Magnetic Confinement Fusion: Ions in ITER – Beams and Surfaces

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Abstract

Ions play a dominant role in the experimental fusion reactor ITER. In this article some aspects of ion interactions will be discussed briefly: fast ion beams to produce fast neutral beams used for current drive and heating, and plasma surface interaction. The fusion plasma with a typical temperature of several tens of keV has to be brought into contact with a physical wall in order to remove the helium produced and drain the excess energy in the fusion plasma. Without cooling, the plasma would degrade the wall and the debris from the wall would extinguish the plasma. Therefore, schemes are developed to cool the plasma edge. The resulting plasma-surface interaction concerned in ITER is facing several challenges including surface erosion, material redeposition and tritium retention.

Contents

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1. Introduction

Ion beams provide unique opportunities to probe and modify matter as witnessed at the ION06 conference and the present volume. Ions created in a dedicated source have properties that can be tailored very precisely in an ion beam. Consequently, such beams have important applications in determining surface and interface structure, and in addition in modification of materials near a surface. Another way to produce quasi unbound ions is in plasma. In the plasma the total charge state is zero, it is quasi neutral, but the properties of the plasma can widely vary because the ion and electron translational temperatures in the plasma can vary from close to zero to phenomenal, multi keV temperatures in laboratory plasma and stars (McCracken and Stott, 2005). In fact, the plasma state is the predominant state of matter in the universe.

Fusion of light elements is the energy source of the stars and when carried out in a controlled fashion would make an almost unlimited amount of energy available on earth. There is two ways to achieve this in principle: inertial confinement fusion where the density and temperature of a small volume are raised to extreme levels during very short pulses, as discussed elsewhere at this conference, and magnetic confinement fusion, in which the nuclei to be fused are magnetically confined in a hot (tens of keV) plasma for so long, that a stable plasma heated internally by nuclear fusion reactions can deliver excess energy to the outside world. Very recently, a major step towards the realisation of fusion energy has been taken. On June 28th 2005, the ITER partners China, the European Union, Japan, the Russian Federation, South Korea and the USA agreed to construct ITER in Cadarache, France. ITER is the large international fusion reactor and a major step on the way (ITER is Latin for "the way") to commercial exploitation of nuclear fusion for the production of electricity. Later India joined the project, and it is expected that before the end of 2006 all treaties bringing ITER into existence will have been signed and ratified by all parties concerned.

ITER is a fusion reactor of the "tokamak"-type, in which a hydrogen plasma is confined in a torus by means of strong magnetic fields (Braams and Stott, 2002; McCracken and Stott, 2005). ITER must demonstrate tenfold power multiplication in a controlled fusion process, at a power level in excess of 500 MW and during pulses of 10 minutes or longer. Experiments with ITER should lead to the solution of the remaining physics problems on the way to fusion (ITER Physics Basis Editors et al., 1999). It will be used to address a number of technological issues that will be important in the construction of commercial reactors (ITER; Ongena and vanOost, 2002; Samm, 2003; Lister and Weisen, 2005).

In ITER's very large vacuum vessel, filled with plasma with a degree of ionization of unity, a very high temperature can be realized. Therefore, ITER is a place where many complex interactions involving ions take place. In this article I will only mention two, the use of ion beams to heat the plasma, and the interaction of the plasma and the physical wall of the device.

2. ITER and Fusion Energy

Nuclear fusion reactions proceed only at temperatures which are roughly six orders of magnitude higher than those required for regular chemical reactions, because of the Coulomb repulsion of the nuclei concerned. The reaction with the lower activation energy is the one between deuterium and tritium. This is the reaction of choice in ITER:

$$^{2}\mathrm{D}^{+} + {}^{3}\mathrm{T}^{+} \rightarrow n + {}^{4}\mathrm{He}^{++} + 17.6\ \mathrm{MeV}.$$

For comparison we list the energetics of carbon monoxide combustion:

$$2\mathrm{CO} + \mathrm{O}_2 \rightarrow 2\mathrm{CO}_2 + 10\ \mathrm{eV}.$$

The six orders of difference in magnitude of the barrier is reflected in the exothermicity of the two reactions. Per unit of mass of the "fuel" the energy release of the fusion reaction is even seven orders of magnitude larger than that of the chemical reaction.

Most (80%) of the energy of the DT-fusion is carried away by the neutron. It is captured in a blanket containing Li, in a reaction in which also tritium is regenerated:

$$^{6}\mathrm{Li} + n \rightarrow {}^{3}\mathrm{T} +^{4}\mathrm{He} + 4.78\ \mathrm{MeV}.$$

This yields the overall reaction equation for a fusion reactor:

$$^{2}\mathrm{D} + {}^{6}\mathrm{Li} \rightarrow 2\,{}^{4}\mathrm{He} + 22.4\ \mathrm{MeV}.$$

This reaction shows that ^{2}D and ^{6}Li constitute the fuel for fusion. ^{2}D and ^{6}Li are abundantly available. The exhaust of a 1 GigaWatt fusion plant is only 250 kg of benign He per year. The latter is to be compared to 7.2×10^{9} kg of CO_2, which is released by a 1 GigaWatt coal fired power plant.

The DT fusion reaction is not a chain reaction, a fusion reactor cannot have an energetic runaway. A fusion reactor is thus inherently safe and will not cause a nuclear explosion. Some components inside the fusion reactor become activated during the operational lifetime, but the total radio toxicity decays rapidly, dropping by four orders of magnitude within the first 100 years, to a level that allows recycling of the material. In addition, operation of a fusion plant does not require transport of radioactive fuel or waste. The fuel is abundant, practically unlimited, very cheap, and available to everyone, which could greatly reduce political tension. Fusion is one of the few options for large scale power generation. In summary, a fusion plant would be a very desirable addition to the world's capabilities to generate energy in a sustainable fashion.

