Ioffe
Institute

"FEC-2023: статус исследований на установках магнитного удержания"

Курскиев Г.С.

Gleb.Kurskiev@mail.ioffe.ru

Содержание

- Обзорные доклады по секциям
  - Burning Plasmas and Long Pulse Operation: ***JET, NIF, EAST, KSTAR***
  - Tokamaks: ***DIII-D, AUG, TCV, HL-2M***
  - Stellarators, Spherical Tokamaks, Private Sector: ***W7X, LHD, MAST-U, ST40***
  - Technology, Long Pulse & Science: ***WEST, Globus-M2***

- Заключение

материалы конференции доступны здесь: https://conferences.iaea.org/event/316/contributions/
для входа на сайт нужно ввести пароль #Fusion23

# -	Burning Plasmas and Long Pulse Operation: JET DTE2, NIF Q>1, EAST, KSTAR

Burning Plasmas and Long Pulse Operation: JET DTE2

ISSN 0029-5515

IAEA

nuclear fusion

Volume 63 Number 11 November 2023

SPECIAL ISSUE

Special Issue of Papers Presenting Results from the JET Tritium and Deuterium/Tritium Campaign

Online: iopscience.org/nf

IOP Publishing

International Atomic Energy Agency, Vienna

Nuclear Fusion

Vol 63, No 11

110201–112015

November 2023

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Guest Editor

Costanza Maggi

Explore the research here

IOP Publishing

Special Issue of Papers Presenting Results from the JET Tritium and Deuterium-Tritium Campaign

https://iopscience.iop.org/issue/0029-5515/63/11

Burning Plasmas and Long Pulse Operation: JET DTE2

JET DTE2 scenarios for high, sustained P_{fus} with Be/W wall

2 paths developed (candidate scenarios for ITER)

Garzotti,
EX-C #1943

JET Baseline scenario (50-50 D-T)

High plasma current

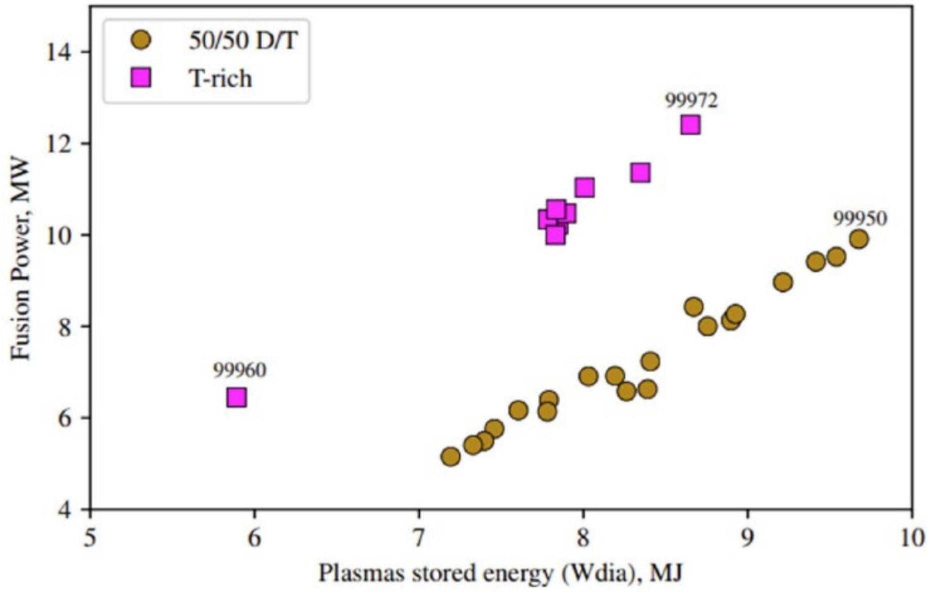
Maximizes thermonuclear neutrons in JET

JET Hybrid scenario (50-50 D-T)

High beta

Hobirk, EX-C #1880

Higher beam-target neutron fraction



Friday afternoon

JET-ILW specific:

JET Hybrid scenario with optimized non-thermal fusion (15-85 D-T)

Maximises beam-target fusion reactions in JET

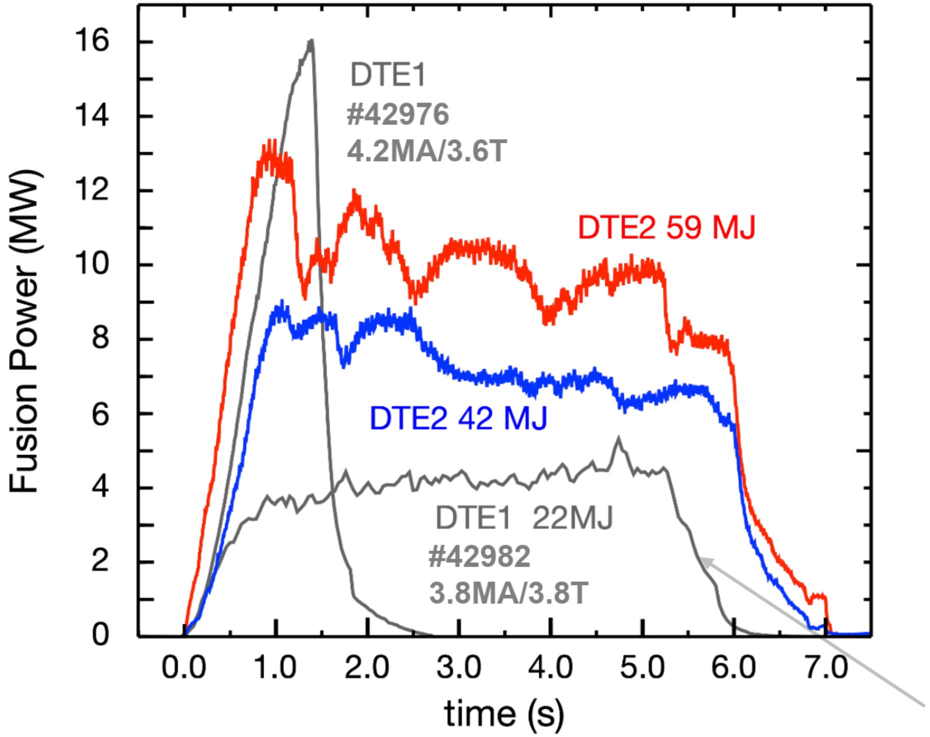
Maslov, EX-C # 1935

High-Z impurity control is essential to achieve sustained, high P_{fus}

Burning Plasmas and Long Pulse Operation: JET DTE2

World fusion energy records obtained with JET hybrid scenarios

Fusion performance with JET-ILW beyond that of DTE1



- Hybrid plasmas performed in D-T for the first time, with Be/W wall
- **Demonstrate compatibility of JET-ILW with sustained high fusion performance**

[Hobirk et al, NF SI on JET T & D-T 2023]

#99869 (2.3MA/3.45T) Hybrid with ~50-50 D-T

#99971 (2.5MA/3.86T) Hybrid with ~15-85 D-T

[Maslov et al, NF SI on JET T & D-T 2023]

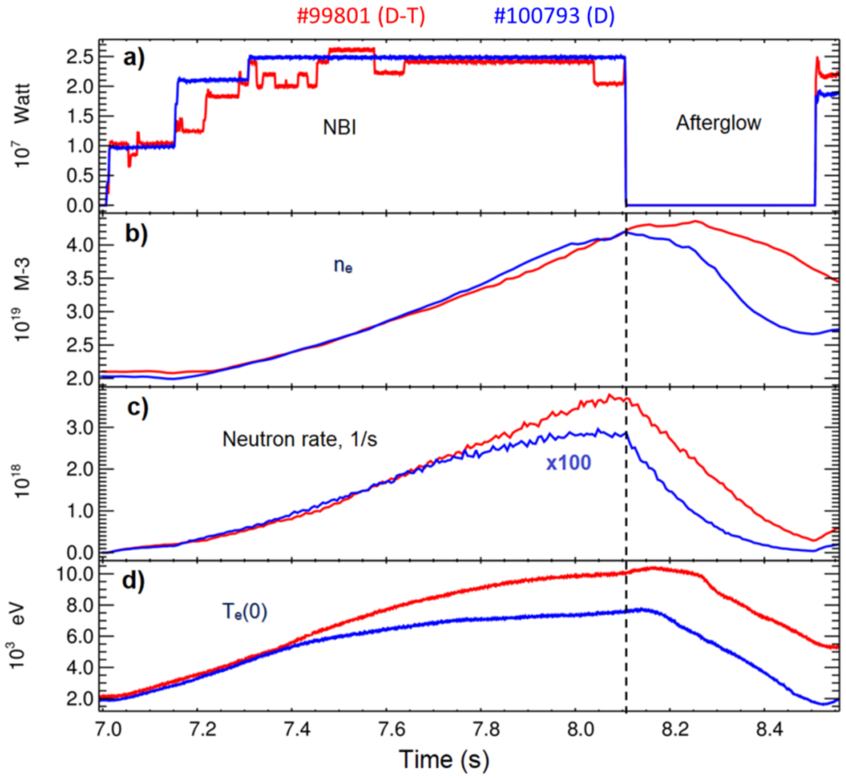
Baseline scenario H-mode (JET-C)
[holding world fusion energy record till DTE2]

Burning Plasmas and Long Pulse Operation: JET DTE2

Direct evidence of electron heating from α-particles

Red: D-T

Blue: counterpart reference D plasma



- **Self-heating of thermonuclear fusion plasma by α-particles observed in JET DTE2**
- Observed in NBI 'after-glow' phase
- Interpretative transport modelling (TRANSP) of D-T and D reference discharges consistent with α-particle heating observation

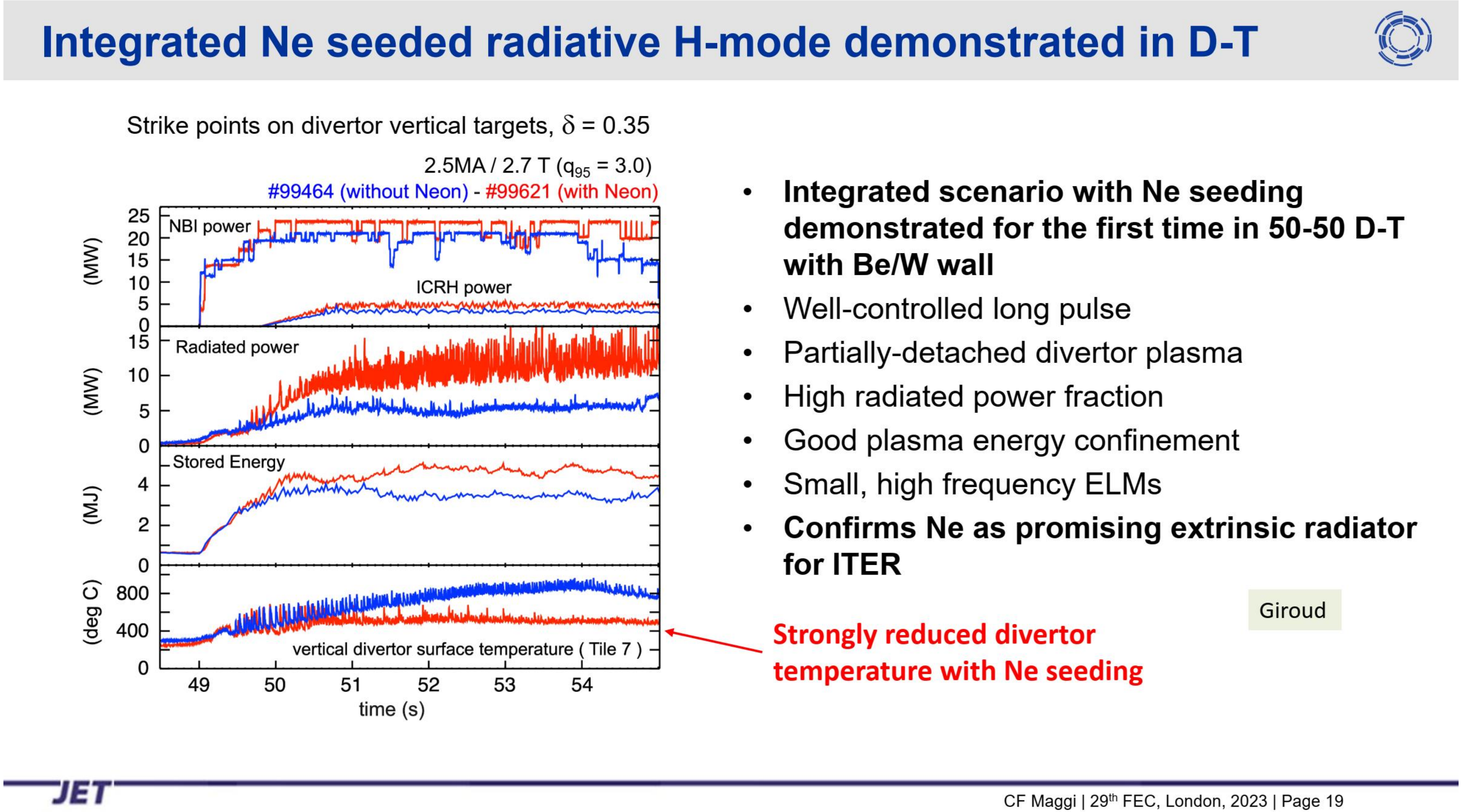
[Kiptily et al, PRL 2023]

Garcia, EX-W # 1814 (Thursday)

JET

CF Maggi | 29th FEC, London, 2023 | Page 21

Burning Plasmas and Long Pulse Operation: JET DTE2



Integrated Ne seeded radiative H-mode demonstrated in D-T

Strike points on divertor vertical targets, $\delta = 0.35$

2.5MA / 2.7 T ($q_{95} = 3.0$)

#99464 (without Neon) - #99621 (with Neon)

- **Integrated scenario with Ne seeding demonstrated for the first time in 50-50 D-T with Be/W wall**
- Well-controlled long pulse
- Partially-detached divertor plasma
- High radiated power fraction
- Good plasma energy confinement
- Small, high frequency ELMs
- **Confirms Ne as promising extrinsic radiator for ITER**

Giroud

Strongly reduced divertor temperature with Ne seeding

JET

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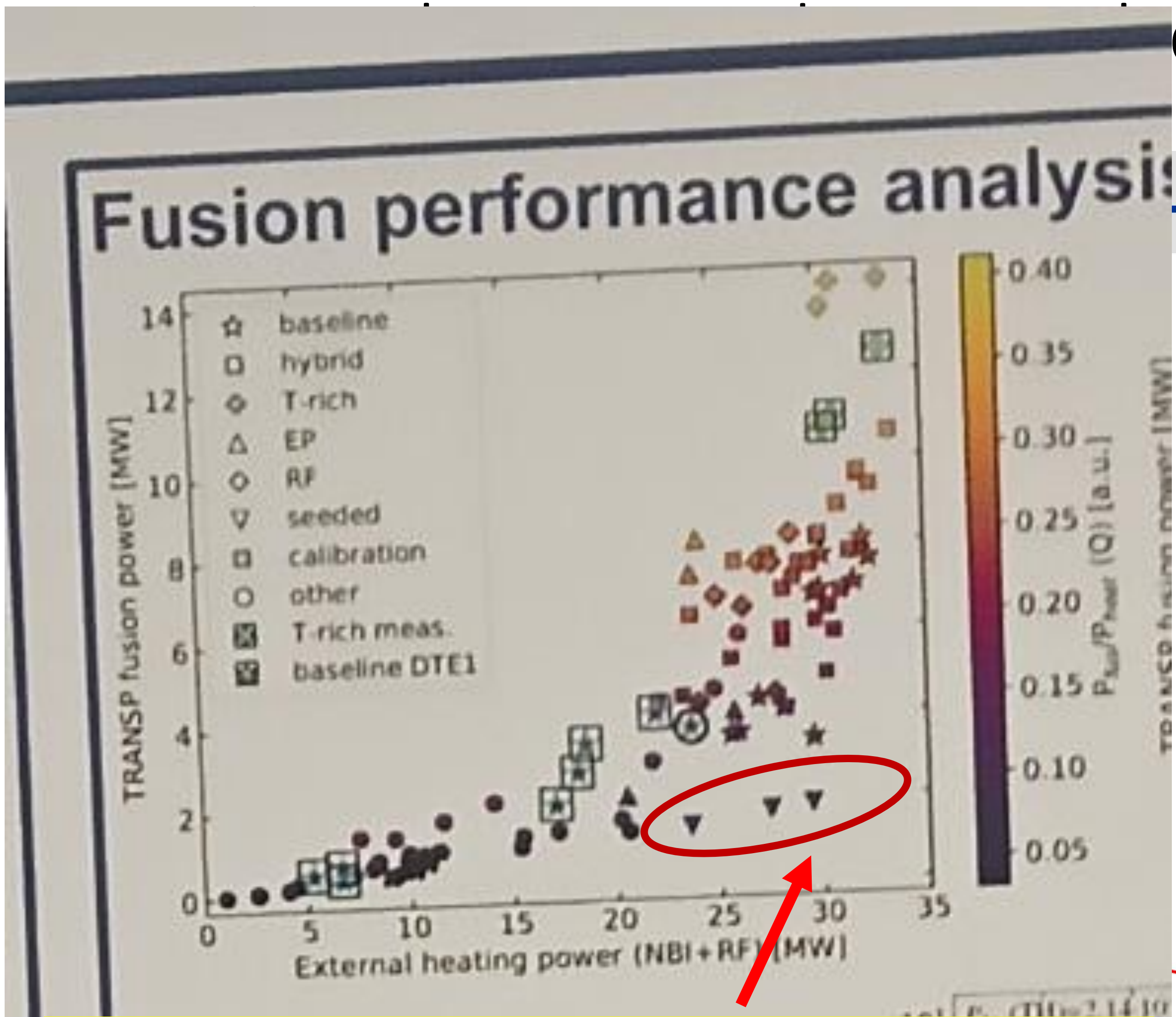

Fusion performance analysis

Разряды с напуском Ne имеют значительно более низкий нейтронный выход

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OVERVIEW OF INTERPRETIVE MODELLING OF FUSION PERFORMANCE IN JET DTE2 DISCHARGES WITH TRANSP

Ž. Štancar^a (ziga.stancar@ukaea.uk), K.K. Kirov^a, F. Auriemma^b, H.-T. Kim^a, M. Poradzinski^a, R. Sharma^a, R. Lorenzini^b, Z. Ghani^a, M. Gorelenkova^c, F. Poli^c, A. Boboc^a, S. Brezinsek^d, P. Carvalho^a, F.J. Casson^a, C.D. Challis^a, E. Delabie^e, D. Van Eester^f, M. Fitzgerald^a, J.M.

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JET

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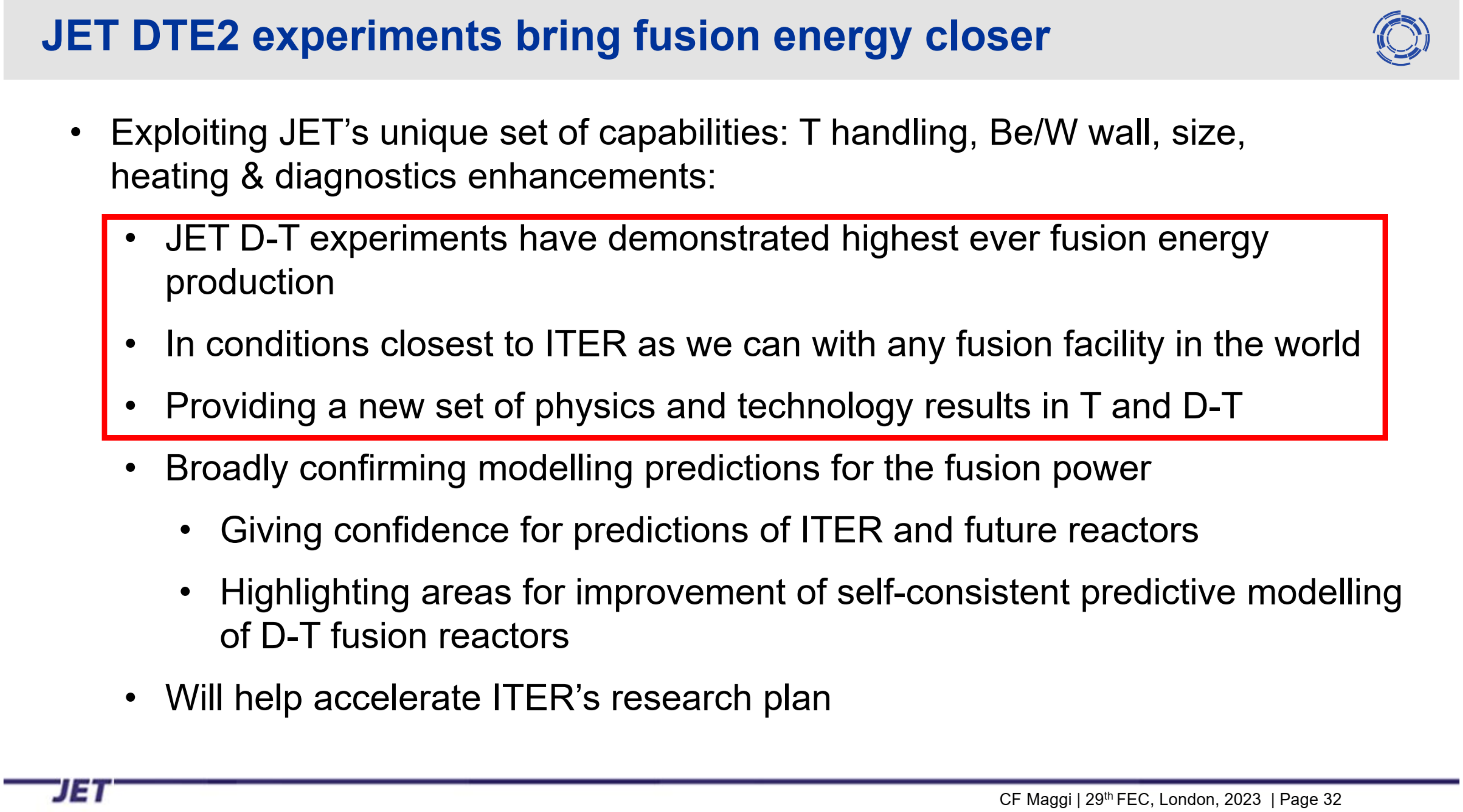
Burning Plasmas and Long Pulse Operation: JET DTE2



JET DTE2 experiments bring fusion energy closer

- Exploiting JET's unique set of capabilities: T handling, Be/W wall, size, heating & diagnostics enhancements:
  - JET D-T experiments have demonstrated highest ever fusion energy production
  - In conditions closest to ITER as we can with any fusion facility in the world
  - Providing a new set of physics and technology results in T and D-T
  - Broadly confirming modelling predictions for the fusion power
    - Giving confidence for predictions of ITER and future reactors
    - Highlighting areas for improvement of self-consistent predictive modelling of D-T fusion reactors
  - Will help accelerate ITER's research plan

JET

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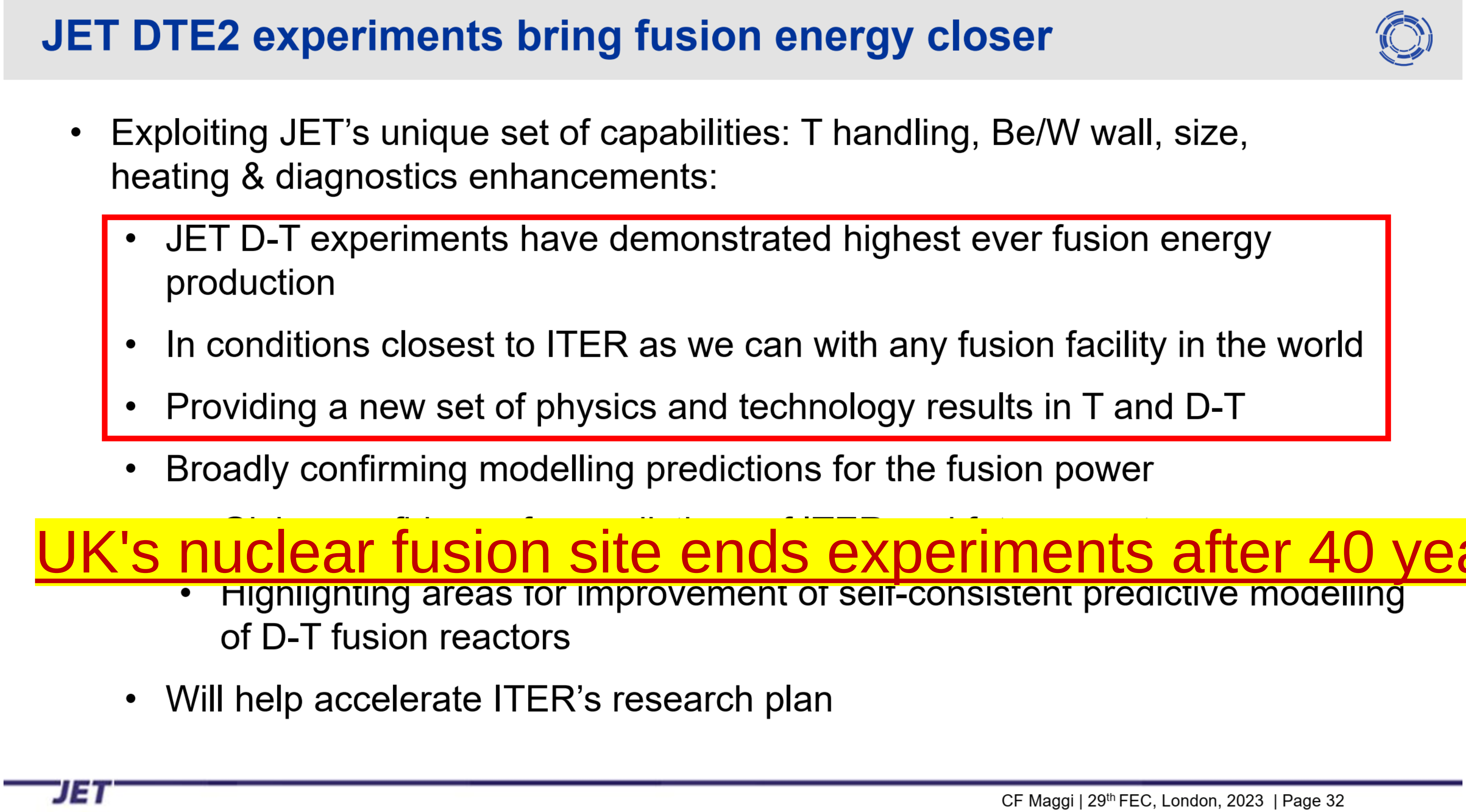
Burning Plasmas and Long Pulse Operation: JET DTE2



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  - Broadly confirming modelling predictions for the fusion power
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  - Will help accelerate ITER's research plan

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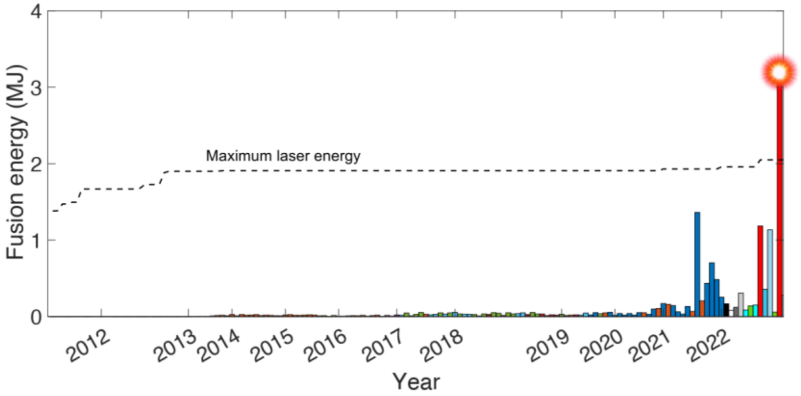
UK's nuclear fusion site ends experiments after 40 years

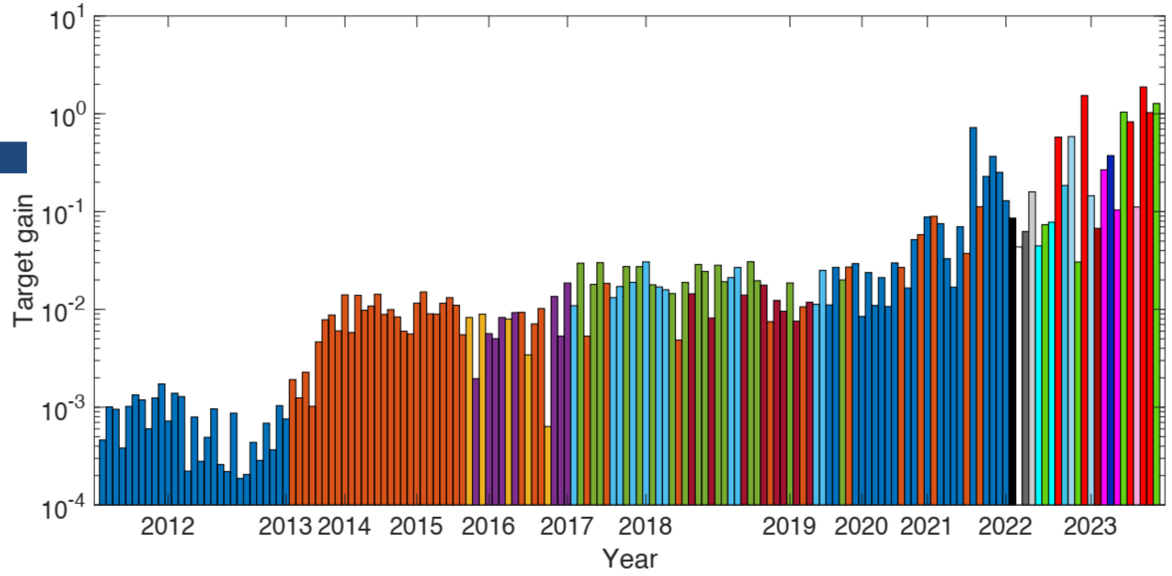
Burning Plasmas and Long Pulse Operation: NIF

The first experiment to exceed the threshold for fusion ignition*
was conducted on Dec. 5th 2022, achieving a target gain of 1.5



3.14 MJ of fusion energy produced from 2.05 MJ of laser energy

challenges remain, but there is proof that with sustained effort,
n accomplish challenging things together



Thanks!

Lawrence Livermore National Laboratory

NNSA
National Nuclear Security Administration

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Burning Plasmas and Long Pulse Operation: EAST

Summary

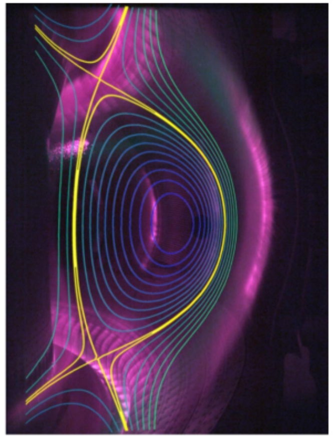
- **Significant progress has been made in long-pulse SSO on EAST**
  - Record of duration ~403s H-mode and a 1056s time scale fully non-inductive plasma demonstrated
  - Development of state steady high-β_p scenarios up to 100s with zero torque injection (β_p~3.0/β_N~1.8, f_{BS}>50%, n_e/n_G~0.82, H_{98y2}~1.5)
  - Extension of fusion performance with high confinement at high density (H_{98y2}~1.5 at n_e/n_G~1.0) and β_N =2.5~4*li near no-wall limit

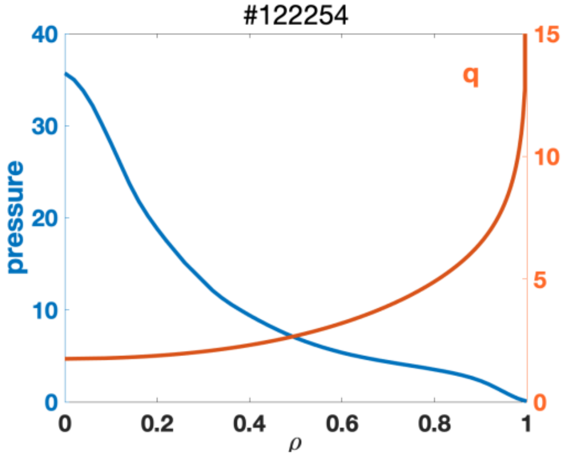
- **Advances on the key issues essential for long pulse SSO, providing supports to ITER and CFETR steady state operation**
  - Improved confinement with stability, broad j(r), Shafranov shift, e-ITB, high efficiency of H&CD at high density, active controls of radiative divertor, small ELM, plasma control, etc.

