Fusion Research in Europe

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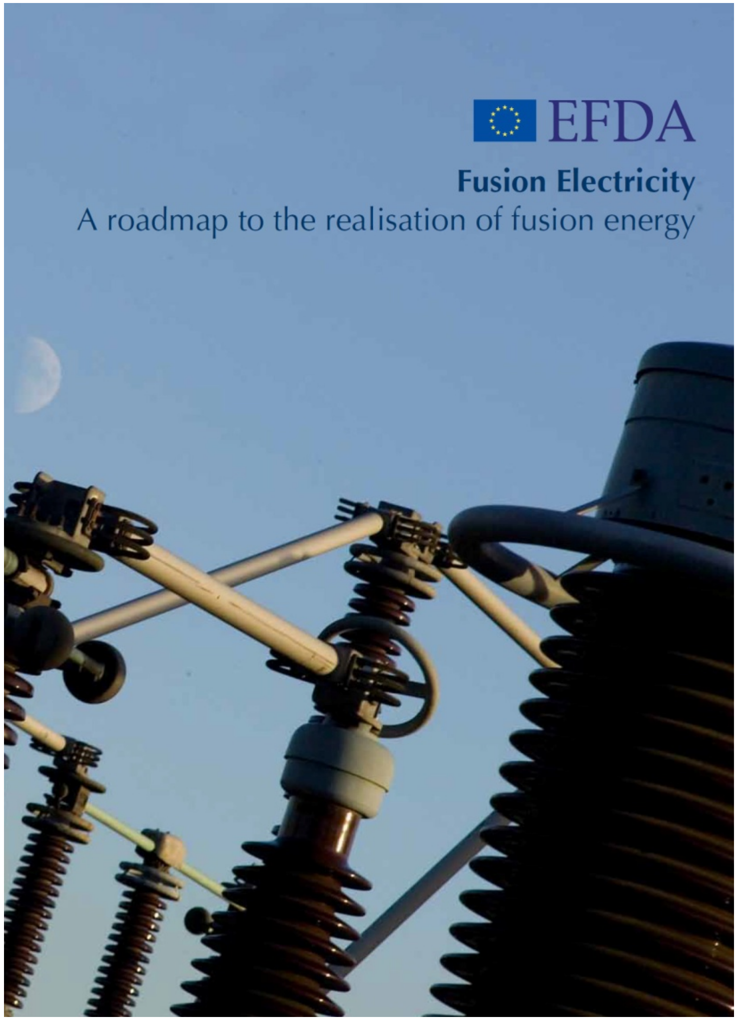
The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Launch event – 9 October 2014

EUROfusion

29 Research Units (+ numerous Third Parties) in 27 European countries working together to achieve the ultimate goal of the Fusion Roadmap

Fusion Roadmap



Eight important missions

- For each mission:
  - overview present status
  - list of unresolved and urgent issues
  - research & development plan
  - estimation of required resources

Three periods

- 2014 – 2020
  (Building ITER & Supporting Experiments)
- 2021 – 2030
  (Exploiting ITER and Designing DEMO
- 2031 – 2050
  (Building and Exploiting DEMO)

Important to intensify the involvement of industry

A roadmap to the realisation of fusion energy

8 Strategic Missions tackle all challenges in two main areas:

ITER Physics

- Risk mitigation for ITER
- JET, Medium Size Tokamaks, Plasma Facing Component devices

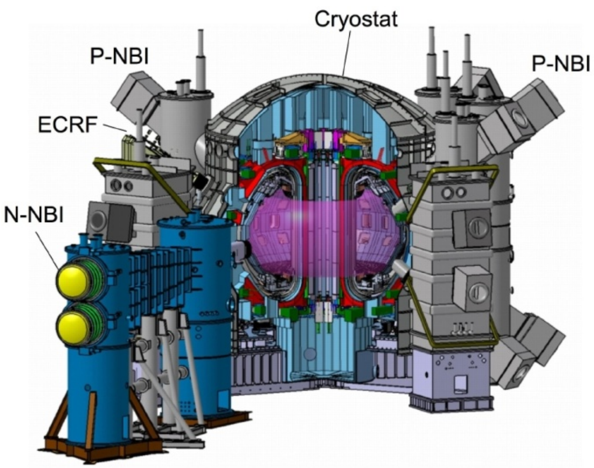
DEMO

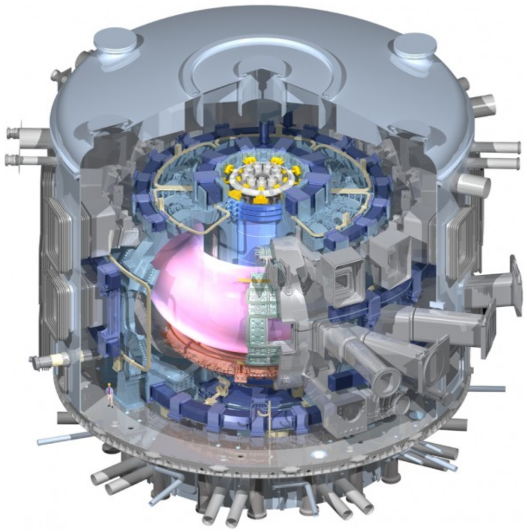
- Conceptual design studies
- A single step to commercial fusion power plants
- Production of electricity with a closed fuel cycle

Back-up strategy

- Stellarator

1: Plasma Regimes of Operation





Main devices:
JET, ASDEX, JT-60SA, ITER

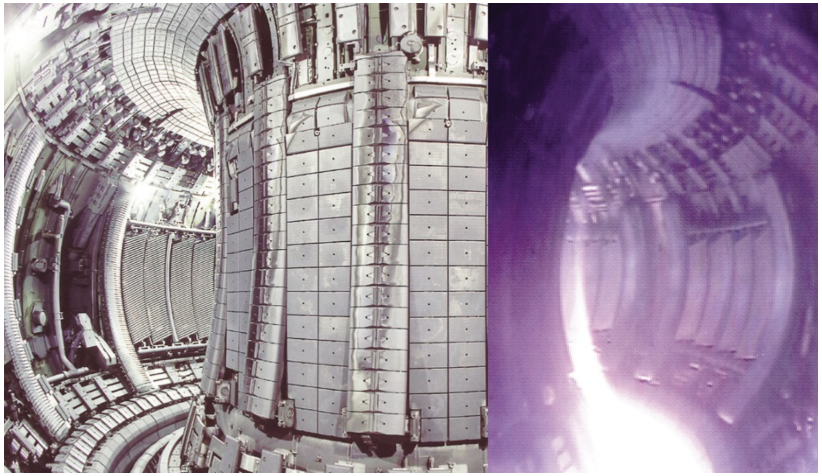
High fusion performance by reducing energy losses by turbulence and by controlling plasma instabilities.

To achieve acceptable power depositions in the divertor, radiate as much as possible power from the plasma without having adverse effects on the performance

Develop active methods the state of divertor detachment

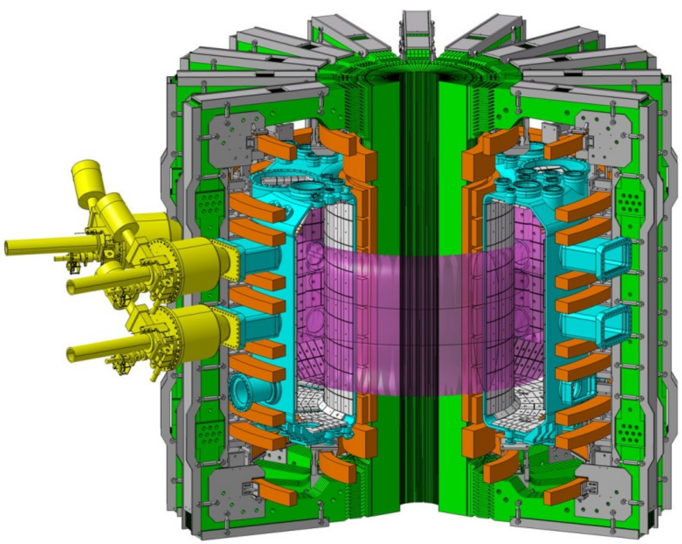
Try to achieve steady state conditions

JET and Medium-Size Tokamaks

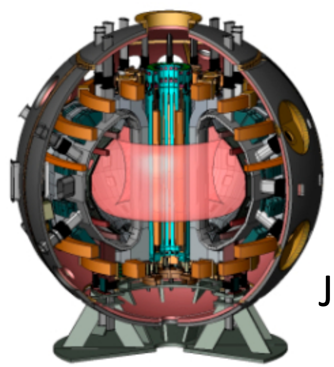


JET

ASDEX-UG



TCV

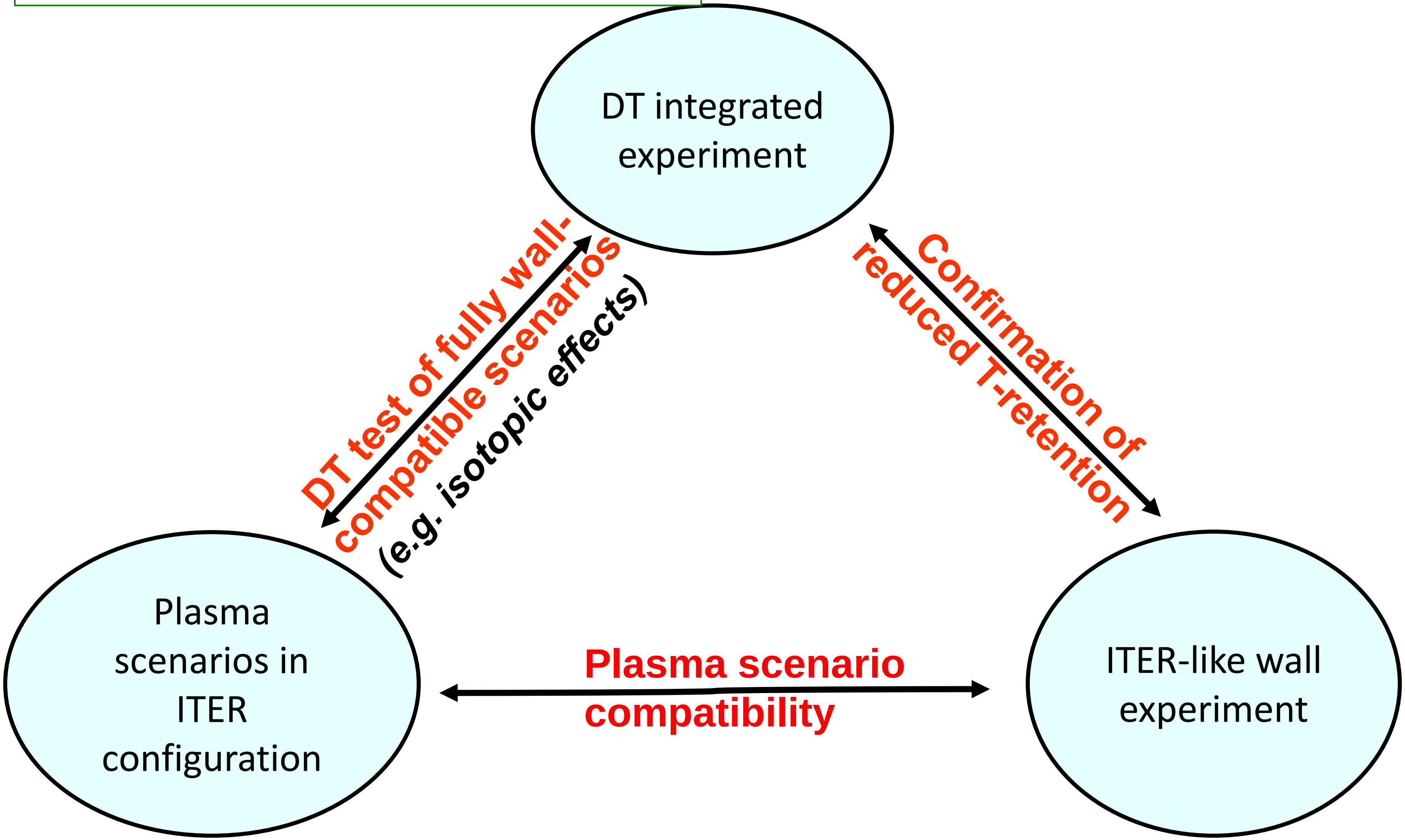


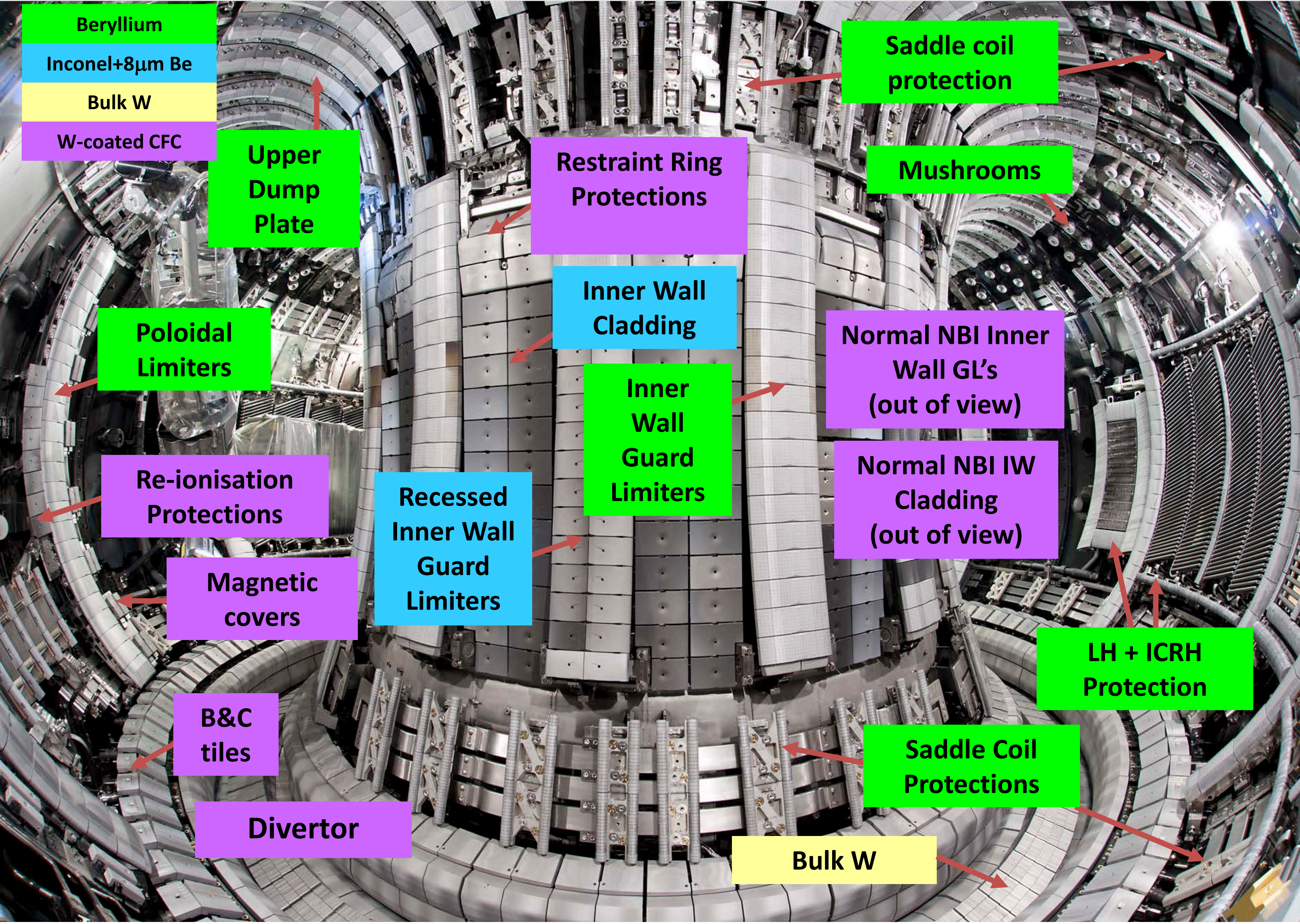
JT-60SA

MAST-UG

The JET Programme in Support of ITER

Paméla et al., Fusion Eng. Des. 82, 590 (2007)



