PLASMA-WALL INTERACTIONS IN MAGNETICALLY CONFINED FUSION PLASMAS

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ABSTRACT

The control of wall loads in fusion devices, in particular with respect to the life time limitations of wall components due to material erosion and migration, will be decisive for the realisation of a fusion power plant operating in steady state, whereas in a pulsed experiment like ITER the primary goal for plasma-wall interaction is the achievement of high availability of the device. The article describes the grand challenges of plasma-wall interactions research along the needs for ITER. Addressed are questions related to material limitations, erosion- and transport processes, tritium retention and transient heat loads.

I. INTRODUCTION

A fusion device cannot work without plasma-wall interactions. Two essential functions have to be provided via the interaction of the plasma with the wall:
a) the exhaust of heating power to the plasma (mainly from alpha-particle heating) and
b) the exhaust of alpha-particles (Helium ash) to avoid fuel dilution.
A large variety of processes is involved in plasma-wall interactions. They are determined by the properties of wall materials, plasma edge parameters (e.g. temperature, density, radiation), heat and particle transport and the plasma species (hydrogen and impurities). Thus the research field of plasma-wall interactions is interdisciplinary and comprises plasma physics, surface physics, physics of atoms and molecules, chemistry and materials sciences. The following describes the concepts for controlling plasma-wall interaction being developed in today's fusion devices.

II. PARTICLE AND HEAT EXHAUST IN ITER

In ITER the total heating power of P_{alpha} = 100 MW plus an external heating of about P_{heat} = 50 MW has to be exhausted via radiation and plasma convection onto wall components. Linked to this is a production rate of Helium of about 2 10^{20} particles per second (about 1 mg/s), which have to be pumped out continuously.

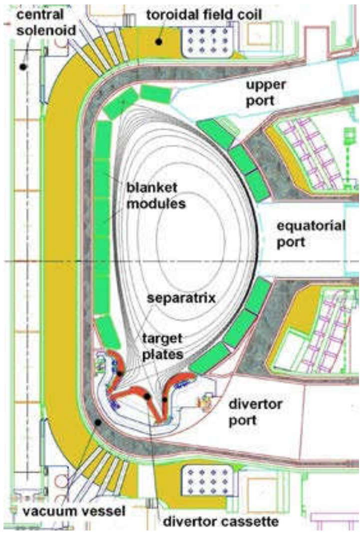

Fig. 1 *Magnetic flux surfaces shown in a poloidal cross section of ITER; the highest loads are located at the target plates*

<u>Helium exhaust</u> will be solved in ITER by pumping the neutral particles from the divertor chamber. The plasma flow into the divertor along the so called scrape-off layer (SOL) and the successive neutralisation of particles on the target plates provides a certain concentration of neutral particles inside the divertor (deuterium, tritium, helium and other impurities). The resulting gas pressure will be sufficient to achieve efficient pumping through channels below the divertor chamber. In ITER it is expected to achieve a characteristic exhaust time for helium of about 15 seconds which will lead in steady state to a helium concentration of about 4 %. This helium concentration in the plasma centre is low enough for providing a significant margin for other impurities, e.g. eroded particles or injected particles for the purpose of radiation cooling.

The required <u>heat exhaust</u> is more difficult to achieve because the radial extend of the SOL is only in the order of a centimetre and thus generates high heat load densities on a rather small area on the target plates. The heat load density on plasma wetted areas can be reduced by up to a factor of 6 by inclining the target plates. Even then, the total area is only 6-8 m^2. In the reference scenario for ITER about half of the total heating power $P_{alpha} + P_{heat}$ is convected to the target plates leading to a power density of up to 10 MW/m^2. Prototype modules for divertor plates which can take these loads under cycling conditions have already been manufactured and tested successfully (Fig. 2).

