The Challenges of Plasma-Surface Interactions for ITER & Beyond

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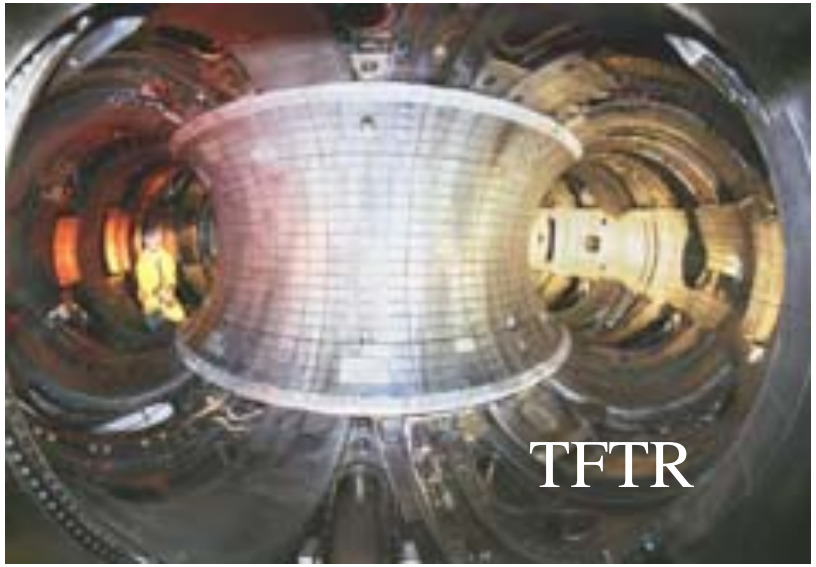
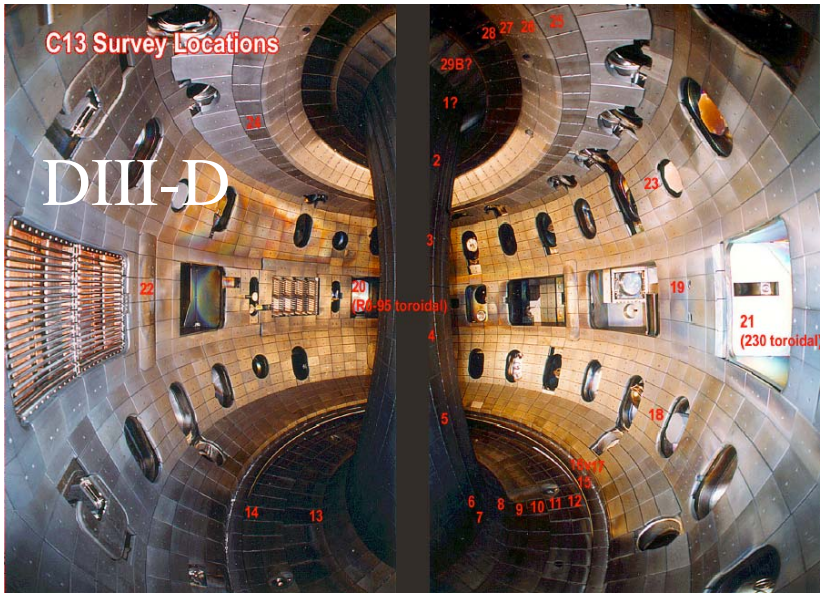
PPPL Colloquium Feb 2008

Present tokamaks require that plasma-facing materials *simultaneously* meet several requirements

- Mechanical strength
  - Large JxB and eddy current forces during disruptions
- Forgiving of temperature transients.
  - Heat flux > GW / m^2
- Erosion/radiation characteristics favorable for a wide variety of exploratory fusion core plasma scenarios.
- To-date low-Z materials, in particular graphite/CFC, have met these challenges.





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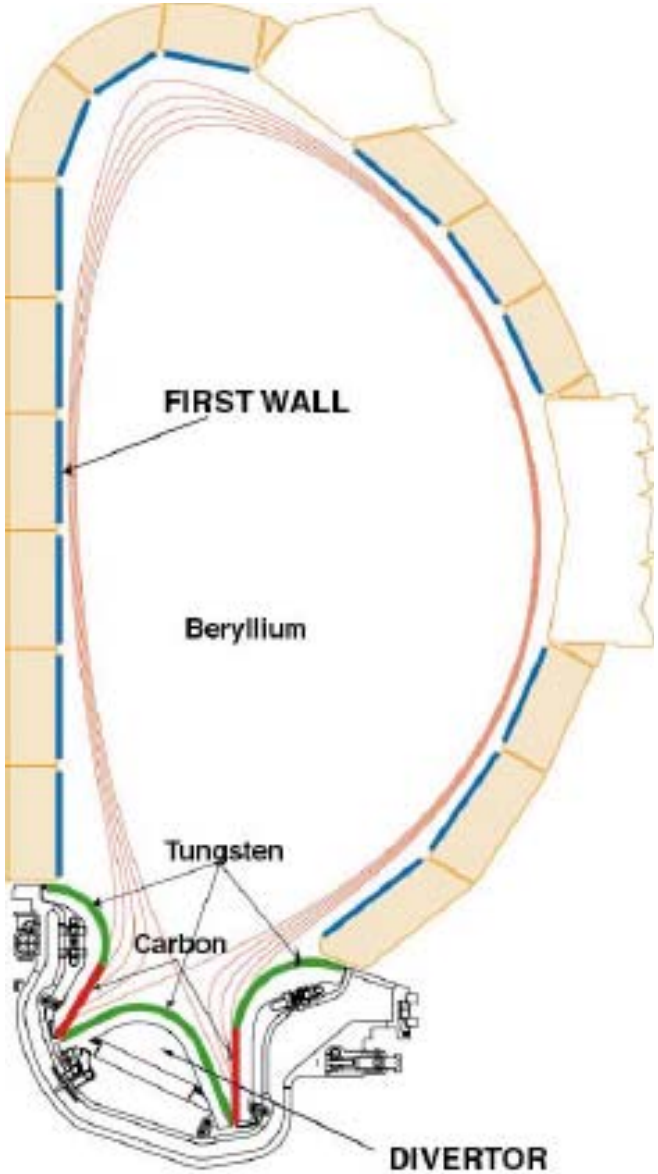
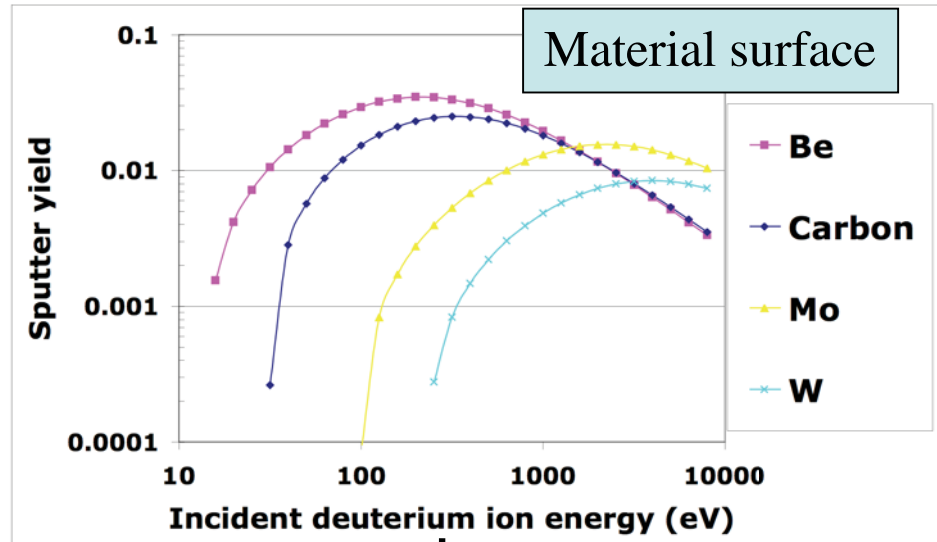
- To-date low-Z materials, in particular graphite/CFC, have met these challenges.
- **Q~1 and fusion power > 10 MW achieved with Carbon walls + low-Z coatings, paving the way for our confidence in ITER.**



So what's the problem?

Lack basic understanding and diagnosis of PSI processes in fusion devices → Uncertain extrapolation

- Intermediate steps uncertain between sputtering and core plasma.
  - Intense power flux density
    - Materials placed near thermal limits.
  - Surface layers of plasma facing materials are rapidly and continually being reconstituted by plasma erosion and redeposition.
    - Peak ion flux ~ 100 kA /m^2
  - Plasma transport ensures large gradients in plasma conditions across magnetic flux surfaces.
  - Turbulent plasma transport
  - While plasma is axisymmetric, real armor geometry leads to 3-D effects.



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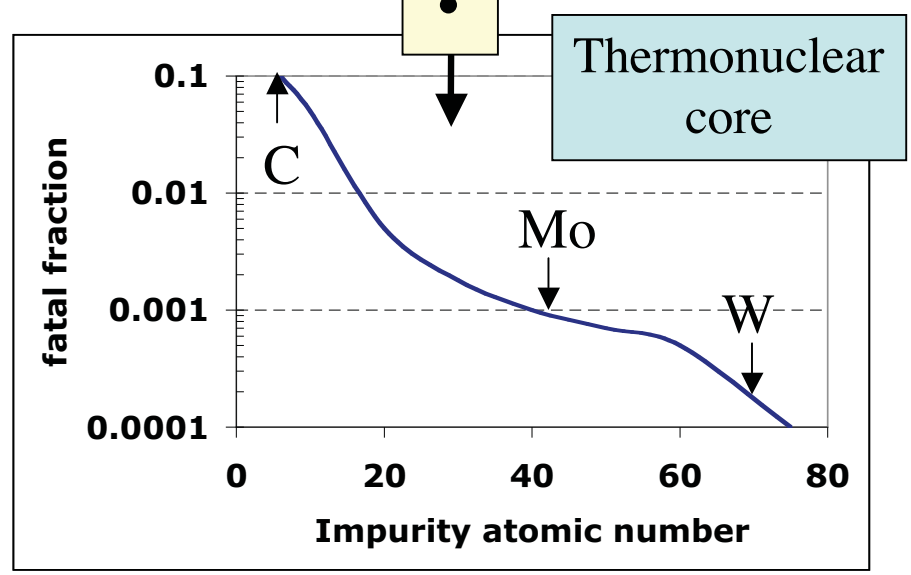


Overview

- Highlight five prominent PSI issues as one steps from present devices to ITER to Demo.
  - Each issue is quantitatively and qualitatively "worse" in Demo.
  - Every issue pushes our PSI knowledge up-to and past our limits.
  - Despite its own challenges, ITER will **not** address most Demo PSI issues.

- Discussion restricted to "conventional" solid plasma-facing materials.

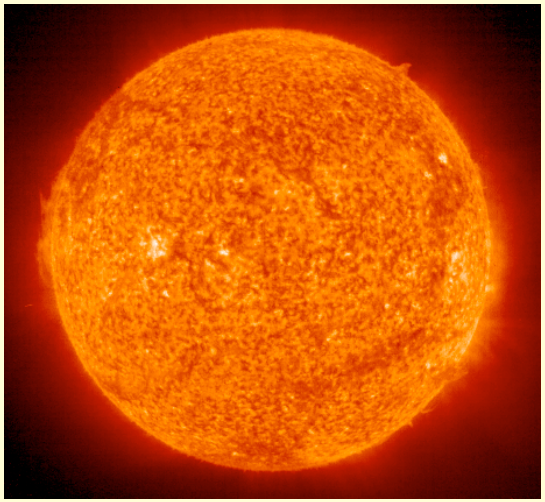
- Ways forward: PSI diagnosis & experiments.

