THE EFFECT OF NUCLEAR RADIATION ON FAST FRACTURE BEHAVIOUR OF METALS

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ABSTRACT

In the past the changes in the yield stress, tensile strength and ductility of metals produced by nuclear radiation have been grouped together under such terms as radiation damage or radiation embrittlement, with the attendant implication that generally they are detrimental and indeed that ultimately they may result in a component becoming unsafe for further service. Recently there has been a growing acceptance of the concept that a large percentage of service failures occur due to the development of cracking and the use of this as a basis for a more realistic and rational model for ultimate failure has led to the formulation of a number of theories of fracture mechanics. However the majority of these are only suitable for use over a limited range of stress and because they are virtually single parameter approaches, are quite incapable of describing the conflicting influences, of say, tensile strength and yield strength.

This paper briefly describes a stress concentration approach to fracture mechanics which has been developed in the UKAEA. By considering the equilibrium of the section along the axis of the crack, the load which would have been carried by the cracked material can be equated to the extra load carried by the high stress regions at the crack tips, and thus to the size of this region and its stress level at failure.

This theory is extended to cover cylindrical pressure components and is used to describe the overall effect of the changes mentioned above. It shows that these changes may either increase or decrease the critical crack size depending on the level of stress in the component.

Using experimental data for the bursting strengths of irradiated and unirradiated Zr Nb pressure tubes, containing artificial flaws, in conjunction with material properties relevant to these materials, it is shown that a reasonably accurate prediction of the fast fracture behaviour of the irradiated tube can be made.

INTRODUCTION

1. The effect of nuclear radiation on the mechanical properties of metals is to produce an increase in yield strength and fracture stress (U.T.S.) and a reduction in ductility. The development of a theory of fracture mechanics [1, 2, 3] has enabled the effect of radiation induced changes on fast fracture behaviour to be predicted.

2. This paper explains the basic behaviour on which the governing equations have been based and shows the predictions of these equations to be in accordance with the observed behaviour.

BASIC THEORY

3. Consider the case of a through crack in a flat plate of infinite area under uniaxial stress normal to the crack axis (see Fig. 1).

Effect of Yield Stress

4. The value of yield stress f_y relative to the general field stress or gross stress f_g determines the size of the plastic zone at the tips of the crack, e.g., when $f_y = f_g$, the size of the plastic zone is infinite - the whole plate is plastic and when the yield stress is equal to the fracture stress the size of the plastic zone is zero.

5. To sum up, increasing the yield stress decreases the size of the plastic zone.

Effect of Fracture Stress

6. The effect of the crack in re-distributing load is shown in Fig. 1. Here the load shed by the cracked material is 2 $f_g l$, but since the crack is symmetrical, only one half need be considered; that is, the load which is shed by one half of the crack is $f_g l$ and because of the need for equilibrium on the crack section, this must be carried by the material at the crack tip. Denoting the fracture stress by F and the average width of the additional crack tip load diagram by S, then at failure at crack tip:-

$$f_g l = (F - f_g)S \qquad .....(1)$$

However S is proportional to plastic zone size (3):-

$$S = \text{plastic zone size} \times \text{shape factor} \qquad .....(2)$$

The shape factor is controlled by the magnitude of the stresses in the thickness direction in the plastic zone, i.e., lateral contraction, and a measure of this can be obtained from a notch ductility test, such as the Charpy V.

7. Thus S at failure is dependent only on yield strength (see paras. 4 and 5) and notch ductility. Re-writing Eqn. (1):-

$$f_g = \frac{F\,S}{l + S} \qquad .....(3)$$

Thus for a given crack size l and constant yield strength and notch ductility the general field stress at failure may be conveniently represented as shown in Fig. 2.

Effect of Radiation Induced Changes in Material Properties

8. The separate effects of yield stress and fracture stress on the load carrying capacity of the crack tip region at failure have been described above and may be summarised as - increase in fracture stress F increases load bearing capacity (see Eqn. (3)) and an increase in yield stress decreases S and therefore decreases loading bearing capacity.

These two effects superimposed are illustrated by the use of published data on 5 ft, 1 in. thick steel pressure vessels for two steels of differing fracture stress and yield point. It will of course, be realised from Eqn. (1) that the fracture stress F is the value of the general field stress f_g at failure when $l = 0$, i.e., the ultimate strength of the flaw free material under the appropriate stress conditions. In the case of steel, and indeed most materials, this coincides with the U.T.S. Figure 3 shows this data plotted in the same form as Fig. 2. The Charpy values for the two materials are nearly equal and so the difference in slope is due to the differing yield stresses. It will be noted that the lines for the two materials cross over and that the material with the higher critical crack opening displacement and fracture toughness, i.e., the 0.36%C steel has the lowest critical crack lengths at high stresses and the highest critical crack lengths at low stresses. Thus, a single parameter approach is incapable of placing these materials in their correct order of crack tolerance over the complete stress range. This observation is not new. As far back as 1960, G. K. Manning pointed out the inability of linear elastic fracture mechanics to correctly evaluate the relative crack tolerance of two high strength steels at short crack lengths [4]. His data are shown on Fig. 4 in original form. Thus, while C.O.D. and L.E.F.M. methods may be alright for describing fracture data over limited ranges at relatively low stress values, they are not suitable for analysing the complete range of radiation effects.

