PLASMA-WALL INTERACTION

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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. In ITER the primary goal for plasma-wall interaction is the achievement of a high availability of this pulsed experiment. The article describes the grand challenges of plasma-wall interaction research along the needs for ITER and the strategies of ongoing research for further optimization of the design. Addressed are questions related to material problems, erosion- and transport processes, tritium retention in deposited layers and problems transient heat loads.

## I. INTRODUCTION

A fusion device cannot work without plasma-wall interaction. Two essential functions have to be provided via the interaction of the plasma with the wall:
a) the exhaust of the heating power to the plasma (mainly 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 interaction. These processes are determined by the choice of wall materials, the magnetic topology, plasma edge parameters (e.g. temperature, density, radiation) and impurities. Thus the research field plasma-wall interaction is interdisciplinary and comprises plasmaphysics, surface physics, atom and molecule physics, chemistry and materials sciences. The following describes the concepts which are presently developed in today's fusion devices to control plasma-wall interaction in ITER.

## II. PARTICLE AND HEAT EXHAUST IN ITER

In ITER a heating power of $P_{alpha}$ = 100 MW plus an external heating of about $P_{heat}$ = 50 MW has to be conducted via plasma-wall interaction to heat sinks inside the 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 pumped out continuously.

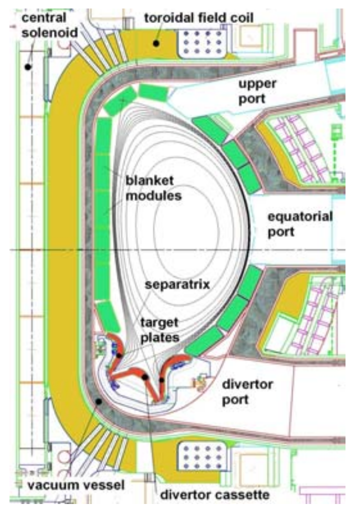


***Fig. 1*** *magnetic flux surfaces shown in the poloidal cross section of ITER; the highest loads are located at the target plates*

Helium exhaust has been solved in ITER by pumping the neutral particles from the divertor chamber. The plasma flow to the target plates in the divertor along the so called scrape-off layer (SOL) provides a certain concentration of particles inside the divertor. Neutralisation of particles at the target plate leads to the pressure of various gases (deuterium, tritium, helium and other impurities) needed for 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 so low that we have still a significant margin for other impurities, e.g. for eroded particles or for injected particles for the purpose of radiation cooling.

Heat exhaust is more difficult because the radial extend of the SOL is only in the order of a centimetre and thus directs the plasma flow to a rather small area on the target plates. By inclined target plates the loaded or wetted area can be extended by about a factor of 6. Nevertheless, the total area is still 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).

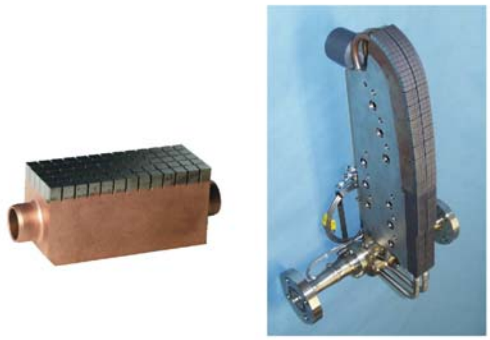

***Fig. 2*** *(left) Monoblock from Graphit (CFC) brazed on copper, as a water cooled element for for the target plate, and (right) a prototype target plate module (vertical target) made from tungsten and CFC; these modules fulfill all requirements for heat exhaust under thermal cycling loads up to 23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

The other half of the heating power is radiated onto the whole inner wall onto 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 positive aspect of the erosion of wall materials and has in this context to be considered when choosing the optimum wall materials. It is important that the radiation from impurities is concentrated mainly on the plasma edge region, which is the case for light impurities. In general 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. However, besides the average heat loads we have also to cope with transient loads, which are much more critical. These loads are caused by plasma instabilities, like disruptions (e.g. at the density limit) or Edge Localized Modes (ELMs)[3].

It is important to minimize the number of disruptions. For the remaining disruptions methods are developed to mitigate the peak loads caused by disruptions [4]. Unlike a fusion power plant, where disruption should be avoided at all, is ITER 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 critical, 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.

