NSTX Research Highlights and Progress Toward NSTX Upgrade*

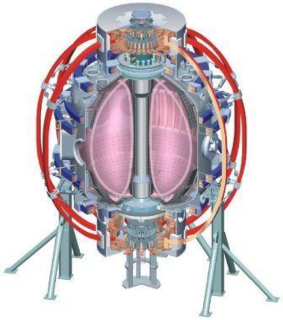
J. Menard,

*J. Canik, J. Chrzanowski, M. Denault, L.Dudek, T. Egebo, S. Gerhardt,
T. Gray, W. Guttenfelder, J. Hosea, S. Kaye, C. Kessel, E. Kolemen, R. Maingi,
C. Neumeyer, M. Ono, E. Perry, R. Ramakrishnan, R. Raman, Y. Ren, S. Sabbagh,
M. Smith, R. Strykowsky, V. Soukhanovskii, T. Stevenson, G. Taylor, P. Titus,
K. Tresemer, M. Viola, M. Williams, R. Woolley, A. Zolfaghari, and the NSTX Team*

24th Symposium on Fusion Engineering (SOFE)
Chicago, IL
June 26-30, 2011

Columbia U
CompX
General Atomics
FIU
INL
Johns Hopkins U
LANL
LLNL
Lodestar
MIT
Nova Photonics
New York U
ORNL
PPPL
Princeton U
Purdue U
SNL
Think Tank, Inc.
UC Davis
UC Irvine
UCLA
UCSD
U Colorado
U Illinois
U Maryland
U Rochester
U Washington
U Wisconsin

Culham Sci Ctr
U St. Andrews
York U
Chubu U
Fukui U
Hiroshima U
Hyogo U
Kyoto U
Kyushu U
Kyushu Tokai U
NIFS
Niigata U
U Tokyo
JAEA
Hebrew U
Ioffe Inst
RRC Kurchatov Inst
TRINITI
NFRI
KAIST
POSTECH
ASIPP
ENEA, Frascati
CEA, Cadarache
IPP, Jülich
IPP, Garching
ASCR, Czech Rep



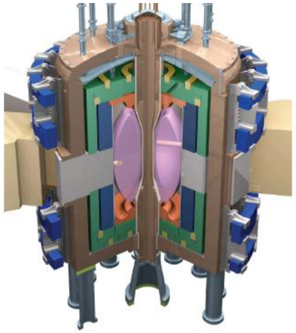
**This work supported by the US DOE Contract No. DE-AC02-09CH11466*

Outline

- NSTX Mission
- Motivation for NSTX Upgrade
- NSTX Research Highlights and Upgrade Design Progress
- Summary

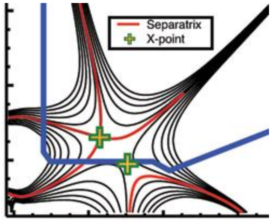
NSTX Mission Elements

- Advance ST as candidate for Fusion Nuclear Science Facility (FNSF)



ST-FNSF

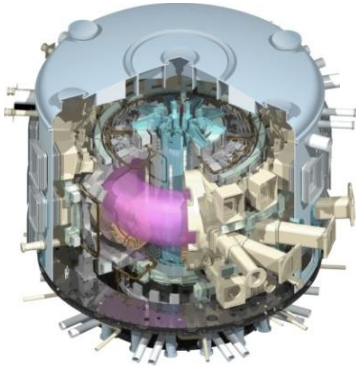
- Develop solutions for plasma-material interface



"Snowflake"

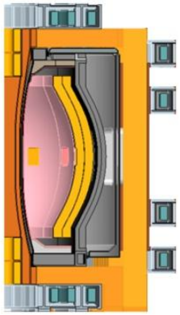
Lithium

- Advance toroidal confinement physics for ITER and beyond



ITER

- Develop ST as fusion energy system



ST Pilot Plant

High-Priority Research Areas for ST-FNSF

ReNeW Thrust 16 (2009): "Develop the ST to advance fusion nuclear science"

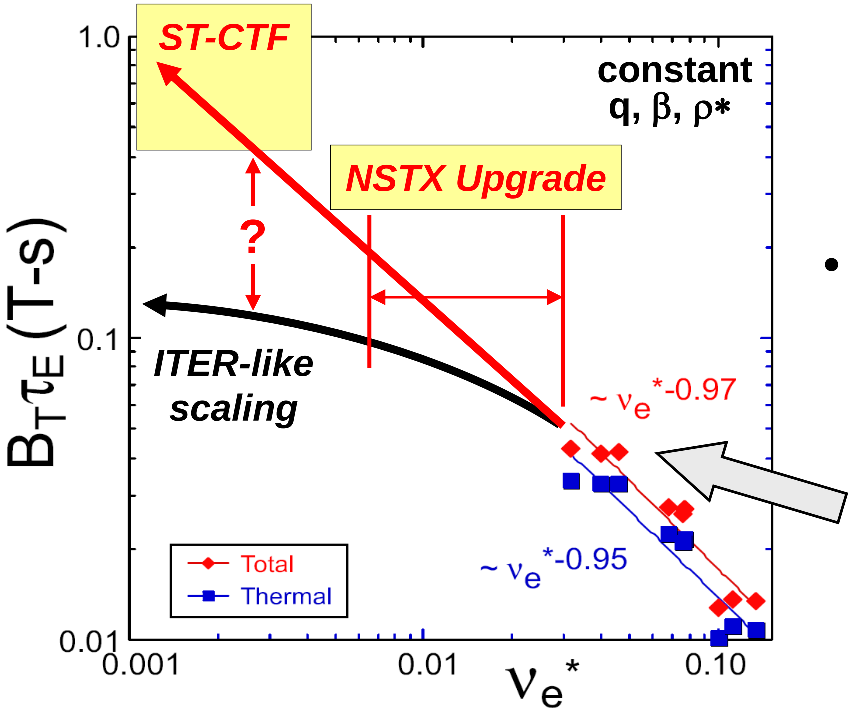
1. Develop **MA-level plasma current formation and ramp-up**
2. Advance **innovative magnetic geometries, first wall solutions**
3. Understand **ST confinement and stability** at fusion-relevant parameters
4. Develop **stability control techniques** for long-pulse, disruption-free ops
5. **Sustain current, control profiles** with beams, waves, pumping, fueling

6. Develop normally-conducting radiation-tolerant **magnets** for ST applications

7. **Extend ST performance** to near-burning-plasma conditions

This talk will focus on how NSTX and NSTX Upgrade address the ST-FNSF physics research needs (1-5) above

Access to reduced collisionality is needed to understand underlying causes of ST transport, scaling to next-steps



- Future ST’s are projected to operate at 10-100x lower normalized collisionality ν^*

Electron collisionality $\nu_e^* \propto n_e / T_e^2$

- Conventional tokamaks observe weak inverse dependence of confinement on ν^*

STs observe much stronger ν^* scaling

  - Does favorable scaling extend to lower ν^* ?
  - What modes dominate e-transport in ST ?

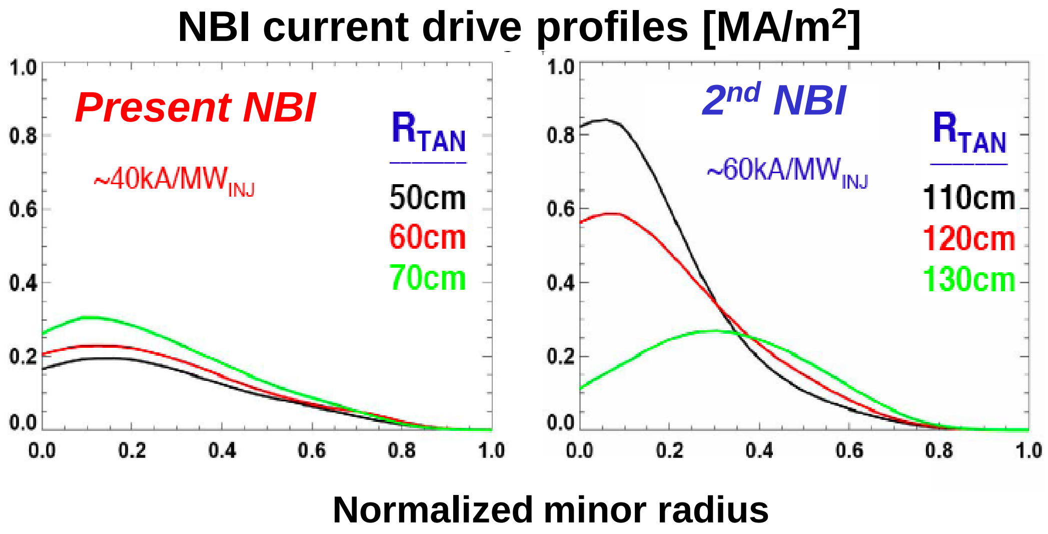
- **NSTX H-mode thermal confinement scaling differs from higher aspect ratio scaling:**

  $\tau_{E,NSTX} \propto B_T^{0.9}\ I_P^{0.4}$ → strong B_T scaling $\tau_{E,98y,2} \propto B_T^{0.15} I_P^{0.93}$ → weak B_T scaling

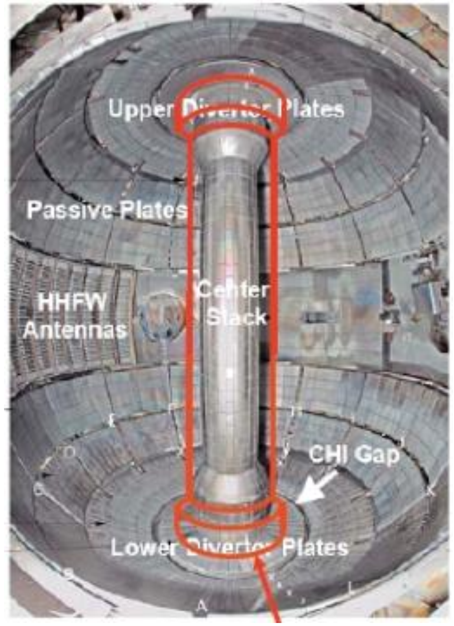
- **<u>Upgrade:</u> Double field and current for 3-6x decrease in collisionality → require 3-5x increase in pulse duration for profile equilibration**

Increased auxiliary heating and current drive are needed to fully exploit increased field, current, and pulse duration

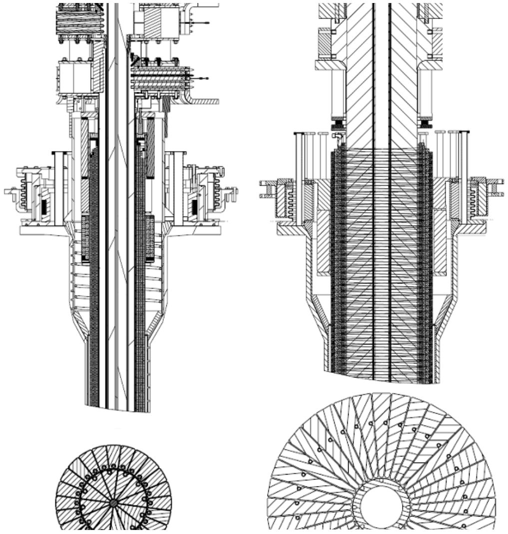
- Higher heating power to access high T and β at low collisionality
  - Need additional 4-10MW, depending on confinement scaling
- Increased external current drive to access and study 100% non-inductive
  - Need 0.25-0.5MA compatible with conditions of ramp-up and sustained plasmas
- **Upgrade: double neutral beam power + more tangential injection**
  - More tangential injection → up to 2 times higher efficiency, current profile control
  - ITER-level high-heat-flux plasma boundary physics capabilities & challenges



NSTX Upgrade consists of two major elements that together bridge the device and performance gaps toward next-steps

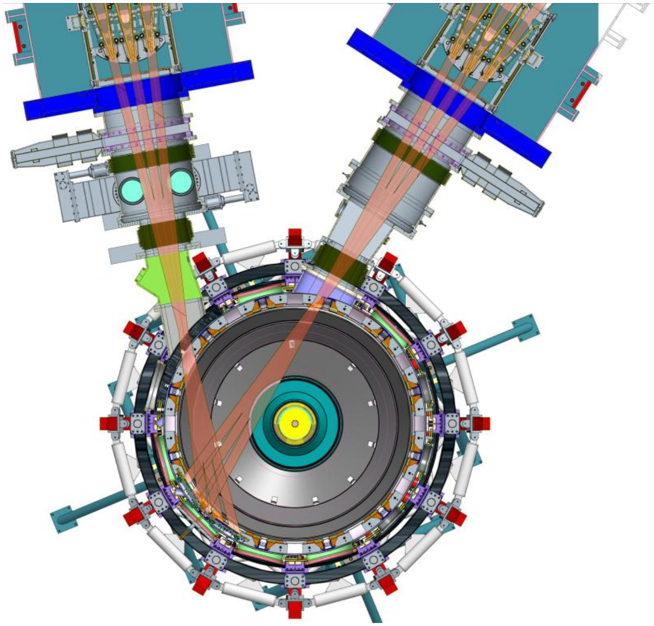


Outline of new center-stack (CS)



