The DTT proposal. A tokamak facility to address exhaust challenges for DEMO: Introduction and executive summary

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HIGHLIGHTS

- The DTT (Divertor Test Tokamak facility) proposal demonstrates the possibility to bridge the technological gap between the present day devices and ITER/DEMO in the area of plasma exhaust.
- DTT retains the possibility to test different divertor magnetic configurations, liquid metal divertor targets, and other possible solutions for the power exhaust problem.
- The edge conditions are similar to DEMO in terms of dimensionless parameters in integrated scenarios, compatible with plasma performance and technological constraints of DEMO.

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ABSTRACT

As indicated in the European Fusion Roadmap, the main objective of the Divertor Tokamak Test facility (DTT) is to explore alternative power exhaust solutions for DEMO so as to mitigate the risk that the conventional divertor based on detached conditions to be tested on the ITER device cannot be extrapolated to a fusion reactor. The issues to be investigated by DTT include:

- demonstrate a heat exhaust system capable of withstanding the large load of DEMO in case of inadequate radiated power fraction;
- close the gaps in the exhaust area that cannot be addressed by present devices;
- demonstrate that the possible (alternative or complementary) solutions (e.g., advanced divertor configurations or liquid metals) can be integrated in a DEMO device.

In this paper, we describe a proposal for such a DTT, presented by ENEA in collaboration with a European team of scientists. The selection of the DTT parameters (a major radius of 2.15 m, an aspect ratio of about 3, an elongation of 1.6-1.8, a toroidal field of 6 T, and a flat top of about 100 s) has been made according to the following specifications:

- edge conditions as close as possible to DEMO in terms of dimensionless parameters;
- flexibility to test a wide set of divertor concepts and techniques;
- compatibility with bulk plasma performance.
- an upper bound of 500 M€ for the investment costs.

This paper illustrates this DTT proposal showing how the basic machine parameters and concept have been selected so as to make a significant step toward the accomplishment of the power exhaust mission.

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1. Introduction

One of the main challenges, within the European Fusion Roadmap [1], is to design a power and particle exhaust system, capable to withstand the large loads expected in the divertor of a DEMO fusion power plant [2,3]. In ITER [4] (the International Fusion experiment under construction in Cadarache) it is planned to test the actual possibilities of a standard divertor working in "detached conditions". However, it is already clear that this solution is very challenging and that, consequently, the power exhaust problem could be a potential "show stopper" of the Fusion Road towards the realization of a Fusion Reactor [5].

For this reason a specific project has been launched, within the European Fusion Roadmap, to investigate alternative power exhaust solutions for DEMO, aiming at the definition and the design of a Divertor Tokamak Test facility (DTT). This tokamak should carry out scaled experiments integrating various aspects of the DEMO power and particle exhaust. DTT should retain the possibility of testing different divertor magnetic configurations, liquid metal divertor targets, and other possible solutions for the power exhaust problem. Hereby, the DTT design proposal presented in [6] refers to a set of parameters selected to reproduce edge conditions as close as possible to DEMO in terms of a set of dimensionless parameters characterizing the physics of Scrape Off Layer (SOL) and of the divertor region, while remaining compatible with DEMO as regards the dimensionless parameters dictating the bulk plasma performance. The parameters of the machine have been obtained consistent with a set of constraints related to the largest possible machine flexibility at a given cost of 500 M€. While the European Programme allocated about 60 MEUR in Horizon 2020, the Italian Government has offered to the European fusion scientific community the opportunity to get complementary funding for a dedicated facility located in Italy. In the international context the power exhaust problem is tackled within different new proposals and/or recently upgraded and ongoing experiments. In United States a smaller, but high toroidal field machine (ADX [7]) has been recently proposed by MIT. In China the superconductive EAST [8] machine has been recently upgraded and already deals with alternative divertor magnetic configurations, although with physics parameters far from reactor conditions. The here presented DTT will tackle the integrated bulk/edge power exhaust problem with physics parameters and power load scenarios as close as possible to the DEMO situation in a scaled experiment. Consequently the final successful target (although not the only one) of the DTT facility will be to propose a "reliable and robust" power exhaust scenario for DEMO; where here "reliable and robust" stands for a power exhaust solution that is fully compatible with the available technology (materials, pumps, coils,.) and the plasma performance (no degradation of the confinement)

2. Power exhaust issues

The confinement in a tokamak is the result of magnetic field configuration forming a set of closed, nested magnetic surfaces that bound the plasma. At the edge a thin region of open field lines is created (the SOL) through which charged particles and heat flowing out of the core plasma are guided into the so-called divertor, where the plasma impinges on the divertor target plates (Fig. 1). The unmitigated heat flux, in the SOL region of ITER and DEMO, is expected to be even higher than on the sun's surface [5].

The current strategy, to be tested on the ITER device, foresees optimizing plasma operations with a conventional divertor based on detached plasma conditions. This strategy relies upon different factors:

- development of plasma facing components to cope with very large power fluxes (> 5 MW/m^2);
- selection of the divertor inclination and of the magnetic flux expansion to reduce the heat flux normal to 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 (5÷10 MW/m^2);
- recycling and increase in the density by lowering the temperature close to the target, with consequent detachment (the temperature drops below ionization's, therefore the particles are neutralized and there is neither direct plasma flux nor power to the divertor targets).

However, on the basis of the current level of knowledge there is a considerable degree of uncertainty about the extrapolation to ITER, DEMO and fusion power plants [5]:

- today's experiments operate with physics SOL 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.

In addition, the risk exists that the baseline strategy pursued in ITER cannot be extrapolated to a fusion power plant for the following reasons [5]

- stability of the detachment front needs to be assessed for ITER and DEMO and fusion power plant 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% when considering bulk, SOL and divertor region)
- compatibility with pumping
- monitoring of erosion, temperature, etc.
- . . .

Moreover, a number of aspects must be taken into account restricting the use of certain materials (i.e. requirements in terms of life expectancy of reactor components, the need of keeping the temperature low in the divertor region in order to take almost vanishing the erosion rate, etc. . .).

Therefore a specific 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. This tokamak should produce scaled experiments integrating most of the possible aspects of the DEMO power and particle exhaust.

