First Wall Issues:
ITER to DEMO

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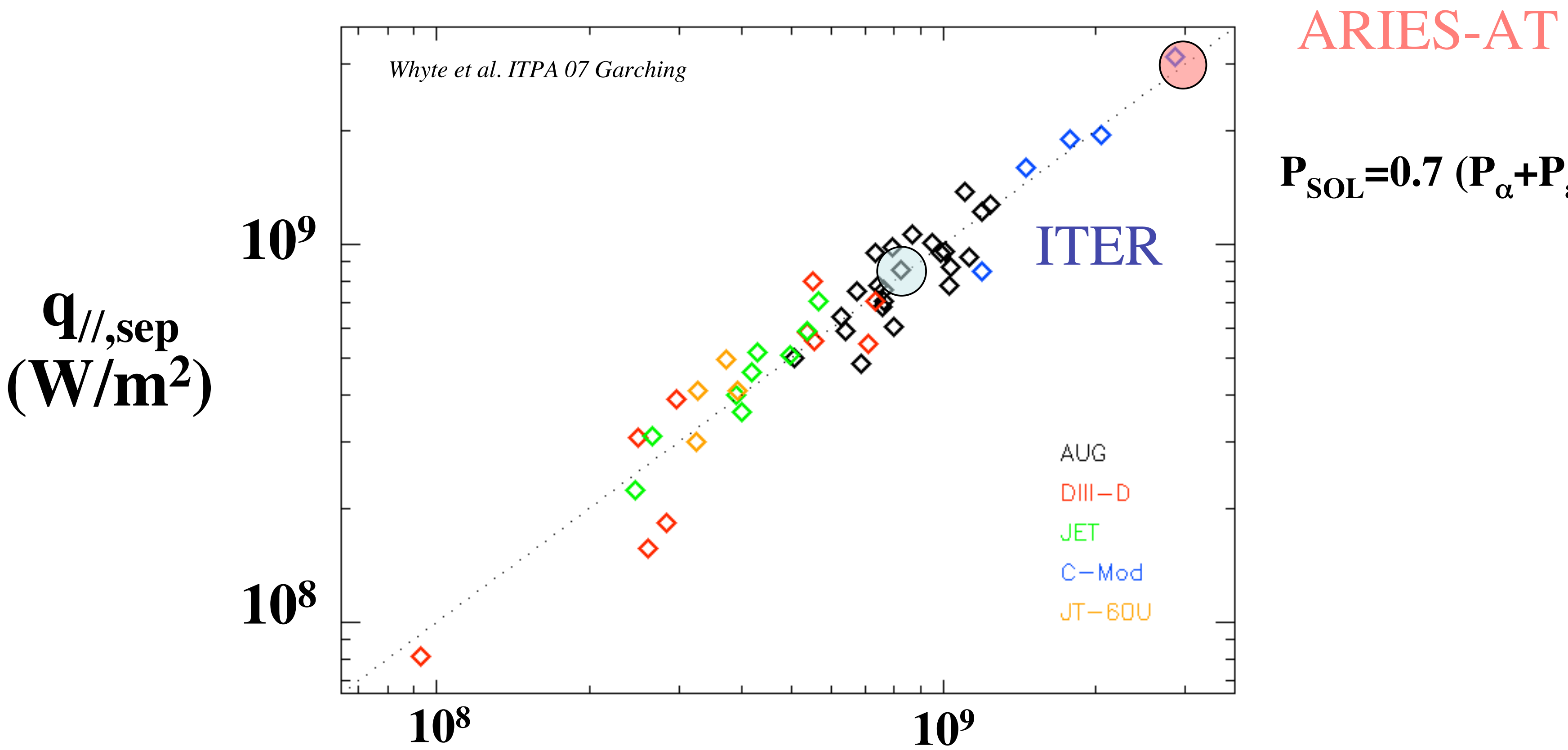
First-Wall & PSI becoming increasingly important and difficult as we move from present tokamaks → ITER → demonstration fusion power plants

Issue / *Parameter*	**Present Tokamaks**	**ITER**	**DEMO**	**Consequences**
Quiescent energy exhaust *GJ / day*	~ 10	3,000	60,000	- active cooling - max. tile thickness ~ 10 mm
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? ~ 60 for C and W - surface distortion
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
Tritium consumption *(g / day)*	< 0.02	20	1000	- Tritium retention in materials and recovery

Upstream peak power exhaust:
SOL Width ~ R leads to $q_{//,max}$ ~ 2-3 GW/m^2 in reactor

Daunting engineering task, but can we access physics in present devices?