Controlling PSI becoming increasingly important and difficult as we move from present tokamaks → ITER → demonstration fusion power plants

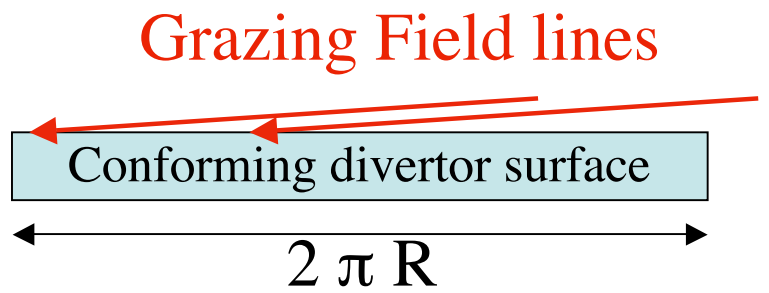
Issue / Parameter	Present Tokamaks	ITER	DEMO	Consequences
Energy exhaust (production) *GJ / day*	~ 10	3,000	60,000	- active cooling - max. tile thickness ~ 10 mm

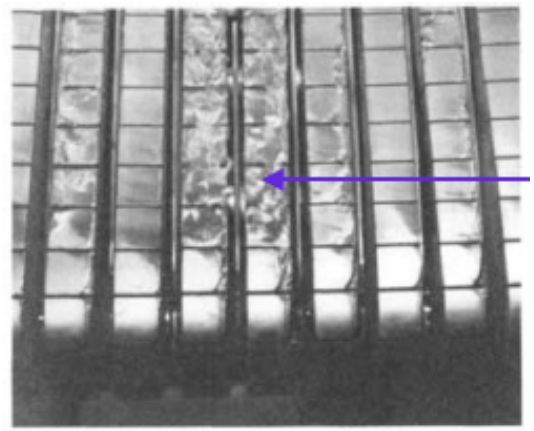
Heat exhaust is primary design point for edge materials, since this is directly related to fusion power density

$$q_{\text{target}} = \frac{\frac{1}{5} P_{fusion}}{A_{divertor}} \sim 10 - 20 \frac{MW}{m^2}$$

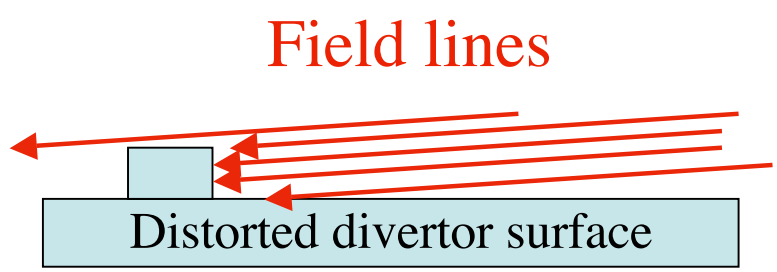


$$\approx 40 \frac{MW}{m^2}$$





Melted Be tiles On JET

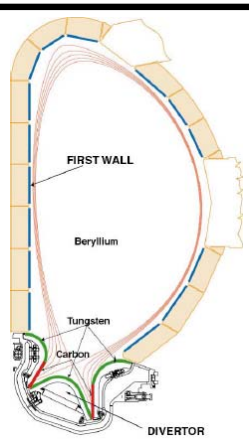


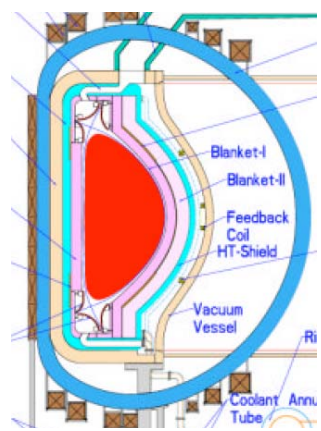
Distorted surface "proud" to the field line receives $q_{//} \sim 500$ MW/m^2 and is immediately melted/ablated.

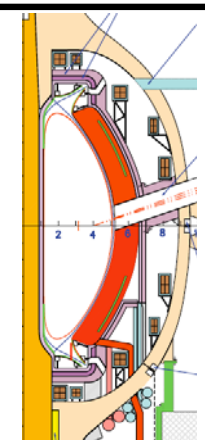
ITER falls substantially short of Demo power density P/S ~ 1 MW/m^2 and energy throughput

	ITER	ARIES-AT	ARIES-RS	ARIES-ST
Duration (s)	400	$3x10^7$	$3x10^7$	$3x10^7$
Ambient T (C)	~ 200	1000	850	> 700
R (m)	6.2	5.2	5.52	3.2
A ≡ R/a	3.1	4.0	4.0	1.6
P_{exh} (MW)	150	390	515	624
P/S (MW/m^2)	0.21	0.85	1.1	0.99
P/A_{div} (MW/m^2)	2.4	10	12	20

$A_{div} / S \sim$ 5-10%

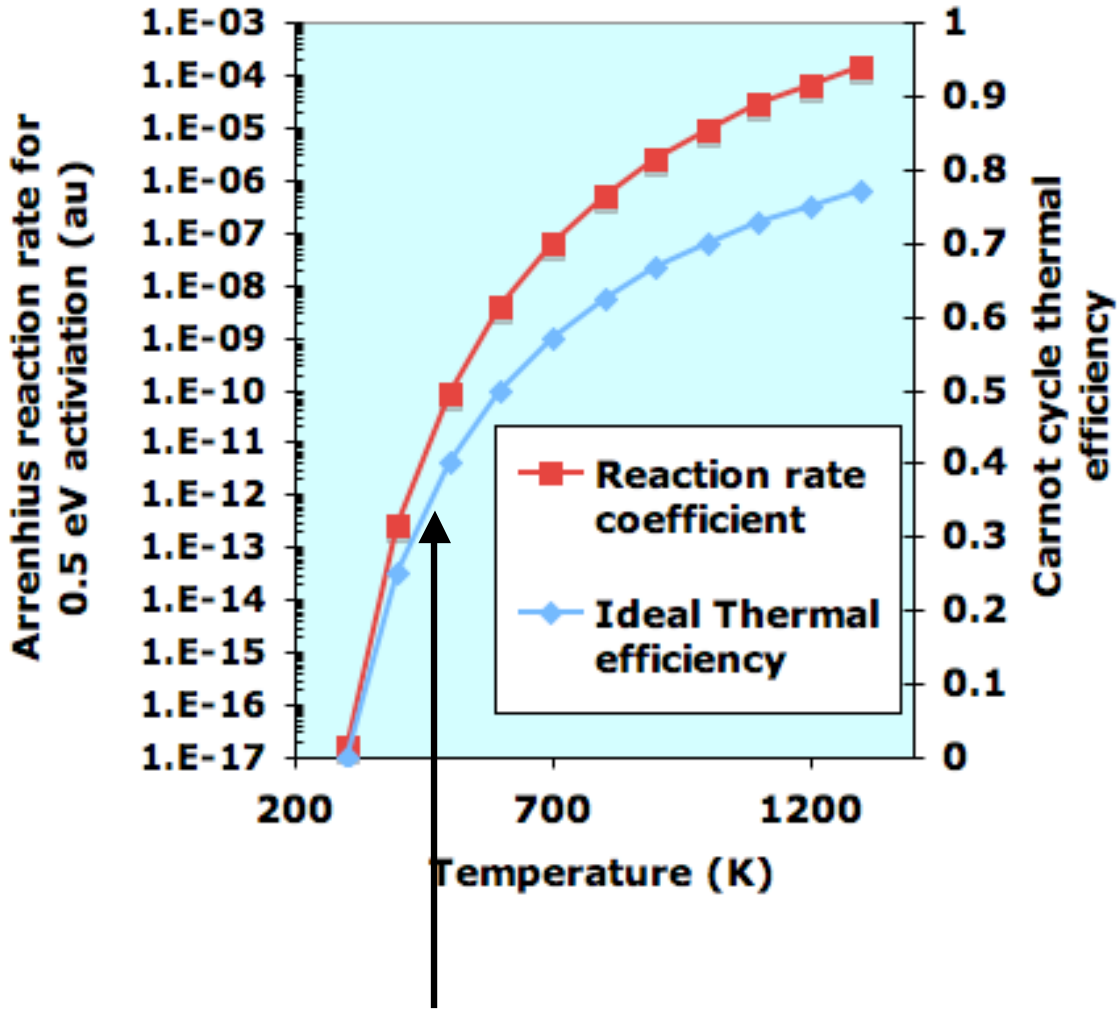






Thermal efficiency dictates high ambient T → Fundamentally different Physical Chemistry regime for wall that is completely unexplored in fusion devices

- Rate equations follow Gibb's / Arrhenius relationship reaction rate ∝ $\exp(-E_o/kT)$
  - Activation energies $E_o \sim 0.5$ -1 eV.
- Precludes water cooling technology in reactor



Limit for Water-cooling

Helium gas (or molten metal) cooling/heating required for hot walls (T > 700 C)

- **Actively controlled temperature of all plasma-facing components.**
- **US Engineering demo peak heat removal ~ 10 MW/m^2**
  - **He and PFC joining sets the limit for peak heat removal in divertor.**
- **The heat removal challenge: P_{exh} / A_{div} ~ 10 MW / m^2 right at the technology limit!!**

Table 2
Test results

Flow rate (kg s^{-1})	Heat flux (MW m^{-2})	Peak surface temperature (°C)	Pumping power (W (% of power removed))
0.022	10	380	157 (0.8)
0.011	6	422	21 (0.2)
0.0064	3	424	3.4 (0.06)

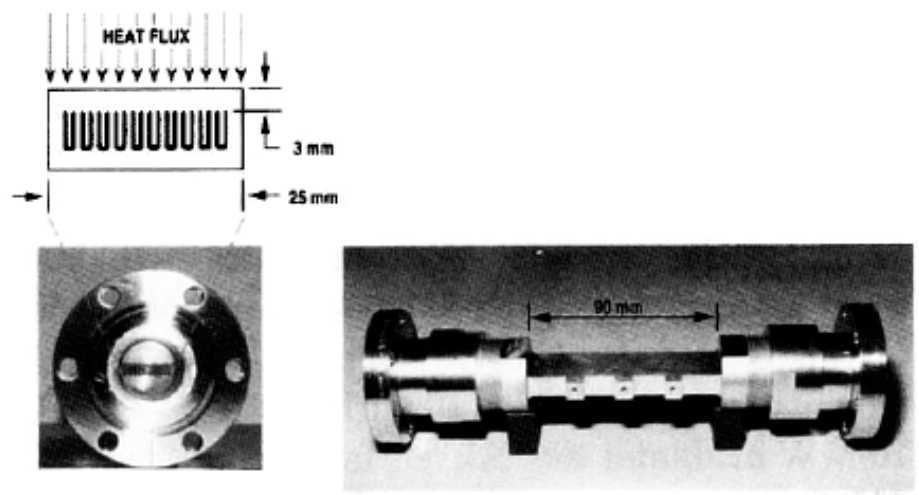


Fig. 1. GA divertor module.

C.B. Baxi / Fusion Engineering and Design 25 (1994) 263–271

Energy sustainment chasm to Component Test Facility (CTF) & DEMO

- **CTF** (Q < 3) cares about energy fluence, period.
  - Power density x Δt: BOTH MATTER!
  - Must also have a closed Tritium fuel cycle!
    - Could we presently get a nuclear license to operate a CTF?

- **DEMO**: Burn Q > 25 + Power/m^2 x Δt.
  - **P_{exh}/S** ~ 1/4 P_n/S ~ **1 MW/m^2 x 3x10^7 s** ~ 1 full-power year
  - Ambient temperature > 700 C for thermal efficiency
    - Therefore CTF must also have high T walls to test components.

- Present track of devices, including ITER, do *not* address energy sustainment issues required for CTF or DEMO
  - P/S and pulse duration too small. Water-cooled, low-T walls.

