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THE EFFECT OF NEUTRON IRRADIATION AND HYDROGEN CONCENTRATION ON TENSILE PROPERTIES OF THE CANDU COOLANT TUBE SPACER ALLOY

by

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Chalk River, Ontario

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SYNOPSIS

The effect of neutron irradiation on tensile properties of the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy has been evaluated for neutron fluxes up to 3×10^{20} n/cm^2 (E>1.0 MeV). Specimen temperature during irradiation was normally 300°C. The material was studied in a heat treated condition, quenched from ≈40°C below the β-transus and aged 6 hrs at 535°C. Hydrogen concentrations up to 250 ppm were used in some tests. The dose dependence of hardening followed the saturation equation proposed by Makin and Minter

$$\Delta\sigma = A[1-\exp(-B\phi)]^{\frac{1}{2}}$$

with a saturation increment of 27,000 psi in the 0.2% yield stress above the unirradiated value of ≈75,000 psi at 300°C. Irradiation up to the maximum studied had little effect on reduction in area, although the uniform elongation was decreased. Increasing the hydrogen concentration had very little effect on the tensile properties in the unirradiated condition, but tended to lower the reduction in area of specimens irradiated to ≈10^{20} n/cm^2.

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1. INTRODUCTION

Coolant tube spacers in the Douglas Point Reactor are fabricated from the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy (1). For this application, the alloy is heat treated by water quenching from a temperature approximately 40°C below the β-transus, and aged 6 hrs at 535°C. In this metallurgical condition the material has a ratio of strength to neutron capture cross-section approximately twice that of 15% cold-worked Zircaloy-2 (in tests over the temperature range of interest, i.e. RT-300°C). Furthermore, in the quenched (and aged) condition, danger from unfavourably oriented hydride is minimized (2). The binary Zr-2.5 wt% Nb alloy has similar favourable strength properties (1), but the copper addition gives improved corrosion resistance in the moist carbon dioxide-air environment in which the spacer is used. The copper addition is also known to accelerate aging in the alloy, and it appeared possible, therefore, that neutron irradiation might have rather more effect on strength and ductility of this alloy than on the Zr-2.5 wt% Nb. The work reported here was undertaken to determine the effect of irradiation on the tensile properties of

the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy, both for material of low hydrogen concentration and with added hydrogen, primarily to establish the amount of embrittlement to be expected from in-reactor service. It has been found that the material is not severely embrittled either by irradiation up to an integrated neutron flux of 3×10^{20} n/cm^2, or by hydrogen concentrations up to 250 ppm.

2. EXPERIMENTAL

2.1 Material

Two batches of alloy were used, both made from sponge zirconium. The ingot analysis of principle constituents is listed in Table I.

TABLE I

ANALYSIS OF INGOT MATERIAL

CRNL Designation	Constituent Analysis Nb wt%	Cu wt%	O wt%
AM	2.52	0.51	0.11
AS	2.57	0.50	0.115

The ingot was fabricated into rod by hot rolling down to 1.5 in. diameter, and subsequently swaging to final size (0.5 - 0.375 in. diameter) at nominal temperatures of 785 and 625°C for AM and AS batches respectively. The tensile properties of material from the two batches were closely similar and results

from them will be discussed collectively throughout the report.

2.2 Metallurgical Condition

Tensile test specimens of gage diameter and length 0.160 and 1 in. respectively were cut in the axial direction of the rod. Specimens receiving a quench and age heat treatment were normally quenched into water from a dynamic vacuum and subsequently aged in evacuated silica capsules. When hydrogen was added, this was done at 800°C and followed by a 72 hr homogenization at this temperature before the quench and age. The latter specimens were solution treated in evacuated silica capsules and quenched by breaking these capsules under water. When hydrogen was not an important variable in the tests the 72 hr homogenization at 800°C was not used. After the quench and age heat treatment, the specimens had a duplex crystallographic texture with basal plane normals both parallel and perpendicular to the tensile axis.

2.3 Irradiation

Specimens were irradiated in fast neutron facilities in the NRX and NRU Reactors at CRNL, Table II. Temperatures during irradiation were monitored by thermocouples peened into some specimens in each charge. Specimens were in contact with air during irradiation, but no effect which could be attributed to the oxide layer formed has been observed. Hydrogen

concentrations were <20 ppm after irradiation, except where added deliberately. The integrated fluxes quoted here were determined either with iron monitors or from reactor power and previous flux calibration. These flux values are considered to have an absolute accuracy of ±30%, with rather higher relative accuracy between individual irradiations.

TABLE II

IRRADIATION FACILITIES

Facility	Reactor	Specimen Temperature Range Used (°C)	Flux Normally Obtained $n/cm^2/sec$	
			E>1.0 MeV	E>500 eV
MK I Transformer Rod	NRX	50-100	10^{13}	3×10^{13}
MK IVA Transformer Rod	NRX	250-325	10^{13}	3×10^{13}
NRU Fast Neutron Facility	NRU	250-325	4×10^{13}	1.2×10^{14}
MK III Transformer Rod	NRX	300-450*	$1\text{-}2 \times 10^{13}$	$3\text{-}6 \times 10^{13}$

* Upper temperature limit not yet established

The 500 eV flux values in Table II are listed to permit comparison with previous reports from CRNL; 1.0 MeV values will be used throughout this report.