Present CS *New CS*

TF OD = 20cm *TF OD = 40cm*



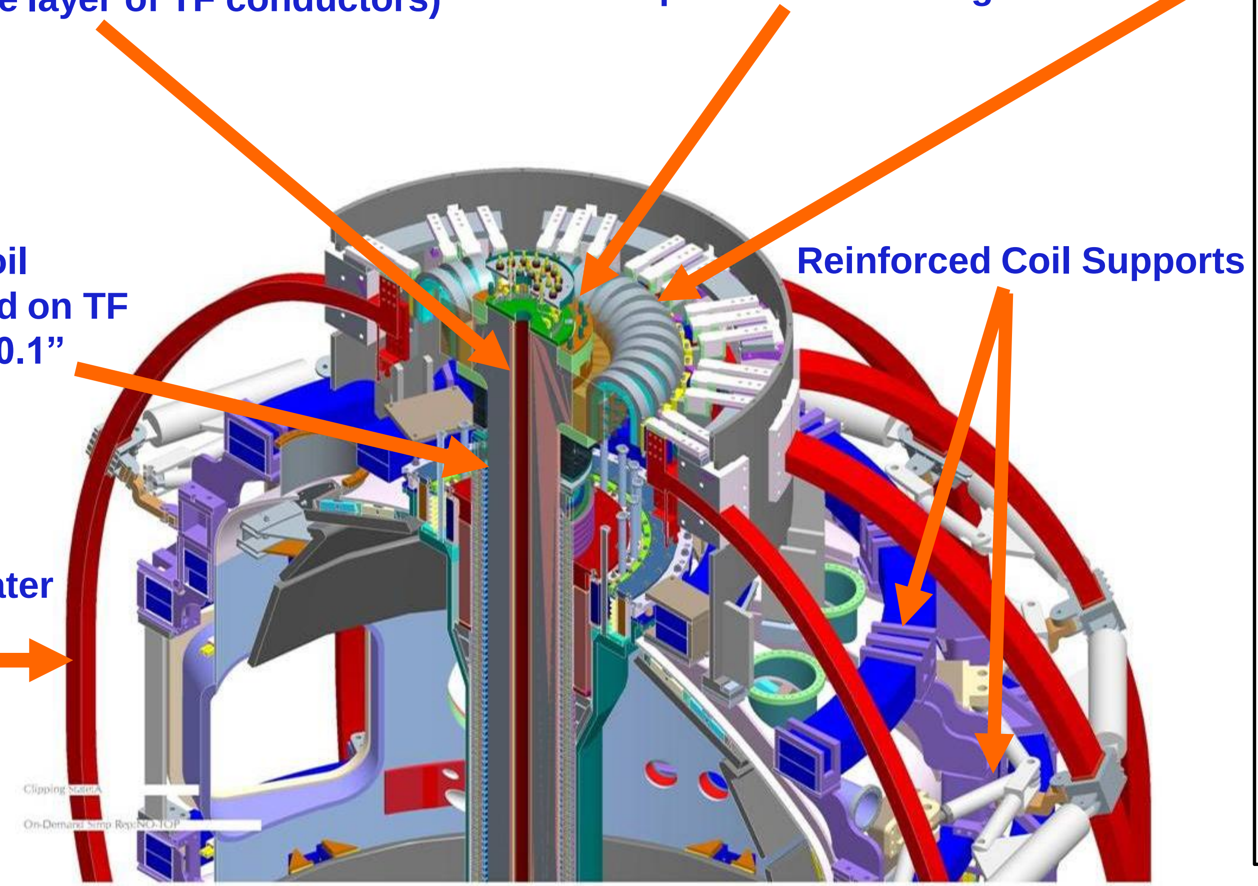
New 2nd NBI *Present NBI*

	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*

Center Stack Upgrade analysis and design are largely complete, and R&D activities are underway

B and J each increase 2x → electromagnetic forces increase 4x



Simpler Inner TF design (single layer of TF conductors)

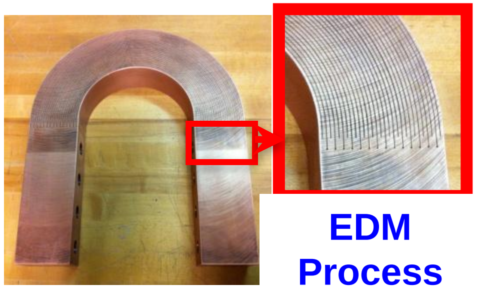
Improved Joint Design

OH coil wound on TF (with 0.1" gap)

Existing outer TF WITH water cooling

Reinforced Coil Supports

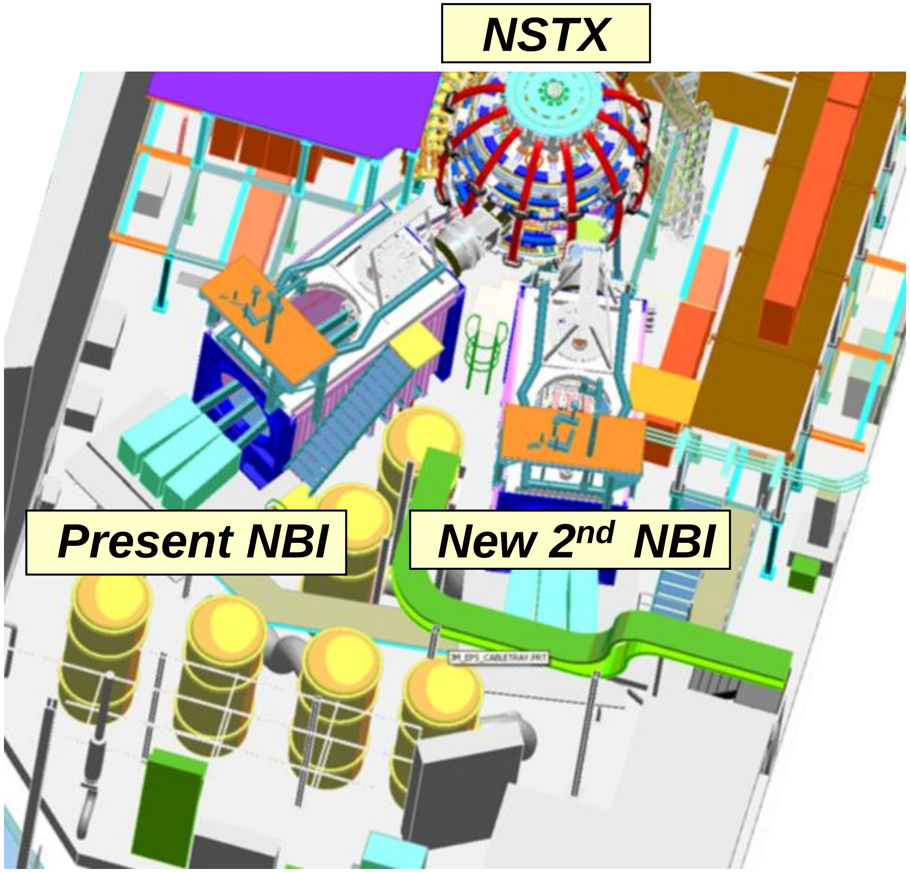
TF Flex Strap R&D

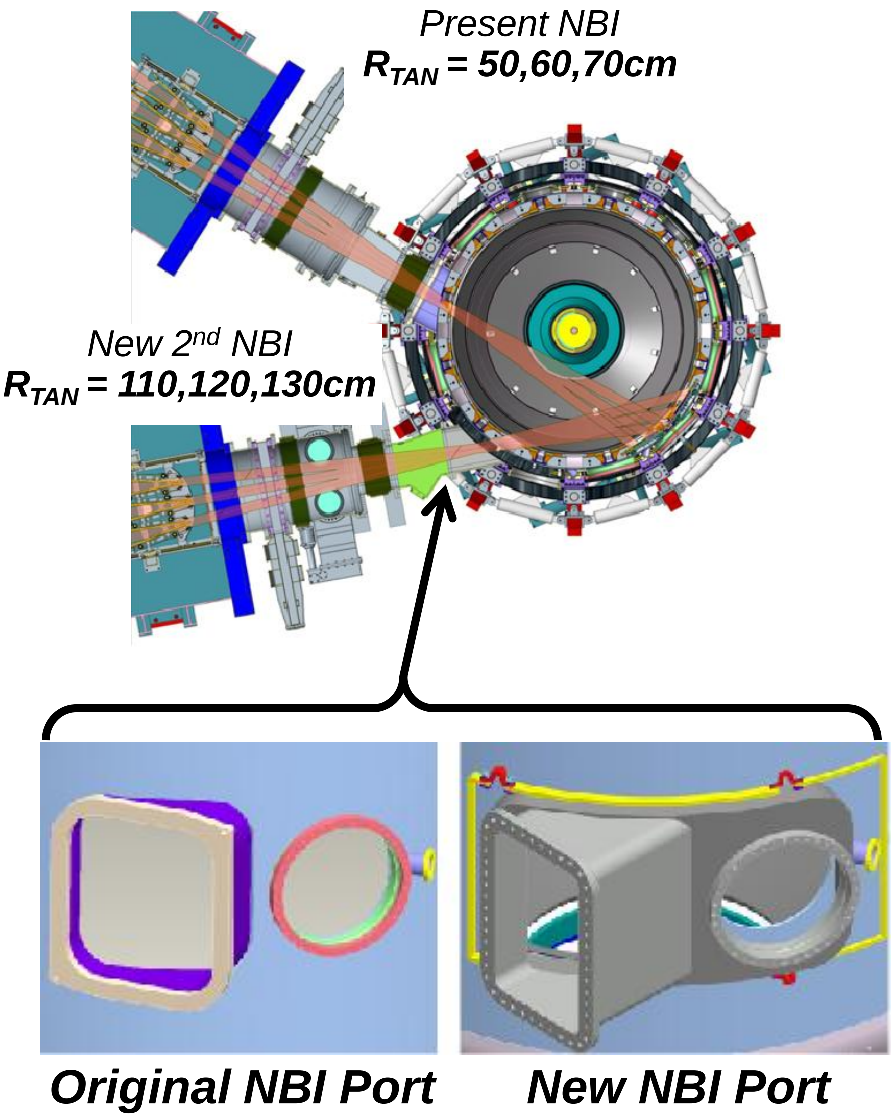


EDM Process

Successfully Tested to 300,000 cycles

2nd NBI requires relocation of a TFTR NBI system to NSTX, diagnostic relocations, new port for more tangential NBI

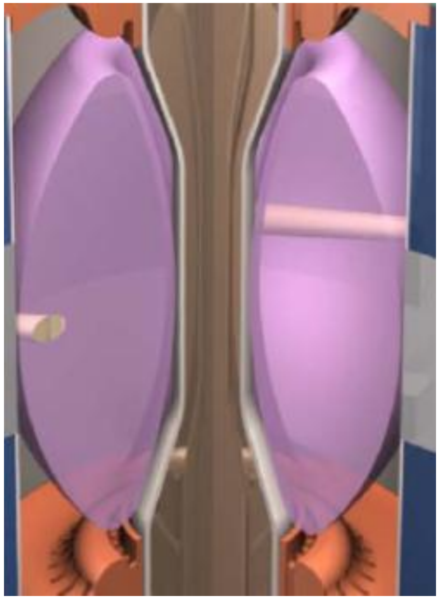


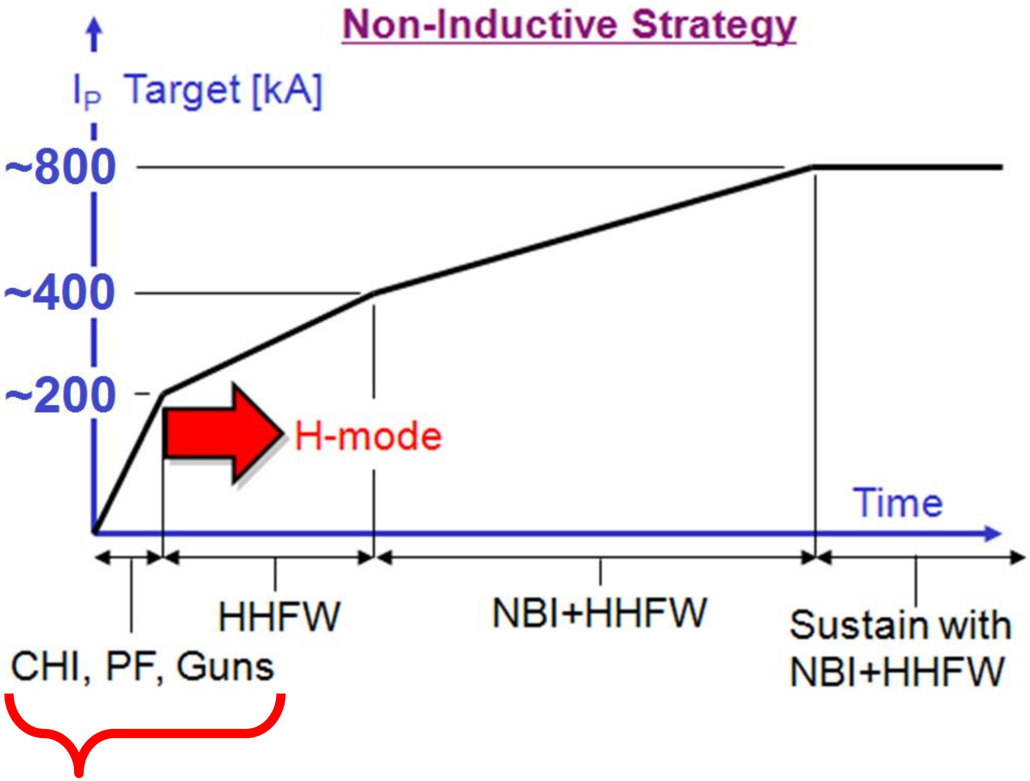


- Decontamination of 2nd Beam line successfully completed in 2010
- Reassembly of 2nd Beam line has started

