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ScienceDirect

Matter and Radiation at Extremes 1 (2016) 153–162

MATTER AND RADIATION AT EXTREMES

www.journals.elsevier.com/matter-and-radiation-at-extremes

Review article

Compact magnetic confinement fusion: Spherical torus and compact torus

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Received 5 April 2016; revised 18 May 2016; accepted 20 May 2016
Available online 7 June 2016

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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Peer review under responsibility of Science and Technology Information Center, China Academy of Engineering Physics.

http://dx.doi.org/10.1016/j.mre.2016.05.004

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 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,

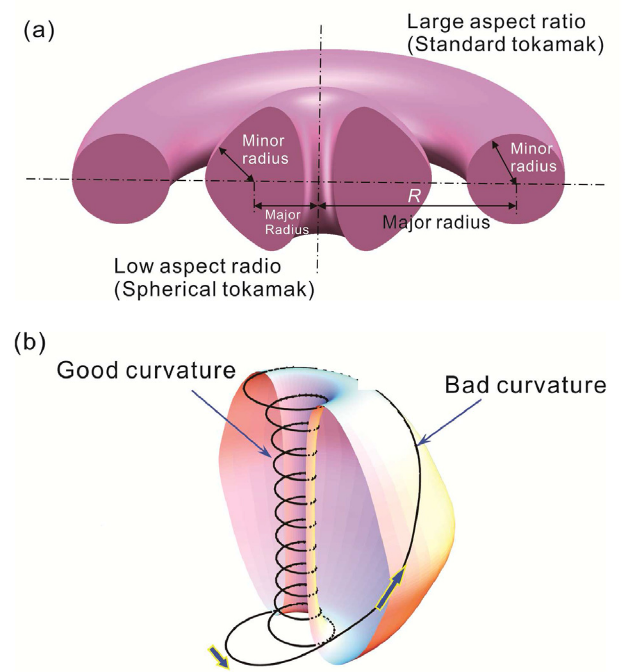

Fig. 1. (a) Comparison of spherical tokamak to standard tokamak (Courtesy of Yi Tan, Tsinghua University) and (b) the magnetic field line in spherical tokamak (Reprint from Fig. 2 of Ref. [5], Copyright 2000 America Institute of Physics).

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

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

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

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].