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ANALYSIS OF THE VESSEL MATERIAL SURVEILLANCE CAPSULES WITHDRAWN FROM LACROSSE BOILING WATER REACTOR DURING THE 1980 REFUELLING

by

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ABSTRACT

The third set of reactor vessel surveillance capsules was removed from the LaCrosse Boiling Water Reactor during the 1980 refuelling outage. The neutron flux results and the neutron embrittlement responses of the surveillance materials, although in good agreement with data from previous analyses, did indicate that the rate of embrittlement is less than previously projected. A revised reference transition temperature vs power generation curve was prepared.

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I. SUMMARY OF RESULTS

The analysis of the data obtained from the vessel material surveillance capsules removed from the LaCrosse Boiling Water Reactor (LACBWR) vessel led to the following conclusions.

1. The irradiated properties of the LACBWR primary pressure boundary materials appear to be adequate to provide continued safe operation of the plant according to present day criteria.

2. The LACBWR vessel plate NP1056 is predicted to control the reference transition temperature (RT_{NDT}) for approximately 17 effective full power years (EFPY) of operation. The projected peak value of RT_{NDT} at the vessel I.D. for the refuelling outages scheduled through 1986 (based on 40,000 MWD_t power generation per year) are as follows.

Refuelling Year	Total Power (MWD_t)	Effective Full Power Years	RT_{NDT} (deg F)
1981	378,000	6.3	125
1982	418,000	7.0	128
1983	458,000	7.6	131
1984	498,000	8.3	134
1985	538,000	9.0	137
1986	578,000	9.6	140

3. The LACBWR weld metal is predicted to control the RT_{NDT} of the primary system after 17 EFPY of operation. At a projected peak vessel fluence of 2.7×10^{19} n/cm^2 (E > 1 MeV) after 20 EFPY of operation, the shift in RT_{NDT}, as controlled by the weld metal, is predicted to be 175°F. Since the initial RT_{NDT} of the weld metal has been taken to be 0°F, the value of RT_{NDT} for the primary pressure system after 20 EFPY of operation is projected to be 175°F.

4. The above projections are based on ΔRT_{NDT} values determined at the 30 ft-lb level and on the results of a two-dimensional discrete ordinates transport calculation of the energy and spatial distribution of the neutron flux between the reactor core and the vessel. The transport analysis showed that assuming 40 percent voids in the steam separators would yield lead factors (ratio of capsule neutron flux to vessel neutron flux) that provide conservative values of neutron flux incident on the LACBWR pressure vessel wall. Based on an analysis of the dosimetry results from the ten specimen capsules and two vessel wall dosimeters removed to date, the LACBWR vessel is projected to receive a peak fast fluence (E > 1 MeV) of 1.35×10^{18} n/cm^2 each EFPY.

5. At a projected peak vessel fluence of 2.7×10^{19} n/cm^2 (E > 1 MeV) after 20 EFPY of operation, the Charpy shelf energy of the vessel weld metal is predicted to be reduced to 50 ft-lb. The Charpy shelf energies of the vessel beltline plates are predicted to range from 54 to 64 ft-lb.

6. The values of RT_{NDT} and toughness at the 1/4 thickness location in the vessel wall are substantially better than those summarized above because the fast neutron flux and fluence at the 1/4 T is 80 percent of that at the vessel I.D. surface.

II. BACKGROUND

For many years, the basis for defining a minimum safe operating temperature of a pressure system had been the Fracture Analysis Diagram (FAD) developed by Pellini and Puzak.[1]* The FAD is keyed to the drop-weight nil-ductility transition (DW-NDT) temperature defined by ASTM Method of Test E 208.[2] The Fracture Transition Elastic (FTE) temperature, above which stresses in excess of yield are required to propagate a large flaw, is indexed at DW-NDT + 60°F.

Until recently, Section III of the ASME Boiler and Pressure Vessel Code had defined the minimum permissible pressurization temperature as 60°F above the higher of (1) the DW-NDT temperature and (2) the minimum temperature at which a set of three Charpy V-notch specimens, representing weld metal and heat-affected zone as well as base material, meet the fracture energy requirements specified by the Code for the particular material. The Charpy V-notch requirements ranged from 15 ft-lb minimum for steels having a specified minimum yield strength less than 35,000 psi to 35 ft-lb minimum for steels having a specified minimum yield strength of 75,000 psi and above.

Currently, the allowable loadings on nuclear pressure vessels are determined by applying the rules in Appendix G, "Fracture Toughness Requirements," of 10CFR50.[3] 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.[4] The RT_{NDT} is defined by Section III of the ASME Code as the highest of the following temperatures:

1. Drop-Weight Nil Ductility Temperature (DW-NDT) per ASTM E 208;[2]

2. 60°F below the 50 ft-lb Charpy V-notch (C_V) temperature; and

3. 60°F below the 35 mil C_V temperature

The initial 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).[5] In addition to a general dependence on neutron fluence, it has been established that tramp elements, particularly copper and phosphorous, affect the radiation embrittlement response of ferritic materials.[6-8]

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* Superscript numbers refer to references listed at the end of the text.

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 parent materials and welds which are located in the core beltline region of the reactor pressure vessel. As a consequence, one or more heats of these ferritic materials must be monitored for radiation-induced changes in RT_{NDT} per the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10CFR50[3] and ASTM E 185[9] which describe the current recommended practice for monitoring the radiation-induced changes occurring in the mechanical properties of ferritic pressure vessel materials.

Allis-Chalmers provided such a surveillance program for the LaCrosse Boiling Water Reactor (LACBWR). The encapsulated C_v specimens are located on the I.D. surface of the thermal shield where the fast neutron flux density is approximately twice that at the adjacent vessel wall surface. Therefore, the increases (shifts) in transition temperatures of the materials in the pressure vessel are generally less than the corresponding shifts observed in the surveillance specimens. However, because of azimuthal variations in neutron flux density, the capsule fluences lead the maximum vessel fluence by varying amounts.

This report describes the results obtained from testing and evaluating the capsules removed during the 1980 refuelling outage at LACBWR. The results obtained from capsules removed during the 1972 and 1975 refuelling outages were reported earlier[10,11], but have been reevaluated as described in Section IV of this report.

III. LACBWR SURVEILLANCE PROGRAM

The LACBWR surveillance program is described in detail in ACNP-66513.(12) Twenty test specimen capsules (10 Type A and 10 Type B) were placed around the periphery of the core on the inner surface of the thermal shield as shown in Figure 1. The capsule supports are located in such a way as to center the capsules axially about the horizontal midplane of the core. In addition, two vessel wall dosimeter capsules (which do not contain mechanical property test specimens) were placed between the vessel wall and the thermal shield to assist in the determination of the acceleration factor for exposure between the test specimen capsule and the vessel wall locations. Capsules 1A and 1B had been removed during the 1972 outage, and the results have been previously reported.(10) Capsules 2A, 7B, 9A, and 9B, along with the two vessel wall dosimeter capsules, were removed during the 1975 refuelling outage, and these results have also been reported.(11) This report covers the testing of specimens from Capsules 3A, 3B, 8A, and 8B, removed during the 1980 refuelling outage, and an analysis of all LACBWR surveillance data.

A. Test Materials and Specimens

Each radiation capsule contains 22 Charpy V-notch specimens machined from the vessel beltline materials, 6 Charpy V-notch specimens machined from a "standard" material, and 6 miniature tensile bars machined from one vessel beltline plate. The vessel beltline materials include 3 ASTM A 302 Grade B vessel plates (NP1054, NP1055, NP1056) and weld metal. The ASTM A 302 Grade B "standard" material, furnished by the Atomic Energy Commission, Chicago Operations Office (AEC-COO), was characterized by Battelle Northwest Laboratories. The available data on chemistries and heat treatments of these materials are given in Table I.

The NP1054 Charpy V-notch specimens were machined from 18 tested DW-NDT specimens. The NP1055 tensile specimens and the NP1055 and NP1056 Charpy V-notch specimens were machined from excess plate material. All specimens were oriented parallel to the rolling direction and were located at the upper or lower quarter-plate thickness. The impact specimen notches were oriented perpendicular to the plate surface. Drawings showing the location of specimens within the sample plates and specimen machining drawings are given in Appendix A.

B. Capsule Design and Loading Arrangements

The radiation capsules are 24 in. long and were fabricated from 1-1/4-in. Schedule 80 TP304 stainless steel pipe with welded closure plugs at both ends. The top closure plug, fitted with a cable and lifting ring assembly, was installed after filling the capsules as described below.

All specimens were cleaned in acetone, arranged in a clean box in the proper groups for each capsule, then bundled and wired. The flux wires and temperature indicators described in Section III.C were also inserted in the

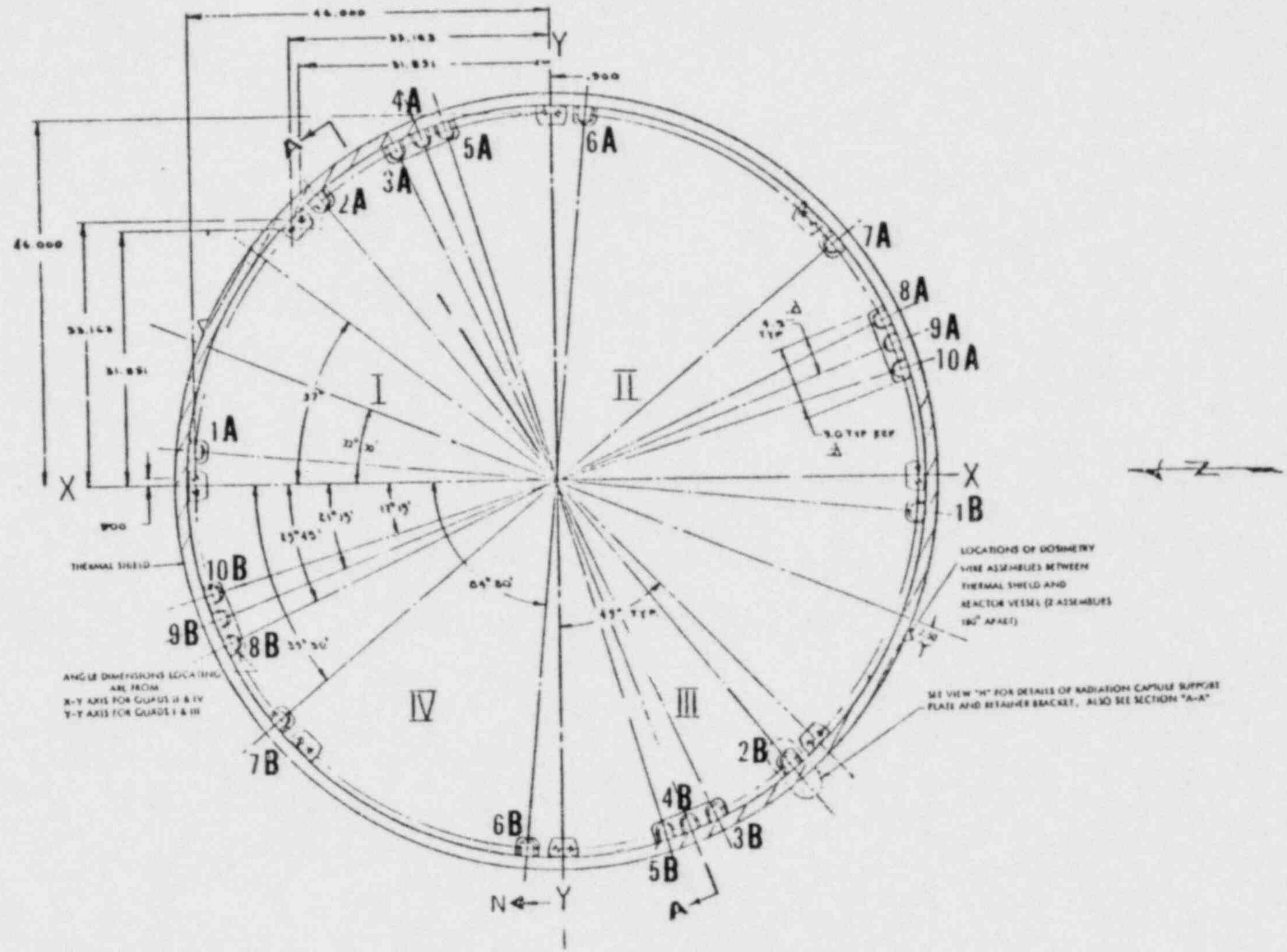

FIGURE 1. INTERNAL THERMAL SHIELD SHOWING LOCATIONS OF 20 RADIATION CAPSULES AND 2 DOSIMETRY WIRE HOLDERS(12)

TABLE I

CHARACTERIZATION DATA ON LACBWR SURVEILLANCE MATERIALS

A. Chemistries

Mat'l Ident.	Heat No.	Data Source	Chemical Analysis (%) C	Mn	P	S	Si	Cu	Mo
NP1054	A5848*	(17)	0.19	1.25	0.009	0.016	0.20	-	0.47
NP1055	A5848*	(17)	0.19	1.25	0.009	0.016	0.20	-	0.47
NP1056	A5852*	(17)	0.20	1.30	0.008	0.022	0.20	-	0.47
NP1056	A5852*	(14)	0.22	1.35	0.007	0.018	0.22	0.11	0.52
Weld	-	(14)	0.10	1.39	0.016	0.006	0.43	0.18	0.55
Weld	-	(19)	-	-	-	-	-	0.15	-
Standard	N31438†	(18)	0.22	1.33	0.017	0.013	0.30	0.07	0.52

B. Mechanical Properties at RT

Mat'l Ident.	Data Source	Tensile (ksi)	Yield (ksi)	Elong. in 2 in. (%)	Charpy V-Notch at 10°F(ft-lb)
NP1054	(17)	82.8	60.6	31	91, 94, 77
NP1055	(17)	86.5	57.5	30	100, 98, 90
NP1055	(12)	87.2	64.1	28	92, 76
NP1056	(17)	83.5	57.5	31	89, 90, 82
Standard	(18)	97.0	76.9	25	60, 58

C. Heat Treatment

LACBWR plates and tests were annealed at 1950-2050°F, then heated to 1725-1775°F, held 1 hour per inch min. and water spray quenched to 500°F, then tempered at 1200-1250°F air cooled; tests were stress relieved at 1100-1150°F (held 2 hours min.). Standard material plate was charged into a 1100°F furnace, heated to 1650°F, held 4 hours, water quenched to below 300°F, recharged into a 750°F furnace, heated to 1200°F, held 4 hours and air cooled.

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* Lukens Steel Company.
† U.S. Steel Corporation.

Charpy V-notch specimen assemblies at this time. Aluminum powder was packed to a depth of 0.88 in. in the bottom of each capsule, the capsules were purged with argon gas, the Charpy V-notch specimen assemblies were inserted, and more aluminum powder was packed around the Charpy V-notch specimen bundles. After the tensile specimen assemblies were inserted, all void space was packed with aluminum powder, and the capsules were weighed to assure that each contained approximately the same quantity of powder. The closure welds were made, then each capsule was subjected to a liquid penetrant inspection. A drawing illustrating a typical capsule arrangement is included in Appendix A. A summary of the test specimen contents and the current removal schedule of each capsule is given in Table II.

C. Flux Wires and Temperature Indicators

Three types of flux wires are contained in the surveillance capsules. These are 0.021-in. diameter pure iron wire, 0.020-in. diameter pure nickel wire, and 0.020-in. diameter aluminum/0.1% cobalt wire. One piece of each wire, approximately 1-1/2 in. long, is located in the V-notch area of each of the seven layers of Charpy specimens in the test specimen capsules.

Two vessel wall flux wire assemblies, each containing iron, nickel, and aluminum/0.1% cobalt wires, were placed 180° apart in the annulus between the vessel wall and the thermal shield. These assemblies, which extend the length of the core, were also fitted with a cable and lifting ring. A drawing of the assembly is included in Appendix A.

Four Charpy V-notch specimens in each capsule contain low melting point eutectic alloys inserted in a hole drilled in the end of the specimens. The four eutectic alloys and their melting points are:

Alloy	Melting Point, °F
2.6 As, 97.4 Pb	554
2.5 Ag, 97.5 Pb	579
0.5 Zn, 99.5 Pb	604
Pure Pb	621

D. Impact and Tensile Properties of Unirradiated Materials

The tensile and impact properties of the LACBWR surveillance materials in the unirradiated condition have been reported previously.(13) The detailed test data are presented in Tables B-1 through B-5 in Appendix B. A summary of the initial DW-NDT temperatures and 30 ft-lb Charpy V-notch "fix" transition temperatures are given in Table III.

TABLE II

LACBWR SURVEILLANCE CAPSULE CONTENTS AND REMOVAL SCHEDULE

Removal Schedule (full power years*)	Surveillance Capsules		Charpy V-Notch Impact Specimens					Tensile Specimens	
	Type	Quantity	NP1054	NP1055	NP1056	Weld	Standard	NP1055	Weld
1.4†	A	1	6	10	-	6	6	3	3
	B	1	-	10	6	6	6	3	3
2.5††	A	2	12	20	-	12	12	6	6
	B	2	-	20	12	12	12	6	6
	VW**	2	-	-	-	-	-	-	-
6†††	A	2	12	20	-	12	12	6	6
	B	2	-	20	12	12	12	6	6
10	A	2	12	20	-	12	12	6	6
	B	2	-	20	12	12	12	6	6
15	A	2	12	20	-	12	12	6	6
	B	2	-	20	12	12	12	6	6
Standby	A	1	6	10		6	6	3	3
	B	1	-	10	6	6	6	3	3

* One full power year equals 60,200 Mwt. Withdrawals to be made during the nearest scheduled refuelling outage.

† Removed during August 1972 outage.

** Vessel wall dosimeters.

†† Removed during May 1975 outage.

††† Removed during November 1980 outage.

TABLE III

INITIAL TRANSITION TEMPERATURES FOR LACBWR SURVEILLANCE MATERIALS

Mat'l Ident.	DW-NDT Temperature (°F) Surface	DW-NDT Temperature (°F) 1/4t	30 ft-lb Charpy "Fix" Transition Temperature (°F)
NP1054	-	10	-10
NP1055	-	-	-75
NP1056	40	50	30
Weld	-	-	-30
Standard	-	-	-60

IV. TESTING OF SPECIMENS AND EVALUATION OF DATA

SwRI utilized the procedure for shipment of the LACBWR surveillance capsules which had been used for the two previous capsule shipments. Four capsules, 3A, 3B, 8A, and 8B, were removed during an outage which began on November 9, 1980. The capsules were then taken to SwRI for analysis. One capsule was opened at a time so that the contents could be examined, identified, and placed in indexed receptacles to prevent mixing with the contents of the remaining capsules.

A. Capsule Disassembly

The first hot cell operation was to remove the closure plug at each end of the capsule with a bandsaw, locating the cut per assembly drawings contained in ACNP-66513.(12) The capsule shells were cut in two lengthwise with a milling machine. The aluminum powder was sintered to a degree similar to that previously reported(11), but the specimens could be broken out quite easily without damage.

The wires binding the assemblies together were removed, and the contents were carefully laid out so that the dosimeter wires could be recovered and identified as to location within the capsule. The specimens were cleaned in an ultrasonic bath, examined to determine the specimen identification number, and placed in an indexed receptacle. Those Charpy V-notch specimens containing low-melting eutectic alloys were examined to determine which temperature indicators had fused during the exposure period, the results being described in Section IV.C of this report.

B. Test Equipment and Procedures

The tensile specimens were tested in a 22,000-lb servo-controlled tension testing machine equipped with a strain gage load cell. Accessories include a set of elevated temperature extensometer arms which attach directly to the specimen gage section, an Instron strain gage extensometer, an electric laboratory furnace, and an X-Y recorder. The calibration of the load cell was verified prior to conducting the tensile tests with an elastic proving ring traceable to the U.S. Bureau of Standards. Tests were conducted on each material at the temperature of the upper knee of the Charpy curve and at 550°F. Elevated temperature tensile specimens were instrumented with two thermocouples wired to the top and bottom of the gage section of the specimen.

The Charpy V-notch tests were conducted on an instrumented SATEC impact machine permanently installed in a warm cell. The calibration of the machine had been checked with a set of USAMMRC standards less than one year previous to the dates of testing. Nonambient specimen temperatures were obtained with a liquid bath. The procedure permitted the operator to remove a specimen from the temperature conditioning bath, place it on the anvil, and break it in less than five seconds.

Nonambient test temperatures (tensile specimens and Charpy conditioning bath) were measured with thermocouples made from calibrated wire and a laboratory potentiometer which is periodically checked against standard voltage sources traceable to the U.S. Bureau of Standards.

The flux wires were weighed on a Mettler laboratory balance then counted with a Ge(Li) solid state detector and a 4084-channel Norland multichannel analyzer. In addition to the unknowns, ^{60}Co, ^{137}Cs, and ^{54}Mn standards were counted to determine the efficiency of the experimental setup as a function of γ-ray energy.

C. Evaluation of Thermal Monitors and Flux Wires

Examination of the thermal monitors revealed that all of the 554°F melting point eutectic alloys had fused. The presence of sintered aluminum powder made it difficult to assess the condition of the 579°F alloy specimens, but none of the 604°F or 621°F alloy specimens had fused. Therefore, it was concluded that the maximum temperature reached by the contents of the four capsules during the operating period of the LACBWR vessel was above 554°F and below 604°F.

