11th International Workshop on Hydrogen Isotopes in Fusion Reactor Materials

Schloss Ringberg May 29 - 31, 2012

Arial view of *"Schloss Ringberg", Conference site of the Max-Planck Society*

The 11th International Workshop on Hydrogen Isotopes in Fusion Reactor Materials is hosted by the "Plasma Edge and Wall" (E2M) Division of Max-Planck Institute for Plasma Physics, Garching.

This workshop is a satellite meeting of the 20th International Conference on Plasma Surface Interactions, which will be held in Aachen, Germany, in the preceding week.

Book of Abstracts

11ᵗʰ International Workshop
on Hydrogen Isotopes in Fusion Reactor Materials
Programme

Tuesday, May 29

until 12:00	Arrival

12:30 - 13:45	Lunch

The length of all presentations (execpt the evening talk on Tuesday 20:00) is 20 min including discussion

13:45 - 14:00		Welcome and Information

| 14:00 - 14:20 | **#1** | *T. Tanabe* |
| | | Revisiting carbon materials as plasma facing materials of a fusion reactor |

| 14:20 | **#2** | *V.G. Stankevich, N.Yu. Svechnikov, Y.V. Zubavichus, A.A. Veligzhanin, V.A. Somenkov, L.P. Sukhanov, B.N. Kolbasov, A.M. Lebedev, and K.A. Menshikov* |
| | | Observation of nanostructural elements of thick homogenous hydrocarbon films redeposited from T-10 tokamak plasma discharge |

| 14:40 | **#3** | *I. Arkhipov, N. Klimov, V. Stankevich, V. Budaev, and K. Vukolov* |
| | | Characterization of Hydrocarbon Films Deposited under Conditions of the Pulsed and Steady-State Plasma Discharges |

| 15:00 | **#4** | *Tamara J. Finlay, James W. Davis, Anthony A. Haasz* |
| | | Hydrogen Isotope Exchange in DIII-D Codepoints |

| 15:20 | **#5** | *Masabumi Nishikawa* |
| | | Recovery Efficiency of Trapped Tritium on Tritium Balance in a D-T Fusion Reactor |

15:40 - 16:00	Coffee Break

| 16:00 | **#6** | *M. Rubel, V. Philipps, A. Huber, D. Ivanova, P. Petersson, B. Schweer and M. Zlobinski* |

#6 Fuel Removal from Plasma-Facing Components: Overview of Methods Tested on Materials from the TEXTOR Tokamak

16:20 **#7** *L. Mercadier, J. Hermann, A. Semerok, C. Grisolia*
Laser-Induced Breakdown Spectroscopy Investigations for Tokamak applications

16:40 **#8** *J. P. Coad, S. Brezinsek, D. Douai, S. Gruenhagen, T. Loarer, G. F. Matthews, R. Neu, V. Philipps2and EFDA-JET Contributors*
Comparative Fuel Retention In JET With The ITER-like Wall And The Carbon Wall

17:00 **#9** *S. I. Krasheninnikov, J. Guterl, A. Yu. Pigarov, R. D. Smirnov, E. D. Marenkov, A. A. Pisarev, and I. V. Tsvetkov*
Modeling of Hydrogen Retention and Coupled Plasma-Wall Interactions

17:20 **#10** *Evgeny Marenkov, V. Kurnaev, A. Lassa, K. Nordlund*
Molecular Dynamic Simulation of KeV Hydrogen Molecular Ions Interaction With Solids

| 17:40 - 18:20 | *time for discussion* |
| 18:30 | Dinner |

20:00 - 20:45 **#11** *After dinner talk by Alexander Pisarev*
Adsorpion of Hydrogen on Metal Surface

| **Wednesday, May 30** | |
| 8:00 - 9:00 | Breakfast |

09:00 **#12** *N. Juslin, F. Sefta, and B. D. Wirth*
Modeling H and He interactions in W

09:20 **#13** *Yu. Gasparyan, V. Efimov, A. Mednikov, and A.Pisarev*
Fine Structure of TDS Spectra of Deuterium from Tungsten

09:40 **#14** *Armin Manhard, K. Schmid, T. Dürbeck, and W. Jacob*
Unambiguous Determination of the Binding Energies for Deuterium in Tungsten

10:00 **#15** *T. Otsuka and T. Tanabe*
Thermal release mechanism of hydrogen implanted in metals by plasma exposure

| 10:20 - 10:40 | Coffee Break |

Wednesday, May 30

10:40   **#16** *Masakatsu Fukumoto, T. Nakano, K. Itami, T. Wada, and Y. Ueda*

Deuterium retention and desorption from vacuum plasma splayed tungsten coating irradiated by deuterium and arbon mixed ion beam

11:00   **#17** *Yuji Hatano, Jie Shi, Teo Nozaki, Tomonori Tokunaga, Hideo Watanabe, Naoaki Yoshida, Takuya Nagasaka, Ryuta Kasada, Akihiko Kimura*

Gas-Driven Hydrogen Isotope Retention in VPS Tungsten

11:20   **#18** *Y. Zayachuk, M.H.J.'t Hoen, P.A. Zeijlmans van Emmichoven, I. Uytdenhouwen, and G. van Oost*

Deuterium retention and surface modification of tungsten and tungsten-tantalum alloys as a result of high-flux deuterium plasma exposure

11:40   **#19** *T. Sizyuk and A. Hassanein*

The Effects of Co-deposited Carbon Impurities as Surface Barrier for Hydrogen Recycling in Tungsten

12:30 - 14:00	Lunch

14:00   **#20** *Tomohisa Taguchi, M. Kobayashi, K. Kawasaki, Y. Miyahara, N. Ashikawa, A. Sagara, N. Yoshida, Y. Hatano, Y. Oya, and K. Okuno*

Correlation between irradiation defects formation and deuterium retention behavior in tungsten under C^+-D_2^+ simultaneous implantation

14:20   **#21** *M.H.J. 't Hoen, B. Tyburska-Püschel, K. Ertl, M. Mayer, A.W. Kleyn, H. Schut, P.A. Zeijlmans van Emmichoven*

Deuterium Retention in Pre-Irradiated Tungsten Exposed to High-Flux Plasmas

14:40   **#22** *O. V. Ogorodnikova, S. Markelj, K. Sugiyama, I. Čadež, Yu. Gasparyan, V. Efimov*

Atomic and Low-energy Deuterium Interaction with Selfdamaged Tungsten

15:00   **#23** *Leon Begrambekov, A. Kaplevsky, Ya. Sadovsky, P. Shigin and S. Vergasov*

The Particle Trapping in Stainless Steel in Oxygen Contaminated Hydrogen Plasma

15:20   **#24** *Y. Sakuma, N. Matsunami, N. Ohno, M, Tokitani and S. Masuzaki*

Depth Distribution Of Deuterium In Fe2O3 Under Low Energy Deuterium Plasma

15:40 - 16:00	Coffee Break

16:00   **#25** *N.P. Bobyr, A.V. Spitsyn, A.V. Golubeva, M. Mayer, Yu.M. Gasparyan, V.S. Efimov, V.M. Chernov*

Deuterium Retention in the V-4Cr-4Ti Alloy

16:20   **#26** *Tatiana Kompaniets, E.Denisov, I.Shikin, A.Yukhimchuk jr.*

Hydrogen Interaction With RAFM Steel Rusfer-EK-181

| 16:40 | **#27** | *A.V. Golubeva A.V. Spitsyn, M. Mayer, Yu.M. Gasparyan, N.P. Bobyr, V.M. Chernov, O.V. Ogorodnikova* |
| | | Deuterium Retention in Low Activated Ferritic-Martensitic Steels |

| 17:00 - 18:00 | | Discussion |

| 18:30 | | Dinner |

Thursday, May 31

| 8:00 - 9:00 | | Breakfast |

| 09:00 | **#28** | *J.S. Hu, G.Z. Zuo, S.Zhen, J.G. Li, B. Cao, D. K. Mansfield, L.E. Zakharov* |
| | | Lithium coating for recycling and H/(H+D) control in EAST/HT-7 |

| 09:20 | **#29** | *F. L. Tabarés, D. Tafalla, E. Oyarzabal and A. B. Martín-Rojo* |
| | | Studies of Isotope Interchange on Lithium in TJ-II |

| 09:40 | **#30** | *J. Roth, B. Wampler, S. van Deusen, M. Oberkofler* |
| | | D retention in and out-gassing from BeO on Be |

| 10:00 | **#31** | *Robert D. Kolasinski, Norman C. Bartelt, Josh A. Whaley, and William R. Wampler* |
| | | Fundamental Hydrogen Interactions with Beryllium Surfaces |

| 10:20 - 10:40 | | Coffee Break |

| 10:40 - 12:00 | | Discussion |

| 12:30 - 14:00 | | Lunch |

| 14:00 | | Departure |

Revisiting carbon materials as plasma facing materials of a fusion reactor

T. Tanabe

Interdisciplinary Graduate School of Engineering Science, Kyushu University, Japan.

