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DTT: a Divertor Tokamak Test facility for the study of the power exhaust issues in view of DEMO

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Abstract

One of the main challenges in the European fusion roadmap is to design a particle and power exhaust system able to withstand the large loads expected in the divertor of a DEMO fusion power plant. Therefore, 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 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 (DTT). This tokamak should be capable of hosting scaled experiments integrating most of the possible aspects of the DEMO power and particle exhaust. DTT should retain the possibility to test different divertor magnetic configurations, liquid metal divertor targets, and other possible solutions for the power exhaust problem. 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 the given constraints. The paper will illustrate 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 to have a 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 SOL (Scrape-Off Layer) in the divertor region, radiation control. The Central Solenoid (CS) and the external Poloidal Field (PF) coils are planned to provide a total flux swing of more than 35 Vs, compatible with a pulse length of more than 100 s. This pulse length, fully compatible with the mission of the study of the power exhaust problem, is obtained using superconducting coils (NbTi for PF coils, Nb3Sn for the CS and the TF coils). Additional heating of 25 MW will be provided in the first phase of the operation using ICRH and ECRH. Afterwards, the ECRH heating power will be increased and NBI launchers will be added up to a planned total power of 45MW. The vacuum vessel is a single 35 mm shell of INCONEL 625, with five ports in each of the 18 sectors. The first wall is made of 5 mm of tungsten coating on a 60 mm stainless steel structure. The tungsten coating is thicker in selected zones of the first wall (where the plasma leans during the limiter phases of ramp-up and shut-down, where the plasma is expected to hit the wall in a disruption, at the upper strike points of double null configurations). Particular attention will be dedicated to the 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€.

1. Introduction

In 2012 EFDA published "Fusion Electricity – A roadmap to the realization of fusion energy" [i], which sets out a strategic vision toward the generation of electrical power by a Demonstration Fusion Power Plant (DEMO) by 2050.

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 metals PFCs (Plasma Facing Components)
- 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 ≤1MA 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 2020's 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." [i].

The radiation baseline strategy will be tested on ITER [ii], it 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÷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 ionization’s, 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 ITER divertor will prove 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…).
The full basket of this problems provide the necessity of a dedicated Divertor Tokamak Test facility, flexible enough to study, test and propose a solution that eventually will be directly used 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 frame, the Work Package WPDTT2 has been organized with several different working groups and divided in 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 in 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 of the WPDTT2 have been completed along the 2015th, the decision of going ahead with the last step (DTT engineering design and construction) is presently under evaluation of EFDA organisms and a final decision should be taken during the second part of 2016th. Meanwhile the Italian Association for Fusion, working within the WPDTT2, has produced a conceptual design for a DTT tokamak facility [iii], that should be able to operate with plasma bulk non-dimensional parameters very close to the DEMO ones and with a divertor region so flexible to be able 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 500M€ has been assumed as a constraint for the facility design. Although the WPDTT2 project has tackled a wider set of aspects of what presented in the Italian DTT proposal (for instance the experiments performed jointly with WPDTT1 on the Chinese EAST tokamak [iv], 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 essential describe what presented within this proposal, as a good synthesis 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 [v, vi]. To overcome this very challenging issue, several different approaches have been proposed [v, vii, viii, ix], 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 15MW/m 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 monoblocks technology [x]. Other two 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 [xi]. 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 for DEMO). Previous works [vi, viii] 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 [vii, viii] 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 [vii], and 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 [vi, xii]. However, it is not possible to simultaneously preserve all these quantities. A strategy has then been proposed, which consists in relaxing in controlled way one of these parameters, [xiii] 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 (ρ^*=Cost $\rightarrow$ B~1/R). Consequently ρ^* is the dimensionless parameter that can be relaxed in the controlled way ($\rho^*_R= \rho^*_S R^\varepsilon$, 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~$B^2R^3 \sim R^{2.75}$, when relaxing in the opportune way ρ^* (ε=0.75). As mentioned, the cost of the additional heating is not included in this scaling; in order to consider it, we can assume to be one third of the maximum cost ratio between the whole machine and the total additional power, Heating_cost/Total_cost≤0.3 (e.g. Machine_cost≈500M€ → Heating_cost≈150M€, where 500M€ 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 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: $R_{Max} \leq 2.3$m [iii].

