NSTX Research Overview: Status and Plans

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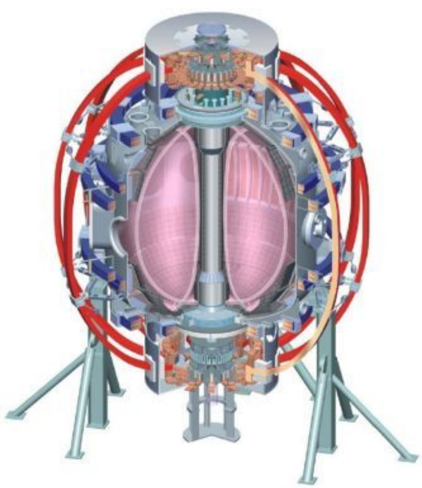
Masayuki Ono, PPPL

For the NSTX Research Team

ISTW 2011, NIFS, Japan
September 27--30, 2011



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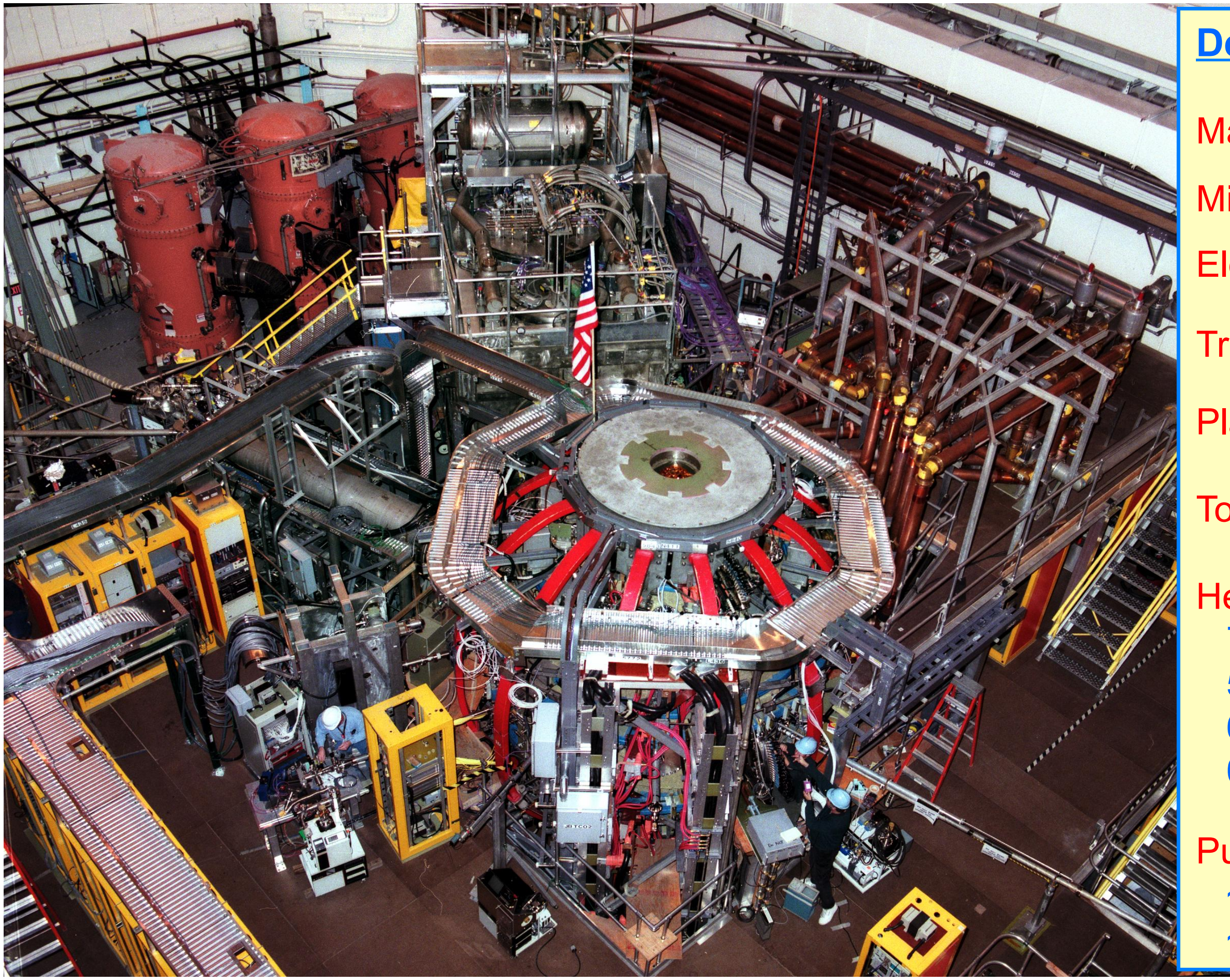
Talk Outline

- Introduction / Tutorial
- Boundary / Lithium Program
- HHFW for Ramp-up and Maintenance
- Energetic Particle Research
- TF-Fault Investigation
- Facility Upgrade Plan
- NSTX Upgrade Project Update
- Conclusion

NSTX is a MA-class ST facility Located at PPPL, USA

NSTX Operated From Feb. 1999 to October 2010



Device Capabilities

Major Radius 0.85 m

Minor Radius 0.68 m

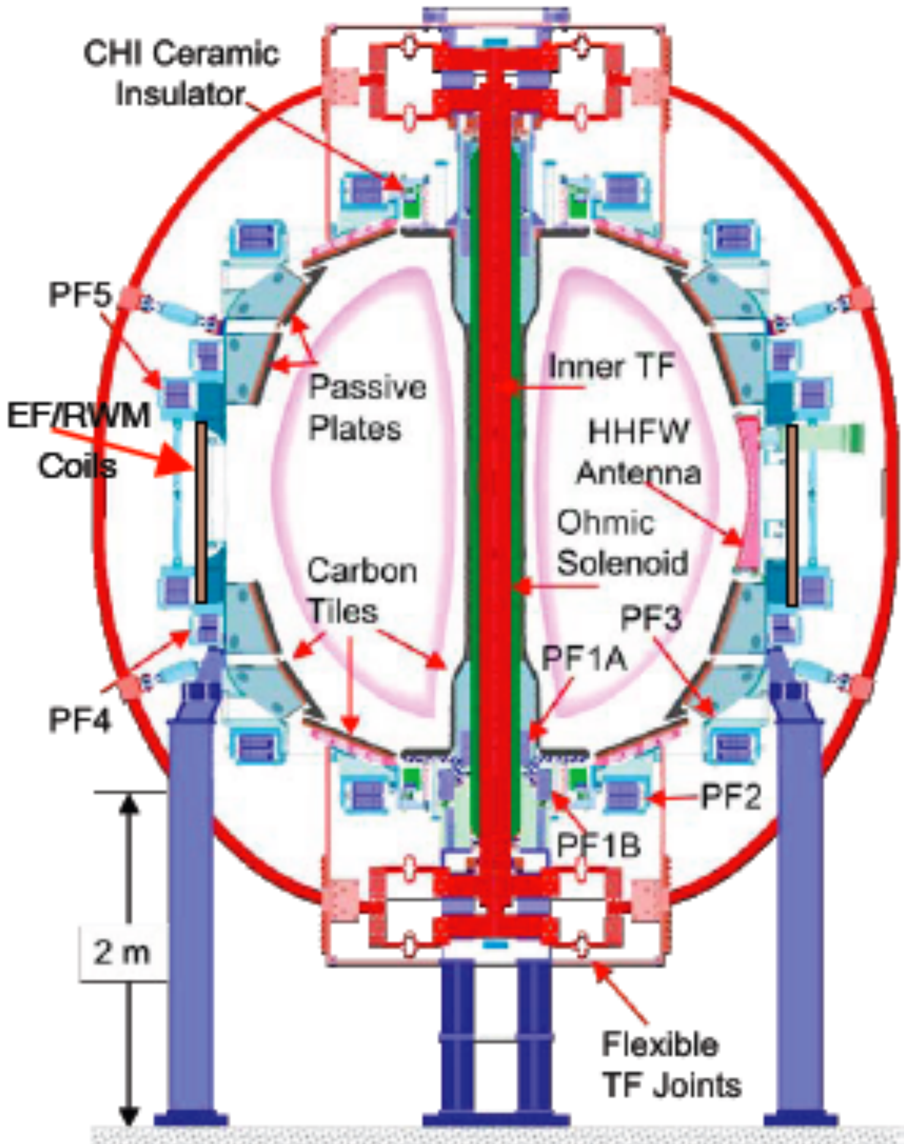
Elongation 1.8 - 3.0

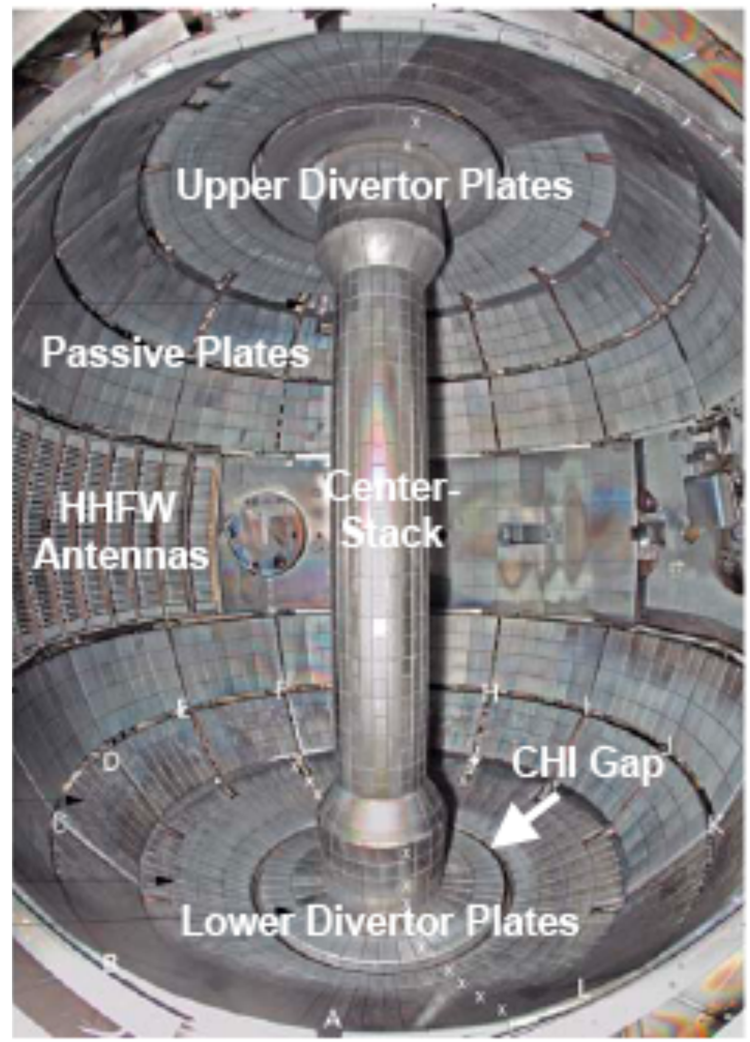
Triangularity 0.2 - 0.8

Plasma Current
1 MA (1.5 MA peak)
Toroidal Field
0.35 - 0.55 T
Heating and CD
7 MW NBI (2 sec)
5 MWNBI (5 sec)
6 MW HHFW (5 sec)
0.2 MA CHI

Pulse Length
~ 1 sec at 0.55 T
~ 2 sec at 0.38 T

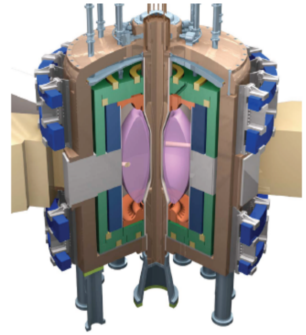
NSTX Device Cross-Section and VV Internal Components
Removable Center-Stack Design





NSTX Mission Elements

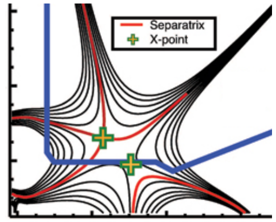
- Advance ST as candidate for Fusion Nuclear Science Facility (FNSF)



ST-FNSF

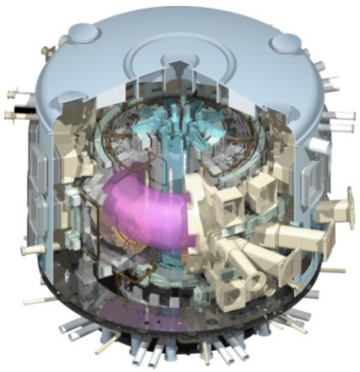
- Develop solutions for plasma-material interface

Lithium



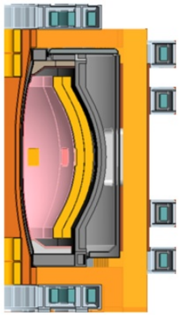
"Snowflake"

- Advance toroidal confinement physics for ITER and beyond



ITER

- Develop ST as fusion energy system



ST Pilot Plant

NSTX Made Significant Progress in All Topical Areas

We Hope to Give You a Comprehensive Update!

