ORNL, Oak Ridge, Tennessee
March 21-23, 2005

Plasma Wall Interactions in ITER

G. Federici,
on behalf of ITER International Team, Garching

- PFC material choice
- Power deposition and erosion during ELMs and disruptions
- Tritium inventory and control
- Mixed-materials effects
- Use of tungsten PFCs
- PFC design and operation strategy / Risks
- R&D Needs
- Conclusions

- **PFC material choice**
- Power deposition and erosion during ELMs and disruptions
- Tritium inventory and control
- Mixed-materials effects
- Use of tungsten PFCs
- PFC design and operation strategy/ Risks
- R&D Needs
- Conclusions

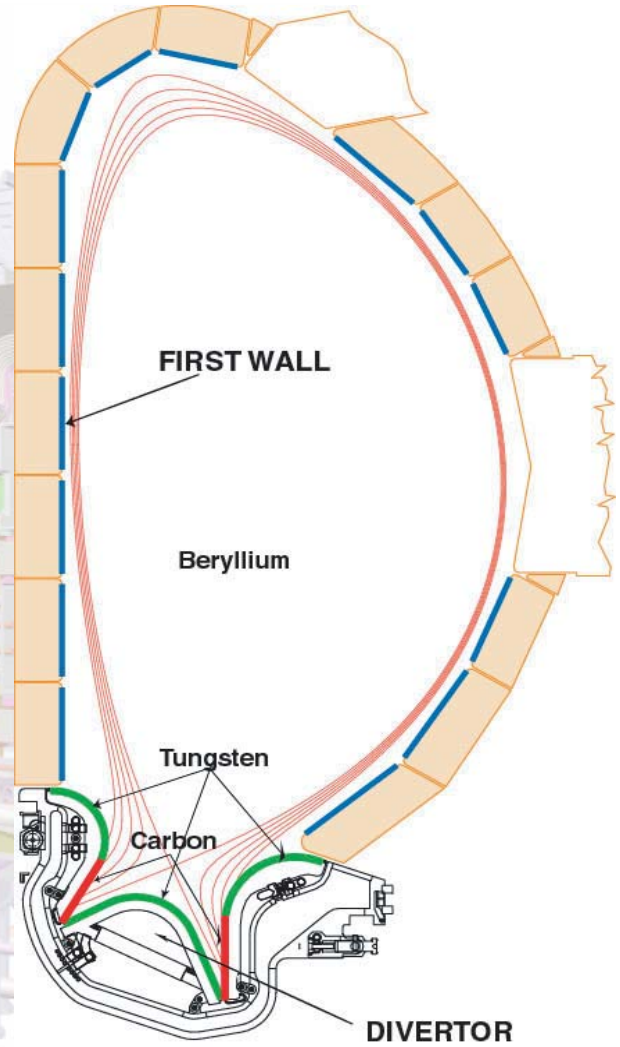
PFC material choice

Three materials to start with

- The choice of plasma facing materials is a key decision for the construction of ITER.

- It impacts design, performance, operation and availability.
  - ~ 700m^2 Be first wall :
    →low Z + Oxygen getter
  - ~ 100m^2 W Baffle/Dome :
    →low erosion, long lifetime
  - ~ 50 m^2 CFC Divertor Target
    →no melting, C good radiator

- **What are the implications of this choice?**
- **What remains to be addressed to demonstrate the validity of this choice?**
- **What are the feasibility issues and operation implications of alternatives?**



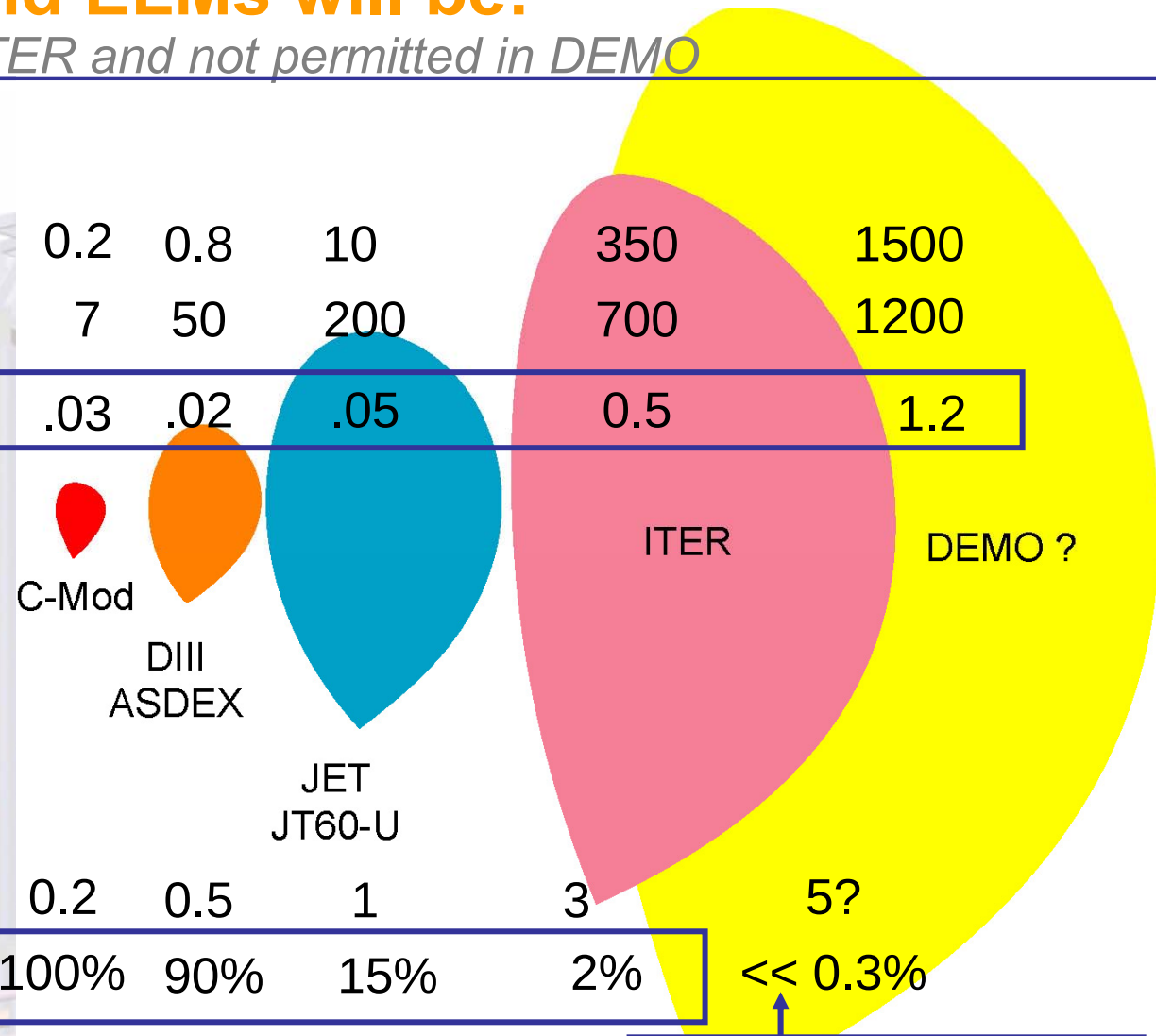
- PFC material choice
- **Power deposition and erosion during ELMs and disruptions**
- Tritium inventory and control
- Mixed-materials effects
- Use of tungsten PFCs
- PFC design and operation strategy/ Risks
- R&D Needs
- Conclusions

Disruptions and ELMs will be:

strictly regulated in ITER and not permitted in DEMO

G. Matthews, PSI 2004

$W_{thermal}$ (MJ) ➔	0.2	0.8	10	350	1500
A_{plasma} (m^2) ➔	7	50	200	700	1200
$W_{thermal}/A_{plasma}$ (MJ/m^2) ➔	.03	.02	.05	0.5	1.2
$A_{divertor}$ (m^2) ➔	0.2	0.5	1	3	5?
$\Delta W_{ELM}/W_{th}$ ➔	>100%	90%	15%	2%	<< 0.3%



Perfect disruption mitigation

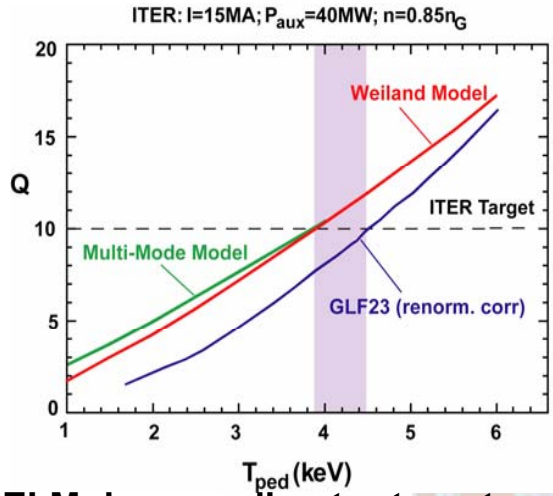
Max. ELM size - target ablation

DEMO - limit set by sputtering threshold

High pedestal pressure required for Q_{fusion}~10; implies that ELM control is required for ITER

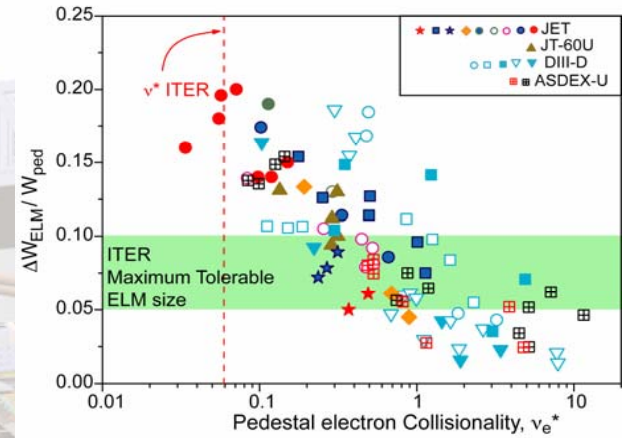
ITER

T_e^{ped} > 3.9-4.5 keV in ITER for Q>10.