- **Conclusion: Need a sustained, reactor-level power density, hot-wall experiment to fill gaps to CTF & DEMO.**

Controlling PSI becoming increasingly important and difficult as we move from present tokamaks → ITER → demonstration fusion power plants

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Transient energy exhaust from plasma instabilities $\Delta T \sim MJ / A_{wall}(m^2) / (1\ ms)^{1/2}$	~ 2	15	60	- require high $T_{melt/ablate}$ - limit? ~ 40 for C and W - surface distortion

Rapid dissipation of plasma thermal energy poses major challenges in any Demo

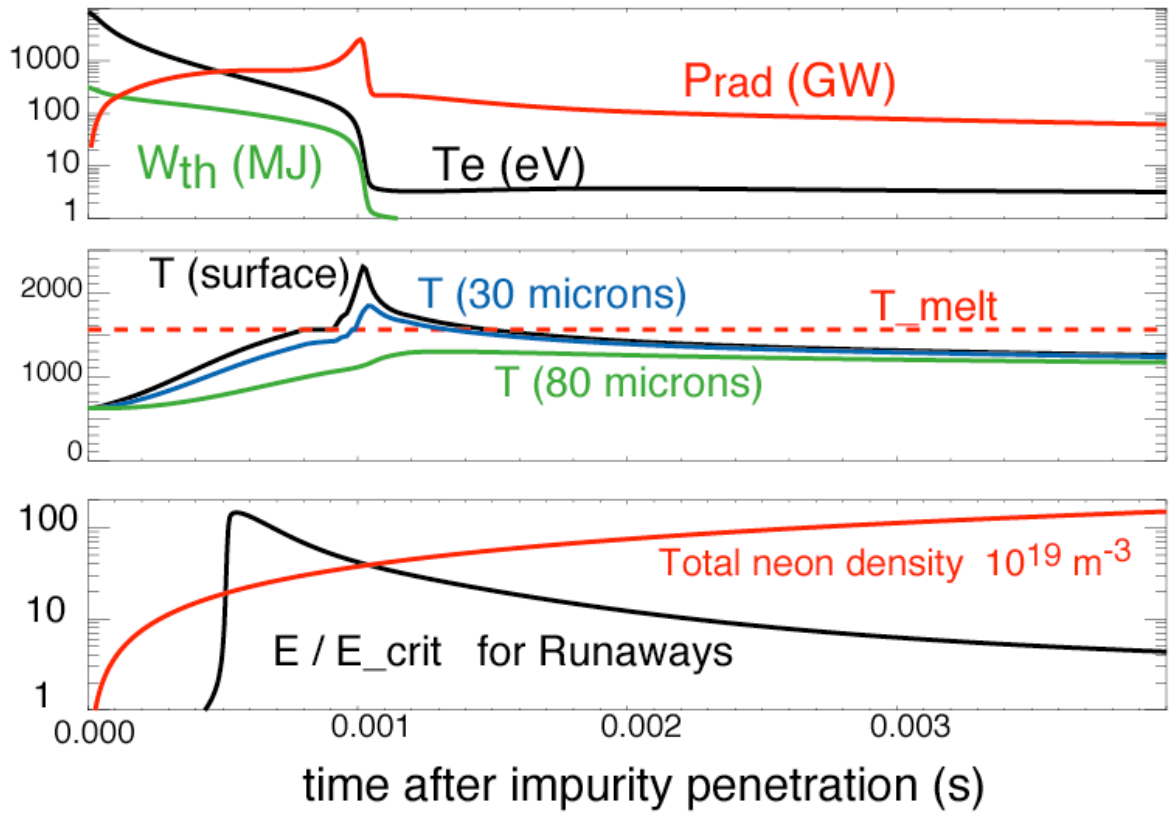
$$T_{surface,\max} = T_{ambient} + C\frac{W_{th}}{A_{wall}\,\tau^{1/2}}$$

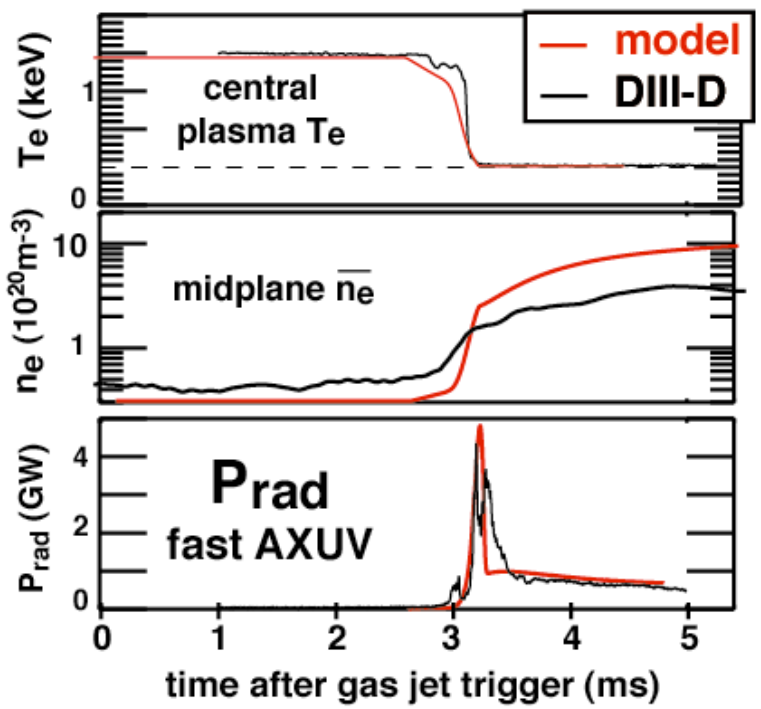
Material	T_{max} (K)	Limit MJ m^{-2} $s^{-1/2}$
Be	1550	8
C	**4000**	**42**
Mo	2900	28
W	**3680**	**45**

Transient thermal limits
($T_{ambient}$ ~ 1200 K)

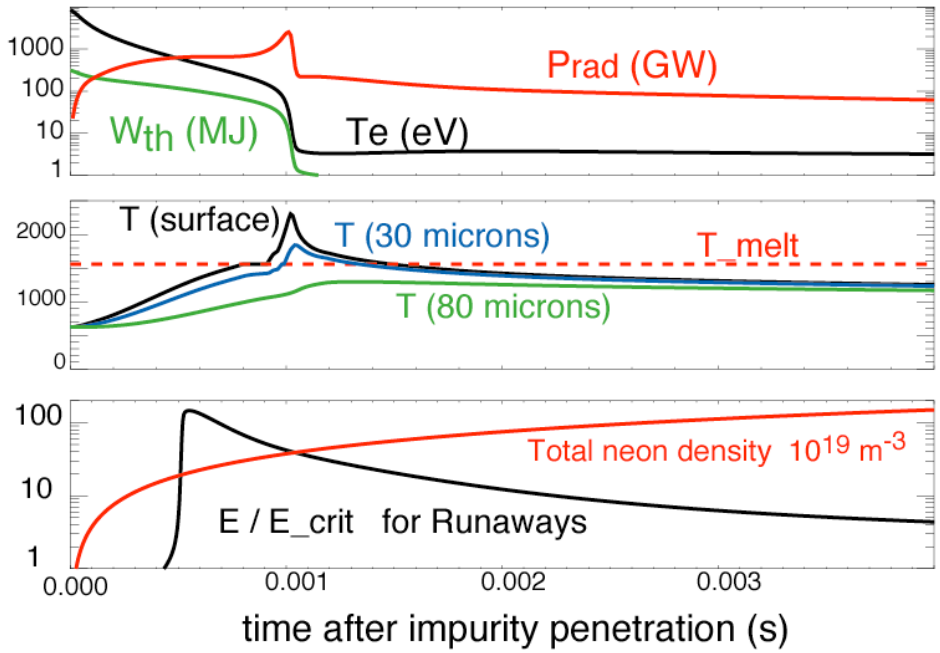
Thermal energy dissipation timescale τ < ms. Set by both atomic physics & MHD

Global energy balance model of ITER disruption mitigation By neon injection

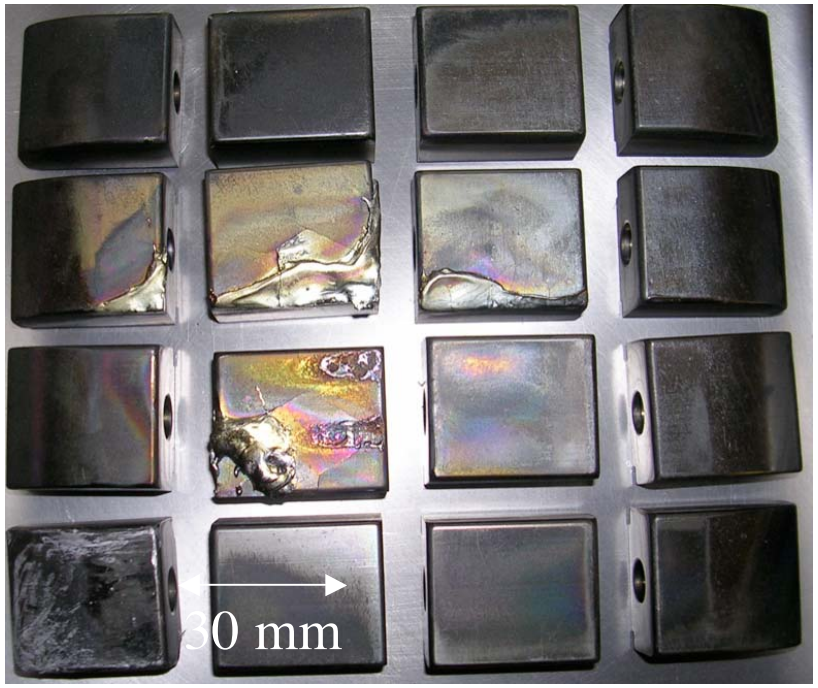




Thermal energy dissipation timescale ~ ms. Can be easily triggered by PSI "failure"



C-Mod Molybdenum (T_{melt}=2900 K) limiter melted during disruptionsc

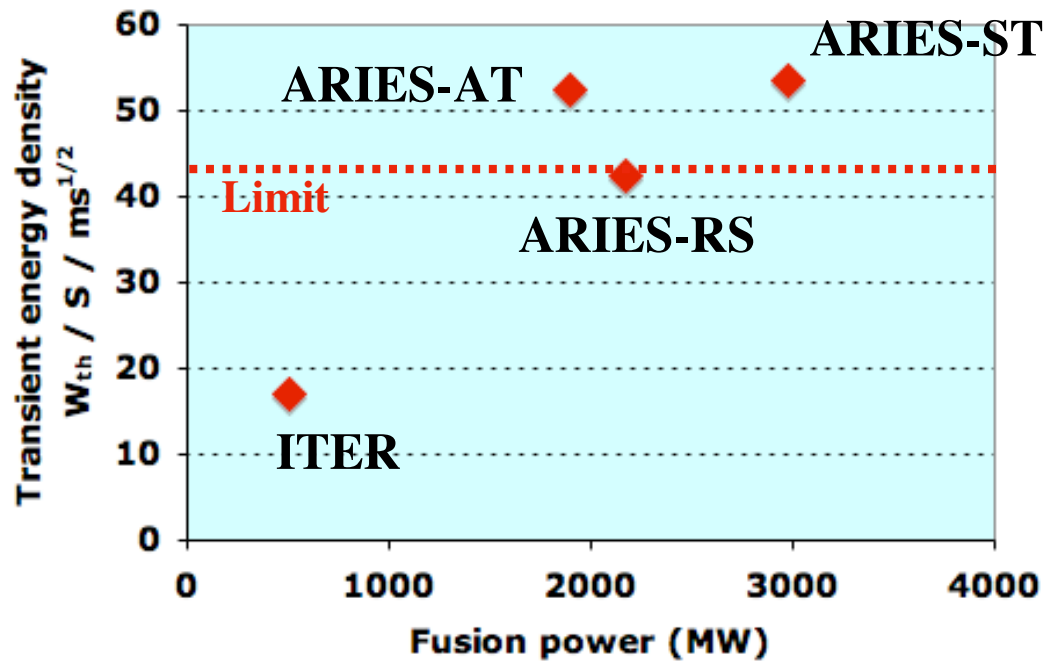


- Dilute MFE plasma (n~10^{20} m^{-3}) extinguished by small particulate
  - 2 mm "drop" of W == $N_{e,plasma}$

