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Reverse Shear Discharge Simulation of EAST Tokamak

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Abstract: EAST Superconducting Tokamak is a non-circular advanced steady-state plasma experimental device, which has been build at ASIPP this year. Predictive simulations of EAST reversed shear configuration are carried out using the 1.5D equilibrim evolve codes. The reversed shear plasmas are sustained by off-axis LHCD profiles. A series of simulations are performed to study the influence of the auxiliary heating applied time to plasma parameters. Based on these simulations, a novel control scheme was proposed and tested for control of the safety factor profile which is very useful in real time profile control in tokamak experiments.

1. Introduction

The mission of the EAST tokamak project [1] is to develop an advanced superconducting tokamak capable of steady state with which to explore advanced plasma configuration scenarios and to establish the scientific and technological bases for an attractive fusion reactor as a future energy source. A strategy for reaching a high performance plasma state in an advanced tokamak scenario is to preform an optimized current density profile in the discharge with off-axis non-inductive currents. Sustained reversed shear configuration, where the safety factor profile has a minimum away from the plasma center, is very promising for all future steady-state tokamak operations as the internal transport barrier usually forms at the location of the minima [2]. Inside this region, the plasma is well confined with reduced particle and heat transport and enhanced transport in the region outside. Thus controlling the q profile and the location of its minimum, which is essentially achieved by controlling the plasma current profile, is of great importance for improved plasma performance in steady state. The objective of this paper is to investigate the reversed shear configuration for EAST and device a control scheme to sustaining and controlling it.

The main EAST parameter is list as follow [3]: R_0/a=1.78m/0.4m, Ip=1MA, B_T=3.5T, elongation κ=1.8, triangularity δ=0.4. The toroidal field is provided by a set of 16 superconducting coils and the plasma equilibrium shape is controlled by a set of 14 up-down symmetric superconducting poloidal field coils.

There are three heating and current drive (H&CD) systems proposed for the EAST: NBI, ICRH and LHCD. The NBI system is required to provide ion heating and core fuelling. The NB H&D system consists of two injectors. One beam line is co- injection with the power of 2MW; another beam line is counter- injection with 2MW. The ICRH system is used to heat plasma ions. The design features for EAST ICRF heating system are: RF output power is 3MW, the frequency range 30MHz-110MHz has been chosen. The LHCD system is required to provide seed current for control of the total plasma current profile in the off-axis part of the plasma as well as for efficient bulk current drive in the low plasma temperature regions. The designed LHCD system on EAST is composed by two sub-systems, one of them delivers 2MW at 2.45GHz and another delivers 1.5MW at 3.7GHz. Both have wave pulse length over 1000 seconds.

In this paper, we firstly investigate in detail the reversed shear configuration with simulations using the coupled 1.5D equilibrim evolve codes and LHCD [4-6] codes, a brief detail of which is given in section 2. The simulations for the reversed shear mode operation are presented in section 3. Section 4 is the control scheme to the safety factor profile. The discussing and conclusions are given in section 5.

2. The Simulation Model

1.5D equilibrium evolve codes evolves a

2-D axisymmetric toroidal plasma equilibrium by solving a set of coupled Maxwell-MHD equations on transport time scales. The initial plasma equilibrium is formed on a rectangular computational domain with boundary conditions set by the PF coil currents. The plasma electron and ion temperatures are evolved by solving a set of flux-surface averaged transport equations.

The LHCD code calculates lower hybrid power and current deposition profiles in tokamak. The wave propagation in the plasma is calculated using a WKB approach, solving appropriate ray-tracing equations, while the current driven is calculated by a 1-D Fokker Planck solver.

Calculations for NBI and ICRH are based on the specified spatial external heat source deposition profile $F_{NBI} = \alpha \frac{D_{BEAM}^2 (1-\hat{\Psi})^{N_{BEAM}}}{(\hat{\Psi} - A_{BEAM})^2 + D_{BEAM}^2}$ and input power profile $S_{ICRH} = \beta \frac{d^2 \hat{\Psi}^{a_1} (1-\hat{\Psi})^{a_2}}{(\hat{\Psi} - a)^2 + d^2}$, where $\hat{\Psi} = (\Psi - \Psi_{min})/(\Psi_{lim} - \Psi_{min})$ is the normalized poloidal flux, A_{BEAM}, D_{BEAM}, N_{BEAM}, a, d, a_1,a_2 are the profile coefficients and α, β are normalization parameters.

3. Simulations of Reversed Shear Configurations

Tokamak plasma operation with negative magnetic shear is now regarded as the most promising way to increase fusion performance. A hollow current density profile, i.e. a reversed q-profile (negative magnetic shear), is one of the key conditions that give rise to the improved core confinement in advanced tokamak scenarios.

