34th EPS Conference on Plasma Phys. Warsaw, 2 - 6 July 2007 ECA Vol.**31F**, O-4.014 (2007)

The First Diverted Plasma on EAST Tokamak

B.J. Xiao, J. Li, J.R. Luo, J.Y. Zhao, X.Z. Gong, and EAST Team[1]
D.A. Humphreys, J.A. Leuer, A. Hyatt, M.L. Walker and DIII-D Plasma Control Team[2]
and D. Mueller[3]

[1]*Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China*
[2]*General Atomics, DIII-D National Fusion Facility, San Diego, CA, USA*
[3]*Princeton Plasma Physics Lab, Princeton, NJ, USA*

Steady-state operation and exploration of the related engineering and physical issues in advanced economically feasible scenarios are research subjects essential to the commercialized fusion energy. The construction of Experimental Advanced Superconducting Tokamak, EAST, has been completed and its engineering commissioning in March 2006 following five-year-long construction phase. EAST is the presently sole operating fully superconducting tokamak in Mega-Ampere plasma current scale. In the first campaign with only temporary in-vessel metal structures for plasma facing components, the first plasma was initiated in September 2006. The first plasma has been successful with all the superconducting coils operated at 4.5 K with the prescribed current limit of 8 kA, which is well below the designed value of 14.5 kA to ensure the sufficient safety for a new machine, however, the engineering commissioning has attained the designed current value for each individual coil [1]. Stable plasma operation with well controlled plasma current, position and density has been achieved repeatedly. The highest plasma current has reached 500 kA [2]. 3 months after first plasma, EAST started its second campaign with goals of achieving diverted plasma and exploring the engineering feasibilities and constraints for the further improvements.

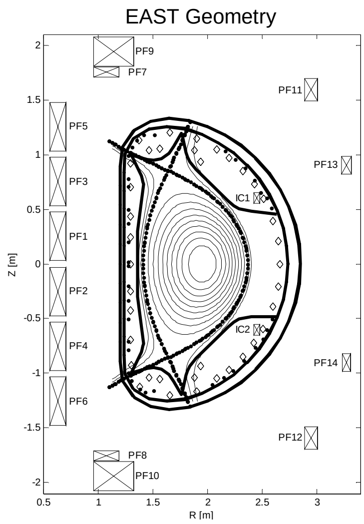
The main parameters of EAST can be summarized in Table 1. It must be noted that the current toroidal field was limited to 2 T for these initial experiments. The TF coils were charged to 8 kA, while the eventual limit is 14.5 kA. The currently available heating method was lower hybrid wave which was mainly used to assist the breakdown and start-up. Fig. 1 shows the position of PF coils and the magnetic diagnostics. EAST has 14 superconductive poloidal field coils. Note that PFs 7 and 9 are connected in series as are PF's 8 and 10 so that a set of 12 independent superconducting PF coil circuits are driven by 12 independent PF power supplies. The vacuum vessel is of double layer SS316-SN structure in thickness of 8 mm each layer. It is toroidally continuous and the one turn resistance is about 74 mΩ. The longest field penetration time corresponding to the lowest-order vessel current eigenmode is about 25 ms. Two internal Copper coils connected in anti-series provide the fast control of the plasma vertical motion.

The magnetic diagnostics consist of current measurements for the plasma, coils and vessel, flux loops for the poloidal flux and magnetic probes for the poloidal fields. 38 magnetic probes (MPs) and 37 flux loops (FLs) were mounted on the vacuum vessel. The plasma current is measured using both an in-vessel Rogowski coil and an ex-vessel one. PF coil currents are measured by both Rogowski coils (used by the plasma control system, PCS) and Hall sensors used for off-line analysis.

Table1. The main parameters of the EAST

Toroidal Field, B_t	3.5 T
Plasma Current, Ip	1 MA
Major Radius, R	1.7 m
Minor Radius, a	0.4 m
Aspect Ratio, R/a	4.25
Elongation, κ	1.6 - 2
Triangularity, δ	0.6 - 0.8
Configuration:	Double/Single-null diverter Pump limiter

Fig.1 PF and magnetic configuration, Circles: FLs; diamonds: MPs



The PCS was adapted from the DIIID PCS [3, 4]. Currently the EAST PCS computer system is comprised of a PC cluster with three real time nodes and one host node, each of which is equipped with a dual- core Intel-Xeon 3.0Ghz processor. The PCS acquires diagnostic data by a real time node equipped with DTACQ digitizers, calculates the controlled parameters and compares with desired waveforms, then controls the parameters using a algorithms and sending the voltage command to the actuators (currently limited to the PF and IC power supplies. The presently implemented control algorithms are the coil current control and limited RZIP control algorithms.