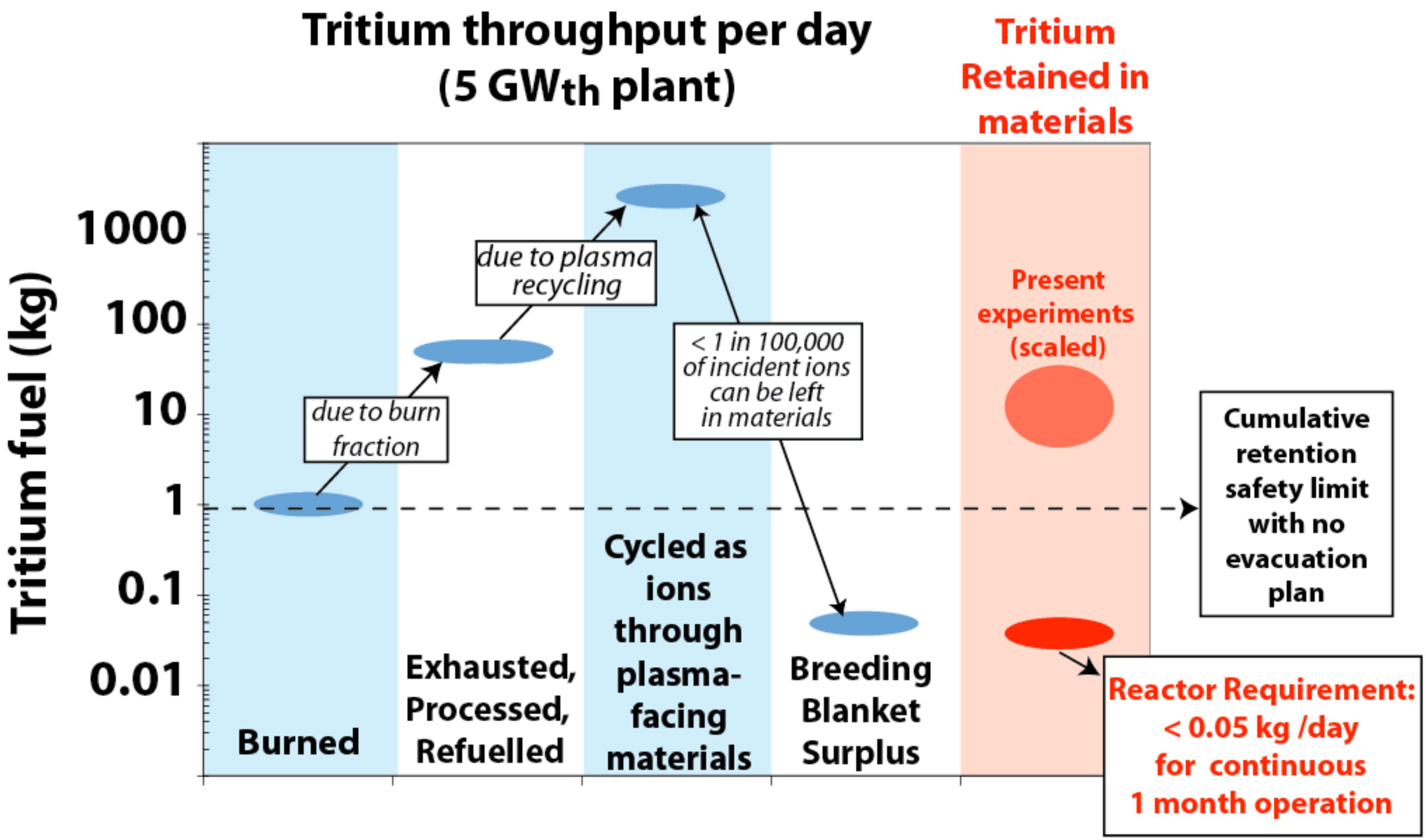
$$q_{//,MAX} \equiv \frac{P_{SOL}}{4\pi R \lambda_{q//}} \left(\frac{B_T}{B_Z}\right)_{sep} \sim \frac{P_{SOL}}{R^2}\frac{q_{cyl}}{\varepsilon} \sim q_{\perp} \cdot q_{cyl}$$



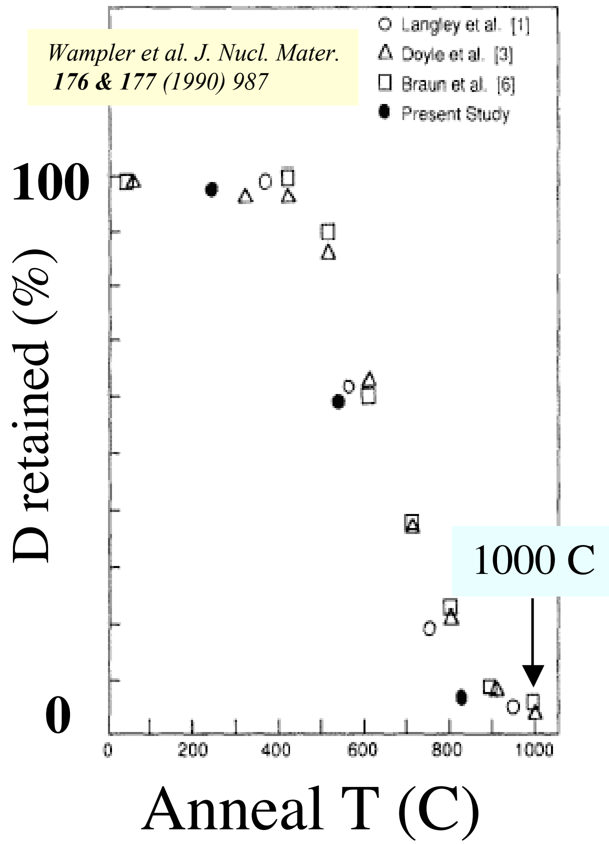
P_{SOL}=0.7 (P_α+P_{aux})

