop (or rack drop) accident, damage to fuel would be assumed until proven otherwise. Therefore the procedure that would be used for a fuel or rack drop accident would be Abnormal Operating Instruction 17.0.2 entitled "Irradiated Fuel Damage in Fuel Storage Building". This procedure requires evacuation of all personnel if radiation monitor alarms are received. All fuel handling would be suspended and fuel movement would not be permitted until further evaluation is performed and permission is obtained from the Operations Manager. After evaluation by Health Physics personnel, re-entry to the Fuel Storage Building to perform damage assessment will occur.

   Coincident with the above actions the required Emergency Plan actions will be evaluated. The event would be classified using the graded classification system for emergencies and actions would be taken to protect the safety of the public, plant personnel and property both onsite and offsite.

b) In the case of a seismic event, Abnormal Operating Instruction 28.0.8 entitled "Earthquake Emergency" would be utilized. The procedure requires an inspection of plant equipment and structures which includes an inspection of the spent fuel pool for possible water leakage following a seismic event. If the water level in the pool is dropping, actions are to be taken to restore normal level. If the normal makeup system to the pool is unavailable, actions are to be taken to utilize available water sources such as the fire protection system. In addition, this procedure requires notification of Con Edison's Plant Structures Engineer and Field Engineering if the seismic event was greater than 0.10g horizontal or 0.05g vertical or if damage to plant structures has occurred. Once this notification occurs, the Structures Engineer and Field Engineering must recommend that further action is not necessary or specify repair procedures for damaged equipment or structures. The spent fuel pool and the racks are designed for a design basis seismic I earthquake (also called a safe shutdown earthquake) which is a 0.15g horizontal and 0.10 g vertical seismic event. Therefore, the condition of the spent fuel pool and racks after a safe shutdown earthquake or less would be within the analysis for safe storage of fuel in the storage racks. This procedure requires Engineering evaluation at a much lower level of seismic event than the pool and racks are designed to withstand. Therefore, this procedure addresses seismic events of concern to the spent fuel pool and racks.

In addition to the procedure discussed above, the severity of the seismic event would be evaluated for activation of the Emergency Plan if necessary.

c) Attachment II contains page revisions to the Consolidated Edison June 20, 1989 request for a license amendment to expand spent fuel storage. As discussed in Attachment II, revision of page 2-2 of Attachment B of the submittal, the Indian Point 2 spent fuel pool was constructed without a leak chase. Loss-of-water from the spent fuel pool is detected by level instrumentation. The level instrumentation has an alarm in the control room which activates when a variation of $\pm$ 6" from normal level occurs. The Alarm Response Procedure for control room panel SGF window 2-2 is for spent fuel pool level. When a spent fuel pool level alarm is received, a direct visual observation of the spent fuel pool level is required. If the water level is low, restoration of normal level using the makeup system is required. The procedure provides two alternative makeup water sources in the event the first choice is not available. The procedure directs that an investigation be initiated to determine the cause of the low water level which includes: refueling cavity leakage, spent fuel pool and purification piping leakage, spent fuel pool building foundations leakage, evaporation, and spent fuel pool cooling system line-up.

ATTACHMENT II

PAGE REVISIONS TO LICENSE AMENDMENT
REQUEST FOR INCREASE IN SPENT FUEL STORAGE CAPACITY
DATED JUNE 20, 1989

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
JANUARY, 1990

Summary of Page Changes

1. Page 2-2

   This revision deletes the reference to a leak chase in the spent fuel pool. The Indian Point 2 spent fuel pool was built and licensed without a leak chase.

2. Page 3-8

   This revision provides the materials for the support leg and failed fuel canister that are now going to be used in rack fabrication.

3. Page 3-13, Figure 3.5

   This revision provides the updated dimensions for the adjustable support.

The racks will be arranged in two regions in the spent fuel pool. Region I will have 269 locations capable of storing unirradiated fuel of up to 5.0 wt% U-235 initial enrichment. Region I has enough locations to store a full core discharge and one-third core of unirradiated fuel. Region II will have 1105 locations for storage of fuel which meets enrichment and burnup criteria developed as part of the rack design. Section 4 of this report addresses this in more detail. In addition, there are two locations for storage of failed fuel canisters. The total number of storage locations, as detailed above, is 1376.

