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Review article

Compact magnetic confinement fusion: Spherical torus and compact torus

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Abstract

The spherical torus (ST) and compact torus (CT) are two kinds of alternative magnetic confinement fusion concepts with compact geometry. The ST is actually a sub-category of tokamak with a low aspect ratio; while the CT is a toroidal magnetic configuration with a simply-connected geometry including spheromak and field reversed pinch. The ST and CT have potential advantages for ultimate fusion reactor; while at present they can also provide unique fusion science and technology contributions for mainstream fusion research. However, some critical scientific and technology issues should be extensively investigated.

PACS Codes: 52.55.Fa; 52.55.Ip; 52.55.Lf; 52.55.-s

Keywords: Spherical torus; Compact torus; Spheromak; Field reversed configuration

1. Introduction

The world magnetic confinement fusion program is entering into the ITER era. The ITER, as a next-generation tokamak, is expected to produce plasma dominated by alpha particle heating and then significant fusion power amplification. If ITER succeeds, the demo fusion reactor will be planned, naturally, based on the tokamak concept. However, even in the ITER era, various plasma confinement configurations rather than the standard tokamak are being investigated, which provide alternative approaches to fusion energy, as well as unique contribution to plasma physics. Major alternative magnetic confinement fusion (MCF) concepts include stellarator, spherical torus (ST), reversed field pinch (RFP), and compact torus (CT). In this paper, two compact MCF concepts, ST and CT will be introduced and their relevance to fusion research is discussed.

The concept of ST is quite clear. Till to now, it mainly indicates the spherical tokamak [1], which is actually the tokamak with a sphere-like shape plasma due to more compact geometry with low aspect ratio and natural D-shape cross section. The compact configuration makes the ST have some significant differences compared to the standard tokamak; therefore it is categorized as a new concept. In principle, both compact stellarator, for example, the suspended NCSX [2], and compact RFP belong to the concept of ST as well; but no such machine is in operation. Comparatively, there is a little confusing in the concept of CT. Spheromak and field reversed configuration (FRC) are always included in the concept of CT. However, sometimes the ST was also considered as a kind of CT [3]; moreover, the RFP and some linear configurations were placed into the category of CT [4]. This confusion is sometimes due to similar literal senses; while sometimes comes from the fact that the concept of CT was extended in some ongoing research/collaboration projects. A definition of CT, which is generally accepted at present, is a toroidal magnetic containment geometry, in which no conductors or vacuum chamber walls pass through the hole in the torus

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plasma. The "toroidal" geometry excludes some concepts such as the Z-pinch and the magnetic mirror, and the "simply-connected" vacuum vessel excludes the ST. Besides spheromak and FRC, some concepts such as particle rings are also dropped into the category of CT, but not being actively investigated any longer now. In Section 2, the configurations of ST, spheromak and FRC will be simply introduced, as well the basic physics due to their unique configurations.

In terms of fusion energy development, both the ST and CT have potential advantages due to their compact geometries, which reduce the unit size and, equally, the overall developmental cost significantly. In addition, as future alternative MCF fusion reactors, the ST and CT have some other advantages in physics and/or in technology. However, the ST and CT are behind tokamak and stellarator in plasma performance at present. In the ITER era, the unique fusion science and technology contribution of the ST and CT to mainstream research should be clarified. In Section 3, potential advantages of the ST and CT for ultimate fusion reactor and their possible contributions to mainstream research are discussed.

The difficulty of fusion research makes it necessary to maintain investigations of multiple MCF concepts simultaneously. The interaction and merging of ideas from different concepts are important to improve plasma performance or to induce new ideas. This point is shown in Section 4 by introducing the ideas of merging ST, screw-pinch ST and magnetized target fusion (MTF). Finally a summary will be given in Section 5.

2. Configurations and fundamental physics

2.1. Spherical tokamak

Spherical tokamak is a kind of tokamak with a very low aspect ratio (A = major radius/minor radius <2) as shown in Fig. 1. As a consequence, the cross section is naturally elongated vertically. However, its magnetic topology remains the same as that of the tokamak, with a toroidal field generated mainly by toroidal field coils and a poloidal field mainly by plasma current. The plasma current is inductively driven by ohmic field coils (or called inner poloidal field coils or center solenoid) or noninductively by neutral beam injection (NBI) or rf waves. In some STs, for example, the LATE device in Japan, the solenoid is removed, but the vacuum chamber in the center still exists [6].

