NRL Report 7917

Further Observations on A533-B Steel Plate Tailored for Improved Radiation Embrittlement Resistance

J. RUSSELL HAWTHORNE

Thermostructural Materials Branch
Engineering Materials Division

September 22, 1975

NAVAL RESEARCH LABORATORY
Washington, D.C.

To: Distribution list for NRL Report 7917, "Further Observations on A533-B Steel Plate Tailored for Improved Radiation Embrittlement Resistance," by J. R. Hawthorne, September 22, 1975.

The following changes should be made:

Table 1c on page 11 contains four errors in the two columns titled "Fluence ($\times 10^{19} n/cm^2 > 1$ MeV)." Under Φ^{cs}, Plate Section A, Experiment ATR-1, the value should read 24.0, not 2.4; under Φ^{fs}, 25.0, not 2.5. Under Φ^{cs}, Plate Section B, Experiment ATR-1, the value should read 71.0, not 7.1; under Φ^{fs}, 75.0 not 7.5.

SECURITY CLASSIFICATION OF THIS PAGE (When Data Entered)

REPORT DOCUMENTATION PAGE		READ INSTRUCTIONS BEFORE COMPLETING FORM
1. REPORT NUMBER NRL Report 7917	2. GOVT ACCESSION NO.	3. RECIPIENT'S CATALOG NUMBER
4. TITLE (and Subtitle) FURTHER OBSERVATIONS ON A533-B STEEL PLATE TAILORED FOR IMPROVED RADIATION EMBRITTLEMENT RESISTANCE		5. TYPE OF REPORT & PERIOD COVERED Interim report on one phase of a continuing NRL Problem
		6. PERFORMING ORG. REPORT NUMBER
7. AUTHOR(s) J. Russell Hawthorne		8. CONTRACT OR GRANT NUMBER(s)
9. PERFORMING ORGANIZATION NAME AND ADDRESS Naval Research Laboratory Washington, D. C. 20375		10. PROGRAM ELEMENT, PROJECT, TASK AREA & WORK UNIT NUMBERS NRL Problem M01-14 Projects AT (49-5)-2110 and AT (49-25)-1003
11. CONTROLLING OFFICE NAME AND ADDRESS Reactor Research and Development Division, Energy Research and Development Administration, Wash., D.C. 20545 and Reactor Safety Research Division, U.S. Nuclear Regulatory Commission, Wash., D. C. 20555		12. REPORT DATE September 22, 1975
		13. NUMBER OF PAGES 19
14. MONITORING AGENCY NAME & ADDRESS(If different from Controlling Office)		15. SECURITY CLASS. (of this report) Unclassified
		15a. DECLASSIFICATION/DOWNGRADING SCHEDULE

16. DISTRIBUTION STATEMENT (of this Report)

Approved for public release; distribution unlimited.

17. DISTRIBUTION STATEMENT (of the abstract entered in Block 20, if different from Report)

18. SUPPLEMENTARY NOTES

19. KEY WORDS (Continue on reverse side if necessary and identify by block number)

A533-B steel
Charpy-V notch ductility
Dynamic Tear (DT) properties
Nuclear reactors
Pressure containment
Pressure vessel steel
Radiation effects
Radiation embrittlement
Radiation sensitivity
Scaleup demonstration melt
Transition temperature

20. ABSTRACT (Continue on reverse side if necessary and identify by block number)

A series of advanced investigations on the radiation performance of four 6-in.-thick plates from a large (30-ton) commercial melt of A533-B steel is described. The melt represented the first commercial-scale demonstration test of improved radiation embrittlement resistance through the control (minimization) of selected residual impurity elements. Melt specifications emphasized a low copper and phosphorus content; one half of the melt was modified however by a copper addition (copper content increased from 0.03% to 0.13%). Initial plate tests described superior 550°F (288°C) radiation resistance, in terms of notch ductility retention, for the primary melt

DD FORM 1 JAN 73 1473 EDITION OF 1 NOV 65 IS OBSOLETE
S/N 0102-014-6601

SECURITY CLASSIFICATION OF THIS PAGE (When Data Entered)

20. Continued

composition and verified the detrimental influence of impurity copper on irradiation behavior.

Promising capability of the primary melt composition for very high fluence ($\sim 2.5 \times 10^{20}$ n/cm^2 $>$ 1 MeV) applications is shown by the current investigations. In addition, a significant effect of copper content on radiation resistance is revealed for a broad range of exposure temperatures. An effect of copper content on postirradiation heat treatment response (notch ductility recovery) is also revealed. Charpy-V versus dynamic tear test performance and tensile strength trends with temperature are examined for low-temperature ($< 450^\circ$F, $< 121^\circ$C) and elevated-temperature (550 to 585°F, 288 to 307°C) irradiation conditions.

CONTENTS

FURTHER OBSERVATIONS ON A533-B STEEL PLATE TAILORED FOR IMPROVED RADIATION EMBRITTLEMENT RESISTANCE

INTRODUCTION

It has been shown conclusively that residual copper and phosphorus impurities in low-alloy structural steels are highly detrimental to nuclear radiation embrittlement resistance. The adverse effects of these elements on postirradiation notch ductility behavior was first isolated with small (300-lb) laboratory melts of A302-B steel using split melt procedures and selective impurity additions [1]. The melts further indicated the possibility for greatly improved radiation resistance by minimizing selected impurities in steel; high-purity laboratory-melt casts showed practically nil sensitivity to moderate neutron exposures ($\sim 3 \times 10^{19}$ n/cm^2 > 1 MeV) at normal reactor vessel service temperatures ($\sim 550°$F, or $\sim 288°$C). To confirm both findings for improving nuclear steelmaking, a commercial-scale (30-ton) demonstration melt of A533-B steel was ordered from Lukens Steel Company to ASTM Specifications but also to NRL supplemental requirements which limited copper, phosphorus, and other impurities of concern [2]. The melt-processing plan again called for a split melt procedure (Fig. 1) whereby a critical test of the copper effect on radiation resistance could be made. Results of the initial radiation assessment of the melt have been reported [2]. The demonstration test fully confirmed the laboratory findings, including the detrimental effect of copper on 550°F (288°C) radiation resistance. The present report presents the results of more recent, detailed evaluations of melt performance and of the copper contribution to apparent radiation sensitivity.

The present set of investigations focused on (a) the temperature dependence of the copper influence, (b) postirradiation heat treatment (annealing) response versus copper content, (c) the high-fluence capabilities of the primary melt composition (low copper), and (d) Charpy-V (C_v) performance relative to dynamic tear (DT) performance for low-temperature ($< 450°$F, $< 121°$C) and elevated-temperature ($\sim 550°$F, $\sim 288°$C) radiation exposures.

MATERIALS

Extensive documentation of the fabrication, heat treatment, and properties of the four 6-in.-thick plates from the demonstration melt is provided in Ref. 2. For this report it is sufficient to note that plate sections A and B were secured from the low-copper (0.03% Cu) ingot 1 and that plate sections C and D were obtained from the copper-modified (0.13% Cu) ingot 2 (Fig. 1). All plate sections received a double-quench, temper, and thermal-stress relief heat treatment sequence typical of that of plates forming welded nuclear-reactor pressure vessels. The heat treatment histories of the Class 1 plate sections A and C were identical; those of the Class 2 plate sections B and D were also identical but different from those of sections A and C.

