Modelling of core tungsten transport and its control in ITER H-mode scenarios

A. Loarte[1], M. Hosokawa[1], A. Polevoi[1], F. Köchl[2], V. Parail[3], M.L. Reinke[4] E. Belli[5], J. Candy[5], R. Dumont[6], D. Zarzoso[7]

[1]*ITER Organization, Route de Vinon-sur-Verdon, 13067 St Paul Lez Durance, France*

[2]*Atominstitut, Technische Universität Wien, Stadionallee 2, 1020 Vienna, Austria*

[3]*CCFE, Culham Science Centre, Abingdon, OX14 3DB, United Kingdom*

[4]*Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831-8072, USA*

[5]*General Atomics, P.O. Box 85608, San Diego, CA 92186-5608, USA*

[6]*CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France*

[7]*PIIM-UMR 7345-Universit´e d'Aix-Marseille-CNRS, 13397 Marseille Cedex 20, France*

1. Introduction

The decision to start ITER operation in the non-active phase (H/He plasmas) with a W divertor has brought increased attention to physics issues related to high Z impurity transport and control. Lack of impurity control could lead to the radiative collapse of plasmas by W accumulation and increased disruptivity, which is detrimental to ITER operation. Control of W in H-mode plasmas requires, as a first step, the control of W production and its transport into the core plasma through the SOL and edge transport barrier. In addition, even when the concentration of W at the pedestal is kept at low levels, unfavourable core W transport can lead to its uncontrolled accumulation and to loss of the H-mode due increased radiation in present experiments (e.g. JET and ASDEX-Upgrade) [1, 2], where schemes have been developed to avoid W accumulation by the application of additional heating to provide stable H-mode operation with W plasma facing components.

An extensive modelling study has been carried out to evaluate the transport of W in the core plasmas and the demonstration of its control by the available heating and fuelling schemes to provide stable H-mode operation in ITER. Stationary H-mode phases have been modelled with the ASTRA and NEO codes and confinement transients (i.e. H-L transitions) have been modelled with the JINTRAC code suite. In these simulations transport is modelled by neoclassical transport plus anomalous transport from the GLF23 model. The study encompasses a range of ITER DT H-mode plasmas from 7.5 to 15 MA. A range of assumptions regarding electron heat transport and DT particle transport in the central region of the plasma (where anomalous transport is found to be negligible) have been considered.

2. Modelling of W transport in stationary phases of ITER DT scenarios

The ITER simulations show that W transport is anomalous for most of the plasma cross section except in the central region (typically $r/a \leq 0.2$-0.3) and is characterized by a large value of the diffusion coefficient with a negligible pinch velocity, as shown in Fig. 1. In the central region W transport is neoclassical and thus determined by the local ion temperature and density gradients, which are found to be modified by the ITER heating and current drive systems even for plasmas dominated by alpha heating; the ITER RF-based heating systems provide peaked power deposition profiles so that the electron (ECRH) and ion (ICRH-He^3 minority) power densities exceed those of alpha heating in the central plasma region (see Fig. 2).

In general, for 15 MA Q ~ 10 plasmas with P_{NBI} = 33 MW and P_{RF} = 20 MW (ECRH or ICRH, with a range of deposition locations), it is found that the level of core W peaking in

ITER is very modest (see Fig. 3 and Fig.4), which is in agreement with ITER-like experimental results in Alcator C-Mod [3]. This is caused by the low central density gradients in ITER due to the very small particle density source rate provided by the NBI system (E_{NBI} = 1 MeV). The ITER particle density source rate is more than 10^2 times lower than that of JET for the same levels of P_{NBI} so that the central density gradient is solely determined by neoclassical transport physics for the DT ions, when ion-scale turbulent processes are considered for anomalous transport. Recently it has been pointed out that electron-scale turbulent processes may have an effect on energy and particle transport [4]. When this effect is considered in the modelling in a simplified way (by assuming an anomalous DT particle transport coefficient proportional to the electron thermal diffusivity from ETG modes) then W density profiles in ITER can become hollow for cases with central ECRH or ICRH heating (see Fig. 4).