ELM loss energy scales w. pedestal energy.

A. Loarte (EFDA/ Garching

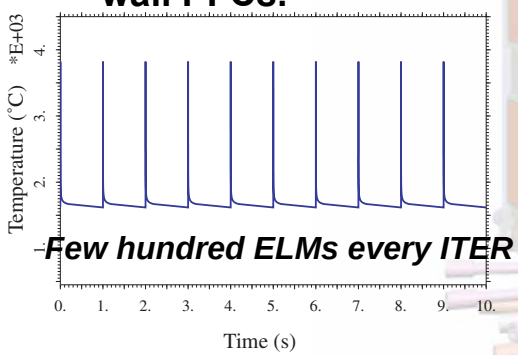


ITER:

- ***Energy: 350 MJ***
- ***Divertor wetted area: 3.1 m² (λ_{mid}~5 mm no broadening)***

→ ELM damage divertor target and, possibly, wall PFCs.

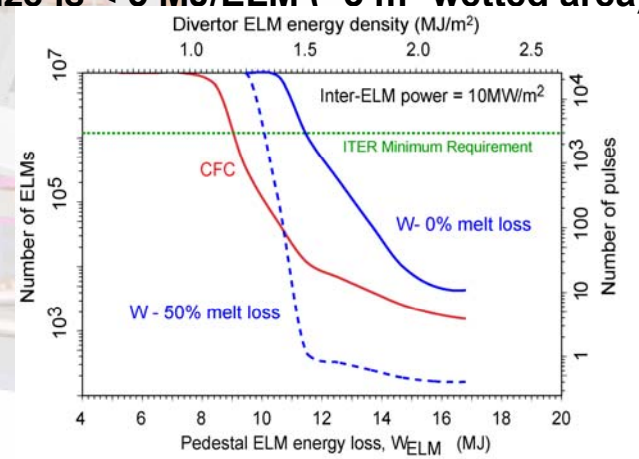
→ Database implies 5-20 MJ/ELM ITER maximum tolerable size is < 5 MJ/ELM (~3 m² wetted area).



Few hundred ELMs every ITER pulse.

Main problems arising for ITER:

- ***Divertor target erosion lifetime.***
- ***Impurity production and plasma contamination.***



G. Federici, Plasma Phys. Control. Fusion 45 (2003) 1523

- **We cannot operate routinely with Elms of ~1MJ/m2**
- **We need to suppress/mitigate large Elms in ITER**
- **Lifetime of a W target is uncertain due to melt layer losses.**

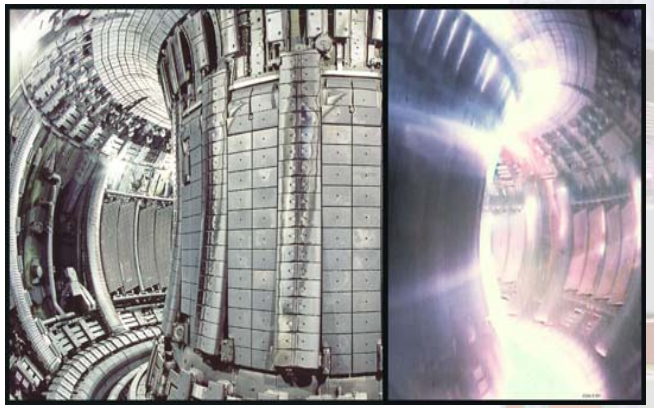
Effects of Type I ELMs at the main chamber wall

Herrmann A. et al. Plasma Phys. Control. Fusion 46 (2004) 971.

ITER

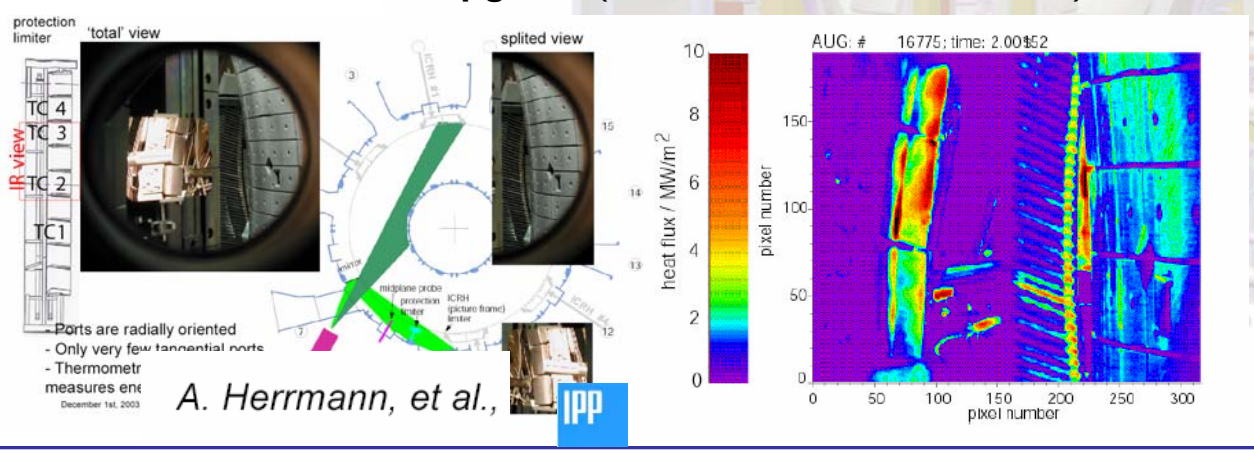
$\Delta W_{ELM}^{div} \sim 50 - 80\ \%$ of ΔW_{ELM}^{dia}

=> A fraction of the ELM energy reaches the main chamber wall

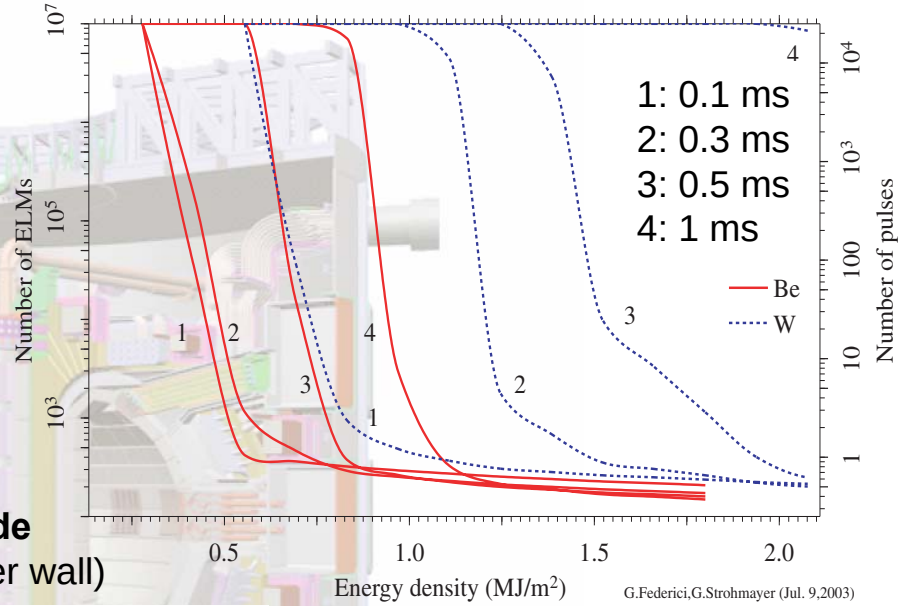


JET – Type I ELM

- **26% of ELM energy mid-plane loss is found outside divertor in ASDEX Upgrade** (14% limiters, 12% inner wall)



- **Strong implications for ITER**



Wetted area?

Protruding surfaces (e.g. start-up limiters (front surface few m²).

