National Spherical Torus Experiment Upgrade – Status and Plans*

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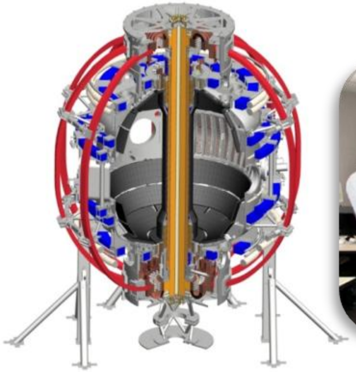
J. Menard, PPPL

For the NSTX-U Team

ANS 20th Topical Meeting on the
Technology of Fusion Energy (TOFE-2012)
Nashville, TN USA
August 27-31, 2012



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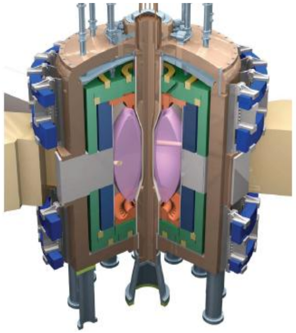
Outline

- NSTX-U Mission
- Planned NSTX Upgrade Capabilities
- Progress of Upgrade Project
- Summary

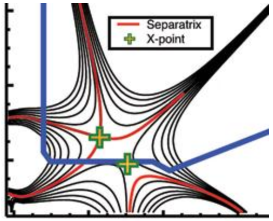
NSTX Upgrade Mission Elements

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



ST-FNSF

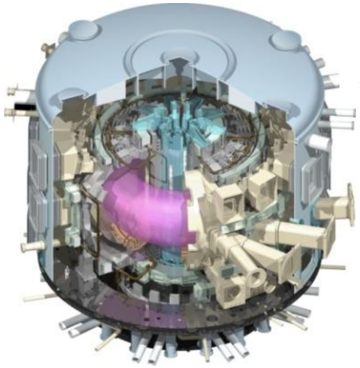
- **Develop solutions for plasma-material interface**



"Snowflake"

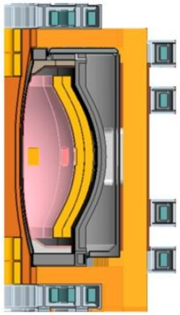
Lithium

- **Advance toroidal confinement physics predictive capability for ITER and beyond**



ITER

- **Develop ST as fusion energy system**

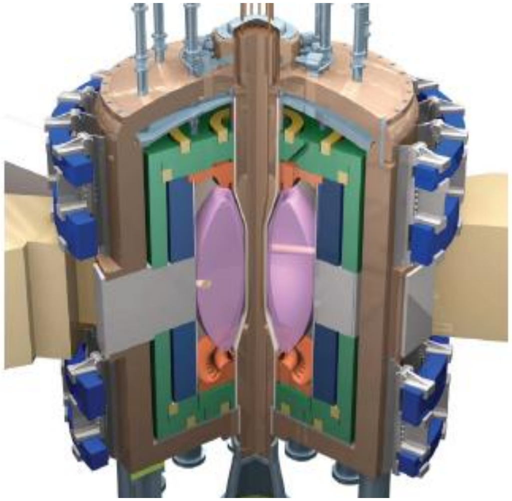


ST Pilot Plant

Mission of ST-FNSF

From M. Peng, ORNL

- **Provide a continuous fusion nuclear environment of copious neutrons to develop an experimental database on:**
  - Nuclear-nonnuclear coupling phenomena in materials in components for plasma-material interactions
  - Tritium fuel cycle
  - Power extraction



ST-FNSF

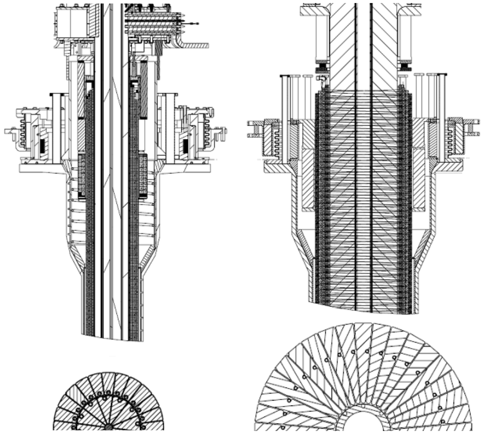
- **Complement ITER, prepare for component test facility (CTF):**
  - Low Q (≤ 3): 0.3 x ITER
  - Neutron flux ≤ 2 MW/m^2: 3 x
  - Fluence = 1 MW-yr/m^2: 5 x
  - t_{pulse} ≤ 2 wks: 1000 x
  - Duty factor = 10%: 3 x

Low-aspect-ratio “spherical” tokamak (ST) is most compact embodiment of FNSF

NSTX Upgrade will address critical plasma confinement and sustainment questions by exploiting 2 new capabilities

Previous center-stack

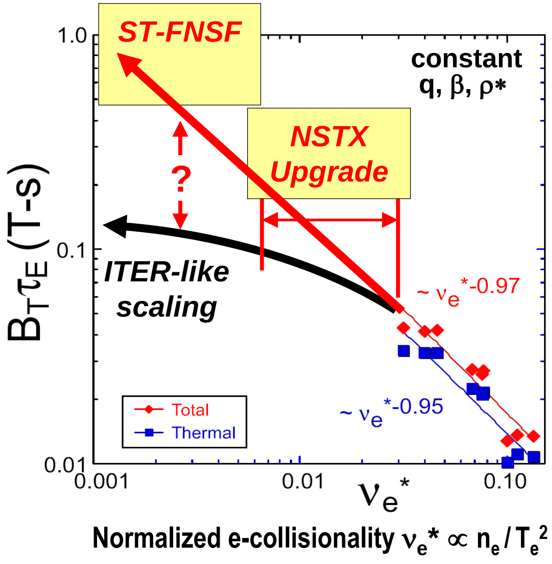
New center-stack

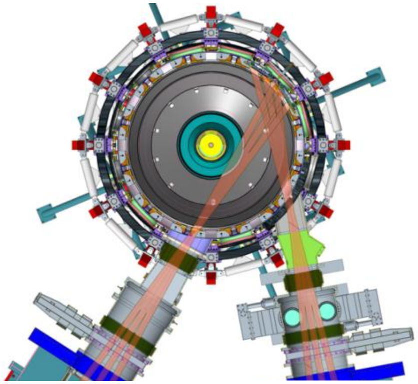


TF OD = 20cm

TF OD = 40cm

- **Higher B_T and I_P increases T, reduces ν^* toward ST-FNSF to better understand confinement**
- **Provides 5x longer pulses for profile equilibration, NBI ramp-up**

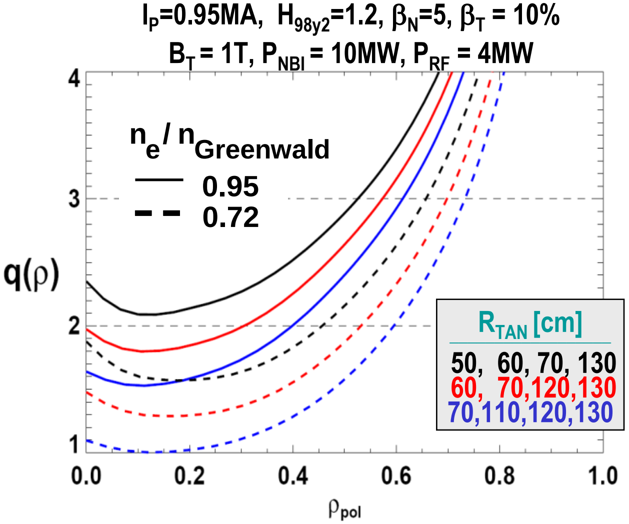




Present NBI

New 2^{nd} NBI

- **2x higher CD efficiency from larger tangency radius R_{TAN}**
- **100% non-inductive CD with q(r) profile controllable by: tangency radius, density, position**

