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Recent plasma control progress on EAST

B.J. Xiao[a,*], Q.P. Yuan[a], D.A. Humphreys[b], M.L. Walker[b], A.W. Hyatt[b], J.A. Leuer[b], G.L. Jackson[b], D. Mueller[c], B.G. Penaflor[b], D.A. Pigrowski[b], R.D. Johnson[b], A.S. Welander[b], R.R. Zhang[a], Z.P. Luo[a], Y. Guo[a], Z. Xing[a], Y. Zhang[a]

[a] *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*
[b] *General Atomics, P.O. Box 85608, San Diego, CA 92186-5608, United States*
[c] *Princeton Plasma Physics Laboratory, Princeton, NJ, United States*

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ABSTRACT

In recent 2 years, various algorithms to control plasma shape, current and density have been implemented or improved for EAST tokamak. These plasma control performances have been verified by either simulated or actual experimental operation, and thus plasma control basis has been established for the long pulse operation and high performance H-mode plasma operation with low hybrid wave (LHW) and ion cyclotron resonance frequency (ICRF) heating. Startup simulation has been done by using TOKSYS code for the plasma breakdown in either 3.1 Wb or 4.5 Wb initial poloidal flux state and the scenarios proved to be robust and used for routine operation. Various shape configurations have been well feedback controlled by using ISOFLUX limited, double-null or single null algorithms based on RTEFIT equilibrium reconstruction. For the long pulse operation, strike point control and magnetics drift compensation have been implemented in the plasma control system (PCS). To improve the operation safety and efficiency, the verification of magnetic diagnostics before plasma breakdown has been demonstrated adequate to prevent a discharge in case of key sensor failure.

1. Introduction

EAST has been designed to be a tokamak with full superconductive toroidal and poloidal field coils aiming at long pulse and high performance operation. EAST PCS [1] was inherited and adapted from DIII-D PCS [2]. Most of the control algorithms in DIII-D have been also inherited and adapted to EAST for plasma current, position and shape control. Some new features have been added in EAST PCS, for example, the magnetics verification to avoid discharge under failure of key sensors and integrator drift compensation for the long pulse operation. These new features are described in Section 5. More new control algorithms are on progress, for example, radiation control for the detached or partially detached plasma for the purpose of long pulse operation. In the 2nd section of this paper, we describe the modeling activities for EAST plasma startup and control. From the 3rd to the 5th section, we summarize the main implemented plasma control algorithms and their applications to the plasma operation.

2. Plasma startup and control modeling

EAST plasma control model has been constructed by using TOKSYS [3] which consists of a series of circuit models with all of the passive conductors such as vacuum vessels and passive plates, active control coils and plasma circuit together with plasma response model, although in TOKSYS code the plasma response model is based on the assumption of rigid radial and vertical plasma displacements, significant variation in plasma poloidal beta, internal inductance, and separatrix configuration are also takes into account. Such a model can be transferred into state space model for the control simulation and development. Fig. 1 shows the EAST poloidal field coil system. The voltage limit for each power supply since 2011 has been also shown. The numbers in brackets represents the voltage limits before 2010. Since 2010 autumn campaign, we increased the power to drive 2 divertor coils (PF7 and 9 are considered a single coil since they are connected in series permanently, PF8 and 10 are connected in the same way, so they are also considered as a single coil) to increase the controllability for the X points and elongation. We also increased the power to drive PF11 through PF14 for the better control of the outer gap or the plasma radial position in particular in the case of the fast plasma ramp up, and the outer squareness as well. By using this EAST-TOKSYS model, the plasma startup in the first plasma campaign has been proven very successful in an initial magnetization state with poloidal flux at

* Corresponding author. Tel.: +86 551 5593350; fax: +86 551 5593350.
E-mail address: bjxiao@ipp.ac.cn (B.J. Xiao).

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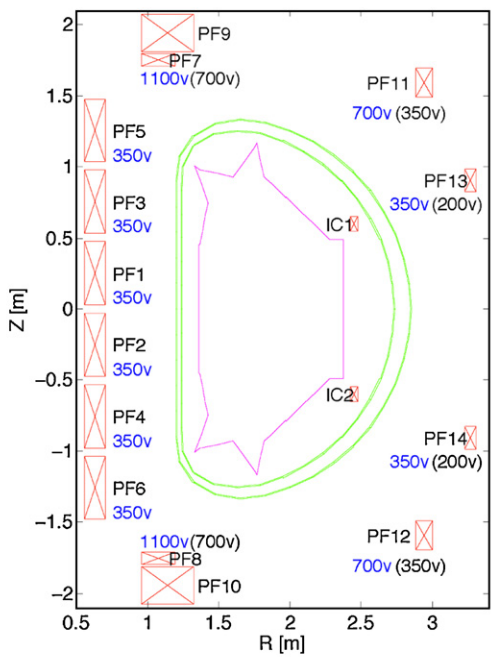

Fig. 1. EAST power supply and PF coil configuration showing the maximum power supply voltage.