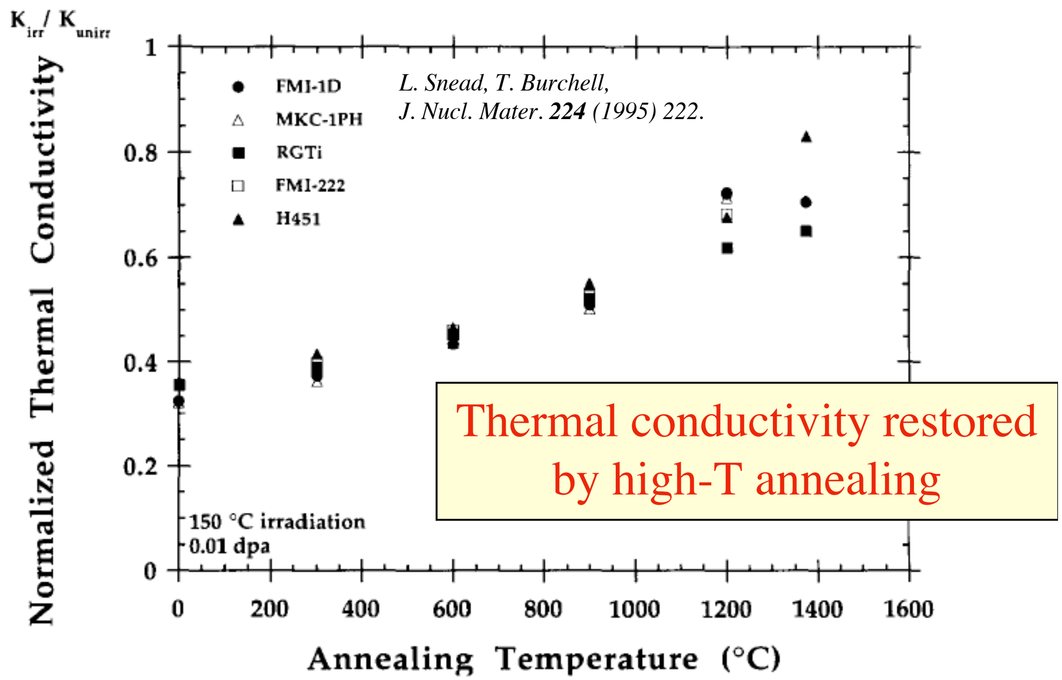
$$q_{//,sep} = 1.43\text{x}10^4\ q_\perp^{0.86}\ q_{cyl}^{1.03}$$

Tritium/fuel control and retention represents the single largest step between present devices/ITER and a DEMO



High materials temperatures seem mandatory to control fuel retention and anneal neutron damage:
What temperature will be required?





JT-60U tokamak (Tanabe et al. ITPA Toronto 06)

Film Location	T_{max} (K)	(H+D) / C
Front-face	>1000	0.01
Behind tile	420	~ 1

Both long pulse AND high temperatures required to reach true particle equilibrium in first wall.