The specific activities of each flux wire, corrected to the plant shutdown date of November 9, 1980 (hereafter referred to as the time of removal--TOR), were determined. The first step in the calculation of the neutron flux is to correct the specific activities at TOR, A(TOR), to infinitely dilute saturated activities at a selected power level, A_s:

$$A(TOR)/A_s = \sum_{m=1}^{m=n} (1 - e^{-\lambda T_m})\, e^{-\lambda t_m}$$

where:

- λ = decay constant for the activation product
- T_m = equivalent operating days at the selected power level for the m^{th} operating period
- t_m = number of days from the end of the m^{th} operating period to TOR.

The daily load charts and operating summaries from the LACBWR monthly operating reports covering the period from July 10, 1967, to November 9, 1980, were utilized to determine values for T_m at 165 MWth and for t_m to the TOR date. The plant operations were divided into 64 operating periods as summarized in Table IV. The resulting saturated activities for each flux wire removed from specimen capsules are given in Table V.

TABLE IV

SUMMARY OF LACBWR PLANT OPERATIONS UP TO 1980 REFUELLING

Operating Period (m)	Dates: Start	Dates: Stop	Shutdown Days	Operating Days	Reactor Power Output (MWDt)	Equivalent Operating Days (T_m)	Decay time After Period (t_m)
1	07-10-67	02-28-68	-	233	24	0.15	4,638
	02-28-68	03-19-68	20	-	-	-	-
2	03-19-68	03-26-68	-	7	66	0.40	4,611
	03-26-68	04-21-68	26	-	-	-	-
3	04-21-68	04-26-68	-	5	61	0.37	4,580
	04-26-68	05-07-68	11	-	-	-	-
4	05-07-68	05-30-68	-	23	368	2.23	4,[illegible]46
	05-30-68	11-20-68	174	-	-	-	-
5	11-20-68	12-24-68	-	34	409	2.48	4,338
	12-24-68	01-14-69	21	-	-	-	-
6	01-14-69	01-31-69	-	17	130	0.79	4,303
	01-31-69	03-27-69	55	-	-	-	-
7	03-27-69	04-03-69	-	7	[illegible]	1.45	4,238
	04-03-69	04-29-69	26	-	-	-	-
8	04-29-69	05-03-69	-	4	398	2.41	4,208
	05-03-69	07-21-69	79	-	-	-	-
9	07-21-69	10-14-69	-	85	9,143	55.41	4,044
	10-14-69	05-02-70	200	-	-	-	-
10	05-02-70	06-18-70	-	47	3,054	18.51	3,797
	06-18-70	07-02-70	14	-	-	-	-
11	07-02-70	09-02-70	-	62	5,670	34.36	3,727
	09-02-70	09-05-70	3	-	-	-	-
12	09-05-70	10-31-70	-	56	8,274	50.15	3,662
	10-31-70	12-20-70	50	-	-	-	-
13	12-20-70	01-20-71	-	31	4,750	28.79	3,581
	01-20-71	01-30-71	10	-	-	-	-
14	01-30-71	02-07-71	-	8	1,207	7.32	3,563
	02-07-71	02-10-71	3	-	-	-	-
15	02-10-71	02-25-71	-	15	2,242	13.59	3,545
	02-25-71	02-28-71	3	-	-	-	-
16	02-28-71	03-09-71	-	9	1,240	7.32	3,533
	03-09-71	03-18-71	9	-	-	-	-
17	03-18-71	09-03-71	-	169	21,659	131.27	3,355
	09-03-71	12-31-71	119	-	-	-	-
18	12-31-71	01-08-72	-	8	992	6.01	3,228
	01-08-72	01-13-72	5	-	-	-	-
19	01-13-72	03-31-72	-	78	10,929	66.24	3,145
	03-31-72	04-03-72	3	-	-	-	-
20	04-03-72	05-19-72	-	46	7,695	46.64	3,096
	05-19-72	06-17-72	29	-	-	-	-
21	06-17-72	06-23-72	-	6	790	4.79	3,061
	06-23-72	06-26-72	3	-	-	-	-
22	06-26-72	07-04-72	-	8	1,081	6.55	3,050
	07-04-72	07-07-72	3	-	-	-	-
23	07-07-72	07-15-72	-	8	960	5.82	3,039
	07-15-72	07-22-72	7	-	-	-	-
24	07-22-72	08-19-72	-	28	4,040	24.48	3,004
	08-19-72	10-14-72	56	-	-	-	-
25	10-14-72	11-02-72	-	19	2,094	12.69	2,929
	11-02-72	11-05-72	3	-	-	-	-
26	11-05-72	11-23-72	-	18	2,264	13.72	2,908
	11-23-72	12-02-72	9	-	-	-	-
27	12-02-72	02-01-73	-	61	8,031	48.67	2,838
	02-01-73	02-04-73	3	-	-	-	-
28	02-04-73	03-30-73	-	54	8,599	52.12	2,781
	03-30-73	06-26-73	88	-	-	-	-
29	06-26-73	06-30-73	-	4	145	0.88	2,689
	06-30-73	07-03-73	3	-	-	-	-
30	07-03-73	07-09-73	-	6	440	2.67	2,680
	07-09-73	07-14-73	5	-	-	-	-
31	07-14-73	07-16-73	-	2	370	2.24	2,673
	07-16-73	07-21-73	5	-	-	-	-
32	07-21-73	09-10-73	-	51	7,234	43.84	2,617
	09-10-73	09-13-73	3	-	-	-	-
33	09-13-73	11-03-73	-	51	6,529	39.57	2,563
	11-03-73	12-26-73	53	-	-	-	-
34	12-26-73	03-01-74	-	65	9,515	57.67	2,445
	03-01-74	03-03-74	2	-	-	-	-
35	03-03-74	05-06-74	-	64	10,090	61.15	2,379
	05-06-74	05-29-74	23	-	-	-	-
36	05-29-74	07-15-74	-	47	6,511	39.46	2,309
	07-15-74	07-16-74	1	-	-	-	-
37	07-16-74	08-28-74	-	43	6,714	40.69	2,265
	08-28-74	09-20-74	23	-	-	-	-
38	09-20-74	09-24-74	-	4	484	2.93	2,228
	09-24-74	10-10-74	16	-	-	-	-
39	10-10-74	01-13-75	-	95	14,288	86.59	2,127
	01-13-75	01-14-75	1	-	-	-	-
40	01-14-75	02-14-75	-	31	4,851	29.40	2,095
	02-14-75	02-17-75	3	-	-	-	-
41	02-17-75	04-16-75	-	58	8,492	51.47	2,034
	04-16-75	04-21-75	5	-	-	-	-
42	04-21-75	05-09-75	-	18	2,631	15.95	2,011
	05-09-[illegible]	08-11-75	94	-	-	-	-

TABLE IV (CONT.)

Operating Period (m)	Dates Start	Dates Stop	Shutdown Days	Operating Days	Reactor Power Output (MWDt)	Equivalent Operating Days (T_m)	Decay Time After Period (t_m)
43	08-11-75	12-11-75	-	122	17,711	107.34	1,795
	12-11-75	12-12-75	1	-	-	-	-
44	12-12-75	02-23-76	-	73	10,871	65.88	1,721
	02-23-76	08-11-76	170	-	-	-	-
45	08-11-76	08-15-76	-	4	8	0.05	1,547
	08-15-76	08-17-76	2	-	-	-	-
46	08-17-76	11-03-76	-	78	10,923	66.20	1,46[illegible]
	11-03-76	11-06-76	3	-	-	-	-
47	11-06-76	11-13-76	-	7	618	3.75	1,457
	11-13-76	11-19-76	6	-	-	-	-
48	11-19-76	02-02-77	-	75	10,498	63.62	1,376
	02-02-77	02-04-77	2	-	-	-	-
49	02-04-77	05-11-77	-	96	9,472	57.41	1,278
	05-11-77	03-09-78	302	-	-	-	-
50	03-09-78	04-27-78	-	49	5,109	30.96	927
	04-27-78	05-10-78	13	-	-	-	-
51	05-10-78	10-18-78	-	161	18,676	113.19	753
	10-18-78	11-08-78	21	-	-	-	-
52	11-08-78	11-20-78	-	12	565	3.4[illegible]	720
	11-20-78	11-24-78	4	-	-	-	-
53	11-24-78	01-13-79	-	50	3,903	23.66	666
	01-13-79	01-22-79	9	-	-	-	-
54	01-22-79	03-25-79	-	62	4,696	28.46	595
	03-25-79	05-26-79	62	-	-	-	-
55	05-26-79	06-02-79	-	7	157	0.95	526
	06-02-79	06-03-79	1	-	-	-	-
56	06-03-79	07-05-79	-	32	3,887	23.56	493
	07-05-79	07-06-79	1	-	-	-	-
57	07-06-79	09-04-79	-	60	8,087	49.01	432
	09-04-79	09-07-79	3	-	-	-	-
58	09-07-79	09-28-79	-	21	2,225	13.49	408
	09-28-79	10-06-79	8	-	-	-	-
59	10-06-79	02-01-80	-	118	15,434	93.54	282
	02-01-80	02-04-80	3	-	-	-	-
60	02-04-80	04-06-80	-	62	8,488	51.44	217
	04-16-80	04-30-80	24	-	-	-	-
61	04-30-80	06-21-80	-	52	7,391	44.79	141
	06-21-80	06-28-80	7	-	-	-	-
62	06-28-80	08-09-80	-	42	5,055	30.64	92
	08-09-80	08-18-80	9	-	-	-	-
63	08-18-80	08-25-80	-	7	323	1.96	76
	08-25-80	09-03-80	9	-	-	-	-
64	09-03-80	11-09-80	-	67	8,426	51.07	0
				Total	337,557	2,045.80*	

* 2,045.80 days = 1.76757 x 10^8 seconds.

TABLE V

MATERIAL SURVEILLANCE CAPSULE DOSIMETRY RESULTS
(Saturated Activity, dps/mg)

Capsule	$^{59}Co(n,\gamma)^{60}Co$	$^{54}Fe(n,p)^{54}Mn$	$^{58}Ni(n,p)^{58}Co$
3AT	.7575E+08	.7027E+04	.9055E+05
3AU	.7877E+08	.6974E+04	.9243E+05
3AV	.8592E+08	.7288E+04	.9334E+05
3AW	.9558E+08	.7274E+04	.9736E+05
3AX	.9532E+08	.5946E+04 (a)	.9348E+05
3AY	.1162E+09	.6915E+04	.9145E+05
3AZ	.8420E+08	.6725E+04	.9153E+05
3BT	.7271E+08	.6967E+04	.8804E+05
3BU	.7800E+08	.4960E+04 (a)	.8944E+05
3BV	.8428E+08	.7089E+04	.9234E+05
3BW	.9066E+08	.6943E+04	.9195E+05
3BX	.8793E+08	.6689E+04	.9010E+05
3BY	.8248E+08	.6638E+04	.8866E+05
3BZ	.8542E+08	.6078E+04	.8586E+05
8AT	.7173E+08	.7018E+04	.8849E+05
8AU	.7697E+08	.7125E+04	.8819E+05
8AV	.8475E+08	.7303E+04	.9155E+05
8AW	.8751E+08	.7353E+04	.9148E+05
8AX	.8327E+08	.6948E+04	.8961E+05
8AY	.8113E+08	.6746E+04	.8803E+05
8AZ	.8028E+08	.6471E+04	.8589E+05
8BT	.7747E+08	.7308E+04	.9270E+05
8BU	.7456E+08	.7375E+04	.9280E+05
8BV	.8917E+08	.7641E+04	.9586E+05
8BW	.9005E+08	.7308E+04	.9525E+05
8BX	.9357E+08	.7627E+04	.9430E+05
8BY	.9404E+08	.7178E+04	.9313E+05
8BZ	.9039E+08	.7025E+04	.9127E+05
	Avg. = .8529E+08	Avg. = .7040E+04	Avg. = .9125E+05

(a) Values not used in computing averages.

The neutron flux density is given by :

$$\phi = \frac{A_s}{N_0 \bar{\sigma}}$$

where:

ϕ = energy dependent neutron flux density ($n/cm^2 sec$)

A_s = saturated activity (dps/mg target element)

N_0 = number of target atoms/mg target element

$\bar{\sigma}$ = spectrum-averaged activation cross section (cm^2).

In the analysis of the LACBWR neutron flux dosimeters, the neutron flux density calculations were based on the results obtained from the iron and nickel wires. (The bare cobalt wires were sensitive to thermal and epithermal flux as well as the fast flux.) The value of $\bar{\sigma}$ was based on two spectra:

1. A fission spectrum-averaged cross section. This was utilized for reference only because much of the early surveillance program data in the literature is based on the use of a fission spectrum-averaged cross section.

2. A calculated spectrum-averaged cross section. The DOT 3.5 two-dimensional discrete ordinates transport code was used to calculate the neutron flux densities and spectra at various points of interest outside the LACBWR core. This information was utilized to determine capsule lead factors (ratio of the neutron flux density at the capsule locations to the maximum neutron flux density incident on the pressure vessel wall) as well as the spectrum-averaged cross sections for the $^{54}Fe(n,p)^{54}Mn$ and $^{58}Ni(n,p)^{58}Co$ reactions. Details of the DOT 3.5 analysis are presented in Appendix C.

As discussed in Appendix C, the neutron flux densities and spectra at the surveillance capsule and vessel wall locations are dependent on the void concentration in the steam separators since the steam separators are located between the reactor core and the surveillance capsules (and, of course, the vessel wall). This analysis indicated that an average void content of 30% may be reasonable, but that a conservative approach would be to assume an average void content of 40% because this leads to a higher calculated value of vessel wall neutron flux density.

The flux values obtained for each capsule were multiplied by 1.768×10^8 seconds (the equivalent operating time at 165 MWth) to determine the corresponding values of neutron fluence. The neutron flux and fluence determinations obtained with each method are summarized in Table VI.

The dosimetry analyses reported earlier for the six previous capsules[11] were based only on the iron activities. Reevaluated fluxes and fluences for these capsules, using both the iron and the nickel data, are also included in Table VI. It is of interest to note that the reevaluated flux for Capsules 1A and 1B agrees well with the SAND-II calculation based on a BWR spectrum reported earlier.[10]

D. Impact and Tensile Test Results

The results of tensile tests conducted on specimens removed during the 1980 outage are given in Table VII. Examination of the 550°F data in Table VII indicates that the plate and weld materials experienced a degree of radiation hardening similar to those in Capsules 2A and 7B.[11] This correlates with the results of the dosimetry analysis since all six capsules received about the same fluence.

The Charpy V-notch impact data obtained on the specimens removed during the 1980 refuelling outage are given in Tables VIII through XII. Charpy V-notch fracture energy transition curves were developed by a least-squares fit of each data set to the relationship:

$$Y = A + B \tanh\left(\frac{T - T_o}{C}\right)$$

where:

Y = C_v function (fracture energy or lateral expansion)

T = C_v test temperature, deg F

A = Intercept when $\tanh\left(\frac{T - T_o}{C}\right) = 0$

B = Slope

T_o = Temperature at transition midpoint, deg F

C = One-half of transition range, deg F.

The resulting transition curves, given in Appendix D, suffered from the lack of lower shelf data. The hand-drawn curves given in Figures 2 through 6 were used to define ΔRT_{NDT} for each material. A summary of the transition temperature shifts at 30 ft-lb, 50 ft-lb, and 35 mil lateral expansion, as well as irradiated upper shelf energies, is presented in Table XIII.

TABLE VI

SUMMARY OF NEUTRON DOSIMETRY RESULTS

Spectrum Type	$\bar{\sigma}$, E > 1 MeV (barns) $^{54}Fe(n,p)^{54}Mn$	$\bar{\sigma}$, E > 1 MeV (barns) $^{58}Ni(n,p)^{58}Co$	Capsule Identification	Neutron Flux Density,[a] E > 1 MeV ($n/cm^2 sec$)	Neutron Fluence,[a] E > 1 MeV (n/cm^2)
Fission[b]	0.113	-	3A	9.95×10^{10}	
	0.113	-	3B	9.53×10^{10}	
	0.113	-	8A	9.90×10^{10}	
	0.113	-	8B	10.40×10^{10}	
				Avg. = 9.96×10^{10}	1.76×10^{19}
DOT 3.5[c]	0.177	0.223	3A	6.15×10^{10}	
	0.177	0.223	3B	5.90×10^{10}	
	0.177	0.223	8A	6.01×10^{10}	
	0.177	0.223	8B	6.32×10^{10}	
				Avg. = 6.10×10^{10}	1.08×10^{19}
	0.176	0.222	1A	6.58×10^{10}	
	0.176	0.222	1B	5.92×10^{10}	
				Avg. = 6.25×10^{10}	2.81×10^{18}
	0.168	0.214	2A	1.06×10^{11}	
	0.168	0.214	7B	1.05×10^{11}	
				Avg. = 1.05×10^{11}	1.02×10^{19}
	0.184	0.231	9A	6.34×10^{10}	
	0.184	0.231	9B	6.70×10^{10}	
				Avg. = 6.52×10^{10}	6.32×10^{18}

(a) Neutron flux densities and fluences subject to a ± 15% uncertainty (1 σ).
(b) $\bar{\sigma}_f$ for fission-averaged cross section based on ASTM E 261. $\bar{\sigma}$, E > 1 MeV = $\bar{\sigma}_f/0.693$.[5]
(c) $\bar{\sigma}$, E > 1 MeV, based on 40% voids in steam separators, per Appendix C.

TABLE VII

IRRADIATED TENSILE PROPERTIES OF LACBWR SURVEILLANCE MATERIAL

Test No.	Material Identification	Capsule Identification	Temperature (°F)	UTS (ksi)	.2% YS (ksi)	Elongation (%)	R.A. (%)
1	Weld	3A	+150	(a)	(a)	(b)	65.9
2	Weld	8A	+150	91.2	76.2	(b)	64.4
3	Weld	8A	+150	88.9	69.6	26.2	52.6
4	Weld	3A	+150	86.0	64.5	28.0	56.5
5	Weld	8B	+150	91.2	74.1	23.5	52.0
6	Weld	3B	+150	87.8	68.3	30.2	61.0
14	Weld	8B	+550	77.4	57.4	19.7	53.8
18	Weld	3B	+550	80.3	61.8	20.3	49.3
19	Weld	8A	+550	88.3	70.4	20.7	55.4
20	Weld	3A	+550	84.1	68.1	(b)	61.1
21	Weld	3B	+550	89.3	72.5	20.4	53.2
22	Weld	8B	+550	88.8	69.4	21.3	57.0
7	NP-1055	3A	+200	88.8	68.4	25.3	67.8
8	NP-1055	3A	+200	88.6	69.2	24.9	67.4
9	NP-1055	8B	+200	91.1	70.9	24.5	66.8
10	NP-1055	3B	+200	91.7	71.0	26.2	68.2
11	NP-1055	8A	+200	93.4	72.9	23.4	68.2
12	NP-1055	8A	+200	91.9	71.6	24.5	67.8
13	NP-1055	8B	+550	91.7	66.5	22.0	61.1
15	NP-1055	3B	+550	89.8	65.5	22.2	64.6
16	NP-1055	8A	+550	91.1	67.6	20.6	60.5
17	NP-1055	3A	+550	85.5	63.2	26.4	63.9
23	NP-1055	8B	+550	90.6	67.3	20.6	60.5
24	NP-1055	3B	+550	90.3	66.8	21.2	63.1

(a) Extensometer malfunction.
(b) Specimen failed outside gage marks.

