TOKAMAK ACCELERATOR

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Abstract

Tokamak accelerator within plasma is analyzed to be implemented in existing machines for speeding the development of fusion energy by seeding fast particles from external high current accelerators — the so-called two-component reactor approach [J. M. Dawson, H. P. Furth, and F. H. Tenney, *Phys. Rev. Lett.* 26, 1156 (1971)]. All plasma particles are heated at the same time by inductively-coupled power transfer (IPT) within an energy confinement time. This could facilitate the attainment of ignition in tokamak by forming high-gain high-field (HGHF) fusion plasma. Tokamak as an accelerator could decrease the circulating power of fusion power plant and HVDC voltage of the external accelerators by simply inserting in-vacuum vertical field coils (IVC) within its vacuum vessel, as is suggested on Experimental Advanced Superconducting Tokamak (EAST) for designing efficiency China Fusion Engineering Test Reactor (CFETR).

INTRODUCTION

Up to 1h long-pulse accelerators with 40A beam current at 1MV are designed for externally heating ITER plasma — safety concern occurs due to its 1MV extra-high-voltage design[1]. The compressed plasma by Magnetic Compression (MC) is thus suggested as High-Gain High-Field (HGHF) plasma for efficiency tokamak heating [2]. In work here, ion accelerator within the plasma is further analyzed as the second accelerating stage for designing China Fusion Engineering Test Reactor (CFETR) with much low external accelerating voltage and circulating power. The safety problem could thus be mitigated by the two-stage accelerating scheme.

Fast particles are further accelerated by Magnetic Compression (MC) in tokamak major radius [3], as is one step of HGHF plasma. Physical process of HGHF is formulated for studying the efficiency plasma on Experimental Advanced Superconducting Tokamak (EAST). EAST can simulate the HGHF fusion plasma of CFETR for high efficiency tokamak with low circulating power requirements by upgrading EAST with MC technology.

MC ANALYSIS TO TOKAMAK ACCELERATOR

Transformer model is not only used for designing tokamak, but also used for analyzing its discharged plasma current as its second winding in real time condition. Recent study [4] suggest the ratio of its bootstrap current, i.e. the fraction of self-generated current in plasma current could reach over 85% by external heating and current drive (H&CD) — it is the nonlinear and non-inductive part of the plasma current. DC mode of operation is thus arrived at the tokamak transformer when the fraction is driven to 100% by the inductive and non-inductive effects, i.e. external H&CD for high plasma pressure where no inductive fraction exist and plasma current is free of inductive limitation with zero-loop-voltage and VS consumptions. Limited by plasma boundary safety factor q, the flat-top amplitude of plasma current and plasma normalized pressure β determine the output power of the tokamak fusion machine where MC could save lots of expensive external H&CD power by simultaneously enhance plasma temperature, density and confinement time, as suggested for EAST in [2]. The safety factor q is the number of toroidal transits of a field line for one poloidal transit [5]. For CFETR, boundary safety factor q is designed to be $\geqslant$ 3.

Formulas for Compressed Plasma

The equivalent circuit of compressed plasma is an inductor- series-connected with a resistor, similar to a lossless superconductor circuit while neglecting its resistance. Because the flux in such a circuit is conserved or constant [2-3], the equations specifying the conservation of toroidal and poloidal flux as well as plasma entropy are derived as

$$\mathrm{a}^2 B_t = const. \qquad (1)$$

where B_t is the toroidal field (TF) and a is the horizontal minor radius of the plasma ellipse model.

The safety factor is

$$q = const. \qquad (2)$$

As the temperature and density constraints of the collisional plasma compression,

$$T n_e^{-2/3} = const. \qquad (3)$$

$$\mathrm{L_p I_p} \cong const. \text{ if } R_p = 0 \qquad (4)$$

where L_p and R_p are the plasma inductance and resistance in an electric circuit with plasma current I_p. L_p=L_i+L_e. L_i and L_e are respectively the internal and external inductances of the current bordering at the plasma boundary of its current flux [2].

After compression, other plasma variables scaling with the major (R) or minor (a) radius are derived as follows [2]:

Poloidal field (PF)

$$B_p \propto \mathrm{a}^{-1} R^{-1} \propto C_a C_R \qquad (5)$$

Where C_a and C_R are, respectively, the compression ratios of the plasma in the minor radius and major radius by external MC fields.

