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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.

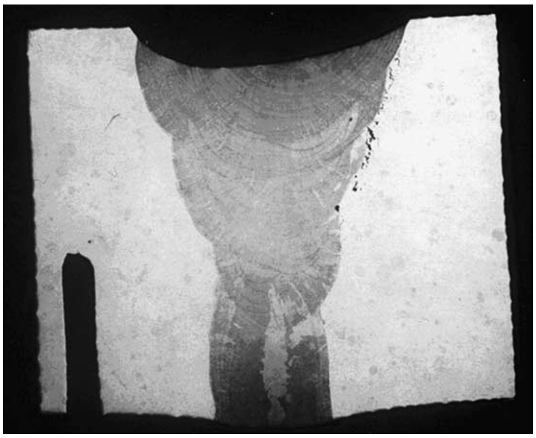

Fig. 1. Metallograph of a re-welded powder hipped stainless steel sample (irradiated material on the right, un-irradiated on the left) showing significant cracking in the HAZ of the irradiated material.

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

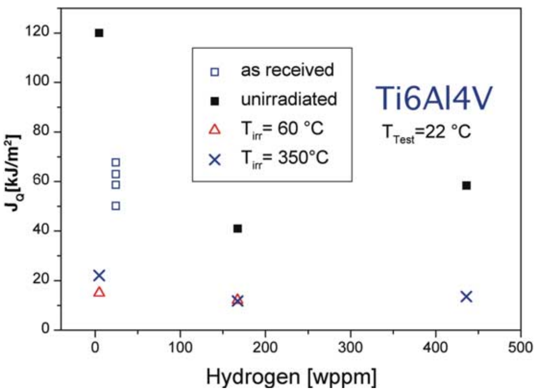

Fig. 2. The effect of hydrogen on the crack initiation fracture toughness of the titanium alloy Ti6Al4V, neutron irradiated to 0.1 dpa.

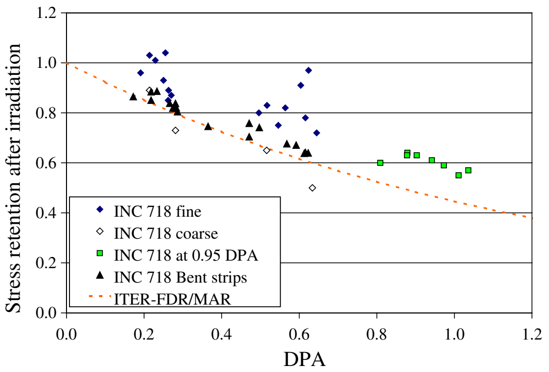

Fig. 3. Stress relaxation values for Alloy 718 against dose, irradiations performed at 300 °C.

relaxation. There are a number of examples where materials are stressed in an irradiation environment, for example in fuel pin assemblies. Experimental work to quantify these effects has been performed using a similar geometry, such as a pressurised tube. This type of measurement has also been made on the Alloy 718 material proposed for use in ITER [4]. To replicate the bolt geometry used in ITER, irradiation induced stress relaxation experiments have also been performed using bolt type specimens. The initial experiments showed a large and unexplainable scatter in the stress relaxation data for 0.22 and 0.73 dpa irradiations in contrast with low scatter for 0.95 dpa data. Further tests have been performed, including some bent strip specimens and these results are shown in Fig. 3. The results show very consistent values for the bent strip samples but again a large scatter for the bolt type specimens. For ITER sufficient margin should be taken to compensate for the irregularities shown here in the bolt specimens. Further details are given in [5].

5. Copper alloys

For the ITER divertor CuCrZr has been selected by ITER as the material of choice for a number of reasons [6], including its relatively good fracture toughness. For the first wall panels the material selection is still open from an ITER perspective. Europe has, however, concentrated its R&D efforts in recent times investigating the possibility of using CuCrZr as the heat sink material for these first wall panels. This effort has included investigating the effect of the different manufacturing routes on the properties of the CuCrZr and investigating the effect of irradiation on the material properties in the manufactured state. In addition investigations have started into the effect of in situ irradiation and mechanical loading on the behaviour of copper, both pure copper and CuCrZr, and investigating the effect of hold time on fatigue lifetime.

5.1. In situ effects

To date most experimental work performed to measure the effect of irradiation on materials has concentrated on measuring these properties on specimens after irradiation. This is not exactly the same as what is experienced in ITER where components are being irradiated during mechanical loading. In situ effects have been investigated before on the fatigue life of 316L steel [7] and no significant effect observed. A number of experiments have been carried to see if this was also the case for the copper alloys in ITER. Two experiments can be reported to date; results from the first set of experiments are presented in Fig. 4, showing the effect of in situ irradiation and mechanical loading on the tensile behaviour of pure copper. As can be seen, the in situ tensile curve lies between the un-irradiated result and the post-irradiation result. The in situ test shows no yield drop effect as is seen in the post-irradiation result, however a significant decrease in total elongation is observed. Experimental details of this work have been described in [8] and detailed experimental results are to be published [9]. The second experiment was performed using a proton beam to produce the irradiation damage. The effect on the fatigue life of in situ irradiation on CuCrZr was measured. Three tests were performed; un-irradiated fatigue test, post-irradiation fatigue test and a fatigue test with concurrent irradiation and cyclic deformation. The results are shown in Fig. 5. More details of the experiment are given in [10]. These experiments show that after the first hundred cycles the behaviour of the three tests followed a similar trajectory. As judged by the criteria N_a (the main crack has

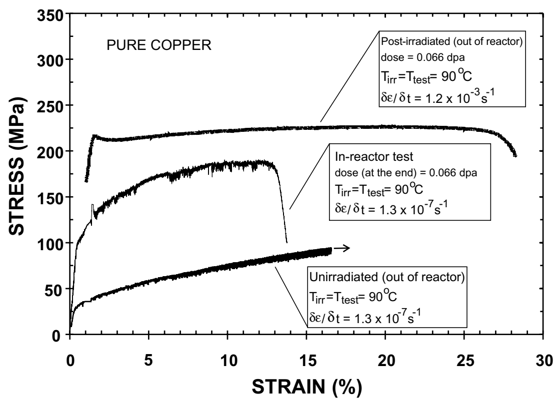

Fig. 4. Stress against strain plot comparing an in situ test with an un-irradiated and post-irradiated test.