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 brings to R_{Min}>1.5m [iii].
2) Reducing too much the plasma size, maintaining fixed the figure P_{SEP}/R>15MW/m, leads to a power flowing towards the first wall larger than the safe figure of about 1MW/m^2 for the power flux on a tungsten FW. Again this type of evaluation lead to R_{Min}>1.5m [iii].
3) The third and last example regards the discharge duration time (τ_S). An accurate discussion about this parameter involves several important points about different technologies and approach to be used in the machine design (for instance the use superconductors or standard copper coils). First obvious assumption is that the discharge must last at least 3÷4 time the diffusion resistive time (i.e. the longest Physics characteristic time); in the proposed DTT (see Table I) $\tau_R \approx 6$ s., leading to a minimum duration time of the order of 20 seconds. But it is quite obvious that, being the DTT 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 integrating in the plasma duration τ_S also the plasma build up and termination a reasonable $\tau_S \approx 100$ s must be assumed. Regardless of the coils used technology this leads to a minimum (when fixed the plasma current) R_{Min}>1.7m [iii].

From these three premises we could roughly estimate a minimum major radius of R_{Min}=1.7÷1.8m. Eventually, the integration of all the just discussed aspects in the design of the DTT tokamak facility leads to a proposal with a major radius R= 2.15m, a minor radius a=0.7m, a plasma current I_p=6MA, a toroidal field B_T=6T and an additional power P_{ADD}=45MW (see Table I).

3. DTT operational programme

Being the DTT a facility mainly dedicated to test innovative ideas to solve the power exhaust problem. A strong effort has been dedicated to verify the possibility to realize the largest possible set of "alternative" magnetic divertor topologies, as shown in Figure 1.
For a fair comparison all the shown equilibria 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 1a shows a standard X point with the machine target plasma current I_p=6MA; Figure 1b illustrates a Snow Flake equilibrium with I_p=4MA, the lower current being constrained by the poloidal coils maximum density current and for a discharge duration of 100 s. When relaxing this parameter the configuration can again be realized with $I_p \approx$ 6MA. Figure 4c shows a Quasi Snow Flake configuration (QSF) [iv], whereas Figure 4d illustrates a double null equilibrium. In both cases (Figures 4c-d) the plasma current is I_p=5MA, but we can the sane consideration realized for the pure SF. The double null configuration is not exactly up-down symmetric, because this would involve a larger machine volume, with the consequent strong increase of the total machine cost.
The presence of a set of small internal coils around the divertor will allow to locally modifying 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 Fig. 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 Fig. 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 to design a new divertor "dedicated and optimized" for an alternative divertor magnetic configuration. The following 7÷8 years will be dedicated to study 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 patent 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 along the previous years. The last part of the experimental activity will be dedicated to the achievement of steady state configurations. Along 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 to 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 by using liquid metal technology [xiv]. A mix of different heating systems will provide the required power (a possible power allocation could be ≈15MW Electron Cyclotron Resonance Heating (ECRH) at 170 GHz; ≈15MW Ion Cyclotron Resonance Heating (ICRH) at 60-90 MHz; ≈15MW 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 [iii]. 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:

- distance not less of 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: $R/a \approx 3.1$, $k \approx 1.76$, $\langle\delta\rangle \approx 0.35$ [xv];
- 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, 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 wounded by 78 turns of Nb3Sn/Cu CIC conductor, carrying 46.3kA of operative current cooled by a forced flow of supercritical Helium, having an inlet temperature of 4.5 K, with a maximum field of 11.4T 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 in 6 independent modules, plus 6 external coils (see Figure 4 and Table 2). The peak field on the central solenoid is 12.5T for a stored flux of ±17.6 Vs. For the 6 PF coils the peak field is 4.0 T. The PF system also includes eight copper in-vessel coils, namely two in-vessel coils for radial and vertical stabilization and control, and four out of six in-vessel coil for magnetic control of SOL and strike point sweeping.

