Modelling of ITER Edge Plasma Dynamics Following Type I ELMs and Consequences for Tokamak Operation

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Abstract. In the H-mode regime of future tokamak ITER the transient events such as the ELMs will cause losses of DT plasma followed by surface damages and a contamination of confined plasma. The current state of computer modelling developed at FZK for the plasma is described, including also the poloidal magnetic field variations after the transients, which can drastically broaden the power footprint at the divertor surface. The two-dimensional MHD codes FOREV and TOKES have been developed for the real tokamak configuration: FOREV calculates plasma processes in SOL, and TOKES is a new code for simulation of confined plasma. The modelling includes multi-species plasma transport, neutral inflow, radiation losses, plasma currents and the poloidal field coils, aiming as the transients as the disruptions mitigation by injections of noble gases. The physics basics employed, the modelling features and the recent results of the research are presented. Latest models incorporated for neutral influxes and magnetic field surfaces are characterized. Deuterium, tritium, helium and carbon species and, as a candidate for ITER divertor armour, the carbon fibre composite (CFC) are implemented into the codes so far. Radiation losses due to carbon impurity are simulated. The computer modelling revealed that a significant impurity contamination of the edge plasma occurs which can cause the collapse of the confinement at lesser ELM sizes than that determined by armour lifetime limitations. The tolerable ELM size was estimated as ~ 1 MJ/m^2 for ELM frequency ~ 1 Hz. To validate the codes, special experimental programme is carried out with the plasma gun MK-200UG in TRINITI in frame of EU-RF collaboration. At MK-200UG first experiments aimed at impurity generation and radiation properties of evaporated carbon were performed, at first the evaporation thresholds for CFC and tungsten surfaces quantified.

1. Introduction

The reference regime for the ITER at the fusion gain Q ~ 10 is the H-mode in which the stationary operation will be repetitively interrupted by outbreaks of the edge-localized mode (ELM) with the duration 0.3 - 0.6 ms. On the basis of scaling from present experiments, due to a high thermal energy of deuterium-tritium (DT) plasma the enhanced losses into the scrape-off layer (SOL) are anticipated with the pulse energies 0.5 – 4 MJ/m^2 [1]. At each ELM the plasma arrives along and across magnetic field lines at the divertor armour and the first wall producing surface damages. The eroded material atoms are ionized and emitted back into SOL, and then they produce a contamination of the confined plasma. The transient energy release concerns also the poloidal field (PF) that undergoes substantial changes broadening the footprint of the power flux.

This work concerns mainly computer modelling developed at FZK for the plasma processes and the magnetic field in question. The goal of adequate calculations for a tokamak implies the enhanced plasma transport across the magnetic surfaces and also including the field lines ending at the vessel walls. Thus uniform algorithms valid in the whole vessel should be implemented. The high energy fluxes and following contamination of the vessel are to be simulated for multi-species plasma and neutrals inflow in real toroidal configuration, accounting for the radiation losses and the variations of the magnetic field due to the electric currents in the plasma and the coils. Moreover, the injection of high pressure noble gases is to be implemented as the disruption mitigation option, which also involves both radiation transport and neutrals modelling.

For the simulation of hot DT plasma impact on the walls and the plasma contamination following the transients due, in particular, to evaporation at the irradiated surfaces the two-dimensional multi-fluid MHD codes FOREV and TOKES have been developed. FOREV calculates plasma processes in SOL and in front of the target, such as the layer of material vapour which decreases the impact (the vapour shield). Actual heat flux and plasma pressure at the surface during an ELM and the following propagation of eroded material into SOL as well as the heat transport in the target material are simulated simultaneously by FOREV in fixed magnetic field for a few ms [2,3]. In contrast, TOKES is a new code developed for simulation of both stationary tokamak operation with the contaminated confined plasma and the long time scale effects caused by the transients, spanning the whole tokamak discharge [4-7]. For a few ms periods of the ELMs the impurity influx is provided by FOREV as a boundary condition at the separatrix. Otherwise the confinement is calculated by TOKES itself accounting for the surface sputtering by the plasma lost from the confinement region. In TOKES the magnetic field evolves together with the plasma. The PF coils automatically controlling the separatrix, the Pfirsch-Schlüter plasma diffusion model with account of neo-classical and anomalous effects, the influx of neutrals from the walls and the heating by alphas and fuelling neutral beams are implemented.

