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Preliminary study on alternative magnetic layout (AML) for tokamaks reactors in the TRUST project framework

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ABSTRACT

The main aim of this study is to explore a conceptual Alternative Magnetic Layout (AML) for the future nuclear fusion reactor based on the Tokamak concept. The proposed concept is eventually applied as a large down scaled study for the new academic tokamak Tuscia Research University Small Tokamak (TRUST), under construction at University of Tuscia (UNITUS). In this proposal the Central Solenoid (CS) placed around the Toroidal Field (TF) coils providing a relevant reduction in the reactor radial build. In order to compensate the increase of the toroidal filed on the coils, that it is intrinsic in such proposal, High Temperature Superconductor (HTS) material is foreseen for the TF coils. An overview of the proposed design and a possible plasma scenario is presented showing the AML advantages and disadvantages. The electromagnetic and structural characterizations for a preliminary design of the TF coils system is included in this paper to prove the engineering feasibility of the solution. Moreover, the involvement of the AML design in the TRUST project framework is presented. TRUST is a flexible and low-cost university-class experiment, to train the next generation of fusion engineers and physicists. Beyond its academic goal, TRUST is designed to allow easy replacement of the plasma facing components and testing innovative technologies (i.e., meta-materials) and materials including a feasible upgrade the magnetic system from copper to HTS.

1. Introduction

The development of nuclear fusion as a practical energy source depends heavily on advancements in magnetic confinement technologies and reactor compactness. Fusion reactors, such as the European DEMOnstration Power Plant (DEMO) [1], aim to bridge the gap between experimental reactors, such as ITER [2], and full-scale commercial fusion reactors, with the goal of delivering net electricity and enhancing reactor efficiency. However, significant challenges persist, particularly in the areas of magnetic confinement and reactor compactness. As outlined in [3,4], while high magnetic fields can theoretically improve plasma confinement efficiency and enable a more compact reactor design, practical engineering limitations impose strict boundaries on this approach. One of the primary obstacles in realizing a compact, high-field tokamak lies in the structural and thermal demands on the magnetic confinement system. Conventional superconductors, such as Nb_3Sn, have critical limitations at higher fields, necessitating large and complex cooling systems to prevent thermal degradation and quenching. Although HTS offer a pathway to support magnetic fields beyond 20 T, their mechanical and structural reliability at such intensities remain challenging. The stresses induced by higher magnetic fields lead to increased loads on TF and PF coils, complicating the design of structural components [5]. Moreover, compact designs introduce further engineering constraints related to neutron shielding and heat dissipation. Reducing the spatial requirements of the reactor limits the available volume for neutron shielding materials, increasing exposure of critical components to neutron flux and affecting their longevity. Simultaneously, compact designs exacerbate thermal loads on plasma-facing components, requiring advanced materials and innovative heat exhaust solutions. Tungsten-based components, for example, have emerged as critical candidates for handling extreme thermal environments while maintaining structural integrity over extended operational cycles [6].

Different possible solutions are under investigation to reduce the

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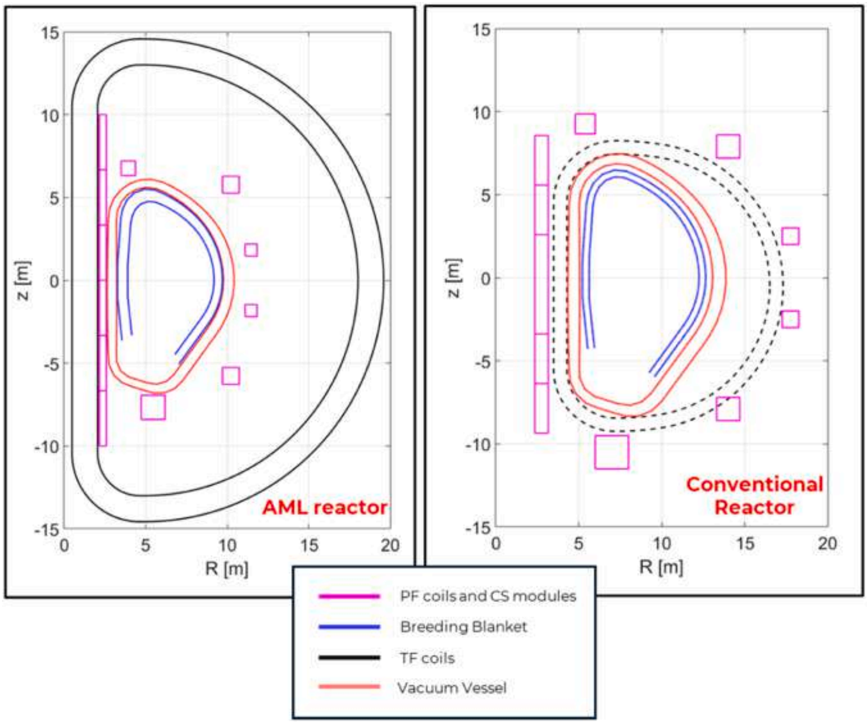

Fig. 1. Comparison between the AML design and a conventional CFETR/DEMO like reactor. The TF coils shape follows the constant tension d-Shape as explained in Section 4.

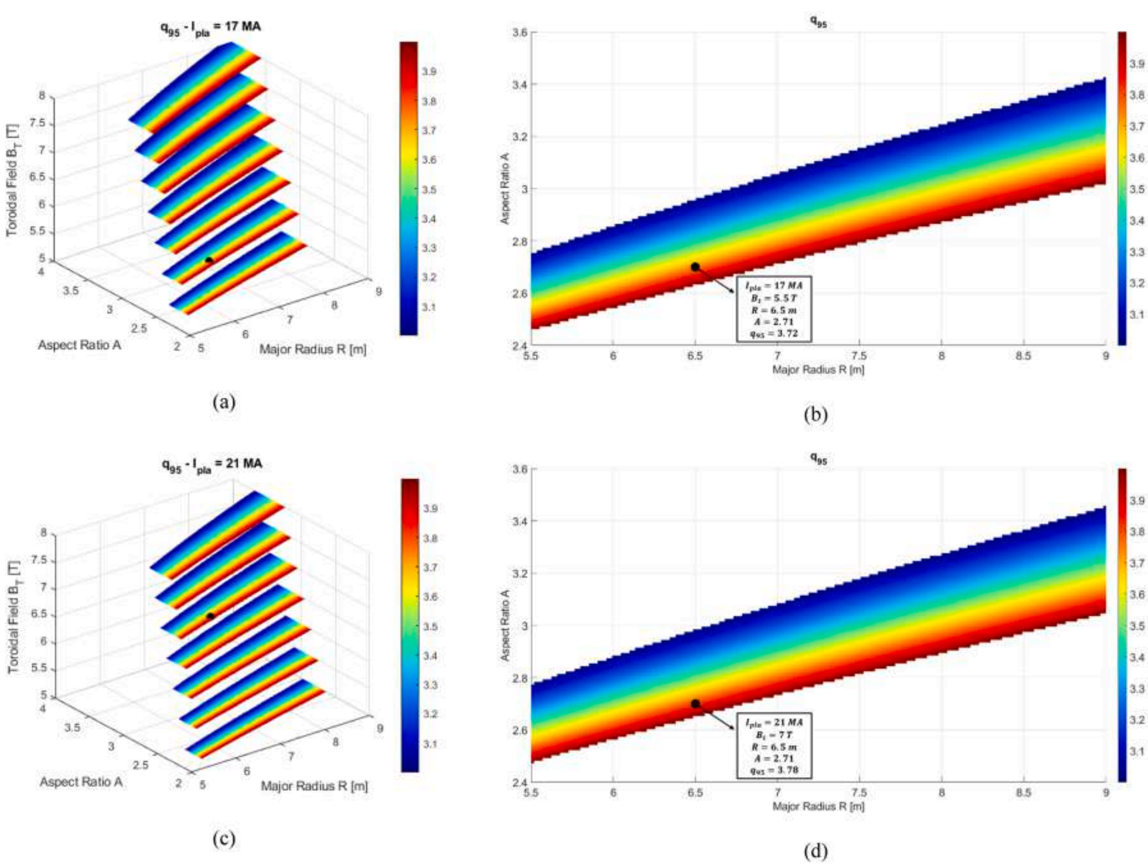

Fig. 2. The q_{95} variation as function of R and A plotted varying the reference plasma current I_{pla} and the reference B_t. The black point represents the possible configuration chosen for the AML design with co-ordinates $R = 6.5\ m$, $A = 2.71$ and fixed for each I_{pla} and B_t considered. In (a) and (b) a configuration with $I_{pla} = 17 MA$ and $B_t = 5.5\ T$ is illustrated. In (c) and (d) a configuration with $I_{pla} = 21\ MA$ and $B_t = 7\ T$ is showed.

