DTT: a divertor tokamak test facility for the study of the power exhaust issues in view of DEMO

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Abstract
In parallel with the programme to optimize the operation with a conventional divertor based on detached conditions to be tested on the ITER device, a project has been launched to investigate alternative power exhaust solutions for DEMO, aimed at the definition and the design of a divertor tokamak test facility (DTT). The DTT project proposal refers to a set of parameters selected so as to have edge conditions as close as possible to DEMO, while remaining compatible with DEMO bulk plasma performance in terms of dimensionless parameters and given constraints. The paper illustrates the DTT project proposal, referring to a 6 MA plasma with a major radius of 2.15 m, an aspect ratio of about 3, an elongation of 1.6–1.8, and a toroidal field of 6 T. This selection will guarantee sufficient flexibility to test a wide set of divertor concepts and techniques to cope with large heat loads, including conventional tungsten divertors; liquid metal divertors; both conventional and advanced magnetic configurations (including single null, snow flake, quasi snow flake, X divertor, double null); internal coils for strike point sweeping and control of the width of the scrape-off layer in the divertor region; and radiation control. The Poloidal Field system is planned to provide a total flux swing of more than 35 Vs, compatible with a pulse length of more than 100 s. This is compatible with the mission of studying the power exhaust problem and is obtained using superconducting coils. Particular attention is dedicated to diagnostics and control issues, especially those relevant for plasma control in the divertor region, designed to be as compatible as possible with a DEMO-like environment. The construction is expected to last about seven years, and the selection of an Italian site would be compatible with a budget of 500 M€.

Keywords: plasma facing components, divertor, tokamak

1. Introduction

In 2012 EFDA published ‘Fusion electricity—a roadmap to the realization of fusion energy’ [1], which sets out a strategic vision toward the generation of electrical power by a Demonstration Fusion Power Plant (DEMO) by 2050.

[a] See appendix A.
[b] See appendix B.

The roadmap elaborates 8 strategic missions to tackle the main challenges in achieving this overall goal. More specifically, two Work Packages:

- WPDTT1—Assessment of alternative divertor geometries and liquid metal plasma facing components (PFCs)
- WPDTT2—Definition and Design of the Divertor Tokamak Test (DTT) Facility

are articulated within Roadmap Mission 2: 'Heat-exhaust systems'.

'*Heat-exhaust systems must be capable of withstanding the large heat and particle fluxes of a fusion power plant. The baseline strategy for the accomplishment of Mission 2 consists of reducing the heat load on the divertor targets by radiating a sufficient amount of power from the plasma and by producing 'detached' divertor conditions. Such an approach will be tested by ITER, thus providing an assessment of its adequacy for DEMO. However, the risk exists that high-confinement regimes of operation are incompatible with the larger core radiation fraction required in DEMO when compared with ITER. If ITER shows that the baseline strategy cannot be extrapolated to DEMO, the lack of an alternative solution would delay the realisation of fusion by 10–20 years. Hence, in parallel with the necessary programme to optimise and understand the operation with a conventional divertor, e.g. by developing control methods for detached conditions, in view of the test on ITER, an aggressive programme to extend the performance of water-cooled targets and to develop alternative solutions for the divertor is necessary as risk mitigation for DEMO. Some concepts are already being tested at proof-of-principle level in $\leqslant$1 MA devices (examples are super-X, snowflake, liquid metals). These concepts will need not only to pass the proof-of-principle test but also an assessment of their technical feasibility and integration in DEMO, perhaps by adjusting the overall DEMO system design to the concept, in order to be explored any further. The goal is to bring at least one of the alternative strategies (or a combination of baseline and some alternative strategy) to a sufficient level of maturity by 2030 to allow a positive decision on DEMO even if the baseline divertor strategy does not work. As the extrapolation from proof-of-principle devices to ITER/DEMO based on divertor/edge modelling alone is considered too large, a gap exists in this mission. Depending on the details of the most promising chosen concept, a dedicated test on specifically upgraded existing facilities or on a dedicated Divertor Tokamak Test (DTT) facility will be necessary. In either case, it will need sufficient experimental flexibility to achieve the overall target. The facility needs to be ready in the early 2020s and is a good opportunity for joint programming among the EURATOM member states and for international collaboration. As the extrapolation to DEMO will have to rely on validated codes, theory and modelling effort is crucial for the success of this Mission and the simulation tools should provide reliable predictions on the behaviour of plasma edge and heat-exhaust systems in the DTT regimes.*' [1].

The radiation baseline strategy will be tested on ITER [2]; it foresees optimizing plasma operations with a conventional divertor based on detached plasma conditions. This strategy relies upon different factors:

- development of PFCs to cope with very large power fluxes (~$5 \div 10$ MW m^{-2})
- selection of the divertor geometry and of the magnetic flux expansion to reduce the normal heat flux on the target, i.e. by distributing the heat over a larger surface
- removal of plasma energy before it reaches the target via impurity radiation by increasing edge plasma density and injecting impurities in the SOL region, so as to decrease the fraction of the heating power that impinges on the divertor, up to a level compatible with the materials technology
- recycling and increase of density lowering the temperature close to the target, with consequent detachment (the temperature drops below that required for ionization, therefore the particles are neutralized and there is no direct plasma flux or power to the divertor targets).

However, the risk exists that the baseline strategy (conventional divertor solution) pursued in ITER cannot be extrapolated to a fusion power plant:

- today's experiments operate with SOL and plasma bulk conditions that are very different from those expected in ITER and DEMO
- simulations with present SOL models and codes are not reliable when extrapolating to ITER and DEMO conditions
- stability of the detachment front needs to be assessed for ITER and DEMO conditions
- problems might arise related to integration of this solution with the plasma core and the other tokamak subsystems, e.g.:
  - impurity contamination of the core with consequent reduction of fusion performance
  - compatibility of bulk plasma with the very high radiation fraction requested (>90%)
  - compatibility with pumping
  - monitoring of erosion, temperature, etc.

In addition, even if the ITER divertor proves to be successful, it will be difficult to extrapolate to DEMO, because of its additional requirements (different first wall material, more nuclear aspects and thus limited use of some materials, requirements in terms of life expectancy of reactor components and thus need of keeping the temperature low in the divertor region with nearly zero erosion, etc…).

This basketful of problems present the necessity of a dedicated Divertor Tokamak Test facility, flexible enough to study, test and propose a solution that will eventually be used directly on DEMO. Consequently, the DTT facility must be able to realize scaled experiments integrating most of the possible aspects of the DEMO power and particle exhaust.

Under this framework, the Work Package WPDTT2 has been organized with several different working groups and divided into two phases. During Phase 1 two steps were tackled, mainly in support of the WPDTT1 Physics activities, of the advanced divertor magnetic configurations and of the liquid metals. The Phase 2 targets were divided into three steps: (a) definition of the DTT technical requirements; (b) DTT conceptual design; (c) DTT engineering design and construction. The first four steps of the WPDTT2 have been carried out during 2015; the decision of going ahead with the last step (DTT engineering design and construction) is presently under evaluation by EUROfusion members and a final

Table 1. Machine comparison.

