PFMC-13 / FEMaS-1

Book of Abstracts

(vers 1.1)

Local Organization Committee

Support

Funding for this conference, in addition to the participant's conference fees, is provided by:

FEMaS-CA — Fusion Energy Materials Science (Coordination Action)
The conference and the research leading to results presented in the conference have received funding from the European Atomic Energy Community's Seventh Framework Programme (FP7 / 2007-2011) under Grant Agreement No 224752

SEVENTH FRAMEWORK PROGRAMME

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The JET ITER-like Wall - Status and experimental programme

I - 01

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The ITER reference materials have been tested in isolation in tokamaks, plasma simulators, ion beams and high heat flux test beds. However, an integrated test demonstrating both acceptable tritium retention, expected to be much lower than for a carbon wall, and an ability to operate a large high power tokamak within the limits set by these materials has not yet been carried out. The ITER-like Wall currently being installed in JET by remote handling comprises solid beryllium limiters and a combination of bulk W and W-coated CFC divertor tiles. In the main chamber an array of thermocouples is being fitted to unambiguously monitor wall temperature.

Work is also well advanced in defining the 2011/12 JET experimental programme. A phased approach will be adopted which maximises the scientific output early in the programme on the basic materials and fuel retention questions whilst minimising the risk associated with operation in an all metal machine. However, re-establishing H-modes at similar power levels to those with the carbon walls is a priority for establishing a reference database. The JET upgrades also include a considerable increase in input power, up to 35MW, this has led to a requirement that the most critical first wall Be and W components are monitored in real time by an appropriate imaging protection system. Safe expansion of operating space will also be a priority. Experiments will have to be carefully managed if they have the potential to jeopardise interpretation of the long term samples which are planned to be removed in a 2012 intervention. Here the concern is that significant mobilisation of molten material could potentially swamp the intrinsic migration due to intrinsic sputtering which is a key part of the baseline migration and fuel retention picture for ITER.

This paper will review the preparation and installation of the ITER-like Wall and give an overview of the experimental programme which is due to start in the summer of 2011. Particular emphasis will be given to the contribution of both aspects to ITER preparation.

This work, part-funded by the European Communities under the contract of Association between EURATOM/CCFE was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission. This work was also part-funded by the RCUK Energy Programme under grant EP/I501045.

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Overview of JET post-mortem results following the 2007-9 operational period, and comparisons with previous campaigns

I - 02

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During 2010 all the plasma-facing components were removed from JET so that the carbon-based surfaces could be replaced with beryllium or tungsten as part of the ITER-like Wall (ILW). This gives unprecedented opportunities for post-mortem analyses of these plasma-facing surfaces, and this paper reviews data obtained so far and relates the information to the studies of tiles removed during previous shutdowns since the JET divertor was first used in 1994.

The tiles removed during a shutdown have experienced a range of plasma conditions during their time in vessel, making the interpretation of post-mortem analyses difficult. Since 2001 supplementary information has been provide by time-resolved diagnostics such as quartz micro-balances (QMB) and rotating collectors. Furthermore, whilst net deposition can be readily assessed, it is more difficult to measure erosion, and JET has experimented with the use of various types of marker tiles since 1999 (following a preliminary test in 1989-1992). Results from the 2007-9 operational period are reported, and are generally in line with the 2005-7 data, however more positive information on erosion is coming from mechanical measurements using a tile profiler system.

There have been a number of JET divertor configurations used: Mk-1 (1994-5), Mk-IIA (1996-8), Mk-IIGB (1998-2001), Mk-IISRP (2001-4) and Mk-IIHD (2005-present time). In all configurations, there has been heavy deposition in the inner divertor channel (typically ~500g per operational period) and a tendency to erosion over much of the outer channel. However, the situation in the outer divertor is complex: plasma conditions can change the outer scrape-off region from erosion to deposition, and the amount of deposition at the outer corner of the divertor has varied in the different operational periods. The greater ion temperatures in the outer divertor affect the nature of the deposits and make the interpretation of QMB data problematic.

It was clear from the Mk-I divertor that most of the material deposited in the inner divertor comes from the main chamber, but the relative contributions from different regions were unknown. The inner wall tiles are known to be an important source, whilst the greatest local erosion occurs on certain Inner Wall Guard Limiter tiles. Tiles from many poloidal locations are being examined to enable comparisons of the erosion/deposition picture for the carbon wall with the new beryllium/tungsten ILW.

Work conducted under EFDA and partly funded by the UK EPSRC under grant EP/G003955 and the EC under the contract of Association between EURATOM and CCFE. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

[1] See the Appendix of F. Romanelli et al., Proceedings of the 23rd IAEA Fusion Energy Conference 2010, Daejeon, Korea

Lithium wall conditioning and surface dust detection on NSTX

I - 03

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Lithium coatings have been applied to the carbon PFCs in NSTX, predominantly by evaporation, although lithium pellet and powder injection have also been used. The lithiumization produced lower edge density and higher edge temperatures resulting in reductions in the plasma collisionality and the number and amplitude of edge-localized modes (ELMs) in H-modes up to the point of complete ELM suppression for periods of up to 1.2 s. The plasma stored energy increased, mostly in the electron channel, and the inductive flux consumption was reduced. The lithium concentration in the plasma remained low (about 0.1%) but during ELM-free H-modes accumulation of carbon and higher Z impurities occurred. Periodic lithiumization has also obviated the need for intershot helium glow discharge cleaning, increasing the number of discharges possible per day.

In 2010 four liquid lithium divertor (LLD) plates were installed; these form a 22 cm wide annulus in the lower divertor. The plasma facing surface is a 0.17 mm layer of molybdenum plasma-sprayed with a 45% porosity on to a 0.25 mm stainless steel liner, brazed to a 1.9 cm copper baseplate. Lithium coating is applied by a pair of lithium evaporators at the top of the vessel. The LLD was operated with temperatures from 20 to ~ 300 °C. No deleterious effects were seen even when the strike point was positioned on the LLD surface. Initial comparisons of the local D-alpha emission and the ion saturation current from Langmuir probes showed that reduced local recycling could be sustained longer by lithium on the LLD than on similarly coated graphite.

Lithium surface chemistry has been explored with a new plasma-material interface probe that exposed material samples to the plasma. XPS and TDS analysis demonstrated that binding of D atoms in graphite is fundamentally changed by lithium – in particular D atoms are weakly bonded in regions near lithium atoms, themselves bound to either oxygen or the carbon matrix. This is in contrast to the strong ionic bonding that occurs between D and pure Li.

The first real-time detection of surface dust inside a tokamak was made using an electrostatic dust detector. Dust particles impinging on a fine grid of interlocking circuit traces biased to 50 V created transient conducting paths and the resulting current pulses were counted electronically. In a separate laboratory experiment, a ‚dust conveyor’ consisting of three concentric spiral-shaped electrodes covered by a dielectric and driven by a high voltage 3-phase waveform was evaluated for potential applicability to fusion reactors. We will present data showing the optimal operating parameters for removing various dust materials.

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Fatigue Lifetime and Power Handling Capability of Actively Cooled Plasma Facing Components for ITER Divertor

I - 04

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One of the most technically challenging components of the next fusion machine like ITER is those directly facing the thermonuclear plasma. Within this framework, extensive R&D programmes have been performed in Europe to develop suitable technologies for high performing actively cooled plasma-facing components (PFCs) for ITER divertor.

The main function of the divertor is exhausting part of the plasma thermal power (including the alpha particle power), as well as minimizing the helium and impurity content in the plasma. The ITER divertor consists of 54 cassette assemblies which include one Cassette Body and three PFCs, namely the inner and outer Vertical targets, and the Dome. The inner and outer targets are the PFCs, which in their lower parts directly interact with the SOL and in their upper parts act as baffles for the neutrals. Due to high energy of the plasma particles, the heat flux received by these components is extremely intense (up to 20 MW/m^2) and requires efficient water cooling system as well as the use of dedicated materials and assembling technologies. For this purpose, high heat flux tests have been performed on different prototypical mock-ups including most recent developments, to assess the performances in terms of power handling capability and thermal fatigue lifetime for the stationary thermal loads expected in the divertor region.

In this paper, recent results are presented and discussed for various types of actively cooled small/medium-scale mock-ups with W/CFC armoured. In particular, the behaviour of mock-ups manufactured by European companies with all the main features of the ITER divertor design was investigated for thermal cycling under heat fluxes higher than 10 MW/m^2, to explore the capability to meet the present ITER requirement close to the strike point conditions in terms of heat flux performances and operational compatibility. Critical heat flux (CHF) experiments were also carried out on the components which survived the above thermal fatigue.

Main results showed promising behaviour with respect to heat flux removal capability up to 15 MW/m^2 and after a limited number of cycles at 20 MW/m^2. Beyond, slight surface erosion takes place on CFC armour material, whereas the embrittlement of W armour materials are still considered unfavourable regarding high temperature deformation and cyclic thermal fatigue. The results of CHF experiments were also rather satisfying and in line with safety margins required for ITER operation, since the tested components sustained heat fluxes in the range of 30 MW/m^2 in steady-state conditions.

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I - 05

Lifetime analysis of ITER first wall under steady state and off normal loads

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The ITER first wall (FW) is made of water cooled, beryllium clad panels. Lifetime is strongly affected by Be armour material loss during plasma operation. Material loss occurs as a result of normal, steady state erosion and off-normal, transient events. Off normal events are associated with intense heat loads, which cause significant evaporation and possibly formation of a melt layer. Material loss is the largest when the melt layer flows away.

ITER has a very large stored energy, and transient heat loads are far in excess of anything accessible in today's tokamaks. This paper addresses in detail the reduction in Be tile thickness associated with a variety of off-normal events (Edge Localised Modes, Disruptions, Vertical Displacement Events), both mitigated and unmitigated. The analysis necessitates the knowledge of the detailed spatial distribution of the energy density on the FW, including peaking due to the panel shaping, introduced into the FW design following the 2007 ITER Design Review [1,2]. The aim of the shaping is to protect leading edges during steady state operation, but necessarily leads to an increased load during transient events. The magnitude of the surface energy density results from parallel plasma heat load, fully defined in the ITER heat and nuclear load specification [3], combined with the incidence angle of the field lines to the shaped surface. As an example, an unmitigated VDE at full plasma stored energy deposits 22 MJ/m² on the beryllium surface during the thermal quench lasting 1 to 4 milliseconds.

For a given energy density, evaporated Be and melt layer thickness are calculated using thermal analysis codes. The analysis is transient, in order to take into account the actual time history of the heat load during the second preceding the thermal quench. Given the tile thickness (8 or 10 mm depending on the location), the allowable number of events is evaluated and compared to the design number. Depending on the degree of melt layer loss, the lifetime prediction spans over several orders of magnitude. Such findings are also common to lifetime evaluations based on detailed simulations of erosion during normal operation [4]. These large uncertainties in Be FW erosion estimates are a good example of the experimental nature of the ITER project and will never be truly known until ITER begins burning plasma operation.

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Detailed Design of a Solid Tungsten Divertor Row for JET in Relation to the Physics Goals

I - 06

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In the frame of the ITER-like Wall (ILW) for the JET tokamak, a divertor row made of bulk tungsten material has been developed for the position where the outer strike point is located in most of the foreseen plasma configurations. In the absence of active cooling of the component by water or helium, this represents a formidable challenge, so much the more so as such plasma-facing components are subject, by virtue of their metallic nature, to much higher electromagnetic (EM) loads than commonly encountered so far. The basic geometry of the divertor is similar to the previous one. The present contribution intends to stress the wide span of different aspects that participated to the conceptual and detailed developments, in relation to the physics goals.

As for the heat load, the solid tungsten row should withstand a steady-state power density of 7 MW/m^2 uniformly distributed over the whole surface for up to 10s. With due consideration of the required segmentation of the tiles and of shadowing effects, local loads can actually exceed 10-20 MW/m^2 so that the nominal specification can only be marginally fulfilled for some plasma scenarios with limited global wetted fraction (GWF).

The electromagnetic loads result from the rate of variation of the magnetic field strength during abnormal events, of the order of 100 T/s (considered an upper limit for standard pulses), and from the halo currents, in the order of 20 kA/module (two tiles). A high level of mechanical pre-loading is correspondingly required.

On the material side, tungsten faces temperature excursions below the ductile-to-brittle transition temperature (DBTT) and above the recrystallisation threshold (~1200°C), which calls for a drastic limitation of the tensile stresses. The possible occurrence of melting on the plasma-facing surface, in particular during transient events such as ELMs, must be taken into account.

These three issues lead to the selected solutions with respect to castellation and assembly of the tiles and to the conception of the underlying carrier, the weight of which is limited. The solutions will be presented with emphasis on the results of the thermal and electromagnetic models and their validation in electron (JUDITH-2) and ion-beam (MARION) facilities. Several operational constraints may arise that force the exploitation to be wall-driven to a large extent. Recommended scenarios encompass sweeping procedures and a power deposition limited to about 60 MJ/m^2. Significant progress in this direction was achieved in the last physics campaign of 2009 [1].

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I - 07

Development of W/Cu Divertor Components for EAST

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EAST has achieved H-mode with a marginal LHW power input around 1 MW and fully C tiles covered plasma facing surfaces. With planned rapid increase in H & CD power in the near future, current bolting structure of C tiles to heat sink will not be capable of removing expected heat flux up to 10MW/m^2. A project to realize a W/Cu divertor on EAST in several years has been launched since last year, aiming at the expected heat removal capability and the feasibility demonstration of the ITER design under practical long pulse tokamak plasmas.

Three kinds of actively-cooled W/Cu-PFCs with direct connection of W PFM to CrZrCu heat sink are under development recently at ASIPP. The first is vacuum plasma sprayed W coatings (VPS-W) on Cu heat sink. Dense W coatings of 1-2mm thick were prepared with good adhesion to substrate (~40MPa) and heat conductivity about 80W/K·m. The second is monoblock type as to be used for the ITER diverter targets. The W monoblocks are lined with a compliant Cu layer of 1-2 mm thick via HIP treatment and then the monoblocks are brazed to a CrZrCu cooling tube to form an element of the W/Cu-PFC. The third is flat type with many flat W tiles brazed directly onto the heat sink.

The EAST W/Cu divertor may consist of ITER-like monoblock targets and flat type or coating dome. The flat type and coating PFCs may also be applied on the first wall area. Once completed, together with the expected long pulse high performance plasmas, EAST will be the best testing device for the ITER divertor physics, design and engineering issues.

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Glass ceramic joined SiC/SiC: properties after nuclear irradiation

I - 08

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Joining of silicon carbide based composites for nuclear applications can be of interest for both future thermo-nuclear fusion reactors and new generation fission reactors components.

In both cases, the main issues are the extreme thermo-mechanical loads on the joined components, the not completely known service conditions and requirements, their resistance to high temperatures, to neutron irradiation and to harsh chemical environment.

Some joining techniques and joining materials for SiC/SiC will be described: results obtained by using non silica based glass ceramics, Ti-Si-C based ceramics and titanium as joining materials will be discussed. Results concerning the use of some pressure-less joining techniques will be shown: slurry and sintering, microwave assisted combustion synthesis and induction heating.

The mechanical characterization of the joints will be discussed in terms of several torsion tests, in particular on joined miniaturized hour-glass shaped specimens, designed to fit irradiation capsules.

Preliminary results from an international collaborative research on joining and mechanical test on joined SiC/SiC will be discussed.

Finally, the behaviour of glass-ceramics as joining materials for SiC/SiC will be shown before and after fast neutron irradiation.

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Recent Progress of Tungsten R&D for Fusion Application in Japan

I - 09

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Tungsten is a leading candidate for plasma facing materials and armour materials for blankets because of low sputtering erosion, high thermal conductivity, high melting point and low T retention. However there still remains important issues under fusion environment such as steady-state and pulsed fluxes of heat and particles, and neutron irradiation. Under these conditions, mechanical property change by heat, particle and neutron, and tritium retention and permeation have to be investigated. In addition, development of new tungsten materials resistant to these conditions have to be performed.

In order to tackle these issues in Japan, several researches are going forward mainly in universities. In this overview talk, present status and future plan of these on-going research projects in Japan will be presented.

Deuterium permeation studies (Osaka Univ.) have showed mixed ion irradiation (D/He and D/C) markedly changes D permeation fluxes compared with pure D cases. In general, addition of He ion decreases and addition of C increases D permeation fluxes (erosion conditions) by an order of two.

Recently, it was found high flux He ion irradiation produces unique surface structure, tungsten nano-structure (fuzz) (Nagoya Univ.). Formation conditions and its effect on initiation of unipolar arcing in high density plasma have been studied.

Understanding and control of neutron irradiation effects on tungsten are the most important issues for fusion application. Retention properties have been investigated for neutron irradiated tungsten (a part of TITAN project [J-US collaboration]). Mechanical and electrical properties of neutron irradiated tungsten and tungsten alloys (W-Re, W-Os) have been studied (Tohoku Univ.).

In order to improve embrittlement caused under fusion environment, fine grained tungsten with TiC dispersoids has been developed (Tohoku Univ.). Embrittlement caused at low temperatures and neutron irradiation have been improved and further investigation on retention and pulsed heat load effects are underway.

Development of tungsten coating technique on low activation materials is a key issue for erosion resistant first walls of blankets. VPS-W coating and W plate joining on F82H and vanadium alloys have been studied in terms of mechanical properties, hydrogen isotope behavior, and steady-state and pulsed heat load effects (mainly Kyoto Univ., NIFS, and Kyushu Univ.).

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I - 10

Tungsten Wire-Reinforced Tungsten Matrix Composites: Proof of principle

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Inherent brittleness and irradiation-induced embrittlement of tungsten impose severe restriction on its structural application for plasma-facing components [1]. In this paper, we propose an innovative toughening method for tungsten which does not require enhanced plasticity. To this end, we utilize reinforcement by tungsten wires with an engineered fiber/matrix interface. The idea is based on incremental dissipation of strain energy by controlled interfacial debonding and frictional sliding. When a propagating matrix crack meets an array of fibers standing perpendicular to the crack front, it deflects along the interfaces, provided that a specific fracture mechanical condition is fulfilled. Then the strong fibers collectively bridge the primary crack suppressing its dynamic extension leading to stress redistribution over volume. The total amount of the absorbed energy is the measure of apparent toughness. Here the fracture mechanical properties of the interface are the key factor. To achieve a maximal overall toughness fracture mechanical properties of the interface have to be optimized by coating films. In this work, we fabricated a number of single-filament model composite specimens with various kinds of interface coatings and evaluated their interfacial parameters such as shear strength and debonding energy by means of fiber push-out tests and inverse fitting to theoretical models [2,3,4]. All the measured parameters satisfied the criterion of crack deflection. The mechanism of interfacial crack deflection was also visually demonstrated by three-point bending tests in SEM and tension tests with in-situ synchrotron tomography using dedicated miniaturized specimens. Extensive microstructure analysis was made to characterize the fractographic feature. Finally, a gas-phase process for fabrication of multi-filament bulk composites was developed. The residual porosity could be kept below 95%. In-situ synchrotron tomography study of 4-point bending test clearly demonstrated that the even fully recrystallized composite specimens, which simulated embrittlement, exhibited desired interfacial debonding and controlled crack extension. The results of present study delivered a strong evidence of metallurgical as well as mechanical feasibility and thus proof-of-principle. Hence, this novel material design concept can be considered as alternative R&D line for tungsten with a potential of breakthrough.

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Cs-corrected STEM investigations of an ODS ferritic steel for fusion applications

I - 11

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Over the last years ODS ferritic steels emerged as one of the major material for fusion reactors. Despite the progress made in their technology, currently available properties still do not meet all the expectations. As a result these steels remain subject of extensive research, recently centred on the possible improvement due to their nano-metric engineering. In particular, technologies are developed for grain size refinement down to nanometres and strengthening by nano-oxides. This in turn calls for nano-scale investigations of the microstructures, which can be efficiently carried out only with the use of high resolution transmission electron microscopy and spectroscopy.
The present paper presents the results of investigations of a novel ODS steel using High Resolution Scanning Transmission Electron Microscope (STEM) equipped with Cs-correction system and EDX+EELS spectrometers. In the investigations Transmitted, TE, and Secondary Electrons, SE, were employed together with HAADF detectors and CCD camera for capturing diffraction patterns. The results acquired with such techniques like imaging in weak beam condition, Z-contrast and nano-diffraction phase analyses are presented. These results show that by application of different techniques one can obtain detailed information on the microstructure of the ODS steels, which is relevant to understanding/shaping their properties.

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Latest achievements utilizing the 3DXRD technique for non-destructive characterization of microstructures

I - 12

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The 3DXRD (Three Dimensional X-ray Diffraction) methodology for non-destructive characterization of individual grains in polycrystalline materials has been around for more than a decade. The first implementation, the 3DXRD microscope, is situated at beamline ID-11 at the European Synchrotron Radiation Facility (ESRF) developed in collaboration between Risoe and ESRF. A wide range of algorithms has been developed to facilitate new data analysis procedures. The ability to monitor the evolution in the local microstructure in materials has proven successful in studying recovery, recrystallization and grain growth as well as deformation mechanisms. The talk will give an overview of the 3DXRD software along with current and future scientific activities.

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Modelling of radiation damage in tungsten including He production

I - 13

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Tungsten is a candidate material for the divertor and for first wall armour of future thermonuclear fusion reactors (ITER and DEMO). In such irradiation conditions it is well known that the microstructure and as a result the properties of the materials will evolve. In this perspective, the fate of irradiation induced defects (helium atoms, vacancies, self interstitials and the complexes they can form) has to be understood. In particular, He migration properties are of fundamental as well as of practical interest as they affect the microstructure evolution and eventually will influence physical and mechanical properties, the most significant example being the high-temperature helium embrittlement.

In this work we show how modeling and carefully designed experimental investigations can provide a route to the understanding of the microstructure evolution of materials in these conditions. Ab initio calculations have been performed to determine the properties of each species as well as the way they interact with each other. This data-base has been used to parameterize an Object Kinetic Monte Carlo (OKMC) code LAKIMOCA which can model the evolution with time of the defect populations and their interaction with impurities. The depth profiles of the irradiation induced defects, which is an input data of the OKMC simulations, are obtained with the binary collision approximation code Marlowe.

LAKIMOCA is then used to investigate their evolution during annealing sequences or He desorption experiments.

This theoretical work is accomplished through a close collaboration with CEMHTI in charge of dedicated experimental investigations.

This work, supported by the European Communities under the contract of Association between EURATOM and CEA, was carried out within the "Fédération de Recherche sur la Fusion par Confinement Magnétique" and the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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I - 14

Irradiation Induced Defects in Alloys Examined by Positron Annihilation Spectroscopy

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Positrons are particularly suited to probe vacancy-like defects in a non-destructive way. For this reason, positron annihilation spectroscopy (PAS) is applied to study the defect distribution near the surface and in the bulk after ion or neutron irradiation. In addition, information about the defect annealing is gained by temperature dependent measurements.

Two instruments for defect spectroscopy are in routine operation at the positron beam facility NEPOMUC at FRM II which provides the world highest intensity of low-energy positrons. The coincident Doppler-broadening (CDB) spectrometer allows the investigation of defects up to a few µm with a lateral resolution of 300µm. In addition, elemental information of the vicinity of defects is gained. Positron lifetime spectra are recorded with the pulsed low-energy positron system (PLEPS) in order to provide information of defect types and its concentration [1].

After a brief review over the applied methods, experiments on irradiated Mg-based alloys and layered samples are presented [2,3]. A recent study about the annealing behavior and the defect kinetics in pure metals with high defect concentration is discussed as well [4].

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Insight into defect properties created by irradiation in Fe-C provided by multiscale modelling coupled with resistivity recovery experiments

I - 15

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It is widely acknowledged that most properties of steels are primarily controlled by the presence of carbon and its interaction with point and extended defects. Resistivity recovery experiments are very useful to investigate these mechanisms occurring at the atomic scale. Indeed, after an irradiation at low temperature, a sequence of isochronal annealing sweeps in ascending order the temperature ranges where the various migration energies and binding energies between defects come into play [1]. However, the interpretation of such experiments is not always straightforward: after a strong debate on its nature, stage III in iron was eventually ascribed to vacancy migration. This has been recently confirmed in pure iron by event-based kinetic Monte Carlo (EKMC) simulations parameterized by ab initio calculations [2].

In the present study we focus on the role of carbon on the resistivity recovery measurements. We follow the same procedure as in the case of pure iron, using ab initio calculations to determine the binding energies of clusters which are then fed into the EKMC code [2]. As the number of carbon atoms remains constant over the whole range of temperature, many events have to be computed in EKMC simulations, which considerably slows down the computations. We have thus used a cluster dynamics (CD) approach at high temperature, using as input the results given by EKMC simulations. This multiscale scheme permits us to simulate the resistivity experiments within reasonable computation times.

Our ab initio calculations reveal that the SIA-C complex is energetically stable, as well as small interstitial-carbon clusters. In addition, not only the vacancy but also small vacancy clusters are strongly trapped by carbon atoms. Using these parameters, mixed EKMC-CD computations show an overall good agreement with experiments at low and high temperature, where the interaction of carbon with interstitials and vacancies dominates respectively. The trapping of SIA and small interstitial clusters on carbon leads to the disappearance of the stage related to the migration of such clusters. A strong effect of carbon is also confirmed on the vacancy migration recovery stage, which is shifted in temperature as the carbon concentration increases.

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Hydrogen in Tungsten as Plasma-facing Material

I - 16

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Materials facing plasmas in fusion experiments and future reactors are loaded from at high fluxes (10^{20} to 10^{24} $m^{-2}s^{-1}$) of H, D, T fuel particles at energies between few eV to keV. In this respect, the evolution of the radioactive T inventory in the first wall, the permeation of T through the armour into the coolant and the thermo-mechanical stability after long term exposure are key parameters determining the applicability of a first wall material.

W exhibits fast hydrogen diffusion coefficients, but an extremely low solubility limit. Due to the fast diffusion of hydrogen and the short ion range most of the incident ions will quickly reach a surface and recycle into the plasma chamber. After plasma operation solute hydrogen will diffuse out and the remaining inventory will consist of hydrogen trapped in lattice defects, such as dislocations, grain boundaries, and irradiation induced traps. In high flux areas the hydrogen energies are too low to create displacement damage. However, under these conditions the solubility limit will be exceeded within the ion range and the formation of gas bubbles and stress induced damage occurs. In addition, simultaneous n fluxes from the nuclear fusion reaction D(T,n)α will lead to damage in the materials and produce trapping sites for diffusing hydrogen atoms throughout the bulk. The formation and diffusive filling of these different traps will determine the evolution of the T inventory.

The presentation will concentrate on experimental evidence for the influence of different trapping sites on the hydrogen inventory in W as studied in ion beam experiments and low temperature plasmas. Based on the extensive experimental data base models are applied to estimate the contribution of different trap to the influence of different traps in future fusion reactors.

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Wet-chemical methods for fabrication of tritium permeation barrier coatings

I - 17

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Mitigation of uncontrolled tritium permeation is critical issue for self-sustaining fuel supply and safety of fusion reactors. Many kinds of tritium permeation barriers and fabrication techniques have been proposed. However, it is still difficult to prepare good barrier layers on large and complicated structures. In addition, materials for barrier layers have to meet low-activation criteria if they are used under neutron irradiation. Wet-chemical methods are suitable to fabricate various types of oxide layers on metal substrates in large and complicated shapes. The coating layers, however, are prone to be porous due to evaporation of solvent and volume reduction of coatings during drying and heat treatment processes. The present authors have developed techniques to prepare dense barrier layers by combination of several different wet-chemical methods, and recent results will be reported in this paper.

Specimens used were sheets of type 430 ferritic steel. First, thin ZrO_2 layers (50 nm) were prepared by a conventional sol-gel (SG) method. As described below, this layer provided only poor barrier effects due to its porosity. Sealing of pores in SG-derived ZrO_2 layer with ZrO_2 was attempted by electrochemical deposition-pyrolysis (EDP) technique [1] and/or with Mg phosphate layer by dip coating (DC) technique [2]. The total thickness of the coating layer was 100 nm after EDP treatment and 200 nm after DC. Barrier effects against permeation were examined with H_2 and D_2 at driving pressures ranging 1.3 to 100 kPa and temperatures from 573 to 873 K. Tritium retention in the coating layers was examined by exposing the specimens to D-18%T mixture gas at 53 Pa and 573 K.

Permeation reduction factor (PRF) provided by SG-derived ZrO_2 layer was only 50. Treatments by EDP technique resulted in significant improvement in barrier effects, and PRF around 1000 was obtained. The concentration of tritium taken up in EDP-treated ZrO_2 layer was significantly higher than that in steel, and it was about 100 appm under the present conditions. Nevertheless, this amount of tritium cannot give serious influence to total tritium inventory due to small thickness of the layer. The preparation of Mg phosphate layer by DC technique also led to improved barrier effects; PRF reached 1000 also in this case. Dissolution of phosphorous into the substrate steel and consequent degradation in barrier effects were observed after long-term operation at temperatures higher than 773 K, but the coating was stable at lower temperatures. In addition, the concentration of tritium taken up in the Mg phosphate layer was about 5 ppm and much lower than that in the ZrO_2 layer.

This study has been supported in part by Grant-in-Aid for Scientific Research on Priority Areas, 476, Tritium Science and Technology for Fusion Reactor from MEXT.

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I - 18

Beryllium Qualification Activity for ITER First Wall Applications

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Beryllium is considered as an armour material for the first wall of ITER. During design activities the specific ITER requirements have been identified and the main design solutions have been developed. Currently the ITER first wall utilizes beryllium as an armour plasma facing surface. Hot Isostatic Pressing and brazing are processes for joining the beryllium tiles to the actively cooled Cu alloy substrate.

During many years of ITER activities the selection of particular beryllium grade(s) has been under study. In the ITER Final Design Report 2001 two grades have been identified: reference S-65C Vacuum Hot Pressed (VHP) from Brush Wellman and DSHG-200 from the Russian Federation. These grades have been selected based on excellent thermal fatigue and thermal shock behavior, high ductility, availability, impurity content, and available comprehensive data base. Later Chinese and Russian Domestic Agencies proposed their new grades: CN-G01 (from China) and TGP-56FW (from Russia) for application as first wall materials for ITER.

To assess the performance of these new grades the ITER organization and Chinese and Russian Parties established a program to perform the characterization of the proposed materials. This program included characterization of the production technologies, studies of main physical and mechanical properties and comparative thermal performance tests with respect to reference grade S-65C VHP.

The program for thermal performance behavior included several tests such as:

- Thermal shock resistance investigations (pre-qualification tests)
- Vertical displacement event (VDE) heat load simulation testing and following thermal shock tests
- Thermal cyclic fatigue tests after VDE simulation testing.

The summary of the main properties of the proposed beryllium grades and results of the qualification thermal tests are presented in the paper. The general conclusion is that the proposed Chinese (CN-G01) and Russian (TGP-56FW) beryllium grades can be accepted onto the ITER list of approved materials. Three grades of beryllium are now available for the application of armour for the ITER first wall, S-65C, CN-G01 and TGP-56FW.

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Deuterium Retention in, and Release from, Beryllium Resulting from High Flux Plasma Exposure

I - 19

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Retention of D in press sintered solid Be (Brush Wellman S-65C) was studied in PISCES-B with pure, and Be seeded, D_2 discharges for very high ion fluences of up to $10^{27}m^{-2}$. Target temperatures of up to 200°C and ion energies of up to 150eV were investigated. For 200°C it could be verified that the saturation values of the D content reported for lower fluences [1] is also valid for these high fluences. However, for low temperature exposure (<100°C), an increase of D retention with fluence was observed. Comparison of D retention in pure, and Be seeded, D plasmas showed the evolution of the surface morphology that is observed at these high fluences could be excluded as a reason for this increase with fluence. Although the morphology changes substantially with Be seeding the total retention as well as the release features are not influenced.

Mulitlayer codeposits were prepared in PISCES-B by collecting Be and D sputtered and/or reflected from a Be target during consecutive D_2 plasmas. Exposure was interrupted for several hours to allow an oxide layer to form on the codeposit. Thermal desorption spectroscopy revealed that this oxide interlayer does not constitute a diffusion barrier and hence is not a potential threat for fuel release in ITER. The sum of the release of the individual codeposits coincides with the release from the multi layer. Likewise there is no crosstalk between multilayers. A deuterium depleted codeposit does not take up D again, when a second codeposit is grown on top of it.

Magnetron sputtered Be films prepared in an Ar/D_2 atmosphere were used as a model system to study D release from codeposited Be films in vacuum. Release at 240°C and 350°C was studied as these are the two baking temperatures foreseen in the ITER baseline for the main wall blanket modules and the divertor, respectively [2]. In both cases longer hold times lead to further release, although the release follows a logarithmic time dependence with time constants of several hours. For the 240°C hold 2/3 of the amount that is left after the initial 0.3K/sec ramp can be released during a 24 h hold. For the 350°C hold 3/4 of the amount that is left after the ramp can be released during a 24 h hold. Nevertheless, even 350°C is not enough to release all the retained D even at this maximum hold time of 24 hours. The D concentrations after baking the films for 24 hours at 240°C and 350°C are 5 at.% and 1.2 at.%, respectively. The large time constants observed in these experiments indicate that short duration temperature excursions applied during dedicated plasma scenarios initiating transient heat loading of the wall, or in photonic cleaning methods, may have difficulty accessing deuterium trapped in these sites.

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I - 20

Analysis of Structural Changes and High-Heat-Flux Tests on pre-damaged Tungsten from Tokamak Melt Experiments

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Tungsten (W) is foreseen as plasma facing material (PFM) for the divertor area in ITER and proposed for DEMO. The main challenges of high-Z PFMs are the plasma radiation losses and the possibility of melting during transients and uncontrolled events. Melting can lead to large W influxes into the plasma associated potentially with unacceptable W accumulation, to reduced component lifetime, and can degrade irreversibly the power handling capability of the PFC due to subsequent surface irregularities. Accidents as occurred in Alcator C-Mod show clearly the limitation of damaged PFCs, disallowing H-mode with the strike points on damaged areas. For studying the power handling capabilities, sets of thermally isolated thin (2 mm) castellated W plates made from rolled W sheets with different gap width and shaping were exposed in the TEXTOR PWI test facility to multiple steady state loads (5 s) with power flux densities of ~45 MW/m^2 (~20 MW at 36˚) resulting in substantial W melting [1]. One of the exposed samples underwent additional ELM simulation in the electron beam facility JUDITH 1 (P = 1.13 GW/m^2, t = 1 ms, 100 repetitions) in order to determine the modified behaviour under ELM-like loads.

In this work, both the material structure after exposure to the edge plasma in TEXTOR and the cracking thresholds and patterns under transient thermal loads were investigated. Depending on the thermal history of the sample, irregular grain structures with grain sizes ranging from the original µm up to the mm range were observed as a result of the steady state heat loads. Furthermore, bubbles in the low µm range and larger voids up to the mm range were found at different regions of the re-solidified part of the W sample. Both have an influence on the thermal properties of the material and therefore on the material's performance during subsequent steady state loads. Due to the limited affected volume, the voids only partially influence the material's behaviour during transient thermal loads. However, the exposure of the molten and recrystallized specimens to transient ELM-like thermal loads reveals a shift of the cracking threshold towards higher base temperatures indicating an increase in the ductile-to-brittle transition temperature (DBTT) up to 400̊C . In addition, surface cracks lose their fabrication related directionality due to the existing grain structure changing towards a more isotropic state. To put these tests into relation with actually damaged PFCs damaged samples from Alcator C-mod are analyzed.The focus lies on melt layer redistribution, changes in power handling as well as material degradation such as grain growth and void formation.

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Transport and screening of ejected tungsten in controlled melt experiments at ASDEX Upgrade

I - 21

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A key issue for the use of a full tungsten (W) divertor in reactor size fusion devices is the possibility of tungsten melting by off-normal transient heat load excursions. Apart from structural damage to the plasma exposed surfaces, release of molten W into the divertor plasma may lead to an unacceptable rise of the W core concentration and corresponding degradation of plasma performance. The motion of ejected molten W droplets in the divertor and the divertor screening efficiency for correspondingly released W atoms were studied by exposure of samples with a protruding W-pin at the outer target plate of ASDEX Upgrade using the divertor manipulator system [1]. The W-pins were exposed to neutral beam heated H-mode discharges and melting at a defined time was induced by establishing plasma flat-top with the outer strike-point far below the W-pin position and then rapidly moving the strike-point towards the pin. The melt event was observed by a fast camera system with 20000fps while the W source in the divertor as well as the W-concentration in the core plasma were measured spectroscopically. Inspection of the retrieved sample revealed that the entire part of the W-pin protruding above tile surface level was molten away. The fast camera showed ejection of W-droplets into the plasma, where they moved along the divertor over toroidal distances of the order of 1m. It was found that all ejected droplets moved downstream, into the direction of the plasma flux, i.e. away from the core plasma. Penetration of W evaporated from the droplets into the confined plasma was compared to that of a main chamber W source by subsequent W laser blow off (LBO) into the same discharge and measuring the respective response of the W main plasma concentration. The W source can be estimated from the measured droplet lifetime, which itself is determined by the power balance between droplet heating due to their exposure to a parallel power flux density of ≈40 MW/m^2 in the divertor and droplet cooling, predominantly by thermal radiation and to a smaller extent by their evaporation. Using published W vapour pressure data, the upper limit for the resulting source of atomic W was estimated to≈10% of the total mass of molten droplets (≈15mg), i.e. 4×10^{18} atoms. Although this is 10 times more than the number of W atoms ablated by LBO, the corresponding increase of the W plasma concentration was only half as much as for the LBO source. This demonstrates that the divertor retains W atoms evaporated deep inside the divertor plasma fan with comparable efficiency to that for W sputtered from target tile surfaces.
To improve the prediction of the consequences of such events, e.g., for the ITER experiment, the experimental results were used to benchmark respective EMC3 simulations [2]. The computed W divertor and edge plasma screening factors are comparable to the values derived from the measured increase of the W concentration. This provides sufficient confidence in the model to advance towards predictions of W melt droplet screening under ITER conditions. It should be noted, however, that the prediction of the W-source expected from melt layers in the divertor will additionally require consideration of the melt dynamics at the material surface [3].

[1] K. Krieger et al., PSI 2010
[2] T. Lunt et al., PSI2010
[3] J. Coenen et al., PSI 2010

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I - 22

ELM-simulation experiments under ITER-like conditions

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Edge Localized Modes (ELMs) are a major concern for the lifetime of the divertor materials in ITER due to the very high localized heat energy deposition. Traditionally, plasma-surface interactions during transient events have been investigated using plasma and electron guns with a focus on the material erosion under such extreme conditions as well as the relevant damaging mechanisms (cracking, melting) [1]. In a fusion device, however, the transient heat/particle pulse associated with an ELM will interact with a plasma-exposed surface i.e. with a gas-loaded near surface and a modified morphology as a result of the high plasma fluence (blisters, 'fuzz',…) [2]. This raises potential concerns about the effect of the plasma exposure on the surface resilience to transient heat loads. In an attempt to address those questions, a new experimental setup has been developed for ELM simulation experiments with relevant steady-state plasma conditions and transient heat/particle source [3].

The setup is based on the Pilot-PSI linear device which produces plasma parameters relevant to the study of steady-state plasma-surface interactions in the ITER divertor. The plasma source has been modified to allow for transient heat and particle pulses superimposed on the steady-state plasma [3]. Peak surface heat fluxes of up to 1 GW.m^{-2} have been generated with pulse duration of about 1 ms (up to 1MJ.m^{-2}). To provide more flexibility, the shape and the duration of the pulse can be adapted to the needs. In addition, a pulsed bias system has been developed to vary the ion energy during the pulse. Importantly, the steady-state and pulsed plasma conditions can be varied independently. The pulsed plasma properties have been studied using Thomson scattering, fast visible and infrared imaging for H, He and Ar operations.

Effect of the combined steady-state/pulsed plasma on tungsten surfaces has been studied for both hydrogen and helium plasmas. Tungsten release, was observed by a fast filtered camera (WI at 400.9nm) operating at 75kHz and was observed to start at 0.2MJ.m^{-2} for hydrogen plasmas 0.3MJ.m^{-2} for helium plasmas. Once normalized to the pulse duration, these numbers are much lower than the melting threshold determined from plasma gun experiments. Significant morphology changes are also observed. Surface roughening of tungsten is observed in hydrogen at energy densities as low as 0.07MJ.m^{-2} and increases strongly with the number of pulses. The formation of helium-induced nano-structure is not affected by the transient heat loads at energy densities up to 0.2MJ.m^{-2}. Exposure of 'fuzzy' surfaces to simultaneous steady-state/pulsed plasmas (both hydrogen and helium) will also be presented.

Those results demonstrate that combined exposure to steady-state plasma and transient events might strongly affect the material resistance to ELMs.

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I - 23

Impact of disruption loads on plasma facing components

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Disruptions are already a potential risk for nowadays larger tokamaks, but are clearly becoming a major issue for ITER. Two main consequences arise from the fast loss of thermal and magnetic energy: extreme heat loads on plasma facing components and currents and high forces in the underlying structure. Heat loads arise from the loss of thermal energy – in ITER of the order of 350 MJ in less than 1 ms with predicted energy load of up to 20 MJ/m^2 – and from runaway electrons with energies that can reach a substantial fraction of the magnetic energy (in ITER about 400 MJ inside the separatrix) which are deposited very localised. Forces are either imposed by eddy currents during the fast plasma current quench and by halo currents developing during vertical displacement events.

Quantification of heat loads includes assessment of peak heat fluxes as well as their distribution in the divertor and the main chamber. Due to the strong increase in radial transport, the wetted area in the divertor increases typically by a factor varying from 3 to 10, relaxing the heat load. On the other hand, this effect can lead to substantial heat fluxes to main chamber components. At JET, for example, disruptions were analysed, where only a small fraction below 10% of the thermal energy arrives in the divertor [1], the rest is either distributed to the main chamber or radiated. In addition, it was also observed at several machines that up to 50% of the magnetic energy can be imposed on PFCs by heat fluxes. Runaways deposit their energy on small areas (a few m^2 in JET [2]). Surface temperatures of up to 700°C have been observed in JET disruption at still moderate runaway currents of 0.5 MA [1]. In low-Z material runaways can penetrate deeply, reducing the load by volumetric power distribution, but increasing the risk of damages in the underlying structure. The radiated power increases during a disruption to values exceeding the steady state values by orders of magnitude - several GW in JET – and asymmetries in the radiation distribution can also cause non-negligible heat loads on the first wall. On the other hand, the 'radiation flash' can be exploited to release fuel from first wall components. This contribution will give an overview on the present knowledge on these issues, based on the work within the EFDA Task Force PWI and the ITPA groups MHD and PWI.

Several mitigation methods are explored amongst which massive gas injection (MGI) is presently the most developed technique [3]. However, although MGI can reduce heat loads significantly, it imposes new implications for PFCs. Strong localised radiation at the injection port could cause Be melting in ITER. The injection of noble gases and/or deuterium has impact on the subsequent discharge with respect to fuel wall loading and impurity contamination. These questions are addressed using JET and TEXTOR data.

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Overview of the Second Stage in the Comprehensive Mirrors Test in JET

O - 01

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Metallic mirrors will be essential components of all optical systems for plasma diagnosis in a reactor-class device. First Mirror Test (FMT) has been carried out at JET on the request of the ITER Design Team [1,2]. Up to date two exposures have been performed in JET with carbon walls: 35 h of plasma operation (Step 1 in 2005-2007 [2]) and recently accomplished Step 2 (2008-2009): 45 h exposure with 32.7 h of X-point operation. That second test was performed with 32 mirrors made of polycrystalline molybdenum (Mo-poly) including 4 specimens coated with a 1 μm thick layer of rhodium (Rh). Mirrors were installed in carriers (cassettes with channels) placed on the outer wall and in the divertor: inner leg, outer leg and base plate under the load bearing tile. Before and after exposure mirrors underwent detailed surface analysis using optical methods, ion beam and microscopy techniques.

The aims of this work are to provide an overview of results obtained in Step 2 and, based on the outcome of the two campaigns, to discuss options for mirror maintenance and cleaning in a steady-state reactor. Essential results are summarized by several points.

Divertor: Reflectivity of all mirrors in the divertor region has been degraded by 80-90 %. It is caused by deposition of thick (> 35 μm), flaking-off coatings on surfaces. The growth of a new layer is observed in places where such thick deposit peeled-off. This leads to dust formation and large local differences in surface roughness and composition.
Outer Wall (midplane) The most important result is that only small reflectivity losses (5-10 %) occur on Mo-poly mirrors at the channel mouth. This is due to the in-situ removal of deposited species by charge exchange (CX) neutrals.
Composition: Deuterium and carbon-12 are the main elements detected on surfaces, but other isotopes are also found in some locations thus indicating differences in the material migration: (i) ^{9}Be on surfaces from the divertor base originates from the re-erosion of beryllium originally deposited on the inner vertical target; (ii) ^{13}C in the outer divertor is related to the local re-deposition of the marker gas [3].
Rh-coatings have initial reflectivity 30% better than Mo-poly. The coatings survived the test without detachment from the Mo substrate, but the implantation of CX neutrals in mirrors on the outer wall caused Rh-Mo mixing and surface texture: resultant reflectivity of pure Mo and Rh-coated mirrors was the same. It may indicate a limited use of coated mirrors in a reactor.
The deposition in channels in the divertor cassettes is pronounced at the very entrance; it sharply decreases with the distance from the plasma, λ □~ 5-7 mm.
The implications of these results for first mirrors and their maintenance in a reactor-class device with carbon components will be discussed. The preparation for the next step of FMT in JET with the ITER-Like Wall will also be presented.

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O - 02

Arcing in DIII-D as a Source of PFC Erosion and Dust Production

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Arcing may contribute significantly to plasma facing component (PFC) erosion and dust production in a tokamak [1]. In DIII-D tokamak featuring inconel vacuum vessel protected with graphite tiles on most plasma facing surfaces, arc tracks are routinely observed during the vessel inspections. Two main types of arc tracks are observed: “unmagnetized” random walk tracks and “magnetized” tracks that are mostly straight and nearly perpendicular to the local magnetic field. Unmagnetized tracks are formed during glow discharges and are mostly found on metallic surfaces near diagnostic port edges. Contribution of those tracks to PFC erosion is negligible, but they may potentially lead to deterioration of diagnostic mirrors. Magnetized tracks are produced by unipolar arcs occurring during normal plasma operation.

In DIII-D, magnetized arc tracks are found predominantly in the upper and lower divertors and on the upper outer divertor baffle tiles. Arc tracks on a number of representative graphite tiles have been analyzed by contact profilometry. The tracks are typically 3–10 μm deep, 50–200 μm wide and 5–20 mm long. By estimating an average track volume and the approximate total number of the tracks, one can estimate the total amount of carbon eroded by arcing over the total exposure time of the tiles (a few run years). An upper bound estimate gives about 1 g of carbon. This is substantially lower than the net erosion of carbon in the lower divertor alone during a single run year, estimated at about 5 g [2]. Therefore, arcing is probably not a significant contributor to the net PFC erosion in DIII-D. However, arcing can not be ruled out as a contributor to dust production, since the dust inventory on the lower divertor surfaces in DIII-D is estimated at ~1 g, and an upper bound estimate of the dust production by disruptions during a run year also gives ~1 g [3].

Erosion by arcing of a thin tungsten layer deposited on graphite was observed in Divertor Material Evaluation System (DiMES) experiments. Tungsten and vanadium stripes ~100 nm thick and 3×30 mm^2 in size deposited on a DiMES sample were exposed in the lower divertor of DIII-D near the outer strike point in two ELMing H-mode discharges for a total of ~7 plasma seconds. Upon the sample extraction, a few arc tracks resulting in substantial erosion were observed on the tungsten stripe. In contrast, no arc tracks were observed on vanadium.

This work was supported by the US Department of Energy under DE-FG02-07ER55917, DE-AC04-94AL85000, and DE-FC02-04ER54698.

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Deposition and Qualification of Tungsten Coatings Produced by Plasma Deposition in WF_6 Precursor Gas

O - 03

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Tungsten is the most promising candidate as plasma-facing component in future fusion devices and is intensively investigated in present fusion devices and lab experiments. In present devices, W coatings on graphite substrates are often used instead of bulk W tiles, due to several constraints with W bulk tiles, such as the large electromagnetic forces and the heavy weight, which are often incompatible with mechanical support structures and also the high costs of bulk W tiles.

To develop a procedure for possible in-situ coatings of plasma-facing materials with tungsten, similar to carbonisation, boronisation and siliconisation, a RF-assisted GDC in a mixture of 95% H_2 and 5% WF_6 has been used to deposit W on fine grain graphite substrate with thicknesses of up to 0.5 µm in a laboratory deposition chamber. The W-coatings have been characterised for its composition and microstructure by various post mortem analysis techniques, such as SEM, SIMS, NRA and EPMA, showing the deposition of a dense W coating with a rough surface structure but free of fluorine. The W coatings have been exposed to HHF tests in the JUDITH e-beam facility and tested for heat shock resistance by laser spot heating. No failure was observed in ELM-like loadings up to 700 MW/m^2 for 1 ms. The W coated samples have been also exposed to the plasma edge in the TEXTOR limiter lock where surface temperatures near to the melting point of W have been reached without any visible damage to the coating. The W release has been monitored spectroscopically on neutral W sputtered by imaging impurities (C, O) and compared with that of a solid W plate under identical plasma conditions. A lower sputtering yield was observed due to the difference in surface roughness.

In further experiments, WF_6 has been puffed through a hole in a Graphite plate positioned near the LCFS in TEXTOR in a number of identical discharges. The local growth of the W layer was observed by in situ W spectroscopy, plasma light and post mortem analysis. The plasma contamination by F injected with the WF6 has been monitored by visible and VUV spectroscopy before, during and after the experiments, demonstrating the disappearance of enhanced F radiation within few shots after the injection.

The experiments proof that plasma deposition of W in a H_2/W mixture is a promising in situ method to produce W coated plasma facing components of fusion devices, by which a full high Z PFC environment can be provided for a certain operational time and that contamination of the device with F remains limited.

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Neutron tomography as a new method for 3D structure analysis of CFC as plasma facing material

O - 04

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Carbon fibre composite (CFC) is used as plasma facing material for the highly heat loaded divertor components of Wendelstein 7-X. These components are based on a water-cooled heat sink structure made of copper, which is protected by CFC NB31 tiles. During operation with high hydrogen particle fluxes the CFC tiles reach surface temperatures in excess of 1000°C. The large mismatch in the coefficients of thermal expansion for CFC and Cu causes high thermally induced stresses and potentially results in failures of the interface and in morphology changes of the CFC material, respectively.

High heat flux tests up to 10.000 cycles at 10 MW/m² and overloading tests up to 32 MW/m² were performed to evaluate the expected operational limits of these components. A slight increase of the surface temperature was measured during the cycling tests. The analysis of the spatial distribution of possible structural material modifications as well as of interface failures after such loading could identify the possible causes. Neutron tomography opens up the possibility to analyse such structures on centimetre-sized samples non-destructively with a high spatial resolution. Different loaded samples were investigated at the ANTARES neutron imaging facility of the FRM II reactor within the framework of FEMaS. A contrast enhancement by vacuum infiltration with a liquid gadolinium agent resulted in a spatial resolution of about 25 µm. The distribution of internal cavities, cracks and the extent of interface delamination was visualized by neutron computed tomography.

The paper describes the method and presents the results of these examinations. The integrity of the CFC structure could be confirmed after the long term cycling tests. However, small localized interface failures were revealed and identified as the reason for the slight surface temperature increase during the loading. The detection of these small failures would have been unlikely by conventional metallographic preparation. The tomographic examination yields an important contribution for the detailed interpretation of spatially extended interface failures. Full high resolution 3D information of the plasma facing material and the interface to the heat sink can be obtained by this method.

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Nanoindentation of Self Ion Implanted Tungsten and Tungsten Alloys for Plasma Facing Applications

O - 05

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There is a great deal of interest in using tungsten alloys as critical plasma facing components in the divertor. For this to occur the mechanical properties, and how they change under the extreme conditions faced in the power plant must be understood. Of particular importance are the effects of neutron damage on mechanical properties. Neutron irradiation in a fusion reactor will cause a large amount of damage to the crystal structure and also transmutation of W to Re and Os, producing He, and hence degrade the mechanical properties of the material. Low-energy neutron irradiation of candidate materials is possible; however it can take several years to create significant levels of damage. Hot cells and remote handling facilities are then required for testing the resulting active samples.

Ion-implantation can be used to mimic the displacement damage produced by neutrons. However the depth over which the damage occurs is only a few hundred nanometres to a few microns in depth. Thus standard mechanical testing techniques are not suitable for measuring changes in mechanical properties due to implantation. Specimens of pure tungsten, W-5wt%Ta, W-1wt%Ta and W-5wt%Re have been irradiated using an ion-beam facility at the University of Surrey, UK. 2 MeV tungsten ions were used, at 500^{o}C, producing a damaged layer of depth ~200nm with doses aimed at generating damage levels of 0.5 displacement per atom (dpa),1dpa, 5dpa,and 15dpa.