Note: Manuscript submitted June 11, 1975.

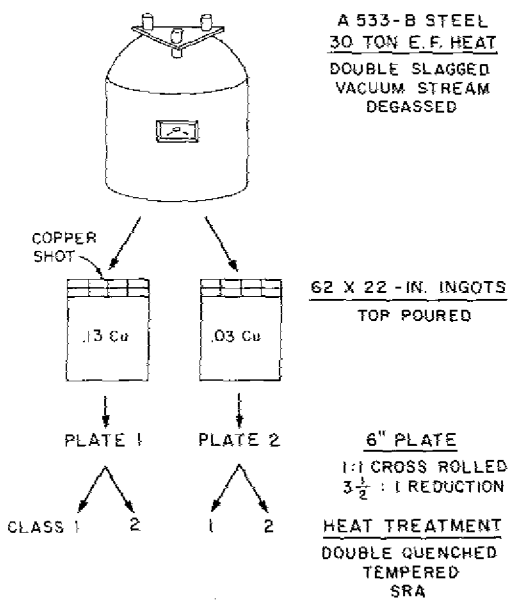

Fig. 1 — Schematic plan for processing the special commercial heat of A533-B steel for radiation sensitivity studies [2]

Chemical composition determinations for ingots 1 and 2 gave the following:

	Cu	*C*	*Mn*	*Si*	*Ni*	*Mo*	*P*	*S*
Ingot 1	0.03	0.17	1.22	0.19	0.58	0.50	0.008	0.008
Ingot 2	0.13	0.17	1.21	0.20	0.56	0.50	0.008	0.007

SPECIMENS

Postirradiation notch ductility investigations involved standard C_v specimens and 5/8-in.-thick DT specimens (MIL-STD-1601 (SHIPS)). The C_v specimens were oriented in the longitudinal (LT, RW) test direction except for C_v vs DT postirradiation comparisons, for which the transverse (TL, WR) test orientation was employed. Postirradiation strength changes were determined using either 0.252-in. diameter or 0.226-in. diameter by 1.250-in.-gage length tensile specimens.

INVESTIGATION 1: TEMPERATURE DEPENDENCE OF THE COPPER EFFECT

Initial evaluations of the demonstration melt involved a simultaneous irradiation of C_v specimens from the four plate sections (A, B, C, D) at 550°F (288°C) to a fluence of 2.4×10^{19} n/cm^2 > 1 MeV (Φ^{cs}, calculated neutron-spectrum fluence). Resultant C_v 30 ft-lb transition temperature increases were 40°F (22°C) and 65°F (36°C) for the 0.03% Cu plate sections A and B respectively and 140°F (78°C) and 125°F (69°C) for the 0.13% Cu plate sections C and D respectively [2]. The objective of the present investigation was to determine relative differences in embrittlement behavior as a function of copper content at successively lower irradiation temperatures and thereby broaden knowledge of the copper contribution and basic postirradiation trends for low-copper material.

Since comparable irradiation responses were originally observed for Class 1 versus Class 2 plates of a given copper content, this investigation proceeded with the two Class 1 plates only (plate A with 0.03% Cu and Plate C with 0.13% Cu). The irradiation experiment was designed to expose three groups of C_v specimens (both plates, random array) at 370, 440, and 500°F (188, 227, and 260°C) to a fluence of 2.0 to 2.3 $\times$ 10^{19} n/cm^2 $>$ 1 MeV. Results for individual as-irradiated conditions are compared in Fig. 2. The upper graph pertains to the 0.03% Cu plate; the lower graph pertains to the 0.13% Cu plate.

The data permit several significant observations:

- For a given irradiation temperature the 30 ft-lb transition temperature increase is consistently less for the 0.03% Cu plate than for the 0.13% Cu plate. Accordingly, a benefit to radiation performance can be derived from a low copper content for a broad range of exposure temperatures below 550°F (288°C).

- A progressive decrease in irradiation effect above 370°F (188°C) is indicated for the 0.03% Cu plate but not for the 0.13% Cu plate. An equivalent irradiation effect is shown for 370°F (188°C) and 440°F (227°C) irradiation conditions for the 0.13% Cu plate. For the 0.03% Cu plate the small difference in irradiation effect between these two irradiation conditions is considered significant because of their fluence difference.

- Irradiation response of both plates was most sensitive to irradiation temperature in the temperature interval 440°F to 500°F (227 to 260°C). From findings of the present study versus the original study the irradiation response of both plates would not appear especially dependent on irradiation temperature in the range of 500 to 550°F (260 to 288°C). A fluence difference of 2.0 versus 2.4 $\times$ 10^{19} n/cm^2 $>$ 1 MeV is noted for the 500 vs 550°F (260 vs 288°C) exposure conditions.

- The increase in transition temperature with 500°F (260°C) irradiation is only 25% of that produced by 370°F (188°C) irradiation for the 0.03% Cu plate; however, the corresponding difference in transition temperature increase is 50% in the case of the 0.13% Cu plate.

INVESTIGATION 2: INFLUENCE OF COPPER CONTENT ON RESPONSE TO POSTIRRADIATION HEAT TREATMENT

The multitemperature irradiation experiment for investigation 1 also provided a limited number of C_v specimens for exploring the influence of copper content on response to postirradiation heat treatment. Postirradiation heat treatment has been shown highly effective toward notch ductility recovery (i.e., embrittlement relief) in some cases.

The heat treatment investigated was 650°F (343°C) for 168 hr. The heat treatment duration was chosen to effect maximum recovery for the heat treatment temperature