Topics	Talks	Posters
Overview	M. Ono / J. Menard	
Boundary / Lithium	M. Ono	V. Soukhanovskii
CHI / Start-up/Ramp-up	R. Raman / M Ono	
MHD	J. Menard	R. Raman
Turbulence and Transport	K. Tritz / J. Menard	Y. Ren / K. Tritz
EP H-Mode	R. Maingi	
HHFW	M. Ono	
Energetic Particles	K. Tritz / M. Ono	
Adv. Scenarios and Control	D. Gates	D. Gates
NSTX Upgrade	M. Ono / J. Menard	
ST Development Paths	J. Menard	

Liquid Lithium Research is High Priority Research For NSTX and NSTX-U

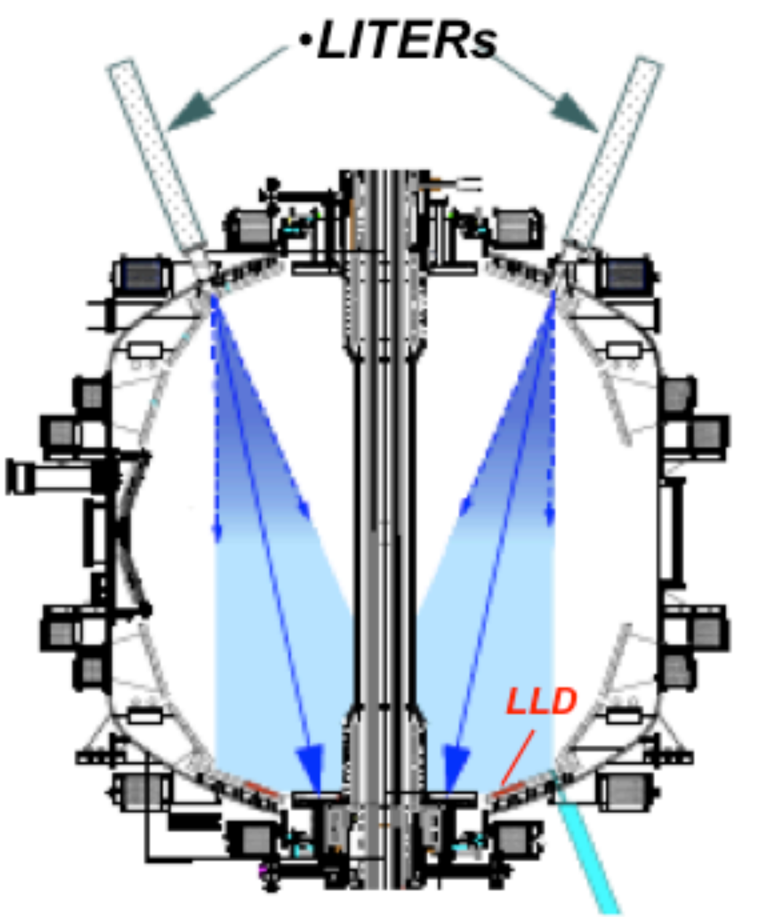
***NSTX Goal:* To investigate effectiveness of lithium for divertor heat and particle control while enhancing plasma performance.**

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

Since 2008, Dual Lithium Evaporators (LITERs) Are Used to Deposit Lithium Coatings on NSTX Lower Divertor



- LITERs aimed toward the graphite divertor. Shown are 1/e widths of the emitted gaussian-like distribution.

- Lithium transported over broad area by wings of LITER distribution and plasma migration.

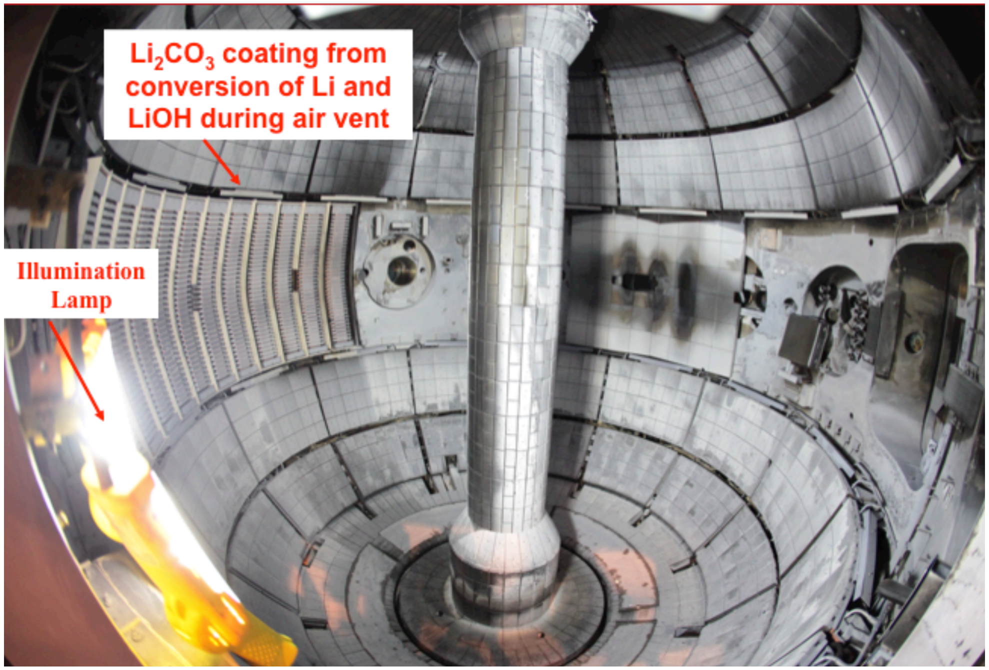


Photo of NSTX interior following 1.3 kg lithium deposition applied during 2010-2011 experimental campaigns indicating extensive lithium coverage due to direct evaporation and plasma transpot

H. Kugel, PPPL

Lithium Significantly Improved NSTX Operations

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

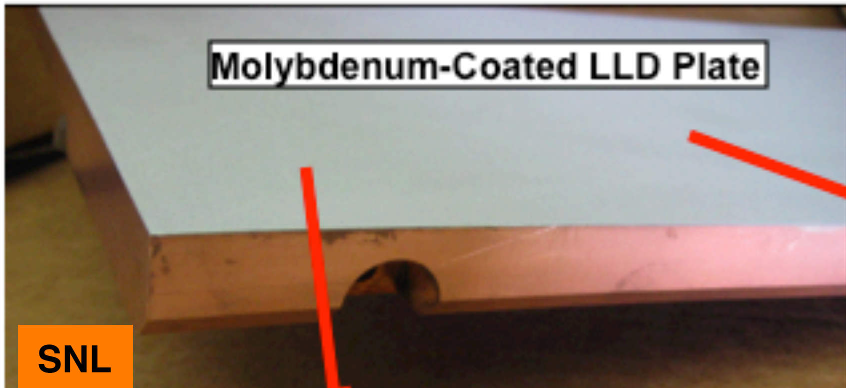
- **Enabled rapid recovery of experimental plasma operation after an extended vacuum vessel opening compared to boronization.**
- **Reduced oxygen impurity level and generally improved plasma reliability and performance.**
- **Conditioned PFCs to produce reproducible shots and eliminated the need for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

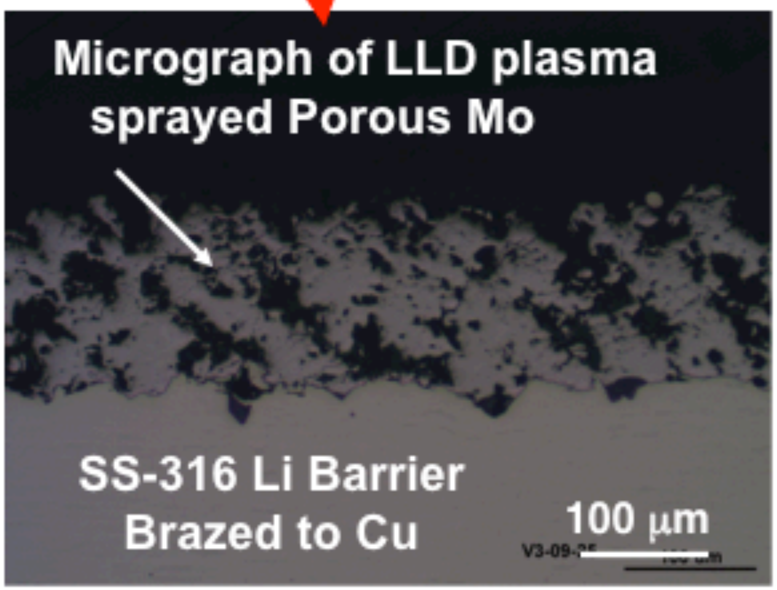
NSTX Plasma Operation Statistics

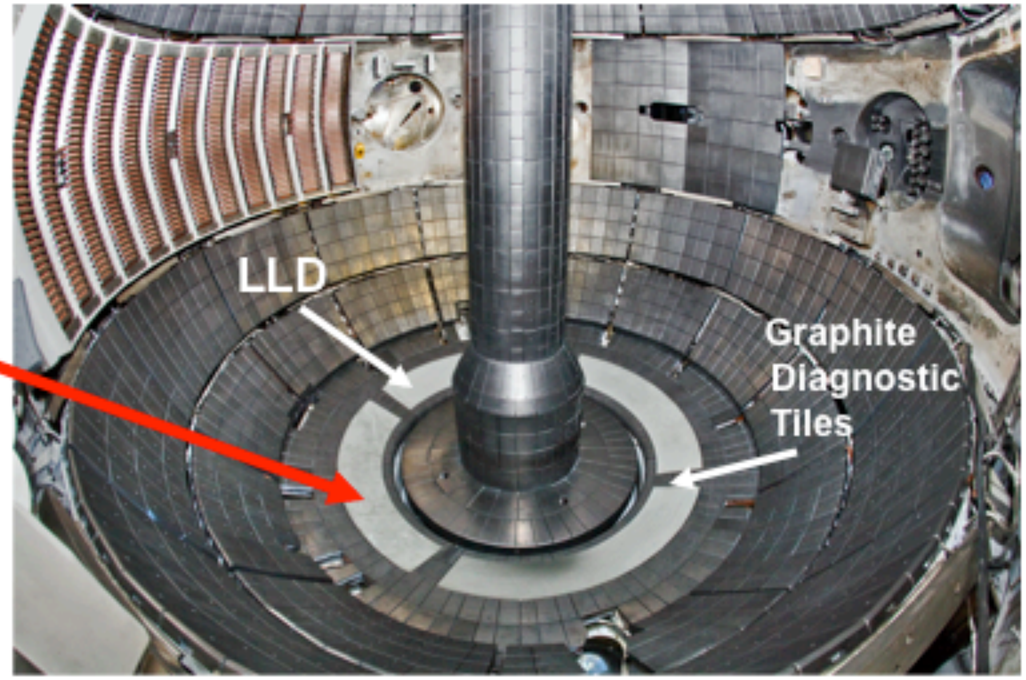
Fiscal Year	Run Weeks	Achieved Shots	Achieved Shots/Week	Lithium Operations (%)
2011	4.2	839	199	~ 100
2010	15.4	2941	191	~ 100
2009	16.8	2750	163	92
2008	16.5	2570	156	46
2007	12.6	1890	150	69
2006	12.7	1615	127	0
2005	18.0	2221	124	0

2010 Liquid Lithium Divertor (LLD) Installed in NSTX with Porous Molybdenum Surface to Retain Lithium

0.165 mm Mo plasma sprayed with 45% porosity on a 0.25 mm SS barrier brazed to 22.2 mm Cu.