High Resolution-EBSD has been used to measure the elastic strains found in the implanted layer as compared to unimplanted regions. Nanoindentation has been performed using an MTS Nano XP. An increase of ≈5% to 30% in hardness in the implanted layer is seen as the dose is increased. The depth at which this occurs over is in agreement with the depth of 200nm predicted by SRIM. The increase in hardness also varies with composition; the W5Ta alloy has a much larger increase in hardness, than the W5Re alloy which in turn has a larger increase than the pure W. As well as the increase in hardness an increase in elastic modulus is also seen. This magnitude of this increase is independent of dose and composition and is ≈15% of the unimplanted value.

Due to the poorly defined stress states around the indenter extracting parameters such as yield stress and work hardening is difficult from indentation experiments. Recently novel micro-mechanical-testing techniques have been developed based on the imaging and loading of FIB machined micro-cantilevers using a nanoindenter. Work is now ongoing using such micro-cantilevers to allow the mechanical properties of the very shallow implanted layers in tungsten alloys to be measured. This will allow yield stress and work hardening of the ion implanted damaged layer to be measured directly without influence of the underlying material. Results will be reported at the conference.

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Comparison of deuterium retention for ion-irradiated and neutron-irradiated tungsten

O - 06

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Plasma facing components research for fusion applications has recently focused on tungsten materials. One of the key issues under evaluation is tritium retention in neutron-irradiated tungsten. In our previous research, it was found by TDS (Thermal Desorption Spectroscopy) that the deuterium retention for the 0.025 dpa neutron-irradiated tungsten was about three times as large as that for un-irradiated one. The TDS spectrum was extended toward higher temperature side, indicating that deuterium was stably trapped in radiation damages. In order to elucidate hydrogen isotope trapping mechanisms in damaged tungsten, comparison of deuterium retention for ion-irradiated and neutron-irradiated tungsten is important. In the present study, iron ion was implanted into tungsten to introduce the irradiation damages and deuterium retention was compared by TDS.

The disk-type samples with 6 mm diameter and ~ 0.2 mm thickness were prepared from a rod of tungsten under stress-relieved conditions supplied by Allied Tungsten Co. Ltd.. The samples were polished mechanically and pre-heated at 1173 K for 30 minutes in vacuum to remove the surface impurities and damages induced by the polishing process. The 2.8 MeV Fe^+ was implanted into the sample to introduce the damages of 0.025 dpa or 0.3 dpa at The University of Tokyo. Thereafter, the sample was installed in TPE (Tritium Plasma Experiment) at Idaho National Laboratory and deuterium plasma irradiation was performed with a flux of ~ 1.0×10^{22} D^+ m^{-2} s^{-1} and a fluence up to of 6.0×10^{25} D^+ m^{-2}. The sample temperature during the plasma exposure was 473 K. After plasma exposure, TDS was applied with heating rate of 0.5 K s^{-1}. The surface morphology was also observed by SEM after TDS experiment.

The D_2 TDS spectra for un-irradiated tungsten showed a large desorption peaks at around 550 K. It was clearly different from that for the 0.025 dpa neutron-irradiated tungsten, which has larger desorption stages at around 700 K. In the case of ion-irradiated tungsten samples of 0.025 dpa and 0.3 dpa, the deuterium desorption behaviour was quite different from that for neutron-irradiated tungsten. The deuterium desorption spectra were extended toward lower temperature side and the deuterium desorption rate was increased as the amount of irradiation damages increased. However, it was found that the D_2 TDS spectra for ion-irradiated tungsten could not represent that for neutron-irradiated one, indicating that the deuterium trapping mechanism for neutron-irradiated tungsten has a difference from that for ion-irradiated one.

This work was performed under the Japan- US joint research project, TITAN.

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In-situ measurement of hydrogen isotope retention using ion beam analyses during plasma exposure

O - 07

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Control of hydrogen isotope retention in plasma-facing components (PFC) is essential to establish steady-state operation. In particular, tritium retention inside the vacuum vessel poses serious problems in the operation of ITER and gives a limit for the number of plasma discharges. Further, the dynamic behaviour of hydrogen retention in PFCs is of importance to investigate because it strongly influences plasma operation. However, direct measurements of the dynamic behaviour of hydrogen retention in PFC are extremely difficult in fusion devices due to the difficulty in the in-situ surface analyses.

We have developed a high density DC plasma source equipped with ion beam analysis devices including Nuclear Reaction Analysis (NRA), Rutherford Back Scattering spectroscopy (RBS), Elastic Recoil Deflection (ERD), which can analyze the surface property in-situ during the plasma exposure. The DC plasma source can produce a high density plasma up to 10^{19} m^{-3} in steady state. A ^3He beam obtained from the Van de Graff accelerator is led to the sample through the plasma in an ambient D_2 pressure of ~0.5 Pa. The retained deuterium near the sample surface was observed by the $D(^3He^+, p)\alpha$ NRA method, and protons are detected by a surface-barrier solid-state-detector(SSD) located in a high vacuum region, which is separated from the sample region by using a mylar film. A $^3He^+$ analyzing beam has the energy of ~1.0 MeV. The beam current can not be monitored at the sample during plasma exposure. Then, a rotating gold plate was equipped between the sample and the beam exit and used for a beam chopper, where the beam current was monitored by using RBS.

An irradiation experiment has been performed in the device using tungsten sample at the surface temperature of 700 K. The density of deuterium on the exposed tungsten surface was measured during and after the plasma exposure. The D-retention increased to 3×10^{16} cm^{-2} during the deuterium plasma exposure with the deuterium fluence of 1.8×10^{18} m^{-2}, and it decreased to one-third of the maximum value within four hours from the plasma irradiation. The results show that the deuterium retention changes significantly during the plasma exposure and after the exposure, representing a clear feature of a dynamic retention.

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Deposition of ^{13}C tracer in the JET MKII-HD divertor

O - 08

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** Appendix of F. Romanelli et al., Proc. 23rd IAEA Fusion Energy Conference 2010, Daejeon, Korea*

Erosion of plasma facing components and deposition on them have a great impact on the tritium retention and the lifetime of the first wall in present and future fusion devices. Post-mortem surface-analysis techniques are widely used to tackle these questions but they can typically provide only campaign-integrated data. Interpreting such results is difficult because a range of different plasma types, geometries and divertor plasma parameters have been used. In contrast, ^{13}C puffing experiments carried out at the end of a campaign are ideal for investigating migration of materials in controlled plasma conditions.

During the last experimental day of the JET experimental campaign in 2009, $^{13}CH_4$ was introduced repeatedly into series of identical plasma discharges (H-mode, I_p= 2.5 MA, B_t= 2.5 T). This way, the ^{13}C atoms can migrate under controlled conditions and the process is not mixed-up with variations of magnetic configurations. In the experiment, the inner strike point was located on the inner vertical target tile 3 and the outer strike point on the load bearing tile 5. A total of 3.3×10^{23} molecules of ^{13}C-labelled methane were puffed into the scrape-off layer through a hole in 48 outer divertor tiles 6, uniformly dispersed in the toroidal direction. After the experiment, a set of CFC divertor tiles was analysed using secondary ion mass spectrometry (SIMS) and Rutherford backscattering (RBS).

^{13}C was observed to migrate both to the inner and outer divertor, and the highest amounts of ^{13}C were found at the bottom of the outer vertical tile 7 and on the top of the outer vertical tile 8. The amount of ^{13}C on the inner vertical tiles 1 and 3 is somewhat lower. On these tiles, the highest amount of ^{13}C was found in the outer SOL close to the inner strike point at the top of tile 3. Also the inner floor tile 4 contains some ^{13}C. Heavy toroidal deposition of ^{13}C was observed near the puffing location on tile 6. The cryo pumps at the divertor were regenerated before and after the ^{13}C puffing experiment, and the gas was collected by the active gas handling system (AGHS) and stored in a 25 litre reservoir for gas chromatography (GC). GC analyses indicated that about 1/3 of the injected ^{13}C was already pumped away and can't be found by the post-mortem analysis. The global ^{13}C migration pattern observed experimentally will be compared with EDGE2D simulations and the local deposition pattern observed near the puffing location on tile 6 will be investigated using the ERO code.

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Predicting time evolution of hydrogen co-deposition in ITER based on self consistent global impurity transport modelling

O - 09

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The prediction of erosion and co-deposition processes for ITER is necessary information for the design and material choice for the first wall. In particular the formation of co-deposited layers determines the tritium retention during the nuclear phase [1]. For the ITER material choice with Be at the main wall, W at the divertor entrance and dome baffles and CFC at the strike points, it can be expected that co-deposited layers containing C, Be and W are formed. From Lab experiments scaling laws have been derived [2] to predict the hydrogen retention in co-deposited layers, but to predict the actual retention in these layers their composition and growth rate has to be determined.

Currently, predictions for co-deposited layer formation in ITER are mainly limited to the divertor area and are based on simulations which only include the erosion/deposition processes in a small plasma volume close to the divertor strike point. These simulations have to make ad-hoc assumptions about the Be impurity influx from the main chamber which results in rather large uncertainties.

This work presents predictions for the hydrogen co-deposition in ITER based on a global erosion deposition model introduced in [4]. This model yields the time and poloidal-position-resolved surface compositions, impurity influxes into the plasma, and re-deposition impurity fluxes in a self consistent way. From this information the growth rate of Be/C/W mixed layers is calculated. Using the local plasma parameters to estimate particle energies and surface temperatures [5] the scaling laws from [2] are used to predict the hydrogen retention.

These new calculations show that in contrast to previous work not only the deposition near the strike points is important for the tritium retention but also co-deposition at the dome baffle and the divertor entrance baffles plays a significant role.
This contribution presents an overview over the main features of the applied global erosion/deposition model and discusses the predicted co-deposited layer formation with respect to its impact on tritium retention in ITER.

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Modelling of Beryllium Erosion/Deposition and Local Transport at ITER First Wall Blanket Modules Using the ERO code

O - 10

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Plasma-surface interaction (PSI) leads to material erosion and, thus, limits the lifetime of plasma-facing components (PFC) in ITER. In addition, retention of tritium demands time consuming cleaning if the amount of in-vessel tritium exceeds the safety limit (~1kg). Thus, modelling of PSI is of importance to understand the underlying physics and to estimate the ITER duty cycle. Predictive modelling can be indispensible for the principal decisions like choice of first wall materials or shaping of PFCs. Previously, the ERO code was successfully applied for PSI in the ITER divertor [1]. Here, the aim is to study the PSI on the blanket modules (BM) to estimate its lifetime by net erosion, tritium retention and the erosion flux of Be.

ERO is a 3D Monte-Carlo impurity transport and PSI code. It tracks the Be atoms eroded from BM surface, their ionization and complicated trajectory in the plasma influenced by electromagnetic field in the complex 3D geometry. If tracked particles impinge the surface ERO calculates deposition/reflection and possible erosion of additional atoms. In the case at hand first modelling was carried out previously [2] with 2D MC guiding centre motion LIM code. The results (net erosion-deposition pattern) are used as benchmark for ERO.

The BMs have a special toroidal shape optimized for the power load handling [3], leading to shadowing of plasma flux coming to certain parts of the BM surface by its other parts and other BM tiles. Shadowing prevents re-erosion and absence of leading edges reduces erosion. The overall BM dimension of about 1m in each direction makes the toroidal and poloidal plasma curvature of importance. Therefore, the 3D pre-calculated plasma parameters and magnetic field must be applied. Two steady state scenarios (high and low density) are available as well as ramp phase. ERO calculations show that gross erosion (depending on yield assumptions) can reach 0.03mm/h. However about 50% or more can be re-deposited.

There are a number of important parameters for the simulations such as physical erosion yields. The details of the model concerning the incidence angle and energy of particles, local flow velocity (friction force), sheath potential, surface temperature and other factors can also have some influence. The main uncertainties and their influence on final modelling results are analysed in a view of many dedicated experiments at existing devices used for benchmarking of ERO e.g. Be sputtering at PISCES-B linear divertor simulator.

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Tungsten and carbon based PFCs erosion and eroded material deposition under ITER-like ELM and disruption loads at the plasma gun facility QSPA-T

O - 11

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This work concerns experimental study of ITER divertor PFCs erosion at the plasma gun QSPA-T (TRINITI) [1] which provides the hydrogen (or deuterium) plasma heat loads relevant to ITER ELM and disruption in the range of 0.2-5 MJ/m^2 and 0.5 ms pulse duration. The primary attention is focused on the following points: a) the experiments at large numbers of pulses (up to 500 ELM pulses and up to 20 disruption pulses); b) the dust and films which are formed after the deposition of eroded material; c) the hydrogen isotopes trapping and retention in the eroded material; d) the cracks formation on the tungsten PFCs under ELM-like plasma heat loads and followed high heat flux thermal fatigue testing.

Under ELM and disruption heat loads the CFC erosion was mainly due to PAN-fibers damage. The significant part of eroded materials deposited on the vacuum chamber. The maximum deposition rate equaled to $2\cdot10^{-2}$ μm/pulse (t_{pulse} = 0.5 ms) was observed in the downstream of plasma at the distance 30-60 cm from the target in the disruption simulation experiments (Q = 2.3 MJ/m^2). The typical deposited film density was varied from 0.5 g/cm^3 (flake-like films) to 2 g/cm^3 (solid compact films). According to the thermodesorption spectroscopy obtained by using MICMA facility (MEPHI) the typical relative concentration of hydrogen isotopes (H+D):C equaled 0.2 for the compact films (density ≥1.5 g/cm^3) and significantly exceeded the value obtained by extrapolation the data received for lower deposition rates ((0.5-2)×10^{-4} μm/s) [2]. So the mechanisms of hydrogen trapping during ELM may differ from mechanisms of hydrogen trapping between ELMs.

Cracks formation is the main mechanisms of tungsten based PFCs damage at the heat loads lower than melting threshold (Q_{melt} = 1.0 MJ/m^2, t_{pulse} = 0.5 ms). The additional process which intensified the crack formation arises due to macrobrush edge melting at heat loads higher than 0.5 MJ/m^2. The cracks which had arisen at the edges then propagated to non-melted material as a result of the following pulses. The tungsten PFCs exposed to ELM-like loads in the QSPA-T were tested afterwards on the TSEFEY electron beam facility (Efremov Institute) to high heat flux thermal fatigue testing. The testing showed significant widening (up to 10 times) of the cracks which arisen as a result of exposure in the QSPA-T.

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Thermal shock response of fine and ultra fine grained tungsten based materials

O - 12

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Among several other critical issues for the development towards a commercial fusion reactor the choice of a suitable plasma facing material seems the most challenging. Thermal loads, either applied in steady state or transient mode, combined He and hydrogen attack, and the impact of high energy neutrons causing material degradation and transmutation pose high demands on the material, particularly on its long-life stability. The most promising materials yet seem to be tungsten or tungsten based materials. Despite their good thermo-physical properties, their low sputter threshold, and the comparably low neutron activation, the thermo-mechanical properties are the weakest point of these materials. Brittleness at low temperatures and an increasing ductile to brittle transition temperature (DBTT) when irradiated with neutrons, after recrystallization and by hydrogen inventory are the main drawbacks that are addressed by various R&D initiatives.

In this work, the focus is set on the investigation of fine and ultra fine grained tungsten base materials doped with 0.5-1.1 wt% TiC that have proven to perform well in mechanical investigations [1] as well as neutron and He-studies [2]. To stretch the characterization to high thermal loads, thermal shock analyses were performed with the means of an electron beam by applying repetitive ELM like loads (n = 100) at various temperatures (RT, 100 °C, 150 °C) with a pulse duration of 1 ms and an absorbed power density of ~1 GW/m^2. By correlating the microstructural properties, i.e. grain size, TiC-particle distribution, and impurity content, with the existence or non-existence of crack formation above a certain temperature, the materials behaviour was qualified and subsequently compared to an extensively characterized reference material [3]. It is shown, that the cracking threshold is significantly reduced even down to RT with increasing TiC-content, with reduced O-content and by applying a post-manufacturing treatment of the material at 1650 °C, which changes the material's microstructure from ultra fine to fine grained.

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P01A

Mass Spectrometry Study of Destruction of Methane by a Radio Frequency Discharge

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In order to understand the kinetics of a-C:H formation in laboratory plasma RF reactors for scavenger experiments, destruction of methane was studied in a low pressure electrodeless radio frequency discharge. The plasma was ignited in pure methane at flow rates ranging from 25 sccm to 140 sccm by means of a radio frequency generator capable of delivering up to 1000 W output power. The generator was coupled to the experimental system by means of a 6 turn, 4 cm diameter coil. Depending on the output power and gas flow rate, the system was operating either in E or H mode. The composition of the discharge was analysed by means of mass spectrometry. It was found that the destruction of methane depended strongly on the discharge power, gas flow rate and operating mode. Radicals created by the discharge were found to recombine and form bigger molecules, such as ethylene and acetylene.

Investigations of Different Carbon-Like-Diamond Layers as Candidates for the Coating of PFCs

P01B

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Structure elements of the first wall of W7-X will be screwed from the plasma side to the cooled support. Recognizing the rather high temperatures caused by the expected stationary power flux of about 50 kW/m^2, TZM (99.39% Mo) was chosen as screw material. It was proposed to cover the exposed heads of the estimated 28000 screws from direct plasma contact by a Diamond-Like-Carbon (DLC) layer to avoid the erosion and influx of high-Z materials into the plasma. The layer has to stand all erosion processes (sputtering by ions and neutrals, chemical erosion, exfoliation, arcing, etc.) to act as a reliable coverage during several experimental campaigns. The layer must not absorb too much ECRH stray radiation and, additionally, has to stand the forces during the screwing. Three candidate DLC materials have been chosen and test-screws were covered by layers of ≈10µm thickness. Because of the high heat load expected (100...250kW/m^2) the layers have to withstand several hundred °C. Although the screw-heads are almost hidden from the streaming plasma itself they are target for neutrals, plasma radiation and particles, as well as for ECRH stray radiation heating and eroding the layers. Furthermore, insulating layers facing a plasma can provoke local breakdowns leading to subsequent arcing.

The investigations of the test layers were carried out in three steps: (i) characterisation of the virgin layers, (ii) loading the layers by heat and particles, (iii) observations on loaded layers. The general aim is to analyse the structure and composition, test the proposed materials for stability and integrity during heat load, sputtering and plasma exposure, and check the adhesion. Microscopy (visible light as well as electrons), ion beam sputtering, thermal desorption, thermo-balance, scratch-testing, mechanical screwing tests and metallography have been applied. Additionally, screws have been exposed to a magnetized plasma having parameters close to those expected near the wall of W7-X.

The DLC-layers tested were found to be dense, homogeneous and of the requested thickness of 10µm. The layer material was amorphous – no signs of long range order could be observed. The adhesion properties of all three layers were reasonable, getting worse after heating or particle bombardment. The virgin a-C:H layer was insulating and would force parasitic discharges in an edge-plasma environment. The two other types of layers had very low resistance (unchanged by heating). All three layer types passed the screwing tests.

For pure C, destruction and delamination of the layer at 750°C was observed.

The dominating process during exposure of the a-C layer to a magnetized plasma was found to be chemical erosion with a yield that agreed well with that from laboratory experiments. The yields for a-C:H and pure C were twice as large but showed the same temperature dependence. An estimate of the lifetimes of such layers for the W7-X environment resulted in several hundreds of working hours.

Thus, for the heads of the TZM-screws in W7-X a coating with a 10µm a-C layer is recommended.

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Deposition and Erosion of Polymer-like Hydrocarbon Layer by Hydrogen Atoms

P02A

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Carbon is due to its high reactivity with hydrogen and therefore high hydrogen retention replaced by tungsten and beryllium as a plasma facing material in current fusion devices. In a background vacuum of the order of 10^{-7} mbar the hydrocarbon impurities may still be present. We have previously studied the hydrogen atom interaction with W, Cu and a-C:H by ion beam methods ERDA and RBS. A steady increase of hydrogen concentration on the surface was observed without any saturation when the sample was exposed to hydrogen atoms at temperatures below 100^{o}C. This increase was identified as deposition of a polymer-like C:H film by incorporating hydrocarbons from the background vacuum [1]. The deposition occurs only when the hydrogen atom beam is on. On the other hand such film is easily removed again by hydrogen atoms but at temperatures above 400 K [2]. Such hydrocarbon deposition could represent an additional problem concerning hydrogen retention, especially in the remote parts of the tokamak, where hydrogen atoms are present.
Here we present results of a further study of this impurity deposition and erosion on Cu, Si and W by ERDA and RBS. A clear transition from hydrocarbon deposition to erosion was observed during exposure of the copper sample to D beam during slow sample heating. Observed transition temperature is around 220^{o}C. Measurement also indicates difference of the deposition mechanism at the room temperature and at the temperature around 100^{o}C. At lower temperature layer growth is accompanied by both [H] and [D] increase while at higher one only [D] continues to increase. This indicates temperature dependence of isotope exchange in the growing film, which is mainly done by incorporating H-containing hydrocarbons from the background vacuum. The carbon concentration is determined from the shift of the Cu-edge in the RBS spectra, which is due to the energy loss of the probing beam in the deposited layer. The presence of hydrogen atom beam clearly induces layer deposition in our system. On the other hand it has been established that hydrogen atoms enhance hydrocarbon deposition in the presence of hydrocarbon radicals [3]. The possible origins of such radicals in our vacuum chamber will be discussed. From the RBS measurement we have also observed W deposition on Cu and Si, which originates from the hot tungsten capillary of hydrogen atom beam source. The deposition rates of both deposits will be presented and discussed. The background vacuum with hydrogen atom-beam on and off was also analyzed by the mass spectrometer.

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Studies of Nanostructures Formed in T-10 Tokamak Using Synchrotron Radiation and Neutrons

P02B

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Hydrocarbon films and flakes are formed under deuterium plasma discharges in T-10 tokamak. Homogenous 20-30 μm thick films, redeposited inside the vacuum vessel far from graphite plasma facing components, may have atomic ratio D/C up to 0.9 and higher. The properties of such films were studied with application of small-angle X-ray scattering using synchrotron radiation, wide-angle X-ray scattering measurements, neutron diffraction and other techniques. According to the X-ray diffraction (XRD) studies, the overall structural pattern of the films resembles the pattern for an amorphous solid, with graphene-like sheets composed of aromatic rings oriented mainly parallel to the film surface. The XRD peak positions showed the presence of structural defects with interplane distances of 0.12, 0.24 and 0.66 nm. The peak widths gave the in-plane sizes of the scattering structures equal to about 1 nm. Comparison of X-ray and neutron diffractograms suggests that the film structure is strongly disturbed by broken bonds of carbon network and by nanopores filled with atomic deuterium. The combination (Raman) scattering studies did not reveal presence of molecular deuterium and protium in the films. The experiments performed using near-edge X-ray absorption fine structure (NEXAFS) spectroscopy have shown that the films contain about 63% of sp^3 and ~37% of sp^2 states. The films display the properties of a wide-band semiconductor with a gap of about 3 eV. X-ray fluorescence spectroscopy employing synchrotron radiation revealed that the films contain at least 12 impurities of Fe, Mo, Cr, Ni, Nb and other transition metals. The studies using electron paramagnetic resonance spectroscopy revealed defects with unpaired spins that refer to unpaired π-bonds in Csp^2-nano-clusters which size is ~4 nm and spin orientation is non-isotropic.

Difference between film properties on its opposite sides was revealed using Fourier-transform infrared spectroscopy and analysis of current-voltage characteristics (CVC). On the wall facing side of the film, aromatic rings Csp^2 dominate, carbon network is distorted. Amount of metallic impurities on this side is higher and concentration of hydrogen isotopes, hydroxyls and C=O groups is smaller than on the wall facing side. CVC is of semiconductor type with resistivity $\rho = 10^5$-10^7 Ω·cm. On the plasma facing side, diamond-like Csp^3 structures prevail, $\rho = 10^8$-10^9 Ω·cm, while CVC is quasi-ohmic. Different types of CVC hysteresis are observed on the opposite sides of the films. They are caused by different types of charge traps, i.e. structural defects serving as centers for hydrogen isotope and hydrocarbon adsorption. Difference in properties of opposite film sides, apparently, is determined by the process of film formation under discharges.

Deuterium retention can be monitored by two groups of vibrational sp^3 modes with different oscillator strengths, depending on the amount of deuterium in films.

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The Structure and Gas Trapping Properties of Plasma Deposited Carbon and Carbon-Tungsten Films

P03A

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The report presents investigation of both formation of the carbon (C) and carbon-tungsten (C-W) films deposited in plasma and trapping of gases in the films.
The experiments were carried out in the gas discharge described in [1]. Residual gas pressure did not exceed P = $4{\cdot}10^{-4}$ Pa. Argon (P_{Ar} = 0.1 Pa) and deuterium (P_D varied from $2.7{\cdot}10^{-4}$ to $1.3{\cdot}10^{-2}$ Pa.) were used as the working gases. Atoms sputtered from W and C targets were deposited on the substrates made from fine grain graphite, stainless steel (SS) and tungsten. The substrates were kept under floating potential, at the temperature 350ºC. C and W atoms were deposited with overall rates $1.1{\cdot}10^{19}$ and $2.8{\cdot}10^{19}$ at/m^2s correspondingly. The average C/W ratio in C-W films was≈2.5. The deposited films were examined by SEM and gas retention in the films was analyzed by TDS in the stand presented in [2].
The C films turned out to be uniform and flat independently of the substrate. C-W films on SS looked similarly until their thicknesses approached 0.4 µm. Then the films started cracking, and exfoliation began when their thickness reached 0.75 µm.
0.5-0.7 µm-thick C-W films on the C and W substrates had columnar structure. Columns with cross section≈3 µm ("big" columns) consisted of the layers. The upper carved layers formed flopping hills on the film surface. Later new type of columns - "small" ones (cross section≈0.5 µm) appeared. They grow mainly around big columns forming subsequently original walls of columns. C/W ratio in the big columns was higher than that in the small ones. The fragments of C films lost their contact with the substrate when their thickness approached 1 µm. The process of exfoliation was not registered, even when film thickness exceeded 2.5 µm. At the same time the 2 µm films on W substrate were intensively destructed.
Deuterium- and hydrogen concentrations (D/C and H/C) in the C films did not depend remarkably on the type of substrate. D/C increased (from 0.001 to 0.01) and H/C decreased (from 0.04 to 0.02), when P_D grow.
H/C ratios in C-W films were higher than D/C ratios as well, but contrary to C films both ratios increased along with P_D. It is interesting to mention that D/C ratio of C-W films was some times smaller than D/C ratio of C films and decreased from C substrate to W substrate.
Experimental results are analyzed. Peculiarities of deposited film growth and influence of the substrate on its formation and exfoliation are discussed. Mechanisms of hydrogen trapping in the plasma deposited films are considered. In particular, it is shown that the water molecules of residual gas sorbed on the film surface are the main source of hydrogen isotopes for trapping in the deposited films.

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Classification of hydrocarbon films obtained in tokamak T-10 under controlled conditions

P03B

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Transport and redeposition of carbon in modern tokamaks result in formation of amorphous hydrocarbon (a-C:H/D) films in remote areas of vacuum chamber. Appearance of the films can lead to degradation of plasma-viewing elements that are used by a majority of plasma optical diagnostics. Therefore study and classification of a-C:H/D films obtained in different tokamaks and other plasma facilities is an actual scientific and practical task.

Properties of a-C:H/D films deposited in tokamaks depend on such factors as temperature, electromagnetic irradiation, intensive fluxes of high-energy ions and neutrals as well as active gases and metallic impurities. Up to now hydrocarbon deposits obtained at uncontrolled conditions mainly in form of flakes and dust were studied in tokamak T-10 [1]. In this work a-C:H/D films were obtained under strictly controlled conditions with using specially constructed diagnostic inserts placed both close and far from graphite limiter and carbon ring diaphragm of tokamak T-10. The films deposited in regime of working pulses (~1 s) and vacuum conditioning were studied. Samples (silicon wafers and metallic mirrors from Mo and stainless steel) were oriented in radial, toroidal and poloidal directions. Experimental data were obtained mainly in the course of campaigns of 2009 - 2010.

Composition, structure, surface morphology and optical properties of a-C:H films were studied by EPMA, IR spectroscopy, ellipsometry, spectrophotometry, SEM and AFM. Deposition and removal rates of obtained films were estimated as well. Studied films were classified and compared with films obtained in other tokamaks and experimental setups. Conclusions about transport and redeposition of carbon in tokamak T-10 were made.

One of the main peculiarities of tokamak T-10 is high fluxes of impurities (including, hydrocarbon radicals) on the first wall surface. Values of these fluxes can by several times exceed those of other present tokamaks. Thus data on carbon transport and a-C:H/D film properties obtained in T-10 can be very useful for impurities control and preventing of first mirrors degradation in ITER.

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Characteristics of co-deposited carbon layers on tungsten nano-structure

P04A

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C and W are plasma facing materials for 1st set of divertor of ITER. C has good thermal and mechanical properties, and low atomic number. However, C is easily sputtered and forms co-deposition layers with T. Therefore, studies on erosion and re-deposition behaviour of C are important for safety in ITER.

Previous reports indicated that W nanostructure (fuzz) produced by high flux He plasma exposure [1] enhanced C deposition, which was observed in TEXTOR tokamak [2]. However, there have been no systematic studies for deposition conditions, structure of deposition layers, and behaviour of hydrogen isotopes by precisely controlled experiments.

In this study, C deposition characteristics on W fuzz by different deposition methods are compared. Structure of C deposition layers was observed by Raman spectroscopy and SEM, and the amount of D retention by NRA and TDS. W fuzz samples were prepared by exposure to high density He plasma. Initial thickness of fuzz was about ~500 nm. Two types of carbon deposition experiments have been done; (1) irradiation by mixed D and C ion beam, (2) deposition by magnetron sputtering. Our purpose is to investigate the effects of fuzz on the structure and formation condition of C deposition layers.

Ion beam irradiation experiments have been done by the high flux mixed ion beam device HiFIT with the C concentration up to about 3%. For the samples irradiated with mixed D and C ion beam (C~1.0%, 450 °C, ion energy~150eV), most of fuzz was sputtered, and very slight C deposition was observed. For the samples irradiated with higher C concentration ion beam (C ~ 1.8%), C deposition layer on the top of the fuzz was observed though some of initial fuzz was eroded. On the other hand, no C deposition was observed on flat W samples under these conditions. This indicates that C deposition takes place preferentially on fuzz surfaces. Similar results were observed on the fuzz samples exposed to TEXTOR edge plasmas [2].

Carbon deposition characteristics by the magnetron sputtering method (D:He=1:1, ~100°C) showed completely different. No clear C deposition layer was observed on fuzz surfaces though C deposition layer was observed on flat surfaces by the magnetron sputtering. Although the reason for the difference in C deposition characteristics on fuzz surfaces between the abovementioned deposition methods is not clearly understood, this indicates that effects of fuzz on C deposition strongly depend on deposition conditions.

In this presentation, we will show the comparison of formation conditions, the structural properties of deposition layers by Raman spectroscopy, and retention properties by NRA and TDS between the different deposition methods.

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Spectroscopic study of plasma produced from CFC targets irradiated by pulsed plasma streams

P04B

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The paper reports on recent research on behavior of carbon fiber composite (CFC) samples exposed to intense pulsed plasma streams, which were generated by a PF-360 plasma accelerator operated at the IPJ in Swierk, Poland. The facility was equipped with two coaxial electrodes of about 300 mm in length, and of 170 mm and 120 mm in diameter, respectively. The inner electrode (anode) was embraced with a ceramic insulator of 80mm in length. Discharges were initiated at the initial deuterium filling up to p_0 = 6 hPa and they were supplied from a condenser bank of 234 μF charged initially to 30 kV/105 kJ. The maximum discharge current amounted to about 1.8 MA, and the pulsed plasma streams were emitted mainly during a characteristic current dip (peculiarity) observed about 5 μs after the discharge initiation.

Three investigated samples were cut of a piece of CFC of the Snecma N11 type (used in the Tore-Supra tokamak). The surface of the irradiated samples was equal to 30 mm x (30-45) mm, and their thickness was 4-6 mm. These samples were fixed successively upon a special support placed at a distance of 30 cm from the PF-360 electrode outlets, and their surfaces were perpendicular to the plasma streams direction. To study characteristics of plasma generated by the erosion of the irradiated CFC targets the use was made of the optical emission spectroscopy. Spectroscopic measurements were performed side-on through a quartz window and an optical collimator coupled with a fiber-cable and a Mechelle®900 spectrometer. The observation axis was almost parallel to the sample surface. The spectroscopic measurements were performed with different acquisition times but most were performed with exposition time of 2 μs at different instants after the discharge current dip. The optical spectra, as recorded mainly in the wavelength range from 350 nm to 700 nm, were analyzed using the NIST database. The distinct deuterium lines as well as carbon lines were identified and analyzed quantitatively. It was estimated that the average plasma density changed from 10^{18} cm^{-3} to 5×10^{15} cm^{-3} during plasma expansion lasting about 20 μs. These values can be useful for further modeling of the CFC erosion by energetic plasma streams.

To investigate the erosion of the irradiated samples they were also analyzed with an optical microscope and weighed with an electronic balance. Some distinct erosion craters were recorded, and it was estimated that one plasma shot can induce a loss of 0.4-0.9 mg of the target material. Taking into consideration that each pulsed plasma stream emitted from the PF-360, contains many micro-beams of accelerated primary deuterons (of energy ranging from about 80keV to even several MeV) as well as high-energy (about 3MeV) fusion-produced protons, it was impossible to identify which particles caused the observed erosion craters. Nevertheless, it has been shown that such pulsed plasma streams can be used to study materials of interested for future fusion machines, and some data needed for their modeling have been collected.

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Modelling of carbon deposition from CD_4 injection in the farScrape-Off Layer of TEXTOR

P05A

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Erosion and re-deposition of wall material in fusion devices are critical issues due to wall lifetime and tritium retention caused by co-deposition. Net layer deposition and thus long-term tritium retention is expected to take place mainly at remote areas.
A dedicated experiment has been carried out at TEXTOR to study transport and carbon deposition from injected CD_4 molecules at remote areas. For this purpose a cylinder equipped with injection tube and quartz microbalance (QMB) diagnostic on its top surface has been exposed to the far Scrape-Off Layer (SOL) – i.e. 5-8cm away from the Last Closed Flux Surface (LCFS) – of the TEXTOR edge plasma. Due to the remote position of the cylinder and the magnetic field almost parallel to its top surface the ion flux to the cylinder top surface and thus erosion due to ions is negligible. Ohmic and NBI-heated discharges have been applied and the injection rate has been varied between 1.2×10^{19} and 3.3×10^{20} CD_4/discharge. Shot-resolved measurements of the deposition on the QMB revealed deposition efficiencies (i.e. amount of deposited carbon atoms on the QMB relative to amount of injected CD_4) between 0.002 and 0.008% depending on the radial position of the cylinder. The difference of deposition between ohmic and NBI-heated discharges is not very significant. Post-mortem analysis of the whole top surface of the cylinder results in a deposition efficiency of about 1%. Details of the experiment can be found in [1].
This contribution focuses on modelling of the experimental findings. The transport of injected CD_4 is calculated with ERO. Observed CD emission pattern can be well reproduced applying measured electron density and temperature profiles from He beam diagnostic in TEXTOR. Overall modelled deposition efficiency on the cylinder surface (∅10cm) for a representative discharge is about 2% applying Molecular Dynamics (MD) data for reflection of returning species from Ohya. The reflection coefficients are rather large (R_{CH4}=R_{CH3}=1, R_{CH2}=0.9, R_{CH}=0.6, R_C=0.3) due to low impact energies smaller ~3eV. As the QMB quartz is located about 1cm recessed from the cylinder surface, ERO only models the flux of particles entering the QMB aperture (∅0.8cm), which is about a factor of 20 larger than the measured deposition on the QMB. The 3D-GAPS code has been applied to study the detailed transport within the QMB housing showing that measured deposition can be reproduced if MD reflection coefficients are used. Thus, important conclusion of the modelling is that – in contrast to plasma-wetted areas - carbon deposition at remote (i.e. plasma-shadowed) areas can be understood without enhanced re-erosion of re-deposits.

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P05B

Modelling of Deposition at the Bottom of Gaps in TEXTOR Experiments

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Fuel retention in fusion devices with carbon based armour materials is to a large extent governed by the co-deposition process. Formation of fuel-rich carbon layers takes place preferentially in remote areas hidden from direct plasma impact. As such, gaps of castellated plasma-facing surfaces act as potential trapping sites for tritium, thus representing a critical issue for continuous and safe operation of next step fusion devices.

A dedicated experiment in TEXTOR using a test limiter with gap-like structures [1] revealed that thick carbon layers are formed not only near the entrance but to a certain extent also at the bottom in gaps. In this case, cleaning of gaps becomes a challenging task. In the experiment it was not clear whether observed deposition on the bottom results from normal discharges or from off-normal events like losses of the plasma position control. Modelling performed with the 3D-GAPS code [2] for normal TEXTOR discharges was not able to reproduce the significant level of deposition in those most plasma remote areas without imposing extreme assumptions on particle transport in gaps.

In order to clarify the deposition in gaps, a dedicated experiment is planned in TEXTOR with injection of quantified amounts of $^{13}CH_4$ molecules in the vicinity of a gap and shot-resolved in-situ measurements of deposition at the gap bottom with sensitive Quartz Microbalance (QMB) diagnostics. Predictive modelling with coupled ERO [3] and 3D-GAPS simulations is applied to estimate the amount of injected carbon capable of reaching the bottom of the gap. ERO calculates the transport of injected $^{13}CH_4$ molecules in plasma and provides fluxes of particles entering the gap. The gap is located approximately 23 mm away from the injection hole and the gap bottom is about 9 mm recessed from the plasma exposed surface. ERO simulations for a representative TEXTOR discharge show that up to 0.7% of injected particles can reach the gap aperture. The 3D-GAPS code is used to follow the transport of these particles down to the QMB surface. Under a very conservative assumption of no reflection and without re-erosion taken into account, only small fraction of particles entering the gap can be deposited at the bottom resulting in carbon deposition efficiency on QMB of the order of 0.01%. With such efficiency, an injection rate of 5×10^{19} molecules per discharge will be sufficient to reach a detectable level of deposition on QMB. Application of an improved particle reflection model with reflection coefficients according to Molecular Dynamics data leads to two times smaller deposition efficiency still affordable from the point of view of QMB sensitivity. Detailed results of refined predictive modelling will be presented in this contribution.

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Ion-surface interactions of seeding gas ions with ITER-relevant armour materials

P06A

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In recent measurements the interactions of seeding gas ions Ar^+ and small hydrocarbon ions with room temperature carbon-fibre-composite surfaces have been investigated on the BESTOF apparatus in Innsbruck [1]. The experimental setup is described in detail in earlier publications [2]. In the first part of this tandem mass spectrometer ions are produced in an electron ionization source before entering a reverse-geometry two-sector-field mass spectrometer. The mass selected beam of projectile ions then is directed onto a surface sample at well defined incident energies ranging from a few eV up to 100eV. Product ions formed in interactions with the surface are analysed in a time-of-flight mass spectrometer at a scattering angle of 90°.
In analogy to similar experiments [3] the data show extensive contributions of sputtered hydrocarbons, which cover the surfaces at room temperature. Despite this, there is no evidence of surface-induced reactions of the projectile ions with surface hydrocarbons forming ArH^+. In addition, the data show contributions of sputtering of alkali contaminants Na^+ and K^+. Experiments of Z. Herman et al. indicate the necessity of heating of the surface to at least 600°C to access measurements of ions with clean, uncovered surfaces [4]. Therefore, the experimental setup is presently being modified to allow for heating of the surface. In this work we present recently measured data on interactions of N_2^+ with CFC and for various surface temperatures ranging from room-temperature to 600°C, hence for both hydrocarbon-covered and clean surfaces.

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Interaction of low pressure hydrogen plasma with materials of interest for fusion: study of the negative ion surface production

P06B

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We are studying negative ion surface production on surfaces in hydrogen and deuterium low pressure plasmas [1-3]. The plasma is created in a helicon reactor powered by a 13.56 MHz RF generator. In the diffusion chamber of the helicon reactor, a negatively biased sample held by a temperature controlled sample holder faces a mass spectrometer placed 35mm away. The sheath in front of the sample accelerates positive ions towards the surface, while the negative ions created on the surface upon bombardment are accelerated towards the plasma. They cross the plasma and are extracted by the mass spectrometer, where they are analysed according to their energy. Negative hydrogen ion distribution functions (IDFs) are studied depending on experimental conditions and surface material, with the aim of better understanding negative ion surface production at the plasma-surface boundary. In this paper, we discuss the influence of extraction voltage on the shape of IDFs, and draw the main mechanisms at the origin of the negative ion creation. Most of studies are performed using graphite sample (HOPG), but we also compare it with other materials of interest for fusion like tungsten or doped diamond.

Hydrogen Retention and Release in the RFX-mod Graphite First Wall

P07A

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RFX-mod is a toroidal experiment (major/minor radius 2/0.46 m) for the confinement of thermonuclear Hydrogen plasmas in the Reversed Field Pinch magnetic configuration. The first wall is entirely covered by tiles made of polycrystalline graphite and it bears a strong thermal power deposition that can locally reach values of the order of tens MW/m^2 and strong particle fluxes from the plasma of about 10^{21}-10^{22} m^2/s. Hydrogen retention in the graphite of the wall has a direct impact on the operation of RFX-mod. In fact, during the discharges, plasma density is entirely sustained by particle in-fluxes from the wall and the control of Hydrogen retention is the way to control plasma density during experiments.

With the aim both to improve our control capability and to study the more general issue of the behaviour of graphite as first wall material for future fusion experiments, we analysed the Hydrogen retention and release under different conditions of the RFX-mod wall. We measured the Hydrogen that is left in the wall during the plasma operation and that is removed during the wall cleaning treatments (baking and Helium Glow Discharge Cleaning (GDC) sessions). Local but more detailed information was obtained by the analyses of graphite samples that have been inserted inside the vessel up to the first wall envelope surface in three different toroidal-poloidal positions by means of dedicated manipulators. Some tiles have been also removed and analysed, giving information about particles absorption under long-term exposure to plasma and cleaning treatments. We found that during operation a progressive accumulation of Hydrogen in the wall occurs and the effectiveness of the cleaning treatments routinely applied is discussed. The differences in the rate of accumulation on clean graphite and on graphite that has been coated with Boron or Lithium are shown and discussed.

Thermal Desorption spectroscopy investigation of co-implanted Hydrogen and Argon in graphite

P07B

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In nuclear fusion research, impurity (N_2, Ne, Ar, Kr…) seeding is receiving a continuous attention because of its implication in plasma detachment and reduction of heat load onto the divertor tiles [1]. Such studies mostly focused on power exhaust, plasma stability and impurity transport. However, little attention is given to the possible synergistic effects resulting from simultaneous bombardment of the impurity with hydrogen or its isotopes on plasma facing components. Therefore, we have undertaken particle beam experiments in order to investigate these fundamental mechanisms.

In this work, 1.5 keV hydrogen (or deuterium) and argon ions were co-implanted in graphite material, by simultaneously feeding a unique ion gun with both species. Several experiments were performed by varying the partial pressure of gases, as well as the ion fluence. Temperature programmed desorption (TPD) was performed and desorption products compared to those resulting from single ion bombardment. It is found that the presence of Ar ions during H bombardment increases the hydrogen desorption yield and also result in a strong downshift of the desorption temperature by nearly 300 K. It is proposed that Ar ions modify the H_2^+ ion range implantation and that defects created by Ar ions assist trapping of H species. These mechanisms should compare well with those implying chemical sputtering of hydrocarbon films that were observed during low-energy Ar+ ion and H atom impact [2].

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Assessment of Fuel Re-absorption by Thermally Pre-treated Co-deposited Layers

P08A

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Reduction of in-vessel fuel inventory is essential in operation with the tritium, especially in the presence of carbon plasma-facing components (PFC). Two basic schemes for fuel removal are currently considered: (i) cleaning surfaces from co-deposits, (ii) only desorption of hydrogen-containing species. The latter approach is based either on surface heating by photonic means or long-term annealing of PFC. As a result, the outgassed deposited layers remain in the vessel and they would be repeatedly exposed to plasma. The question is: how does the fuel-depleted surface respond to plasma during a subsequent exposure?

Systematic studies have been performed to address fuel re-absorption by carbon deposits exposed to plasma after cleaning procedures. The investigation has been done with graphite tiles from ALT-II, i.e. the main limiter at the TEXTOR tokamak. Pure graphite plates are used as a reference material. The experimental program comprises: (i) pre-characterisation of specimens; (ii) D desorption by baking the tile at 1000 ^{o}C or by μs laser pulses (surface temperature exceeds even 1200 ^{o}C) accompanied by gas-phase control; (iii) surface analyses of the fuel depleted layers; (iv) exposure to deuterium plasmas under laboratory conditions (glow discharge or ECRH plasma) or in the tokamak; (v) quantitative deuterium determination in the re-exposed materials. Analyses are done by means of gas-phase and material research methods: thermal desorption spectrometry (TDS), ion beam techniques (e.g. NRA: nuclear reaction analysis) and microscopy. The most important findings of the still on-going research are listed below:

(a) TDS and NRA measurements of the D content in the original co-deposits and after repeated exposures show good agreement between these methods (within 10%) thus indicating that most fuel is in the surface layer.

(b) Thermal treatment causes strong modification of the deposit structure. It enhances surface roughness and layer brittleness leading eventually to flaking and detachment of co-deposits.

(c) Fuel content in original layer is 4.7×10^{18} D·cm^{-2}, whereas the re-absorption of fuel during the subsequent exposure is on the level of 1.5×10^{17} D·cm^{-2}, i.e. 30 times lower. Moreover, the D uptake by the fuel-depleted layer is only 10-20 % greater than measured on the reference graphite plate. This result is most probably attributed to the differences in surface roughness.

(d) The desorption spectra of fuel species from the original and the re-exposed surface are different and show that a lower fraction of fuel is bound in hydrocarbons after the exposure to laboratory plasma.

The experiments performed to date indicate that the fuel re-absorption by thermally cleaned carbon surfaces may not be significant and will not lead to an immediate re-saturation of deposits. Also the dust produced in connection with surface heating will contain less fuel than observed after laser-induced ablation [1].

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Tritium uptake in graphite tiles exposed to EAST plasma and then tritium gas

P08B

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Doped graphite GBST1308 (1%B_4C, 2.5%Si, 7.5%Ti) with thick SiC gradient coatings (SiC/C) is now being used as plasma facing materials (PFMs) on the Experimental Advanced Superconductive Tokamak (EAST) in the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). In our previous work [1], tritum uptake behavior of the unirradiated graphite tile samples was investigated. In this paper, the amount of adsorption and the depth profile of tritium have been studied using β-ray-induced X-ray spectrometry (BIXS) [2, 3] in the irradiated graphite tiles from EAST with following exposure to tritium gas.

Samples were cut from graphite tiles exposed to EAST plasmas, including erosion and deposition ones and the third one from unirradiated tile for comparison, and were then exposed to tritium gas after thermal pretreatment under the given conditions. Each sample was analyzed by means of BIXS, and changes in the X-ray spectrum with time were observed for a maximum time of 144 hours. To examine tritium distribution in the bulk, the cross-section of tritium-exposed samples was also measured by an IP method for comparison with the results of BIXS measurements. The surfaces of the samples were observed by a digital microscope, and analyzed by a small angle XRD. It was confirmed from XRD analyses that the surfaces of the three samples mainly consist of SiC. The results of BIXS indicated the significant amount of tritium was absorbed in the deposition sample in comparison with other samples, which was supported by the IP measurements. In addition, it was found that a variety of metallic elements existed in the deposition and erosion samples, indicating that the plasma-facing materials are partly eroded by plasmas. Drastic decrease in tritium retention appeared by lowering exposure temperature, and the trapped tritium was maintained stable with time.

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Influence of deuterium ion and atomic exposure on dehydrogenation of C:H films

P09A

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The paper investigates the processes initiated by deuterium ions and atoms irradiation (D) in hydrocarbon films (ratio H≈C0.2, the thickness ≈ 2.5 μm) deposited by PCVD technique.

Energies of 50 eV/D and 400 eV/D were selected for irradiating D ions which were extracted from deuterium plasma. Ion flux and fluence equaled 10^{16} D·cm^{-2}·s^{-1} and $5\cdot10^{19}$ D·cm^{-2} respectively. Average energy of atomic deuterium (D^0) was about 0.2 eV. Atomic D^0 flux and fluence were kept near the same as ion ones. Substrate temperature was maintained at 550 K in all experiments. The irradiated samples were subjected to sputtering in argon plasma up to different depths in the range of 20-900 nm. Argon ions with energy 300 eV were used. Then D and H atom contents in remaining part of the films were measured by thermal desorptional spectrometry, and D and H atoms depth distributions were calculated.

Experimental results showed that implanting D ions penetrated into and initiated H atoms release from the layers much thicker than the ion stopping zone. As this took place D retention did not compensate H diminution. $9.1\cdot10^{16}$ and $1.3\cdot10^{17}$ D·cm^{-2} atoms were retained and $1.6\cdot10^{17}$ and $1.7\cdot10^{17}$ cm^{-2} H atoms were removed during respectively 50 eV and 400 eV ion D irradiation. Decrease of H concentration in the layer of the order of stopping zone was stronger, when 50 eV/D ions were used.

Atomic deuterium irradiation led to deuterium trapping mainly in the top surface layers and did not cause to any noticeable hydrogen desorption from the films.

Calculations gave the value $3\cdot10^{-14}$ cm^2·s^{-1} for D diffusion coefficient in the depth interval 0 - 200 nm, and $2\cdot10^{-13}$ cm^2·s^{-1} for interval 200 - 900 nm in the ion irradiated films. These values are much higher than the maximum value of diffusion coefficients for dense graphites at 550 K, which were found to be 10^{-15} - 10^{-17} cm^2·s^{-1} [1]. Slower D diffusion in the interval 0-200 nm could be explained by influence of irradiation induced defects penetrating from the stopping zone into underneath layers.

D atom irradiation did not produce remarkable amount of radiation defects, and D atoms diffusion coefficient for such films in accordance with above conclusion was found to be close to the value obtained for interval 200 - 900 nm of plasma irradiated films.

Conclusion is made that radiation defects and stresses initiated by them rather than isotope exchange processes were the causes of liberation of H atoms from the traps and their release.

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P09B

Time dependent low-energy deuterium interactions with lithiated graphite plasma-facing surfaces

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Lithium has been utilized as a plasma-facing conditioning surface (PFS) in TFTR, CDX-U, FTU, T-11M, TJ-II and NSTX as a means of achieving enhanced plasma performance. In part, improved plasma performance is a result of reduced deuterium recycling immediately following lithium wall conditioning. Work at Purdue University has investigated the fundamental mechanisms responsible for deuterium retention using surface characterization techniques such as *in-situ* X-ray and ultraviolet photoelectron spectroscopy (XPS and UPS). XPS has been used to identify Li-O-D interactions at 533.0 ± 0.6 eV and Li-C-D interactions at 291.2 ± 0.6 eV in the O 1s and C1s energy ranges, respectively [1]. Previous studies have investigated time-dependent deuterium irradiation of lithiated graphite with fluences up to 7 x 10^{17} cm^{-2}. Results suggest that a nominal lithium dose of 2 μm will saturate with deuterium at fluences between 3.8 to 5.2 x 10^{17} cm^{-2} [2]. Wall conditioning in NSTX typically calls for around 200 mg lithium to be deposited in between discharges [3], approximately equal to a lithium dose of 200-300 nm. In addition, future upgrades will allow NSTX-Upgrade to sustain a 5 sec plasma pulse, thus exposing the PFS to deuterium fluences much higher than the one-second pulses currently sustained [4]. In the present work we study the saturation limits of retained deuterium by lithium-conditioned graphite PFS under more relevant fusion-device conditions. The possible saturation of smaller lithium doses is examined for current deuterium fluences (~10^{17} cm^{-2}), as well as higher deuterium fluences (>10^{18} cm^{-2}) for larger lithium doses.

Controlled experiments utilize a monoenergetic ion source to introduce deuterium. Fusion plasmas have a wider energy distribution than those supplied by conventional ion guns. XPS spectra of lithiated graphite samples irradiated with 300 eV deuterium (150 eV/amu for D2) are within ± 0.1 eV of spectra from samples irradiated with 1 keV deuterium (500 eV/amu for D2). This behavior indicates that surface chemistry occurs at the ion end of range and that chemistry is not energy dependent. However, because ion penetration depth is a function of ion energy, lithiated graphite is conjectured to saturate faster at lower ion energies than at high ion energies.

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Work supported by USDOE Contract DE-FG02-08ER54990

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P10A

Characterization of temperature-induced changes in amorphous hydrogenated carbon thin films

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Redeposition of eroded carbon along with codeposition of hydrogen in fusion experiments operating with carbon as a first wall material results in the build-up of hydrogenated carbon films in deposition-dominated areas. Heating of these films during plasma operation can modify them and, especially, drive out hydrogen, thus decreasing the hydrogen isotope retention associated with the formation of these films.

