Liquid Lithium Divertor Development for the National Spherical Torus Experiment

Laura Berzak

Practicum Advisor: Richard Nygren, SNL/NM

Thesis Advisors: Robert Kaita and Richard Majeski, PPPL

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Outline

* Brief overview of magnetic fusion energy (MFE)
* Overview of the benefits of liquid lithium usage in MFE
* Description of the National Spherical Torus Experiment (NSTX)
* Brief survey of NSTX Li experiments
* Description of the Liquid Lithium Divertor (LLD)
* Overview of LLD material testing
* Current status of NSTX and the LLD

Fusion Energy

Goal: Generate power, like stars, through fusion reactions

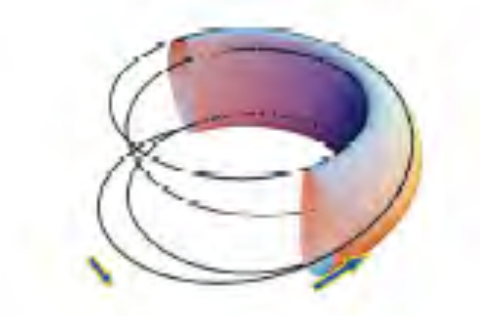
Example Reaction: $D + T \rightarrow n + \alpha + 17.6\ \text{MeV}$

Benefits:

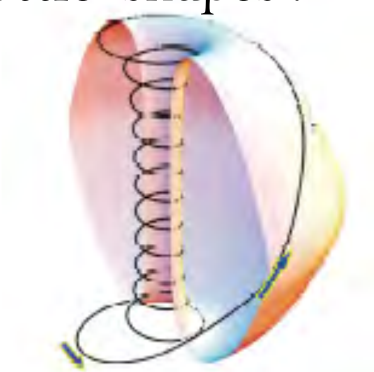
* Widely available input fuel source – sea water
* Clean energy production – no greenhouse gas emission
* No risk of runaway chain reaction leading to nuclear accident
* No high-level radioactive, nuclear waste

Magnetic Fusion Energy

* Fusion requires a balance of density, temperature, and confinement time
* Required temperature too high for physical containment → magnetic bottle
    * Form a plasma inside a physical vacuum vessel
    * Constrain plasma particles to closed magnetic field lines
    * Supply external heating in the form of neutral beam injection or radiofrequency heating
    * Highly energetic particles collide leading to fusion reactions and energy production
* Several potential magnetic bottle 'shapes':



Advanced Tokamak



Spherical Torus

Compact Stellarator

First Wall Challenges

* Even with magnetic confinement, neutral particles (e.g. neutrons) and some energetic plasma particles escape and strike inner surface of vacuum vessel

* Energetic plasma particles on open field lines will strike limiter or divertor (structure defining plasma last closed flux surface)

* This leads to:

  * Neutron activation of physical structure

  * Ablation and sputtering of inner chamber wall

  * Introduction of cold impurities into plasma → peaked temperature profiles, flattened density profiles → gradient-driven plasma instabilities

100 nm (VPS W on C) (TEM)

* Typical first wall materials – C, W, Be, Mo, stainless steel – act as mirrors to plasma particles that strike them (high-recycling surfaces) and are difficult to replace when damaged

Liquid Lithium Usage

* Liquid metal (particularly lithium) usage as plasma-facing surface:
    * Reduce structural activation by reducing neutron fluence to solid physical chamber
    * Potential to regenerate sputtered wall particles
    * Low Z and low ionization potential of Li mean reduced concerns over wall material introduction into plasma
    * Liquid Li pumps hydrogenic species in nearly 1:1 ratio – plasma particles which escape confinement and strike Li surface do not reenter plasma → peaked density profile, flattened temperature profile → enhanced confinement