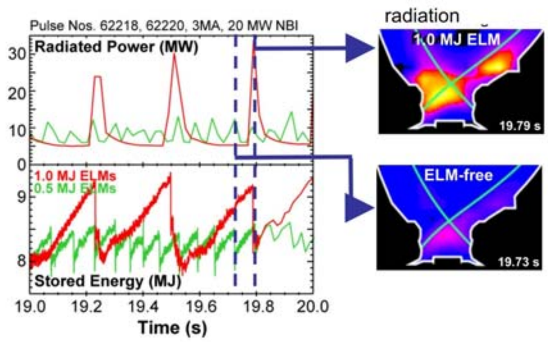


***Fig. 3*** *Plasma discharges in JET with ELM-activity; comparison of two discharges with a) large low frequency ELMs (heat exhaust 1 MJ per ELM, dark line) and b) small and fast ELMs (0,5 MJ per ELM); the curves down left show the energy content (pressure)at the plasma boundary and the upper left 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 lead to an load on the target plate of up to 0,2 $MJ/m^2$, causing a transient increase of the target temperature up to 2500 °C and correspondingly to the sublimation of carbon. 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 to optimize plasma-wall interaction 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. Another attempt is the development of methods for ELM-mitigation: A pace-maker technique employing the injection of pellets can trigger ELMs before they acquire too much energy[7]. Another method is the ergodization of the magnetic field at the boundary by external perturbation coils (chaotic field lines) [8]. A concept which is also studied with the new Dynamic Ergodic Divertor on the tokamak TEXTOR [9]. The promising attempts to solve the ELM-problem is a matter of ongoing research in present fusion experiments.

## EROSION AND DEPOSITION OF WALL MATERIALS

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

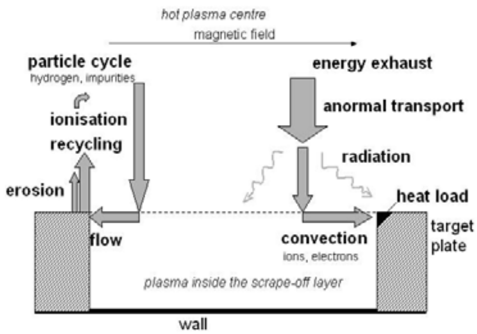


*Fig. 4 Schematic 2-d view of particle recycling and heat exhaust; inside the SOL the plasma flows to both sides along the magnetic field lines towards the target plates*

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

Physical sputtering of wall materials is caused by the bombardement by ions from the plasma. The sputtering yield depends on the energy of the ions, the mass ratio of projectiles and target and the surface binding energy of the 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 that 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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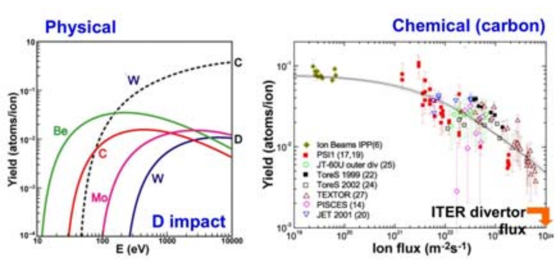


*Fig. 5 Erosion yields of wall materials for physical sputtering by deuterium as a function of the energy of the projectiles (left)* [10] *and chemical erosion of graphite as a function of the flux density of deuterium (right)* [11]*; the dashed line shows the physical sputtering of tungsten by carbon*

In present experiments the effective yields für 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 mainly by impurities as carbon or injected impurities for as neon radiation cooling [12].

For some wall materials also chemical erosion processes are significant, as is the case for graphite, where mostly chemical erosion is as important as physical sputtering. Here are the particle flux density and the surface temperature of the material the most important parameters. The flow density dependence became only recently clear from a multi-machine comparison. The strong decay of the yield at the highest flux densities was shown clearly at the tokamak TEXTOR, where special means (limiter locks) allow measurements with high flux densities close to the conditions of ITER (Fig. 5)[13].

While the erosion in the divertor plays the main role for the life time of wall components, the erosion on the large inner wall also contributes to the global impurity flow and to the impurity contamination of the main plasma. The erosion on the main 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 again 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 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).