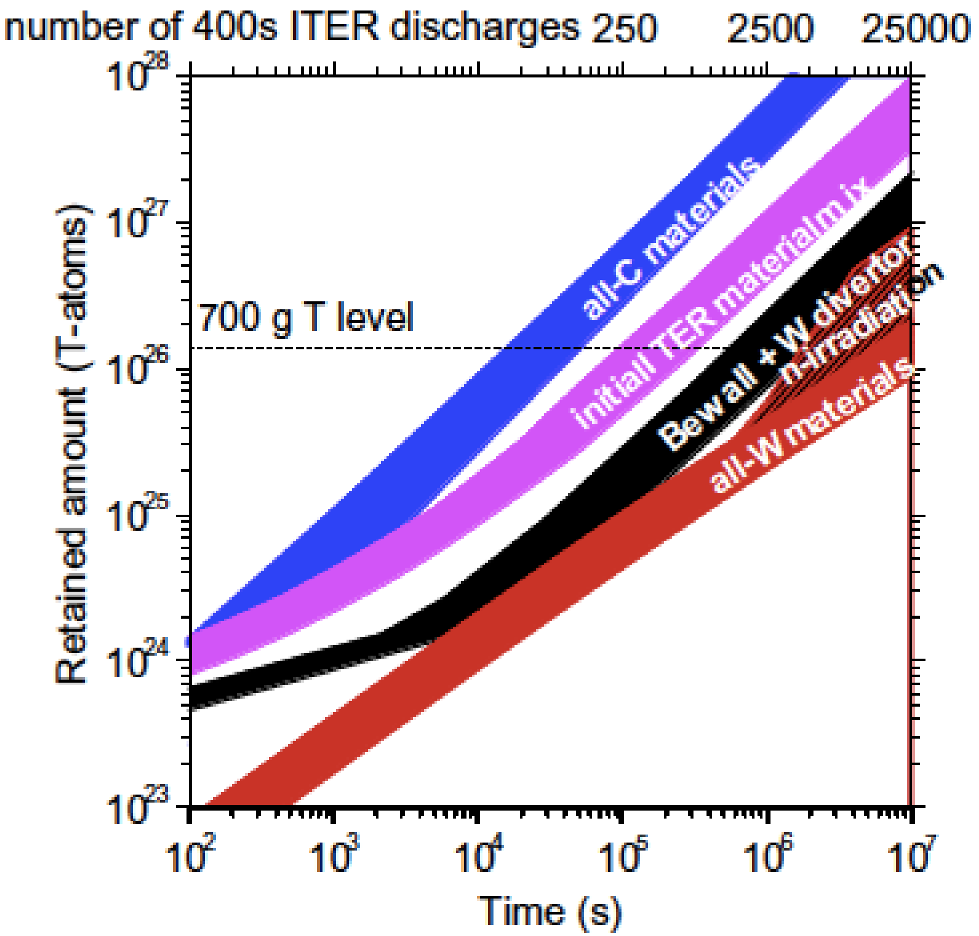


ILW = 2880 installable items, 15828 tiles (~2 tonnes Be, ~2 tonnes W)

1.1: Fuel retention with ITER like wall

Roth et al., J. Nucl. Mat. 390-391, 1 (2009)

In ITER, retention of tritium fuel due to co-deposition with carbon is expected to be unacceptably high.

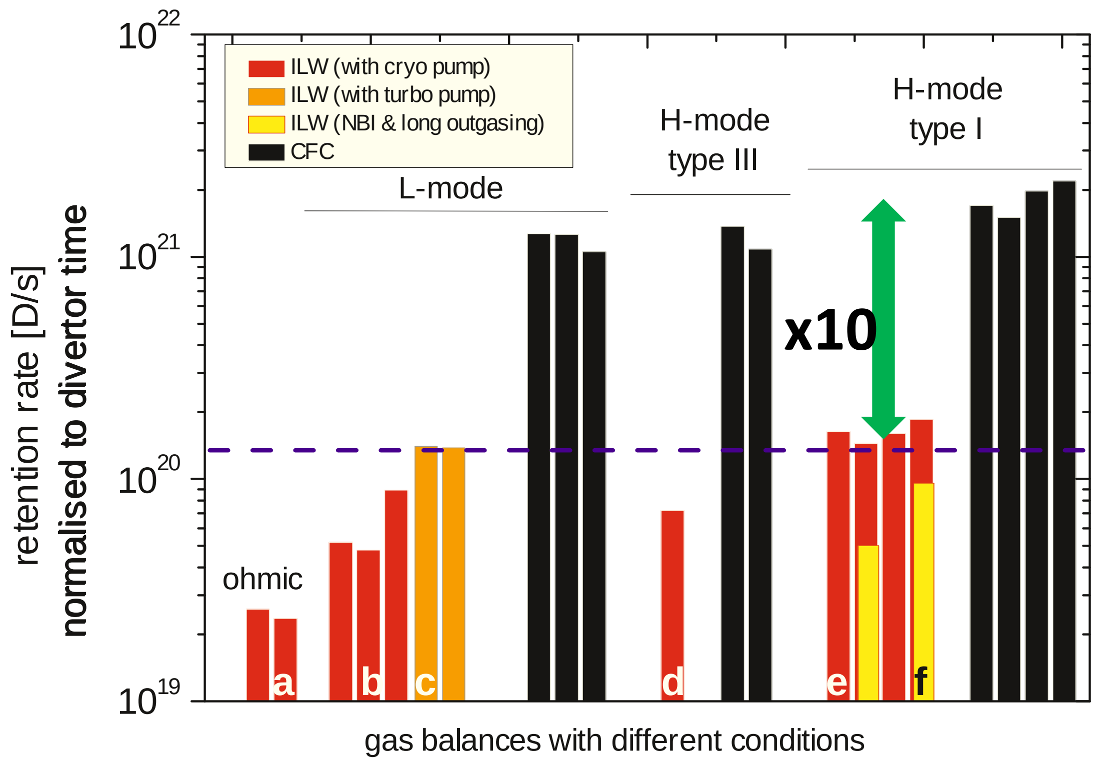


1.1: Fuel retention with ITER like wall

Loarer et al., J. Nucl. Mater. 438, S108 (2013)

Measured fuel retention is more than an order of magnitude lower with the ILW, consistent with predictions made before the wall was installed.

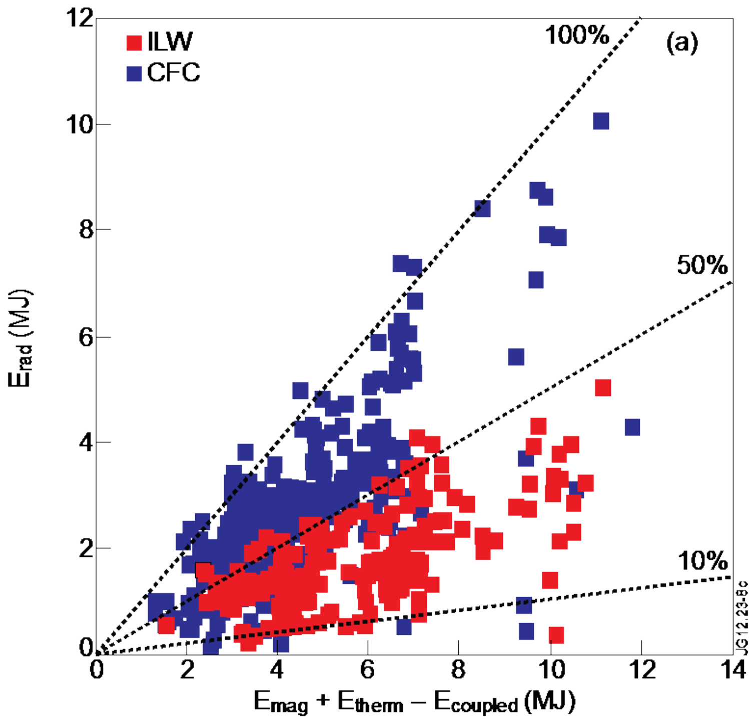
Residual retention consistent with co-deposition in Be layers.

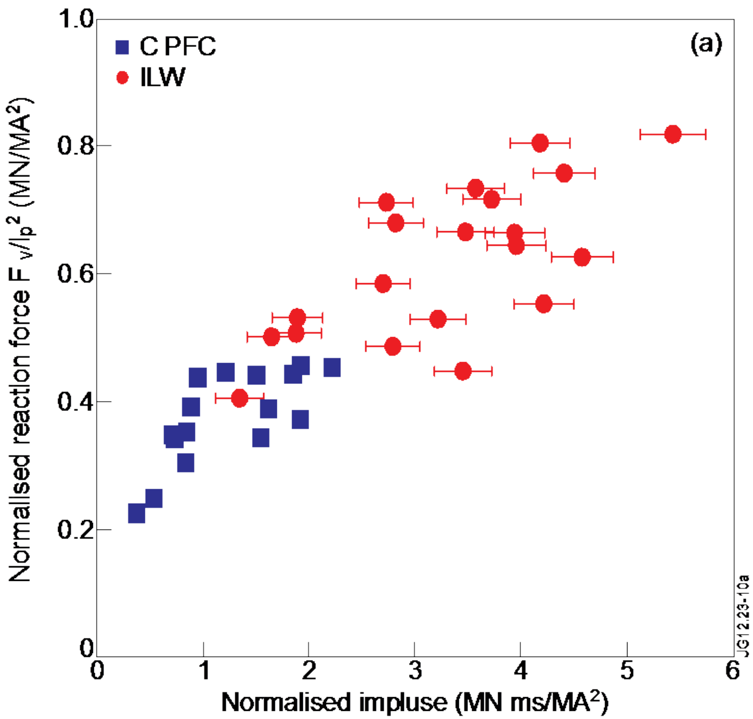


1.2: Disruption mitigation

Lehnen et al., J. Nucl. Mater. 438, S102 (2013)

- The dynamics of disruptions are very different with the ILW
  - ➢ Higher plasma purity → lower radiation during disruption
  - → slower current quench
  - → higher heat loads and halo currents
  - → higher reaction forces on the vessel

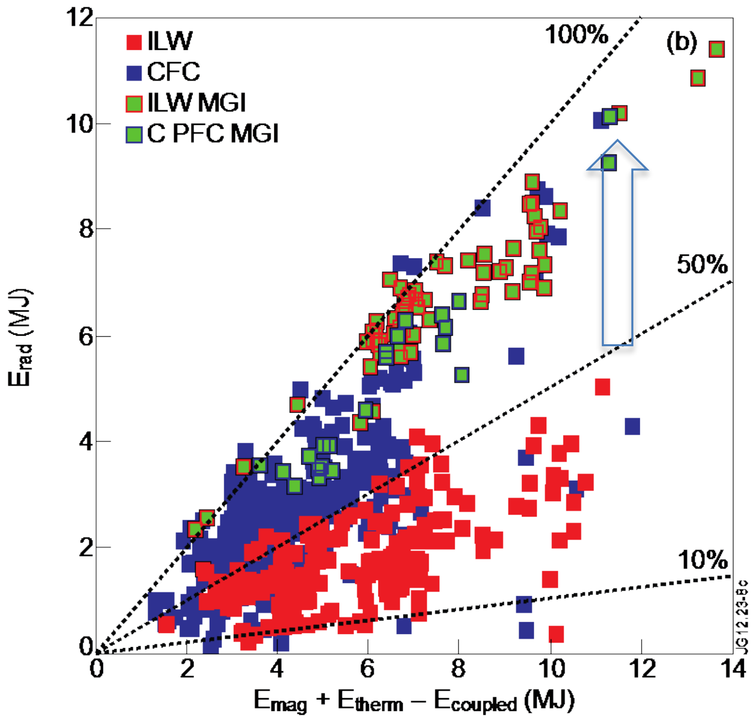


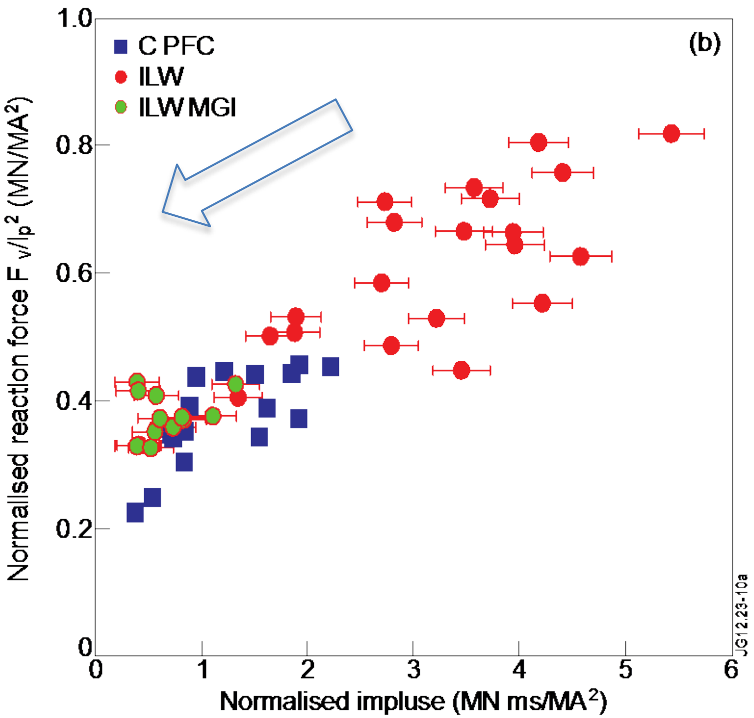


1.2: Disruption mitigation

Lehnen et al., J. Nucl. Mater. 438, S102 (2013)

- Massive gas injection as a disruption mitigation tool is now mandatory for JET experiments at or above 2.5 MA (as it will be for ITER)
- With the mitigation, the forces and power loads resulting from disruptions are returned to the level observed with C wall

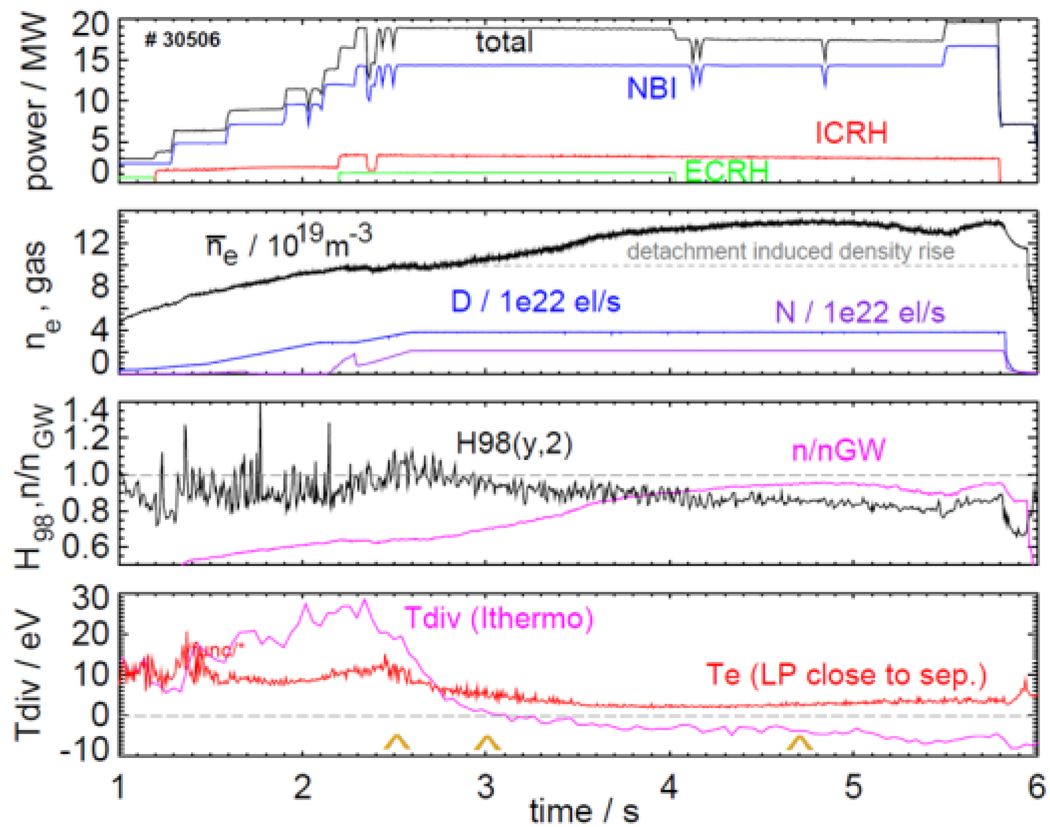




2014: aim at highest P_{sep}/R with N seeding

- I_p=1.2MA, P_{tot}=20 MW
- High neutral pressure (1/3 cryo)
- Partial detachment achieved from 3 s
- P_{rad}/P_{tot}~60%
- P_{sep}/R ~ 9MW/m

(2/3 of ITER range 15 MW/m)