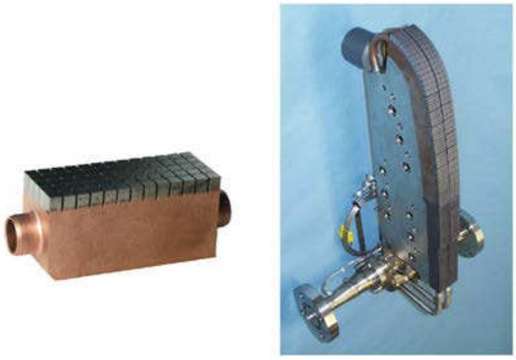

Fig. 2 *(left) Monoblock from graphite (CFC) brazed on copper, as a water cooled element for the target plate, and (right) a prototype target plate module (vertical target) made from tungsten and CFC; these modules fulfill all requirements for heat exhaust under thermal cycling loads up to 23 MW/m^2s* [1]

The other half of the heating power is radiated onto the whole inner wall on an area of 680 m^2 (radiation cooling). The corresponding power density of 0,11 MW/m^2 can be extracted rather easily via water cooled wall elements. A significant fraction of the radiation cooling is provided by electromagnetic radiation from excited impurity species in the plasma. Thus radiation cooling from impurities can be seen as a beneficial aspect of erosion of wall materials. This effect has to be taken into account when different wall materials are considered. It is important that the radiation from impurities is concentrated mainly on the plasma edge region, which is the case for light impurities. Normally the amount of eroded material is not sufficient to provide the necessary radiation level, in particular not in the case of heavy metals as wall material. By injecting additional impurities, preferably noble gases, we have the possibility to adjust the radiation level and thus to control the heat load to the target plates [2].

With these concepts we can exhaust the average heat loads in ITER reliably. Much more critical are the transient loads. These loads are caused by plasma instabilities, like disruptions (e.g. at the density limit) or Edge Localized Modes (ELMs)[3]. Transient loads lead to enhanced erosion, possible excess loads (melting) or fatigue effects on thermo-mechanical properties.

Methods are developed to mitigate the peak loads caused by disruptions [4]. ITER is in contrast to a fusion power plant, where disruptions should be avoided at all, an experiment in which a certain number of disruptions are unavoidable when exploring the operational limits of the device.

Periodic events, like ELMs, are more difficult to cope with, since they are linked closely to the plasma scenario and the corresponding energy confinement. For ITER the standard scenario is the H-Mode plasma where ELM activity is always present. An example for ELM activity in JET [5] is shown in Fig. 3.

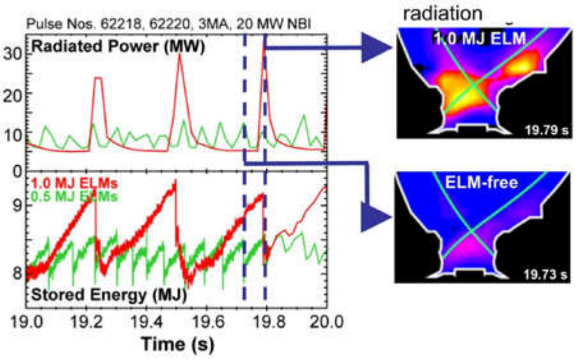

Fig. 3 *Plasma discharges in JET with ELM-activity; comparison of two discharges with a) large low frequency ELMs (heat exhaust 1 MJ per ELM, dark line) and b) small and fast ELMs (0,5 MJ per ELM); the curves down left show the energy content (pressure)at the plasma boundary and the upper left curves show the corresponding radiation level; on the right the 2-dimensional view of the radiation pattern from carbon inside the divertor is shown during and before an ELM-crash* [6]

Large ELMs in JET can lead to a load on the target plate up to 0.5 MJ/m^2, causing a transient increase of the target temperature up to 2500 °C as observed for the operation with graphite target plates[6]. Extrapolations to ITER show that the transient loads due to ELMs must be limited in order to achieve a sufficient life time of the target plates. In ongoing research different strategies are pursued to achieve this goal.

There is a link between good energy confinement (and steep pressure gradients at the boundary) and ELM energy. A possible way for optimizing plasma-wall interactions with ELMs is to choose plasma-scenarios with an optimum balance between good energy confinement and ELM energy, with the consequence of accepting some decrease in energy confinement and thus energy amplification. In this respect, a possible way is the use of small type III ELMs induced by strong gas injection[7].