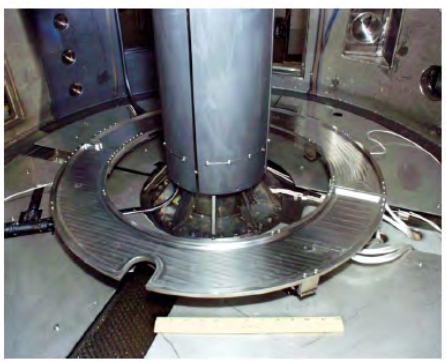
* Change boundary condition from 'mirror' to 'sponge' – reflective (high-recycling) to absorptive (low-recycling) surface as seen by plasma particles

Previous Results

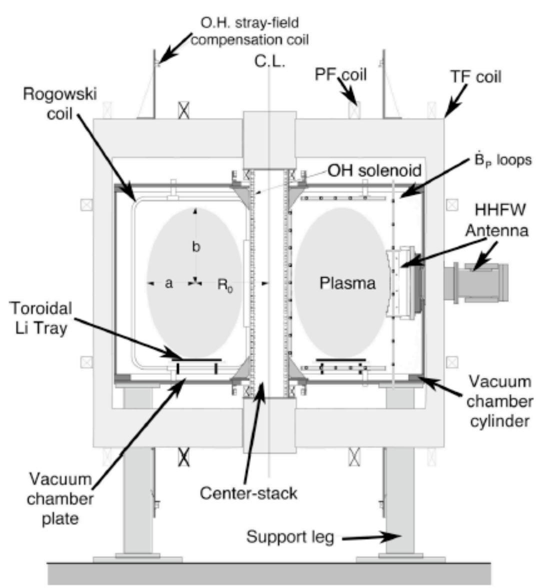
* Extensive theoretical predictions of reduced recycling yielding enhanced plasma stability and performance (Krasheninnikov, Zakharov, and Pereverzev, *Physics of Plasmas*, 2003)

* Experimental confirmation of plasma performance enhancement with liquid lithium usage:

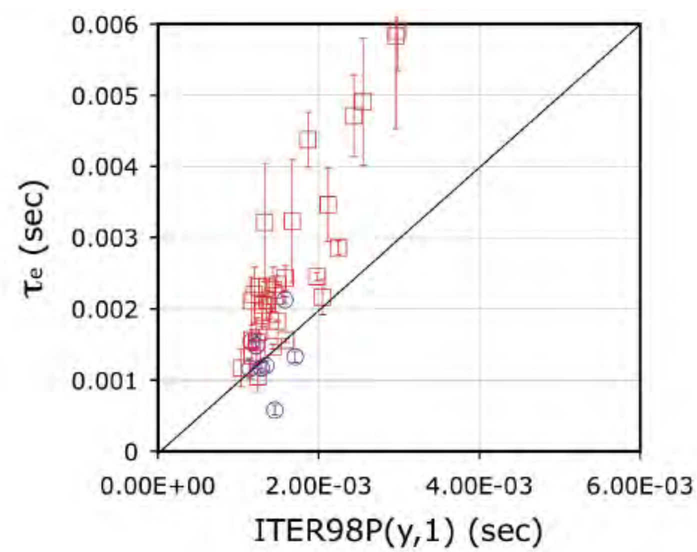
  * Dramatic improvements even without a fully liquid lithium first wall – e.g. tray limiter on spherical torus CDX-U