However, the compact geometry makes the ST different from the standard tokamak. In the overall shape, the standard tokamak is like a donut or a wheel, with a large hole in the middle; while the ST is more like a cored apple with a slim hole. In the magnitude of fields, the toroidal field is much stronger than the poloidal field in the standard tokamak, while in the ST the poloidal field in the outer region can even compare to the toroidal field. This significant change in the ratio of field components indicates a higher edgy safety factor (that is, increasing the number of toroidal turns of the field line for each poloidal turn as shown in Fig. 1), which benefits the

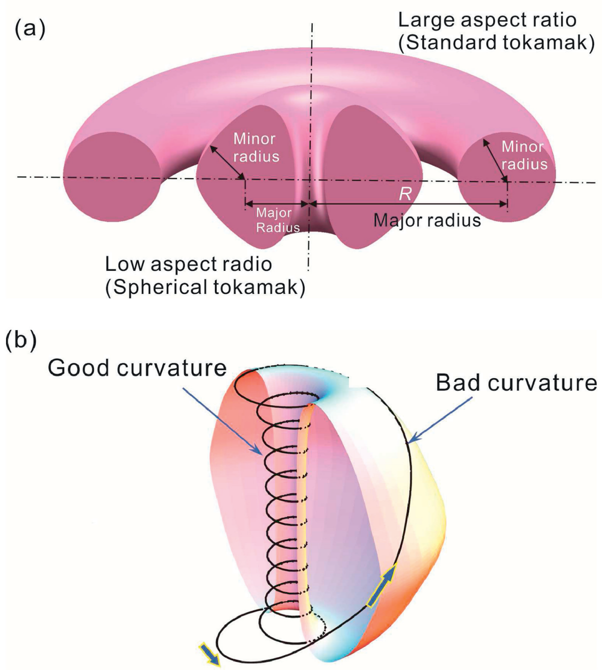

Fig. 1. (a) Comparison of spherical tokamak to standard tokamak (Courtesy of Yi Tan, Tsinghua University) and (b) the magnetic field line in spherical tokamak (Reprint from Fig. 2 of Ref. [5], Copyright 2000 America Institute of Physics).

MHD stability and/or allows larger plasma current for a given toroidal field. These benefits can be seen from the significant increase of two factors: the ratio of plasma pressure to magnetic pressure β and the ratio of plasma current to toroidal coil current I_p/I_{TF}. The feature of high β in ST can be understood from this relation,

$$\beta = \beta_N I_p/(aB_{t0}) = 5\beta_N\left(1+\kappa^2\right)/2Aq^* \quad (1)$$

where the first equality, the so-called "Troyon limit", gives a constant β_N (although in fact the ST database shows a higher β_N[7]), the cylindrical safety factor q^* can also be considered as a constant. Due to low aspect ratio, A, and large elongation, κ in the ST, the plasma β can reach tens of a percent, while it is only a few percent in standard tokamak. Another important quantity is I_p/I_{TF}, which is described as

$$I_p/I_{TF} = \left(1+\kappa^2\right)/\left(2A^2q^*\right) \quad (2)$$

Clearly, it shows that in the ST the toroidal coil field can support larger plasma current than in standard tokamak. Typically I_p/I_{TF} can reach about unity in the ST.

However, it is mentioned that the field at the plasma center are not so strong, which limits the plasma pressure even β is high. Theoretically there is the same constraint of maximum toroidal field on the inner leg of the toroidal coils in the ST and the standard design. However the field reduces dramatically over the plasma volume due to the stronger $1/R$ effect at low aspect ratio [8].

The physics of ST is also almost the same as that of tokamak. The plasma equilibrium is controlled by the Grad-Shafranov equation,

$$\frac{\partial^2\psi}{\partial R^2}-\frac{1}{R}\frac{\partial\psi}{\partial R}+\frac{\partial^2\psi}{\partial Z^2}=-\mu_0 R j_\phi=-\mu_0 R^2\frac{\partial p(\psi)}{\partial\psi}-\frac{\partial I^2(\psi)}{2\partial\psi} \quad (3)$$

where ψ is the poloidal magnetic flux; p and I/R are the pressure and the toroidal field, respectively. The MHD stability in the ST is rather good, however, more Alfven eigenmodes and energetic particle modes are easily excited due to low Alfven velocity $V_A = B/\sqrt{\mu_0\rho} = v_{ti}/\sqrt{\beta}$ [9,10].

Two specific physics are especially focused on. One is the noninductive plasma startup and current drive, which are two related but different problems. Due to little space for central solenoid, research on non-inductive plasma startup is urgent in spherical tokamaks. Approaches may include compression and merging [11], helicity injection [12] and rf assistance [5,13]. In addition, the ramp-up of plasma current and simultaneous heating from the initial startup phase are also challenging. The problem of current drive is the same as in tokamak, mainly by NBI or rf drive [14]. However, with a relative high density and a low magnetic field in ST plasmas, the Alfven velocity is lower approaching to the ion thermal velocity and plasma frequency exceeds the electron cyclotron frequency and, therefore, the noninductive current drive becomes more difficult than in standard tokamak. The low hybrid current drive (LHCD) and electron cyclotron current drive (ECCD), which are successfully applied in standard tokamak, encounter difficulties in plasma accessibility. Efforts of special design of LHCD for better accessibility are underway. The high harmonic fast wave current drive (HHFWCD) and mode converted electron Bernstein wave current drive (EBWCD) are proposed, but the results are not always robust. The other big physics in the ST is plasma transport. Although the ST has reasonably good energy confinement as the tokamak, the dominant transport processes seem to be somewhat different. The transport in the ST is mainly through the electron channel [15,16], while the ion-channel is more important in standard tokamaks. The reason might be that the strong $\boldsymbol{E}\times\boldsymbol{B}$ shear in the ST stabilizes the ion–scale turbulence. On the mechanism of electron transport, micro-tearing modes seem to play a more important role in ST [17] besides drift modes. Moreover, since in present STs the magnetic field is rather weak, the transport level is a little bit high comparing to that in present large tokamak devices.

