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Assessment of creep in reactor-irradiated CuCrZr alloy intended for the ITER first wall panels

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ABSTRACT

CuCrZr alloy is candidate heat sink material for the ITER blanket, first wall, and divertor. During ITER operation it will be exposed to a combination of elevated temperatures, heat flux, and intense fast neutron radiation. This environment will challenge the performance of components and joints based on CuCrZr. To address this issue, mechanical tests were performed with irradiated and reference specimens of CuCrZr and its joints with 316 L(N)-IG (ITER Grade) stainless steel made by hot isostatic pressing. The reactor exposure up to ∼0.7 dpa was performed in the BR2 reactor at SCK•CEN, in water at a temperature of 257 °C A special design was used to allow irradiation of specimens axially pre-stressed at different strain levels. The post-irradiation examination included: (i) tensile test, (ii) measurements of plastic deformation of samples axially loaded during irradiation (in situ creep test); (iii) thermal creep tests on irradiated samples. The fracture surfaces were examined in a hot cell using a Scanning Electron Microscope (SEM). The results were compared with data obtained from mechanical tests and SEM/EDX fracture surface analysis on non-irradiated reference samples. The level of a possible creep under irradiation is below the experimental uncertainty.

1. Introduction

One of the major challenges for the establishment of fusion energy is the development of appropriate in-vessel materials to be used in ITER and later on in DEMO. These materials shall resist the neutron, heat and electro-magnetic loads at elevated temperatures for the whole installation lifetime. CuCrZr alloy is currently candidate heat sink material for the ITER blanket, first wall, and divertor [1,2]. A layer of this material will be placed between the plasma facing armor and the main structure. In addition to high thermal conductivity, this layer and its joints to adjacent material must also be mechanically strong to sustain thermal and mechanical stresses. CuCrZr brazed onto water cooled steel pipes is used as a heat sink material in Wendelstein 7-X [3].

During ITER operation the first wall will be subjected to significant neutron loads up to ∼0.3 MWa/m^2 [4], corresponding to a damage of ∼0.7 dpa in CuCrZr alloy. Such neutron fluences affect both thermal and mechanical properties of CuCrZr and CuCrZr joints with both armor and the main structure. Degradation of heat conduction and mechanical strength can be a significant problem for the ITER operation resulting in shorted maintenance periods for first wall components. It is therefore important to check under relevant conditions and with representatively manufactured material mock-ups the scale of a possible impact on the performance of first wall material and joints.

Information on neutron radiation influence on mechanical properties of CuCrZr alloy at ITER relevant conditions is rather limited. In the early 2000’s irradiation tests were performed with the former heat sink candidate, the dispersion-strengthened copper (DS-Cu) [5]. Unfortunately, those results are not directly representative for our mechanical tests. However, a higher than normal erosion after neutron irradiation observed in thermal shock experiments could be considered as an indirect indication of a possible decrease of ductility after irradiation. On the other hand, high heat flux testing of mock-ups neutron irradiated up to 0.35 dpa at 350 and 700 °C (to ∼0.6 dpa at 200 °C) showed that brazed Be/Cu flat tile mock-ups fulfilled the operational requirements for first wall components.

The effect of neutron irradiation on the performance of high heat flux components for ITER was studied in [6]. Mock-ups made of carbon-reinforced materials and W-1%La_2O_3 tiles joined to CuCrZr tubes by hot

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isostatic pressing (HIP) were irradiated at 200 °C up to 0.6 dpa in tungsten and 1.0 dpa in carbon. The thermal conductivity of the mock-ups was strongly reduced after neutron irradiation, especially for carbon. However, no significant effect on the HIP joints with the CuCrZr tubes was observed. The monoblocks did not show any degradation of the fatigue performances after irradiation and all survived 1000 cycles at 18 MW/m^2.

Influence of neutron irradiation with doses of 0.15 and 1.5 dpa on tensile and fracture toughness properties of precipitation-hardened CuCrZr alloy was reported in [7]. The mechanical testing was performed at room temperature. Significant irradiation hardening and plastic instability at yield occurred with a saturation dose of ~0.1 dpa.

In the framework of beryllium qualification for the use as the ITER first wall plasma facing material, mock-ups using CuCrZr/Cu as a heat sink were neutron irradiated up to 0.75 dpa and subsequently high heat flux tested in the JUDITH-1 facility [8]. It was found that neutron irradiation influences the thermal performance by possibly decreasing thermal conductivity of the plasma facing and heat sink materials or by a degradation at the interface between beryllium and Cu.