Alcator C-Mod

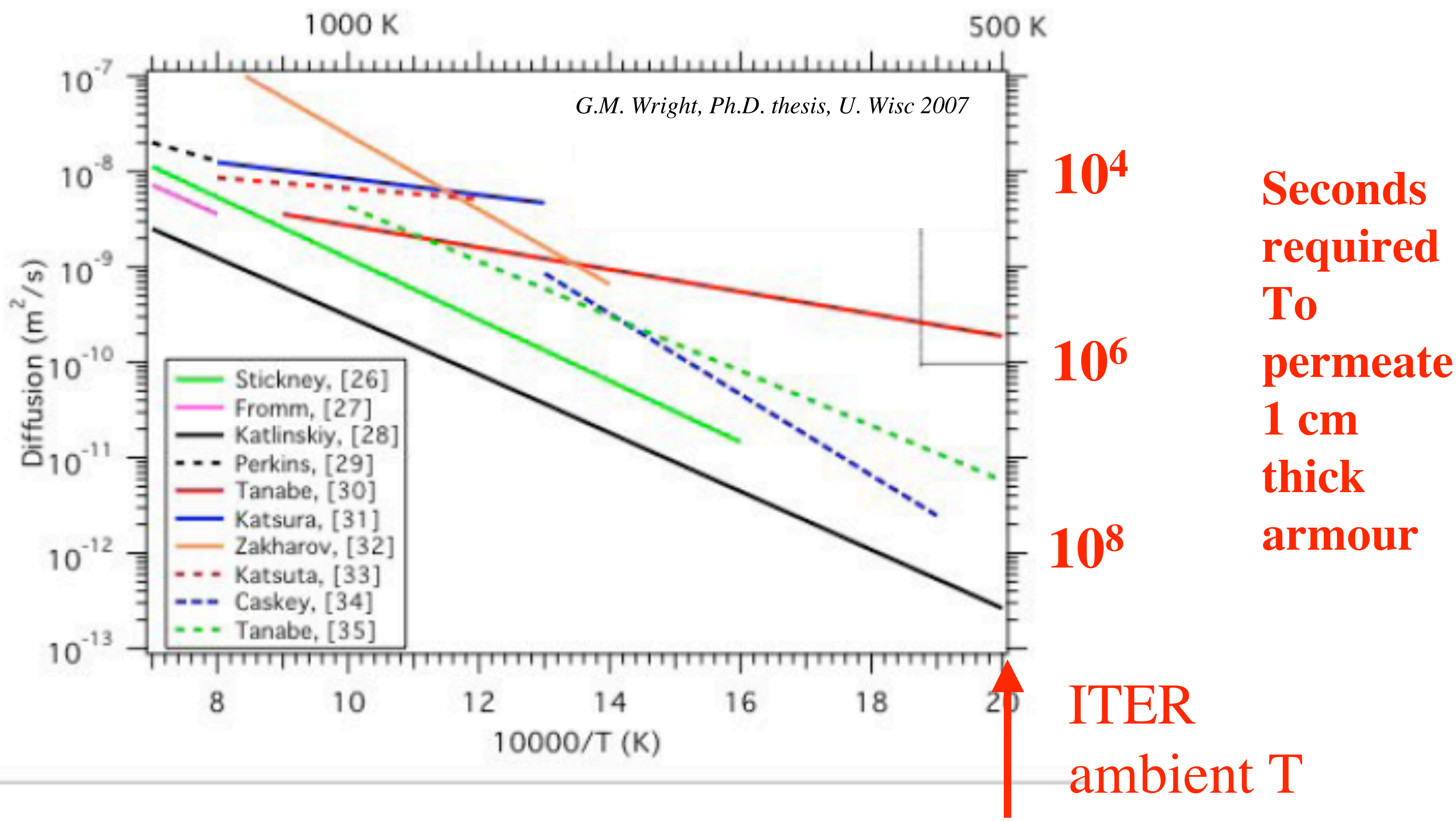


Figure 2.1 A collection of experimental fits for the diffusivity of hydrogen/deuterium in Mo as a function of temperature.

Summary

- The requirements for a sustainable fuel cycle and wall viability push us strongly to investigate first walls with much higher ambient temperatures
  - Tritium recovery, suppression of retention in codeposits.
  - Walls and particles in equilibrium
  - Annealing of neutron damage
  - Technological requirement: gas cooling + peak heat load removal

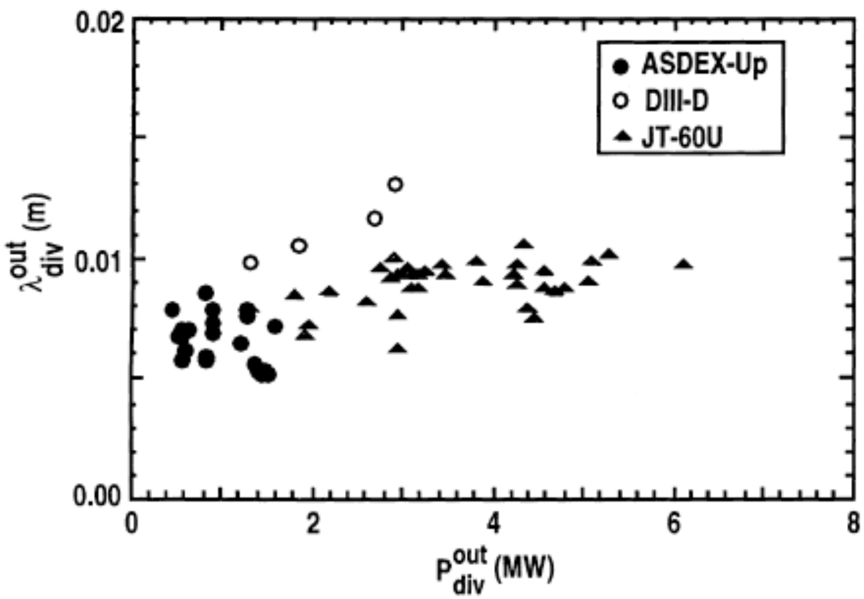
- A key physics / design issue will be the selection of the ambient temperature that provides the appropriate trade-offs between temperature limits (e.g. heating from transients), desired high power density, and high temperature benefits such as tritium recovery.
  - Exponential T dependences demand experimental demonstration / testing.

- "Hot-wall" tokamak operation would investigate a physics, operational and technological path that seems vital to fusion's success, but which no one else, including ITER, is pursuing.
  - Must be coupled with a concentrated effort on PSI science and diagnosis + sustainment physics.