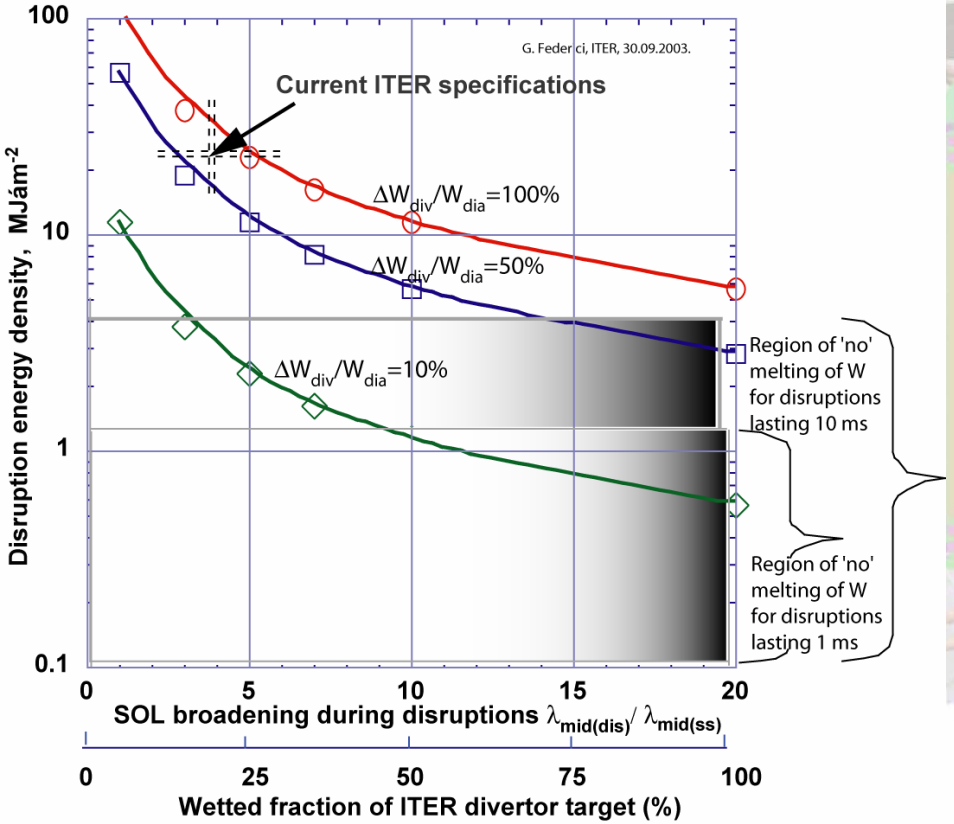
=> tolerable ~0.3-1 MJ/m² depending on material and ELM duration (if ≥ 1 ms W doesn't melt).

Energy deposition during disruptions

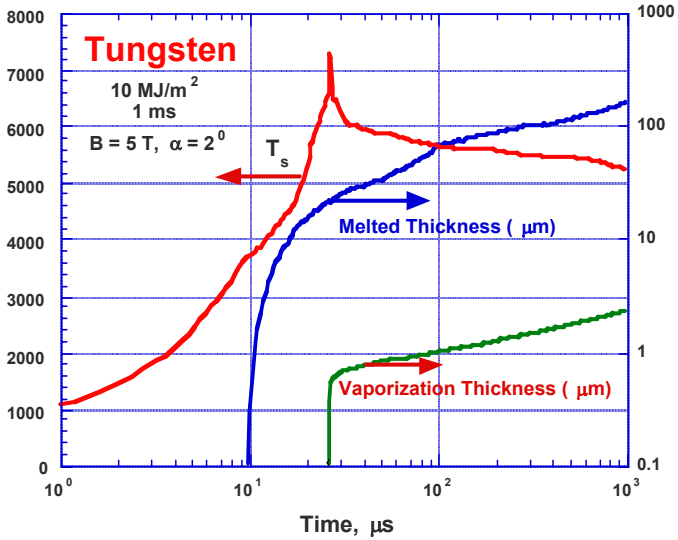
Present ITER specifications for thermal quench

ITER

- $W_{tht.q.}$ ~ 350 MJ (for Q_{DT} = 10)
- $t_{t.q.}$ ~ 1 ms
- $A_{divt.q.}$ ~ 3 $A_{divs.s.}$



A. Hassanein, J. Nucl. Mater. 273 (1999) 326.



New experimental results

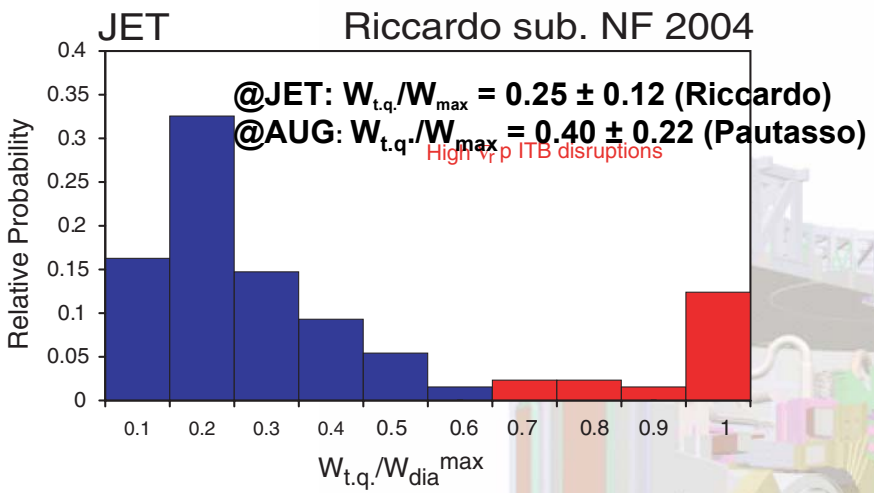
- $W_{tht.q.}$ is smaller than W_{th} with good confinement
- Only part of the energy is seen in the divertor
- Large broadening

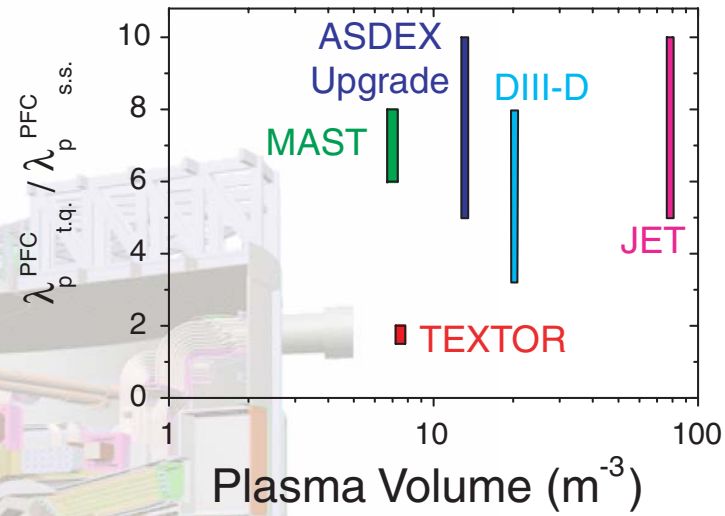
→ Implications for ITER

Energy deposition during disruptions

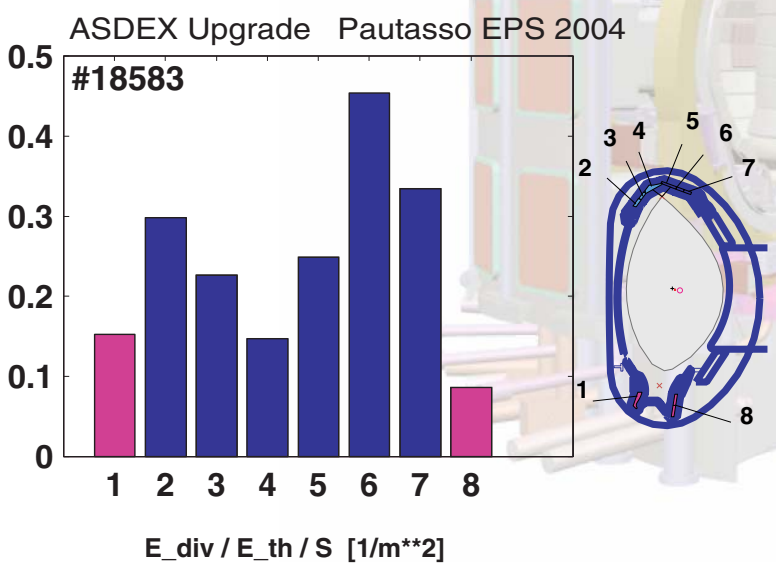
New experimental results & design implications

ITER





Good news for divertor but not for the wall!



