

Introduction of Fusion Research Activities at SWIP

China Fusion Energy Co., Ltd.

Southwestern Institute of Physics

ZHONG Wulyu

2026.7.20

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01

Introduction of CFEC & SWIP

- Founded in 1955, former Ministry of Nuclear Industry
- Centrally managed state-owned enterprise approved by the State Council
- Key player in the national nuclear science and technology industry
- The only comprehensive nuclear industry enterprise among the Fortune Global 500

China National Nuclear Corporation



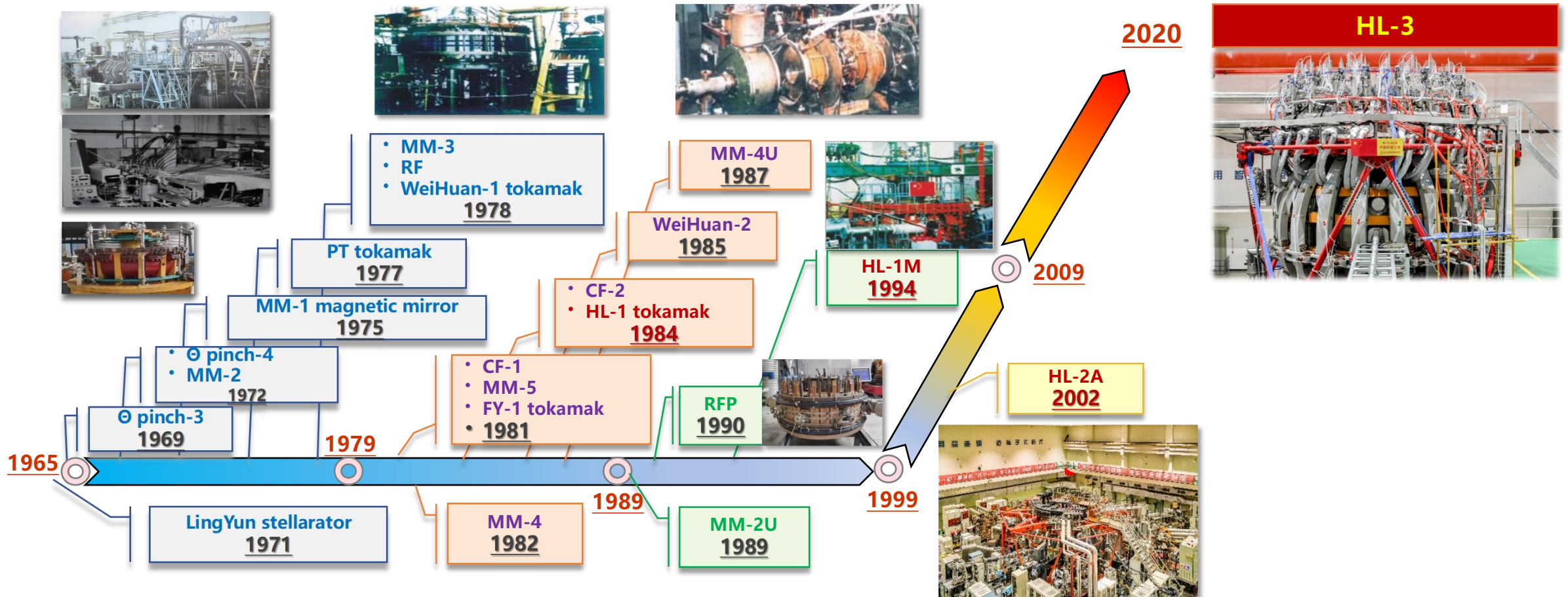
- Mission: Promoting the development and utilization of nuclear energy, nuclear engineering construction, and the application of nuclear technology.
- Total assets has reached approximately 1.7 trillion RMB Yuan.
- > 170,000 employees.

Southwestern Institute of Physics (SWIP)

Affiliated with China Fusion Energy Co., Ltd (CFEC), CNNC

Earliest institute for fusion energy research in China, founded in 1965 with staff > 2000

SWIP has developed **more than 20 fusion research devices** of various magnetic configurations



Based on 60-year fusion energy R&D of **SWIP**

China Fusion Energy Co., Ltd. (CFEC) established in 2025, affiliated to CNNC



CFEC established in Shanghai in July 2025

Shareholders include:

- China National Nuclear Corporation
- China National Petroleum Corporation
- National Green Development Fund
- Shanghai Future Fusion Energy Technology Co., Ltd
- Zheneng Electric Power Co., Ltd
- Chengdu Industry Co., Ltd

R&D layout with 3 Campuses in 2 Cities



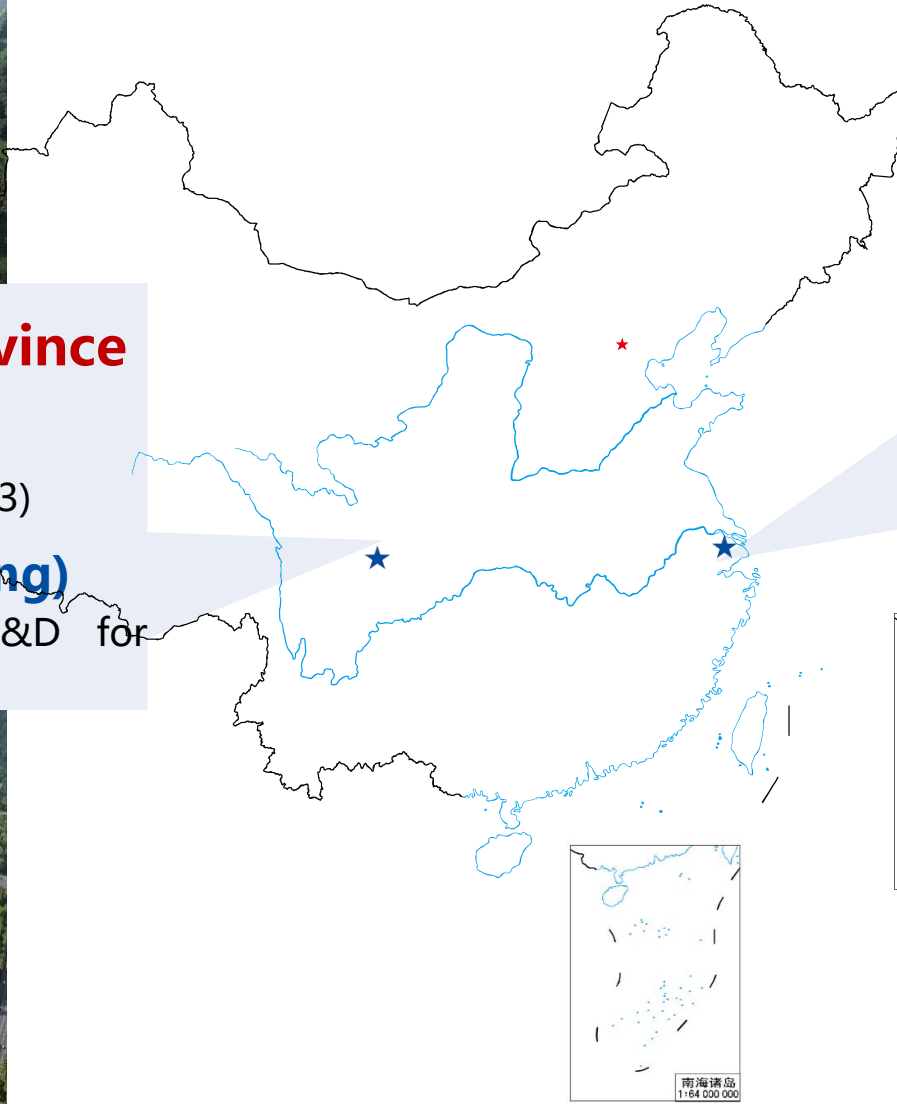
📍 Chengdu, Sichuan Province

■ Shuangliu District (Baijia)

Frontier scientific experiment (HL-3)

■ Tianfu New Area (Yongxing)

Key engineering technology R&D for fusion reactor



📍 Shanghai

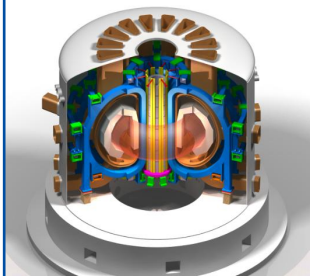
■ Minhang District (Maqiao)

- Fusion reactor design, high-temperature superconducting (HTS) magnet, digital tokamak
- High-field tokamak with HTS magnet (HL-4)



Construction started in Dec. 2025
and will be completed in 2030

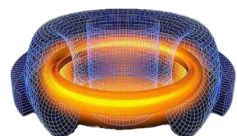
Fusion Engineering Design Center



Integrated design of fusion reactors

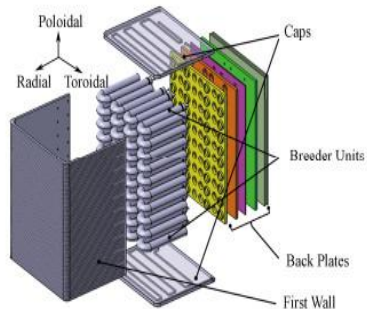


High temperature superconducting (HTS) magnets R&D

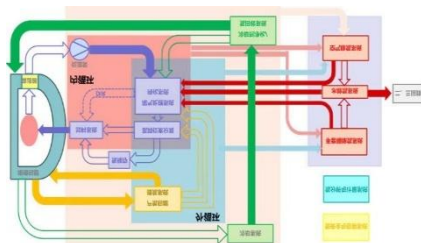


Digital tokamak

Fusion Technology R&D Center



First Wall R&D



Fuel cycling system

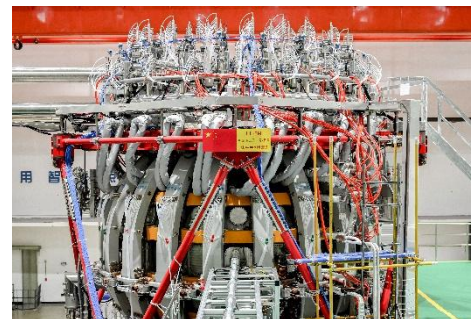


Power supplies for special applications



Heating and Current Drive R&D

Fusion Scientific Experiment Center



HL-3 Tokamak



Frontier Scientific Experiments

Plasma Technology Application Center



Plasma Environmental Governance Technology

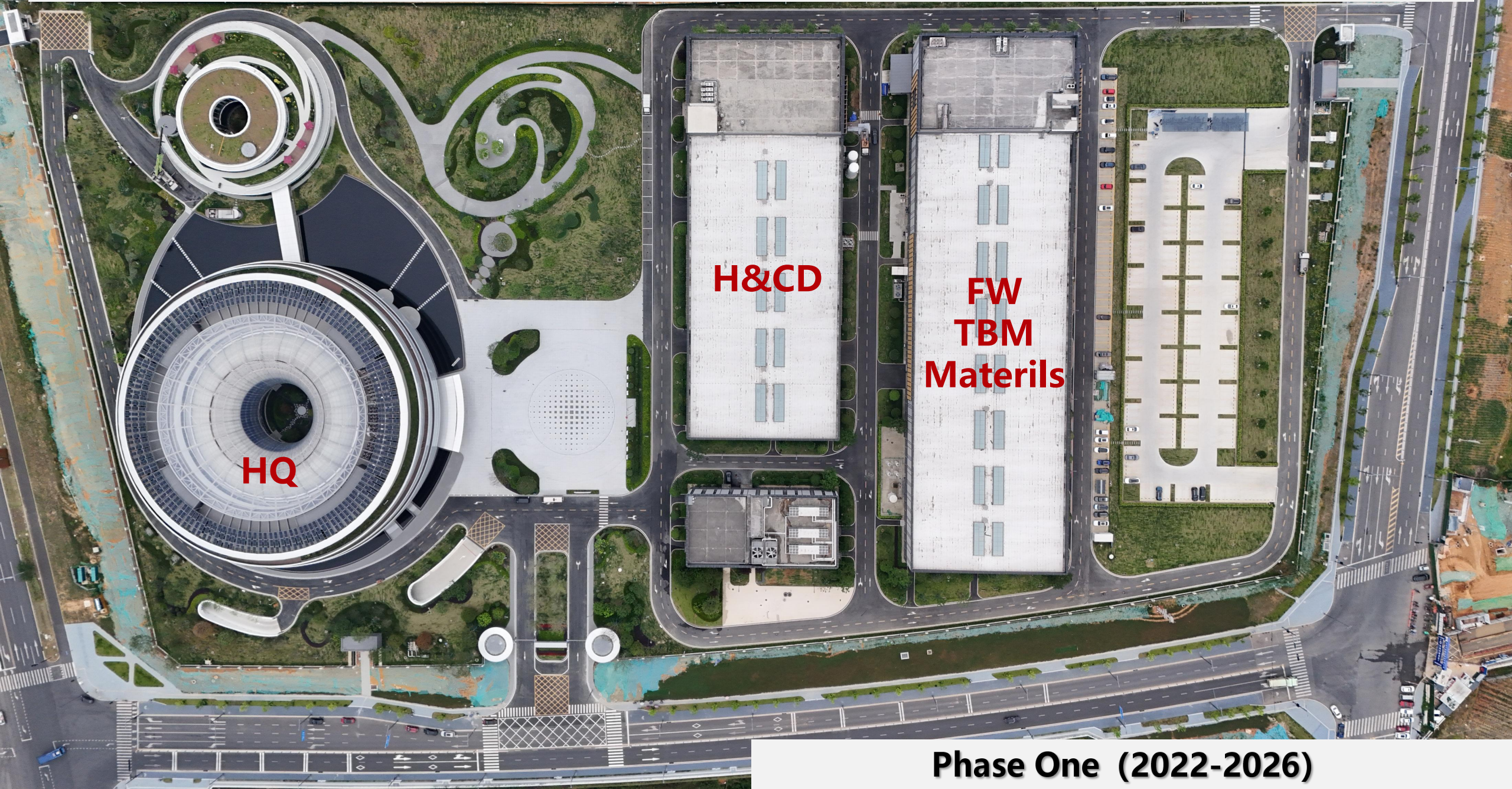


Plasma Surface Engineering Technology

Fusion Scientific Experiment Center (Baijia Campus, Chengdu) (~ 5 hectares)



Nuclear Fusion Technology R&D Center (Yongxing Campus, Chengdu) (~ 33 hectares)



Phase One (2022-2026)



Interior of Zhengwu Building (HQ)

Maqiao Campus in Shanghai

For fusion reactor design, HTS magnet, digital tokamak, HL-4... (about 13 hectares)



HL-4

02

HL-3 Tokamak

Tokamaks developed by SWIP

Size
Parameters

$n T_i T_e$



HL-1 (1984)

Radius 1.02 / 0.20 m

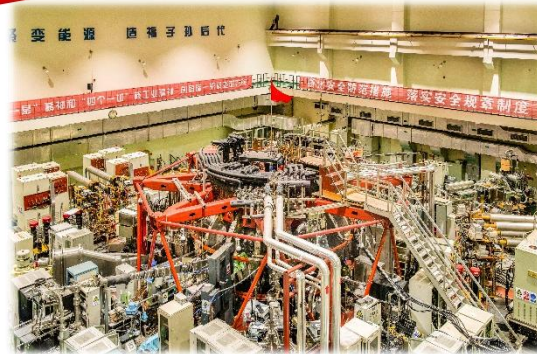
First large scientific and engineering fusion device in China.