3. DTT role and objectives

3.1. The role of DTT in the frame of european fusion research

The development of a reliable solution for the power and particle exhaust in a fusion reactor is recognized as one of the major challenges towards the realization of a fusion power plant [1,5].

There is the risk that the solution to adopt a conventional divertor (to be tested in ITER) could not be extrapolated to DEMO. In order to mitigate this risk, alternative solutions must be developed.

While several alternatives, such as the cooled liquid Li limiter in FTU [9], the Super-X divertor in MAST-U [10] are being investigated

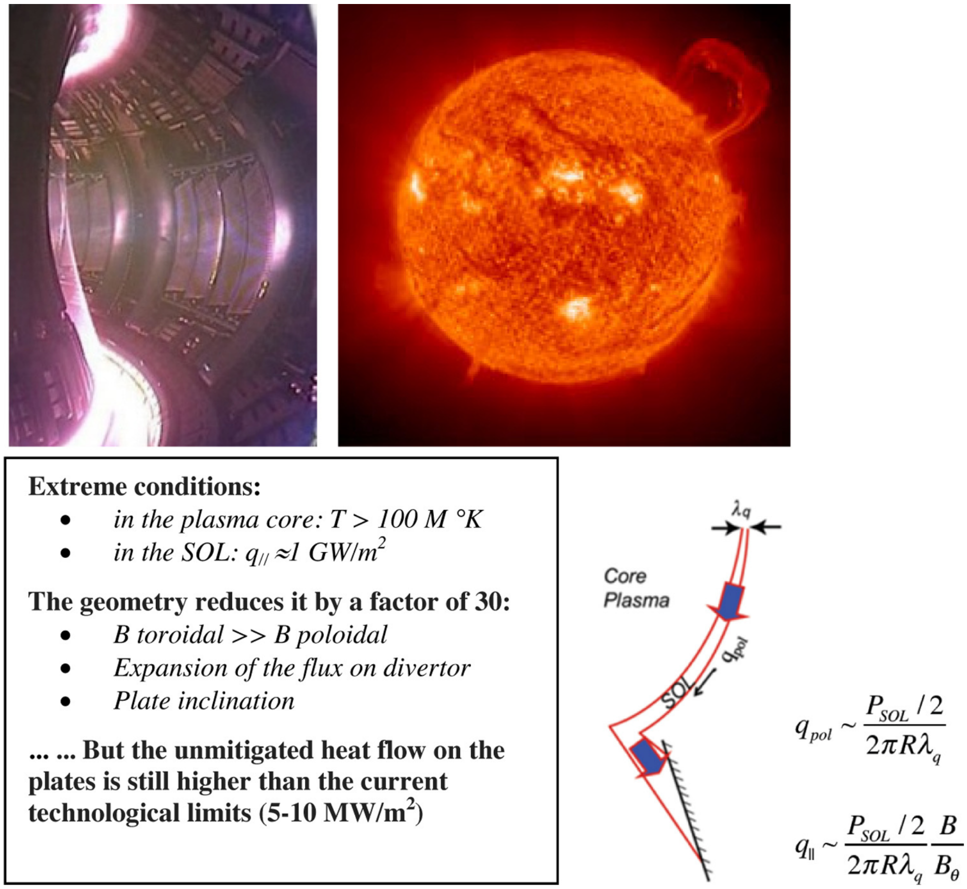


Fig. 1. Power flux on the divertor.

in presently operating tokamaks, the extrapolation from present devices to DEMO is considered not reliable [1].

The DTT project is part of the general European program in fusion research, which includes many other R&D issues (plasma experiments, modeling tools, technological developments for liquid divertors, etc. . .). The specific role of the DTT facility is to bridge the gap between today's proof-of-principle experiments and the DEMO reactor. DTT should, in particular, have the capability to bring such solutions to a sufficient level of maturity and integration from both physics and technology points of view [11].

3.2. *The main objectives of DTT*

The DTT facility will be able to test the physical and technological feasibility of various alternative divertor concepts that can confidently be extrapolated to DEMO. In this way it will possible to integrate the knowledge about the concepts of a number of divertor presently in testing operation on existing machines, with the implementation requirements of DEMO.

The main objectives of DTT, as reported in a number of official European documents [1,2,11], can be summarized as follows:

- demonstrate whether the heat exhaust system proposed for DEMO is able to withstand the strong thermal load acting if the fraction of radiated power turns out to be lower than expected;
- improve the experimental knowledge in the heat exhaust scientific area that cannot be addressed by present devices;
- demonstrate whether the possible (alternative or complementary) divertor solutions (e.g., advanced divertor configurations or liquid metals) can be integrated in a DEMO device.

In particular, DTT will host experimental tests addressed to assess whether:

- the alternative divertor magnetic configurations are viable in terms of power and particle exhaust as well as plasma bulk performances;
- the alternative divertor magnetic configurations are viable in terms of poloidal coils constraint (i.e., currents, forces, . . .);
- the various possible divertor concepts are compatible with the technological constraints of DEMO;
- the divertors based on the use of liquid metals are compatible with the characteristics of the edge of a thermonuclear plasma;
- liquid metals are applicable to DEMO.

4. Main parameters

Aim of DTT is to be a reduced size model, able to study the problems of the "Scrape Off Layer" (SOL) of DEMO. The reliability of the extrapolation to DEMO would increase with the DTT dimensions and physics parameters. However, a limiting factor in this design approach is related to the cost constraint. DTT should achieve its objectives with an available budget of 500M€ for the basic machine, i.e., the funds requested by Italian Government, plus the financial resources planned by EUROfusion in Horizon 2020 for WPDTT2 (Work Package "Definition and Design of the Divertor Tokamak Test Facility).

A machine with a plasma major radius of approximately 2.15 m is able to ensure a region of the divertor sufficiently broad to allow the testing of different magnetic configurations and various materials, including metals liquids. The relatively high toroidal field ($B_T = 6T$) will give the possibility to achieve plasma perfor-

Table 1
Main DTT parameters for a reference standard x point configuration.

