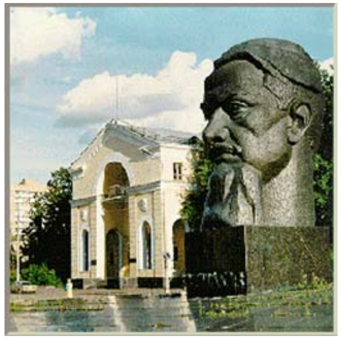
National Research Centre “Kurchatov Institute”



Progress in Magnetic Fusion Technology Summary on FIP, FNS, MTS and SEE sessions

B. Kuteev

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Acknowledgment:

K. Sakamoto, S. Eckstrand, J. Snipes, E. Surrey,
P. Goncharov, B. Kolbasov, V. Sergeev, A. Sivak and
ALL Participants of Fusion Technology sessions

IAEA FEC-25, 13-18 September 2014, St. Petersburg, Russia

INTRODUCTION TO MFT SUMMARY

- **FEC-25 collected 153 contributions on Magnetic Fusion Technology**

  FUSION ENGINEERING, INTEGRATION&POWER PLANT DESIGN
  FUSION NUCLEAR SCIENCE
  MATERIAL TECHNOLOGY SYSTEMS
  SAFETY, ECONOMIC, ENVIRONMENT
- **Overview contributions on ITER project status, construction and IAEA TM&CRP** activity added 1 O, 15 OV and 2 OV/P
- **Sessions statistics**

Oral sessions presented	32 contributions
ITER Technology	8
Heating and Disruption	10
New Devices and Technology	8
Next Step Fusion Nuclear Technology	6
Poster sessions presented	115 contributions

  ITER technology (3) , DEMO design (13), New Devices and Technology (8), Magnets (15), VV&TS (4), Divertor (15), Blanket (10), Heating &CD (15), Diagnostics (12), MPT (25), SEE (9)

- **TRENDS** – ITER shifts to full scale manufacturing of prototypes and parts

  Higher activity in DEMO and FNSF
  Growth of FT contributions on MFT issues in OV&OV/P
  Larger number of MTS and SEE contributions
  Russia announces tighter interlinks of Fusion and Fission

1. ITER
2. DEMO DESIGN
3. NEW DEVICES
4. HEATING
5. MATERIALS
6. SAFETY

Major Achievements

Physics

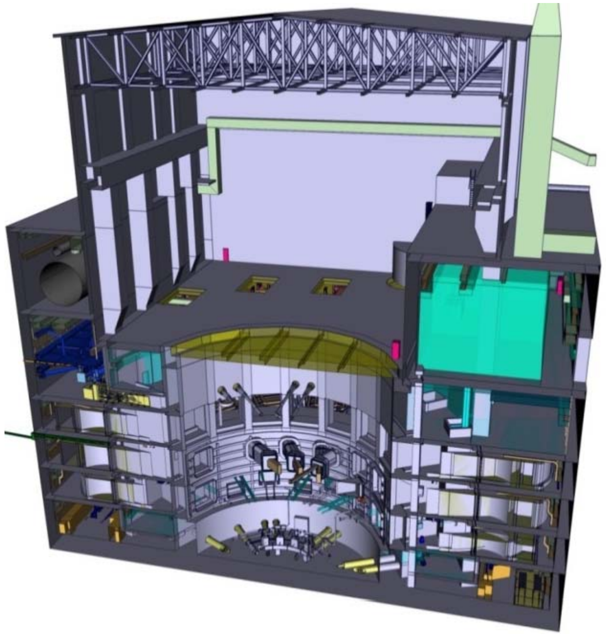
- Overview of Diagnostics Status
- New ITER inner wall shape
- Heating System, NBI, EC etc
- Access to high $Q_{DT} = 10$
- Edge Plasma MHD Stability
- Disruption Mitigation – ITER requirements

Manufacturing

- Vacuum Vessel and Cryostat (EU, KO, IN)
- Poloidal Field Coils: PF Coils (EU & RF); Dummy Conductor (CN)
- Toroidal Field Coils: Conductors : 6 DAs, Coils: EU & TF Coils
- Central Solenoid (US & JA), Correction Coils (CN)
- Central piping procurement :Tokamak Cooling Water System (US)
- First delivery of Plant Components
- Test Convoys

Tokamak Complex Buildings

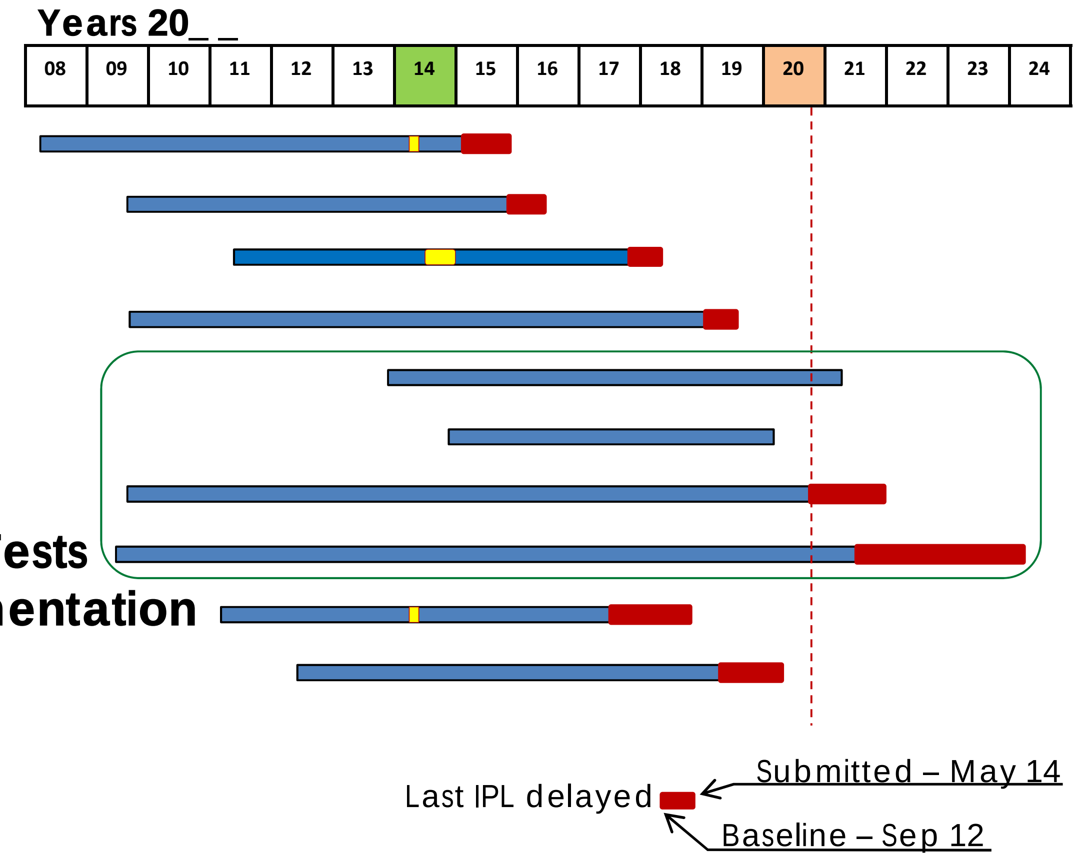
- Dimensions 80*110*60^{ht} m (-16m underground, 350,000tons)
- 493 Seismic Isolation Pit completed on 18 April 2012
- Main B2 slab completed (~14, 000m^3 concrete) on 27 August 2014
- Start erection of walls in October 2014



Tokamak Complex

B2 Slab

RFDA Procurements execution / Tokamak systems



1. **TF Conductors**
2. **PF Conductors**
3. **PF Magnet 1**
4. **Upper Ports**
5. **Blanket First Wall**
6. **Blanket Module Connectors**
7. **Dome divertor**
8. **Plasma Facing Component Tests**
9. **SN, FDU, DC Busbar & Instrumentation**
10. **EC Gyrotrons**

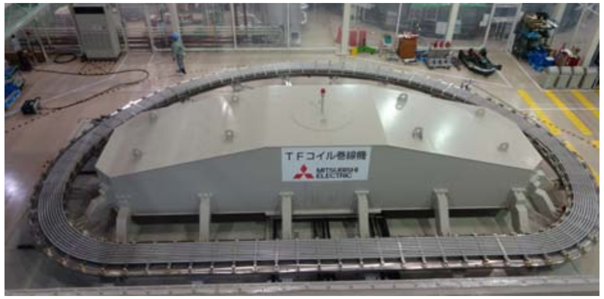
On-schedule
AWP Delayed

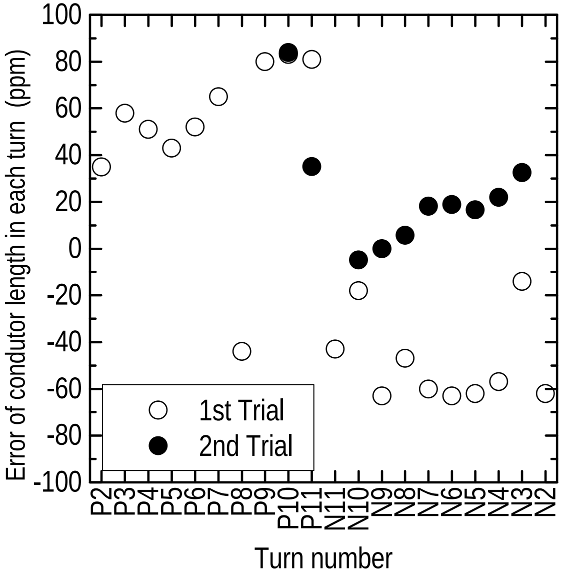
Assembly after FP

Full-scale trial results to qualify optimized manufacturing plan for ITER Toroidal Field coil winding pack in Japan

FIP-1-3
N. Koizumi et al.

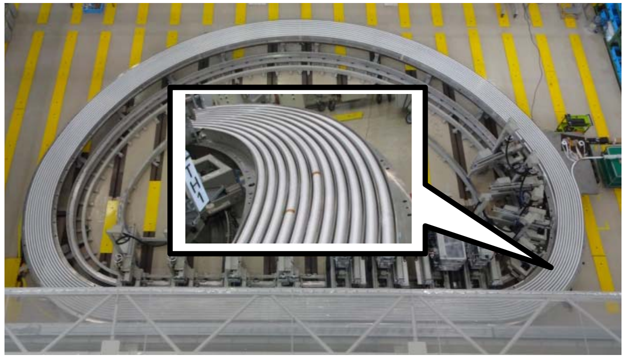
Dummy double-pancake (DP) winding was completed.





Target tolerance of ±0.01% in conductor length was achieved.

Transfer of RP between dummy DP was completed.



Conductor could be transferred into RP groove after turn insulation.

Heat treatment trial of dummy windings was carried out.

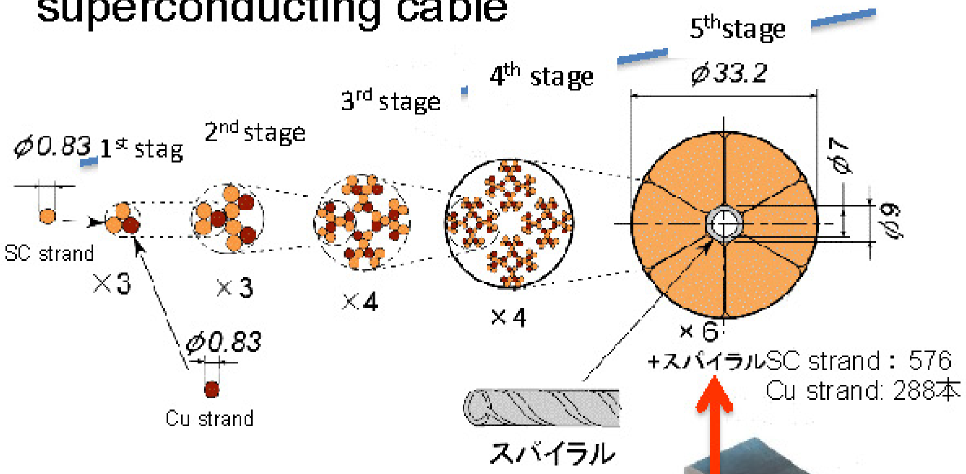
Elongation of heat-treated conductor was evaluated to be about 0.06% with scatter smaller than 0.01%. This enables highly accurate prediction of conductor elongation by heat treatment to determine the winding dimension.

- These successful results allow JADA to start the first TF coil fabrication. 4 DP winding was completed and the 1st DP was successfully heat-treated.

