SPHERICAL TOKAMAK WITHOUT EXTERNAL TOROIDAL FIELDS

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Abstract

A spherical tokamak design without external toroidal field coils is proposed. The tokamak is surrounded by a spheromak shell carrying requisite force free currents to produce the toroidal field in the core. Such equilibria are constructed and it is indicated that these equilibria are likely to have robust ideal and resistive stability. The advantage of this scheme in terms of a reduced ohmic dissipation is pointed out.

1. INTRODUCTION

The most severe design constraints on a spherical tokamak(ST) arise because of the copper centre post, which is excessively stressed because of intense ohmic dissipation, exposure to neutron irradiation, strong mechanical forces due to halo currents *etc.*. In this paper we propose a new method for generating the toroidal field of a spherical tokamak without the use of external coils. The method consists in surrounding the ST plasma with a spheromak shell plasma (carrying force-free currents) with enough poloidal current to generate the required toroidal field in the ST core. The spheromak shell could be sustained in the steady state by coaxial helicity injection whereas the currents in the ST core could be driven by a combination of bootstrap and RF/NBI current drive mechanisms. This scheme has several advantages.

(i) All the problems related to heat removal from the centre post, its protection from the heat and neutron fluxes and stresses on it due to halo currents are eliminated. The heat dissipated in the shell is removed through divertor plates in a manner similar to the removal of heat flux conducted from the core to the edge while the force-free nature of shell currents eliminates the issue of stresses due to halo currents.

(ii) All the external coil systems now reside on the outside in a simply connected geometry and can be made superconducting with adequate neutron shielding.

(iii) As we will show later the present scheme improves the ratio of fusion power(P_F) to power dissipation in *plasma coil*(P_C)

(iv) The force-free currents surrounding the tokamak core are known to improve the β limits[3].

2. EQUILIBRIUM

We now demonstrate the existence of such equilibria with inflected q - profile by solving numerically the Grad-Shafranov equation given by $R\frac{\partial}{\partial R}\left(\frac{1}{R}\frac{\partial\Psi}{\partial R}\right) + \frac{\partial^2\Psi}{\partial Z^2} = -\mu_0 R J_\phi$; $J_\phi = R\frac{dp}{d\Psi} + \frac{F}{R\mu_0}\frac{dF}{d\Psi}$ for given pressure and toroidal field profiles as function of Ψ the flux function as $p(\Psi) = p_0\Psi_s^{\alpha_p}$ and $\Psi_s = \left(\frac{\Psi_L - \Psi}{\Psi_L - \Psi_m}\right)$ where Ψ_L,Ψ_m are the Ψ values at the limiter and magnetic axis respectively . The profile for F is $F(\Psi) = R_0^2 B_0^2\left[1 - c_1\Psi_s^{\alpha_p} - sech^2\left(\frac{-\Psi_s}{\delta_F}\right)\right]$. p_0 and B_0 is the peak pressure and toroidal field at the axis and R_0 is the major radius. The constant c_1 decides whether plasma is diamagnetic ($c_1 > 0$) or paramagnetic ($c_1 < 0$), while δ_F controls the gradient of F near the edge. This equation is solved by specifying A, R_0, δ_f, β_0, B_0 and β_{pc} (the poloidal beta in the core) which determines c_1. We choose $\beta_{pc} \simeq \beta_0$ to ensure the paramagnetism of the core [1]. In fig. 1 we show an equilibrium solution of G-S equation with parameters related to a pilot plant and fig.2 shows the equilibrium profiles. This equilibrium is characterised by low A $\simeq 1.1$ (A $= R_0/a$ with R_0 and a respectively the major and minor radii), large elongation $\kappa \leq 2.5$, an inflected q-profile and the ratio of core toroidal current to the poloidal shell current is nearly one while the shell thickness is one-third of the core radius. In this solution the core is ST like *i.e*, core aspect ratio $A_T \leq 1.5$, monotonically increasing q and paramagnetic regime with $\beta_{pc} \simeq \beta$. In this case, naturally diverted SOL can be clearly seen.