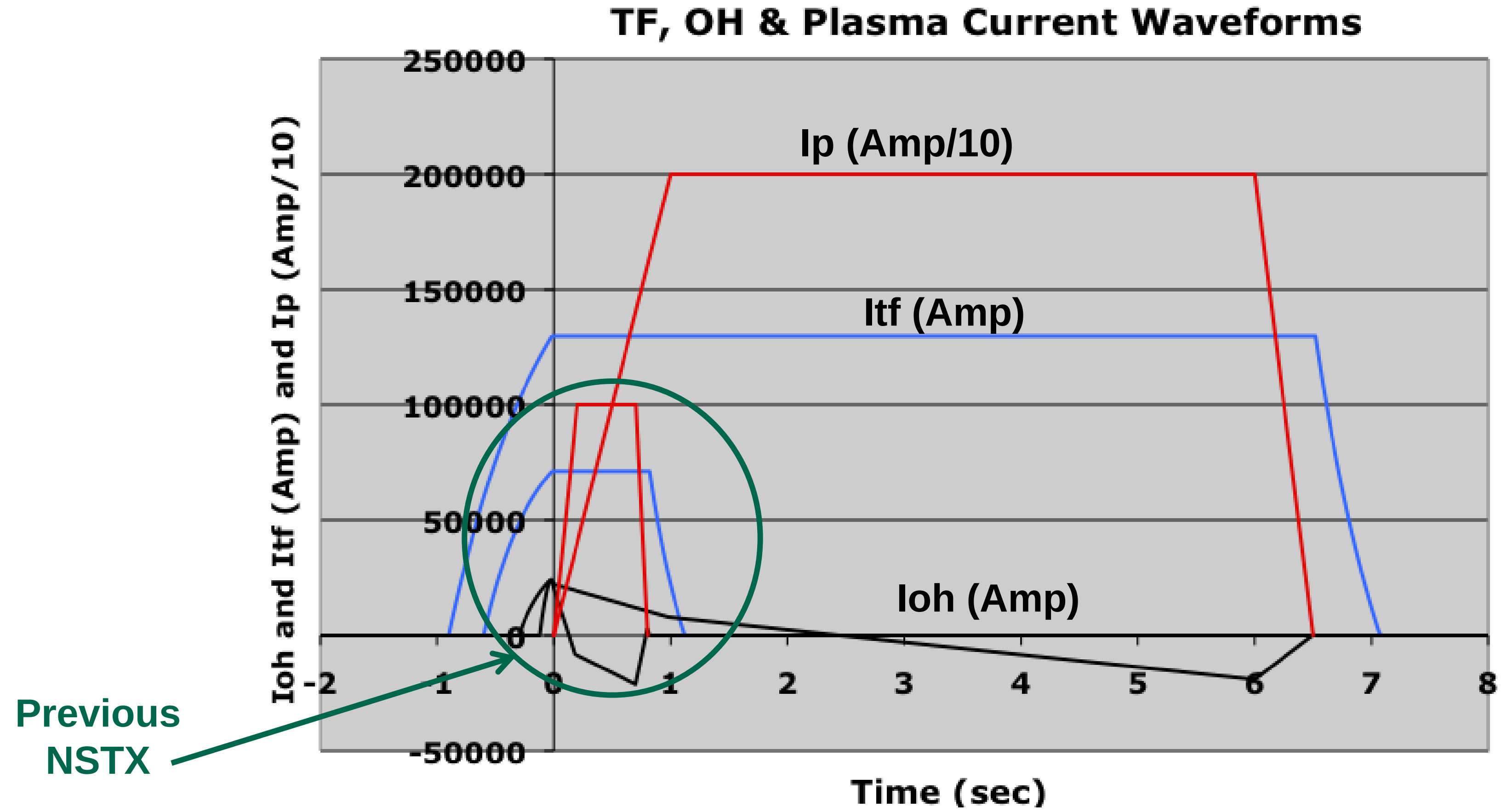


Upgrade substantially increases B_T, I_p, P_{NBI}, τ_{pulse}

Field and current will be within factor of 2 of initial operation of ST-FNSF

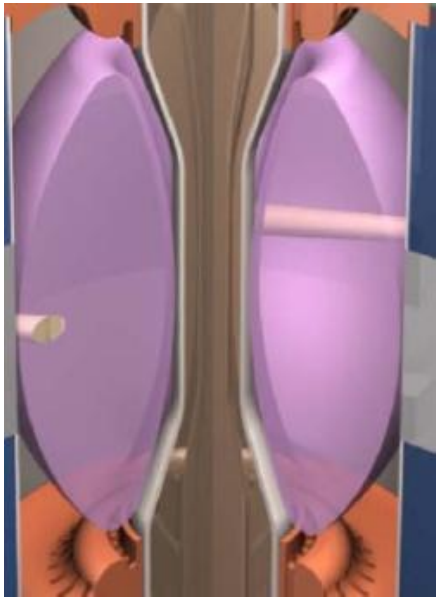
Relative performance of Upgraded NSTX vs. Base:

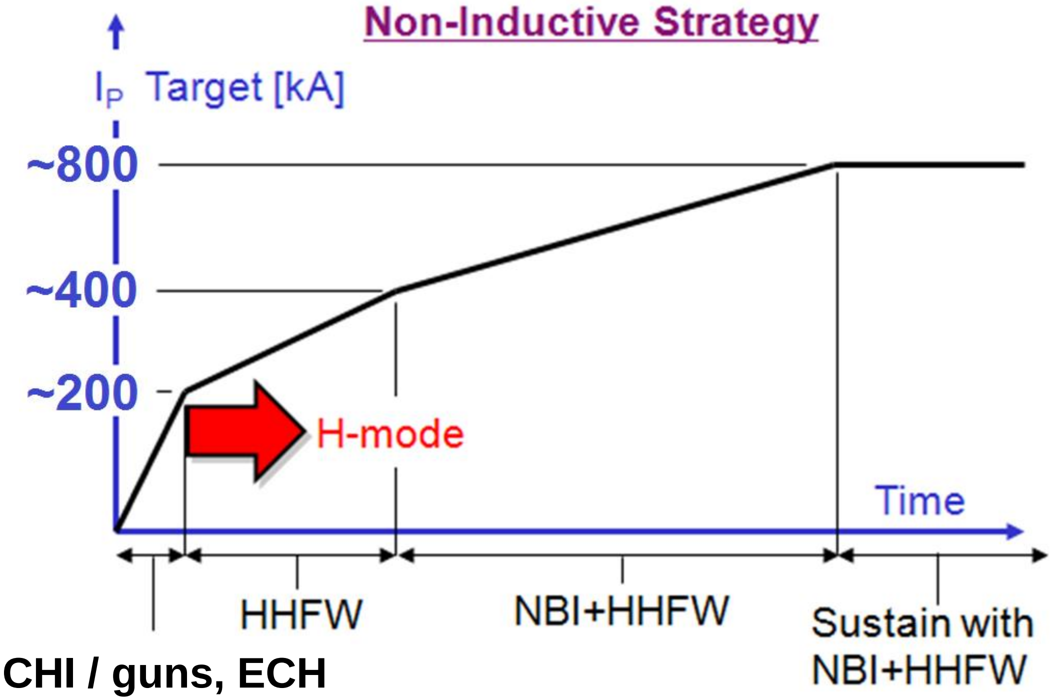
- I_P = 1 → 2MA, B_T = 0.5 → 1T (at same major radius)
- Available OH flux increased 3x, 3-5x longer flat-top
- NBI power increased 2x (5→10MW for 5s, 15MW 1.5s)
- Plasma stored energy increased up to 4x (0.25→ 1MJ)



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

ST-FNSF has no/small central solenoid





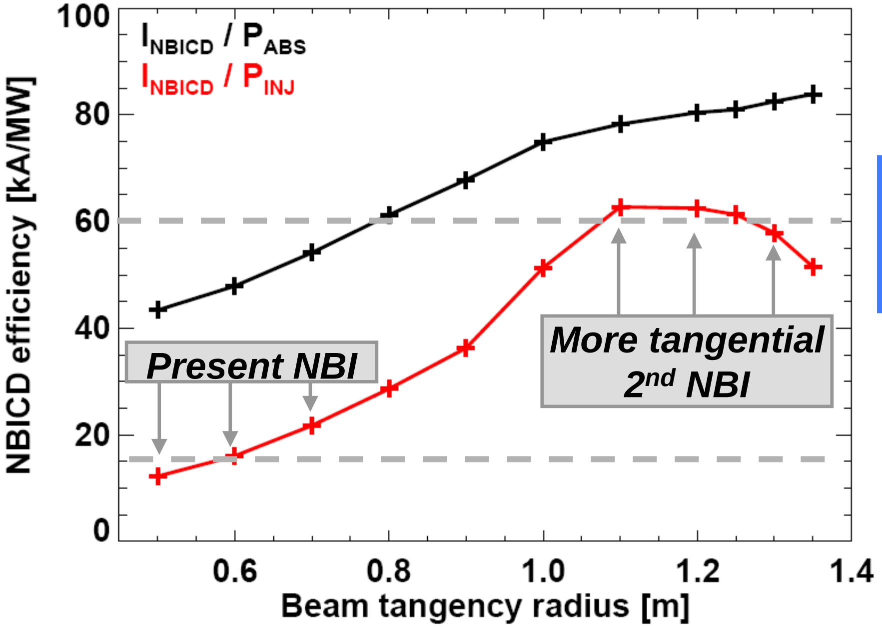
- **NSTX-U goals:**
  - Generate ~0.3-0.4MA full non-inductive start-up with helicity injection + ECH and/or fast wave heating, then ramp to ~0.8-1MA with NBI
  - Develop predictive capability for non-inductive ramp-up to high performance 100% non-inductive ST plasma → prototype FNSF

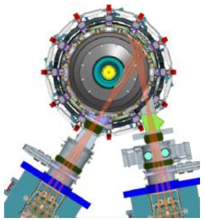
Non-inductive ramp-up from ~0.4MA to ~1MA projected to be possible with new centerstack (CS) + more tangential 2nd NBI

- **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

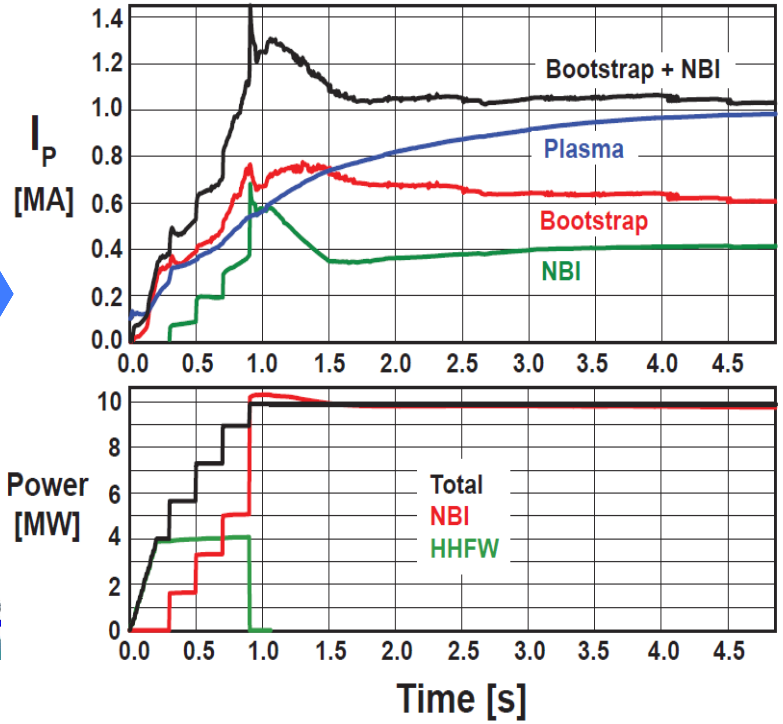
E_{NBI}=100keV, I_P=0.40MA, f_{GW}=0.62

$\overline{n}_e = 2.5\times10^{19}$m^{-3}, $\overline{T}_e$ = 0.83keV

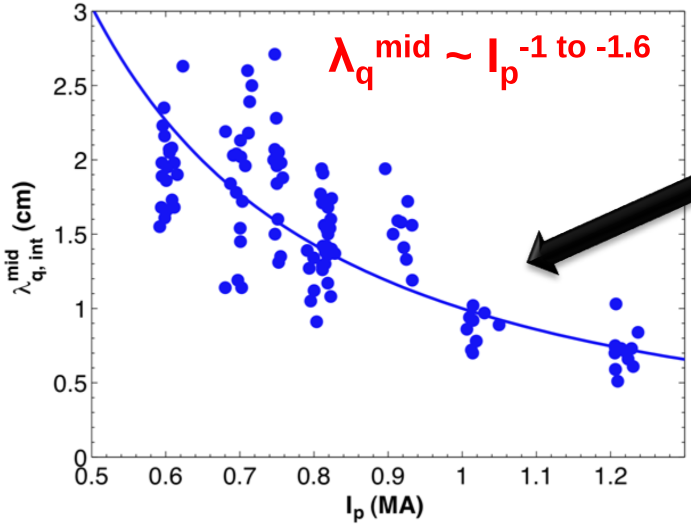




TSC simulation of non-inductive ramp-up from I_P = 0.1MA, T_e=0.5keV target at B_T=1T