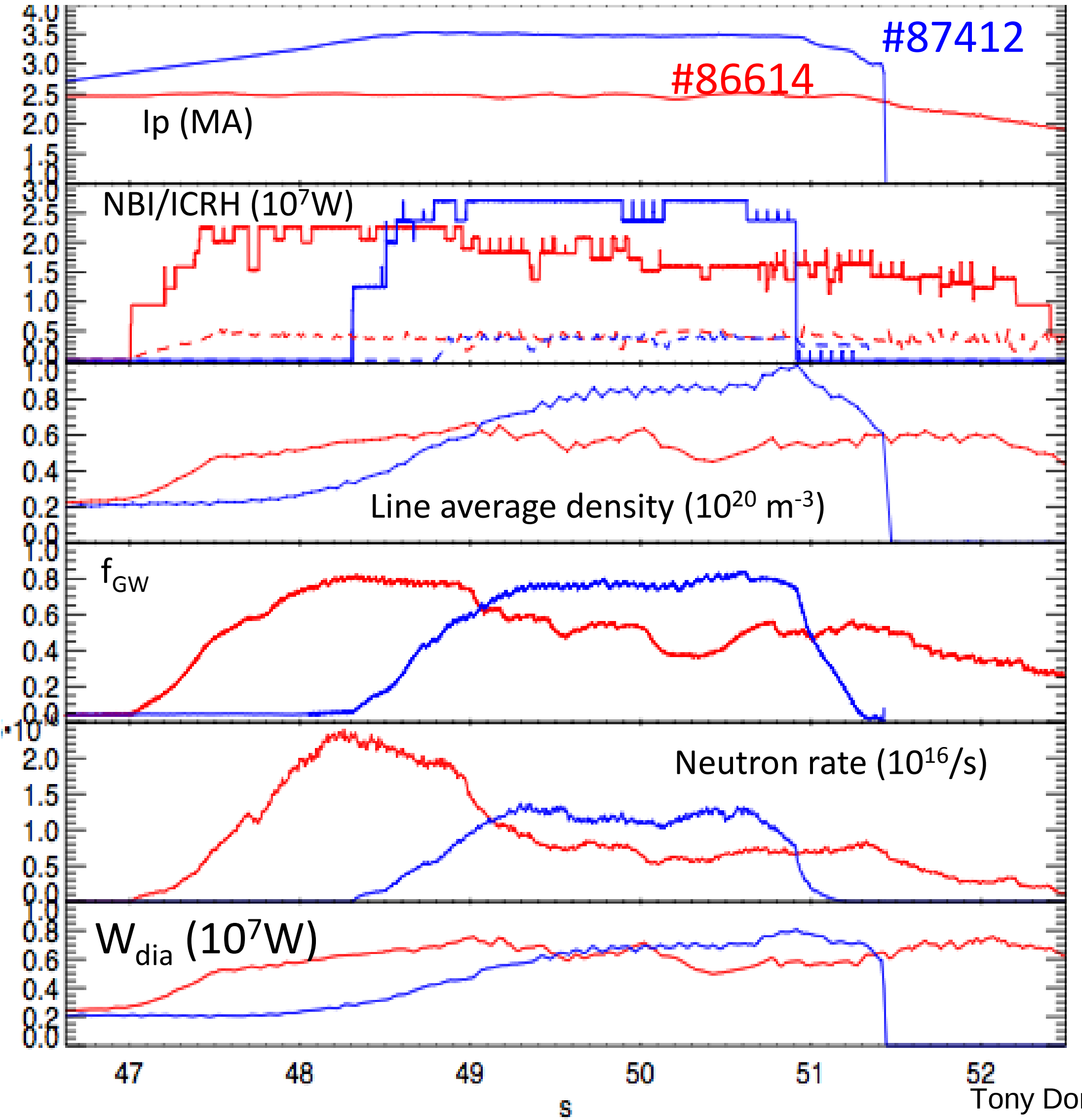
Aim 2015:

increase P_{in} to 25 MW to P_{sep}/R ~ 12MW/m

(in 2014 domestic programme following MST1 campaign P_{sep}/R~10 achieved)

1.3 JET ELMY and HYBRID scenario preparation for DT

Target : W_{dia} >12MJ, P_{fusion}~15MW for 5s



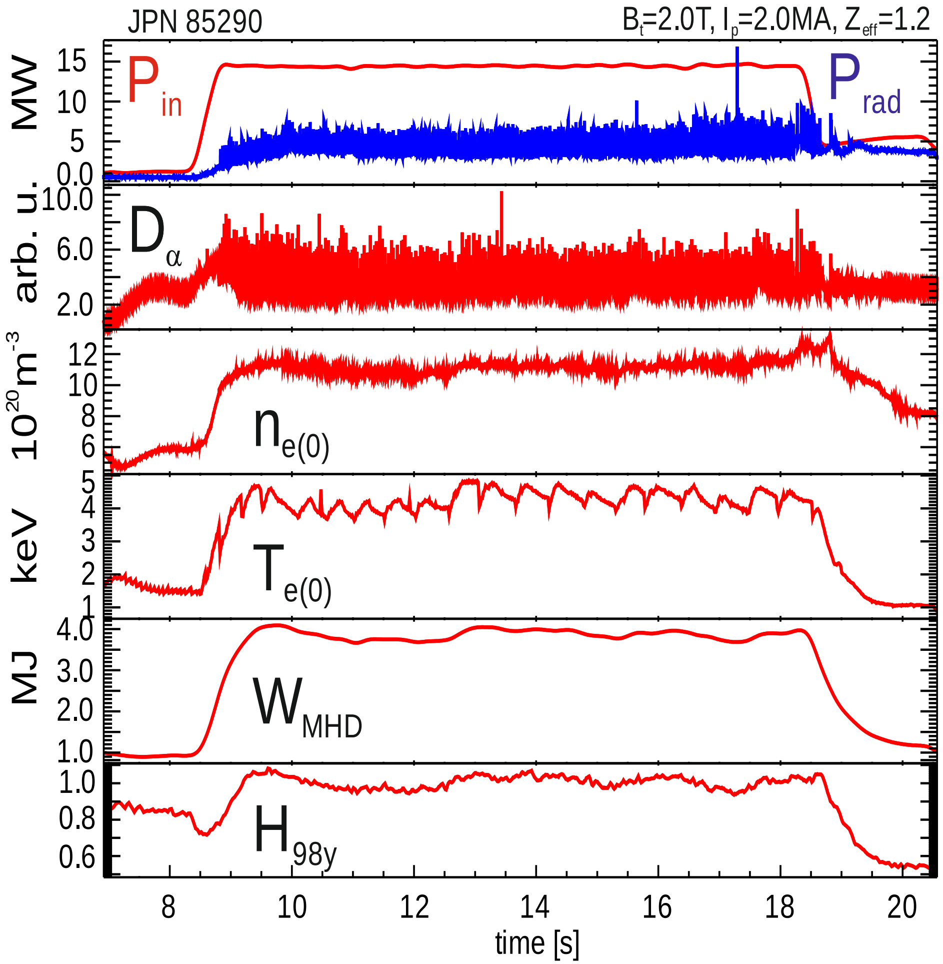
- hybrid: 2.5MA/2.9T (q_{95}=3.7) Transient good confinement phase hampered by MHD and divertor temperature limit
- baseline: 3.5MA/3.3T (q95=3) Stationary plasmas hampered by temperature limit on divertor

Develop methods for divertor exhaust control:

impurity seeding (Ne) & strike point sweeping

1.3 JET : DEVELOPMENT of STATIONARY H-mode Operation in ILW at H~1

Operation with strike-points close to pump duct entrance



Operational window narrower with JET-ILW with Limited access to low gas fuelling (best performance in JET-C)

- Stationary type I ELMy H-modes in low and high triangular plasmas requires
  - Gas fuelling (to reduce W source)
  - Minimum ELM frequency (to flush W)
  - Central heating (to avoid W peaking)
  - Heat exhaust control

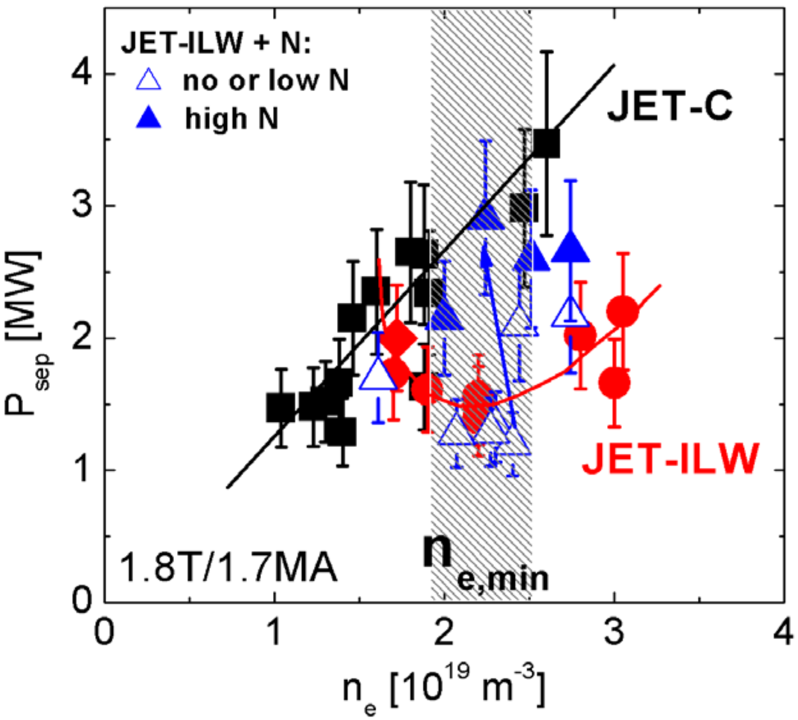
otherwise W-accumulation might occur

W concentration below C_W~5x10^{-5}

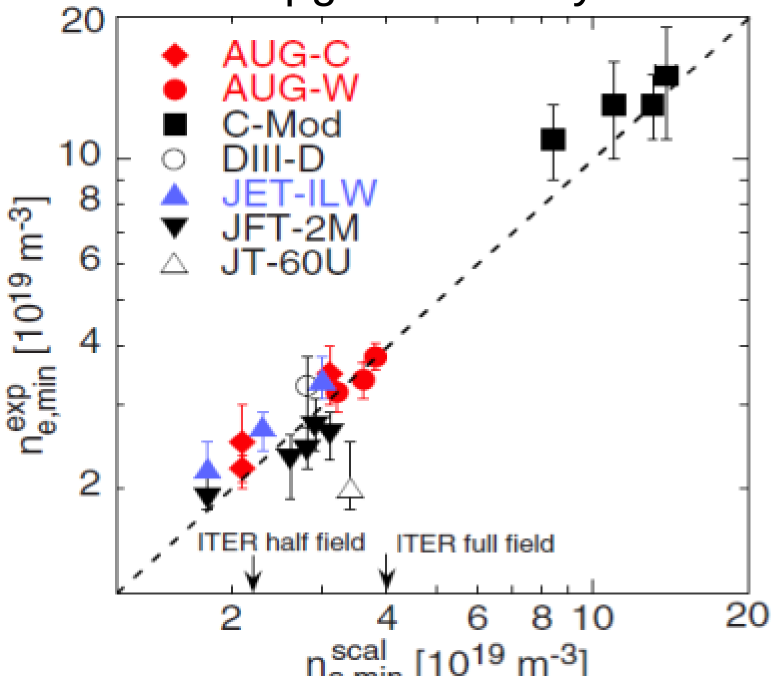
1.3 H-Mode access

- R&D on key players for H-mode transition ?
  - H mode access at reduced power and resolve deviations from scaling (low n_e), turbulence measurement
- Understanding to determine optimum H-mode access/exit strategy for ITER
  - Minimum density in He H-modes ? , role of Z_{eff} ? ion heating ?

JET - C. Maggi – Nuc Fus/EPS 2014

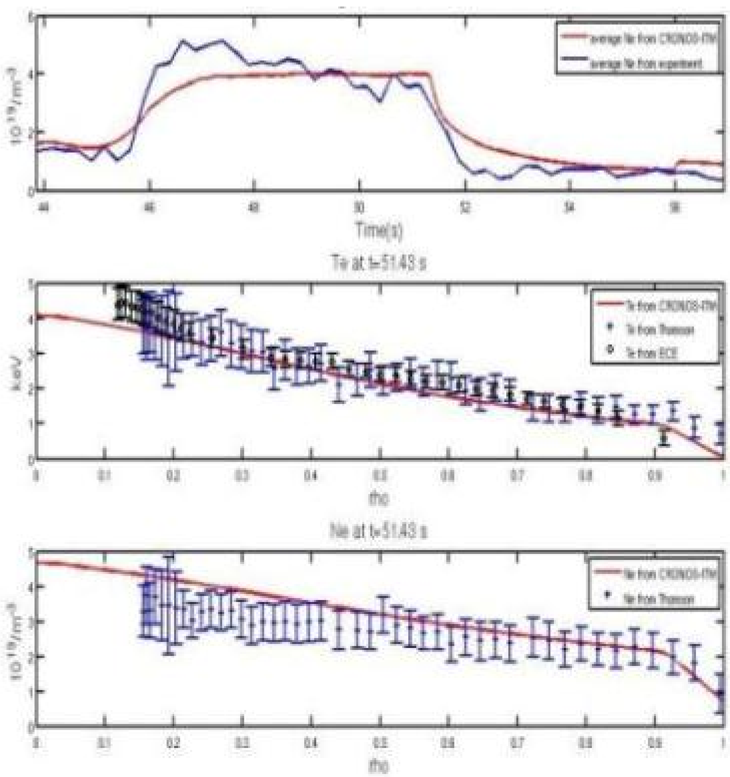


ASDEX-Upgrade – F. Ryter NF - 2014



1.4: Modelling support

- WCD/JET/MST: Release of ETS core simulator (fixed and free boundary) with full hierarchy of transport models and full integration of H&CD, including synergies (NB, IC and alpha); deployment on JET & exploitation in JET/MST.
- JT-60 SA : Core-equilibrium modelling of flat-top phases of JT-60SA scenarios.



- **ETS Simulation covering the full JET plasma discharge duration (13s).**
- **Good agreement between experimental data and simulation**

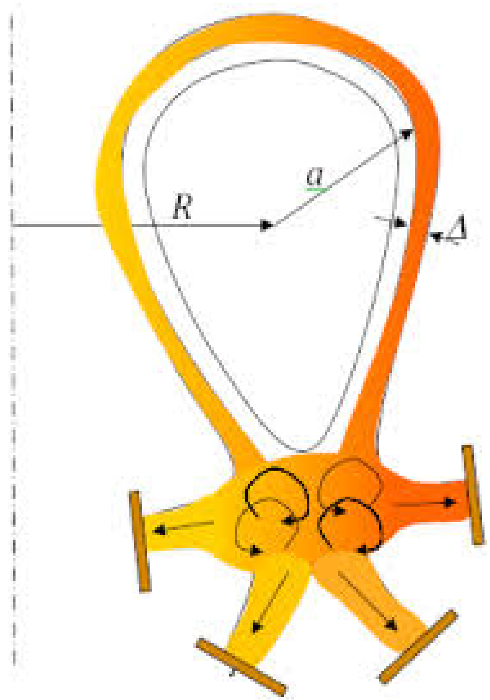
2: Heat Exhaust Systems

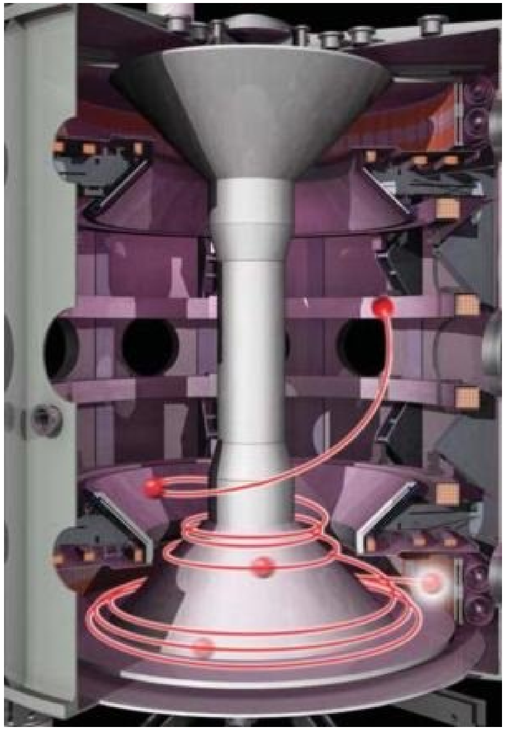
Research in alternative divertor solutions (Super-X, snowflake, liquid metal divertors)

