Fusion Reactor Related Technologies Development in ASIPP

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Abstract. Fusion research in the world has nowadays been focused on the R&D of future fusion reactor technologies, such as technologies of large scale superconducting magnets, RF/NBI heating and current drive, blanket and divertor, plasma control and diagnostics, tritium recycling and radiation resistant materials. Some key technologies have been developed by the projects of HT-7 in operation and EAST being built in the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). The technologies, such as long pulse discharges for plasma initiation with low loop voltage, wall conditioning, plasma facing material (PFM), non-inductive current drive and heating, plasma control and data acquisition and processing etc., have been developed very well on the HT-7 superconducting tokamak in the past five years. The R&D programs of EAST have been quite successful, which have been especially focusing on the superconducting technologies development, such as design, fabrication and test technologies of large scale magnets and cable-in-conduit conductor. A multi-element doped graphite and thick gradient SiC coatings on carbon based materials (CBM) have been developed as plasma facing materials of the EAST device. Detailed investigation of the composition has been done. High-Z materials, e.g. Mo and W, and functionally graded materials for PFM have also been studied in China. As a primary candidate of structural materials of blanket, the Chinese version of the RAFM (Reduced Activation Ferritic/Marntensitic) steels, e.g. CLAM (China Low Activation Ferritic-Martensitic) steel, are being developed in ASIPP. In this paper, some key technologies developed in ASIPP are introduced.

1. Introduction

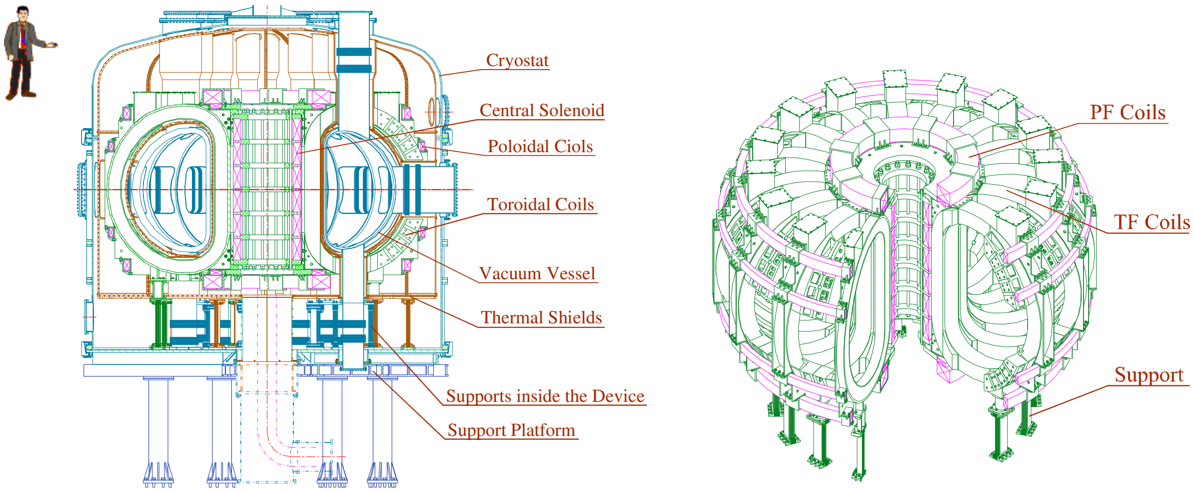
After fifty years research and development, fusion research nowadays in the world has been mainly focused on the fusion energy and reactor technologies. Especially with promotion of the ITER (International Thermal-nuclear Experimental Reactor) project, some fusion reactor related key technologies were developed pretty well in the past ten to fifteen years, such as technologies of large scale superconducting magnets, non-inductive heating and current drive, blanket and divertor, plasma control and diagnostics, tritium recycling and radiation resistant materials etc..

