EAST and Magnetic Confinement Fusion Programs in China

Jiangang LI for EAST team and international collaborators

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7) Fusion Research Center, UT at Austin, USA
8) Princeton Plasma Physics Laboratory, USA
9) Tri Alpha Energy, California, USA
10) National Institute for Fusion Sciences, Toki, Japan
11) Association Euratom-CEA, CEA Cadarache, France
12) Natiaon Fusion Research Institute, Daejeon, Korea
13) ORNL, USA

Outline

Technical improvements

- C PFCs with active water-cooling
- Basic Diagnostics
- H&CD systems
- Wall Conditioning

Experiments

- Physical engineering capability
- Plasma initiation and Ramp-up
- Plasma control
- Divertor experiments
- LHCD experiments
- Long pulse discharges

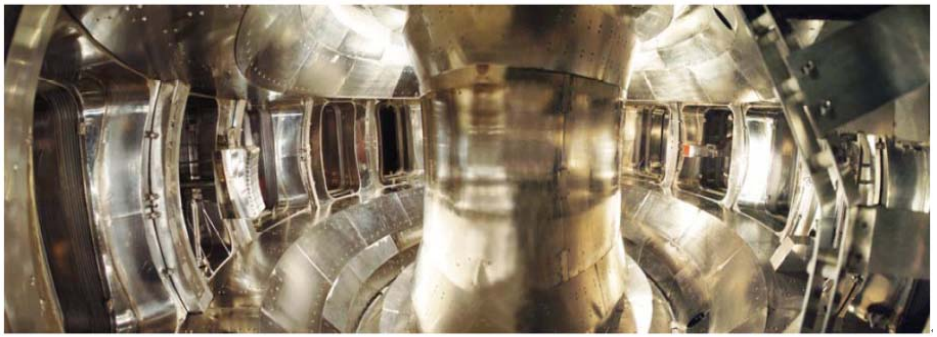
Research Plan in next 2-5 years

Future Plan of CN-MCF program

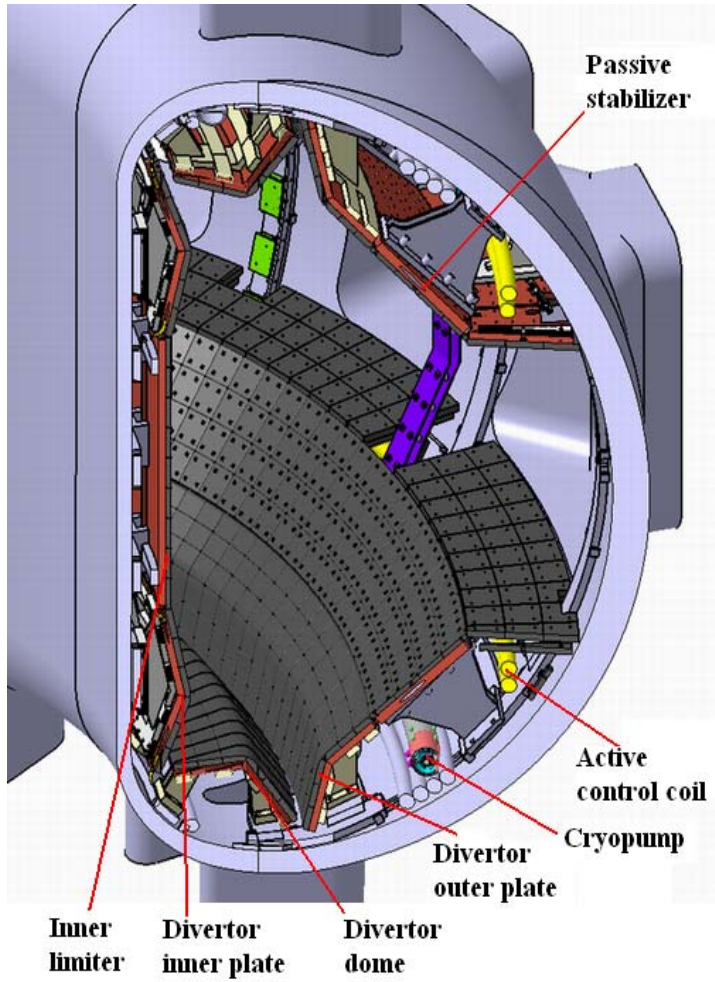
Summary

Actively Water-Cooled PFCs

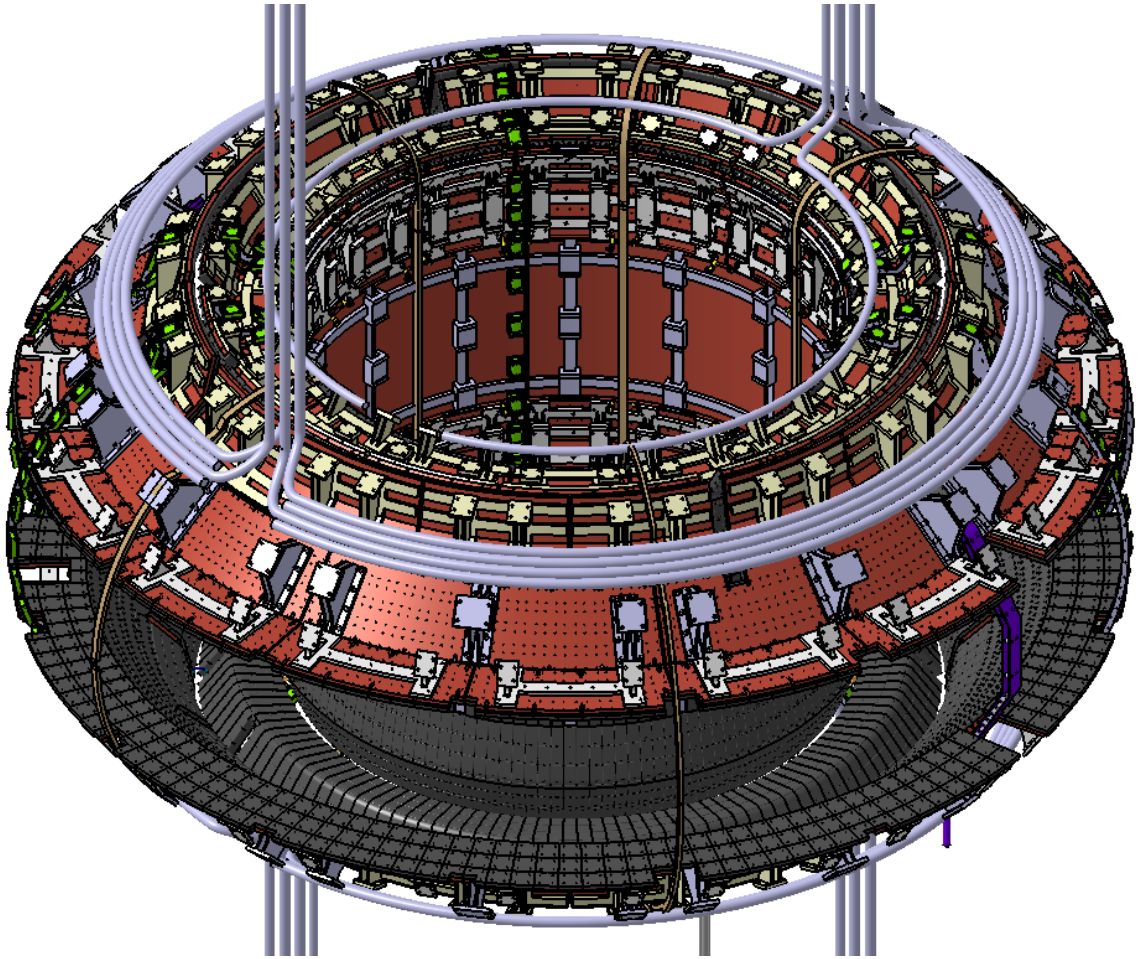
1st and 2nd Campaign with SS PFCs



Since the 3rd Campaign changed to full graphite's



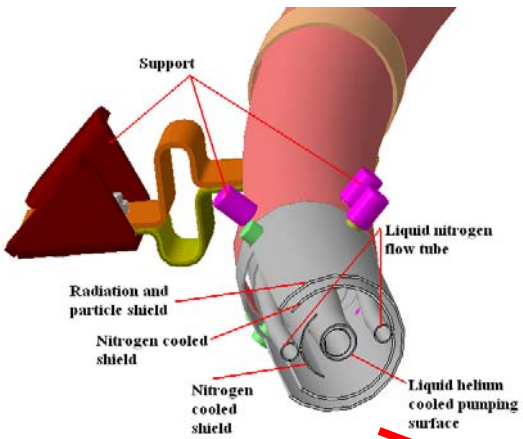
Integration of PFC



- Magnetics
- Cryo-pump
- Thermal couple
- Water cooling
- Anodes of DC GD
- Internal coils
- RF antenna
- Poloidal limiters
- Divertor probes
- Support structures
- Heat sink
- Graphite titles

Key elements in-vessel

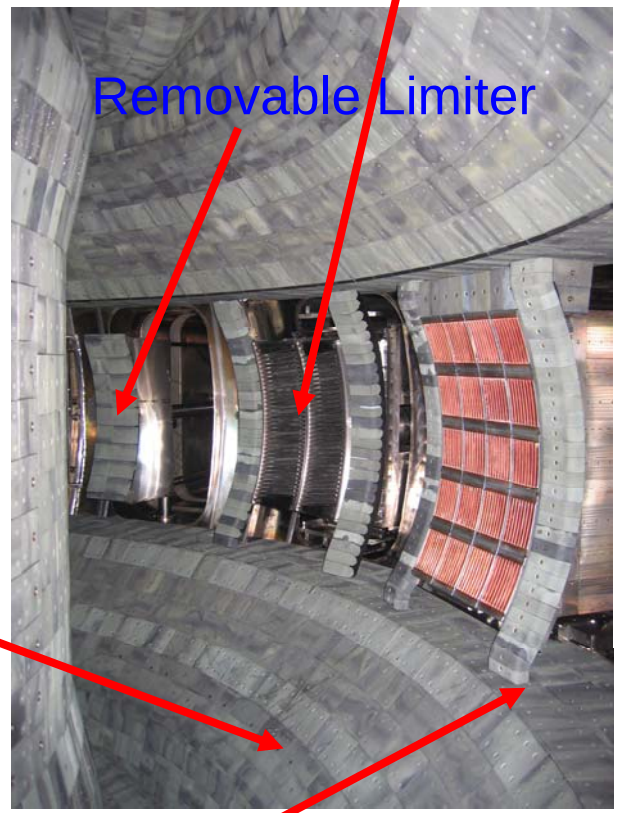
Internal Cryo-Pump



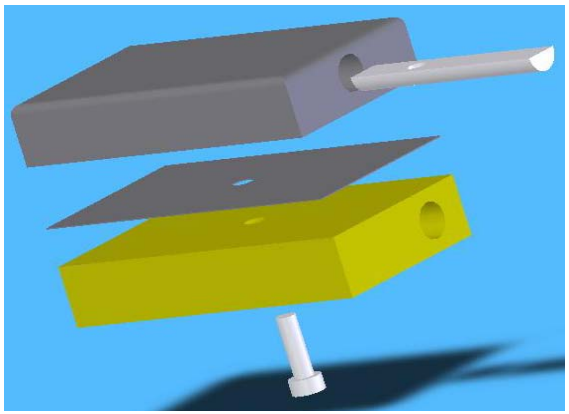
75,600L/S for D2
107,000L/S for H2

- Actively-cooled PFC (~9000 tiles)
- Internal Cryo-Pump
- **LHCD: 2.45GHz, 2MW**
- **ICRF: 30-110MHz,1.5MW**
- Magnetic sensors
- 2 Removable limiter