4.3. Vacuum vessel and first wall

The design of the Vacuum Vessel (VV) includes a shell of INCONEL 625 (Figure 7a-b). The 18 sectors are joined by welding. The maximum thickness of the shell is 35 mm, while the 5 ports per sector are 25 mm thick. Its L/R time constant is about 40 ms.
These features of the vacuum vessel ensure the keep the parameters of the vertical within a range that can be controlled using the internal coils C5 and C6 with a maximum current of 25 kA (growth rate of $20 \div 70\ s^{-1}$ with a stability margin of $0.4 \div 0.8$) 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 INCONEL 625 admissible stress limit.
The first wall (Figure 7a-c) consists of a bundle of tubes armored with plasma-sprayed tungsten. The plasma facing tungsten is about 5 mm, the bundle of copper tubes (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), a remote handling system (Figure 7d) 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 takes into account also 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 8a). Furthermore the design of VV, ports and RH devices shall be compatible with the application and testing of a liquid metal divertor (figure 8b).

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 is being considered as the main candidate for a subsequent power upgrade. Along the experimental exploitation the total amount of heating will be upgraded up to 45 MW; the final sharing will be decided on the basis of the gained experience, however NBI is being considered as the main candidate for a subsequent power upgrade up to 15MW.
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 ±800V (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 for plasma control in the divertor region, designed to be as compatible as possible with a DEMO-like environment (Table III).

4.6 Site and licensing

Taking into account the role of DTT as "European facility", the proposed site is Frascati. In this view, 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 much easier the authorization and licensing procedures of the new machine. 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 9a 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 9b 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.

5. Conclusions

This proposal for a DTT facility demonstrates the possibility to set up a facility able to bridge the technological gap between the present day devices and ITER/DEMO. The DTT scientific project is well framed within the European fusion development roadmap [i], 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 [iii], on the activity carried out inside the EUROfusion work package WPDTT2 [xvi], and a presentation given at the 2nd IAEA DEMO Programme Workshop [xvii].
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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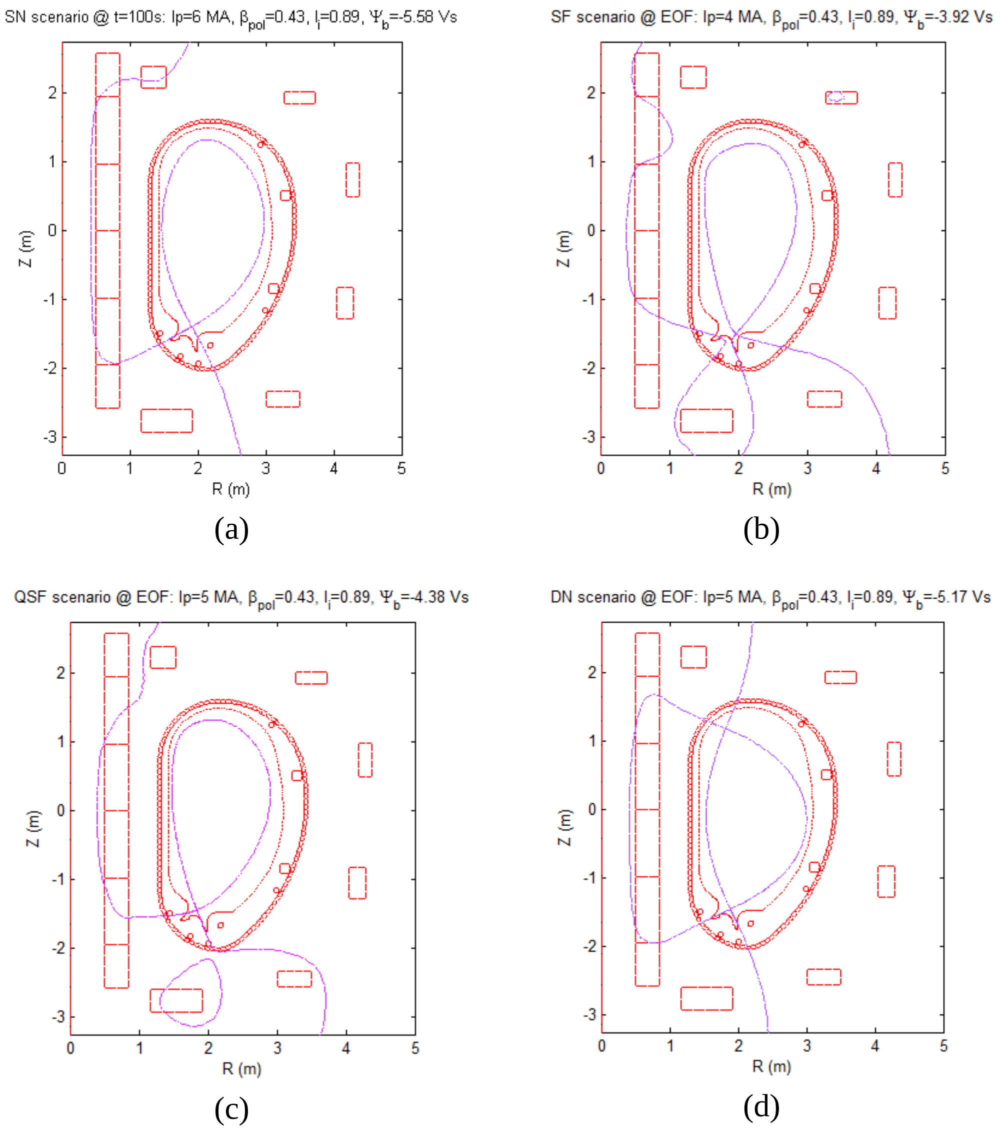


