Journal of Physics: Conference Series **100** (2008) 062003 doi:10.1088/1742-6596/100/6/062003

Plasma-wall interaction: a complex combination of surface processes critical for thermo-nuclear fusion

Joachim Roth[1], Emmanuelle Tsitrone[2], Alberto Loarte[3]
[1] Max-Planck-Institut für Plasmaphysik, EURATOM-Association, 85748 Garching, Germany
[2] Association EURATOM-CEA, CEA/DMS/DRFC CEA Cadarache, 13108 Saint Paul lez Durance, France
[3] EFDA-Close Support Unit Garching, 85748 Garching, Germany

E-mail: joachim.roth@ipp.mpg.de

Abstract. Handling particle and heat loads on the plasma-facing in vessel components constitutes a major engineering problem in thermo-nuclear plasma devices. The choice of the plasma-facing materials is a crucial point in future devices, such as ITER (International Thermonuclear Experimental Reactor), which will determine the plasma facing components (PFCs) lifetime or the tritium (T) inventory build up in the vessel, which must be limited for safety reasons. In order to address the most critical issues, the European Task Force on Plasma Wall Interaction has been implemented in the frame of EFDA (European Fusion Agreement). The EU PWI TF brings together the efforts of 24 European associations in the following fields of investigation:

- Material erosion and transport in tokamaks
- Tritium inventory and removal techniques
- Transient heat loads on plasma facing components
- Dust production and removal techniques
- Associated modelling and diagnostic development

This paper describes the organisation of the EU PWI TF. It provides examples for the multitude of surface processes involved, ranging from sputtering of PFCs by the plasma, to eroded material transport in the discharge, sticking of atoms and molecules on PFCs, and D/T trapping in the material. In particular, the present status of knowledge concerning material erosion and hydrogen retention for the choice of ITER materials (Beryllium, Carbon and Tungsten) is presented.

1. Introduction

ITER (International Thermonuclear Experimental Reactor) is a joint international research and development project that aims to demonstrate the scientific and technical feasibility of fusion power. ITER will be constructed in Europe, close to the research centre of Cadarache in the South of France.
The collaborative result of decades of fusion research on energy production in magnetic fusion devices world-wide has resulted in a design which should allow a long-pulse (400 s) burning fusion plasma at an energy amplification factor, *Q*, of at least 10 ($Q = P_{fusion}/P_{heating}$ is the ratio of power produced by fusion reactions versus the power required to heat up the plasma by external means),
However, questions remain open, in particular in the field of plasma wall interactions. Indeed, in comparison with present day devices, ITER represents a major step forward in terms of plasma wall

interactions, namely due to the increase in the plasma energy and extended duty cycle. While the size of ITER is about two times larger than JET [1], the largest present experiment, the stored energy in the plasma is a factor of 35 and the ion wall fluences per discharge a factor of 1000 higher. For surfaces in direct plasma contact (divertor plates, see fig. 1) the constraints on the plasma facing components (PFCs) are severe. The design has

- to ensure heat exhaust at significant power loads (up to 10 MWm^{-2} to extract continuously on the divertor target plates), as well as handling of higher transient heat loads. In any case excessive sublimation or melting of the target materials must be avoided.
- to achieve sufficient lifetime of the plasma-facing components (>3000 ITER full performance shots). This goal is largely related to the control of transient heat loads due to events such as edge localized modes (ELMs), which are fast repetitive heat and particle expulsion from the plasma occurring during the high performance scenario foreseen for ITER, and disruptions, which is a loss of control of the plasma resulting in a fast energy peak on PFCs.
- to ensure an acceptable level of contamination of the plasma by He ash and eroded wall material.
- to stay below the long-term tritium (T) inventory limit, which is set by safety considerations to 350g.

Extrapolation from present day devices with all carbon walls show that the T inventory limit could be reached within less than 100 full performance discharges [2]. The main retention processes identified so far are linked to carbon erosion, transport and re-deposition, leading to fuel trapping in the re-deposited layers [3]. The material mix foreseen for ITER (Be for the first wall (700 m^2), W on the upper divertor regions and the dome (100 m^2), C on the divertor plate (50 m^2), see figure 1) should reduce this constraint compared with full carbon machines. However, predicting T retention under these conditions remains challenging. Finally, if the T retention remains unacceptably high with C divertors, a full metal device [1,4] could be contemplated in a second phase.