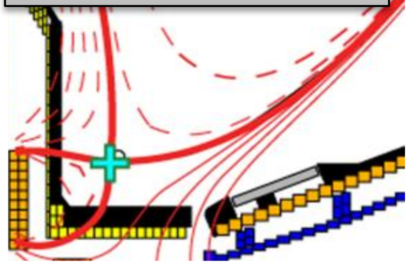
NSTX-U will investigate high flux expansion snowflake divertor + detachment for large heat-flux reduction



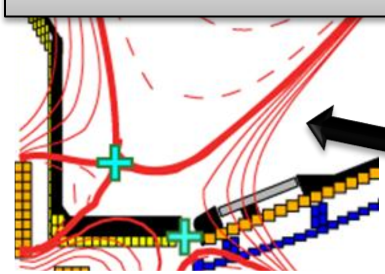
- Divertor heat flux width decreases with increased plasma current I_P
  - Major implications for ITER, FNSF

→ **NSTX Upgrade with conventional divertor projects to very high peak heat flux up to 30-45MW/m^2**

Standard Divertor



Snowflake Divertor



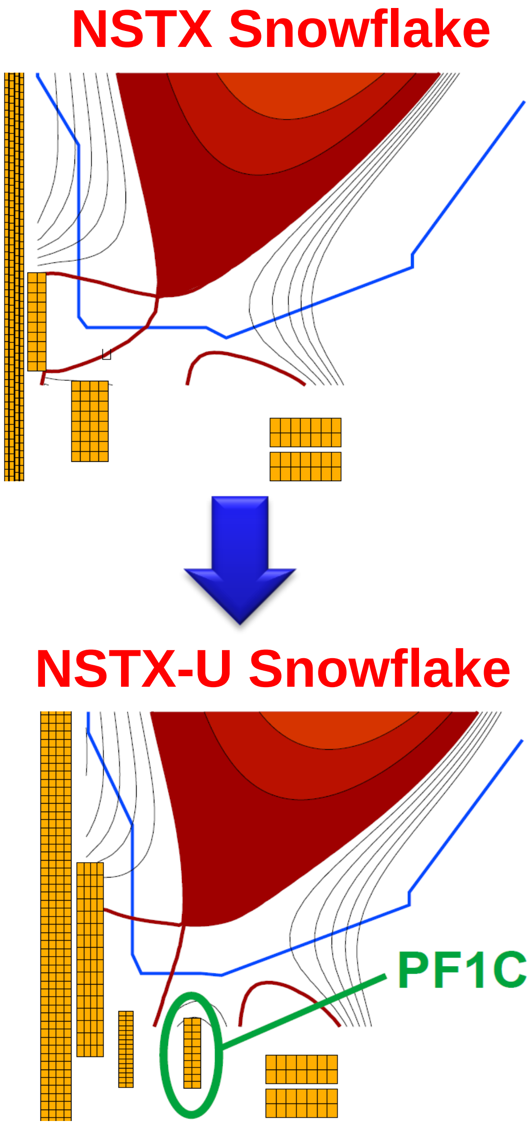
- Divertor heat flux inversely proportional to flux expansion over a factor of five

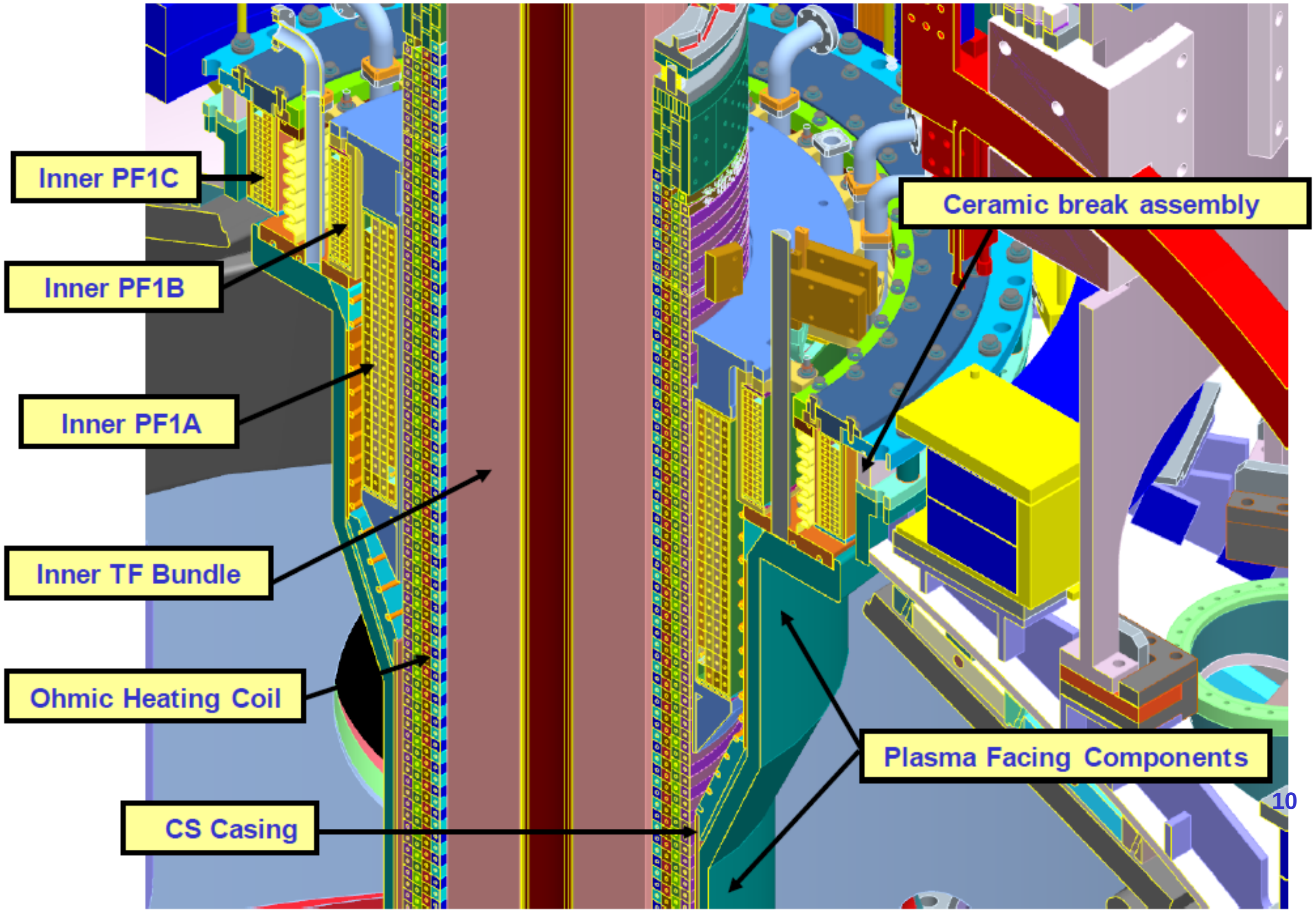
  Snowflake→ high flux expansion 40-60, larger divertor volume and radiation

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

→Partial detachment → Additional ~2x reduction in NSTX

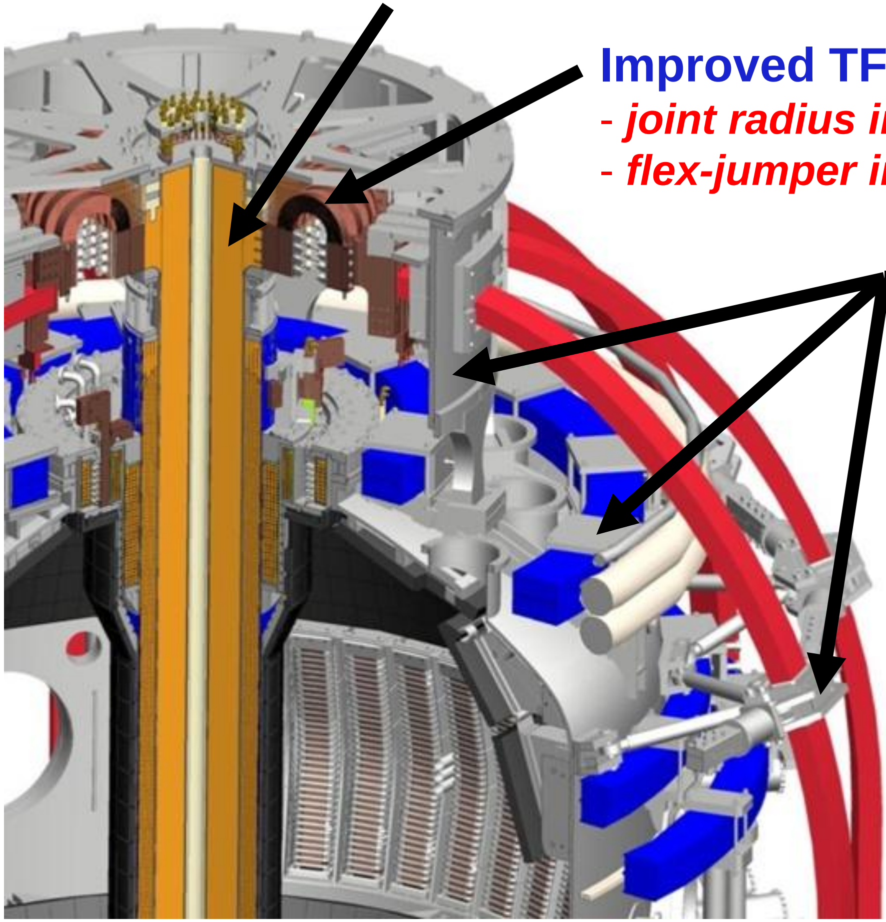
Upgrade CS design provides additional coils for flexible and controllable divertor including snowflake, and supports CHI





NSTX-U centerstack and vacuum vessel analysis/design are complete, component fabrication/installation has begun