Thermal quench energy balance

- **complex experimental evaluation**
- **Divertor toroidal symmetry @ thermal quench?**
- **$W_{mag} \rightarrow W_{div}$ @ thermal quench (AUG ~ 15%) ?**
- **$W_{t.q.} \rightarrow W_{wall}$?**
- **Modelling of IR measurements ?**

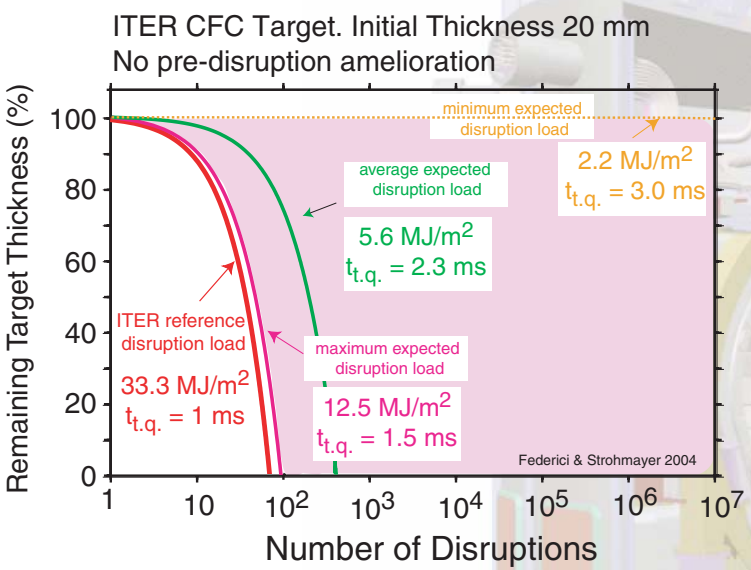
DIII-D (Whyte)	$W_{div}/W_{t.q}$
Disr. type	
VDE	0.88
Grad-P	0.73
MHD Mode	0.80
n_e-limit	0.16

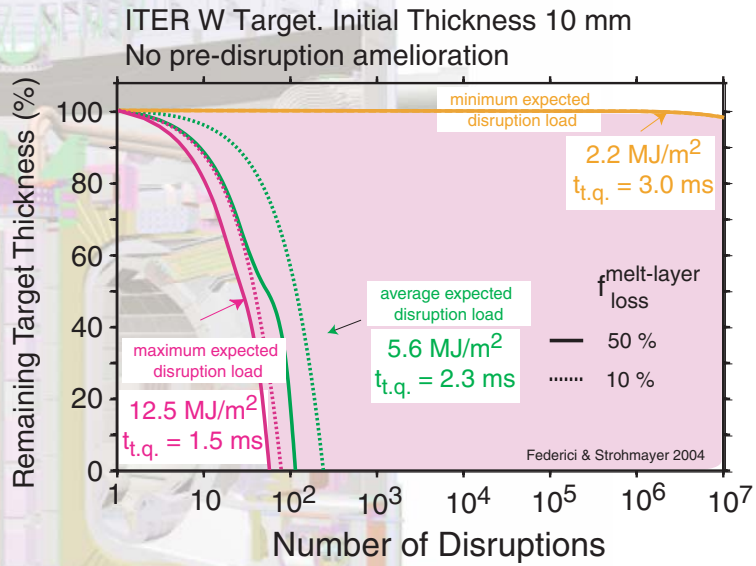
Energy deposition during disruptions

Estimates of erosion lifetime

ITER

- ITER plasma energy @ thermal quench < 0.6 * 350MJ
- Thermal quench energy flux timescales > 1.5 ms
- Divertor wetted area @ thermal quench > 18 m^2





- **Lifetime ~ 100–1000 full plasma performance disruptions (W → $f_{melt\text{-}layer\ loss}$?)**
- **Unresolved: thermal quench energy balance and divertor asymmetries →divertor flux peaking factors & fluxes to main chamber Be PFCs**
- **Needs: more disruption experiments/measurements in ITER-relevant regimes.**

Damage/erosion during ELMs

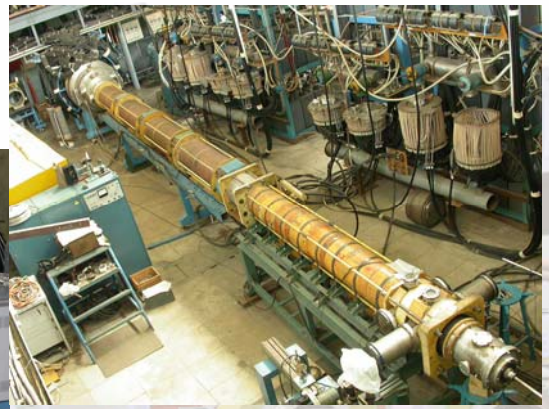
G. Federici, A. Zhilukhin et al.,
J. Nucl. Mater. 337-339 (2005) 684.

ITER

Experiments in Russian plasma guns

Russian plasma guns located in SRC RF TRINITI

Mk-200UG

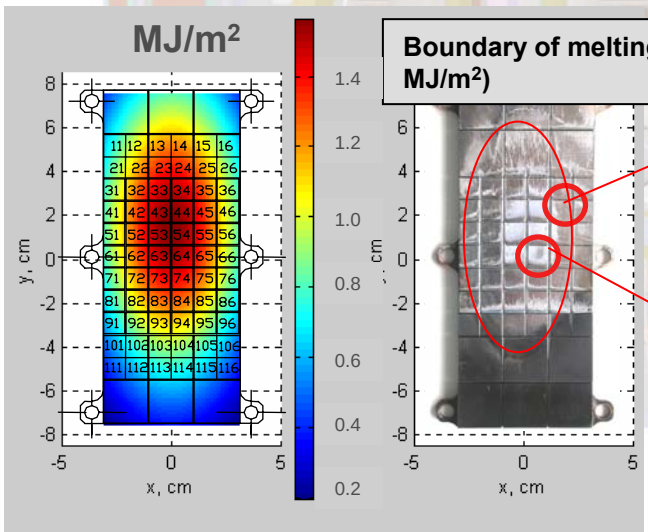


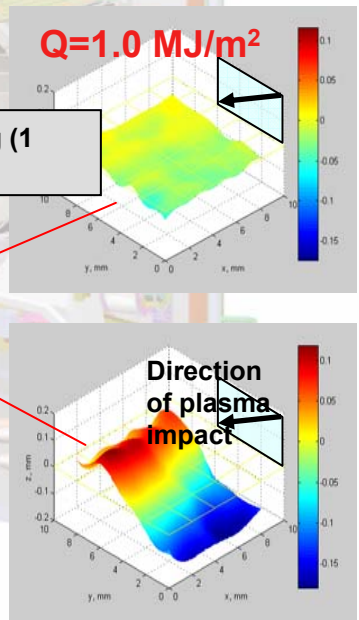
QSPA

Plasma parameters (ELMs):

	QSPA	MK200UG
Heat load (MJ/m^2)	**0.5-2**	**0.2–1**
Pulse duration (ms)	**0.1–0.6**	**0.04–0.06**
Plasma stream ϕ (cm)	5	6-10
Magnetic field (T)	**0**	**0.5-1.2**
Ion impact energy (keV)	**<0.1**	**1.5**
Electron temp. (eV)	**<10**	**100-200**
Plasma density (m^{-3})	< 10^{22}	(2-5)x10^{21}

150x60x10 mm^3





- **W melt layer forms at the edge and shifts within an individual macro-brush element along plasma stream direction but does not close the gaps;**
- **average erosion did not exceed 0.2 µm/shot (evaporation, no melt losses);**
- **maximal roughness of surface reaches 0.3 mm after 100 pulses @ ~1.5 MJ/m^2;**
- **modelling prediction in good agreement with experimental results.**

- PFC material choice
- Power deposition and erosion during ELMs and disruptions
- **Tritium inventory and control**
- Mixed materials effects
- Use of tungsten PFCs
- PFC design and operation strategy/ Risks
- R&D Needs
- Conclusions

→Talks of Charles Skinner (Wednesday), Tony Haasz (Tuesday)

Predictions of tritium co-deposition in ITER

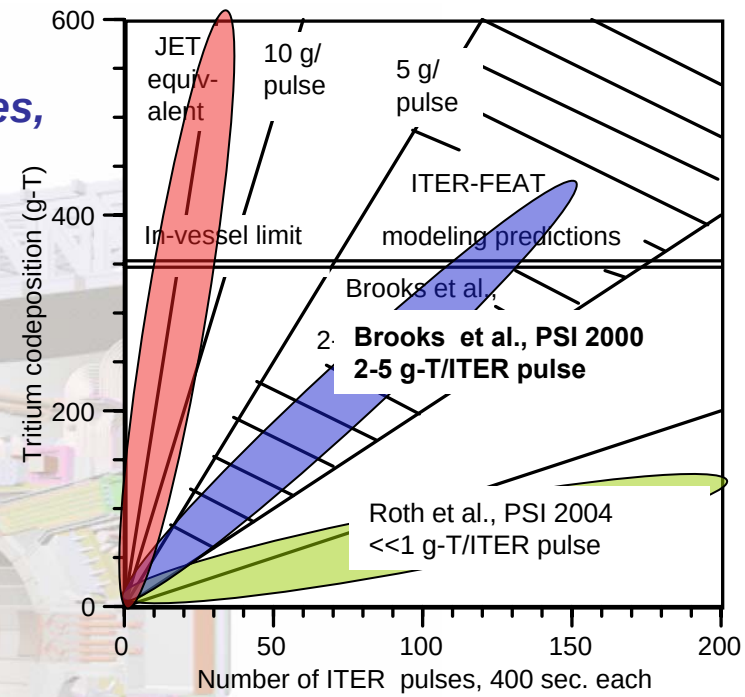
Details discussed in the talk of Charles Skinner