9. Figure 5 depicts the effect of radiation-induced increases in U.T.S. and yield stress on the critical crack length-stress relationship. It will be noted that due to the increase in slope and increase in U.T.S., cross over is inevitable; thus, there is an improvement in crack tolerance at high stress levels and a deterioration at low stress levels.

Effect of Cylindrical Geometry

10. If the member containing the crack is a pressure component instead of a flat plate, then additional stresses, due to the bending action of the pressure along the unsupported edges of the crack, are set up at the crack tip. When the length of the crack approaches the same order of size as the diameter of the component these additional bending stresses augment the in-plane stresses sufficiently to cause failure to occur at significantly lower values of internal pressure than would have been necessary in their absence. This type of behaviour has been analysed in Ref. 2, for cylindrical geometry and for a number of materials. It is shown that Eqn. (1) can be extended to:-

$$F - f_g \frac{l}{S} - f_g Kl^2 = f_g \qquad \text{.....(4)}$$

which can be represented as shown on Fig. 6. The term $f_g Kl^2$ is the bending stress component, i.e., the amount of 'droop' from the straight line relationship. It is further shown in Ref. 2 that K may be broken down to:-

$$K = \frac{k}{D^2} \qquad \text{.....(5)}$$

where D is the diameter of the cylinder and $k \propto f_y^{\,3}$.

Thus, the effect of an irradiation induced increase in f_y is to cause an increase in K, i.e., to cause the f_g v $f_g l$ characteristic to droop at an increased rate as depicted in Fig. 7.

CORRELATION WITH BEHAVIOUR OF ZIRCONIUM - 2½% NIOBIUM ALLOY PRESSURE TUBES

11. The burst tests carried out on one of the experimental Zr-Nb pressure tubes (G 09), withdrawn from Winfrith SGHW reactor in mid-1969, are probably the most comprehensive and valid full scale tests carried out to date on an irradiated pressure component of an operational reactor. The tube was cut into 16 in. lengths and these each had a through thickness slit of different length machined in it in the axial direction. Burst data on the same size of tube with the same composition, but in the unirradiated condition is also available. The data from the unirradiated and irradiated tests are given in Tables I and II respectively. Both sets of tests were carried out at 300°C.

12. Insertion of the test data in Eqn. (4) gave sets of simultaneous equations which allowed the material constants F, $\frac{1}{S}$ and K to be determined for both the unirradiated and irradiated cases.

For the unirradiated case:-

$$44 = f_g\,(0.81l^2 + 1.53l + 1) \qquad \text{.....(6)}$$

For the irradiated case:-

$$68 = f_g\,(2.18l^2 + 3.45l + 1) \qquad \text{.....(7)}$$

(f_g in t.s.i. l in inches)

The constants used in the equations are average values.

13. In order to check the validity of Eqn. (4) and the fit of Eqns. (6) and (7) with the experimental data $\left(\frac{f_u}{f_g} - 1\right)/l$ was plotted against l for these equations and for the experimental data, on Fig. 8. Since a test was not carried at l = 0 for the irradiated case the value of F or f_u used for plotting the experimental data was taken as that obtained by solving the equations, i.e., 68 t.s.i. A good fit between the data and the straight line representing the equations is obtained in each case. Equations (6) and (7) are superimposed in Fig. 9 (individual points are not shown as they cannot be distinguished from the plot of the equations).

PREDICTION OF EFFECT OF IRRADIATION FROM CHANGES IN MECHANICAL PROPERTIES MEASURED BY SMALL SPECIMENS

14. In making a prediction of irradiated behaviour from the unirradiated behaviour described by Eqn. (6), the material constants must be modified according to the changes in mechanical properties [5].

Thus:-

$$K_i = K_{ui}\left(\frac{f_{yi}}{f_{yui}}\right)^3 \text{ (see para. 10)} \qquad \text{.....(8)}$$

where suffix $\quad i \quad$ denotes the irradiated value

and suffix $\quad ui \quad$ denotes the unirradiated value

$$\left.\begin{aligned} f_{yi} &= 109 \text{ KSi} \\ f_{yui} &= 75.5 \text{ KSi} \end{aligned}\right\} @ \ 300^{\circ}\text{C}$$

Therefore:- $\quad K_i = 0.8 \times \left(\frac{109}{75.5}\right)^3$ (From Eqns. (6) and (8))

$$= 2.4$$

Again slope $A = \frac{1}{S}$

$$\frac{A_i}{A_{ui}} = \frac{\frac{1}{S_i}}{\frac{1}{S_{ui}}} = \frac{S_{ui}}{S_i} = \left(\frac{\phi_{ui}}{\phi_i} \times \frac{f_{yi}}{f_{yui}}\right)^{\frac{1}{2}} \times \frac{\beta_{ui}}{\beta_i} \quad \text{(see Ref. 3)} \qquad .....(9)$$

where $\quad \phi$ = Charpy energy (ft lbs)

f_y = Yield stress

$\beta = (1 - 0.7R)^{-1}$

R = Fractional reduction of area in tensile test

$R_i = 0.06$

$R_{ui} = 0.5$

$\frac{\phi_{ui}}{\phi_i} = \frac{2.2}{1.5}$

[Ref. 5

Irradiated data relevant to material with 200 ppm H_2 and 4 to 7 x 10^{20} neutrons/cm^2 (> 1/Mev)]