Another attempt is the development of methods for ELM-mitigation: A pace-maker technique employing the injection of pellets or the recently developed technique of short vertical plasma displacements (kicks) can trigger ELMs before they acquire too much energy[8]. Another method is based on resonant magnetic perturbations (RMP) imposed by external coils. By this method it has been shown that the magnetic field at the boundary is ergodized (chaotic field lines) leading to a change in transport and eventually to mitigation or even suppression of ELMs[9,10]. This concept is also studied with the Dynamic Ergodic Divertor on the tokamak TEXTOR [11].

III: EROSION AND DEPOSITION

Plasma-Wall Interactions lead to significant erosion processes at plasma wetted areas. Some erosion mechanisms are caused directly by excess heat loads, like melting or sublimation. Normally the more important erosion mechanisms are linked to the particle fluxes to the wall: physical sputtering and chemical reactions.

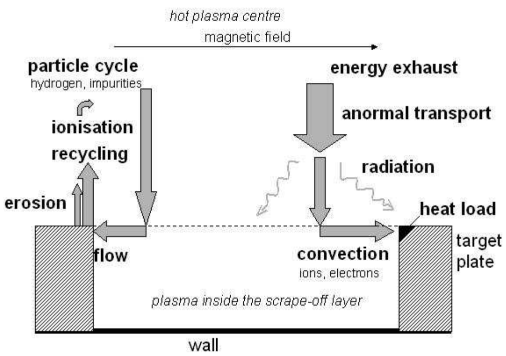

Fig. 4 *Schematic 2-d view of particle recycling and heat exhaust; inside the SOL the plasma flows to both sides along the magnetic field lines towards the target plates*

Erosion processes release particles from the wall, which are then ionized when entering the plasma or dissociated and ionized in the case of molecules. Transport processes in the plasma lead the impurity particles back to the wall - in most cases even to the place of their origin due to the specific guiding effect of the magnetic field lines. In contrast to the energy, the particles are cycling (Fig. 4). Hydrogen is neutralized at the target plate and is re-emitted as atom or molecule – this is called hydrogen-recycling. Impurities are eroded and deposited.

Physical sputtering of wall materials is caused by the bombardement of plasma ions. The sputtering yield depends on the energy of the ions, the mass ratio of projectile and target and the surface binding energy of the target particles. Fig. 5 shows some yields for deuterium on various materials as a function of the energy of the projectiles.
The energy with which the ions impinge on the target is primarily given by the plasma temperature. Typical values are between 2 and 100 eV. However, the sheath potential accelerates the ions proportional to their charge state (typically about four times for singly charged ions). As a consequence, in relatively cold plasmas, typical e.g. for high density divertors, already rather small amounts of impurities can dominate the overall sputtering yield compared to the deuterium/tritium ions.

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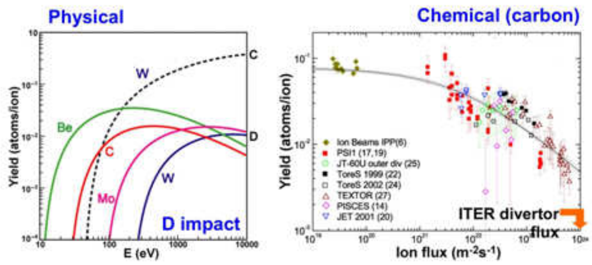

Fig. 5 *Erosion yields of wall materials for physical sputtering by deuterium as a function of the energy of the projectiles (left)* [12] *and chemical erosion of graphite as a function of the flux density of deuterium (right)* [13]*; the dashed line shows the physical sputtering of tungsten by carbon*

The effective yields for physical sputtering of carbon are in the range of 1-2 %. In case of tungsten the yields are much lower down to values of 0.01 to 0.001 % governed by impurities as carbon or injected impurities as neon for the purpose of radiation cooling[14].

For some wall materials also chemical erosion processes are significant, as is the case for graphite, where chemical erosion is as important as physical sputtering. Particle flux density and surface temperature of the material are the most important parameters. The yield of chemical erosion shows a strong decay with high flux densities as was demonstrated in a multi-machine comparison (Fig. 5). The highest flux densities were provided by the tokamak TEXTOR, where special means (limiter locks) allow measurements with very high flux densities close to the conditions of ITER[15].