For plasma initiation, the currents in poloidal field coils are optimized to achieve the maximum flux, desired PF and V_{loop} in the plasma breakdown region with the consideration of the vacuum vessel field penetration retarding effect. The PF coils are precharged and discharged through resistors in order to have sufficiently high loop voltage for the plasma breakdown. A feed-forward voltage was added in the PCS discharge waveform during the resistor switch-on period of 50 ms for field adjustment. The maximum magnetic flux is around 3 volt-seconds at the present prescribed current PF limit of 8kA. Initial magnetization state optimization suggests the initial PF currents to be 6.886, 6.978, 8.247, 3.232, 1.274 and 0.765 in unit of kA for PF 1, 3, 5 , 7, 11 and 13, respectively. It is expected that reliable breakdown will happen at an electric field of around 0.5 V/m for the EAST day-one-plasma in the case of the center toroidal field at 2 T under the condition of the metallic first wall and the new machine. The historic EAST plasma discharge is shot 1144, which is first plasma discharge with Ip over than 100 kA. The subsequent shot 1149 has an adjustment on the vertical field and soon made Ip over than 200 kA. The breakdown loop voltage was around 5-6 Volts. The maximum PF ramp-down rate is about 19.5 kA/s at PF1, which is below the prescribed limit for the EAST poloidal coil conductor under the breakdown condition (short period). This limit is due to the consideration of the thermal effects due to eddy current heating of the coils. The reconstruction showed good null field at breakdown. The reliable and repeatable plasma breakdown was achieved using such scenario with additional breakdown assistance using lower hybrid wave at powers up to 50 kW. It is believed that the null field can be improved further and the loop voltage requirement can be reduced with the additional assistance of RF wave. Such optimization remains a future research topic. The well controlled R, Z, and I_P was achieved at a plasma current level of 300 kA on shot 1767 and repeatable.

The second campaign, aiming at divertor configuration, was begun in early 2007. Aside from the availability of the fast inner coils for vertical control, no important changes were made in the machine since the first campaign. After achieving reliable position and current control of the vertical stable plasma, the fast control power supply of ICs was added to the vertical control loop in addition to the outer large superconductive coils PFs11-14 that can perform only slow position control. The control parameters of the vertical position control using the ICs were adjusted to stabilize the plasma vertical motion. After the well controlled vertical motion, the plasma elongation was increased by increasing the currents of the PF7(9) and PF8(10) to stretch the plasma vertically. The gain matrix (M-matrix) was also changed in order to increase the pull force by using PF5 and 6 and push the plasma by further decreasing the currents of PF1 and 2, so that the plasma was made more D- like and diverted to adapt to the vacuum vessel space and the divertor structure. The elongated, diverted plasma was achieved using the coil current control feature of the PCS and by changing the PF currents appropriate for elongated discharge. Actual close-loop feedback of diagnostic estimated elongation and triangularity is planned for next campaign.

Fig. 2 compares the reconstructed magnetic surfaces using the EFIT code and a CCD image of shot 3692 which is a typical diverted discharge. The reconstructed plasma edge shows a good agreement with the CCD image. The longest duration of the stable diverted discharge was about 9 seconds with I_p = 200 kA. The typical elongation and triangularity can achieve 1.9 and 0.5, respectivelly. The maximum I_p reached in the diverted configuration was 500kA. The plasma density range is 1-5 $*10^{19}/m^3$.

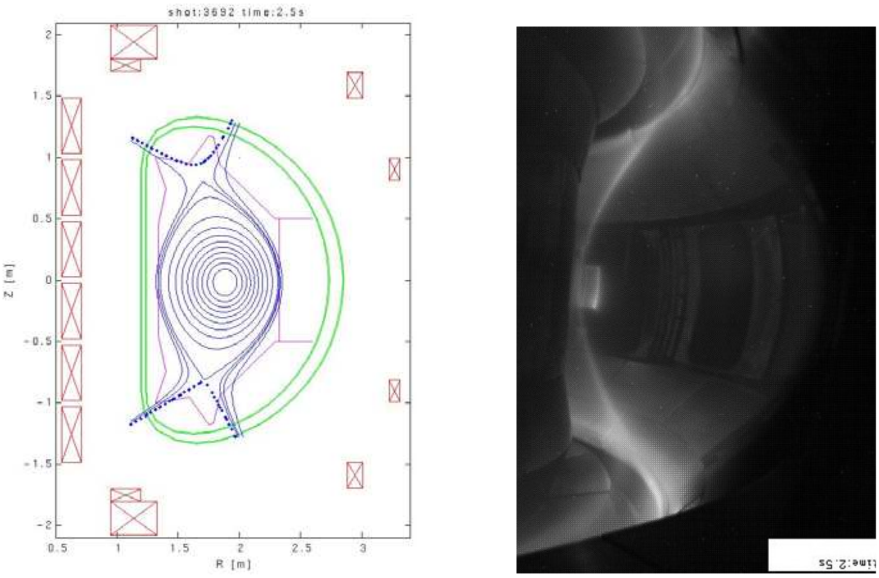


Fig. 2 Reconstructed plasma surfaces and CCD image of shot3692 @2.5s

The EAST next target is the plasma shape control. This will be done by isoflux control with real time EFIT equilibrium reconstruction, which is expected to occur in early 2008.

Acknowledgement

One of the authors, D. Mueller, was supported by U.S.D.O.E. Contract No. DE-AC02-76CH03073.

References:

1. Y. Wan, et al., 21th IAEA Fusion Energy Conf., OV/1-1, Chengdu, China, 2006
2. Wan B., et al., 21th IAEA Fusion Energy Conf., PD P6-5, Chengdu, China, 2006
3. M.L. Walker, J.R. Ferron, D.A. Humphreys, R.D. Johnson & J.A. Leuer et al., , Fusion Engineering and Design 66_/68 (2003) 749_/753
4. J.R. Ferron, B. Penaflor, M.L. Walker, J. Moller, & D. Butnera, Proc. of the 16th IEEE/NPSS Symp. on Fusion Engineering, Champaign, Illinois (Institute of Electrical and Electronics Engineers, Inc., Piscataway, 1996) Vol. 2, p. 870)