Plasma thermal energy linked to fusion power efficiency. Dissipation will be major challenge in any Demo, including stellarators

- Only flexibility appears to be in τ.
  - Will high-Beta large-scale tokamaks & stellarators have τ > ms?
  - Can we trick the plasma into having τ > ms, for example using opacity?

$$\frac{W_{th}}{A_{wall}\,\tau^{1/2}} \sim \frac{pV}{A(R/c_s)^{1/2}} \sim P_{fusion}^{1/2}\,\varepsilon R^{1/2}$$



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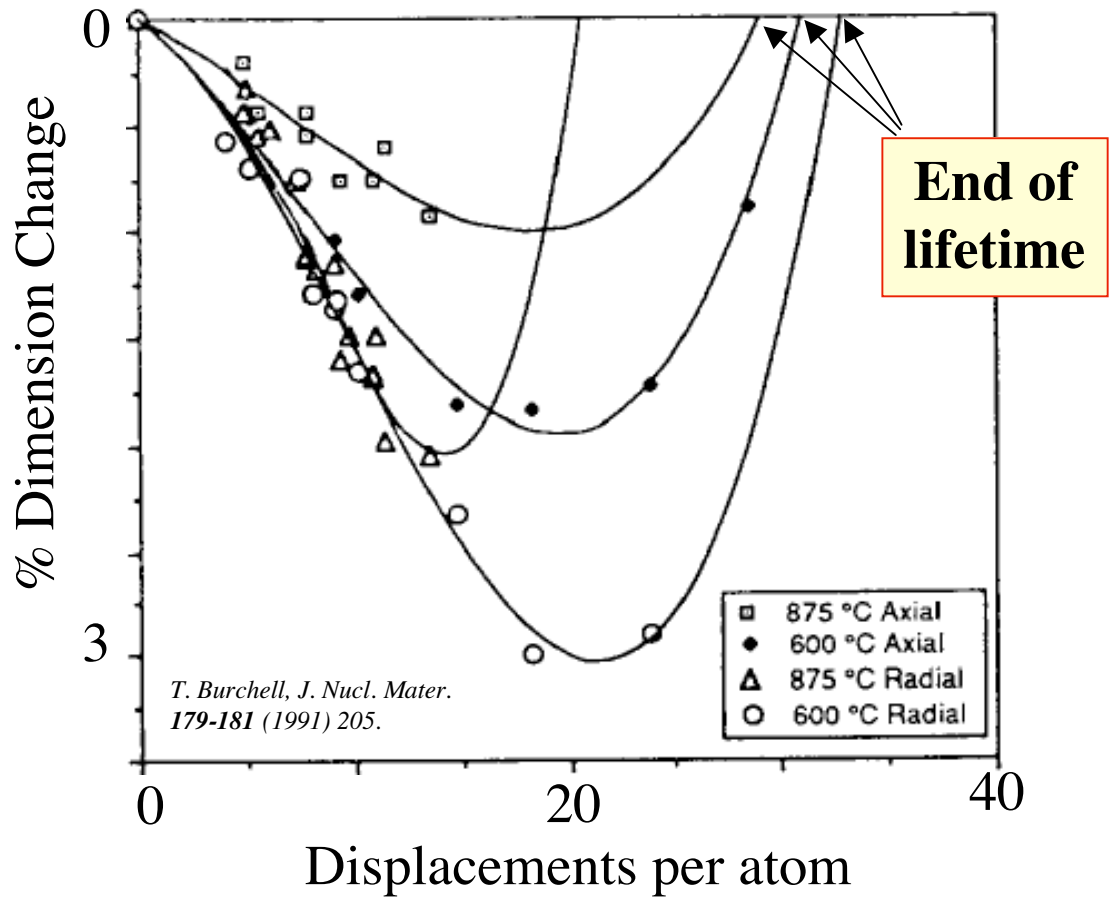
Yearly neutron damage in plasma-facing materials *displacements per atom*	~ 0	~ 0.5	20	- evolving material properties: thermal conductivity, swelling, traps for tritium

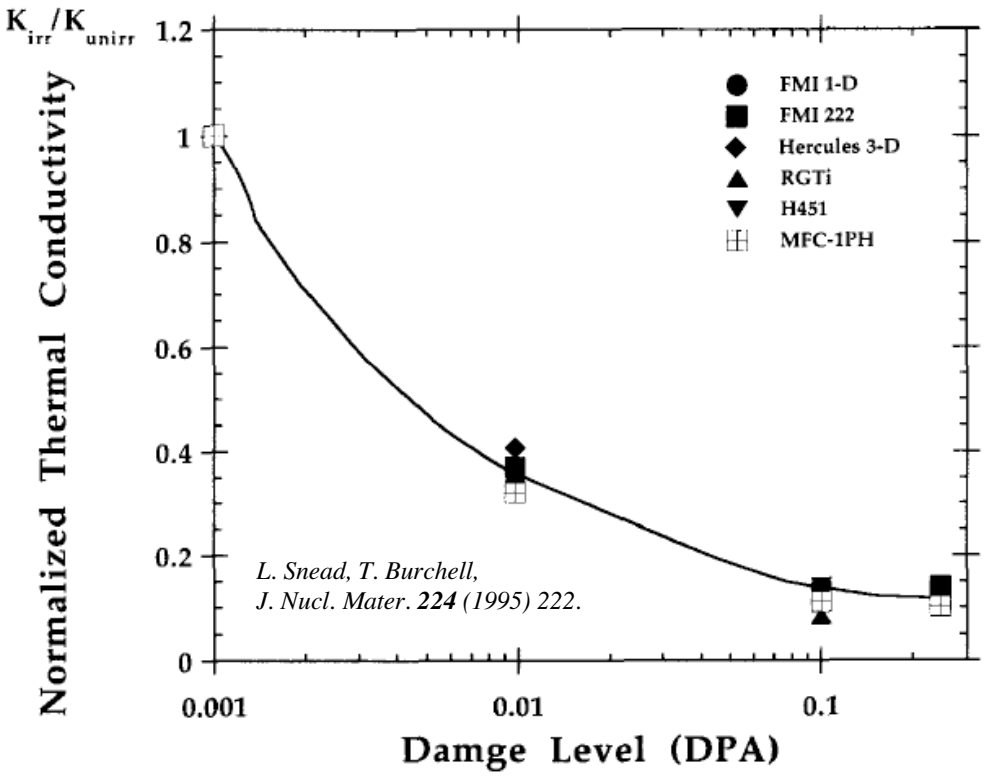
14 MeV neutron-induced damage set lifetime limits for graphitic fusion materials

Neutron induced shrink/swell in N3M graphite

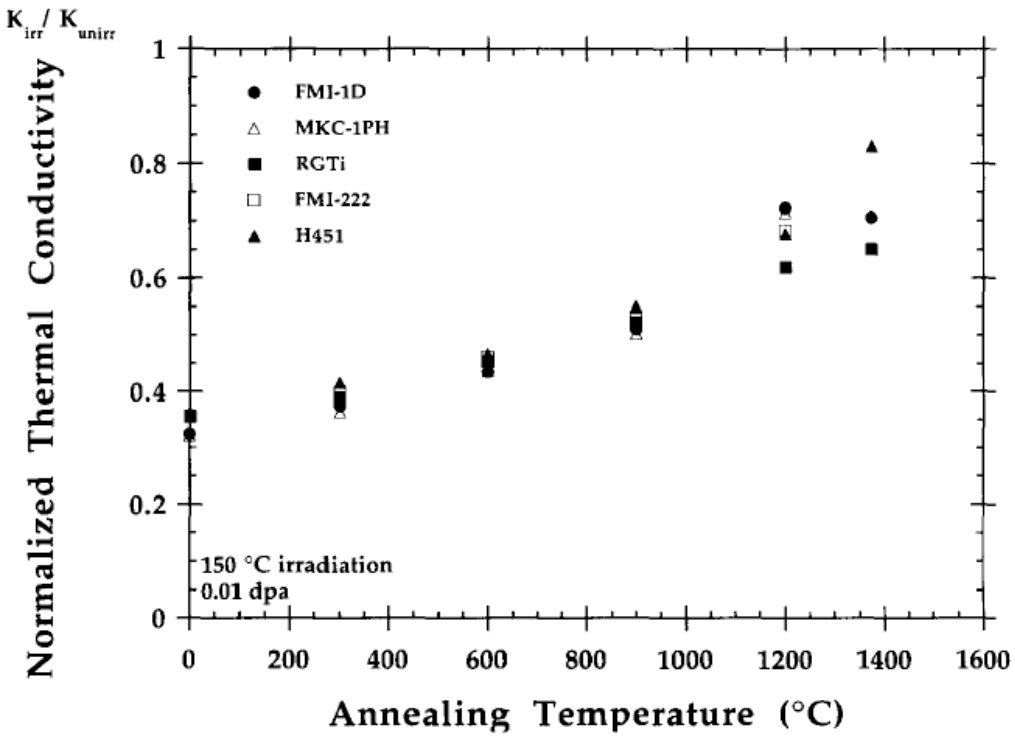


"Self-annealing" of neutron damage: High ambient temperatures required + energetic fluxes

Neutron-induced defects scatter phonons, greatly reducing C thermal conductivity



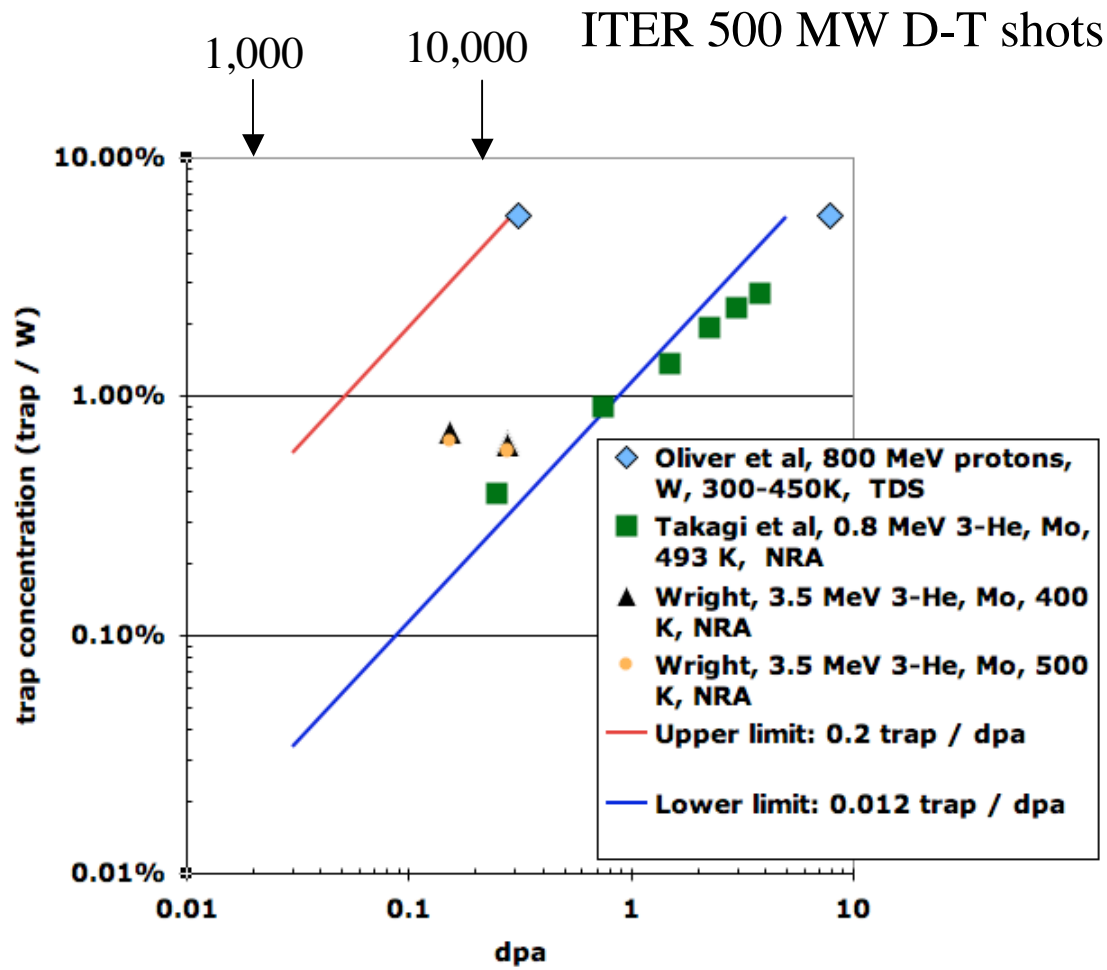
Thermal conductivity restored by high-T annealing



An emerging area of concern: Neutron damage producing Tritium trap sites in refractory metals

- Neutron-induced displacements produce damage "trap" sites for D/T fuel in bulk of material
  - Lab studies ~ 0.1 - 0.3 trap / dpa that saturate 1% traps / atom.