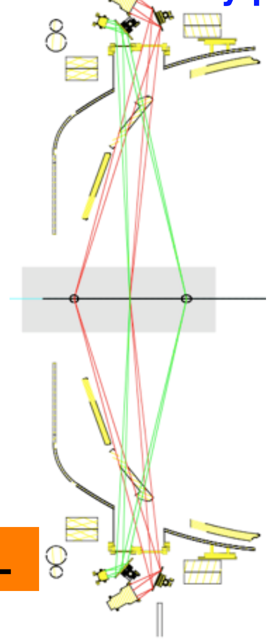


- 4 heated plates (80°each) separated by graphite diagnostic tiles. Each section electrically grounded at one location to control disruption induced currents
- LLD loaded by LITER evaporation
  - LLD has 37g Li capacity (100% full)
  - 2010 tests with LLD up to 200% full
  - 5% of LITER output reaches LLD

Enhanced Diagnostics for LLD and Boundary Physics

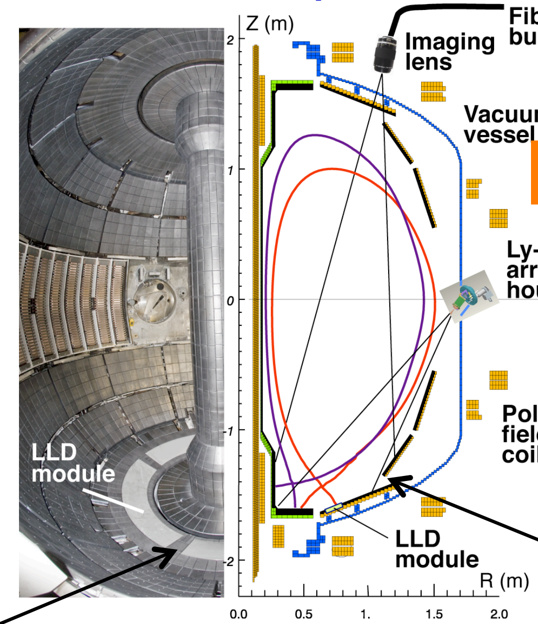
Crucial for Assessing LLD - Multi-Institutional Contributions

Lithium CHERS
Lithium density profile



PPPL

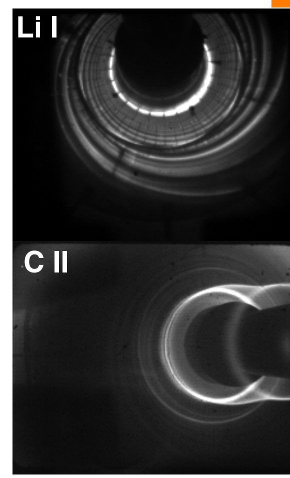
Divertor Imaging Spectrometer



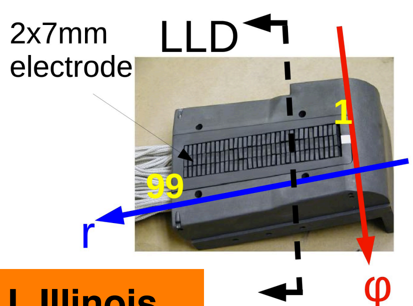
LLNL

Two fast 2D visible and IR cameras with full divertor coverage

LLNL, ORNL

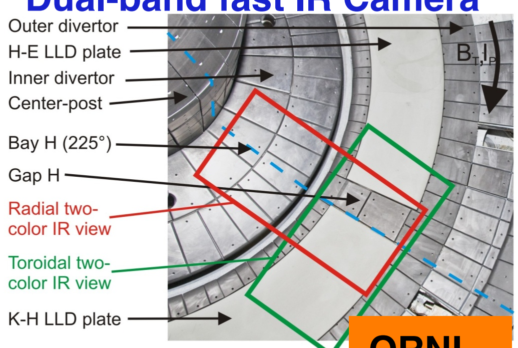


Dense Langmuir Probe Array



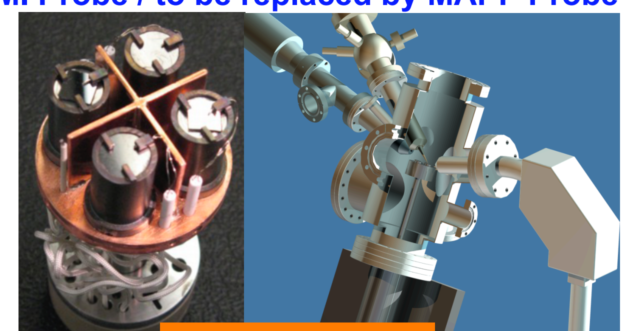
U. Illinois

Dual-band fast IR Camera



ORNL

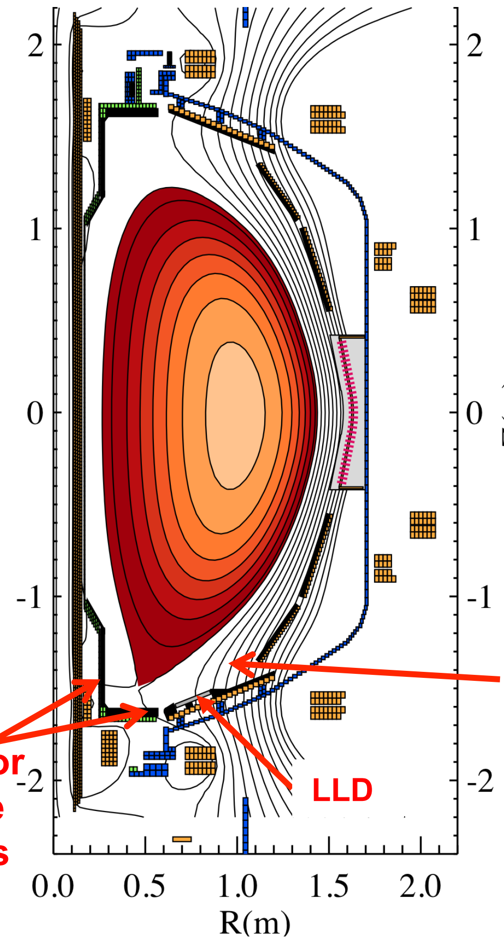
PMI Probe / to be replaced by MAPP Probe



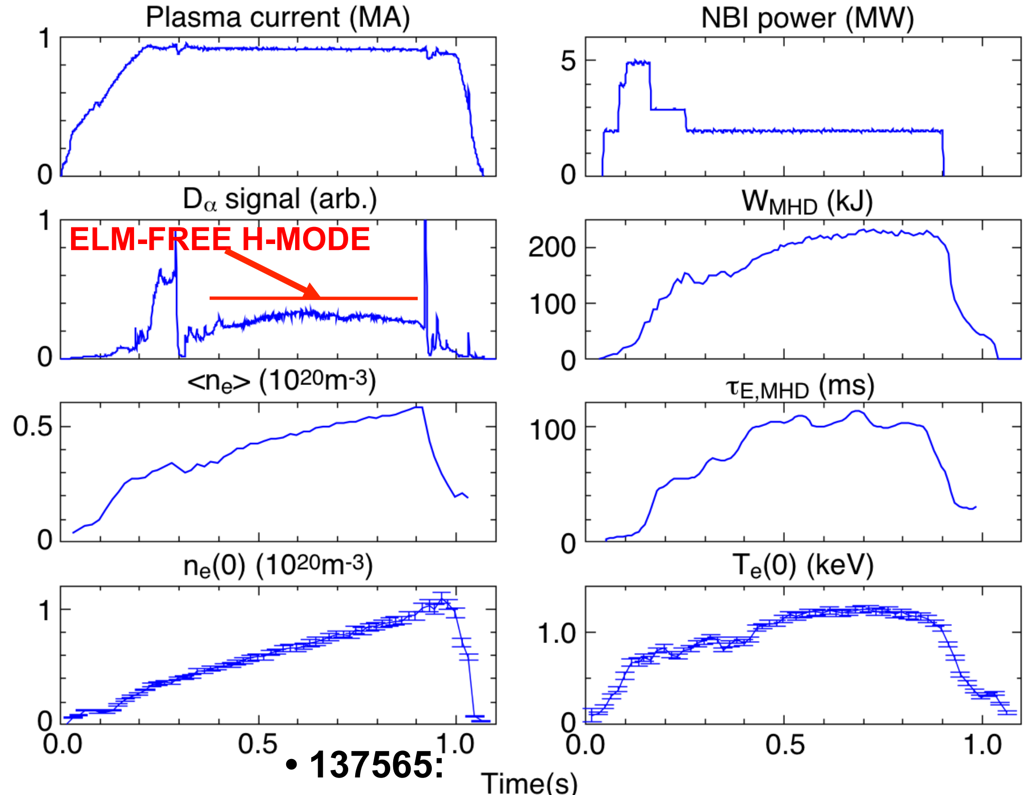
Purdue U.

LLD Yielded Improvements Similar to Lithiated Graphite

LLD Surface Temperature Varied from Solid to Liquid States



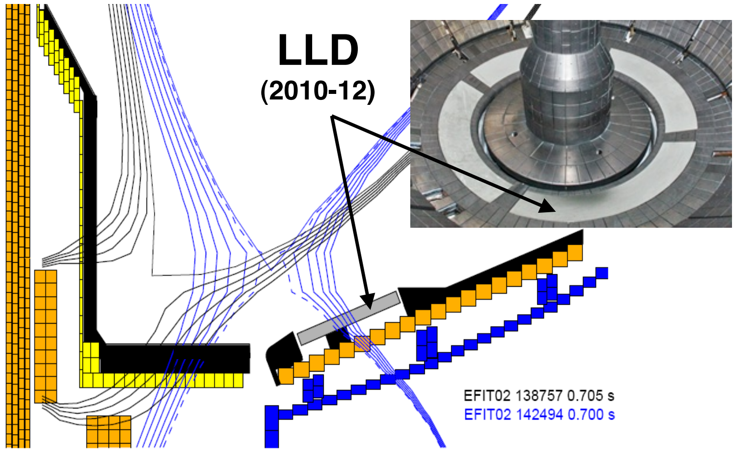
TYPICAL DISCHARGE OBTAINED WITH LITHIUM WALL CONDITIONS



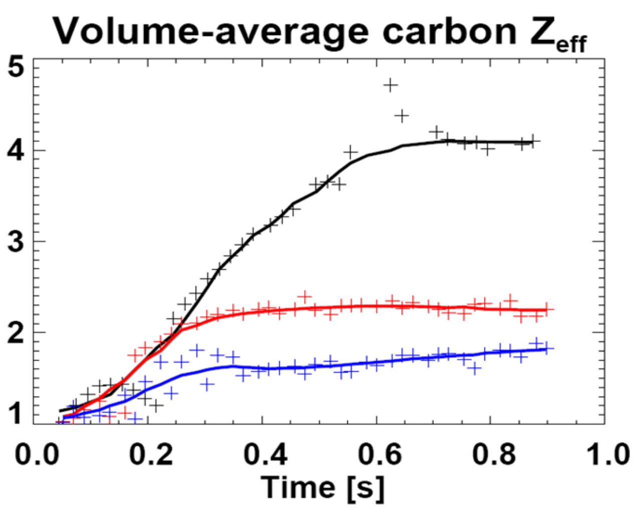
- **137565:**
  - **OSP=0.50m on graphite with open field lines on LLD with Li fill <5%**
  - **LLD T_0 = 319°C, LITER 21mg/min**

The required fueling and resultant edge conditions were about the same as when using lithiated graphite over entire lower divertor

Operation with outer strike-point on Mo LLD (coated with Li) compatible with high plasma performance, low impurities



- **LLD did not increase global D pumping beyond that achieved with LiTER**
  - **Solid Li on C pumps D quite efficiently**
  - **Liquid Li may react rapidly w/ background gases**
  - **C on LLD may have impacted D pumping**
- **Divertor T_e increases when $T_{LLD} > T_{Li\text{-}melt}$**
- **No evidence of Mo from LLD in plasma during normal operation**
- **Operation with strike-point (SP) on LLD reduced core impurities (due to ELMs?)**



◀ SP on inner carbon divertor – high δ (no ELMs)

◀ SP on LLD - moderate δ, $T_{LLD} < T_{Li\text{-}melt}$

◀ SP on LLD – moderate δ, $T_{LLD} > T_{Li\text{-}melt}$ (+ fueling differences)

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

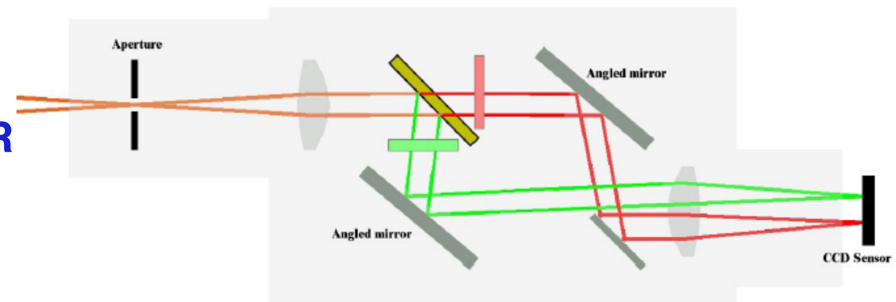
→ High-Z impurities also reduced, $\beta_N > 4$ sustained

Dual-band IR Camera Enabled LLD Surface Temp. Monitor
Clamping of the temperature observed near Li melting temp.