Additional Materials

Cross-device study that showed no dependence of λ_q with R, also revealed a scaling of $\lambda_q \sim P^{0.5}$ that results in favorable extrapolation to ITER & reactors

Alcator C-Mod



Scaling H-1 (with measured divertor power)

$$\lambda_q^{\text{H-1}}(\text{m}) = (5.2 \pm 1.3)10^{-3} P(\text{MW})_{\text{div}}^{0.44\pm0.04} B(\text{T})_{\phi}^{-0.45\pm0.07} q_{95}^{0.57\pm0.16}$$

ARIES-AT: $q_{peak} \sim 6$ MW/m^2

Table 1
Extrapolated values of power flux peak and width for ITER-EDA from the scalings derived in this paper

Regime	$\langle n_e \rangle$ (10^{19} m^{-3})	P_{net} (MW)	P_{div} (MW)	λ_q (cm)	$P_{\parallel}^{peak}$ (MW/m^2)	P_{div}^{peak} (MW/m^2)
L-mode (L-1)	5.0	100	40	$2.7 \begin{pmatrix} +0.8 \\ -0.8 \end{pmatrix}$	$61 \begin{pmatrix} +26 \\ -14 \end{pmatrix}$	$2.2 \begin{pmatrix} +0.9 \\ -0.5 \end{pmatrix}$
L-mode (L-2)	5.0	100	40	$2.1 \begin{pmatrix} +0.9 \\ -0.7 \end{pmatrix}$	$79 \begin{pmatrix} +40 \\ -24 \end{pmatrix}$	$2.9 \begin{pmatrix} +1.5 \\ -0.9 \end{pmatrix}$
H-mode (H-1)	10.0	200	50	$2.5 \begin{pmatrix} +0.2 \\ -0.2 \end{pmatrix}$	$83 \begin{pmatrix} +7 \\ -6 \end{pmatrix}$	$3.0 \begin{pmatrix} +0.3 \\ -0.2 \end{pmatrix}$

A. Loarte et al. / Journal of Nuclear Materials 266–269 (1999) 587–592

Target heat removal is the highest priority in edge design, but the strategy forward is confused by the lack of consistent or compelling empirical scalings

Alcator C-Mod

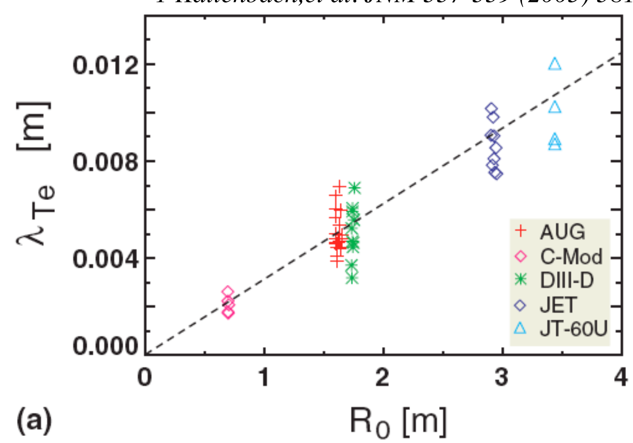
Table 1. Scalings for dependence of SOL heat flux width and peak target heat flux with total target (or divertor) power load for a number of machines and multi-machine databases.

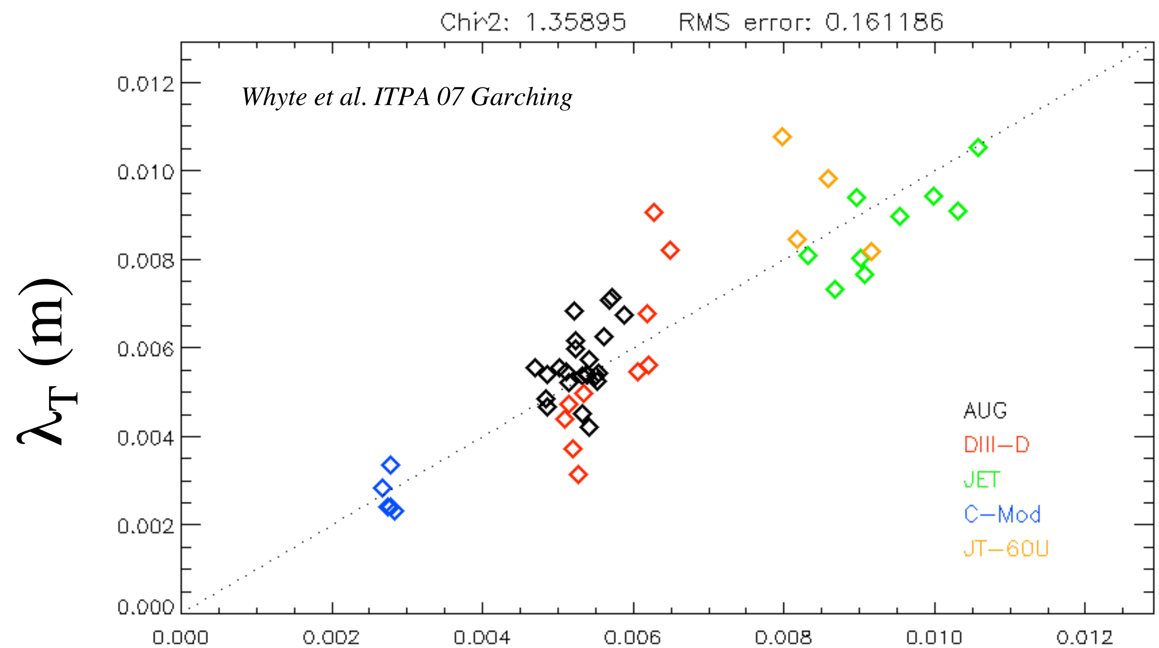
	λ_q	$q_{\max}$	Comment
Multi-machine: JT60-U, DIII-D and ASDEX-Upgrade (DIVI) [28]	$P_{\mathrm{div}}^{0.44\pm0.04}$	—	Scaling for λ_q mapped to outer mid-plane
Multi-machine: JT60-U, ASDEX-Upgrade [1]	$P_{\mathrm{target}}^{0.35\pm0.05}$	—	—
ASDEX-Upgrade (DIVI) (IR data) [3]	$P_{\mathrm{target}}^{0.52\pm0.05}$	$P_{\mathrm{target}}^{0.5\pm0.05}$	Type I and III ELMs included and partially detached plasmas.
ASDEX-Upgrade (DIVII) (IR data) [3]	$P_{\mathrm{target}}^{-0.1}$	$P_{\mathrm{target}}^{1.1\pm0.06}$	Type I ELMs and attached plasmas only.
DIII-D [30]	—	P_{div}^{1}	—
JET (IR data) [31]	$P_{\mathrm{target}}^{-0.13\pm0.08}$	P_{target}^{1}	Inter-ELM. λ_q from FWHM of profile at outer target.
JET (TC data) [18]	$P_{\mathrm{target}}^{-0.48\pm0.09}$	—	ELM-averaged. D (Type I ELMs) and He (Type III ELMs) discharges included. λ_q from the integral width of profile at outer target.

