NSTX-U 5-Year Plan for Materials and Plasma Facing Components

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NSTX-U 5-Year Plan Review
PPPL – B318
May 21-23, 2013

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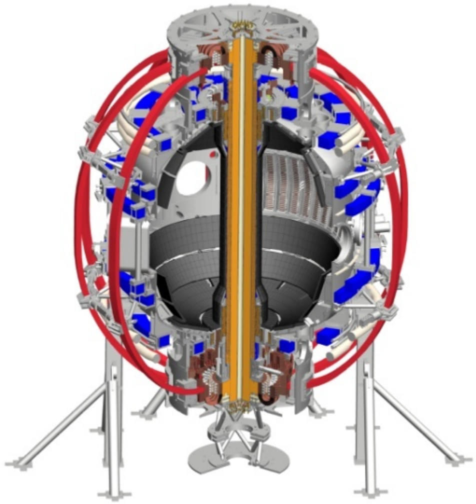

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High-level goals for NSTX-U 5 year plan

(see Program Overview Talk)

1. Demonstrate 100% non-inductive sustainment at performance that extrapolates to ≥ 1MW/m^2 neutron wall loading in FNSF

2. Access reduced ν^* and high-β combined with ability to vary q and rotation to dramatically extend ST physics understanding

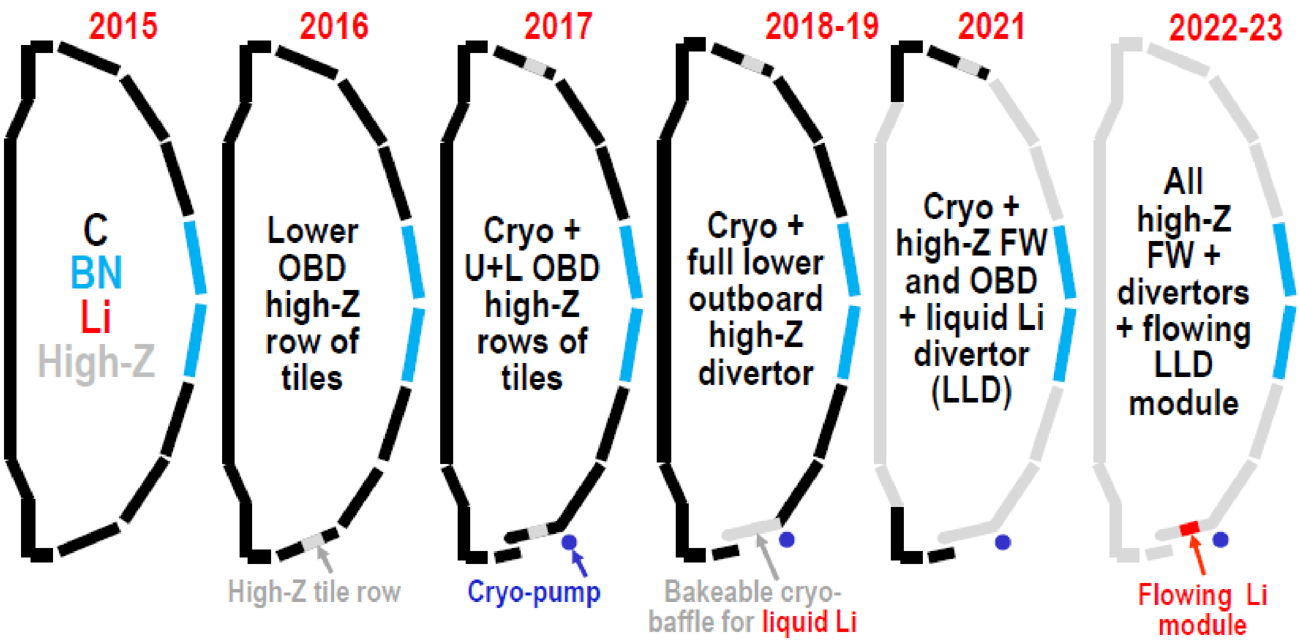
3. Develop and understand non-inductive start-up and ramp-up (overdrive) to project to ST-FNSF with small/no solenoid

4. Develop and utilize high-flux-expansion "snowflake" divertor and radiative detachment for mitigating very high heat fluxes

5. Begin to assess high-Z PFCs + liquid lithium to develop high-duty-factor integrated PMI solutions for next-steps

NSTX-U long-term objective is to perform comparative assessment of high-Z and liquid metal PFCs (see Maingi talk)

- Conversion to all-metal PFCs provides opportunity to examine role of PFCs (including liquids) on integrated scenarios with good core, pedestal, divertor
- NSTX-U has two emphases for addressing power exhaust and PMI issues for next step devices
  - Magnetic topology, radiative divertor
  - **Self-healing/replenishable materials (e.g. liquids)**
- Significant uncertainties in both solid- and liquid-PFCs motivates parallel research



M&P research will develop understanding of material migration and heat-flux handling of high-Z and liquid Li PFCs

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  - Assess impact of more complete Li coverage
  - Use the new Material Analysis and Particle Probe (MAPP) and laboratory studies to link tokamak performance to PFC surface composition
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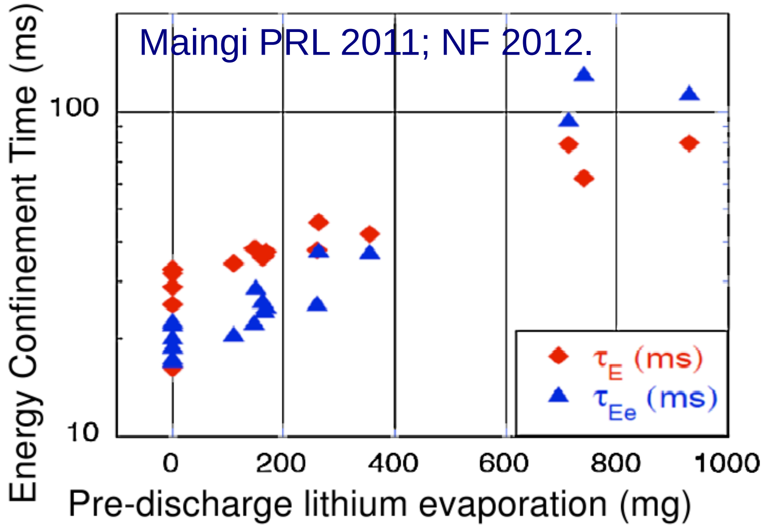
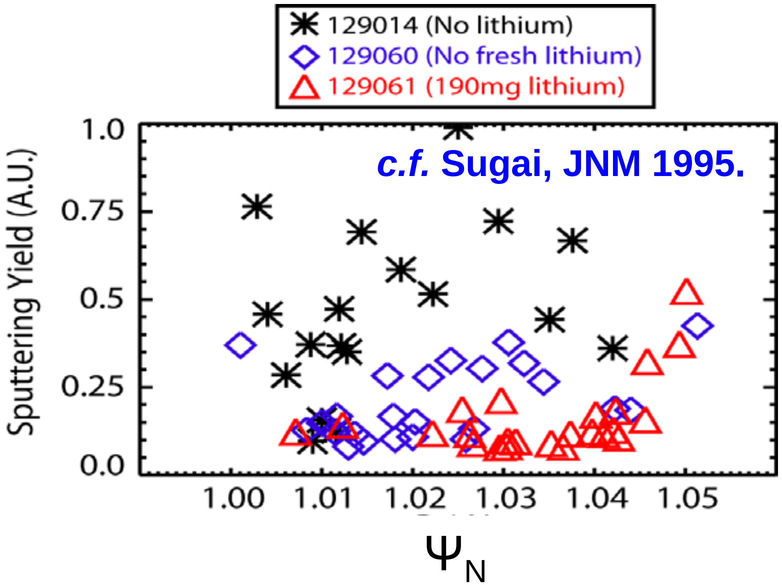
Li wall conditioning results in confinement improvements and reduction in divertor carbon source

- Confinement increases continually with pre-discharge lithium application with downward facing evaporators

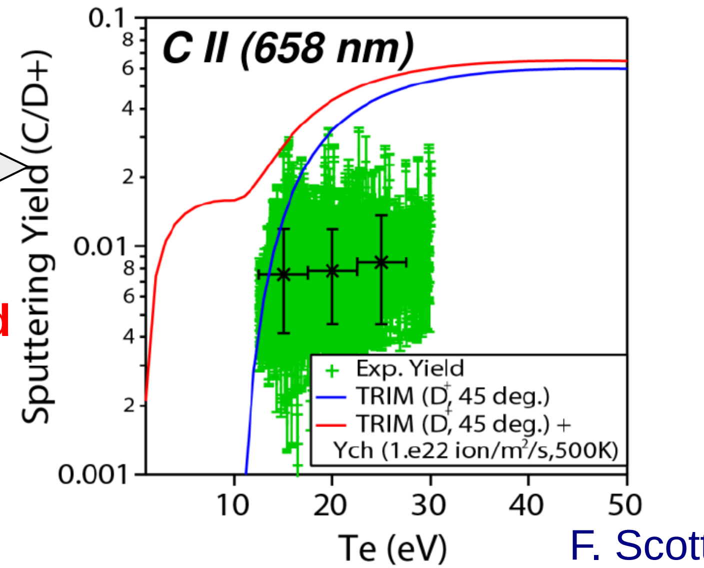
- Carbon influx in divertor reduced after application of lithium conditioning





Plasma spectroscopy and target Langmuir probes provide gross yield

Carbon sputtering yield with lithium

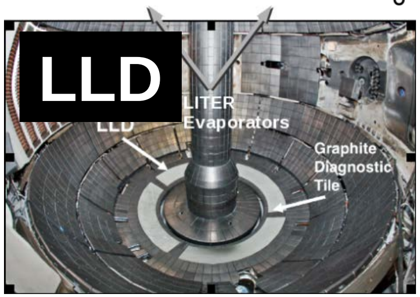


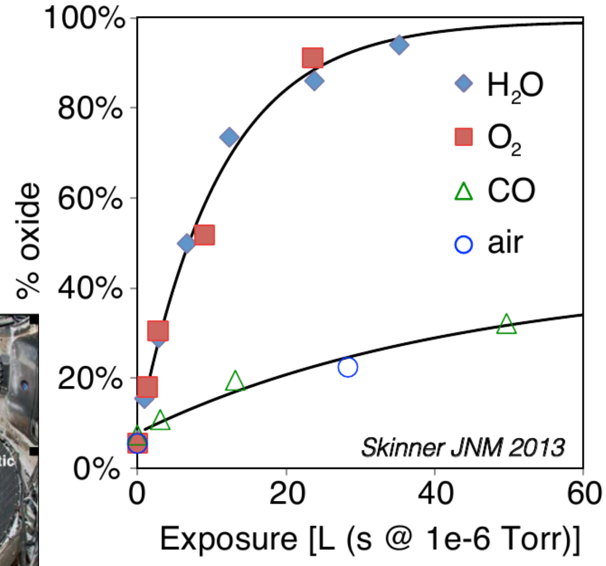
F. Scotti

Oxygen recently identified as important to lithium chemistry and sputtering

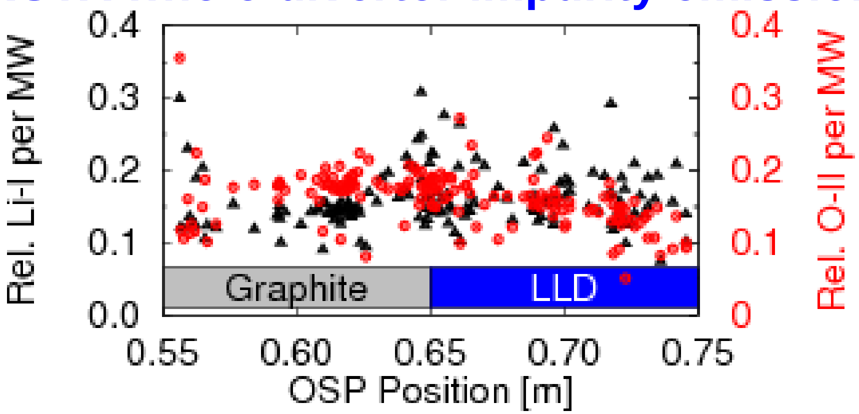
- Oxygen uptake by lithium films quantified in laboratory experiments
  - Oxide layer formation in ~200s in NSTX (~600s inter-shot time)
  - Consistent with Liquid Lithium Divertor (LLD) results showing little change in impurity emission
- Influence of oxygen contaminants being investigated in laboratory
  - Molecular dynamics simulations of Li-C-O show increased D uptake (Krstic, PRL 2013)
  - Non-zero **oxygen** sputter yield from contaminated surfaces
  - No indications of O degrading plasma performance (so far)
- Rapid impurity accumulation motivates flowing systems

Oxygen uptake by Li coating on Mo





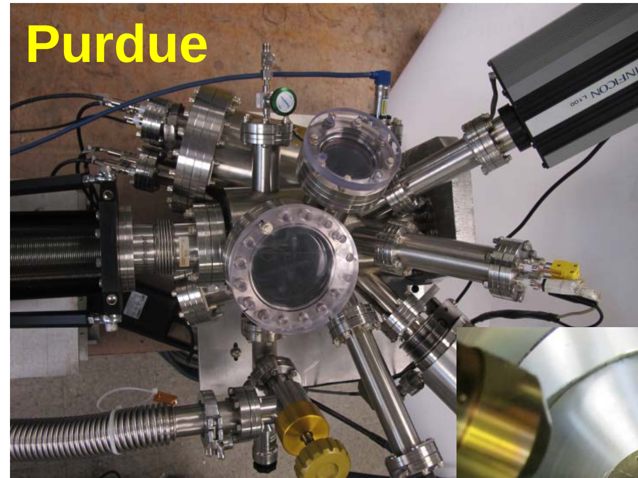
NSTX whole-divertor impurity emission



Jaworski, IAEA 2012

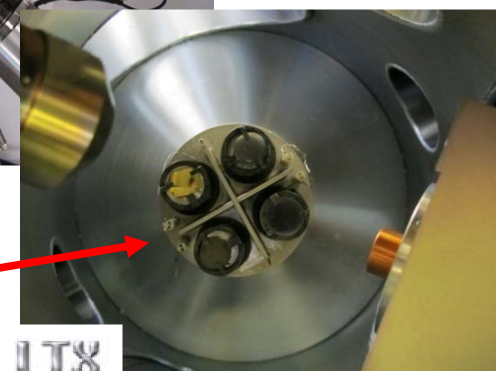
MAPP will be a key diagnostic for bridging the gap between discharge performance and lab-based surface science

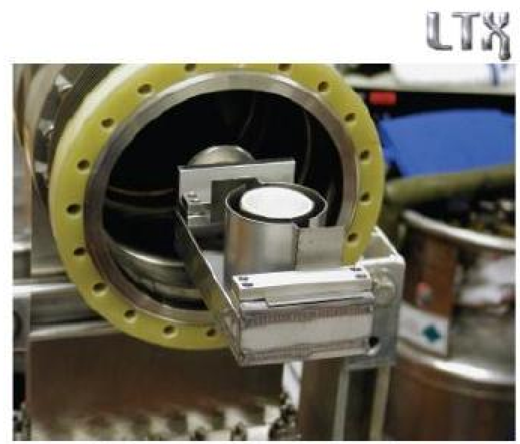
- **M**aterial **A**nalysis and **P**article **P**robe will determine material composition and surface chemistry inside tokamak
  - Exploit MAPP capabilities to link with surface science labs at PPPL, Purdue, U-Illinois
  - Identify role of contaminants in Li PMI
  - Being prepped for use in LTX this year
- Optimize areal coverage of wall conditioning techniques
  - Diffusive evaporations and upward-facing evaporators
  - Examine energy confinement, impurity production, particle control...