HL-1M (1994)

Radius 1.02 / 0.24 m

Plasma current >300 kA
Plasma fueling by SMFI proposed for the first time.



HL-2A (2002)

Plasma current 0.45 MA

Radius 1.65 / 0.4 m

First diverted tokamak in China.
Important milestones achieved in fusion field.



HL-3 (2020)

Plasma current 2.5 (3) MA

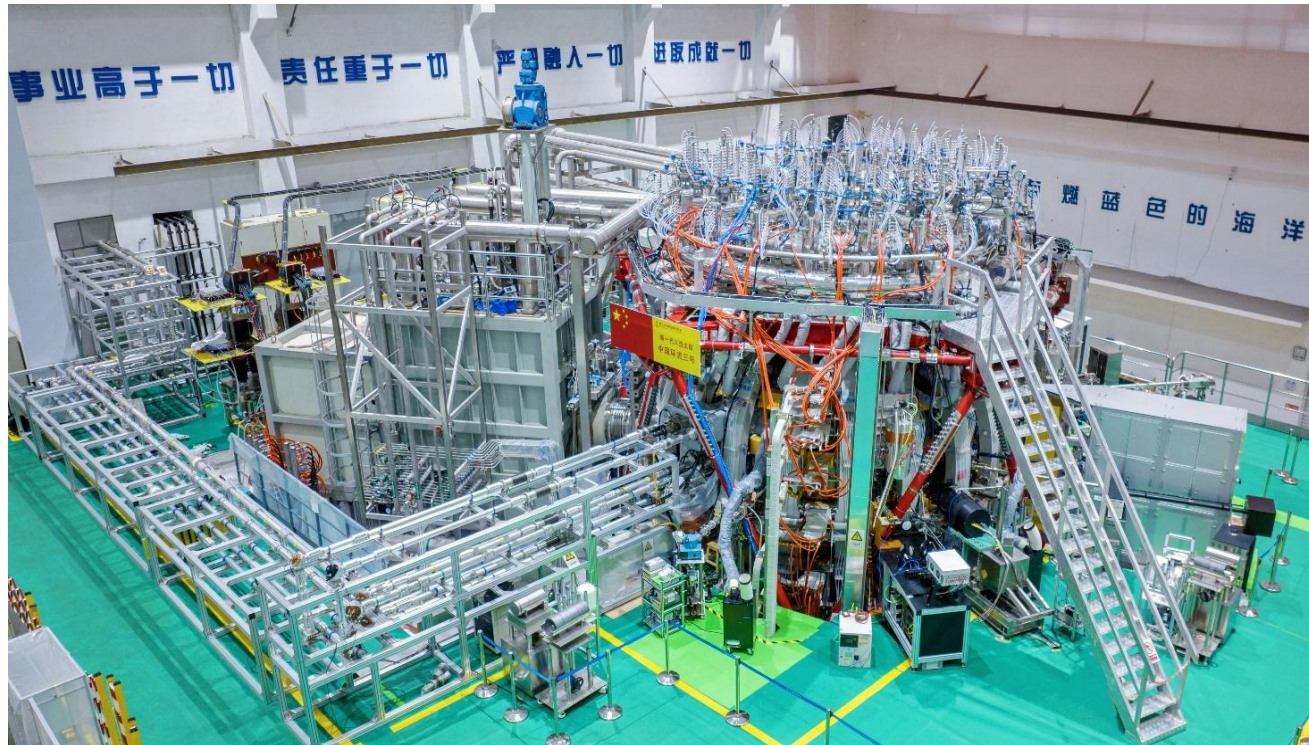
Radius 1.78 / 0.65 m

Largest tokamak in China with highest parameter and advanced divertor.

Mission of HL-3 tokamak

■ CN largest and highest parameter Tokamak to High Performance Plasma Operation

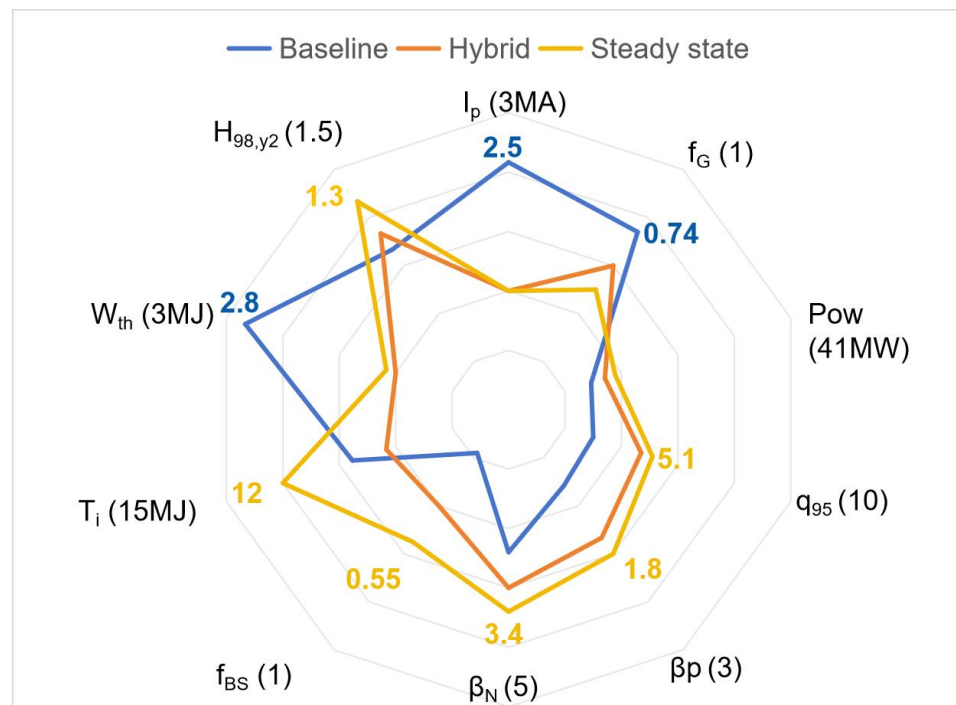
- High-performance scenarios compatible with low heat load
- Deep understanding of frontier plasma physics
- Tests and validation of high heat flux plasma-facing components



HL-3 Main Parameters

Major radius	$R = 1.78 \text{ m}$
Minor radius	$a = 0.65 \text{ m}$
Aspect ratio	$R/a = 2.8$
Plasma current	$I_p = 2.5 \text{ (3) MA}$
Toroidal field	$B_t = 2.2 \text{ (3) T}$
Elongation	$\kappa = 1.8\text{-}2$
Triangularity	$\delta > 0.5$
Heating power	NBI (20) + ECRH (11) + LHCD (4) + ICRH (6) MW

■ Comprehensive operation scenarios



□ Baseline Operation

$$I_p = 2.5 \text{ MA}, q_{95}=3, H_{98,y2} = 1.0$$

□ Hybrid Operation

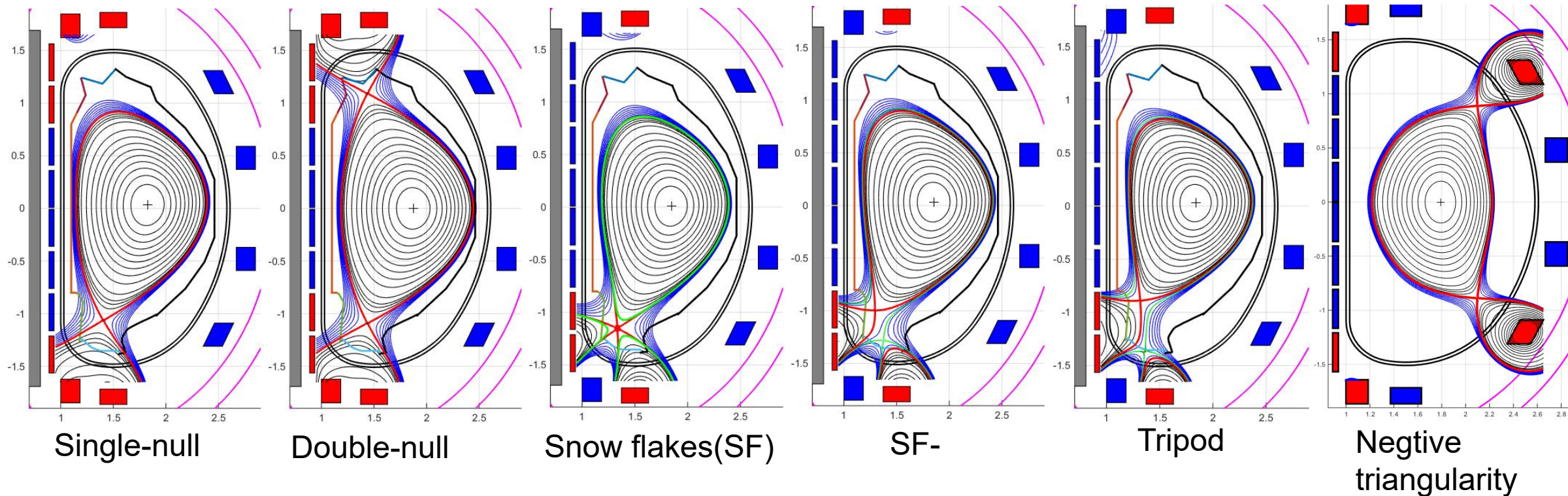
$$I_p = 1.0 \sim 1.4 \text{ MA}, f_{BS} \sim 40\%, f_{ni} \sim 70\%, \beta_N > 2.5$$

□ Steady-State Operation

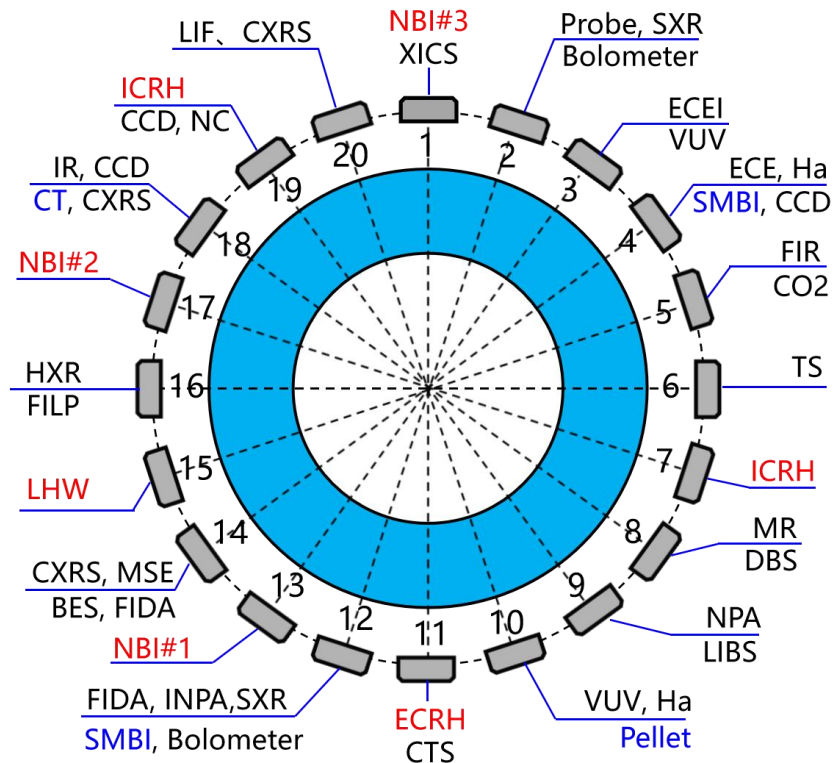
$$I_p = 1.0 \sim 1.2 \text{ MA}, \text{fully non-inductive current drive}, \beta_N > 3.0$$

Key features

- Comprehensive operation scenarios
- Flexible magnetic configurations



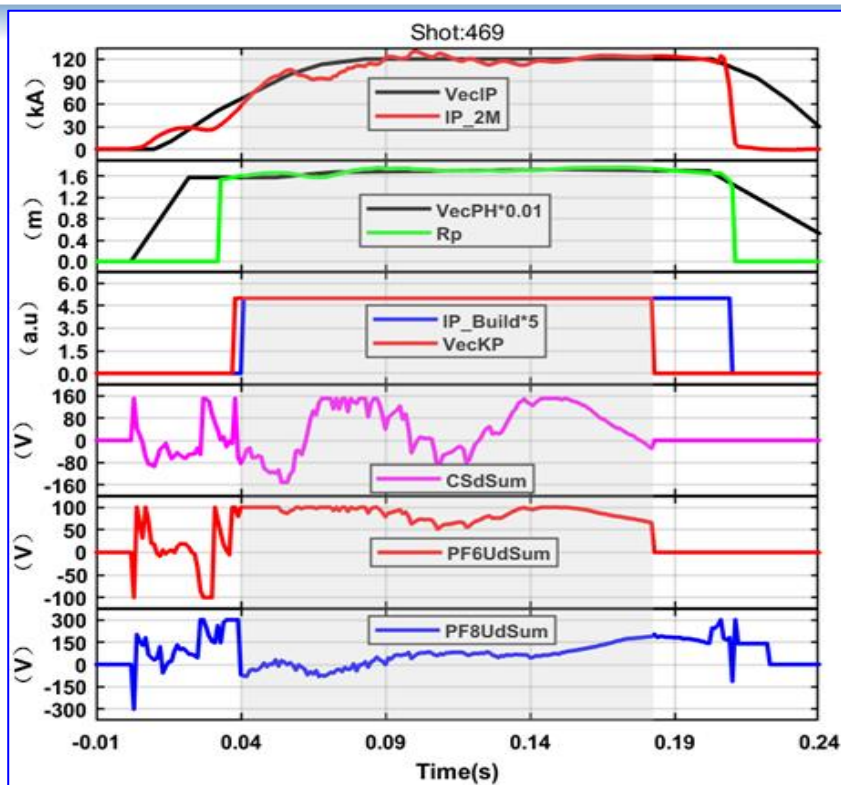
- Comprehensive operation scenarios
- Flexible magnetic configurations
- High accessibility for experiments



- 128 ports: diagnostics, fueling, heating and current drive systems
- Flexible capacity on heat and particle exhaust: actively cooling divertor and strong pumping speed cryopump (38 m³/s)