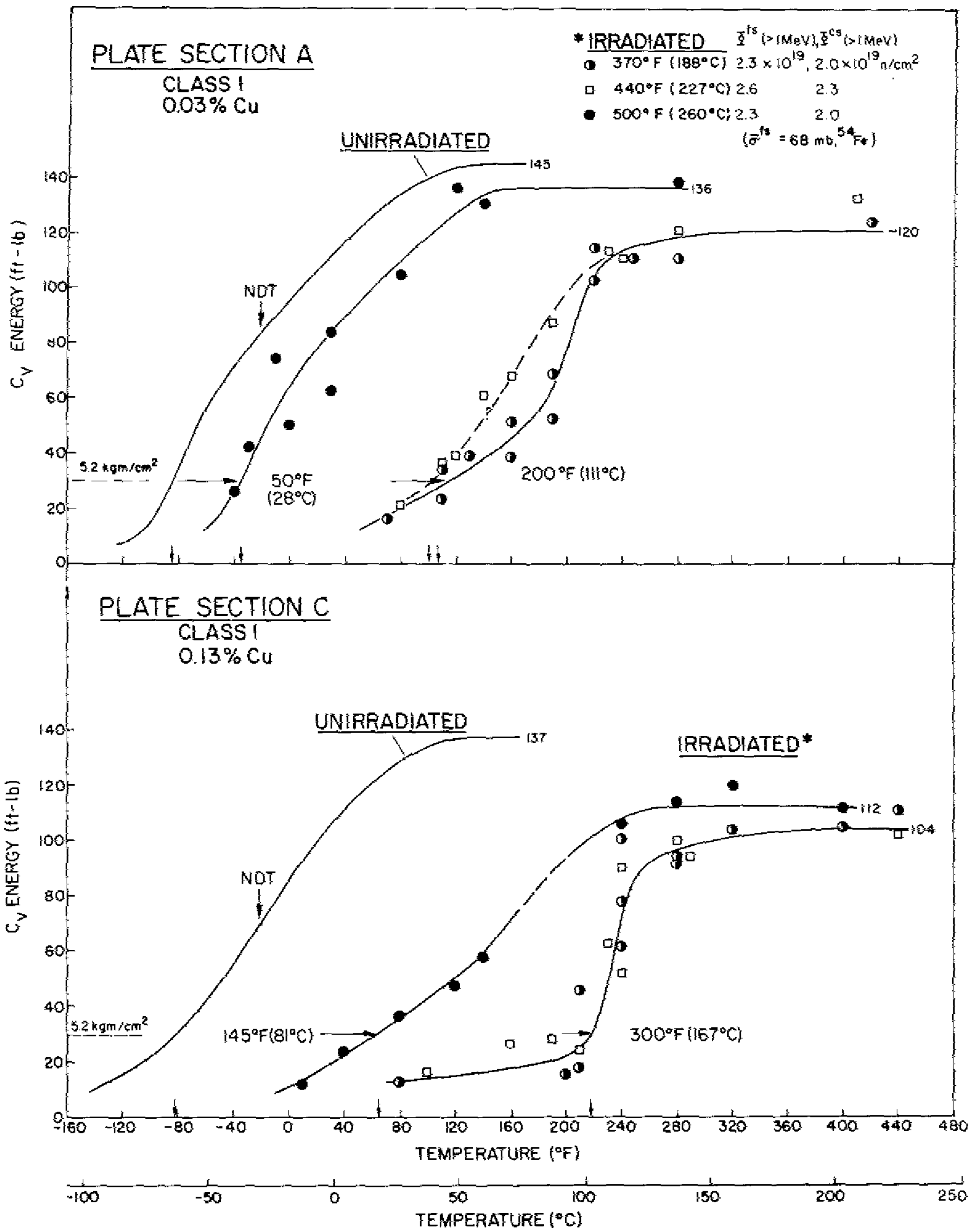

Fig. 2 — Charpy-V notch ductility of plate section A (primary melt composition) and plate section C (copper-modified composition) before and after irradiation at three temperatures. A higher copper content is shown to be detrimental to radiation resistance at each temperature. Neutron fluences for a calculated neutron-energy spectrum (Φ^{cs}) and for an assumed fission spectrum energy distribution (Φ^{fs}) are both given for reference.

selected based on other studies. Results of the investigation are presented in Fig. 3. Reference curves for the as-irradiated condition from Fig. 2 are separately identified by the coded data symbols in parentheses. The following observations can be made:

- Significant recovery was achieved by annealing the 440°F (227°C) irradiated condition but not the 550°F (260°C) irradiated condition for the 0.03% Cu plate. On the other hand, comparable residual embrittlement in terms of transition temperature elevation after the anneal is depicted for both irradiated and annealed conditions.

- Significant recovery was achieved by annealing both the 440°F (227°C) and 500°F (260°C) irradiated conditions of the 0.13% Cu plate. However recovery for the 440°F (227°C) irradiated condition was approximately twice that for the 500°F (260°C) irradiated condition. A slightly lower residual embrittlement after annealing for the 500°F (260°C) irradiated condition compared to the 440°F (227°C) irradiation condition is inferred by the data but may be data scatter (in view of the datum for the 370°F (188°C) irradiated and annealed condition).

- Residual embrittlement after annealing was significantly greater for the 0.13% Cu plate than for the 0.03% Cu plate (75 to 80°F vs 25 to 30°F, or 42 to 44°C vs 14 to 17°C).

In a separate experiment, hardness recovery following 550°F (288°C) irradiation was also explored for the Class 1, 0.03% Cu and 0.13% Cu plates. In this case 1-hour heat treatments at temperatures from 300 to 1050°F (149 to 566°C) were used. Results are given in Fig. 4. The increase in hardness by irradiation, as noted, was much greater for the 0.13% Cu plate than for the 0.03% Cu plate. The trends in recovery indicate that for both plates the reduction in postirradiation hardness begins when the original irradiation temperature is exceeded and the return to the preirradiation hardness level occurs at about the same annealing temperature (~ 1025°F, ~ 551°C).

Other studies have shown that the contribution of copper content to apparent radiation sensitivity is through a mechanism enhancing the yield strength elevation by irradiation [3,4]. Accordingly a reduction of strength toward preirradiation levels by annealing is one mode of recovery, as demonstrated by Fig. 4. Of greater importance, however, the results of Fig. 3 (500°F, 260°C, irradiated condition) and of Fig. 4 demonstrate that response to annealing is proportional, in part, to the amount of prior radiation hardening (i.e., yield strength elevation).

INVESTIGATION 3: HIGH-FLUENCE ASSESSMENTS OF PRIMARY MELT COMPOSITION

High-fluence assessments of the Class 1, 0.03% Cu plate were performed to investigate the full potential of primary melt composition for severe nuclear service applications. Currently a trend toward much higher end-of-life fluences is apparent in nuclear reactor vessel applications. The potential of ferritic alloys for high-fluence service in new reactor systems such as the fast breeder reactor is also of increasing interest. Investigation 3 centered on the irradiation of C_v specimens in five experiments in the Advanced Test Reactor in a water-cooled loop. Irradiations were at 530, 550, and 600°F (277, 288, and 316°C). Fluences ranged from 1.5 to 6.6 $\times 10^{20}$ n/cm^2 > 1 MeV. Tensile specimens were included in one 550°F (288°C) experiment.