Figure 1. Conventional and alternative magnetic configurations that can be obtained using the DTT PF system: a) conventional single null (SN); b) snow flake (SF); quasi snow flake (QSF); d) double null (DN).

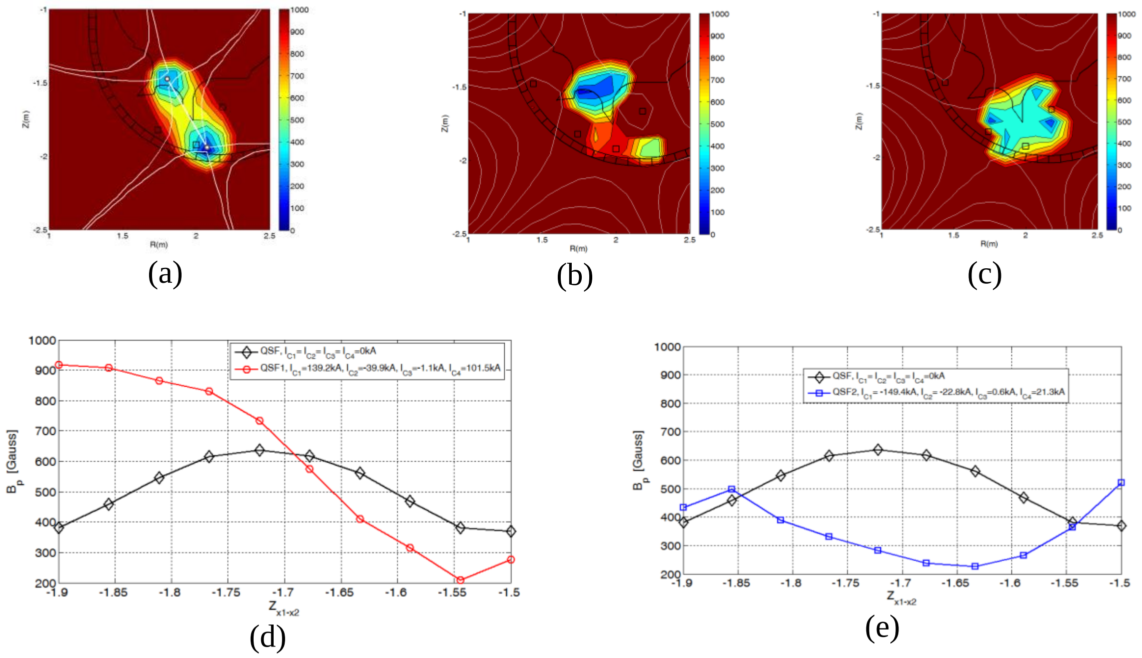


Figure 2. Use of in-vessel coils for the local modification of the magnetic topology and behavior of the poloidal magnetic field B_p in the region between the two nulls as a function of the vertical co-ordinate z: a) reference QSF configuration obtained by the external coils; b) modified QSF1 configuration; c) modified QSF2 configuration; d) monotonic slope of $B_p(z)$ in QSF1 compared to the hillock-like behavior of QSF; e) dip-like behavior of $B_p(z)$ in QSF2.