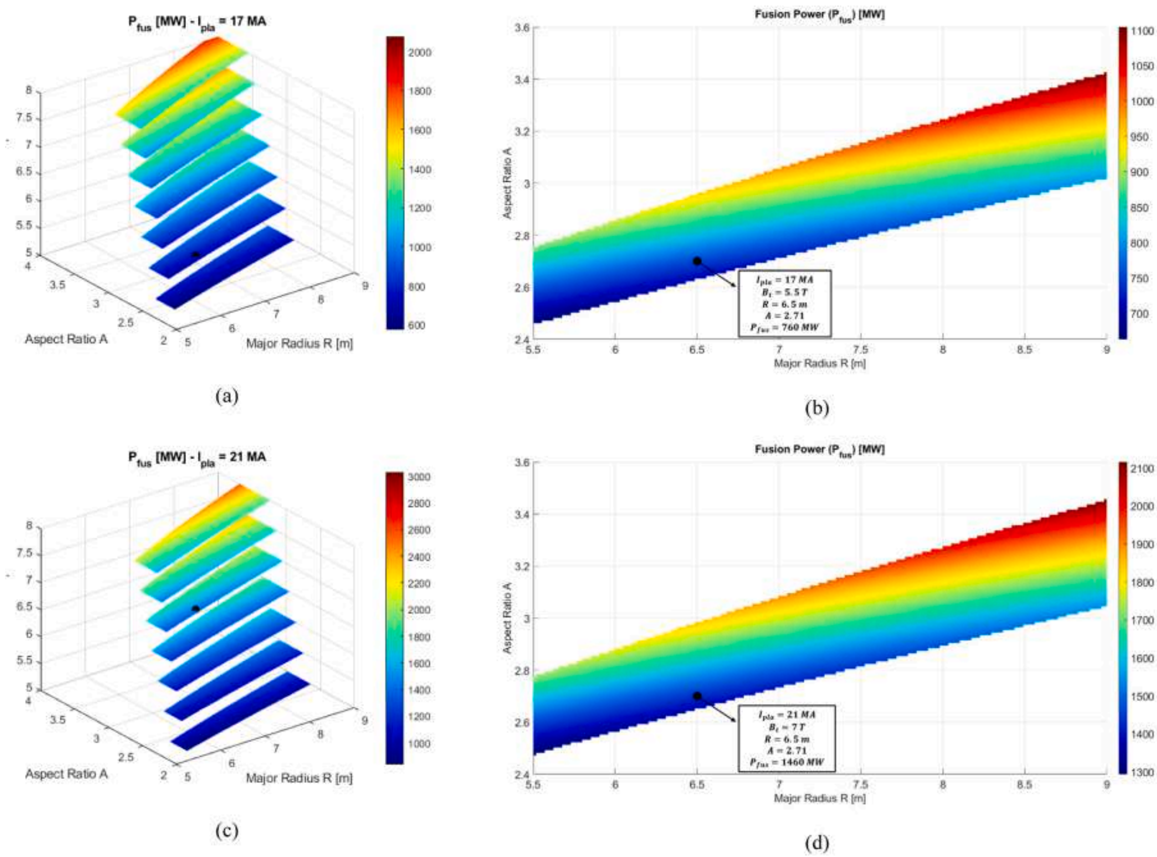

Fig. 3. The $\boldsymbol{P_{fus}}$ variation as function of $\boldsymbol{R}$, $\boldsymbol{A}$ plotted varying the reference plasma current $\boldsymbol{I_{pla}}$ and the reference $\boldsymbol{B_t}$. The black point represents the possible configuration chosen for the AML design with co-ordinates $\boldsymbol{R} = 6.5\ \boldsymbol{m}$, $\boldsymbol{A} = 2.71$ and fixed for each $\boldsymbol{I_{pla}}$ and $\boldsymbol{B_t}$ considered. In (a) and (b) a configuration with $\boldsymbol{I_{pla}} = 17\boldsymbol{MA}$ and $\boldsymbol{B_t} = 5.5\ \boldsymbol{T}$ is illustrated. In (c) and (d) a configuration with $\boldsymbol{I_{pla}} = 21\ \boldsymbol{MA}$ and $\boldsymbol{B_t} = 7\ \boldsymbol{T}$ is showed.

overall size of a future fusion reactor. Compact high-field tokamaks, such as SPARC [7], wants to demonstrate the potential to overcome the aforementioned challenges by leveraging advancements in HTS technologies. Commonwealth Fusion Systems (CFS), in particular, is trying to develop the magnetic and superconducting technology, which offers superior current-carrying capacity and quench detection capabilities compared to conventional superconductors. These possible innovation aims to enable the generation of the intense magnetic fields necessary for plasma confinement while maintaining manageable structural and thermal stresses within the reactor core [8,9]. Spherical tokamaks represent another promising approach to achieving compact fusion. Devices such as the National Spherical Torus Experiment (NSTX) [10] and the UK's STEP [11] initiative utilize low-aspect-ratio configurations to improve plasma stability and achieve compact designs without compromising performance. However spherical tokamaks face unique challenges, including increased stresses on the central column [12] and limited space for neutron shielding. Despite these challenges, their compact nature and efficiency make them strong candidates for commercial fusion energy systems [13,10]. China has also emerged as a leader in fusion research, making significant strides in the development of compact tokamak reactors. The Chinese Fusion Engineering Test Reactor (CFETR) represents the next step in their roadmap, designed to bridge the gap between ITER and future DEMO reactors. CFETR aims to achieve steady-state operation with fusion power outputs of 200 MW in its initial phase, scaling up to over 1 GW in later stages. These developments underline China's commitment to advancing compact fusion reactor technology and highlight the potential of their designs to address challenges related to size, efficiency, and scalability [14].

In this context, the aim of this study is to explore an Alternative Magnetic Layout (AML) for the future nuclear fusion reactor. Another fundamental issue, on the road map for the achievement of the nuclear fusion as a reliable energy source, it is the setting up of a solid University system capable to prepare at the best the future generation that will be necessary to further develop the future reactor and to maintain them. Bearing in mind this idea the proposed AML is even applied as a conceptual scaled study for the new academic tokamak Tuscia Research University Small Tokamak (TRUST), under construction at University of Tuscia (UNITUS) [15]. The alternative magnetic layout has the CS placed around the TF coils central column providing a relevant reduction in the reactor radial build as explained in Section 2. To provide the required increase of the toroidal field on the coils HTS material is foreseen for the TF magnet. The main plasma parameters and a preliminary plasma scenario are reported in Section 3. A preliminary electromagnetic and structural characterization for the TF coils is performed to prove the feasibility of this solution and reported in Section 4. Finally in Section 5, the implication of the TRUST project in the development of AML reactor is explained.

2. AML design

The transition from conventional to the studied conceptual AML configurations introduces significant changes in the overall reactor design, with the primary objective of achieving a reduced radial build while maintaining performance metrics critical to magnetic confinement and plasma stability. The AML reactor (illustrated in Fig. 1, compared with a standard possible future reactor) demonstrates a compact arrangement of the PF coils and structural components, achieved by repositioning the CS around the central column of the TF coils. Indeed, it is also possible to notice in Fig. 1 as the overall AML reactor size remains comparable to the conventional one (except for the total