CDX-U tray limiter



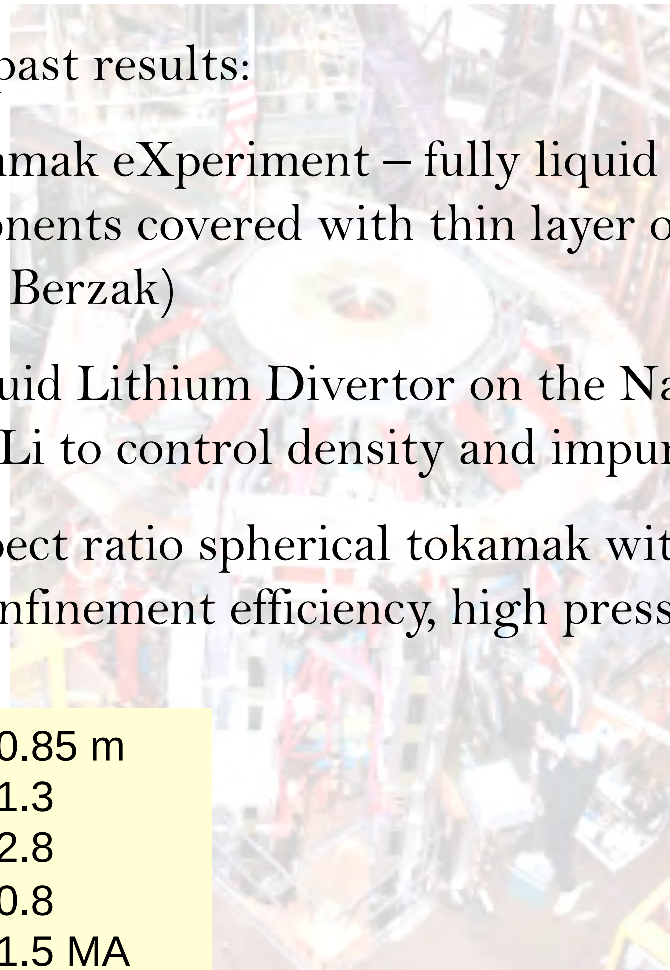
CDX-U schematic



Red: Li, Blue: no Li

Largest improvement in energy confinement time for ohmic (resistively heated) tokamak

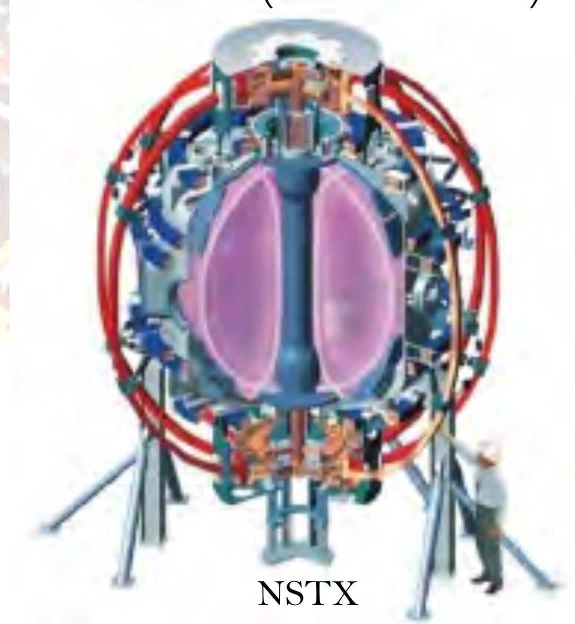
Next Steps



- Expand upon promising past results:
  - LTX – Lithium Tokamak eXperiment – fully liquid lithium first wall, ~90% plasma-facing components covered with thin layer of liquid lithium (poster tomorrow, L. Berzak)
  - LLD in NSTX – Liquid Lithium Divertor on the National Spherical Torus eXperiment - utilize Li to control density and impurity influxes
    - NSTX – low aspect ratio spherical tokamak with divertor, strong toroidal shaping, high confinement efficiency, high pressure-related ('self-driven') plasma current

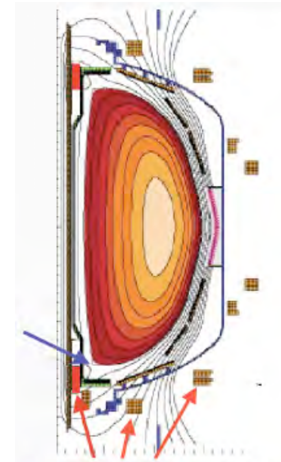
Major Radius R_0	0.85 m
Aspect Ratio A	1.3
Elongation κ	2.8
Triangularity δ	0.8
Plasma Current I_p	1.5 MA
Toroidal Field B_T	0.55 T
Pulse Length	1.5 s
NB Heating (100 keV)	7 MW
$\beta_{T,tot}$	up to 40%