For its large populations and rapid economic growing, China could be the country that is the eagerest to use fusion energy in the world. In 1998 China government approved a national Mega-project that is the EAST (Experimental Advanced Superconducting Tokamk) project, even though China is a developing country. In 2003, China government decided to joint ITER negotiation in a short time. By the existing superconducting tokamak HT-7, the technologies of long pulse discharges for plasma initiation with low loop voltage, wall conditioning, plasma facing material, non-inductive current drive and heating, plasma control and data acquisition and processing have been developed very well. The R&D programs of EAST have been quite successful, which have been especially focusing on the superconductivity engineering development, such as design, fabrication and test technologies of large scale magnets. A multi-element doped graphite and thick gradient SiC coatings on carbon based materials (CBM) have been developed as plasma facing materials of the EAST device. High-Z materials, such as tungsten, for PFM have also been studied in China.

As a primary candidate of structural materials of blanket, the Chinese version of the RAFM (Reduced Activation Ferritic-Marntensitic) steels, China Low Activation Martensitic (CLAM) steel, are being developed in ASIPP. The current researches on CLAM steel have covered smelting, heat treatment, property tests, fabrication of steel and the effects of irradiation and corrosion etc..

2. Large Scale Superconducting Technologies Developed with the EAST Project

EAST features superconducting toroidal field (TF) system and poloidal field (PF) system, non-inductive current drive and plasma heating systems, flexibility and reliability of plasma shaping control, J(r) and P(r) control to satisfy advanced operation requirements of EAST, replaceable plasma facing components and divertors for power and particle handling study in steady-state operation and advanced diagnostic measurements [1]. The main components of the EAST superconducting tokamak are the TF and PF superconducting magnet systems, the double-wall vacuum vessel, the thermal shields, the cryostat and the support system [2]. The total weight of the EAST machine is 400 ton. Its overall size is 8 m in diameter and 10 m high including a support frame. A cross-section view of the EAST device is shown in Fig. 1 (a).



(a) *(b)*

Fig. 1 The cross-section of the EAST device (a) and a cutaway view of the EAST superconducting magnet system (b)

The EAST superconducting magnet system, as showing in Fig. 2 (b), is consisted of a PF magnet system and a TF magnet system. The PF magnet system includes a central solenoid (CS) assembly and other three pairs. Two biggest PF coils have the same diameter of 6.65 m. The maximum energy stored in the PF magnet system is about 50 MJ. Sixteen toroidally arrayed coils form the TF magnet system. Each of them has size of 3.5 m × 2.6 m and weighs 10 tons. The energy stored in the TF system is about 300 MJ with 14.5 kA nominal current. The D-shaped coil composed by five arcs and a straight line is designed based on bending moment free D.

A so-called cable-in-conduit conductor (CICC) is adopted for its obvious advantages, such as good self-support structure, conventional insulation design and technology, especially its multi-cabling and wrest-free structures benefit lower AC losses and operation in pulse magnet fields. But CICC stability mechanism is much more complicated and less experienced. The suitable copper fraction of the superconducting cable is one of the key factors affecting the CICC stability and must be optimized. The optimization principle of copper ratio is to get maximum current density in CICC with satisfied stability.

To make use of existing superconducting strands made by NbTi superconducting material originally for accelerator, a so-called CICC with separated copper strands as stabilizer (CICC-SCSS) has been suggested, because the copper ratio of the strand is about 1.38. Otherwise to get high copper fraction in the cable, such as 0.5, more superconducting strands have to be used. It will result in a lower current density and higher cost. Several versions to use separated copper strands were considered and finally the configuration of (2SC+2Cu)×3×4×5 and (1SC+11Cu)×4×5×6 was proposed. Considering low limitation current and up limitation current and transitional stability, the relationship among the current density inside cable and copper ration may be indicated by following expressions [3]:

$$J_{Lim}^{Low} = \frac{4K_p h}{\rho d}\frac{(T_c - T_{op})}{J_c}.\frac{f_{cu}}{f_{sc}}(f_{cu}+f_{sc})COS\theta \qquad (1)$$

$$J_{Lim}^{up} = COS\theta\sqrt{\frac{4K_p h}{\rho d}(T_c - T_{op})f_{cu}(f_{cu}+f_{sc})} \qquad (2)$$

$$J_{\Delta E}^{w\cdot c\cdot} = f_{sc}J_c[1-\frac{f_{cu}+f_{sc}}{1-f_{cu}-f_{sc}}\frac{\Delta E}{\overline{C}(T_c-T_{op})}]COS\theta \qquad (3)$$

$$J_{\Delta E}^{tr} = J_{Lim}^{up}[1-\frac{f_{cu}+f_{sc}}{1-f_{cu}-f_{sc}}\frac{\Delta E}{\overline{C}(T_c-T_{op})}]^{0.5} \qquad (4)$$

