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Uses of zirconium alloys in fusion applications

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Abstract

Calculations are presented showing the generation of hydrogen and helium gas and cascade damage (expressed in displacements per atom) for zirconium alloys. After a typical first wall exposure of five years, the hydrogen content of a Zr-alloy increases by a factor of ~1.5 over that initially present, suggesting that hydrogen embrittlement is unlikely to be a life-limiting factor. Structural analyses of a typical first wall using zirconium alloy indicate that Zr-alloys may have satisfactory properties, and should be reconsidered for use in fusion applications.

1. Introduction

Zirconium alloys have been used in a number of water-cooled fission reactor types since the late 1950s due to their excellent aqueous corrosion resistance, low thermal neutron absorption cross-section and good mechanical properties. By contrast, the fusion community has apparently judged Zr-alloys to be unsuitable for use as either structural support or for cooling channel material in future fusion power plants [1]. Part of this disregard stems from the notion that Zr-alloys may suffer excessive hydrogen embrittlement arising from the several hydrogenic sources present in the fusion environment. Of concern also is the possible high level degradation of mechanical properties, as a result of the combined helium production and displacement damage in the hard neutron spectrum typical of a fusion first wall. In this paper, evidence is presented suggesting that neither hydrogen embrittlement nor radiation damage effects are serious impediments to the deployment of Zr-alloys in fusion applications.

2. Hydrogen content

Zirconium and Zr-alloys have an extremely high affinity for hydrogen, being able to tolerate up to 0.7% at.% (~64 wppm) in solid solution at 300°C [2]. This solubility decreases rapidly with decreasing temperature, falling to about 1 appm (10 wppb) at 20°C. Hydrogen in excess of the solubility limit is precipitated as plates of zirconium hydride, which may adversely affect several mechanical properties in some crystal orientations. However, some mitigation may be achieved through careful thermomechanical processing, which aligns the $ZrH_{1.5}$ phase in an optimally textured dispersion, retaining good properties. The main sources of hydrogenic species in a fusion power plant are likely to be:

1. hydrogen liberated from aqueous corrosion processes;
2. deuterium and tritium implanted from the plasma;
3. tritium permeation from the breeder;
4. hydrogen, deuterium and tritium generated by transmutation;
5. hydrogen initially present in the fabricated material.

The relative importance of these sources is briefly discussed. Corrosion in Zr-alloys in fission plants is now well understood. The modest levels of hydrogen pickup that accompany corrosion, and after long service time result in embrittlement, are unlikely to pose significant problems. This was a problem in the early power plants that used Zircaloy-2 (nickel was responsible for the uptake), but the problem was addressed by special alloy formulations (nickel replaced by iron) which have been shown to perform extremely favourably, with the Zr-4 alloy showing greatly reduced hydrogen uptake relative to earlier Zr-alloys [3]. For the operating conditions typical of a water-cooled fusion power plant, hydrogen embrittlement due to corrosion processes is therefore

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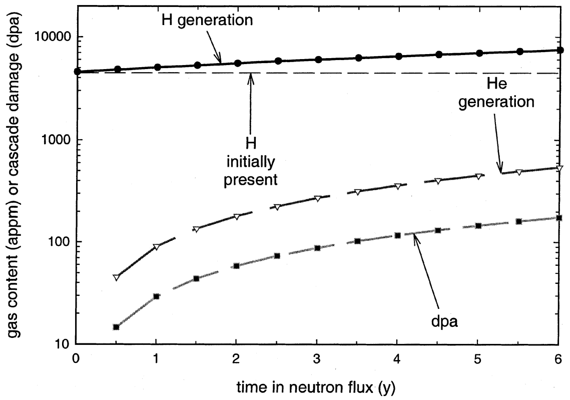

Fig. 1. Evolution of hydrogen and helium gas production, and cascade damage for a Zircaloy-2 first wall.

probably unimportant. Regarding ion implantation from the plasma, it is assumed that a bare Zr-alloy first wall will never be employed in any foreseeable power plant. The D and T ions will be intercepted by an armour material, and thus protect the Zr-alloy from hydrogen pickup. For reasons of good tritium economy, permeation barriers will be employed in many blanket structural components, thus minimising loss, or conversely, pickup of tritium from the breeder. It is therefore assumed that these three sources will have negligible contribution to the hydrogen population in the alloy.

For the remaining sources, assuming a Zircaloy-2 composition containing 4500 appm (50 wppm) of hydrogen as a result of the fabrication process, the additional generation of all isotopes of hydrogen through exposure to neutrons in a first wall location, is determined. A Monte Carlo model was used to determine the first wall neutron flux spectrum; and the EASY-99 [4] code system, the build-up of hydrogenic species as a function of time under flux, see Fig. 1. After five years the hydrogen concentration rises by ~50%, to 7000 appm (~77 wppm). Therefore, it is reasonable to assume that if the material hydrogen content was considered fit for purpose before irradiation, then the relatively small increase during irradiation should not seriously compromise the mechanical properties further. It is concluded that hydrogen embrittlement of Zr-alloys need not be a life-limiting phenomenon.