	JET	AUG	EAST	DIII-D	ITER	DEMO	JT-60SA	WEST	TCV	ADX	DTT
R (m)	2.98	1.65	1.7	1.67	6.2	8.77	3.0	2.5	0.88	0.73	2.15
a (m)	0.94	0.5	0.4	0.67	2.0	2.83	1.2	0.5	0.24	0.2	0.70
I_p (MA)	3.5	1.6	1.4	2.0	15	20	5.5	1	0.45	1.5	6.0
B_T (T)	3.2	2.4	3.4	2.1	5.3	5.8	2.3	3.7	1.45	6.5	6.0
V_p (m^3)	82	13	10	19	853	2218	141	15	1.85	0.9	33
$\langle n \rangle$ (10^{20} m^{-3})	0.9	0.9	1.0	0.85	1.0	0.9	0.9	0.8	1.2	4.5	1.72
$\langle n \rangle / n_G$	0.7	0.5	0.4	0.65	0.85	1.1	0.8	0.7	0.5	0.4	0.45
P_{Tot} (MW)	30	25	30	27	120	450	41	16	4.5	14	45
τ_E (s) ($H_{98} = 1$)	0.49	0.07	0.07	0.11	3.6	3.4	0.62	0.05	0.027	0.05	0.47
$\langle T \rangle$ (KeV)	3.3	2.5	3.3	2.8	8.5	12.6	3.4	2	0.8	1.7	6.2
τ_R (s)	3.1	0.6	0.7	1.5	73	120	8	0.4	0.04	0.2	5.9
β_N	1.8	2.4	2.2	2.9	1.6	2.1	2.4	2	2.7	2.2	1.5
υ^* (10^{-2})	8.6	8.4	7.4	4.0	2.3	1.3	4.1	35	65	13.1	2.4
ρ^* (10^{-3})	4.0	8.5	8.5	7.2	2.0	1.6	4.5	5.0	17	7.7	3.7
T_{Ped} (KeV)	1.7	1.3	1.7	1.4	4.3	7.0	1.7	0.5	400	1.3	3.1
n_{Ped} (10^{20} m^{-3})	0.7	0.7	0.9	0.7	0.8	0.7	0.7	0.5	0.9	3.8	1.4
υ^*_{ped} (10^{-2})	22.6	22	21	10	6.2	2.8	11	92	170	35	6.3
ELMs En. (MJ)	0.45	0.06	0.07	0.13	24	140	1.1	0.2	0.03	0.02	1.2
L-H Pow. (MW)	9.5÷12	3÷4	3.5÷4.5	3.0÷4.0	60÷100	120÷200	10÷12	4÷6	0.6÷0.8	4÷6	16÷22
P_{Sep} (MW)	21	18	21	18	87	150	29	10	3	9.5	32
P_{Sep}/R (MW m^{-1})	7	11	12	11	14	17	9.5	4	3.4	13	15
λ_{int} (mm)	3.2	3.7	2.6	3.6	2.2	2.2	3.7	3	5.5	1.7	1.7
P_{Div} (MW m^{-2}) (no Rad)	28	44	62	45	55	84	24	25	7.3	110	54
P_{Div} (MW m^{-2}) (70% Rad)	8.6	13	19	13	27	42	7.4	7.5	2.2	33	27
$q_{//} \approx P_{Tot}B/R$ (MW T m^{-1})	32	44	60	40	100	290	22	23	5	125	125
Pulse length (s)	≈20	≈6	??	≈6	400	7000	100	1000	5	3	100

decision is planned for the end of 2016. Meanwhile the Italian Association for Fusion, working within the WPDTT2, has produced a conceptual design for a DTT tokamak facility [3], that should be able to operate with plasma bulk dimensionless parameters very close to the DEMO ones and with a divertor region sufficiently flexible to test quite different magnetic divertor topologies (i.e. standard *X* point (SN), double null configurations, snow flakes (SF), expanded divertor configurations (XD), …) and different divertor materials (i.e. from tungsten to liquid metals). A total cost of 500 M€ has been assumed as a constraint for the facility design. Although the WPDTT2 project has tackled a wider set of aspects than presented in the Italian DTT proposal (for instance the experiments performed jointly with WPDTT1 on the Chinese EAST tokamak [4], and/or a deeper analysis on the possibility of liquid metals for the divertor), since the mentioned proposal has been realized within the WP and since it has involved more than fifty scientists from different European countries, in the rest of the paper we will essentially describe what is presented within this proposal, as a good synopsis of the full WPDTT2.

2. Rationale for the choice of DTT parameters

It is well recognized that to simulate the complete behavior of DEMO the only solution would be to realize DEMO itself [5, 6]. To overcome this very challenging issue, several different approaches have been proposed [5, 7, 8, 9], either considering the divertor and the SOL as regions completely independent of the bulk plasma, or focusing the interest also on the core. Since any DTT experiment is finalized to study the power exhaust, the first parameters to preserve are those connected with the divertor and the SOL regions. A key parameter characterizing these two regions is P_{SEP}/R, whose values should be around 15 MW m^{-1} to be DEMO relevant (where P_{SEP} is the power flowing through the plasma boundary). This figure is constrained by using an actively cooled tungsten monoblock technology [10]. Two other important parameters are the upstream poloidal (q_θ) and parallel ($q_{//}$) power fluxes: $q_\theta = P_{SEP}/\lambda_q 2\pi R$, where $\lambda_q \sim B_\theta^{-1}$ is the decay length of the mid-plane heat channel and the inverse poloidal field dependence comes from the Eich scaling [11]. Since the parallel heat transport is dominant, it follows that $q_{//} \sim q_\theta B/B_\theta \sim P_{SEP}B/R$ (>110 MWT m^{-1} for DEMO). Previous works [6, 8] have shown that, even considering the edge plasma as an insulated region, a complete 'self-similarity scaled down experiment' cannot be realized, but that it could be approximated [7, 8] by fitting five dimensionless parameters: T_e (with a suitable normalization), $\upsilon^* = L_d/\lambda_{ei.}$, Δ_d/λ_0, ρ_i/Δ_d, β, where L_d is the divertor field line length, λ_{ei} is the electron-ion collision mean free path, Δ_d is the SOL thickness, λ_0 is the neutrals mean free path, ρ_i is the ion Larmor radius, β is the plasma pressure normalized to the magnetic one. Some of these parameters are intrinsically linked with the divertor 'magnetic topology' and/or with the actual divertor geometry [7], and

