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REVIEW ARTICLE

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Magnetic confinement fusion: a brief review

Abstract Fusion energy is considered to be the ultimate energy source, which does not contribute to climate change compared with conventional fossil fuel. It is massive compared with unconventional renewable energy and demonstrates fewer safety features compared with unconventional fission energy. During the past several decades, never-ceasing efforts have been made to peacefully utilize the fusion energy in various approaches, especially inertial confinement and magnetic confinement. In this paper, the main developments of magnetic confinement fusion with emphasis on confinement systems as well as challenges of materials related to superconducting magnet and plasma-facing components are reviewed. The scientific feasibility of magnetic confinement fusion has been demonstrated in JET, TFTR, JT-60, and EAST, which instigates the construction of the International Thermonuclear Experimental Reactor (ITER). A fusion roadmap to DEMO and commercial fusion power plant has been established and steady progresses have been made to achieve the ultimate energy source.

Keywords fusion energy, magnetic confinement, tokamak, structural material, superconducting magnet

1 Introduction

The development of human civilization depends on the utilization of energy production and the sustainable development of modern society requires environmentally friendly solutions for energy production. Currently, more than 85% of the energy production originates from irreversible fossil fuels, i.e., oil, gas, and coal, which produce carbon dioxide and other greenhouse gases, and thus change the climate system on Earth. Development of clean as well as sustainable unconventional energy technology poses the greatest challenge to modern society. The renewables, nuclear fission and nuclear fusion are candidates to replace the current massive use of conventional fossil fuels. The renewable energy sources, such as solar, wind, and hydro energy, are intermittent and limited compared with nuclear energy, including both fission and fusion energy. Both nuclear fission energy and nuclear fusion energy are based on Einstein's famous formula $E = mc^2$, where E is energy, m, the mass, and c, the velocity of light in vacuum. Under proper conditions the light (heavy) nuclei will react to convert mass to huge energy ($E = (\Delta m)c^2$) via nuclear fusion (fission) reactions. This is the principle of atom-bomb and hydrogen-bomb which produce huge energy in a moment and in an uncontrolled mode through nuclear fission and fusion reactions, respectively. The awesome power of nuclear fission and fusion reactions can be harnessed for peaceful purpose if the nuclear reactions take place in a controlled mode. Since the first nuclear fission reactor was switched on in 1947 in the UK, more than 440 nuclear power plants have been in operation in more than 30 countries and fission reactors have produced about 16% of the global electricity and about 6% of the primary energy consumption so far [1,2]. However, the Fukushima Dai-ichi nuclear power plant accident that occurred in 2011 made people worry about nuclear safety and some nations even want to reconsider the future policy for fission nuclear energy. Compared with nuclear fission, nuclear fusion has the advantage of the absence of long-lived radioactive waste and of almost inexhaustible fusion fuels.

Fusion reactions occur in several ways, proton with proton, deuteron with triton, deuteron with deuteron, and deuteron with helium-3 (^{3}He), to name only a few. Among them, the deuteron-triton (D/T) reaction forms a helium nucleus (α-particle) and a neutron, which simultaneously produces 17.6 MeV of energy. The energy produced in the

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D/T fusion reaction is in the form of kinetic energy of the neutron (14.1 MeV) and the alpha particle (3.5 MeV). The fusion reaction rate, which is defined as a product of densities of reacted species and cross section, can be determined and the results demonstrate that the D/T fusion reactivity is much greater than that of other fusion reactions like the D/D and the D/^{3}He. Deuterium can be extracted from seawater, which exists at 0.0153% and thus is an essentially infinite fuel source. Tritium is radioactive with a half-life of 12.3 years, which rises to the challenge in reserve. Moreover, the supply must be periodically replenished in case external tritium is used as fuel in a fusion reactor. However, the capture of neutron generated from the D/T reaction in lithium (^{6}Li or ^{7}Li), which is quite abundant in nature, forms tritium. A fusion machine can breed its own fuel when blanket containing lithium is placed around the fusion chamber and thus deuterium is the unique fuel. This is the reason why pursuing D/T fusion is the principal goal of the present phase of magnetic confinement fusion research. The reserves of conventional fossil fuels, unconventional renewable energy, and nuclear fission/fusion energy are summarized in Table 1 [3]. It is clearly seen that fusion is almost inexhaustible compared with other energy sources. In addition, as tabulated in Table 2, the mass required to produce 1 GW electric power for different energy sources, fusion energy has the advantage of least mass consumed. The ultimate energy source to maintain human civilization will be the one that does not occupy much space, with a virtually inexhaustible supply, safe, neither releasing any carbon oxide into the atmosphere, nor leaving any long-lived radioactive waste. The fusion energy is considered to be the ultimate energy source.