The rate coefficient $\langle \sigma v \rangle$ for the DT- reaction peaks at a value of 10^{-21} m^{-3}s^{-1}. The peak occurs at a Maxwellian temperature of 70 keV. For the fusion power output also the ion density plays an important role. For a given product of density and temperature (pressure) we find that the maximum output of fusion power is given around an operation temperature of a fusion reactor of 10–30 keV. At those temperatures all light atoms are fully stripped of their electrons. Obviously, the contact of the hot plasma with a material wall has to be avoided, because the wall will be evaporated, the evaporated matter will be ejected into the hot plasma, and the plasma will be extinguished by the resulting fast cooling. In ITER, and other so-called Tokamak reactors this is done by confining the plasma in a doughnut-shaped magnetic field. In the picture of ITER in Figure 1, the doughnut-shaped plasma chamber, surrounded by magnets, can clearly be seen. The magnetic field is so strong that the ions and electrons can only move along the field lines, reducing the plasma transport perpendicular to the magnetic field lines by 14 orders of magnitude. This brings the thermal conduction of the hot plasma to the wall down so much, that a temperature difference of 100 Million K over a distance of about a meter can be sustained. The magnetic field is produced by superconducting coils, which implies that the vacuum vessel of ITER containing the hottest (macroscopic) volume on earth is placed inside the worlds largest liquid-He cryostat.

Perhaps the most crucial element of magnetic confinement fusion is the stability of its confinement. Most efforts of the fusion community in the past have been

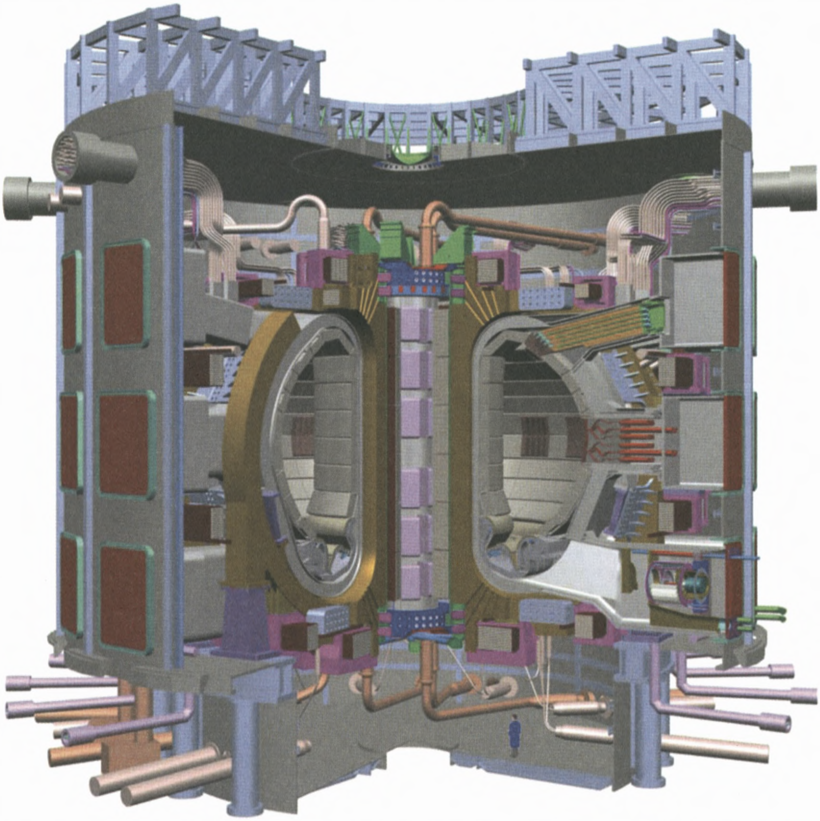

Figure 1. Artists view of ITER. The toroidal plasma chamber is clearly visible. Some parameters are: major radius 6.2 m; minor radius, 2 m; vertical elongation, 1.86; plasma current, 15 MA; magnetic field, 5.3 T; plasma volume, 850 m3; fusion power, 500 MW; power multiplication Q, 10. From (ITER).

devoted to proper confinement of the plasma and major breakthroughs have been realised. This is nicely illustrated in Figure 2, where the experimental confinement time of plasma in many different tokamaks is plotted against the confinement time as derived from various models. From the figure it is clear that ITER is a direct extrapolation from existing machines and scaling such as the one shown here demonstrate that ITER will be built on very solid grounds. Nevertheless, ITER is a scientific experiment and the last step between fusion science and fusion reactor engineering. It is obvious, that this article is not the place to discuss the scientific issues for ITER in any detail. The reader is referred to other sources, notably

