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Fuel retention and transport in fusion components

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Tritium (T) transport through the first wall into the coolant is a major concern in fusion reactor studies. When irradiated by plasmas, hydrogen permeation flux through in-vessel components would be significantly higher than that of gas-exposure cases. To support reactor design studies, low energy plasma-driven hydrogen isotope permeation through the first wall has been extensively investigated. This review will introduce our recent research progress in three relevant topics:
(1) Surface damage effects on hydrogen isotope permeation [1,2]. In-situ measurements of low energy deuterium (D) through helium pre-damaged tungsten (W) has been done. With the increases of helium (He) pre-irradiation fluence, the D permeation flux was found to reduce effectively.
(2) The role of W-structural materials interface [3]. The transport behavior of D in W-Cu joining sample was explored using gas driven permeation and thermal desorption spectroscopy. A large number of D atoms were found to be trapped by dislocations and impurities at the intermediate layer.
(3) Isotope effects [4,5]. The co-permeation experiments of H and D isotopes through~mm thickness materials. The H/D ratio of the steady state permeation fluxes was found to be close to the classical theoretical. Hydrogen isotope exchange is an effective method for T removal in fusion reactor materials.

References
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