First plasma on Dec. 4th, 2020

First plasma: >100 kA, >100 ms



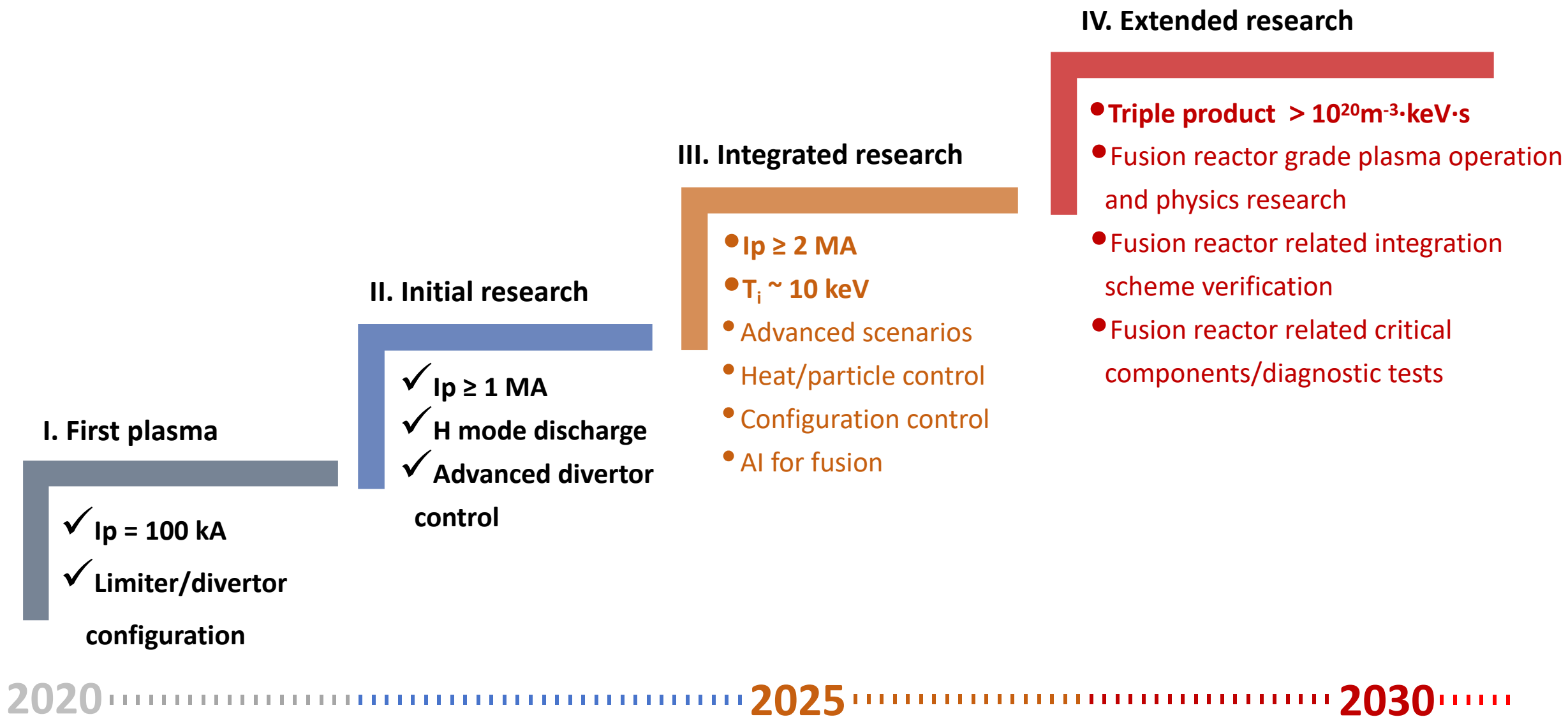
Reported by CCTV News on Dec. 4th, 2020



Selected as:

"Top 10 News of Scientific and Technological Progress in China in 2020"

Staged research plan of HL-3



Heating and current drive systems

Plasma heating, current drive,
plasma instability control, and
plasma start-up assist

■ Neutral Beam Injection

(NBI): 20 MW / 120 keV / 5s

■ Electron Cyclotron Resonance Heating

(ECRH): 11 MW / 105 GHz, 140
GHz, and 175 GHz / 10s

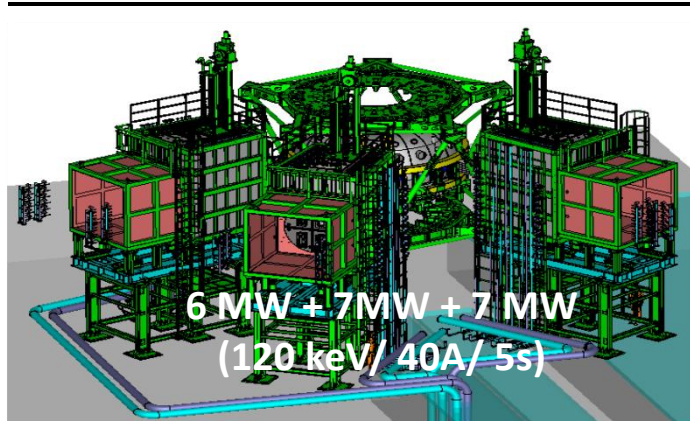
■ Ion Cyclotron Resonance Heating

(ICRH) : 6 MW / 25–50 MHz / 5s

■ Lower Hybrid Current Drive

(LHCD): 4 MW / 3.7 GHz / 5s

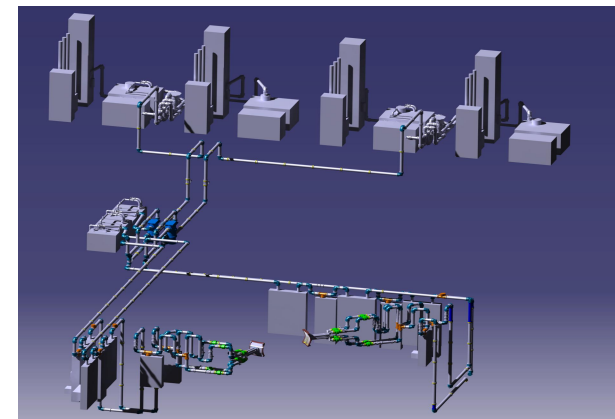
NBI



3 systems

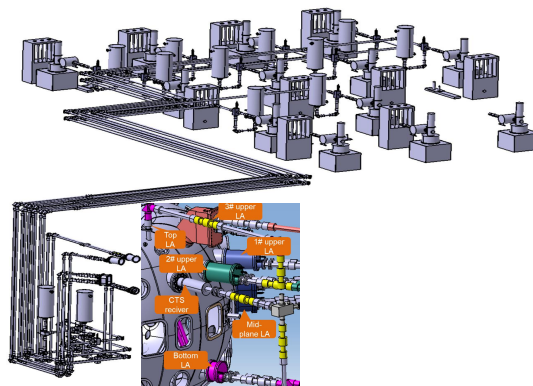
1 # (2023), 2#(2025), 3#(2026)

ICRH



4 × 1.5 MW (25–50 MHz)

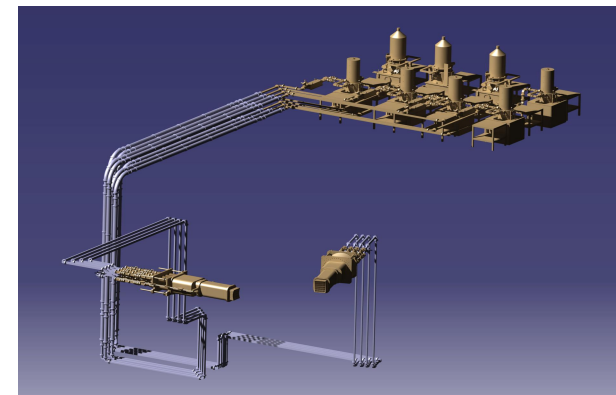
ECRH



EL: -20~15° NTM control

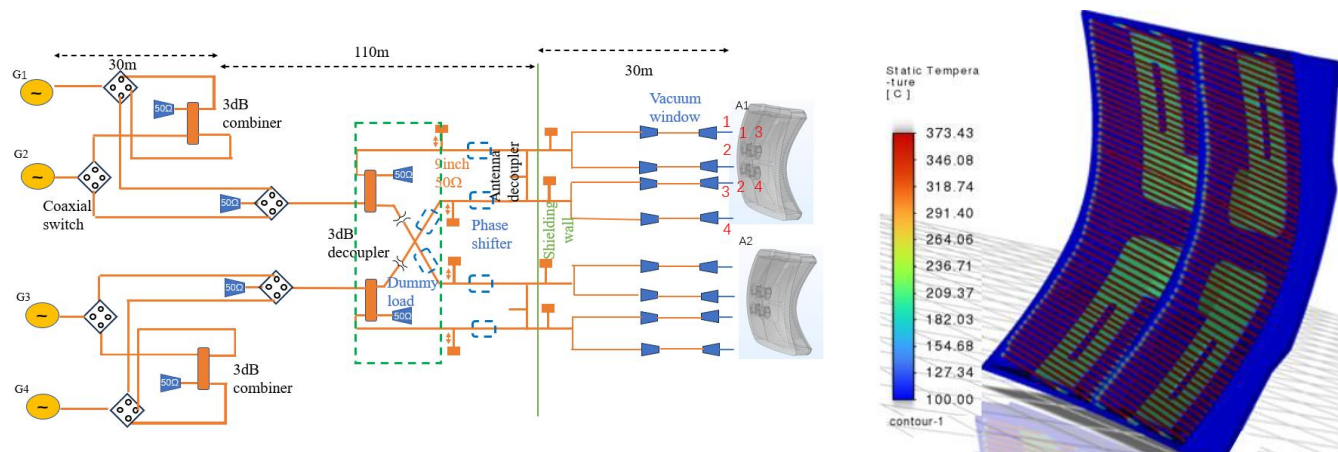
UL: for CTS

LHCD

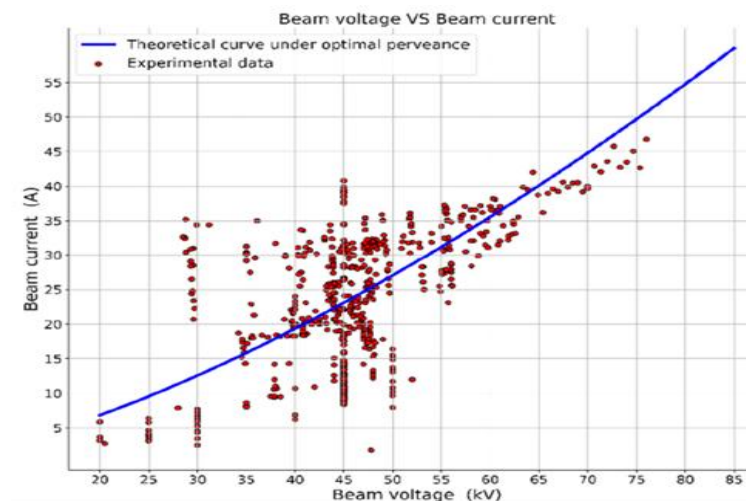


TE01 circle WG technology

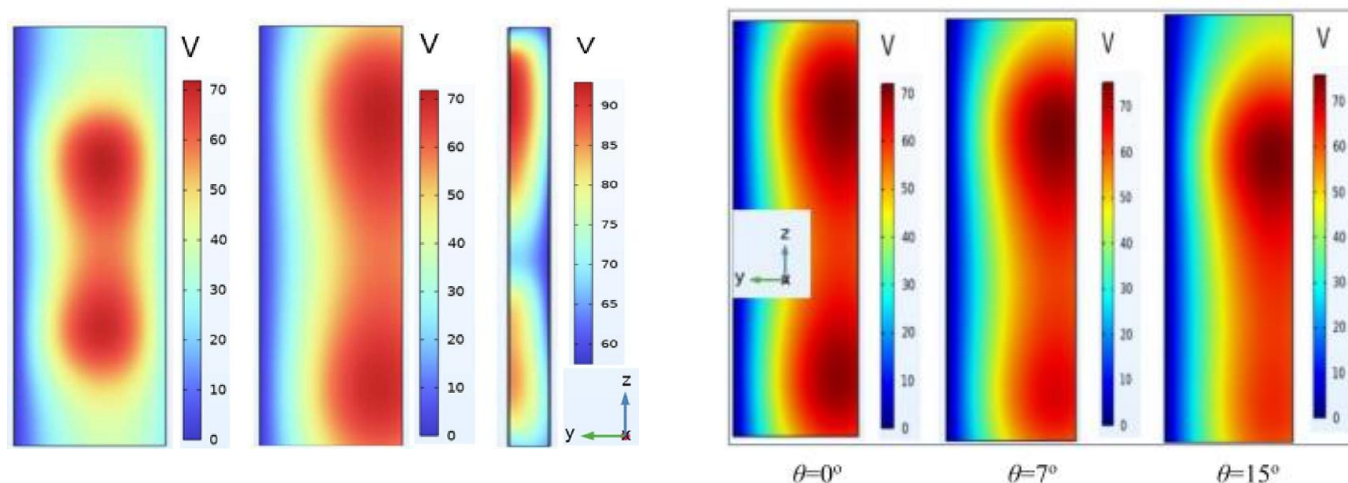
Highlights on H&CD design& experiment



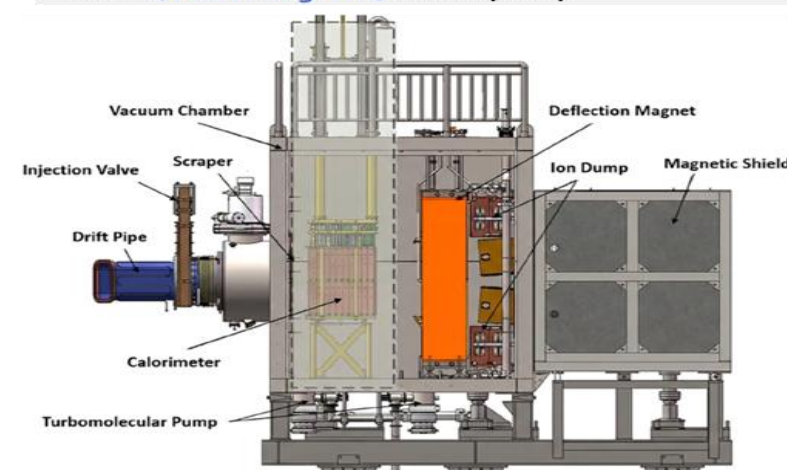
ICRF system for the HL-3 tokamak,
L. F. Lu, *Nucl. Fusion*, 10.1088 (2026)



Development and operational progress of ionsource for
the second neutral beam injector on HL-3 tokamak,
H.Y.Luo, *Fusion Eng. Des.*, 115823 (2026)



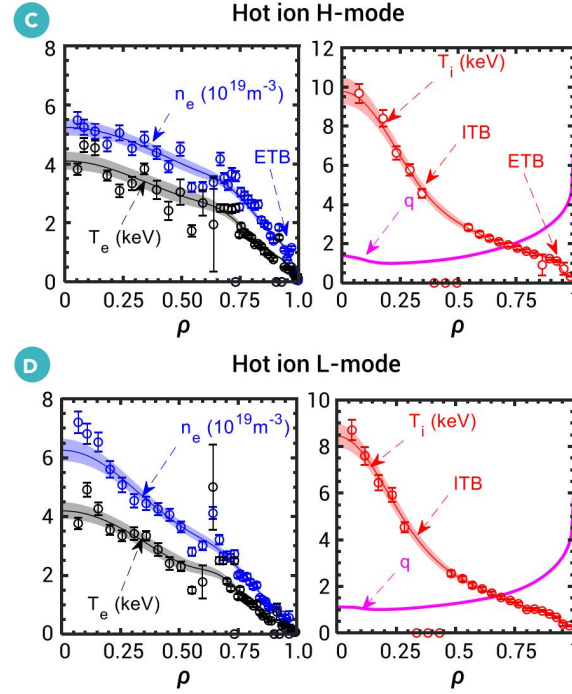
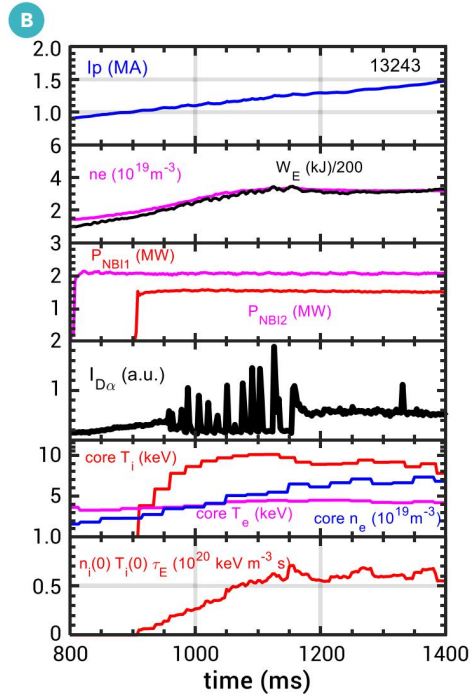
Self-consistent modelling of radio frequency sheath in 3D with realistic
ICRF antennas,
L. F. Lu, *Nucl. Fusion*, 126013(2024)