A coordinated European research activity has been established (EU Task Force on Plasma–Wall Interaction) with the aim "to provide ITER with information concerning lifetime-expectations of the divertor target plates and tritium inventory build-up rates in the foreseen starting configuration and to suggest improvements, including material changes, which could be implemented at an appropriate stage." This paper outlines the strategy of the EU-PWI-TF, described in section 2, and illustrates the need for basic PWI research, focussing in section 3 on the issue of tritium retention.

2. Strategy of the EU Task Force on Plasma-Wall Interaction

In order to promote the collaboration between European research groups in this field the EU-PWI-TF was established in 2002. High priority issues have been identified and special expert working groups (SEWGs) have been set up to work on well defined subtopics (for more details see list of issues and SEWGs on the TF website www.efda-taskforce-pwi.org). SEWGs define coordinated experiments to be performed in fusion devices, or laboratory experiments (plasma simulators, ion beams), and gather the associated results for comparative analysis. Meetings at the working level and exchange of scientists to participate in collaborative experiments are encouraged through mobility funding by EFDA. New SEWGs can be created if needed, or ended after completion of the related work.

A 3 years work programme is established from the commitments of each association in the various working areas. This work programme, based on the voluntary contributions, is periodically reviewed and can be found on the web-page. Contact persons, nominated in each association, are essential to coordinate the PWI work within the association, to transfer the knowledge from the association to the TF and vice versa and to participate actively on the TF meetings.

Finally, EFDA Task Agreements can be issued on critical topics, defining activities eligible for preferential support. These tasks have well defined deliverables, deadlines (typically 1 year duration) and budget. New Task agreements are sent to all Heads of Associations and TF contact persons to ensure widespread information and contributions. Final reports are being made available on the EU TF web page.

A general TF meeting is organised once a year, where progress in the SEWGs, EFDA technology tasks and reports from the associations are presented. Overview of the EU PWI TF activities has been

presented (e.g. [5]) and more than 70 publications with results obtained within the TF were published in 2006.
To illustrate the TF activities, the present status of knowledge on tritium retention is shortly reviewed in the next section. After presenting the main ITER plasma parameters, the different processes involved, namely implantation and co-deposition, are described.

3. Strategy of the EU Task Force on Plasma-Wall Interaction

3.1 Parameters of the plasma in contact with plasma facing components in ITER [6]

Particle and energy fluxes on ITER plasma facing components will span a wide range [7]. Due to the magnetic configuration of the tokamak, the power carried by the charged particles, following the field lines, is concentrated on the specially designed divertor target plates (see figure 1), while the rest of the first wall is less loaded. Scenarios with enhanced dissipative power losses through radiation in the plasma boundary have been developed, resulting in low divertor plasma temperatures (~ 5 eV and lower). This "detached divertor regime" [8],allows to keep peak power fluxes on the divertor targets under ~ 10 MWm^{-2}. The corresponding ITER plasma parameters along the outer divertor target estimated from modelling [9, 10] are shown in fig. 2.

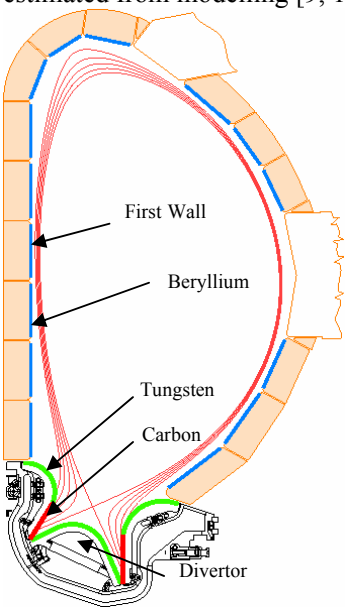

Figure 1. Cross-section of ITER with different armour materials. Magnetic surfaces are also shown.

