Plasma operation with high-Z environment

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Abstract

High-Z materials appear to be the most realistic choice for reactor plasma facing components (PFCs) because they exhibit the lowest erosion and in the case of tungsten acceptable radiological properties. Results from present day devices and from laboratory experiments confirm the advantages of high-Z materials but also point to operational restrictions, when using them as PFCs. They are associated with the central impurity concentration being crucial for central power losses, which is determined by the source, the penetration and the transport of the impurities within the confined plasma. In this respect, the compatibility of ion cyclotron frequency heating (ICRH) in a high-Z PFC environment has still to be proven and a high enough level of anomalous central transport compared to neoclassical transport is necessary to avoid central high-Z accumulation. Remedies are being developed to ameliorate the impact of the restrictions, mainly relying on reducing the source as far as possible and increasing anomalous impurity transport to keep the central high-Z plasma concentrations low.

1 Introduction

ITER [1], which has been officially launched in 2006, is designed to demonstrate the scientific and technical feasibility of magnetic fusion. Demo reactors (see for example [2]), which are being designed currently and which should be constructed in about 20 years from now, will incorporate all technical solutions necessary in a commercial reactor. By doing these steps, emphasis shifts from purely plasma oriented research to an integrated approach which has to aim not only at the optimization of the plasma performance but also at the adaptation to the technical boundary conditions.
Optimisation of the core plasma performance was the main driver for the implementation of low-Z carbon based materials as plasma facing materials (PFM) in almost all fusion devices during the last two decades. A large

operational experience and database exists with these materials as plasma facing components (PFCs) [3], which allows a reliable prediction of the core plasma performance for future devices. However, for economic reasons a sufficient lifetime of the first wall components is essential. Chemical erosion [4] leads to significant erosion yields even under low temperature, cold plasma conditions and can seriously limit the lifetime of carbon based components. Another critical issue connected to this large erosion is the long term retention of the radioactive tritium by co-deposition with C at remote areas [5, 6]. This negatively impacts the fuel supply and safety of fusion energy [7].
The most promising alternative category of plasma facing materials are high-Z materials with tungsten (W) in particular [8]. These materials have acceptable thermo-mechanical properties, the possible advantage of very low or negligible erosion at low plasma temperatures and a moderate uptake of tritium [9]. These advantages compete with their strong poisoning effect of the plasma due to cooling by radiation losses, if the impurity source is too high and/or impurity transport leads to accumulation in the central plasma. After the first negative experiences due to strong central cooling through tungsten at the Princeton Large Torus (PLT) [10, 11] high-Z material was no longer used, except for high field/high density devices using molybdenum (Mo) for their PFCs (see for example [12, 13]). Driven by the needs of a reactor, experiments using W as PFC were resumed in TEXTOR using W test and poloidal limiters (see for example [14]). ASDEX Upgrade performed in 1995 a campaign with a complete W divertor [15] and just recently was equipped with all PFCs made of tungsten [16, 17].
Alternative modes of operation, especially the use of a magnetic divertor and the application of special techniques to suppress excessive high-Z influx and accumulation, allowed to operate successfully with high-Z PFCs in these devices. Further confirmation is expected from the 'ITER-like wall experiment' in JET, which will use Be, W and CFC in a configuration similar to that foreseen in ITER [18].
This paper will concentrate on recent results on the operation of fusion devices with high-Z PFCs and especially W components – a more general overview can be found in [19]. In the next section investigations on sputtering and influx will be presented, Sec. 3 is dedicated to transport issues. Sec. 4 will conclude the paper and give some extrapolations to the use of W in ITER.