Considering recent data on hydrogen retention obtained in JT-60U [1-3], which was usually operated at 573K, and lots of laboratory date at rather higher temperatures, we demonstrate possible use of carbon materials as PFM in a reactor. The benefit of the carbon materials is their quite small hydrogen diffusion, which inhibits hydrogen penetration into graphite grains or carbon particles, even though the materials are porous. In addition, the hydrogen retention in redeposited carbon layers could be very small, if the layers were produced at above ~800K, near reactor operating temperature. Moreover, the redeposited carbon layers at such high temperature are rather graphitized and retain less hydrogen (below a few %) and/or inhibit hydrogen uptake afterwards. Consequently hydrogen retention in carbon materials seems more easily saturated compared to those for any metal materials in which no saturation seems to occur owing to hydrogen penetration and permeation in depth..

It is well known that neutron irradiation degrades thermal conductivity of carbon materials and gives dimensional change. However, their high temperature usage would relax the limitation. In particular, their usage as armor tiles of PFM surfaces allows temperature rise and even the dimensional change. Furthermore the temperature increase due to the loss of thermal conductivity by neutron irradiation would reduce tritium retention and anneal the damage. A reactor needs periodic maintenance for safety and exchangeable structure for fixing the carbon armor tiles to the cooling base would greatly help to replace the amours if the replacement was required, compared to ITER like large divertor cassette structure. Still a problem remains how to fix the carbon armor tiles to the cooling base.

Redeposition of carbon at plasma shadowed area such as tile gaps and remote area like pumping ducts is unavoidable. However, the redeposition in tile gaps in main chamber of JT-60U and JET was found to be much smaller than that in the remote area. Furthermore, the redeposition occurred at limited locations and could be removed mechanically.

[1] T. Tanabe, K. Masaki and K. Sugiyama, Physica Scripta, **T138** (2010) 14006
[2] M. Yoshida, T. Tanabe, T. Hayashi, et al. Physica Scripta, **T145** (2011) 014023
[3] M. Yoshida, T. Tanabe, K. Sugiyama, et al. **415** (2011) S752-S756

Observation of nanostructural elements of thick homogenous hydrocarbon films redeposited from T-10 tokamak plasma discharge

V.G. Stankevich[*], N.Yu. Svechnikov, Y.V. Zubavichus, A.A. Veligzhanin, V.A. Somenkov, L.P. Sukhanov, B.N. Kolbasov, A.M. Lebedev, and K.A. Menshikov

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Homogeneous hydrocarbon thick films ("flakes") with a high D/C ratio ~0.5, redeposited by sputtering of the first wall plasma-facing components during D-plasma discharge within a T-10 tokamak onto the vacuum vessel regions, have been studied by means of synchrotron radiation-based small- and wide-angle X-ray scattering (SAXS + WAXS) as well as neutron diffraction at scattering wave vector range $q = 0.04$–100 nm^{-1}. Previous works [1,2] used a number of techniques for studying electronic and vibration structure of such films: X-ray diffraction, optical spectroscopy and luminescence, infrared and Raman spectroscopy, X-ray absorption EXAFS and NEXAFS, thermal desorption spectroscopy, X-ray fluorescence, electron paramagnetic resonance, volt-ampere characteristics.

Experimental data are consistent with the following model of an amorphous carbon film. The film is composed of fractal aggregates with a size of 32 nm composed of primary structural units of ca. 3 nm. The primary structural units are random sets of small graphene-like islands (effectively, three fused benzene rings) located from each other at a distance of 0.6–0.7 nm and interlinked into a porous carbon network with linear C-C and C=C bonds. Unsaturated sites of the linear carbon chains are occupied by chemically bound H/D atoms (with a binding energy E_b~1.3 eV). Another type of H/D atoms occupy positions atop graphene islands forming weaker (E_b~0.65 eV) π-type bonds.

[1]. N.Yu. Svechnikov, et al., Plasma Devices and Operations, **14** (2006) 137-157.
[2] V. I. Krauz, et al. Physics – Uspekhi, **53**(10) (2011) 1015.

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Characterization of Hydrocarbon Films Deposited under Conditions of the Pulsed and Steady-State Plasma Discharges

I. Arkhipov[1], N. Klimov[2], V. Stankevich[1], V. Budaev[1], and K. Vukolov[1]

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Elements of construction of modern tokamaks manufactured from carbon materials undergoing to intensive erosion during working (pulsed) and cleaning (steady-state) plasma discharges. Codeposition of erosion products (carbon, hydrocarbon radicals) and hydrogen leads to growth of amorphous hydrocarbon films which can influence on many important aspects of tokamak working. In this work, comparative analysis of surface morphology, electronic structure and composition of such films have been carried out.

"Pulsed" hydrocarbon films were deposited under ITER-relevant conditions in plasma gun QSPA-T and tokamak T-10. Pulse duration of QSPA-T is equal to about 0.5 ms which relevant to transient events of tokamaks such as ELMs and disruption. In tokamak T-10 the films were obtained both in regime of stable working pulses with 1 s duration and regime of disruptions with thermal quench duration up to 2 ms. It should be noted that deposition rate during disruptions was in about three times more than for the films deposited at the stable working pulses.

Magnetron deuterium discharge in laboratory setup and deuterium inductive Taylor-type discharge in tokamak T-10 were used to obtain "steady-state" type of hydrocarbon films. It was shown that deposition rate of these films much smaller than that of "pulsed" films (about 0.008 and 0.5 nm/s, respectively).

Amount of hydrogen, composition and optical parameters (refractive index and thickness) were analysed by TDS, XPS, AXS and ellipsometry. Surface morphology was investigated with using optical, scanning and atomic-force microscopes.

It is known [1] that small fluctuations (the noise) of deposition flux can lead to growth of amorphouse films of irregular shape. Redeposited films with such structure are expected in the ITER. In this work, analysis of surface rough of hydrocarbon composite films obtained in tokamak T-10 and plasma gun QSPA-T has been made.

Electronic structure of hydrocarbon films was characterized with using of a ternary phase diagram in terms of the atomic fractions of sp^2 - sp^3 hybridized carbon atoms and hydrogen. Such comparison was successfully used for correction of infrared analysis data [2] and description of diamond-like amorphous carbon films as well [3]. In our case, some features of hydrocarbon films which stayed beyond the scope of this classification have been noted also.

[1] V. Budaev, et al., Physica A **382** (2007) 359
[2] W. Jacob, W. Möller, Appl. Phys. Lett. **63**(1993) 1771
[3] J. Robertson, Mater. Sci. Engineering R **37**(2002) 129

Hydrogen Isotope Exchange in DIII-D Codeposits

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The codeposition of tritium with sputtered light elements (Be and/or C) is currently foreseen as the primary contributor to in-vessel tritium inventory in ITER. In the case of carbon-based codeposits various chemistry-based means have been considered for releasing the tritium from such deposits; these include various forms of oxidation [1], but also isotope exchange. Previous attempts at isotope exchange in carbon-based codeposits; i.e., in JET following the DTE1 campaigns [2], were not particularly successful. However, these were carried out at low D_2 pressure (1– 10 Pa) and for short duration (~ 25 min).

In the current experiments, we have exposed DIII-D divertor codeposits to H_2 gas at elevated temperature and have observed the decrease in D content as a function of H_2 exposure time. Experiments were carried out at pressures of 15 – 150 Torr (2 – 20 kPa) H_2, over a temperature range of 350 – 450°C. At 450°C and 150 Torr, it was found that ~ 85% of the D was released after a few hours. Reference experiments involving only vacuum baking showed no release of D at this temperature. The temperature dependence of the process suggests a thermally activated process with an activation energy of ~ 1.1 eV (105 kJ/mol). If this value is correct, then the isotope exchange process would have a time constant of several hundred hours at 250°C.

Isotope exchange is made possible by the porous structure of carbon-based codeposits, which gives H_2 gas molecules access throughout the depth of the deposit. This is consistent with our interpretation of the oxidation of such codeposits as being a volumetric process [3].

The presentation will give an overview of the current results, and discuss the implications for ITER.

[1] J.W. Davis and A.A. Haasz, J. Nucl. Mater. 390-391 (2009) 532.
[2] P. Andrew et al. Fus Eng Design, 47 (1999) 233
[3] C. Tsui, A. A. Haasz, J. W. Davis, J. P. Coad and J. Likonen, Nuclear Fusion, 48 (2008) 035008.

Recovery Efficiency of Trapped Tritium on Tritium Balance in a D-T Fusion Reactor

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The amount of tritium bred in the breeding part of the blanket system of a fusion reactor must exceed the amount of tritium consumed in the reactor and the remainder is accumulated for the initial inventory of the next reactor to be constructed [1, 2]. In other words, the tritium breeding ratio attainable in the blanket system must be larger than the tritium breeding ratio required evaluated from the tritium balance in a power plant and tritium preparation. Recent plasma-wall interaction studies have revealed [3] that no small amount of tritium is trapped to the re-deposition layer and the bulk of the first wall materials of the plasma vessel. It is also anticipated that some amount of tritium is lost from the plasma driven permeation through the first wall when the wall temperature becomes higher as in the case of demo or power plants. Furthermore, several ways are considered to be required to control ELM [4] or to moderate the heat load to divertor in the power plant and addition of the gas or liquid is proposed for these purposes.

This paper discusses the effect of addition of tritium to the pedestal plasma region to control the ELM and the effect of the recovery efficiency of tritium trapped to the media applied to moderate the heat load on the tritium balance in a D-T power plant. Estimation in this study shows that tritium introduced to the pedestal region gives some contribution to the plasma-wall interaction phenomena though it gives no contribution in D-T burning at the core region. Accordingly, the ELM control of this way gives no preferable effect on the tritium balance.

