The new Divertor Tokamak Test facility

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Abstract. Appropriate disposal of the non-neutronic energy and particle exhaust in a reactor is universally recognized as one of the high priority challenges for the exploitation of fusion as an energy source. The new Divertor Tokamak Test (DTT) facility, which will be built in Italy, is a tool to address that challenge in high-field, high performance tokamak with complete integration between core and edge plasma scenarios.

Background. The controlled exhaust of energy and particle from a fusion reactor is a difficult issue that has to be solved before starting the design of DEMO. According to experimental and theoretical work (see for example [1]) one of the major risks comes from the size of the scrape off layer (SOL) power flow decay length λ_q, which – from data taken in existing experiments - scales as [2]:

$$\lambda_q\ (mm) = 1{,}35\ P_{SOL}^{-0.02}\ R^{0.04} B_p^{-0.94} \varepsilon^{0.42} \qquad (1)$$

where P_{SOL} is the power in the scrape-off-layer, R the major radius of the machine, B_p the poloidal magnetic field and ε the inverse aspect ratio. Activity is ongoing to fully assess both theoretically and experimentally the behaviour of λ_q, taking into account the role of turbulence and of the main plasma parameters (see for example [3]) If this scaling holds true for ITER, it means that in that device λ_q is expected to be of the order of 1 mm. Considering the Q=10 scenario, with 500 MW of net fusion power (400 MW brought by neutrons and 100 MW heating the plasma and lost through radiation and thermal/particles losses) the expected power exhausted on the divertor in a low radiation case is about 90 MW. This means a heat flux on the divertor of the order of 50 MWm^{-2}, a value far above the limit of present target materials. To cope with these challenges and following the recommendations of the EUROfusion roadmap [4], the Italian fusion community proposed in 2015 the DTT experiment.

The Divertor Tokamak Test facility. The Divertor Tokamak Test facility [5] (DTT, figure 1) is a superconducting tokamak with 6 T on-axis maximum toroidal magnetic field carrying plasma current up to 5.5 MA in pulses with total length (including start-up, flat-top and ramp-down phases) up to 100 s. The D-shaped device is up-down symmetric, with major radius R=2.14 m, minor radius a=0.64 m and average triangularity 0.3.

The auxiliary heating power coupled to the plasma at maximum performance is 45 MW, shared between the three heating systems used in ITER and foreseen in the present version of DEMO: ion and electron cyclotron resonance heating and negative ion beams. In particular DTT will use 170 GHz ECRH, 60-90 MHz ICRH and 400 keV negative ion beam injectors. DTT will start operation with 8 MW ECRH, which will increase to 25 MW within two years from the beginning of plasma operation. The full 45 MW heating power is planned to be reached within 6 years. The precise sharing between the three heating sources will be defined later. An input power of 45 MW allows matching the P_{SEP}/R values, where P_{SEP} is the power flowing through the last closed magnetic surface, with those of ITER and DEMO. DTT is in fact designed to reach P_{SEP}/R =15 MW/m, assuming P_{SEP}=32 MW. Fast particles up to 700 keV (simulating the alfa generated by nuclear fusion reaction) will be generated through ICHR power while the 400 keV NNBI allows for producing a super-Alfvenic population to study fast particle collective transport and wave particle interaction.

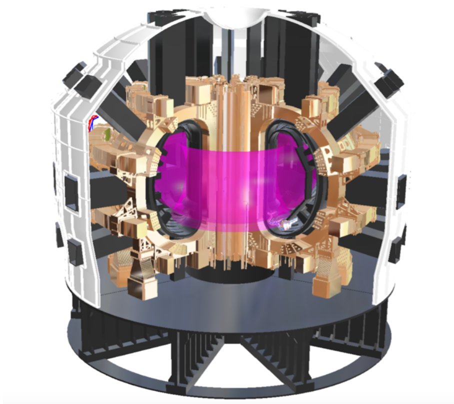

Figure 1: the DTT experiment

The plasma facing material is tungsten, sprayed on the first wall and bulk in the divertor. The entire machine is up-down symmetric to allow the study of double null (DN) divertor configurations and will be maintainable through remote handling. The central solenoid, the toroidal and the poloidal field coils will all be superconducting (mostly Nb_3Sn), with the possibility of adding in a second phase a high-temperature superconducting insert in the central solenoid to test this emerging technology in a fusion environment. Sets of copper internal coils will be used for vertical stabilization, divertor control and for ELM and MHD stability control. A dedicated disruption mitigation system will be installed not only to support plasma operation but also to make DTT a facility to support ITER with dedicated disruption and runaway mitigation experiments at high current and high power.