Substituting these values and values of f_{yi} and f_{yui} in Eqn. (9) gives:

$$\begin{aligned} A_i &= A_{ui} \times 1.9 \\ &= 1.53 \times 1.9 \\ &= 2.92 \end{aligned}$$

15. The data used in the above prediction of A_i and K_i has been for typical material as distinct from values from test pieces cut from the remnants of the burst tube. Such test pieces have been machined from the irradiated tube but have not yet been tested.

16. In addition this also means that in determining F_i from the relation:-

$$F_i = F_{ui} \times \frac{UTS_i}{UTS_{ui}} \qquad .....(10)$$

only typical values of UTS are available.

17. These give, in conjunction with F_{ui} from Eqn. (6):-

$$F_i = 44 \times \frac{130}{90} = 64 \text{ t.s.i.}$$

18. Thus the equation predicted for the irradiated tube, from the equation for the unirradiated tube and measured changes in the mechanical properties of typical material is:-

$$64 = f_g\,(2.4\,l^2 + 2.92\,l + 1) \qquad .....(11)$$

and in the range of interest this predicts failure stresses and critical crack lengths which are within 10% of the experimentally determined values.

DISCUSSION

19. From Eqns. (6) and (7) and Fig. 8, it would appear that the failure initiation behaviour of the unirradiated and irradiated pressure tubes is in close agreement with the basic theory described in paras. 3-10. The shape of the curves (see Fig. 9) makes it obvious why the expression:-

$$f_g l = \text{Const.}$$

is an approximate fit to the data over a limited range, i.e., in the "knee region".

20. Also it is fairly easy to see why burst experiments carried out in the region l = 0.4 in. give the impression that irradiation does not have much effect on fracture behaviour, sometimes giving a slight increase in fracture strength and sometimes a slight decrease.

21. From the curves on Fig. 9 it would appear that the cross over point for this tube size and material will always be above design stress level and therefore from the practical viewpoint the effect of irradiation is to reduce crack tolerance in this case.

22. However, in the more general case where bulging stresses are not significant, e.g., in a reactor pressure vessel, the characteristics are as in Fig. 5 and the cross over point can be in a stress range which is of practical interest.

23. In testing the ability of Eqn. (4) to predict the effect of changes due to irradiation it would of course, be preferable to have burst tests and mechanical properties for the same tube, thus eliminating tube to tube variations. However, this was not possible and although it has been necessary to use typical values for the bursting strengths of unirradiated tubes and also typical values for the changes in mechanical properties, the degree of accuracy of the prediction is satisfactory.

24. The strong influence of U.T.S. and yield strength clearly shows the importance of including tensile specimens in reactor material monitoring programmes.

REFERENCES

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[4] Proceedings of the Seventh Sagamore Ordnance Materials Research Conference Transcribed Discussion - Fourth Session. P IV-186/187. Aug. 16-19th, 1960.

[5] COWAN, A. RNML, Private Communication, 1970.

[6] COWAN, A., KIRBY, N. "The application of C.O.D. Measurement to Large Scale Test Behaviour", Proceedings, Symposium on Fracture Toughness Concepts for Weldable Structural Steel. UKAEA, Risley, April, 1969.