Modelling for 7.5 MA/2.65T plasmas shows similar results regarding W transport and W density profiles to those for 15 MA Q = 10 plasmas. In this case the differences between the various auxiliary heating combinations explored is larger (P_{NBI} = 33 MW and P_{RF} = 0 and 20 MW (ECRH or ICRH)) with somewhat more peaked W profiles for the configurations with lower central power densities. This is expected because of the low level of alpha heating in these 7.5 MA plasma conditions. On the other hand, as $\langle n_e \rangle$ is lower in 7.5 MA than in 15 MA plasmas, the electron and ion temperatures are less coupled leading to a larger T_e/T_i ratio at 7.5 MA than at 15 MA in ITER. Thus, when a residual effect of ETG turbulence on DT particle transport is assumed in the simulations, the DT profiles become very flat in the central region and correspondingly the W density profiles are very hollow. In summary we can conclude that, on the basis of our study, W transport is expected to be similar for 7.5 MA and 15 MA plasmas in ITER so that the results of the development of W control strategies at the beginning of operations at low plasma currents will be applicable to 15 MA high Q plasmas.

3. Modelling of effects of toroidal rotation and fast particles on neoclassical W transport

The calculations carried out with ASTRA and JINTRAC contain a simplified model for neoclassical W transport (NCLASS) that does not describe effects which have been identified to be of importance for high Z impurity transport [5]. One of them is the centrifugal force effect due to the toroidal rotation of impurities with the main DT ions (which can become supersonic for high Z impurities). The other is the effect of high energy particles which are ubiquitous in ITER plasmas (3.5 MeV alpha particles, 1 MeV NBI ions and ICRH accelerated minority ions). These two effects have been modelled with the NEO code [6] using plasma parameters (including toroidal rotation profiles) and fast particle energies from ASTRA as well as impurity minority energy distributions from EVE/AQL [7]. NEO simulations show that both the diffusion and pinch velocity for W increase in the same proportion with the plasma toroidal rotation velocity so that the stationary W core density peaking remains constant for a wide range or Mach numbers (up to M = 0.4); for larger Mach numbers the pinch velocity decreases and can reverse sign, which is beneficial for W accumulation control. The toroidal rotation Mach number predicted by ASTRA is very low for ITER (M = 0.02) due to the momentum diffusivity assumed ($\chi_\phi = \chi_i$) and the low momentum input provided by the 1MeV NBI. Thus no major effect of toroidal rotation on W transport is expected (i.e. the simulations carried out are appropriate) unless intrinsic toroidal rotation is very large in ITER, which seems unlikely.

Analysis of the effects of fast particles on W transport shows that these are negligible for alpha particles and NBI ions due to their large energy (low collisionality with W ions), low density and moderate density/energy radial gradients. On the contrary, the effects associated with He^3 ICRH accelerated ions are significant because of their moderate density ($n_{He}{}^3/<n_e> \sim$ 2%) and their very steep radial energy gradients due to the localized ICRH power absorption in the resonant layer. The screening of W provided by these steep energy radial gradients acts positively on core W transport and can even reverse the sign of the inwards pinch (i.e. leading to hollow W density profiles) for the highest levels of P_{ICRH} considered (20 MW), as shown in Fig. 5. However, this positive effect requires fine-tuning of the ICRH scenario (P_{ICRH}, $n_{He}{}^3/<n_e>$, f_{ICRH}) for each plasma condition considered in order to ensure that the collisionality of the He^3 minority ions with W and their radial energy gradients in the area of interest (plasma centre) are optimized for their beneficial effect on W transport.