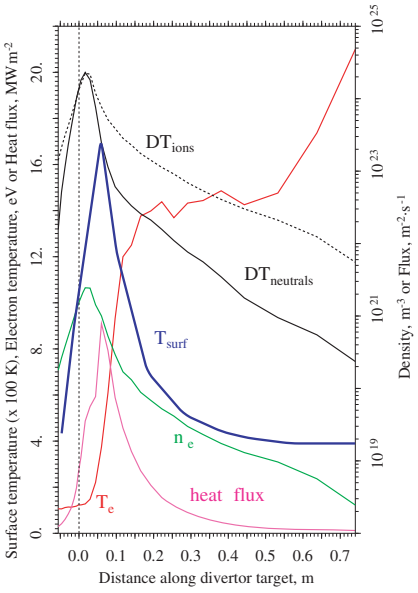

Figure 2. Plasma temperature Te, density ne, D/T ions and neutrals particle fluxes, heat fluxes and surface temperature along the ITER outer divertor target. x = 0 to the strike point position.

Near the plasma strike point the typical plasma/neutral fluxes reach values larger than 10^{24} $m^{-2}s^{-1}$ (leading to a total fluence > 10^{26} m^{-2} for each ITER pulse) with plasma densities in the order of ~10^{21} m^{-3} and plasma temperatures of ~ 3 eV. This corresponds to an ion impact energy of ~15 eV, due to acceleration in the plasma sheath potential. Despite these low impact energies, the power flux deposited on the water cooled ITER divertor target is significant, leading to an expected surface temperature of ~ 1500 K around the strike points [10]. Asymmetries between inner and outer divertor are observed in present tokamaks with lower power fluxes and plasma temperatures in the inner divertor.
In contrast to the divertor, the present understanding of particle and energy fluxes, at the first wall of ITER is more uncertain. Present experimental and modelling results indicate that the flux of neutrals, produced by charge-exchange processes between the incoming cold recycling D/T neutrals and hot edge plasma ions, is in the range of 10^{19}-10^{21} $m^{-2}s^{-1}$ with typical energies ~ 8 – 300 eV, but extending into the keV range. In addition to neutral particles, ion fluxes can also reach the first wall, in particular

due to plasma turbulence which results in transport perpendicular to the field lines [11]. On different areas of the ITER walls the conditions will range from net surface erosion to predominant deposition of impurity atoms eroded elsewhere. In present devices, the main vessel and outer divertor walls are typically erosion dominated, while large deposition occurs on the walls of the inner divertor [12].

3.2 Implantation

Implantation of low-energy hydrogen ions into materials selected for ITER, such as carbon [13,14], tungsten [15] and beryllium [16,17], has been investigated in detail. Ion ranges, reflection coefficients and hydrogen retention profiles have been compared to Monte Carlo code calculations. Measured depth distributions are well reproduced and the underlying physics in electronic and nuclear stopping is reasonably well understood [18].

For carbon, retention depends on the material structure. In pyrolytic or fine grain graphite with low porosity, hydrogen does not diffuse and after reaching a local concentration in the implantation range of D/C= 0.4 further hydrogen is reemitted such that the total inventory saturates [19].In more porous materials [20], like carbon fibre composite (CFC) materials [21] selected for ITER due to their good thermo-mechanical properties, the behaviour is different. Experiments at high fluences have been performed in the frame of the TF in ion beams, linear plasma devices and tokamaks. Results show no saturation of the total implanted amount as a function of fluence. Instead, the retained amount increases as the square root of the ion fluence (see fig. 3) due to diffusion deep into the bulk. This inward diffusion shows a very low activation energy and details of the transport process are poorly understood [21]. This could have implication for long pulse/high fluence machines such as ITER, and probably plays a role to explain the retention rate observed in the long pulses of Tore Supra [22].

In many metals, such as W and Mo, deuterium is highly mobile and is only retained in radiation damage or defects of the crystal lattice [23,24,25]. After saturating available traps in the ion induced damage profile, inward diffusion and subsequent trapping at lattice defects increases the total amount of trapped hydrogen similarly to the case of porous graphites. The resulting retention depends critically on the crystallinity of the substrate. The retention increases from single crystals, to well annealed polycrystalline material and can reach high values in plasma-sprayed W coatings [26]. As the build-up of the inventory is diffusion limited, it increases only with a square-root of the fluence and stays tolerable at high fluences contemplated for ITER (fig. 3). For Be the diffusion of hydrogen is slow and retention is limited to the saturation of traps within the ion range [17,27], although the database is much less broad. Figure 3 presents a comparison of implantation and retention in the three candidate materials, as recently reviewed in [28].

