§36. Study on Surface Modification of Tungsten Plasma Facing Materials

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It is of importance to clarify phenomena of implantation, retention, diffusion and permeation of tritium on surface of the armor materials of the first wall/blanket and the divertor from a viewpoint of precise control of fuel particles, reduction of tritium inventory and safe waste management of materials contaminated with tritium. In addition, it is well known that re-deposited layer, which includes the first wall components emitted by sputtering and residual gases such as oxygen, is formed. On the other hand, tungsten would be used as armor material of the first wall and divetor in demo reactor. Therefore, clarification of behavior of tritium on surface exposed by plasma in all metallic first wall and divertor needs to be made. In the present work, surface analyses have been carried out for long term installed samples on first wall in spherical tokamak QUEST, which is an all metallic first wall device.

Samples have been installed on vacuum chamber of spherical tokamak QUEST in Kyushu University. The vacuum vessel, and an armor of divertor and center stack of QUEST are made of SUS316L and tungsten, respectively. After the plasma discharge experiments, the samples have been examined using XPS, RBS and ERD. Quantitative analyses of depth profiles of composition and the implanted H in the materials deposited on sample have been carried out by means of Rutherford back scattering (RBS) and elastic recoil detection (ERD). Depth profiles of H were measured by using an $^4He^{2+}$ analyzing beam ERD technique with an energy of 2.8 MeV. The incident angle of the analyzing beam was 72° from the surface normal to the specimen. The scattered ^4He atoms were detected with the RBS detector placed at an angle of 170° to the incidence direction. The recoiled H atoms were detected by the ERD detector at an angle of 30° to the analyzing beam direction. An Al film 12 μm thick was placed in front of the ERD detector to absorb the He ions scattered from the specimen surface. In this fiscal year, W installed on vacuum chamber of spherical tokamak has been investigated. In addition, tritium exposure experiments have been carried out using a tritium (T) exposure device.

Figure 1 shows depth profiles from XPS analyses of W installed in the 9th cycle (from 2012/11 to 2013/3). These results show that re-deposited layer was formed and main composition was Fe, O, W and Cr. Fe and Cr is considered to be sputtered on the vacuum vessel made by SUS316L. On the other hand, W is considered to be emitted from the divertor armor, the center stack and W protector. In addition, O, which is residual gas, was co-deposited in the re-deposited layer.

Figure 2 shows ERD depth profile from W installed in the 9th cycle (from 2012/11 to 2013/3). H is also detected on the W surface and is considered to be also co-deposited in the re-deposited layer.

Amount of T on the re-deposited sample which temperatures of pre-heating and T exposures were both 100 °C (same temperature of wall during plasma discharge experiment in QUEST) measured by IP method was 1.1 times higher than that of non-exposure sample in QUEST. This indicates that amount of T of surface layer of the W before and after the plasma exposure is almost same. Distribution of T from IP on the non-exposed W was not uniform and large T amount area existed spotty, however, distribution of T on the exposed W was almost uniform. In the case of W used, W was strongly rolled because a thickness of W used was 0.1mm. This resulted in slightly uneven surface. As a result, non-uniform T distribution is considered to be formed. On the other hand, in the case of the W sample which re-deposited layer was formed, T is considered to be retained uniformly in the re-deposited layer.

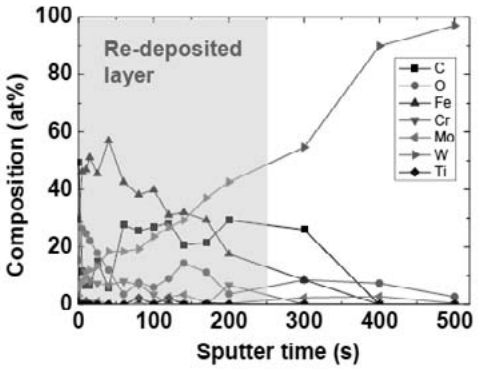

Fig. 1. Depth profiles of surface composition of W exposed to the 9 th cycle in QUEST.

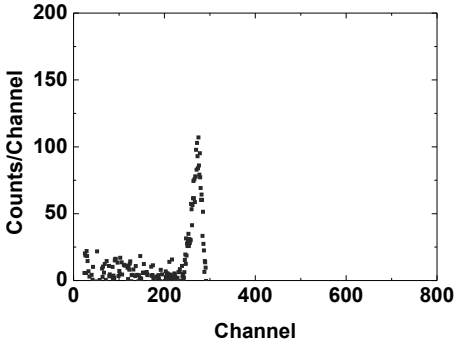

Fig. 2. ERD depth profile of W exposed to the 9 th cycle in QUEST