In the following chapters the physics basis employed to model the transient loads in ITER and summarise the main studies carried out to determine the implications for the tokamak operation are described. So far the carbon fibre composite (CFC) only, as one of candidates for ITER divertor armour is implemented into the codes. Radiation transport through carbon and He contaminated DT plasma in toroidal tokamak geometry and radiation heat loads at the vessel wall following the ITER ELMs are simulated. The modelling using FOREV (see Chapter 2) revealed that significant impurity contamination of edge plasma can occur, with edge carbon densities significantly exceeding the edge DT plasma density. The rich impurity can cause large radiation losses from the core and eventually lead to the collapse of the confinement at lesser ELM sizes than that determined by armour lifetime limitations [8]. Chapter 3 describes the TOKES simulations using the FOREV input in which the tolerable ELM size was estimated as 1 MJ/m^2 for ELM frequency ~ 1 Hz [3]. Latest TOKES developments [5,6], presented in Chapter 4, include implementation of magnetic field surfaces, the vessel surface and plasma wall interactions involving neutrals simulation, as well as the first validations for TOKES reported in [7].

Earlier verifications of radiation transport of FOREV using experiments on the tokamak JET and the facility GOL-III were carried out. In order to further validate the code, special experiments are carried out with the plasma gun MK-200UG [9] in TRINITI in frame of EU-RF collaboration. The main experimental achievements that concern the vapour creation and the plasma in front of the targets are described in Chapter 5. The targets manufactured in EU for ITER divertor armour have been exposed to the hydrogen plasma streams with the energy peak density relevant to the Type I ELMs and the disruptions of ITER. The experiments were carried out on the interaction of hot magnetized plasma streams with CFC and W targets. First experiments aimed at impurity generation were performed at rather low loads in order to quantify the melting threshold for W and evaporation thresholds for CFC and W. The multiple ionization of carbon ions, their temperature, density and velocity have been measured as functions of plasma load.

2. ELM induced carbon contamination of SOL

In earlier simulations [2] a rather simplified ELM scenario had been employed in which hot DT plasma of the pedestal region looses through the separatrix with a constant loss rate during the ELM duration time $t_{\rm ELM}$ and then it appears in SOL having constant temperature T and exponential density profile $n_{\rm C}(y)$ along the local axes y normal to the separatrix. The plasma expands along the magnetic field lines hitting then the surface; the diffusion along y was not taken into account. However, after the impact power reaches the vaporisation threshold the vaporized mass rate becomes strongly dependent on load details such as time and spatial dependences of power flux, which requires improvements of the scenario. In the new scenario the varying power flux at the divertor surface is calculated using a plasma diffusion transport model in the fixed magnetic field [3].

The new phenomenological model assumes that the ELM caused energy loss $W_{\rm ELM}$ is due to a sudden increase of cross-diffusion coefficient D, differently for the pedestal and the SOL. To simulate D, the data measured at modern tokamaks and extrapolated to ITER have been used [10,11]. The measured hot DT plasma flux was reproduced approximating D by suitable functions of y and the time t. It is to note that the heat flux to the divertor measured e.g. at the tokamak JET does not cause the vaporisation. Therefore to ensure the dependencies the FOREV calculations have also been done for ITER ELM parameters switching off the vaporisation. As the fitting parameters, the pedestal diffusion coefficients $D_{\rm ped}(t,y)$ and the SOL diffusion coefficients $D_{\rm SOL}(t,y)$ are chosen, because $D_{\rm ped}$ controls the amount of lost DT plasma and $D_{\rm SOL}$ the fractions of heat loads delivered to the first wall and into the divertor. The functions reach their maxima at the outer torus midplane, because the outer midplane vicinity is the most unstable region. Dimensionless time dependence of $D_{\rm ped}$ and $D_{\rm SOL}$ and also the calculated heat fluxes at the separatrix strike positions (SSP) at the both divertor targets without the vaporization are shown in Fig. 1.

