A Selected Summary Report

A bi-annual meeting on Plasma Facing Materials and Components
First one to include structural materials in the presentations

PFMC-13 Workshop
May 9-13, 2011 Rosenheim, Germany

Clement Wong
General Atomics

Plasma Facing Components Meeting
August 10-12, 2011, ORNL, TN, USA

PFMC-13 Group Photo
May 9-13, 2011 Rosenheim, Germany

PFMC-13 covered many specific PFM areas relevant to ITER with mostly EU participants ~280, >30 orals, >200 posters

Carbon, beryllium, and tungsten based materials	
JET-ILW, C-13 transport	Culham
W-design for JET, W for EAST, R&D	Culham, **EAST, Osaka**
H in W, W-coating, Self-implanted W, in-situ H measurement	IPP, Julich, Oxford, **Nagoya**
ITER	ITER
Be-Tritium	IPP
W-transport, D-retention in W, arcing	IPP, **Sizuoka, UCSD**
CFC 3-D tomography	Munchen
Mixed materials, Be/W, C/W, Be/C	Many posters
Erosion and redeposition	Culham + posters
High heat flux component development	
Testing, plasma gun, thermal shock	Julich, Triniti
Benchmarking of radiation damage & modelling	
Radiation experiment, SiC/SiC	Di Torino, Germany
Modeling, co-deposit transport, Be-transport	France, CEA, IPP, **Triniti**
ELM	Netherlands
Disruption	Julich
Others:	
Li	**NSTX**
ITER divertor, FW life-time	ITER
ODS-FS	Warsaw
W/W composite	IPP
T permeation barrier	**Toyama**
Diagnostics…mirror	Culham

Distribution of FW Panel Design Heat Load

Steady state:

$q_{\|} \sim 8\ MWm^{-2}$, $\lambda_{q\|} > 4.0$ cm
$q_{\|} \sim 24\ MWm^{-2}$, $\lambda_{q\|} > 2.5$ cm (ELMs)

Disruptions

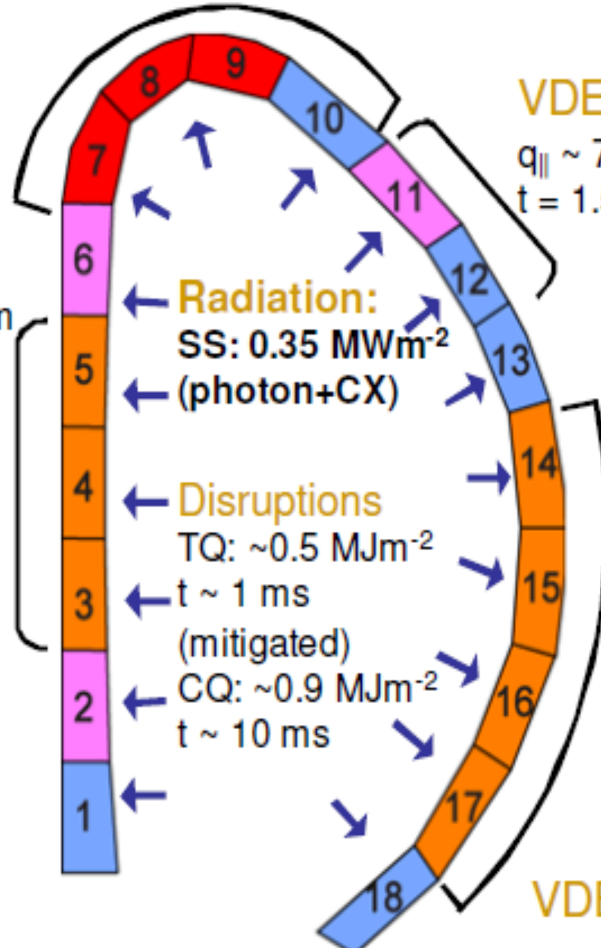
$q_{\|} \sim 45\text{-}120\ MJm^{-2}$, $\lambda_{q\|} > 20$ cm
t = 3.0-6.0 ms

Start-up:

$q_{\|} \sim 25\ MWm^{-2}$, $\lambda_{q\|} \sim 5.0$ cm
Several seconds

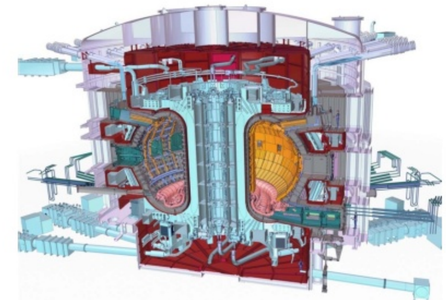
Confinement transients

$q_{\|} \sim 250\ MWm^{-2}$, ~2-3 secs



VDE (up):

$q_{\|} \sim 70\text{-}270\ MJm^{-2}$, $\lambda_{q\|} > 3.0$ cm
t = 1.5-3.0 ms



Start-up and rampdown:

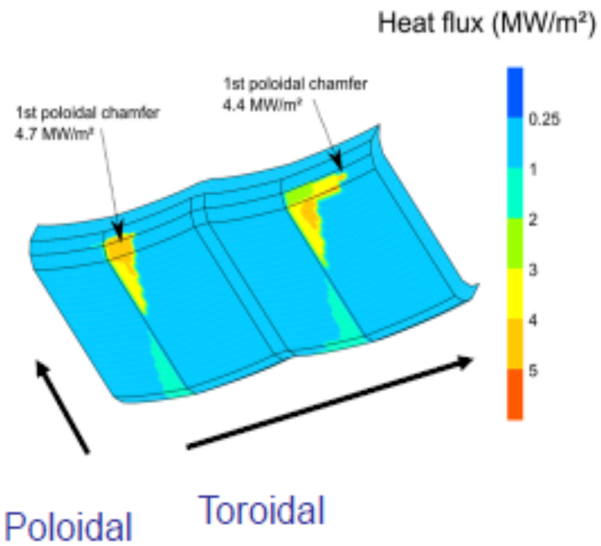
$q_{\|} \sim 40\ MWm^{-2}$,
$\lambda_{q\|} > 1.2$ cm
Several seconds

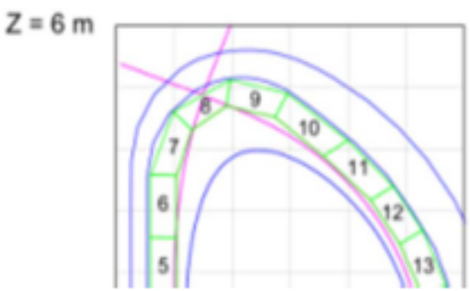
VDE (down):