Fusion energy is very hard to operate. There are mainly three ways for fusion reactions to occur, i.e., the gravitation confinement, the inertial confinement, and the magnetic confinement. Gravitation makes the sun shine mainly through the fusion reaction of proton with proton, based on weak interaction and the reaction exhibits very low reaction cross-section. Proton-proton reaction is impossible to be used for any practical application on earth, because the fusion reaction uniquely takes place under extreme conditions, i.e., temperature and pressure of 15 million degree and 150 billion bar respectively. The second way is fast compression of a tiny D/T fuel cell with high energy beams, for instance laser, and fusion reaction takes

Table 1 Approximate world energy estimates

Energy types		Reserves*/ZJ	Resources*/ZJ	Technical potential/ZJ (per year)
Fossil fuels	Coal	20	290–440	-
	Oil	9	17–23	-
	Natural gas	8	50–130	-
Nuclear fission	U-238 + U-235	260	1300	-
	Th-232	420	~3 × Uranium	-
Nuclear fusion	Deuterium	-	1.60×10^{10}	-
	Lithium in ocean	-	1.40×10^{10}	-
	Lithium on land	-	1700	-
Renewable Energy	Biomass	-	-	0.16–0.27
	Geothermal	-	-	0.8–1.5
	Hydro	-	-	0.06
	Solar	-	-	62–280
	Wind	-	-	1.3–2.3
	Ocean	-	-	3.2–11

Notes: *—Reserves denote those that can be recovered economically, whereas resources are greater, but may be much more expensive; 1.0 ZJ = 10^{21} J, and 1.0 ZJ = 31.7 TW-years; The total world energy consumption rate in 2010 is around 17 TW and it will be about 20 TW in 2020.

Table 2 Mass required to produce 1 $GW_{electric}$ (or 3 $GW_{thermal}$) for different energy sources

Energy sources		Mass	Area
Conventional	Coal/MT	2.7	-
	Oil/ MT	1.8	-
Unconventional	Solar plant	-	70 km^2 of solar panels
	Wind power plant	-	3000 × 1 MW windmills (~100 km^2)
	Uranium (fission reactor)/T	25	-
	Mixture of D/T (fusion reactor)/kg	350	-

place when the inertia maintains a sufficient pressure [4,5]. The third way is the magnetic confinement, which aims to achieve controlled fusion reaction on the earth as the inertia confinement. In this way, the plasma can be confined in a steady-state and it takes place at a lower pressure (several bars) but a higher temperature (150 million degree). The famous Lawson criterion on necessary confinement fusion ignition conditions, published in the early 1950s, reveals that the triple product of ion density (n), confinement time of fuel ions (τ_E), and the ion temperature (T) shall be larger than a constant related to fusion reaction. The relationship between the triple product ($nT\tau_E$) and the plasma power amplification factor Q_p, which is defined as the fusion power divided by the external heating power that must be provided in addition to the fusion power to maintain the fusion reaction, has been obtained afterwards. A high Q_p requires achievements of a high plasma pressure and a long plasma energy confinement time. $Q_\mathrm{p} > 1$ constitutes a definition of break-even, which means that the fusion reaction produces a larger amount of thermal power than the amount of external power. The ultimate goal of the fusion reactor is the achievement of a sufficiently good confinement that partial of fusion power is sufficient to maintain the plasma at thermonuclear temperature without any external power. This condition is defined as ignition with practical definition corresponding to $Q_\mathrm{p} > 1$, whereas the $Q_\mathrm{p} > 10$ criterion can be considered as a practical definition for scientific feasibility of fusion machine economically producing net electrical power. In the present paper, the magnetic confinement fusion with emphasis on device developments as well as on superconductors and structural materials challenges are concisely reviewed.

