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The DTT proposal: introduction and executive summary

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The DTT proposal: introduction and executive summary

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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.

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 the Italian proposal showing how the basic machine parameters and concept have been selected so as to accomplish the DTT mission.

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. The proposal is among the projects submitted to the 315 billion € European Fund for Strategic Investments plan (EFSI).

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 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, the risk exists that the baseline strategy pursued in ITER cannot be extrapolated to a fusion power plant for the following reasons [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;
- 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.
    - …

Moreover, a number of nuclear 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].

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

While several alternatives, such as the cooled liquid Li limiter in FTU [7], the Super-X divertor in MAST-U [8] are being investigated 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 programme 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 [9].

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, 9], can be summarized as follows:

- demonstrate that 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 that the possible (alternative or complementary) divertor solutions (e.g., advanced divertor configurations or liquid metals) can be integrated in a DEMO device.

In particular it will be possible 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 of DTT

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 can achieve its objectives with an available budget of 500M€, 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 performances 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 lowing 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 [10-11] 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 [10-11]. Therefore, to limit the size and the cost, the solution proposed by DTT project team [5, 11-13] is to relax in a controlled way one of these parameters (the normalized Larmor radius) while preserving the remaining physics aspects.
Table I reports the main DTT parameters for a reference standard single null scenario.

5. DTT operational programme

Figure 2 shows a schematic planning of the DTT operations [12, 14]. A first phase will be aimed at the realization and installation of the various components of the machine. In a subsequent phase about a year and a half long, the machine will reach the operative capability in modality robust H-mode (i.e. operating regimes characterized by configurations of single-null divertor type, top performance and with all the additional power installed). The next phases will be reserved to test a number of alternative divertor solutions, including new magnetic configurations and innovative technologies in the liquid metal.

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 those of DEMO (R/a≈3.1, k≈1.76, <δ>≈0.35) [15]. The typical plasma parameters are shown in Table I and Fig. 3 for the standard and advanced configurations.

7. DTT basic machine

A schematic view of the DTT facility is shown in Fig. 4.

7.1 Cryostat

The basic machine is surrounded by the Cryostat Vessel (CV), a 40 mm thick vacuum tight container, which provides the vacuum 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, 12, 16]. The present design is with a number of 18 toroidal field (TF) coils, with 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 46.3kA of operative current, cooled by a forced flow of supercritical Helium, having an inlet temperature of 4.5K.

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:

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

7.2 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, 12, 16]. 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 room 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 coil for local changes the magnetic topology in the divertor region.

7.3 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, 17]. 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 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].

7.4. 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, 17].

7.5. 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.

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].

7.6. 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. 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, 15MW of ICRH (in the range 60- 90MHz) 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. 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, 12, 20].
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, 12, 21].
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 100s/3600s.
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, 12, 22].
All the remaining subsystems are described in [6, 23].

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 30x20x28 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.
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, 12, 23].
Figure 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. Figure 7b shows the location of the DTT in the new hall. Figure 8 shows the organizational scheme of DTT, whereas the planned licensing scheme is illustrated in Fig. 9 [23-24].
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 II and, in more detail, in [6].

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.

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.

Acknowledgments

This paper is largely based on the Italian proposal for DTT prepared with contributions of European and international experts [6], the activity carried out inside the EUROfusion work package WPDTT2, and Ref. [12] related the a presentation given at the 2nd IAEA DEMO Programme Workshop.
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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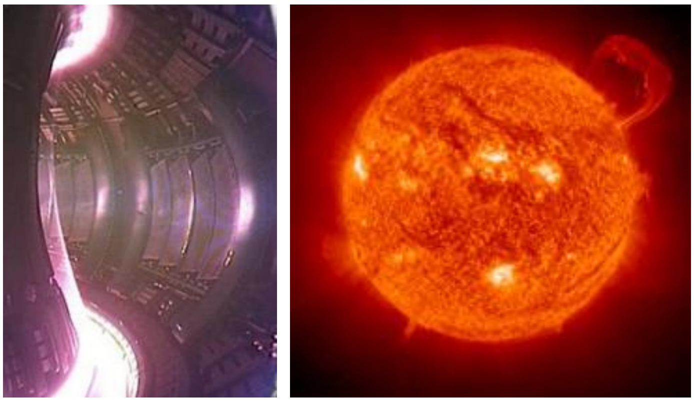


