P 17: Plasma Wall Interaction II

Time: Thursday 16:15–18:15 Location: KH 02.016

Invited Talk P 17.1 Thu 16:15 KH 02.016

Energetic proton damage for simulating fusion relevant neutron damage on reactor materials — •Rahul Rayaprolu — Forshungszentrum Jülich, Jülich, Germany

As fusion research progresses from experimental devices to reactors, it is imperative to comprehend the effects of fusion neutron damage and its impact on plasma-facing materials (PFMs). At present, the study of neutron damage is predominantly conducted through the use of fission reactors, which exhibit a distinct neutron-energy flux spectrum. Hence, the fission neutron damage leads damage ingrowth as compared to that of a fusion reactor. Ion irradiation is a well-established method and is frequently used as a surrogate for neutron damage. It is a more accessible method than nuclear reactors and simultaneously offers superior control over experimental conditions. Until recently, the focus has been on the use of heavy ions to replicate the displacement effects of neutron damage on the PFMs lattice. However, the heavy-ion irradiation method has a very shallow damage depth and is unable to take macroscopic effects into consideration. Energetic protons offer a solution and have been shown to deliver damage in depths ranging from 500 um to 1 mm. They produce a combination of displacement and nuclear-transmutation damage that is comparable to fusion neutrons. However, the irradiated area is constrained to the beam spot size, and the samples like fission irradiations are radioactive. In the absence of a fusion neutron source, this method can be used to study the development and influence of fusion relevant neutron damage in PFMs.

P 17.2 Thu 16:45 KH 02.016

Influence of the presence of deuterium on damage evolution in tungsten — •Zeqing Shen[1,2], Thomas Schwarz-Selinger[2], Mikhail Zibrov[2], Armin Manhard[2], and Martin Balden[2] — [1]Technische Universität München, 85747 Garching, Germany — [2]Max-Planck-Institut für Plasmaphysik, 85748 Garching, Germany

A systematic investigation was conducted to examine the effects of ion flux during deuterium (D) exposure of self-ion-damaged tungsten (W). The samples were heated to four temperatures within the range of 470–770 K during the plasma loading process. Two plasma conditions were applied during annealing: a low flux of 6e19 D/m2/s and a high flux of 5e20 D/m2/s, both using an ion energy of 5 eV/D. For comparison, annealing experiments were also carried out in vacuum. The depth distribution of deuterium was determined by 3He nuclear reaction analysis (NRA), while its total inventory was evaluated using both NRA and thermal desorption spectroscopy (TDS). Morphological modifications at the surface were analysed by confocal laser scanning microscopy (CLSM). For the flux condition of 6e19 D/m2/s, the results revealed a decrease in deuterium retention with increasing annealing temperature for both plasma annealing and vacuum annealing. The presence of D during annealing has only a small stabilizing effect on the defects. In contrast, exposure to a higher flux of 5e20 D/m2/s induces additional damage, producing micrometer-wide blisters only a few tens of nanometers high.

P 17.3 Thu 17:10 KH 02.016

Hydrogen Isotope Exchange in Tungsten Displacement Damaged at High Temperature — •Laurin Hess[1,2] and Thomas Schwarz-Selinger[2] — [1]Technical University Munich, Munich, Germany — [2]Max-Planck-Institute for Plasma Physics, Garching, Germany

Retention of hydrogen fuel in tungsten is an active area of research, as it is an integral part of modelling the tritium inventory and certification of future fusion reactors. It has been shown that hydrogen retention increases significantly due to displacement damage produced by 14 MeV fusion neutrons. Over the last years, a basic understanding of the behaviour of hydrogen in point defects was acquired. However, damage at high temperatures can also produce nm-sized voids. To gain understanding of hydrogen in nm-sized voids, tungsten single crystals were self-damaged by irradiation with 20 MeV tungsten ions at 1370 K and decorated with <5 eV deuterium (D) from a low-temperature plasma. The retained deuterium was then exchanged for protium (P) using the same plasma loading process. D depth profiles were measured after D loading and periodically during P loading using ^{3}He-NRA. After the loading, the retained hydrogen isotopes were measured using TDS. This showed that hydrogen in voids does not exchange at temperatures up to 370 K. At higher temperatures exchange starts to happen, with almost all hydrogen exchanged within hours during exposure at 460 K. The results were compared to the model proposed by Zibrov and Schmid [1].

[1] M. Zibrov, K. Schmid, Nucl. Mat. Eng. 30 (2022) 101121

P 17.4 Thu 17:35 KH 02.016

Effect of high displacement damage at different dose rates and temperatures on Deuterium retention in Eurofer97 — •Abdulrahman Albarodi[1,2], Thomas Schwarz-Selinger[2], Dina Mergia[3], and Dimitrios Papadakis[3] — [1]Technical University of Munich, Garching, Germany — [2]Max Planck Institute for Plasma Physics, Garching, Germany — [3]NCSR Demokritos, Athens , Greece

Open-volume defects in irradiated metals are attributed to increased gas retention. In this study, deuterium (D) retention was used as an indicator for the evolution of open-volume defects at different temperatures and dose-rates. EUROFER97 samples were irradiated with a defocused continuous 11.6 MeV Au beam with a dose rate of 2.5-16.5 dpa/hr using ions to peak displacement damage doses of 50-100 dpa at 100-400°C. Rutherford-backscattering (RBS) with 4 MeV helium was used on aluminum surrogates to characterize the Au implantation profile. The Au depth profile showed a region of high displacement damage of 500 nm that is gold-free for uncontaminated defect analysis. X-ray diffraction showed clear differences in the lattice constant and the width of the Bragg peaks between damaged samples and undamaged reference. The samples were subsequently exposed to D-plasma at 100°C and low energy (<5 eV/D) until full D defect decoration. D depth profiles were measured using 3He nuclear reaction analysis. D-retention results are comparable with previous low-dose (0.6 dpa) irradiated samples at 100°C. For the sample damaged at 300-400°C, it is comparable to undamaged EUROFER97. NRA and XRD results show recovery of open-volume defects at 400°C.

P 17.5 Thu 18:00 KH 02.016

Developing Tritium Diffusion Barrier Materials for Plasma-Facing Components in Fusion Reactors — •Markus Hermanski[1,3], Liang Gao[1], Arkadi Kreter[1], Jan Willem Coenen[1], Rongxing Yi[1], Sebastijan Brezinsek[1,2], and Christian Linsmeier[1,2] — [1]Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management — Plasmaphysics (IFN-1), Jülich 52425, Germany — [2]Ruhr-Universität Bochum, Faculty of Physics and Astronomy, Bochum 44801, Germany — [3]RWTH Aachen University, Faculty of Georesources and Materials Engineering, Aachen 52072, Germany

Fuel permeation and retention in wall materials of fusion reactors, especially Tritium (T), are important aspects to be prevented to achieve commercial viability of fusion power in the future. For this reason, T diffusion barrier materials have been developed, however, focusing on applications at the coolant side. Preventing T permeation into the coolant, they cannot address the degradation of wall materials induced by H isotopes from plasma. Here, a diffusion barrier made of tungsten carbides (WxC) is proposed and will be developed for applications at plasma-facing surfaces.

WxC coatings will be fabricated via field assisted sintering technique (FAST) on W and G91 steel substrates, followed by deuterium (D) plasma exposure, where laser induced breakdown spectroscopy (LIBS), nuclear reaction analysis (NRA) and thermal desorption spectroscopy (TDS) measurements will be employed to measure the D depth profile after plasma exposure.