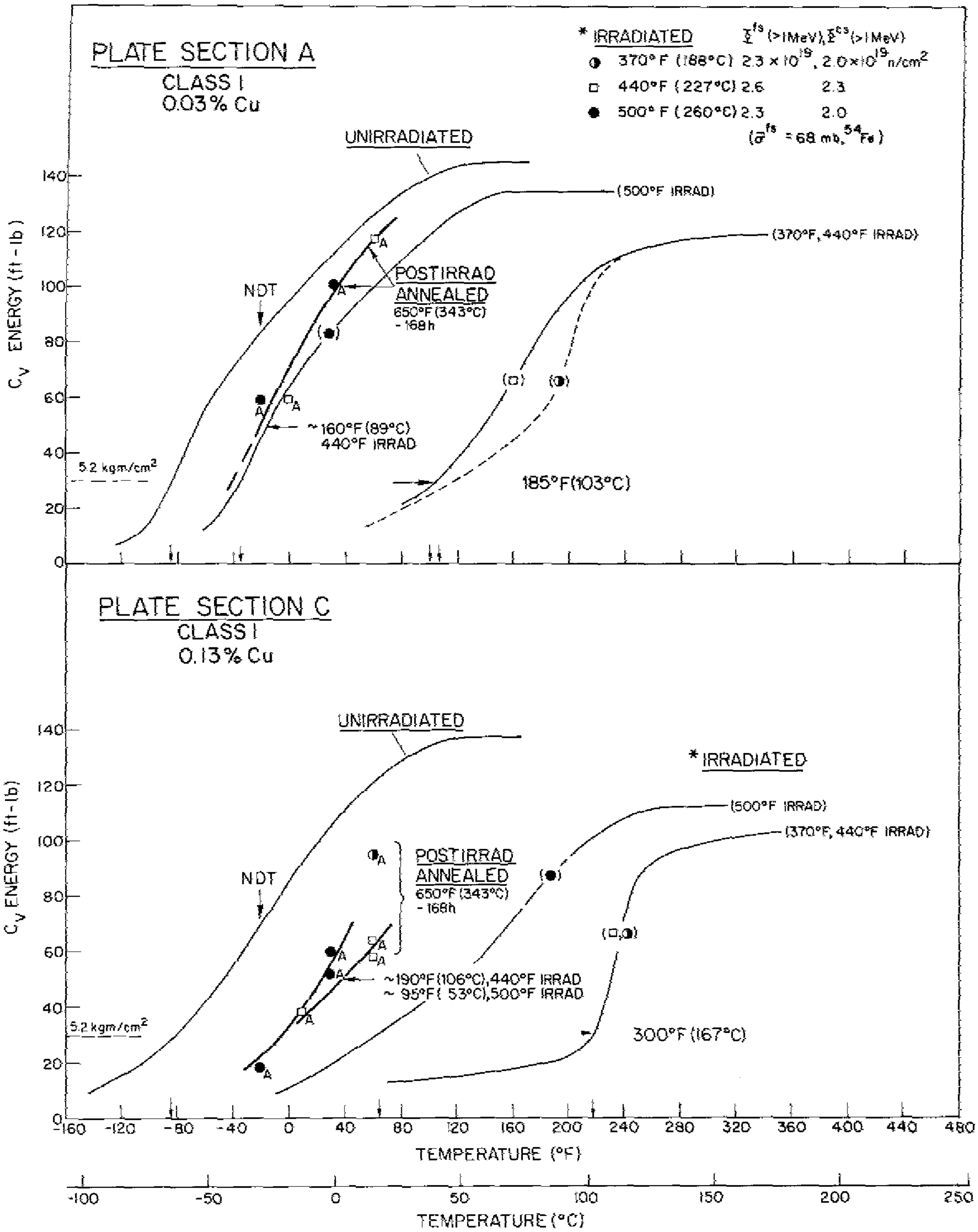

Fig. 3 — Charpy-V notch ductility recovery by plate sections A and C with 650°F (343°C), 168-hr heat treatment following irradiation at three temperatures. Greater residual embrittlement after heat treatment is observed for the higher-copper content plate section C. (Specimens representing the 370°F (188°C) irradiation condition of plate section A were not available for heat treatment studies.)

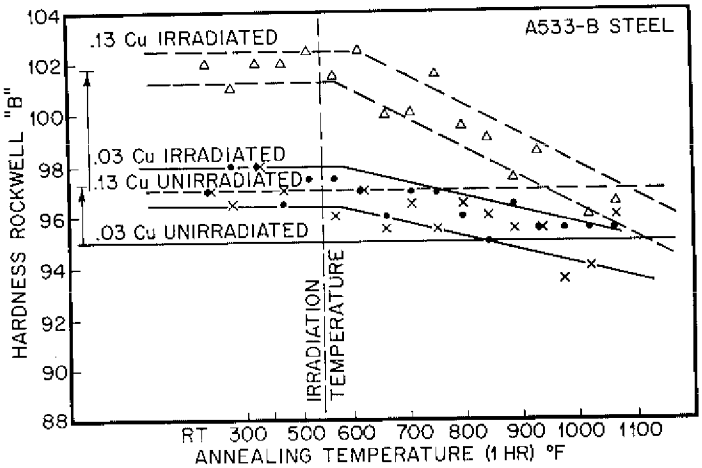

Fig. 4 — Hardness recovery by plate sections A and C with isochronal heat treatments following 550°F (288°C) irradiation. The hardness of unirradiated material and the radiation-induced increase in hardness are shown respectively by the horizontal reference lines and the vertical arrows.

Postirradiation C_v performance (Fig. 5) reveals a strong potential of the material for high-fluence service, at least to 2.5 × 10^{20} n/cm^2 at the temperatures investigated. For example, a fluence of 2.3 × 10^{20} n/cm^2 at 530°F (277°C) did not raise the C_v 30 ft-lb transition temperature above 250°F (121°C). Similarly, a fluence of 3.8 × 10^{20} n/cm^2 at 600°F (316°C) elevated the 30 ft-lb transition to only 320°F (160°C). Relatively high upper-shelf energy retention is also indicated.

Of special interest, radiation embrittlement saturation is not evident in the data for either 530°F (277°C) or 600°F (316°C) irradiation conditions. Also, unlike observations at fluences a decade lower (investigation 1), irradiation temperature differences in the range 530 to 600°F would appear quite significant to notch ductility behavior. Specifically, the difference in transition temperature increase for irradiation at 550°F (288°C) vs 600°F (316°C) (to 2.6 and 3.8 × 10^{20} n/cm^2 respectively) was somewhat larger than expected. However, little is known on progressive embrittlement with increasing neutron exposure at this general fluence level for low-alloy steels.

Postirradiation strength determinations following 550°F (288°C) irradiation to 2.5 × 10^{20} n/cm^2 indicated yield strengths of 84.5 ksi at 250°F (121°C), 80.8 ksi at 400°F (204°C), and 77.8 ksi at 550°F (288°C) (Table 1c), compared to preirradiation values of 64.8 ksi at 75°F (24°C) and 56.8 ksi at 550°F (288°C). The available data describe a linear relationship between yield strength and temperature for both irradiated and unirradiated conditions; more importantly the linear relationship appears essentially unchanged by irradiation except for the increase in yield strength. Accordingly accurate projections of yield strength behavior for temperatures in the range 75 to 550°F (24 to 288°C) should be possible from only limited postirradiation data.

Yield strength and C_v upper-shelf performance for the high-fluence condition was evaluated using Ratio Analysis Diagram (RAD) procedures [5]. The RAD analysis indicates capability of the material to retain plastic fracture resistance qualities even in