Research in order to understand detached divertor conditions

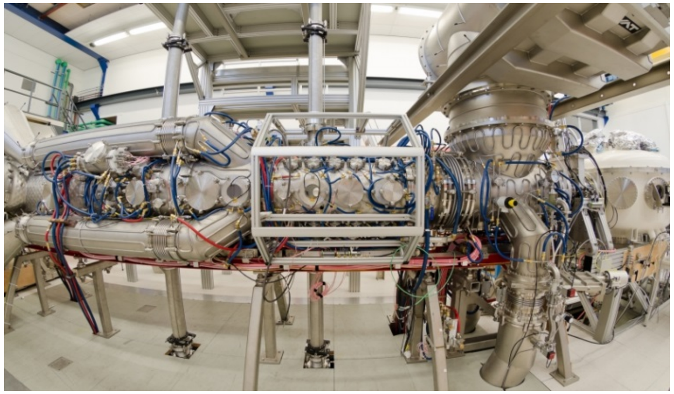
Research to find more robust materials

Main devices:
MAST, TCV,
Linear devices
Potentially a
Divertor Test Tokamak

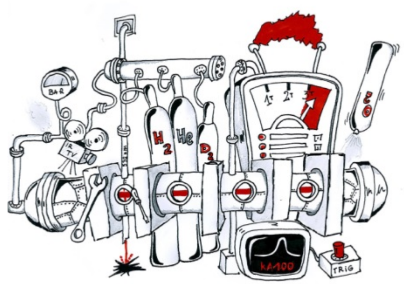




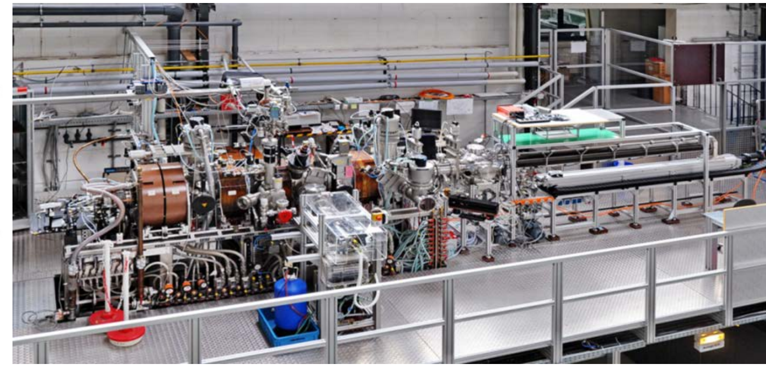
2: PFC Devices



Magnum-PSI



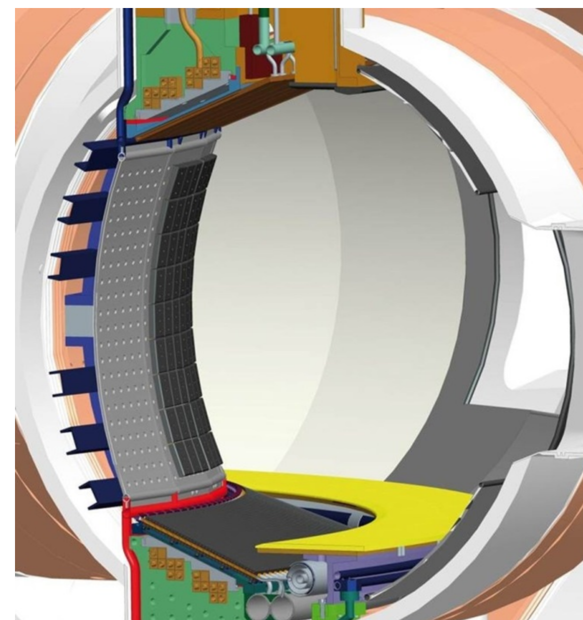
Pilot-PSI



PSI-2

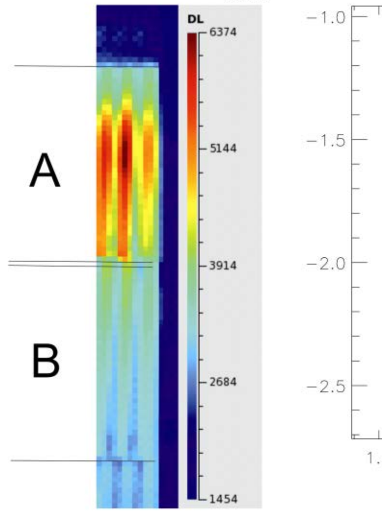
JUDITH-1/2

WEST

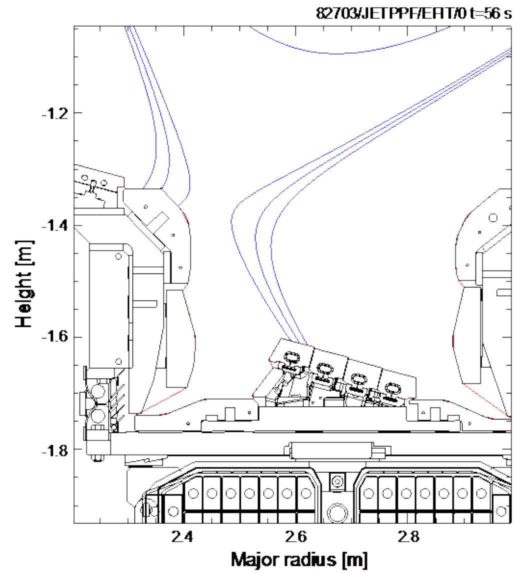


2.1: Tungsten melt experiments

82703, t =56.178s, T_{max} = 236°C

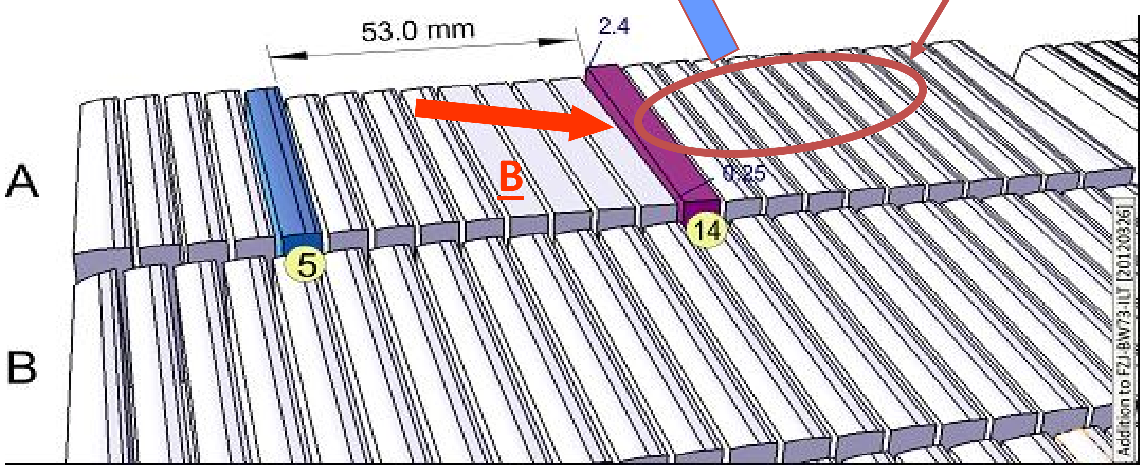


– KL9B –
[NBI ~10MW, 2s]



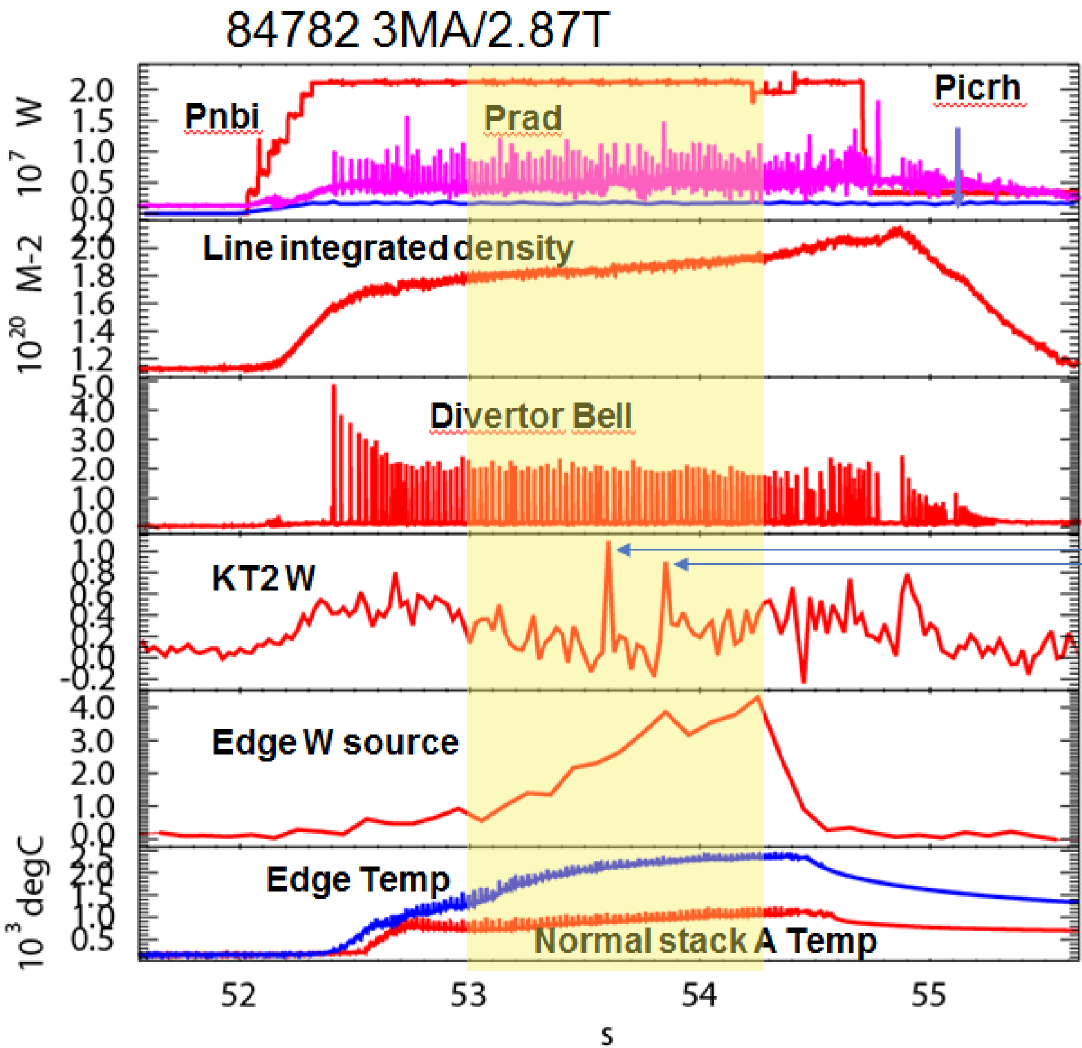
Melt motion?

Large potential melt volume



2.1: Tungsten melt experiments

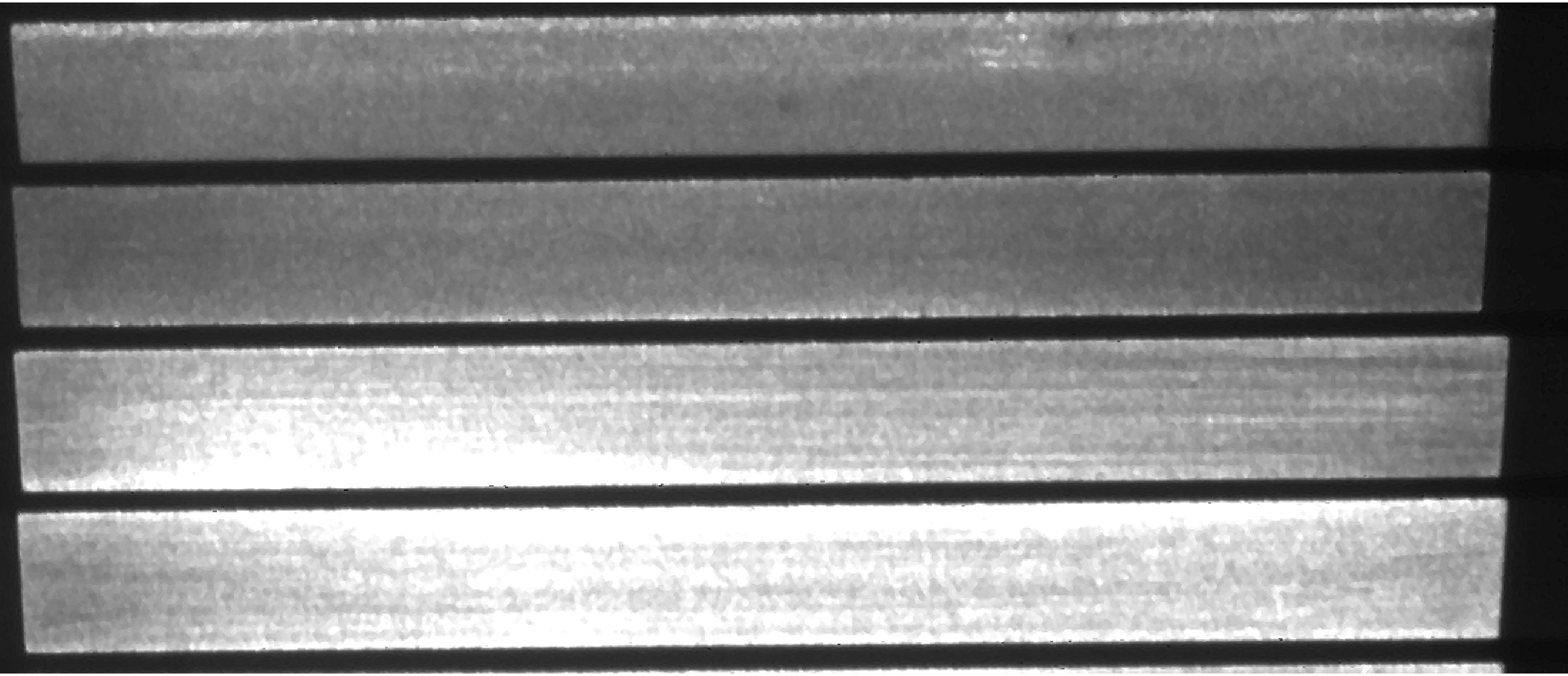
Coenen, ITPA DivSOL, Kanazawa, Jan.2014



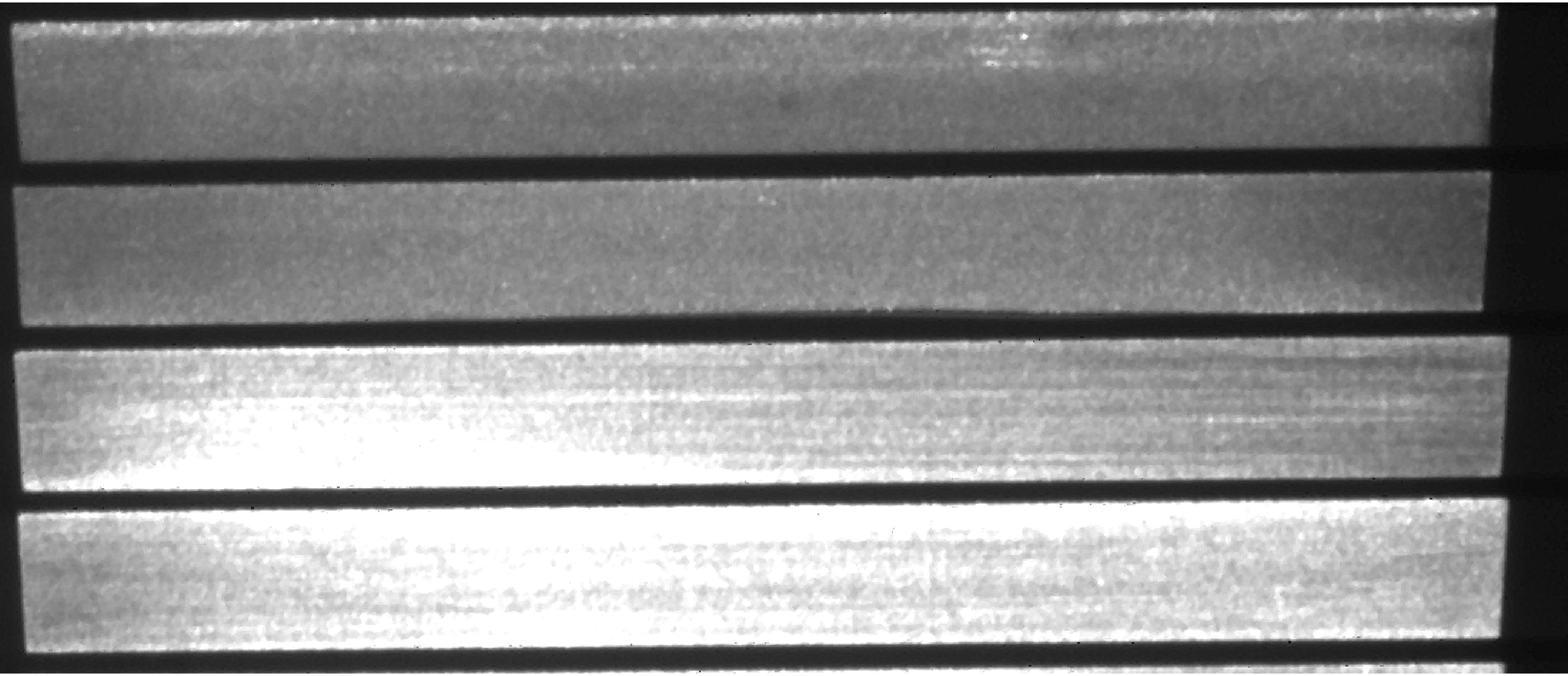
ELM analysis suggests
ΔW=300kJ with
$q_{\parallel}$ = 0.5-1 GW m^{-2}