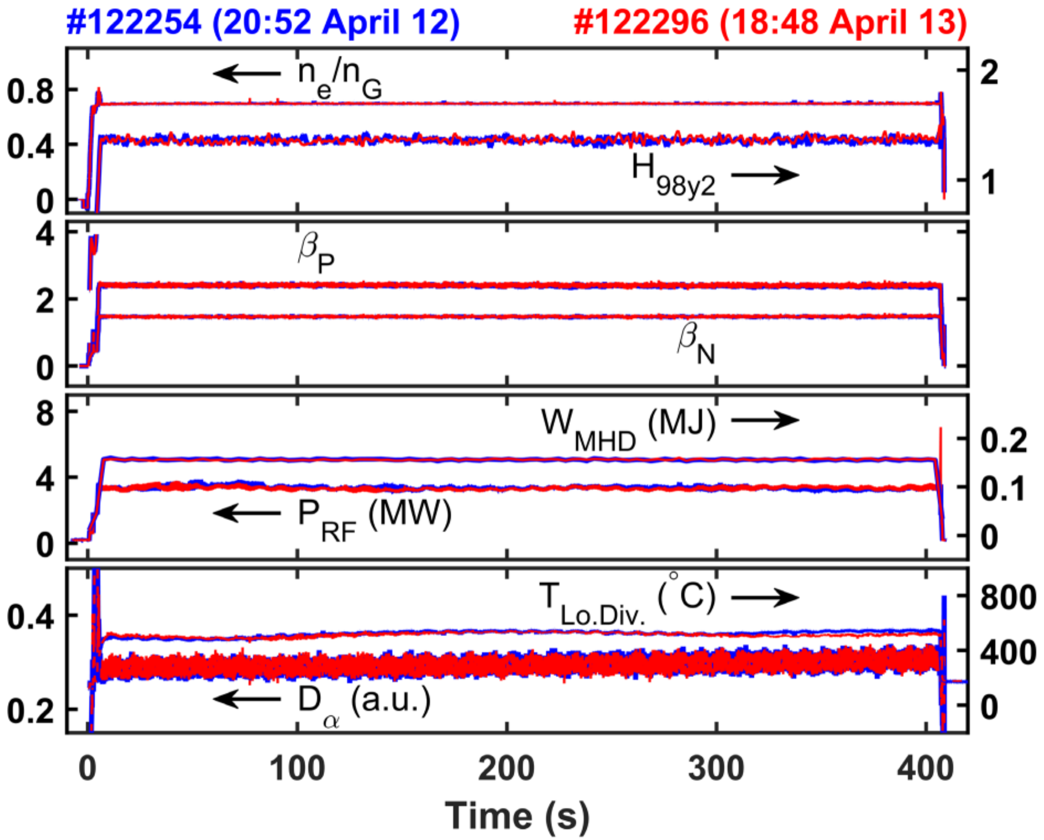
- **Near-term plan with upgrade of inner components and augmented H&CD systems**
  - 1000s long-pulse H-mode operation with high bootstrap current fraction
  - Demonstrate SSO with **extended fusion performance** at 15-20 MW power injection

Burning Plasmas and Long Pulse Operation: EAST

New Record of Reproducible 403 Seconds H-mode Plasmas Demonstrated on EAST with Tungsten Divertor







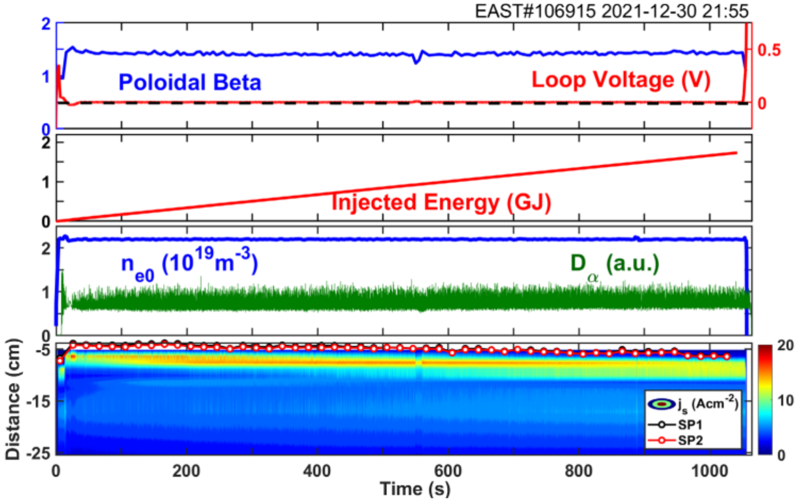
New Milestone and Tremendous Step Forward

- A full non-inductive at f_{GW}~0.7 with f_{BS}>50% by RF heating with zero torque injection
- $H_{98,y2}$ ~1.35 with ITB by electron dominant heating
- Stationary control on particle exhaust and heat load with actively cooling W-divertor
- Small ELMs throughout discharges with high core performance

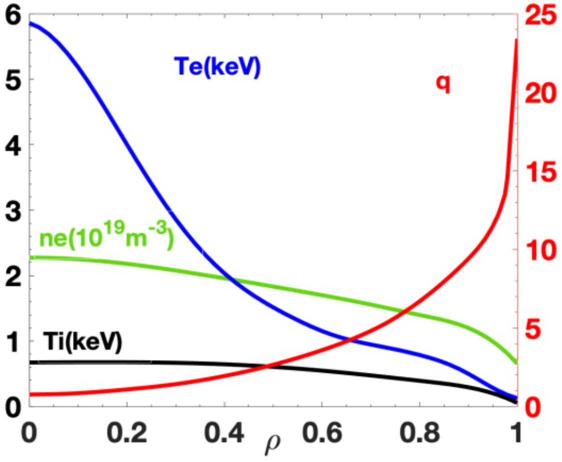
ASIPP

Burning Plasmas and Long Pulse Operation: EAST

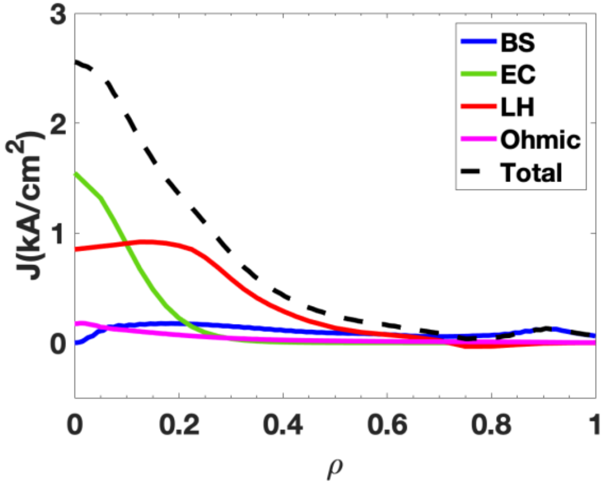
Demonstration of 1056-second Steady-state Plasma with Improved Confinement on EAST



Total injected energy ~ 1.73 GJ



Y. Song, Sci. Adv., 2022



https://www.iter.org/newsline/-/3740

- Fully non-inductive f_{RFCD}~70%, f_{BS}~30%, H_{89} ~1.3, V_{loop}~0 with feedback control by LHW
- Enhanced power exhaust with new water-cooled lower W-divertor
- Lower recycling control with real-time wall conditioning
- Good control of plasma equilibrium and position over long time scales

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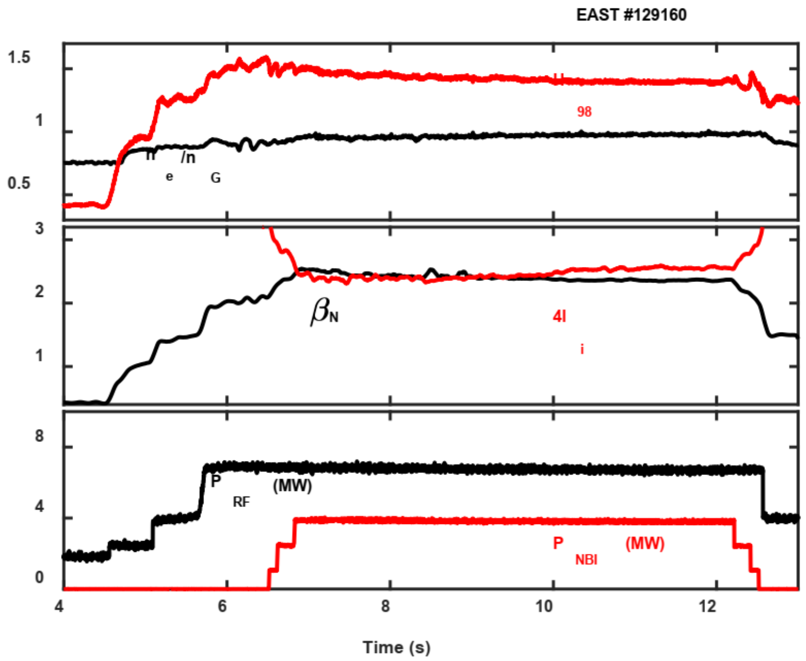
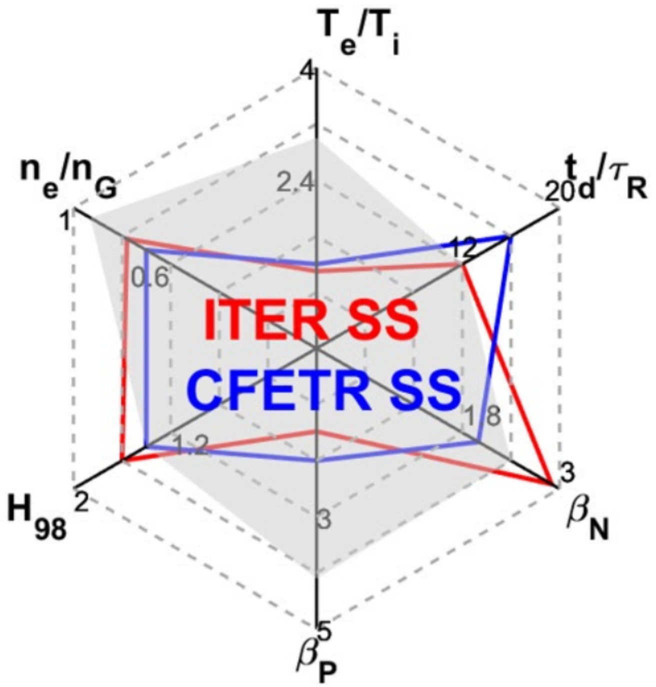
ASIPP

Burning Plasmas and Long Pulse Operation: EAST

Demonstration of High Confinement, High Density and High f_{BS} towards ITER Steady-state Operation





- **Normalized parameters close to ITER/CFETR steady state**
  - H_{98y2}~1.5, n_e/n_G~1.0, β_N~2.5~4xl_i
  - Broader q-profile (weak shear/NCS) and q_{min}>2 to avoid NTM

Burning Plasmas and Long Pulse Operation: EAST

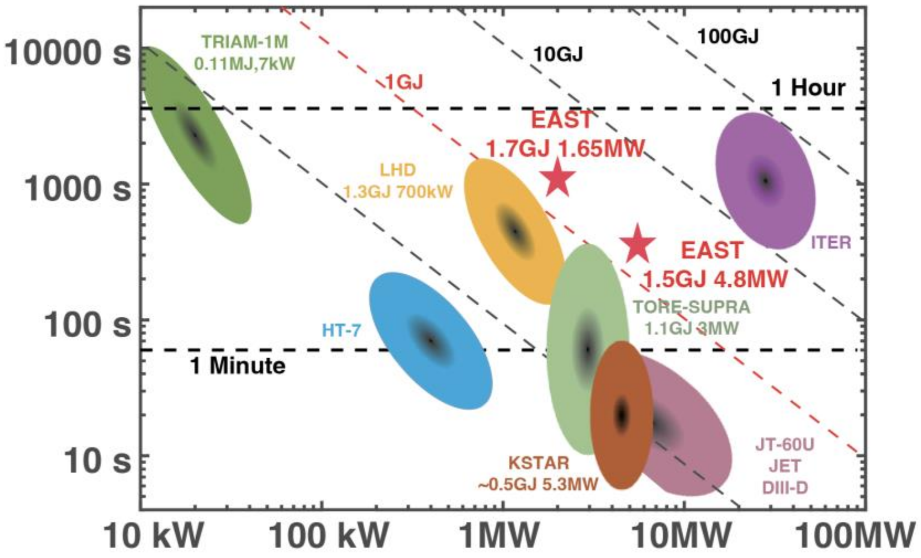
Strategies to Establish the Scientific Basis Integrated Solutions of for SSO in Support of Future Fusion Devices

S1: Enhance H/CD efficiency & relevant fundamental physics understanding and key diagnostics

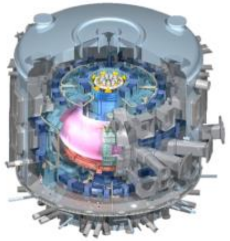
S2: Demonstrate long-pulse (≥400s) H-mode plasmas and develop fully non-inductive high-β scenarios

S3: Extend EAST operation regime to demonstrate steady-state high performance plasmas and deliver relevant physics for ITER and CFETR

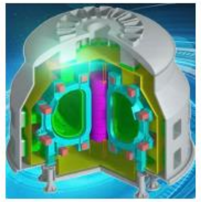
Total injected energy up to 1.73GJ



ITER

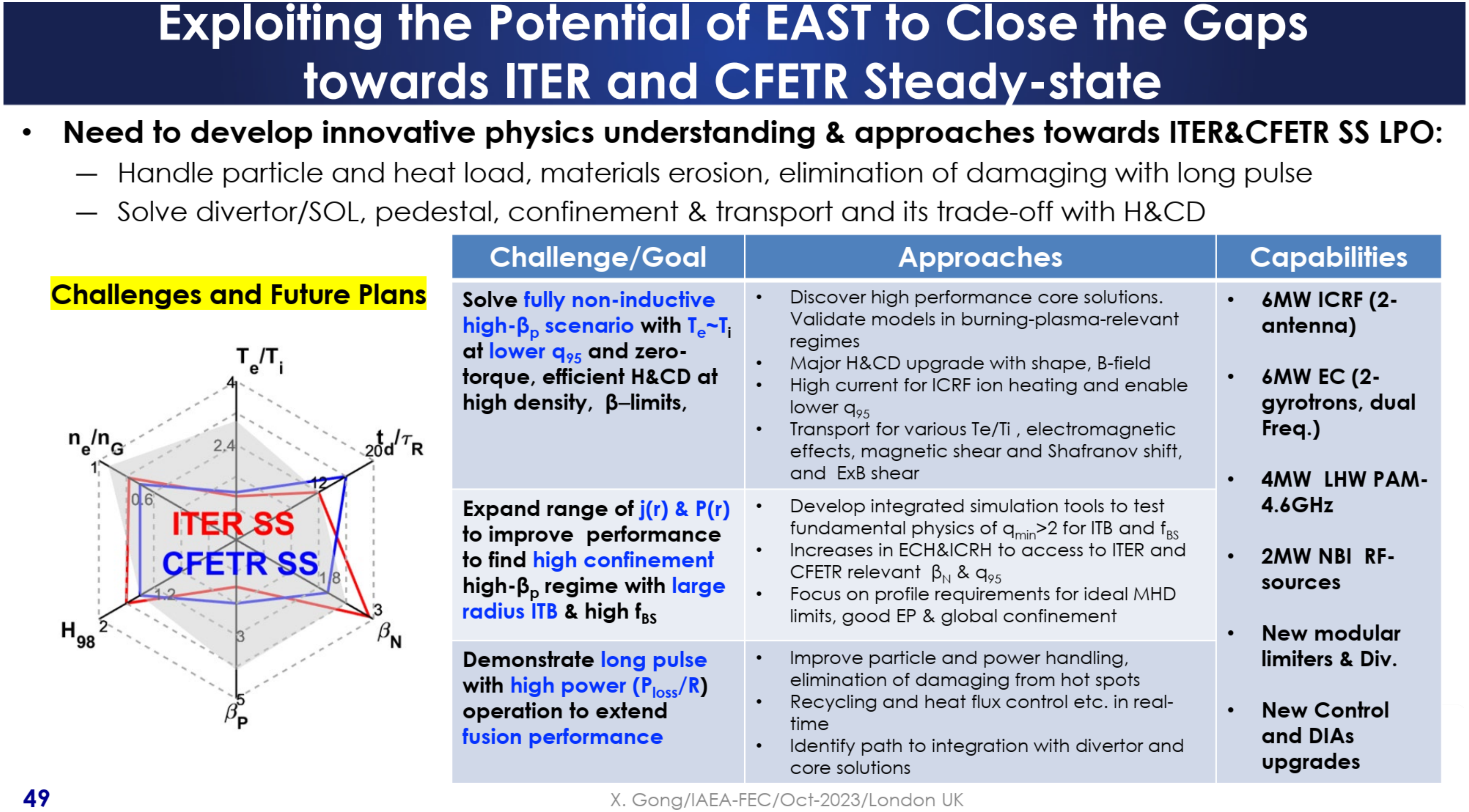


BEST



CFETR

Burning Plasmas and Long Pulse Operation: EAST



Exploiting the Potential of EAST to Close the Gaps towards ITER and CFETR Steady-state

- **Need to develop innovative physics understanding & approaches towards ITER&CFETR SS LPO:**
  - Handle particle and heat load, materials erosion, elimination of damaging with long pulse
  - Solve divertor/SOL, pedestal, confinement & transport and its trade-off with H&CD

Challenge/Goal	Approaches	Capabilities
Solve fully non-inductive high-β_p scenario with T_e~T_i at lower q_{95} and zero-torque, efficient H&CD at high density, β–limits,	• Discover high performance core solutions. Validate models in burning-plasma-relevant regimes • Major H&CD upgrade with shape, B-field • High current for ICRF ion heating and enable lower q_{95} • Transport for various Te/Ti , electromagnetic effects, magnetic shear and Shafranov shift, and ExB shear	• **6MW ICRF (2-antenna)** • **6MW EC (2-gyrotrons, dual Freq.)** • **4MW LHW PAM-4.6GHz** • **2MW NBI RF-sources** • **New modular limiters & Div.** • **New Control and DIAs upgrades**
Expand range of j(r) & P(r) to improve performance to find high confinement high-β_p regime with large radius ITB & high f_{BS}	• Develop integrated simulation tools to test fundamental physics of q_{min}>2 for ITB and f_{BS} • Increases in ECH&ICRH to access to ITER and CFETR relevant β_N & q_{95} • Focus on profile requirements for ideal MHD limits, good EP & global confinement	
Demonstrate long pulse with high power (P_{loss}/R) operation to extend fusion performance	• Improve particle and power handling, elimination of damaging from hot spots • Recycling and heat flux control etc. in real-time • Identify path to integration with divertor and core solutions	

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Burning Plasmas and Long Pulse Operation: KSTAR

Summary

- **Scenario Development Toward High Beta Steady-state Operation**

  KSTAR has demonstrated the potential of **steady-state high β_N operation and achieved high $\ell_i \sim 1$ and $\beta_N \sim 3$ for 12 s.**

  A new stationary ITB scenario with self-organized high Ti (~ 10 keV) has been established and **Gyrokinetic simulations** explain that inverted main ion density gradient contributes significantly to turbulence suppression with fast ion fraction for ITB formation in KSTAR.

- **Topics Resolving Uncertainty for ITER**

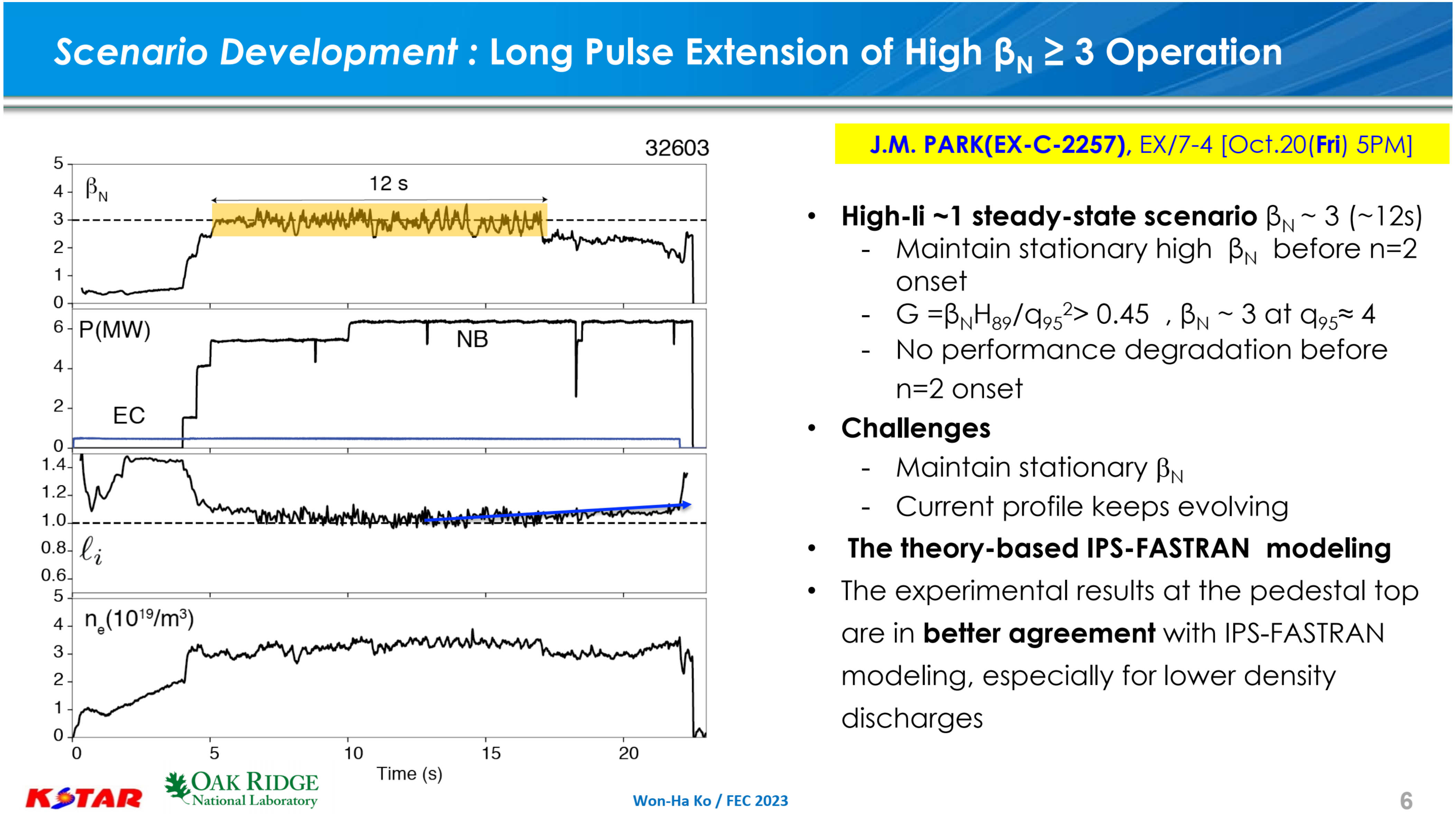
  Advanced ELM suppression achieved β_N (~ 2.65) record and the longest (~ 45 s) discharges with adaptive control and ERMP optimization.

  Real-time **feedback control of the detachment** achieved long pulse for 22s in KSTAR.

- **Better Understanding on the Fundamental Processes**

  High fidelity simulations and advanced fluctuation analysis provide better insight for transport and turbulence in KSTAR.

Burning Plasmas and Long Pulse Operation: KSTAR

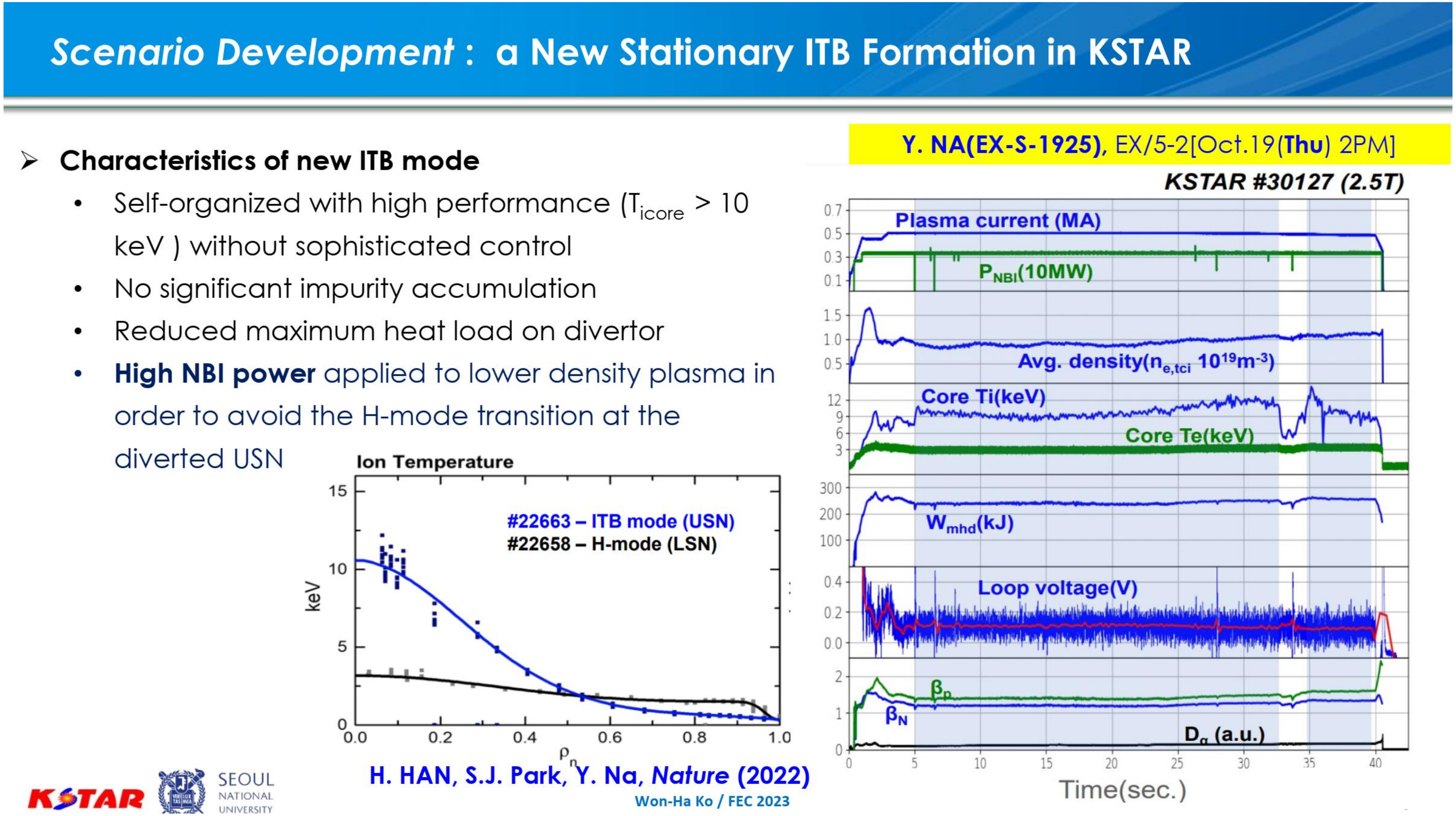


Scenario Development : Long Pulse Extension of High $\beta_N \geq 3$ Operation

J.M. PARK(EX-C-2257), EX/7-4 [Oct.20(**Fri**) 5PM]

- **High-li ~1 steady-state scenario** $\beta_N \sim 3$ (~12s)
  - Maintain stationary high β_N before n=2 onset
  - $G = \beta_N H_{89}/q_{95}^2 > 0.45$, $\beta_N \sim 3$ at $q_{95} \approx 4$
  - No performance degradation before n=2 onset
- **Challenges**
  - Maintain stationary β_N
  - Current profile keeps evolving
- **The theory-based IPS-FASTRAN modeling**
- The experimental results at the pedestal top are in **better agreement** with IPS-FASTRAN modeling, especially for lower density discharges

Burning Plasmas and Long Pulse Operation: KSTAR



Scenario Development : a New Stationary ITB Formation in KSTAR

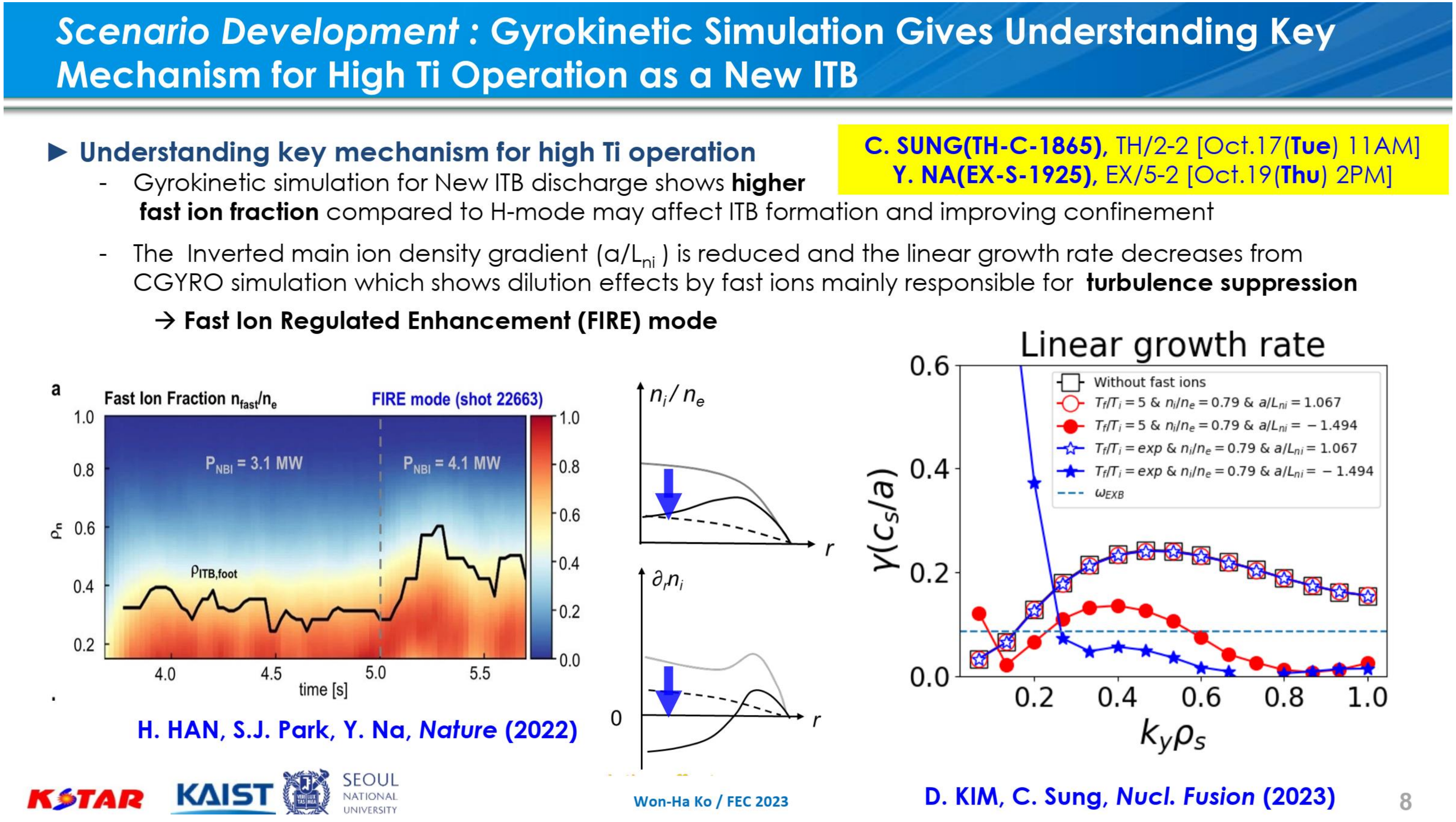
- **Characteristics of new ITB mode**
  - Self-organized with high performance (T_{icore} > 10 keV) without sophisticated control
  - No significant impurity accumulation
  - Reduced maximum heat load on divertor
  - **High NBI power** applied to lower density plasma in order to avoid the H-mode transition at the diverted USN

Y. NA(EX-S-1925), EX/5-2[Oct.19(**Thu**) 2PM]

H. HAN, S.J. Park, Y. Na, *Nature* (2022)

Won-Ha Ko / FEC 2023

Burning Plasmas and Long Pulse Operation: KSTAR



Scenario Development : Gyrokinetic Simulation Gives Understanding Key Mechanism for High Ti Operation as a New ITB

C. SUNG(TH-C-1865), TH/2-2 [Oct.17(**Tue**) 11AM]
Y. NA(EX-S-1925), EX/5-2 [Oct.19(**Thu**) 2PM]

▶ **Understanding key mechanism for high Ti operation**

- Gyrokinetic simulation for New ITB discharge shows **higher fast ion fraction** compared to H-mode may affect ITB formation and improving confinement
- The Inverted main ion density gradient (a/L_{ni}) is reduced and the linear growth rate decreases from CGYRO simulation which shows dilution effects by fast ions mainly responsible for **turbulence suppression**

→ **Fast Ion Regulated Enhancement (FIRE) mode**

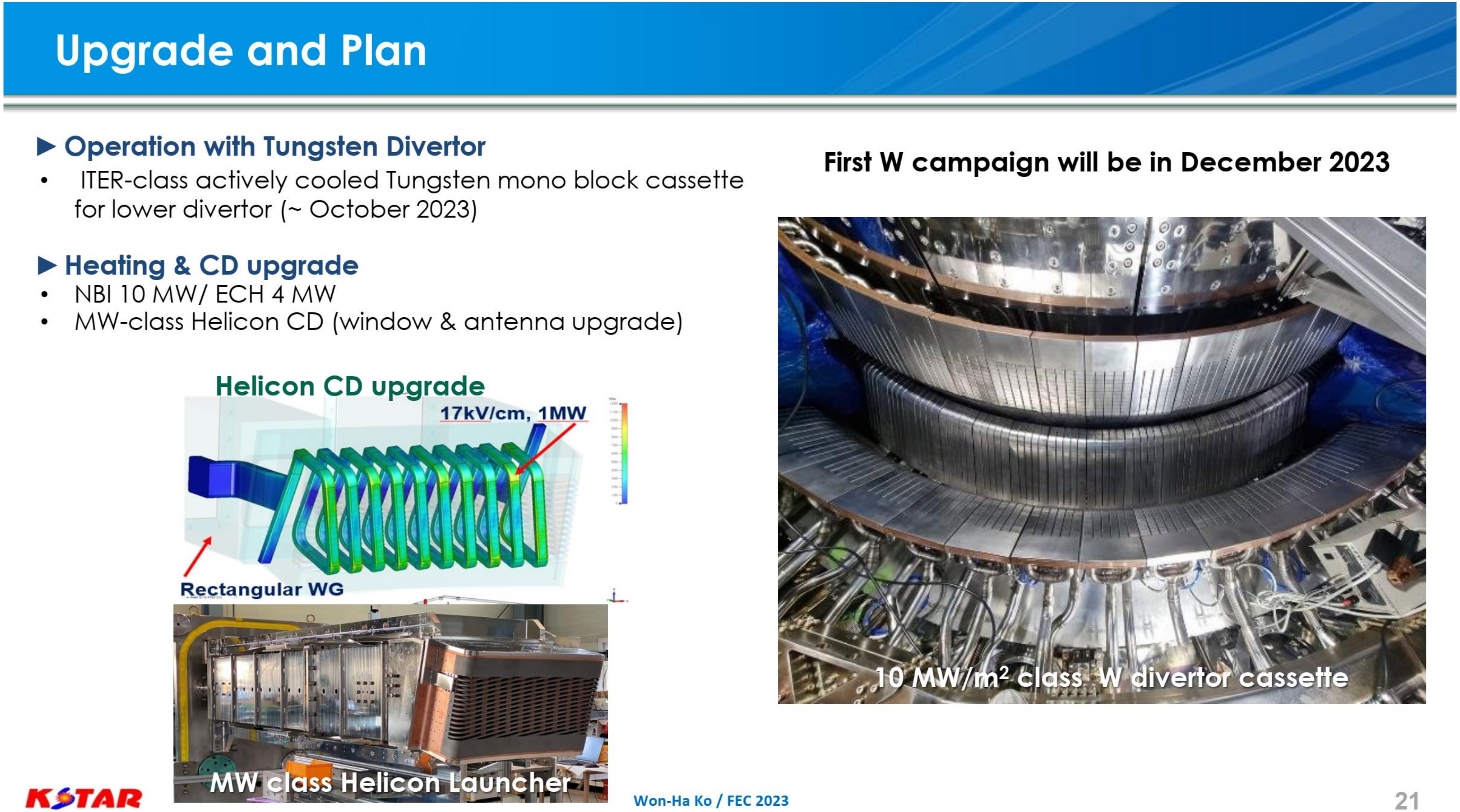
H. HAN, S.J. Park, Y. Na, *Nature* (2022)

Won-Ha Ko / FEC 2023

D. KIM, C. Sung, *Nucl. Fusion* (2023)

KSTAR KAIST SEOUL NATIONAL UNIVERSITY

Burning Plasmas and Long Pulse Operation: KSTAR



Upgrade and Plan

► Operation with Tungsten Divertor

- ITER-class actively cooled Tungsten mono block cassette for lower divertor (~ October 2023)

► Heating & CD upgrade

- NBI 10 MW/ ECH 4 MW
- MW-class Helicon CD (window & antenna upgrade)

Helicon CD upgrade

First W campaign will be in December 2023

KSTAR

Won-Ha Ko / FEC 2023

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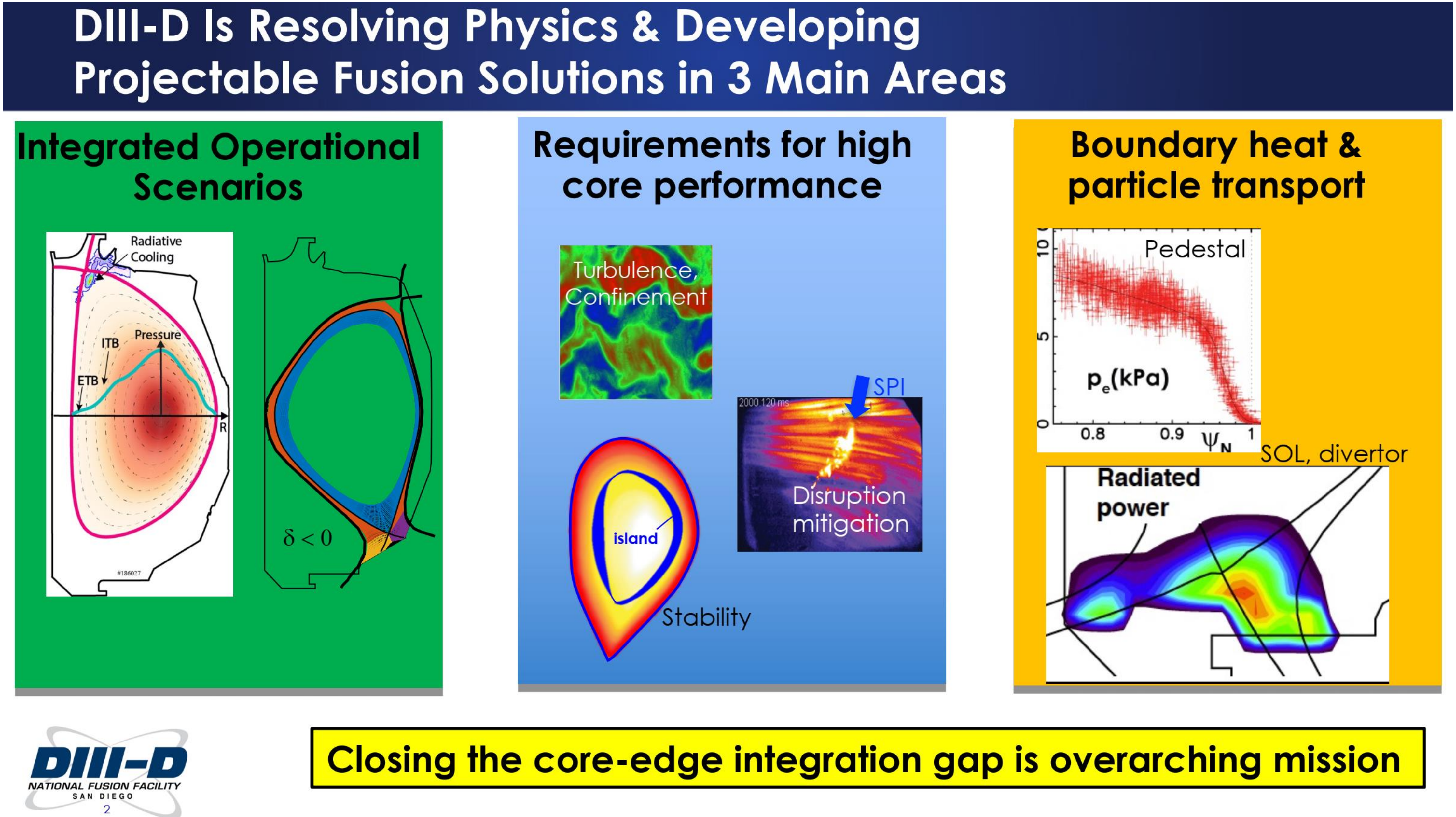
# -	Tokamaks:
DIII-D, AUG, TCV, HL-2M

Tokamaks: DIII-D



DIII-D Is Resolving Physics & Developing Projectable Fusion Solutions in 3 Main Areas

