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Recent experiments towards to the steady-state operation in the EAST and HT-7 superconducting tokamaks

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Abstact

The first plasma has been successfully obtained just after EAST engineering construction and followed by achievement of diverted plasma configuration in the second campaign under the full metal wall condition. To support the long pulse diverted plasma discharges, new capabilities including the fully actively water cooled in-vessel components, current drive and heating powers, diagnostics, real-time plasma control algorithm were developed. These developments were primarily validated in recent experimental campaign, such as realization of RTEFIT/ISOFLUX control algorithm, LHCD long pulse discharges over 20 seconds, etc. Recent experiments in HT-7 focused on long pulse discharges under different scenarios. The long pulse discharges up to 400s renews the records in HT-7 after a series of system modification.

Key words: superconducting tokamak, long pulse discharge, lower hybrid current drive

1. Introduction

EAST (Experimental Advanced Superconducting Tokamak) was completed its engineering construction and achieved the first plasma in 2006[1,2]. First divertor plasma in EAST was obtained in the second campaign in 2007. As first full superconducting tomakak with highly shaped plasma configuration, EAST is aimed to the high performance plasma and relevant technologies under steady state condition. To meet these requirements, EAST has the actively cooled plasma facing components (PFCs) and is equipped with lower hybrid current drive (LHCD) and ion cyclotron resonant heating (ICRH) systems. It could address a number of issues regarding the technology of steady-state divertor control and physics of long pulse operation with non-inductive current drive.

In last few years, HT-7 experiments were strongly oriented to support the EAST project both physically and technically[3], such as long pulse discharges, high performance plasma investigation, etc. Experience and experiments from the HT-7 become one important part of the EAST basis and certainly speeds up the EAST procedure. This paper will report the main progress of machine modification and operation on EAST and long pulse experiments on HT-7.

2. System Development

The long pulse high performance operation of a tokamak requires specific in-vessel structures and PFCs, which should be capable to handle the particle and heat fluxes in a variety of operation scenarios under steady-state condition. The in-vessel structure on EAST is a complicated integration of multi-systems as shown in Fig.1. They are composed of the fully actively water cooled PFCs and supporting structures, a full set of magnetic inductive sensors for machine operation and plasma control, the divertor cryopump, the actively water cooled internal coils for vertical

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stabilization control, divertor probe arrays, baking system and thermal coupler etc. The system of the actively cooled PFC is a key element in construction of the new in-vessel structure. Fig.2 shows the EAST in-vessel structure together with ICRF antenna and LHCD launcher after full construction. The in-vessel geometry is designed as top–bottom symmetry to accommodate both double null or single null divertor configuration.

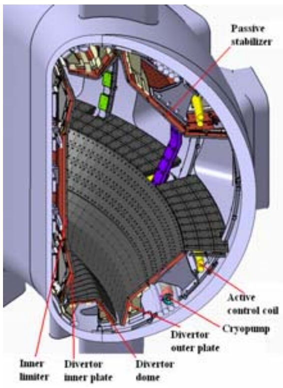

Fig.1 Elevation view of EAST in-vessel structure

Fig.2 Picture of in-vessel together with the ICRF antenna and LHCD launcher

Present PFCs structure can handle a peak heat flux up to 3.6MW/m^2 on the divertor plates and 0.5MW/m^2 on other plates. Bolted tiles affixed to an actively cooled heat sink are employed in the initial PFCs engineering. All plasma facing surface are made by one kind of multi-element doped graphite materials[**4**]. Such a structure with 2 t/h water mass flow rate for inner target, outer target and dome can maintain plasma facing surface temperature around 800℃ under peak heat load up to 3.6MW/m^2 according to the thermo-hydraulic analysis. Each of the upper and lower divertor structures consist of three high heat flux targets: inner, outer and private baffle (dome) plates. The vertical targets and dome form a "V" shape. Two gaps between inner, outer target and dome were provided with total 180m^3/s gas conductance for particle and impurity exhaust by cryopump, which was installed behind first wall and diverter plates. All PFCs were divided into 16 modules in toroidal direction for easy maintenance and modification. Two movable molybdenum limiters have been installed, which allow radial movement from 2.26m to 2.42m. In-vessel coils close to plasma with fast power supplies are utilized for vertical stabilization, which are also shown in Fig.1. The coils have been actively water cooled and can be operated at a current up to 20kA/turn under steady-state condition.

EAST vacuum chamber has double layer structure and can be baked up to 250℃ by high pressure hot nitrogen gas. The PFCs can be baked up to 350℃ also by hot nitrogen gas flowing in the cooling lines of the heat sinks. Other components including the 4 DC glow discharge anodes and 2 RF conditioning antenna were installed for wall conditioning. The RF conditioning is used normally for boronization and wall cleaning. The thermal couplers are attached on the chamber and port extension walls and embed in the graphite titles of the liner, limiter and divertor for wall conditioning and also plasma discharges.