2 Plasma Wall Interaction

Because of the higher atomic mass of high-Z materials substantial sputtering by deuterium occurs only at relatively high energies (e.g. a sputtering rate 10^{-3} for 400eV ions) so that a sufficiently cold deuterium plasma in the SOL and in front of the target plates does not lead to substantial erosion [20]. In addition, because of their low speed, W-neutrals sputtered from the wall are ionized over distances small compared to their large gyro radius. This leads to a prompt re-deposition of a high fraction of sputtered particles first observed in [21, 22]. In present day devices the sputtering of the high-Z materials is governed by low-Z impurities. Usually boron (from frequent boronisations), carbon (from remaining C PFCs or as an intrinsic impurity of metals) and oxygen (from water and residual leaks) exist in concentrations of up to about one percent. These impurity ions have typically the charge $Z = 2+ - 4+$ and gain additional energy in the sheath potential in front of the targets shifting the thermal energy distribution by $3ZkT_e$ (T_e: electron temperature). As a consequence the sputtering threshold of W is lowered considerably to $T_e \approx 5$ eV compared $T_e \approx 25$ eV for pure D plasmas.
In tokamaks with graphite surfaces, carbon plays an important role as a radiator in the plasma boundary leading to a substantially reduced heat load on the target plates. With a first wall of a high Z-material, the carbon radiator has to be replaced by Argon or Neon for example [23, 24]. While with graphite surfaces, a degree of self regulation between erosion and radiation can be observed, the addition of a noble gas has to be carefully controlled. To this end, the measurement of thermo currents through the target plates appears to be a robust method for the control of the divertor electron temperature [23]. The introduction of noble gases primarily leads to an increase of the W yield but at the same time the reduction of the edge temperature by radiative cooling can compensate the effect. If the necessary impurity concentration is assumed to vary with the nuclear charge Z as $1/Z$, there is virtually no difference in W sputtering rates for Ne, Ar and Kr at their corresponding (ITER relevant) concentration in the range of typical SOL and divertor temperatures [23, 25]. Experimentally, a reduction as well as an increase of the W yield has been observed in ASDEX Upgrade [26, 27] in discharges with Ne injection depending on the actual divertor plasma parameters. A strong increase of the Mo source in the divertor was observed in Alcator C-Mod when using Ar puffing [28]. Under limiter conditions, a reduction of the edge temperatures by neon seeding reduced the erosion yields

in TEXTOR, but the overall tungsten release did not decrease as additional sputtering of tungsten by neon occured [29].
Currently the ELM type I H-Mode is foreseen as ITER baseline scenario. Recent investigation with high temporal resolution to resolve the erosion flux during ELMs [30, 31] at the low field side ICRH and guard limiters and in the divertor of ASDEX Upgrade reveal that a large fraction of the W erosion occurs during ELMs. Fig.1 a) shows the W fluence during type-I ELMs at an ICRH guard limiter versus the ELM energy loss of the plasma W_{ELM} for discharges with NBI heating and additional ICRH heating [30]. It rises linearly with the ELM energy, which is on the other hand almost inversely proportional to the ELM frequency $1/f_{ELM}$ [32], and therefore the fraction of W erosion during ELMs is almost constant over the ELM frequency ($70 \pm 10\%$). Fig. 1 b) gives the effective W yield during and in-between ELMs, demonstrating that not only the larger particle flux during the ELMs leads to a larger W flux, but also the energy of impinging particles is larger, which is reflected in a higher W yield [30]. In the divertor, the fraction of in-ELM erosion varies more strongly because the inter ELM T_e can vary much stronger. Here the ratio of in-ELM to inter-ELM W erosion fluency amounts 50% - 90% [31].

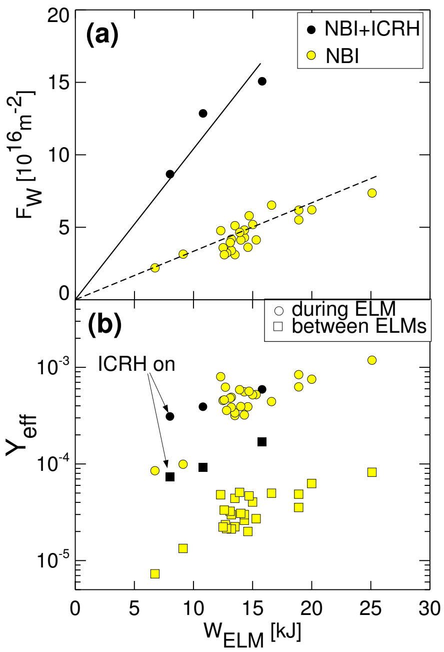


Figure 1: a) Tungsten fluence F_W during type-I ELMs at an ICRH guard limiter versus the ELM energy loss of the plasma W_{ELM} for discharges with NBI heating and additional ICRH heating in ASDEX Upgrade. b) Effective tungsten yield during and in-between ELMs [31].

