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Recent Progress of NSTX Lithium Program and Opportunities for Magnetic Fusion Research

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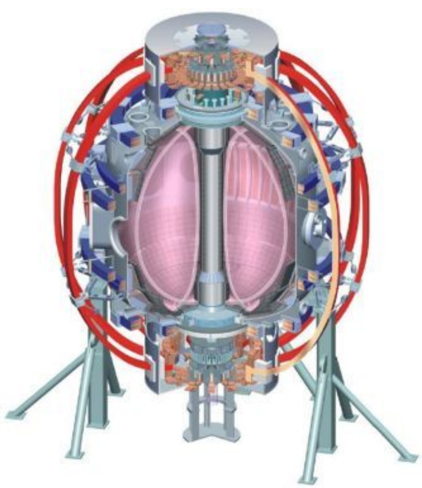
M. Ono, PPPL

For the NSTX Research Team

The 2nd International Symposium of Lithium Application for Fusion Devices

PPPL, April 27 -29, 2011



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Growing World Lithium Experimental Program!
(NSTX, LTX, FT-U, T11M, TJ-II, EAST, RFX, KTM)

NSTX Goal: To investigate effectiveness of lithium for divertor heat and particle control while enhancing plasma performance.

Lithium in NSTX proved to be an exceptionally powerful tool for H-mode plasma performance:

- *Global confinement improved through electron confinement improvement by ~ 20 – 30% with strong lithium pumping. Contributed to the highest confinement H-mode with H98y2 $\leq$ 1.7.*
- *H-mode power threshold significantly reduced by ~ 20 – 30%. Completely stabilized ELMs.*
- *Very little core lithium contamination (< 1%) found.*
- *Improved HHFW and EBW (RFs) performance by controlling edge density. Contributed to the non-inductive CHI start-up success by controlling impurities.*
- *Improved plasma shot reliability: shots / week increased ~ 40% over pre-lithium by controlling impurities.*

Fundamental understanding needed to predict toward future devices

Lithium Improved NSTX Operations

Plasma Shot Rate Improved by ~ 50% compared to pre-lithium

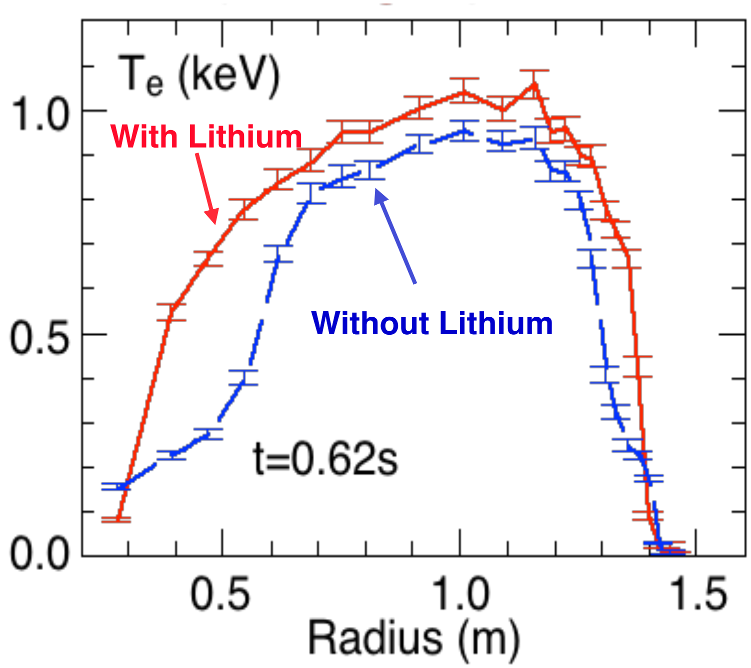
- **Enabled rapid recovery of experimental plasma operation after an extended vacuum vessel opening compared to boronization.**
- **Reduced oxygen impurity level and generally improved plasma reliability and performance.**
- **Conditioned PFCs to produce reproducible shots and eliminated the need for helium GDC between shots.**

NSTX Plasma Operation Statistics

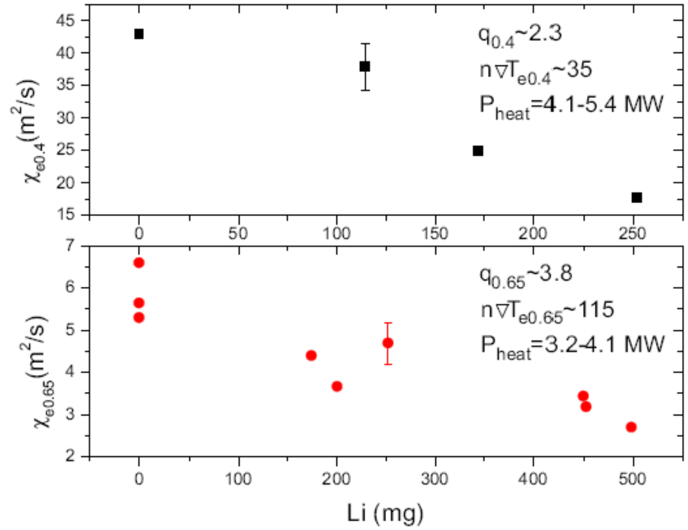
Year	Weeks	Shots	Shots/week	Lithium %
2010	15.4	2941	191	~ 100
2009	16.84	2750	163	92
2008	16.5	2570	156	46
2007	12.6	1890	150	69
2006	12.7	1615	127	0
2005	17.97	2221	124	0
2004	21.1	2460	117	0

We do not understand quantitatively why lithium is so effective for vent-recovery / plasma start-up (compared to boronization).