All magnetic sensors are newly manufactured and installed in vacuum chamber as a part of the integration of whole in-vessel components. They provide sufficient information for machine operation, plasma control and physics analysis. Two Langmuir probe and mach probe arrays were integrated on the specific divertor and dome modules. They were made by graphite rode and cooled through thermal conducting to the heat sink of the module, which is actively water-cooled. The probes were installed both in top and bottom diverter plates and dome respectively. The individual probe tips can be easily configured as triplet probe array along the divertor plates with a poloidal resolution of 2cm. The mach probe has 4 tips, which are oriented for toroidal and poloidal velocity measurements.

An advanced x-ray imaging crystal spectrometer (XCS) in collaboration with PPPL and NFRI has been installed at the end of the main pumping duct for ion and electron temperatures measurements[5]. There is a single channel laser interferometer with the vertical sight line at R = 1.9 m, which provide a line integrated density for density feedback control. Two visible CCD camera look at plasma in tangential sight line to monitor plasma discharges. Additional 20 diagnostics installed presently can provide measurements of electron temperatures, surface temperature of the liner or divertor plates, radiation power, and information of soft-X ray, visible to near UV radiation of impurities, H_{α} radiation etc. Hard-X ray and neutron flux measurements are also available for LHCD experiments and monitoring the runaway electrons.

The RF systems at ICRF with 1.5MW, 30~110MHZ and at LHF with 2MW, 2.45GHz are available for heating and current drive experiments as well as wall conditioning and discharge pre-ionization.

3. The experiments with new PFCs

EAST as a full superconducting tokamak has new features compared to conventional tokamak and also those tokamaks only with the toroidal superconducting coils. The key issue is to reduce the PF current variation rate by optimizing plasma operation scenarios, particularly, during plasma current buildup phase, which is important for stability and safety of the superconducting magnets with fixed cooling capability under steady-state condition. To achieve reliable break down and plasma current ramping up, the RF cleaning and boronization were used as routine wall conditioning on EAST. The working gas was hydrogen in 2007 and switched to deuterium for recently campaign in 2008.

The experiments have firstly performed using pre-programming shape control and feedback control for plasma position and current with the internal control coils (ICs) for vertical stabilization. To verify the shaping capabilities of the poloidal field system and vertical stabilization of ICs, highly shaped plasma at various configurations has been stably produced. The following configurations have been stably produced: double null configuration with kappa=1.9 and delta=0.50; top or bottom single null configurations with kappa=1.7 and delta=0.64 with a plasma current up to 0.6MA. Above results almost cover all designed configuration in EAST and confirm the capabilities of the poloidal field systems including ICs and their power supplies for the present operation regime. Such experiments provide the important basis for algorithm development and optimization of real time plasma shape control.

The full reconstruction of the equilibrium has be performed by using EFIT[6] code routinely between shots. This kind of reconstruction was made to be real-time (RTEFIT) and sufficiently fast for the real-time shape control in DIII-D by using a fast loop and a slow loop calculations on separate CPUs. While RTEFIT has been done at a control cycle, the control reference points were determined at first. The flux difference between measured and pre-defined at the reference points were controlled to be zero based on the called RTEFIT/ISOFLUX algorithm, This control algorithm was primarily realized on EAST in 2008 campaign, which provides good basis for RF coupling.

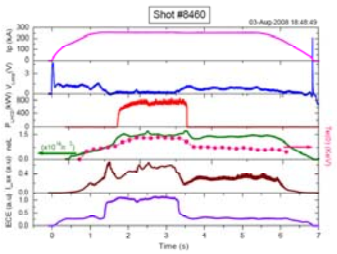

Fig.3 delivered 800kW LHCD power for nearly full non-inductive current drive

The LHW was used for current drive both in sustaining plasma discharges and assisting the plasma start-up. The plasma shape and position were optimized to maximize the wave coupling into the plasma. Nearly 0.8MW LHW at a fixed $N_{\|}^{peak}$=2.3 has been successfully delivered, from what about 0.65MW power has been coupled into the plasma shown in Fig.3. This power can nearly sustain a fully non-inductive plasma discharge at Ip=250kA and line averaged density of ~ $1.5*10^{19}m^{-3}$. The current driving efficiency under this condition is about $0.8*10^{19}$ $Am^{-2}W^{-1}$ shown. Significant electron heating by LHW has been observed, while ion heating is very weak. The plasma discharges can be sustained over 20 seconds in such operation scenarios (Fig.4) under present condition. The LHCD experiments were also performed at different plasma current. At present conditions, LHCD can sustain plasma discharges of 400kA and line averaged density of ~ $1.5*10^{19}m^{-3}$ for a duration longer than 10s. Maxima current drive efficiency up to $1.0*10^{19}$ $Am^{-2}W^{-1}$ has been achieved. It means the possibility to sustain fully non-inductive plasma discharges at 500kA and line averaged density of ~ $2.0*10^{19}m^{-3}$ with presently available LHW power of 2MW, which will be a next goal.

