Recent experiments in the EAST and HT-7 superconducting tokamaks

Baonian Wan for the EAST and HT-7 teams and international collaborators *
Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China

e-mail contact of main author: bnwan@ipp.ac.cn

Abstract: First diverter plasma configuration in EAST was obtained in the second campaign after the last IAEA meeting. To support the long pulse diverted plasma discharges, new capabilities including the fully actively water cooled in-vessel components, current drive and heating power, diagnostics, real-time plasma control algorithm were developed. Pre-program shape and RZIP feedback control produce a variety of shaped plasma, which cover almost designed configuration. RTEFIT/ISOFLUX control algorithm was primarily realized. A number of operational issues, such as plasma initiation, ramp up and configuration control with constraints of superconducting coils were investigated. The physical engineering capability on the superconducting magnetic system was assessed by simulating discharges. Since the last IAEA meeting, 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. The investigation of sawtooth activities in HT-7 by 2D ECE image and high resolution soft-X arrays support the turbulence model instead of the fast reconnection of the m=1 magnetic island. Coexistence of electron mode and ion mode in high density ohmic plasma has been observed by 2D ECEI in HT-7.

1. Introduction

Achievement of the first plasma has demonstrated that EAST (Experimental Advanced Superconducting Tokamak) engineering construction is completely successful[1,2]. First diverter plasma configuration in EAST was obtained in the second campaign just after the last IAEA meeting. EAST is presently equipped with actively cooled plasma facing components (PFCs) and with 8~10MW radio frequency (RF) heating and current drive systems in next 2 years. It is currently a unique facility to explore some of critical issues relating to steady-state operation with the shaped plasma cross-section. It could become a good test bench, especially regarding the technology of steady-state diverter control and physics of long pulse operation with non-inductive current drive and full superconducting machine operation.

The institute of plasma physics, Chinese Academy of Sciences (IPP/CAS or ASIPP) has EAST as new project and simultaneous the HT-7 superconducting tokamak in operation. Significant progress has been achieved in the EAST construction/operation and the HT-7 experiments with many contributions from broad international collaboration. In last two years, HT-7 experiments were strongly oriented to support the EAST project both physically and technically[3,4].

This paper will report the main progress of machine modification and operation on EAST and long pulse experiments on HT-7. Section 2 briefly describes the development of systems on EAST, mainly, the new in-vessel structure. The experimental progress with the new in-vessel structure is given in section 3. The experimental progress of long pulse discharges in HT-7 are reported in section 4 followed by a summary and near future plan in section 5.

2. System Development on EAST

The first EAST plasma was achieved under the full metal PFCs condition[2]. EAST as a full superconducting tokamak is aimed for long pulse (60-1000s) high performance operation,
which requires specific in-vessel structures and PFCs. It should be capable to handle the particle and heat fluxes in a variety of operation scenarios under steady-state condition. The in-vessel structure is a complicated integration of multi-systems as shown in Fig.1. They include the fully actively water cooled PFCs and their 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 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 geometry is designed as top–bottom symmetry to accommodate both double null or single null divertor configuration. This section only gives a very brief summary of the newly developed system around EAST.

![Elevation view of EAST in-vessel structure](image1)
![Picture of in-vessel together with the ICRF and LHCD launcher](image2)
![Elevation view of EAST toroidally continuous cryopump](image3)

### 2.1 Plasma Facing Components

In the first stage of the machine operation, the peak heat flux will not be more than 3.6MW/m². From economical and technical consideration, brazed tiles are not employed in the initial PFCs engineering instead of using bolted tiles. The PFCs consist of a plasma facing surface affixed to an actively cooled heat sink. All plasma facing surface are one kind of multi-element doped graphite materials\[5\]. The 15mm or 20mm (especially for divertor) thick graphite tiles are bolted to the copper alloy (CuCrZr) heat sink and retrained through the spring washers that allow limited deformation during thermal expansion. A thin graphite sheet of 0.38mm is used between the tile and the heat sink to improve the thermal contact. Water-cooling channels are drilled holes directly along the 20mm thick heat sink plates. 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°C 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). The vertical targets and dome form a “V” shape. Two gaps between inner, outer target and dome were provided with total 180m³/s gas conductance for particle and impurity exhaust by cryopump. Other PFCs except the divertor region have the same structure but with lower cooling capability and allow maximum heat load of 0.5MW/m². All PFCs were divided into 16 modules in toroidal direction, respectively for easy maintenance and modification. Halo current is considered to be 50% of plasma current and its
toroidal asymmetry factor is 2 in the design. Two movable molybdenum limiters have been installed, which allow radial movement from 2.26m to 2.42m.

