30aP74

Modeling Fuel Retention in Tungsten Plasma-Facing Materials under Realistic Tokamak Operation including Plasma Impurities

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Abstract

This study outlines a new methodology in modelling in-vessel Tritium inventory in tungsten by using near surface solute concentration to parameterize transport behaviour. It is empirically based on near surface solute concentration measurements from laboratory mixed ion-driven permeation experiments, and their changes due to nitrogen and helium impurities. For nitrogen, we present two limits important for modeling hydrogen retention behavior in tungsten. The limits are described using near surface solute concentration vs. Temperature diagram. At divertor relevant fluxes (10^{24} D/m^{2}s), the concentration at T < 700 K is determined by a material limit due to precipitation effects for an impurity free case. Consequently, the inward diffusion flux can be decoupled from changes in plasma parameters. At T > 700 K, the concentration is fixed by plasma parameters and scales linearly to the incident flux (i.e. diffusion limited). In the presence of N impurities at T < 700 K, co-deposition of D with N controls the solute concentration. However, at T > 700 K, the concentration is again diffusion limited. For helium, front diffusivity and recombination coefficients were determined from fitting ion driven permeation experiment data using TMAP7 code. For D-only case, the T retention estimated in this work was lower by approximately two orders of magnitude than previous estimate due to difference in the front surface recombination coefficient used in calculations. For D-He case, good agreement with previous estimate was observed. The effect of He on T transport can be modeled by changes to both an increase in the front recombination coefficient and diffusivity of W. Such He effects was larger under first wall conditions than divertor conditions. These results will open the way to adopting a more physically sound model for hydrogen-tungsten interactions and will thus improve our estimate and prediction for fuel retention in future burning plasma machines. It constitutes an example of a new methodology to model fuel retention in tungsten plasma facing materials in a forward manner that can includes changes to surface properties.

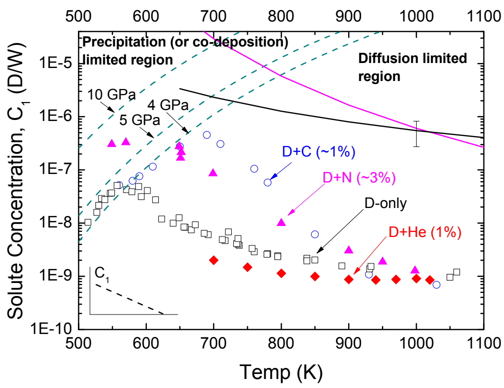

Fig. 1. Near surface solute concentration of D plotted as function of irradiation temperature for various mixed D+X (He, C, N) impurity irradiation. Also plotted are the precipitation limits in dashed blue lines for various pressures. The solid black and magenta lines correspond to extrapolation to divertor relevant fluxes (10^{24} D/m^{2}s) for D-only and D+N case, respectively.

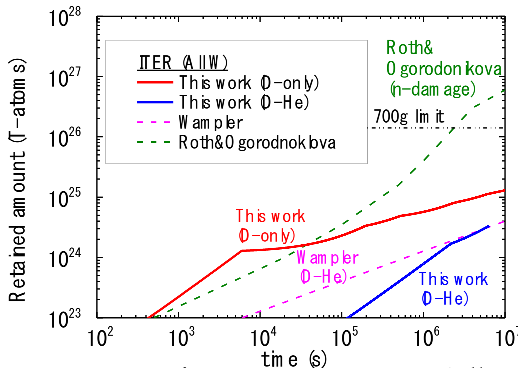

Fig. 2. Estimate of T retention in ITER (All W case). The solid lines indicate T retention for D-only and D-He case.