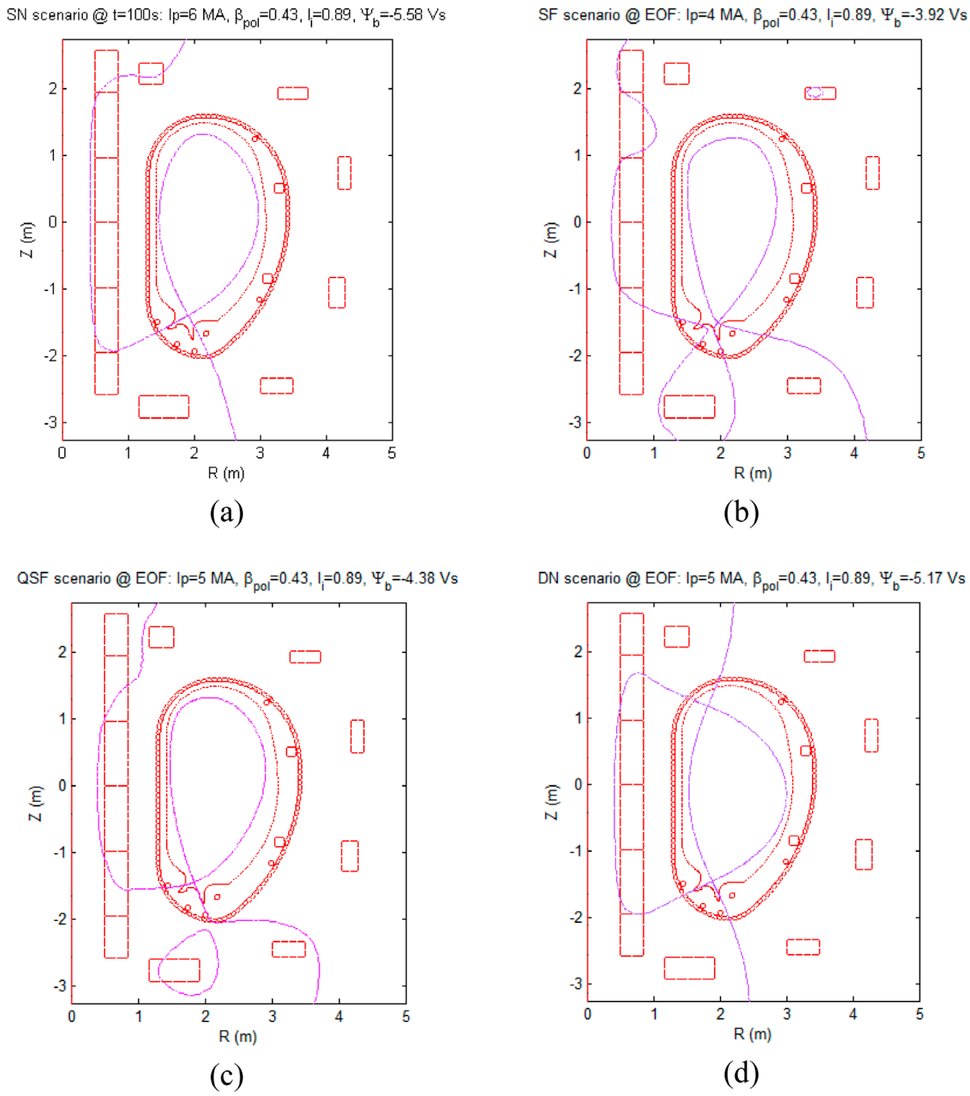


Figure 1. Conventional and alternative magnetic configurations that can be obtained using the DTT PF system: (*a*) conventional single null (SN); (*b*) snow flake (SF); quasi snow flake (QSF $-$ SF$^+$); (*d*) double null (DN).

this fact immediately poses a first strong constraint for the DTT design: the necessity of having a very flexible divertor ‘region/configuration’ to study and optimize the role played by the various topologically linked parameters.

Eventually, the machine dimension and the plasma bulk performances should guarantee an exhaust solution extrapolating to a reactor-graded plasma. It is well known that the plasma physics properties (bulk and edge) are completely determined by the dimensionless parameters υ^* (normalized collisionality), ρ^* (normalized Larmor radius), β and T [6, 12]. However, it is not possible to simultaneously preserve all these quantities. A strategy has therefore been proposed, which consists of relaxing one of these parameters in a controlled manner, [13] so as to down-scale the main physics properties of a reactor-like experiment (i.e. ITER, DEMO) on a smaller experimental device, while preserving all the main physics aspects. Since $\rho^* \sim T^{0.5}/BR$, it is practically impossible to exactly preserve this parameter without using machine and plasma parameters requiring magnetic fields that are not technologically achievable ($\rho^* = \text{Cost} \rightarrow B \sim 1/R$). Consequently, ρ^* is the dimensionless parameter chosen to be relaxed in a controlled way ($\rho^*_{\rm R} = \rho^*_{\rm S} R^{\varepsilon}$, the subscripts R and S indicate respectively the ‘reactor’ and the ‘scaled’ device, ε is the ‘controlling’ scaling parameter).

When fixing the machine dimension, on top of the technical and physical criteria already discussed, we must introduce another important constraint, i.e. the cost containment. The cost of a Tokamak (without using Tritium and not including the additional power) scales as the machine magnetic volume, Cost $\sim B^2R^3 \sim R^{2.75}$, when relaxing in the opportune way ρ^* ($\varepsilon = 0.75$). As mentioned, the cost of the additional heating is not included in this scaling; in order to consider it, we can assume the maximum cost ratio between the whole machine and the total additional power to be 30%, i.e., Heating_cost/Total_cost $\leqslant 0.3$ (e.g. Machine_cost $\approx$ 500 M€ $\rightarrow$ Heating_cost $\approx$ 150 M€, where 500 M€ is the total cost foreseen for the Italian proposed DTT, see the Introduction). By fixing an opportune ε value a rough estimation of the machine cost (not