Advances in superconductors for ITER

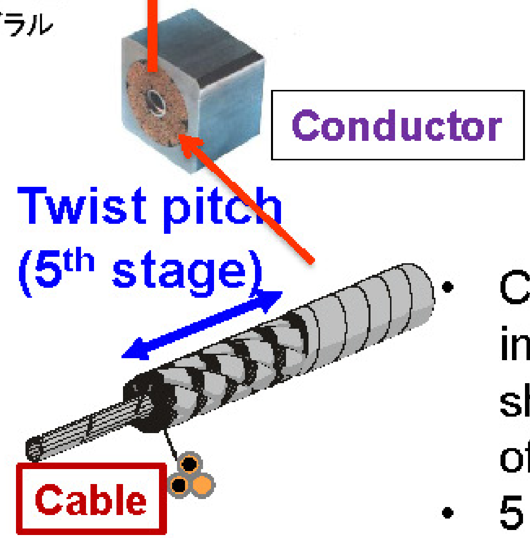
P. Bruzzone et al. FIP/1-4Ra; V. Vysotsky et al. FIP/1-4Rb; Y. Nunoya et al. FIP/P4-21;

ITER CS conductor superconducting cable

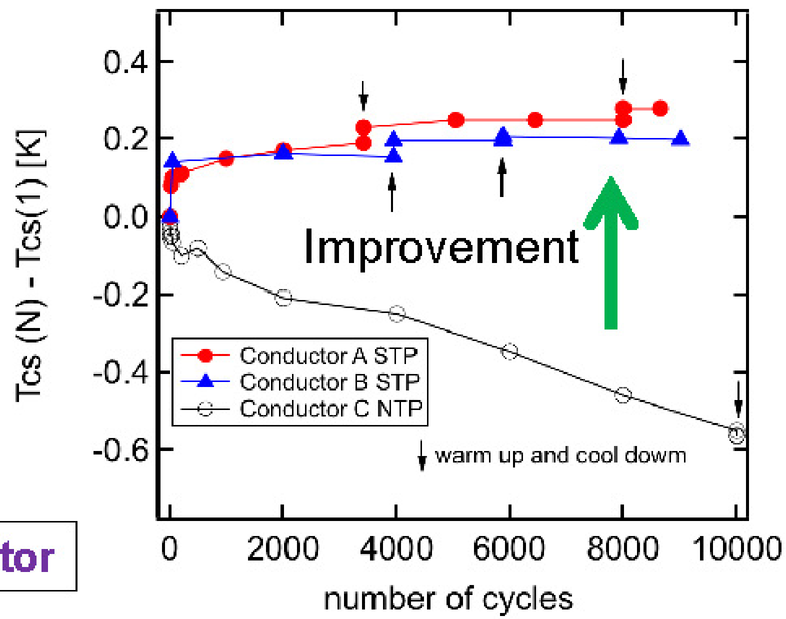


	Twist Pitch[mm]	
Cabling stage	Original cable	Improved cable (A/B)
1st	45	23/20
2nd	85	48/45
3rd	145	88/80
4th	250	150
5th	450	452/450

Twist pitch: the longitudinal length necessary for one twist



Conductor performance test at SUTAN facility



- CS conductor performance was improved by the cable optimization to shorten the twist pitch, enabling start of CS cable manufacturing in Japan.
- 5 conductors was completed and delivered to US in June, 2014.

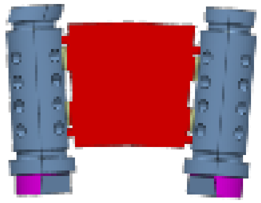
➢ The production line of VNIIKP successfully passed all qualifications procedures

Manufacturing Design and Progress of the First Sector for ITER Vacuum Vessel (FIP/1-6Rb)

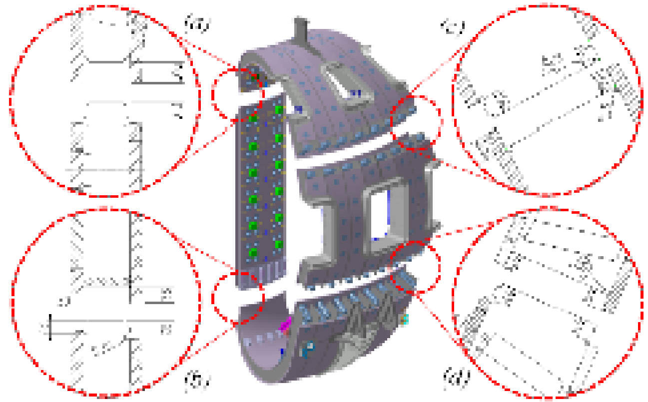
H.-J. Ahn et al.

☐ Manufacturing Design of the First Sector

- The manufacturing design of the first sector has been developed in accordance with the RCC-MR code and the regulatory requirements by HHI as a supplier.
- The design of Korean VV sectors introduces special concepts like a self-sustaining welded IWS rib and cup-and-cone type segment joints to minimize welding distortion.



Self-Sustaining IWS Rib



Cup-and-Cone Type Segment Joints

☐ Manufacturing Progress in Korea

- Several real scale mock-ups had been constructed to verify and develop the manufacturing design and procedures.
- The first sector has been manufacturing slowly at the front of ITER project as a nuclear component since 2012.
- All poloidal segments for the first sector are being fabricated simultaneously. Fabrication speed could be getting better after solving current issues.

Progress of Upper Segment for the First Sector

Progress in the Design and Manufacture of High Vacuum Components for ITER

FIP/1-6Ra C. Sborchia

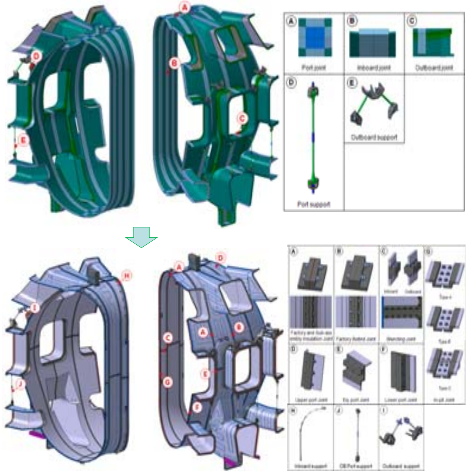
Manufacture of VV Sector#6 and lower port inner shells (courtesy of KO DA)

Manufacture of Cryostat base pedestal ring and sandwich structure (courtesy of IN DA)

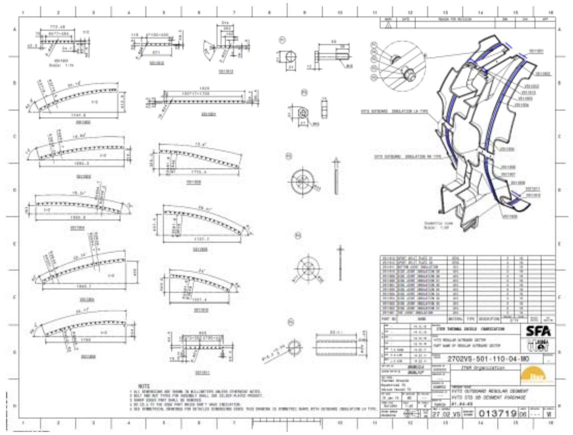
Design Finalization and R&D Activities before the Start of Manufacture of ITER Thermal Shield

(FIP P4-12)

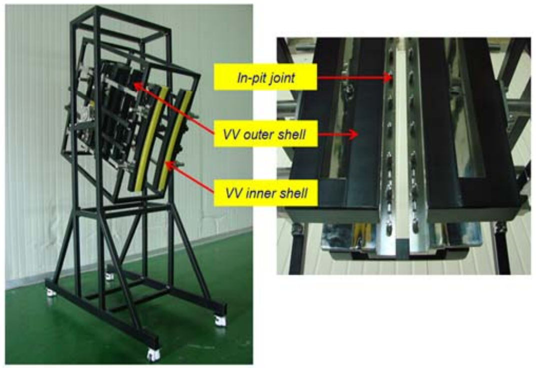
- Authors: W. Chung et al, ITER Korea
- Highlights
  - The final design of the TS was completed in Sep. 2012 and the manufacturing design was then followed to make manufacturing drawings.
  - Manifold design for the coolant supply to the TS was performed and its structural integrity was verified.
  - Two kinds of sector field joints were made and their assemble feasibilities were checked. Complex shape of cooling tube routing for VVTS lower port was made by a novel bending method.
  - Full-scale mock-up for VVTS 10 degree section was made before the start of the TS manufacture.



VVTS design update



VVTS manufacturing drawing



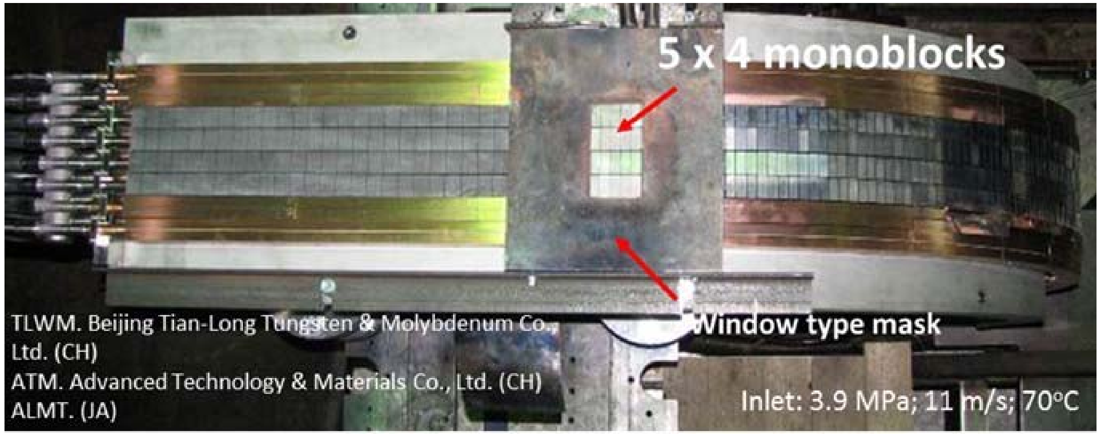
In-pit joint test mock-up

Full scale mock-up of VVTS 10 degree section

FIP/1-1

Development of Tungsten Monoblock Technology for ITER Full-Tungsten Divertor in Japan

Y. SEKI, K. Ezato, S. Suzuki, K. Yokoyama, K. Mohri (JAEA), T. Hirai, F. Escourbiac (ITER Org.), V. Kuznetsov (NIIEFA)



- The full-W divertor qualification program has been implemented by JAEA. As the first phase, the technology validation and demonstration of the full-W divertor, the full-W small-scale mock-ups were manufactured and HHF tested at IDTF in Saint Petersburg. JAEA succeeded in demonstrating the durability of the W divertor for a repetitive heat load of 10 MW/m^2 × 5000 cycles and 20 MW/m^2 × 1000 cycles.
- JAEA demonstrated first in the world that W monoblock technology is capable of withstanding the heaviest heat loads specified for the ITER full-W divertor without macroscopic crack, melting and degradation of the heat removal capability.

Technical achievements demonstrated by JAEA provided an essential boost for full-W divertor.

- IC-13 (Nov 2013) endorsed the STAC recommendation on full-W divertor as the first divertor.

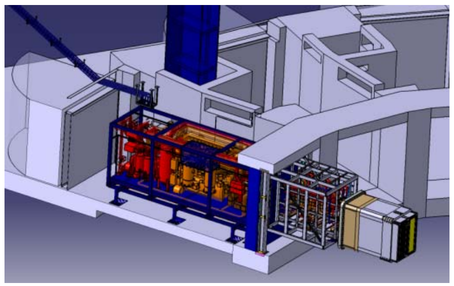
Current Status of Chinese HCCB TBM Program

Presented by: K.M. Feng, SWIP/China FIP/3-5Ra

Summaries:

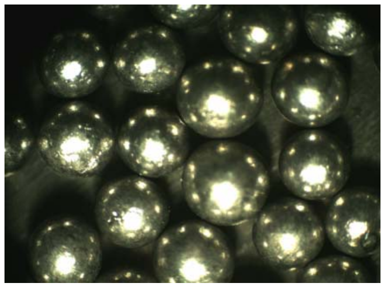
- . Helium-cooled ceramic breeder (HCCB) test blanket module will be the primary option of the Chinese ITER TBM program.
- The Conceptual Design Review (CDR) for HCCB TBS was hold in July 2014 in ITER IO.
- Related R&D on key components, materials, fabrications and mock-up test have being implemented.
- 4.5 tons ingots and 2.5 dpa of neutron irridiation data for Chinese RAFM (CLF-1) have been obtained.
- The ceramic breeder pebble Li4SiO4 of kg-class was fabricated by using the melt spraying method.
- The Be pebble of kg-class was fabricated by using the REP method.

Signing TBMA for CN HCCB TBS

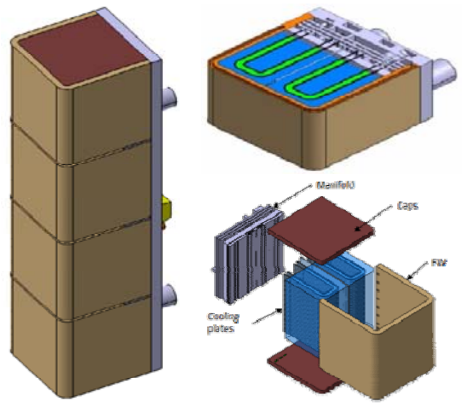


HCCB TBS Sub-system

Li4SiO4 Pebble



Be Pebble

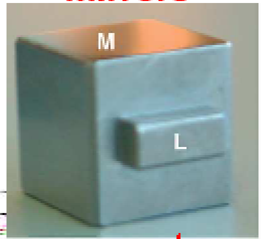


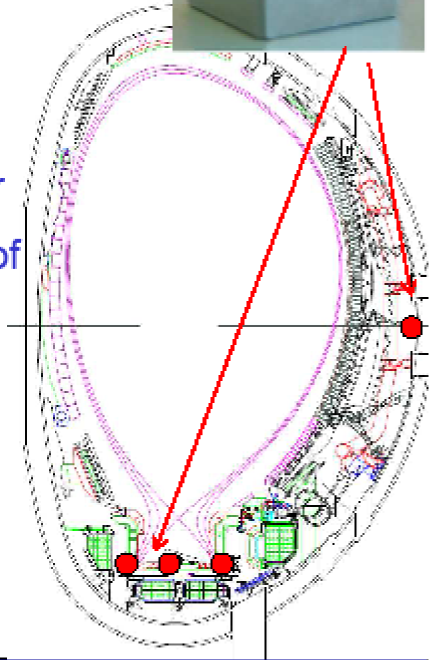
HCCB TBM Design

Comprehensive First Mirror Test for ITER at JET with ITER-Like Wall

- All optical diagnostics in ITER will rely on first mirrors.
- First Mirror Test has been carried out since 2002 in JET-C and JET-ILW over 90 specimens in the divertor and on the main chamber wall.