Lithium Coating Improved Electron and Total Confinement Enabled NSTX to achieve high performance plasmas



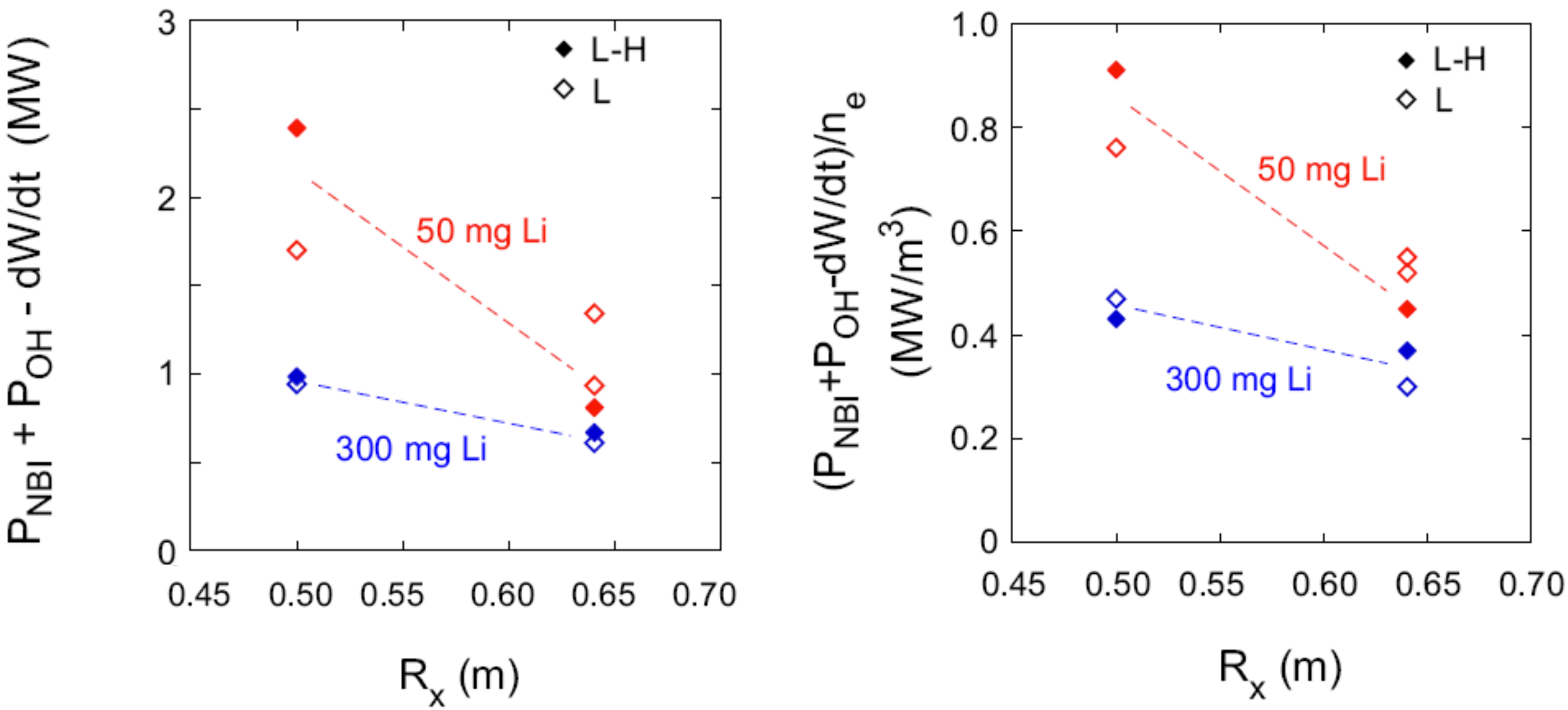
Electron confinement improving with lithium



S. Ding et al. published in PPCF 52 (2010) 015001

- **Lithium confinement improvement in NSTX is consistent with the common observation that reduced recycling improves confinement.**
- **High-k fluctuation measurements indicate that ETGs are not solely responsible for transport in Li conditioned plasmas.** **Y. Ren, APS, 2010**

Lithium Significantly Reduces H-mode Power Threshold Provide a Margin for ITER Particularly in the Early Phase

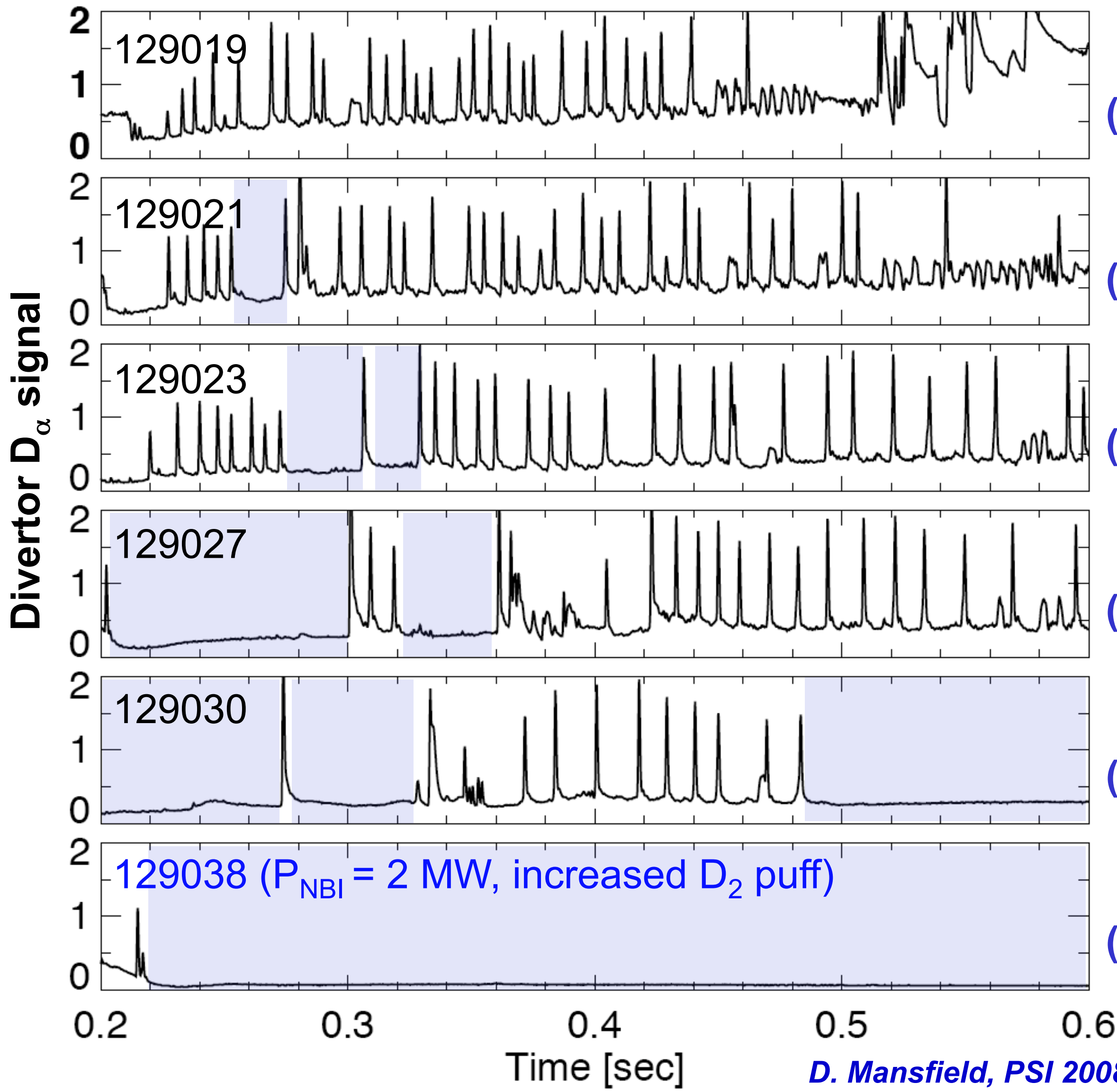


S.Kaye, D. Battaglia, R. Maingi et al., IAEA (2010)