ICRH antenna



LHCD antenna

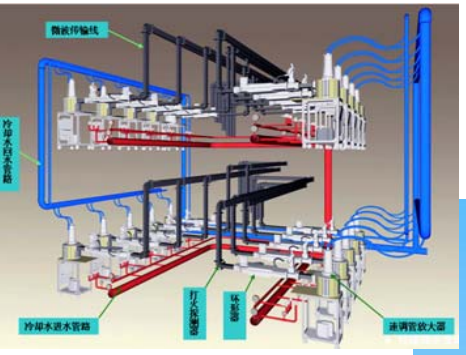


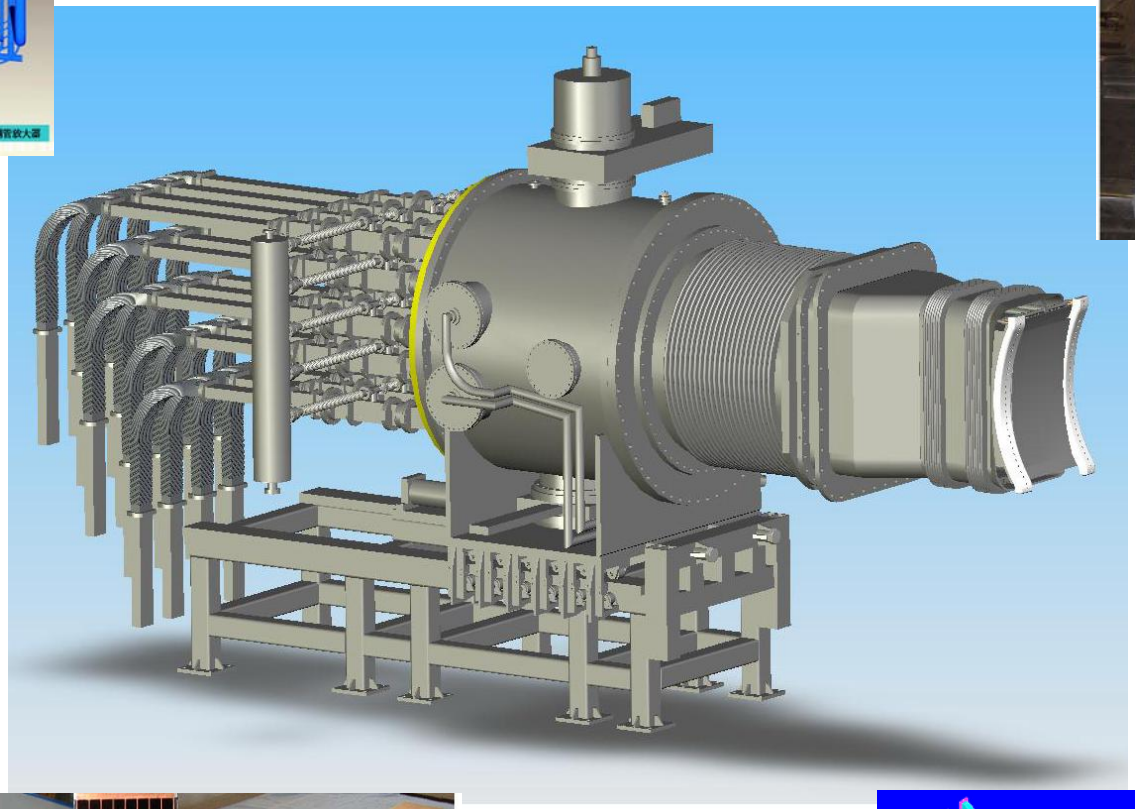
High heat flux region 2MW/m^2

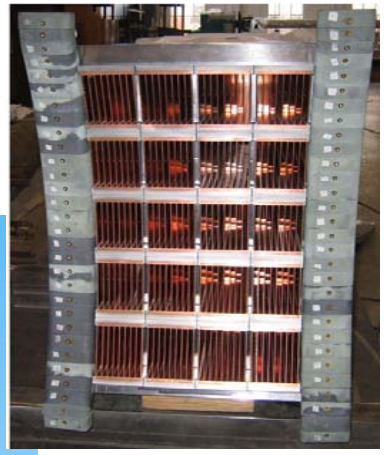
Total 37 flux loop

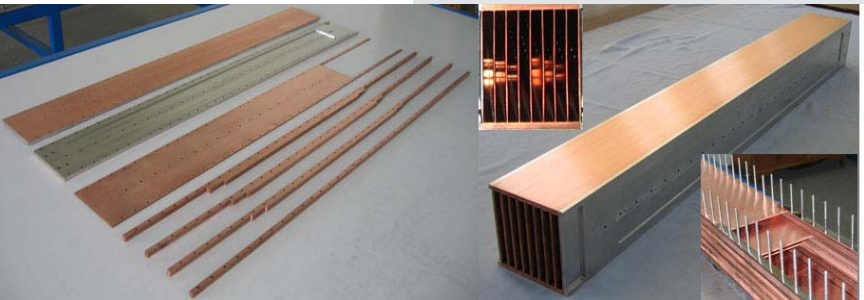
CW LHCD system

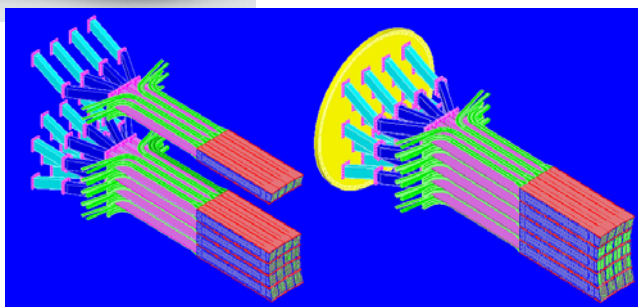
2MW，2.45GHz, n=1.8-2.4





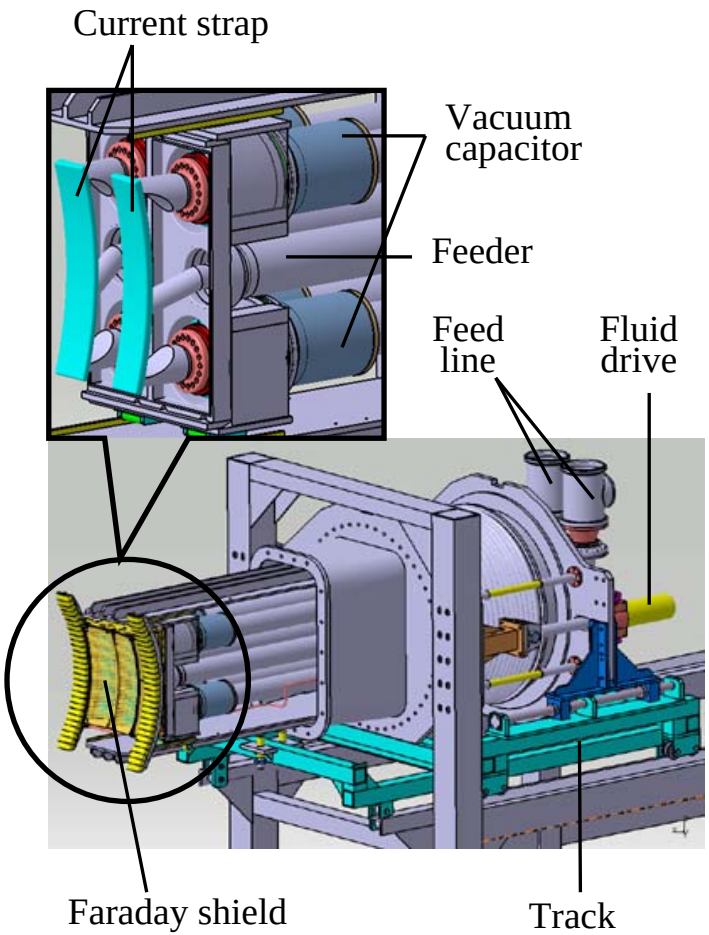






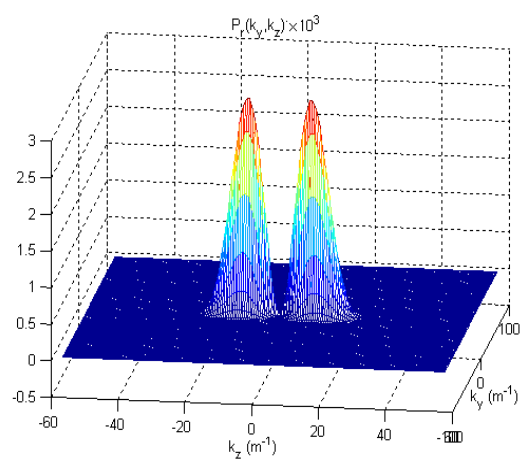
ICRH system

RDL antenna

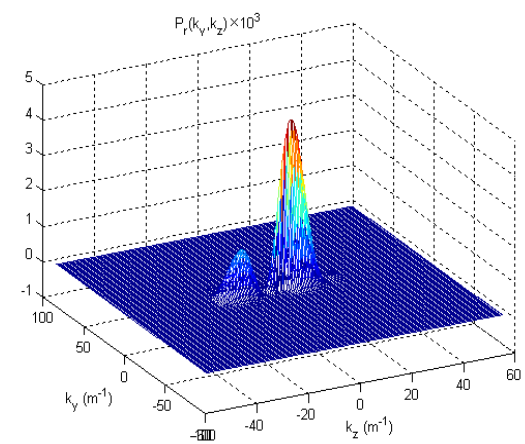


In collaboration with CEA, TS

- Maximum output power: 2.5MW.
- Maximum pulse duration: CW (1.5MW)
- Frequency range: 25MHz-100MHz.
- Output impedance: 50 Ω .
- Bandwidth with 50 Ω load: 2MHz (-3dB).
- Fast shut off time: ⩽10 μ s.
- Phasing capability.



0,pi phasing



0,pi/2 phasing

EAST Diagnostics

EAST

ASIPP

- **Diagnostics in initial two campaigns, total 20**

  ----Machine protection and plasma control, description of the basic plasma performance with SS wall, H

  5 new constructed, 15 moved from HT-7 with minor modification

- **Diagnostics in 3rd campaign, total 25**

  ----Same function above with graphite tile wall, D

  8 new constructed, 10 modified for DN plasma configuration

- **Diagnostics from 4th campaign, total 30**

  ----Evaluation and optimization of the plasma performance

  ----Focusing on key profile parameters

Measurements for machine operation, protection and plasma control

EAST

Function	Measurements	Purpose
Machine operation and control		
Plasma initiation, shaping	Magnetic diagnostics	Ip, Vp, real-time magnetic surface recons. , position and shape feedback control
	FIR DCN laser interferomet.	Plasma line-averaged density
	Vacuum gauge	Vacuum pressure
Operation optimization	Tangential visible CCD Thermocouples	Plasma image, in-vessel comp. insp. Wall temperature uniformity
Machine protection-------		
Disruption avoidance	Magnetic diagnostics	Forces on in-vessel components MHD modes, Locked mode
	SXR array	Sawtooth and core MHD
Damage avoidance	AXUV	Radiation power profile
	Hard X-ray emission	FEB profile
	Tangential visible CCD	Plasma image, in-vessel comp. inspect.
	IR camera	Heat spot
	Bolometer	Total radiation power and profile
	DNB+CXRS, XCS	Plasma rotation
Run away avoidance	NaI/BGO detector	Run away electrons
Operation safety	3He, fission chamber	Neutron flux monitor
Impurity species and control	OMA, UV-VIS, VB	Impurity line radiation, spectrum, Zeff
Divertor optimization	Divertor probe, …	Key Divertor parameters (density, temperature, heat flux)

Measurements for plasma performance evaluation and optimization

EAST

Function	Measurements	Purpose
Te profile	TS, ECE radiometer, XCS (X-ray crystal system), SX spectrum PHA	Auxillary heating effects and optimization profile control, transport
line-averaged density	FIR DCN laser interferometer	Fuelling optimization, transport
Ti profile	2-D XCS, DNB+CXRS	Auxillary heating effects and optimization
Radiation profile	SXR array, AXUV Tangential hard X-ray Ha/Da array VB array Bolometer	Core plasma, radiation power profile Fast events understanding & control FEB emission, LHW deposition & coupling Recycling, edge plasma, Divertor behaviors Zeff (line-averaged) profile Radiation loss mechanism
Plasma-wall interact.	Tang. visible CCD, IR camera	Deposition of heat load, filament, ripple…
Impurity line radiation, spectrum	OMA, UV-VIS, Fibre spectrometer SXS spectrum PHA	Impurity species and density, dilution heavy impurity concetration
Runaway behaviors	NaI + BGO detector	Run away electron dynamics
MHD instability	Magnetic diagnostics (Mirnov coil) SXR array	Plasma performance, Beta, li, W, Sawtooth and core MHD instability
Divertor param. (Ne/Te profile, Impurity & influx)	Divertor probe, Mach probe, thermocouple, Ha/Da, bolometer…	Divertor optimization, plasma flow
Neutron flux, spectrum	3He, BC501, fission chamber	Spectrum and emission profile
Vac. P of vessel diver.	Vacuum gauge	Oper. scenario optimiz., cryo-pump physics
Plasma rotation	DNB+CXRS, XCS	Performance improvement, ML avoidance
q(r) profile	SXR + EFIT reconstruction	

PSI RF Conditionning

1. ICR conditioning were successfully carried out in EAST, a divertor SC tokamak with metal/C walls.
2. ICR cleaning, recycling control, boronization and oxidation have been carried out and compared with GDC.
3. High pressure and RF power are favorable for removal of hydrogen and impurities.
4. Wider operation widows (EAST: 5-30kW, 10^{-4}-10Pa HT-7: 10-60kW, 10^{-4}-10^{-2}Pa) and higher removing rate were obtained.
5. RF-Boronization has been routinely used for all campaions with about 200nm thinkness. 30-60 min. He RF conditioning was used for control recycling. Very good plasma performance can been easily obtained.
6. RF-Li coating has been carried out. Higher plasma parameters and lower recycling were obtained.

He/O-ICR in EAST

EAST

1. He/O-ICR were successfully carried out in EAST, a divertor tokamak with metal walls. O-GDC would lead arcing, and it is difficult to be controlled.
2. High pressure and conditioning power in He/O-ICR are favorable for removal of hydrogen and impurities.
3. O-RF on a SS walls are beneficial for both H and C removal. Highest removal rate are 7.8×10^{22}H-atoms/h,4.2×10^{22}C-atoms/h (20kW 7×10^{-2}Pa), which were higher than that in He-ICR by a factore of 5 and a few tens respectively.
4. During oxidation, C was removed by the formation of CO and CO_2 and most of hydrogen released in the form of water molecules.
5. Both He-ICR or He-GDC are effective way to remove oxygen.
6. Both O-RF and O-GDC could lead contamination on various materials (W\Pure Graphite\Deposits\SS\Silicon).

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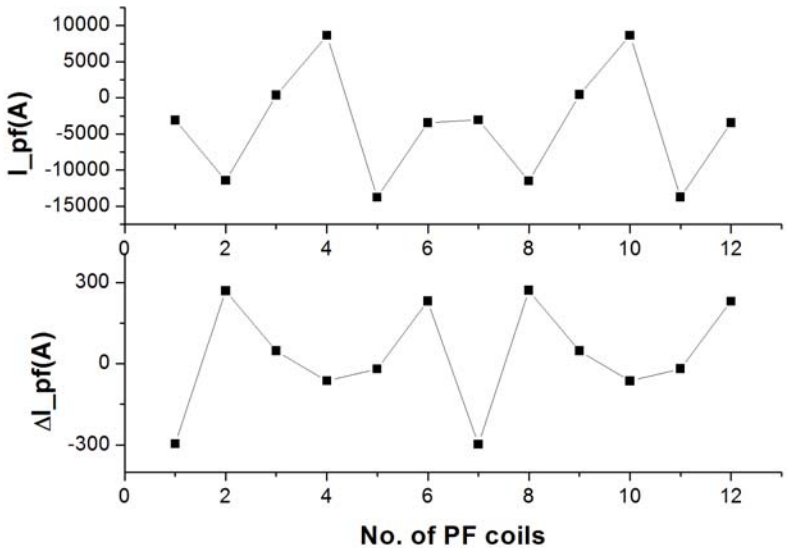
Summary

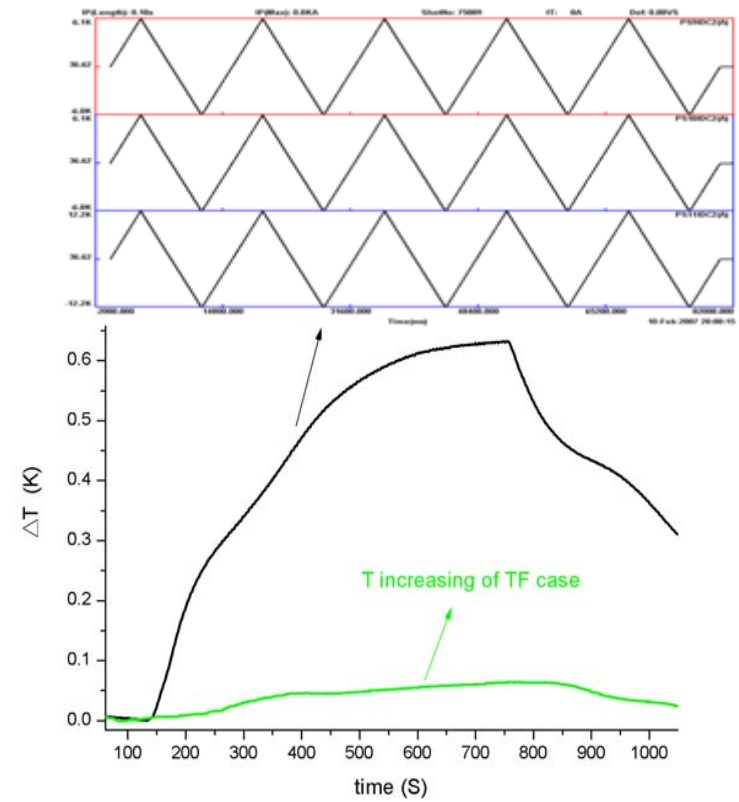
Physical engineering capability

Evaluation of superconducting magnets and related systems for steady-state plasma discharges. Key issue→AC loss



Any event causes betaP 1.24→1.10, if the LCFS shape was kept same. For a period variation of betaP (ST, ELM…), PF coils will experience the continuous current variation.

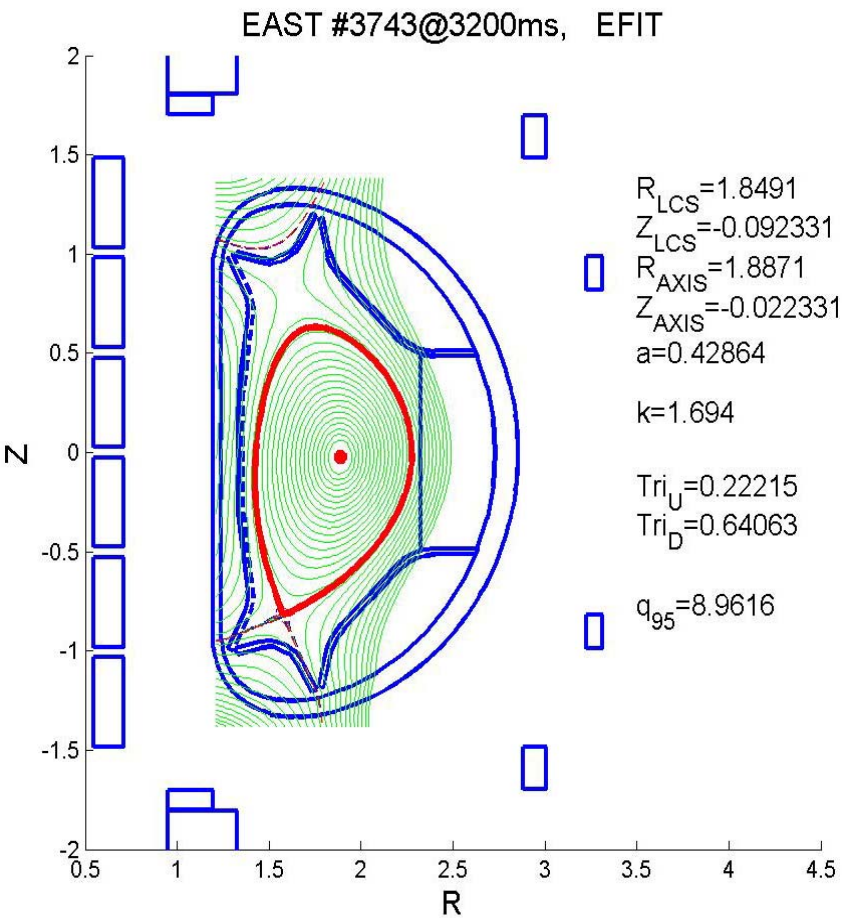
$dI_{pf}/dt|_{max}$ ~3kA/s for period of 0.1s