Extreme conditions:

- *in nucleus: T > 100 M °K*
- *in the SOL: $q_{//} > 1$ GW/m^2*

The geometry reduces it by a factor of 30:

- *B toroidal >> B poloidal*
- *Expansion of the flux on divertor*
- *Plate inclination*

... ... But the heat flow on the plates is still higher than the current technological limits (5-10 MW/m^2)

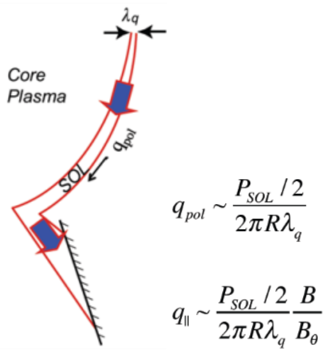


Figure 1. Power flux on the divertor.

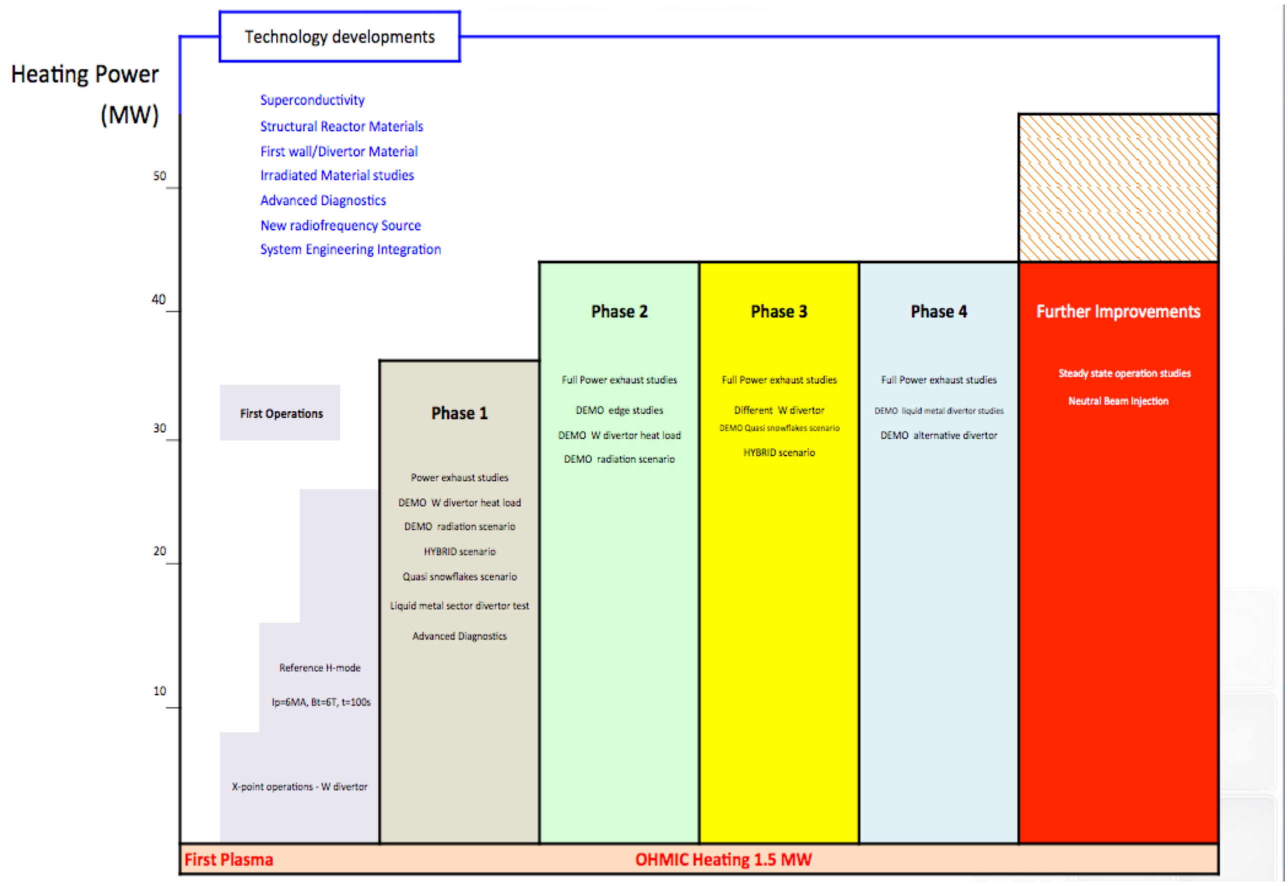


Figure 2. Schematic planning of the DTT operations.