2.2 In-vessel Coils and Cryopump
For a full superconducting machine, coupling between the PF coils and plasma is relative weak. On another hand, the current variation rates in the superconducting PF coils are limited due to the AC losses, which affect the stability of the magnets. These lead to insufficient capability to stabilize the highly shaped plasma. In-vessel coils with active water cooling close to plasma with fast power supplies are utilized as a solution for vertical stabilization, which are also shown in Fig.1. The corresponding power supply is operated in current mode with a temporal response of 100 μs.

A cryopump are installed behind first wall and divertor plates to avoid facing plasma directly and fixed to vacuum vessel wall as shown in Fig.1. It is located in a plenum near the gap between the doom and outer targets with pumping speed > 15 m³/s. The pump is continuous tubes with slot open to vacuum vessel wall and are supported by flexible supports (Fig.3).

2.3 Elements for wall conditioning
EAST vacuum chamber has double layer structure and can be baked by high pressure hot nitrogen gas up to 250°C. The PFCs can baked also by hot nitrogen gas flowing in the cooling lines of the heat sinks up to 350°C. Other components including the 4 DC glow discharge anodes and 2 RF conditioning antenna were installed for wall conditioning. The RF conditioning is used for boronization and wall cleaning between the shots. The thermal couplers are installed on various components for wall conditioning and plasma discharges.

Fig. 4 a) poloidal distribution of flux loops. B) poloidal distribution of magnetic probes
Fig.5 a comparison of measurement (triangle points) and calculation (line) by exciting the PF coils.
Fig.6 A helium-like Argon spectrum taking from the central sight line

2.4 Diagnostics
All magnetic sensors are newly manufactured and installed in vacuum chamber as a part of the integration of the in-vessel components. These magnetic sensors provide sufficient information for machine operation, plasma control and physics analysis. The distribution of flux loops and magnetic probes along the poloidal direction is shown in Fig.4. All magnetic sensors are carefully mounted and calibrated with the newly developed low-drift integrator together to assure the accuracy needed for plasma configuration reconstruction. Uncertainties of most magnetic probes and flux loops are less than 10 Gaussian and 2% respectively, which an example of the testing result is shown in Fig.5.

Two Langmuir probe and mach probe arrays were designed as an important integration of the specific divertor and dome modules. They were made by graphite rode and cooled through
thermal conducting to the heat sink of the modules. 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 of EAST, which can record temporally and spatially resolved spectra of helium-like ions from multiple sightlines through the plasma for the measurement of ion and electron temperature profiles [6,7]. Figure 6 gives an example of the helium-like Argon spectrum taking from the central sight line by puffing Ar gas as seed impurity into an ohmic plasma on EAST.

There is a single channel laser interferometer to provide a line integrated density and is used 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, Hα radiation etc. Hard-X ray and neutron flux measurements are available for LHCD experiments and monitoring the runaway electrons.

2.4 Plasma Control System
Plasma discharges are controlled by a plasma control system (PCS) built in collaboration with GA, which is similar to the PCS of DIII-D, but with new EAST features such as the coil current ramping rate limitations due to the eddy current heating on the superconducting cables, IC power supply command algorithms, etc [8].

2.5 RF systems
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 pre-ionization. A actively cooled resonant double loop antenna with two straps was designed in collaboration with Tore Supra teams, France. Two current straps can be powered with phase shift for heating and current drive, which provides the capabilities for various operation scenarios. The LHCD system is similar to the system used in HT-7 but with higher power and narrower FWHM of 0.2 [9].

