Hydrogen Trapping in Neutron-Irradiated Graphite

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Hydrogen recycling and tritium inventory are the critical issues for next step fusion reactors which aim to realize long pulse or steady state operation. In the case of graphite and CFCs, which are used for plasma facing walls or divertor plates, raise the hydrogen recycling rate and tritium inventory due to their affinity for hydrogen. Moreover, codeposited layers will be the major source of hydrogen and tritium even in a device mainly lined with beryllium and tungsten. Previous studies by the authors of bulk hydrogen retention in neutron-irradiated graphite showed that the amount of retained hydrogen increased by 20-50 times larger than that for unirradiated samples.

In the present study, the hydrogen absorption behavior and the nature of trapping sites have been investigated for newly irradiated graphite samples. The major interests are trapping site identification, pre-annealing treatment of the sample, hydrogen pressure and absorption temperature, which were not confirmed in the previous works.

From the results obtained in this study, two kinds of hydrogen trapping sites may exist and be additionally created by neutron-irradiation, which seem to be interstitial cluster loop edge sites (trap 1) and carbon dangling bonds at edge surfaces of crystallites (trap2). Neutron irradiation preferably produced trap 2 sites at lower fluences up to 0.011 dpa, and trap 1 sites at higher fluences above 0.047 dpa. Trap 2 tended to be annealed out by heat treatment at higher temperatures prior to hydrogen absorption, although trap 1 was hardly reduced even at 1873 K. Hence, it is anticipated that highly damaged graphite or carbon may keep high hydrogen retention even at a very high temperature. Changes of hydrogen absorption behavior and the characters of these trapping sites for the samples irradiated at various fluences will be discussed in the paper.

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