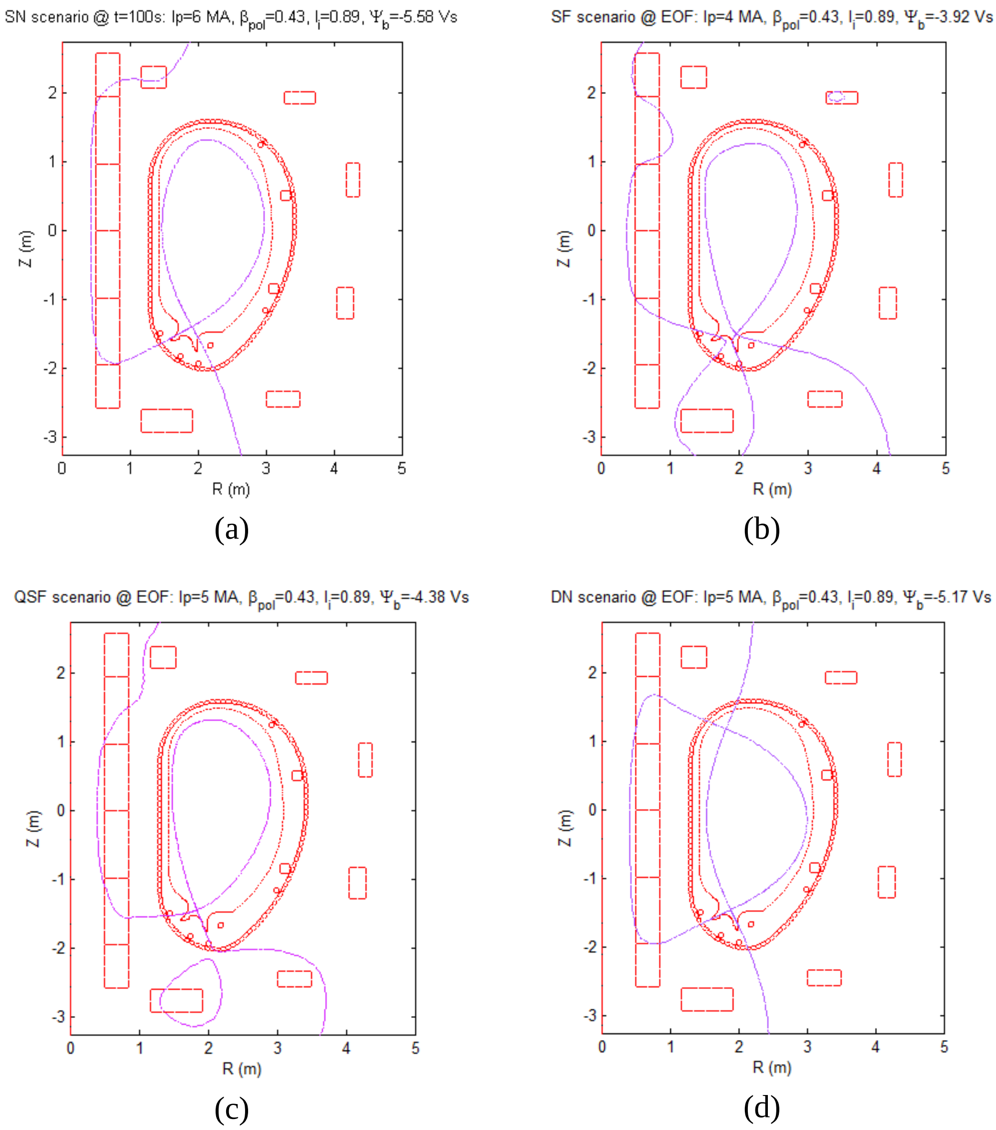


Figure 3. Conventional and alternative magnetic configurations obtained using the DTT PF system: a) conventional single null (SN); b) snow flake (SF); c) quasi snow flake (QSF); d) double null (DN).