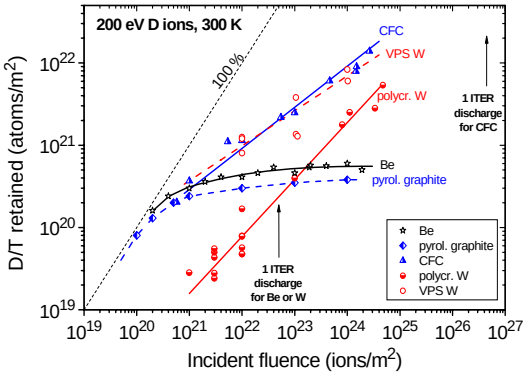

Figure 3. Deuterium retention in pyrolytic graphite and CFC materials, polycrystalline and vacuum plasma sprayed (VPS) tungsten, and beryllium irradiated with 200 eV D ions at room temperature as a function of incident ion fluence. The data for Be are taken from Ref. [17] and from recent measurements [27]. Arrows indicate expected D/T ion fluences incident on areas covered with Be, W and CFC materials, respectively, during one ITER discharge.

3.3 Implantation

3.3.1 Erosion of plasma facing materials

The first step in the chain of processes leading to fuel inventory build-up by co-deposition is the erosion of the wall material. In the keV energy range physical sputtering of solids is well described by theory [29], while in the range below 1 keV, especially for light ions, threshold effects have to be

considered [30,31]. A broad experimental data base exists today and even down to yields of the order of 10^{-5} the data are well reproduced by Monte Carlo codes simulations, such as SPTRIM [32]. In view of uncertainties in the edge and divertor plasma parameters in future fusion devices, the precision of the sputtering yield data is sufficiently good for wall lifetime and tritium inventory estimates.
However, for carbon based materials, chemical interaction with the hydrogen plasma leads to enhanced erosion yields [33,34] (fig. 4). Chemical erosion is a complicated, multi-step process which depends on surface temperature, ion energy (see figure 4 [35]) and ion flux, and still exhibits many unresolved details. The main features are (see review by A. Kleyn [36] and T. Zecho [37] in this volume for details):

- Enhanced yields (~ 0.1) at elevated temperatures with a maximum around 600-800 K)
- Emission of hydrocarbon molecules and radicals in compositions varying with ion energy [38,39].
- At least two erosion processes are involved: a thermal reaction process (no threshold, weak isotope effect) [40] and a kinematical process at room temperature and low ion energies (energy threshold observed in plasmas [41] and MD calculation [42]).
- A flux dependence becoming obvious at fluxes above 10^{22} D/m^{2}s and decreasing the chemical erosion yield to values well below 10^{-2} [43].

After cross examination and recalibration of data from different experimental devices performed in the frame of the TF, the large experimental data base could be unified, and an analytic description [43] including the flux dependence could be developed for extrapolation to ITER conditions. Critical, and insufficiently known parameters are the threshold behaviour [44], the contribution of heavier hydrocarbons and radicals [45], the influence of the chemical reactivity due to simultaneously incident energetic impurity ions [46,47] and the origin of the flux dependence.