Integrated Operational Scenarios

Requirements for high core performance

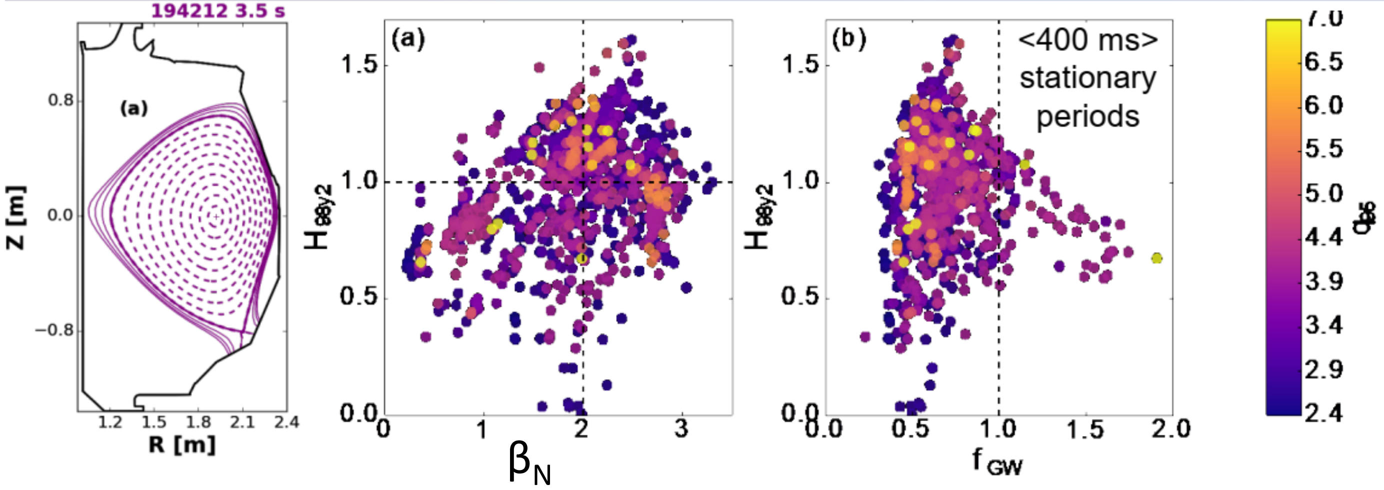
Boundary heat & particle transport

Closing the core-edge integration gap is overarching mission

DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

Tokamaks: DIII-D

Negative Triangularity Campaign Achieved High Performance With a Diverted Non-ELMing Edge



- **Achieved:**
  - Simultaneous $H_{98}>1$ & $\beta_N>2.5$
  - $f_{GW}\sim1.9$ at high q_{95}
  - $q_{95}<3$ at low f_{GW}

- **With $\delta_{ave}\sim-0.5$, maintain non-ELMing edge with small T_e pedestal**
  - ∇p limited by high-n ballooning modes set by geometry induced $\hat{s}$

- **Sustained detached divertor for ~1 s with $\beta_N\sim2.2$ & $H_{98}\sim1$**

DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

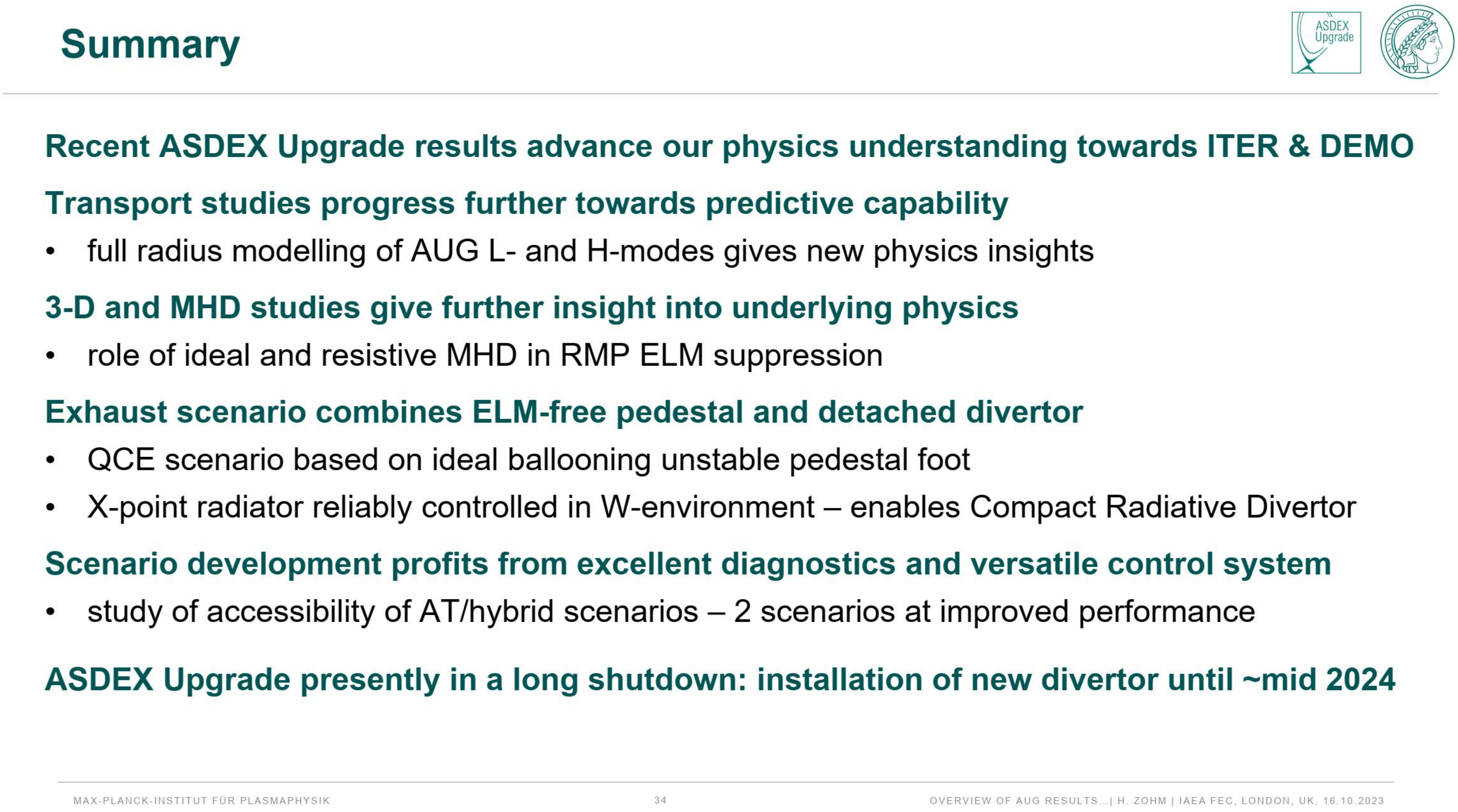
Possible highly attractive core-edge integration solution for reactors

Thome EX-P, Tue. AM
Nelson poster Fri. AM
Casali poster Sat. AM
Marinoni poster Tue AM

Negative triangularity – модная тема

1) 2000. NONLINEAR GYROKINETIC MODELLING OF HIGH CONFINEMENT NEGATIVE TRIANGULARITY PLASMAS Dr Alessandro Marinoni (Massachusetts Institute of Technology)
2) 2355. Assessment of Negative Triangularity as a Reactor Scenario in DIII-D Kathreen Thome (General Atomics)
3) 1863. NEGATIVE TRIANGULARITY TOKAMAK OPERATION IN TCV Dr Olivier Sauter (SPC-EPFL)
4) 1703. Negative triangularity scenarios: from TCV and AUG experiments to DTT predictions Alberto Mariani (Istituto per la Scienza e la Tecnologia dei Plasmi, CNR, Milano (Italy))
5) 2093. MITIGATION OF TOROIDAL ALFVEN EIGENMODES IN NEGATIVE TRIANGULARITY PLASMAS AT TCV
6) 2499. [REGULAR POSTER TWIN] Assessment of Negative Triangularity as a Reactor Scenario in DIII-D Kathreen Thome (General Atomics)
7) 2504. [REGULAR POSTER TWIN] MITIGATION OF TOROIDAL ALFVEN EIGENMODES IN NEGATIVE TRIANGULARITY PLASMAS AT TCV Pablo Oyola (Universidad de Sevila)
8) 2496. [REGULAR POSTER TWIN] Negative triangularity scenarios: from TCV and AUG experiments to DTT predictions Alberto Mariani
9) 2498. [REGULAR POSTER TWIN] NEGATIVE TRIANGULARITY TOKAMAK OPERATION IN TCV Dr Olivier Sauter (SPC-EPFL)
10) 1776. GEOMETRY MEETS FEEDBACK LOOPS: SHEARING AND TURBULENCE SELF-REGULATION IN NEGATIVE TRIANGULARITY TOKAMAKS Dr Rameswar Singh
11) 2100. Effects of neutral transport and negative triangularity on plasma scrape-off layer turbulence in gyrokinetic simulations Tess Bernard (General Atomics)
12) 1769. Non-linear gyro-kinetic Ion Temperature Gradient and Trapped Electron Modes turbulence modelling in X-point geometry with 3D fields, Edge Localized Modes and at negative or positive triangularity shapes. Dr Marina Becoulet (IRFM/CEA)
13) 2247. SIMULATED EQUILIBRIUM OF PLAMSAS WITH NEGATIVE-TRIANGULARITY IN TOKAMAK ADITYA-U USING THE IPREQ MHD CODE Mrs Deepti Sharma (IPR)
14) 1997. ROBUST L-MODE EDGE BEHAVIOR IN HIGH PERFORMANCE NEGATIVE TRIANGULARITY PLASMAS: FROM EXPERIMENTS TO REACTORS Dr Andrew Oakleigh Nelson (Columbia University)
15) 2283. EXPLORING THE NEGATIVE-TRIANGULARITY PATHWAY TO FUSION WITH MANTA Andrew Nelson (Columbia University)
16) 2608. First Integration of negative triangularity plasmas with high core radiation fraction Prof. Livia Casali (The University of Tennessee)
17) 1632. Plasma rotation effects on the resistive wall modes in the negative triangularity tokamaks Dr Linjin Zheng (University of Texas at Austin, Institute for Fusion Studies)

Tokamaks: ASDEX-Upgrade



Summary

Recent ASDEX Upgrade results advance our physics understanding towards ITER & DEMO

Transport studies progress further towards predictive capability

- full radius modelling of AUG L- and H-modes gives new physics insights

3-D and MHD studies give further insight into underlying physics

- role of ideal and resistive MHD in RMP ELM suppression

Exhaust scenario combines ELM-free pedestal and detached divertor

- QCE scenario based on ideal ballooning unstable pedestal foot
- X-point radiator reliably controlled in W-environment – enables Compact Radiative Divertor

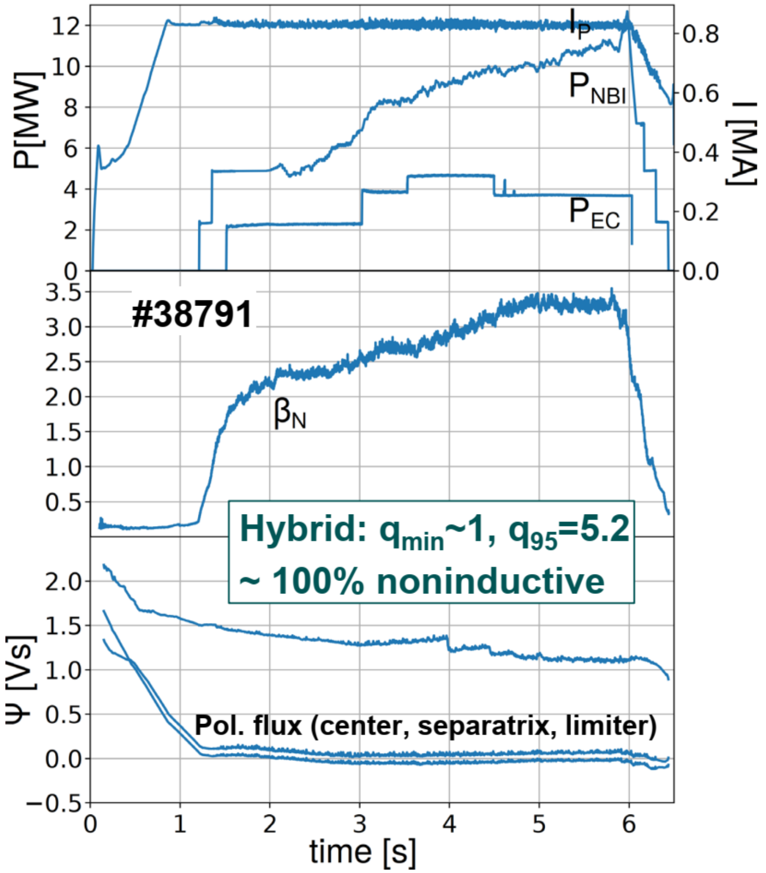
Scenario development profits from excellent diagnostics and versatile control system

- study of accessibility of AT/hybrid scenarios – 2 scenarios at improved performance

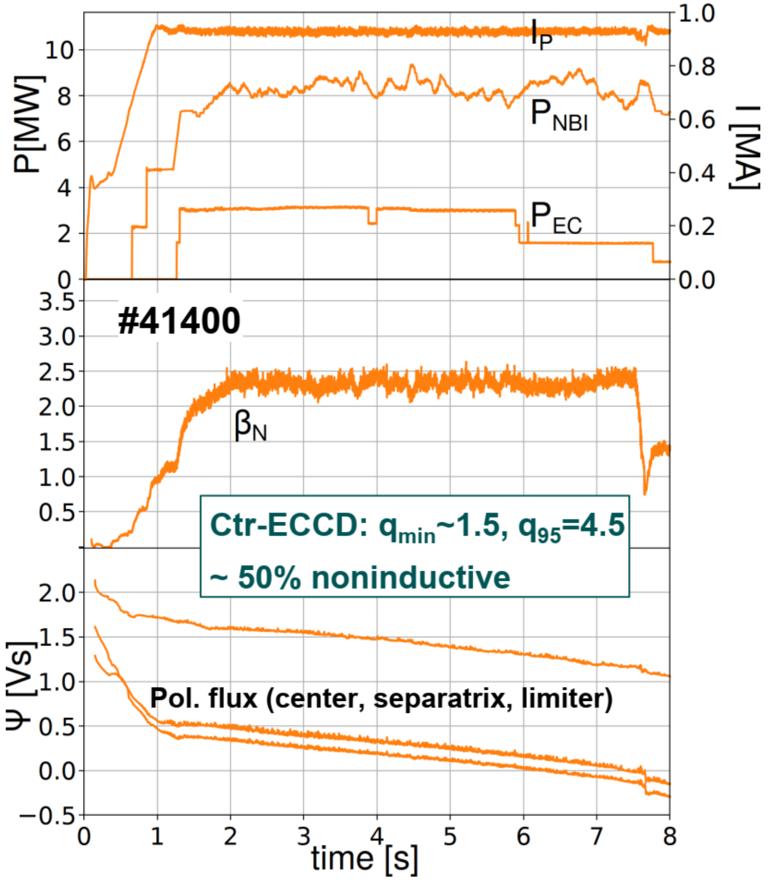
ASDEX Upgrade presently in a long shutdown: installation of new divertor until ~mid 2024

Tokamaks: ASDEX-Upgrade

AT Scenario development: progress towards steady state



Hybrid: q_{min}~1, q_{95}=5.2
~ 100% noninductive

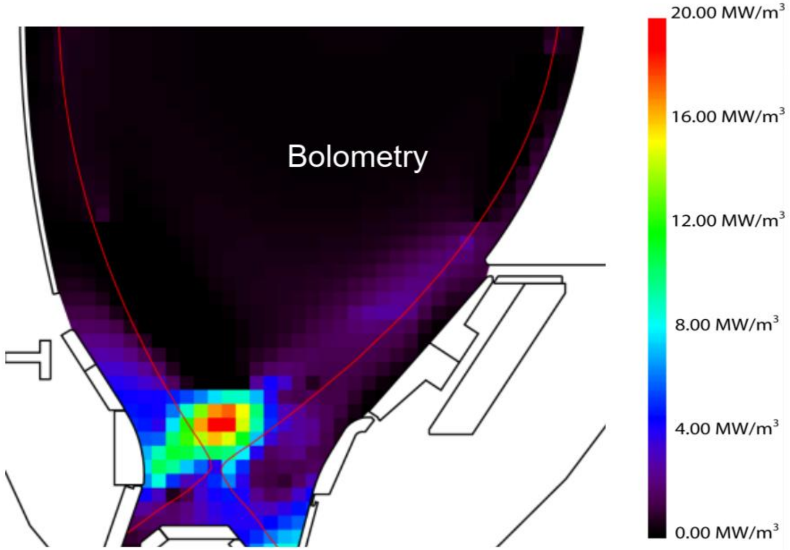


Ctr-ECCD: q_{min}~1.5, q_{95}=4.5
~ 50% noninductive

Both the ‘hybrid’ and the ‘elevated q’ scenario are now accessible in full-W AUG

Tokamaks: ASDEX-Upgrade

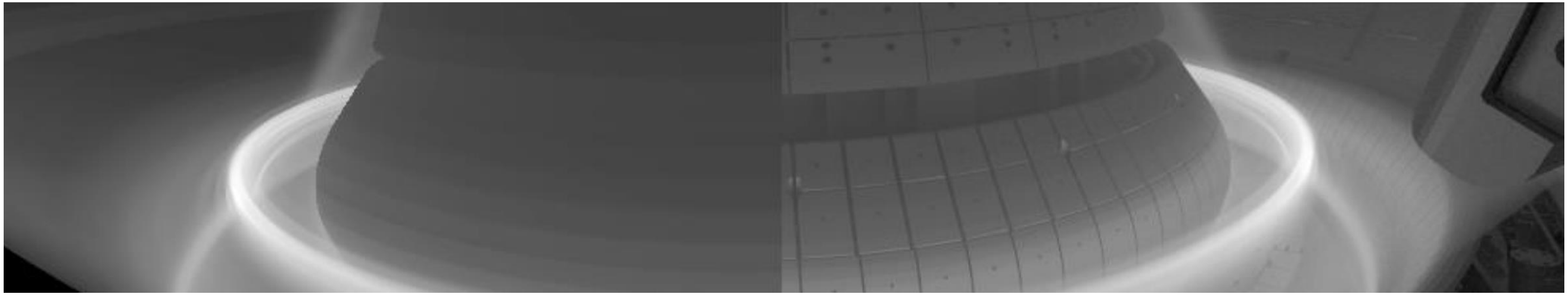
The X-point radiator (XPR):
a cold, dense, strongly radiating volume inside confined plasma



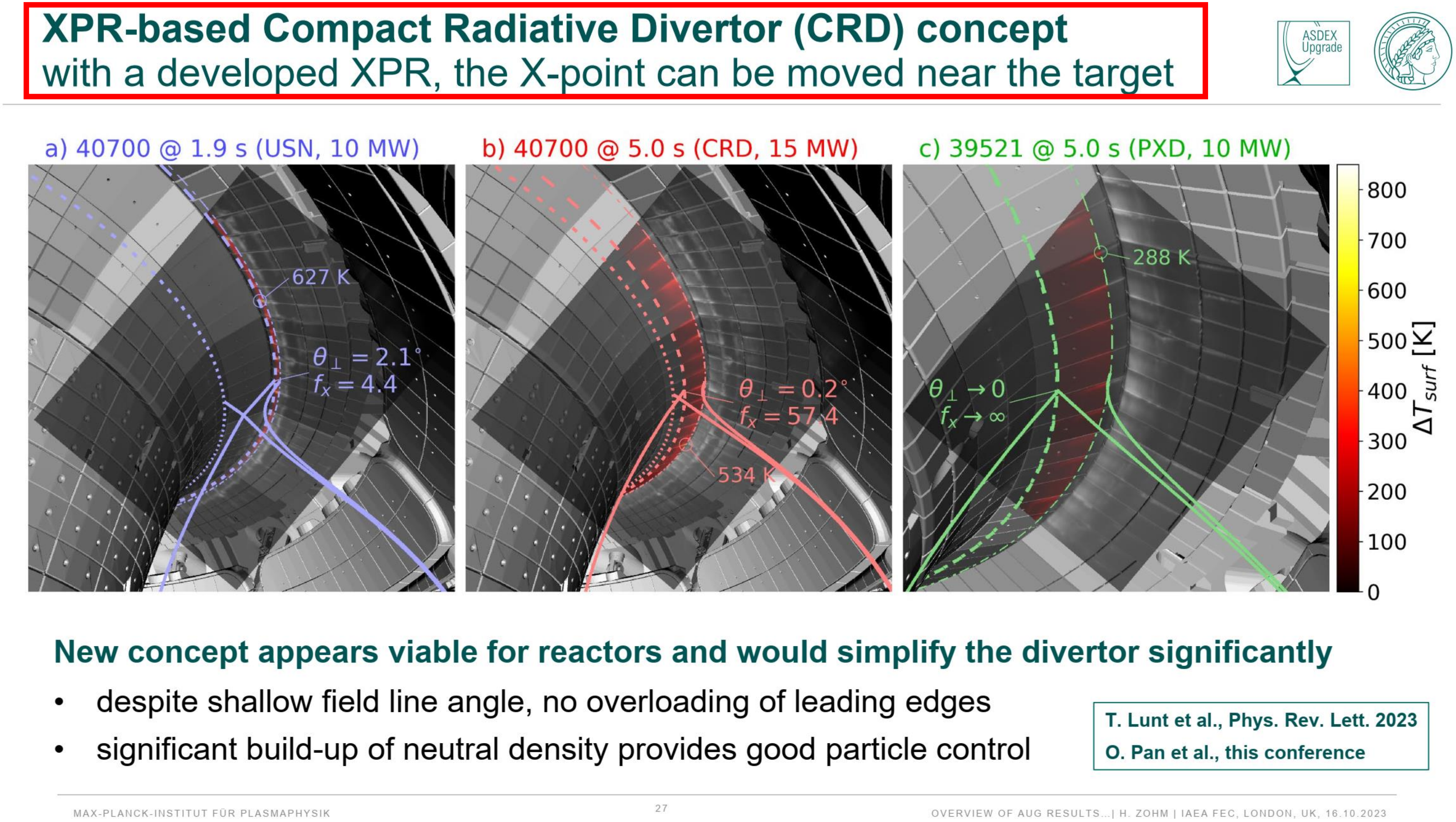
Important element for detachment control

- radiates power fraction up to 90 %
- high impurity & neutral compression
- stable & controllable up to 15 cm above X-point
- analytical model predicts stable operational window

M. Bernert et al, Nucl. Fusion 2021
U. Stroth et al., Nucl. Fusion 2022



Tokamaks: ASDEX-Upgrade



XPR-based Compact Radiative Divertor (CRD) concept
with a developed XPR, the X-point can be moved near the target

a) 40700 @ 1.9 s (USN, 10 MW)

b) 40700 @ 5.0 s (CRD, 15 MW)

c) 39521 @ 5.0 s (PXD, 10 MW)

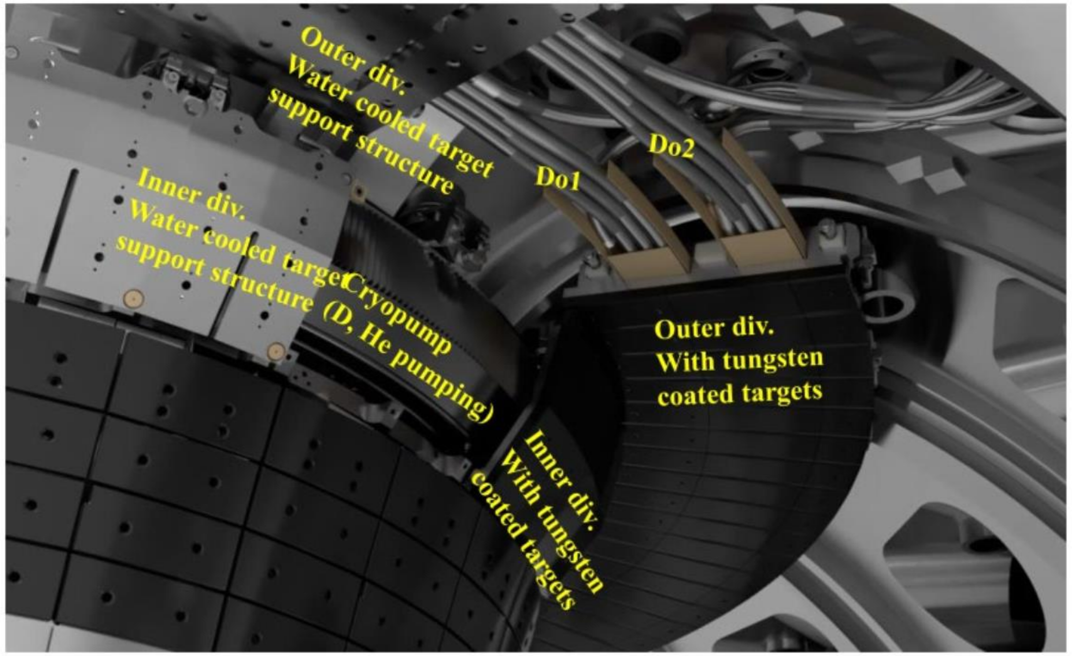
New concept appears viable for reactors and would simplify the divertor significantly

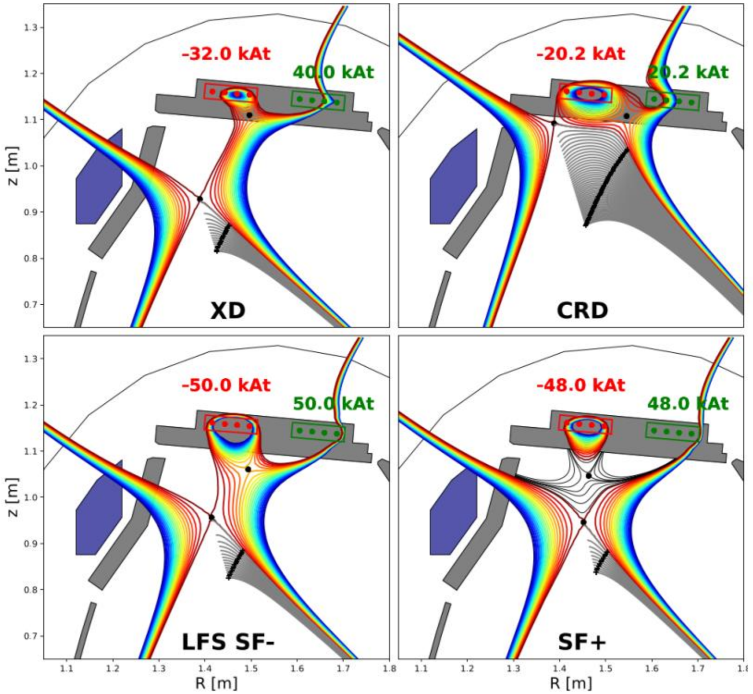
- despite shallow field line angle, no overloading of leading edges
- significant build-up of neutral density provides good particle control

T. Lunt et al., Phys. Rev. Lett. 2023
O. Pan et al., this conference

Tokamaks: ASDEX-Upgrade

Long shutdown for installation of new upper divertor (DivIIo)

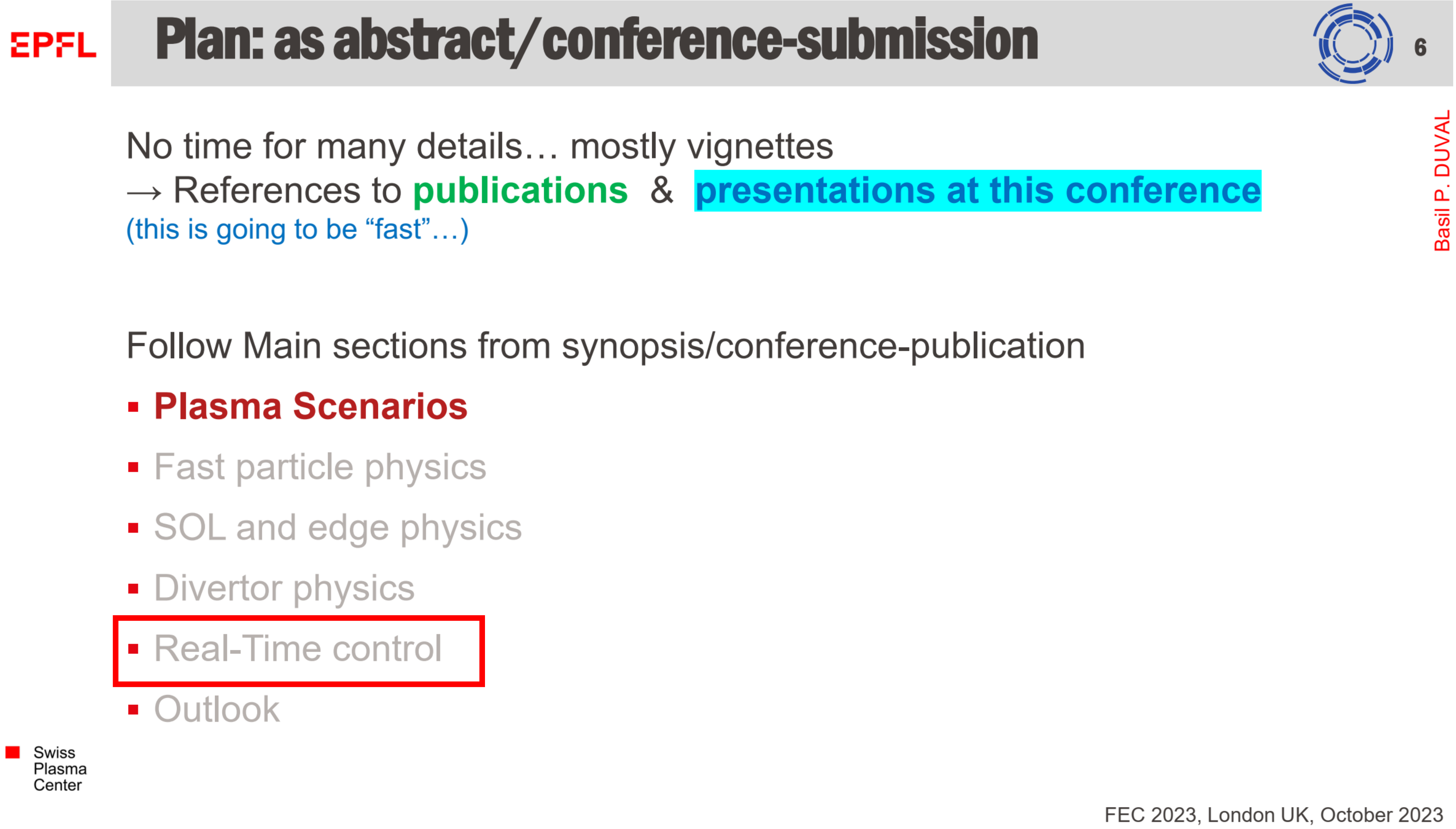




ASDEX Upgrade presently in a ~2 yr shutdown for installation of the new divertor

- will allow to study the physics element of advanced divertors (SF, X-Pt, XPR)
- coils are being wound inside the AUG vessel as we speak

Tokamaks: TCV



EPFL

Plan: as abstract/conference-submission

6

Basil P. DUVAL

No time for many details… mostly vignettes
→ References to **publications** & **presentations at this conference**
(this is going to be “fast”…)

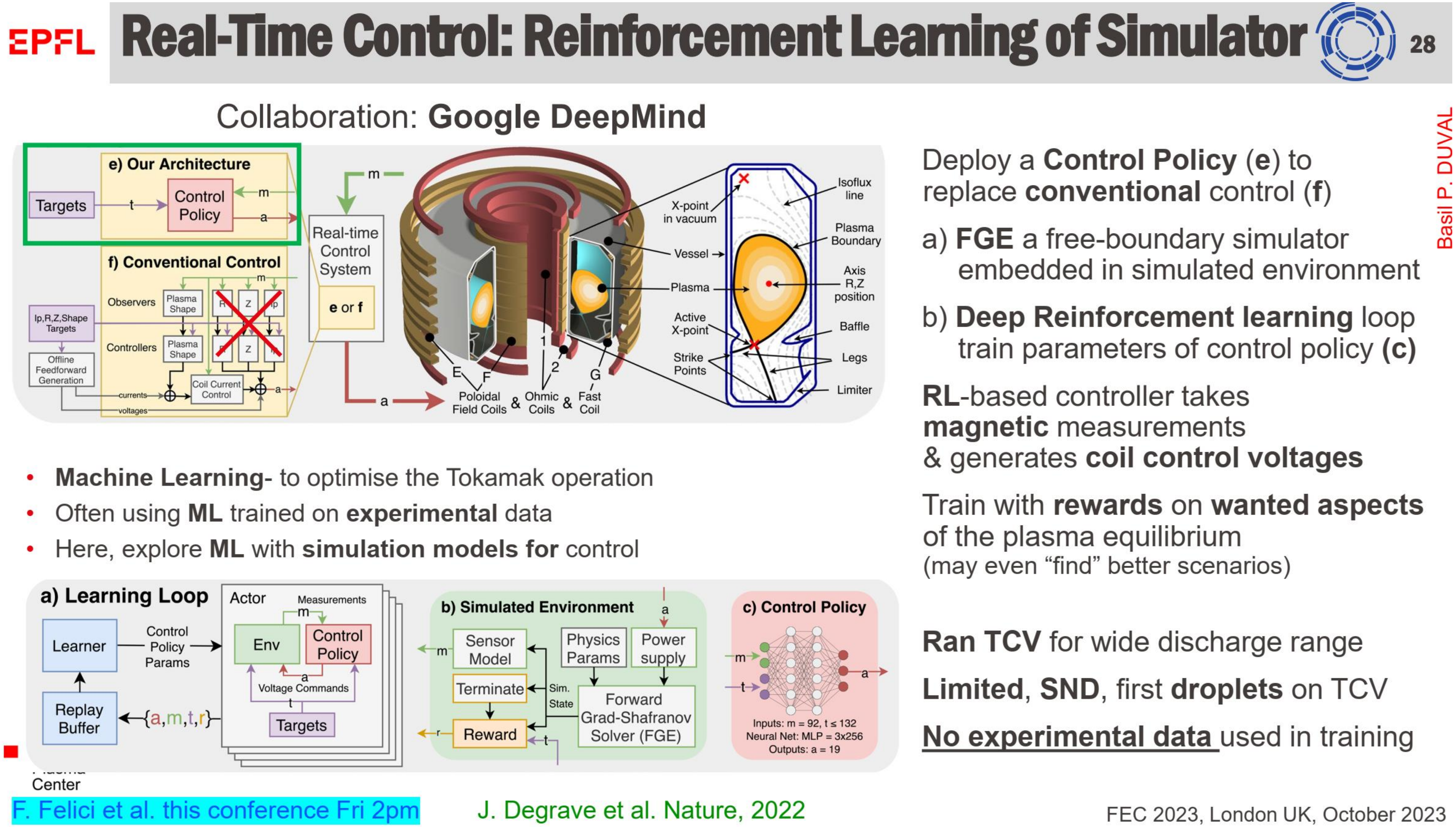
Follow Main sections from synopsis/conference-publication

- **Plasma Scenarios**
- Fast particle physics
- SOL and edge physics
- Divertor physics
- Real-Time control
- Outlook

Swiss
Plasma
Center

FEC 2023, London UK, October 2023

Tokamaks: TCV



Real-Time Control: Reinforcement Learning of Simulator

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Collaboration: **Google DeepMind**

- **Machine Learning**- to optimise the Tokamak operation
- Often using **ML** trained on **experimental** data
- Here, explore **ML** with **simulation models for** control

Deploy a **Control Policy** (**e**) to replace **conventional** control (**f**)

a) **FGE** a free-boundary simulator embedded in simulated environment

b) **Deep Reinforcement learning** loop train parameters of control policy **(c)**

RL-based controller takes **magnetic** measurements & generates **coil control voltages**

Train with **rewards** on **wanted aspects** of the plasma equilibrium (may even “find” better scenarios)