3. The first experiments with new PFCs

Fig.7 Operation space for highly shaped plasma discharges
Fig.8 Operation space for vertical plasma stability
Fig.9 confinement scaling for ohmic plasma discharges

EAST as a full superconducting tokamak has new features compared to conventional tokamak.
These issues, particularly, the limitation of current ramping rate in PF coils, relative weak coupling between plasma and PF coils, PF field penetration through the vacuum vessel and thermal shielding into the plasma etc affecting the machine operation have been discussed briefly[2]. The key issue is to reduce the current variation rate by optimizing plasma operation scenarios, particularly, during plasma current buildup phase, which is important for stability of the superconducting magnets with limited 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. The principal goal of this experiment was achieved with the appearance of stably controlled diverted plasmas with sufficient elongations and tri-angularities. To verify the shaping capabilities of the PF system and vertical stabilization of ICs, highly shaped plasma at various configurations has been stably produced. The result is summarized in Fig.7 for two campaigns of 2007 with full metal wall and 2008 with the new PFCs. The relation of elongation and internal inductance for well stabilized and shaped plasmas is given in Fig.8. Above results almost cover all designed configuration in EAST. The discharges with plasma current \( I_p = 0.2-0.6 \text{MA}, B_t = 2-3 \text{T} \) show the confinement consistent with Neo-Alcator scaling for ohmic plasmas shown in fig.9. Such experiments provide the basis for algorithm development and optimization of real time plasma shape control.

![Fig.10 Left: reference point for iso-flux control, right: plasma shape controlled by iso-flux from 3s](image_url)

**Fig.11** main controlled parameters at the isoflux control algorithm

**Fig.12** delivered 800kW LHCD power for nearly full non-inductive current drive

The full reconstruction of the equilibrium has be performed by using EFIT[9] 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. 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 be controlled to be zero based on the called RTEFIT/ISOFLUX algorithm [10]. Under the collaboration with DIII-D, EAST also adapted DIII-D plasma control software system[8]. RTEFIT/ISOFLUX was primarily realized on EAST in 2008 campaign shown in Fig.10. Figure 11 shows the main controlled parameters at the isoflux control algorithm. It can be seen that the flux errors were well controlled below 5 mill-Vs for most of the discharge time under shape control which started from 3 s. The X point position control is rather good for up X-point, but in the range of
several centimeters for lower X-point. In order for X positions to be well controlled with much lower error, the shape control has to be sufficiently consistent with the fast vertical position control. This leaves to be fulfilled at the next experimental campaign.

3.2 LHCD experiments

The LHW was used for current drive both in sustaining plasma discharges and assisting the plasma start-up. LHCD experiments were performed by pre-programming control of plasma shape and feedback control of RZIp. 0.8MW LHW at a fixed $N_{\text{peak}}=2.3$ has been successfully delivered, from what about 0.65MW has been coupled into the plasma shown in Fig.12. The current driving efficiency under this condition is about $0.8 \times 10^{19} \text{Am}^{-2}\text{W}^{-1}$ shown in Fig.13. Significant electron heating by LHW has been observed both by soft X-ray pulse height analysis and XCS, while ion heating is very weak. Figure 14 shows the electron and ion temperature profiles derived from XCS for two LHCD discharges of 400 kW and 650 kW injection powers. Electron heating dominates in the plasma core, which implies the central power deposition of LHW consistent with the code prediction.

The plasma discharges can be sustained over 20 seconds in such operation scenarios (Fig.15). The main limitation for higher power and longer pulse is the unstable coupling due to pre-programming shape control and the power supply stability of the LHCD system. The LHCD experiments were performed at different plasma current. At present conditions, LHCD can sustain plasma discharges of 400kA and line averaged density of $\sim 1.5 \times 10^{19} \text{m}^{-3}$ for longer than 10s. Maxima current drive efficiency up to $1.0 \times 10^{19} \text{Am}^{-2}\text{W}^{-1}$ has been achieved. It means possibility to sustain fully non-inductive plasma at 500kA and line averaged density of $\sim 2.0 \times 10^{19} \text{m}^{-3}$ with presently available LHW power of 2MW.

3.3 Startup of Plasma discharges

Full use of the PF capability can create a toroidal voltage of $\sim 6.5 \text{V}$ 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 shown in Fig.16. The maxima current ramping rate in PF coils is 10kA/s with a voltage only half of the normal operational voltage for break down, which increases the safety margin of PF coils significantly and reduce the loss of poloidal magnetic flux to 0.1V*S.