In this work, hard hydrogenated amorphous carbon thin films with an initial hydrogen content of about 30% were heated in vacuum to different temperatures and held at these for about 30 minutes. Afterwards, the cooled-down samples were analyzed by various techniques. Fairly strict and reproducible correlations were found between all the determined parameters and the heating temperature. Single-wavelength ellipsometry showed that the real part of the refractive index of the films at 632.8 nm wavelength decreased with annealing temperature while absorption increased. It also showed swelling of the films with a thickness increase of about 50% for films heated to approximately 1000 K. Ion beam analysis showed that hydrogen was released from the films during heating with only about 5% of the initial H remaining at 1300 K while no significant loss of carbon could be detected. The losses of hydrogen (H_2) during heating were monitored by thermal effusion spectroscopy and they are in good agreement with the IBA results. Both results together indicate that hydrogen from these hard films is mainly released as H_2 in contrast to soft a-C:H films where a substantial amount of hydrogen is released as hydrocarbons [1]. Raman spectroscopy, sensitive to the sp^2 carbons in the film, delivered evidence of an aromatic domain size increasing under heat treatment. All observed changes set in at about 700 K and the quickest change with temperature is observed around 900 K.

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Hydrogen-carbon deposit removal with a multi-pin-plane plasma device

P10B

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This project aims to study the feasibility of a plasma cleaning method, previously tested in the context of the plasma-assisted destruction of combustion-engine exhaust, namely VOCs and soot particles. This was achieved by means of a corona plasma discharge, ignited between a multi-pointed electrode and an electrically conducting substrate. In one instance, a gas mixture similar to that encountered in the exhaust of combustion engines (atmospheric pressure oxygen-poor air and ppm quantities of pollutants such as naphthalene, acetylene (soot precursor), etc...) is allowed to flow in the space between the electrode and substrate. In another, soot particles are deposited on the substrate and exposed to the plasma discharge, in the presence of oxygen-poor air. The plasma produces chemically active species (ozone, atomic oxygen) that can then interact with the VOCs and soot to break them down into smaller, and, ideally, less harmful molecules.

We wish therefore to test these ideas in a fusion context. More specifically, both the issues of detritiation and dust removal from the vessel walls of a fusion device can be considered. In that case, we would envisage the electrode as mounted onto a robotic arm capable of reaching the relevant areas of the walls in need of cleaning, along with the provision of appropriate feed gases.

As a proof-of-concept experiment, we plan to build a fixed electrode facing a substrate holder in an enclosure where various feed gases at different pressures can be introduced. The substrates to be exposed to the plasma would include some castellated or grooved surfaces, in order to measure the penetration of the cleaning plasma into narrow trenches, as will be necessary for the ITER and JET-ILW device walls. Hydrogen-rich dust or harder deposits obtained from the LIMHP CASIMIR device would be placed on the substrate and exposed to the plasma. We would then study the influence of discharge energy, exposure time, feed gas composition and pressure, and any other relevant process parameters on the dehydrogenation of the dust particles, the lifting of the dust from the substrate or its decomposition, the penetration depth of the plasma cleaning action along the grooves, etc... If the method looks promising in our laboratory set-up, a scaled-up device to be introduced in a tokamak such as Tore Supra or JET would be envisioned in the future.

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Influence of H-H discharge on hydrogen isotope retention in boron films

P11A

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In the Large Helical Device (LHD) of the National Institute for Fusion Science (NIFS), boronization has been applied as a first wall conditioning technique in order to keep impurity concentration low in plasma. During plasma operation, hydrogen isotopes would be implanted into the boron films, and interact with the impurities. Retention behaviors of hydrogen isotopes in the boron films contained impurities should be clarified for the stable plasma operation and tritium recycling. From our previous studies, the impurities were found to exist in forms of O-B and C-B bonds, free oxygen and free carbon in boron film. It was also found that O-B and C-B bonds trapped deuterium to form B-O-D and B-C-D bonds, although free oxygen and free carbon formed water and hydrocarbons which reduce the deuterium retention in the boron films [1]. To simulate tritium recycling in actual fusion environment, the boron films should be exposed to H-H discharge in plasma devices and complex interactions with the impurities on hydrogen isotope retention should be clarlified.

The boron films were deposited on the Si substrates by the Plasma Chemical Vapor Deposition (P-CVD) apparatus at Shizuoka university. These samples were exposed to H-H discharge at LHD. The chemical compositions of the boron films were measured by X-ray Photoelectron Spectroscopy (XPS) and the hydrogen isotope retention behavior in the boron film exposed to H-H discharge was investigated by Thermal Desorption Spectroscopy (TDS). These results were compared with those by ion implantation

The results of XPS showed that the concentrations of boron, carbon and oxygen in the boron films were changed from 92% to 31%, from 2% to 41% and from 2% to 26% by H-H discharge, respectively, indicating that the trapping of impurities by the boron film was proceeded. However, no contamination of impurities was found for the deuterium implantation, showing that an introduction of impurities in boron films was derived from the H-H discharge. In the H_2 TDS spectrum for the H-H discharge exposed sample, H_2 was desorbed in the temperature range of 400-1100 K. On the other hand, in the D_2 TDS spectrum for the deuterium implanted sample, D_2 was desorbed in the temperature range of 300-800 K.It was indicated that the hydrogen trapping by impurities would make a large influence on hydorgen retention in boron films, which is almost consistent with the XPS results. Therefore, it is suggests that the H-H discharge have a large impact on chemical states in boron films and the retention behaviors of hydrogen isotopes.

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O-carborane boronization of MST during pulsed discharge cleaning

P11B

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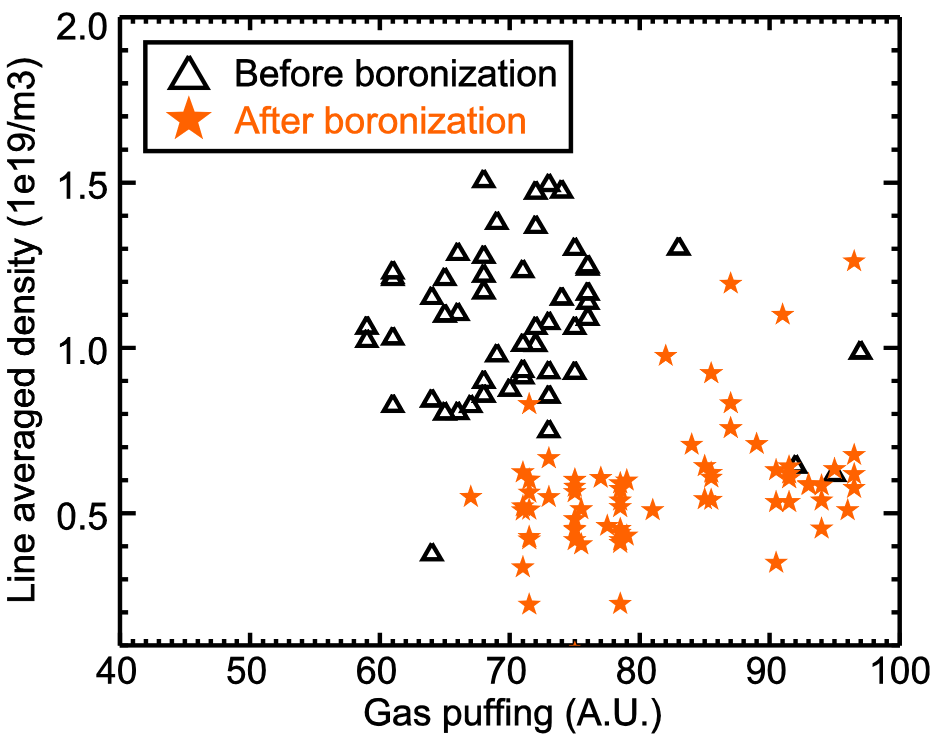
A technique for o-carborane boronization of the Madison Symmetric Torus (MST) reversed field pinch (RFP) has been developed. Gaseous o-carborane, created by heating the powder up to 100 °C, is injected during a pulsed discharge cleaning process. The pulse repetition rate is being optimized to increase the fuel (o-carborane) efficiency. Depth profiles for the aluminium coupons exposed to the boronization process are routinely obtained by x-ray photoelectron spectroscopy with Ar ion beam etching. The characteristics of the deposited film, such as thickness and hardness measured by profilometry and ellipsometry, respectively, will also be identified. Significant reduction in wall recycling is observed after the boronization followed by helium conditioning discharge for hydrogen removal, which is indicated by the amount of deuterium puffing (See the figure below). In addition, long-lasting (up to 50 % of the current flattop in a discharge) spontaneous enhanced confinement (EC) periods are often observed after the boronization. The effective charge of the plasmas before and after the boronization will be analyzed and compared using x-ray spectroscopy.



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P12A

Interaction of high flux D_2/N_2 plasmas with Be

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Reliable edge plasma temperature control and radiative cooling are considered crucial to protect plasma facing components in high performance discharges in current and future divertor machines. Positive results from ASDEX Upgrade with nitrogen (N_2) seeding in an all-tungsten tokamak [1] lead to the question whether N_2 seeding would be a viable option within the ITER-like-wall project at JET [2]. One of the open issues is the compatibility of N_2 seeding with Beryllium (Be) as the main wall material.

In order to study N_2-Be interaction in realistic divertor plasma conditions we exposed Beryllium samples to high flux D_2/N_2 plasmas in PISCES-B. At ion fluxes of ~$4x10^{18}$ $cm^{-2}s^{-1}$, fluences of approximately $1.5x10^{22}$ $ions/cm^2$ were accumulated per exposure. For pure D_2 sputtering yields of $4.2x10^{-3}$ per ion at -100 V bias voltage and $1.5x10^{-3}$ per ion at -50 V sample bias were calculated from mass loss measurements. Adding approximately 10% N_2 to the discharge reduced the target erosion rates by a factor of 3 (-100 V bias). Similar results were obtained for exposures with ~4% N_2. At -50 V bias, the addition of N_2 resulted in mass losses close to our detection limit. This reduced erosion is probably due to a passivation of the target surface by the N_2 leading to charging of the isolating target surface and hence a reduction of the energy of the impinging ions. The formation of thin nitride layers on the target samples was confirmed by XPS and AES. SEM imaging of the exposed target surfaces showed a strong influence of the N_2 on the surface morphology. While samples exposed to pure D_2 show “grass like” structures with nm sized tips, samples exposed to N_2/D_2 plasmas show a smoother surface with µm sized erosion dents. Additionally, arc tracks along the surface were visible on samples exposed at -100 V bias, 10% N_2 plasmas but not for the -50 V bias exposures or for 4% N_2.

In addition to the Be target samples, non plasma wetted tungsten witness plates were used to collect material sputtered from the Be targets. The deposits collected from N_2 seeded plasmas showed smooth, non porous surfaces and are insulating. Thermal desorption measurements (TDS) on these deposits showed small D_2 and HD release peaks at about 540K. Normalized to fluence and target mass loss, witness plates exposed to N_2/D_2 plasmas retain roughly 50x less D_2 than those exposed to pure D_2 plasmas. During TDS different N_2 containing species were released at temperatures between 500 and 800 K.

Altogether our results show that introducing nitrogen into the plasmas can significantly reduce Be erosion and D_2 codeposition but the formation of insulating layers could lead to problems in tokamak operation.

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Effects of transient heating events on plasma facing materials in a steady-state plasma environment

P12B

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In the International Thermonuclear Experimental Reactor (ITER) and other long-pulse diverted tokamaks such as EAST and K-STAR, edge-localized modes (ELMs) will result in large thermal transient loads on plasma-facing components (PFCs). These components will have been exposed to long duration or steady-state plasmas, resulting in a material surface saturated with hydrogen isotopes and with helium. Nearly all heat-pulse tests of plasma facing materials (PFMs) have been completed in vacuum environments without the presence of simultaneous plasma exposure. Thus combined effects of thermal transients on materials undergoing plasma exposure need to be explored.

Heat-pulse experiments have been conducted in the PISCES-A device using a short-pulsed laser in divertor-like plasma conditions. Previous results [1] indicate that the erosion of tungsten PFMs is enhanced as compared to transient-only or plasma-only experiments, and the threshold energy for material removal by a transient heat pulse in a steady-state plasma background is reduced. These results have been confirmed in DIII-D experiments [2] where synergistic mass loss is three to five times greater. NRA of exposed samples will be presented to examine a possible link between D depth profiles and these effects. There exists a direct correlation between fluence incident between transients and W-I line emission in the plasma. We will additionally discuss the effects of He+D plasmas on similar PFMs.

A new laser capable of pulse durations of 0.3-10msec, similar to ELMs, has been utilized in PISCES-B to evaluate the effects of transient heating of beryllium PFCs. We calculate that transient heating by laser pulses at 1Hz during steady-state deuterium plasma exposure (Γ~4E18 cm^{-2} sec^{-1}, Te~8eV) heat the sample from 35ºC to several hundred degrees. Initial results for beryllium (Fluence~1.5E22 cm^{-2}) indicate that retention is reduced under these conditions and the temperature at which deuterium is released is altered. In addition, the surface structure is changed although transient heating is well below the normal melting temperature of beryllium. When higher irradiance laser pulses are used, there is a direct correlation between peak temperature and Be-I line emission. Progress of a new fast-response surface temperature diagnostic will also be discussed.

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Depth distribution of deuterium implanted into beryllium and heating rate-dependent desorption

P13A

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This contribution is a further step in the continuing effort to better understand the retention and release mechanisms of hydrogen isotopes implanted into Be at keV energies. An energetic D ion impinging on the Be target transfers its energy to atoms of the metal matrix. This initiates a collision cascade in the course of which defects are created and the D atom is slowed down. Various experiments point to the fact that implanted hydrogen is trapped within the ion range at defect sites created by the implantation itself (e.g. [1,2]).

The first part of this contribution deals with the D depth distribution after implantation at room temperature. A mirror polished single crystalline Be sample with (0001) surface orientation is implanted with D at 3 keV to a fluence well above 10^{17} cm^{-2}. Saturation of the Be lattice by D occurs beyond this threshold fluence [1]. The depth profile is calculated in a simulation with the program SDTrim.SP [3]. The distribution can be approximated by a smeared-out step function which extends up to the ion range. The calculation result is compared to the measured depth distribution resulting from nuclear reaction analysis (NRA). For the NRA investigations a beam of ^{3}He ions at various energies is used to probe the D-implanted sample. The experimental assessment of the D depth profile is performed with the help of state-of-the-art computational tools: The optimal ^{3}He energies are chosen by means of Bayesian experimental design [4] and a statistically sound evaluation of the depth profile is performed with recently developed software [5].

In the second part of this contribution temperature programmed desorption (TPD) spectra of D from polycrystalline Be are presented. D is implanted with fluences in the order of $3*10^{15}$ cm^{-2}. Implantation energies from 0.4 to 3 keV are used. Temperature ramps vary from 0.2 to 4.0 K s^{-1}. Variations in desorption temperatures are observed. They are connected to the diffusive transport of D from trapping sites to the surface. Interpretation of the spectra is aided by simulations with the recently introduced CRDS code [2].

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Study of Helium Bubbles Evolution in Highly Neutron Irradiated Beryllium by X-Ray Micro-Tomography and Metallography Methods

P13B

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Beryllium is planned to be used as a plasma facing material and as a neutron multiplier in the test blanket module (TBM) in ITER, and also as a neutron multiplier in the European Helium Cooled Pebble Bed (HCPB) blanket concept of DEMO. Any case beryllium will accumulate some amount of helium due to nuclear threshold reactions with high energy neutrons going under neutron irradiation at fusion reactoe operation. The helium accumulation strongly depends on neutron fluence or damage dose in the irradiated beryllium. For example the production of about 80 dpa, up to 25 000 appm of helium and also 700 appm of tritium is planned in beryllium pebbles from the HCPB blanket of DEMO. At present beryllium is widely used as a neutron reflector or moderator in research nuclear reactors. Prominent features of the beryllium use in research reactors are low temperature operation parameters (50-70 °C) and a high neutron dose accumulation to the end-of-life of the beryllium blocks. It seems very attractive to use fragments of the irradiated beryllium blocks with following post-irradiation high temperature annealings for reproducing of fusion reactor parameters on temperatures and helium accumulations. In the work the fragments of the BR2 reactor beryllium matrix irradiated at 50 °C up to 22000 appm of helium accumulation were investigated. The irradiated beryllium fragments were annealed at 850 °C and 1000 °C both for 0.5, 1, 5 and 10 hours in vacuum. The prepared beryllium samples were investigated by optical (OM) and scanning (SEM) electron microscopes at KIT, and by X-Ray micro-tomography at ESRF.

Because beryllium for the use in a reactor is manufactured by powder metallurgy, i.e. by hot pressing or hot extrusion methods, this leads to formation of pores and beryllium oxide particles on grain boundaries. Under neutron irradiation, owing to anisotropic growth of grains in different crystallographic directions, an evolution to increase of grain boundary porosity occurs. Post-irradiation high temperature annealing involves to the process redistribution of helium atoms with formation of helium bubbles. This leads to the increase of beryllium swelling having different values for different parts of the annealed beryllium samples depending on the distance to the external surface. The evolution of helium bubble sizes and amount takes place on variety of the anneal parameters (temperature and exposure). In particular, the increase of the parameters leads to increase of swelling by means of the increase of bubble sizes and amount. The presented results are discussed and summarized as a model describing the evolution of helium bubbles in a connection with microstructure defects (boundaries, dislocations, point defects).

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Quantum studies for plasma facing components for ITER

P14A

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Quantum theory within the DFT frameworks provides very powerful tools in analyzing the fundamental chemical processes governing the reactivity of plasma facing surfaces in nuclear fusion devices. In this contribution we present results on *plane-waves* DFT calculations on the reactivity of beryllium and lithium doped graphite surface. Atomic or molecular oxygen are always present in the tokamaks (because of accidental leaks) or in laboratory experiment. Beryllium and lithium are both good oxygen getter, Be cover the largest part of the inner ITER wall, and Li is sometimes used to dope the graphitic components.

Since the reactivity of Be toward hydrogen can hardly be studied in situ, many experiments are developed in laboratories. Unfortunately the beryllium surface can be very easily oxidized even at relatively low temperature and under ultrahigh vacuum conditions. This reaction dramatically pollutes the experimental measures and more specifically the TDS (Thermal Desorption Spectroscopy) ones.

The problem of oxygen adsorption and dissociation on beryllium surfaces is still largely under discussion, by analogy with oxidation of other sp-metals such as aluminum or magnesium, it is generally agreed that O_2 adsorption, and then dissociation, is an activated process. DFT is carried on in order to evaluate the barriers to dissociation of the oxygen molecule above the special symmetry points of the Be(0001) surface. Then the interaction of molecular and atomic hydrogen with this partially oxidized surface is studied in comparison with metallic beryllium surface.

Deposition of a lithium thin layer onto graphite was found to considerably suppress physical sputtering. DFT calculations are developed on lithium interaction with pristine and defective graphite surfaces, oxidized or not, in order to evaluate hydrogen retention and restitution. Li is found to significantly enhance H/D adsorption. Oxygenation or/and perturbation (single vacancies) is shown to give rise to complex structures modifying hydrogen trapping on surface and inducing Li diffusion towards the inner graphite layers.

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Transient effects during sputtering of a-C:H surfaces by nitrogen ions

P14B

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In fusion devices nitrogen seeding is used to radiatively cool the plasma edge in order to reduce the power load on highly exposed plasma facing components. Hence the interaction of nitrogen ions with amorphous hydrogenated carbon (a-C:H) layers, which are built up by co-depostion of carbon and hydrogen isotopes in carbon containing fusion devices, is of considerable interest.
Sputtering yields of polymer-like a-C:H thin films by N_2^+ molecular ions were studied in-situ and real-time using a highly sensitive quartz crystal microbalance (QCM) technique developed at the technical University of Vienna [1]. 370 nm thick a-C:H films with a hydrogen content of 50% were deposited onto the gold electrode of a quartz crystal in an ECR methane plasma [2]. The mass change of the a-C:H layer under N_2^+ bombardment was deduced from monitoring the change in the resonance frequency of the SC cut quartz driven at its thickness shear mode. A highly accurate electronics allows detecting mass changes of 10^{-2} a-C:H monolayers per second [3].
When bombarding a fresh plasma-deposited a-C:H layer with nitrogen ions we observe a sputtering yield, which decreases exponentially with fluence until a steady state value of approximately 1/3 of the initial sputtering yield is reached after a typical fluence of some 10^{15} N_2^+ ions per cm^2. The fact that reaching this steady state roughly corresponds to the removal of a surface layer of a thickness similar to the ion penetration depth suggests a correlation of this transient phase to the formation of a hydrogen depleted, nitrogen containing modified surface layer.
By using a set of rate equations, which take into account chemical sputtering and depletion of hydrogen as well as the implantation and chemical sputtering by nitrogen projectiles, both the transient as well as the steady state sputtering yield can be described quite well.

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DIVIMP simulation of W transport in the SOL of JET H-mode plasmas

P15A

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The performance of future reactors highly depends on the impurity content of the core plasma. The impurity contamination of the core plasma should be kept low enough in order to avoid extensive radiation. The requirement is particularly stringent for high Z materials, such as tungsten, for which the core concentration should be kept below 10^{-5}. To address the effect of wall materials on these parameters experimentally, the first wall of the JET tokamak has been exchanged to the ITER-like wall materials, i.e., beryllium in the main-chamber and tungsten in the divertor.

In this study, the Monte-Carlo trace-impurity code DIVIMP is used to predict the tungsten concentration in the steady-state inter-ELM phase of a JET ELMy H-mode plasma (2.7 T, 2.5 MA, P_{in} ~ 16MW, δ ~ 0.4). In DIVIMP, tungsten sputtering at the target plates and its transport in the SOL are calculated on background plasmas computed by the 2-D fluid code EDGE2D/EIRENE. The tungsten concentration is studied in different SOL regimes to investigate the screening of the divertor tungsten source by the SOL, and thereby to map out the feasible operating space in JET. The investigated SOL regimes are modelled with EDGE2D/EIRENE starting from a low-fuelling case via increased deuterium fuelling, including carbon sputtering and transport [1]. Tungsten sputtering in DIVIMP was imposed by C^{+4} being the impacting species and tungsten self-sputtering. The effect of ELMs to the tungsten sputtering is not included.

The tungsten concentration at the core boundary of the lowest fuelling case was 10^{-4}. Increasing the deuterium fuelling with factors 2 to 3 in the EDGE2D/EIRENE simulations lowers the peak target temperature and thus results in a reduction of the tungsten concentration at the core boundary to $5*10^{-6}$. The peak outer target temperature was reduced from 50 eV to below 10 eV, at the maximum fuelling rate. Thus, with low Z impurities present, keeping the peak target temperature below 10 eV is required to maintain the core tungsten concentration below 10^{-5} in a steady-state inter-ELM plasma.

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[1]See the Appendix of F. Romanelli et al., Proceedings of the 23rd IAEA Fusion Energy Conference 2010, Daejeon, Korea

Assessing Erosion of JET Tiles

P15B

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Erosion/deposition studies have been ongoing in JET since operations commenced in 1983. As a result of such studies it is known that material deposited in the shadowed regions of the divertor (including tiles and cooling louvres) comes from erosion of the first wall via physical and chemical sputtering (in the case of a carbon first wall) and also transient events [1]. Through post-mortem analysis of tiles the amount of deposit per JET campaign has been calculated in excess of 500 g, however a corresponding quantification of erosion is still required.

The introduction of tiles coated in a marker material has improved information on erosion. In the 2005-2007 JET campaigns CFC tiles were coated with tungsten [2] to evaluate tungsten erosion for the forthcoming ITER-like Wall project, however little information on carbon erosion can be inferred from this data. In addition the total thickness of marker coatings is limited by the sampling volume of non-destructive analysis techniques, such as ion beam analysis. Consequently marker layers are typically only a few microns thick depending on the coating material. The fact that the layers are thin limits their usefulness as erosion markers in high erosion zones.

Erosion from the JET divertor has also been evaluated through a series of micrometer measurements on modified CFC tiles. The results revealed small amounts of erosion on the vertical outer divertor tiles
[3] for the operating period 1999-2001. Following the success of this experiment a tile profiling instrument was developed and built at the Culham Centre for Fusion Energy which allows the front surface of tiles to be profiled at points specified by a grid before and after installation in JET. A series of tiles located in the divertor and the first wall were measured using the instrument prior to installation in 2005 and 2007. These tiles were removed from JET in 2010 and are available for re-profiling. Results of profiling an IWGL tile and a divertor tile will be presented and the amount of erosion/deposition evaluated and compared with complementary erosion/deposition diagnostics and analysis.

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This work, part-funded by the European Communities under the contract of Association between EURATOM/CCFE was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission. This work was also part-funded by the RCUK Energy Programme under grant EP/I501045.

[1)] See the Appendix of F. Romanelli et al., Proceedings of the 23rd IAEA Fusion Energy Conference 2010, Daejeon, Korea

Parameter studies and benchmarking of the global near-wall impurity transport

P16A

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The long term re-distribution of first wall materials by the plasma-wall-interaction plays an important role in multi-element fusion experiments such as ITER or JET-ILW (ITER-like wall). The iterative erosion / deposition cycles during discharges change the impurity source distribution, thus influencing the plasma performance and leading to net-erosion or deposition zones with massive changes of the wall material. This can pose restrictions on the life time of armour material or lead to safety issues due to T accumulation [1]. Therefore, interpretative and consequently validated predictive models are required that allow an assessment of critical scenarios.

For these purposes we developed the code WALLDYN, which can simulate the time evolution of the first wall surface under global re-distribution by plasma transport of the wall materials based on a self-consistent analytical description [2]. The code evaluates numerically the local flux balances along the first wall yielding the time evolution of the surface composition. The coupled processes changing the surface concentrations include sputtering by background- and impurities fluxes, charge-resolved re-distribution of elements and temperature induced processes such as chemical phase formation, sublimation, and oxidation by background gas [3]. If we assume, that the complex plasma-wall-interaction in a tokamak environment can be described as this balance approach (i.e. coupling of elementary processes), then the single processes can be validated separately by comparison with established codes (e.g. TRIM for sputtering) and experimental results.

It turns out that the plasma transport of impurities very close to the main wall is a critical part of the model and has been treated in a simplified manner in most simulations so far by “teleporting” particles from the simulation grid edge to the wall over a cm gap (standard grid) [4]. If the impurity transport is modelled up to the first wall by bridging the gap with an extended grid, the global redistribution model is very sensitive to the plasma parameters in the simulation cells close to the wall. First attempts to benchmark the plasma and impurity transport in the near-wall region were performed in [5]. To further benchmark the plasma transport near the main wall we compare WALLDYN simulations including impurity transport based on extended grid DIVIMP plasma (sol28) with JET experimental results [4]. We characterize the influence of local plasma properties such as temperature gradients, density or cross field diffusion on the global re-distribution and the time evolution of the Be erosion.

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Erosion and re-deposition of W and Ni in the divertor and midplane regions of ASDEX Upgrade

P16B

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Key factors that determine successful operation of future fusion reactors are a sufficiently long lifetime of their plasma-facing components and a low accumulation of tritium in the reactor vessel. These topics have been addressed by studying erosion and re-deposition of W and Ni in the divertor and low field side midplane regions of the ASDEX Upgrade tokamak during the 2008 and 2009 experimental campaigns. In both campaigns, ASDEX Upgrade was operated with a fully W-covered first wall and regular conditioning of the vessel by boronizations.

Erosion/deposition patterns for W and Ni were determined using Rutherford backscattering spectroscopy (RBS) and secondary ion mass spectrometry (SIMS). In the divertor zone, campaign-integrated data were extracted from inner- and outer-divertor marker tiles, having poloidal W, Ni, and graphite regions on them; the Ni stripes were located only at the outer divertor. At the outer midplane, discharge-resolved information about erosion was obtained by inserting graphite probes with 50-nm thick W, Ni, Al, and carbon stripes into the boundary plasma for typically 3—5 subsequent identical discharges.

In both campaigns, the outer divertor was a net erosion zone for W and Ni while the inner divertor was found to be a net deposition region for W. The average erosion rate of both elements peaked close to the outer strike point and gradually decreased, even turned into marginal net deposition, poloidally towards the outer-divertor baffle. Re-deposition, on the other hand, was the largest in the private flux region vertically below the inner strike point. In 2008, significant re-deposition was also measured around the outer strike zone. We observed that the erosion of Ni is 5—10 times higher than that of W and that deposition of these elements is two times larger on graphite than on the metallic marker stripes. Furthermore, in 2009 both erosion and deposition rates were 1.5—2 times larger than in 2008 due to higher particle and power loads during plasma operations.

The midplane probe experiments reveal an exponential decrease of erosion as a function of the radial distance from the separatrix, and at approximately 5 cm radially outside the plasma edge, the erosion of Ni is 2—3 times higher than that of W. The erosion pattern has been qualitatively reproduced by the ERO code simulations but the amount of erosion is underestimated by several orders of magnitude, indicating an erosion threshold effect. Fast ions can explain part of the discrepancy and also the addition of C and other impurities (e.g. Ar) from preceding discharges have been noticed to move the ERO results into the direction of the measurements.

This work was carried out within the framework of the EFDA Task Force on Plasma Wall Interactions.

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P17A

Long-term evolution of tungsten surfaces in ASDEX Upgrade

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As was shown previously with erosion marker stripes [1], the outer divertor of ASDEX Upgrade is a net tungsten erosion area. The long-term evolution of tungsten surfaces at the outer divertor strike point of ASDEX Upgrade was investigated after the discharge campaign 2009 with 5275 plasma seconds by scanning electron microscopy, focused ion beam cross-sectioning, transmission electron microscopy, and various ion beam analysis techniques. The initial surface consisted of a 10 µm thick tungsten coating on fine-grain graphite deposited by the CMSII technique and showed a crystalline roughness in the µm range due to the growth of large tungsten grains and a larger-scale roughness due to the initial substrate roughness.
Despite the fact that net erosion of tungsten occurs, a complicated pattern of net erosion and net redeposition areas is observed on the microscopically rough surface. Net erosion areas are observed on faces of the rough surface inclined towards the magnetic field, while net redeposition is observed in shadowed areas, such as valleys. This pattern can be explained by the gyro-motion of eroding species due to the combined acting of the magnetic field and the electric sheath potential, which impinge preferentially on plasma-inclined faces, and the prompt redeposition of tungsten, which is almost homogeneously distributed [2].
The initially rough surfaces get considerably smoother due to preferential net erosion of tungsten on plasma-exposed faces and net redeposition of tungsten in the shadowed areas. Redeposited tungsten layers have a spongy, foam-like structure with a large number of pores, the pore diameters range from a few to a few ten nm. The redeposited layer reaches thicknesses of up to 3 µm. These redeposited layers incorporate also impurities, especially B, C, N, O and Fe. The consequences of these long-term morphology and compositional changes are discussed.

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Carbon erosion and deuterium retention of tungsten-doped amorphous carbon films exposure to deuterium plasma

P17B

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One of the most crucial plasma-wall-interaction issues in unclear fusion is tritium retention via tritium co-deposition with eroded carbon [1]. For a next-step device such as ITER, the use of carbon materials together with metallic plasma-facing materials (PFMs) will lead to cycles of erosion of the PFMs by hydrogen and subsequent deposition of mixed carbon-metal layers. These layers will be partly on areas subjected to further erosion during the lifetime of the plasma-facing components [2]. Therefore, the erosion behavior and deuterium retention of such mixed layers need to be investigated.

We investigated the erosion of tungsten-doped amorphous carbon (*a*-C:W) films in deuterium plasmas and the retention of deuterium in these films. *a*-C:W films with 0-6.5 at.% tungsten concentration were produced by magnetron sputtering. Films with different thicknesses varying from 0.3 to 1 μm were investigated. They were exposed to deuterium plasmas with incident ion energies of 30 and 100 eV per deuterium atom. The carbon erosion rates were determined from the carbon areal density changes measured by Rutherford backscattering spectrometry. The total amount of retained deuterium was obtained by the D(^{3}He,p)^{4}He nuclear reaction analysis at 700 keV.

Carbon erosion yields of tungsten-doped amorphous carbon films are clearly lower than the yields of pure amorphous carbon and decrease strongly with increasing W concentration. The carbon removal rates from *a*-C:W films decreases with increasing deuterium fluence due to tungsten enrichment at the surface. At lower deuterium energy (30 eV per D), the total removable amount of carbon saturates in all tungsten-doped films for D fluence above about 10^{24} m^{-2}. This means that further erosion of carbon stops. For 100 eV per D this saturation was only observed for W concentration higher than 6.5 %. For this higher energy we also observed a slight sputtering of W due to the presence of a small fraction (a few percent) of D^+ ions in the plasma which impinge on the surface with energy of 300 eV.

For 30 eV per D retention in 2.8 % *a*-C:W films saturates at a D fluence above 6×10^{23} m^{-2}. The D retention is in this case about 7×10^{20} m^{-2} which is noticeably higher compared to a pure tungsten film. D retention in 5 % and 6.5 % *a*-C:W films increases monotonically with increasing fluence. At higher deuterium energy (100 eV per D) retention is higher than at 30 eV D. For fluences above 2×10^{24} m^{-2} we find a decreasing D retention with increasing fluence. This is attributed to the observed tungsten sputtering.

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XPS and SIMS Analyses of Mixed Deposition Materials

P18A

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The generation of mixed deposition materials has been identified as a major topic of concern for ITER. JET provides a source of Be-C-W mixed materials, and in future mixed materials such as tungsten beryllides are expected to be generated in JET ILW. Compositional information will already be collected from continuing analyses programmes using Ion Beam Analysis techniques, SEM and EDS. However, these methods do not address other film properties such as chemical states, structure and thermal/electrical properties. Further techniques such as XPS/AES and SIMS are needed that may add information on film properties, either for predominantly carbon-beryllium films present on current JET tiles, or in preparation for surface films that may be expected from the Be/W wall now being installed.

Measurements were performed on post-mortem samples as parts of current JET tiles, as well as on mixed films that may by expected after the JET ITER-like wall operation. The mixed films were prepared using Thermionic Vacuum Arc (TVA) method based on the electron-induced evaporation [1]. The behaviour of these films at different temperatures is crucial for the fuel retention and release. For this purpose some graphite and silicon substrates were coated with Be, Be-W and Be-CW films and annealed at 273 K, 573 K and 673 K. Secondary ion mass spectrometry (VG IX70S double focusing magnetic sector SIMS) was applied for the depth profiling of the Be, O, C and W atoms, as well as BeO molecule. 5keV O_2^+ primary ions, (current 250 nA, sputtered area 300 x 220 μm^2) were used in the analyses. In order to infer Oxygen content and formation of BeO molecule Xe^+ primary ions (5keV, ion current 250 nA) were used. The sputtering rates were: 0.56nm/s (Be-W) and 0.73nm/s (Be-CW). The SIMS results were compared with XPS measurements and a good agreement was highlighted.

Acknowledgement: This work was supported by EURATOM and carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

[1]. C. P. Lungu, I. Mustata, V. Zaroschi, et al., Phys. Scripta 128, 157(2007)

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**See the Appendix of F. Romanelli et al., Proceedings of the 23rd IAEA Fusion, Energy Conference 2010, Daejeon, Korea

Micro-distribution of fuel and metal in carbon based plasma facing materials

P18B

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Most of present-day devices have plasma-facing components made of graphite and/or Carbon Fibre Composites (CFC) due to the excellent power handling capacity under high heat load. For ITER a metal wall is planned, except for the strike points were CFC is suggested, to minimize carbon tritium co-deposits

The aim of this work was to investigate the relation between deuterium and heavy impurities such as metals on the surface of CFC (e.g. NB41 which is an EU reference material for divertor tiles in ITER). Samples were exposed to deuterium plasmas in the TEXTOR tokamak in Jülich Germany. Thick deposits from TEXTOR and on CFC from Tore Supra at Cadarache France were also investigated.

The resulting distributions were determined by Nuclear Reaction Analysis (NRA), Rutherford Back Scattering (RBS) and Particle Induced X-ray Emission (PIXE). With a $^{3}He^{+}$ ion beam the reaction $^{2}D(^{3}He,p)^{4}He$ was used for deuterium, $^{12}C(^{3}He,p)^{14}N$ for carbon and x-rays induced by the incoming beam and back scattered particle for detection of heavier elements. Measurements were made both with a 1 mm beam and by a micro-beam focused down to 10 µm spot size and scanned over the sample to obtain maps of the different elements.

For samples from Tore Supra metal (i.e. steel components) distributions in the range of 1-7×10^{20} atoms/m^2 in the erosion zone of the toriodal pump limiter and 3-7×10^{20} atoms/m^2 within the accessible depth in the deposition zone. The simultaneous measured distribution of the deuterium concentrations were 6-23×10^{21} atoms/m^2 for the erosion zone and 3-25 ×10^{22} atoms/m^2 in the deposition zone. For samples exposed in TEXTOR the amount of deuterium after exposure was measured (~6×10^{20} D/m^2) and compared to the amount after laser cleaning (<5×10^{19} D/m^2). During this type of measurements any changes in the metal distribution can be used as an indicator of whether the substrate surface has been modified by the laser pulse or if only the deuterium content has been affected as desired.

Maps of the different distributions are presented and discussed for the different situations. Good knowledge of the local variation can help in the knowledge of both fuel retention and possible formation of small particles i.e. dust.

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Influence of Impurities on Deuterium and Helium Retention in Carbon Materials

P19A

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The first generation of the ITER divertor will employ carbon based plasma-facing materials for the strike point areas. The divertor target will be subjected to mixed species fluxes such as hydrogenic isotopes, beryllium eroded from the main wall, helium produced in DT reactions and argon seeded for cooling of the divertor plasma. It is necessary to test how carbon materials perform with respect to hydrogen retention under the realistic mixed species plasma conditions.

The PISCES linear plasma laboratory offers a unique test bed for material testing under the ITER-relevant conditions. Our earlier work [1] showed that the impurity seeding can significantly influence the deuterium retention in carbon materials. However, the effect of impurities depends strongly on exposure parameters, such as incident ion energy E_i and sample temperature T_s. The retention and its behaviour were similar in different grades of CFCs and fine-grain graphites.

The aim of this study was to investigate the influence of impurities on the deuterium retention under the variation of exposure parameters in a systematic manner. Samples of fine-grain graphite ATJ were exposed to plasmas containing (i) deuterium, (ii) deuterium and helium, (iii) deuterium and argon, (iv) deuterium, helium and beryllium and (v) deuterium, argon and beryllium. The fractions of He and Ar were kept at a level of 10% and of Be at 0.3%. The exposures for each species mixture were carried out at E_i of 35 and 120 eV and T_s of 470 and 700 K, except for both cases with Be, where targets were kept at 700 K to allow the formation of a carbide layer. For a direct comparison, all plasma runs were stopped at a total fluence of 1×10^{26} D/m^2. Thermal desorption spectrometry (TDS) resolving D_2 and He was used to measure the amount of retention in the samples.

The presence of He and Ar tends to reduce the amount of retained deuterium in comparison to the pure D plasma. The dependences of the D retention on E_i and T_s appear to be interconnected, so that no clear trend was observed by variation of either of both parameters. The beryllium seeding results in a redistribution of TDS peaks, pointing towards different D trapping mechanisms. However, the total amount of the D retention is only weakly affected by the addition of Be. The amount of retained He strongly increases with E_i. The presence of Be supports the He retention. E.g., for E_i = 35 eV and T_s = 700 K the addition of Be increased the amount of stored He by a factor of 40, which was then 5 times higher than the amount of D in the same sample.

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P19B

Chemistry of the first wall of future fusion devices: Interaction of energetic oxygen ions with the beryllium tungsten alloy Be_2W

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The three elements beryllium, carbon and tungsten are going to be used as armour material covering the inner wall of the vacuum vessel of the future fusion device ITER. Contamination of the surfaces and the plasma by oxygen is inevitable after the vacuum vessel is exposed to air for e.g. maintenance. Plasma processes lead to formation of ‘mixed materials’ due to erosion, transport and redeposition. Newly formed phases show different physical properties compared to the original surface, e.g. melting point, thermal conductivity and hydrogen retention. Therefore fundamental research of solid state reactions is required. Previous studies have dealt with various binary systems containing Be, C and W. Binary systems have been subject to various experiments. [1] Next step is increasing the number of reaction paths by adding more elements.
In this study, the interaction of energetic oxygen ions with the beryllium tungsten alloy Be_2W is investigated by depth-resolved X-ray photoelectron spectroscopy (XPS). [2] The sample is prepared in-situ in the preparation chamber ‘LAICA’ which is directly interconnected to the analysis chamber ‘SurICat’. A 4 nm tungsten layer is deposited on a beryllium plate. The sample is alloyed at 900 K for 60 min. The O-implantation energies for these experiments are chosen such, that implantation range is still accessible via depth-resolved XPS. According to simulations with SDTrim.SP implantation energies of 500 and 1,000 eV satisfy the experimental conditions for O-fluences of $5 \cdot 10^{14}$ cm^{-2}. Between the single implantation steps the sample is heated for 30 min at 600 K. Photoelectron spectra are taken after each experimental step. High resolution spectra of core levels of every element present are taken at up to six different information depths. In addition survey scans at two different photon energies are performed.
The W 4*f* and Be 1*s* spectra reveal formation of beryllium tungstate $BeWO_4$ already at room temperature (r.t.) in significant amounts due to energy deposition of implanted oxygen ions. The spectra of the W 4*f* region show formation of $BeWO_4$ mainly near the surface. This is in qualitative agreement with the simulated implantation depth profiles. No indication for oxygen diffusion at r.t. is provided. In previous experiments in a Be-rich environment the ternary compound has proven to be stable to temperatures up to 1,100 K. Contrary to these results, annealing at 600 K already leads to decomposition of $BeWO_4$. This decrease of decomposition temperature is attributed to the excessive amounts of Be. After decomposition at 600 K oxygen stays in the sample bound as BeO.

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Deuterium retention in co-deposited C-W-D films formed in a magnetron deuterium plasma discharge

P20A

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Co-deposition of tritium with sputtered wall material in the inner divertor and in remote areas is considered as the main channel of tritium accumulation in fusion devices with a carbon wall. A high concentration of hydrogen isotopes is usually observed in C-H codeposits. A mixed carbon and tungsten divertor is planned for ITER, however hydrogen retention in mixed carbon/tungsten codeposits is not well investigated.

Deuterium retention in mixed carbon-tungsten-deuterium (C-W-D) films co-deposited during a deuterium magnetron discharge running with a mixed carbon-tungsten cathode was studied in this work. Films were deposited on molybdenum and graphite substrates installed in various positions in the discharge chamber. This resulted in films with different C:W:D ratios. Some substrates were grounded, while other substrates were biased with a negative potential. The temperature of the substrates was not higher than 100 °C. Plasma parameters were measured with a Langmuir double probe.

Films morphology and composition were analyzed by secondary electron microscopy and ion beam analysis. The tungsten concentration was determined from Rutherford backscattering (RBS). Deuterium and carbon concentrations were measured by nuclear reaction analysis (NRA) using $D(^{3}He,p)^{4}He$ and $^{12}C(^{3}He,p)^{14}N$ reactions. The concentration of other contaminations was derived from RBS spectra of the films on the graphite substrates. Typically few percents of O were observed, while other elements were below the detection limit. Thermal desorption spectroscopy (TDS) of samples was performed in an UHV TDS stand with temperatures rising up to 1300°C.

The ratio of W:C in the films varied from 0.1 to 1. Total deuterium retention increased both with C and W content, but the ratio D/(C+W+D) remained at a level of 20-30 %, which is only slightly less in C-D films.

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Fuel Retention in Mixed Tungsten / Carbon Layers

P20B

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Deuterium retention in mixed layers deposited on surfaces of the three ITER plasma facing material (PFMs) has been investigated for all relevant binary material combinations. However, only few data exist on the influence of the layer structure, composition and thickness on the D-retention in both the layers and the deep bulk regions below. Influence of simultaneous impact of D and He (typical of a DT fusion reactor) on deuterium retention in such systems is also an issue.

Co-deposited tungsten and carbon layers have been deposited using a specific magnetron sputtering design [1]. Experimental parameters such as the carbon / tungsten deposition rates, the incident particle energy, and the substrate temperature could be investigated separately. Deposition was carried out using a mixture of deuterium and argon as process gas. Detailed characterizations of the mixed tungsten/carbon films have been carried out by in situ X-Ray Photoelectron Spectroscopy (XPS) and Secondary Electron Microscopy (SEM). These films were exposed to different plasma conditions in the Pilot-PSI linear device. Samples were exposed to pure deuterium and mixed D/He plasmas to study the effect of the incident flux and surface composition (C/W ratio) on the retention. The latter was measured by thermal desorption spectroscopy (TDS).

The W/C films could be deposited with or without deuterium gas process. The deuterium retention in such co-deposited layers was investigated as a function of the deposition conditions and film composition.

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Effect of atomic Hydrogen and atomic Nitrogen densities near substrate surface on microstructure of plasma nitrided layer

P21A

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The use of nitrogen seeding to reduce the edge plasma temperature has raised the question of interaction of nitrogen with a tungsten first wall [1]. In presence of hydrogen, in addition to nitrogen, these interactions become even more complex. The present investigation reports the effect of ratio of atomic hydrogen and atomic nitrogen density on the microstructure of the plasma nitride layer. An optical emission spectroscopic (OES) investigation was carried out on N_2-H_2 plasma for typical nitriding conditions. EN41B, SS8 and SS410 substrate samples were plasma nitrided for different N_2-H_2 gas mixtures. It was observed that atomic nitrogen and $H_{\alpha,}$ were the main species in close proximity to the substrate surface [2]. The atomic nitrogen and atomic hydrogen density depends on dissociation rate of N_2^+ and H_2^+ ions respectively that are reaching the substrate with translational energy gaining from the ion sheath. From qualitative analysis of the emission spectral lines near the cathode surface the concentration of [N] and [H] was estimated and established that microstructure of nitrided layer has direct correlation with availability nitrogen and hydrogen atoms on the substrate surface.

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Glow-Discharge Optical Emission Spectroscopy for Plasma-Surface Interaction Studies

P21B

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Glow-discharge optical emission spectroscopy (GDOES) is a technique to measure depth profiles of constituent elements in a solid sample by detecting emissions from atoms accommodated in plasma by sputtering of sample surface. In most cases, Ar is used as plasma work gas. High flux and low energy incident Ar ions provide large sputtering rates (several tens of nanometers per second) with small sample damages. Depth resolution is few nanometers. These characteristics of GDOES are in general suitable for analysis of deposited layers on plasma facing materials (PFMs) and measurements of depth profiles of implanted particles. There are, however, several particular requirements in the field of plasma-surface interactions (PSI). In this paper, we examined the applicability of GDOES for PSI studies.

(1) To distinguish hydrogen isotopes

Secondary ion mass spectroscopy (SIMS) is commonly employed for this purpose. However, secondary ion yield is strongly dependent on nature of chemical bond in material (ionic, covalent and metallic), and hence this technique is not suitable for analysis of interface between dissimilar materials such as deposited layers and PFMs. GDOES is free from this type of problem, and resolution of its grating is sufficiently high to separate H emission from that of D. However, isotopic measurement by GDOES has been scarcely reported. We prepared H and D containing oxide layers on metal substrates by oxidizing in H_2O and D_2O as model samples, and analyzed with GD Profiler 2 (HORIBA Jobin Yvon Co.) by adjusting optics arrangements to detect H or D emission. H and D could be clearly distinguished from each other, and detailed profiles of hydrogen isotopes at oxide-metal interfaces could be measured.

(2) He measurement

Measurement of He with GDOES is relatively difficult because the energy for He excitation ($>$ 20 eV) is high compared with the first ionization potential of Ar (15.8 eV). Hence, we used high power, high pressure Ne plasma because of higher ionization potential of Ne (21.6 eV). The depth profile of He implanted into tungsten at 8 keV up to fluence of 3×10^{21} m^{-2} at room temperature was successfully measured with large diameter anode.

From these results, we concluded that GDOES is a powerful tool for PSI studies. Details of the above-mentioned measurements will be reported in the presentation.

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Quantification of Tungsten Sputtering at W/C Twin Limiters in TEXTOR with the Aid of Local WF_6 Injection

P22A

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Tungsten is the most promising material candidate for plasma-facing components at the divertor strike-point area of ITER in the activated operation phase. Apart from the critical issue of power handling, in particular under impact of uncontrolled transients described in detail in [1], the most important issue for the qualification of W as plasma-facing material is the quantification of the W source strength, which is connected to the components lifetime and, finally, to the W core accumulation. Thereby W sputtering is dominated by impinging intrinsic impurities such as O or C or extrinsic species for seeding like N and not by the fuel species itself.
A dedicated experiment to study the W and C sputtering under plasma impact was carried out in TEXTOR with the aid of a spherical twin limiter with one half made of W and the other of C. The limiter was installed in the TEXTOR vacuum lock system, positioned in the near scrape-off layer and exposed to a set of plasma discharges with variations of edge plasma parameters owing to strong gas puffing: T_e from 25eV to 85eV and n_e from $5.0x10^{18}m^{-3}$ to $1.5x10^{19}m^{-3}$. 180° rotation of the limiter between discharges allowed a direct comparison of W and C sputtering, local transport and W/C mixing. Impurity fluxes and penetration depths were measured by optical spectroscopy with the aid of 2D cameras with interference filters, an imaging spectrometer as well as a high resolution survey spectrometer (363nm-720nm).
Access to the W impurity influx is given by spectroscopy of neutral and singly ionised W of which the corresponding emission lines are preferable in the ultraviolet and blue spectral region [2]. Experimental and modelled photon efficiencies have so far been only available for T_e<20 eV [3], whereas in the range of 20eV-100eV no experimental data exist so far. In order to calibrate in-situ the W photon fluxes of different spectral lines, WF_6 has been injected in a second experiment into the plasma boundary through a gas inlet installed in the vacuum lock system. WF_6 is quickly dissociating when entering the plasma and the emission pattern of different W lines (e.g. WI at 400.9nm, 505.3nm, 522.5nm etc.) has been recorded. Effective inverse photon efficiencies, so-called $\left\| S/XB \right\|_{WI}^{WF_6 \Rightarrow W}$ for different lines have been determined simultaneously, thereby a constant value of 100 for the prominent WI line at 400.9nm has been measured for T_e>50eV which is in good agreement with GKU modelling.
These experimental photon efficiencies have been used to calculate the W sputtering flux in the twin limiter experiment as function of local T_e and n_e. The W sputtering yield decreases by factor 4 with decrease of T_e from 85eV to 25eV and simultaneous increase of the impinging deuterium flux by 50%. Contrary, the C sputtering yield remains almost unchanged under identical experimental conditions.

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P22B

Sputtering of tungsten surfaces by N^+ and N_2^+ ions – investigation of transient and molecular effects

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Plasma edge cooling by nitrogen seeding is used in present fusion devices to improve the overall plasma performance and reduce erosion of plasma facing components especially in full tungsten devices like ASDEX Upgrade. When a plasma containing nitrogen interacts with a tungsten surface the formation of a tungsten-nitride surface layer within the ion penetration depth was observed [1]. As a result of the nitrogen accumulation in these layers a reduced partial sputtering yield of tungsten as compared to pure tungsten surfaces was found.
With a highly accurate quartz crystal microbalance (QCM) developed at the University of Technology in Vienna [2] we have determined the total sputtering yield of tungsten (at 460 K) during both the bombardment of N^+ atomic and N_2^+ molecular ions. Nitrogen ions in an energy range of some hundred eV up to 2 keV were produced in an ECR ion source and selected according to their m/q ratio in a sector field. A SC cut quartz coated with a 500 nm tungsten film was exposed to the beam. The change of the resonance frequency of the quartz is a direct measure of the total mass change of the surface film. In this way, total sputtering yields can be determined in situ and thus transient effects such as the formation of a modified surface layer upon ion impact can be studied.
In our measurements we observe that the sputtering yield of tungsten decreases exponentially with fluence, which confirms that a transition from a pure tungsten to a nitrogen enriched layer takes place. A steady state sputtering yield of roughly one third of the initial sputtering yield is reached after several 10^{15} ions per cm^2. The steady state sputtering yield corresponds well to results of dynamic TRIDYN simulations with a maximum nitrogen concentration of 50% in the tungsten-nitride layer [1].
Furthermore when comparing steady state results from atomic N^+ bombardment to molecular N_2^+ sputtering a distinct molecular effect was found, i.e. the sputtering yield per nitrogen atom for N_2^+ molecular ions is approximately 10% higher than that of two equally fast N^+ ions. In the investigated energy regime a N_2^+ ion can thus not be considered as two independent nitrogen atoms impinging on the surface. In the molecular case the overlapping collision cascades have higher energy densities than those initiated by the atomic ions, leading to a sputtering enhancement in dense cascades [3].

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Self-passivating bulk tungsten-based alloys manufactured by powder metallurgy

P23A

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Tungsten is presently the main candidate material for the plasma exposed areas of future fusion reactors (DEMO) due to its low erosion yield by sputtering with plasma particles and its good thermo-mechanical properties. However, the use of tungsten as first wall material implies an important safety concern in case of an accident with loss of coolant and air ingress into the reactor vessel. In this situation, the high temperatures achieved in the in-vessel components within 10 to 30 days due to the decay heat [1] would lead to a strong exothermic reaction owing to tungsten oxidation with the release of volatile radioactively activated tungsten oxides.