Erosion in the divertor plays the main role for the life time of the target plates and contributes to the impurity contamination of the plasma. Also the erosion on the large main chamber contributes to the impurity contamination. Erosion on the main chamber wall can be caused by fast neutrals (via charge exchange processes) or by impinging ions due to enhanced radial transport. The latter may go mainly via convective cells which are formed due to instabilities in the edge plasma. This kind of turbulent transport is an important issue of ongoing research and is also the cause for some uncertainty in the extrapolations of global erosion results to ITER.

The eroded impurities can have a substantial influence on the plasma characteristics. The particles are ionized and excited by electron collisions, which can lead depending on the kind of impurities and their concentration to substantial radiation and thus cooling of the plasma. A lowered plasma temperature can have repercussion on the erosion yields. The choice of wall material and the characteristics of the plasma close to the wall are therefore coupled nonlinearly.

The deposition of eroded wall material plays an important role for the extension of the life time of wall components. In some areas the deposition rates are smaller than the erosion rates (net-erosion zones), in other areas they are larger (net-deposition zones) and at the boundary of these areas there is even balance between erosion and deposition (Fig. 6).

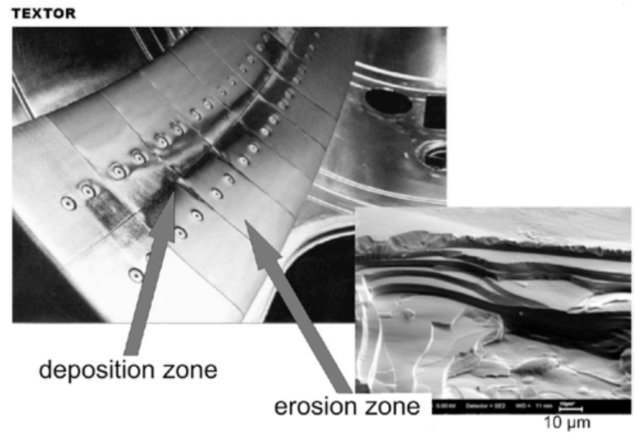

Fig. 6 *Formation of layers by deposition of eroded carbon on limiters in the tokamak TEXTOR; the electron microscope image shows layers structures of about 4 µm thickness* [16]

Generally the local and global deposition processes lead to a strong reduction of net erosion - the deposition processes therefore represent an important self healing mechanism for highly loaded components, like divertor target plates or limiters.

In the zones with net deposition, however, accumulative layers develop. These can store deuterium and tritium by co-deposition. In fusion experiments, in particular at less loaded surfaces (remote areas), carbon layers with 100% hydrogen content have been observed. Based on these facts it is criticized using carbon at all for plasma facing components in ITER.

IV. WALL MATERIALS FOR ITER

A combination of different materials is intended for the plasma-loaded surfaces in ITER. A first design has foreseen the materials beryllium, tungsten and graphite, as they fulfill the requirements with their specific characteristics for the very different loads at different wall areas (Fig. 7).

This material choice was found on the basis of the following criteria:

Beryllium: Radiation by impurities in the plasma centre, as it can occur by heavy elements, is unwanted. By the lining of the inner wall with as light an element as possible this condition can be fulfilled. Beryllium is the lightest metal, these wall components can be made of. Additionally, the property of beryllium as an oxygen getter, turned out to be advantageous for the vacuum characteristics and thus for the tokamak operation.

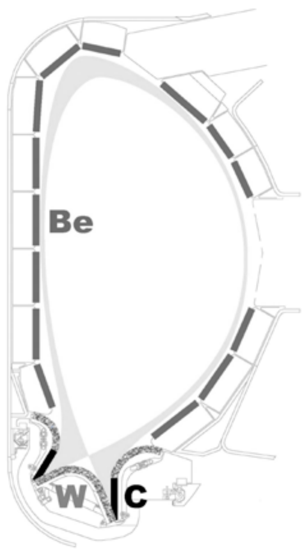

Fig. 7 *Combination of wall materials for plasma-loaded surfaces in ITER according to the first design: Beryllium in the main chamber, graphite for the target plates in the divertor and tungsten for the remaining surfaces in the divertor*

Graphite: With a melting point of only 1560 K beryllium cannot be used in the most loaded areas of the divertor. Here graphite is ideal a material. It is “forgiving”, because it does not melt with overloading but only sublimates (3825 K). This characteristics is in particular of high relevance in case of experiments like ITER, which will go to the operational limits. The largest disadvantage of graphite is its rather strong erosion, and associated to this the tritium retention by co-deposition. Therefore, the use of graphite should be restricted to a minimum area of high loads where the properties of graphite are required.