TABLE VIII

CHARPY V-NOTCH DATA ON PLATE NP-1055
(Removed from LACBWR in 1980)

Capsule No.	Temp. (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Fracture Appearance (% shear)
8A	- 40	22.0	16	nil
8B	- 40	18.0	14	nil
3A	- 40	21.0	17	nil
3B	- 40	20.0	18	nil
8A	- 20	27.0	21	nil
8B	- 20	39.0	29	nil
3A	- 20	16.0	15	nil
3B	- 20	45.0	35	nil
8A	0	53.0	22	nil
8B	0	9.5	19	nil
3A	0	43.0	20	5
3B	0	56.0	29	10
8A	0	28.5	40	nil
8B	0	22.5	10	nil
3A	0	22.5	35	nil
3B	0	39.0	43	nil
8A	+ 20	36.5	29	5
8B	+ 20	15.0	15	nil
3A	+ 20	33.5	29	5
3B	+ 20	63.5	52	15
8A	+ 40	51.0	44	15
8B	+ 40	26.5	27	nil
3A	+ 40	76.0	61	5
3B	+ 40	64.0	52	25
8A	+ 74	75.0	58	35
8B	+ 74	81.0	59	50
3A	+ 74	91.0	70	50
3B	+ 74	83.5	65	40
8A	+110	103.5	85	100
8B	+110	97.0	78	100
3A	+110	110.0	85	100
3B	+110	88.5	71	70
8A	+150	96.5	80	100
8B	+150	112.5	90	100
3A	+150	112.0	77	100
3B	+150	101.0	82	100
8A	+200	102.5	87	100
8B	+200	112.5	91	100
3A	+200	118.5	90	100
3B	+200	103.0	86	100

TABLE IX

CHARPY V-NOTCH DATA ON PLATE NP-1054
(Removed from LACBWR in 1980)

Capsule No.	Temp. (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Fracture Appearance (% shear)
8A	- 20	37.0	30	nil
3A	- 20	23.5	19	nil
8A	+ 40	70.5	58	10
3A	+ 40	74.0	67	25
8A	+ 74	84.0	65	40
3A	+ 74	64.5	53	30
8A	+110	102.5	82	100
3A	+110	94.0	82	90
8A	+150	107.5	89	100
3A	+150	108.0	89	100
8A	+200	107.0	90	100
3A	+200	102.0	88	100

TABLE X

CHARPY V-NOTCH DATA ON PLATE NP-1056
(Removed from LACBWR in 1980)

Capsule No.	Temp. (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Fracture Appearance (% shear)
8B	+ 74	35.0	25	nil
3B	+ 74	16.5	18	5
8B	+110	16.5	22	nil
3B	+110	37.5	39	5
8B	+150	41.0	45	40
3B	+150	41.0	43	20
8B	+200	46.0	48	20
3B	+200	74.0	65	65
8B	+250	91.0	75	100
3B	+250	78.0	71	95
8B	+300	88.5	79	100
3B	+300	91.5	84	100

TABLE XI

CHARPY V-NOTCH DATA ON WELD METAL
(Removed from LACBWR in 1980)

Capsule No.	Temp. (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Fracture Appearance (% shear)
8A	+ 40	15.0	16	20
8B	+ 40	14.5	16	nil
3A	+ 40	23.0	25	15
3B	+ 40	7.5	9	5
8A	+ 55	18.0	21	5
8B	+ 55	23.0	26	25
8A	+ 74	30.0	31	30
8B	+ 74	35.5	30	25
3A	+ 74	26.0	42	25
3B	+ 74	38.5	24	5
3A	+ 74	42.5	22	25
8A	+110	44.5	44	65
8B	+110	47.5	45	65
3A	+110	24.0	28	20
3B	+110	45.0	44	100
3B	+110	31.0	35	60
8A	+150	55.5	51	95
8B	+150	54.0	54	80
3A	+150	56.5	55	10
3B	+150	53.5	54	95
8A	+200	55.5	58	100
8B	+200	56.0	56	100
3A	+200	60.5	63	100
3B	+200	52.5	52	100

TABLE XII

CHARPY V-NOTCH DATA ON STANDARD MATERIAL
(Removed from LACBWR in 1980)

Capsule No.	Temp. (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Fracture Appearance (% shear)
8A	0	14.5	14	5
8B	0	14.5	14	5
3A	0	17.5	18	10
3B	0	16.0	14	5
8A	+ 40	22.0	22	10
8B	+ 40	20.5	20	15
3A	+ 40	44.5	37	20
3B	+ 40	28.5	25	10
8A	+ 74	46.5	38	25
8B	+ 74	43.5	35	10
3A	+ 74	55.0	51	95
3B	+ 74	42.0	36	15
8A	+110	53.0	48	100
8B	+110	52.0	47	100
3A	+110	69.5	58	100
3B	+110	56.0	51	100
8A	+150	61.5	61	100
8B	+150	66.5	64	100
3A	+150	74.0	64	100
3B	+150	69.0	67	100
8A	+200	65.0	59	100
8B	+200	71.5	60	100
3A	+200	81.0	66	100
3B	+200	72.0	62	100

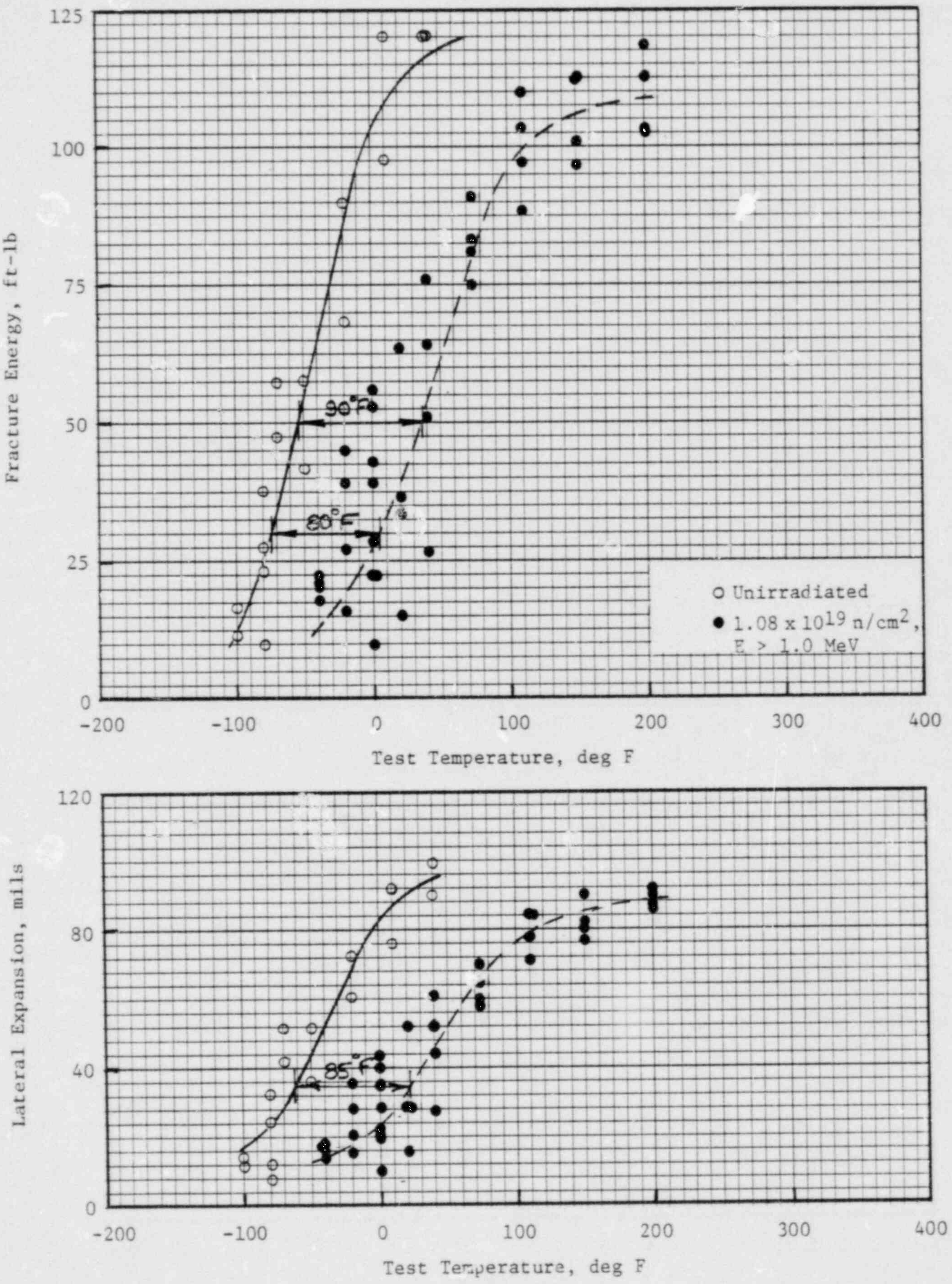

FIGURE 2. CHARPY V-NOTCH PROPERTIES OF PLATE NP-1055

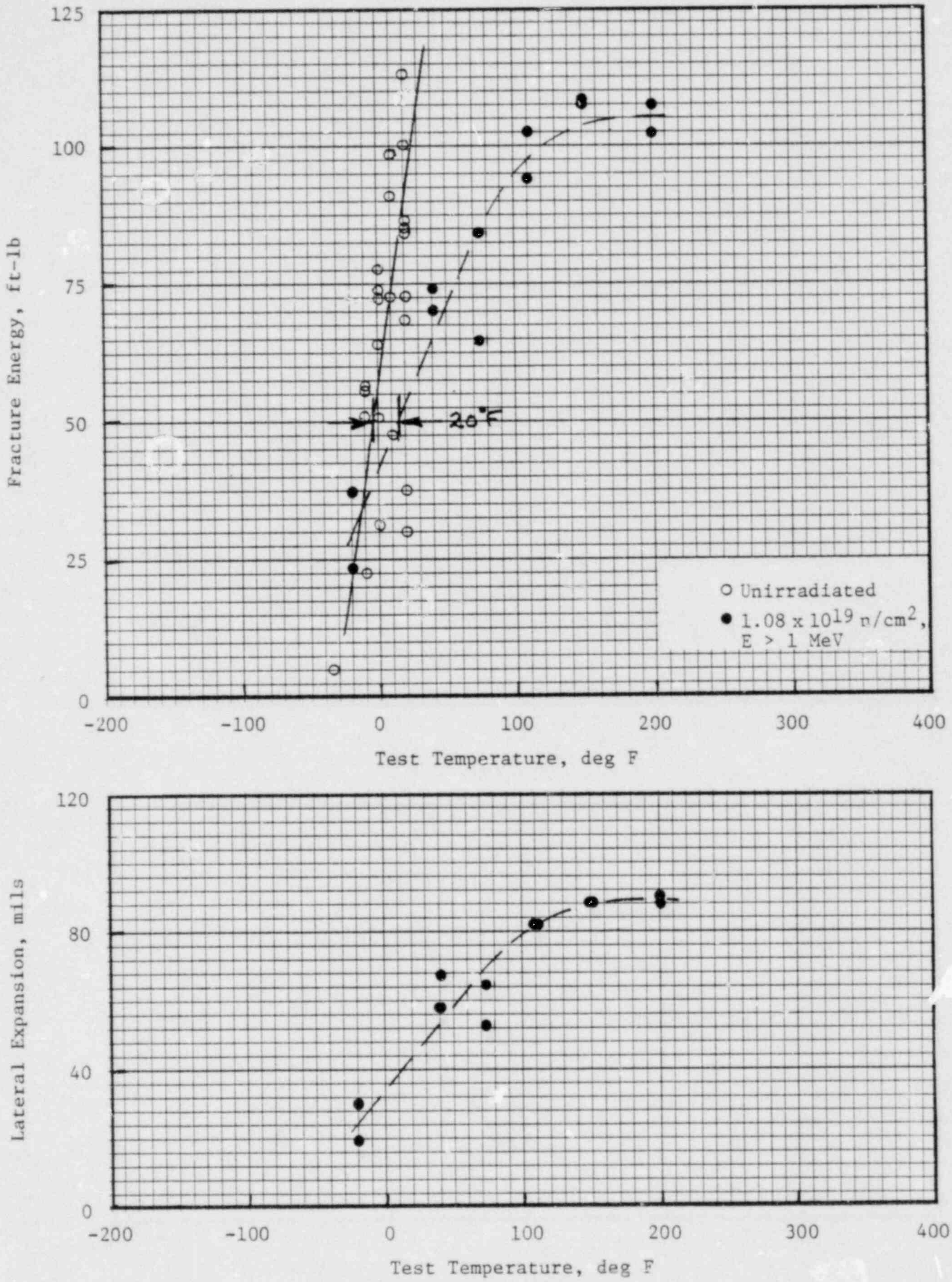

FIGURE 3. CHARPY V-NOTCH PROPERTIES OF PLATE NP-1054

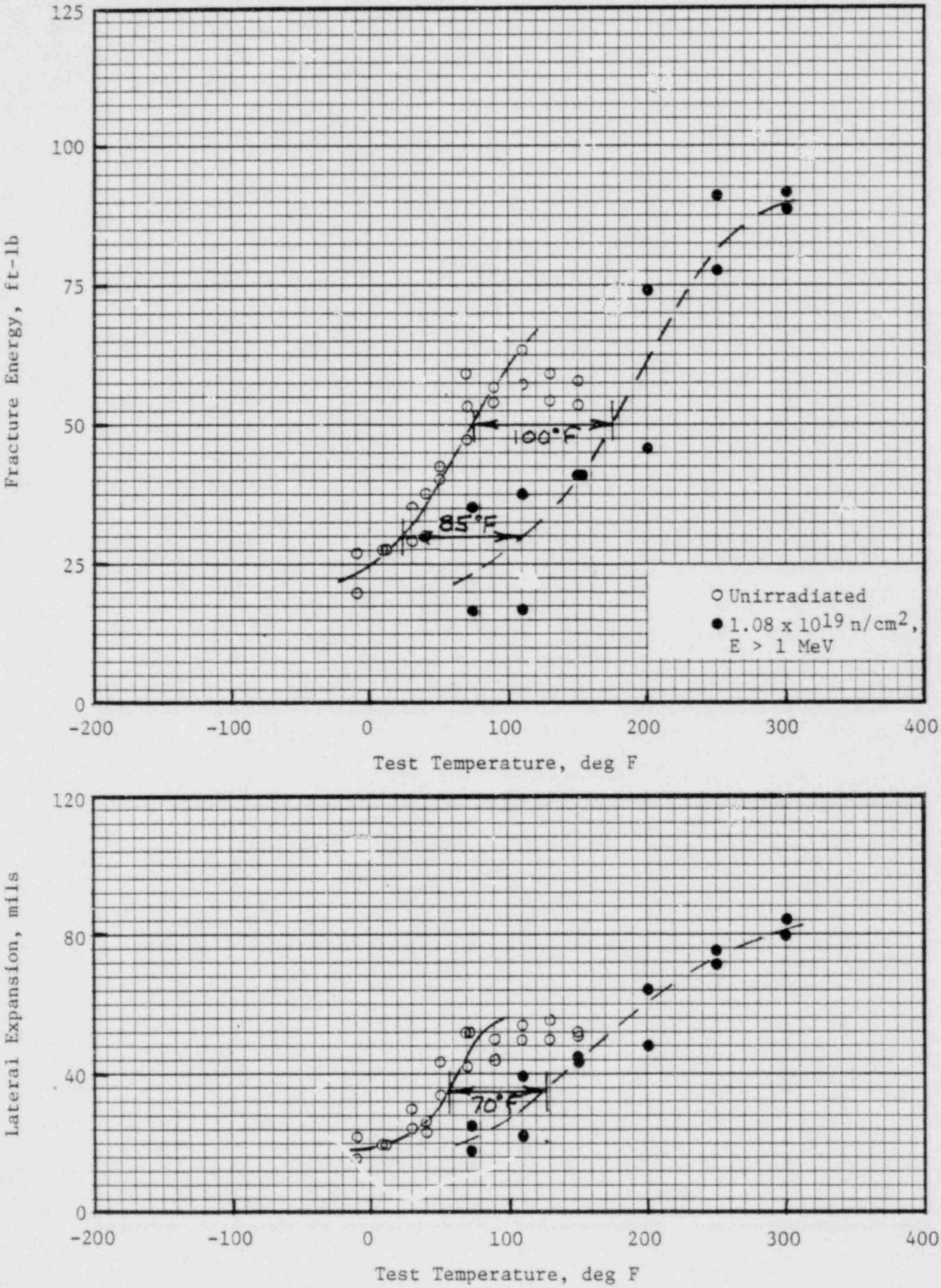

FIGURE 4. CHARPY V-NOTCH PROPERTIES OF PLATE NP-1056

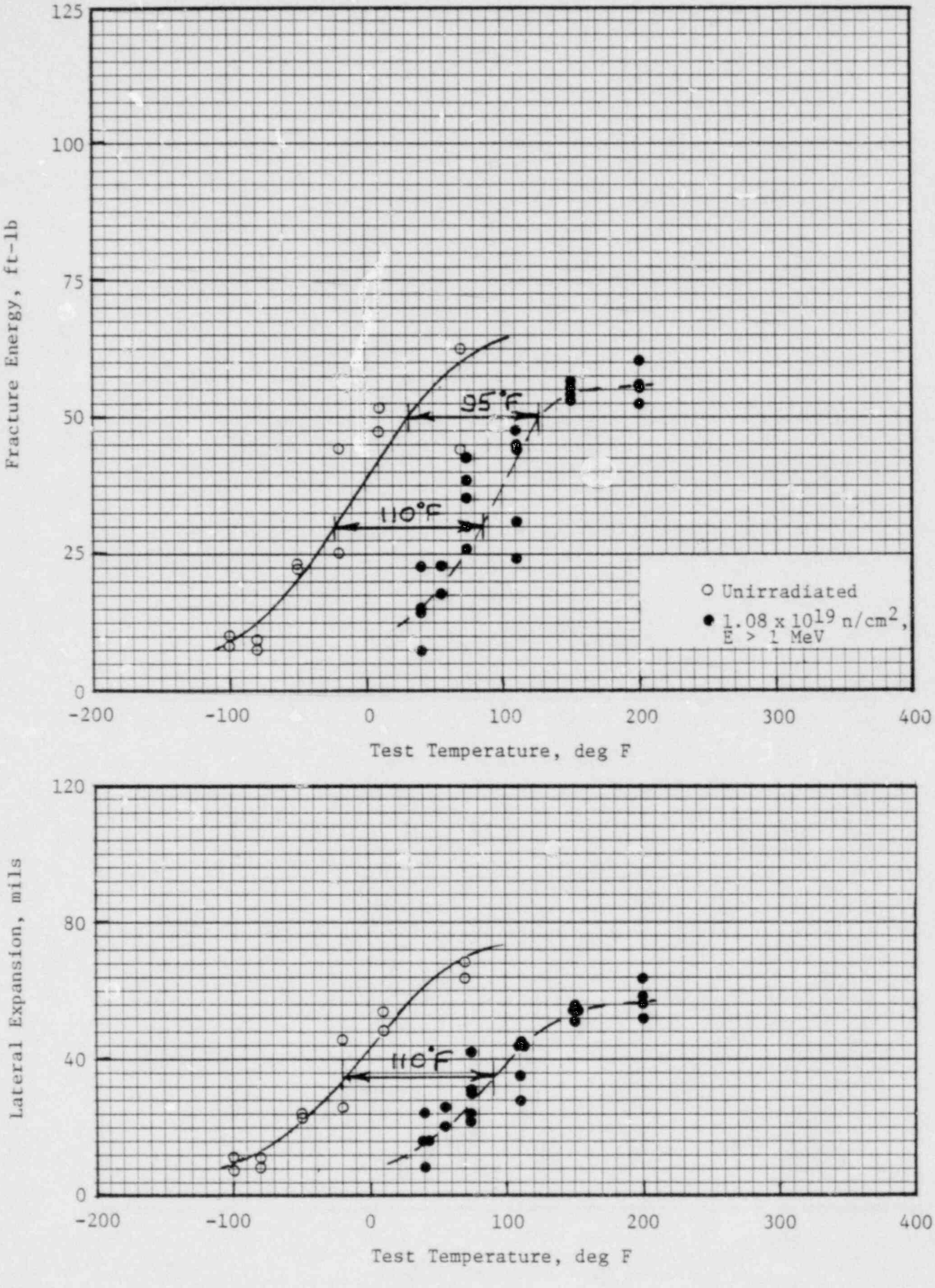

FIGURE 5. CHARPY V-NOTCH PROPERTIES OF LACBWR WELD METAL

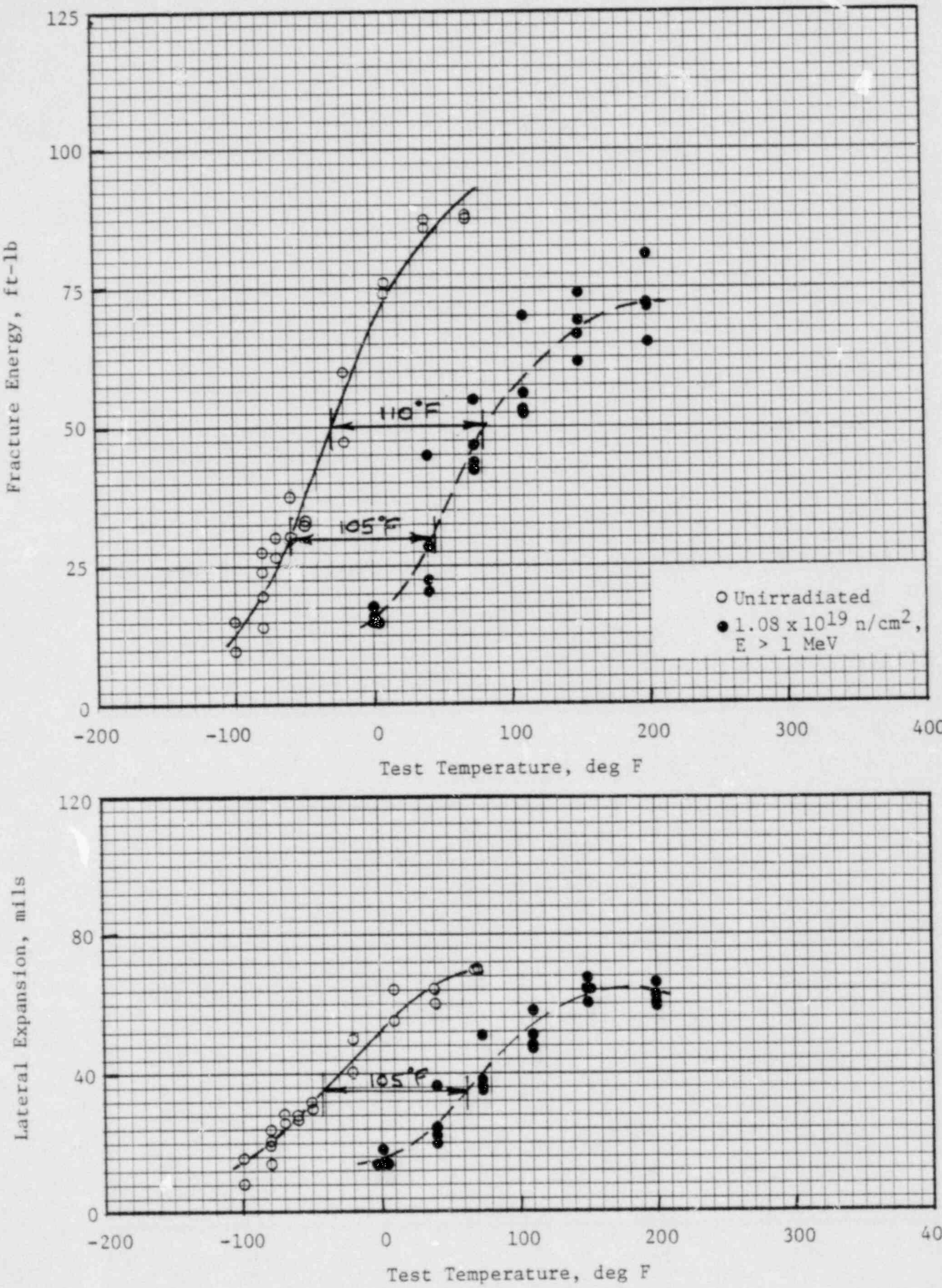

FIGURE 6. CHARPY V-NOTCH PROPERTIES OF STANDARD MATERIAL

TABLE XIII

EFFECT OF NEUTRON IRRADIATION ON LACBWR VESSEL SURVEILLANCE MATERIALS

Material Identification	Neutron Fluence n/cm^2, E > 1 MeV	30 ft-lb TT Increase, deg F	50 ft-lb TT Increase, deg F	35 mil TT Increase, deg F	RT_{NDT} [(a)] Increase, deg F	Irradiated C_v Shelf, ft-lb
NP-1055	1.08×10^{19}	80	90	85	80	106
NP-1054	1.08×10^{19}	nil	20	(b)	nil	105
NP-1056	1.08×10^{19}	85	100	70	100	90
Weld Metal	1.08×10^{19}	110	95	110	110	56
Standard	1.08×10^{19}	105	110	105	105	66

(a) ΔRT_{NDT} determined at 30 ft-lb level per ASTM E 185-79.
(b) Not determined. No lateral expansion data available for unirradiated material.