[1] M. Nishikawa, Fusion Sci. and Technol., **57** (2010) 120.
[2] M. Nishikawa, Fusion Sci. and Technol., **59** (2011) 350.
[3] J. Roth et al., J. Nucl. Mater., **390-391** (2009) 1.
[4] H. Takenaga et al., Fusion Sci, and Technol., **57** (2010) 94.

Fuel Removal from Plasma-Facing Components:
Overview of Methods Tested on Materials from the TEXTOR Tokamak[*]

M. Rubel[1], V. Philipps[2], A. Huber[2], D. Ivanova[1], P. Petersson[1], B. Schweer[2] and M. Zlobinski[2]

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Curtailing of long-term fuel inventory in plasma-facing components (PFC) is one of the most critical and challenging issues to be resolved in order to ensure safe and economically viable operation of a reactor, especially if the use carbon (C) target plates is considered. Methods for removal of hydrogen isotopes and co-deposited layers on PFC are being developed and tested.

This paper provides an overview of results obtained with thermal, photonic, oxidative and plasma-assisted fuel removal processes studied on PFC and special probes originating from the TEXTOR tokamak: graphite and metal (steel or tungsten) substrates. Comparative systematic experiments were also carried out in laboratory plasma devices. Three aspects have been taken into account in the assessment of each approach: (i) efficiency of removal of fuel and co-deposits, (ii) effect on the surface condition of the PFC and (iii) dust formation caused by destruction and disintegration of co-deposits. To control the gas phase and surface morphology during subsequent stages of treatment a set of analysis techniques was used: e.g. thermal desorption spectrometry, optical and X-ray spectroscopy, ion beam analysis and microscopy. The main results are:

(a) Oxidative methods remove fuel and carbon but the rate strongly depends on the overall film composition; the presence of non-volatile impurities (B, Si, metals) slows the process.

(b) Nitrogen-assisted ICWC discharges do not lead to a noticeable fuel removal [1].

(c) Desorption at 623 K, i.e. at the maximum baking temperature of the ITER divertor, removes only 10-15% of hydrogenic species. Effective fuel release requires heating above 850-900 K, but baking at such conditions results in cracking and peeling-off of co-deposits thus leading to the dust formation. The same is observed under oxidation at that temperature level.

(d) Laser-induced ablation generates large amount of dust which may still contain substantial amount of fuel. The irradiation may result in substrate damage [2].

(e) Efficient local fuel release from C layers occurs with low-power laser-induced desorption.

The ongoing study shows that no single method alone would provide solution to efficient fuel removal especially from shadowed regions in a reactor. The whole activity indicates that hybrid scenario, involving mechanical methods and hovering of dust, would be needed to reduce fuel inventory. Fuel re-absorption by depleted layers is also now examined.

[1] P.Sundelin et al., J. Nucl. Mater. **390-391** (2009) 647
[2] D.Ivanova, et al., J. Nucl. Mater. **415** (2011) S801

Laser-Induced Breakdown Spectroscopy Investigations for Tokamak applications

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During Tokamak operation, hydrogen isotopes are trapped in Plasma Facing Components. For safety reasons, the total in-vessel amount of tritium is limited and has to be measured. To achieve this goal, laser-induced breakdown spectroscopy (LIBS) is a potential technique that can be considered. In order to develop quantitative measurements of both the total amount of hydrogen isotopes and the D/T isotopic ratio, the plasma produced by laser ablation of a sample from *Tore Supra* is studied.

Fast imaging is applied to investigate the expansion dynamics of the ablation plume into the surrounding argon atmosphere. The atomic and ionic plume species are observed via time- and space-resolved optical emission spectroscopy. It is shown that the laser plasma consists of two distinct regions of significantly different temperatures. The high temperature peripheral region is attributed to shockwave heating and collisional heating at the vapour/gaz interface. It contains ionic species and hydrogen. The plasma core is formed by the colder ablated material vapour. The knowledge of this temperature gradient enables to better interpret the LIBS spectra.

To overcome the need of LIBS calibration for quantitative measurements, a procedure based on the calculation of the spectral radiance of plasma in local thermal equilibrium is developed and its applicability is critically discussed. From this result and assuming laser ablation stoichiometry, the composition of a Tore Supra samples is evaluated with its accuracy. The time- and space-evolution of the H_α and D_α spectral line intensities is measured. A mass dependent segregation of plume species under low pressure argon atmosphere is observed: the relative abundance of deuterium in the plume decreases with time. In addition, the role of hydrogen diffusion into samples submitted to multi-pulse nanosecond laser irradiation is investigated by in-depth LIBS measurements of multilayered samples. The limit of nanosecond LIBS is emphasized and a backup solution proposed.

Finally, the integration of the LIBS technique in Tokamak is critically reviewed in this presentation.

Comparative Fuel Retention In JET With The ITER-like Wall And The Carbon Wall

J. P. Coad[1], S. Brezinsek[2], D. Douai[3], S. Gruenhagen[4], T. Loarer[3], G. F. Matthews[4], R. Neu[5], V. Philipps[2] and EFDA-JET Contributors*

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In a major shutdown lasting from late-2009 to mid-2011 JET was transformed from a full carbon-dominated device with all Plasma-Facing Components (PFC) made of Carbon-Fibre Composites (CFC) to a full metallic device with the ITER material choice for the DT phase, i.e. Beryllium (Be) PFCs in the main chamber and Tungsten (W) surfaces in the divertor. This so-called ITER-Like Wall (ILW) experiment at JET provides an ideal test bed in which to provide vital scientific and operational information in support of ITER. Primary goals are to demonstrate plasma compatibility with the new metallic wall and the expected reduction in fuel retention in comparison to the previous data obtained with the carbon wall.

A number of plasma scenarios have been chosen for fuel retention measurements in the ILW programme that closely match scenarios that were used with the carbon wall prior to the changeover. Gas balance measurements are made by monitoring the gas input and collecting all the gas released over a day of operations with the Advanced Gas Handling System (AGHS). This involves regenerating the cryopumps before and after the operating day: the number of pulses is limited to about 15 by the capacity of the analysis system in the AGHS of 2500 Pa.m^3 (1.2 10^{24} D). Typical L-mode discharges (0.5MW RF heating) and type III ELMy H-mode plasmas (5.0MW NBI heating) in high triangularity (δ=0.2 and 0.4) for discharges with the carbon and the metal wall configurations have been compared. Normalising to the integral divertor time, a drop in the retention rate for L-mode pulses from 1.27x10^{21}D/s to 4.8x10^{19}D/s for the ILW, and for H-mode discharges from 1.37x10^{21}D/s in the carbon configuration to 7.2x10^{19}D/s for the ILW, have been measured.

The dynamic retention increases in the limiter phase with the Be first wall in comparison with CFC, but the outgassing after the discharge has also risen in the same manner and compensates this transient retention. The overall amount of D retained in the series of Type III ELMy H-mode discharges (I_p =2 MA, B_t=2.4 T) was 2.5 10^{22} (input 1.2 10^{24} D), corresponding to a fuelling efficiency of 2.1%. High dynamic retention is clearly associated with Be, as analogous behaviour was observed when Be was first introduced into JET in 1989. Although only small amounts of Be were evaporated onto the CFC surfaces the required D fuelling increased by factors of 5-10. However, D retention per pulse was not significantly affected (being dominated by co-deposition with carbon), indicating reduction in fuelling efficiency also by a factor of 5-10.

Modeling of Hydrogen Retention and Coupled Plasma-Wall Interactions[*]

S. I. Krasheninnikov[1], J. Guterl[1], A. Yu. Pigarov[1], R. D. Smirnov[1], E. D. Marenkov[2], A. A. Pisarev[2], and I. V. Tsvetkov[2]

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The processes in the first wall play important roles in both performance and design of any fusion reactor [1]. The main focus of our research in the study of the wall physics effects was on the modeling of hydrogen retention and dynamic response of the wall to plasma bursts associated with ELMs. In this report we highlight our main results.

Hydrogen retention. Substantial amount of experimental data on hydrogen interactions with ITER wall materials (beryllium and tungsten) under various plasma and wall operational regimes has been accumulated [1]. Among the issues requiring better understanding are observed depth profiles and temperature dependence of the retained hydrogen in the wall material. We approached these issues with recently modified WallPSI [2] and newly developed FACE (First wAll simulation CodE) codes. Both are based on the reaction-diffusion models but while WallPSI is mostly suited for the simulation of the implantation layer, FACE is more focused on the prediction of hydrogen penetration into the bulk. We report on the results of the benchmarking of the codes against available experimental data on hydrogen retention in Be and penetration depth of hydrogen into bulk of material for different sample temperatures.

Dynamic response of the wall. The H-mode, which is a primary candidate for the operation of ITER, is a subject of ELM bursts accompanied by a fast degradation of H-mode plasma pedestal. It is widely assumed that the pedestal recovery is determined solely by plasma transport processes. However, the reduction of total amount of hydrogen in tokamak after the ELM crash is due to accumulation of hydorgen in the first wall. Therefore, it is plausible that for some cases the pedestal recovery can be strongly impacted by hydrogen retention. We considered a simplified model of hydrogen retention and studied the dynamics of hydrogen accumulation in the wall under periodic bursts of plasma flux. We found that for some parameter range the pedestal recovery indeed can be controlled by hydrogen outgassing processes [3].