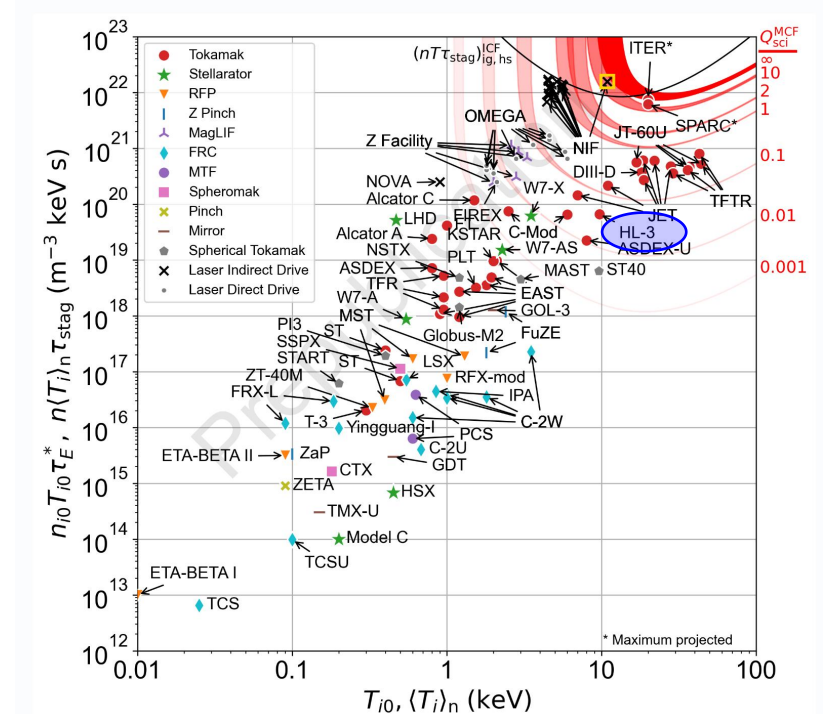
Development of the calorimetry target on second neutral
beam injection beam line of HL-3 device, B.W.Zhou, *Nucl.
Tech.* 050012 (2024)

High performance operation aided by NBI

- Early NBI injection during the current ramp optimizes the q profile for ITB formation while facilitating H-mode access, enabling integrated high-performance operation.



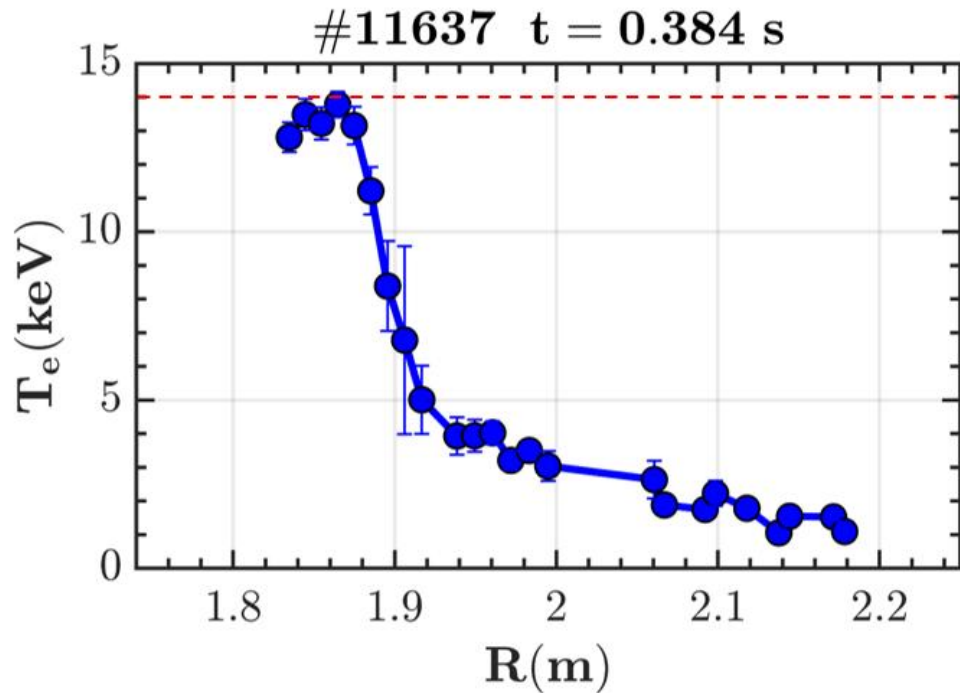
W.L. Zhong, The innovation. 7 (2026) 101167



- Early NBI modifies the q profile and promotes strong ion ITB formation.
- High auxiliary heating power enables H-mode access and formation of ETB.
- Simultaneous achievement of $T_i > 10$ keV, and a triple product of $0.67 \times 10^{20} \text{ m}^{-3} \cdot \text{s} \cdot \text{keV}$.

eITB Control via Optimized Off-axis ECRH

- Systematic optimization of EC power and deposition location established a reliable operating window for reproducible eITB formation.

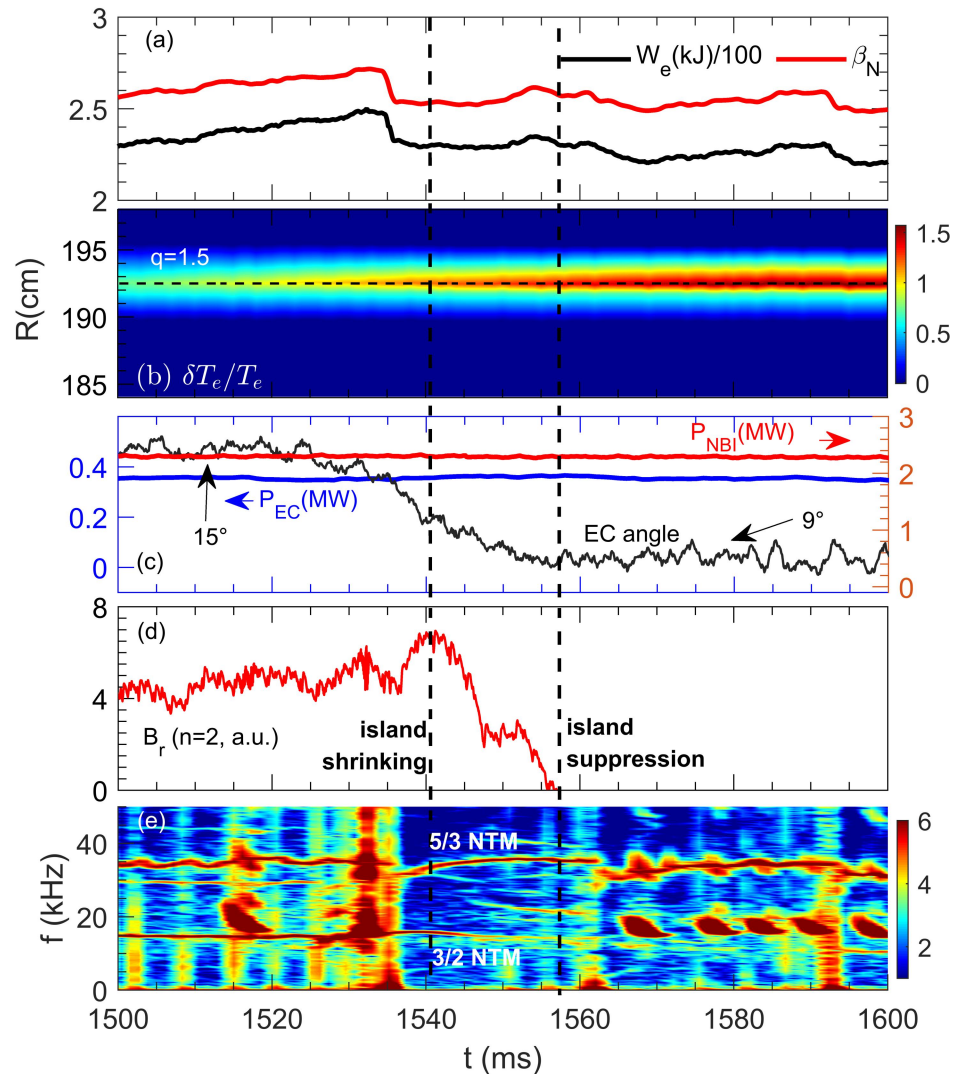


Shot	I_p (MA)	P_{EC} (MW)	EC Configuration	Peak T_e (keV)
11621	0.6	1.2	Off-axis (+16°) ECCD	9–13
11622	0.6	1.5	Off-axis (+16°) ECCD	~12
11626	0.6	1.7–1.5	Off-axis (+16°) ECCD	12–13
11631	1.0	1.4	Off-axis (+16°) ECCD	11–14
11632	0.6	~1.65	Off-axis (–16°) ECCD	14
11637	0.6	2.5 → 2.1	Off-axis (–16°) ECCD	~14

- Optimized ECRH coupling significantly improved electron heating efficiency.
- Stable and reproducible eITBs were established using off-axis ECRH.
- Peak electron temperatures over 10 keV were achieved with a plasma current range of 0.6 - 1.0 MA

NTM feedback control by ECCD

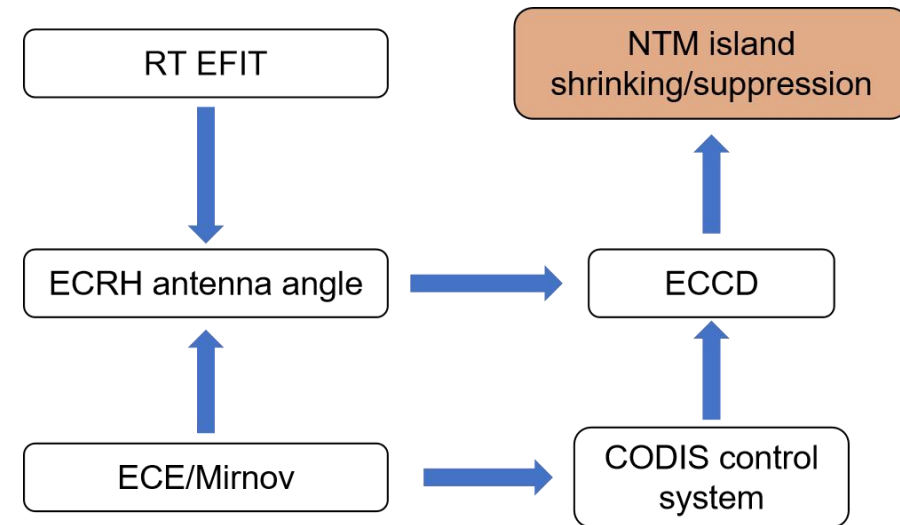
Real-time neoclassical tearing mode (NTM) feedback control by ECCD



NTM identified
by ECE

ECCD injected with
particular angle

NTM suppressed

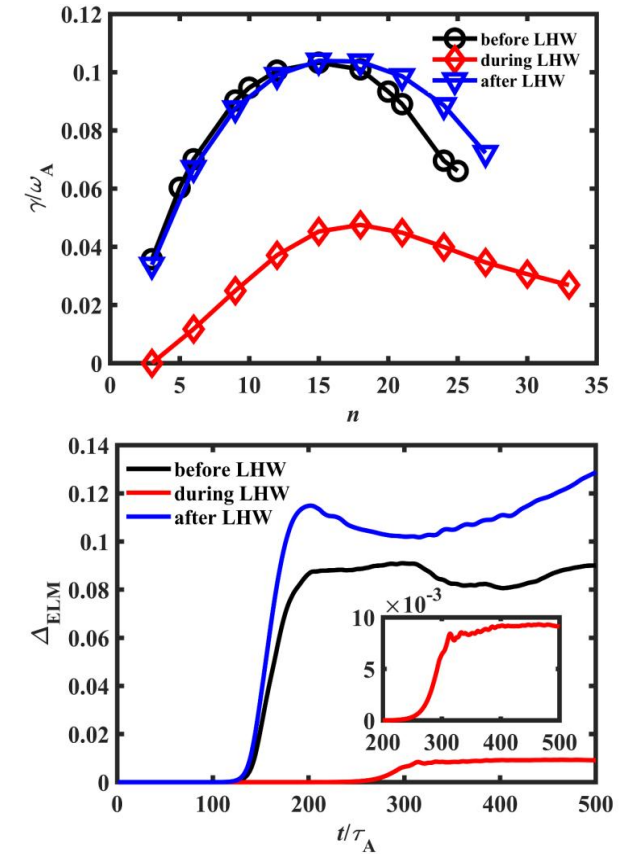
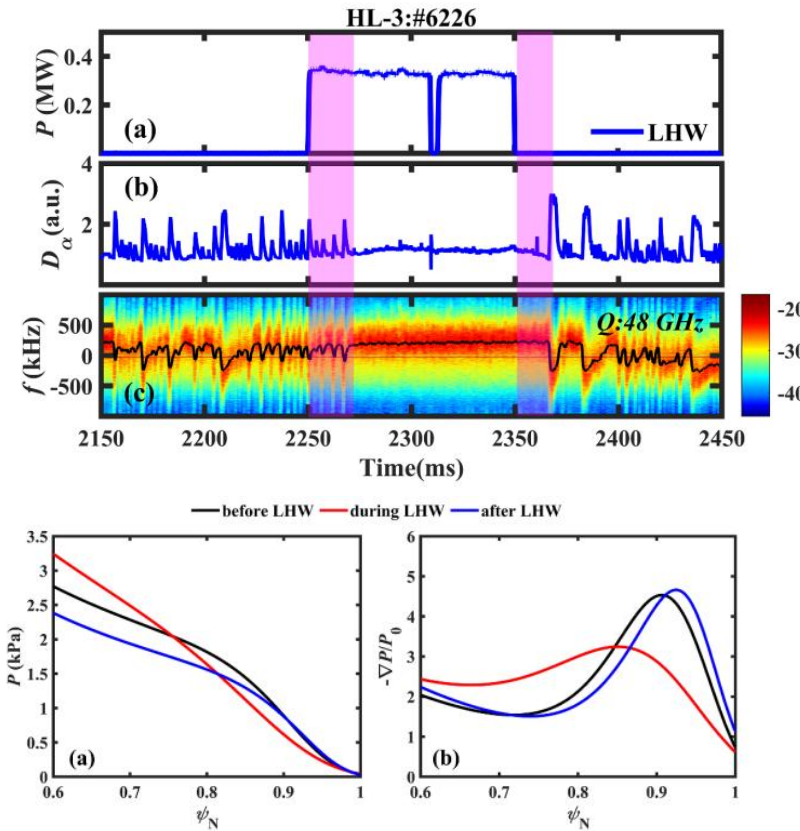
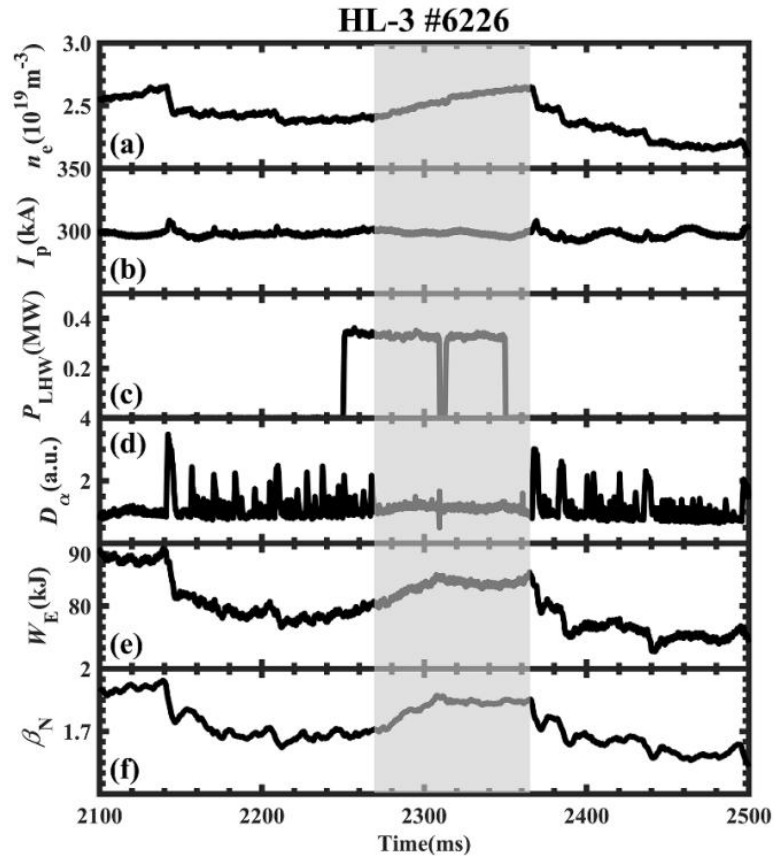