E. Check Chemical Analyses

The copper content of one broken Charpy V-notch specimen, representing each material type, was determined with an X-ray fluorescent technique.[15] The results, summarized below, agree well with those reported earlier, see Table I.

Material Identification	Copper Content (wt. %)
Plate NP-1054	0.10
Plate NP-1055	0.14
Plate NP-1056	0.14
Weld Metal	0.14
Standard Material	0.07

V. ANALYSIS OF RESULTS

The analysis of the data obtained from the LACBWR vessel surveillance specimen data 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. Determine the increase in RT_{NDT} as a function of reactor power generation. 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 LACBWR was calculated as described in Appendix C. The calculated lead factors for each of the LACBWR surveillance capsules are presented in Table XIV. The vessel I.D. surface lead factors vary from 2.13 for specimen capsules 2A, 2B, 7A, and 7B to 1.37 for specimen capsules 4A, 4B, 9A, and 9B. The neutron flux densitites at the vessel wall dosimeter capsule locations are very nearly equal to the maximum neutron flux density incident on the pressure vessel I.D. surface.

A. Reference Temperature Projections

An independent program for evaluating the response of the LACBWR pressure vessel material to accelerated neutron irradiation was carried out by the Naval Research Laboratory[16] at the request of the Atomic Energy Commission and Allis Chalmers Manufacturing Company. Figure 7, taken from Reference 16, summarizes the results obtained on specimens of LACBWR pressure vessel steels irradiated in the Oak Ridge Low Intensity Test Reactor at a controlled temperature of 550°F. (The ASTM reference material referred to in Figure 7 is not the same heat of A302B steel which is being used as a reference material in the LACBWR surveillance program.)

The transition temperature shifts obtained on the corresponding materials which have been removed from LACBWR have been added to Figure 7 for comparison. The 30 ft-lb transition temperature shifts are plotted at fluences based on a fission spectrum-averaged cross section to be consistent with the procedures employed by the Naval Research Laboratory at that time. The results obtained from the LACBWR surveillance program show good agreement with the results reported in Reference 16.

TABLE XIV

CALCULATED NEUTRON FLUX DENSITY LEAD FACTORS FOR
LACBWR VESSEL MATERIAL SURVEILLANCE CAPSULES

Capsule Identification	Lead Factor[a]		
	I.D. Surface	1/4 T[b]	3/4 T[c]
1A, 1B	1.62	2.02	3.60
2A, 2B	2.13	2.66	4.73
3A, 3B	1.66	2.07	3.68
4A, 4B	1.37	1.71	3.03
5A, 5B	1.50	1.87	3.33
6A, 6B	1.62	2.02	3.60
7A, 7B	2.13	2.66	4.73
8A, 8B	1.66	2.07	3.68
9A, 9B	1.37	1.71	3.03
10A, 10B	1.50	1.87	3.33
V.W.	0.96	1.20	2.10

(a) Ratio of neutron flux density, $E > 1$ MeV, at the capsule location to the maximum incident on or within the pressure vessel wall with 40% voids in the steam separators.

(b) 1/4 T is 1 in. within the pressure vessel wall. Neutron flux density is attenuated to 80% of that at the vessel I.D. surface.

(c) 3/4 T is 3 in. within the pressure vessel wall. Neutron flux density is attentuated to 45% of that at the vessel I.D. surface.

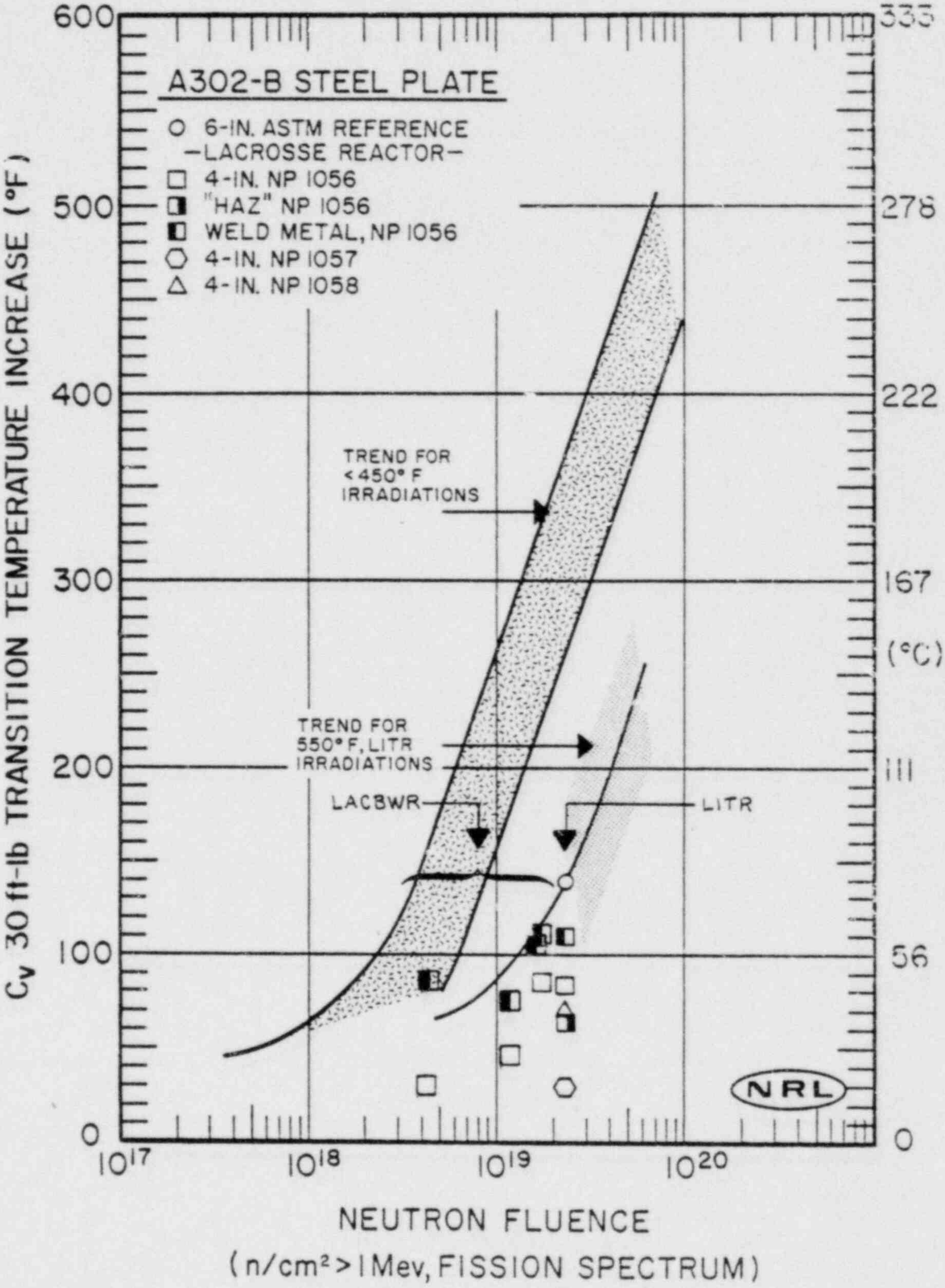

FIGURE 7. TRANSITION TEMPERATURE RESPONSE OF LACBWR VESSEL SURVEILLANCE MATERIALS TO NEUTRON IRRADIATION(16)

Trend curves could be constructed on Figure 7, but a more appropriate method is to employ Regulatory Guide 1.99.[8] In Figure 8, the curve identified as "upper limit" is taken from Figure 1 of Regulatory Guide 1.99, and the other two solid curves are computed for the LACBWR weld metal and Plate 1056 using the expression $A = [40 + 1000(\% \text{ Cu} - 0.08) + 5000 (\% \text{ P} - 0.008)] [f/10^{19}]^{1/2}$. The plotted data points represent results obtained from the 1980 surveillance capsules plus the reevaluated embrittlement data from the previous capsules (References 10 and 11). Figure 8 shows that the weld metal is the most radiation sensitive of the LACBWR vessel surveillance materials but that the measured transition temperature shifts fall below the calculated trend curve. The LACBWR vessel plate material, on the other hand, appears to follow the calculated trend curve quite well.

Averaging the dosimetry results obtained from the capsules tested to date indicates that the peak neutron flux, E > 1 MeV, incident on the pressure vessel wall is:

$$\frac{1}{4}\left(\frac{6.25 \times 10^{10}}{1.62} + \frac{1.05 \times 10^{11}}{2.13} + \frac{6.52 \times 10^{10}}{1.37} + \frac{6.10 \times 10^{10}}{1.66}\right) = 4.30 \times 10^{10} \text{ n/cm}^2\cdot\text{sec}$$

The maximum neutron fluence, E > 1 MeV, incident on the pressure vessel wall per EFPY is 4.30×10^{10} n/cm^2·sec x 3.15×10^7 sec/yr = 1.35×10^{18} n/cm^2.

The next step is to predict the RT_{NDT} of the LACBWR vessel as a function of reactor operation. The values of initial (unirradiated) RT_{NDT} given in Table XV have been established for this purpose. As indicated by Table XV, the minimum pressurization temperature was initially controlled by materials in the main steam and forced recirculation systems.[17] Because these materials are not exposed to a significant amount of neutron radiation, Plate NP-1056 became the controlling material within the first EFPY, since it had the highest initial RT_{NDT}.

The analysis of the data obtained from the 1975 capsules indicated that the weld metal would control the primary system RT_{NDT} after 3 EFPY. However, that analysis[11] defined ΔRT_{NDT} at the 50 ft-lb level (as required by Appendix G of 10CFR50 in effect at the time) which resulted in an overestimate of the weld metal ΔRT_{NDT} because of the proximity of 50 ft-lb to the upper shelf energy. All ΔRT_{NDT} values shown in Figure 8 were redetermined at the 30 ft-lb level per ASTM E 185-79[9] as currently practiced by the NRC.

The projection of the value of RT_{NDT} as a function of plant operations is shown in Figure 9. This projection is based on both the LACBWR weld metal and vessel plate trend curves of Figure 8 using the average pressure vessel fluence rate of 1.35×10^{18} n/cm^2, E > 1 MeV, per EFPY. Also included in Figure 9 is the RT_{NDT} vs. EFPY curve currently in the LACBWR Technical Specifications. The reevaluated vessel material surveillance data indicate that (1) the embrittlement rate is less than previously predicted, and (2) the vessel plate NP-1056 will control the primary system RT_{NDT} for more than 15 EFPY of operation because of its high initial value of RT_{NDT}.

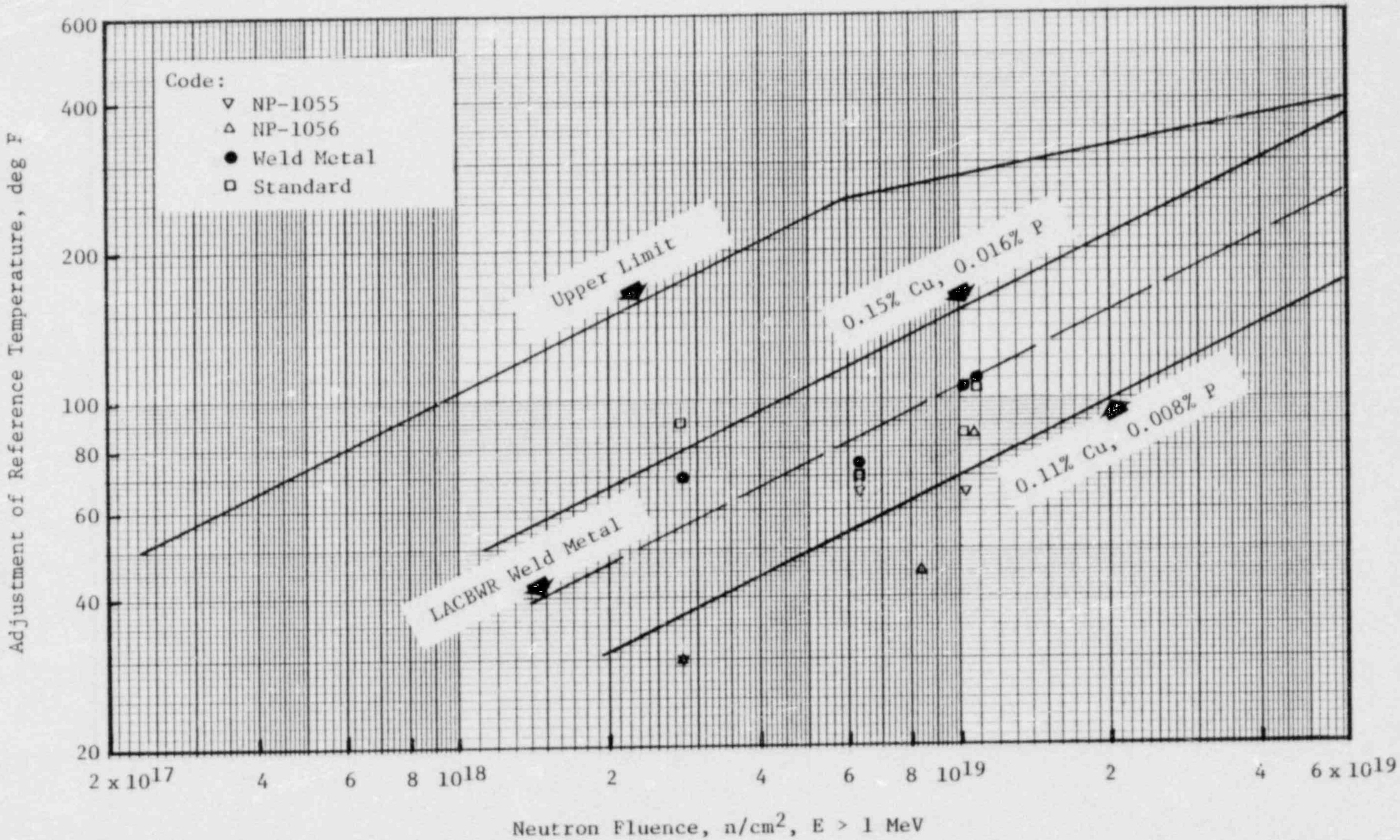

FIGURE 8. EFFECT OF NEUTRON FLUENCE ON RT_{NDT} OF LACBWR VESSEL SURVEILLANCE MATERIALS

TABLE XV

INITIAL TRANSITION AND RE ERENCE TEMPERATURES
FOR LACBWR PRESSURE B UNDARY MATERIALS

A. Main Steam and Forced Recirculation Materials(19)

Component	Material	Heat No.	Drop Weight NDT (°F)	20 ft-lb Charpy V-Notch Transition Temperature (°F)
20-in. Piping	A-335, P-11	B-2795	-30	-27
20-in. Piping	A-335, P-11	B-3064	+10	-
20-in. Piping	A-335, P-11	B-3080	+20	-
Pump Casing	A-217, C-5	5-272	+30	-
20-in. Fittings	A-217, WC-6	2-1823	>+10	+70
Valve Casing	A-217, WC-6	C-842	+50	-
Roto-valve Casing	A-216, WCB	C-853	+10	-
Roto-valve Casing	A-216, WCB	C-861	+30	-
Roto-valve Casing	A-216, WCB	C-863	+10	-
Roto-valve Casing	A-216, WCB	C-903	+30	-

B. Pressure Vessel Surveillance Materials

Material Ident.	DWNDT	30 ft-lb	50 ft-lb	35 mil	Initial RT_{NDT} deg F
NP1055	-	-75	-55	-65	0[a]
NP1054	10	-15	-5	-	10[b]
NP1056	50	15	90	55	50[b]
Weld	-	-20	30	-15	0[a]

(a) Since DWNDT tests were not run, RT_{NDT} = 30 ft-lb C_v TT or 0°F, whichever is higher.(18)

(b) RT_{NDT} is the higher of (1) DWNDT, (2) 60°F below the 50 ft-lb C_v TT (increased by 20°F because specimens are longitudinally oriented), and (3) 60°F below the 35 mil LE C_v TT (increased by 20°F because specimens are longitudinally oriented).(18)

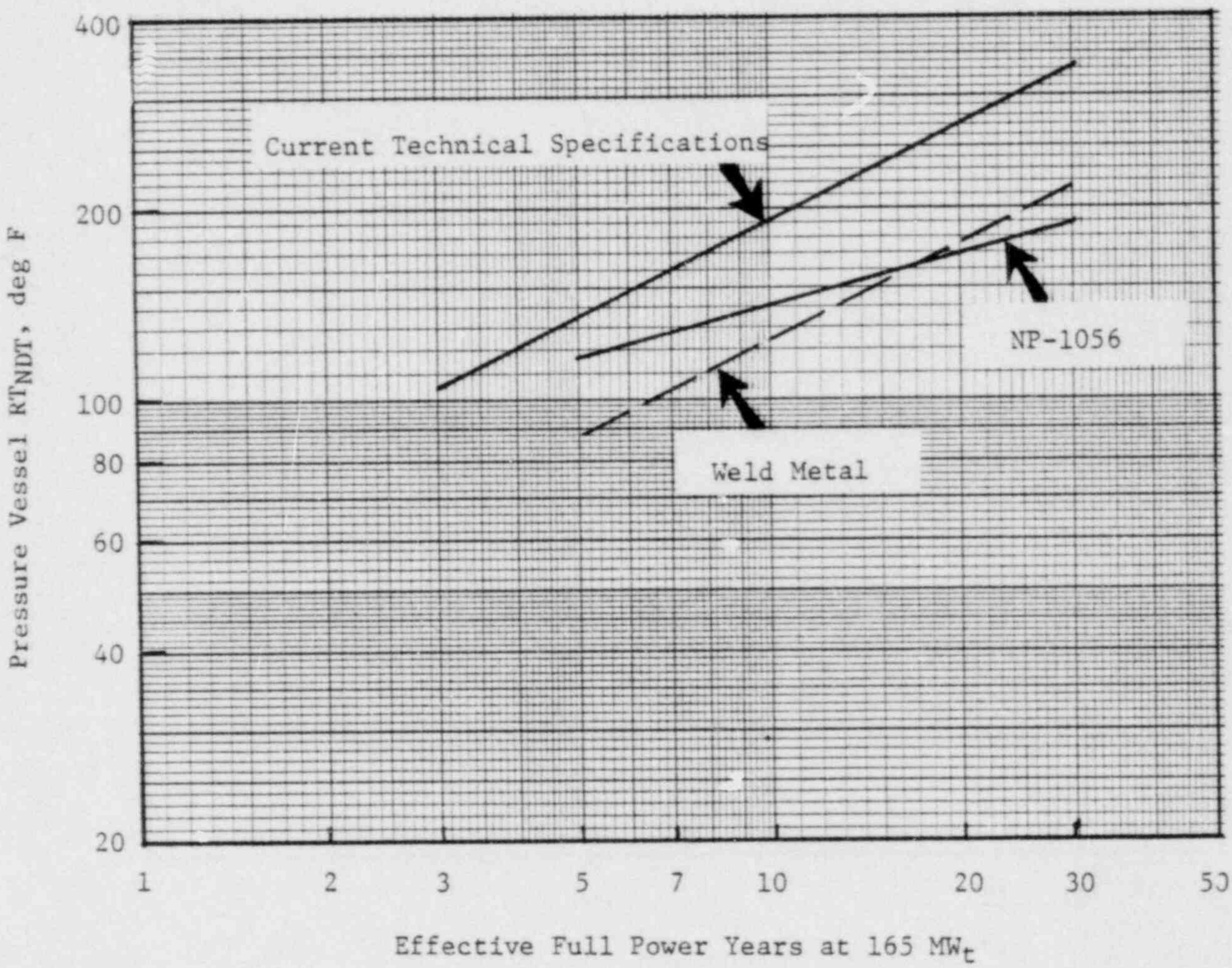

FIGURE 9. COMPARISON OF CURRENT AND REVISED CURVES RELATING THE REFERENCE TRANSITION TEMPERATURE TO PLANT OPERATION

B. Material Toughness Projections

Appendix G of 10CFR50 requires that the primary pressure boundary materials retain a Charpy V-notch upper shelf energy of at least 50 ft-lb through the life of the plant. The LACBWR pressure vessel plate materials have high C_v upper shelf energies and appear to be relatively insensitive to radiation embrittlement. The irradiated upper shelf energies for the LACBWR plate materials given in Table XIII are for longitudinally-oriented specimens, but if they are reduced by 35 percent as recommended in the NRC Standard Review Plan,(18) the values for material irradiated to 1.08 x 10^{19} n/cm^2 (E > 1 MeV) range from 58 ft-lb for Plate NP-1056 to about 69 ft-lb for Plate NP-1055. The weld metal exposed to the same fluence has retained a Charpy shelf energy of 56 ft-lb.