Mo and Rh-coated mirrors





Main Results

❖In JET-C reflectivity of all tested mirrors was degraded.

❖In JET-ILW molybdenum mirrors on the main chamber retained reflectivity in the visible and IR range.

❖30% reflectivity loss of Rh-coated mirrors in the main chamber

❖Divertor mirrors in JET-ILW lost reflectivity by co-deposition of Be, W, C, N with fuel species.

Concluding Remarks and Outlook

❖*Refreshing of mirror surfaces on the main chamber wall by periodic evaporation of Mo layer.*

❖*Need for mirror replacement in the ITER divertor.*

❖*Mirror test in the ITER-relevant geometry from 2015.*

1. ITER
2. DEMO DESIGN
3. NEW DEVICES
4. HEATING
5. MATERIALS
6. SAFETY

EU DEMO Design Point Studies – FIP/3-3

R. Kemp, D. J. Ward, G. Federici, R. Wenninger and J. Morris

- Conceptual DEMO designs are created and optimised using the systems code PROCESS
- These studies allow identification of the most significant physics and technology limitations in improving the design, and demonstrate the trade-offs required for optimisation.
- Operating space – pulse length and net electrical power – has been explored to find a performance target with maximum likelihood of successfully demonstrating net electrical power from fusion.
- The EU is carrying out a comprehensive scan in aspect ratio for a pulsed DEMO, looking at the performance and cost variation, and the knock-on effects on engineering issues.
- A steady-state DEMO is also being investigated, with technological and physics enhancements over pulsed DEMO.
- Reducing the toroidal field ripple to an acceptable value in DEMO imposes additional engineering constraints on the design.

Physics and Engineering Studies of the Advanced Divertor for a Fusion Reactor

FIP/3-4Ra N. Asakura, K. Hoshino, H. Utoh, et al. JAEA, Toshiba Co., Nagoya Univ., NIFS

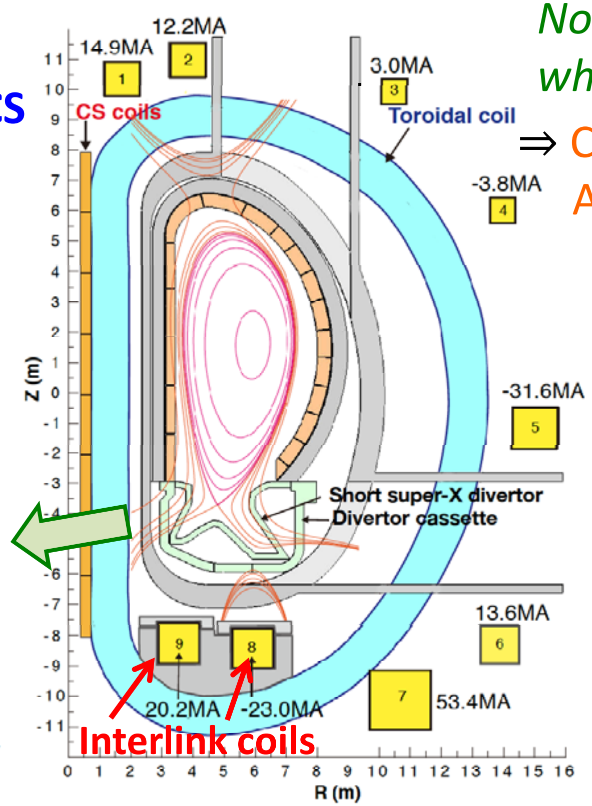
Advanced divertor study will provide new options of the divertor configuration:
Physics advantages and Engineering issues of "*Short Super-X divertor*" *(short SXD)*
has been studied in SlimCS (FP: 3GW, R_p: 5.5m, I_p: 16.7MA).

- **Interlink divertor coils are required: Nb_3Al SC is preferable for React&Wind**
  ⇒ SC filament size should be reduced, and EM-force on IL-coil support is required.
- f_{exp} and $L_{//}$ *to target* are increased along the divertor leg: max. 19 times and 2 times.
- **Power handling were investigated by SONIC for P_{FP}= 3GW reactor (P_{out}=500MW)**
  ⇒ Radiative area is narrow poloidally, and efficient to produce full detachment:

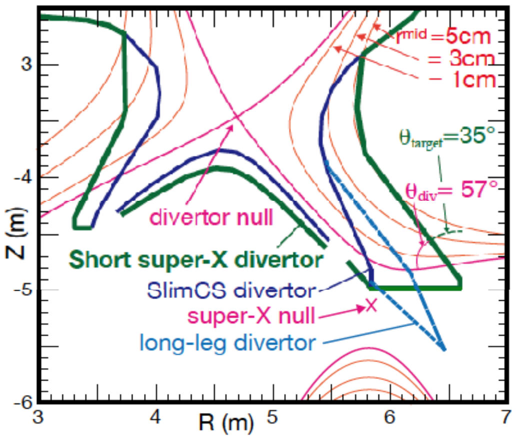
Note: Total peak heat load is ~10MWm^{-2}, where Surface recombination is dominant.

⇒ Conventional divertor is the first choice: Advanced div. is studied for alternative.

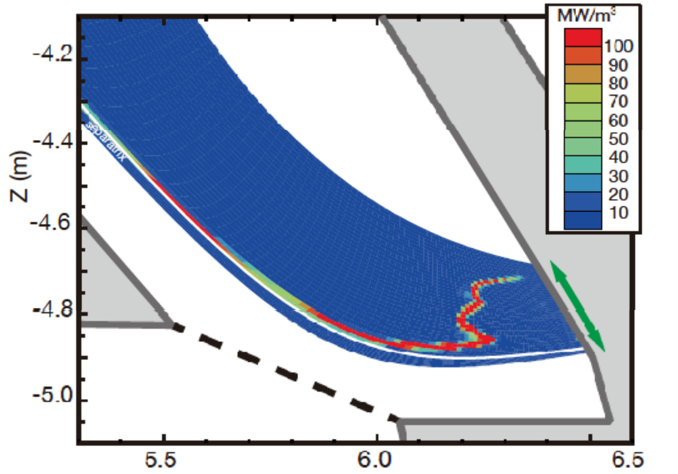
2 Interlink coil arrange for SlimCS



Magnetic configuration of short-SXD and Conventional divertors



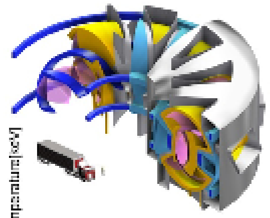
Radiation distribution in short-SXD



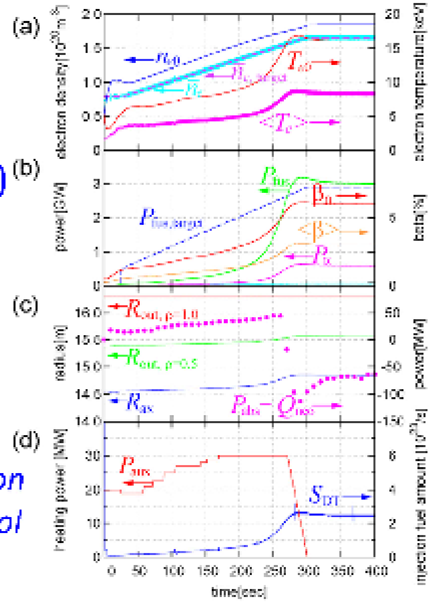
FIP/P7-16

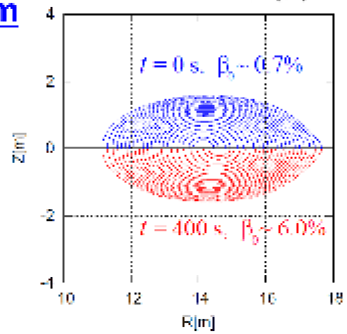
Integrated Physics Analysis of Plasma Operation Control Scenario of Helical Reactor FFHR-d1

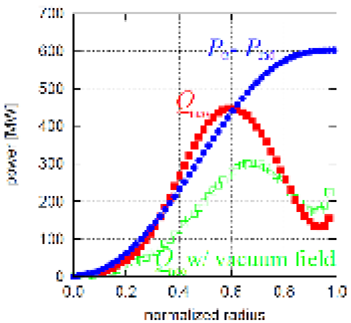
by Takuya GOTO *et al.*, (NIFS, Japan)



- Conceptual design of LHD-type helical fusion reactor FFHR-d1 has been advanced
  - Take full advantage of the characteristics of **net-current-free plasma** (no disruption, no current drive)
  - Detailed physics assessment of the core plasma and 3D CAD design have been carried out
- Plasma operation control scenario of FFHR-d1 has been discussed
  - **Stable control with a small number of simple diagnostics** can be realized (*by fuelling control based on line-averaged electron density and heating power control based on edge electron density and fusion power*)
  - **Startup scenario consistent with MHD equilibrium and neo-classical transport** can be achieved with adequate vertical field control
- More precise physics analysis is needed to confirm this scenario
  - MHD stability, alpha particle confinement, energy transfer from electrons to ions, etc.



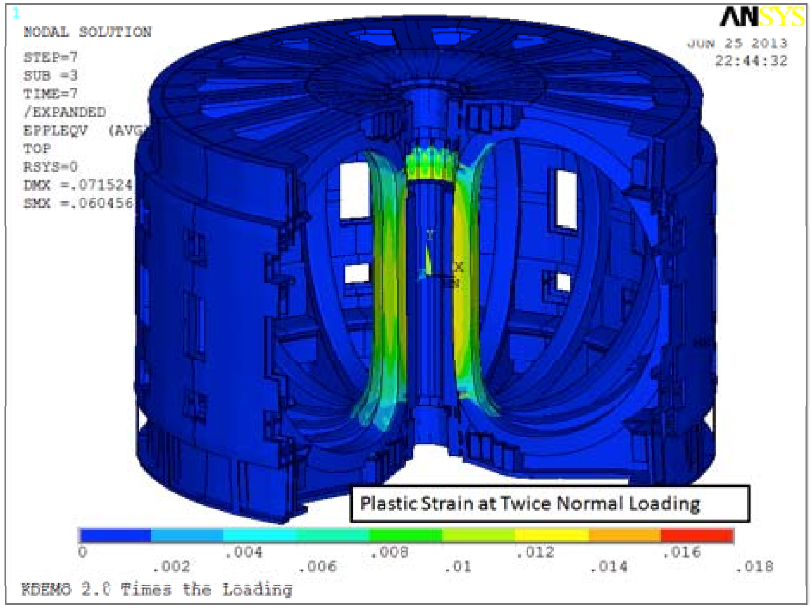


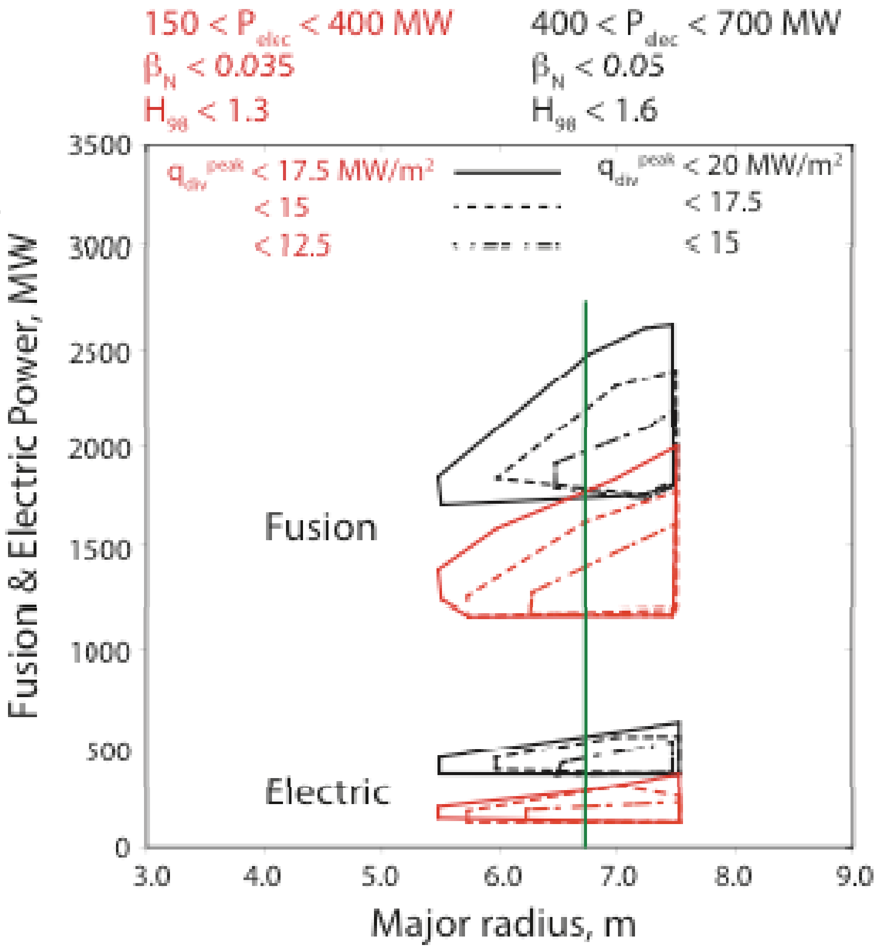


Physics and Engineering Assessments of the K-DEMO Magnet Configuration

G. Nielson et al
FIP/P7-2
K.Kim, FIP/3-6

- K-DEMO design point at R = 6.8 m, B = 7.4 T provides operating space with margin against physics uncertainties.
- Nonlinear, elastic-plastic analysis at 2 × normal load shows adequate structural margin.