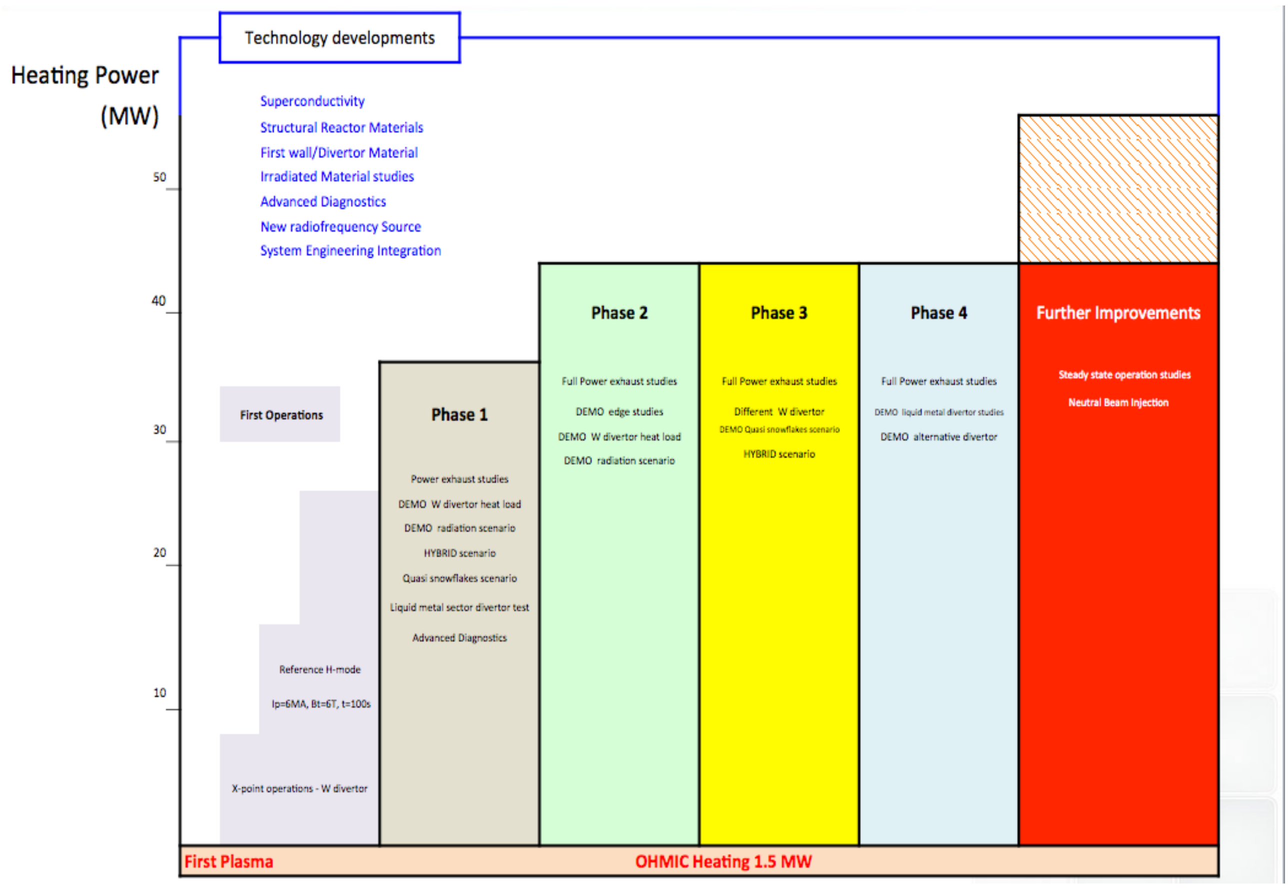


Figure 3. . Schematic planning of the DTT operations.

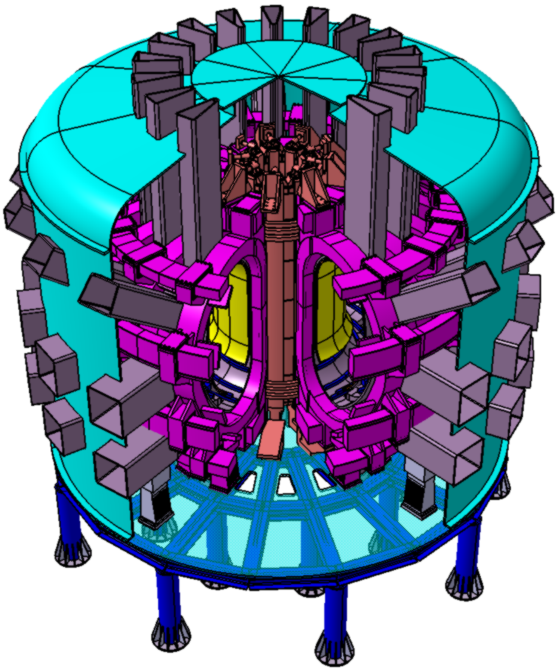


Figure 4: View of the DTT machine: the basic toroidal machine is entirely contained inside a cryostat vessel, which provides the vacuum for the superconducting magnets.