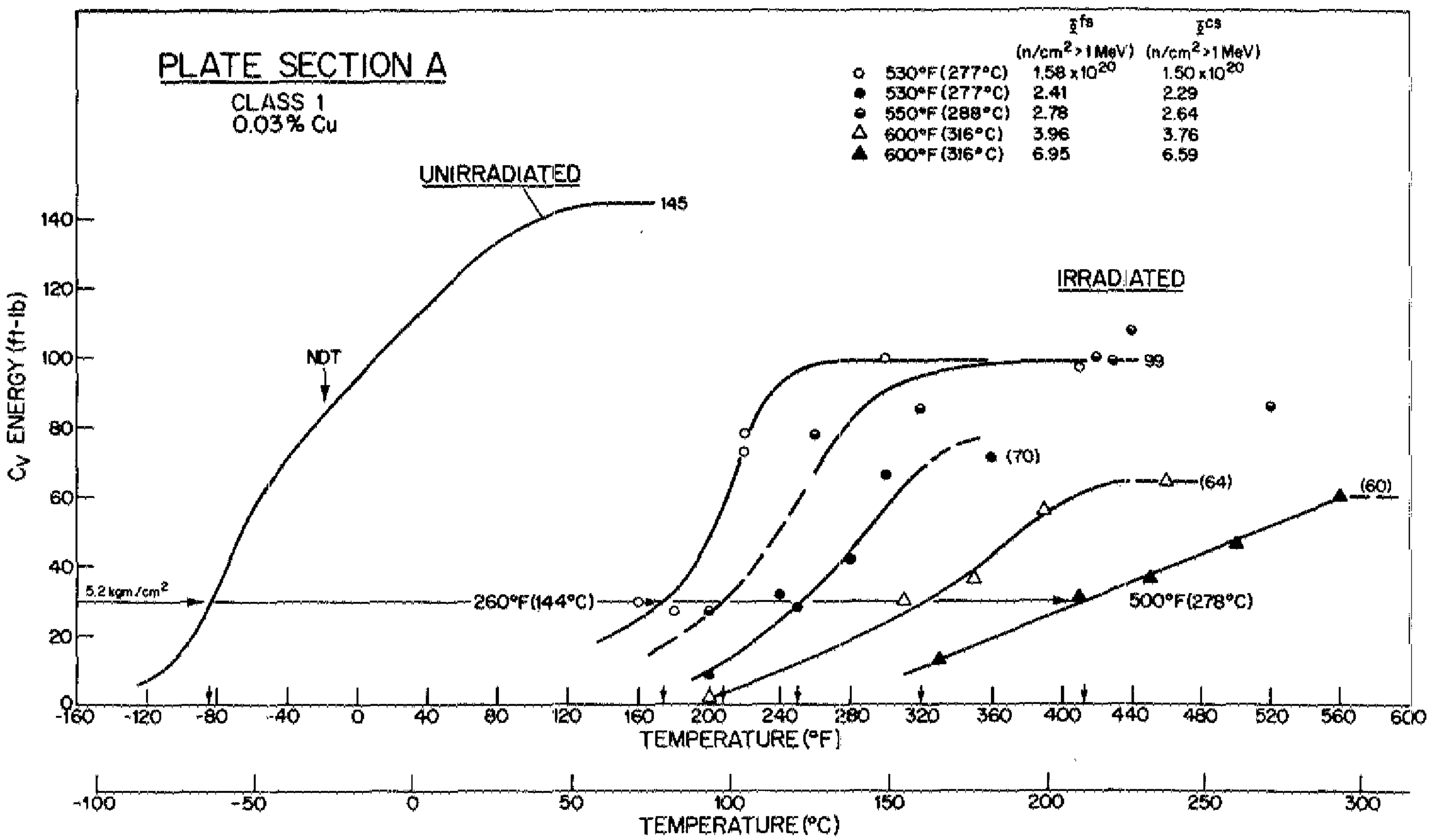

Fig. 5 — Charpy-V notch ductility retention of plate section A (primary melt composition) following high-fluence exposure at elevated temperatures. A capability for high-fluence service is clearly indicated.

thick (6-in.) component sizes after 2.5×10^{20} n/cm^2 at 550°F (288°C). The C_v upper-shelf energy levels described in Fig. 5 are also noted to be above the minimum level specified by the AEC (10 CFR 50, Appendix G) for conventional fracture toughness evaluation.

INVESTIGATION 4: COMPARISON OF C_v vs DT PERFORMANCE WITH IRRADIATION

Investigation 4 was conducted to compare C_v and DT indications of radiation embrittlement for 0.03% Cu and 0.13% Cu melt versions and to test the correlation of the two test methods themselves for the irradiated condition. The DT test procedure provides a means for rational interpretation of fracture toughness whereby structural performance can be projected. This structural translation generally is not possible with the C_v test, but an apparent correlation of C_v to DT notch toughness has been observed for some materials [6,7].

Relative upper-shelf degradation and relative transition temperature elevation were examined. For the latter the irradiation effect was indexed to the midenergy transition temperature. For the DT test the midenergy transition approximates a yield criterion for the material: the lowest toughness level which still permits general through-thickness yielding in the neighborhood of a flaw [8,9].

A series of experiments involving the simultaneous irradiation of C_v and DT specimens were conducted in the Engineering Test Reactor (ETR) at $<$ 300°F ($<$ 149°C), the Union Carbide Research Reactor (UCRR) at 550°F (288°C), and the Big Rock Point Power Reactor (BRPR) at 585°F (307°C). Tensile specimens were included in most experiments. Test results are summarized in Tables 1a and 1b and are illustrated in Figs. 6 through 9. Where noted, fluence differences between C_v and DT specimens were due to flux gradients in the experiment.

The aggregate results permit several observations:

- The data give additional verification of the detrimental effect of copper content on radiation resistance at low and elevated temperatures.

- The radiation-induced elevation in the C_v 30 ft-lb transition temperature (ΔC_v 30TT) matches or slightly exceeds the elevation in C_v midenergy transition temperature (ΔC_v 50TT) for each of the exposure temperature conditions used.

- General agreement of ΔC_v 50TT and ΔDT 50TT with irradiation is indicated. Differences in actual C_v 50TT and actual DT 50TT are on the order of 70 to 130°F (39 to 72°C) after irradiation for all plate sections.

- A reasonable approximation of the percent reduction in DT upper-shelf energy by irradiation is given by the percent reduction in C_v upper-shelf energy (considering the limited number of upper-shelf data points available).

TABLE 1a
C_v and DT Notch Ductility Properties of Irradiated Plate Sections A (0.03% Cu, Class 1), B (0.03% Cu, Class 2), C (0.13% Cu, Class 1) and D (0.13% Cu, Class 2)

Plate Section	Experiment*	Irrad. Temp.		Fluence† (× 10^{19} n/cm² > 1 MeV)‡		Brittle/Ductile Transition Temperature								
						C_v 30 ft-lb TT			C_v 50 ft-lb TT			DT 50 ft-lb TT		
						Irrad.		ΔT (°F)	Irrad.		ΔT (°F)	Irrad.		ΔT (°F)
		°F	°C	Φ^{cs}	Φ^{fs}	°F	°C		°F	°C		°F	°C	
A	ETR-1	< 300	< 149	2.6	2.4	240	116	330	245	118	295	380	193	325
	UCRR-1	550	288	2.7	3.1	-20	-29	70	25	-4	75	115	46	60
	BRPR-1	585	307	1.6	2.5	5	-15	75	40	4	75	—	—	—
				2.0	3.1	—	—	—	—	—	—	120	49	65
B	ETR-2	< 300	< 149	3.7	3.3	260	127	345	270	132	290	400	204	325
	UCRR-2	550	288	2.5	2.9	-30	-34	55	35	2	55	100	38	25
C	ETR-3	< 300	< 149	1.9	1.7	270	132	360	275	135	320	345	174	280
	BRPR-1	585	307	1.6	2.5	60	16	140	135	57	155	—	—	—
D	ETR-4	< 300	< 149	2.2	1.9	320	160	390	325	163	335	430	221	330

*ETR = Engineering Test Reactor, UCRR = Union Carbide Research Reactor, and BRPR = Big Rock Point Power Reactor.
†Φ^{cs} = calculated spectrum fluence and Φ^{fs} = fission spectrum fluence.
‡For UCRR, $\Phi^{cs} > 0.1$ MeV = 2.0 × $\Phi^{cs} > 1$ MeV; for BRPR, $\Phi^{cs} > 0.1$ MeV = 1.66 × $\Phi^{cs} > 1$ MeV.