- High temperature MUST heal these traps since in a D-T reactor ~ 1 kg Tritium can be stored in (Traps / atom) ~ 10^{-4}
  - Surpassed in a few hours in demo due to high permeability of the D/T in tungsten at high ambient temperature.



[1] Oliver et al J. Nucl. Mater. ***307-311*** *(2002) 1418.*
[2] Takagi et al. Fus. Sci. Tech. ***41*** *(2002) 898.*
[3] Wright, G, Ph.D. thesis, submitted to J. Nucl. Mater.

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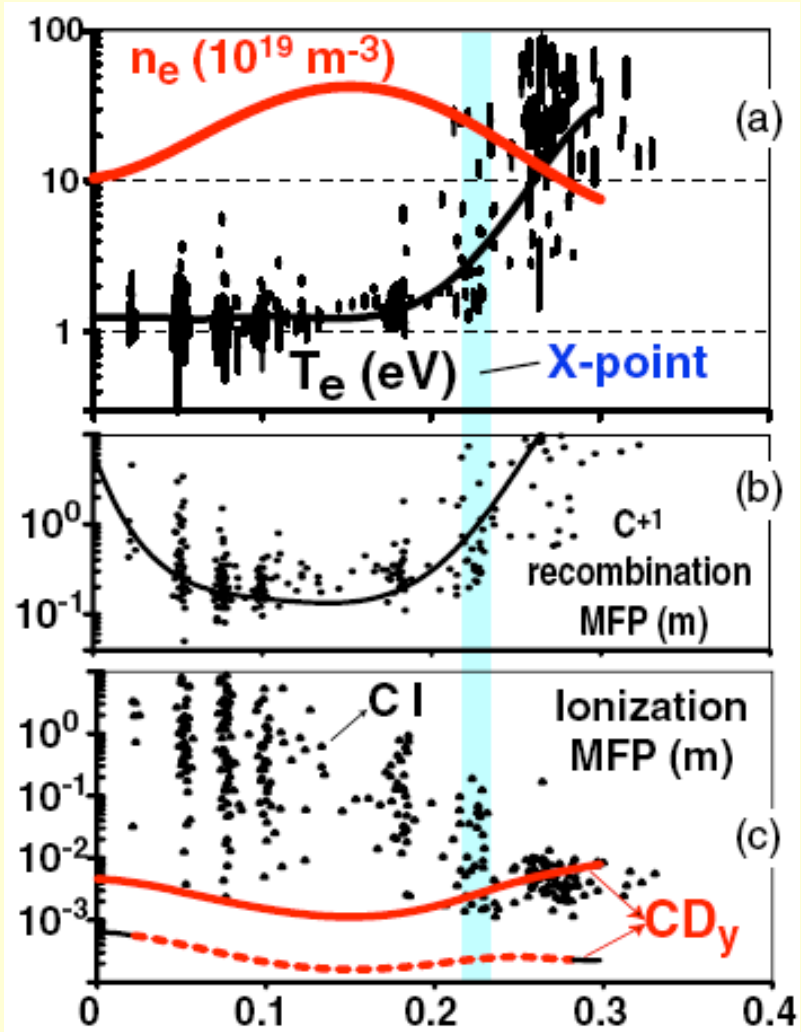
Yearly neutron damage in plasma-facing materials *displacements per atom*	~ 0	~ 0.5	20	- evolving material properties: thermal conductivity & swelling
Max. gross material removal rate with 1% erosion yield *(mm / operational-year)*	< 1	300	3000	- must redeposit locally - limits lifetime - produces films

Tokamak edge plasmas feature extreme spatial gradients and fluctuations levels, making erosion prediction and control very difficult

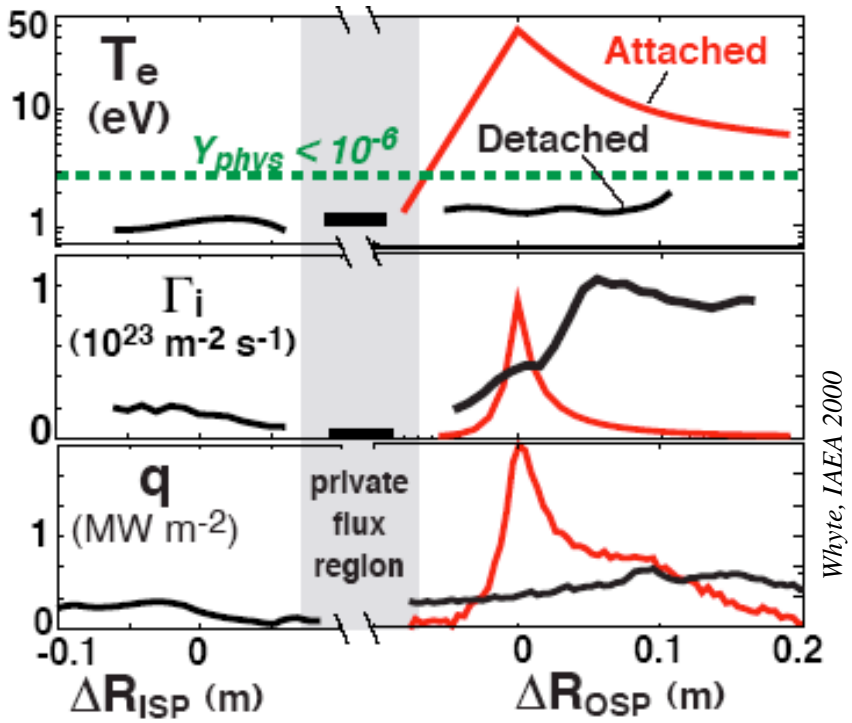
- Extreme range (100 eV → 1 eV) is possible for plasma temperatures
  - Highly ionizing → recombining.
  - Physical sputtering → chemical removal.
- Develops large-scale sonic flows to surfaces.
  - Particle flux density ~ 10^5 A / m^2
- **Key result: PFC species have ionization distances << linear size of divertor targets**
  - **Every atom removed from surface has already been removed and replaced by plasma many times over.**
  - **Plasma and surfaces are strongly coupled to each other**



Vertical distance from divertor surface (m)

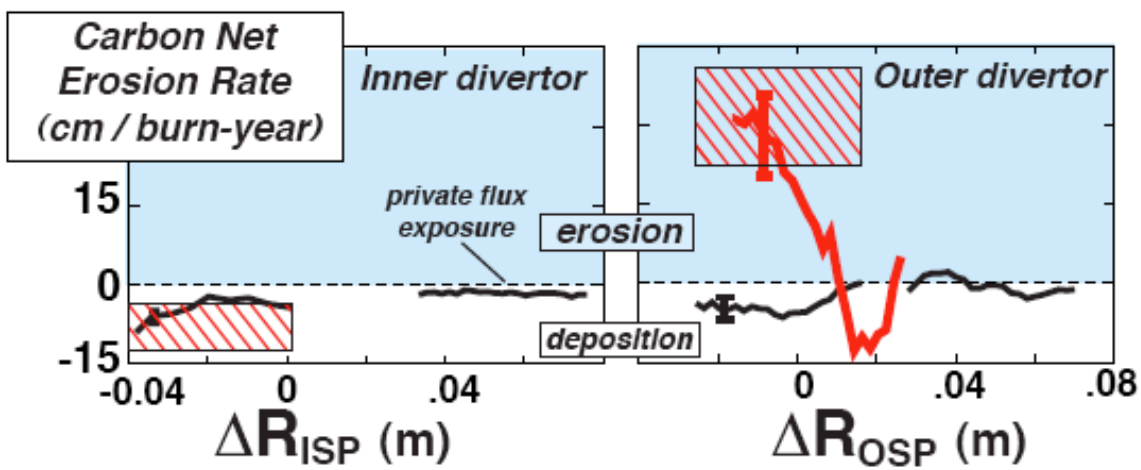
Whyte et al. Nucl. Fusion ***41*** *(2001) 1243*

One can exploit the non-linear nature of the divertor plasma recycling to access low T_e regimes where erosion is suppressed.

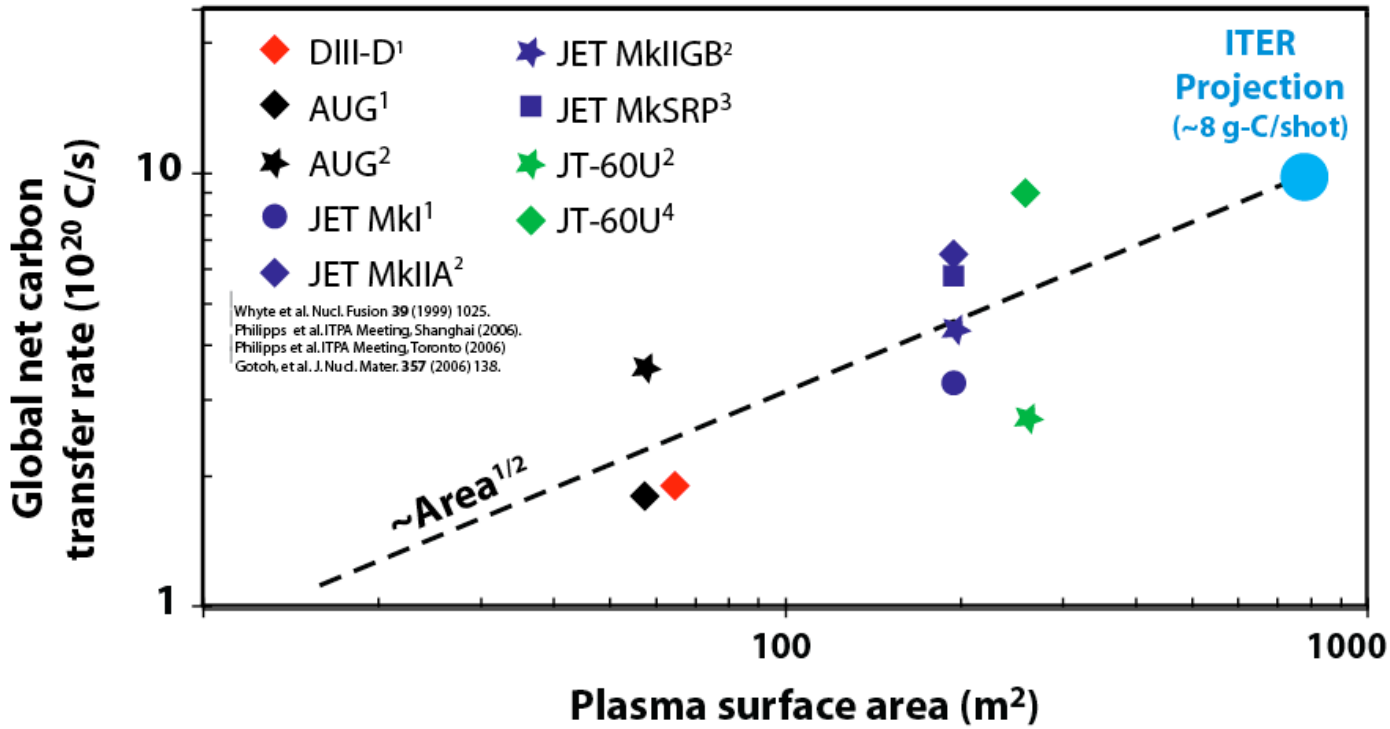


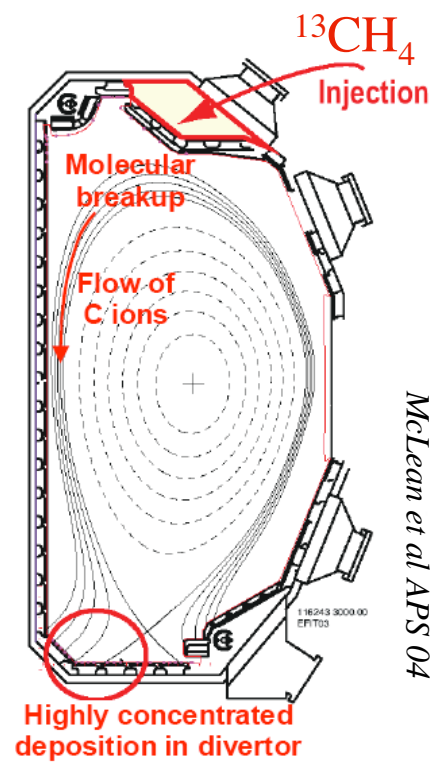
- Key insight: one has more control knobs with the plasma parameters than with the material properties!!