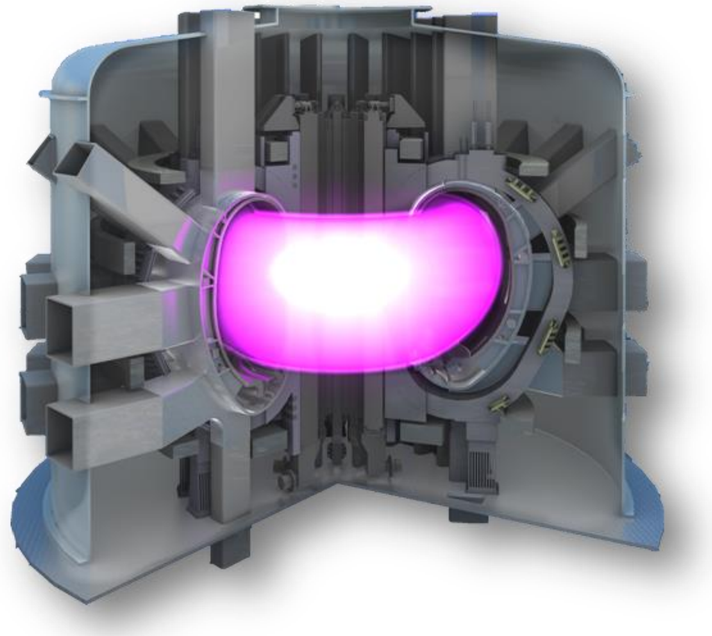


Figure 4: DTT view.

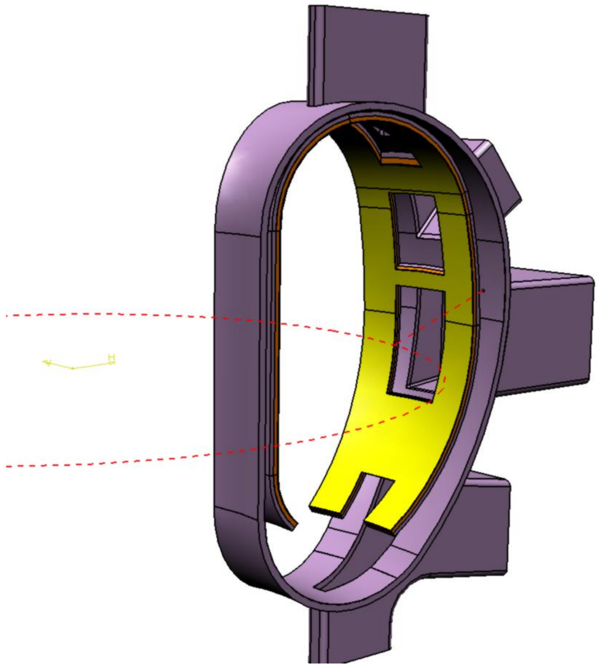
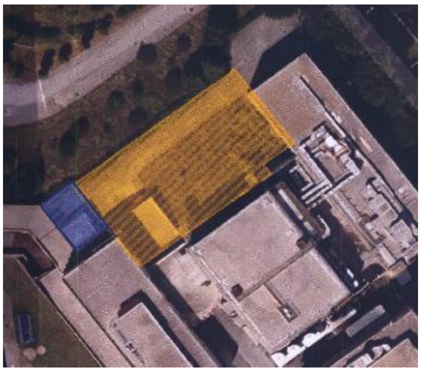


Figure 5. View of the DTT vacuum vessel and first wall.

