Progress in the conceptual design of the CFETR toroidal field coil with rectangular conductors

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Abstract
A comprehensive research facility project was approved in December 2018 with funding of 345 million EUR, to support the research and development of the China Fusion Engineering Test Reactor (CFETR). As part of this project, a full-size CFETR toroidal field (TF) coil will be designed, manufactured and tested by the Institute of Plasma Physics Chinese Academy of Sciences. Two options are being explored in parallel for the TF coil design, using either circle-in-square or rectangular cable-in-conduit conductors (CICCs). The rectangular CICC has been reported to have some merits for a DEMO TF coil, as reported in designs of the EU-DEMO and K-DEMO. First this paper presents the progress in the conceptual design of the CFETR TF coil with rectangular CICCs, according to the most recent reference single-null configuration and radial build of CFETR. Then, electromagnetic analyses are performed to give the magnetic field distributions, toroidal field ripple, in-plane and out-of-plane Lorentz loads. Finally, 3D global and 2D local mechanical analyses are conducted, and the detailed mechanical behavior of the TF coil is illustrated and discussed. Our analyses indicate that the present TF design is reasonable, considering the ITER criteria, and provides valuable insight into the mechanical behavior of the CFETR TF coil system.

Keywords: tokamak, DEMO, superconducting magnet, TF, cable-in-conduit conductor, electromagnetic load, mechanical analysis

1. Introduction

ITER is the world’s largest experimental reactor under construction and is scheduled for the first plasma in 2025. After ITER, construction of a demonstration fusion reactor (DEMO) will be a necessary next step to develop the scientific and technological viability of a commercial fusion power plant. Conceptual designs have been conducted for EU-DEMO [1, 2], JA-DEMO [3], K-DEMO [4–6] and CFETR [7–9], etc.

CFETR aims to initially demonstrate fusion energy production of 200 MW and eventually reach a DEMO relevant power of 1 GW. The most recent design of CFETR plasma is shown in figure 1, with comparisons to other relevant devices. Note that obsolete designs of CFETR with major radius of 5.7 m and 6.6 m have previously been considered [8]. The updated major and minor radii of CFETR are now $R = 7.2$ m and $a = 2.2$ m, the plasma volume is $1249 \text{ m}^3$ (840 m$^3$ for ITER).

The reference lower single-null configuration of CFETR issued in June 2019 is shown in figure 2. In this configuration, the toroidal field at major radius (7.2 m) is $B_t = 6.5$ T, the plasma elongation is 2 and the plasma current is 14 MA.
The magnet system consists of 16 toroidal field (TF) and 6 poloidal field (PF) coils, a central solenoid (CS) and a divertor coil (DC).

Figure 3 provides the updated radial build of CFETR issued in March 2019. The CS has an outer radius of 2.27 m and a thickness of 1.18 m. A space of 0.94 m is allocated to the inner blanket for tritium breeding and neutron shielding. The TF inboard leg is restricted between 2.29 m and 3.48 m. It should be noted that the radial space allocated to the TF coil is limited and, in general, results in mechanical issues for its design.

In support of the research and development for CFETR, a comprehensive research facility project has been approved with funding of 345 million EUR (2.7 billion CNY). This project will advance the development of key components for CFETR, including the blanket, heating system, divertor and superconducting magnet system.

A full-size CFETR TF coil will be designed, manufactured and tested in this comprehensive research facility. For its design, two options are being considered, one uses circle-in-square CICCs [10], the other uses rectangular CICCs. In this paper, first, the conceptual design of the TF coil with rectangular CICCs is introduced, next, the numerical model is built and the electromagnetic analysis is conducted, then, the 3D and 2D mechanical analyses are performed, finally, a conclusion is given.

2. Conceptual design of the TF coil with rectangular CICC conductors

2.1. Toroidal field consideration

The TF coil is one of the most important superconducting coils, it defines the shape of the tokamak, the major radius and the toroidal field. With \( R = 7.2 \) m, \( N = 16 \) (number of CFETR TF coils) and TF coil dimensions restricted in the aforementioned radial build, many schemes of the winding pack (WP) layout have been studied, taking into account different rectangular conductor cross-sections, different numbers of layers/turns and different operating currents. These schemes are summarized in figures 4(a) and (b), where \( B_t \) (toroidal field at \( R = 7.2 \) m) and \( B_{\text{max}} \) (maximum field on TF conductors) are plotted as a function of the total current \( N I \). By simplifying a WP into a current filament, the TF coil system (consisting of \( N \) coils, each carrying a current \( I \)) can be modeled as in figure 4(c), and the relations can be expressed as the following formulas [11–13]:

\[
B_t = \frac{\mu_0}{2\pi R} \left[ 1 + \frac{1}{(R/R_1)^N - 1} + \frac{1}{(R_2/R)^N - 1} \right] \cdot NI \quad (1)
\]

\[
B_{\text{max}} = \frac{\mu_0}{2\pi R_1} \cdot NI. \quad (2)
\]