Dual-band IR camera can largely eliminate effects of variable surface emissivity such as the case for LLD:

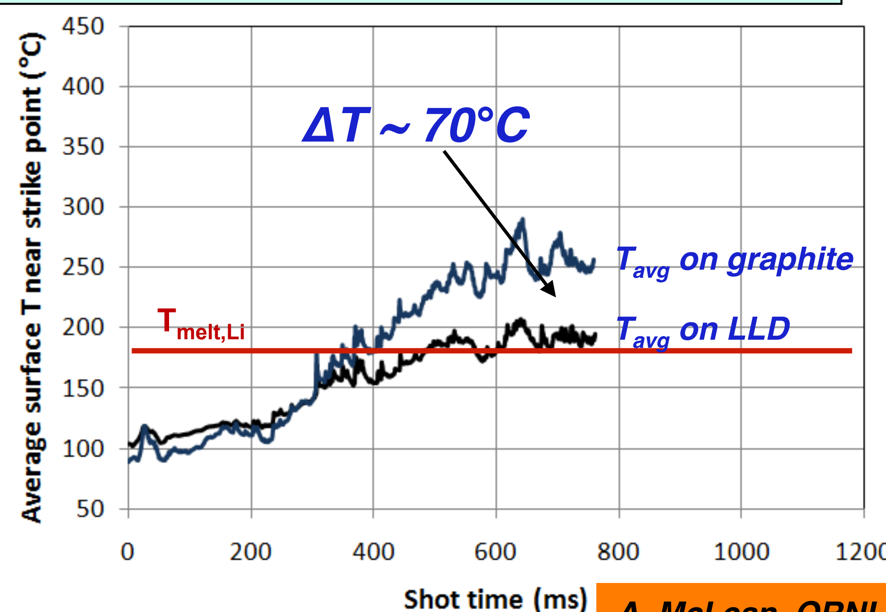
- An optical splitter is inserted between the IR camera and lens
- Projects separate IR wavelengths side-by-side on the camera's detector

Dual-band IR adaptor layout



T_{avg} near Outer Strike Point on LLD and graphite tile at equal radii suggests that Li in the LLD is causing clamping of the temperature!

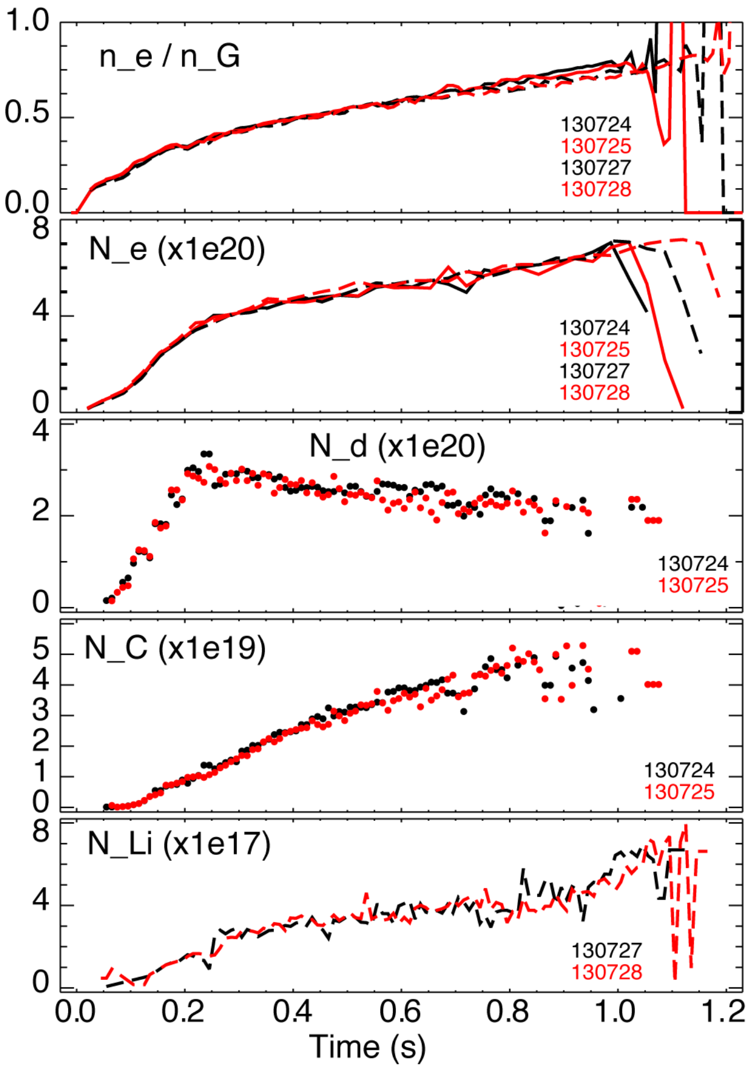
- Series of 10 repeat discharges with outer strike point on the LLD
- T_{avg} on graphite gap tile increases through all shots in SQRT(t) fashion
  Average $T_{surface}$ of ~250°C
- T_{avg} plotted at same radius, but on LLD
- T_{avg} on LLD surface gravitates at $T_{melt,Li}$
  - Efficient heat removal in liquid Li?
  - Li radiation?
  - Vapor shielding playing a role?



A. McLean, ORNL

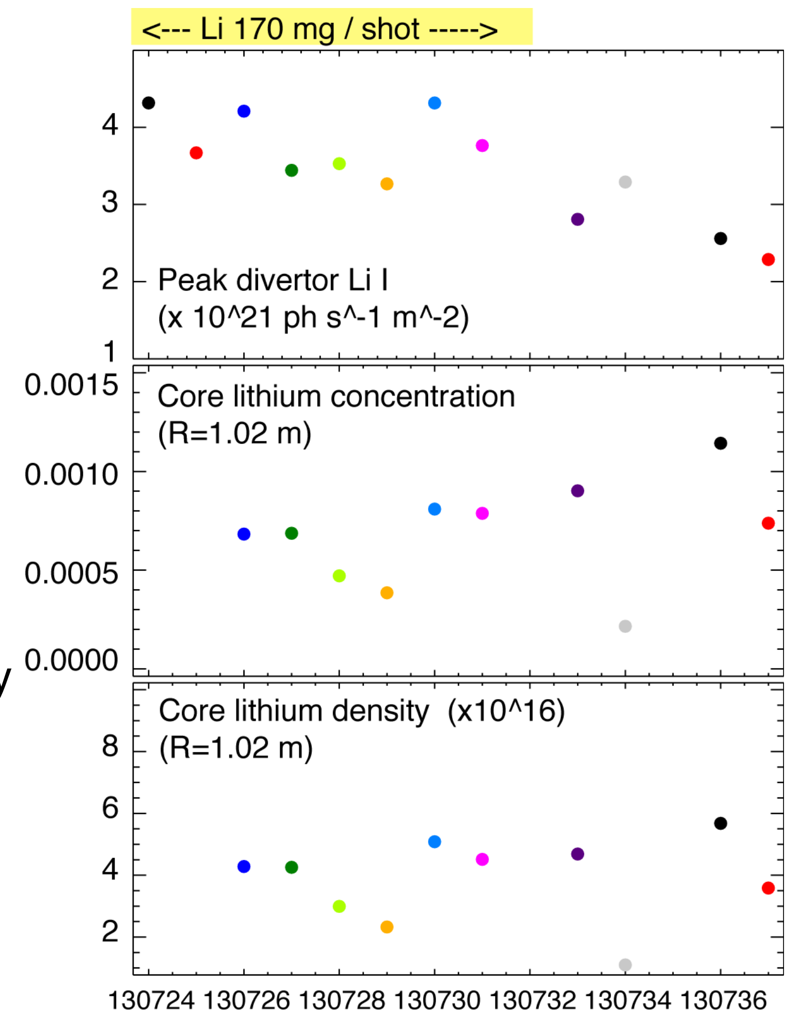
Core Li Density Very Low, No Dependence on Li Source

Lithium screening efficiency high, penetration factor N_{Li} / Γ_{li} ~ 0.0001



Impurity density profiles from CHERS

- C VI, n = 8-7, 529.1 nm
- Li III, n = 7-5, 516.7 nm

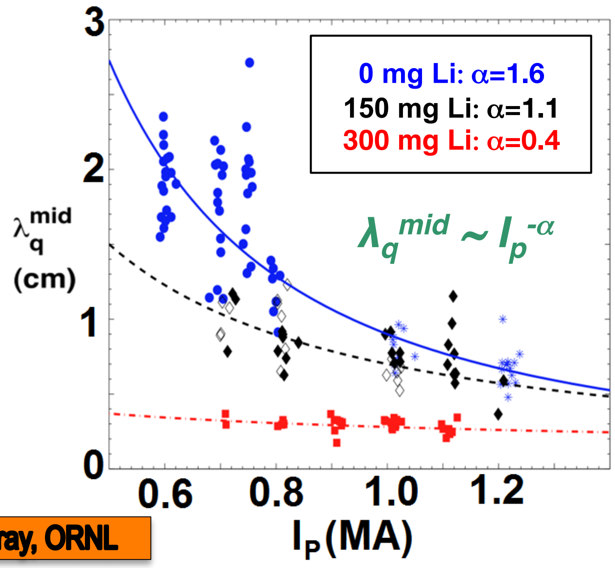


R. Bell (PPPL)

V. Soukhanovskii (LLNL)

Divertor Heat Flux Measured and Characterized

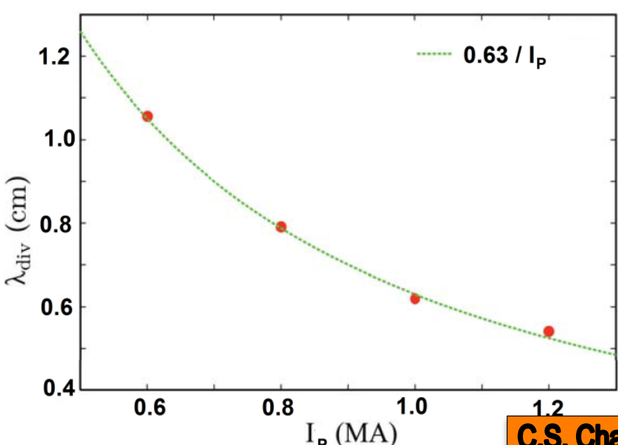
Ip scaling and Li effects were surprising results



T. Gray, ORNL

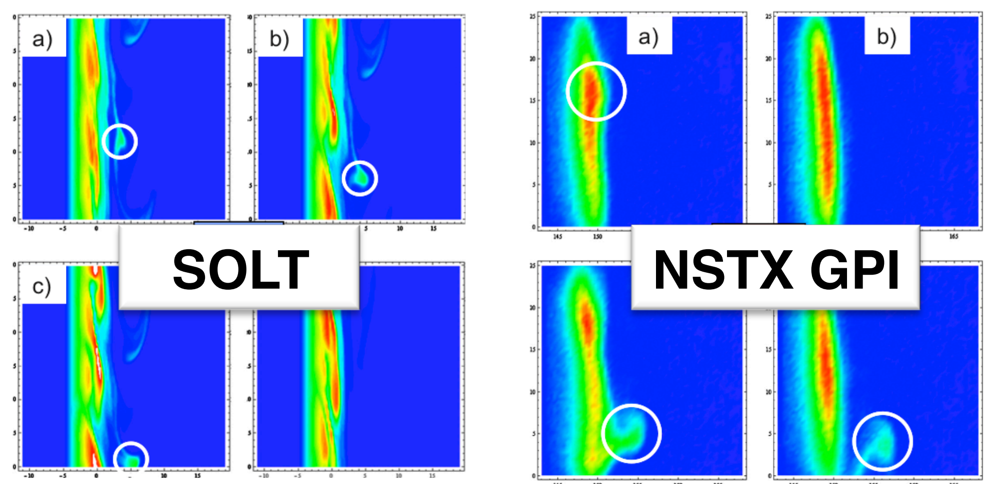
- **Divertor heat flux width λ_q^{mid} decreases with increased plasma current I_P**
  - Potentially major implications for ITER
  - NSTX: λ_q^{mid} further decreases with Li
  - Physics mechanisms not yet fully understood
- For NSTX-U parameters: λ_q^{mid} = 3 ± 0.5 mm

- **XGC0 kinetic neoclassical consistent with ~ $1/I_P$ scaling**



C.S. Chang, PPPL

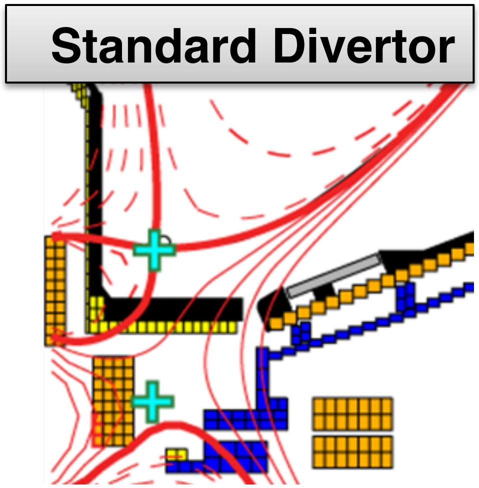
Blob formation in SOLT is similar to NSTX Gas Puff Imaging (GPI) diagnostic data