NSTX parameters



NSTX

Lithium Experiments on NSTX



* Lithium campaign in three-phase approach:
    I. Lithium Pellet Injector (2005-2009)
    II. Lithium Evaporator (2006-2009)
    III. Liquid Lithium Divertor (2009-2012)

* Improvements in stored energy and pulse length with Li led to installation of dual LIThium EvaporatoRs (LITERs) in 2008

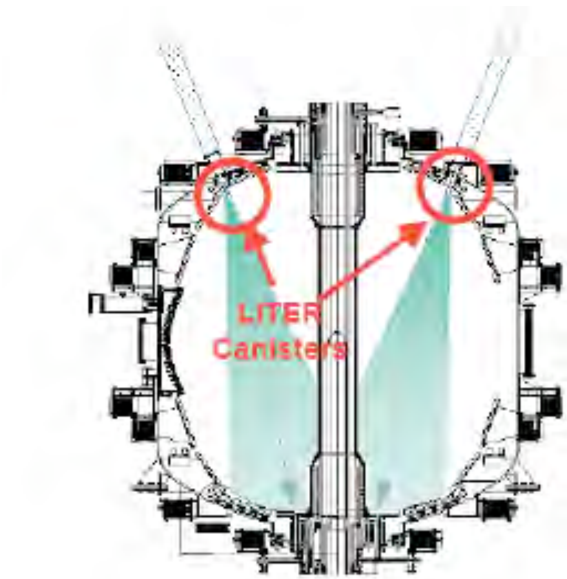


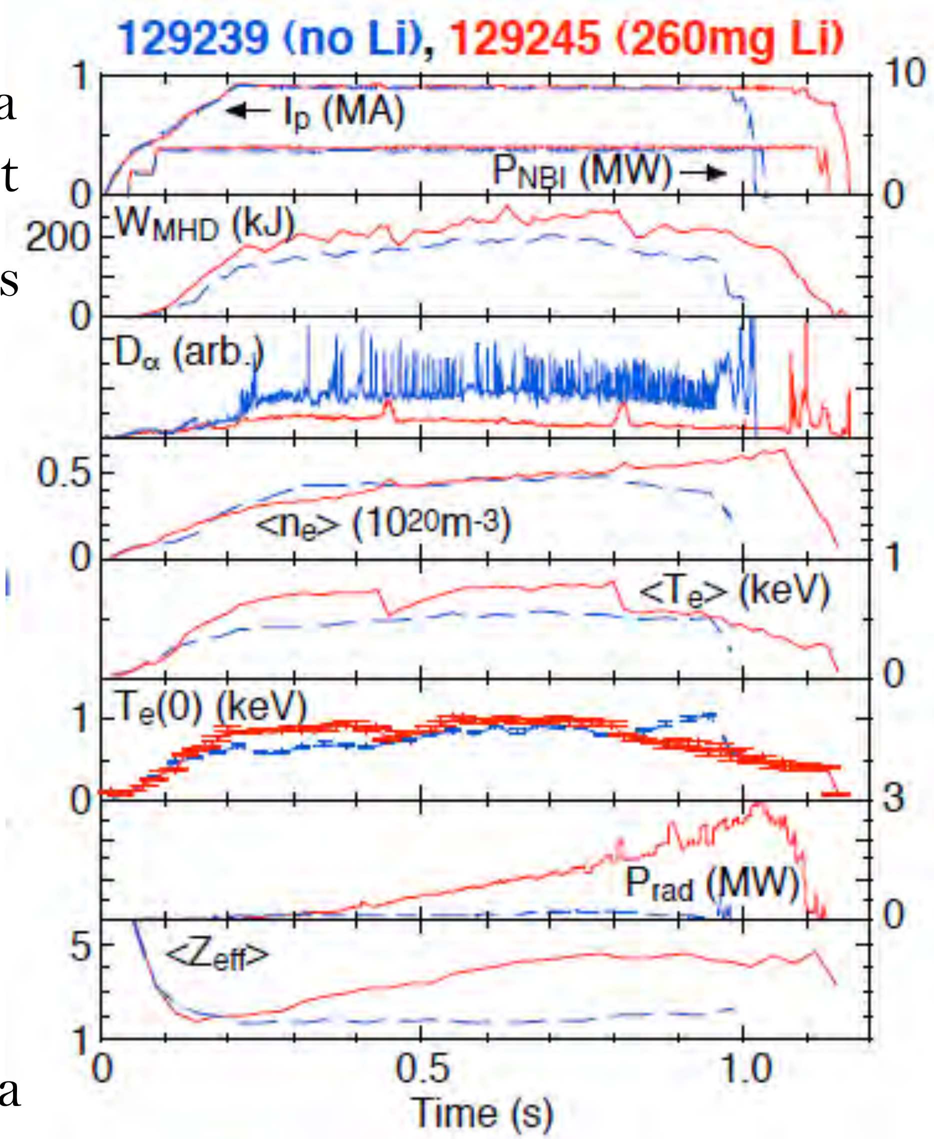
Diagram of dual LITER locations

Allows for full coverage of lower divertor region and high deposition rate (10-70 mg/min)