Tungsten: A combination of high melting point and very small sputtering yield is given by tungsten. The sputtering yield in the range of 10^{-5} to 10^{-4} for those particle energies as expected in ITER. Tungsten can be used for medium and high load areas instead of graphite, however, at the risk of melting in case of off-normal operation.

At present the risk of tungsten melting seems to less dangerous compared to the problem of tritium retention in connection with graphite as wall material, because an accumulation of the mobilizable tritium inside the deposited layers could exceed the maximum allowed amount (at present 700 gr). In this case the machine would have to be shut-down for the period of cleaning or conditioning with possibly substantial disadvantages for the availability of the whole experiment[17]. Thus, graphite as a divertor material is still an option, but most likely the ITER divertor will be covered fully by tungsten.

In any case some wall components (e.g. the divertor plates) will have a limited life time and have to be exchanged occasionally. However, a too large replacing frequency would impair also the availability of the plant. The main uncertainties lie here in limited knowledge about the erosion by transient loads, like ELMs or disruptions. In ITER the exchange of the divertor after some years of operation is made possible by a cassette design for the divertor.

V. STRATEGIES FOR MINIMIZING THE RISKS

The risk of insufficient availability is to be minimized by a coordinated effort with experiments and calculations for plasma-wall interactions. This task is addressed worldwide with joint plasma-wall interaction experiments on many facilities facilities[18]. The strategy for minimizing the risk contains both the improvement of the wall concept with the selected materials (W/Be-wall in JET) and the parallel development of alternative material combinations (e.g. full metal wall in ASDEX-upgrade).

Carbon

The re-erosion of deposited carbon must still be understood better, in order to derive from it concepts for the minimization of hydrogen containing layers. We can forecast the net-erosion of the target plates in ITER based on our current knowledge by computer models. Fig. 8 shows such a calculation of the ERO code for the outer divertor[19].

The main zones of net-erosion and net-deposition concentrate on a few centimeters in the proximity of the magnetic separatrix, which intersects the target plate. Eroded particles are transported preferentially in the direction of the magnetic field lines and are deposited with a small offset on the target plate again. Finally 94% of the eroded carbon particles are re-deposited locally. The remaining local loss rate amounts to maximally 3 nm/s. This means that more than 3000 ITER discharges of 8 minutes duration are possible until a 5 mm thick layer is eroded at the target plate. According to this calculation we do not have a life time problem.

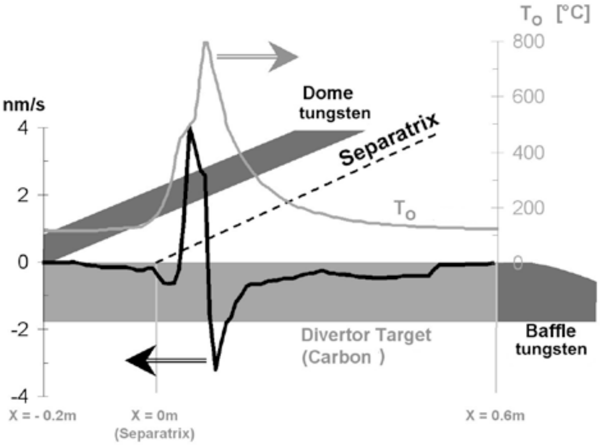

Fig. 8 *Calculation of erosion, deposition and surface temperatur on an ITER target plate ITER with the computer code ERO; the direction of the magnetic field line is indicated by the separatrix*

About 6 % of the eroded carbon particles leave the target region and deposit at other surfaces. These in such a way formed layers will be mainly responsible for the unwanted tritium retention, contrary to the deposited layers on the target plate, which can take up little tritium only because of the high surface temperatures.

The computations still suffer from uncertainties due to limited knowledge in our understanding of some processes:

- Control and effect of transient loads by ELMs and disruptions
- Effect of eroded beryllium from the main chamber on the erosion behavior of carbon in the divertor
- Data on the probability of re-erosion of deposited carbon and its migration behavior to remote or hidden surfaces
- Understanding of mixed material systems in tokamaks
- Retention of tritium by beryllium layers.