TABLE I

Burst Data for Unirradiated Zr-2½% Nb Pressure Tubes

5.14 in. Int. Diameter, 0.16 in. Thick, Test temperature = 300°C

l inches	0	0.5	1.0	1.5	1.5	2.0
f_g t.s.i. @ Failure	43.3	22.8	13.2	8.7	8.3	6.05

TABLE II

Burst Data for Irradiated Zr-2½% Nb Pressure Tube G 09

5.14 in. Int. Diameter, 0.174 in. Thick, Test temperature = 300°C

NEUTRON DOSE (> 1 MeV) 5 to 6 x 10^{20} n/cm^2

l inches	0.25	0.5	1.0	1.5
f_g t.s.i. @ Failure	34	21	10.3	6.15

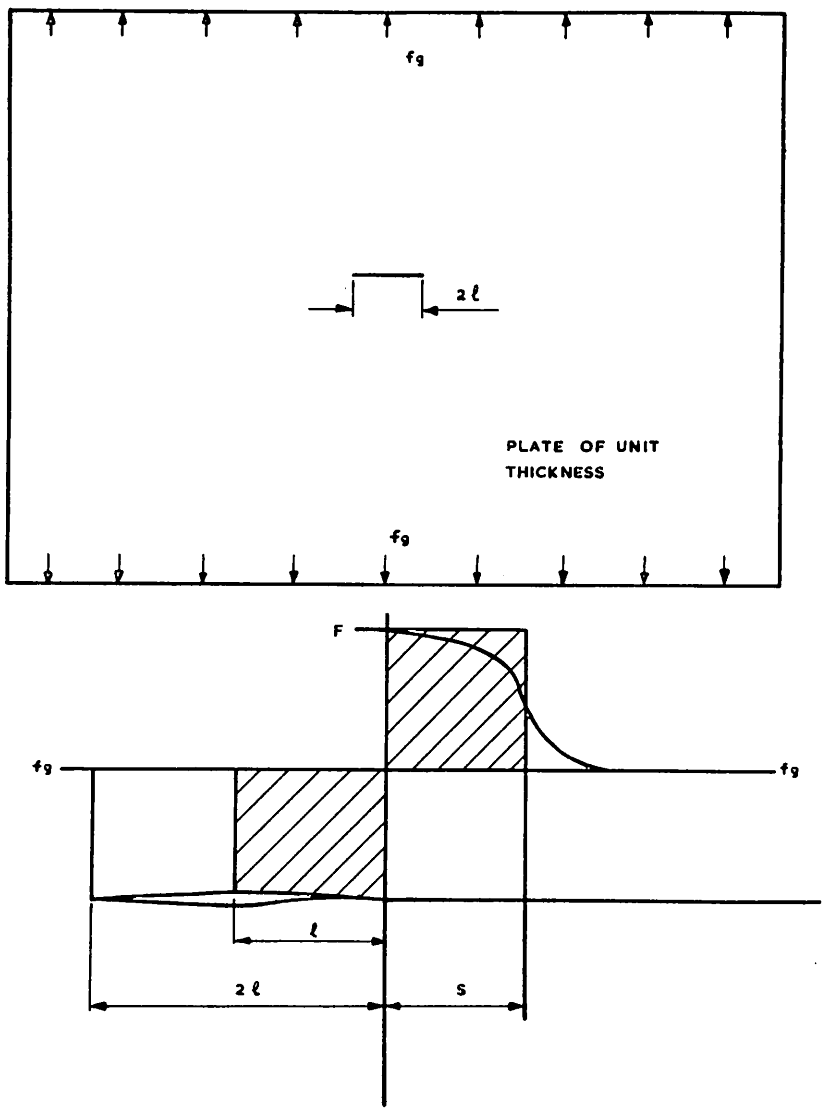


FIG. I

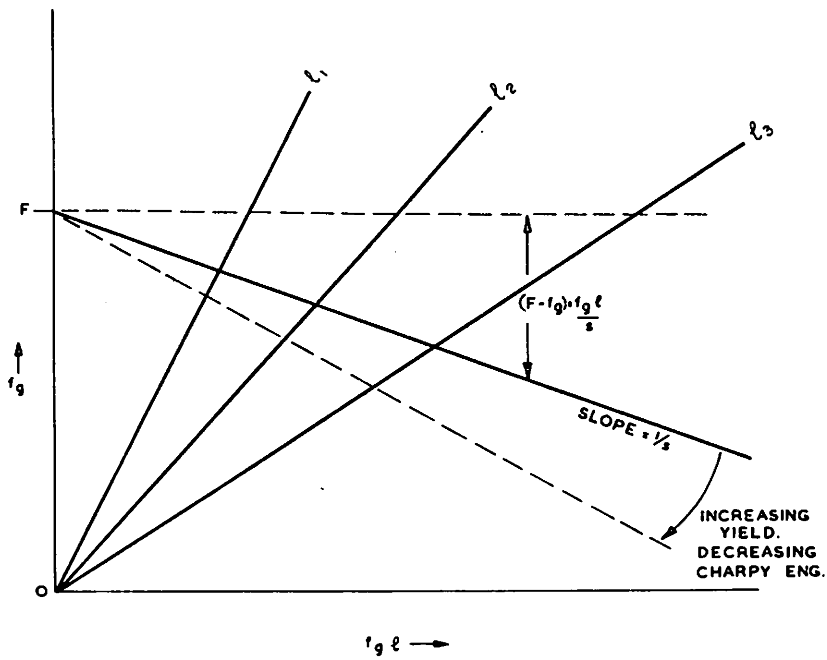


FIG. 2.