According to the fittings, it can be obtained that the \( B_t \) and \( B_{\text{max}} \) is proportional to \( NI \) with a slope of 2.8 and 6.2 respectively, which means that if we want to increase the toroidal field of CFETR, the maximum field on TF conductors will increase 2.2 times faster. The higher field on conductors and more severe mechanical issues hinder a higher toroidal field. Actually, we have tried to consider schemes of TF coil with a \( B_t \) of 7.5 T or 7 T, but the mechanical performances cannot satisfy the design criteria. Therefore, a \( B_t \) of 6.5 T is chosen for the CFETR TF coil.

2.2. The TF structure design

The poloidal cross-section of a CFETR TF coil and dimensions of the WP centerline are displayed in figure 5(a). This centerline has been reached through an iterative process,
Figure 3. Radial build of CFETR issued in March 2019. The main components include the central solenoid (CS), toroidal field coil (TFC), thermal shield (TS), vacuum vessel (VV), and blanket.

Figure 4. (a) and (b) The toroidal field $B_t$ and the max. field on conductors $B_{\text{max}}$ as a function of total current $NI$ for CFETR. (c) The simplified filament model for a TF system consisting of $N$ coils, each carrying a current $I$.

Figure 5. (a) Poloidal cross-section view of the CFETR TF coil and dimensions of the WP centerline. (b) Overview of the CFETR magnet system.
basing on the constant-tension (bending-free) Princeton-D shape [12]. In the inboard leg, the inner part of each TF coil case is thicker to withstand the significant centering force.

The overview of the present CFETR magnet system is given in figure 5(b), the TF coil system has a radius of 16.3 m and a height of 21.7 m. The inboard legs of TF coils are wedged and thus can support each other, while the outboard legs are supported by the outer inter-coil structures (OIS) in the toroidal direction.

The TF coil cases, which enclose the TF WP, are the main structural components of the CFETR magnet system and have to withstand a combination of in-plane and out-of-plane Lorentz forces. The in-plane (poloidal) forces are mainly reacted by the wedging of the inboard straight legs, while the out-of-plane (toroidal) forces are mainly reacted by the stiff rings formed by the three sets of OIS.

2.3. The TF WP design

The layout of the WP and magnetic field on the cables of conductors are shown in figure 6, the total number of turns is 154 and the main parameters for the preliminary design of rectangular conductors are listed in table 1. With a conductor current of 95.6 kA, a toroidal field of 6.5 T is produced and the peak field on conductors is 14.7 T.

In the plasma facing side of the WP, the magnetic field is higher while the stress level is lower. On the contrary, in the CS side of the WP, the field is lower while the in-plane radial forces are accumulated. Under these conditions, a layer-wound WP is designed composed of three grades (high-performance Nb$_3$Sn, ITER-like Nb$_3$Sn and NbTi) of rectangular CICC, thus optimizing the use of superconducting conductors and hence the cost. This layout also allows grading of the jacket thickness. In the high-field region (L1 to L3, peak filed 14.7 T), high-performance Nb$_3$Sn CICC with a jacket thickness of 7 mm is employed. In the middle-field region (L4 to L8, peak filed 11.1 T), ITER-like Nb$_3$Sn CICC with a 10 mm thick jacket is used. In the low-field region (L9 to L12, peak filed 6.6 T), NbTi CICC with a 13 mm thick jacket is used.

The insides of the jackets are optimized to oval shapes in order to reduce the stress concentrations, which usually occur at the inside corners. This jacket shape is hard to manufacture with the pull-through and compaction technique, and a longitudinal-welding method should be more applicable. Detailed designs of the three grades of rectangular conductors are still in progress, and thorough research and development activities are still needed. The performance requirements of the high-performance Nb$_3$Sn strand are listed in table 2. The Nb$_3$Sn sub-WPs will be fabricated by the wind-and-react technique.

