§97. Effect of Carbon Impurity on Tritium Permeation and Retention Behavior for Tungsten First Wall for Helical Reactor

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i) Introduction

Tungsten (W) is considered to be a candidate for plasma facing materials (PFMs) in future fusion devices. During the plasma operations, high energy hydrogen isotopes, carbon (C) and helium (He) will be irradiated and irradiation defects will be introduced into W. In particular, a tungsten-carbon (W-C) mixed layer will be formed on the surface of W by C^+ irradiation and He bubbles will be produced into W by He^+ irradiation. They will act as trapping sites of hydrogen isotopes, indicating that hydrogen isotope retention for W will be increased. In addition, the temperature of PFMs will be estimated higher than 800 K during the plasma operation. Therefore, it is necessary to elucidate the heating effect for the defect behaviors. In this study, to understand the hydrogen isotopes trapping effects of defects which were introduced by simultaneous C^+-He^+ implantation at higher temperature, heating for He^+ irradiated W at various temperature were performed.

ii) Experimental

Polycrystalline tungsten (10 mm^{φ}×0.5 mm^{t}) purchased from A.L.M.T. Corp. was used. The samples were heated at 1173 K for 30 minutes under ultrahigh vacuum to remove the surface impurities and damages introduced during the polishing processes. After preheating, the 3.0 keV He^+ was irradiated into the samples with the fluence of 1.0 × 10^{21} He^+ m^{-2} at room temperature. Thereafter, these samples were heated at 873 K and 1173 K. To evaluate the deuterium (D) retention behavior for these samples, the 3.0 keV D_2^+ implantation with fluence of 1.0 × 10^{22} D^+ m^{-2} and thermal desorption spectroscopy (TDS) measurements were performed. Furthermore, to clarify the defect states of the sample surface, the Transmission Electron Microscopy (TEM) observation was also done for the sample which heated at various temperatures after He^+ irradiation.

iii) Results and discussion

Fig. 1 shows TEM images for He^+ irradiated W and its heating behaviors at various temperatures up to 1173 K. By He^+ irradiation at room temperature, dislocation loops were introduced, and even if the sample temperature was reached at 1173 K, dislocation loops were not completely recovered due to higher thermal stability of He-trapped dislocation loop by pinning effect.[1)] In addition, He bubbles were also formed by He^+ implantation. In particular, as the sample temperature was risen above 1073 K, the size of He bubble became larger than 10 nm. Fig. 2 shows the D_2 TDS spectrum for He^+ irradiated W with the D^+ fluence of 1.0 × 10^{22} D^+ m^{-2}. This spectrum consists of four D desorption peaks located at around 390, 430, 550 and 700 K, namely Peaks 1, 2, 3 and 4, respectively, attributing to the desorption of D adsorbed on the surface[2)], trapped by dislocation loops[3)], vacancies[3)] and voids[4)], respectively. Fig. 3 shows the D_2 TDS spectra for various heating He^+ irradiated samples. It is found that the D retention at Peak 1 would be increased as 1.6 times large as that of only D^+ implantation sample. However, there is no large difference for D desorption behavior despite the growth of He bubbles by heating up to 1173 K. This fact indicated that the growth and the size of He bubbles would not affect the D retention behavior in W.

1) I. -S KIM, et al.: J. Nucl. Mater., 280 (2000) 264.
2) O. V. Ogorodnikova, et al.: J. Nucl. Mater., 313-316 (2003) 469-477.
3) H. Eleveld and A. van Veen,: 191-194 (1992) 433-438.
4) G.N. Luo, et al.: Fusion Eng. Des., 81 (2006) 957.

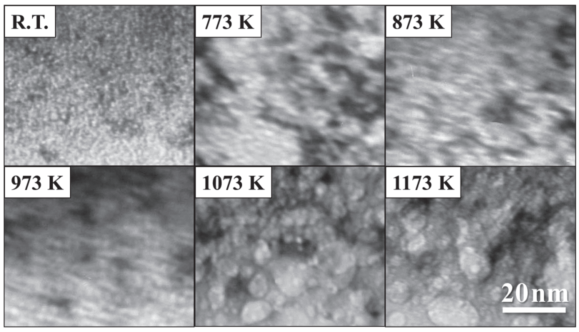

Fig. 1. TEM images for He^+ implanted W and its heating behaviors at various temperatures.

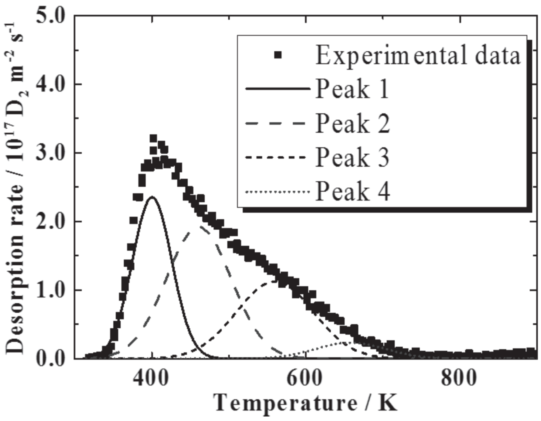

Fig. 2. D_2 TDS spectrum for He^+ and D_2^+ sequential implantation for W.

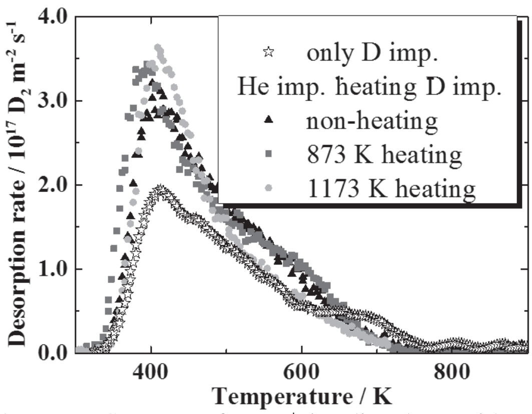

Fig. 3. D_2 TDS spectra for He^+ irradiated W with various heating treatment