A possible way for avoiding this important safety issue would be the addition to tungsten of alloying elements forming stable oxides, in such a way that at high temperature in the presence of oxygen a self-passivating layer is formed protecting material from further oxidation. In previous works [2] different binary and ternary tungsten alloys have been manufactured via magnetron sputtering, demonstrating that tungsten thin films containing 10 wt.% Si and 10 wt.% Cr exhibited excellent self-passivating behaviour when exposed to air at temperatures up to 1000°C. However, the PVD route is not applicable to DEMO because for the blanket first wall coatings or tiles with a thickness of several mm are required.

Powder metallurgy is a suitable route for the production of bulk tungsten alloys with tailored composition and microstructure while being a route of relatively low cost. In this paper, results on the manufacturing of W-Cr-Si and W-Cr-Ti alloys by powder metallurgy (mechanical alloying (MA) + hot isostatic pressing (HIP)) are presented. First oxidation tests performed on preliminary W-Cr-Si bulk samples produced in previous work [3] with starting powders milled in a SPEX mill, demonstrated a similar self-passivating behaviour than W-Cr-Si thin films. In this work MA was performed in a planetary ball mill. Different MA parameters were studied to find the best balance between lowest possible amount of contaminants and effective milling. After HIPing, densification close to 100% was obtained. The microstructure and the existing phases were observed by FEG-SEM, FIB cross sectioning, EDX mapping and XRD, and the mechanical properties were explored by micro- and nano-indentation. Furthermore, the thermal conductivity of some samples was measured as a function of temperature.

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Oxidation behaviour of silicon-free tungsten alloys for use as first wall material

P23B

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Self-passivating tungsten-based alloys have been developed as first wall materials to avoid the formation and evaporation of activated tungsten oxides in a future fusion power reactor in case of a loss-of-all-coolant-accident with coexistent air ingress. The current approach at IPP is to use alloyed tungsten with the intention to ensure the formation of a passivating (i.e. adhesive and tungsten diffusion-tight) oxidation layer out of the alloy, which leads to a reduction of the oxidation rate of about three orders of magnitude compared to pure tungsten in the temperature-regime between 600 and 1000 °C. This behaviour is observed on films of W-Cr-Si alloys, which show strongly decreased oxidation rates, when compared to thin layers made out of pure tungsten [1, 2].

The elements currently used for the alloying of the tungsten involve silicon, which renders the formation of WSi_2 or W_5Si_3 at elevated temperatures possible. These tungsten silicides could compromise the mechanical processabilitiy of the resulting material because of intermetallic-phase brittleness. As for bulk tungsten or tungsten alloy-material the mechanical properties are crucial both during construction and operation of a fusion reactor, a search for alternatives is currently undertaken at IPP.

Investigations on potential chemical compositions for the self-passivating alloys have been performed. A system containing tungsten, chromium and titanium was identified as a promising path to follow during further work on this topic. These types of silicon-free tungsten alloys exhibit similar or even slightly superior performances in respect of thermal stability of the oxide-layer and the reduction of the oxidation rate when being compared to the classical W-Cr-Si. Furthermore, as a decisive advantage of these silicon-free tungsten alloys, they may show a higher ductility, less brittleness and therefore a better mechanical processability, than their silicon containing counterparts.

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Morphology, Structure and Composition of Dust Formed in Globus-M Tokamak

P24A

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Morphology, structure and composition of graphite (RGTi) in-vessel plasma facing components and dust formed during operation of Globus-M spherical tokamak were studied by means of scanning electron microscopy (SEM), X-ray diffraction (XRD) and electron probe microanalysis (EPMA), respectively. Currently about 90% of the inner vacuum vessel (austenitic stainless steel) surface area is covered by RGTi tiles (614 pieces) with sizes of 12×12×1 cm^3. The tiles were manufactured from recrystallized graphite doped with 2 at.% Ti and 0.3-0.7 at.% Si and were protected by B/C:H films obtained by boronization from carborane $C_2B_{10}H_{12}$ (7-8 procedures per year, ~30 in total). The dust was collected from bottom divertor tiles exposed to direct plasma impact as well as from shadowed zones located underneath them after about 10000 pulses (~1000 s). The typical plasma pulse in Globus-M is characterized by high power deposition onto the RGTi tiles (up to 5-10 MW/m^2). It was observed dust of two kinds: carbon and metallic.

As follows from SEM, the carbon dust collected from the bottom divertor tiles had a shape of broken stones (~10 μ) that was caused by thermocycling and brittle fracture of graphite under interaction with plasma. Identical thickness (~1 μ) of graphite debris could be connected with deuterium embrittlement of surface layers due to deuterium accumulation at a depth about 1 μ. Deuterium content in the tiles was found to be of 10^{22} D/m^2 [1]. XRD and EPMA showed that the dust consisted of well graphitized and amorphous carbon and TiC inclusions. The amorphous phase could be formed as a result of chemical (or physical) erosion of carbon and its subsequent redeposition on the tiles. The crystalline sizes of 100-300 nm were the same for initial and damaged graphites.

The carbon dust collected from the shadowed zones consisted of smooth spheres (10-30 μ), free-form dense agglomerates and sponge agglomerates with size up to 100 μ. Formation of the carbon spheres is probably connected with their nucleation on metallic particles and catalytic growth in plasma. Ti, Cr, Fe, and Ni were found in the spheres. In any case the spherical shape indicates that a particle of dust was presented in plasma for some time. The free-form and sponge agglomerates composed of B, C, and O and could be formed during boronization or consequent discharges with condensation of gaseous products onto primary dust.

The metallic dust (so-called “metallic ice”) collected from the shadowed zones had a shape of plates with sizes of 10-1000 μ and thicknesses of 2-4 μ and contained mainly Fe, Cr, Ni, Ti (stainless steel components) and O. The steel was oxidized and quantity of ferrite phase was larger than in the initial steel. There were a lot of traces from microarcs that resulted in steel melting and cohesive tearing-off the plates.

This work is supported by the Presidium of the Russian Academy of Sciences (program No. 27; 2010).

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Characterization of mixed material dusts in JT-60U

P24B

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Dust control is an important issue related to the accumulations of radioactive impurities in core plasmas and the tritium retention in thermonuclear fusion devices with magnetic confinements. Thus, for understanding characteristics of dusts, their performance and compositions have been investigated in tokamaks and stellarator/heliotrons.
The dust collections in JT-60U were carried out in 2010 by a polycarbonate membrane filter, which is a polymer film with pores of 0.1 μm. Dusts at plasma facing areas, underneath the plasma facing tiles, divertors and baffle plates in different toroidal section have been investigated. 12 tungsten-coated CFC tiles have been installed at the upper part of the outer divertor since 2005, and irradiated by the divertor plasma. These tungsten coatings less than 50 micron were made by the vacuum plasma spray (VPS) method. Compositions of the collected dusts are analyzed by scanning electron microscope (SEM) with energy dispersive X-ray spectrometry (EDX) and X-ray photoelectron spectroscopy (XPS).
An analysis of carbon-tungsten mixed dusts in JT-60U has been difficult with EDX, because the peak of the tungsten is located close to that of silicon. EDX results by a curve fitting showed that the observed peak is the combination of those of tungsten and silicon. The purpose of this study is to enhance resolution of the XPS measurement to analyze quantitatively the composition of the dust flakes. In this study, an indium film was newly used as the substrate instead of carbon tape to reduce the silicon background.
Large dust flakes of several 100 microns collected at P-8 under the dome were selected for the analysis to obtain sufficient intensities for XPS. XPS measurements of the dust flakes showed signals of the plasma facing materials, such as carbon, tungsten and iron with clear tungsten peaks. The new analysis showed that the dust flakes contains about 1% of tungsten in carbon. Compositions of tungsten-carbon mixed dusts at different poloidal positions will be reported using the present technical method.

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Influence of the plasma and surface conditions on the formation of dust on plasma exposed graphite

P25A

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The issue of gross and net erosion of carbon under ITER relevant plasma conditions remains an open question, especially the importance of local re-deposition and the structure of the deposits formed under such conditions. It has been observed previously [1] that thick co-deposits are readily formed on the surface of graphite targets exposed to ITER-relevant hydrogen plasmas in Pilot-PSI. These co-deposits consist of cauliflower-like dust particles and, surprisingly, accumulate in the region exposed to the peak particle and heat flux. The diameter *D* of the particles varies from sub-micrometer range to ~60 μm. The dust formation mechanism as well as the influence of the plasma conditions on the dust production requires further studies.

Polycrystalline graphite targets were exposed to hydrogen, mixed hydrogen/argon and nitrogen plasmas in Pilot-PSI. The influence of the electron temperature, ion energy, plasma composition and surface temperature on the evolution of the surface morphology was investigated. In addition, CH_4 injection through a molybdenum target was carried out to study the influence of the substrate material on the re-deposition process. SEM and TEM techniques were used to characterize the particle structure, size distribution, surface coverage and get some insight into the formation mechanism.

Observations show that increasing the ion energy in hydrogen plasma prompted formation of large (D>30 μm) dust particles, which are not detected on the surface of floating targets. Addition of argon into the hydrogen plasma beam, on the other hand, shifted the particle size distribution towards smaller values (D_{max}≤10 μm). Co-deposits formed during CH_4 injection experiments are similar to that observed on plasma-exposed graphite surfaces. In addition to cauliflower-like microparticles, TEM analyses of graphite targets revealed the formation of spherical nanoparticles and chains of nanoparticles consisting of agglomerations of graphitic nuclei (10-20nm). This strongly suggests formation of dust particles in the plasma.

In pure nitrogen plasma surface morphology of graphite evolves completely differently. Pyramidal structures were formed on the surface of floating graphitic targets. Some of them have nano-scaled objects attached to their tops. Negative biasing of the target led to growth of densely-packed filaments (~10-50 nm in diameter). At the same time, no spherical-shaped dust microparticles were observed in nitrogen.

These results point out the possibility of dust formation in the plasma of ITER divertor and its accumulation on the target surface.

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Characteristics of different dust sampling techniques in view of the classification of particles from full-tungsten ASDEX Upgrade

P25B

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Quantification of the impact of dust produced by plasma-wall interaction processes in fusion experiments in order to assess the impact of different types of dust on operation and safety of a future fusion power plant requires detailed investigation of the physical and chemical properties of these classes of dust.
A first preliminary diameter based classification of about 500 arbitrarily selected dust particles collected on silicon wafers during the 2008 ASDEX-Upgrade campaign was presented in [1] revealing B, C and W as dominating elements. This classification scheme was considerably refined to a total number of 7 classes after careful analysis of the outer morphology and elemental composition of more than 50000 dust particles trapped on 5 silicon wafer collectors during 2009 campaign [2]. The therein described spheroid metallic particles were also observed in close vicinity of vacuum arcing sites [3] indicating these particles originate from local melting triggered by both, off-normal plasma events and arcing. The sectioning of several particles typical for each of the 7 identified dust classes via focussed Ga ion beam revealed inner cavities and sponge like structures stressing the need to quantify also the porosity for each dust particle class [4]. Subsequently a comparative study of the outer morphology and elemental composition of particles trapped during three consecutive campaigns of ASDEX-Upgrade with full tungsten first wall configuration on silicon wafers was carried out introducing a more detailed particle classification scheme using non-dimensional parameters to describe the shapes of particles attributed to each class. The influence of off-normal events such as loss of coolant accidents and first wall boronization on the dust composition was assessed. In this study three different sampling methods are compared using this refined classification: filtered vacuuming technique, adhesive carbon tape method and silicon wafer collection at the example of 2009 AUG campaign samples. The particle size selectivity and the sampling sensitivities for each dust particle class are discussed and at the example of tungsten dominated spheroids the variation of estimated accumulated mass in this campaign is assessed. Furthermore, an online service for the analysis of scanning electron microscopy images of particles developed in close cooperation with IAEA is presented and a proposal for a common standard on dust classification is made to guarantee the production of data sets directly usable as input for dust transport and radiation energy loss studies.

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First results from dust detection during plasma discharges in Tore Supra

P26A

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During tokamak operation, material from plasma facing components is eroded, transported and finally re-deposited in the form of layers that can trap a significant fraction of the plasma fuel, whose inventory must be limited in future fusion devices. These layers produce flakes and dust that can penetrate the plasma discharge, sometimes leading to disruptions. In-vessel dust has been recognized as a safety and operational issue for next step devices such as ITER even though this is not the case in present day tokamaks [1]. As a consequence, it is essential to develop methods for dust detection and dust removal. At present dust measurement techniques are still immature.
An electrostatic dust detector has been developed by Princeton Plasma Physics Laboratory [2] to detect dust particles on a remote surface. With its ability to perform long pulses, Tore Supra offers a unique opportunity to test this detector. Therefore a set of electrostatic dust detectors has been installed in one of the pumping ducts of the Tore Supra vacuum vessel.
This detector is made of a fine grid of interlocking circuit traces biased to 50V. Every conductive particle which reaches the sensitive surface creates a temporary short circuit, before the particle is ejected or vaporized by the current flowing across. The resulting current pulses are recorded and sorted by level and duration. The pulse duration can be related to the size of the dust particles. This detector provides real-time information on dust generation and a preliminary correlation of the dust signals with plasma events has been obtained.
One of the detectors has been covered with a film of Kapton to protect its sensitive surface from dusts. It is used as a reference to discriminate pulses created by particles from electrical pickup.
The good correlation which exists between the dusts detected by the electrostatic detector and the particles observed on the visible CCD camera will be shown. The first results show that dust particles are detected systematically after disruptions, and from time to time during plasma current ramp up and during plasma current plateau.
After disruptions, we observed that dusts are measured for up to 5 seconds, and their number can reach several hundreds for a detector sensitive surface of ~25 cm^2. Finally an evaluation of the dust collected over the plasma campaign in late 2010 will be presented and discussed.

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ITER In-Vessel Dust and Tritium Control Implementation Status and Plan

P26B

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In order to obtain the licensing for construction and later for operation, ITER has to demonstrate that the radioactive doses to the staff and to the public and the environmental releases in normal and postulated accidental conditions are minimized and well within the general safety objectives.
The main potential sources of such doses are the activated dust produced and accumulated in the vacuum vessel, the tritium retained there and the activated corrosion products that could be mobilized during an in-vessel accident of water coolant (LOCA).
A systematic safety analysis has assessed the radiological risks for ITER in normal conditions (e.g. chronic release during in-vessel maintenance) and accident scenarios (e.g. mobilization of inventories and releases through the various confinement barriers in the event of LOCA or a hydrogen/dust explosion with air). Maximum admissible inventories of in-vessel dust and tritium have been fixed.
The strategy for control of the in-vessel dust/tritium inventory is mainly based on their measurement and removal.
All this has been presented in the Preliminary Safety Analysis Report that is being assessed by the French Safety Authorities. Limits and strategy are commitments for ITER: it has to be demonstrated continuously that the machine is operated inside such limits and that sufficient margins are maintained.
The paper will recall the strategy adopted and will focus on the relevant tools, now part of ITER baseline design, foreseen to control dust and tritium inventory. In particular the paper will concentrate on the R&D results obtained so far on some systems such as the various kinds of diagnostics for measurement, the removal techniques particularly that based on baking of the divertor at 350°C and the tritium tracking procedure. The ITER In-Vessel Viewing System (IVVS) that allows for in-vessel inspection of plasma facing surfaces to look for possible damage caused during plasma operations, shall also be used for metrology measurements of the plasma chamber and its components and the main results will be presented too.
The relevant uncertainties will be pointed out as well as the main lines of further R&D necessary to validate tools and procedures.

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P27A

Advances in Dust Detection for Tokamaks

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Dust production in next-step magnetic fusion devices will be significantly higher than in contemporary devices due to the more intense plasma wall interactions and the increase in erosion levels. Methods to measure the inventory of dust particles and to remove dust if it approaches safety limits will be required in next-step tokamaks such as ITER. An electrostatic dust detector, based on a 5 cm x 5 cm grid of interlocking circuit traces biased to 50 V, has been developed to detect dust on remote surfaces. Impinging dust particles created a temporary short circuit and the resulting current pulse was recorded by counting electronics [1]. The total number of counts was proportional to the mass of the impinging dust. The detector was calibrated with carbon and lithium particles in air and vacuum environments and was successfully tested for the first time on the National Spherical Torus Experiment (NSTX). The sensitivity in vacuum for carbon particles with a count median diameter of 2.14 μm was found to be 0.15 ng/cm^2/count.

Typically 90% of the total number of particles that landed on the detector are vaporized by the current pulse and ejected from the detector, however about 10% may remain on the surface of the detector [2]. These may produce signals at a later time, complicating efforts to correlate the dust signal with plasma events. A helium puff system was developed to clear the residual dust from the electrostatic detector and any incident debris or fibers that might cause a permanent short circuit. Helium puffs were delivered by three nozzles of 0.45 mm inside diameter. An angle of 30º with respect to the surface of the detector and a helium backing pressure of 6 bar was found to be the optimal configuration. Two consecutive helium puffs cleared carbon particles from the entire surface of the detector.

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Carbon deposition on beryllium substrates and subsequent delamination

P27B

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Beryllium and carbon are foreseen as materials for plasma facing components of future fusion devices. Erosion, re-deposition and annealing arising from heat-load events during reactor operation will produce mixed material layers and compounds on the plasma facing surfaces, leading to changes in local melting point, sputtering behaviour, hydrogenic species retention and dust formation due to delamination.

In order to mimic the erosion/deposition processes, carbon layers have been evaporated onto beryllium plates and annealed in the 373 to 1073 K range for 90 min. Ion beam measurements revealed a smooth beryllium and carbon interdiffusion at the samples surface up to 773 K. A carbide formation reaction front became apparent for higher temperatures in scanning electron microscopy observations, with the volume fraction of Be_2C crystals resulting also evident in X-ray diffraction patterns. The annealing treatments induced delamination of large surface areas through telephone cords blistering attributed to strain energy release. At 1073 K cracking occurred preferentially along blister boundaries. This fracture behaviour seems caused by the different thermal expansion coefficients of the phases. The results show that delamination of re-deposited layers in PFCs is a natural mechanism of dust formation.

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Computer Simulations of Plasma-Carbon and Lithiated Carbon Interface

P28A

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We have studied chemical sputtering, hydrogen retention and other properties in the least known, intermediate-to-low range of impact energies (1-30 eV), with H, D, T atomic and molecular projectiles of hydrogenated amorphous carbon surfaces [1]. Successfully combining atomistic, molecular dynamics (MD) modeling using massive computation with "in house" particle beam experiments [2] provided a benchmark for the MD approach, enhancing its predictive capabilities. The best results are achieved when preparation of the hydrogenated carbon target surfaces mimics the particular experimental conditions as a function of impact fluence, energy and type of projectile species and of the rovibrational state (when applicable). The energy distributions of ejected molecules confirm the partial thermalization of the impact cascade. Sputtered hydrocarbon molecules have rovibrational energies in the range 1.5-2 eV, with relatively "cold" translational and rotational motion, close to 0.5 eV. In contrast, translational and rovibrational energies of sputtered deuterium molecules are close to 1 eV, with approximate equipartition between rotational and vibrational modes**.** We also show that number of terminal hydrocarbon moieties created by cumulative bombardment at the interface exhibits only a weak dependence on the isotopic mass. However, the sputtering yields exhibit a clear mass dependence which originates in the increased energy transfer for increasing projectile mass [1,3]. To study synergetic effects of plasma irradiation, we extended the MD simulation using predefined distributions of impact particles in energy, angle, and angular momentum, leading to encouraging comparison with available plasma-irradiation experimental data. Finally, for carbon mixed with a highly polarizable lithium, we were able to explain the experimentally found specifics [4] of the chemistry and dynamics for deuterium bonding in lithiated/oxidated carbon, employing the quantal Self-Consistent-Charge Density-Functional Tight-Binding (SCC-DFTB) method [5]. This opens new prospects for improved simulations of the mixed-materials plasma-surface interface.

We acknowledge support of DOE OFES, NICS, NCCS and LDRD program of ORNL.

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Simulations of low energy D bombardment of W and WC alloys

P28B

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Tungsten (W) is expected to be one of the basic plasma facing materials in ITER and future fusion devices. Due to plasma wall interactions, such as sputtering and re-deposition processes, tungsten-carbides are expected to be present in the reactor walls. The aim of our work is to simulate these plasma-wall interactions in order to study sputtering and fuel retention processes, as well as the damage caused in the W-based materials by the bombardment of low energy D.

Molecular Dynamics (MD) has been used for cumulative bombardment of 3000 D ions, with the following energies and substrates: 10eV, 30eV, 50eV, 100eV, 200eV and 300eV for pure W, using 2 different parametrizations of the interatomic potential [1,2]; 10eV, 20eV, 30eV, 50eV and 100eV for tungsten-carbide (WC), with both W and C surfaces; 20eV and 100eV for amorphous W2C, and 10eV, 20eV, 30eV, 50eV and 100eV for crystalline W2C, again with both W and C surfaces.

As expected, W sputtering is zero below its physical sputtering threshold (~340eV). This result confirms that, so far, the code simulates properly the sputtering behavior of W-based materials. The C and C-cluster sputtering in WC shows a peak around 30eV-50eV; detailed analysis of the atom trajectories showed that the erosion can be explained by the swift chemical sputtering mechanism [4], verifying that other sputtering mechanism than the physical one is important also in WC at low energies. To get a better picture of the damage and retention of hydrogen isotopes in the material, D trapping and the release of D-clusters have also been analyzed.

The aim for the near future is to continue simulating the sputtering and damage processes in the plasma-facing materials; different mixtures of W, C and Be, irradiated by different plasma components and impurities (D, Ar, C...). The data obtained can be used in simulation codes in need of detailed information about sputtering behavior of materials in the low energy regime.

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P29A

Simulation for Dynamical Process of Hydrogen Retention in Graphite Material under Continuous Injection

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The divertor plate of nuclear fusion reactor, which consists of graphite tiles or carbon fiber composites, is bombarded with hydrogen plasma. The hydrogen plasma erodes the divertor plate, yielding H_2 and other hydrocarbon molecules such as CH_x and C_2H_x, which are undesirable impurities in plasma confinement experiments. To understand the nature of chemical and physical interactions between hydrogen plasma and the divertor plate, it is important to clarify the elementary processes of the reactions.

Molecular dynamics (MD) simulation with modified Brenner’s reactive empirical bond order (REBO) potential [1] is a powerful tool to investigate plasma wall interaction on divertor plates in a nuclear fusion device. However, the size of MD simulation box is generally set less than several nm because of the limits of a computer performance. To extend the size of the MD simulation, we are developing a hybrid simulation code between MD code using REBO potential and binary collision approximation (BCA) code. Using the BCA code instead of computing all particles with a high kinetic energy for every step in the MD simulation, considerable computation time is saved. It is possible to simulate submicrometer size materials by the hybrid simulation code.

In our previous researches [2, 3], we investigated the interactions between an ideal graphite crystal and single hydrogen atom by the hybrid simulation. In those researches, the hybrid simulation code could not treat the continuous injection of hydrogen atoms.

In this paper, we improve the hybrid simulation code to treat the continuous hydrogen injection into a graphite material. The continuously injected hydrogen atoms destroy the bonds between carbon atoms. As the result, the ideal graphite material becomes an amorphous hydrogenated carbon material. We investigate the dynamical process of hydrogen retention in graphite material whose structure is changed by continuous hydrogen injection.

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Hydrogen diffusion in tungsten: Comparison of DFT, MD and Experimental Results

P29B

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A thorough understanding of tungsten-hydrogen interaction is mandatory for the first-wall development of fusion reactors for which tungsten is one of the most promising materials. While hydrogen retention in undamaged tungsten is presently considered as tolerable the diffusion of hydrogen into the bulk towards the interface may lead to severe interface problems. Extrapolation of present day measurements towards a fusion reactor requires a proper physics based modelling of the relevant processes.
Here we report results from a comparison of DFT-based modelling of hydrogen diffusion in bulk tungsten with classical molecular dynamics simulations. The respective minimal activation energies for diffusion are estimated by the direct saddle-point search and by the nudged elastic band method, showing relevant discrepencies of up to 0.2 eV between the different approaches. The rate constant for diffusion is calculated within the harmonic transition state theory approximation. The simulation results are discussed and compared with experimental data.

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Molecular Dynamics Simulations of Hydrogen Atoms on SiC Surface

P30A

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SiC has much lower chemical erosion, high temperature sputtering and hydrogen recycling compared with pure graphite. Therefor thick SiC coatings on graphite should lead to better plasma performance for a sufficiently long life compared with much thinner boron or silicon films prepared by RF boronization and siliconization. SiC coatings of 200 μm thickness have been developed on the multi-element doped GBST1308 (1%B_4C, 2.5%Si, 7.5%Ti) graphite tiles using chemical vapour reaction (CVR) combined with chemical vapour infiltration (CVI), and now are being applied to the first phase of the Experimental Advanced Superconducting Tokamak (EAST) at the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) [1]. In fusion society, the investigation on the interactions between plasma and SiC materials has been scarcely performed up to now and the mechanism and the processes of the interactions have not been well understood. In this work, molecular dynamics (MD) simulations were carried out to study the surface effects of SiC under hydrogen ions irradiation.

The initial computational cell consists of 4000 silicon atoms and 4000 carbon atoms in lattice sites of β-SiC crystal with crystal size $10a_0\times10a_0\times10a_0$, where a_0 = 0.436 nm is the lattice constant. In order to form a surface, periodic boundary conditions were removed in the z direction and the atoms in the lowest three atomic layers were kept fixed at their original positions all times. Atomic H projectiles with a series of kinetic energies from 0.1 to 100 eV were used to interact with SiC target. In this work, a multicomponent potential for C-Si-H systems was adopted [2], which is a semiempirical potential developed for modeling both the chemistry and the bulk properties of C-Si-H systems based on the Tersoff formulation. At relatively lower incident energy, the projected H atoms tend to be absorbed on the surface of the simulated cell. The absorbed sites with different formation energy have been analyzed. With increasing the incident energy, the defects emerged in the crystal cell under successive irradiation of H atoms and the incident H atoms could travel a rather long distance below the surface. We also have investigated the interactions between the H atoms with different defects in the crystal cell, namely C vacancy, Si vacancy and some interstitial configurations.

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Molecular Dynamics Simulations of the Sputtering Behaviour of Mixed Be/W/C-Materials

P30B

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The interactions between ions with kinetic energies of the order of 1 – 100 eV with materials are an interesting borderline between physics and chemistry. Since the energies involved are comparable or only slightly higher than the strengths of chemical bonds, chemical effects can be expected to give a significant contribution to material erosion in this energy regime. This issue is particularly important due to the ongoing development of the ITER fusion reactor, where ions in this energy range will interact with the complex Be/W/C-based plasma materials surface.

Several types of chemical effects have indeed been reported in these systems. Numerous experiments have shown that any carbon-based first wall material in the bottom part of the reactor (the divertor) erodes with high yields at energies clearly below the physical sputtering threshold. While this anomalous erosion can at high temperatures be understood by hydrogen-induced formation of volatile species that desorb by thermal activation, the effect does not show any temperature dependence between liquid nitrogen and room temperature, showing that a thermally activated mechanism cannot be the full explanation.

Molecular dynamics (MD) computer simulations have shown that both the carbon and beryllium erosion can be explained by a special type of chemical sputtering, where the incoming energetic ion enters between two carbon atoms, forcing them apart if its kinetic energy is low enough that it spends a substantial amount of time between the atoms. Also reflection of low-energetic species (less than about 10 eV) can be described with MD much better than with codes based on the binary collision approximation (BCA).

In order to realistically model the complex multi-scale behaviour of the plasma-wall interactions, a joint effort encompassing different modelling techniques is necessary. For example, 3D Monte-Carlo (MC) methods have been successfully used to simulate impurity transport, taking into account erosion and deposition processes. The plasma transportation is well described in such models, e.g. in the MC code ERO, but the modelling of the plasma-wall interaction processes is not complete. For instance, the sputtering yield is often calculated with the help of codes based on BCA, which is inaccurate for low-energy bombardments, and assumptions for the sticking of molecules and the re-erosion of re-deposited layers must be done. By combining the two methods, MD and MC, one can achieve a higher level of accuracy.

We present results from recent MD simulations of the sputtering and reflection behaviour of mixed Be/W/C-based materials, with incident ions within the energy range of 1 – 100 eV. These results can be used as input for e.g. plasma impurity transport codes, to make realistic predictions for future fusion reactors.

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P31A

Simulating the Plasma-Induced Evolution of Surfaces

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Tungsten is a leading candidate material for the divertor in ITER and other future nuclear fusion reactors, as chemical sputtering is virtually absent and physical sputtering is not expected at the relevant ion potentials (tens of volts). However, helium plasma bombardment experiments have demonstrated that surface defects and bubbles form even at modest ion energies, and in some cases “fuzz” [1] and “coral” [2] like surface features exist after a few hours of exposure. We investigate the formation mechanisms behind these surface features using molecular dynamics and kinetic Monte Carlo simulations, focusing on surface defect formation and bubble formation, growth, and rupture. Sub-surface helium bubbles, in particular, are found to leave behind relatively stable surface defects after they burst. We quantify the helium density that causes bubbles to burst as a function of temperature, surface crystallography, and bubble depth. These quantities will be used as input to larger-scale simulations of surface evolution. We also find that ad-atom/surface vacancy defects can form at energies as low as 25 eV on the [100] surface, but more stable crystal planes such as [110] and [111] give considerably higher activation energies for ad-atom formation. We also study the migration behavior of ad-atom/surface vacancy pairs, revealing a slight tendency of these defects to cluster together and suggesting a path to the larger structures observed experimentally. These results, combined with bubble rupture and formation data, are used in a kinetic Monte Carlo calculations to simulate the formation of surface defects over length scales inaccessible by traditional atomistic simulations.

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Simulation of He and H defect clusters in W

P31B

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Tungsten is considered a viable divertor candidate material to face the extreme particle and heat flux conditions in a fusion reactor. The divertor will be subject to high intensity, low energy hydrogen isotope and helium bombardment from the plasma, which can lead to modification of surface and bulk properties, erosion, swelling, fuel retention, etc. Helium plasma exposure experiments have demonstrated that surface defects and bubbles form even at modest ion energies, and in some cases “fuzz” [1] and “coral” [2] like surface features exist after a few hours of exposure. Neither the mechanisms of tungsten fuzz formation, nor the impact on device performance in a fusion environment are currently known.

Molecular dynamics (MD) simulations are well suited to describe the time and length scales associated with formation of small defect clusters and plasma–surface interaction events and can provide input for longer time and larger length scale Monte Carlo (MC) simulations to investigate tungsten nanoscale fuzz formation. In this presentation, we specifically describe interatomic potentials developed to describe the W-He-H system under conditions of plasma interaction and atomistic simulation results to examine the interactions between helium, hydrogen and small point defect clusters, as well as the behavior of helium bubbles in tungsten. In particular, a newly derived pair potential for W-He describes the energetics of He-vacancy defects in W in good agreement with recent ab initio results.
The energetics, mobility and stability of clusters are important quantities to parameterize the input for future MC simulations, which will be used to further study long term effects of helium in tungsten and provide insight into the formation of nano-fuzz.

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Construction of fuel retention model for full carbon devices

P32A

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Although fuel retentions in the redeposited carbon layers on the inner divertor tiles and the plasma shadowed area have been extensively studied, little work has been devoted to those in eroded divertor area and main chamber (first wall).
We have been examining retention of all hydrogen isotopes of H, D and T in the redeposited layers on plasma facing tiles of JT-60U. In this study, we have measured hydrogen isotopes retentions in eroded tiles located in divertor, baffle and first wall by means of TDS and SIMS. The results are analyzed in terms of incident flux, tile temperature and discharge time. Taking previous results of the retention in the redeposited layers into account, the mechanism of hydrogen retention including isotopic exchange effects is clarified and a model to derive total retention of all hydrogen isotopes in JT-60U is constructed.
The most important finding is that the hydrogen retention at surface and near surface was saturated with the saturation concentration determined by the tile temperature, irrespective of redeposited or eroded tiles. Owing to surface depression by the erosion, the thickness of the saturated layers on eroded tiles increased quite slowly with fluence dependence of $\Gamma^{0.13\sim0.16}$, while the thickness of the redeposited layers in which hydrogen retention was saturated with the same saturation concentration as that of near surface regions, increased linearly with the fluence, i.e. $\Gamma^{1.0}$. Another important finding is that bulk retention was also nearly saturated. The bulk retention is attributed to hydrogen retention at near surface layers of carbon particles (grains) which is easily saturated and very slow hydrogen diffusion in the particles hardly allow the growth of the thickness of the saturated regions.
Thus the total hydrogen retention in the JT-60U vacuum vessel is separated into three components, (1) hydrogen retention in the redeposited layers on divertor and plasma shadowed area, (2) hydrogen in the surface saturated regions of eroded tiles (most tiles of divertor, baffle and fist wall, and (3) bulk retention. Because of the huge volume of the first wall tiles, the last one occupies a significant part of the total retention in earlier time but is saturated soon. Then the second becomes appreciable, also owing to slow but continuous increase and large surface area of the first wall. Finally, the retention in the redeposited layers dominates the total retention. Since the saturated concentration in the redeposited layers decreases with increasing temperature, the total retention could be significantly reduced by higher temperature operation. According to this retention mechanism and supposing all plasma facing tiles to be carbon in ITER, the integrated amount of retained hydrogen are calculated to be more than one order of magnitude less compared to the current estimation made by Roth et al [1].

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Multi-Scale Modeling of Hydrogen Retention in Co-deposits

P32B

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Fuel retention in co-deposits is one of the potentially limiting constraints in the steady-state operation of the future fusion devices [1]. New generation tokamaks such as ITER, will use a combination of high and low Z materials as plasma facing components at different parts of the firstwall. This leads to multi-material co-deposition [2]. Understanding and quantification of retention in these co-deposits is required to estimate the total in-vessel fuel inventory.

In the present work, we discuss a multi-scale model for the estimation of the fuel retention in hydrocarbon co-deposits and the possibilities of extension to multi-material deposits. The model is based on the insights derived from the simulations performed at different time and space scales. MD simulations were performed to understand the fundamentals of the growth of hydrocarbon co-deposits. These results were implemented in an equilibrium Monte Carlo model for simulating the structure of the deposits [3,4]. The diffusion of the hydrogen within these films were studied using time-dependent Monte Carlo simulations. The comparison of the model predictions with experimental data will also be presented in the presentation. The extension of this model for carbon-tungsten systems will be discussed in the second part of the presentation.

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Tracing of hydrogen isotopes with the ERO code

P33A

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Tritium retention will be a crucial issue in next step fusion devices. Predictions for the in vessel tritium content are indispensable for the design of ITER and DEMO. Reliable predictions can, however, only be made by developing sophisticated computer models, benchmarked by dedicated experiments. The VISION I plasmatron - refurbished in 2008-2009 at the SCK•CEN in Mol (Belgium) - is a very promising experiment in this context [1]. Recently, modelling activities for VISION I were started. During the first stage of the project, plasma-surface interaction will be modelled by the ERO code. ERO is a 3D Monte Carlo code, calculating erosion, transport and deposition of plasma impurities for a given background plasma interacting with a target [2]. The code is now being adapted in order to meet the requirements for its use in the VISION I experiment.

Up to now, hydrogen was not traced in ERO and the contribution of hydrogen from hydrocarbon break-up to the erosion of the target could not be taken into account. As a first step in adapting the code, therefore, the tracing of hydrogen was implemented. Now hydrogen atoms, molecules and ions originating from the gas puff or the background plasma are followed. Like other traced particles, hydrogen hitting the surface can either be reflected or deposited (TRIM database). Besides physical sputtering (Bodhanski and Yamamura formulas), hydrogen and hydrocarbon ions also induce chemical erosion (user specified yield, typically 5-10%).

This new feature in ERO can also help improving interpretation of experimental results in the TEXTOR tokamak. $^{13}CH_4$ puffing experiments in TEXTOR were already studied before with ERO [3,4]. The new runs show that background and puffed hydrogen fluxes are comparable (respectively 9.02 and 5.74 10^{16} $cm^{-2}s^{-1}$) and that erosion effects related to hydrogen from $^{13}CH_4$ break-up significantly lower the modelled ^{13}C deposition efficiency. Including these effects, without changing any parameters (reflection coefficients 1.0 and 0.1 for respectively neutral and ionized hydrocarbons and enhancement factor 5 for both chemical and physical erosion), lowers the sticking efficiency on a graphite roof limiter from 6.1% to 1.8%. However, similar to the previous studies, enhanced re-erosion of deposited carbon still has to be assumed in order to reproduce the very low experimental ^{13}C deposition efficiencies around 0.2% for this case. A more detailed parameter study is now being performed. For benchmarking, the modelled deposition and optical emission profiles will be compared with experimental data from TEXTOR.

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On the thermal instability caused by plasma-wall coupling

P33B

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Hydrogen isotopes recycling and retention in the first wall materials are among major issues in fusion devices and in tokamaks, particularly. Interaction between the edge plasma and surrounding surfaces can strongly influence properties of the core plasma and, consequently, the tokamak discharge [1]. High energy/heat loads can lead to damage of the first wall and to release of trapped hydrogen, which is vitally important for the discharge.

A qualitative analysis based on 0D model [2] demonstrates that plasma-wall interaction can lead to appearance of a thermal instability, which may cause intensive heating of the wall and release of a large amount of trapped hydrogen. The instability development involves several processes, which are sequentially arranged in a positive loop leading to the temperature increase. An initial increase of the wall temperature results in a strong increase of the hydrogen desorption rate, and that leads to enhancement of the charge exchange and radiation losses, which increase the heat flux to the wall and prompt further increase of the wall temperature.

In this work we present an analytical study of this instability, and formulate the conditions it is possible. Plasma particle and energy balance is treated within the framework of the simple 0D model proposed in [2]. Description of hydrogen and heat transport in the wall is considered by using a 1D model. The cases of metal and carbon walls are considered separately because the processes involved in hydrogen-wall interactions are different for these materials. Analysis of the dispersion equation shows that instability is possible only for the graphite wall and only at high (more than 1000 K) temperatures. The solution for graphite is stable for smaller temperatures. For the metal wall, the steady-state solution is stable over a very wide temperature range.

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Integrated scenario for controlling tritium retention in carbon co-deposits in fusion devices

P34A

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The use of carbon materials in the high flux area of ITER is a concern due to the formation of carbon codeposits [1]. Several uncertainties exist about the number of pulses allowed before reaching the inventory limit of 700 g [2]. The presence of Be in the first wall [3] and the large amount of gaps [4] in the present design of ITER are key issues that will have a strong impact on the retention. There is a broad experience on the use of carbon facing materials in fusion devices, and their flexibility in handling high power and transient events [1] make carbon a candidate for the DT phase if tritium retention problems can be obviated. Hence, some strategies should be developed to face these problems.

In the present contribution, a global overview about the capabilities for co-deposit cleaning using mainly nitrogenic species is shown. Several techniques including plasma cleaning, thermooxidation and cleaning of gaps are addressed. In particular, those involving the use of NO_2 have shown a higher capability for the removal of carbon layers than those using oxygen, both in plasmas and thermooxidation [5]. Water production will be also addressed because of the existing limited capability for tritiated water handling in ITER.

In order to increase the availability of the reactor, it will be needed to develop techniques able to reduce the tritium retention during normal operation. Nowadays the scavenger technique is the only one able to inhibit the formation of co-deposits in remote areas of the divertor during plasma shots [6]. The integration of this technique in an ITER scenario will be also addressed.

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Deuterium-retention in iron oxides under low energy D_2^+-plasma exposure

P34B

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Stainless steel (SS) is widely used for the nuclear fusion devices. When O_2 plasma discharge is performed for tritium removal, formation of iron-oxides such as Fe_2O_3 will take place [1]. Hence, the retention of hydrogen isotopes or deuterium (D) in the oxide layers is important for designing the devices. In this study, we present the D-retention in Fe_2O_3 films and compared it with D-retention in SS and iron (Fe).

Fe_2O_3 films were prepared by oxidation of Fe deposited on SiO_2 substrate and Fe sheet (99.99% purity and 0.2 mm thickness) in air. According to X-ray diffraction (XRD) and Rutherford backscattering spectroscopy (RBS) of 1.8 MeV H^+, hexagonal α-Fe_2O_3 (hematite) films are formed on SiO_2 substrate at 400ºC with stoichiometric composition (Fe:O=2:3) throughout the film, while cubic γ-Fe_2O_3 (maghemite) is dominantly formed on Fe at 400ºC (a very thin layer, ~10 nm [2], of Fe_2O_3 on top of ~1.5 μm of mixed layers of Fe_2O_3 and Fe), and mixture of γ- and α-Fe_2O_3 (~0.7μm on top of mix layers over 10 μm of predominantly Fe_2O_3 and a small fraction of Fe) on Fe at 500ºC.

Samples were exposed to D_2-plasma (glow discharge in 0.4 Torr D_2, with AC-applied voltage of 1.5kV for 15 min [3]). The projected range of D at the maximum energy of 1.06 keV is 12 and 8 nm in Fe_2O_3 and Fe [4]. Nuclear reaction analysis (NRA) D(^{3}He,α)H with 1 MeV ^{3}He was employed to analyze the amount of D (an estimated error of 20 %). We find the followings. (1) D-retention in α-Fe_2O_3 on SiO_2 is 17-20x10^{15} cm^{-2}, nearly independent of the oxide layer thickness (0.16-0.82μm). This indicates small diffusivity of D in the Fe_2O_3. (2) D-retention in Fe_2O_3 (γ-phase dominant) and Fe_2O_3 (mixture of γ- and α-phases) are 47 and 36x10^{15}cm^{-2}. Both results imply that D-retention in γ-Fe_2O_3 is larger than that in α-Fe_2O_3. Dependence of D-retention on crystal structure is to be investigated. (3) D-retention in SS and Fe is obtained to be 28 and 16x10^{15}cm^{-2}, comparable with that in α-Fe_2O_3. Measurements of oxide layer thickness on SS and Fe and their contribution to D-retention, diffusivity of D in iron oxides, DC-discharge for exposure of D with well defined energy, dynamic retention of D, thermal release of D, D-retention in Fe_3O_4 are under way.

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DITS project: mapping of the topology of the erosion and deposition zones at the surface of the toroidal pump limiter

P35A

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Understanding and preventing in-vessel fuel retention is essential for next step fusion devices since the tritium inventory is limited due to safety reasons. The “Deuterium Inventory in Tore Supra” project (DITS) was initiated to gain insight on the retention mechanisms and to compare the deuterium (D) inventory obtained from particle balance and post mortem analysis. The vessel walls of Tore Supra were loaded with deuterium in a dedicated experimental campaign. A cumulated D inventory of 10 g was estimated from particle balance measurements in addition to the previous ~35 g present in the machine.

After plasma operations, N11 carbon fibre composite (CFC) tiles were dismantled from the toroidal pump limiter (TPL) and the poloidal inner bumpers, and prepared for post mortem analysis in a collaborative European effort.

By D sensitive techniques – nuclear reaction analysis and thermal desorption – a robust estimation of the D inventory trapped in the main plasma facing components (CFC N11 Sepcarb) was calculated. The dominant retention mechanism is codeposition (~95% of the D inventory) from which a significant amount was deposited on the gap surfaces (~30%).

By microstructural means – SEM, TEM, Raman and AFM – analysis of topology and morphology of those deposits constitute another possibility to evaluate D and C content, their transport in the SOL and can contribute to the global post mortem study.

Major results have been obtained on deposit properties on the top of the TPL which is a fingerprint of plasma interactions with CFC (e.g. erosion and deposition zones):

- (i) loosely attached thick deposits presenting tips pointing toward low field side (LFS)
- (ii) tightly attached thin heterogeneous deposits with tips oriented toward LFS
- (iii) eroded tiles presenting striation due to rippling of the surface (cf this conference C.Martin et al.)

Both tips and striation are signatures of ion flux direction and ion interaction with TPL tiles.

We present in this poster a mapping of the TPL surface's topology thanks to deposit thickness and net erosion measurements, with a complementary mapping of the orientation of striation (erosion ripple) and tips growth. The thickness map will be useful to calibrate confocal measurements and to access to the total mass of carbon eroded from the TPL. It will be also of prime interest to normalize NRA profiles to the total plasma operating time.

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Post-mortem measurements on fuel retention at JET in 2007-2009 experimental campaign

P35B

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Long lifetime of plasma-facing components and fuel retention are critical issues of future fusion devices. Carbon or carbon fibre composite (CFC) is a plasma-facing material in many present-day fusion devices because of excellent power-handling capabilities. The major disadvantage of carbon based materials is its chemical erosion under hydrogen bombardment and associated to this the ability to trap large amounts of tritium. This is especially dangerous in the case of deuterium-tritium operation as it may lead to an unacceptable inventory of radioactive tritium. In ITER, a retention of the injected tritium would lead to the in-vessel tritium safety limit of 1000 g set by nuclear licensing less than 1000 full performance ITER discharges without any cleaning effort. Determination of deuterium retention in plasma-facing components is crucial for the assessment and overall fuel inventory in plasma-facing components.

In the period 2007-2009 JET operated with the MkII-HD divertor. The deuterium inventory in divertor tiles exposed in 2007-2009 has been determined using Ion Beam Analysis (IBA), Secondary Ion Mass Spectrometry (SIMS) and optical microscopy. IBA analyses give quantitative D/C ratio near the surface region (up to depth ~7 μm) whereas SIMS provides information from greater depths. Thickness of the co-deposited layers was determined both with SIMS and optical microscopy.

Deposits on the inner divertor vertical Tiles 1 and 3 have high Be/C ratio. On Tile 1 the Be/C ratio is bigger than one and on Tile 3 it is less than one which indicates that there is somewhat less deuterium on Tile 1 than Tile 3. The thickness of the deposit decreases from the apron of the Tile 1 to the bottom of the Tile 1 and then increases on Tile 3. The thickness of the co-deposited layer on Tile 3 is larger than in Tile 1 resulting in a higher D retention. Both floor Tiles 4 and 6 have very thick deposition on the sloping part. Deuterium is retained mainly in the shadowed area of Tiles 4 and 6. On the outer divertor Tiles 7 and 8 deuterium is retained mainly due to ion implantation so the deuterium amount is small. From IBA and SIMS results the total retained amount of D will be estimated by assuming toroidal uniformity of deposition and by integrating over the torus. These results will be discussed in the paper.

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Nanostructural analysis of eroded CFC tiles of Tore Supra

P36A

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The Deuterium Inventory in Tore Supra project is aimed at studying retention in the carbon Plasma Facing Components (PFCs). After a D-loading PFCs campaign with an in-situ trapped D inventory monitored through particle balance[1], a sector of the Toroidal Pump Limiter (TPL) has been dismantled for an extensive post-mortem analysis of the CFC tiles, combining techniques devoted to deuterium inventory [2] and to structural and chemical characterisations. Due to the different rates of the erosion/deposition processes, the TPL surface exhibits a pattern defining different zones of interest for the analyses, in particular an erosion dominated zone.
Thanks to the coupling of electron microscopy (SEM and TEM), atomic force microscopy (AFM) and Raman microspectrometry, we performed a multiscale analysis of the structure of the tile surfaces extracted from different locations of the erosion zone (2.4 m^2). We analysed the topography, thickness, roughness, porosity filling and atomic structure of both gap and top surfaces of CFC tiles. These measurements give information on C-deposition and D-retention inside gaps and also give some clues on particle transport in the SOL. We focused on erosion processes at the surface characterising the carbon structure. The results are compared with those obtained by laboratory plasma exposures performed on pure graphite and CFC surfaces, with the aim of studying erosion processes at atomic scale.
Major results have been obtained on deposit properties inside gaps such as (i) the tip morphology with well defined orientation similar to that previously observed for deposits previously collected on Tore Supra and TEXTOR neutralisers [3], (ii) a strong low field / high field toroidal asymmetry in deposit thickness and (iii) an up-stream/down-stream poloidal asymmetry in roughness and deposit depth profile [4]. On the other hand, the top surfaces of erosion zone have revealed to be a fingerprint of the interaction between the edge plasma and the CFC fibre or matrix. First, TEM and Raman have shown that a smooth and thin layer (< 50 nm) of amorphous carbon covers the surface, due to the amorphization of the original graphitic CFC material, its thickness being related to the energy of ions impinging on the surface. Second, SEM and AFM measurements have shown that ion fluxes induce a ripple pattern with striation at the micrometer scale and, in addition, they have clearly revealed a differential fibre/matrix erosion. All the phenomena observed on Tore Supra TPL are investigated by combining these post-mortem analyses and laboratory plasma experiments on HOPG graphite and CFC surfaces, to deduce relevant parameters to model ion-induced erosion and to better describe the erosion/deposition processes at micro- and nano-meter scales.

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P36B

Hydrogen Retention Properties of V-4Cr-4Ti Alloys

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V-Cr-Ti alloys are promising constructive materials for nuclear and fusion reactors due to their low activation by fast neutrons. In the Bochvar Institute a base V-4Cr-4Ti (with 4% wt. Cr and 4%wt Ti) was produced with thermomechanical properties that would allow it's use in fusion devices. The hydrogen transport and accumulation in the alloy was never investigated before but is expected to be a critical property due to the potential high retention of hydrogen in vanadium.
In the present work we analyze the interaction of hydrogen with the base V-4Cr-4Ti alloy produced in Russia (Bochvar Institute) under exposure to a deuterium gas atmosphere and under irradiation by 300 eV deuterium plasma in the fluence range of 10^{19} – 10^{21} D/cm^2 and in the temperature range of RT-400 ^{0}C. Deuterium depth profiles were determined by means of nuclear reaction analysis (NRA) using the D(^{3}He,p)□ reaction at various incident energies from 700keV to 5500 keV, thus allowing to measure the depth profile until a depth of about 15 µm, and methods of getter properties investigation. The deuterium concentration after deuterium plasma irradiation was high (of the order of 5 at%) and exceeded far beyond the maximum analyzable depth, indicating a strong diffusion and retention of hydrogen in this material. It was found, that annealing does not led to activation of the alloy as a getter, while after combination of annealing and argon glow discharge treatment V-4Cr-4Ti alloy shows getter properties. That means that under these conditions a first wall made of V-Cr-Ti may act as a hydrogen getter pump absorbing hydrogen from a gas even at room temperature. The use of these alloys in a tritium-containing environment seems questionable without additional protecting permeation barriers.
This work is supported by the joint Russian Foundation for Basic Researches - Helmholtz-Gemeinschaft grant HRJRG-216 and contract with Russian Ministry of Education and Science (14.740.11.413)

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Deuterium retention in bulk tungsten exposed to the outer divertor plasma of ASDEX Upgrade

P37A

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Tungsten (W) is a promising material for the plasma-facing components in future fusion reactors, and has been extensively studied in ASDEX Upgrade (AUG), which has been operated with a full-W plasma-facing wall since 2007. There are two options for use of W, either as W-coatings which are currently used in AUG or as bulk W material. Because plasma-facing components with W coatings suffer from a low damage threshold power flux density, components in future devices like ITER will consist of bulk tungsten. Bulk W tiles are, therefore, going to be tested in also current fusion devices, i.e. installed in JET as the horizontal divertor plate of the ITER-like wall configuration in coming campaigns, and it is also planned to use bulk W tiles in the next step divertor configuration (divertor III) of AUG. For this material, however, one concern is the possibility of hydrogen diffusion and trapping / accumulation deep into the W bulk, which can lead to significant tritium uptake in the plasma-facing wall. As current studies with W layers are difficult to extrapolate to bulk W, D retention in bulk W exposed to the AUG divertor plasma has been investigated to improve the understanding of tritium inventory in W under fusion reactor conditions.

Four W blocks with a plasma-facing cross-section of 8 mm x 8 mm and a thickness of 25 mm were simultaneously exposed to the outer divertor plasma by using the AUG divertor manipulator system. The plasma-facing surface of one block was mirror polished while the other three had machined surfaces with different roughnesses. After exposure, D retention in each block was measured by nuclear reaction analysis (NRA) using the $D(^3He, p)^4He$ reaction and by thermal desorption spectroscopy (TDS). NRA gives the D depth profile and the amount in the W surface up to ≈ 7 μm in depth, while TDS provides the D release behaviour and the total D retention in the W block. The surface morphology was examined by scanning electron microscopy in order to check any depositions, surface damages and blister formation.

D surface depth profiles measured by NRA showed that the maximum D concentration was at the top surface (several hundred nm) with a concentration of several at.% in all samples. The D concentration decreases with depth, and has a plateau with ≈ 10^{-1} at.% in the sub-surface region (1 - 2 μm in depth) in the case of technical surface samples, while the mirror-polished sample showed one order of magnitude less D concentration in that region, indicating that the surface finalization of the bulk material has some influence on the hydrogen retention. Eventually, all samples showed a D diffusion tail into deeper regions (> 3 μm) with the concentration in the range of 10^{-2} ~ 10^{-3} at.%. Based on these results, the consequences for fuel retention in bulk W materials under fusion plasma conditions will be discussed.