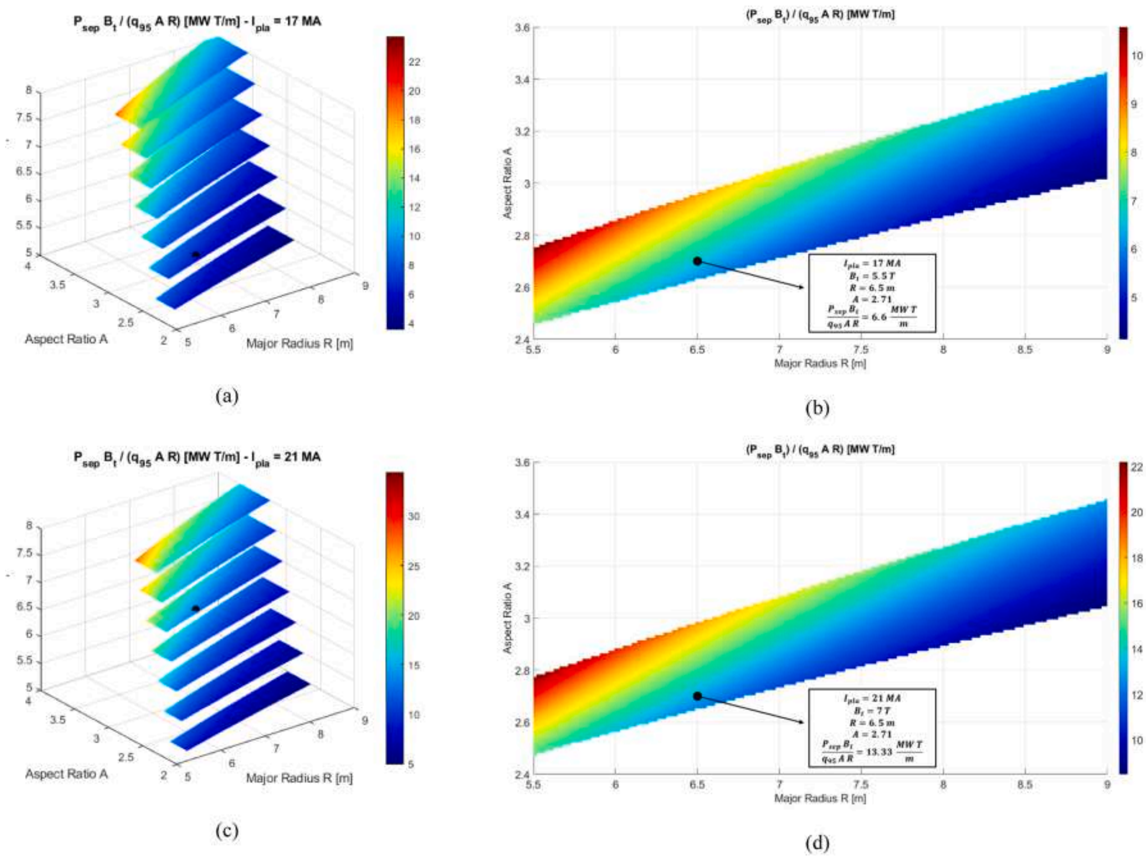

Fig. 4. The $\frac{P_{sep}\ B_t}{q_{95}\ A\ R}$ variation as function of $\boldsymbol{R}$, $\boldsymbol{A}$ plotted varying the reference plasma current $\boldsymbol{I_{pla}}$ and the reference $\boldsymbol{B_t}$. The black point represents the possible configuration chosen for the AML design with co-ordinates $\boldsymbol{R} = 6.5\ \boldsymbol{m}$, $\boldsymbol{A} = 2.71$ and fixed for each $\boldsymbol{I_{pla}}$ and $\boldsymbol{B_t}$ considered. In (a) and (b) a configuration with $\boldsymbol{I_{pla}} = 17\boldsymbol{MA}$ and $\boldsymbol{B_t} = 5.5\ \boldsymbol{T}$ is illustrated. In (c) and (d) a configuration with $\boldsymbol{I_{pla}} = 21\ \boldsymbol{MA}$ and $\boldsymbol{B_t} = 7\ \boldsymbol{T}$ is showed.

Table 1
Main parameters for the AML reactor design point.

R [m]	A	I_{pla} [MA]	B_t [T]	q_{95}	$<n>/n_{GW}$	κ_{95}	δ_{95}	li	β_p	H_{98}
6.5	2.71	21	7	3.78	0.87	1.67	0.36	0.8	0.62	0.9

Pulse length [s]	P_{fus} [MW]	Volume [m^3]	P_{sep} [MW]	$\frac{P_{sep}\ B_t}{q_{95}\ A\ R}$ [MW T/m]	P_{add} [MW]
7200	1460	1037	127	13.33	70

height that increase for the proposed design). In fact, despite the compact radial build, the larger TF coils footprint in the AML concept impacts on the reactor main dimensions.

Nevertheless, the rearrangement of the CS and TF coils is a strategic design choice that improve the radial space utilization on the central column, while maintaining critical magnetic performance. By moving the TF coils central column toward the reactor axis (radial coordinate decreased) and the CS toward the plasma (radial coordinate increased), the AML configuration achieves a more compact radial build design. In fact, the target flux ψ_{CS} could be achieved with a thinner CS due to its quadratic dependence with respect to the radial coordinate of its centroid [16]:

$$\psi_{CS} = \frac{\pi B_{CS}}{3}\left(R_i^2 + R_i R_e + R_e^2\right) \tag{1}$$

Where B_{CS} is the maximum magnetic field, R_i is the CS inner radius and R_e is the CS outer radius. This flux dependency allows the CS to maintain effective inductive drive even with a reduced footprint, ensuring adequate plasma current generation and duration.

The displacement away from the plasma of the TF coils, however, reduces the toroidal magnetic field B_T due to its inverse dependence on the radial position:

$$B_t(R) = \frac{\mu_0\ N_{TF}\ I_{TF}}{2\pi R} \tag{2}$$

Where μ_0 is the magnetic permeability, N_{TF} is the number of TF coils, I_{TF} is the current in the single TF coil and R is the radial coordinate. To maintain a fixed toroidal field, the TF coils current must be increased and as a result the current density in the magnet will also increase [5]. This requirement underscores the critical role of HTS in the AML design. HTS materials enable the TF coils to sustain high current densities and magnetic fields while maintaining manageable coil dimensions. Unlike Low Temperature Superconductors (LTS), HTS materials could operate at higher magnetic field intensities, higher current density and higher temperatures increasing the electromagnetic load acting on the magnets [17]. From Eqs. (1) and (2) it comes immediately out that, since $B_t \approx 1/R$, whilst $\psi_{CS} \approx R^2$, a clear reduction of the machine dimension can be obtained, whilst maintaining the same engineering parameters.

A clear implication of the proposed AML is the necessity of different assembly strategies, respect to conventional tokamak [18], to place the

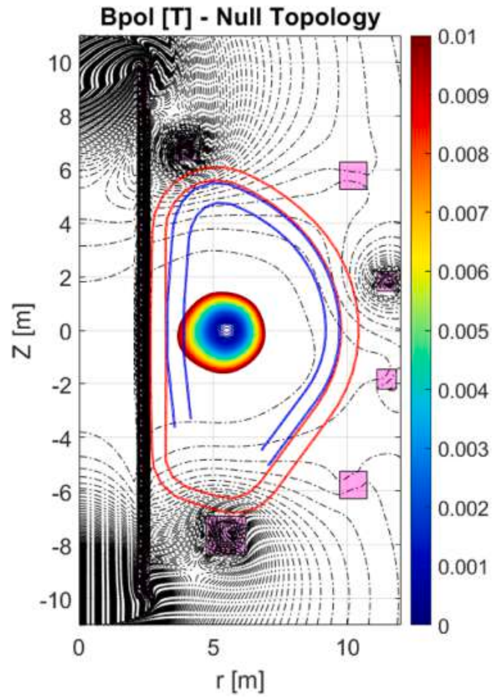

Fig. 5. Poloidal field null generated to reach the plasma breakdown conditions inside the vessel.

CS modules and the PF coils inside the TF coils:

- A prominent strategy involves the use of demountable electrical joints for the TF coils, which theoretically allow modular assembly and simplify maintenance. The need of electrical joint may represent a showstopper for the AML configuration, in fact different designs and solutions to perform reliable HTS joints are under study for a future fusion reactor [19–21] but no one reached the maturity level needed yet. Assuming a qualified technology to realize reliable electrical junction between the HTS conductors, the TF coils inner legs would be mounted in the tokamak pit separately respect the TF coils outer legs. In the aforementioned preliminary solution, the first components assembled in the tokamak pit is the CS modules, realized and wounded before the assembly phase. Subsequently, the Vacuum Vessel (VV) sectors, including the In-Vessel components and the Blanket, are located and assembled in the tokamak pit around the CS modules. In the AML configuration, the PF coils are considered structurally supported by the VV and assembled directly on the VV

Table 2
Main parameters for the plasma scenario.