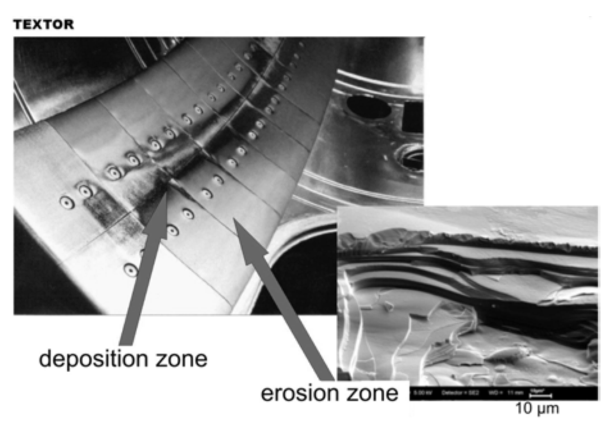


*Fig. 6 Formation of layers by deposition of eroded carbon on limiters in the tokamak TEXTOR; the electron microscope image shows layers structures of about 4 µm thickness* [14]

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 also. In fusion experiments, in particular at less loaded surfaces, carbon layers with 100% hydrogen content have been observed. This is presently considered as one of the most critical problems in plasma wall interaction for ITER. We discuss this in the following for the concepts for ITER in greater detail.

**WALL MATERIALS FOR ITER**

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

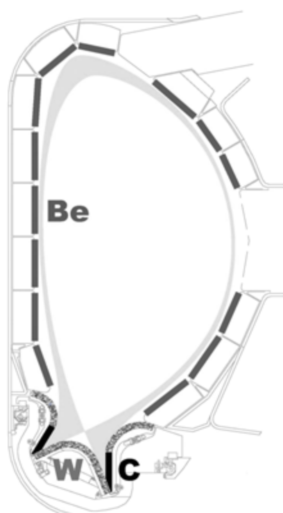


***Fig. 7*** *Combination of wall materials for plasma-loaded surfaces in ITER: Beryllium in the main chamber, graphite for the target plates in the divertor and tungsten for the remaining surfaces in the divertor*

The current material choice was found on the basis the following criteria:

Beryllium: Radiation by impurities in the plasma center, as it can occur by heavy elements, is unwanted. By the lining of the inner wall with as light an elements 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.

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 physical and chemical erosion, and associated to this the tritium retention by co-deposition. Therefore, the use of graphite should be minimized.

Tungsten: In the range of medium loads graphite is not necessary and beryllium is not yet possible. The combination of high melting point and small sputtering yield is fulfilled only by heavy materials. The choice fell on tungsten, which has sputtering yields in the range of $10^{-5}$ to $10^{-4}$ for those particle energies as expected in ITER. This small yield and the relatively small surfaces made of tungsten guarantee that no significant radiation in the plasma center arises.

This material choice is the basis for the concept for ITER for the control of heat exhaust, helium removal and erosion processes. From the point of view of plasma-wall interaction effect there are no obstacles for the primary goal of ITER - 500 MW fusion power for 8 minutes of pulse duration and significant alpha-particle heating.

In addition, beyond that still further goals are to be pursued with ITER. In particular a high availability of the experiment is worthwhile. In this context there are some critical questions related to plasma-wall interaction with relevance also for the continuous operation of a fusion reactor aimed at on a long-term basis. The critical questions are concerned particularly with the still existing uncertainties about the expected life time of wall components and the minimization of the tritium retention in deposited layers.

Tritium retention is considered as one of the largest problems in connection with graphite as wall material, because an accumulation of the mobilizable tritium inside the deposited layers could exceed the maximum allowed amount (presently 350 gram). In this case the plant would be have to shut-down for the period of cleaning or conditioning with possibly substantial disadvantages for the availability of the whole experiment.

Some wall components (e.g. the divertor plates) will have (due to erosion) a limited life time and will be occasionally exchanged. A too large replacing frequency would impair however also the availability of the plant. The main uncertainties lie here in limited knowledge about the erosion by transient loads, like ELMs or disruptions. This concerns in particular the question whether graphite can be totally replaced e.g. by tungsten, because transient loads can melt the metal contrary to graphite and possibly prematurely destroy thus the components.

**STRATEGIES FOR MINIMIZING THE RISK**

The risk of insufficient availability is to be minimized by a coordinated effort with experiments and calculations for plasma-wall interaction. This task is addressed e.g. by the European "task Force Plasma-Wall Interaction", under whose co-ordination the critical questions are investigated at the European facilities[15]. The strategy for minimizing the risk contains both the improvement of the wall concept with the selected materials and the parallel development of alternative material combinations.

The Re-erosion of deposited carbon must be still better understood, 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[16].

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 again 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 before a 5 mm thick layer is eroded at the target plate. According to this calculation we do not have a life time problem.