Interior view of NSTX

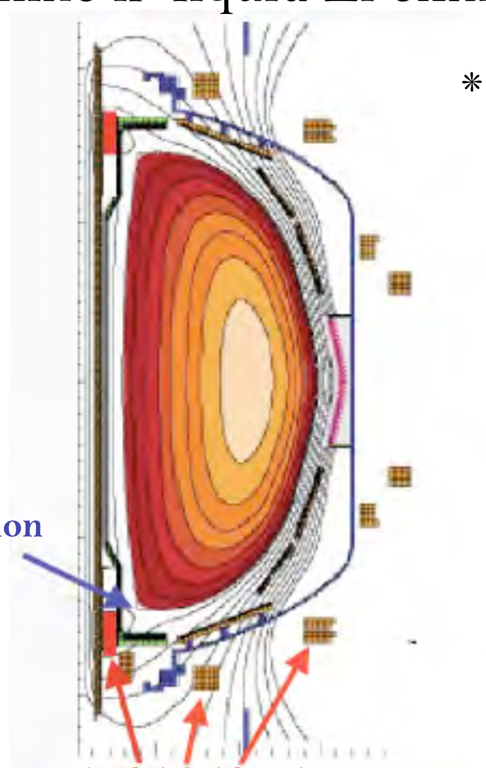
LITER Results

* Solid Li evaporated coatings reduce deuterium recycling, suppress plasma instabilities and improve confinement
* Broader electron temperature reduces internal inductance and improves efficiency of driving plasma current
* Suppression of edge plasma instabilities leads to
  * Increase in duration of plasmas relative to discharges without Li evaporation
  * Impurity accumulation increase seen in higher radiated power (P_{rad}) and higher 'effective' plasma charge (Z_{eff})

