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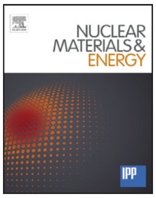

Effect of dual ion beam irradiation (helium and deuterium) on tungsten–tantalum alloys under fusion relevant conditions

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ABSTRACT

The selection of tungsten (W) as a divertor material in ITER is based on its high melting point, low erosion, and strong mechanical properties. However, continued investigation has shown W to undergo severe morphology changes in fusion-like conditions. Recent literature suggests alloying W with other ductile refractory metals, *viz.* tantalum (Ta) may resolve some of these issues. These results provide further motivation for investigating W–Ta alloys as a plasma-facing component (PFC) for ITER and future DEMO reactors. Specifically, how these alloy materials respond to simultaneous He^+ and D^+ ion irradiation, and what is the effect on the surface morphology when exposed to fusion relevant conditions. In the present study, the surface morphology changes are investigated in several W–Ta targets (pure W, W-1%Ta, W-3%Ta, and W-5% Ta) due to simultaneous He^+ and D^+ ion irradiations. This comprehensive work allows for deeper understanding of the synergistic effects induced by dual ion irradiation on W and W–Ta alloy surface morphology. Pure W and W–Ta alloys were irradiated simultaneously by 100 eV He^+ and/or D^+ ions at various mixture ratios (100% He^+, 60% D^+ + 40% He^+, 90% D^+ + 10% He^+ ions and 100% D^+ ions), having a total constant He fluence of 6×10^{24} ion m^{-2}, and at a target temperature of 1223 K. This work shows that slight changes in materials composition and He/D content have significant impact on surface morphology evolution and performance. While both the pure W and W–Ta alloys exhibit very damaged surfaces under the He^+ only irradiations, there is a clear suppression of the surface morphology evolution as the ratio of D^+/He^+ ions is increased.

1. Introduction

The magnetic confinement fusion project ITER is leading the way for fusion as future commercial energy source. With the decision to move to a full-tungsten (W) divertor in ITER, the study of W as plasma facing components (PFCs) under fusion environments has become a key issue for the fusion community [1]. W has mainly been selected for its desirable thermo-mechanical properties such as high melting temperature [2], good thermal conductivity [3], and low erosion under ion bombardment [2]. Despite these excellent advantages, recent studies have shown W to undergo severe morphology evolution in response to both low-energy helium (He) and deuterium (D) ion irradiations. W surfaces exhibit blistering after low-energy deuterium irradiation at surface temperatures under 700 K [4–6], and blisters [7,8], pores [9,10], and eventually ‘fuzz’ [11–14] after low energy helium ion irradiation at surface temperatures between 800 and 2000 K. This surface evolution has been shown to degrade key PFC attributes such as thermal conductivity [15,16] and erosion rate [13,17,18], and these adverse effects have driven research into innovative alternative PFC materials which are resistant to extreme surface modification under relevant fusion conditions.

One area that has shown some promising PFC enhancements is the formation of W alloys. The alloying of W with certain materials like Rhenium (*Re*) has been shown to improve ductility [19,20]. Recently, it has been suggested that although the alloying of W with tantalum (Ta) does not provide the same ductility enhancement as that of the *Re* case; it prevented the crack propagation under certain grain orientations [21]. This result is supported by further research on W–Ta alloy’s response to thermal shock via transient heat loading which has shown a significant improvement when compared to pure W materials [21,22]. Along with mechanical enhancements, W–Ta alloys have exhibited a significant reduction in retention of hydrogen (H) isotopes [23–25], as well as a significant resistance to morphology evolution [26].

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There has been significant work done towards understanding how W responds to low energy He^+ ion irradiation. Previous work showed also the effect of pre-irradiation or sequenced He^+ and D^+ irradiation of materials on their properties such as deuterium retention [27]. However, in a real fusion environment PFC materials will be subjected to dual He^+ and D^+ ion bombardment simultaneously. There have been a few studies looking at the effect of mixed plasmas, but their focus has been mainly on D retention. Studies were performed on W exposed to "D + He" mixture plasma with various He concentrations to investigate the impact on D retention [28,29]. The results suggest that He competes with D for the near surface trapping sites creating deep networks of bubbles in nanometer scale range. These bubbles then act as a diffusion path to the surface for implanted D thus significantly reducing the D retention. Similar work has also been done on the effect of mixed species on surface morphology and D retention [30,31]. This work and studies discussed earlier [29,30] showed similar results demonstrating a significant reduction in D retention due to mixed plasma species effects. It was shown also that surface morphology evolution appeared to be slower as a result of the mixed plasma irradiations [31]. However, this effect was attributed to a dilution of the He flux rather than a mixed plasma effect [31].

The goal of the present work is to focus on the understanding of the morphology responses when both pure W and W–Ta alloys are exposed to simultaneous He^+ and D^+ ion irradiation at elevated temperatures. By changing the mixture ratio of the irradiation species (He^+ and D^+ ions) the synergistic effects of dual ion irradiations are investigated. SEM imaging of ion-exposed samples reveal that He induced microstructures are suppressed due to the presence D, and that the magnitude of this suppression is dependent on the D^+/He^+ ion ratio and the Ta concentration. These results suggest that W based PFCs may respond differently to the fusion environment than previously expected when synergistic effects are taken into account.