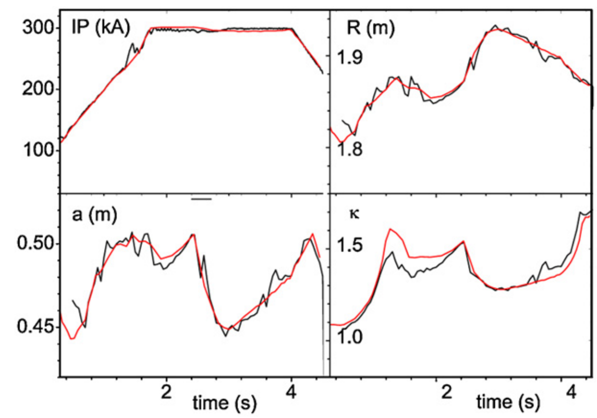

Fig. 2. Comparison of TSC modeled (in red color) plasma current, major radius, minor radius and elongation with the values reconstructed by EFIT (in black color) from the magnetic diagnostics in the experiment. (For interpretation of the references to color in this figure legend, the reader is referred to the web version of the article.)

3.1 Wbs [4]. This startup scenario had been used for all the routine plasma operation before 2010. In the 2010 autumn campaign, in order to provide more flux for the plasma Ohmic current drive and heating, we have further applied the TOKSYS model to simulate the plasma breakdown at 4.5 Wb initial poloidal flux. In order to have as high as 0.5 V/m toroidal electric field for plasma breakdown and initial current ramp-up, the breakdown resistors were calculated to be 73.7, 74.5, 55.9, 350, 413 and 288 mΩ s for PF 1, 3, 5, 7, 11 and 13, respectively. The breakdown null field, breakdown loop voltage and plasma initiation got almost the same performances as the previous scenario [4] with initial magnetization state at poloidal flux of 3.1 Wb.

In comparison with the 0 dimensional and rigid model used in TOKSYS for the control design, we also performed 1 dimensional plasma discharge simulation by using TSC (Tokamak Simulation Code) [5]. In the simulation, all the circuit parameters are consistent with the EAST TOKSYS model. We report in this paper a simulation of a typical Ohmic shot (17127). For this shot, plasma current ramped up to 300 kA at 1.73 s. TSC simulates the plasma evolution starting from 0.2 s, when the plasma has been ramped up to ∼90 kA and RTEFIT [9] could provide reliable equilibrium reconstruction. The simulation ends at 4.45 s, when the plasma current ramped down. In the simulation, PF currents are set to reproduce the measured ones, while plasma current, position, shape and parameters evolves by solving the transport equations. Plasma transport parameters and density distribution parameters have been chosen to fit the experiment. The detailed discussion of such modeling and comparison can be found in [6]. The modeled results agreed well with electron density, electron temperature, and even magnetics surface evolution in addition to the measured PF currents which are strong constraints.

Fig. 2 shows the time evolution of the modeled plasma current, radial position, minor radius and elongation in comparison with those in the experiment reconstructed by EFIT [7] from magnetic measurement. These modeled parameters agreed well with those in the experiment. This proves that the EAST TSC model has been matured to a state to reproduce the experiment well and has the potential to be used for the prediction of the future discharge and the development of the advance discharge scenarios if appropriate plasma heating and current drive models been applied. Moreover, it could be also used as a test bed for future plasma control algorithm development as well.

3. Plasma current, position and density control

In Ref. [1], plasma current and position control has been described in detail. Since 2008, PID operations on the plasma current and position errors have been changed from the voltage requests to the power supplies as shown in Eq. (2) in Ref. [1], to the PF current requests. Then PCS has additional loop to control each PF coil current to meet the request. For the control of plasma current, the distribution of requested PF currents is rather straightforward. PF currents distribute in a way as close as possible to generate uniform poloidal flux in the plasma region. For the control of the plasma position, we use PFs 11 through 14 which are located in the outer region of the vacuum vessel. In particular, vertical position is also controlled by the fast control coils, IC1 and IC2, that are located inside of the vacuum vessel and connected in anti-series. Fig. 3 demonstrates the plasma current and position control performance. In these 2 shots, plasma was in circular shape. It can be seen that plasma current was tracked perfectly. Radial position control response had a rise time about 50 ms due to vacuum vessel vertical field penetration time in the order of ∼25 ms, power supply response and other delays. Vertical control response is much faster because the vertical position is also additionally controlled by the

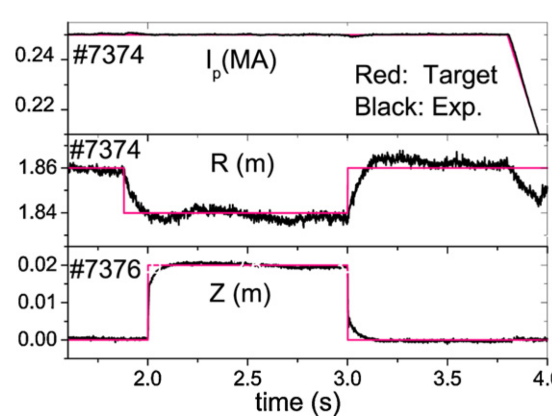

Fig. 3. Demonstration of RZIP control performance.