3. EXPERIMENTAL RESULTS

3.1 Tests with Low Hydrogen Content

When the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy was quenched from ≈40°C below the β-transus followed by aging 6 hr at 535°C, a hardness typically 260 VPN was obtained. The yield stress at 300°C was ≈75,000 psi, with uniform elongation and reduction in area 4% and 70% respectively. No yield points were revealed in the stress strain curves over the temperature range from room temperature to 450°C, Figure 1.

Some apparently significant variations in tensile properties were observed for the same nominal condition, e.g. yield stress at 300°C varied from 70-80,000 psi. These differences appeared to be due to heat treatment procedure and irradiation results in this report, therefore, are compared with material heat treated in the same batch. Detailed tensile properties, considered typical for the unirradiated material, are shown in Figure 2 for the temperature range of interest here.

During irradiation at 50-325°C, the material normally lost a large part of its uniform elongation, but underwent only moderate increases in yield strength -- $\leqslant$25% for an integrated flux of 5-6 x 10^{19} n/cm^2, Table III. Changes in fracture stress and reduction in area were $\leqslant$20% at this flux,

TABLE III

EFFECT OF NEUTRON IRRADIATION ON THE TENSILE PROPERTIES OF THE Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY

Material quenched from ($\alpha+\beta$)-phase and aged 6 hrs at 535°C prior to irradiation. Percentage changes measured to be <10% are listed as zero.

Irradiation Temp. °C	Integrated Flux n/cm^2 E>1.0 MeV	Test Temp. °C	Change in Property Percent of Unirradiated Value				
			0.2% Yield Stress	Ultimate Tensile Strength	Fracture Stress	% Uniform Elongation	% Reduction In Area
<100	5.3×10^{19}	RT	+17	+13	0	-80	0
<100	5.3×10^{19}	300	+25	+19	+13	-55	0
250-325	6.2×10^{19}	RT	+14	+10	0	-66	0
250-325	6.2×10^{19}	300	+16	+15	-20	-28	0
375	1.2×10^{20}	RT	+10	+10	0	-25	-12
375	1.2×10^{20}	300	+20	+27	+13	-18	0

Table III, or for irradiation up to 3×10^{20} n/cm^2. The yield stress was an increasing function of integrated flux up to 3×10^{20} n/cm^2, Figure 3, but appeared to follow a saturation relation of the form

$$\Delta\sigma = 27{,}000\ [1-\exp(-1.02 \times 10^{-20}\phi)]^{\frac{1}{2}}$$

where $\Delta\sigma$ and ϕ are yield stress increment in psi and integrated flux in n/cm^2 respectively. When the irradiation temperature was raised to 375°C, the increase in yield strength and decrease in uniform elongation tended to be lower, Table III. Reducing the irradiation temperature to <100°C, as compared to 250-325°C, made little difference to the results, except perhaps giving more loss in uniform elongation, Table III.

3.2 <u>Tests with Added Hydrogen</u>

Hydrogen concentrations up to 250 ppm had very little effect on tensile properties of the unirradiated material for the test conditions used, Table IV. Irradiation of hydrided specimens to 9×10^{19} n/cm^2, however, tended to result in a significant decrease in reduction of area on subsequent tests, Table IV, but had little other effect on the properties.

The quench and age heat treatment resulted in a structure of equiaxed alpha grains in a matrix of transformed beta phase, Figure 4(a). The addition of hydrogen decreased the amount of equilibrium alpha phase relative to the material

TABLE IV

EFFECT OF HYDROGEN CONTENT AND IRRADIATION ON TENSILE PROPERTIES

OF THE Zr-2.5 wt% Nb - 0.5 wt% Cu

Metallurgical Condition	Nominal Hydrogen Concentration ppm	Test Temp. °C	0.2% Yield Stress psi x 10^{-3}	Ultimate Tensile Strength psi x 10^{-3}	Fracture Stress psi x 10^{-3}	Reduction in Area %	Uniform Elongation %
As heat treated	20	RT	102	113	174	52	4.5
	100	RT	113	123	178	49	3
	250	RT	113	122	174	48	3
Held 3 months at 300°C in air	100	RT	106	117	172	49	4
	250	RT	102	115	168	49	4
Irradiated at 270°C to 9 x 10^{19} n/cm^2	20	RT	124	134	177	45	1.5
	100	RT	135	140	177	42	1.5
	250	RT	138	143	168	30	1.5
As heat treated	20	300	73	83	152	67	3
	100	300	81	88	142	61	2
	250	300	79	86	146	65	2.5
Held 3 months at 300°C in air	100	300	73	82	127	57	3
	250	300	73	82	144	65	3
Irradiated at 270°C to 9 x 10^{19} n/cm^2	20	300	97	102	171	68	1.5
	100	300	98	102	139	50	1.75
	250	300	101	104	134	45	1.25

of low hydrogen content, Figure 4(b) compared to Figure 4(a). No hydrogen was apparent in the alpha phase, Figure 4(b), confirming previous observations on this material (2). Roy (3) has confirmed the validity of this observation with an autoradiographic technique using tritium in place of hydrogen.

3.3 Stability Tests

Specimens in the ($\alpha+\beta$)-quenched and aged condition were held in air at 300°C for periods up to 650 days, both for comparison with material irradiated at elevated temperatures and to check on the long term stability at this temperature. Tensile tests on specimens at intervals throughout this period showed no apparent trend in change of property, Table V. Specimens held for the full 650 days developed a film of grey oxide that evidently had little effect on tensile properties.

