# §19. Study on Hydrogen Retention and Hydrogen Recycling in Radiation Damaged Tungsten

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Hydrogen recycling is one of the important issues for stable steady state operation of fusion plasma. Especially, study of hydrogen isotope retention in plasma facing materials is indispensable from the view point of particle control and safety (i.e. tritium inventory). We have investigated hydrogen retention properties of damaged tungsten material.[1,2] Tungsten has excellent properties such as high melting temperature, high thermal conductivity, low sputtering yield and low hydrogen isotope retention as a plasma facing material. However, such properties should change during the steady state operation due to the plasma-wall interaction. As for the hydrogen isotope retention, it is well known that the retention increases due to the neutron irradiation, but there is less database of neutron irradiation effects on the hydrogen isotope retention. In this study, heavy ion (2.4 MeV $Cu^{2+}$) irradiation to the recrystallized tungsten (RC-W) has been done as a surrogate irradiation of neutron.

Figure 1 shows estimated depth profiles of displacement damage and implanted ion range distribution in W calculated using the SRIM code. The position of damage peak is ~ 400 nm beneath the surface and the damage ranges up to 1000 nm. The TEM (transmission electron microscope) observation of another W sample which were irradiated by 2.4 MeV $Cu^{2+}$ ions revealed that interstitial dislocation loops, nano-voids and fine vacancy clusters existed beneath the surface.[3]

Recrystallized W irradiated by 2.4 MeV $Cu^{2+}$ ions with a different damage level (i.e. 0.1, 0.4, 4 dpa) and non-irradiated W (0 dpa) were exposed to low energy and high flux D plasma up to ~2 x $10^{25}$ D $m^{-2}$ by using a PWI simulator APSEDAS. The flux of the D plasma was ~ 3.7 x $10^{21}$ D/$m^2$s and the ion energy was about ~ 30 eV. The surface temperature during the plasma exposure was ~ 480 K. After the D plasma exposure, the D retention in W was evaluated by thermal desorption spectroscopy (TDS). The W sample was heated with 1 K/s up to 1173 K. As shown in Fig. 2(a), two desorption peaks exist at ~560 K and ~ 740 K for the non-irradiated W. As for the irradiated W, on the other hand, a new desorption peak appears at ~ 840 K in addition to the two peaks and the desorption rate increased due to the $Cu^{2+}$ ion irradiation. It is considered that the new peak at the highest temperature (~ 840 K) is attributed to nano-voids and vacancy clusters, which are produced by the $Cu^{2+}$ ion irradiation as observed by TEM. Figure 2(b) shows dependence of retention on the damage level. Retention in the irradiated W increases with the damage level but it saturates around 0.4 dpa, suggesting that newly introduced defects may be cancelled by already existing vacancies and voids with high density.

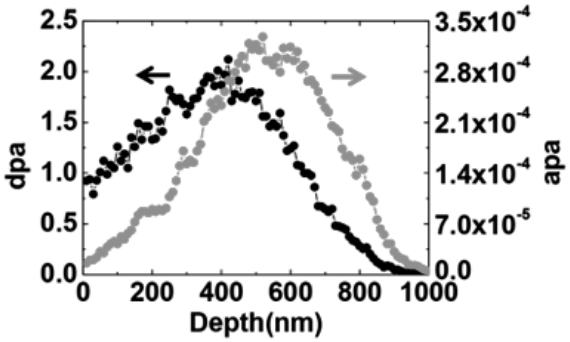


Fig. 1 Estimated depth profiles of displacement damage and implanted ion range distribution in W calculated using the SRIM code.[3] The energy of $Cu^{2+}$ is 2.4 MeV and the flux is 1 x $10^{19}$ $m^{-2}$.

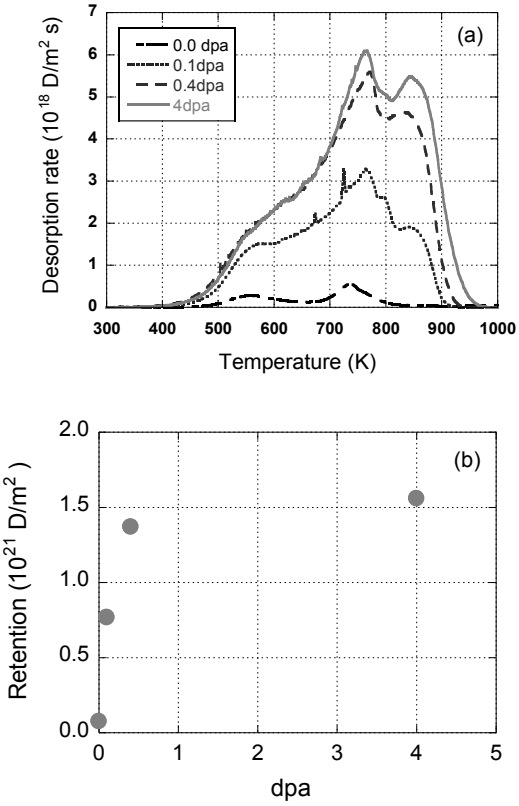


Fig.2 (a) TDS spectra of irradiated W and non-irradiated W and (b) dpa dependence of D retention.

1) M. Sakamoto et al, 15th International Conference on Plasma –Facing Materials and Components for Fusion Applications, 18-22 May 2015 (Aix-en-Provence, France), P-92.

2) H. Tanaka et al, 32nd JSPF Annual meeting, 24-27 Nov. 2015 (Nagoya) P-92.

3) H. Watanabe, N. Futagami, S.Naitou, N.Yoshida: J. Nucl. Materials **455** (2014) 51-55.