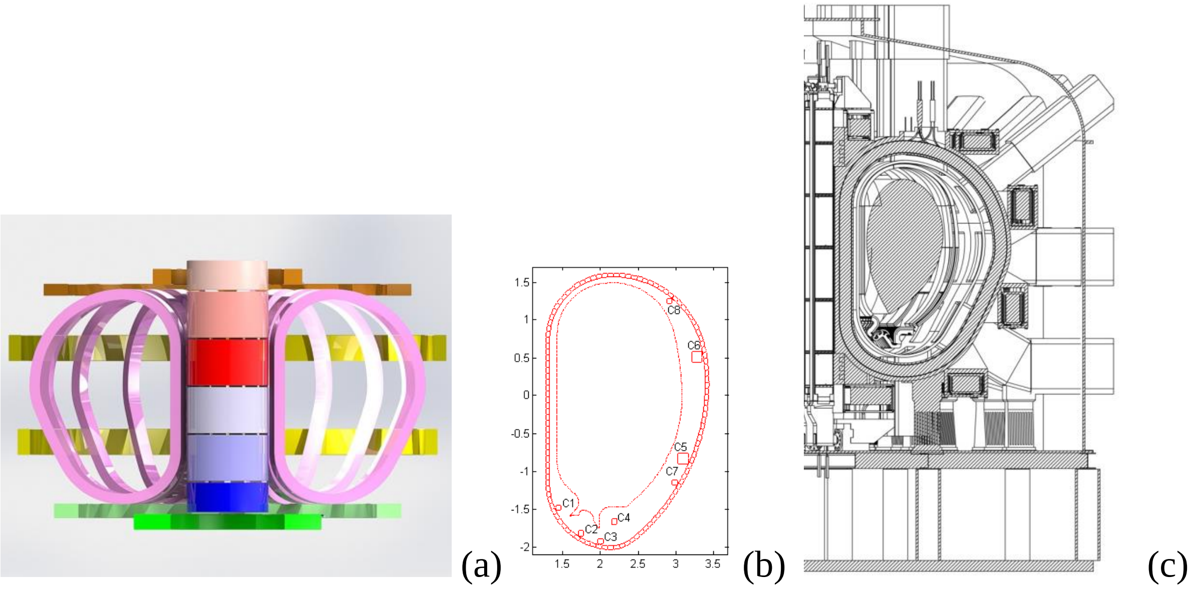


Figure 5. The magnet system of DTT: a) artistic view, with the complete TF, CS, and PF coils system; b) location of in-vessel copper coils; b) schematic