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Summary

Thermonuclear fusion has the potential of large-scale sustainable energy production. A large amount of energy is released during nuclear fusion of the hydrogen isotopes deuterium and tritium into helium and neutrons. In a plasma of deuterium and tritium that is heated to temperatures of 150 million degrees Celsius the Coulomb repulsion can be overcome, so that fusion can occur. The most advanced method for achieving these conditions on earth is via magnetic confinement of hot plasma in a toroidal geometry: ‘tokamak’. ITER is a scientific research machine in such a tokamak configuration and is being built at this moment to demonstrate that fusion is a feasible energy source.

An important component of ITER is the divertor, the ‘exhaust’. Its function is to extract impurities and helium from the plasma. The divertor has to be capable of withstanding heat and particle fluxes of respectively 10 MW m$^{-2}$ and $10^{24}$ m$^{-2}$s$^{-1}$. This high-flux ITER divertor regime cannot be accessed by present-day tokamaks. The linear plasma generator Piot-Psi, is able to create such extreme plasmas. Furthermore, a linear plasma generator has the advantage over tokamaks that it is very well accessible, relatively quick and that its parameters can be tuned independently. This makes them very suitable for investigation of the interaction between plasma and material.

Recently, it has been decided that ITER will operate with a full tungsten divertor from the start. Tungsten is a beneficial material to use in the divertor, because of its good thermal properties and high sputtering threshold for hydrogen. Tritium retention in the reactor wall is one of the key concerns of ITER. Tritium is radioactive and has a half-life of 12.3 year. For safety considerations, the tritium inventory limit that is allowed in ITER is set at 700 g. The tungsten divertor will be subjected to high energy neutrons produced by the fusion reaction. These neutrons create defects in the metal lattice that act as traps for tritium and enhance tritium retention significantly.

The underlying motivation of our work is to predict tritium retention in neutron damaged tungsten under high-flux plasma exposure. Since tritium is radioactive, deuterium, which has similar chemical properties, was employed to investigate hydrogen retention. High-energy tungsten ions were used instead of neutrons for damaging tungsten since the damage is similar and since they do not activate the material. The main topic of this PhD project was to investigate deuterium retention in pre-irradiation damaged tungsten under high-flux plasma bombardment. The three important aspects of the research are: pre-irradiation damage, high-flux plasma exposure and deuterium retention correlated to surface modifications.
Deuterium retention in radiation damaged tungsten was studied by exposing polycrystalline tungsten samples with different levels of pre-irradiation damage to high-flux deuterium plasmas. The experiments were performed at a relatively low temperature of \(~500\,\text{K}\). The retained deuterium was found to saturate at a concentration of 1.4 at.\%. This saturation originates in the high energy ion damaging mechanism and was not affected by the high plasma flux. At saturation, the concentration of vacancies and interstitials is at such a high level that every newly created vacancy or interstitial automatically recombines with one already present in the material. This was shown by using a simple geometric model that assumes that the saturation solely originates in the tungsten pre-irradiation and that explains it in terms of overlapping saturated volumes. The average saturated volume per incident MeV ion amounts to \(3 \times 10^4\,\text{nm}^3\).

The tungsten divertor tiles of ITER are predicted to operate at surface temperatures of around 600 – 1300 K. The above experiment was therefore also carried out in the temperature regime of 800 – 1200 K, which is above the temperature where vacancies in tungsten become mobile (\(>550\,\text{K}\)). The deuterium retention saturated at the same damage level, but the absolute level of deuterium retention was strongly decreased. This reduction originates in the mobility of vacancies that results in their annealing and clustering. The contribution of increased deuterium mobility that causes a lower occupation level of the defects was found to be small.

We studied the effect of fluence (plasma flux multiplied by exposure time) on deuterium retention and found that only very low fractions (\(10^{-5} – 10^{-7}\)) of deuterium from the incoming plasma beam are retained in tungsten. The radiation-damaged material was used to monitor the diffusion in tungsten. To explain the low penetration, we proposed a mechanism in which the deuterium atoms form a chemisorbed layer at the surface. Incoming deuterium ions (\(~5\,\text{eV}\)) do not directly enter the material, but interact with the deuterium covering the surface. Energy is redistributed between incoming ions and atoms at the surface, so that they cannot directly penetrate. A thermally activated process is needed to introduce chemisorbed deuterium into the tungsten.

The experiments so far were carried out at self-biased conditions. ITER is anticipated to operate in detached plasma mode, which corresponds indeed to low energy ion energies. However, part of the divertor will experience higher energies as well. We therefore investigated high-flux plasma exposure while the target was biased at a certain negative potential thereby accelerating the ions. It is known that high energy deuterium ions can create surface modifications in tungsten. A comparative study was made to investigate the explicit effects of plasma flux, plasma fluence (time) and pre-irradiation damage on surface modifications and deuterium retention. We distinguished three types of surface modifications: blisters originating in inter-granular cavities, protrusions arising from intra-granular cavities and structures on the nanometer scale. Micrometer-sized blisters form at high-flux plasma exposure and grow in size and quantity with exposure time. We found that pre-damaging with MeV ions decreases the formation of these blisters. On the other hand, protrusions, which are typically hundreds of nanometer in size, were enhanced after exposure to high-flux (\(~10^{24}\,\text{m}^{-2}\,\text{s}^{-1}\)) plasmas.
In the research described above, all experiments were performed on polycrystalline tungsten. Present-day tokamaks like JET also make use of thin tungsten films. The retention properties are however largely unknown. Also, for fundamental studies of retention behaviour and diffusion in low density tungsten, layers with varying tungsten density and crystallite sizes are of interest. Therefore, we exposed micrometer thick pulsed laser deposited layers to high-flux plasmas. These thin films generally withstand the interaction maintaining overall integrity. However, thin films show considerably more retained deuterium than bulk polycrystalline tungsten, even compared to the pre-damaged tungsten.

In conclusion, the retained fraction of deuterium in polycrystalline tungsten after exposure to high-flux ($\sim 10^{24} \text{ m}^{-2}\text{s}^{-1}$) plasmas is very low. Pre-damaging with MeV ions and increase of the energy of the plasma ions both enhance the retention by approximately an order of magnitude each. The overall retention level though, remains low.