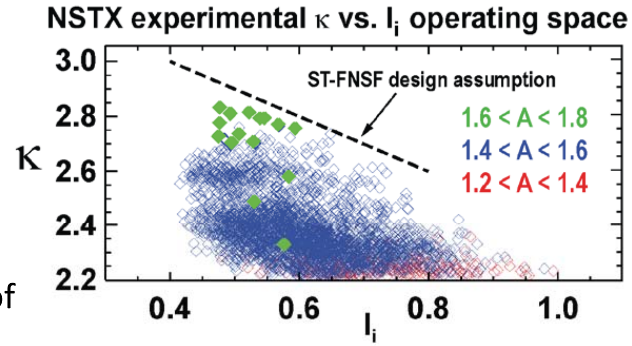




Configuration Studies for an ST-based Fusion Nuclear Science Facility FNS/1-1

J. Menard/L. El-Guebaly et al.

$$\kappa_{x\text{-point}} = \kappa_{max\text{-ST}}(li) \equiv 3.4 - li$$



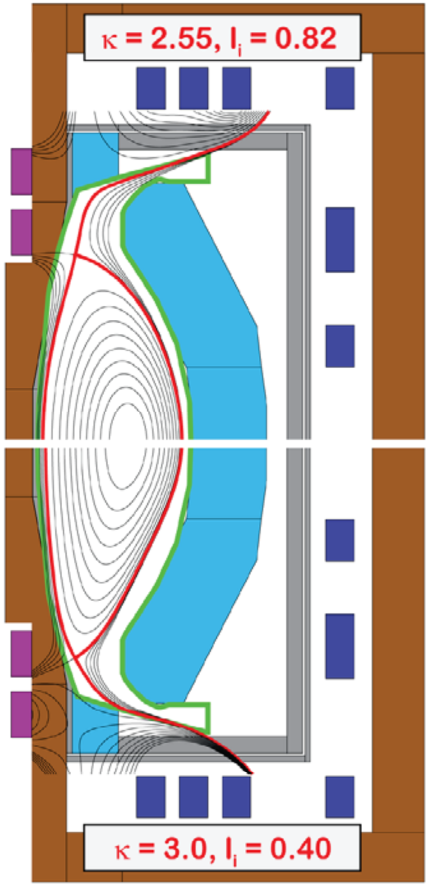
During the past two years, U.S. studies have for the first time developed ST configurations simultaneously incorporating:

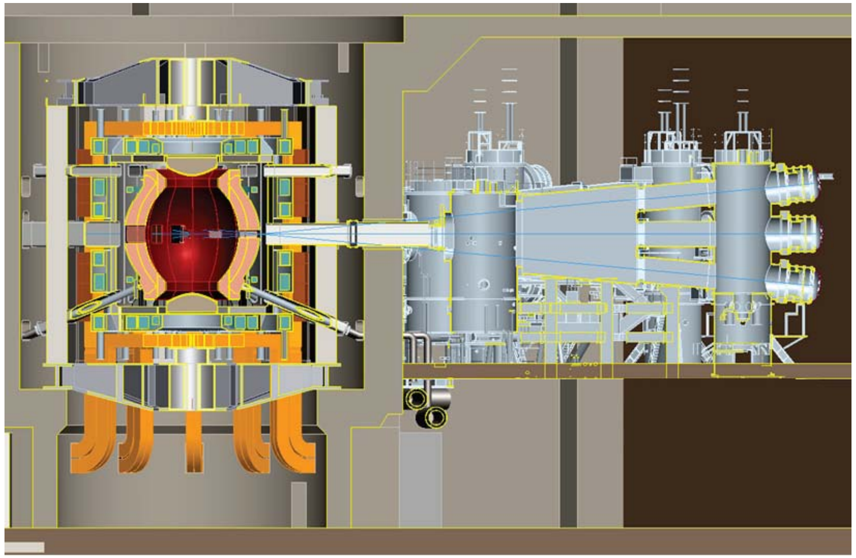
(1) a blanket capable of TBR ~ 1 with ports provided for test modules and heating and current drive,

(2) (2) a poloidal field (PF) coil set supporting high κ and δ for a range of li and βN values consistent with NSTX/NSTX-U operation,

(3) (3) a long-legged / Super-X divertor [8] analogous to the planned MAST-U divertor [9] which substantially reduces projected peak divertor heat-flux and has all outboard PF coils outside the vacuum chamber and as superconducting to reduce power consumption, and

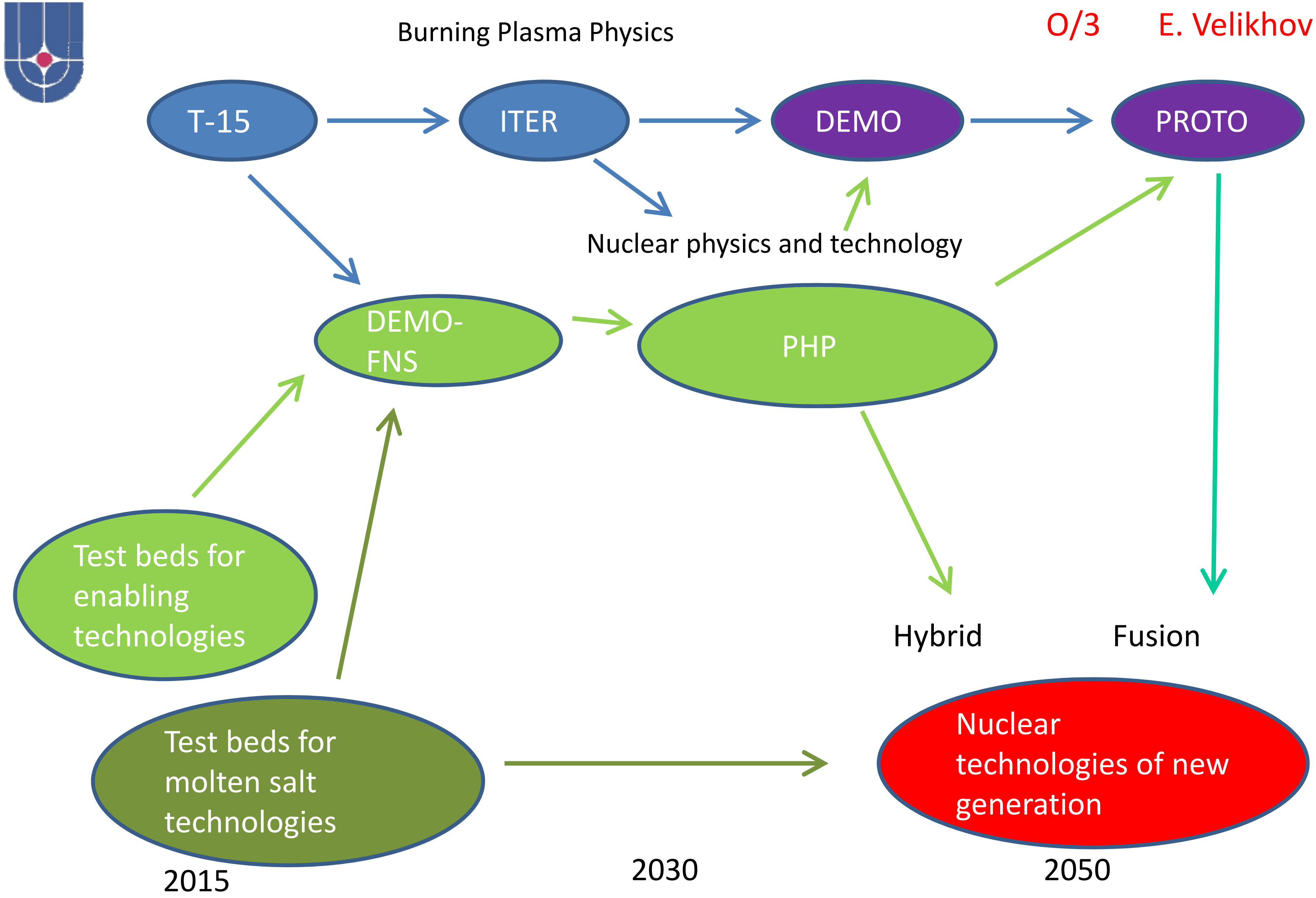
(4) (4) a vertical maintenance scheme in which blanket structures and the centerstack (CS) can be removed independently.





Progress in these ST-FNSF mission vs. configuration studies including dependence on plasma major radius R0 for a range R0 = 1 – 1.6m was described.

Strategy 2013 for Fusion-Fission development in Russia



Major facilities on the path to Industrial Hybrid Plant

O/3 E. Velikhov

Test beds	Russian Tokamaks	DEMO-FNS	
Steady State Technologies		**DT neutrons**	**MS blankets**

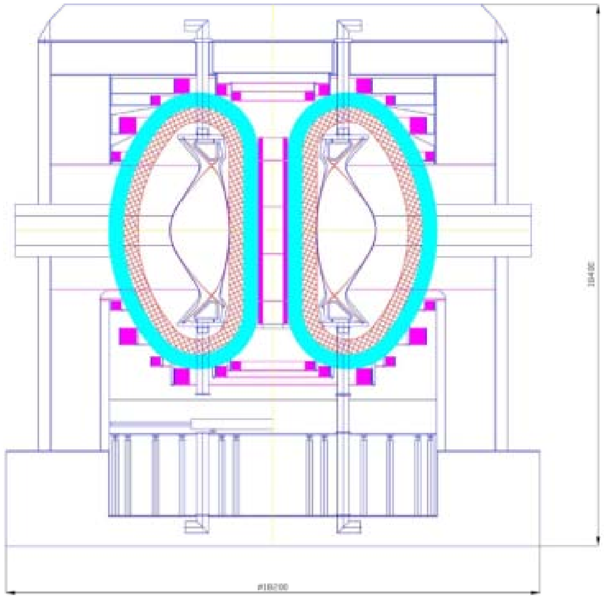
- Magnetic system
- Vacuum chamber
- Divertor
- Blanket
- Remote handling
- Heating and current drive
- Fuelling and pumping
- Diadnostics
- Safety
- Molten salts

•Integration

•Materials

•Components

•Licensing



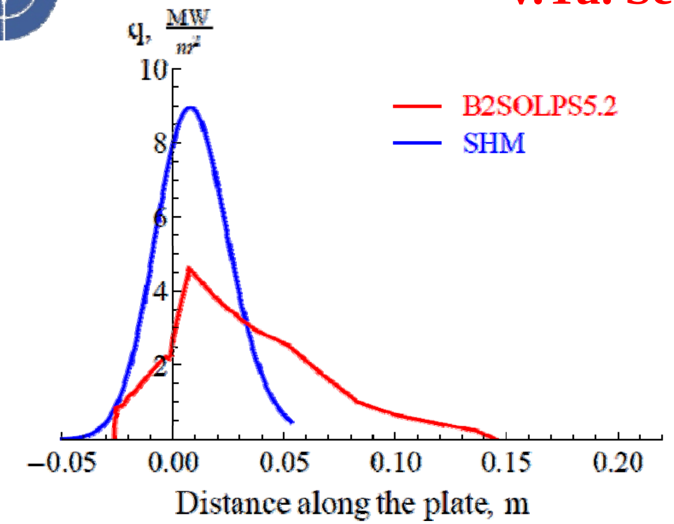
•Hybrid Technologies

Pilot Hybrid Plant construction by 2030 **P=500 MWt, Q_{eng} ~1**

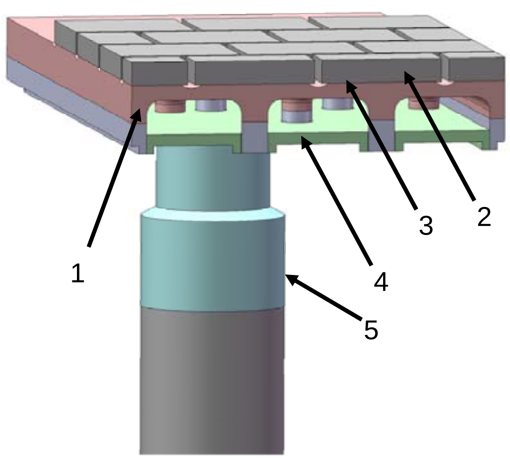
Industrial Hybrid Plant construction by 2040

P=3 GWt, Q_{eng} ~6.5, P=1.3 GWe, P=1.1 GW(net), MA=1t/a, FN=1.1 t/a

Design of Divertor and First Wall for DEMO-FNS
V.Yu. Sergeev et al., Paper FIP/P7-9



- Double Null divertor with long external leg and V-shaped corner is accepted.
- Beryllium tiles with liquid lithium is used for edge plasma control.
- Maximal heat flux density of 5-9 MW/m^2 is evaluated by B2SOLPS5.2 and the Semianalytical Hybrid Model (SHM) codes for configuration with small plasma-separatrix gap. Neon puff in vicinity of strike point is foreseen for detachment.
- Mock-up of the water-cooled first wall element of DEMO-FNS with beryllium tiles was successfully tested. No tiles lost the mechanical and thermal contacts at both 5 MW/m^2 (sustained 1000 cycles) and at 10.5 MW/m^2 (sustained 100 cycles).



