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A novel Plasma Source concept for negative ion generation in neutral beam Injectors for Fusion applications

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INTRODUCTION

Negative ion sources used for Fusion applications produce up to **~ 50 A of Hydrogen or Deuterium ion current** from a low-temperature (few eV) plasma. Being the negative ion current typically ~ 200 - 300 A/m² at the extraction apertures (where gas pressure is kept < 0.3 Pa, to avoid excessive neutral gas flow in the accelerator), these sources can be quite large and their input power ranges from some tens to **hundreds of kW.**

Types of Negative Ion Sources currently used for Fusion experiments:

- the Japanese "kamaboko" sources, where hot biased filaments produce an arc plasma discharge, which is confined inside a "plasma box" by a multi-cusp magnetic configuration, generated by permanent magnets [1]
- the RF (radio-frequency) plasma-driver sources, mainly developed in EU, where a coil operating at about 1-2 MHz induces a plasma discharge inside a cylindrical volume surrounded by thick metallic shields [2]

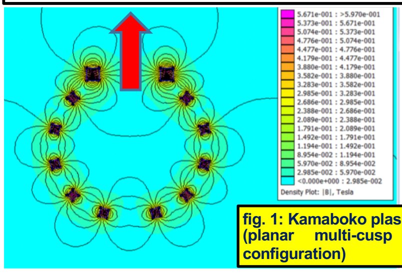

fig. 1: Kamaboko plasma source (planar multi-cusp magnetic configuration)

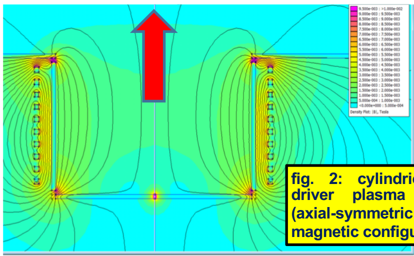

fig. 2: cylindrical RF-driver plasma source (axial-symmetric magnetic configuration)

The generated plasma expands towards a cesiated Plasma Grid (PG), where some of the impinging particles are converted into negative ions and then accelerated through the grid apertures.

In both types of source, a power of about 2-4 MW/m2 is necessary for achieving the required negative ion current at the grid. The main reasons for this are:

- the magnetic field configuration does not provide an efficient confinement of the low-temperature plasma with respect to the walls of the plasma chamber;
- a large fraction of the electric power of the RF coils is also dissipated by eddy currents induced on the metallic structures of the RF driver.

In principle, a **current-free plasma** (such as the one considered in the plasma sources) could be **confined using a purely poloidal magnetic field.** Such configuration] has the **clear advantage** that **the purely poloidal (dipole) magnetic field lines do not cross the solid walls** of the device and thus most of the electron trajectories are not intercepted by the solid walls.

In addition, since the dipole configuration can achieve plasma pressure equilibrium with zero net plasma current, the MHD stability with limited energy transport.

The concept of the plasma confinement in a dipole configuration has already been widely applied for plasma confinement in the **LDX (Levitating Dipole eXperiment)** [4].

A similar configuration has also been proposed in the **Polomac** device [5]

Finally, a small device for magnetron sputtering has been realized using the dipole configuration produced via permanent magnets [6]

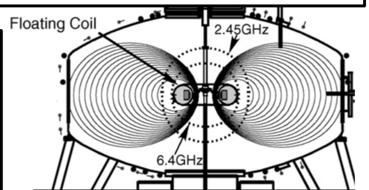

fig. 3: schematic of the LDX (Levitating Dipole eXperiment) operated at MIT [4]

The proposed **new plasma Driver configuration** consists of an **"central cylinder" coil**, positioned inside an **"external stewpot" coil** (see fig. 5) and can be considered a toroidal version of the "hard-core Z-pinch" [3].

The central coil produces a poloidal field profile $B_\theta(r)$ which is high inside the cylinder and ~1/r outside (fig. 6): this stabilizes the m=1 mode which is the major problem of the Z-pinch [3].

Plasma pressure starts from zero just outside the central cylinder coil, then peaks and decreases again for large r.

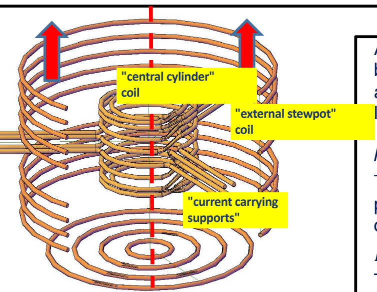

fig. 5: CAD view of the of proposed **Plasma Driver**,:

- **"central cylinder" coil** constituted by water-cooled current-carrying conductors
- **"external stewpot" coil**, also constituted by water-cooled conductors.
- **"current-carrying supports"** also provide magnetic shielding to reduce heat load on the supports.

