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Design and definition of a Divertor Tokamak Test facility

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The effective treatment of the heat and power exhaust is a critical issue in the road map to the realization of the fusion energy. In order to provide possible, reliable, well assessed and on-time answers to DEMO, the Divertor Tokamak Test facility (DTT) has been conceived and projected to be carried out and operated within the European strategy in fusion technology. This paper, based on invited paper and seven posters presented at the 29th SOFT Conference held in Prague in 2016, provides an overview of the scientific proposal, which is deeply illustrated in a number of papers.

Keywords: Tokamak Reactor; Heat and Power exhaust; Divertor facility.

1. Introduction

The European strategy toward the realization of the Fusion Energy [1] is articulated in 8 missions specifically aimed to face with the main challenges in achieving the overall goal to be studied by ITER [2] and assessed in the Demonstration Fusion Power Plant (DEMO) [3].

Mission #2 is specifically focused to develop a heat and power exhaust system able to withstand the large loads expected in the divertor of a fusion power plant.

A specific activity is in progress to optimise a conventional divertor based on detached conditions to be tested in ITER. However, to prevent the negative effects on DEMO of possible unforeseen, technical or technological difficulties, a new facility has been proposed, the "Divertor Tokamak Test facility" (DTT). Aim of DTT is to assess a set of possible alternative solutions for DEMO, including advanced magnetic configurations and liquid metal divertors.

The DTT proposal, worked out by an International European Team of experts [4-6], demonstrates the possibility to set up a flexible and effective facility able to bridge the power handling gaps between the present day devices with ITER and DEMO experiments, within the European fusion development roadmap.

The status of the project, its strengths and weaknesses, the European and international meaning of initiative, have been discussed in detail at the SOFT 2016 Symposium through an invited talk and 7 posters presentations [7-14]. A thorough presentation of the project will be made on a Special Issue on DTT to be published in 2017 on Fusion Engineering and Design. Therefore, this paper is not intended to be original nor exhaustive. It just makes the point of testify the main aspects of the discussion on the possible DTT contribution to the problem of the power exhaust that animated the Fusion scientific community attending SOFT.

The paper is organized as follows. In Section 2 the main goals of the DTT project are presented together with its main specifications. Section 3 illustrates its main subsystems. Section 4 provides some information about the scheduling of construction and operations. Finally, Section 5 summarizes the status and the perspectives of the project.

2. Project key lines

2.1 Design Specifications

DTT has been conceived as a flexible test bed capable to tackle plasma exhaust issues in a fairly integrated fashion, according to EU Fusion Roadmap Mission suggesting a specific Divertor Tokamak Test (DTT) facility.

The main specifications are here summarized:

(i) the capability to assess the performance of a conventional divertor:

- with dimensionless parameters similar to ITER and DEMO
- with large values of the parameter P_{sep}/R (15 MW/m) and very large radiation (up to 90%), where P_{sep} is the power flowing through the last closed magnetic surface and R is the plasma major radius
- benchmarking the accuracy of the SOL code predictions in ITER and DEMO relevant range of parameters
- testing the effectiveness of closed loop control system of plasma detachment including diagnostics, control algorithms and actuators

(ii) the capability to assess the performance alternative materials and new divertor concepts:

- including liquid metal materials;
- in presence of new or not yet consolidated scenario solutions, including X-Divertor (XD), Snow Flake (SF), Super-X (S-X), negative triangularity or "long leg"
- benchmarking the accuracy of the theoretical models used to model SOL, core and LM targets
- testing the effectiveness of a further optimization process to improve the new divertor concepts for selected configurations

2.2 Physical & Technological Requirements

In order to pursue the objectives of the project, DTT fulfils a number of physical requirements, including:

- preservation of 4 DEMO relevant parameters: T_e, $\nu^*=L_d/\lambda_{ei}$, Δ_d/λ_0, β
- relaxation on normalized Larmor radius: value of (ρ_i/Δ_d) different, but not very far from that of DEMO,
- integrated scenarios: solutions compatible with plasma performance of DEMO

where Te is electron temperature, L_d is the divertor field line length, λ_{ei} is the electron-ion mean free path, Δ_d is the SOL thickness, λ_0 is the neutrals mean free path, β is the normalized plasma pressure.