3.3.2 Re-deposition of hydrocarbon radicals

The eroded particles will enter the plasma, get ionised and be transported along the magnetic field to divertor surfaces, where they can either stick or be re-eroded and transported until they reach remote areas of the vacuum vessel where they can be co-deposited with tritium. While the T inventory in deposited layers on the divertor target is small due to the high surface temperatures (800 to 1700 K [10]), the D,T/C ratio can reach values of 1 in remote layers at lower temperature.
The sticking coefficient depends strongly on the kind of hydrocarbon molecules and the deposition conditions. For a CH_3 particle beam alone the sticking coefficient on a typical hydrogen saturated layer is of the order of 10^{-4}, while simultaneous bombardment by incident hydrogen ions provide bonding sites for CH_3 by hydrogen removal, such that the sticking coefficient increases into the 10^{-2} range [48].
A collection of sticking coefficients for different hydrocarbon molecules measured experimentally [49] and calculated using molecular dynamics (MD) codes as function of incident ion energy [50] can be found in ref [51]. At thermal energies the sticking coefficient depends strongly on the electronic configuration of the hydrocarbon molecules decreasing from sp1 to sp2 and sp3 in reasonable agreement with MD calculations, which predicting sticking coefficients close to 1 at energies above 10 eV. Data for sticking coefficients on realistic plasma-facing materials were presented recently [52] and are further studied in the framework of the EU PWI Task Force.

3.3.2 Re-deposition of hydrocarbon radicals

A major effect in the transport of hydrocarbons in fusion devices is their multiple deposition and re-erosion by the impact of plasma ions and atomic hydrogen. Modelling of the erosion/redeposition balance indicates that re-deposited hydrocarbon layers in tokamaks are subject to an “enhanced” chemical erosion with yields in the range of ~ 10 % under plasma impact [53]. This is in agreement with laboratory experiments of the chemical erosion of soft, hydrogen-rich a-C:H layers showing much higher erosion yields by thermal atomic hydrogen when compared to graphite [54]. The rate for re-erosion of a-C:H layers by neutral, atomic hydrogen is temperature dependent as demonstrated in

laboratory experiments, while the hydrocarbon deposition is not [55,56]. Thus, it is generally found that the combination of the complex processes involved in the final deposition of hydrocarbons results in a prevalence of hydrocarbon deposition at low surface temperatures and of erosion at high temperatures.

For example, studies with temperature controlled samples exposed in the private flux region of the divertor in ASDEX Upgrade show that carbon deposition can be reduced by a factor of 200 when going from 300 to 500 K [57]. This is in qualitative agreement with the experimental results in lab experiments, but the temperature of transition from deposition to net erosion depends strongly on the respective ratio of hydrocarbon to atomic hydrogen flux.

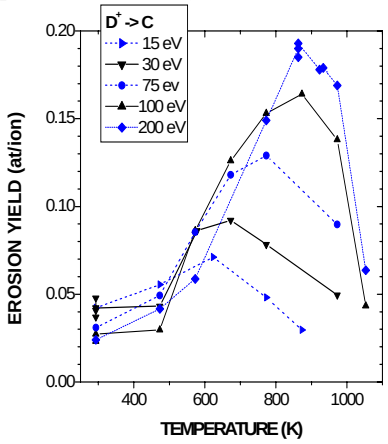

Figure 4. Chemical sputtering yield of C due to deuterium ions as function of surface temperature for different ion energies [35].

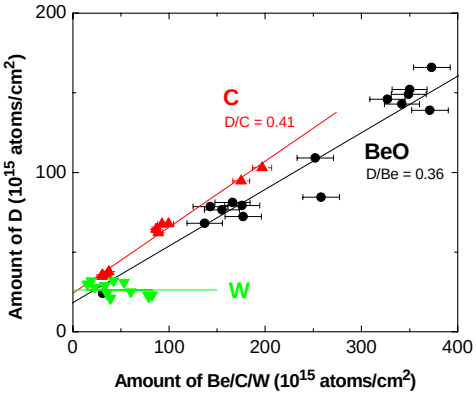

Figure 5. Co-deposited amounts of deuterium on a collector in front of different samples sputtered by 1 keV D+ ions [60].

3.3.4 Co-deposition of hydrogen atoms

The final step in the co-deposition process is the incorporation of hydrogen during deposition in remote areas, not reached by the plasma for further re-erosion. Impurity atoms or molecules are deposited together with a flux of energetic or thermal neutral hydrogen atoms. For carbon deposition, amorphous hydrogenated carbon(a-C:H) layers are formed, where the hydrogen concentration and the hardness of the layer depend critically on the energy of the simultaneously incident flux of hydrogen. Energetic ions lead to the deposition of hard films with hydrogen concentrations H/C of about 0.4, while low energy thermal hydrogen leads to the formation of soft films, with H/C concentrations exceeding 1 [58,59]. In fig. 5 the amount of retained hydrogen in deposited films of C, Be and W is shown on collectors mounted in front of targets irradiated with 1 keV D ions [60]. Thus, together with the deposited sputtered atoms, a flux of reflected energetic deuterium atoms hits the collector. Under these conditions the carbon film contains 0.41 D/C. In contrast to carbon, for deposited metals a very low retention is expected due to the fast out-diffusion of hydrogen. This is indeed the case for W, while for Be the oxidation of the layer to BeO is responsible for a deuterium uptake which can be similar to carbon.