GRAPHICAL REPRESENTATION OF EQN. $f_g \ell = (F - f_g) S.$

CHARPY ft.lbs.	STEEL	COD X 10^3	YIELD STRESS	G	REF
8	Mo B	8	33	520	6
12	0·36 % C	25	15	840	6&1

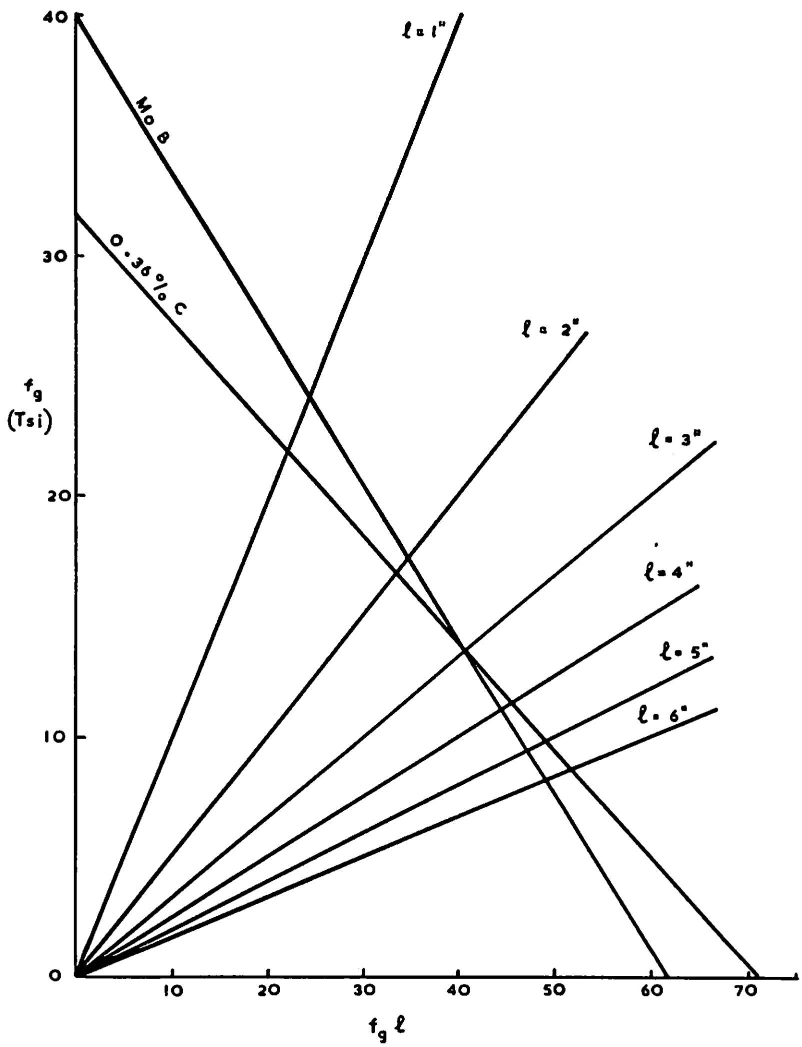


FIG. 3

TENSILE PROPERTIES	4340	H-11
Y.S. psi	225,000	225,000
T.S. psi	260,000	270,000

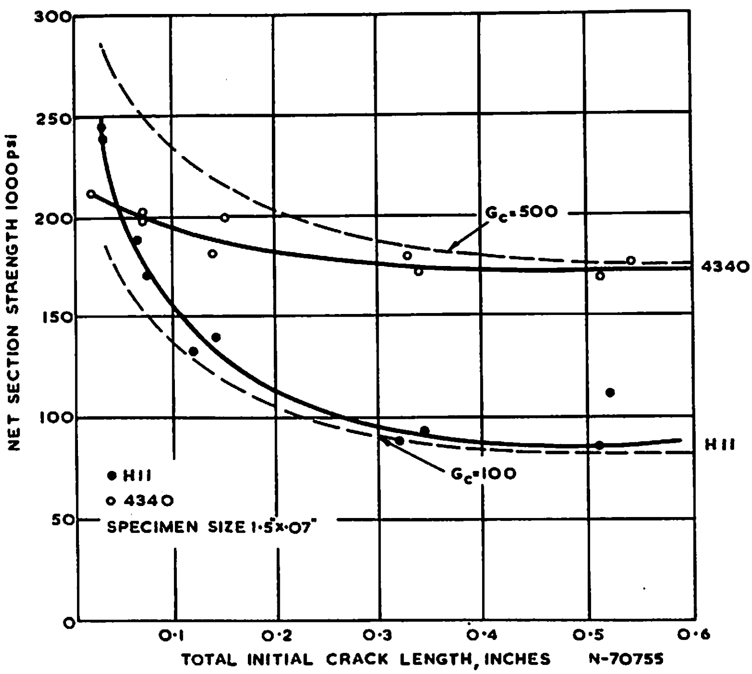


FIG. 4

NET SECTION STRENGTH VS. INITIAL CRACK LENGTH (4340 VS. H-11)

