EVOLUTION OF CONFINEMENT PHYSICS AND MOST PROBABLE COMPACT IGNITION TEST DEVICE IN MAGNETIC FUSION

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In magnetic fusion, demonstration of the ignition state is an urgent task prior to any DEMO design in order to quantify various critical plasma parameters and engineering issues. These include transition physics from external to internal α-heating for sustained ignited state and sufficient neutron yields for self-sustained tritium production and first wall material test. Among various fusion reactions, the most practical fusion reaction in the laboratory is the Deuterium and Tritium (DT) reaction which has sufficient cross-section at T_i~10-20 keV ion temperature with a reasonably high density and adequate energy confinement time. In this paper, possible practical actuators for the edge and core confinement improvement and a compact tokamak plasma with $n_i\tau_E T_i \cong Q$=5-6 for $T_i > T_e$, that can test the ignition state based on half a century experimental data is suggested.

Following discovery of high confinement (H-mode) mode [1] when a diverted magnetic configuration was introduced in ASDEX, Germany, instead of limiter configuration, the improved confinement region was dominantly at the edge region and it was understood as an Edge Transport Barrier (ETB). The turbulence suppression mechanisms like velocity shear due to E_rxB effect were dominant approaches to understand the ETB formation without clear source of E_r. Here, a potential source of E_r due to ∇B drift in the vicinity of x-point where poloidal field (B_P) ≅0, is suggested but turbulence inside the Last Closed Flux Surface (LCFS) may not be responsible for the ETB formation as shown in Fig, 1. However, the amplitude of turbulence is not affected by the presence of velocity shear in pedestal region in H-mode (1d) and the reduced normalized turbulence level in pedestal is due to high pedestal density (1c). A finite impedance model between LCFS and divertor plate is introduced for H-mode phase whereas it is "short-circuit" in limiter plasma. This model can be applied to both tokamak as well as stellarator plasmas and explains why the edge confinement of the stellarator is inferior to that of the tokamak. The formation of ETB (H-mode phase) is due to increased impedance (reduction of density and turbulence from inflow plasmas from divertor plates) which is a result of conditioning of divertor plate with outflow plasmas in L-mode phase (i.e., interplay between outflow and inflow plasmas). A smart divertor system that can be a good heating sinking system as well as recycling control will be the best actuator for the edge confinement improvement (i.e., "super-x" and "snow flake").

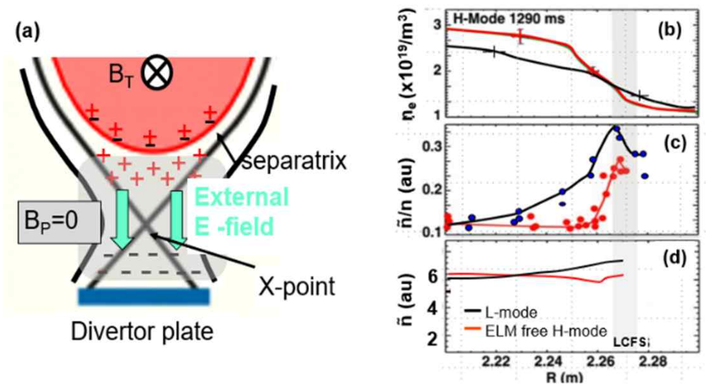

Fig.1 (a) Potential source of the electric field (E_r) formation inside the LCFS in response to the external E-field arising from ∇B drift in the zero-shear region ($B_P \cong 0$) is shown schematically. (b) Edge density profiles of the L-mode phase (black) and H-mode phase (red) (b) Corresponding normalized turbulence profiles for L and H mode phase. (c) Corresponding turbulence amplitude of L and H mode phase. Figures (b) and (c) are courtesy from ref. [2].

Improved confinement in the core region of the plasma (i.e., a steep pressure gradient) was observed in both magnetic configurations (i.e., toroidal plasmas with limiter and divertor) and it was understood as an internal transport barrier (ITB). The ITB can be formed in an ion channel with an ion heating system and an electron channel with an electron heating system. The ITB formation has been also attributed to turbulence suppression (i.e. like η_i marginality or breaking of streamers due to E_rxB effect) or q-profile shear. η_i marginality cannot explain the observed wide variation of η_i values and models based on E_rxB effect only exist in theoretical models so far. Correlation between ITB position and q_{min} position is likely the result of the off-axis PNB current, not the cause of ITB. The practical actuator for ITB can be a smart heating system and combination of ITB and ETB such as “Super H-mode” [3] would be ideal for ignition test. Auxiliary heating system is essential to achieve high ion temperature (T_i>10 keV) for optimum α-particle power necessary for ignition in toroidal devices and methods can be divided into two categories; electron heating and ion heating. A direct ion heating by Positive Neutral Beam (PNB) has been very effective but application to a large device and/or high-density plasmas is challenging. On the other hand, application of electron heating in high density and/or large devices can be effective but there are unresolved physics issues like T_i clamping [4] and density pump-out [5]. Also, high-power application (~up to 50MW) is an engineering challenge and antenna structures may not be compatible with the fusion reactor. Note that ITER plans to have ~60MW of electron heating system only.

The confinement time scaling and performance data accumulated for half a century are examined to project a logical path for the most probable compact ignition test device in magnetic fusion as shown in Fig.2. A tokamak with the plasma volume of 240-270 m^3 with PNB power <40 MW is a good candidate. The plasma volume provides an adequate confinement time of ~2.5s with an H-factor of 1.5 and effective core PNB heating will sustain T_i>10keV. This compact device will produce ~200-250 MW fusion power at a moderately high density (~$1.5x10^{20}/m^3$). At his level of fusion power, 14 MeV neutron flux can be used for material testing and α-particle power is up to ~50MW which may be sufficient to test sustainment of the ignition state including transition physics from external to internal heating.

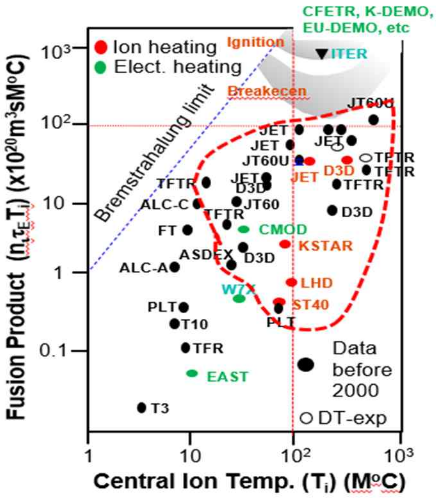

Fig. 2 Performance data accumulated for the last half a century is illustrated. The y-axis is $n_i t_E T_i$ (≅Q for T_i>T_e) and the x-axis is ion temperature (T_i). Black circle data are up to year 2000 including highest performance data from three large tokamaks (TFTR, JT-60U and JET). Projection of ITER in ignition territory is shown together with other large devices (CFETR, EU-Demo and K-Demo). New data points after the 2000s are shown; the green data points are discharges heated or will heated by ECH and/or ICRF (electron heating; T_e>T_i) and the red circles are discharges heated by PNB system (ion heating; T_i>T_e). Territory with red dotted line is discharges heated by PNB system.

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