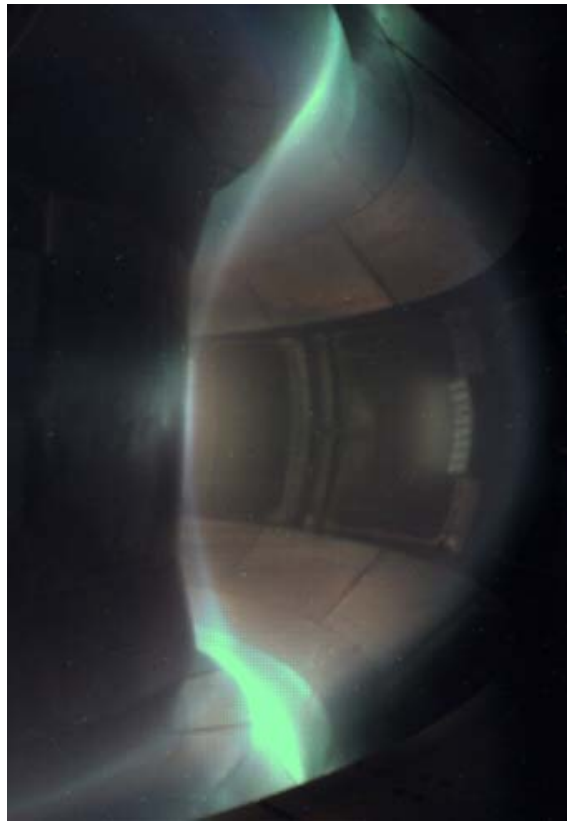


Outlet temperature rise of CS at excitation rate of 1kA/s

Contol of divertor plasma

DN, SN, Ip = 0.5 MA, Bt = 2-3.5 T, t ~ 10s, k=1.6-1.96,D=0.3-0.6

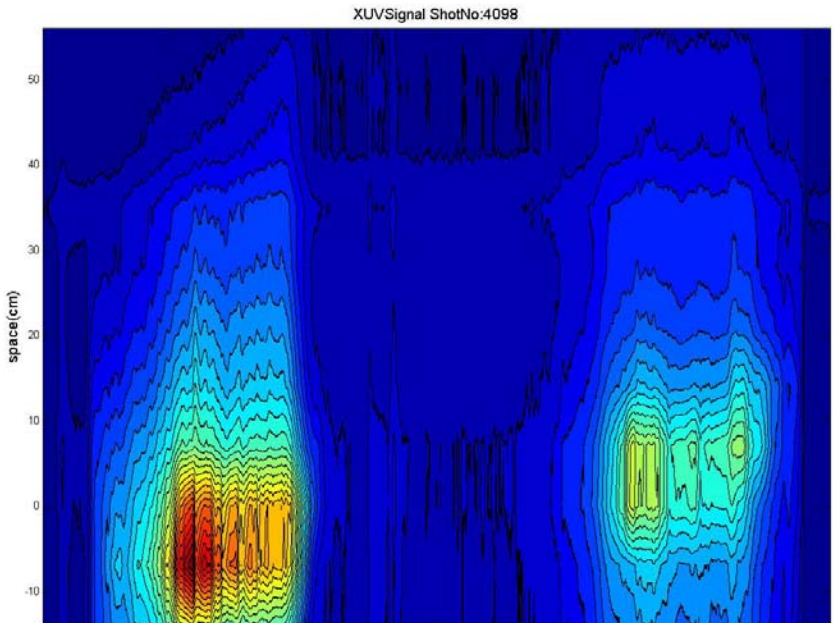


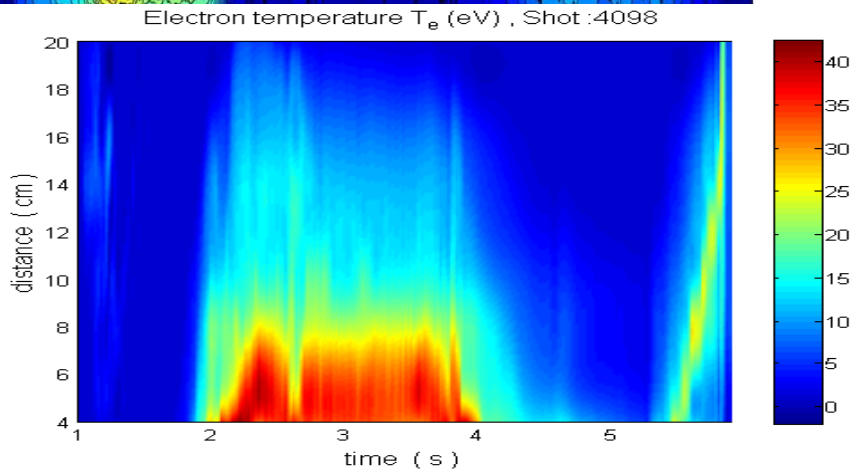


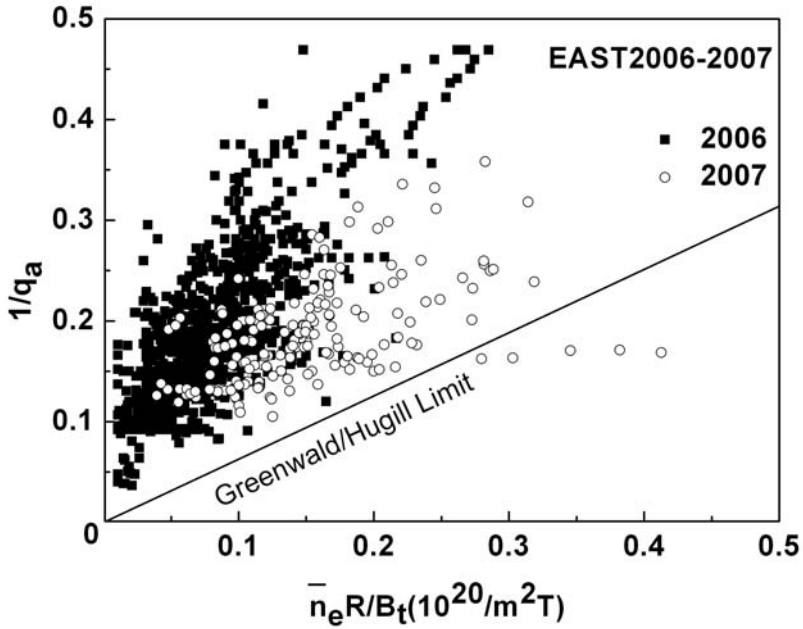
k = 1.8,D=0.5

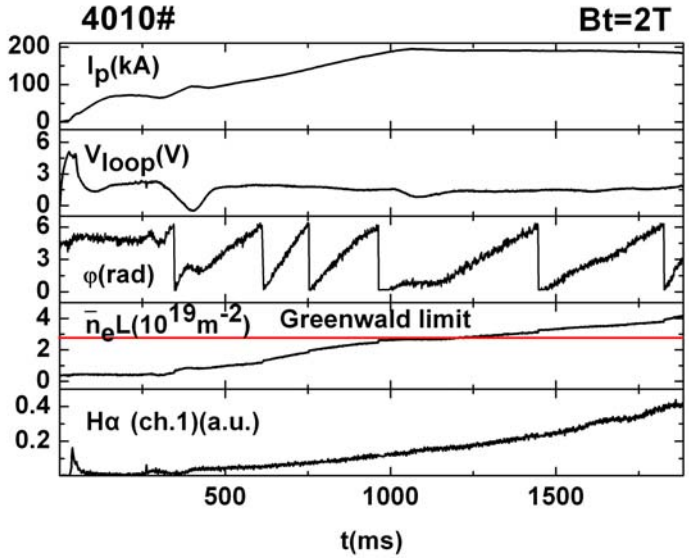
k = 1.96,D=0.3

High density plasma ~1.4Ne_{GW}





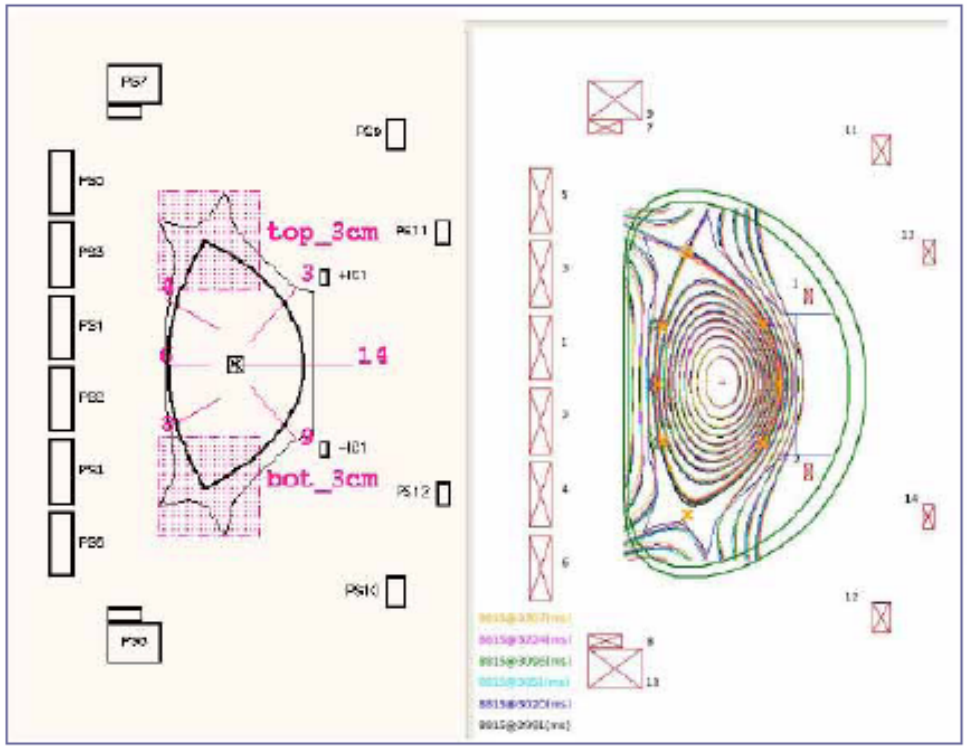


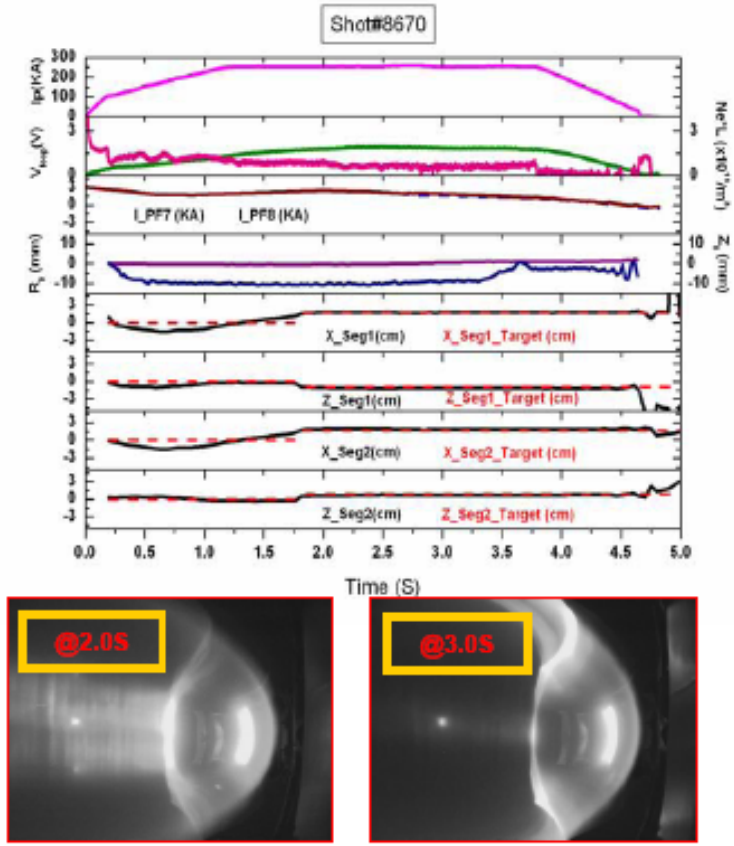


Impurity screen by diverter

Realization of Iso-flux control

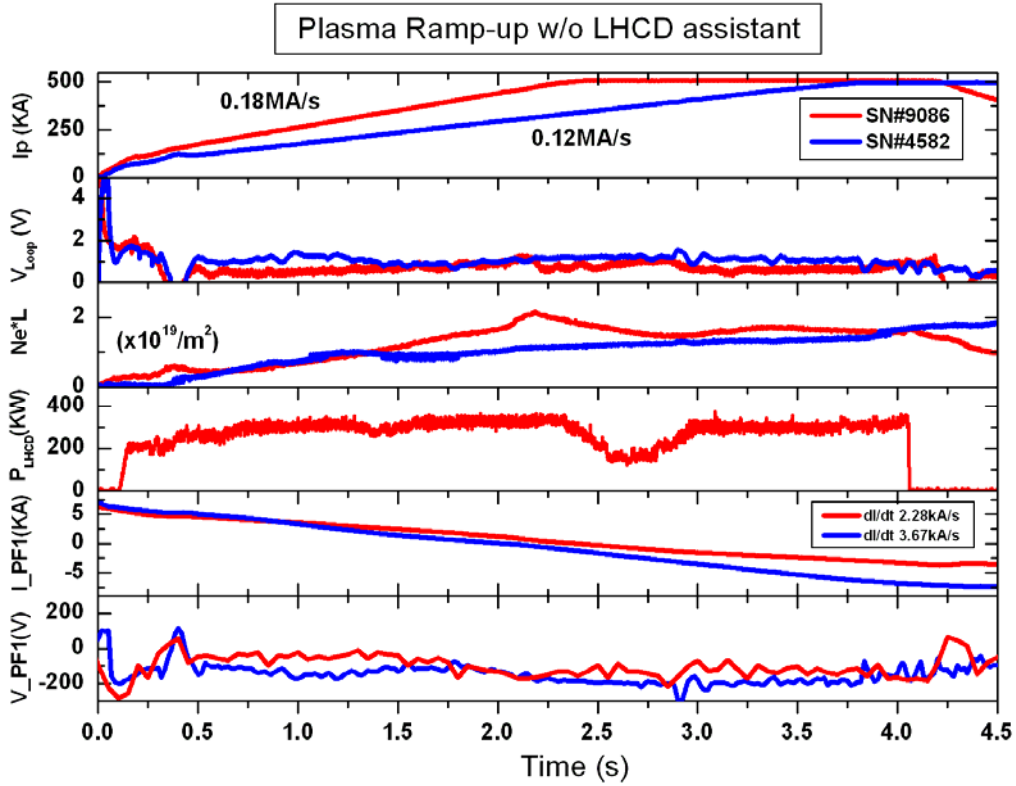
In collaboration with GA





Primarily realization of RTEFIT/ISO-flux control algorithm produced well controlled plasma shapes and provide basis for further experiments

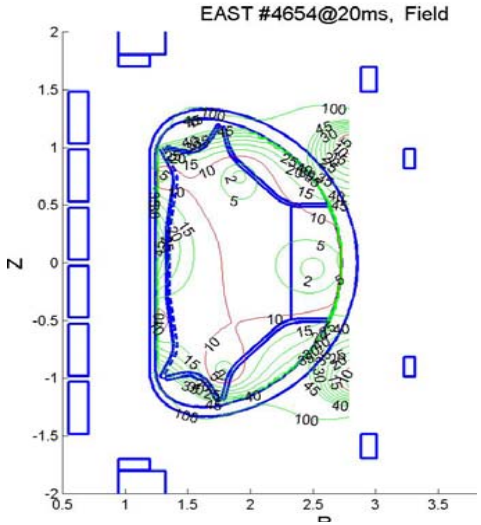
Plasma Ramp-up w/o LHCD

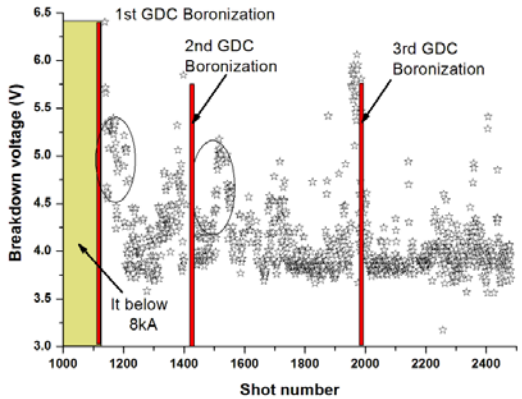


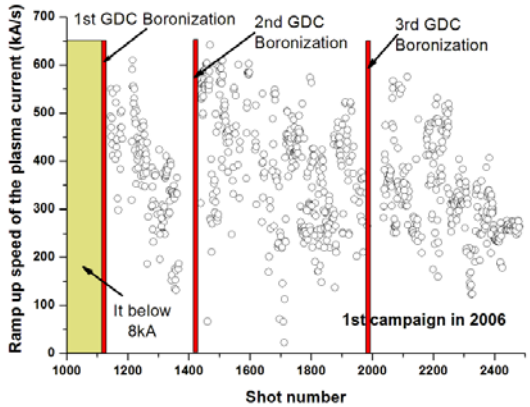
- Pre-program controlled
- RZIP feed back
- LHW assisted

LHCD applied at plasma start up phase can significantly reduce the current ramping rate in PF coils or voltage applied at PF coils, which increases the safety margin of SC magnets and provide larger margin for plasma control

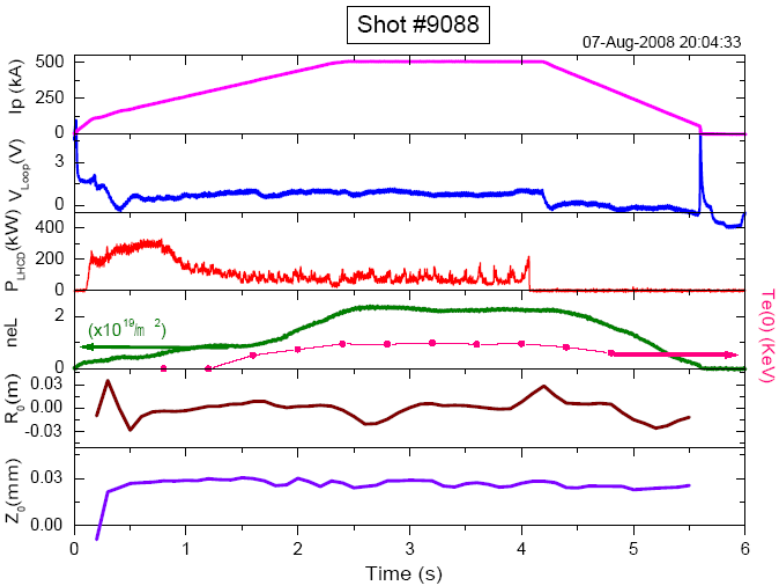
Optimization of Start up and Termination