2. Experimental methods

The experimental work discussed here studied four different W-based materials, 99.95% pure W and three W–Ta alloys with 1, 3, and 5 wt% of Ta. The 2 mm thick sheets of the W–Ta alloys were sintered at 1500 °C and both the W and Ta powder had an average particle size (APS) of less than 10 µm. When referring to these samples going forward the following name convention will be used: W, W-1Ta, W-3Ta, and W-5Ta will denote the pure W and the 1, 3, and 5 wt% of Ta, respectively.

Samples of the W, W-1Ta, W-3Ta, and W-5Ta were cut from the same sheets into 10 mm × 10 mm × 2 mm samples. A total of 16 W and W–Ta samples were mechanically polished to a mirror finish prior to irradiation. He^+ and D^+ ion exposures were conducted at the UHFI-II chamber located in CMUXE lab at Purdue University [32]. Fig. 1 shows a schematic illustration of the experimental setup used during the irradiation experiments. Four combinations of "He^+ and D^+ ion mixtures" have been used for all the W–Ta samples; He^+: D^+:: 100: 0 (hereafter pure He^+ ion), He^+: D^+:: 40:60 (hereafter 60% D^+ ion), He^+: D^+:: 10:90 (hereafter 90% D^+ ion) and He^+: D^+:: 0:100 (hereafter pure D^+ ion). Note, in an ideal case, the He flux would remain constant for all the mixtures and only adjustment to the D^+ flux would be needed to achieve the necessary ratios. However, the upper limit on the achievable flux for D^+ proved to be 1.4×10^{21} ions m^{-2}-s^{-1}. This means that the He^+ flux at the surface needed to be suppressed to 1.5×10^{20} ions m^{-2}-s^{-1} in order to get the 10:90 He^+-D^+ ratio, and the same fluence and flux for He is used in order to isolate the effect of D on the damage process.

Specifically, 100 eV He^+ ion flux of 4.0×10^{20} ions m^{-2}-s^{-1} at 1223 K, for 4.17 h was used for the experiments with pure He^+ and 60% D^+ ion beams. The He^+ flux was reduced to 1.4×10^{20} ions m^{-2}-s^{-1}, for the experiments with 90% D^+ ion beams, and the irradiation time was increased to get the same total He^+ fluence. The experiments with pure 100 eV D^+ ion irradiation used flux of 6.0×10^{20} ions m^{-2}-s^{-1} at 1223 K, for 4.17 h. Table 1 shows the flux and fluences for each mixture case. After ion irradiation experiments, the samples were taken out from the UHV chamber.

Following irradiation, field emission (FE) scanning electron microscopy (SEM) was performed to monitor the He^+ ion-induced surface modifications. Optical reflectivity measurements were performed over spectra of incident light (using a combination of halogen and deuterium light source and a beam diameter of ~1 mm) ranging from 200 to 1100 nm wavelengths. Before the reflectivity measurements began, the spectrometer was calibrated with a reference plate having 100% reflectivity. Note, the observed reflection in our optical reflectivity system is mainly specular. A specular reflection is a reflection of a mirror-like surface (keeping in mind that different surfaces to different wavelengths may or may not be mirror-like). Specular reflection will result when the surface roughness is smaller than the applied wavelength of light (and diffuse reflection will result when the surface roughness is larger than the wavelength). A specular reflectance of 100% would correspond to an ideal mirror; typical specular reflectance is less than the maximum value. For collecting the reflected light, a "*reflection probe*" has been used which can collect light at the same angle as it illuminates, and can be used for either specular or diffuse reflection measurements. The "*reflection-probe*" is made of 6 illumination fibers around a single read fiber (in the center), which results in a 25° full angle field of view. Each illumination fiber project a cone of light from the source and all of them overlap at the sample in the center, exactly where the central read fiber is situated. Thus, in principle the reflectivity for this ideal mirror will be ~100%. During our measurements the "*reflection probe*" was placed at 90° to the sample surface (along the sample surface normal). The distance between sample and "*reflection probe*" was ~1 mm.

3. Results and discussion

3.1. Field emission scanning electron microscopy (FE-SEM) studies

Fig. 2 depicts the FE-SEM images of 4 W–Ta samples exposed to 100 eV He^+ irradiation only, These ion-exposures represent the base case (reference) for He^+ induced damage; the subsequent mixed ion-species exposures will be compared with these samples. As seen in the four FE-SEM images there is a noticeable morphology difference that is dependent on the Ta concentration. These results are in good agreement with the results observed by our group very recently [27], where we have shown that the alloying of W with Ta alters the crystallographic structure of W causing it to have slightly larger lattice parameter spacing. It appears that the extra lattice spacing provides more available room for the He accumulation before surface damage is observed (in other word, effectively delaying the fuzz morphology evolution of the surface) [27]. This trend is consistent with the FE-SEM images in Fig. 2 where the surface modification is most extreme for pure W and least extreme for W-5Ta.

Fig. 3 shows the FE-SEM images of W–Ta samples irradiated using 100 eV, dual ion (D^+ and He^+) beams, having ion fluxes of 6.0×10^{20} and 4.0×10^{20} ions m^{-2} s^{-1}, for D^+ and He^+ respectively. The images show that the addition of the D^+ flux leads to significant differences in the resulting morphology. First, for the pure W case, the SEM images exhibit a rough porous surface. This contrasts heavily with the tendril, fuzz-like surface as seen in Fig. 2. Second, the W–Ta alloy samples not only show reduced surface damage, but also the appearance of grain boundaries. This is especially clear in the W-3Ta and W-5Ta case. It appears that the

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