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Behaviour of tritium in plasma-sprayed tungsten coating on steel exposed to tritium plasma

P37B

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As a plasma-facing wall in a future fusion reactor (DEMO) of Japan, a reduced activation ferritic/martensitic stainless steel (F82H) is planned to be used with tungsten (W) surface coating deposited by plasma-spray techniques [1]. Although the W coating is porous, one may expect that the coating or the interface between the coating and F82H substrate works as T permeation barrier. This study has been devoted to examine the role of the W coating on T permeation through the W coated F82H exposed to T plasma by surface and depth profiling of T with using Tritium Imaging Plate Technique (TIPT) [2].

The samples used here were two types of plasma-sprayed W coating on a F82H substrate. The coatings were made by an atmospheric plasma spray technique (APS-W/F82H) and a vacuum plasma spray technique (VPS-W/F82H). The thickness of the W coating was 1 mm. Particle sizes used for plasma spray were 17 and 50 μm for the VPS-W and the APS-W coatings, respectively. The porosity of the coating was 0.6 % for the former and 6 % for the latter. Hydrogen including T ($T/H=10^{-4}$) was loaded by exposing the sample to DC glow discharge (DCGD) plasma at 533 K for 2 h. After the exposure, the sample was rapidly cooled to 233 K to avoid hydrogen release and diffusion from/in the sample. T distribution on the surface of the W coatings was measured by the TIPT at the same temperature. Subsequently the sample was cut perpendicular to the surface of the W coating to get the cross-section on which T distribution or T depth profile in the sample was measured by TIPT [2].

Except very near surface region within a few hundred μm where T was highly segregated, T distributed rather homogenously in the whole thickness of the coating for both VPS-W and APS-W. In the F82H substrate, T penetrated into a few mm with a decaying concentration in depth. The surface segregation was likely caused by T injection from the plasma and the segregated amount of T was higher for the VPS-W coating than for the APS-W coating, probably due to the larger surface area of the former than the latter. The homogeneous T distribution suggests the penetration of T gas through open pores in the coating. At the interface, the T concentration in the F82H substrate was about a half of that in the W coating. Both samples showed nearly the same decaying concentration profiles of T in the F82H substrates. This also suggests that T gas penetrates through the open pores in the coating and directly faces to the part of the F82H substrate to give the same diffusion (decaying) profiles. In other words, the W coatings did not work as the permeation barrier but restricted T gas penetration only through the open pores.

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P38A

Concept Design of the ITER Glow Discharge Cleaning System

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The present baseline design for ITER Glow Discharge Cleaning system (GDC) is DC GDC. Following venting of the vacuum vessel, GDC will be carried out for a couple of hundred hours to remove impurities (especially oxygen) from the near surfaces of in-vessel plasma-facing components. This is a routine operation like in many nowadays tokamaks. Before the venting of the vacuum vessel, GDC may also be conducted to decrease the hydrogen isotopes inventory on the in-vessel surfaces.
Due to the restriction on the space, ITER GDC has to share the same penetration through the vacuum vessel structure and blanket with the In Vessel Viewing System (IVVS). The two systems are highly integrated into one assembly in each port they're sharing.
Another important function of GDC's is to provide efficient shielding, especially nuclear shielding, for the structures behind.
These features impose several restrictions on the electrode design. This paper will summarize the requirements and concept design of the ITER GDC.
(1) Total glow current will be about 150 A, giving an averaged current density of about 0.15 A/m2, matching standard values achieved on most past and present tokamaks. This will be provided by a total of 6 GDC electrodes distributed at the lower port level to provide good toroidal uniformity of the glow plasma within the constraints imposed by availability of port access points.
(2) During GDC operation, the electrodes will be inserted right above the divertor dome structure to achieve clear access to the main chamber and divertor regions.
(3) During plasma operation, the electrode will be retracted in a cut-out of the blanket module and vacuum vessel triangular support, meeting the requirements on nuclear shielding and reducing the neutron load to the structures and systems hidden behind.
(4) Each GDC electrode will be integrated with an IVVS to form a single assembly sharing the same access trajectory through the vessel support and blanket. When the IVVS is needed for inspection, the GDC electrode will be retracted to a "parking position" to clear the penetration path.
A series of difficult questions on the GDC electrode need to be answered during the concept design stage, including: exact locations in different states, the working temperature of the electrode, material selection. These are closely related with the heat load and cooling design, which also put a number of obstacles to the design of the service lines (such as cooling circuits and electrical cables) and the deployment. These several meters long flexible structure will be in operation in a nuclear environment, therefore shall be highly reliable as well. The electrode structure, deployment system and support structures need to be robust and be optimized for remote handling maintenance.
This paper will give an over-all description of the above-mentioned issues as a summary of the concept design of ITER GDC.

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Ion beam analysis of ^{13}C and deuterium deposition in DIII-D and their removal by in-situ oxygen bake

P38B

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In response to concerns about potential erosion/deposition in the ITER main chamber, an experiment was conducted in 2009 in DIIII-D to examine carbon deposition when a secondary separatrix is near the wall. The magnetic configuration for this experiment was a biased double-null, close to the one foreseen for ITER, but with the primary separatrix at the top. ^{13}C methane was injected toroidally symmetrically from the lower pumping plenum near the secondary separatrix. ELMy H-mode deuterium plasmas were used. ^{13}C deposition was measured by nuclear reaction analysis, on plasma-facing surfaces of 37 tiles to obtain the full poloidal distribution, and on the edges of selected tiles to examine depositon in the gaps between tiles. Here we present the results from the measurements of ^{13}C and of deuterium on these tiles. 48% of the injected ^{13}C was found (assuming toroidally symmetric deposition), mainly near the secondary outer strike point. Deposition of ^{13}C in the gaps was small. These results show that very little of the injected ^{13}C was deposited at the primary separatrix, whereas a large fraction of injected ^{13}C was deposited close to the point of injection near the secondary separatrix, in contrast to previous studies with single null plasmas.

In April 2010, six of the tiles were put back into DIII-D where they were baked in oxygen (2 Torr O_2 and 8 Torr He) for 2 hours at 350°C to examine the efficacy of oxygen baking to remove codeposits. Subsequent ion beam analysis of these tiles showed that about 20% of the ^{13}C and 53% of the deuterium, had been removed by the bake.

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration, under contract DE-AC04-94AL85000.

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Characterisation of hydrocarbon and mixed layers in TEXTOR by Laser-Induced Ablation Spectroscopy (LIAS)

P39A

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Laser based methods are investigated presently with the goal to create an in-situ spatially and temporally resolved diagnostic for the characterisation of the first wall in fusion devices. This contribution reports on laser ablation studies in lab experiments on fusion relevant materials and layers and on Laser Induced Ablation Spectroscopy (LIAS) experiments on similar materials and layers in the TEXTOR tokamak.

The basic concept of LIAS is that wall material (or deposits on it) is evaporated during running plasma discharge by intensive laser radiation and the spectral line radiation of ablated species is measured absolutely over the whole observation volume. With known edge plasma parameters, the line intensities are converted into fluency of the released particles. From this the composition of the particles ablated from the surface can be inferred. With consecutive laser pulses on the same spot also the depth profile (thickness) of the deposited layer can be determined.

A detailed laboratory analysis of laser ablation was performed involving post mortem crater analysis of the laser spot, Laser Induced Breakdown Spectroscopy (LIBS), analysis of the ablation plasma, microbalance measurements and time of flight quadrupole mass spectroscopy analysis of ablated species. (TOF-QMS).

The investigations show that the particle source becomes more stable for higher fluence ($F \geq 10$ J/cm^2). For typical spot sizes of A=0.07 cm^2 on fine grain graphite, a volume of typically 3.1*10^{-7} cm^3 is ablated per shot. For a:C-H layers and high laser fluence a volume of around 1.0*10^{-5} cm^3 (A=0.062 cm^2) is removed.

The ablation process and plume is analysed for the composition of neutrals and ions, the ion and neutral particle velocity and angular distribution with TOFQMS. Mainly carbon C1-neutrals escape the ablation plume with a strong forward direction, with velocities of $\approx$ 20 km/s , independent of the laser fluence. Also a smaller fraction of C_2 and C_3 molecules have been observed.

LIAS Experiments have been started in TEXTOR on carbon bulk material and W/C/Al/D_2 layers deposited by magnetron sputtering at the University of Basel. Samples positioned in the SOL of TEXTOR were irradiated at F=8.2 J/cm^2 during discharges with densities between 2.1-3.1×10^{19} m^{-3} and I_p=355 kA. Spectroscopic analysis allowed the identification of the mixed material composition. For carbon bulk material ablation, CI-III line radiation was observed as well as C_2- and C_3- molecule band emission.

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I

In-situ Measurements of Fuel Retention by Laser Induced Desorption Spectroscopy (LIDS) in TEXTOR

P39B

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Material erosion and deposition and the associated long-term fuel retention by co-deposition is a critical issue for ITER and future fusion devices and requires both fuel retention control and mitigation techniques. As a prerequisite a space resolved diagnostic to monitor fuel retention is required. For ITER, laser based methods in combination with spectroscopy are proposed as possible in-situ methods to characterise fuel retention and material deposition. This contribution reports about the experimental methods and results obtained in the TEXTOR tokamak to determine in-situ the fuel retention in dedicated wall samples by laser induced desorption spectroscopy (LIDS).

A Nd:YAG laser pulse is guided via a 35 m fibre to the TEXTOR tokamak. With laser intensities of 700 MW/m^2 a 4 to 7 mm^2 spot on wall components located in the TEXTOR limiter lock system is heated to about 1800 K within few milliseconds [1]. The limiter is observed from the side and top with spectrometers and a synchronized CCD camera using interference filters. The hydrogen Balmer lines are measured quantitatively and via conversion factors the absolute local fuel inventory is evaluated. This in-situ method is regularly cross-checked against ex-situ laser induced desorption quantified by quadrupole mass spectrometry (LID-QMS) and other methods. It has been found that the in-situ and ex-situ methods agree within a factor < 2.

For the studies of long-term fuel retention by LIDS, thick deposited fuel containing layers with fuel inventory up to 10^{23} /m^2 have been analyzed by LIDS in TEXTOR under various plasma edge, heating and desorption positions. For the conversion of the H_α photons to desorbed atoms, conversion factors for the actual plasma conditions were chosen. The accuracy of the data, its reproducibility and dependence on plasma parameters and the optical observation system will be discussed.

In addition, studies of the dynamic short-term fuel retention, which only builts up during the plasma loading, were done. For this purpose first wall materials have been positioned under erosion dominated conditions in the SOL and the fuel retention was measured by LIDS for various accumulated fluencies by rapidly repetitive laser shots. By an optimised limiter design and timing scheme the dynamic retention signal could clearly be separated from the background light. The data show that only a short plasma exposure time (< 500 ms) is needed to refuel the laser depleted surface in case of graphite walls.

The experimental experiences and data will be used to give an outlook to the potential application of LIDS in ITER.

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P40A

Studies on laser heating and laser ablation of ITER-like surfaces

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Laser heating and laser ablation of deposited layers with high tritium content is seen as a promising method for detritiation of the plasma facing components in fusion reactors [1]. In this work laser heating and laser ablation of ITER-like samples were studied. The samples consist of stainless steel or tungsten substrate with deposited layer composed of tungsten, aluminium or diamond-like carbon (DLC) with hydrogen and deuterium content.

For laser ablation, a focused ytterbium fiber laser beam (1.06 µm wavelength, 1 mJ pulse energy, 120 ns pulse duration, 20 kHz repetition rate, 10 J/cm^2 fluence, 100 µm laser spot diameter at e^{-1} intensity level) was applied. The heating measurements were performed using a laser with 10 ms pulses and a fast pyrometer (10 µs response time, 1.58–2.20 µm spectral range of sensitivity). To calculate the sample heating temperatures, a computer code developed in our laboratory was applied [2].

The ablation thresholds and cleaning performances were found for 800 mm/s laser beam scanning regime. Laser heating temperatures were measured and compared with calculated ones. A good agreement was obtained between the measured and the calculated temperatures, indicating that the developed computer code can be applied with an appropriate tritium diffusion model to simulate laser detritiation of ITER-like surfaces.

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Characterisation of Wall Components in Fusion Devices by Laser-Induced Breakdown Spectroscopy

P40B

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Plasma-facing components (PFC) in the main chamber (beryllium), the baffle region (tungsten) and the divertor (carbon) of ITER are eroded by physical and chemical sputtering (in case of C) and evaporation. The material is transported from net erosion areas by the edge plasma to areas with net deposition, forming layers of mixed materials with co-deposition of tritium. Knowledge of the distribution, thickness and composition of these layers, which will be strongly inhomogeneous, especially in poloidal direction, is essential for the safe operation of ITER. *In situ* characterisation of deposition layers (tritium quantity and surface distribution, thickness, composition) are of major importance for the operation of fusion devices.

Laser-Induced Breakdown Spectroscopy (LIBS) is a well-known laser method for quantitative analysis of surface matter composition and is proposed as possible method for characterisation of the wall conditions in ITER. The principle of the technique is to ablate the co-deposited layer with a laser pulse with $P \geq 0.5$GW/cm^2 and to analyse the light emitted by the plasma created by the laser-matter interaction by spectroscopic means. The typical extension of the laser plasma plume ($n_e \approx 10^{22}$m^{-3} and $T_e \approx 1$-2eV) is in the order of 1cm and the plasma lifetime (decay time) is below 1μs.

The feasibility of the LIBS method has been studied in laboratory experiments in a vacuum chamber with a base pressure below 10^{-4}Pa. The laser-induced plasma plume from a Q-switched ruby laser with 1J maximum energy and 15ns pulse duration is analysed by a high resolution cross-dispersion Echelle spectrometer (resolving power $\lambda/\Delta\lambda \approx 20000$) in the spectral range between 375nm and 715nm. The obtained material ablation rates are in the range of ~0.3÷0.6μm/shot at the energy density range of F~2.6÷7J/cm^2 and saturate at about ~0.7μm/shot for higher energies. Stable conditions of the laser plasma have only be achieved at laser energy densities above 8J/cm^2, otherwise the fluctuation of a single line can reach about 100% independently of the type of C-layer. Therefore, the operation at F<8J/cm^2 is affected by uncertainties that are too large for analysis of the ablated layer in a single laser pulse (not recommended for ITER operation). The application of the LIBS method on well-known prepared samples, supported by absolute calibrated spectroscopy, will allow quantitative determination of the layer components. Laboratory results shows that for F≥10J/cm^2 on fine grain graphite (EK98), the ratio of the ablated C atoms to the number of CII line photons resulting from the single ionised C is $C_f = N^C/I(\text{photons}) \approx 10^6$. The high conversion factor implies that to resolve the LIBS signal with good photoelectron statistic $1/\sqrt{N_{el}} \leq 3\%$, about 10^{18}C atoms (content of C atoms in a 100 nm thick layer) must be ablated from a C layer. This lower limit seems to be fully appropriate for ITER, since one full power ITER pulse will provide a deposition similar to e.g. a JET operational period in which layers much larger than 1μm are deposited. The first results of the LIBS diagnostic at TEXTOR are presented and discussed.

⍀ work supported under EFDA contract **WP09-DTM-01-3.1**
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Dependence of power density and energy fluence on ablation rates and LIBS observation thresholds for laser-irradiated ITER relevant materials

P41A

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In research on ablative laser removal of fuel and co-deposits from PFCs, the parameters of removal are commonly expressed in the dependence of the laser energy flux defined as J/cm^2 and calculated for single laser pulses or series of pulses. Due to variety of recently available laser systems for removal applications, which can provide pulses from broad ranges of energy and time duration, this approach is no longer sufficient. The limitations of the fluence model is especially clear when comparing the removal with the use of short pulse Nd:YAG lasers operating in single nanoseconds and a part of J regime and fiber lasers operating in the regime of one hundred nanoseconds pulse duration and single mJ energy.
To complete the fluence model and make it applicable for assessment of the laser removal efficiency, the information on the laser power density should be included into it. It is obvious, that especially for longer and low energetic pulses, the decreasing power density can have deteriorating impact on the ablation rate even for high energy fluence. While the removal process can be monitored by the LIBS method, the power density threshold for the correct LIBS characterization should be determined.
The contribution includes a comparative experimental study on the influence of the laser power density and energy flux on the ablation rate for Nd:YAG and Yb:fiber laser systems. The ablation rate in function of fluence is estimated for various power densities from a wide range from 10^6 to 10^{10} W/cm^2 by the means of profilometry of laser induced craters. The thresholds for LIBS observation are estimated with the Mechelle 5000 spectrometer equipped with ICCD iStar. As the targets for the laser irradiation there were used various calibrated samples with clean C, W, Al (as Beryllium analogue) surfaces as well as with deposited layers containing various mixes of these materials also with hydrogen contamination.

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Hydrogen permeability in pure Fe metal and FeC alloy

P41B

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Samples of pure Fe metal and FeC alloy with controlled chemical alloying element contents and microestructure, supplied by the European Fusion Development Agreement (EFDA), were experimentally analysed in the facilities located at the University of the Basque Country (UPV-EHU) in collaboration with CIEMAT.

In this work, the hydrogen transport parameter of permeability was experimentally measured in pure Fe metal and FeC alloy by means of the gas evolution permeation technique. The experimental temperature range explored was from 423 K to 823 K and the high purity hydrogen loading pressures from 10^3 Pa to $1.5{\cdot}10^5$ Pa. We observed that the permeability obtained for the two materials in this temperature range follows an Arrhenius law. The resultant diffusive permeability for pure Fe metal and FeC alloy were found to be $\Phi_{Fe} = 3.60{\cdot}10^{-8}$ exp(-35.6 (kJ mol^{-1})/RT) and $\Phi_{FeC} = 9.17{\cdot}10^{-8}$ exp(-37.4 (kJ mol^{-1})/RT), respectively. The resulting activation energies turn to be very close, whereas the permeation prefactor increases for the FeC alloy, in comparison to pure Fe. These results are in good agreement with values found in the literature for pure Fe metal and for low carbon content FeC alloys. According to the results, the influence of the metallurgical composition of C in Fe alloys in the diffusion of hydrogen is discussed.

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Deuterium ion driven permeation in tungsten with different microstructures

P42A

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Buildup of tritium (T) inventory in tungsten (W) plasma facing components is a safety concern for ITER and DEMO. Tritium inventory in W depends on the concentration and distribution of various defects, and the rate at which tritium diffuses and reaches such defects and is trapped. From a wide number of irradiation-desorption studies, it is well known that a large part of the retained hydrogen in W is trapped in the bulk of the material. Specifically, the W microstructure was found to strongly influence the trapped H distribution and total retention [1]. However, such studies themselves do not yield direct information on the H diffusion behavior. In comparison, ion driven permeation experiments can investigate diffusion and trapping parameters that influence retention. The data on hydrogen ion driven permeation in tungsten is very limited with only one previous study having examined different W microstructures [2]. This paper presents deuterium (D) ion driven permeation and its dependence on different tungsten microstructure.

Three different W specimens were investigated. Two polycrystalline specimens of 99.99% purity with different grain sizes were obtained by annealing at 1300 ºC (W1) and recrystallization at 2000 ºC (W2). A third polycrystalline specimen of similar purity with the smallest relative grain sizes was provided by IPP, Garching (W3). The microstructure was characterized by taking specimen cross sections using SEM microscopy. Permeation experiments were performed using the HiFIT device for D ion irradiation at Osaka University. In comparison to previous ion driven permeation studies, the parameter space explored both higher fluxes (10^{20} D/m^{2}s) and a wider temperature range (550-1050 K). The maximum energy of the incident D ions was 1 keV.

Steady state permeation through the recrystallized W2 was lower by one order of magnitude in comparison to the 1300 ºC annealed W1 specimen, while the lag times were comparable. This indicates that the bulk trapping properties were similar but the lattice concentration at the irradiation side was significantly lower for the case of recrystallized W2 specimen. In contrast, the lag times of IPP specimen W3 was longer by one order of magnitude in comparison to W1, while the steady state permeation was similar. This indicates different bulk trapping properties, but similar lattice concentration at the irradiation side. We present diffusivity data calculated from simple diffusion theory for all three W specimens. Furthermore, we discuss if the two most common assumptions in modeling and interpreting hydrogen diffusion and trapping in W hold in the case of ion driven permeation. Namely: 1) hydrogen diffuses via a single site (interstitial) with Frauenfelder's [3] diffusivity, and 2) traps are in local thermodynamic equilibrium and their concentrations are small in comparison to the hydrogen concentration in the lattice.

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Gaseous hydrogen permeation through CMSII coated and ITER-grade bulk tungsten

P42B

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Tungsten has been identified as a crucial plasma-facing material in future fusion reactors. It is considered particularly suitable for plasma facing high heat flux materials due to its high melting point, high thermal conductivity, low vapor pressure, very high threshold for sputtering, and presumably low tritium retention property.

The only large-area application of tungsten to existing fusion reactors has been done in ASDEX Upgrade and is currently in progress in JET ("ITER-like wall"). Combined Magnetron Sputtering and Ion Implantation (CMSII) technology was used for W coating of carbon based materials (Carbon Fibre Composite - CFC and fine grain graphite) since only this coating method provided no deterioration during demanding thermo-mechanical tests [1]. The coating thickness was 10-15 μm or 20-25 μm depending on the position of the tile at the wall. While the thermo-mechanical and structural properties of such coating are well known, not much is known about its hydrogen interaction. The later is particularly important for the assessment of tritium retention. Due to the low hydrogen diffusivity and very small volume of W in the coated layer, the gaseous hydrogen permeation measurement at 400°C was selected for the experimental technique, where increasing & decreasing transient and steady state permeation flux was monitored. Problems that could arise with the CFC membrane sealing were overcome by deposition of the identical W layer on the 0.5 mm Eurofer substrate. Two such double-layer membranes were investigated. Obtained hydrogen permeability in tungsten layer (~10^{-13} mol H_2 /m s $Pa^{0.5}$) is comparable to the upper range of published data. Measured diffusivity (~10^{-14} m^2/s) is several orders of magnitude lower compared to the average of published data for tungsten, while the measured solubility (~1 mol H_2/m^3 $Pa^{0.5}$) is several orders of magnitude higher. The explanation can be given in terms of hydrogen trapping that has significant impact on hydrogen migration.

The membrane made from bulk ITER-grade tungsten was also investigated in our experimental setup since a considerable degree of scatter in hydrogen transport parameters is present in published data. The results obtained on CMSII coated and ITER-grade bulk tungsten are compared and discussed.

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Phase formation of Erbia coatings on Eurofer, phase stability and deuterium permeability

P43A

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Erbia is a promising material for reduction of hydrogen permeation in future fusion reactors, since it reduces permeability by a factor up to almost 1000 [1]. There exist two crystallographic modifications: the cubic phase and the monoclinic phase [2]. Both phases were investigated concerning hydrogen permeation and phase stability under high energy ion irradiation.

Thin films of Erbia were deposited on Eurofer steel substrates by a filtered cathodic arc, varying the substrate temperature (RT – 600 °C) and sample bias (0 – -450 V). Deposition at 600 °C without bias led to solely formation of the cubic Er_2O_3 phase. Thin films of the monoclinic B-phase can be prepared in good crystallinity when a negative bias voltage of 250 V is applied at 400°C [3]. The crystallographic phases were analyzed by X-ray diffraction.

Neutron irradiation was simulated by 5.5 MeV Au++ irradiation in a special beamline of IPP tandem accelerator, which led to max. 100 dpa within 5 hours. During the irradiation the cubic phase changed partly into the monoclinic phase. The monoclinic phase showed no changes.

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Influence of Silicon and Temperature on Hot-dip Aluminizing Process and Subsequent Oxidation for Preparing Tritium Permeation Barrier

P43B

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In ITER D-T fuel will be used for fusion plasma. Therefore a material capable of acting as a tritium permeation barrier on stainless steel is required. It is well known that thin alumina layer can reduce the tritium permeation rate by several orders of magnitude.

A technology is introduced here to form a ductile Fe/Al layer on the 316LN steel with an alumina over-layer, which consists of two main steps, hot-dip aluminizing (HDA) and subsequent oxidation process. According to the experiments that have been done in pure Al, the coatings were inhomogeneous and too thick. Additionally, a large number of cracks and porous band could be observed.

In order to solve these problems, the element silicon was added to the aluminum melt with a nominal composition. The influence of silicon on the aluminizing and following oxidation process was investigated. With the addition of silicon into the aluminum melt, the coating became thinner and more homogeneous. The effort of the silicon on the oxidation behavior was observed as well concerning the suppression of porous band and cracks. The influence of temperature on the formation of ductile phases and alumina over-layer in the oxidation treatment has been also studied. At the temperature of 900℃, the brittle Fe-Al intermetallic layer and aluminum coat transformed into ductile phase (FeAl and α-Fe(Al)).

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An advance process of aluminum rich coating as tritium permeation barrier on 321 steel workpiece

P44A

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Preparation of tritium permeation barrier (TPB) with highly performance to reduce tritium penetration through on complex structures is a major problem in fusion engineering. The best solution was identified in aluminium rich coatings, which form Al_2O_3 at their surface [1]. We proposed an advance three-step process, Al-electroplating in ionic liquid followed by heat treating and artificial oxidation, preparing such aluminum rich coating [2].

In present work, the advance process was applied to 321 steel workpieces. In the Al-electroplating, pieces were coated Al coatings by galvanostatic electrodeposition at 20mA/cm^2 in aluminum chloride ($AlCl_3$)-1-ethyl-3-methylimidazolium chloride (EMIC) ionic liquid. The Al layers on those steel components all display attractive brightness and well adhere to the surface. Within the aluminizing time from 1 to 30h, a series of experiments were carried out to aluminize 321 steel pieces with Al 20μm coating at 700 °C [3]. After heat treated for 8h, a 30μm thick aluminized coating appears homogeneous, free of porosity, and exhibit a three-layer structure consisting of an outer (Fe,Cr,Ni)Al_2 layer, transitional (Fe,Cr,Ni)Al layer and inner $(Fe,Cr,Ni)_3Al$ layer, and then was selectively oxidized in argon gas at 700 °C for 50 h to form Al_2O_3 scale. The finally fabricated aluminum rich coating has a double-layered structure consisting of an outer γ-Al_2O_3 layer with thickness of 0.2 μm and inner (Fe,Cr,Ni)Al/$(Fe,Cr,Ni)_3Al$ layer of 50 μm thickness, without any visible defects. Deuterium permeation tests were taken to examine hydrogen isotope permeation property of such coating on one end closed piece (Φ80X2, L150mm). The deuterium permeation rate through the coated piece is reduced by 2-3 orders of magnitude at 600~727 °C, .i.e., the PRF value is about 131 at 727 °C and about 2262 at 600 °C.

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P44B

Ion-driven Permeation of Deuterium in Tungsten by Deuterium and Carbon Mixed Ion Irradiation

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Tungsten (W) is a candidate for plasma facing material in future fusion reactors and ITER. Although tungsten has low hydrogen retention under hydrogen isotopes (H, D, T) irradiation, hydrogen will enter from the W surface and diffuse towards the coolant structures. Diffusion affects hydrogen refuelling rates, retention, and permeation, which are important in determining the operational cycle of a reactor. Since tritium is a radioactive material, hydrogen diffusion also affects safety aspects. The diffusion behaviour of hydrogen isotopes in W can be evaluated by studying the permeation of deuterium in W.

In ITER, the existence of other plasma facing materials such as carbon-fibre composites will result in the release of carbon (C) impurities through erosion and transport processes. This results in the simultaneous irradiation of W by hydrogen and carbon impurity ions. At present, only single D ion-driven permeation experiments in W or C layers deposited on W have been performed, but the study of the D permeation in W under D+C mixed ion irradiation has yet to be carried out. In order to study such effects, we simultaneously irradiate W with D and C ions, and compare the difference in permeation with D-only irradiation.

The experiments were performed with a high flux ion beam test device (HiFIT) [1], where the incident energy of the ions was 1 keV at a flux of 10^{20} $m^{-2}s^{-1}$. The W specimens used were of thickness of 30 μm, with purity of 99.99%. The irradiation chamber and the permeation chamber were isolated by sealing the W specimen with a copper gasket. The D permeation flux was measured by using a quadrupole mass-spectrometer. The main experimental parameters varied were the specimen temperature (550 – 1050 K) and the C fraction in the incident flux (0.1 – 3%). Following D+C permeation experiments, depth profiles of the implanted C at the front surface were measured.

In comparison to D-only irradiation, D+C mixed ion beam irradiation (~1% C) resulted in a higher steady state D permeation flux, indicating surface modification by incident C ions. The increase in the D permeation flux varied from a factor of two to two orders in magnitude depending on the irradiation temperature. Under simultaneous D+C irradiation, the system develops into two distinct steady state regimes of C layer growth or formation of a dynamic C-W mixed layer formation [2]. We characterize the front surface of the W specimens from the measured C depth profiles, and correlate the increase in steady state D permeation flux with the respective modified front surfaces.

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Influence of He implantation and W self-implantation on deuterium permeation through tungsten

P45A

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During the operation of ITER as plasma-facing material tungsten will be subjected to high fluxes of low energy particles including He and 14 MeV neutrons originating from the nuclear reaction. Tungsten was chosen as a candidate material for the ITER divertor due to its good thermal properties and low sputtering yield. In previous studies [1,3] it was found that hydrogen retention in the material is strongly affected by He pre-implantation [1]. The accumulation of radiation damage due to the neutron bombardment is supposed to change the hydrogen retention in tungsten. Various studies of the influence of radiation damage on hydrogen retention using heavy ion implantation to model the neutron irradiation of the material were recently published [2]. However, in these studies the diffusion properties of deuterium in damaged tungsten were obtained from depth profile measurements using IBA methods. The ion-driven permeation experiment gives a direct information about diffusion of deuterium in material and it's trapping behaviour.

The effect of neutron bombardment was modelled by implantation of the target with a high energy tungsten ion beam (so-called self-implantation) [2]. Tungsten samples cut from a cold-rolled tungsten foil with a purity of 99,9% produced by Plansee (Austria) were implanted by 5,5 MeV W^{2+} and 20 MeV W^{6+} ion beams at the tandem accelerator (IPP, Garching, Germany) to the different fluences. Measurements of deuterium permeation through these damaged samples were then performed at the HiFIT (Osaka, Japan) and PERMEX (Garching, Germany) devices. Depending on the experiment the damaged zone was situated at the front or the rear side of the specimen.

The influence of He on hydrogen permeation was also studied in two steps. First the samples were exposed to a He plasma (500 eV, implantation fluence 10^{20} cm^{-2}) at the KESCABO facility (IPP, Garching, Germany), the permeation measurements were then performed in the PERMEX device using a low energy (200 eV/D^+) D^{3+} ion beam. The He exposed side in this measurements was on the front. The amount of He was determined twice - before and after the permeation measurement using the ERDA technique. The specimen surfaces after He and after D implantation were investigated by SEM.

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P45B

Isotopic exchange in ITER-grade tungsten exposed sequentially to low-energy, high-flux deuterium and protium plasmas

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Deuterium retention in polycrystalline ITER-grade tungsten (A.L.M.T. Corp., Japan) exposed both to a deuterium plasma and sequentially to deuterium and protium plasmas have been studied. This work was performed to reveal how isotopic exchange during plasma operation is effective to remove tritium retained in the plasma facing surface in a DT reactor.

The linear plasma generator (JAEA, Tokai) was used for delivering plasma beams at various exposure temperatures. A bias voltage of -80 V was applied to the W sample, resulting in an incident energy of 76 eV for D_2^+ or H_2^+ (38 eV per D or H), taking into account the plasma potential of about -4 V as measured by a Langmuir probe. The incident deuterium and protium ion fluxes and fluences were fixed at 10^{22} D(H)/m^{2}s and 10^{26} D(H)/m^2, respectively. The D retention and D depth profiles in the W samples were examined correspondingly by thermal desorption spectroscopy (TDS) and the D(^{3}He,p)^{4}He nuclear reaction at a ^{3}He energy varied from 0.69 to 4.0 MeV (NRA) allowing determination of the D concentration up to a depth of 7 μm.

After exposure to the D plasma, the deuterium retention increases with the exposure temperature, reaching a maximum value of about 2.7×10^{22} D/m^2 at 490 K, and then decreases as the temperature rises further. The high D concentration (about 0.3 at.%) at depths of several micrometers, observed after exposure to pure D plasma at temperature of the maximum D retention, is due to accumulation of D_2 molecules in voids created during the D plasma exposure. After sequential H plasma exposure at about the same temperatures as for D plasma exposure, the D retention reduces by a factor of 4-10, depending on the exposure temperature. The removal efficiency for D as function of depth in the W sample indicates large diffusion depths for the sequentially implanted hydrogen at temperatures above 490 K.

Stress-induced plastic deformation caused by deuterium super-saturation within the near-surface layer and formation of superabundant deuterium-vacancy clusters are suggested as mechanisms for nucleation and growth of microscopic cavities which are responsible for deuterium trapping. At sequential protium plasma exposure, further plastic deformation leads to exchange of considerable proportion of deuterium by protium.

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Deuterium implanted into polycrystalline tungsten: Novel TPD investigations

P46A

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Deuterium retention in tungsten is often studied by temperature programmed desorption (TPD) in order to determine the total deuterium inventory and the binding energies of the deuterium to the trap sites. However, in the analysis of TPD results it has to be kept in mind that the resulting D_2 release spectrum is influenced not only by the binding energy distribution but also by the depth distribution of the trapped deuterium and the heating ramp of the TPD experiment. Especially for elevated temperatures during implantation, deuterium can diffuse deep into the bulk of the sample. This can lead to a change of the shape as well as the position of the release peaks, which makes the interpretation of the release spectra difficult.

The effects of deuterium release from trap sites, diffusion and the TPD heating ramp can be separated by comparing the release spectra of identical deuterium loaded tungsten samples for different heating ramps and for interrupted heating ramps. This ramp and hold technique has already been successfully applied, e.g., for the determination of the binding energy spectrum of hydrogen in amorphous hydrocarbon thin films [1].

Previous experiments on D retention in W show that the deuterium inventory is typically three to four orders of magnitude lower than the implantation fluence. With increasing fluence the amount of retained D increases monotonically and approaches saturation at several 10^{20} D/m^2 around fluences of 10^{24} D/m^2. D_2 desorption is characterized by two desorption peaks at 520 and 680 K at a heating ramp of 1.4 K/s. Deuterium depth profiling by nuclear reaction analysis (NRA) shows a sharp surface peak and a broader secondary peak at a depth of approximately 1 µm with a subsequent exponential decay into the bulk [2].

This presentation shows the results of such ramp and hold experiments for polycrystalline tungsten that was mechanically polished and subsequently stress-relieved and degassed. This grade of tungsten was thoroughly characterised in prior work [2]. The deuterium implantation was performed in a fully quantified deuterium plasma source [3] at a specimen temperature of 370 K. Samples were implanted with a fluence of 6×10^{24} D m^{-2}. The deuterium depth profile was determined by nuclear reaction analysis (NRA) with a ^{3}He beam prior to the TPD experiments and subsequently used for the modelling of the D_2 release spectra.

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Modelling thermal desorption from tungsten surfaces containing hydrogen precipitates: a continuum-scale approach

P46B

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Exposing tungsten to high-flux hydrogen plasmas leads to the formation and growth of sub-surface gas bubbles under appropriate conditions. When precipitation is favored thermodynamically, trapping by bubbles can deplete the concentration of hydrogen in solution and reduce the diffusing flux to traps located deep within the material [1]. Hence, accurately modelling hydrogen precipitation in tungsten is particularly important for predicting the tritium inventory in the ITER divertor. Much of the existing understanding of hydrogen uptake has been derived from thermal desorption spectroscopy experiments. As a pathway toward modelling the effects of precipitation on retention, we simulate the release of hydrogen from bubbles during heating after plasma exposure. It is common practice to fit desorption peaks by modelling the release from atomic trap sites of different energies and adjusting the trap concentrations to match the experimental spectra. However, in this work we are able to simulate the release of hydrogen from bubbles and show how it also contributes to desorption spectra. Simulations of thermal desorption that take into account precipitation can therefore improve the interpretation of existing laboratory results and are important for testing the present understanding of retention mechanisms.

To simulate hydrogen release from bubbles, we leverage our previously developed continuum-scale model for tungsten which incorporates bubble growth by dislocation loop punching [2]. Based on recent post-mortem focused ion beam profile studies [3,4], this growth mechanism appears predominant in recrystallized tungsten after plasma exposure at moderate temperatures (< 700 K). Our 1-D diffusion model incorporates an equation of state that takes into account real gas effects at high pressure. This relationship is used to determine the stresses exerted on the surrounding metal. We apply this same equation of state to calculate changes in the pressure and fugacity of the hydrogen within the bubbles during a thermal desorption ramp. The aforementioned microscopy studies provide guidance on the expected bubble distributions. For polycrystalline metals with smaller grain sizes, the growth of gas-filled subsurface structures appears more complex, possibly indicating that crack propagation could be a mechanism for expansion. Using our model, we evaluate mechanisms that contribute to the growth of these structures.

Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under Contract DE-AC04-94AL85000.

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Deuterium retention in Tungsten-Tantalum alloys

P47A

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The use of tungsten (W) as a first wall material is hampered by its brittle nature which makes it not very well suited for withstanding the cyclic power loading conditions at the first wall of a magnetic confinement fusion device. To alleviate this problem alloyed W with a higher ductility is being developed. Promising candidates are W-Ta alloys with an alloying fraction in the range of several % of Ta. However, when choosing a first wall material one must consider not only its thermo-mechanical properties, but also its compatibility with the D-T plasma environment. One important requirement for a first wall material is a low level of hydrogen retention. While W is known for its low retention the addition of a hydride former like Ta might increase retention in the W+Ta alloy compared to pure W.

To investigate the influence of % levels of Ta in W on the retention of D, pure well characterized [1] W samples and W+Ta samples were exposed under identical conditions to energetic deuterium bombardment. The experiments were performed both with mass separated D ion beams and with D ions from a D plasma, accelerated by a DC bias. The use of ion beams and plasmas allowed to span wide energy (keV to eV), flux (10^{18} - 10^{22} $m^{-2}s^{-1}$) and fluence (10^{23} – 10^{25} m^{-2}) ranges.
After implantation the retained D was analysed using thermal desorption (TDS) and nuclear reaction analysis (NRA). TDS thereby yields the total amount of retained D and gives information about the D binding states in the materials. NRA complements TDS by allowing to measure quantitative depth profiles of D from within the first ~10 μm.
Due to the importance of the material microstructure on D retention, the W+Ta samples were also analyzed by SEM & FIB cross sectioning and by XRD to obtain information about the crystallite size and the quality of alloying.

We will present the ratio of D retention in W+Ta alloys to that in pure W which, depending on implantation conditions, ranges from 5 to 10. This difference in retention will be discussed based on the binding energies obtained from TDS, the D depth profiles from NRA and the structural information from SEM & XRD.

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P47B

Deuterium retention in tungsten-tantalum alloys

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Tungsten is considered as a material for the divertor of ITER, and the divertor and the first wall of DEMO [1]. The practical use of tungsten is hindered by its high DBTT and therefore high brittleness at the temperatures of operation. In order to improve the ductility and thus the machinability tungsten alloys are considered. It was demonstrated that tungsten-tantalum alloy has superior thermo-mechanical properties comparing to pure bulk W. One of the issues still to be clarified is the retention of hydrogen isotopes (including radioactive tritium) in this alloy and the influence of various factors (such as the presence of helium ash, neutron irradiation, thermal shock etc.) on it. While there exists a considerable database related to hydrogen isotopes retention in pure tungsten (e.g., [2, 3]), no data are available concerning retention in binary alloys and in W-Ta alloy in particular.

W-Ta samples were exposed to deuterium plasma in low flux, high temperature plasma simulator VISIONI (*V*ersatile *I*nstrument for *S*tudy of *ION* *I*nteraction), installed at Belgian Nuclear Research Centre SCK-CEN. This device is able to reproduce conditions expected at the first wall, as well as the divertor dome and baffles, of ITER [4, 5]. Investigated materials included two grades of W-Ta alloy, namely one with 1% Ta content and one with 5% Ta, as well as reference pure W and Ta. Subsequent studies of deuterium retention were performed with thermal desorption spectroscopy (TDS) and secondary ion mass spectroscopy (SIMS). Modification of surface morphology due to plasma exposure was studied by scanning electron microscopy (SEM).

It was demonstrated that deuterium retention in the alloy is strongly influenced by the presence and the amount of Ta. Comparison of retention data obtained for grades with different Ta contents and reference data obtained for pure materials is presented in the paper. The mechanism of interaction between D atoms and Ta, dissolved in a W matrix, is discussed.

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Mechanical, Erosion and Permeation Properties of Nanostructured W and W-Ta Coatings

P48A

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Full W first wall may represent one of the solutions foreseen for next generation tokamaks.
The features of W, and particularly W coatings, are strongly influenced by the structure and morphology of the layer at the nanometric scale (see e.g. [1]). In addition, interaction with plasma determines morphological modifications like the formation of porous open tungsten structures called “fuzzy tungsten” that exhibit strongly different properties if compared with the normal film [2].
In order to investigate this problem we deposited W and W-Ta coatings with different structure and morphology, on relevant substrates (EUROFER, CFC), exploring the possibilities offered by the Pulsed Laser Deposition. Thanks to the flexibility of this technique it is possible to study a very wide range of coating properties, like composition, structure, crystalline growth and morphology (from compact to open, fuzzy-like structures), by changing the process parameters, e.g. laser fluence, background gas pressure, target composition, substrate temperature and subsequent annealing. The deposited films have been characterized by high resolution SEM, XRD, EDS and XPS.
We deposited both highly crystalline oriented W micrometric-thick coatings with a columnar structure, amorphous like films with a random growth, porous “fuzzy like” films and multilayered systems. We also produced W-Ta films with different fraction of Ta, to begin an investigation of W-Ta alloys.
Mechanical and adhesion properties have been tested using scratch test (adhesion) and micro/ nano-indentation (mechanical properties).
Permeation measurements on W and W-Ta films on EUROFER substrate are presented. This campaign allows precise determination of hydrogen interaction with W films.
In order to investigate the erosion effects on such systems the W coatings will be exposed to ion flux from plasma produced by capacitively coupled RF plasma (CCP) using hydrogen and noble gases. The erosion phenomena will be evaluated by measuring the thickness change of the layers by profilometer. Additional plasma exposures are foreseen.

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Deuterium retention in various pre-damaged materials as a function of the exposure temperature

P48B

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During ITER operation, 14 MeV fusion neutrons will not only introduce new traps for tritium in tungsten but also, after one year, will transmute it into an alloy containing about 3% Re [1]. To account for these effects, deuterium retention in different batches of irradiation damaged tungsten materials, including 97W-3Re samples, has been investigated as a function of the exposure temperature. All samples were self-implanted above the saturation dpa (displacements per atom), i.e. 0.25 dpa (for E_{th} = 90 eV), using 20 MeV W ions and then exposed to high-flux, low-energy D plasma at various temperatures ranging from 350 to 750 K.

The deuterium depth profiles were measured by means of NRA using the D(^{3}He,α)p reaction, and based on these results the dependence of the D concentration in the radiation-induced defects on the exposure temperature was obtained for all materials.

The results indicate that in all investigated materials, D concentration in the radiation-induced defects decreases with increasing exposure temperature. In all other samples, the D concentration is comparable, but for 97W-3Re it drops somewhat faster with the temperature, a behavior which was not observed in undamaged W. The reason for that can be found in TEM observations of neutron-irradiated pure and Re-containing tungsten [2]. These studies show that in the presence of even small amounts of rhenium, the number of defects created during damaging is lower in comparison with pure tungsten.

Based on these and our previous studies [3], one can derive a function for the trap density dependence on the plasma exposure time and wall temperature which can be used for a tritium inventory prediction for ITER.

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Interaction of D and He Plasmas with Tungsten in Fuego-Nuevo II

P49A

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Tungsten is one of the main candidate materials for plasma-facing components in a future fusion power plant. The experience with tungsten as plasma facing material is less than with graphite, and a need for a comprehensive database has been identified [1]. Many studies simulate the plasma wall interaction using ion beams [2], while only a few use plasma simulators [3,4] . Recently the question of the interaction of neutrons added to the interaction of mainly helium plasmas with tungsten is gaining relevance [5].The Fuego-Nuevo II [6] is a plasma focus device which can produce dense magnetized helium and deuterium plasmas. Additionally it produces a significant amount of high energetic neutrons during deuterium operation.

In this paper we present preliminary results of tungsten targets exposed to deuterium and helium plasmas in the Fuego Nuevo II device..

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The TRitium Ion Implantation eXperiment (TRIIX) for hydrogen isotopes retention measurements in plasma facing materials

P49B

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The TRitium Ion Implantation eXperiment (TRIIX) was built in the early 1980's at the Idaho National Laboratory and successfully operated until the late 1990's [1-5]. The system has recently been refurbished to perform the study of retention and desorption behavior of hydrogen isotopes implanted in plasma facing component materials used in fusion reactor. TRIIX is currently capable of generating an intense beam of deuterium ions with tunable mass, energy (between 0.4 ~ 10 kV) and flux density (between 10^{18} ~ 10^{20} ion/m^2.s). The beam is generated by a commercial Duoplasmatron ion source [6], an ion accelerator and decelerator, a mass analyzer and two electromagnetic lenses. The focused beam is directed into an implantation chamber where targets are maintained in UHV conditions and at controlled temperature up to 700^{o}C. The chamber includes a vacuum manipulator to load samples in vacuum and a pumping system providing ultra high vacuum for the ion beam, implantation, and sample transfer volumes (background vacuum typically below 10^{-6} Pa). The target chamber is also equipped with a Faraday cup coupled with a final focusing aperture to control the beam size and measure its intensity.
TRIIX is located in the Safety and Tritium Applied Research (STAR) facility [7], a US DoE National User facility operated by the INL Fusion Safety Program that is capable to handle both tritium and radioactive samples. It is currently used in the frame of the US/Japan TITAN[7] collaboration to investigate deuterium retention in plasma facing materials, in particular tungsten. Results of tungsten exposure to deuterium ions and subsequent Thermal Desorption System (TDS) analysis are reported in this paper mainly to validate facility operation by comparison with reference data. Planned activities to investigate the synergistic effect of neutron and deuterium irradiation by testing samples that have been irradiated in the ORNL HFIR reactor are outlined. In particular, neutron damaged nickel samples will be tested to validate modeling results of neutron damage effects on deuterium retention with experimental data.

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Hydrogen Retention in Tungsten Exposed to High-Flux Plasmas

P50A

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In future magnetic fusion devices as ITER, tungsten is foreseen as material for the divertor. Tritium retention in W 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 on the deuterium retention in tungsten exposed to high-flux, low temperature plasmas.

Polycrystalline, W targets were annealed for 1 hour at 1000°C and pre-irradiated with 12.3 MeV W^{4+} ions as proxy for neutron damage. Damage profiles, simulated by TRIM, extend to 1.5 μm below the surface and show peak damage levels of 0.1-1.0 dpa (displacements per atom). The damaged targets have been exposed in Pilot to plasma fluxes of ~10^{24} ions $m^{-2}s^{-1}$. To investigate the influence of the surface temperature, exposures were done at two different temperature regimes (<500K and 650-950K) by using two different ways of cooling.

Positron Annihilation Spectroscopy was used to study the behaviour of the vacancies by pre-irradiation and plasma exposure. Non-irradiated and irradiated targets showed a clear distinct Doppler broadening profile. The targets exposed to Pilot-plasmas showed a similar behaviour of the bulk material as the non-exposed targets, but with a longer diffusion length.

The deuterium retention has been studied by Nuclear Reaction Analysis and Thermal Desorption Spectroscopy. TDS and NRA show saturation in deuterium retention at 0.2-0.5 dpa and seem to depend on target temperature. The values found for the saturation level are in good agreement with Tyburska et al.[1]. Since their experiments have been carried out at much lower plasma fluxes, it can be concluded that the saturation level is not significantly affected by the high fluxes used in our experiments.

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Deuterium Depth Profile in Neutron-Irradiated Tungsten Exposed to Plasma

P50B

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Plasma-facing components will be exposed to 14 MeV neutrons from the D-T fusion reaction, and is expected to receive a neutron dose of 0.7 displacement per atom (dpa) by the end of operation in ITER. The effect of neutron-irradiation damage has been mainly simulated using high-energy ion bombardment. The ions, however, are limited in range to only a few microns into the surface. Hence, some uncertainty remains about the increase of trapping at radiation damage produced by neutrons, which penetrate much farther into the bulk material. With the Japan-US joint research project: Tritium, Irradiations, and Thermofluids for America and Nippon (TITAN), the tungsten samples (99.99 % pure from A.L.M.T., 6mm in diameter, 0.2mm in thickness) were irradiated to high flux neutrons at 323K (50C) and to 0.025 dpa in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). Subsequently, the neutron-irradiated tungsten samples were exposed to a high-flux deuterium plasma (ion flux: 10^{21}-10^{22} $m^{-2}s^{-1}$, ion fluence: 10^{25}-10^{26} m^{-2}) in the Tritium Plasma Experiment (TPE) at Idaho National Laboratory (INL). First result of deuterium retention in neutron-irradiated tungsten exposed in TPE was reported previously [1]. This paper will present the latest results in our on-going work of deuterium depth profiling in neutron-irradiated tungsten exposed to plasma at 373K (100C), 473K (200C), and 773K (500C) via nuclear reaction analysis (NRA). The experimental data is compared with the result from non neutron-irradiated tungsten, and is analyzed with Tritium Migration Analysis Program (TMAP) to elucidate the hydrogen isotope behavior such as retention and depth distribution in neutron-irradiated and non neutron-irradiated tungsten.

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Hydrogen gas filled cavities under surface extrusions on hydrogen-implanted tungsten

P51A

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Hydrogen-implanted tungsten is extensively investigated, e.g., regarding hydrogen retention and surface morphology changes. The reported surface extrusions vary from spherical, more classical blisters [1] to stepped, high-domed structures [2,3]. Further, in the subsurface a spectrum of different crack and cavity features is observed [3]. The presence and absence of features under similar exposure parameter (fluence, flux, ion energy, implantation temperature, specimen pre-treatment) as well as the dramatic variation in their size draw a confusing picture of the morphological modifications described in the literature.

In this study, the three-dimensional morphology of surface extrusions on various tungsten grades and the gas filling of cavities beneath the surface are analysed applying scanning electron microscopy (SEM) assisted by cross-sectioning with a focused ion beam (FIB) and by calibrated residual gas analysis with a quadrupole mass spectrometer.

On polycrystalline hot-rolled tungsten, spherical blisters are formed by deuterium (D) bombardment (33 eV per D) up to several 10^{24} D/m^2. Their size depends on implantation temperature (300–600 K) and ranges from several µm up to a few hundred µm. Individual blisters were sequentially punctured with FIB and imaged with SEM until an elastic relaxation was observed. A D_2 spike was observed simultaneously with the relaxation if the volume of the blister cavities was large enough [4]. By measuring the blister volume and the amount of released D_2 molecules, the pressure can be determined to be of the order of 1 GPa. The subsequent cross-sectioning shows that delamination along grain boundaries roughly parallel to the surface occurred. A blister cap of more than 10 µm was observed for blister with a lateral expansion of >100 µm. The cap contains about 10 layers of grains.

On recrystallized tungsten, the extrusions formed by deuterium implantation range from small in-grain crack systems to large stepped features with large cavities at grain boundaries. The cavities are in many cases not located directly under the extruded material, but also shifted laterally. Because their formations mechanism was elucidated as gliding along the low-indexed slip systems {110}<111> [3], the position of the cavity could be proposed and, therefore, the targeted degassing of the cavity with FIB can be performed. The results of such degassing measurement will be discussed together with classification regarding the proposed formation mechanisms of the various surface extrusions.

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P51B

Hydrogen Interaction With Tungsten Surface

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Tungsten is the material used for high heat load PFC in both current and future fusion devices, such as ASDEX Upgrade and ITER. The interaction of hydrogen with tungsten is important in order to understand the retention and recycling of hydrogen on the walls of reactor. We are studying the processes occurring on tungsten surface, which is exposed to neutral hydrogen atoms and molecules. The binding energy of hydrogen atom on tungsten is usually taken to be 2.9-3 eV. Still tungsten exhibits high hydrogen saturation concentration above 1 ML (1 ML - one hydrogen atom per one tungsten atom) and hydrogen atoms are adsorbed in several binding sites having different desorption energy. Polycrystalline tungsten is characterized by a mixture of the binding sites observed for the single-crystal surfaces [1].

We have studied the interaction of hydrogen atoms and molecules by ion beam methods ERDA and RBS. A polycrystalline tungsten sample (Plansee - ITER grade, 99.99% purity) was exposed to hydrogen atom beam and the surface was monitored by ERDA and RBS in real time. The sample was initially cleaned by hydrogen atom beam at high temperature (300^{o}C), where it is known that atoms erode the hydrocarbon [2] and also oxygen layer [3]. Three different saturation concentrations were found in different temperature ranges that can be correlated to three desorption energies. This is in agreement with the measurements of Tamm and Schmidt [1]. The time variation of the hydrogen concentration is calculated by a kinetic model of Jackson et al. [4], extended by including the Langmuir-Hinshelwood recombination contribution. The hydrogen atom recombination with hydrogen atoms occupying different binding sites also explains our experimentally obtained vibrational temperature of recombined molecules produced on tungsten. Namely, the vibrational populations for tungsten are higher and show a considerable overpopulation of highly excited vibrational states, than predicted by assuming only single site with binding energy of 2.9-3 eV. In order to obtain an insight into the recombination mechanism with more than one binding site per unit cell, a Monte-Carlo simulation was performed, confirming the above interpretation [5].