Table 2.3 gives the essential storage cell data for all racks. As noted, the storage cells are 8.75" (internal dimension) for Region I and 8.80" for Region II which accommodates the standard Westinghouse fuel assembly or equivalent fuel.

The module's four support legs are remotely adjustable. Thus, the racks can be made vertical and the top of the racks can easily be made co-planar with each other. The rack module support legs are engineered to accommodate variations of the pool floor. The placement of the racks in the spent fuel pool has been designed to preclude any support legs from being located on the liner welds. Support pads have been provided to bridge any obstructions which could potentially interfere with placement of a rack support leg.

2.1.2 Poison Material

Boraflex has been selected as the neutron absorber material for the new high density spent fuel storage racks.

e. Other References

(1) NRC Regulatory Guides 1.13, Rev. 2 (proposed); 1.29, Rev. 3; 1.31, Rev. 3; 1.61, Rev. 0; 1.71, Rev. 0; 1.85, Rev. 22; 1.92, Rev. 1; 1.124, Rev. 1; and 3.41, Rev. 1.

(2) General Design Criteria for Nuclear Power Plants, Code of Federal Regulations, Title 10, Part 50, Appendix A (GDC Nos. 1, 2, 61, 62, and 63).

(3) NUREG-0800, Standard Review Plan, Sections 3.2.1, 3.2.2, 3.7.1, 3.7.2, 3.7.3, 3.8.4.

(4) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and the modifications to this document of January 18, 1979.

3.5 MATERIALS OF CONSTRUCTION

Storage Cell:	SA240-304
Baseplate:	SA240-304
Support Leg:	SA479-304
Support Leg (male):	Ferritic stainless (anti-galling material) SA564-630
Poison:	Boraflex
Failed Fuel Canister	SA312-304