In a second type of stability test, specimens irradiated (at 250-300°C) and control specimens of material quenched from the ($\alpha+\beta$)-phase (and aged) were annealed for 1 hr at a series of temperatures before test at RT. Under these experimental conditions, tensile properties of the irradiated material did not change significantly until the annealing temperature was 500°C, Figure 5. At this temperature, however, the unirradiated material also softened. After 1 hr at 800°C, both yield stress and uniform elongation of the as-heat-treated material and material at the two levels of irradiation had returned to the values for fully annealed (slow cooled from ($\alpha+\beta$)-phase) alloy, Figure 5.

TABLE V

TENSILE PROPERTIES OF Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY

AFTER HOLDING VARIOUS TIMES AT 300°C

AM material, quenched from 825°C and aged 6 hrs at 535°C prior to further treatment

Days Held at 300°C	Test Temp. °C	0.2% Yield Stress psi x 10^{-3}	Fracture Stress psi x 10^{-3}	% Uniform Elongation	% Reduction in Area	Ultimate Tensile Strength psi x 10^{-3}
As heat treated	RT	108	188	6	54	123
80	RT	104	183	5	55	121
160	RT	108	188	4	55	123
420	RT	105	180	5	54	116
650	RT	106	180	5	53	118
As heat treated	300	78	147	3.5	66	88
20	300	77	145	3	69	86
40	300	77	152	3	72	89
80	300	78	160	2.5	74	89
160	300	79	191	2	77	91
420	300	78	168	2.5	78	87
650	300	79	163	2.5	74	86

4. DISCUSSION

4.1 Strengthening Mechanisms in the Alloy

Very little detailed study of the quench and age hardening mechanisms in the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy has been reported. It is assumed (4) that these mechanisms are closely similar to those operating in the Zr-2.5 wt% Nb alloy system, with the copper merely altering the kinetics of metallurgical changes. The quench hardening then is associated with niobium in the distorted hexagonal lattice of the transformed β-phase, and further hardening is associated with the formation of a niobium-rich precipitate in this phase. Overaging would follow from continued growth of these precipitates, with depletion of niobium in the transformed β-phase. Alternatively, overaging may result from developing incoherency between the matrix and niobium-rich precipitate (5).

The basic mechanism of irradiation hardening of pure zirconium or zirconium-rich alloys has not yet been identified. In an examination of irradiated zirconium and Zircaloy-2 using transmission electron microscopy, Howe (6) concluded that the irradiation-induced defects must be smaller than can be resolved by the technique he was using. Later, similar experiments by Roy (7) revealed evidence of irradiation in the form of unidentified 'black dots,' which appeared to pin some dislocations. Roy concluded that this pinning would result in yield point

formation, citing work in his laboratory as well as Howe and Thomas (8) in support. Although Bement (9) has also observed yield drops induced by irradiation in annealed Zircaloy-2, recent work (10) established that the reverse effect can occur in zirconium-rich alloys as previously reported for other metals, and hence no generalization can be based on Roy's observations.

The increment in yield strength due to irradiation, Figure 3, follows a saturation equation of the form (11)

$$\Delta\sigma = A[1-\exp(-B\phi)]^{\frac{1}{2}} \qquad \text{..........} \qquad (1)$$

This equation can be derived on the assumption that each neutron generates a fixed number of obstacles to deformation, with the increment in yield stress proportional to the square root of the total number of obstacles generated. If saturation is due to effective overlapping of neutron generated obstacles, and to obstacles already present in the material, the theoretical equation for the yield stress increment has the form

$$\Delta\sigma = K\left(\frac{1-V_o}{V}\right)^{\frac{1}{2}}[1-\exp(-aV\phi)]^{\frac{1}{2}} \qquad \text{..........} \qquad (2)$$

where V is the volume associated with each neutron induced obstacle, V_o is the volume fraction initially forbidden to obstacle formation, a is the number of obstacles generated per neutron in unit distance, ϕ is the integrated flux, and K is the proportionality constant between $\Delta\sigma$ and number of obstacles for small values of ϕ.

From tensile tests alone it is not possible to establish values of V and a, although comparison with results on annealed Zr-2.5 wt% Nb - 0.5 wt% Cu alloy indicates that V_o is $\geqslant$0.75 for the heat treated material (10).

Although there are considerable discrepancies between results reported from various laboratories on post-irradiation annealing tests (6, 8, 12, 13 and 14) of Zircaloy-2, there appears to be a significant trend for irradiation damage in this alloy to anneal out at lower temperatures than in alloys containing niobium. This trend was first pointed out by Cupp (15), in comparing his results on Zr-2.5 wt% Nb alloy with those of Howe and Thomas (8) on Zircaloy-2. Howe (6) subsequently confirmed the original results on Zircaloy-2, and the current results on Zr-2.5 wt% Nb - 0.5 wt% Cu are quite similar to those found by Cupp for the binary alloy. Other experiments by Lemaire, et al. (12) on Zircaloy-2, by Jung-Konig, et al. (13) on Zircaloy-2 and Zr-3 wt% Nb - 1 wt% Sn and by Bement (14) on Zircaloy-2, used annealing temperatures <400°C, and definite comparison with the Zr-2.5 wt% Nb alloys over the temperature range for which annealing occurs in these latter alloys is not possible.

4.2 The Effect of Hydrogen

Work already reported on the effect of hydrogen on tensile properties of polycrystalline zirconium alloys has outlined three pertinent variables affecting the properties:

1. amount of hydride (16),
2. location of hydride, whether intragranular or at grain boundaries (17), and
3. orientation of hydride platelets with respect to the tensile axis (18, 19).