Where, θ is the average twist angular, J_c is critical current of NbTi, K_p is the wet perimeter reduced coefficient due to the cabling, d is the strand diameter, f_{cu} and f_{sc} is the copper fraction and NbTi fraction in the cable separately, f_{He} is the void fraction of the cable. $\overline{C}$ is the helium average volume specific heat from T_{op} to T_c. Not only the classic calculation and analyses were performed, but also a code called Gandalf was used for quench simulations [4].

According to the sub-cable tests of two configurations mentioned above, (2SC+2Cu)×3×4×5 was accepted. To investigate the function of the separated copper strands, the influence of the strand surface to stability and the real energy margin, four short samples were fabricated and tested. We are glad to see all conductors have enough ability against transient disturbance at the nominal operating condition. We also find higher mass flow rate may improve transient stability. Based on the test results, (2SC+2Cu)×3×4×5+1 central copper cable (CCC)

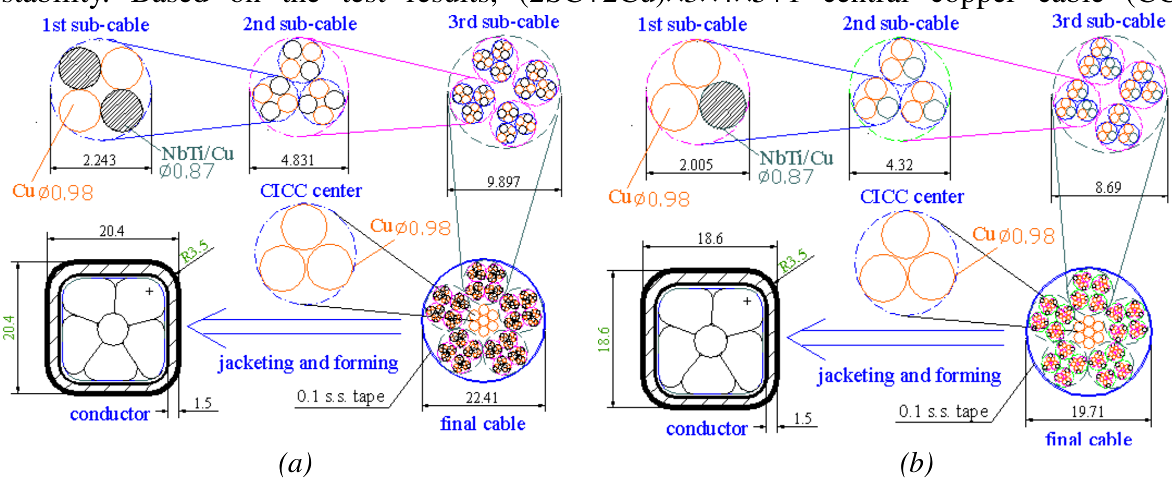
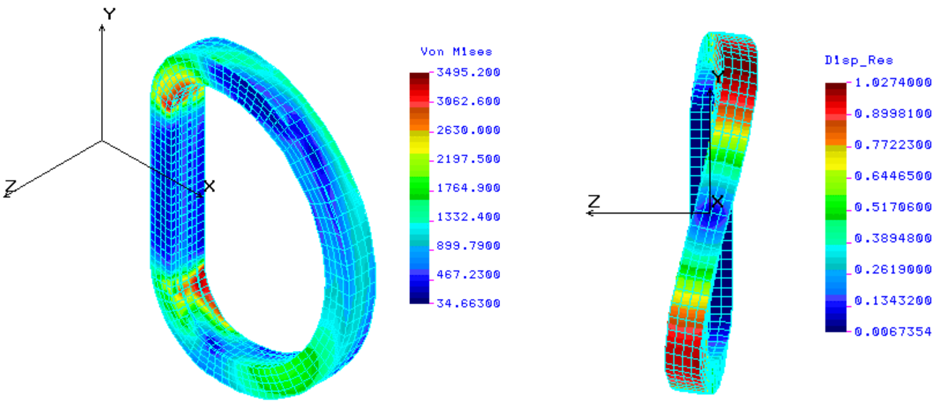