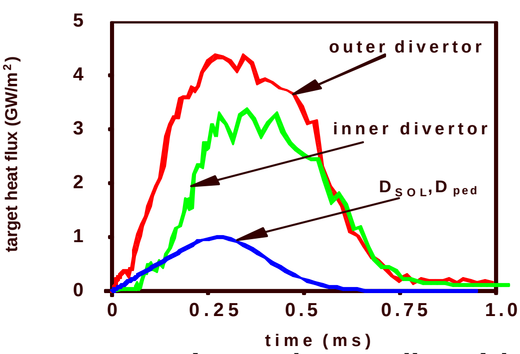

FIG. 1. D_{SOL} and D_{ped} schematically and the SSP heat fluxes modelled by FOREV without vaporization (W_{ELM} = 12 MJ, t_{ELM} = 0.5 ms)

Then the calculations had been repeated with the same $D_{\rm ped}$ and $D_{\rm SOL}$ switching on the vaporisation, which decreases the wall load above the vaporization threshold obtained at $W_{\rm ELM}$ = 4.0 MJ. Fig. 2 shows the carbon density along SOL. The FOREV simulations for ITER SOL contamination have been done in the range $W_{\rm ELM}$ = 3.5 - 12.1 MJ. The evaporated carbon plasma fills SOL with the densities $n_{\rm C} = 10^{20}$ to 10^{21} m^{-3} that depends on $W_{\rm ELM}$. The simulations demonstrated rather uniform density distribution along SOL at $t > 2t_{\rm ELM}$. After a few ms the plasma temperature T in SOL drops to 1-2 eV, which is due to the radiative cooling. Assuming the classical cross-diffusion of the post ELM carbon plasma into the pedestal, the carbon influx is estimated as being limited by $j_{\rm C}$ [m^{-2}] =

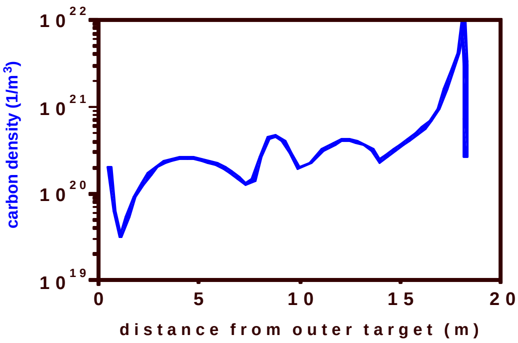

FIG. 2. Separatrix carbon density along SOL at 1.1 ms after start of ELM (W_{ELM} = 12.1 MJ, t_{ELM} = 0.5 ms)

$1.4\times10^{-21}(Z_C n_C[\mathrm{m}^{-3}])^2/T[\mathrm{eV}]$, where the charge state $Z_C = 6$. This estimation fits to the half of calculated carbon amount remaining in the SOL after the ELM.

3. Multiple ELMs induced carbon contamination of the core

With the output data of FOREV, the core contamination was calculated for the whole ITER discharge using TOKES as mentioned in the Introduction. Carbon ions having penetrated after the ELM into the periphery of confinement region diffuse into the core and re-radiate electron thermal energy from the whole plasma volume. Preliminary TOKES simulations for the carbon penetration into the core used a simplified approach in which a rare impurity diffuses into single-fluid plasma [4]. Now an elaborated model is implemented in TOKES. The diffusion process is modelled based on collisional interactions among arbitrary number of different ion species present in the fusion reactor: D^+, T^+, He^{+2} and C^{+6} so far, and a large value of n_C obtained using FOREV became adequate.