Ran TCV for wide discharge range

Limited, **SND**, first **droplets** on TCV

<u>**No experimental data**</u> used in training

Basil P. DUVAL

F. Felici et al. this conference Fri 2pm

J. Degrave et al. Nature, 2022

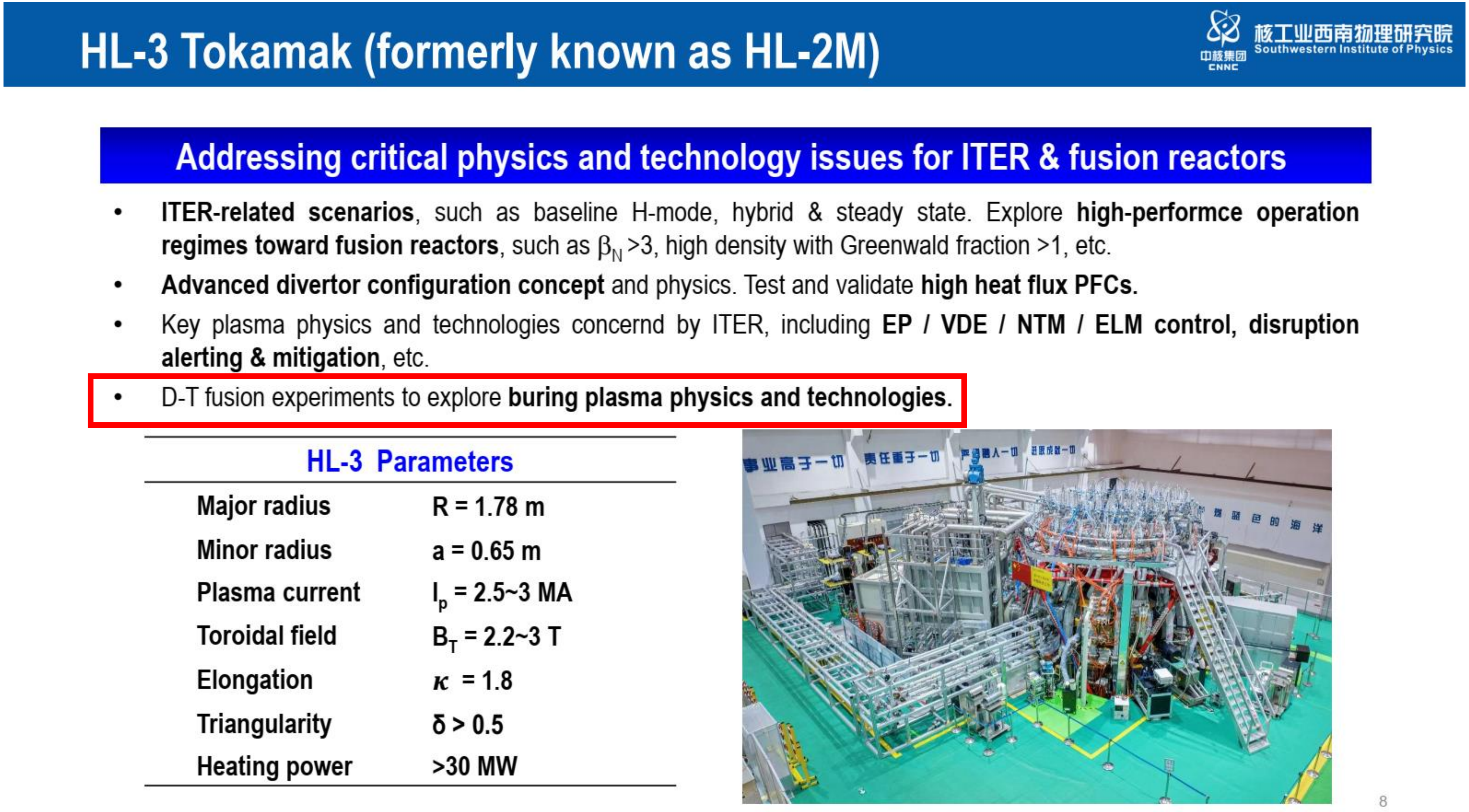
FEC 2023, London UK, October 2023

Tokamaks: HL-2M

For further details relevant to SWIP research activities, please refer to contributions:

Presenter	Session, Time	Presenter	Session, Time	Presenter	Session, Time
W.L. Zhong	EX/5-6 19/10, 15:25	W. Chen	2080, 20/10, 10:00	T. Long	2235, 20/10, 14:40
G.L. Dong	TH/8-2 20/10, 14:17	W.L. Zhong	2542, 20/10, 10:00	R.R. Ma	2135, 20/10, 16:10
J.M. Chen	TECH/2-4 18/10, 13:08	W.H. Lin	2118, 20/10, 10:00	N. Wu	2141, 20/10, 16:10
X.R. Duan	2485, 16/10, 16:14	J.Q. Xu	2081, 20/10, 10:00	T.F. Sun	2185, 20/10, 16:10
N. Zhang	2246, 17/10, 10:00	Y.P. Zhang	2120, 20/10, 10:00	P.Y. Li	2209, 20/10, 16:10
Y.R. Zhu	2191, 17/10, 10:00	M.K. Han	2129, 20/10, 10:00	G.L. Xiao	2214, 20/10, 16:10
Y. Zhang	2056, 17/10, 10:00	L.M. Yu	2180, 20/10, 10:00	D.A. Kang	2218, 20/10, 16:10
G.Z. Hao	2114, 17/10, 10:00	L. Xue	2138, 20/10, 10:00	S. Qu	2225, 20/10, 16:10
B.P. Gong	2216, 19/10, 10:00	C.Z. Chao	2219, 20/10, 10:00	J. Du	2337, 20/10, 16:10
D.M. Fan	2376, 19/10, 10:00	T. Wu	2222, 20/10, 10:00	X.J. Zhang	2370, 20/10, 16:10
J.M. Chen	2562, 19/10, 10:00	H.T. Chen	2336, 20/10, 10:00	L. Zhang	2402, 20/10, 16:10
P.W. Shi	2050, 20/10, 10:00	J.Q. Li	2131, 20/10, 13.20	B. Li	2134, 21/10., 07:50
J. Zhang	2052, 20/10, 10:00	A.S. Liang	2210, 20/10, 14:20	G.Q. Dong	2521, 21/10, 10:23

Tokamaks: HL-2M



HL-3 Tokamak (formerly known as HL-2M)

中核集团 CNNC 核工业西南物理研究院 Southwestern Institute of Physics

Addressing critical physics and technology issues for ITER & fusion reactors

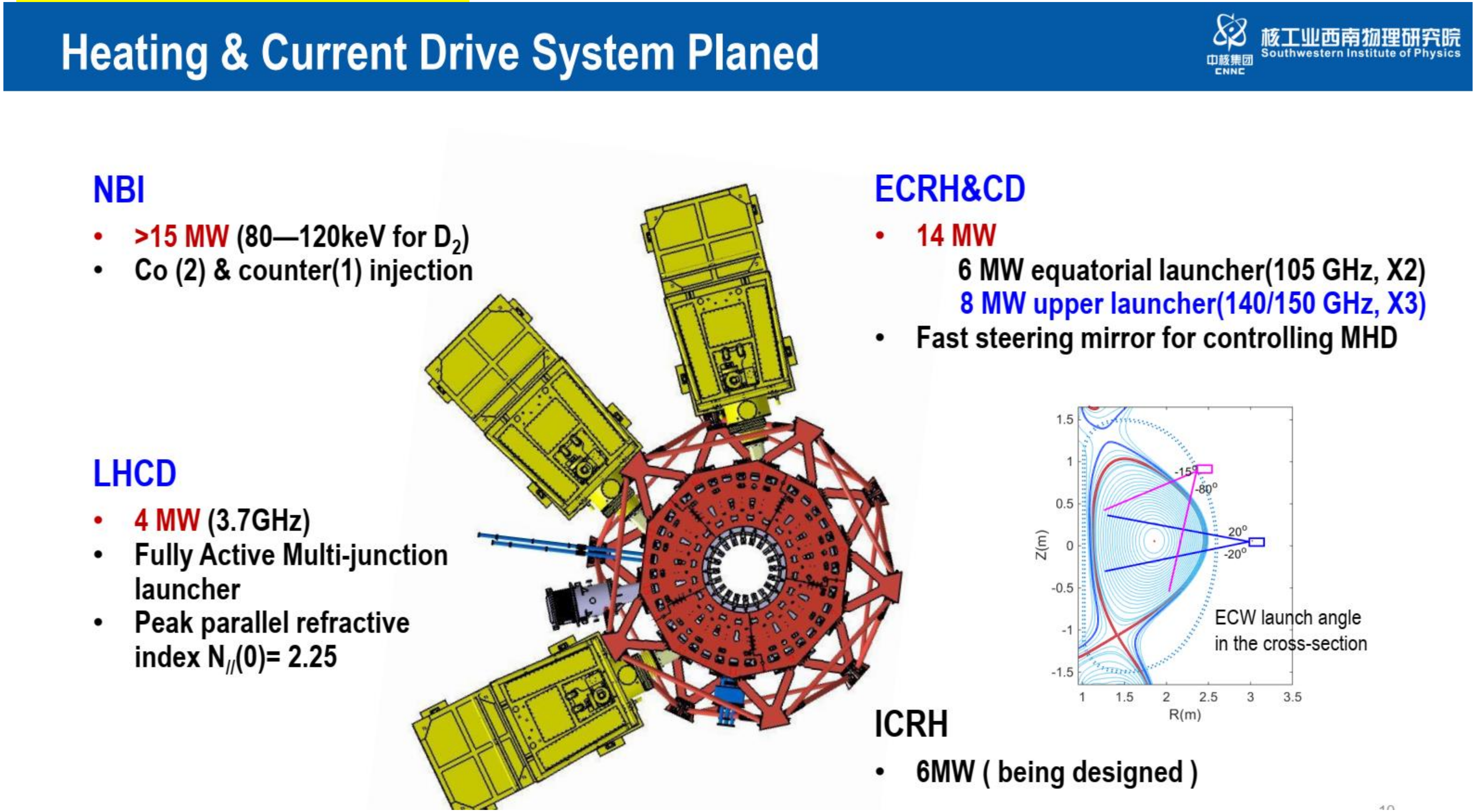
- **ITER-related scenarios**, such as baseline H-mode, hybrid & steady state. Explore **high-performce operation regimes toward fusion reactors**, such as β_N >3, high density with Greenwald fraction >1, etc.
- **Advanced divertor configuration concept** and physics. Test and validate **high heat flux PFCs.**
- Key plasma physics and technologies concernd by ITER, including **EP / VDE / NTM / ELM control, disruption alerting & mitigation**, etc.
- D-T fusion experiments to explore **buring plasma physics and technologies.**

HL-3 Parameters

Major radius	$R = 1.78$ m
Minor radius	$a = 0.65$ m
Plasma current	$I_p = 2.5\sim3$ MA
Toroidal field	$B_T = 2.2\sim3$ T
Elongation	$\kappa = 1.8$
Triangularity	$\delta > 0.5$
Heating power	>30 MW

8

Tokamaks: HL-2M



Heating & Current Drive System Planed

核工业西南物理研究院 Southwestern Institute of Physics
中核集团 CNNC

NBI

- **>15 MW** (80—120keV for D_2)
- Co (2) & counter(1) injection

LHCD

- **4 MW** (3.7GHz)
- Fully Active Multi-junction launcher
- Peak parallel refractive index $N_{//}(0)$= 2.25

ECRH&CD

- **14 MW**
  - 6 MW equatorial launcher(105 GHz, X2)
  - 8 MW upper launcher(140/150 GHz, X3)
- Fast steering mirror for controlling MHD

ECW launch angle in the cross-section

ICRH

- 6MW (being designed)

Tokamaks: HL-2M

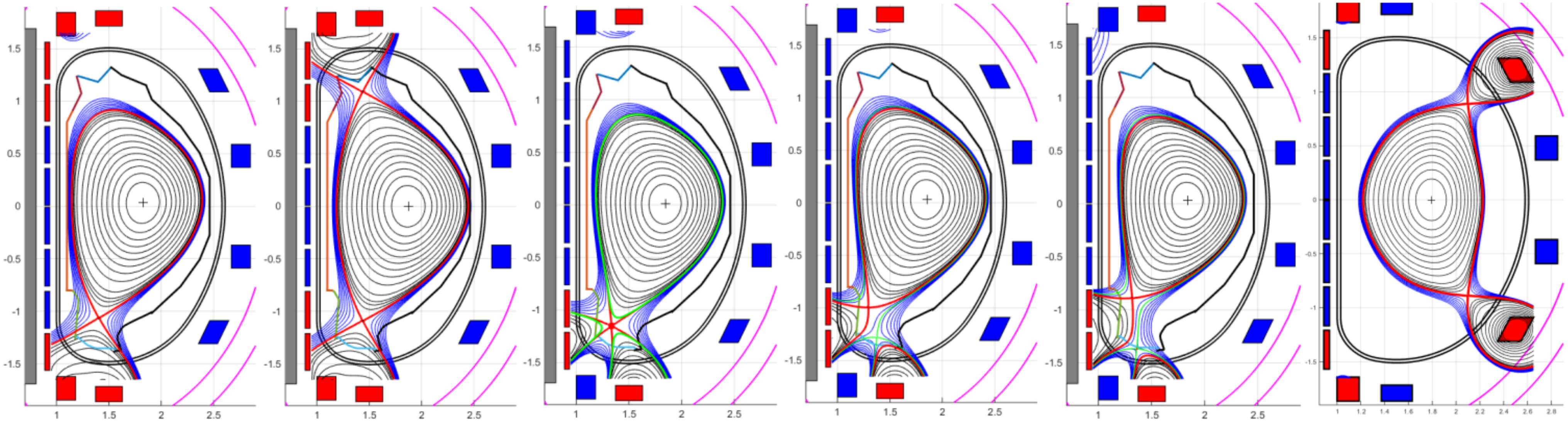
Provide Divertor High Heat Flux Solution

核工业西南物理研究院 Southwestern Institute of Physics

中核集团 CNNC

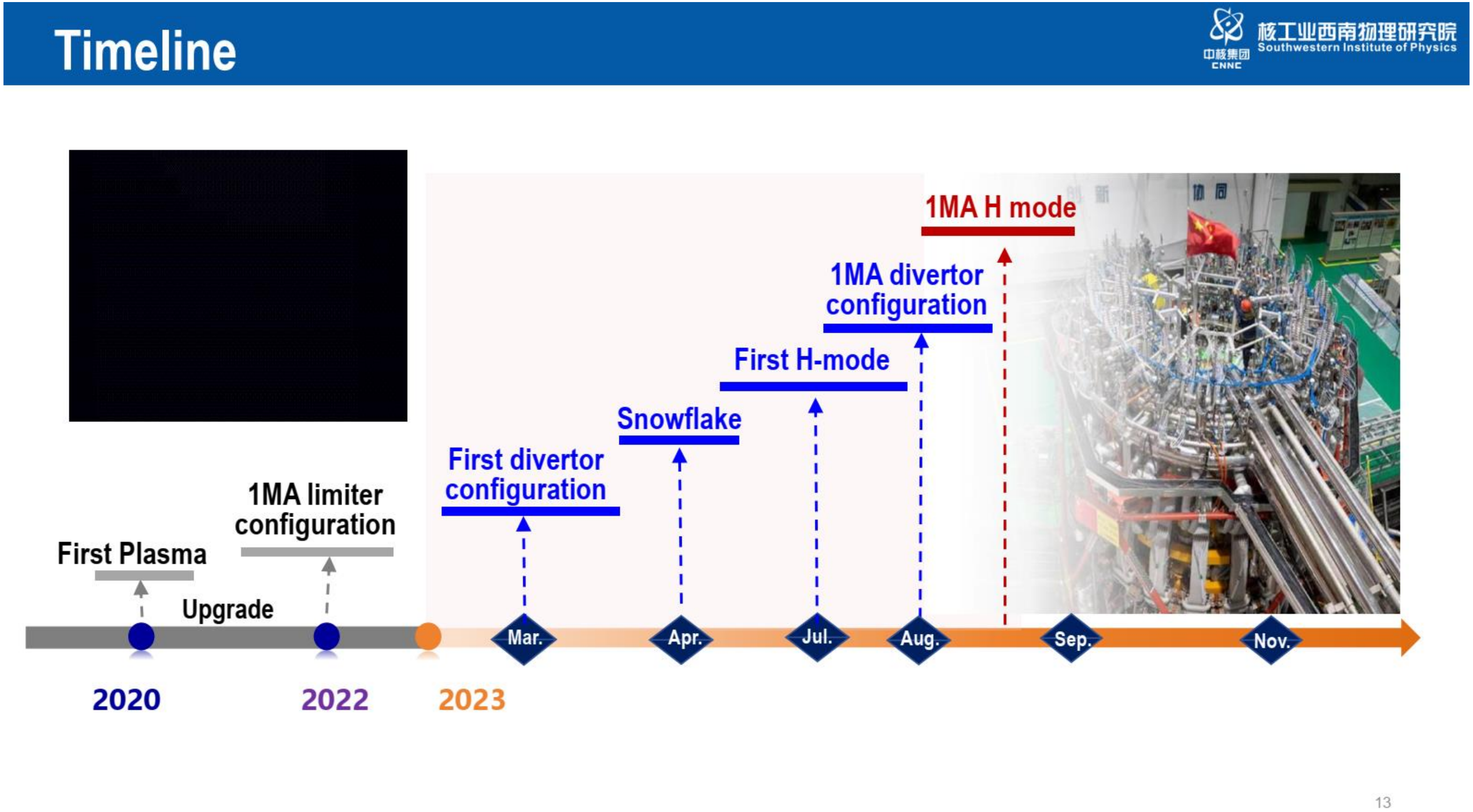
Extreme divertor heat flux handling capibility, compatible with heating >30 MW.

- Advanced divertor configuration (snowflake, tripod, fishtail, etc.)
- Detached divertor operation during ELMy H-mode



Single-null Double-null Snow flakes(SF) SF- Tripod Negtive triangularity

Tokamaks: HL-2M



Timeline

[redacted]

1MA H mode

1MA divertor configuration

First H-mode

Snowflake

First divertor configuration

1MA limiter configuration

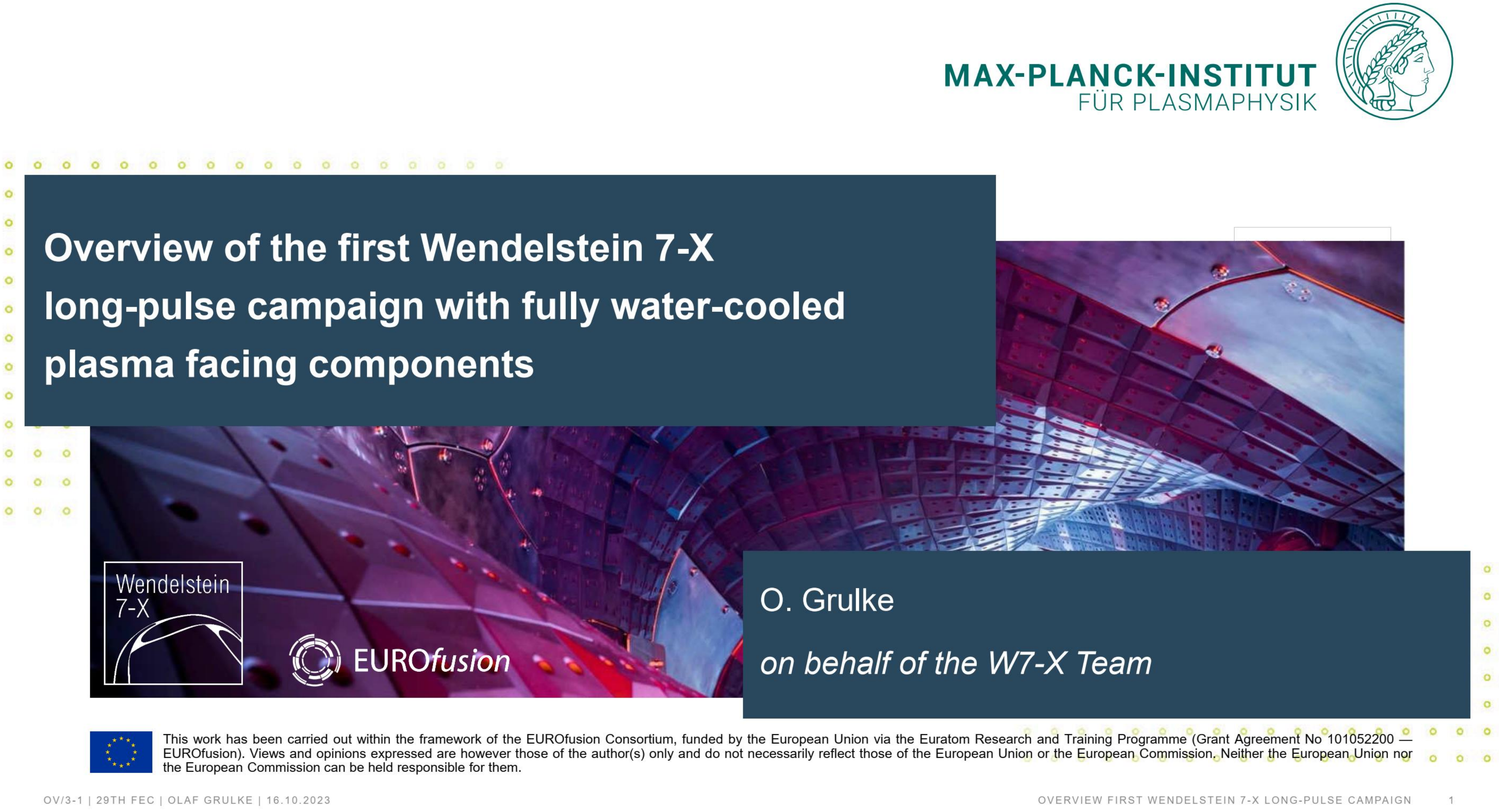
First Plasma

Upgrade

13

- Stellarators, Spherical Tokamaks, Private Sector: W7X, LHD, MAST-U, ST40

Stellarators, Spherical Tokamaks, Private Sector: W7X



Overview of the first Wendelstein 7-X long-pulse campaign with fully water-cooled plasma facing components

O. Grulke

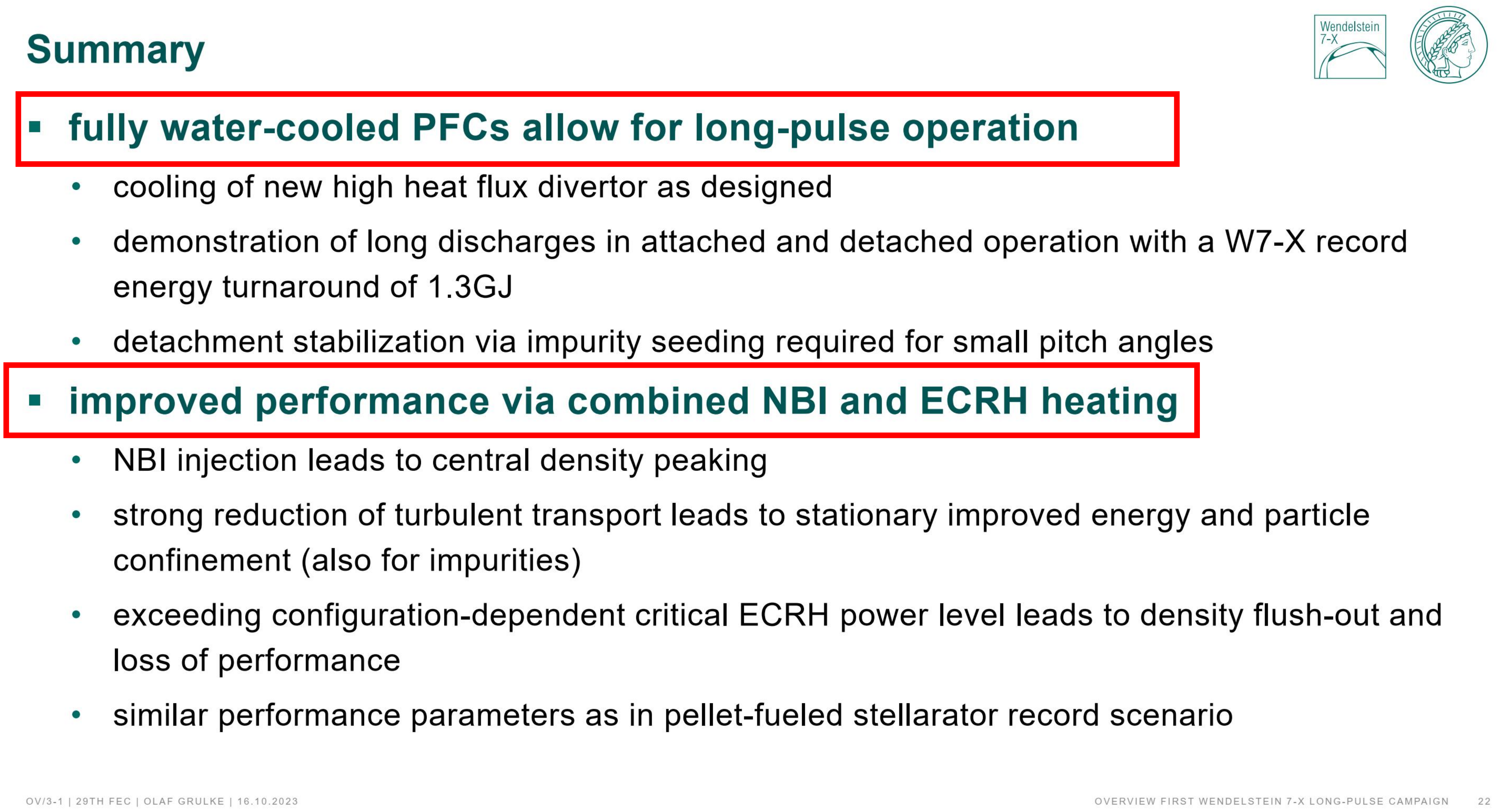
on behalf of the W7-X Team

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

OV/3-1 | 29TH FEC | OLAF GRULKE | 16.10.2023

OVERVIEW FIRST WENDELSTEIN 7-X LONG-PULSE CAMPAIGN 1

Stellarators, Spherical Tokamaks, Private Sector: W7X



Summary

- **fully water-cooled PFCs allow for long-pulse operation**
  - cooling of new high heat flux divertor as designed
  - demonstration of long discharges in attached and detached operation with a W7-X record energy turnaround of 1.3GJ
  - detachment stabilization via impurity seeding required for small pitch angles
- **improved performance via combined NBI and ECRH heating**
  - NBI injection leads to central density peaking
  - strong reduction of turbulent transport leads to stationary improved energy and particle confinement (also for impurities)
  - exceeding configuration-dependent critical ECRH power level leads to density flush-out and loss of performance
  - similar performance parameters as in pellet-fueled stellarator record scenario

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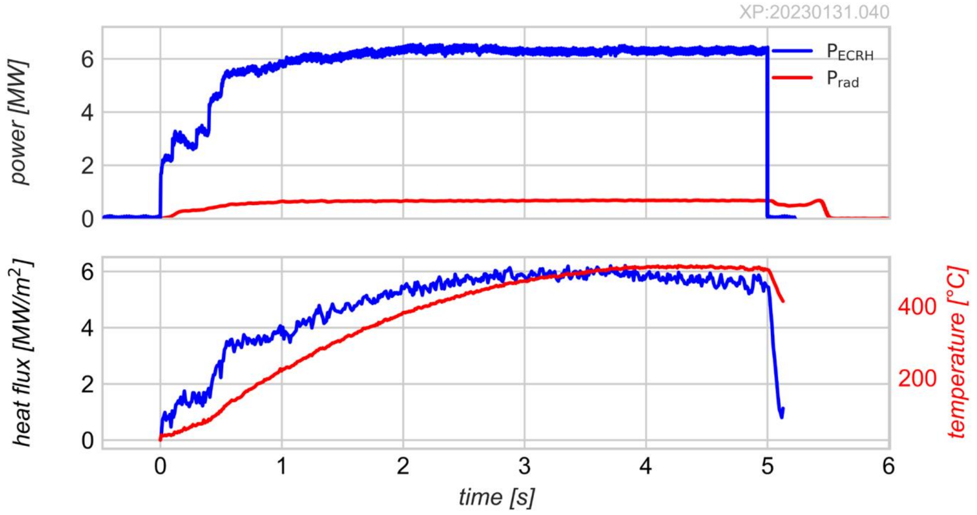
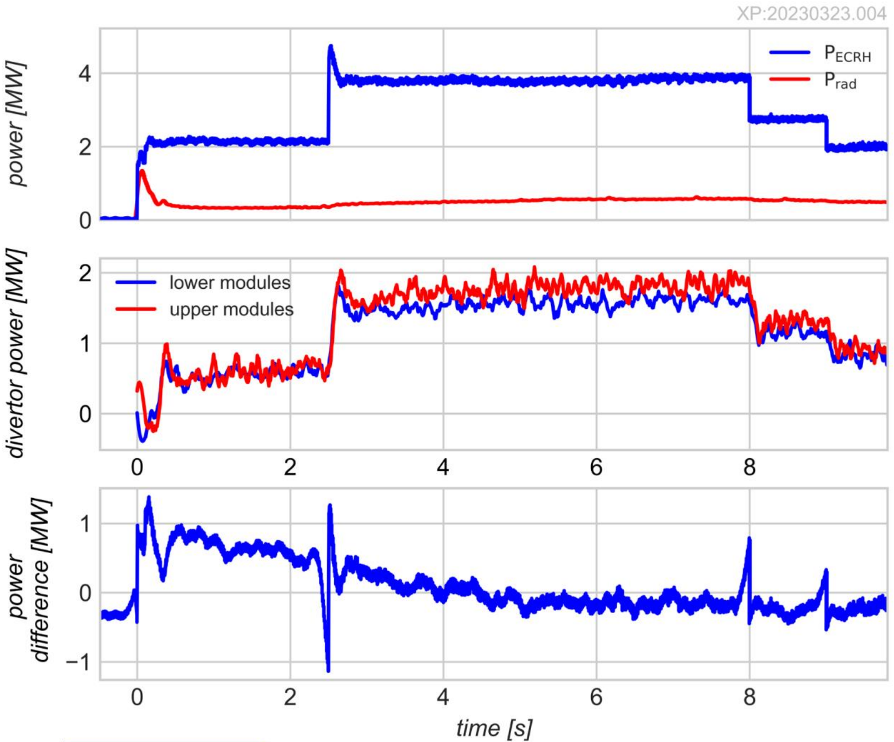
OVERVIEW FIRST WENDELSTEIN 7-X LONG-PULSE CAMPAIGN 22

Stellarators, Spherical Tokamaks, Private Sector: W7X

Performance of high heat flux divertor



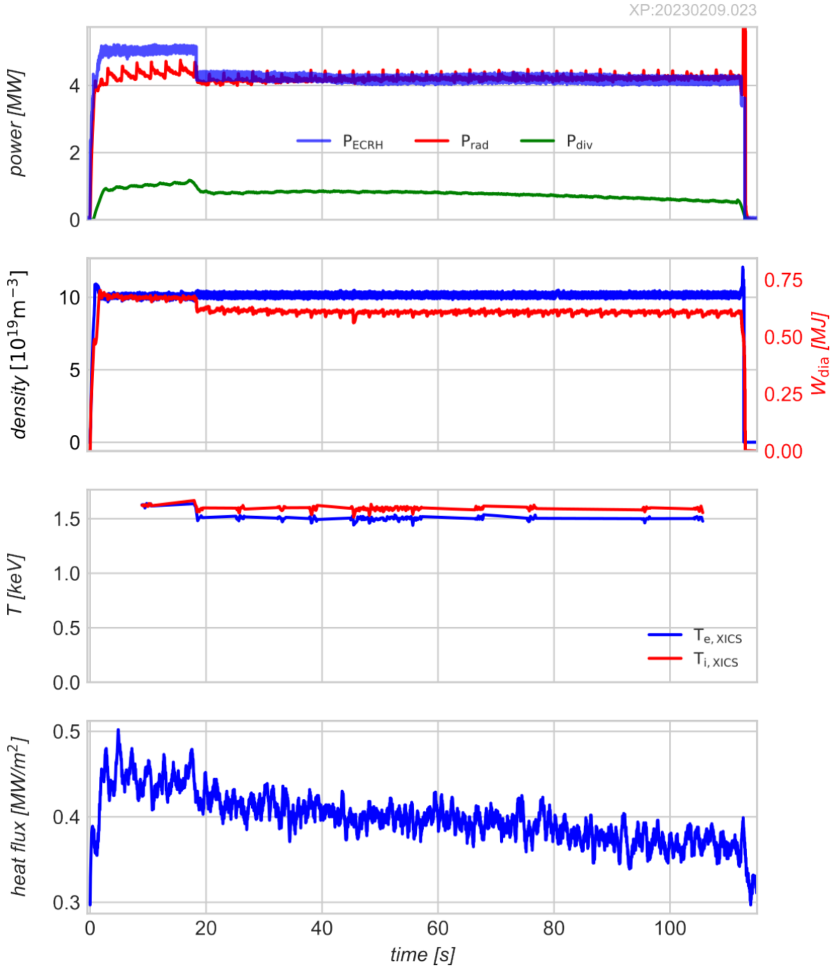


- operated at steady-state heat fluxes up to 6MW/m^2
  - ⇨ divertor surface temperature equilibrates as expected after 4-5s
- with error-field correction, no significant asymmetries in divertor power loads are observed
- little convective power loads to other PFCs

P2-1752
P6-1771
P7-1767

Stellarators, Spherical Tokamaks, Private Sector: W7X

Long-pulse detachment scenario



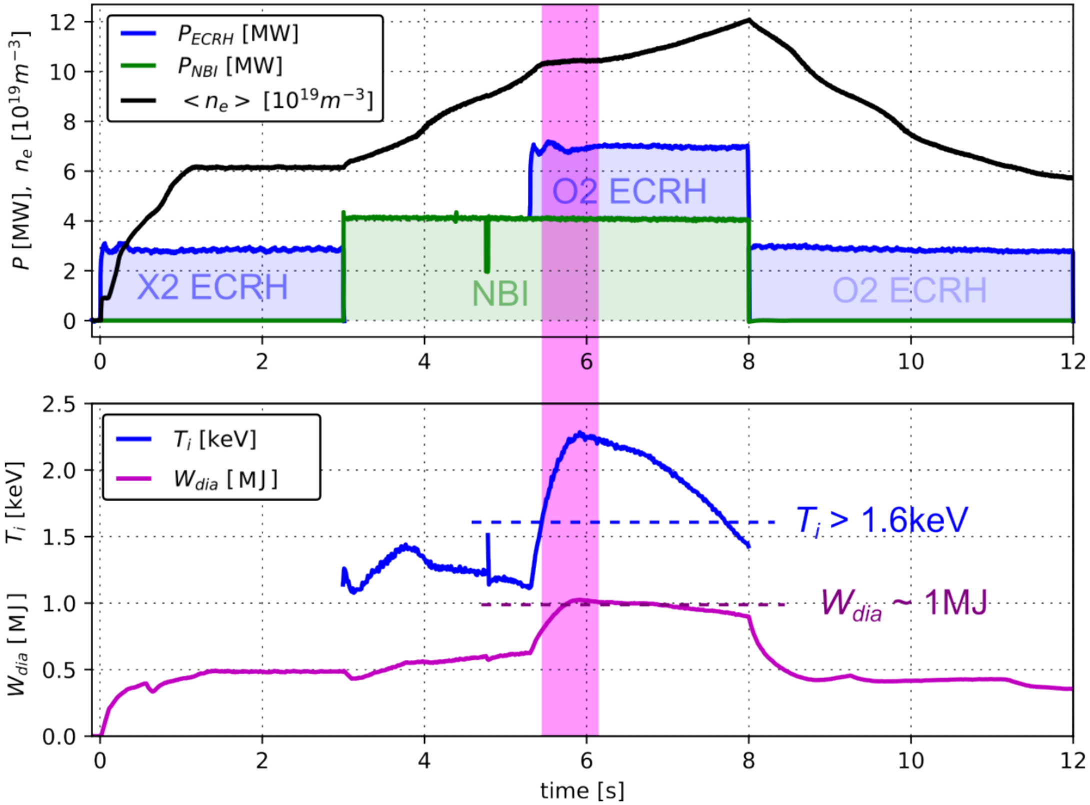
- feed-forward Ne seeding to stabilize detachment
  - ➪ puff of ~$8 \cdot 10^{17}$ Ne atoms every 2.5s to keep radiated power constant at f_{rad}=0.8
  - ➪ no influence on density control or core plasma parameters (Z_{eff}=1.5 throughout entire discharge)
- strongly reduced divertor heat flux at P_{heat}≈1MW/m^2
- programmed pulse length of 110s achieved
  - ➪ stability of scenario allows for much longer pulse lengths (will be addressed in upcoming experimental campaign)
  - ➪ it is foreseen to introduce feedback on radiative power levels to counteract changes in impurity radiation changes in long pulse scenarios