When a moderate amount of hydride is present, i.e. <500 ppm hydrogen concentration, then the orientation of the platelets is of paramount importance (18, 19). Results at the 200-250 ppm hydrogen concentration level from several studies are presented in Table VI, along with current results transcribed from Table IV. Many values in the table are approximate, especially when the absolute values of uniform elongation are quite small, and percentage changes calculated to be <10% are listed as negligible. Apart from the critical case of the highly oriented hydride at room temperature, the results of previous work indicate that the effect on strength is small, the effect on uniform elongation is variable and the effect on reduction in area is much more marked at room temperature than at 300°C. The difference in ductility between room temperature and 300°C is more than can be attributed to increased solubility of hydrogen at the higher temperature; Parry and Evans (19) have suggested that the hydride itself may have some ductility at temperatures above 150°C.

TABLE VI

THE EFFECT OF HYDROGEN CONCENTRATION ON TENSILE PROPERTIES OF ZIRCONIUM ALLOYS

Alloy	Metallurgical Condition	Hydrogen Concentration ppm	Test Temp. °C	Percent Change in Property Compared to Hydrogen Concentration of <20 ppm			
				0.2% Yield Stress	Percent Uniform Elongation	Percent Reduction in Area	Reference
Zircaloy-2	Annealed at 730°C	200	RT	negligible	---	-36	Burton (16)
		200	300	negligible	---	negligible	
	α-phase anneal before hydrogenation - hydrogen at grain boundaries	250	RT	negligible	negligible	-40	Babyak, et al. (17)
		250	260	negligible	-30	negligible	
	α-phase anneal before hydrogenation - hydrogen intragranular	250	RT	negligible	negligible	-15	Babyak, et al. (17)
		250	260	negligible	-30	negligible	
	β-quench before hydrogenation - hydrogen at grain boundaries	250	RT	negligible	-50	-50	Babyak, et al. (17)
		250	260	negligible	perhaps -15	negligible	
	β-quench before hydrogenation - hydrogen intragranular	250	RT	negligible	-50	-25	Babyak, et al. (17)
		250	260	negligible	probably negligible	negligible	
	Cold-worked 50% - hydride plates normal to tensile axis	200	RT	---	-100	-100	Parry and Evans (19)
		200	300	negligible	---	negligible	
Zr-2.5 wt% Nb	Quenched from α+β - phase (and aged) after hydrogenation	250	RT	negligible	-20	-61	Sawatzky (20)
		250	300	+10	negligible	-13	
	Quenched from α+β - phase (and aged) after hydrogenation and irradiated at 270°C to ≈3 x 10^{20} n/cm^2	250	RT	negligible	perhaps -20	-77	Sawatzky (20)
		250	300	negligible	perhaps -30	-50	
Zr-2.5 wt% Nb-0.5 wt% Cu	Quenched from α+β - phase (and aged) after hydrogenation	250	RT	+12	-33	negligible	Current results (Table IV)
		250	300	negligible	-15	negligible	
	Quenched from α+β - phase (and aged) after hydrogenation and irradiated at 270°C to 9 x 10^{19} n/cm^2	250	RT	+11	negligible	-33	Current results (Table IV)
		250	300	negligible	-15	-34	

The current results show that the response of the heat treated Zr-2.5 wt% Nb - 0.5 wt% Cu alloy to the 250 ppm hydrogen addition is similar to that of other zirconium alloys in strength and uniform elongation at both RT and 300°C, and in reduction of area at 300°C. However, at room temperature the decrease in reduction of area is seen to be surprisingly low, and equalled only by the annealed Zircaloy-2 treated to have all hydrogen intragranular. Weinstein and Holtz (21) have suggested that cracks formed in the prior β-phase of hydrided Zr-2.5 wt% Nb - 0.5 wt% Cu alloy are blunted on propagation into the softer equilibrium alpha phase grains. This mechanism would also be operative in the high ductility Zircaloy-2 obtained by Babyak, et al. On the other hand, it should apply equally well to the binary Zr-2.5 wt% Nb alloy in the $\alpha+\beta$ - quenched and aged condition; the data in Table VI indicate this is not so. The reduction in area value for the binary alloy at room temperature listed in Table VI is an average from six tensile specimens, and results from specimens containing 100 ppm hydrogen were consistent in that some reduction in ductility for this concentration was also observed. In the present work, twelve specimens of 4M material hydrided to 250 ppm were pulled at room temperature. This number of tests indicates that the difference observed is real, but no explanation for it has been established thus far.

For both binary and ternary Zr-Nb alloys, the combined effect of hydrogen and irradiation was to induce rather more

embrittlement than would have been predicted from tests in which only one of these variables was present, Tables IV and VI. The mechanism of interaction between the hydrogen and irradiation induced defects, however, has not been worked out.

5. CONCLUSIONS

The present study indicates that when the Zr-2.5 wt% Nb - 0.5 wt% Cu alloy is quenched from the $(\alpha+\beta)$-phase and over-aged down to a hardness of ≈260 VPN the material is stable at 300°C. Weinstein and Holtz (21) have shown that in this metallurgical condition, the unirradiated alloy is not notch sensitive at room temperature with randomly oriented hydrogen concentrations up to 200 ppm. The present results demonstrate that the material is not severely embrittled by irradiation up to an integrated fast neutron flux of 3×10^{20} n/cm^2: Even specimens prehydrided to 250 ppm retained 45% reduction in area at 300°C after 0.9×10^{20} n/cm^2 (the highest exposure for such highly hydrided specimens).