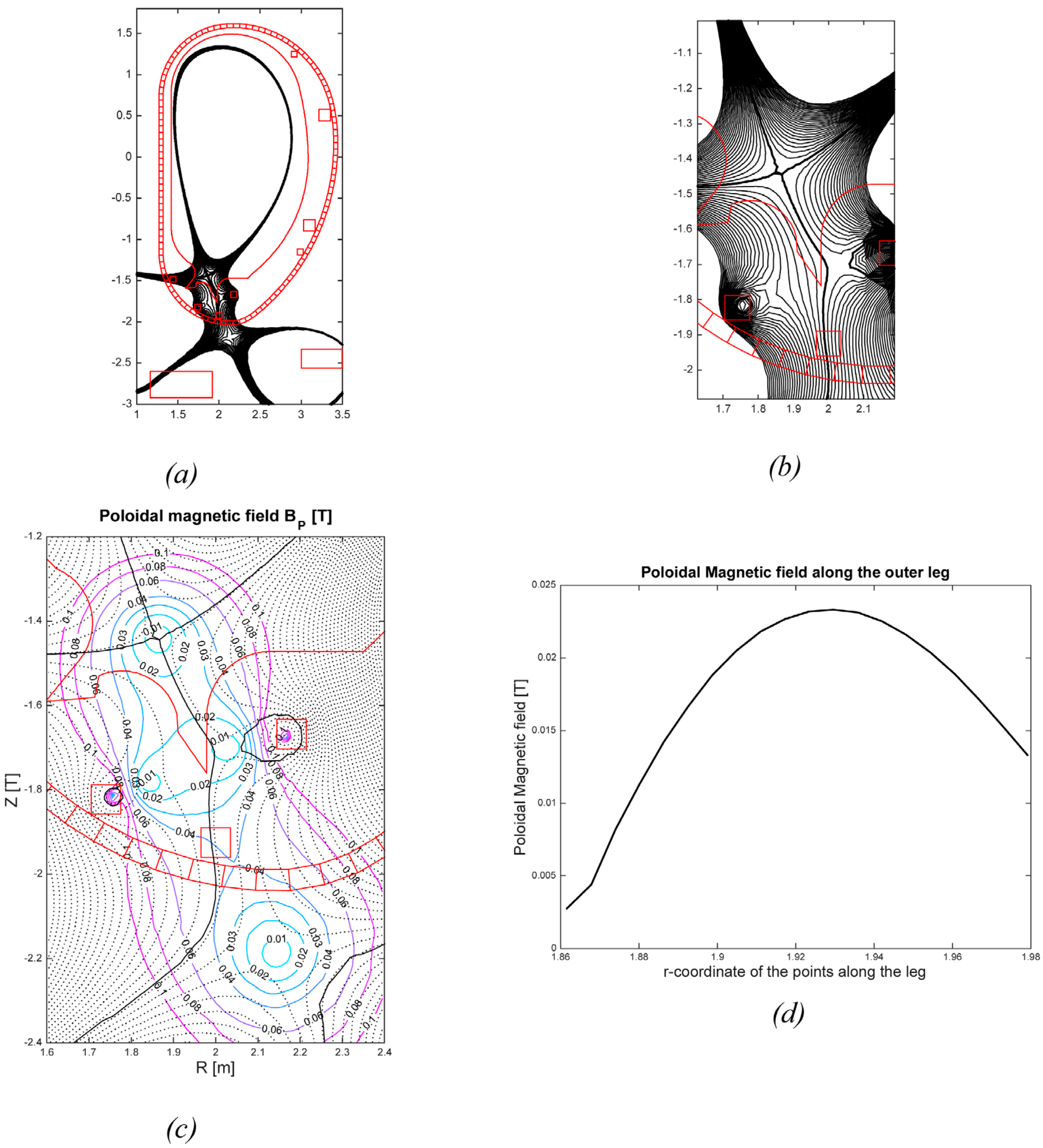


Figure 2. Use of lower internal coils C1–C4 for the modification of the SF^+ configuration into a XD configuration: (*a*) XD equilibrium generated from the reference SF^+ with the use of the internal coil currents (C1: −2.6 kA, C2: −43.0 kA, C3: −0.5 kA, C4: 57.3 kA); (*b*) XD equilibrium in the divertor region; (*c*) poloidal magnetic field of the XD configuration in the divertor region; (*d*) Poloidal magnetic field along the outer leg as a function of the *r*-coordinate of the leg points from the *X*-point to the target.

including the heating) can be evaluated, by using the mentioned scaling, versus the machine major radius. Eventually all these considerations indicate that the maximum machine radius cannot exceed 2.3 m: $R_{\mathrm{Max}} \leqslant 2.3$ m [3].

The previous reasoning indicates an upper bound for the major radius, but it gives no indication about a minimum size. The definition of this minimum radius will be a compromise among several different factors. Here we will only quote three points, which give some strong indication about the quantification of a minimal machine dimension.

(1) The main machine target (i.e. to study quite different divertor magnetic topologies) makes it necessary to introduce a small set of internal coils, to modify the reciprocal position of the main *X* point and of a secondary magnetic field null. Considerations about the necessary magnetic field produced by these coils lead to $R_{\mathrm{Min}} > 1.5$ m [3].
(2) Reducing the plasma size too much, keeping the term $P_{\mathrm{SEP}}/R > 15$ MW m^{-1} fixed, leads to more power flowing towards the first wall than the safe figure of about 1 MW m^{-2} for the power flux on a tungsten FW. Again this type of evaluation leads to $R_{\mathrm{Min}} > 1.5$ m [3].
(3) The third and last example regards the discharge duration time (τ_{S}). An accurate discussion on this parameter involves several important points about different technologies and approaches to be used in the machine design (for instance the use superconductors or standard copper coils). The first obvious assumption is that the discharge must last at least 3–4 times the diffusion resistive time

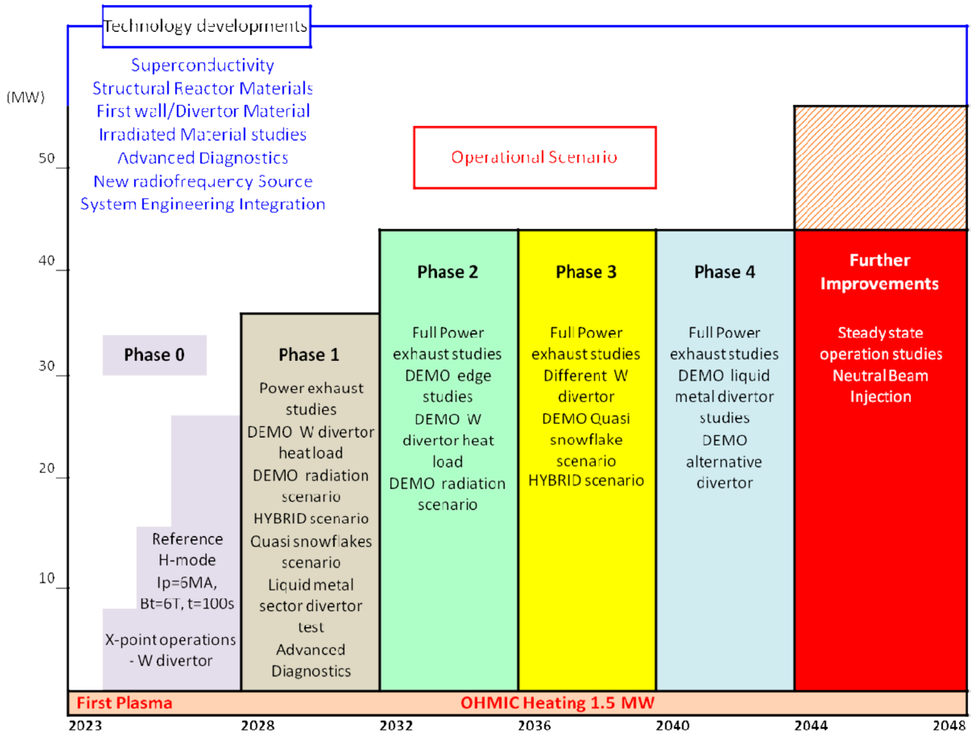


Figure 3. Schematic planning of the DTT operations.

(i.e. the longest Physics characteristic time); in the proposed DTT (see table 1) $\tau_R \approx 6$ s, leading to a minimum duration time of the order of 20 s. But it is quite obvious that, the DTT being dedicated to the integrated studies of the physics and of the materials technology, this physics longest time must be only considered as the 'zero' time to study the thermalization time of the materials. Consequently, a plasma current plateau time should be at least a factor of two longer than τ_R; when including the plasma build up and termination in the plasma duration τ_S, a reasonable $\tau_S \approx 100$ s must be assumed. Regardless of the coils technology this leads to a minimum (when fixed the plasma current) $R_{Min} > 1.7$ m [3].

From these three arguments we could roughly estimate a minimum major radius of $R_{Min} = 1.7$–1.8 m.

Eventually, the integration of all the aspects just discussed in the design of the DTT tokamak facility leads to a proposal with major radius $R = 2.15$ m, minor radius $a = 0.7$ m, plasma current $I_p = 6$ MA, toroidal field $B_T = 6$ T and additional power $P_{ADD} = 45$ MW (see table 1).

3. DTT operational programme

The DTT being a facility mainly dedicated to testing innovative ideas to solve the power exhaust problem, a strong effort has been dedicated to verifying the possibility of realizing the largest possible set of 'alternative' magnetic divertor topologies, as shown in figure 1.