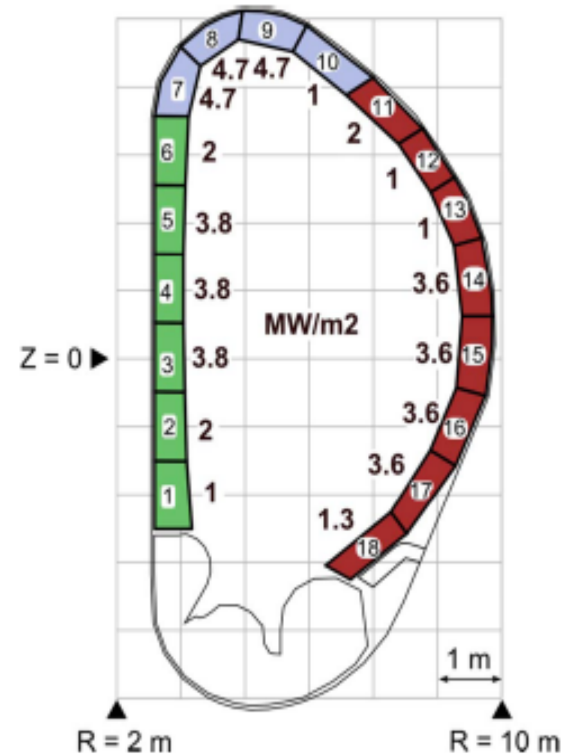
$q_{\|} \sim 90\text{-}300\ MJm^{-2}$, $\lambda_{q\|} > 3.0$ cm

Design Heat Load on Blanket

Outboard flat top heat load (550 MW)





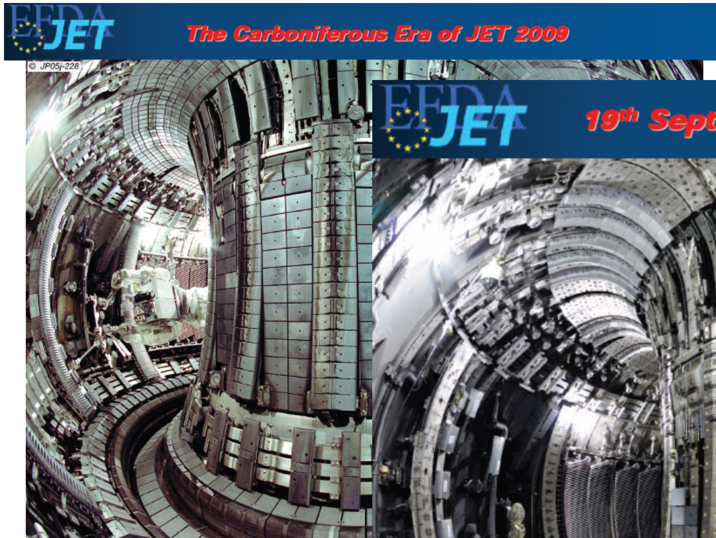


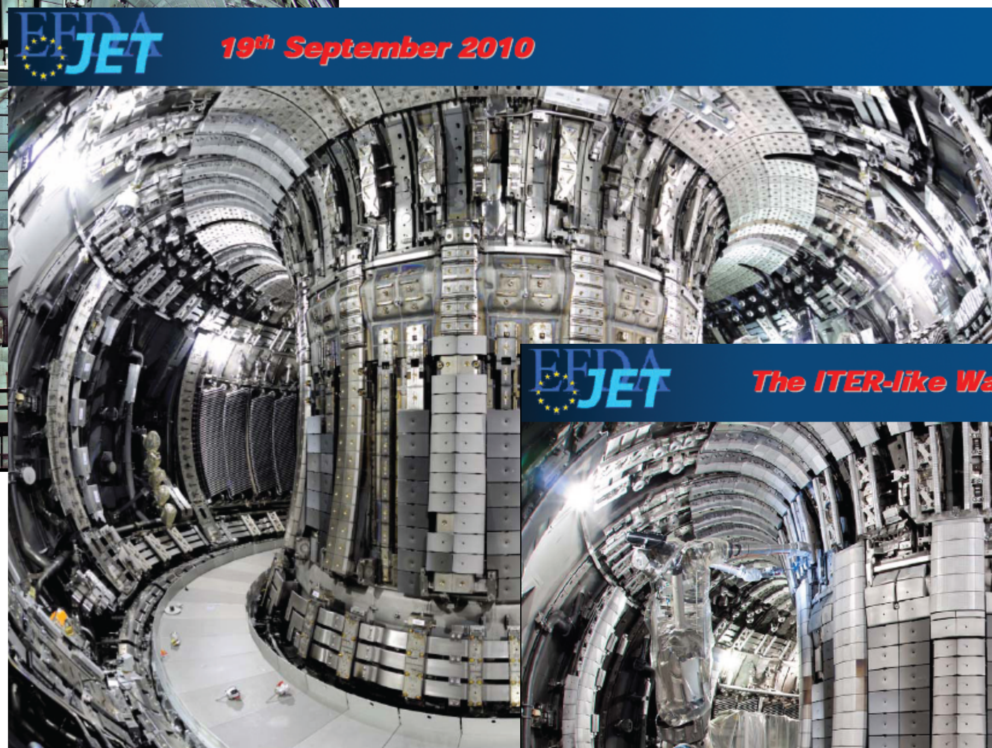
- Group 1 : 1 – 2 MW/m²
  Normal heat flux panels
- Group 2 : 3.5 – 5 MW/m²
  Enhanced heat flux panels