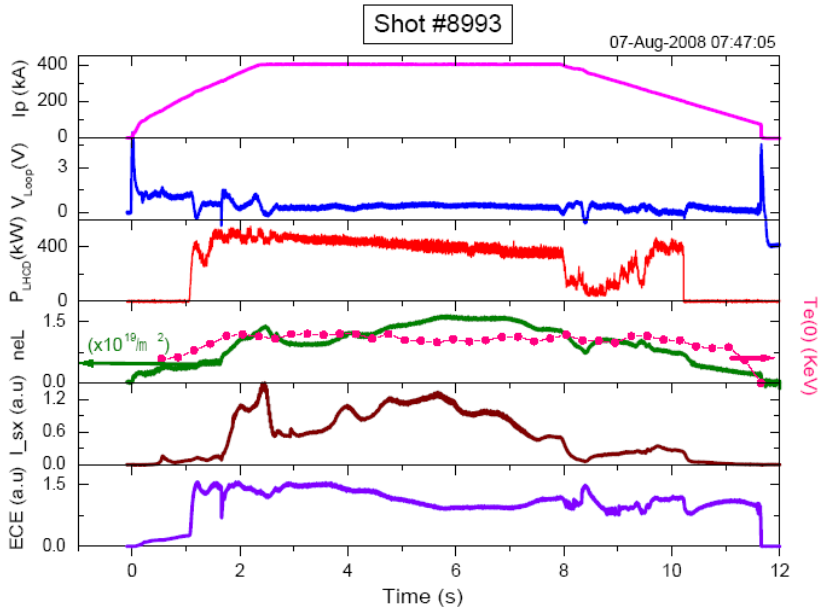






3.5 V < V_{loop} < 6.5 V **0.1 MA < dIp/dt < 0.65 MA/s**



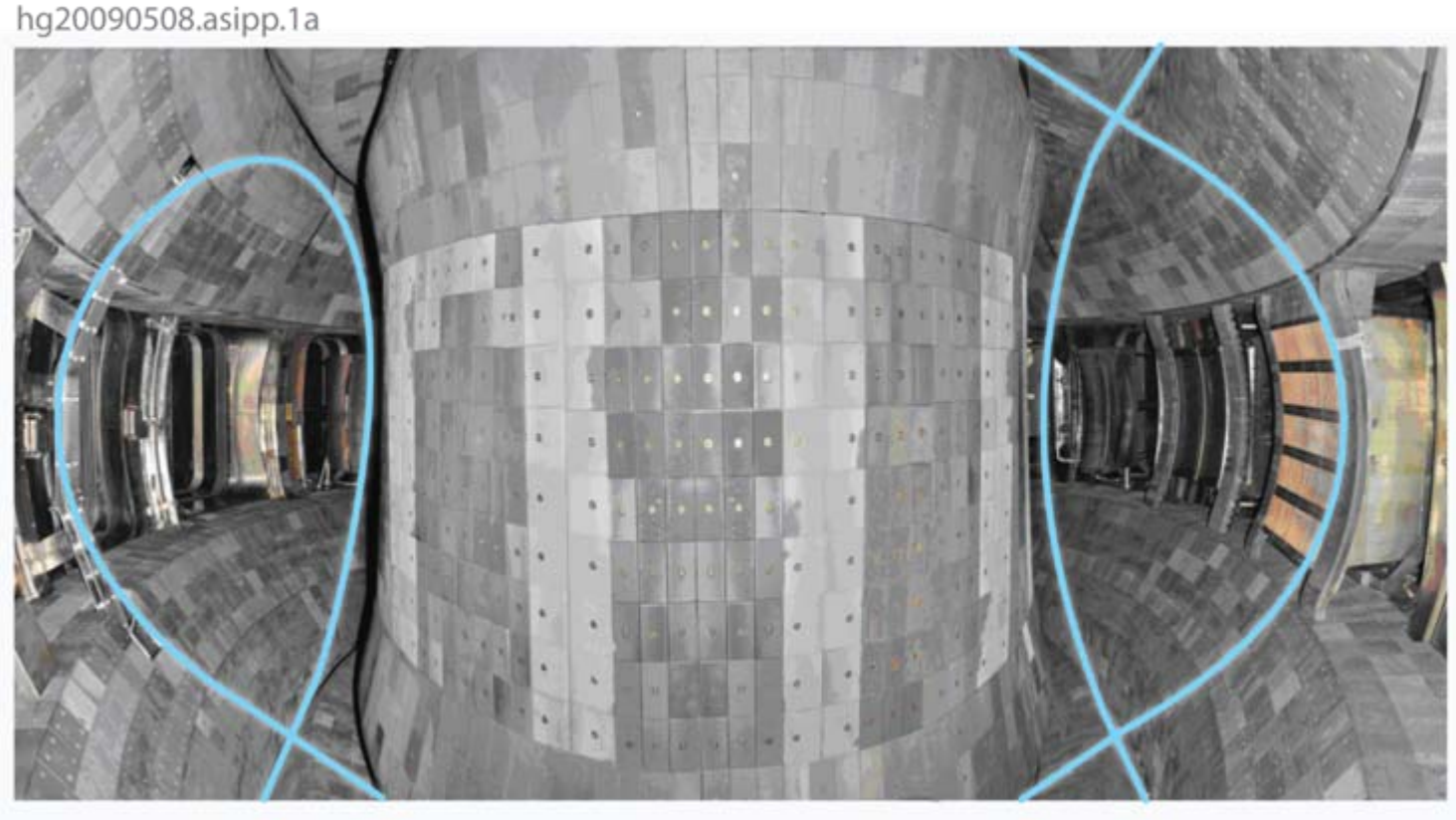


Shaping for minimum heat load on PF coils

Divertor Physics Experiments

- **Assessment of basic divertor plasma behavior**
- **Effect of divertor configurations** ***– Comparison between single null and double null***
- **Divertor asymmetry and drift effects** ***– Comparison between normal and revered toroidal fields***
- **Effect of gas puff locations on divertor asymmetry and fuelling efficiency**
- **Divertor screening for intrinsic carbon by CH_4 puffing**
- **Active control of divertor heat flux by Ar puffing**
- **Effect of divertor cryopump**

EAST Divertor Configurations



EAST has flexible poloidal field control system, allowing both single null and double null divertor configuraions.

Divertor Plasma Detachment Was Clearly Demonstrated on EAST

Sheath-Limited

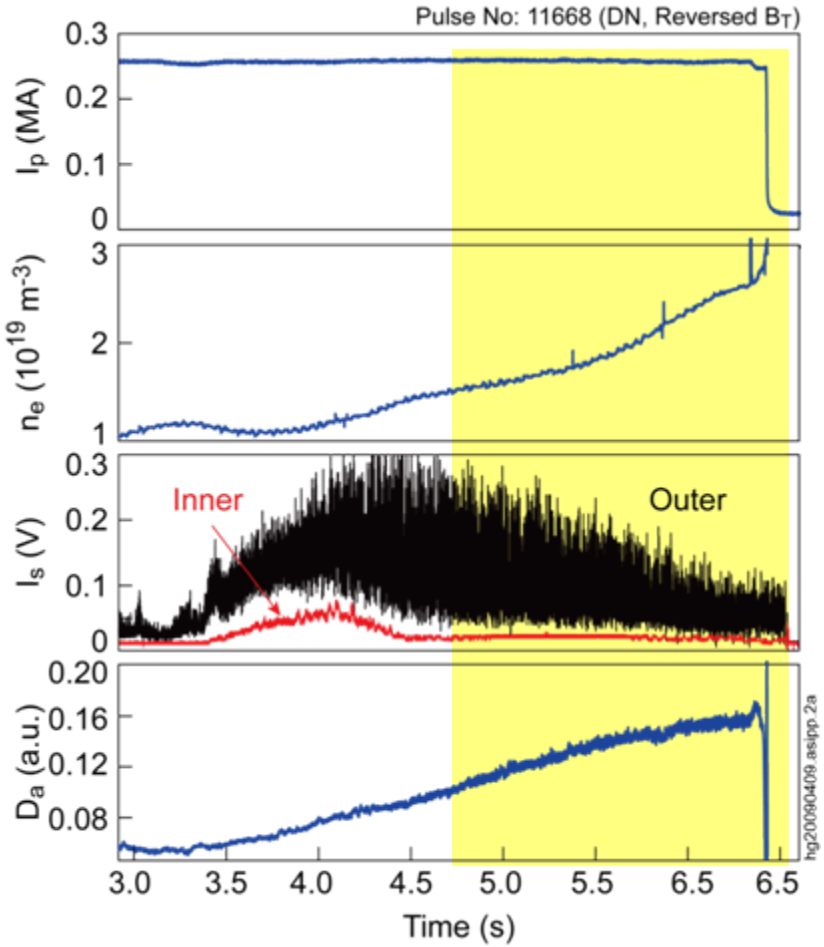
- Ion saturation current I_s (particle flux) increases with density n_e

Conduction-Limited

- I_s further increase until roll over

Detachment

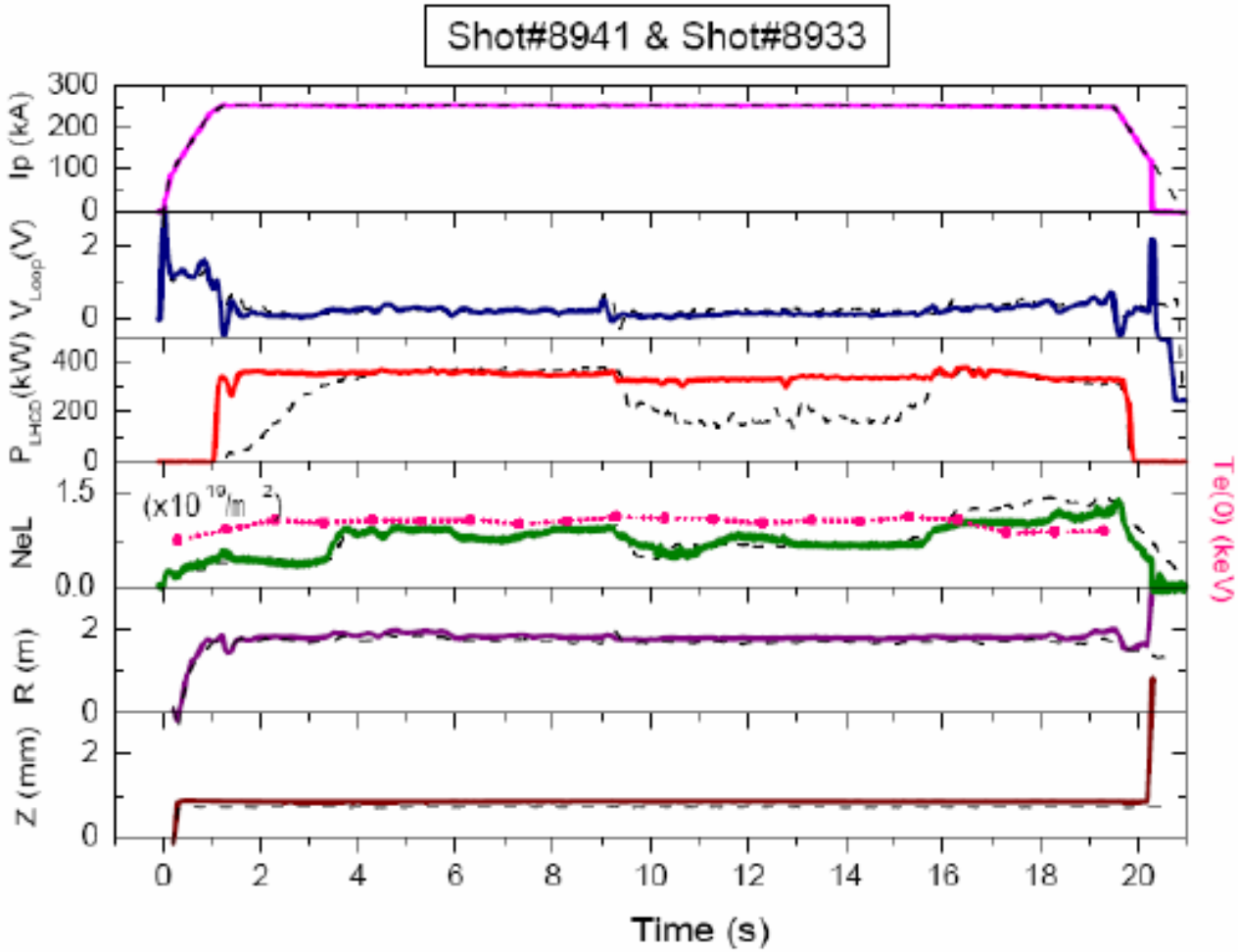
- Particle flux starts to decrease as n_e increases



Plasma detachment reduces peak particle & heat fluxes, as well as associated material damage, essential for steady-state operations.

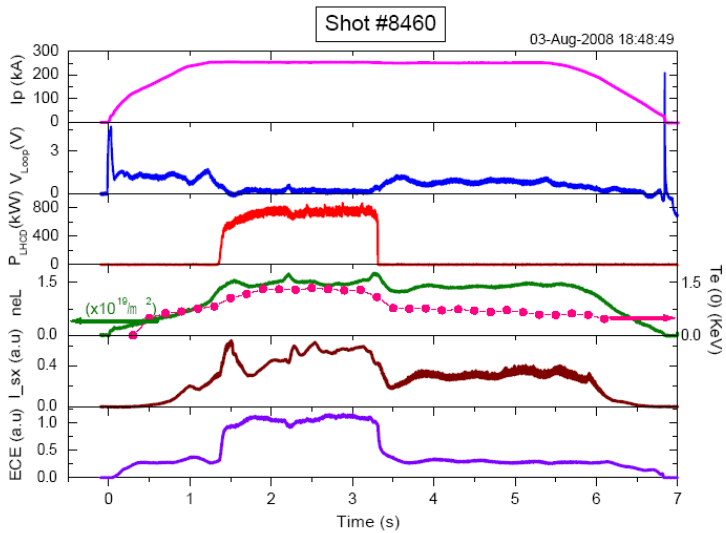
Long Pulse Discharges

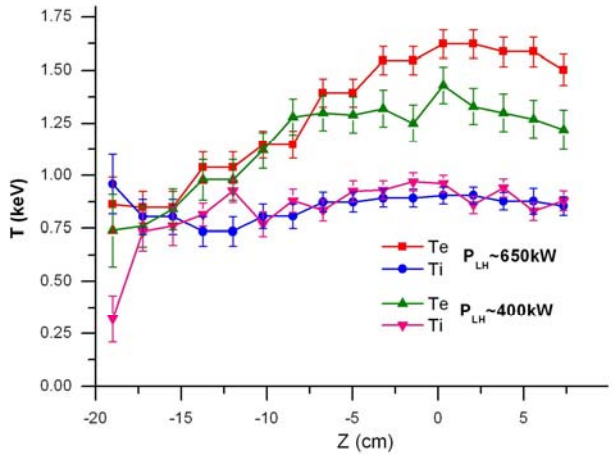
ASIPP

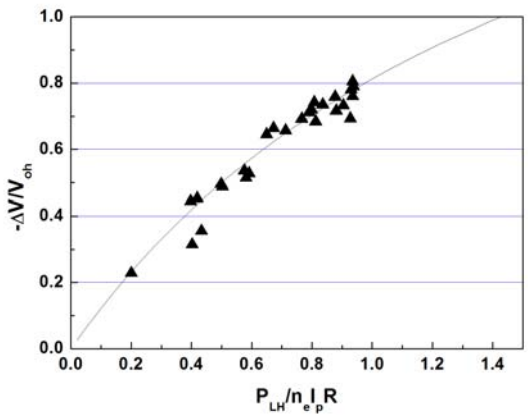


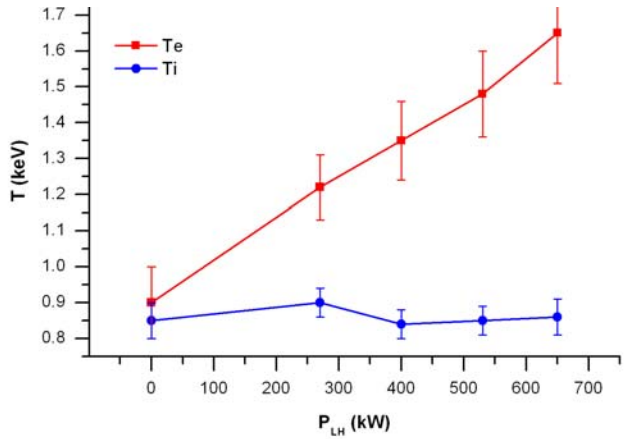
Repeatable long pulse discharges up to 23s has been achieved in elongated plasma. Main limitation…

First LHCD experiments



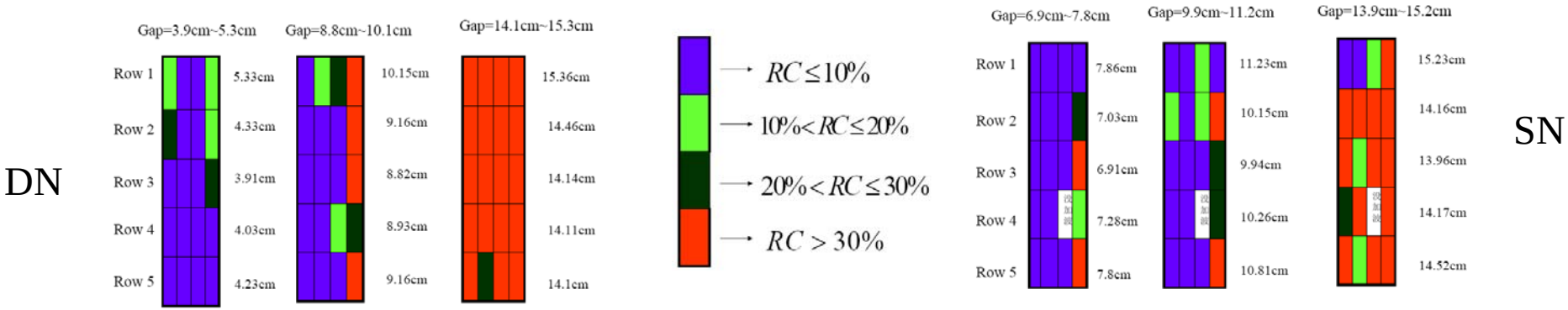


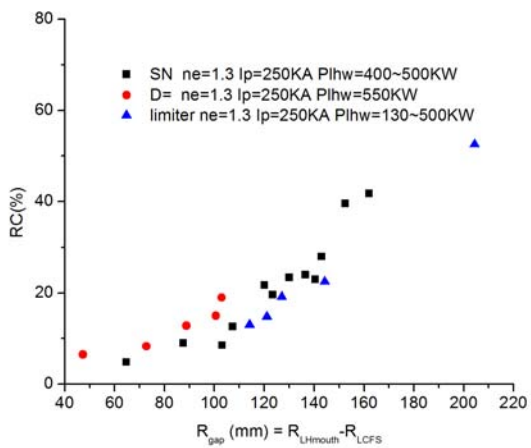


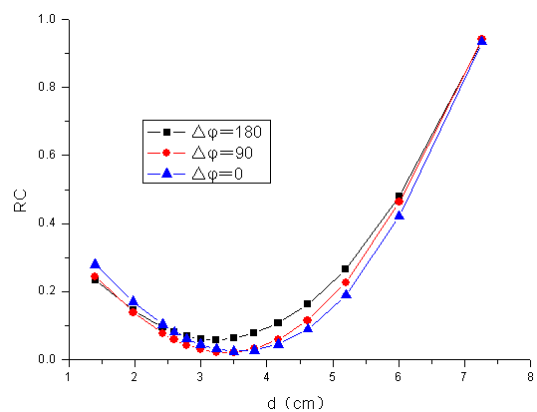


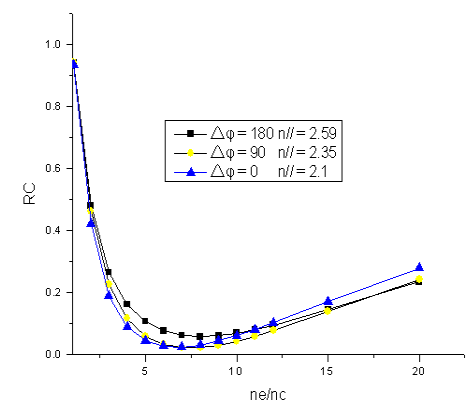
LHCD up to 800kW was injected into plasma at Ip=250kA in 2008.
The current driving efficiency 0.8*1019 Am-2W-1 at Ip=250kA
Significant electron heating has been observed.