The initial (unirradiated) values of Charpy V-notch upper shelf energies were not well established for the LACBWR vessel beltline materials. However, the trend bands for decrease in shelf energy given in Regulatory Guide 1.99(8) can be used to estimate the Charpy shelf energies at the end of the 20 EFPY design life (E.O.L.) as shown in Figure 10. Regulatory Guide 1.99 estimates that the shelf energies of the weld metal, 0.15% Cu plate (NP-1055) and the 0.10% Cu plate (NP-1056) have been reduced by 29%, 24%, and 19%, respectively, at a fluence of 1.08 x 10^{19} n/cm^2, E > 1 MeV). At the peak 20 EFPY vessel fluence of 2.7 x 10^{19} n/cm^2 (E > 1 MeV), the guide predicts that the shelf energies of these materials will have been reduced by 36%, 30%, and 24%, respectively. The projected E.O.L. shelf energies are:

(1) Weld Metal:

$$C_v \text{ Shelf at E.O.L.} \approx 56\left(\frac{1-0.36}{1-0.29}\right) = 50 \text{ ft-lb}$$

(2) Plate NP-1055:

$$C_v \text{ Shelf at E.O.L.} \approx 69\left(\frac{1-0.30}{1-0.24}\right) = 64 \text{ ft-lb}$$

(3) Plate NP-1056:

$$C_v \text{ Shelf at E.O.L.} \approx 58\left(\frac{1-0.24}{1-0.19}\right) = 54 \text{ ft-lb}$$

The LACBWR material surveillance program does not include specimens representing the HAZ material. However, the companion program conducted by NRL(16) did include HAZ specimens machined from Plate NP-1056. The results obtained indicated that the radiation sensitivity of the HAZ material, as measured by the increase in the transition temperature, was similar to that of Plate NP-1056. Since the transition temperature of the unirradiated HAZ material was nearly 100°F below that of the base plate, and since the longitudinal C_v shelf energy was in the 90 ft-lb range (i.e., 60 ft-lb in the transverse direction) after being irradiated to 2.1 x 10^{19} n/cm^2 (E > 1 MeV, based on a fission spectrum-averaged cross section), it is concluded that the properties of the HAZ material will not pose a problem to LACBWR operations.

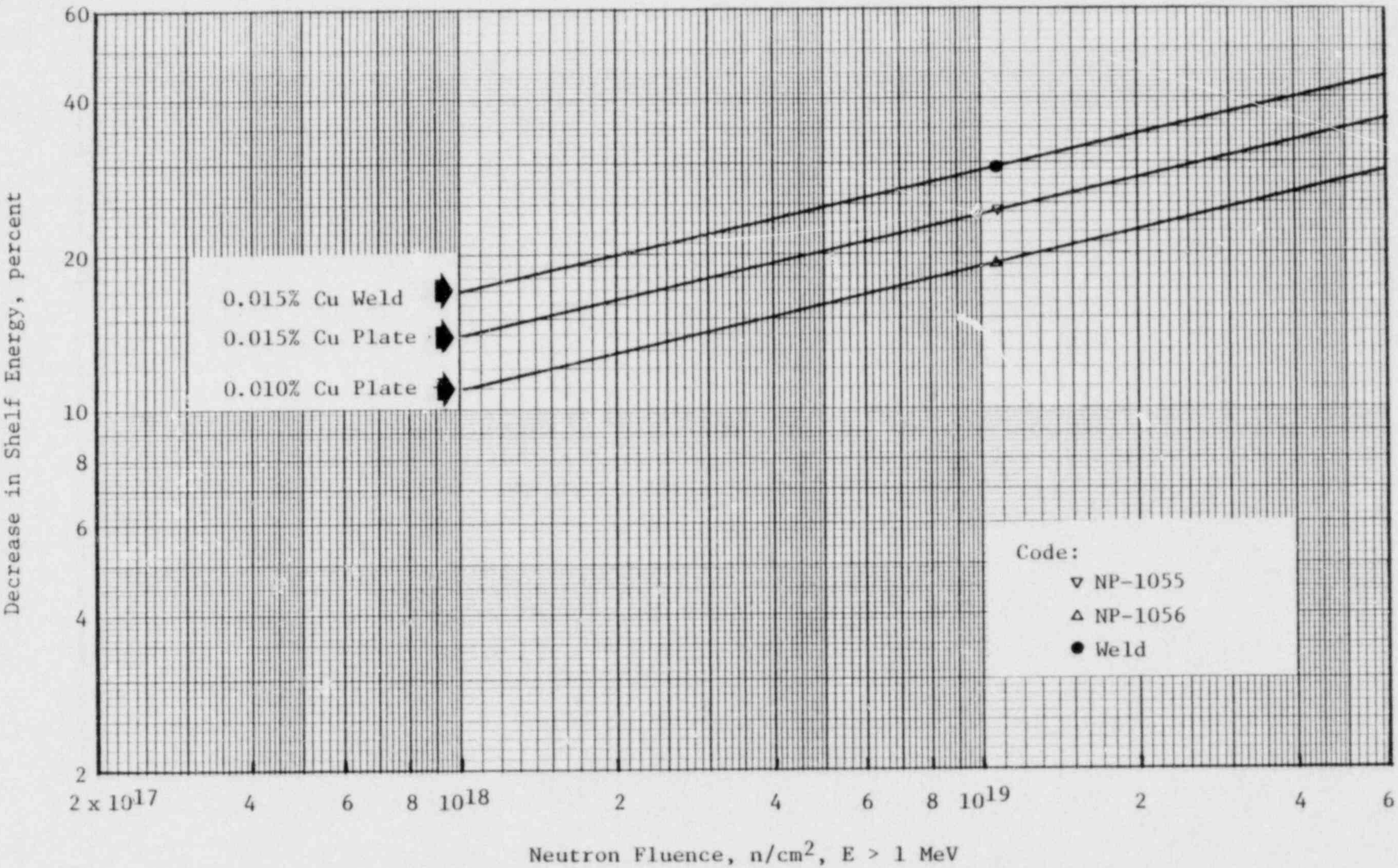

FIGURE 10. SHELF ENERGY DEGRADATION PROJECTIONS

It should be pointed out that this analysis has been based on the maximum fluence incident on the LACBWR pressure vessel I.D. surface. For the 4-in. thick pressure vessel wall, the fluence at the 1/4T position would be only 80 percent of that at the surface. Therefore, the results obtained on weld metal toughness from the 1980 capsules would be at a fluence equivalent to 10 EFPY at the 1/4T position. Therefore, there appears to be no need to modify the vessel material surveillance capsule removal schedule at this time. It is recommended that Capsules 5A, 5B, 10A, and 10B be removed after approximately 10 EFPY.

VI. REFERENCES

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2. ASTM E 208-69, "Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," 1972 Annual Book of ASTM Standards, Part 31.

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

4. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1974 Edition.

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

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13. Norris, E. B., "Tensile and Impact Properties of LACBWR Reactor Vessel Plate and Control Material," SwRI 1228-7-35, August 12, 1966.

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15. ASTM E 322, "Standard Method for Spectrochemical Analysis of Low Alloy Steels and Cast Irons Using an X-ray Fluorescence Spectrometer," 1974 Annual Book of ASTM Standards.

16. Hawthorne, J. R., et al, "Irradiation Effects on Reactor Structural Materials," Quarterly Progress Report, 1 August 1967 - 31 October 1967, NRL Memorandum Report 1833, November 15, 1967.

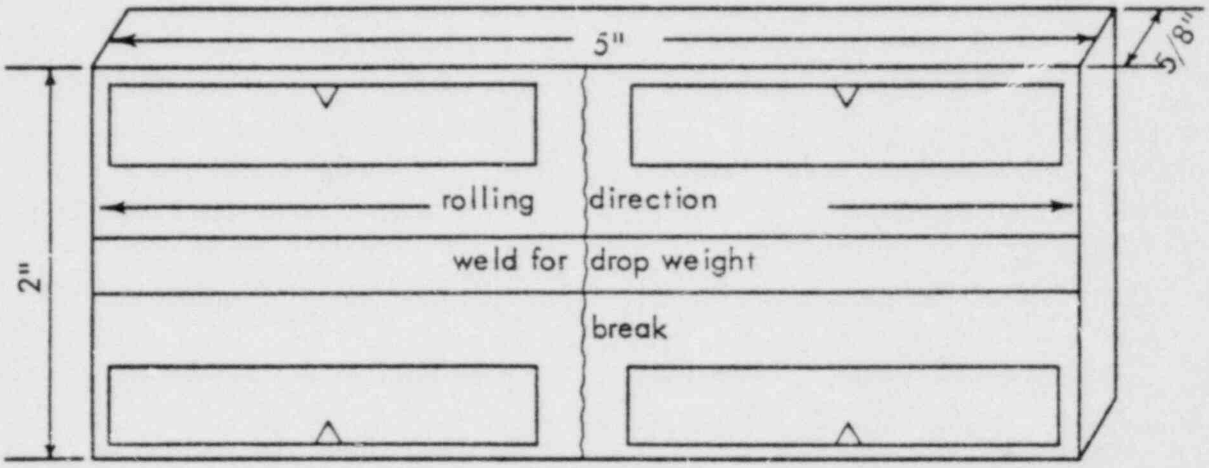
17. Norris, E. B., and Wylie, R. D., "Transition Temperatures of Selected Materials from LACBWR Main Steam and Forced Recirculation Systems," USAEC Report SwRI 1228 P7-37, June 28, 1967.

18. NRC Standard Review Plan, NUREG-75/087, November 24, 1975.

19. "LaCrosse Boiling Water Reactor--Reactor Vessel Materials, Fabrication and Inspection," ACNP-65534, May 10, 1965.

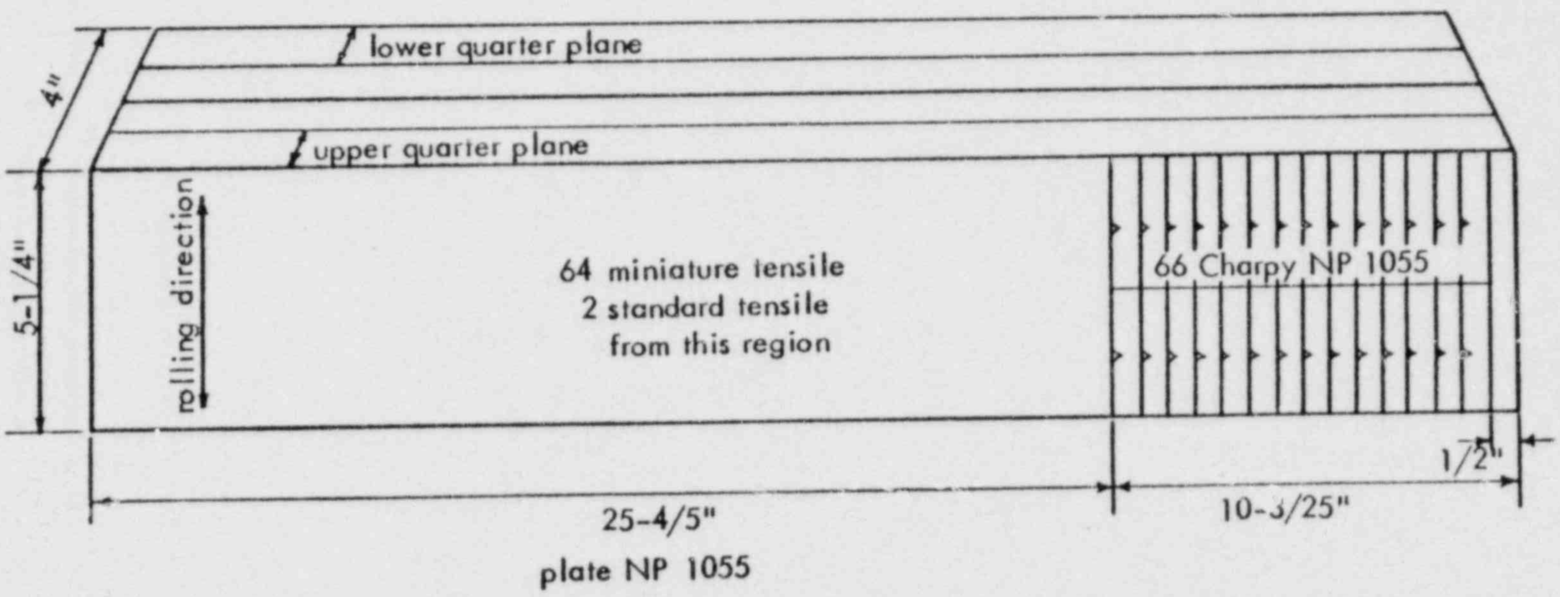
APPENDIX A

SKETCHES AND DRAWINGS FROM ACNP66513



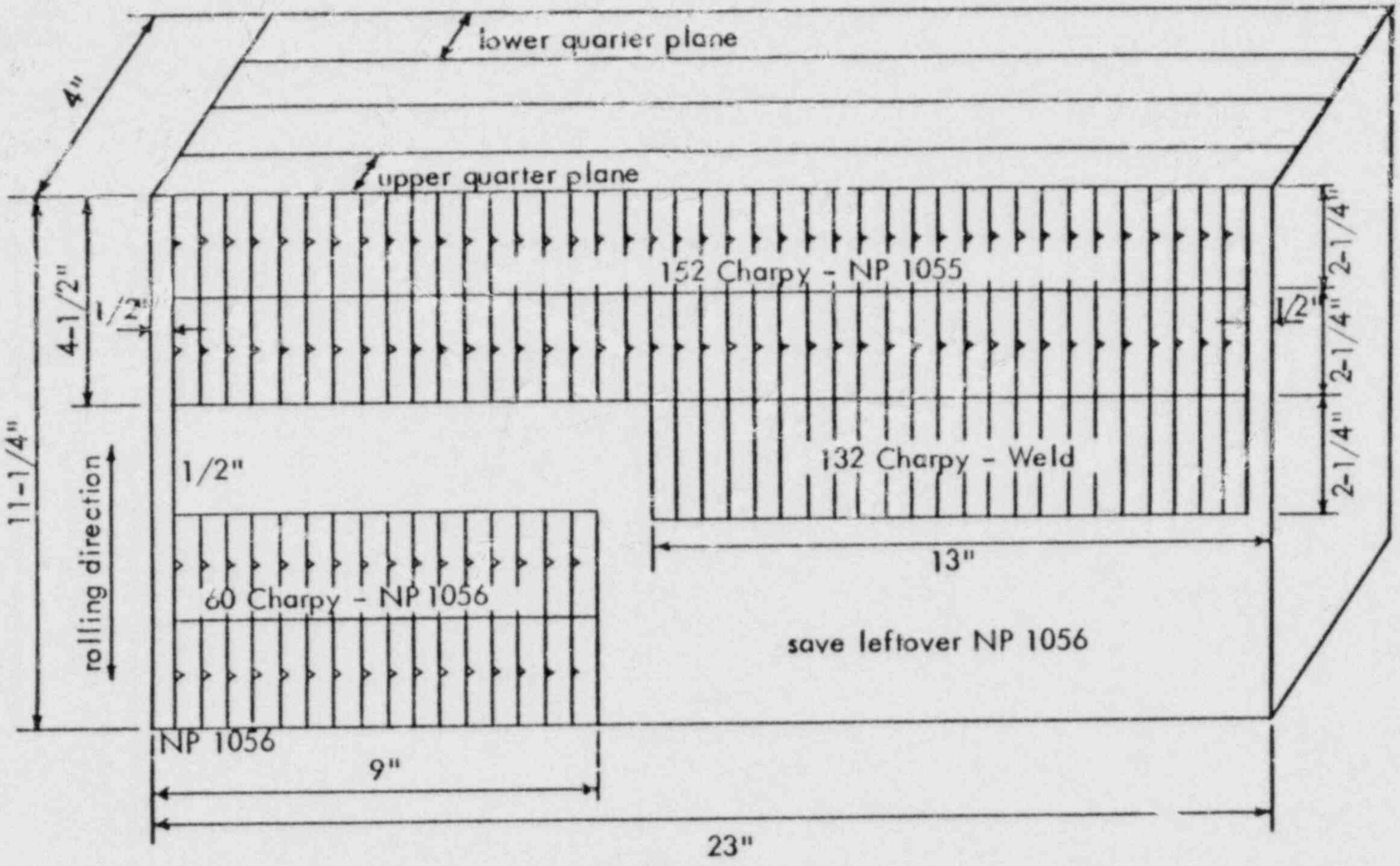
Typical for 18 NDT drop weight plates total of 72 Charpy V-notch specimens

CHARPY V-NOTCH TEST SPECIMENS MACHINED FROM BROKEN
NDT DROP WEIGHT TEST SPECIMENS FROM LACBWR
REACTOR VESSEL PLATE NP 1054



Charpy and tensile specimens taken parallel to rolling direction and upper and lower quarter plane only

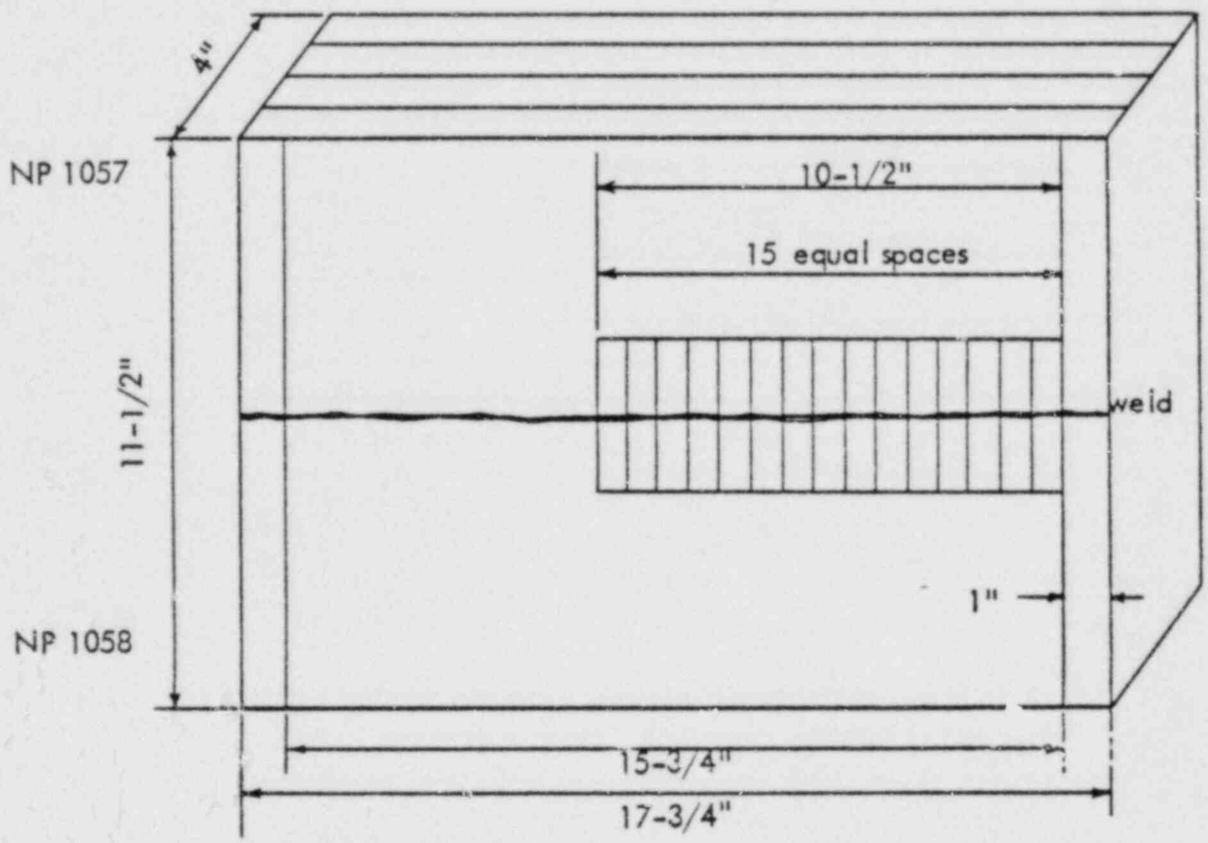
CHARPY V-NOTCH AND TENSILE TEST SPECIMENS
MACHINED FROM LACBWR REACTOR VESSEL PLATE NP 1055



Charpy in the plate taken from upper quarter plane and lower quarter plane parallel to rolling direction as shown, with notch oriented as shown. Charpy in the weld between the two plates taken 6 per thickness, as shown.