- **The L-H transition power threshold is not yet understood quantitatively.**
- **Lithium likely to be playing a role due to lower edge collisionality, improved edge confinement, and higher edge temperature.**
- **Further experimental scan and gyro-kinetic simulation are being performed.**

ELMs Stabilized by Lithium Application

Consistent with Peeling-Ballooning Mode Stability Theory



D. Mansfield, PSI 2008

Lithium deposited (accumulated) (mg)
0 (0)
110 (110)
150 (426)
170 (1056)
264 (1624)
715 (5355)

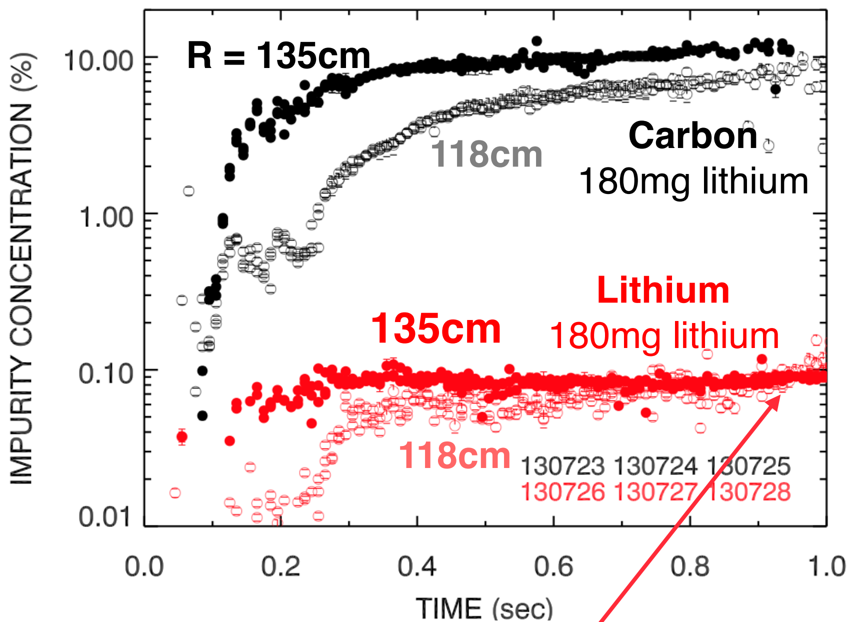
Status:

- **ELM stabilization is consistent with the edge profile modification by lithium particularly the edge density reduction**
- **Lithium induced ELM-free discharges tend to accumulate carbon and higher Z impurities**
- **Impurity reduction being explored**

R. Maingi et al., PRL 2009

Lithium Concentration in Plasma Core Remains Very Low Compared to Higher Z Carbon

- **Quantitative measurements of C^{6+}, Li^{3+} with charge-exchange recombination spectroscopy**
- **n_C/n_{Li} ~ 100**
- **Hollow profiles early for both C and Li fill in as time progresses**



No sign of Li accumulation in core

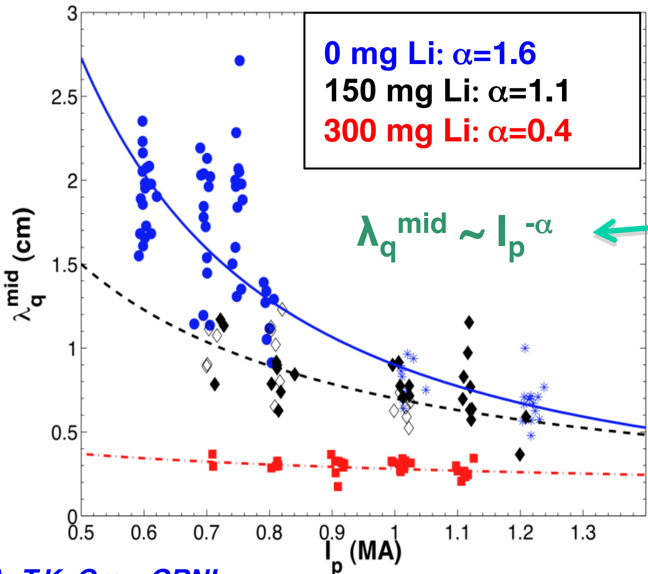
Status:

- **Low lithium core contamination continues to hold true for LLD operation**
- **Very good news for lithium based divertor concepts**
- **Low level of lithium accumulation consistent with neo-classical theory (C.S. Chang *et al.*)**
- **A quantitative model is still lacking**

R. E. Bell (PPPL)

Divertor Heat Flux Width Decreased with Lithium

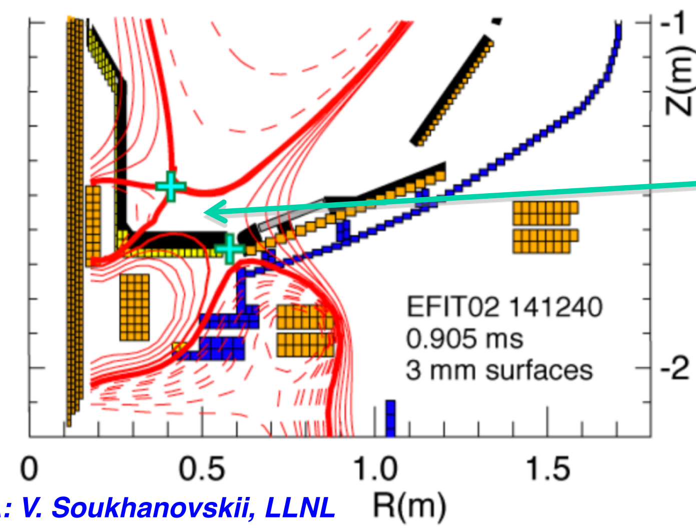
"Snowflake" Configuration Greatly Reduces Heat Flux