"All than can be strongly concluded from Table 1 [scalings for SOL heat flux] is that there is a need for improved experimental measurements and a theory-oriented approach for making extrapolations for the target heat flux.."
Tokamak Physics Update: Power and Particle Control, A. Loarte, et al. Nucl. Fusion **47** (2007) S203.

As Kallenbach[1], we find $\lambda_T \sim R$. Further regression analysis shows λ_T is invariant with P_{SOL} and insensitive to other global and separatrix parameters

1 Kallenbach,et al. JNM 337-339 (2005) 381.



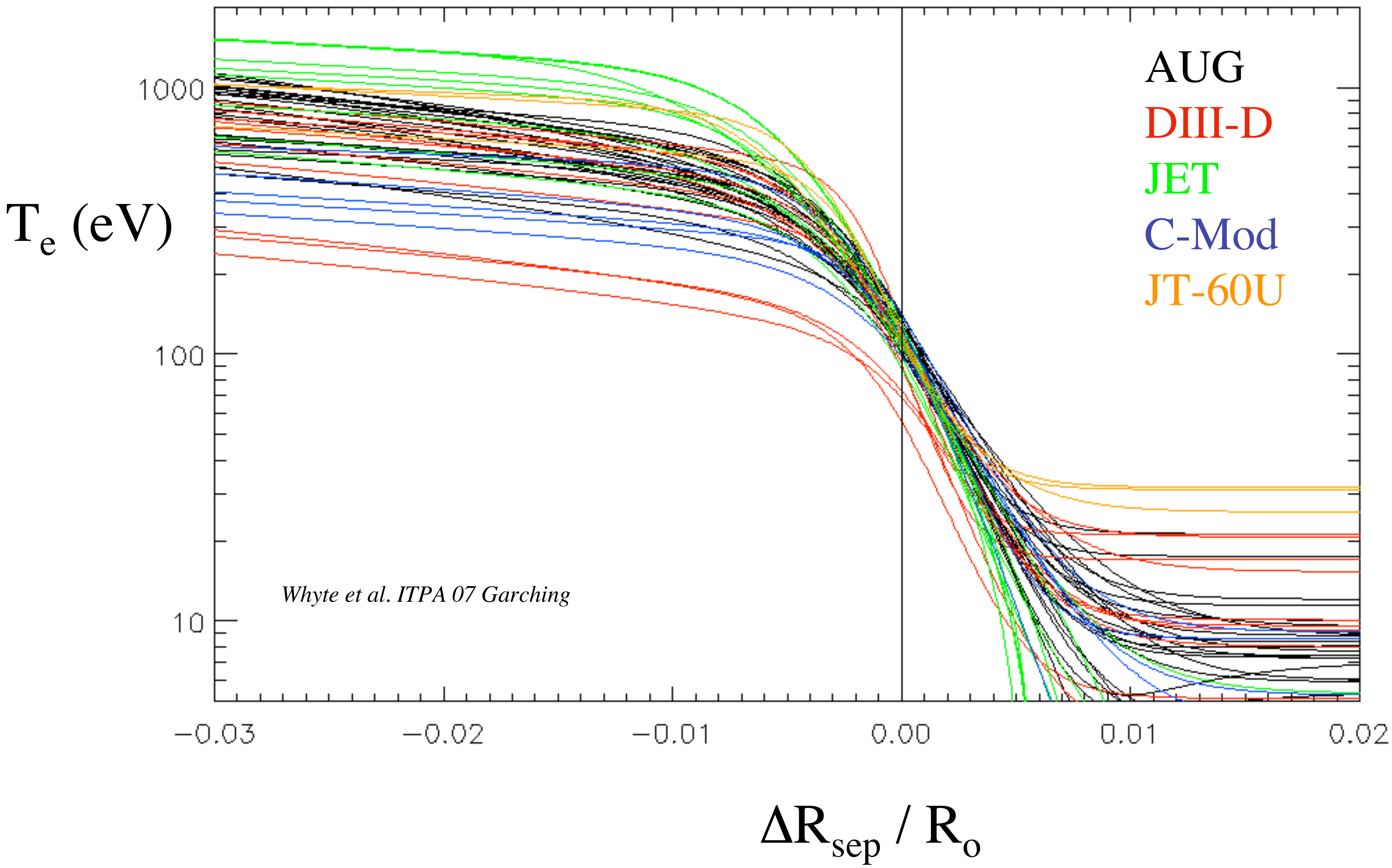


$$\lambda_T \sim R^{1.1\pm0.1}\ q_*^{0.15\pm0.2}\ P_{SOL}^{0.01\pm0.05}\ n_{sep}^{0.2\pm0.06}$$

The invariance with P_{SOL} counter-indicates the scaling expected from $\perp$ conduction