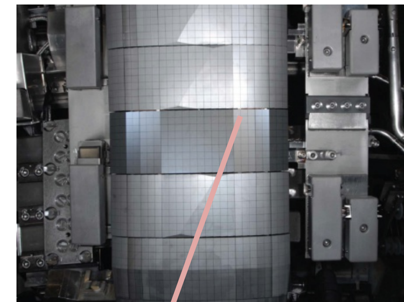
See talk by R. Mitteau

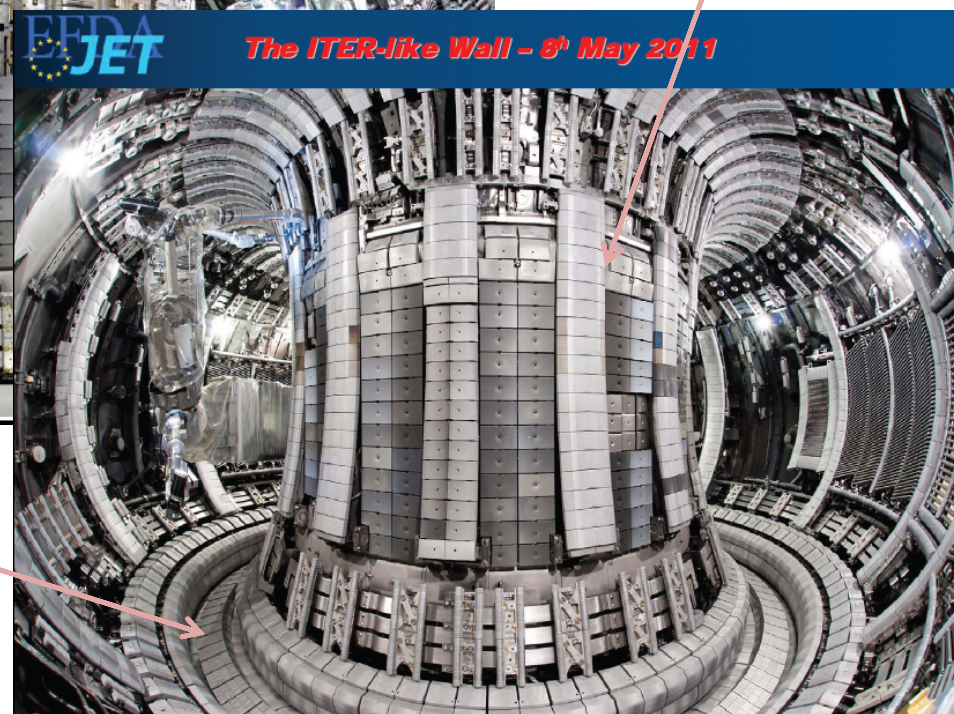
JET with the ITER like Wall with Be-tiles, W-lamella and W-coated Tiles

G. Matthews





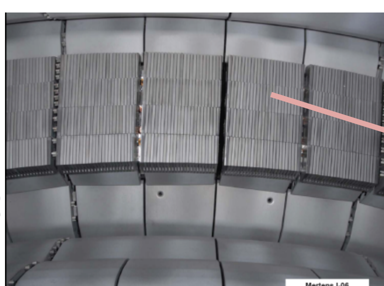




~60M€

5,384 Be tiles (~2 tons Be / ~ 1m^3)
1,288 W-coated CFC tiles
9,216 W-lamellas (~2 tons W / ~ 0.1m^3)
15,828

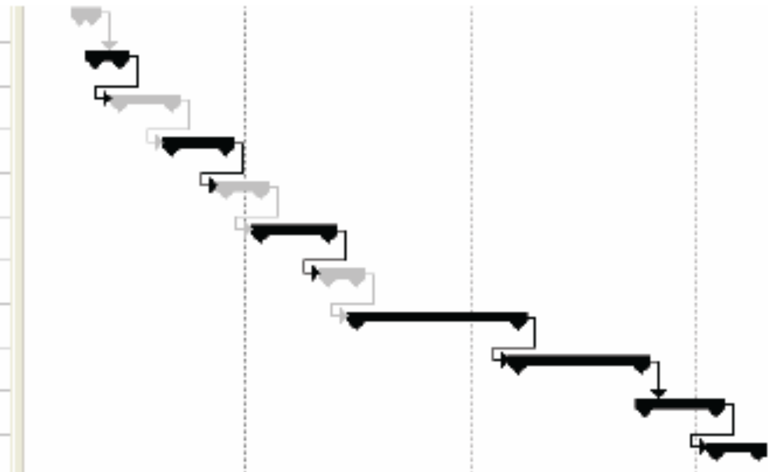
82,273 counting bulk W modules as one part



Overall structure of the 2011/12 programme

Gradual expansion of operating space

Restart 1 - including conditioning studies
C28A Ohmic studies - first material migration/mixing
Restart 2
C28B L-mode Studies and initial H-mode
Restart 3
C28C Establish and characterise first H-modes
Restart 4
C29 Establish and exploit robust H-modes and ELM mitigation
C30A Expansion of operating space including hybrid modes
C30B Exploitation of available operating space
C30C Operation prior to long term sample retrieval