For a fair comparison all the equilibria shown have been studied at the same β_p and the same l_i, but the plasma current is not the same for the four cases. Figure 1(*a*) shows a standard *X* point with the machine target plasma current $I_p = 6$ MA; figure 1(*b*) illustrates an SF equilibrium with $I_p = 4$ MA, the lower current being constrained by the poloidal coils maximum density current and for a discharge duration of 100 s. When

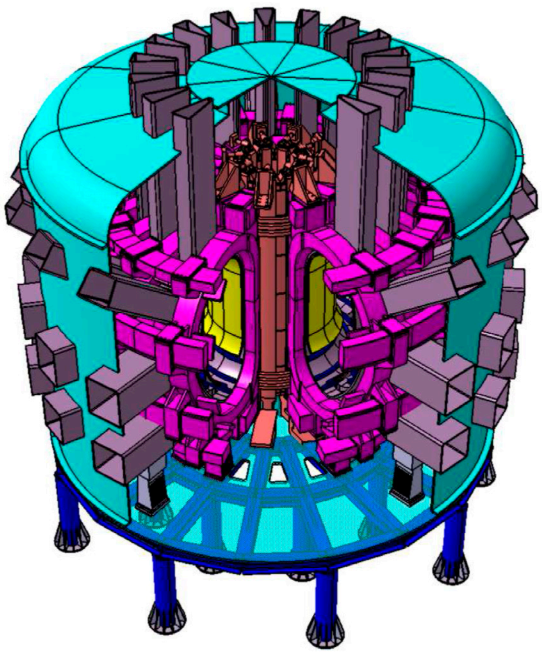


Figure 4. View of the DTT machine: the basic toroidal machine is entirely contained inside a cryostat vessel, which provides the vacuum for the superconducting magnets.

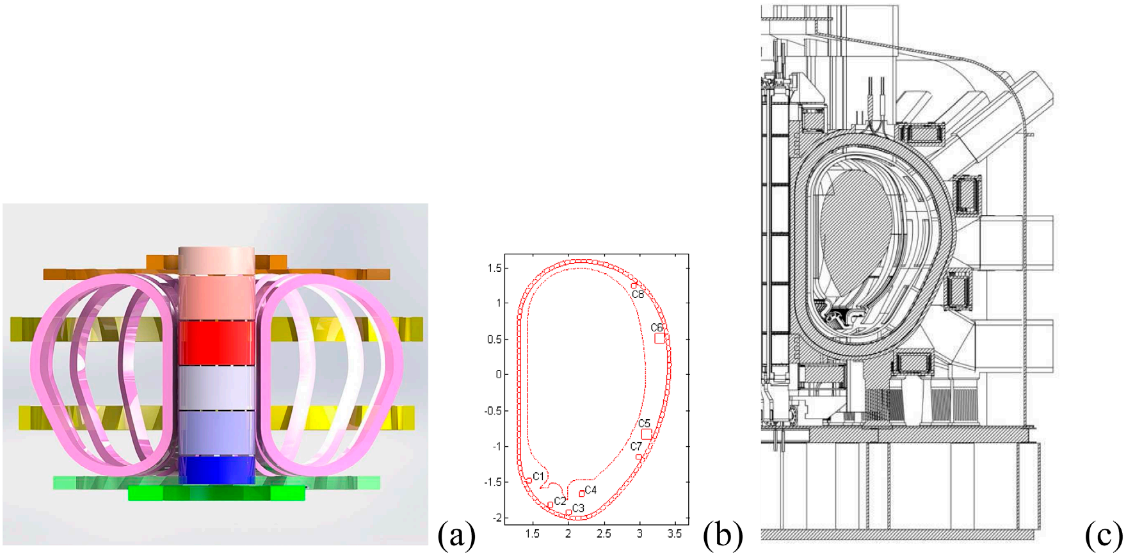


Figure 5. The magnet system of DTT: (*a*) artistic view, with the complete TF, CS, and PF coils system; (*b*) location of in-vessel copper coils; (*c*) schematic view of superconducting magnets and main structures.

relaxing this parameter, the configuration can again be realized with $I_p \approx 6$ MA. Figure 4(*c*) shows a quasi SF configuration (QSF) [4], whereas figure 4(*d*) illustrates a double null equilibrium. In both cases (figures 4(*c*) and (*d*)) the plasma current is $I_p = 5$ MA, but we can make the same considerations as the ideal SF. The double null configuration is not exactly up–down symmetric, because this would involve a larger machine volume, with consequent strong increase of the total machine cost.

The presence of a set of small internal coils around the divertor will allow local modification of the magnetic topology, when a second null has already been realized by the external poloidal coils, without affecting the rest of the plasma boundary. This will allow performing detailed studies about the role of the divertor magnetic topology, in reducing the power flow on the divertor plates, either affecting the local energy transport properties and/or the local radiation. An example of such a possibility is shown in figure 2.

The DTT facility is planned to operate on a very long time period, accompanying the ITER experiment and operating at least until the beginning of the DEMO realization. In figure 3 we show a possible indicative and schematic planning of the DTT operations, where possible shut down periods are included in different phases. In the first 4–5 years DTT will operate in standard divertor configuration with the target to get good plasma performance at high power and with physics parameters close to DEMO. This activity will permit the design of a new divertor 'dedicated and optimized' for an alternative divertor magnetic configuration. The following 7–8 years will be dedicated to studying the new divertor scenarios in combination with the highest planned additional power and with synergies with high radiative plasmas. Mainly during these phases important technological targets could be achieved, including some possible patents on new materials. In the following phase it will be possible to design and realize a liquid metal divertor; consequently, this type of solution will be tested with the 'optimal' divertor geometry obtained during the previous years. The last part of the experimental activity will be dedicated to the achievement of steady state configurations. During this period several technological innovations will be implemented and tested; as an example, some poloidal low temperature superconductor coils could be replaced with new generation high temperature superconductor coils. This would allow European industries to remain on the front line on a huge series of old and new market sectors outside fusion (e.g. medical applications, electric power transmission).

DTT will be equipped with a set of external poloidal coils able to guarantee a large set of different divertor magnetic configurations. The presence of a set of small internal coils will allow locally modifying the magnetic configuration, so as to produce a very large set of quite different topologies. The large space allocated at the bottom of the machine will easily allow the installation of a divertor realized using liquid metal technology [14]. A mix of different heating systems will provide the required power (a possible power allocation could be $\approx$15 MW electron cyclotron resonance heating (ECRH) at 170 GHz; $\approx$15 MW ion cyclotron resonance heating (ICRH) at 60–90 MHz; $\approx$15 MW neutral beam heating (NBI) at 300 keV).

4. DTT technical features

A schematic view of the DTT is shown in figure 4. A complete technical description of the DTT proposal is reported in [3]. Here we illustrate the most important features.