2 Magnetic-confinement fusion

As a result of Lorentz force, a charged particle moves in a helical (corkscrew) orbit around the field line in a straight and uniform magnetic field, as shown in Fig. 1. The resolution of the vectorial motion results in a circular motion and a linear motion, i.e., a motion perpendicular to the magnetic field and a motion along the magnetic field. The radius of the circular motion, i.e., the gyroradius, is inversely proportional to the intensity of the magnetic field. Configuring magnetic field can confine the macroscopically neutral collection of ions and unbound electrons-plasma. In this way, the magnetic field can be used to confine a plasma within in a space without contacting with the chamber wall. It is impossible to maintain the thermonuclear temperature for a plasma if it is allowed to come in contact with a chamber wall. The magnitude of plasma pressure that can be achieved associates with the confining magnetic field pressure magneto hydro dynamic (MHD) instability limits (β). The output power density in a magnetic confinement D/T reaction is proportional to β^2

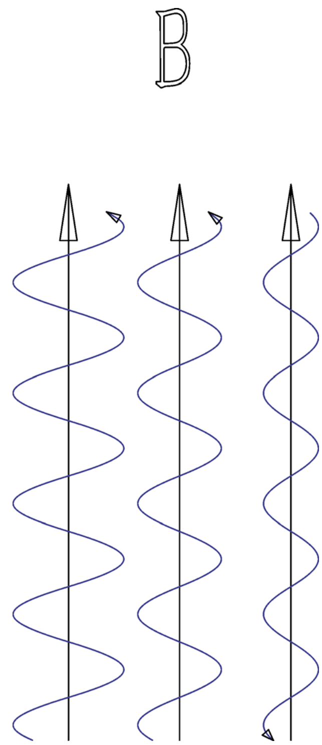

Fig. 1 Orbit of charged particles in a uniform straight magnetic field (The straight line represents a magnetic-induced field line, and the radius of gyration (also known as Larmor radius) depends on the mass of the particle.)

and B^4, where B is the magnetic field strength. Because of the limitation of permanent magnet, increasing attention has been paid to superconducting magnets. The magnetic-confinement fusion puts a severe demand on superconducting materials as well as superconducting magnet technology, which simultaneously and significantly extends cryogenic technology.

2.1 Open confinement system–magnetic mirror

The magnetic confinement falls into open confinement systems and closed toroidal confinement systems [6]. A straight cylinder with two magnetic mirrors at both ends forms a magnetic bottle of plasma, and then a simple mirror machine can be developed as illustrated in Fig. 2. The magnetic mirror concept was the strongest proposal for plasma confinement devices when fusion research initiated in the early 1950s. However, good confinement of plasma is hard to be achieved with mirror machines mainly due to the instabilities generated by end losses. The reason for this is that, only particles with large perpendicular velocity component can be reflected by the end mirrors, whereas particles with a large component along the magnetic field line are hardly reflected by the end mirrors and thus escape from the confinement, namely an open confinement system. Although the researchers in Kurtchatov Institute developed a mirror with a quadru polar field

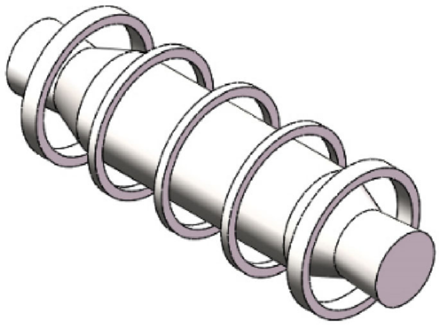

Fig. 2 Schematic diagram of a simple magnetic mirror

which improved the magnetic field configuration and impeded large scale instabilities, the "microinstablities" resulting from the depletion of particles with large velocity component along the magnetic field cannot be eliminated in a satisfactory way. Several mirror machines were developed by Russia (GOL-3, GDT, AMBAL-M), Japan (GAMMA-10), Korea (HANBIT) and the US but the magnetic-mirror approach was abandoned by the US government in 1986 after stopping the Mirror Fusion Test Facility B (MFTF-B). The main achievements of tandem mirror machines represented keV (tens of millions °C) level plasma temperature and an energy confinement time of ~0.05 s. During the past several decades, considerable efforts have been taken to improve the mirror machines and the mirror concept has demonstrated potential for a materials test facility [6–8].