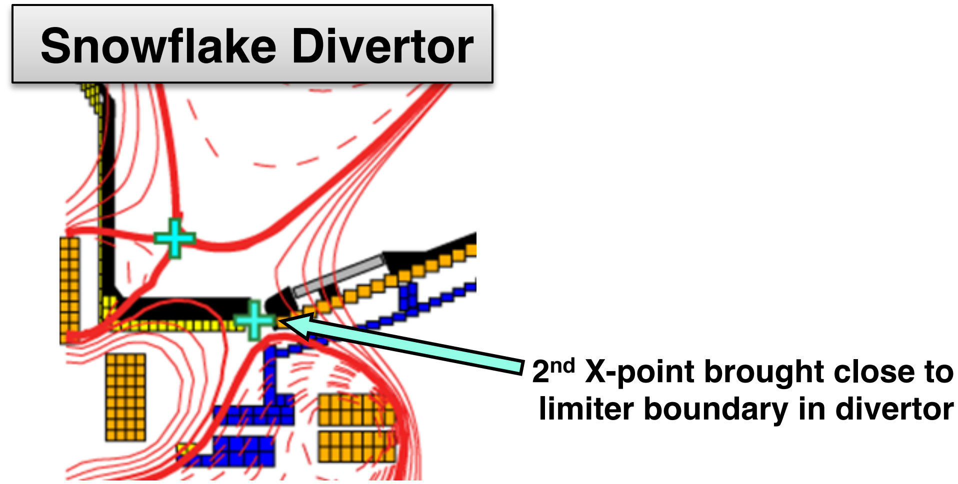


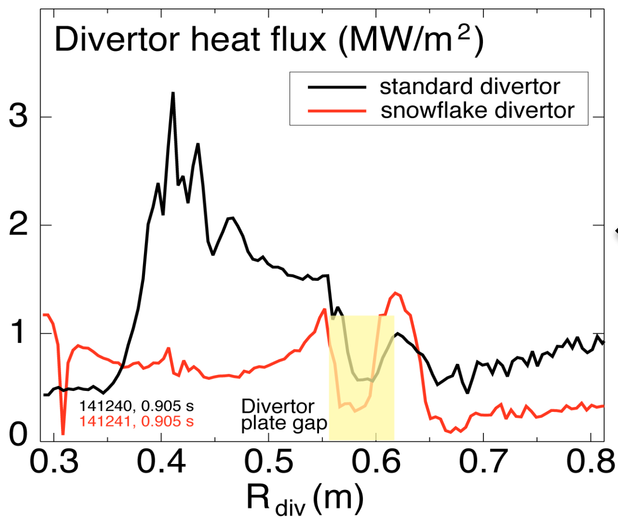
- **SOLT scaling weaker than $1/I_P$**

J. Myra, Lodestar

"Snowflake" divertor configuration provides significant divertor heat flux reduction and impurity screening







V. Soukhanovskii (LLNL)

Higher flux expansion (increased div wetted area)
Higher divertor volume (increased div. losses)

- **Peak heat flux reduced by 2-3×**
- Snowflake H-mode τ_E similar to standard divertor
- Core and pedestal carbon reduced by 50%
- Double-null snowflake is baseline divertor in Upgrade to maintain heat flux < 10MW/m^2 for 2MA, 15MW plasmas

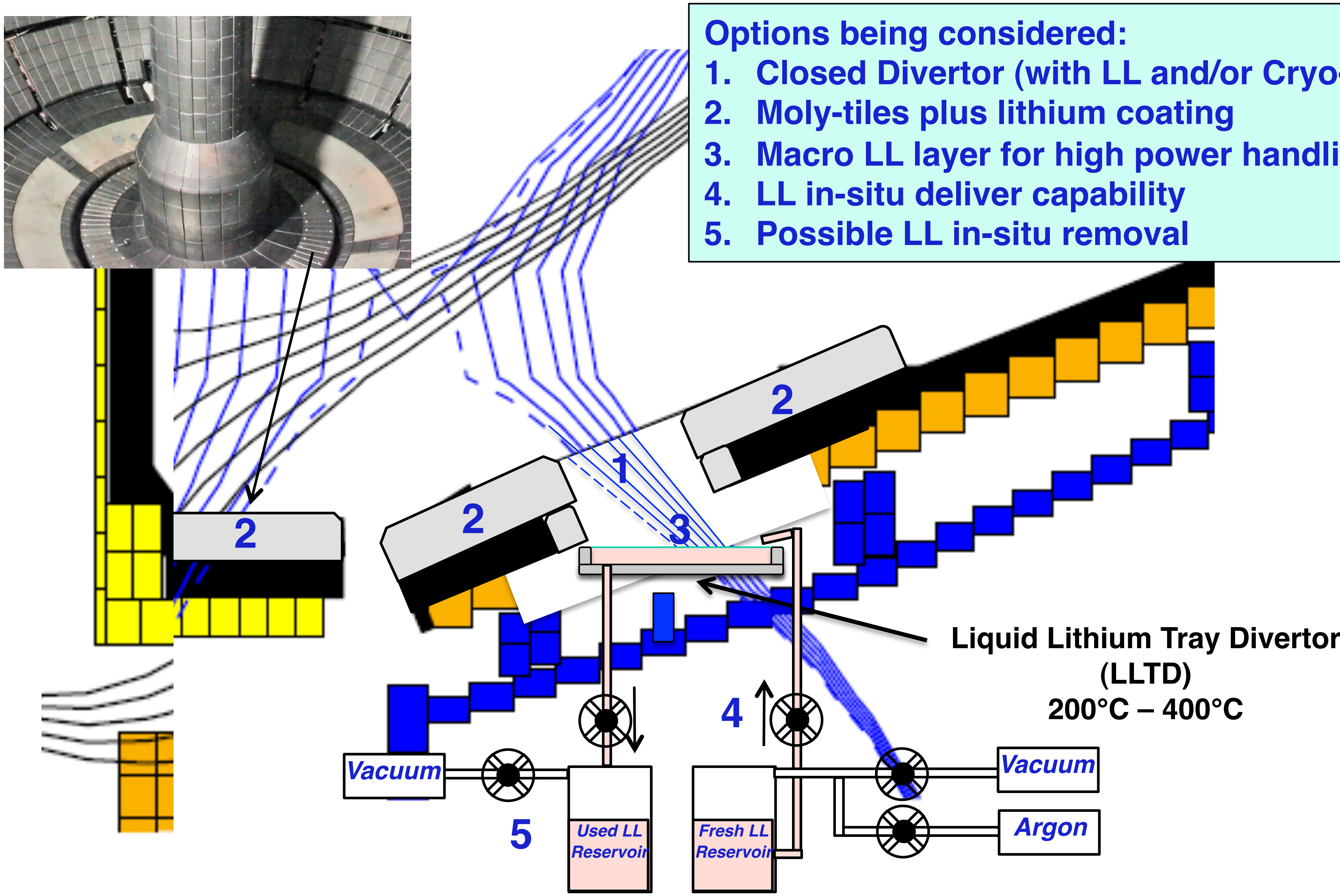
- **Assess control of U/D snowflake, possible synergies with Li, radiation**

For Post-Upgrade, Divertor Upgrade Are Being Considered

Example below: a “Closed” Divertor System with Liquid Lithium Divertor

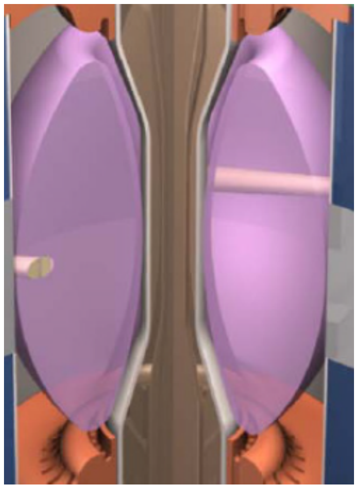
Options being considered:

1. Closed Divertor (with LL and/or Cryo-pump)
2. Moly-tiles plus lithium coating
3. Macro LL layer for high power handling
4. LL in-situ deliver capability
5. Possible LL in-situ removal

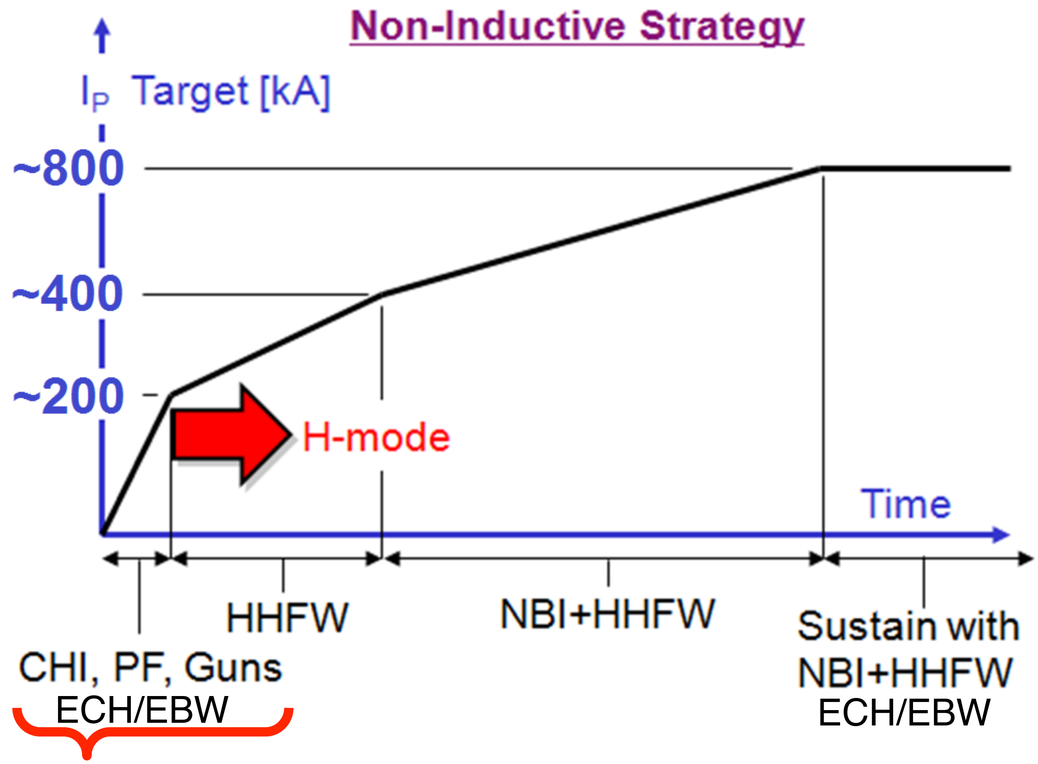


Plasma initiation with small or no transformer is unique challenge for ST-based Fusion Nuclear Science Facility

ST-FNSF has no/small central solenoid

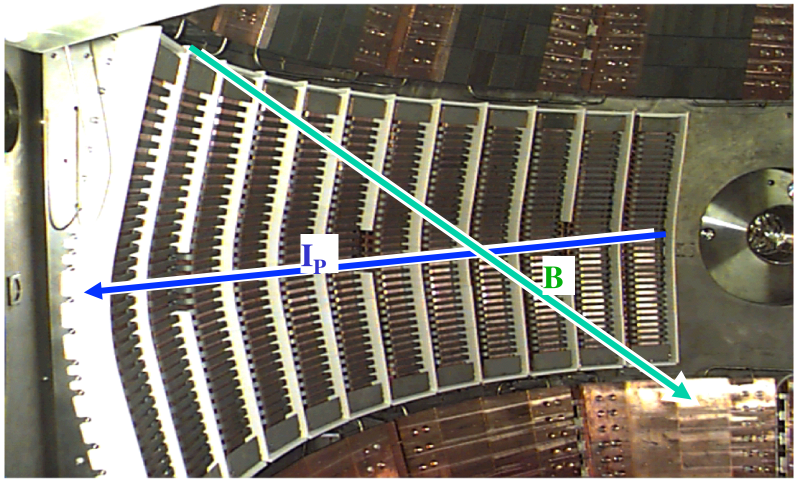


Iron core and/or thin mineral insulated conductor transformer may be able to provide 1-2MA FNSF start-up current