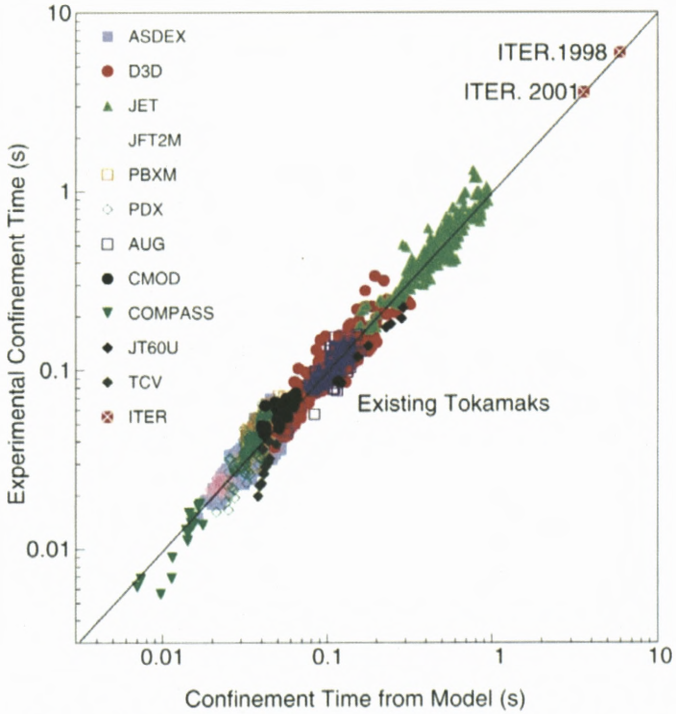

Figure 2. Experimental data from all the major tokamaks in the international fusion program, showing how the measured confinement time fits model calculations of the confinement time. Extrapolating this scaling, the confinement time is predicted to be almost an order of magnitude higher for the present and previous ITER design. The confinement time is a measure how much external energy needs to be provided to run the tokamak in steady state. From McCracken and Stott (2005).

also on the web (ITER; Samm, 2003). In this report we limit ourselves to briefly introducing neutral beam heating and plasma-surface interaction.

3. Neutral Beam Heating

The very high temperatures of many keV in a fusion reactor can only be created by extensive heating systems. In Tokamaks several heating schemes are used (Braams and Stott, 2002; McCracken and Stott, 2005). Each of these schemes should be able to deposit tens of MW's into the plasma. Using a coil around the inner structure of the tokamak, inside the doughnut, as the primary winding of a transformer, currents can be induced in the plasma, which lead to Ohmic heating. Besides, electromagnetic radiation of several frequencies can be employed to resonantly

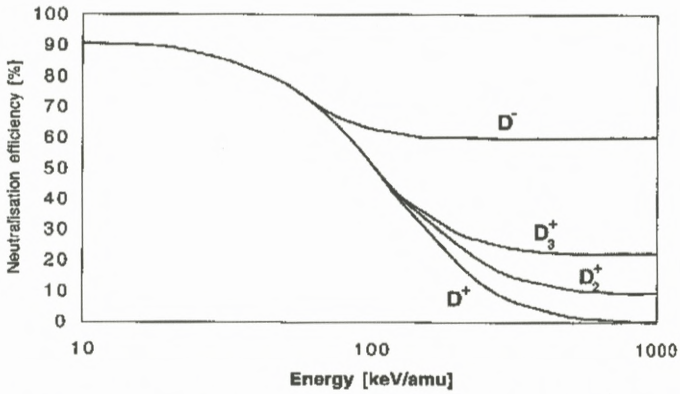

Figure 3. Neutralization rate for hydrogenic ions in a gas neutraliser as a function of the ion energy. The figure demonstrates that for high neutral beam energies, negative ion beams have to be used. From Koch (2006).

heat modes of motion of either the electrons or the ions. Finally, the plasma can be heated and directional current can be driven by injecting energetic hydrogenic ions into the plasma (Pamela, 1995). Due to the very strong magnetic field of tokamaks these particle beams can not be charged when entering the tokamak through the magnetic field. The ions would be deflected away from the plasma. Therefore, ion beams need to neutralized before passing through the magnetic field into the plasma. In the plasma the neutral beam will be gradually ionized. The easiest way to neutralise ion beams is by using a gas neutraliser. The efficiency of neutralization in this case is plotted in Figure 3 (taken from Koch, 2006). It is clearly seen that for positive ions the neutralization efficiency drops markedly above 100 keV. The reason for this is that at the corresponding velocities the nuclear velocity is much higher than the classical orbit velocity of an electron in a hydrogen atom, and those electrons would have to "jump" on to the moving deuteron. Conversely, a negative ion can easily detach its most weakly bound electron at any velocity. Since each neutral beam heating system has to deposit several MW into the core of plasma, neutral beam energies of 1 MV or more are foreseen to obtain the proper penetration, and those cannot be realized starting by using positive ion sources. Negative ion beam based neutral beam heating is applied on several machines and a representative example is shown in Figure 4, where the neutral beam heating system of the large helical device (LHD) in Japan is shown (Kaneko et al., 2003). The LHD is not a tokamak but a stellerator, another type of magnetic confinement. From the overview drawing it can be seen that the neutral beam injection system consists of a negative ion source, followed by a

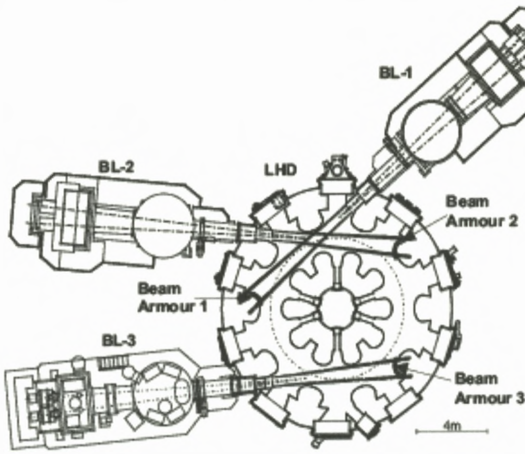

Figure 4. Arrangement of beam lines and the location of beam armour plates in the vacuum vessel of the Large Helical Device in Japan. Three tangential beam lines are installed, and each beamline has two negative ion sources side by side, followed by a single neutraliser. Note the scale in the lower right. From Kaneko et al. (2003).

gas neutraliser. In 2002 the total power of the three beams was about 10 MW, at acceleration voltages of 160–180 keV, and pulse lengths of 2 s. The interaction between the beam and the plasma is a complex issue not to be addressed here. In fact, the stopping power of the fusion plasma is a property analogous to the well know stopping power for ions in solids. If under circumstances not all the power available in the beams is transferred to the plasma, the walls might be damaged. Therefore, special armour is installed to protect the vacuum vessel to the very high power load imposed by the beams. For ITER the building of megavolt accelerators of multi-ampere D^- beams will be a major challenge.