LH wave coupling





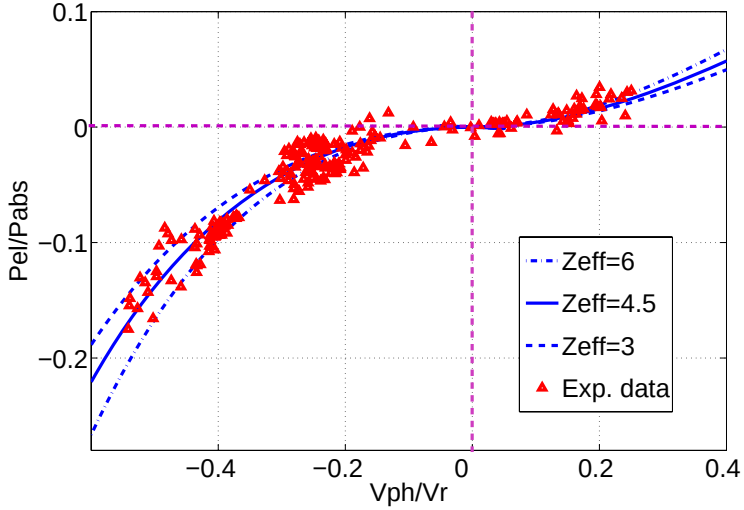


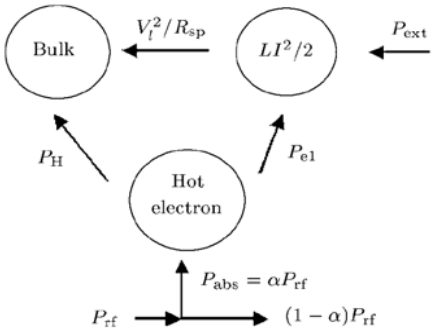


LHW coupling optimization by regulation of gap between the launcher mouth and the last closed flux surface and plasma shape according to plasma density

otal RC: 5~7% has been achieved for routinely LHCD experimen

LHCD efficiency

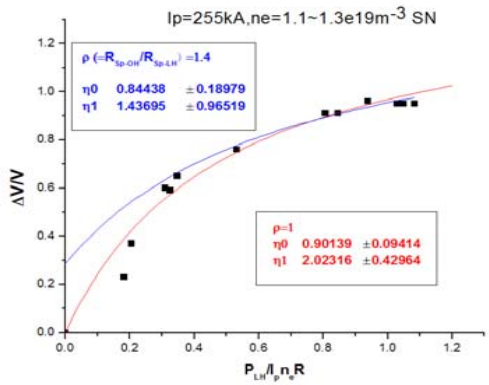


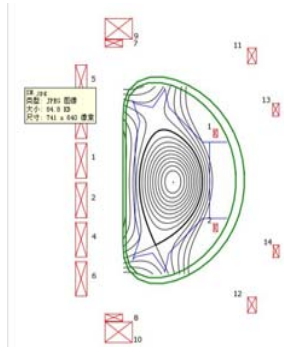


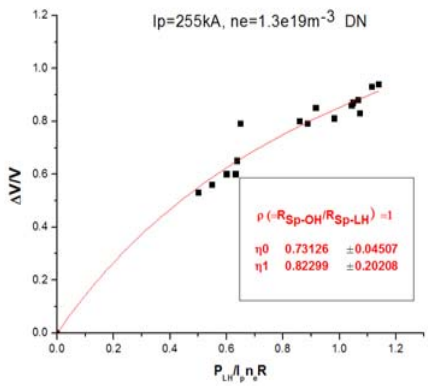
Fitting parameters:

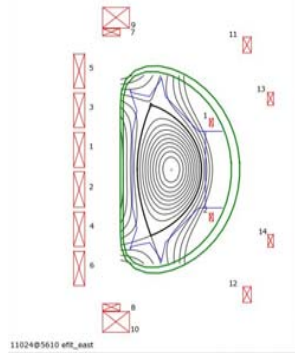
$P_{abs}/P_{inj}=0.78$

$N_{||abs}/N_{||launch}=1.6$





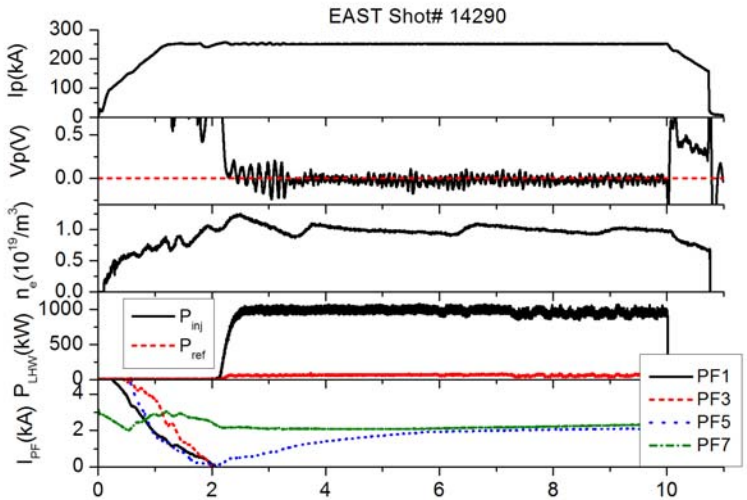


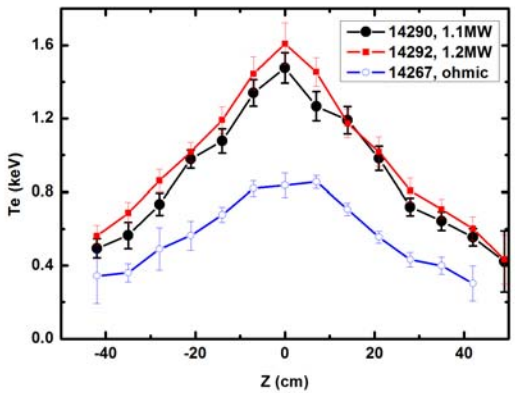


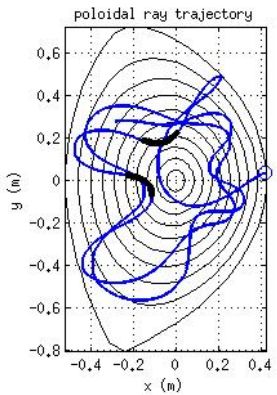
LHCD efficiency derives an injected power of 2MW for 0.5MA plasma current

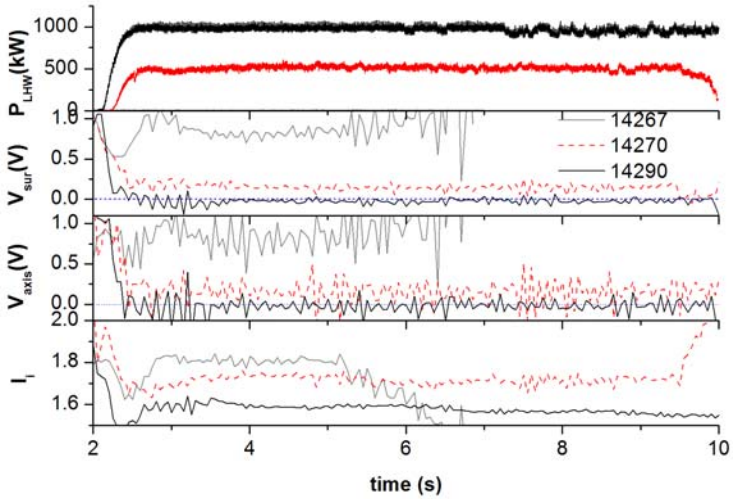
Full LHCD experiments

- Fully non-inductive discharge has been achieved both in DN and SN configuration.
- Higher injected LHW power results lower li, meaning off-axis CD.
- Power deposition peaked at rho~0.5
- The electron temperature profile was not significantly changed



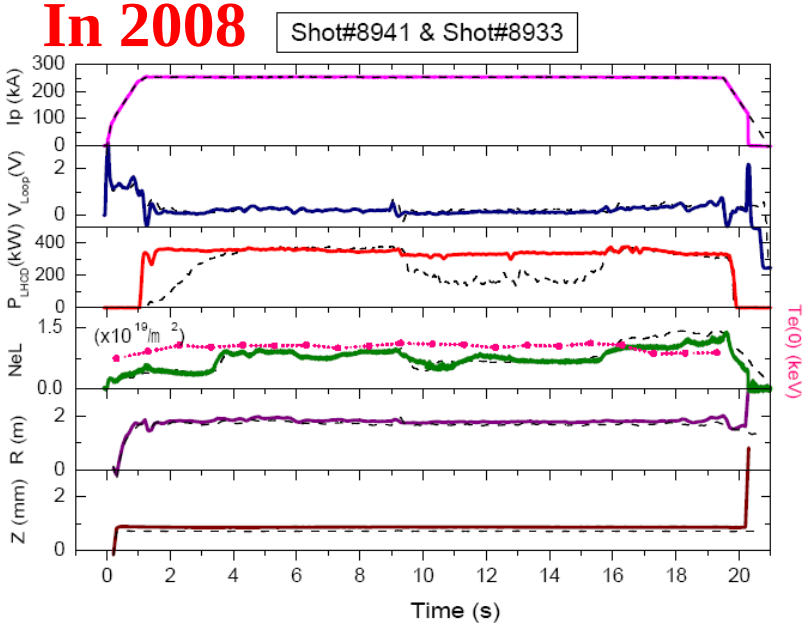




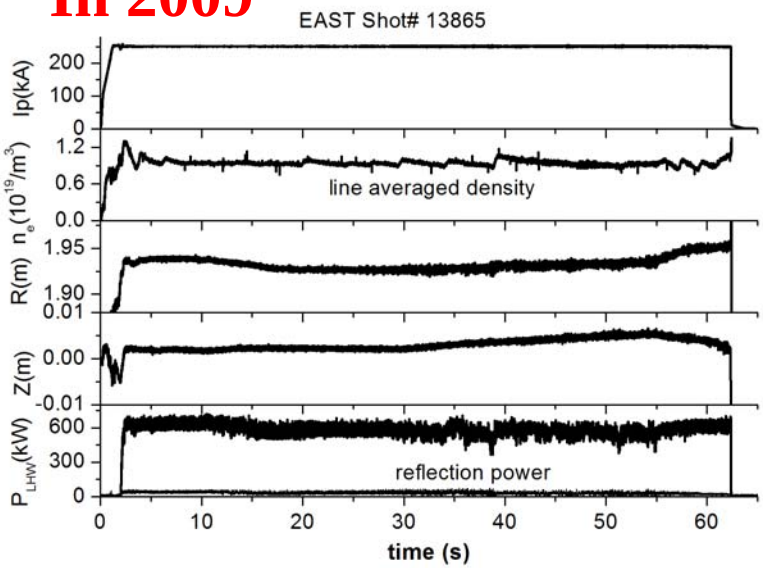


Long Pulse Discharges

In 2008



In 2009



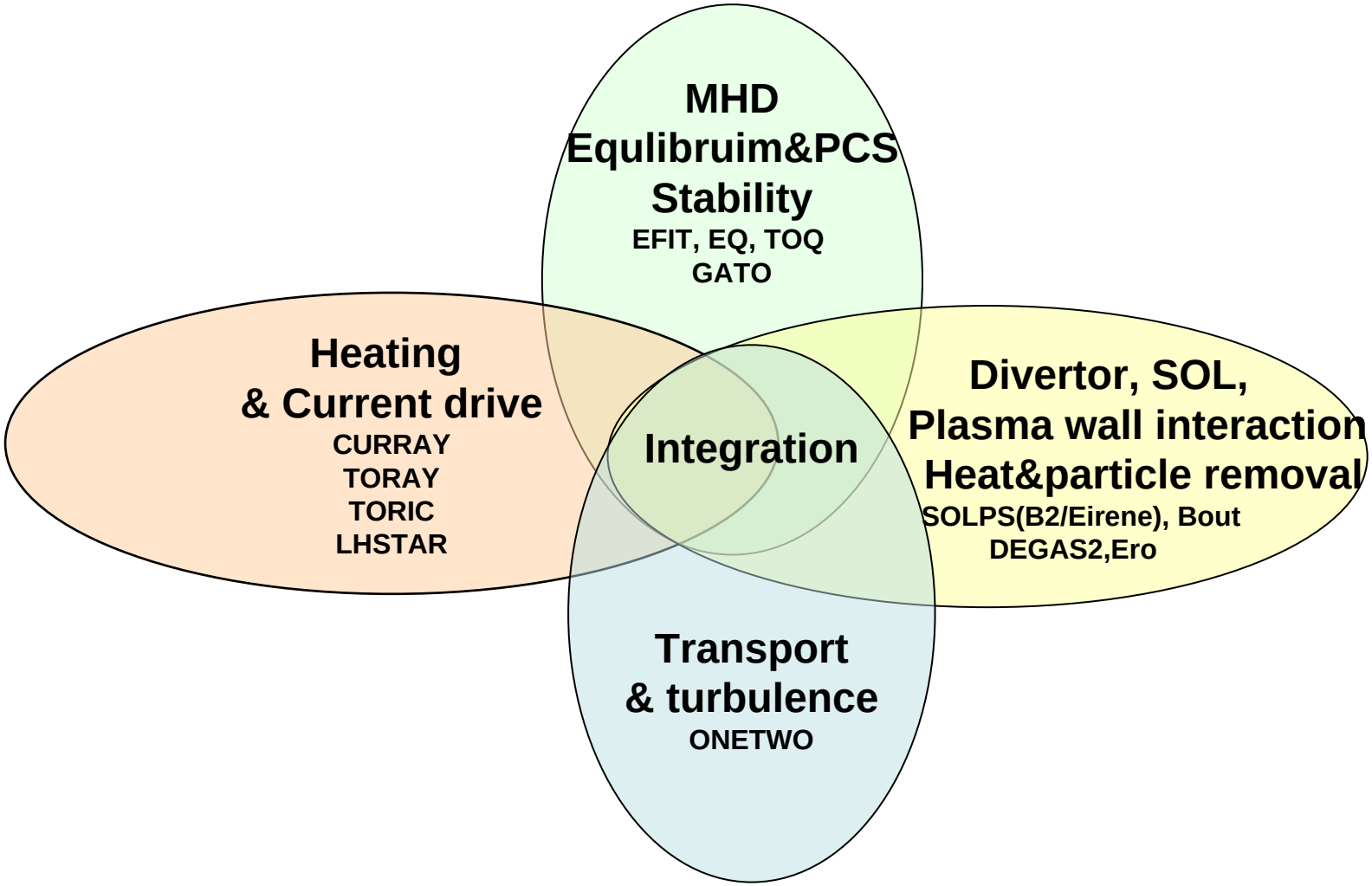
- Improved LHCD system, particularly control of power supply
- Optimized LHW coupling under reliable plasma shape control based on EFIT/Iosflux control algorithm
- Higher fuelling efficiency from Dome (see divertor experiments) reduced the fuelling gas amount for same line averaged density, e.g. retention of gas