The data of fig. 5 were obtained from collectors at room temperature. The concentration in the deposited films is strongly dependent on surface temperature. Especially for metals and BeO the concentration decreases already above 500 K [61]. For carbon the concentration decreases only at temperatures exceeding 700 K, and values on D/C between 1 in remote areas and 0.01 on high temperature surfaces are measured in fusion devices [62, 63].

The hydrogen concentration in deposited films, with its dependence on plasma conditions, geometry with respect to the eroded surfaces, impurity content and surface temperature, is still one of the largest uncertainties in the estimate of tritium inventory in ITER.

3.3.5 Influence of material mixing

An important complication in the evaluation of the tritium inventory is the intermixing and chemical interaction of the different materials used in ITER. In the binary interaction of C, Be and W, the formation of carbides [64] and Be-W alloy phases [65] were reported and the temperature limits for different stoichiometries of the compounds assessed. However, for tritium inventory, most of the available data are obtained for pure materials, from which it is difficult to predict the behaviour in mixed materials.
From the scarce literature [24, 60, 61,66] it appears that superficial carbide layers on W strongly increase the retention of implanted deuterium [24]. On Be a carbon layer retains deuterium up to temperatures of 800 K ,when a stoichiometric Be_2C phase is formed, which releases the deuterium [67]. The interaction of Be with W leads to the formation of different beryllides with much lower melting temperature than pure W [65], but no information on the influence on hydrogen retention is reported, yet. More work is needed in this area.

4. Summary

Plasma-wall interaction processes are critically defining the plasma-facing material choice for ITER. The EU PWI TF concentrates the knowledge of 24 European associations on these problems. Critical issues are covered by coordinated work performed in SEWGs. Limited financial support is available for collaborative work on well defined tasks identified as priorities by the TF.
One of the critical issue covered by the TF programme is the tritium inventory build up in the vessel, which must remain under the safety limit. For extrapolations from present day devices to ITER, improved understanding is necessary on several individual steps in the tritium retention chain:

- implantation, diffusion and retention in ITER relevant material grades
- Erosion, transport and re-deposition of the eroded material on vessel walls
- Co-deposition of hydrogen isotopes with eroded materials

In particular, influence of mixed material formation on all the above processes needs further investigation.