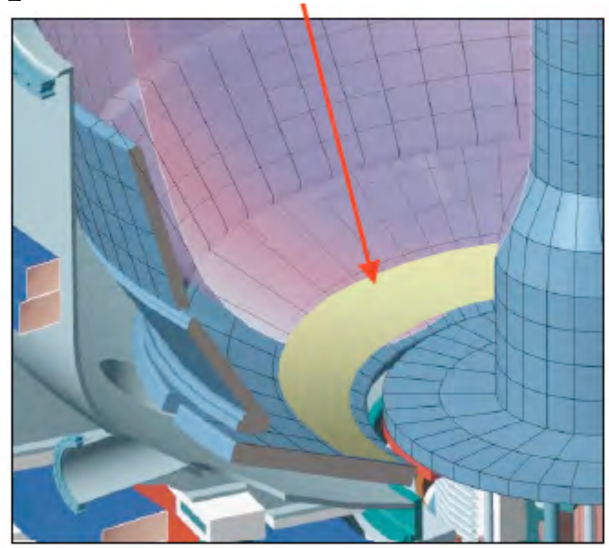


Liquid Lithium Divertor

* High-recycling observed in divertor region
* Promising results achieved with LITER have motivated next step in Li campaign → LLD
* Replace section of divertor region with full, toroidal liquid lithium surface
* Determine if liquid Li exhibits more deuterium pumping than solid coatings



* Utilize LLD for density control and eventually for power handling in long pulse, high heat flux operation



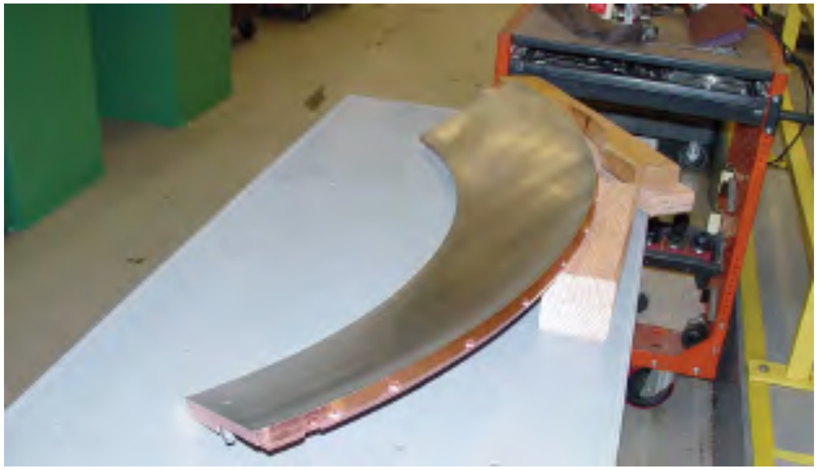
LLD Materials Testing

* Materials testing is a key component to optimizing LLD functionality
* Analyses include temperature and IR data from bench-top testing as well as vacuum testing of material
* Choice of materials will affect:
  * Deuterium retention
  * Self-cooling and heat conduction
  * Potential flow rate or rate of capillary action
* Considerations include:
  * Size of Li inventory in machine
  * Magnetic fields near Li
  * Length of time power deposited in material
  * Amount of power deposited

LLD Materials Testing

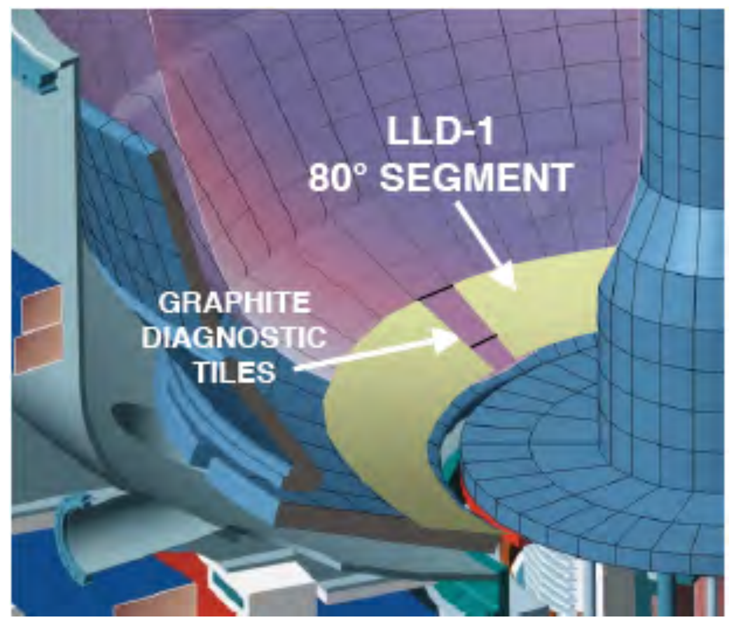
* Initial concept: utilize porous Mo foam to contain and 'wick' Li to surface
  * Allows consistent surface of liquid Li to face plasma
  * Could provide ability to continually replenish Li available for plasma interaction
  * Search for material which allows Li to 'wet' the surface
* Current concept: utilize plasma-sprayed Mo surface on Cu substrate with thin stainless steel liner
  * Forgo Li replenishment with more complicated porous Mo in first LLD version
  * Mo surface enables satisfactory 'wetting' by liquid Li and small total thickness of substrate (Mo and SS) permits Cu base to serve as heat sink