ITER

- Predictions still uncertain due to
  - ***chemical erosion yields at high temp. and fluxes,***
  - ***effects of type I ELMs (ablation)***
  - ***effects of gaps,***
  - ***effects of mixed materials,***
  - ***lack of code validation in detached plasma.***

Note:

- *ITER predicted rate is 10x less than that in JET*
- *Model underestimates JET retention by factor x40.*



- **T issues will be heavily scrutinised by licensing authorities.**
- **Scale-up of removal rate required is 10^4.**
- **Potential options for T removal techniques for ITER.**

1) Remove whole co-deposit by:
   - ***oxidation (maybe aided by RF)***
   - ***ablation with pulsed energy (laser or flashlamp).***
2) Release T by breaking C:T chemical bond:
   - ***Isotope exchange***
   - ***Heating to high temperatures e.g. by laser, or ...***

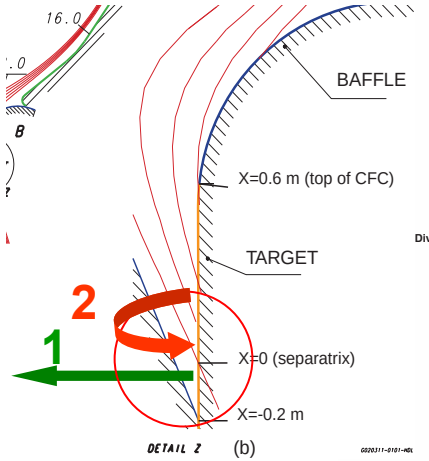
- Constraints:
  - 6.1 Tesla field at inner divertor
  - 10,000 Gy/hr gamma field from activation, 3 h after shutdown.
  - Access difficult, especially to hidden areas

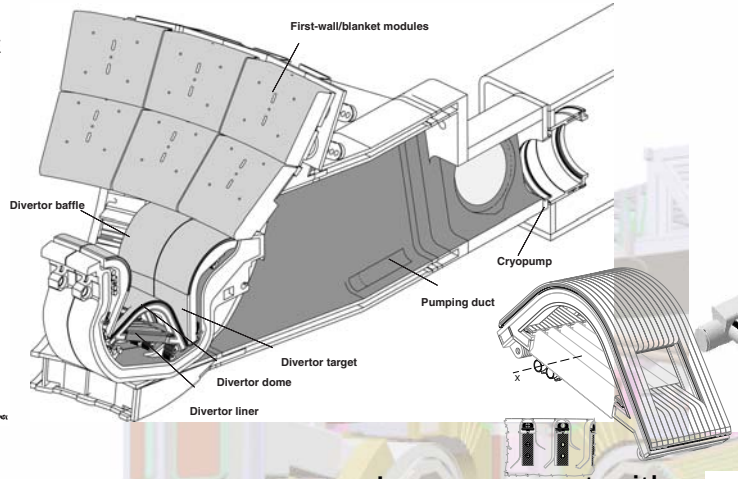
C -deposition in remote areas

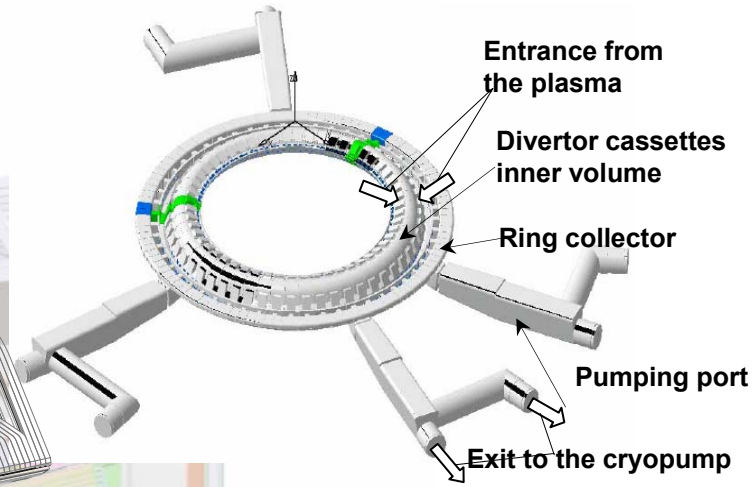
Divertor structures/ tile gaps/ ducts

G. Federici et al.,
J. Nucl. Mater. 337-339 (2005) 40.

ITER







to private region

back on plate

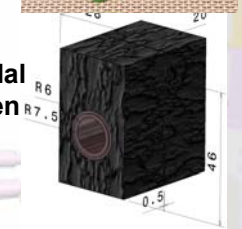
In agreement with experimental findings one finds that:

- ✓ *majority of radical species entering the divertor private regions will stick on the nearby surfaces;*
- ✓ *only a small fraction of low sticking probability species can enter the pumping duct;*
- ✓ *only species with very low sticking probabilities ($\leq 10^{-3}$)reach the pump.*

Retention in the gaps:

- **Ignored by modelling**

Zero poloidal gap between CFC tiles



0.5mm toroidal gap between CFC elements

10mm toroidal gap between cassettes or for diagnostic cassettes ~ 30mm toroidal gap

- **~ 54 cassettes.**
- **Outer VT: 22*-23 poloidal elements;**
- **Inner VT: 16*-17 elements;**
  - **In diagnostic cassettes one poloidal element is removed from each target to allow integration of Langmuir probes, microhorn horns etc.**
- **Total surface area of gaps > 140 m^2**
  - **35 g-T/µm (assuming uniform film and 0.5 T/C)**
  - **We do not need much to reach inventory limit**

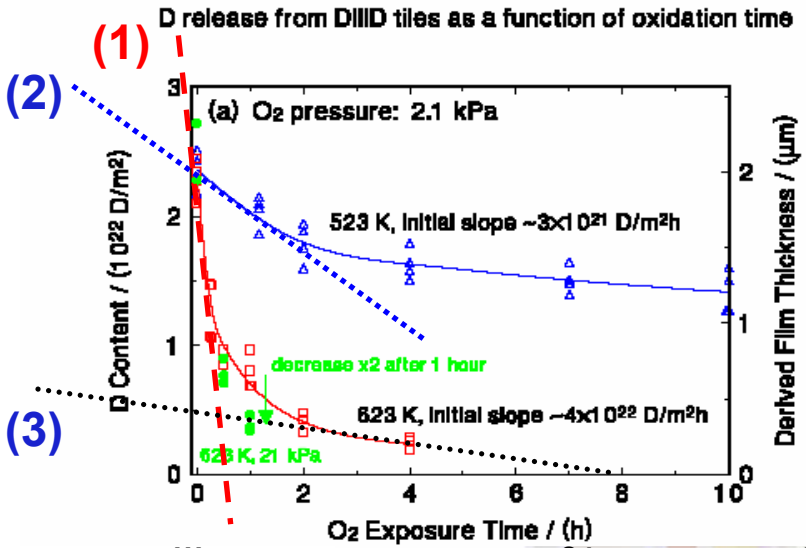
Tritium removal from co-deposited films

Required T recovery efficiency >>90%

ITER

→Talk of Charles Skinner

J.W. Davis and A.A. Haasz, J. Nucl. Mater. 266-269 (1999) 478.