MAPP capabilities
TDS
XPS
LEISS
DRS

Up to 4 samples exposed





Y_2O_3 crucible, Ta heater
➢Tested to 700 °C

Upward evaporator under development for NSTX-U

Thrust MP-1 research plan will use multiple tools to unravel lithium surface-science during transition to high-Z PFCs

- FY14 – Surface science laboratory and LTX experiments
  - Utilization of MAPP to examine high-Z substrates in LTX
  - Measurements of deuterium retention in lithium on high-Z substrates in laboratory experiments (PU collaboration)
- FY15 – Comparison of boronization with more complete PFC coverage by Li, establish baseline performance data sets
  - Diffusive evaporation and MAPP to identify surface chemistry changes during transition to lithium wall conditioning
  - Prepare high- and low-triangularity discharges to prep for high-Z tiles
- FY16 – High-Z tile installation, upward evaporation
  - Examine role of evaporation *rate* on Li efficacy
  - Determine how high-Z substrate affects coating performance
- FY17-18 – High-Z tiles and vapor shielding
  - Determine high-temperature (T>500C via plasma heating), coated PFC performance and role of impurities in lithium PMI throughout machine

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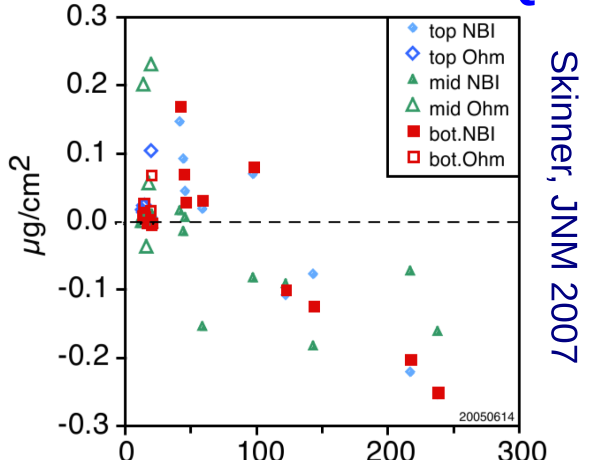
Complementary erosion diagnostics will enable identification of erosion/redeposition patterns

- Simple scaling estimates indicate factor of 10 increase in wall erosion in NSTX-U

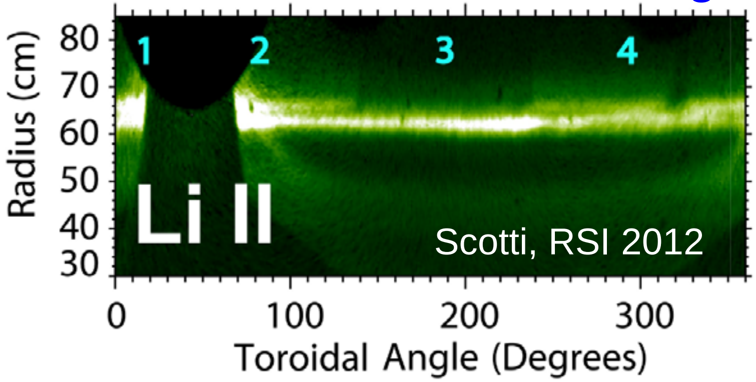
$$\Gamma^{Gross}_{sputt.} \approx Y_{sputt,cx} \frac{f_{cx} P_{heat} / E_{cx}}{S_{plasma}}$$

- NSTX-U will expand coverage of erosion diagnostics
  - QCMs and witness samples at multiple first-wall locations (4x poloidal locations, 20 witness plates in NSTX)
  - Marker tiles used in high-heat flux regions
  - Upgraded MAPP with QCM will measure mass and composition simultaneously
  - **Net erosion** diagnosed with mixture of inter-shot and campaign-integrated diagnostics
- Expanded suite of plasma diagnostics to constrain plasma models
  - Langmuir probes for local n_e, T_e, potential
  - **Gross erosion** via plasma spectroscopy

Mass-loss measured on QCMs



Full divertor camera coverage



Thrust MP-2 research will quantify material erosion/migration on first-wall and in divertor for both high-Z and low-Z surfaces

- FY14 – Data analysis and test-stand experiments to prepare for tokamak experiments
  - Continue analysis of NSTX discharges to optimize diagnostics
  - Magnum-PSI experiments to measure gross and net erosion at high temp.
- FY15 – Make initial assessment of material erosion and migration in NSTX-U discharges, compare B vs. Li discharge conditions
  - Utilize MAPP to measure composition of films, compare to Magnum-PSI material evolution data and models; measure gross and net erosion
  - Prepare for high-Z tile upgrade with low- and high-triangularity data sets
- FY16 – Establish first-wall erosion scalings, protection of high-Z substrate and compatibility with high-performance discharges (e,g. high-Z impurity accum.)
  - Examine dependence on edge neutral pressure, input power, pulse length
  - Determine impact of synergistic operations on material migration (e.g. impurity seeded divertors, snowflake configuration)
- FY17-18 – Determine impact of upper divertor high-Z tiles and the impact of vapor-shielded regime on material migration

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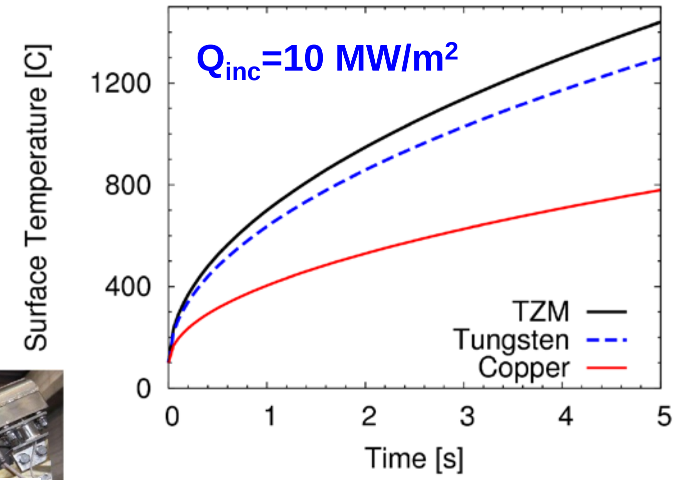
High temperature lithium surface may be able to provide a self-healing surface and intrinsic low-Z impurity radiation source

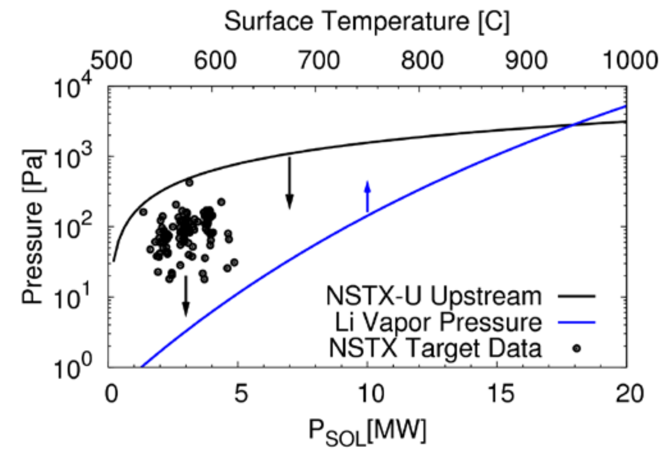
- Lithium vapor cloud can potentially provide effective power and pressure loss
  - Non-coronal Li radiation
  - Li vapor pressure vs. plasma pressure
- Capillary-Porous System targets have dissipated large incident heat fluxes - tested to 25MW/m^2 limited by Li inventory (Evtikhin JNM 2002)
- **What is $T_{max,surf}$ for a lithium PFC?**
- Diagnosis in NSTX-U via complementary diagnostics
  - Langmuir probes for target pressure, n_e, T_e
  - Optical, VUV emission and bolometry for P_{rad}
  - DBIR thermography and TCs for heat flux and energy deposited
- Preliminary experiments already performed at Magnum-PSI (see backup)

Front-face Temperature Rise



$$p_u = p_t(1 + M_t^2) + p_{Li}$$

$$q_t^{plasma} = \gamma \Gamma_{sat}^{+} T_e = \gamma n_{es} c_s T_e = \gamma c_s p_t$$

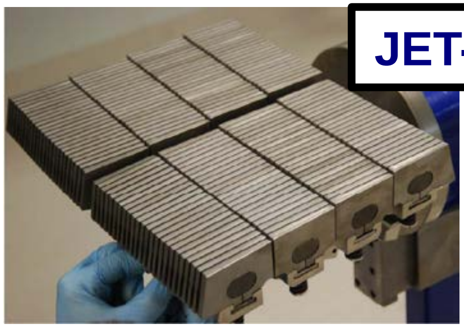


Thrust MP-3 will establish the existence of vapor-shielded regime and assess compatibility with integrated scenarios

- FY14-15 – Assess the high Li influx regime (w/ Thrust 2)
  - Complete high-Z tile design for installation in year 3
  - Conduct high temperature experiments on Magnum-PSI
  - Validate plasma transport and atomic physics data bases for high-density, high-Li fraction plasmas
- FY16 – Validate high-Z substrate design and reference performance
  - Boron vs. Lithium experiments diverted directly onto high-Z tiles
  - Assess power and pressure balance in the NSTX-U SOL
- FY17 – Extend operational space of vapor-shielded regime
  - Determine core compatibility with vapor-shielded divertor plasmas
  - Determine vapor-shield performance with varying SOL pressures, input powers, connection length
  - Assess transient loading response (ELM loads)
- FY18 – Determine flowing-system replenishment needs due to net erosion to extend vapor-shielding regime beyond 1s tests

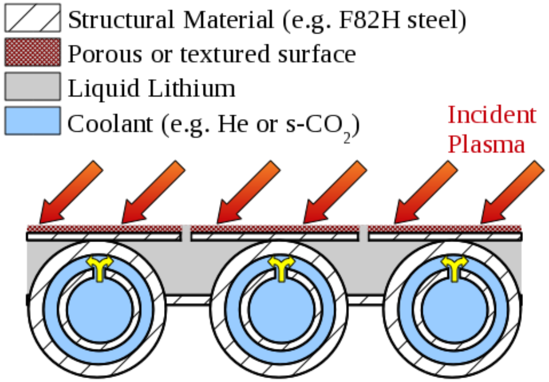
Parallel plasma-facing component research and development to support high-Z tile upgrade and vapor-shielding studies

- High-Z PFC assessment
  - Lamellae demonstrated technique (e.g. C-MOD and JET-ILW) to reduce stress
  - Both W and Mo are compatible with Li
- Flowing liquid lithium PFC development underway
  - Conceptual design and initial engineering assessment of long-pulse, LM-PFCs underway
  - Basic liquid-metal loop technologies being assembled for testing
- Long-term NSTX-U goal to implement LM divertor target

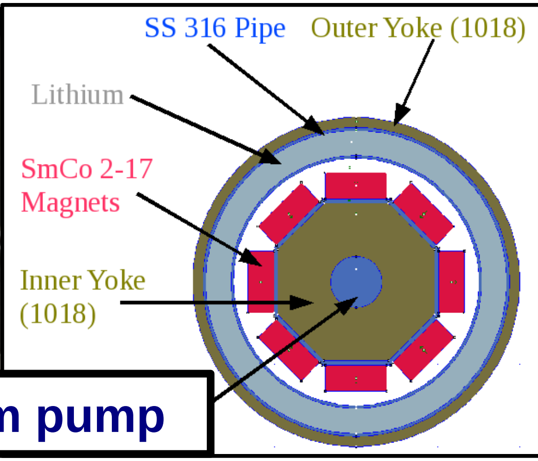


JET-ILW Divertor Cassette

Ph. Mertens, 13th PFMC, 2011



Actively-supplied, capillary-restrained, gas-cooled LM-PFC



PPPL Liquid lithium pump

NSTX-U Five-year plan will begin assessment of a high-Z PFC and liquid lithium integrated PMI solutions for next steps

- M&P research will contribute to the understanding of material migration and evolution to prepare for next-step devices
  - Study mixed material issues with Li, C, O and high-Z studies
  - Examine whole machine material erosion, migration, re-deposition
  - Begin assessment of high-Z PFCs with low-Z coatings

- M&P research will advance liquid metal PFCs as an innovative solution to handling fusion exhaust power and particle loads
  - Establish the science of continuous vapor shielding with high-temperature, liquid lithium PFCs and determine the compatibility with high-performance discharges
  - Perform side-by-side comparison of a high-Z PFC vs. a liquid metal PFC to inform next-step devices

Backup

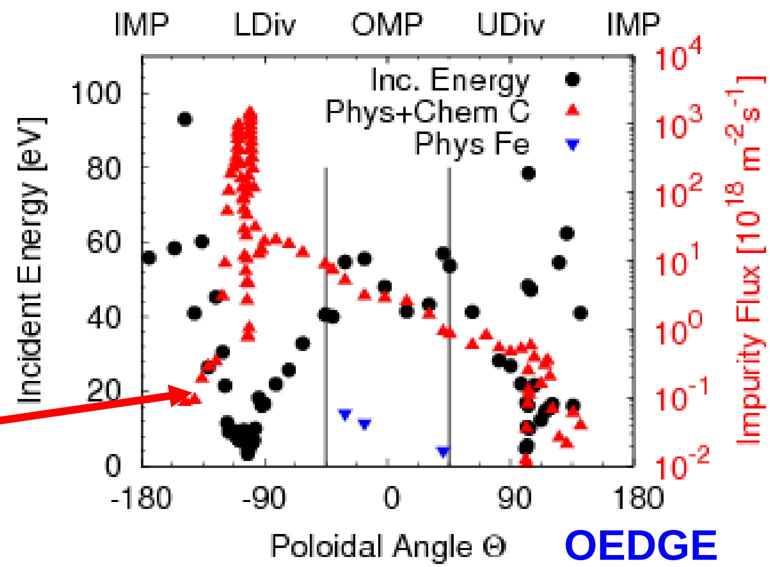
NSTX discharges already indicate significant first-wall erosion and NSTX-U will extend this erosion by factor of 10

- Simplified estimates indicate large amounts of wall erosion in next-step devices
  - Estimates based on charge-exchange neutrals at the edge
  - Several simplifications (e.g no neutral pressure dep., poloidal uniformity)
- Plasma + neutral transport codes enable modeling of these processes
  - Fluid codes: UEDGE, SOLPS, OEDGE; Neutral codes: EIRENE, DEGAS2; impurity codes: DIVIMP
  - Mean neutral energies in NSTX discharge 40-90eV
  - Flux to walls poloidally non-uniform (peaked near outboard LSN)
  - ~20% of carbon flux from first wall

$$\Gamma^{Gross}_{sputt.} \approx Y_{sputt,cx} \frac{P_{cx}/E_{cx}}{S_{plasma}}$$

$$\approx Y_{sputt,cx} \frac{f_{cx} P_{heat}/E_{cx}}{S_{plasma}}$$

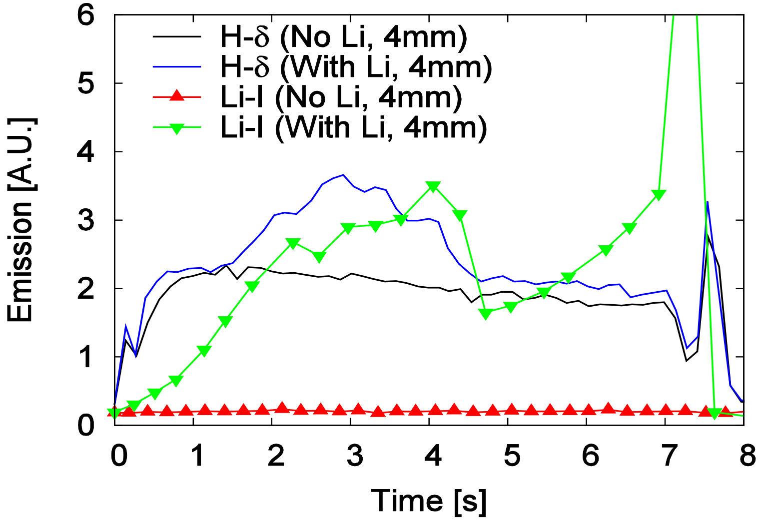
Machine	P_{max}/S [MW/m^2]	τ_{annual} [s]	Yield [kg/yr]
DIII-D[1]	0.4	10^4	0.08 (C)
NSTX	0.2	3×10^3	0.012 (C)
NSTX-U	0.5	10^4	0.1 (C)
ITER[1]	0.15	10^6	92 (W)
ST-Pilot	0.98	10^7	1800 (W)
ARIES-AT	0.98	3×10^7	8000 (W)



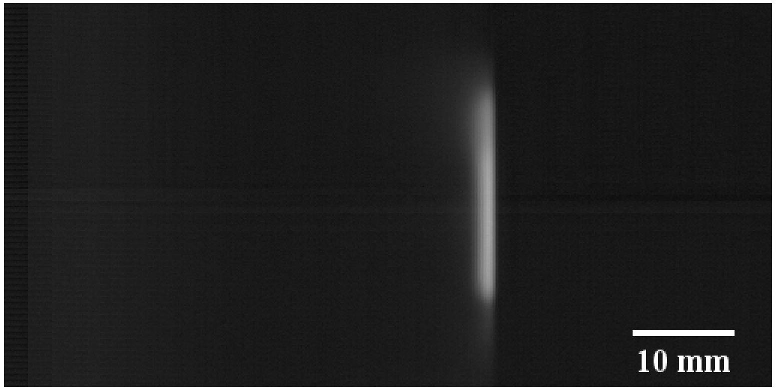
[1]P.C. Stangeby, JNM 2011.