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Molecular Dynamic Simulation of KeV Hydrogen Molecular Ions Interaction With Solids

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If parameters of protons and deuterons interaction with plasma faced materials (PFM) are well known, existing data on interaction with solids of molecular hydrogen ions are still very poor. Usually it is supposed that interaction with solids of molecular ions with kinetic energy much higher than the dissociation energy of the molecular ion can be described as interaction with the solid of molecular ions constituents at the same velocity moving independently. This approach is commonly accepted for simulation of molecular ions impact on PFM.

Nevertheless, experiments showed that the effect of molecular ions interaction with solids differs from a sum of constituents contributions. For example, bombardment of thin carbon films by 2-12 keV H^+, H_2^+, H_3^+ ions [1] has shown, that the larger an amount of ions constituents, the greater the width of energy loss spectra peaks for transmitted particles (a "molecular effect"). The same effect takes place in scattering of the keV molecular ions from surface [2]. Therefore, one can expect that such characteristics as stopping ranges, energy and angular distribution of scattered particles, defining plasma-wall interactions in fusion devices for molecular ions can deviate from data for single atoms.

We present a molecular dynamic simulation of H^+, H_2^+ transmission through thin carbon layers. The calculations were carried out using the PARCAS code [3]. The Brenner's potential [4] was used to describe all kinds of elastic C-H interactions while the classical Lindhard's model was employed to account for the electronic stopping of protons in the film [5]. Simulated energy loss spectra for H^+ penetration are in reasonable agreement with both experimental and SRIM-simulated results. We see that the widths of simulated energy loss spectra for H_2^+ are larger than those for H^+ in correspondence with experimental findings. The origin of energy spectra broadening is H_2^+ disintegration in result of which the ion fragments get the kinetic energy of order of several eV in the center of mass frame. It corresponds to energy straggling of fragments in the laboratory system of reference about of 100 eV, which finally causes the peaks broadening.

Calculations for molecular ions with lower energies more relevant to the tokamak conditions are also performed.

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Adsorpion of Hydrogen on Metal Surface [*]

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Adsorption on the surface is an intermediate process that prefaces absorption in the bulk of solids from gas and desorption from the bulk of solids into gas. The number of publications on hydrogen adsorption is extremely large, and there are many overviews on this topic [1-8] starting from that made by Tamm and Schmidt in 1969. Though, researches are very active, uncertainties and controversial conclusions still exist even in simple cases. This makes the researchers to revisit various areas with improved experimental and theoretical techniques.

Adsorption is a delicate effect, which depends on many factors and which takes place just at the interface between gas and solid in a region of a nanometer thick. Therefore, its features are investigated in refined conditions of ultrahigh vacuum, controlled impurities, flat single crystalline surfaces, and low hydrogen pressure. Any complications of the experiment can lead to a danger of loosing control over the experimental conditions and getting misleading conclusions from experimental observations.

An overview on fundamental aspects of hydrogen adsorption on the metal surface is given in this report. Potential energy diagram, features of physical and chemical adsorption, and electronic nature of adsorption are considered. Elementary processes taking part during adsorption including reflection, dissociation, diffusion, formation of hot species, desorption, and isotopic exchange are discussed. Elementary structure of the H-Me adsorption complex, formation of hydrogen super lattice, reconstruction effects, and long-range interaction are analyzed. General approach to description of the rates of adsorption and desorption as well as equilibrium isotherms are shortly touched.

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Modeling H and He interactions in W

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Tungsten is a candidate material for the divertor in fusion reactor materials. It will be subject to intense, low energy (1-100 eV) hydrogen isotope and helium bombardment from the plasma. He and H in a material can cause changes in thermal and mechanical properties, such as swelling, ductile to brittle transition temperature, bubble formation and nanofuzz formation. Fuel retention, in particular of tritium, is a serious issue. Molecular dynamics (MD) is a valuable tool to study energetics and structures of H and He clusters and many radiation damage pheonomena that happen on short time and length scales, up to nanoseconds and millions of atoms.

The quality of the inter-atomic potentials is vital to the reliability of MD results. We will briefly review available potentials, as well as ongoing development work. For small H-He-vacancy clusters literature ab initio results are available and we will show that while the overall agreement is decent there are structures with large discrepancies between MD and ab initio. MD results with current potentials tend to over-estimate binding energies compared to DFT, in particular H-H and vacancy-vacancy binding.

For larger He and He-H bubbles in W, the system is too big for electronic structure calculations, and MD can help us understand bubble formation and composition, as well as provide input to longer time scale simulations such as Monte Carlo and cluster dynamics. We have studied the formation of He bubbles ranging in size from a few atoms up to a radius of several nm, noting trends used to evaluate energetics and amount of He that can fit in a bubble. The He bubbles affect H retention. Hydrogen introduced in bubbles of different size and composition quickly diffuse towards the edge of the bubbles, but stay bound around the first atomic layer of the surrounding W matrix. Only at high temperatures, 1500 K and above, a significant amount of H escapes further than a few Å from the bubble. With increasing number of H, as the energetically favourable positions at the edge of the bubble fill up, a large H/vacancy fraction can stay inside the bubble. While most of these results are for helium and hydrogen in bulk tungsten, we will also compare with simulations with a surface.

Fine Structure of TDS Spectra of Deuterium from Tungsten[*]

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Typical TDS spectra from tungsten have two or three broad peaks. However, some features (small peaks, shoulders) are often observed that give an idea about a fine structure of the spectra. This work is devoted to analysis of a possible fine structure basing on experiments with a specific procedure of TDS measurements.

All steps of experiments (annealing, implantation, and TDS measurements) were performed in the ultrahigh vacuum installation MEDION with the base pressure of 5×10^{-9} mbar without air exposure in between. Samples were cut from 50 μm thick tungsten foil (Plansee) and annealed at the temperature of 1750 K before every implantation. These conditions provide excellent repeatability of experiments.

A number of identical implantations by the 10 keV D_3^+ ion beam up to the fluence of 10^{21} D/m^2 have been performed. Heating of the samples was done after each implantation with the constant rate of 2 K/s. After heating from RT to a given temperature T_i, the sample was cooled down back to RT, and then heated again to the maximum temperature of 1750 K. Several experiments were made with the temperature T_i varied in the range of 350-700 K with a step of 20-30 K.

Experimental spectra give clear evidence that the broad peaks that are usually observed in standard TDS runs have a fine structure and consist of several unresolved narrow peaks. Analysis of experimental curves is supported by calculations based on TMAP 7 code. Therefore, one can conclude that variety of traps with different trapping energies exist in tungsten.

Unambiguous Determination of the Binding Energies for Deuterium in Tungsten

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Thermal desorption spectroscopy is a popular method to determine not only the total amount of deuterium retained in tungsten, but also the binding energy distribution of the trapped deuterium. Commonly, experiments are performed once for a certain type of sample with a fixed temperature ramp, which is assumed to be linear. The measured desorption spectrum of mass 4, i.e., D_2 molecules, is then compared with a forward calculation performed with a diffusion-trapping model. By this method, binding energies are assigned to the individual features of the desorption spectrum (see, e.g., [1,2]).

Unfortunately, this method introduces some ambiguities. As, e.g., Venhaus and Causey demonstrated, a model with a complex binding energy distribution with a simple depth distribution can yield the same result as a model with only one binding energy and a complex depth distribution [3]. Therefore, it is necessary to include the actual depth distribution of deuterium in a given tungsten sample into the modeling of the desorption spectrum. This is by now widely recognized. Apart from this, the correlation of a certain desorption peak with a binding energy depends also on the pre-exponential factor of the detrapping rate, i.e., the detrapping attempt frequency. However, this point is usually not even addressed in publications.

There are several methods known in surface science how the binding energy and the pre-exponential factor can be determined independently. One possibility is to use several identical samples and degas them with different heating rates. Following the evaluation procedure described by Falconer and Madix [4], the binding energy can be determined from the shift of the desorption maximum for different heating rates, without any knowledge of the pre-exponential factor. Once the binding energy is known, the corresponding pre-factor can also be calculated.

We will present an adaption of this analysis method to deuterium implanted into tungsten that includes the effect of diffusion of deuterium from larger depths to the surface. We will show that this does not affect the derived binding energy, but requires forward calculations with a numerical diffusion-trapping model to correctly derive the pre-exponential factor. The model used here will be discussed as well as the results for hot-rolled, polished and stress-relieved tungsten exposed to a deuterium plasma with an ion energy of 38 eV/D, an ion flux of 10^{20} D/m^{2}s and to a fluence of 6×10^{24} D/m^2 at a temperature of 370 K.

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Thermal release mechanism of hydrogen implanted in metals by plasma exposure

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Tritium (T) release from plasma facing metals after the plasma exposure plays a critical role to determine T inventory. In order to investigate the release (desorption and permeation) mechanisms of hydrogen implanted in metals, we have profiled T depth distributions in metals by using the tritium imaging plate (TIP) technique just after the T loading by plasma exposure [1] and its sequential changes with time and temperatures under controlled surrounding atmospheres.