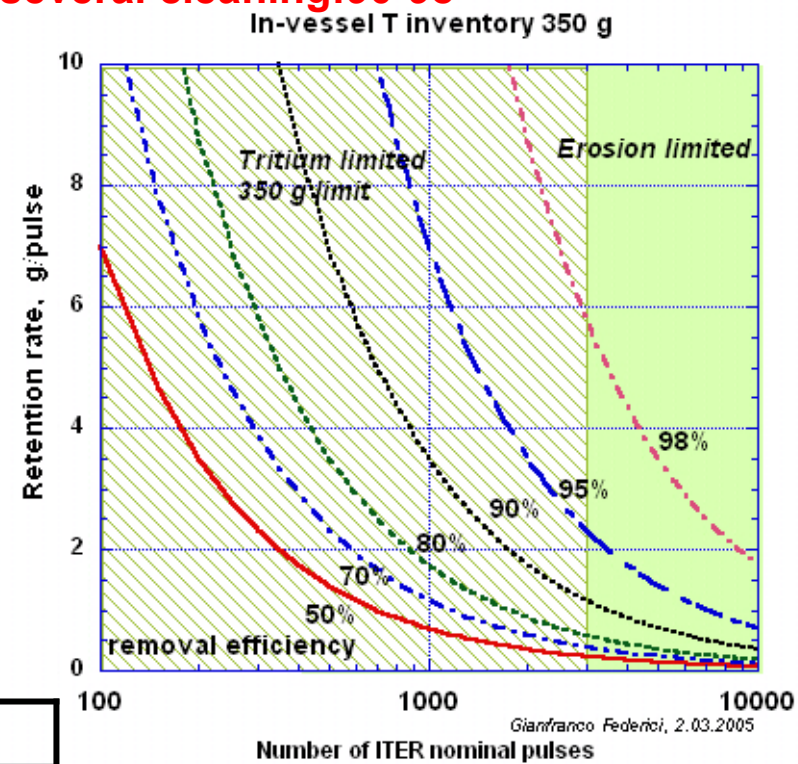


- Fuelling rate=50 Pa m³/s ~48 g-T/ 400 s
- 20 pulses per day
- Tritium limit (mobilisable) ~350g

Year	4	5	6	7	8	9	10	
HH								
DD	2000	1300		500				
DT	1	750	1000	1500	2500	3000	3000	11751

- **The fraction of T co-deposit left over after each cleaning plays a major role**

→ Required T recovery efficiency for several cleaning:90-98



- **Equivalent no. nominal pulses (400 s) during the first 10 years of operation**

- PFC material choice
- Power deposition and erosion during ELMs and disruptions
- Tritium inventory and control
- **Mixed-materials effects**
- Use of tungsten PFCs

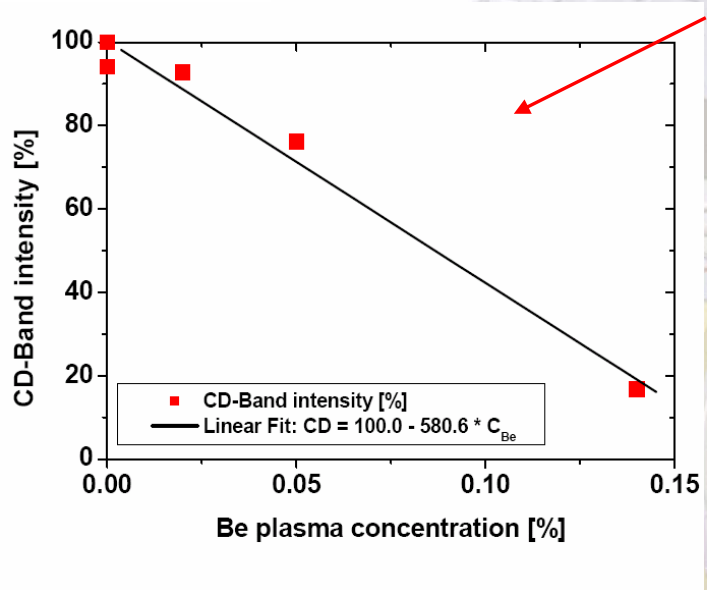
  →Talk of Russ Doerner (Tuesday)
- PFC design and operation strategy/ Risks
- R&D Needs
- Conclusions

Mixed-material effects

Erosion of C by D/Be in PISCES-B

ITER

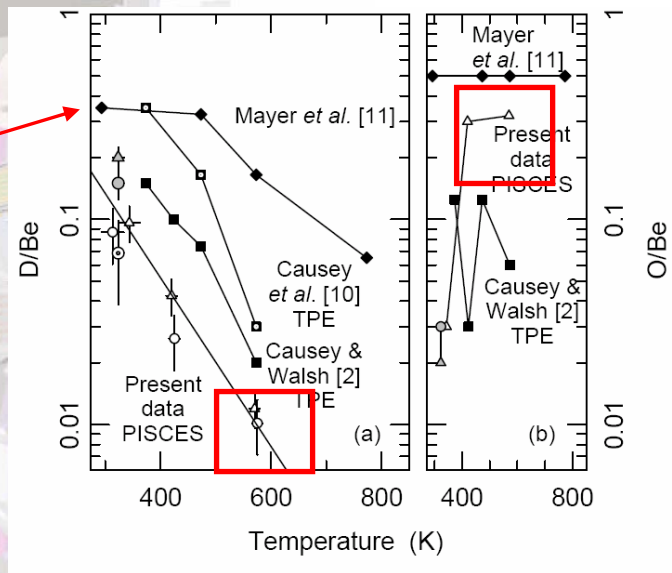
A small Be impurity concentration (~0.2%) in the plasma causes Be surface layers to form, suppressing both chemical and physical erosion of carbon



Be suppresses C chemical erosion

D concentration in Be/O layers very low at T ~ 600 K

R.P. Doerner et al., *Carbon Erosion Mitigation by Beryllium Layer Formation in ITER*, IT/P3-18, FEC 2004 IAEA, Vilamoura, Portugal, Nov. 2004.



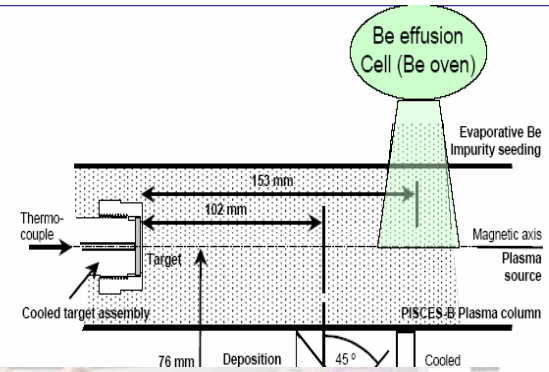
Still several open questions remain:

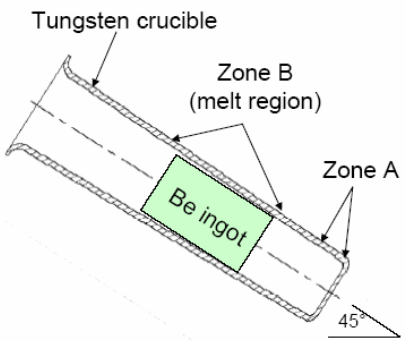
- **Be-deposition/Be-diffusion (high T_{surf})** ⇨ ***A tokamak demonstration under ITER relevant conditions* is urgently needed**
- **Behaviour for (Ne, Ar) seeding**
- **Power handling/transient power loads**
- **In/out coverage asymmetries ?**

Mixed-material effects

Observations of Be/W interactions in PISCES-B

- **Be impurities from the ITER first wall will form Be surfaces on the tungsten baffle plates.**
- **Be can alloy with W**
- **Resulting alloys ($Be_{22}W$ and $B_{12}W$) have lower melting temperatures (~1500C)**

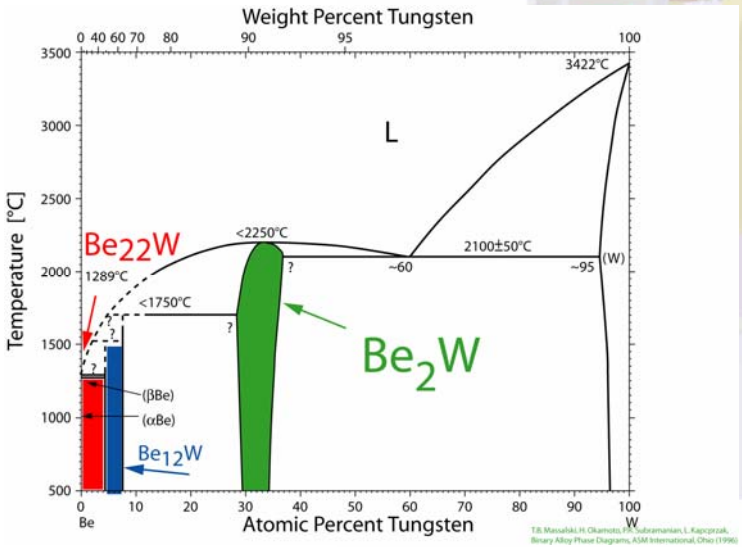


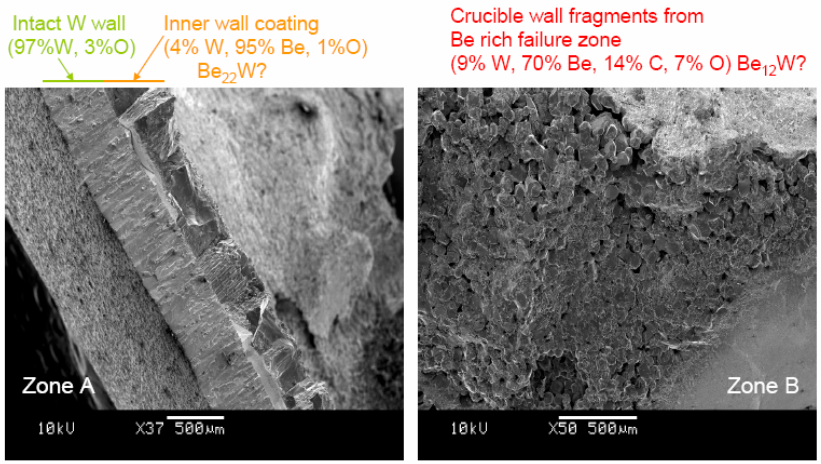


The W crucible cracked.