Initial experiments at Magnum-PSI demonstrate vapor cloud production under divertor-like plasma conditions

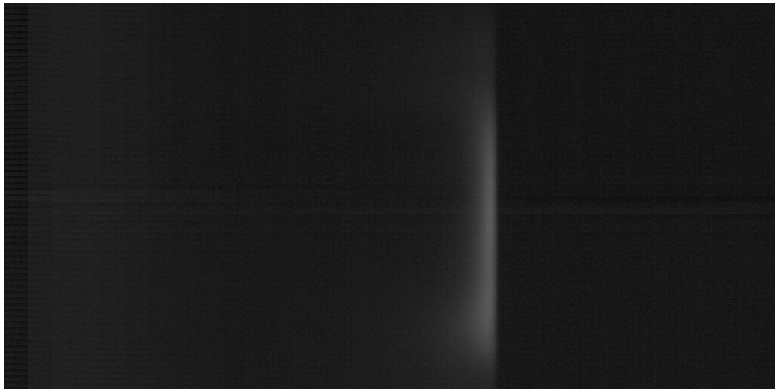
- High-Z substrate target tested
  - 2-3x10^{20}m^{-3}, 2eV plasma
  - 1200C surface by end of 7s
- Persistence of cloud indicates very large (R~1) redeposition fraction (c.f. Brooks JNM 2001)



Li-I emission, t=2.5s



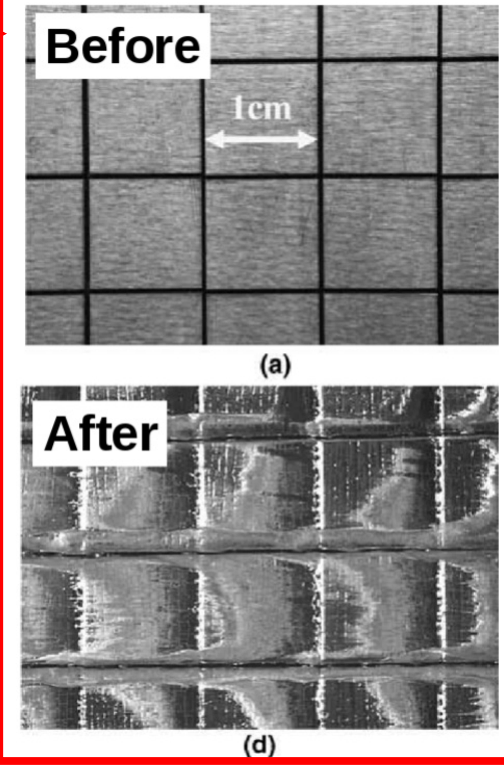
Li-I emission, t=6s

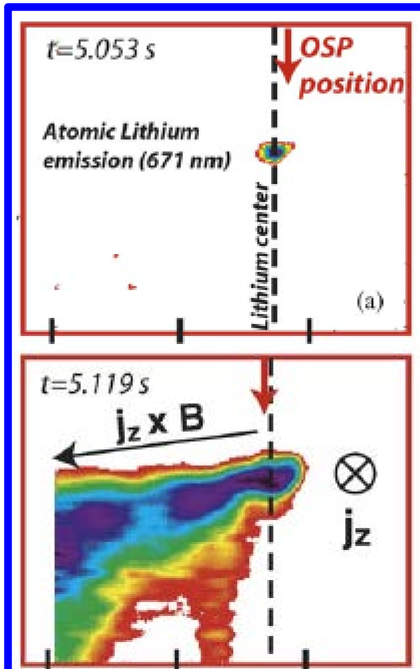


Taming the plasma-material interface is a grand-challenge for magnetic fusion energy

- Creation of economical fusion energy depends on component lifetime
- Significant uncertainty in how existing PFC candidates will extrapolate
  - Solids look promising but...
  - Liquids look promising but...
- PMI challenge must be met to enable a high-power-density, long-pulse facility such as FNSF

Tungsten melting under ELM-like bombardment (Federici, JNM 2005)

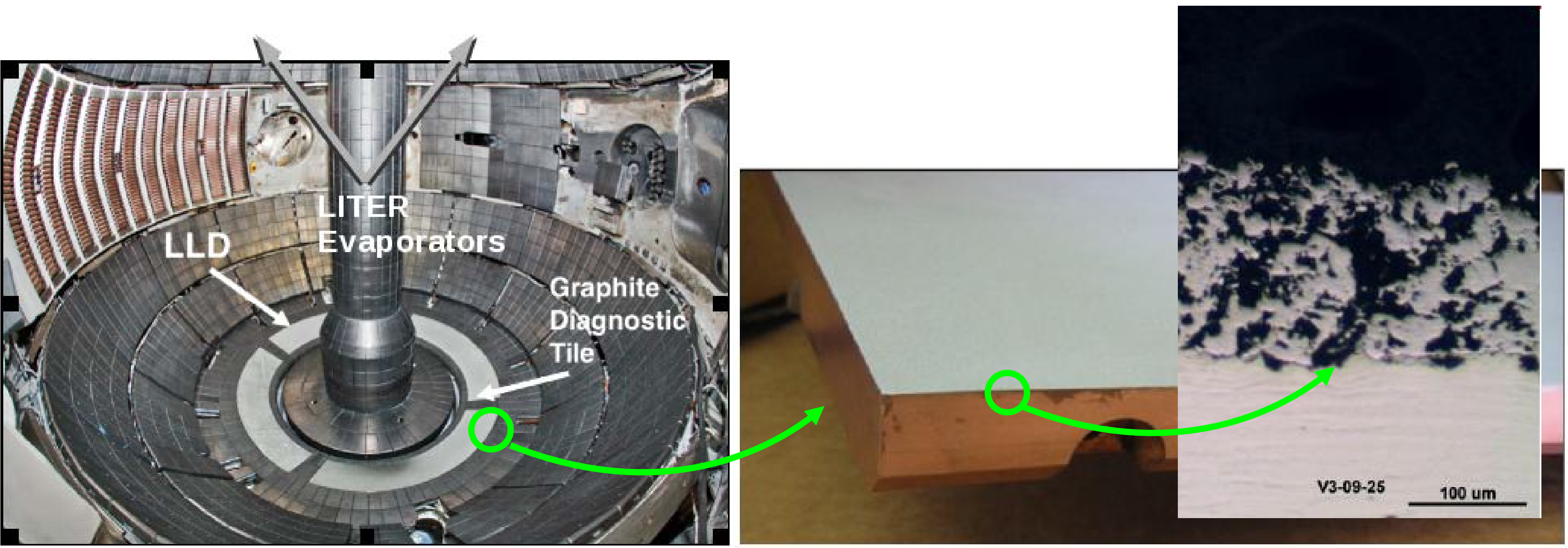




Lithium ejection in DIII-D (Whyte, FED 2004)

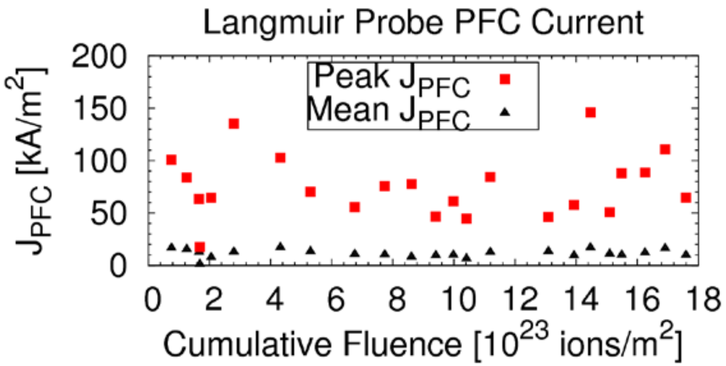
NSTX performed liquid-metal PFC experiments with the Liquid Lithium Divertor (LLD)...

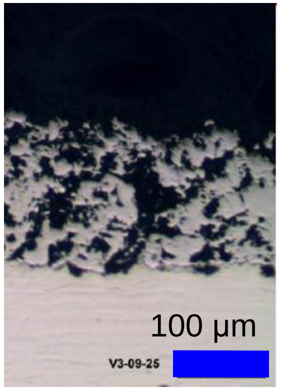
- Liquid lithium divertor installed for FY2010 run campaign
- 2.2cm copper substrate, 250um SS 316, ~150um flame-sprayed molybdenum porous layer; LITER loaded
- 37g estimated capacity, 60g loaded by end of run



...and demonstrated stable liquid metal PFC operation in a diverted tokamak

- Large transient currents measured with Langmuir probes, LLD porous geometry limits wavelength

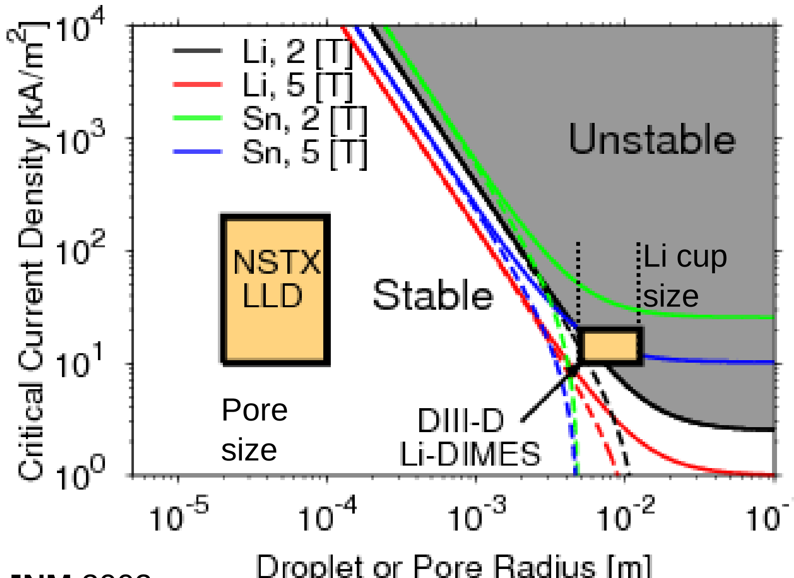




- Raleigh-Taylor analysis provides marginal stability curves; NSTX LLD stable

$$k_{Cr} = \sqrt{\frac{jB - \rho g}{\Sigma}}$$

For the fastest growing modes

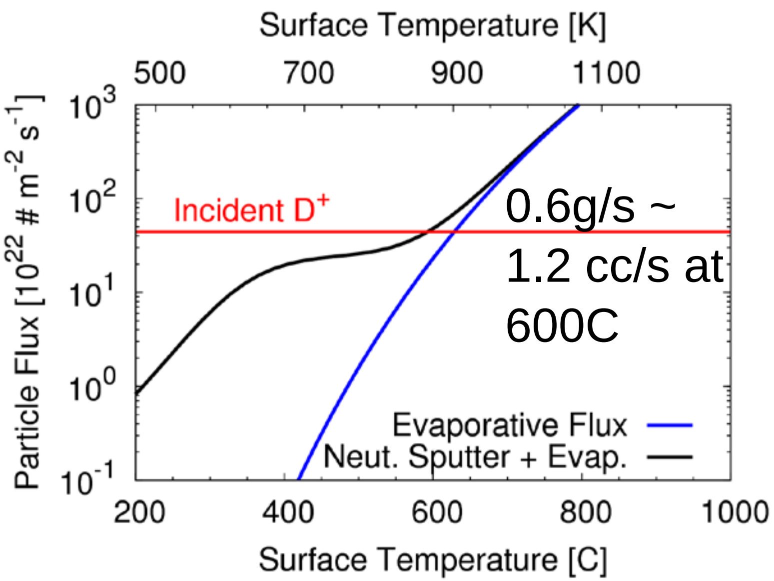


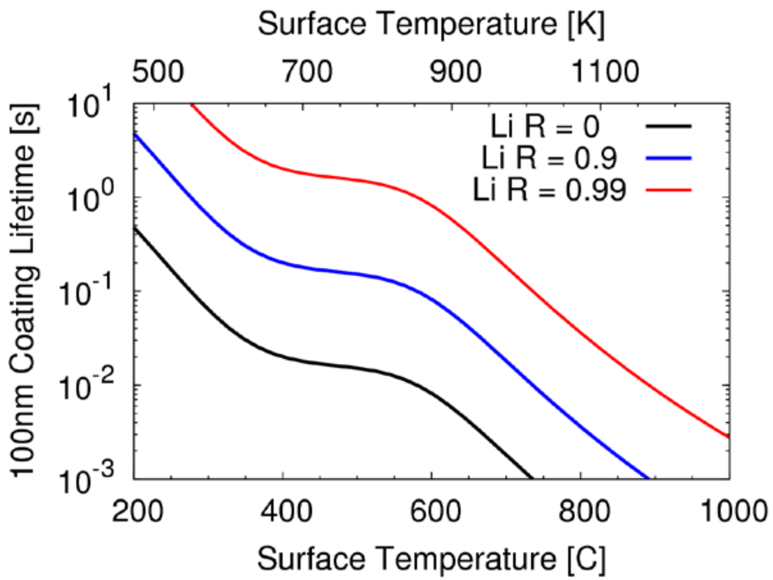
- CPS tests also reduced droplet ejection with smaller pore sizes*

Jaworski **JNM** 2011, Jaworski IAEA FEC 2012, Whyte **FED** 2004, *Evtikhin **JNM** 2002

Thin coatings can provide protection of high-Z substrates

- Ion penetration depth of ~10nm means plasma will only interact with coating material
- Lithium erosion rate is large and highly temperature dependent
  - Erosive fluxes could quickly remove protective layers
  - No experimental data of erosion yield above ~500C, at these fluxes[1,2]
- Coating lifetime is extended by redeposition fraction
  - Motivates flowing systems to replenish coating
  - No temperature dependence in boron sputter yield[3], τ~2s at R=0