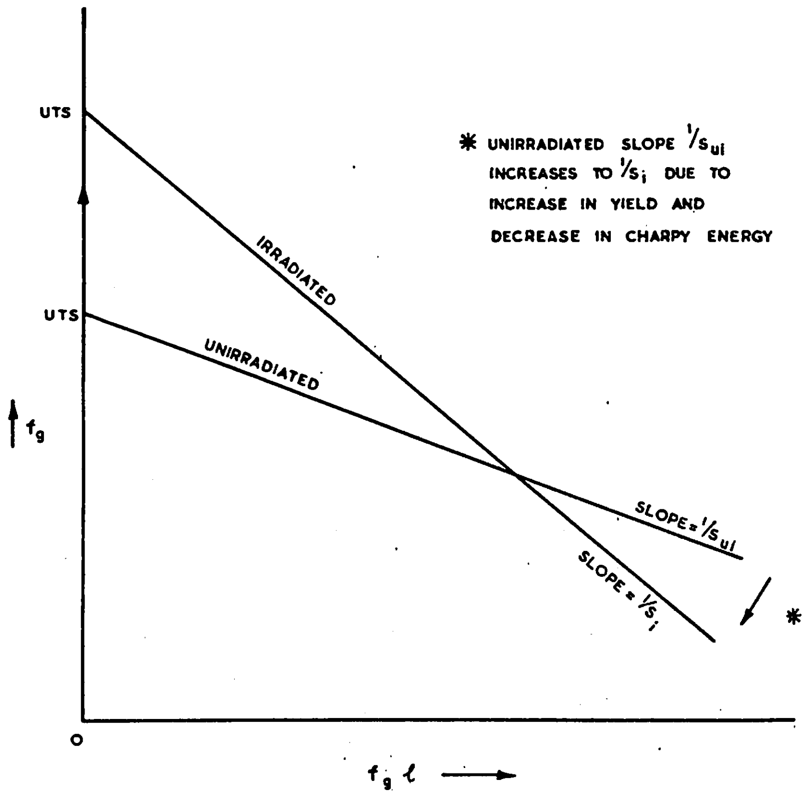


FIG. 5

EFFECT OF IRRADIATION INDUCED CHANGES IN MECHANICAL PROPERTIES ON FAST FRACTURE BEHAVIOUR (FOR ZERO BULGING)

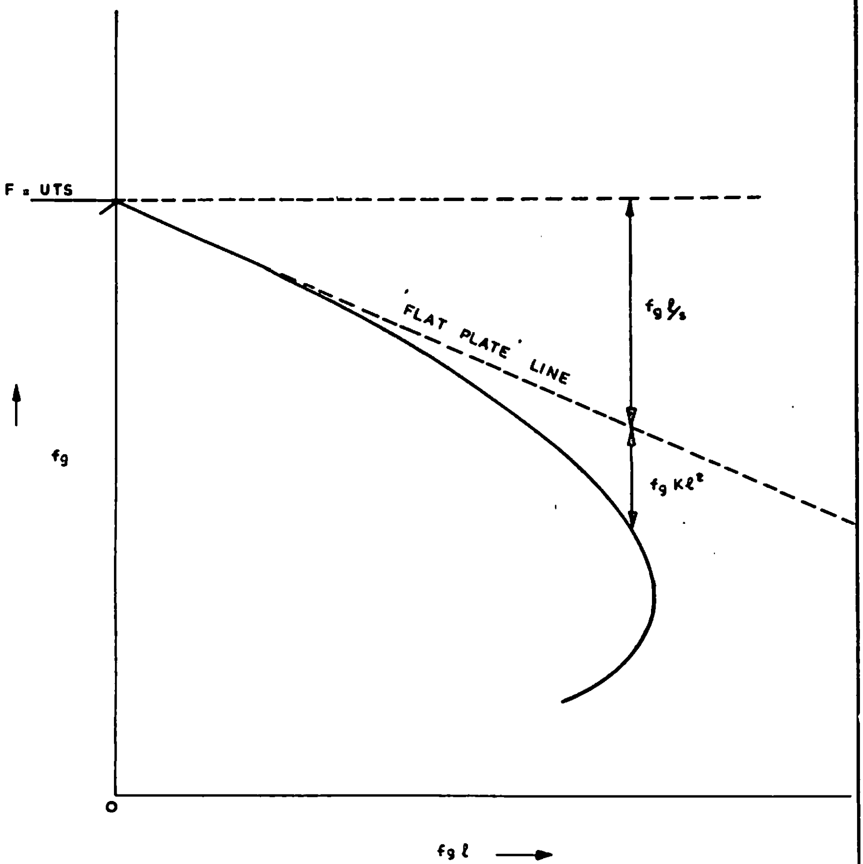


FIG. 6.

GRAPHICAL REPRESENTATION OF EQN. $F - \frac{f_g \ell}{s} - f_g K\ell^2 = f_g$

SHOWING 'DROOP' DUE TO BULGING STRESS $f_g K\ell^2$

CAUSED BY CYLINDRICAL GEOMETRY.