3. STABILITY

The general consensus on low A spherical tokamaks is that natural shaping along with low A effects improve the MHD stability limits on plasma β. In the present scheme this is further likely to improve on account of shell carrying force-free currents[3]. To get some insight into the ideal stability of these equilibria we consider the large A regime where an analytic expression for ideal MHD perturbed energy δW can be examined. Normally, a monotonically increasing q-profile with $q_0 > 1$ is stable to ideal internal kinks with $m = 1$, $n \geq 1$ (m and n respectively the poloidal and toroidal mode numbers). In the present case, however, the shell introduces additional kink modes with resonant surfaces in the shell region. The stability of these modes can be examined by an eigenfunction which is $\xi = \xi_0 =$ constant for $0 \leq r \leq r_s$ and $\xi = 0$ for $r_s \leq r \leq a$ where r $= r_s$ is the location of the resonant surface. For modes with $m = 1, n > 1$, δW is given by[4] $\delta W = \xi_0^2 \int_0^1 \left[\frac{2k^2r^2}{s}\left(\frac{dp}{dr}\right) + \frac{k^2 r B_\theta^2 t}{s^2}\right] dr + \int_1^{r_s} \frac{k^2 r B_\theta^2 t}{s^2} dr$ with $s = (1 + k^2r^2)$ and $t = (1 - nq)(k^2r^2 - 1 - nq(3 + k^2r^2)$, where contribution due to shell region between $1 \leq r \leq r_s$ is written explicitly and $k = -n/R_0$. We first discuss the stability neglecting pressure. The most unstable modes lie in the range $k^2r^2 \leq 1$ and since $(nq - 1) > 0$ in $0 \leq r \leq r_s$, all these modes are stable [fig. 3]. The stabilising contribution due to shell given by second integral can be clearly seen. This is an interesting result which shows that by eliminating the field reversal region normally present in reverse field pinches(RFP), we have eliminated internal kinks in this region[5]. In RFP these modes impose a very stringent condition on the position of the conducting wall. A finite pressure gradient will tend to destabilise these modes and limit the β_p in the core. For large aspect ratio, this limit is roughly $\beta_p < 1$. We next discuss $m = n = 1$ mode, for which toroidal effects are important. In this case $\delta W = \frac{3r_s^4}{R_0^2}(1 - q_0)\left(\frac{13}{144} - \beta_p^2\right) + \delta W_s$ where the first term is contribution from the core region derived by Bussac *et al.*[6] and δW_s is the cotribution from the shell. The detailed calculation of δW_s is beyond the scope of this paper, however we note that if $q_0 > 1$, then $m = n = 1$ kink is stable for arbitrary large $\beta_p > \sqrt{\frac{13}{144}}$, provided there is no additional destabilisation from the shell. Such a destabilisation is unlikely to happen for two reasons; first since shell is cold, there is no destabilisation due to pressure; second, since shell is able to relax to force free configuration quickly , there is no free energy due to current gradients. Thus if $q_0 > 1$, ideal $m = n = 1$ kink continues to be stable for large β_p. Turning next to the case of external kinks we note that with $q_0 > 1$, the only possible external kinks are $n = 1$, $m > q_{max}$ [fig. 3]. In ST regime, because of natural elongation q_{max} can be as high as 15 to 20. Thus only possible external kinks are $m \gg 1$, $n = 1$. As is known[4,7], such high m kinks are localised, have small growth rates and hence are not prohibitive for β limits. This removal of external kink constraint, as can be seen is solely due to the shell. In normal tokamaks with $q_0 > 1$, low m external kinks with gross radial structure determine the Troyon limit on plasma β. In the present case, because of the shell, these modes have become stable internal kinks with resonant surface within the plasma. On the other hand the $n \to \infty$ ballooning stability of the core is unlikely to be seriously affected by the shell which is cold. Thus the β limit in these equilibria will be determined by $n \to \infty$ ballooning stability of the core. This is a definite advantage as β limits due to ballooning modes are generally soft and less restrictive. The localised Mercier interchange modes are stable because in the tokamak core $q_0 > 1$ while in part of the shell where $q < 1$, the plasma is pressure less[8]. We now turn to resistive MHD and kinetic stability of these equilibria. Recent experiments on START have shown that the major disruption events which are generally related to $m = 1, 2, 3$ resistive tearing modes are remarkably absent in the ST regime. Preliminary theoretical studies [9] have shown that the saturated island width around $q = 3/2$ and $q = 2$ surfaces decreases strongly with the decrease of A. Furthermore, for $m = n = 1$ modes, the results show a ten-fold increase in the reconnection time in the A $\leq$ 1.5 regime. With the inflected q-profile of our configuration, we may expect some double tearing modes; however, as they are similar in character to $m = n = 1$ modes (when the two resonant surfaces are close to each other) we expect them to be stabilised in the low A regime. The kinetic stability in a low A regime is robust for two reasons. First, reduction in orbit averaged bad curvature instability drive and second reduction in the trapped particle fraction due to omnigeneity and magnetic well in the outboard region. The tilt and shift modes can be stabilised by external coils.