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



Simplified inner TF design

- single layer of TF conductors

Improved TF joint design

- *joint radius increased → lower B*
- *flex-jumper improved*

Reinforced umbrella structure and PF and TF coil supports

Upper TF/ OH Ends

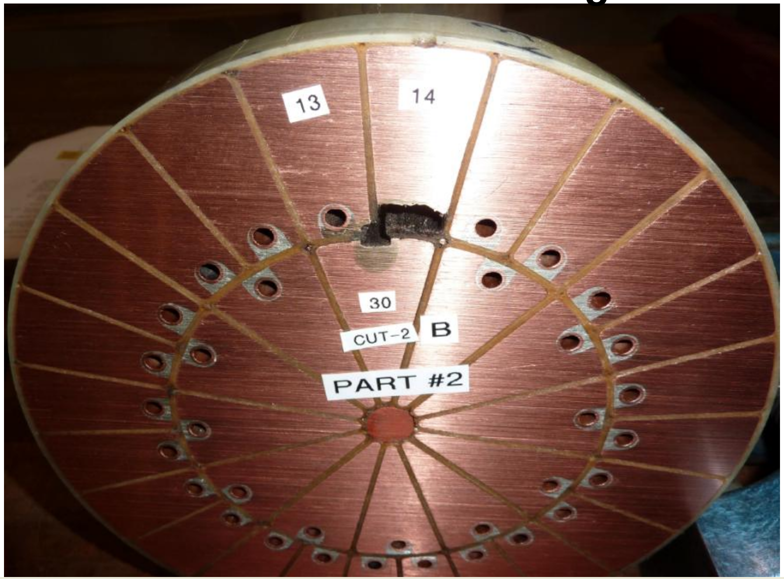
Coaxial and bottom OH lead minimizes error-fields

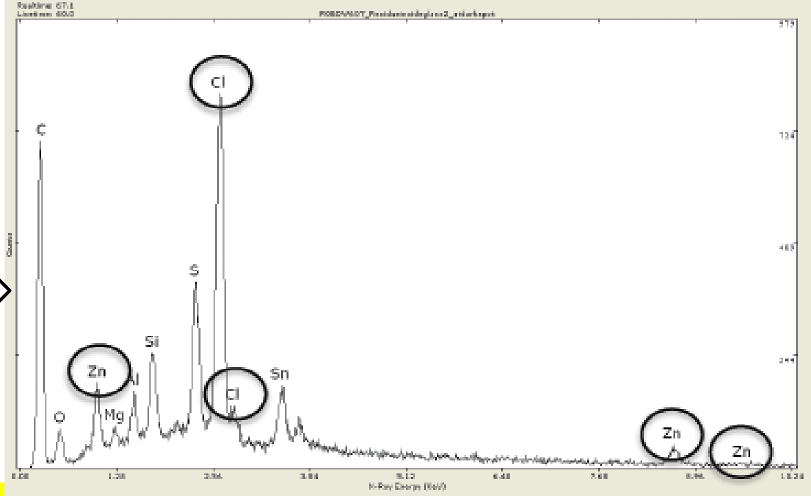
NSTX TF Fault Occurred on July 20, 2011

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

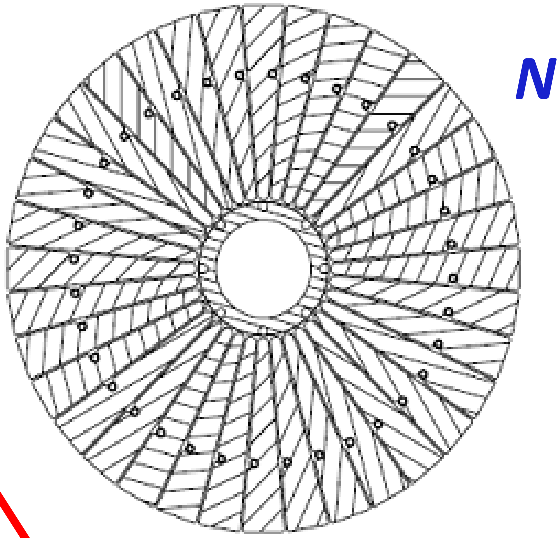


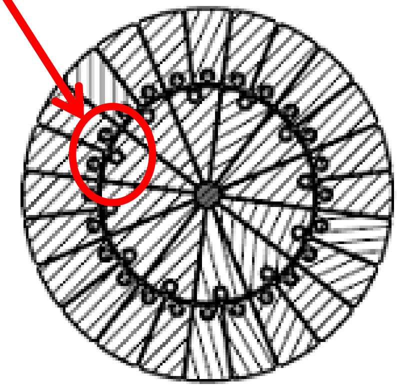


TF Upgrade will use resin flux and improved procedures for removing the flux residues

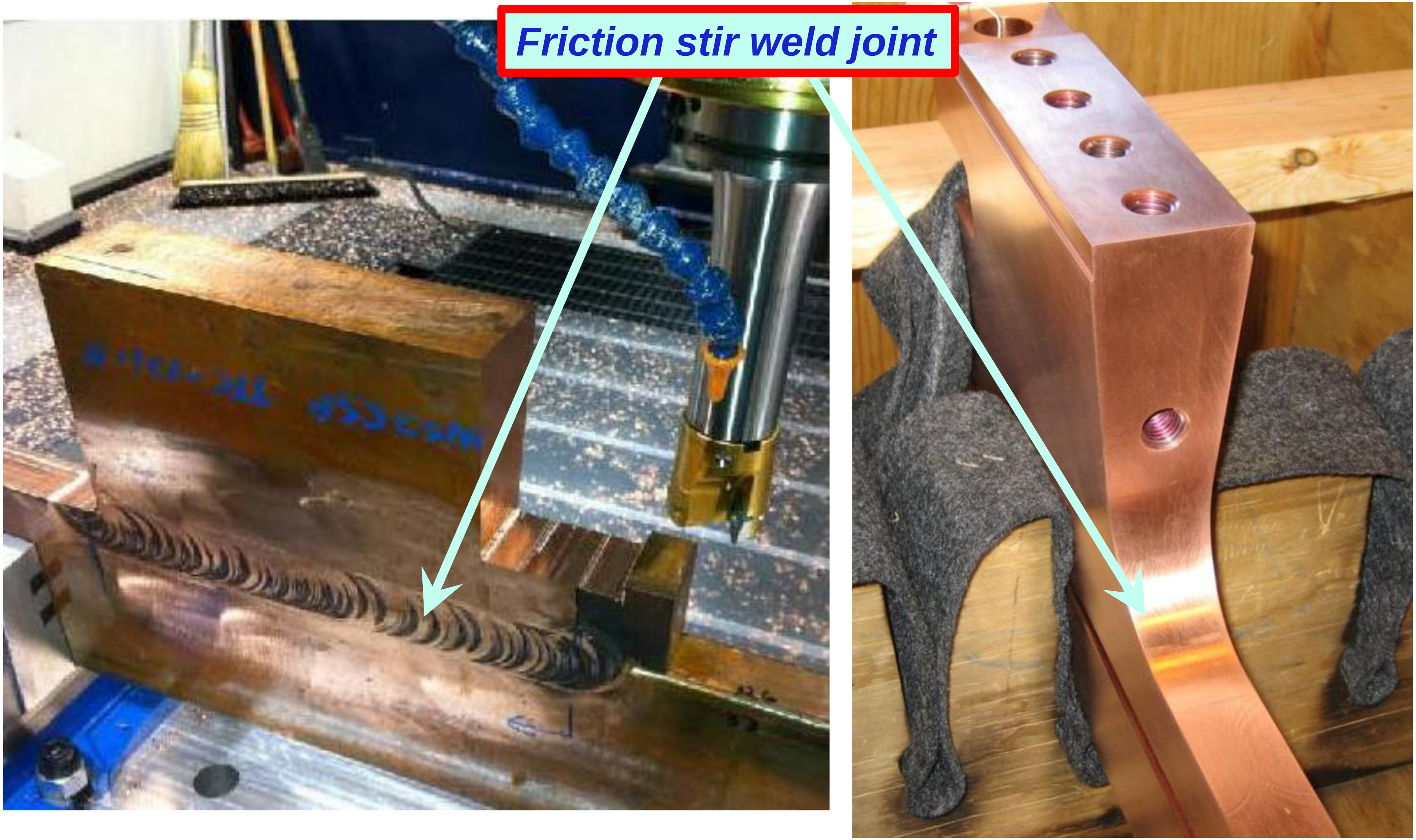
The NSTX-U center-stack design incorporates improvements that address factors contributing to NSTX center-stack failure

- Single-layer vs. double layer design
  - Reduced voltage stress between conductors (30 volts)
  - Terminal voltage (1 kV) is across quadrant segments where there is increased insulation
- VPI vs B-Stage glass resin system
  - More homogenous insulation system without voids
- Bundle manufacturing improved to address residual solder flux
  - Less corrosive flux
  - Post-soldering bakeout



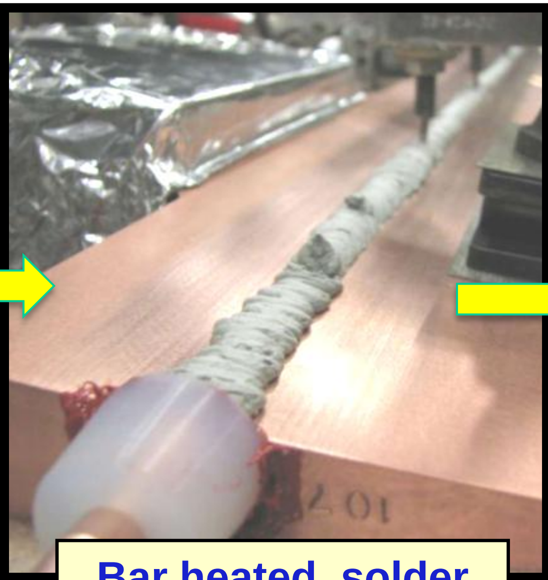


Friction stir welding of TF flags to vertical TF conductors is producing high-quality joints



Improved soldering and flux removal process for TF cooling tubes has also been developed

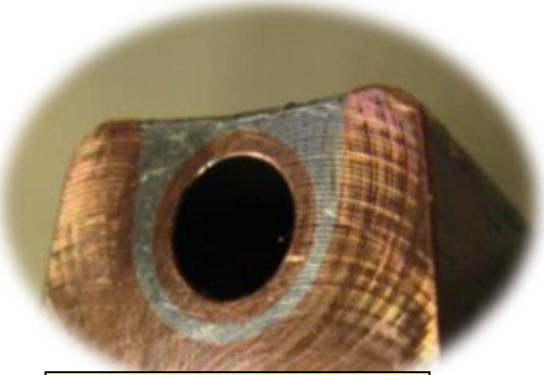
Bar placed on heat plate, cooling tube inserted into grove