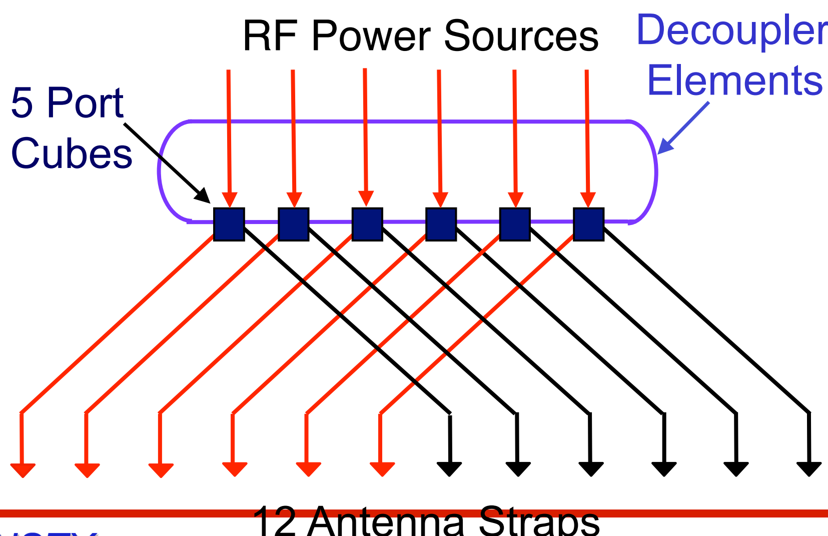


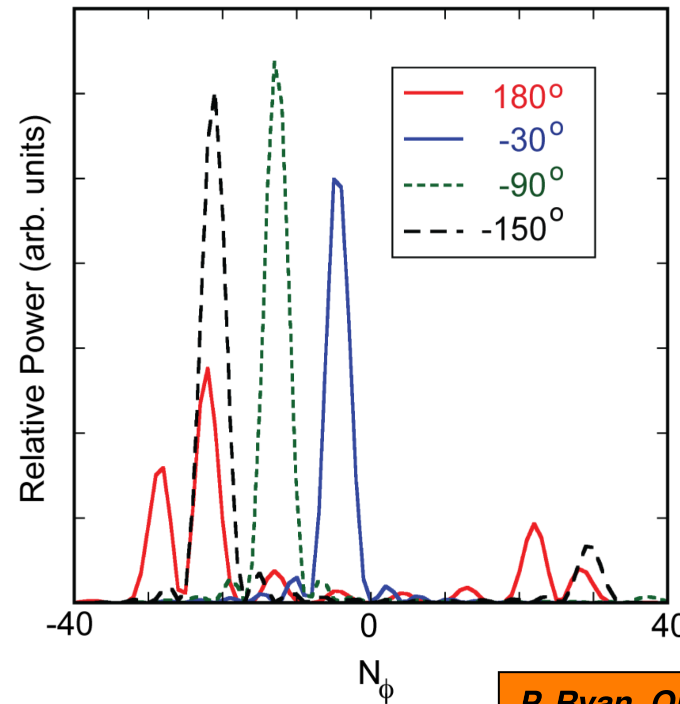
- Upgrade Goal:
  - ~0.3-0.4MA fully non-inductive start-up with CHI + Gun + ECH/EBW + HHFW
  - Use NBI current drive to ramp-up from 0.4MA to 0.8-1MA
  - More tangential 2nd NBI of upgrade has much higher CD efficiency at low I_P
  - Provide physics basis for non-inductive ramp-up to high performance 100% non-inductive ST plasma → prototype FNSF

NSTX HHFW antenna has well defined spectrum, ideal for studying dependence of heating on antenna phase



HHFW antenna extends toroidally 90° 6 MW available at 30 MHz





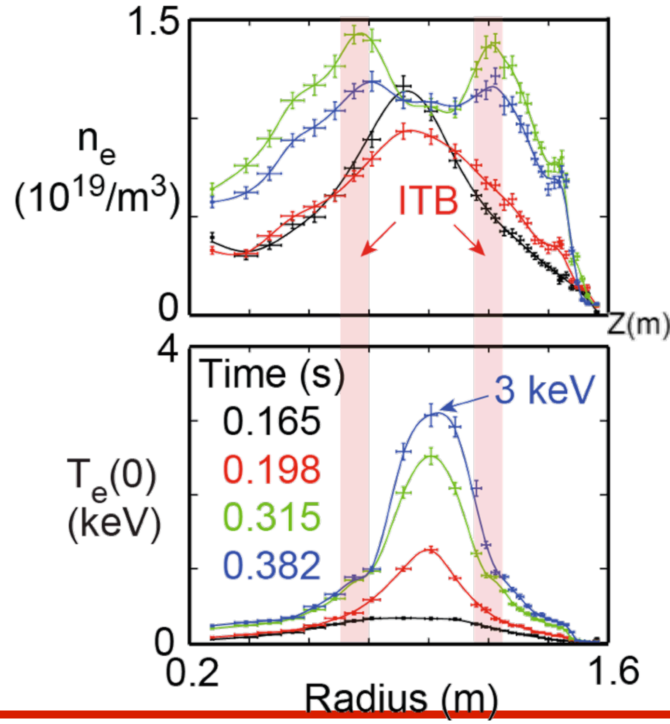
P. Ryan, ORNL

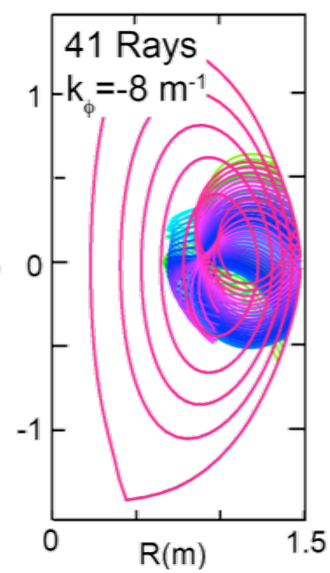
Phase between adjacent straps can be adjusted in real time between 0° to 180°

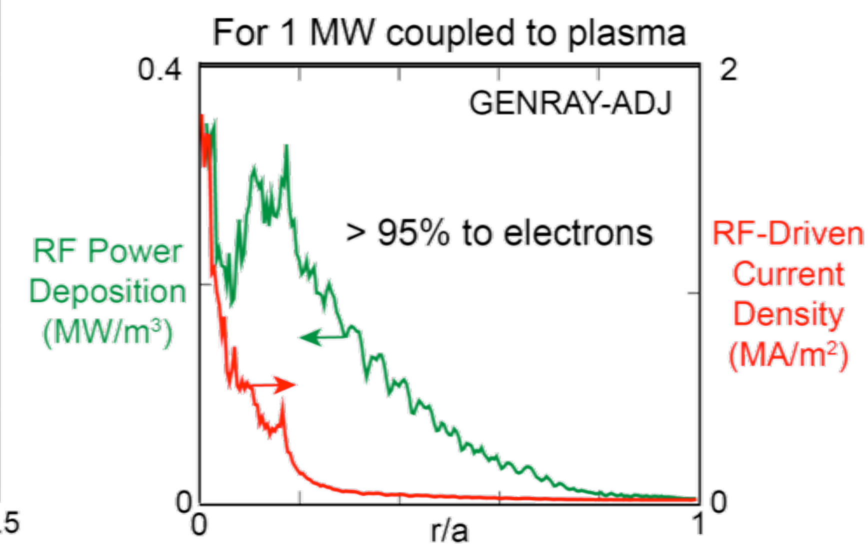
Large B pitch affects wave spectrum in plasma core

Progress made in sustaining HHFW heating during I_p=300 kA RF-only H-mode plasma; $T_e(0)$ = 3keV with only 1.4 MW

- **RF-only H-mode sustained in deuterium plasma:**
  - **Better plasma-antenna gap control due to reduced PCS latency**
  - **Modeling predicts I_{RFCD} ~ 70 kA, $I_{Bootstrap}$~ 130 kA → f_{NI} ~ 65%**
  - **High f_{NI} enabled by positive feedback between ITB, high $T_e(0)$ and RF CD**
  - **f_{NI} ~ 100% requires P_{RF} ~ 3 MW**



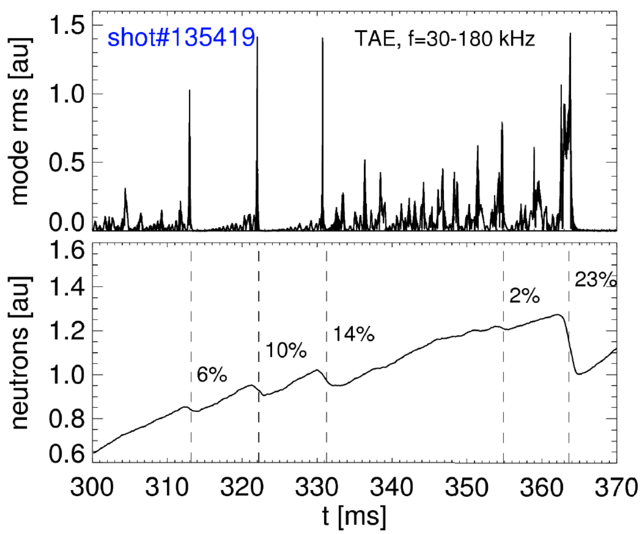


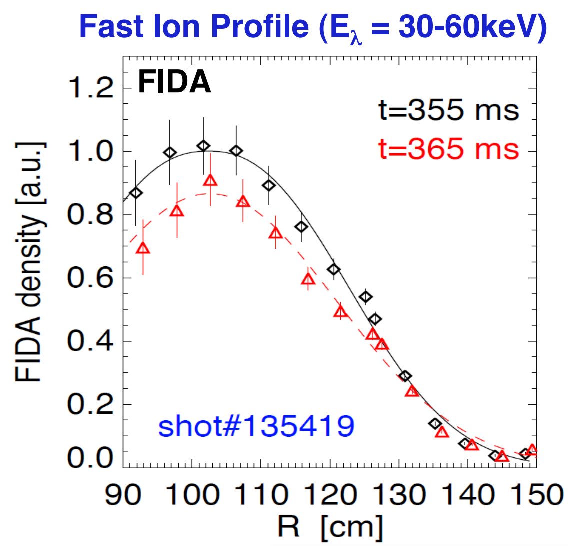


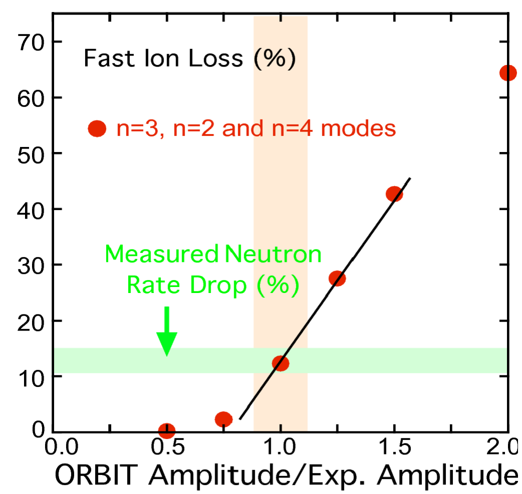
G. Taylor (PPPL)

TAE-avalanche induced neutron rate drop modeled successfully using NOVA and ORBIT codes

- Toroidal Alfvén Eigenmode (TAE) avalanches in NBI-heated plasmas associated with transient reductions in D-D (beam-target) neutron rate







M. Podesto, UC Irvine

- Change in beam-ion profile measured with Fast-ion D_α (FIDA)
- Modeled using NOVA-K + ORBIT codes

UCLA

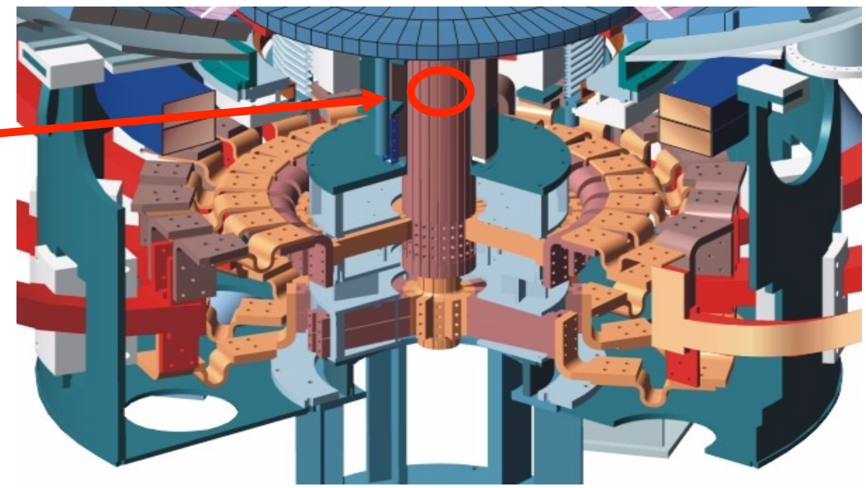
  - Mode structure obtained by comparing NOVA calculations w/ reflectometer data
  - Fast ion dynamics in the presence of TAEs calculated by guiding-center code ORBIT
- Improve predictive capability for TAE/GAE/CAE with self-consistent and advanced codes (M3D-K, HYM, SPIRAL)