Charpy V-notch test specimens machined from LACBWR reactor vessel welded plates NP 1055 and NP 1056.

WELDED PLATES NP 1055 and NP 1056

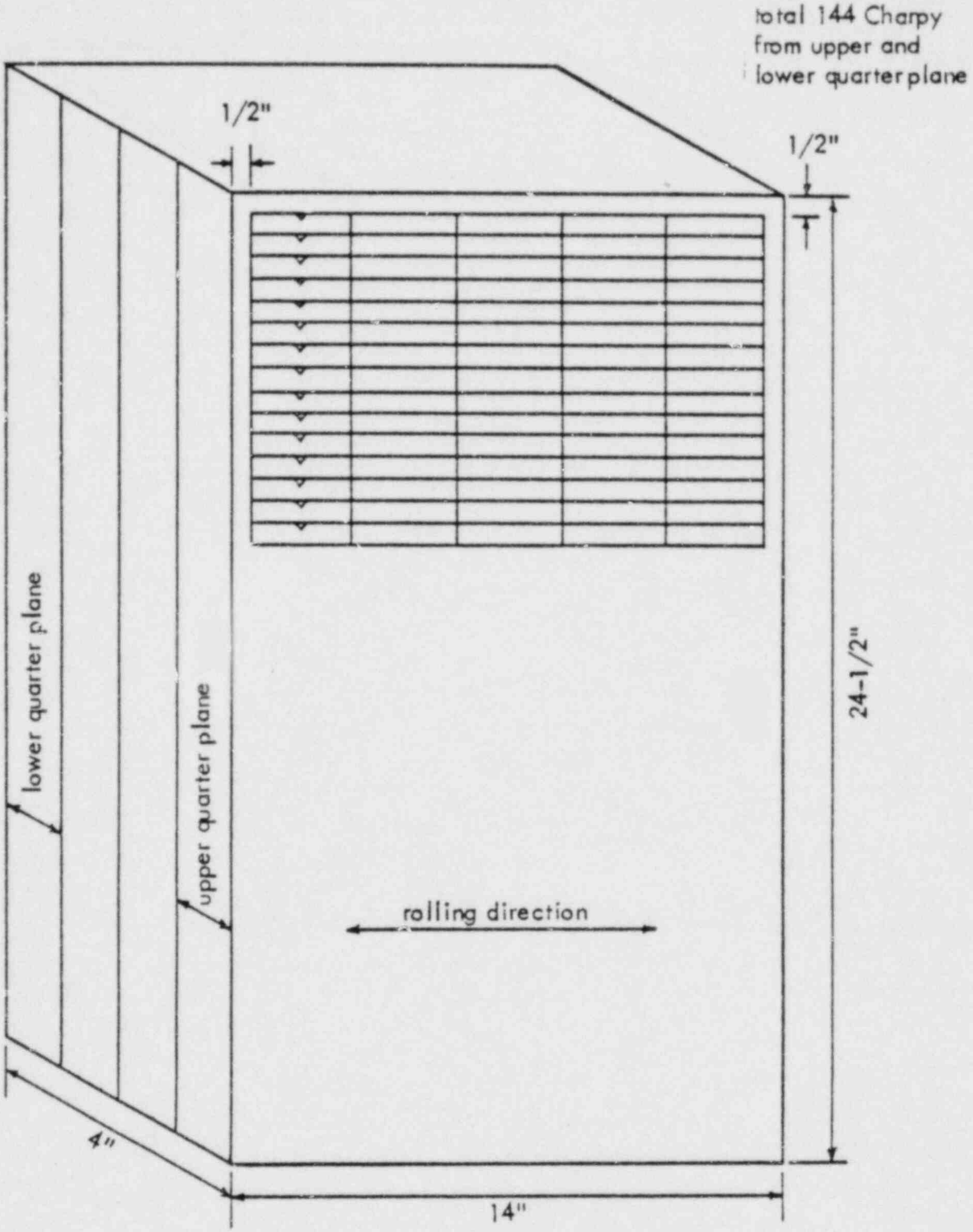


Tensile Weld Specimens

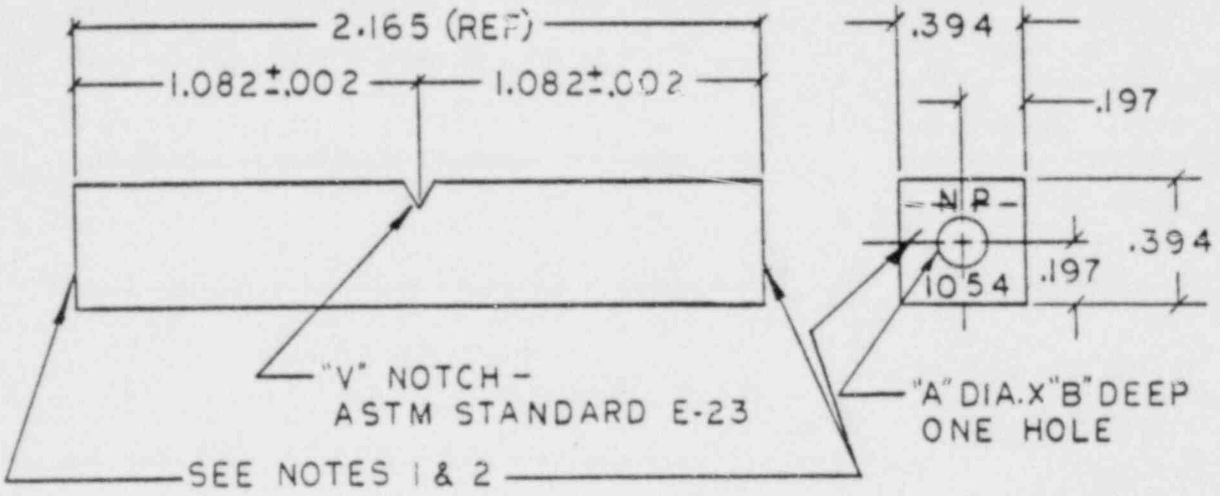
15 tensile specimens in each quarter plane of weld (4 planes full thinness)

60 tensile taken, 15 in each quarter plane of weld

CHARPY V-NOTCH AND MINIATURE TENSILE TEST SPECIMENS
MACHINED FROM LACBWR REACTOR VESSEL WELDED PLATES NP 1057 and NP 1058



CHARPY V-NOTCH TEST SPECIMENS MACHINED FROM
BATTELLE NORTHWEST STANDARD OR CONTROL MATERIAL (Ref. BNWL-CC-236)



MARK NO	MATERIAL IDENTIFICATION	"A"	"B"
001	NP-1054	—	—
002	NP 1056	—	—
003	NP 1055	—	—
004	NP 1055	.125	.250
005	WELD	—	—
006	STD.	—	—

NOTES:

1. EACH SPECIMEN IS TO BE STAMPED ON BOTH ENDS INDICATING ITS RESPECTIVE MATERIAL IDENTIFICATION.
2. STAMPING IS TO BE DONE WITH LOW STRESS STAMP.

UNLESS OTHERWISE NOTED MACHINING TOLERANCES ARE
1 PLACE DECIMAL ± .060
2 PLACE DECIMAL ± .030
3 PLACE DECIMAL ± .010
ANG. MACHINED SURFACE ± 30'
CHAMFER & WELD PREP ± 2°
MACHINED SURFACE TEXTURE 63 ✓
TOLERANCES DO NOT APPLY TO COMMERCIAL STOCK OR PARTS. DIMENSIONS SHOWN FOR SUCH ITEMS ARE NOMINAL.

DWG STATUS

AS BUILT

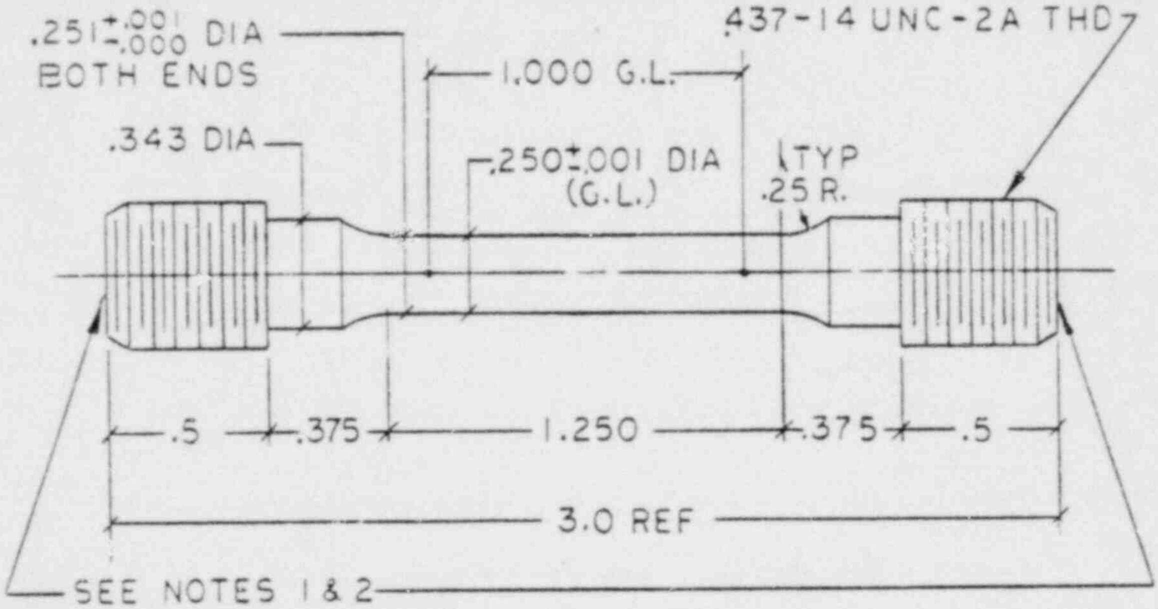
APPROVAL	DATE
DR. R.E. MILLER	9-3-65
CH'D. Sartori	9-9-65
APP.	
APP.	
APP.	

SCALE 2:1
BM
USED ON NEXT ASSY. 41-100-384-SC, 41-100-384-[illegible]
PROJECT LACBWR

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND

DESCRIPTION
MATERIAL SEE TABLE
WT R. F.
PART NAME IMPACT SPECIMEN
DWG. NO. 41-100-385-006 (T)
REV. 01

01	9-9-65
REVISIONS	



MARK NO	MATERIAL IDENTIFICATION
001	NP-1055
002	WELD

NOTES:

1. EACH SPECIMEN IS TO BE STAMPED ON BOTH ENDS INDICATING ITS RESPECTIVE MATERIAL IDENTIFICATION
2. STAMPING IS TO BE DONE WITH LOW STRESS STAMPS.

UNLESS OTHERWISE NOTED MACHINING TOLERANCES ARE

1 PLACE DECIMAL ± .060
2 PLACE DECIMAL ± .030
3 PLACE DECIMAL ± .010
ANG. MACHINED SURFACE ± 30'
CHAMFER & WELD PREP ± 2°

MACHINED SURFACE TEXTURE 63√

TOLERANCES DO NOT APPLY TO COMMERCIAL STOCK OR PARTS. DIMENSIONS SHOWN FOR SUCH ITEMS ARE NOMINAL.

APPROVAL	DATE
DR. R.E. MILLER	9-7-65
CH'D. Sartori	9-9-65
APP. ELR	9-9-65
APP.	
APP.	

SCALE 2:1

BM

USED ON NEXT ASSY 41-100-384-501, 41-100-384-502

PROJECT LACBWR

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
BETHESDA MARYLAND

DESCRIPTION

MATERIAL SEE TABLE

WT R. F.

PART NAME TENSILE SPECIMEN

DWG. NO. 41-100-386-002 (T)

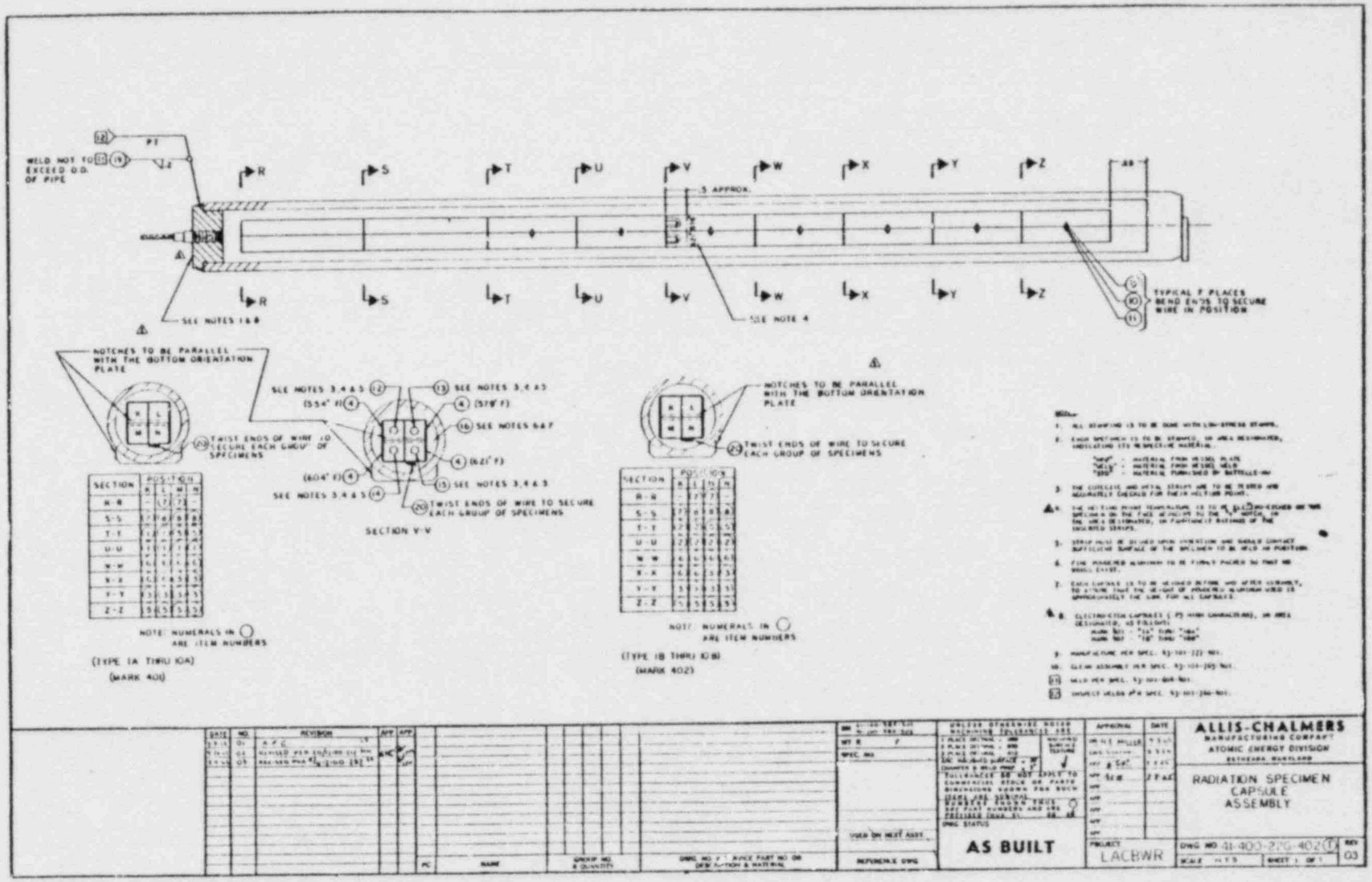
REV. 01

DWG STATUS

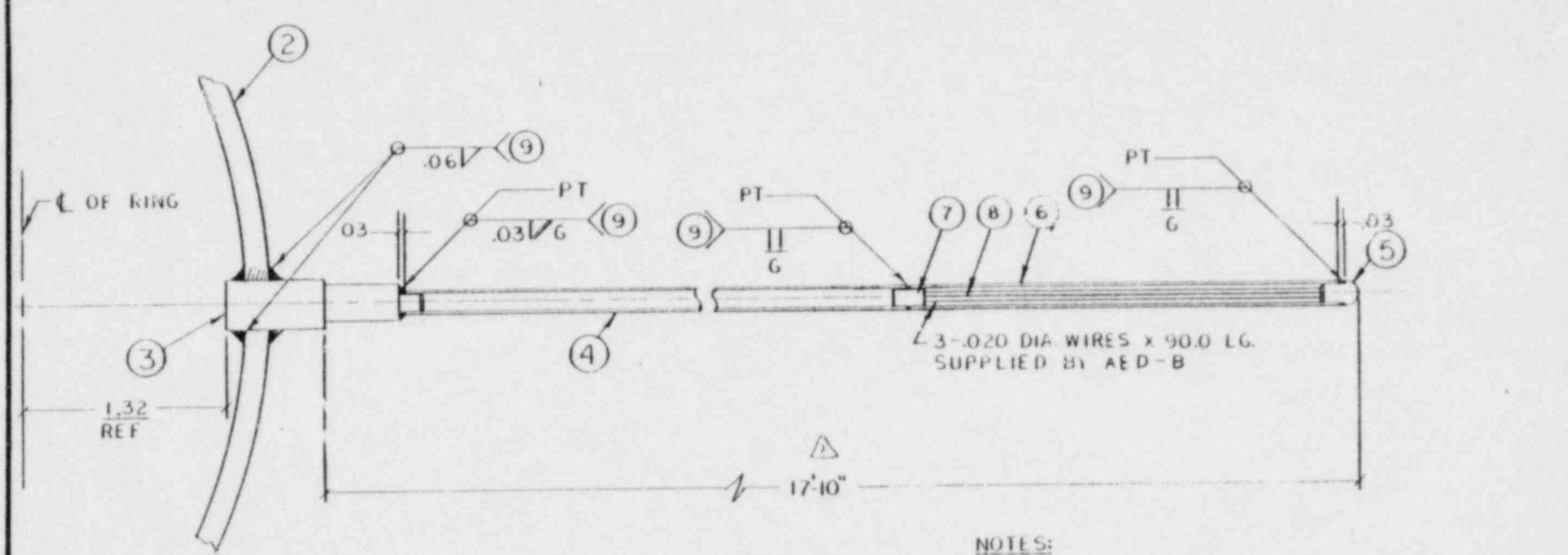
AS BUILT

01	9-9-65

REVISIONS



WELD NOT TO EXCEED O.D. OF PIPE
SEE NOTES 1 & 8
.5 APPROX.
SEE NOTE 4
TYPICAL 7 PLACES BEND ENDS TO SECURE WIRE IN POSITION
NOTCHES TO BE PARALLEL WITH THE BOTTOM ORIENTATION PLATE
TWIST ENDS OF WIRE TO SECURE EACH GROUP OF SPECIMENS
NOTE: NUMERALS IN ◯ ARE ITEM NUMBERS
(TYPE 1A THRU 10A)
(MARK 401)
SEE NOTES 3,4 & 5
SEE NOTES 6 & 7
SECTION V-V
(TYPE 1B THRU 10B)
(MARK 402)
AS BUILT
ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
RADIATION SPECIMEN CAPSULE ASSEMBLY
LACBWR
03



NOTES:

2. CLEAN FINAL ASSEMBLY PER SPEC. 43-101-765-401

2 ASSEMBLIES REQD

REVISIONS		
02	10-26-66	1/60006B-11
	As-shown Revised ⚠(D-4) DIM WAS 10-0	
01	9-9-65	

UNLESS OTHERWISE NOTED MACHINING TOLERANCES ARE
1 PLACE DECIMAL ± .060
2 PLACE DECIMAL ± .030
3 PLACE DECIMAL ± .010
ANG MACH. SURFACE ± 30'
CHAMFER & WELD PREP ± 2°
MACHINED SURFACE TEXTURE ✓
TOLERANCES DO NOT APPLY TO COMMERCIAL STOCK OR PARTS. DIMENSIONS SHOWN FOR SUCH ITEMS ARE NOMINAL.