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Simulation of Hydrogen Retention and Re-emission from Tungsten Exposed to Divertor Plasmas

P52A

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In addition to co-deposition with eroded Be from the first wall, implantation and retention of plasma ions in tungsten is important for tritium inventory in the divertor of ITER. In tungsten, hydrogen isotopes are highly mobile and are retained in radiation damage sites or defects of the crystal lattice. After saturating available traps, inward diffusion and subsequent trapping increase the inventory. In order to study such mechanisms, the diffusion process [1] was coupled to a Monte Carlo code, EDDY [2]. After benchmarking with an existing TDS experiment and other codes (PIDAT [3] and ACAT-DIFFUSE [4]), tritium retention profiles on inner and outer targets and dome regions, made of tungsten, are calculated for an ITER diverter plasma configuration. Parameter studies on tritium retention and re-emission from the plasma-facing tungsten are also presented with probable range of the material parameters.

Slowing down process of impinging ions and the depth profile of thermalized particles are calculated with EDDY (based on the binary collision approximation) for pseudo particles, each of which represents a differential ion fluence. The depth profile is used as input profile in DIFFUSE, where a diffusion process is calculated with a diffusion time (1 s), and the profile after diffusion serves as input for the calculation with the next ion fluence. This sequence is performed N (e.g., 400) times to simulate the time evolution of retained tritium in tungsten. In the diffusion process, trapping in and detrapping from different trap sites and surface recombination with given rate coefficients are taken into account.

The parameters of a real-sized divertor plasma of ITER are taken from a B2/EIRENE code calculation. Using velocities of deuterium (D) ions along magnetic field lines and the angles intersecting target plates, the ion fluxes are calculated. In ITER, the toroidal magnetic field lines intersect the target surface at very shallow angles between 1^o and 3^o, where the gyro-motion influences the angle of incidence of the impinging ions. The angular distribution of the ions is calculated using a one-dimensional PIC simulation with three dimensions in velocity space, where the gyro-motion of the ions is taken into account [5].

The EDDY/DIFFUSE calculation reproduced well the observed temporal evolution of released thermal D flux in the experiment, as did by other codes. Nevertheless, strict comparisons with the experiment and predictions of long-term tritium retention in the divertor are strongly influenced by material properties, which are input parameters in the calculations.

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Effect of nitrogen seeding into deuterium plasma on deuterium retention in tungsten

P52B

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Tungsten (W) is the currently used plasma-facing material for ASDEX Upgrade and a reference material for the high-flux, low-ion-energy region of the divertor in ITER. To reduce the power flux onto the divertor and limiter tiles cooling by impurity seeding into the plasma is necessary. In ASDEX Upgrade nitrogen (N) puffing through nozzles in the divertor roof baffle reduces the divertor temperature and power flux as well as the ELM size. In addition, the energy confinement is also improved [1]. However, the question about an influence of N on the fuel (deuterium (D) and tritium) retention in tungsten is still open. Therefore, dedicated laboratory experiments have been performed to investigate the interaction of low-energy N-seeded D plasma with W. Tungsten samples were exposed to 1-5% N-seeded D plasma generated by electron-cyclotron-resonance (ECR) plasma source. Different DC bias voltages were chosen and applied to the substrate holder to accelerate the ions in the sheath leading to an energy of about 60, 100 and 200 eV per ion. Irradiation temperatures were ranging from 300 to 800 K.

The D and N retention in each sample was subsequently analysed 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 and N. It is found that the N amount in W is (7-9)x10^{19} N/m^2 in the temperature range between 300 and 650 K and slightly decreases down to 6x10^{19} N/m^2 at 800 K. This means that the nitride compound formed upon N-seeded D plasma exposure of W is thermally stable and does not decompose at least up to 800 K. Highest nitrogen concentration of 10^{20} N/m^2 was observed at highest DC bias used in the present investigation. The saturation of the W surface with N within the implantation zone was observed already after 30 minutes of plasma exposure.

Changes of morphology and composition in the surface layer by plasma exposure were investigated by analytic scanning electron microscopy. It is shown that seeding of nitrogen into D plasma does not prevent blister formation and even results in the increasing of the size of blisters in some cases. This is in agreement with depth profile measurements which show an enhancement of the D diffusion into the bulk and, consequently, an increase of the D retention in W in the presence of the N-seeding in D plasma compared to pure D plasma. Therefore, it is assumed that N-containing compounds on the W surface act as a diffusion barrier for D desorption leading to an increased D diffusion into the bulk.

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Ab-initio Modelling of Point Defects and Anisotropic Elasticity Effects in Tungsten-Vanadium and Tungsten-Tantalum Alloys for Fusion Application

P53A

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W-V and W-Ta alloys are candidates for DEMO divertor applications because of their high melting point and expected improved ductility and fracture toughness in comparison with tungsten. We have recently identified the lowest energy inter-metallic compounds, which should form at low temperatures, by using first-principles density functional theory (DFT) calculations in combination with Monte-Carlo simulations [1,2]. The predicted temperature of order-disorder phase transformations is relatively low and at high temperature it is found that short-range orders are present for both alloys.

In this work, a systematic DFT study of point defects (vacancy and self-interstitial atoms (SIA)) as function of vanadium and tantalum concentrations in these binary alloys has been carried out. *Ab-initio* calculations show that vanadium atoms strongly trap SIA defects in W-V alloys, whereas Ta atoms in W-Ta alloys have very little effect on either the formation energy or thermally activated mobility of self-interstitial atom defects. Starting from predicted ground-state configurations predicted for different alloy compositions, mono-vacancy formation energies have been calculated. It is shown that in W-Ta alloys, they are very sensitive to the concentrations and depending on local environment of vacancy sites, their values can be changed from 3 to 5 eV whereas in W-V the corresponding change in vacancy formation energies is found to be small. Furthermore, elastic constants calculations were performed for all the ground states as well as for some meta-stable cubic configurations in order to study the composition effect of alloying on mechanical properties in W-Ta and W-V alloys. Using the general expression for Young modulus and Poisson's ratio, we are able to investigate in detail the anisotropic elasticity behaviour for W-V and W-Ta across the all range of composition and therefore to understand if alloying can improve the ductility of tungsten-based alloys.

This work is carried out within the European Fusion Development Agreement (EFDA) materials programme on W&W alloys. Work at CCFE is funded by the RCUK Energy Programme and EURATOM under EFDA.

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P53B

Modeling Helium migration and Helium-vacancy cluster nucleation at grain boundaries in α-Fe

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The role of He on embrittlement of ferritic/martensitic steels is not fully understood. Modeling can provide some insight on the basic processes occurring at grain boundaries in the presence of Helium and defects produced by the irradiation. In this work we present density functional theory (DFT) calculations of interstitial helium and vacancy migration at grain boundaries in bcc Fe. These simulations show that helium migrates slower at the grain boundaries than in the bulk while vacancies migrate faster. Stabilities of different Helium-vacancy clusters at grain boundaries have also been calculated with DFT. These values are used in a kinetic Monte Carlo model to study the nucleation and growth of Helium-vacancy clusters for different temperatures and He/vacancy ratios. Helium-vacancy clusters are considered to be the nuclei for the formation of bubbles and voids. Calculations show that near a grain boundary the concentration of these nucleation sites, He-vacancy clusters, is reduced compared to the bulk. This is in agreement with the experimental observation of void depletion close to grain boundaries. We also identify different mechanisms of He accumulation at grain boundaries depending on temperature: at low temperatures (~100K) no He-vacancy clusters are formed at the grain boundaries, however there is a large supersaturation of He that could contribute to grain boundary decohesion or embrittlment. At high temperatures (~600K) He-vacancy clusters can grow by addition of new vacancies and these clusters could then be responsible for helium embrittlement at this temperature.

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Resistivity recovery of proton irradiated Fe-Cr alloys

P54A

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High chromium ferritic/martensitic steels are important materials for application in fusion devices due to their superior resistance to irradiation, in terms of low swelling and low damage accumulation. The knowledge of fundamental properties of point defects in these materials is important in order to quantitatively describe their behaviour under long term irradiation. For this purpose, the study of simpler model alloys, as Fe-Cr in the present case, can provide valuable information, allowing at the same time an easier interpretation. Although extensive work has been performed on dilute alloys, fewer experimental data are available to date on concentrated Fe-Cr alloys [1,2]. In view of recent computational progress towards the understanding of defect behaviour in this alloy system [3], more detailed experimental results are currently needed.

In this work pure Fe-Cr model alloys with Cr concentrations between 5 and 15 at% have been irradiated at cryogenic temperatures (~ 20K) with protons of energy 5 MeV. Damage production and subsequent recovery of the defects created during the irradiation have been studied by means of electrical resistivity measurements. Resistivity recovery spectra exhibiting several annealing stages have been recorded up to a temperature of 600K. Changes in the recovery spectra with Cr concentration reveal the effect of Cr on the migration of point defects.

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Plasma Impact on Materials at Displacement Damage Condition

P54B

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Changes in plasma-surface interaction effects arising from the fusion neutron irradiation may be anticipated for plasma facing materials suggested for application in fusion devices such as ITER and future reactors. Investigation of radiation damage probable influence on PFMs erosion properties and on tritium in-vessel inventory may be considered actually as an experimental task of primary concern. Such an experimental work is being done at Kurchatov Institute on the basis of ion accelerator and plasma linear simulator LENTA. Two kinds of materials are under study: carbon based materials including CFC and tungsten. High-energy ions accelerated with cyclotron were used to produce displacement damage in the materials (C^+ and He^{++} at 3-5 MeV). The displacement damage induced in the samples covered the range from 1 dpa to 80 dpa. The samples were then subjected to deuterium steady-state plasma exposure. Multiple exposures of each sample have been done to reach deuterium ion fluence of 10^{25}-10^{26} ion/m^2 at 100-250 eV of ion energy. Erosion rate has been measured and erosion yield evaluated. While significant influence of the damage occurring in the surface layer on erosion rate was observed for carbon materials, this was not found in the case of tungsten. Changes in the surface microstructure were analyzed with Scanning Electron Microscopy technique; comparison was made of damaged and non-irradiated materials. Surface modification has been observed on the damaged tungsten with formation of blister-like features, cavities, and column structures. Concentration of gases retained in tungsten after plasma exposure has been analyzed with nuclear reactions methods. Deuterium retention in irradiated tungsten was measured by ERDA. Deuterium concentration in tungsten was determined at different depths of the damaged layer. Increase of deuterium retention was found on the damaged tungsten with the largest integral deuterium uptake measured $1.7\cdot10^{17}$ D/cm^2. The helium pre-implanted by high-energy irradiation has also been detected in the damaged tungsten with maximum of 10% at.

The work was supported by RFBR projects Nr.08-08-12151 and Nr.10-08-01170-a.

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Spatially-dependent cluster dynamics modeling of microstructure evolution in low energy helium irradiated tungsten

P55A

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In fusion reactors, plasma facing materials will be irradiated with high ion fluxes. One key plasma facing component particularly, the divertor, will face high fluxes of low energy (~100 eV) helium and hydrogen. Tungsten is being considered as one of the leading candidates as a material for the divertor. However, the behaviour of tungsten under high dose irradiation with coupled helium/hydrogen exposure remains to be determined.

The irradiation behavior and in particular, performance changes are intimately related to microstructural changes, such as the formation of point defect clusters, helium and hydrogen bubbles or dislocation loops. Computational materials modelling will investigate the mechanisms controlling microstructural evolution in tungsten following high dose, high temperature radiation exposure.

The aim of this study is to understand and predict primary defect production and defect diffusion, clustering and interaction close to the inner surface of the divertor due to low energy helium irradiation. These defects can be interstitial clusters, vacancy clusters, helium interstitials and helium-vacancy clusters.

We report results from a spatially-dependent cluster dynamics model based on reaction-diffusion rate theory to describe the evolution in space and time of all of these defects. The inputs to the model (Diffusion coefficients, migration and binding energies, initial defect production) are key parameters and are determined from a combination of atomistic materials modelling and available experimental data.

We then compare our modelling predictions with experiments available in the literature, which are mainly thermal desorption spectrometry experiments.

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TEM characterization of self-ion damaged polycrystalline W and W-alloys

P55B

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Transmission electron microscopy has been used to study self-ion irradiation damage in polycrystalline W and W-based alloys, which are possible candidate materials for plasma facing components in a TOKAMAK. Polycrystalline W (99.99wt%, 99.95wt%), W-5Re and W-5Ta were irradiated with 2MeV W^+ ions at nominally 500°C to a dose of 1.05×10^{18} W^+ m^{-2}, aimed at simulating and thus exploring the displacement damage effect of fusion neutrons. Features of irradiation-induced dislocation loops were extracted from series of diffraction contrast experiments, including loop Burgers vectors, loop nature (interstitial or vacancy), and size distributions and number densities. The influence of grain orientation, material purity and alloying elements are demonstrated and the underlying mechanisms are discussed.

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Thermal Property of Neutron Irradiated Tungsten and Its Alloys

P56A

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Tungsten (W) is one of the candidate materials for plasma facing components of fusion reactor because of its high melting point, high resistance to sputtering, and high thermal conductivity. During fusion reactor operation, plasma facing components are exposed to high flux 14 MeV neutron irradiation, therefore, it is important to study the effect of neutron irradiation on thermal properties of W. The objective of this study is to investigate the thermal conductivity change due to neutron irradiation in W and its alloys.

The addition of rhenium (Re) in W not only improves the low temperature ductility of W, but also increases its high temperature creep strength [1], and it is one of the major transmutation elements of W. Arc melted W and W-Re alloys were examined in this work. The neutron irradiation experiments were performed in JOYO (fast reactor) and JMTR (mixed spectrum reactor) at JAEA (Japan Atomic Energy Agency), and HFIR (mixed spectrum reactor) at ORNL (Oak Ridge National Laboratory). Irradiation fluence and temperature ranges were 1-12 x 10^{25} n/m^2 and 400-800ºC, respectively.

Thermal conductivity was evaluated by two methods. One was the thermal conductivity converted from the thermal diffusivity by laser flash analysis. The other was the thermal conductivity converted from the electrical resistivity by four-probe method.

Thermal conductivity of W and W-Re alloys decreased with neutron irradiation. In the case of 750^{o}C/1.54dpa irradiation in JOYO, thermal conductivity of W was decreased to 80% of unirradiated W. In the case of W-Re alloys, thermal conductivity was decreased with Re content, but reduction by neutron irradiation was smaller than that of W. In the case of unirradiated W-5Re, thermal conductivity was decreased to 50% of unirradiated W. In HFIR irradiated specimen, reduction of thermal conductivity was larger than that of specimens irradiated in JOYO and JMTR.

The effects of irradiation damage level and Re content to thermal conductivity of W and W-Re alloys will be discussed. The mechanism of thermal conductivity changes will be also discussed considering the microstructure changes and composition change due to transmutation.

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The Dependence of Thermal Diffusivity on Measurement Temperature Affected by Irradiation Damage in Ceramic Materials

P56B

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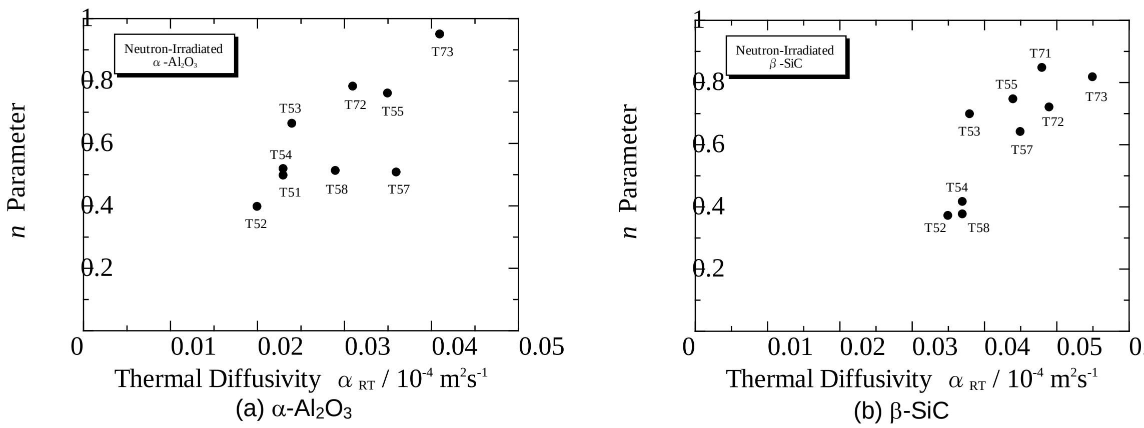
Thermal conductivity and also thermal diffusivity is one of the most important parameters for ceramic materials used in fusion reactors. Neutron irradiation induces many defects in crystals, and they decreases thermal diffusivity severely. The decrement was almost saturated with neutron dose around 30 dpa, but changed with the irradiation temperature. On the other hand, thermal diffusivity of ceramics decreases by phonon scattering as the temperature rises even in the perfect crystal. Typical structural ceramics, α-Al_2O_3, AlN, β-Si_3N_4 and β-SiC specimens were irradiated in the experimental fast reactor JOYO to 4-73 dpa at 650-1050 K. Thermal diffusivity was measured at 123-413 K via the laser flash method. The result was described by an approximation function $\alpha=k/T^n$, where k and n are constants, T [K] is measurement temperature and α [m/s^2] is thermal diffusivity at each temperature. Using the function, the thermal diffusivity at the irradiation temperature was determined for each specimen with several assumptions [1,2].

In addition, the n parameter in the function was changed with the irradiation condition. For unirradiated specimen, this n parameter is almost 1, but it is turned out that it decreased with decline of thermal diffusivity at room temperature that represents the amount of defects (Fig.1). This n parameter decides the thermal diffusivity at the irradiation temperature, so the estimation of this parameter enables more reliable evaluation of thermal diffusivity during the irradiation.



(a) α-Al_2O_3 (b) β-SiC

Fig.1 The n parameter in the approximation function $\alpha=k/T^n$ changed with the thermal diffusivity of that specimen at room temperature.

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Radiation Induced Conductivity for Er_2O_3 coatings on EUROFER under 1.8 MeV electron irradiation

P57A

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The use of some type of electrical insulating ceramic coating will be necessry in order to mitigate MHD (magnetohydrodynamic) pressure drop of liquid Li breeder/coolant in future blanket systems [1]. Previous studies have shown that Er_2O_3 coatings maintain the chemical compatibility with highly corrosive high temperature liquid Li. However it is expected that during operation under the influence of the intense radiation field the electrical behaviour of the insulating coating will be severely modified. In particular, enhancement of the electrical conductivity due to RIC (Radiation Induced Conductivity) [2] and RIED (Radiation Induced Electrical Degradation) [3] must be examined.

Cubic phase Er_2O_3 coating on EUROFER substrate was produced at IPP Garching using the method of cathodic arc deposition. Average thickness of the erbia coating was 800 nm and it was deposited at 700 C. After being produced the sample was delivered to CIEMAT where the sample was irradiated with 1.8 MeV electrons in order to evaluate the electrical behaviour of the sample within the radiation field. Before irradiation the electrical conductivity of the erbia coating was measured in vacuum as a function of temperature from room temperature up to 450 C. Then the sample was irradiated at 450 C and 700 Gy/s and the RIC was continiously measured during irradiation. After being irradiated a clear increase in electrical conductivity was observed together with electrical instability induced by both irradiation and thermal treatment.

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Database Development and Coordinated Research Projects at the IAEA on Plasma-Material Interaction and Neutron-Induced Processes

P57B

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Within the IAEA Division of Physical and Chemical Sciences (NAPC) the Nuclear Data Section is devoted to the development and dissemination of atomic and nuclear data for applications. In connection with materials in fusion energy research our activities are concentrated on plasma-material interaction processes and on fusion-relevant neutron cross-sections. Plasma-material interaction data are included in the ALADDIN numerical database and the AMBDAS bibliographical database. For neutron processes the key databases are ENDF for evaluated data, EXFOR for experimental data and CINDA for bibliographical data. The Section encourages development of new data and new methods by way of Coordinated Research Projects (CRPs). A CRP normally runs for about 4 years and involves 10-15 research groups. The ongoing and planned CRPs in the area of fusion materials are:

- Data for surface composition dynamics relevant to erosion processes. This CRP (2007-2011) is concerned with the behavior of mixed materials, such as C-Be-W, in a fusion vacuum vessel.

- Characterization of size, composition and origins of dust in fusion devices. (2008-2012.) Dust and tritium retention in dust are critical safety concerns for ITER and for a reactor.

- Fusion Evaluated Nuclear Data Library FENDL 3.0. A primary objective of this CRP is to provide an evaluated nuclear data library for fusion applications, extending the present FENDL-2.1 library toward higher energies and including incident charged particles and the evaluation of related uncertainties.

A CRP on “Erosion and tritium retention in beryllium surfaces” is being proposed for the timeframe 2012-2016, overlapping with the operation of the JET ITER-Like Wall and in time to prepare data before the start of operation of ITER. The databases and information about the research projects are on the Nuclear Data Services and Atomic and Molecular Data Unit web pages [1,2] and will be described further in the poster.

[1] Nuclear Data Section web page: http://www-nds.iaea.org/
[2] Atomic and Molecular Data Unit web page: http://www-amdis.iaea.org/

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Erosion of Tungsten and Its Brazed Joints with Bronze Substrates Irradiated by Pulsed Deuterium Plasma Flows

P58A

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Results on erosion of mono- and polycrystalline tungsten and brazed joints of tungsten plates and bronze substrates under irradiation by high-temperature pulsed (τ_p~20 μs) deuterium plasma (HTPP) flows, with the energy density Q=0.4−1.3 MJ/m^2 and the pulse number N=2−30, imitating the expected plasma disruptions in fusion reactors are presented.
The surface erosion character and heat-resistance of tungsten and brazed joints were investigated by SEM, and erosion coefficients were determined by target mass loss.
It is found that for the both types of tungsten the surface starts to significantly crack even under relatively weak irradiation regimes (Q=0.4 MJ/m^2, N=2). At that, surface melting is not observed. Local meltings become visible with an increase of Q up to 0.5 MJ/m^2. In addition, there is formation of blisters with a typical size of 1−2 μm on the surface of monocrystalline samples and craters up to 2 μm in diameter on polycrystalline samples. An increase of Q up to 0.6 MJ/m^2 and higher results in intensive melting of the surface layers and their additional cracking along the boundaries of newly formed 50–150 nm submicrocrystalline grains as a result of melting and subsequent hardening. Besides, craters ~10–30 μm in diameter are formed on defects similar to those observed under unipolar arcs.
The dependence of the erosion coefficients from Q and N is identical for poly- and monocrystalline tungsten. A significant decrease of the erosion coefficients and their saturation with an increase of both parameters are observed. At that, the erosion coefficients change in ranges of 0.2–0.3 μg/(J·cm^2) for mono- and 0.2–0.7 μg/(J·cm^2) for polycrystalline samples.
The heat-resistance of brazed joints of tungsten plates with bronze under irradiation by deuterium plasma flows is found to depend on the type of bronze. It is found that at Q=1.0 MJ/m^2, the brazed joints of monocrystalline tungsten with bronze of Cu-0.6% Cr-0.08% Zr have the highest heat-resistance.

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An Investigation of Non-Destructive Examination (NDE) Methods for the Analysis of Be to Cu alloy Bonded Samples

P58B

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The task of bonding the dissimilar materials within the First Wall Panels (FWP), such that components will survive the extreme environment of a fusion reactor, presents challenges. Bonds must remain intact under high temperatures, exhibiting a resistance to failure through static and cycling thermal loads, whilst also being tolerant to irradiation. Commercialisation of manufacture in preparation ITER brings with it a requirement for guaranteed repeatability and high quality of output. This is important considering the high costs of replacement and potential damage to underlying components, resulting from failure of the FWP components during service.

One way to evaluate the quality of individual bonded components would be through the destructive testing or cutting of samples from completed components. However, sampling and destructive testing is not ideal given the high cost of fabrication for the FWP components and this will only statistically demonstrate the integrity of the components, which leaves the possibility that flawed components could remain un-revealed. Non-destructive examination (NDE) methods are therefore attractive for evaluation of completed components, both prior to use and during service.

The work reported evaluates the effectiveness of several non-destructive examination (NDE) methods when used for inspection of bond between the Beryllium (Be) plasma facing material and the Copper alloy (CuCrZr) heatsink of the FWPs. This is the main area of interest as this bond has been found to present the greatest technological challenge and consequently requires the most attention within the FWP manufacturing process. Failure of this bond in service could result in further damage to the first wall and may preclude further tokamak operation due to excessive plasma contamination with high Z elements.

Small samples of HIP-bonded Be-CuCrZr-SS were produced for this investigation, following the current manufacturing route for the FWP. NDE techniques investigated include ultrasonic, x-ray and electro-resistance methods, with the goal of identifying defects both directly and through the measurement of local bond residual stresses. A detailed characterisation of the bond region using SEM has also been performed. Following tests using these techniques, consideration is given to the practicalities of scaling up to full size components.

This initial work is intended to demonstrate the NDE methods that can be used to inspect the Be-CuCrZr bond. Subsequent work will focus on relating measured quantities to process control and structural integrity.

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Fabrication and Test of KO Preliminary Semi-Prototype for ITER First Wall Qualification

P59A

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The ITER First Wall (FW) includes the beryllium armour tiles joined to a CuCrZr heat sink with stainless steel cooling tubes. This first wall panels are one of the critical components in the ITER machine with the surface heat flux of 5 MW/m^2 or above. So, a qualification program needs to be performed with the goal to qualify the joining technologies required for the ITER First Wall. Based on the results of these tests, the acceptance of the developed joining technologies will be established. The results of this qualification test will affect the final selection of the manufacturers for the ITER First Wall.

The detailed fabrication process of the KO qualification semi-prototypes (SP) will be described as follows. [1-3] For the CuCrZr and stainless steel, the canned materials are placed into the HIP furnace. HIP (Hot Isostatic Pressing) was conducted at 1,050 °C and 100 MPa for 2 hours with the heating rate of 5 °C/min and the furnace cooling. During the heating process, the temperature was held at 900 °C for 210 min for pressure control and the homogenizing of the materials. And, in the case of beryllium to CuCrZr HIPping, the canned materials are placed into a HIP furnace. HIP was conducted at 580 °C and 100 MPa for 2 hours with the heating rate of 4 °C/min and the furnace cooling. The canning plates were removed by electro-discharge machining. The surface of HIPped SP was mechanically machined, and the materials were cleaned in ethyl alcohol by using an ultrasonic cleaner.

For the non-destructive tests (NDT) of the fabricated SP, visual and dimension inspections were performed whenever needed in the fabrication process, ultrasonic test (UT) was performed with ultrasonic probes. Destructive tests (DT) for the qualification SP were performed on a small mockup which was fabricated together with these SP. The small mockup for the destructive test has one small beryllium tile without the cooling pipes. It is assumed that the small mockup has the same properties as SP since the test mockup and SP are fabricated simultaneously with the same manufacturing process by using the same facilities.

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Micro-chemical analysis of highly heat loaded CFC/Cu interfaces from Tore Supra and Wendelstein 7-X

P59B

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A comparison of high heat flux (HHF) test results of Tore Supra and Wendelstein 7-X plasma-facing components was performed to study the fatigue behaviour of the CFC/Cu bonding. Both types of actively water-cooled elements are made of CFC as plasma-facing material bonded by active metal casting onto a CuCrZr structure. The manufacturer Plansee improved the applied bonding technique in the frame of pre-series development for the W7-X divertor targets. [1]

Samples with three different CFC materials were compared. One sample was cut from a Tore Supra limiter element fabricated with N11-92 CFC material which fully meets the technical specification. Another sample was taken from a Tore Supra element covered with N11-98. This CFC material slightly deviates from the requested properties [2]. The third investigated sample was cut from a W7-X prototype element covered with NB 31.

All three components were cyclically loaded with up to 3000 pulses (10 s each) at 10 MW/m² at the HHF test facility GLADIS. The components showed stable performance without significant surface temperature increase, except for the elements covered with N11-98. Suddenly occurring defects of the N11-98 bonding resulted in a collapse of bonding within the following 100 – 200 pulses.

A reliable bonding of CFC onto Cu structures requires the formation of carbides to wet the carbon surfaces by liquid copper. To check whether the bonding itself could be made responsible for the failure. Micro-chemical investigations were performed on metallographically prepared cross sections of the respective joints. XPS was applied to determine the carbidic and metallic chemical states of titanium. SIMS investigations visualize the spatial distribution of the alloying elements.

The paper discusses the micro-chemical investigations with respect to the results of the heat load test. The Ti 2*p* peak position in XPS measurements reveals that titanium is present mainly in carbidic form in all three analysed samples. The lateral elemental distribution visualised by SIMS 2d images differs reasonably between the older Tore Supra samples and the improved bonding W7-X sample, where a thick titanium interlayer is found all along the interface. In between the two Tore Supra samples no qualitative difference can be found. Only little titanium is detected in higher concentration at the interface itself but on both samples titanium is found also in a distance to the interface diffused inside CFC pores.

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Image based modelling of silicon carbide composites (SiC_f/SiC) for use in a fusion power plant's breeding blanket

P60A

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Silicon carbide has been identified as a candidate material for use in fusion reactors. The intrinsic mechanical behaviour of monolithic forms of SiC is poor, hence it is primarily considered as a constituent in composites. SiC composites are being evaluated because of their high operating temperatures and good activation properties. The purpose of this work was to develop a mechanistic understanding of the behaviour of SiC_f/SiC composites under thermal and mechanical loading.

The microstructures of a tube made of 3D woven Hi nicalon fibres with CVD SiC matrix were characterised in 3D by X-ray tomography. Imaging was performed using the Metris 320kV custom bay at the Henry Moseley Imaging Facility at the University of Manchester, UK. Meshes for finite element models were generated in the Simpleware software package directly from these data. Input mechanical and thermal data for the models were determined by; nanoindentation and appropriate interfacial tests, laser flash, differential scanning calorimetry and thermal gravimetric analysis, respectively. The quasi-static and thermal behaviour of the composites were modelled in ABAQUS and ParaFEM and validated by experiment.

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Effects of Heat Treatment Conditions on the Micro-Structure and Impact Properties of Eurofer 97 ODS Alloy

P60B

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High-chromium oxide dispersion strengthened ferritic-martensitic steels (RAFM) are potential candidate materials for first wall application in the next nuclear fusion device to be built, ITER (International Thermonuclear Experimental Reactor). The reduced activation material produced in this study is the ODS variant of the Eurofer 97 alloy, containing a dispersion of yttrium oxide (0.3 wt.%). To date, one of the most pressing challenges is the requirement of a material capable of withstanding high levels of neutron irradiation whilst maintaining sufficient mechanical properties. The present study focuses on the ductile to brittle transition temperature (DBTT), which should be low enough to counteract the expected shift to higher temperatures caused by neutron irradiation embrittlement. In order to evaluate the effects of heat treatment conditions on micro-structure and DBTT properties, samples of Eurofer 97 ODS, produced were produced by means of mechanical alloying in hydrogen atmosphere and hot isostatic pressing (HIP). The samples were then normalised and tempered. A range of twelve heat treatments, with normalising temperatures between 980°C and 1300°C were investigated. These were chosen to identify any influence of heat treatment upon the grain size, carbide distribution and composition, and grain boundary character. This information was then used to indicate which heat treatment parameters yielded the material with the most favourable impact properties, and how the microstructure was affected. The characterisation of the various microstructures produced was peformed by means of scanning electron microscopy, electron back-scattered diffraction microscopy and transmission electron microscopy. The Charpy V-notch samples were tested using an impact testing machine modified for use with sub-size specimens, with an atmospherically controlled chamber and instrumented displacement sensor. Results showed that independently from the heat treatment conditions, all the microstructures shared the same features: fine grained ferritic matrix (sub-micron); fine and uniform carbide population decorating grain boundaries and, to a lesser extent, within the grains; fine and uniform yttria dispersion; randomly distributed regions of larger ferritic grains. The grain boundary character distribution showed to be .unchanged except at the highest normalising temperature and the slowest cooling rate. This heat treatment led to a microstructure containing a high fraction of low angle boundaries, which displayed very encouraging DBTT behaviour: a high USE and a DBTT downward shift compared to the as-received material of 150°C.

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Analytical TEM investigation of boron alloyed EUROFER 97

P61A

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Martensitic tempered steels are established as high temperature material for the use in turbines or power plants. The 8-10%Cr-steels are one development for the future application as component in a Fusion reactor. For the thermal stability of the material and precipitates, it is necessary to alloy the steel with a small quantity of boron. The European reduced activation ferritic-martensitic (RAFM) reference material EUROFER 97 [1] was alloyed with ^{10}B and natural boron to study the influence of He on the mechanical properties of irradiated material [2]. Helium will be generated by the nuclear reaction $^{10}B(n, \alpha)7Li$. The boron forms several precipitates of 200-500nm size. The application of modern analytical methods shows that the precipitates are in most cases complex. Detailed analysis was possible using combined electron energy loss spectroscopy (EELS) and Energy dispersive X-ray analysis (EDX) spectroscopic imaging. EELS spectroscopy conducted in the transmission electron microscope (TEM) represents a powerful tool not only for the detection of light elements such as boron or nitrogen but also for determination of chemical phases and even its orientation to the electron beam. The possibility of direct measurement of chemical bonding effects at the sub nanometer level has aroused considerable interest in the materials science community.

The elemental mapping shows the distribution of different precipitates in the specimen. Altogether four types of precipitates have been detected: $M_{23}C_6$ precipitates which are visible in the Cr map, the TaC and VN particles which are visible in the Ta and V maps respectively. Large precipitates of 300nm-500nm size have been identified as boron nitrides.

The chemical analysis of these precipitates has been performed using EDX and EELS. EDX shows the chemical ratio of 3d metals and “heavy” elements such as Cr, Ta or V inside precipitates. EELS has been used for detection of light elements such as N, C and O.

Spatially resolved EELS and EDX methods were used for imaging and detailed elemental analysis of different precipitates in the boron alloyed EUROFER 97.

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Three-dimensional imaging and metrology of yttria dispersoids by electron tomography

P61B

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Microstructure of the Oxide Dispersion Strengthened (ODS) alloys for high temperature application [1-2] as well as the ODS steels for fusion were studied mainly by scanning- and transmission electron microscopy (SEM, TEM). The precise qualification of the shape, size and chemical composition of dispersoids (mostly Y-Al oxide particles), which are the strengthening particles of these materials, is of great importance for determination their influence on material properties.
Electron tomography allows for generation of 3D model (image) of the investigated object(s) from the multiple 2D projection images, obtained over a range of viewing directions. Promising results were achieved using electron tomography with EFTEM or HAADF-STEM imaging [3].
We have performed electron tomography studies of the Y-Al dispersoids in ferritic ODS alloy, Incoloy MA956, after isothermal annealing at 1350 °C up to 1000 hours. The samples of dispersoids were prepared as extracted double-replicas. Tilt series were acquired semi- automatically at FEI Tecnai G2 microscope using FEI 3D software. Oxide Al/Y ratio was determined by EDS.
The results show precisely three-dimensional morphology of the oxide dispersoids, their chemical composition (namely Al/Y ratio) and their changes during isothermal annealing at 1350 °C. Several different types of Y-Al particles were observed and a dependency between their chemical composition and their shape was established.

Therefore another tomography technique, FIB tomography, was used for detailed characterisation and metrology of Y-Al dispersoids in the MA956 alloy.
FIB (Focused Ion Beam) tomography is based on a serial sectioning procedure employing a FIB/SEM dual beam workstation. Repeated removal of layers as thin as a few nm allows to explore at total a volume of some μm^3. 3-D mapping of dispersoids by serial FIB slicing and SEM imaging was performed using a 30 kV Ga$^+$ ion beam to get a precise *in-situ* milling. The 3-D reconstruction and visualization led to observe several types of dispersoids present in the matrix and allowed for precise measurement of their size. The results achieved confirm the ability of FIB tomography to get 3D reconstruction of the objects of 100 nm or even smaller. Such 3D reconstructions can serve as a basis for quantitative analyses of complex structures in materials down to the nano-scale.

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Characterization of Y_2O_3 and La_2O_3 nanoparticles in W-V and W-Ti ODS tungsten alloys by small angle neutron scattering

P62A

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W alloys and oxide dispersion strengthened (ODS) W alloys are being considered as potential structural materials for the modular helium-cooled divertor of the fusion reactor DEMO [1,2]. Among the properties that make W a suitable material for plasma facing material are its high melting point, good thermal conductivity, low sputtering rate and low tritium retention. However, there are two important deficiencies that could limit its use in the future fusion reactors, i.e. its high ductile-brittle transition temperature (DBTT), which is in the temperature range 373-673 K, and its recrystallization temperature (RCT) around 1500 K [3]. The DBTT and RCT of W can be improved by addition of some impurities. For instance, the addition of Re lowers DBTT of W and enhances its mechanical characteristics at high temperatures [4-6]. However, the W-Re alloys undergo severe embrittlement under neutron irradiation. On the other hand, it appears that the the DBTT and RCT for W can be modified via a uniform dispersion of oxide nanoparticles as such ThO_2, La_2O_3 or Y_2O_3. It has reported that the addition of Ti or V by itself, or combined with either Y_2O_3 or La_2O_3 can moderately enhance the strength and fracture toughness of W in the temperature range below ~ 873 K or 1073 K, as well as its microhardness noticeably, in comparison to pure W produced by the same procedure [7, Ref de los workshops].

W and W-V and W-Ti alloys containing a dispersion of stable oxide, La_2O_3 or Y_2O_3, have been prepared by mechanical alloying and subsequent consolidation by hot isostatic pressing. The mechanical behaviour of these alloys at high temperature appears to be related to the presence of a fine dispersion of oxide particle. In this work, small angle neutron scattering (SANS) measurements performed on these alloys will be presented. The analysis of the data has let us extract information about the morphology and size distribution of the La_2O_3 and Y_2O_3 nanoparticles present in the ODS W alloys.

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Microstructure analysis of tungsten materials produced by different fabrication routes

P62B

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Present design concepts for nuclear fusion reactors include high heat flux components which have to be operated at extreme physical conditions. Due to a complex mix of requirements, the most promising materials for shield as well as for structural applications are tungsten alloys. Their use as structural material is limited by the onset of recrystallization and/or loss of strength at high temperatures – typically above 1200°C. Since mechanical properties are defined by the underlying microstructure, refractory alloys can behave quite different, even if their chemical composition is the same [1,2].

Therefore, a screening study of the microstructure characteristics was performed with various commercial tungsten half products. Grain size, orientation, anisotropy, particle form and distribution were investigated using transmission electron microscope (TEM), and a focused ion beam (FIB) system.

With the help of slice and view techniques in the FIB [3], it was possible to generate 3D models of oxide dispersion particles, which were visualized by surface rendering revealing needle or platelet shaped particles depending on the processing conditions: In one refractory alloy lanthanum the oxide particles have nearly a spicular structure and 3D visualization of 3D reconstruction clearly indicates a considerable fraction of the aligned particles is longer than 20µm. In contrast, a differently processed lanthanum oxide tungsten refractory material investigated earlier exhibits sheet-like lanthanum oxide particles. These results indicate the importance to correlate structure and processing conditions.

Analytical TEM investigations were performed to study the presence of Lanthanum on the grain boundaries and for imaging of lanthanum oxide in tungsten microstructure, for example by TEM pictures and 2D EDX mappings.

Based on these investigations dislocations and defects in structure were documented.

This presentation describes and discusses the correlation between the microstructure and the specific fracture modes of common tungsten materials.

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TEM investigation of W in the as-received and forged condition

P63A

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Tungsten is considered as a primary candidate to be used as a first wall material in future tokamaks such as DEMO and beyond. The optimal production parameters and the full characterisation (microstructure, thermo-mechanical properties) of the material is still under investigation. In particular, the thermal shock behaviour and the cracking resistance need to be characterised. For this investigation, pure tungsten blocks were produced and provided by Plansee AG, Austria. The main steps in the production process are isostatic pressing followed by sintering to obtain a cylindrical block. This block is forged first in the radial direction and then along a cylindrical direction. The last step is a stress relief treatment at 1000°C to remove residual stress. Besides the purely stress relieved material also a recrystallized material was characterized simulating a microstructural change of the material due to long term operational conditions in a fusion reactor. The recrystallization treatment was performed at 1600°C for 1 hour by Plansee AG.

The aim of this production method was to obtain a tungsten material with isotropic grains and no porosity. However, the metallography performed on this material showed that the grain structure is not as homogeneous as expected. In the as-received condition, larger grains were observed near the edge of the cylindrical block compared to the center and also differences between the top of the block and the center were stated. Moreover, almost no porosity was present at the edge of the block, but 1-2% porosity consisting of pores up to 5µm were found in the center. The recrystallization treatment resulted in a grain growth, mainly at the edge of the block.

In this presentation, the influence of the production method and recrystallization on the microstructure of the material, analyzed with transmission electron microscopy (TEM), will be discussed. In total 8 specimens, 4 from the as-received and 4 from the recrystallized material, were taken from the edge and center of the block, both at the top and in the bulk. It was found that microstructure did not vary significantly with the position of the specimen in the block, but that the recrystallization does have a big influence. In all specimens from the as-received material, a large amount of small angle tilt boundaries were observed. These boundaries were removed during the recrystallization treatment and for these specimens only a limited amount of dislocations were observed.

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Subsurface structures on rolled and re-crystallised W after D bombardment

P63B

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Tungsten is a promising candidate for a plasma facing material in the main chamber and also for divertor areas in fusion reactors. The erosion of this high-Z material by plasma is acceptably low and the D retention is intensively investigated by many international groups. The absolute D retention in W seems to be tolerable for the next generation of nuclear fusion devices like ITER or DEMO. But still clarification is needed how and under which conditions D enters the material and where D is mainly accumulated.

In this work two types of W specimens were investigated: i) rolled tungsten with a mirror-finish polished surface (ITER grade Japan, grain sizes 2-100 µm) and ii) the same W grade but after additional re-crystallisation at 2070 K for 2 h under H atmosphere (~50 µm grain size). The W samples were exposed in a linear plasma generator at elevated temperatures (320 – 700 K) by D bombardment with an energy of 38 eV/D and with fluences up to 10^{27} D/m² at a flux of ~10^{22} D/m²s. The samples were analysed with SEM combined with FIB for iterative cross-sectioning to obtain a 3D reconstruction. Additionally, EBSD was applied to determine the grain orientation and deformation.

The D transport is driven by diffusion under super-saturation conditions and depends strongly on temperature. The re-crystallised samples show at low temperatures bizarre surface extrusions and the subsurface region is cracked and plastically deformed (up to depth of 10 µm). At temperatures around 600 K stepped extrusions appear, the grains remain undamaged, but large cavities are created at the first grain boundary up to a depth of 50 µm with the same volume as the extrusion. The extruded material moves along the <111>{110} slip system, which is activated above the DBT temperature [1]. The positions of cracks and pores are determined by 3D reconstruction of cross-section slices with the Amira software. Their positions and elongations are compared with the lattice orientation and the grain cleavage planes. At temperatures above 700 K no extrusions are visible.
In contrast, the observed extrusions on the rolled tungsten are blisters (typical 2 µm diameter) and nearly similar over the whole temperature range, even above 750 K. They relax partly by a FIB cut made several µm away. Obviously, networks of channels exist between blisters which are gas filled and partly elastic [2].

Further first experiments were performed on a D loaded sample to investigate an expected accumulation of D in grain boundaries on the surface by Nano-SIMS. It is also tried to determine the lateral distribution of D on cross sections.

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P64A

Elasticity and Screw Dislocations in W-Re and W-Ta Alloys

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Using a first-principles approach, the elastic properties as well as the core structure of the 1/2<111> screw dislocation in W-Me (Me=Ta, Re) alloys are investigated from the atomistic point of view. For a range of Ta/Re concentrations the lattice parameter, bulk modulus, and elastic constants are calculated and compared with pure W to study the influence of solute atoms on the elastic properties. A periodic quadrupolar arrangement of the dislocation is employed to model the core structures. We show that W and W-Ta alloys at all concentrations exhibit a symmetric core structure. In contrast, W-Re alloys exhibit a gradual transition to asymmetric cores. Furthermore, the critical stress which has to be applied to move the dislocation at 0K (Peierls stress σ_p) is calculated to determine the mobility of dislocations. The reduction of σ_p and a change of slip plane explain the brittle to ductile transition in W upon Re alloying. However, for W-Ta alloys the reduction of σ_p is found only with high Ta concentrations. Finally, we investigate the correlation between the core symmetry and the γ-surfaces for both W-Me cases.

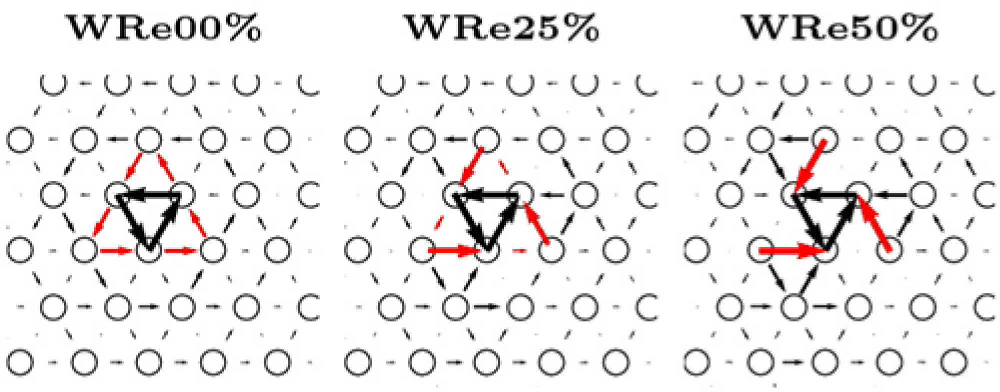

Fig.1 Core structures of 1/2<111> screw dislocation for WRe alloys

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Visco-elastic model of the “fuzz” growth

P64B

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Recent experiments on the irradiation of Tungsten with helium-hydrogen plasma [1-3] have shown the formation of “fuzz”, filled with nano-bubbles, on the front surface of the sample. The “fuzz” growth was observed, within some temperature range of the sample, where the thickness of the “fuzz” was increasing as a square root of the time of the irradiation. The rate of “fuzz” growth depends on the temperature of the sample as well as on the rate of helium ion flux. However, at relatively large helium fluxes the rate of “fuzz” growth saturates.

We present theoretical model [4] describing all main features observed in experiments. This model is based on plastic deformation of the “skin” of the “fuzz” fibers caused by newly growing nano-bubble on the tip of the fiber. The main idea of the model can be described as follows. Newly glowing bubble having high helium pressure inside creates an excessive force on surrounding tungsten “skin” of the fiber and forming pressure difference between base and nose of the fiber. As a result, tungsten “flows” through the “skin” from the base to the nose.

This model predicts $t^{1/2}$ growth of length of the fibers, strong temperature dependence of the growth rate, and the saturation of the growth with ion helium flux to the substrate.

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Characterization of powder metallurgy processed and hot rolled W-TiC alloys

P65A

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Tungsten seems to be one of the most promise candidate plasma facing materials for the divertor and the first wall in fusion devices because of the low sputtering yield, high thermal conductivity, high strength at elevated temperatures and low tritium inventory [1]. Unfortunately, these advantages are coupled with serious embrttilement in several regimes, i.e., low temperature embrittlement and recrystallization embrittlement which are strongly dependent on the chemical composition and the fabrication method of tungsten based materials.

In order to improve both the low toughness and the resistance to embrittlement by recrystallization of tungsten, W-TiC alloys have been fabricated by powder metallurgy and plastic deformation. W-TiC composite powders were prepared by wet-planetary ball milling, and subsequently the milled powders were consolidated by hot isostatic pressing (HIP). Some of the as-HIPped samples were hot rolled at the temperature 1600°C. The effect of the fabrication method on the microstructure and mechanical properties of W-TiC alloys was investigated. The low-temperature toughness was measured using a single-edge-notched specimen under a three-point bending test. The recrystallization behaviour investigated by annealed at temperatures from 1300°C to 2000°C. The etched sample surface and fracture surface after bending testing are investigated by Optical Microscopy (OM), Scanning Electron Microscopy (SEM) and Transmission Electron Microscopy (TEM).

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Application of transmission electron microscopy and FIB tomography to microstructure characterisation of tungsten based materials

P65B

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Tungsten and its alloys are considered as probable candidate for plasma facing material due to high thermal conductivity, high melting point, good mechanical properties and small coefficient of thermal expansion. The requirement of divertor design is the development of tungsten alloys having ductile–brittle transition temperature below 600 °C and recrystallization temperature around 1300 °C [1]. One of new material under consideration is W-1.7%TiC. Addition of carbides stabilizes the tungsten microstructure at elevated temperature and should contribute to higher tungsten strength and increases both temperatures [3].
The aim of the present work is detailed microstructural characterization of the two investigated materials pure W and W-1.7%TiC alloy, particularly determination of phases as well as particles' size and shape. The microstructural analyses were performed by light microscopy (LM), analytical scanning- and transmission electron microscopy (SEM, TEM).
Very important problem connected with TEM investigation of these materials was extremely difficult preparation of the specimens, due to the high atomic mass of W the electron transparency is very low and requires very thin specimens. After many trials by various techniques, a new method using special cutting of the sample and its electrolytic polishing in 10% NaOH solution was successfully elaborated. The TEM investigation of pure tungsten shows that grains with a low dislocation density, various size and shape were characteristic features of its microstructure.
For characterization of W-1.7%TiC microstructure, FIB tomography and TEM investigations were used. FIB tomography based on a serial sectioning procedure employing a FIB/SEM dual beam workstation. The results show that the TiC particles have mostly globular shape and their size can be estimated as about 0.1-0.2µm. Particles are fairly uniformly distributed in the matrix. Additionally the microstructure shows a high porosity with irregularly shaped pores of various sizes. Further investigations and precise particle measurements are in progress.

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The study was performed under co-operation between AGH University of Science and Technology and Forschungszentrum Jülich within the bilateral project on fusion energy materials. The financial support of the Ministry of Science and Higher Education (project No. 28.28.110.7018) is kindly acknowledged. The authors would like to thank K.Plonska- Niznik and A. Gruszczynski, MSc for assistance in sample preparation and FIB tomography.

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Thermal and Mechanical Characterizations of W-armoured Plasma Facing Components after Thermal Fatigue

P66A

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ITER is a future fusion device, aims at demonstrating the scientific and technical feasibility of fusion power. In the domain of the ITER Plasma Facing Component (PFC) design, monoblock geometry has been chosen to support high heat flux loads on the vertical targets of the divertor component.

The current ITER Baseline foresees the use of carbon in the strike-point region of divertor for the initial non-active phase of operation with hydrogen and helium plasmas followed by an exchange to a full tungsten divertor for the nuclear phase with deuterium and deuterium-tritium (D-T) plasmas. Hence, the tungsten (W) is increasingly considered as a prime candidate armour facing the plasma for ITER and a better knowledge of effects due to cyclic thermal fatigue and high temperature usage on its long term is required.

Recently, high heat flux tests were performed on W monoblock-type mock-ups to evaluate their performances in terms of thermal fatigue lifetime under the conditions expected in the divertor strike-point region of ITER [1]. Main results show that 10 MW/m² is currently the limit before appearance of surface alterations. Beyond, an embrittlement of W armour near the loaded surfaces occurs (roughening aspect, cracks) and later melted W droplets apparition due to high temperature deformation and cyclic fatigue. Current numerical simulation tools are not able to predict correctly these experimental results. A better characterization of structural damage of W armour material, due to fatigue solicitations, is required for a better understanding and prediction of fatigue lifetime related to W-armoured components.

In order to provide first results on the fatigue effect, thermal and mechanical characterizations were performed on a batch of several tens of actively cooled components with W armour after an intensive thermal fatigue campaign carried out in the FE-200 electron beam facility [2]. Several scenarios and events (e.g. located overheating, melting surface, failure, large cycling) happened during the high heat flux test campaign enabling to define different categories of W samples demarcated by specific threshold temperatures (typically, annealing, recrystallization and melting temperatures). Based on these relevant categories, thermal diffusivity assessments by flash lamp method and Vickers Hardness measurements were performed on these different samples to investigate the effects of the thermal fatigue on each category.

In this paper the main results of measurements performed on these different samples are reported. In addition, the systematic grain size assessment, carried out to correlate the thermal conductivity to the structural modification is also discussed.

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Numerical Simulation of Tungsten Melt Layer Erosion caused by JxB force at TEXTOR

P66B

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Tungsten is foreseen as one of the armour materials for plasma facing components (PFCs) in the ITER divertor and dome and as the main material of DEMO. During the transients expected in tokamaks (disruptions, ELMs, and VDE) the armour will be exposed to hot plasma streams and localized impacts of runaway electrons (RE). The heat fluxes are expected to be so high that they can cause severe erosion of PFCs thereby limiting their lifetime. During the intense transients the melting, melt motion, melt splashing and surface evaporation are seen as the main mechanisms of metallic armour erosion. In case of RE impact and long time transients (VDE) a melt layer can exist up to several seconds [1]. Experiments at TEXTOR [2] with long time plasma action at the target surface in the strong magnetic field demonstrated that the JxB force generated by thermo-emission electrons dominates in the acceleration of the melt layer and leads to a high target erosion (up to 1 mm per event).

