# The Influence of Hydrogen Embrittlement on Safety and Life-Time of Reactor Pressure Vessels

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In addition to specific factors that affect the radiation damage to materials, the combined effect of irradiation and a number of other factors, such as thermal and deformation ageing, hydrogen embrittlement, corrosion and low-cycle fatigue, continue to draw substantial interest.

Harries and Broomfield /1/ studied hydrogen embrittlement and hound that the only source of hydrogen in light-water RPV´s is the corrosion reaction on the internal RPV surface and also that, when a stainless steel lining is used on the RPV walls, the damage caused by hydrogen embrittlement can be neglected.

A number of mechanisms have been proposed, to explain the effect of hydrogen on the properties of steel:

(1) The mechanism proposed by Zappf. Hydrogen recombines on the internal defects in the material to form molecules whose pressure is added to the applied strain, thus decreasing the fracture strain.

(2) The mechanism proposed by Petsch. Adsorption of hydrogen decreases the surface energy, which is decisive for crack deformation.

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(3) Dissolved hydrogen decreases the cohesive forces in the crystal lattice, and facilitates nucleation and crack propogation.

(4) Interaction of hydrogen with dislocations. Two different models have been derived. The first assumes that hydrogen facilitates dislocation motion and thus simplifies plastic deformation in the macro-regions. The second considers the bonding of hydrogen in the dislocations which limits its mobility, leading to local accumulation of hydrogen and embrittlement of the crystal lattice.

The hydrogen embrittlement of steel in RPV´s of the VVER type has been studied in Czechoslovakia for a number of years, especially considering the static plastic properties of these steels /2-5/. The following results were obtained: (a) the range of electrochemical parameters in which hydrogen embrittlement appears for the given type of steel was determined; (b) the effect of neutron irradiation on hydrogen absorption in steel and the effect of irradiation embrittlement on hydrogen embrittlement were clarified. Hydrogen was introduced experimentally by cathodic generation in a 1 N $H_2SO_4$ solution (with 30 mg . $dm^{-3}$ $As_2O_3$) at room temperature for 1 h, with graduated current densities. The hydrogen content was found by vacuum extraction of the fused samples using a modified Balzers EAN-220 analyzer. The samples were irradiated in the VVRS reactor at the Institute for nuclear Research in a probe with a helium environment by a neutron fluence of $2.5 - 4.9 \times 10^{23}$ n . $m^{-2}$ ($E > 0.5$ MeV) at low temperatures.

The following conclusions can be drawn from the tests that have been carried out:

(1) CrMoV steel in both the irradiated and nonirradiated states is very sensitive to hydrogen embrittlement at hydrogen contents above 2.5 ppm. (Fig. 1)

(2) After irradiation at 130 and 180° C at a fluence of greater than $2.6 \times 10^{23}$ n . $m^{-2}$ ($E > 0.5$ MeV), the superposition of hydrogen and radiation embrittlement is marked and, at hydrogen contents of greater than 10 ppm ( $>100$ A . $m^{-2}$), almost complete loss of plasticity can occur. (Fig. 2)

(3) On the other hand, after irradiation at a temperature of 290° C, the increase in the radiation embrittlement is very small and the basic mechanism involves hydrogen embrittlement.

(4) The results of fractographic analysis (plastic fracture in samples with lower hydrogen contents, quasicracking in samples irradiated at a temperature af 290° C, intercrystalline defect formation in samples irradiated at 180° C and 130° C) are in agreement with concepts of the continuity of the structural state of the experimental material and causes of embrittlement.

## REFERENCES

/1/ Steele,L.E. (1975). Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels. Technical Report Series. No. 163, IAEA, Vienna.

/2/ Koutský,J., Otruba,J., Šplíchal,K. (1983). Radiation and Hydrogen Embrittlening of Low Alloy Ferritic Steels for Reactor Pressure Vessels. Collection 7th International Conference on Structural Mechanics in Reactor Technology, 62/9 Chicago.

/3/ Koutský,J., Šplíchal,K. (1983). Hydrogen Embrittlement of Ferritic Steels for Reactor Pressure Vessels. Collection symposia "VTT 43", Espoo, pp. 271.

/4/ Koutský,J., Šplíchal,K. (1986). Int. J. Press. Ves. a. Piping 24, pp. 13.

/5/ Koutský, J., et al. (1987). Theoret. a Appl. Fract. Mech. 8, pp. 21.