TABLE 1b
C_v and DT Notch Ductility Properties of Irradiated
Plate Sections A, B, C, and D

Plate Section	Experiment	Irrad. Temp.		Fluence ($\times 10^{19}$n/cm^2 > 1 MeV)		Upper-Shelf Energy (ft-lb)			
						C_v		DT	
		°F	°C	Φ^{cs}	Φ^{fs}	Irrad.	Δft-lb	Irrad.	Δft-lb
A	ETR-1	< 300	< 149	2.6	2.4	~ 103	~ 42	> 600	ND‡
	UCRR-1	550	288	2.7	3.1	139	< 10	1150	170
	BRPR-1	585	307	1.6	2.5	~ 130	< 10	—	—
				2.0	3.1	—	—	1010	120
B	ETR-2	< 300	< 149	3.7	3.3	~ 94	~ 35	720	380
	UCRR-2	550	288	2.5	2.9	123	< 10	1040	60
C	ETR-3	< 300	< 149	1.9	1.7	80	61	730	330
	BRPR-1	585	307	1.6	2.5	124	13	—	—
D	ETR-4	< 300	< 149	2.2	1.9	79	⩾ 25	605	465

TABLE 1c
Tensile Properties of Irradiated Plate Sections A, B, C, and D

Plate Section	Experiment*	Irrad. Temp.		Fluence ($\times 10^{19}$n/cm^2 > 1 MeV)†		Test Temp.		Yield Strength (ksi)	Tensile Strength (ksi)	Red. of Area (%)	Elong. (%)
		°F	°C	Φ^{cs}	Φ^{fs}	°F	°C				
A	ETR-1	< 300	< 149	2.6	2.4	75	24	116.8	116.8	50.8	ND‡
	UCRR-1	550	288	2.7	3.1	75	24	70.2	90.2	68.5	26.3
	BRPR-1	585	307	1.6	2.5	250	121	67.6	84.4	69.6	23.7
						585	307	63.4	87.6	ND	ND
	ATR-1	550	288	2.4	2.5	250	121	84.5	96.5	ND	ND
						400	204	80.0	97.2	ND	ND
						550	288	77.8	97.2	ND	ND
B	ETR-2	< 300	< 149	3.7	3.3	75	24	130.7	130.7	64.1	13.5
						550	288	74.8	93.0	60.2	24.7
	UCRR-2	550	288	2.5	2.9	75	24	87.8	102.5	ND	ND
						300	149	77.6	93.8	ND	ND
						550	288	74.6	96.2	ND	ND
	ATR-1	550	288	7.1	7.5	400	204	130.6	139.6	ND	ND
C	ETR-3	< 300	< 149	1.9	1.7	75	24	117.1	117.1	ND	ND
						450	232	82.0	87.6	ND	ND
	BRPR-1	585	307	1.6	2.5	250	121	74.2	90.8	67.3	25.3
						585	307	68.8	91.0	ND	ND
D	ETR-4	< 300	< 149	2.2	1.9	75	24	140.4	140.6	60.2	12.1
						550	288	93.4	101.7	55.0	19.7

*ATR = Advanced Test Reactor.
†For ATR, $\Phi^{cs} > 0.1$ MeV = $2.15 \times \Phi^{cs} > 1$ MeV.
‡ND = not determined.

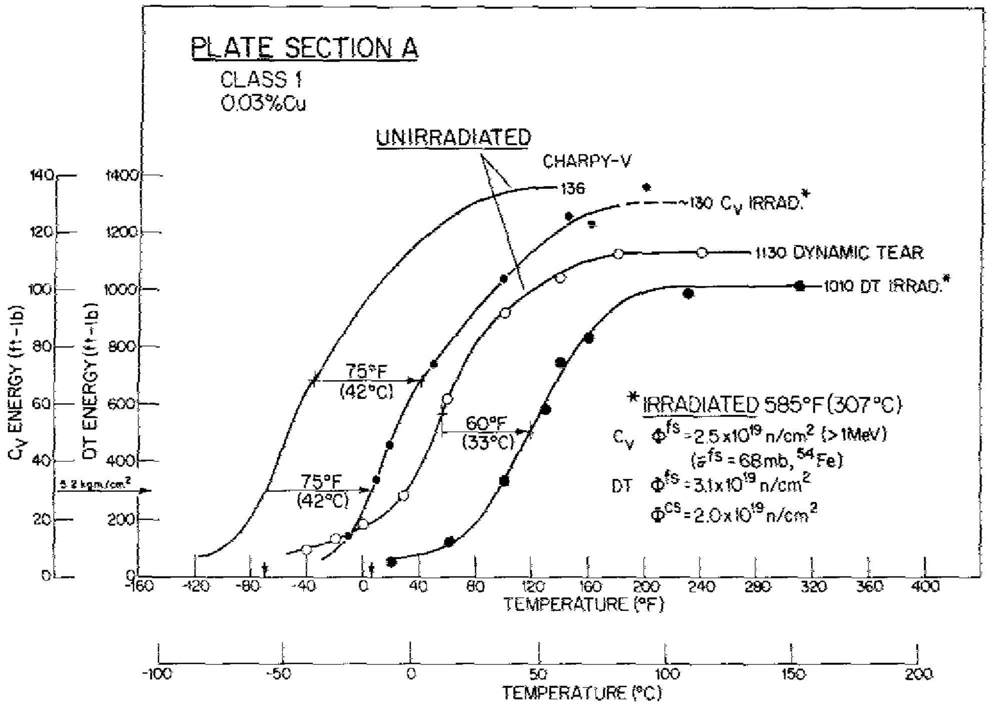

Fig. 6 — Comparison of Charpy-V and dynamic tear test performance of plate section A after 585°F (307°C) irradiation in the Big Rock Point Power Reactor. Good agreement is indicated in C_v and DT midenergy-range transition temperature increase by irradiation. The separation of the C_v and DT energy curves is typical. The toe of the DT curve corresponds with the nil-ductility transition (NDT) temperature by the drop-weight test; the NDT temperature physically denotes the beginning of the temperature region of sharply increasing notch toughness with temperature.

• The range of C_v to DT upper-shelf energy ratios for the preirradiation condition (7.5-9.1:1, or 8.4:1 avg) compares well with the range for the postirradiation condition (7.7-9.1:1, or 8.2:1 avg). Accordingly a single correlation of C_v:DT upper-shelf energy is suggested for both conditions.

• Yield strengths of the Class 1 plate sections were not elevated appreciably by 1.6×10^{19} n/cm^2 > 1 MeV at 585°F (307°C), but major elevations in strength were recorded for < 300°F (< 149°C) irradiation for Class 1 and Class 2 plates. The effect of irradiation on the yield strength was less for low-copper content plate than for the copper-modified (0.13% Cu) plate.