See video

EAST near future Plan

	2010--4	2010--11	2011	2012	2013
Ip(MA)	0.5-0.8	0.5-1.0	0.5-1.0	0.5-1.0	0.5-1.0
R(m)	1.85	1.85	1.85	1.85	1.85
a(m)	0.45	0.45	0.45	0.45	0.45
K	1.5-1.9	1.5-1.9	1.5-1.9	1.5-1.9	1.5-1.9
Δ	0.3-0.6	0.3-0.5	0.3-0.5	0.3-0.6	0.3-0.6
ICRF(MW)					
30-110MHz	1	1	4.5	4.5	4.5
20-70MHz	4	4	4.5	4.5	4.5
LHCD(MW)					
2.45GHz	3	4	4	4	4
4.6GHz				4	6
NBI(MW)					4
ECRH(MW)				4	4
Diagnostics	30	30	35	40	40
configuration	SD, DN	SD, DN	SD, DN	SD, DN	SD, DN
Ptotal(MW)	8	9	13	20	25
t(s)	10-200	10-400	10-400	10-400	10-400
t(H-mode)s	5-20s	30s	40s	60s	100s
tplasma(s)	0.5MA 60s	0.8MA 100s	1.0MA 60s	1.0MA 100s	1.0MA 400s

Try to Build a Close Coupling between Theory and Experiments



New wall Conditioning techniques

ICR conditioning has been successfully carried out in EAST

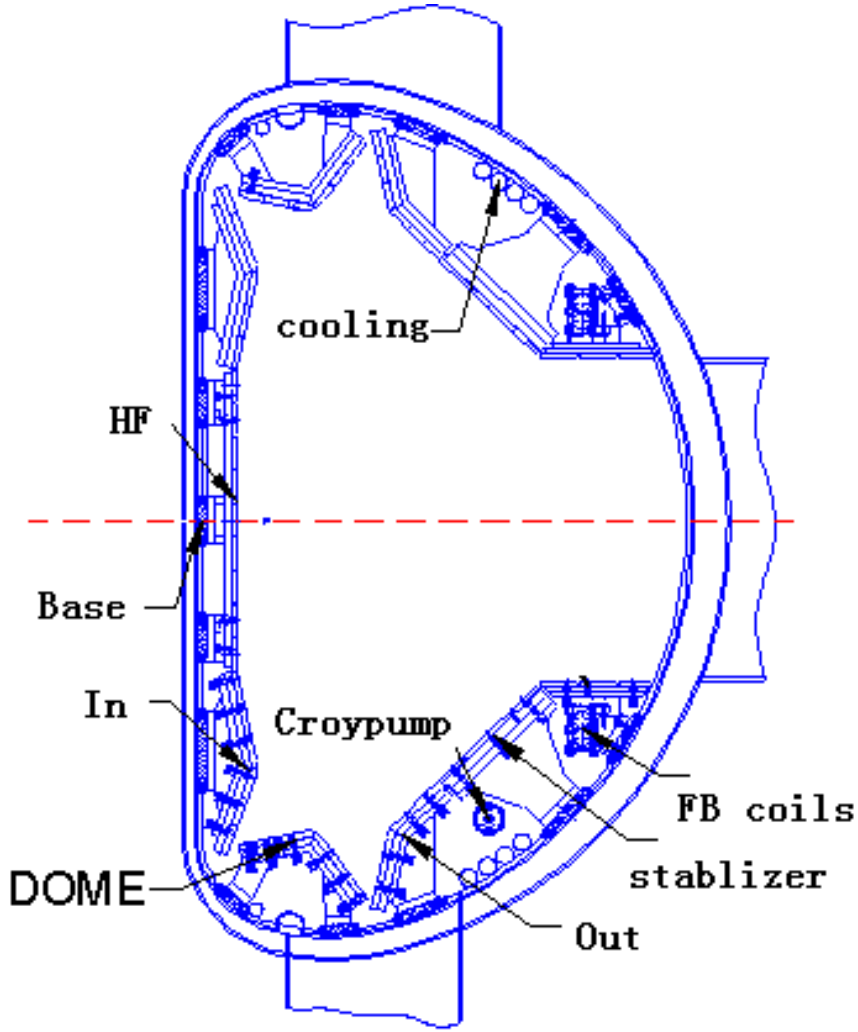
- **recycling control, boronization, Li coating and oxidation in comparison with GDC.**
- ✓ **High pressure and RF power are favorable for removal of hydrogen and impurities.**

HF conditioning was tested with very simple arrangement

Both are promising for ITER

Providing Experiences for ITER Main PSI Issues

- Full C PFC (-2010): 2 MW/m^2, 350 °C
  W/Cu divertor(-2014): 400 °C, 10 MW/m^2, liquid LI divertor
  Full W PFC (-2016), > 500 °C
- RF wall conditioning techniques in divertor devices (cleaning, isotopic control, boronization).
- Develop Tritium removal techniques that could be applicable to ITER.
- Steady-state erosion, redepositior and dust formation.
- In-situ control of T-codeposition and inventory by surface temperature control.



Providing Experiences for ITER AT SSO Issue

- Technologies for SS H&CD (LHCD, RF, NBI, ECRF)
- Coupling to the AT high performance plasma.
- Extend pulse length to steady-state condition with fully non-inductive current drive (LHCD + Bootstrap).
- SS ELMy control and mitigation

Outline

Technical improvements

- C PFCs with active water-cooling
- Basic Diagnostics
- H&CD systems
- Wall Conditioning

Experiments

- Physical engineering capability
- Plasma initiation and Ramp-up
- Plasma control
- Divertor experiments
- LHCD experiments
- Long pulse discharges

Research Plan in next 2-5 years

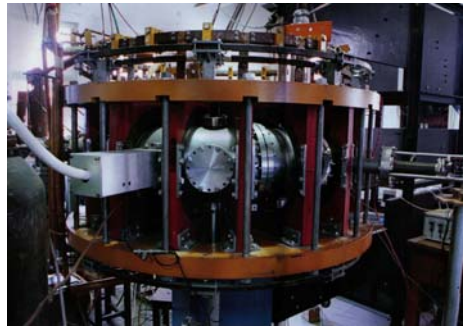
Future Plan of CN-MCF program

Summary

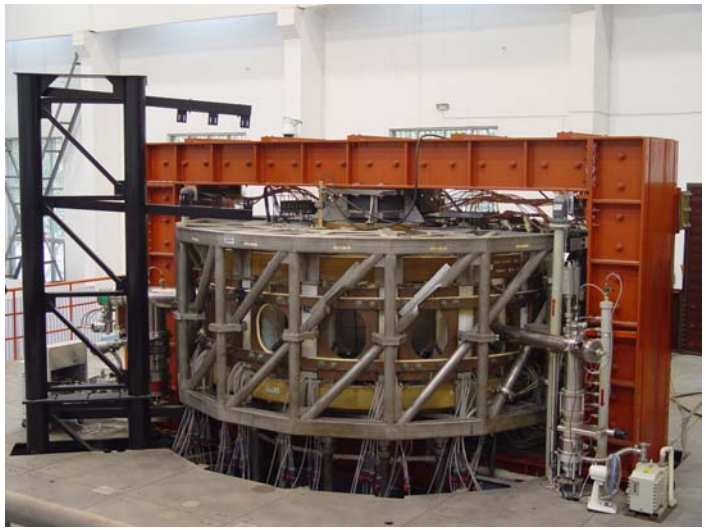
3 Tokamaks in university

For education

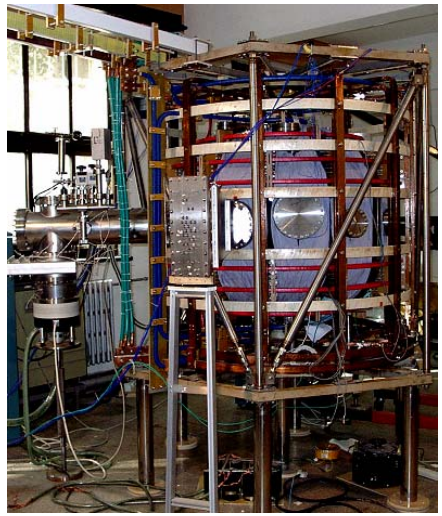
There are 30 stuff +40 students for tokamak experiments.
Theory group: 10+10
Material: 10+10
Others:20+20



KT-5 in USTC

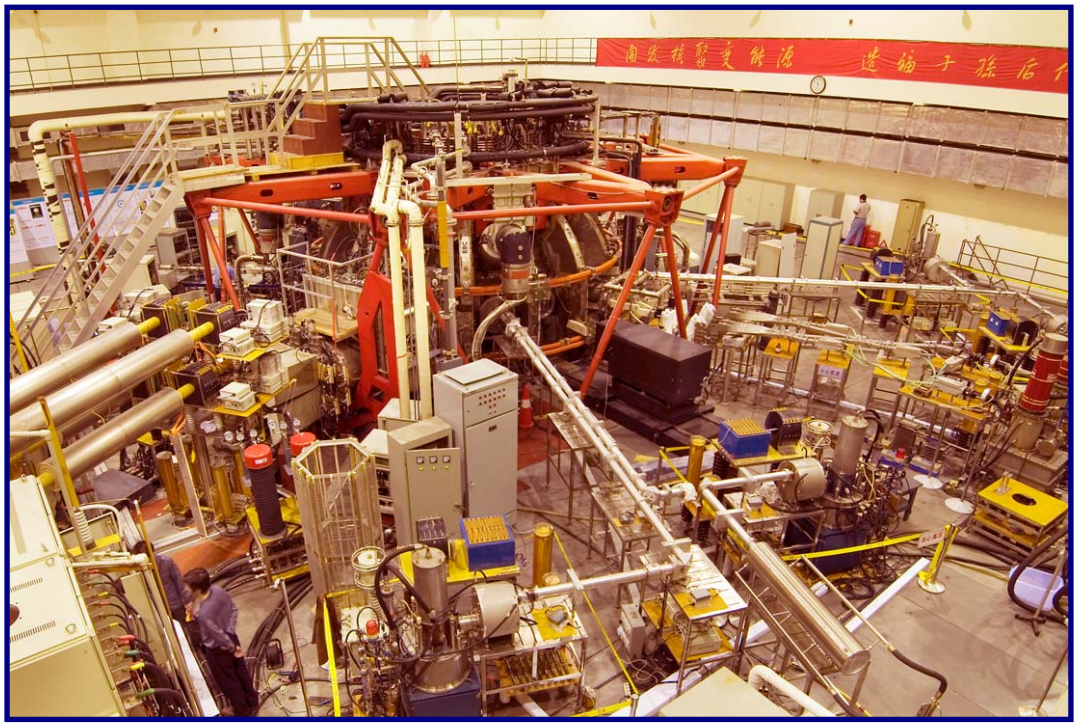


J-TEXT in Huazhong Univ.



SUNIST in Tshinghua

Status of HL-2A



- *R:* 1.65 m
- *a:* 0.40 m
- *Bt:* 1.2~2.8 T
- *Configuration:*
  Limiter, LSN divertor
- *Ip:* 150 ~ 480 kA
- *ne:* 1.0 ~ 6.0 x 10^{19} m^{-3}
- *Te:* 1.5 ~ 5.0 keV
- *Ti:* 0.5 ~ 1.5 keV

Auxiliary heating:

ECRH/ECCD: (3+2)MW

(6/68 GHz/500 kW/1 s)

modulation: 10~30 Hz; 10~100 %

NBI(tangential): 1.5 MW

LHCD: 1 MW

(2/2.45 GHz/500 kW/1 s)

Fueling system (H_2/D_2):

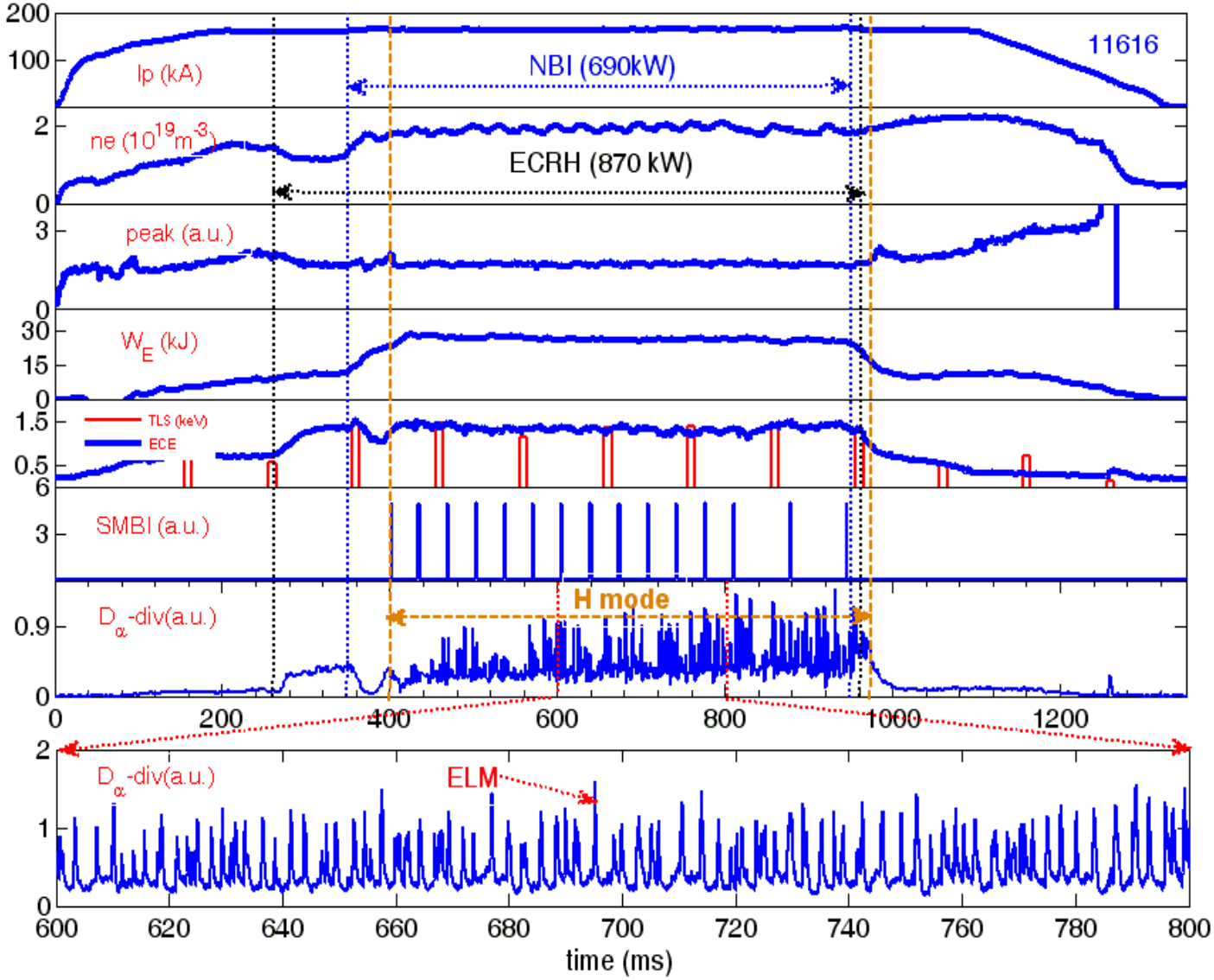
Gas puffing (LFS, HFS, divertor)

Pellet injection (LFS, HFS)

SMBI (LFS, HFS)