4. W transport in the exit phase of ITER Q = 10 H-modes

While the situation regarding core W control during the stationary phases of ITER H-modes is rather positive, core W control is found to be more complex in our first studies of the termination phases of ITER H-modes. For optimum radial position control of the plasma in ITER, it is desirable that the plasma energy decreases with timescales of $t_{decay\text{-}min} \sim 5$ - 10 s in the termination phase of high Q H-modes. The $t_{decay\text{-}min}$ timescale is determined by the penetration of the vertical field through the ITER vacuum vessel and the time constants of the ITER Poloidal Field Coils. The decrease of the plasma energy in such timescale can be achieved by a slow ramp-down of the auxiliary heating power, which together with alpha heating, extends the H-mode phase and thus the timescale of plasma energy decay over $t_{decay\text{-}min}$ [8]. Modelling of W transport with the JINTRAC code has been carried out for two such Q = 10 H-mode terminations: one in which the auxiliary heating (P_{NBI} = 33 MW and P_{ECRH} = 20 MW) is abruptly switched-off and one in which it is ramped-down gradually in 10 s. The results of the simulations for two core DT particle transport assumptions (see Fig. 6) show that strong W accumulation (with timescales of ~ 5 s) can occur when the auxiliary heating power is gradually decreased, which is optimum for radial position control. Further high Q H-mode termination studies will be carried out to optimize the ramp-down of the auxiliary heating power (as well as the type of heating scheme applied in this phase) and of the pellet fuelling so that the requirements regarding radial position and W control can be fulfilled simultaneously.

It is important to note that the quantitative features of the ITER simulations in this paper depend on physics assumptions on: a) the residual level of ion and electron thermal transport in the core region, where anomalous transport is low and b) the particle transport physics in the core region, in particular on the magnitude of the neoclassical DT pinch and on whether there is a residual low level of particle transport ($D >> D_{neo} \sim$ few 10^{-2} m^2/s) in this region or not. It is thus crucial to improve the central plasma transport physics basis and its experimental validation to refine the predictions for core W transport and its control in ITER.

Disclaimer: The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

6. References

*[1] C Angioni, et al, Nucl. Fusion **54**(2014)083028.*
*[2] R Dux, et al, Pl. Ph. Cont. Fus. **45**(2003)1815.*
*[5] FJ Casson, et al, Pl. Ph. Cont. Fus. **57**(2015)014031.*
*[6] E Belli, et al, Pl. Ph. Cont. Fus. **54**(2012)015015.*

[3] A Loarte, et al, Phys. Plas. 22(2015)056117.
*[4] N T Howard, et al, Phys. Plas. **21**(2014)112510.*
*[7] RJ Dumont, et al, Nucl. Fusion **53**(2013)013002.*
*[8] A Loarte, et al, Nuclear Fusion **54**(2014)123014.*

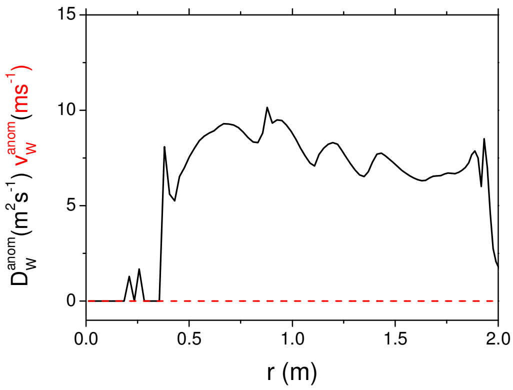

Figure 1. Anomalous diffusion coefficient and pinch for W transport in typical ITER Q = 10 plasma conditions.

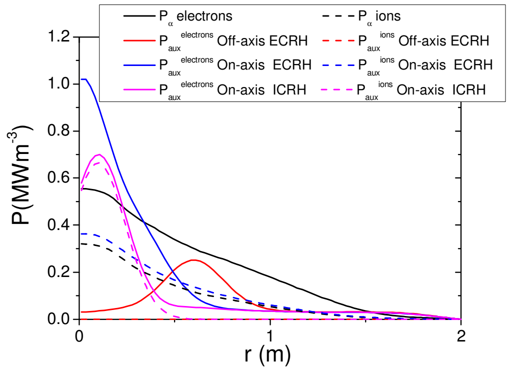

Figure 2. Ion and electron power density profiles for alpha heating and NBI+ECRH or NBI+ICRH (He^3 minority 50:50 electron-ion) heating for ITER Q = 10 plasma conditions.