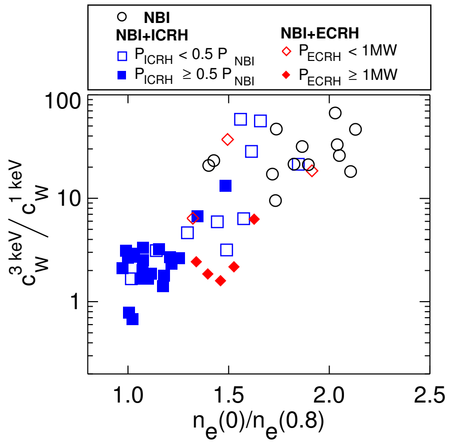
A specific effect in the main chamber are filaments with a helical structure [33] transporting plasma to the wall. Impurities in this typically $T_i \approx 50$ eV plasma [34] have the potential to sputter tungsten. However their contribution to the total impurity flux is not yet clear, due to the strong temporal and spatial variation making their diagnostic difficult. Erosion by CX-particles may also lead to substantial tungsten fluxes if, in some confinement scenarios, the plasma edge temperature is relatively high [24]. Another source for erosion are fast particles from additional heating. Fast deuterium ions generated by neutral beam injection (NBI) can be lost due to drifts or MHD-effects, like ELMs [30]. In contrast, for the case of ICRH, tungsten erosion is not dominated by fast particles [35, 36, 30, 37]. Local observation on limiters shows tungsten influxes created immediately after the ICRH is switched on, which suggests that an increase of the sheath potential in the neighbourhood of the antenna is responsible. However, variations of the field line angle with respect to the Faraday screen - which should lead to a strong variation of the electric potential - do not show a clear effect.
The erosion of high-Z material by ICRH (and of course by other sources) can strongly be reduced by a boronisation of the surfaces [38, 35, 39]. However dedicated experiments in Alcator C-Mod revealed that the B coating is removed after a few discharges on heavily loaded areas [35, 36]. Experiments in ASDEX Upgrade showed that a global equilibrium for the W influx and W concentration is reached after about 100 discharges after boronisation, being smaller than the typical distance in-between boronisations of about 200 discharges [16]. This value is consistent with estimates on B erosion based on measured deuterium fluxes and B layer thicknesses, using B erosion rates of 5×10^{-4} in the divertor and 10^{-2} at main chamber PFCs. In order to consolidate the results on the W behaviour in ASDEX Upgrade, the deposits at a major part of the PFCs have been removed an the machine start-up was recently performed without boronisation. As expected, no major difference compared to operation with an aged boronisation is observed [17].

3 Edge and Core transport

Impurity transport can be divided in the region of field lines intersecting with PFCs, the so called scrape off layer (SOL), and the confined plasma. Experimental investigations on transport in the SOL region are scarce and rather indirect. By use of a sublimation probe, W was injected in the divertor and at the midplane SOL of ASDEX Upgrade [40], revealing a divertor retention of 16 in a medium density H-mode discharge. This experiment elucidates the observation that although the divertor is usually the largest gross source in Alcator C-Mod [13] as well as in ASDEX Upgrade [31], the impurity content is governed by the main chamber sources.

In H-modes, one has to subdivide the confined plasma into the pedestal and core regions. In the reactor relevant regime with type-I ELMs, the pedestal, with its steep pressure gradient, collapses during an ELM and a substantial part of the pedestal plasma is ejected (see also Sec. 2). In between ELMs, tungsten moves into the pedestal region due to a strong inward particle drift [41]. If the next ELM comes in due time then this tungsten is removed, before further penetrating towards the central plasma. An increase of the ELM frequency and a reduction of the W content can be achieved by external means, so called ELM pace making [42, 24, 25].

In the core plasma, an inward particle drift can lead to accumulation in the centre. In a simple picture, neoclassical theory predicts a diffusion coefficient which scales with the ion charge as $1/Z^2$ while the inward drift is proportional to $1/Z$. If the deuterium density profile is not particularly flat, this, in general, leads to accumulation of high-Z elements in the core. The neoclassical accumulation has been experimentally observed in several occasions [43, 25, 44, 45, 46]. However, if the heat flow in the core is sufficiently high, anomalous transport can easily exceed the neoclassical effects, especially that of high-Z ions because of the $1/Z$ and $1/Z^2$ scaling, respectively. One can experimentally demonstrate this effect by, for example localized central ECRH which stimulates the anomalous turbulent transport [47, 25, 48]. Fig. 2 presents the peaking of the W concentration (c_W) as a function of background density peaking at ASDEX Upgrade. Discharges with pure NBI heating (black circles) show strongest peaking, whereas central ECRH reduces the c_W peaking significantly already at low additional heating power [25]. There are two mechanisms anomalous transport can influence the W-profile. On the one hand it can overcome the Ware-pinch of the main ions, thus flattening the deuterium density profile, while on the other hand anomalous transport can directly determine the W flux and W density profile. Both of these effects have been experimentally substantiated using the trace impurity Si [49].