4. PRODUCTION AND SUSTENANCE

Next, we briefly discuss how such equilibria can be produced and sustained experimentally . One way of producing the desirable configuration would be to start with a spheromak and to simultaneously *increase* the pressure and *decrease* the toroidal current density in the central core region by a combination of neutral beam heating and/or RF current drive methods. The q in the central region can thus be lifted to tokamak like values and shaped by current profile control; note that this is possible only while the plasma is still resistive. As the plasma β becomes significant, the current in the core can be predominantly maintained by bootstrap effects. The maintainance of the external spheromak like region can be ensured by a steady co-axial helicity injection in the periphery[10]. Recent experiments on HIT have successfully demonstrated tokamak startup and sustainment phases in low A spherical tokamaks. Typically currents upto $\sim$200 kA in the core was sustained for $\sim$10 ms in a configuration of A = 1.2, $\kappa = 2$. As stated earlier toroidally rotating $n = 1$ perturbations were observed which are thought to be responsible for inward flux diffusion. Extrapolation of this method to reactor regimes have been recently projected.

5. OHMIC DISSIPATION

We finally discuss the issue of ohmic dissipation(P_C) in the 'TF Coil'. The ratio of fusion power to ohmic dissipation power for an ST reactor (P_F/P_C) is given by[2] $G = \frac{P_F}{P_C} = \frac{(0.88\lambda)\pi^5(0.2)^4}{\eta_C}\left[\frac{0.36(1+\kappa^2)}{2\beta_p}\right]^2 \frac{R_c^4 J_C^2(A-1)^2}{A^2}\left[1-\frac{\delta}{a}\right]^5$ where η_C is the resistivity of the centre post while R_C is its thickness given by $R_C = R_0 - a - \delta$. The term δ takes into account the width of SOL and shielding which may be needed to protect the copper centre post from the core fluxes. For usual ST,δ/a can be taken to be $\simeq 0.1$ while in the present scheme $\delta = 0$ as the plasma shell is in contact with the core. The increase in the gain G is given by $\frac{G(\delta=0)}{G(\delta=0.1)} = \frac{1}{\lambda}\left(\frac{R_0-R_C}{R_0-R_C-\delta}\right)^5$. Thus with the present scheme ohmic dissipation can be reduced by a factor of two, if η_C for plasma shell is taken to be the same as that of copper.

To summarise, we have proposed a novel compact magnetic confinement configuration of the spherical tokamak type in which the toroidal field in the tokamak core is provided by a current carrying spheromak shell. This coupled compact configuration combines the attractive stability features of force-free currents in spheromaks with excellent confinement properties of the tokamak like core and eliminates a number of complex design issues related to the excessively dissipative centre post in conventional spherical tokamaks. Experimental attempts to set up such configurations and to in v estigatetheir stability, confinement and sustenance requirements will therefore be v ery worthwhile.