Metals used here were pure tungsten (W) and the Japanese reduced activation ferritic/martensitic steel (F82H). They were exposed to DC-glow discharged hydrogen plasma including tritium (T/H=1.3 x 10^{-4}) at a constant temperature ranging from 453 K to 673 K. After the exposure, they were stored in the air at room temperature (RT) for 300 days or annealed in vacuum of $10^{-5\sim6}$ Pa at 1273 K, and the change of T depth profiles with time or that induced by the annealing was monitored by TIP of which detail was given in [1].

The T depth profile in W and F82H was decomposed into following three components; two were trapped ones at the surface and in the subsurface region within several hundreds μm, and the last was dissolved one in the bulk. During the storage in the air, the dissolved component was released by normal diffusion keeping its depth profile given by a simple diffusion theory, while the trapped ones showed little changes in their profiles [1]. After annealing in vacuum at 1273 K for 2 h, most of the trapped component in the subsurface region was released, while the trapped one at the surface kept its initial profile. The three components were attributed to surface trapping at surface impurities like oxygen (oxide) and carbon (carbide), the subsurface trapping at defects introduced by plasma exposure or implanted hydrogen itself and the dissolved hydrogen in the bulk. The relative amounts of these three components were different for metals and exposed plasma conditions. This means that without knowing depth profiles, thermal desorption analysis could give wrong interpretation on desorption processes. We have tried to make a comprehensive release model for hydrogen loaded by plasma exposure at different temperatures, investigating surface and depth profiles of tritium in W and F82H given by various conditions, i.e. release (stored) temperatures and atmospheres.

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Deuterium retention and desorption from vacuum plasma splayed tungsten coating irradiated by deuterium and arbon mixed ion beam

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In a fusion reactor, tungsten coating to the blanket first wall is planned for the purpose of suppression of sputtering erosion by energytic particle bombardment. In the case carbon is used for a part of plasma facing materials such as high heat load components, the tungsten coating is siumultaneously irradiated by carbon ions, originating from these components, and tritium ions. Although it is pointed out that simultaneus injection of carbon and hydrogen ions increase hydreogen retention in tungsten coating, dependence of carbon concentration in the hydrogen ions on the characteristics of hydrogen retention and desorption from the tungsten coating are not understood yet. In the present work, in order to investigate this dependence, the tungsten coating was irradiated by deuterium and carbon mixed ion beam with a different carbon concentration.

The tungsten coating with a thickness of ~50 μm were produced by vacuum plasma splaying on a rhenium/tungsten multi-inter-layer on a carbon fiber composit substrate. Flux density and fluence of the mixed ion beam were ~10^{20} D^+/m^2s and ~1.4×10^{24} D^+/m^2, respectively. The temperature of the tungsten coating was 700 K. The carbon concentration in the mixed ion beam was changed to ~0.1, ~0.8 and ~3.2 %. After the mixed ion beam irradiation, the amount of the gas moleculs, D_2, HD and D_2O, desorbed from the tungsten coating were measured by thermal desorption spectroscopy (TDS). Deuterium depth profiles were measured by secondary ion mass spectroscopy (SIMS).

In the case the carbon concentration in the mixed ion beam was ~0.1, ~0.8, and ~2.8 %, the deuterium retention in the tungsten coating determined by TDS was ~4.2×10^{20}, ~0.4×10^{20}, and ~6.7×10^{20} D/m^2, respectively. While no clear D_2 desorption peak was found in the case of the carbon concentration of ~0.1 and ~0.8 %, D_2 desorption peaks between 550 and 1300 K were found in the case of ~2.8 %. Curve fitting of the D_2 desorption spectrum using a series of Gaussian functions indicated three desorption peaks at a temperature of ~810, ~1020 and ~1160 K. These desorption temperatures were corresponding to those observed from the carbon tiles used in the JT-60U divertor. Hence it is probable that the deuterium was retained as a form of C-D bond in the tungsten coating.

In the case the carbon concentration in the mixed ion beam were ~0.8 and ~2.8 %, the depth integral of the deuterium signal intensity measured by SIMS were ~3.4 and ~10 times higher than that in the case of ~0.1 %. This is inconsistent with the deuterium retention determined by TDS as descrived above (~0.1 and ~1.6 times, respectively, higher than that in the case of ~0.1 %). Because the present TDS measurement did not count deuterium atoms desorbed in the form of hydrocarbons, the deuterium reteniton determined by TDS was possibly underestimated. However, this discrepancy indicates that the retained deuterium in the tungsten coating is desorbed mainly in the form of hydrocarbons when small concentration of carbon ions is included in hydrogen plasmas.

Gas-Driven Hydrogen Isotope Retention in VPS Tungsten

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Measurements of gas-driven absorption are suitable to evaluate hydrogen isotope retention in relatively thick and porous materials because hydrogen isotope molecules can penetrate in deep regions through connected pores while ions and atoms cannot. In the present study, we have examined deuterium retention in vacuum plasma spray tungsten (VPS-W) after exposure to D_2 gas.

A layer of VPS-W was prepared on a plate of F82H (Japanese reduced activation ferritic/martensitic steel) substrate at around 873 K by sweeping plasma gun emitting 42 kW Ar-H plasma with melted W powder particles. The average size of powder particles was 18 μm. The thickness of the coating was about 0.6 mm. The coated plate was cut into smaller pieces and exposed to D_2 gas in a conventional vacuum device. First the specimens were heated in vacuum at 873 or 1273 K for degassing, and then exposed to D_2 gas at 10 kPa and 673–873 K for 2 hours. The retention of D was measured by thermal desorption spectroscopy (TDS).

After degassing at 873 K, relatively large amounts of D (1–2×10^{21} D m^{-2}) were retained in the VPS-W layer at the above-mentioned exposure temperatures and pressure. The desorption peaks in TDS spectra were observed at around 1100 K. By assuming that D was distributed uniformly through the coating thickness, the average D concentration in VPS-W was evaluated to be 250–500 appm. These values were larger by orders of magnitude than those expected from the solubility of hydrogen isotopes in W [1]. Such high D retention was ascribed to trapping effects of defects and/or impurities. Nevertheless, the D retention was reduced to ca. 1/100 after degassing at 1273 K. The optimization of preparation conditions and measurements of D retention are in progress. The results for advanced coatings will be reported in the presentation.

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Deuterium retention and surface modification of tungsten and tungsten-tantalum alloys as a result of high-flux deuterium plasma exposure[*]

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In this contribution we present the results of a comparative study of the interaction of high-flux deuterium plasmas with tungsten and tungsten-tantalum alloys. Exposures of the investigated grades were performed at the linear plasma generator Pilot-PSI. The maximum flux on all targets was in the range of $8*10^{23} - 1*10^{24}$ m^{-2}s^{-1}, the fluence $5*10^{25} - 10^{27}$ m^{-2}, surface temperature $460 - 510$ K, ion energy ~ 50 eV. Retention measurements were done with thermal desorption spectroscopy (TDS), surface modification was studied with scanning electron microscopy (SEM).

The comparison of deuterium retention in samples of W-5%Ta with that in pure W is presented, demonstrating that there is no systematic difference in the retention in the lower fluence range, while at higher fluences retention in W-5%Ta is consistently lower compared to pure W.

Heavy blistering was observed on the surfaces of both W and W-5%Ta. For both materials blistering patterns are strongly locally non-uniform, being closely related to the grain structure. For pure W there is an observable evolution of the blistering pattern as a function of fluence: surfaces of the samples being exposed to low deuterium fluence exhibit high density of small (up to about 100 nm) blisters while at high fluence considerably larger structures (up to about 10 μm) are formed, having much lower surface density. On the other hand, the blistering pattern on the surface of the W-5%Ta samples does not noticeably change with fluence in the studied fluence range.

We demonstrate that the desorption spectra of both W and W-5%Ta strongly change their shapes with the exposure fluence, and the dynamics of this change is different for the pure material and for the alloy. In particular, the positions of the release peaks of the spectra of both materials are close in case of low-fluence exposures, while at higher fluences they become considerably different, indicating that plasma exposure induces considerably different modification of the W-5%Ta as compared to W.

We also demonstrate that there exists an impact of the previous plasma exposures on the deuterium retention in the subsequent ones (even though the ion energy is considerably below the displacement damage threshold) which manifests itself as an increase of total retention in the samples with the history of prior exposures (which we refer to as the "history effect"). This increase of total retention is found to be present in case of both W and W-Ta. We also demonstrate that the history of prior exposures has different impact on the evolution of the release spectra D_2 and HD, indicating that it influences not only the total retention, but also the dynamics of surface processes governing the formation of molecular species.

The Effects of Co-deposited Carbon Impurities as Surface Barrier for Hydrogen Recycling in Tungsten

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Complex structure of the divertor in ITER design with carbon and tungsten surfaces on inner and outer vertical targets, beryllium walls, and presence of gases in chamber can lead to mixed materials formation and significant changes in the properties of plasma facing materials. This can significantly influence the erosion/recycling during normal operation as well as the melting/evaporation dynamics during various plasma instabilities.