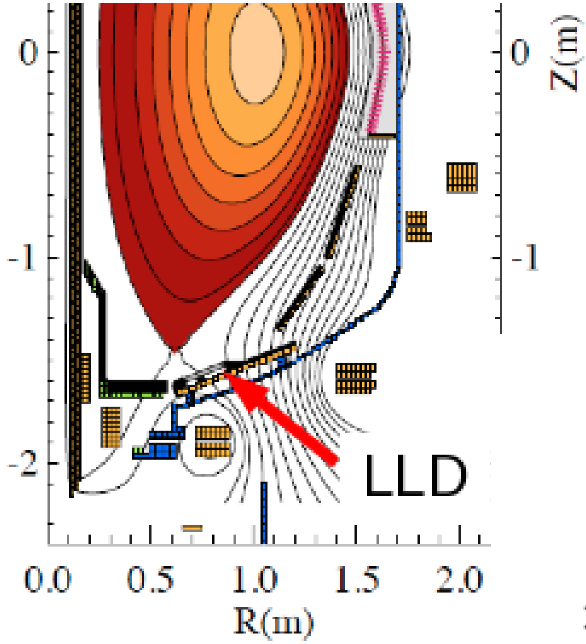




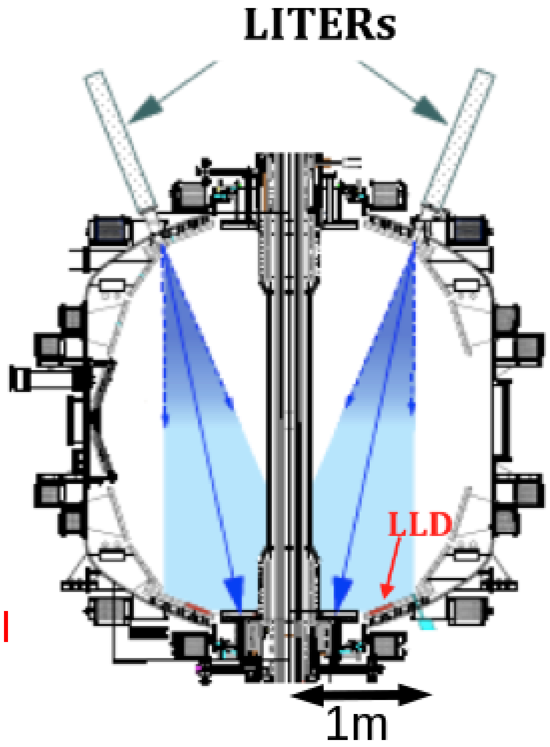
[1]Doerner, JNM 2001. [2]Allain, JNM 2003. [3]Hechtl, JNM 1992

Overview of experiments

- Experiments diverting onto the LLD occurred throughout run campaign
- Either diverted onto LLD or just inboard on ATJ graphite
- LITER only available filling method for the LLD
  - 7% filling efficiency estimated
  - Always coating entire lower divertor in addition to LLD
- Database of shots taken throughout run year

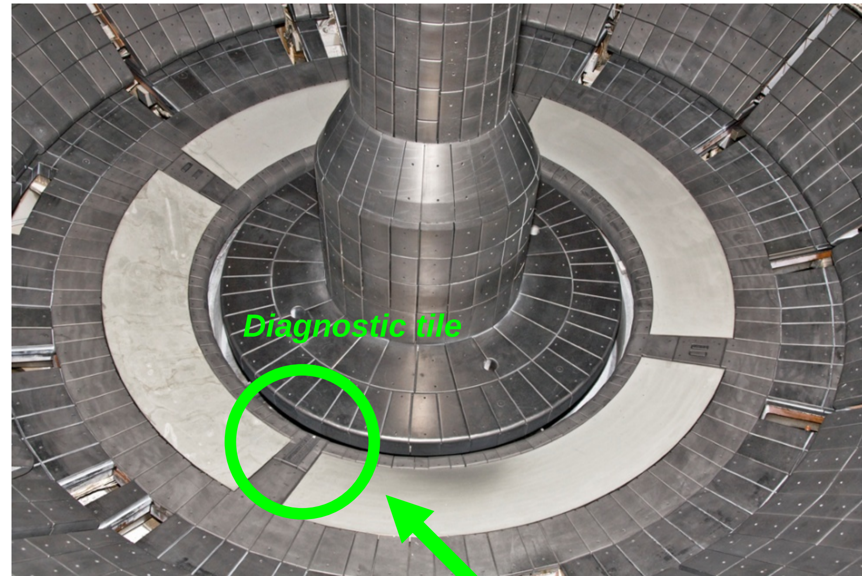


No boronization campaign prior to lithium introduction
Database already starts with 60g inventory in vessel

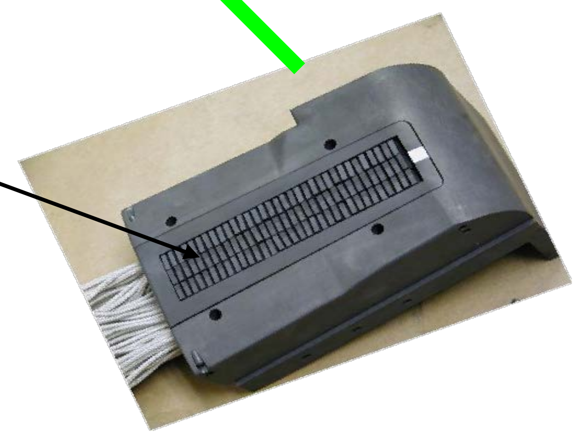


High-density Langmuir probe array installed for divertor plasma characterization

- Liquid Lithium Divertor (LLD) installed to study lithium plasma-material interactions
- Probe array characterizes local plasma properties in a range of experiments
- Provides high spatial density of measurements
- Oblique incidence yields smaller effective probe size



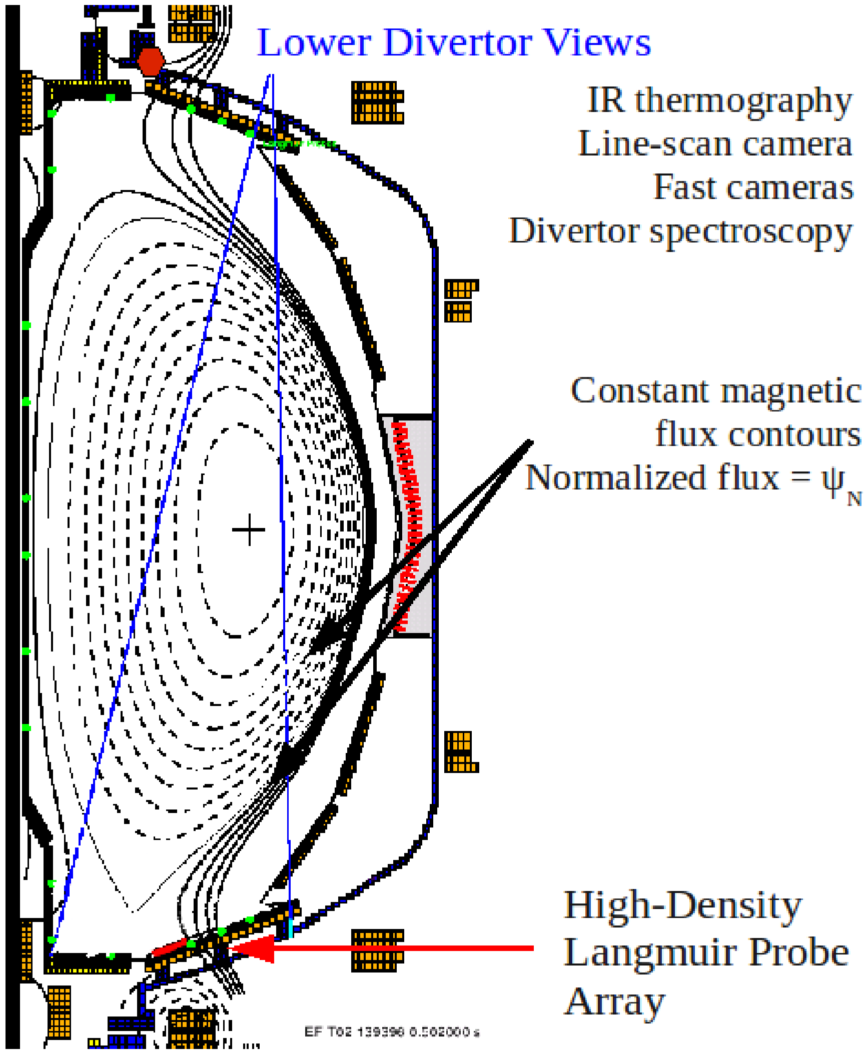
2x7mm electrode



J Kallman, RSI 2010
MA Jaworski, RSI 2010

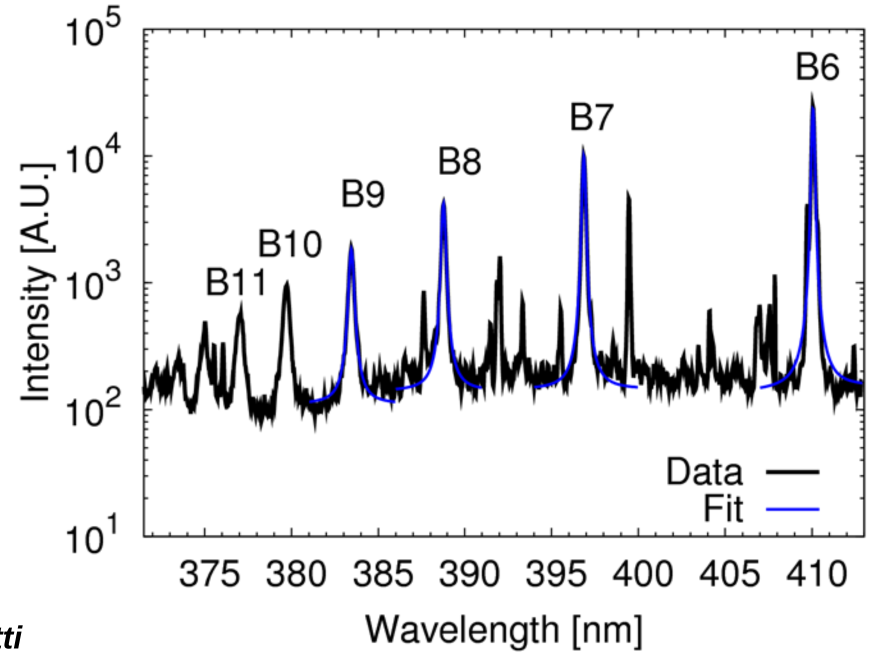
Consistency between diagnostics demonstrated with empirical plasma reconstruction framework

- Utilizes measured data points as starting point in constraining plasma models to fill the gaps between diagnostics
- Solution improves as more and more data constrains background
- OEDGE code suite used here: Onion-Skin Method (OSM2)+EIRENE+DIVIMP
    - OSM2 solves plasma fluid equations
    - EIRENE performs Monte Carlo neutral hydrogen transport, iteratively coupled to OSM2
    - DIVIMP performs Monte Carlo impurity transport
- Utilized here to compare probe interpretation methods against other diagnostics

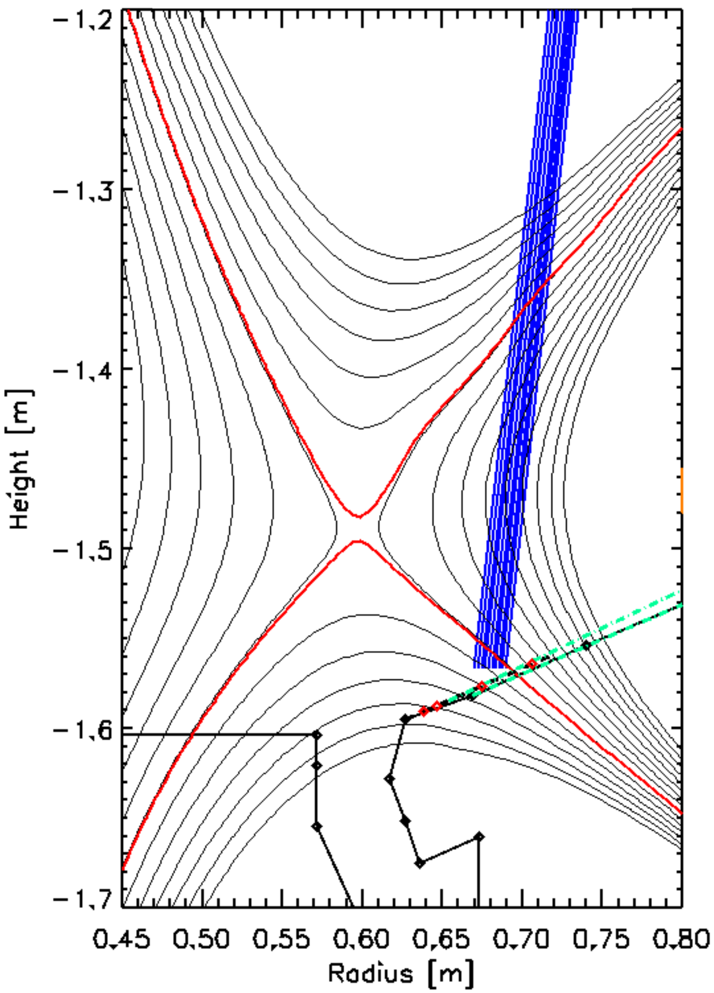


Density measurement from spectroscopy confirm kinetic probe interpretation

- Divertor spectrometer viewing strike-point region during discharge
- Deuterium Balmer lines shown in spectra
- Pressure broadening analysis indicates dneisty of 3.6e20 m^{-3}
  - Existence of high-n Balmer lines indicates low temperature



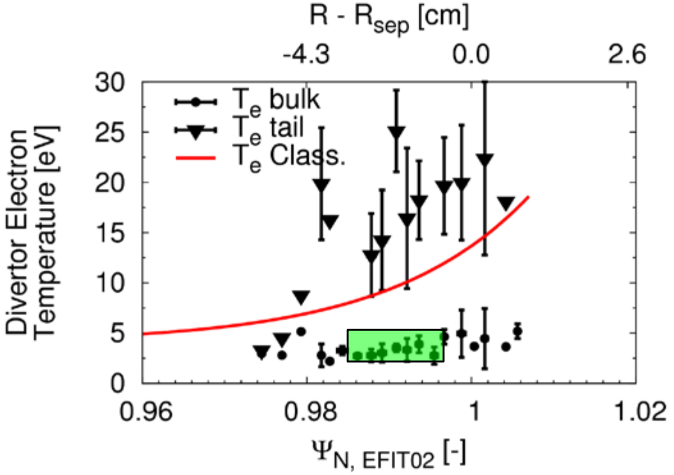
Divertor spectrometer view

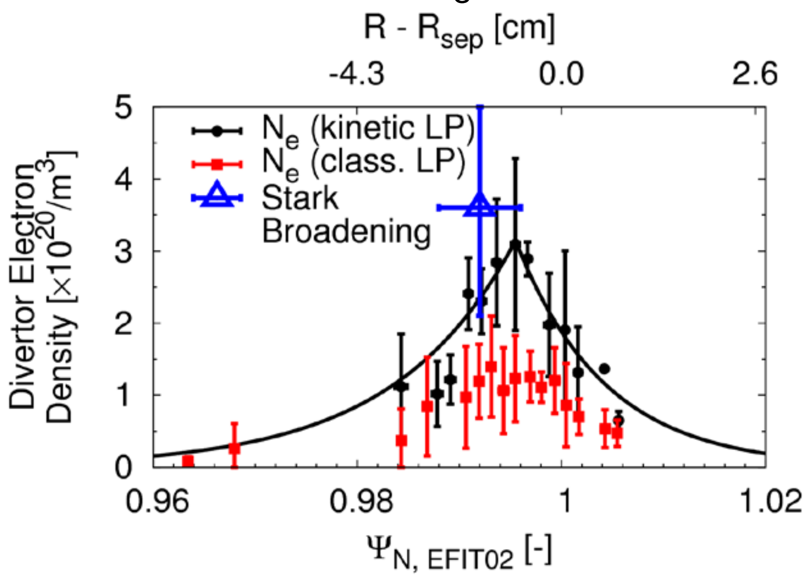


F. Scotti

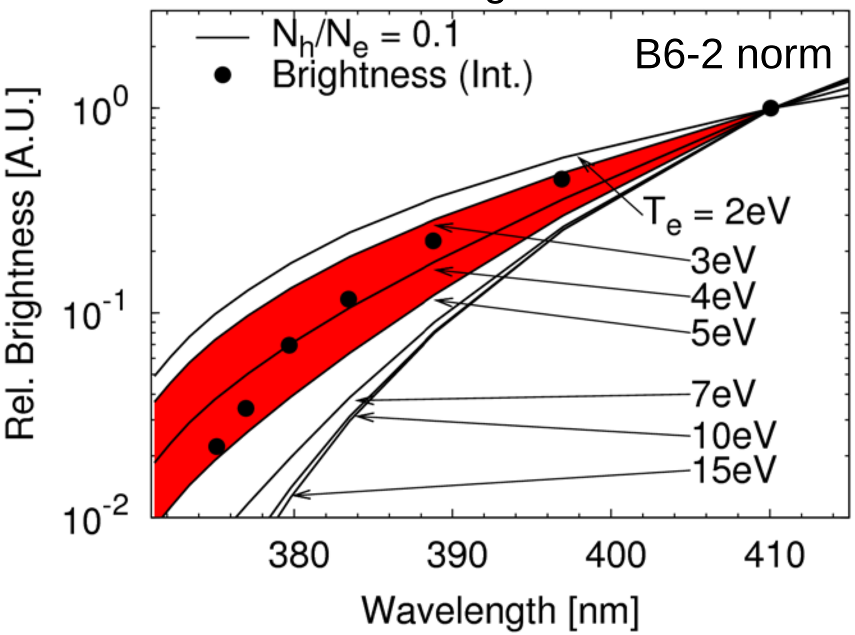
Broadening measurement and modeling of hydrogen spectrum consistent with kinetic probe interpretation