Fig. 2 The TF/CS CICC-SCSS configuration (a) and the big PF CICC configuration (b)

configuration was used for TF, central solenoid and divertor coils. But the TF conductor strands were coated by Pb-30Sn-2Sb and other two PF conductors strands were coated by nickel. The four big PF coils configuration is (1SC+2Cu)×3×4×5+1CCC for their work field are lower than 2 T. Fig. 2 shows the TF/CS CICC-SCSS configuration (a) and the big PF CICC-SCSS configuration (b). The main parameters of the EAST CICC-SCSS are listed in Table 1.

TABLE 1: THE MAIN PARAMETERS OF THE EAST CICC-SCSS

	TF	CS/Divertor	Big PF
Rated operation field	5.8 T	4.5 T	2 T
Rated operation current (I_{op})	14.3 kA	14.5 kA	
Rated operation temperature	4.2 K		
Cable configuration	(2SC+2Cu)×3×4×5+1CCC		(1SC+2Cu)×3×4×5+1CCC
Conductor dimension	20.4× 20.4 mm^2	20.3× 20.3 mm^2	18.5× 18.5 mm^2
Conduit thickness	1.5 mm		
Number of SC strand	120		60
Number of Cu strand	120+21		
Diameter of SC strands	0.85~0.87 mm		
Diameter of copper strands	0.98 mm		0.98/0.87 mm
RRR of Cu strands	> 100		
Coating materials	Pb-30Sn-2Sb	Nickel	Pb-30Sn-2Sb
Solder thickness on strands	2-3 μm	2 μm	3 μm
Cu fraction (f_{cu})	0.54		0.44
Helium fraction (f_{he})	0.34	0.35	0.359
I_{op}/ I_c	0.28	0.224	0.31
Temperature margin (T_{cs}-T_{op})	1.88 K	2.54 K	2.29 K
Energy margin (ΔE)	250 mJ/cm^3	350 mJ/cm^3	400 mJ/cm^3



(a) *(b)*

Fig. 3 The stresses (a) and deformation (b) distribution on the TF case

Each TF coil is housed in a stiff case to withstand very strong in-plane forces and out-plane forces. All case structures are made by 316LN stainless steel which presents more than 1500 MP yield stress at lower than 80 K. In the front of the TF straight leg, the case plate thickness is 58 mm to withstand more than 1300 ton centripetal forces [5]. To support out-plane moment up to 332 ton·m, the case is designed as a box structure with reinforced ribs and big keys between two cases. To investigate the TF structure design, the finite element analyses (FEA) have been performed. The FEA results show the maximum stress on the case is less than 350 MPa and the maximum deformation is about 10 mm. Fig. 3 shows the stresses and deformation distribution on the TF case. In the Table 2, the main parameters of the TF magnet are listed.

TABLE 2: THE MAIN PARAMETERS OF THE TF MAGNET

Magnetic field at R_o=1.7 m	3.5 T	4.0 T
Maximum field on the coil	5.8 T	6.5 T
Ripple from R_o=1.3 to 2.3 m	<1.5 %	
Number of TF coils	16	
Total Ampere-turns	30 MAT	34 MAT
Operating current	14.31 kA	16.35 kA
Total inductance	2.92 Henry	
Total stored energy	298.39 MJ	389.71 MJ
CICC length of each coil	1200 m	
Total weight of the TF magnet system	160 ton	

All EAST superconducting coils are designed in pancakes structure. In the region formed by four corners of the CICC-SCSS, four wires are filled along with CICC-SCSS for quench detection. Conductors and pancakes of the coils are insulated by 0.5 mm and 1 mm fiberglass, separately. Each coil is ground insulated by more than 6 mm multi-layer glass tape and finally impregnated with epoxy resin by vacuum pressure impregnation (VPI). All PF coils are even wrapped by glass/kapton tapes. Because of 600 m of the each CICC-SCSS, stub structure on conductor is used for reducing the cooling channel length less than 200 m. To separate the helium circuit with electric circuit, potential break technology has been developed successfully.