Possible W droplet events size estimated at 100μm

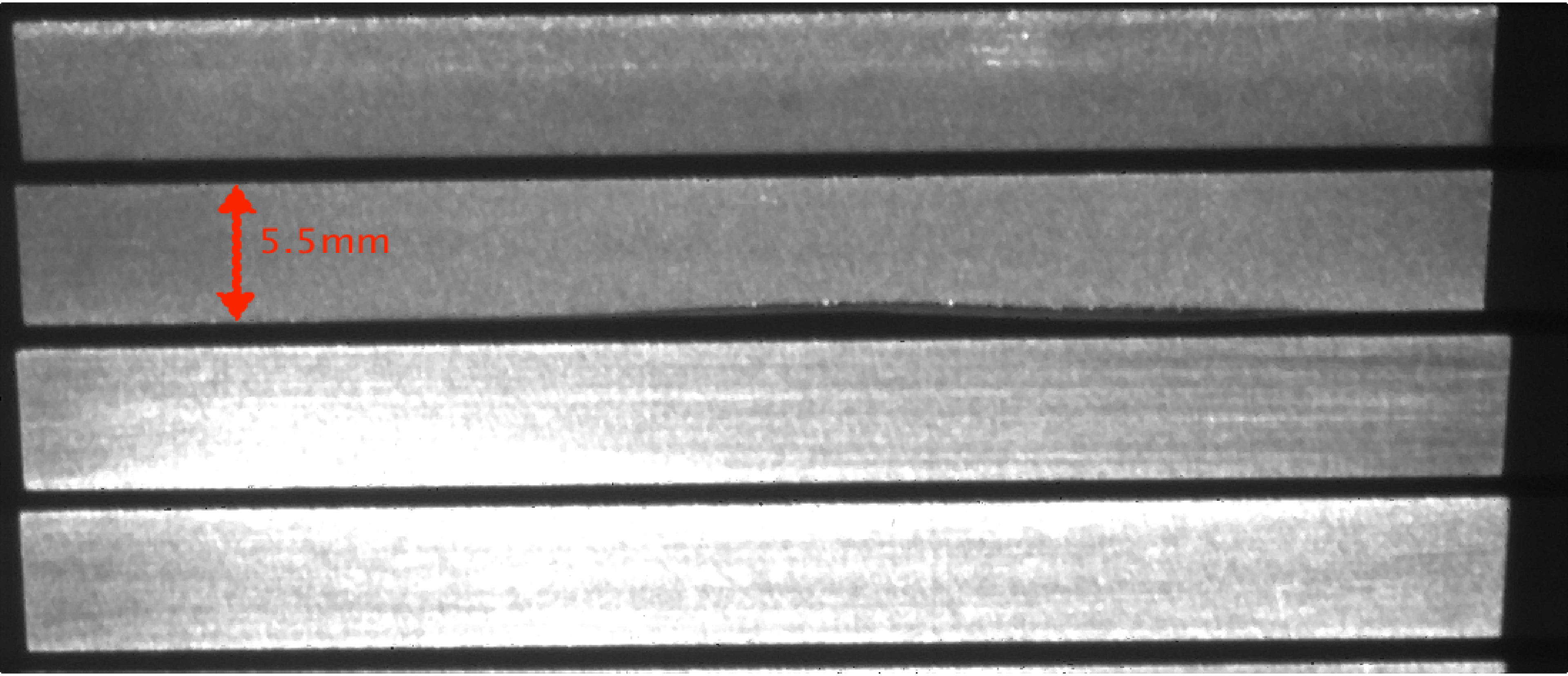
2.1: Tungsten melt experiments after #84686



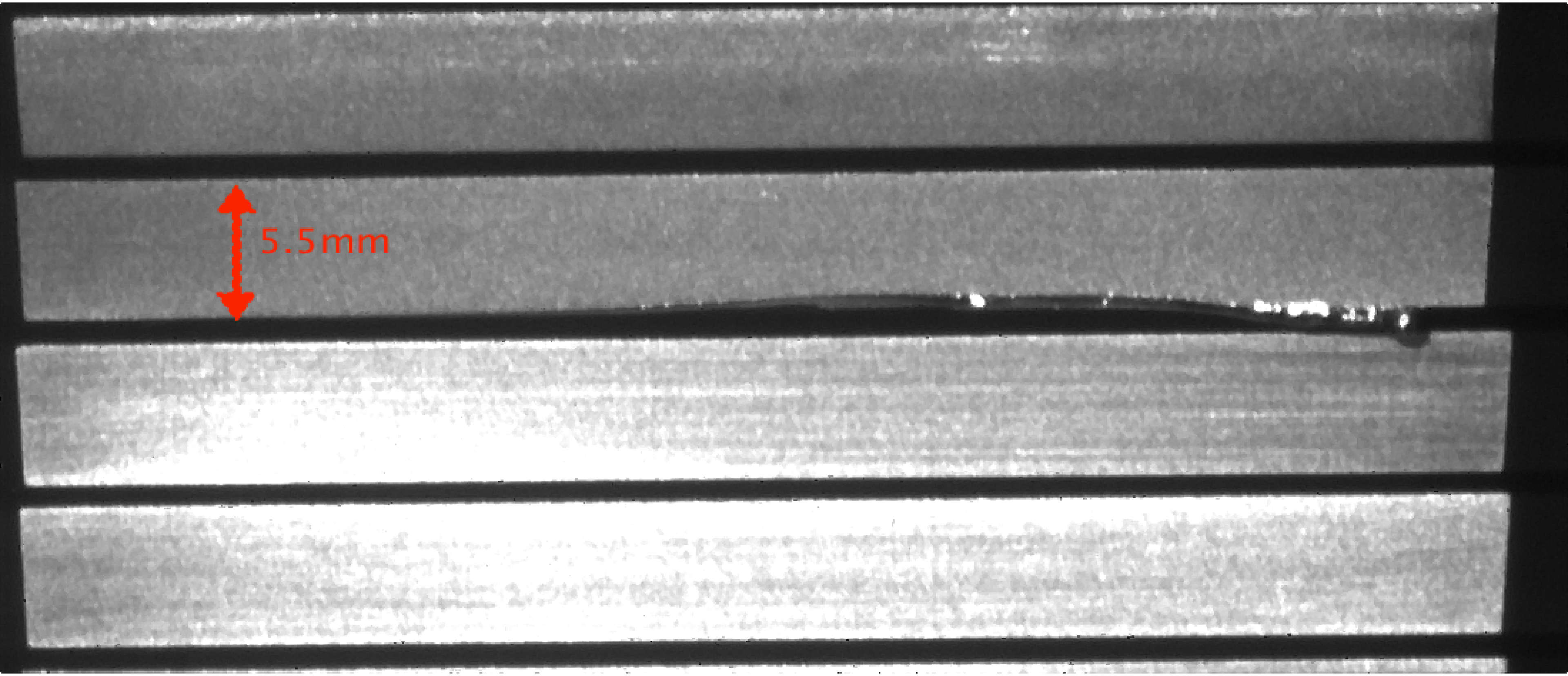
2.1: Tungsten melt experiments after #84724



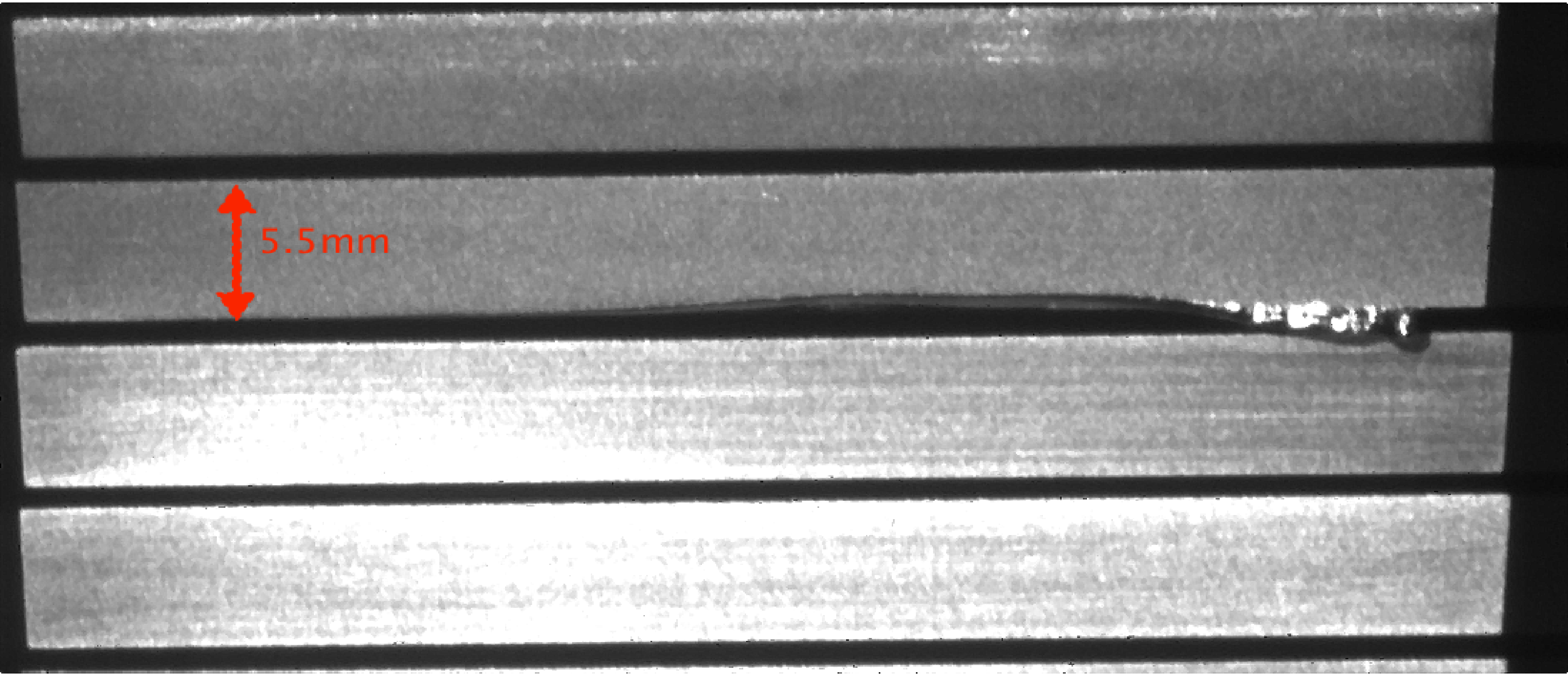
2.1: Tungsten melt experiments after #84778



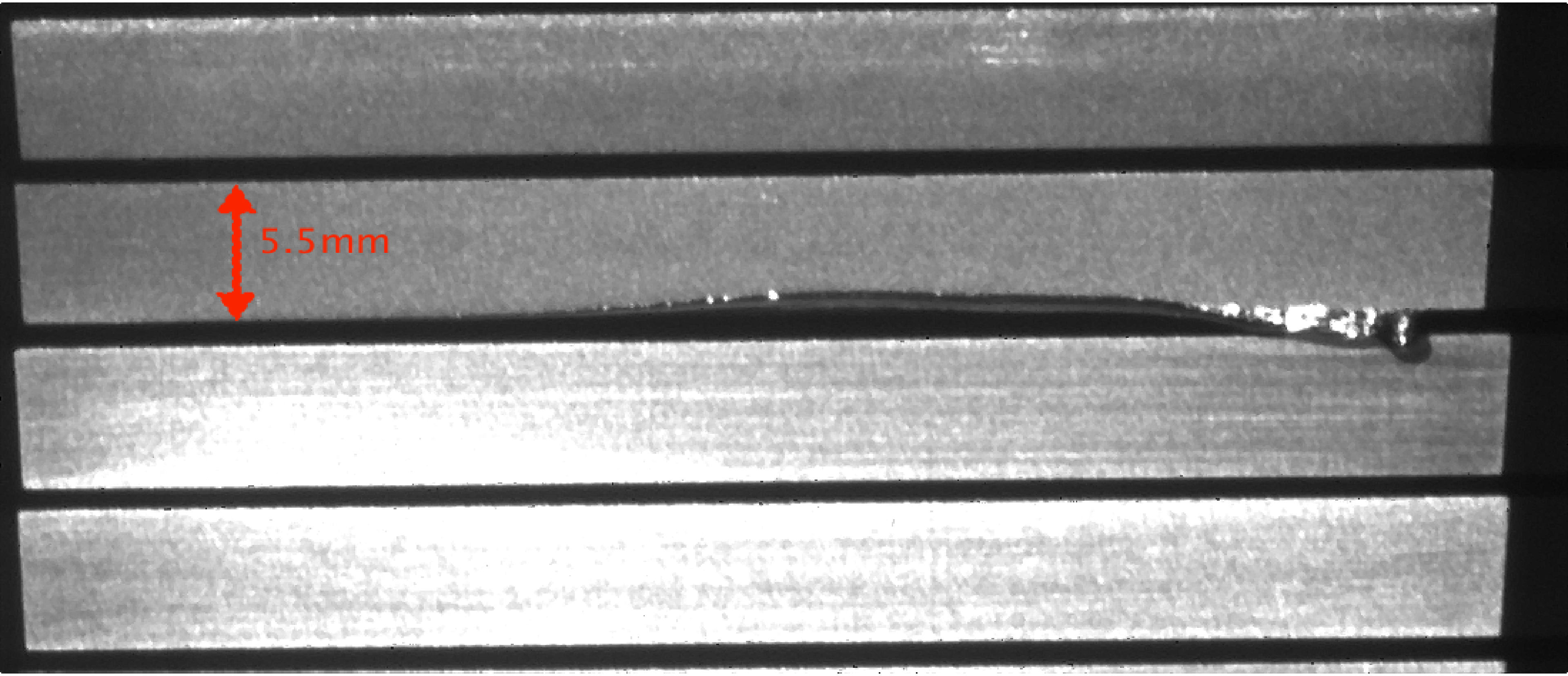
2.1: Tungsten melt experiments after #84779



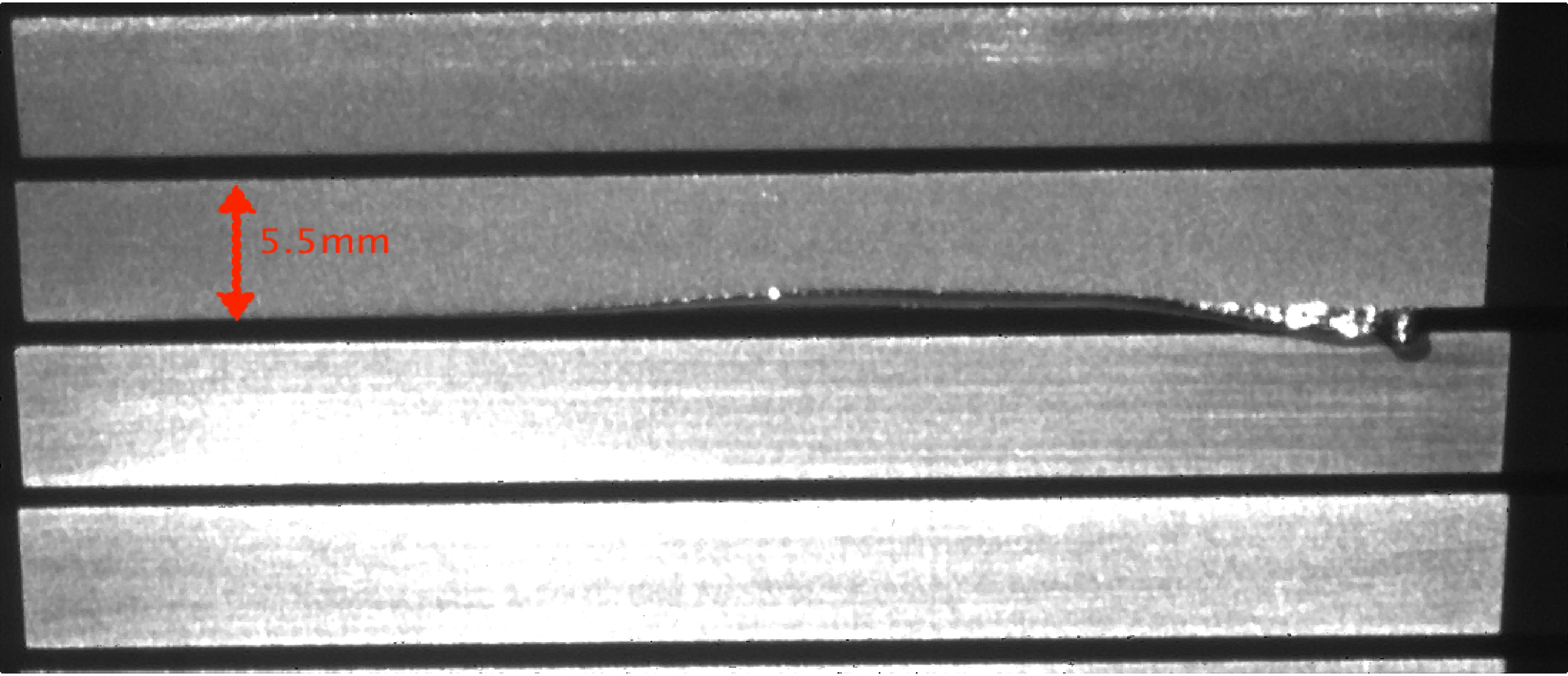
2.1: Tungsten melt experiments after #84781