There are more than twenty ST devices operating worldwide, with the number of machines possibly only smaller than that of standard tokamak and plasma performance lower than tokamak and stellarator. The NSTX [18] and MAST [19] are the largest two facilities with plasma major radii of about 1 m, mega-ampere plasma currents, toroidal field of 0.5–0.6 T. Comprehensive study of ST physics and engineering are performed in these two machines. Now an upgrade of NSTX was just finished and the MAST is undergoing its upgrade, both with the aim of toroidal field up to about 1 T, plasma current of about 2 MA, and advanced divertor for better power exhaust. Other medium-size and small-size devices in the world include Globus-M [20], QUEST [21], Pegasus [22], LTX [23], HIT-II [24], TST-2 [25], LATE [5], TS-3/TS-4 [26], UTST [27], HIST [28], SUNIST [29], VEST [30], ETE [31], Multi-Pinch [32], GLAST [33] and so on, which investigate the plasma startup and current drive, particle recycling and thermal flux control, fundamental physics of toroidal plasma with low aspect ratio, and so on. The physics and recent progress of ST can be found in those overview papers [5,7,34].

2.2. Spheromak

Spheromak is another kind of toroidal confinement with both toroidal and poloidal magnetic fields, as illustrated in Fig. 2. It is easy to confuse spheromak with spherical tokamak due to the similar names. However, there are significant differences between these two concepts. Firstly, in the shape, spheromak plasma has a simply connect surface without external toroidal field coils and a vacuum vessel in the center part. If we said the spherical tokamak like a cored apple, the spheromak is a core-less apple. In principle, the aspect ratio of spheromak can reach unity although the central part is not the confinement zone in actual experiments. Secondly, the spheromak has no toroidal field (TF) coil; therefore the toroidal field is generated completely by plasma currents, although the external magnet is usually used when the spheromak is initially generated. As a result, the toroidal field vanishes at the outer surface. In other words, the current is totally toroidal at the core and totally poloidal at the surface. Also it is natural that the toroidal field has the same order of magnitude as the poloidal field. Thirdly, the magnetic field and current are almost parallel in spheromak, which results in no electromagnetic force in the plasma, $\nabla p = \boldsymbol{j}\times\boldsymbol{B} = 0$. This indicates that the spheromak plasma has relaxed to the minimum-energy state under the conservation of magnetic helicity $K \equiv \int \boldsymbol{A}\cdot\boldsymbol{B}\mathrm{d}V$, where $\boldsymbol{A}$ is the vector potential (noted the scalar A is the aspect ratio). This minimum-energy state is the so-called Taylor relaxation state with $\nabla\times\boldsymbol{B} = \lambda\boldsymbol{B}$ [35,36]. In fact, taking $\partial p/\partial\psi = 0$ in Eq. (3), one can get

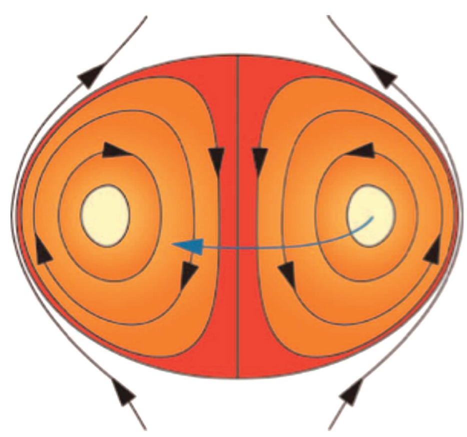

Fig. 2. Spheromak configuration. (Courtesy of Lawrence Livermore National Laboratory).

$$\frac{\partial I(\psi)}{\partial \psi} = \frac{\boldsymbol{J}}{\boldsymbol{B}} = \text{const.} \tag{4}$$

Therefore $I(\psi)$ is proportional to the flux function and then the Grad-Shafranov equation can be analytically solved with certain boundary conditions.