### Table 1. Main parameters for preliminary design of the WP rectangular conductors.

| WP conductors | Grade-1 | Grade-2 | Grade-3 |
|---------------|---------|---------|---------|
|               | High-perf. Nb$_3$Sn | ITER-like Nb$_3$Sn | NbTi    |
| Layer         | L1      | L6      | L10     |
| Peak field (T)| 14.7    | 9.6     | 6.6     |
| No. of turns  | 15      | 13      | 11      |
| Turn length (m)| 43.3   | 45.1    | 46.2    |
| Jacket ext. dim. (mm)| 46 × 70 | 52 × 70 | 58 × 70 |
| Jacket thk. (mm) | 7      | 10      | 13      |

### Table 2. Performance requirements of the high-performance Nb$_3$Sn strand.

| Parameter                           | Value |
|-------------------------------------|-------|
| Minimum piece length (m)            | 1000  |
| Cr plated strand diameter (mm)      | 1.0 ± 0.005 |
| Cr plating thickness (µm)           | 2.0 ± 0.1 |
| Cu: non-Cu                          | 1.0 ± 0.1 |
| Twist pitch (mm)                    | 15 ± 2 |
| RRR between 273 K and 20 K          | >100  |
| $J_c$ at 14 T, 4.2 K (A mm$^{-2}$)  | >1550 |
The merits of using rectangular CICCs to face the significant electromagnetic loads are that the strain of Nb3Sn strands could be reduced during winding, and the cable deformation could be reduced during operation, hence the performance of conductor could be improved in a DEMO TF coil [1, 2, 14–17]. In the design of the EU-DEMO TF coil, WP#1 with 12 grades of rectangular CICCs is proposed by Swiss Plasma Center (SPC) [16, 18, 19], WP#2 with six grades of rectangular CICCs is proposed by Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA) [20–22], and prototype rectangular CICCs have been developed and tested [15, 16]. Moreover, French Alternative Energies and Atomic Energy Commission (CEA) proposed the WP#3 with square CICCs [23, 24].

The largest thickness of the CFETR TF coil case is 323 mm (205 mm for ITER), and the lowest thickness is 62 mm (41 mm for ITER). A 1.5 mm turn-insulation, 1 mm pancake insulation, 3 mm layer-insulation, 8 mm ground-insulation and a 10 mm insertion-gap has been considered.

3. Numerical model

3.1. Finite element model

A 2D local finite element model of the inboard leg cross-section in the equatorial plane and a 3D global model of the TF coil is built, as displayed in figure 7. Mapped meshing is used as much as possible for the 3D global model, the final mesh contains 116000 elements and 372000 nodes.

For both the 2D and 3D model, a 16-fold cyclic symmetry is applied, and frictional contacts are considered between the coil cases and the WP insulations. The displacements of the support post in all directions are constrained.

3.2. Material properties and load conditions

Material properties retrieved from the ITER database are used for the analysis [25–27]. 316 LN stainless steel is utilized for the coil case, support structures and the conductor jackets. Orthotropic epoxy glass fiber G10 is utilized for the insulations.

Because the 3D global model is too large to model in detail, the three (high/mid/low field) parts of the WP are treated as equivalent homogeneous orthotropic materials. The smeared properties of each part of the WP are calculated by finite-element method and listed in table 3.

Mechanical analyses are performed for the following load conditions: (i) cool-down from room temperature to 4 K; (ii) energization of TF coil (TF-Only); and (iii) start of flattop (SOF), middle of flattop (MOF) and end of flattop (EOF). Note that since the CFETR scenario design is not mature, during plasma operation, only the load cases of SOF/MOF/EOF have been defined.

3.3. Structural limits

For the static structural assessment of the TF coil system, allowable stresses are determined by referring to the ITER design criteria [28, 29], based on the maximum shear stress theory using the Tresca stress intensity. The primary membrane stress should not exceed two-thirds of the yield strength of 1000 MPa ($P_m < 667$ MPa), the allowable value increased by a factor of 1.3 for the combined membrane and bending stresses ($P_m + P_b < 867$ MPa), and primary plus secondary stress ($P_m + P_b + Q$) should not exceed 1000 MPa [25, 29].

The ITER criteria for the allowable shear stress of insulation is between 42 MPa and 68 MPa, depending on the normal compression stress [25, 28].

4. Electromagnetic analysis

The inductance of the TF coil system is 34.9 H, the stored magnetic energy is 159.6 GJ with a total current of $NI = 16$.
\[ 154 \times 95.6 = 235.6 \text{ MA}. \] Note that for the ITER TF coil system, the inductance is 17.7 H and the stored energy is 41 GJ with a current of 164 MA [25].

The magnetic field distributions at TF-only (cross-sectional view) and MOF (cross-sectional and 3D view) are illustrated in figure 8, where the peak field on the TF WP is 14.6 T. The maximum fields on outboard segments of the WP are between 7 T and 11 T, which should be taken into account for the design of the conductor joints.

The toroidal field ripple of the present TF coil design, calculated as the relative amplitude of toroidal field variation \((B_{\text{max}} - B_{\text{min}})/(B_{\text{max}} + B_{\text{min}})\), is plotted in figure 9. We can see that the peak ripple over the plasma region is 0.36%.