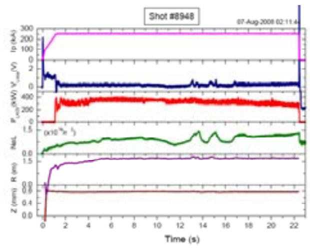

Fig.4 Long pulse discharge sustained by LHCD for 22s

Plasma initiation, ramp up and control are very important issues for a full superconducting machine. Full use of the PF capability can create a loop voltage of ~6.5V with a reasonable null field configuration, which will produce vessel currents up to 150 kA [2] at breakdown and lead to a loss of poloidal magnetic flux of about 0.3 V*s. Low loop voltage startup has beneficial not only for safety of the machine operation, but also for reduce loss of poloidal magnetic flux due to vessel current. Break down at a toroidal electric field of 0.3V/m has been achieved by optimizing the null field configuration, gas pressure and assistance of the LHW of 100kW and well conditioned wall, which increases the safety margin of PF coils significantly due to reduced the PF current ramping rate. At the same time, the loss of poloidal magnetic flux due to vessel current (<100kA) is only 0.1V*S. Very low plasma ramp rate of 0.12MA/s during plasma current ramping up phase have been obtained with assistance of LHCD, which can significantly reduce the current ramping rate in PF coils or voltage applied at PF coils for the same plasma current ramping rate. This operation mode, on one hand, minimized the heat deposition on the PF coils caused by AC losses, and hence, increases safety of machine operation. On another hand, it allows better plasma control, particularly, during shaping phase due to the larger voltage regulation margin of PF power supply.

4. Experiments in HT-7

In last few years, experiments in the HT-7 tokamak focused on long pulse discharges under different scenarios to support EAST experiments both physically and technically. To meet the long pulse operation requirements, several systems around HT-7 were modified. The plasma control algorithm was implemented based on real time magnetic equilibrium reconstruction with the improved magnetic diagnostics. The iron core is simplified by using the "spool" model [7] and gaps of the last closed flux surface from the PFCs is adopted for the plasma control. New heat sink technology and material were utilized to replace the belt limiter at high field side for validation and supporting construction of the EAST in-vessel components.

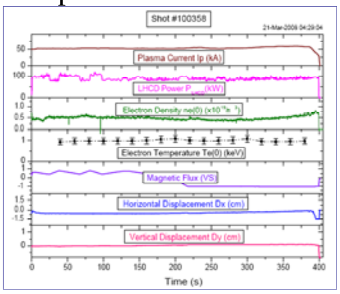

Fig.5 A long pulse discharge for 400s

The long pulse experiment was performed by driving the plasma current fully non-inductively through the use of LHCD, which was realized in two different scenarios. The first one is via feedback control of the magnetic swing flux of the transformer at a constant. The second way is so-called transformer-less discharges, which was realized by over current drive up to reversed saturation of the transformer and then switch off the current in the central solenoid. Experimental observation by hard X-ray pulse height analysis indicates that the cut-off energy of supra thermal electrons are smaller in the transformer-less discharges than in the first scenarios. These techniques has been used successfully to sustain the plasma discharges for 400s at Ip~50 kA shown in Fig 5, which is the new record in HT-7. The magnetic flux of the transformer was controlled at a constant during first 150s in these three shots and then over current drive led to switching off the central solenoid at about 200s, 210s, and 270s respectively. There was no observable hot spot during transformer-less discharges, which might be correlated with vanish of further acceleration of fast electrons driven by LHCD. In such an operation mode, the surface temperature of the belt limiter could be well controlled below a certain value for whole plasma discharge duration as shown in Fig.20. These long pulse experiments indicate success of the new built belt limiter, and more important is to validate the same heat sink material and structure applied for the EAST PFCs. The well controlled surface temperature also suggests important role of supra thermal electrons on the heat load at the limiter. The limitation for even longer pulse is due to uncontrollable density rise caused by out-gassing mainly from the first wall, which was heated by plasma radiation and not actively cooled.

5. **Summary and near future plan**

Significant progress in construction of the fully actively water-cooled in-vessel components and plasma control in obtaining highly shaped plasma has been made on EAST. The primary achievements, particularly, the experiences from last two years provide us confidence that the highly shaped plasma with relevant performance could be sustained by RF powers for long duration, although some improvements are needed for reliable machine operation and effective experiments.

Totally 8MW RF systems will be available in 2009 and additional heating and current drive power of 8MW is expected before end of 2010. The flexibilities of heating scenarios and current drive in controlling power deposition and current density profile provide the possibilities to operate EAST in high performance regime with edge and/or internal transport barrier, which allow investigation focused on advanced scenarios for long pulse. In the next two years, diagnostics on EAST will provide measurements of all key profiles, which should be sufficient to describe the basic plasma performance and for integrated modeling and data analysis.

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