FEASIBILITY STUDY OF AN ACTIVELY COOLED TUNGSTEN DIVERTOR IN TORE SUPRA FOR ITER TECHNOLOGY TESTING

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The current ITER baseline foresees the use of a full tungsten (W) divertor for the nuclear phase with deuterium and deuterium-tritium plasmas. However, the required high heat flux technology has never been tested in the demanding environment of a tokamak under the steady state plasma heat fluxes (~10 MWm^{-2}) expected in ITER. In order to mitigate the risks for ITER, it is proposed to equip Tore Supra with a full W divertor, benefitting from the unique long pulse capabilities of the Tore Supra platform, the high installed power and the long history of experience with actively cooled high heat flux components at the IRFM.
The transformation from the current circular limiter geometry of Tore Supra to the required X-point configuration will be achieved by installing a set of copper poloidal coils inside the lower and upper parts of the vacuum vessel. A wide range of plasma equilibria will become possible, from lower single null to upper single null passing through double null geometries. Plasma volume will be reduced by 30% (from 25 m^3 to 17 m^3) while the maximum plasma current for long pulse operation (several minutes) will be I_p ~1 MA.
The new configuration will allow for easy H-mode access, providing relevant plasma conditions for plasma-facing component (PFC) technology validation. Furthermore with the installed RF power (15 MW), attractive steady-state regimes are expected to be achievable. Simulations with the CRONOS code show, for example, that I_p = 0.65 MA could be fully non-inductively sustained at 0.9 Greenwald density fraction with β_N above 2 and bootstrap current fraction above 50%.
The lower divertor target design will be closely based on that currently envisaged for ITER (W monoblocks), while the upper divertor region will be used primarily to qualify the main chamber wall heat sink technology adopted for the ITER blanket modules (CuCrZr copper/stainless steel) with a tungsten coating (in place of the Be tiles which ITER will use). The in-vessel refurbishment will require existing components, including diagnostics to be modified.
Five years is expected to be required for manufacture of the W divertor elements. This industrial-scale manufacturing and the associated quality assurance testing will alone be of great use to ITER. The final divertor configuration will complement existing W divertor experiments in Europe (e.g. JET and ASDEX-Upgrade) by adding the important component of long pulse capability and actively cooled surfaces to the operational experience being gathered elsewhere. Extended plasma exposure provides access to ITER-critical issues such as PFC lifetime (melting, cracking etc.), tokamak operation on damaged metallic surfaces, real time heat flux control through PFC monitoring, fuel retention and dust production etc.