IAEA: T.K. Gray, ORNL

- Divertor heat flux width, magnetically mapped to the midplane, shows a strong decrease as I_P is increased
  - Potentially major implications for ITER
  - NSTX: λ_q^{mid} further decreases with Li

- **Divertor heat flux scaling is an active topic.**
- **No commonly accepted model has emerged.**
- **Does heat flux with lithium represent a "floor"?**



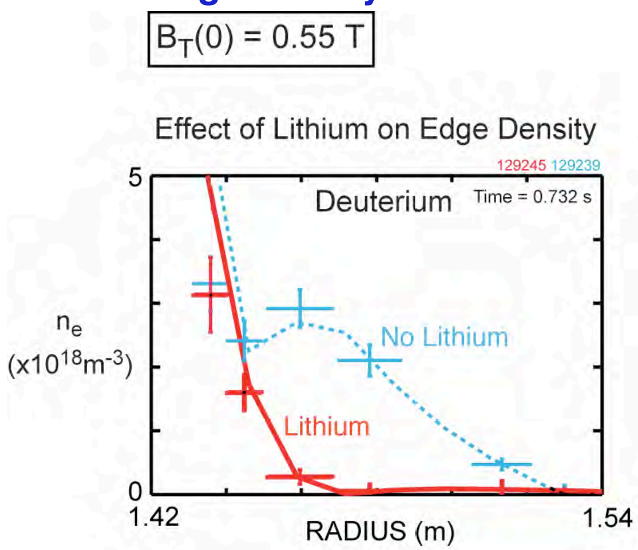
IAEA: V. Soukhanovskii, LLNL

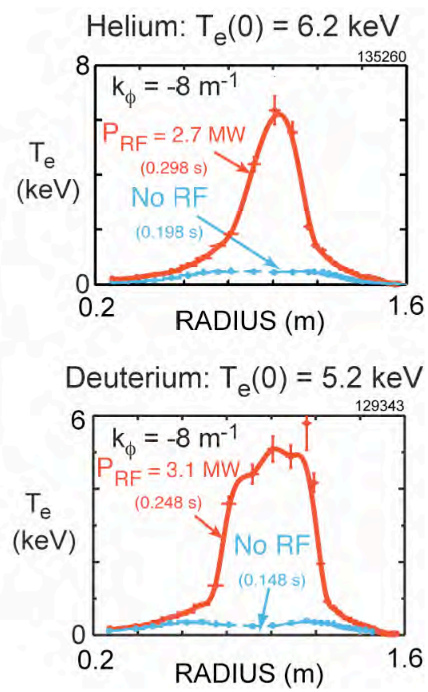
- Divertor heat flux inversely proportional to flux expansion over a factor of five
- Snowflake→ high flux expansion 40–60, larger divertor volume and radiation

→ U/D balanced snowflake divertor projects to acceptable heat flux < 10MW/m² in Upgrade at highest expected I_P = 2MA, P_{AUX}=15MW

HHFW Operation Benefited From Lithium Initially but Encountered Power Limit After Heavy Lithium Use in 2010

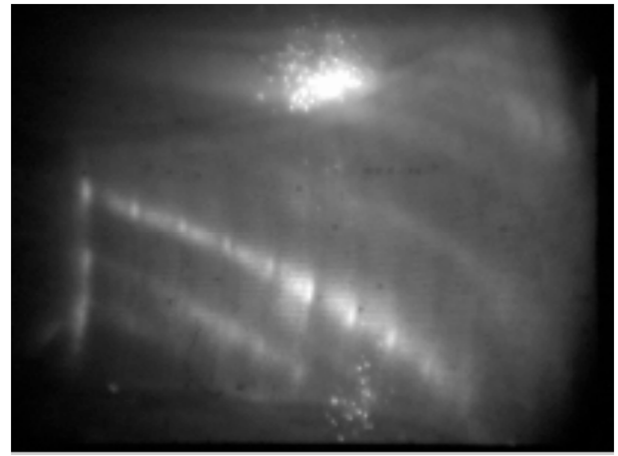
In 2009, heating efficiency improved by controlling the edge density with lithium





- **In 2010, air contamination during vessel vent with argon may have caused lithium dust formation resulting in arcing (lithium dust found on antenna)**
- **Issues need to be resolved with early HHFW operation and improved antenna conditioning**

