P 8: Plasma Wall Interaction I/HEPP IV

Time: Tuesday 17:00–19:10 Location: CHE/0091

Invited Talk P 8.1 Tue 17:00 CHE/0091

Fuel retention and removal in the JET tokamak — •Dmitry Matveev[1], David Douai[2], Tom Wauters[3], Sebastijan Brezinsek[1], and JET Contributors[4] — [1]Forschungszentrum Juelich GmbH, EURATOM Association, 52425 Jülich, Germany — [2]CEA Cadarache, IRFM, F-13108 Saint Paul Lez Durance, France — [3]ITER Organization, Route de Vinon-sur-Verdon, CS 90 046, F-13067 St Paul Lez Durance Cedex, France — [4]See the author list of J. Mailloux et al, Nucl. Fusion 62, 042026 (2022)

The control of fuel retention remains a critical issue for future fusion reactors due to tritium fuel self-sufficiency and related radiation safety requirements. This talk will cover fuel retention studies in the JET tokamak over the past decades, from the carbon wall configuration and the first deuterium-tritium experiment (DTE1) to the beryllium-tungsten ITER-Like Wall (ILW) configuration and the recent second deuterium-tritium campaign (DTE2). Fuel retention mechanisms, the aspects of long-term and short-term fuel retention, and post-discharge outgassing of hydrogen isotopes from tokamak wall materials, as well as wall cleaning techniques and respective fuel removal experiments will be addressed.

P 8.2 Tue 17:30 CHE/0091

Early stages of He cluster formation in pristine and displacement-damaged tungsten — •Annemarie Kärcher[1,2], Vassily V. Burwitz[2], Thomas Schwarz-Selinger[1], Lucian Mathes[2], Wolfgang Jacob[1], and Christoph Hugenschmidt[2] — [1]Max-Planck-Institut für Plasmaphysik, 85748 Garching, Germany — [2]Technische Universität München, 85748 Garching, Germany

In future fusion reactors, tungsten as plasma-facing material will be subjected to intense fluxes of helium (He). While the consequences of high He fluxes on the surface properties of tungsten have already been thoroughly studied, there are no experiments that could clarify the process of early He cluster formation. To understand the initial steps of the interaction of He with W, especially the impact of pre-existing defects, annealed, polycrystalline W samples were irradiation-damaged to various damage levels. Then, these samples were exposed to a low-temperature He plasma at fluxes between 10^{17} and 10^{19} He/m^{2}s and various fluences using implantation energies of 50 and 100 eV. The samples were measured by positron annihilation spectroscopy for defect characterization and elastic recoil detection analysis (ERDA) for quantification of the He retention. For the depth distribution of He, a novel method was applied: thin surface layers of the sample were subsequently removed followed by ERDA measurements in between the erosion steps. The removal was performed by by electrochemical oxidation and dissolution of the oxide in NaOH. The results show a higher He retention in pre-damaged samples by factors up to 10 and a deeper reaching distribution of He in undamaged samples.

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Low Energy Ion Scattering investigation of dynamic surface segregation of chromium in the WCrY SMART material — •Pawel Bittner, Hans Rudolf Koslowski, Andrey Litnovsky, and Christian Linsmeier — Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung, 52425 Jülich,Germany

Self-passivating Metal Alloys with Reduced Thermo-oxidation (SMART) are promising candidates for the first wall of the DEMOnstration power plant (DEMO). These materials should feature an increased oxidation resistance during accidental conditions and tolerate plasma loading during regular operation of the power plant. In this work, the effects of segregation, diffusion and sputter erosion on the surface Cr concentration of a tungsten-chromium-yttrium SMART alloy (WCrY - 68 at% of W, 31 at% of Cr and 1 at% of Y) are studied with low energy ion scattering (LEIS) measurements at 800 K, 900 K and 1000 K and numerical simulations. The LEIS is operated with He^+, Ne^+ and Ar^+ ions at 1 keV in sputter mode. The time resolved measurements show a build-up of Cr at the surface directly after increasing the temperature, followed by a slow decrease with evolving time. A comparison to a discrete layer model, in which the segregation enthalpy, entropy and atomic mobility are taken into account, indicates that this decrease is caused by a slower bulk diffusion rate compared to the rates of sputtering and surface segregation.