- Pressure broadening yields density
- OEDGE plasma+neutral solution provides local parameters
- Collisional-radiative model by D. Stotler calculates excited state populations
- Brightness ratios normalized to B6-2 consistent with $3<T_e<5$eV





Balmer-Series Brightness Ratios

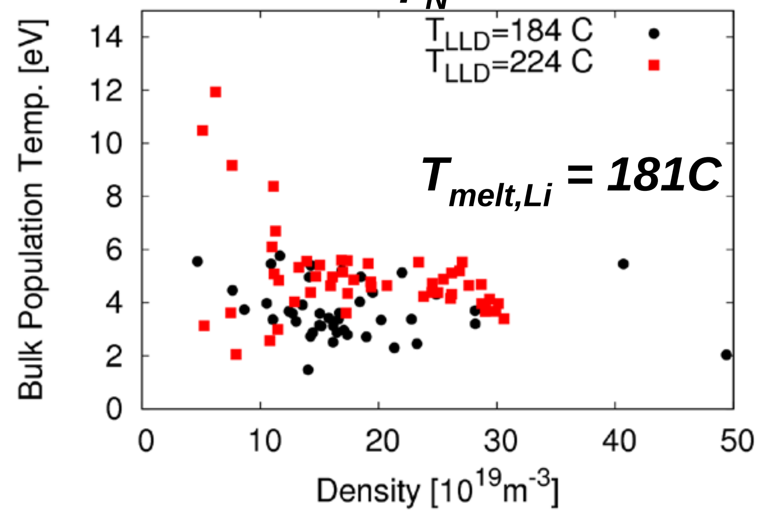


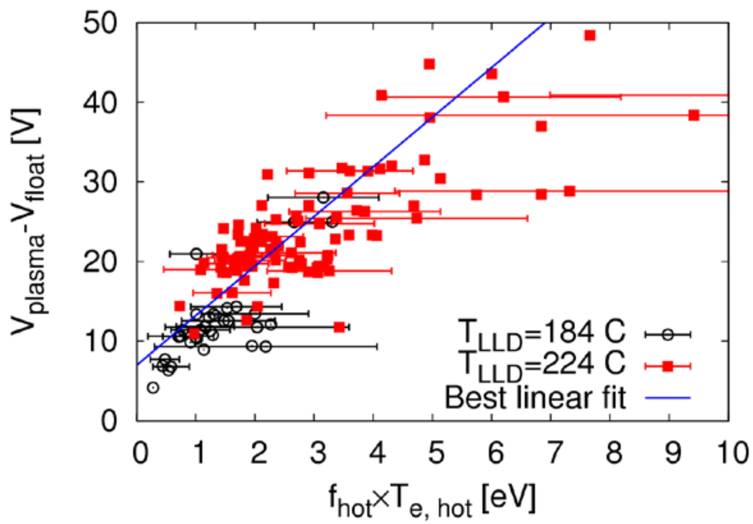
Jaworski, et al., 20th PSI, Aachen, Germany, June 2012.

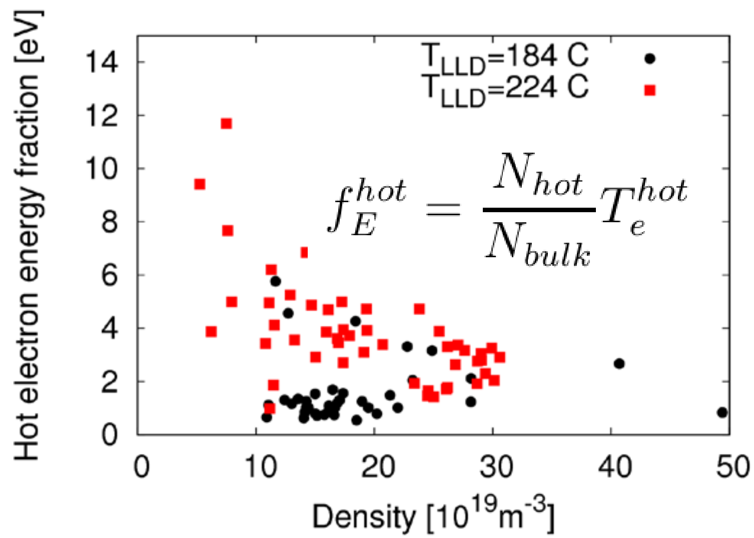
Distribution function analysis indicates some local changes in plasma conditions on plasma-heated LLD

- Discharge sequence repeatedly heated and plasma-conditioned the LLD surface
- Local plasma temperatures elevated with hotter LLD surface temperature (T_{LLD} > $T_{melt,Li}$)
- Increase in plasma temperatures correlated with increase in V_p-V_f potential difference[1]
- **Local changes raise the question whether large-scale global changes are also observed...**

Comparisons made on identical ψ_N locations



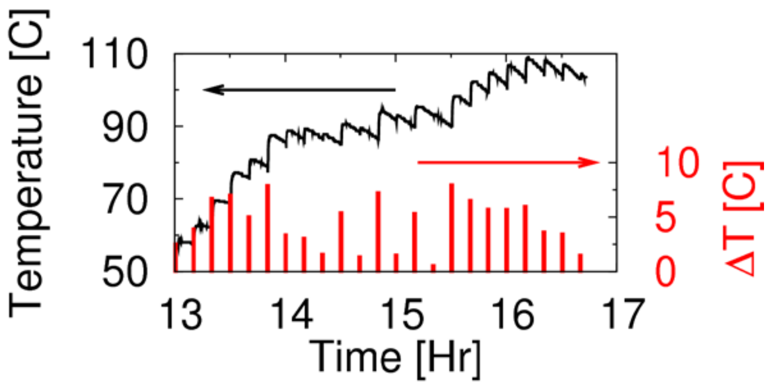


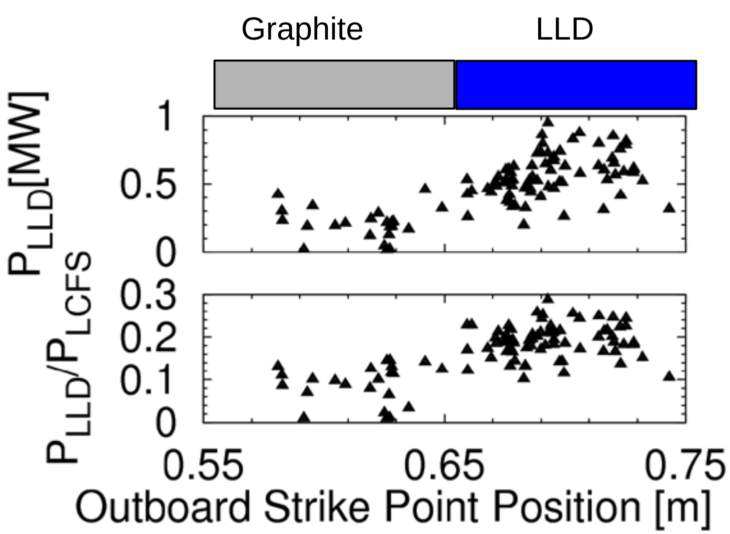


[1] *Jaworski et al., Fusion Eng. Des. 87 (2012) 1711.*

Significant power onto LLD measured

- Embedded thermocouples provide measure of temperature changes from before and after discharge
- Each plate is 43kg of copper
    - $\Delta E = mc_p \Delta T$ per plate
    - $P_{LLD} \sim 4\Delta E / \tau_{pulse}$
    - $P_{LCFS} = P_{NBI} + P_{OHM} - P_{RAD} - dW/dt$
- LLD absorbing about 25% of exhaust power (P_{LCFS})
    - ~1MW in some cases
- No molybdenum observed in the plasma after melted (Soukhanovskii, **RSI**, 2010)





Jaworski, et al., IAEA FEC 2012

Surface contamination indicates this was not a "fair" test of a liquid lithium PFC

- Divertor filterscopes provide indicator of impurities
  - Relative fraction of impurity should be reflected in sputter yield
  - Particle flux proportional to power
- Normalization against flux indicates no difference diverted onto the LLD
- Plasma cleaning in PISCES-B did show oxygen reduction*
  - 400s, T>600K
  - LLD transiently exceeded these temperatures, but not steady
- $\tau_{intershot} \sim \tau_{oxidize}$ indicates oxidation likely (see GO6.008, A. Capece)

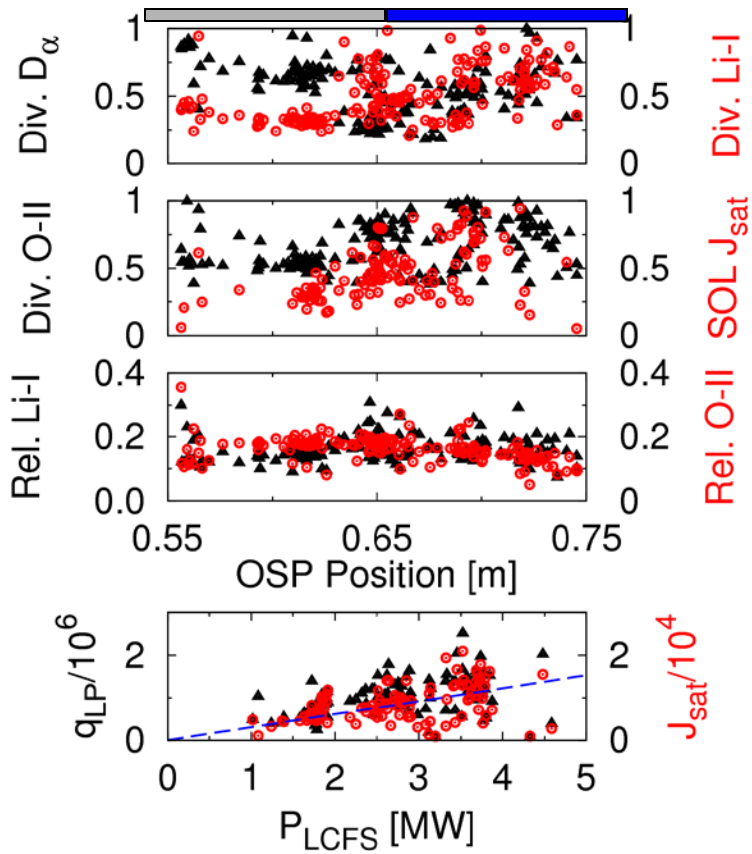
$$\varepsilon_{imp} \propto N_{imp} \propto Y_{imp} J_{sat}$$

$$Y_{imp} \propto Y^0_{imp}\theta_{imp}$$

$$P \propto \gamma J_{sat} T_e$$

$$\theta_{imp} \propto \frac{\varepsilon_{imp}}{P}$$

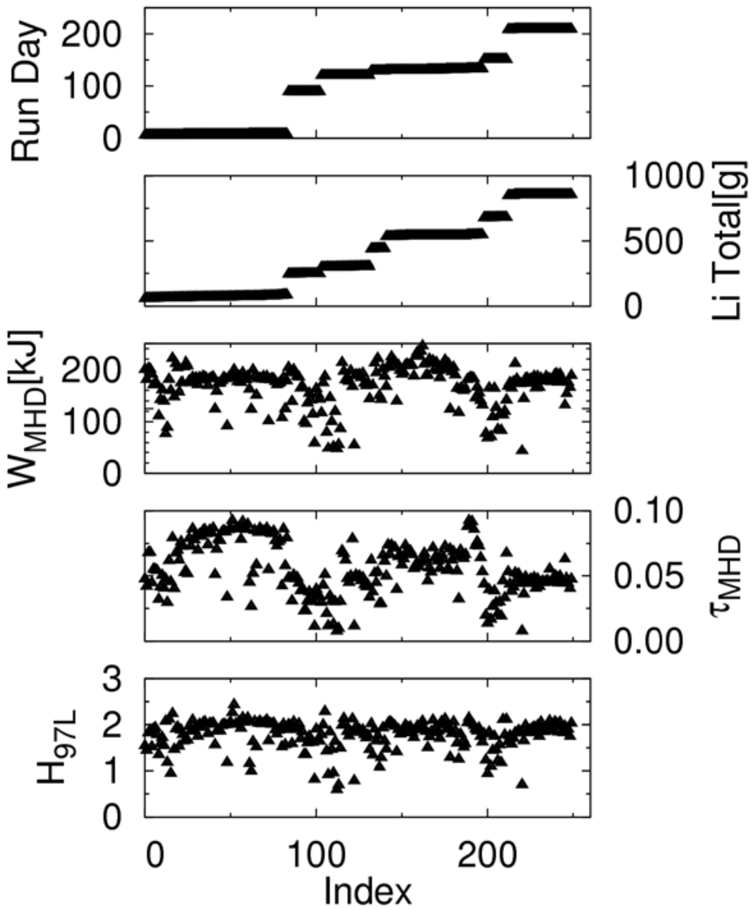
Emission ϵ
Coverage θ
Divertor Power P



Jaworski IAEA FEC 2012, *Baldwin **NF** 2002.

Performance should be independent of lithium quantity *if* surface contamination is key variable

- FY2010 LLD experimental set
  - Experiments span 60g to nearly 1kg of deposited lithium
  - Includes 75hr deposition at mid-year
  - Calculate ITER 97L H-factor *average* from 400-600ms for each discharge
- Discharges look about the same between start and end of run
  - Consistent with surface contamination hypothesis



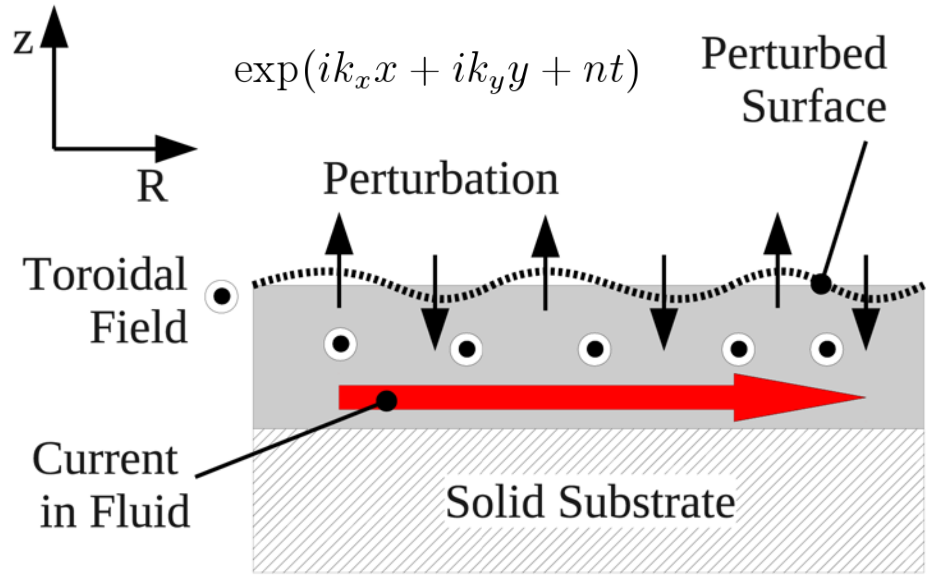
Fully-flowing PFC can provide a means of sweeping away gettered material and creating “stationary” surface conditions.

