Optimization of EAST Plasma Start-Up for Simulations of ITER with Low Loop Voltage

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Abstract. The Experimental Advanced Superconducting Tokamak (EAST) was the first shaped tokamak of mega-ampere plasma scale utilizing a fully superconducting poloidal and toroidal field coil system. Owing to its similarities in coil configuration and constraints to ITER, it can readily determine issues of breakdown, plasma formation and shaping, and plasma current ramp specifically relevant to ITER. Plasma start-up has been optimized to minimize superconducting PF coil AC lose for simulating ITER safe operation in future. EAST achieved reliable Ohmic plasma breakdown and initial ramp-up at toroidal electric field of 0.5 V/m. In 2009, we successfully realized the plasma initiation assisted by lower hybrid wave under toroidal electric field ~ 0.16 V/m. We describe both routine plasma startup scenarios and plasma startup in low electric field aided by LHW in EAST. Control design tools and plasma control algorithms utilized during the first campaign are briefly discussed.

1. Introduction

Most of the existing tokamaks, with poloidal field coils in normal conductor type, were designed with large margins on the key breakdown parameters such as breakdown electric field. However, the tokamaks with poloidal field coils in superconductive states are strictly restrained to their end voltage due to the insulation of the cold structure and the eddy current heating in the coils which would reduce the coil stability. This voltage limitation allows narrower breakdown electric field margin. For example, ITER was designed to be breakdown under the electric field of 0.3 V/m [1]. Robust Ohmic breakdown requires almost twice larger than this field and thus auxiliary pre-ionization or heating method must be applied in addition to the good control of the stray field to ensure sufficient connection length. For the purpose of reducing the startup loop voltage, many existing tokamaks developed their breakdown scenarios in various ways to demonstrate the feasibility of ITER startup scenarios. After improving the control of the stray field, DIII-D realized reliable Ohmic breakdown at 0.3 V/m, and further achieved breakdown at 0.15 V/m with the assistance of the electron cyclotron resonance frequency (ECRF) heating for pre-ionization and to following burn-through [2]. JT-60 even achieved the plasma breakdown under the electric field lower to 0.08 V/m with Helium discharge assisted by low hybrid wave [3].

For EAST, all the field coils are superconductive with the conductors in Nb-Ti cable-in-conduit type cooled by supercritical liquid helium. To avoid the overheating to coils and the possible damage to the insulation layer by high end voltage, the current ramp-rate of all coils is restrained to be less than 20 kA/s in a short time scale such as 50 ms to 100 ms. This limits the toroidal electric field to be less 0.6 V/m on the expected plasma center. Well-controlled null field in EAST four years experiment reveals that breakdown can be robustly achieved under this electric field level. In order to increase the robustness of the plasma startup, low hybrid wave can be used to assist the breakdown and following plasma burn-through. In section 2 of this paper, we report the normal plasma breakdown procedure. Then in section 3 we report the attempt for the breakdown under lower electric field assisted by LHW. In section 4, the plasma current ramp-up and plasma shaping will be discussed. Finally a brief summary will be given.

2. Normal EAST plasma startup

As shown in Fig. 1, EAST poloidal field (PF) coil system consists of 6 center solenoid coils (PF1 to 6), a pair of divertor coils(PF7 and 9 are connected permanently in series, so are PF8 and 10) and 4 large coils (PF11 to PF14) for position and outer boundary control. Major radius of the plasma is about 1.8 m and minor radius is about 0.4 m. Each PF coil is in superconductive state with designed current limit up to 14.5 kA. Each PF coil is driven by a separate individual power supply. For the fast control of the vertical motion of the elongated plasma, a pair of inside coils in normal state are connected in anti-series and driven by a fast power supply.