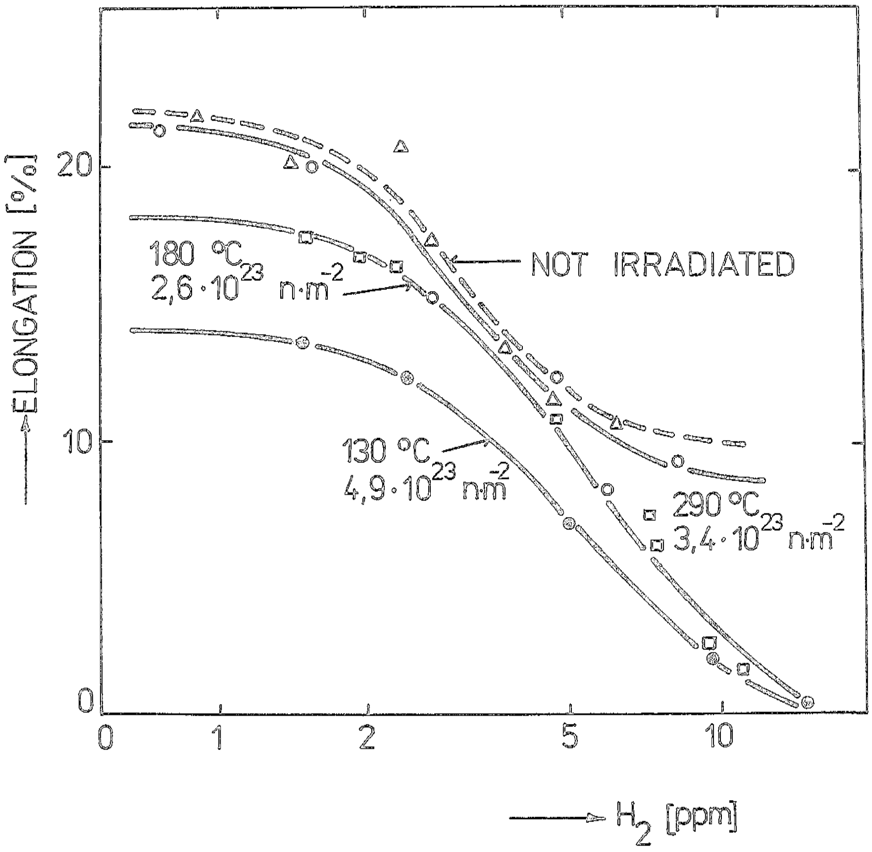

Fig. 1. The whole ductility of irradiated and non-irradiated CrMoV steel in dependance on hydrogen content.

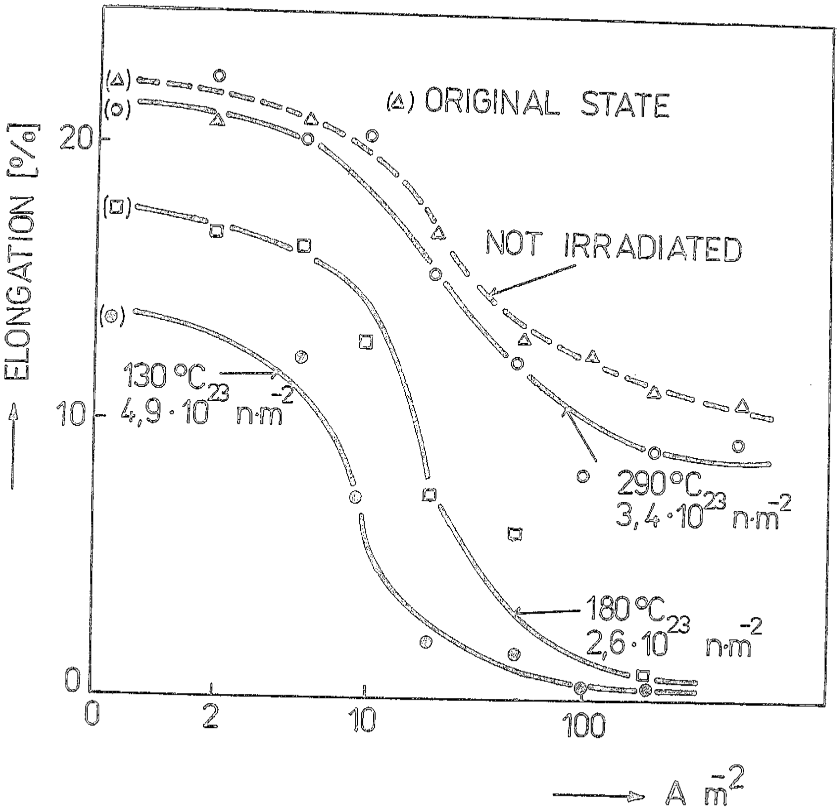

Fig. 2. The whole ductility of irradiated and non-irradiated CrMoV steel in dependance on current density.