Jaworski, et al., IAEA FEC 2012

Vertical body forces can destabilize free surface

- Net result of radial currents is to produce vertical forces
  - Currents in SOL that close in the PFC
  - Disruption eddy currents

- Net body force upward has the potential to create Raleigh-Taylor instability
  - Must overcome gravity and surface tension
  - Must overcome magnetic tension (depending on orientation)



$$\frac{\partial u_i}{\partial x_i} = 0$$

$$\rho\left(\frac{\partial u_i}{\partial t} + u_j \frac{\partial u_i}{\partial x_j}\right) = \rho X_i - \frac{\partial p}{\partial x_i} + \mu \nabla^2 u_i$$

$$n^2 = k(jB/\rho - g)\left[1 - \frac{k^2\Sigma}{(jB/\rho - g)\rho} - \frac{B^2 k_x^2}{2\pi\mu_0 (jB/\rho - g)\rho k}\right]$$

Jaworski **JNM** 2011

Porous-MHD effects alter liquid-metal wicking

- Wicking into porous material described with Darcy Eqn.
  - Pressure head provided by capillary pressure

$$u = \frac{k \Delta P}{\mu L} \qquad P_c = \frac{2 \Sigma \cos\alpha}{r_p}$$

- Addition of MHD pressure losses and rearrangement creates porous-MHD version of the Lucas-Washburn eqn.

$$L u = L \frac{dL}{dt} = \frac{2 \Sigma \cos\alpha}{r_p} \cdot \frac{1}{\mu / k + \sigma B^2}$$

$$= \frac{2 \Sigma r_p \cos\alpha}{\mu \left(r_p^2 / k + Ha^2 \right)}$$

- Solution yields "sorptivity", S, of porous material including MHD effects

$$L(t) = \left(\frac{4 \Sigma r_p \cos\alpha}{\mu \left(r_p^2 / k + Ha^2 \right)} t \right)^{1/2} = S \sqrt{t}$$

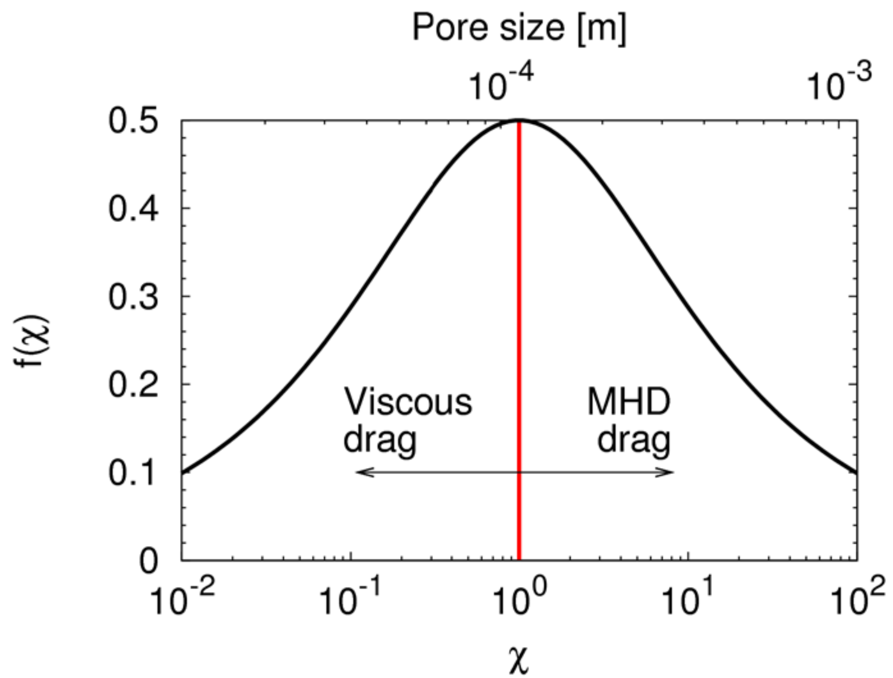
Wicking into material can be optimized with pore size

- Re-arrangement provides permeability-enhanced Hartmann parameter
  - Permeability of packed-bed used for illustration[1]
  - Liquid lithium material properties in 1T field yield 130μm pore

- Sorptivity no longer isotropic as in many hydrodynamic systems
  - $S^2/2$ = 8, 19 cm^2/s (perp, para) Li, $r_{p,opt}$=130μm
  - $S^2/2$ = 6, 20 cm^2/s (perp, para) Sn, $r_{p,opt}$=290μm

$$L\frac{dL}{dt}=\left(\frac{2\Sigma\cos\alpha}{B\sqrt{\sigma\mu}}\right)\left[\frac{\epsilon^3}{45(1-\epsilon)^2}\right]^{1/2}\frac{\sqrt{\chi}}{1+\chi}$$

$$\chi=\frac{k}{r_p^2}Ha^2 \qquad r_{p,opt}=\left[45\frac{(1-\epsilon)^2}{\epsilon^3}\cdot\frac{\mu}{\sigma B^2}\right]^{1/2}$$



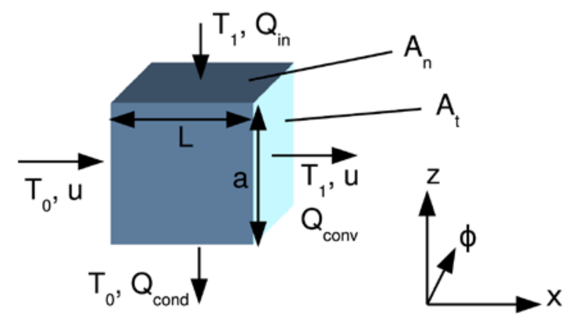
[1]Scheidegger, A.E. *The physics of flow through porous media*, University of Toronto Press, Toronto, Canada, 3rd ed. 1974.

Conduction dominated thermal transport makes conventional cooling relevant

- Control-volume analysis illustrates relevant thermal transport regime[1]

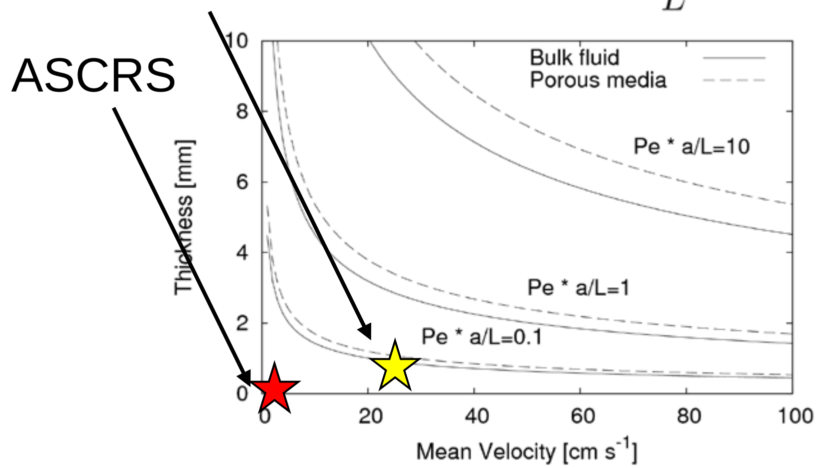
- Thin, slowly-moving liquid metal can be considered a solid in thermal anaysis

- Conventional gas cooling techniques applicable to these types of LM-PFCs



$$Q_{in} = Q_{cond,l} + Q_{cond,s} + Q_{conv}$$

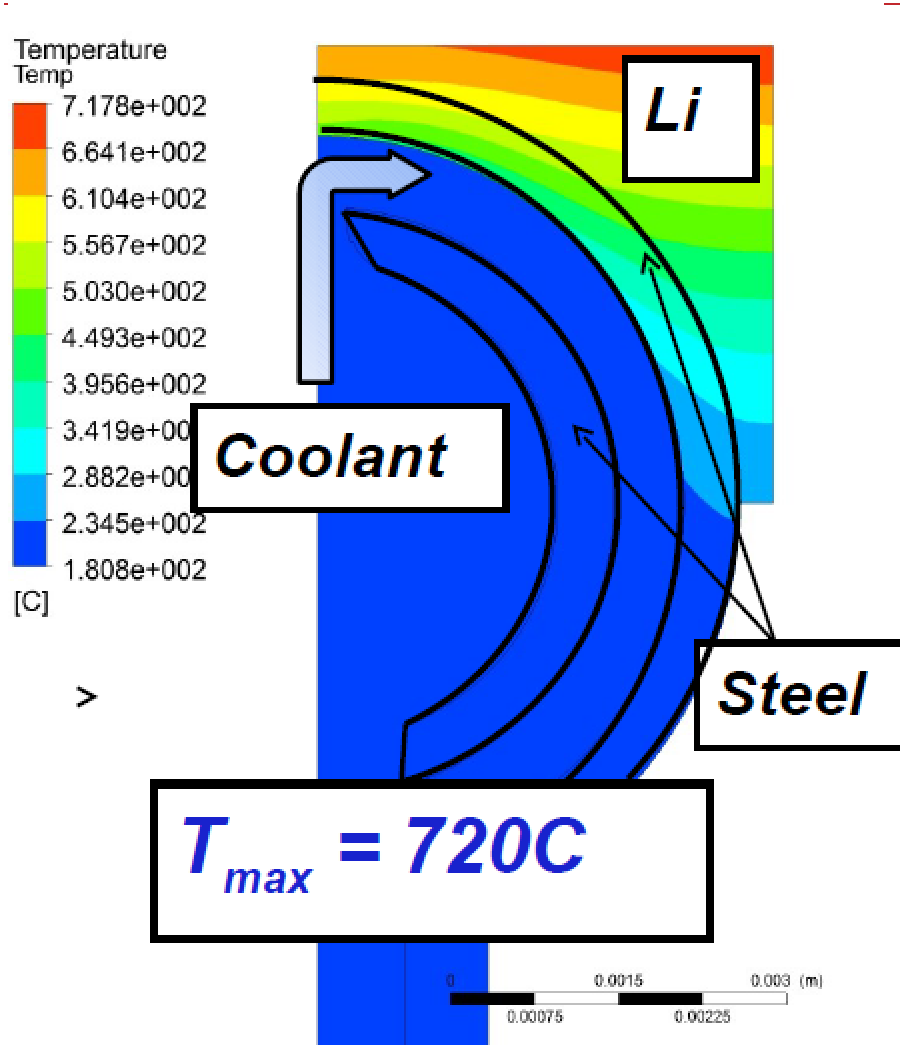
$$\frac{q_{in}}{k_{\text{eff}}\Delta T/a} = 1 + \frac{f\rho_l c_{p,l}}{k_{\text{eff}}}\frac{a^2 u}{L} = 1 + \frac{au}{\alpha_{eff}} \cdot \frac{a}{L} = 1 + \mathbf{Pe}\frac{a}{L}$$



[1]Jaworski, et al., **JNM** 2009, [2]Ruzic, et al., NF 2011.

Advanced cooling techniques can be optimized for LM-PFCs

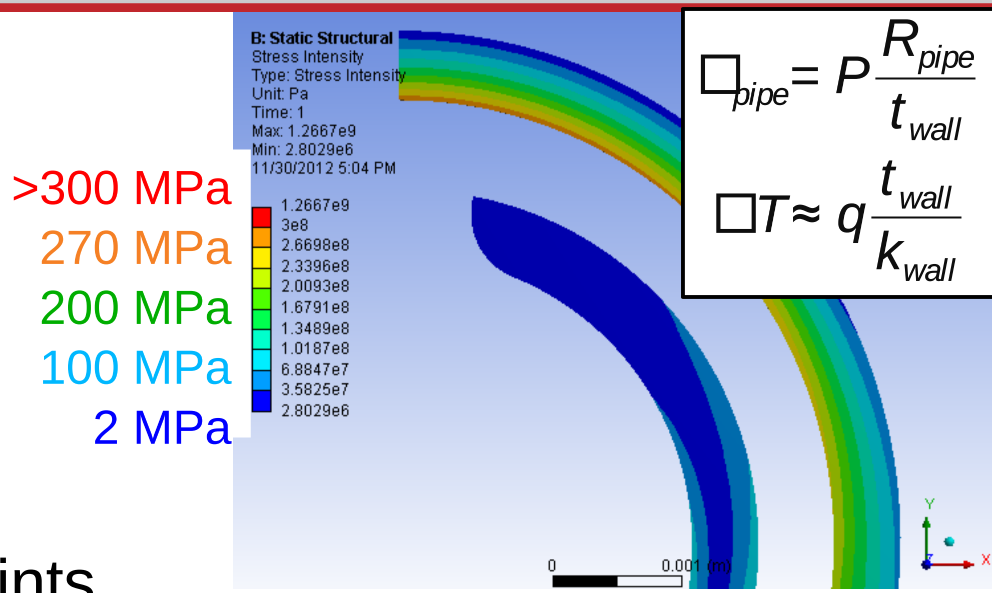
- T-tube[1] uses impinging gas jets to increase local heat transfer coefficient
- Helium jet peak heat transfer is near ~40 $kW/m^2/K$ in original T-tube design
- Altered T-tube for these simulations to have:
  - Smaller radius
  - Steel structure, s-CO2 coolant (No tungsten)
  - 10 MW/m^2 incident



[1]Abdel-Khalik FST 2008.

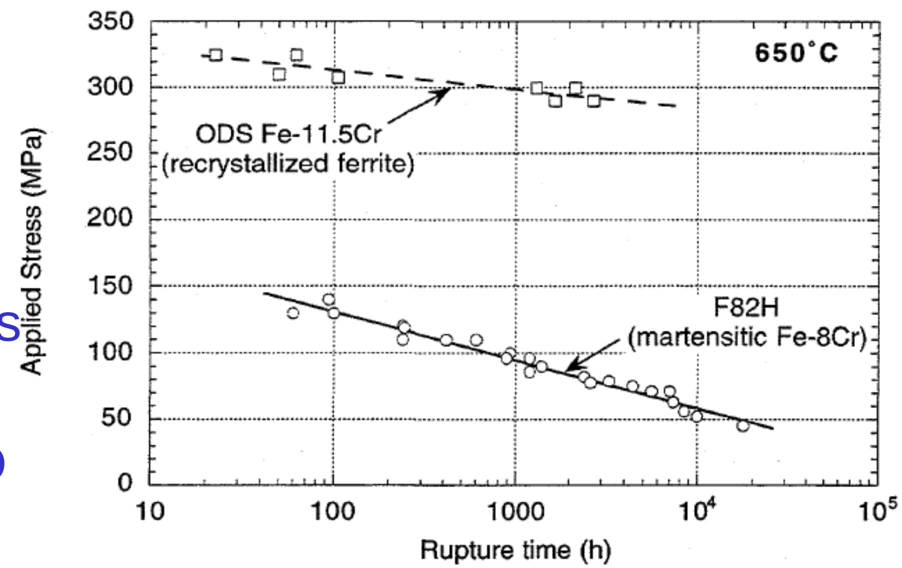
Reduced sizes can benefit power extraction

- At constant stress, radius reduction can reduce pipe-wall thickness
  - Rely on liquid metal protection of substrate
  - Manufacturing challenges...
- Yield stress and creep deformation provide design points
- ODS-RAFM should avoid rupture for >2yrs
  - $T_{max,steel}$~610C
  - Highest stress at lowest temperatures (500-550C)
  - Further optimization to be done in 3D to address margins of safety, etc.