Bar heated, solder paste added

Solder flowing. Supplemental heat applied by torch

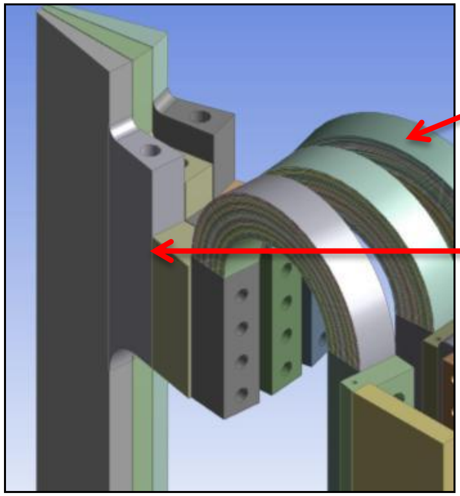
Bar ground smooth



Close up views of solder joint

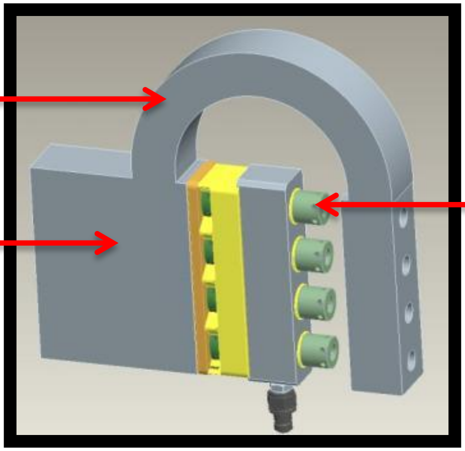
Good wetting of both the tube and copper bar, indicating effectiveness of flux

Features of TF inner/outer flex strap connector

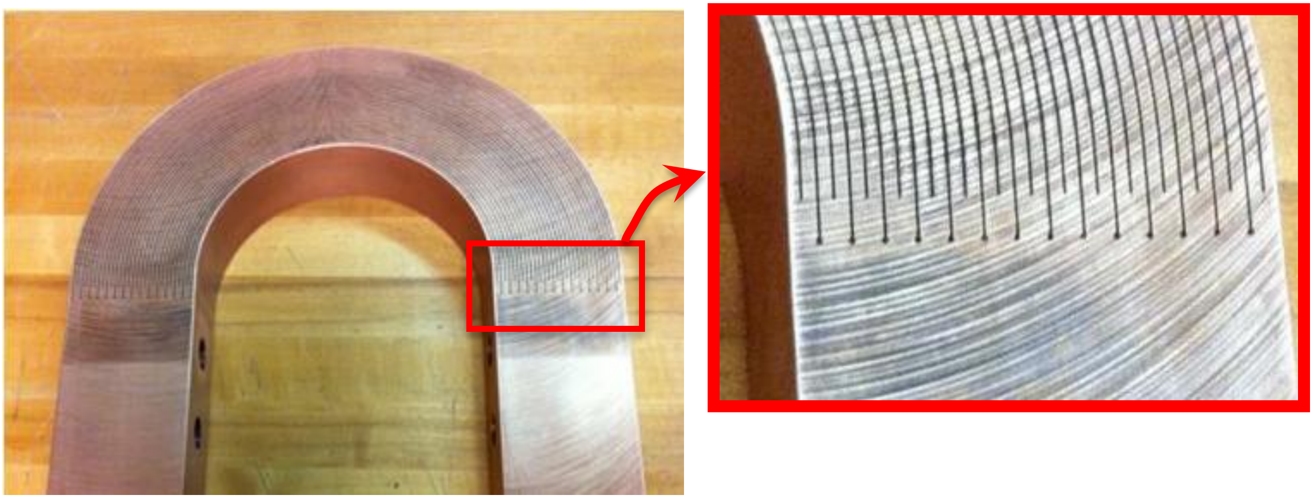


Flex strap

Inner TF



Supernuts® to be used to facilitate assembly



Wire EDM instead of laminated build

Testing (60,000 cycles)

Center-stack fabrication is now underway

Conductor bar being removed from oven after post-solder bake out

Bar being wrapped with insulation

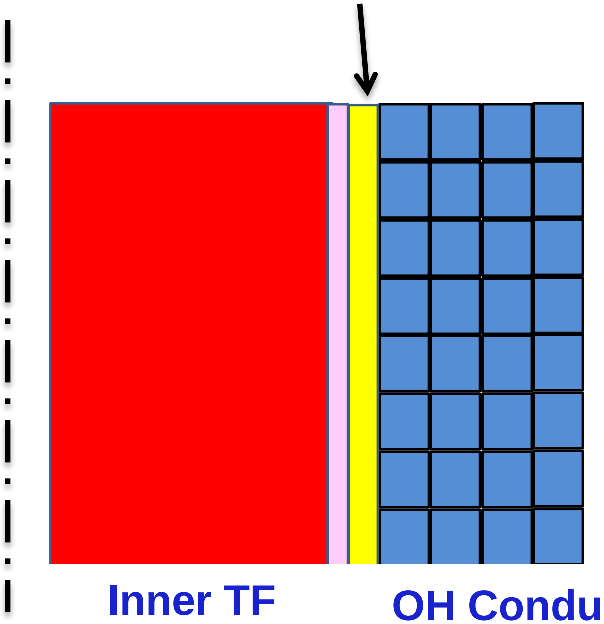
Now entering the riskiest stage of project → inner TF and OH fabrication and VPI – will VPI 1st quadrant in Sept/Oct

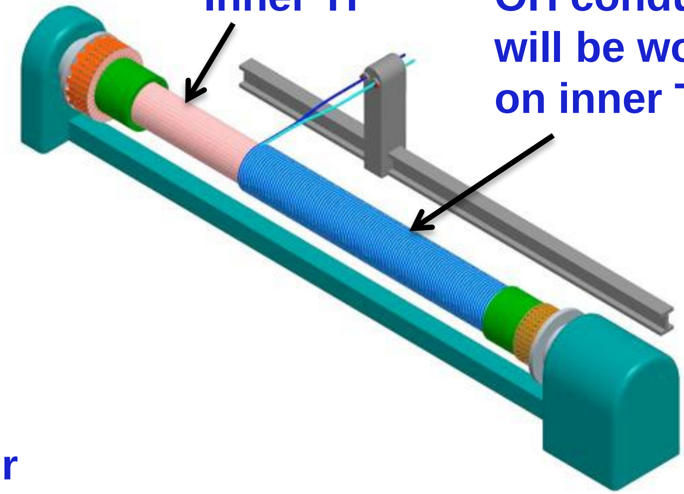
Fabrication techniques for the TF and OH coils

- Epoxy VPI (CTD-425: special cyanate-ester blend) required for shear strength will be used for the inner TF assembly

Recent successful VPI trials

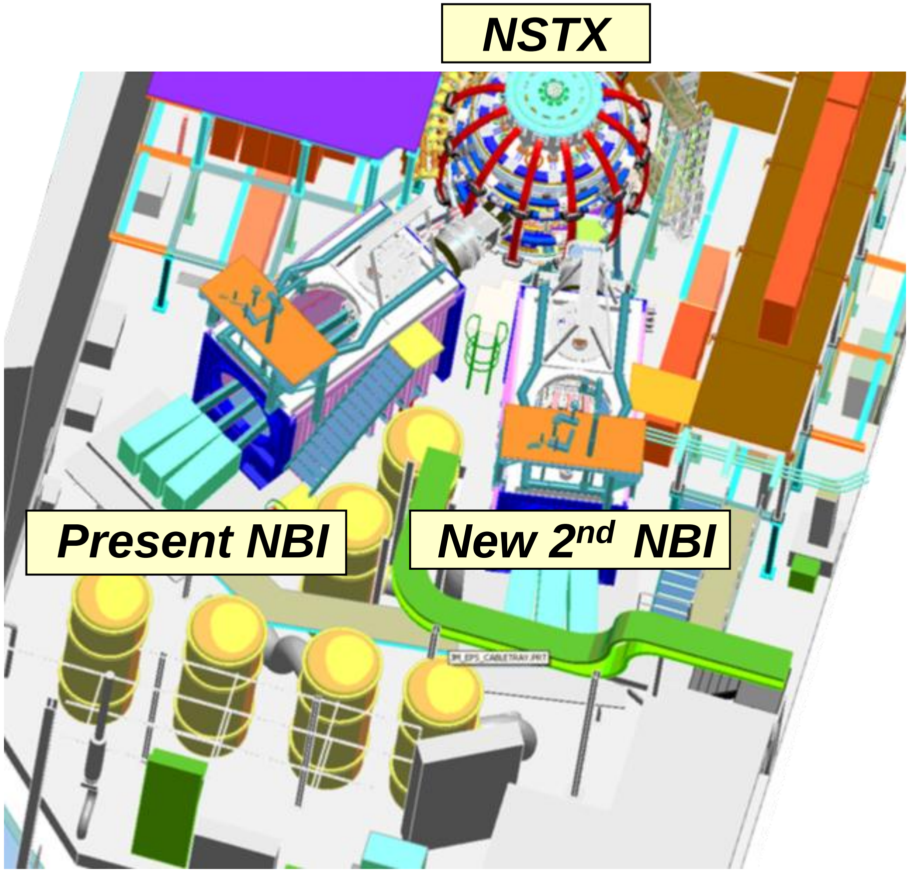
- Aquapour™ will be used as a temporary winding mandrel material to maintain gap between inner TF and OH of 0.1”