DIII-D: Map of divertor Erosion / deposition



"Archeological" deposition measurements:
Tokamak plasmas effectively net "transfer" carbon from one location of the wall to another

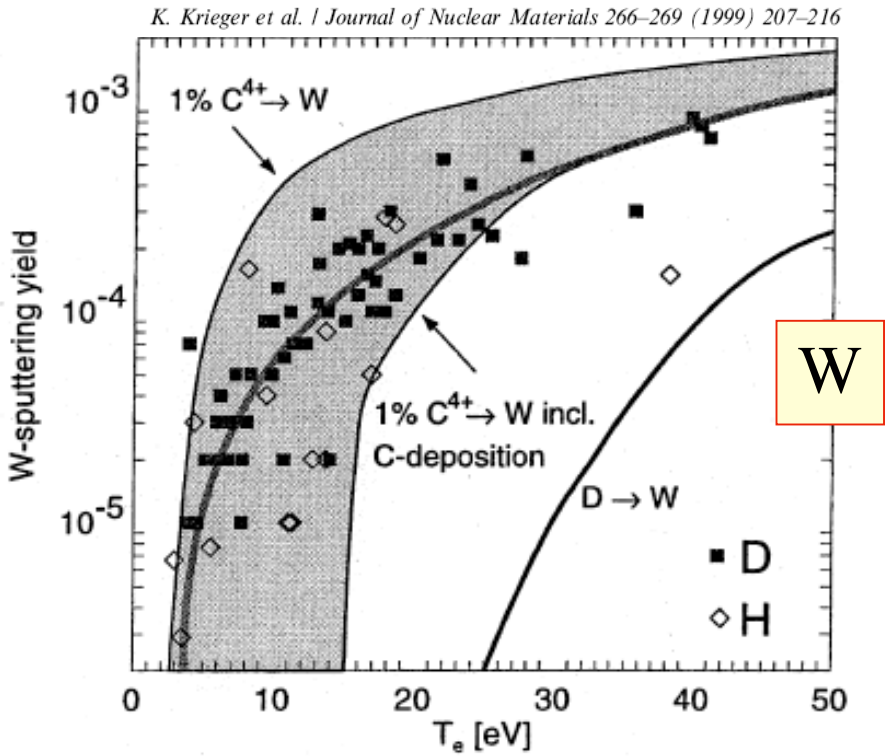


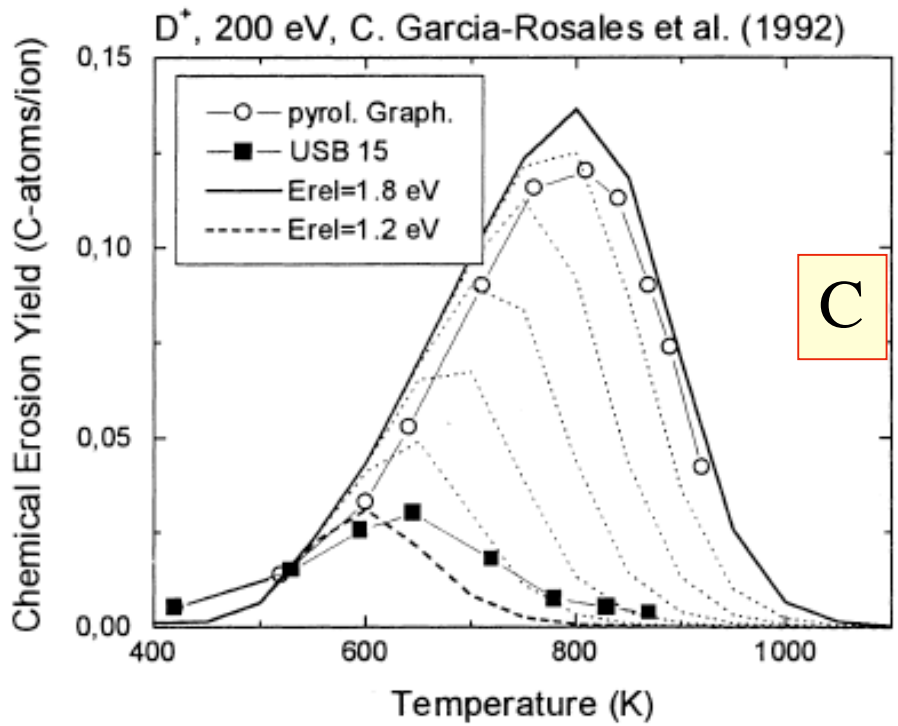


McLean et al APS 04

- Controlling mechanisms of erosion sources, long-range transport and deposition balance are not well understood.

Erosion control should be very different with hot walls.





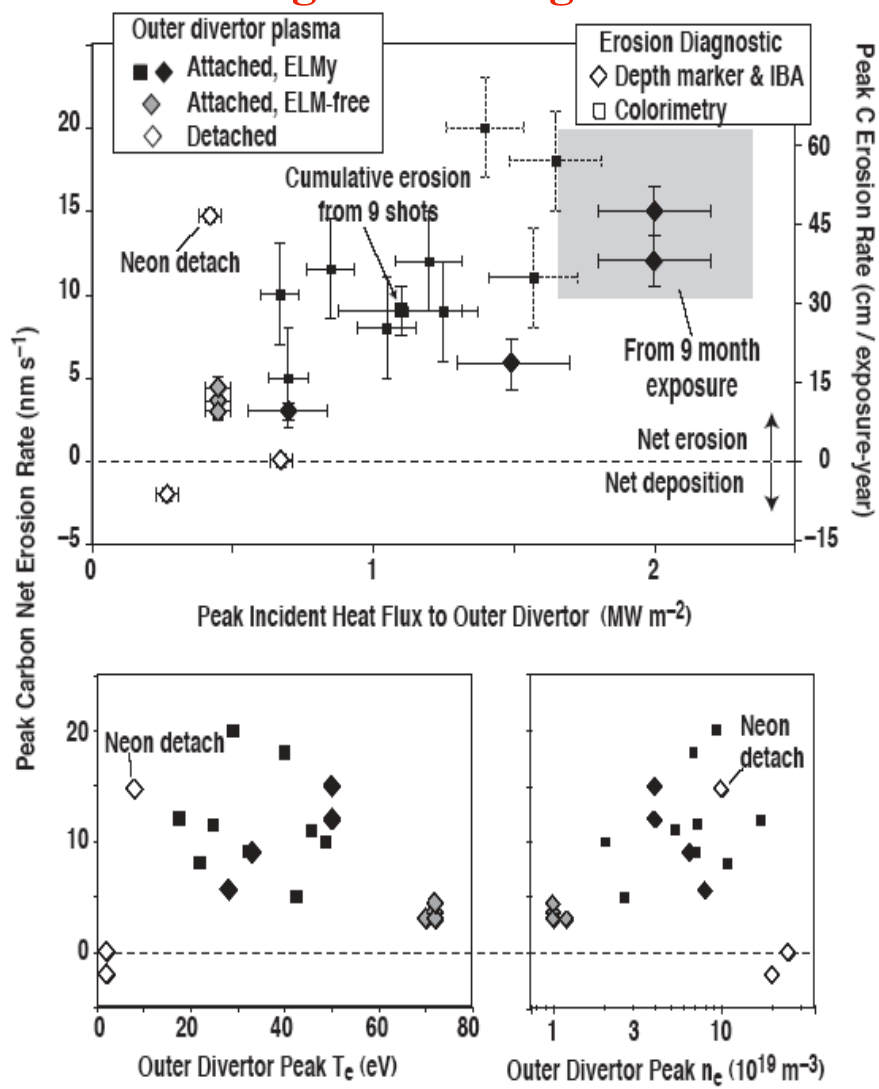
- Tungsten divertor erosion caused by low-Z C in Asdex.
- Elimination of low-Z vacuum impurities --> elimination of W erosion with hot walls?

- High temperature forces strong H_2 recombination in carbon
- Minimal C chemical erosion with hot walls.

The harsh tokamak environment has limited the deployment of in-situ surface diagnosis, but are vital to improve our understanding

- Quartz MicroBalance widely deployed
  - Recessed regions only
  - No element or H/D/T discrimination.
- Colorimetry
  - Requires in-situ film deposition
- Marker tiles
  - Require dedicated operation time and physical access to materials.

Survey of DIII-D erosion results Using in-situ diagnostics



A new generation of innovative in-situ PSI diagnostics are being developed on C-Mod

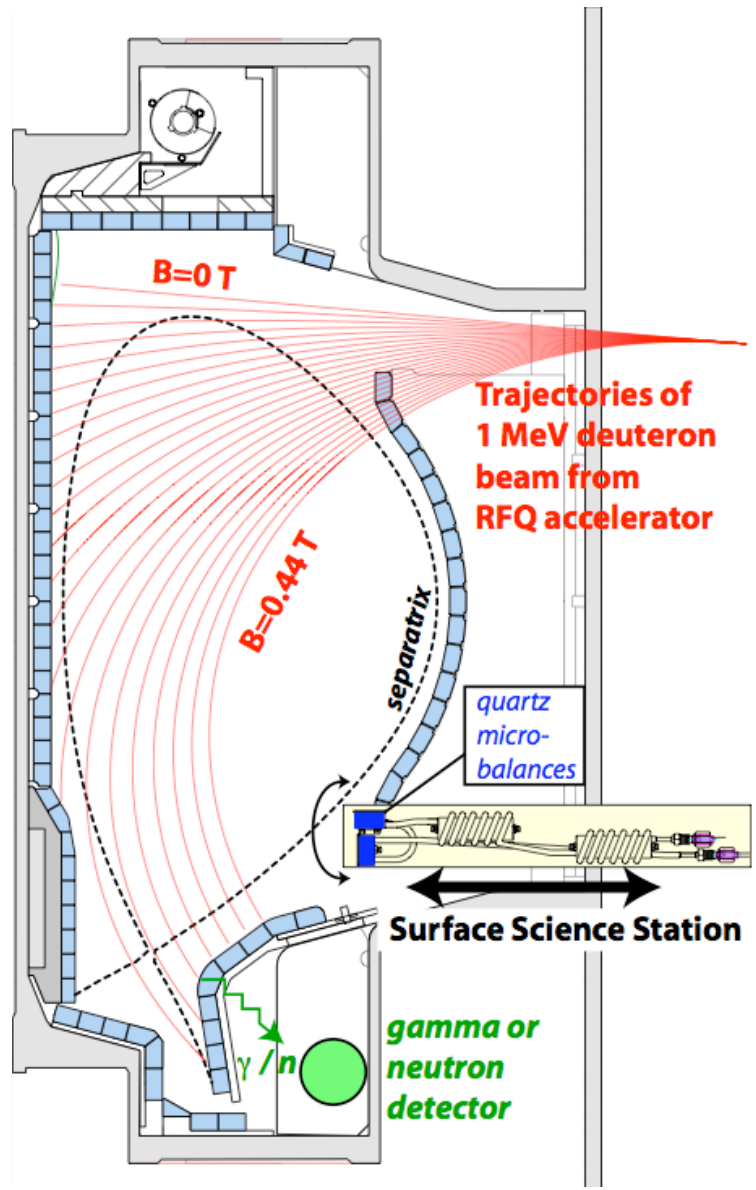
Example: RFQ accelerator

- High-current cw RFQ accelerator attached to the tokamak

- Innovation: Exploit intrinsic magnetic fields to steer beam to any poloidal (toroidal) location.