$$\sigma_{pipe} = P\frac{R_{pipe}}{t_{wall}}$$

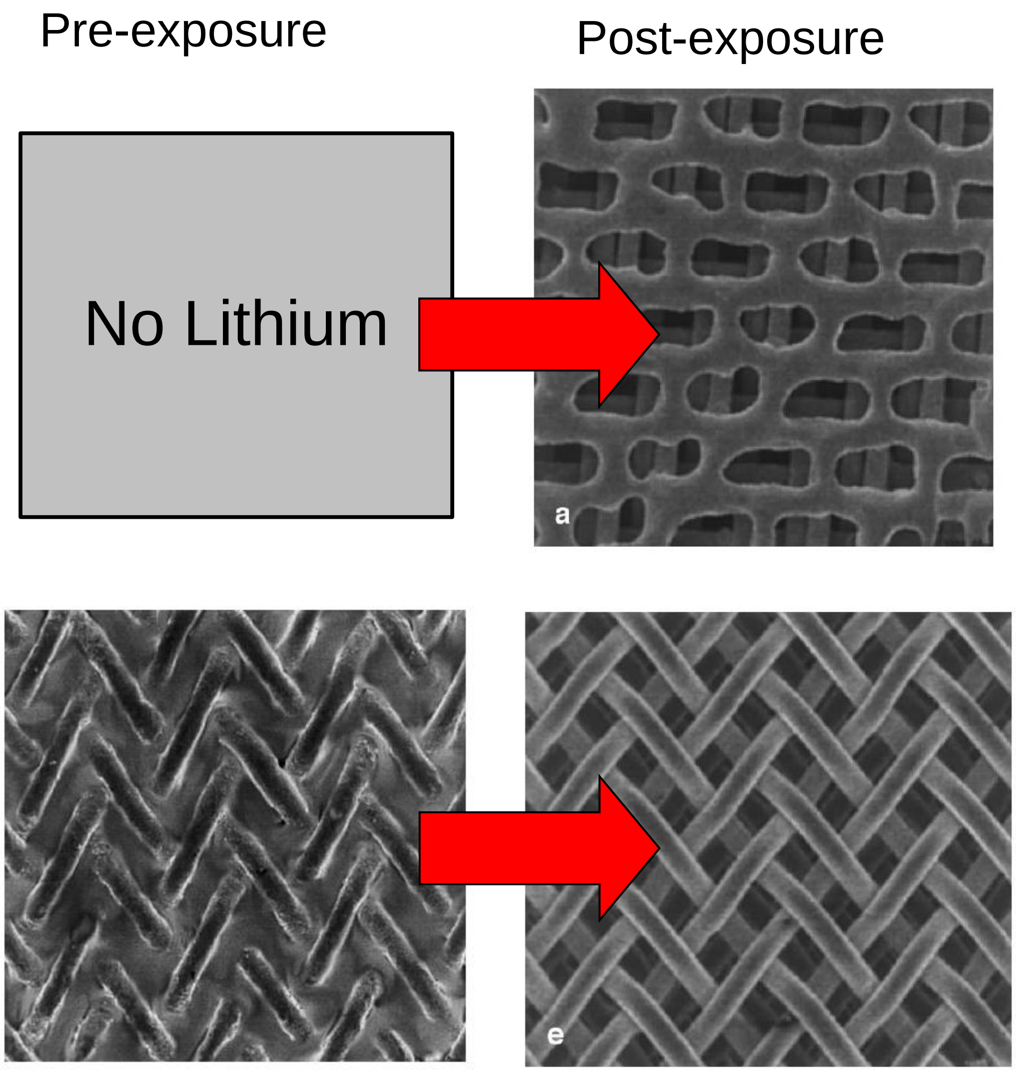
$$\Delta T \approx q\frac{t_{wall}}{k_{wall}}$$



Zinkle, et al., Fusion Materials Report, June 1998, p.135.

Liquid lithium demonstrated to protect fragile substrate

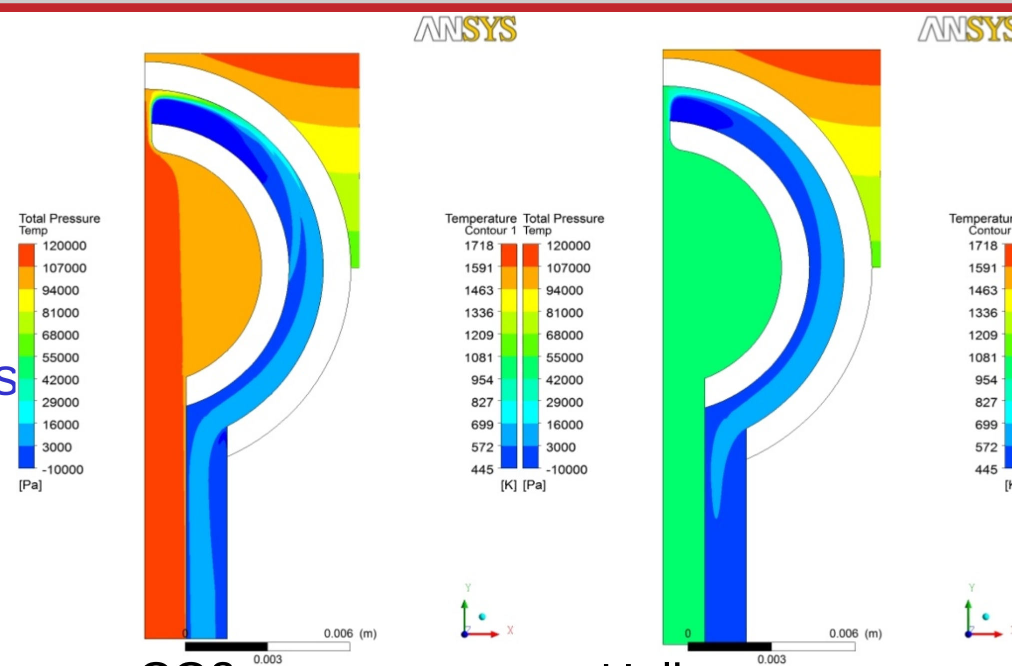
- Red Star Capillary-Porous-System (CPS) long-since shown to substrate damage under ELM-like disruptions ($5MJ/m^2$, 0.5ms in QSPA)
- Able to withstand $25MW/m^2$ heat fluxes via strong evaporation and vapor shield
- In principle, all PFCs in fully-flowing system will return to an equilibrium position (i.e. self-healing)



Evtikhin, et al., J. Nucl. Mater. **271-272** (1999) 396.

Supercritical CO2 is a more effective coolant than helium

- Replacement of He with s-CO2 in base T-tube design reduces pumping power by 30%
  - Identical front-face temperatures
  - 2x pressure drop, 1/3 required flow
- Better thermal efficiency at lower temperatures than He Brayton cycle[1]
  - S-CO2 w/ 550C turbine inlet has 45% thermal efficiency
  - He w/ 700C turb. inlet had 43% (ARIES-CS)[2]



s-CO2	Helium
T_{inlet} = 600 C	T_{inlet} = 600 C
T_{outlet} = 647 K	T_{outlet} = 677 K
V_{inlet} = 8 m/s	V_{inlet} = 24 m/s
ΔP = 108 kPa	ΔP = 50 kPa

$$W_{pump} = Q\Delta\Delta \rightarrow \frac{W_{CO2}}{W_{He}} \approx \frac{8\times108}{24\times50} \approx \frac{2}{3}$$

[1] Dostal, et al., Tech. Report MIT-ANP-TR-100, MIT, March, 2004.
[2] Raffray, et al., **FST** 2008.

Material compatibility determines substrate

- Liquid surface absorbs incident plasma, substrate material absorbs neutrons

- Liquid lithium compatible with steel, vanadium alloys, refractory metals[1]

- Liquid tin compatible with refractories, not compatible with steel above 400C (unknown compatibility with vanadium alloys)[1]

- Porous structure can be produced by various methods (e.g. laser texturing[2], flame-spraying[3], foam CVD[4])

[1] Zinkle and Ghoniem, **FED** 2000. [2] Lin, et al., **JNM** 2013. 3 Kugel, et al., **FED** 2011, [4] Jaworski, et al., **JNM** 2008.

More advanced options could exist

- T-tube first example considered with well documented design for extension and modification
  - Still requires significant absolute pressure and wall thicknesses
  - Continued size reduction becomes difficult to manufacture
  - ~10 MW/m^2 may be the limit with steel structures
  - Integrated PFC-power cycle analysis is on-going work...

- Liquid metal heat-pipes another option being pursued
  - Reduces pressure at the target front-face allowing thinner structures and lower stress levels (may not need ODS)
  - Can effectively spread heat-flux over larger area reducing requirements on gaseous cooling
  - Porous-MHD issues already under study with free-surface work

Other open issues

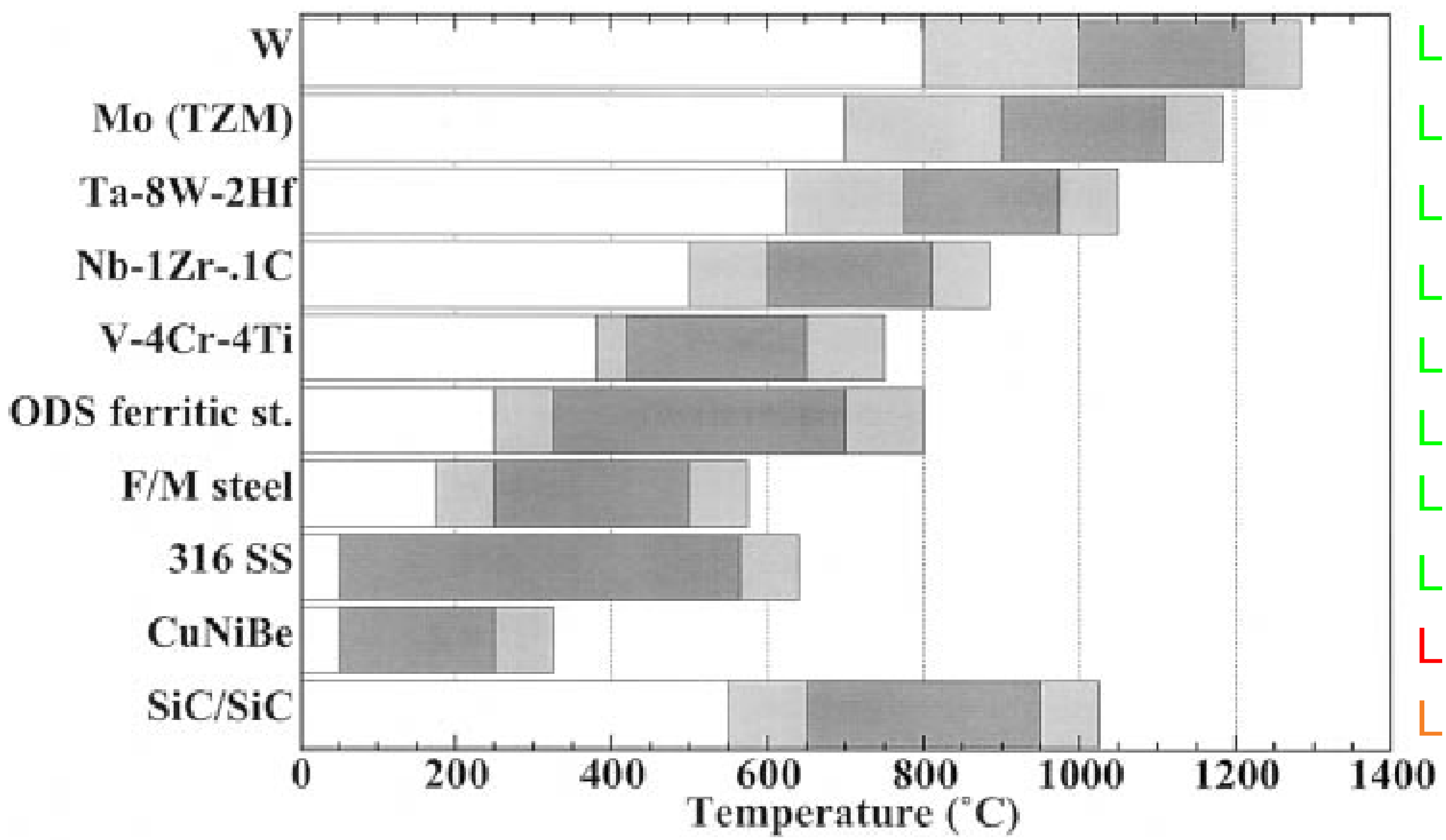
- Tritium processing and closing the liquid lithium loop
  - Requires confinement device experiments to demonstrate re-capture of migrating materials
  - Could prove to be lithium "Achilles heel" due to on-site inventory[1]
- Liquid metal protection of substrate material requires demonstration
  - Thin walls for better heat transfer rely on sacrificial liquid layer
  - Runaway electron beams? Other disruption events?
- Integrated core performance with high-temperature, liquid-metal PFCs
  - High-temperature, high evaporation/erosion lithium not demonstrated in divertor (encouraging results with limiter on FTU[2])
  - High-temperature liquid tin PFCs never tested to date
- Plasma modeling...

[1]M. Nishikawa, "Tritium in a fusion reactor (effect of Li system on tritium)" Presentation at PPPL ST tokamak discussion, March 27, 2012.
[2]M.L. Apicella, et al., Plasma Phys. Control. Fusion 54 (2012) 035001.

DEMO challenges considerable but progress is being made to determine if a technically feasible LM-PFC option exists

- For 10MW/m^2 peak divertor heat loads incident on target in "Pilot-plant" ST- and AT-DEMOs
  - Actively-supplied, capillary-restrained system prevents ejection
  - Liquid lithium on ODS-RAFM structure with s-CO2 cooling looks encouraging ***(eliminates net-reshaping of PFC)***
  - Some additional optimization to be done with full 3D design
  - Need data with high temperature lithium surface in divertor-like plasma
- Experimental demonstration and additional analysis will address open issues over coming years
  - @PPPL – internal lab funding, NSTX-U base program, other sources pending
  - Collaboration underway with Magnum-PSI, NSTX-U, and EAST
- Still several open issues forcing talk titles with the word "possible" but progress is being made; your input is welcome!

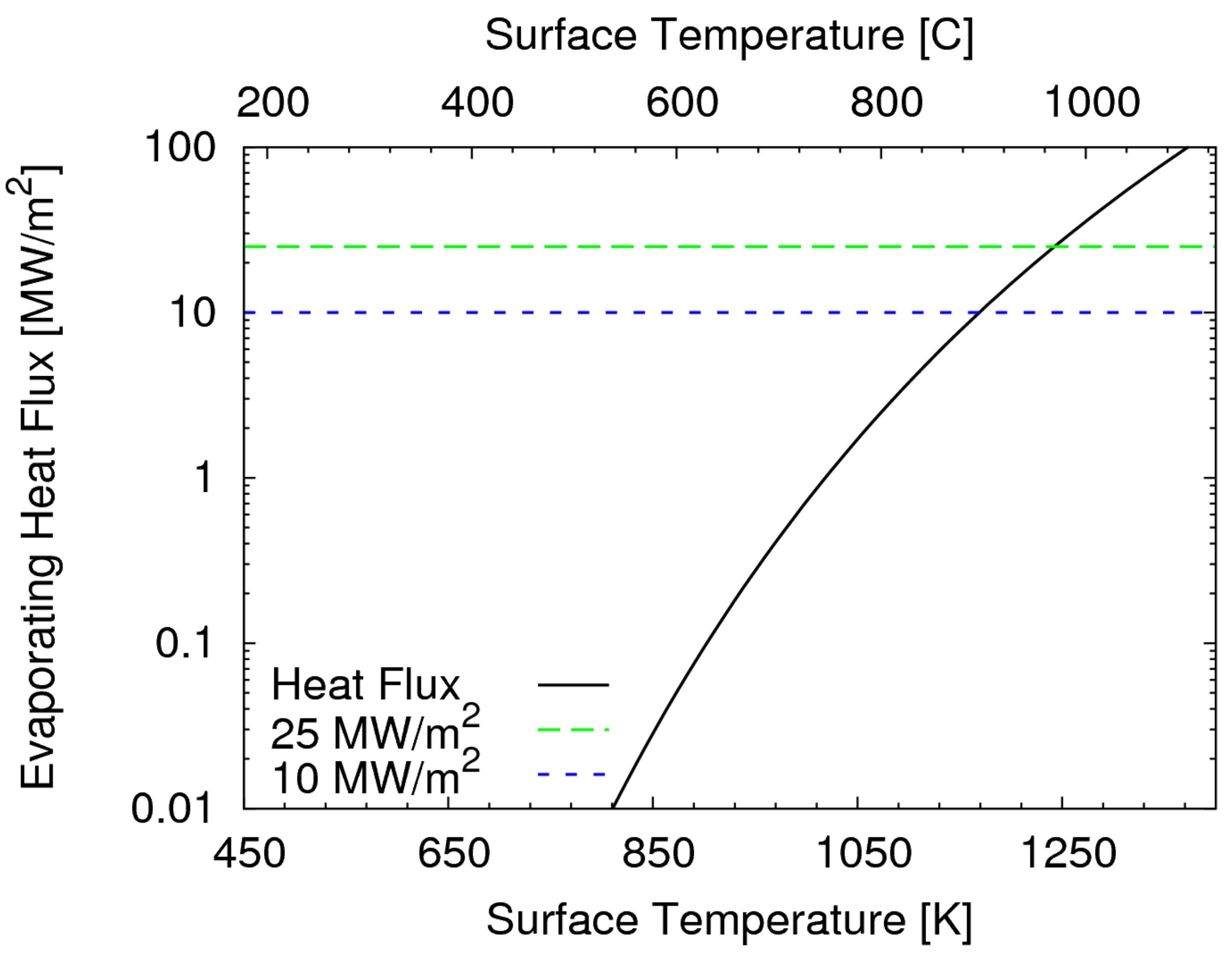
Liquid metal-structural compatibility



Li, Sn, Ga
Li, Sn, Ga
Li, Sn, Ga?
Li, Sn, Ga?
Li, Sn?, Ga?
Li, Sn, Ga
Li, Sn, Ga
Li, Sn, Ga
Li, Sn, Ga
Li?, Sn, Ga