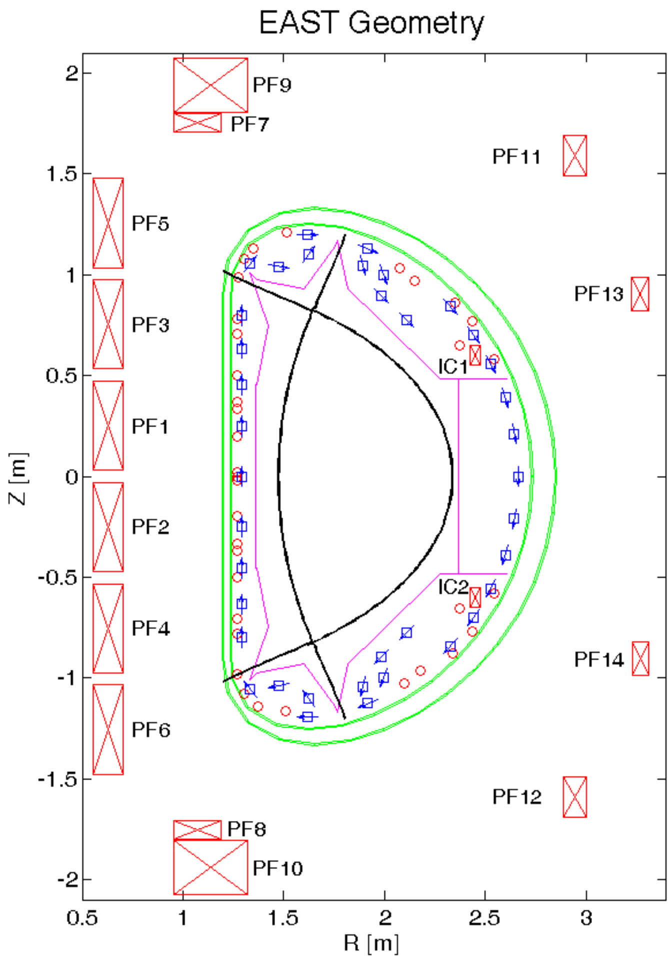


Fig.1 EAST geometry showing poiloidal field coils (red boxes), vacuum veseel (double layer, in green), magnetic probes (small blue squares), flux loops (small red open circles), plasma boundary and first wall or limiter (pink)

Unlike most of the existing tokamaks, EAST has no separate PF coil system for the flux generation and control. Ohmic flux and equilibrium field must be generated by controlling each coil current separately. For the normal breakdown, the power of the power supply is not enough to provide sufficient voltage for driving PF current down as quick as to generate desired breakdown electric field. So in a specified short period, usuallly about 50 -10 ms, individual resistor is switched on to be connected to each PF coil circuit for the resistor discharge. After the breakdown, resistors are switched off and then allow the power supply only to drive coil current.

Before the PF discharge to generate the plasma, each PF coil must be energized to provide sufficient flux for Ohmic heating. Null PF field region inside the vacuum vessel must be large enough to ensure the sufficient connection length for easy plasma breakdown. After the breakdown, the plasma current ramping rate and the equilibrium field must be adjusted in a synchronous mode. For these purpose, an optimization scheme is done to decide each coil current trajectory and the appropriate breakdown resistors. The optimization is by a least square method with constrains such as current derivatives limit and voltage limit targeting at the desired flux and field patterns in the predefined plasma region. Reference [4] have a detailed discussion on this modeling method and tools for such optimization. We use it to predetermine the initial magnetization state and the following PF current ramp-down to generate plasma and its initial plasma current ramp-up. The model takes into account all the current conducting structures either passive conductors or active coils such as PF coils, passive plates and vacuum vessels. This breakdown optimization was included in a MATLAB simulation package, TOKSYS, which is specially designed for generic tokamak plasma control design and simulation [5].

TABLE I: Initial magnetization state and breakdown resistances.

PF	Current (kA)	Resistors (ohm)
1 and 2	7.079	0.122
3 and 4	7.357	0.121
5 and 6	8.297	0.062
7 and 8	3.369	0.485
11 and 12	1.177	0.572
13 and 14	0.7737	0.360

Table 1 shows the normal EAST breakdown initial magnetization state and the breakdown PF resistors. This breakdown scenario has been used for all of the routine plasma operations. The initial flux is 3.1 vs. This breakdown scenario can give rise to maximum of central toroidal electric field of ~ 0.5 V/m without exceeding the constraint of the PF current ramp-down rate below 20 kA/m. The null field in the plasma breakdown region can be well controlled [4,6]. The toroidal field at plasma center is 2 T for all the shots mentioned in this paper.