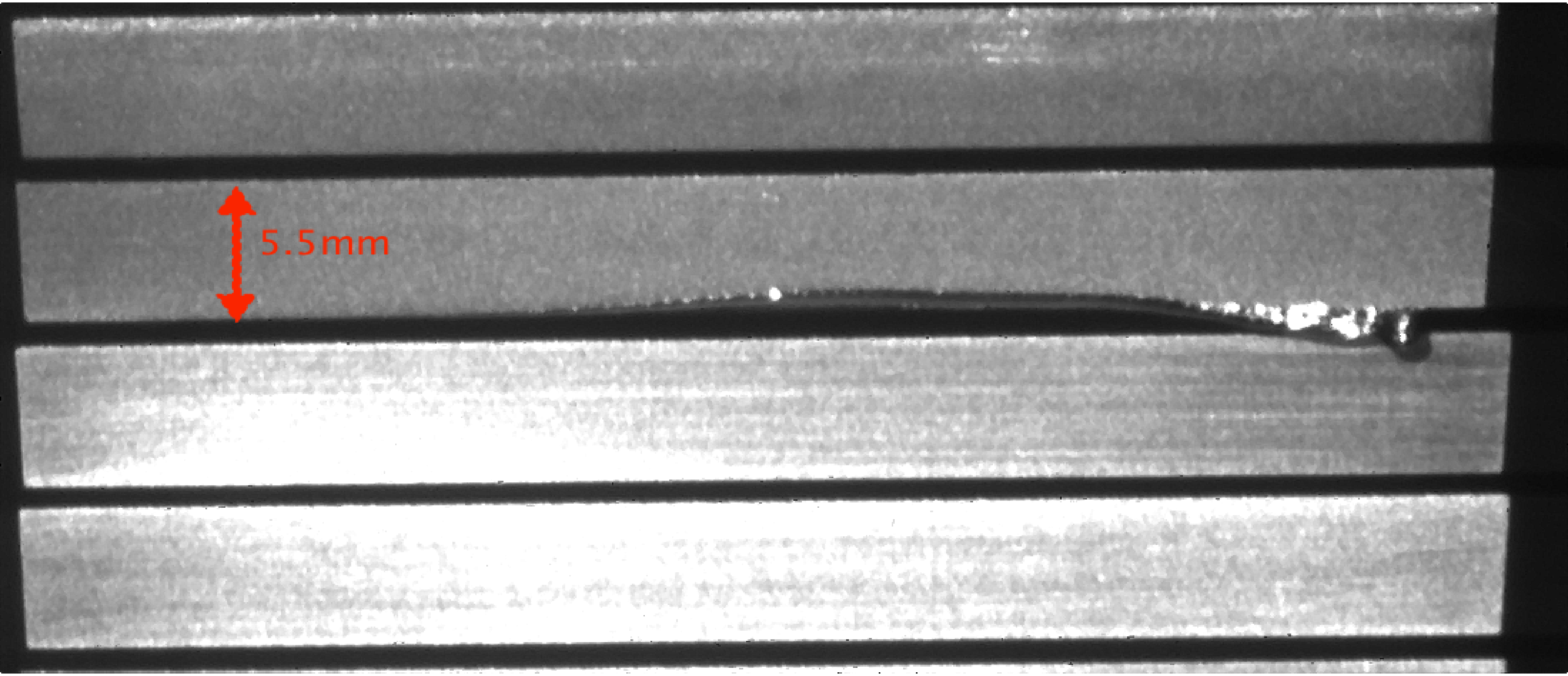
2.1: Tungsten melt experiments after #84782



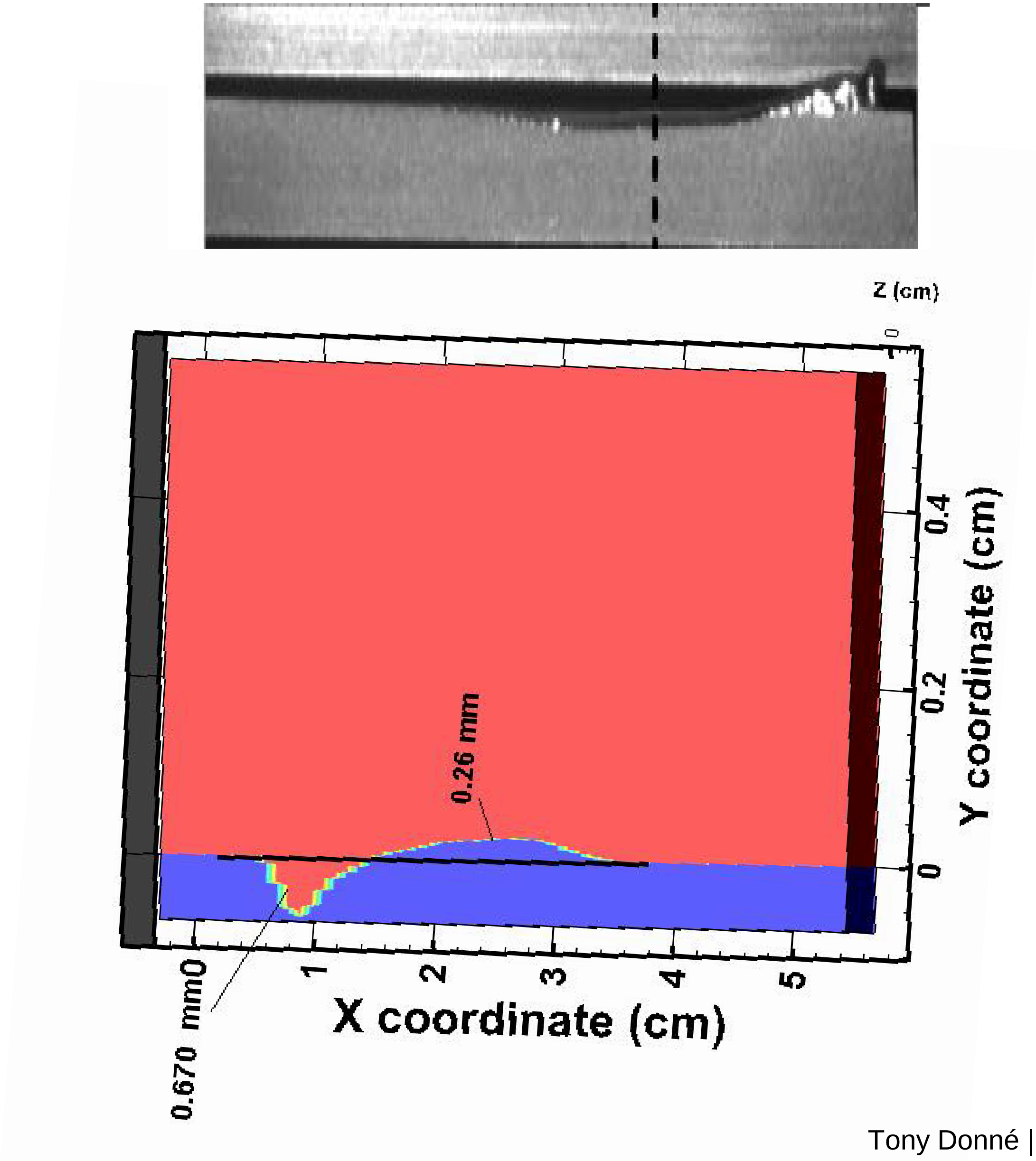
2.1: Tungsten melt experiments after #84783



2.1: Tungsten melt experiments after #84785 (final)



2.1: Tungsten melt motion

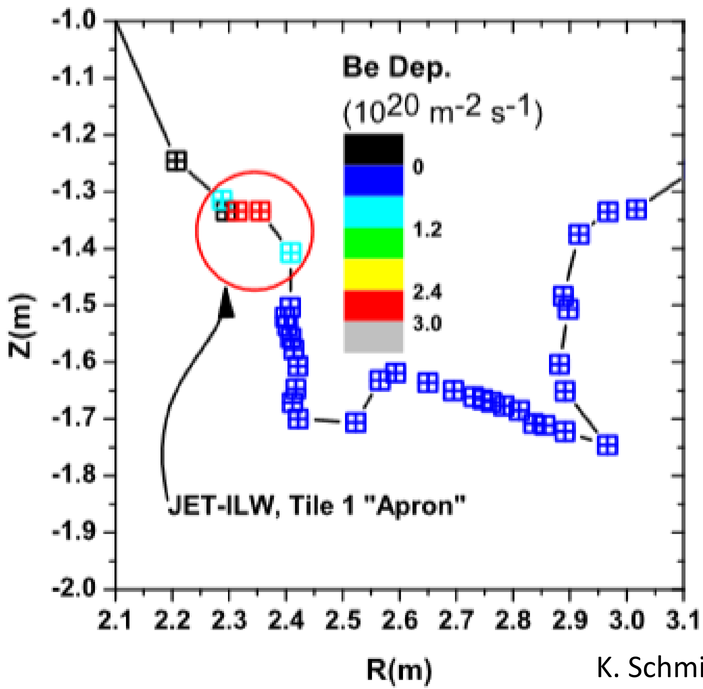


When the temperature evolution in MEMOS is fixed to measured data melt motion is correctly modelled

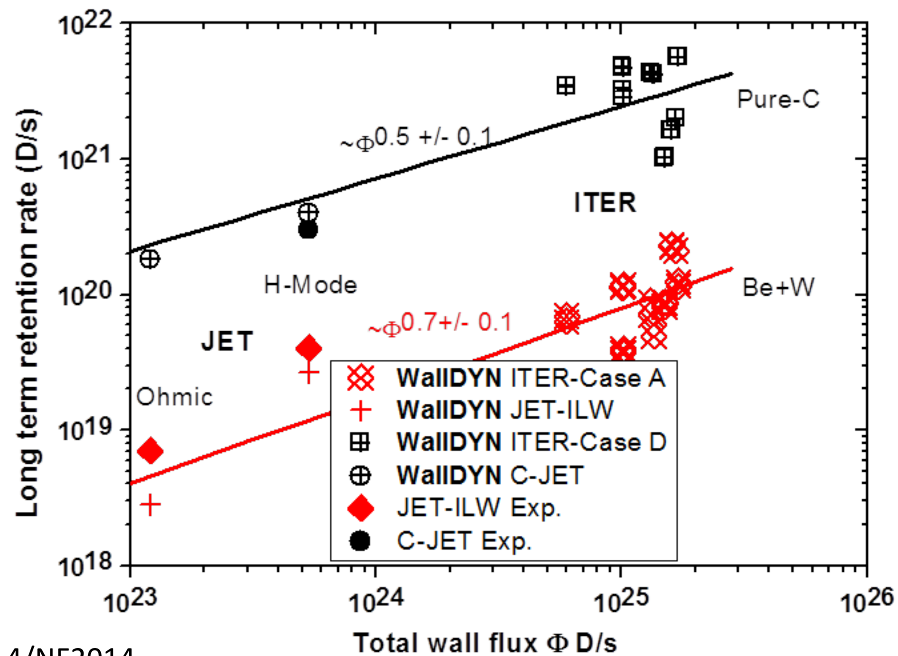
2.2 ITER Fuel Retention Predictions

- WallDYN code validated with JET-1 (gas balance) and JET-2 (post-mortem analysis)
- Extension of the code input within PFC (mixed materials, outgassing , impact of neutrons on fuel retention, seeding gases etc.)

WallDYN confirms material migration path in JET in divertor configuration



WallDYN confirms fuel retention rate (absolute values) in JET-C and JET-ILW



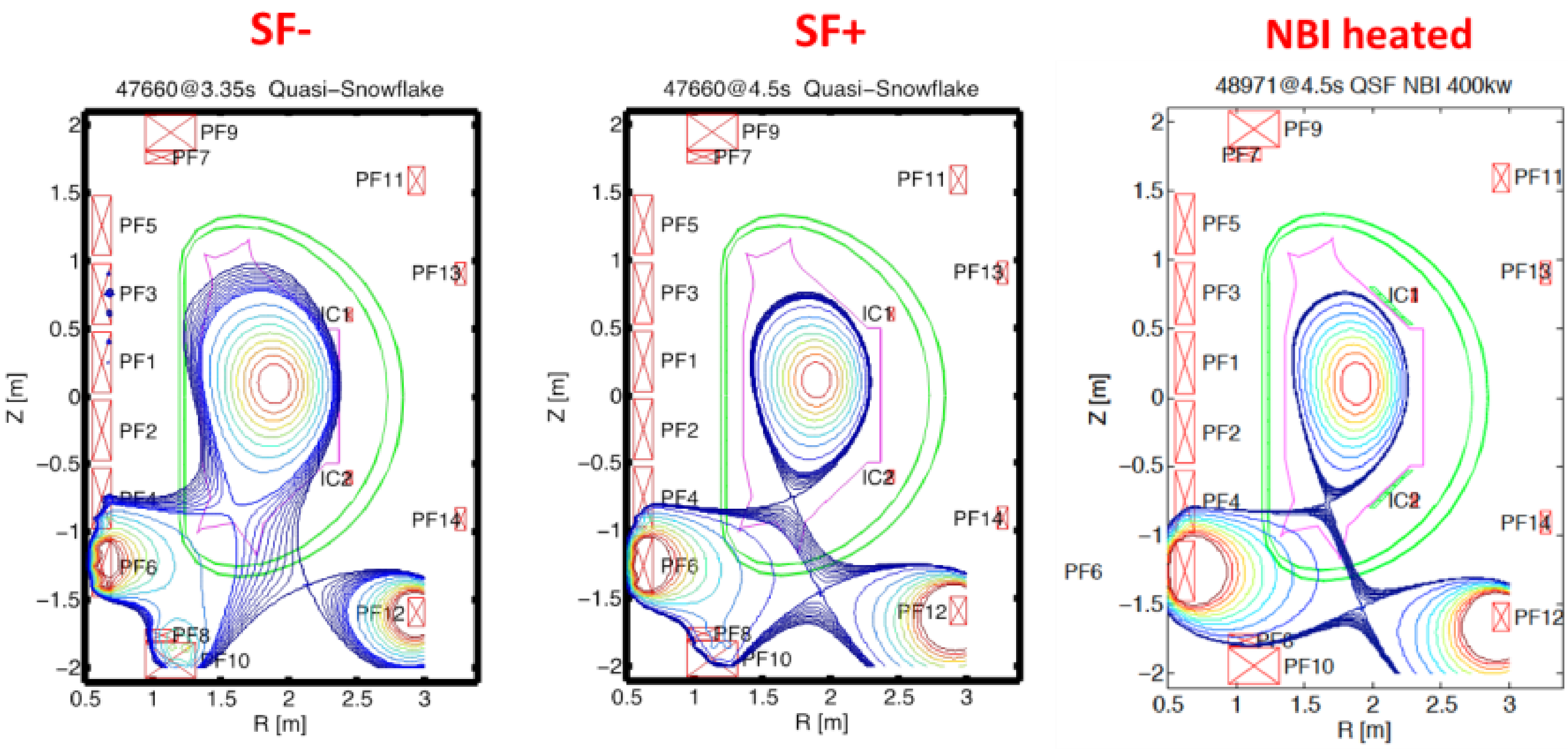
K. Schmid et al. PSI2014/NF2014

Prediction of the fuel retention limit in ITER:

- ITER with Be+W walls 3000-20000 discharges (400s DT plasmas)
- Variation depends on ITER background plasma!