LLD Design



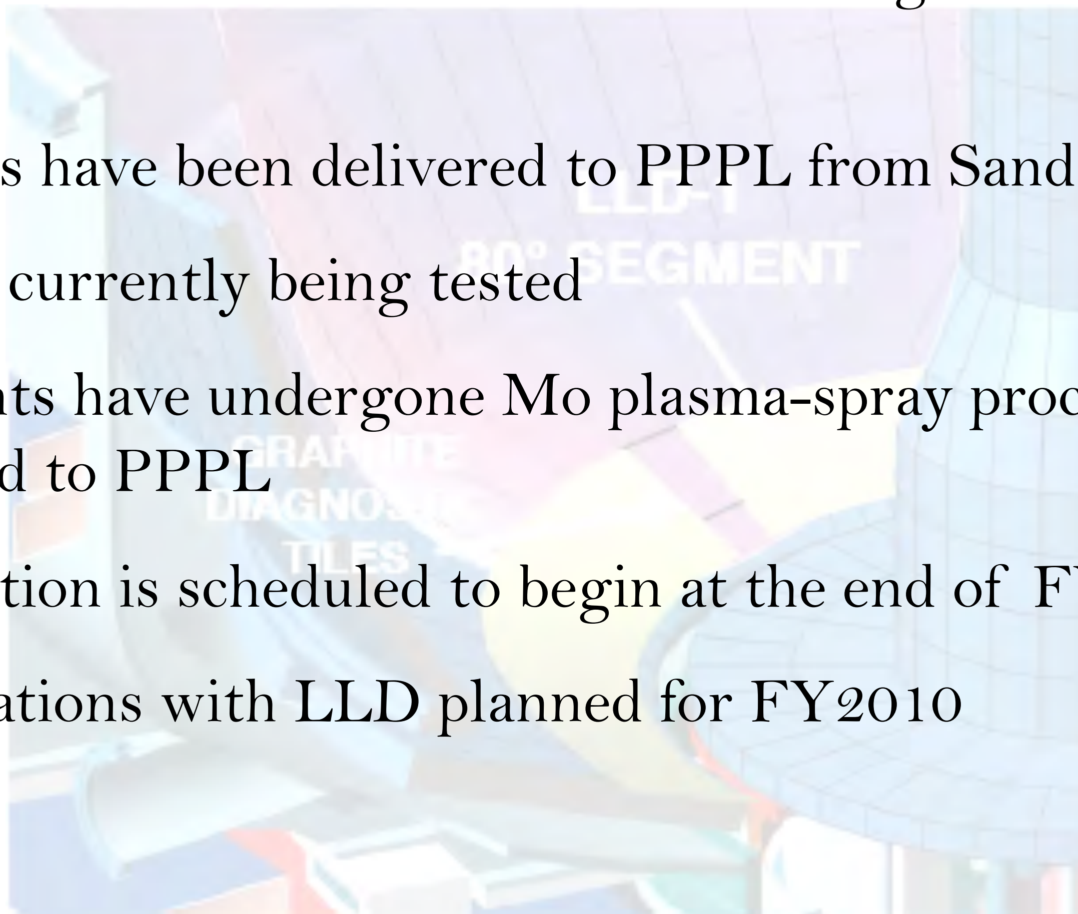
* Thin Li film on porous, high-Z (Mo) substrate
* Maintain film liquidity by heated Cu substrate
* Narrow graphite tile transition regions between LLD sections contain thermocouples, Langmuir probes, and magnetic and current sensors