Sketch of mock-up : (1) heat carrier plate made of chromium–zirconium bronze, (2) sectioned beryllium tiles, (3) brazing layer, (4) stainless-steel base (vacuum vessel), (5) pipe for heat carrier (water) flow.

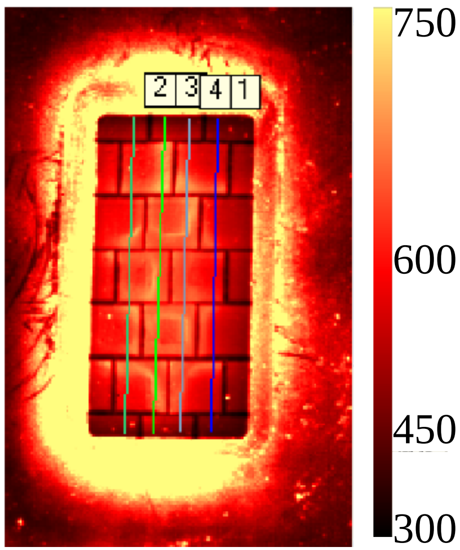


Photo of the mock-up by infrared camera during cycle of heat loads (15 sec -load/15 sec - pause): Load - 10.5 MW/m^2.

1. ITER
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PROGRESS OF THE CEA CONTRIBUTIONS TO THE BROADER APROACH PROJECTS

DE LA RECHERCHE À L'INDUSTRIE
cea

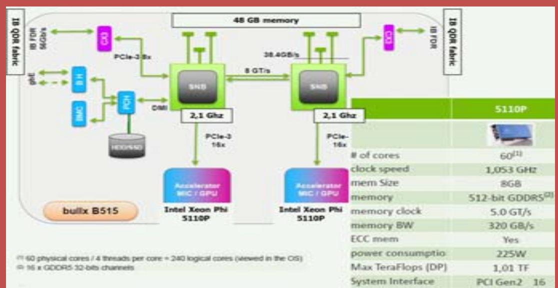
FIP/P7-21; FNS/1-2Ra

IFERC CSC

Helios Supercomputer
Bullx B515 HPC

Scientific exploitation started
Jan.2012 1.24 Pflop/s (Linpack)

85% < Usage < 90%



Mid 2014 Helios Upgraded
to 1.98 PFlop/s (peak)
with 0.43 PFlop/s of new Intel
Xeon Phi processors

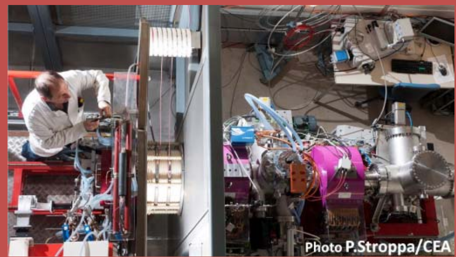
IFMIF-EVEDA

IFMIF Intermediate Design Report
delivered (BA partners)

LIPAc prototyping:

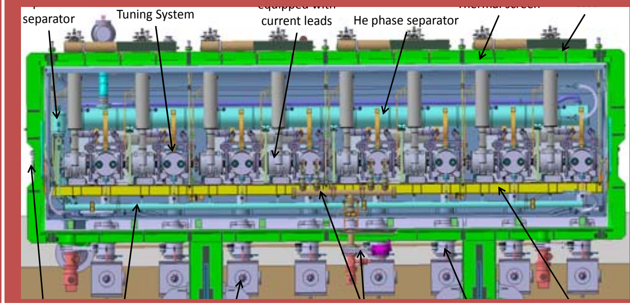
Beams dynamics studies and
LIPAc design completed

D^+ Injector delivered 140mA-100 keV



Beam diagnostics delivered

SRF Linac Design completed



JT-60SA

TF coils manufacture started

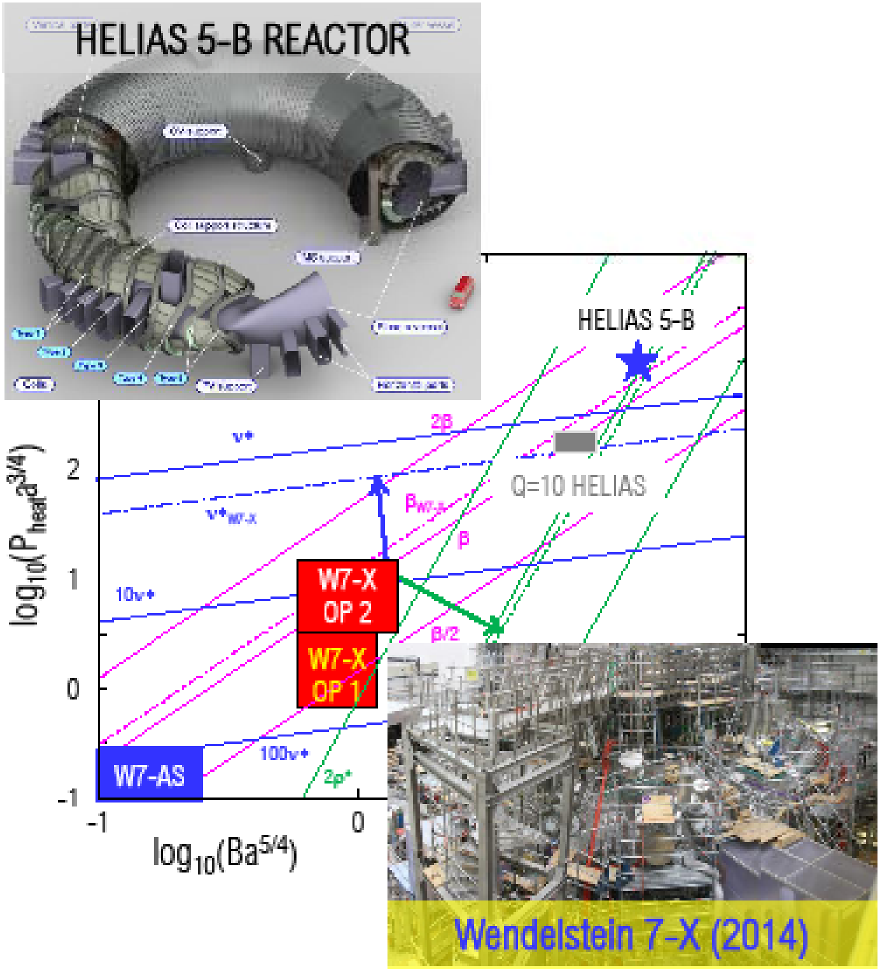
TF coils Cold Test Facility
assembled

Cryogenic System completed

TFcoil structures manufacture
started

Magnet Power Supplies design
& manufacture started

The Road to a HELIAS Reactor



W7-X Initial Operation Phases (OP)

2015 - OP1.1 (short-pulse limiter phase)

- verify good flux-surfaces
- first plasma

2016/17 - OP1.2 (pulsed, un-cooled divertor)

- prepare for steady-state/high-power operation
- assess impact of stellarator optimization
  1) increase density & develop scenarios
     - fuelling/density control, heating
     - prepare safe divertor operation
  2) use configuration flexibility
     - study effects of optimization
  3) <u>begin</u> to address physics topics
     - divertor physics
     - impurity transport/PWI
     - transport: neoclassical, turbulence, …
     - fast ion generation
     - high-beta & MHD

>2019 – OP 2 (steady-state divertor)

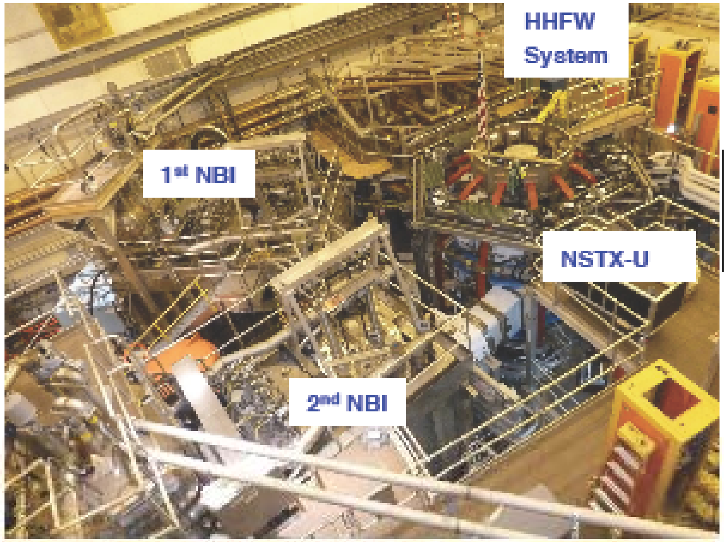
- to integrated, steady-state, high-power/high-density operation

FIP/P8-30: NSTX-U First Plasma Scheduled in February 2015

To provide data base to support ST-FNSF designs and ITER operations

Key issues to resolve for next-step STs

- Stability and steady-state control at high β
- Confinement scaling (esp. electron transport)
- Non-inductive start-up, ramp-up, sustainment
- Divertor solutions for mitigating high heat flux



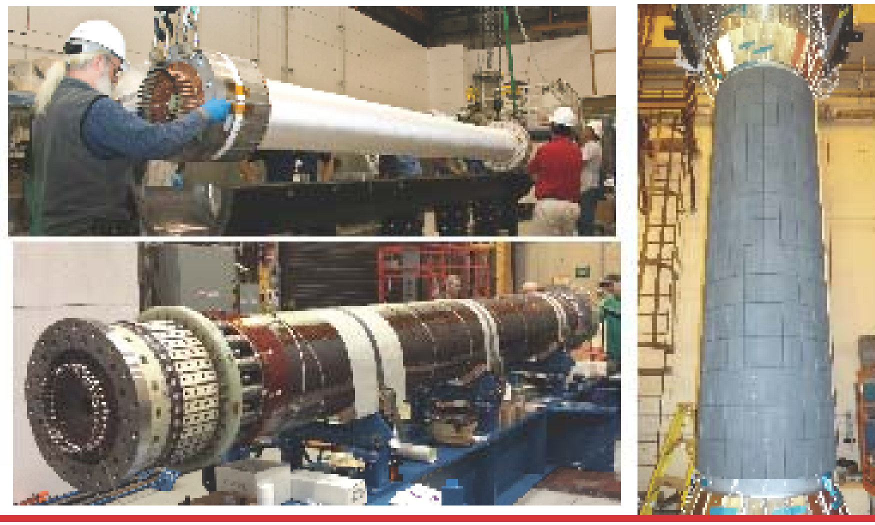
	R_0 (m)	A_{min}	I_p(MA)	B_T (T)	T_{TF}(s)	R_{CS} (m)	R_{OS} (m)	OH flux (Wb)
NSTX	0.854	1.28	1	0.55	1	0.185	1.574	0.75
NSTX-U	0.934	1.5	2	1	6.5	0.315	1.574	2.1

~ X 5 - 10 increase in $n\tau T$ from NSTX

NSTX-U average plasma pressure) ~ Tokamaks

2nd NBI Being Installed

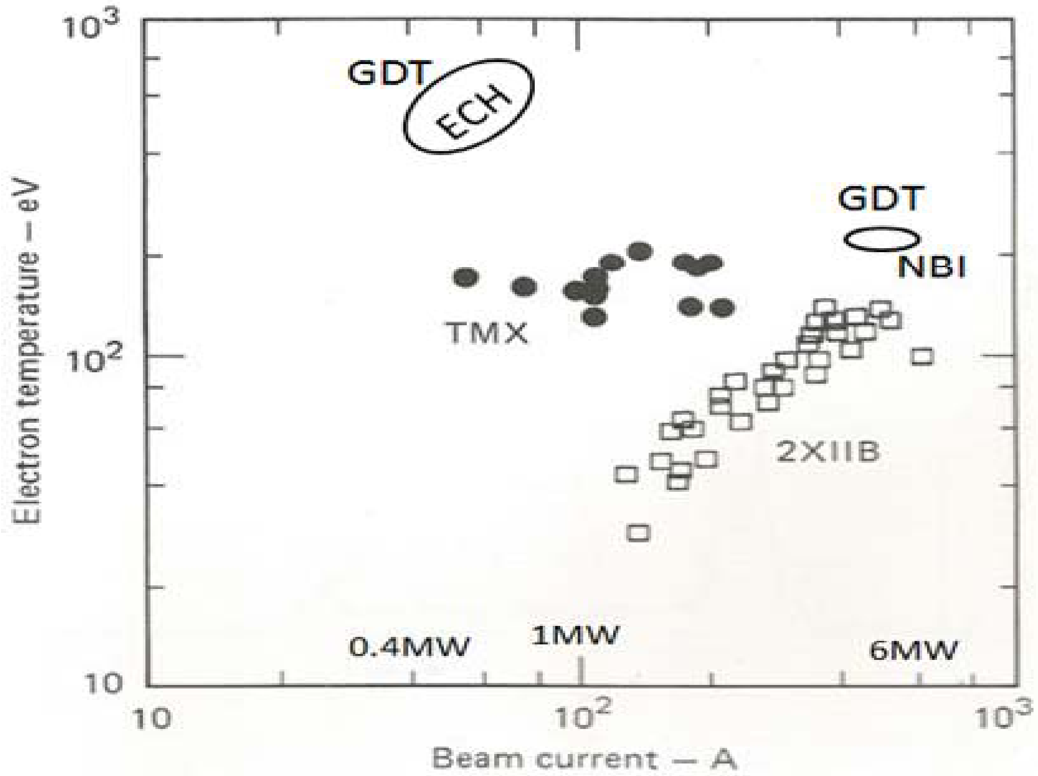
New Center-Stack Nearing Completion

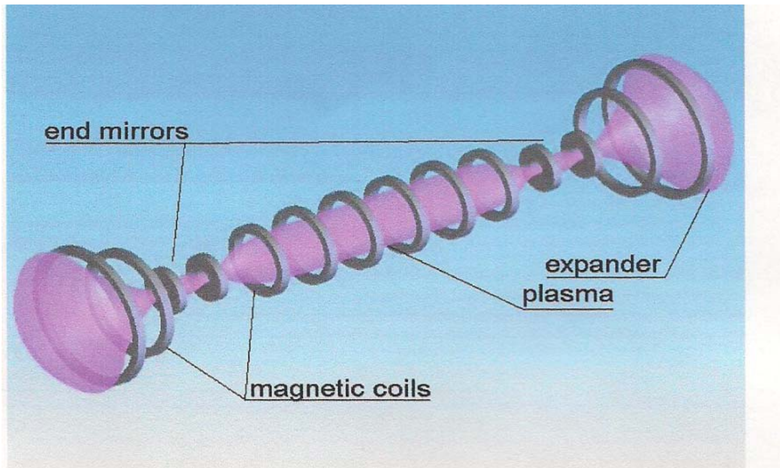


GDT With ECH Te = 750 eV

Sufficient for a D-T Fusion Neutron Source

- Magnetic Mirror Device Historic Thompson Data:
  - 2XIIB (1977), TMX(1980), GDT(2014)





EX/P4-21 P. Bagryansky et al.

