Physics basis for the ITER tungsten divertor

R. A. Pitts[1], X. Bonnin[1], F. Escourbiac[1], T. Hirai[1], J. P. Gunn[2], A. S. Kukushkin[3], M. Lehnen[1], V. Rozhansky[4], E. Sytova[1,4], G. De Temmerman[1]

[1] *ITER Organization, Route de Vinon-sur-Verdon, CS 90 046, 13067 St Paul Lez Durance Cedex, France*
[2]*CEA Cadarache, F-13108 St Paul lez Durance, France*
[3]*NRC Kurchatov Institute, 123182 Moscow, Russia*
[4] *Peter the Great St.Petersburg Polytechnic University St.Petersburg, 195251 St.Petersburg, Russia*

richard.pitts@iter.org

Building on about 20 years of physics simulation, engineering design and component testing, the ITER tokamak divertor is the largest and most complex ever to be constructed. At the time of the last report to the PSI Conference Series on the ITER divertor status in 2012, the strategy to begin operations with full-tungsten (W) armour had been proposed by the ITER Organization (IO) and was under study. The decision was taken formally in 2013, since when the physics basis in support of the final design has been further developed, with invaluable and numerous contributions from the research community within the ITER Parties. On the eve of component procurement, this paper will discuss the present basis, beginning with a reminder of the key elements defining the overall design, and outlining relevant aspects of the Research Plan accompanying the new “4-staged approach” to ITER nuclear operations which fix the overall lifetime constraint of the first divertor.

The main focus will be on steady state and transient power fluxes in both non-active and DT phases, the main drivers for design and future divertor operation. Stationary loads are obtained from simulations using the 2-D SOLPS-4.3 and SOLPS-ITER plasma boundary codes, assuming the use of the low Z seeding impurities N, Ne and now, for the first time, including fluid drifts, allowing more realistic accounting for in-out target power asymmetries. Imposed by the power handling requirement, the use of W monoblock technology on the divertor targets introduces gap edge heat loading, which must be prevented by surface shaping. This in turn increases the surface heat flux density for given thermal plasma power $q_{||}$ arriving parallel to field lines, with further potential increases due to drifts and narrower than assumed SOL heat flux channel widths. Avoidance of W recrystallization sets an upper limit on the allowed stationary power flux density and since increased fuel gas puffing will increase upstream densities beyond those which may be compatible with disruptive stability, stronger seeding is the principal mitigation route to decrease $q_{||}$ in the event that margins are too eroded. There are, however, limits on allowable impurity concentrations before confinement is compromised and/or full detachment occurs, a process often experimentally observed to be rapid and therefore undesirable from the control point of view. Moreover, whilst N seeding is found to be preferable on today’s all-metal tokamaks regarding performance, the divertor compression of both N and Ne on ITER is predicted to be similar, an important physics issue occupying current R&D. Work is also progressing on the assessment of power loading in the presence of magnetic perturbations for ELM control using the EMC3-Eirene 3-D code suite.

The issue of tolerable limits for transient heat pulses is still an open question and will be addressed here. Although a new scaling for ELM power deposition has shown that there may be more latitude for operation at higher current without ELM control, the ultimate limit is likely to be set more by material fatigue under large numbers of sub-threshold melting events. In the case of disruptions, recent simulations have shown that W vapour shielding should provide significant surface power flux mitigation.