A magnetic configuration was created by TOKES for the PF coils of ITER design and then fixed. In this simulation the plasma temperature is also assumed be fixed and equal for all plasma species, dropping as the parabolic dependence of the poloidal magnetic flux from 12 keV at the magnetic axis $\mathbf{r}_0$ to 7 keV at the separatrix, which in carbon-free plasma corresponds to the calculated a-fusion power W_{fus} = 80 MW. In the scenario, with a period t instantaneous injections of N_C carbon ions into the last plasma layer at the periphery occur. Fig.3 demonstrates the contamination by C after single injection at $N_C = 2.45\times10^{20}$. During one second the ions penetrate into the core and then are entrained with the diffusing plasma back to the periphery.

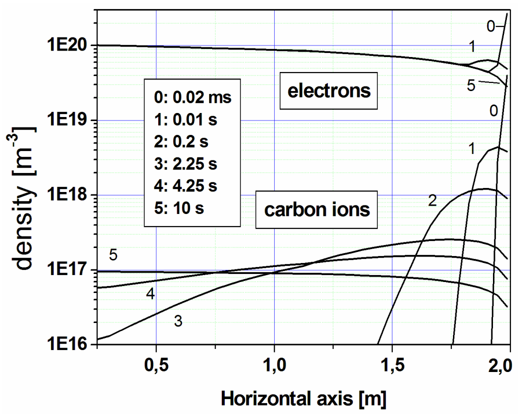

FIG. 3. Electron and carbon densities on a horizontal coordinate originated at $\mathbf{r}_0$.

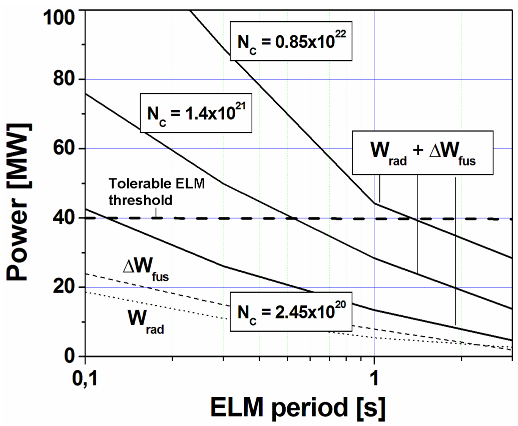

FIG. 4. The radiation and fusion power losses as functions of ELM period

The repetitive ELMs produce more and more carbon in the bulk, but due to simultaneous entrainment the core carbon density and thus the radiation loss power gradually saturate and the fusion power as well. The carbon contamination decreases W_{fus} and increases the radiation losses, so that the total loss power due to ELMs is the sum of the radiation power and the induced fusion power drop ΔW_{fus}. After each ELM the radiation power gets a sharp short increase negligible in the energy balance, and then it decreases to some inter-ELM value W_{rad}. The tolerable ELM period t_{min} corresponds to the loss power $\Delta W = W_{rad} + \Delta W_{fus}$ below some value ΔW_{max}, which seems reasonably to assume as $\Delta W_{max} = W_{fus}/2$. Fig. 4 shows the saturated values of W_{rad} and ΔW_{fus} for $N_C = 2.45\times10^{20}$, and also ΔW in the range of $N_c = 2.45\times10^{20} - 0.85\times10^{22}$ as functions of t. In the FOREV data the tolerable ELM energy fluxes Q for N_C in this range correspond to $Q = 0.8 - 1.1$ MJ/m^2 and thus $t_{min} < 1.5$ s. Hence, the non-tolerable ELM sizes at $t_{ELM} \sim 0.5$ ms are above 1 MJ/m^2.

4. Implementation of neutrals, vessel surface and magnetic surfaces in TOKES

In TOKES the vessel surface and the magnetic surfaces obey the toroidal symmetry, being of arbitrary poloidal cross-section shape. Neutral influx into the confinement regions is modelled using a special triangular grid coupled with the magnetic surfaces, which provides fast calculations of plasma-neutrals interactions [5]. In the poloidal cross-section, the PF flux contours are approximated with line segments situated in the triangles. Due to the same algorithm for each wall segment, uniform calculations for the whole vessel surface are achieved including the divertor, the dome and the main wall, and the PF flux coordinates uniformly cover the whole vessel including the regions of confined plasma, SOL and the divertor legs. For modelling the neutral particles (atoms, photons and neutrons) the Monte-Carlo method is applied. They are spread over the volume as random beams localized in specified solid angles, for instance the wall emitted atoms inside $2p$ (half-isotropic).