ADX: a High Field, High Power Density, Advanced Divertor Test Tokamak

The MIT PSFC and collaborators are proposing a new high-field (6.5 tesla), high power density ($P/S \sim 1.5$ MW/m^2) *A*dvanced *D*ivertor e*X*periment to perform critical R&D on the pathway to a DEMO:

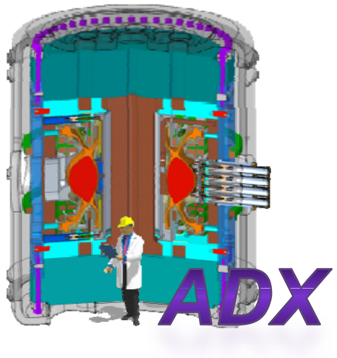
1. **Demonstrate robust divertor power handling physics solutions, at DEMO-level heat flux densities**
2. **Demonstrate nearly complete suppression of divertor erosion**
3. **Demonstrate low PMI, efficient, RF current drive and heating technologies that scale to steady-state**
4. **Achieve 1, 2, 3 with core plasma performance *compatible with obtaining a burning plasma***

- **Background**
  - **Recent results [1] project to *very narrow power exhaust channel widths* for ITER and future DEMOs, $\lambda_q \sim 1$ mm. Parallel heat fluxes, $q_{//}$, scale as**

$$q_{//} \sim P_{SOL} B/R$$

Power exhaust for a DEMO will be 3-4 times higher than ITER, with the additional need to completely suppress divertor erosion. *New divertor solutions are required*.

  - **Just as important: efficient, low PMI, *RF current drive and heating* technologies that scale to steady state must be developed for a DEMO.**



- **ADX employs high-field, demountable toroidal field magnet technology of Alcator C-Mod**

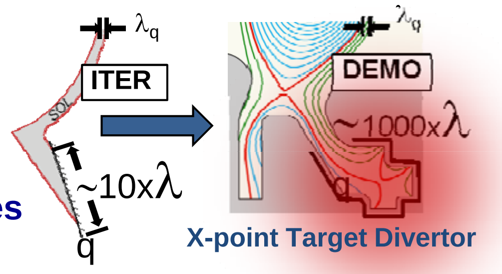
ADX will uniquely access reactor-level $q_{//} \sim P_{SOL}B/R$ (~ 125 MW-T/m), with the unique-in-the-world ability to operate with key divertor parameters (B, $q_{//}$, n_{div}, $\lambda_{debye}/\lambda_{ion}$, ρ_z/λ_{ion}, ...), *identical to a reactor*, while implementing advanced magnetic divertor topologies with an internal poloidal field coil set.

- **Innovative Divertor Solutions for a DEMO**

ADX will test advanced divertor topologies, including *Super-X*, *X-point Target* long-leg divertor concepts with options for *heated and liquid metal targets*.

- **Innovative RF Current Drive/Heating Solutions for a DEMO**

ADX will employ *high-field-side-launch Lower Hybrid current drive and ICRF systems* – for the first time in a diverted tokamak – which project very favorably to high current drive efficiency and dramatically reduced PMI. ADX will develop RF physics/technology at the magnetic fields (6.5T) and densities of a DEMO.



X-point Target Divertor

[1] Eich, *et al.*, NF 53 (2013) 093031.

B. LaBombard, E. Marmar, J. Irby, J. Terry, R. Vieira, D.G. Whyte, S. Wolfe, S. Wukitch, *et al.*, paper FIP/P7-18.

1. ITER
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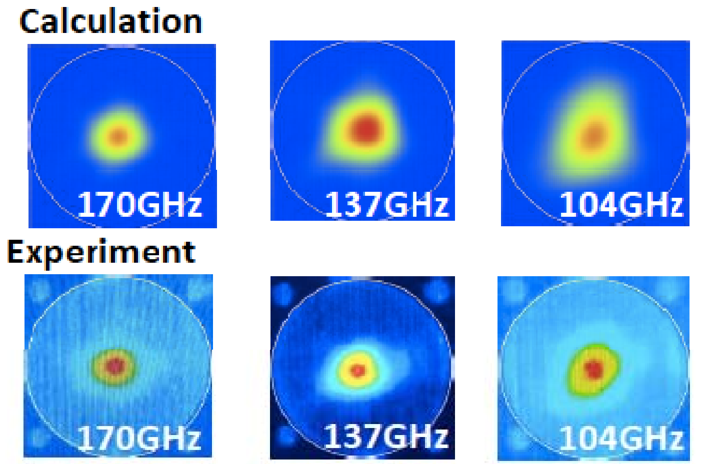
Prototype Development of the ITER EC System with 170 GHz Gyrotron

FIP/2-5Ra Y. Oda

(1) Multi frequency gyrotron:
Output power of 1 MW was demonstrated in 3 frequencies.

(2) Multi frequency RF Power transmission :
High Efficiency power transmission at ITER-Relevant transmission line:
91%/90%/85% at **170GHz/137GHz/104GHz .**



Output beam profiles of multi freq. gyrotron

(3) Gyrotron operation using anode switch :
Power supply system with anode switch for ITER gyrotron was developed. Gyrotron operation with anode switch and **5kHz modulation** were **successfully demonstrated**.

New ideas

PD/1-2 G. Denisov

- **Wavebeam switching inside a gyrotron**. Short pulse experiment with 170 GHz, TE28.12 mode successful (!) **2.5s pulse duration at 1.5MW**

- **Frequency/Phase Locking** of MW gyrotrons in order to provide the same frequency/phase of many gyrotrons, enhance gyrotron efficiency and radiation spectrum. Experiments are going on. Proof-of–principle experiment performed for 35 GHz short–pulse gyrotron.

Success may change essentially EC systems.

IAP RAS

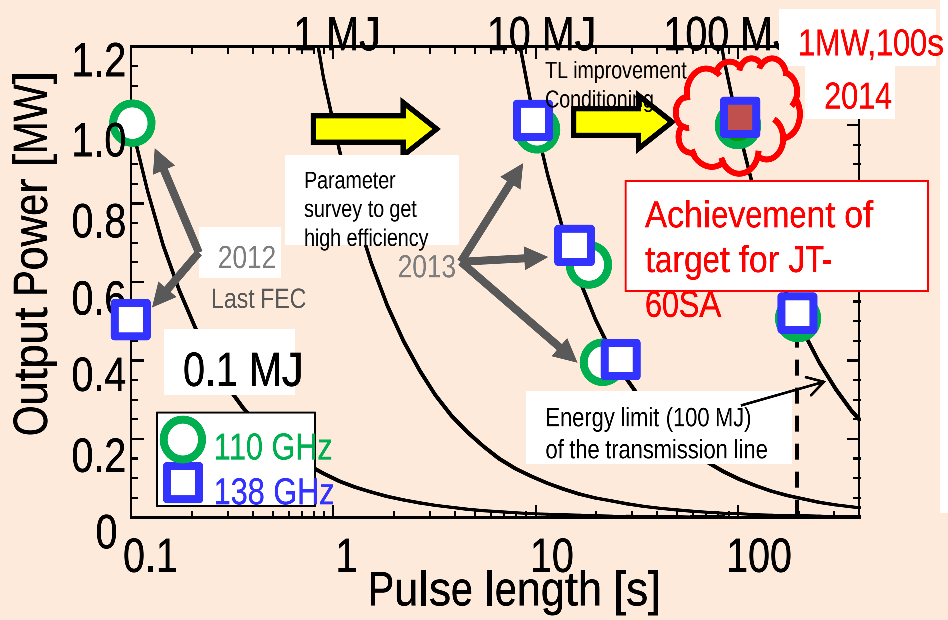
Development of Multi-Frequency Gyrotrons

FIP/2-2Rb, T.Kobayashi

138 GHz : $TE_{27,10}$ 110 GHz : $TE_{22,8}$

Progress in long-pulse operations

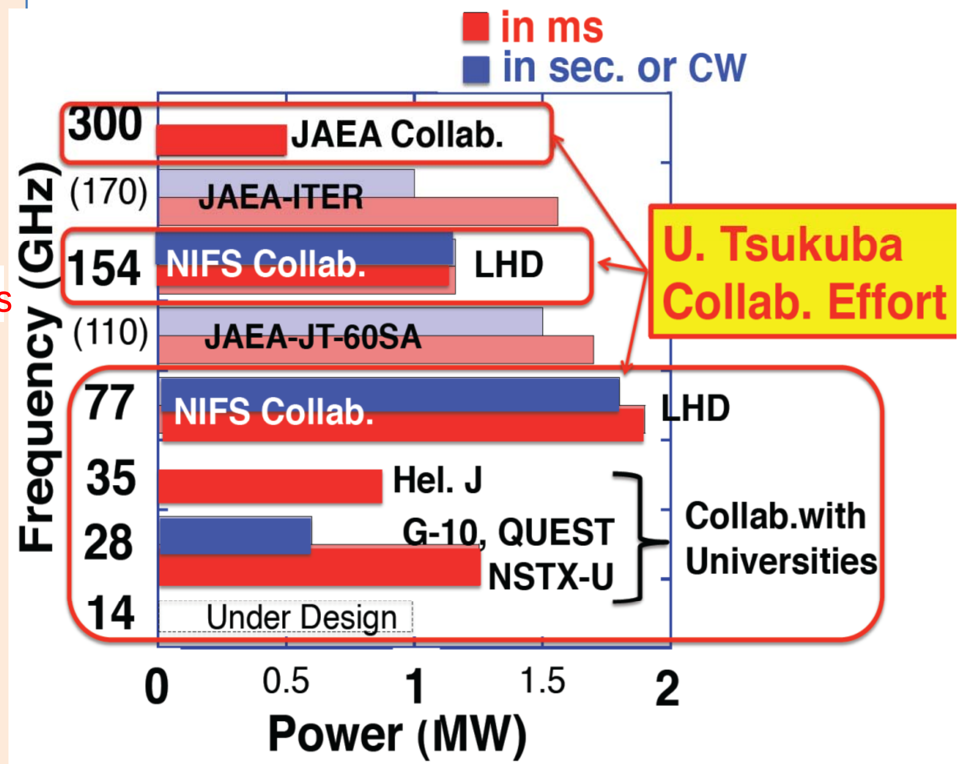
New record as a dual-frequency gyrotron



Successful oscillations of 1 MW for 100 s at both 110 GHz and 138 GHz.

The target for JT-60SA fully satisfied.

FIP/2-2Rc, T.Imai



Univ. of Tsukuba has been developing MW gyrotrons of 14GHz to 300 GHz in collaboration with JAEA, NIFS, TETD, Kyushu & Kyoto Universities, and PPPL.

FIP/2-5Ra; H. Tobari (JAEA), FIP/2-5Rb; A. Kojima (JAEA)

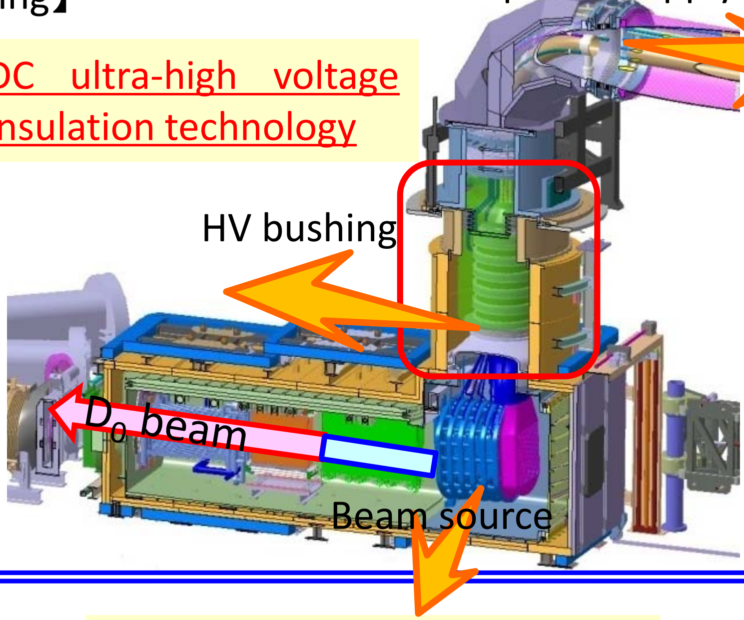
R&Ds in JAEA toward Neutral Beam Injector for ITER and JT-60SA

Key technologies for ITER and JT-60SA have been developed in the past two years.