3. He-dpa considerations

In general, there are two parameters that are important and control the behaviour of material under irradiation conditions: the displacement damage rate and the production rate of transmutant helium. The ratio of helium production rate to damage rate plays a significant role in the microstructural behaviour of the materials under irradiation. In a typical fast fission neutron spectrum, this ratio is around 0.35 appm/dpa, but for most materials in a fusion spectra is expected to be in the range 1–20 appm/dpa across the first wall and blanket of the machine, with the highest values observed in the first wall. This has a direct impact on the material ductility.

The results for hydrogen, helium production, and damage, are shown in Fig. 1. The damage and helium production response functions are used to produce the damage rate G_{dpa}, in dpa per second (dpa/s) and helium production rate G_{He}, in atom parts per million per second (appm/s), for the material. The calculated damage rate for wall load of 2.4 MW/m^2 is 9.28×10^{-7} dpa/s, and the helium production rate is 2.86×10^{-6} appm/s, which results in a (He/dpa) = 3.1. The first wall (He/dpa) ratio for steels is somewhat higher at ~16, suggesting that zircaloys in a usion neutron spectrum will have a ductility limit better than steels, by a factor of at least 2, see arguments in [5] and references within.

4. Irradiation creep

Irradiation creep has been extensively studied for various types of materials used in fission power plants, e.g., stainless steels [6–11] and zirconium based alloys [3,12]. The majority of data gathered are not fusion specific although some attempts have been made to simulate the fusion environment [13–16] and then draw conclusions about material behaviour. In these studies, theoretical analysis, semi-empirical analysis, and data suggest that the material ductility in a fusion neutron spectrum would be a function of the following variables:

$$\varepsilon_{\mathrm{f}} = \text{ function } \{D, G_{\mathrm{dpa}}, T, \sigma, (\mathrm{He/dpa})\}. \tag{1}$$

An expression for irradiation creep strain ε_{c} for Zircaloy-2 based on fission data [12] is

$$\varepsilon_{\mathrm{c}} = 3.672 \times 10^{-10} t \mathrm{e}^{-1683.84/T} \phi^{0.85} \sinh(0.01668\sigma), \tag{2}$$

with T the temperature in K, t the time in seconds, ϕ the neutron flux in n/cm^2 s (>1 MeV) and σ the stress in MPa. This equation can also be written in terms of the dose and damage rate, with $t = D/G_{\mathrm{dpa}}$.

The accurate prediction of lifetime and prevention of component failure is important, especially for first wall structures, which are subjected to a variety of loading conditions. Design curves of stress versus time can be constructed by combining irradiation and thermal creep, provided a limitation is set for creep strain. Assuming fracture/design strains of 1% and 0.5%, the stress-time design curves can be obtained, and for the zircaloy are shown in Fig 2, for $T = 300$°C (fission design codes recommend averaged across a section strain limit values of 1%). The results indicate that for fusion conditions,

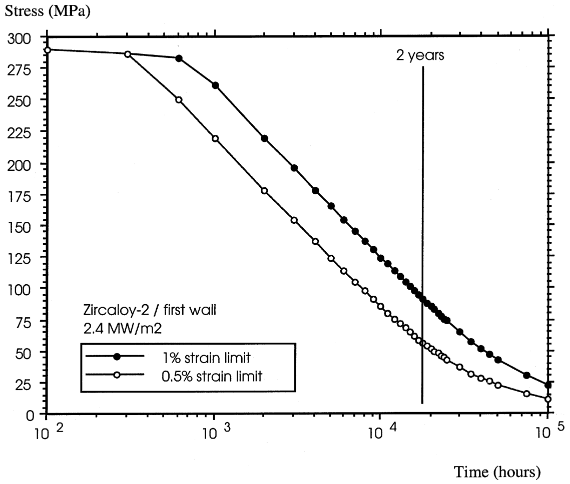

Fig. 2. The proposed zircaloy-2 design curve including irradiation creep effects.

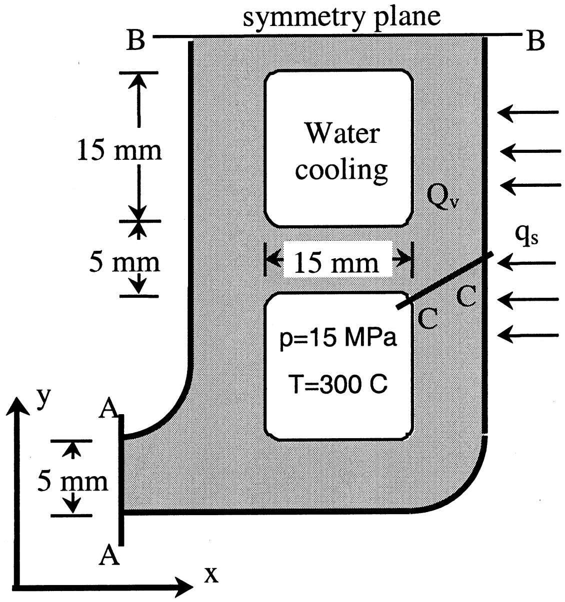

Fig. 3. Schematic of the water cooled first wall.

the irradiation based design curves are quite restrictive as compared to the thermal creep to failure curves.