Plasma configurations in DTT. DTT is being designed with a high level of flexibility, in particular as far as divertor scenarios are concerned. From a magnetic point of view the external coils together with a set of four internal coils will allow to control and optimize the local

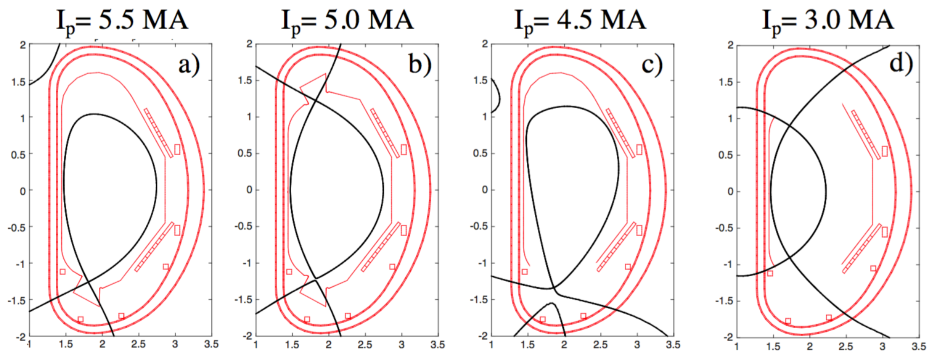

Figure 2: divertor magnetic configurations in DTT. (a) reference single null, (b) double null, (c) snowflake, (d) double super-X

magnetic configuration in the vicinity of the divertor target. Figure 2 shows the main divertor magnetic topologies, which can be produced in DTT. The reference single null, double null and snowflake configurations can be produced at (or close) to the maximum target plasma current of 5.5 MA, while double super-X may be feasible only at significantly lower current (and with vertical stability issues still to be solved). DTT will also have the possibility to study liquid metal divertors.

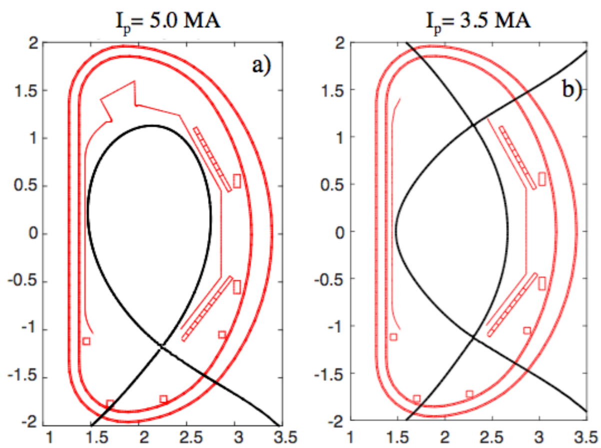

Figure 3: negative triangularity in DTT. a) single null @5 MA, b) double null @3.5 MA

The DTT poloidal field coils system also allows for the realization of scenarios with negative triangularity. Figure 3 shows a 5 MA single null scenario with $\delta = -0.13$ and $\delta_{lower} = -0.16$ and a double null scenario at 3.5 MA with $\delta = -0.38$. Configurations with negative triangularity shape are studied since they can achieve high beta levels and H-mode level confinement while being ELM free [6,7,8]. DTT can provide significant contributions to this line, the performance of which can be optimized with a proper dedicated modification of the first wall and the vessel shells without modifying the toroidal field coils.

The DTT facility is designed to advance the research on divertor physics and technology and on plasma exhaust management in regimes relevant for ITER and DEMO and in regimes where plasma core and edge behaviours are full integrated. Far from being merely an experiment dedicated only to plasma exhaust, DTT will provide a facility to study high performance tokamak physics and to address also core confinement and stability issues, and the management of transient events like disruptions and ELMs. The 45 MW auxiliary heating is significantly higher than the threshold power for the L–H transition, which according to [9] is set at 7.4 MW for operation 3MA/3T and at 21 MW for operation 5.5MA/6T - with Greenwald density fraction of 0.5 for both cases. Moreover, in terms of power DTT – given also its high magnetic field - guarantees agile access to the ELM-free I–mode regime, which potentially has a significant operational range in power (a factor of at least ~ 1.5) while avoiding the H-mode, for $\langle n_e \rangle$ =0.2-0.4 n_{GW} at I_p = 3 MA for high q_{95} operation or for high current I_p = 5.5 MA at MA at q_{95} =3 [10]. For both low ($P_{rad,core}/P_{tot}$ ~ 1/3) and high ($P_{rad,core}/P_{tot}$ ~ 2/3) levels of core radiation fraction the levels of scrape-off layer (SOL) loading in DTT, as evaluated with simple 0-d modelling [10], are unique in the operational space of present and future device and are relevant for both ITER and DEMO. It is important to note that the high-performance design point of DTT is set at $\langle n_e \rangle$ =0.45 n_{GW}, which therefore allows ample space for exploring different regimes. Preliminary fluid modelling of power exhaust is being done for a variety of tungsten divertor configurations both in pure deuterium plasma and with argon and neon seeding. In pure deuterium detachment conditions are obtained with P_{SOL}~10 MW for single null divertor and at highly values for snow flake configurations (P_{SOL}~15-20 MW for SF). Operation of DTT at maximum input power calls therefore for high radiation fractions (P_{rad}/P_{sol} ~ 80-90%), which can be obtained with impurity seeding. Both at low and high separatrix density ($n_{e,sep}$~0.5×10^{20} m^{-3} and ~1×10^{20} m^{-3}) P_{rad}/P_{sol} ~ 90% can be obtained in single null, double null and snow-flake configurations. Zeff at the separatrix =2.3 is needed for high density single null configurations.

Acknowledgments. The authors thank the whole DTT team for the precious contributions.

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