	BD	SOF	EOF
I_{pla} [MA]	0.5	21	21
R [m]	5.02	6.50	6.50
a [m]	1.03	2.34	2.35
B_t [T]	9.15*	7.07	7.07
κ	1.08	1.75	1.72
δ_{low}	−0.001	0.26	0.28
β_{pol}	0.10	0.62	0.62
li	0.88	0.73	0.73
q_{95}	20.2	3.52	3.53
Volume [m^3]	112.3	1105.6	1100.9
$\psi_{boundary}$ [Vs]	290.3	150.26	- 144.76

* The magnetic field is evaluated at the plasma centroid which is at higher magnetic field respect the target R for the SN.

Table 3
CS modules currents during the plasma scenario.

	Current @BD [MA• Turns]	Current @SOF [MA• Turns]	Current @EOF [MA • Turns]
CS1	44.861	2.981	−1.712
CS2	44.861	8.328	−33.240
CS3	44.861	7.655	−34.380
CS4	44.861	7.131	−34.460
CS5	44.861	10.260	−35.050
CS6	44.861	4.177	−18.010

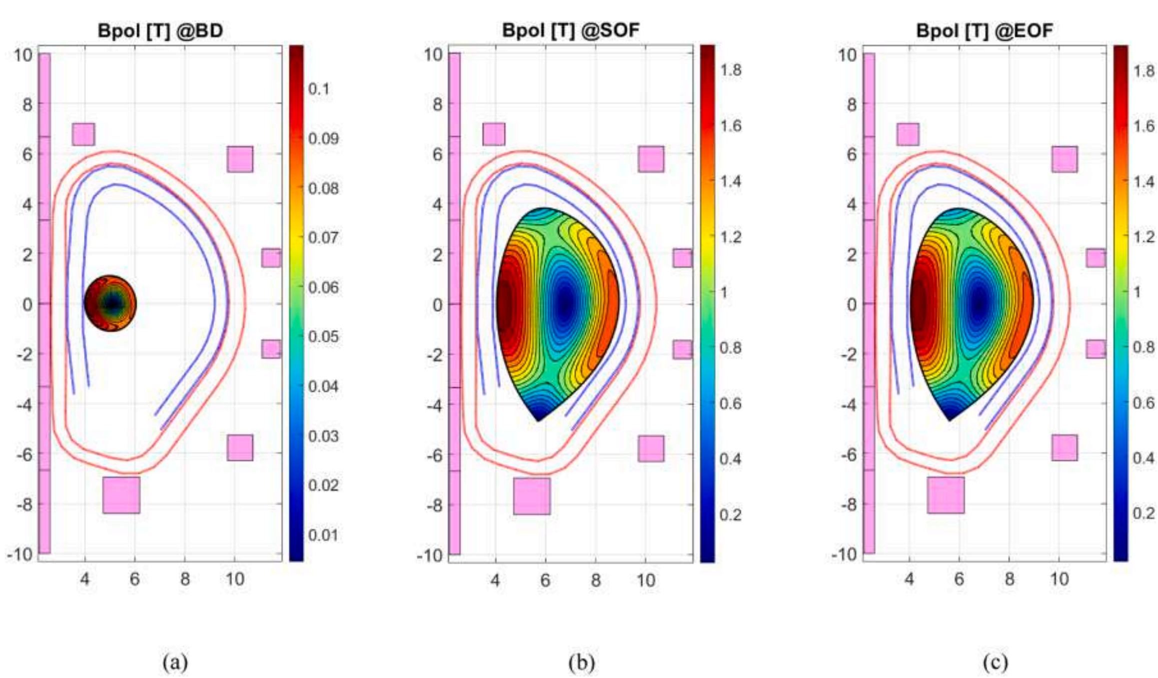

Fig. 6. Different plasma equilibrium for the three time instant considered in the plasma scenario. Respectively the equilibrium (a) at the breakdown, (b) at the Start of Flattop and (c) at the End of Flattop are reported.

Table 4
PF coils currents during the plasma scenario.

	Current @BD [MA• Turns]	Current @SOF [MA• Turns]	Current @EOF [MA • Turns]
PF1	4.320	12.110	−1.640
PF2	3.795	−4.049	−7.289
PF3	−1.215	−2.080	−2.035
PF4	1.445	−2.034	−2.442
PF5	0.471	−8.023	−9.931
PF6	6.937	15.120	7.372

Table 5
Main parameters for the analysed TF coil configuration.

I_{TF} [MA]	$B_{t,max}$ [T]	$R_{inb-ext,TF}$ [m]	A_{WP} [m^2]	J_{WP} [MA/m^2]	N_{TF}	$F_{z,TF}$ [MN]	$F_{r,TF}$ [MN/m]
19	22	2.07	0.592	32	12	379	209

sectors during this phase. Finally, the TF coils inner legs are located in the tokamak pit and connected through the electrical joint to the TF coils outer legs closing the d-shape.

- An alternative strategy configuration could benefit from the innovative assembly strategies inspired by projects like the Volumetric Neutron Source (VNS). As part of the VNS project, the feasibility of an in-situ winding process will be explored, which could serve as a foundation for AML tokamak assembly [22]. In this approach, the vacuum vessel and PF coils are pre-assembled, and the TF coils are wound directly around these completed structures. This eliminates the need for prefabricated TF coil modules and could simplifies alignment and integration. Indeed, the needs of electrical joint on the TF coils (as in the aforementioned solution) could bring to alignment problems during the assembly between central column and external legs. Moreover, such type of fabrication procedure might result in an easier integration of the components inside the TF coils. The in-situ winding approach could be also explored for the manufacturing and assembly of the CS modules and for the PF coils. In this condition the TF coils is considered manufactured and then pre-assembled in the tokamak pit. Subsequently, the in-situ winding for the CS modules is foreseen. Finally, the VV sectors are assembled and the PF coils are wound in-situ on the supporting structures. The in-situ winding process feasibility, as for the TF coils as for the PF coils, depends on several factors and should not be considered as a certain solution for the AML concept. For instance, the winding machine must be accommodated with the available space, especially inside the tokamak central column and the HTS cable technology must be compatible with the coils bending radius.

A clear solution for the assembly of proposed AML design it is not yet provided and it is currently under investigation, reflecting a broader issue faced by all future fusion reactors under development. For instance, ITER has encountered significant assembly difficulties, including defects identified in major components like thermal shields and vacuum vessel sectors, which have impacted its construction timeline [23]. Similarly, CFETR is facing challenges related to complex interfaces and the integration of physical and engineering requirements, necessitating extensive research and development efforts [24]. In addition, the preliminary AML design includes increased complexity due to the possible size for the TF coils. The managing of the large TF coils during the manufacturing phase and the high energy in the superconducting cables during the operation are critical and challenging aspects of the proposed concept. At the same time, the AML design could present a significant mechanical advantage by reducing the structural loads on the top and bottom curvature of the TF coils and therefore on the possible electrical joints of the TF coils. In principle, the design allows taller TF coils and geometrically optimized to reduce their interaction with the poloidal magnetic field. This tailored shaping of the TF coils could mitigate the electromagnetic forces generated by the coupling with poloidal fields. A preliminary quantitative evaluation on the mechanical stress of the TF coils and its implications for the overall AML configuration are discussed and illustrated in Section 4. TF Coils Characterization Moreover, the TRUST project is proposed as testbed experiment to investigate possible downscaled engineering solutions.

These examples underscore the intricate nature of assembling advanced fusion reactors and the necessity for ongoing research to develop effective solutions. Solving this issue could lead to advantages not only related to the assembly and maintenance of the reactor, but also benefits related to its operation and performance. For the proposed AML configuration a concrete example is the possibility to design the PF coils closer to the plasma compared to conventional tokamak configurations. This strategic arrangement enhances the magnetic coupling efficiency, enabling the PF coils to generate stronger shaping and stabilizing fields with lower currents. The shaping field (B_{PF}) produced by the PF coils is inversely proportional to the distance from the plasma, as expressed by:

$$B_{PF} \approx \frac{\mu_0 \, I_{PF}}{2 \, \pi \, d_{PF}} \tag{3}$$

Where I_{PF} is the current in the PF coils, and $d_{PF}$is the distance respect the plasma. The PF coils position shown in Fig. 1 clearly indicates that reducing d_{PF} increases B_{PF} for a given plasma current, thereby improving the precision of plasma shaping and enabling better control of