2.3: alternative magnetic configurations on EAST

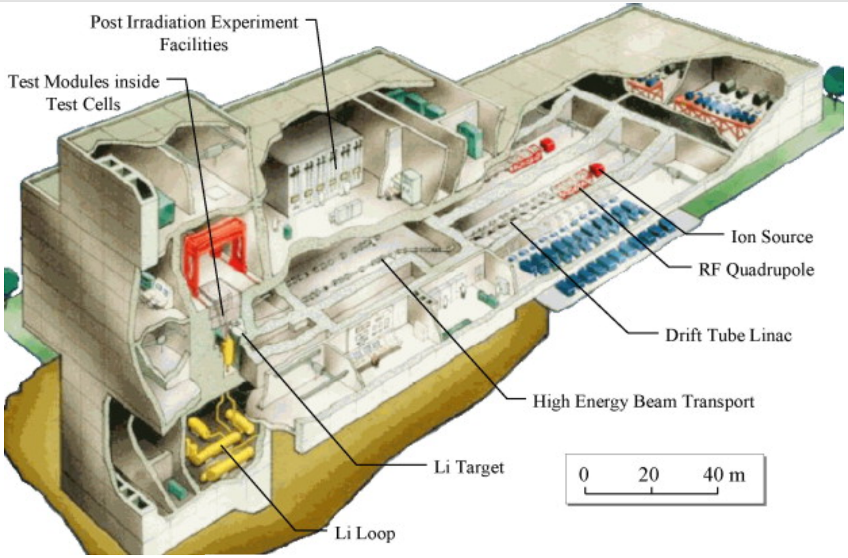
- CREATE-NL tools to optimise quasi-snowflake configuration in EAST
- First EAST experimental test of advanced configurations
  - Plasma current of 250kA ~50% below PF coil current limits

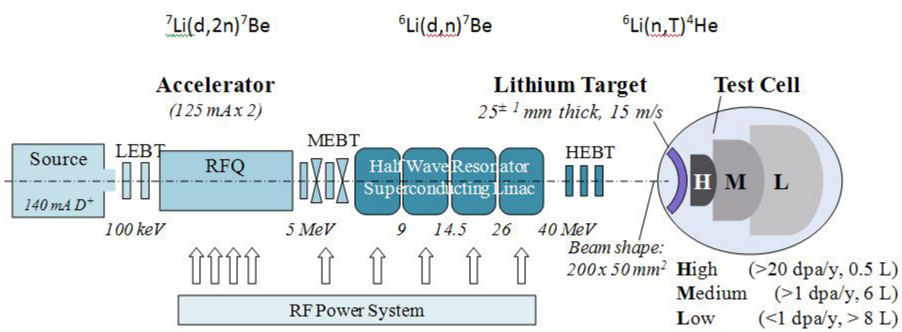


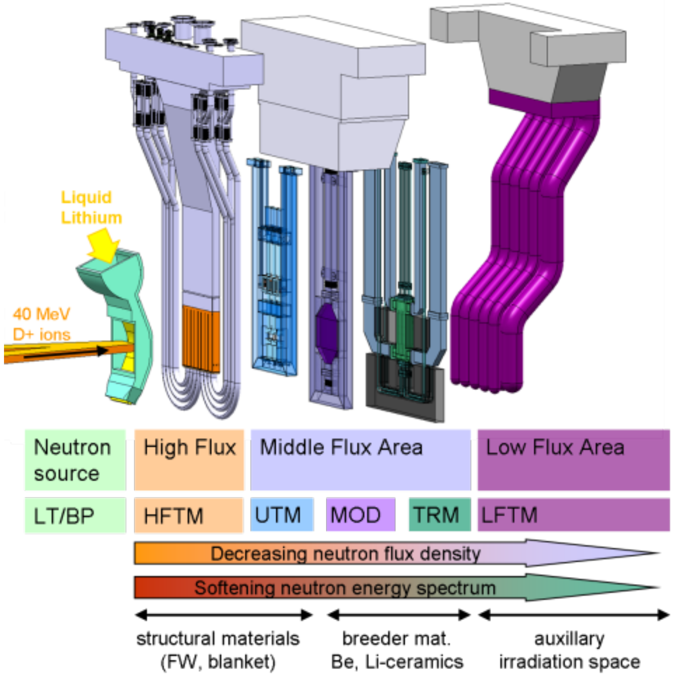
3: Neutron Resistant Materials

Full characterization of the baseline materials for DEMO:
EUROFER as structural material
Tungsten as Plasma Facing Component
Copper-alloys for cooling

Expand the operational range of these materials (e.g. EUROFER has an operational range of 350 – 550 °C





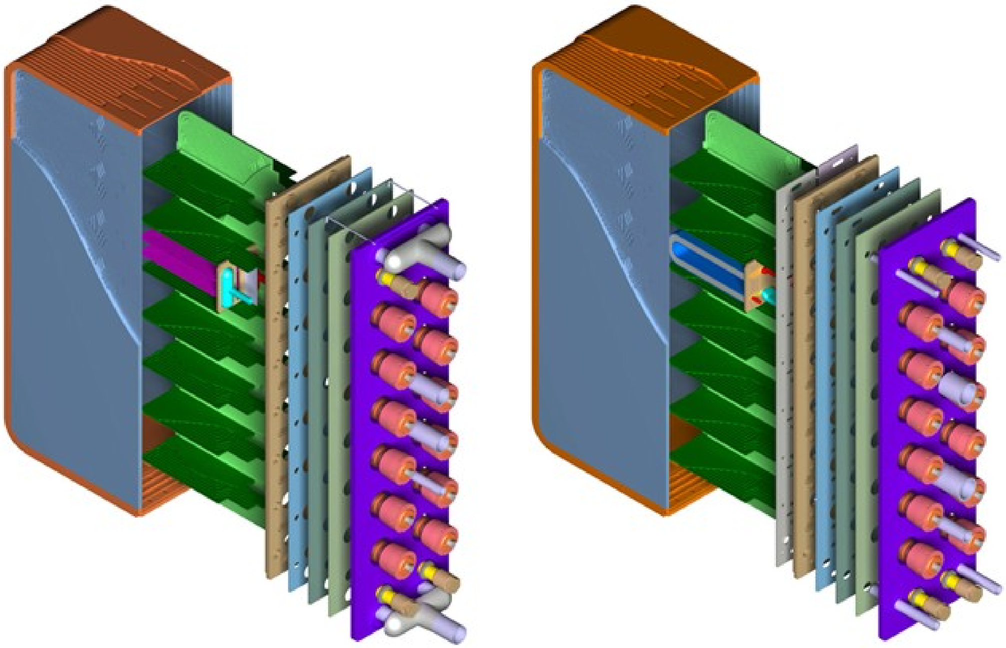


Main devices: IFMIF, Early Neutron Source, Irradiation facilities

4: Tritium self-sufficiency

Main question is whether a fusion reactor can produce enough tritium for its own fuel supply

Research concentrated on Two test blanket modules in ITER



Research in extraction of tritium from the blankets

Main devices (on ITER):

- TBM based on eutectic Pb-16Li and TBM on ceramic material; both using He as coolant
- Possibly also research in water-cooled Pb-16Li

5: Intrinsic safety features

A relatively small mission to study the specific nuclear licensing procedures for DEMO and to study how the amount of radioactive waste can be reduced as much as possible.

Differences between ITER and DEMO in this respect are the much higher neutron and tritium fluences

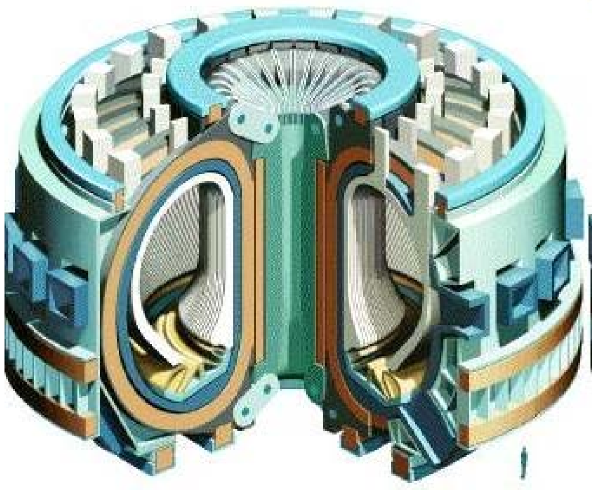
Main device: ITER

6: Integrated DEMO design and system development

Find ways to reduce degradation of superconducting cables under continuously changing loads

Study application of high T_c superconductors

Increase gyrotron frequencies for ECRH and ECCD to ~230 GHz

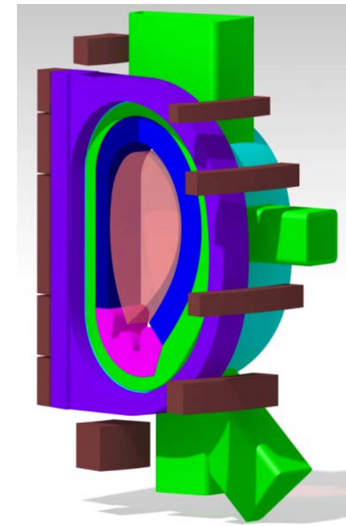


Optimize remote handling and remote maintenance strategies

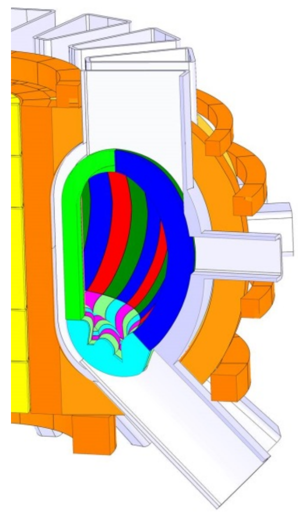
Develop control strategies for underdiagnosed plasmas

6.1 Evolution of the DEMO CAD geometry

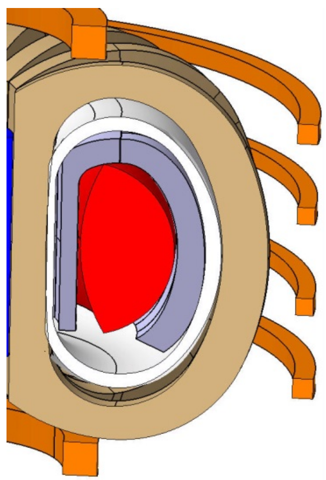
2014



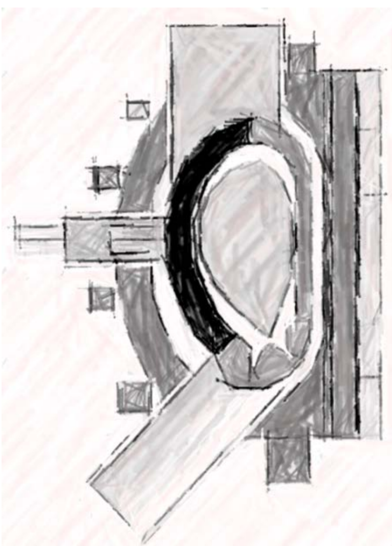
2013

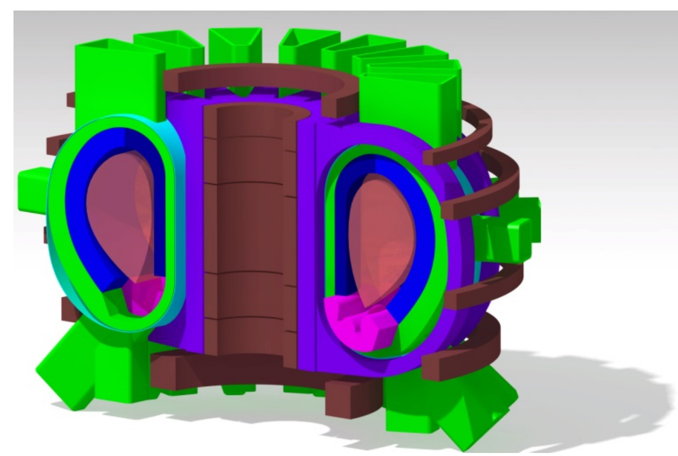


2012



2011

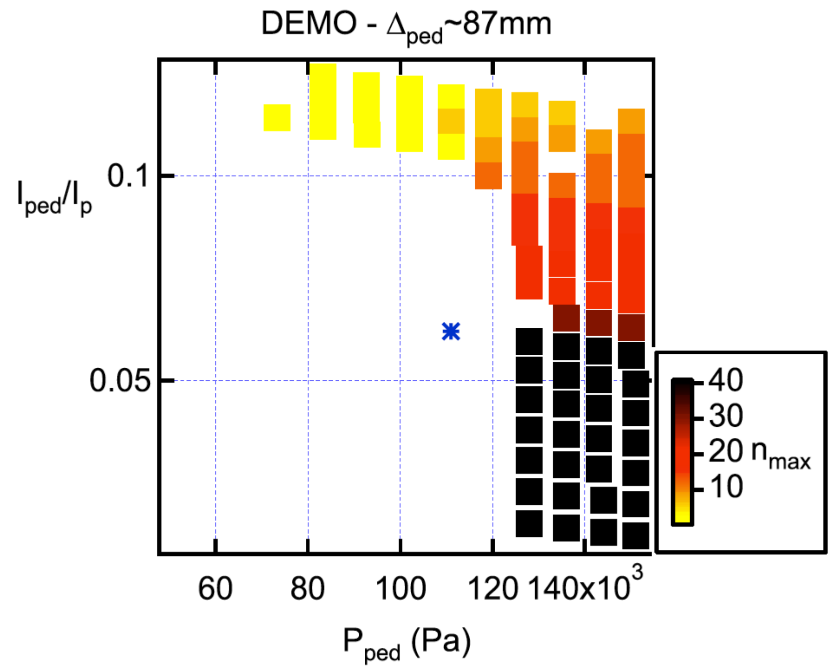




Advancing the DEMO Physics Basis

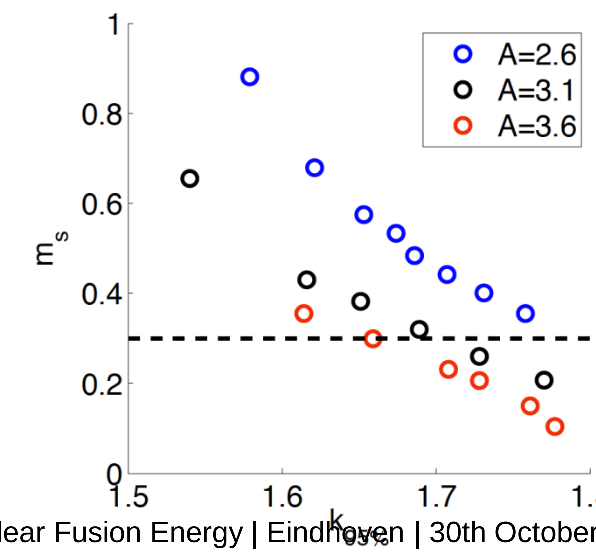
Currently investigated:

- Scenario Modeling
- Transport
- MHD
- H&CD
- Fast Particles
- Plasma Wall Interactions
- Disruptions



Examples:

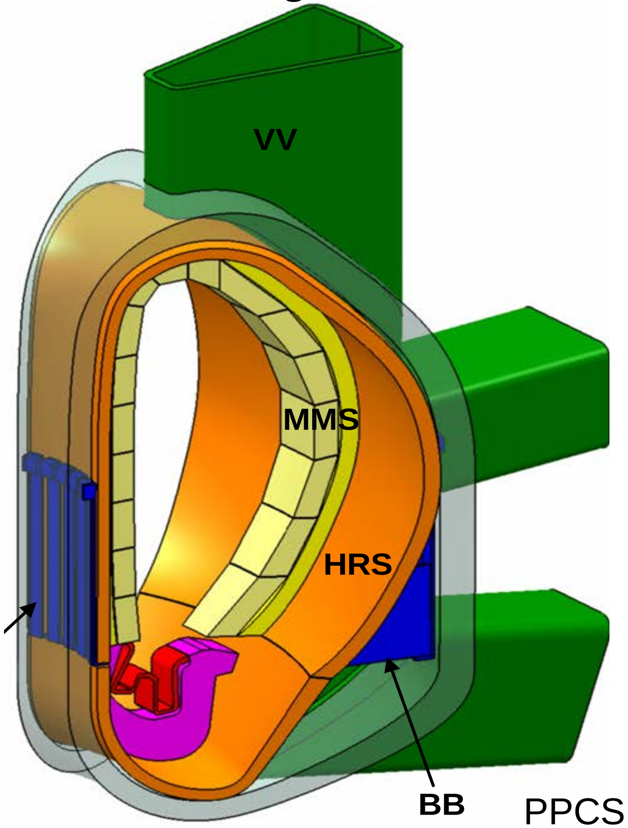
- p_{ped} of a pulsed DEMO has been predicted to be 130kPa (5.6keV at $n_{ped}=0.85n_{GW}$)
- Maximum k_{95}(A) with acceptable active and passive stability features has been evaluated in a selfconsistent way