Very low plasma ramp rate of 0.12MA/s during plasma current ramping up have been
obtained under a very well boronized wall condition. Such low ramping rate led to operation of the PF coils and power supplies in the region far from their limits. LHCD applied at plasma start up phase 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. Typical results with and without LHCD start up assistants are compared in Fig. 17. The main issue applying the LHCD during start-up phase is the coupling optimization.

![Fig 16. Low voltage start-up with assistant of LHW](image)

![Fig.17 Plasma current ramp-up with/without LHCD.](image)

![Fig. 18 Outlet temperature rise of CS at excitation rate of 1kA/s](image)

### 3.4 Physical engineering effect

In instance of frequent variation of the plasma performance such as large ELM burst, monster sawteeth, etc, the control demands response of the PF currents to maintain the plasma configuration. For typical EAST plasma at current of 500kA, the maxima PF current variation of about 300A is needed to keep the configuration if the poloidal beta is changed by 10%. This will cause a continuous PF coils current variation rate of about 3kA/s with a time scale of 0.1s. The continuous heat deposition cased by AC loss on the PF coils can affect steady-state operation of superconducting machine. These issues have been investigated by simulating discharges or during plasma discharges. Plasma disruption seems not be a direct safety constrain for the superconducting PF coils due to the strong shielding effect from the vacuum vessel[2]. But such algorithm has to be adopted in controlling the PF current ramping down rate after termination of the plasma discharge to avoid higher thermal load on the magnets and cryogenic system. The energy deposition on TF coils, TF cases and PF coils due to the AC losses caused by varying PF currents has been investigated at different current ramping rates for a sufficient duration up to saturation of the outlet liquid helium temperature in the PF coils. Figure 18 shows an example for the outlet temperature rise of the CS during the excitation at constant rate of 1kA/s in triangular waveform of the PF current. The temperature rise seems to be saturated for sufficient excitation duration and is still below the safety margin of 1K. To keep the magnet temperature below the safety margin at higher PF current variation rate require more cooling power. The capability of the cryogenic system on EAST is sufficient to keep the magnet temperature much lower than the marginal values if the current varying rates in PF coils do not exceed the designed specification. More issues about the safety operation for the superconducting magnets have been discussed elsewhere [12].

### 4. Experiments in HT-7

Since the last IAEA meeting, experiments in the HT-7 tokamak focused on long pulse
discharges under different scenarios. 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 [13] 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.

**4.1 Long pulse discharges**

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. These techniques has been used successfully to sustain the plasma discharges for 400s at Ip~50 kA shown in Fig 19, which is the new record in HT-7. The magnetic flux of the transformer was controlled at a constant for first 150s and then over current drive led to switching off the central solenoid current. 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. 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 limitation for even longer pulse is due to uncontrollable density rise caused by out-gassing mainly from the first wall, which was not actively cooled.

**4.2 Sawteeth investigation**

Sawtooth oscillation is one of the most important MHD instabilities in tokamaks. Recently, progress has been achieved by newly developed 2D ECEI and high-resolution soft-x-ray multi-arrays to verify new models[14,15]. There are developed 5 high-resolution soft-x-ray arrays and a 2D ECEI system 8(radial)×16(vertical) channels on HT-7 for investigation of sawtooth activities and temperature turbulence. Islands coalescence is observed in ohmic plasmas with Bt=1.9T, Ip=170kA and central
line-averaged density of $2.4 \times 10^{19}/m^3$ before reconnection during m=1 sawtooth crash by the 2D ECEI system. A representative view of the sequence of 2D ECEI images during the reconnection process is provided in Fig 21. The reconnection happened on the q=1 radius in the LFS makes the magnetic field line open. As a result, the heat flow escapes to the outside of the inversion radius collectively. Similarly, magnetic reconnection is also found by 2D image in the HFS. The characteristic of magnetic reconnection near inversion radius during sawtooth oscillation is analogous to that observed on TEXTOR [14]. Fig. 22 shows the temporal evolution of m=2 islands coalescence in q~1 surface during precursor phase of sawtooth. The first island rotates anticlockwise into the diagnostic region in the frame 1 and 2 and still stays in the frame 3, while simultaneously another island rotates into the diagnostic region below. The two islands reconnect near the mid-plane in only about 10 microseconds as shown in the frame 3 and 4 and finally, the two islands coalesce into one island as m=1 mode in the frame 5. After the formation of the m=1 island, the heat in the core escape to the outside of the inversion radius during sawtooth crash during the frame 6 to 8.