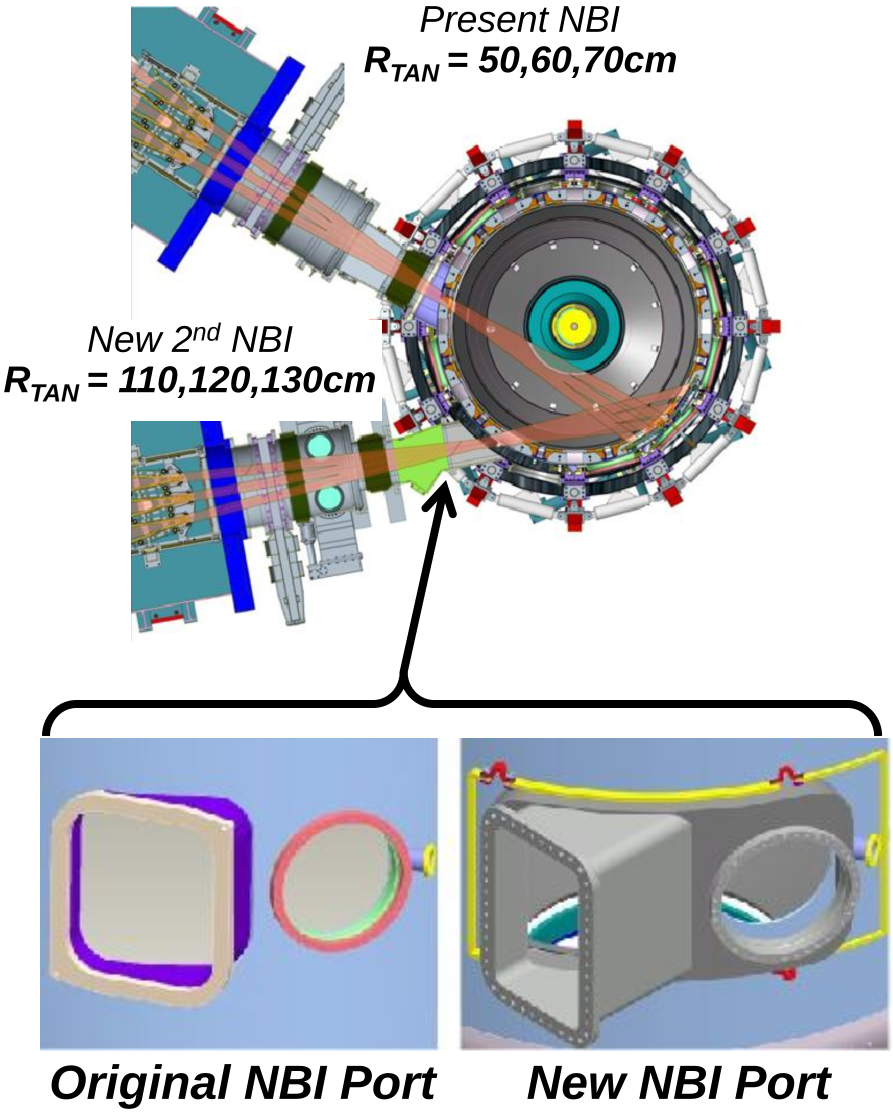




OH conductor will be wound on inner TF

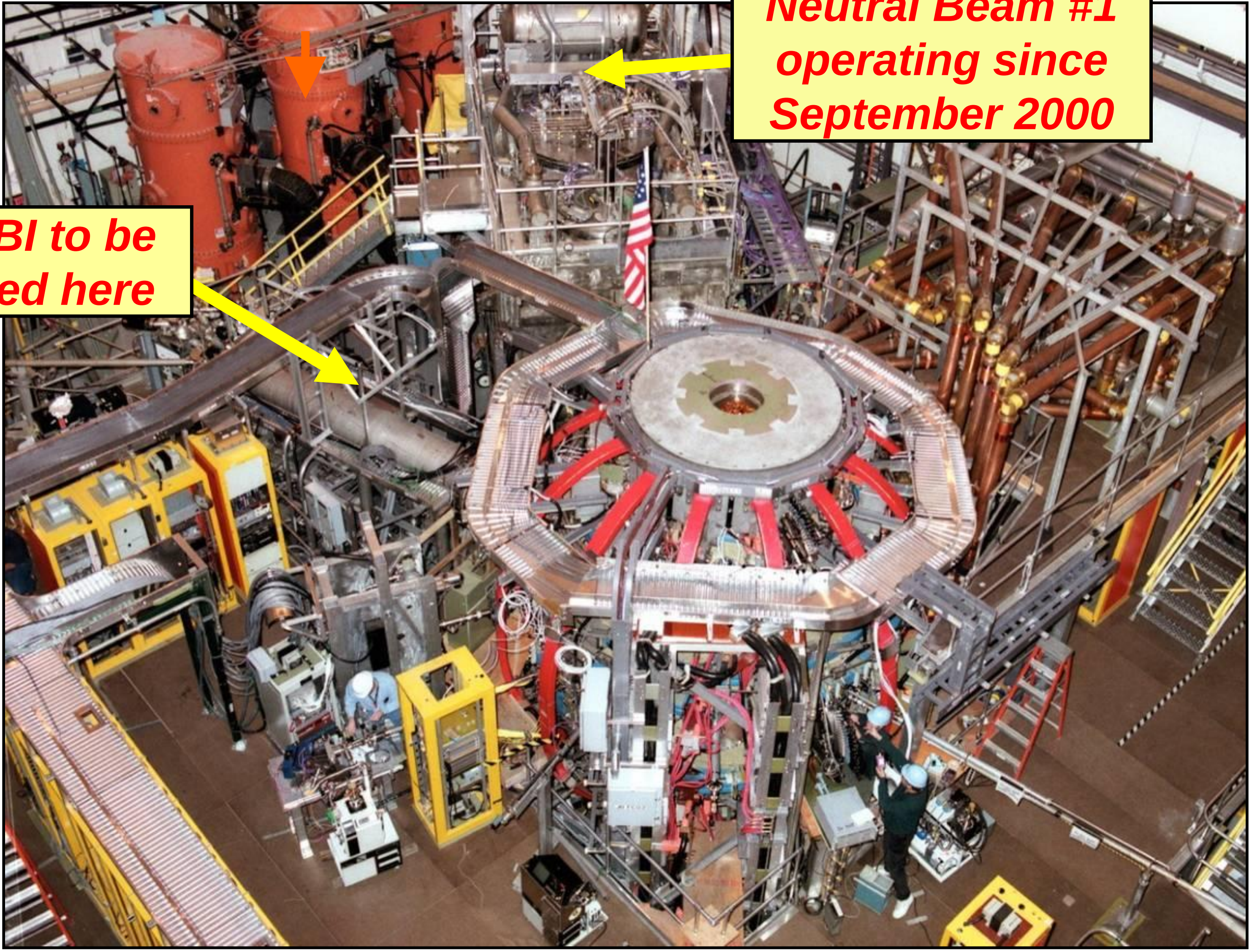
2nd NBI requires relocation of a TFTR NBI system to NSTX, diagnostic relocations, new port for more tangential NBI





- **Decontamination of 2^{nd} Beam line successfully completed in 2010**

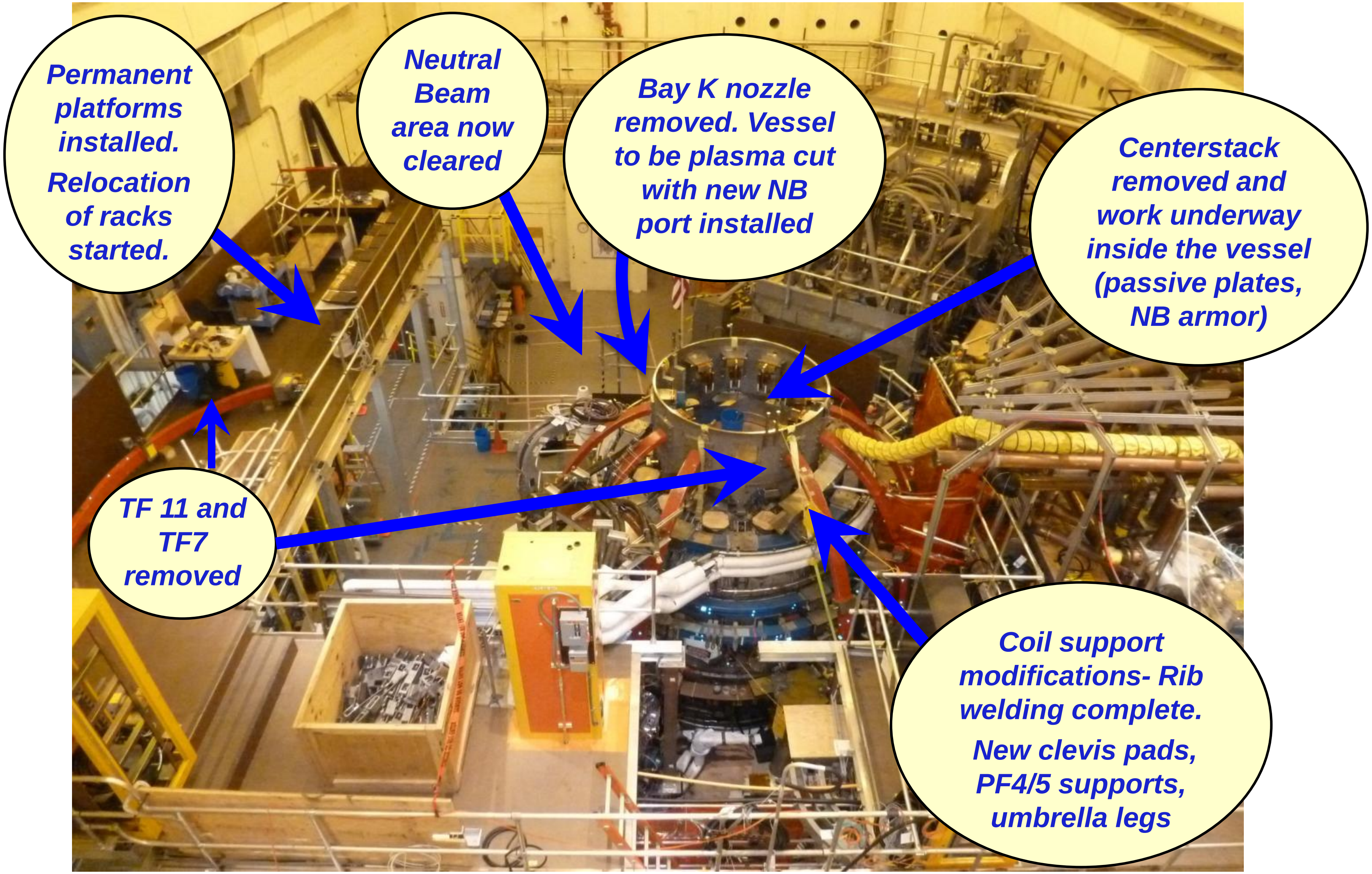
NSTX circa 2010



Neutral Beam #1 operating since September 2000

2nd NBI to be located here

Test-cell progress since September 2011



Permanent platforms installed.