The expected erosion of ITER PFCs under short time transients has been properly estimated using the code MEMOS validated against plasma gun target erosion experiments in cases of short time transients, in which the JxB force is practically negligible. To simulate large time scale melt motion TEXTOR experiments the code was significantly updated, in particular acquiring some additional 3D features. Several models of space-charge limited thermo-emission were implemented.

In this work new MEMOS simulations for the TEXTOR experiments on tungsten targets damage under long time plasma heat loads up to several seconds with heat fluxes up to 40 MW/m^2 in a strong magnetic field are performed, with taking into account 3D geometrical peculiarities of the experiments. The melt layer damage is calculated for multiple plasma loads up to 3 shots. Main attention is focused on investigation of melt layer erosion caused by the JxB force generated by thermo-emission electrons. Different models of space-charge limited thermo-emission current are considered. Numerical simulations carried out for the heat loads in the range 15 – 30 MW/m^2 on the timescale of 5 s demonstrated reasonable agreement with TEXTOR experimental data on tungsten target erosion.

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Annealing behavior and transient high heat loading performance of fine grained W fabricated by fast resistance sintering technology

P67A

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Tungsten samples with fine grain size have been fabricated by resistance sintering under ultra-high pressure. The relative density of the fabricated pure tungsten can reach ~ 98.5 % when using 10 μm size tungsten powders, and ~ 93.5 % when using 200 nm tungsten powders as the starting materials. The grain size is retained nearly the same to the starting powder size. Annealing tests show that although the recrystallization temperature of pure fine grained tungsten is not very high, between 1150 to 1300 °C. The increase of grain size is less than 20 μm after annealing at 1750 °C for 2 h. For evaluation the transient thermal loading response of these fine grained tungsten, a high-intensity pulsed ion beam (HIPIB) apparatus was used for providing the high heat flux onto the surface of tungsten at a heat flux parameter of up to 160 MW m^{-2} $s^{1/2}$. The surface morphologies of fine grained tungsten samples after the HIPIB exposure were investigated using scanning electron microscope (SEM).

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Understanding of crack formation in a double forged tungsten grade induced by thermal shock loading

P67B

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Tungsten and tungsten alloys are promising metals as protective materials for the armour in future fusion reactors. These metals exhibit the highest melting point, superior thermo-mechanical properties, low erosion, acceptable tritium retention and moderate neutron activation properties. The main drawback is their intrinsic brittleness at room temperature and their low recrystallization temperature.

During thermal shock events in ITER, tungsten materials will exhibit various crack formations and failure mechanisms. The extensive heat loads on the surface of the material will create high thermal stresses, huge temperature rises and therefore large strain rates in the subsurface layers.

The microstructure of the double forged tungsten grade was examined by SEM, TEM and metallography [1] in both the radial and longitudinal orientation to determine the grain size, porosity level and grain tilt boundaries.

To better understand the crack formation and propagation at various temperatures, very high mechanical tensile tests were performed (up to 2000°C). Because the brittleness (DBTT: ductile to brittle transition temperature) of the tungsten material also depends on the strain rate, a yield strength model based on a thermally activated slip process was used to describe the flow properties combining both temperature and strain rate effects. In addition the influence of the grain orientation and recrystallization on the ductility is addressed. This is particularly important for long term steady state operation where recrystallization of the tungsten armour may occur.

A strong decrease in the flow properties after recrystallization was found, but this was compensated by an increase in ductility at least for temperatures above the DBTT up to 1000°C. An additional effect of creep was found already at 1500°C and a large degradation for the as-received material was detected at 2000°C which is correlated to the recrystallization. Depending on the strain rate, the crack formation changes as a function of the temperature.

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Effect of Off-Normal Events on Reactor First Wall

P68A

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In the paper we analyse the energy deposition and the consequent erosion of different functional materials like tungsten armour and possible tungsten alloy composites for the first wall due to the various off-normal events impact. Runaway electrons and vertical displacements events (VDE) represent a potential threat to the integrity and availability for fusion reactor. Originated during the natural disruption or in the pre-emptive disruption phase caused by the massive gas injection they eventually release their kinetic and magnetic energy onto the plasma facing component structures. The simulations were performed for 10-100 MeV of electron energy with energy deposition to the wall of 10 MJm^{-2} and incident angles in the range of about 1^o -20^o. The magnetic energy carried by the RE beam will be a substantial part of energy for DEMO, and eventually it will be also deposited to the wall due to the radiation and dissipation of eddy currents. The stopping power and the attenuation time of runaway electrons in materials targets are calculated by means of the ENDEP Monte Carlo code, which takes into account the production of secondary electrons and polarization of solid media (density effect). The energy deposition of RE into FW materials, the level of erosion, caused by RE impact and energy deposition profiles where calculated for DEMO relevant conditions by using the code MEMOS [1]. Calculations show that .both VDE and RE energy deposition on the first wall depends on a number of parameters, including the plasma parameters and assumed deposition area. For the VDE case, under the assumption of 1 GJ of plasma energy deposited over a ~ 0.5 m toroidally continuous band, the resulting energy density on the FW is ~ 60 MJ/m2 including toroidal and poloidal peaking factors. For the RE case, the kinetic energy in the runaways is about 50 MJ for a 15 MA runaway current and 10 MeV average runaway energy. However, in the case of slow VDE's (~0.3 s), most of the remaining plasma magnetic energy can be converted to runaway kinetic energy, resulting in up to 300 MJ of deposited energy on the FW. The exact energy density will depend on the number of modules seeing the total energy.

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High Heat Load Testing of W-Y_2O_3 Armour Materials Fabricated by Chemical Method

P68B

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Tungsten based materials are considered as promising candidates for plasma facing materials in a future fusion reactor. Since tungsten is not yet mature enough, particularly for the improvement of the material's ductility and strength, nanostructured tungsten powders dispersed with yttrium oxide have been fabricated through a novel processing route. Therein, a chemical bottom-up process is followed by thermal processing in a controlled reducing environment. The obtained high purity nanopowders were uniformly dispersed with Y_2O_3 and were sintered into oxide dispersed strengthened (ODS)-W composites using spark plasma sintering technique (SPS) at relatively low temperatures in order to limit the grain growth. Microstructural characterization revealed uniform dispersion of the ultrafine oxide particles at grain boundaries as well as nano-sized particles inside tungsten grains.

For the evaluation of the developed material for plasma facing armour application in the fusion reactor, high heat load tests have been performed in an electron beam test facility. Surface effects, i.e. roughening and crack formation in dependence of temperature and power density, were determined for an applied number of 100 ELM like loads with a pulse duration of 1 ms. The crack paths at the surface and particularly versus the bulk material were investigated to determine the resistance of the material to the formation of cracks parallel to the surface which finally would limit the thermal transfer and lead to local overheating and probably melting. Furthermore, the thermal stability of material, i.e. the resistance to recrystallization was determined by thermal annealing up to 1800 °C and during the electron beam tests by applying heat loads that lead to a surface temperature increase of > 2000 K.

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Evolution of tungsten degradation under combined high cycle ELM and steady state heat loads

P69A

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In ongoing research, investigating the impact of thermo shock loads on divertor materials, usually up to 10^3 shocks are applied. Within the ITER divertor lifetime 10^6 or more transient events are expected during H-mode operation due to edge localised modes (type I ELMs) [1]. In order to simulate ELM-like heat loads the electron beam facility JUDITH 2 (Juelich divertor test facility in hot cells) in Forschungszentrum Jülich was used. A method was developed to apply the loads with sufficiently high power density (~0.5 GW/m^2), ELM-like pulse duration (0.5 ms) and a frequency (25 Hz) close to the ELM frequency expected for ITER (several Hz) and fast enough to enable experiments with 10^4-10^6 loads within a reasonable time.
Beam diameter and beam guidance are the main parameters to be under control in order to achieve the desired conditions. The developed circular beam loading method [2] provides stable conditions in case of small parameter fluctuations (e. g. vacuum pressure) influencing the beam diameter and hence the power density. The digital beam guidance system facilitates the application of additional steady state heat loads during the interpulse time between successive ELM loads. Hence synergistic effects originating from combined transient ELM-loads and steady state heat loads (SSHL) can be evaluated at different base temperatures of the plasma facing material.

For the tests, tungsten tiles with a geometry of 12x12x5 mm^3 were brazed to an actively cooled copper heat sink and loaded with ELM intensities of 0.14 GW/m^2, 0.27 GW/m^2 and 0.41 GW/m^2 (t = 0.5 ms) as well as 0, 5 and 10 MW/m^2 additional SSHL. Experiments were performed applying 10^3-10^6 ELM pulses. Depending on ELM intensity, pulse number and SSHL different degradation levels can be observed and evaluated by a number of diagnostics: Optical inspection and light microscopy as well as laser profilometry for surface modifications (e. g. roughening), SEM and metallographic analyses for cracks and erosion or melting of crack edges.

No material damage was found at 0.14 GW/m^2 (independent of the SSHL) for up to 250,000 pulses. This value is therefore considered as damage threshold. For 0.27 GW/m^2 surface modifications occurred at 10^5 pulses with and without SSHL. At 10^6 pulses (without SSHL) thermal fatigue induced cracks with eroded crack edges were found indicating a continuous material degradation with increasing pulse number. For 0.41 GW/m^2 cracking already occurs at 10^4 pulses (without SSHL). Detailed analyses of the results and the resulting material degradation are provided in this paper.

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Comparison of the thermal shock performance of different tungsten grades and the influence of microstructure on the damage behaviour

P69B

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One serious concern for the realisation of a thermonuclear fusion reactor is the choice of the plasma facing materials (PFMs). These have to withstand very high steady state heat loads (up to 20 MW/m²) and transient events like disruptions, vertical displacement events (VDEs) and edge localised modes (ELMs) with heat loads up to several MJ/m². One candidate as a PFM, beside beryllium and CFC, is tungsten. Its advantages are a high thermal conductivity, high melting point and low tritium inventory. However there are some drawbacks like the brittleness at low temperatures and the high atomic number.

During a transient event like an ELM thermally induced stresses in the PFM cause damages, e.g. surface modifications due to plastic deformation and crack networks. How severe these damages are depends strongly on the materials mechanical and thermal properties and on their microstructure. The grain orientation thereby influences the crack propagation through the material. The formed cracks, particularly when oriented parallel to the surface, could reduce the thermal conductivity and lead to melting and erosion of whole surface parts of the PFM.

In order to understand the mechanisms which influence the damage behaviour and the crack pattern, particularly related to the specific material properties, several tungsten grades were tested in the electron beam facility JUDITH 1 (Juelich Divertor Test Facility in Hot Cells) [1]. This work focuses on three single forged tungsten grades, i.e. W-UHP (ultra high purity tungsten), WTa1, and WTa5. The latter contain 1 and 5 weight% of tantalum, respectively [2]. All three materials were tested at different base temperatures and power densities for two different grain orientations and in their recrystallised state.

The results indicate that WTa5 offers the best performance, which is most probably due to the highest mechanical strength of all tested tungsten grades and almost unaffected by the inferior thermo-physical properties. Furthermore, the cracking and damage thresholds vary depending on the testing orientation which corresponds to the materials anisotropic properties. Annealing and recrystallisation leads to a change of damage and cracking thresholds and to a better conformity of the different material orientations.

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Material performance of tungsten coatings under transient heat loads

P70A

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First wall components for applications in future nuclear fusion devices need to fulfill special requirements, e.g. a good thermal conductivity, a reasonable strength value as well as a good compatibility with a deuterium/tritium plasma. Especially transient and/or cyclic thermal loads in magnetic confinement experiments like ITER have a severe impact on the material damage of the plasma facing components.

Tungsten coatings are being assessed for use instead of bulk tungsten components. Within the ITER like wall project, realized at JET, a part of the thermally loaded wall will consist of tungsten coated Carbon Fiber reinforced Carbon (CFC) modules. The coating with a thickness of about 25 µm was produced by a Combined Magnetron Sputtering and Ion Implantation (CMSII) coating technique in Romania [1].

In order to quantify the material degradation under transient ELM-like (Edge Localized Modes) heat loads, small specimens of tungsten coated CFC substrates were exposed to 100 short thermal pulses in the electron beam material test facility JUDITH 1 (Juelich Divertor Test Facility in Hot Cells) for various power densities and base temperatures. The damage threshold for the tested layer system of a W/Mo/W/Mo coating was found to be approximately 158 MW/m^2 for pulse durations of 1 ms. The influence of base temperature in a range between room temperature and 400°C is negligible. Additionally, the failure mechanism was investigated in light microscopy as well as in SEM analysis of cross sections. These investigations have shown that coating degradation is mainly dependent on the fiber orientation of the CFC substrate.

Finally, high cycle thermo shock fatigue tests (ELM like heat loads) on tungsten coated divertor tiles for the ITER like wall project with cycle numbers up to 10^5 pulses for different power densities were performed in the electron beam test facility JUDITH 2. The results of the experiments confirm that increasing cycle numbers lead to an accumulation of damage.

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P70B

ITER ELM's Simulation with Plasma Accelerators

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Divertor armor response to the repetitive plasma impacts during the transient events in ITER and DEMO remains one of the most important issues that determine the tokamak performance. Erosion of plasma-facing components in fusion devices leads to contamination of the hot plasma by heavy impurities and can produce a substantial amount of the material dust.

Experimental simulations of ITER transient events with relevant surface heat load parameters (energy density and the pulse duration) as well as particle loads were performed with quasi-steady-state plasma accelerator QSPA Kh-50 that is largest and most powerful device of this kind. Pulsed plasma gun PPA and rod type injector IBIS were used also for comparative studies of an initial stage of PSI, evaporated impurities dynamics, features of surface damages appearing under varying plasma parameters and sort of plasma ions.

Performed studies of plasma-surface interaction include measurements of plasma energy deposited to the material surface as a function of the impacting energy and angles of plasma streams incidence for W, C and adjoined W-C surfaces under repetitive ELM-like plasma exposures. Dynamics of the W- and C-ions in near surface plasma was studied by means optical spectroscopy. Particularly, Mechelle®900 spectrometer equipped with a CCD-camera and operated in the wavelength range (300 nm-1100 nm) with different exposition times was in use. Some issues of the droplet splashing at the tungsten surfaces and the formation of hot spots upon the graphite surface, which can be sources of the enhanced evaporation and significant amount of dust, are also discussed basing on dust dynamics and droplets monitoring with high speed camera.

Influence of material modification by plasma exposures on cracking thresholds of tungsten is emphasized. It was found that increasing number of exposures shifts the energy threshold for crack development to essentially lower values. Cracks evolution and changes in their thickness with increasing exposition dose is studied for deformed W grade and compared with sintered tungsten. Analysis of X-ray diffraction peaks intensity, profiles, and their angular positions was applied to evaluate the texture, the macro-strain and the lattice parameters. Non uniform changes of both stress-free lattice spacing and half-width of diffraction maximum are observed under heat loads above the tungsten melting threshold. Differences in evolution of tungsten substructure after exposures with helium and hydrogen plasma streams of different duration are analyzed.

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P71A

Experimental Investigation of Radiation Resistance of a Number of the Tokamak Plasma Facing Ceramic Materials by means of Plasma Focus Devices

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For experiments simulating conditions realized on the first wall of tokamak chambers we used 4 devices of the Dense Plasma Focus (DPF) types – PF-1000, PF-6 (both are in operation at IPPLM, Poland), PF-5M (IMET, RF) and “Bora” (ICTP, Italy). Researches undertaken on evaluation of dynamics of plasma and beams of fast ions of deuterium have shown that the parameters of the above streams are very close to that taking place near the tokamak’s wall during the so-called Edge Localized Mode events (plasma temperature, energy of fast ion beam, etc.). Researches on investigation of radiation resistance were provided for CFC types of ceramics and for BN and Al_2O_3. The first ones are used in divertors of tokamaks whereas the last two – in TAE Antenna coil on MAST spherical tokamak. We undertake experiments to compare their radiation resistance. Samples of the materials (blocks of CFC, bulk for BN and 20-μm film on Al substrate for Al_2O_3) were exposed on the axis of the DPF devices to intense streams of hot plasma (T_{pl}≤1 keV, v≥107 cm/s and N_{pl}~1018 cm-3) and fast deuterons (E_i~100 keV). Time duration of the most intense phase of the pulses is ~ 0.2...1.0 μs subsequently decreasing till lower figures (e.g. down to T_{pl}~2 eV) during 100 μs. The irradiation process was diagnosed by fast optical cameras, laser interferometry and optical spectrometry. Experiments were performed at power flux densities during first phase of action 109…1010 and 108…109 W/cm2. Irradiated specimens were investigated by optical microscopy (OM), X-Ray structure analysis (XRSA) and photometric analysis of structural images. Results:
1. Characteristic features of damage of material based on a carbon-fiber composite (CFC) with additions of SiC of 8 and 40 vol.% were investigated. It was found that at q = 10^9 W/cm^2 an erosion of the surface layer is mainly associated with the processes of sputtering and evaporation of the material. Degree of erosion depends on orientation of fibers in relation to the direction of the streams and on the percentage of the SiC dopant.
2. At 1010 W/cm2 Al_2O_3 coating is completely evaporated.
3. Direct comparison of samples of Al_2O_3 film and bulk of BN irradiated at 108-W/cm2 provided by means of OM has shown a wave-like structure that becomes apparent more profoundly in Al_2O_3. Yet weighing of samples has shown that evaporation of BN was ~ 2 times higher than that for Al_2O_3.
4. The XRSA has shown no evidence of cracking of Al_2O_3. So the insulation properties of Al_2O_3 did not decline, and it may be potentially more beneficial if it is kept below melting point.

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Development of PWI-plasma Gun as Transient Heat Load Source for ELM-simulation experiments

P71B

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The divertor plasma-facing materials (PFMs) in ITER are exposed to both the steady-state divertor plasma and the intense transient heat flux during edge localized modes (ELMs). In ITER, the ELM heat loads are predicted to be 0.2–2 MJ/m^2 during 0.1–1 ms on the divertor plate during each cyclic event. These intense transient heat loads (>10^3 MW/m^2) are more critical when compared with the static heat loads during normal operation which are estimated to be up to levels of 20 MW/m^2 for the divertor targets. Thus, it becomes important to perform heat load tests for R&D of ITER PFMs under these realistic conditions. Earlier transient heat load tests with plasma guns simulators in Russia (e.g., QSPA-T) [1] demonstrated various erosion processes of a tungsten (W) surface such as melt layer splashing, cracks and brides. In Japan, we have recently started the ELM simulation experiments using the magnetized coaxial plasma gun (MCPG) [2] under research collaboration with several institutions (UCSD, Osaka Univ., Nagoya Univ., Kanazawa Univ., JAEA and NIFS).

The present MCPG in Univ. of Hyogo provides that the electron density and ion flux of the plasma is ~1 × 10^{21} m^{-3} and ~5 × 10^{25} $m^{-2}s^{-1}$, respectively. The time of flight measurement of magnetic fields provides the velocity of ~50 km/s, corresponding to the ion energy ~30 eV for deuterium (D) ion. The initial W irradiation experiment reported that although cracks were formed on a W surface at the energy density of ~0.7 MJ/m^2, no melting of the W surface was observed under a single pulse exposure [3]. We are now developing the well-controlled MCPG with higher performance to increase the energy density up to 1.5~2.0 MJ/m^2 that may make it possible to cause droplet ejection from a melting W surface. The capacitor banks energy W_{bank} will be increased from 24.5kJ (C=1 mF) to 70.6 kJ (C=2.9 mF) at the same charging voltage V_g=7kV. We have already obtained the energy density ~1.2 MJ/m^2 of MCPG with W_{bank} = 30 kJ (C=2.4 mF, V_g =5 kV) in a preliminary examination. This paper presents primarily the results from examining performances of the improved MCPG facility and diagnosing plasma parameters. We will discuss there dynamics of the gun-produced plasma stream, a cloud formation of dense vapor plasma in front of a target surface and similarities of the plasma parameters with ELM plasmoids [4].

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Optimization of QSPA-Be plasma gun facility for ITER ELM, disruption, and mitigated disruption simulation experiments. Preliminary results of Be erosion under ELM-like plasma heat loads

P72A

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The first wall PFCs erosion under ITER transient plasma events such as ELM, disruption and mitigated disruption is expected to determine the PFCs lifetime and amount of erosion products in a form of dust particles and films. The magnitude of ITER plasma heat loads during transient plasma events are not achieved in existing tokamaks so other devices are used for armour testing. The quasistationary plasma guns such as the QSPA-T [1, 2] and QSPA-Be facilities provide the hydrogen (or deuterium) plasma heat loads relevant to ITER ELM and disruption in the range of 0.2-5 MJ/m^2 and pulse duration 0.5 ms. Because of specific safety requirements QSPA-Be facility was installed in Bochvar Institute and was licensed to work with beryllium targets.

In this work the primary attention is focused on the following points: a) optimization of QSPA-Be power supply system to obtain power pulse form relevant to different transient plasma events of ITER; b) calibration of QSPA-Be which include the measurements of plasma velocity, pressure, and heat loads depend on operating parameters (gun voltage, gas consumption); c) experimental study of plasma stream energy transformation to radiation for mitigated disruption simulation; d) experimental study of beryllium erosion under ITER ELM-like plasma heat loads up to 1 MJ/m^2.

Plasma gun facility operating mode corresponding to ELM and disruption-like plasma heat loads on the target was determined. The plasma flow velocity, plasma pressure, absorbed energy distribution on the target surface as well as radiation spectrum were measured as a function of plasma gun voltage.

As a result of the experiments the different type of beryllium (TGP56-PS and S-65C) were exposed by hydrogen plasma flow in the heat loads range of 0.2-1MJ/m^2, 0.5 ms pulse duration and inclined plasma action. Measured melting threshold for both type of beryllium equal to 0.5 MJ/m^2. The specific mass loss and erosion rate determined as a result of mass loss measurement decreased with pulse number increasing. After the second pulse the specific mass loss equal 3,5 g/m^2/pulse (erosion rate 2 µm/pulse), after the pulse number 50 the specific mass loss decreased down to 0,5 g/m^2/pulse (erosion rate 0.3 µm/pulse). The mass loss decreasing is a result of edges smoothing. According of the surface investigation by means of optical and electron microscope the main erosion mechanisms of Be at the heat load 0.5-1 MJ/m^2 is a melt layer movement and splashing.

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Dynamic Response of Refractory Metal Electrode to ~GW/m^2 Plasma Heat Load in the Stabilized Arc Discharges

P72B

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Tungsten, which will be used as a divertor target in ITER D-T operational phase, has many advantages compared to CFC, such as low physical sputtering, low tritium retension, high thermal conductivity, high melting temperature(3695 K) and so on. The usage of tungsten, however, brings another problem different from CFC. The surface modification by hydrogen and/or helium plasma irradiation at elevated temperature has been studied extensively and it has been discussed what are benefits and drawbacks when tungsten is used as plasma-facing components [1].
Study of the dynamic response of tungsten surface to the transient and extremely high plasma heat load in type-I ELM's and disruptions (>100MW/m^2)[2] requires experimental approaches different from those in steady state and low heat flux experiments (<10 MW/m^2). Simulation experiments of material erosion in ITER using high power plasma guns have been intensively studied [3]. In the present experiments, high current stabilized arc plasmas with ~GA/m^2 are used as a high heat flux pulse and steady state plasma source. The plasma heat flux onto the cathode surface is several MW/m^2 in steady state and is several GW/m^2 in arc ignition phase. These properties of high heat flux arc plasmas are very useful to study the transient behaviour of the divertor materials during ELMs and disruptions in fusion reactor complementally with other ELM/disruption experiments. In Fig. 1 the abrupt ejection of molten hafnium (Hf) from the cathode surface observed by high speed color camera is shown. The bright area in the figure is the hot molten cathode with T=3,500 K ~ 4,000 K. The cathode surface temperature is estimated from RGB intensities of the color camera image under black body approximation. In the experiments disruptive ejection of the cathode molten materials, which may related to some fluid dynamic instabilities, is observed in the steady state arc in addition to the arc ignition phase with pulse ~GW/m^2 heat flux.

Fig. 1 Successive pictures of Hf cathode with 150 A arc current during abrupt disruptive instability. The cathode area is about 2 mm^2. Each time interval is 125 μs.

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Experimental devices for PFM testing in NRI Rez, plc.

P73A

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NRI Rez plc continues to perform fusion-related tasks from EFDA/F4E which are in general aimed on the testing of PFM mock-ups. This paper is dealing with high-heat flux generating devices, which are developed and operated under TW6-TVM-TFTEST, TW4-TVB-TFTEST2 and TW3-TVB-INPILE tasks, respectively. TW6-TVM-TFTEST and TW4-TVB-TFTEST2 tasks are focused on development and operation of out-of-pile testing loop called BESTH device (BEryllium Sample THermal testing device), while TW3-TVB-INPILE aims for development and operation of in-pile testing rig, so far known as 'TW3 rig'. Although both devices are used for generation of almost same heat fluxes to Beryllium-coated mock-ups, the development of BESTH device was already finished and the loop was put in full operation, where six First Wall Qualification Mock-ups (FWQM) were already tested. The development of TW3 rig is ongoing and the rig is planned to be inserted to the LVR-15 research reactor's core by Q2/2011.
The paper describes both experimental devices, it capabilities, faced obstacles, operational experiences and it also briefly deals with results, unless restricted under commercial contracts.

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Recent Results in High Thermal Energy Load testing of beryllium for ITER Fist Wall application

P73B

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Beryllium will be used as a plasma facing material in the ITER First Wall and Port Limiter. During ITER operating the beryllium armour of first wall will be exposed by (1) steady heat loads (normal event) that will provoke both permanent and cyclic fatigue stresses of moderate level and (2) plasma instabilities of different kind (“off-normal” event), which will stimulate high local stresses (disruptions, VDE, ELMs). All these events may lead to surface melting, cracking, evaporation and erosion of beryllium. For Be armour, thermal fatigue/shock resistance is the most important factor, because cracking may lead not only to the intensified armour erosion but to damage of its joint with the heat sink structure.

This paper presents recent results of a complex HHF testing of two modification of Russian beryllium grade TGP-56FW in comparison with reference S-65C grade. Testing was performed in the frame of ITER IO qualification procedure. The thermal loading of beryllium was performed using two actively cooled Be/CuCrZr brazing mock-ups, each of them was armored with four beryllium tiles of $40\times24\times10$ mm^3. Each tile of every mock-up was loaded in the electron beam facility JUDITH 1 (FZJ) in the following way: 1) VDE simulation test at 40 MJ/m^2, 1 shot, 0.3 s; 2) Disruption simulation at 3 MJ/m^2, 2 shots, Δt = 5 ms; 3) Low cycle fatigue test at 80 MW/ m^2, 1000 shots, Δt = 25 ms. In addition to 1) - 3) tests, the second mock-up was exposed to 1000 cycles at 2 MW/m^2, Δt = 15 s heating /15 s cooling. The results of metallographic studies of microstructure and cracks morphology in Be tiles after this thermal loading will be reported and discussed. The overall results of TGP-56FWgrade qualification have demonstrated the reliable performance capabilities for application as armor of ITER First Wall.

Other important task of ITER R&D activity is the investigation of beryllium first wall erosion in the ITER-like plasma-wall interaction conditions. To obtain the experimental data for evaluation of the beryllium armor lifetime and dust production under ITER-relevant transient loads (ELMs, disruptions, radiative heat) the new plasma gun QSPA-Be facility has been prepared to operating in Bochvar Institute. The results of first experiments with berylluim will be reported. Beryllium targets ($80\times80\times10$ mm^3) were tested by hydrogen plasma steams (5 cm in diameter) with pulse duration 0.5 ms and heat load 0.5 – 2 MJ/m^2. Experiments were performed at RT initial temperatures. The investigations of evolution of surface microstructure and rate of erosion process exposed up to 100 shots will be presented and discussed. The targets

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Modelling of Massive Gas Injection with tokamak code TOKES for ITER Disruption Mitigation Design

P74A

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In the future tokamak ITER the damage to the wall after the disruptions can be mitigated using preventive massive gas injection (MGI) of noble gases into confined plasma during the thermal quench. The gas gets ionized in the plasma, and then the ions dump into the scrape-off layer (SOL) and impact on the target. The contamination of core plasma results in fast loss of plasma energy by radiation. The radiation distributes over the first wall which decreases the damage to divertor plasma facing components. However, enhanced radiation load in e.g. vicinity of gas jet entry is an issue for ITER design that can be addressed numerically.

For the modelling the tokamak code TOKES is applied. Preliminary results on modelling of MGI with TOKES have been presented in [1]. Two-dimensional toroidally symmetric plasma model allowed estimations of radiation fluxes and the expansion of noble ions both across and along the magnetic surfaces. However, a simplified quasi-stationary radiation model employed in [1] had not allowed validation of numerical results against tokamak experiments. The processes only in the confined region inside the tokamak separatrix were taken into account and the plasma surface interaction neglected.

In this work, significantly improved non-stationary radiation model for 2D plasma implemented in TOKES is employed. The new model was recently validated against the tokamak DIII-D [2]. Also 2D modelling of plasma and neutrals in the whole tokamak vessel with account of wall processes is achieved. Due to this the contribution of atoms eroded and emitted from the wall surface during MGI to the plasma contamination is numerically estimated. Elaborated predictive simulations for MGI into ITER deuterium confined plasma with assessments of plasma fluxes and radiation fluxes from the contaminated plasma onto the facing components are presented.

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Development of Transient Tolerant Plasma Facing Material

P74B

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Plasma facing material (PFM) is a critical element of the high performance DT tokamak reactor design. It is the interface between the plasma and the high performance first wall and divertor components. Presently, solid W is projected as the preferred PFM due to its low physical sputtering, high thermal performance at elevated temperatures, and high neutron fluence tolerant properties. Unfortunately, the commonly proposed material W could suffer radiation damage from implantation of alpha charged particles and experience blistering at the first wall and the formation of submicron fine structure at the divertor. Furthermore, it will melt under Type-I edge localized modes (ELMs) and disruption events. A potential remedy was demonstrated using linear machine material exposure which showed that, with a background of low-Z material, the W-surface damage could be prevented or significantly reduced. On transient tolerant, as a conservative engineering design, the first wall and divertor PFM must withstand a few unanticipated disruptions even when the disruption and ELM mitigation techniques are fully engaged. Using a low-Z sacrificial material, like Si, deposited on the W-surface will allow W to withstand type-I ELMs and disruptions without serious damage while retaining the capability of transmitting high heat flux for power conversion. An equivalent Si thickness of 10 μm is sufficient to form a shielding layer during a disruption that protects the W-substrate from serious damage. Accordingly, transient tolerant PFM surface test buttons have been fabricated and initial results have been obtained with exposure in the DIII-D divertor. This paper is a status report on the development of the Si-W transient tolerant PFM design.

This work was supported in part by the US Department of Energy under DE-FC02-04ER54698 and DE-FG02-08ER54917.

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Estimation of dust production rate from tungsten armour after repetitive ELM-like heat loads

P75A

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The edge localised modes (ELMs) of plasma instabilities, intrinsic for the ITER reference operational scenario, produce short periodic pulses of heat flux at the divertor armour, being the most heat loaded part of the tokamak. The heat deposition to the divertor armour reaches up to 1-5 MJ/m^2 for ~0.1-0.5 ms during unmitigated ELMs of type I, causing the divertor armour cracking or even melting.
Carbon fibre composite (CFC) being the reference material for ITER divertor armour for first He-H stage of the reactor operation does not fit the safety requirements for the thermonuclear stage of the ITER operation. Most suitable material for the armour operating with D-T plasma is tungsten. It can withstand the stationary heat flux and have tolerably low erosion rate and tritium retention. Further tungsten investigations as the divertor armour material concerned its behaviour under action of ELMs and disruptions. Simulations of the tungsten armour cracking due to the type I ELMs of various sizes have been successfully performed in [1] and [2] using the PEGASUS-3D code. The results of these investigations assume that the giant ELMs are intolerable and should be mitigated for the industrial type reactor.
The ITER armour produced from tungsten with grains elongated perpendicular to the armour surface to avoid the cracks going parallel to the heated surface. Cracking of this armour due to the ITER-size ELMs is unavoidable, but it does not decrease the effective armour thermo-conductivity because these cracks are perpendicular to the heated surface. Most dangerous consequence of the armour cracking is production of tungsten dust, which can contaminate the thermonuclear core of the reactor decreasing its power gain or even initiating the disruption. In contrast with the CFC, which is the reference ITER divertor armour material, contamination of the reactor core with tungsten needs several orders of the magnitude less density in the reactor core for the catastrophic consequences.
Estimation of tungsten dust production rate during the ELMs is extremely important issue for ITER performance. With this aim a dedicated series of experiments have been performed in the quasi-stationary plasma accelerator QSPA Kh-50. Tungsten targets behaviour under the repetitive ELM-like plasma exposures have been studied in these simulation experiments and the dust production rate have been measured under the ITER ELM-like plasma shots of 0.25 ms time duration. The experimental results valid for the ELMs of this duration are extrapolated for ELMs of different sizes and time durations using the results of PEGASUS-3D simulations.

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Influences of ELM-like Pulsed Plasma Bombardment on Deuterium Retention in Tungsten

P75B

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In-vessel tritium (T) retention in ITER is limited to 700 g for safety reasons, and thus significant efforts have been made to accurately predict T retention [1]. Experimental investigations on this issue have been conducted mainly in linear divertor plasma simulators and ion beams under steady-state conditions. However, plasma-facing materials (PFMs) in ITER will be subjected to cyclic transient plasma loads induced by edge localized modes (ELMs) in addition to steady-state loads, and influences of transient plasma loads on T retention in PFMs have not been clarified well.

In this study, stress-relieved pure tungsten (W) samples (25.4 mm in diameter and 1.5 mm in thickness) were first exposed to steady-state deuterium (D) plasma (ion fluence ~ 5×10^{25} m^{-2} at sample temperature ~ 573 K) in the linear divertor plasma simulator PISCES-A [2], and then bombarded by ELM-like pulsed D plasma (ion fluence ~ 7.5×10^{21} m^{-2} per shot, surface absorbed energy density ~ 0.5 MJ m^{-2} at pulse width ~ 0.5 ms) in a magnetized coaxial plasma gun (MCPG) [3]. The base temperature of W samples before pulsed plasma bombardment was ~ 300 K, and the peak surface temperature during bombardment is calculated to be ~ 1900 K by numerically solving the heat conduction equation. Steady-state D plasma exposure created D blisters (~ 1 μm in diameter) on the surface, which cracked at the edge of the sample following 10 plasma gun shots. D retention properties of W samples were examined by thermal desorption spectroscopy (TDS). With no plasma gun shot, a large desorption peak at ~ 850 K and a small one at ~ 670 K were observed. After 10 plasma gun shots, the small peak disappeared, while a high temperature peak at ~ 1000 K emerged. This high temperature peak has been also observed in a high temperature (773 K) ion beam irradiation experiment [4]. Regardless of the change in the desorption spectra, it is found that the total D retention does not significantly alter by pulsed plasma bombardment: 9.7×10^{20} and 9.9×10^{20} D m^{-2} with and without 10 transient shots, respectively.

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Analysis of molybdenum erosion and transport for toroidal limiter in FTU using the ERO code

P76A

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In fusion devices, the erosion of wall surfaces by plasma contact can limit the lifetime of plasma-facing components (PFC). The eroded wall materials can be transported into the main plasma, which seriously affect the plasma quality. Therefore, it is important to understand the erosion and transport processes to make reliable predictions for future devices, such as ITER.

The 3D Monte-Carlo code ERO [1], which simulates surface erosion, impurity transport and deposition in a given background plasma, has been used to estimate the target lifetime and tritium retention of the ITER divertor [2]. However, further code development and benchmarking with existing experiment is still necessary. In this contribution, ERO has been applied to simulate the erosion and re-deposition of the toroidal limiter material in the FTU tokamak. In the near future application of ERO code to FAST, the satellite tokamak proposed by Italian association on Fusion, is foreseen too.

The full toroidal limiter in molybdenum, located at the inboard wall, is the main limiter in high-magnetic-field FTU tokamak [3]. Its geometry has been implemented into the ERO code. About half of the poloidal length is nearly tangential to the last closed flux surface for circular or slightly elongated discharges and faces the core plasma directly, where a constant cross field particle flux is assumed. Only the poloidal component of incident plasma flux is considered and the molybdenum influx into plasma, due to physical sputtering, is calculated according to the local incident angle. The dependence of magnetic field, the electrical field parallel to the magnetic field as well as the plasma flow velocity on the location in the scrape-off layer (SOL) is also taken into account. The analysis of molybdenum erosion and re-deposition along the limiter surface will be given. It is found that more than 50% of eroded molybdenum is re-deposited. The net erosion rate of molybdenum is relatively low. The light emission of molybdenum atom and ions are calculated using corresponding effective photon emission coefficients and are compared with spectroscopic measurements.

In addition, the increase of Z_{eff} due to the presence of molybdenum impurity in the core plasma is analyzed, which is compared with diagnostics. Detailed modelling results will be presented in comparison with experimental observations.

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Divertor Studies for FAST, a proposed European satellite tokamak for the fast track to fusion

P76B

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One of the objectives of the proposed FAST tokamak is to investigate the plasma-wall interaction in ITER and DEMO relevant conditions. The FAST main parameters (R=1.82 m, B_T≤8.5 T, I_p≤8 MA, P_{add}~30÷40 MW) make it very well suited for ITER and quite well positioned for DEMO. Indeed the SOL reference quantity, P/R^x with 1≤x≤2, is ~22 MW/m for x=1 (as in ITER) and ~11.9 MW/m^2 for x=2 (as in DEMO with P_α=0.54 GW, P_{add}=0.3 GW, R=8.5 m). Clear important contribution to DEMO relevant technical issues derives from the full W wall and divertor and from the also foreseen liquid metal (Li) divertor target. For a consistent design it is necessary to have a reliable estimate of the thermal loads on the divertor targets by predicting the SOL behavior for the different FAST scenarios. This has been done in two steps. Firstly a self-consistent SOL description is performed with the coupled edge-core code COREDIV, 1D in the core and 2D in the SOL. Next a more detailed analysis is carried out with the code EDGE2D/EIRENE. Indeed, this code describes in more details the SOL and divertor geometries and treats more reliably the neutral physics through a Monte-Carlo technique, even though it is fully decoupled from the core. The fixed inputs to the SOL that are requested by the code are derived from the outputs of COREDIV. In this first stage we mainly aimed at optimizing the divertor design and at identifying the conditions under which acceptable divertor thermal load are attained. Particular attention is given to the configuration and the local conditions that can favor the plasma detachment from both outer and inner targets. Five different ideal geometries of the divertor have been considered, varying the tilting angle of the outer and inner divertor plates (one or both) and/or the closeness of the divertor chamber in order to modify the neutral dynamics.
The numerical investigations with the EDGE2D/EIRENE have scanned the reasonable ranges of the main parameters that COREDIV foresees for the envisaged FAST scenarios. They are $n_{s,out}=0.3\text{-}1.5\times10^{20}$ m^{-3} for the plasma density at the outer midplane, and P_{SOL}=10-30 MW for the power input into the SOL. The extreme values refer respectively to the full non-inductive case with an averaged plasma density $\langle n_e\rangle=0.7\times10^{20}$ m^{-3} and significant impurity seeding and to the extreme H-mode with $\langle n_e\rangle=5\times10^{20}$ m^{-3} and no injected impurity.
In general, detachment in the inner divertor is more easily attained than in the outer divertor targets. On both it can be attained even with no impurity and relatively low radiation losses at the highest density, but the intermediate density regimes require an increase in the SOL and/or bulk radiation loss in order to alleviate the thermal load on the outer target. To this purpose the effect induced by injecting Ar and Ne is studied and will be extended also to N. However also the plate tilting angle and the neutral dynamics are crucial factors, a careful optimization process of the divertor geometry would be quite important. Also very important are the transport coefficients, here assumed similar to ITER, with a significant benefit if they are increased, specially for $D_\perp$. The first simulations of the ELM load are also reported.

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W thick coatings on CuCrZr and steel for plasma facing components

P77A

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Characterization of W thick (up to 4 mm) coatings deposited on CuCrZr alloy (ELBRODUR) via Plasma Spraying, with optimised composite interlayer of Ni, Al, Si and W, has been carried out. The structure of the system coating-interlayer and the distribution of chemical elements have been investigated by SEM observations with EDS microanalitical mapping. Coating and interlayer appear compact without macro porosity and no long range diffusion of the elements forming the interlayer occurs in the W coating.
The mechanical characteristics of the coatings have been investigated by means of instrumented indentation tests at increasing temperatures up to 500 °C.
X-ray diffraction measurements carried out in the temperature range from 25 to 425 °C showed the presence of strains in the interlayer whereas they are negligible in coating and substrate.
The results of present experiments confirm that Plasma Spraying is a reliable technique for realizing W thick coatings for plasma facing components.
Recent results on W coating of stainless steel (AISI 316) substrates by using different interface layers consisting of a grading mixture of W and SS powders, a mixture of NiAl, SiAl and W and SiAl/W only will be also discussed.

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EDGE2D-EIRENE calculations of JET ILW pedestal and SOL near radiative collapse

P77B

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EDGE2D is a fluid SOL code which models W in 6 super-stages. EIRENE is a Monte-Carlo code which calculates the neutral deuterium and impurity neutrals. One difficulty with tungsten is the high radiation per tungsten atom. Radiative collapse occurs when tungsten impurity concentrations lead to the tungsten radiation being significant in the electron energy balance.

In this paper, EDGE2D calculated effects due to radiative collapse on an hypothetical JET H-Mode SOL, pedestal, and divertors. The tungsten was introduced by injection (not sputtering) and the amount was increased until W radiation was significant in the energy balance. The fraction of tungsten in the pedestal rose linearly with the injection rate of the tungsten. Typically, about 100 times more tungsten must be injected into the divertor compared to the outer mid-plane in order to achieve the same pedestal tungsten density.

The primary effect on the pedestal electron energy balance is the reduction in separatrix power flow. The energy balance in the outer and inner divertors, or SOL indicate that the W radiation itself is insignificant in those regions. However, the reduced power inflow has several effects. The electron energy balance in the divertors indicates that the atomic deuterium processes (ionization, atomic and molecular radiation) become the dominant and larger than the power flowing to the targets.

The radiative collapse also reduced the plasma temperatures in front of the targets. Consequently, when the tungsten radiation becomes large, the target temperatures are reduced, lowering the W sputtering, and reducing the tungsten radiation. Thus the W sputtering has a self-regulating nature. ELMs have not yet been considered but are expected to be important.

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ASCOT simulations of ^{13}C transport in ASDEX Upgrade

P78A

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In ITER, co-deposition of plasma fuel with carbon may lead to substantial retention of tritium on the plasma-facing components. Thus, the atomic and molecular physics of carbon and other impurities as well as their migration in a tokamak environment have to be understood. A new simulation tool has been developed that avoids the shortcomings of the fluid approach usually applied to the impurity migration problem.

ASCOT is a Monte Carlo guiding-center-following code employing magnetic and plasma backgrounds that can originate from experimental sources. Today, ASCOT is the most complete code of its kind in Europe. It accurately accounts for all the neoclassical physics and can now be operated with 3D wall structures and magnetic fields. Full orbit following can be adopted in situations where finite-Larmor-radius effects are anticipated to play a significant role.

Migration and deposition of carbon is studied by trace-element injection experiments. In 2007 in ASDEX Upgrade (AUG), $^{13}CH_4$ (methane) was injected into the torus and a set of wall tiles was removed for surface analyses with secondary ion mass spectrometry. Less than 10% of the injected carbon could be accounted for by assuming toroidally symmetric deposition [1]. In simulations with the DIVIMP code, deposition in the inner divertor was only achieved by imposing a scrape-off-layer (SOL) flow pattern suggested by experimental measurements [2]. The significance of the DIVIMP results was also limited by other weaknesses: the simulation grid did not extend all the way to the first wall, the ion drifts were not yet implemented in the code, and both first wall structures and the magnetic field were assumed toroidally symmetric.

To overcome these shortcomings, the ASCOT code has been upgraded to permit impurity transport studies in SOL, including imposed plasma background flow and atomic physics for carbon.

In this contribution we introduce ASCOT-PWI and report results from the first simulations of the 2007 carbon injection experiments at AUG. According to these results, the assumption of a toroidally symmetric wall is particularly insufficient for predicting locations of high deposition in the main chamber. In particular, protruding wall structures, such as port limiters near the injection location, are found to cause very localized deposition patterns in toroidal direction. However, it should be noted that at this stage no re-erosion was included in the simulations.

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P78B

Relevant processes for hydrocarbon transport, break-up, and light emission in an ITER divertor-relevant plasma

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Plasma-surface interaction in the ITER divertor including erosion of carbon target plates and tritium retention due to co-deposition limits the availability of ITER. Therefore, modeling of carbon erosion/deposition processes is of importance. However, the model itself should be validated by dedicated experiments at existing devices.

The chemical erosion of carbon, which produces hydrocarbons, is often diagnosed by emission of the CH A-X Gerö band around 430 nm. At electron temperatures below 2 eV, such as expected close to the ITER divertor, assumptions underlying the interpretation of such measurements become uncertain. In particular, the dissociative recombination of CH_4^+, CH_3^+, and CH_2^+ yielding (electronically excited) CH is expected to become an important contributor to light emission [1].

In the present work, simulated CH A-X molecular band emission intensity profiles from the 3D Monte Carlo particle tracing code ERO [2] are benchmarked against experiments in the linear plasma generator Pilot-PSI with T_e ~ 1 eV, n_e ~ 10^{20} m^{-3} at the plasma axis and B=0.4 T. Using a target made of molybdenum, methane was injected both through a hole in the center of the target [3] and into the edge of the plasma where the electron temperature is lower. The contribution of dissociative recombination to the production of excited CH has been included in the respective simulations, and it is confirmed that this process dominates at electron temperatures below 1 eV. From fluid modeling using the B2.5 code, we have indications that the electron density and temperature decay in the final millimeters before the target. Including this in ERO modeling was found to reduce simulated CH emission by a factor ~ 2, underlining the importance of a thorough understanding of the plasma. Finally, carbon deposition profiles from simulation and experiment were compared, and an inventory of the final destiny of the injected methane was produced from modeling. In the simulation, depending on the electron temperature, between 53% and 68% of the methane injected through the target center returns to the target. Experimentally, much lower values are found in hydrogen plasma; this could be explained by the high chemical erosion of carbon.

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Plasma induced surface modification in the divertor strike point region of ASDEX Upgrade

P79A

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Tungsten (W) is a candidate material for the ITER divertor. In current fusion experiments it is used both as bulk material (TEXTOR) and for coatings (evaporated and plasma-sprayed) on carbon substrates (TEXTOR, ASDEX Upgrade and JT-60U) (see for example [1]). Tungsten-coated tiles are also foreseen for the divertor and shine-through protection plates of JET in the ITER-like Wall Project [2, 3]. In this context, a detailed understanding of the behaviour of W-coated plasma-facing components (PFCs) is essential for assessing the changes of materials during operation by the damage by plasma interactions and mixed material formation, related to tritium retention and components lifetime.

The present investigations were carried out on 13 samples from outer divertor Tile 1 of ASDEX Upgrade, with s coordinates from 1038 to 1231 mm, at different positions with respect to the strike point. This tile was installed in the machine for the whole 2009 campaign, during which three boronizations took place. The samples differed in received flux, which can be specified based on the s-coordinates. The tile was initially coated with a 10 µm W layer deposited on fine grain graphite with a 2-3 µm Mo interlayer. The aim of these post-mortem examinations was to describe material mixing and plasma-induced damage of tungsten coatings.

It has been generally observed that the surface morphology of the tungsten coating after the exposure shows a directional character of erosion and deposition. The deposits form mostly in the shadowed areas. Their thickness, according to FIB/STEM examinations varies from 200 nm to 1.5 µm, depending on the sample location and amount of received flux. The investigations revealed mostly W and W/O deposits. The structure of the re-deposited tungsten is highly porous and resembles a foam or sponge.

The XPS and AES examinations confirmed that the main constituents of the deposits are tungsten (40-60 at.%), oxygen (15-40at.%) and carbon (15-20at.%). Residues of iron were also observed. The amount and morphology of the boron deposition varied strongly with the position of the sample.

The roughness measurements confirmed surface smoothing caused by erosion.

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WEST (tungsten-W Environment in Steady-state Tokamaks), a project aiming at testing ITER Plasma Facing Components in Tore Supra

P79B

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The current ITER baseline foresees a full tungsten (W) divertor for the nuclear phase with deuterium and deuterium-tritium plasmas. However, the required high heat flux technology has never been tested in the demanding tokamak environment under the steady state heat fluxes (~10 MWm^{-2}) expected in ITER. In order to mitigate the risks for ITER, it is proposed to equip Tore Supra with a full W divertor, benefitting from its unique long pulse capabilities and the experience with actively cooled high heat flux components.

A wide range of plasma equilibria (from single to double null) will be possible while the new configuration will allow for H-mode access for several minutes at plasma current of ~0.8MA, providing relevant plasma conditions for plasma-facing component (PFC) technology validation. The lower divertor target design, the materials and the assembly technologies will be closely based on that currently envisaged for ITER divertor (W monoblock), while the pumping baffle and the upper divertor region will be equipped with CuCrZr copper/stainless steel heat sink technology with tungsten coating. Monitoring of the surface temperature of the divertor and of the RF antenna will be ensured using endoscopes and cameras coupled to automatic detection of thermal events for real time protection of PFCs.

Five years are expected to be required for manufacturing of the full toroidal W divertor elements. This industrial-scale manufacturing with ~ 500 actively cooled high heat flux components (i.e. total of 15,000 W monoblocks) and the associated quality assurance tests with a fully relevant ITER design, materials and technologies will consolidate the manufacturing process. In addition, this will offer a unique opportunity to anticipate any difficulties in terms of series fabrication and integration, but also to reduce the risks for the procurement of the second ITER divertor. The final divertor configuration will complement existing W divertor (e.g. JET and ASDEX-Upgrade) by adding the long pulse capability and actively cooled surfaces to the operational experience being gathered elsewhere. Extended plasma exposure provides access to ITER-critical issues such as PFC lifetime (melting, cracking etc.), tokamak operation on damaged metallic surfaces, real time heat flux control through PFC monitoring, fuel retention and dust production etc.

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A solid tungsten divertor for ASDEX Upgrade

P80A

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ASDEX Upgrade is a mid-size tokamak fusion experiment that was stepwise transformed from a carbon to a tungsten first wall experiment. Starting with the experimental campaign 2007 ASDEX Upgrade was operated as a full tungsten experiment. It could be shown that tungsten and ITER like plasma performance are compatible as long as the central heating of the plasma is high enough to suppress tungsten accumulation in the core[1].

The transition from carbon to tungsten was realized by coating fine grain graphite with W-PVD up to 4 µm thickness in the main chamber and with W-VPS of 200 µm thickness at the outer strike line module of the lower divertor where the highest heat load and highest erosion rate of up to 1 µm/campaign is expected. Unfortunately tungsten coatings have a lower heat receiving capability compared to carbon due to the thermal stress that is induced by high surface temperatures resulting accidentally in a delamination of the coating [2]. The thick coatings in the lower outer divertor were replaced in 2008 by 10 µm W-CMSII to reduce the effect of delamination onto the plasma but reducing the lifetime in the outer divertor.

The lower strike line module of ASDEX Upgrade will be replaced in 2012 by targets made from powder metallurgy (PM) tungsten to expand the operational space with a full tungsten first wall. This PM tungsten divertor Div-III overcomes the problem of delamination and/or erosion of tungsten coatings. In addition, it allows to investigate erosion and deuterium retention at solid tungsten.

The design of Div-III is compatible to the existing divertor support structure. The additional load due to the higher specific weight of the tungsten of about 0.5 kN/sector is small compared to the load caused by halo currents of about 50 kN sector. FEM modelling of the divertor support structure has shown that also for the worst case of 450 kA halo current at 3.5 T toroidal field strength the induced van Mieses stress is below 100 MPa. More critical are eddy currents induced by the temporal variation of the poloidal and radial magnetic field component during disruptions due to the higher electrical conductivity of tungsten compared to graphite. The need to keep the eddy currents and the mass low results in a sandwich design consisting of a 15 mm thick solid tungsten plate clamped together with a graphite interface plate of about 15 mm thickness to the cooling structure. Target tilting is realized by shaping the graphite interface plate rather than the tungsten plate that is flat to minimize the manufacturing effort and to maximize the flexibility of divertor shaping. Two sandwich targets are installed after a successful test of up to 30 MW/m^2 for 1.8 s in the high heat flux test facility GLADIS.

The paper will present the design criteria, the qualification procedure of Div-III as well as first experience from ASDEX Upgrade operation.