- Location - lower outer divertor in four 90°sections.
- Width - 20 cm starting 5 cm outboard of CHI gap.
- Shape - replaces present graphite tiles.
- Structure - 0.01cm Mo flame-sprayed on 0.02 cm SS brazed to 1.9 cm Cu. Resistive heaters and cooling lines maintain 200-400°C.
- Li Loading - 2 lithium evaporators.



NSTX and LLD Current Status

* Final design review for LLD and associated diagnostics was successful
* LLD controls have been delivered to PPPL from Sandia
* Controls are currently being tested
* LLD segments have undergone Mo plasma-spray process and are being shipped to PPPL
* LLD installation is scheduled to begin at the end of FY2009
* Plasma operations with LLD planned for FY2010



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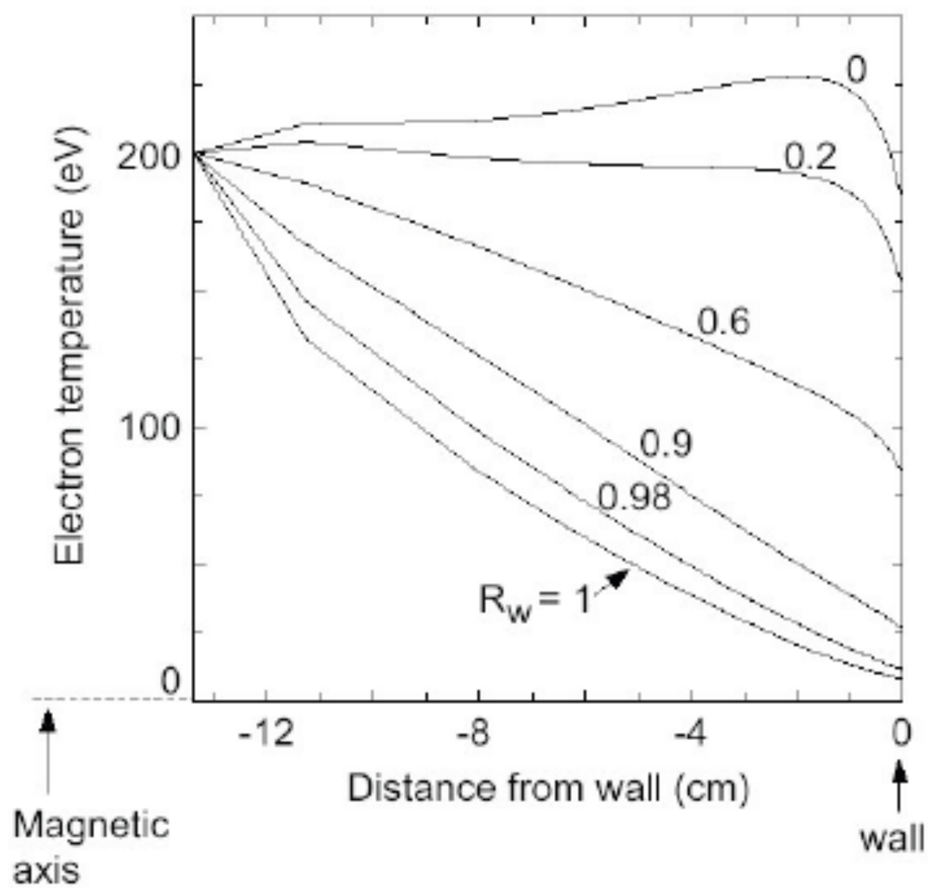
Supplemental Slides



Theoretical calculation of temperature profiles with various recycling coefficients