The key technologies of the CICC-SCSS fabrication developed in ASIPP are butt joint of conduits, cable inserting into the 600 m conduit, extruding CICC-SCSS with high accuracy of the size and 600 m CICC-SCSS collection. Any leakage means the failure of the superconducting facility. Several kinds of non-destroyed checking technologies were used for joint check, such as eddy current, ultrasonic, X-ray, surface viewing, vacuum and pressure. Each CICC has been tested under liquid nitrogen temperature inside a vacuum tank. A pre-bending and continuous winding technology were used and realized by adjusting distances between multi-rollers by computer. It may control the radius within 0.1 mm and overall size within 0.5 mm. Four big PF coils had to be fabricated inside the EAST hall for their diameters are more than 6 m. Conductors and pancakes of the coils are insulated by 0.5 mm and 1 mm by multilayer fiberglass for turn-to-turn and pancake-to-pancake insulation, and finally by more than 8 mm ground insulation and impregnated with epoxy resin. A special VPI technology for large scale superconducting coil has been developed. A VPI facility with a big multi-function (vacuum/pressure /oven) vessel was set up for coils of less than 4 m in diameter. For bigger coils, VPI on site was performed without multi-function oven. A so-called soft jig was used for VPI on site.

To test the coils at cryogenic temperature before installation inside the EAST machine, a test facility system was set up, which includes a big cryostat of 3.5 m in diameter and 6 m high, 500 W/4.2 K cryogenic and refrigerator system, 80 MW power supply system and data collection and control system. Except four big PF coils, all other coils were tested with up to nominal currents. A model coil made by the same CICC as four big PF coils was fabricated and tested. The prototype TF coil and the central solenoid prototype coil were tested with not only nominal currents, but also quench currents. The test current of the TF prototype coil is based on the same $\vec{B}\times\vec{I}$ as the operation. But due to the current limitation of the power supply system the quench current of the TF prototype coil was detected at 16.4 kA with 6.8 K operation temperature. The extrapolated quench current at 3.8 K with peak field of 6.72 T can be calculated to be 58 kA, which indicates I_{op}/I_c is still less than 0.3. The typical test current scenario of the CS prototype coil is based on the same $\int \dot{B}^2 dt$ as the operation scenario of the EAST plasma initiation. It is well-known that the superconducting strands are quoted as its weight no matter how much copper ratio of the strand. The fine results of CICC-SCSS could be very useful to develop CICC with high performance-price ratio.

3. Plasma Facing Materials and Components Developed for EAST

Low erosion, high heat flux components of carbon based materials (CBM), such as doped graphite and carbon fiber enforced carbon composite (CFECC) with high thermal conductivity and adequate mechanical strength are the main requirements of plasma facing components (PFC). The EAST stationary heat flux on the divertor target could be up to 5 MW/m^2. New materials with improved properties, for example, erosion behavior and heat flux capability are being developed and characterized [6]. The research works were mainly concentrated on boron doped graphite, silicon doped graphite, titanium doped graphite and multi-element (B, Si, Ti) doped graphite, together with the methods to apply some thick gradient distributed low-Z carbide coatings (B_4C, SiC) on doped graphite. Material performance-evaluations were tested with physical sputtering, chemical erosion, radiation enhanced sublimation, thermal shock resistance, out-gassing and thermal desorption. The doped graphite has shown characteristics of high density, low open porosity, high strength and high thermal conductivity.

By optimizing the proportion of doped elements and the preparation technique, a name of GBST1308 (1%B, 2.5%Si, 7.5%Ti) doped graphite has been successfully developed and used as the main belt toroidal and poloidal limiter materials of the HT-7 superconducting tokamak as shown in Fig. 4. The plasma performance has been significantly improved. The thermal conductivity of GBST1308 is up to 150 W/m·K at room temperature and very stable with the temperature rise. The erosion experiment also indicated that chemical sputtering yield of GBST1308 by 50 eV and 1 KeV D^+ bombardment was lower than that of pure fine grain graphite by a factor of 30% and 5, respectively. GBST1308 also has good vacuum engineering properties with low degassing rate of 3×10^{-12} Torr·L/s·cm^2 at room temperature and lower recycling. The mechanical bending strength of GBST1308 is higher than 40 MPa. By a method of chemical vapor reaction (CVR) combined with chemical vapor infiltration (CVI), all the carbon tiles can be coated with about 200 μm SiC gradient coatings. Its thermal and mechanical properties are much stable when surface temperature is not more than 1100^0C. The surface temperature of GBST1308 is less than IG-430U and CX-2002U when heat flux not more than 4 MW/m^2 with active water-cooling (copper alloy as heat sink and super carbon sheet as the compliant layer between the interface). The material of GBST1308 with 200 μm SiC gradient coatings is one attractive first wall material in EAST device. GST38 (2.5% Si, 7.5% Ti) is being considered to be used as the divertor material for the EAST device due to its thermal conductivity high up to 250 W/m·K at room temperature.