- Shielded neutron + gamma detection from MeV D beam nuclear reactions with PFCs

- Shot-to-shot "maps" of erosion, redeposition and tritium retention, depth resolved to penetration distance of beam (~10 microns).



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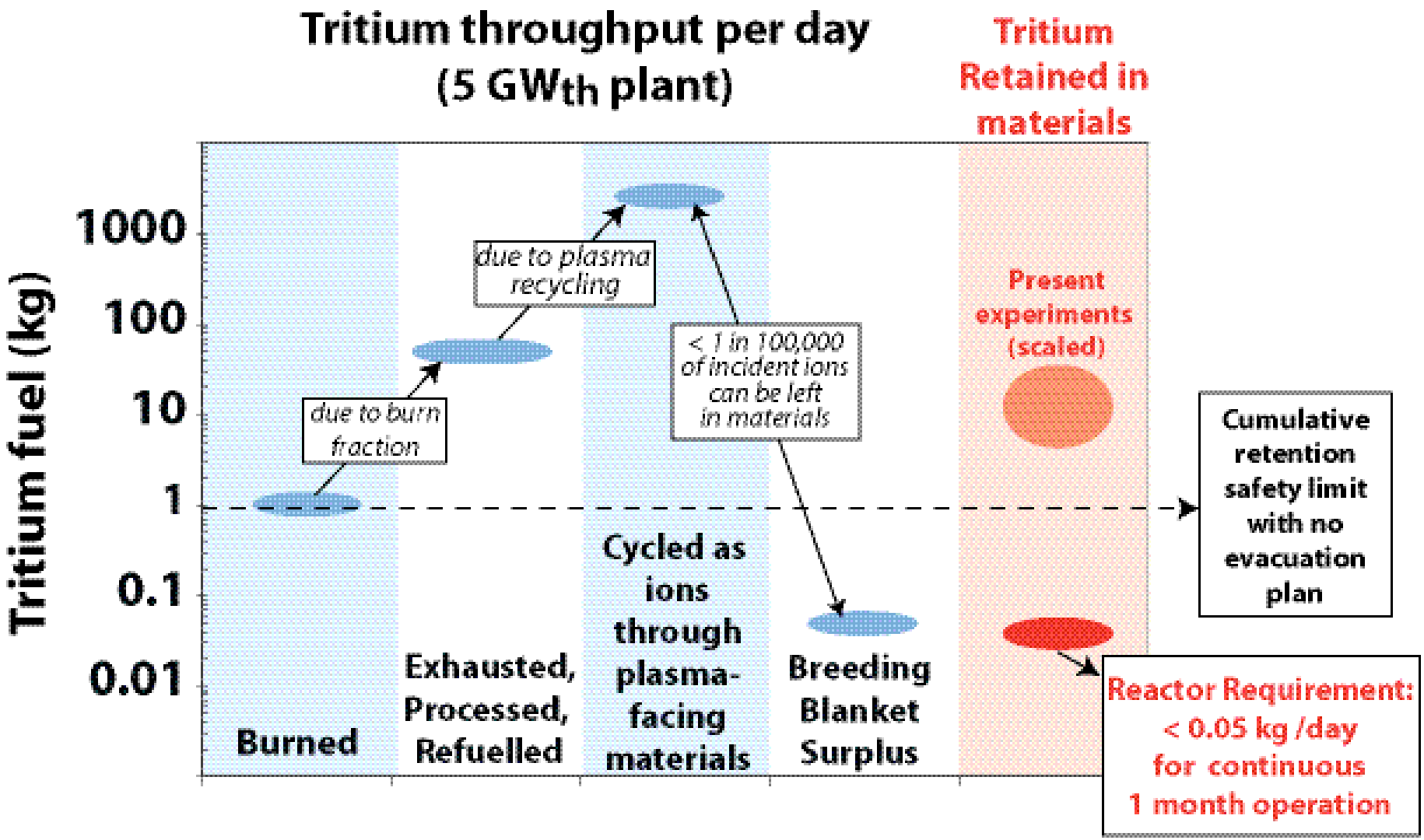
Tritium consumption *(g / day)*	< 0.02	20	1000	- Tritium retention in materials and recovery

Tritium retention in Demo must satisfy fuel cycle and regulatory limits: Orders of magnitude improvement required.



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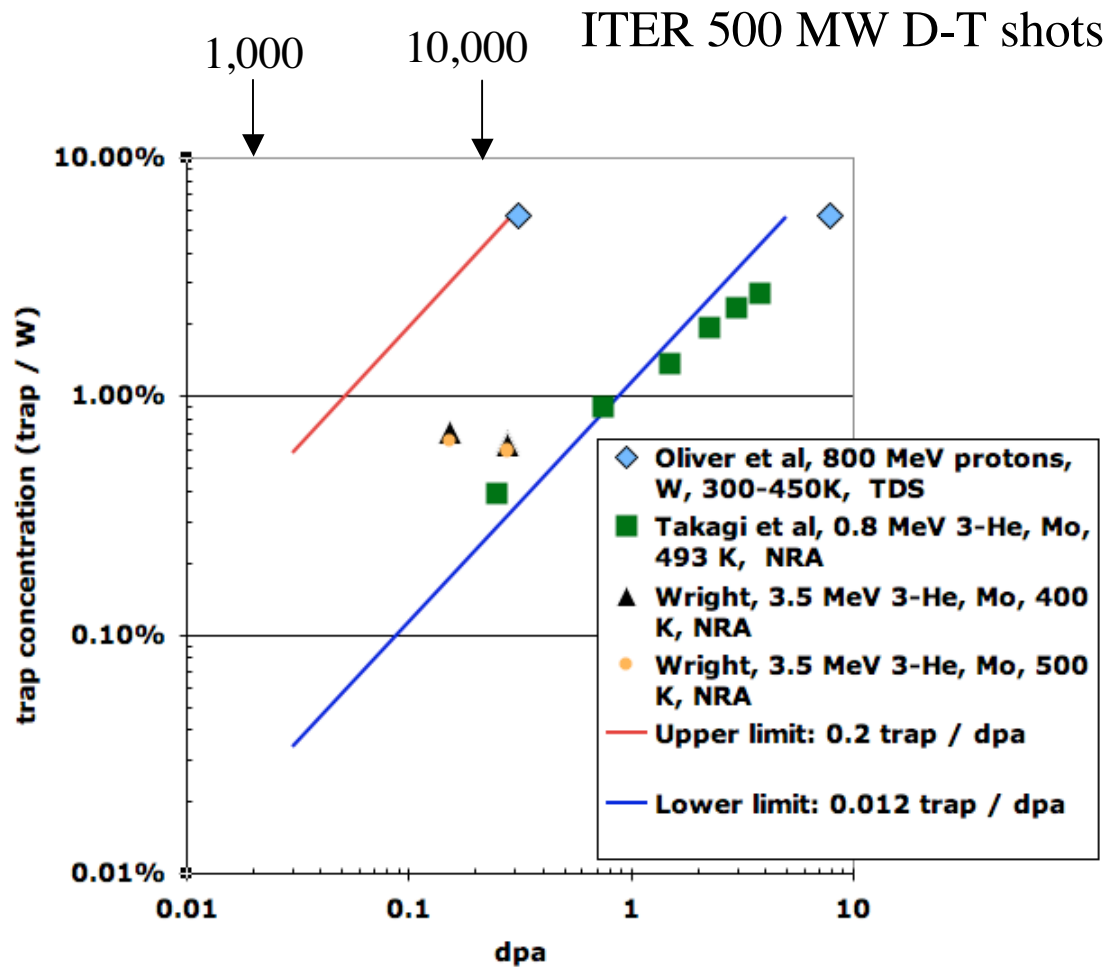
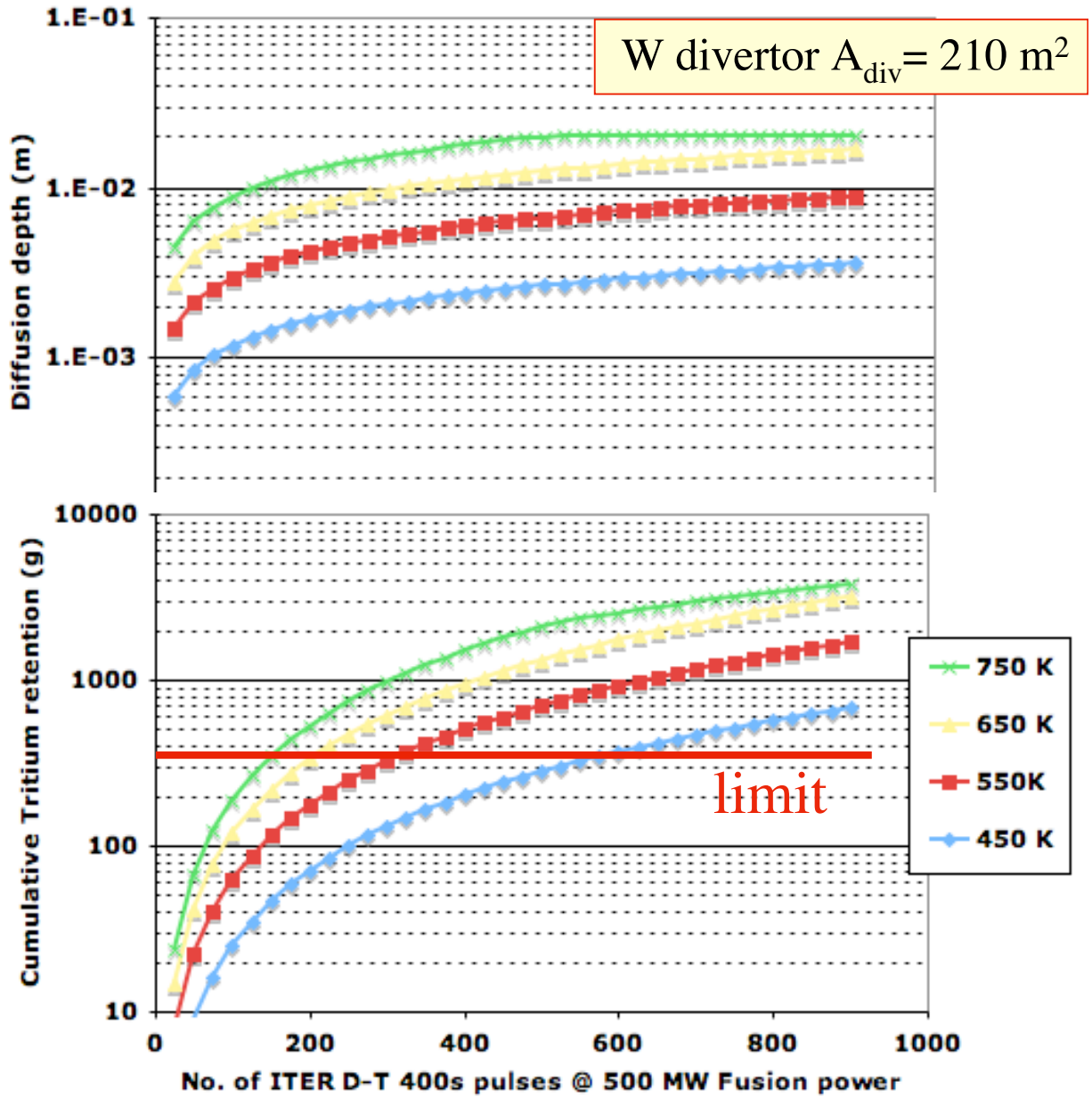
- High temperature MUST heal these traps since in a D-T reactor ~ 1 kg Tritium can be stored in (Traps / atom) ~ 10^{-4}
  - Surpassed in a few hours in Demo due to high permeability of the D/T in tungsten at high ambient temperature.