- Optimal EC antenna angle scanned
- Real-time NTM identification and EC injection achieved, 3/2 NTM island shrank and finally suppressed when EC deposited near q=1.5 surface
- Next step: Real-time angle control to suppress multi-modes.

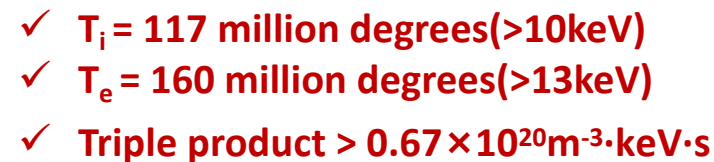
ELM suppression by LHW Injection

- LHW suppresses ELMs by enhancing pedestal particle transport and broadening the pedestal, reducing pressure gradient and peeling-ballooning instability.

Y.M. Zhang, Nucl. Fusion 65 (2025) 096019



LHW injection $\rightarrow$ enhanced pedestal turbulent transport $\rightarrow$ broader pedestal $\rightarrow$ decreased pressure gradient $\rightarrow$ reduced PB instability $\rightarrow$ ELM suppression



- ✓ 1.5 MA H-mode Operation
- ✓ H-mode in Tripod Divertor Configuration
- ✓ High Density Operation ($\sim 1.2 \times n_{GW}$)

- ✓ **1MA H-mode Operation**
- ✓ **Advanced Divertor (Snowflake & Tripod) Operation**



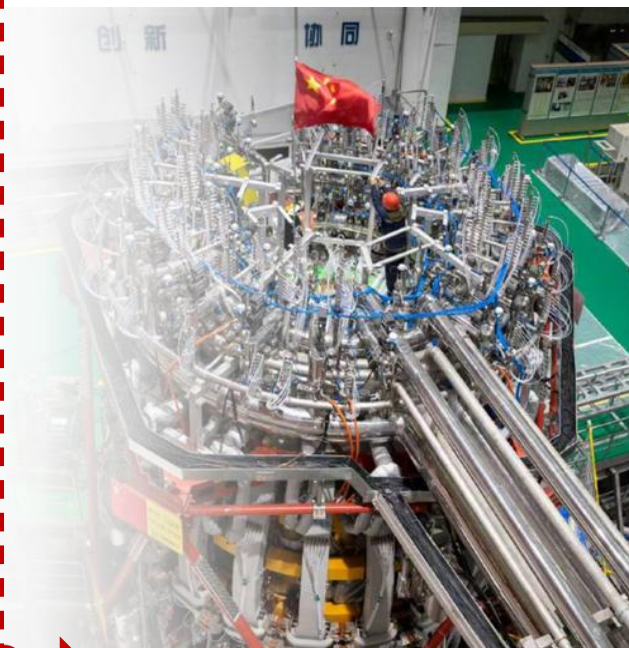
2020

2022

2023

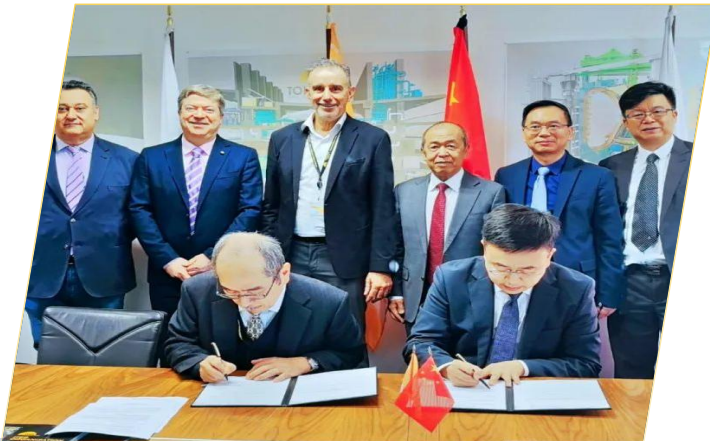
2024

2025



HL-3 Global Opening

In 2023, HL-3 opened as ITER satellite device to support ITER physics research and operation
In 2024, HL-3 was announced as facility opened to the world for international cooperation



- ◆ SWIP and IO signed agreement for deepened cooperation and HL-3 opened as ITER satellite device (2023, IO)



- ◆ The 68th IAEA annual general conference side meeting (Sept., 2024, Vienna)

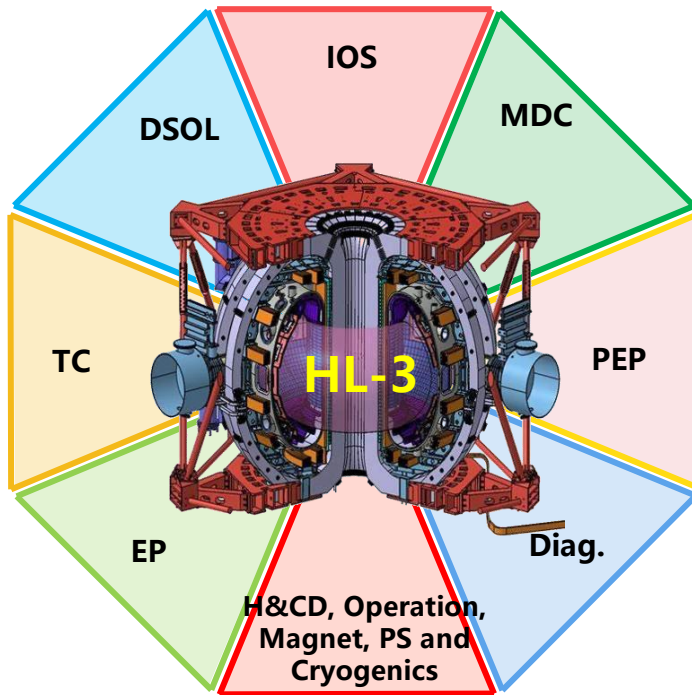
- ◆ The 17th China International Nuclear Industry Exhibition (Mar., 2024, Beijing)



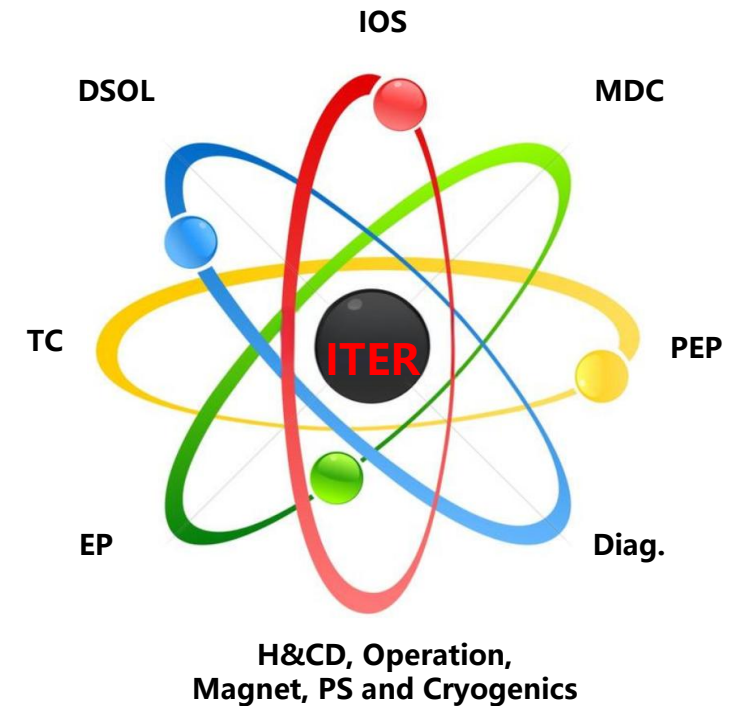
- More than **80 scientists** from **ITER**, Japan, Korea, France, Portugal, US, etc., participated
- Focus on **high beta plasmas, advanced divertor physics, ETB/ITB regimes, disruption physics**, etc.



Taskforces were set up in accordance with ITPEA Topical Groups



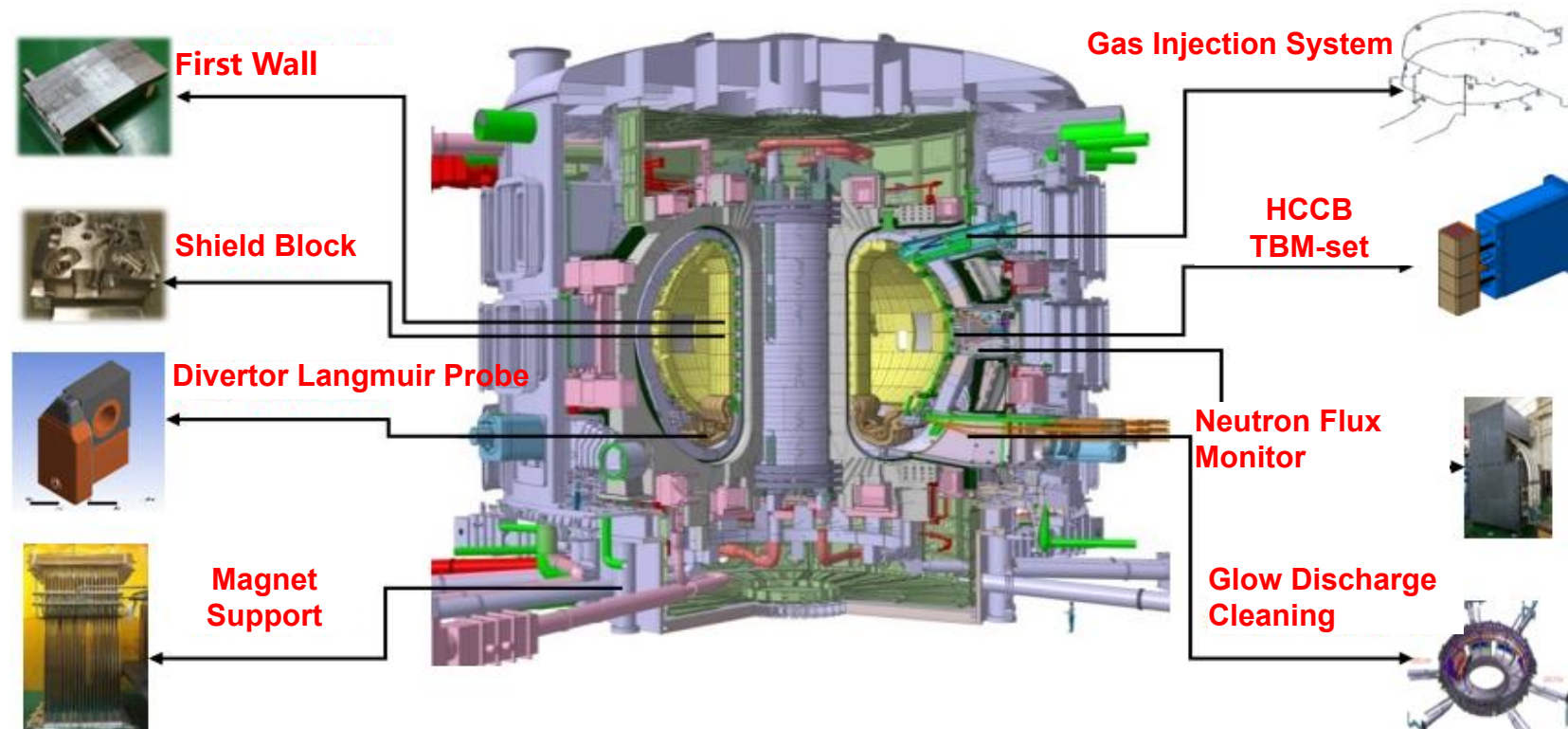
Corresponding Tasks
Synchronized Research
International Collaboration
Joint Experiments



03

ITER-Related Projects

SWIP is responsible for 9 PAs of ITER Key Components,
including TBM / First Wall / Shielding Block / GIS & GDC / Magnet Support /
Neutron Flux Monitor / Divertor Langmuir Probe...