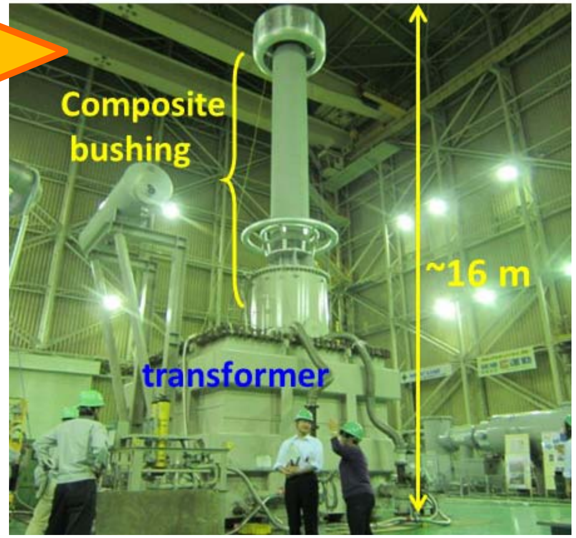
【Mockup test of HV bushing】

DC ultra-high voltage insulation technology

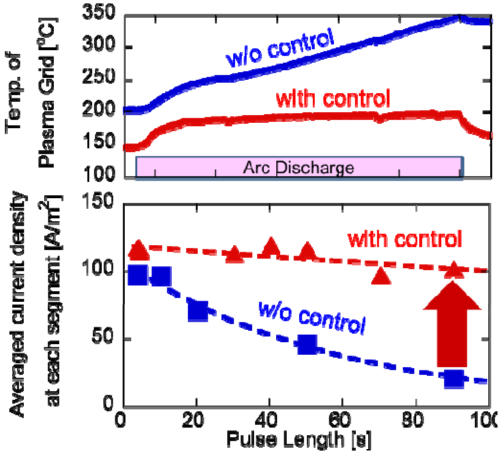
1 MV power supply



【DC 1MV insulating transformer】

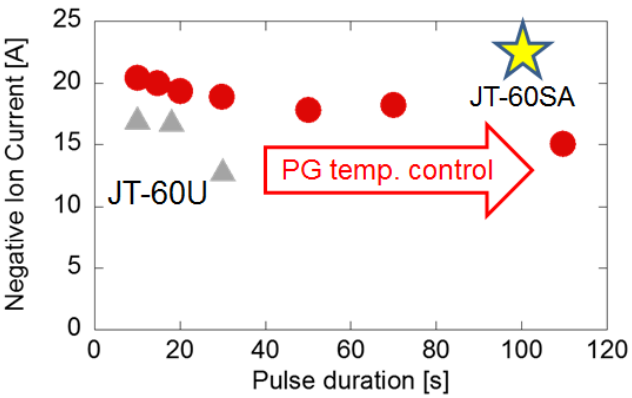


Active control of PG temp.

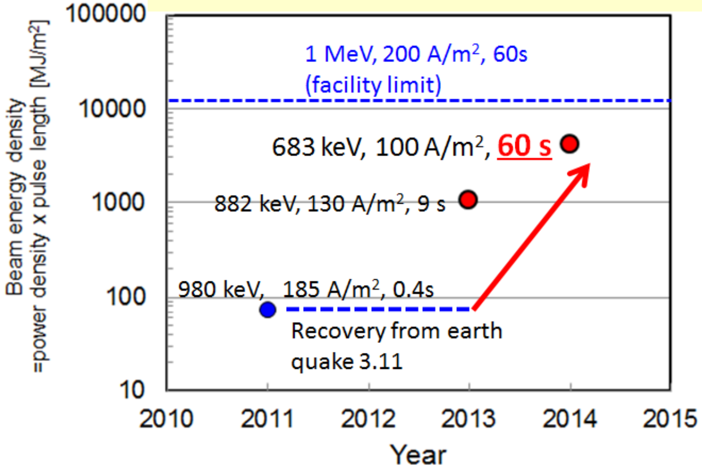


Long pulse beam production

100 s beam production at 15 A

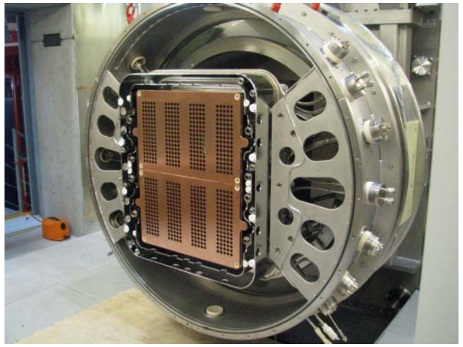


Long pulse beam acceleration



A. Masiello et al., Progress Status of the Activities in EU for the Development of the ITER Neutral Beam Injector and Test Facility

FUSION FOR ENERGY



The PRIMA buildings at Consorzio RFX Padova – Italy.

SPIDER manufacturing is being completed and installation has started.

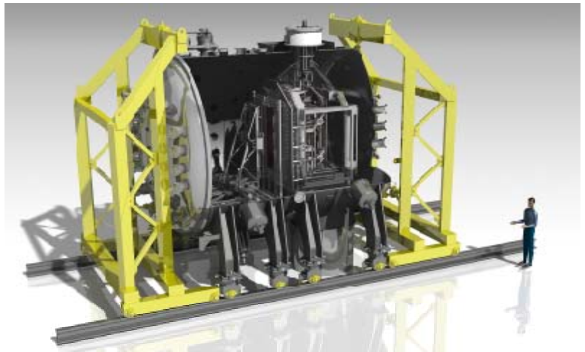
MITICA design is being finalised.

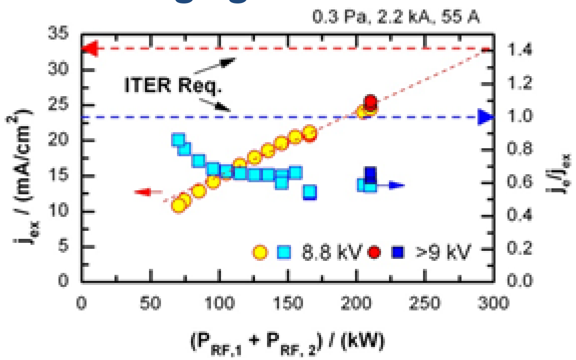
Several R&D and demonstration activities are on-going

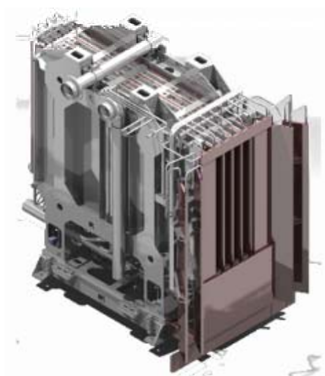
ELISE started in 2012

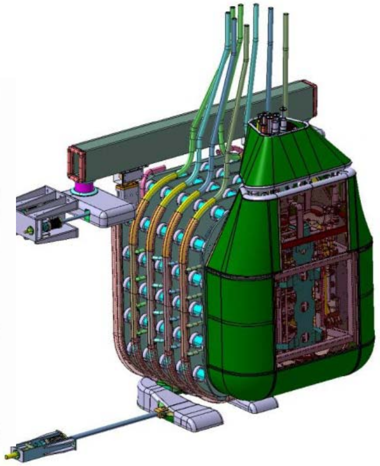
Good results in H: ¾ of the current density achieved

D operations still challenging









Development of the HNB components is well underway!

FIP/2-4

1. ITER
2. DEMO DESIGN
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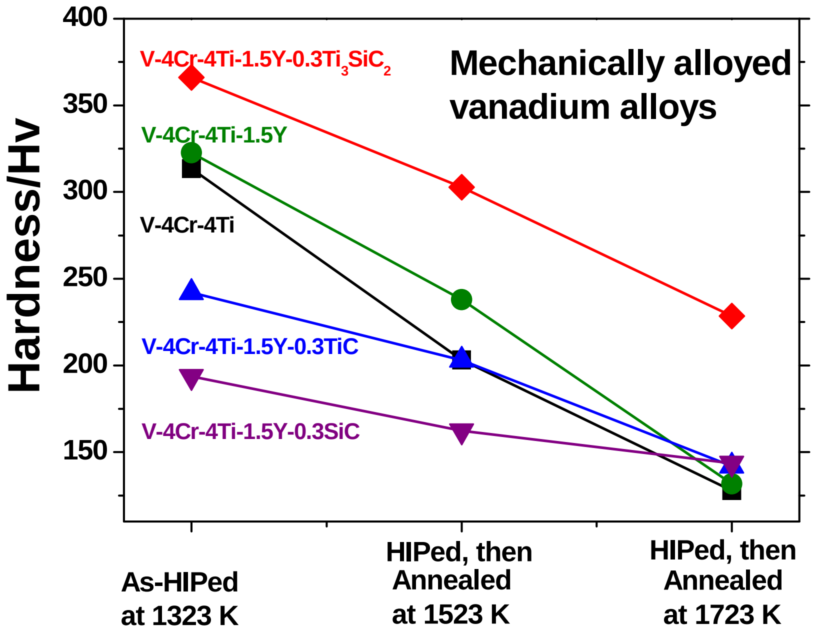
MPT-1. Structural materials

Vanadium alloys:

- **V-Me(Cr, W)-Zr-C** strengthened by **ZrO_2** nanoparticles (**internal oxidation**) (MPT/P7-31-Chernov, JSC "VNIINM", Russia)
- **V-4Cr-4Ti** strengthened by Y, Ti, SiC, **Ti_3SiC_2** nanoparticles (**mechanical alloying**) (MPT/P7-32-Zheng, MPT/1-2-Liu, SWIP, China)



Obtained mechanical properties of the advanced vanadium alloys are significantly higher than those achieved so far for the referenced alloy V-4Ti-4Cr.

Reduced Activation Ferritic/Martensitic (RAFM) steels:

- **9Cr RAFM steel CLF-1** (MPT/P8-7-Wang, MPT/1-2-Liu , SWIP, China)
- **8Cr RAFM steel F82H** (MPT/P7-38-Oyaidzu, JAEA, Japan)

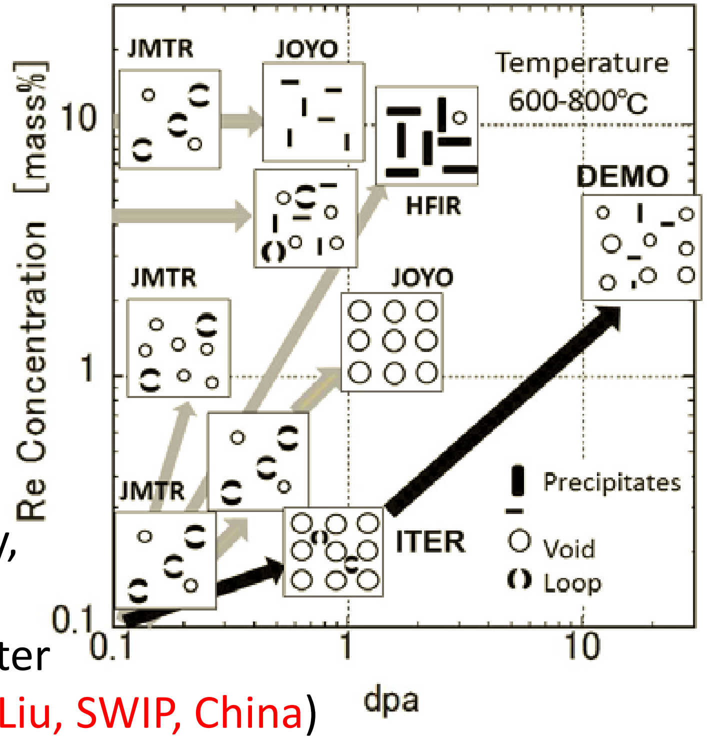
The property data base is being established, including creep tests by more than 11000 h and neutron irradiation data at 0.3-1 dpa.