Clean tungsten plasma facing materials have low solubility and high diffusion of hydrogen isotopes and will result in high recycling of hydrogen. Retention in bulk tungsten will be controlled only by defects such as intrinsic lattice imperfections and neutron induced damage. However, tungsten surfaces in the divertor area are subjected to plasma ions mixed with impurities which can lead to significant changes in hydrogen recycling, retention, and permeation behavior. Several parameters can influence these processes including particle fluence, surface temperature, impurities content and concentration. Detailed dynamic analysis is required to evaluate, for example, the influence of co-deposited carbon impurity on hydrogen retention and sub-surface blisters formation in tungsten. The dynamic self-consistant analysis is required since many processes are involved in this case such as chemical erosion, diffusion, surface molecular recombination, compounds formation, as well as processes of carbon surface segregation that also influence target erosion.

Dynamic tracking of surface evolution at nano/micro layers subjected to various plasma species was performed using Monte Carlo binary collision approximation models that include most important processes of ions/atoms interactions and time-dependent evolution of target composition. The dynamic version of ITMC (Ion Transport in Materials and Compounds), i.e., ITMC-DYN code [1] includes several implemented interatomic potentials for modeling elastic atomic collisions; combination of several models for inelastic electronic energy loss; dynamic time-dependent update of target composition; implanted atoms diffusion and mixing; molecular surface recombination and desorption; chemical erosion and surface segregation. We studied different parameters that influence hydrogen isotopes recycling and bulk penetration and compared the simulation results with recent experiments [2].

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Correlation between irradiation defects formation and deuterium retention behavior in tungsten under C^+-D_2^+ simultaneous implantation

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Combination usage of tungsten and carbon has been considered for the divertor region of plasma facing components in D-T fusion reactors. The divertor region is exposed to energetic particles, and carbon is sputtered and mixed into plasma. Tungsten is exposed to energetic hydrogen isotopes and carbon simultaneously and then W-C mixed layer is formed. It is predicted that retaintion behaviors of hydrogen isotopes in the layer will be changed dramatically by the simultaneous implantation in order to the non-thermal equilibrium reaction with energetic carbon and hydrogen isotopes. Therefore, it is important to elucidate the hydrogen isotope retention behaviors under the carbon simultaneous implantation in tungsten. In this study, phenomena of C^+-D_2^+ simultaneous implantation were compared with those of only D_2^+ and pre-C^+ implantation. The deuterium retention behaviors and the chemical states of the tungsten surface were investigated by the thermal desorption spectroscopy (TDS) and X-ray photoelectron spectroscopy (XPS), respectively. The irradiation defects in tungsten were observed by Transmission Electron Microscopy (TEM).

The disk-type W samples with 10 mm diameter and 0.5 mm thickness purchesed from Allied Material Co. Ltd. were used. After preheating at 1173 K, only D_2^+, pre-C^+ and C^+-D_2^+ simultaneous implantations were performed at R.T. The energies of D_2^+ and C^+ were 3 keV and 10 keV, respectively. The ion fluxes and fluences of D^+ were 1.0×10^{18} D^+ m^{-2} s^{-1} and $(0.03\text{-}1.8) \times 10^{22}$ D^+ m^{-2}, respectively. Those of C^+ were, 2.0×10^{17} C^+ m^{-2} s^{-1} and $(0.06\text{-}3.6) \times 10^{21}$ C^+ m^{-2}, respectively. After implantations, XPS and TDS measurements were carried out. The observation of irradiation defects was performed by TEM.

The D_2 TDS spectra for D_2^+, pre-C^+ and C^+-D_2^+ simultaneously implanted tungstens were consisted of three D_2 desorption stages around 400 K, 550 K and 650 K. These stages were attributed to the desorption stages of deuterium adsorbed on the surface, trapped by vacancies and migrated into the bulk of tungsten, respectively. Comparing to only D_2^+ implantation, the retentions of deuterium adsorbed on the surface were increased by pre-C^+ and C^+-D_2^+ simultaneous implantations, whereas those in the bulk region were decreased. The W-C mixed layer was formed on the surface region by pre-C^+ and C^+-D_2^+ simultaneous implantations. In addition, surface area was increased due to the tungsten sputtering by C^+ implantation. It was considered that deuterium trapping capacity of the surface was increased and deuterium diffusion to the bulk region was inhibited by the W-C mixed layer. The retention of deuterium trapped by vacancies was saturated in the fluence of 3.0×10^{21} D^+ m^{-2} in both cases of pre-C^+ and C^+-D_2^+ simultaneous implantations. However, the amount of deuterium trapped by the vacancies in the simultaneous implantation case was larger than that in the pre-C^+ implantation one. Therefore, the deuterium was trapped by the vacancies formed by C^+ implantation, which would prevent the recovery of vacancies. It was indicated that the amount of vacancy was increased due to development of dislocation loops in simultaneous implantation. As the result, the amounts of deuterium trapped by vacancies were increased by C^+-D_2^+ simultaneous implantation.

Deuterium Retention in Pre-Irradiated Tungsten Exposed to High-Flux Plasmas.

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In ITER, tritium retention in the tungsten divertor should be kept as low as possible for safety reasons and to prevent fuel-loss. Unfortunately, continuous bombardment with MeV neutrons degrades material properties and introduces damage into the material. In this contribution we show the effect of radiation damage and its effect on the deuterium retention in tungsten exposed to high-flux, low temperature plasmas.

Polycrystalline, annealed tungsten targets were pre-irradiated with 12.3 MeV W^{4+} ions to various damage levels as proxy for neutron damage. Deuterium (D) was implanted using high-flux plasmas in Pilot-PSI ($>10^{24}$ $m^{-2}s^{-1}$). The D retention was studied after D plasma exposure at surface temperatures below 520K. These temperatures are below the vacancy mobility threshold of about 550K, so that any effect of vacancy migration could be excluded. Under these conditions, scans in pre-irradiation damage and in D implantation fluence (up to $5 \cdot 10^{27}$ m^{-2}) were performed. Additionally, a series of pre-irradiated tungsten targets were exposed to D plasmas at surface temperatures between 600-950K to investigate the effect of vacancy migration during D implantation.

Deuterium retention was studied by nuclear reaction analysis and by thermal desorption spectrometry. We found that D retention is strongly enhanced by the tungsten bombardment. Saturation of pre-irradiation damage occurs at a W^{4+} pre-irradiation fluence of about $3 \cdot 10^{17}$ m^{-2}. At this damage level a scan in fluence of the D implantation was made by changing the time of plasma exposure. Diffusion of deuterium into the tungsten was measured, at a D fluence of $2 \cdot 10^{27}$ m^{-2} all traps created by the tungsten pre-irradiation were occupied.

Positron Annihilation Spectroscopy was used to study the behaviour of the vacancies as a function of temperature. Non-irradiated and irradiated targets showed a clear distinct Doppler broadening profile in the depth range of the ion-implantation. A significant effect of subsequent exposure of the pre-irradiated targets to deuterium plasma was observed in the first 50 nm below the surface.

ATOMIC AND LOW ENERGY DEUTERIUM INTERACTION WITH SELF-DAMAGED TUNGSTEN

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Tungsten is a reference material for the high-flux, low-ion-energy region of the divertor in ITER and a candidate plasma-facing material for DEMO. Experimental data on the influence of displacement damage on hydrogen accumulation, recycling and permeation in materials are very scarce, and the detailed theoretical understand of hydrogen transport through damaged materials is incomplete. These data are strongly required not only for ITER, but also for future fusion power reactors such as DEMO. To understand and predict tritium transport and retention in complicate irradiation conditions, interaction of atomic and low energy deuterium (D) with polycrystalline tungsten (W) is investigated in the present paper under well-defined laboratory conditions. To simulate the fast neutron damage produced in fusion reactors, W was irradiation by 20 MeV W ions. Following to the damage production by the irradiation at different dpa levels, samples were exposed to thermal atomic flux (~0.2 eV per D) in JSI and to low energy deuterium plasma (~5 eV per D) in IPP. The D retention in each sample was subsequently analyzed by various methods such as nuclear reaction analysis (NRA) for the depth profiling up to 6 μm and thermal desorption spectroscopy for the determination of total amount of retained D. The rate of deuterium decoration of radiation-induced damage was investigated by exposure at different fluences and different temperatures. The saturation concentration of D in radiation-induced traps in W is the same for both atomic and plasma deuterium exposure but the rate of deuterium decoration of traps is different. The difference between atomic and plasma exposure is discussed.

The diffusion model with dynamic trap formation during irradiation was applied to the assessment of binding energies of deuterium with natural and radiation-induced defects and density of the defects. The nature of traps produced by heavy ion implantation in W, and its influence on retention behaviour under fusion reactor conditions is discussed.

The Particle Trapping in Stainless Steel in Oxygen Contaminated Hydrogen Plasma

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The paper presents peculiarities and discusses the mechanism of particle trapping and retention in stainless steel exposed in (D_2+O_2) plasma. The experiments were performed in the thermal desorption stand (TDS-2) [1] intended for sample irradiation in gas discharge with heated cathode and thermal desorption analysis of irradiated samples.

