§92. Tritium Retention and Release Property of Tungsten

Nobuta, Y. (Hokkaido Univ.),
Hatano, Y., Matsuyama, M., Abe, S. (Univ. Toyama),
Sagara, A.

From the point of view of tritium (T) safety and fuel hydrogen recovery, hydrogen retention behavior of W needs to be investigated for its use in fusion reactor. Tungsten (W) is most promising plasma-facing material in ITER and beyond. Almost all studies on hydrogen retention property for W have been done just after hydrogen uptake procedure. In terms of more accurate estimation of T inventory and reduction/removal of T retained in W, the long-term T release behavior after T irradiation also needed to be clarified. In addition, helium (He) particle produced by D-T nuclear reaction is implanted into plasma-facing material, which should influence the T retention/release behavior. In this study, T ion irradiation was conducted for He pre-implanted W, and the amount of retained T in W and its long-term release behavior were evaluated.

First, W samples (Nilaco co., 99.9% purity) were irradiated by He ion with energy of 5 keV at fluences varing from 1×10^{16} to 1×10^{18} He/cm^2. After the He irradiation, the surface morphology was investigated with secondary electron microscopy (SEM). After the He irradiation, DT^+ ion with energy of 1.0 keV (0.6 keV for T) were irradiated with fluence of 4.5 x 10^{14} T/cm^2 (9.0 x 10^{16} D+T/cm^2). The samples temperature during all the irradiations were room temperature. After these irradiations, the amount of retaiend T in the samples was measured quantitatively with BIXS. After that, the samples were preserved in vacuum. In order to evaluate the reduction of retained tiritum during the vacuum preservation, the amount of retained T in the samples was periodically measured with an IP technique.

SEM images of surface morphology of W after He pre-irradiation with fluence of 1×10^{17} He/cm^2 and 1×10^{18} He/cm^2 are shown in Fig.1 (a) and (b), respectively. At 1×10^{17} He/cm^2, no change in surface morphology was seen. On the other hand, He blisters with 2-3 μm in diameter were observed after He irradiation with fluence of 1×10^{18} He/cm^2. It has been reported that when He blister is formed on the surface of metal after He irradiation, He bubbles containing high-pressured He gas are formed just beneath the surface and interconnected channels and interbubble fracture can be formed due to the high pressure [1-3]. These changes of micro structure caused by helium implantation could influence T retention and long-term release behavior of retained T as described below.

Figs.2 and 3 shows the amount of retained T as a function of pre-irradiated He fluence and time evolution of the amount of T during vacuum preservation for T-irradiated W, respectively. As the He fluence increased up to 1×10^{17} He/cm^2, the amount of retained T increased (Fig.2) and the long-term release rate of retained T became lower (Fig.3). On the other hand, in the case of 1×10^{18} He/cm^2, the amount of retained T became smaller (Fig.2) and the long-term release rate became higher (Fig.3) compared to 1×10^{17} He/cm^2 case.

These results indicate that the formation of He bubbles (blisters) and the change of micro structures caused by He irradiation could siginificantly influence the T retention and its long-term release properties in W.

[1] Y. Lifshitz and E. Cheifetz, J. Nucl. Mater., 137 (1986) 139-143.
[2] J. H. Evans, J. Nucl. Mater., 76 & 77 (1978) 228-234.
[3] W. Jager and J.Roth, J. Nucl. Mater., 93 & 94 (1980) 756-766.

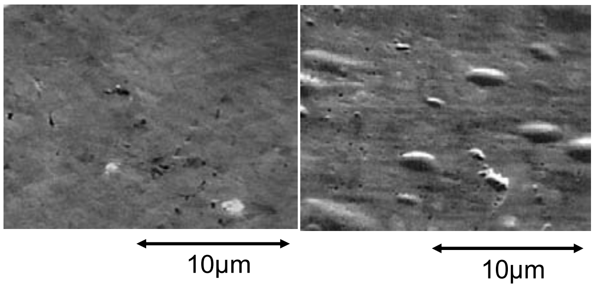

Figure 1 SEM images of surface morphology of W irradiated with pre-irradiated He fluence of (a) 1×10^{17} He/cm^2 and (b) 1×10^{18} He/cm^2.

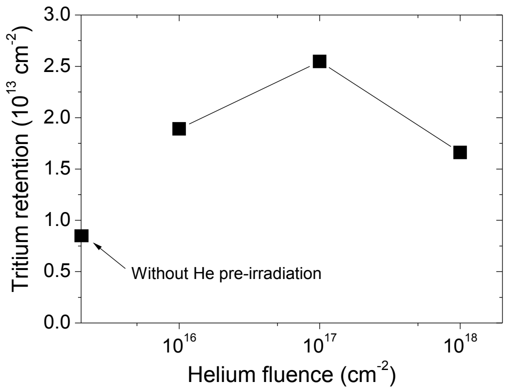

Figure 2 Dependence of T retention on pre-irradiated He fluence.

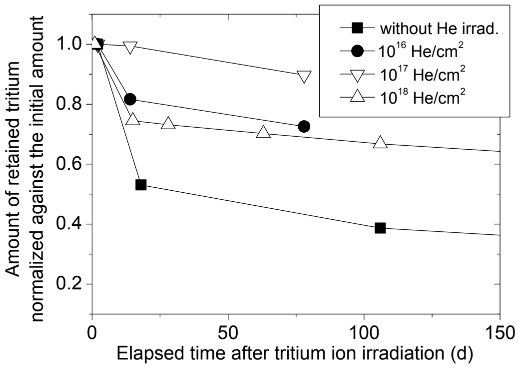

Figure 3 Time evolution of amount of retained T for He pre-irradiated W during preservation in vacuum.