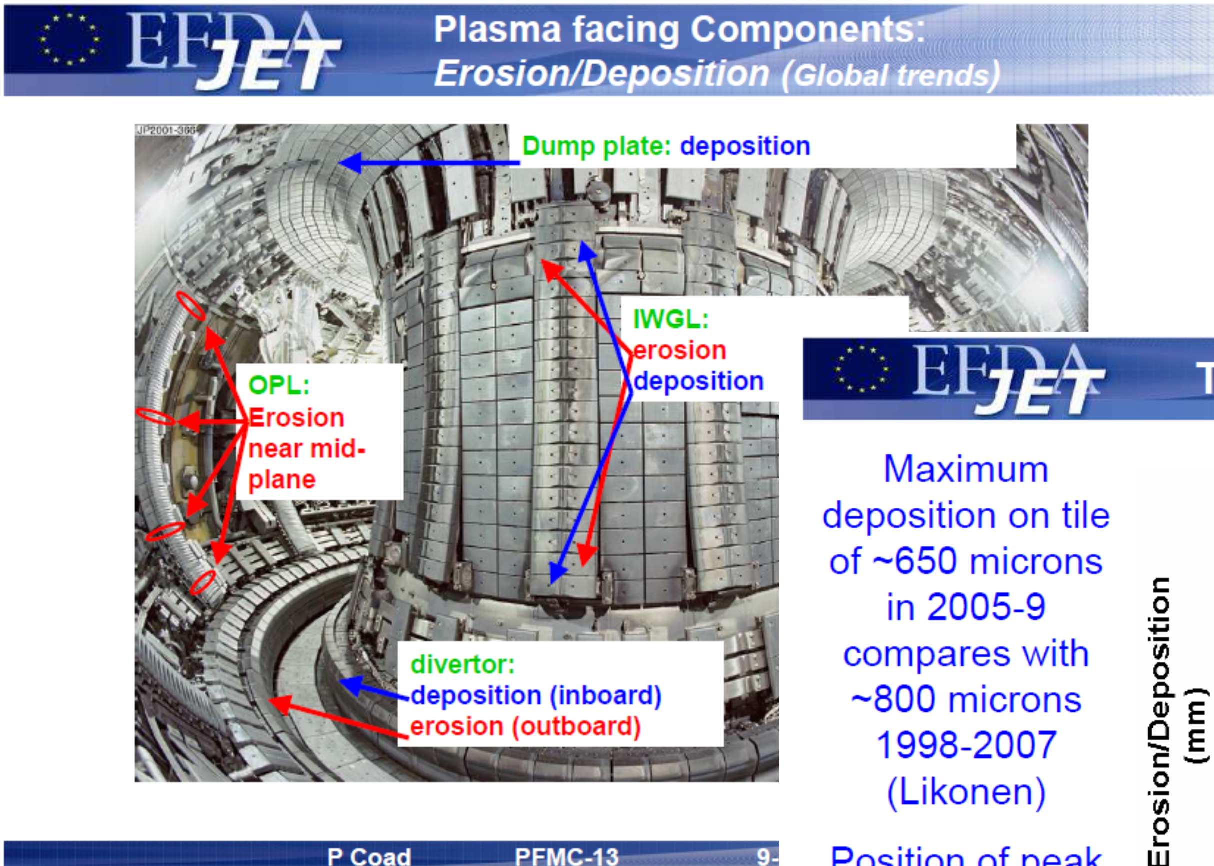


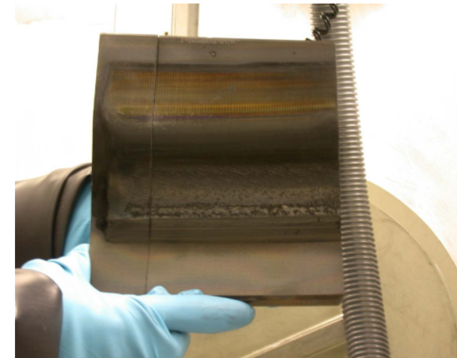
- Restart blocks interleaved with Campaign C28 blocks.
- Plan is based on five-day double-shift operation (Restart and Campaign C28-29)
- 182 experimental days in C28-C29-C30.
- Detailed plan established for C28-C29.
- C30 to be consolidated in Nov 2011 in a Programme meeting

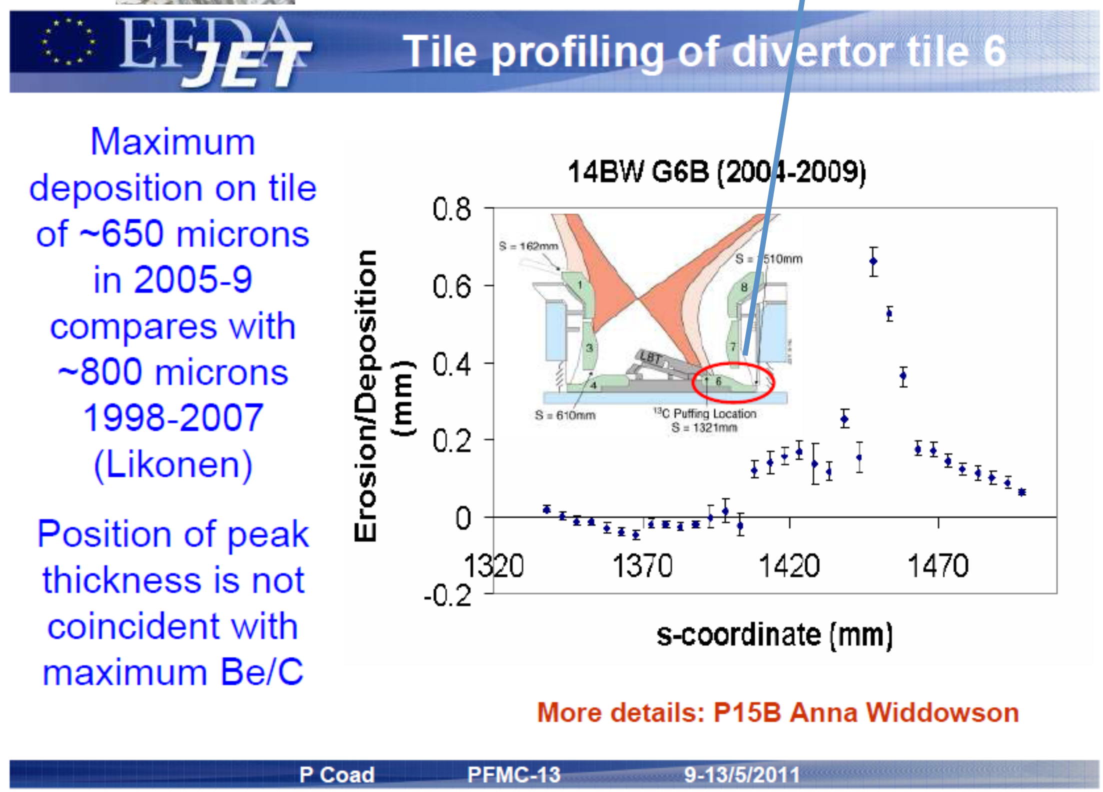
First plasma due early August

JET Post-mortem Results Following the 2007-9 Operational Period

P. Coad

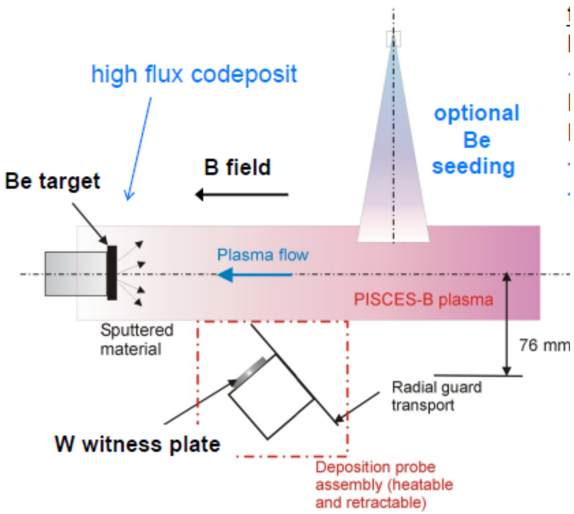


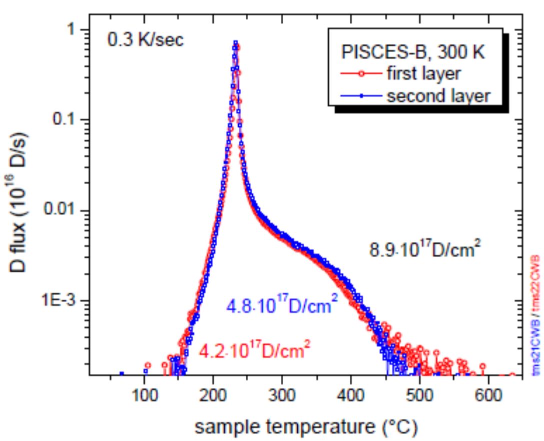




D retention in Be   T. Schwarz-Selinger from IPP

Concern about tritium inventory and question on: Do BeO layers influence D release?