E. Fredrickson, PPPL

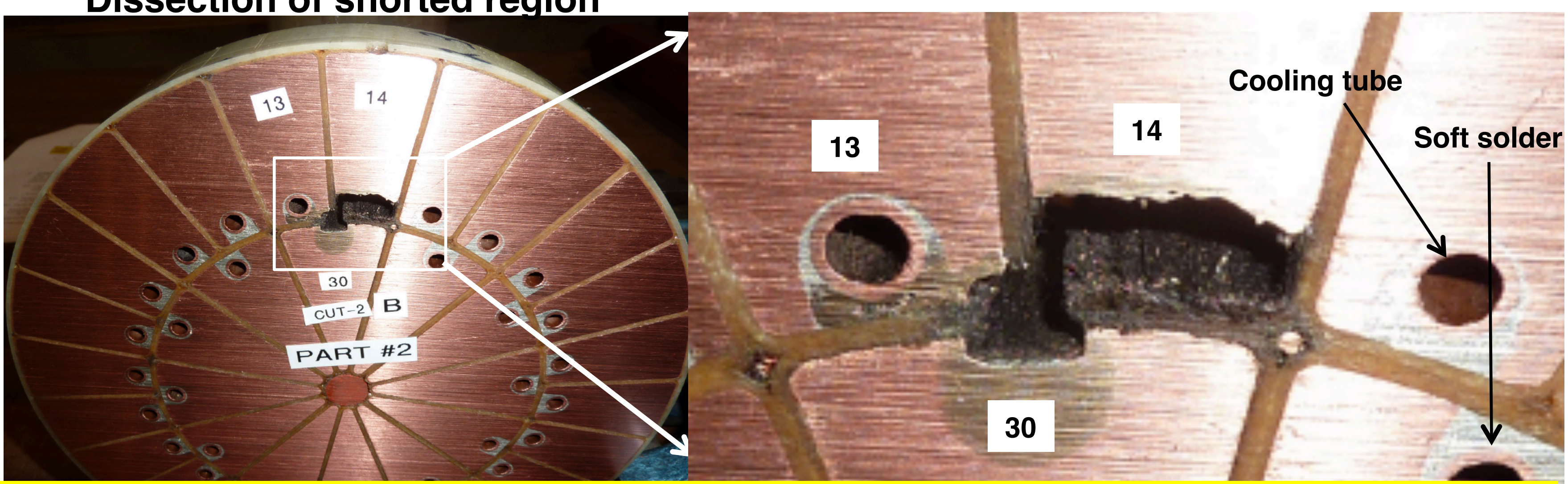
NSTX TF Fault Occurred on July 20, 2011

TF Bundle Operated for 7+ years for 20,000 shots

- TF bundle short occurred ~ 2 feet from the bottom in a relatively low mechanical stress area.
- TF bundle dissection and analyses showed no sign of fatigue.
- **Zinc chloride based flux** used for cooling water tube soldering **was the cause** of insulation failure.

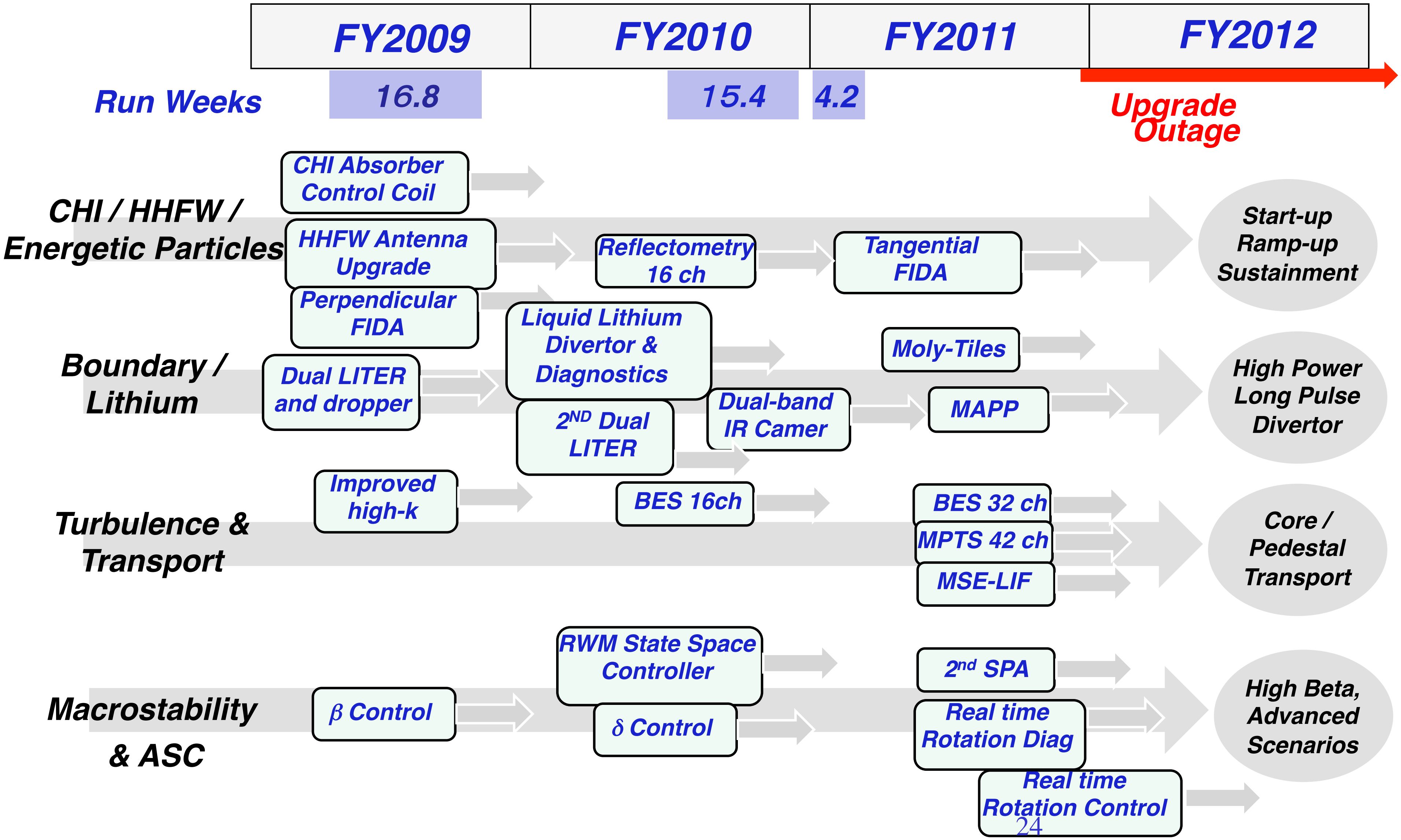


Dissection of shorted region



TF Upgrade to use "Rosin" flux and change the procedures for removing the flux residues

NSTX Facility Overview
To Support NSTX Mission Elements and Upgrades

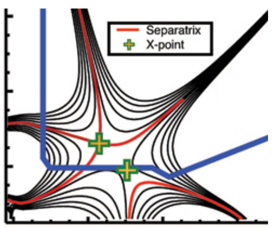


NSTX Upgrade Outage Started

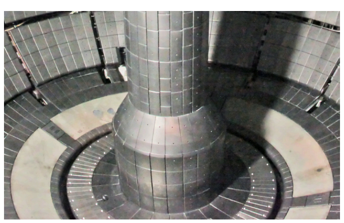
Due to TF Fault, Upgrade was Accelerated by ~ 6 Months

2009	2010	2011	2012	2013	2014	2015	2016	2017	2018
1 MA Plasma			Upgrade Outage		2 MA Plasma				

- HHFW Upgrade
- LLD
- Moly-tile
- CHI Control Coils
- New Center-Stack
- 2nd NBI
- 1 MW ECH/EBW
- Long–pulse Divertor
- 0.5 MA CHI
- 0.5 MA Plasma Gun
- 1 MA CHI / Plasma Gun
- NCC Upgrade

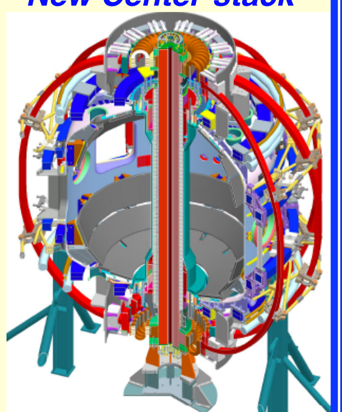


"Snowflake"

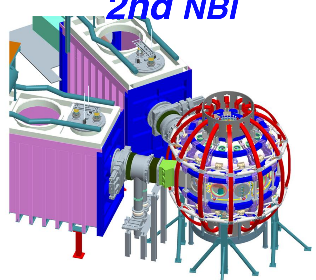


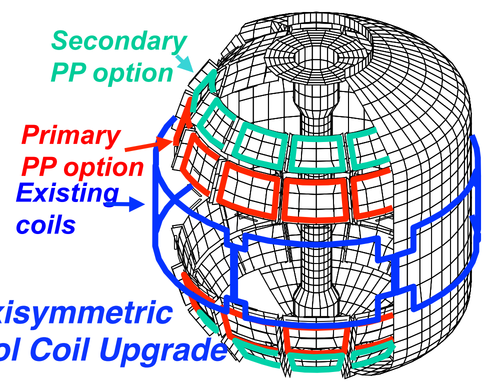
Lithium

New Center-stack



2nd NBI





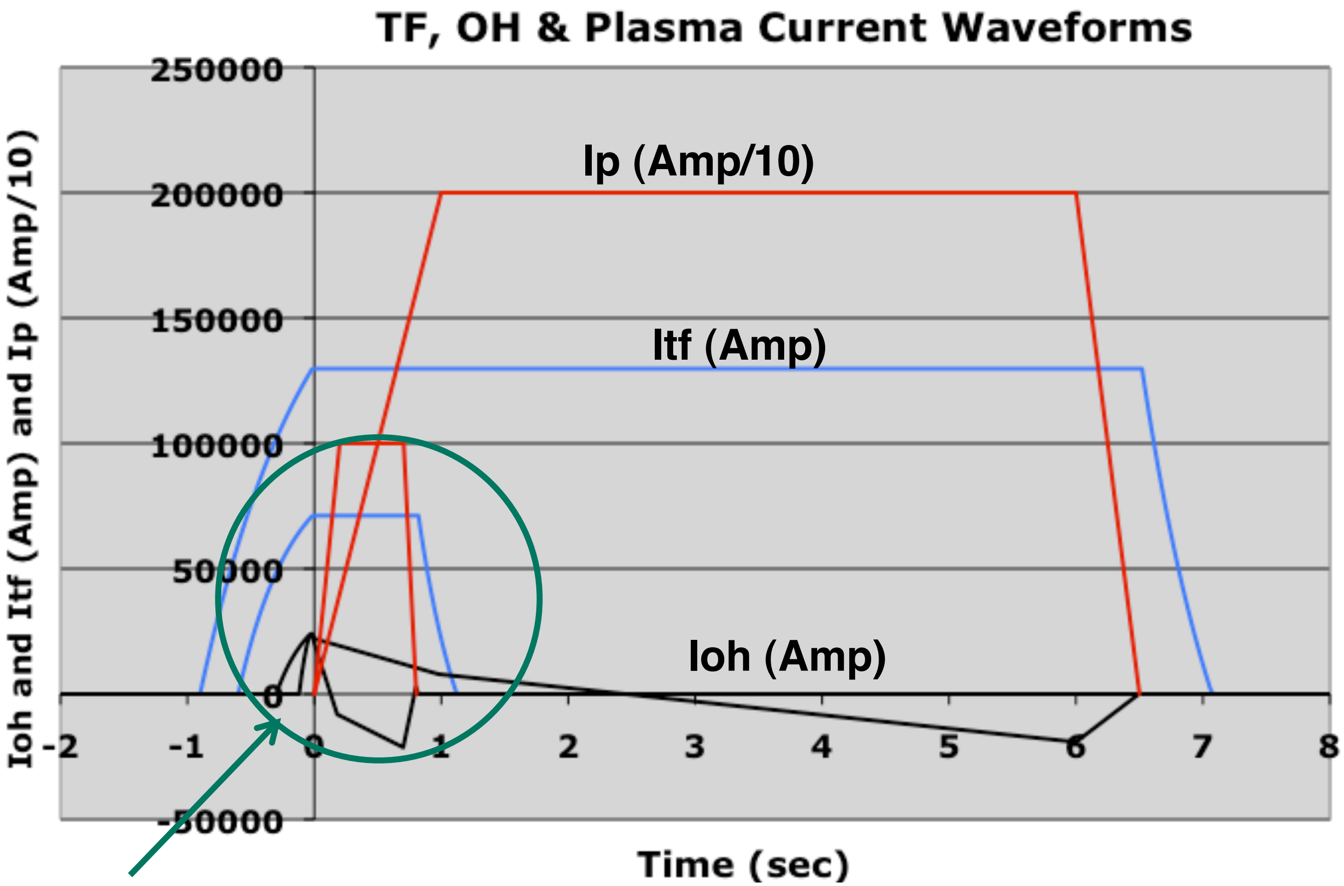
Nonaxisymmetric Control Coil Upgrade

NSTX Research Goals

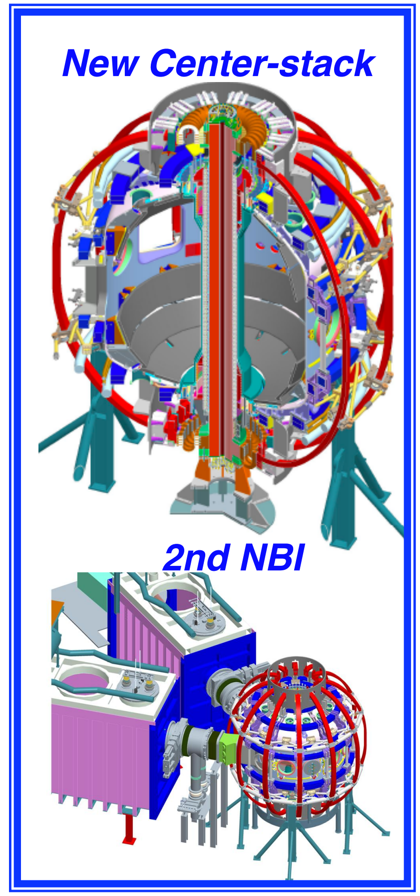
- Low Collisionality Regimes
- Fully Non-Inductive Operations
- Long-pulse High Power Divertor
- High Beta Advanced Scenarios