2.2 Closed confinement systems — Tokamak and stellarator

Configuring magnetic field lines completely within a confinement chamber can eliminate depletion of particles as within an open confinement system. Technically, this type of magnetic confinement can be achieved by a proper choice of position and currents in a set of magnetic coils to have a configuration with a toroidal shape and this type of confinement is defined as a closed toroidal confinement system [6].

The simplest configuration of a closed confinement system is a torus in which a set of coils produce a toroidal field and particles following along the closed toroidal field lines and moving within the toroidal confinement chamber. However, this kind of curvature and non-uniformity of the toroidal field results in the drift motions of particles. The drift motions are radially outward and cause the particles to contact with the chamber wall, which leads to a plasma instability. To confine all particles, the drift can be compensated by means of configuring field lines winding around the torus instead of being a simple circular. A poloidal magnetic field can be superimposed upon the toroidal magnetic field to compensate for these drifts in such a way that a helical magnetic field which is entirely contained within the toroidal confinement chamber forms. Since the early days of fusion research, the stellarator concept and the tokamak concept have been developed.

In the stellarator concept, the necessary poloidal field is generated by external coils. The stellarator confinement concept has the advantage of having no inherent limitation to the length of operation as well as no disruptive termination of the discharge [6,9]. Early stellarator configurations lacked good confinement properties but it was optimized by means of a complex set of coils. The stellarator machine is devilishly hard to build due to the helical coils. Both the stellarator and the magnetic mirror were one of the first to be investigated but the success of the tokamak concept in the late 1960s drew researchers' attention away. Stellarator machines are currently operating in the US (CAT, HSX), Japan (LHD, CHS), Germany (W7-X, WEGA), Spain (TJ1U, TJII, UST1), and Australia (H-1). The stellarator machine NCSX was stopped by the US government because of manufacturing difficulties. The LHD located in Toki, Japan, which was the first large-sized stellarator, started its operation in 1998. The construction of the largest optimized stellarator W7-X finished in 2015 and started its operation in 2016 [9–11]. The main parameters of the LHD and W7-X are listed in Table 3. The first physics results of the W7-X have proven that the complicated and delicate magnetic topology with a required accuracy of better than 1/100000, which plays a crucial role in good confinement, can be achieved with the stellarator concept [10].

Table 3 Parameters of LHD and W7-X [12]

Parameters	LHD	W7-X
Location	Toki Japan	Greifswald Germany
R_o*/m	3.5–3.9	5.5
a*/m	0.6	0.53
B/T	2–3	3
Pulse length/s	$> 10^3$ at low density	1800
$nT\tau_E/(\mathrm{m^{-3}\cdot keV\cdot s})$	4.4×10^{19}	-

Notes: *—R_o is major radius and a is minor radius.