Equation 5 predicts that the poloidal field of the plasma could be linearly-extended to a high value by compressing the plasma in minor or major radius for HGHF. The plasma parameters are derived as [2],

Density $$n_e \propto a^{-2}R^{-1} \tag{6}$$

Temperature

$$T \propto n_e^{2/3} \propto a^{-4/3}R^{-2/3} \propto C_a^{4/3}C_R^{2/3} \tag{7}$$

Plasma current $$I_p \propto R^{-1} \propto C_R \tag{8}$$

Greenwald density

$$n_G = I_p/(\pi a^2) \propto R^{-1}a^{-2} \propto C_R C_a^2 \tag{9}$$

Toroidal field (TF) within plasma is derived as

$$B_t \propto a^{-2} \propto C_a^2 \tag{10}$$

Equation 9&10 predicts that the threshold of the Greenwald density limit could be extended to a high value by compressing the plasma in the minor radius or major radius for the HGHF [2].

The plasma normalized pressure in toroidal direction,

$$\beta_t = 8\pi nT / B_t^2 \propto 8\pi a^{2/3}R^{-5/3} \tag{11}$$

And in poloidal direction

$$\beta_p = 8\pi nT/B_p^2 \propto a^{-4/3}R^{1/3} \tag{12}$$

The thermal energy-confinement time is thus derived as

$$\tau_{E,th} \propto aR^{-1} \propto C_a^{-1}C_R \tag{13}$$

In a beta-limited tokamak as in Ref. [3], the gain of fusion power with DT plasma is derived as [2]

$$P_{DT} \propto \beta_t^2 B_t^4 a^2 Rk \propto a^{-14/3}R^{-7/3} \tag{14}$$

In the same vacuum vessel of fusion plasma, the output power gain of compressed plasma is thus derived as

$$G_p \propto C_a^{14/3}C_R^{7/3} \tag{15}$$

Equations 10 and 15 predict the high-gain nature of fusion plasma in the high field in the same vacuum vessel of a tokamak if the plasma is compressed in the minor radius or in the major radius as suggested for EAST [2]. The gains observed during testing of ATC and TFTR are, slightly better than the predictions by equation 15 [2].

By combining equations 6, 7 and 13, the parameter gain of the Lawson criterion within the compressed plasma is derived as

$$G_L \propto n\tau_E T \propto C_a^{7/3}C_R^{8/3} \tag{16}$$

Equation 16 predicts that HGHF plasma also has a high gain in the Lawson criterion by the above magnetic compression; the gain rate is also faster than the rate of volume decrease according to equation 16. These characteristics will facilitate access to the ignition condition by existing tokamak, such as EAST, DIII-D or JET.

Due to conservation of the tangentially-injected beam-ion angular momentum about the major axis [6], the energy gain of fast particles moving along magnetic field lines with major-radius-compression is derived and validated as,

$$G_E \propto C_R^2 \tag{17}$$

For an ohmically heated tokamak, the saturation limitation of plasma density is empirically scaled as [3]

$$n_{sat}(m^{-3}) = 0.06x10^{20} I_p RM^{0.5}k^{-1}a^{-2.5} \tag{18}$$

Where I_p is the plasma current, M is the atomic mass of the ions in amu and k is the plasma elongation, b/a.

Ohmic heating is not enough to drive the plasma to fusion temperature, thus more powerful external heating with beams of an RF wave and neutral particles are developed — but testing found that all heating will shorten the energy confinement time of tokamak plasma. In 1982 on ASDEX at condition of strong external heating, the confinement time was favorably extended to a state compared to the Ohmic heating phase, originally dubbed as H-type discharge [7] and later named as H-mode. The H-mode was implemented as ITER scaling in 1998 [2-3], i.e., IPB_{98y2}. For IPB_{98y2} scaling, the thermal energy-confinement time is described as [3]

$$\tau_{E,th}^{IPB98y2} = 0.0562 I_p^{0.93}B_t^{0.15}P^{-0.69}n_e^{0.41}M^{0.19}R^{1.97}\varepsilon^{0.58}\kappa^{0.78} \tag{19}$$

Where ε is the plasma inverse aspect ratio (a/R).

For compressed HGHF plasma, equations 1 to 12 can be inserted into equation 19. Thus the thermal energy-confinement time of IPB_{98y2} scaling is derived as

$$\tau_{E,th}^{IPB98y2} \propto R^{0.05}a^{-0.54}\kappa^{0.78}P^{-0.69} \tag{20}$$

Equation 20 predicts that the energy-confinement time of compressed plasma has strong relations with plasma minor radius, heating power and elongation, as has been purposely explored by tokamak society since 1982.