- **Be oven temperature never exceeded 1550C (avg. 1200C).**
- **Lifetime of crucible 100 hrs.**
- **Crucible thickness ~1 mm**

A. Wiltner and Ch. Linsmeier, PSI 16, 2004



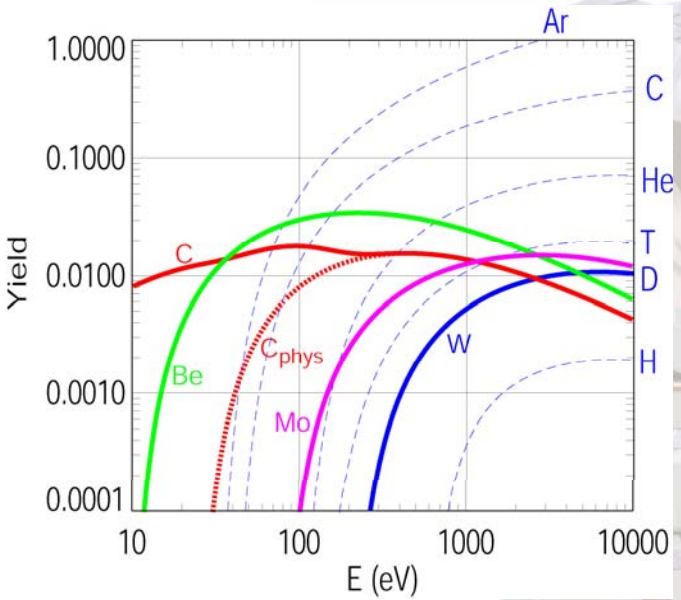


- PFC material choice
- Power deposition and erosion during ELMs and disruptions
- Tritium inventory and control
- Mixed-materials effects
- **Use of tungsten PFCs**
- PFC design and operation strategy/ Risks
- R&D Needs
- Conclusions

What are the main problems with High-Z?

Plasma contamination and melting

High sputter threshold for high Z ➔ low erosion



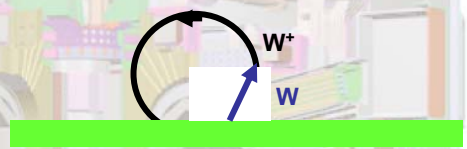
Control of steady-state and transient power loads essential in ITER

G. Matthews, PSI 2004

Ion impact energy

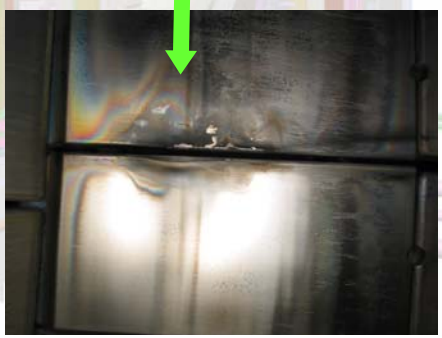
$$E_i = 3ZT_e + 2T_i$$

Prompt re-deposition also helps high Z at divertor



ASDEX, C-Mod

Melting



ASDEX Upgrade W Upper divertor tile

- Very low target erosion in C-Mod (Mo) ➔ 4.5mm per year (continuous) at outer strike point
- In ASDEX and C-Mod erosion dominated by impurity sputtering

C-Mod Mo divertor baffles

Use of tungsten in ITER

Prospects are good but more work is needed

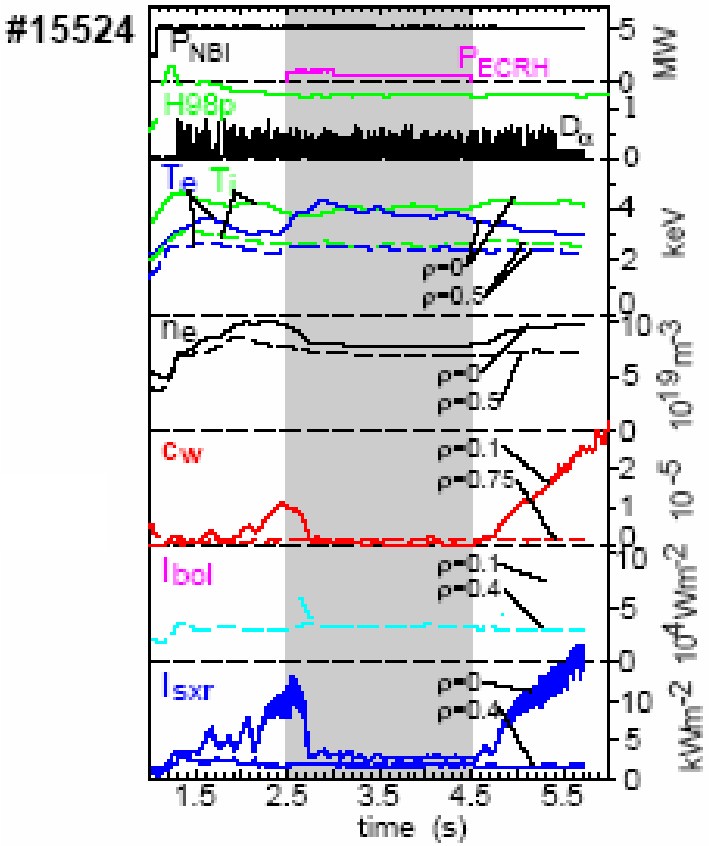
- Limited but benign divertor tokamak operational experience.
- Plasma compatibility and constraints on accessible plasma scenarios.
- Control of central impurity accumulation.
- Uncertainties of projection to ITER operation.
- Divertor detachment and control of power exhaust w/o. C-radiation.
- Enhanced impurity production from irregular/damaged surfaces and implications on plasma operation and performance.
- Damage arising from repetitive ELMs and limited disruptions. Melt layer loss behaviour ?
- Mixed material issues.
- Expand use of tungsten in present tokamaks
  - ASDEX-U full W device first possible in ’06/07
  - W divertor planned in JET
  - Some W planned for C-mod.

Operating experience in ASDEX Upgrade

Summary of results and extrapolations to ITER

ITER

- In most discharges no problem (including W divertor operation)
  - ➢USN and LSN discharges have similar W contentdivertor W source is not dominant.
- Impurity problems if:
  - *Density peaking (neoclassical impurity pinch)*
  - *Limiter operation*
  - *ELM-free phases in H mode.*
- Fast particle play important role for low field side W erosion
- Impurity seeding compatible with W-PFCs
  - *stable integrated scenario available*
- No W accumulation expected for ITER reference scenario.
- Further extension of W surfaces under way, full W device first possible in ’06/07



- PFC material choice
- Power deposition and erosion during ELMs and disruptions
- Tritium inventory and control
- Mixed-materials effects
- Use of tungsten PFCs
- **PFC design and operation strategy/ Risks**
- R&D Needs
- Conclusions

PFC design and operation strategy

Start with CFC in the divertor and Be on the wall but …..

ITER

- Test PMI issues with H/D plasmas. This implies the availability of:
  - reliable and accurate H/D-retention and erosion diagnostics and validated modelling tools (*which is a big challenge per se*), for accurate estimates of H/D retention and projections of retention during T-operation.
  - maintaining C during the DT phase, is contingent on the availability of fast and efficient methods for T-removal, which are still not available
- Start with W divertor & develop compatible operating scenarios. This implies:
  - reliable methods to control/mitigate transient events (ELMs, disruptions).
  - accepting greater restrictions on ITER operating space.

→In both cases ELMs mitigation/suppression systems, need to be developed.

- Rationale for using Be on the 1st wall hinges on the absence of intense energy deposition, and feasibility of wall exchange. However,
  - Localised transient power deposition to the main chamber wall during ELMs, disruptions is observed in existing tokamaks and in ITER would lead to damage.
  - Favourable operation with W in AUG and in JET (Be/W) and tests in C-MOD,
  - A W wall shall be considered for later reactor orientated operation.

Remaining caveats

Physics and Technology

- A change of divertor and wall material during operation requires development of new plasma scenarios (e.g., no intrinsic impurity radiation, more high Z core impurity contamination, lower tolerance to disruptions or ELMs etc.) and it is questionable how much of the experience gained in the carbon phase would still be relevant.
- Because of long lead-times (3-5 y) to manufacture a new W divertor, one must procure a full-W divertor early during ITER construction.
  - **Delaying procurement until T retention proves as large as feared and if no satisfactory removal schemes are devised would create long delays**.
- Need (probably) to remove C deposits formed in various locations during H/D and D/T phases, to minimise adverse effects, from residual C.
- If we want to retain flexibility , feasibility of wall exchange and not only divertor is essential. It is a very challenging task, and a critical technical assessment is needed.