In the tokamak concept, the poloidal field is produced by a toroidal current flowing in the plasma, which is different from that of a stellarator. The poloidal field component of a tokamak concept results from the toroidal current following in the plasma itself, whereas the current is induced inductively, like a transformer, by varying the magnetic flux in the primary transformer coils. In a tokamak machine, the heart of the magnet system will be tens of toroidal field coils. Additional coils such as poloidal coils and central solenoid coils are also necessary to counteract the expansion forces of the plasma-current loop and of the plasma pressure, and to shape the plasma [13], as exhibited in Fig. 3. The operation of a tokamak shall be pulsed unless the electromotive force with waves/fast particles or thermos-electrical forces are exploited. The tokamak concept has been the most extensively investigated worldwide and is the most advanced. So far, tens of experimental tokamak machine operate worldwide, such as the EFTR (USA), ALCMOD (USA), DIII-D (USA), T-10 (Russia), JET (UK), ASDEX-U (Germany), ToreSupra (France), JT-60SA (Japan), EAST (China), and K-STAR (South Korea). Several scientific milestones have been achieved on tokamak machines. In 1991, the JET demonstrated the first large-scale test of the D/T reaction could produce a maximum of about of 1.7 MW in a short pulse of about 2 s (corresponding to a Q_p of around 0.15). Several years later, the fusion power of more than 10 MW with a Q_p of around 0.27 was produced on the TFTR tokamak machine. In 1997, the JET tokamak machine produced a maximum fusion power of over 16 MW with 50%D/50%T plasma (corresponding to a Q_p of 0.65). The equivalent steady-state Q_p becomes 0.94 after correcting for transient effects and this value is almost close to the break-even status. These are the highest fusion powers and Q_p values which have been produced by a fusion machine so far. Some of the large tokamak and their main achievements are presented in Table 4. It should be noted that the EAST, which was the first fully superconducting tokamak machine in the world, recorded 61 s in fully-non-inductive *H*-mode in 2016. In the *H*-mode confinement, which presents a high confinement condition compared with the normal low confinement regime (*L*-mode), the confinement time is significantly enhanced at a factor of 2 or more. In the same year, K-STAR demonstrated again that operation in high-performance *H*-mode condition lasted for more than 1 min in the tokamak concept[1)].

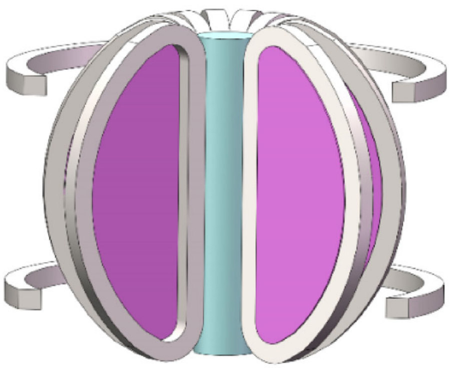

Fig. 3 Schematic diagram of a tokamak, mainly consisting of central solenoid coils, toroidal field coils, and poloidal field coils

This breathtaking achievement of tokamak machines has resulted in the construction of ITER, which naturally utilizes the best performances obtained in the tokamak machines and officially launched in 2006. The ITER, which is still an experimental reactor and will be the world's largest tokamak machine as a joint project by the EU, China, India, Japan, South Korea, Russia, and the US at a cost of over €15 billion, has a planned power output of 500 MW in pulses of 400 s and a Q_p value of 10. The main objective of the ITER are to demonstrate the feasibilities of a burning fusion plasma characterized by a fraction of self-heating maintained over 10 min and of tritium breeding from lithium. The ITER are under construction and fusion power will be generated by around 2028 [14].

Alternative confinement concepts, including the spherical torus and the reversed-field-pinch, can be considered as the optimization of the tokamak concept. The spherical torusis, basically a tokamak with a low ratio of major radium (R_o) and a minor radium (a) of 1.5, has improved the central region to maintain plasma pressure with a lower magnetic field strength than in a conventional tokamak. In other words, the design of the spherical torus improves the β value compared with that of a conventional tokamak. Moreover, the spherical torus lends itself to high plasma current driven by the plasma pressure gradients compared with a conventional tokamak. There are several spherical

Table 4 Some large tokamaks[2)] [6,12]

Parameters	Tokamaks				
	EAST	D III-D	JT-60U	JET	ITER
Location	Hefei, China	San Diego, USA	Naka, Japan	Culham, UK	Cadarache, France
R_o*/m	1.7	1.7	3.4	2.96	6.2
a*/m	0.4	0.67	1.1	0.96	2.0
B/T	3.5	2.1	4.2	4.0	5.3
I/MA	1.0	2.1	5.0	6.0	15
Main achievements	First fully superconducting machine; 61s in fully-non-inductive *H*-mode (2016).	World's record β = 12.5% for tokamak machine.	Long pulse (28 s) in a steady-state, equivalent $Q_p > 1$ (2005); being updated to "JT-60SA."	World's only tritium compatible machine; world's record fusion power 16.1 MW (1997).	Expect Q_p~10; planned fusion power 500 MW.