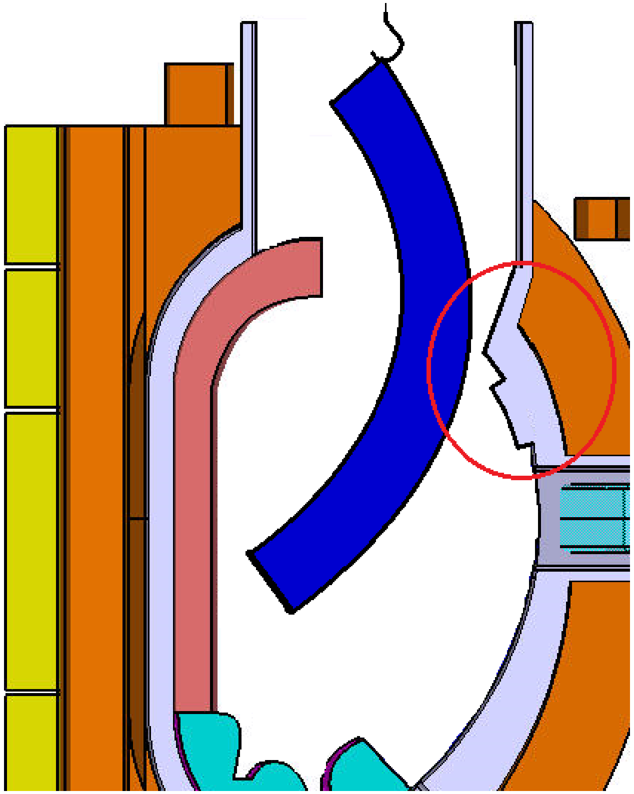
In-vessel shield

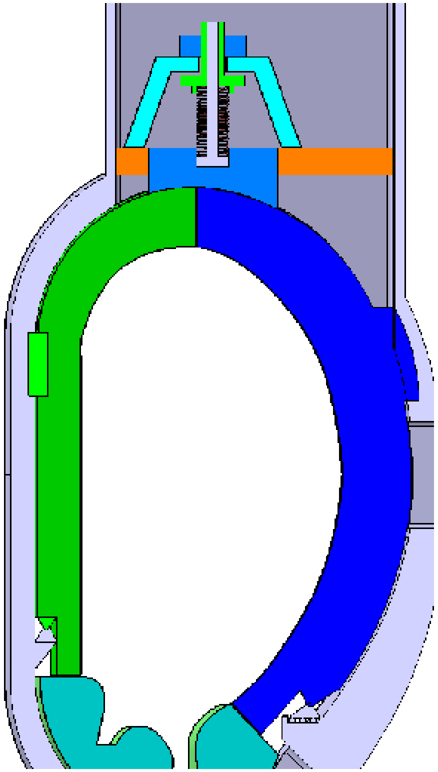
The in-vessel shield (orange, HRS) present in the PPCS and other power plant studies (e.g. JA DEMO), would not be required in case the blanket could be directly supported from the vessel.

- Promising direct attachment concepts were developed, no show-stopper was identified.
- Shielding function of the shield would be transferred to the vessel



PPCS



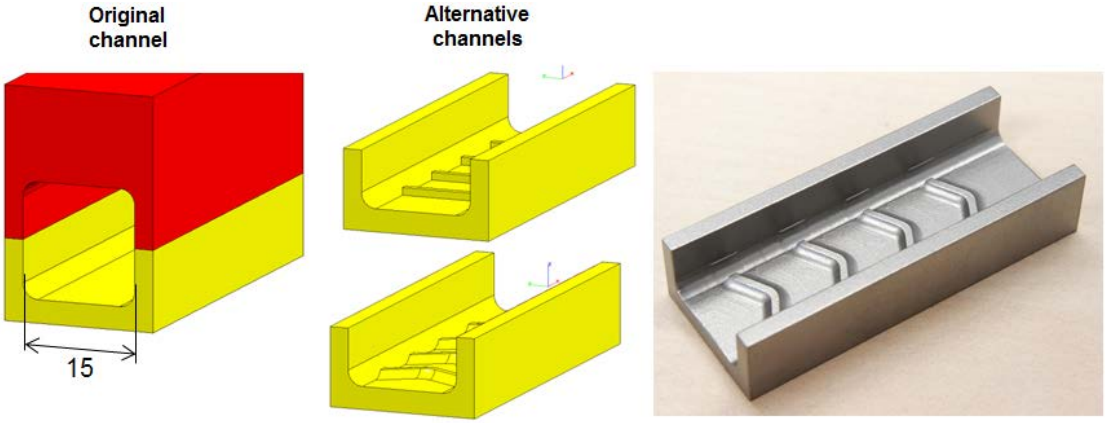


In-vessel component design

First-Wall Design

- Large uncertainty on the thermal load specifications (interactions with IPH)
- FW engineering HF limit << ITER specs (2-5 MW/m²)
    - 0.7-1.0 MW/m² (helium-cooled)
    - 1.0 – 1.7 MW/m² (water-cooled)
- Assess Plasma Protection Limiters in DEMO

Development of helium-cooled FW channel geometries with improved thermohydraulic performance:



7: Competitive cost of electricity

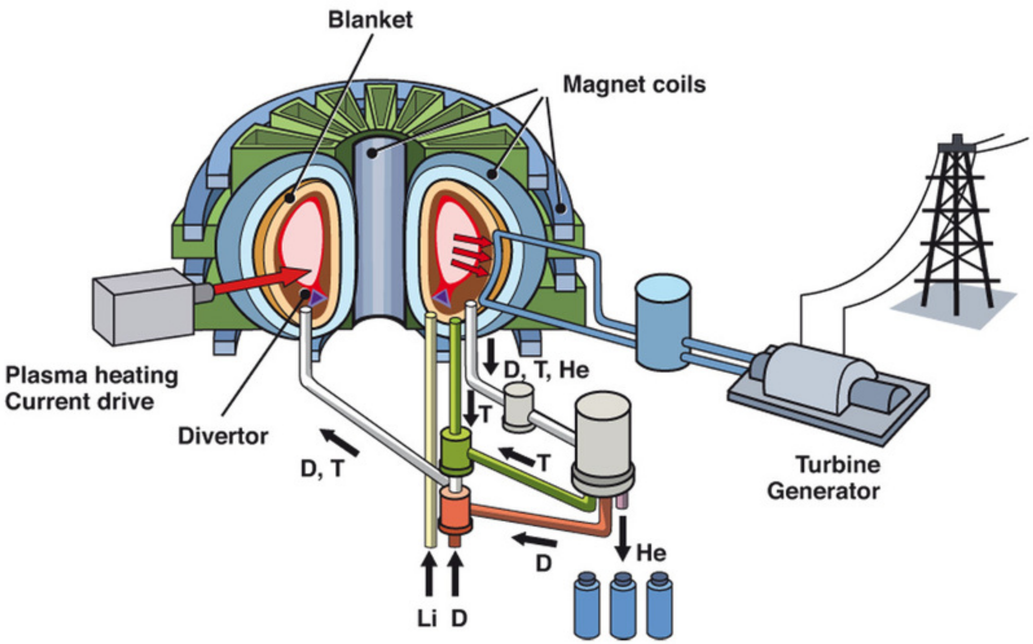
Which impact do design choices for DEMO have on the ultimate price of electricity:

Cheap and straightforward design solutions

Components with long life-expectancy

High machine availability

High temperature superconductors?

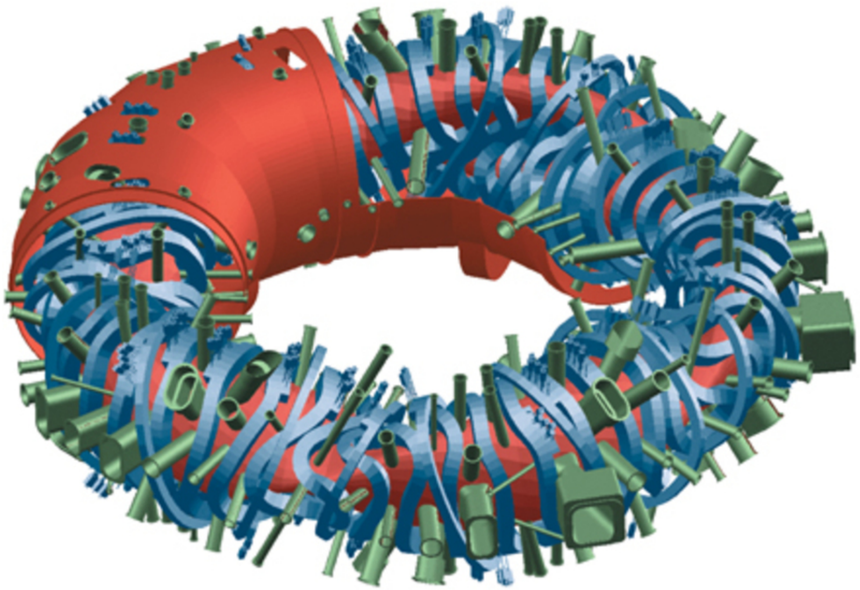


8: Stellarator

Stellarators are behind tokamaks performance wise

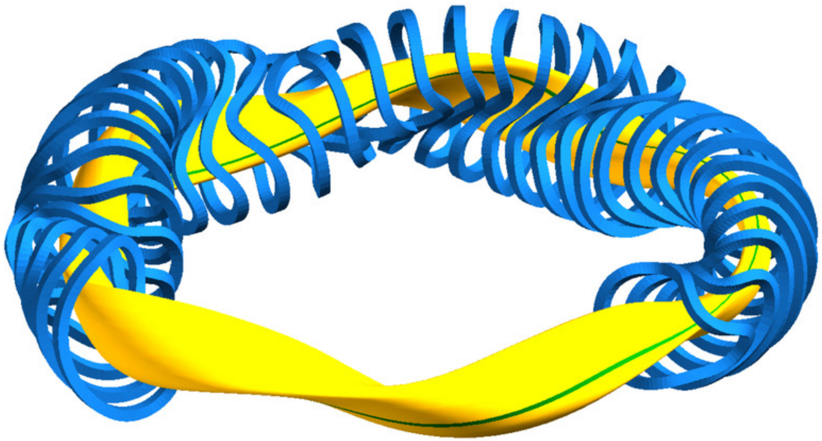
Stellarators are technically complicated

But, stellarators are by definition stable and steady state and they offer a number of important advantages for a fusion reactor



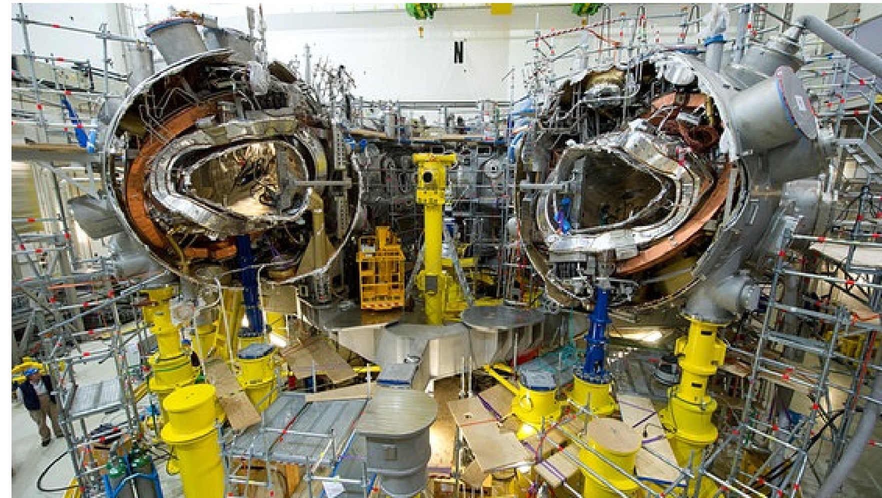
Main device: Wendelstein 7-X

8.1 Stellarator



Wendelstein 7-X

First operation in 2015



Proposal for risk mitigation of ITER

ITER has significant delays

☞ Use contemporary fusion devices as risk mitigation for ITER

☞ Reduce the ITER non-active phase by proper preparation and training elsewhere

Proposal to next ITER Council

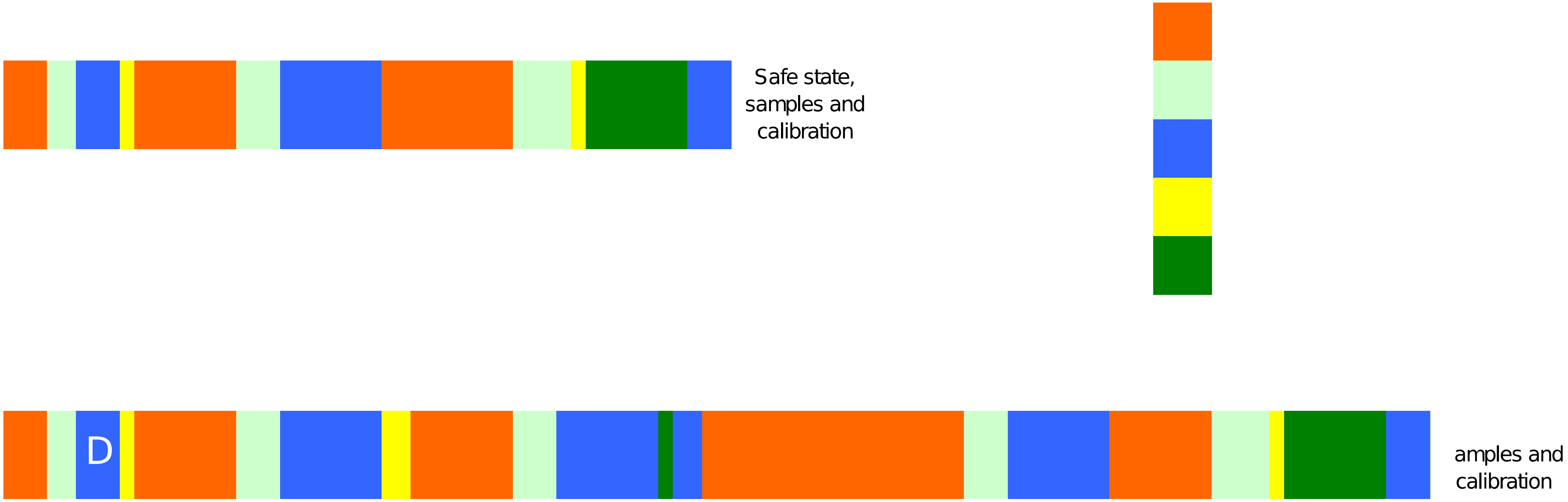
☞ Establish an Int. Task Force to make a detailed proposal how present machines can be used to bring DT phase of ITER forward

Specific JET contributions:

Only machine with DT, with ITER-like wall (Be-handling), full remote handling, organizing truly int. scientific campaigns, training

Future of JET - Internationalization

Long term JET plan depends on the success of the internationalization process (Wagner Panel on Strategic Orientation)



*Details of the Alternative Scenario are not yet agreed

The feasibility of two further major enhancements (RMP Coils for ELM Control, ECRH) have been studied

DT Campaign options

DT Campaign options		**Full DT phase**	**DT phase ~DTE1**	**100% tritium only**	**Trace tritium**
	14 MeV budget	**$1.7x10^{21}$**	**$2.5x10^{20}$**	**$5.0x10^{19}$**	**$5.0x10^{18}$**
ITER Scenarios in DT*	Baseline	20	8	200	
	Hybrid	40	2	200	
	Steady State	20	0	50	
Technology	Tritium retention				
	14 MeV calibration				
	Use 14 MeV Fluence				
Physics	Retention removal				
	Isotope scaling				
	α-particle effects				
	Fuelling & DT mix control				transport

Number of high power (>25MW, 5s) pulses in DT (or 100% tritium) is indicated.

ITER risk mitigation

Impact of DT at JET

Maximum

Limited

No

$1.7x10^{21}$ budget: Full exploitation of JET for mitigating the risks for ITER.

Conclusion

EUROfusion programme is in full swing

All members and linked third parties can participate

For Netherlands:

- FOM-DIFFER (full member)
- NRG, TU/e, CWI, UT and TUD (linked third parties)