The ECEI presents clearly the characteristics of the temperature fluctuation during the sawtooth crash phase. However, the observation is limited by a small region. Tomography of high-resolution soft-x-ray arrays can give the sawtooth crash structure. The Fourier-Bessel inversion method is used in the reconstruction by choosing M=2-3 and L=6-8 [15]. The sawtooth activity in LHCD plasma is analyzed due to the large fluctuating amplitude for distinguishing the crash and precursor stage. In order to obtain the picture of variation of the plasma pressure due to sawtooth oscillations, Singular Value Decomposition (SVD) technique is employed to extract the perturbation components from the signals, which is used to reconstruct the mode structure of the oscillations.

By using this method, the reconstructed sawtooth crash process from perturbation signals is shown in the bottom frames of Fig. 22. A large heat flow transfer across the x point of (both the m=1 and m=2) islands (C-E) and then the transferred energy gradually spreads poloidally over the peripheral region near the q=1 surface (E-F), while the reconstructed sawtooth crash process from total signals, as shown in the middle frames in Fig. 22, seems to be consistent.
with the Kadomtsev’s model, which was also concluded in Ref [15]. The asymmetric heat flow can occur both in low field side and high field side, which is different from the observations in high $\beta$ plasmas on TFTR [16]. The heat flow characteristics near the x point during sawtooth crash phase are consistent with the observations by 2D ECEI. Another type of sawtooth crash is shown in Fig. 23, where the heat flow transferring is not across the x point of the m=1 island instead of one of the x points of the m=2 magnetic islands. This suggests that the large m=1 component in the soft-x-ray signals is due to the large asymmetric heat flow during sawtooth crash phase rather than the abruptly increasing of the m=1 magnetic island. Another interesting observation shown in Fig. 24 is that the heat flow transfer across both the two x points of the m=2 magnetic islands, with a little asymmetry in low field side and high field side. The x point of the m=1 islands is overlapped with one of the O point of the m=2 magnetic islands. This result again demonstrates that the sawtooth crash is not caused by the fast reconnection of the m=1 magnetic island.

The observation above strongly suggests that the sawtooth crash is not caused by the fast reconnection of the m=1 magnetic island, while the picture is in agreement with the turbulence model[17]. For turbulence model, the large heat flow and hence the crash is the result of the fast energy and particle diffusion in the magnetic stochastic region around the separatrix of the magnetic islands.

4.3 Turbulence measurements

Electron mode and ion mode coexistence in high density ohmic plasma $n_e \geq 4.5 \times 10^{19} \text{ m}^{-3}$ has been observed by 2D ECEI. The local wavenumber-frequency spectrum by two-points correlation technique are shown in Fig 25. The two modes are found coexist at $r \sim 10 \text{ cm}$ in the low field side as well as at $r \sim 13 \text{ cm}$ in the high field side (HFS). Ion temperature gradient mode in density fluctuation was observed by far-infrared (FIR) scattering and beam emission spectrum (BES), but has not been reported in the temperature fluctuation yet. Further experiment and analysis are needed to clarify this observation.

![Fig.25a The wavenumber-frequency spectrum of temperature fluctuation at LFS](#93532,93533)

![Fig.25b The wavenumber-frequency spectrum of temperature fluctuation at HFS](#93572,93573)

5. Summary and near future plan

Since last IAEA meeting, 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. Presently, a 2MW lower hybrid current drive system at 2.45GHz and a 1.5MW ICRF system at 30-110MHz are in operation. A new 4.5MW ICRF system at 25-70MHz will be available in 2009. The present LHCD system is planned to be upgraded to 4MW. The total RF heating and current drive power will be 10MW in CW before end of 2010. This power is much higher than the H-mode threshold for standard EAST operation scenario at 1MA plasma current and toroidal field strength of 3.5T. 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. Some of them will be built via international collaboration. These diagnostics should be sufficient to describe the plasma performance and for integrated modeling and data analysis. 