Notes: *—R_o is major radius and a is minor radius.

1) www.firefusionpower.org
2) www.tokamak.info

torus experiments at this moment, including NSTX, PEGASUS, and HIT-II in the US, MAST in the UK, GLOBUS-M in Russia, TST-2 and LATE in Japan, and SUNIST in China. The largest spherical tokamaks are the NSTX in the US, and MAST in the UK, both of which have a major radius of 0.85 m and a minor radius in the range of 0.65–0.85 m. The spherical torus concept drew researchers' attention and some companies even initiated construction this kind of fusion machine [15].

The reversed-field pinch concept also can be considered as a modification of the conventional tokamak concept in that it has a toroidal field produced by external coils and a poloidal field produced by an axial current induced in the confined plasma by external transformer coils. The reversed-field pinch concept characterizes a reversal of direction of the toroidal field in the outer region of the plasma compared with that in the conventional tokamak concept. The unique feature improves the plasma stability and thus relaxes the limitation on allowable plasma current. In addition, this concept introduces the feasibility of ohmic heat as well as improvement of β limits. Several reversed-field pinch experiments have been conducted, such as the MST (US), RFX (Italy), EXTRAP T2R (Sweden), and KTX (China), and experimental data obtained over two decades convincingly have demonstrated the establishment of the field reversed configuration. The MST experiment has achieved a plasma current density and ion density (n) of 0.6 MA and 10^{19} m^{-3} respectively at $\beta = 10\%$. Moreover, a maximum value of $\beta = 26\%$ has been demonstrated in the same machine [16], which confirms improvements of β limits for the reversed-field pinch concept.

The race toward the generation of fusion power has largely been a story of tokamak research so far and therefore, the magnetic-confinement fusion research is still strongly focused on tokamak development. The magnetic mirror concept and stellarator concept have never been abandoned but still require further development. After ITER, the tokamak concept has been chosen in future demonstration (DEMO) reactor designs with a planned power output of 2000–4000 MW. The DEMO can generate electrical power and test the capability of tokamak to operate reliably for extended periods of more than 80%. Moreover, the EU, Japan, and South Korea have established their own fusion roadmaps and launched conceptual design of tokamak DEMO reactors [17–19]. In 2012, the conceptual design of China Fusion Engineering Test Reactor (CFETR) initiated, which aimed at a complementary with ITER, a fusion power of 50–200 MW, a duty cycle time of 30%–50%, and breeding tritium by blanket [20]. The full superconducting magnet system of the CFETR is displayed in Fig. 4. The DEMO will be the transition before fusion power plant in a fusion roadmap, which will complete several missions such as resolving the remaining physics and technical issues in a power plant, demonstrating production of several 100 s MW of electricity, and verifying tritium breed.

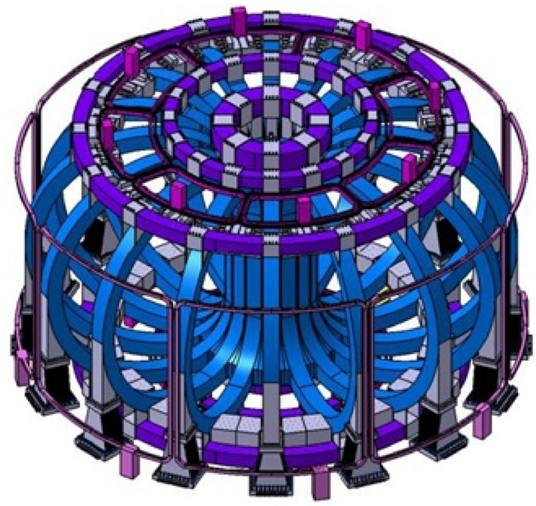

Fig. 4 Magnet system for CFETR, consisting of TF coils, PF coils, CS coils and several pairs of correction coils

The past several decades have witnessed a number of striking advances in magnetic confinement fusion, especially tokamak machines. Despite the scientific feasibility of magnetic confinement fusion has been demonstrated in JET, TFTR, and JT-60, which instigates the construction of ITER. However, the engineering, practical and economic feasibilities of fusion energy still need to be investigated and demonstrated by ITER and DEMO. There are still several engineering and practical challenges in fusion energy research to construct a practical fusion machine, for instance, the remote-handling technology for treatments of in-vessel components with radioactivity [21]. In addition, there are also several challenges in the materials aspects including both magnet and plasma-facing components.