All ions arriving at the wall during one time step are assumed to recombine at the surface. They and other arriving atoms are collected as computer data with regard to their numbers and the energy they brought with, separately for each kind of atom and wall segment. The recombination energy is added to some data item Q_w that accumulates the heat absorbed in a wall segment; the incoming electromagnetic radiation is also added to Q_w. The energy of atoms accumulated during a time step t is divided by their collected number in order to obtain the averaged impact energy E_a, which is used for the calculation of the sputtering yield. The striking particles may produce significant wall heating up to the vaporization threshold. Using Q_w, the wall temperature $T(t,x)$ is simulated with a finite-difference heat conduction equation as a function of time t and the local depth x.

The toroidal symmetry of the distribution of atoms emitted from the wall surface allows the reduction of calculations to one starting toroidal angle z. Simulation for one straight trajectory that starts at $z = 0$ and stretches into the vessel (representative ray) describes the whole symmetric set of the trajectories parameterized by z. A ray either stretches until full absorption in the plasma or penetrates the vessel striking the opposite wall. Fig. 5 demonstrates the triangle grid (dashed lines) and the straight rays seen in the poloidal plane as the (solid) curves. The emitted atoms propagate until being absorbed either by the walls or ionized at the plasma periphery. The interaction of atoms with the plasma is calculated sequentially along the ray, triangle by triangle. The ionization and charge-exchange algorithms are implemented in TOKES, but only preliminary simulations are achieved so far.

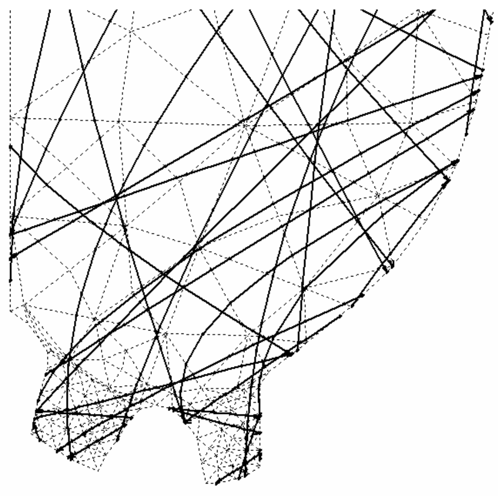

FIG. 5. ITER poloidal plane with the rays representing neutrals on the triangle mesh

The TOKES calculates at each time step the plasma currents and the coil currents assuming toroidal symmetry about the main axis z. The plasma is assumed to be distributed over a set of "magnetic layers" that follow the magnetic surfaces. In the poloidal plane the lines of magnetic layer contours, like the ray lines, are ending at the triangle sides. This structure provides optimal coordination of the rays and the plasma. The PF flux w is calculated only at

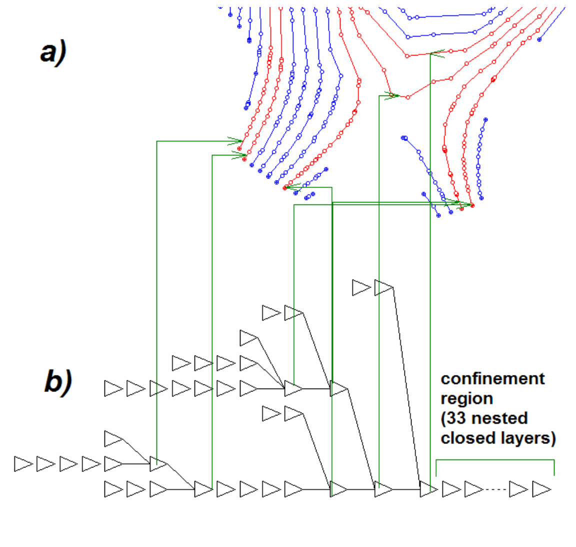

FIG. 6. A fragment of ITER vessel is shown, with the magnetic layers (a) and the respective graph (b) in which one (turned) symbol D stands for one magnetic layer.