After energization (TF-only), each TF coil experiences an in-plane bursting force as well as a resultant centering force towards the center of the tokamak. This in-plane force arises from interaction between the TF coil current and the toroidal field. As shown in figure 10, the in-plane bursting force can be decomposed into a radial \((F_r)\) and a vertical \((F_z)\) component, with a maximum \(F_r\) of \(-143\) MN m\(^{-1}\) and a maximum \(F_z\) of 88 MN m\(^{-1}\). The resultant centering force is 1090 MN/coil (403 MN for the ITER TF coil [30, 31]).

During plasma operation, interaction of the TF coil with the poloidal field of CS and PF coils gives rise to the time-dependent out-of-plane force. As illustrated in figure 11, the out-of-plane forces at SOF/MOF/EOF vary along the coil perimeter and the maximum value is 25.4/18.8/21.7 MN m\(^{-1}\). The resulting overturning moment about the radial axis at SOF, MOF and EOF is 560.567 and 631 MNm, respectively (the maximum overturning moment is 134 MNm for the ITER TF coil [32, 33]).
5. 3D global mechanical analyses

During cool-down from room temperature to 4 K, the maximum radial and axial displacements are $-48.2 \text{ mm}$ and $-63.3 \text{ mm}$ for the TF coil.

At TF-only and SOF/MOF/EOF, the peak stress intensity of the TF coil system amounts 794 MPa, as we can see in figure 12.

Note that the maximum values and the stress distributions in the inboard region are the same for the load cases of TF-Only/SOF/MOF/EOF, which suggests that the mechanical strength of the TF coil is mostly determined by the in-plane centering forces. Actually, the in-plane radial force $F_r$, resulting in the centering force, reaches its maximum at the inboard wedging region (figure 10), that is why the peak stress occurs at this region.
In fact, the in-plane forces are shared between the WP and the coil case [34]. Therefore, if the thicknesses of conductor jackets are small and consequently the poloidal modulus of the WP is low, then more in-plane forces will be shared by the coil case and thus leading to a higher stress in the inboard region. In other words, the softer the WP is, the more load will be endured by the coil case, which results in higher stress level in the inboard legs.

Furthermore, it can also be observed in figure 12 that the time-dependent out-of-plane forces cause different stress distributions in the outboard region, and local stress maximums occur at the connections between the OIS and the TF case.

The out-of-plane displacements at SOF (back view), MOF (side view), and EOF (back view) are shown in figure 13, the maximum value is 20.13/20.05/19.38 mm at SOF/MOF/EOF. The maximum toroidal displacement occurs at the upper outboard region for the reason that the vertical maintenance ports of CFETR are located here. Note that the maximum toroidal displacement of ITER TF coil is 21.7 mm at EOB + PD [25].

6. 2D local mechanical analyses

In order to obtain information on the mechanical behavior of the rectangular conductors and insulations, a mechanical analysis at TF-only with the 2D local model has been performed. Figure 14 shows the stress intensity distribution in the coil case and conductor jackets. For the coil case, the peak stress intensity is 626 MPa. For the conductor jackets, a peak stress of 758 MPa (membrane plus bending stress of 624 MPa) occurs at the internal fillet of the jacket (see inset of figure 14(b)). We
note that the large internal fillet radius could reduce the peak stress of the jackets.

The shear stress in the insulations is below 43.9 MPa as shown in figure 15.

The radial displacement distribution in the cross-section of the inboard leg is presented in figure 16. It can be seen that the WP is compressed inward by the centering force, and a gap up to 3 mm appears between the coil case and the WP.

7. Conclusions

According to the updated reference single-null configuration and radial build of CFETR, a conceptual design of the TF coil with graded rectangular conductors has been conducted. Electromagnetic analyses show that the inductance is 34.9 H (2 times of ITER TF) and the stored energy is 159.6 GJ (4 times of ITER TF). The resultant centering force is 1090 MN/coil (2.7 times of ITER TF), and the overturning moment amounts to 631 MNm (4.7 times of ITER TF).

3D global and 2D local mechanical analyses have been performed, the high stress regions have been identified as the inboard straight leg of the TF coil case and the curved region of the OIS structures. The maximum out-of-plane displacement is 20.1 mm at SOF (21.7 mm at PD for ITER TF). Our analyses show that the mechanical behavior and the stress levels of the TF coil satisfy the ITER design criteria. However, the present design of CFETR TF coil still needs to be updated according to the detailed design of the rectangular conductors in the future. For instance, the thickness of the TF conductor jackets could be reduced to provide more space for the cables and/or the coil case.

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