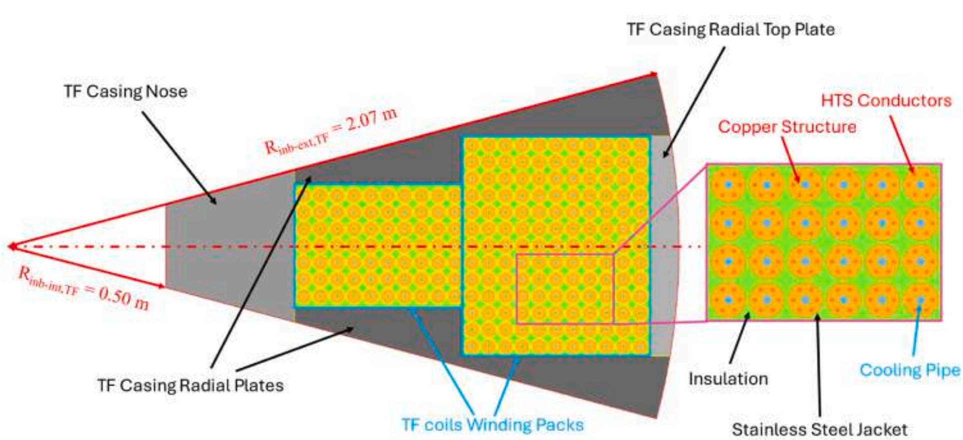

Fig. 7. TF coils design proposed for in the AML design. The preliminary concept includes a stainless-steel casing divided in four components. The HTS cables are encapsulated in an insulating material and bundled in Windings Packs.

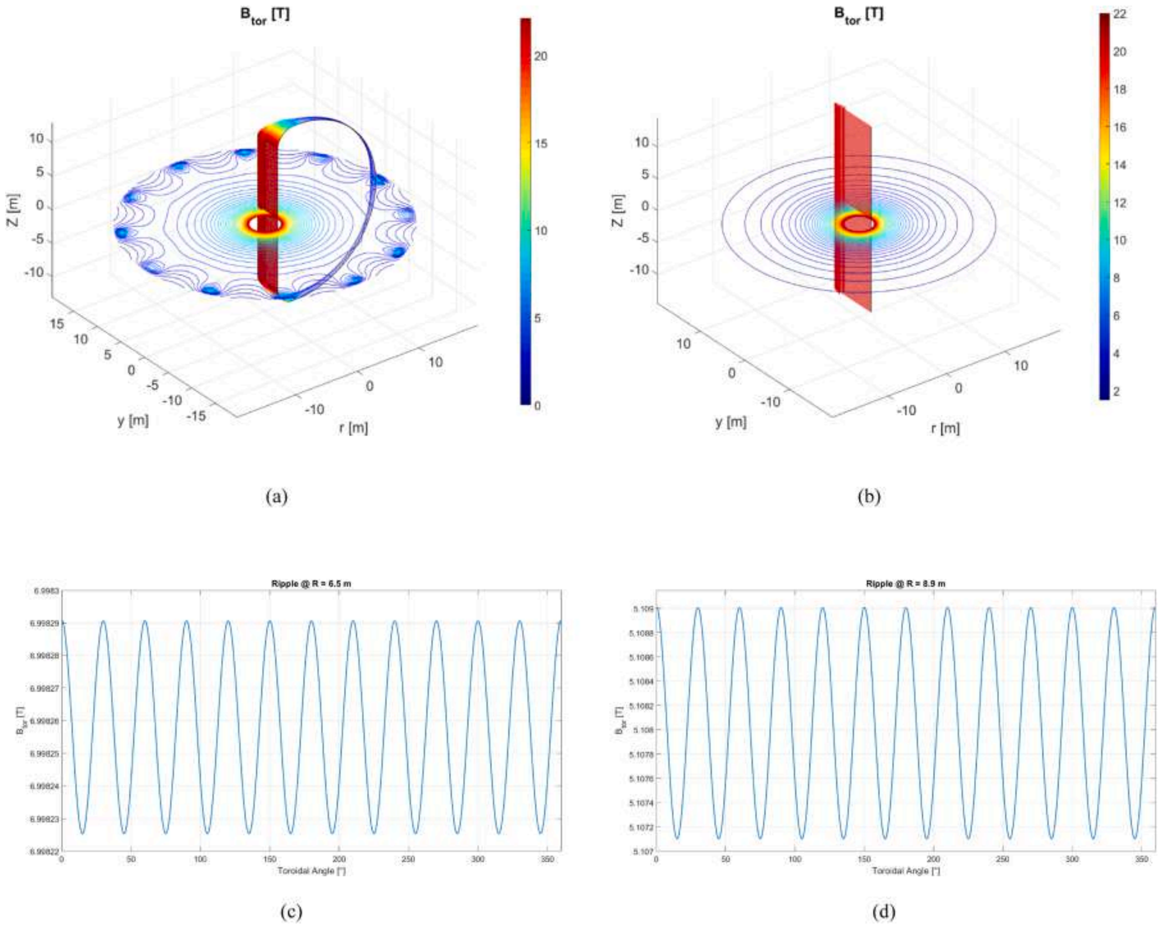

Fig. 8. The toroidal field produced by the magnets is reported in (a) compared with the magnetific field produced by infinitely long straight conductor (b). The comparison between the magnetic ripple at the plasma axis $R = 6.5$ m(c) with the magnetic ripple on the plasma edge $R = 8.9$ m (d).

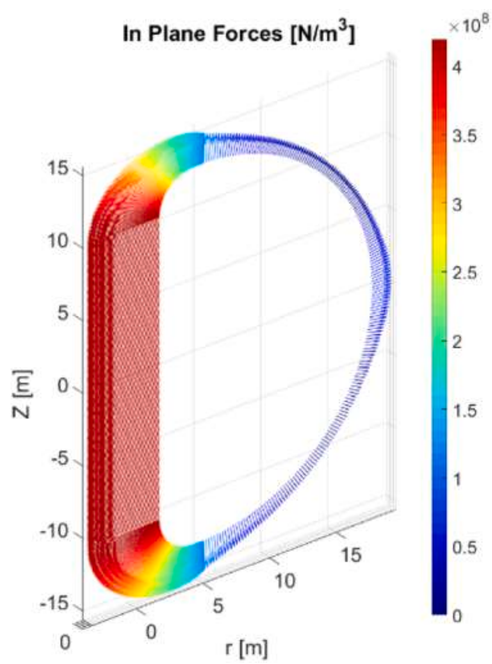

Fig. 9. In plane forces distribution along the TF coil reference line.

elongation (κ) and triangularity (δ). Consequently, there is a noticeable space for improving the position of the PF coils in order to minimize their current density. Additionally, the closer positioning of the PF coils enhances the system's ability to counteract vertical displacement events (VDEs). The stabilizing vertical force ($F_{vertical}$) depends on the coupling efficiency, which scales with d_{PF}, and is given by:

$$F_{vertical} \approx \frac{\partial M_{PF}}{\partial z} \delta I_{PF} \tag{4}$$

Where M_{PF} is the mutual inductance between the plasma and the PF coils, and δI_{PF} is the current variation in the coils. Smaller d_{PF} increases $\frac{\partial M_{PF}}{\partial z}$, enabling a stronger vertical stabilization force acting on the plasma considering the same VV radial penetration [25]. Therefore, considering the same delay due the typical time constat of the VV, it is possible to reach a higher value of $F_{vertical}$ given the same variation in current. These control advantages are further supported by the AML's compact design, which minimizes energy losses and operational costs, as detailed in the subsequent sections. These control advantages are inherently linked to the AML design, in which the PF coils are directly supported to the vessel structure. A dedicated thermal shield surrounding the VV will be essential to thermally insulate the PF coils, as foreseen the under-construction tokamaks such as DTT [26] and ITER [27]. Moreover, additional shielding material could be integrated around the poloidal field coils and their casings to enhance thermal insulation between the high-temperature vacuum vessel and the cryogenic environment required by the coils.