In addition, a number of technological requirements has been considered, including

- Psep/R ≥ 15 MW/m;
- flexibility in the divertor region in order to test several solutions;
- possibility to test alternative magnetic configurations;
- possibility to test liquid metals;
  integrated scenarios compatible with technological constraints of DEMO.

Of course an additional constraint has been imposed on the construction budget that, for opportunity reasons, has been limited within 500 M€.

2.3 Main Design Parameters

The DTT project proposes a facility characterized by a major and minor radius of 2.15 and 0.70 m, respectively, with a plasma current of 6 MA and a toroidal magnetic field of 6T.

A detailed list of the design data can be found in [4,6-14]. Here in order to have a direct comparison with ITER and DEMO devices, the main parameters are reported in Table I.

Table I Comparison among DTT, ITER and DEMO main parameters [4-6, 14]

	DTT	ITER	DEMO
R (m)	2.15	6.2	8.77
a (m)	0.70	2.0	2.83
I_p (MA)	6.0	15	20
B_T (T)	6.0	5.3	5.8
V_p (m^3)	33	853	2218
<n> ($10^{20}m^{-3}$)	1.72	1.0	0.9
<n>/n_G	0.45	0.85	1.1
P_{Tot} (MW)	45	120	450
τ_E (s) (H_{98}=1)	0.47	3.6	3.4
<T> (KeV)	6.2	8.5	12.6
β_N	1.5	1.6	2.1
υ^* (10^{-2})	2.4	2.3	1.3
ρ^* (10^{-3})	3.7	2.0	1.6
T_{Ped} (KeV)	3.1	4.3	7.0
n_{Ped} ($10^{20}m^{-3}$)	1.4	0.8	0.7
υ^*_{Ped} (10^{-2})	6.3	6.2	2.8
ELM energy (MJ)	1.2	24	140
L-H Pow. (MW)	16÷22	60÷100	120÷200
P_{Sep}/R (MW/m)	15	14	17
λ_{int} (mm)	1.7	2.2	2.2
P_{Div} (MW/m^2) (70% Rad.)	27	27	42
$P_{Tot}B/R$ (MW T/m) $\propto q_{//}$	125	100	290
Pulse length (s)	100	400	7000

3. Main Subsystems

The sectional view of DTT is reported in Fig. 1.

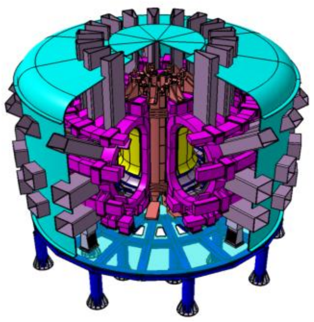

Fig. 1. Sectional view of DTT.

Here a brief overview of the main subsystems of the facility is given; more detailed information can be found in [4, 6-14].

3.1 Magnetic System

The magnetic system includes

- a toroidal system with 18 coils (B_{peak}: 11.4 T, B_{plasma}: 6.0 T, 65 MAt);
- a poloidal system with 6 coils (B_{peak}: 4.0 T, Σ_k $|N_k I_k|$ =21 MAt);
- a central solenoid with 6 coils (B_{peak}: 12.5 T, Σ_k $|N_k I_k|$ =51 MAt; poloidal flux: ±17.6 Vs).

Moreover an additional copper system with 8 coils inside the vessel has been included be used for plasma control or local modifications of the magnetic configuration in the divertor region.

In Figs. 2-3 a number of different configurations are reported, also showing the flexibility and the effectiveness of the Pcoil system.