4. Plasma-Surface Interaction in Fusion Devices

Another major area where ion interactions play a crucial role is where energetic ions hit physical surfaces. This is part of the well studied field of plasma-surface interaction (PSI). This is an area of very exciting research, where PSI in ITER will be radically different from PSI in its predecessors, and in other areas such as plasma etching (Winters and Coburn, 1992). A typical plasma pulse in a contemporary, non-superconducting tokamak lasts at most tens of seconds. An ITER pulse will last at least 500 seconds and continuous operation is foreseen. While the electron, ion and power fluxes to the wall in ITER will be only a factor of 2–3 higher, the accumulated particle and energy loads of the surfaces concerned

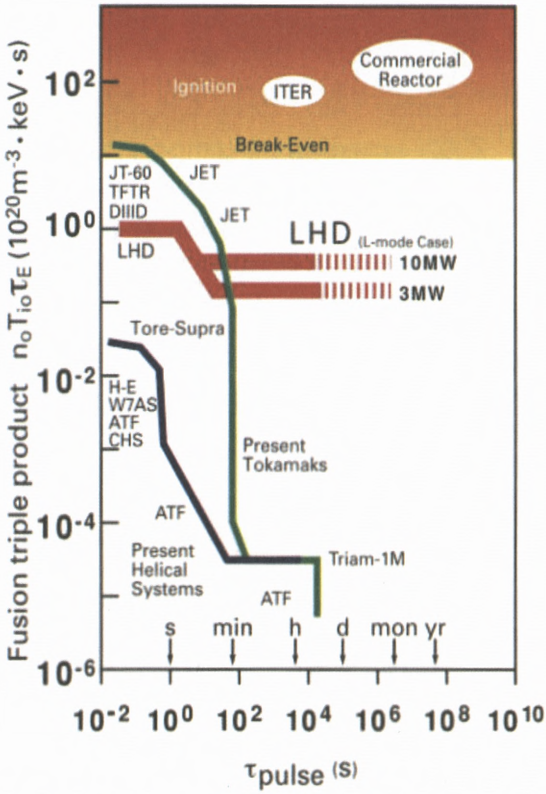

Figure 5. Parameter space for fusion devices plotted against fusion triple product, a measure for confinement, and pulse duration. It is clear that most devices operate at short pulses and that up to now the path towards a working reactor was chosen to follow an increase in triple product first, only later to be followed by long pulse experiments. From the website of the large helical device: http://www.lhd.nifs.ac.jp/en/home/lhd.html.

in ITER will be up to 4 orders of magnitude higher than for the earlier machines. In terms of wall-load, a single ITER pulse is comparable to at least a year of operation of JET, presently the largest fusion device in the world. It is a very large challenge to construct walls that can sustain the loads to be expected in ITER. It is a very interdisciplinary problem (Kleyn et al., 2006b).

The development of fusion devices and PSI therein can be summarized very nicely with Figure 5. Here is plotted the performance of various machines as a function of the fusion triple product, the key figure of merit for reaching ignition in a magnetic confinement device, and the pulse duration (Braams and Stott, 2002; Motojima et al., 2002, 2004; McCracken and Stott, 2005). It is seen that most effort has been devoted to reach very high plasma densities, high temperatures and sometimes even break even for a short time. It is also seen that there are devices such as Triam in Japan, Tore Supra in France, and LHD in Japan, where plasma operation for extended periods was achieved, but at lower plasma performance (see e.g. Mutoh et al., 2006). In ITER the ambitious goal is sustained burn with a power amplification of 10, which is a formidable job as far as PSI is concerned.

The first PSI question to be addressed here is: where does the plasma hit the surface and why (Federici et al., 2001, 2003; Philipps, 2002, 2005; Philipps et al., 2003 Samm, 2002, 2003, 2005)? The primary wall is supposed to be completely protected against impact of the plasma by the confining magnetic field. Diffusion across this field is strongly suppressed. But some diffusion across the field always occurs. Delicate equipment inside the toroidal plasma chamber is protected from plasma impact by so-called limiters. Impact on those devices is also unlikely under normal conditions, but they are designed to take a large power load in exceptional cases. In the so-called divertor the plasma is deliberately brought in contact with the wall. The reason for this is simple: in a burning DT-plasma He is produced. If the He remains in the core of the plasma, it will gradually dilute the burning DT mixture, and eventually extinguish the nuclear fire. In addition, the heating power released into the plasma volume by the He^{++} formed and the initial external heating has to be exhausted via the wall.

4.1. Divertor Physics

In Figure 6 a cross section perpendicular to the toroidal field shows the divertor. The magnetic field surrounding the plasma core is designed to block any transport across it to the wall. The field lines form nested flux surfaces as shown on which the particles run around the torus according to their thermal speed. The outmost of such magnetic flux surfaces is called "last closed flux surface" or LCFS. Below the LCFS the so-called X-point is shown. The magnetic field lines outside the LCFS are designed to intersect the wall in the divertor region. Plasma that by diffusion has moved outside the LCFS in the divertor region will eventually hit the divertor surface. The angle between magnetic field lines and divertor surface is very small, a few degrees only, to reduce the specific heat load.