Finally, also the CS design for the AML concept presents challenges due to the high toroidal magnetic field acting on it, which can reach up to 20 T. Although the mechanical support structure and vertical

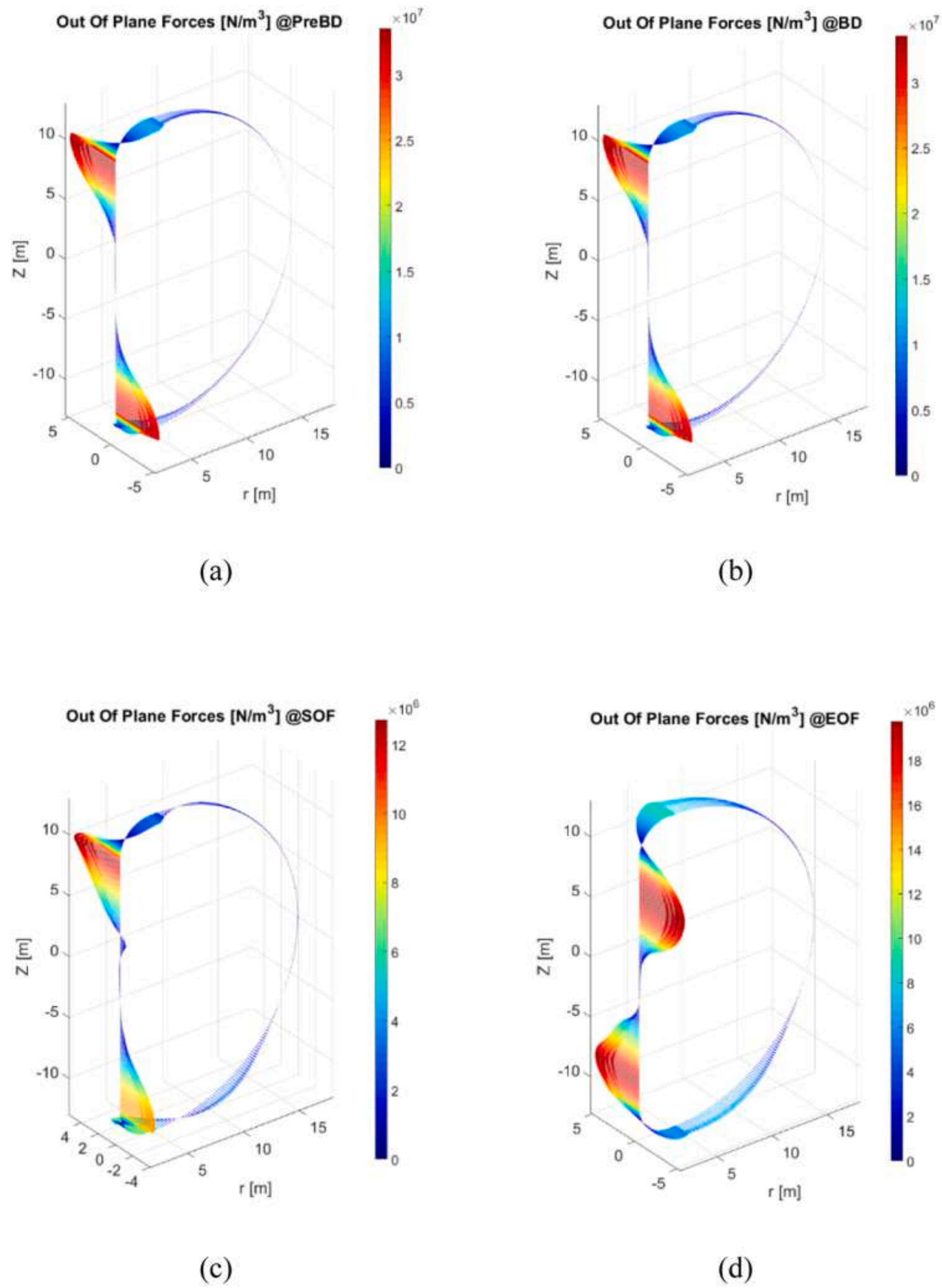

Fig. 10. Out plane forces distribution along the TF coil reference line for plasma scenario instant studied: (a) pre-breakdown phase, (b) breakdown phase, (c) start of flattop instant and (d) end of flattop instant.

precompression required for the CS are expected to be similar to those of conventional tokamaks [28], any misalignment between the CS conductors and the toroidal field can result in significant electromagnetic forces. While these forces may be manageable within the pancake windings of the CS (where the angle between the current and the toroidal field is relatively small) they could pose critical issues at the level of electrical joints and current feeders. These components, which surround the CS and connect it to the power supply and cooling systems, may experience strong interactions with the toroidal field. This interaction could represent a potential showstopper, making the optimization of the current feeder routing essential to minimize electromagnetic coupling with the toroidal field. Partially, also the conventional tokamaks have already dealt with such type of problems [29]. The engineering design for the CS is still under investigation and is therefore not included in the preliminary concept proposed.

3. Plasma parameters and scenario

A preliminary investigation of the operational conditions for the AML reactor is reported in this section. The AML plasma parameters are explored to obtain high-performance plasma operation within a compact and efficient tokamak layout. Specifically, a preliminary 0D model was used according to the ITER scaling and according to the benchmark proposed for JET and DTT [30]. The parameters are evaluated to balance confinement, stability, and engineering constraints while supporting long-pulse operation. The analysis of operational plasma parameters for the AML reactor is developed by the evaluation of critical metrics such as q_{95}, fusion power (P_{fus}), and the normalized heat

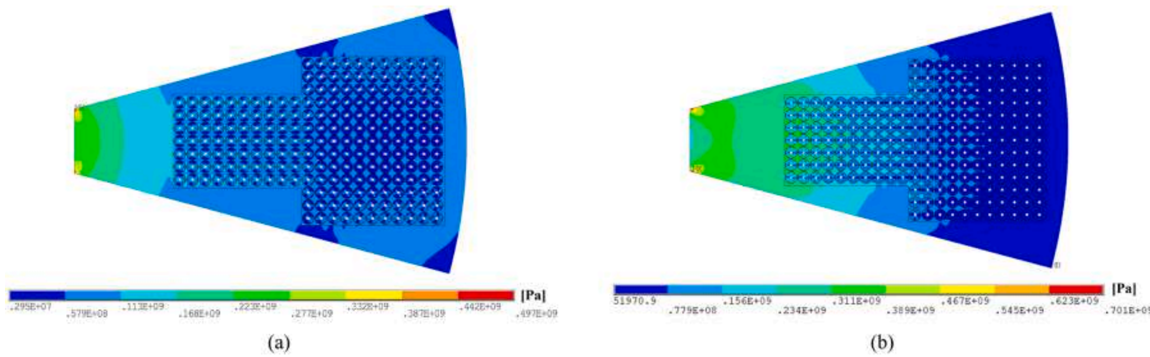

Fig. 11. Von Mises stress contour plot for the 2D FE model considering the in plane radial force for the (a) wegded solution and (b) bucked solution.

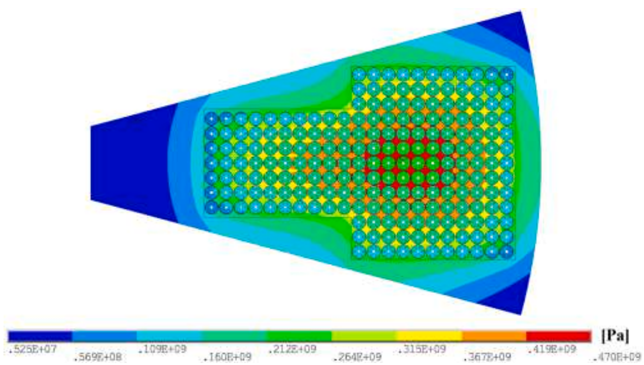

Fig. 12. Von Mises stress contour plot for the 3D FE model considering the in plane vertical force.

flux parameter $\frac{P_{sep}}{q_{95}}\frac{B_t}{A\ R}$.

The value of q_{95} was confined to the range $3 < q_{95} < 4$ (Fig. 2), which is critical for preventing magnetohydrodynamic (MHD) instabilities that bring to plasma off-normal events such as edge-localized modes (ELMs) and vertical displacement events (VDEs). These stability requirements are well-documented in tokamak research and have been validated in device like JET [31]. Within Fig. 2, as in the following Figs. 3 and 4, the black point in the sub-figures (c) and (d) represents the chosen values for R, A, I_{pla} and B_t as indicated in Table 1. For the design of the plasma scenario the following plasma shape parameters have been assumed: elongation $\kappa \sim 1.65$ and triangularity $\delta \sim 0.29$. These values align with those observed in high-confinement (H-mode) scenarios, providing robust equilibrium conditions that support high plasma performance [32]. Conservative assumptions were made for the Greenwald fraction ($f_{Green} \sim 0.87$) and the confinement enhancement factor relative to ITER scaling ($H_{98} \sim 0.9$) [33]. These values were selected based on operational regimes already achieved in JET, ensuring a realistic baseline for AML's performance projections.

