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Overview of recent European materials R&D activities related to ITER

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Abstract

An overview is given of the wide range of activities contained within recent R&D being performed within Europe on In-vessel materials for structural, heat sink and plasma facing purposes for ITER. The effect of creep-fatigue interaction on the fatigue life of CuCrZr and the effects of irradiation on over-aged CuCrZr are given. In addition the lifetime of ITER components has been further investigated by the performance of in situ experiments with both neutrons and high-energy protons. The effect of hydrogen on the crack initiation fracture toughness of Ti is reported at a range of irradiation temperatures. The irradiation induced stress relaxation of Alloy 718 used for bolting applications is being studied and initial results will be described. Work in the area of plasma facing materials and re-welding issues will also be presented.

1. Introduction

The materials chosen for ITER have been selected, as far as possible, from conventional, well-characterised materials. Given this fact, however, a significant amount of materials R&D activities were needed for ITER. Some of the main reasons for this have been: the relatively high neutron fluence at the First Wall, the evolving nature of the ITER design, the selection of materials from competing alternatives proposed by the different parties and the development of new manufacturing routes for In-vessel components. Changes in the design of some of the components aimed at reducing the cost of construction has also had an impact on some of the materials used. As the construction of ITER imminent it is an appropriate time to review the materials data available to support the design of components for ITER. This review is being performed in the ITER framework [1].

However, R&D results are still becoming available and this paper will describe the recent R&D performed within Europe for ITER in the vessel/In vessel field and indicate where work is still under way. The order of the work described will start at the outside of the machine and work inwards towards the plasma from the vacuum vessel to the plasma facing materials. The paper will

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concentrate on the effect of irradiation on the material properties.

2. Reweldability of 316L(N)-IG

The issue of re-weldability of thicker section material after irradiation has been addressed with a study of the re-weldability of 316L(N)-IG and powder HIP stainless steel of a similar composition (apart from a higher oxygen content). The materials investigated were 5 and 10 mm thick with He contents after irradiation in the range of 2–7 appm; the heat input was in the range 700–1000 J/mm per weld pass. All welds showed HAZ cracking in the metallographic cross sections, consistent with previous results but the results also showed that HAZ cracking appears to develop and increase in severity during filling of the weld. The powder HIP material showed greater sensitivity to re-welding compared with the plate material for equivalent He contents. Fig. 1 shows the metallographic section of a multi-pass weld for the powder hipped material. The figure shows increasing severity of the cracking as the weld is filled. Further details of this work can be found in [2]. The reason for the increased sensitivity of the powder hipped material is not known.

Modelling of these phenomena has been performed and a good understanding of the reasons for the increased bubble formation developed. These appear to be the repeated heating to high temperatures, thermal stresses and an increase in constraint during multi-pass welding. These results indicate that the He content of the base material is only one factor among many that determine if a material is re-weldable or not after irradiation.

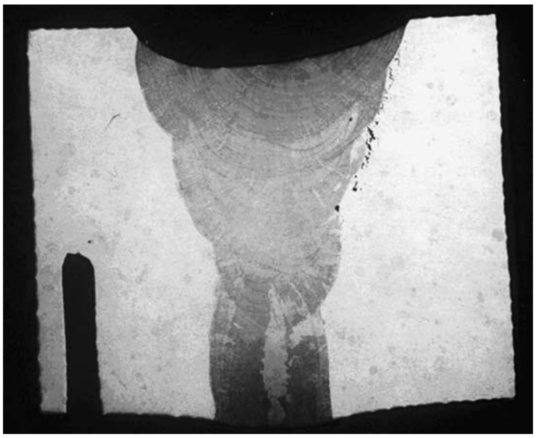

Fig. 1. Metallograph of a re-welded powder hipped stainless steel sample (irradiated material on the right, un-irradiated on the left) showing significant cracking in the HAZ of the irradiated material.

3. Titanium alloys

A titanium alloy has been selected for the flexible cartridges, used to support the first wall blanket modules, due to its excellent elastic and strength properties. Two titanium alloys have been investigated in Europe, the $(\alpha + \beta)$ alloy Ti6Al4V and the α alloy Ti5Al2.5Sn. Detailed characterisation of the effect of irradiation on these alloys has previously been published [3]. Recent work has concentrated on the effect of irradiation on the crack initiation fracture toughness of hydrogen loaded Ti alloys. This effect has been studied with two different hydrogen contents, one below and one above the solubility limit of hydrogen in these alloys. The results of the latest tests, shown in Fig. 2, indicate that the presence of hydrogen further decreases the crack initiation fracture toughness values J_Q of the alloy when tested at room temperature. No dependence of J_Q on the hydrogen content is observed when the samples were tested at 350 °C, possibly as a result of the higher solubility of hydrogen at higher temperatures. Despite the low J_Q values reported, no evidence of brittle fracture has been observed in the materials tested. Further work is underway to determine the J_Q values for hydrogen loaded samples irradiated at 150 °C, the present expected operational temperature of the Ti flexible cartridges in ITER, and thereby confirm its applicability.

4. Irradiation induced stress relaxation

The preferred solution in ITER for the mechanical attachment of the blanket modules to the vacuum vessel uses pre-loaded bolts of high strength Alloy 718, a nickel based alloy. These bolts will be exposed to an irradiation field and hence subject to irradiation induced stress

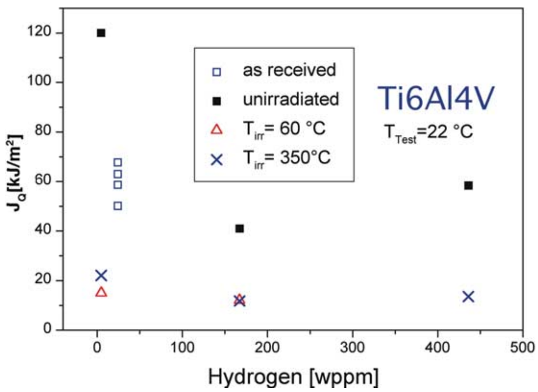

Fig. 2. The effect of hydrogen on the crack initiation fracture toughness of the titanium alloy Ti6Al4V, neutron irradiated to 0.1 dpa.