At the divertor plate the ions from the plasma are neutralised on the surface. The neutralization step will result in electron or photon emission and surface heating. In addition, neutral particles will be formed on the surface, that leave the divertor plates as atoms, molecules, clusters or even dust particles and are reionized when entering the plasma. Once ionised the magnetic field guides the ions and the plasma flow forces them to return to the surface, where again neutralisation occurs. Resonant charge transfer reactions with neutral gas produces

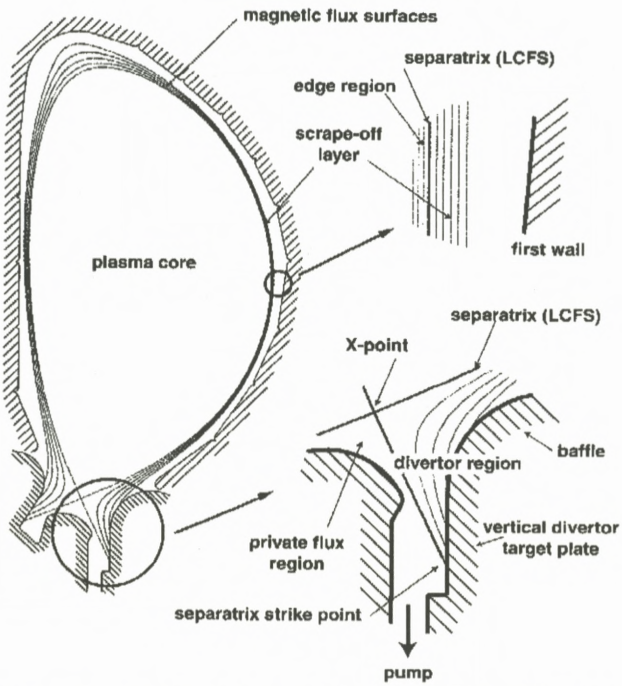

Figure 6. Poloidal cross section of a tokamak, defining the various regions of the plasma and the boundary walls. Important regions are: the plasma core, the edge region just inside the separatrix, the scrape off layer outside the separatrix and the divertor region, which is an extension of the scrape off layer plasma along field lines into the divertor chamber. The divertor structure is designed to prevent neutrals from leaving the divertor. In the magnetic flux region below the X-point, the magnetic field lines are spiralling downward to intersect the wall at the vertical divertor target plates, and are isolated from the rest of the plasma. In the divertor region intense plasma surface interaction will take place. From Federici et al. (2001).

hot, electronically excited neutral atoms, which exhaust a significant fraction of plasma energy by radiation onto large sections of the walls when leaving the plasma. In addition, neutrals and impurity ions are excited by electron impact which leads to more electromagnetic radiation. Electrons and ions originating from the core plasma are thus effectively cooled by radiation and charge exchange processes from the keV range to preferably below 1 eV (Samm et al., 1993; Samm, 2002). The plasma ions can undergo many neutralization-reionization cycles before they leave the plasma regions as neutrals towards the vacuum pumps under the divertor. At plasma temperatures around a few eV inside the divertor the plasma is in a so-called detached state (Stageby, 2000). The surface is thus protected by a dense plasma with a low degree of ionization. This dissipative layer

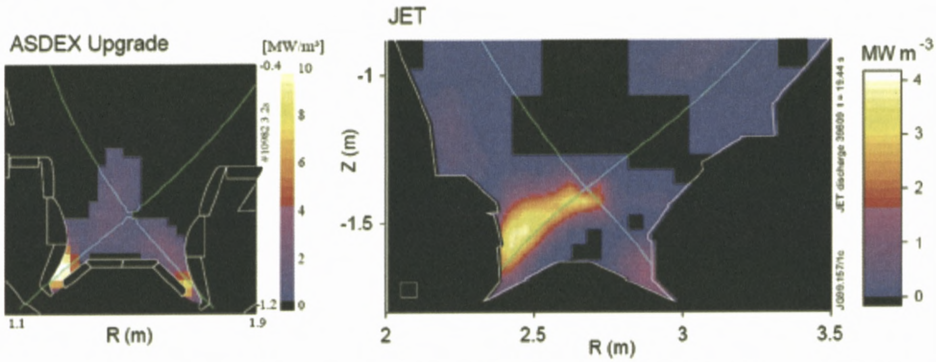

Figure 7. A vertical cross section through the ASDEX and JET tokamaks, with a measurement of the emission from the plasma. It is clear that below the X-point the plasma is efficiently cooled by radiation, and that the maximum of the radiation is detached from the divertor surfaces. From Kallenbach et al. (1999).

that radiates large amounts of the power carried by the incident plasma has been studied extensively, and a nice view on it is reproduced in Figure 7 (Kallenbach et al., 1999). It is seen that most of the power radiated does not come from the surface region but slightly above it. The plasma is detached from the wall and the wall is protected by it. Nevertheless, the power load on the surface of the divertor in steady state is still formidable in ITER, up to 10 MWm^{-2}.

4.2. Wall Materials and Lifetime

The materials of the various walls are of critical importance, because they might be emitted in some form into the divertor plasma, contributing to the plasma chemistry and to the radiation level in the plasma. The role of the divertor surface and its material composition require much more study. Requirements for divertor materials are:

- Good thermal and electrical conductivity,
- Low probability of ending up in the core plasma, and
- If ending up in the core plasma: low Z.