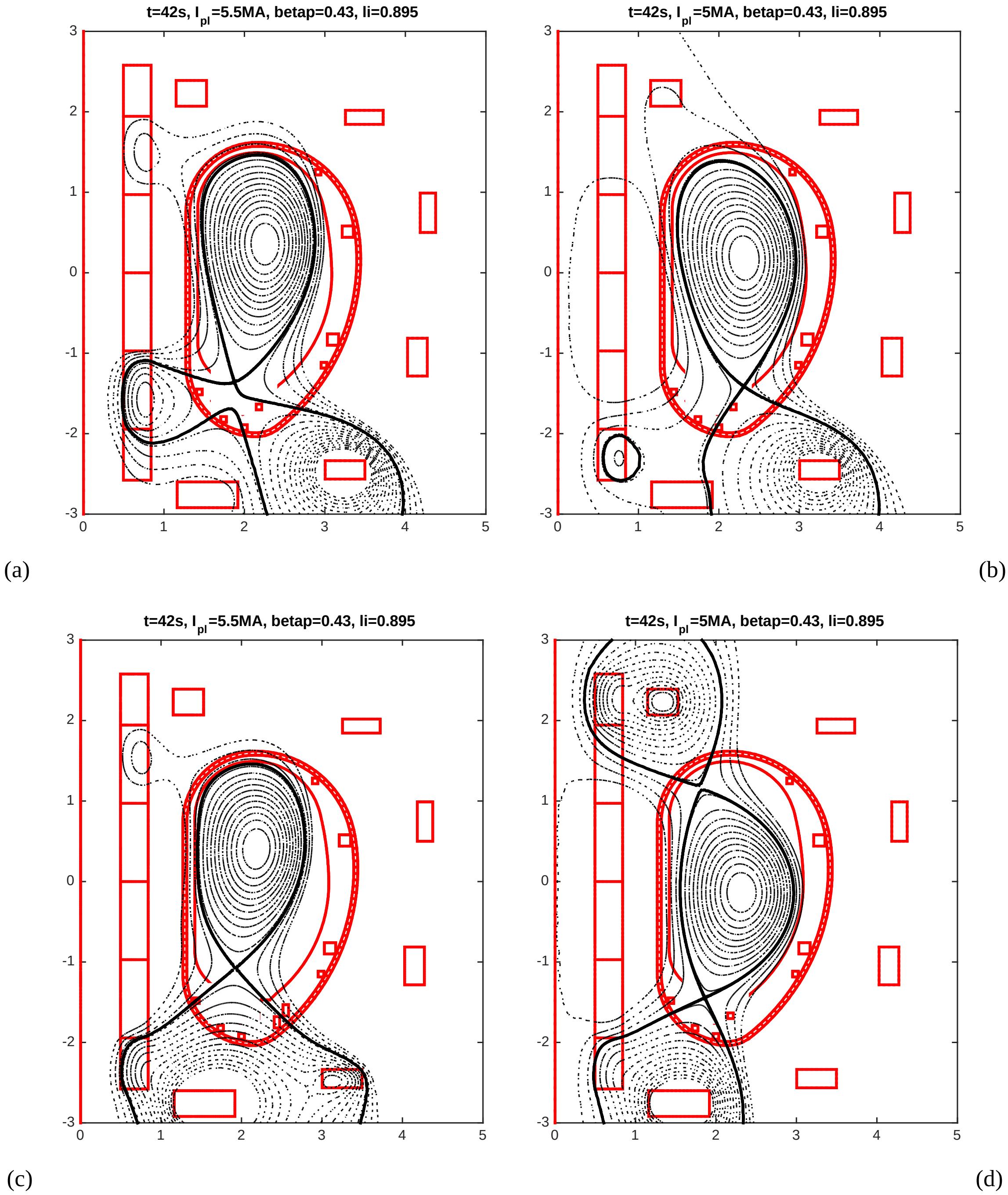

Fig. 2. Examples of alternative configurations that can be achieved at the start of flat top using only the external PF coils: a) Snow Flake Plus (SF^+); b) Reverse Triangularity; c) Super-X like; d) Double Null.