Upgrade Substantial Increases B_T, I_p, τ_{pulse}, P_{NBI}

Higher B_T and Ip narrows gaps to Fusion Neutron Science Facility



Present NSTX



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)

Center Stack Upgrade and Related Enhancements

Detailed Design, Analyses, and R&D Are Now Well Advanced

Since B and J increase x 2, the E&M forces increase x 4

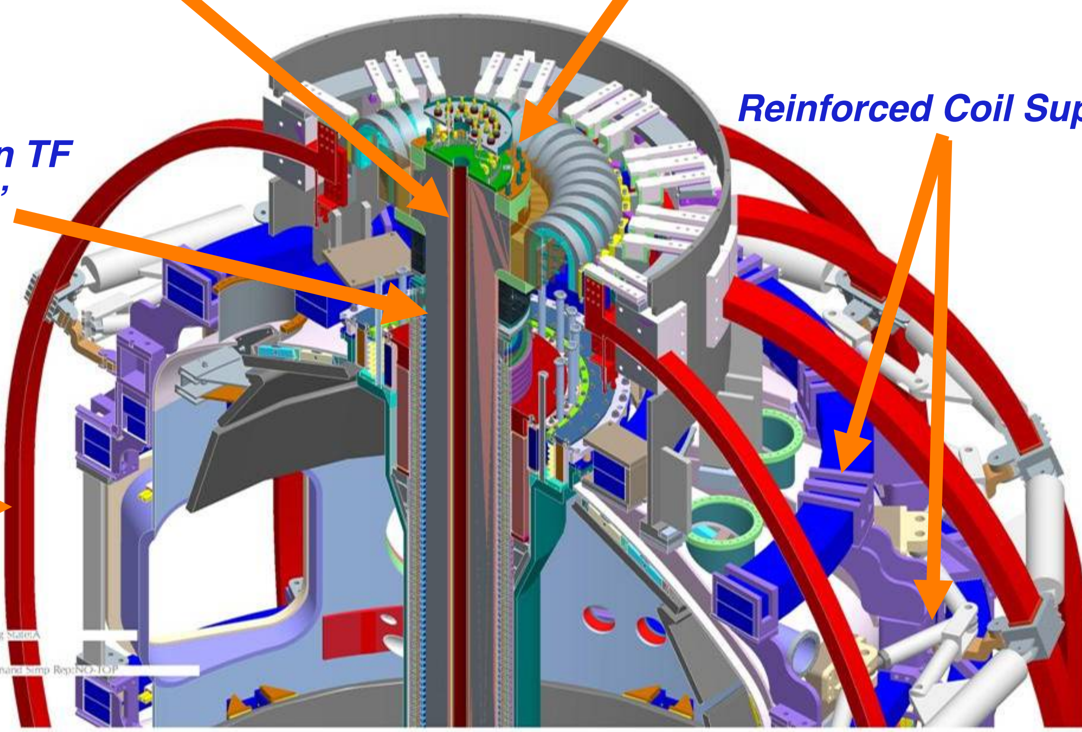
Simpler Inner TF design (single layer of TF conductors)

Improved Joint Design

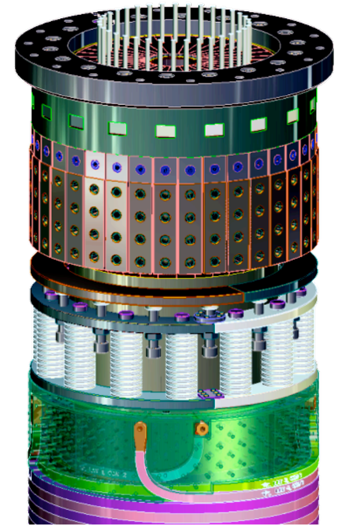
OH coil wound on TF (with 0.1" gap)

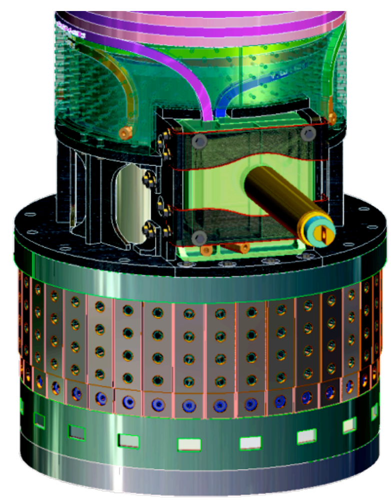
Reinforced Coil Supports

Existing outer TF WITH water cooling



Upper TF/ OH Ends





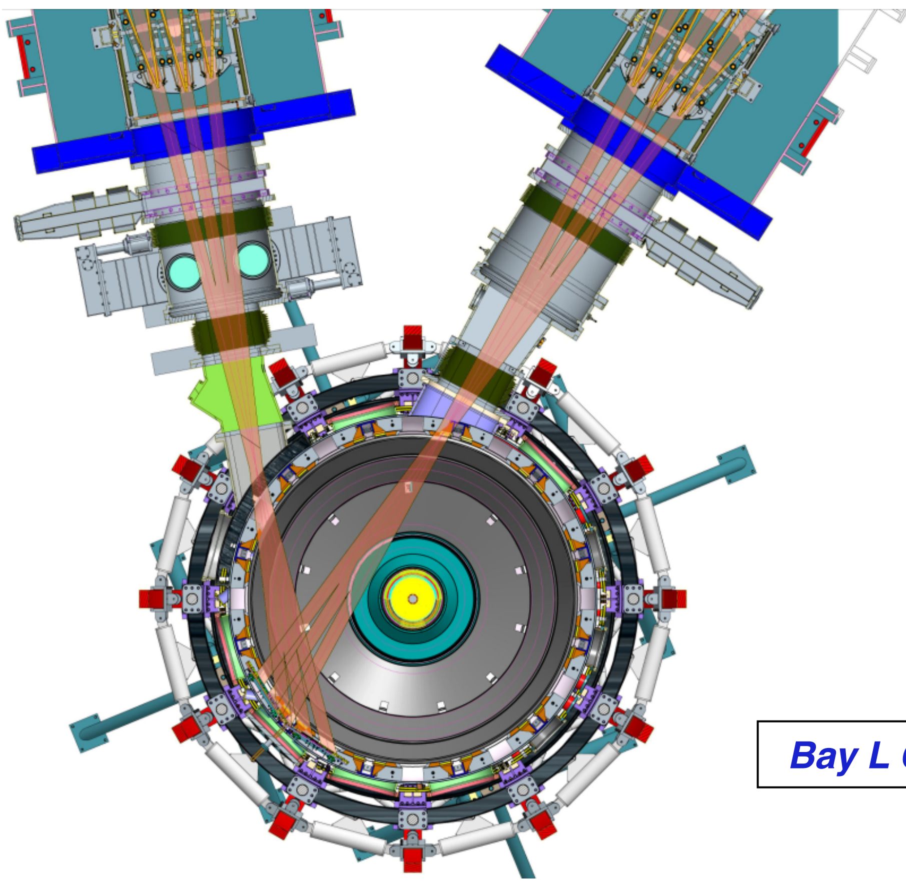
Lower TF/ OH Ends

2^{nd} NBI requires relocation of a TFTR NBI system to NSTX and relocation of NSTX diagnostics from Bay K to Bay L

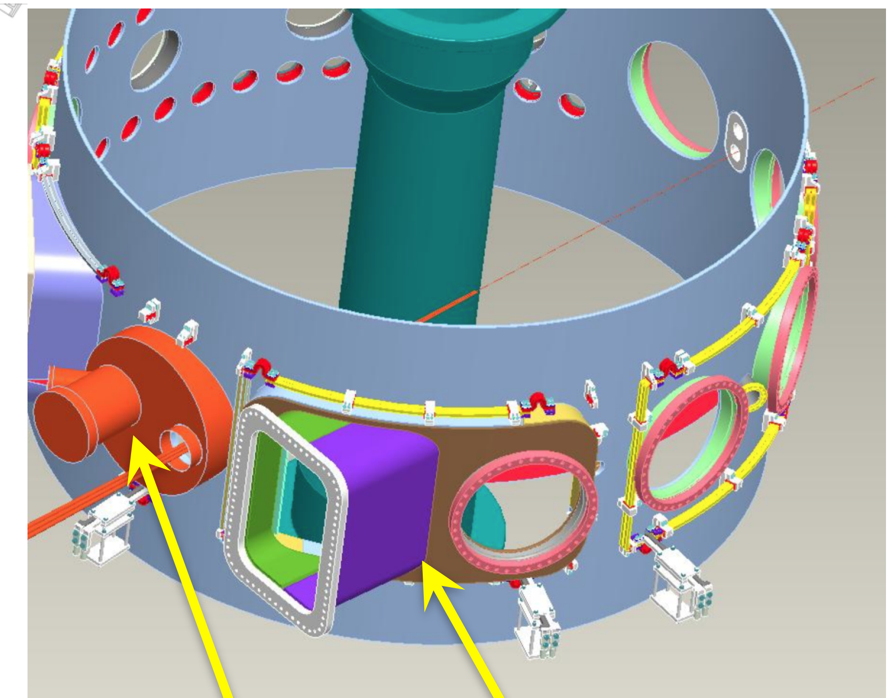
- Decontamination of 2^{nd} Beam line Successfully Completed in 2010
- Reassembly of the 2^{nd} Beam line has started

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

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



Bay L Diagnostics & Bay K Weldment



Bay L CHERS, MPTS, & others

Bay K-J weldment installed for NBI BL2

Summary

- NSTX Program aims to contribute to near term and longer term fusion challenges (FNSF, PMI, ITER, Pilot Plant…)
- NSTX made significant progress in all science areas
- Liquid Lithium Divertor System successfully operated (2010-2011)
  - LL continued to show the benefit of lithium coating
  - LL reduced carbon influx and core accumulation
  - An intriguing LL phase transition behavior observed
  - Lithium core fraction remain very low
  - Lithium improved the NSTX operation efficiency by ~ 50%
- "Snow-flake" divertor demonstrated to significantly reduce divertor heat flux
- HHFW demonstrated at low current for current ramp-up
- TF-fault occurred on July 20, 2011 due to degraded insulation from flux used in soft solder of cooling water
- Due to TF fault, Upgrade Project accelerated by 6 months – first plasma planned in March 2014

We look forward to enhanced collaborations in the future!!

Other NSTX Presentations at STW-2011

- J. Menard (PPPL), "NSTX Research Progress towards NSTX Upgrade and Next-Step STs", 29-1-1i
- R. Raman (University of Washington), "Demonstration of Tokamak Inductive Flux Saving by Transient Coaxial Helicity Injection on NSTX", 27-3-1
- K. Tritz (Johns Hopkins University), "Global Alfven Eigenmodes Induced Electron Thermal Transport in NSTX", 29-2-1
- R. Maingi (ORNL), "Energy confinement enhancement and pedestal growth triggered by an ELM in NSTX", 29-1-3
- D. Gates (PPPL), "Advanced Scenario and Control Development on NSTX", 30-1-3

POSTERS:

- R. Raman (University of Washington), "Disruption Mitigation Studies in NSTX"
- K. Tritz (Johns Hopkins University), "Multi-Energy Soft X-ray Array Diagnostic for Electron Temperature and Impurity Measurements on NSTX"
- V. Soukhanovskii (LLNL), "Plasma-Material Interface Development for Future Spherical Tokamak based Devices in NSTX"
- Y. Ren (PPPL), "Recent progress in transport and turbulence research at NSTX"