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Strategies for first wall power flux management during plasma current ramp-up on ITER

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The early phases of tokamak plasma current ramp-up are often of very short duration and of little concern on current devices. On ITER, the combination of costly, actively cooled plasma-facing components (PFC) and relatively long timescales ($\sim 10s$) before the transition to X-point configuration, means that power flux management is key if PFC lifetime is not to be compromised. This paper will provide a comprehensive description of the strategies being put in place at ITER to ensure that this critical phase of each plasma discharge is properly managed.

As in many present tokamaks, early ramp-up on ITER will be performed in limiter configuration on the central column, benefiting from the proximity to the resonance location of Electron Cyclotron (EC) start-up assist and lower 3D stray fields produced by currents induced in the vacuum vessel, or due to port openings and ferritic inserts on the low field side [1]. Simulations of plasma magnetic control, performed with the DINA code, are used to design the ramp-up phase [2]. The transition to divertor configuration is typically made $\sim 10s$ after breakdown when $I_p \sim 3.5MA$, and usually assumes some EC heating power at the level of a few MW [2]. Assuming a simple scrape-off layer (SOL) model for parallel heat flux, and taking into account the shaping of the beryllium first wall panels (FWP), this kind of scenario satisfies the constraints of acceptable FWP heat loads and minimizes poloidal flux consumption such that the burn duration will be maximized at high fusion gain.

It has, however, recently become clear that the near SOL heat flux channel width, $\lambda_{q,near}$, may be much narrower than previously thought [1], posing a problem for wall heat loading if FWP alignment is not tightly controlled, and/or the power conducted into the SOL (P_{SOL}) is too high. Although multi-machine scalings of $\lambda_{q,near}$, and of the main SOL heat flux width, $\lambda_{q,main}$, have been used to optimize the inner wall FWP toroidal shaping [1], so called longwave (LW) departures from lack of concentricity of the FW with the toroidal field (TF), significantly increase FWP heat fluxes over the expected values. The current engineering specification for blanket assembly assumes an n=1 LW misalignment and requires that the inboard FW be aligned to a target of Δ_{LW} = $\pm 5mm$. In addition, the ITER Heat Load Specifications [3] set maximum values of $I_p = 5MA$ at $P_{SOL} = 5MW$ for inboard limiter plasmas.

Detailed examination of this situation (see Fig. 1), using 3D field tracing, together with the expected radial SOL heat flux profile and properly accounting for power sharing on the inner wall, shows that for $\Delta_{LW} = 5mm$, the maximum allowable stationary surface heat flux on the inner midplane FWPs will be largely exceeded when additional penalties are included for FWP front surface faceting and tilting. The situation can be further worsened if the assumed SOL transport turns out to be more severe than expected (corresponding to high values of parameter R_q in Fig. 1). This analysis suggests the baseline LW (n=1) requirement may have to be tightened to a new, more challenging target of $\Delta_{LW} \sim \pm 3mm$ to provide sufficient margin for heat loads.

To ensure that inboard FWP power loading during current ramp-up can be properly managed, ITER is adopting a 3-fold strategy:

1. Continuous refinement of the accuracy with which the FW can be aligned to the Tokamak Axis Datum (TAD) on the basis of as built and as-assembled components.
2. Development of a diagnostic system for precision measurements of the TF structure.
3. Study of alternative current ramp-up schemes.

Concerning 1), statistical tolerance accumulation analysis based on a large ensemble of "virtual tokamak builds"finds that, for the case of an n=1 LW misalignment (which assumes the presence of a magnetic "centreline"), the baseline $\Delta_{LW} = 5mm$ criterion can be met if the displacement between the centreline and the TAD is below $3 - 4mm$. If the LW requirement is reduced to $\pm 3mm$, alignment to TAD cannot be achieved. However, if a direct magnetic measurement of the TF structure at the inboard equatorial region can be made with an accuracy better than $\sim 1.5mm$, then the tighter target can be met.

In fact, finite element (FE) simulations of the energization and locking of TF coils show that the real perturbation of the TF structure is likely to be more complex than a simple n=1. In which case the new alignment target will be local rather than global, making a measurement of the field structure even more important. Extensive design activities are now underway for the provision of such a measurement using an array of nuclear magnetic resonance (NMR) sensors (Fig. 2). They will be deployed during the First Plasma commissioning phase (before installation of the main blanket) and target a measurement at half nominal $B\phi$ (2.65 T). An analytic model has been developed to guide this TF Mapping diagnostic design. Using the cylindrical approximation, and tested against full 3D numerical simulations, it yields the perturbed field at given radial location produced by an ensemble of misaligned TF coils. Taking as an example the realistic field structure generated from the FE coil locking simulations, the model shows that a set of 18 toroidally distributed NMR probes in the region of the inboard vacuum vessel wall will be sufficient to reconstruct the TF structure within an error of $\pm 0.5mm$ at the radial position of the key start-up FWPs. Estimates at this stage of the diagnostic design anticipate that the target $\sim 1mm$ measurement accuracy in the spatial location of the chosen field magnitude can be satisfied.

Regarding alternative ramp-up schemes, an option is under study [4] in which I_p is increased up to $\sim 2MA$ in circular plasma configuration at the same rate as in the standard scenario, but is then maintained constant for $\sim 10s$, during which elongation is increased in preparation for the X-point transition. Reducing the level of EC power in this phase also helps to reduce FWP heat loads, at the expense of reduced burn duration.

References

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Figure 1: Left: full bore limited equilibrium at $t \sim 10s$, just before the X-point transition in standard ITER current ramp-up scenarios. Centre: corresponding 3D field line traces of surface heat flux on the inboard midplane FWP#4 where heat fluxes ($q_{\perp}$) are highest for cases with and without n=1 LW misalignment. Right: variation of ($q_{\perp,peak}$) with R_q (ratio of power conducted in the narrow and main SOL heat flux channels) for $\Delta_{LW} = 0 - 7mm$, $I_p = 5MA$, $B_{\phi} = 5.3T$, $P_{SOL} = 5MW$, $\lambda_{q,nearIW} = 4mm$, $\lambda_{q,mainIW} = 50mm$ (IW = inner wall).

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Figure 2: Composite of CAD images for the conceptual TF Mapping diagnostic. Upper: NMR sensor boxes attached to the vacuum vessel wall inter-modular keys used to centre the Blanket Modules (BM). Lower: the cabling from port plug feedthrough to sensor, showing the toroidal distribution of the 27 probes, with 18 on the inner midplane and 9 vertically displaced upwards by 1 BM. These are largely foreseen as risk mitigation in case of failure of any probes on the midplane row.

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