R (m)	2.15	$T_e(0)$ (KeV)	10.2
a (m)	0,7	H_{98}	1
I_P (MA)	6	β_P	0.5
B_T (T)	6	β_T(%)	2.1
V (m^3)	33.0	β_N	1.5
k	1.7	τ_{Res} (s)	8
<δ>	0.3	V_{Loop} (V)	0.17
Pulse length (s)	100	Z_{eff}	1.7
$q_{(95\%)}$	3.1	P_{Rad} (MW)	13
P_{ADD} (MW)	45	P_{Sep} (MW)	32
$P_{L\text{-}H}$ (MW)	15–18	T_{Ped} (KeV)	3.1
$<n_e>$ (10^{20} m^{-3})	1.7	n_{Ped} (10^{20} m^{-3})	1.4
n_e/n_{eG}	0.45	P_{Div} (MW/m^2) (No Rad)	~100
$<T_e>$ (KeV)	6.2	P_{Sep}/R (MW/m)	15
τ (sec)	0.47	$P_{Tot}B/R$ (MW T/m)	125
$n_e(0)$ (10^{20} m^{-3})	2.2	λ_q (mm)	~2.0

mances not far from those in DEMO in plasma divertor condition with the figure $P_{SEP}/R \approx 15$ MW/m (where P_{SEP} is the power flowing through the plasma boundary and R the major radius) DEMO relevant [1,2,5,6].

In addition, a number of dimensionless parameters can be identified [12,13] including $T_e{}^*$, $\upsilon^* = L_d/\lambda_{ei}$, Δ_d/λ_0, ρ_i/Δ_d, β, where L_d is the divertor field line length, λ_{ei} is the electron-ion mean free path, Δ_d is the SOL thickness, λ_0 is the neutrals mean free path, ρ_i is the ion Larmor radius, $T_e{}^*$ is the electron temperature T_e normalized to a suitable reference value. It is well known that the only solution that preserves all the bulk and/or SOL dimensionless parameters (ρ^*, β, υ^*, $T_e{}^*$) yields a unit scale factor [12,13]. Therefore, to limit the size and the cost, the solution proposed in [5,13–15] is to relax in a controlled way one of these parameters (the normalized Larmor radius) while preserving the remaining physics aspects. Of course, as widely discussed and recognized [12], in any scaled down experiment something is lost as regards the desired target. It has been discussed and shown [16] that $\rho^* \propto (R\,B_T)^{-1}$ cannot be preserved in a dimension scaled down experiment because it would require a too high value for the magnetic field. Apparently, a different value of ρ^* might imply that also the width of the upper stream energy decay length (λ_E) is not preserved, but luckily the situation is a bit different. Preserving ρ^* would also imply a much larger value of the poloidal field B_p and, consequently, a quite smaller value of $\lambda_E \propto (B_p)^{-1}$. On the contrary an opportune relaxation of ρ^* it implies to preserve λ_E (see Table 1). The selected value of ρ^* ($3.7 \cdot 10^{-3}$) is in any case closer to expected ITER ($2.0 \cdot 10^{-3}$) and DEMO ($1.6 \cdot 10^{-3}$) values than any relevant device in operation (e.g., JET, for which ρ^* is about $5.0 \cdot 10^{-3}$ or, in principle, $4.0 \cdot 10^{-3}$ if working with a toroidal field of 3.5 T) [14]. It is even important to underline that, although preserving the non dimensional parameters is definitely the best choice for scaling the plasma bulk, this choice is not so ‘robust’ for the scaling the plasma edge. This is the main reason why the first focus of the facility will be to study and test different magnetic and technical solution for the divertor region. For this purpose the machine has been designed to be as flexible as possible in that region.

Table 1 reports the main DTT parameters for a reference standard single null scenario assuming a good H mode confinement ($H_{98} = 1$).To show the influence of the power load radiation we also quote the limit case assuming no radiation losses at all (P_{NoRad}). The discharge is lasting around 100 s, with the obvious consequence of the necessity of using superconductor coils. Within Ref. [6,15] the rational leading to a discharge lasting around 100 s is widely discussed from several points of view. The most important is strictly connected with the main DTT target to realize a DEMO relevant, integrated bulk and edge scenario. This leads to a plasma with a resistive time $\tau_{Res} \approx 8$ s (Table 1). To have a steady state plasma bulk configuration, we must at least to have a plateau of 25 ÷ 30 s

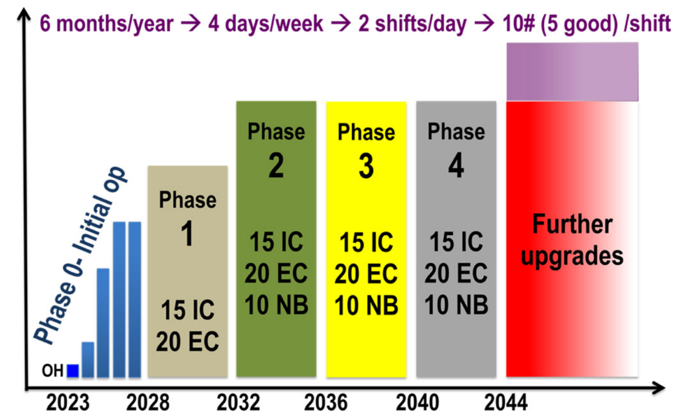


Fig. 2. Schematic planning of the DTT operations. The planned timing is shown on the top. The diagram shows a possible experimental program:.
Phase 1). Power exhaust studies, DEMO W divertor heat load, DEMO radiation scenario, HYBRID scenario, quasi snowflakes scenario, Liquid metal sector divertor test, Advanced Diagnostics.
Phase 2). Full Power exhaust studies, DEMO edge studies, DEMO W divertor heat load, DEMO radiation scenario
Phase 3). Full Power exhaust studies, Different W divertor, DEMO quasi snowflakes scenario, HYBRID scenario
Phase 4). Full Power exhaust studies, DEMO liquid metal divertor, DEMO alternative divertor
Phase 5). Further enhancements: i.e. extra additional power by using new RF sources, tet of high temperature superconductors,. . .

and that only at that point we can assume to be able to study the integrated conditions and the material properties. Considering the possibility of having comparative tests and adding up the plasma ramp-up and ramp down we arrive to a discharge time of the above length.