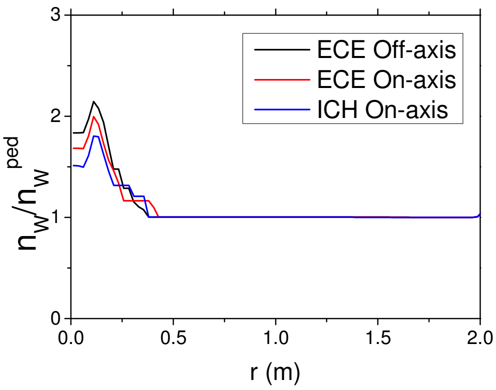

Figure 3. W density profiles, normalized to the pedestal value, for ITER Q = 10 plasmas with 33 MW of NBI and a range of RF heating schemes. Core particle and energy transport assumptions are those of neoclassical transport for all species (ions and electrons).

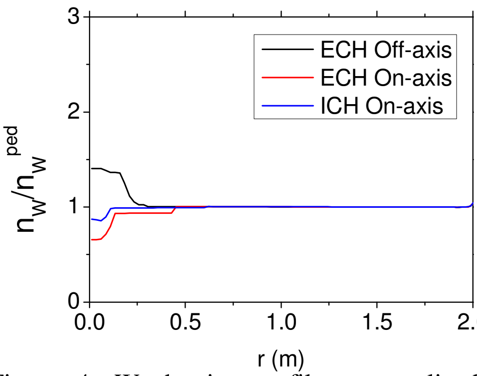

Figure 4. W density profiles, normalized to the pedestal value, for ITER Q = 10 plasmas with 33 MW of NBI and a range of RF heating schemes. Energy transport assumptions are: $\chi_{i,e} = \chi_{i,e}^{neo} + \chi_{i,e}^{GLF}$. Particle transport assumptions are: $D_{DT} = D_{DT}^{neo} + D_{DT}^{GLF} + 0.1\ \chi_e^{ETG}$, where the last term accounts for particle transport driven by ETG turbulence.

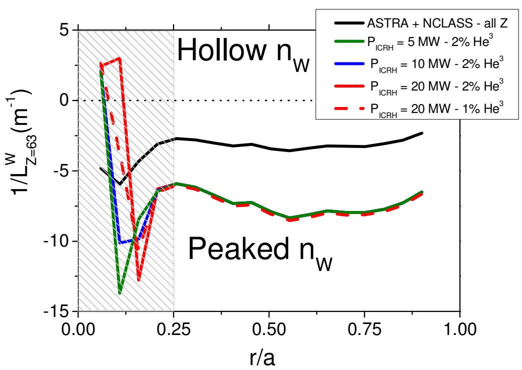

Figure 5. Inverse scale-length of the W density profiles evaluated on the basis of neoclassical transport with NCLASS (thermal plasma) and NEO including the effect of high energy He^3 minority ions at 2% level plus one case with 1% (not self-consistently evaluated). The dashed region corresponds to the plasma core where neoclassical transport alone is found to describe W transport.

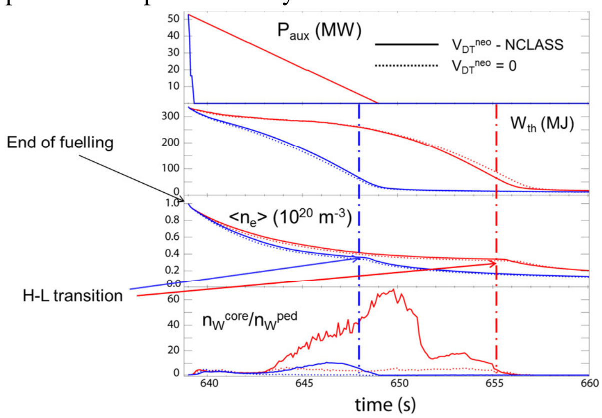

Figure 6. Evolution of auxiliary heating power and plasma parameters (energy, density and W peaking) in the termination phase of ITER Q=10 plasmas with a fast and a slow ramp of the auxiliary heating power. Full lines correspond to simulations which include a neoclassical pinch for the DT ions and dashed lines to those that do not include it.