It is easily understood that there is no free energy in Taylor state with uniform λ and therefore it is always MHD stable. However, when λ is not uniform, for example, due to different decay rates of poloidal and toroidal fields, current driven kink modes will be excited to release free energy towards the final Taylor state. The axisymmetric force-free equilibrium results in a non-zero but small safety factor, $q < 1$. Therefore the suppression of current-driven instability needs the response from the conducting wall. Also, a realistic spheromak has finite pressure. The pressure-driven modes are stabilized by magnetic shear in spheromak, which give an MHD beta limit on the order of 0.1.

The formation of spheromak is a process of injecting linked toroidal and poloidal magnetic fluxes (i.e. magnetic helicity) into a vessel with conducting walls and then rearranging the fluxes into the spheromak configuration with fixed helicity. The commonly-used methods [37] include magnetized coaxial plasma gun with poloidal field and radial discharge current, Z-discharge theta pinch with toroidal inductive current and axial discharge current and flux core with both toroidal and poloidal inductive currents.

Due to the absence of TF coils, it is difficult to generate large toroidal fields in spheromak. In the SSPX spheromak experiments [38],with mega ampere toroidal currents magnetic fields of 1 T have been generated and tokamak-like transport was measured by suppressing fluctuations, including core electron thermal conductivities in the range of 2−10 m^2/s, and electron temperature peak value of 0.5 keV. It is acceptable good confinement although not as good as in advance scenarios of tokamaks. Also, due to the absence of ohmic coils, it is also difficult to obtain long pulse discharges. In other words, a good confinement is hard to achieve simultaneously with an efficient current drive. In fact, the drive of plasma current in spheromak needs to break the magnetic surfaces by the dynamo effect; while this will induce high-level heat losses. This intrinsic problem is still key issue in spheromak research.

Spheromak is easily formed in laboratories; therefore, there are many spheromak experiments in the world. The largest spheromak device is the closed SSPX [39,40] and other major devices may include SPHEX [41], BCTX [42], SSX [43], CTIX [44], HIT-SI [45], Caltech spheromak [46], and TS-3/TS-4 [47]. Spheromaks have been thoroughly reviewed in the book [48] and review papers [49].

2.3. FRC

The FRC is a toroidal confinement with poloidal magnetic field only. Like spheromak, magnetic fields in FRC are generated almost entirely by plasma current and no external toroidal field coil and vacuum vessel is inserted in the plasma. The FRC is usually cylindrical in the overall shape, more like a magnetic mirror rather than a toroid. However, the concept does not originate from the idea to locally reverse the axial field in a mirror to mitigate end-losses, but in experiments with theta pinches. A schematic drawing of FRC is shown in Fig. 3.

The bulk currents of the FRC are diamagnetic, leading to a high beta approaching unity. However, the equilibrium in FRC is more complicated than in spheromak or in tokamak. It is noted that the FRC's magnetic helicity is zero and thus its Taylor state corresponds to a vacuum field without any pressure and flow. The Grad-Shafranov equation in Eq. (3) cannot describe the FRC equilibrium as well. Therefore the advanced theory rather than the MHD theory, for example, the two-fluid theory or kinetic theory, is required. With the conservation of a general helicity $K \equiv \int (\boldsymbol{A} + m\boldsymbol{v}/q) \cdot (\boldsymbol{B} + m\boldsymbol{\Omega}/q) \mathrm{d}V$, a two-fluid theory can predicts a relaxation state with finite pressure and sheared flows, which can roughly reproduce the FRC characters from experimental observations [51−53].

FRC's magnetic topology has another significant feature, that is, an averaged "bad curvature" without rotational transform and magnetic shear. Therefore, the FRC should be unstable to most ideal MHD modes, including rotational instabilities due to the centrifugal effect of rotational flow and tilting instabilities driven by bad curvature. However, the robust stability can be obtained in some experiments [54−56]. Finite Larmor radius (FLR) effect possibly contributes the stability [57,58], but this effect is not effective in large-size FRC with high s, the ratio of plasma radius to the ion Larmor radius. Sheared flows [59,60] and large orbit of energetic ions [61] may improve stability at large s, but theoretical explanations and experimental evidences are still not well convincing. The FRC provides a big challenge in equilibrium and stability of plasmas at the extreme.

FRC configuration can be formed by different methods. The conventional idea is using theta-pinch technology [62,63]. After pre-ionization, a large electric field is applied to reverse the axial field. The name of "field reversed configuration" just comes from this character. An easily confused concept is the so-called RFP, reversed field pinch, which is an axisymmetric torus like the tokamak but the toroidal field reverses its direction at the edge of the plasma. Accompanying with the radial compression, the field line in the end is reconnected to

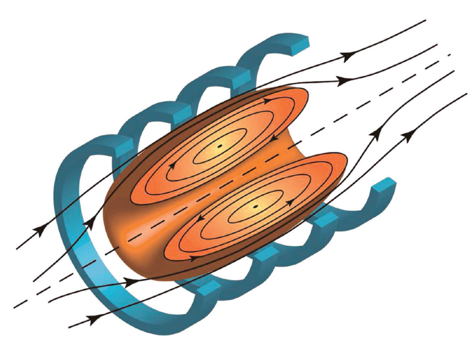

Fig. 3. Field reversed configuration (Reprint from Figs. 2−5 of Ref. [50]).