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4 Discussion and Outlook

Experiments at the full W device ASDEX Upgrade have demonstrated that in high density scenarios with sufficient central heating the tungsten erosion of the surfaces and its concentration in the central plasma can be kept sufficiently low, even under un-boronised conditions. However, a large increase of the high-Z particle influx during ICRH can lead to strong radiation losses and even to the degradation of confinement when used as sole additional heating (Alcator C-Mod, [36]), or - when used in combination with NBI - to a back-transition to L-Mode (ASDEX Upgrade [17]). The only operational regime found up to now for full tungsten ASDEX Upgrade, where the ICRH is able to increase the plasma energy significantly, is the operation at high gas puffing rates with an increased gap between separatrix and antennae and the limited theoretical understanding poses a challenge for the combination of ICRH and a high-Z wall.

Although an extrapolation towards a burning plasma device is far from being straight forward, several arguments support the applicability of high-Z PFCs in such a device:

Recent DIVIMP calculations based on B2-Eirene simulations suggest edge W concentrations below 10^{-4} even for a full W coverage of the ITER PFCs [50].

Frequent ELMs as a necessary prerequisite for sufficiently low tungsten content in the plasma go in line with the demand of ITER or DEMO to keep the energy per ELM small. Suppression of ELMS by increasing the overall edge transport with edge resonant magnetic field perturbations [51] may also be a solution, however the influence of this method on impurity transport is not yet clear, since a higher Z_{eff} is reported compared to similar discharges without edge perturbation [52].

Transport simulations for ITER with the ASTRA code using the GLF23 model [53] suggest that the anomalous particle transport should be significantly larger than the neoclassical one, leading only to very moderate peaking of the W concentration [54, 25]. Moreover, recent theoretical work [55] shows two dominant turbulent transport mechanisms for high Z-ions. Neither of them results in a substantial accumulation of tungsten with respect to the deuterium density.

In summary, this leads to the expectation that peaked W concentration profiles are unlikely to exist in the type-I-ELMy H-Mode of a burning device. Combined with a high density, low temperature edge plasma, an all tungsten device seems to be feasible [24], although the details of the radial transport still have to be quantified. In the future, emphasis has to be put on the development and optimization of reactor relevant scenarios which incorporate the experiences with and the boundary conditions set by the high-Z PFCs.

References

[1] Aymar R, Barabaschi P and Shimomura Y 2002 *Plasma Phys. Control. Fusion* **44** 519

[2] Toschi R *et al.* 2001 *Fus. Eng. Des.* **56-57** 163

[3] Philipps V *et al.* 2000 *Plasma Phys. Control. Fusion* **42** B293

[4] Roth J *et al.* 2005 *J. Nucl. Mater.* **337–339** 970–974

[5] Federici G *et al.* 2001 *J. Nucl. Mater.* **290-293** 260

[6] Federici G *et al.* 2003 *J. Nucl. Mater.* **313–316** 11

[7] ITER physics basis editors 1999 *Nucl. Fusion* **39** 2137

[8] Bolt H, Barabash V, Federici G, Linke J, Loarte A Roth J and Sato K 2002 *J. Nucl. Mater.* **307-311** 43

[9] Yoshida N 1999 *J. Nucl. Mater.* **266 – 269** 197

[10] Suckewer S and Hawryluk R J 1978 *Phys. Rev. Letters* **40** 1649

[11] Arunasalam V *et al.* 1978 *Proc. 8th Conf. EPS, Prague 1977* vol 2 (Geneva: EPS) pp 17

[12] Apicella M *et al.* 1997 *Nucl. Fusion* **37** 381

[13] Lipschultz B, Pappas D, LaBombard B, Rice J, Smith D and Wukitch S 2001 *Nucl. Fusion* **41** 585

[14] Pospieszczyk A *et al.* 2001 *J. Nucl. Mater.* **290–293** 947

[15] Neu R *et al.* 1996 *Plasma Phys. Control. Fusion* **38** A165

[16] Neu R *et al.* 2007 *J. Nucl. Mater.* **363-365** 52

[17] Neu R *et al.* 2007 *submitted to Plasma Phys. Control. Fusion*

[18] Matthews G *et al.* 2007 *Physica Scripta* **T128** 137

[19] Neu R 2006 *Physica Scripta* **T123** 33–44

[20] Eckstein W, Bohdanski J and Roth J 1991 *Atomic and Plasma-Material Interaction Data for Fusion* vol Vol.1, Suppl. to Nucl. Fusion (Vienna: IAEA) p 51

[21] Naujoks D, Roth J, Krieger K, Lieder G and Laux M 1994 *J. Nucl. Mater.* **210** 43