Together to the plasma stability, a minimum fusion power output of $P_{fus} \geq 1400$ MW is considered for the AML design point. This parameter is essential for economic feasibility and meeting energy production goals. Fig. 3 shows the dependence of P_{fus} on R, A, B_t and I_{pla}. By reducing the main dimension for the reactor (R and/or A) to reach the same fusion power it is necessary to increase the plasma current and the toroidal magnetic field as clearly indicated by Fig. 3c and Fig. 3d, at lower plasma current and lower toroidal field the fusion power does not result sufficient fixing the R and A.

Also the power that flow through the separatrix (P_{sep}) was evaluated to address thermal loads on the divertor. This parameter, scaled by the ratio $\frac{P_{sep}}{q_{95}}\frac{B_t}{A\ R}$, provides indication for the heat flux to the divertor [34]. Fig. 4 highlights the parametric evaluation of this metric as a function of R, A, B_t and I_{pla}. Of course, moving the design point to smaller major radius and smaller aspect ratio involves the increase on the normalized heat flux parameter. The aim of this study is not focused on address in detail the heat flux mitigation on the divertor, but different studies are exploring solutions for reactor-class tokamak [35,36].

A possible design point is chosen to analyse its behavior in AML reactor showing a possible compact design respect a conventional tokamak reactor. The main parameters for the design point are indicated in Table 1:

A preliminary plasma scenario and a plasma equilibrium, based on

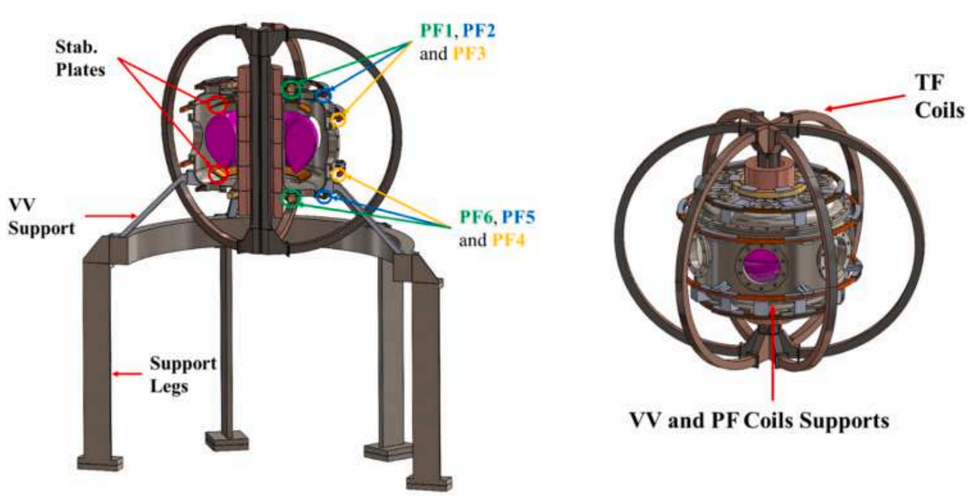

Fig. 13. TRUST conceptual design assembly including all the main components.

the chosen design point, are designed using MAXFEA code. The PF coils and CS modules position is maintained with a distance of at least 1.5 m respect to the plasma facing material. This space is mandatory to accommodate the Vacuum Vessel (VV) and the Breeding Blanket (BB) under relevant neutronics and electromagnetic load conditions as expected in a nuclear fusion reactor [37]. In addition, the PF coils centroid position and the CS modules extension is obtained to ensure the needed topology to get an optimised quasi null magnetic field during the plasma break down (BD) (Fig. 5).

An assisted BD is foreseen, indeed different solutions are under development for large scale tokamak as expected for the Divertor Tokamak Test (DTT) [38]. During the non-ohmic breakdown it is assumed no flux consumption. At the end of the breakdown phase a first limiter plasma is considered with CS modules at the maximum current. The equilibrium for the first limiter plasma is reported in Fig. 6 and the main plasma parameters obtained are report in Tables 2-4. The available flux at the start of flattop (SOF) is evaluated according to [39]:

$$\psi_{boundary\ @SOF} = \psi_{boundary\ @BD} - \left(\frac{1}{2}\ li\ \mu_o\ R\ I_{pla} + C_{ejima}\ \mu_0 R\ I_{pla}\right) \tag{5}$$

where the flux available at the SOF $\psi_{boundary\ @SOF}$ depends by the flux available at the breakdown $\psi_{boundary\ @BD}$ (total flux from the CS), the plasma normalized internal inductance li and the Ejima coefficient C_{ejima} assumed equal to 0.4 [39]. The plasma current rise is assumed equal to be 0.4 MA/s, leading to $I_{pla} \approx 21$MA in 51.25 s. The detailed evolution of the plasma shape needed to reach the target SN configuration is currently neglected.

A Single Null plasma configuration is considered as target for the AML reactor scenario. The resistive voltage drop during the flattop considering the plasma performance is equal to 0.041 V allowing to reach around 7117 s for the flattop duration.

4. TF coils characterization

To check the feasibility of the proposed AML design, a preliminary characterization for the TF coils system is provided. The constant tension d-Shape [40] is used as first design for TF coils to evaluate the overall dimension and the electromagnetic loads acting on the HTS conductors and on the structural materials. The main parameters for the TF coils are reported in Table 5.

From the target toroidal magnetic field (B_t), it is possible to derive the needed current flowing in the TF coils central column, and consequently the current flowing in the single TF coil I_{TF} through the Biot-Savart law (Eq. (2)). Given the CS modules geometry from the plasma equilibrium and scenario (Section 3), the TF coils HTS conductors and the structural material are accommodated in remaining available space in the central column. The proposed solution includes a TF coils cassette composed by a nose, two radial plates and a top plate as showed in Fig. 7. The TF coils Winding Packs (WP) are considered formed by Conductor on Round Core (CORC®) HTS cables [12,19] surrounded with insulation material. Inside each HTS cables the amount of copper needed to perform a safety discharge in case of quench is considered allowing to reach the hot spot temperature at the end of the fast discharge [41]. The limit time constant τ_{lim}, for the proposed TF coils preliminary design, is approx. equal to 30 s to avoid high stress condition on the VV. Considering 66 MA/m^2 of current density on the stabilizer copper and for each TF coil slightly >200 turns are assumed with an operating current around 90 kA, it is possible to sustain the coil discharge avoiding hot spot temperature in windings. The energy stored in the TF coil is approx. equal to 12 GJ with a voltage discharge of about 9 kV,with a similar order of magnitude expected in EU-DEMO [5]. The adopted procedure is similar to that proposed and well explained in different studies [42–44]. The proposed design is not intended to be an engineering optimized solution for the TF coils, but it should be considered as a preliminary feasibility study to for the proposed AML design.

The radial overall dimension for the central column is around 1.5 m located between $R_{inb-ext,TF}$ and $R_{inb-int,TF}$. The electromagnetic forces acting on the TF coil could be roughly estimated [3]:

$$F_z = \frac{1}{4}\ B_t\ R\ I_{TF}\ \ln\left(\frac{R_{out,TF}}{R_{inb-ext,TF}}\right) \tag{6}$$

$$F_r = \frac{1}{2}\ I_{TF}\ B_{t,max} \tag{7}$$

Where F_z (evaluated in N) and F_r (evaluated in N /m) are respectively the vertical and radial force acting on the TF coil, $R_{out,TF}$ is the outboard radial coordinate of the TF coil reference line and $B_{t,max}$ is the maximum toroidal field acting on the TF coils ($B_{t,max} = B_t \frac{R}{R_{inb-ext,TF}}$). These parameters are useful for a first dimensioning and as comparison with different design.

An analytical and a Finite Element (FE) model are developed to evaluate in a more precise way the In Plane Forces (IPF), namely F_z and F_r, and the Out of Plane Forces (OPF). The latter will be strictly dependent by the PF coils and CS modules position and current.

The IPF are evaluated considering a filament TF coils model with an analytic approach. The TF coils are modelled as multiple segment filament passing through the geometrical d-shape TF coils profile where the current density centroid lies (called midline of the winding pack). For each filament, a current of value equal to I_{TF} is set to evaluate the magnetic field B in the space of interest including maximum toroidal field acting on the winding pack. The maximum field acts on the d-shape TF coils profile facing the plasma area (called TF coil reference line).