form closed surface; and then the FRC equilibrium is established with further axial contraction. In this process ion temperatures are typically high, usually with a few keV, due to reconnection heating and shock heating in fast implosive formation. However, this kind of FRC has a quite short duration due to the limitation of poloidal flux provided by theta-pinch. Another method to form FRCs is by merging two spheromaks with oppositely directed helicities [64]. Since the magnetic reconnection converts magnetic energy to thermal energy, high beta is obtained with the formation of FRC. However, the merging process is also pulsed and the flux limitation is also expected. The third technique to form FRCs is driving the cross-field current with rotating magnetic fields (RMF) [65]. In practice, a steady-state FRC can be formed and sustained by RMF, where the temperature is lower than the reconnection cases [66]. The strong tangential NBI can also be used to drive the cross-field current in a mirror and then form and sustain a long-lived FRC configuration [67]. Here, the tangential NBI also provide the large-orbit energetic particles to stabilize the plasma.

The major FRC experiments may include TCSU [66,68], PFRC [69], FRX [70], Colorado FRC [71], SSX [72], PHD [73], MRX [74], TS-3/TS-4 [48], Rotamak-FRC [65] and so on. Also, some experimental programs in private companies also use the FRC concept [67]. The FRC concept is reviewed in some papers [75,76]. An overall introduction to CT, including FRC and Spheromak, can also be found in some books [77] or review papers [78].

2.4. Relation between ST and CT

The relation between ST and CT was previously described in Ref. [5]. Intuitively the outboard magnetic field line of the ST becomes similar to that of the CT, though the inner field line quite different. Moreover, toroidal field coils vanish in CT. This trend is partially consistent to the fact that the I_{TF}/I_p, shown in Eq. (2), can approach a small value when the aspect ratio is reduced to unity and the elongation is increased in ST. Therefore, the ST plasma shows some self-organization behaviors sometimes. Some device, such as Pegasus [22], a ST with an extreme low aspect ratio, is used to explore this physics at extreme cases.

In fact, if a central rod is inserted into a CT or a toroidal field is pre-existing, a ST-like configuration is formed. As a result, the stability of the plasma is improved, which was observed in some CT experiments [65,79−82]. Some device, such as TS-3, where the ST, FRC and spheromak configurations can be generated and transformed, is suitable to investigate the physics during the conversion process. On the other hand, considering a plasma arcing current to replace the central rod, a new ST concept with a simple-connected geometry is reached [32], which will be introduced in Section 5.

When a strong toroidal field is applied or a ST plasma is pre-existing, the CT plasma can be injected and merge to the ST plasma as a method of fueling or current drive, which will be described in the next section.

3. ST and CT for fusion

Considering the ultimate fusion reactor, the ST and CT have a similar potential advantage due to their compact geometries to reduce the unit size and, equally, the overall developmental cost significantly. Especially in the CT, removing toroidal field (TF) coils and simply-connected chamber seems to make the engineering so simple [83]. There are other scientific and technical advantages for specific configuration. For example, in FRC the cylinder geometry with a linear divertor outside the separatrix may ease the engineering constraints for impurity control and power exhaust; high beta in FRC and ST may lower the requirement for external magnets for burning plasma; good MHD stability in ST may improve the safety margin against disruptions, and so on.

However, there are realistic trade-offs for these potential advantages. For ST, the efficient techniques for non-inductive start-up and current drive are more urgent and stronger

Table 1
Advantages and disadvantages of ST, spheromak and FRC for fusion.

	ST	Spheromak	FRC
Scientific advantages	• High beta • Good MHD stability • Good confinement	• Force free	• High beta (~1)
Scientific disadvantages	• Difficulty in non-inductive startup and current drive	• Relative low plasma performance • Difficulty in the coexistence between good confinement and effective current drive	• Relative low plasma performance • Difficulty in realizing long life time for high pressure plasma • Unclear physics in equilibrium, stability and transport
Technological advantages	• Compact • High effective TF coils	• Compact • Simply-connected vacuum vessel and no TF coils	• Compact • Simply-connected vacuum vessel and no TF coils • Natural linear divertor
Technological disadvantages	• Relative low magnetic field at axis • Narrow center post • Intense wall loading		