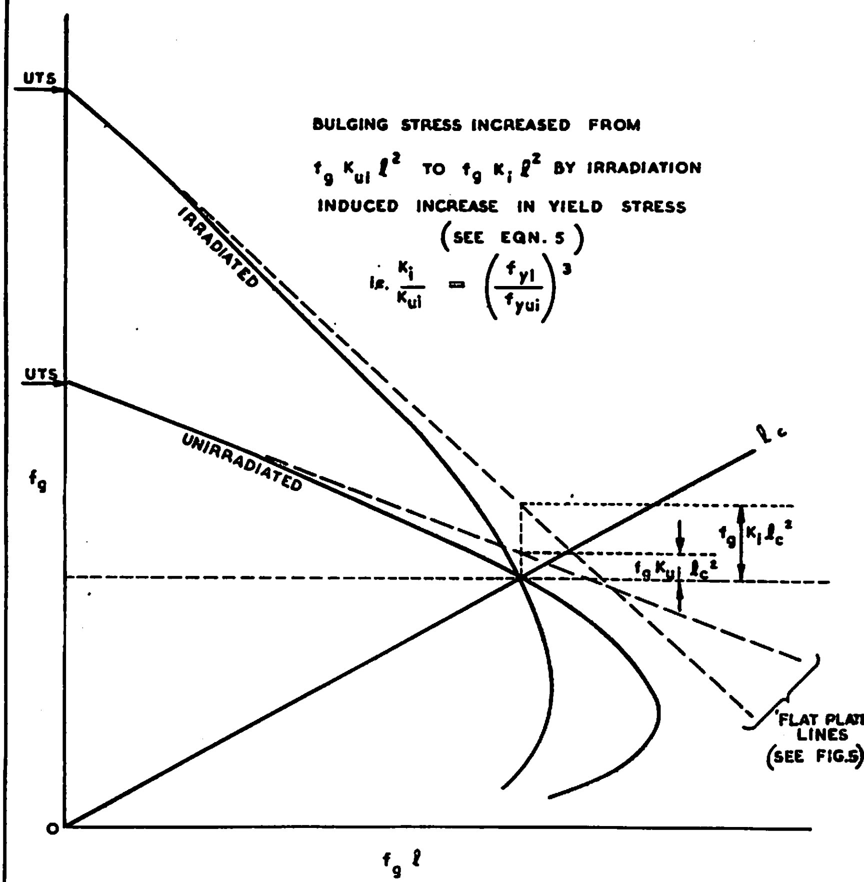
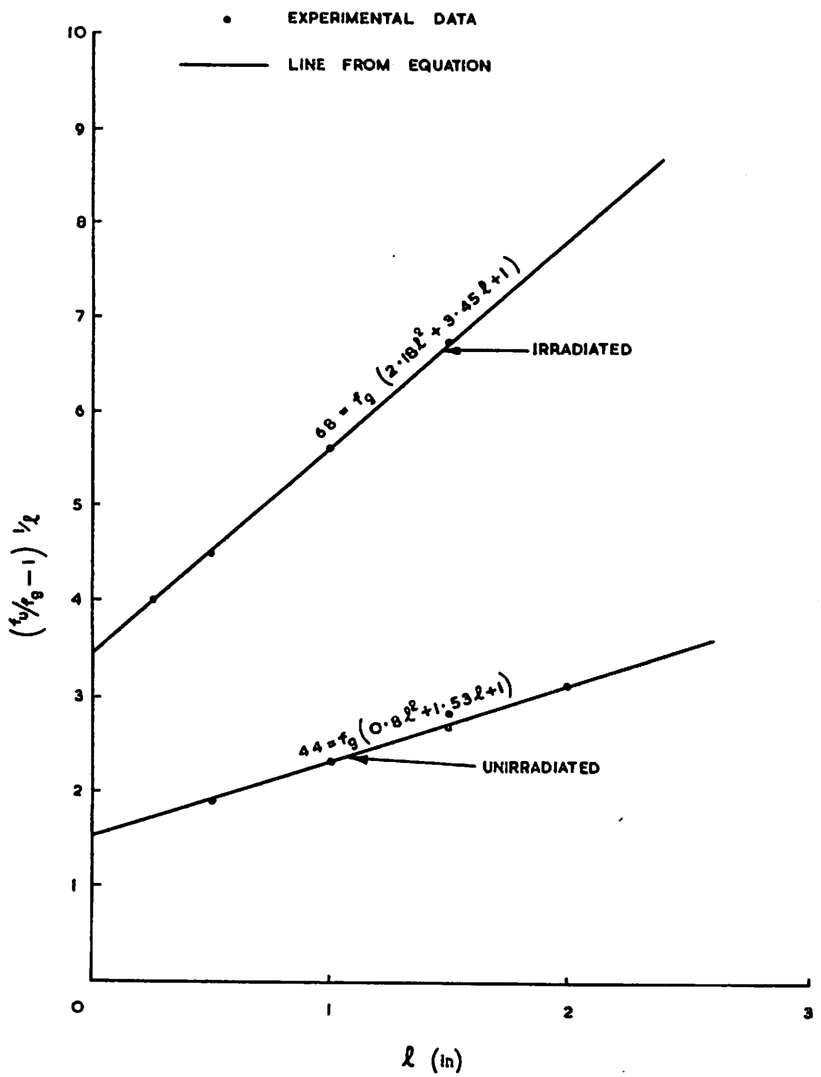


FIG. 7

EFFECT OF IRRADIATION INDUCED INCREASE IN YIELD STRESS ON 'DROOP' DUE TO BULGING STRESS $f_g K \ell^2$



EXPERIMENTAL DATA
LINE FROM EQUATION
IRRADIATED
UNIRRADIATED
68 = fg (2·18ℓ² + 3·45ℓ + 1)
44 = fg (0·8ℓ² + 1·53ℓ + 1)
(fu/fg − 1)^1/ℓ
ℓ (in)
10
9
8
7
6
5
4
3
2
1
0
1
2
3