Many efforts have been made to improve the surface characteristics of graphite by use of boron or silicon containing materials, which have a strong affinity for elemental oxygen. The ICRF Boronization and Siliconization coating technology has been successfully developed in the HT-7 superconducting tokamak. But the coating lifetime is short, so a variety of thick B_4C and SiC coatings have been developed and tested. SiC has low chemical and high-temperature sputtering and is capable of oxygen gettering and low hydrogen recycling. Thick SiC coating on graphite does not have a deleterious effect on the good thermal characteristics of the graphite. A SiC coating technique, such as CVR combined with CVI, has been developed and applied to the HT-7 limiter carbon tiles. SiC gradient coatings are formed by the infiltration of reaction gas through open pores. A test stand with diagnostics was established in ASIPP for PFM test experiments, which can simulate steady state high heat loads with the energy of 20 KeV electron beam irradiation, and disruptions impact on PFM within the time scale of 0.1-3 ms. Beside it, the HT-7 operation can simulate long pulse high heat loads. Thick SiC gradient coatings on doped graphite showed excellent durability under high heat flux and plasma irradiation laboratory experiments and the HT-7 operation. The obtained results indicated that the multi-element doped graphite-GBST1308 improves the durability of thick SiC gradient coatings significantly. It is especial important that the carbon tiles with SiC coated after experiments can be recoated again, which provides a possibility of PFM restore technique.

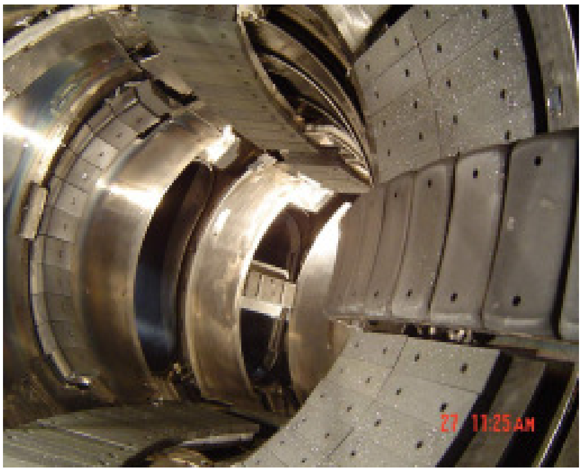


Fig.4 The HT-7 toroidal and poloidal limiters

Some mock-ups of carbon and copper alloy heat sink with actively cooling condition have been evaluated. The specimens were mechanically joined to copper heat sink with super carbon sheet as a compliant layer in between. The experiments have been performed by a 100 kW deflection-type electron beam heating apparatus, which was supported by Chinese-Japan Core University Program. The experimental results, shown in Fig. 5, are very encouraging that the surface temperature of GBST1308 is less than 1000^0C when heat flux is not more than 6 MW/m^2. The primary results indicate that the mechanical-joined material system by proper design, such as thin tile, super compliant layer and GBST as copper alloy heat sink, can be used as the most part of PFC in the first phase of EAST device. Brazing techniques by co-operated with China Institute of Aerospace Material and Technique is now under way.