Plasma initiation with small or no transformer is unique challenge for ST-based Fusion Nuclear Science Facility

ST-FNSF has no/small central solenoid

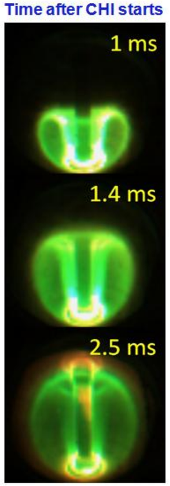


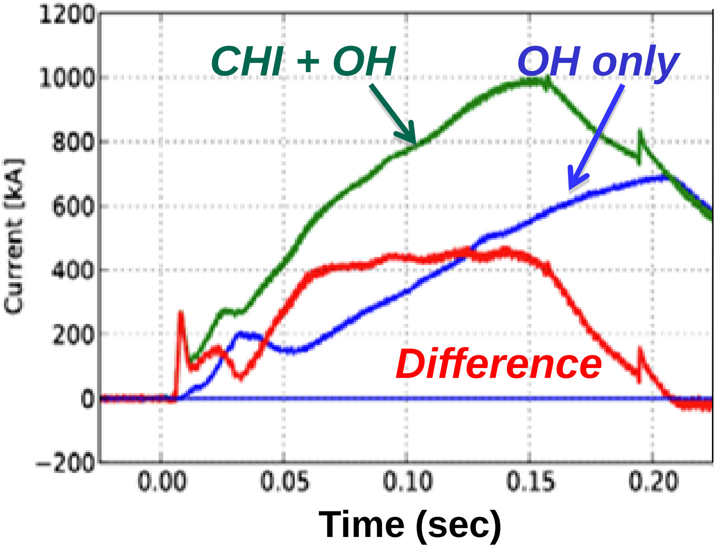


- Near-term NSTX Goal: Generate ~0.3-0.4MA full non-inductive start-up with Coaxial Helicity Injection + fast wave heating + NBI (need Upgrade)
- Upgrade goal: Provide physics basis for non-inductive ramp-up to high performance 100% non-inductive ST plasma → prototype FNSF

Achieved substantial progress on Coaxial Helicity Injection (CHI) and fast wave heating of low-current plasmas in 2010

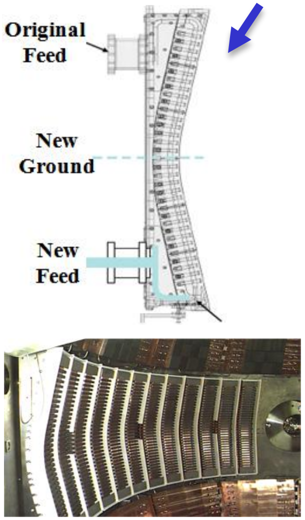
- **Early in shot: produce 150-200kA**
- **Generated 1MA using 40% less flux than induction-only case**
  - **Low $l_i \approx 0.35$, and high elongation > 2 → suitable for advanced scenarios**

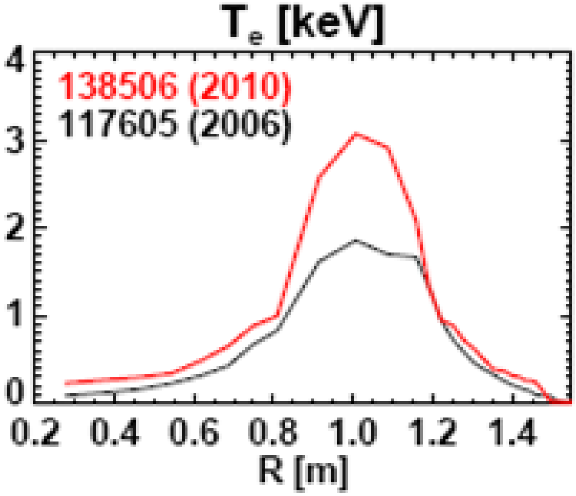




- CHI-driven current scales linearly with B_T → **2x higher in Upgrade**

- **Achieved high $T_e(0)$ ~ 3keV at I_P=300kA w/ only 1.4MW of HHFW**
  - **- Previous best was $T_e(0)$ ~ 1.5keV at twice the RF power**
  - **- Enabled by 2009 antenna upgrades**





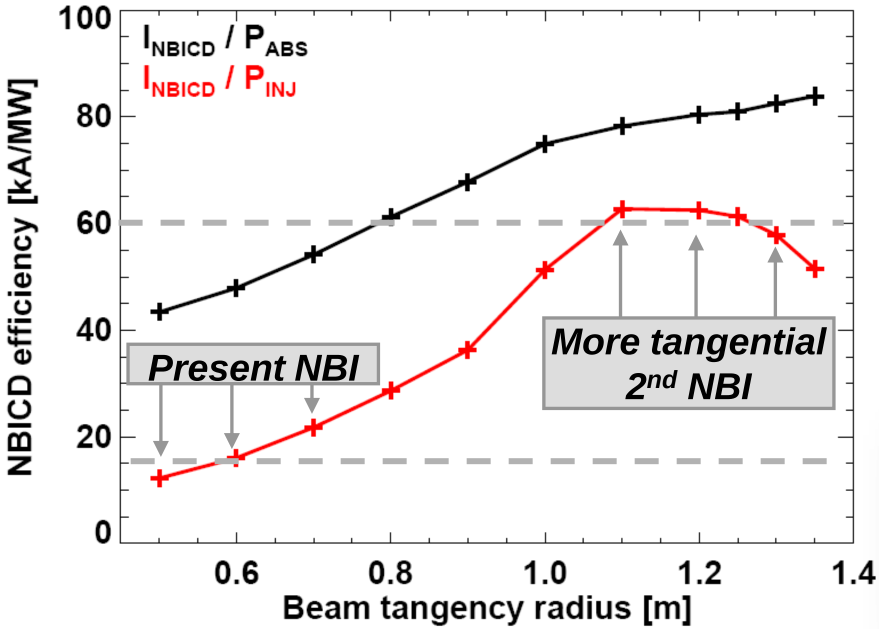
- Non-inductive fraction ~60-70% with 25-30% from RFCD from high $T_e(0)$
- **Projects to ~100% NI at P_{RF} = 3-4MW**

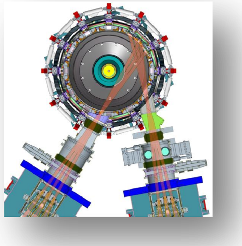
Non-inductive ramp-up from ~0.4MA to ~1MA projected to be possible with new CS + more tangential 2nd NBI

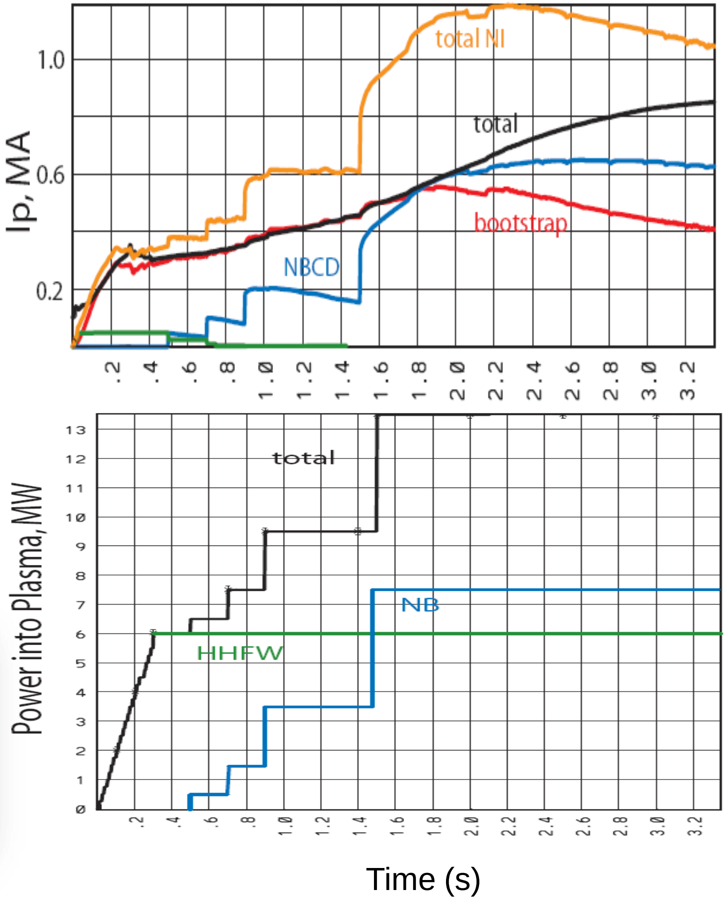
- **New CS provides higher TF (improves stability), 3-5s needed for J(r) equilibration**
- **More tangential injection provides 3-4x higher CD at low I_P:**
  - 2x higher absorption (40→80%) at low I_P = 0.4MA
  - 1.5-2x higher current drive efficiency

E_{NBI}=100keV, I_P=0.40MA, f_{GW}=0.62

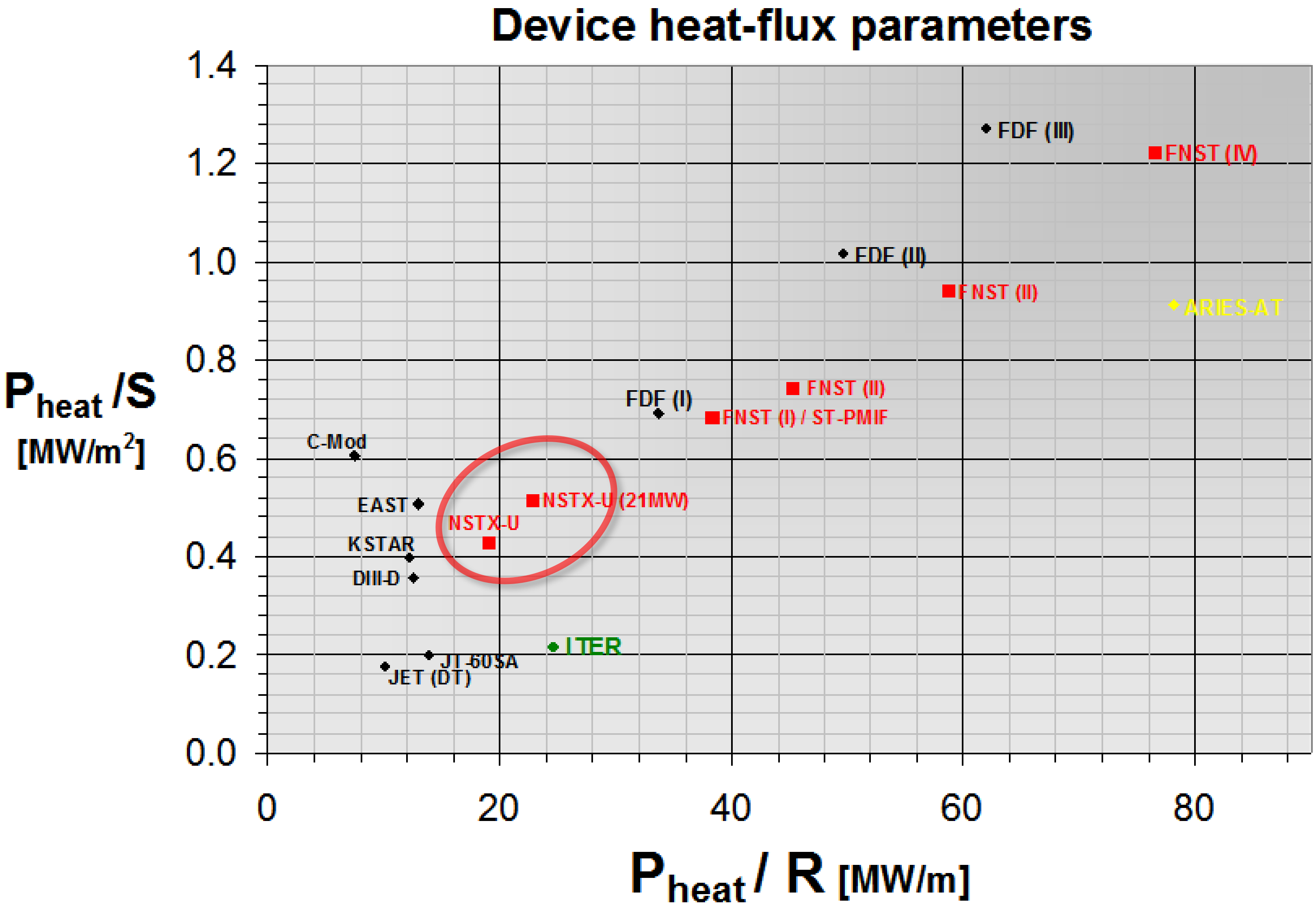
$\overline{n_e}$ = 2.5x$10^{19}$$m^{-3}$, $\overline{T_e}$ = 0.83keV