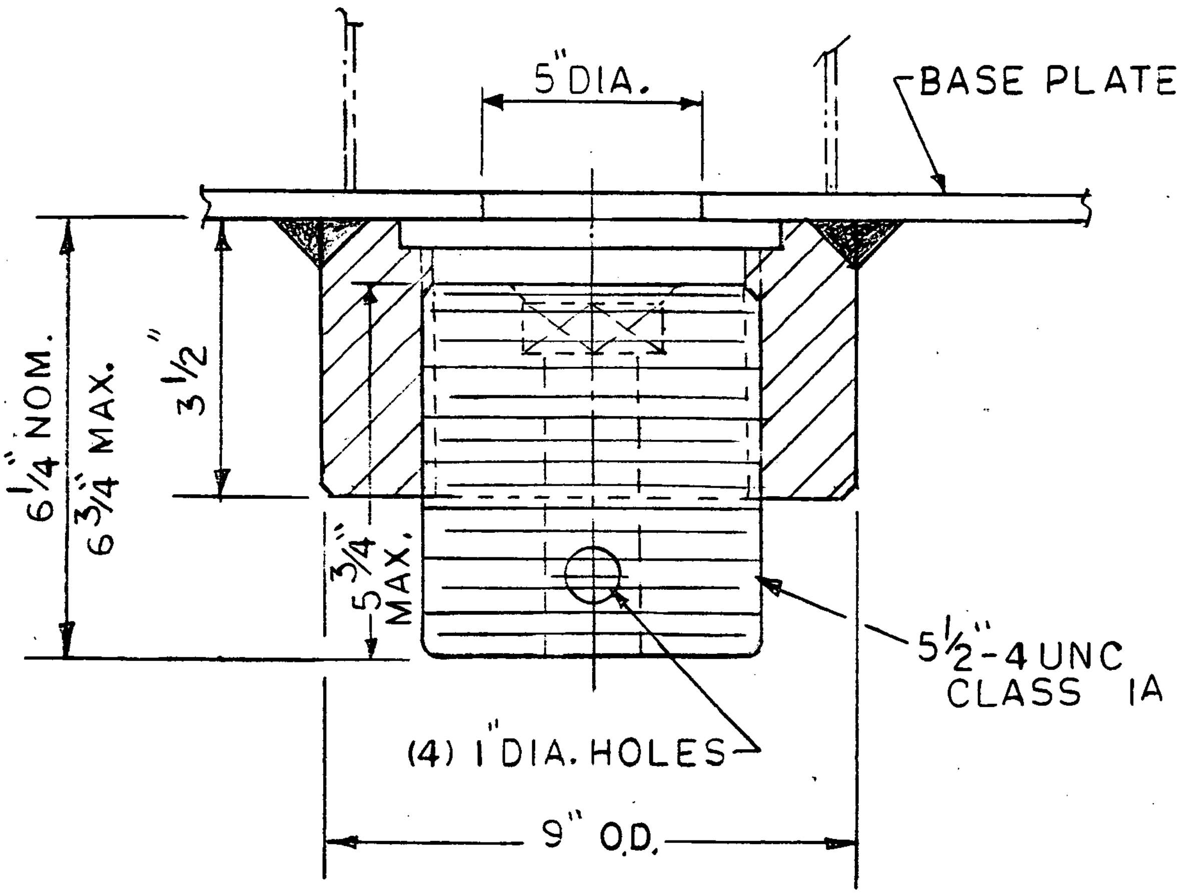

FIGURE 3.5 ADJUSTABLE SUPP'T.

ATTACHMENT III

REPORT ON ANALYSIS OF AN ISOLATED
5 W/O FUEL ASSEMBLY IN WATER

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
JANUARY, 1990

I. INTRODUCTION

After completion of the analysis for the licensing report in Attachment B to the June 20, 1989 letter to the NRC requesting a license amendment to modify spent fuel storage requirements, an additional case involving a 5 w/o fuel assembly was identified as requiring further analysis. This case involved the keff, including uncertainties, of a 5 w/o fuel assembly in pure water in the spent fuel pool when not located in a storage rack. Upon review of other spent fuel storage modifications at other facilities, it became apparent that this case had not been addressed before. Therefore, Con Edison proceeded with the analysis for the fuel used at Indian Point 2, Westinghouse 15x15. This report provides the analytical methodology and results and the subsequent conclusions.

II. ANALYTICAL METHODOLOGY

A. Reference Fuel Assembly

The design basis fuel assembly is a 15x15 array of fuel rods with 21 rods replaced by 20 control rod guide tubes and 1 instrument thimble. Table 1 summarizes the design specifications and the expected range of significant variations. The fuel assembly grid spacers and miscellaneous hardware were neglected and are considered to have only a minor and conservative effect on reactivity.

B. Calculational Models

The primary criticality analyses were performed with a two-dimensional multi-group transport theory technique, using the CASMO-2E(1) computer code. Independent verification calculations were made with a Monte Carlo technique utilizing the AMPX-KENO IV computer package(2), with the 27-group SCALE* cross-section library(3) and the NITAWL subroutine for U-238 resonance shielding effects (Nordheim integral treatment). These codes have previously been benchmarked and determined to have a bias of 0.0013 with an uncertainty of ± 0.0018 for CASMO-2E and 0.0106 ± 0.0048 (95%/95%) for NITAWL-KENO. In addition, a check calculation was run with KENO Va to independently confirm the KENO IV calculation.

Casmo-2E was also used to evaluate the reactivity consequences of temperature and the tolerances on fuel density and enrichment.

In the geometric model used in the calculations, each fuel rod and its cladding were described explicitly and reflecting boundary conditions (zero neutron current) were used in the axial direction and at the centerline of the water space between assemblies. The model assumed fuel assemblies on a 21 inch lattice spacing. Diffusion theory calculations (with constants edited from CASMO-2E) confirmed that the model adequately represents an isolated fuel assembly. Because of the high scattering and low absorption crossections of the large volume of water, it was necessary to use a large number of neutron histories (75,000) to obtain acceptable statistics in the KENO calculation.

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*"SCALE" is an acronym for Standardized Computer Analysis for Licensing Evaluation, a standard cross-section set developed by ORNL for the USNRC.

III. ANALYTICAL RESULTS

A. Reference Calculations

Calculations for a single isolated fuel assembly in pure water at 20°C gave the following results:

CODE	CALCULATED k_∞	BIAS CORRECTED k_∞
CASMO-2E	0.9552	0.9565 ± 0.0018
KENO-IV	0.9426 ± 0.0047	0.9532 ± 0.0067
KENO-Va	0.9428 ± 0.0071	0.9534 ± 0.0086

Including the effect of fuel tolerance uncertainties and a small temperature correction, the maximum, k_∞ for both CASMO-2E and KENO IV becomes 0.961 (see Table 2).

B. Tolerance Uncertainties

The reactivity effect of fuel tolerances were determined from differential CASMO-2E calculations. These uncertainties were found to be ± 0.0028 for fuel density (± 2% in density) and ± 0.0011 for fuel enrichment (± 0.05 in % enrichment).