In order to have the desired q-profile with all its beneficial effects during the high power and plasma performance phase of a tokamak discharge, a successful preparation phase is required to create the appropriate target q-profile. The preparation phase is the preheating phase and defined as the time between plasma initiation and the large increase in the heating power (called main heating or high power phase). It is the phase when most of the current ramp up occurs. Current ramp up plays an important role in establishing the appropriate q-profile. The purpose of the preheating phase is to bring the plasma to an optimum state for experiments to be conducted at high power phase which further takes advantage of the created q-profile via the long current diffusion time at high electron temperature.

In our simulation, Plasma current is ramped up from 100 kA to a flattop maximum of 1.0 MA for 4s (Fig. 1). 1MW of ICRH power is applied to the plasma until the plasma shape is formed at 4s and, then, the power is raised to 3MW. 3.5 MW of LHCD power is applied from 1s to optimize plasma current density profile. 4MW of NB power is applied from 4s (Fig. 2). Fig. 1 also shows the lower hybrid driven current and bootstrap current. Of the total plasma current of 1MA, I_{LHCD} = 0.75MA, I_{BS} = 0.19MA. Fig. 3 and 4 plot the time evolution of main plasma parameters: Te(0), Ti(0), n(0), β_p and li/2. It is clearly seen that the applying of lower hybrid wave make the electron temperature increased from 2keV to 14keV and, then decreased to 8keV due to the added plasma density. At the flattop duration, β_p = 0.9, li = 0.96. Fig. 5 is the lower hybrid driven current profile and bootstrap current profile. The peak value of lower hybrid driven current appears at ρ = 0.25. Fig. 5 shows the final q profile, which is negative magnetic shear.

To study the influence of the auxiliary heating applied time to plasma parameters, we have performed a series of simulations (not shown). We find that the time auxiliary heating injected has little influence on some plasma parameters at flattop stage such as β, plasma density, temperature, loop voltage, but great influence on q0, internal inductance, volt-second，bootstrap current value.

4. Scheme for Controlling the q-profile

On the basis of the reversed shear simulations presented above, we propose a novel control algorithm for control of safety factor profile. We select the control actuators that can influence the safety factor profile in EAST are the total NBI, ICRH and LHCD powers. We select the control variables as q_{min}: the minimum of the safety factor, ρ_{min}: the location of q_{min} and q_0: the central safety factor. The control model can be written as **P=K• QQ** being the vector of the control parameters: **Q**=$\left[\delta q_{min}, \delta\rho_{min}, \delta q_0\right]$ with the δ's representing the differences between referenced and desired values and **P** the actuator vector: **P=** $\left[\delta P_{NBI}, \delta P_{ICRH}, \delta P_{LHCD}\right]$, where $\delta P_{NBI}, \delta P_{ICRH}, \delta P_{LHCD}$ are the incremental power of NBI, ICRH and LHCD. The kernel matrix **K** is the transfer function between the actuator signals and the control variables that we propose to determine using principal component analysis on the database generated in the simulations of the EAST plasma evolution. Similar control techniques have been reported for JET experiments [7,8,9]. We make four discharge simulations to determine the plasma response matrix **K**. We select case 2 described above with NBI power stepped down to 2MW at t=7s as the reference discharge simulation. In the other three discharge simulations, the LHCD, NBI and ICRH powers are alternatively stepped down with respect to the reference ones, always at t=7s. The LHCD power is stepped down to 0.5MW, NBI is stepped down to 3.5MW (but this corresponds to a step-up with respect to the 1.5MW of the reference discharge simulation) and ICRH is stepped down to 0.5MW. Through analysis the response data, we obtain the following approximate control matrix: $\mathbf{K}=\begin{bmatrix} 5.47 & -2.63 & -0.33 \\ -2.71 & -1.78 & 3.79 \\ 2.57 & 0.82 & -1.50 \end{bmatrix}$ where we have normalized the input power differences and the q-differences to their reference values [i.e. we consider the vector elements $\mathbf{P}_i' = \delta\mathbf{P}_i/\mathbf{P}_{i,ref} = (\mathbf{P}_i - \mathbf{P}_{i,ref})/\mathbf{P}_{i,ref}$ and $\mathbf{Q}_j' = \delta\mathbf{Q}_j/\mathbf{Q}_{j,ref} = (\mathbf{Q}_j - \mathbf{Q}_{j,ref})/\mathbf{Q}_{j,ref}$.