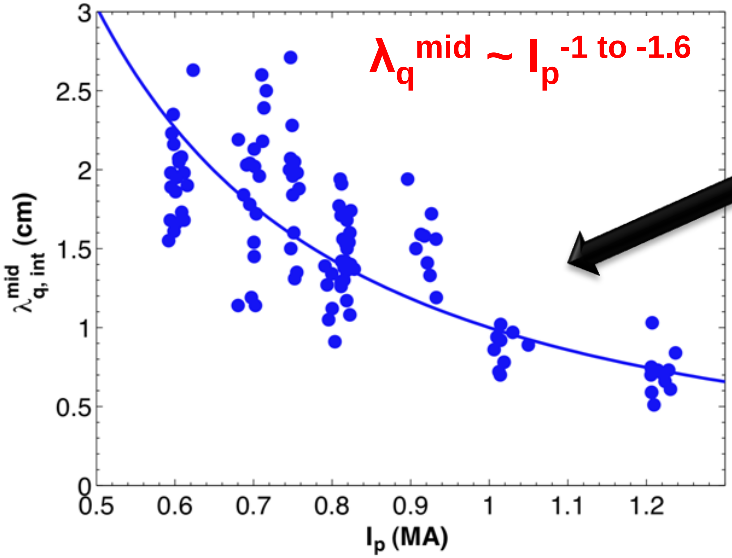




NSTX Upgrade will extend normalized divertor and first-wall heat-loads much closer to FNS and Demo regimes



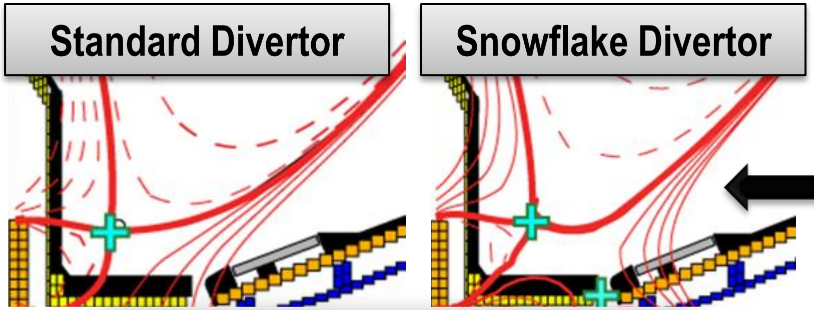
NSTX has contributed strongly to divertor heat flux width studies*, and is developing new heat-flux mitigation methods



**Joint Research Milestone (3 U.S. Facilities)*

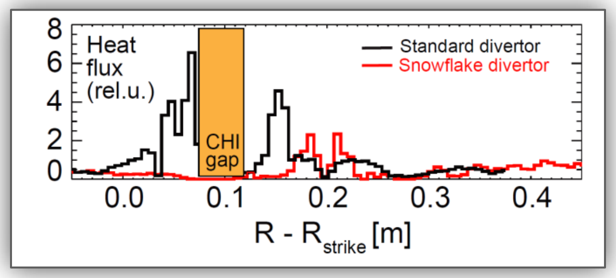
- Divertor heat flux width decreases with increased plasma current I_P
  - – Potentially major implications for ITER

→ NSTX Upgrade with conventional divertor projects to very high peak heat flux up to 30-45MW/m²



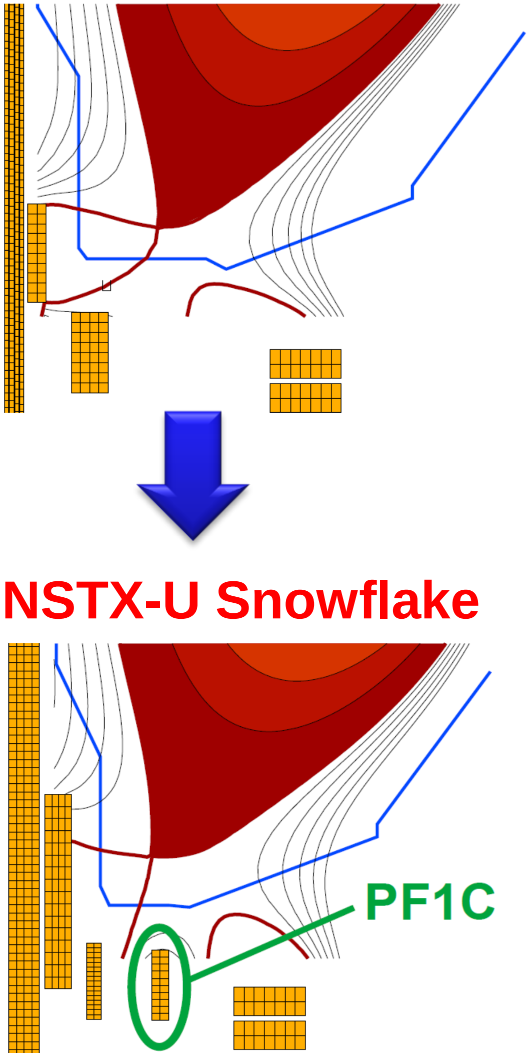
- Divertor heat flux inversely proportional to flux expansion over a factor of five

Snowflake→ high flux expansion 40-60, larger divertor volume and radiation



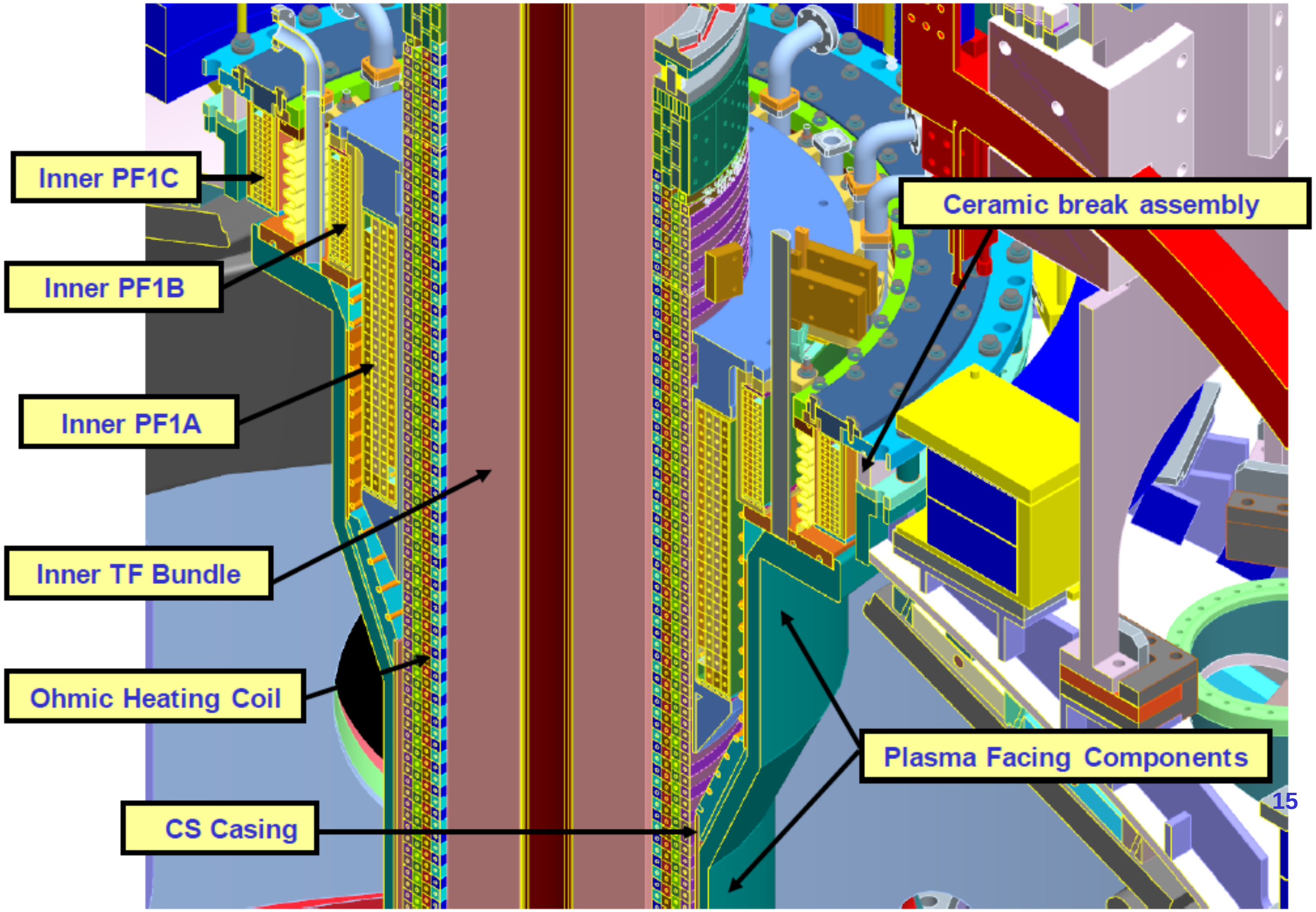
→ U/D balanced snowflake divertor projects to acceptable heat flux < 10MW/m² in Upgrade at highest expected I_P = 2MA, P_{AUX}=10-15MW

Upgrade CS design provides additional coils for flexible and controllable divertor including snowflake, and supports CHI



NSTX Snowflake

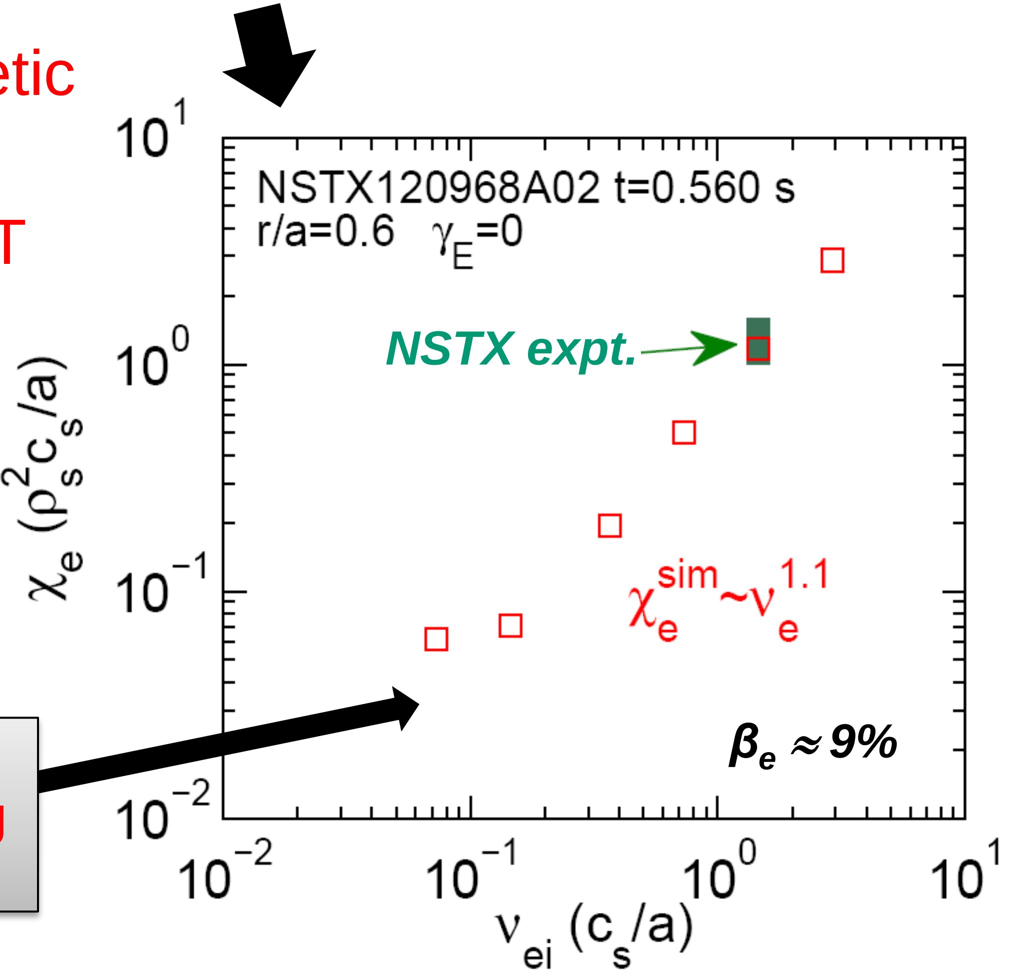
NSTX-U Snowflake



New NSTX turbulence simulations are advancing the understanding of ST energy confinement

- Non-linear gyrokinetic turbulence simulations of micro-tearing instabilities predict $\chi_e \propto \nu_e^* \Rightarrow \tau_E \propto 1/\nu_e^*$

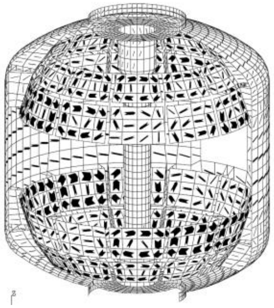
- Predominantly electromagnetic turbulence – result of high β
- Candidate explanation for ST confinement scaling



Lower ν* accessible in Upgrade can clarify roles of micro-tearing vs. ETG, TEM in ST e-transport

NSTX is 1st tokamak to implement advanced resistive wall mode state-space controller, utilized it to sustain high $\beta_N \sim 6$

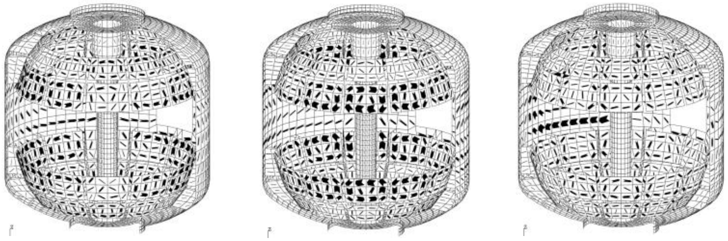
Full 3-D model



~3000+ states

State reduction (< 20 states)