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P80B

Thermal stability of W/Mo JET divertor coatings on CFC substrate

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Thin tungsten coatings produced by combined magnetron sputtering and ion implantation (CMSII) with additional 3μm thick molybdenum interlayer deposited on carbon fibre reinforced carbon (CFC) substrates were selected as the first wall material for the divertor in the ITER-like Wall Project at JET. Elevated temperature in the divertor region during the operation time shall promote carbon diffusion from the substrate into the deposited layers. Due to the fact that both Mo and W easily form carbides, brittle carbide phases may be created at the interface, strongly affecting the thermomechanical performance of the coatings, once a critical carbide layer thickness is exceeded. This was experimentally confirmed by high heat flux testing of pre-annealed samples.

In the present study, a set of samples were annealed at different temperatures. The Focused Ion Beam (FIB) technique was used for both sample preparation for electron microscopy examination as well as first structural change investigation. Electron diffraction analysis performed at STEM microscope revealed the formation of both W_2C and WC carbides in the W layer, as well as Mo_2C carbide in the Mo layer. The thickness of carbide layers was measured using SEM/STEM microscopy, providing information about the kinetics of coatings degradation via transformation of metal layers into carbide phase.

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Development and Characterisation of Reference Tungsten / Tungsten alloys, and SiC_f/SiC for DEMO Fusion Applications

P81A

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Tungsten and Tungsten alloys, as well as SiC_f/SiC are considered as potential materials in the DEMO fusion reactor. The present work is a summary of the results obtained for these materials within the European Fusion Development Agreement (EFDA) work programme.

Tungsten (W) and W alloys are currently considered as an attractive solution for the helium cooled modular diverter components manufacturing in the future DEMO reactor because of their strength at high temperatures, good thermal conductivity, and low sputter rates. Nevertheless, lots of effort should be done towards improving the W toughness at low temperature range. This work gives a glance at the first results obtained on pure W and W alloys (ODS W) produced by Metal Injection Moulding (MIM) technique at ECN, The Netherlands. Mechanical and physical material characterization together with detailed microstructural examination were performed so far and the results are reported hereafter. Additionally, the conducted research aims to propose a fabrication technology and to demonstrate its potential for mass production of components.

SiC_f/SiC is identified as a potential structural material due to its corrosion resistance, low neutron activation, chemical stability, attractive mechanical properties up to 1100°C, reasonably good thermal conductivity, and last but not least, SiC composite is the only non-magnetic material among the candidates. Despite its promising properties, there are issues to be addressed under intensive neutron irradiation, such as: thermal conductivity decrease, swelling and mechanical properties degradation. The work addresses mechanical and physical properties characterisation of new unirradiated 2D and 3D Tyranno SA 3 fiber reinforced Silicon Carbide composites (SiC_f/SiC), known as EU reference material. An innovative tensile set-up was developed at NRG to allow SiC_f/SiC tensile testing in compression. Additionally, the conducted research serves as a pre-irradiation investigation and to qualify test procedures to be used for post-irradiation examination.

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Interface engineering of the Tungsten-Fibre-Reinforced Tungsten composites

P81B

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Enhancing the properties of tungsten materials for their practical use in the divertor of thermonuclear fusion reactors is very challenging. There is an urgent need to explore novel toughening mechanisms for tungsten to retain its toughness even under conditions of embrittlement.

In this work, such a toughening method for tungsten is proposed based on the reinforcement of tungsten fibers (W_f/W_m composites) and engineered interfaces. The underlying toughening mechanism is analogous to that of a fiber-reinforced ceramic matrix composite (FCMC), which relies on energy dissipation by controlled debonding and friction at the fiber/matrix interfaces. The fracture properties of the engineered interface are the key factors controlling the overall composite toughness. In this work, intensive analysis has been performed on the fracture behavior of various engineered interfaces of W_f/W_m composites for exploring the feasibility of producing a toughened W_f/W_m composite.

Commercial tungsten wires (fibers) were selected as reinforcement, while dense tungsten was chosen as the matrix. Based on the brittle zirconia and erbia, the ductile copper, the lubricating carbon, and their combinations, 14 types of coatings were used as interfaces. The interfacial parameters were calibrated by means of a fiber push-out test on a single-fiber W_f/W_m composite. The results showed that the interfacial fracture energy of employed interfaces satisfied the fracture criteria for the crack deflection. Microscopic analysis of the interface structures was carried out before and after the push-out test, indicating the interfacial debonding location were in accordance with the interfacial parameter calibration results. The interfacial crack deflection was directly demonstrated by a three-point bending (3PB) test. Mechanical property prediction of the W_f/W_m composites with multiple fibers indicated that the stress-strain curves of the involved W_f/W_m composites were of typical 'tough' material type, which agreed well with the interfacial parameter calibration results and the interfacial crack deflection demonstration results, supporting the primary motivation of this work.

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Synthesis of tungsten-fibre reinforced tungsten composites and in situ observation of cracking by high energy synchrotron tomography

P82A

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The inherent brittleness of tungsten below the ductile-to-brittle transition temperature strongly restricts its use as plasma facing material of fusion reactors. Metallurgical efforts to enhance the toughness (e.g. grain refinement, alloying) still face stability problems during operation. An alternative approach for enhancing the damage tolerance is the application of the so called pseudo toughness mechanism. Tungsten is reinforced by tungsten fibres (W_f/W_m-composites) which enable controlled crack deflection at engineered fibre/matrix interfaces. This leads to internal energy dissipation by interface debonding and friction. Local stress can be reduced and thus the global fracture toughness is improved [1, 2]. As this is a purely mechanical effect there is reasonable hope that the toughness will withstand material embrittlement.

Conventional processing routes of tungsten (e.g. powder metallurgy) were not applicable for W_f/W_m-fibre-composites. This resulted in the development of the chemical vapour infiltration process of tungsten (WCVI). This technique combines the chemical vapour deposition of tungsten with an additional infiltration step. First Drawn tungsten wire is arranged in a regular pattern. Subsequent tungsten hexafluoride (WF_6) is infiltrated and by its decomposition the tungsten matrix is formed. The advantages of this technique are low substrate temperature and a negligible mechanical impact which are essential for maintaining the fibre pattern. By combining controlled gas flow with optional temperature distribution it was possible to produce samples with a density up to 92%. Additional redensification steps either by repeated WCVI or hot pressing led to further density improvements up to 99%.

The key factor for the toughness improvement in W_f/W_m-composite is the interaction of a possible crack with the microstructure (fibre, matrix, and interface). For an in-situ observation of this effect high energy synchrotron tomography is the most adequate technique. Therefore a possibility of tungsten tomography up to a diameter of 1 mm was established and the crack formation and propagation was studied during mechanical testing. The experiments were conducted at the European Synchrotron Radiation Facility (ESRF) in Grenoble, France. The interaction between crack and microstructure was clearly detectable. Crack deflection at the interface was observed both for single-fibre and multi fibre samples. Hence the theoretical model of crack deflection is validated.

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P82B

Production of copper-μdiamond composites for first wall heat sinks

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Copper alloys have been selected as first wall heat sinks due to their favorable thermal conductivity and radiation resistance [1]. However, the demand for operation temperatures above the range proposed for ITER first wall (<300°C) poses extra challenges, especially regarding thermal conductivity and mechanical strength [**Fehler! Textmarke nicht definiert.**]. The extremely high thermal conductivity of diamond turns its dispersions into excellent candidates for thermal management applications. Additionally, particle dispersions can be used as reinforcement for increased strength and, furthermore, Cu-Diamond composites with enhanced thermal conductivity will also exhibit lower thermal expansion [2,3] mismatch with plasma facing W-based materials than copper alloys. Natural diamond is known to have the highest thermal conductivity, 2000 W/(m.K), which compares with 390 W/(m.K) at 20°C for copper [4]). In the present work, natural micro diamond (μD) has been selected to reinforce copper in composites produced by mechanical alloying due to its strong resistance to amorphization and graphitization as compared, for example, with nanodiamond. Moreover, phonon scattering in diamond occurs for submicrometer crystallite size with a concomitant thermal conductivity attenuation. On the other hand, electrons dominate heat conduction in copper, whereas phonons control it in diamond. Hence, composite heat conduction requires energy transfer between electrons and phonons at the interfaces, which critically affects the material thermal behavior. The aim of the present study is to develop Cu/μD composites for heat-sinks integrated in first wall panels of nuclear fusion reactors. Powder mixtures of Cu and μD have been mechanically alloyed, and the reinforcement dispersion was monitored for different milling times by X-ray diffraction and electron microscopy. The load transfer at the interfaces was evaluated by microhardness measurements, allowing inferring the quality of the interfaces and its effect on the thermal properties of the material.

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Thermal stresses and fatigue damage in monofilament reinforced copper composites for heat sink applications in divertor elements

P83A

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Monofilament reinforced Cu composites (MFRM) are developed as heat sink materials for divertor elements where a plasma facing W plate is attached to a CuCrZr heat sink. Under pulsed operation, required for the Tokamak reactor type, high thermal mismatch stresses between W (~ 5 ppm/K) and Cu (~ 16 ppm/K) lead to delamination at the interface. SiC fiber and W wire reinforced copper composites are applied as interlayer material, which combine the high thermal conductivity of the matrix with low thermal expansion of the fibers. With this approach the thermal mismatch stresses are transferred from the macroscopic W-Cu interface of the component into the composite interlayer. Thermal micros stresses between the matrix metal and the stiff fibers in the MFRM are evaluated and their effect on the long term stability under service conditions is studied.

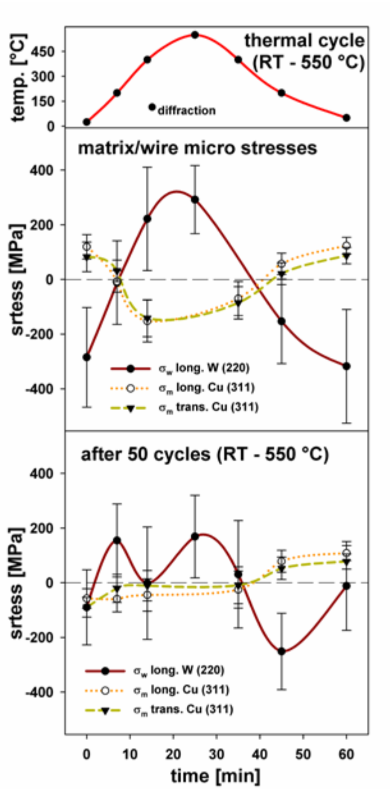

Fig. 1.: Neutron diffraction:
[2] Thermal stresses in Cu/W/20m during thermal cycling (R–550°C) before and after 50 ex-situ cycles, HZB, Wannsee, Germany.

In-situ neutron and synchrotron diffraction were applied to measure the micro stresses in SiC fiber and W wire reinforced Cu. Different interface bonding qualities were investigated during thermal cycling [1]. The thermal stresses were measured in-situ during thermal cycling as well as after further ex-situ cycles (Fig. 1). Complementary in/ex-situ synchrotron tomography revealed information on fatigue damage mechanism and propagation (Fig .2).

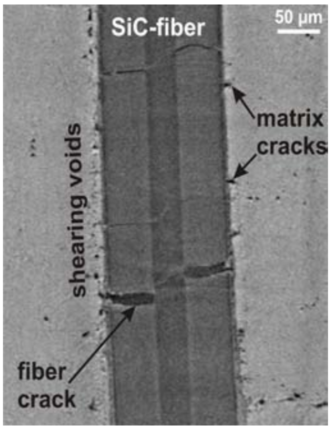

Fig. 2.: Synchrotron tomography: [3] Thermal fatigue damage in Cu/SiC/20m after 50 ex-situ cycles (RT–550 °C), ESRF, Grenoble, France.

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P83B

Sigma fibre reinforced copper for heat sink application

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SiC fibre reinforced copper shows a great potential as heat sink material in the divertor of future fusion reactors, where heat loads reaches up to 15 MW/m². This leads to an interface temperature between tungsten as plasma facing material (PFM) and CuCrZr as heat sink material of up to 550°C. This temperature exceeds the maximum operation temperature of the CuCrZr alloy. SiC fibre reinforced copper matrix composite (CuMMC) can be used between the PFM and the CuCrZr alloy where it combines high thermal conductivity (200 W/m*K) with a sufficient mechanical strength (300 MPa) at elevated temperatures up to 550 °C.

Sigma SM1140+ fibres (TISICS Ltd) were coated by magnetron sputtering with a titanium interlayer for better fibre/matrix bonding. The copper matrix was deposited by electroplating, where the deposition time defines the fibre volume fraction. The coated fibres were heat treated with a slow heat rate of 0.5 K/min at 550°C with a dwell of 2 h for hydrogen degassing. Finally the composite was consolidated by hot–isostatic pressing.
The ultimate tensile strength and also the Young's Modulus of the SiC fibres were determined by single fibre tensile tests. The uncoated fibres exhibited an average strength of 3200 MPa and a Young's Modulus of around 370 MPa. The tests were repeated with coated and heat treated fibres to study the influence of interlayer to reinforcement. Push-out tests verified the sufficient fibre/matrix bonding.

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Microstructural Characterization of SiC Ceramic Multilayers Using X-ray Laboratory Microtomography

P84A

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SiC based materials offer attractive properties for future fusion power plants. The spatial distribution of porosity and density of these materials has to be determined for assessing their performance at high temperatures and/or intense irradiation fields. Further, SiC with controlled distribution of porosity and density is a promising concept for high thermal load applications as this offers directional control of the thermal conductivity.
In the current work we show that X-ray laboratory microtomography can be used to determine simultaneously the local porosity and density of SiC ceramic multilayers, fabricated by tape casting, de-binding and sintering. Different architectures layers have been examined to establish relationships between manufacturing parameters and material microstructure/properties. The method is nondestructive and accurate down to the resolution of the instrument (~2 µm) and it provides detailed morphological information such as pore shape, spatial distribution, and connectivity. Relevant defects like cracks, voids, inclusions or delaminations can be quantified and correlated with different manufacturing phases. By comparison with global methods of porosity and density measurement and with direct observation with electron microscopy one get substantial new information and results gain in confidence.

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Computed Tomography Inspection of the CFC Monoblock section of ITER Inner Vertical Target Prototypes after Production, after High Heat Flux Testing and after Critical Heat Flux Testing

P84B

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One of the technically most challenging components for the ITER (International Thermonuclear Experimental Reactor) machine is the divertor. Its main function is to extract the power which has been directed into the scrape-off layer of the plasma while at the same time maintaining plasma purity. The Inner Vertical Target (IVT) as part of the divertor comprises plasma facing materials which are either carbon fibre reinforced carbon composites (CFC) or tungsten. The ITER relevant design is a full monoblock armouring being actively cooled during operation by a heat sink tube made of a precipitation hardened CuCrZr alloy. The tungsten monoblocks are joined to the cooling tube by copper casting and hot isostatic pressing, while for the CFC monoblocks active metal casting and hot isostatic pressing are applied.

This paper focuses on the straight CFC section of IVT prototype components that have been high heat flux tested at 20 MW/m² for 1000 cycles, followed by critical heat flux testing. In order to investigate defects as well as defect creation and evolution in the materials and their interfaces, computed tomography (CT) inspection was applied. This non-destructive test method enables precise determination of position and size of defects and structures in the CFC and at the CFC/Cu interface. CT was performed at three different stages of a component life time, namely after production, after high heat flux test and after critical heat flux test. The results of these investigations are comparatively presented and discussed.

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R&D activities for the production of 1,0 mm thick molybdenum armour layer on copper substrates

P85A

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In the framework of the activities for the development of the Neutral Beam Injector (NBI) for ITER, the design of the Radio-Frequency plasma source has been carried out. The most critical components of the plasma source are the rear vertical plates facing the plasma, since they are hit by the back-streaming positive ions that are generated, mostly for stripping losses, inside the 1 MV electrostatic accelerator. Such high energetic particles, impinging the rear vertical surfaces of the plasma source, cause heat deposition and physical sputtering. As consequence, the need of an armour layer having low sputtering yield was established to be necessary. Molybdenum and tungsten are the most suitable materials; nevertheless the required thickness is around 0.5 mm.
Different technologies for the manufacturing of such a thick armour layer, that is not common, have been investigated and samples have been manufactured by explosion bonding and atmospheric plasma spraying. Samples have then been tested: microscopic, outgassing, delamination, thermal shock, and thermal fatigue analyses have been carried out. The results of tests performed on the explosion bonded samples are presented in the paper, giving an overview on the critical technological aspects and open issues.

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Centerstack PFCs for NSTX Upgrades

P85B

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The National Spherical Torus Experiment (NSTX) is a low aspect ratio, spherical torus (ST) configuration device which is located at Princeton Plasma Physics Laboratory (PPPL). This device is presently being updated to enhance its physics by doubling the TF field to 1 Tesla and increasing the plasma current to 2 Mega-amperes. The heart of the upgrade involves a new Centerstack Assembly (CSA). The CSA consists of the inner legs of the Toroidal Field (TF) windings, the Ohmic Heating (OH) solenoid, three pair of inner Poloidal Field (PF) coils, thermal insulation, diagnostics and an Inconel casing which forms the inner wall of the vacuum vessel boundary. The outside surface of the Inconel casing is protected from the heat loads by a layer of carbon fiber tiles

The upgrade of the Centerstack Assembly calls for an increase in the casing diameter which required the replacing of all the carbon-based Plasma-Facing Components (PFCs). Additionally, due to other upgrades to the operating capacity of the device, the PFCs needed to be improved to withstand higher operating heat fluxes and disruption forces. A combination of 2D and 3D carbon fiber composites have been proposed to provide adequate thermal shielding as well as the sufficient mechanical properties to tolerate higher, thermally-induced stresses within the tiles. The attachment scheme for this upgrade presents several changes from the original design: replacing the weld stud with a weld nut, changing the soft joints to a hard design in order to facilitate the implementation of the Spiralock thread technology, and finally, the removal of Grafoil from the system in order to thermally isolate the Centerstack Casing.

This paper will describe the details of the design features of the PFCs and elaborate on the major changes from the original layout, emphasizing the challenges of adapting the present PFC configuration to a higher operating platform. (Note: Upgrade work supported by U.S. DOE Contract No. DE-AC02-09-CH11466)

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NSTX Vacuum Vessel Protection Armour for Neutral Beam Injection Heating

P86A

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In support of a planned upgrade for the National Spherical Torus Experiment (NSTX) a second large Neutral Beam Injector (NBI) is being added to the experiment. Each of the two NBI systems can deliver approximately 7.5 MW of heating power to the plasma. After transiting the main torus vacuum vessel, neutral beams from the sources of both of these devices land on plasma facing plates currently armoured with ATJ™ grade isotropic graphite. Due to the additional beam line this armour must be repositioned, reattached, and upgraded to protect the vacuum vessel wall from incident heating from neutral beam shine through, or in the case of an off-normal fault, absorb the full energy of the NBI. Interlocks are designed to prevent the firing of the NBI into the vessel without a plasma; however, the armour is the final barrier to damage to the experiment. This paper will detail the redesign and analysis of the armour, attachment, and services.

Early on in the design the decision was made to consolidate the footprint of both NBI systems on the same area of armour in order to preserve space at the mid-plane of the torus for plasma diagnostics. This presents the problem of twice the neutral beam energy input to the same area. Complicating this further, the geometry of desired tangency radii and a fixed 4 degree spread in the NBI sources create areas of overlap where two high heat flux zones of up to 2.8kW/cm^2 from each beam source intersect. Various operational scenarios and carbon reinforced and graphite materials have been analyzed to qualify the new armour designs for the higher heat loads. Additionally, the overall attachment of the armour was redesigned to accommodate new disruptive loads and provide for easier installation.

Work Supported by United States Department of Energy Contract No. DE-AC02-09-CH11466

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Residual Stress Measurements of a CuCrZr Tube Brazed to a Tungsten Tile Using Neutron Diffraction

P86B

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In plasma facing components for nuclear fusion reactors tungsten or carbon based tiles need to be cooled through a heat sink. The joint between the PFC and the heat sink can be realized using a brazing process. The main problem of bonding tungsten or carbon based material to CuCrZr is the large difference in their coefficient of thermal expansion (CTE). This difference creates very large residual stresses at the interface and may result in damage of the joint simply from cooling it from the joining temperature. A low yield strength material compliant layer, like pure copper, can be introduced between the protective material and CuCrZr in order to lower the residual stresses by plastic deformation. Another solution is to introduce a high yield strength material interlayer with CTE matching the protective material, like molybdenum, in order to transfer the residual stresses from the carbon/metal interface to the strongest metal/metal one. Experimental verification of the induced stresses during the brazing process is of vital importance.
In the present work neutron diffraction measurements have been employed in order to measure the strains and residual stresses in a CuCrZr tube brazed to tungsten tile, in the side of the tungsten tile. Three geometries have been used in order to measure the axial, radial and tangential strains, at various distances from the weld. Also, neutron radiography measurements were performed in order to assess the integrity of the brazing process.

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Advanced X-ray Imaging of Metal Coated/Impregnated Plasma-Facing Composite Materials

P87A

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Due to their excellent thermo-mechanical properties, Carbon Fiber Composites (CFCs) are one of the candidate armour materials for ITER's plasma facing components. The heat flux received by these components is intense and requires active water cooling. This is achieved through a welding (active metal casting, AMC) between the metallic water loop and the CFC material. The CFC porosity plays a major role in both fabrication and operating of PFCs. Firstly, the quality of the AMC of the monoblock geometry developed for ITER depends on the metal impregnation inside CFC. Secondly, a part of the fuel retention is due to the codeposition mechanism inside CFC porosity.

The present work describes analysis results of relevant PFC materials coated with refractory metals (W, Mo) and/or welded on heat sink structures (Cu, CuCrZr). The experimental characterization of macroporosity is performed using an X-ray micro-tomography technique. Numerical 3D images are used to describe the distribution of macroporosity with respect to the position of the carbon fibre bundles and its correlation with the impregnation pattern of the heat sink material. The identification of the material composition was certified by X-ray fluorescence analysis.

Tungsten erosion and redeposition are of great interest, because a full tungsten divertor is foreseen to be used during the deuterium-tritium operational phase of ITER. Upgrade and work is currently in progress to completely replace the existing JET CFC tiles with tungsten-coated tiles within the JET ITER-like wall project. The need for a fast and nondestructive method which allows the quantitative determination of the thickness of a tungsten coating on a carbon material on large areas led us to develop a combined absorption/fluorescence X-ray technique. The method provides fast analysis, high spatial resolution and a material selective detection of deposited layers and intrusions. It was applied at the post-mortem analysis on W coated fine graphite tiles from the divertor of ASDEX-Upgrade and on JET ITER-like wall samples.
The combination of these X-ray Imaging techniques can be used for the quality control monitoring of the new CFC ITER reference materials.

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Modification of the structure and optical characteristics of the surface of tungsten mirrors under bombardment with 38 eV deuterium ions

P87B

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The results of temperature effects on modification of surface morphology and optical properties of tungsten mirror samples exposed to low energy deuterium ions (38 eV) up to ion fluence 10^{26} D/m^2 are presented. It was found for samples recrystallized at 2073 K in hydrogen atmosphere after they were cut and polished, that the surface state weakly depends on the exposure temperature in the range 320-695 K with the exception of a rather narrow region around 535 K, where drastic changes of all optical characteristics occur, according to optical reflectometry and ellipsometry data; this range of exposure temperature is characterized by strong blistering [1].

It was found that the reflectance obtained in direct measurements at normal incidence (optical reflectometry) drops for the W samples exposed at 535 K, whereas the estimations of reflectance using the ellipsometry data demonstrate some increase. On our point, the reason of this difference is that both methods, reflectometry and ellipsometry, are based on different physical effects. In reflectometry of specular reflection, the full energy specularly reflected from the sample is measured, thus the surface defects result in an increase of the diffusive component and, correspondingly, to a decrease of the specular reflectance. Ellipsometry is based on investigation of the change of the state of polarization of the specular component only and, therefore, it provides only information on the specularly reflected fraction of the light, without taking into account the fraction scattered by the blisters.

In the case of the strongly blistered surface of the specimen exposed at 535 K, ellipsometry yields information about those parts of the surface that still remain free from blisters. Thus the strong modification of the ellipsometric characteristics means significant modification of the electronic structure for this particular specimen as distinct from those exposed at other temperatures.

Concluding, on the surface of the specimen exposed to D^+ ions at 535 K two processes are realized: appearance of blisters and some modification of the electronic structure of a near-surface layer.

It is worthy to note that similar behavior was not observed for mirror samples fabricated from polycrystallized ITER-grade tungsten. The reason of this difference will be discussed in the presentation.

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Simulation of first mirror surface composition change under ITER relevant irradiation

P88A

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ITER optical and laser diagnostic systems will implement metallic mirrors as plasma-viewing elements. The charge-exchange neutrals irradiation and impurity transport can lead to the significant decrease in performance of mirrors due to the erosion of their surfaces and changing of its composition profile. The optical properties degradation of diagnostic mirrors will directly affect the signal quality and the efficiency of the respective ITER diagnostics. First mirror material choice and impact of erosion and deposition on their performance is intensively studied experimentally and numerically [1-3]. However, these numeric simulations do not include the relevant dynamic surface composition change under irradiation, which is important for estimation of surface erosion.

In the current work TRIDYN-like code SCATTER is used for simulations of metallic mirror surface composition modification under irradiation with ITER relevant fluxes of CX-neutrals and ions [4]. The upper and midplane port location of the first mirror is considered. Carbon and beryllium impurities deposition is also included in the model. Comparison between the areas with prevailing erosion and deposition is presented as well as the doze dependencies of the surface composition profiles.

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Passive protection of the ITER first mirrors

P88B

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First mirrors which face the plasma directly will be used in ITER to redirect light to the protected optical diagnostic instruments. The mirrors reflectivity and thus diagnostic performance can severely deteriorate because of erosion due to fast particles and deposition of impurities [1]. Numerical simulations [2] have shown that the first mirror erosion can be made acceptably low if the mirrors are installed sufficiently far behind the first wall. However, the tolerable deposit thickness of 10 nm/a which would not lead to serious loss of reflectivity can be exceeded even in the long ducts.

Recent experimental results at LHD [3] have indicated significant reduction of the deposit thickness on samples protected in cylinders with fins. In the present paper the efficiency of such fin structures for protecting first mirrors in ITER is analyzed.

The same model as in [2] is used to calculate the energy and angular distribution of the incident particles at the entrance aperture of the diagnostic duct. The model for impurity atom transport in the duct has been refined. In [2] an assumption of the 100 % reflection from the duct side walls was applied to account also for re-erosion. In the present model a reflection probability $R_N<1$ is used but re-erosion of the deposit from the duct wall due to fast atoms is modeled explicitly.

For the simple cylindrical tubes even in the case of the length (L) to diameter (D) ratio L/D=30 and $R_N=0.9$ the model yields less than a factor of 10^3 reduction of the impurity (Be) flux at the mirror compared to that at the entrance aperture (in the ITER main chamber). For the tube with the same optical aperture and L/D, but equipped with periodic fins (ring-shaped diaphragms) more than a factor of 10^4 impurity flux reduction is obtained (for the same R_N). Such an efficient attenuation would mean that the mirror deposit thickness could be kept below 10 nm after several hundreds ITER shots even under most pessimistic assumptions that aperture impurity fluxes reach values of 10^{20} m^{-2}/s, as indeed observed in some experiments [4].

The much stronger attenuation of the incident impurity flux in ducts with fins compared to simple ducts obtained in the present simulations backs observations of [3]. The possibility of the mirror self-cleaning in short ducts due to re-erosion of deposits, as observed experimentally [4], will be also addressed in the paper.

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Experimental modeling of film growth on diagnostic mirrors in ITER-relevant conditions

P89A

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Most of the optical diagnostics in ITER contain mirrors for light transfer from plasma to detectors. Estimations show that the first wall at the upper and equatorial diagnostic ports will be subjected to hydrogen flux of ~10^{16} cm^{-2}s^{-1} and carbon and beryllium fluxes of ~10^{14} cm^{-2}s^{-1} [1]. Therefore there is a risk of deposition of CH films with Be inclusions on the first diagnostic mirror looking to plasma that can significantly reduce the reflectivity of the first mirror [2, 3]. Estimation of the mirror's lifetime requires experimental modeling of deposition process in ITER-relevant conditions including X-ray irradiation and mirror's temperature elevation.

Such conditions have been simulated in two magnetron sputtering devices. In the first one (Kurchatov), formation of soft CH films under X-ray irradiation of 0.1 Gy/s was investigated. Mirrors made of polycrystalline Mo and stainless steel SS-316 were exposed to carbon flux about $4\cdot10^{14}$ cm^{-2}s^{-1} for 2 hours. The expositions were accompanied with mirrors heating to ITER-relevant temperatures, which are expected to be within 100-150˚C in the operational mode. Under gas pressure of 18 Pa with X-rays on CH film growth was observed at the temperatures below 140˚C resulting to degradation of mirror's reflectivity by 7-30%. Though heating the mirror to the temperature over the threshold (140˚C) prevented film growth, the reflectivity of the mirror degraded by 7%. Turning off X-ray irradiation reduced the temperature threshold down to 120˚C. Lowering gas pressure to 6 Pa raised the threshold of films growth up to 170˚C and no influence of X-rays was observed in this case.

Another device (BMSTU) was used to examine the growth of metalized CH films on the mirrors of the same type under gas pressure of 0.4 - 1 Pa and particle flux up to $4\cdot10^{14}$ cm^{-2}s^{-1}. In experiments toxic Be was replaced by Al due to similar chemical properties. Exposition for 2 hours at the temperature below 170˚C leads to a thick film growth that decreased mirror's reflectance by 15-40%. Heating the mirror reduced the rate of film growth but did not prevent it and degradation of mirrors reflection was about 10%.

The experiments show that X-ray irradiation can stimulate film growth though its influence becomes insignificant at low pressure expected in diagnostic ports. Combined C, H and Al flux leads to formation of metalized CH films that cannot be prevented by temperature elevation. Additional methods of mirrors protection and cleaning must be developed to achieve acceptable lifetime of the first mirrors in ITER.

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P89B

First diagnostic mirror in ITER designed for in-situ plasma cleaning treatment

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The maintenance of the first optics operability is one of the most challenging issues in designing optical diagnostics for ITER. In the areas featuring deposition-dominated conditions the major threat stems from the intensive contamination with products of the erosion of first-wall elements and divertor tiles. The problem is of special concern in the divertor area and particularly in the ports used for divertor plasma diagnosing. To prevent the deposition-induced degradation of the first collecting mirror of the divertor Thomson Scattering system we suggest complex approach including the choice of proper mirror design [1] and special cleaning discharge in immediate proximity of mirror surface [2]. The implementation of cleaning discharge in ITER and the requirements to the plasma to be used for mirror cleaning from hydrocarbon deposits, are discussed along with our recent results on testing mirrors prototypes under ITER relevant conditions.

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Deviated Reflectivity Spectrum of Molybdenum upon Low Temperature Deuterium Plasma Exposures

P90A

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Metallic mirrors are foreseen to play a crucial role for all optical diagnostics in ITER. First mirrors have to maintain a good reflectivity both in erosion and deposition zones in the harsh ITER environment of charge exchange neutrals, UV and X-ray radiation. Molybdenum is one of the most important candidates for the first mirrors, due to its low sputtering yield under deuterium exposure. Molybdenum mirrors exposed to low temperature (4-5 eV) deuterium plasma exhibit reflectivity spectra different from that of bulk molybdenum. This difference is both due to implanted deuterium acting as voids in the metal and enhanced surface scattering due to point defects on the metal surface. The results presented show that these reflectivity changes are similar for single- and nanocrystalline molybdenum mirrors [1]. Moreover, exposure of magnetron sputtered nanocrystalline molybdenum films to deuterium plasma revealed that after a certain deviation in the spectrum has been reached, the reflectivity remains constant upon further exposure. Exposures were carried out in a range of fluences between 1.2×10^{19} to 6.5×10^{20} ions/cm^2 corresponding to up to 385 ITER discharges in equatorial ports and 11 discharges in the upper ports in the diagnostic ducts close to first wall. Constant conditions of -200 V bias and 150 ºC temperature were maintained on the sample. Further exposures performed in tokamaks [2] or using an ion gun for higher flux result in reflectivity changes that are comparable to those obtained with deuterium plasma exposure. No mechanical damage, such as blistering and increase in roughness, are observed on the coated molybdenum films upon any of the mentioned exposures. Deuterium (200 eV) is calculated to have a 0.00047 atoms/ion sputtering yield on coated nanocrystalline molybdenum films in comparison to the theoretical one: 0.001 atoms/ion.

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Active control over carbon deposition by gas feeding for protection of diagnostic mirrors in ITER

P90B

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Future fusion devices such as ITER will require accurate and reliable measurements of a wide range of plasma parameters necessary for the machine protection, basic plasma control and envisaged research program [1]. ITER will be equipped with an extensive set of spectroscopic and laser diagnostics using metallic mirrors as first plasma-viewing elements. Erosion of the mirror surface and deposition of impurities will change drastically optical properties of mirrors leading to an immediate impact on the quality and reliability of detected signals and on a long-term performance of respective diagnostics. The lifetime of mirrors becomes a critical issue for ITER operation [2].

A gas feeding in the vicinity of mirrors represents a promising technique for mitigation of impurity deposition. Series of experiments were performed in the TEXTOR tokamak with a prototype of a diagnostic duct (the so-called periscope system) equipped with molybdenum mirrors [3]. After the first exposure deposits having 400 nm in thickness were revealed at the surface of the first mirror. For the newest experiments a half of each mirror was pre-coated with a 60 nm-thick amorphous carbon film (a-C:D). The periscope was exposed in the scrape-off layer plasma of TEXTOR under deposition-dominated conditions. Chemically reactive deuterium and non-reactive helium gases were fed in the vicinity of mirrors at comparable rates to overbalance the incoming adverse particle flux. The aim of this study was to determine the relative contribution of physical and chemical erosion in deposition mitigation. In recent experiments mirrors were exposed at elevated temperature about 460 °C.

Exposure with deuterium feeding was performed to enhance the chemical erosion of a-C:D layers by deuterium atoms and ions. After exposure, a full suppression of deposition on the first mirror and even complete removal of the pre-deposited a-C:D film were detected. The reflectivity of the mirror was restored. In case of exposure with helium feeding, deposition on the first mirror surface was significantly suppressed but still not sufficiently enough to protect the mirror completely. A part of the mirror surface was covered with an 11 nm thick re-deposited layer causing the decrease of the reflectivity.

In this contribution, experimental data will be presented along with results of modeling of plasma-gas interaction inside the periscope system. The applicability of gas feeding technique for the mitigation of deposition at ITER diagnostic mirrors will be addressed.

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Nanostructured Rhodium Films for Advanced Diagnostic Mirrors Produced by Pulsed Laser Deposition

P91A

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The fabrication of nanostructured/amorphous metallic films with structural properties different from the corresponding crystalline materials represents a research area of continuously increasing interest [1,2]. In particular the challenge of producing metallic films with high reflectivity on a wide wavelength range (200-1000 nm) and improved thermo-mechanical and resistance properties needs to be faced for the fabrication of first mirrors working in many diagnostic systems of thermonuclear magnetic fusion machines.
Here we present a mechanical, annealing and thermal fatigue investigation made on Rhodium mirrors produced by Pulsed Laser Deposition (PLD). We already explored the possibilities offered by PLD for the production of Rhodium films with controlled nanostructure [3]. The produced mirrors (characterized by SEM, XRD, and profilometry) are few microns thick Rh films, deposited on, Mo substrate, having diameter of 18 mm and proper planarity (λ/2 at 200 nm) and roughness. We deposited films with different structures, both crystalline and amorphous-like. In addition we also explored the deposition of multilayer systems and functionally graded (i.e. continuous variation from crystalline to amorphous) films. Important properties of the mirrors have been tested, namely: mechanical and adhesion, with scratch and micro indentation tests, thermal fatigue using a 532 nm pulsed laser onto the Rh films and annealing at 200-400 °C. Mirror optical properties have been characterized by ellipsometry. In addition exposure tests are foreseen (e.g. FTU and IFP-CNR, Milan).

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Fracture mechanics approach to Be/bronze joint structural assessment

P91B

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Approach to a structural assessment of the joint of a Be tile and bronze sink of the ITER FW is proposed. This approach is based on the linear elastic fracture mechanics that is similar to methods used for structural assessments of components containing cracks. It combines the following steps: (1) determination of analytical asymptotic formulas valid in the singularity zone and (2) numerical simulation using the finite element method to calculate the stress state in the vicinity of the singularity zone. At the analytical step, asymptotic formulas are obtained for the geometry in question. These formulas establish relationship between stresses (σ) and stress intensity factor (*K*) in the singularity zone of the joint. It is assumed that stresses in the singularity zone depend on the single stress intensity factor *K* which is an unknown value. Asymptotic formulas provide the mode of dependence between σ and *K*. Geometry of the joint, properties of used materials and boundary conditions are the input data for this step. The numerical simulation step is used for determination of *K* using calculated stress distribution σ(*r*) for the given load; *r* is the distance from singularity point.
Proposed approach can be used for: (1) qualitative comparative analysis of the different joints from the static strength point of view, (2) analysis of joints with defects, (3) establishing the equivalent loading states for the joint (in geometrical/structural singularity) for experimental testing of various geometries.

Two variants of the ITER FW mock-ups with different Be armour layouts have been assessed using the proposed approach. The experiments are planning and intended to obtain the values of stress intensity factor at joint's failure. These experimental data will be used for validation of the proposed approach.

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Assessment of CFC grades under thermal fatigue for the ITER inner vertical target

P92A

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The ITER divertor system is aimed at controlling the plasma density, exhausting the alpha particles and reducing the impurity content of the plasma. It consists essentially of two parts: a massive support structure called cassette body and the Plasma Facing Components (PFC). The cassette body is aimed at supporting the PFC, routing the water coolant into the PFC and providing neutron shielding whilst the divertor is composed of 54 cassettes placed in a circular array. PFCs are actively cooled thermal shields devoted to sustain the heat and particle fluxes under operational conditions in the range of 10-20 MW/m^2. PFCs of the divertor are the dome, the particle reflector plates, the outer vertical target and the Inner Vertical Target (IVT) constituted of units. European agency is in charge of the fabrication of the IVT. Each unit of the IVT is composed of W monoblocks (baffle region) and CFC monoblocks (strike point region). Thermal fatigue is one of the most important damaging mechanisms for these PFCs due to the high number of operating cycles (several thousands) and to the expected surface heat loads. ITER requires that CFC monoblocks of the first IVT set, have to sustain 10 MW/m^2 in steady state for 3000 pulses and 20 MW/m^2 in quasi stationary condition (10 seconds) for 300 pulses. At the moment, CFC grade choice has not been yet decided. In this frame, assessment of different CFC grades under thermal fatigue was launched by European agency.

To assess thermal fatigue behaviour of European CFC grades, monoblock small-scale components with different CFC grades were delivered to CEA/IRFM by two European manufacturers (PLANSEE and ANSALDO Ricerche). CFC grades were fabricated by two suppliers SNECMA (NB41, NB31 and N11 CFC grades) and DUNLOP (DMS780, 3D and Megaggard CFC grades). Quality of the assembly between CFC and Cu was assessed with an Infrared Thermography non-destructive technique (SATIR test bed located in CEA Cadarache). After the initial examination, components were tested by means of the electron beam facility FE200 (European facility located at Le Creusot, France) with heat fluxes in steady state conditions in 5-20 MW/m^2 range. The impact of thermal fatigue effect on component lifetime is detailed in this study. In addition, the reported analysis relies on calculations based on the FEM simulations and on the comparison with results obtained with non destructive examination (IR thermography) undergone before, and after the thermal tests.

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Mechanical characterization of EUROFER97 in tension at high strain rate

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In a real fusion reactor, plasma disruptions are expected to occur that will yield disruption stress peaking in about 1ms: that represents the typical loading rate of dynamical tests. Thus, up to now, not enough attention has been paid to characterize both the dynamic constitutive behaviour and dynamic fracture toughness behaviour of the tempered martensitic steels. As a first step to fill that gap, this study has been undertaken to investigate the tensile properties, yield stress and strain-hardening, from static to highly dynamic regime at room temperature of Eurofer97 steel. Those data are necessary to calculate the stress/strain field around the crack tip by finite element simulations to model the toughness-temperature behaviour in the transition region
This paper presents an experimental investigation on the strain rate sensitivity of reduced activation steel Eurofer97 under uniaxial tensile loads in the strain rate range from 0.001s^{-1} to 1000s^{-1}. Round undamaged specimens of this material having gauge length 5 mm, diameter 3 mm, were tested in universal machine to obtain its stress-strain relation under quasi-static condition (0.001s^{-1}), and in modified Hopkinson bar to study its mechanical behaviour at high strain rates (300 s^{-1}, 1000 s^{-1}) respectively.
This tempered-martensitic stainless steel shows a quite high strain rate sensitivity.

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Mechanical properties of ultra-fine grained ODS Fe-Cr based alloys

P93A

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Irradiation resistance, high-temperature strength and reduced activation are the main properties demanded for the structural materials to be used in the future nuclear reactors in order to build devices with improved efficiency and safety. Among the candidate materials, oxide dispersion strengthened (ODS) reduced activation ferritic (RAF) steels appear to be the most promising candidates with such properties. The increasing interest in developing ODS RAF steels relies on the experimental results, which have shown improved irradiation resistance, high-temperature properties and thermal stability. The microstructure and composition of ODS materials, which depend on how they are processed, are the key for achieving the desired properties. The aim of the present work is the development of an ODS RAF Fe-Cr model alloy with an enhanced ultra-fine grained microstructure and the assessment of its mechanical properties.

ODS and non-ODS alloys with target compositions: Fe-14Cr, Fe-14Cr-0.3Y_2O_3 and Fe-14Cr-2W-0.3Ti-0.3Y_2O_3 alloy (% wt) have been produced and characterized. The blends of elemental powders and nanosized Y_2O_3 powder were mechanically alloyed in a planetary ball mill under an atmosphere of either helium or hydrogen. The milled powder was analysed by scanning electron microscopy (SEM), energy dispersive X-ray spectroscopy (EDS), X-ray diffraction (XRD) and laser diffraction (LD) techniques. The alloyed powder was canned, degassed and consolidated by hot isostatic pressing (HIP) for 2h at 1373 K and 200 MPa. After consolidation, the material was forged at 1323-1423 K and finally heat treated at 1123 K for 2h. The microstructure has been investigated by X-ray diffraction (XRD) and transmission electron microscopy (TEM) techniques. Tensile tests in the temperature range 273-973 K have been performed for all the alloys. A homogeneous dispersion of oxide nanoparticles was observed in the ODS alloys, and the tensile properties were enhanced in comparison to the Y_2O_3 free alloy. The obtained results demonstrate that the powder metallurgy route applied in the present work can produce ODS ferritic Fe-Cr alloys with enhanced microstructure and mechanical properties.

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Spark plasma sintering of model ODS ferritic steels

P93B

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Reduced activation ferritic (RAF) steels are being considered as fusion material candidates for structural components of the first wall. A nanosized dispersion of stable oxide particles in these alloys is expected to enhance their radiation resistance besides increasing their working temperature due to the high creep strength attributed to the stable nanodispersoids. In order to develop these alloys with the demanded mechanical properties and radiation resistance, the manufacturing route still needs to be optimized.
In this work the spark plasma sintering (SPS) technique has been explored as an alternative consolidation route for producing ultra-fine grained Fe-14Cr model alloys containing a dispersion of oxide nanoparticles. Elemental powders of Fe and Cr, and nanosized Y_2O_3 powder have been mechanically alloyed in a planetary ball mill and rapidly sintered in a spark plasma furnace. In this sintering process an electrical current is applied simultaneously with an uniaxial pressure for achieving homogeneous high temperatures using very high heating rates. Two alloys, with nominal compositions Fe-14%Cr and Fe-14%-0.3%Y_2O_3 (% wt), have been fabricated and their microstructure and mechanical properties investigated. The results have been compared with those obtained for other powder-metallurgy processed alloys of the same composition but consolidated by hot isostatic pressing.

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Mechanical Properties of Ferritic-Martensitic Steel 1Cr13Mo2NbVB and Austenitic Steel 12Cr18Ni10Ti Irradiated in the BOR-60 Reactor up to a Maximum Damage Dose of 108 dpa

P94A

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Examinations of fast reactor core internals can give a big array of data on properties of the materials irradiated at a high dose within a temperature range to be reached during operation of the first wall structural elements of the DEMO fusion reactor under design. The paper focuses on studies of mechanical properties of the 1Cr13Mo2NiVB ferritic-martensitic steel and the 12Cr18Ni10Ti austenitic steel irradiated up to a maximum damage dose of 108 dpa as the material of structural elements in the BOR-60 reactor core. Results of metallographic and TEM examinations of the 12Cr18Ni10Ti steel and fractographic investigations of the specimen fractures after mechanical tests are presented.

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Characterisation of Poly- and Singlecrystalline Tungsten by Instrumented Indentation

P94B

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Instrumented indentation is a common test method to determine materials properties in an almost non-destructive way. For interpreting the load-displacement curves exists a classical analysis technique, developed basically by Oliver and Pharr [1] and followed by the international standard DIN EN ISO 14577. Based on this approach, Tyulyukovskiy and Huber [2] developed a neural network analysis method. The training of the neural networks was done with data from finite element simulation of indentation tests. For the simulation a continuous, viscoplastic material behaviour was assumed. This way the neural network software solves the inverse problem of material characterisation, furnishing material parameters by analysing load-displacement data.
In the present work, this method is used for identification of plastic behaviour of single- and polycrystalline tungsten from macroindentation experiments at room temperature. For both materials a remarkable creep could be observed. It is shown that the continuum-based material model is able to characterise the isotropic tensile properties of polycrystalline tungsten. Even if the material model cannot point out the plastic anisotropy of singlecrystalline tungsten – due to the nonlinear tri-axial deformation – all load-displacement-curves can be evaluated with the neural networks. Further experimental study was made using SEM and profilometry in order to observe the pile-up behaviour and corresponding plastic anisotropy of tungsten for different crystal orientation. Investigations at elevated temperatures will follow.

[1] W. Oliver and G. Pharr, J. Mater. Res **7**, 1564 (1992)
[2] E. Tyulyukovskiy and N. Huber, J. Mater. Res **21**, 664 (2006)

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Influence of Processing and Alloying Method in Fracture Behaviour of Tungsten-Vanadium Alloys for ITER

P95A

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The purpose of this research is to evaluate the mechanical behaviour of three W-V alloys and compared them with a reference pure W. They are processed by hot isostatic pressing (HIP). In two of them, with a content of 2 and 4 %V by weight, the processing conditions are similar, this way we can determine the influence of the content of V. The other one has been manufactured with a different procedure and with a content of 2 %V in order to analyze the influence of the manufacturing method.

Additionally, we have developed a new experimental method that allows us to obtain a very small notch tip radius around 5-7 µm though the development of a new machined notch much more similar to a crack.

We have obtained the fracture toughness, mechanical strength, yield strength and modulus of elasticity as function of temperature. The mechanical characterization was performed by three point bending tests in an oxidizing atmosphere in a temperature range between -197 °C (immersion tests in liquid nitrogen) and 1000 °C.

Finally, we have examined by optical and scanning electron microscopy the microstructure and the fracture surfaces. This way it has been possible to determine and analyze the relationship between the macroscopic mechanical properties and the micromechanisms of failure involved, depending on the temperature and the dispersion of the alloy.

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On the brittle fracture of tungsten and tungsten alloys and its impact on the material's usage for fusion applications

P95B

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Within the last years, tungsten based materials experienced high attention, as they are within a small number of materials applicable for highly heat loaded regions in future fusion power plants. One of the major drawbacks, when working with tungsten, is its brittleness at ambient temperatures hampering this applicability. The low fracture toughness seems to be an intrinsic property of the material; nevertheless, it is possible to control it up to a certain value with specific alloying agents, rhenium probably the best known, or adjustment of a desired microstructure, ending up with a material having in most cases good fracture properties in one or two testing directions. Formation and stabilization of a desired microstructure is supported by low-alloying content (La, Y, Ti…) or doping (K) of tungsten. The goals worth heading for are manifold now: Understanding the strong ductilizing effect of rhenium alloying in detail would pave the way for new tungsten alloys, relying on more abundant and cheaper elements. Density Functional Theory calculations are of great help, as alloys of various compositions, also metallurgical non-accessible ones, are easily at hand. Using a completely different technique, micrometer-sized bending-, fracture-, and tensile experiments are an adequate tool to gain more knowledge on the basic mechanisms of deformation and fracture. By design of various different experiments, it is possible to change the size and shape of the specimen with an accuracy of about hundred nm, the type of loading and hence the stress state as a superimposition of crack tip stress field and stress field that would be existent without any crack. The crack velocity is strongly dependent on these factors; stable and unstable fractures have been observed for different configurations. One topic of this paper will be the detailed analyses of these experiments and the discussion of the consequences to understand better the fracture process in tungsten alloys. The above-mentioned topic of deformation respectively distinct microstructure is maybe the way most worthwhile to follow. Recent experiments [1,2] show that the fracture toughness and even more the fracture behaviour are strongly dependent on grain size and shape, texture and dislocation density, to a lesser amount on impurity concentration. These facts provide the basis for the work on highly deformed tungsten alloys made by high pressure torsion after subsequent powder compaction. The restrictions to alloying, maybe present at an industrial scale, are less stringent, opening a wide field of work, solely confined again by the small sample dimensions currently accessible. By use of a adequate heat treatment of this new W-X-composites, it is possible to end up with a dense tungsten alloy of desired composition. The analyses of the fracture behaviour of the different types of alloys and microstructures will be used to show the potential of improvement in fracture toughness and show novel ways of better microstructural design for fusion application.

[1] B. Gludovatz, Fracture Behavior of Tungsten, PhD thesis (2010)
[2] D. Rupp, S.M. Weygand, Phil. Mag., **90** 4055 (2010)

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Finite element crystal plasticity analysis and microindentation tests of single crystal tungsten

P96A

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In this work, a combined FEM simulation and experimental microindentation approach was taken to determine the micro-mechanical behaviour of single crystal tungsten incorporating a crystal plasticity constitutive model. This model was implemented in the finite element code ABAQUS/Standard using the user material subroutine UMAT. Numerical simulations and microindentation tests were conducted on single crystal tungsten in three crystallographic orientations [(100), (110) and (111)] using a spherical indenter. Load-displacement relationships and out-of-plane displacements were obtained from both simulations and experiment. Appropriate crystal plastic parameters were determined by correlating the simulated and the measured load-displacement curves. The pile-up patterns under the spherical indenter showed four-fold, two-fold, and three-fold rotational symmetries for the (100), (110), and (111) orientations, respectively. The comparison of microindentation impressions obtained from the experiments with simulation results showed a reasonably good agreement.

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Multi-colour Pyroreflectometry performance with parasitic reflexion

P96B

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ITER environment raises the issues related to surface temperature measurements of plasma facing components which will be metallic with low and changing emissivity. Indeed assuming that the emissivity is known, if the reflected flux is not taken into account in the standard process of infrared measurement, the temperature of the surface will be wrong. In this paper we asses the performance of the multi-colour pyroreflectometry method in such environment by simulating parasitic reflexion. Parasitic reflexion is simulated in laboratory facing two blackbodies: the first one to heat up a metallic surface, the second one to simulate parasitic source. The experiment is performed on INOX. Experimental approach is completed by numerical simulation in order to validate a model and extrapolate for surfaces with different optical properties.

The multi-colour pyroreflectometry methods have been developed at the CNRS-PROMES at two wavelengths [1]. The principle of the method is as following: the illumination is provided by pulsed laser diodes working at 1.3 µm and 1.55 µm. From the reflected light the bidirectional reflectivities at both wavelengths are determined. From the emitted light, luminance is measured at both wavelengths. The method introduces the diffusion factor η_d (the ratio of bidirectional reflectivity $\rho_{FD}^{\vec{i}_0,\vec{j}_0}$ to hemispherical reflectivity $\rho^{\vec{j}_0,\cap}$) which is assumed to be independent with the wavelength. Therefore, the absolute temperature as well as the emissivity can be deduced from the resolution of a system of two linear equations [1]. This method is routinely used at the Test bed Facility FE200 (AREVA) and has been also tested at ASDEX on tungsten sample [2].

Numerical simulation of a temperature measurement by bicolour pyroreflectometry has been performed adding a reflexion term in the total luminance received by the detector in order to assess the error made when we do not take into account the reflexion. In the case of tungsten heated at 800 °C and assuming a blackbody environment at a surface temperature of 800°C, 1100 °C and 1600°C, the overestimation is respectively of about ~ 0%, 24 % and 70 %. It is worth noting that the overestimation starts when the temperature of the parasitic source is higher than the surface temperature of the object that we observe. In the case of a temperature environment lower than the target, the temperature measured can be slowly underestimated. This is characteristic of a bicolour measurement. Indeed when the target is heated up to 1600 °C and the parasitic source radiates at 800°C and 1100°C, the temperature is underestimated respectively of about ~ 1 % and ~ 2 %.

In this paper simulations are compared with laboratory experiments using a blackbody to simulate a parasitic source. A tricolour approach is also tested to assess reflexion contribution by introducing a background perturbation temperature as a new parameter and reduce the uncertainties on surface temperature.

[1] D. Hernandez et al., Rev. Sci. Instrum. 66 (1995) 5548.
[2] Report Contract CEA and PROMES-CNRS. Ref: V3448.001-4

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