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Influence of the Microstructure of Tritium Permeation Barrier Layers on Hydrogen Isotope Retention and Permeation — •Jonah Lennart Book, Anne Houben, and Christian Linsmeier — Forschungszentrum Jülich GmbH, 52425 Jülich, Germany

For the safe and efficient operation of fusion reactors, tritium permeation barriers, or TPBs, are required to prevent fuel loss through first wall materials. Yttrium oxide is chosen as a TPB due its favorable neutron activation behavior compared to other candidates. Different Y_2O_3 layers several hundred nanometers thick are deposited onto a steel substrate using RF magnetron sputtering and studied using scanning electron microscopy. The samples are annealed at 550°C to obtain the favorable cubic phase of Y_2O_3, which is verified by X-ray diffraction. Permeation measurements are performed by gas-driven deuterium permeation experiments from 25 mbar to 800 mbar at 300°C to 550°C. The calculation of the single layer permeability is introduced to obtain a comparable value of the permeation reduction effect for the different coatings. In addition, in lag time measurements the diffusivity of the sample is determined separately from the permeability. The permeation results and layer permeabilities are compared for the different microstructures. Furthermore, the hydrogen isotope retention of the different layers is measured using nuclear reaction analysis and evaluated with their permeation reduction performance.

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Ex-Situ Ion Beam Analysis of ^{13}C on Plasma-Facing Components of Wendelstein 7-X — •Christoph Kawan[1], Sebastijan Brezinsek[1], Timo Dittmar[1], Sören Möller[1], and the W7-X team[2] — [1]Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung - Plasmaphysik, Partner of the Trilateral Euregio Cluster (TEC), 52425 Jülich, Germany — [2]See author list of T. Klinger et al. (2019) Nucl. Fusion 59 112004

At the end of OP1.2B 4.5 * 10^{22} ^{13}C - methane molecules were injected to study carbon transport in W7-X and generate benchmark data for material migration codes. Here we present the results of a dedicated NRA analysis using the ^{13}C(d, p_0)^{14}C reaction on 24 divertor target elements of different toroidal positions. The majority of the deposition was on the divertor half module where the carbon was injected (60%), with layers up to 100 μm in a 5 cm radius around the injection location. The reminder of the ^{13}C was deposited on the other divertor modules close to the strike line.

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Experimental Determination of Irradiation-Induced Stress Relaxation in Thin Tungsten Wires — •Alexander Feichtmayer[1,2], Max Boleininger[3], Raphael Colson[1,2], Bailey Curzadd[1,2], Sebastian Estermann[1,2], Till Höschen[1], Johann Riesch[1], Thomas Schwarz-Selinger[1], and Rudolf Neu[1,2] — [1]Max Planck Institute for Plasma Physics, Boltzmannstr. 2, 85748 Garching, Germany — [2]Technical University Munich, Boltzmannstr. 15, 85748 Garching, Germany — [3]Culham Centre for Fusion Energy, Abingdon, OX14 3DB, Oxfordshire, UK

The development of suitable materials for the highly loaded plasma facing components is a major challenge in the development of a future fusion power plant. The influence of neutron irradiation on the mechanical properties is particularly difficult to measure, since there is no suitable neutron source available. A widely used technique to simulate neutron irradiation is the use of high energy ions, since these can produce similar dislocation damage as neutrons. For this purpose, a dedicated device has been developed to allow simultaneous ion irradiation as well as mechanical testing. This device and the latest upgrades, as for example a laser-based strain measurement system, will be presented. The setup of a stress relaxation experiment on 16 μm tungsten wires, to study the synergistic effects between mechanical stress and irradiation damage, will also be presented. For this the wires were preloaded with up to 2 GPa and simultaneously irradiated with 20.3 MeV tungsten ions. The resulting force drop (10-30 mN) and the ion current across the sample (0.1-0.8 nA) was measured.

P 8.7 Tue 18:55 CHE/0091

First trials to regenerate the surface of plasma-facing components by wire based laser metal deposition — •Jannik Tweer[1], Robin Day[2], Thomas Derra[2], Daniel Dorow-Gerspach[1], Christian Linsmeier[1], Thorsten Loewenhoff[1], Ghaleb Natour[3,4], and Marius Wirtz[1] — [1]Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung - Plasmaphysik, 52425 Jülich, Germany — [2]Fraunhofer-Institut für Produktionstechnologie IPT, 52074 Aachen, Germany — [3]Forschungszentrum Jülich GmbH, Zentralinstitut für Engineering, Elektronik und Analytik (ZEA-1), 52425 Jülich, Germany — [4]Lehrstuhl und Institut für Schweißtechnik und Fügetechnik, RWTH Aachen University, 52074 Aachen, Germany

The harsh conditions inside a nuclear fusion reactor put high demands on the plasma-facing materials and components. Tungsten is the preferred material for lining the inner walls of future fusion reactors. It is considered as such due to its exceptionally high melting point, excellent thermal conductivity, low tritium retention and high erosion resistance during plasma exposure. However, even plasma-facing components made of tungsten get damaged during reactor operation, thereby limiting the lifetime of these components. It is envisioned to counteract these erosion losses by local deposition of tungsten using the wire based laser metal deposition process (LMD-w). During this process new material gets fused to the substrate, enabling in-situ repair of damaged plasma-facing components. Several experiments were conducted to find suitable process parameters and methods to create layers of new material by placing several melt tracks next to each other.