Certain metallurgical conditions should be avoided. If the alloy is taken into the β-phase immediately prior to the final metallurgical or fabrication operation, then it becomes markedly susceptible to severe embrittlement. Conditions in which very low values of reduction in area have been observed are:

a) Quenched from the β-phase and aged at 400°C (10),

b) Irradiated in either the β-quenched condition or after either slow or rapid cooling from the β-phase (10),

c) Quenched from the $(\alpha+\beta)$-phase after quenching from the β-phase (22), and

d) Cold drawn after cooling from the β-phase (22).

Also, Weinstein and Holtz have shown that after quenching from the β-phase (and aging), the presence of 500 ppm hydrogen will induce both notch sensitivity and delayed failure. This latter result came from material with hydrogen introduced prior to quenching from 825°C, so that the hydrogen had lowered the transformation temperature below this value. The effect of hydrogen addition after heat treatment, say by corrosion reactions, has not been established.

The favourable behaviour of material in the preferred metallurgical condition has been observed from specimens with a duplex crystallographic texture of basal plane normals parallel and perpendicular to the tensile axis. This texture gave at least a pseudo-isotropy of tensile properties (22). If the fabrication sequence were to give anisotropic properties. then the possibility of reducing impact strength in certain direc ns exists (23). Any large deleterious effect of texture in reactor components, however, would probably arise from textural influence on hydride platelet orientation.

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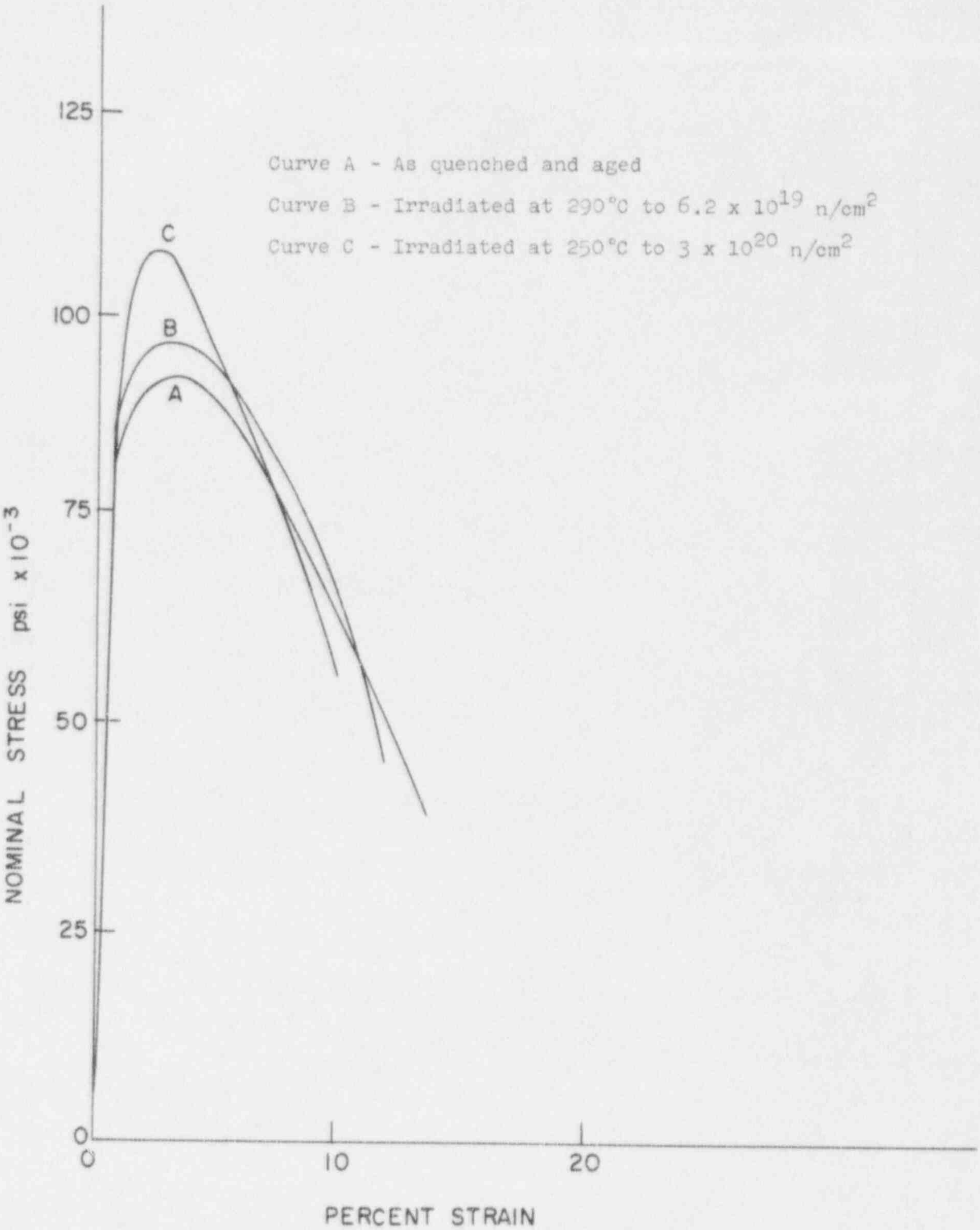


Fig. 1 NOMINAL STRESS-STRAIN CURVES OF Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY QUENCHED FROM ($\alpha+\beta$)-PHASE AND AGED 6 HRS AT 535°C