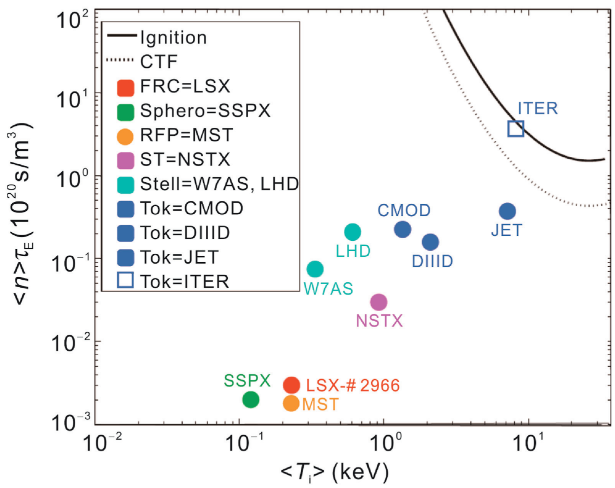

Fig. 4. Plasma performance achievements of various concepts (Reprint from Figs. 2–6 of Ref. [50]).

toroidal fields are also required to increase the plasma performance further. For spheromak and FRC, many questions regarding the stability, sustainment, and confinement need to be solved as mentioned in the last section. As a common issue, the combination of compact geometry and high power density provides great challenges on heat removal and radiation shielding. The advantages and disadvantages of ST, spheromak and FRC in science and technology are listed in Table 1.

The most important fact is that the tokamak plasma has a superior performance than other concepts, which is clearly shown in Fig. 4. It is almost the only key issue when the fusion ignition is still not realized. This status is, of course, due to the fact that the tokamak has more extensive research programs in history and now operating largest size machines; but it should not be forgotten that this choice was also decided depending on scientific achievements of each options in history. Anyway, at present the ST and CT are treated as "alternative concepts" to the tokamak. Other promising alternative concepts include the stellarator and RFP. In US FESAC's report in 1996 [84], five stages of fusion development in each concept are clearly stated, that is Concept Exploration; Proof-of-Principle; Proof of Performance and Optimization; Fusion Energy Development; and Fusion Demonstration Power Plant. At each stage, one should face different scientific and engineering problems. It is not significant to evaluate the advantage and disadvantage of one concept beyond its developing stage. The tokamak is moving from the third stage to the fourth stage, however, other concepts are behind. The ST is in the Proof-of-Principle stage, while the CTs are probably still in the Concept Exploration stage. It is a pity that the status seems not to be changed much after two decades.

The development of one concept from one stage to next stage needs increasing finance and human power support, usually marked by larger-sized machines. However, at present the mainstream fusion research is on the tokamak. The ITER and large domestic tokamak program have occupied larger resources. In this case, alternative concepts, such as the ST and CT, should find or prioritize its unique fusion science and technology issues for mainstream research, rather than its potential advantages as a fusion energy concept.

In fact, two planning activities: Fusion Energy Sciences Advisory Committee (FESAC), Toroidal Alternates Panel (TAP) [51] and The Burning Plasma Organization Research Needs Workshops (ReNeW) [85] were performed under the requirement of US DoE about ten years ago. Although these activities are initially for supporting the evaluations and strategic decision in fusion science and technology in US, the critical issues described in these documents are good references for all of us. For example, in FESAC-TAP report in 2008 [28], ITER-era goals are proposed and evaluated for each concept. For the ST, the ITER-era goal is "*to establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility that provides high heat flux, neutron flux, and duty factor needed to inform the design of a demonstration fusion power plant*". This goal aims to support the mainstream fusion development beyond the ITER. Therefore, the key physics and technology closely related to the design of component testing facility (CTF, now this concept is extended to fusion nuclear science facility FNSF) are emphasized, such as noninductive startup and current drive in over-dense plasmas, power exhaust by innovative divertors, electron energy transport at high temperature and low collisionality and single-turn TF magnet and insulators with high neutron fluence. There are three major ST-based FNSF design studies [86–88], with quite different engineering considerations. For CT, the ITER-era goal is "*to demonstrate that a compact toroid with simply connected vessel can achieve stable, long pulse plasmas at kilovolt temperatures, with favorable confinement scaling to proceed to a pre-burning CT plasma experiment*". In fact, this goal has already far from the mission of CT-based fusion reactor, but it is still evaluated as a "*highly ambitious*" goal since a great deal of issues such as MHD stability, confinement, and sustainment should be dealt with both experimentally and theoretically.

In fact, CT plasmas have many interesting physics, and some can contribute to mainstream fusion research. One example is helicity injection. The toroidal current is built by helicity injection in spheromak, while the helicity injection mentioned has been used for startup in ST [12]. Another example is CT injection to toroidal plasma. This method can be employed to fuel a tokamak [89,90] and to mitigate the tokamak disruption if high Z species are used [91]. Also, the injected CT may change the local magnetic field of the target plasma [92] or trigger a shear flow [93] and provide a method to control plasmas profile. Understanding and advanced application of these innovative ideas may be strong motivations for CT research.

