§91. Tritium Retention for Neutron Irradiated Tungsten at Higher Temperature

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i) Introduction

For future fusion reactor, it is important to evaluate the fuel retention, especially tritium retention, for plasma facing W material under operation temperature. In especially, various energetic ions, like neutron, helium, hydrogen isotopes will be dynamically implanted into W materials, which will introduce various trapping sites. We have already revealed that the D retention for 0.025 dpa neutron irradiated W is not reduced by D plasma exposure even at 800 K.[1)] This indicates that the distribution of damages would control the D retention. In addition, the recovery of damages will proceed at the temperature above 800 K. Therefore, this study focuses on the elucidation of D retention behavior for damaged W at room temperature and accumulate the fundamental knowledge for hydrogen isotope retention in W exposed to complex circumstance at higher temperature.

ii) Experimental

The mirror finished disk-type tungsten samples with the size of 10 mm in diameter and ~ 0.5 mm in thickness were exposed to 6 MeV Fe^{2+} to introduce the damages of 3.0×10^{-4} dpa to 1.0 dpa at TIARA facility in JAEA. Thereafter, the samples were picked up and transfer to Shizuoka University. The 1.0 keV deuterium ions (D_2^+) were additionally implanted into these samples with the flux of 1.0×10^{18} D^+ m^{-2} s^{-1} up to the fluence of 1.0×10^{22} D^+ m^{-2} to evaluate the D retention behavior in W using TDS.

iii) Results and discussion

Figure 1 shows the D_2 TDS spectra for W with various damage concentrations. The D desorption stages consisted of three stages, namely Stage 1 at ~ 400K, Stage 2 at ~ 600 K and Stage 3 at above 700 K. Based on the previous reports, Stage 1 was assigned to be the desorption of D adsorbed on the surface or trapped by dislocation loops.[2, 3)] The amount of D desorbed at Stage 1 was almost the same among all the damaged W with different damage concentration, although the D desorption of Stage 1 for the damaged W was higher than that for the undamaged W, indicating that the concentration of dislocation loops would be almost saturated at the damage concentration of 3.0×10^{-4} dpa. However, the D retention of Stage 2 was increased as the damage concentration increased. No large D desorption of Stage 2 for undamaged W was found, indicating that the Stage 2 should be the desorption of D trapped by vacancies, whose concentration was increased as the damages are accumulated. For Stage 3, no D desorption was found for the sample with the damage concentration less than 3.0×10^{-2} dpa, showing that the dense damage would initiate additional trapping site in W. In our previous study, the same desorption behavior was found for W with higher D fluence.[4, 5)] The TEM observation showed that void was formed in W in this condition, leading the stable D trapping site. The accumulation of void in W would shift the D desorption temperature toward higher temperature side.

These desorption behavior was simulated using trapping & diffusion model. It was found that the experimental results were good agreement with the simulation results, especially Stages 2 & 3. The temperature shift of Stage 3 toward higher temperature side would be caused by the diffusion from bulk to surface. In future work, more detail evaluation of D trapping and diffusion behaviors will be elucidated.

1) M. Shimada et al., Fusion Eng. Des., 87 (2012) 1166.
2) H. Eleveld et al., J. Nucl. Mater, 191 (1992) 433-438
3) H. Iwakiri et al., J. Nucl Mater, 307 (2002) 135-138
4) M. Kobayashi et al., Fusion Eng. Des., 88 (2013) 1749-1752.
5) Y. Oya et al., Mater. Trans., 54 (2013) 430.

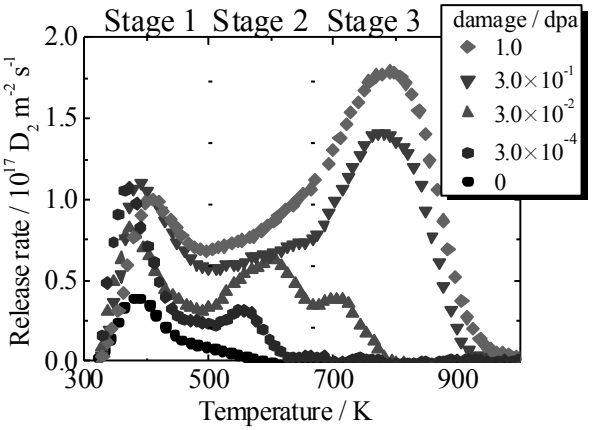

Fig. 1 D_2 TDS spectra for W with various damage concentrations