AM material, tested at 300°C

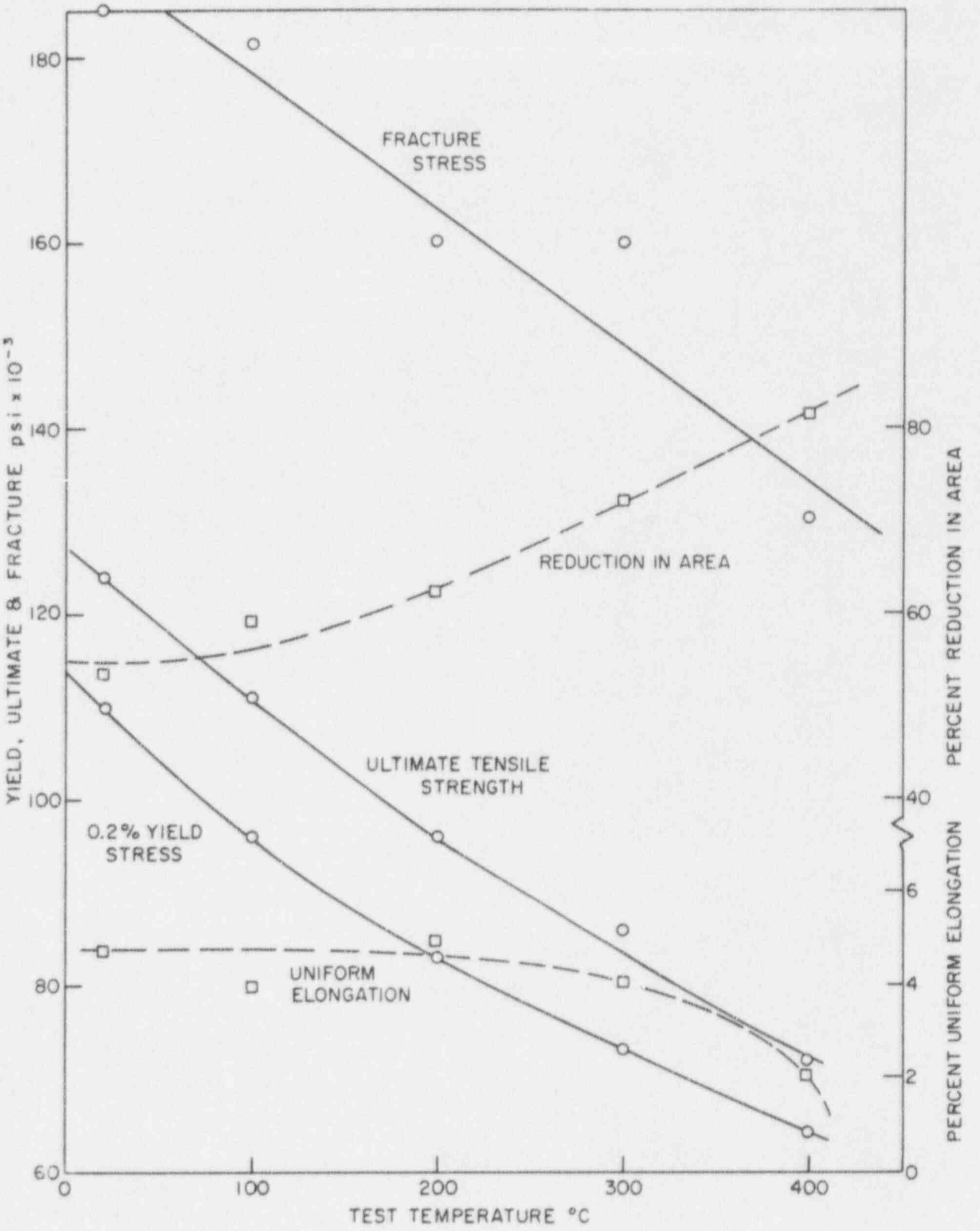


Fig. 2 TENSILE PROPERTIES OF THE Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY IN THE QUENCHED AND AGED CONDITION