Li coatings contribute to arcs occurring where it accumulates at top and bottom

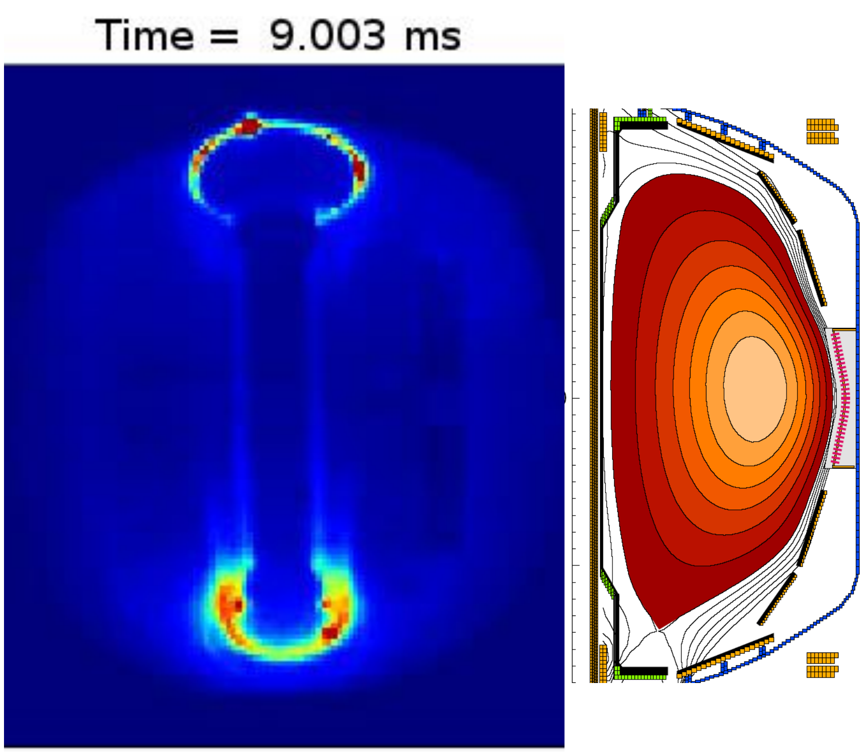


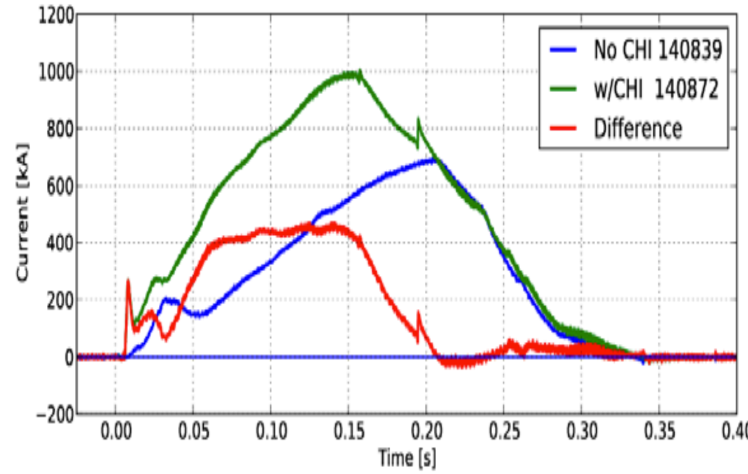
HHFW Antenna before and after cleaning

P. M. Ryan (ORNL) et al., US-Japan (2011)

After lithium application, Coaxial Helicity Injection produced lower density, lower inductance start-up

Low Z impurity reduction during CHI produced OH compatible plasmas





- **Discharge cleaning of lower divertor plates or electrodes**
- **Avoidance of absorber arcs by control coils**
- **Lithium evaporation of lower divertor surfaces**

R. Raman, B. Nelson et al., U Washington, PRL 2010

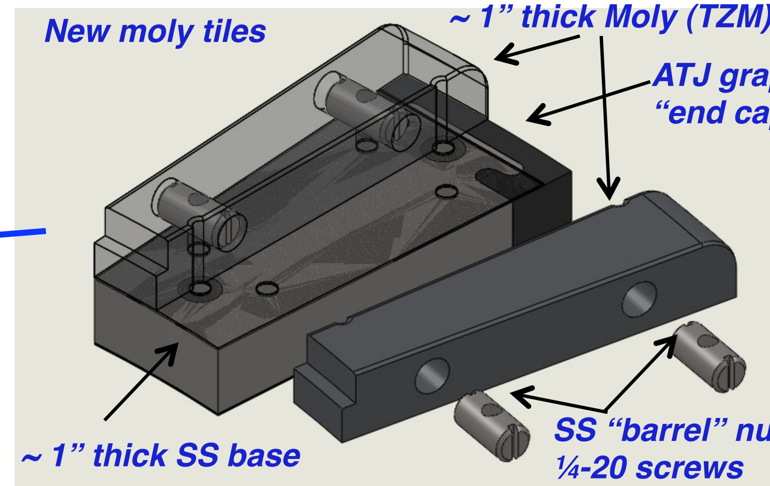
2011-2012 Plans
- **Molybdenum cathode with Lithium coating for higher CHI current ~ 0.5 MA**
- **CHI + HHFW scenarios with higher T_e**
- **Provide low-inductance target for stability and pulse-length optimization**

Molybdenum Tiles Installed on Inboard Divertor Supplement Molybdenum Surface of Liquid Lithium Divertor

Aim to Reduce Carbon Influx

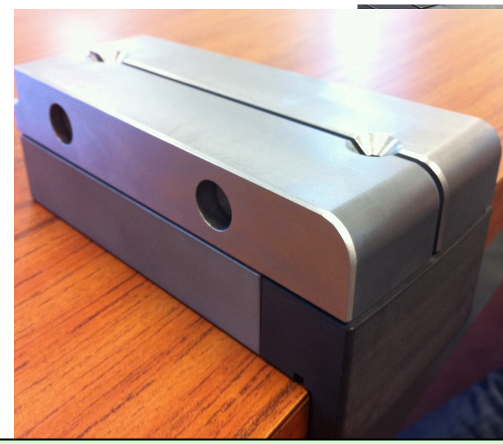
LLD Plates: R. Nygren, SNL

Split-top Moly on SS tile satisfies requirements



Molybdenum tiles on inboard divertor

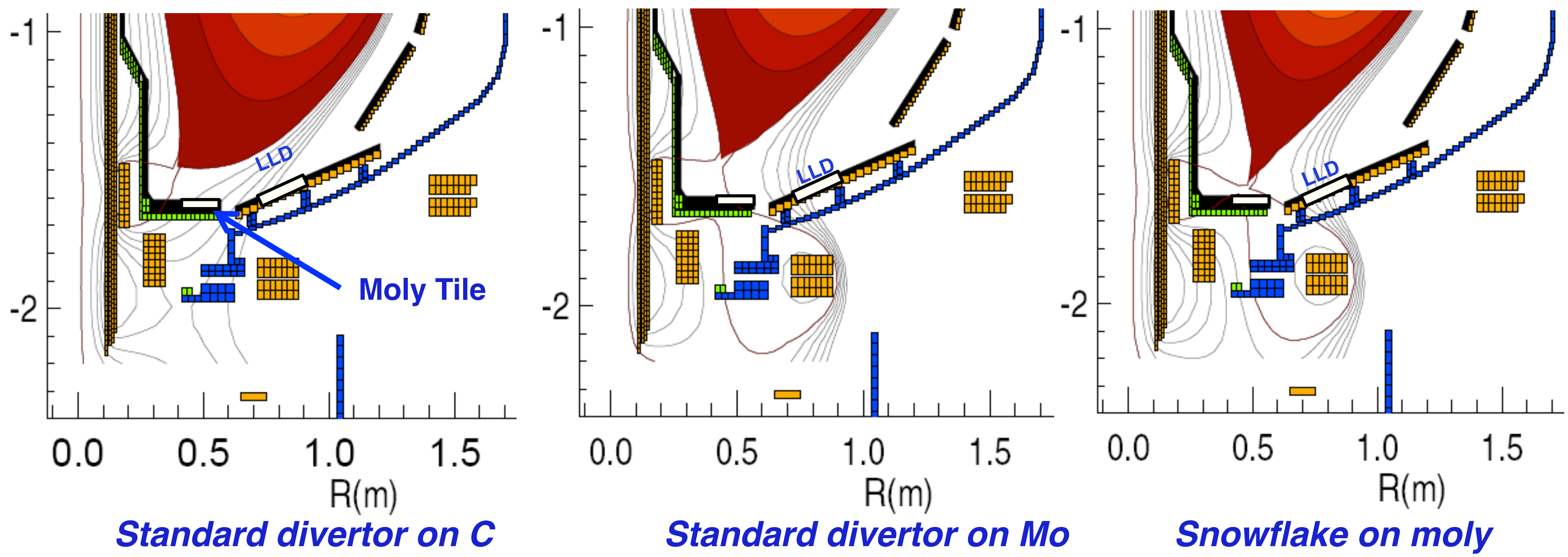
- Replace 48 second row tiles with 1" moly tiles
- Includes three tiles with embedded diagnostics
- Lithium coating with LITER ~2 x outer LLD rate
- Plasma heating can liquify lithium surface