PISCES-B at UCSD

⇒ magnetron sputtered Be/D films are suited as a model system to study retention and release

⇒ release of D from Be at 240°C and 350°C has a very large time constant

⇒ even a several hours bake at 240°C or 350°C does not release all D (a D/Be of 0.8% or 0.2% remains, respectively) (deeper trap sites that cannot be drained)

⇒ codeposits grown in different ways (energies, growth rates) show similar release features

⇒ the total amount of D released above 350°C is the same for all codeposits investigated, equivalent to D/Be ≈ 1%

⇒ multilayer codeposits show the same release as single layer codeposits (BeO interface is no transport barrier)

For screening tests

High heat flux test facilities

J. Linke

JÜLICH FORSCHUNGSZENTRUM

(simulation of quasi-stationary heat loads – thermal fatigue)

	facility	particle type	particle energy [keV]	beam power [kW]	max. loaded area [m²]	power density [GWm⁻²]	remarks	institute *ITER-partner*
A	TSEFEY	e^-	30	60	0.25	0.2	scanned beam, ϕ = 20 mm beryllium compatible	Efremov *RF*
A	Tsefey-M Since 2008)	e^-	40	200	1.0	1.0	scanned beam, ϕ = 8÷20 mm beryllium compatible hot water- & hot He cooling loop	Efremov *RF*
A	IDTF (ITER Divertor Test Facility)	e^-	60	800	2.25	1.0	scanned beam, ϕ = 15÷50 mm hot (ITER-like) water cooling loop	Efremov *RF*
B	JUDITH1 JUDITH2	e^-	120 30 - 60	60 200	0.01 0.25	10	irradiated samples beryllium	FZJ *EU*
C	FE 200	e^-	200	200	1.0	60	scanned beam, ϕ ≈ 2 - 3 mm hot coolant loop	CEA *EU*
D	JEBIS	e^-	100	400	0.18	2	beam sweeping ϕ ≈ 1 - 2 mm	JAEA *JA*
E	EB 1200	e^-	40	1200	0.27	10	scanned beam, ϕ ≈ 2 - 12 mm hot coolant loop	SNLA *US*
F	DATS	H^+, He^+	50	1500	0.1	0.06	2 ion sources à 0.75 MW ϕ ≈ 150 mm	JAEA *JA*
G	GLADIS	H^+	50	2200	0.3	0.05	2 ion sources à 1.1 MW ϕ ≈ 70 mm	IPP *EU*
H	MARION	H^+, He^+	60	5000	0.01	0.12	1 ion source à 5 MW ϕ ≈ 200 mm	FZJ *EU*

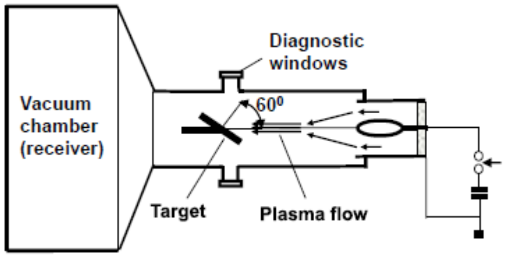
Other HHF test facilities:	IR test stands (≤ 1MW/m2), solar furnaces plasma wind tunnel (reenty vehicles), burner rigs (TBCs)

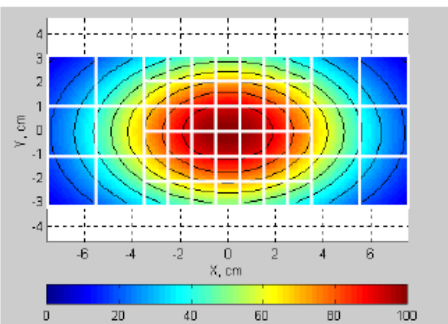
Surface Damage Effects from ELMs

J. Linke

Simulation of short transient heat pulses

Quasi Stationary Plasma Accelerator (QSPA)





The energy density distribution on W surface,%

QSPA plasma parameters (ELMs):

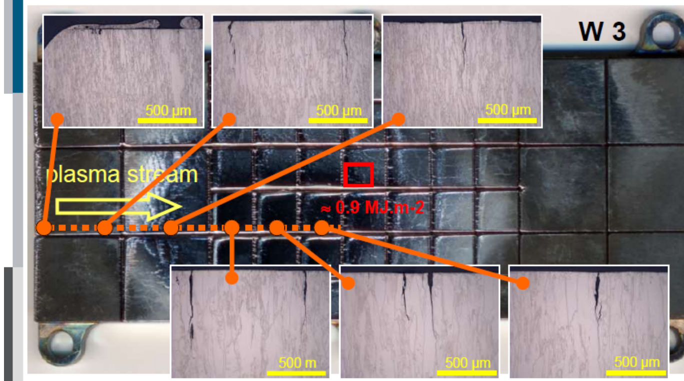
• Heat load	0.5 – 2 MJ/m^2
• Pulse duration	0.1 – 0.6 ms
• Plasma diameter	5 cm
• Magnetic field	0 T
• Ion impact energy	≤ 0.1 keV
• Electron temperature	< 10 eV
• Plasma density	≤ 10^{22} m^{-3}