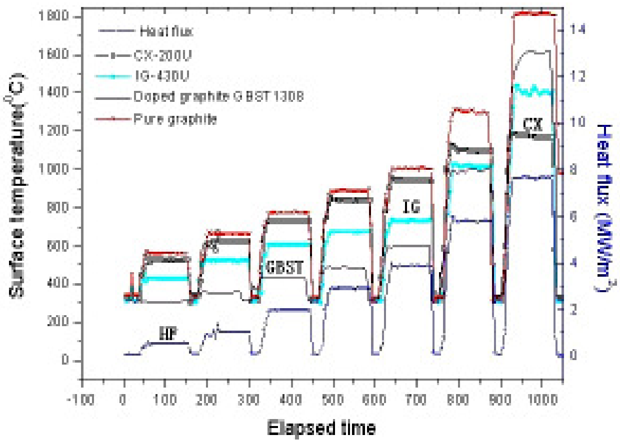


Fig.5 The surface temperatures under SS-HHF

In China two kinds of doped carbon fiber composites (CFC) have been developed, one is the 3D-CFC (I) (doped 5% B_4C), and another is 3D-CFC (II) (doped 3% B_4C). The thermal shock resistance and chemical sputtering tests show good results compared with un-doped

CFC-CFC (III). The parameters of doped CFC and their comparable materials are shown in Table 3. From Table 3, it can be seen that doped CFC has excellent mechanical and thermo-physics properties, but its thermal conductivity is not sufficient for using as high heat flux components for the EAST device. In the second phase of EAST project, high thermal conductivity of CFC should be more than 300 W/m·K at room temperature. A new CFC with SiC doped or un-doped is being developed by cooperation with the China Institute of Aerospace Material and Technique.

TABLE 3: MAIN PARAMETERS OF DOPED CFC AND UN-DOPED CFC

Item	CFC (I)	CFC (II)	CFC (III)	Remark
Density (g/cm^3)	2.07	2.03	1.97	-
Thermal conductivity (W/m·K)	120	150	180	300 K
Elastics modulus (GPa)	25	27	29	300 K
Thermal expansion coefficient ($\times 10^{-6}$/K)	0.9	1.2	1.1	300-1273K
Bending strengh (MPa)	115	97	88	300 K
Electric resistance (μΩ.m)	1.8	1.5	1.2	300 K
Porosity (%)	2.6	3.3	4.5	-

Tungsten coatings on doped graphite and CFC are also proposed to be used for the EAST device for its lower ionization potential and sputtering than carbon. In the later phase of EAST, tungsten directly coated on copper alloy and stainless steel heat sink is also under consideration. Two kinds of tungsten coatings on doped graphite have been developed. Samples with 15~20 μm Re layer as a diffusion barrier to prevent from tungsten reacting with carbon at elevated temperature were made by a double electrode glow discharge (DGD) diffusion deposition method, which is patented by China Institute of Northwest Nonferrous Metal. On the rhenium (Re) layer, tungsten coatings can be realized by chemical vapor deposition (CVD) or DGD diffusion deposition. Two kinds of sample have sustained pulsed power of 5 MW/m^2 with 0.25 Hz for several hundreds cycles, and with steady state heat flux of 5 MW/m^2 for 300 s several times. The surface temperature is even high up to 1800 ^{0}C, the coatings systems show durability and long life.

4. Research and development of CLAM

In recent years, much effort has been paid to the development of China Low Activation Martensitic (CLAM) steel in ASIPP with wide cooperation with other institutes and universities in China. The composition of CLAM, as shown in Table 4, has been proposed based on the results of other RAFMs (Reduced Activation Ferritic/Martensitic steel) in the world, such as EUROFER97, F82H, JLF-1 and 9Cr-2WVTa etc.. During heat treatment, CLAM is austenitized for 30 minutes at 980 ^{0}C followed by air cooling and then tempered for 90 minutes at 760 ^{0}C followed by air cooling.

TABLE 4: CLAM SPECIFICATIONS OF CHEMICAL COMPOSITION

Element	Fe	Cr	Mn	C	W	Ta	V	Si	N
wt %	Bal.	9.0±0.1	0.45±0.05	0.10±0.02	1.5±0.1	0.15±0.03	0.20±0.02	<0.1	<0.02
Element	Nb	Mo	Ni	Cu	Al	Co	Ti	O	
wt %	<0.001	<0.005	<0.005	<0.005	<0.01	<0.005	<0.01	<0.002	

Several kilograms of CLAM ingots have been produced in vacuum induction furnace from 2002 to 2003, and tens of kilogram ingots were produced in 2004. Some mechanical properties have been tested and analyzed, such as tensile properties, impact ductility, Ductile-to-Brittle Transition Temperature (DBTT) etc.. The tensile properties are similar with Eurofer97 and DBTT is about -100℃ [7]. In the future, the composition and heat treatment procedure will be optimized further. In addition, other mechanical properties, such as creep, fatigue, and irradiated properties etc., are going to be tested.