5. Operational program

Fig. 2 shows a schematic planning of the DTT operations [14,17]. The facility life can be schematically divided in 6 phases lasting more or less five years each. A zero phase will be aimed at the realization and installation of the various components of the machine. During the first phase the facility will be fully operative in H-mode aiming at the following targets: power exhaust studies, DEMO W divertor heat load, DEMO radiation scenario, hybrid scenario, snowflakes scenarios, liquid metal sector divertor test, advanced diagnostics. The next phases will be reserved to full power exhaust studies, DEMO edge studies, DEMO W divertor heat load, DEMO radiation scenario. The fourth phase will be dedicated to full power exhaust studies with DEMO liquid metal divertor and DEMO alternative configurations. Eventually, on a longer time scale, further technological enhancements could be tested, including additional power by using alternative RF sources [18] and the test of an upper poloidal coil by using high temperature superconductor.

This nominal program is only an indication of the DTT potentiality in terms of performances and timing. The experimental program will be finalized by an international scientific committee, taking in account the performance achieved in the initial phases of the operations and the needs expressed by the scientific community. If necessary and allocating the necessary funds, some tests (e.g., tests at full power, or test of a full liquid metal divertor) can be anticipated.

6. Plasma performance

The DTT machine is able to host various configurations with a plasma-wall clearance of at least 40 mm [3,6]. The plasma shape parameters for reference single null configuration are similar to

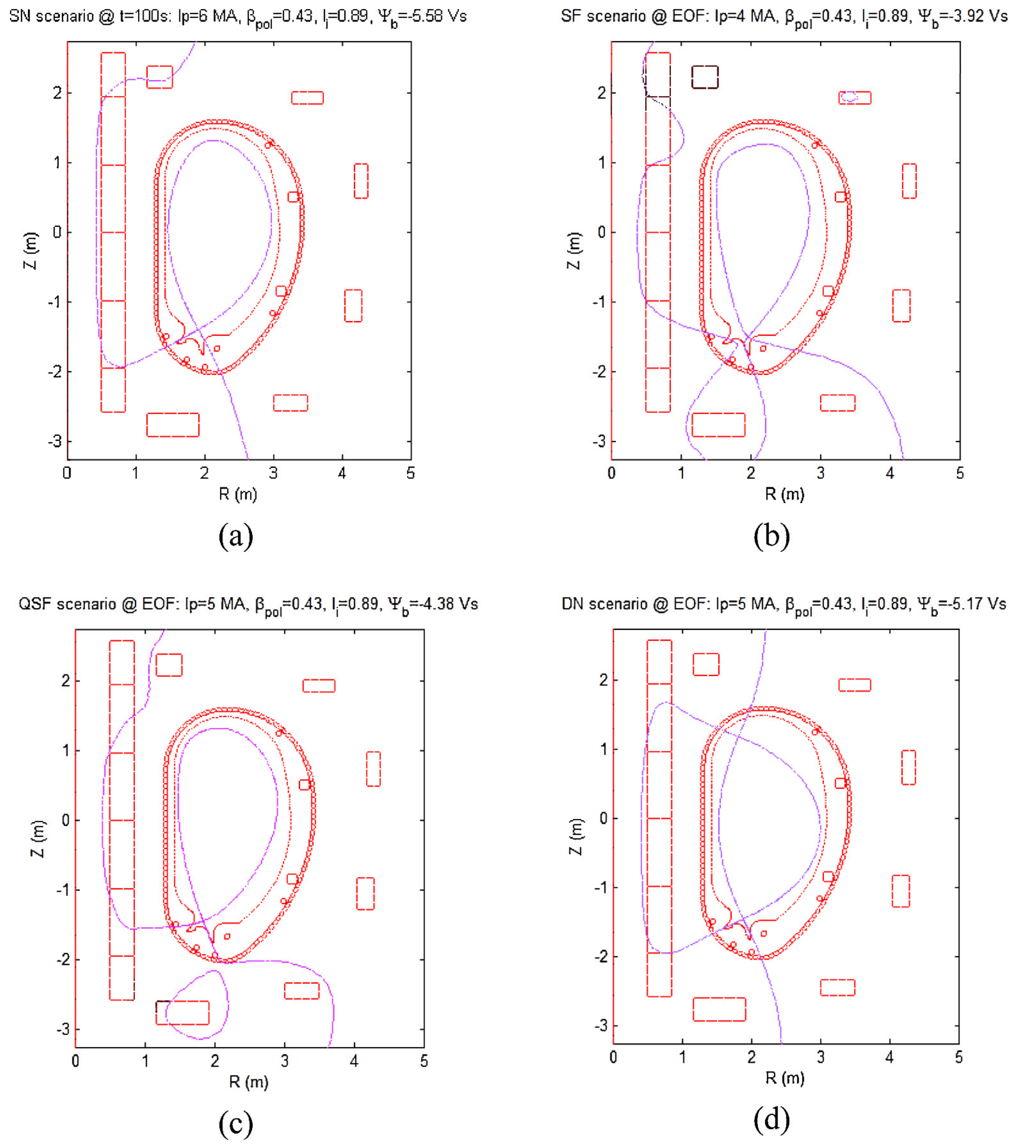


Fig. 3. Conventional and alternative magnetic configurations obtained using the DTT PF system: a) conventional single null (SN); b) snow flake (SF); c) quasi snow flake (QSF); d) double null (DN).

those of the EU DEMO (R/a ≈ 3.1, k ≈ 1.76, <δ> ≈ 0.35) [19]. The typical plasma parameters are shown in Table 1 and Fig. 3 for the standard and advanced configurations (mainly focused on X-Divertor and snow flake equilibria). It has to be noted that the machine is not up and down symmetric. Although the poloidal field system allows obtaining a symmetric double null plasma configuration (Fig. 3d), the divertor geometry will not be symmetric. On the machine top the remote handling system will allow installing a very simplified divertor, capable to manage the power flux, but not optimized at all from the geometry point of view. In the DTT proposal [6], there was no mention about the DTT possibility to realize "long leg" and/or Super X configurations, since the study of these configurations had been given a secondary priority. However, as illustrated in [19], DTT will have the possibility to realize "long leg" configurations at I_p = 5MA with a "standard" divertor geometry. A Super X configuration will be feasible, on the outboard side, at I_p = 3MA and $R_{Strike}/R_X \approx 1.8$, but it will require a dedicated divertor. The final realization of the DTT standard scenario, lowering the power flux on the plates to about 10 MW/m^2, would indicate by itself the full success of the facility. But lowering the No Radiation power flux (≈ 100 MW/m^2) to this safe technical figure it implies the success of the full project: a) the synergy between radiation and alternative divertor magnetic configuration (and/or materials); b) the integration of point a) with maintaining the high quality plasma confinement properties.