RWM eigenfunction (2 phases, 2 states)



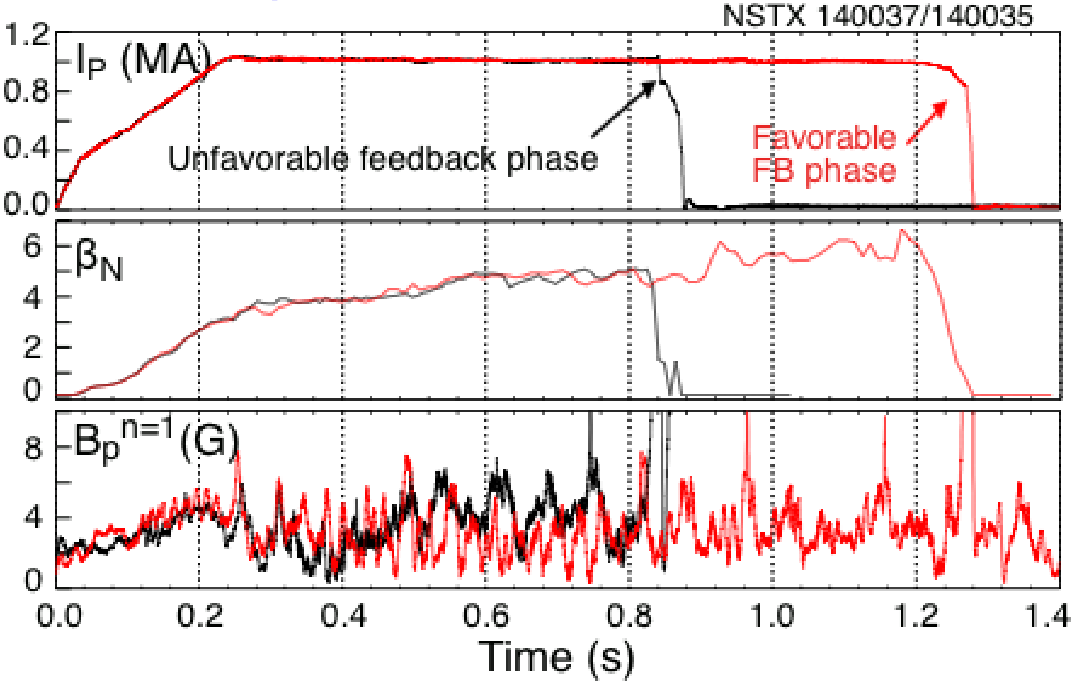
$(\hat{x}_1, \hat{x}_2)$ $\hat{x}_3$ $\hat{x}_4$ $\hat{x}_N$

truncate

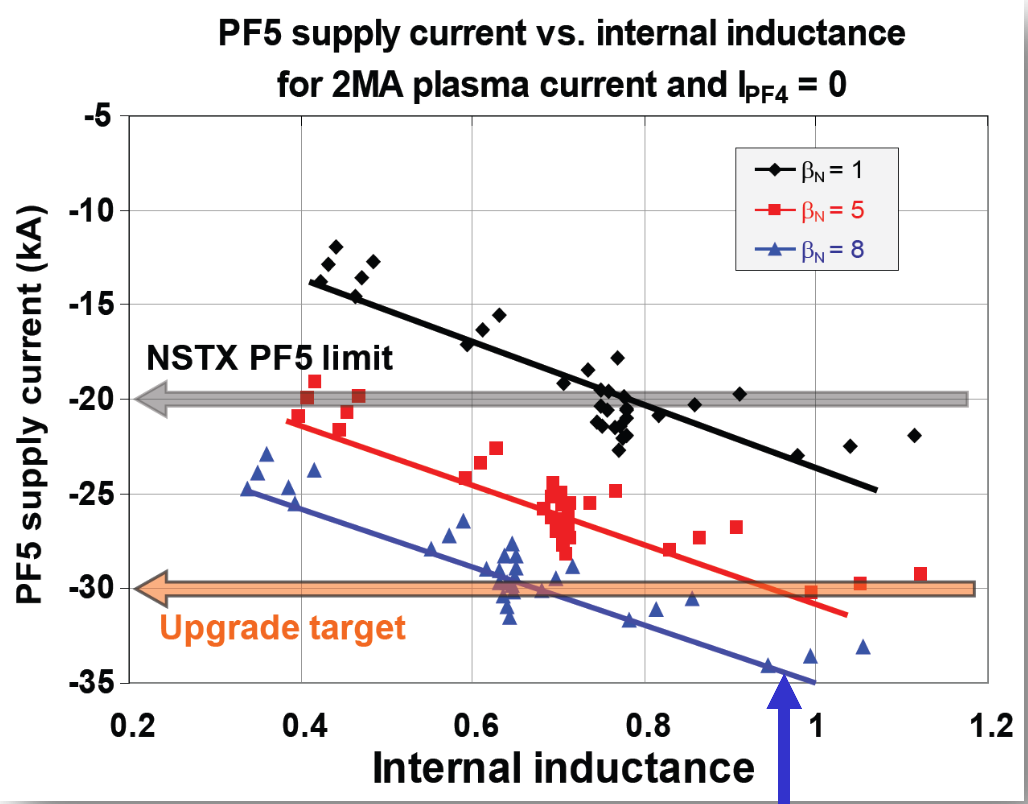
- ***Device R, L, mutual inductances***
- ***Instability B field / plasma response***
- ***Modeled sensor response***

- **Controller can compensate for wall currents**
  - **Including mode-induced current**
  - **Examined for ITER**
- **Successful initial experiments**
  - **Suppressed disruption due to n = 1 applied error field**
  - **Best feedback phase produced long pulse, $\beta_N = 6.4$, $\beta_N / l_i = 13$**

State space feedback with 12

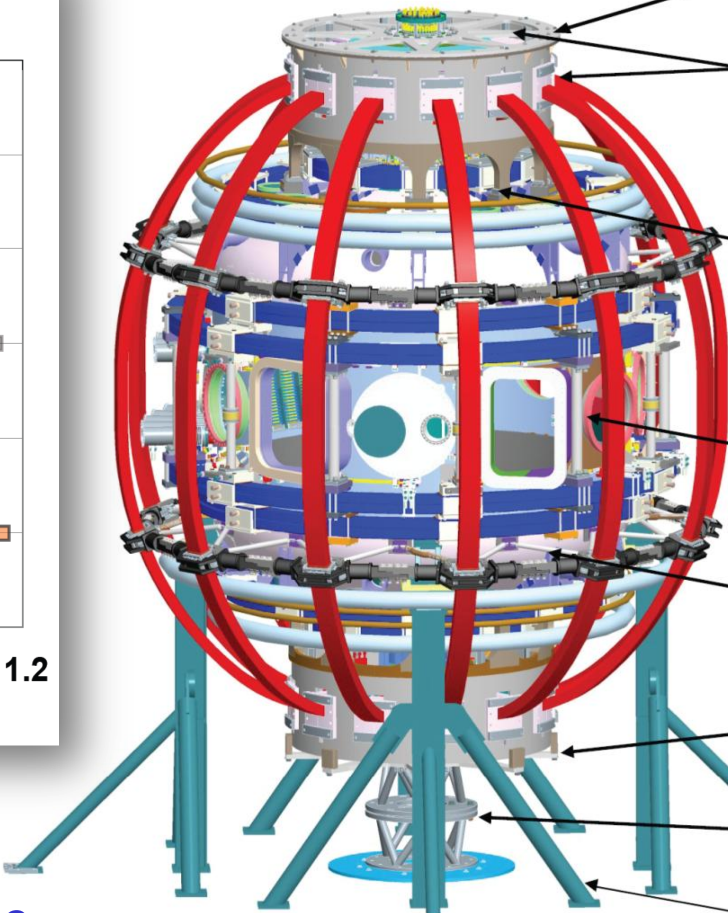


Upgrade structural enhancements designed to support high β at full I_P = 2MA, B_T=1T: $\beta_N = 5, l_i \leq 1$ and $\beta_N = 8, l_i \leq 0.6$



High l_i, high-β_N scenarios determine the maximum vertical field (PF5) current required

NSTX Upgrade



New Umbrella Upper Lid

Upper & Lower Umbrella Reinforcements

Additional PF2 Clamp, PF2/3 Support Upgrades

New TF Leg Support Ring

New (additional) PF4/5 Supports

New Clevis & Connecting Rods

New Umbrella Lower Lid

New Pedestal

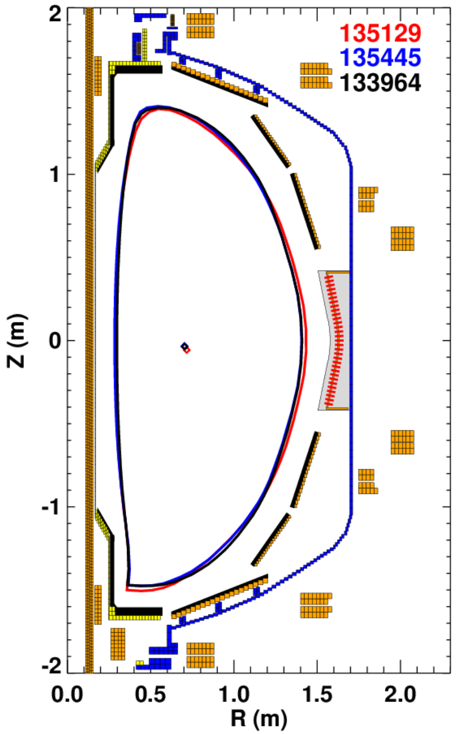
VV Foot Reinforcement

In 2009-10, NSTX demonstrated sustained high-elongation configurations over a range of currents and fields

High-β_T
$q^*=2.8$
$B_T=0.44$ T
$I_P=1100$ kA

Long Pulse
$q^*=3.9$
$B_T=0.38$ T
$I_P=700$ kA

High-β_P
$q^*=4.7$
$B_T=0.48$ T
$I_P=700$ kA



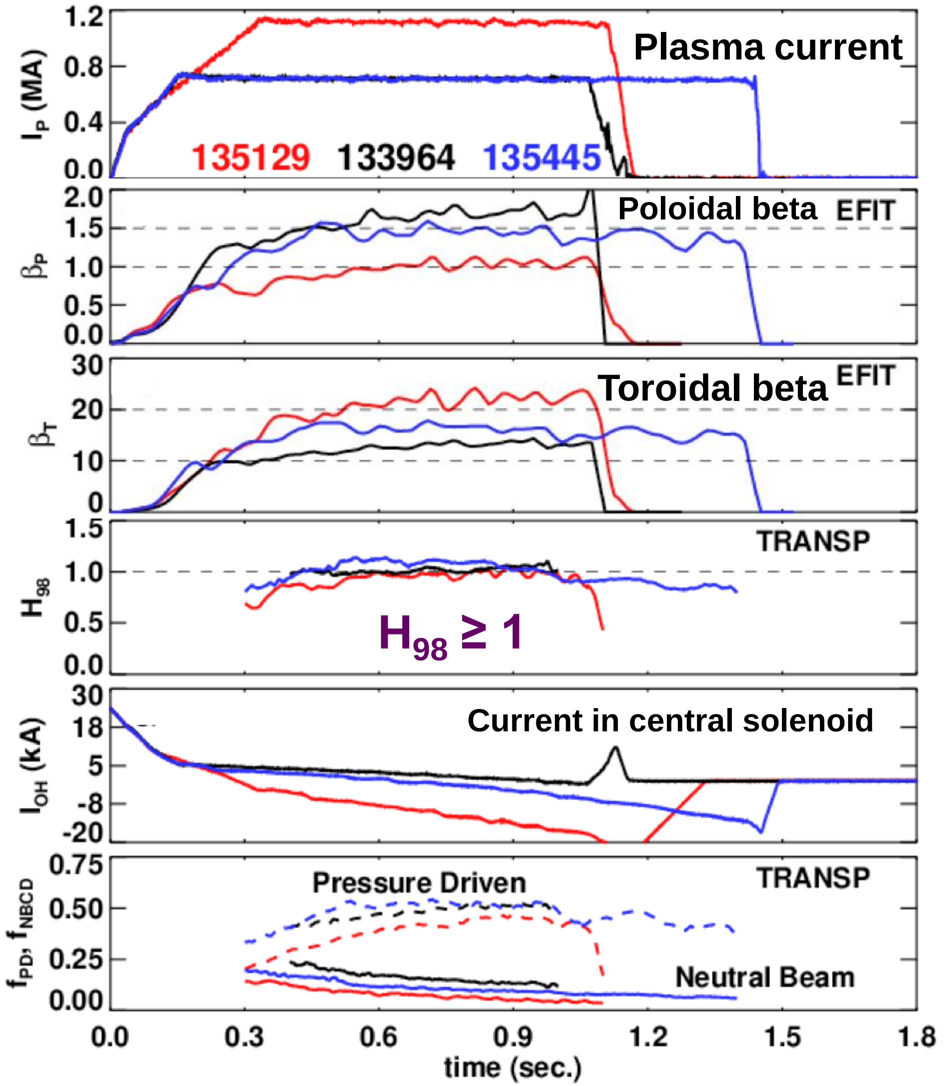
$$q^* = \frac{\varepsilon\left(1+\kappa^2\right)\pi a B_{T0}}{\mu_0 I_P}$$