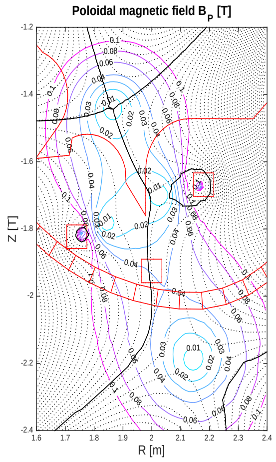

Fig. 3. Use of the in-vessel copper coils to locally modify the topology of the field lines in the divertor region: flux map and poloidal field in an X-D configuration characterized by the flaring of the field line in the vicinity of the target.

3.2 Additional heating

A mix of different heating systems will provide a 45MW power contribution, including ≈15MW ECRH at 170 GHz, ≈15MW ICRH at 60-90 MHz and ≈15MW NBI at 300 keV. During the initial DTT plasma operations 15 MW of ICRH and 10 MW of ECRH will be available.

3.3 Power Supply

The power supplies for the PF coils are based on AC/DC converters are four quadrants, thyristor based, 12 pulses with current circulating able to provide an output DC current ±25 kA and output DC voltage of ±800V for some coils and ±1kV for the others. Just the supply of the coils specifically devoted to the control the vertical position of plasma are IGCT based to be fast enough for the specific real-time needs.

A new connection to the national extra high voltage (EHV) grid at 400 KV has been foreseen by an intermediate dedicated electric substation 400KV/150KV in proximity to an important node with adequate power.

3.4 Vacuum vessel

The Vessel includes

(i) the main vessel: a “D” shaped, 35 mm shell in INCONEL 625, segmented in 18 modules joined by field welding;

(ii) the system of ports for maintenance of the in-vessel components (divertor cassette, first wall) and allocation of diagnostic and heating equipment: 5 access ports for each module.

3.5 Divertor

The main objective of the DTT project is to test several divertor design and configurations, from the standard single null (SN) plasmas to alternative configurations like X Divertor (XD) Snow Flake Divertor (SFD).

This need has inspired the whole project of the facility. In particular the design of VV, ports and RH devices takes into account application and testing of a Liquid Metal Divertor.

Therefore one of the key characteristic pursued by the design has been the flexibility in installing and testing different divertor modules; therefore

- VV and In-vessel components design, independent on the divertor design;
- high modularity and easiness in replacing the divertor by remote handling in a relatively short time;
- VV ports designed to allow easy replacement of different divertor design

DTT will initiate operating with standard SN configuration but the testing with all the possible different magnetic divertor configurations, with the full available additional power, is planned.

3.6 Additional Subsystems

A number of additional subsystems play important roles in the DTT device, including the First Wall; the Shield and Cryostat; the system for Data acquisition, Diagnostics and Control; the Remote handling; the Cooling systems; the Pumping and Fuelling components; the Auxiliary components.

4. Scheduling

According to the European roadmap, the DTT experiment should start its operation in 2023. To be coherent with this plan, the realization of the device will cover a time of around 7 years, starting from the first tender (in 2016 or 2017) up to full commissioning and the first plasma (during 2023). The operations should then cover a period of more than 20 years, up to the initial phases of the DEMO realization.

5. Status and perspectives.

DTT is an European Facility, fully open to international cooperation, projected to face with one of most challenging issue in the fusion road map.

It has been conceived to tackle in a fairly integrated fashion the power handling issues in view of a reliable DEMO design by experimenting innovative plasma operating scenarios and new technologies.

As an additional strategic goal, DTT would like also to test effective schemes in managing and control complex construction and operation machine. In addition it will support the development of a new generation of physicists, engineers, technologists duly trained by highly experienced persons on a up to date device.

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