Figure 6 shows a control simulation using this control matrix in which we tried to shift the minimum of the q profile ρ_{min} from an initial minimum at ρ = 0.25 to 0.4, with no demand on the other control parameters, i.e., **Q**=[0 0.15 0]. The final q profile obtained has a minimum at ρ = 0.36, which is reasonably good. However, these simulations prove, that such techniques can be very useful in real time profile control in tokamak experiments.

The control actuators that can influence the safety factor profile in EAST are not only the total NBI, ICRH and LHCD powers, but also the power profile launched. The next we will study the relation between the q profile and the different of the power profile launched.

5. Conclusions

We have simulated reversed shear operation scenarios in EAST tokamak using the 1.5D equilibrium evolve codes. The reversed shear plasmas are sustained by off-axis LHCD profiles. Then we perform a series of simulations to study the influence of the auxiliary heating applied time to plasma parameters. We see that the time auxiliary heating injected has little influence on some plasma parameters at flattop stage such as β, plasma density, temperature, loop voltage, but great influence on q0, internal inductance, volt-second，bootstrap current value. Based on these simulations, we propose and test a novel control scheme for control of the safety factor profile. The results of the test prove that such techniques can be very useful in real time profile control in tokamak experiments.

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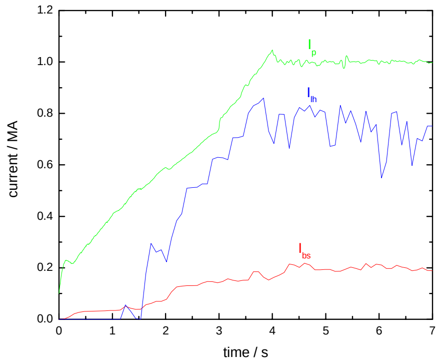

Figure 1. Total and LH, bootstrap currents

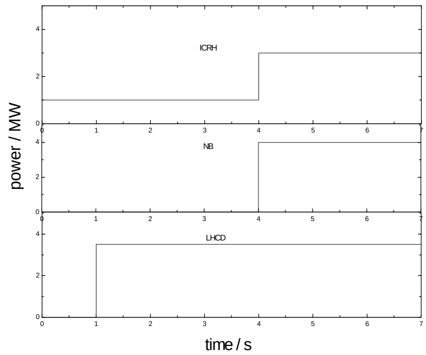

Figure 2. Auxiliary heating power

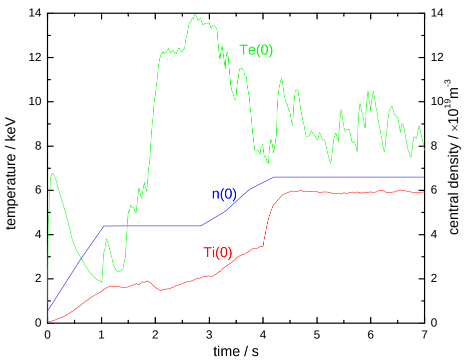

Figure 3. Central Te, Ti and density

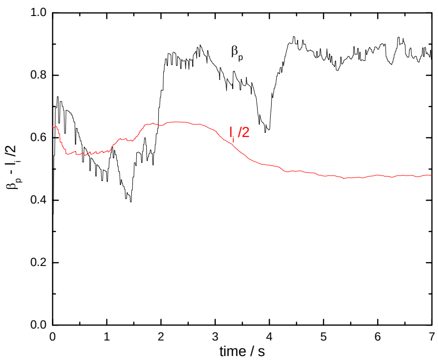

Figure 4. Evolution of β_p and li/2

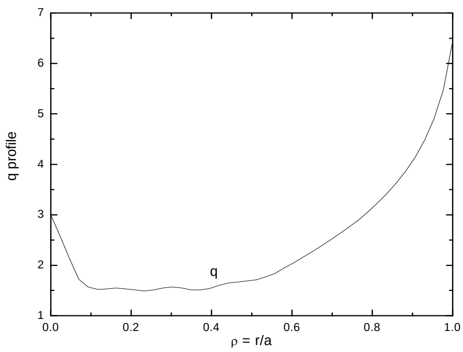

Figure 5. The final safety factor profile

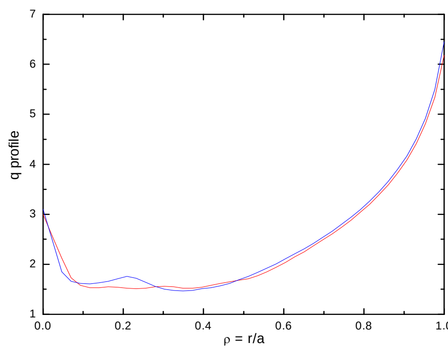

Figure 6. Initial (red) and final (blue) q profile in the control experiments.