$\kappa \sim 2.6\text{-}2.7$

$\delta \sim 0.8$

Double Null

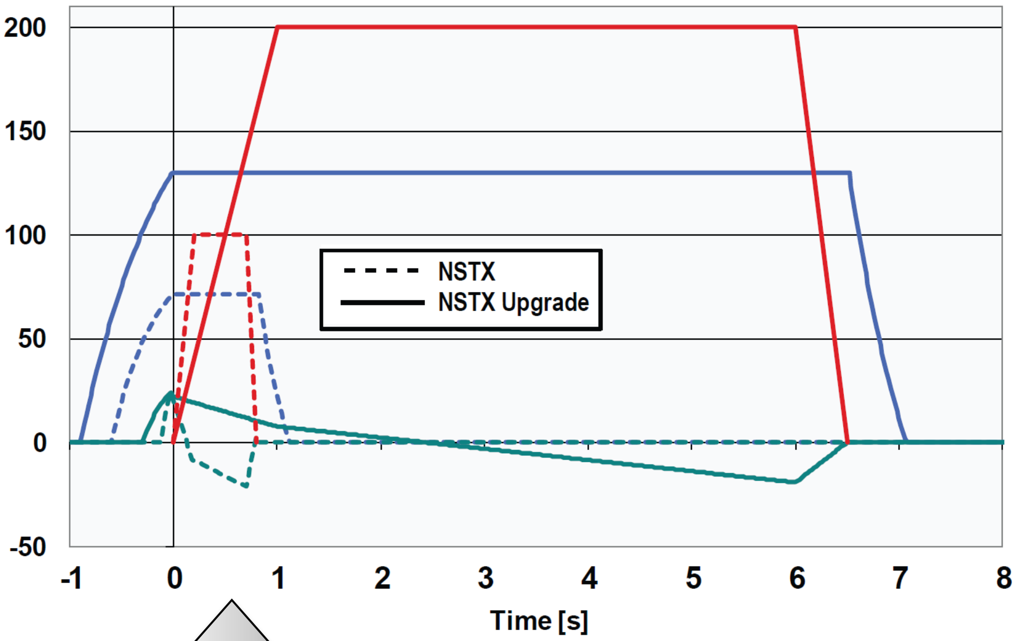
Pulse-lengths limited by OH, TF coil heating limits



NSTX Upgrade supports 5x longer pulses and 100% non-inductive current drive, ultimately with *q* profile control

TF, OH, and Plasma Current

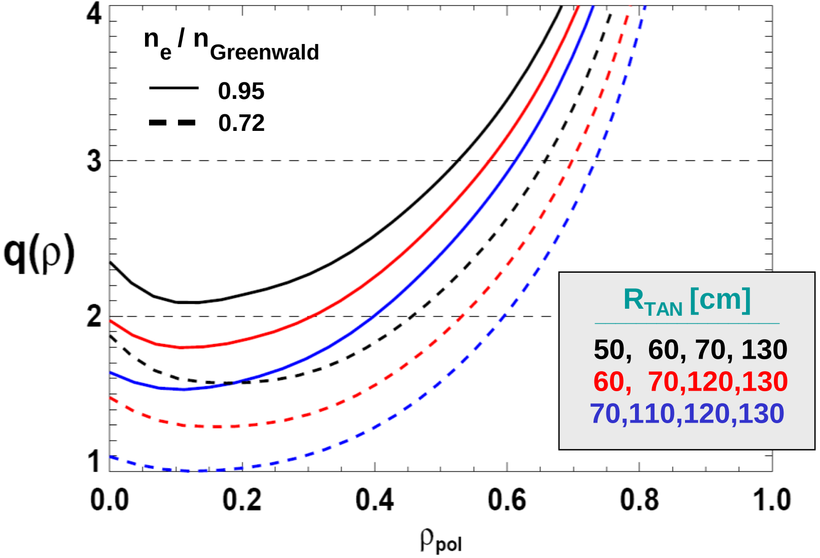
Units: I_{OH} and I_{TF} [kA], I_P [kA/10]



Fully non-inductive scenarios with q(r) profile will be controllable by:

- NBI source tangency radius
- plasma density
- plasma position (not shown)

B_T = 1T, P_{NBI} = 10MW, E_{NBI} = 110keV



I_P = 0.95MA, H_{98y2} = 1.2, β_N = 5, β_T = 10%, 4MW RF

Changes from NSTX to NSTX Upgrade:

- I_P and B_T 2x higher, 3x OH flux, flat-top 5x longer, W_{TOT} up to 4x higher
- Minimum Aspect Ratio A = 1.3 → 1.5, inter-shot time increased ~2x

Summary: NSTX and NSTX Upgrade strongly support FNSF development, Materials/PMI, and ITER

- **NSTX Research Highlights:**
  - CHI+OH plasma current savings up to 400kA, RF heating of low I_P to 3keV
  - Established divertor heat flux scalings, snowflake divertor for mitigation
  - Non-linear simulations suggest micro-tearing may influence ST transport
  - High β_N ~ 6 sustained with advanced RWM control
  - Long-pulse plasmas developed – duration limited by magnet capabilities
- **NSTX Upgrade Progress:**
  - Design supports CHI/start-up, PMI, transport, high-β, 100% NICD research
  - New center-stack design and analysis complete – fabrication beginning
  - 2nd NBI relocation/installation – ready to begin during Upgrade outage
- **NSTX Upgrade Schedule:**
  - Project base-lined (CD-2) December 2010, Final Design Review last week
  - NSTX operation to be completed February 2012
  - NSTX Upgrade outage to begin April 2012
  - NSTX Upgrade first plasma → end of 2014

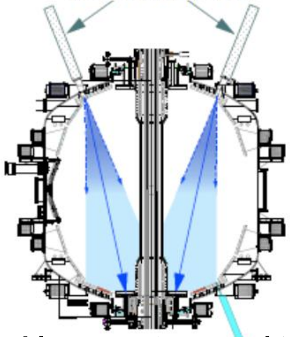
BACKUP

NSTX is a world leader in assessing lithium plasma facing components as a possible PMI solution for magnetic fusion

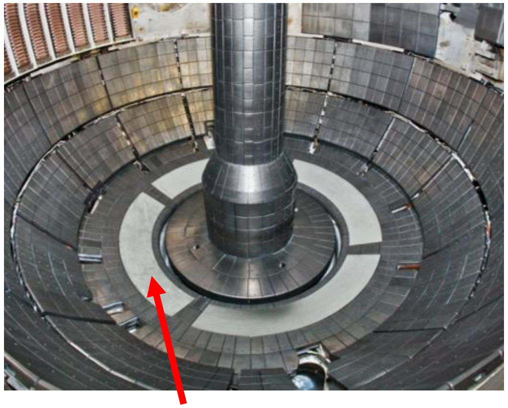
- **<u>Solid Li surface coatings</u>:** Pump D, increase confinement, stored energy, pulse length, eliminate ELMs, reduce core MHD instabilities

- **<u>Liquid Lithium Divertor (LLD) motivation</u>:** provide volume D pumping capacity (> solid Li coatings) to provide longer pumping duration + potential for handling high heat flux

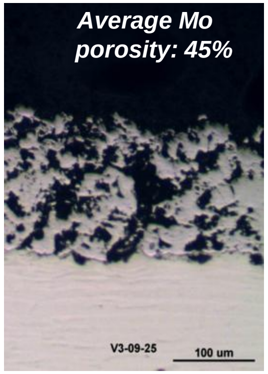
Dual Liquid Lithium Evaporator
For Li wall coatings
Now routinely used



Li evaporators used to coat lower divertor, LLD

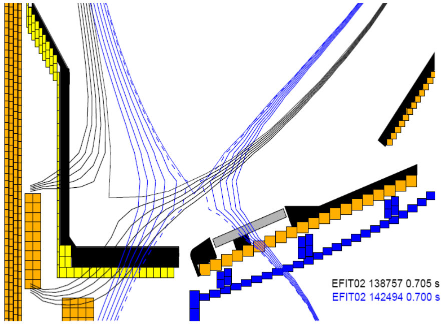


4 heatable LLD plates (Mo on Cu)
Surface temp: 160 – 350+ °C



Average Mo porosity: 45%

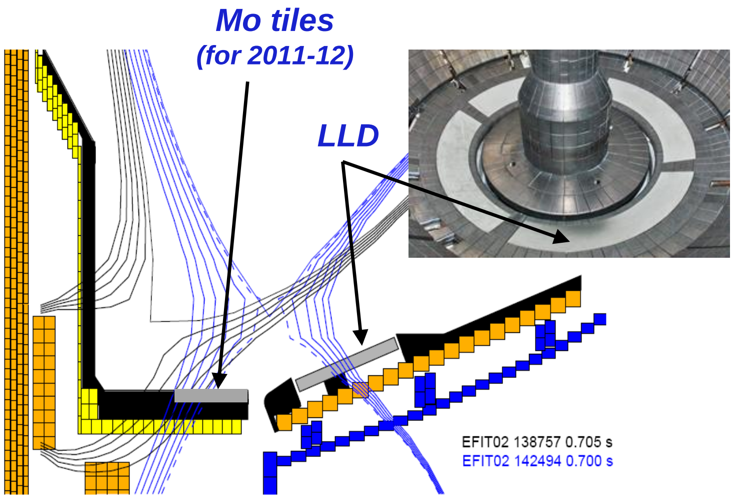
LLD surface cross section: plasma sprayed porous Mo



<u>Controlled scans of strike-point location:</u>
On inboard divertor
On LLD (outboard divertor)

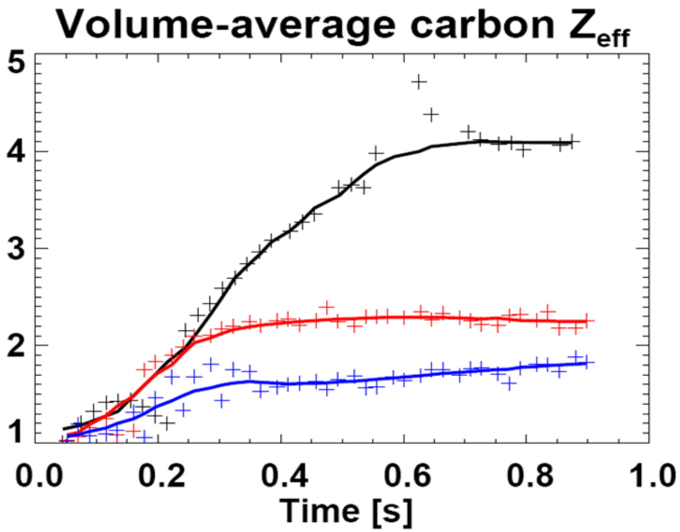
- FY2010: First LLD tests - filled with 67g Li by evaporation (2x needed to fill porosity)

Operation with outer strike-point on Mo LLD (coated with Li) technically successful, achieved high plasma performance



LLD FY2010 results:

- **LLD did not increase D pumping beyond that achieved with LiTER**
  - **Solid Li on C pumps D quite efficiently**
  - **Liquid Li reacts rapidly w/ background gases (LTX)**
  - **C on LLD may have impacted D pumping**
- **No evidence of Mo from LLD in plasma during normal operation**
- **Operation with strike-point (SP) on LLD <u>reduced</u> core impurities**



◄SP on inner carbon divertor (no ELMs)

◄SP on LLD, $T_{LLD} < T_{Li\text{-}melt}$

◄SP on LLD, $T_{LLD} > T_{Li\text{-}melt}$ (+ fueling differences)

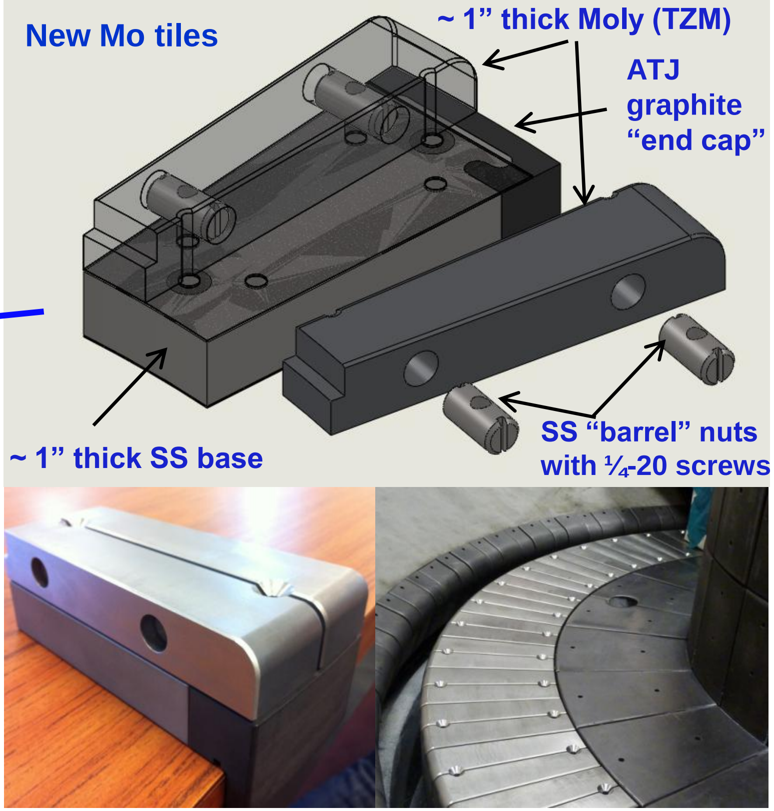
- **No ELMs, no → small, small → larger**