5. Thermal-structural analysis

Thermal and structural analysis is performed on a typical first wall section for power plant relevant conditions, to determine the behaviour when irradiation creep is included in the finite element structural analysis. The time dependent non-linear analysis using temperature dependent material properties was performed with the finite element analysis code COSMOS/M [17].

A schematic of the geometry, with loading conditions and dimensions used in the thermal and structural calculations, is shown in Fig. 3. Structural analysis assumes plane strain conditions with no bending in the out of plane direction. Section AA movement is restricted, and section BB is a symmetry plane. The zero thermal stress temperature is taken to be the coolant temperature and is $T_{ref} = 300°C$. The heating loads at full power conditions are: volumetric heat of 16.9 MW/m^3 and surface heat flux of 0.4 MW/m^2. The calculated damage rate of 29.3 dpa/year, is comparable to the damage rates of the austenitic and martensitic steels at the same wall load.

The results with Zircaloy-2 are compared with results for the 316 stainless and martensitic steels, for the same geometry and loading conditions. The thermal analysis shows the maximum temperature occurring in the area between the cooling channels and facing the plasma (at the edge of section CC). The temperatures across CC, attained by the Zircaloy-2, vary between 310°C and 493°C. The corresponding temperature variation for the 316SS is 319–461°C and for the martensitic steel is 317°C–412°C. Results for the averaged stress and strain variation with time, across section CC, are shown in Fig. 4, for 40,000 h of operation. The Zircaloy-2 first wall shows complete relaxation in ~10 000 h, from a stress of 85 MPa to a stress of 35 MPa, with simultaneous strain enhancement due to creep, see Fig 4.

From the structural point of view, the major concern of creep induced stress relaxation is operation of the structural material close to the ductility limit, which is reduced due to embrittlement (the helium production and displacement damage effects). The problem with the structure at this 'relaxed' stress state is its reduced ability

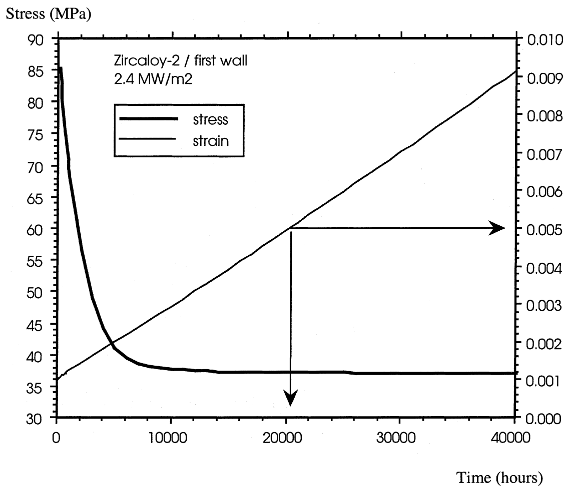

Fig. 4. The averaged cross section CC stress and strain variation with time.

to accommodate any further load fluctuations. For example, a power transient (under power), will result in the structure wanting to resume its original shape, thus introducing a stress reversal which can lead to structural failure.

Accumulation of creep damage in the structure is fast at the beginning of operation but eventually reaches a steady value. At the beginning of power plant operation the stress is high and the time to failure t_r is low, resulting in large values of the damage ratio ($\delta t/t_r$). As time goes on, stress relaxes, and t_r progressively increases, resulting in decreasing values of the damage ratio and minimal contributions to the sum (or accumulation of damage).

Assuming there are no power transients, the calculated in-service time for a zircaloy-2 first wall (see Fig. 4), results in ~21 000 h (2.4 years, 70.3 dpa) for a design strain of 0.5% and ~44 000 h (5.02 years, 147 dpa) for 1%. If in-service time is based only on irradiation creep strain, a comparison of Zircaloy-2 with steel [5], assuming all three materials have the same design strain limit, gives:

In-service time (0.5% *strain*)

Material	Years	dpa
316SS	3.17	71
Martensitic steel	5.28	107
Zircaloy-2	4.0	117

Many more factors should be considered in the calculation of in-service time, but the present results for Zircaloy-2 imply that structural performance in fusion conditions is acceptable and will not exclude these alloys from being chosen as structural material in the design of fusion power plants.

6. Conclusions

A brief examination of the likely sources of hydrogen species in a fusion power plant has indicated that hydrogen embrittlement in Zr-alloys is unlikely to be a life-limiting problem. Most of the hydrogen in a Zr-alloy component will have been present since the fabrication process.

The structural analysis of a typical first wall section, using Zircaloy-2 as the material, shows that behaviour, performance and in-service time is comparable or better to that of steels in the fusion neutron environment, and it will not be prohibitive if these alloys are chosen for future use in fusion power plant design.

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