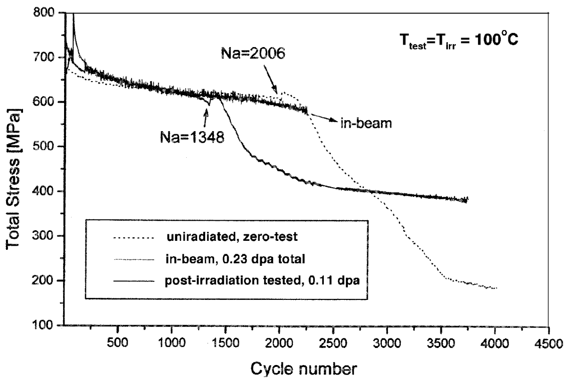

Fig. 5. Applied stress range against number of cycles for an in beam fatigue test compared with un-irradiated and post-irradiated specimens.

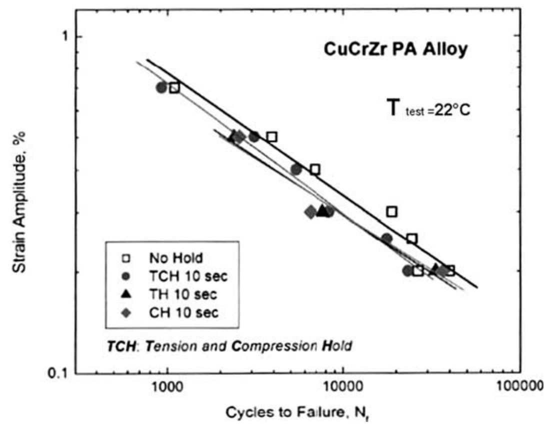

Fig. 6. Strain amplitude against Cycles to failure for CuCrZr alloys with different hold combinations.

propagated through the specimen wall), the concurrently irradiated and tested specimen lasted longer than the post-irradiation tested specimen.

Given the limited statistics available for in situ testing it is difficult as yet to draw a firm conclusion regarding the dynamic effect of irradiation and mechanical loading on materials performance and lifetime, however, it does appear that assessment of the irradiation behaviour of materials in ITER using post-irradiation testing gives conservative values. However, a fuller analysis is needed of component behaviour taking these results into account, which has yet to be done. Further in situ work is planned including fatigue and fatigue with hold time tests.

5.2. Effect of irradiation on over-aged CuCrZr

The manufacturing techniques that are used to manufacture first wall panels with CuCrZr [11] lead to an over-ageing of the material. This is, in particular, due to the Be bonding step, although attempts are being made to reduce this over-ageing. The effect of irradiation on over-aged material has been investigated by looking at samples, which have been heated for different times and temperatures replicating the effect of over-ageing prior to irradiation. The effect of this over-ageing treatment, 600 °C for 1 h, on the tensile properties of the material after irradiation is to reduce the yield strength of the material by 10% in comparison to the prime-aged material, a similar effect to the effect of over-ageing on the un-irradiated tensile properties.

5.3. Fatigue with hold time

The combination of creep hold times with fatigue has been shown to have a significant effect on the fatigue lifetime of copper alloys [12]. The effect appears to be influenced by whether the hold is in tension or compression. This effect has been further studied and some of the results of this study are shown in Fig. 6. At the test temperature investigated there appears to be only a limited impact of the hold time and no effect of whether the hold is in compression or tension. Further results on this series of tests are given in [13].

6. Plasma facing materials

Significant work is continuing in Europe to investigate the effect of irradiation on the physical and mechanical properties of plasma facing materials. An example of this work is shown in Fig. 7, which shows the reduction of thermal conductivity of tungsten with increasing dose after irradiation at 200 °C. Further de-

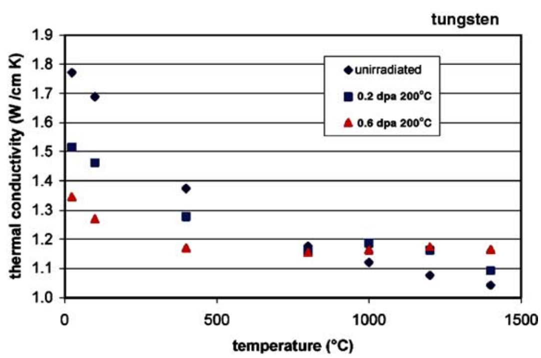

Fig. 7. The thermal conductivity of tungsten as a function of temperature after irradiation to different doses at 200 °C.

tails of the recent work of plasma facing materials is given in [14].

7. Conclusions

A number of experimental campaigns instigated several years ago are now reaching their conclusion and the experimental results are becoming available. These results, together with results from other ITER parties, confirm that the selected materials and design solutions achieve the required performance at ITER specified conditions. As ITER moves towards construction these experimental results will be consolidated in a materials database and assessed. The assessed data will then be used to support the component design and acceptance process. Detailed assessment, analysis and testing of mock-ups will also be required.

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