Residual gas (H_2O – 97%, H_2 – 3%) pressure in plasma chamber did not exceed 1.5×10^{-5} Pa, working gas (D_2+O_2) pressure was $(4-7)\times10^{-1}$ Pa, and O_2 concentration in working gas varied from 0 to 6%. Irradiation ion energy changed from 20 to 700 eV, ion flux density was 2×10^{19} at/m^{2}s, irradiation dose was $1,6\times10^{23}$at/m^2 with the exception of the measurements of trapping dose dependence. Plasma chamber was preheated before plasma ignition at the temperature 570 K during 2-3 hours, and samples were annealed at the temperature 1450 K during 5 minutes. During irradiation the samples were kept at 450-500 K, temperature rump up during TDS was 5 K/s. The samples of 12X18H10T type stainless steel (SS) (0.12% C, 18% Cr, 10% Ni, ≤1% T) and of nickel were used in the experiments.

It was found that irradiation of SS walls of plasma chamber with both, deuterium atoms (which appeared in the working gas when tungsten cathode had been heated) and plasma particles initiates generation of "water" molecules (H_2O, HDO и D_2O) on the wall surface from oxygen and deuterium of residual gas and from hydrogen coming from SS.

Properties of oxidized surface layer of SS essentially depend on oxygen concentration in plasma and in turn determine the mechanisms of particle trapping and types of retained particle traps. In particular, when working gas consisted of $D_2+2\%O_2$ about 75% of hydrogen and deuterium trapping in SS under deuterium atom irradiation and (40-50)% of trapping under plasma ion irradiation were provided by the "water" molecules sorbed on SS surface through the mechanism of "potential" trapping [2].

Trapping in Ni did not depend on oxygen concentration in plasma and proceeded by "kinetic" mechanism [2].

Dependence of trapping on irradiating ion energy passed through the minimum at the ion energy ≈100 eV and then grew practically linearly up to 700 eV. Under low energy irradiation trapped particles were bound predominantly with chromium atoms within oxidized layer. When ion energy exceeded 100 eV the particles were retained mainly behind oxidized layer in the traps related to iron atoms.

Retention of deuterium in SS irradiated with 100 eV plasma ions grew linearly and approached 4.3×10^{20}at/m^2 at the irradiation dose 1.5×10^{23}at/m^2. Retention in Ni in similar conditions saturated and did not exceed 0.7×10^{20}at/m^2.

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Depth Distribution Of Deuterium In Fe₂O₃ Under Low Energy Deuterium Plasma

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Deuterium (D)-retention in Fe_2O_3 has been investigated [1], considering the possoblity of oxidation of stainless steel (SUS, the structure material of fusion device), during O_2 plasma discharge or heating in O_2 for tritium removal. It has been found that D-retention in α-Fe_2O_3 (hematite) is larger by a factor of 1.6 and ~5 than in SUS and Fe, and the diffusivity of D in the oxide is very small near room temperatute. Use of the oxide as a D-permeation barrier is of interest for fusion application. In this study, we have investigated the depth distribution of D in α-Fe_2O_3 under low energy D-plasma, and the result has been compared with the calculated range-distribution of D (TRIM code).

Samples were prepared by thermal oxidation of Fe film on SiO_2 and characterized by X-ray diffraction. The film thickness is evaluated to be ~100 nm by Rutherford back- scattering spectroscopy. Samples were exposed to D-plasma in glow discharge of D_2 gas with AC-applied voltage of 1.5 kV [1]. The depth distribution of D was evaluated by nuclear reaction analysis (NRA), $D(^3He, \alpha)H$ with 1.0 MeV $^3He^+$, α-particles being detected at 90 deg. in order to achieve a better depth resolution (estimated to be ~40 nm).

Figure 1 shows the experimental depth distribution of D in α-Fe_2O_3 (●) and compared with the calculated range-distribution (- - -), in which the fractions of D_3^+, D_2^+ and D^+ in D-plasma (60, 38 and 2 %, respectively [2]) are taken into account. It is found that D's are distributed deeper than the calculated range-distribution and the depth resolution, however, more than 90 % of D's are located in the film. This result indicates that the diffusivity of D in Fe_2O_3 is very small. D-retention is evaluated to be 50 x 10^{15} cm^{-2} by integrating the D-density and taking the ^3He beam induced desorption into account, and this value agrees with the reported value [1]. Meaurements of diffusivity at high temperature, dynamic retention of D and D-distribution in γ-Fe_2O_3 are under way.
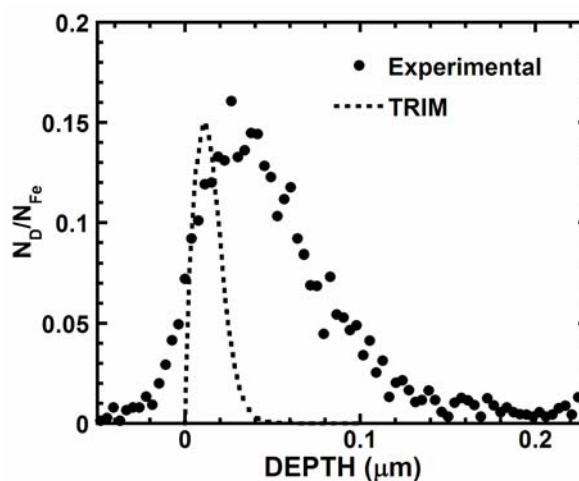

Figure 1. Depth distribution of D-density ιν α-Fe_2O_3 normalized to Fe density.

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DEUTERIUM RETENTION IN THE V-4Cr-4Ti ALLOY[1]

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V-Cr-Ti alloys are promising constructive materials for fission and fusion reactors and fusion neutron source due to their low activation by fast neutrons and compatibility with molten lithium. In the A.A. Bochvar High-technology Research Institute of Inorganic Materials (Russia) a base V-4Cr-4Ti alloy (with 4% wt. Cr and 4%wt Ti) was produced with thermomechanical properties that would allow the use of this alloy in fusion devices. However, hydrogen transport and accumulation in this alloy is expected to be a critical property due to the potential high retention of hydrogen in vanadium.

The deuterium retention and the absorption rate were measured under gas loading at different sample temperatures from room temperature to 1000 K, and various gas load pressures from $1 \cdot 10^{-2}$ to 10^4 Pa, and after low temperature plasma irradiation in the fluence range of $10^{19} - 10^{21}$ D/cm^2 at sample temperatures from room temperature to 700 K.

The retention was measured by the gas pressure drop in a volume filled with deuterium gas. The results were verified by thermodesorption spectrometry (TDS) and nuclear reaction analysis (NRA) using the D(^{3}He, p)α reaction.

The absorption rate and deuterium retention at gas loading increases exponentially with increasing the sample temperature and reaches 0.0025 l/s·cm^2 and $5 \cdot 10^{18}$ at.D/g respectively at 900 K. These results were obtained only after sample preconditioning (heating and surface irradiation with argon ions). Without this procedure, the sorption properties was much less.

The deuterium concentration after deuterium plasma irradiation was high in the material (of the order of 5 at %) and exceeded far beyond the maximum depth detectable with NRA, indicating a fast diffusion in this material.

Hydrogen Interaction With RAFM Steel Rusfer-EK-181

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Hydrogen and helium concentration in materials of the in-vessel components of the next generation fusion reactors (DEMO) can reach thousands of appm. Therefore studying of the effects of these gases on in-service properties should be considered as a primary task in selecting and characterization of structural materials for DEMO. This problem can not be solved without comprehensive study of hydrogen sorption-desorption processes.

Because of high heat and neutron loads in DEMO, ferrite-martensite (FM) steels are considered for application as structural materials These steels have better thermal properties and a lower radiation swelling in comparison to austenite steels. Reduced-activation FM steels (RAFM) were manufactured in many countries for lowering the long term radioactivity. Available FM steels are insufficiently studied not only from the viewpoint of hydrogen isotope effects on their in-service properties, but also in respect of hydrogen interactions. Data available in literature mainly relate to room temperature sorption, scatter considerably, and do not allow to choose the experimental conditions for studying hydrogen effects on mechanical properties at elevated temperature, and for a proper application of expensive "tritium trick" technique for ^{3}He buildup.

Hydrogen interactions with Russian RAFM steel RUSFER-EK-181 were studied in the present work by means of thermal desorption spectroscopy. In contrast to austenite steel 12Cr18Ni10Ti, for which two states of hydrogen were always observed after sorption at various temperatures [1], the shape of thermal degassing spectra for FM steel strongly depended on sorption temperature and heat treatment of the sample before hydrogen sorption. This implies that hydrogen sorption is very sensitive to structural changes, which occur in FM steel at heat treatment. In order to have reproducible conditions, the samples were heated to austenization temperature and then rapidly cooled before each test. The high rate of cooling provides martensite structure. After that the samples were annealed at various temperatures and exposed to hydrogen. Maximal hydrogen sorption was observed after annealing in the temperatures range 500-650 C, and was, most probably, related to formation of fine carbide and nitride precipitates. Annealing at 200 C and 720 C results in minimal hydrogen sorption. In the first case, the precipitates are not yet formed, and in the second case, an increase of the precipitate size hinders hydrogen sorption. The amount of sorbed hydrogen rapidly decreases with the increase of annealing time at 720 C, and stays almost unchanged after annealing for more than 30min.

The obtained results suggest that the attempts to improve mechanical properties by formation of fine precipitates with the usage of a hot rolling [2,3] or hydrostatic extrusion [4] can result in an increase of hydrogen sorption, and as a consequence enhance hydrogen embrittlement of FM steel.