Relocation of racks started.

Neutral Beam area now cleared

Bay K nozzle removed. Vessel to be plasma cut with new NB port installed

Centerstack removed and work underway inside the vessel (passive plates, NB armor)

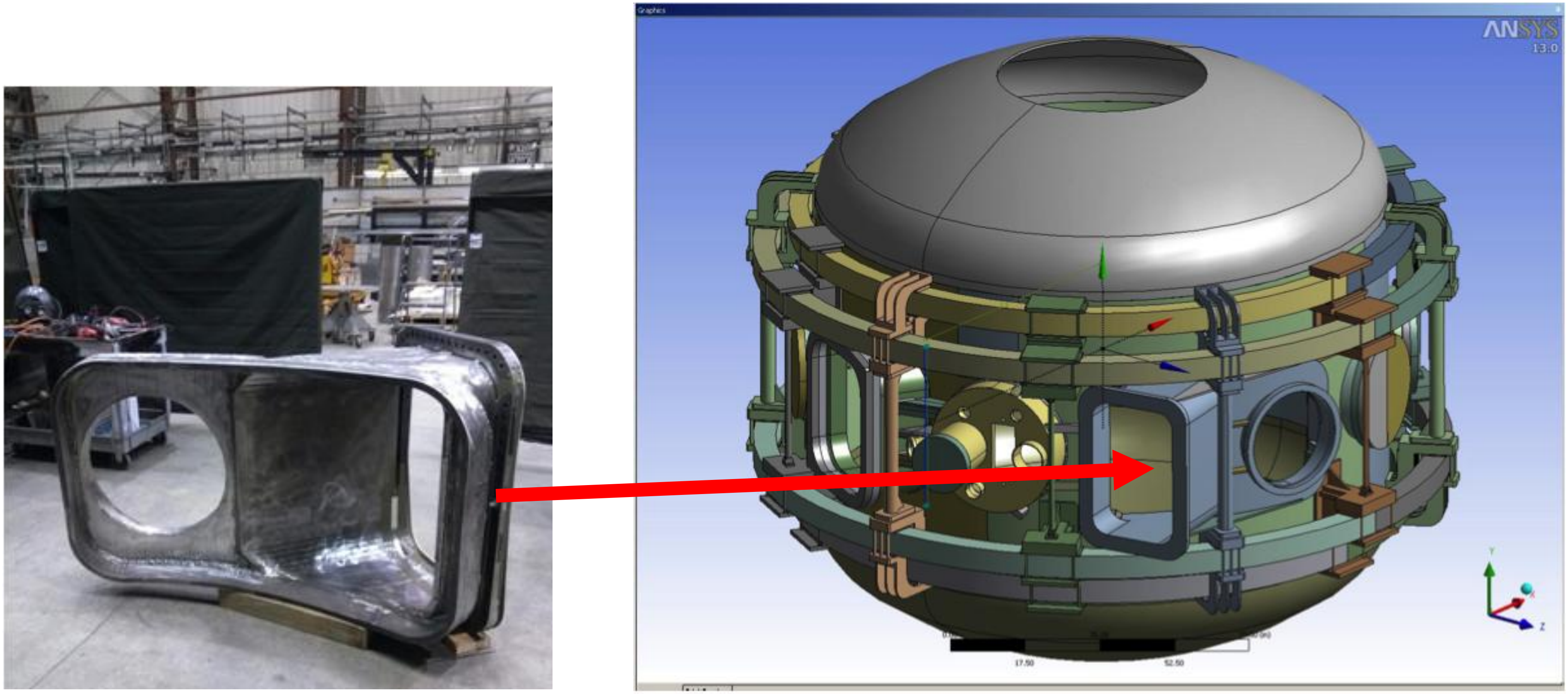
TF 11 and TF7 removed

Coil support modifications- Rib welding complete.

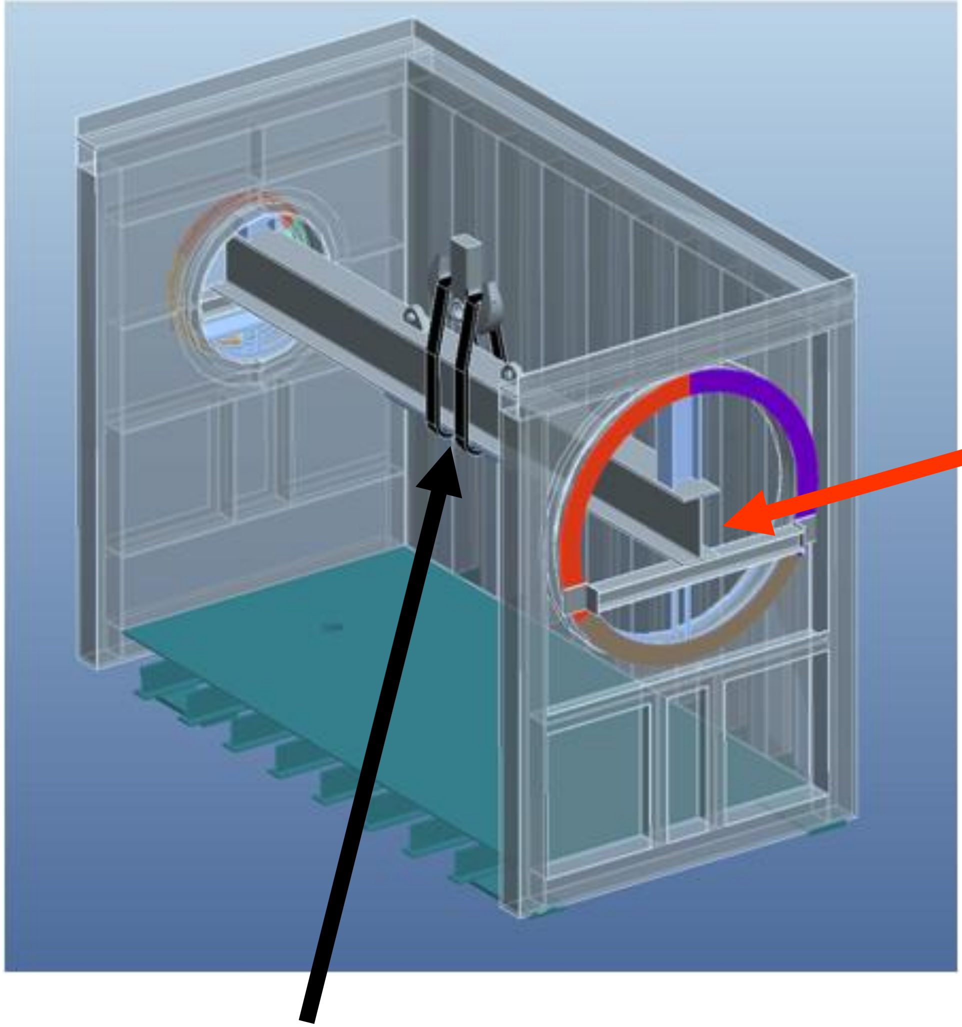
New clevis pads, PF4/5 supports, umbrella legs

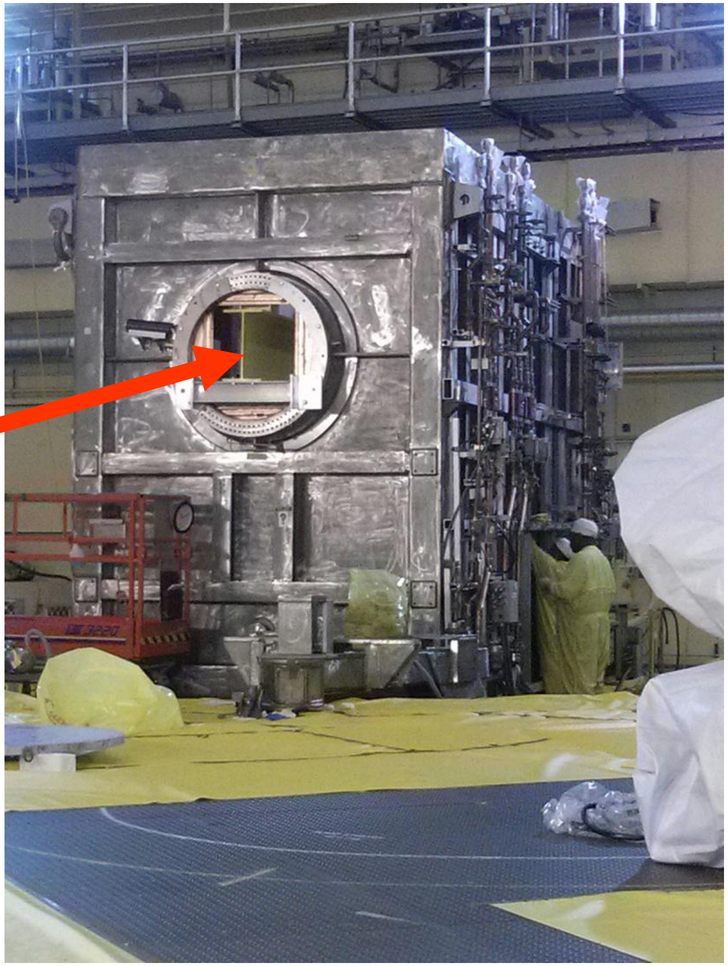
New NBI port-cap has been received

- Materials, machining meet spec (but welds being re-worked)
- Preparing to plasma-cut hole in vessel for cap installation