DWG STATUS

AS BUILT

APPROVAL	DATE
DR R E M.	
CHD	9-9-65
APP E R	9-9-65
APP A C H	9-9-65
APP	

SCALE 2:1
BM 41-100-390-501
USED ON NEXT ASSY
PROJECT LACBWR

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND

DESCRIPTION
MATERIAL
WT R F
PART NAME DOSIMETRY WIRE HOLDER
DWG NO. 41-200-294-401
REV 02

APPENDIX B

UNIRRADIATED CHARPY V-NOTCH AND TENSILE DATA

TABLE B-1

TENSILE TEST RESULTS ON HEAT NP1055 AT AMBIENT TEMPERATURE
(Unirradiated)

Specimen No.	Gage Diameter (in.)	0.2% *YS* (ksi)	*UTS* (ksi)	*RA* (%)	Elong. (%)
1	0.506	64.1	85.3	70.9	26.7
2	0.505	62.8	85.8	70.8	27.5
3	0.250	64.7	87.6	72.5	30.9
4	0.250	63.6	88.1	74.9	27.7
5	0.250	64.2	87.6	72.1	27.3
6	0.250	65.2	88.6	71.3	27.6

TABLE B-2

RESULTS OF CHARPY V-NOTCH TESTS ON HEAT NP1055
(Unirradiated)

Temp. (°F)	Absorbed Energy (ft-lb)	Fracture Appearance (% Shear)	Lateral Expansion (mils)
70	120+	100	87
70	120+	100	93
40	120.0	100	99
40	120.0	100	90
10	120.0	100	92
10	97.5	60	76
−20	68.5	30	61
−20	89.5	25	73
−50	58.0	15	51
−50	41.5	10	37
−70	57.0	8	51
−70	47.0	8	42
−80	27.5	2	25
−80	38.0	2	33
−80	9.5	2	7
−80	23.5	2	12
−100	16.5	1	14
−100	11.5	1	11

TABLE B-3

RESULTS OF CHARPY V-NOTCH TESTS ON BNW REFERENCE HEAT
(Unirradiated)

Temp. (°F)	Absorbed Energy (ft-lb)	Fracture Appearance (% Shear)	Lateral Expansion (mils)
70	88.0	100	70
70	87.5	100	70
40	86.0	100	64
40	87.0	100	59
10	74.0	95	55
10	76.0	90	65
20	47.0	30	41
20	59.5	35	50
-50	32.0	20	30
-50	33.0	25	32
60	37.0	15	27
60	30.5	18	29
70	26.5	15	26
70	30.5	18	29
80	14.0	10	14
80	19.5	8	19
80	24.0	10	21
80	27.5	10	24
100	9.5	5	9
100	15.0	5	15

TABLE B-4

RESULTS OF CHARPY V-NOTCH TESTS ON WELD METAL
(Unirradiated)

Temp. (°F)	Absorbed Energy (ft-lb)	Fracture Appearance (% Shear)	Lateral Expansion (mils)
70	43.5	70	69
70	62.5	80	63
10	51.5	60	54
10	47.5	65	49
-20	44.0	30	46
-20	25.5	25	26
-50	22.0	20	23
-50	23.5	18	24
-80	7.5	5	9
-80	9.5	5	11
-100	8.5	2	11
-100	10.5	5	7

TABLE B-5

RESULTS OF CHARPY V-NOTCH TESTS ON HEAT NP1056
(Unirradiated)

Temp. (°F)	Absorbed Energy (ft-lb)	Fracture Appearance (% Shear)	Lateral Expansion (mils)
150	53.5	95	51
150	58.0	90	52
130	54.0	90	50
130	59.0	90	55
110	57.0	90	50
110	63.5	90	54
90	54.0	90	45
90	56.5	80	50
70	53.5	60	51
70	59.0	50	42
70	47.0	60	51
50	42.5	40	43
50	40.5	35	34
40	29.5	30	26
40	38.0	30	33
30	29.0	25	24
30	34.5	30	31
10	23.0	20	20
10	23.0	20	20
−10	19.5	10	15
−10	26.5	10	22

APPENDIX C

DISCRETE ORDINATE TRANSPORT ANALYSIS

APPENDIX C

DISCRETE ORDINATE TRANSPORT ANALYSIS

A. Background

The LACBWR Safety Review Committee, at their meeting of October 4, 1973, questioned if the fast neutron irradiation of the vessel is proceeding at a faster rate than envisioned in the design calculations given in the LACBWR Safeguards Report ACNP65544. The committee's Recommendation #16 was to resolve this question of vessel NDT changes near the steam separators. This is a very important consideration because information on the neutron flux distribution is the primary link relating surveillance capsule and pressure vessel material property changes.

An increase in the NDT temperature of the vessel steel may be expected when the fast neutron fluence ($E > 1$ MeV) to which the steel is exposed exceeds a threshold value of approximately 10^{17} n/cm^2, $E > 1$ MeV. The fast neutron flux intensity in the vicinity of the pressure vessel boundary varies axially, radially and azimuthally. The only portion of the vessel which is expected to receive a fast neutron fluence above the threshold value for radiation damage during the design life is the vessel wall opposite the core, the maximum exposure generally occurring opposite the vertical center of the core. However, local perturbations in fast neutron flux within this region result from geometric as well as material differences.

Since it is difficult, if not impossible, to place the material surveillance capsules exactly at one or more points of maximum fast flux, the reactor design calculations should provide the lead factor(s) which relate the fast neutron flux at the surveillance capsule locations to the maximum fast neutron flux expected anywhere on the vessel wall I.D. ACNP65544 indicates that the incident neutron flux on the pressure vessel at the core centerline is expected to be 1.53×10^{10} n/cm^2 sec > 1 MeV, while that on each capsule is predicted to be 3.03×10^{10}. Thus, a single acceleration factor of 1.98 is defined for all surveillance capsules.

The answer to DRL Question III-18 confirms that the effects of steam voids in the separator and downcomer regions were considered in calculating the neutron exposure of the pressure vessel. However, by coincidence, the estimated 15 percent increase in fast flux due to void distribution was cancelled by the 15 percent overestimate of fast flux obtained from the P1MG program because of the energy group structure selected.

The material surveillance capsules installed in the LACBWR vessel are located between the steam separators, but they are shielded from the core by the steam separators to varying degrees, as indicated by Figure 2 in ACNP66513, February 1966. Therefore, all capsules are not located at the position assumed in the original reactor design calculations. For example, the first two capsules removed (1A and 1B) are located about 5.5° from the true center between two steam separators. Examination of Figures 1 and 2 in ACNP66513 indicates that the vessel wall dosimeters (located outside the thermal shield) are also positioned between steam separators so that they cannot necessarily be relied upon to provide an experimental confirmation of maximum vessel wall neutron flux intensity.

It was recommended by SwRI that the original LACBWR reactor design calculation be reviewed to determine the limitations of the calculated neutron flux values contained in the LACBWR Safeguards Report. This review should also be directed to satisfy the recommendation in SwRI Topical Report No. 1, "Analysis of the First Vessel Material Surveillance Capsule Withdrawal from LaCrosse Boiling Water Reactor," concerning the LACBWR flux spectra calculations. It was also recommended that should the original design calculations not be available, or prove to be of insufficient detail, the question of vessel NDT changes near the steam separators can be resolved by performing a new set of calculations using a spectral computer code such as P3MG at radial positions between the steam separators and through the steam separators.

After it was determined that the original design calculations were unavailable, a list of available computer codes was compiled. The two-dimensional Discrete Ordinates Transport code, DOT 3.5, was selected from the Radiation Shielding Information Center (RSIC) computer code collection. A 40-group coupled neutron and gamma-ray cross section package, CASK, was also obtained from RSIC.

B. DOT 3.5 Analysis

The DOT 3.5 code was used to calculate the neutron spatial and energy distribution in the LACBWR vessel with 0 percent, 20 percent and 40 percent voids in the steam separators. The results were compared for consistency with data obtained from the vessel material surveillance program neutron dosimeters to determine which level of voids was most appropriate. In the performance of these calculations, the LACBWR vessel and internals were modeled two-dimensionally in a plane perpendicular to the vertical core axis. A one-eighth segment, with one boundary parallel to a compass point and the other 45° to the compass point, was taken to be representative because of the symmetry involved. The boundaries of the core, steam separators, thermal shield and pressure vessel were then described in R-θ coordinates, as shown in Figure C-1.

The core was subdivided into two regions, an inner region with the operating control rods inserted and an outer region with all control rods withdrawn. The core materials within each region were homogenized over their respective areas, assuming that one-half of the shrouds were stainless steel and the remainder were Zircaloy. The stainless steel steam separators, homogenized over the area they enclosed, and the stainless steel thermal shield were taken as a mixture of 18 percent Cr, 8 percent Ni and 74 percent Fe. The pressure vessel was assumed to be 98 percent iron and the coolant as pure water.

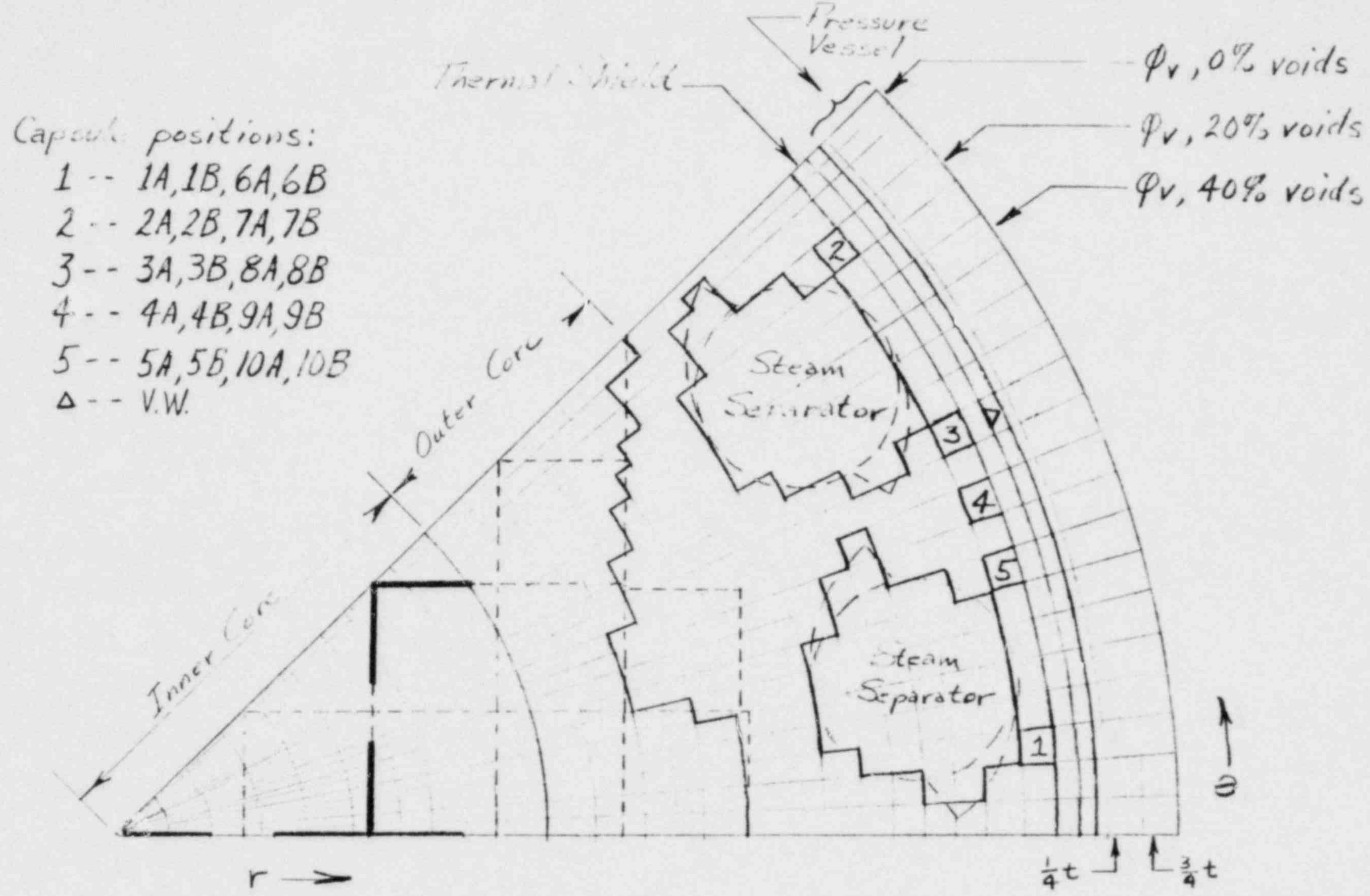


FIGURE C-1. TWO-DIMENSIONAL MODEL OF LACBWR VESSEL

The following information was computed for each level of void content:

1. ϕ_v, the maximum neutron flux density (E > 1 MeV) incident on the vessel I.D. surface.

2. ϕ_c, the neutron flux density (E > 1 MeV) for each surveillance capsule location.

3. $\bar{\sigma}$, the spectrum-averaged cross section for the $^{54}Fe(n,p)^{54}Mn$ reaction at each surveillance capsule location.

Then ϕ_m, the measured neutron flux density for each removed surveillance capsule, was calculated using the iron dosimeter activities and the appropriate $\bar{\sigma}$.

The azimuthal location of the maximum neutron flux density incident on the vessel I.D. surface, ϕ_v, is affected by the void content in the steam separators, see Figure C-1. With 0 percent voids, the azimuthal location of ϕ_v is at the point on the vessel having the closest approach to the core boundary, as would be expected. The presence of voids in the steam separators, however, reduces the attenuation of neutron flux density by the coolant which in turn results in the moving of the peak flux density to a region behind the nearest steam separator as well as an overall increase in the calculated neutron flux density on the vessel wall.

The effect of void content in the steam separators on the lead factors (ϕ_c/ϕ_v) for the surveillance capsules which have been removed is shown in Table C-1. This effect is relatively small because the presence of voids in the steam separators has similar effects on both the capsule and vessel neutron flux densities.

A comparison of calculated (ϕ_c) and measured (ϕ_m) neutron flux densities as a function of void content in the steam separators is presented in Table C-2 and Figure C-2. Referring to the latter, several features are of interest:

1. The assumption of 0 percent voids leads to completely inconsistent results between the measured and calculated neutron flux densities at the locations from which capsules have been removed.

2. There is an apparent difference in behavior between those capsules removed in 1972 and those removed in 1975. This might be explained in three ways: (1) the average void content before the 1972 refuelling was lower than that after the 1972 refuelling; (2) the void content varies from steam separator to steam separator; (3) the model may be more consistent with

TABLE C-1

NEUTRON FLUX DENSITY LEAD FACTORS AS A FUNCTION OF VOID CONTENT IN THE STEAM SEPARATORS

Capsule Identification	Lead Factor[(a)]		
	0% Voids	20% Voids	40% Voids
1A, 1B, 6A, 6B	1.41	1.60	1.62
2A, 2A, 7A, 7B	2.15	2.28	2.13
3A, 3B, 8A, 8B	1.51	1.67	1.66
4A, 4B, 9A, 9B	1.46	1.49	1.37
5A, 5B, 10A, 10B	1.38	1.52	1.50
V.W. Dosimeter	0.86	0.94	0.96

(a) ϕ_c/ϕ_v

TABLE C-2

COMPARISON OF CALCULATED AND MEASURED NEUTRON FLUX DENSITIES
AS A FUNCTION OF VOID CONTENT IN THE STEAM SEPARATORS

Capsule Ident.	Iron Dosimeter ASAT, dps/mg	0% Voids				20% Voids				40% Voids			
		ϕ_c (a)	$\bar{\sigma}$ (b)	ϕ_m (c)	ϕ_v (d)	ϕ_c	$\bar{\sigma}$	ϕ_m	ϕ_v	ϕ_c	$\bar{\sigma}$	ϕ_m	ϕ_v
1A	7.705×10^3	4.18	.189	6.50	4.61	5.87	.183	6.71	4.19	8.48	.176	6.98	4.31
1B	6.902×10^3	4.18	.189	5.82	4.13	5.87	.183	6.01	3.76	8.48	.176	6.25	3.86
2A	1.196×10^4	6.37	.179	10.65	4.95	8.39	.174	10.96	4.81	11.16	.168	11.35	5.33
7B	1.109×10^4	6.37	.179	9.88	4.60	8.39	.174	10.16	4.46	11.16	168	10.52	4.94
9A	7.806×10^3	4.32	.198	6.28	4.30	5.47	.192	6.48	4.35	7.15	.184	6.76	4.93
9B	7.902×10^3	4.32	.198	6.36	4.36	5.47	.192	6.56	4.40	7.15	.184	6.85	5.00
V.W.	4.466×10^3	2.54	.168	4.24	4.93	3.46	.160	4.45	4.73	5.00	.154	4.62	4.81
V.W.	4.600×10^3	2.54	.168	4.36	5.07	3.46	.160	4.58	4.87	5.00	.154	4.76	4.96
Average ϕ_v, excluding 1A and 1B					4.70				4.60				5.00

(a) Neutron flux density, $n/cm^2/sec \times 10^{-10}$, calculated with DOT 3.5 Code
(b) Spectrum averaged cross section, barns, for $^{54}Fe(n,p)^{54}Mn$ from DOT 3.5 Code
(c) Neutron flux density, $n/cm^2/sec \times 10^{-10}$, using A_{SAT} and $\bar{\sigma}$
(d) Maximum neutron flux density incident on vessel wall using ϕ_m and lead factor from Table I

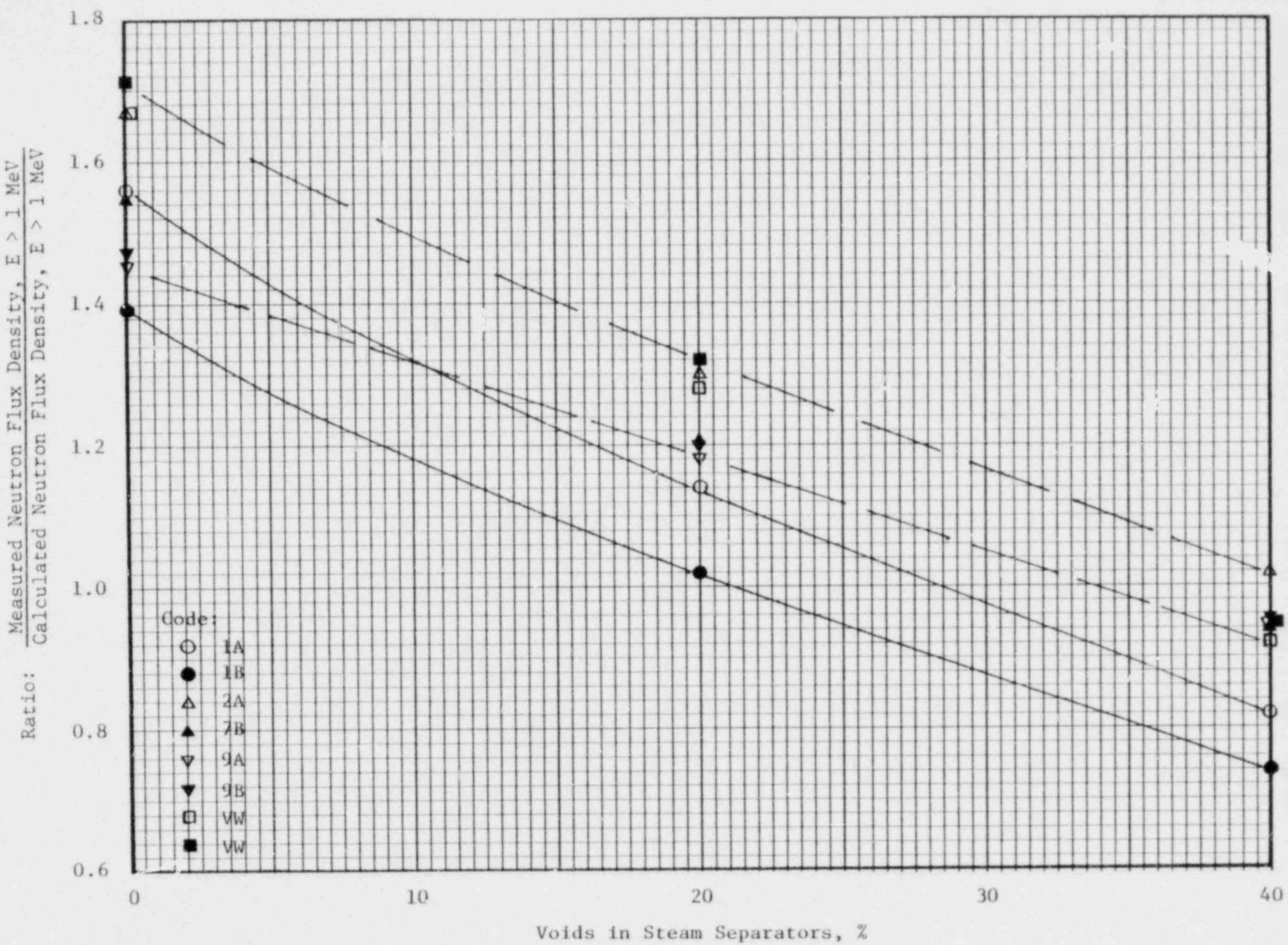


FIGURE C-2. EFFECT OF STEAM VOID CONTENT ON MEASURED-TO-CALCULATED NEUTRON FLUX RATIO actual operation after 1972. (Before 1972 the center of the core was less heavily rodded and, therefore, the power, on the average, was probably shifted to the core center and away from the vessel wall.)

3. If it is assumed that the average void content was the same in all steam separators prior to and since the 1972 refuelling, an average value of 30 percent voids would provide for the most consistency between calculated and measured neutron flux densities for all capsules removed to date.

4. It appears that assuming average void contents in excess of 40 percent would lead to inconsistencies similar to those noted in 1. above.