$$\lambda_T = C_0 \cdot \left[\left(f_{shape} \right) \left(\frac{B_T}{B_Z} q_* \right)^{2/7} \right]^{7/9} \left(n\chi \right)^{7/9} \cdot \left(R^{14/9} \right) \cdot \frac{1}{P_{SOL}^{5/9}}$$

The separatrix seems to exhibit a critical gradient scale length set by R

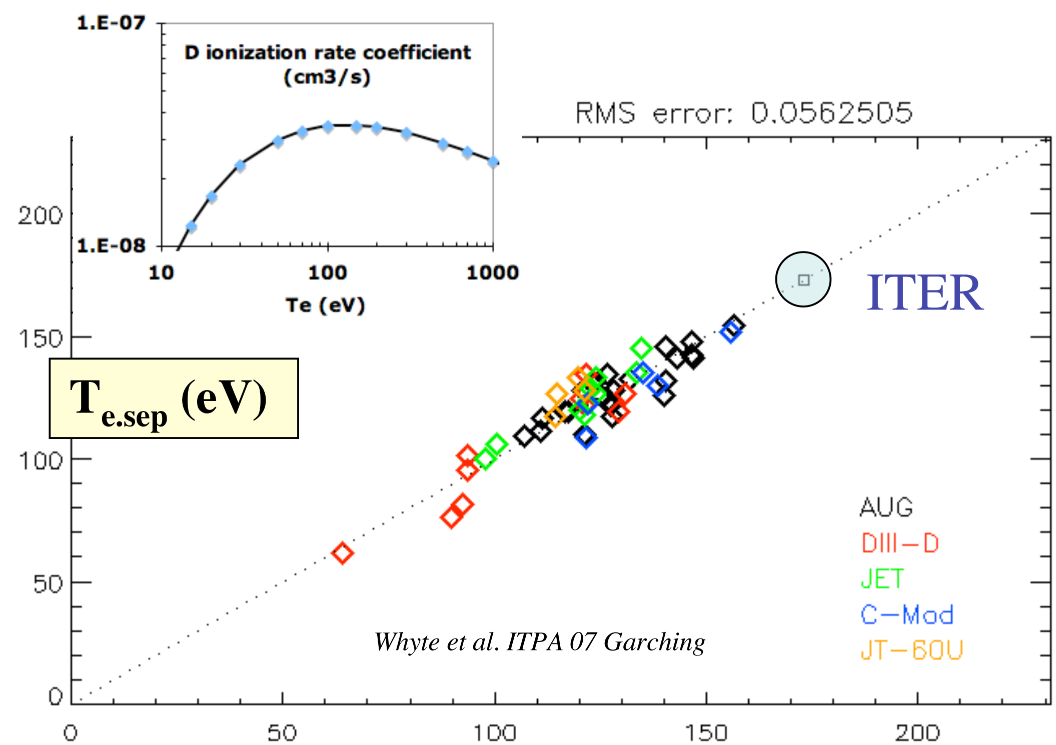


The scaling gives $(P/R)^{2/7}$ as the figure-of-merit to set upstream separatrix T_e

$$T_{e,sep} \cong \left(\frac{49}{16\kappa_{e,0}} \frac{P_{SOL}}{\lambda_T} \left(\frac{B_T}{B_Z} \right)_{sep} q_* \right)^{2/7} \sim \left(\frac{P_{SOL}}{R} \right)^{2/7}$$

- Consistent with original derivation by Lackner[1], who argued T_e / E_{atom} should be matched for edge similarity if SOL width scaled as R.