The low Z requirement follows from the fact, that ions which are not fully stripped act as a heat sink in the plasma, due to continuous excitation/de-excitation cycles of inner shell electrons by plasma electrons. If the plasma can be tailored such, that divertor material will never end up in the core, the low Z requirement can

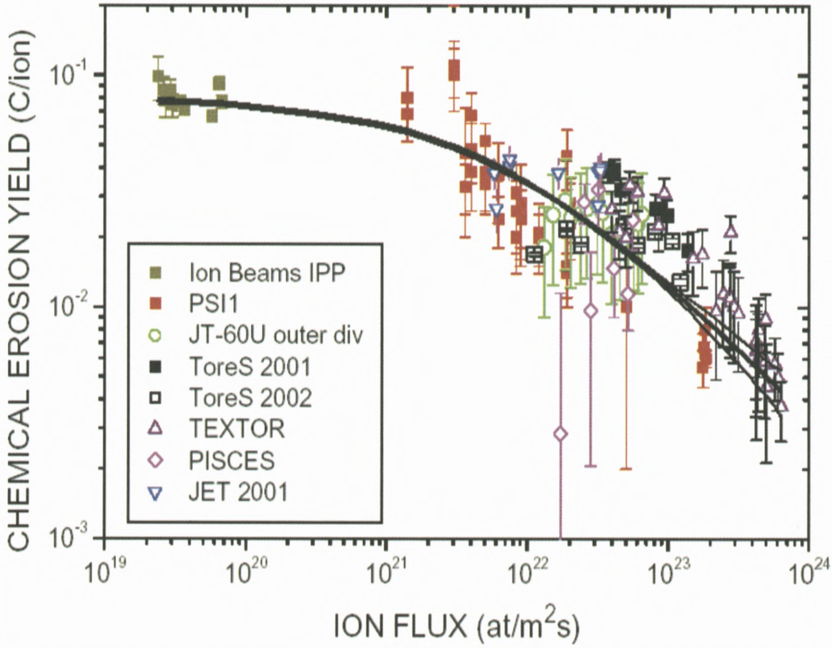

Figure 8. Erosion rates for Carbon surfaces by low temperature plasma. From Roth et al. (2005).

be relaxed. Because this is not yet completely certain, the provisional choice of materials for ITER is:

- W for limiters,
- Be for the primary wall, and
- Carbon for the divertor target plates.

The argument in general for this choice is beyond the present paper, and the reader is referred to other papers (Federici et al., 2001, 2003; Samm, 2003). Some arguments will be touched on later in this article.

The erosion rate for carbon surfaces as a function of the incident plasma flux is shown in Figure 8 (from Roth et al., 2005). What is seen is that the erosion rate drops remarkably with the incident flux. This might be at first sight very surprising, because several erosion mechanisms such as physical sputtering or chemical etching can be expected to be linear in the flux of the etching particle. This is because the PSI enters into the so-called strongly coupled regime (Kleyn et al., 2006a). This means that the mean free path of the plasma particles and the etch products is much smaller than the system size. The particles can simply be atoms, but also hydrocarbon molecules and even carbon containing dust. Dust particles in the critical size range of 1–10 nanometres are thought to play an essential role (Hollenstein et al., 1996; Hollenstein, 2000; Winter, 2000). The particles produced at the surface are interacting in the plasma, getting reionized and are redeposited at the surface, because these ions are confined by the strong magnetic field available. The eroded particles are thus effectively recycled near the surface. The higher the plasma flux, the higher the recycling probability and thus the lower the effective erosion yield. In fact, in certain regions deposition will be the result of the recycling in the strongly coupled region and wall material can effectively be transported inside the machine. This has important consequences when tritium is present in the machine.

4.3. Tritium Retention

Because tritium is radioactive, most of the tritium should be in the plasma volume or in the gas processing plant, but not retained or trapped elsewhere. Hydrogen and its heavier isotopes deuterium and tritium can be adsorbed with long residence time in deep pores in the walls. In addition, they readily form compounds with carbon. In these compounds the deuterium and tritium present in a fusion reactor are not available for the fusion reactions in the plasma, but are retained somewhere in the vacuum vessel or walls. For hydrogen and deuterium this is only an operational problem, because there is hardly an upper limit to the amount of hydrogen in a tokamak. For tritium the retention in the walls is a serious problem, because the amount of tritium allowed in the reactor is small.

From present tokamak experience it is definitely concluded that the overwhelming majority of the long term tritium retention is due to co-deposition of tritium along with eroded carbon forming tritium saturated carbon co-deposits. Like H and D, T is very reactive with carbon and can form a variety of molecules. To evaluate the scientific basis of this process and to improve our predictions for future devices comprehensive action is needed. The reactivity of D and T will be considered to be similar, so that the majority of studies can be carried out with D.