the triangle nodes, and linear interpolation is employed for getting intermediate $w(r,z)$ at either r- and z-coordinates of the cylindrical frame, providing a unique set $\{k\}$ of magnetic layers with different constant magnetic flux values w_k. Multiple-mapping spatial variations of w_k are possible in the code, which enables also the calculations in presence of magnetic islands. Going away from the magnetic axis of the core, w_k of sequential contours change monotonically up to the separatrix, outside of which the contours are unclosed. The PF coils produce there several regions (maps) of monotonic decrease or increase of w_k near the wall surfaces.

The neighbour regions of monotonic behaviour of w_k are described ordering the maps in a graph structure that is automatically generated in the code after each time step [6], as Fig. 6 demonstrates it. One original (right side) layer Δ is followed by chains of its descendants, "dads" and "sons" in TOKES terminology. Each dad but the last one of a chain has only one son, and the last Δ of a chain either has several sons or none. In such a graph structure, each separate region of monotonic behaviour of w_k is represented with one Δ-chain, for instance the whole confinement region.

The confined plasma diffuses into the periphery regions where the longitudinal transport is assumed be comparable with anomalous cross-diffusion. At each time step the code calculates layers geometrical parameters such as volume and the safety factor and the interface areas between magnetic layers, which are necessary for calculating the plasma fluxes. A special algorithm described in [6] allowed the plasma diffusion throughout the graph. In addition, some other algorithms for the graph-organized magnetic layers, for the thermal conductivity and convective contributions to the transport equations, have also been developed.

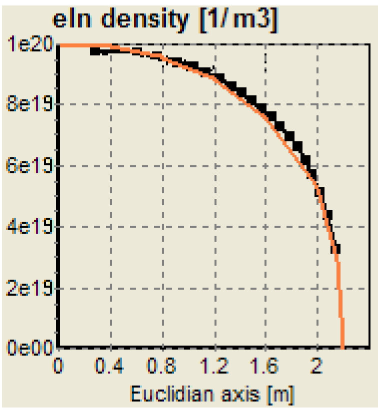

FIG. 7. Analytical (thin curve) and TOKES solutions for D_r.

The algorithms need validations. So far only the numerical scheme developed for plasma diffusion was checked comparing TOKES results with several alternative solutions [7]. Instead the ITER configuration, confined plasma of circular cross-section and a very large major radius ($r_0 = 10^3$ m) has been modelled. The large r_0 allowed independent analytical solution of diffusion equation. The plasma diffusion was also simulated with TOKES after creating deuterium plasma of the temperature $T = 10$ keV and density $n = 10^{20}$ m^{-3} on the closed magnetic layers. Three different diffusion models were investigated both numerically and analytically: those with the classic diffusion coefficient D_c, the Bohm diffusion (D_B) and some artificial reference diffusion (D_r). The D_c is proportional to n, D_B is constant for this model, and $D_r \propto n^2$ is assumed because in this case the density profile is similar to that of H-mode. The comparison showed a good fitting of numerical approach to the analytical one (as example see Fig. 7). These tests were also verified with the MHD transport code ASTRA for the same plasma configuration and diffusion coefficients, and the corresponding plasma density profiles calculated by ASTRA are obtained to be in a good agreement with that of TOKES.