Enhanced Heat Flux (EHF) First Wall (FW)

Technical development and qualification programme of Enhanced Heat Flux (EHF) First Wall (FW), including small mock-ups, semi-prototype to full-scale prototype (FSP)

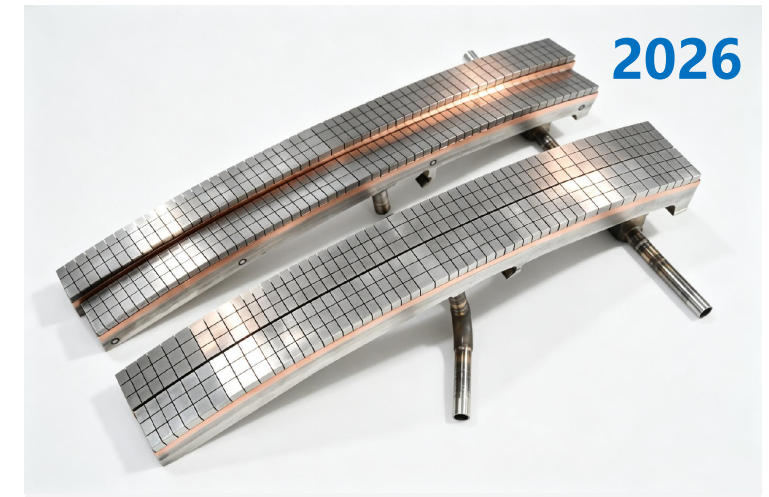
- Full scale prototype (Beryllium armor) developed, qualification passed in 2023
- Serial Production (57) Contract of W-armor FW signed in Dec. 2025
- Manufacturing Process Qualification approved in June 2026



1st Prototype of FW (beryllium armor)



1st Prototype of FW (tungsten armour)



Qualification of Tungsten Armour Bonding technology completed

Magnet Support (MS)

Manufacturing of **ALL** ITER Magnet Supports, including 18 sets of Gravity Supports (GS), 108 sets of Poloidal Field Coil Supports (PFCS), and 6 sets of Correction Field Coil Supports (CCS)

- ITER' s key structural safety component withstanding 20,000 tons of complex loads
- First key components delivered by China, >1600 tons
- All products delivered and contract to be closed this year



Gravity Support



All Products delivered in Nov. 2023



PFCS Assembly

Technical support to all the 220 SBs

- 1st ISO standard in fusion R&D, 'hot helium leak testing method' in 2023
- First batch of 48 blocks delivered in 2024 and the rest will be done in 2027
- 2 hot helium leak test facilities developed for test of the blocks

INTERNATIONAL
STANDARD

ISO
4233

Reactor technology — Nuclear fusion reactors — Hot helium leak testing method for high temperature pressure-bearing components in nuclear fusion reactors

Technologie du réacteur — Réacteurs à fusion nucléaire — Méthode de contrôle d'étanchéité par détection de fuite d'hélium à chaud pour les composants sous pression à haute température de réacteurs à fusion nucléaire



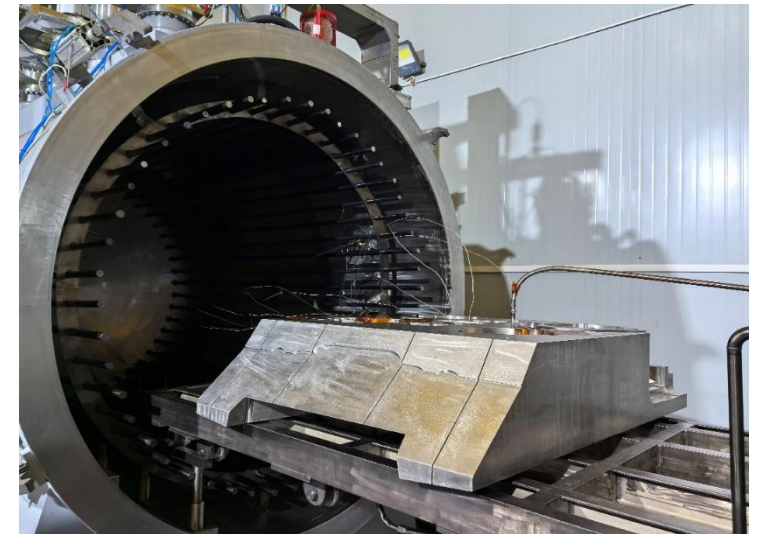
Reference number
ISO 4233:2023(E)

© ISO 2023

1st fusion ISO international standard, 2023



1st batch delivered in 2024



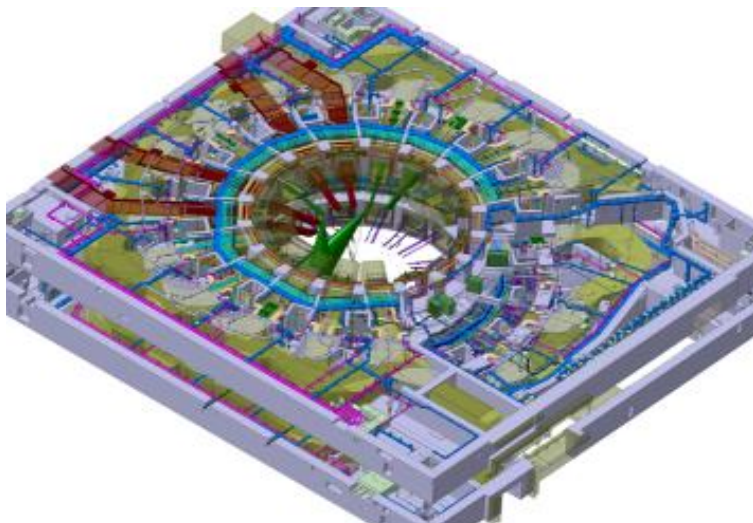
Ready for Hot helium leak test

Gas Injection System (GIS): Design of the whole ITER GIS system and the manufacture of the ITER GIS except GDS Manifold

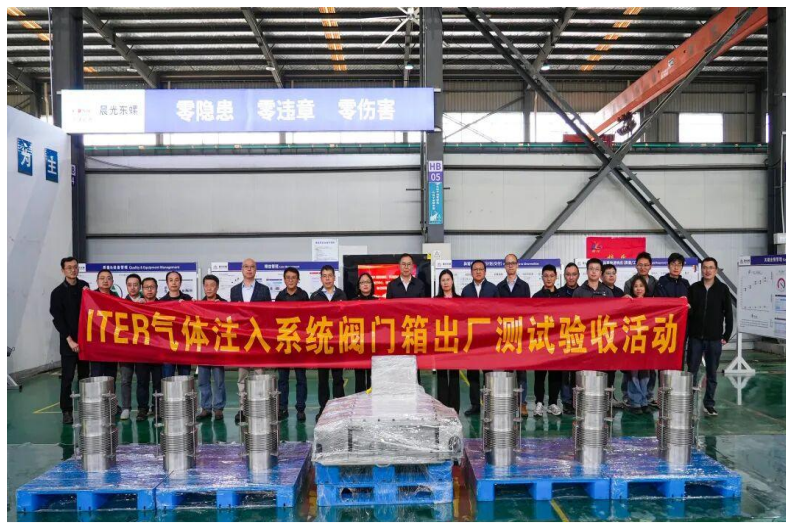
- First batch of GF GVBs (4 sets) delivered to ITER in May, 2026

Glow Discharge Cleaning (GDC): All 11 glow discharge cleaning electrodes for ITER

- GDC PD Contract passed the acceptance of CNDA in Dec. 2025



Distribution of ITER GIS



First batch of GF GVBs (4 sets) delivered to ITER



GDC PD Contract passed the acceptance

Divertor Langmuir Probe (DLP):

R&D of all 400 sets of Langmuir Probe systems

- FDR completed in Dec. 2025
- 100% pass rate of high heat load test

Neutron Flux Monitor (NFM):

4 sets of NFM systems in Equatorial Port 1#, 7#, 8#, and 17#

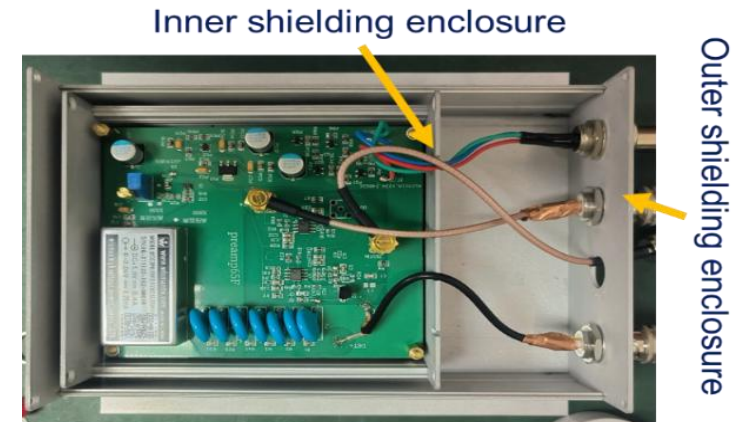
- Double-layer shielded preamplifier developed with excellent performance



DLP under high heat load test



NFM7# support Frame installed at ITER in 2020

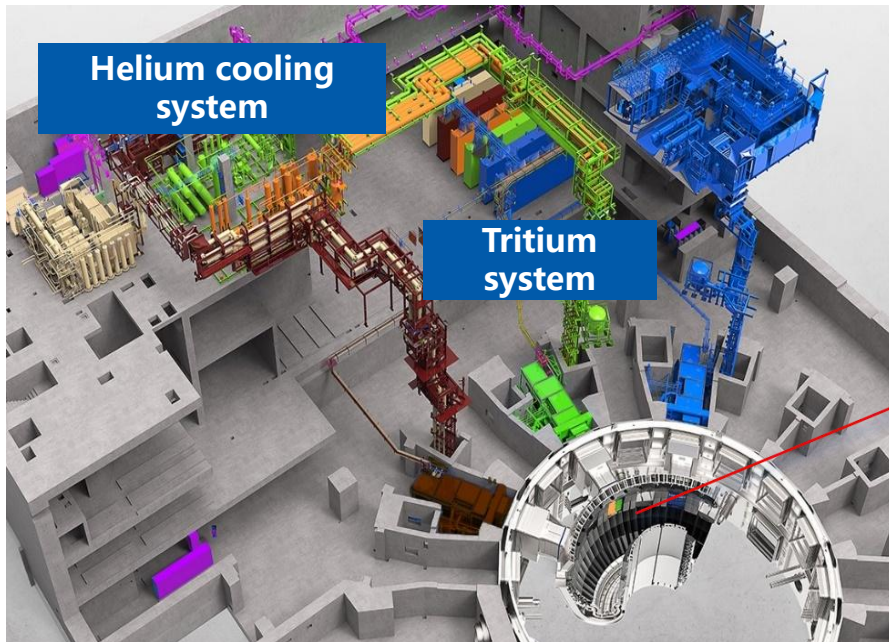


double-layer shielded preamplifier

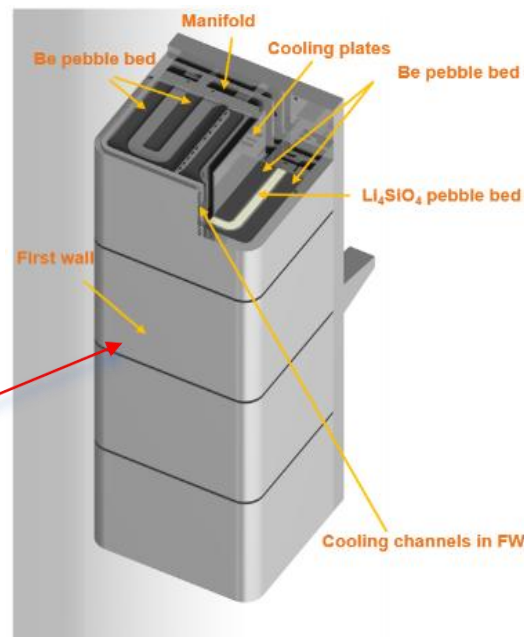
Helium-Cooled Ceramic Breeder Test Blanket System (HCCB-TBS)

One of **major engineering goals of ITER** is to test and verify the **tritium breeding** and energy extraction technologies of fusion reactors

- Completed the first TBS conceptual & preliminary design
- Developed the first semi-prototype module
- First TBM Shield Block developed in 2025; HCCB-TBS PDR held in July 2026



Test Blanket System



Test blanket module



TBM Shield Block



Semi-Prototype
HCCB TBM Module

SWIP is one of the China Nuclear Power Engineering (CNPE) consortium members undertaking the ITER Assembly Contracts

2019

TAC1



2024

SMSA



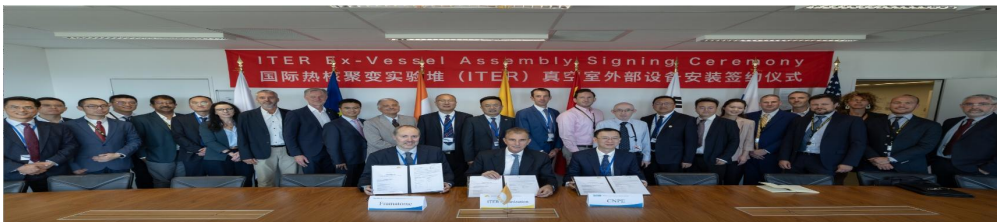
2024

SMPA



2025

WPP



2026

EVA



Acceptance, inspection, assembly, packaging, transportation and on-site testing of **58 divertors** and on-site acceptance of **205 shielding modules**

- Divertor components received from JADA
- IO, F4E, RFDA, and JADA witnessed the unpacking and inspection of Prototype Divertor components in May
- Procedures and specialized tooling for the SB Factory Acceptance Test prepared



The series OVT from JADA arrived at Integration Site in May



The unpacking and inspection of prototype components witnessed by IO delegation



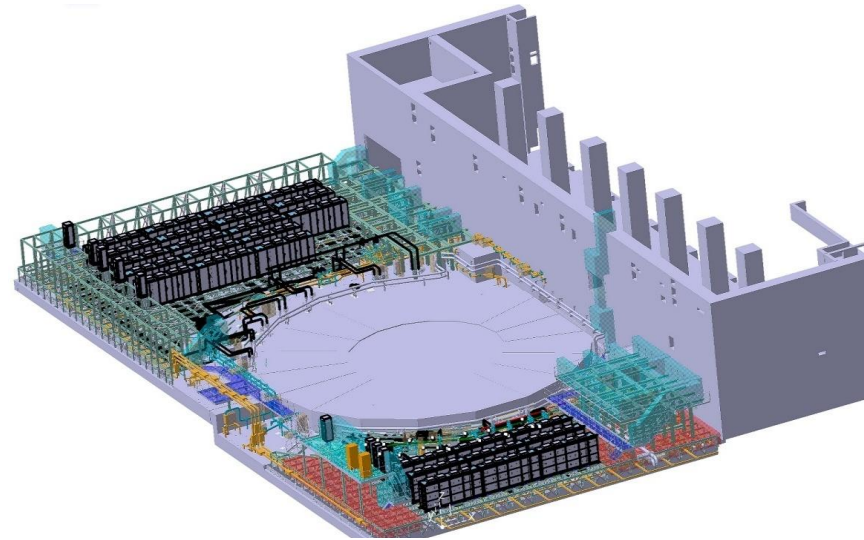
Special tools for shielding modules acceptance

ITER ELM Power Supply Contract

Consortium led by SWIP and joint by top Chinese institutions and enterprises

27 sets of adjustable output power supplies for 3 independent ITER in-vessel coils

- **Contact signed on Jan.22**
- **Preliminary design and preparation of Static Magnetic Fields (SMF) Qualification Program is on-going**