4.1. Plasma scenario requirements

All the plasma configurations (including standard single null and advanced configurations, see figure 1) satisfy the following constraints:

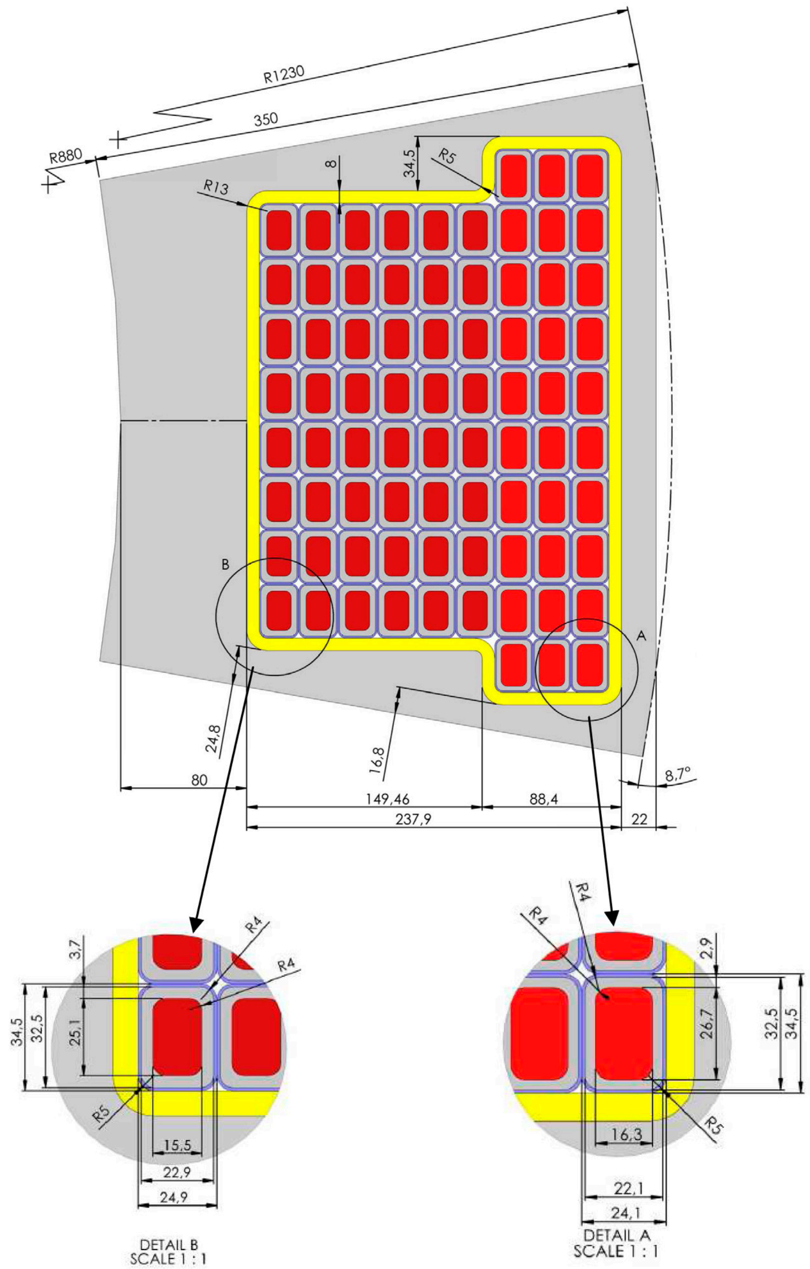


Figure 6. The DTT TF winding pack and details of high (right) and low field side conductors.

- distance of at least 40 mm between the plasma last closed surface and the first wall, in order to minimize the interaction between the plasma and the vacuum chamber; as a matter of fact, the power decay length at 6 MA is about 2 mm at the outboard midplane;
- plasma shape parameters similar to those of the present design of DEMO: $R/a \approx 3.1$, $k \approx 1.76$, $\langle \delta \rangle \approx 0.35$ [15];
- pulse lasting more than 100 s (total available flux about 45 Vs, CS swing about 35 Vs).

4.2. Magnet system

The requirements of the plasma scenarios suggest the use of superconducting windings. The DTT magnet system design (figure 5) is based on Cable-In-Conduit Conductors (CICCs) made of low temperature superconducting (LTS) materials, such as Nb3Sn (for the toroidal magnet and the central solenoid) and NbTi (for the external poloidal coils), copper and stainless steel.

The toroidal magnet consists of 18 D-shaped coils wound by 78 turns of Nb3Sn/Cu CICC, carrying 46.3 kA of operative current cooled by a forced flow of supercritical helium, having an inlet temperature of 4.5 K, with a maximum field of 11.4 T and a conductor cable current density of about 120 MA m^{-2}. Details of the winding packs are shown in figure 6.

The poloidal system includes a central solenoid divided into six independent modules, plus six external coils (see figure 4 and table 2). The peak field on the central solenoid is 12.5 T for a stored flux of $\pm$17.6 Vs. For the six PF coils the peak field is 4.0 T.

Table 2. PF coil system shown in figure 5 and currents needed for the magnetic configurations depicted in figure 1 (end of flat top configurations with a poloidal beta of 0.43 and an internal inductance of 0.89).

Name	R (m)	Z (m)	ΔR (m)	ΔZ (m)	N	NI (MAT) 6 MA SN	NI (MAT) 4 MA SF	NI (MAT) 5 MA QSF	NI (MAT) 5 MA DN
CS3U	0.6685	2.2615	0.343	0.635	270	−0.76	−5.69	4.46	2.48
CS2U	0.6685	1.458	0.343	0.972	420	−9.18	3.54	−1.94	−5.95
CS1U	0.6685	0.486	0.343	0.972	420	−9.18	−8.74	−9.18	−9.18
CS1L	0.6685	−0.486	0.343	0.972	420	−9.18	−9.18	−9.18	−9.18
CS2L	0.6685	−1.458	0.343	0.972	420	−9.18	0.04	−3.21	−9.18
CS3L	0.6685	−2.2615	0.343	0.635	270	1.31	0.98	3.91	0.87
PF1	1.34	2.23	0.377	0.32	130	0.27	−3.04	−2.80	2.81
PF2	3.49	1.931	0.468	0.174	108	−0.71	0.86	−0.55	−2.62
PF3	4.28	0.745	0.192	0.49	112	−2.05	−2.27	−1.37	−0.97
PF4	4.15	−1.049	0.245	0.469	140	−2.37	−2.78	−3.45	−1.14
PF5	3.25	−2.45	0.494	0.228	152	−1.51	2.38	1.75	−3.17
PF6	1.541	−2.76	0.754	0.32	260	3.99	−2.32	−1.62	4.89
C1	1.44	−1.481	0.07	0.07	1	—	—	—	—
C2	1.74	−1.823	0.07	0.07	1	—	—	—	—
C3	2	−1.925	0.07	0.07	1	—	—	—	—
C4	2.18	−1.668	0.07	0.07	1	—	—	—	—
C5	3.1	−0.83	0.14	0.14	4	—	—	—	—
C6	3.285	0.51	0.14	0.14	4	—	—	—	—
C7	2.988	−1.15	0.07	0.07	1	—	—	—	—
C8	2.915	1.25	0.07	0.07	1	—	—	—	—

The PF system also includes eight copper in-vessel coils, specifically two in-vessel coils for radial and vertical stabilization and control, and four out of six in-vessel coils for magnetic control of SOL and strike point sweeping. The concept of a hollow DTT copper coil with a diameter of 45 mm is aligned with that of the ITER VS in-vessel coils [16], with internal cooling by water (flowing in a 30 mm diameter tube), surrounded by a 5 mm thick layer of compacted MgO powder, and a 2 mm thick SS 316L jacket. A thermal analysis considering a total heat load of 1.2 kW—due to the Ohmic power in the copper and the first wall temperature up to about 400 °C—shows that a water mass flow rate of 7.31 kg s^{-1}, corresponding to a 10.3 m s^{-1} water velocity, is able to keep the copper temperature well below 150 °C.