7. Basic machine

A schematic view of the DTT facility is shown in Fig. 4. Given the foreseen neutron flux the machine will have the possibility to remotely operate on all the internal systems. Given the importance of this point the machine design has been realize including, since the very beginning, all the Remote Handling constraints and/or request. Given the Italian laws the remote handling will not be necessary during the first years of operations, but given its importance, the present planning foreseen to have the remote handling full operative since the very first day, in order to set up and test it during a long time period.

7.1. Cryostat

The basic machine is surrounded by the Cryostat Vessel (CV), a 40 mm thick vacuum tight container, which provides the vacuum

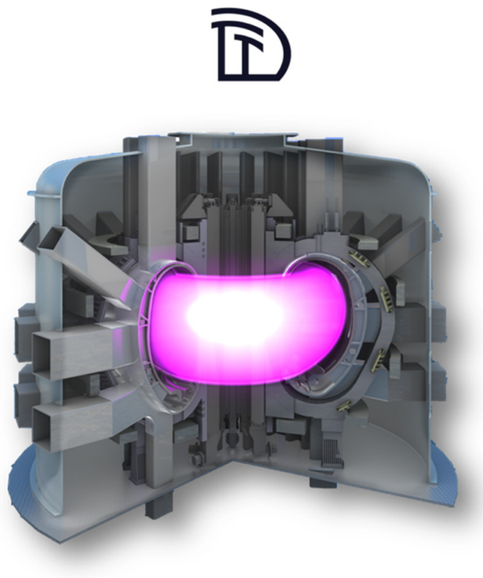


Fig. 4. DTT view.

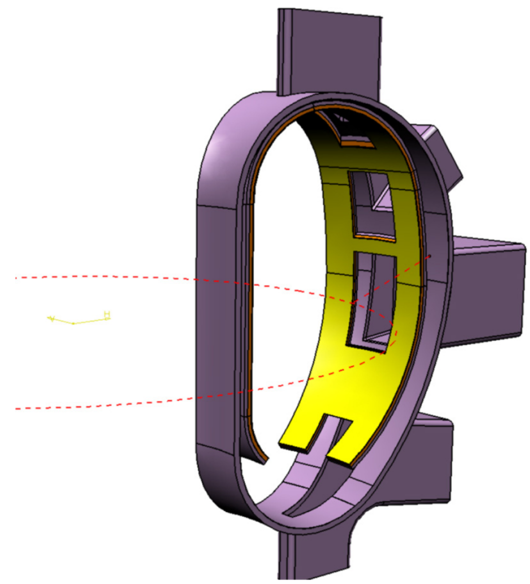


Fig. 5. View of the DTT vacuum vessel and first wall.

for the superconducting magnets and forms part of the secondary confinement barrier. The vacuum environment is intended to avoid excessive thermal loads from being applied to the components that are being operated at cryogenic temperatures by gas conduction and convection. The CV provides ports and penetrations, with proper bellows, to the vacuum vessel. The external diameter of the stainless steel CV is 10 m, the internal height is 8.5 m, and the total weight is around 150 tons.

7.2. Toroidal field coils

The need of a long duration of the flat top (about 100 s) has suggested the use of superconducting windings [6,14,20]. The present design is with a number of 18 toroidal field (TF) coils, which allow sufficient space for the ports and at the same time keep the TF ripple below the threshold of 1%.

Each of the 18 D-shaped coils is wound by 78 turns of Nb3Sn/Cu Cable-In-Conduit (CIC) conductor, carrying 46.3 kA of operative current, cooled by a forced flow of supercritical Helium, having an inlet temperature of 4.5 K.

In order to optimize the allocation for both the stainless steel (SS) and the superconducting (SC) material, the winding pack (WP) is designed in a graded solution, combining two different Nb3Sn CIC conductor layouts. In particular, a low field (LF) section of the WP is constituted of 48 turns with conductors characterized by thicker jacket and lower SC strand number, whereas a high field section (HF) is wound by a more performing conductor in 30 turns arrangement. Each of the two sections is wound in pancakes, in order to reduce the He path and thus better manage the expected nuclear heat load.

The TF coil design features are hereafter summarized:

- $B_{\text{plasma-axis}}$: 6.0 T;
- $B_{\text{peak-HF}}$: 11.4T for high field (HF) grade;
- $B_{\text{peak-LF}}$: 7.6T for low field (LF) grade;
- total current flowing in the 18 coils: 65 MAt.

7.3. Central solenoid and poloidal field coils

The poloidal field (PF) coil system includes a central solenoid (CS), six external PF coils, and eight in-vessel coils (Fig. 3) [6,14,20]. Due to the long duration of the pulse, the CS and the external PF coils are superconducting.

The CS assembly consists of a stack of six circular coils, named modules (CS3U to CS3L), 4 of which (CS2U to CS2L) identical, and the other two slightly shorter but with the same radial dimensions; a pre-compression structure is foreseen. The CS operates at a peak field of 12.5 T, so it relies on Nb_3Sn as superconductor material. The conductor concept is that of a rectangular CIC conductor with low void fraction, cooled by supercritical helium, manufactured by deformation from a round tube of constant thickness. The CS design provides an available poloidal flux swing of ±17.6 Vs.

The 6 external PF coils are designed to operate in a not-challenging range of parameters (peak field of 4.0 T). Therefore the superconducting NbTi material has been chosen, and the differences in the six conductors are mainly driven by the need to find the best trade-off between the space availability and the requested performances.