Tungsten - Beryllium

Recently the JET main chamber wall has been covered fully by beryllium and the divertor is now made fully of tungsten (the “ITER-like wall”: ILW) [20,21,22]. The main driver for the installation of the JET-ILW was the expected reduction in fuel retention. The expectations have been met: with the JET-ILW the long-term fuel retention has been reduced by more than one order of magnitude with respect to JET-C[23]. The remaining retention mechanisms are now co-deposition with Be and, of less importance, hydrogen implantation in Be and W. A strong reduction of divertor radiation has been found with the ILW. This is a consequence of the absence of carbon radiation and the low radiation potential of beryllium and deuterium inside the divertor[24]. With incident power densities of 5-10MW/m^2 or higher, divertor target plates of large tokamaks are driven close to the material limits. Moreover, the possible occurrence of thermo-mechanical fatigue cannot be ignored.

All High-Z metal wall

In view of the necessities for DEMO also the avoidance of Beryllium is an important matter. This could be achieved by using a full tungsten lining of the inner wall and the divertor. The Tokamak ASDEX-upgrade took the leading task developing an integral solution with main chamber wall and divertor covered fully with tungsten. (Fig.. 9) [25].

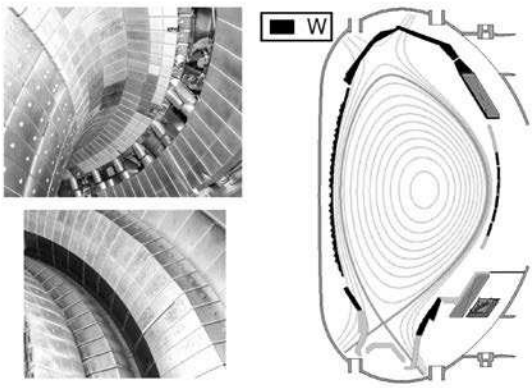

Fig. 9 *Lining of the inner wall of ASDEX-Upgrade with tungsten*

As a main result the transport of eroded material to remote areas has been reduced significantly. The deposited layers in the inner divertor were reduced by a factor of ten. The deposits are mainly B and C from boronization and residual C in the machine[26]. The absence of carbon radiation requires external impurity seeding for achieving a sufficient power exhaust via radiation. This has been demonstrated successfully by feed-back controlled in jection of different gas species[27].

VI. CONCLUSIONS

The primary goal of ITER - 500 MW fusion power for 8 minutes of pulse duration and significant alpha-particle heating - will be achieved with the available concepts for the control of plasma-wall interaction. The open questions of plasma-wall interactions refer to problems with the realization of a continuously operated fusion device. In ITER these questions have only relevance for the availability of the experiment.

A high availability is crucial for the economy of a fusion power plant. Fusion research enters now into a new era, in which the main question is not anymore whether we can produce a burning fusion plasma – this we know in principle based on the results of JET[28]. Now the new goal is the demonstration of an economical long-term operation. The research field plasma-wall interaction is for this a key topic.

The current research in the area of plasma-wall interactions concentrates in the coming years on decisions related to the construction of ITER. On the long run the research will contribute to the improvement of the ITER operation as well as on the preparatory work for the next step – the construction of the first fusion power station DEMO.

The stellarator development will play an important role for the plasma-wall interaction research regarding the development of concepts for steady state operation of fusion devices. Stellarators, like the Wendelstein 7-X, work contrary to tokamaks not in a pulsed manner. This makes stellarators particularly relevant for investigations of plasma-wall interactions during continuous operation. The concept of an island divertor will impose new questions about heat and particle exhaust. However, in general the problems of plasma-wall interactions are alike in stellarators and tokamaks.

Experiments with new wall concepts on a large scale are very important. However, the complex questions of plasma-wall interaction with its various aspects in plasma physics, surface physics, atomic and molecule physics, chemistry as well as material sciences cannot be solved alone on the large facilities, like JET or later ITER. Flexible smaller and medium sized plasma devices (tokamaks, stellarators, linear plasmas) as well as specialized laboratory equipment (e.g. test facilities for thermal loads, material laboratories) represent the actual backbone for the study of plasma-wall interaction.