• The change in yield strength with increasing test temperature is much greater for the < 300°F (< 149°C) irradiation condition than for the 550°F (288°C) or 585°F (307°C) irradiation conditions.

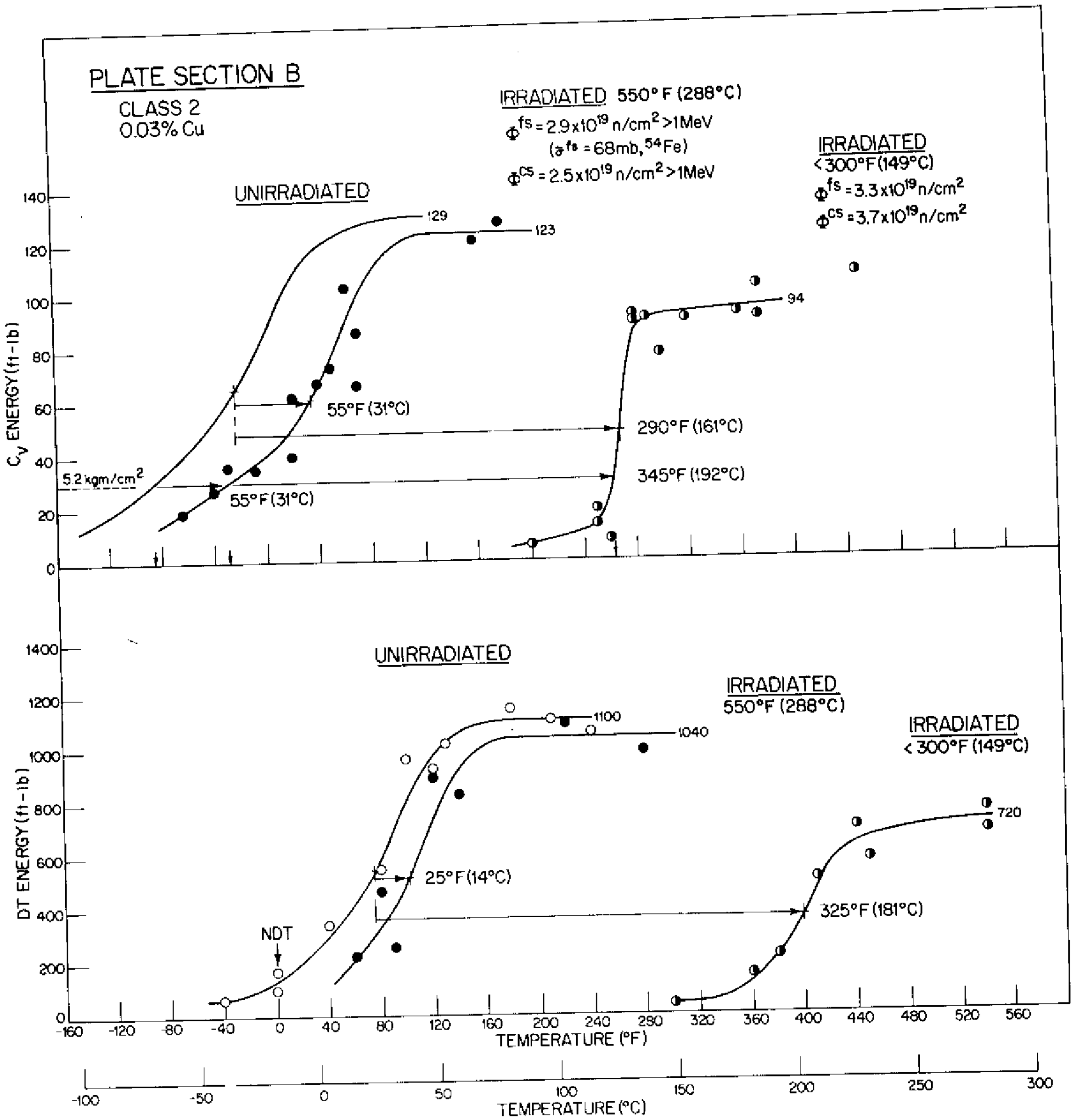

Fig. 7 — Comparison of Charpy-V and dynamic tear test performance of plate section B after < 300°F (< 149°C) irradiation in the Engineering Test Reactor and after 550°F (288°C) irradiation in the Union Carbide Research Reactor

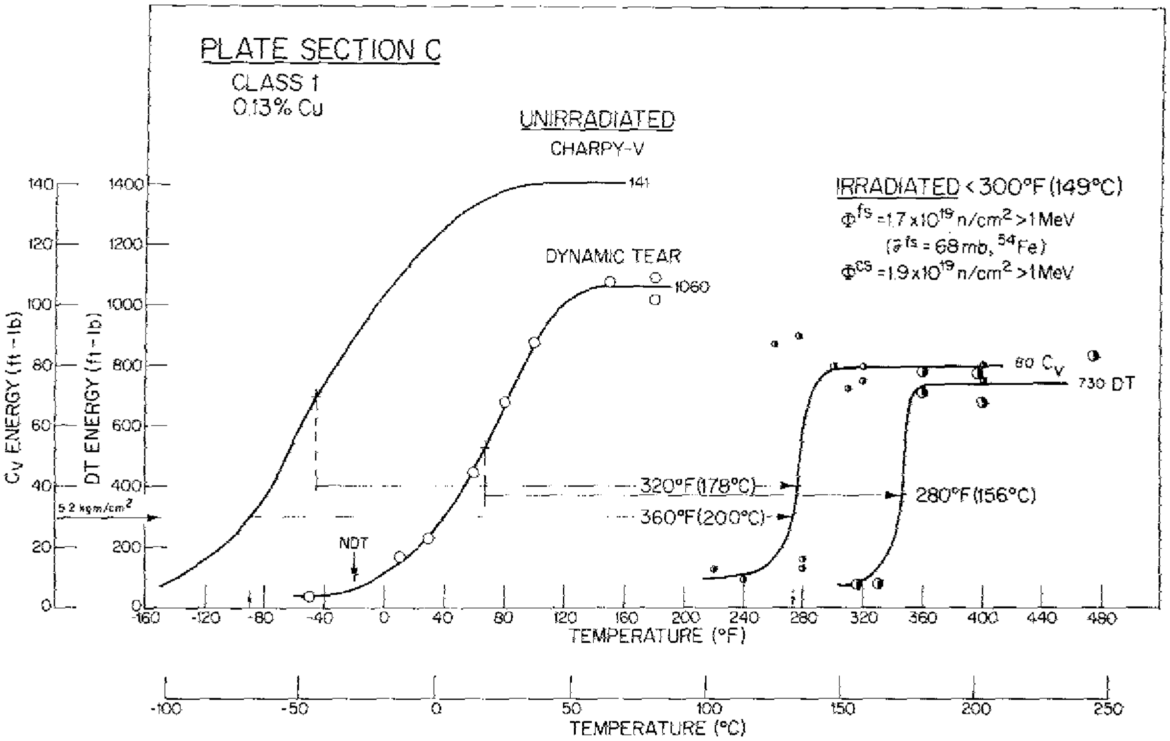

Fig. 8 — Comparison of Charpy-V and dynamic tear test performance of plate section C after < 300°F (< 149°C) irradiation in the Engineering Test Reactor