The PF system also includes eight copper in-vessel coils; in particular:

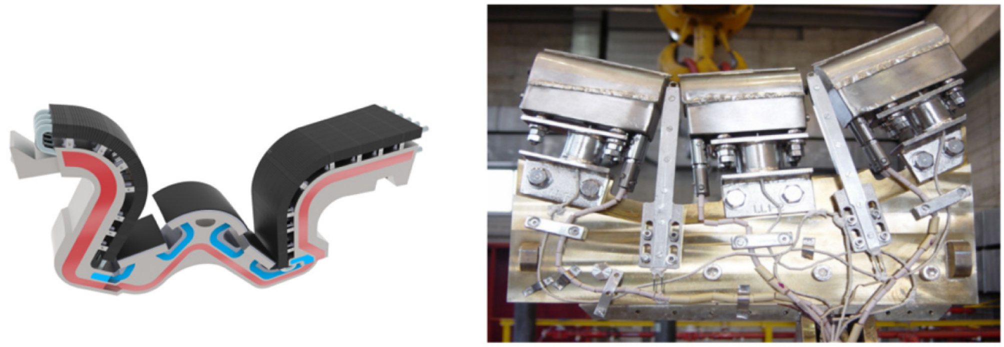
- two in-vessel coil for radial and vertical stabilization and control;
- four out of six in-vessel coils for local changes of the magnetic topology in the divertor region.

These in-vessel copper coils are water cooled and carry modest currents and therefore they have the capability of running through the whole pulse for all scenarios.

7.4. Vacuum vessel and shield

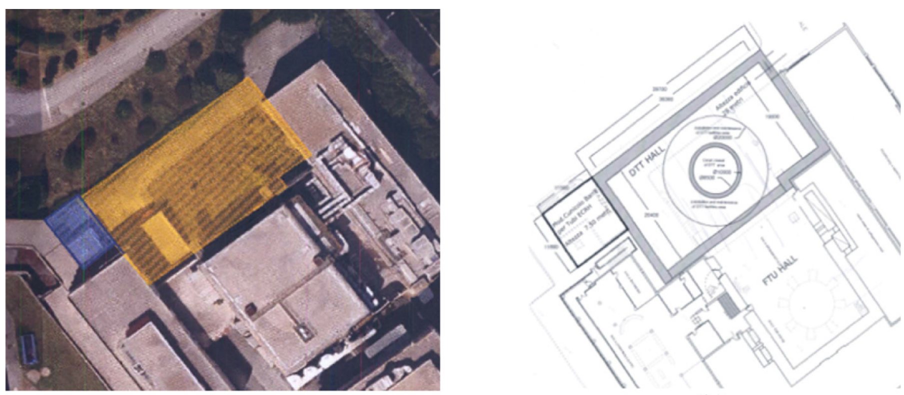
The vacuum vessel (VV) is located inside the magnet system. It provides an enclosed, vacuum environment for the plasma and, in addition, acts as a first confinement barrier. It is composed by 18 sectors joined by welding. The main components are the main vessel, the port structures and the supporting system.

The design of the VV includes a wall of INCONEL 625 (Fig. 5). The maximum thickness of the shell is 35 mm, while the 5 ports per sector are 25 mm thick [6,21]. The L/R time constant is about 40 ms. These features ensure to keep the parameters of the vertical instability within a range that can be controlled using the internal coils C5 and C6 with a maximum current of 25 kA (maximum



(a) (b)

Fig. 6. DTT divertor: a) a W-shaped tungsten divertor, compatible with both the single null and snow flake configurations; b) liquid lithium limiter in FTU.



(a) (b)

Fig. 7. Proposed DTT site in Frascati: a) aerial view on of the present FTU buildings, with the necessary upgrades for DTT highlighted in yellow; b) design of the new hall and the present FTU hall.

growth rate of 70 s^{-1} with a stability margin of 0.4) in case of 1.2 MJ ELMs or VDEs detected after more than 40 mm displacements.

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

Although DTT is designed to operate without tritium, the assessment of the radiation fluxes, loads and radiation damage is crucial in the design of the machine as a significant DD neutron yield is expected (in the order of $1.3{\cdot}10^{17}$ n/s for the reference H-mode scenario). Considering the attenuation capabilities of a B_4C shield and the available space between VV and TF case (50 mm), the total nuclear loads on the TF coil is expected to be about 5 kW. A reduction of this figure to about 2.5 kW can be achieved by increasing the shielding thickness (acting on the VV design and/or the operational density) [6].

7.5. First wall

The first wall (FW) surrounds most of the vessel wall. Heat loads on the FW in normal operation include radiation and particle bombardment from the burning plasma. The power transported by neutrals from charge-exchange is important only locally near neutral particle sources for fuelling. Its temperature will be kept around 300 ÷ 400 °C in order to avoid impurities adsorption.

The FW consists of a bundle of tubes armored with plasma-sprayed tungsten (W). The plasma facing tungsten is about 5 mm thick (except for the equatorial and upper inboard segments where the tungsten layer is about 10 mm thick), the bundle of stainless steel tubes (coaxial pipes in charge of cooling operation) is 30 mm thick, and the backplate supporting the tubes is 30 mm thick of SS316L(N) [6,21].

7.6. Divertor

One of the main objectives of the DTT project is to test several divertor concepts and configurations. Initially, the machine will operate with a standard single null configuration. Afterwards, advanced configurations and liquid metal divertors should be tested. So far no experience exists on relevant plasma scenarios with liquid metal divertors. Consequently, several problems (for instance possible accumulation in the SOL and or in plasma) are still waiting for simulations carried out in realistic geometries and confirmed by some experiment. Recently some preliminary studies [22,23,24] have started to evaluate the fluid dynamics of a conceptual liquid metal divertor for the DTT geometry.

For the first operation phase, the basic machine design includes a tungsten divertor, with W-shaped modules, distributed along the VV; the design is compatible with both single null and advanced magnetic configurations. Furthermore the design of VV, ports, remote handling devices, and additional power systems is compatible with the application and testing of a liquid metal divertor (Fig. 6) [6,17,18,19].