[1] J. Paméla, G.F. Matthews, V. Philipps and R. Kamendje, J. Nucl. Materials, **363-365** (2007) 1-11
[2] T. Loarer, C. Brosset, J. Bucalossi, P. Coad, G. Esser, J. Hogan, J. Likonen, M. Mayer, P. Morgan, V. Philipps, V. Rohde, J. Roth, M. Rubel, E. Tsitrone A. Widdowson and JET EFDA Contributors Nucl. Fusion **47** (2007) 1112–1120
[3] M. Mayer, R. Behrisch, H. Plank, J. Roth, G. Dollinger, C.M. Frey, J. Nuclear Materials **230**, 67-73 (1996).
[4] R. Neu, R. Dux, Ch. Hopf, A. Kallenbach, Th. Pütterich, V. Rohde, A. Herrmann, K. Krieger, H. Maier, J. Nucl. Materials, **363-365** (2007) 52
[5] V. Philipps, J. Roth and A. Loarte, Plasma Phys. Control. Fusion **45** (2003) A17
[6] Progress in the ITER Physics Basis : Chapter 4: Power and Particle Control, B. Lipschultz, A. Loarte et al., Nuclear Fusion **47** (2007) S203
[7] ITER Technical Basis, ITER EDA Documentation Series No. 24, IAEA, Vienna 2002
[8] A. Loarte et al., Nucl. Fusion 38 (1998) 331
[9] A.S. Kukushkin et al., Nucl. Fusion 42 (2002) 187
[10] G. Federici et al., J. Nucl. Mater. 313–316 (2003) 11
[11] A. Kallenbach, R. Dux, J. Gafert, G. Haas, et al., Nucl. Fusion 43 (2003) 573-578
[12] J. P. Coad et al., J. Nucl. **290–293 (**2001) 224
[13] W.R. Wampler, C.W. Magee, J. Nucl. Materials **103&104** (1981) 509
[14] B.M.U. Scherzer, M. Wielunski, W. Möller, A. Turos and J. Roth, Nuclear Instr. & Meth. in Phys. Res. B33, (1988) 714-718
[15] R. Causey, K. Wilson, T. Venhaus, W.R. Wampler, J. Nucl. Mater. **266-269** (1999) 467
[16] W. Möller, B.M.U. Scherzer, J. Bohdansky, *Retention and release of deuterium implanted into beryllium*, IPP-JET Report No. 26 (1985)
[17] R.A. Anderl, R.A Causey, J.W. Davis, et al., J. Nucl. Mater. **273** (1999) 1
[18] J.F. Ziegler, J.P. Biersack, and U. Littmark, *The Stopping and Range of Ions in Solids*, Vol. 1 Pergamon Press, New York, 1985
[19] B.L. Doyle , W.R. Wampler, D.K. Brice, S.T. Picraux, J. Nucl. Materials **93/94** (1980) 551
[20] A.A. Haasz, J. Davis, J Nucl. Mater. **209** (1994) 1140