3 Challenges of materials

3.1 Superconducting materials and structural materials of magnet

In magnetic confinement fusion designs, superconducting magnets are used to generate the magnetic fields that confine high temperature plasma. Since the output power density of magnetic confinement fusion depends on the magnetic field strength B to the power 4, high field superconducting magnets are always desired for a high-performance fusion reactor. Superconductors fall into low temperature superconducting and high temperature superconducting materials. Low temperature superconducting materials, mainly NbTi (with a critical temperature T_c of 10.7 K and an upper critical field B_{c2} of 17 T @ 4.2 K) and Nb_3Sn (T_c of 18.3 K and B_{c2} of 22 T @ 4.2 K) have currently been widely utilized in large-scale magnet systems. The poloidal field coils and correction coils of

ITER rely on NbTi strands, whereas toroidal field coils and central solenoid coils rely on Nb_3Sn strands. Two methods, the internal tin and bronze, have been developed to form Nb_3Sn superconductor. The intrinsic brittle property of the superconducting Nb_3Sn prefers wind and the react process which challenges the manufacturing magnetic coils as well as the conductor jacket material and insulating materials. The superconducting strand bundle and conductor jacket form the cable-in-conduit conductor (CICC), which is preferred option for fusion magnet [22]. Totally, around 560 T of Nbs are required for the superconductors in ITER, which is a small part of the total weight of coils. The reason for this that superconductor requires copper as stabilizer with a copper/superconductor ratio of over 1. Moreover, large-scale, high-field superconducting magnets require extremely strong structural materials as reinforcement, such as conductor jacket, and coil case.

Future fusion machines demand superconducting materials with a high critical temperature, a high upper critical field as well as a high critical current density. For the low temperature superconducting material Nb_3Al, the T_c and B_{c2} are 18.9 K and 29.5 T respectively. Moreover, Nb_3Al demonstrates an improved strain sensitivity compared with Nb_3Sn, which will be a promising superconducting material for large-scale and high-field magnet [23].

Conventional low temperature superconductors like NbTi, Nb_3Sn, and Nb_3Al suffer dramatic drops in current carrying ability at high magnetic fields. Magnets with magnetic fields above 20 T are beyond the reach of present low temperature superconductors. The high temperature superconductors, which have undergone the first generation like $Bi_2Sr_2CaCu_2O_{8-\delta}$ or $Bi_2Sr_2Ca_2Cu_3O_{10-\delta}$ (BSCCO) and the second generation like $REBa_2Cu_3O_{7-\delta}$ (*RE*BCO, *RE* = rare earth atom e.g. Y, Gd, Nd) are potentials for high-field fusion magnets [24,25]. The high temperature superconducting $Bi_2Sr_2CaCu_2O_{8-\delta}$ (Bi-2212), which is a unique cup rate superconductor that can be made into round wire and thus is suitable to develop a cable-in-conduit conductor, is a promising materials for the development of large-scale superconducting magnets with a peak field in the range of 20–30 T [26]. Moreover, high-T_c superconductors (HTS) improve the temperature margin and the REBCO is the only HTS materials which can be potentially used at magnetic fields over 12 T and at a high temperature of over 30 K [25]. $Bi_2Sr_2Ca_2Cu_3O_{10-\delta}$ (Bi-2223) tapes have been successfully used as current leads for ITER at the moment [27]. One of the obvious shortcomings of HTS is the cost and all HTS conductors are several times the cost of Nb_3Sn in volumetric cost. It has been estimated that the material cost of Bi-2212 cable which is the cheapest of HTS materials for DEMO operating at 5 K and 13.5 T is still higher than that of Nb_3Sn by a factor of 2 to 5 [25]. It is expected that low cost production will be instigated by the huge demand of fusion magnets in the future.