view of superconducting magnets and main structures.

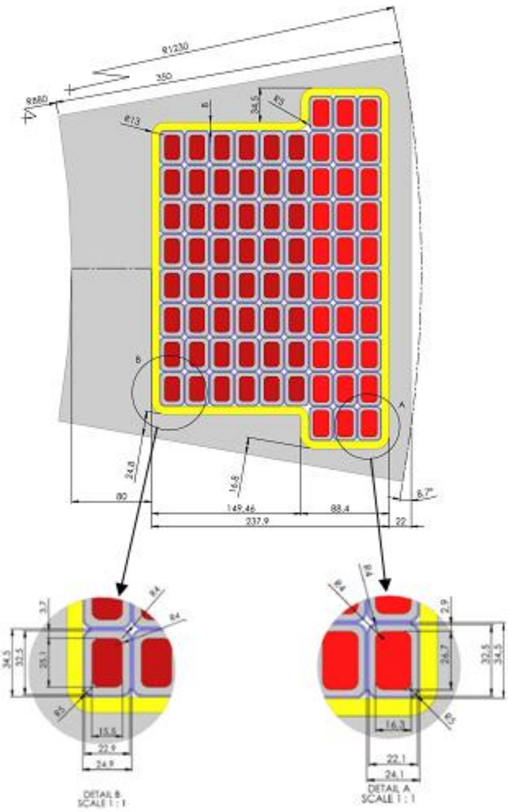
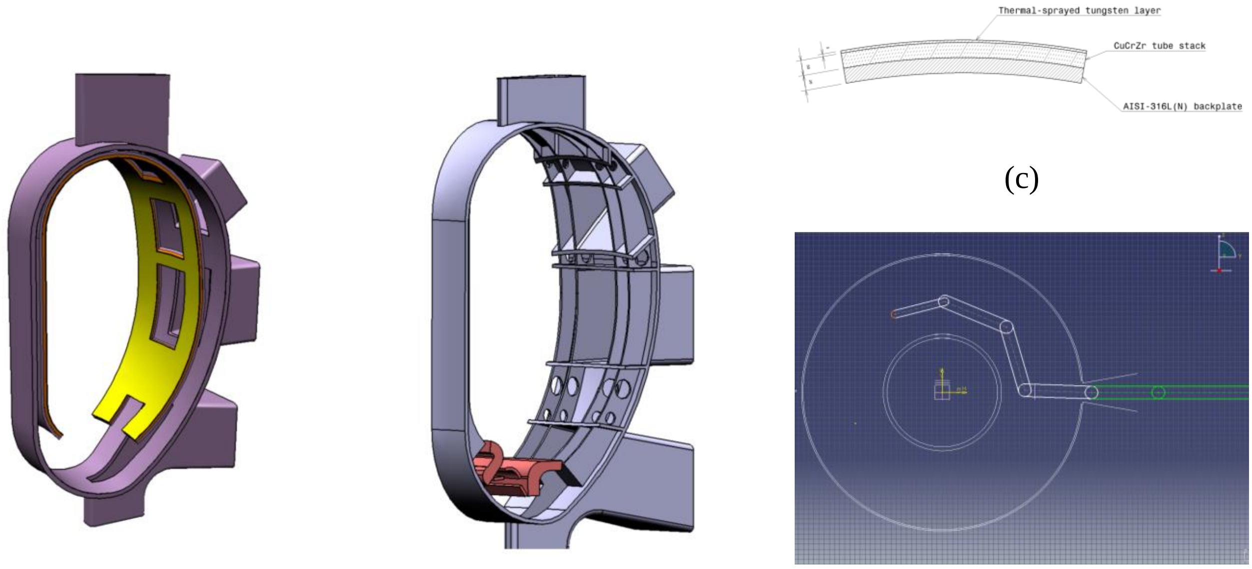
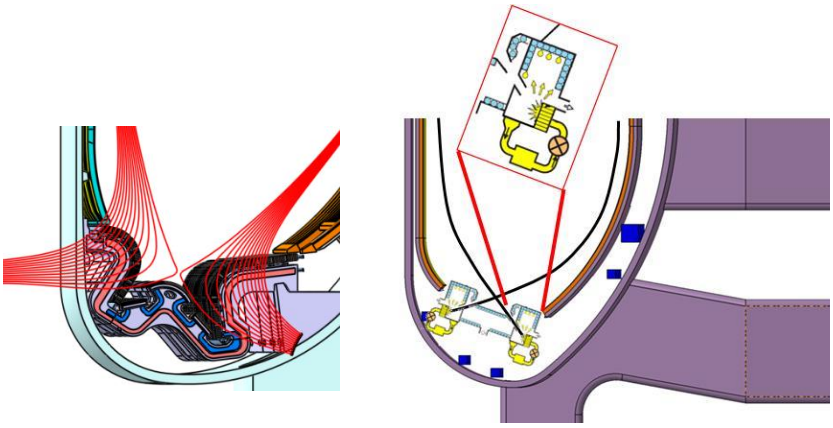