C. Summary

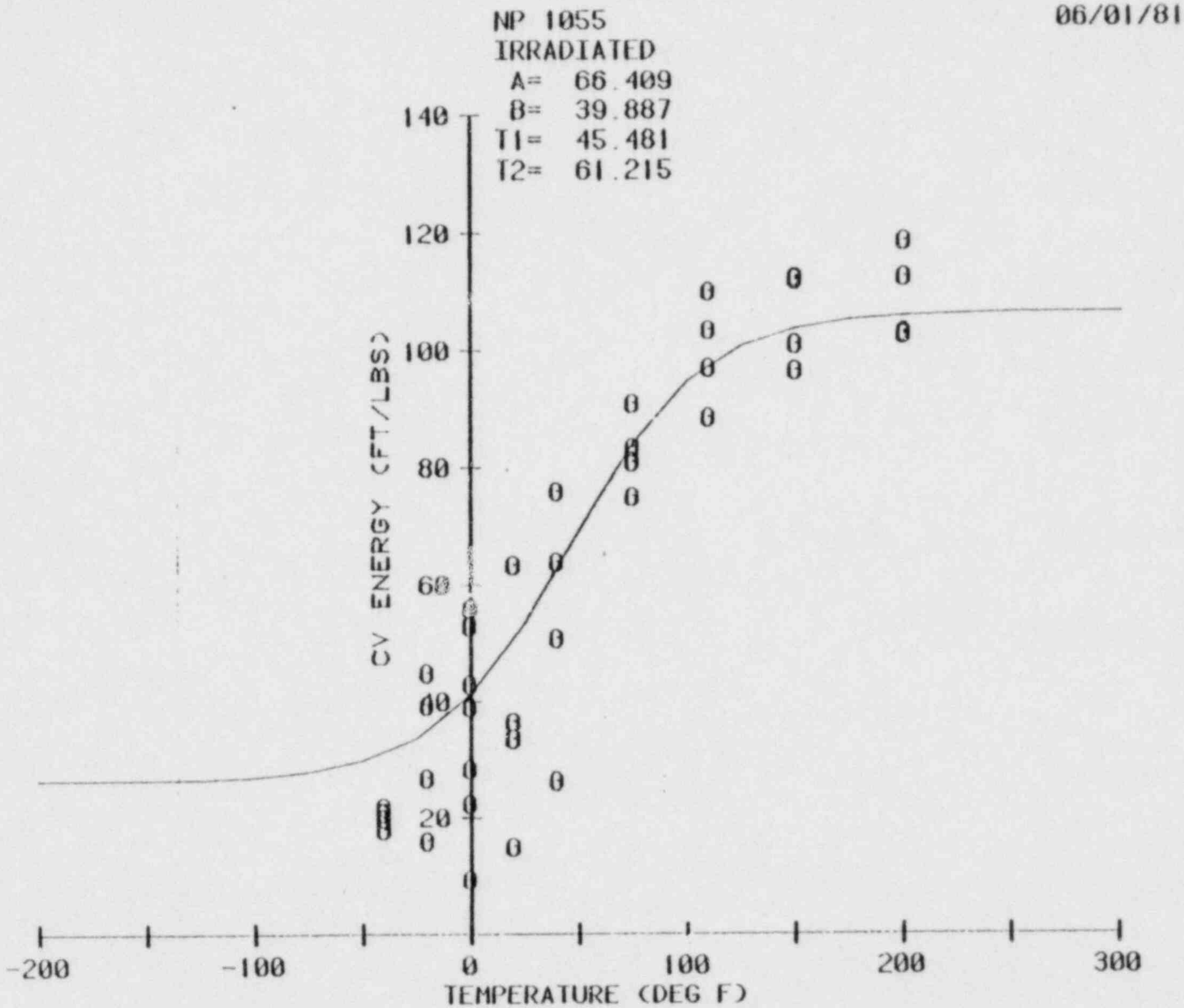
One cannot expect to obtain perfect agreement between calculated and measured results since operations would be expected to vary and the dosimeters integrate the effect of operating variables over the exposure period. Based on the calculations made, the conservative approach would be to assume an average void content of 40 percent in the analysis of the capsules removed during the 1975 refuelling outage because this leads to a higher value of vessel wall flux, as shown in Table C-2.

Assuming an incident flux of 5.0×10^{10} ($E > 1$ MeV) on the I.D. surface of the LACBWR vessel, the projected fast fluence after 20 full power years of operation is 3.2×10^{19} n/cm^2 ($E > 1$ MeV). This is very close to that estimated from the analysis of capsules 1A and 1B (Topical Report No. 1, SwRI Project 02-3467, "Analysis of the First Vessel Material Surveillance Capsule Withdrawal from LaCrosse Boiling Water Reactor," March 23, 1973), but more than three times that predicted in ACNP-65544.

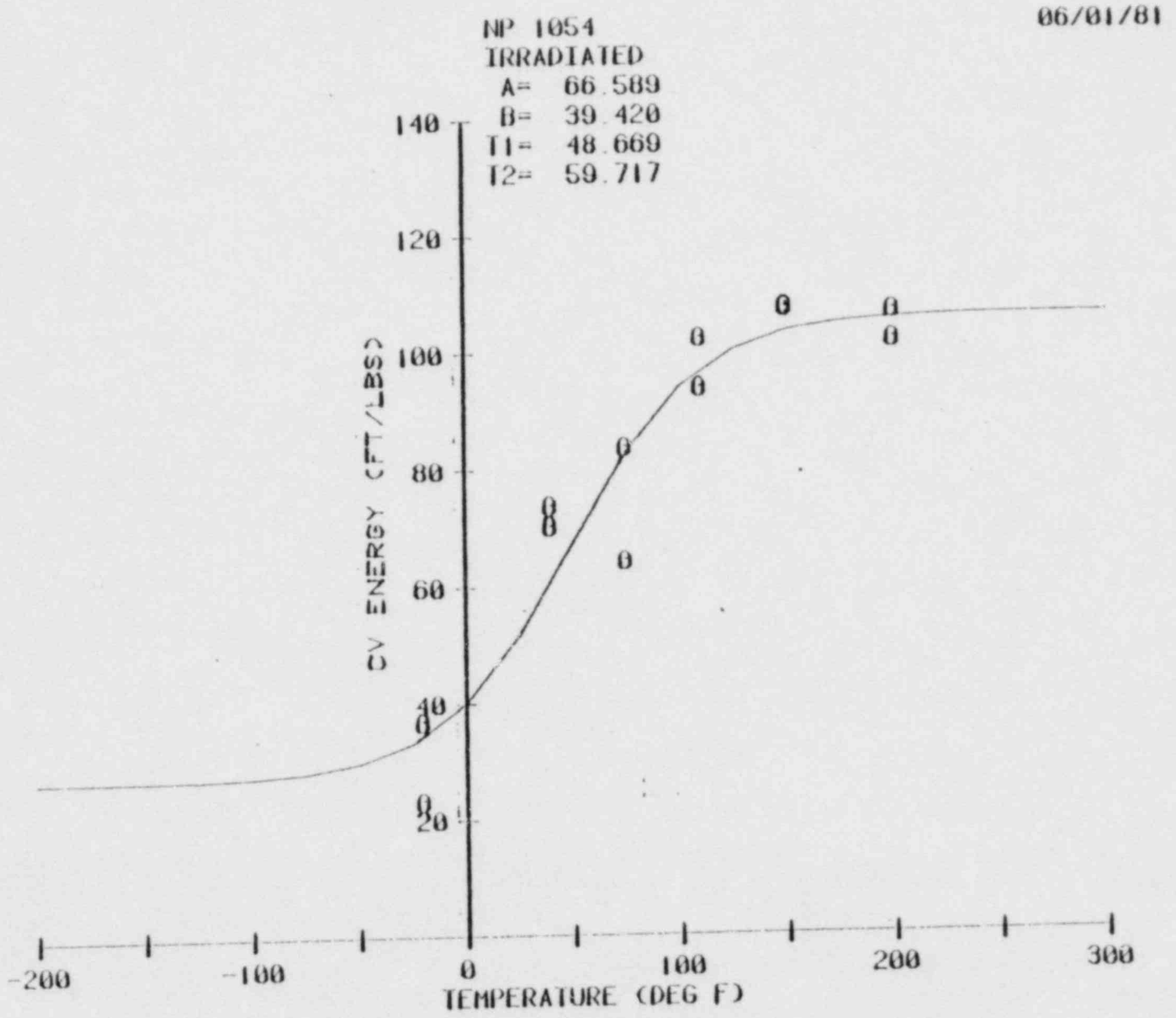
Except for capsules 2A, 2B, 7A and 7B, the lead factors computed in this analysis are quite different from the value of 2 suggested in ACNP-65544. However, the neutron flux density at the vessel wall dosimeter locations was computed to be nearly equal to the maximum value incident on the pressure vessel wall as planned. The variation in flux density at the capsule locations predicted with the DOT 3.5 code are supported by the variations in activities of the neutron dosimeters contained in the surveillance capsules removed to date.

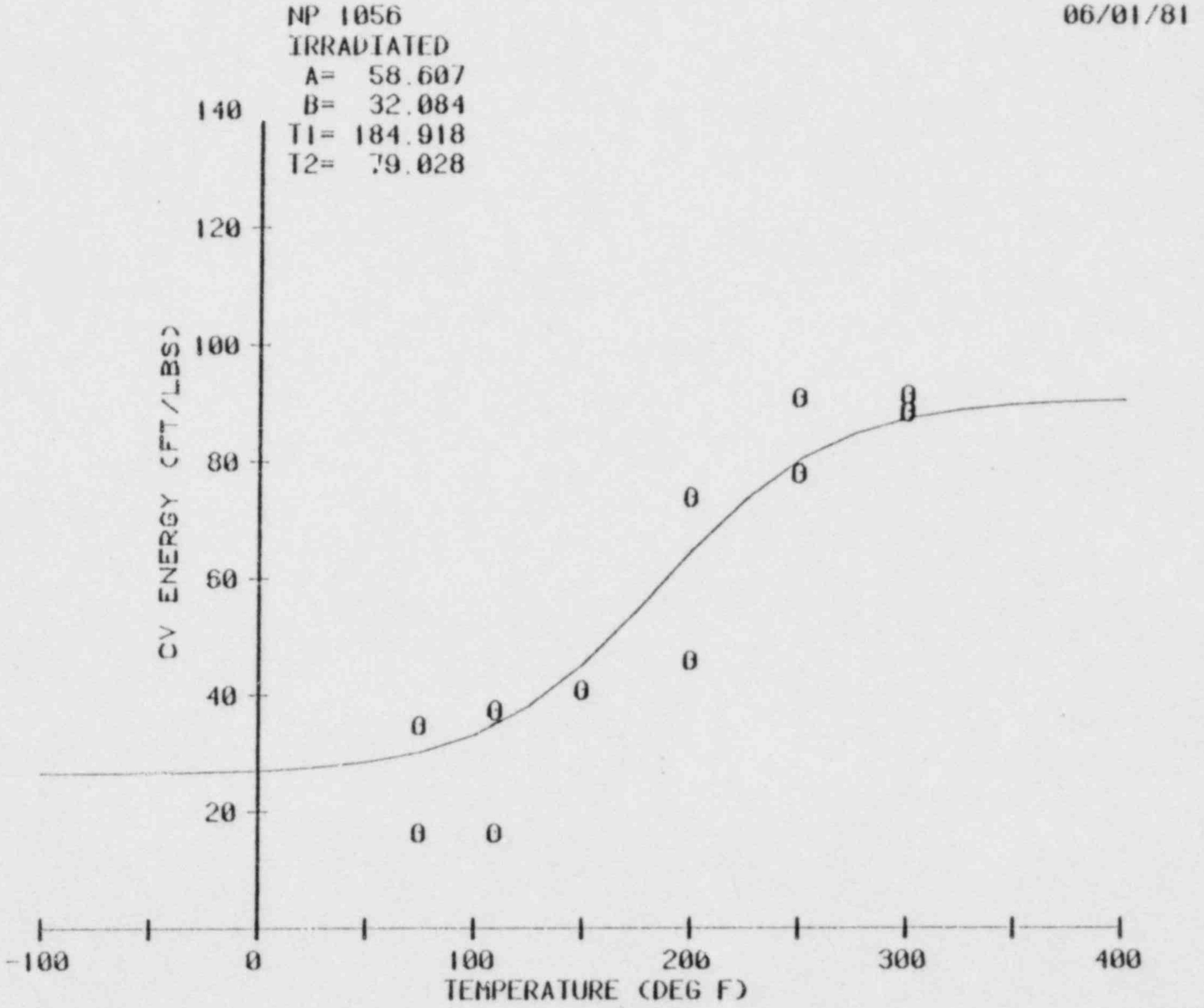
APPENDIX D

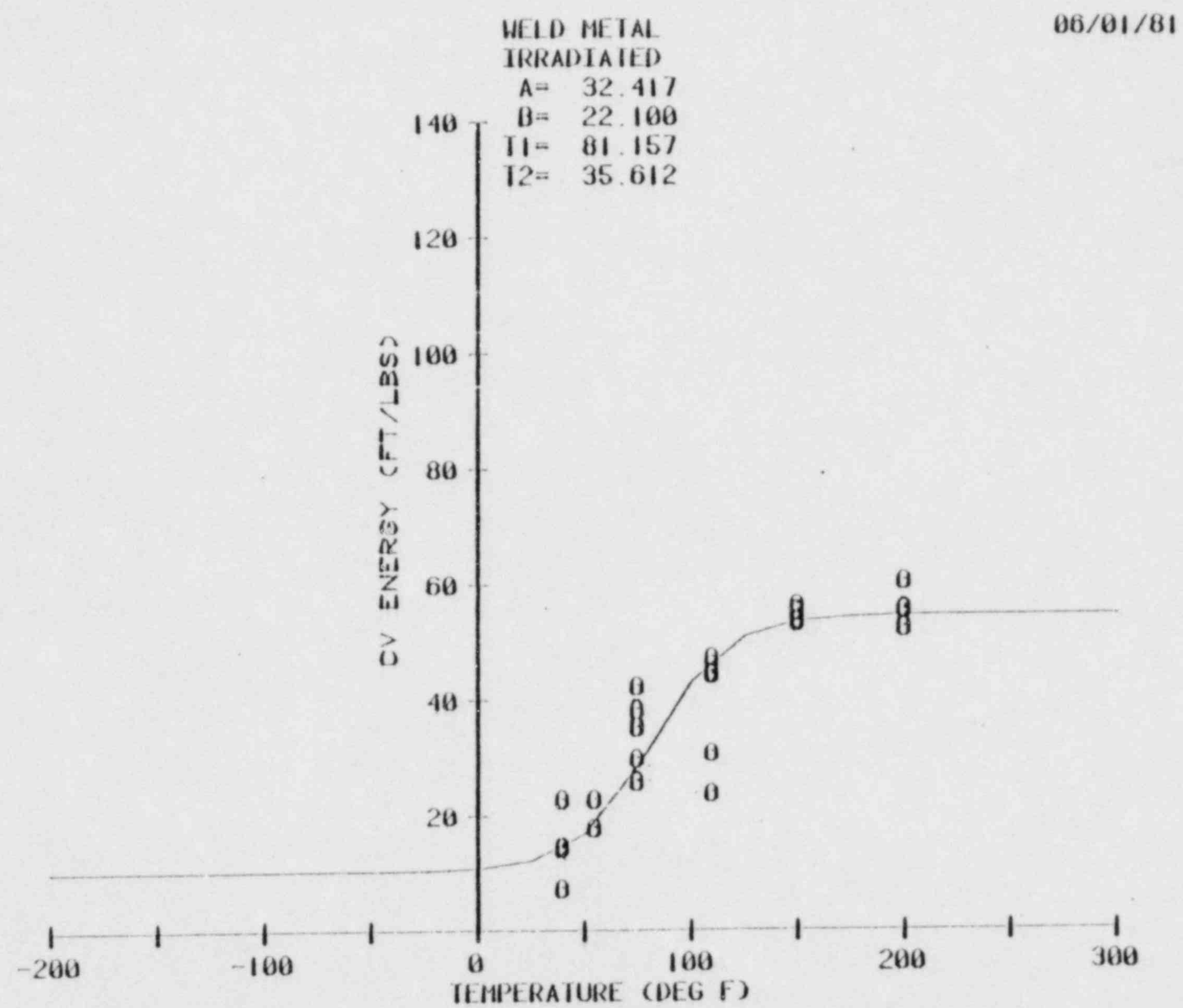
TANH-FIT CHARPY CURVES
(CAPSULES 3A, 3B, 8A, AND 8B)

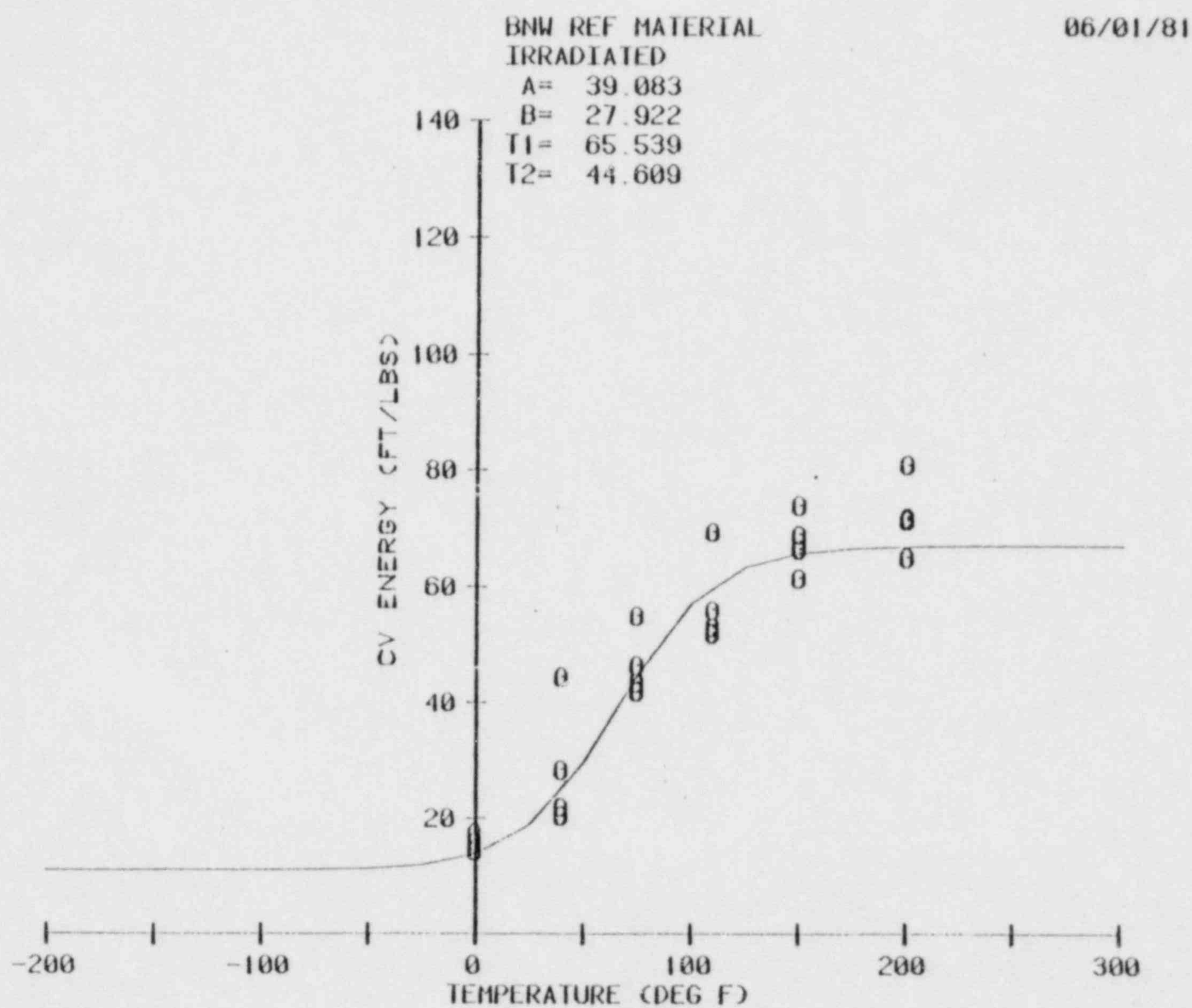


NP 1055
IRRADIATED
A= 66.409
B= 39.887
T1= 45.481
T2= 61.215
06/01/81
CV ENERGY (FT/LBS)
TEMPERATURE (DEG F)


NP 1054
IRRADIATED
A= 66.589
B= 39.420
T1= 48.669
T2= 59.717
06/01/81
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)


NP 1056
IRRADIATED
A= 58.607
B= 32.084
T1= 184.918
T2= 79.028
06/01/81
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-100
0
100
200
300
400
TEMPERATURE (DEG F)


06/01/81
WELD METAL
IRRADIATED
A= 32.417
B= 22.100
T1= 81.157
T2= 35.612
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)


BNW REF MATERIAL
IRRADIATED
A= 39.083
B= 27.922
T1= 65.539
T2= 44.609
06/01/81
CV ENERGY (FT/LBS)
140
120
100
80
60
40
20
-200
-100
0
100
200
300
TEMPERATURE (DEG F)