For a magnetic-confinement tokamak design, the field strength limits are primarily determined by the maximum allowable stresses in the structural components like the conductor jacket, and not just by the intrinsic limits of the superconductors. The requirements on the mechanical properties of ITER CS conductor jacket after wind and react treatment to function Nb_3Sn are at least 850 MPa in 0.2% proof strength ($R_{p0.2}$), 1150 MPa in tensile strength (R_m), 130 MPa.m$^{1/2}$ in a plane strain fracture toughness (K_{Ic} or $K(J)_{Ic}$) at 4.2 K based on strength and fracture mechanics design philosophies. Structural materials with at least 1500 MPa in 0.2% proof strength ($R_{p0.2}$) but maintaining good ductility and good fracture toughness at 4.2 K is required for future DEMO machines, which will pose a considerable challenge to traditional austenitic stainless steels and Ni-alloys [28]. Moreover, the effect of wind and react treatments required by Nb_3Al or Bi-2212 on mechanical degradation of conductor jacket structural materials will be a big challenge in material selection [23,29].

3.2 Plasma-facing structural materials

Materials for the first wall, the breeding-blanket, and the divert or components of tokamak machines, which will directly contact with high energy neutrons, plasma particles, and electromagnetic radiation (mainly gamma ray), require suitable high temperature strength, high resistance to radiation-induced damage, and low neutron-induced radioactivity. The radiation damage poses multi-faceted challenge to structural materials in fusion machines. The neutrons produced by the fusion reactions dislodge substantial numbers of atoms in structural materials from their lattices sites over the projected operating lifetimes of fusion reactors, which can be quantified in terms of displacements per atom (DPA). The value of neutron damage level in DPA of ITER is less than 10 DPA, whereas it will be in a range of 50–200 DPA in future DEMO and commercial reactors [30]. For ITER, the high-purity Be brazed to a copper heat sink and type 316 stainless steel was chosen as the first-wall component, and tungsten bonded to a copper alloy heat sink was chosen as the divert or component. The ITER does not have a tritium breeding blanket. However, these materials are far away from the requirements of future DEMO and commercial reactors. Extensive research results indicate that the low-activating ferritic-martensitic steels, vanadium alloys, and SiC/SiC ceramic composites are promising candidates of plasma facing advanced materials including the first wall, the tritium breeding-blanket, and the divertor [31,32]. From the safety and waste disposal aspects, the SiC/SiC ceramic composites are preferred but practical structural designs are immature [33]. The low-activating ferritic-martensitic steels are considered to be potential candidate blanket materials for ITER and CFETR due to its high strength, high heat conductivity, high resistance to irradiation, as well as low activation features [34,35].

Related research of plasma-facing components has been delayed because of historical lack of a fusion-relevant neutron source for materials testing. The reason for this is that current neutron sources including fission reactors, spallation sources or accelerator-driven systems produce neutron energy spectra that are different from those expected in a fusion reactor. For example, neutrons in fission reaction have an average kinetic energy of around 2 MeV whereas it will be around 14.1 MeV in a D/T fusion reaction. The International Fusion Materials Irradiation Facility (IFMIF), presently in its engineering validation and engineering design activities phase, will provide a high neutron intensity neutron source with a suitable neutron spectrum to fulfill the requirements for testing materials under fusion reactor relevant irradiation conditions [36,37].

4 Conclusions

The past with more than half a century has witnessed steady progresses in magnetic confinement fusion toward the goal of ultimate energy source. Although there are still several challenges in engineering, practical and economical aspects, roadmaps to fusion electricity have been established by several nations and union. Moreover, international recognition as well as international collaboration have been established, which will resolve remaining challenges and accelerate the peaceful utilization of the awesome fusion energy. The commercial fusion power would become available, as scientist Lev Artsimovich, one of the founders of the tokamak concept, said: “Fusion will be ready when society needs it, maybe even a short time before that.”

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