As shown in Fig.2, in a short period of 30-40 ms after time 0 there is sharp PF current drop due to resistor discharge aiming at the generation of high electric field for the plasma breakdown. Plasma breakdown usually happens around 20 ms after the discharge. After breakdown the resistor still remains 20 ms more in each circuit to provide high voltage for the quick initial rise of the plasma current to pass through burnthrough. The highest plasma ramp rate at breakdown is about 1 MA/s. Vertical field must be adjusted experimentally to be consistent with the actual plasma current rise rate, which was done by adjusting mostly the current trajectories of PF13 and PF14.

It must be noted that the modeling tools and the simulation process introduced in ref. [4] gave a very consistent agreement with what appeared in the experiment. In the actual experimental startup scenario development, only slight adjustment to the predetermined values is needed. This made east day-one- plasma achieve good performance in only a few shots [4,6].

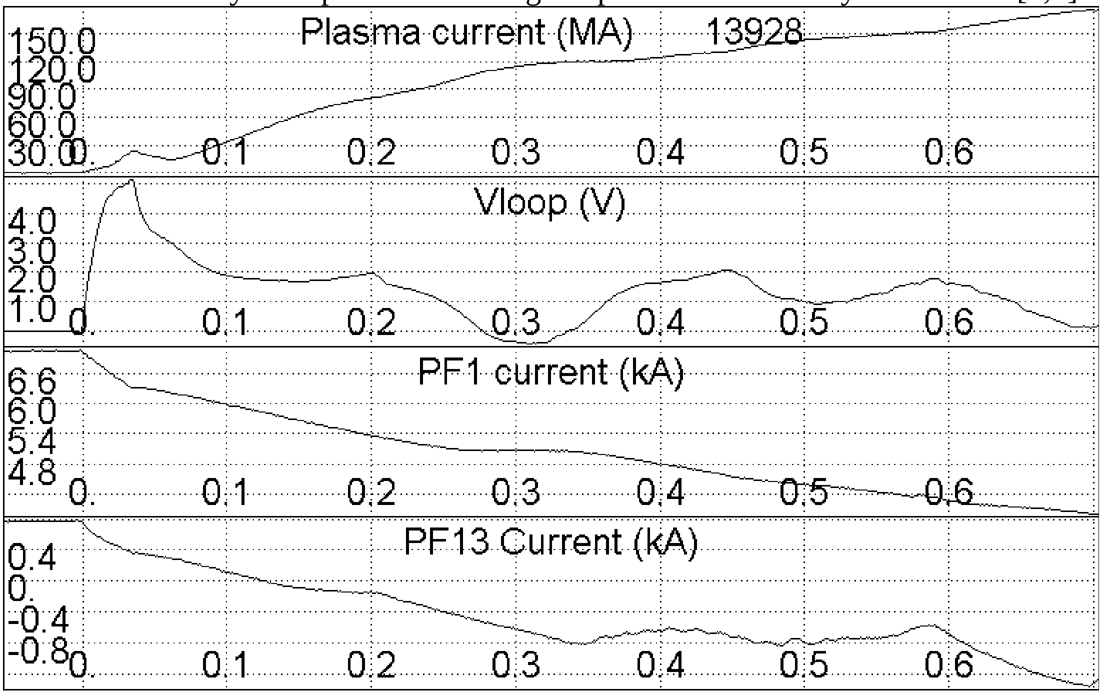


Fig. 2 Typical plasma discharge with breakdown loop voltage about 5 V.

3. Plasma startup under low electric field

As shown in the above section, normal EAST plasma breakdown requires loop voltage at about 5 V to 6 V or the toroidal electric field at about 0.5 V/m. The current ramp-down rate of the central solenoid touched the design limit of 20 kA/s. This electric field is almost twice as high as the ITER designed value. Another fact is that we need to use resistors to be connected for breakdown in order to achieve such high loop voltage. If the electric field can be lowered down to 0.2 V/m, power supply has sufficient voltage to drive current down in a desired rate. In this case, resistor discharge can be avoided which makes the discharge scenario simpler and more robust.