Source: A. Zhitlukhin, 17th PSI, Hefei, 2006

QSPA-Be facility

For Be testing

Crack formation on tungsten in QSPA



@ 0.9 MJ/m^2, 100 pulses

Bridging of gaps due to melt motion

100 shots @ E = 1.6 MJ/m^2

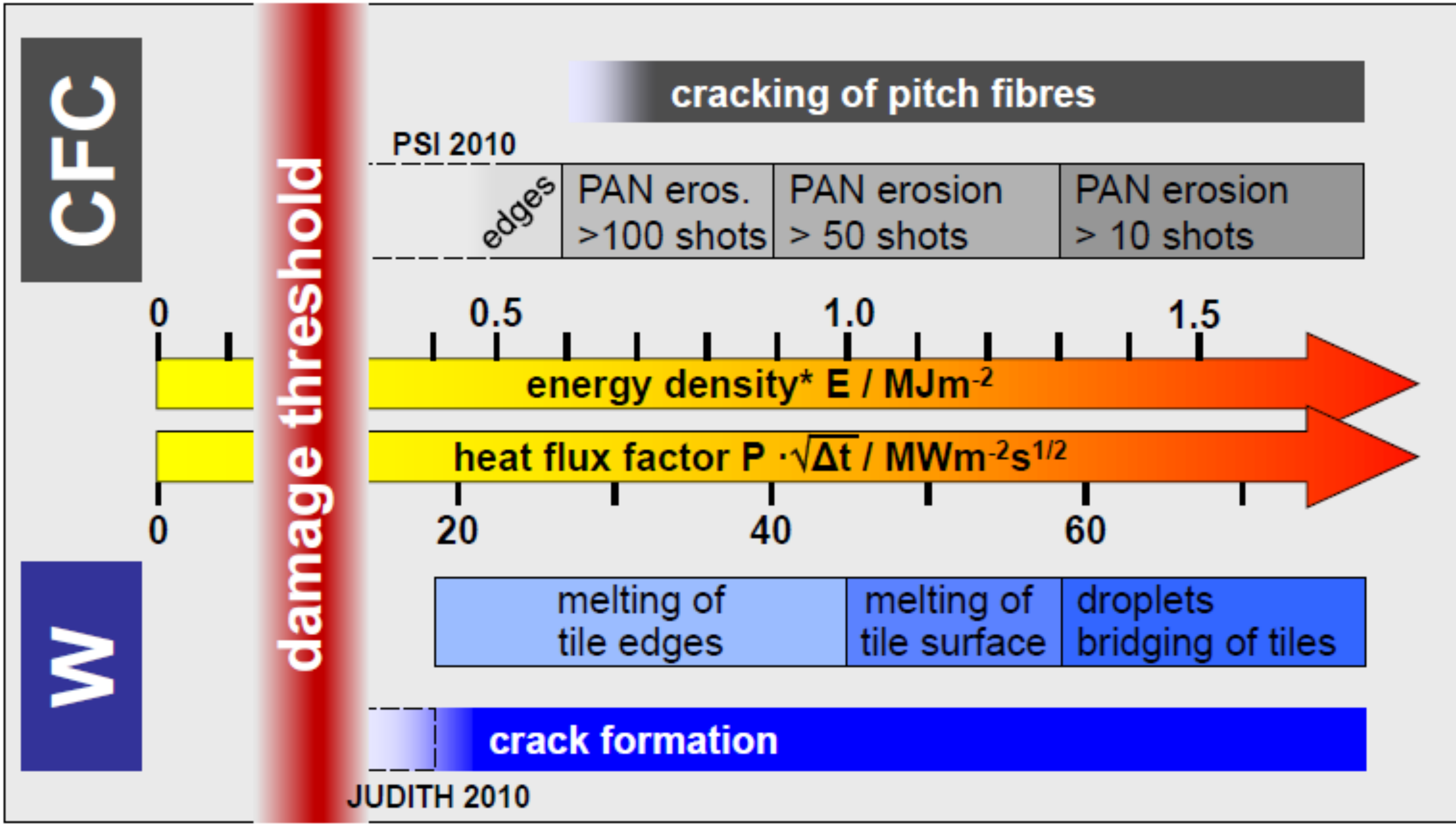
Source: A. Zhitlukhin et al., SRC RF TRINITI, Troitsk

@ 1.6 MJ/m^2, 100 pulses

Troitsk Institute for Innovation and Fusion Research

ITER: 0.5 MJ/m^2,
0.2-0.5 ms, ≥ 1 Hz

Threshold values for ELM loads



* Δt = 500 µs
T_0 = 500°C
CFC: NB31
W: forged rod material

source: PSI 2006

Testing of Plasma Facing Materials and Components

J. Linke

Summary

Materials characterization

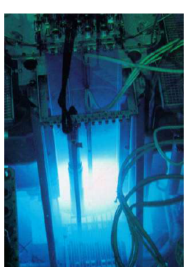
- an extensive data base is required including microstructure and all physical properties (mechanical, thermal, electrical, optical etc.)
- these parameters are required for monolithic materials, coatings and interlayers for a wide temperature range & different material treatment

Thermal fatigue and thermal shock

- technical solutions for cyclic thermal loads up to ~20 MWm^{-2} are available (CFC- or W-monoblocks represent a very robust design solution)
- off-normal events such as VDEs or disruptions result cause damage (melting, crack formation, ...) – effect of ELMs needs further analyses
- dust formation is a serious safety issue (codeposition of tritium, toxic Be dust, activated tungsten particles)

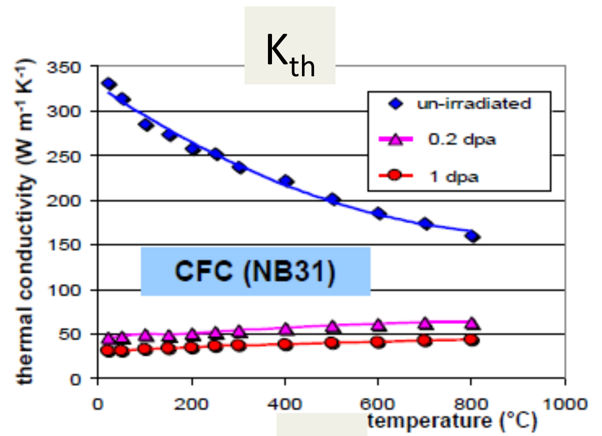
Material degradation by energetic neutrons

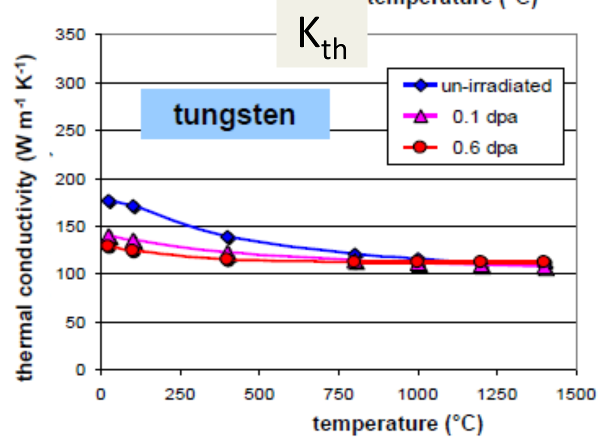
- the thermal conductivity is decreased significantly (e.g. graphite / CFC)
- the surface temperature of carbon based high heat flux components is significantly increased after neutron irradiation



High Flux Reactor (HFR)
Petten, The Netherlands

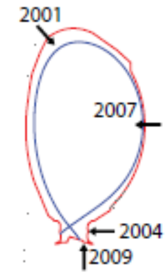
Neutron induced effects:
- Activation...Co, Ag
- Transmutation..Re, Cd, He...etc
- Degradation...K_{th}, hardening, embrittlement





^{13}C Experiment in JET MKII-HD Divertor

J. Likonen



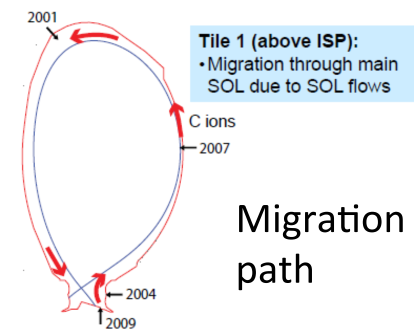
^{13}C amount (% of puffed amount):

	2001	2004		2007	2009
Inner div.	45	3.2	2.9	7.7	1.6
Floor	0.9	6.3	7.5	5.6	8.8
Outer div.	0.4	17	16.4	5.5	4
Main wall			2.7	4.1	0.4
Pumped amount	n/a	n/a	n/a	n/a	33
Total	46.2	26.5	29.4	22.9	14.8