7.7. Additional heating, power supplies and auxiliary subsystems

About 40–45 MW of heating power are foreseen to guarantee the achievement of the design parameter P_{SEP}/R of 15 MW/m, when assuming ≈ 30% core radiation (i.e. no seeding). Of course this radiation scenario is different from DEMO, where to achieve the same

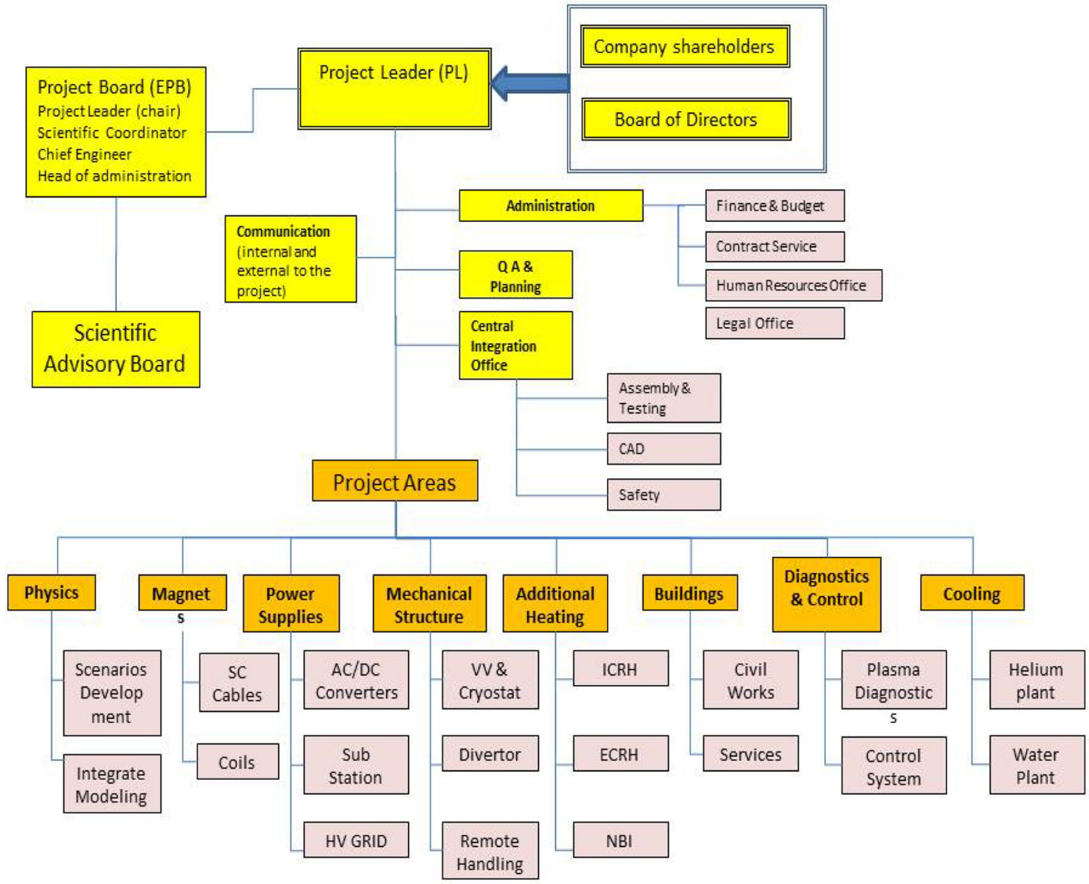


Fig. 8. DTT organization scheme.

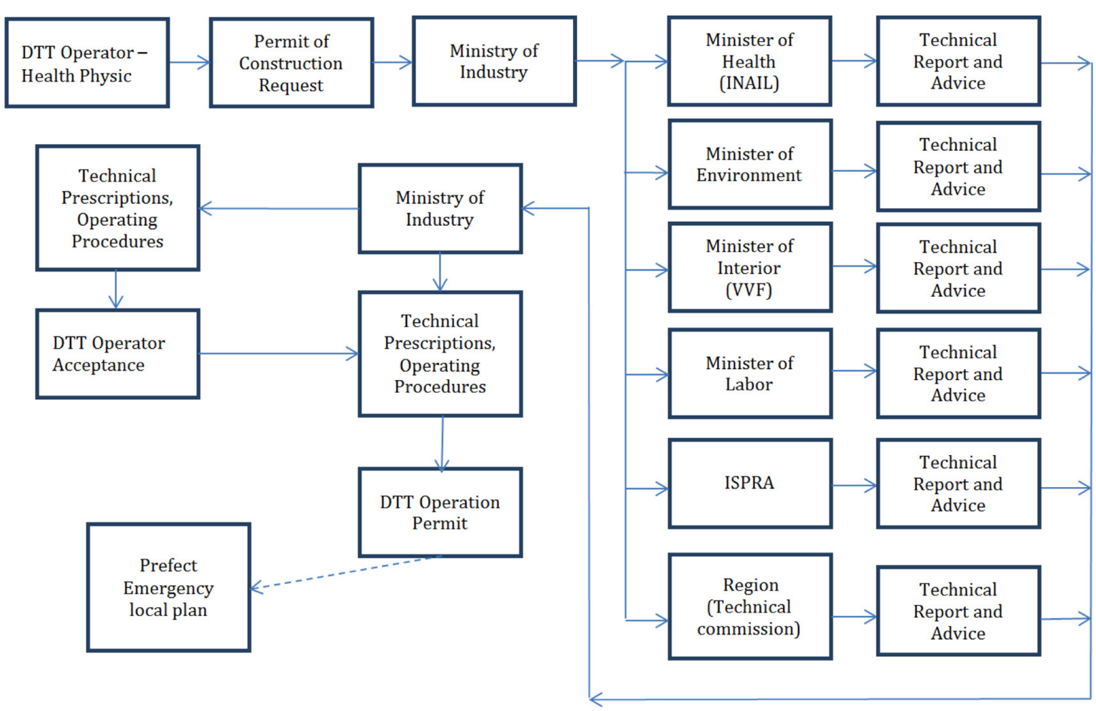


Fig. 9. DTT Licensing Scheme.