(a)

(b)

Figure 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)

Figure 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.

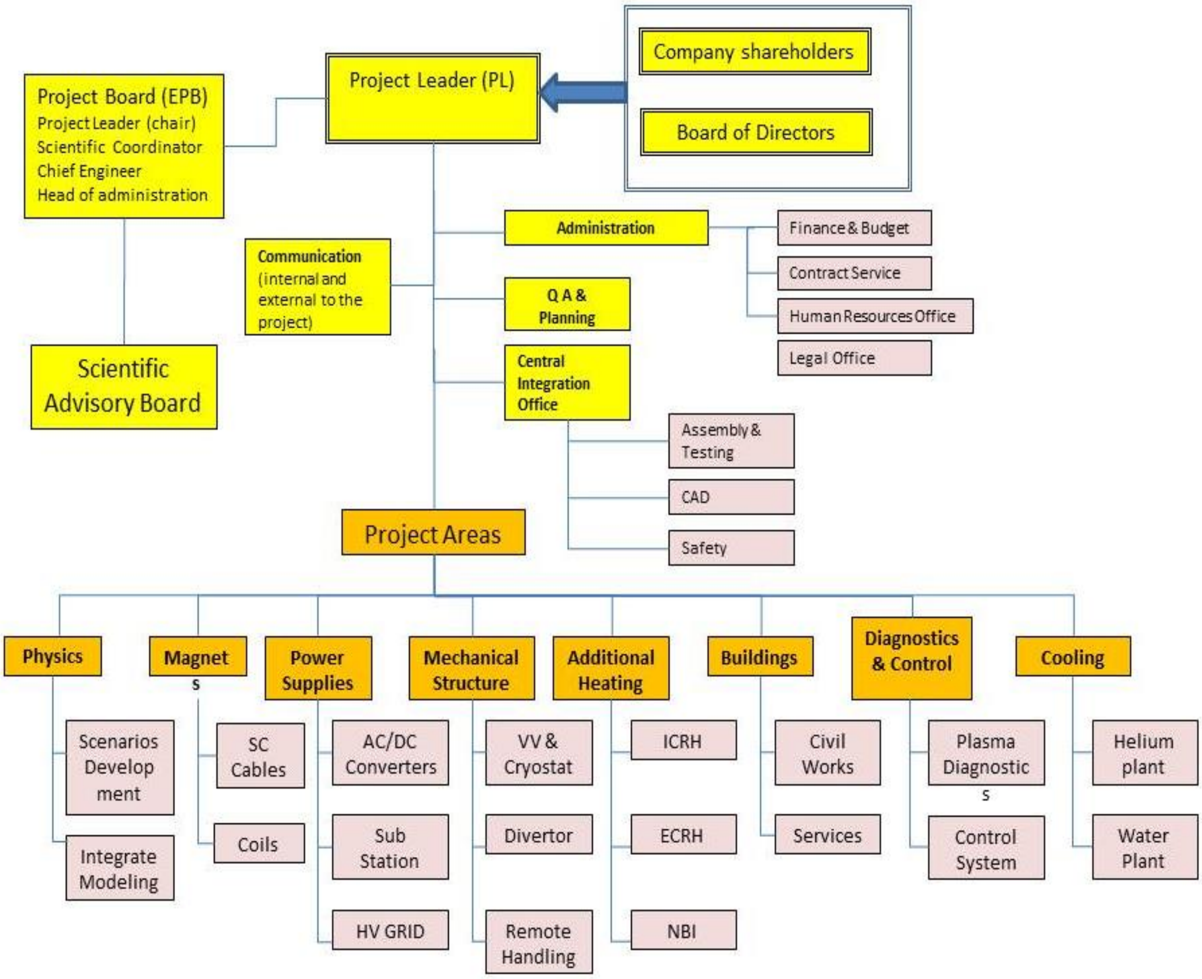


Figure 8. DTT organization scheme.