Potential major PFC/PWI risks

Risk	Potential consequence	Mitigation techniques: Design changes /R&D
Divertor damage by large ELMs, and unmitigated disruptions.	Premature divertor failure, long down time to replace it.	Tokamak ELM research worldwide. Lifetime predictions still uncertain. Mitigation/suppression techniques.
Damage of Be limiters, main chamber wall.	Longer down time for ITER wall replacement, success questionable.	Be/W planned on JET 2009. Modeling still immature. Replace Be wall with W everywhere. But feasibility of wall exchange need to be demonstrated.
Tritium removal unsuccessful	ITER limited to ~ few hundred pulses	Abandon carbon and install a full W divertor.
Large W influx to core plasma	Failure to achieve Q = 10	ASDEX Upgrade investigates successful control using central heating, full W wall by 2007; Be/W planned on JET 2009.
Unforseen mixed materials effects	PFC failure, long down time.	Be/W planned on JET 2009. More R&D in PISCES. Replace a Be wall and install W clad ding everywhere.
Radioactive/toxic/explosive dust particles accumulate	Measure dust inventory and remove it before restarting operation	Measurement and removal techniques still need to be developed.

- Background / material choice / operation strategy
- Power deposition of transients energy during ELMs and disruptions
- Tritium inventory and control
- Mixed-materials effects
- Use of tungsten PFCs
- **R&D Needs**
- Conclusions

Main material/PWI issues for ITER

Still some time to go (not much!) and you know what we need!

- The high priority issues for ITER that must be addressed in tokamak experiments or laboratory simulations aided by modelling are:
    - **tritium co-deposition and effective control of inventory**
    - **plasma operation with a beryllium first wall**
    - **the use of tungsten as a plasma facing component**
    - **the effects of material mixtures**
    - **the mitigation/suppression of ELMs and disruptions**
- The answers to these questions must be forthcoming in a time scale not exceeding the ?2010?.
- Main elements of a strategy in this directions are:
    - **ASDEX Upgrade (entire W device);**
    - **JET with a Be wall and W and/or C divertor;**
    - **Work on ELMs/disruption damage on CFC & W targets in Russian plasma guns;**
    - **Work on Be/C and Be/W mixed materials in PISCES-B.**

Schedule for transition to a full W ASDEX Upgrade device

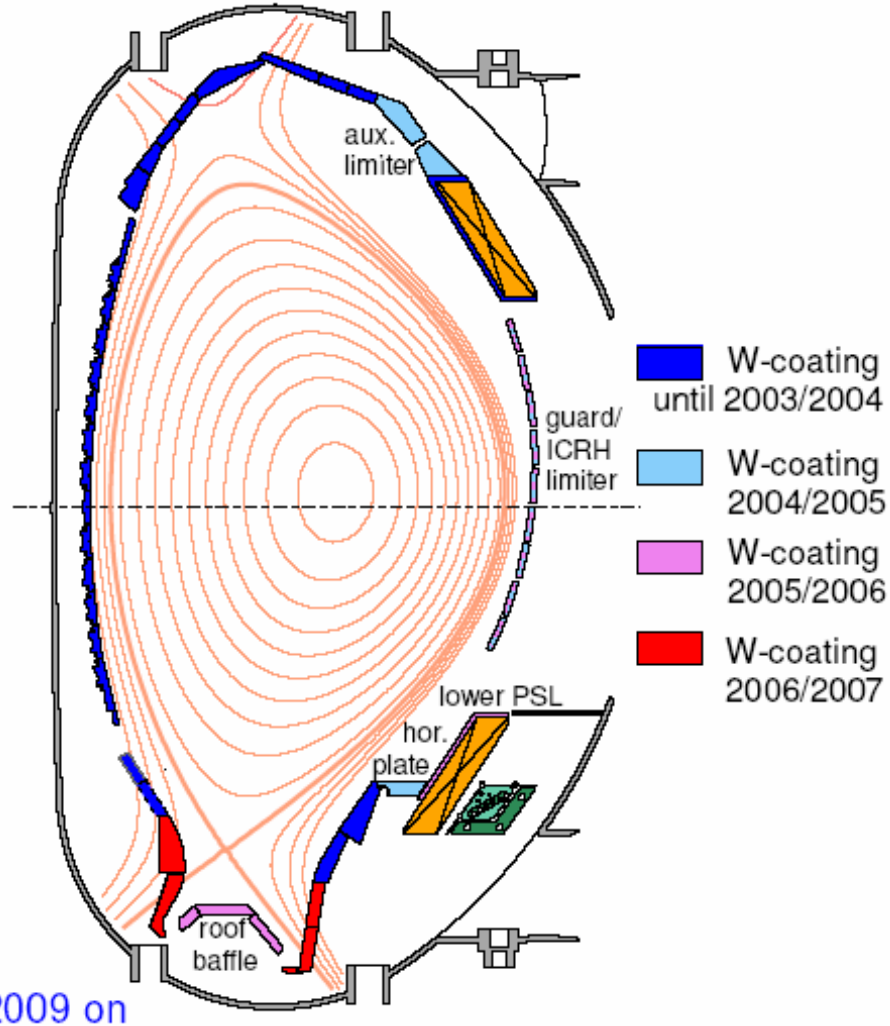
technique / thickness of coatings for guard limiters / lower divertor need to be veryfied:

- operational results of this campaign
- test of thick (VPS) coatings
- marker erosion in lower divertor

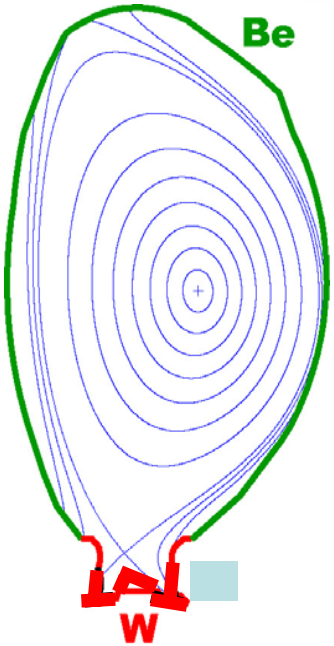
- tile manufactoring for next years coatings in progress / to be launched
- layout of lower divertor has to be decided early next year (coating after nect campaign)

⇒ full W device possible in 2006/2007

⇒ results may impact ITER design



JET will probably have W divertor from 2009 on



Option 1: Reference

ALL Tungsten divertor

PROPOSED STRATEGY:

- **Prepare for Option 1: build a complete set of new tiles with W coating**
- **If in 2007 the "new" ITER teams request test of option 2 rather than 1: exchange only the relevant tiles and keep CFC tiles for the target**

Be wall / all W divertor - Option 1
⇨ ITER fall back option

Objectives

- Demonstrate low T retention
- Test de-tritiation techniques
- Study effect of Be on W erosion
- Study melt layer loss – wall + divertor
  ⇨ ELMs and disruptions
- Refine control / mitigation techniques
  ⇨ Limit disruption / ELM damage
- Operate without C - radiation

Demonstrate routine / safe operation of fully integrated ITER compatible scenarios at 3-4MA
⇨ NBI power upgrade

Conclusions

Opportunities and challenges

- The choice of plasma facing materials is a key decision for of ITER.
- The main uncertainties/risks in ITER are identified. But we are still looking for solutions.
  - Erosion due to Type-I ELMs and disruptions and ELM mitigation methods as well as the control of T co-deposition with eroded C are most challenging issues.
  - The slow rate of progress in the area of tritium removal, together with favorable results from divertor tokamaks with high-Z, suggest that a renewed emphasis on all-metal machines would be prudent.
- According to current construction plans, few years are still available to conduct the critical R&D to improve confidence in the final material choices for ITER.
- Bearing in mind all the uncertainties, and the impact that PWIs may have on performance and operation, replacement of divertor and, in particular, of wall components should be an important operational feature of the ITER design.
- The feasibility and risk of such replacements need to be further reviewed and the outcome of these studies could (in my view still) impact the material mix selected at the beginning of operation.

Further info in:

- *G. Federici, C. H. Skinner et al "Plasma material interactions in current tokamaks and their implications for next step fusion reactors". Nucl. Fusion 41 (2001)1967*
- *C.H. Skinner, J. P. Coad and G. Federici 'Tritium removal from carbon plasma facing components' Physica Scripta T111, 92-97, 2004.*
- *G. Federici, 11th European Fusion Physics Workshop (Heraklion Crete Dec. 2003) ed. D. Campell to be published in PPCF.*
- *G. Federici, et al., "Key edge physics and plasma-material interaction issues of the ITER divertor and PFCs", J. Nucl. Mater. 313-316 (2003) 11-22.*