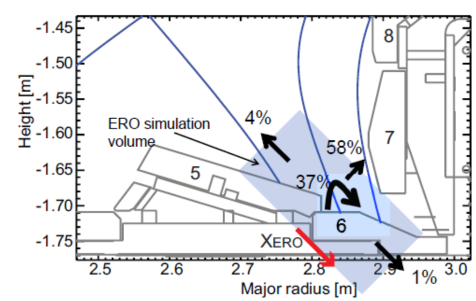
- More balanced pattern in 2004, 2007 and 2009 experiments than in 2001
- This is most likely due to longer migration path in SOL
- "Missing" ^{13}C possibly in gaps and shadowed regions
- In 2009 ~ 33 % of puffed ^{13}C pumped instantly by cryopump
- This 33% for pumped ^{13}C amount is perhaps upper limit

- Full poloidal set of divertor tiles analysed with RBS and SIMS
- ^{13}C deposited mainly near puffing location on Tile 6, and at outer divertor (bottom of Tile 7 and top of Tile 8)
- Main wall is important source for long-term deposition on inner divertor
- ~1/3 of puffed ^{13}C pumped instantly by cryopump in 2009
- ^{13}C deposition on C/W surfaces similar
- Completion of tile analysis (Tile 5, Tiles 6, main wall…)

- Simulation results (EDGE2D, DIVIMP, ERO) are preliminary
- Migration pathways identified with EDGE2D and ERO calculations
- Qualitative features of global and local ^{13}C migration reproduced by EDGE2D, DIVIMP and ERO codes but quantitative not
- More realistic grid and plasma background required, re-erosion/re-deposition phenomena have to be included, scanning of parameters (plasma parameters, diffusion coefficient…)



Migration path



ERO simulation

Analysis of Structural Changes and High-Heat-Flux Tests on Pre-damaged Tungsten from Tokamak Melt Experiments

Power handling with a metal wall is a severe issue
Resolidified materials decrease tolerable limits

Melt motion significantly worsens the situations
Forces include pressure and jxB (melt thickness , timescale)

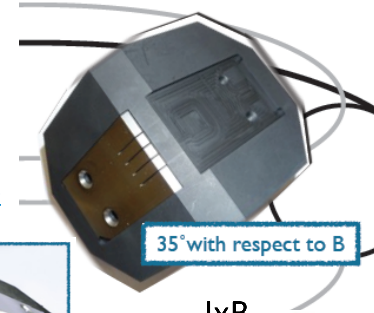
Material structure departs strongly from design values
Additional transient heat flux is less tolerable (brittle)

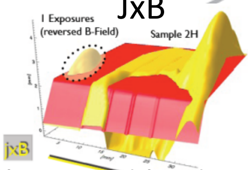
Material suffers severly and material loss can significantly impact machine and plasma operation

‚Real' melt damage and modeling confirm experimental results

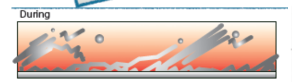
Do we need new Concepts when going beyond ITER ?

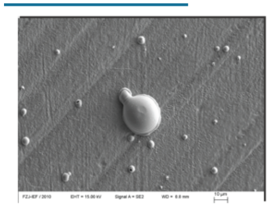
TEXTOR
Typical
~20 mW/m^2





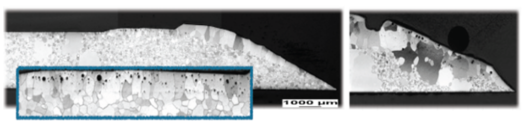
Melting and bridging





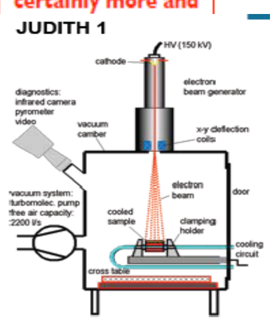
Local W source consistent with evaporation

Local boiling may explain Spraying



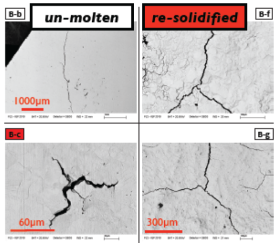
Melted layer with Bubble like structure

certainly more and



Transient
E-beam test

Exposed crack
Under transients
Seems more benevolent
yet much more brittle

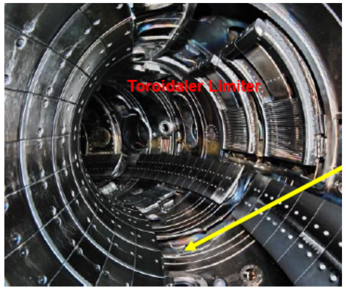


Cracks Formed
Transient heat flux
@1.13 GW/m^2
> 1 ms pulse

L.W. Coenen

TEXTOR was able to Demonstrate the in-situ RF Coating of W onto a Large Area Graphite Tiles with WF_6 Achieving a Thickness of 10-25 Microns (V. Phillips)

Goal: Develop an in situ method to deposit W- coatings on the first wall of fusion devices



W layers have been deposited on graphite by plasma deposition in WF_6 and H_2

Layers with sufficient purity and very low amount of Fluor have been deposited with good adhesion on graphite and promising thermal shock behaviour

Injection of smaller amount of WF6 in running TEXTOR shots has resulted in local deposition of pure W layers

Increased Fluor plasma contamination disappeared in less then 20 shots

In a new RF deposition arrangement, local deposition of an C film was achieved with no deposition on the rest of the wall

RF plasma deposition of W layers with DC ion acceleration on graphite appears a promising technique for in situ local W coating of wall tiles

Further optimisation ongoing

Preparation for TEXTOR W coating ongoing

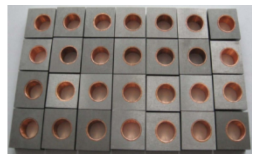
Large scale test facility
One octant of TEXTOR

EAST has a Very Aggressive Chamber Wall Program

Simplified schedule

HT-7/ EAST

2011	2012	2013	2014	2015
Structure Design				
	Technology optimization of PFCs			
	Mockup testing HHF & EAST			
		Manufacturing & Assembly (lower divertor)		
			First plasma campaign	
			Assembly (upper divertor)	
				Second plasma campaign
				Assembly (FW)