FIG. 8

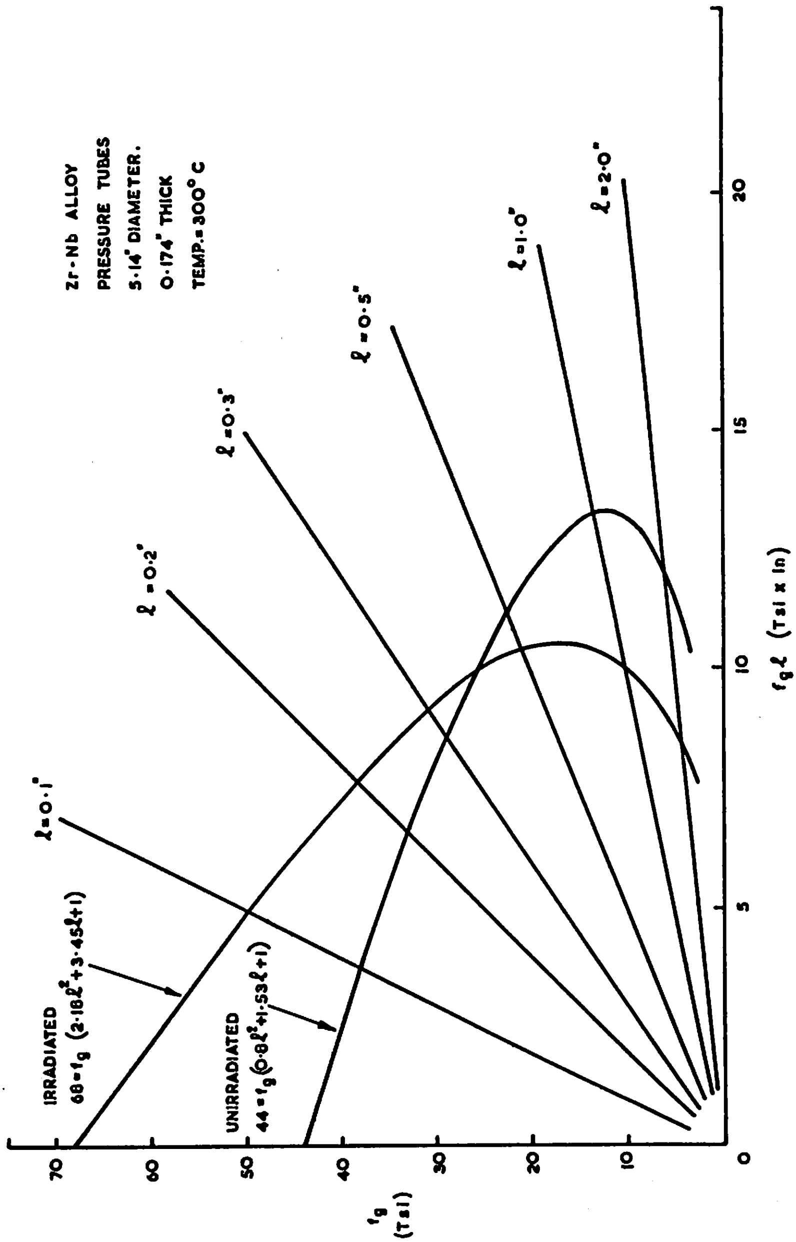

Zr-Nb ALLOY
PRESSURE TUBES
5·14" DIAMETER.
0·174" THICK
TEMP.= 300° C
ℓ = 0·1"
ℓ = 0·2"
ℓ = 0·3"
ℓ = 0·5"
ℓ = 1·0"
ℓ = 2·0"
IRRADIATED
68 = f_g (2·18ℓ² + 3·45ℓ + 1)
UNIRRADIATED
44 = f_g (0·8ℓ² + 1·53ℓ + 1)
f_g
(Tsi)
f_g ℓ (Tsi x In)
0
5
10
15
20
10
20
30
40
50
60
70

FIG. 9

DISCUSSION

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Figure 9 of your paper indicates that in the tube test the curves reach peak values of about 10 to 12, $-f_g l$ (Tsi x in) at values of l in the range 0.5 to 2.0 in. Do you consider the tube test to be too sensitive to geometric effects to adequately demonstrate the material fracture behaviour, or more specifically, a change in material property due to irradiation ?

A W. H. IRVINE, U. K.

The overall effect of radiation on the hoop stress-critical crack length relation for Zr. Nb pressure tubes is summarised in Fig. 9 of my paper. However, the separate effects of increase in UTS and yield stress and decrease in critical plastic zone size are perhaps more clearly brought out by Fig. 8, and here the considerable change in behaviour is obvious. As is explained on page 2 of the paper the opposing effects of increase in UTS and decrease in critical plastic zone size, inevitably lead to cross-over of the f_g-$f_g l$ characteristics even in flat plate tests. Thus individual tests carried out near this cross-over will give the impression of little change in properties and I believe that this, rather than the influence of tubular geometry, is to blame for the <u>apparently</u> small changes in fracture behaviour due to irradiation.