The effects of **tritium** on passivation of **F82H** steel is investigated, exotic corrosion of metal is predicted

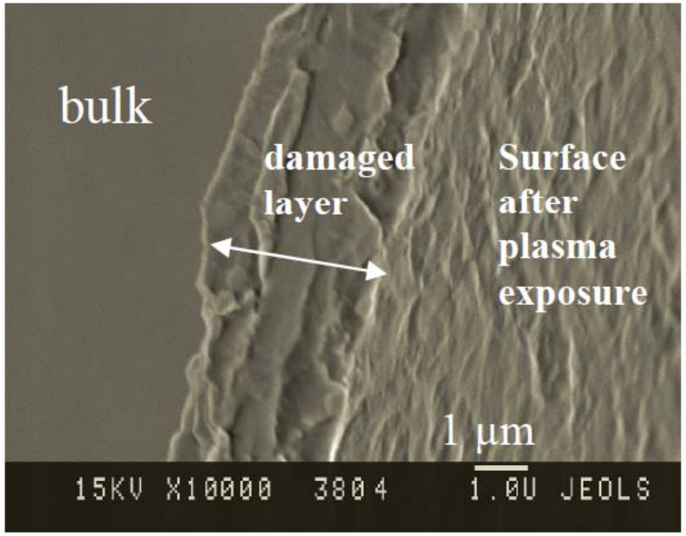
MPT-2. Plasma-facing materials

W and its alloys

Microstructural data of neutron irradiated W (up to 1.5dpa) was compiled. Qualitative prediction of the damage structure development and hardening of W in fusion reactor environments was made taking into account the solid transmutation effects for the first time. (MPT/1-4-Hasegawa, Tohoku Univ., Japan)



Several kinds of tungsten based materials are developed, such as oxides and carbides dispersion strengthened W alloy, and a fast CVD-W coating. They shows higher cracking thresholds at transient heat loading. CVD-W indicates a better crack suppression effect at elevated temperature.(MPT/1-2-Liu, SWIP, China)

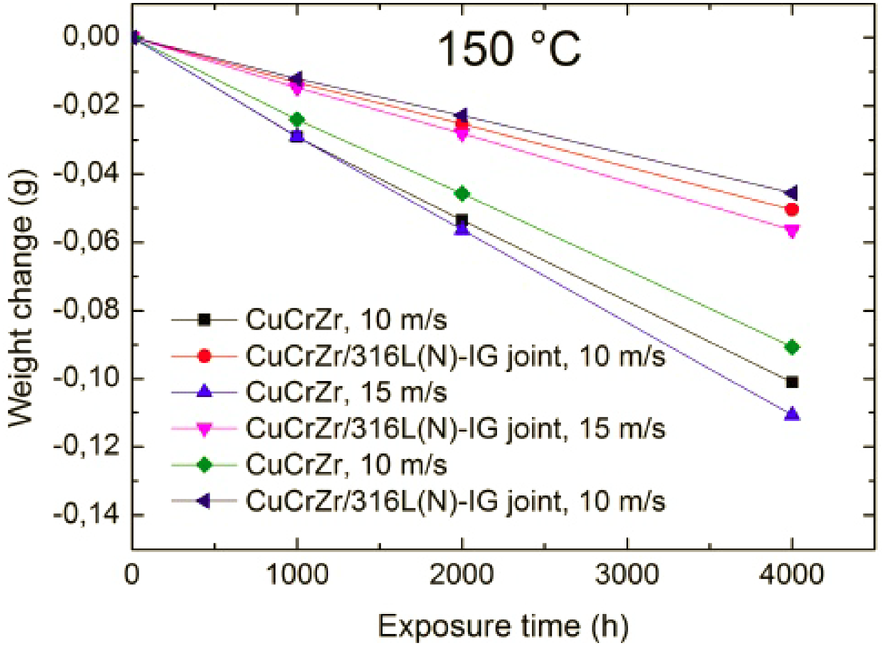


High-energy ion beams (C-ions, 10 MeV) have been applied to produce high-level displacement damage (up to 100 dpa) in W and the irradiated material has been studied under plasma impact. While no correlation of erosion yield could be attributed to damage influence, clear effect of the damage on deuterium retention in plasma exposed tungsten was demonstrated. (MPT/P7-37-Koidan, NRC KI, Russia)

MPT-3. Functional materials

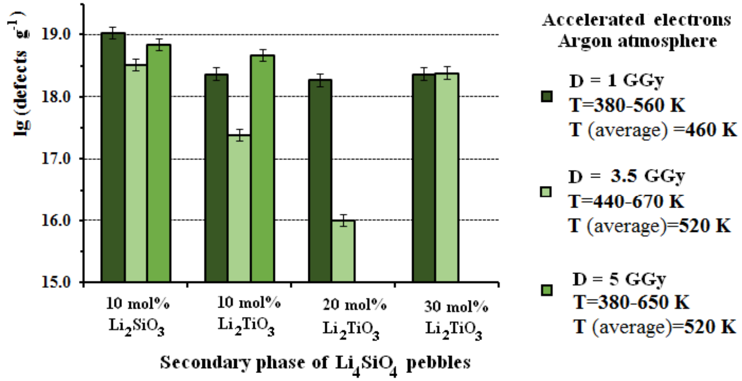
Heat sink materials: CuCrZr

Erosion corrosion rates under simulated conditions relevant for the ITER coolant system are disturbingly high (25 µm/year at 110 °C, 37 µm/year at 150 °C, 1600 µm/year at 250 °C). Erosion corrosion of CuCrZr can thus potentially cause serious problems for the ITER coolant systems. (MPT/P4-23-Wikman, F4E, Spain)



Tritium breeder materials: Li_2SO_4

The modified Li_4SiO_4 pebbles with 10-30 mol% Li_2TiO_3 have slightly higher radiation stability in comparison to the reference Li_4SiO_4 pebbles with 10 mol% Li_2SiO_3.
The modified pebbles have the potential to combine the advantages of Li_4SiO_4 and Li_2TiO_3 as a tritium breeding ceramic for the HCPB TBM.
(MPT/P8-5-Zarins, Univ. of Latvia)



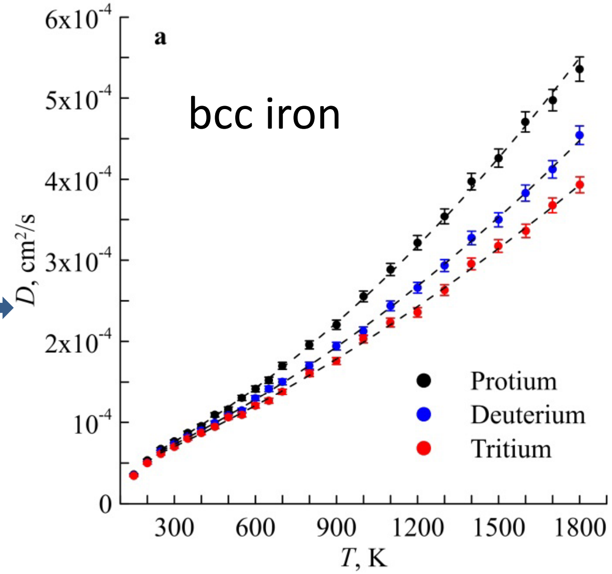
Total concentration of paramagnetic radiation-induced defects and radiolysis products in the different samples after irradiation.

MPT-4. Multiscale modelling

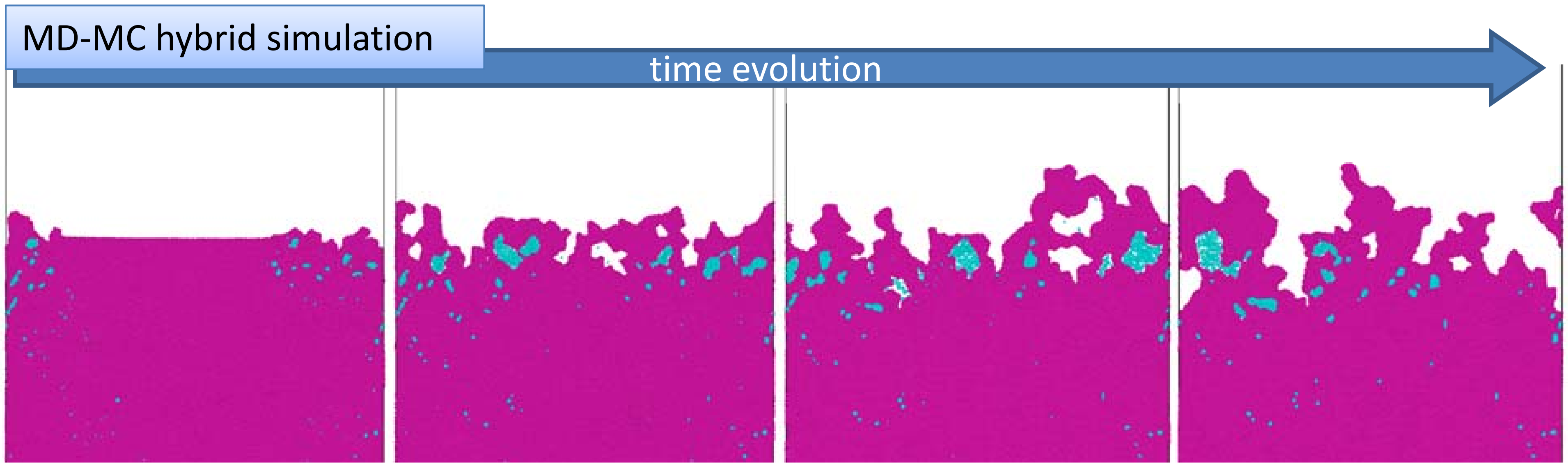
H and He effects in Fe, V, W:

MD and DFT calculations of energetics of self-defects, impurities, their complexes in metals:

- H isotopes in Fe (MPT/P7-33-Sivak, NRCKI, Russia),
- H in W (MPT/P7-36-Kato, NIFS, Japan),
- He in W (MPT/1-3-Ito, NIFS, Japan).



◆ Molecular dynamics and Monte-Carlo (MD-MC) hybrid simulation achieved to represent the formation process of the fuzzy nanostructure by helium plasma irradiation.



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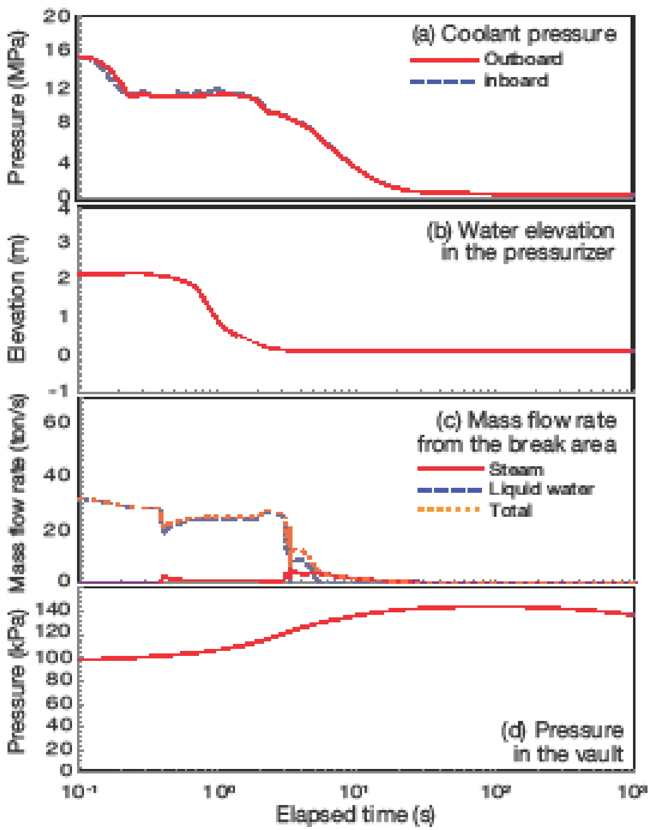
SEE/P5-10 M. Nakamura

Summary slide of "Analysis of Accident Scenarios of a Water-Cooled Tokamak DEMO"

Large in-vessel and ex-vessel loss-of-coolant accidents of a water-cooled tokamak DEMO have been analyzed.



Ex-VV LOCA analysis

- ❖ We have identified the event sequences following an ex-VV large (double-ended) pipe break of the primary cooling system.
- ❖ The load onto the confinement area covering the broken primary cooling loop was found to be so large that it is difficult make a large volume, such as the tokamak building, pressure-tight.
- ❖ The analysis result suggests that measures to protect the confinement area will be needed.
  - ✓ A possible way is to implement a small vault of pressure-tightness or with a pressure suppression system, covering the primary cooling pipes.

Review of the Safety Concept for Fusion Reactor Concepts and Transferability of the Nuclear Fission Regulation to Safety Concept for Fusion Reactor Concepts

- **Achievement**

  - A thorough **literature survey** of the **fusion safety concept** was carried out and it was exemplarily checked against German safety requirements for nuclear power plants.

- **Current status**
  - The **fusion safety concept** is based on the concept of **defence in depth**, which is **necessary to guarantee** the confinement of the radioactive inventory.
  - In principle, the (German) **safety requirements** for **NPPs** can be applied to **FPPs**. However, there are specific differences between the implementations of the safety concept of FPPs and NPPs. In principle, the **fundamental safety functions are applicable.**
- **Next steps**
- Together with an increased level of detail of the plant designs of future FPPs
  - a **systematic assignment** of measures and installations to the different **levels of defence**
  - **potential releases**
  - **external events** (e. g. earthquakes and flooding) and very rare **man-made external hazards** (crash of a large air plane)
- have to be analysed in more detail.

FEC-25 Fusion Technology Conclusions

- ITER project develops sustainably and remains the leader of Burning Plasma Physics & MFT
- Enabling technologies become closer to ITER technical requirements demonstrating full scale prototypes, parts and construction site progress
- New concepts of DEMO and FNS facilities explored under IAEA auspices are effective drivers of steady state technologies and FNS
- Materials and neutron test facilities, compatibility with neutron environment, maintainability and equipment lifetime are currently the major challenges and concerns on the path to DEMO & FNSF
- Russia strongly participates in the ITER project additionally developing a new strategy with tighter interlinks of Fusion and Fission to accelerate the implementation of fusion technologies to mutual benefit of the two branches of nuclear power