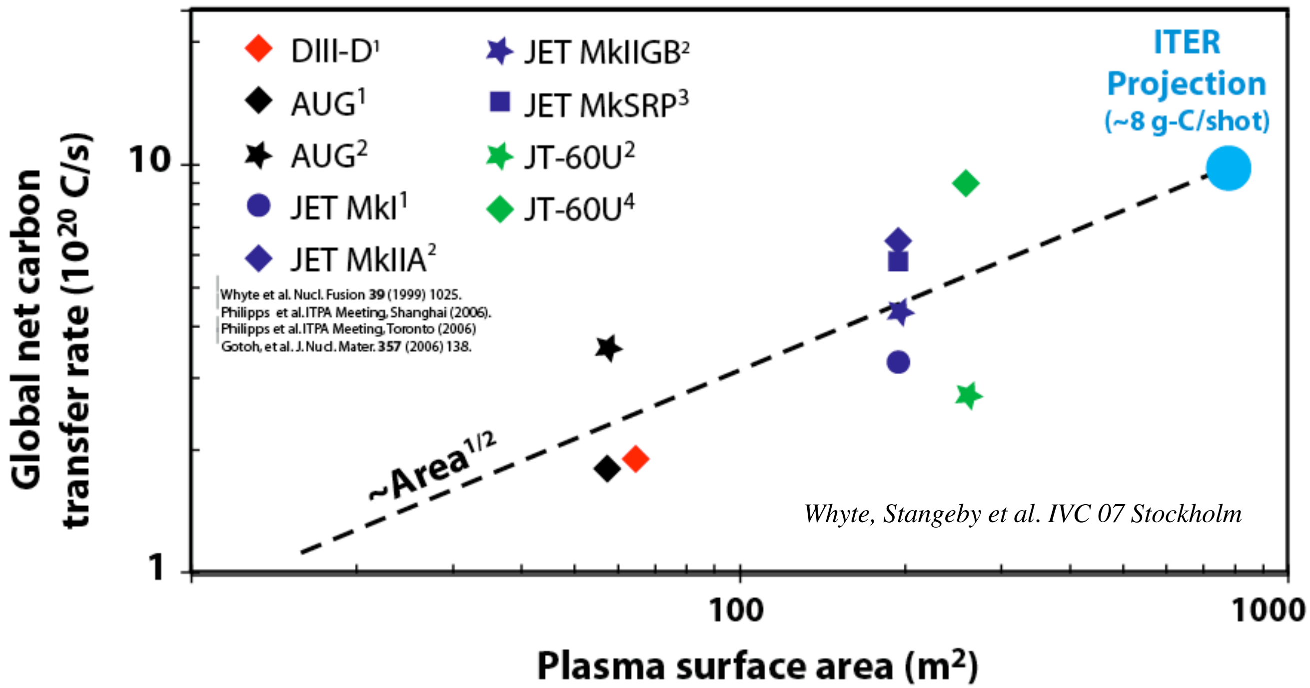
- Upstream atomic rates vary weakly with T, e.g. D ionization rates match within few percent between present devices & ITER..
    - Upstream atomic physics constraint can probably be dropped from similarity requirements, as it is in core.

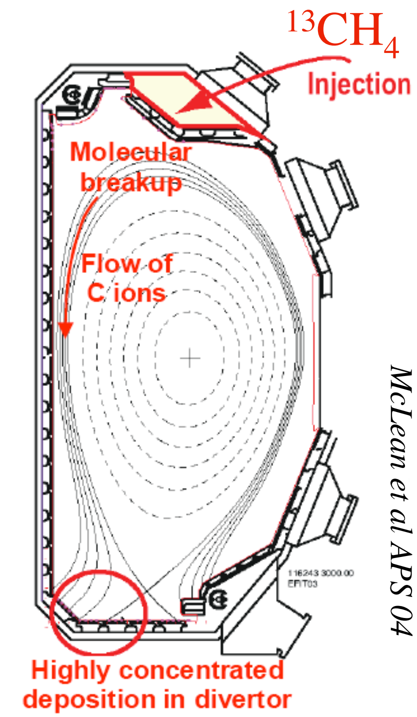


$$T_{e,sep} = 1.09\, (P_{SOL}/R)^{0.28\pm0.02}\ q_{cyl}^{0.6\pm0.07}$$

[1] *Comm. Plasma Phys. Controlled Fusion* **15** (1994) 359.

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





- 13-C isotope tracer experiments support idea that C is transferred from main-wall "limiters" to inboard divertor
- Controlling mechanisms of main-wall erosion sources, long-range transport and deposition balance are not well understood.

Alcator C-Mod

Operational consequences of 10^{21} C/s global transfer rate demonstrate necessity for high ambient temperature to control Tritium retention in carbon films.

<table>
<tr><th>Ambient wall temperature</th><th>T/C</th><th># Pulses to reach 350 g T limit in ITER*</th><th># days to T-limit in DEMO [1]</th><th># days lifetime for “limiters” in DEMO [2]</th></tr>
<tr><td>~ 400 K</td><td>~ 0.3</td><td>500</td><td>~ 6</td><td rowspan="2">~240</td></tr>
<tr><td>> 1000 K</td><td>~ 0.01</td><td>>10^4</td><td>~ 230</td></tr>
</table>

Whyte et al. IVC07 Stockholm

* ITER is water-cooled
$T_{ambient}$ ~ 400K

1 Assumes 1 kg safety limit
2 Assumes 20 m^2 limiter surface

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

Neutron induced shrink/swell in N3M graphite