The objective of the present work is to assess the effect of neutron radiation on the mechanical performance for the current candidate material CuCrZr alloy and its joints. Two boundary conditions were taken into account. First, the material characterization according to industrial codes and standards is usually based on standard size test specimens. However, available material irradiation facilities are limited in irradiation volume with a uniform dose deposition with the consequence that a reduction of the dimensions of the test specimens is required. Secondly, mechanical or thermal testing in situ is impossible or at least very difficult and expensive. Therefore, the following approach, which is a rather common practice, was applied: small-scale mock-ups were irradiated in a fission reactor and then post-irradiation examination and testing were performed in dedicated test facilities. Considering the relatively small cross section of the test specimens (∅ 2.3 mm and 3 mm), the compliance of the material quality with the grain size limits plays a significant role in obtaining reliable results on the mechanical performance. The irradiation was performed in the BR2 reactor of the SCK•CEN at 257 °C up to a fast($E > 0.1$ MeV)/thermal neutron fluence of 9.5E20/6.6E20 n/cm^2, equivalent to 0.7 dpa in Cu. Corresponding reference tests on non-irradiated materials were performed under comparable conditions.

The work focuses on mechanical testing of irradiated and reference specimens made of CuCrZr base material and HIP joints of CuCrZr/CuCrZr and CuCrZr/SS316 L(N)-IG. The following tasks were performed:

- assessment of in-pile creep relaxation at 257 °C.
- thermal creep tests at 250 °C on irradiated samples.
- tensile test on an irradiated sample at RT;
- accompanying reference tests after heating equivalent to the irradiation conditions.

The temperature requirement of 250 °C comes from the lengthy-bake out cycle of the blanket components, which will be performed at 240 ± 10 °C [9].

The results of this work should be considered as a conservative estimation of the level of a possible creep under irradiation. Firstly, the temperature of the CuCrZr components during ITER operation will be below the test temperature of 250 °C. Secondly, the post-irradiation tests were performed on materials and joints having accumulated damage corresponding to the neutron irradiation fluence corresponding to the ITER operation life-time.

2. Experimental

2.1. Irradiation rig

The reactor irradiation was performed using the Multi-purpose Irradiation System for Testing of Reactors Alloys (MISTRAL) irradiation rig at SCK•CEN. This rig allows high neutron fluence exposure of metallic samples in a demineralized water environment, at elevated temperatures controlled with accuracy better than ± 5 °C. The high neutron dose is achieved by inserting the in-pile section (IPS) into a BR2 driver fuel element. The use of a non-standard fuel element type allows for a rig with an external diameter of 34 mm. The high temperature under irradiation is obtained by combining electrical and radiation heating. The nuclear radiation produces a significant amount of heat – up to 18 W/g. However, this heat may not be sufficient to maintain a temperature of 250 °C during irradiation. In some cases, it is also important to start irradiation with the samples preheated up to a desired temperature. In this situation the heating element operates also before the radiation. In addition, the heater provides a compensation of the radiation heat generation non-uniformity in the vertical direction. Without the heater samples located at the extremities of the basket would be irradiated at lower temperatures. The combination of electric and nuclear heating allows for a temperature field being nearly the same over the full length of the irradiation basket.

During irradiation the rig is filled with water, which provides temperature stabilization at the point when water in the rig starts to boil. The boiling temperature is controlled by adjusting the pressure in the rig i.e. the steam pressure controls the temperature.

Test samples are attached to the irradiation basket using springs, specially designed to simplify loading and unloading operations performed in a hot cell. The basket is instrumented with thermocouples used to control and measure the temperature at different levels in the irradiation rig. Activation dosimeters are inserted in specific positions, providing neutron fluence mapping based on the activation analysis.

2.2. Sub-size specimen design

According to the ASTM standard [10] the length of a standard tensile specimen is ~145 mm. Due to the limited irradiation volume such specimens could not be used in the present experiment and we had to use smaller, sub-size specimens. The specimens of ∅5 × 27 mm^2 were designed and fabricated, so that they fit into the irradiation basket, Fig. 1. Dimensions of the sub-size specimens are in agreement with small-scale test technology (SSTT) applied for ITER EUROFER material testing. For the *in situ* creep relaxation test the specimens were uni-axial pre-stressed using a sleeve locked by two screw nuts. PIE showed that the nuts remained tightly screwed and fixed after the end of the irradiation. Combined with the absence of detectable elongation of the specimens, this allows us to conclude that the stress state did not change during the whole irradiation cycle.

High temperature water under irradiation is a highly oxidizing environment. Direct contact of Cu-containing samples with IPS water would result in corrosion and coolant contamination. Therefore, before pre-stressing all the samples were coated with a ~15 µm thick Ni layer.

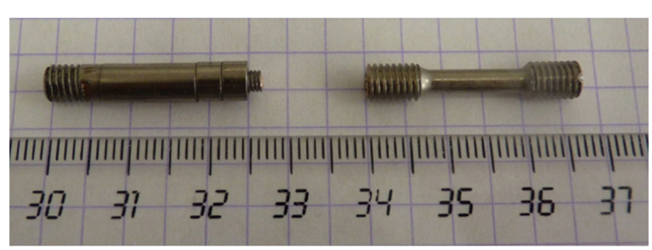

Fig. 1. Uni-axial pre-stressed and locked creep relaxation (left) and thermal creep test (right) Ni-coated specimens before irradiation.