Simulation of material migration in tokamaks

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Plasma-wall interaction (PWI) processes such as erosion, resulting impurity transport and redeposition, are critical issues in fusion research due to limited wall lifetime, plasma contamination with impurities and radioactive tritium retention by means of co-deposition in re-deposited layers. In addition to experiments, modelling of these processes is important with respect to predictions for future machines such as ITER or DEMO.

This contribution presents selected examples for modelling of PWI experiments in the existing fusion experiments TEXTOR and PISCES-B. The modelling is performed applying various numerical simulation codes such as ERO \cite{1} and EDDY \cite{2} for impurity transport and TriDyn \cite{3} and Molecular Dynamics (MD) based calculations for ion-solid interaction. Comparing of the modelling results with experimental observations helps to understand the underlying physical processes and serves as benchmarking of the modelling codes. Finally, predictive modelling for ITER is presented focusing on estimations of long-term tritium retention. As the allowed amount of in-vessel-tritium is limited due to safety reasons, tritium retention will strongly influence the availability of ITER (time consuming and complicated cleaning of the machine is necessary after having reached this limit).

Local impurity transport studies have been performed in TEXTOR by injection of \textsuperscript{13}CH\textsubscript{4} through test limiters exposed to the edge plasma. In general very low \textsuperscript{13}C deposition efficiencies are measured on the test limiter surfaces \cite{4}. Applying EDDY and ERO to these experiments results in much larger deposition efficiencies - up to factor of 100 - if standard assumptions for reflection & physical sputtering (based on TriDyn) and chemical erosion are used \cite{5}. However, measured light emission patterns from the injected impurities can be well reproduced by the codes indicating that the impurity transport is well described, see figure 1. Thus, the simulations suggest that additional surface processes, such as enhanced (compared to bulk material) re-erosion of re-deposited species under simultaneous ion

Figure 1. Observed (left) and modelled (right) CH light emission after CH\textsubscript{4} injection through test limiter exposed to edge plasma of TEXTOR.
bombardment at plasma-wetted areas, are important. In contrast, CD$_4$ injection experiments performed in TEXTOR at remote areas (at least 5 cm away from last closed flux surface, LCFS) and thus under negligible plasma contact, can be modelled without the assumption of enhanced re-erosion. In this case reflection of impinging species has to be calculated by MD simulations as the impact energy is too low to be treated with binary collision approximation as used in TriDyn-based models. The carbon deposition from the injection has been measured for each discharge by means of a quartz microbalance (QMB). The results are summarized in figure 2 together with the modelled deposition efficiencies.

As the first wall of ITER will be made of beryllium (Be), experiments with this toxic material are of great importance. The linear plasma device PISCES-B is able to perform experiments with beryllium under ITER-like divertor plasma conditions. One striking result of these experiments is the mitigation of chemical erosion of carbon with very small Be concentrations in the plasma [6]. According ERO simulations can reproduce measured light emission but it is seen that the mitigation of chemical erosion cannot be explained by a simple mixing or coverage of carbon with beryllium as the Be plasma concentrations are too low [7]. Whereas carbide formation can in principal explain full mitigation of chemical erosion at low Be plasma concentrations, the observed time constant of mitigation is much larger than the modelled one unless a characteristic carbide formation time is introduced.

Predictive modelling of target lifetime and long-term tritium retention has been performed with ERO for carbon divertor plates and beryllium main wall under steady state plasma conditions [8]. It is seen that target lifetime is much less critical than tritium retention. As seen in figure 3, within parameter variations studied for enhanced re-erosion and plasma conditions the number of possible discharges before reaching the safety limit is between 200 and 450.

References