The maximum magnetic field acting on the TF coils as expected is $\sim$ 22 T. The ripple on the plasma axis is verified reaching a maximum value $<$ *0.1* ‰ as for the ripple on the plasma edge that achieve a maximum value $<$ *0.4* ‰ according to the operational limits [45]. To obtain this ripple condition the number of TF coils required is obtained according to [45]:

$$\delta = \left(\frac{R}{R_{outer}}\right)^N + \left(\frac{R_{inner}}{R}\right)^N \tag{8}$$

Where δ is the ripple amplitude, R is the coordinate along the major radius, N is the number of TF coils, R_{outer} and R_{inner} are the major radii of the outer and inner legs of the TF coils. The proposed TF coils preliminary design is also used to evaluate the influence of the TF coils outer legs respect the straight and inner legs. In fact it is possible to observe as the needed toroidal magnetic fields for the plasma confinement is generated only by the TF coils central column. In fact, TF coil outer leg is mandatory to close the conductors winding pack, but it does not have a relevant impact on the magnitude of the magnetic field on the plasma axis (approx. $\sim$ 0.3%). Therefore, increasing the radial extension of the TF coils on the outer side can reduce the magnetic ripple, allowing a lower number of TF coils while approaching the ideal case of an infinitely long straight conductor, as illustrated in Fig. 8b Indeed, lower N brings to simplification in the assembly phase (i.e., a smaller number of electrical joints and an increased space to explore alternative assembly strategy) and critical issues in the manufacturing phase (Section 2). Moreover, the AML configuration with all the PF coils and CS modules located inside the TF coils d-shape could also bring to an optimized shape that minimize the interaction with the poloidal field lines.

By the interaction between the current in the TF coils and the toroidal field it is possible to obtain the Lorentz Forces ($J\ x\ B$) acting on the winding pack. The maximum electromagnetic force density acting on the conductors is approx. equal to 420 MN/m^3 (Fig. 9) and it will be considered constant along the TF coil winding pack as conservative choice to evaluate the stress state on the conductor and on the TF coil casing.

The poloidal field acting on the TF coil is evaluated with a FE model reproducing the plasma equilibrium configuration in the three-time instants considered in Section 3 and the pre-breakdown phase (PreBD).

The worst condition is obtained in the PreBD instant (Fig. 10a), that is when all the CS modules and the TF coils are at the maximum current and no plasma current is in the tokamak. This critical condition, in terms of Lorenz Forces, affect as the TF coils central column as the curvature area (where the electrical joint is expected). All the force density distribution evaluated for the TF coils are referred to the magnet reference line, i.e. the TF coils profile toward the plasma.

The preliminary stress assessment on the TF central column is provided through a 2D Finite Element (FE) model. Two different configurations are reported and compared: a wedged and a bucked design.

The FE analysis includes a 2D model in the condition of plain strain under the radial force load due to the IPF in a linear elastic regime. The radial force acting on the TF coil inner legs is evaluated considering the maximum in plane force density 420 MN/m^3. This is applied on the with a linear distribution along the winding pack. In the wedged configuration (Fig. 11a) the stress state in the stainless-steel casing and the WP is better than the bucked solution (Fig. 11b).

It is important to note that, compared with a conventional reactor design, the bucked configuration adopted in the AML concept is in principle different. In particular, due to the placement of the CS modules outside the TF coils central column, the IPF and OPF acting on the magnets cannot be transmitted to the CS outer surface. In the AML design, there is available space inside the TF coil vault to accommodate a dedicated supporting structure. The latter can be designed to withstand the loads acting on the TF coils, effectively decoupling them from the CS radial expansion. The engineering design of this dedicated supporting structure, intended to sustain the TF coils electromagnetic loads, is under investigation and will be addressed in future studies. The stress analysis, proposed in this Fig. 11b, is preliminary simulated considering a fixed constraint in the TF casing nose inner edge. This represents a conservative assumption, leading to the worst-case scenario in terms of the stress state on the components. Moreover, the positioning of the toroidal magnets in the AML design enables the use of additional structural material closer to the tokamak axis, without increasing R or A of the reactor. As a matter of fact, this solution may also reduce the stress conditions on the CS superconducting material that is foreseen in the conventional bucked design (e.g., the frictional contact between TF coils and the CS causes transmitting the shear forces to the CS itself) [46].

To complete the stress state characterization, the vertical tension load is also taken into account, as it can be superimposed linearly on the stress state produced by the radial load. A 3D FE model of a section of the TF central column is carried out with $F_z/2$ applied as external load. The maximum Von Mises stress on the FE model are: $\sim$ 500 MPa on the structural material and $\sim$ 120 MPa on the HTS cables for the wedged solution (Fig. 11a); $\sim$ 700 MPa on the structural material and $\sim$ 170 MPa on the HTS cables for the bucked solution (Fig. 11b); $\sim$ 470 MPa on the structural material and $\sim$ 220 MPa on the HTS cables for the 3D tension analysis (Fig. 12). The proposed analyses performed to characterize the TF coils, in the AML design, are a first step to prove the feasibility of the solution included in this work bringing to comparable values respect what is foreseen in the under design nuclear fusion reactors [5].

5. Trust project

The TRUST project idea was conceptualised in the summer 2022 by UNITUS research group on the nuclear fusion (UNITUS4FUSION) and it is based on three main pillars: creation of a new pole on nuclear fusion in Italy capable of training a class of engineers with specific skills on all the aspect connected with the technical of a fusion reactor plant; activation of collaborations with the industrial world for their direct involvement right from the design phase of components and technologies applicable in the world of nuclear fusion; dealing with the frontier of technology and engineering challenges testing innovative materials and alternative solutions for the design of a future nuclear fusion reactors. To accomplish its aim, TRUST is already under its design phase (Fig. 13) by PhD students with the following parameters: single null (SN) configuration with $R_0 = 0.3$ m, $A = 2.6$, $I_{pla} = 0.110$ MA, $B_t = 0.8$ T, $q_{95} = 3.3$. The target parameters for TRUST are a trade-off between the need to build an academic and economic tokamak fully design and operated by students and the intention to test advanced technologies for nuclear fusion. The major radius of 0.3 m guarantees a compact device affordable for a university in terms of economic resources, facility area and manpower to be involved. Actually, the cost of the machine is basically expected to scale like the magnetic energy, i.e. $C_{tot} \sim B^2R^3$ [47]. At the same time, with the designed TRUST performances, it will be possible to reach relevant plasma scenario for testing materials. Moreover, the design of a small aspect ratio tokamak (close to a spherical one) is attractive to studies engineering and physical challenges for the nuclear fusion science, such as the integration of all necessary components within the central column of a tokamak. In fact, with this experiment the will be possible test technological solutions for the AML design and will be possible validate the expected results in terms of electromagnetic loads on the TF coils and their coupling with the poloidal fields. More detail on the TRUST project and its engineering and physics aspect are reported in [15].

6. Conclusion

Among all the several different challenging problems, that the Nuclear Fusion will have to tackle in coming years, two of them are really of a fundamental importance. The setting up of a solid and permanent University system capable to prepare at the best the new generations. The necessity to optimise the future reactor system to minimise its complexity, the costs and the maintainability. Within this work we have proposed a way to tackle together these two challenging problems. A promising alternative magnetic concept has been proposed as a study field for new generations of engineers, that will further develop the concept and that will check the possible advantages, drawbacks and eventually the possible way out through the realization of a small experiment, where even test experimentally the problems and the possible solutions. Future work should focus on refining these designs, enhancing experimental capabilities, and fostering collaborations with industrial partners to address the remaining engineering and scalability challenges.

CRediT authorship contribution statement

S. Carusotti: Writing – review & editing, Writing – original draft, Validation, Supervision, Project administration, Methodology, Investigation, Funding acquisition, Formal analysis, Data curation, Conceptualization. **R. Lombroni:** Validation, Methodology, Investigation, Formal analysis, Conceptualization. **M. Scarpari:** Writing – original draft, Validation, Methodology, Investigation, Formal analysis, Data curation, Conceptualization. **M. Notazio:** Writing – original draft, Validation, Methodology, Formal analysis, Data curation. **A. Pizzuto:** Supervision, Methodology, Conceptualization. **F. Crisanti:** Writing – review & editing, Writing – original draft, Supervision, Project administration, Methodology, Investigation, Funding acquisition, Data curation, Conceptualization. **P. Fanelli:** Supervision, Resources, Project administration, Funding acquisition, Conceptualization. **G. Calabrò:** Supervision, Resources, Project administration, Funding acquisition, Conceptualization.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Data availability

Data will be made available on request.

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