  → High-Z impurities also reduced, $\beta_N > 4$ sustained

Understanding roles of δ, C, Mo, Li, ELMs motivates Mo tiles on inboard divertor

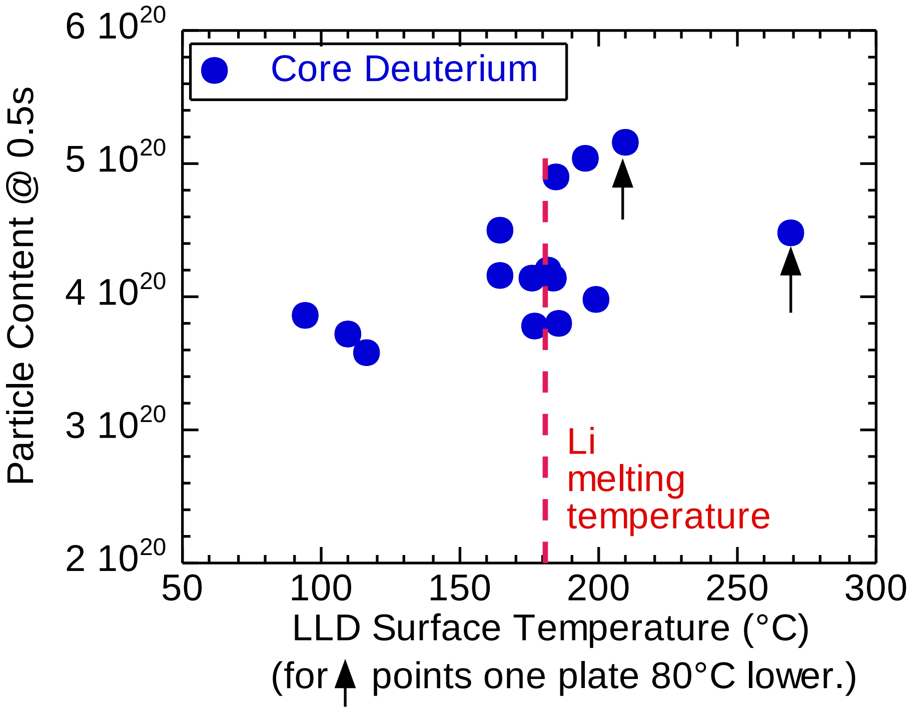
Mo tiles have recently been installed on inboard divertor to inform decisions on Mo as candidate PFC for Upgrade

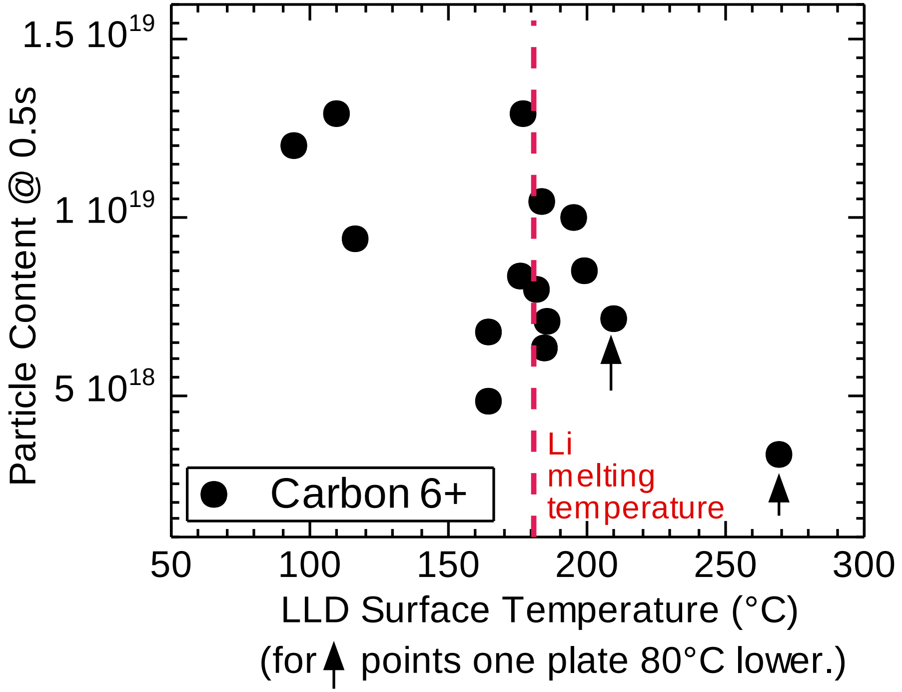
- **Do Mo tiles provide improved LLD?**
  - Li coating with LiTER ~ 2x outboard LLD rate
  - Plasma heating can melt Li during shot



- **2011-12: Assess Mo thermal, mechanical, impurity production performance**
  - **Does Li help protect Mo thermally? (2010: LLD exhibited temperature saturation)**
  - **Do Mo PFCs help reduce core C impurity accumulation in Li ELM-free H-mode?**

LLD pumping similar above or below Li melting temperature





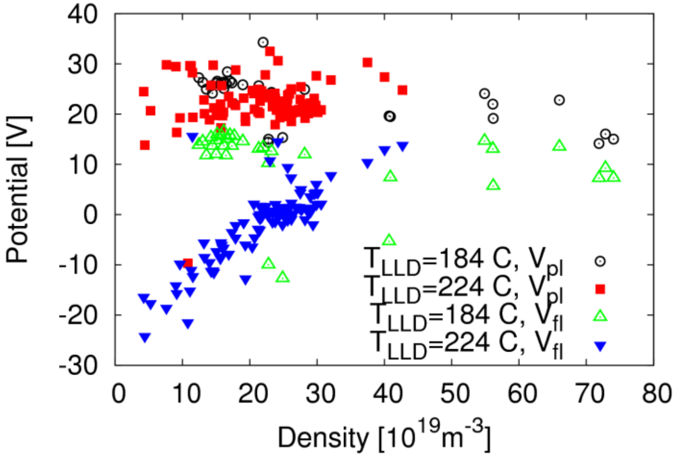
- Constant deuterium fueling for LLD 100% Li fill conditions, 4 plates air heated.
- As LLD surface temperature transitioned from solid temperatures to the liquid regime, the plasma electron and deuterium content remain relatively constant.
- Core carbon C6+ content decreased - may be due in part to increased ELMing and edge turbulence.

- No systematic trend in D-alpha, wall inventory, or ion pumping with a transition above the Li melting temperature.

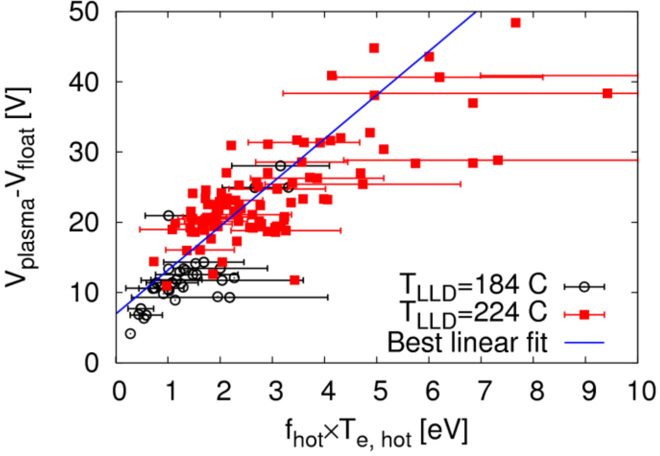
Higher local electron temperature during LLD melting sequence despite increased fueling

- Langmuir probes indicate increase in near-surface electron temperature during LLD experiments
  - Discharge sequence indicated decreased fueling efficiency during LLD heating (gas increased, N_e constant)
  - Non-local and classical probe interpretations applied, increase in T_e consistent with increase in V_p-V_f difference
  - Temperature rise occurs in hot-electron population of the distribution function
  - Observations consistent with plasma-absorbing PFC

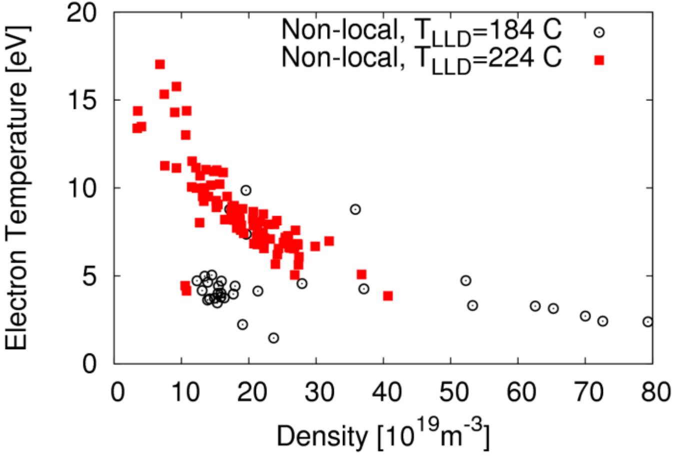
Floating and Plasma Potentials



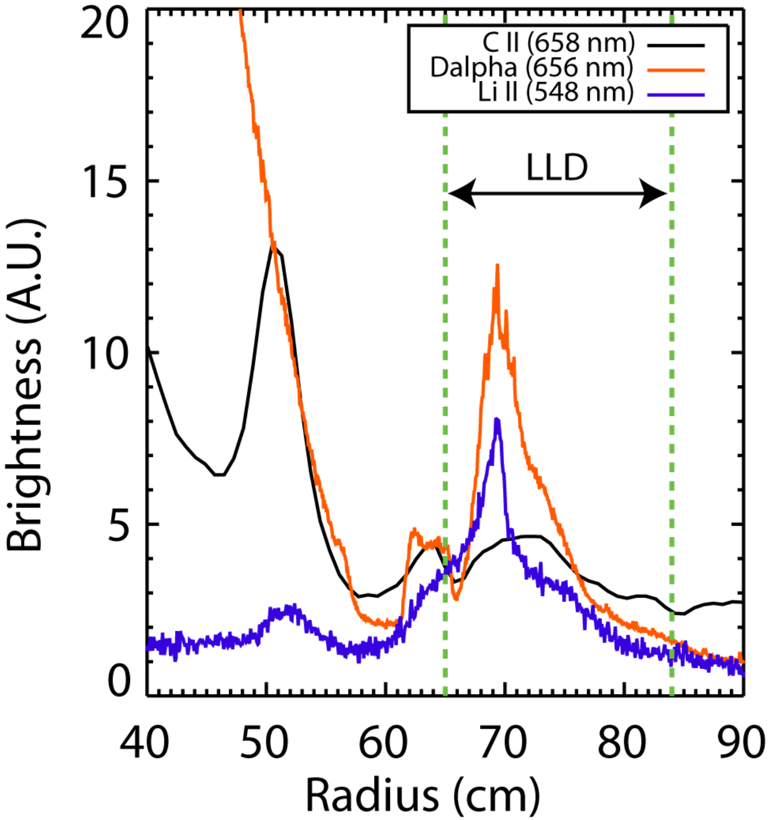
Difference in potentials most strongly correlated with hot-electron energy fraction



Inferred temperature in near-SOL

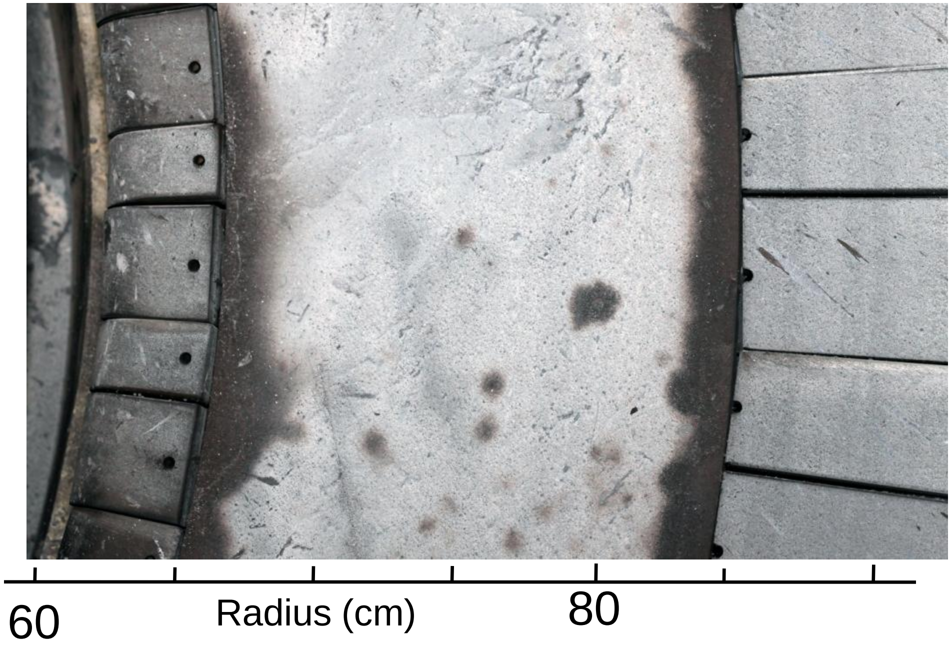


LLD surface was not pure Li



- Carbon, lithium, and deuterium emission extends across LLD surface after overnight Li evaporation.
- No marked change at LLD location