HT-7 will be still operated before EAST is equipped with sufficient heating and current drive power and diagnostics. Its experiments will focus on issues related with the plasma and wall interaction under long pulse condition and some of specific topics, such as, MHD instabilities, transport and turbulence by Langmuir probes, newly developed CO2 laser scattering and 2D ECE image systems.

Acknowledgements

This work was supported by the National Natural Science Foundation of China under Grant No. 10725523 and No. 10721505 and partially by the Core-University Program of Japanese Society of Promote Sciences.

Reference

Appendix,
EAST and HT-7 teams:
Baonian Wan 1), Jiangang Li 1), Yuanxi Wan 1), Xianzhu Gong 1), Bingjia Xiao 1), Yuntao Song 1), Yu Wu 1), Jiafang Shan 1), Yanping Zhao 1), Liqun Hu 1), Liman Bao 1), Yanfang Bi 1), Lei Cao 1), Jiafeng Chang 1), Junling Cheng 1), Xu Deng 1), Bojiang Ding 1), Jiayi Ding 1), Siye Ding 1), Shijun Du 1), Yanmin Duan 1), Peng Fu 1), Jia Fu 1), Daming Gao 1), Ge Gao 1), Wei Gao 1), Xiang Gao 1), Zhe Gao 4), Shiyi Ge 1), Liansheng Huang 1), Zhensan Ji 1), Yinxian Jie 1), Guiming Li 1), Shiyao Lin 1), Bingjian Liu 1), Jinfeng Liu 1), Jialin Liu 1), Junling Cheng 1), Xiaoyang Sheng 1), Yuejiang Shi 1), Youwen Sun 1), Ang Ti 1), Xiaogang Wan 1), Huazhong Wang 1), Linsen Wang 1), Mao Wang 1), Keping Wu 1), Hao Wu 1), Jiefeng Wu 1), Junshuan Wu 1), Songtao Wu 1), Weiyue Wu 1), Yichun Wu 1), Zhenwei Wu 1), Weibing Xi 1), Chunyi Xie 1), Guosheng Xu 1), Liuwei Xu 1), Ping Xu 1), Tiejun Xu 1), Xiaoyuan Xu 5), Diye Xue 1), Lei Yang 1), Li Yang 1), Qingwei Yang 3), Zhongshi Yang 1), Chanxuan Yu 5), Chunyan Yuan 1), Qiping Yuan 1), Junyu Yuan 1), Qing Zhan 1), Qiuyong Zhang 1), Wei Zhang 1), Xiaodong Zhang 1), Xinjun Zhang 1), Guanhua Zhen 1), Fali Zhong 1), Guoqiang Zhong 1), Yanfei Zhu 1), Youhua Zhu 1), Ping Zhu 1), Ming Zhuang 1)

International collaborators:
Paul Anderson 6), Vincent Chan 6), Dave Humphreys 6), Al Hyatt 6), Lang L. Lao 6), Jim Leuer 6), Ali Mahdavi 6), Gary Jackson 6), Ben Penaflor 6), David Piglowski 6), Michael Schaff 6), Mike Walker 6), He Huang 7), Kenneth Gentle 7), Perry Philippe 7), William Rowan 7), D. Mueller 8), M. Bitter 8), K. W. Hill 8), D. M. Mastrovito 8), Houyang Guo 9), R. Kumazawa 10), T. Mutoh 10), K. Saito 10), T. Watari 10), B. Bremond 11), K. Vueillie 11), S. G. Lee 12), J. G. Bak 12)

1). Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China
2). Donghua University, Shanghai, China
3). South West Institute of Physics, Chengdu China
4). Tsinghua University, Beijing, China
5). University of Science and Technology of China, Hefei, China
6). General Atomic, San Diego, USA
7). Fusion Research Center, UT at Austin, USA
8). Princeton Plasma Physics Laboratory
9). Washington University, Washington, USA
10). National Institute for Fusion Sciences, Toki, Japan
11). Association Euratom-CEA, CEA Cadarache, France
12). Natiaon Fusion Research Institute, Korea Basic Science Institute, Daejeon, Korea