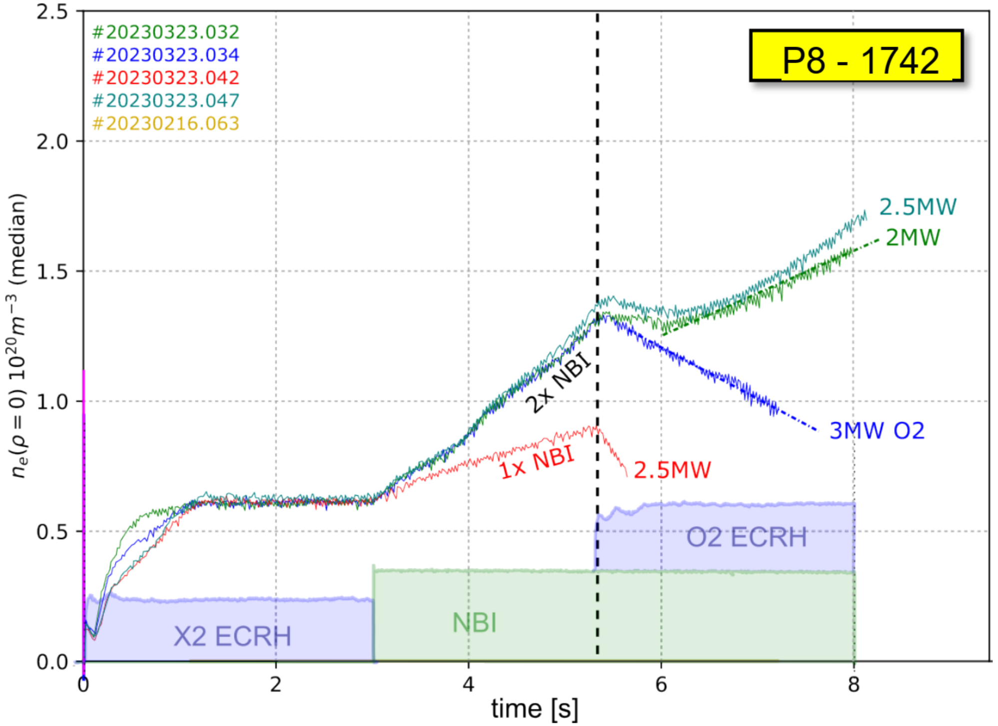
P2-1764

Stellarators. Spherical Tokamaks. Private Sector: W7X

Development of high-performance scenarios

XP: 20230323.047



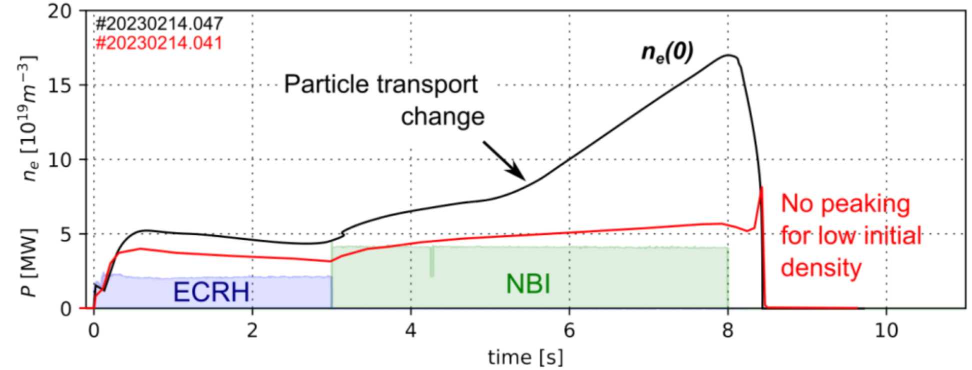


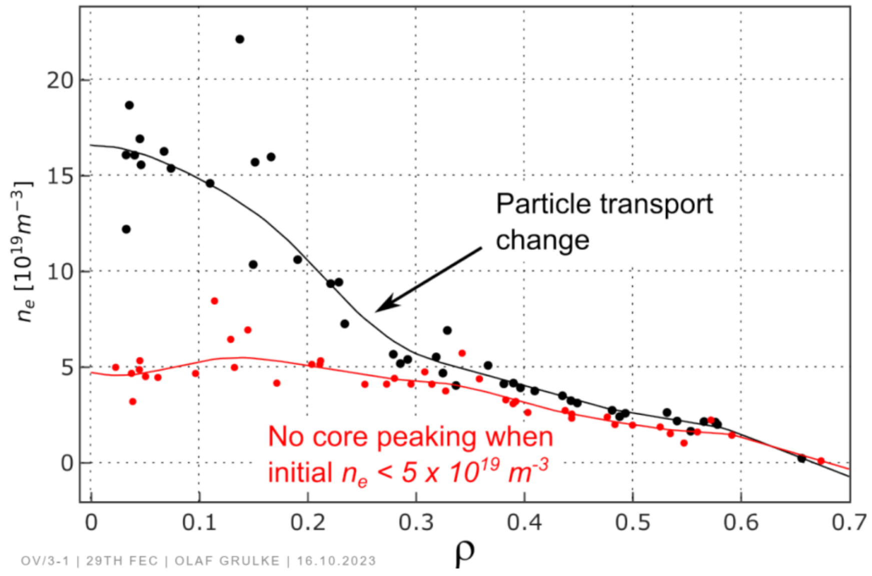
O.P. Ford et al, Nucl. Fusion (submitted)

- combined NBI and ECRH heating in O2 polarization leads to exceptionally high performance parameters
  - ⇨ stationary increase of ion temperatures, stored energy, confinement times
  - ⇨ however, critical (magnetic configuration dependent) ECRH power level observed

Stellarators, Spherical Tokamaks, Private Sector: W7X

Performance associated with plasma density peaking





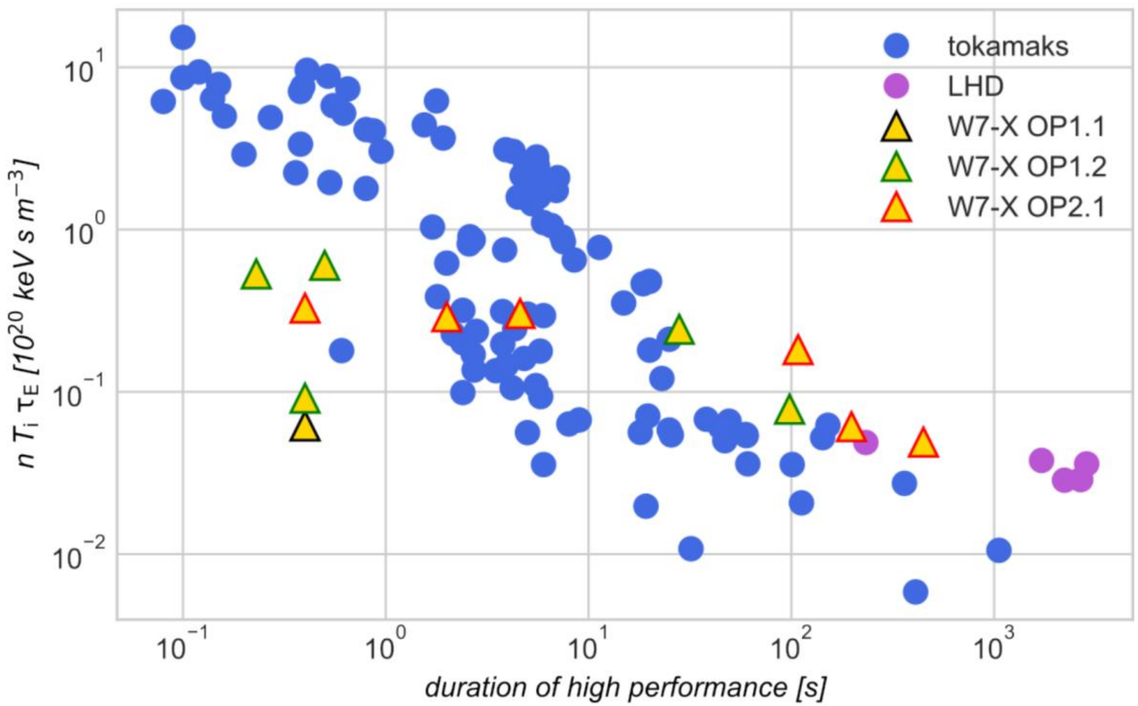
- NBI injection leads to central plasma density peaking
  - ➩ minimum initial plasma density required
  - ➩ peaking of density profile within $\rho<0.5$
- central density gradients considerably reduce turbulent transport
  - ➩ support of previous observations using pellet fueling
  - ➩ impurity transport reduces to neoclassical level; associated accumulation in the plasma center
- loss of improved confinement associated with an increase of turbulent fluctuations
  - ➩ causality not clear and requires results from global turbulence simulation

OV/3-1 | 29TH FEC | OLAF GRULKE | 16.10.2023

OVERVIEW FIRST WENDELSTEIN 7-X LONG-PULSE CAMPAIGN 17

Stellarators, Spherical Tokamaks, Private Sector: W7X

Figure of merit



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OVERVIEW FIRST WENDELSTEIN 7-X LONG-PULSE CAMPAIGN 21

Stellarators, Spherical Tokamaks, Private Sector: LHD

Conclusion

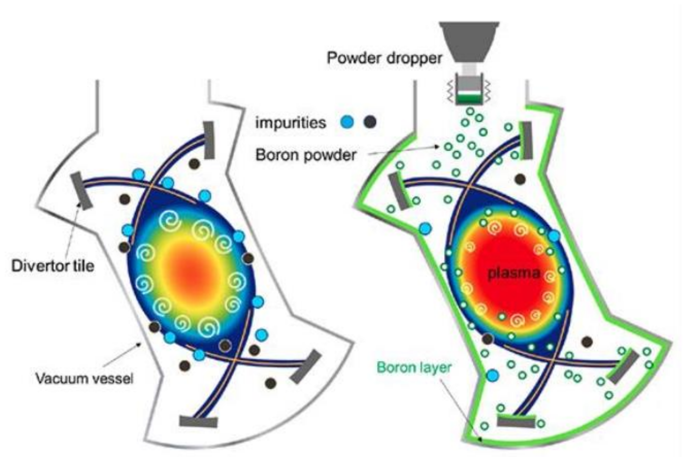
Significant progress in scientific research has been made in experiments on turbulence/transport, magnetic island, energetic particle based on international and domestic collaborations.

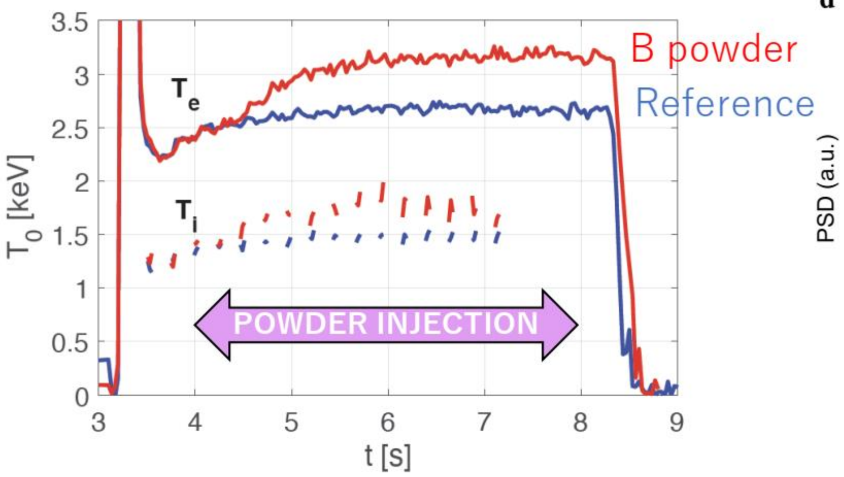
LHD experiments provide the following important observations on
1) Collision-less energy transfer via Landau and transit-time damping
2) Isotope mixing and isotope effect on transport
3) Reduction of peak heat load on diverter plate by turbulence spreading and detachment enhanced by magnetic island
4) Turbulence propagation faster than avalanche model prediction
5) Deep particle fueling with suppression of plasmoid drift by neon-doped pellet
6) Improvement of confinement by boron power drop
7) First trial of $p^{11}B$ fusion

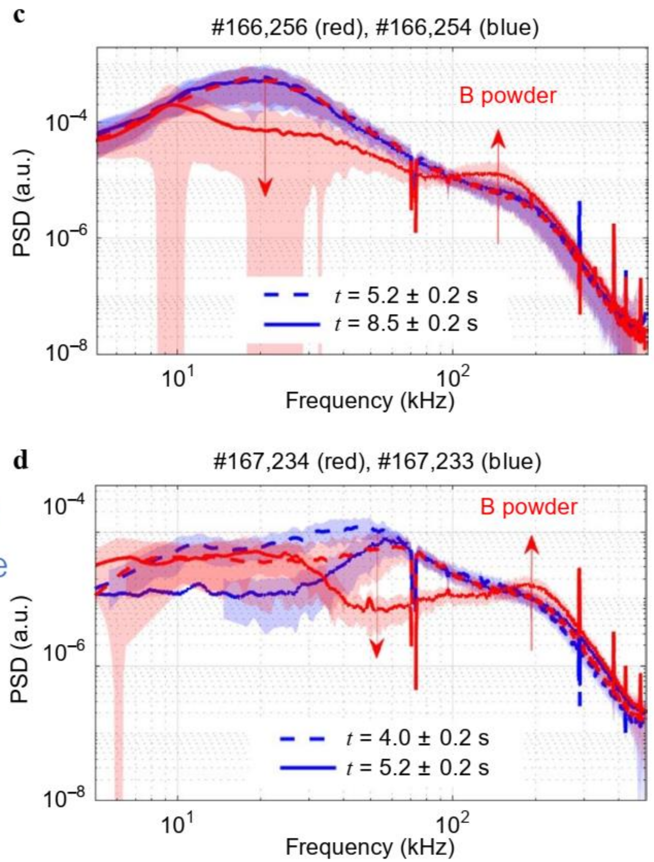
These results give new insights for solving the physics issue of burning plasma.

Stellarators, Spherical Tokamaks, Private Sector: LHD

Boron powder injection makes plasma temperature increase







During B powder injection, plasma electron and ion temperatures increase, along with stored energy and confinement time

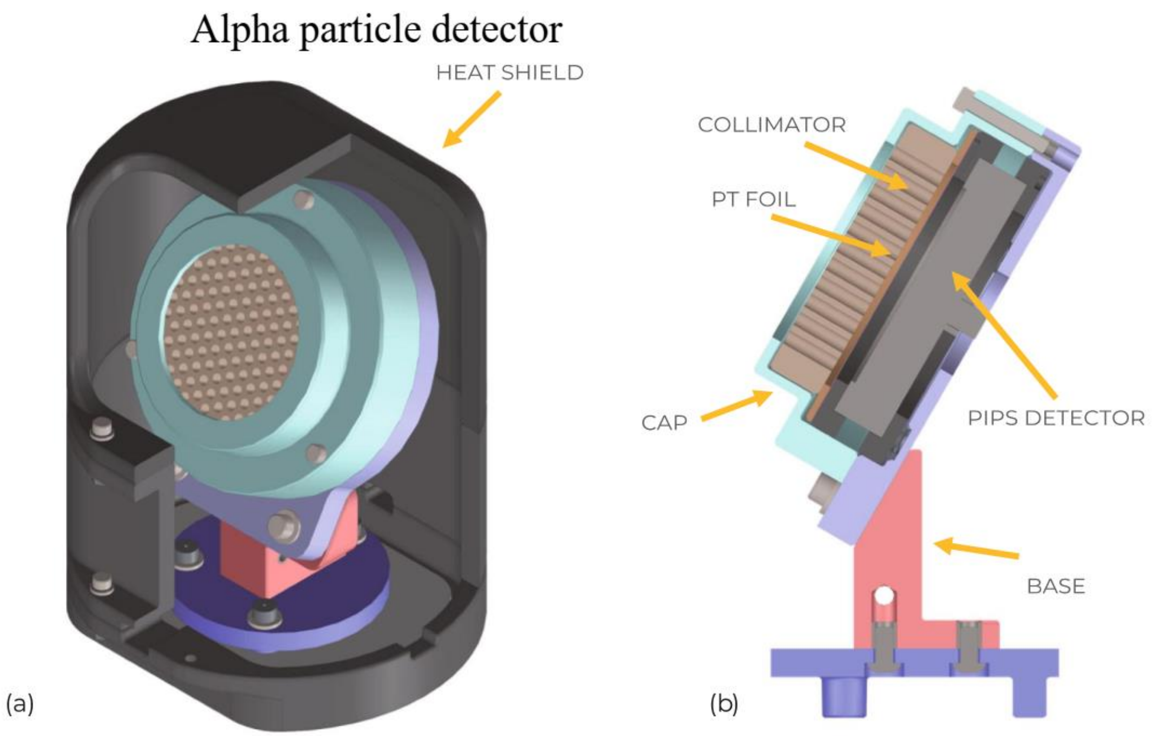
Turbulence fluctuation amplitude during injection period is reduced at low frequency (< 100kHz) regime

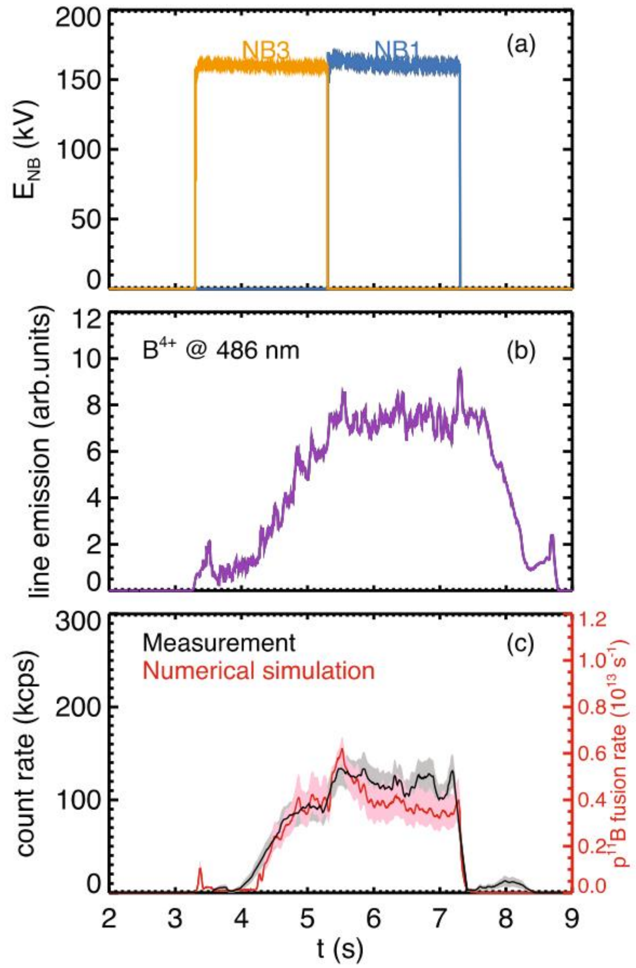
F. Nespoli et al., Nat Phys 18 (2022) 350

F.Nespoli IAEA-FEC 10/19 14:51 EX/5-4

Stellarators, Spherical Tokamaks, Private Sector: LHD

First measurements of p^{11}B fusion in a magnetically confined plasma

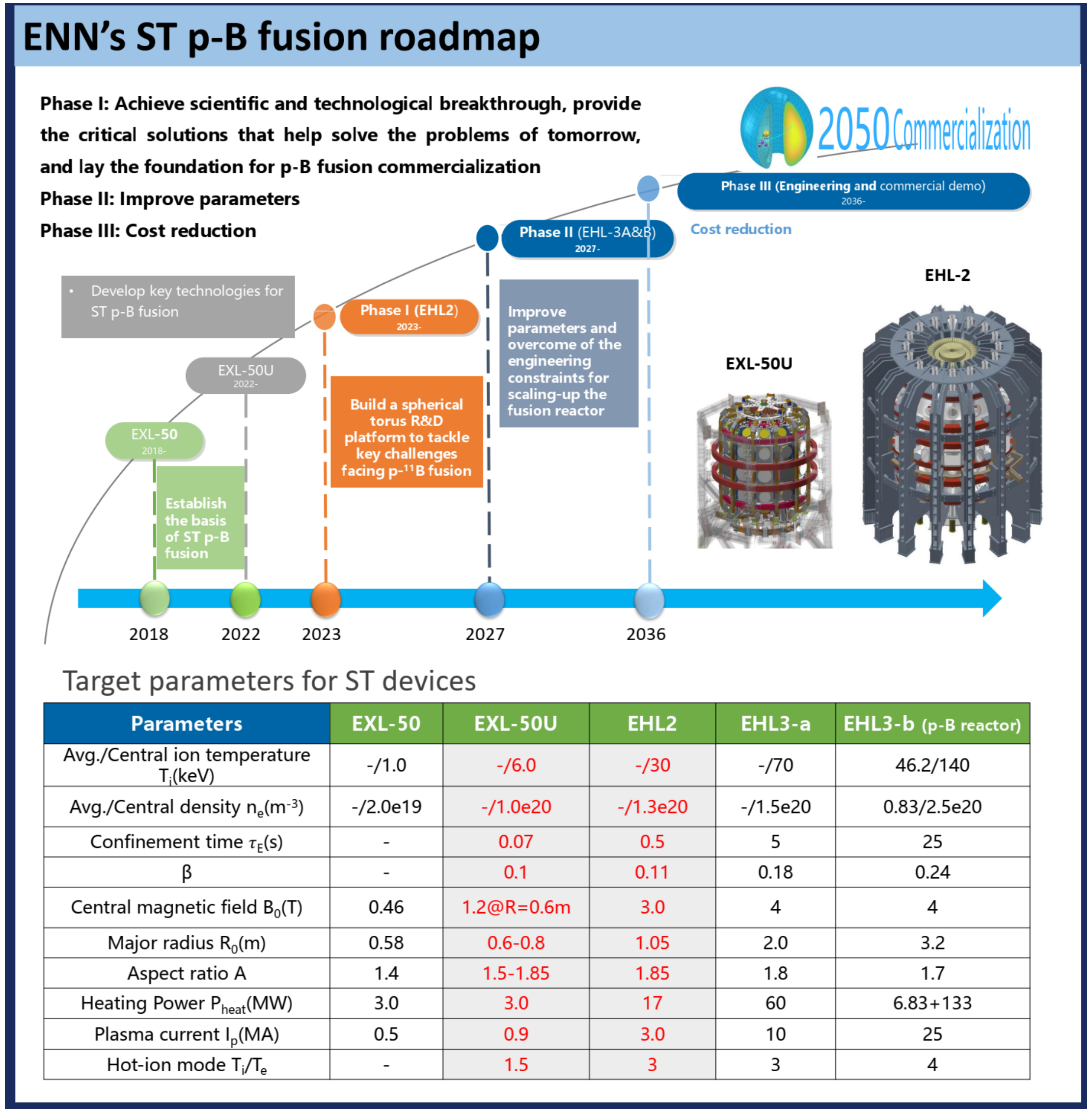




Significant levels of alpha particle emission is observed when Boron powder is injected to LHD plasma with 160keV hydrogen beam. The measured flux is consistent with that predicted by numerical simulation.

R,M,. Magee, et. al., Nat. Commun. 14 (2023) 955

Реактор p-B на базе сферического токамака в Китае к 2036



ENN’s ST p-B fusion roadmap

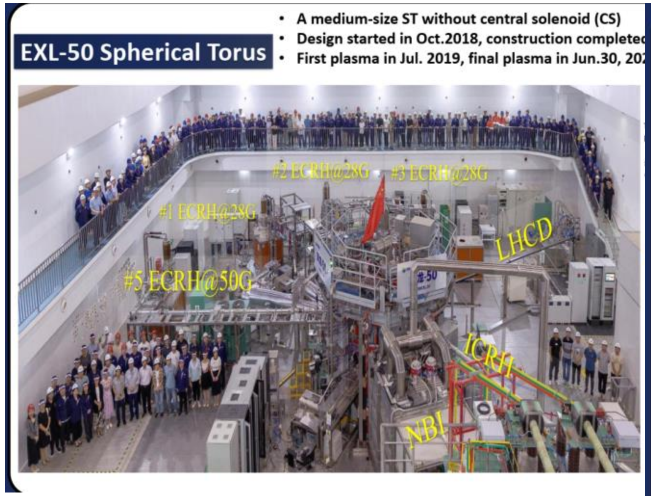
Phase I: Achieve scientific and technological breakthrough, provide the critical solutions that help solve the problems of tomorrow, and lay the foundation for p-B fusion commercialization

Phase II: Improve parameters

Phase III: Cost reduction

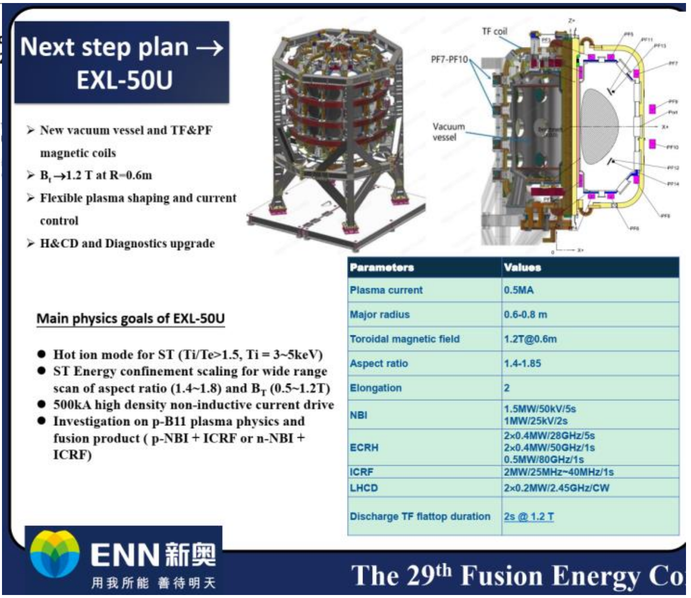
Target parameters for ST devices

Parameters	EXL-50	EXL-50U	EHL2	EHL3-a	EHL3-b (p-B reactor)
Avg./Central ion temperature T_i(keV)	-/1.0	-/6.0	-/30	-/70	46.2/140
Avg./Central density n_e(m^{-3})	-/2.0e19	-/1.0e20	-/1.3e20	-/1.5e20	0.83/2.5e20
Confinement time τ_E(s)	-	0.07	0.5	5	25
β	-	0.1	0.11	0.18	0.24
Central magnetic field B_0(T)	0.46	1.2@R=0.6m	3.0	4	4
Major radius R_0(m)	0.58	0.6-0.8	1.05	2.0	3.2
Aspect ratio A	1.4	1.5-1.85	1.85	1.8	1.7
Heating Power P_{heat}(MW)	3.0	3.0	17	60	6.83+133
Plasma current I_p(MA)	0.5	0.9	3.0	10	25
Hot-ion mode T_i/T_e	-	1.5	3	3	4



EXL-50 Spherical Torus

- A medium-size ST without central solenoid (CS)
- Design started in Oct.2018, construction complete
- First plasma in Jul. 2019, final plasma in Jun.30, 20



Next step plan → EXL-50U

- New vacuum vessel and TF&PF magnetic coils
- $B_t \rightarrow 1.2$ T at R=0.6m
- Flexible plasma shaping and current control
- H&CD and Diagnostics upgrade

Main physics goals of EXL-50U

- Hot ion mode for ST (Ti/Te>1.5, Ti = 3~5keV)
- ST Energy confinement scaling for wide range scan of aspect ratio (1.4~1.8) and B_T (0.5~1.2T)
- 500kA high density non-inductive current drive
- Investigation on p-B11 plasma physics and fusion product (p-NBI + ICRF or n-NBI + ICRF)

Parameters	Values
Plasma current	0.5MA
Major radius	0.6-0.8 m
Toroidal magnetic field	1.2T@0.6m
Aspect ratio	1.4-1.85
Elongation	2
NBI	1.5MW/50kV/5s 1MW/25kV/2s
ECRH	2×0.4MW/28GHz/5s 2×0.4MW/50GHz/1s 0.5MW/80GHz/1s
ICRF	2MW/25MHz~40MHz/1s
LHCD	2×0.2MW/2.45GHz/CW
Discharge TF flattop duration	2s @ 1.2 T

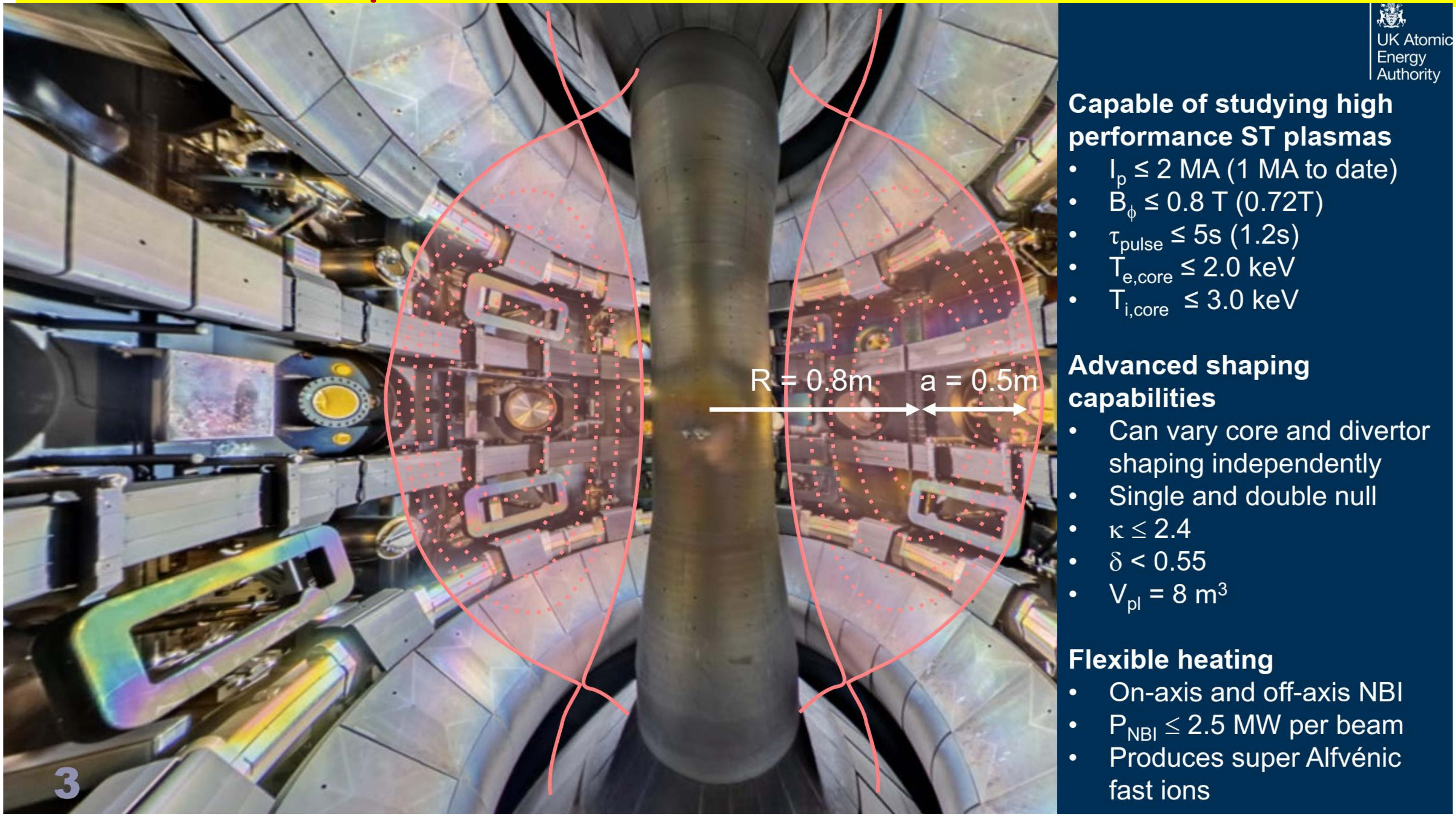
ENN 新奥
用我所能 善待明天

The 29th Fusion Energy Co

2178. ENN's Roadmap for Proton-Boron Fusion Based on Spherical Torus Minsheng Liu

1641. Overview of EXL-50 Research Progress and Future Plan Prof. Yuejiang Shi

Stellarators, Spherical Tokamaks, Private Sector: MAST-U



Capable of studying high performance ST plasmas

- $I_p \leq 2$ MA (1 MA to date)
- $B_\phi \leq 0.8$ T (0.72T)
- $\tau_{pulse} \leq 5$s (1.2s)
- $T_{e,core} \leq 2.0$ keV
- $T_{i,core} \leq 3.0$ keV

Advanced shaping capabilities

- Can vary core and divertor shaping independently
- Single and double null
- $\kappa \leq 2.4$
- $\delta < 0.55$
- $V_{pl} = 8\ m^3$

Flexible heating

- On-axis and off-axis NBI
- $P_{NBI} \leq 2.5$ MW per beam
- Produces super Alfvénic fast ions

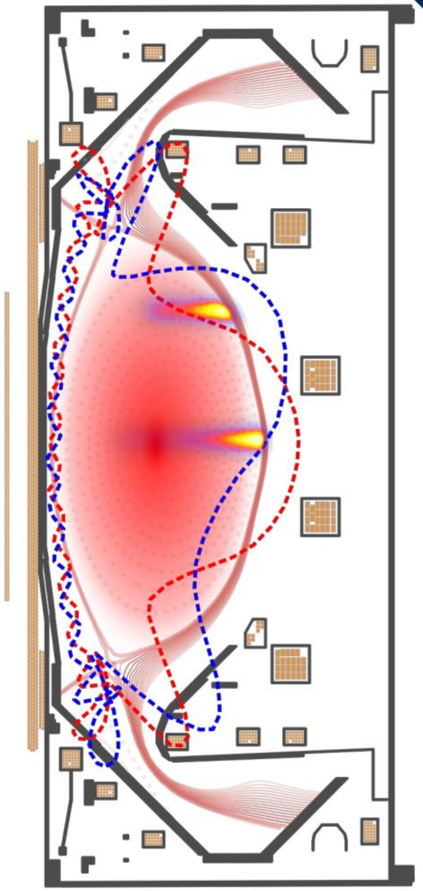
Stellarators, Spherical Tokamaks, Private Sector: MAST-U

Outline

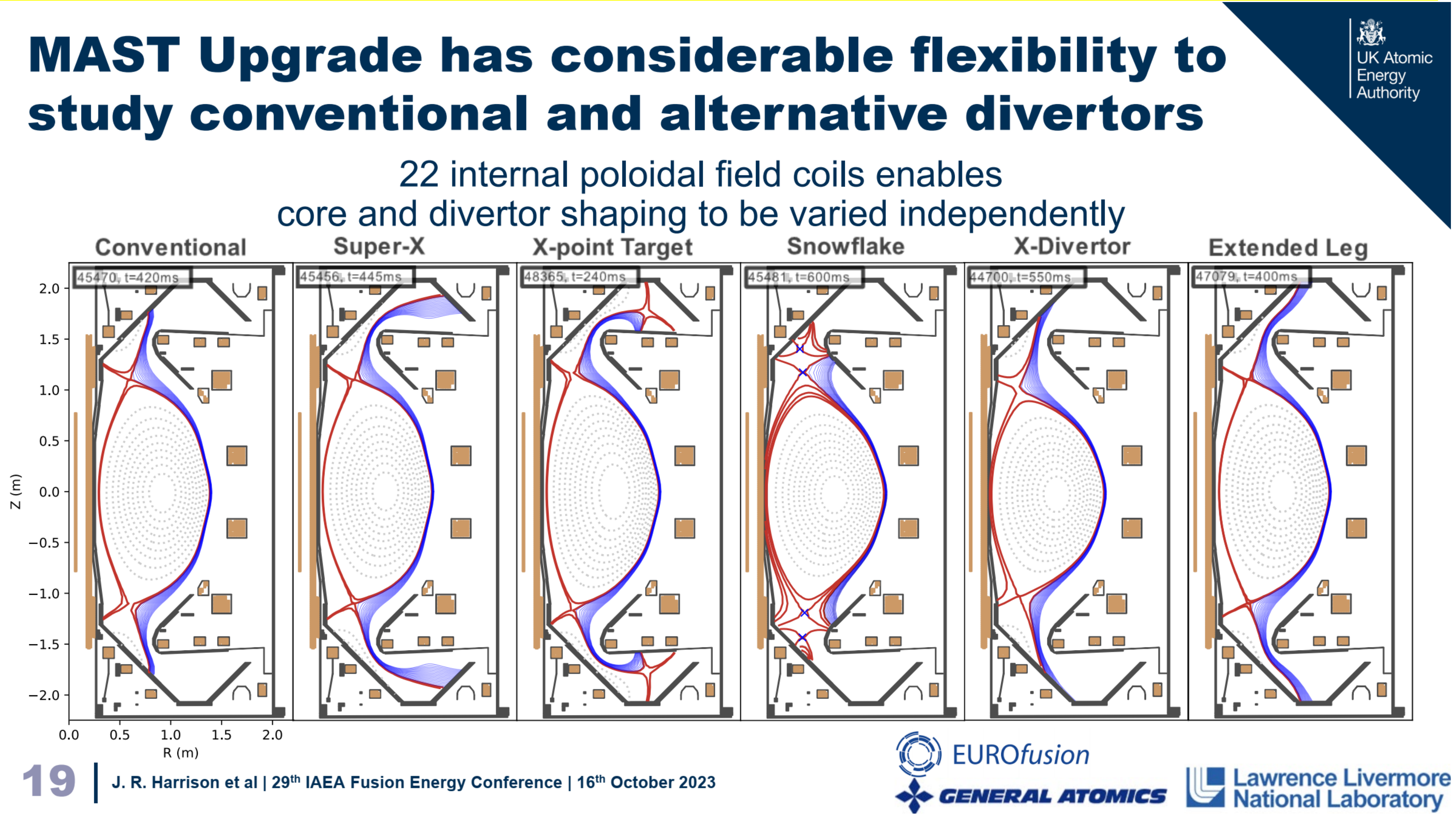
UK Atomic Energy Authority

MAST Upgrade results deepen our understanding of key physics issues for the operation of ITER and design of future fusion power plants:

- Fast ion confinement
- Avoidance of MHD instabilities
- Maximising pedestal confinement
- Mitigating divertor power and particle loads
- **Scenario integration with alternative divertor configurations**



Stellarators, Spherical Tokamaks, Private Sector: MAST-U



MAST Upgrade has considerable flexibility to study conventional and alternative divertors

UK Atomic Energy Authority

22 internal poloidal field coils enables
core and divertor shaping to be varied independently

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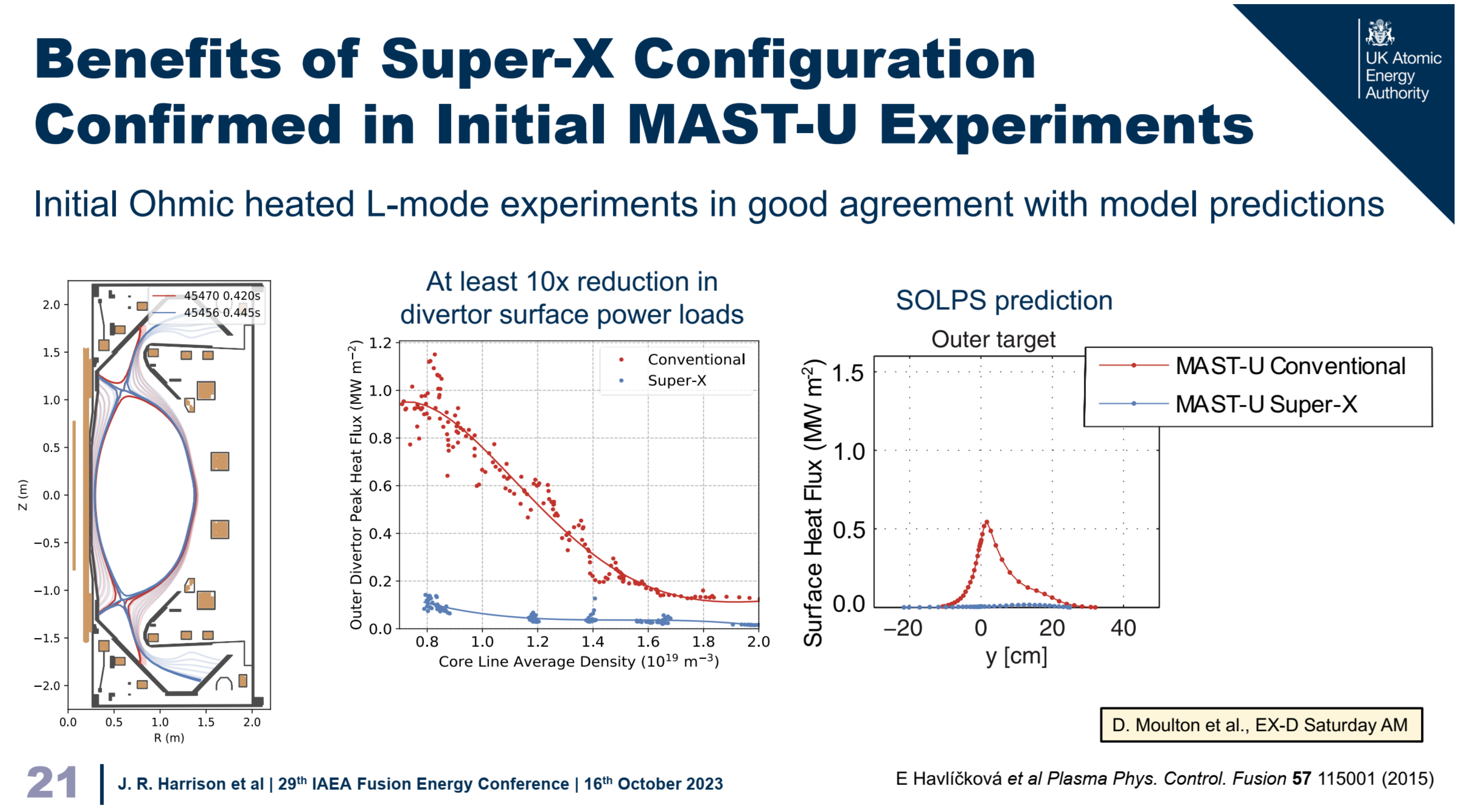
GENERAL ATOMICS
Lawrence Livermore National Laboratory

Stellarators, Spherical Tokamaks, Private Sector: MAST-U



Benefits of Super-X Configuration Confirmed in Initial MAST-U Experiments

Initial Ohmic heated L-mode experiments in good agreement with model predictions

At least 10x reduction in divertor surface power loads

SOLPS prediction

D. Moulton et al., EX-D Saturday AM

E Havlíčková *et al Plasma Phys. Control. Fusion* **57** 115001 (2015)

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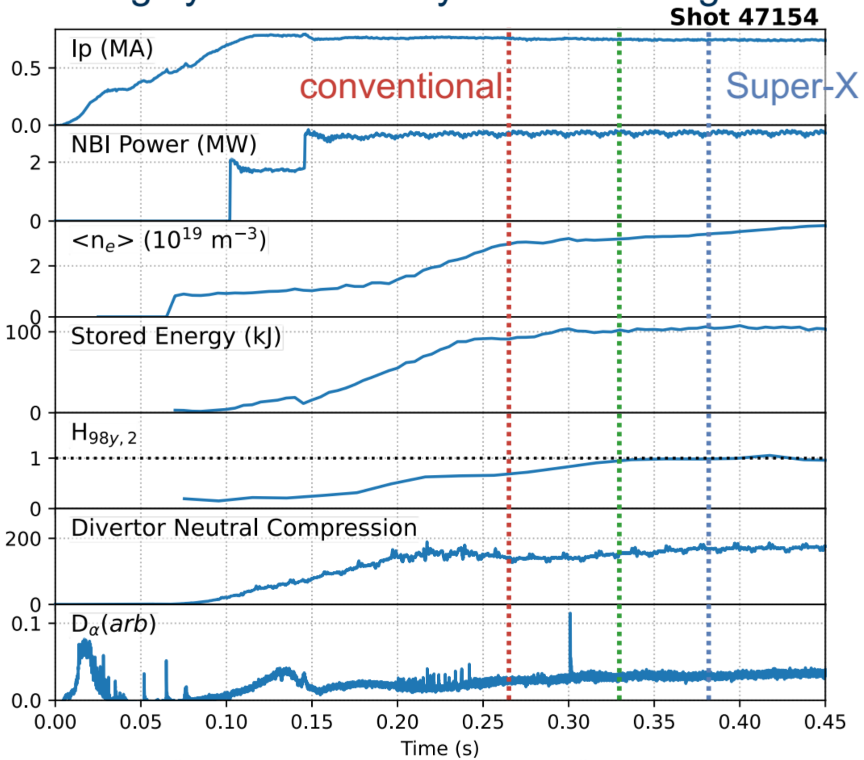
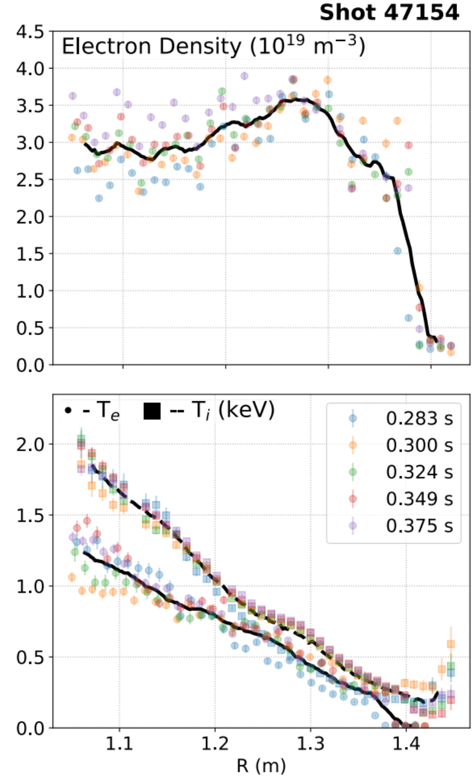
Stellarators, Spherical Tokamaks, Private Sector: MAST-U

UK Atomic Energy Authority

Confinement Maintained in H-mode With Detached Divertors in Super-X

Core profiles largely unaffected by divertor configuration within natural evolution between ELMs





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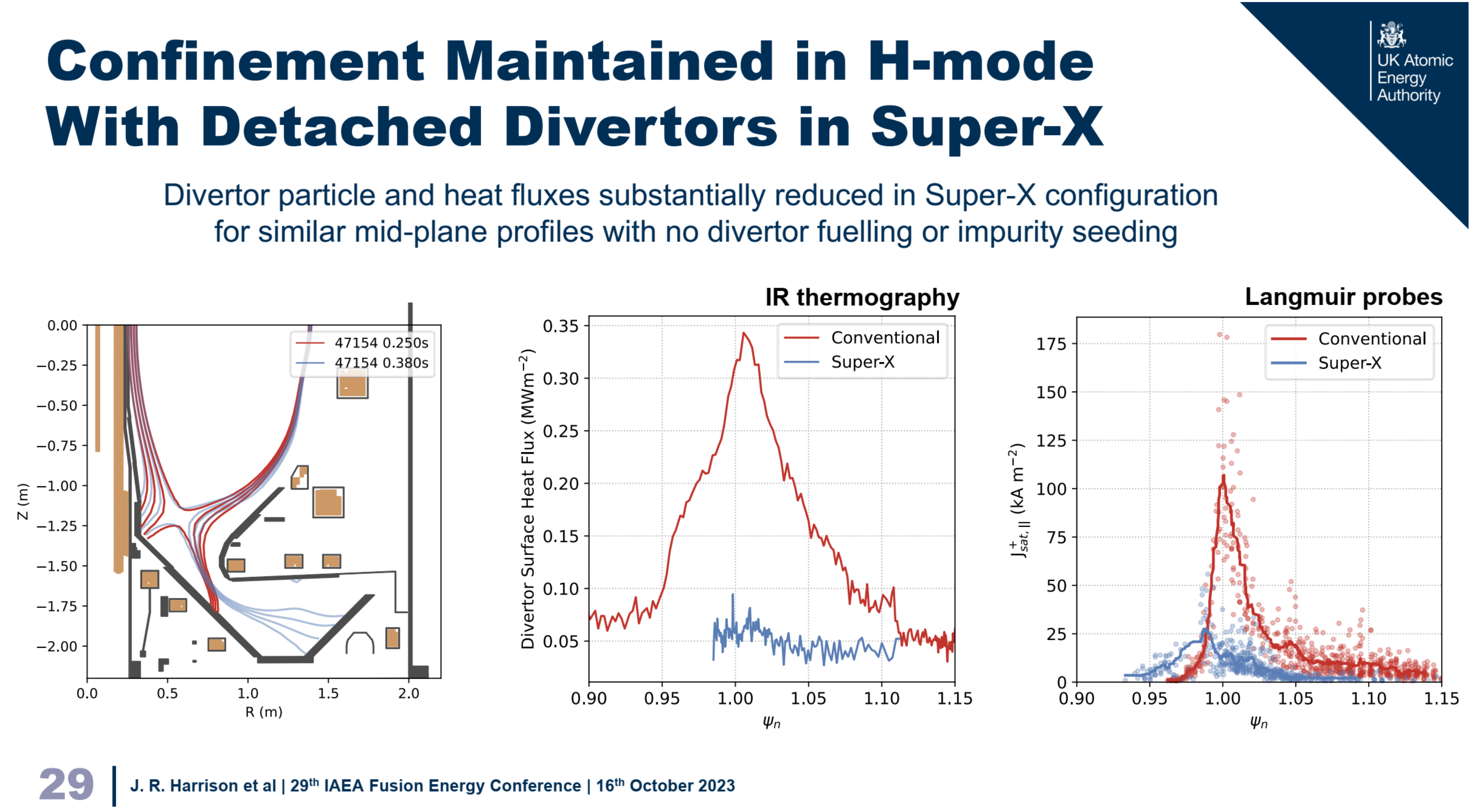
Stellarators, Spherical Tokamaks, Private Sector: MAST-U



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Confinement Maintained in H-mode With Detached Divertors in Super-X

Divertor particle and heat fluxes substantially reduced in Super-X configuration for similar mid-plane profiles with no divertor fuelling or impurity seeding

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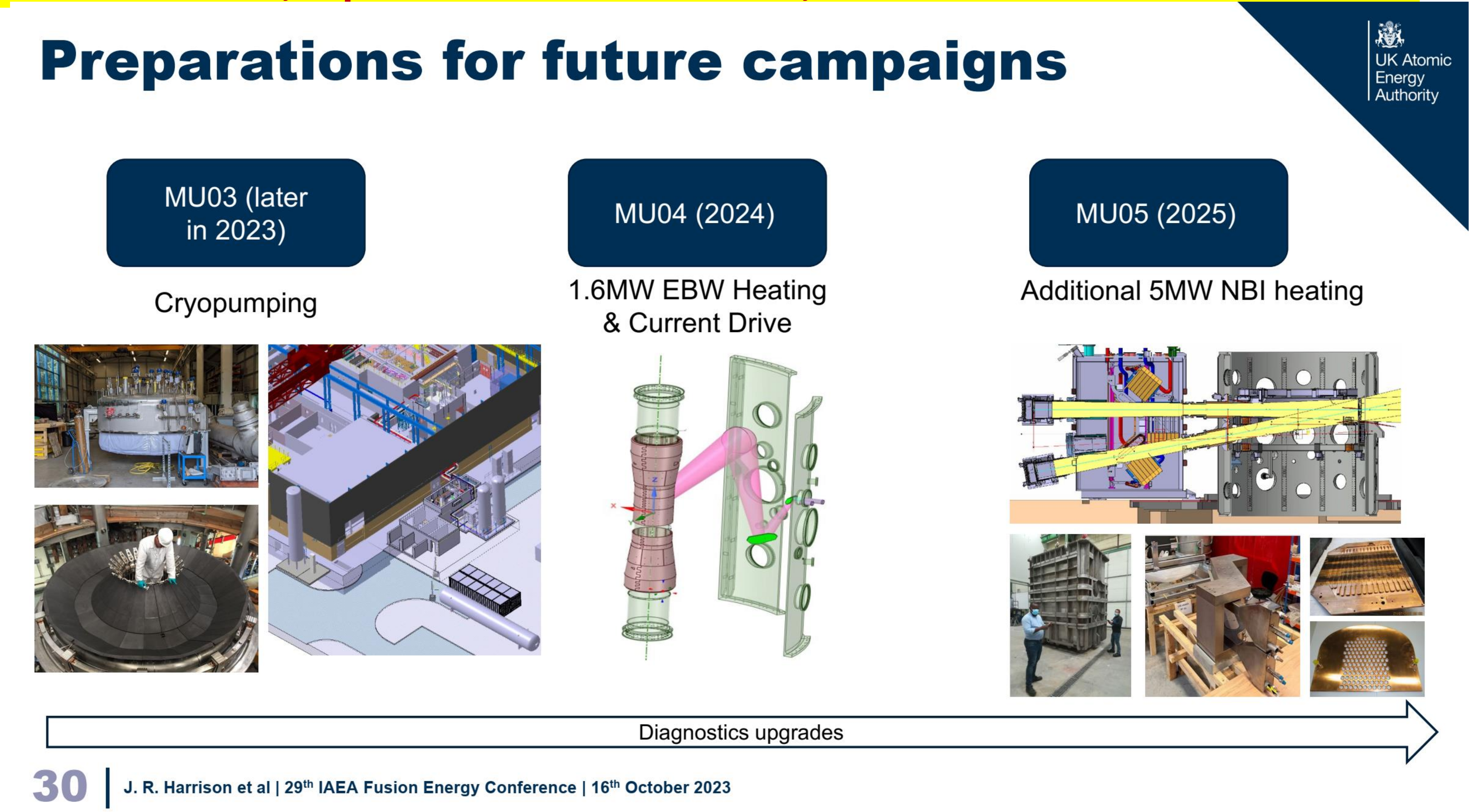
Stellarators, Spherical Tokamaks, Private Sector: MAST-U



Preparations for future campaigns

UK Atomic Energy Authority

MU03 (later in 2023)

Cryopumping

MU04 (2024)

1.6MW EBW Heating & Current Drive

MU05 (2025)

Additional 5MW NBI heating

Diagnostics upgrades

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Stellarators, Spherical Tokamaks, Private Sector: ST40

Tokamak Energy: developing spherical tokamak fusion pilot plants with HTS magnets for deployment in the 2030s

Approach

High-field spherical tokamak (ST) using magnet made from high temperature superconductor (HTS)

Team of 250+

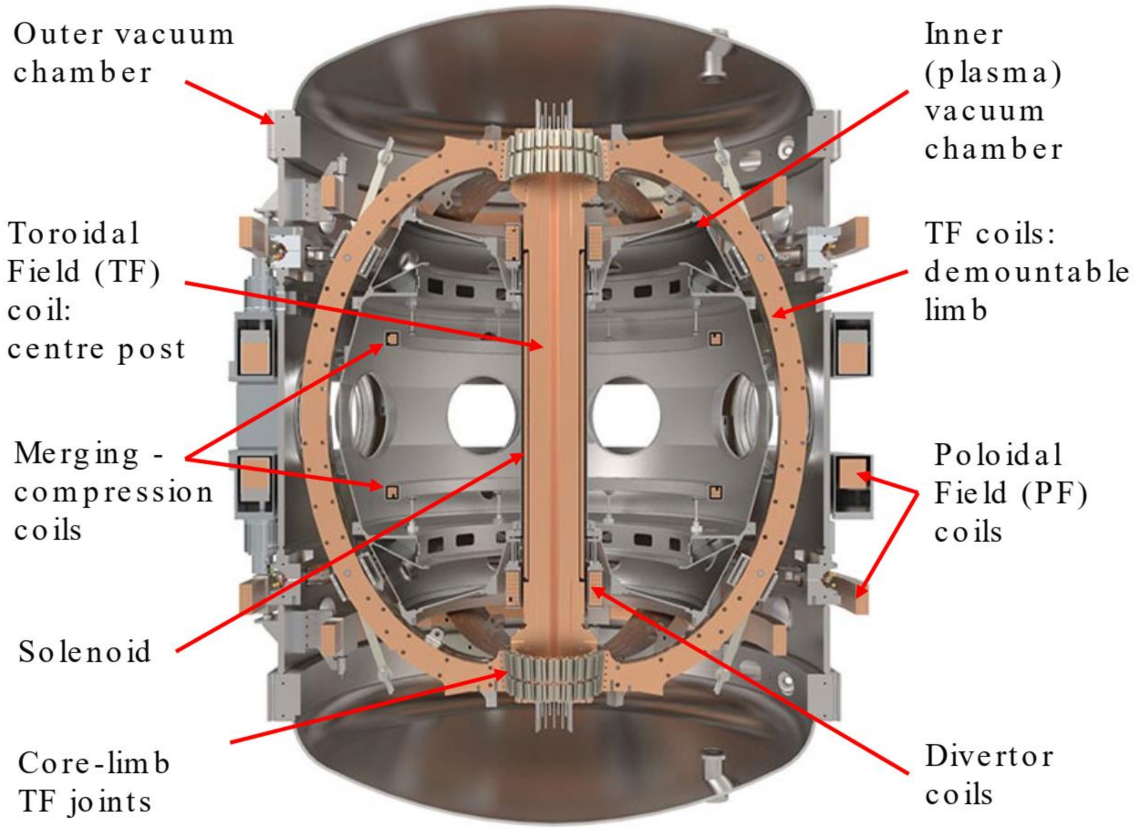
World-class scientists, engineers and commercial specialists

$250M raised to date

Financial backing from private capital and government grants

Stellarators, Spherical Tokamaks, Private Sector: ST40

The ST40 high-field spherical tokamak



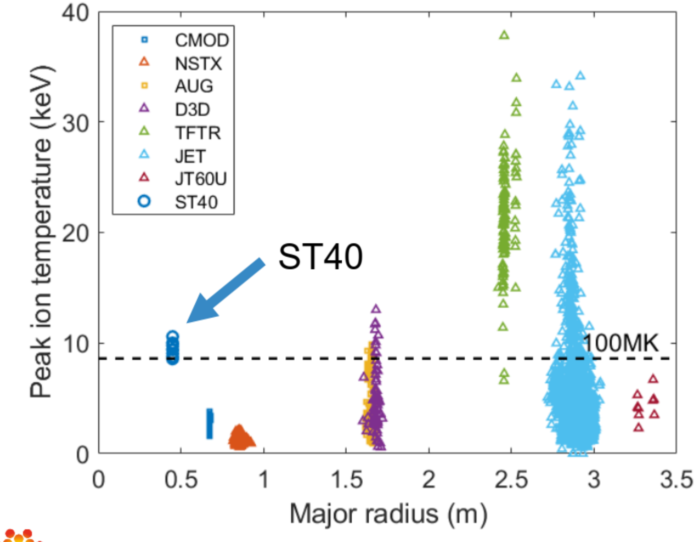
© 2023 Tokamak Energy

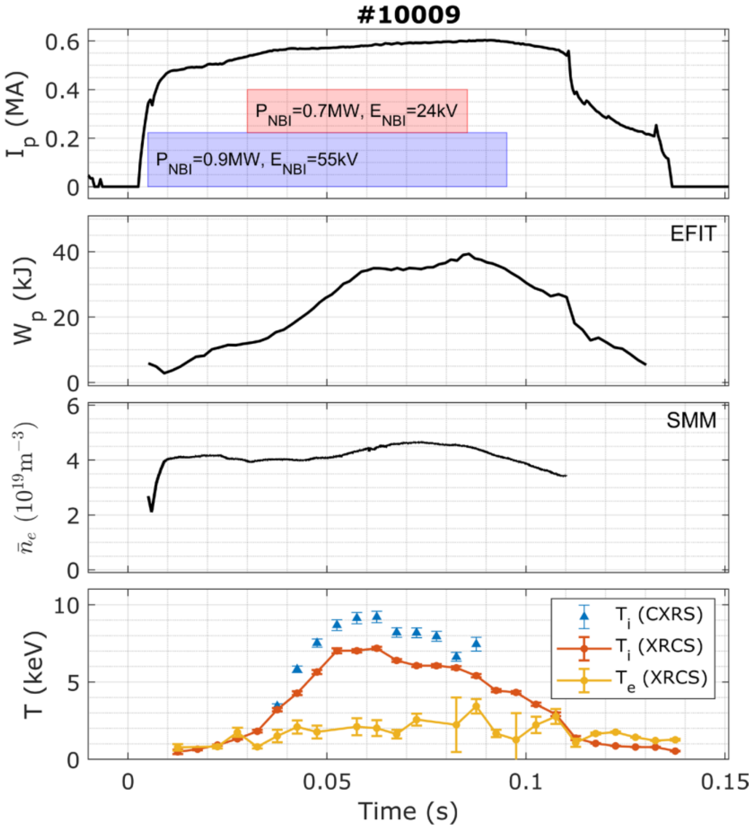
Parameter	Range
B_T [T]	0.9 – 2.1
I_P [MA]	0.3 – 0.8
R_{Geo} [m]	0.4 – 0.5
A / κ	1.6 – 1.9 / ≤ 2
P_{NB}/E_{NB} [MW/kV]	0.8/24, 1.0/55
Start-up	Merging-compression
ψ_{sol} [mWb]	200
New diagnostics	TS, divertor-IR, Langmuir probe arrays

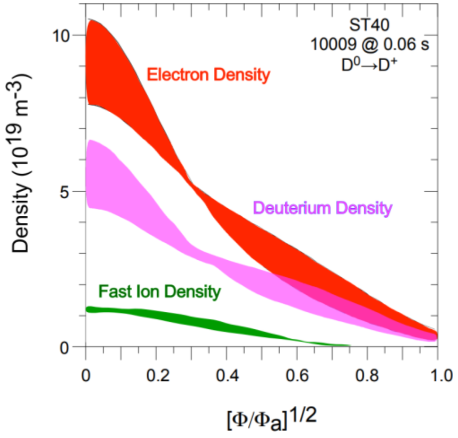
Stellarators, Spherical Tokamaks, Private Sector: ST40

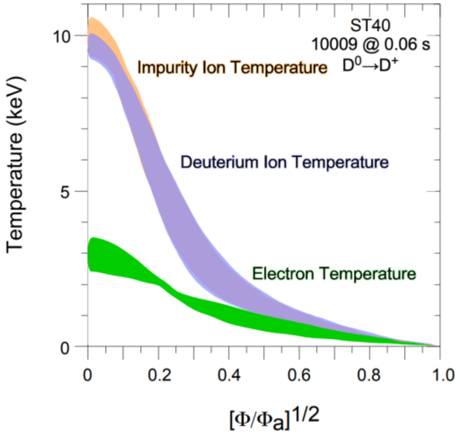
Record ion temperatures achieved in compact high -field ST

- Central ion temperatures of 9.6 ± 0.4 keV achieved in hot-ion mode ($T_i \gg T_e$) with R_{Geo}=0.45m, A=1.65, B_T=1.9T, I_p=0.6MA, and P_{NB}=1.6MW
- Corresponding triple product $n_{i0} T_{i0} \tau_E \approx 6 \pm 2 \times 10^{18}\,\mathrm{m^{-3}\,keV\,s}$ at 9.6keV









S. McNamara et al., NF 2023

S. Kaye et al., PPCF 2023

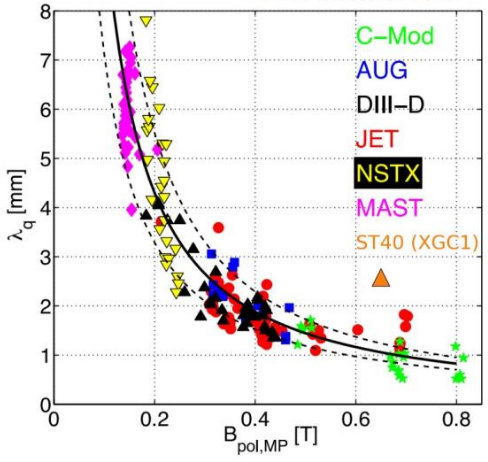
Stellarators, Spherical Tokamaks, Private Sector: ST40

Potential for scrape-off layer width broadening in ST40 plasmas

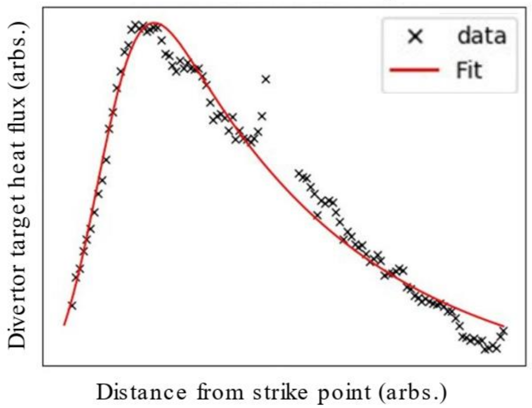
- Predictive scenarios developed using flight simulator coupled to plasma control system.
- XGC1 simulations (PPPL) show factor of 2-3 broadening above Eich scaling at I_p = 1MA.
- First heat flux measurements with divertor IR camera and Langmuir probes taken. Work ongoing to account for geometric effects.

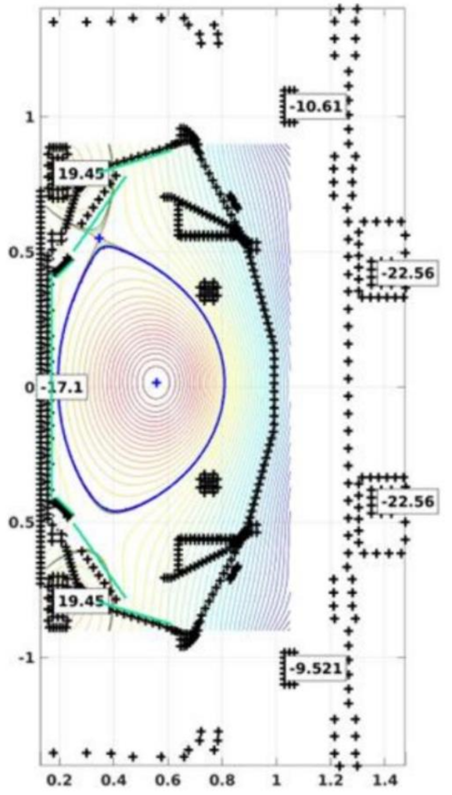
→ poster by S. Janhunen – Thursday 8:30

T. Eich et al., NF 2013

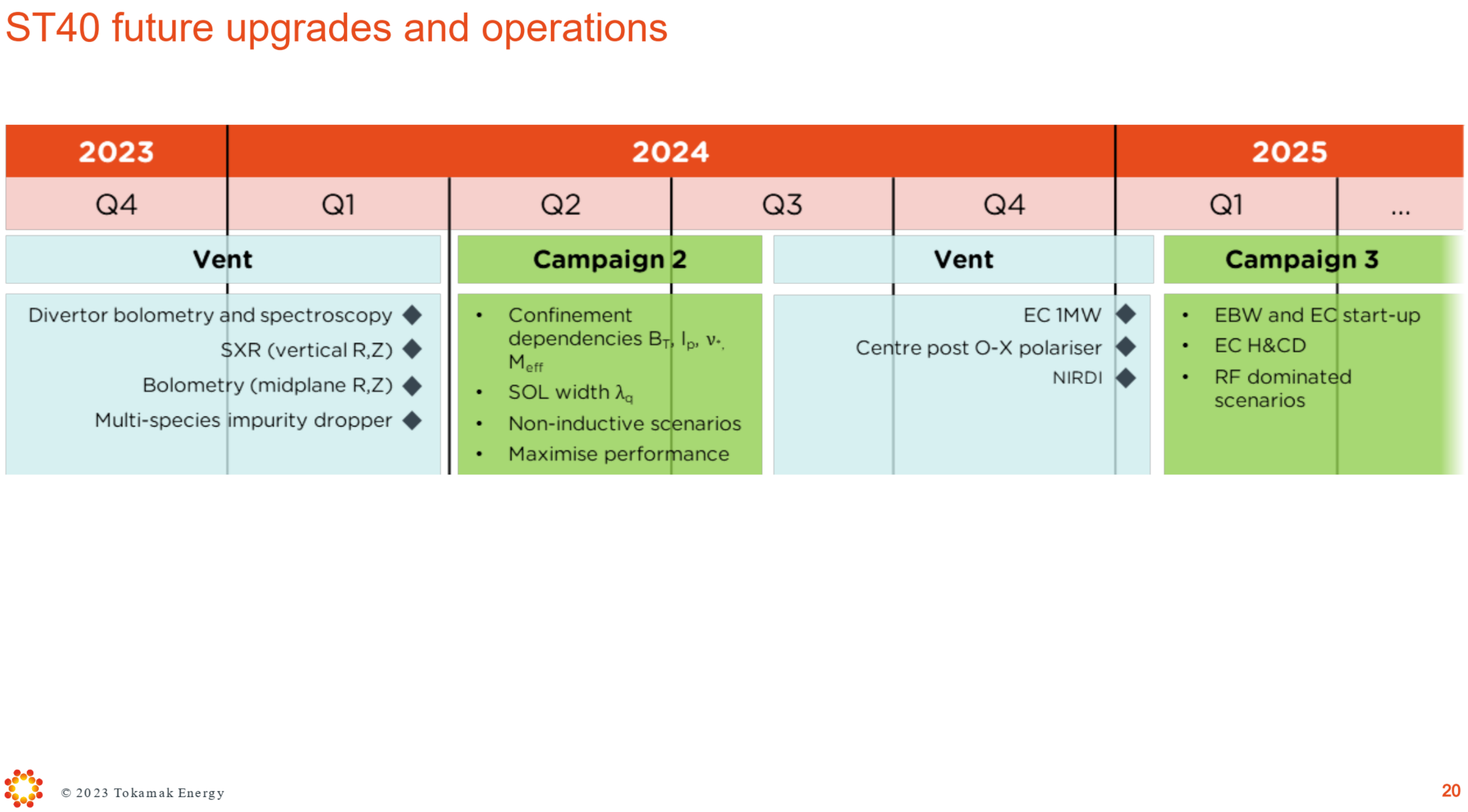


IR camera divertor heat flux profile





Stellarators, Spherical Tokamaks, Private Sector: ST40



ST40 future upgrades and operations

2023	2024				2025	
Q4	Q1	Q2	Q3	Q4	Q1	...
Vent		**Campaign 2**		**Vent**	**Campaign 3**	
Divertor bolometry and spectroscopy ◆ SXR (vertical R,Z) ◆ Bolometry (midplane R,Z) ◆ Multi-species impurity dropper ◆		• Confinement dependencies B_T, I_p, ν_*, M_{eff} • SOL width λ_q • Non-inductive scenarios • Maximise performance		EC 1MW ◆ Centre post O-X polariser ◆ NIRDI ◆	• EBW and EC start-up • EC H&CD • RF dominated scenarios	

- Technology, Long Pulse & Science: WEST, Globus-M2

Technology, Long Pulse & Science: WEST

cea irfm

WEST full tungsten operation with an ITER-grade divertor

J. Bucalossi, on behalf of the WEST team*

*http://west.cea.fr/WESTteam

West

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Technology, Long Pulse & Science: WEST

WEST: preparing ITER operation focusing on plasma exhaust

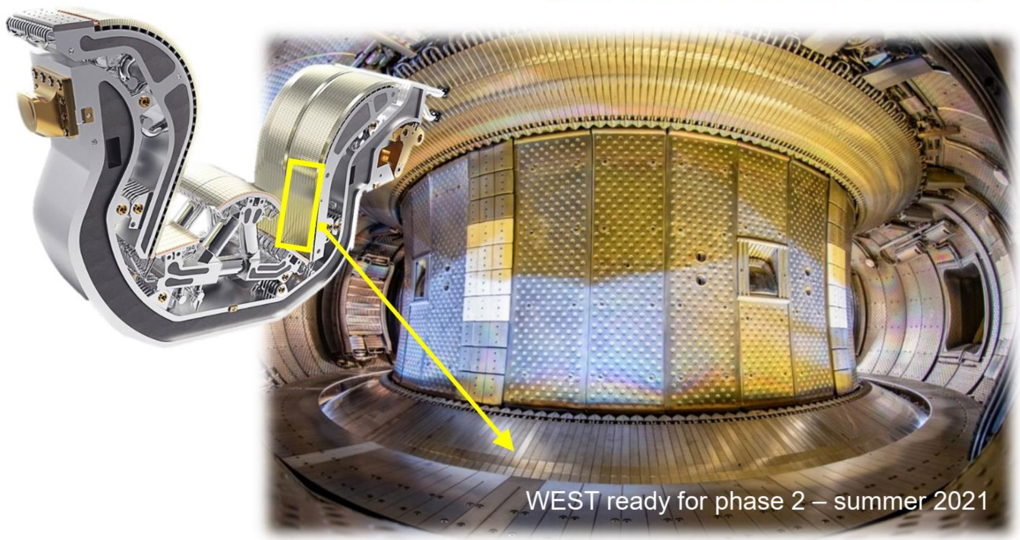
B	I_p	R	A	V_p	κ / δ	P_{RF}	Magnetic conf.
3.7 T	1 MA	2.5 m	5-6	15 m^3	1.4 / 0.5	16 MW	LSN, USN, DN

[J. Bucalossi et al., NF 2022]

1 MA superconducting tungsten tokamak

WEST phase 1, with inertially cooled divertor and a set of ITER-grade prototypes, completed:

- Up to ~9 MW of injected RF power (LHCD+ICRH)
- Up to 6 MW/m^2 of SS divertor peak heat load
- First H-mode transitions observed, consistent with Martin scaling: $P_{L\text{-}H}$ ~ 3-4 MW at 3.7 T → not sustained, operation too close to L-H threshold: density ↗, bulk radiation ↗, separatrix power ↘



WEST ready for phase 2 – summer 2021

This talk: First results of WEST phase 2 operation, with the full ITER-grade divertor, focusing on long pulse operation and PFC lifetime assessment

Technology, Long Pulse & Science: WEST

In summary

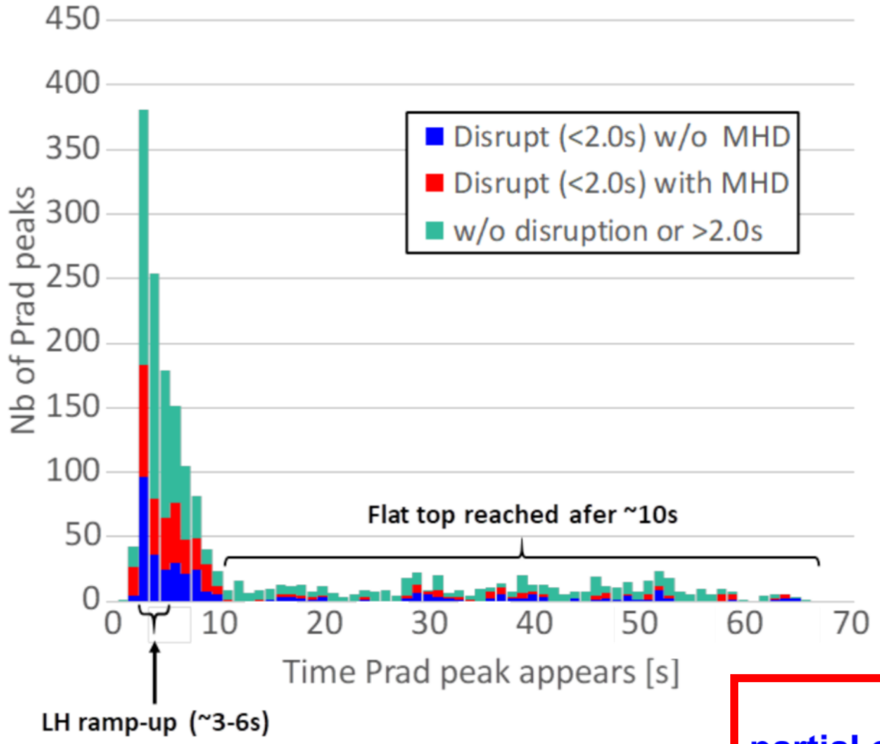
- WEST phase 2 started in December 2022 with **full ITER-grade divertor** (15 000 MB)
- **100 s** long pulses achieved, **~300 MJ** injected energy in a single pulse
- **Controlled X-point radiator** (XPR) regime in L-mode plasmas (feedback controlled)
- Improved confinement (+15-30%) and **reduced central W contamination** (-40%) in XPR
- Strong **resilience of the radiated fraction** for a given plasma configuration (wall clearance)
- First **3D modelling of the plasma-wall interaction** confirms this resilient self-organization

- High fluence campaign (L-mode, attached plasma): ~3 hours plasma **without boronisation**
- **Deposited layers building up**: on the HFS and in the monoblock bevel shadow
- Pulse length hampered by **increasing UFO occurrence**, originates mainly from deposited layers on the divertor ➔ issue for continuous operation? divertor cleaning required?