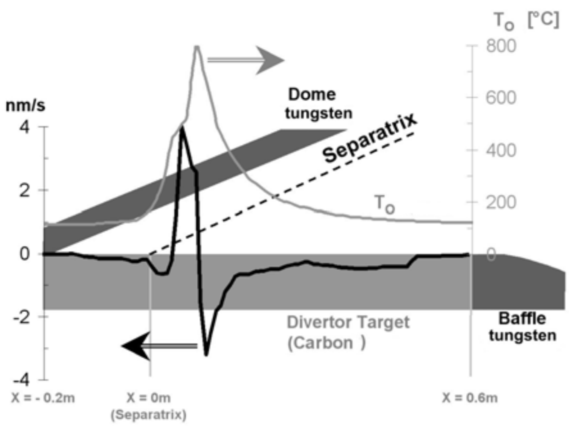


***Fig. 8*** *Calculation of erosion, deposition and surface temperatur on an ITER target plate ITER with the computer code ERO; the direction of the magnetic field line is indicated by the separatrix*

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 impact 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.

The development of methods for a fast removal of deposited layers may also help to make graphite more acceptable as a wall material. Ongoing research concentrates on different procedures, e.g. laser ablation of the surface, outgassing of the layers by local heating by means of photo-flash lamps or chemical removal by oxygen. Likewise it is necessary to develop in-situ methods for measuring the layer thicknesses.

So far still no tokamak with the material combination Be-W-C is operated. Therefore one decided to equip the tokamak JET with the same combination of wall materials as in ITER. The installation will take place in the year 2008. This global test of an ITER-like wall has great importance for further decisions about materials for the ITER divertor. It is still discussed whether the divertor ITER starts with, and whose construction will start approximately in 2009, should be tested in JET or whether it would be more meaningful to prepare already at JET for later years the planned second divertor generation for ITER. The installation of a new generation of divertor is made possible by a cassette design. This gives flexibility and offers the opportunity to make use of all improved knowledge about plasma-wall interaction at a later stage for an improved divertor.

The alternative to graphite and thus the avoidance of tritium retention would be a full tungsten lining of the inner wall and the divertor. The parallel development of this alternative is an important element of the European fusion research. The Tokamak ASDEX Upgrade took the leading task to develop for the ITER reference scenario an integral solutions with pure tungsten walls. For this the coverage of the walls with tungsten-coated bricks is performed stepwise, with the goal of reaching a complete coverage with tungsten in the year 2006/7 (Fig.. 9).

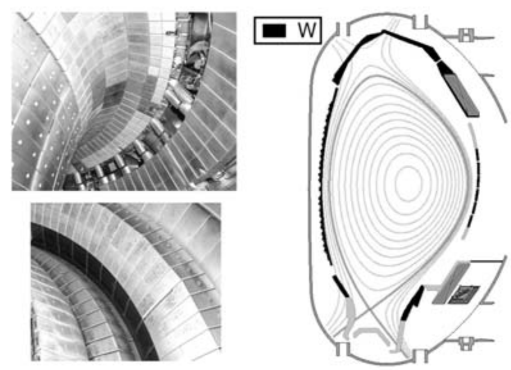


***Fig. 9*** *Lining of the inner wall of ASDEX-Upgrade with tungsten (65% coverage in 2004/5)*

So far the experiments could test successfully integrated scenarios, whereby the feared tungsten accumulation in the plasma center could be avoided [17].

Without carbon in the divertor the desired radiation cooling can be achieved only by injection of additional impurities, e.g. argon. It was shown on ASDEX-Upgrade that the admixture of argon led to a still tolerable increase of tungsten erosion with the favorable side effect that the peak loads by ELMs was reduced. However still carbon is in the system, so that only the complete wall lining can give a final answer. The employment of a complete tungsten wall puts also high demands on the permitted plasma scenarios. Only plasma conditions will be tolerable, which exclude a melting of wall components.

**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 without doubt. The still open questions of plasma-wall interaction refer to problems with the realization of a continuously operated fusion power station. In ITER these questions have relevance for the availability of the experiment.

A high availability is crucial for the economy of a fusion power station. Fusion research enters now into a new era, in which the main question is not anymore whether a burning fusion plasma can be produced – this we know in principle based on the results of JET[18]. 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 interaction concentrates in the coming years on forthcoming 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.

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 presently being built, work contrary to tokamaks not in a pulsed manner. That makes stellarators particularly relevant for investigations of plasma-wall interaction during continuous operation. In principle the problems of plasma-wall interaction are alike in stellarators and tokamaks.

Experiments with new wall concepts on a large scale, as now intended for JET, are very important. 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 however 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.

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