- No sign of significant influx of moly even strike point was on LLD
- Plasma heating turned out to be very effective surface heater for LLD

Addition of IBD Mo Tiles Enabled Important Divertor Studies and Extend Liquid Lithium & Moly Divertor Research

- Help quantify fraction of core C coming from lower divertor for high-δ shapes
- Potentially reduce C content of Li ELM-free scenarios
- Characterize Mo performance to inform choice of div/CS PFC in Upgrade
- Apply Li (LiTER) to divertor moly surfaces for partial/full liquid lithium
- Provide metal cathode surface for CHI to reduce impurity generation



2010 IAEA: V. Soukhanovskii, LLNL

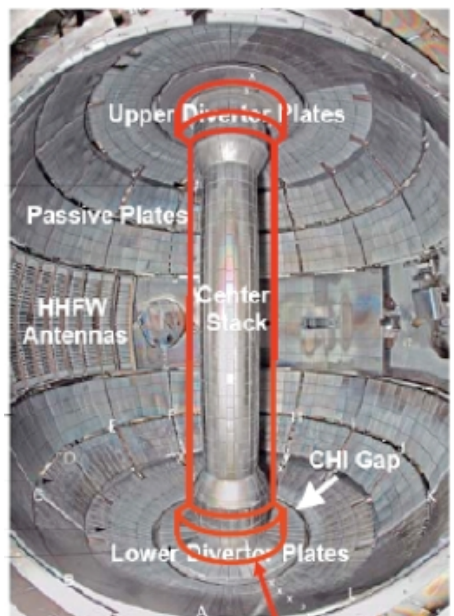
Moly-tile implementation for FY 2011-2012 run would provide valuable information for the post-upgrade PFC options

Upgrades provide a major step toward FNSF

Access to low collisionality and fully non-inductive operations

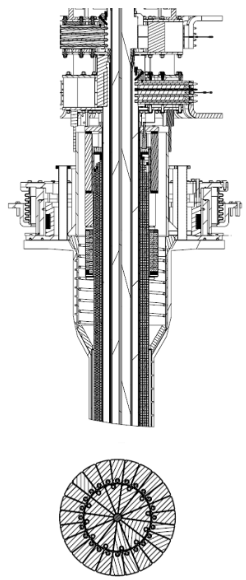
	NSTX	NSTX Upgrade	Fusion Nuclear Science Facility
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.5
Plasma Current (MA)	1	2	4 → 10
Toroidal Field (T)	0.5	1	2-3
P/R, P/S (MW/m,m^2)	10, 0.2*	20, 0.4*	30 → 60, 0.6 → 1.2

** Includes 4MW of high-harmonic fast-wave (HHFW) heating power*



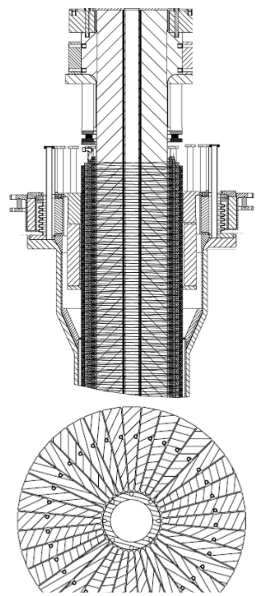
Outline of new center-stack (CS)