Technology, Long Pulse & Science: WEST

UFOs triggered primarily during LH power ramp up

Average pulse length affected by appearance of UFOs and increased disruptivity



- UFO detection: peak > 250 kW on P_{rad} from bolometry (~1800 UFOs)
  - Disruption < 2s, no MHD
  - Disruption < 2s, with MHD
  - No disruption (or > 2s)
- ~70% of detected UFOs appears during the ramp-up phase of the discharge before 10 s (mainly during LHCD ramp-up) ➔ actively cooled PFC may enhance stresses in deposits?
- Most frequent process: UFO ingress ➔ increased core radiation ➔ “cold branch” regime ➔ MHD ➔ disruption

Link between UFOs, deposits and disruptions ➔ partial cleaning of the divertor performed for the next campaign

Technology, Long Pulse & Science: WEST

Controlled X-point radiator regime achieved in compact divertor

- X-point radiator (XPR) regime successfully achieved using divertor & midplane nitrogen seeding, as in AUG *[M. Bernert et al., NF 2021]*
- Test of **control scheme** using divertor interferometry line of sight
- **XPR regime maintained for 18 s**, feedback controlled for 12 s

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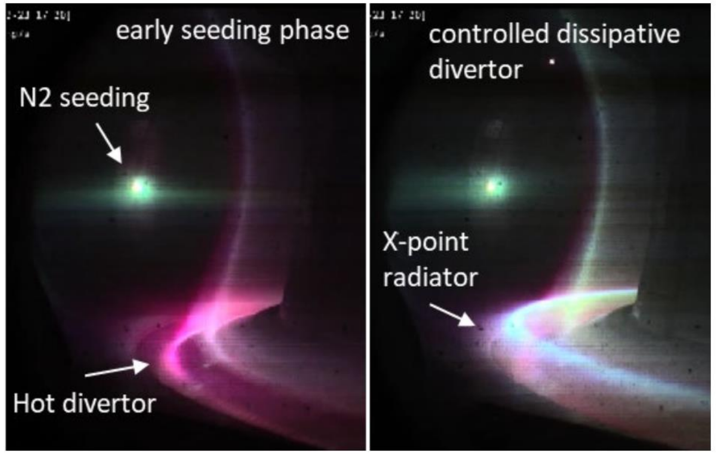
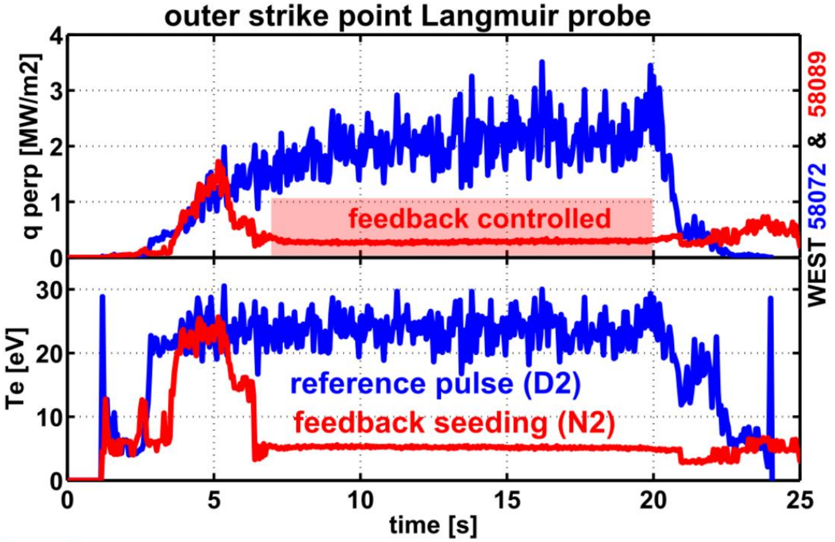
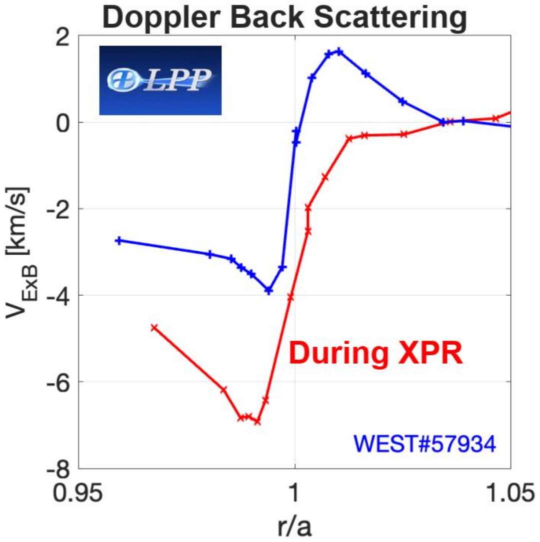






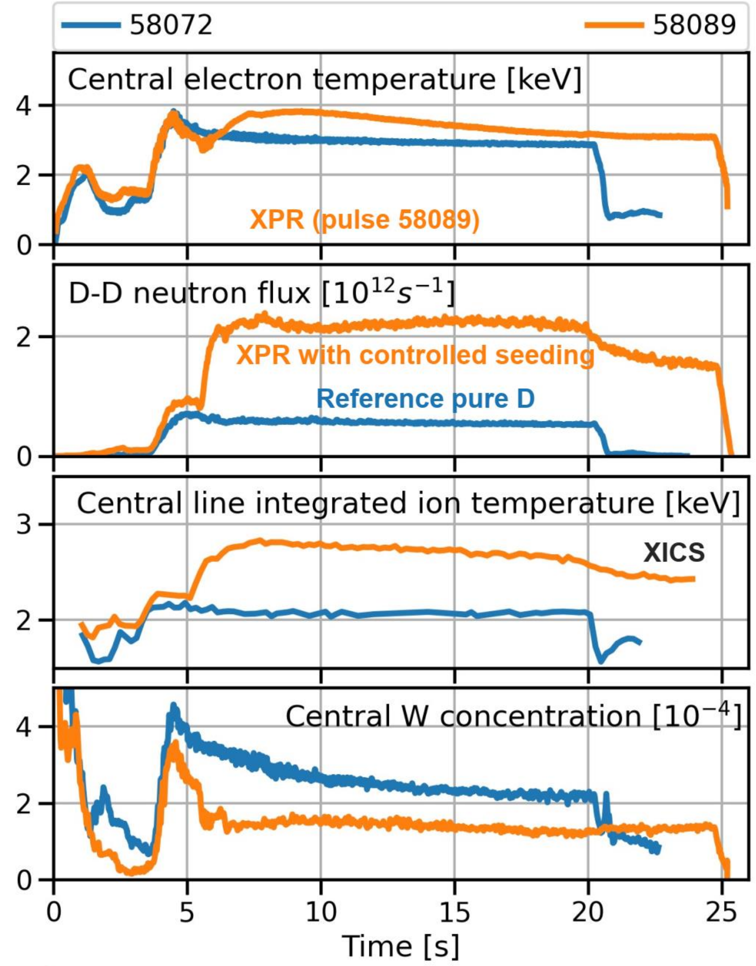
- Deeper V_{EXB} well inside separatrix
- Flattens to ~0 in the Scrape-off Layer (SOL)

➔ Coherent with low electron temperature at divertor

[D. Brida et al., NME 2022]

Technology, Long Pulse & Science: WEST

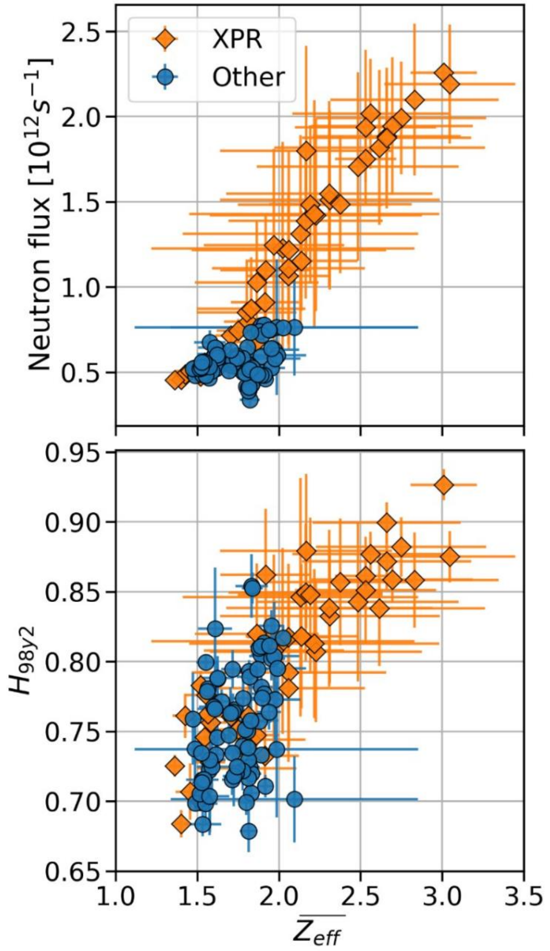
Improved confinement in X-point radiator regime



- **Increased confinement:** +15% to +30%
- **Higher central electron temperature:** +20%
- **Higher central ion temperature:** +35%
- Neutron flux: x 4
- **Reduction in central W contamination:** -40%
- Important effect of Zeff

Similar improvements observed with Boron Powder Dropper

LUNSFORD, Fri PM Poster



Technology, Long Pulse & Science: Globus-M2

Ioffe Institute

CONFINEMENT, HEATING AND CURRENT DRIVE IN SPHERICAL TOKAMAK GLOBUS-M2 WITH HIGH MAGNETIC FIELD

G.S. Kurskiev on behalf of the Globus-M2 team

<u>Ioffe Institute, Russian Academy of Sciences, St. Petersburg, Russia</u>

Budker Institute of Nuclear Physics of Siberian Branch Russian Academy of Sciences (BINP SB RAS), Novosibirsk, Russia

Efremov Research Institute of Electrophysical Equipment, St. Petersburg, Russia

Ioffe Fusion Technology, St. Petersburg, Russia

Lomonosov Moscow State University, Moscow, Russia

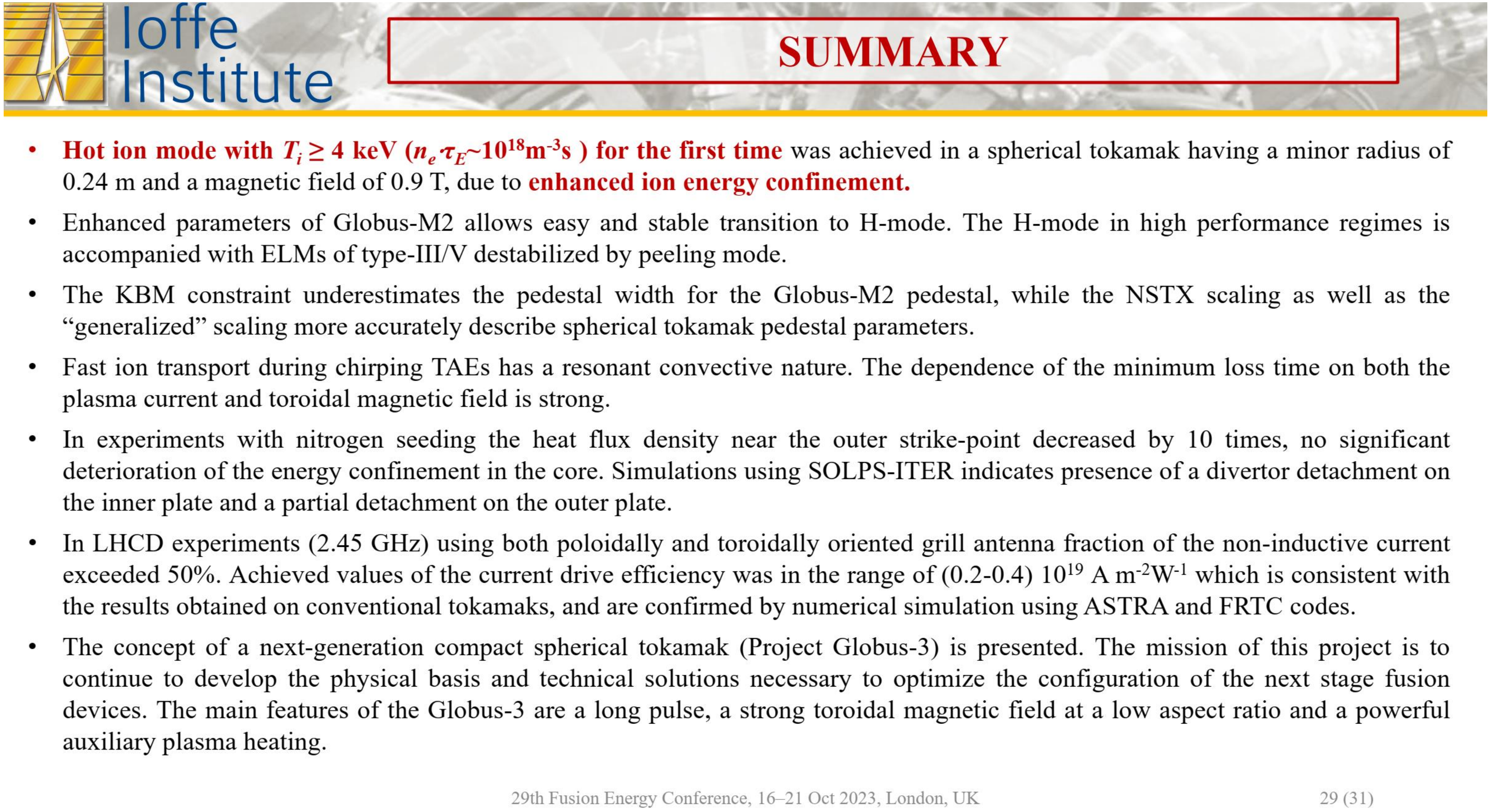
Peter the Great St. Petersburg Polytechnic University, St. Petersburg, Russia

V.A. Trapeznikov Institute of Control Problems, Russian Academy of Sciences, Moscow, Russia

Konstantinov Petersburg Nuclear Physics Institute, National Research Center "Kurchatov Institute," St. Petersburg, Russia

Gleb.Kurskiev@mail.ioffe.ru

Technology, Long Pulse & Science: Globus-M2



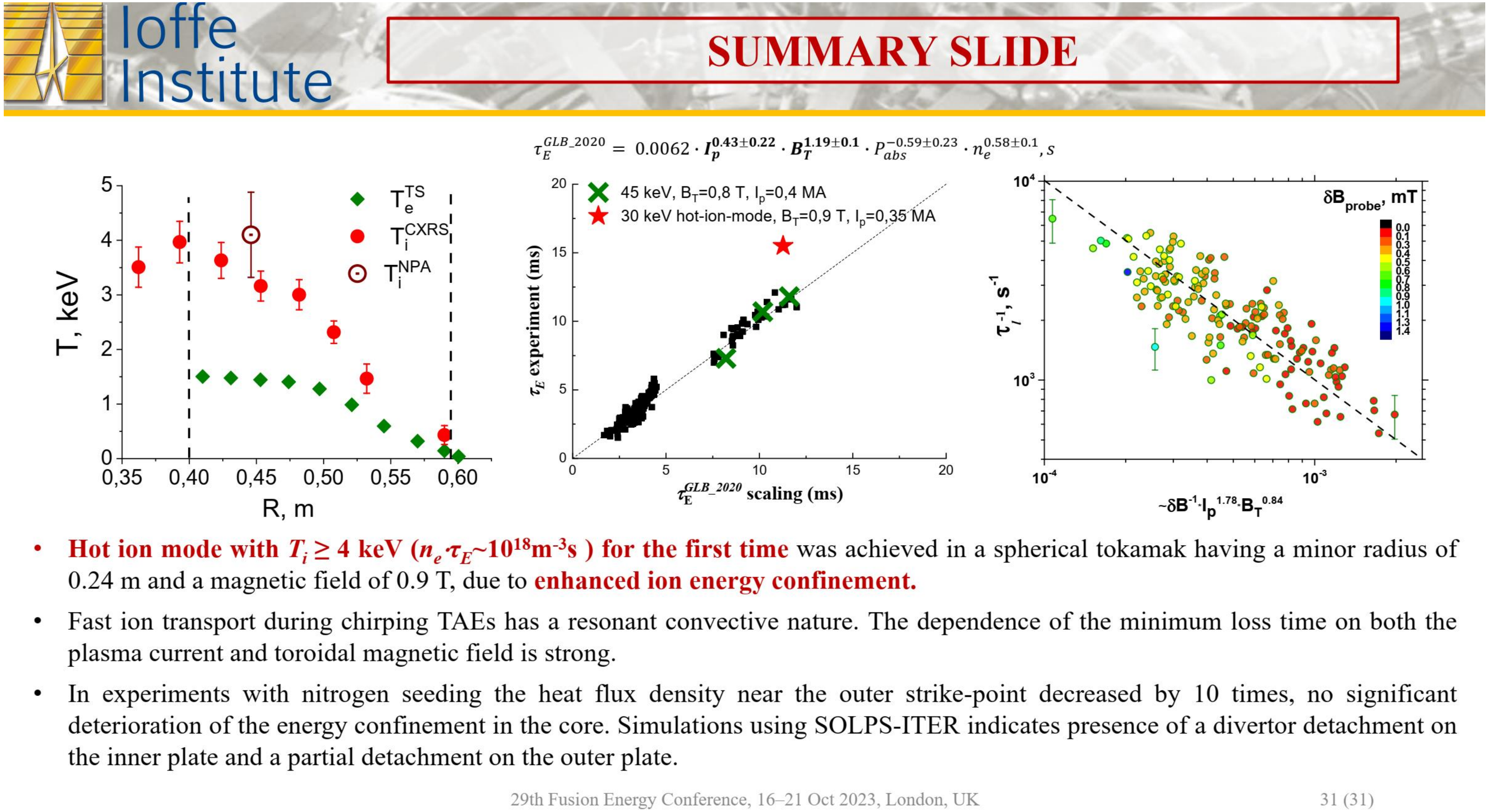
Ioffe Institute

SUMMARY

- **Hot ion mode with $T_i \geq 4$ keV ($n_e \cdot \tau_E \sim 10^{18}$m^{-3}s) for the first time** was achieved in a spherical tokamak having a minor radius of 0.24 m and a magnetic field of 0.9 T, due to **enhanced ion energy confinement.**
- Enhanced parameters of Globus-M2 allows easy and stable transition to H-mode. The H-mode in high performance regimes is accompanied with ELMs of type-III/V destabilized by peeling mode.
- The KBM constraint underestimates the pedestal width for the Globus-M2 pedestal, while the NSTX scaling as well as the “generalized” scaling more accurately describe spherical tokamak pedestal parameters.
- Fast ion transport during chirping TAEs has a resonant convective nature. The dependence of the minimum loss time on both the plasma current and toroidal magnetic field is strong.
- In experiments with nitrogen seeding the heat flux density near the outer strike-point decreased by 10 times, no significant deterioration of the energy confinement in the core. Simulations using SOLPS-ITER indicates presence of a divertor detachment on the inner plate and a partial detachment on the outer plate.
- In LHCD experiments (2.45 GHz) using both poloidally and toroidally oriented grill antenna fraction of the non-inductive current exceeded 50%. Achieved values of the current drive efficiency was in the range of (0.2-0.4) 10^{19} A m^{-2}W^{-1} which is consistent with the results obtained on conventional tokamaks, and are confirmed by numerical simulation using ASTRA and FRTC codes.
- The concept of a next-generation compact spherical tokamak (Project Globus-3) is presented. The mission of this project is to continue to develop the physical basis and technical solutions necessary to optimize the configuration of the next stage fusion devices. The main features of the Globus-3 are a long pulse, a strong toroidal magnetic field at a low aspect ratio and a powerful auxiliary plasma heating.

29th Fusion Energy Conference, 16–21 Oct 2023, London, UK 29 (31)

Technology, Long Pulse & Science: Globus-M2



Ioffe Institute

SUMMARY SLIDE

$$\tau_E^{GLB\_2020} = 0.0062 \cdot \boldsymbol{I_p^{0.43\pm0.22}} \cdot \boldsymbol{B_T^{1.19\pm0.1}} \cdot P_{abs}^{-0.59\pm0.23} \cdot n_e^{0.58\pm0.1}, s$$

- **Hot ion mode with $T_i \geq 4$ keV ($n_e \cdot \tau_E$~10^{18}m^{-3}s) for the first time** was achieved in a spherical tokamak having a minor radius of 0.24 m and a magnetic field of 0.9 T, due to **enhanced ion energy confinement.**
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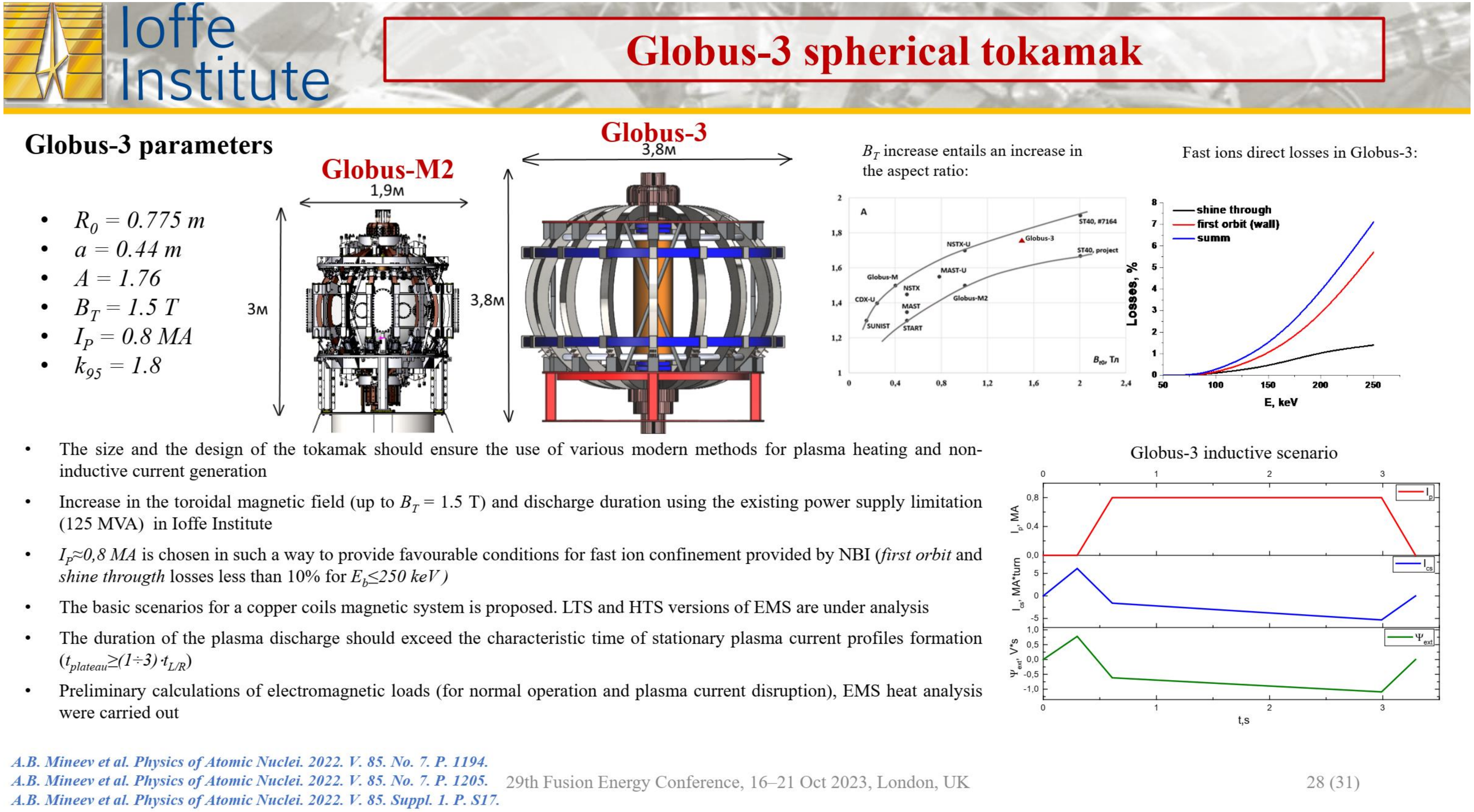
31 (31)

Technology, Long Pulse & Science: Globus-M2



Globus-3 spherical tokamak

Globus-3 parameters

- $R_0 = 0.775\ m$
- $a = 0.44\ m$
- $A = 1.76$
- $B_T = 1.5\ T$
- $I_P = 0.8\ MA$
- $k_{95} = 1.8$

B_T increase entails an increase in the aspect ratio:

Fast ions direct losses in Globus-3:

Globus-3 inductive scenario

- The size and the design of the tokamak should ensure the use of various modern methods for plasma heating and non-inductive current generation
- Increase in the toroidal magnetic field (up to B_T = 1.5 T) and discharge duration using the existing power supply limitation (125 MVA) in Ioffe Institute
- $I_P \approx 0,8\ MA$ is chosen in such a way to provide favourable conditions for fast ion confinement provided by NBI (*first orbit* and *shine througth* losses less than 10% for $E_b \leq 250\ keV$)
- The basic scenarios for a copper coils magnetic system is proposed. LTS and HTS versions of EMS are under analysis
- The duration of the plasma discharge should exceed the characteristic time of stationary plasma current profiles formation ($t_{plateau} \geq (1 \div 3) \cdot t_{L/R}$)
- Preliminary calculations of electromagnetic loads (for normal operation and plasma current disruption), EMS heat analysis were carried out

A.B. Mineev et al. Physics of Atomic Nuclei. 2022. V. 85. No. 7. P. 1194.
A.B. Mineev et al. Physics of Atomic Nuclei. 2022. V. 85. No. 7. P. 1205.
A.B. Mineev et al. Physics of Atomic Nuclei. 2022. V. 85. Suppl. 1. P. S17.

29th Fusion Energy Conference, 16–21 Oct 2023, London, UK

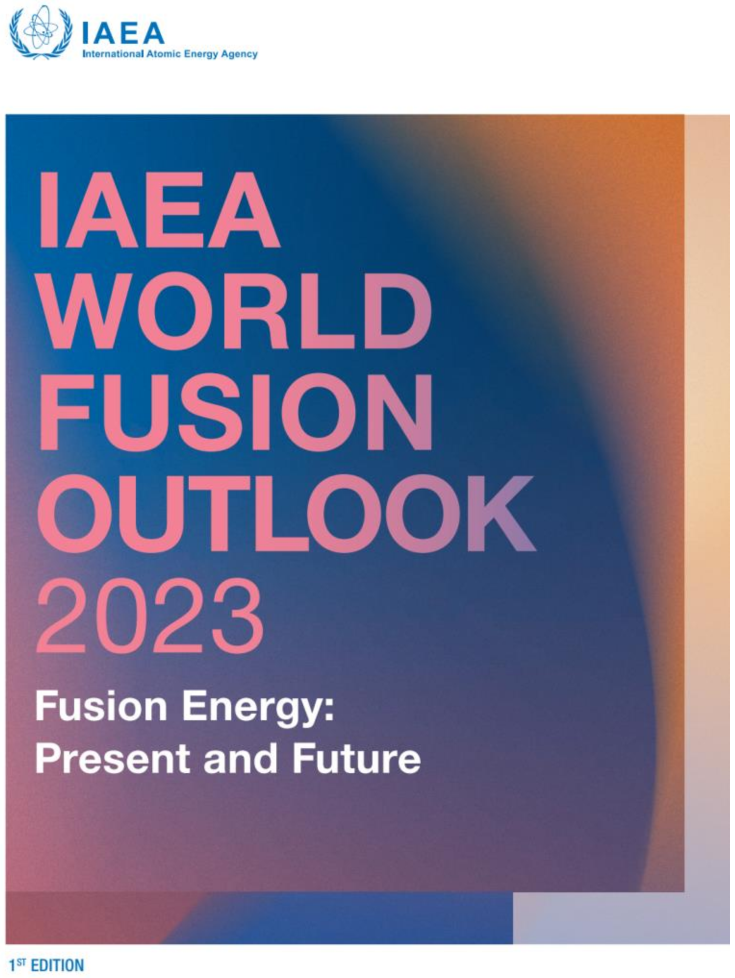
28 (31)

Заключение

- В условиях W-Be первой стенки полученная мощность термоядерного выхода не уступает результатам предыдущей кампании на JET-C.
- Вопрос получения длительных режимов при большой мощности нагрева плазмы по прежнему является наиболее актуальной проблемой, на решение которой смещен фокус экспериментов ведущих установок.
- Отрицательная треугольность выглядит привлекательно: DIII-D, TCV, AUG– будет ли развитие?
- Сферические токамаки - получение плазменных разрядов с высокой температурой и плотностью потока нейтронов возможно на малых установках. Направление активно развивается во всем мире.

Запасные слайды

IAEA World Fusion Outlook 2023

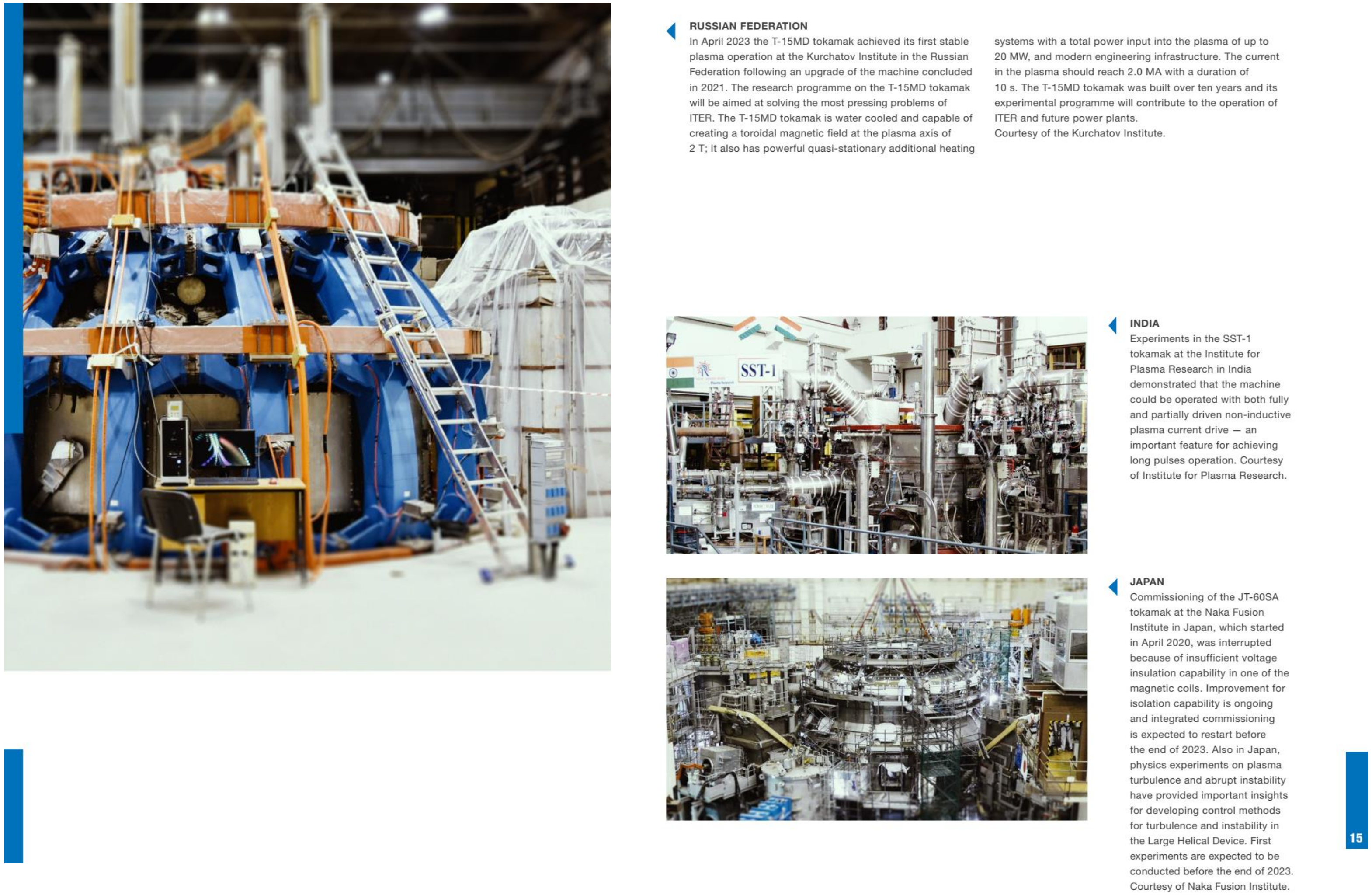


IAEA
International Atomic Energy Agency

IAEA
WORLD
FUSION
OUTLOOK
2023

Fusion Energy:
Present and Future

1ST EDITION



RUSSIAN FEDERATION

In April 2023 the T-15MD tokamak achieved its first stable plasma operation at the Kurchatov Institute in the Russian Federation following an upgrade of the machine concluded in 2021. The research programme on the T-15MD tokamak will be aimed at solving the most pressing problems of ITER. The T-15MD tokamak is water cooled and capable of creating a toroidal magnetic field at the plasma axis of 2 T; it also has powerful quasi-stationary additional heating systems with a total power input into the plasma of up to 20 MW, and modern engineering infrastructure. The current in the plasma should reach 2.0 MA with a duration of 10 s. The T-15MD tokamak was built over ten years and its experimental programme will contribute to the operation of ITER and future power plants. Courtesy of the Kurchatov Institute.

INDIA

Experiments in the SST-1 tokamak at the Institute for Plasma Research in India demonstrated that the machine could be operated with both fully and partially driven non-inductive plasma current drive — an important feature for achieving long pulses operation. Courtesy of Institute for Plasma Research.