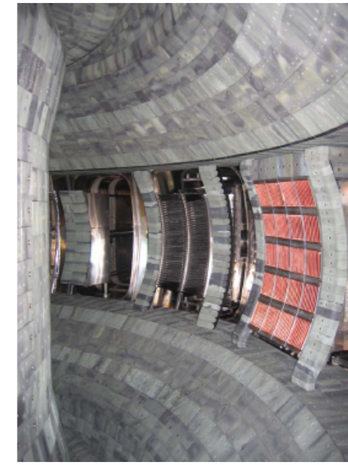
Key controlling factors

- Technology optimization

  two types of W blocks welding to CuCrZr heat sink, and VPS-W coating PFCs

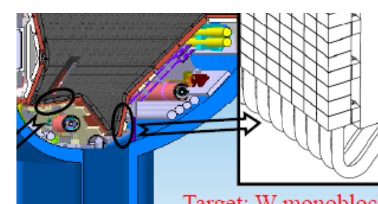
- Plasma optimization

  plasma heating and control, H-mode (type I ELMs) and divertor physics



- VPS-W/Cu PFMC
- Monoblock and Flat-type W/Cu-PFMC

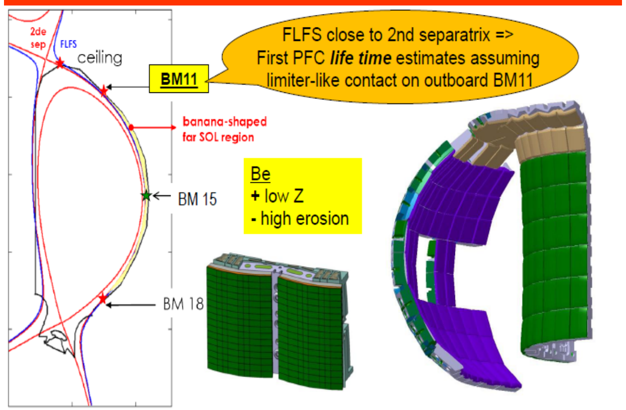
All-W wall actively cooled



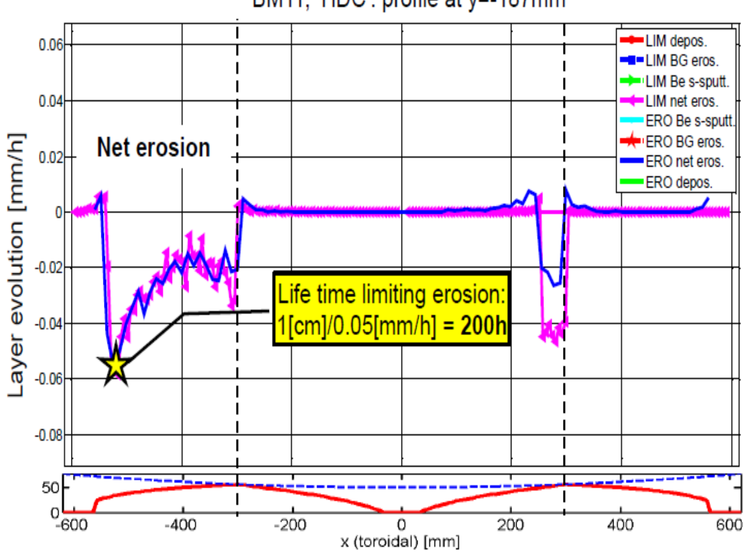
ITER Be Wall Erosion/deposition Modeling

D. Borodin

Aim – predictive modelling of ITER, including first wall life time



Code	ERO	LIM
Type	Monte-Carlo impurity tracing (BG plasma import)	
BM implementation	Shape, shadowing, plasma parameter, etc. - SIMILAR	
Geometry	3D	2D
Test-particle tracing	resolving gyro-motion	guiding centre
Intrinsic Be impurity	concentration in D^+ flux	possible
Collisions with surface	resolved angle and energy	sheath potential
Multiple BM tiles	periodic boundary	particle "mirrors"



The net erosion in LIM and ERO is in a very good agreement

Using same input plasma parameters, shadowing geometry, Be sputtering yields, ERO (3D) in excellent agreement with LIM (2D)

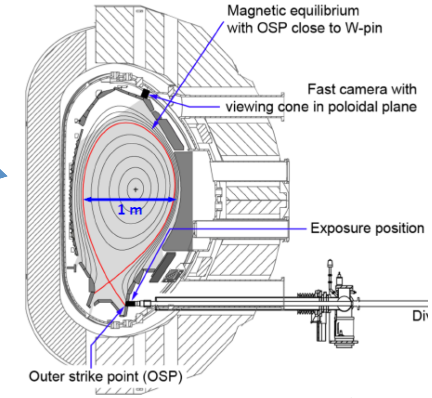
→ LIM low limit for FW panel erosion lifetime reproduced by ERO (~**1500** ITER reference Q_{DT} = 10 discharges)

Large range of erosion lifetime dominated by uncertainties in input plasma parameters and Be sputtering yields

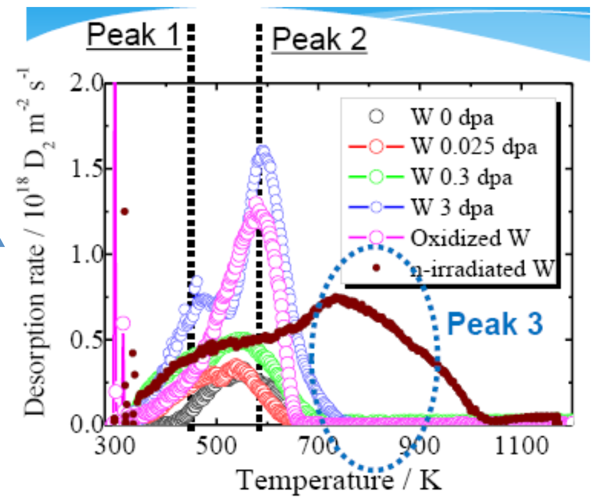
→ inclusion of different sputtering models in ERO yields variation of lower limit from 1100 ↔ 4200 reference discharges for BM11, HDC. This variation can also be influenced by other assumptions e.g. intrinsic Be impurity.

Others…Far from a Complete List

- Controlled W melting and droplet ejection using their divertor manipulator in ASDEX Upgrade; droplets can survive travelling several m toroidally
- D. Rudakov showed arcing occurred at different surfaces of DIII-D, but the estimated amount of eroded material is not a significant contributor to the total erosion of surface material
- US-Japan TITAN collaboration compared Deuterium retention between ion-irradiated and n-irradiated tungsten up to 0.025 dpa; surface morphology was clearly different therefore D trapping and desorption mechanisms would be different …by Y. Oya
- Lehnen of Julich reviewed different impacts of disruption loads on ITER PFC components and found that the runaway electrons constitute the most critical load and that the damage could reach the Be-Cu interface
- On the W-fuzz due to helium damage, S. Krasheninnikov of UCSD gave a credible visco-elastic model with helium getting into the W-fuzz/hair, causing upward growth of the hair
- W. Wampler's C-13 experiment in DIII-D



K. Krieger



D_2 TDS spectrum for damaged W