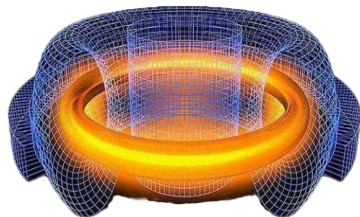
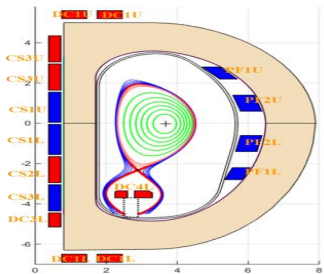


04

Fusion Technologies

Developed kinds of technology to support the fusion research

- Plasma theoretical research and simulation
 - Diagnostics
 - Heating (NBI, ECRH)
 - Power supply
 - Control
 - Vacuum
 -
- High temperature superconductor
 - Materials
 - FW and blanket fabrication
 - Welding and UDT
 - Remote handling
 - Testing platforms
 -

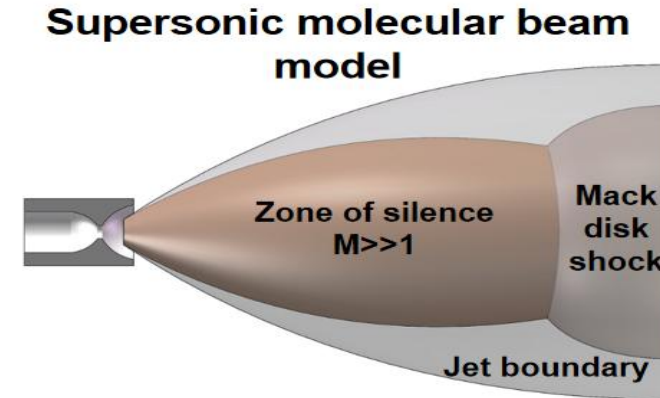


Tri-Band High-resolution Spectrometer



- Integrated and compact structure
- Simultaneously measuring the profiles of temperature, rotation speed and concentration of carbon and helium ions

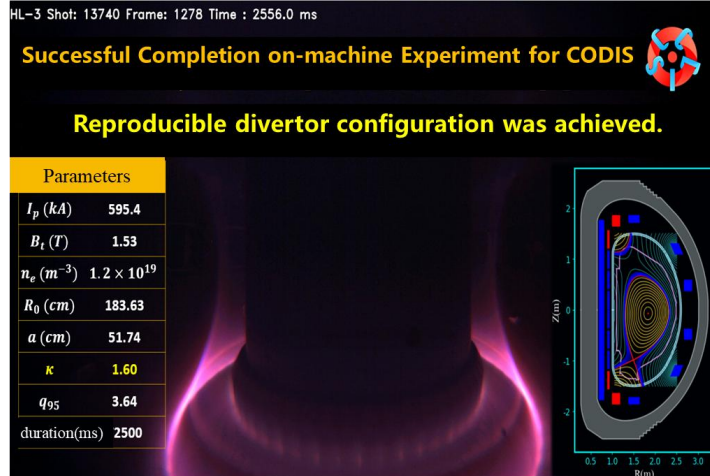
Supersonic Molecular Beam Injection



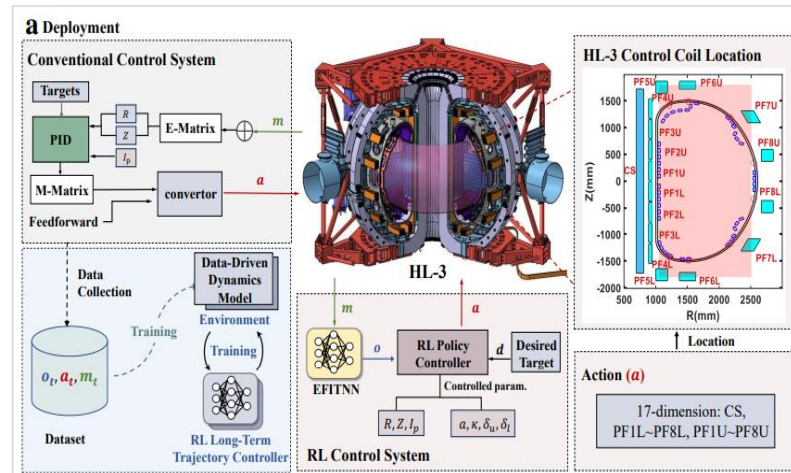
- Higher fuelling efficiency, lower neutral recycling
- IOS standard approved
- Injection speed > 10 Mach
- plasma fueling and control

Plasma Control & AI Application

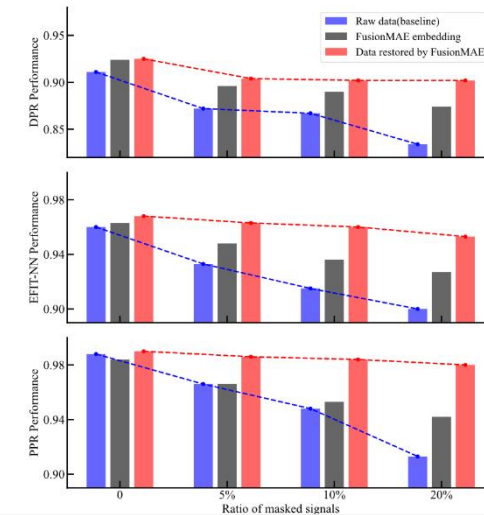
- **The Fusion Device Operating System (CODIS)**, integrating control, operation, data, and intelligence, successfully developed and validated on HL-3
- Undertook the IAEA Coordinated Research Project F13022 (**AI for Accelerating Fusion R&D**)
- **Two key AI capabilities demonstrated**: reinforced learning magnetic control and the large-scale pre-trained model **FusionMAE**



CODIS validation in HL-3 experiment



Diagnostic model, FusionMAE, took over control of magnetic configuration



FusionMAE maintains downstream task performance with missing diagnostics

R&D Platforms for Blanket and Materials



**Helium Cooling
Experimental Loop**



**Liquid Metal
Integrated Research Platform**



Hot Helium Leak Testing System



**CNAS-approved Fusion Material
Testing Center**



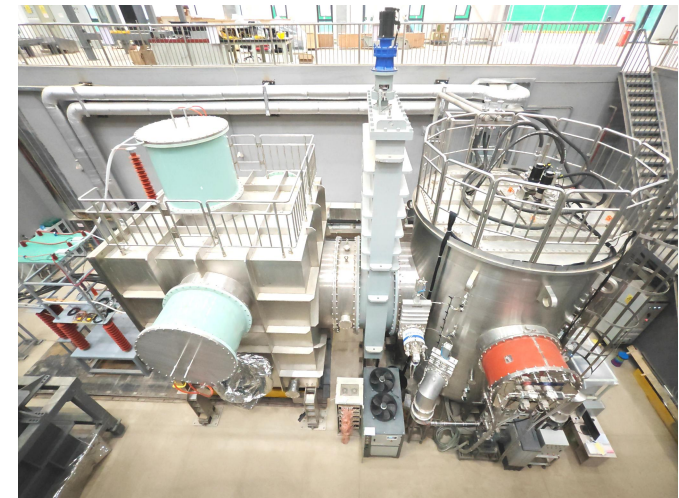
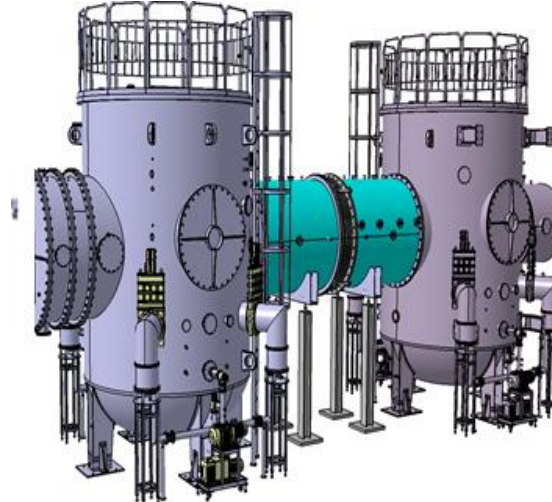
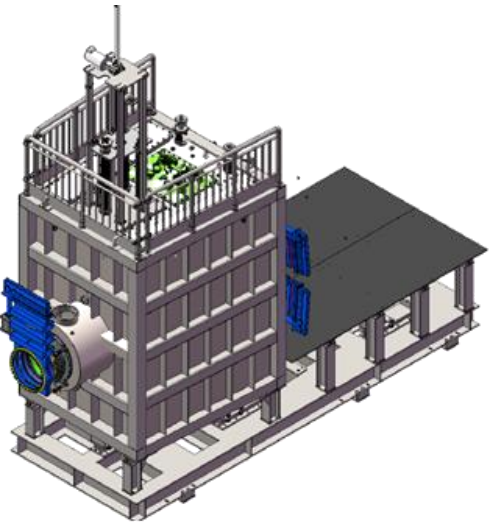
**ITER-approved
High Heat Load Testing platform**



Neutron Activation System

NBI Platforms for R&D

- Full-sized ion source commissioning platform and RF negative ion source platform have been constructed in Tianfu fusion R&D site
- Self-developed 7MW NBI beamline, NBI power highest in China
- RF negative ion source: Achieving pulse width > 5000s, Max current Density 317A/m^2 , Max beam Energy 202keV
 - Full-sized commissioning platform
(Filament positive ion source $120\text{ keV} / 80\text{A}$)
 - RF negative ion source platform
(RF negative source $400\text{ keV} / 20\text{A}$)



■ ECRH R&D platform

- Self-developed 105GHz gyrotron:
515kW / 1.02s
- Multi-frequency, 1000-s, MW gyrotron
and transmission line (>100m) test
platform



High-Power Long-Pulse 100-Meter-Class Transmission Line System Test Platform



UHV Exhaust Station



Welding Process Platform



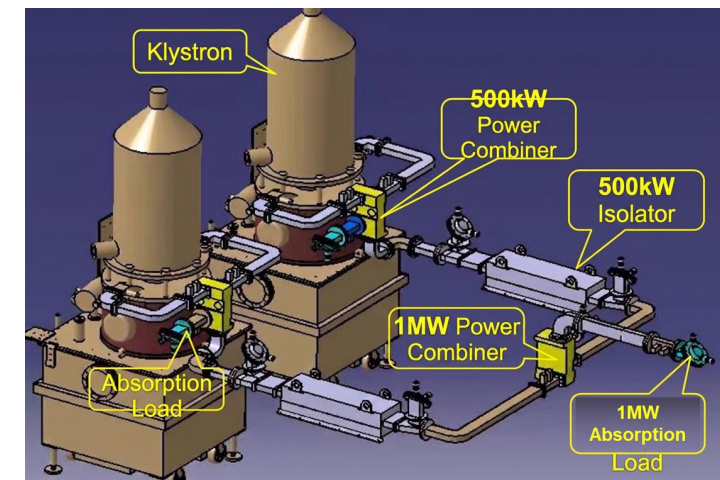
Wave Source Test Platform

■ LHCD R&D platform

- 4MW/3.7GHz high-power dual-tube synthesis
test platform
- Localized production and validation of high-
power klystrons
- TE01 circle WG technology



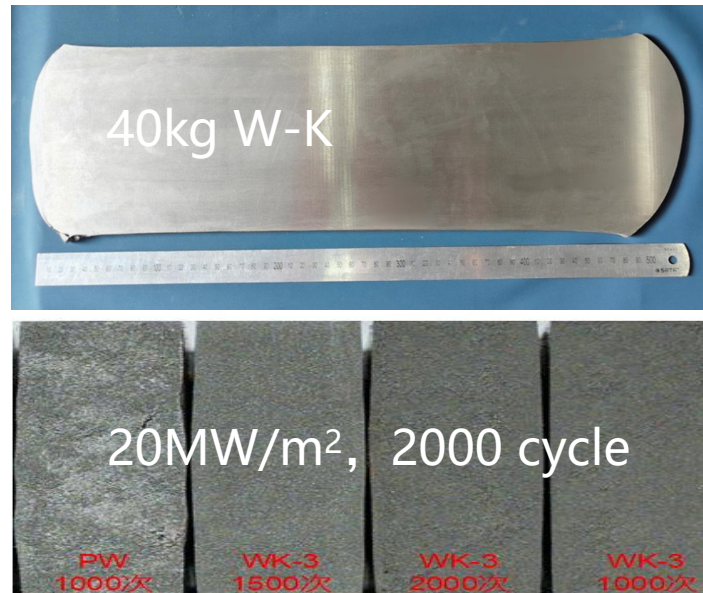
LHCD dual-tube synthesis test platform



- **Reduced Activation Ferritic Martensitic (RAFM) steel - CLF-1**
Large-scale ingot: up to **13 ton**.
- **Plasma facing materials**
Advanced **W-K** alloy (**40kg**), The high flux performance tolerate **20MW/m²**, 2000 cycle.
- **Functional Materials (beryllium pebble & Li₄SiO₄ pebble)**
Completed **engineering qualification**, Li₄SiO₄ production capacity **≥10kg/batch & >2 t/y**.



13 ton-scale CLF-1 steel ingot



W-K alloys (40kg scale) and mock up



MSM Facility for Li₄SiO₄

The production line for HTS stacked armored conductors has been completed

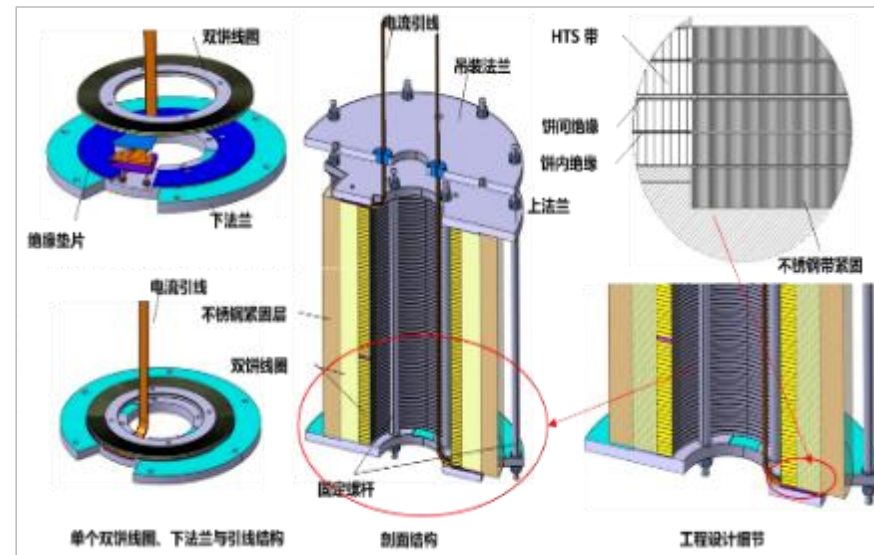
The 300mm large-diameter, 20T HTS solenoid magnet is under development