C. Temperature Effect

Calculations were made at several temperatures by CASMO-2E, with the following results:

TEMPERATURE	k_∞
20 °C	0.9552
40 °C	0.9558
65 °C	0.9538

Although the reactivity is nearly insensitive to temperature over the expected range of pool water temperatures, the maximum value at 40 °C was used to determine a small correction (+ 0.0006 k) to the base calculations at 20 °C. Above 40 °C, the temperature coefficient of reactivity is negative and higher temperatures will therefore result in lower reactivities.

IV CONCLUSIONS

Results of the analysis confirm that a single assembly of 5w/o enrichment, immersed in clean unborated water, would exceed a k_{eff} of 0.95 when not in storage. As summarized in Table 2, the maximum calculated reactivity (k_∞) was 0.961, including uncertainties at the 95% probability, 95% confidence level. Independent calculations by CASMO-2E and by KENO (both versions IV and Va) were in agreement and confirmed the maximum k_∞ of 0.961.

Although a k_{eff} of 0.95 is exceeded, no immediate criticality safety concern exists since (1) there is a substantial subcriticality margin (-0.04 Δk) and (2) the soluble boron actually present in the pool water will assure reactivity limits will be met (a concentration of only 100 ppm boron is estimated to be adequate to reduce the reactivity below 0.95).

The proposed Indian Point 2 Technical Specification page revisions that were submitted to the NRC in Attachment A to the June 20, 1989 letter from Con Edison contain a requirement for a minimum boron concentration in the spent fuel pool at all times. Proposed Technical Specification 3.8.D.2 states "At all times the spent fuel storage pit boron concentration shall be at least 1500 ppm." The required 1500 ppm far exceeds the approximately 100 ppm required to reduce the keff to less than 0.95. Therefore, with the proposed Technical Specification 3.8.D.2 in effect, the reactivity of the spent fuel pool for this case will be well below a k_{eff} of 0.95.

V REFERENCES

1. A. Ahlin, M. Edenius, H. Haggblom, "CASMO - A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (proprietary).

   A. Ahlin and M. Edenius, "CASMO - A Fast Transport Theory Depletion Code for LWR Analysis," ANS Transactions, Vol. 26, p. 604, 1977.

   M. Edenius et al., "CASMO Benchmark Report," Studsvik/ RF-78-6293, Aktiebolaget Atomenergi, March 1978.

2. Green, Lucious, Petrie, Ford, White, Wright, "PSR-63/AMPX-1 (code package), AMPX Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B", ORNL-TM-3706, Oak Ridge National Laboratory, March 1976.

3. R.M. Westfall et al., "SCALE: A Modular Code System for performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, 1979.

Table 1
DESIGN BASIS FUEL ASSEMBLY SPECIFICATIONS

FUEL ROD DATA

Outside diameter, in.	0.422
Cladding thickness, in.	0.0243
Cladding inside diameter, in.	0.3734
Cladding material	Zr-4
Pellet density, % T.D.	95
Pellet diameter, in.	0.3659
Maximum enrichment, wt % U-235	5.00 ± 0.05
Maximum stack density, g UO_2/cc	10.31 ± 0.21

FUEL ASSEMBLY DATA

Fuel rod array	15x15
Number of fuel rods	204
Fuel rod pitch, in.	0.563
Number of control rod guide and instrument thimbles	21
Thimble O.D., in. (nominal)	0.546
Thimble I.D., in. (nominal)	0.512

Table 2

CRITICALITY ANALYSIS

OF AN ISOLATED FUEL ASSEMBLY IN WATER

	CASMO-2E	AMPX-KENO IV
Fuel Enrichment, wt% U-235	5	5
Temperature for analysis	20°C (68°F)	20°C (68°F)
Calculated k	0.9552	0.9426
Temperature correction (40°C)	+0.0006	+0.0006
Calculational bias, k	0.0013	0.0106
Sum	0.9571	0.9538
Uncertainties		
Bias	±0.0018	±0.0048
Monte Carlo Statistics	NA	±0.0047
Fuel enrichment	±0.0011	±0.0011
Fuel density	±0.0028	±0.0028
Statistical combination of uncertainties	±0.0035	±0.0074
Reference k	0.9571 ± 0.0035	0.9538 ± 0.0074
Maximum Reactivity (k)	0.9606	0.9612

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