4.4. Research on ITER Relevant Plasma Surface Interaction

Of course ITER itself will serve as an important test bed for the divertor design. However, supporting experiments in smaller devices are indispensable in approaching a deeper understanding of the processes. The issue of PSI and wall materials in reactor conditions is far from settled. Tokamaks are needed to study the complex interplay of main chamber plasma and divertor plasma in toroidal geometry. The most important large scale experiment in this context is planned to be conducted on JET by modifying the JET wall with an ITER-like mix of materials (www.JET.org). On a smaller scale, but addressing the processes in more detail with specialised diagnostics, experiments on plasma-wall interaction are performed on other tokamaks, like e.g. on TEXTOR in Jülich. TEXTOR is operated by the Trilateral Euregio Cluster collaboration (TEC: FOM Institute for Plasma Physics Rijnhuizen The Netherlands, Institute for Plasma Physics Forschungszentrum Jülich Germany, Royal Military School Brussels Belgium). However, most present tokamaks are short-pulsed compared to ITER. Their rather small duty cycle and lack of easy access for PSI diagnosis give rise to uncertainties with respect to long term erosion and deposition processes. From this problem the need for steady-state experiments with the relevant PSI parameters will be obvious. Such experiments should allow addressing the issues discussed above in an open and well-accessible, steady state flexible laboratory environment. The accessibility allows the use of *in-situ* real time plasma and surface diagnostics, so that processes can be studied while they happen, with the plasma on. Samples should be transferable, if necessary under vacuum, to surface analysis facilities. Modifications, changes to materials etc should be introduced relatively quickly. Thus, smaller scale laboratory experiments, with steady-state capability and heavily equipped with diagnostic tools, will complement the studies of plasma-wall interaction in tokamaks.

5. Experiments to Study ITER Relevant Plasma-Surface Interaction

The ideal experiment is sketched in Figure 9. Essentially, a small slice out of the circular ITER divertor is taken. Please note that the ITER magnetic field is almost perpendicular to the plane of the paper. In fact, in ITER the magnetic field intersects the divertor plates at an angle of a few degrees. Above it, a plasma generator needs to be build delivering a plasma of ITER-like characteristics with a temperature and density that is high enough that the plasma will diffuse perpendicular to the field lines and intersect the surfaces to be studied with the ITER-like powers and fluxes. Basically, this is not possible, because such a plasma generator does not exist, other than a real Tokamak that the smaller scale experiment tries to mimic but not to copy. The only way to realise such an experiment is to use the geometry given in Figure 10.

A plasma source delivers the plasma the required energy and particle density. This plasma is magnetically confined and impinges on a target at grazing incidence. The device has opportunities for heating the plasma by radio-frequency radiation, and by biasing the target. It should be noted that the similarity to the ITER case and Figure 10 is limited. The aim in the linear experiment of Figure 10

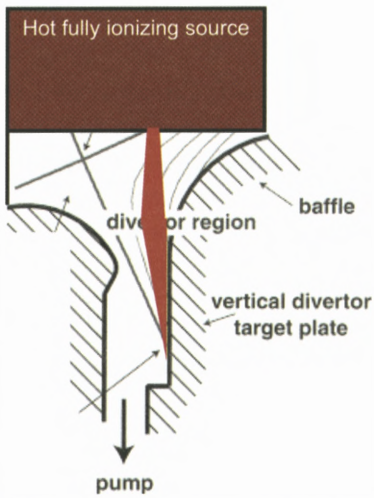

Figure 9. Schematic diagram showing how to construct an experiment to mimic interactions at the ITER divertor. The divertor region is taken from Figure 6.

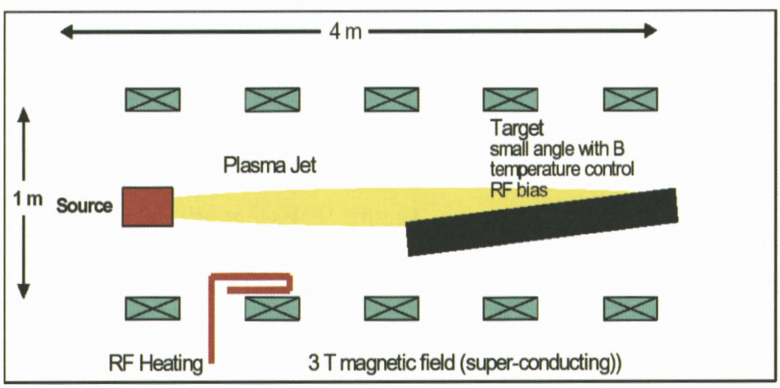

Figure 10. Schematic drawing of a linear experiment to study plasma surface interaction.

is to have conditions similar to those in ITER in the last 1–2 centimetres before the target plate.

As part of the TEC collaboration and within the framework of Euratom the FOM Institute is building a new machine, Magnum-psi, providing an important new experimental facility in the range of experiments that are available to PSI research for ITER and reactors beyond ITER (Groot et al., 2005; van Eck et al., 2005). The uniqueness of Magnum-PSI lies first in its ability to access simultaneously the multitude of aspects of PSI in the combination of which ITER differs essentially from present day experiments:

1. Large ion fluence and continuous operation, which leads to "macroscopic" modification of plasma-facing surfaces.
2. High power density (5–10 MWm^{-2}) with low plasma temperature (<5 eV) such that materials are close to, or at the energy threshold for sputtering, but have high surface temperature and are therefore near their materials limits for stress/strain, etc.
3. Strong plasma-surface coupling: the high plasma density leads to short mean free paths for dissociation/ionisation of eroded atoms or molecules in comparison with the linear dimensions of the plasma.
4. Access to plasma diagnostics and *in-situ* surface analysis.

With a steady-state high flux of up to 10^{24} ions $m^{-2}s^{-1}$ at a plasma temperature in the eV range, a magnetic field of 3 T, and large beam diameter, Magnum-psi will be a unique experiment, bringing the relevant parameters typically an order of magnitude beyond what is presently available in linear plasma devices, and into the realm of the ITER divertor.