LFS: f =1~80 Hz, pulse duration > 0.5 ms
gas pressure < 3 MPa

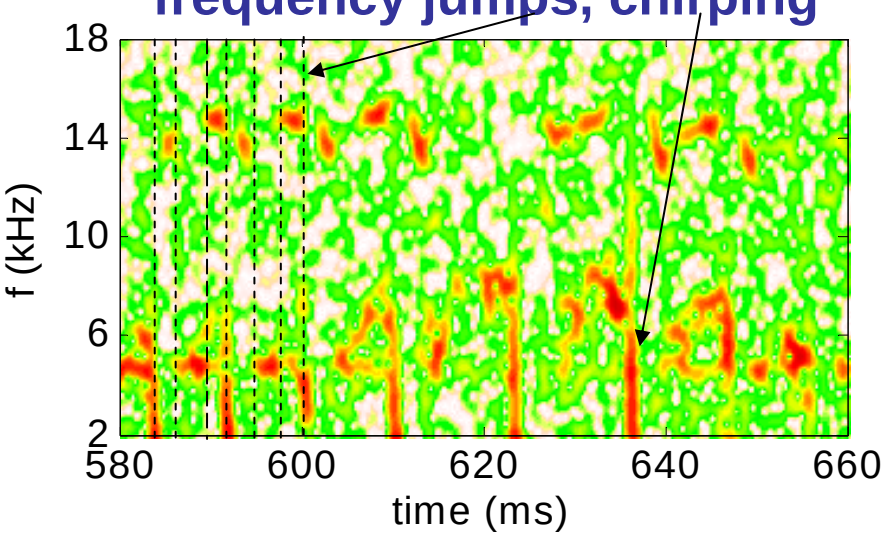
ELM H-mode on HL-2A



Physics of energetic particles

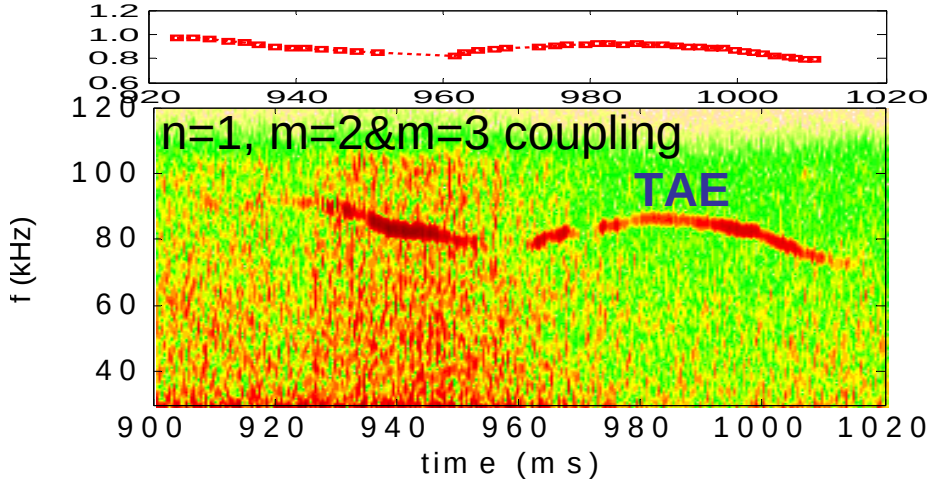
E-fishbone on HL-2A during ECRH

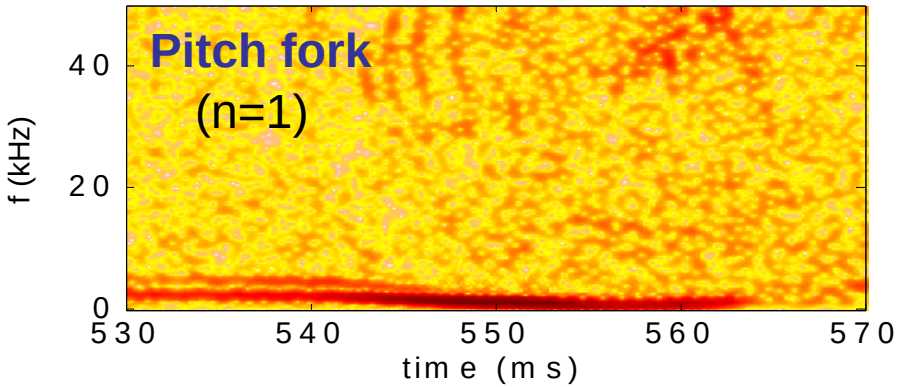
frequency jumps, chirping



Toroidalcity induced Alfvén eigenmode (TAE) during NBI

$$\omega_{TAE} = v_A / 2qR$$

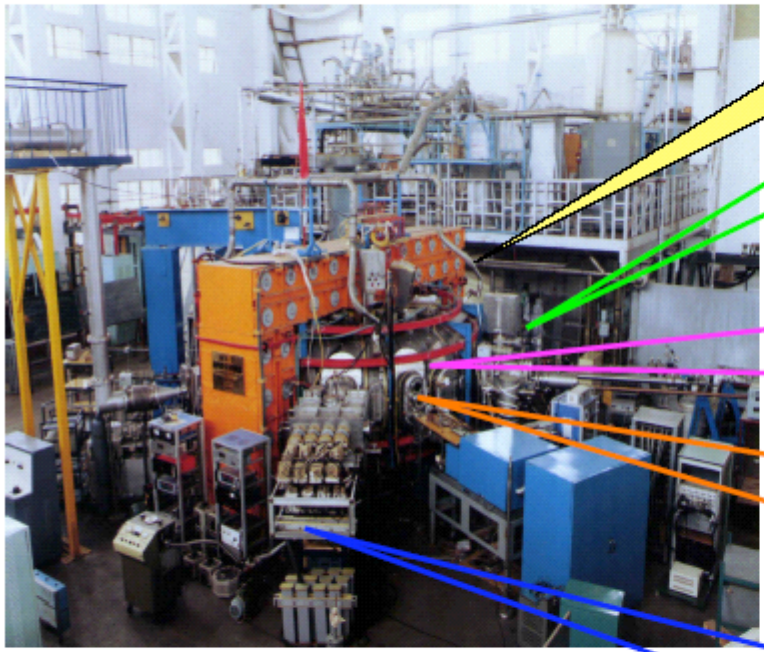




ASIPP

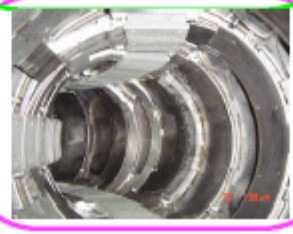
HT-7 experiments

Main Mission: Steady-state high performance operation and related physics and technologies

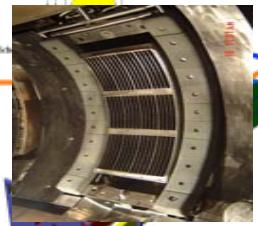


SS real-time PF PS&control

RF: 18-30MHz
350 kW CW
IBW, FW antenna



Graphite (P, T)
Water cooled



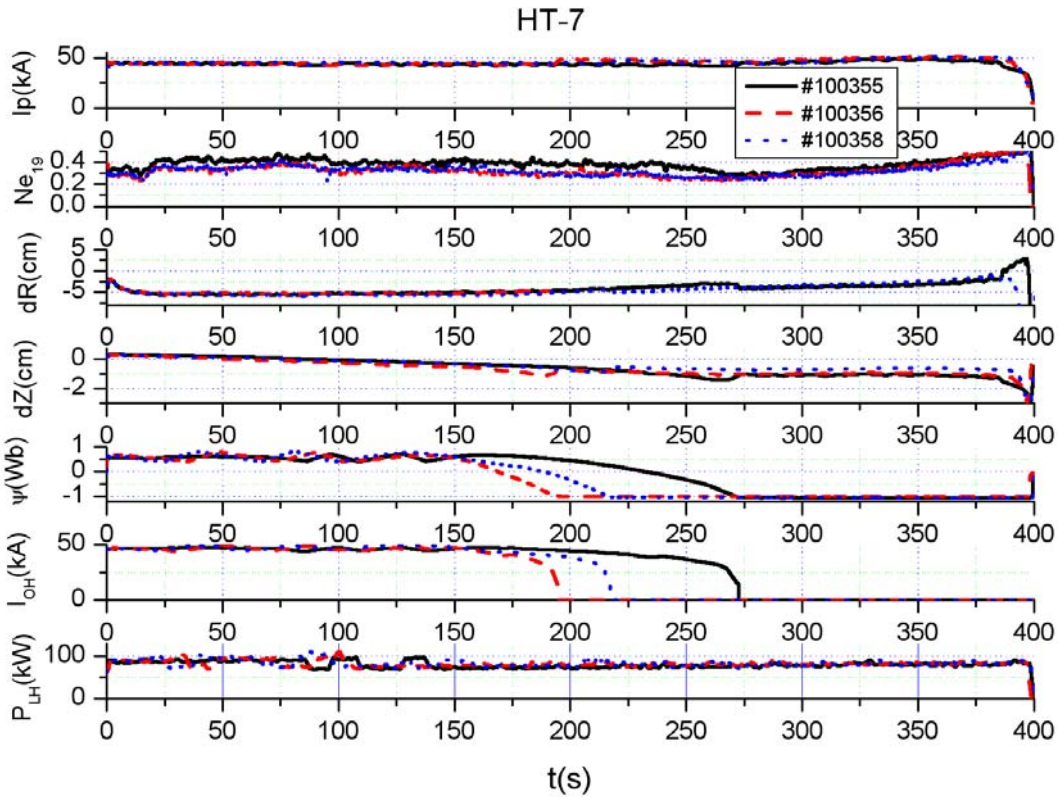
>30 diagnostics
Steady-state

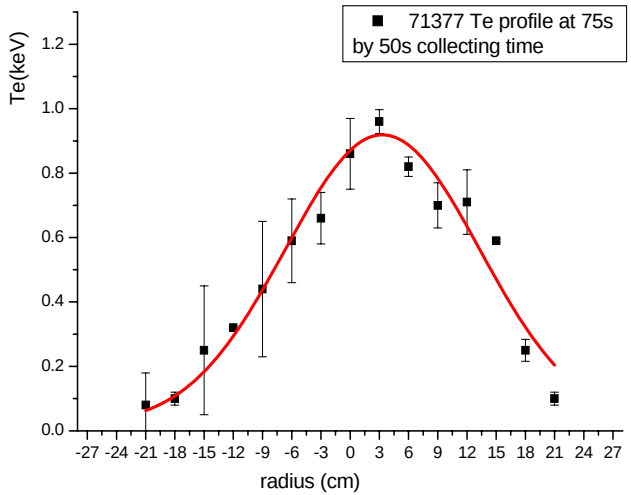
f: 2.45 GHz
1.2 MW CW
Multijunction

R = 1.22m, a = 0.27m ,

Ip = 100~250 kA, B_T = 1~2.5T

Repeatable 400s discharges (T_{e0}~1.0keV)





CN-ITER Home Teams

- Join the negotiation from 2003.2
- ITER-CN national committee
- Join ITER and ITPA activities
- CN-ITER home team (Hefei-Chendu branches) has been established.
- CN-Package: Feeders, Correction coils, Magnetic supports, PF conductor, Blanket, Transfer cask, AC-DC converter, Diagnostics, GDC and fueling, TBM

ITER-RD

CICC conductor

SC magnets

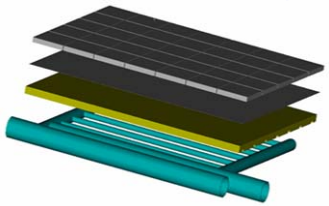
VPI

machining

SC magnet testing

R&D for ITERrelated technologies

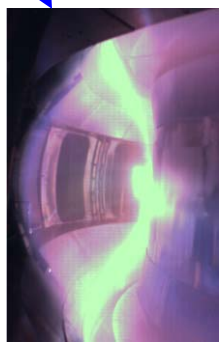
Power supply



PFC

HTc feeder

Plasma control



Cryogenic

ITER-Conductor

Multi-Vender activities

Wire: NICNC,Oxford

Coating:Shenghai Ltd

Wire testing:ASIPP

Central tube: Tai Steel,

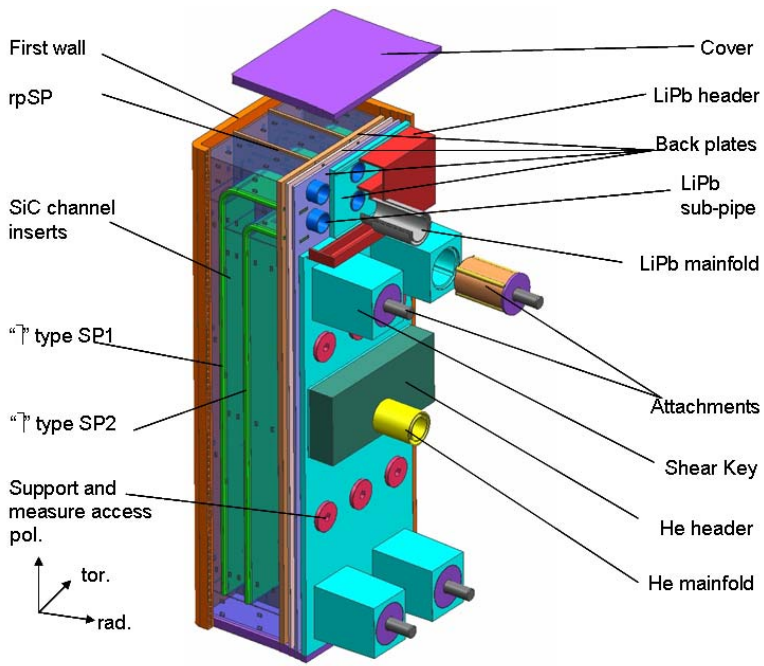
Cabling: Basheng Ltd,

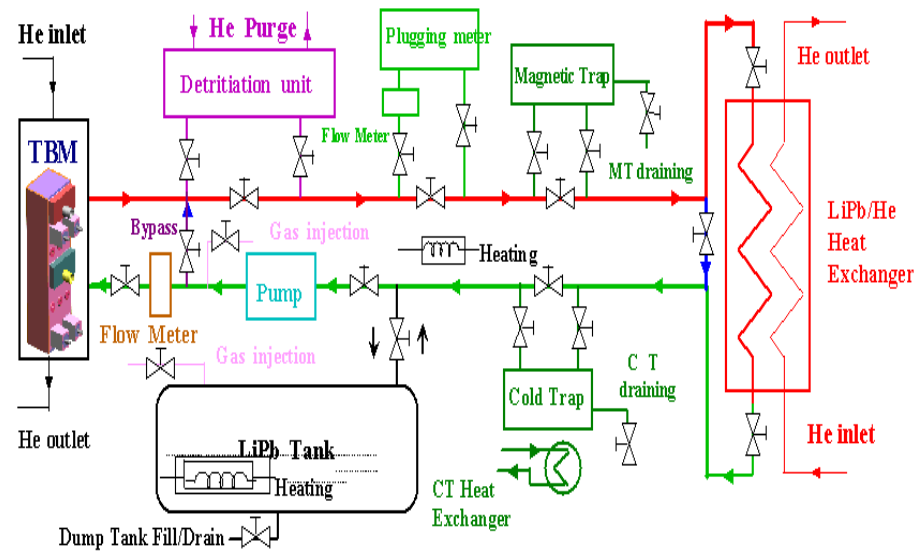
316LN Tube:

Integration:ASIPP

1000m jacketing line
In ASIPP

R&D of TBM

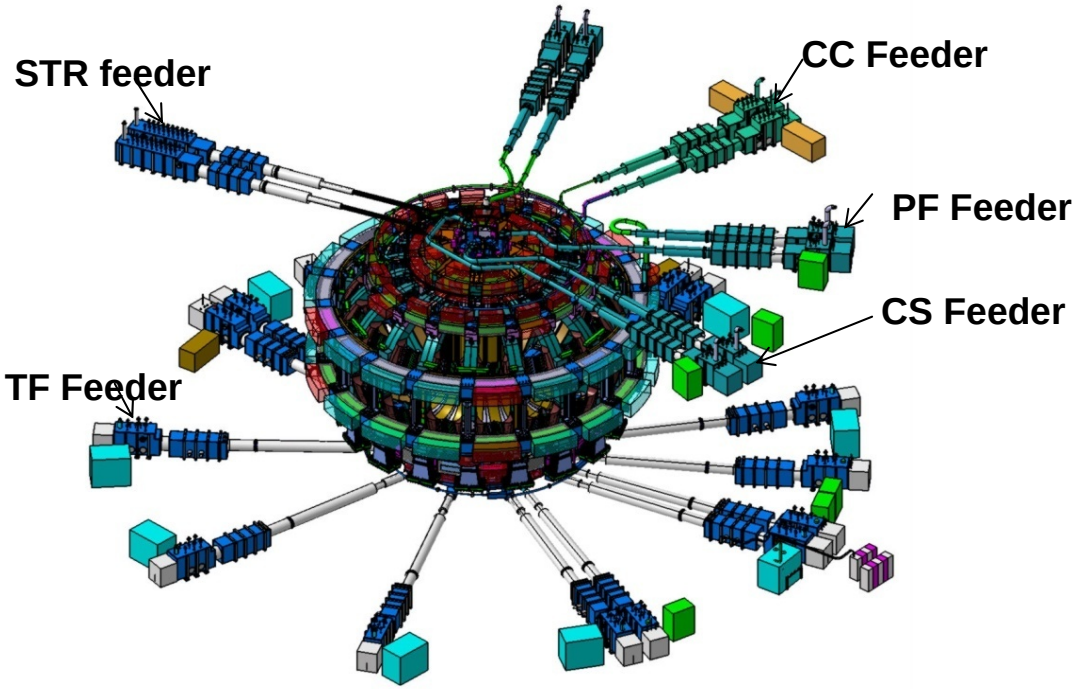




Dual-cooled He/LiPb (DLL) TBM

Feeder Team

Feeder



TF feeder：9

PF feeder：6

CS feeder：6

CS/TF joint feeder：3

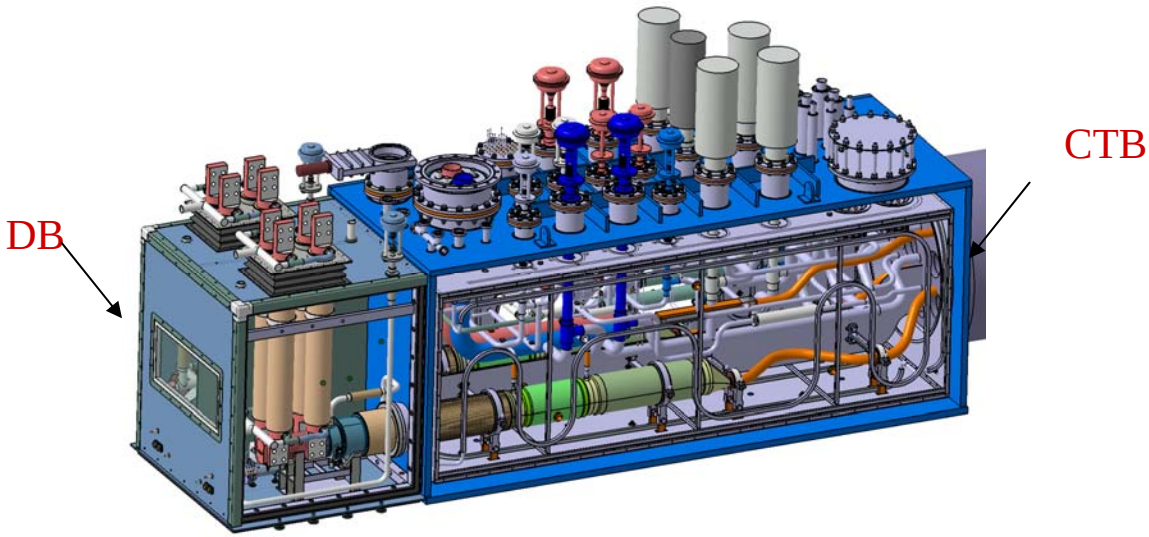
CC feeder：5

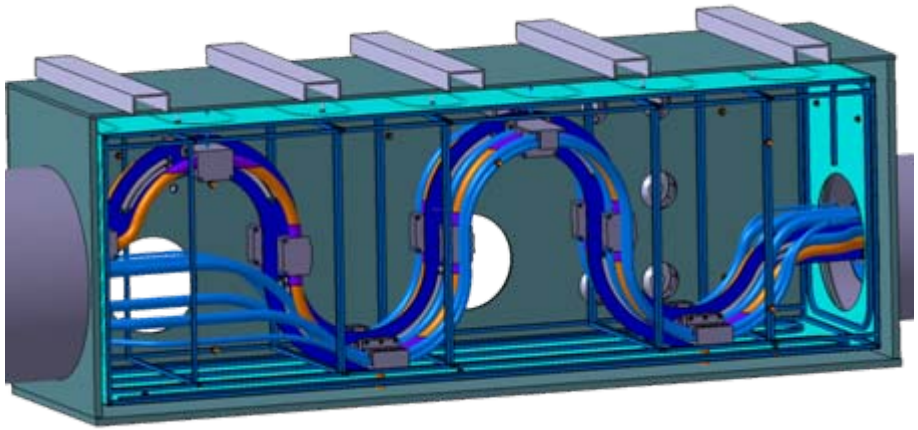
Diagnostic feeder：2

Total：31 different feeders

Feeder Team

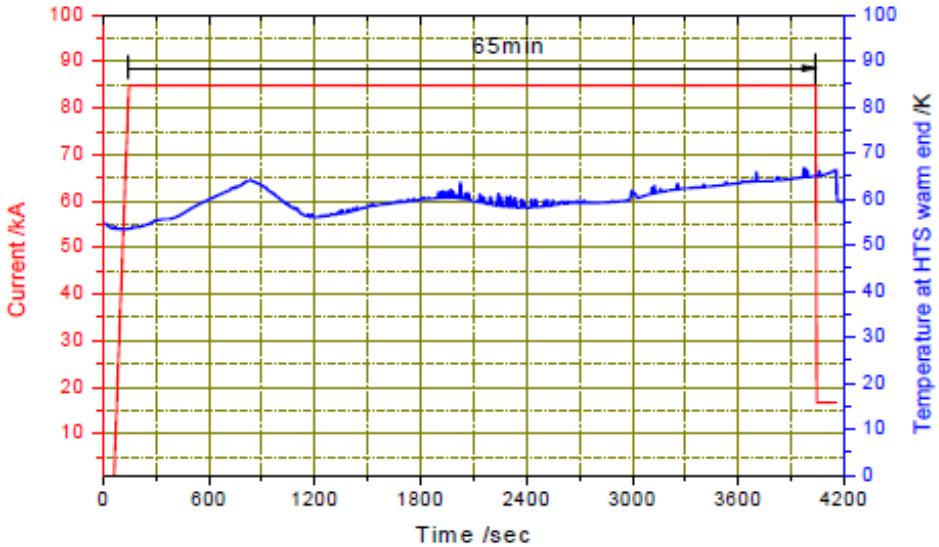
Detail design is nearly finished





HTc current leads

ITER

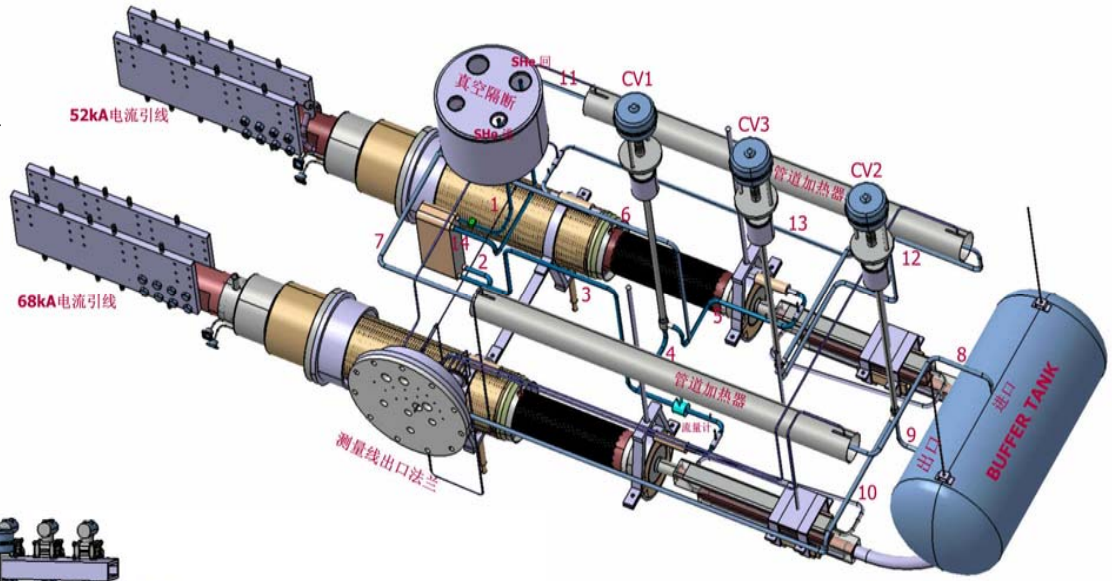


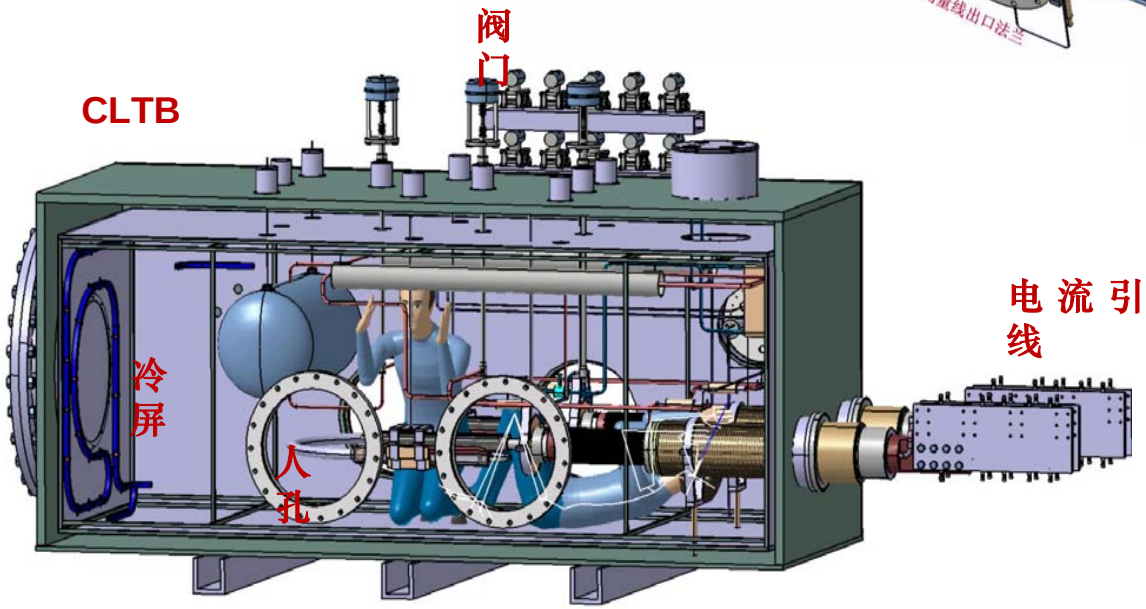
额定电流 kA	52	68
HTS 制造商	中国 INNOST	美国 AMSC
稳态通电电流与时间，在 65K HTS 热端温度	57 kA/ 90 min	85 kA/ 64 min
最大电流和通电时间 冷却条件	90kA/4 min >60kA，用 5K 氦流强化冷却	90kA/4 min 65K HTS 热端温度
HTS 5K 端与分流器接头电阻 nΩ	0.345	0.538
分流器 5K 端与低温超导线接头电阻 nΩ	5.18	5.12
HTS 65K 端接头电阻 nΩ	3.22	2.25
与低温超导母线接头电阻 nΩ	0.88	1.38
失冷后继续通额定电流至失超的时间	360	320
失超后继续通额定电流至 HTS 最高温度 160K 时间	21	22
HTS 温端 65K 运行时的温度裕度 K	17.5	25.4
0－额定电流－0 电流循环	19 次 ＋ 10 次脉冲电流	15 次

Feeder Team

R&D is going
(Proto-type HTc feeder testing)

内部布管图





Near term CN-MCF Plan (2020)

ITER construction

- **ASIPP: Feeders (100%), Correction Coils (100%), TF Conductors (7%) , PF Conductors (69%), Transfer Cask System(50%), HV Substation Materials (100%), AC-DC Converter (62%)**
- **SWIP: Blanket FW (10%) &Shield (40%), Gas Injection Valve Boxes+ GDC Conditioning System (88%), Magnetic Supports (100%),**
- **Diagnostics (3.3%)**

Enhance domestic MCF

Upgrade EAST, HL-2M

ITER technology

TBM

University program

DEMO design

Material

Y. WU et.al, J. of Fusion Engineering and Design, 2006, 81(23-24): 2713-2718

FDS Series Fusion Reactors & Blankets Conceptual Design for Plants

- **FDS-I: Fusion-driven Subcritical System**

  for early applications of fusion (multi-function)

  e.g. waste transmutation, fuel breeding etc.

- **FDS-II: Fusion Power Reactor**

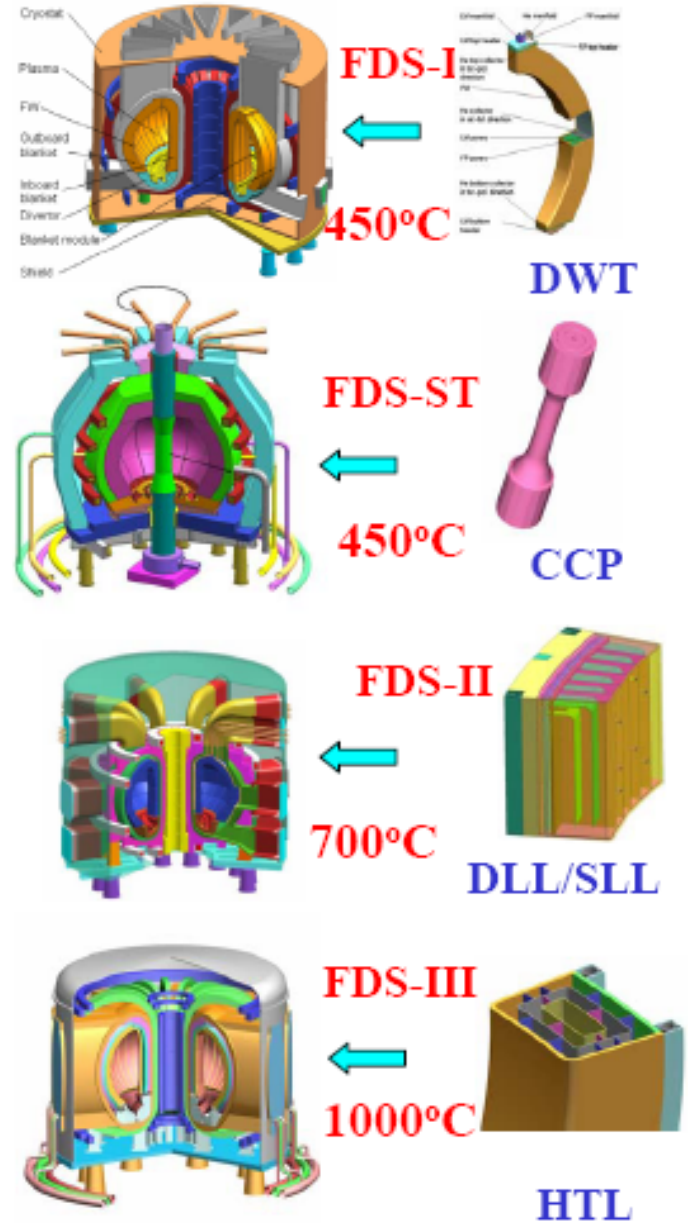
  for highly efficient electricity generation

- **FDS-III: High Temperature Fusion Reactor**

  for advanced applications, e.g. hydrogen production

- **FDS-ST: Spherical Tokamak-based Reactor**

  for exploiting and assessing innovative conceptual path



Development of LiPb Loop Technology

Operation/Fabrication/Design:

- Thermal convection loop (compatibility)
  - DRAGON-I : 316SS, 500℃ (in operation)
  - DRAGON-II : Inconel, 700℃ (in operation)
  - DRAGON-III: SiC_f/SiC, 700-1000℃ (in fabrication)
- Forced convection loop (MHD)
  - DRAGON-IV: 480-700℃ (in fabrication)
- Auxiliary system for TBM in EAST
- Auxiliary system for TBM in ITER

DRAGON-I

✓ Design Objectives:

- Thermal convection loop (500℃)
- Compatibility of CLAM and 316L steel with LiPb.

✓ Main Parameters:

- Loop Size : 0.5m×0.5m
- Structural Material : SS316L
- Inner-diameter : 22mm
- LiPb inventory : 1 liter
- Temperature : 420 ~ 480℃
- Flow rate : ~ 0.08m/s

DRAGON-II

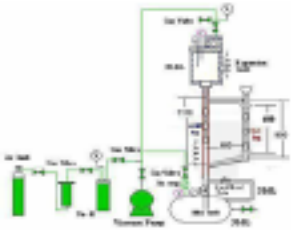
✓ Design Objectives:

- High temperature (700℃) thermal convection loop
  - hot leg: 700℃
  - cold leg: 480-640℃
- Obtain corrosion results for the TBM-DLL concept

✓ Key issues:

- Material selection (700℃)
- Fabrication technology
- Smelting of Large quantity LiPb

DRAGON-III



✓ Design Objectives:

- High temperature (700-1000℃) thermal convection loop
- Pursue the manufracture of SiC_f/SiC materials for fusion
- Compatibility of SiCf/SiC with LiPb

Quartz Loop Base

SiC_f/SiC Loop

✓ Key issues:

- Fabrication of SiC_f/SiC Loop
- Connection technology
- Heating method of high temperature (up to 1000℃)

DRAGON-IV

✓ Design Objectives:

- MHD Experiment (>2T, 1m/s);
- Gerenral corrosion in flowing LiPb: -800℃;
- Stress corrosion: max~50kN.

✓ Key issues:

- Structureal Materials
- LiPb Purification
- Loop construction and measurement technology

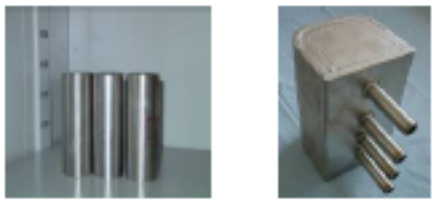
Design and Analysis Tools Development

1. Multi-functions neutronics calculation code system: VisualBUS
2. Multi-physics (neutronics, thermalhydraulics, MHD) coupling simulation codes: NTC/MTC
3. System Engineering (safty/economy) analysis codes: RiskA, SYSCODE
4. Fusion virtual assembly system: FVAS
5. material database and component reliability database management system: FUMDS, RiskBase

- 50~100 man-years each program

Key Tools for Fusion/Fission Reactor Design & Analysis

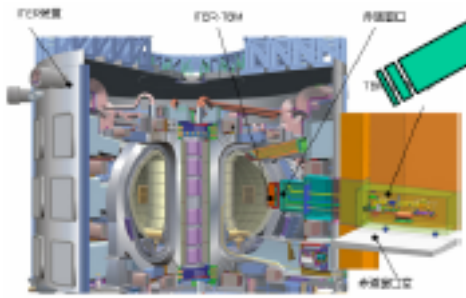
Roadmap of Fusion to DEMO



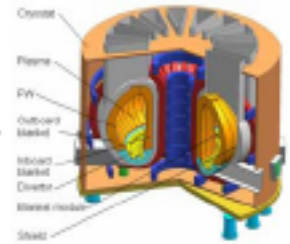
TBM & Materials R&D

EAST

Concept & Eng. design

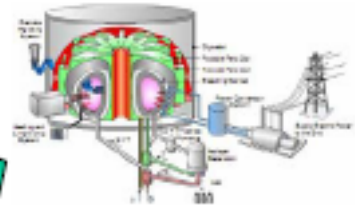


ITER



Hybrid

DEMO



FPP

- Waste Transmutation
- Fuel Breeder
- Energy Multiplication
- Materials Test

~15 yr earlier

Now~2015 | 2015~2025 | 2030~2040 | 2050

Other Activities in ASIPP

Main research Activities

- Magnetic confinement fusion and high temperature plasma physics
- Fusion engineering and technologies Reactor design
- High magnetic field
- Ion beam implant biology
- Low temperature plasma and application
- Superconductor and cryogenic engineering
- R&D of solar cell

Manpower and Team

• Permanent stuff	450
• Contract	100
• Average age	42
• Senior researchers	46
• Associate professors	80
• Graduated students	416
• Post doctors	6
• Divisions	14
• Centers	2
• R&D Companies	2

ASIPP

Summary

- **With HL-2A, HT-7 and EAST operation, China will make more efforts for fusion development.**
- **With ITER project, China will work more closely with other 6 parties for fusion research.**
- **More close cooperation between China and USA will beneficial fusion research.**
- **ASIPP would like to have close cooperation with ORNL for the early use of fusion energy.**

Thanks