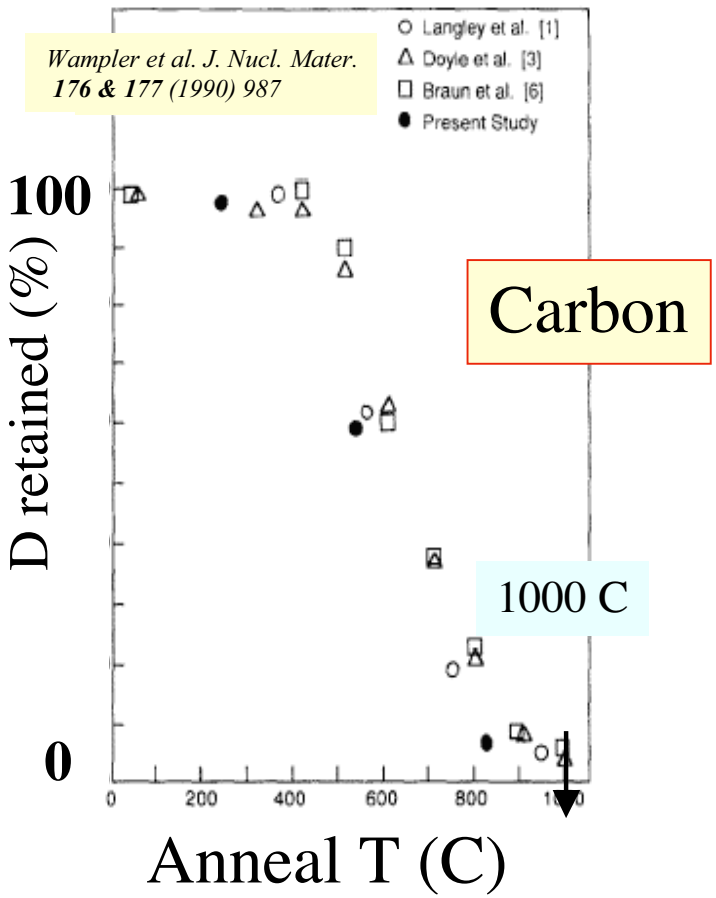
[1] Oliver et al J. Nucl. Mater. ***307-311*** *(2002) 1418.*
[2] Takagi et al. Fus. Sci. Tech. ***41*** *(2002) 898.*
[3] Wright, G, Ph.D. thesis, submitted to J. Nucl. Mater.

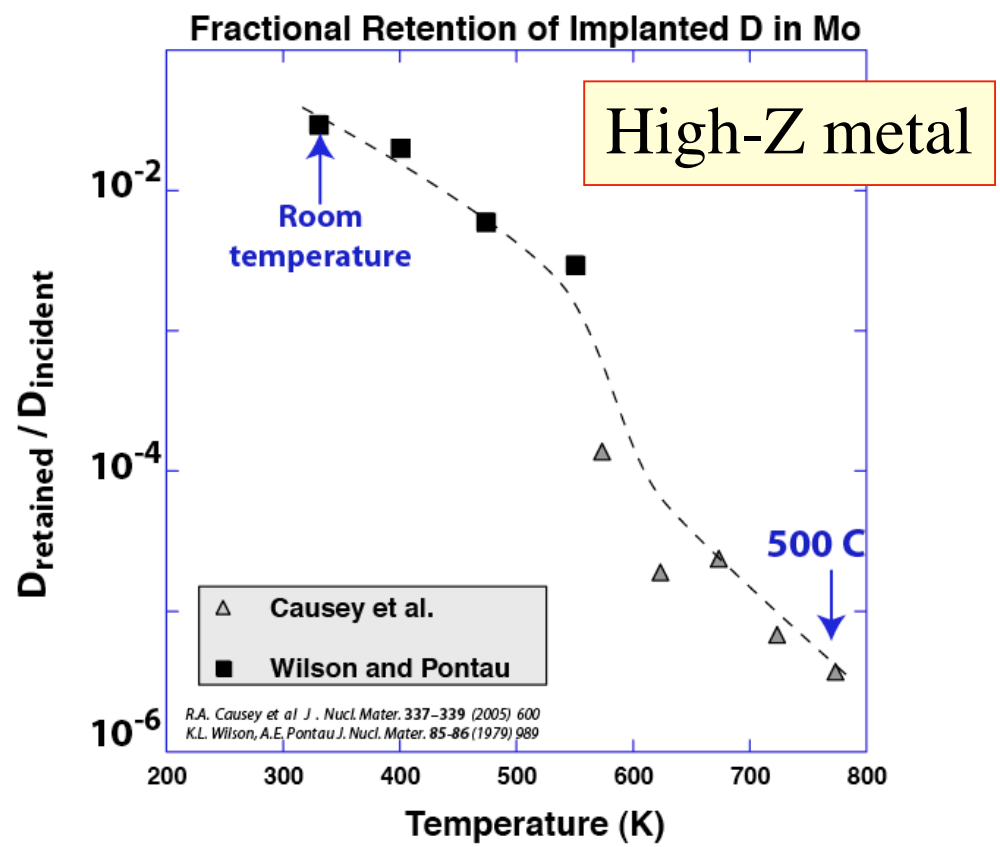
Simplest diffusion analysis + volumetric trap production rate illustrates the maximum possible T retention rate in ITER W divertor

- The solute D/T can only gain access to volumetric traps up to the "diffusion depth" $d=(2Dt)^{1/2}$
- **Assumes $n_{D,0} \geq n_{trap}$** , so retention is not limited by incident flux [Stangeby].
- **Limit reached < 1000 shots in all cases.**
- **Illustrates the complexity of understanding "evolving" material properties in a real fusion environment.**



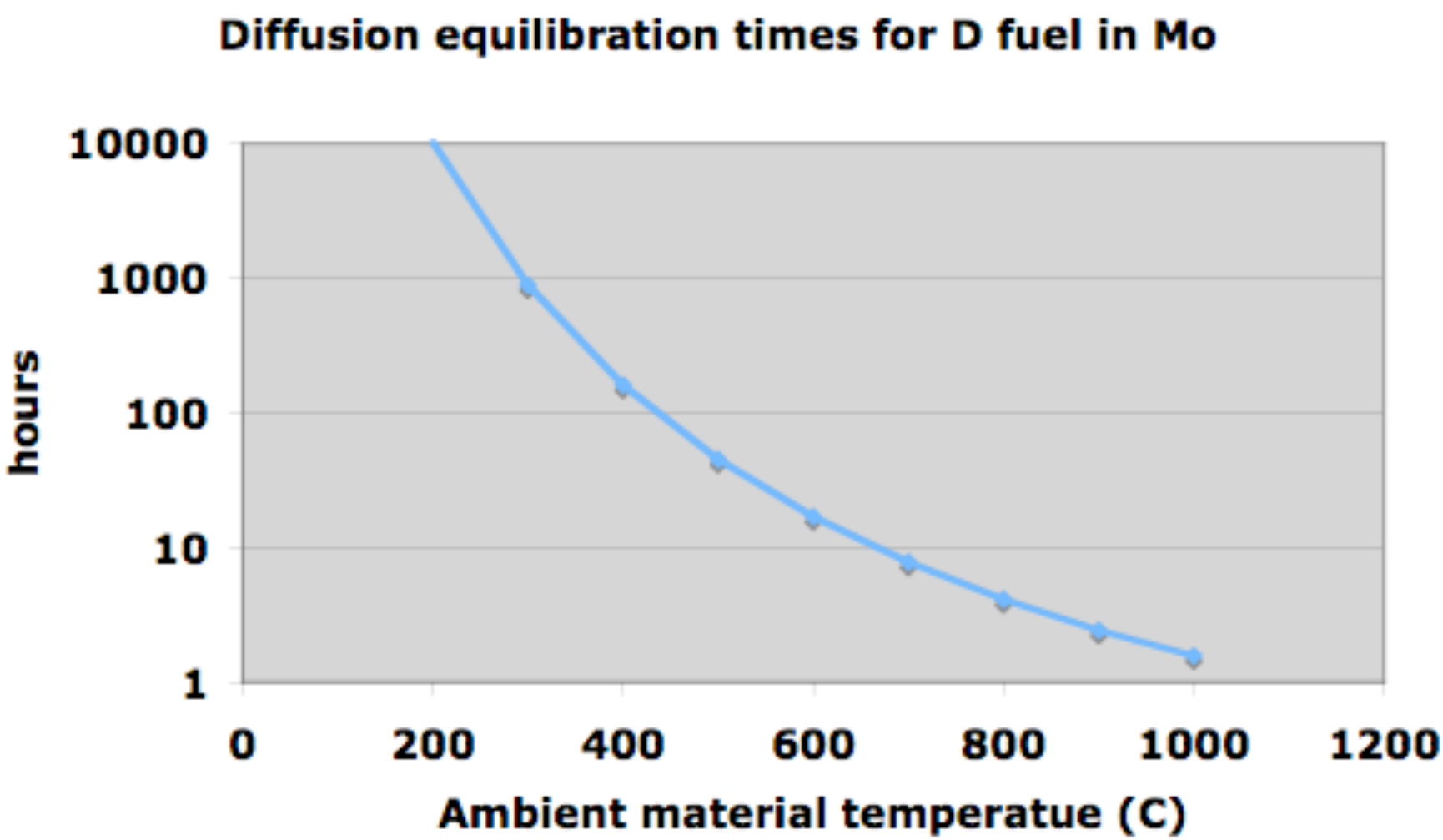
It is hard to overstate the importance of high temperature for fuel control / retention





- Orders of magnitude reduction in fuel retention direct cause of orders of magnitude changes in physical chemistry rates.

High temperature walls are synergistic with steady-state equilibrium mission: e.g. to ensure fuel equilibrium in PFC in "reasonable" time



30+ years of experience in confinement devices tells us we should be worried/excited about all this, and in particular the effects of having "hot walls"

- Every major (and minor) modification to the wall surfaces had profound effects on core performance.
  - E.g. lithium layers (TFTR, NSTX), He discharge cleaning (TFTR, DIII-D, etc), boronizations (DIII-D, C-Mod, etc.), ad infinitum

- **Can we be so naïve that ~10 orders of magnitude modifications to boundary condition of wall will not have profound effects on the core?**

- Must assess effects experimentally but can make educated guesses
  - **Fuelling balance: surface strongly desorbed of H_2 --> no retention?**
  - **Recycling: depleted walls --> ~100% particle albedo?**
  - **Safety: flakes/dust fully T depleted, reactivity?**
  - **Impurity control: ~ ZERO vacuum impurities (H_20, C_2O)**
  - **High-Z materials: fuel permeated through wall, no sputtering?**
  - **Erosion control: Hydrogen activity with materials fundamentally modified.**

Conclusion: The PSI chasms from ITER to Demo are so large we must start to address them now

- Every PSI issue pushes us to our science and technology limits
  - Power density (P/S ~ 1 MW/m^2) x duration (30,000,000 s)
  - Transient energy dissipation: all materials at thermal limits.
  - Erosion and fuel control
  - Evolving quasi-equilibrium due to neutron damage.

- Where do we start?
  - We must assess materials in the proper physical chemistry range (T_{wall})
  - Establishing quasi-equilibrium between PFC materials and plasma will be a key scientific advancement.
  - A new generation of innovative in-situ PSI diagnostics are essential.

- The fusion community must realize this is not a technology issue, but rather a grand "fusion science" challenge since it deals with nearly every aspect of plasma and material science.