REFERENCES

/[1]/ G. Janeschitz et al., "Divertor design and its integration into ITER", Nucl. Fusion **42** (2002) 14-20

/[2]/ U. Samm et al., "Radiative edges under control by impurity fluxes", Plasma Phys. Contr. Fusion **35** (1993) B167

/[3]/ A Loarte et al., "Characteristics of type I ELM energy and particle losses in existing devices and their extrapolation to ITER", Plasma Phys. Control. Fusion **45** (2003) 1549-1569

/[4]/ M. Lehnen et al., „Runaway genereation during disruptions in JET and TEXTOR",J. Nucl. Mater. 390-391 (2009) 740

/[5]/ http://www.jet.uk/

/[6]/ A. Huber et al., "Plasma radiation distribution and radiation loads onto the vessel during transient events in JET", J. Nucl. Mater. 390-391 (2009) 830

/[7]/ J. Rapp et al., "Integrated scenario with type-III ELMy H-mode edge: extrapolation to ITER", 2009 Nucl. Fusion **49** 095012

/[8]/ P.T. Lang et al., "ELM pace making and mitigation by pellet injection in ASDEX Upgrade", Nucl. Fusion **44** (2004) 665-677

/[9]/ T.E. Evans et al., "Edge stability and transport control with resonant magnetic perturbations in collisionless tokamak plasmas", Nature Physics 2 (2006) 419

/[10]/ Y. Liang et al., "Active Control of Type-I Edge-Localized Modes with n=1 perturbation fields in the JET tokamak", PRL **98** 265004 (2007)

/[11]/ M. Lehnen et al., "Transport and divertor properties of the dynamic ergodic divertor", Plasma Phys. Control. Fusion **47** (2005) B237

/[12]/ W. Eckstein, „Computer simulation of ion-solid interaction“, Springer Verlag, Berlin (1991)

/[13]/ J. Roth et al., „Flux dependence of carbon erosion and implications for ITER“, J. Nucl. Mater. 337-339 (2005) 970

/[14]/ A. Kallenbach et al., „Tokamak operation with high-Z plasma facing components”, 32nd European Physical Society Conference on Plasma Physics (EPS) 2005 Tarragona, Spain, to be published in Plasma Phys. Contr. Fusion

/[15]/ A.Pospieszczyk et al., Phys. Scr. T 81 (1999) 48

/[16]/ P. Wienhold et al., „Short and long range transport of materials eroded from wall components“, J. Nucl. Mater. 313-316 (2003) 311

/[17]/ G Counsell et al., “Tritium retention in next step devices and the requirements for mitigation and removal techniques”, Plasma Phys. Control. Fusion 48 No 12B (2006) B189

/[18]/ J. Roth et al., “Recent analysis of key plasma wall interactions issues for ITER”, J. Nucl. Mater. 390-391 (2009) 1-9

/[19]/ A. Kirschner et al., "Simulation of the plasma-wall interaction in a tokamak with the Monte-Carlo code ERO-TEXTOR", Nucl. Fusion **40** 5 (2000) 989

/[20]/ V.Philipps V. et al 2010 Fusion Eng. Des. 85 1581

/[21]/ S.Brezinsek S. et al 2011 J. Nucl. Mater. 415 S936

/[22]/ G.Matthews G.F. et al 2012 Phys. Scr. T145 014001

/[23]/ T.Loarer T. et al 2013 J. Nucl. Mater. 438 S108

/[24]/ A.Huber, A. et al. 2013 J. Nucl. Mater 438 S139

/[25]/ A. Kallenbach et al., „Spectroscopic investigation of carbon migration with tungsten walls in ASDEX Upgrade“, J. Nucl. Mater. 363-365 (2007) 60

/26/ M. Mayer et al., “Carbon balance and deuterium inventory from a carbon dominated to a full tungsten Asdex Upgrade”, J.Nucl.Mater. 390-391 (2009) 538

/27/ A.Kallenbach, A et al. 2012 NF 52 122003

/28/ U. Samm, “Controlled Thermonuclear Fusion at the Beginning of a New Era”, Contemporary Physics **44** 3 (2003) 203-217