4. Concept interaction and improvement

The fusion community, with no doubt, benefits from the coexistence of multiple fusion concepts. There are many examples in history that the ideas from one concept can be borrowed to apply to another concept. For example, the divertor, originally in the stellarator, now is widely employed

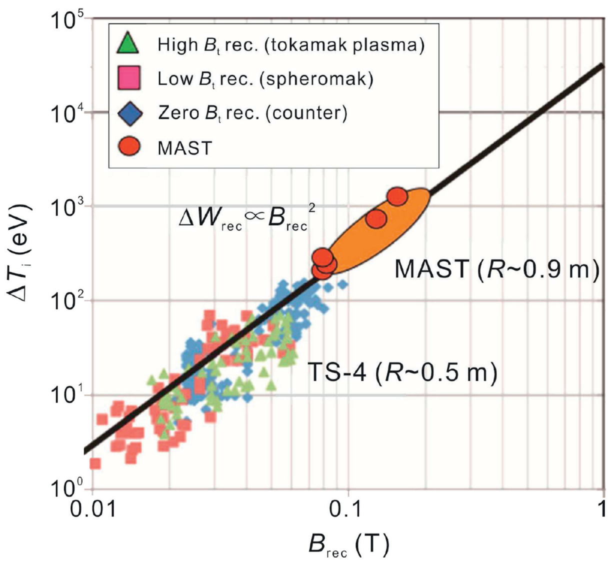

Fig. 5. The scaling law of ion temperature increments versus the reconnecting field in the merging tokamak experiments (Reprint from Fig. 14 of Ref. [95]. Copyright 2015 America Institute of Physics).

in the tokamak for particle and heat flux control; the axis-symmetry is referred to design the quasi-symmetric stellarator with optimized transport. As mentioned in the last section, the method of injecting CT to tokamak plasma for current drive, fueling or immigrating disruption is also an outstanding example. Furthermore, mutual reference and integration among different concepts may propose new innovative ideas for fusion. Here three activities are presented, that is, merging ST, multi-pinch ST and MTF.

4.1. Merging ST

Merging is a usual method to form CT and it is also demonstrated to form the ST [11]. In small device, one may pay much attention to the evolution of magnetic configuration (i.e. current drive) during the reconnection although the compression heating effect has been also observed. Experiments on TS-3/TS-4 predicted a scaling law [94] that the ion temperature increment increases with the square of reconnecting field, which is roughly the poloidal field. Recent experimental results on the MAST [95] agree well with the same scaling line, where the magnetic field is higher by one order roughly, which is shown in Fig. 5. This gives us more confidence that, if the poloidal field can be over 0.5 T, it is possible to increase ion temperature over 10 keV to burning plasma only depending on the reconnection heating. In this process the toroidal plasma is formed and heated simultaneously, somewhat like the high efficient inductive method. If it is realized, the requirement for heating physics and technology will be significantly released, even removed completely.

Here, physics involved the reconnection heating should be focused. Programs to upgrade the TS-3/TS-4/UTST are approved to investigate the physics. A private company proposed a more ambitious experimental program to build a high

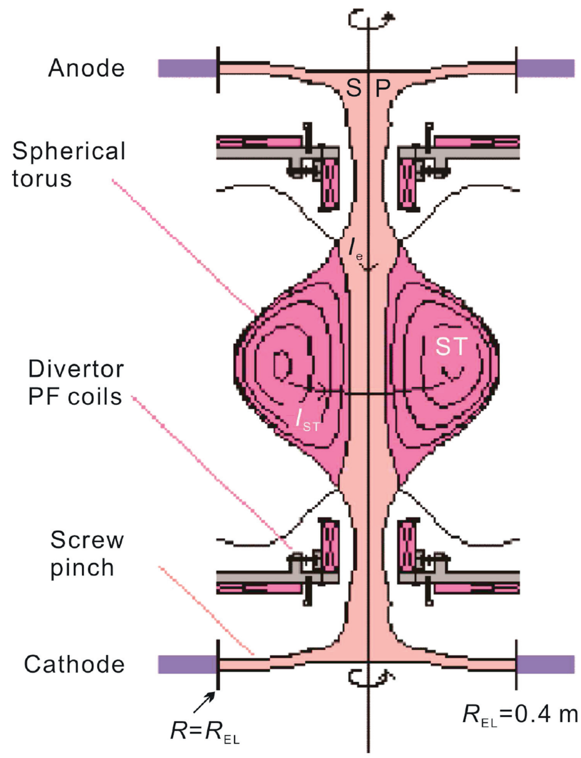

Fig. 6. Schematic of PROTO-SPHERA (Reprinted from Fig. 1 of Ref. [32]. Copyright 2006 International Atomic Energy Agency).

field ST based on high temperature superconductivity technology, where the toroidal field is designed to 3 T and the plasma current to 2 MA, and then the ignition is expected [96].

4.2. Screw-pinch ST

The ST has a central post for toroidal field coils and central solenoid, which induces some difficulties in engineering. The CT has a simply connected configuration but its plasma performance cannot be as good as that of ST. In Italy, a new type of ST, named as PROTO-SPHERA, was proposed [32], where a plasma arc, shaped as a screw pinch, was driven by electrodes to replace the central post current. The schematic drawing of PROTO-SPHERA is shown in Fig. 6. Since the central current is not the plasma current inside the closed flux surface, its configuration is more like ST rather than spheromak, although it has a simply connected configuration as well.