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Deuterium Retention in Low Activation Ferritic-Martensitic Steels[*]

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Low-activation materials are necessary for the development of fusion reactors with high fluxes of neutrons (like DEMO or Fusion Neutron Source (FNS)).

A new reduced activation ferritic martensitic steel (RAFMS) EK-181 (RUSFER) has been developed recently in Bochvar's Institute, Russia. The aim of the present work is to compare the hydrogen retention properties of Rusfer and Eurofer steels after exposure to deuterium gas and low temperature deuterium plasma.

Deuterium retention in the materials was investigated by gas loading in a pressure range of 10^{-2}–10^4 Pa and in the temperature range of RT-900 K. The samples of Rusfer and Eurofer were simultaneously saturated by gas at the same conditions.

The retention was measured by nuclear reaction analysis (NRA) using the $D(^3He, p)\alpha$ reaction at several energies from 0.7 to 4.5 MeV and by thermodesorption spectrometry (TDS). NRA allows to measure the depth profile until a depth of about 15 μm.

It was shown that at the same conditions of gas exposure near-surface deuterium concentration in the Rusfer is twice higher than in samples of Eurofer. The deuterium concentration weakly depends on time of exposure of samples in the gas for both materials investigated and strongly depends on the sample temperature, significantly decreasing at temperature of exposure in gas above 400 K.

Plasma irradiation results in much higher deuterium concentrations in the near-surface layer. The materials were exposed to a 300 eV deuterium plasma in the fluence range of 10^{19} – 10^{21} D/cm^2 at sample temperatures in a range of RT-700 K. It was found that hydrogen retention strongly decreases at temperatures above 450 K and does not exceed 2×10^{15} D/cm^2. At T<400 K amount of hydrogen retained is an order of magnitude higher.

The structure changes in the materials investigated at annealing in vacuum and hydrogen atmosphere are also discussed.

Lithium coating for recycling and H/(H+D) control in EAST/HT-7

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In the recent years, lithium wall conditioning has been successfully carried out in HT-7 and EAST superconducting tokamaks.

Encouraged by some positive results of lithium coating in EAST in 2009 [1], new great progresses were obtained with an upgraded lithium coating system on EAST in 2010. Lithium coating by evaporation, associated by GDC or ICRF discharge, from two ovens, or actively coating from a NSTX type lithium dropper during plasma discharge was successfully carried out. Lithium coating in EAST seemed as a most effective method for the suppression of impurities and the reduction of hydrogen content and particle recycling, compared to ICRF/GDC cleanings, boronization($C_2B_{10}H_{12}$+He, $C_2B_{10}H_{12}$+D_2), siliconization (SiH4, SiD4) [2]. Specially, in the 2010 campaign of EAST with full carbon walls, by lithium coating, the H/ (H+D) ratio could be easily decreased to about 5%, which had successfully improved ICRF minority heating efficiency in autumn campaign of 2010. Due to reduce the H/(H+D) ratio and particle recycling, reduced power threshold of H mode, improved plasma confinement and the first H mode plasma were obtained. With increasing accumulation of deposited lithium, a few new milestones of EAST, such as 6.4s H mode, 100s long pulse plasma and 1MA plasma, were successfully achieved in autumn campaign of 2010.

Besides lithium coating, liquid lithium experiments with CPS structure and re-filling system at outside of plasma vessel have been successfully carried out in HT-7 with carbon limiters and full Mo limiters. A few of key techniques were mastered and some very positive results for recycling and H/(H+D) control were obtained. Now, a flowing liquid lithium limiter is testing in a bench and will be installed in HT-7. In the future, a flowing lithium divertor is also planned for EAST according the results from HT-7 if the PFMS changed to full W. Flowing lithium PFCs would be served as a new alternative choice for future fusions.

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Studies of Isotope Interchange on Lithium in TJ-II

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Lithium is becoming a material of high potential for Plasma Facing Components (PFC) in a Fusion Reactor [1]. The reasons for that are its low atomic number, high capability of particle and power handling, in particular in its liquid form, and its low melting point, thus opening the possibility of developing liquid PFC concepts at moderate temperatures. To date, a direct relation between the enhanced performances of Li based plasma devices and the associated low recycling of cold Li surfaces (T<400ºC) has been postulated [2]. However, hot wall operation is demanded in a fusion reactor from simple thermodynamic considerations. It is expected that D and T recycling in liquid lithium could become unity at high enough temperatures (450 ºC), so that a compromise between high recycling and low vapour pressure in the range 400-500 ºC must be achieved. At present, it is unknown whether the positive effects on plasma confinement will be lost under high recycling conditions.

In previous works [3], we have addressed the release of He and H from the lithiated walls of TJ-II in H and He plasmas respectively. Hints of diffusion limited processes and larger than expected interaction range were found. The corresponding cross sections were evaluated. In the present work, a liquid lithium limiter (LLL), based in the Capillary Porous System (CPS) has been exposed to H and D plasmas in TJ-II. Outgassing of the limter upon exposure to several plasma shots in a separated chamber has allowed for the estimation of fuel uptake and isotope exchange, while mass spectrometry in TJ-II was used for the particle balance of H/D when plasmas were produced on the solid, lithiated walls alone.

The results will be presented and discussed in terms of tritium inventory control under lithium PFC's in a fusion reactor.

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D retention in and out-gassing from BeO on Be

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Beryllium constitutes the dominant plasma-facing material in the main chamber of the next step fusion device ITER as well as in JET since the installation of the ITER-like wall. Thermal oxidation of Be surfaces and the temperature-dependent retention of hydrogen isotopes in Be oxide are an important issue for these experiments. Operational constraints and safety considerations give rise to questions about the retention of hydrogen in BeO and methods of removal.

Tritium retention in Be due to implantation saturates within the ion range [2,3] as no inward diffusion of hydrogen isotopes was observed. However, the uptake of hydrogen isotopes co-deposited with Be layers was established to dominate retention in fusion devices with a Be first wall. At present, outgassing of divertor cassettes up to 350°C is the method of choice once the tritium in-vessel limit of 700g is reached [4]. This method appears to be promising for pure Be and clean co-deposited layers, but due to surface oxidation higher temperatures are needed for hydrogen desorption [5,6]. D implanted into Be at high fluences is release in a sharp peak around 180°C, while desorption from BeO proceeds in a broad temperature distribution extending to above 700°C.

In the present investigation the desorption of implanted D from Be with a superficial oxide layer is studied. Samples were oxidised at 600 to 660°C resulting in oxide layers of different thicknesses. The samples were implanted with a fluence of $1 \cdot 10^{18}$ D/cm^2 at different energies using a mass analysed ion beam. Three different conditions were obtained:
• 100 nm BeO on Be implanted at 2 keV/D (all D deposited in BeO)
• 80 nm BeO on Be implanted at 3 keV/D (most D deposited in BeO)
• 45 nm BeO on Be implanted at 5 keV/D (more than 50% D deposited in the Be substrate)
The total retention as well as the D depth distribution were analysed by nuclear reaction analysis (NRA) using D(^{3}He,α)p. The thermal desorption was performed in situ by step-wise annealing for 20 min and analysis at room temperatue. The highest annealing temperatures were 600°C.

The total retained amount of D decreases smoothly with annealing temperature in the case of implantation into BeO. 350°C annealing temperature, as proposed for ITER, may not be sufficient for tritium recovery. For the conditions, where part of the D was deposited in the Be substrate, a sharp decrease of the reained amount of D occurs already below 200°C. Correspondingly, the depth distributions show that above 200°C D is only retained in the BeO layer. D deposited into the substrate is released preferentially. The superficial BeO layer does not act as diffusion barrier for D.

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Fundamental Hydrogen Interactions with Beryllium Surfaces [*]

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Increasingly, basic models such as density functional theory and molecular dynamics are being used to simulate different aspects of hydrogen recycling from plasma facing materials [1,2]. These models provide valuable insight into hydrogen diffusion, trapping, and recombination from surfaces, but their validation relies on knowledge of the detailed behavior of hydrogen at an atomic scale. Despite being the first wall material for ITER, basic single crystal beryllium surfaces have been studied only sparsely from an experimental standpoint. In prior cases researchers used electron spectroscopy to examine surface reconstruction or adsorption kinetics during exposure to a hydrogen atmosphere [3]. While valuable, these approaches lack the ability to directly detect the positioning of hydrogen on the surface. Low energy ion scattering (LEIS) and direct recoil spectroscopy (DRS) are two of the only experimental techniques capable of providing this information.

In this study, we applied both LEIS and DRS to examine how hydrogen binds to the Be(0001) surface. Our measurements were performed using an angle-resolved ion energy spectrometer (ARIES) to probe the surface with low energy ions (500 eV - 3 keV He[+] and Ne[+].) We were able to obtain "scattering maps" of the crystal surface [4], providing insight on how low energy ions are focused by surface atoms. Once we completed a characterization of the clean surface, we dosed the sample with atomic hydrogen using a heated tungsten capillary. A distinct signal associated with adsorbed hydrogen emerged that was strongest along open surface channels. To aid in the interpretation of the experimental results, we developed a computational model to simulate ion scattering at grazing incidence [5]. For this purpose, we incorporated a simplified surface model into the Kalypso molecular dynamics code [6]. This approach allowed us to understand how the incident ions interacted with the surface hydrogen, providing insight into the binding configuration.

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