Hot Isostatic Pressing (HIP) is a promising fabrication technology with many advantages as a non-melting bonding technique and is selected for the ITER blanket as well. Some preliminary HIP experiments and tests on CLAM such as tensile strength, charpy impact value are ongoing. At the same time, some other welding approaches such as TIG (Tungsten Inert Gas) welding, LBW (laser beam welding), EBW (electron beam welding) etc. will be studied to find appropriate way to joining the components of blanket. A small-sized mock-up of Test Blanket Modules (TBMs) will be fabricated based on the optimized HIP conditions and techniques to establish a proper fabrication procedure for ITER TBM of China. Then a middle-sized TBM will be made and tested in the EAST machine before the full-sized ITER TBM of China is tested.

A lot of work on simulations and analysis on the activation characteristic of CLAM has been done in recent years based on (Fusion Driven System) FDS series designs [8]. The neutron spectra were calculated with MCNP/4C code and data library FENDL/2.0, the inventory code FISPACT and date library FENDL/2.0 etc. were used to perform the activation calculations with the neutron spectrum as one of its input files. In FDS-I system, the activation calculations and the impurities controlling level for CLAM were carried out. The dose rate for CLAM as the First Wall is shown in Fig.5. To meet the requirement for remote handling, the required control levels for impurities such as Nb, Ag, Ni and Ho in CLAM are relatively strict, but those for Mo and Co are slightly lower.

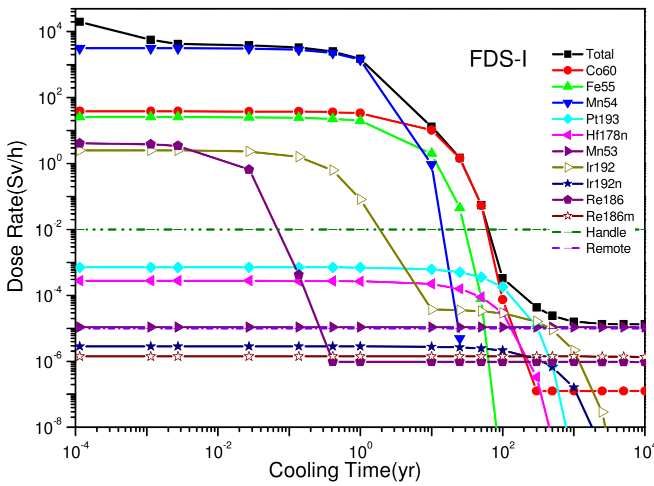


Fig.5 Dose rates of CLAM and its dominants

Activation of other RAFMs under the same irradiation conditions as CLAM in FDS design were carried out and compared with each other to see the difference among their activation characteristics [9]. It shows that there is almost no difference in total activation values among the RAFMs for shorter cooling time, but the difference among them become larger with longer cooling time.

Thin insulating coatings on CLAM steel are proposed as the solution to mitigate MHD pressure drop to acceptable levels. The coating will also be served as the tritium permeation barriers to reduce the permeation of tritium from Li17Pb83 breeder material into the coolant and to reduce the releasing of tritium to other components of tokamaks and also to the environment. It is also a kind of corrosion barrier to permit higher temperature operation. The α-Al_2O_3 layer on CLAM steel is chosen as the main coating material in the FDS designs. A

CVD method has been used to form aluminum based coatings on CLAM steels. Some coating properties, such as thickness, electrical resistance, micro hardness and density were tested [10]. The compatibility examination of the coating will be carried out in a liquid LiPb loop which is built at ASIPP soon. At the same time, other methods, such as hot dip aluminizing and self-propagating high-temperature synthesis, are underway. The coating made with these methods would have better engineering characteristics compared with CVD method and may be much more suitable for ITER and fusion reactors in the future.

5. Summary

Under the support of the China national project of the EAST superconducting tokamak, some key technologies related on fusion reactors, such as the large scale superconducting magnet, CBM for PFC and low activation martensitic steel CLAM etc., have been developed in China.

Acknowledgment

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