AS material, quenched from 850°C and aged 6 hrs at 535°C

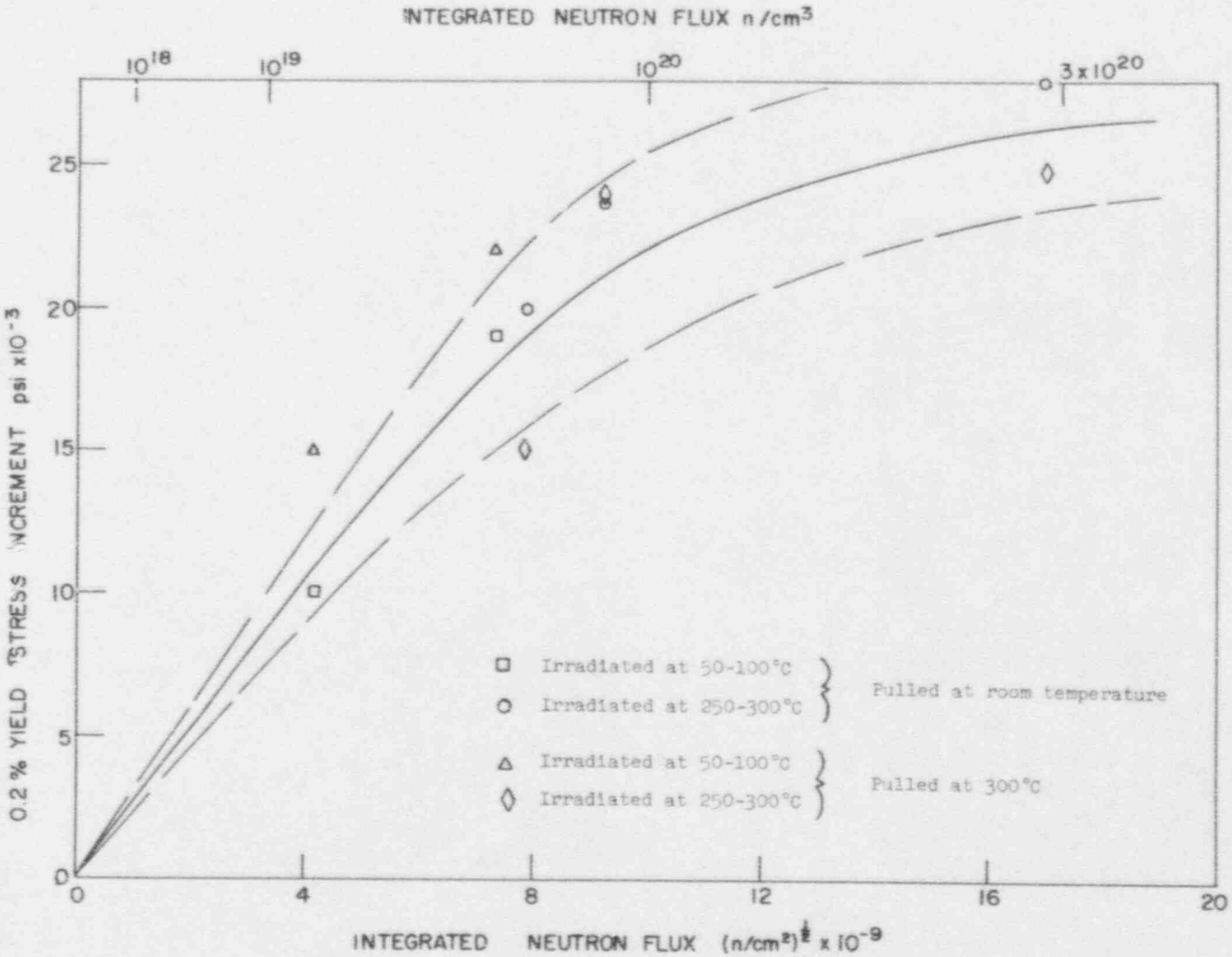
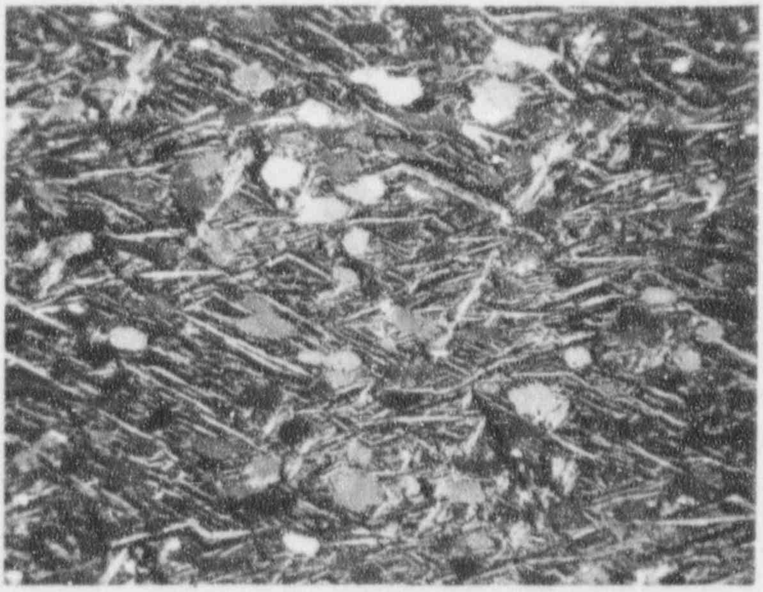


Fig. 3 INCREMENT IN 0.2% YIELD STRESS AS A FUNCTION OF IRRADIATION FOR Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY QUENCHED FROM $(\alpha+\beta)$-PHASE AND AGED 6 HRS AT 535°C

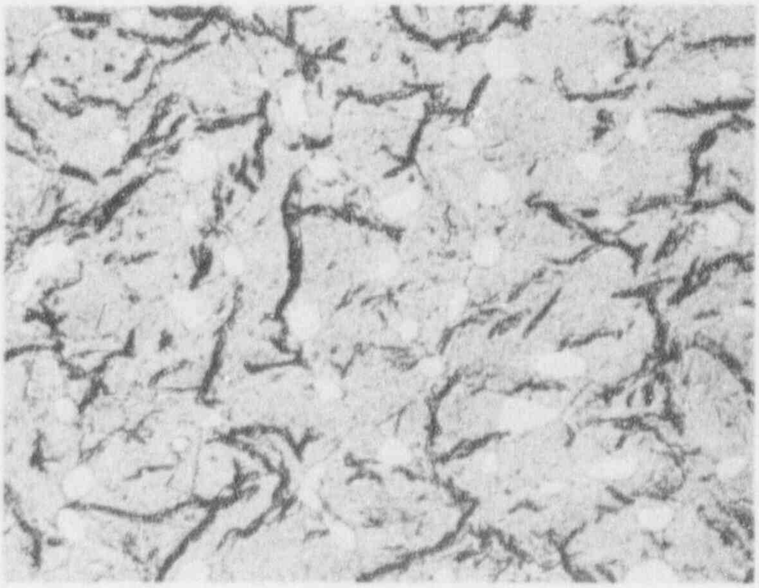
The full line is calculated from the equation
$\Delta\sigma = 27{,}000\ [1-\exp(-1.02 \times 10^{20}\phi)]^{\frac{1}{2}}$, the dashed lines are calculated from the equations
$\Delta\sigma = (27{,}000 \pm 10\%)\left\{1-[\exp(-1.02 \pm 10\%)\ 10^{20}\phi]\right\}^{\frac{1}{2}}$

Fig. 4(a)



AM material quenched from 825°C and aged 6 hrs at 535°C. Chemically polished, anodized, photographed in polarized light. Structure of equiaxed prior alpha in transformed beta. Photographed to reveal martensitic appearance of transformed beta-phase.