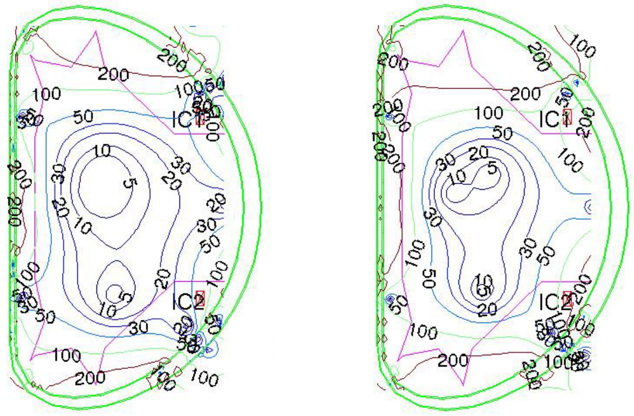
The most effective way in assisting the plasma breakdown including pre-ionization and plasma burn-through is to use ECRH such as that used in DIII-D experiment and ITER in the future. ECRH aided plasma startup can reduce the electric field requirements by a factor of 2, resulting in significant reduction of the eddy current heating and coil end voltage. Flux consumption can also be saved. However, for EAST the only possible pre-ionization tool is to use LHW.

In the resistor discharge scenario, the PF current isn’t well feedback controlled during the short breakdown period. In the low loop voltage startup, we control each PF current in a rate,

$$\dot{I}_{PF} \propto I_{PF}^{IM}, \qquad (1)$$

where I_{PF}^{IM} is the current of PF coils in their initial magnetization states. Before the plasma breakdown if neglecting the current of the vacuum vessel and passive structures, the null field region must be able to be controlled in the same pattern with those in the initial state. The ohmic field penetration time constant of the EAST vacuum vessel is about the 25 ms so after ~ 25 ms we ask PF13/14, may also PF9/10 if PFs 13 and 14 power supplies reach voltage saturation, to drop steeper to provide appropriate vertical equilibrium field. This drop rates can be adjusted shot by shot according to the actual plasma position for the experimental scenario exploration.

Simpler formulation for the PF current trajectory in eq. (1) did give good null field configuration in the experiment. Fig.3 shows the reconstructed vacuum poloidal field at two time stamps in an attempt for the low loop voltage startup. It can be seen that the null field region can be found on the plasma center and the region with $B_p < 20$ Gauss is quite large. In our scenario, LHW was added since the discharge began. The LHW power is about 100 kW. Fig.4 shows the time history of the plasma breakdown and plasma initial ramp-up for shot 14596. It can be seen the plasma breakdown happened at ~ 30 – 40 ms. After the breakdown, we adjusted the vertical field by tuning the PF13/PF14 current. It can be seen that plasma ramp-up smoothly.