Figure 6. The DTT TF winding pack and details of high (right) and low field side conductors.



(a) (b) (c) (d)

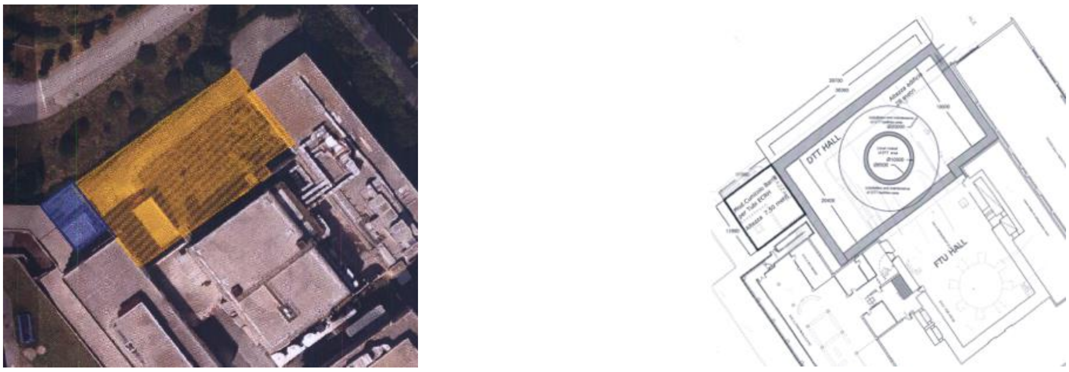
Figure 7.. DTT vacuum vessel (VV) and first wall (FW): a) 3D view showing the 5 access ports per sector; b) 3D view of the FW support structure; c) details of the FW layers; 4) remote handling system.



(a) (b)

Fig. 4.4: a) A possible tungsten divertor, compatible with both the SN and SF configurations; b) Liquid lithium box divertor.

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



(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) design of the new hall and the present FTU hall.

	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
<n> ($10^{20}m^{-3}$)	0.9	0.9	1.0	0.85	1.0	0.9	0.9	0.8	1.2	4.5	1.72
<n>/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
<T> (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)	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)	32	44	60	40	100	290	22	23	5	125	125
Pulse Length (s)	≈ 20	≈ 6	??	≈ 6	400	7000	100	1000	5	3	100

Table I: Machine comparison

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	-	-	-	-

Table II: 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).

	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

Table III: Main components of the real time control system of DTT.

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