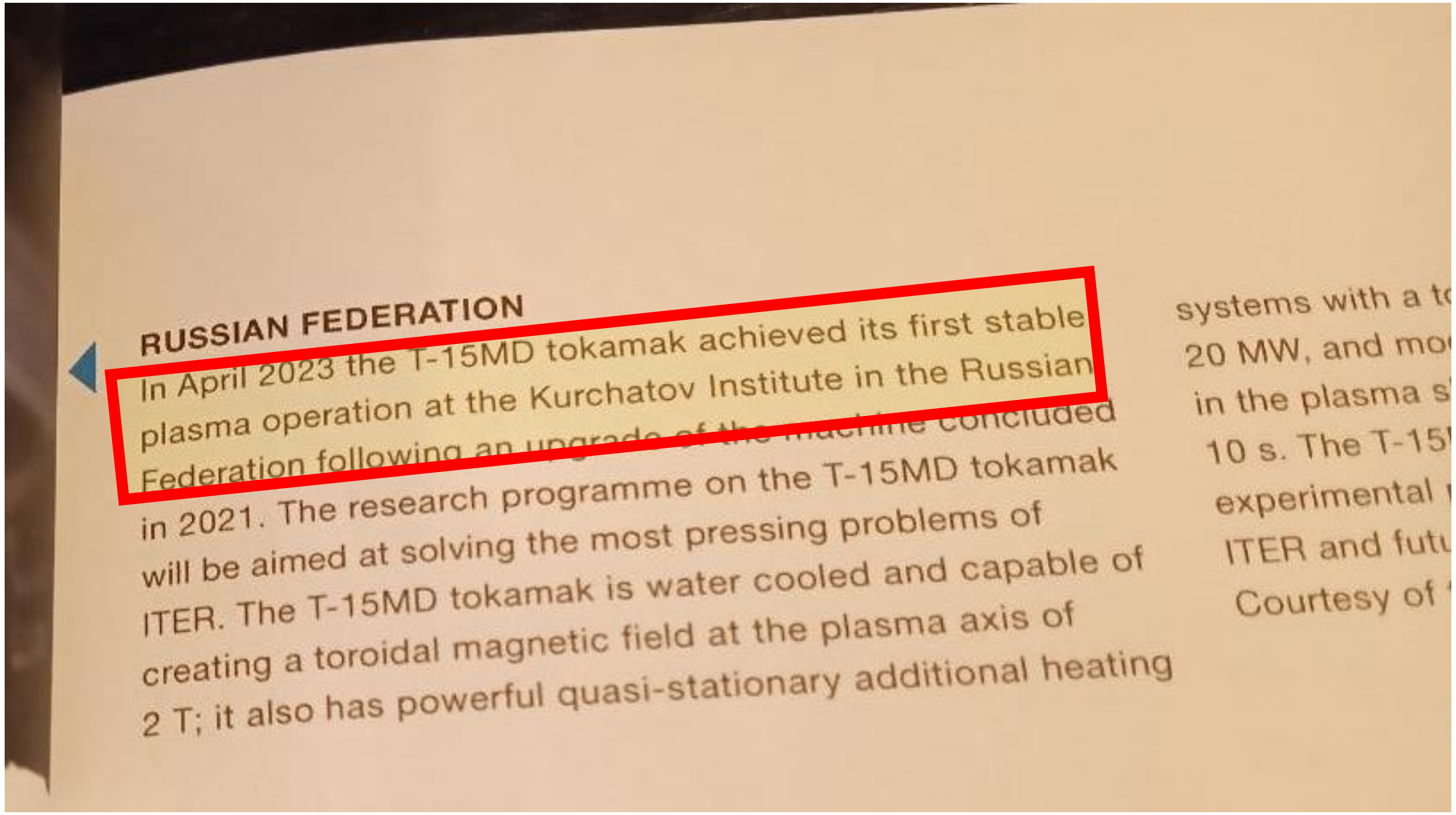
JAPAN

Commissioning of the JT-60SA tokamak at the Naka Fusion Institute in Japan, which started in April 2020, was interrupted because of insufficient voltage insulation capability in one of the magnetic coils. Improvement for isolation capability is ongoing and integrated commissioning is expected to restart before the end of 2023. Also in Japan, physics experiments on plasma turbulence and abrupt instability have provided important insights for developing control methods for turbulence and instability in the Large Helical Device. First experiments are expected to be conducted before the end of 2023. Courtesy of Naka Fusion Institute.

15

https://www.iaea.org/publications/15524/iaea-world-fusion-outlook-2023

IAEA World Fusion Outlook 2023



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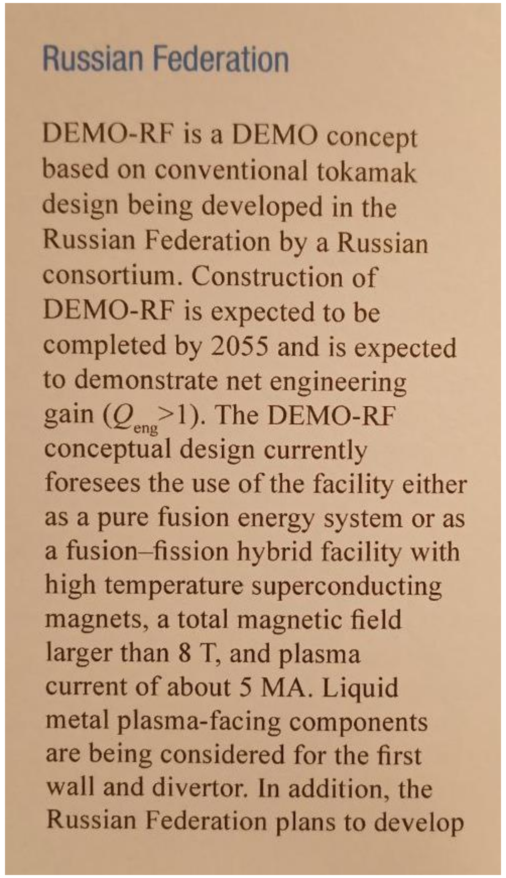


Russian Federation

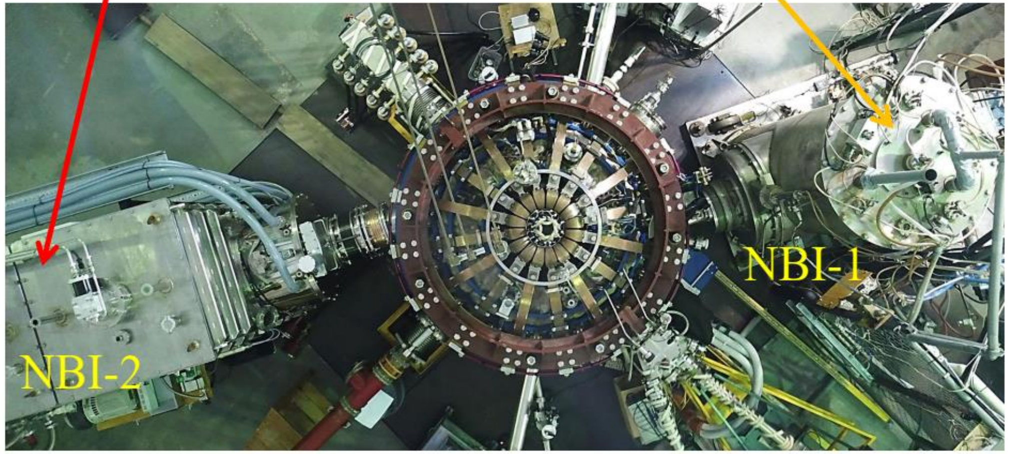
DEMO-RF is a DEMO concept based on conventional tokamak design being developed in the Russian Federation by a Russian consortium. Construction of DEMO-RF is expected to be completed by 2055 and is expected to demonstrate net engineering gain ($Q_{eng}>1$). The DEMO-RF conceptual design currently foresees the use of the facility either as a pure fusion energy system or as a fusion–fission hybrid facility with high temperature superconducting magnets, a total magnetic field larger than 8 T, and plasma current of about 5 MA. Liquid metal plasma-facing components are being considered for the first wall and divertor. In addition, the Russian Federation plans to develop

https://www.iaea.org/publications/15524/iaea-world-fusion-outlook-2023

Experimental set-up and TS upgrade

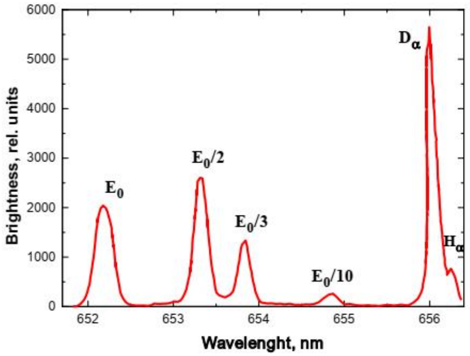
NBI-2 allows significantly increase heating power

NBI-1	0.5 MW, 18-30 keV	50 ms	5×12 cm
NBI-2	1 MW, 25-50 keV	1000 ms	11 cm



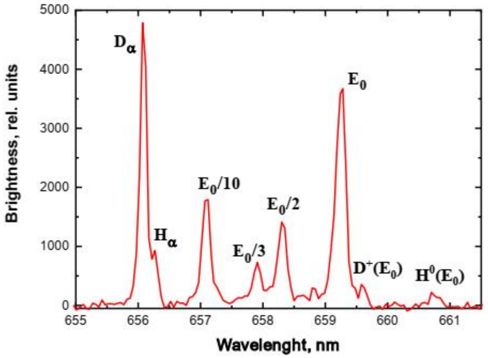
Doppler-shifted radiation spectra for deuterium atomic beams:

NBI-2 with E ≈ 50 keV



$P(E):P(E/2):P(E/3) = 0.56: 0.32: 0.12$

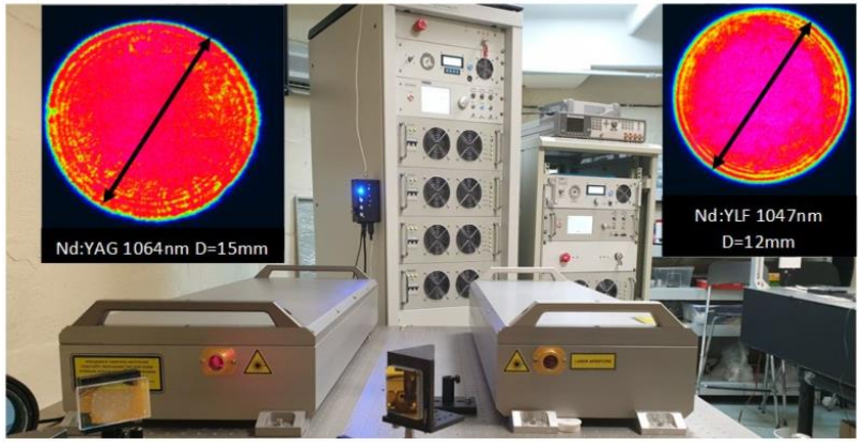
NBI-1 with E ≈ 28 keV

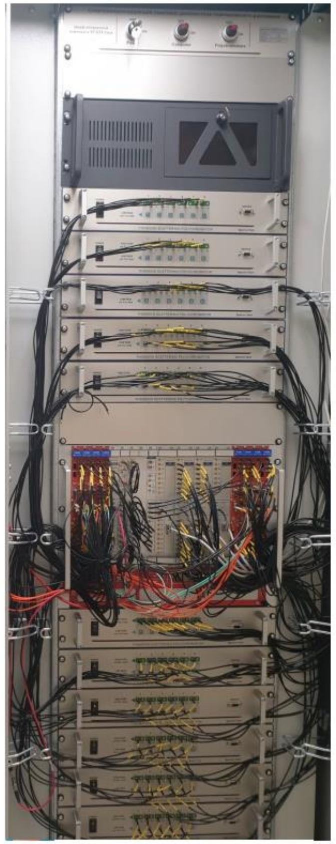


$P(E):P(E/2):P(E/3) = 0.82: 0.13: 0.05$

[P.B. Shchegolev et al 2023 Plasma Physics Reports]

Thomson scattering diagnostics upgrade





4x CAEN v1743
SAMLONG sca
12-bit, 3.2 (GS/s)

ITER grade *pulsed steady-state* lasers:

Diode-pumped Nd:YAG laser 1064,5 nm, 330 Hz 3 J, 10 ns;

Diode-pumped Nd:YLF laser 1047,3 nm, 50 Hz 2 J, 3 ns

Accurate measurements of T_e and n_e profiles in 10 spatial points from LCMS to magnetic axis

Reliable single-pulse T_e measurements in SOL for n_e<3·10^{18} m^{-3} , E_L = 0.7 J and L = 10 mm

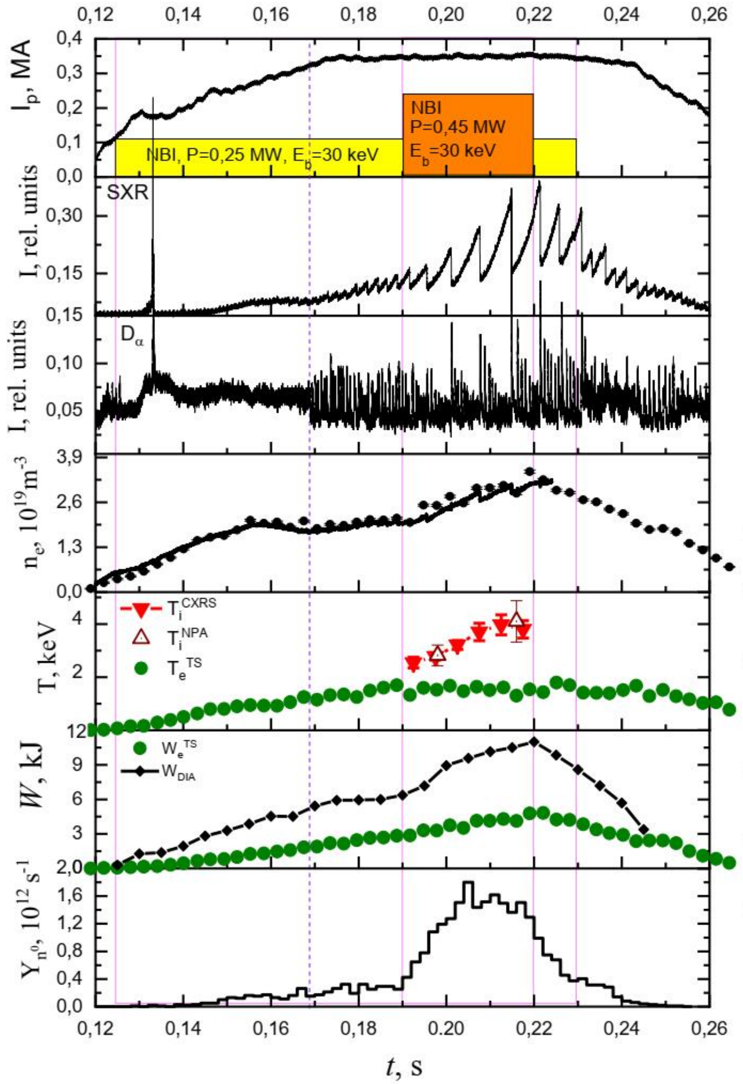
Ultra-wide density range: $n_e = 5\cdot10^{17} \div 3\cdot10^{20}$ m^{-3}

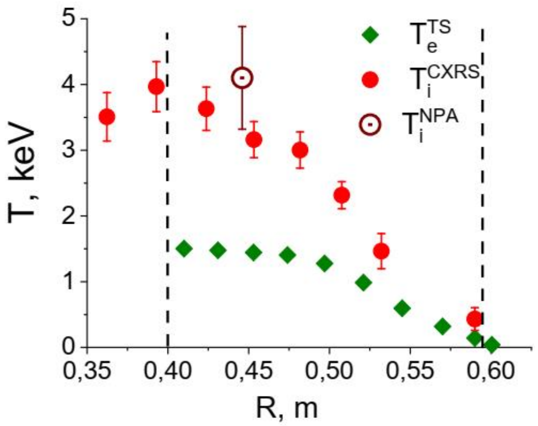
Real-time operation mode, latency <Δt> = 2.0 ms (Δt$_{max}$ < 2.5 ms) matches ITER requirements for CPTS

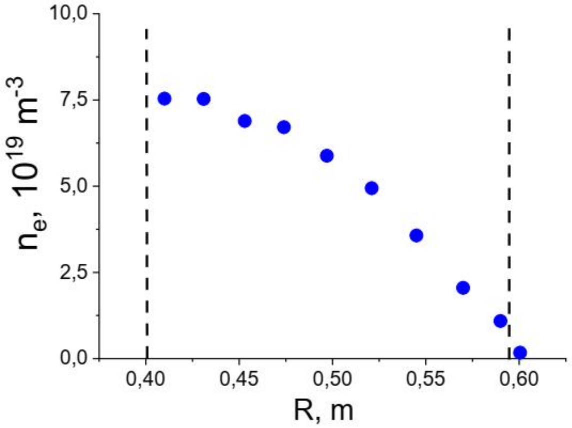
[G. S. Kurskiev et al, 2021, Technical Physics Letters]
[N. S. Zhiltsov et al, 2023, Nuclear Fusion]

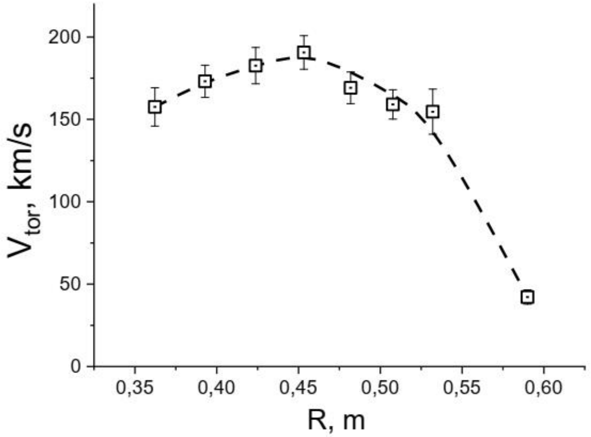
Hot ion mode in Globus-M2 plasma

Scenario that utilizes both NBI-1 and NBI-2 dramatically changes plasma heating efficiency

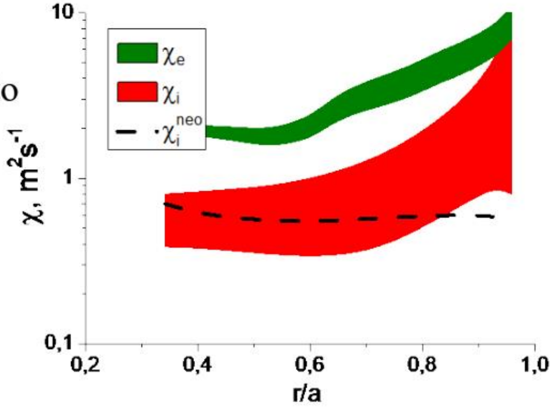








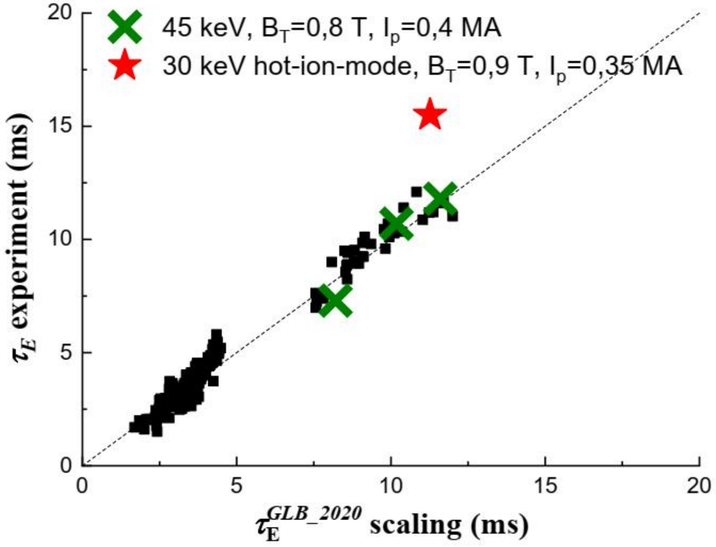
- Robust L-H transition at 169 ms;
- $T_i(0) \geq 2{*}T_e(0)$, high toroidal rotation in the plasma center up to 180 km/s;
- Ion temperature measured by CXRS diagnostic T_i^{CXRS} corresponds well to the NPA data: $T_i\,(0)$=4 keV;
- The neutron flux reaches 1.8×10^{12} neutrons/s during the NBI-1;
- Energy confinement time τ_E reaches 13-18 ms;
- Ion heat transport is close to neoclassical level

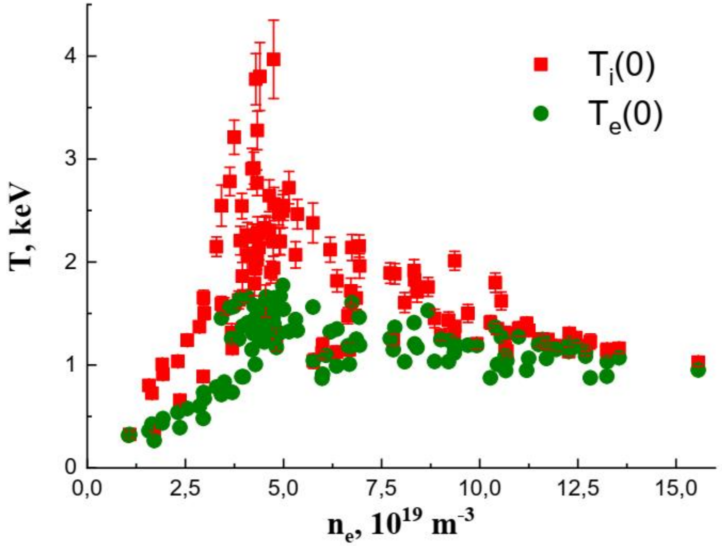


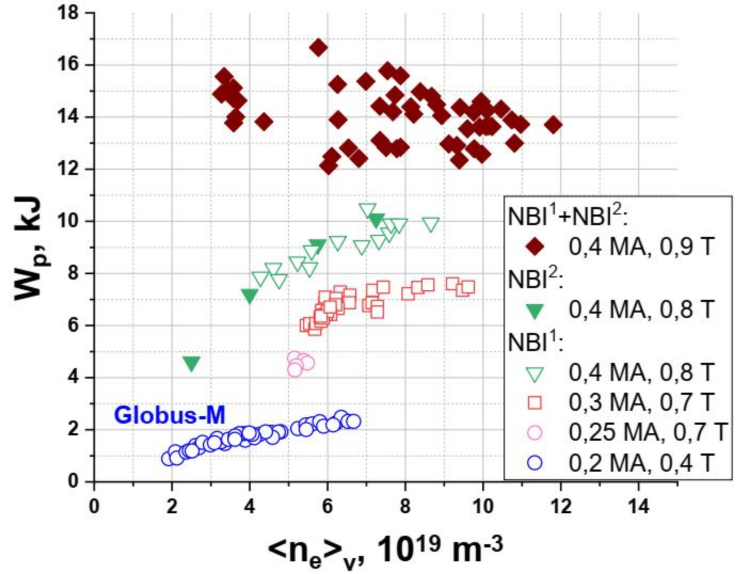
Kurskiev G.S. et al, 2022, Nucl. Fusion 62, 104002

Hot ion mode in Globus-M2 plasma

$$\tau_E^{GLB\_2020} = 0.0062 \cdot \boldsymbol{I_p}^{\mathbf{0.43\pm0.22}} \cdot \boldsymbol{B_T}^{\mathbf{1.19\pm0.1}} \cdot P_{abs}^{-0.59\pm0.23} \cdot n_e^{0.58\pm0.1}, s$$





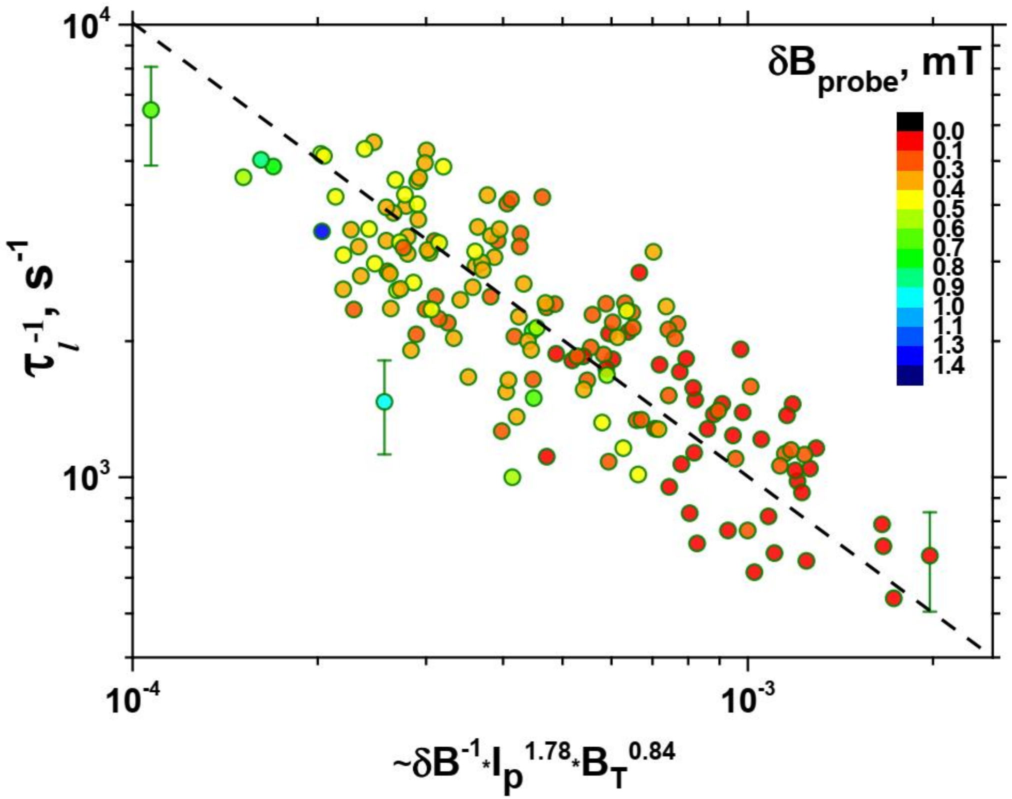


- Hot ion mode is obtained in a wide range of Globus-M2 parameters I_p=0,25-0,4 MA, B_T=0,7-0,9 T;
- NBI-2 ($E \leq$ 45 keV, $P_{NBI} \leq$ 0,75 MW) was switched on in the current ramp-up phase, while NBI-1 ($E \leq$ 28 keV, $P_{NBI} \leq$ 0,45 MW) was switched at the current flat-top;
- Measurements performed by CXRS and NPA diagnostics indicates: T_i = 4 keV, T_i =2.5 · T_e at $<n_e>$ = 5·10^{19} m^{-3};
- T_i exceeds T_e in a wide range $<n_e>$ = 1.6-10.0·10^{19} m^{-3};
- The record (for Globus-M2) stored plasma energy W is 17 kJ and energy confinement time τ_E is 13-18 ms.

[G.S. Kurskiev et al, Plasma Physics Reports, 2023, Vol. 49, No. 4, pp. 403–418]

TAE-induced fast ion transport

Transport from the plasma center using active NPA



TAE

	Experiment	**Modeling**
Transport from center to edge, $\tau_l^{-1}\sim$	$I_p^{-1.78\pm0.06}B_T^{-0.84\pm0.04}$	$I_p^{-1.9\pm0.08}B_T^{-1.18\pm0.37}$
Losses from edge ,$\tau_l^{-1}\sim$		$I_p^{-1.42\pm0.11}B_T^{-0.43\pm0.07}$

First orbit losses in MHD-free discharge

Losses from edge,$\tau_l^{-1}\sim$		$I_p^{-1.72\pm0.06}B_T^{-0.53\pm0.04}$

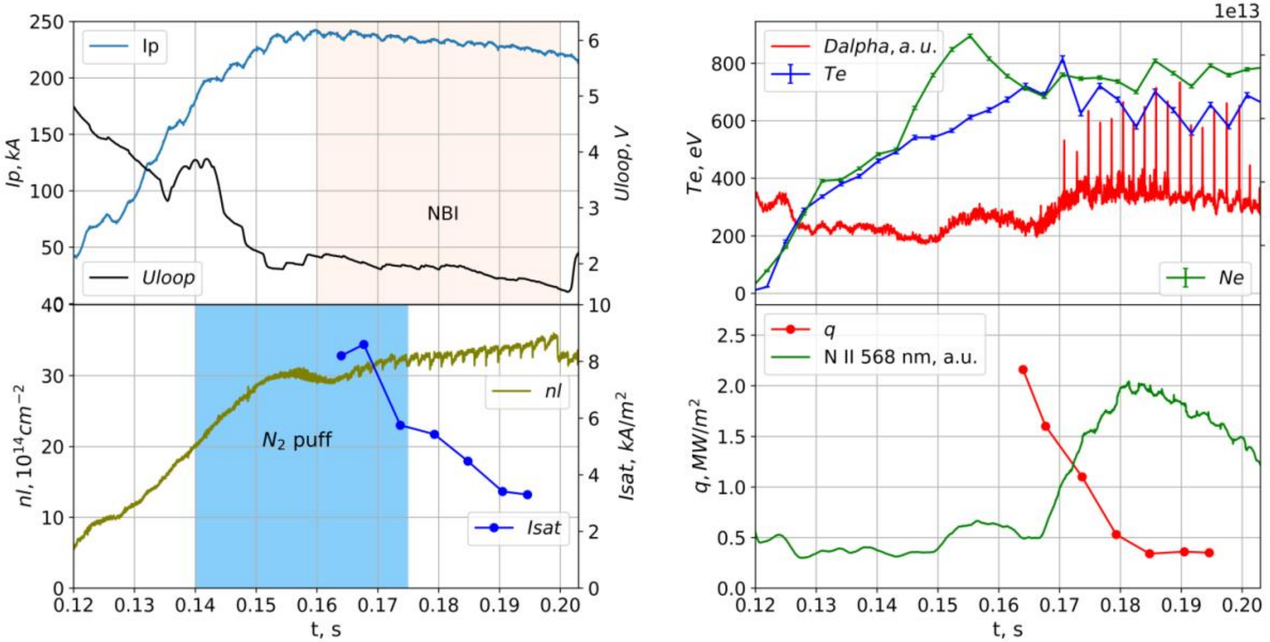
- During TAE resonant transport of fast ions to the periphery is observed
- Size of the loss region shrinks with I_p and B_T increase -> fast ion losses decrease
- This effect is similar to the decrease of the NBI first orbit losses decrease with I_p and B_T rise

The observed dependence of the fast ion loss rate on the plasma current and the toroidal magnetic field due to TAE is favorable for the next generation of spherical tokamaks.

[Bakharev N.N. et al. Physics of Plasmas 30, 072507 (2023)]

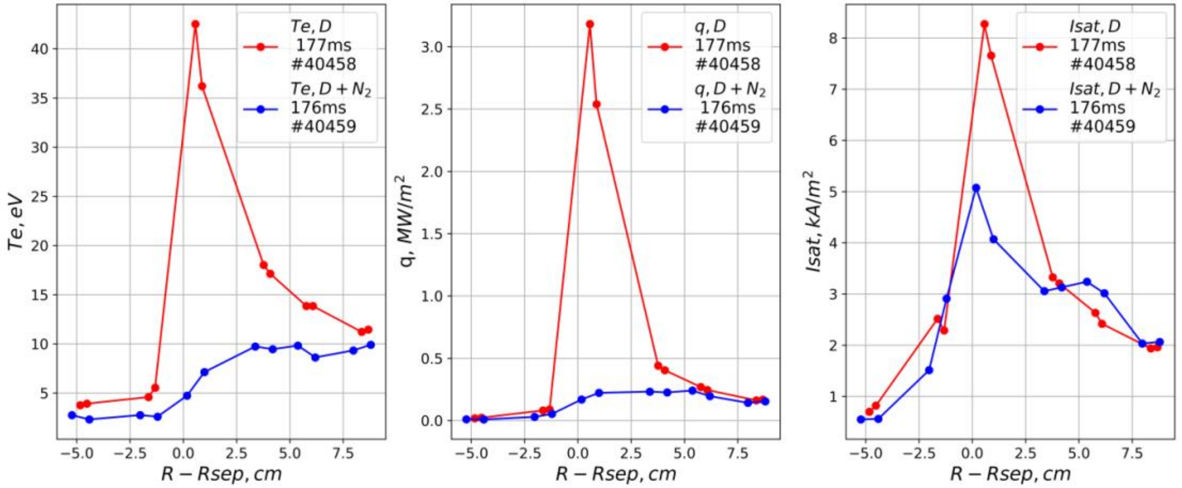
Nitrogen-seeding experiment

Time traces for a typical discharge (#41028) with nitrogen seeding.



- Nitrogen was injected through a capillary on the lower dome in a private flux region according to a prescribed program.
- The nitrogen line sensor that was directed to the divertor region showed a significant increase in intensity 30 ms after turning on the radiating impurity gas injection valve, while the maximum heat flux density to the outer target sharply decreased.
- During the seeding process the parameters of the core plasma measured using Thomson scattering diagnostics did not undergo significant changes.

#40458(D), #40459(D+N_2) $B_t = 0.7$ T, Ip = 230 kA, $P_{NBI} = 700$ kW

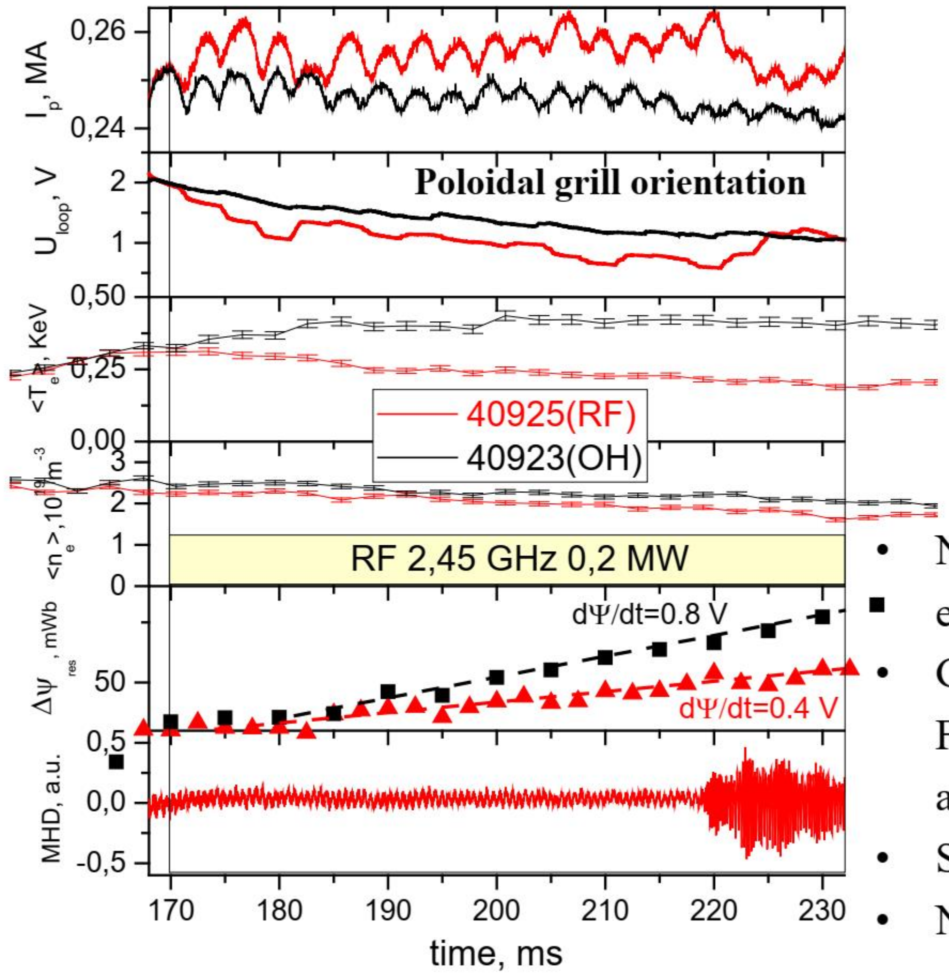


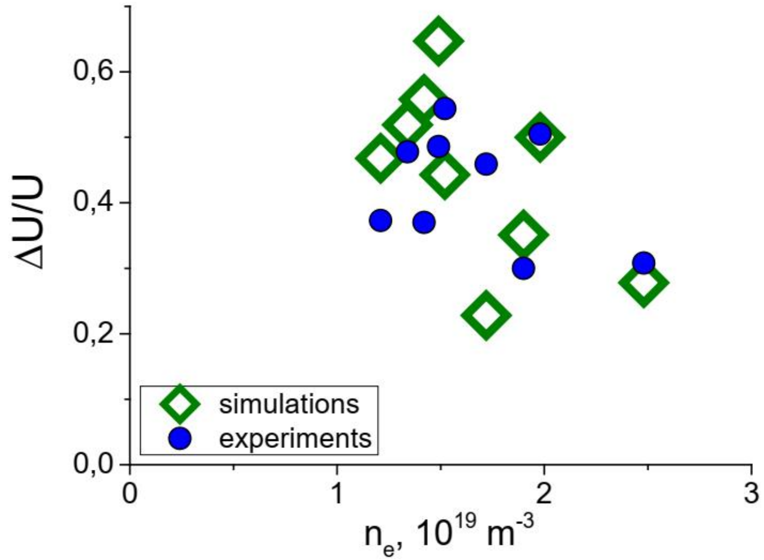
Electron temperature, heat flux density and ion saturation flux density along the outer lower target measured with flush-mounted divertor Langmuir probes.

Conditions for partial detachment have been achieved.

N_2 seeding lead to decrease in the heat flux density by more than an order of magnitude w/o significant confinement degradation in the plasma core.

LHCD simulations compared to experiments





- Non inductive current generation with poloidal antenna orientation is clearly observed in experiments
- Grill3D*, ASTRA** and Fast Ray Tracing Code*** combined to time-dependent 1D Fokker-Planck solver codes are used for simulations of the discharge dynamics with applied RF power
- Satisfactory agreement between ΔU/U ratio in simulations and experiment
- Non inductive current fraction decreases with density rise, $\eta = (0.2–0.4)\cdot 10^{19}$ $A\cdot m^{-2}/W$

***M. A. Irzak and O. N. Shcherbinin 1995 Nucl. Fusion 35 1341**

****G.V.Pereverzev and P.N. Yushmanov Automated System for TRansport Analysis 2002 IPP-Report IPP 5/98**

***** A.N. Saveliev 2017 EPJ Web of Conferences 157 03045**