P_{SEP}/R a radiation of around 90% is required; but, eventually DTT will have (by using seeding and operating at high density) to realize very high radiative scenarios, that, of course, will strongly low P_{SEP}/R. For a robust and reliable heating, a mix of the three heating systems presently proposed for ITER has been chosen, assuring the necessary flexibility in scenario development. An ECRH system at 170 GHz will provide 10 MW at plasma for several tasks, such as: bulk electron heating to bring the plasma in the high confinement regime, current profile tailoring by localized CD, avoidance of impurity accumulation, MHD control and current ramp up and ramp down assistance. In addition, 15 MW of ICRH (in the range 60–90 MHz) will provide the remaining bulk plasma heating power, on both electrons and ions. ICRH, in minority scheme, will produce fast ions, with an isotropic perpendicular distribution, allowing the study of fast particle driven instabilities like alphas in D-T burning plasmas. The heating schemes foreseen in DTT are 3He and H minority as well as Deuterium 2nd harmonic. Additional 15 MW of NBI, to be included later in the project, could provide a mainly isotropic parallel fast ion distribution to simulate the alpha heating scheme of a reactor [16]. The NBI primary aim is to support plasma heating during the flat top phase when the need of central power deposition and the minimization of the shine-through risk suggest selecting a beam energy around 300 keV. In the first phase of the DTT operation the available power will be at least 25 MW, to be increased during the lifetime of the machine [6,14,25].

The total electric power demand for magnets, additional heating and auxiliary systems is about 180 MW (active power). The power supplies for CS and PF coils include 4-quadrant 12-pulse AC/DC converters in series to quench protection circuits and, in most cases, switching network units. The voltages and currents to be provided by the converters are estimated applying the reference scenarios to a model of the PF circuits, taking into account the mutual couplings and the SNU contributions [6,14,26].

The independent evaluation of the electrical requirements of each PS system led to the definition of the active, reactive and apparent power scenarios. Due to the pulsed PSs (serving CS, PF, ECRH, ICRH, NBI), the 100-MVA continuous load can reach 350 MVA with a duty cycle of 100 s/3600 s.

To pursue the aims of the program, particular attention has been devoted to the diagnostics and control issues, especially those relevant for plasma control in the divertor region, anyway having in mind the requirement of a strong compatibility with the operating conditions in DEMO [6,14,27].

It is planned to have a remote handling (RH) equipment operating since the beginning for the following reasons:

- the Italian laws on radioactivity are relatively stringent, preventing any hand contact above a threshold that, depending on the plasma performance, might also be reached during the first phase;
- a test and training time will be in any case requested before RH becomes necessary;
- RH, avoiding the need to wait for the decay of the dose rate, will make faster any maintenance intervention that may be required.

All the remaining subsystems are described in [6,28].

8. Cost, schedule, site and licensing

The project includes the analysis of the site requirements from several points of view; among other alternatives the ENEA Frascati Research Center (FRC) has been indicated on the basis of technical, scientific, organizational and economics considerations. FRC is well suited from this point of view. Since 1960, FRC hosts most of the Italian fusion research. Presently the FTU machine is in operation at FRC. For the DTT plant requirements it will be possible to adapt the complex FTU buildings except the DTT hall and the cryoplant. The DTT hall will be an extension of the present FTU 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. The dimensions of the new hall are $30 \times 20 \times 28$ m on three levels. On the lowest one, the cold boxes for the electrical connection of the superconductive coils will be placed while in the intermediate level the diagnostic using the bottom ports will be arranged. The third level starts at the cryostat bottom and will host all the additional heating system and the diagnostics. The machine is particularly demanding in terms of power supplies and the grid requires an extension of the 150 kV line. Discussions are in progress with the operators for energy transmission. The tunnel solution is recommended to prevent possible environmental impact.

Table 2
Costs for the basic machine.

Main Components	Cost (M€)
Load Assembly	224.10
Auxiliary Heating Systems	96.00
Principal diagnostic systems	8.00
Controls and Data Acquisition System	4.50
Cooling System	27.40
Power Supply	78.00
Remote Handling	14.00
New buildings	11.00
Assembly	11.00
Contingency	25.00
Total	**499.00**

The ENEA FRC has the possibility to realize the DTT facility, given its capability to meet the various technical requirements. The presence of FTU Tokamak facility would make much easier the authorization and licensing procedures of the new machine [6,14,28].

Fig. 7a 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 infrastructures and will be re-used for DTT with some minor internal modifications. Fig. 7b shows the location of the DTT in the new hall. Fig. 8 shows the organizational scheme of DTT, whereas the planned licensing scheme is illustrated in Fig. 9 [28,29].

The facility needs to be ready in the early 2020s, in order to be able to bring at least one alternative divertor strategy to a suitable level of maturity by 2030 for a positive decision on DEMO. The nominal duration of the construction of DTT from the “green light” to the beginning of the initial operational phase is expected in about seven years. The realization of the DTT project is a top priority for the world of European research, since it represents a crucial step towards the realization of a DEMO reactor. The DTT scientific program was included in the list of projects submitted for funding of 500 M€ as part of the 315 billion € European Fund for Strategic Investments Plan. The amount claimed is consistent with the costs summarized in Table 2 and, in more detail, in [6]. Referring to the DTT foreseen schematic scientific program (Fig. 2) a yearly averaged budget of about 40 M€ is foreseen. This figure will cover the operation and the planned enhancements (Fig. 2).

9. Conclusions

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

DTT main parameters have been worked out in order to realize a facility capable to approach, as much as possible within a reasonable time frame and a reasonable budget, scenarios DEMO relevant conditions in terms of plasma Physics parameters and Power Exhaust performances. The machine will be very flexible in the divertor region giving the possibility to test different magnetic configuration, different divertor geometries and/or materials, including liquid metals. But the main and most important DTT feature will be the possibility to actually test an integrated plasma bulk and plasma edge solution in condition not far from a reactor one.

The design of the machine is not frozen. Future upgrades are already planned in the proposal [6], including possible replacement of the divertor and first wall modules, double null divertor and increase of plasma heating capabilities. In addition, the interaction with the EUROfusion activities might lead to a revision of the machine design, with slight modifications still compatible with the construction schedule, but able to improve some aspects related to the advanced configurations, the temperature of the first wall, the pumping capabilities, the dimensions and the costs of some components.

The ultimate DTT target will be to propose a "reliable and robust" power exhaust scenario for DEMO: a scenario fully compatible with the available technology (materials, pumps, coils,...) and the plasma performance (no degradation of the confinement).

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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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