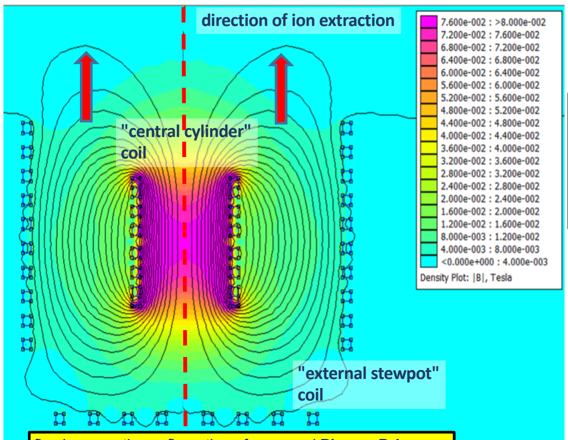

fig. 4: magnetic configuration of proposed **Plasma Driver** (axial-symmetric geometry)

A first **estimate of β** is given from global pressure balance relating the central coil current (~1kA · 8 turns) and the plasma diamagnetic current, assumed to be less than 1% of the central coil current (~80A) [3]:

$$\beta = 1 - \left(\frac{I_{coil}}{I_{coil}+I_{plasma}}\right)^2 \sim 2\%$$

This formula, obtained in the case of the hard-core Z-pinch, is valid also for the pure dipole, where the field depends on the magnetic moment M according to:

$$B_\theta = \frac{\mu_0\, M}{2\pi^2 r^3}$$

The corresponding β is a function of the coil radius r_c and plasma radius r_p :

$$\beta = 1 - \left(\frac{1}{1+(r_c/r_p)^2}\right)^2$$

Using the LDX values of r_c=0.6 m and r_p=0.8 m, one obtains β=60%, which justifies the use of the dipole concept for a reactor, as initially suggested by Hasegawa [9].

In the proposed configuration for the ion source (r_c=0.04 m and r_p=0.08 m), the estimated beta would be up to β=35%. But this would require a plasma diamagnetic current of 2 kA.

We might expect an operating space of the proposed ion source configuration with values of **beta in the interval β≈2-35%** and corresponding diamagnetic plasma current I_plasma ≈ 80 - 2000 A.

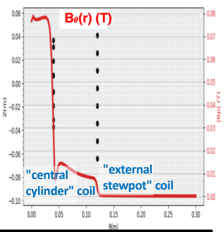

fig. 6: radial profile of **poloidal magnetic field** $B_\theta(r)$ at the equatorial plane

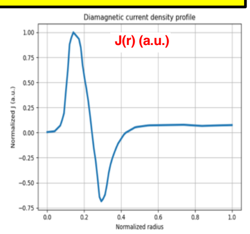

fig. 7: Assumed **diamagnetic current density profile** J(r)

Force balance static equilibrium:

Since there is no "toroidal" magnetic field, the plasma current is purely diamagnetic, and is only related to the pressure gradient:

$$\boldsymbol{J} = -\frac{\nabla P \times \boldsymbol{B}}{B^2}$$

Thus, a purely poloidal vacuum field gives rise to a purely toroidal diamagnetic current, which in turn adds to the poloidal field.

The toroidal current density is assumed to follow the "trifurcated model" used in LDX experiment [7], with a linear transition between the negative toroidal current density inside the pressure peak to the positive toroidal current density outside the peak corresponding to a parabolic pressure profile (Fig. 7) .

Note that, while the direction of the current density must change sign in the plasma, there is always a net plasma current that flows in the same direction as the central coil current in order to provide toroidal force balance [8]. This current is estimated to be less than 1% of the total central cylinder coil current.

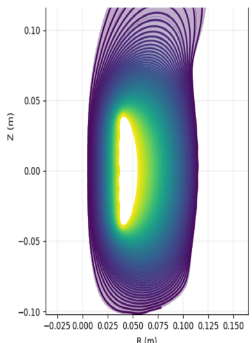

fig. 8: **poloidal flux surfaces** obtained by solving the Grad-Shafranov equation

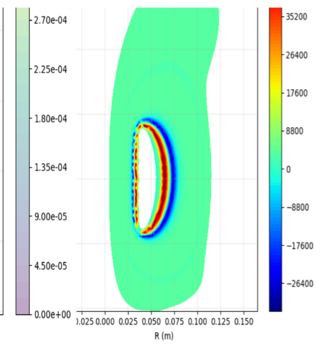

fig. 9: plasma **diamagnetic current density** obtained by solving the Grad-Shafranov equation

Given the value of the flux at the coil boundary and at the plasma boundary, the equilibrium problem can be solved for two cases:

i. Vacuum, where no plasma current exists, equation
ii. Grad-Shafranov, equation, where the plasma diamagnetic current density profile is assumed (Fig. 7) [7].

For the solution (ii), the boundary conditions at plasma and coil boundaries have been updated during the iterative solution with the contribution of the plasma diamagnetic current. The poloidal flux surfaces for case (ii) are shown in fig. 8; the plasma diamagnetic current density is shown in fig. 9.

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