[22] Naujoks D *et al.* 1996 *Nucl. Fusion* **36** 671

[23] Kallenbach A *et al.* 2005 *J. Nucl. Mater.* **337-339** 732

[24] Kallenbach A *et al.* 2005 *Plasma Phys. Control. Fusion* **47** B207

[25] Neu R *et al.* 2005 *Nucl. Fusion* **45** 209

[26] Thoma A *et al.* 1997 *Plasma Phys. Control. Fusion* **39** 1487

[27] Krieger K, Maier H, Neu R and ASDEX Upgrade Team 1999 *J. Nucl. Mater.* **266–269** 207

[28] Kurz C *et al.* 1995 *J. Nucl. Mater.* **220-222** 963

[29] Unterberg B *et al.* 1999 *J. Nucl. Mater.* **266–269** 75

[30] Dux R *et al.* 2007 *J. Nucl. Mater.* **363-365** 112

[31] Dux R, Pugno R, Pütterich T, Bobkov V, Kallenbach A, Neu R and ASDEX Upgrade Team 2007 *Europhysics Conference Abstracts (CD-ROM, Proc. of the 34th EPS Conference on Plasma Physics, Warsaw, 2007)* (Geneva: EPS) pp P–1.052

[32] Herrmann A 2002 *Plasma Phys. Control. Fusion* **44** 883

[33] Kirk A *et al.* 2006 *Plasma Phys. Control. Fusion* **48** B433

[34] Herrmann A, Kirk A, Schmid A, Koch B, Laux M Maraschek M and Mueller H W o 2007 *J. Nucl. Mater.* **363-365** 528

[35] Lipschultz B *et al.* 2006 *Physics of Plasmas* **13** 056117

[36] Lipschultz B *et al.* 2007 *J. Nucl. Mater.* **363-365** 1246

[37] Dux R *et al.* 2005 *J. Nucl. Mater.* **337-339** 852

[38] Apicella M *et al.* 2003 *J. Nucl. Mater.* **313-316** 269

[39] Bobkov V V *et al.* 2007 *J. Nucl. Mater.* **363-365** 122

[40] Geier A, Maier H, Neu R, Krieger K and ASDEX Upgrade Team 2002 *Plasma Phys. Control. Fusion* **44** 2091

[41] Dux R, Giroud C, Neu R, Peeters A G, Stober J, Zastrow K D, contributors to the EFDA-JET work programme and ASDEX Upgrade Team 2003 *J. Nucl. Mater.* **313–316** 1150

[42] Lang P T *et al.* 2005 *Nucl. Fusion* **45** 502

[43] Rapp J *et al.* 1997 *Plasma Phys. Control. Fusion* **39** 1615

[44] Dux R, Giroud C, Zastrow K D and contributors to the EFDA-JET work programme 2004 *Nucl. Fusion* **44** 260

[45] Pacella D, Fournier K, Zerbini M, Finkenthal M, Mattioli M, May M and Goldstein W 2000 *Phys. Rev. E* **61** 5701

[46] Rice J E, Bonoli P, Marmar E, Wukitch S, Boivin R L, Fiore C, Granetz R, Greenwald M *et al.* 2002 *Nucl. Fusion* **42** 510

[47] Neu R *et al.* 2002 *Plasma Phys. Control. Fusion* **44** 811

[48] West W *et al.* 2002 *Physics of Plasmas* **9** 1970

[49] Dux R, Neu R, Peeters A G, Pereverzev G, Mück A, Ryter F, Stober J and ASDEX Upgrade Team 2003 *Plasma Phys. Control. Fusion* **45** 1815

[50] Schmid K, Krieger K, Kukushkin A and Loarte A 2007 *J. Nucl. Mater.* **363-365** 674

[51] Evans T, Moyer R, Watkins J, Osborne T, Thomas P, Becoulet M *et al.* 2005 *Nucl. Fusion* **45** 595

[52] Fenstermacher M *et al.* 2007 *J. Nucl. Mater.* **363-365** 476

[53] Pereverzev G N, Angioni C, Peeters A G and Zolotukhin O V 2005 *Nucl. Fusion* **45** 221

[54] Dux R *et al.* 2005 *Proc. of the 20th IAEA Conference Fusion Energy (CD-Rom), Vilamoura, Portugal, November 2004* vol IAEA-CSP-25/CD (Vienna: IAEA) pp IAEA–CN–116/EX/P6–14

[55] Angioni C and Peeters A G 2006 *Phys. Rev. Lett.* **96** 095003–1