4.3. Vacuum vessel and first wall

The design of the vacuum vessel (VV) includes a shell of INCONEL 625 (figures 7(*a*) and (*b*)). The 18 sectors are joined by welding. The maximum thickness of the shell is 35 mm, while the five ports per sector are 25 mm thick. Its *L*/*R* time constant is about 40 ms.

These features of the vacuum vessel keep the parameters of the plasma vertical instability within a range that can be controlled using the internal coils C5 and C6 with a maximum current of 25 kA (growth rate of 20 ÷ 70 s^{-1} with a stability margin of 0.4 ÷ 0.8) in case of 1.2 MJ ELMs or VDEs detected after more than 40 mm displacements.

Analyses of TF coil discharges and plasma disruptions show that the maximum von Mises stress is lower than the INCONEL 625 admissible stress limit.

The first wall (figures 7(*a*) and (*c*)) consists of a bundle of tubes armored with plasma-sprayed tungsten. The plasma facing tungsten is about 5 mm, the bundle of finned tubes in stainless steel (coaxial pipes for cooling operation) is 30 mm thick, and the SS316LN backplate supporting the tubes is 30 mm thick.

Since a non-negligible neutron flux is expected (about $9 \cdot 10^{-11}$ n cm^{-2} s^{-1} @ inboard midplane with a cumulative fluence exceeding the threshold of $4 \cdot 10^{12}$ n cm^{-2} after a few pulses at high power), a remote handling system (figure 7(*d*)) will be used for the maintenance of the in-vessel structures and a thermal shield will be placed so as to reduce the load on the TF coil system.

4.4. Divertor

The main goal of the DTT project is to build a facility for testing several divertor concepts and configurations. Therefore the design of the VV, the ports and the additional heating system also takes into account the constraints related to the testing of liquid metal divertor targets.

The ‘first day’ design includes a tungsten divertor, realized with W-shaped modules, distributed along the VV; the design is fully compatible with advanced magnetic configurations (figure 8(*a*)). Furthermore, the design of VV, ports and RH devices shall be compatible with the application and testing of a liquid metal divertor (figure 8(*b*)).

4.5. Additional heating and other subsystems

For the first DTT phase, the additional heating system will provide 15 MW with ICRH and 10 MW with ECRH. NBI

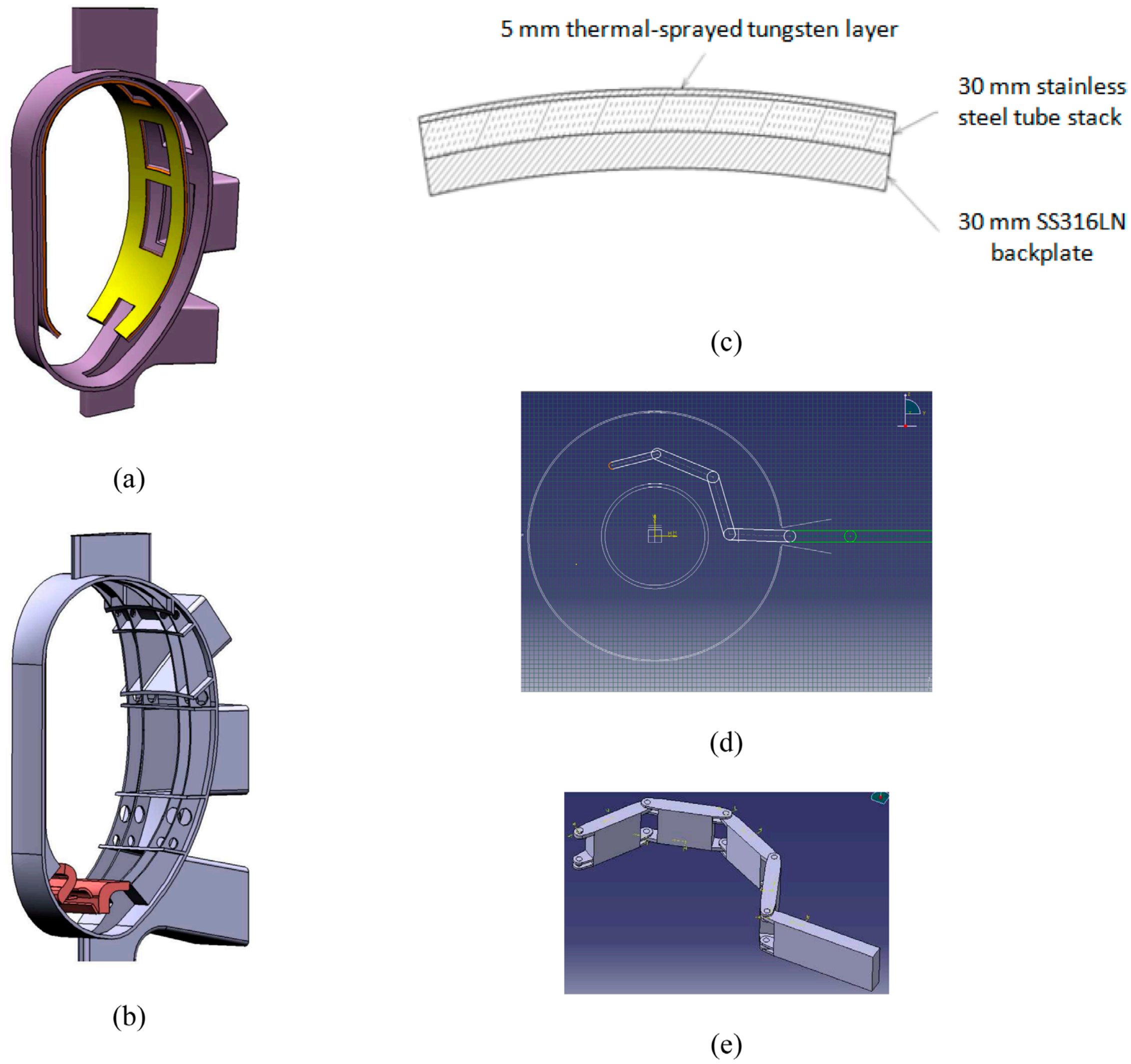


Figure 7. DTT vacuum vessel (VV) and first wall (FW): (*a*) 3D view showing the 5 access ports per sector; (*b*) 3D view of the FW support structure; (*c*) details of the FW layers; (*d*) schematic representation of the remote handling system for the first wall; (*e*) detail of the 'snake' part of the remote handling system.