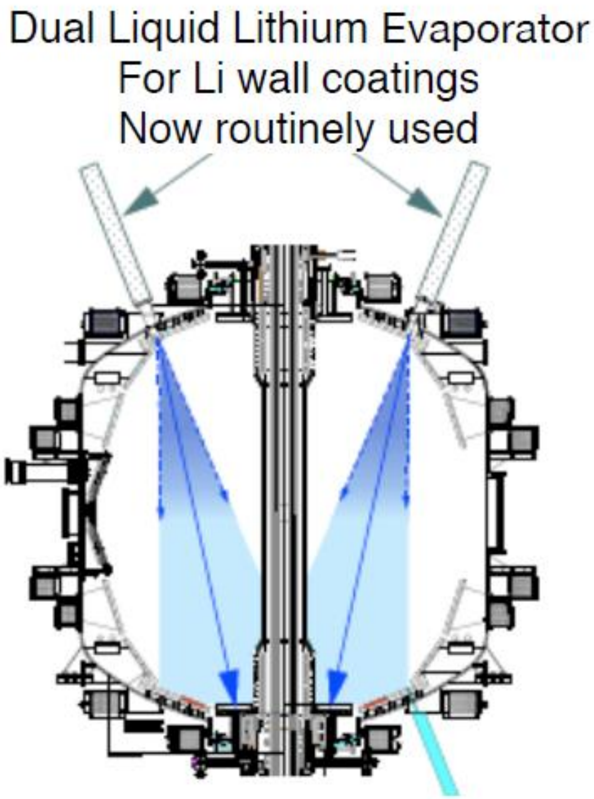
LLD after vent at end of run

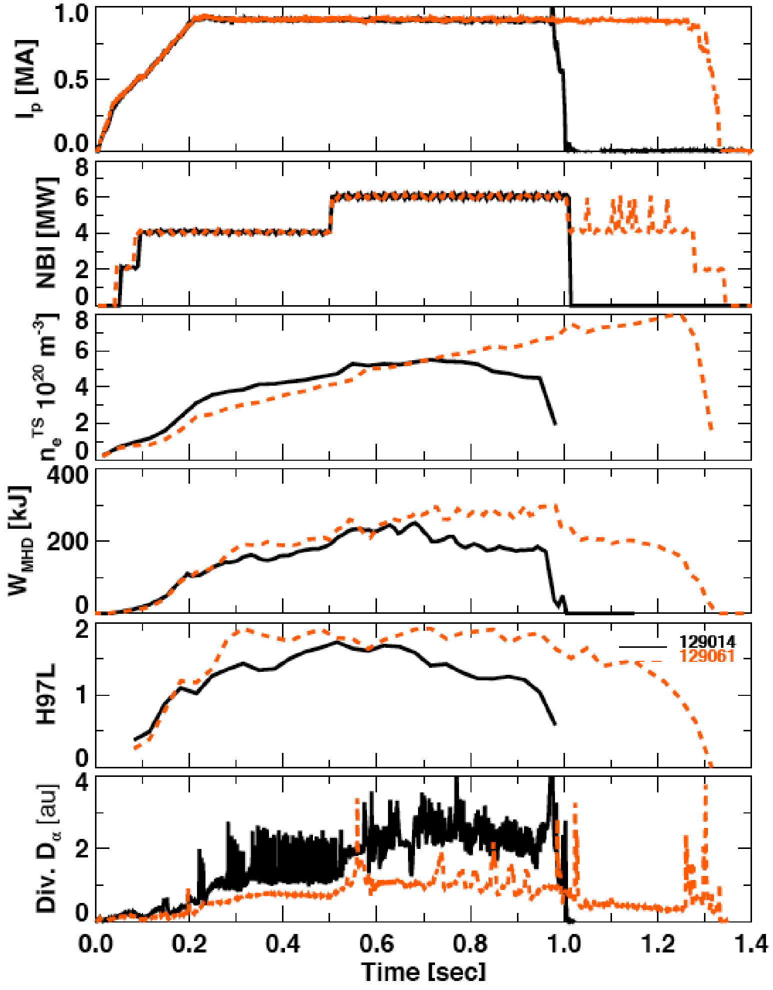


- LLD surface converted to Li_2CO_3 following vent
- LLD edges exhibit evidence of sputtered graphite from plate to graphite tile (vessel-ground) arcing.
- Acetic acid tests on the LLD after run suggests that Li does wick into Mo pores and is depleted from the surface at blackened region.
- Reactions with residual gasses also likely

Solid lithium surface coatings pump D, increase confinement, stored energy, and pulse length, and eliminate ELMs

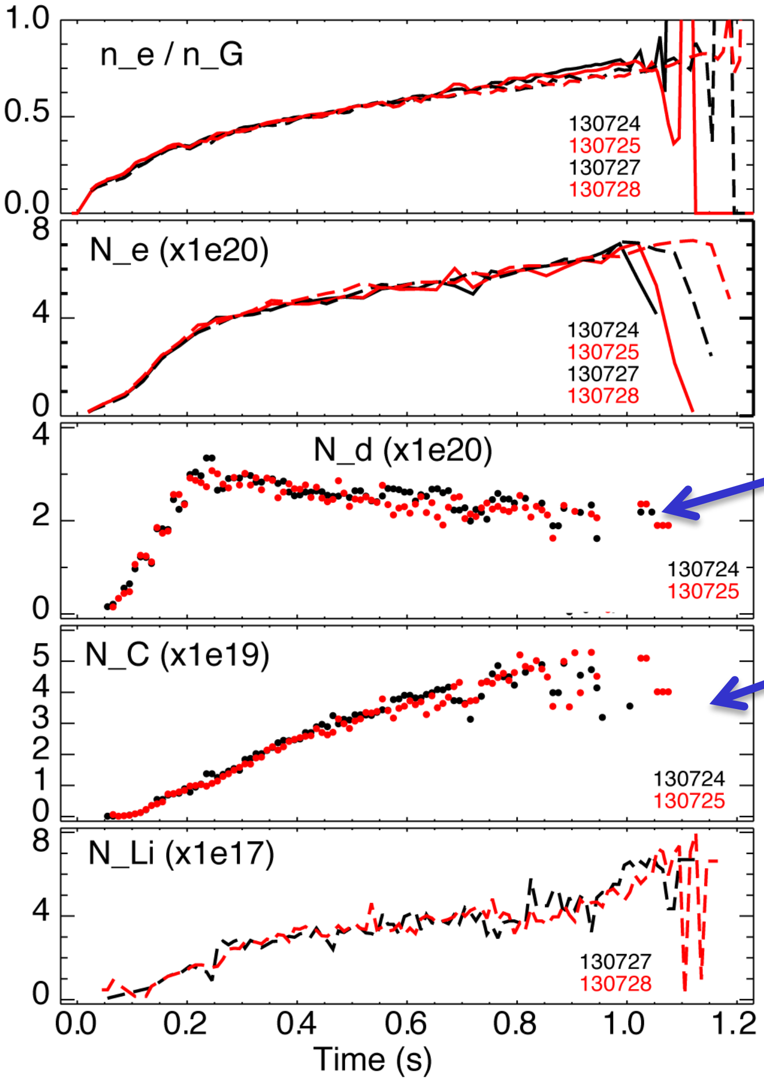
- **2009: Lithium evaporation became baseline wall-conditioning tool**


Dual Liquid Lithium Evaporator
For Li wall coatings
Now routinely used



- **No longer perform between-shot He glow**
- **Typically deposit 50 to 300mg Li between shots**

With lithium coating pumping, deuteron inventory is constant or even decreasing, C accumulates, Li saturates



R. Bell, M. Podestà (PPPL) V. Soukhanovskii (LLNL)

Electron density rises monotonically in lithium induced ELM-free H-mode

Strongly Li-pumped discharges exhibit D inventory pump-out

but C inventory increases

Li inventory increases slowly, remains at very low value

- **Lithium screening efficiency is high, penetration factor is low: $N_{Li} / \Gamma_{Li} \sim 0.0001$**

As a result, NSTX is shifting emphasis from D inventory control to C impurity reduction

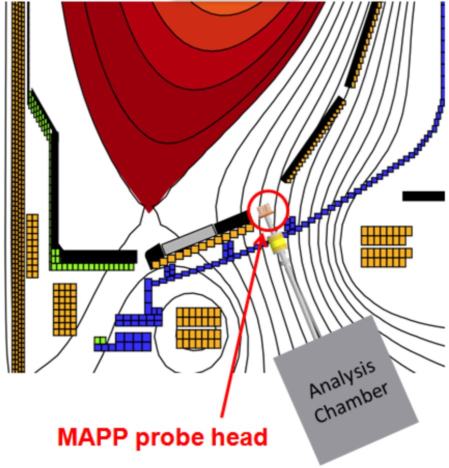
New diagnostics will be used to investigate relationship between Li-conditioned surface composition & plasma behavior

- Chemistry of Li on C/Mo critical, complex, under-diagnosed
- Li very chemically active → *prompt* surface analysis required to characterize the lithiated surface conditions during a shot

- An in-situ materials analysis particle probe (MAPP) being installed on NSTX to provide prompt surface analysis
  - Ex-vessel but in-vacuo surface analysis **within minutes** of plasma exposure using state of the art tools

- Li experiments will utilize MAPP to study:
  - Reactions between evaporated Li and PFCs, gases
  - Correlation surface composition and plasma behavior, comparisons to lab experiments, modeling
  - Characterizations of fueling efficiency, recycling

Purdue



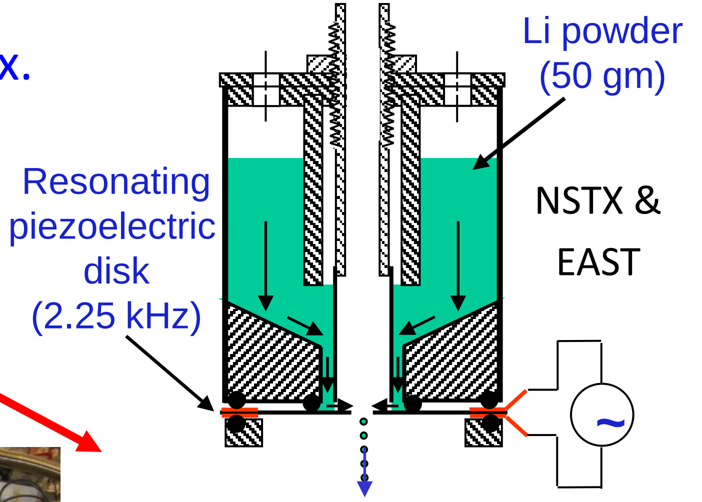
MAPP probe head

Lithium Technology developed for NSTX needs

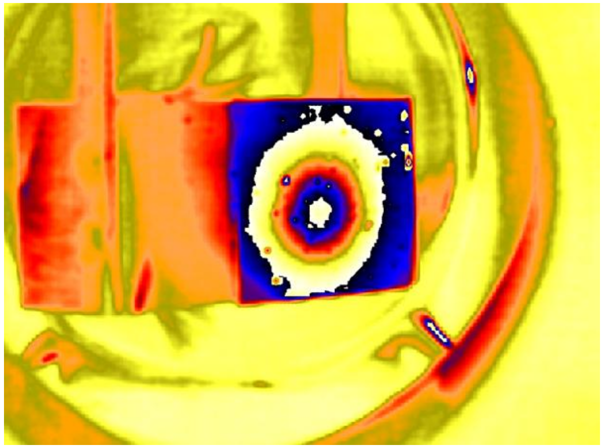
- R&D continuing for LLD performance at high heat flux.
- Continuous Li replenishment systems:

Li delivery by: powder

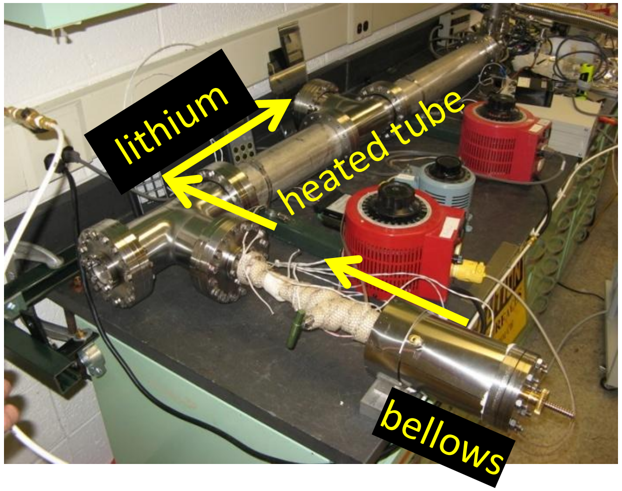
granules



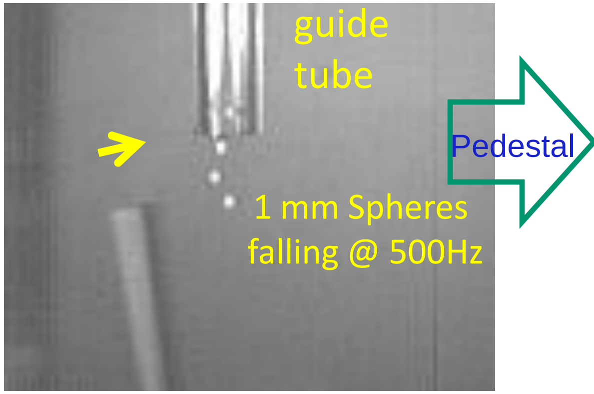
High heat flux test



False color image of LLD sample during NB exposure $T \leq 225$ °C @ 1.5 MW/m^2 - 3s. Potential PFC for upgrade



10 g of molten Li moved 1.1m in vacuum and ejected 7.6 cm from nozzle



500 Hz Impeller Rotating @ 95 m/s

Midplane injection for ELM pacing