Zinkle and Ghoniem, **FED** 2000. (Sn and Sn-Li used interchangeably)
“The Liquid Metal Handbook” Liquid-metals handbook”, United States Office of Naval Research. U.S. Govt. Print. Off. 1950. (Gallium estimates)

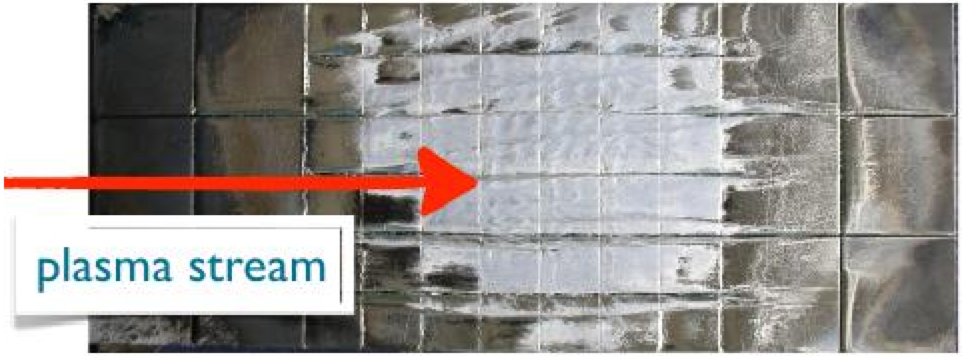
Evaporative self-cooling by Lithium



Why liquids? Because solids may not extrapolate

- Two major failure modes for solids that are known:
  - Melting (transient heat loads)
  - Net-reshaping (erosion, migration, redeposition)
- Some speculative failure modes:
  - Neutron-PMI synergistic effects (aside from bulk material changes)
  - Steady-state, self-regulating walls?

B. Lipschultz, et al., "Tungsten melt effects on C-MOD operation & material characteristics", 20-PSI, Aachen, Germany, May, 2012.



Coenen, et al., "Evolution of surface melt damage, its influence on plasma performance and prospects of recoverhy", 20-PSI, Aachen, Germany, May, 2012.
Klimov, et al., JNM **390-391** (2009) 721.

Wall erosion/redeposition not mitigated by divertor configuration

Table 1
Rough estimate of net erosion rate of main walls based on assumptions in text. Assumes 100% wall coverage by Be, B, C or W.

Device	P_{heat} (MW)	τ_{annual} (s/yr)	E_{load}^{year} (TJ/yr)	Beryllium net wall erosion rate (kg/yr)	Boron net wall erosion rate (kg/yr)	Carbon net wall erosion rate (kg/yr)	Tungsten net wall erosion rate (kg/yr)
DIII-D	20	10^4	0.2	0.13	0.11	0.08	0.16
JT 60SA	34	10^4	0.34	0.22	0.19	0.15	0.27
EAST	24	10^5	2.4	1.6	1.2	0.82	1.8
ITER	100	10^6	100	77 (29)[a]	64	44 (53)[a]	92 (41)[a]
FDF	100	10^7	1000	610	500	340	740
Reactor	400	2.5×10^7	10,000	6500 (21,000)[b]	5300	3700	7900 (5000)[b]

*P.C. Stangeby, et al., JNM **415** (2011) S278.*

- Charge-exchange processes create steady wall-flux
- Low density plasma at first wall reduces local redeposition
- **1000s of kgs** of eroded material migrating around tokamak vessel
- Likely to redeposit in locations where cooler plasmas exist or behind baffled areas of machine
- Do PFCs remain functional with large amounts of redeposited material?
  - Need very high duty-factor to even study the problem!

Magnum-PSI plasmas similar to NSTX divertor conditions

Parameter	Magnum-PSI	NSTX discharges with heavy lithium (Liquid Lithium Divertor)
Power [kW]	60	4 MW NBI **(15MW NSTX-U)**
Pressure source [Pa]	10^4	N/A
Pressure target [Pa]	<3	~0.1-1 (OEDGE modeling)
Ti target [eV]	0.1-10	1-50?
Te target [eV]	0.1-10	1-15 (non-Maxwellian)
Ni target [m^{-3}]	10^{20}-10^{21}	5×10^{20} at SP
Ion flux target [$m^{-2}s^{-1}$]	10^{24}-10^{25}	2×10^{23} at SP
Power flux [MW m^{-2}]	10	2-5 at ~5 deg. Incl. **(25 unmit.)**
B [T]	1.9	0.6 **(1T NSTX-U)**
Beam diameter [cm]	10-1.5	~4cm FWHM
Pulse length [s]	12-110	1s **(5s-10s)**
Target size [cm]	3cm – 60x12	Order~10cm
Bias [V]	$-100 < V_{target} < 0$	$-20 < V_{floating} < 20$

Several high-Z PFC fabrication concepts will be developed in parallel w/lab studies; demonstrated readiness affects pacing

- High heat flux regions (strike-point regions)
  - TZM or W lamellae, or TZM tiles (if workable)

- Intermediate heat flux regions (cryo-baffles, CS midplane)
  - TZM tiles or TZM/W lamellae

- Low heat flux regions (passive plates, CS off-midplane)
  - W-coated graphite

- Additional pulse-length extension (10-20s) at high power (~15MW) would require actively-cooled divertor PFCs

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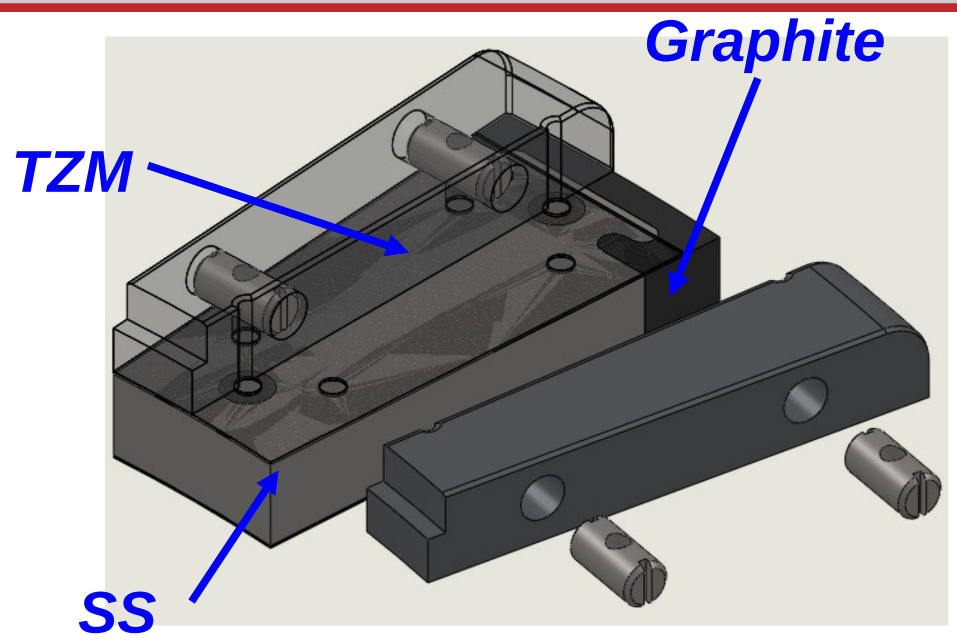
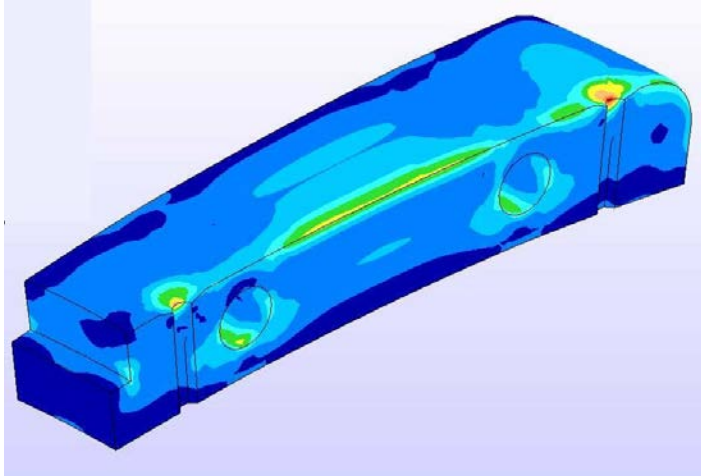
FY2011 tile design

- Design developed for FY2011 run campaign, inboard horizontal divertor (next to CHI)
- Split-top molybdenum to reduce eddy-current forces (NSTX design points)
- 2MW/m^2 for 2s average surface heat flux expected to be acceptable for avoiding fatigue limits of TZM



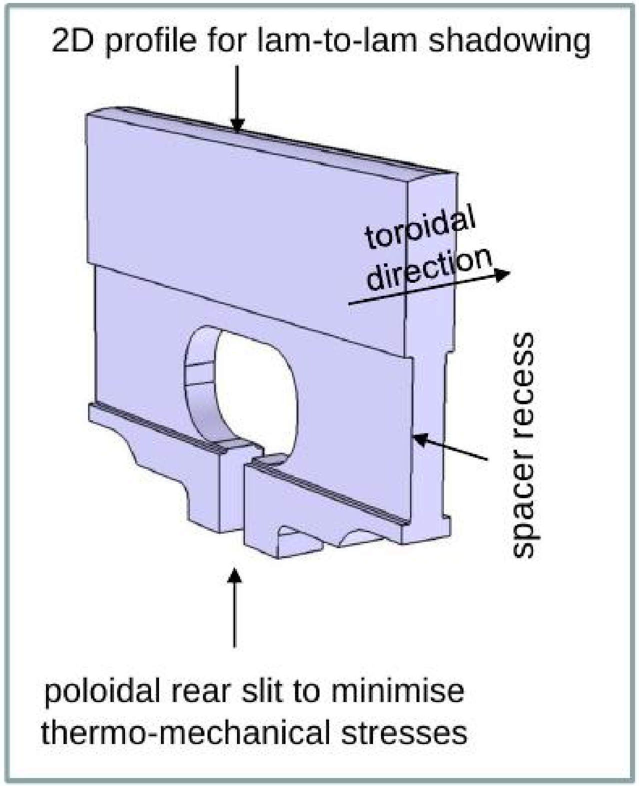


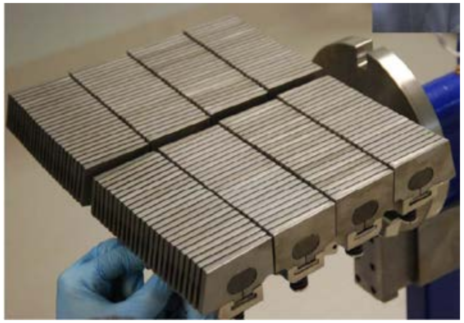
Thermo-mechanical stress

R. Woods Mo-tile FDR

Lamellae used on JET and CMOD divertors

- Designed for $7 MW/m^2$ (uniform) for <10s ($60 MJ/m^2$ total energy deposition)
- Lamellae depth determines thermal reservoir
- Cuts in toroidal and poloidal directions minimize eddy-currents and thermo-mechanical stress
- Complex shaping used to eliminate leading-edge effects

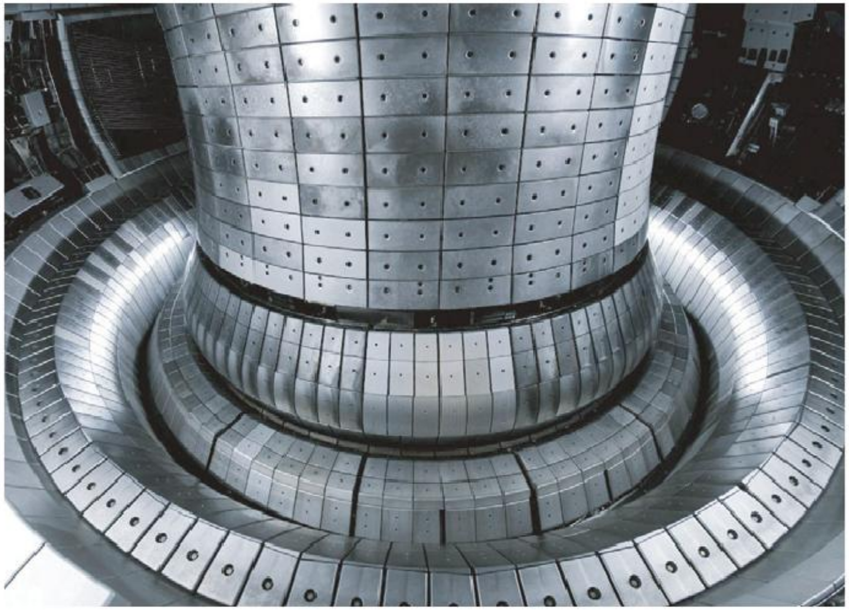




Ph. Mertens, 13th PFMC, 2011

Coatings of graphite substrates

- ASDEX converted machine to tungsten with the use of coatings
- Wall components coated with ~4 μm
- Divertor targets coated with ~200 μm
  - Despite extensive tests, still delaminated under peak heat fluxes >10MW/m^2
  - Switch in coating technologies due to repeated delaminations
  - Necessitated radiative divertor development



Neu, Phys. Scr. 2009.

Initial thoughts without rigorous engineering assessment of concepts

- NSTX-U parameters will make bulk tiles difficult to implement
  - Bulk-tile difficult to reduce thermal stresses and maintain thermal capacity
  - Lamellae seem to offer all the appropriate features
  - Not considering actively cooled targets (yet)

- Low-heat flux areas can probably use coatings
  - Assuming already fabricated ATJ to be used
  - Some batch testing recommended to ensure any CTE mismatch not “life-threatening”

Possible NSTX-U high-Z development plan

- FY 13 – Perform more rigorous engineering assessment of lamellae vs. bulk-tile for NSTX-U conditions (much of this would likely require ~1 FTE engineer, ~0.5 FTE designer/drafting + some tech time per year)
  - Identify coating technology (e.g. PVD vs. VPS) for use on ATJ tiles
  - Identify heat-flux facility for cyclic testing
- FY 14 – Fabricate prototype PFC tile for thermal testing at suitable facility
  - Test small lots of coated samples
  - Test PFC prototype
- FY 15 – Determine PFC interfacing issues with existing mounting hardware – final designs, procurements
  - Begin scenario development to control PFC energy deposition
  - PFC prototype testing to failure to establish absolute limits
- FY 16 – fabrication installation
  - Complete scenario development for high-Z protection
- FY 17 – operation with all high-Z

M&P diagnostic needs for high-Z upgrade on accelerated/incremental schedule

- M&P program written against baseline funding and consistent diagnostic set
  - Diagnostics consistent, PFCs transition
  - Expected modest implementation in first 5 years.
- With aggressive transition to high-Z walls the following would be advantageous
  - Expanded coverage of first-wall elements (passive plates) with particle sensors (probes) and spectroscopic coverage
  - Ensure core x-ray spectrometers are ready for operations to support high-Z transport studies and core accumulation