More bootstrap currents could be produced in compressed high-pressure-gradient plasma, which requires less power for the current drive [2]. Thus the power of the current drive is reduced. If the scaled-down power multiple of the current drive is assumed to be the ratio of the bootstrap current fraction of the total current in post-compression plasma, it can be simply scaled as [2-3]

$$C_p \propto f_{bs} \propto \sqrt{a/R}\beta_p \propto C_a^{-1/2}C_R^{1/2}C_a^{4/3}C_R^{-1/3} \propto C_a^{5/6}C_R^{1/6} \tag{21}$$

Due to the scaling down of the external heating power, equation 20&21 predicts that the energy-confinement time of compressed plasma could also be improved by compressed plasma.

It is interesting to be noted here that above HGHF plasma share the same principle of flux-conserving tokamak and large shift of the plasma magnetic axis as that of unity-beta plasma approach originally started from Oak Ridge in 1977 [8-11]. But many differences still exist between them as listed in Table 1. HGHF plasma is much more economic than the conventional unity-beta approach in [10] due to its decreased amplitude of plasma current.

Table 1: Differences between HGHF Plasma and Unity-beta Plasma

Parameters	HGHF plasma approach	unity-beta plasma approach
Plasma current	$\mathrm{I_p} \propto R^{-1}$	Constant plasma current
Safety factor q	Fixed within plasma	Varied
Plasma boundary	Moving by MC	Nearly fixed
Toroidal field	Amplified by C_a^2	Nearly fixed
Poloidal field	Amplified by $C_a C_R$	Varied with plasma current
Volume	Small but at high field	Nearly full within vacuum vessel

MC DESIGN FOR UPGRADING EAST

Based on the flux-conserving principle and above formulations, using EAST tokamak for designing efficiency CFETR are listed as follows.

MC in-Vacuum Coil Design for EAST

MC accelerator functions are realized by powering the inserted MC coils within EAST vacuum vessel as suggested in figure 1 of Ref. [2]. Its right red coils are used for ion accelerating within high gain plasma by major-radius compression with equation 17.

Mapping EAST Tokamak for Breakeven

The plasma parameter of EAST is marked as green line on existing map of Lawson criterion as shown in Figure 1 [12].

If upgrading EAST by MC parameter $\boldsymbol{C_a=3/C_R=1.39}$ and inserting them into above formula as suggested in Ref. [2], EAST will be mapped to breakeven by two-step MC at red line at Q≈1. In this MC case by equation 22, H&CD power for breakeven is only 1/2.639=0.379 times that of conventional approach due to the gain of the bootstrap current fraction. Circulation power within tokamak could thus be decreased than that of the conventional design [12]. For the tangentially-injected beam-ions, the energy gain of fast particles within plasma is 1.93 by equation 17 with MC parameter $\boldsymbol{C_R=1.39}$.

CONCLUSION

For designing CFETR, Tokamak as an accelerator could path the quick and safe way by developing efficiency HGHF plasma on EAST — low circulating power simulation is thus enabled by MC technology which will decrease the voltage of the external heating accelerators of NBs, such as that of ITER at 1MV by equation 17.

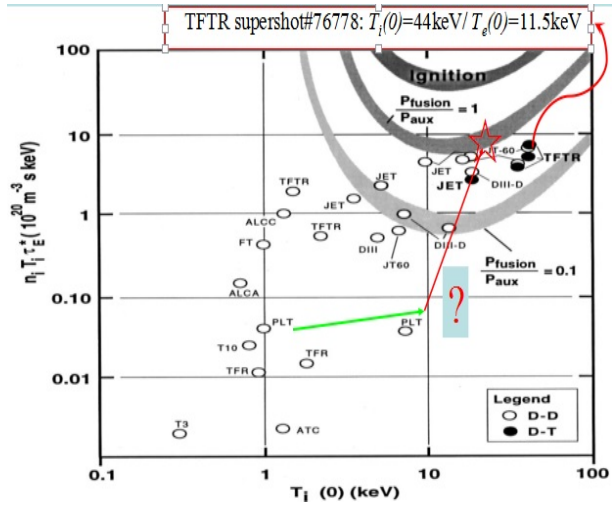

Figure 1: Mapping EAST tokamak for breakeven by MC.

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