Plasma-Facing Material Development Strategy in ITER

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The plasma-facing components (PFCs) of ITER, particularly divertor targets will have to withstand high heat loads both during steady state operation and transient events. These off-normal events, which deposit impulsive heat loads on PFCs, include disruptions, vertical displacement events (VDEs) and ELMs (Edge-Localised Modes) deposit. In ITER, where the plasma stored energy will exceed that found in the largest current tokamaks by one to two orders of magnitude, off-normal events may result in serious consequences for PFCs and operation of the device itself. The concern is that the lifetime of certains PFCs could be significantly shortened if no mitigation or avoidance measures are taken. Quite apart from lifetime issues, the impurity influxes associated, for example, with the transient ELM power fluxes could have considerable influence on the discharge evolution. Dust generated by transient plasma-wall interaction (PWI) under severe disruptions, if produced in the quantities currently estimated would have a strong impact on machine operation and availability. Schemes for prediction and mitigation or avoidance of disruptions and ELMs must therefore be developed during construction and demonstrated in the early phases of ITER operation when plasma stored energies are lower. The search for operational regimes with smaller ELMs in current devices continues to be an important priority.

ITER is a nuclear device and safety considerations form a major component of design and operational guidelines. In the area of PWI, the control of dust and tritium retention is mandatory in the D-T phase. Although carbon-fibre-composite (CFC) is an attractive PFC material, providing flexibility for a large range of operational parameter space, it may be associated with high levels of dust production and tritium co-deposition and is not a long term option for the nuclear phase. However, in order to maximize the flexibility of the machine during initial operation (H and D phases), CFC will be used as divertor target material, with tungsten in other areas where lower heat fluxes are expected. In order to minimize the tritium retention, W will replace CFC also in the high heat flux areas during the DT phase.

Being a unique preparatory experiment for Demo, ITER is expected to demonstrate routine operation in plasma regimes which lead to acceptable PFC lifetime together with the elimination or significant mitigation of off-normal events, ultimately demonstrating the compatibility of fusion plasmas with reactor-relevant PFCs, such as tungsten. Many aspects of plasma-wall interactions addressed in ITER will not only be of relevance to tokamak reactors, but to fusion reactors in general.