Present CS



TF OD = 20cm

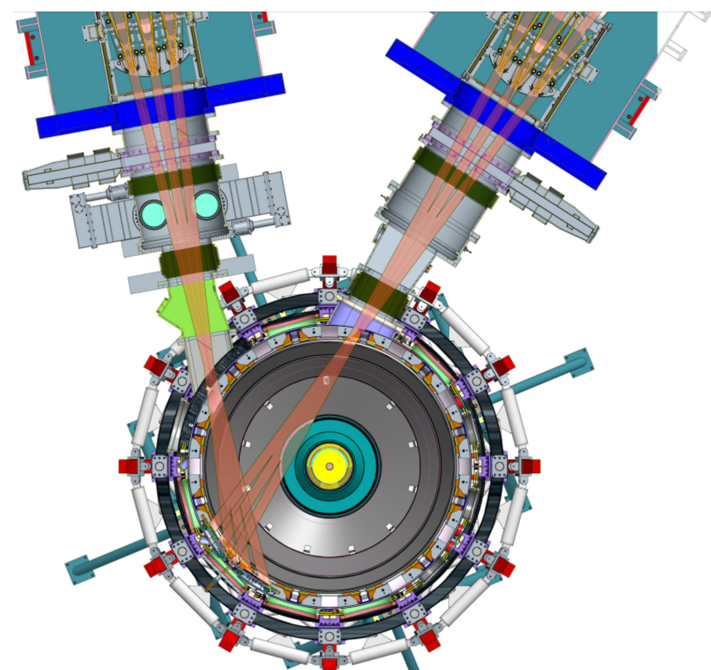
New CS



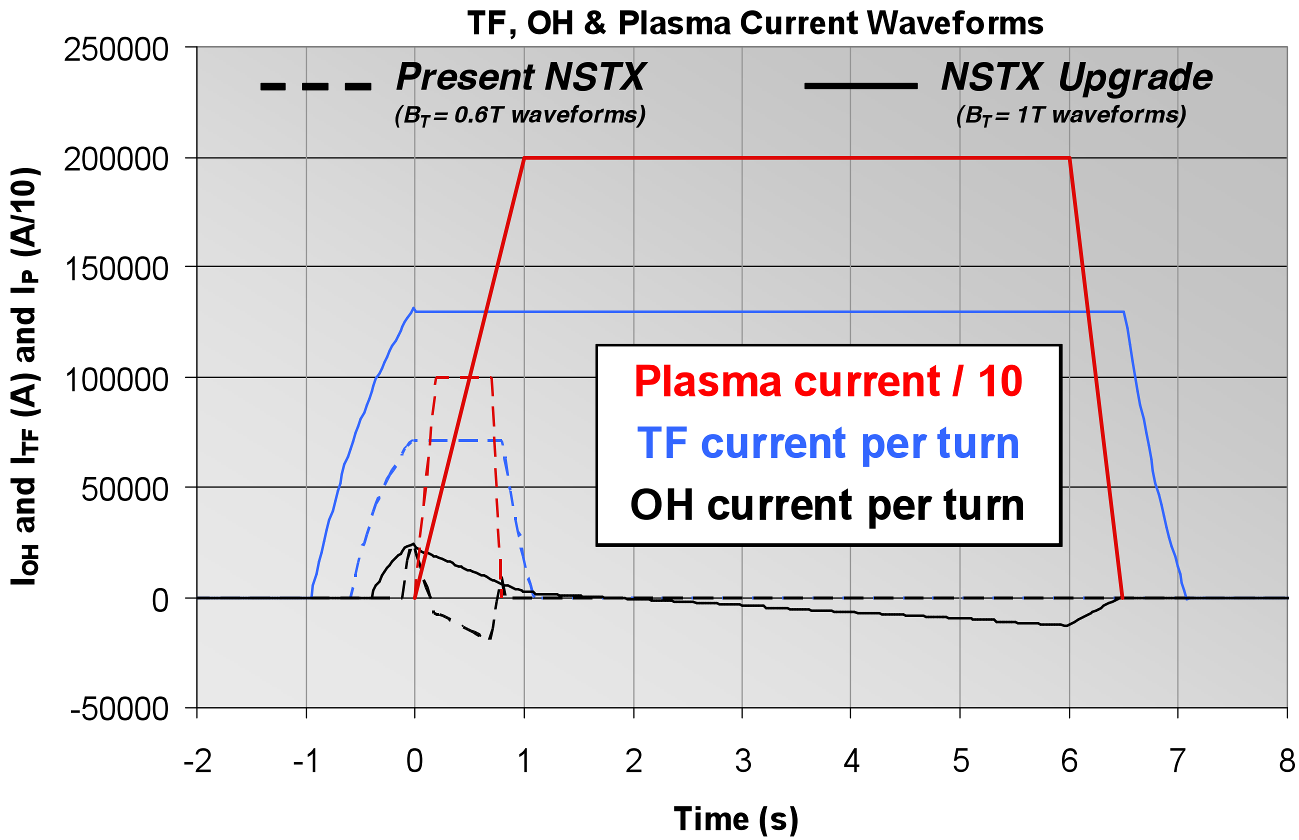
TF OD = 40cm

New 2nd NBI (R_{TAN}=110, 120, 130cm)

Present NBI (R_{TAN}= 50, 60, 70cm)



Upgrade provides substantial increase in device performance
An order of magnitude enhancement in $n\tau T$



	Base NSTX	NSTX Upgrade
R_0 [m]	0.854	0.934
Min. aspect ratio	1.28	1.5
I_p [MA]	1	2
B_T [T]	0.55	1
T_{pulse} [s]	1	5
$T_{repetition}$ [s]	600	1000
R_{center_stack} =R_0-a [m]	0.185	0.315
$R_{antenna}$=R_0+a [m]	1.574	1.574
Total OH flux [Wb]	0.75	2.1

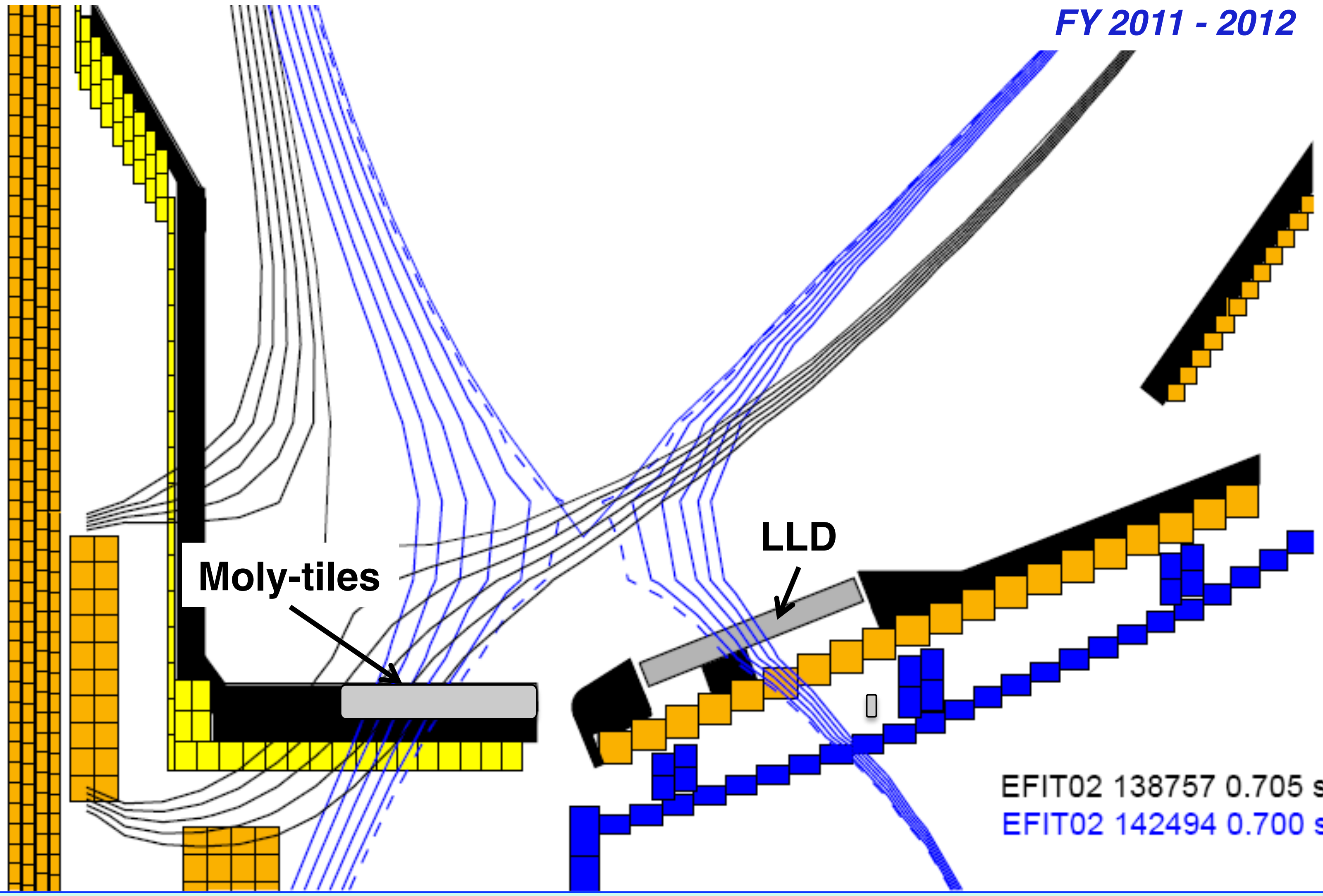
Relative performance of Upgraded NSTX vs. Base:

NBI power increased 2 x
Available OH flux increased 3x, 3-5x longer flat-top
I_P increased 2x, B_T increased 2x at same major radius
Plasma stored energy increased up to 4x (0.25→ 1MJ)

Aim to Reduce Carbon Influx with Moly-tiles and LLD

All moly divertor surfaces + lithium to simplify the chemistry

FY 2011 - 2012

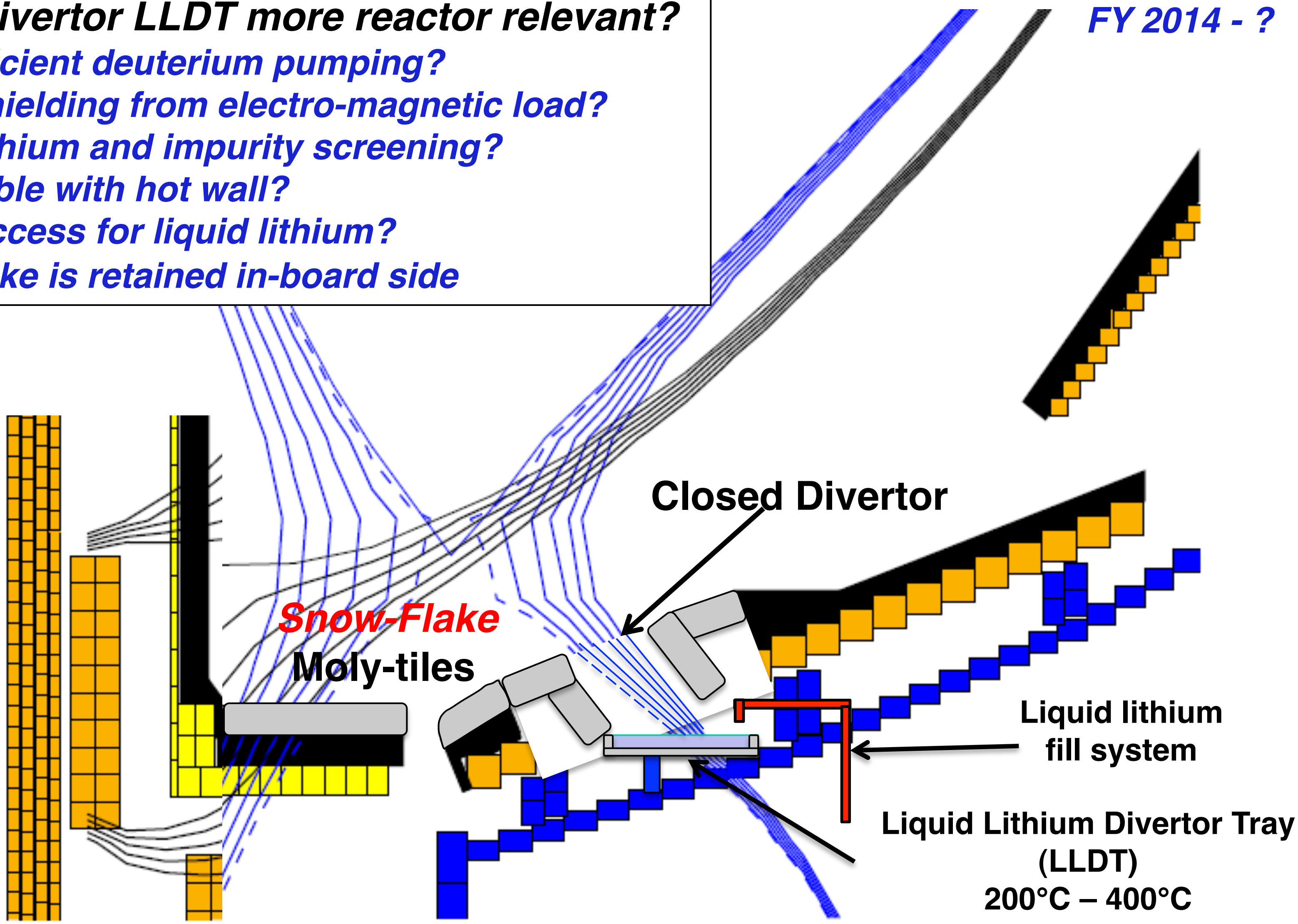


However, no LLD for post-upgrade due to ~ x 4 electro-magnetic forces

For Post-Upgrade, Divertor Upgrade Will Be Examined
a "Closed" Divertor with Liquid Lithium Divertor Tray Possible?

Closed divertor LLDT more reactor relevant?
- *More efficient deuterium pumping?*
- *Better shielding from electro-magnetic load?*
- *Better lithium and impurity screening?*
- *Compatible with hot wall?*
- *Easier access for liquid lithium?*
- *Snow-flake is retained in-board side*

FY 2014 - ?



NSTX Research Benefited Greatly from Lithium Application

Lithium Has Exciting Near-Term and Longer Term Potentials

- **NSTX tested applications of lithium in diverted tokamak configuration. Potential importance for fusion energy development is summarized:**
  - **Electron energy confinement increased for improved plasma performance. Improved electron thermal confinement at edge.**
  - **Reduction in H-mode power threshold.**
  - **ELM stabilization through edge electron pressure profile modification**
  - **Lower edge density and impurity control benefited RF heating and non-inductive tokamak start-up.**
  - **Low lithium core dilution demonstrated, enhancing lithium utilization for the challenging divertor solutions.**
  - **Improved NSTX operational efficiencies.**
  - **Narrow divertor heat flux width, however.**
- **NSTX experimental results suggest potential benefits for near-term and longer term tokamak/ST fusion development path.**

We need to develop fundamental understanding of effects on lithium on plasma performance and assess its applicability for fusion reactors.