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ヘリウム混合照射におけるW中のトリチウムリテンションモデリング
Modeling tritium retention in tungsten under mixed helium irradiation

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1. Introduction

Tungsten (W) is a plasma facing material in ITER divertor and candidate armor material for DEMO. The evaluation of tritium (T) retention is important due to its impact on safety and operation. T retention in ITER has been estimated by the previous studies. In the case of D-T phase, neutrons and He ash will be present. On the one hand, neutrons create additional trap sites resulting in an increase in T retention. On the other hand, He can strongly reduce hydrogen retention in W [1] by limiting T diffusion.

In this paper, T retention in W under mixed He irradiation is modeled using TMAP7. In contrast to previous studies, simulations were performed using hydrogen transport parameters benchmarked from permeation experiments [2]. First, the recombination coefficient was determined from D-only irradiation case. Next, recombination coefficient and diffusivity required for mixed D-He irradiation case was fitted. Finally, the benchmarked transport parameters and trap parameters from recent neutron damage study [3] were used to estimate T inventory in ITER-like conditions.

2. Procedure

The detail of the benchmarked experiments is shown. Specimens used in the permeation experiment were polycrystalline W with thickness of 30 μm supplied by A.L.M.T. Corp., Japan. D ion irradiation (D-only case) as well as mixed D-He ion irradiation (D-He case) at 1 keV was performed at 550 K $\leq$ T $\leq$ 1050 K. The incident flux varied from 10^{19}- 10^{20} $m^{-2}s^{-1}$. The He concentration was fixed at 2 %.

To model the experimental data, the parameters were determined as below. Diffusivity, D, and solubility, S, were taken from Frauenfelder [4]. Initially, Anderl's [5] recombination coefficient was used. However, to fit the experimental data for D-only case, the front segment recombination coefficient, K_1, needed to be varied. Modification of hydrogen transport during mixed He-D irradiation was modeled by varying the front segment recombination coefficient, K_2, and diffusivity, D_2.

Finally, Simulation of ITER-like conditions was done by assuming full W divertor and first walls.

3. Result

Tritium retention with and without He irradiation under ITER-like condition was estimated using TMAP7 code. The surface temperature of W divertor and W first wall were taken as 500 K and 400 K, respectively [6]. The incident flux of W divertor and W first wall were set to be 2.0 x 10^{23} m^{-2} s^{-1} and 1.4 x 10^{19} m^{-2} s^{-1}, respectively. The hydrogen transport parameters were benchmarked against ion driven permeation experiments.

Fig. 5 shows the calculated T retention with and without He irradiation. Without He irradiation, the T retention estimated in this work was lower by approximately two orders of magnitude than previous estimate by Roth et al. [6]. This was due to different trap parameters resulting in different trap filling dynamics and due to traps filled sequentially (strong trapping) using 1.8 eV traps. With He irradiation, x 5 reduction in T retention observed compared to D-only case. The calculations were in good agreement with a previous estimate by Wampler et al.[7]

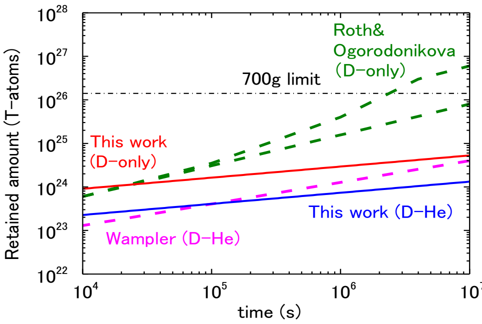

Fig.1 Estimate of T retention in ITER (All W case). The solid lines indicate T retention with and without He irradiation. Plotted in dashed line are previous studies by Roth et al [6] and Wampler and Doerner [7].

Reference

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