A device providing the parameters as described above is currently not available to the magnetic fusion community although a number of smaller devices do exist, following the basic design of Figure 10. These include the PISCES experiment at the University of California in San Diego (Hollmann et al., 2002), the PSI-2 experiment at the Humboldt University in Berlin (Grote et al., 1997), the NAGDIS-II at the Nagoya University (Hollmann et al., 2001), and the Pilot-PSI experiment at FOM Rijnhuizen (de Groot et al., 2003, 2005). The latter device, Pilot-PSI, is shown in Figure 11. The plasma generator, a high pressure cascaded arc source, can be seen clearly at the left (VanDeSanden et al., 1992). The cylindrical vacuum vessel is surrounded by magnetic field coils (blue and yellow). The target is at the right of the vessel, and not visible in the figure. The vacuum (roots) pump is at the far right. This experiment has a number of characteristics of the Magnum-PSI experiment. The ITER-like flux needed in those experiments has been demonstrated recently at our laboratory and dramatically high etching rates have been shown. These rates are way off the curve of Figure 8. It demonstrates that in the small linear device Pilot-PSI the erosion products are not recycled, and real ITER relevant experiments can only be carried out if recycling takes place in the experiment. In Magnum-PSI the dimensions and density of the plasma will be sufficiently large to ensure recycling. Another difference between Pilot-PSI and the ITER divertor region is, that the pressure in the Pilot vessel can considerably exceed

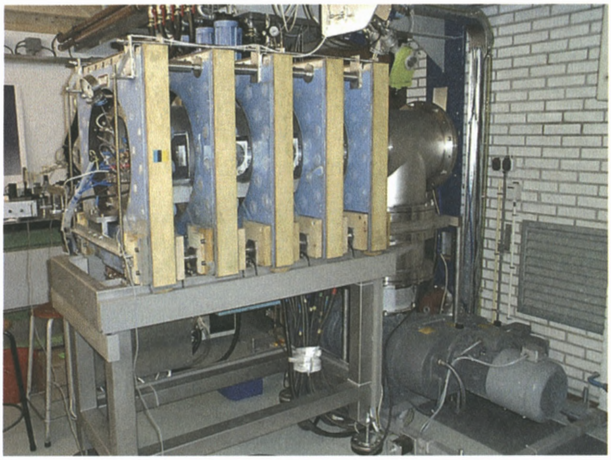

Figure 11. Picture of the Pilot-PSI experiment.

the one near the ITER divertor, a few Pa, because the degree of ionization of the plasma as it exits the source is only 10% or so, and in contrast to ITER, where it is 100%. Therefore, a linear simulator based upon high pressure cascaded arc plasma generators needs differential pumping to make sure, that the neutral gas around the target plates is only created by neutralization of the plasma, not by a steady flux from the source. In the design of Magnum-PSI this differential pumping has been incorporated, as can be seen in a schematic drawing in Figure 12. In this drawing one sees the vacuum vessel, moved into a large bore superconducting magnet. To the right of the magnet the target chamber with tubing to its pumps and an analysis chamber for the retractable target can be seen. In the main vacuum vessel one sees from left to right the source, a skimmer leading to the heating chamber followed by the target irradiation chamber. Magnum-PSI is described in more detail in other papers (vacuum) (de Groot et al., 2005; van Eck et al., 2005).

6. Conclusions

Plasma-surface interaction will be one of the areas determining the availability of ITER and the ultimate viability of generating fusion power under steady state conditions. Erosion and redeposition, handling the steady state power, and preventing tritium retention by the walls, are issues to be solved for and by ITER. Although a lot of knowledge is available an extension of our knowledge base at all levels is necessary for the ultimate success of ITER.

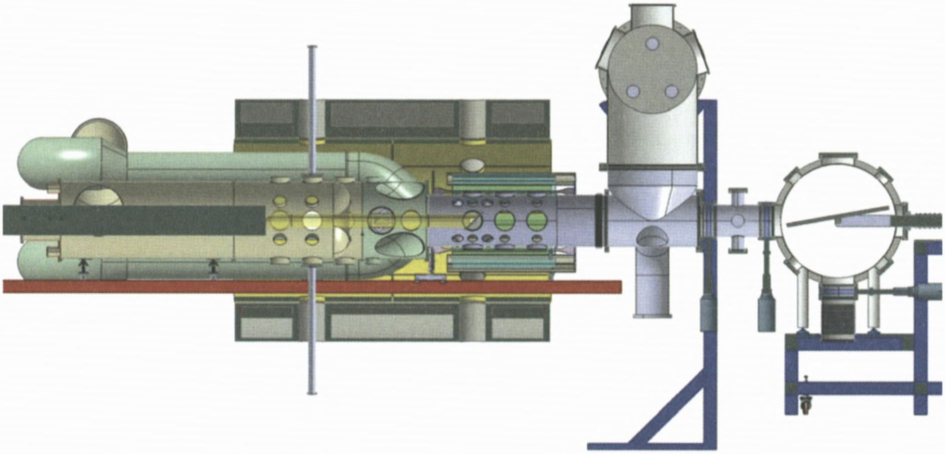

Figure 12. Total overview of the Magnum-PSI experiment with target station and target manipulator. Shown are (from left to right) the source-, heating- and target chamber with pump ducts. Next to these, the pumping station for the third stage is shown. On the right hand side, the target station with target and target manipulator are visible. In the target analysis station, the targets can be analyzed in detail with surface analysis equipment.

Acknowledgements

The author thanks all members of the Pilot-PSI and Magnum-PSI teams for their contributions to this article. He thanks Dr. Tony Donne and Dr. K. Tsumori for their comments on the manuscript. This work, supported by the European Communities under the contract of Association between EURATOM and FOM, was carried out within the framework of the European Fusion Programme with financial support from NWO. The views and opinions expressed herein do not necessarily reflect those of the European Commission

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