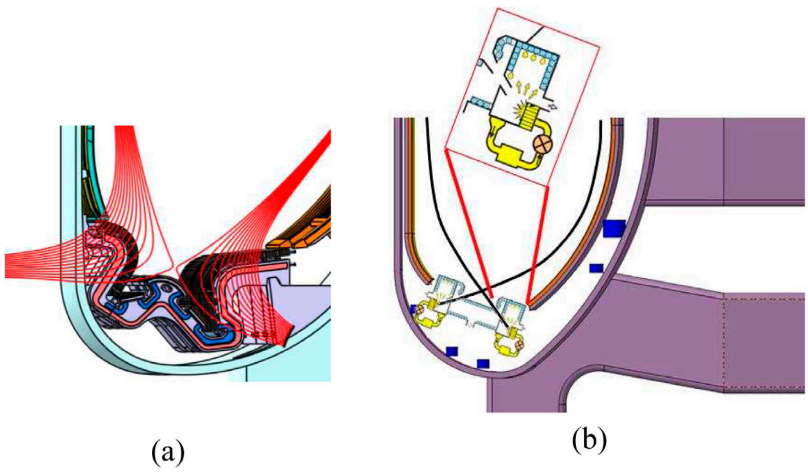
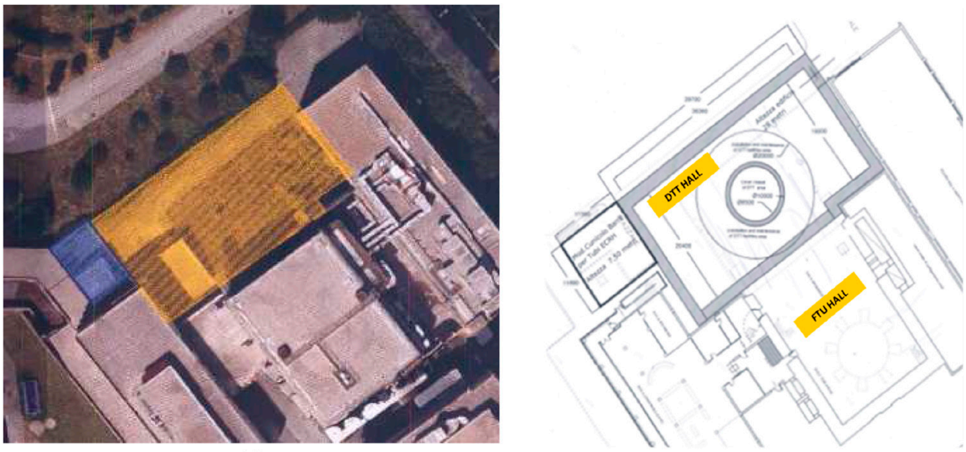


Figure 8. DTT divertor: (*a*) a possible tungsten divertor, compatible with both the SN and SF configurations; (*b*) liquid lithium box divertor.

is being considered as the main candidate for a subsequent power upgrade. During experimental use the total amount of heating will be upgraded to 45 MW; the final sharing will be decided on the basis of the experience gained—however NBI is being considered as the main candidate for a subsequent power upgrade up to 15 MW.

Table 3. Main components of the real time control system of DTT.

	Diagnostics	Actuators
Plasma current	Rogowsky coils	Magnetic flux
Axisymmetric equilibrium	Magnetic sensors	PF coils
Electron density	Interferometer	Gas valves/Cryopumps
MHD/NTM	Pick-up coils/ECE / SXR	ECE/Control coils
ELM control	Da, stored energy	Control coils, plasma shape control, vertical kicks, pellets, RMP's
Power exhaust	IR cameras/thermocouples/ CCD cameras/spectroscopy	Divertor and main plasma gas valves/ impurity gas valves/in-vessel coils



(a) (b)

Figure 9. Proposed DTT site in Frascati: (*a*) aerial view on of the present FTU buildings, with the necessary upgrades for DTT highlighted in yellow; (*b*) sketch of the new hall and the present FTU hall.

The total electric power demand for magnets, additional heating and auxiliary systems is about 180 MW (active power). Most power supplies for the magnet system have output DC current ±25 kA and output DC voltage ±800 V (except PF3, PF4, IC5 and IC6 PSs that have an output DC voltage ±1 kV). These AC/DC converters are four quadrants, thyristor based 12 pulses with current circulating and sequential control to reduce the reactive power, except IC5 and IC6 PSs that are IGCT based to be fast enough to control the vertical position of plasma.

Particular attention will be dedicated to the diagnostics and control issues, especially those relevant to plasma control in the divertor region, designed to be as compatible as possible with a DEMO-like environment (table 3).

4.6. Site and licensing

Taking into account the role of DTT as 'European facility', the proposed site is Frascati. In this regard, this proposal primarily considered the accessibility of the site and its attractiveness for the interested people from several European and international countries (researchers, scientists, engineers) that will contribute to the project/construction and operational activity, providing an important beneficial impact also on the scientific and technical performance.

The ENEA Research Center has the possibility to realize the DTT facility, given its capability to meet the various technical requirements. The FTU Tokamak is still in operation inside the ENEA Frascati site. The presence of such a facility would make the authorization and licensing procedures for the new machine much easier. Moreover, the FTU buildings can host DTT with some modifications, already discussed with local authorities. The required upgrade of the grid requires the extension of the 150 kV line.

Figure 9(*a*) shows an aerial view of the present FTU buildings highlighting the modifications planned to install the DTT tokamak. The other buildings are now part of the FTU infrastructure and will be re-used for DTT with some minor internal modifications. Figure 9(*b*) shows the location of the DTT in the new hall. The machine would be preassembled in a modular way inside the present FTU hall, which, on a longer time scale, should host the NBI injector.

4.7. Costs

The costing of the facility is based on direct experience gained during the construction of several devices/systems and an industry survey. The investment costs sum up to 499 M€, including:

- hardware and infrastructures (444 M€ for magnets, vacuum vessel, cryostat, in-vessel components, layout, auxiliary heating systems, principal diagnostic systems, controls and data acquisition system, cooling system, power supplies, remote handling, buildings, assembly);

- personnel cost (30 M€ for the engineering design and construction phase, corresponding to 300 ppy, taking into account the additional support of research institutions and industries);
- contingency (25 M€, in excess of 5% of the investment cost).

The replacement of an additional divertor (possibly optimized in accordance with the experimental evidence of different needs coming from alternative magnetic configurations) has been included in the 'operating cost'. The cost has been estimated to be similar to the cost of the initial divertor (30 M€). If we assume the same dimensions and shape as the tungsten divertor, then in the liquid divertor the tungsten tiles will be replaced by a continuous CPS structure based on a mesh of tungsten wires with a thickness of only 1 mm. So, in terms of raw materials, the cost of a module based on a liquid metal target will be less than the W-shaped tungsten divertor, whereas the structure is expected to be more complex. At the current stage, according to JET and ITER experience, the replacement of a full divertor using remote handling (if necessary with double 8 h shifts per day) should not exceed 6 months, whereas the replacement of a single faulty cassette should not exceed 2 months.

5. Conclusions

This proposal for a DTT facility demonstrates the possibility to set up a facility able to bridge the technological gap between present day devices and ITER/DEMO. The DTT scientific project is well framed within the European fusion development roadmap [1], which plays a crucial role for the development of one of the most promising technologies for an alternative, safe and sustainable new energy source.

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This paper is largely based on the Italian proposal for DTT prepared with contributions of European and international experts [3], on the activity carried out inside the EUROfusion work package WPDTT2 [17], and a presentation given at the 2nd IAEA DEMO Programme Workshop [18].

This proposal is synergic with the activities carried out within the EUROfusion work packages:

- WPDTT1—Assessment of alternative divertor geometries and liquid metals PFCs;
- WPDTT2—Definition and Design of the Divertor Tokamak Test Facility.

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