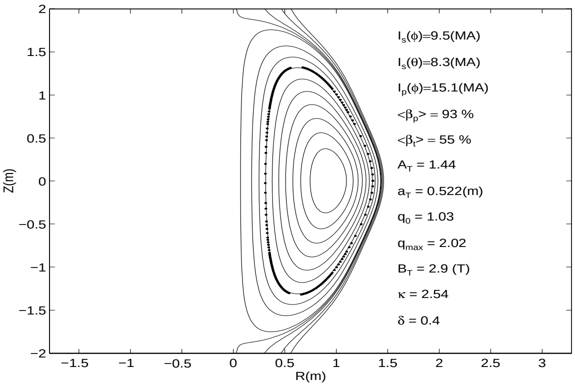

Fig 1. shows the flux surfaces of Pilot plant diverted plasma and relevant plasma parameters. Dotted line shows the boundary between hot core and shell region.

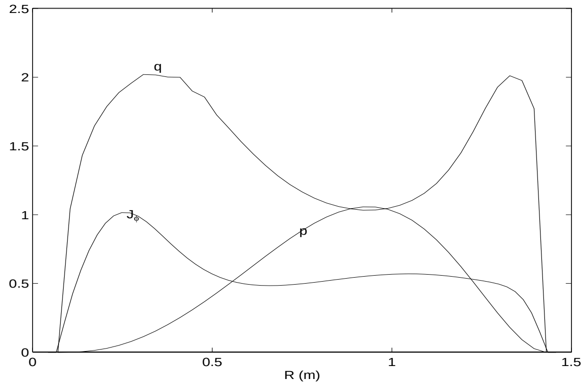

Fig 2. Profiles of q, J_ϕ and pressure. Normalization of J_ϕ is 13 MA and that of pressure is 3 MP.

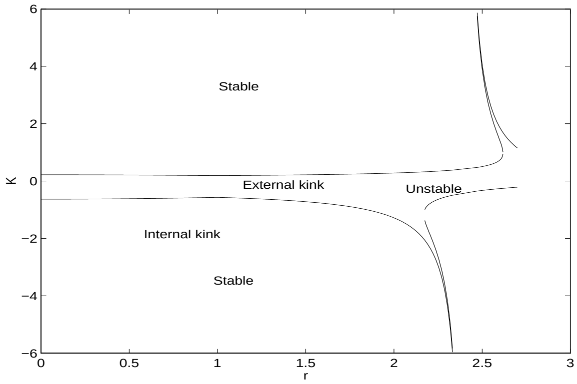

Fig 3. shows stability diagram for internal and external kink modes for large A.

6. REFERENCES

[1] Y. K. M. P engand D. J. Strickler, Nucl. F usion,**26**, 769 (1986).

[2] R. D. Stambaugh et al. in F usion EnergyProceedings of 16^{th} Conference, Montreal, canada, 7-11 Oct. 1996 (IAEA, Vienna, 1996) Vol. II 395 pp.

[3] D. A. D'Ippolito et al., Phys. Fluids, **21**, 1600 (1978).

[4] J. P .F reidberg,Ideal Magnetohydrodynamics (Plenum Press, NY & London 1987) p. 222.

[5] D. C. Robinson, Plasma Phys. **13** 439 (1971).

[6] M. N. Bussac *et al.*, Phys. Rev. Lett. **35**, 1638 (1975).

[7] F. Tro y on*et al.*, Plasma Phys. **26**, 209 (1984).

[8] A. Sykes *et al.*, in Controlled Fusion and Plasma Phys. 11^{th} European Conf., Aachen, Germany **14**, 645 (1983).

[9] B. A. Carreras *et al.*, in Plasma Phys. and Controlled Nuclear Fusion research 1986, 11^{th} Conf. Proceedings, Kyoto, 13-20 march 1986 (IAEA, Vienna, 1987) Vol. II, p. 57.

[10] T. R. Jarboe ref. 2 pp. 243.