The first phases of the PROTO-SPHERA were commissioned recently [97], which was only a primary attempt to form a ST configuration. Furthermore, the efficiency and stability of pinch formation should be investigated, as well as the impurity problem.

4.3. Magnetized target fusion (MTF)

For obtaining high pressures in MCF, the usual idea is to improve its confinement time and/or to enlarge the machine size due to the existence of density limit. The idea of ICF is to

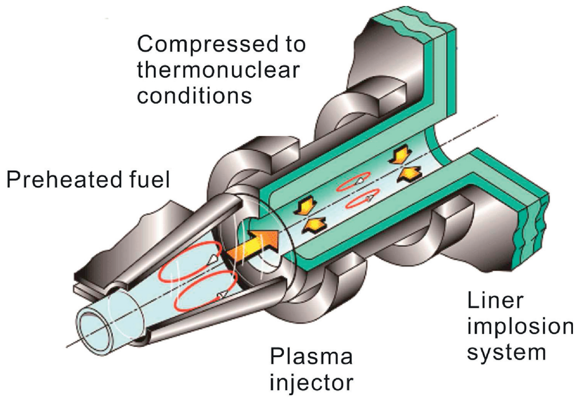

Fig. 7. Schematic of MTF concept. (Courtesy of G. A. Wurden, Los Alamos National Laboratory).

compress the small size plasma to ultra-high density. However the stability degrades the compression efficiency. The so-called MTF is the idea that involves a liner imploding system for inertial containment, compressing and heating a target of magnetic confine plasmas [98]. It is easily estimated that at higher density the requirements on transport and total thermal energy are much relaxed comparing to the MCF [99]. Also, acknowledging to partial contribution of magnetic confinement the plasma density need not to be compressed to the ultra-high density as in ICF, therefore, the least-understood physics and/or simpler inertial containing engineering are possible.

The CTs, especially the FRC, without external coils and vacuum vessels inside, can be considered as good candidates to be target plasma for implosion. The FRC have a high-beta equilibrium and usually a straight cylinder shape. It may be formed and guided into a liner with little loss of plasma energy by adjusting the bias fields, which is shown in Fig. 7.

It is an innovative idea but still many uncertain issues exist in the road to ultimate fusion energy development. Besides the challenge on linear imploding technology, there are similar scientific problems in standard FRCs, for example, to ensure the stability during forming and compression; to generate FRCs targets with hot temperature and sufficient flux; and to control particle inventory for obtain very high initial densities. For example, the initial density of MTF is 10^{23} m^{-3}, which is much higher than the value reached in present FRC experiments. The lifetime of FRC is also a problem since the steady FRC technology seems difficult to be applied in the MTF. Efforts both from the compact-torus research and the liner explosion technology are needed.

A program named ARPA-E [100] was approved in US from 2009 for investigating the MTF concept, including spherically imploding plasma liners for target compression, staged magnetic compression of FRC, ion beam driver with micro-electromechanical systems technology, staged Z-pinch target for fusion, fuel magnetization and laser heating tools for magnetized liner inertial fusion (MagLIF) concept, and plasma jets into heavy gases or metal walls.

5. Summary

The ST and CT are promising MCF alternatives since the compact geometry can reduce the unit cost of fusion reactor significantly after their intrinsic scientific and technology issues are solved.

The ST is a tokamak with low aspect ratio, which obtained similar performance as standard tokamak with similar size. The physics that the ST faces is also similar to tokamak physics, but more challenging in noninductive plasma startup and current drive, electron thermal transport and heat handling. Researches on these topics should be addressed for the ST community. Moreover, the FNSF based on the ST characters of high heat flux and neutron flux may contribute significantly to mainstream fusion research.

The CTs, including the FRC and spheromak, occupy unique regions of fusion configuration and then provide many interesting physics such as plasma relaxation, reconnection and turbulence. These configurations and physics involved not only induce some potential advantages for fusion, such as high beta and easy engineering, but also result in some critical issues such as long life time for high performance plasma which prevent the CT concept being practical fusion reactors. For the FRC, the most important issue is to achieve and to understand the stability at high s. To increase the plasma performance by reducing the transport and to enlarge the lifetime of FRC plasma without using RMF are also very critical. For spheromak, the current drive accompanying with good confinement, which is critical for plasma sustainment and realization of high magnetic fields, should be concerned. On the other hand, the CT research can still play active roles in fusion research through the contribution such as helicity injection current drive and fueling. Extensive research on this topic is also needed.

From the ST and CT concepts, more innovative concepts may be proposed based on the mutual reference and integration among different concepts. However, overall, it is the most important thing to improve the plasma performance in the ST and CT.

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