HTS Cable



HTC Conductor



20T large-bore HTS solenoid magnet

Test facilities for high-field magnets research

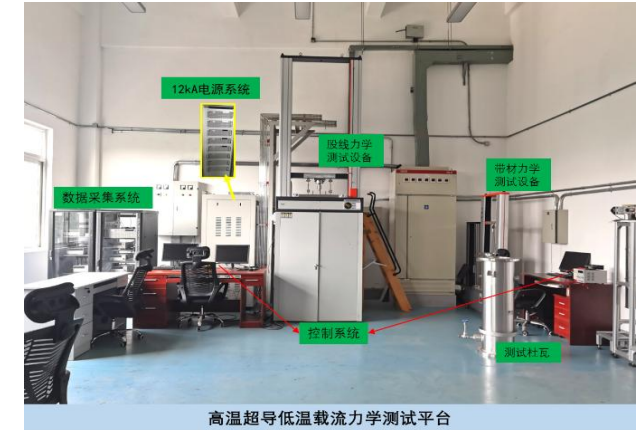
- ✓ Testing facilities for HTS strand and large-scale conductor — critical current, electro-mechanical and high field performance (4.2K-77K, 0-12T, 0-100kN, and 0-40kA)



12T high field magnet (4.2K-77K, 12kA/40kA)



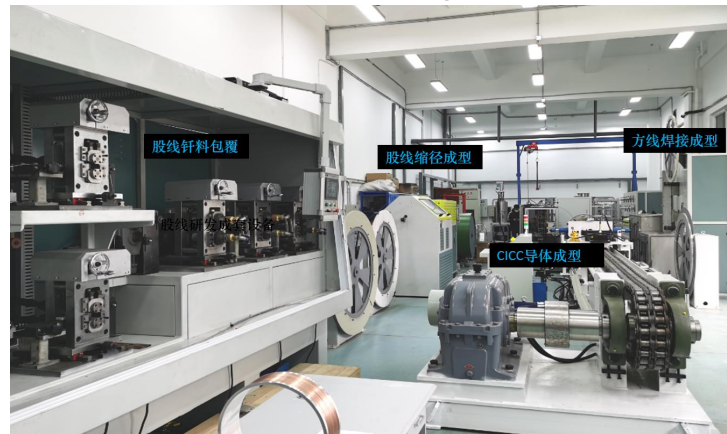
Tensile testing machine



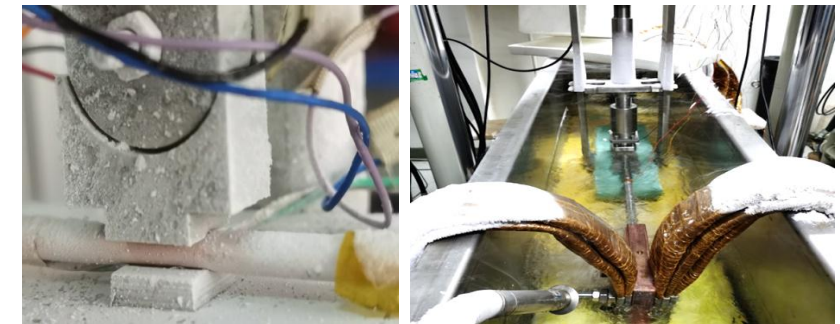
Tensile and twist testing machine



Fatigue testing machine

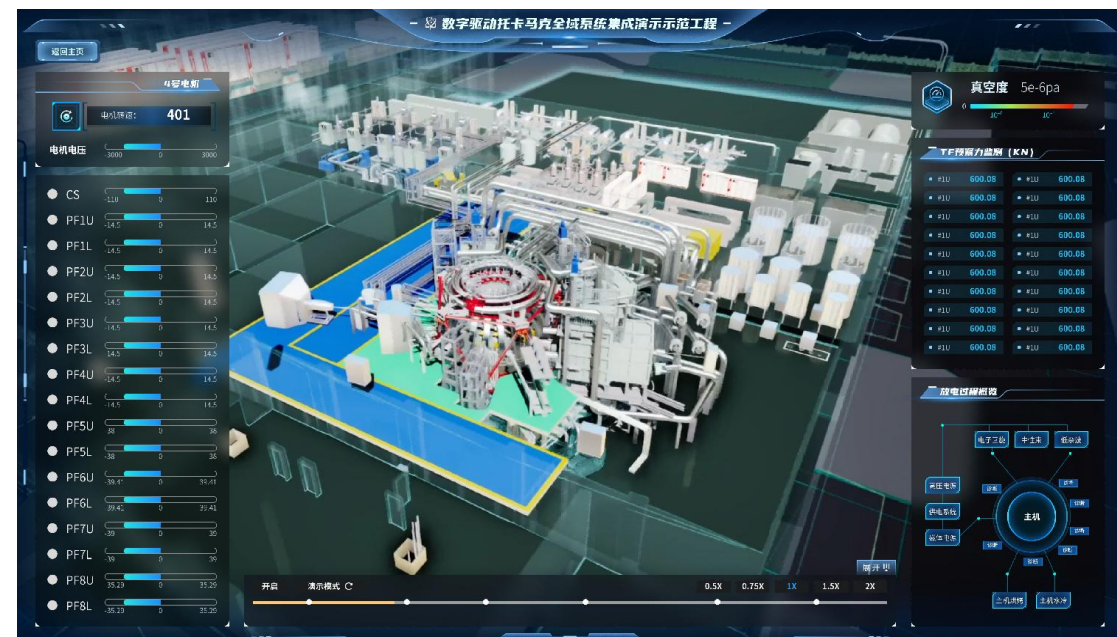
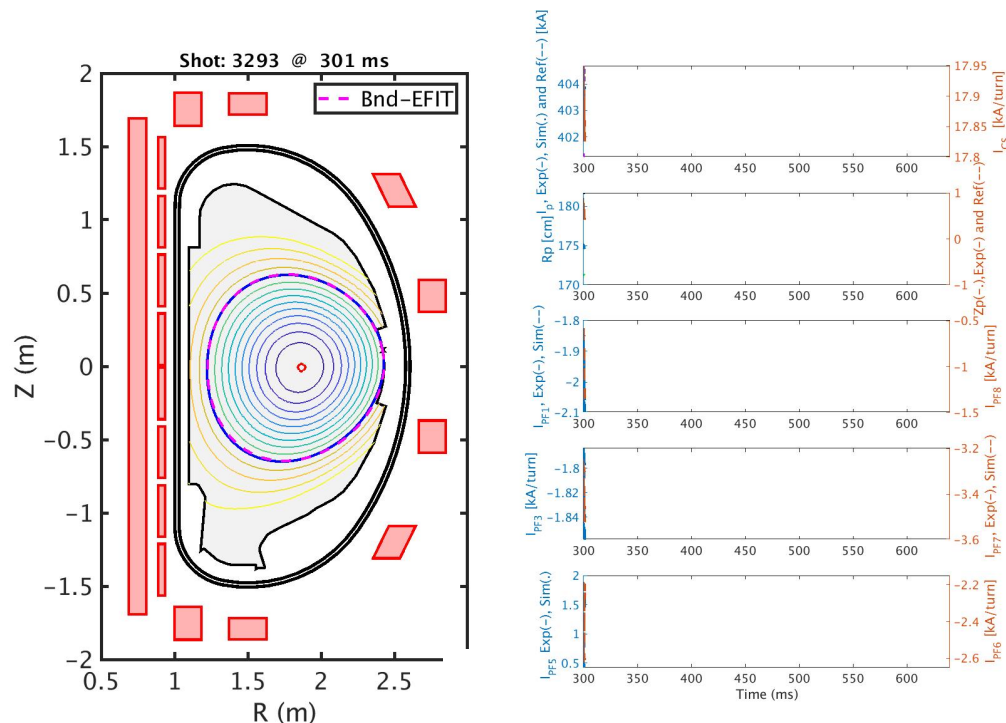


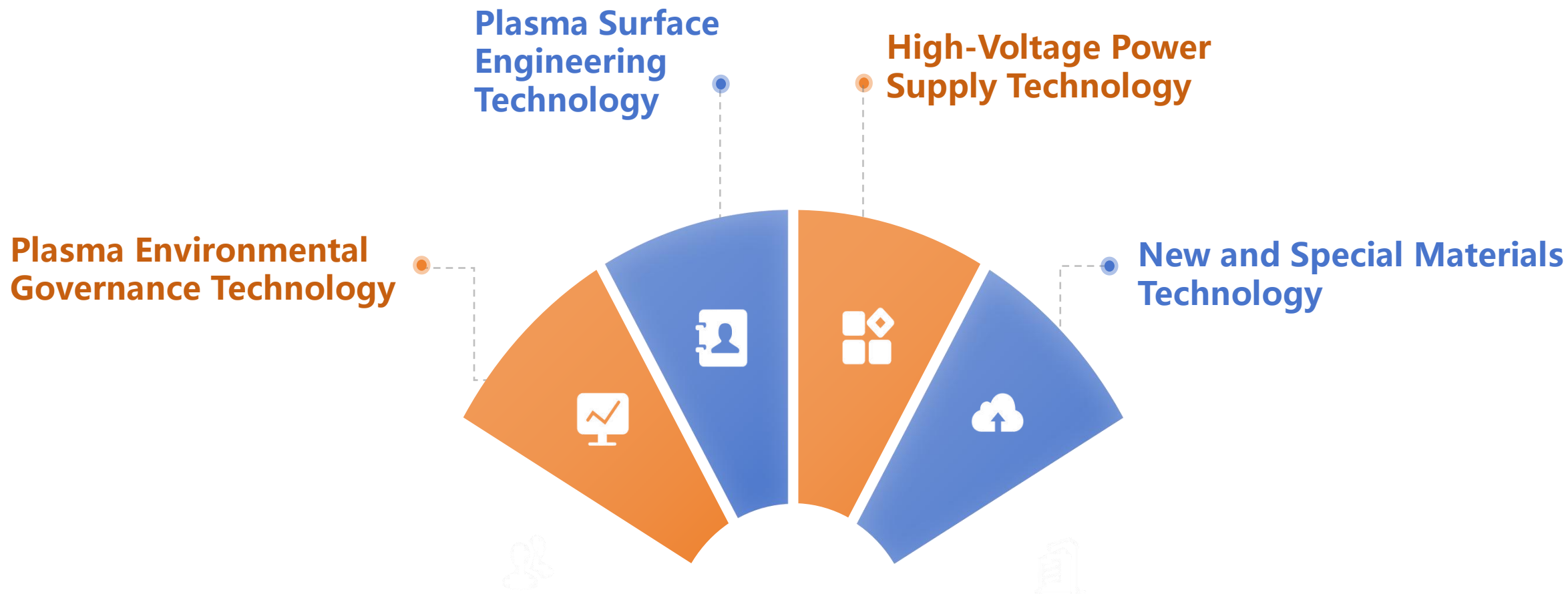
Conductor manufacturing equipment



Electro-mechanical tests of cable strand and CICC cable

- ❑ Hardware-in-the-loop testing prototype developed. Introduce **real-data disturbance & delay from hardware**. Test & optimize plasma control algorithm **without real plasma**, decreasing VDE risk.
- ❑ Integrate experimental data to achieve digital visualization of multi-system status. Lay the foundation for further research on digital twins.





Plasma Environmental Governance Technology

Long Yan-1: China's First domestic plasma high-temperature melting treatment equipment for low-level radioactive waste from nuclear power plants (applied in nuclear power plants in Hainan Province)



Mobile modular waste discharge treatment device for on-site oxidative melting treatment of radioactive combustible and non-combustible solid wastes



Plasma Surface Engineering Technology

Plasma Surface Engineering Technology for **surface metallization of non-metallic materials** and **surface strengthening of metallic materials**. This coating technology can significantly improved tool performance in automotive, machine tools, general machinery, aerospace, among others



Plasma coating equipment developed by SWIP

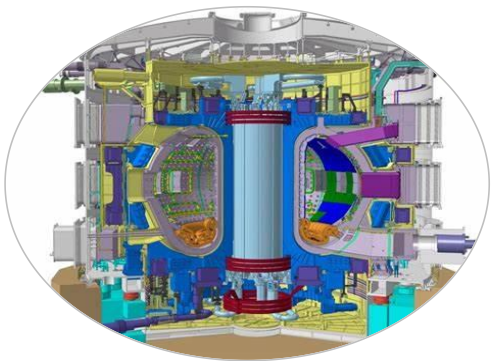
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Prospects of Fusion Research

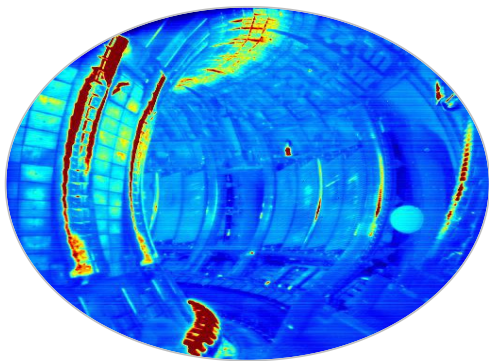
Major Scientific, Engineering and Technical Challenges



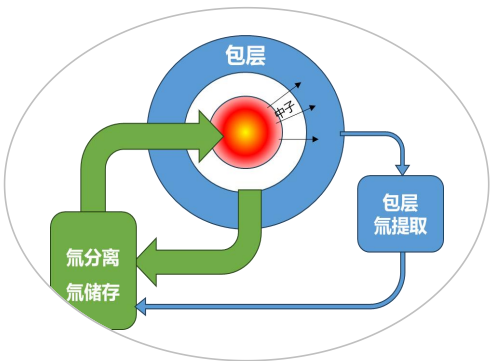
3 Major Scientific and Technological Challenges



Steady-state and self-sustained operation of burning plasma



Materials resistant to high-energy neutron and high heat load

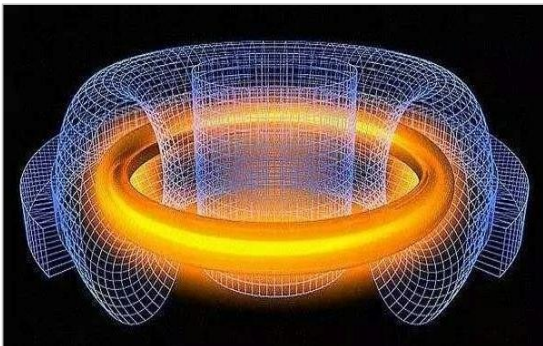


Tritium breeding

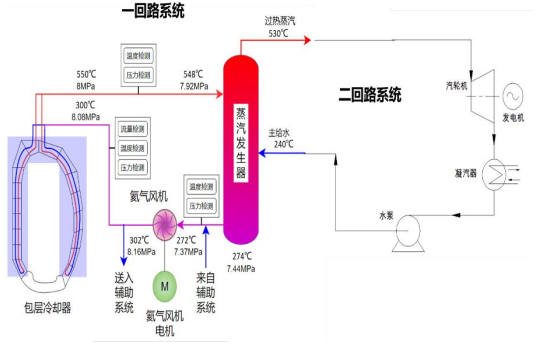
3 Major Engineering and Technical Challenges



High field HTS magnets



Plasma operation and control



Heat extraction

- 1. To accelerate fusion energy R&D, aiming for its earliest practical and economical application, is currently the core objective of the fusion community.**
- 2. Global fusion development benefits from ITER. SWIP will actively promote the construction and operation progress of ITER.**
- 3. Two new campuses of CFEC and SWIP in two cities will greatly support the R&D of Fusion technologies. The D-T experiments on HL-3 tokamak is the next key mission.**
- 4. The establishment of the Innovation Consortium and CFEC is of strategic significance in fully leveraging and utilizing the technological accumulations and advantages of China's industrial sector, including the experience accumulated by CNNC in nuclear fission energy R&D and application.**

Thanks for Your Attention!