5. Experimental activities at the facility MK-200UG

At the plasma gun MK-200UG the experiments on the interaction of hot magnetized plasma streams with carbon-fibre composite (CFC) and tungsten (W) were carried out. Earlier the plasma material interactions were investigated at plasma loads relevant to hard disruption. The materials have been tested under impact of hydrogen plasma streams with the following peak parameters: energy density $Q = 15$ MJ/m^2, pulse duration 0.05 ms, impact ion energy 2.5 keV and the magnetic field B up to 3 T. It was shown that the powerful plasma stream causes an evaporation of a thin surface layer and produces a cloud of evaporated and ionized material, which acts as a thermal shield that prevents further excessive evaporation. Due to the shielding effect, surface damage caused by evaporation is quite small. However, plasma - induced erosion results not only from evaporation but also from macroscopic erosion processes such as brittle destruction for CFC and melt layer splashing and motion for W. The erosion products are emitted as carbon grains or metal droplets that penetrate into the plasma.

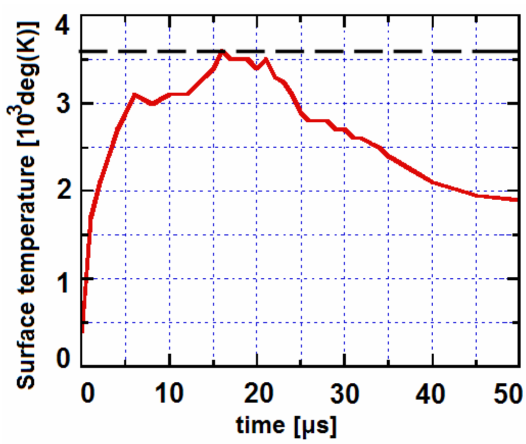

FIG. 8. Surface temperature at MK-200UG in vicinity of tungsten melting threshold

In recent years the responses of the armour materials to the heat fluxes relevant to ITER ELMs have been studied, in particular aiming at production of impurities and their transport in the plasma. The facility has been updated for lower plasma loads. At present the materials are tested with Q from 0.1 to 1 MJ/m^2 and B up to 2 T. Fast infrared pyrometer is applied for online temperature measurements on the target surface at varying heat fluxes, and time- and space-resolving visible spectroscopy for determination of the moment when evaporated material appears near the surface. For CFC, vapour plasma was studied by visible and VUV spectroscopy near the target surface up to the distance 15 cm.

For carbon plasma, the ion ionization states, the plasma temperature and density and the velocity of carbon vapour have been measured as functions of Q. As a reference result, the excessive evaporation was measured at $Q \geq 0.2$ MJ/m^2. These investigations are in progress. For W-target, the melting threshold Q_{melt} and the evaporation threshold Q_{evap} have been determined. The reference results are Q_{melt} = 0.3 MJ/m^2 and Q_{evap} = 0.7 MJ/m^2. Fig. 8 demonstrates the W surface temperature as function of time during the shot with Q = 0.28 MJ/m^2, which is close to the melting threshold (the melting temperature is shown with dashed line). Those findings are important for validation of numerical codes developed at FZK.

6. Conclusions

The improved FOREV simulations of carbon inflow into ITER pedestal combined with elaborated TOKES model for penetration of carbon impurity into the core allowed a prediction of tolerable ELM energy with taking into account the capabilities of ITER plasma to withstand the ELM caused wall damage. However, for more accurate predictions the opacity data for radiation transport and the transport coefficients need to be improved. The implementation of ITER surface shape and the main wall processes including heat transport, surface evaporation, and sputtering, seems a significant step of TOKES development. The random neutral fluxes propagating through the vessel and the neutral-plasma coupling achieved by means of triangular meshes which overlap the multi-mapping magnetic surfaces of arbitrary poloidal plane shapes provided the computational flexibility and universality necessary for future use of TOKES in two-dimensional integrated tokamak modelling. The next urgent steps for elaborating the plasma wall interactions include further improvements of radiation transport algorithms in FOREV, accounting for each ionization stage at plasma simulations, implementation of other wall materials (beryllium and tungsten) and improvements of data for different wall processes. To provide the data necessary for validation the codes, the experiments at the plasma gun MK-200UG with evaporated material plasmas of CFC and tungsten should be continued.

Acknowledgements

This work, supported by the European Communities under the contract EFDA/05-1305 of Association between EURATOM and Forschungszentrum Karlsruhe, was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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