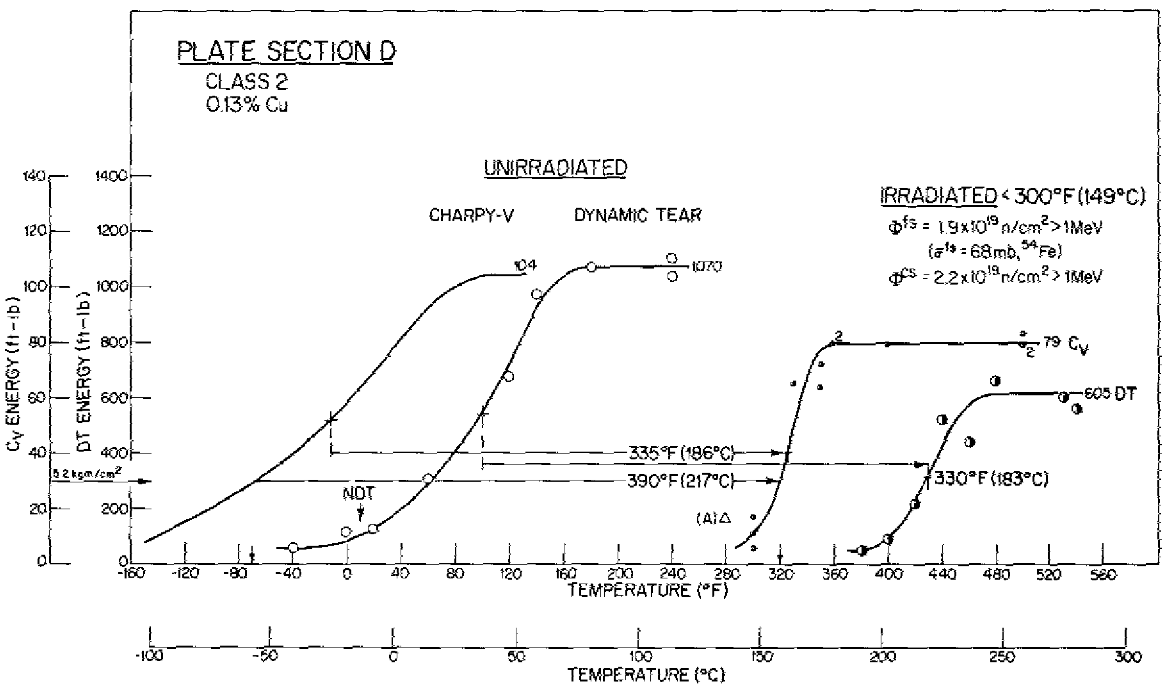

Fig. 9 — Comparison of Charpy-V and dynamic tear test performance of plate section D after < 300°F (< 149°C) irradiation in the Engineering Test Reactor

DISCUSSION

The latest experimental results for the demonstration melt clearly support a case for limiting copper, phosphorus, and other harmful impurity elements in steels for reactor structures, whether service temperatures are near ambient temperature as in the case of support structures or at elevated temperatures as in the case of pressure vessels. An improvement to elevated-temperature radiation resistance has similarly been recorded for an improved-composition, submerged-arc demonstration test weld (A533-B parent plate) [10]. In response to these and other observations, the development of supplemental specifications for limiting the impurity element contents of reactor steels and filler metals has been undertaken by ASTM Committee A-1 with assistance by ASTM Committee E-10 and by the AWS Filler Metal Committee.

A potential for high-fluence, elevated-temperature applications has been revealed by the investigations for improved-composition, ferritic structural steels. A need for further definition of property-change trends as functions of fluence and exposure temperature is pointed out however. In particular the significance of small differences in exposure temperature requires clarification.

SUMMARY

The investigations have demonstrated that a low copper content in A533-B steel is beneficial to radiation embrittlement resistance in both low-temperature and elevated-temperature service applications. In addition, specially tailored structural steel is shown to have promise for high-fluence advanced-reactor applications. The investigations observed a correspondence between C_v and DT test methods in radiation effects determinations for 0.03% Cu and 0.13% Cu plates and found the yield strength trend with temperature to differ with irradiation temperature condition (< 300°F (< 149°C) vs 550°F (288°C) to 585°F (307°C)).

ACKNOWLEDGMENTS

The investigations were sponsored by the Division of Reactor Research and Development and the Division of Reactor Safety Research of the U.S. Atomic Energy Commission. The continuing support of both divisions is deeply appreciated.

The author also expresses his gratitude to the members of the NRL Operations Section of the Thermostructural Materials Branch (H. E. Watson, W. E. Hagel, and B. H. Menke) for their several contributions to experimental phases of the program.

REFERENCES

1. U. Potapovs and J.R. Hawthorne, "The Effect of Residual Elements on 550°F Irradiation Response of Selected Pressure Vessel Steels and Weldments," NRL Report 6803, Nov. 22, 1968; also in Nuclear Applications 6, 27-46 (Jan. 1969).
2. J.R. Hawthorne, "Demonstration of Improved Radiation Embrittlement Resistance of A533-B Steel Through Control of Selected Residual Elements," NRL Report 7121, May 29, 1970.

3. F.A. Smidt, Jr. and J.A. Sprague, "Property Changes Resulting from Impurity-Defect Interactions in Iron and Pressure Vessel Steel Alloys," American Society for Testing and Materials Special Technical Publication 529, 1973, 78-91.

4. J.R. Hawthorne and E. Fortner, "Radiation and Temper Embrittlement Processes in Advanced Reactor Weld Metals," *Trans. ASME, J. Eng. Ind.*, 94B (No. 3), 807-814 (Aug. 1972).

5. W.S. Pellini, "Advances in Fracture Toughness Characterization Procedures and in Quantitative Interpretations to Fracture-Safe Design for Structural Steels," NRL Report 6713, Apr. 3, 1968.

6. F.J. Loss, J.R. Hawthorne, and C.Z. Serpan, Jr., "A Reassessment of Fracture-Safe Operating Criteria for Reactor Vessel Steels Based on Charpy-V Performance," NRL Report 7152, Sept. 8, 1970.

7. J.R. Hawthorne, "Postirradiation Dynamic Tear and Charpy-V Performance of 12-in. Thick A533-B Steel Plates and Weld Metal," *Nuclear Engineering and Design*, 17 (No. 1), 116-130 (Aug. 1971).

8. J.R. Hawthorne and F.J. Loss, "Fracture Toughness Characterization of Shipbuilding Steels," NRL Report 7701, July 10, 1974.

9. F.J. Loss, J.R. Hawthorne, and W.S. Pellini, "Generalized Procedures for the Use of Plane Strain and Elastic Plastic Fracture Mechanics Options in the Modernization of Nuclear Standards," Proceedings 2nd International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, 10-14 Sept. 1973, G5/1, (*in Nuclear Engineering and Design* 27, 77-93 (1974)).

10. J.R. Hawthorne, "Radiation Resistant Weld Metal for Fabricating A533-B Nuclear Reactor Vessels," *Welding J. Res. Supl.* 51 (No. 7), 369s-375s (July 1972).