Fig. 4(b)



AM material with 250 ppm hydrogen added, homogenized at 800°C, quenched from 825°C, and aged 6 hrs at 535°C. Chemically polished, bright field illumination. Structure of equiaxed prior alpha and hydride platelets in matrix of transformed beta.

Fig. 4

STRUCTURE OF UNDEFORMED Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY IN $\alpha+\beta$ QUENCHED AND AGED CONDITION

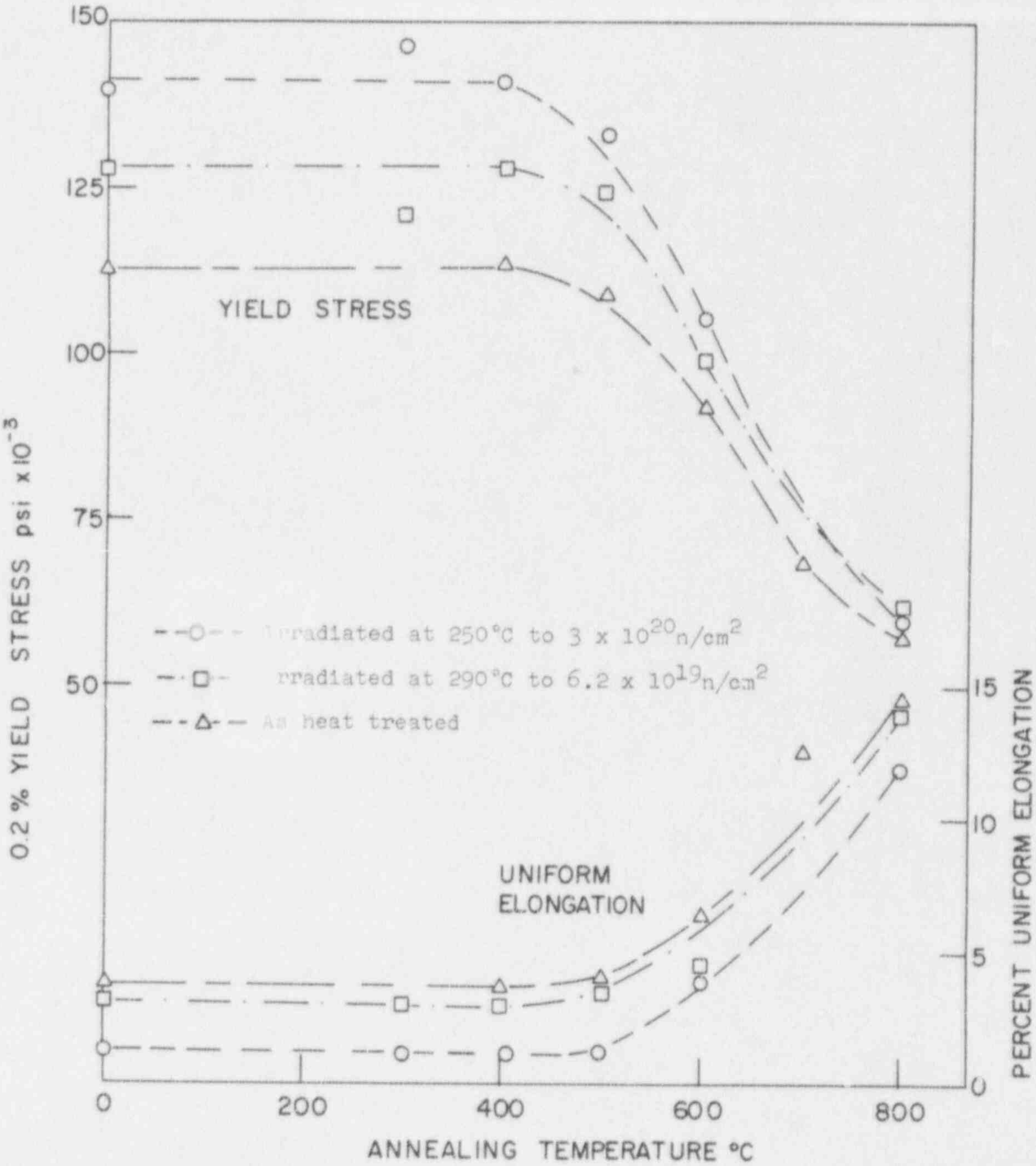


Fig. 5 EFFECT OF POST-IRRADIATION ANNEAL ON TENSILE PROPERTIES OF Zr-2.5 wt% Nb - 0.5 wt% Cu ALLOY

AM material quenched from (α+β)-phase and aged 6 hrs at 535°C prior to irradiation

Specimens annealed 1 hr at temperature shown and pulled at room temperature