Bp @ 20 ms Bp @ 40 ms

Fig.3 poloidal field (Bp) contours, shot 14596

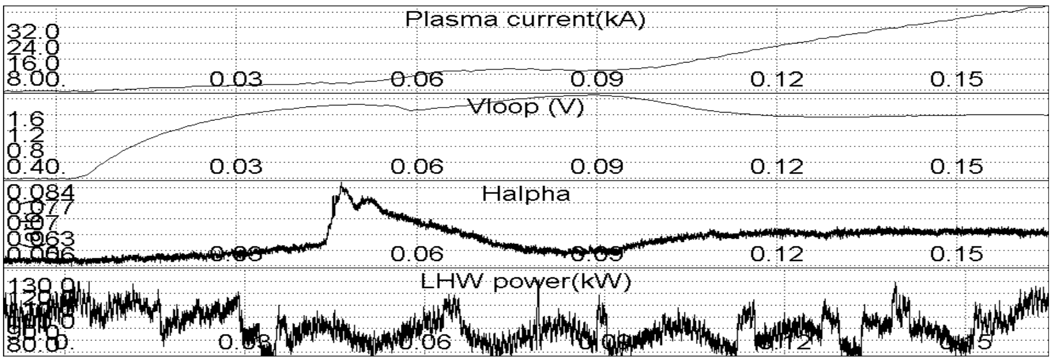
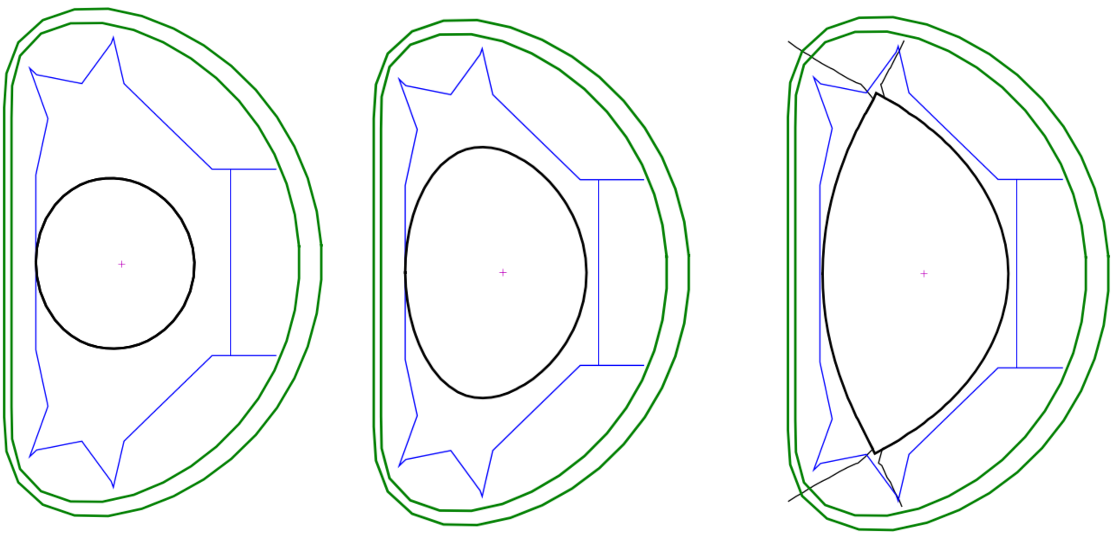


Fig. 4 Low loop voltage startup assisted by LHW @shot 14596, from top to bottom: Plasma current (kA), loop voltage (V), Hα, and LHW power.

4. Plasma ramp-up and shaping

Section 2 and 3 showed that either normal discharge with resistors or the discharge in low loop voltage assisted by LHW can ensure robust plasma breakdown and initial plasma ramp-up. In our control scheme, the poloidal field current is program controlled until the point when the plasma current can be significantly high. After that the estimation of the plasma position are accurate enough and the plasma is strong enough to make the position feedback feasible which doesn't cause much PF current oscillation. The plasma must be gradually shaped from its circular breakdown shape, be elongated and finally reach its pre-specified diverted shape.

An RTEFIT scheme [7] was used for fast equilibrium reconstruction. The reconstruction uses signals from 35 flux loops and 38 tangential aligned magnetic probes as shown in Fig.1 and Rogowski measurements for plasma current, vessel current and PF currents. This reconstruction was done slightly later than the plasma current and position feedback control. After a stable current and position being well controlled, the plasma control is shifted to the RTEFIT/ISOFLUX control algorithm. Plasma is at first limited, gradually elongated and then finally diverted. Fig. 5 shows this plasma shape evolution in shot 13928. For this shot since .5 s, plasma began to be elongated by controlling X points using divertor coils. In the limited configuration, plasma is always limited to inner wall. After 2 s, plasma reached stable double null configuration in an elongation about 1.95.



0.5 s 1.5s 2.1s

Fig.5 Plasma boundaries at 0.5s, 1.5s and 2.1s @shot 13928

Due to the limitation of power supplies' voltage, the vertical field variation speed can't catch up fast plasma current ramp-up rate so that the variation window of the plasma ramp-up rate is narrow. In the pure Ohmic discharge, most of the plasma current ramp-up rate after early breakdown phase is less 0.4 MA/s.

5. Summary

In summary, EAST has demonstrated its startup both in pure Ohmic and the assistance by LHW. By well controlling the null field and applying mild LHW pre-ionization and heating power, EAST can reach robust plasma startup in an electric field less than 0.16 V/m. EAST is upgrading some power supplies to increase their power limit which allows more robust equilibrium control. With such update, the startup scenario can be systematically explored in the future.

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