[21] J. Roth, V.Kh. Alimov, A.V. Golubeva, R.P. Doerner, J. Hanna, E. Tsitrone, Ch. Brosset, V. Rohde, A. Herrmann, M. Mayer, J. Nucl. Materials **363-365** (2007) 822-826
[22] E. Tsitrone, J. Nucl. Materials, **363-365** (2007) 12
[23] R.A. Causey, T.J. Venhaus, Physica Scripta T94 (2001) 9
[24] O. Ogorodnikova, J. Roth, M. Mayer, Journal of Nuclear Materials **313-316** (2003) 469
[25] G. Wright, D. Whyte, Retention in Mo, in press
[26] A.V. Golubeva, V.A. Kurnaev, M. Mayer, J.Roth, Int. Symp. on Hydrogen in Matter, (Eds.) Myneni, G.R.; Hjörvarsson, B. AIP Conference Proceedings **837**. American Institute of Physics, Melville, NY (2006) 12-21
[27] V. Kh. Alimov and J. Roth, Private communication (2006).
[28] V. Alimov, J. Roth , Physica Scripta **T128** (2007) 6
[29] P. Sigmund, Phys. Rev. 184 (1969) 383
[30] J. Bohdansky, J. Roth and H.L. Bay:, J. Appl. Phys. 51 (1980) 2861
[31] J. Roth, J. Bohdansky, A.P. Martinelli, Rad. Eff. 48 (1980) 213
[32] W. Eckstein, *Computer simulation of ion solid interactions, Springer Series in Materials Science* (Springer Verlag, Berlin and Heidelberg, 1991), 1st ed.
[33] J. Roth, J. Bohdansky, W. Poschenrieder and M.K. Sinha, J. Nucl. Mat. 63 (1976) 222
[34] W. Jacob, J. Roth: *Chemical Sputtering*, in Topic in Applied Physics, ed. W. Eckstein, R. Behrisch (Springer Verlag, Berlin 2007), in press
[35] M. Balden, J. Roth J. Nucl. Mater. (2000)
[36] A.W. Kleyn, Shumack, A.E., Westerhout, J., Vijvers, W.A.J.; Brezinsek, S, Lopes Cardozo, N.J., Goedheer, W.J., de Groot, B., van der Meiden, H.J., Schram, D.C., Whyte, D.G., van Rooij, G.J.: this volume
[37] T. Zecho, Ch. Fischer, G. Ehrenhaft, J. Küppers, this volume
[38] B. V. Mech, A. A. Haasz, and J. W. Davis , J. Appl. Phys. **84** (1998) 1655
[39] E. Vietzke, K. Flaskamp, V. Philipps, J. Nucl. Mater. **128&129** (1984) 545
[40] A. Horn A. Schenk, J. Biener, B. Winter, C. Lutterloh, M. Wittmann, J. Küppers, Chem. Phys. Lett. **231** (1994) 193
[41] U. Wenzel, M. Laux, R. Pugno, K. Schmidtmann, J. Nucl. Materials 290-293 (2001) 352
[42] E. Salonen, K. Nordlund, J. Keinonen, and C. H. Wu, Phys. Rev. B 63 (2001) 195415
[43] J. Roth, R. Preuss, W. Bohmeyer,et al., Nuclear Fusion **44** (2004) L21
[44] D.G. Whyte, W.P. West, C.P.C. Wong, et al., Nucl. Fusion **41** (2001) 1243
[45] E. Vietzke, J. Nucl. Mater. **145-147**, 443 (1987)
[46] K. Schmid, M. Baldwin, R. Doerner, J. Nucl. Materials **337-339** (2005) 862
[47] W. Jacob, C. Hopf, M. Schlüter, and T. Schwarz-Selinger, J. Nucl. Materials **337-339** (2005) 839
[48] M. Meier, A. von Keudell, J. Chem. Phys. **116** (2002) 5125
[49] W. Jacob, J. Nucl. Materials 337-339 (2005) 839
[50] D. A. Alman, D. N. Ruzic, Physica Scripta **T111** (2004) 145
[51] J. Roth, E. Tsitrone, A. Loarte, Nucl. Instr. Meth. **B258** (2007) 253
[52] W. Schustereder, B. Rasul, N. Endstrasser, V. Grill, P. Scheier, T.D. Märk, IISC NIM B (2007)
[53] A. Kirschner, et al., Journal of Nuclear Materials **328** (2004) 62
[54] E. Vietzke, A.A. Haasz, in: W.O. Hofer, J. Roth (Eds.), Physical Processes of the Interaction of Fusion Plasmas with Solids, Academic, San Diego, 1996
[55] Bohmeyer, W., et al., Journal of Nuclear Materials **337-339** (2005) 89
[56] Bohmeyer, W., et al., Proc. 30th EPS Conf. on Controlled Fusion and Plasma Physics, St. Petersburg, Russia, 2003, Vol. 27A, p. 3.184
[57] M Mayer, V Rohde and the ASDEX Upgrade Team, Nucl. Fusion **46** (2006) 914
[58] T. Schwarz-Selinger, A. von Keudell, and W. Jacob, J Applied Physics 86 (1999) 3988
[59] W. Jacob, Thin Solid Films 326 (1998) 1
[60] M. Mayer, R. Behrisch, H. Plank, J. Roth, G. Dollinger, C.M. Frey, Journal of Nuclear Materials **230**, 67-73 (1996)
[61] M.J. Baldwin, K. Schmid, R.P. Doerner, A. Wiltner, R. Seraydarian, Ch. Linsmeier, J. Nucl. Mater. 337-339 (2005) 590
[62] M. Mayer et al., *Mechanism of Hydrocarbon Layer Formation in Remote Areas of Fusion Devices*, EPS St. Petersburg (2003)
[63] T. Nakano, N. Asakura, H. Takenaga, H. Kubo,K. Shimizu, H. Kawashima and the JT-60 Team: *Particle control under wall saturation in long-pulse discharges in JT-60U*, J. Nucl. Materials (2006) accepted for publication
[64] Ch. Linsmeier, J. Roth, K. Schmid, *Formation and erosion of mixed materials*, Atomic and Plasma-Material Interaction Data for Fusion **12** (2003) 79
[65] Ch. Linsmeier, K. Ertl, J. Roth, A. Wiltner, K. Schmid, F. Kost, S.R. Bhattacharyya, M. Baldwin, R.P. Doerner: J. Nucl. Materials **363-365** (2007) 1129-1137
[66] A. Wiltner, Ch. Linsmeier, New Journal of Physics 8 (2006) 181
[67] J. Roth, W.R. Wampler, W. Jacob, J. Nucl. Mater (1999)