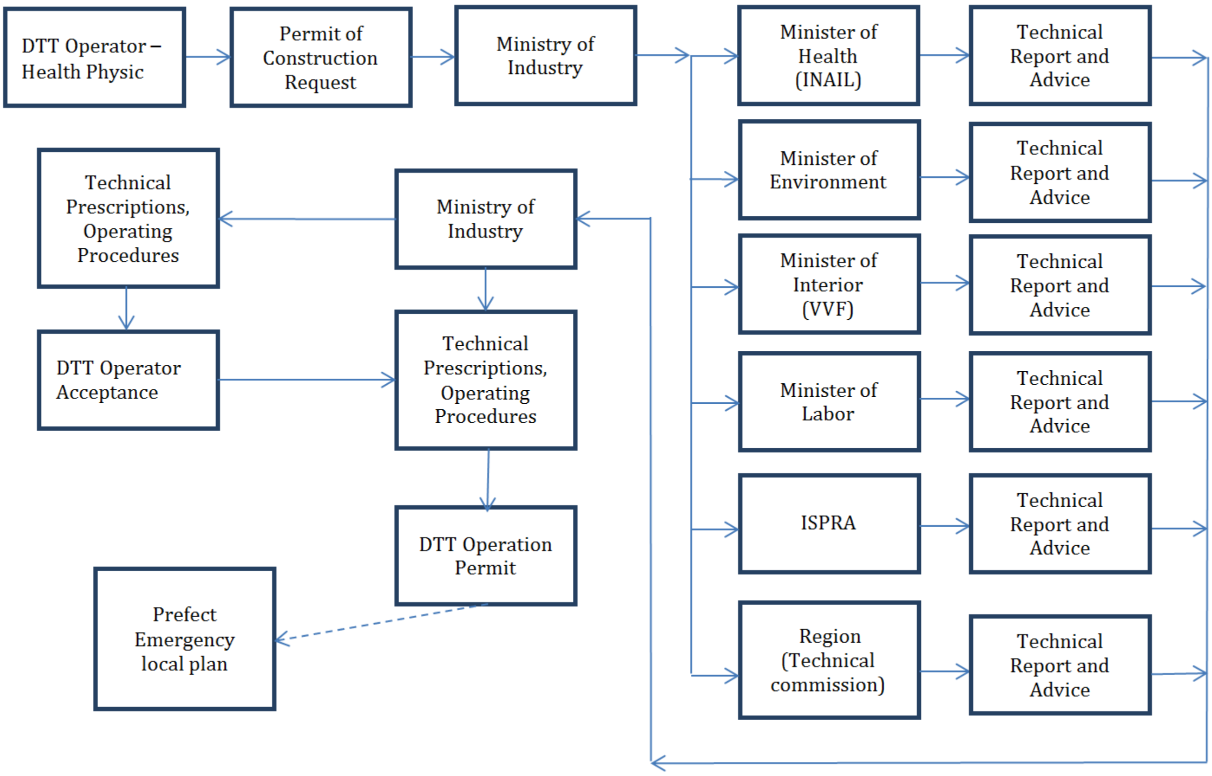


Figure 9. DTT Licensing Scheme.

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

R (m)	2.15
a (m)	0,7
I_P (MA)	6
B_T (T)	6
V (m^3)	33.0
P_{ADD} (MW)	45
H_{98}	1
$<n_e>$ (10^{20} m^{-3})	1.7
n_e/n_{eG}	0.45
$<T_e>$ (KeV)	6.2
τ (sec)	0.47
$n_e(0)$ (10^{20} m^{-3})	2.2
$T_e(0)$ (KeV)	10.2

β_N	1.5
τ_{Res} (sec)	8
V_{Loop} (V)	0.17
Z_{eff}	1.7
P_{Rad} (MW)	13
P_{Sep} (MW)	32
T_{Ped} (KeV)	3.1
n_{Ped} (10^{20} m^{-3})	1.4
β_p	0.5
P_{Div} (MW/m^2) (No Rad)	~ 55
P_{Sep}/R (MW/m)	15
$P_{Tot}B/R$ (MW T/m)	125
λ_q (mm)	~ 2.0

Table II. DTT investment costs

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**

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