§91. Helium Irradiation Effects on Tritium Retention and Long-term Tritium Release in Polycrystalline Tungsten

Nobuta, Y. (Hokkaido Univ.),
Hatano, Y., Matsuyama, M., Torikai, Y. (Univ. Toyama)

The hydrogen retention property of tungsten should be studied for future use in fusion devices. Tungsten materials have been used as plasma-facing material and are a candidate plasma-facing material in ITER and a demo reactor. Almost all studies on the hydrogen retention properties of tungsten have been done just after hydrogen uptake. From the point of view of a more accurate estimation of tritium retention and reduction/removal of tritium retained in tungsten, the long-term tritium release behavior after tritium uptake also needed to be clarified. In addition, helium particles produced by the deuterium-tritium nuclear reaction are irradiated to the plasma-facing material. Thus, helium bubbles and ion-induced defects created by helium irradiation should have an influence on tritium retention and long-term release behavior. In this study, tritium ion irradiation was performed on helium pre-implanted tungsten, and the tritium retention in the tungsten was evaluated using an imaging plate (IP) technique and also using β -ray induced X-ray spectroscopy (BIXS). After the irradiation experiments, the tungsten samples were preserved in vacuum at room temperature. By taking advantage of the fact that the IP and BIXS methods are non-destructive techniques with no influence on retained tritium, the change in tritium retention after preservation in vacuum was repeatedly measured.

Before tritium ion irradiation, samples were irradiated with helium ions with an energy of 5 keV using an ECR ion source at Hokkaido University. The helium ion flux estimated by measuring the sample current was approximately 2×10^{13} He/cm^2/s, and the helium ion fluence varied from 1×10^{16} He/cm^2 to 1×10^{18} He/cm^2. The samples were irradiated at room temperature. After the helium irradiation, tritium ion (DT^+) irradiation was performed at the University of Toyama. The incident energy of DT^+ ion was 1.0 keV (0.6 keV for T). The ion fluence was estimated to be 5 x 10^{14} T/cm^2 (1 x 10^{17} D+T/cm^2) by measuring the sample current. The samples were at room temperature during the irradiation. After the irradiation, the amount of retained tritium in the samples was measured quantitatively with BIXS. The samples were then preserved in vacuum. In order to evaluate the reduction of retained tritium during the vacuum preservation, the amount of tritium in the samples was periodically measured with an IP technique. The tritium-irradiated samples were simultaneously measured against standard samples containing a known quantity of tritium and a reduction of tritium in the tritium-irradiated samples was estimated by comparing the PSL intensity of the two sample types.

Figure 1 shows the time evolution of tritium retention during vacuum preservation of tritium-irradiated tungsten. The tritium retention in the figure is normalized against those just after tritium irradiation. In the case of no helium pre-irradiation, tritium retention rapidly dropped off to approximately half of the initial amount after the first couple dozen days of preservation in vacuum, and then continued to slowly decrease afterwards. It has been pointed out by the authors that the trap energy of tritium in materials greatly influences the long-term release behavior [1]. Also, hydrogen retained in tungsten has several trapping energies because of being trapped in different trap site types. The tritium implanted in the tungsten of this study was thought to be trapped by different types of trap sites, such as pre-existing sites (grain boundary, point defect, etc.) and ion induced sites. Tritium trapped in tungsten with relatively low trap energy would be released at an early stage of vacuum preservation while that with a higher trapping energy would be released during later stages. The long-term release behavior was influenced significantly by helium fluence. At 1×10^{17} He/cm^2, the trapped tritium was released much more slowly than the case of 1×10^{16} He/cm^2. Since larger amounts of helium bubbles and defects are created in the case of 1×10^{17} He/cm^2 as compared to 1×10^{16} He/cm^2, the implanted tritium would be more easily trapped in the bubbles and ion-induced defects. Therefore, the release rate of retained tritium decreased as the helium fluence increased up to 1×10^{17} He/cm^2. On the other hand, the long-term release rate at a helium fluence of 1×10^{18} He/cm^2 increased compared to the lower helium fluence cases. At 1×10^{18} He/cm^2, helium blisters with diameters of several μm were observed by SEM observation. It is possible that the micro cracks can be caused by high helium bubble pressure at 1×10^{18} He/cm^2 [2),3)]. These cracks can act as a short pass to the surface for the retained tritium during vacuum preservation. This might be one of the possible reasons for the long-term release behavior at 1×10^{18} He/cm^2. The results obtained in this study indicate that the long-term tritium release should be taken into account for more precise estimation of tritium retention in the long-term use of tungsten in fusion devices.

1) Nobuta, Y. et al.: Fus. Eng. Des., **89** (2014) 1516.
2) Y. Lifshitz. Et al.: J. Nucl. Mater., **137** (1986) 139.
3) J. H. Evans, J. Nucl. Mater., **76 & 77** (1978) 228.

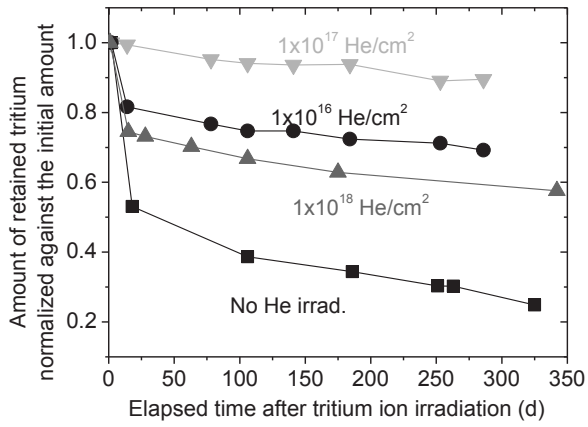

Figure 1 Time evolution of tritium retention in helium pre-irradiated tungsten during vacuum preservation.