2nd NBI to move to NSTX-U test-cell in Sept/Oct

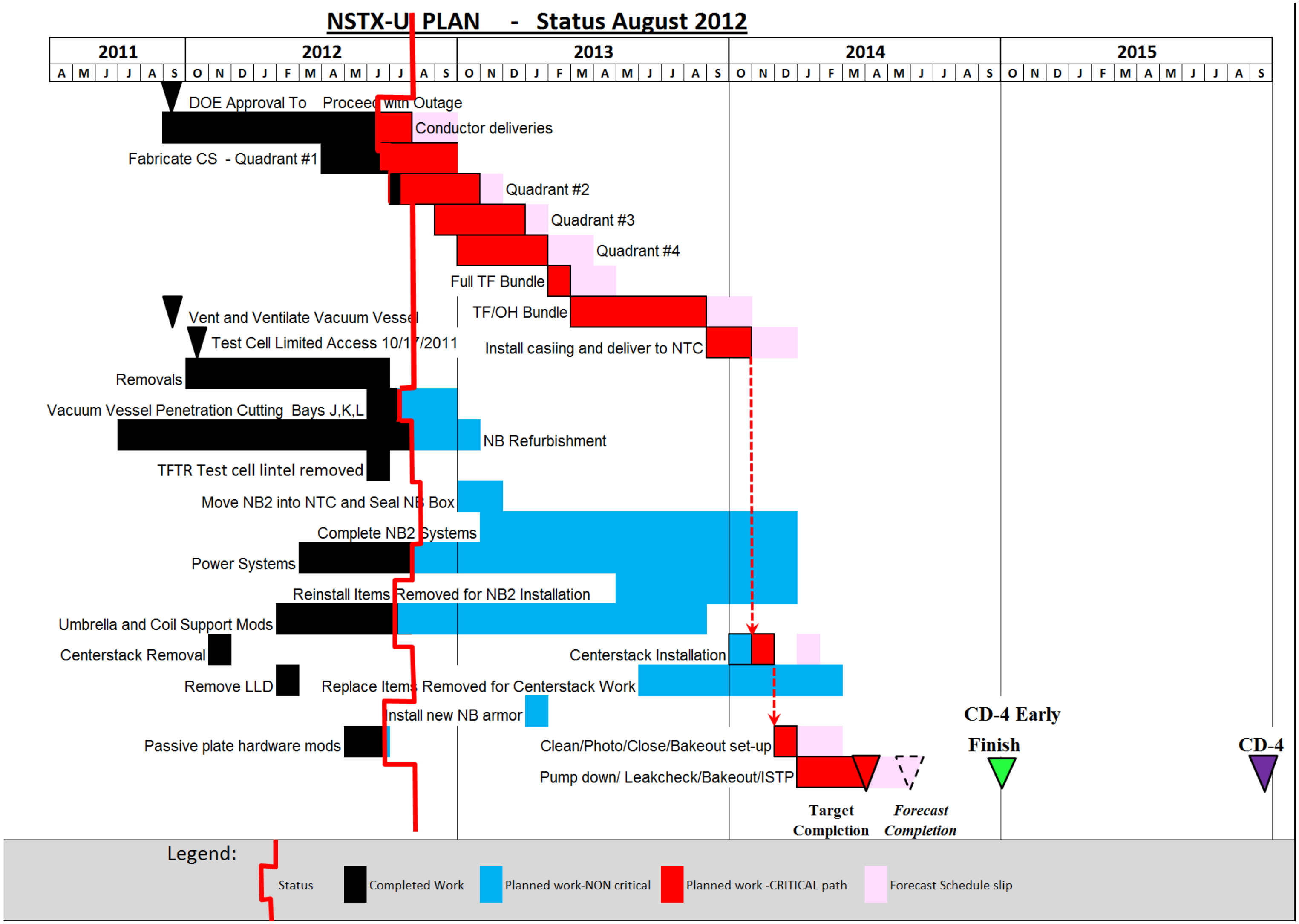




Reentrant hook lift fixture designed to increase clearance over NSTX test-cell wall

Lift fixture installed, tested, and ready to go…

Project on-track for (early) completion: Apr-Jun 2014



President's budget for FY2013 would delay completion by ~1 year

Summary

- NSTX-U device and research will narrow many performance and understanding gaps to next-steps
- The Upgrade Project has made good progress in overcoming key design challenges
- The Project is on schedule and budget
- NSTX-U team now formulating next 5 year plan (2014-18) to access new ST regimes including follow-on staged and prioritized upgrades

Backup slides

Backup slides

Program Plans

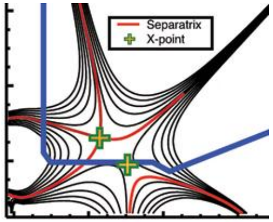
Formulating FY2014-18 5 year plan to access new ST regimes with Upgrade + additional staged & prioritized upgrades

2009	2010	2011	2012	2013	2014	2015	2016	2017	2018
1 MA Plasma			Upgrade Outage		1.5 → 2 MA, 1s → 5s				

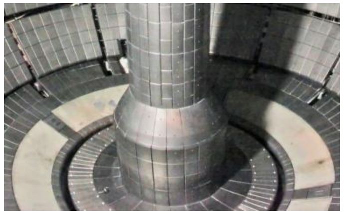
● CHI Control Coils

● LLD ● Moly-tile

● HHFW Upgrade



“Snowflake”

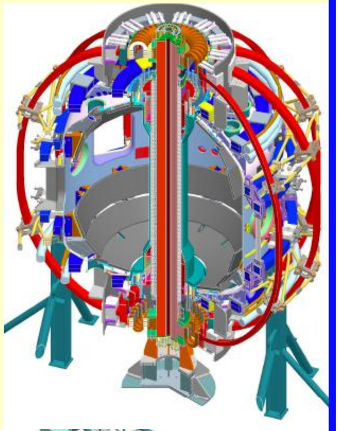


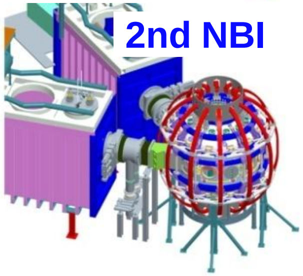
Lithium

New Center-Stack ●

2nd NBI ●

New Center-stack





2nd NBI

0.5 MA CHI ●

● 1 MA CHI / Plasma Gun

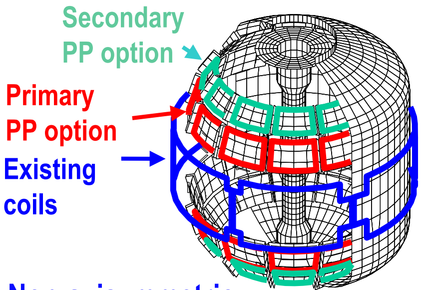
ECH/EBW ● 1MW → ● 2 MW

0.5 MA Plasma Gun ●

Long-pulse cryo-pumped divertor ●

Flowing Li module ●

NCC Upgrade ●

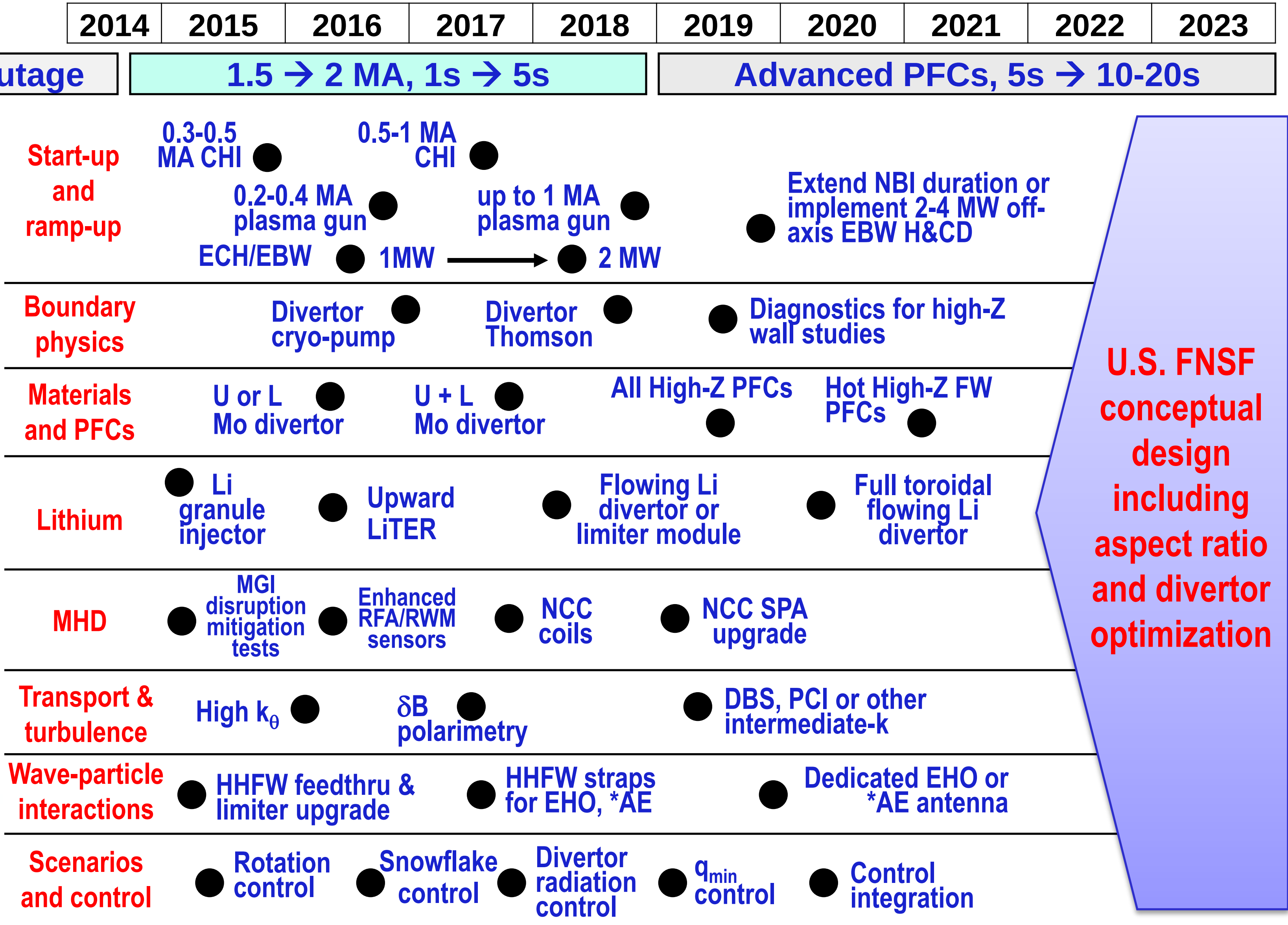


Non-axisymmetric Control Coils (NCC)

NSTX Upgrade research goals in support of FNSF and ITER

- Low collisionality plasma regimes
- 100% non-inductive operation
- Long-pulse, high power divertor
- Advanced high-β scenarios

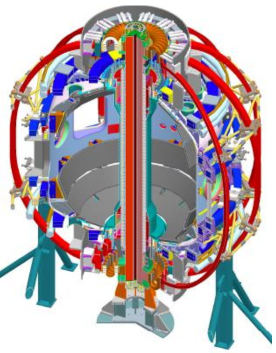
Developed comprehensive long-range plan for NSTX-U supporting ITER and FNSF – next step is to down-select based on priorities and budgets



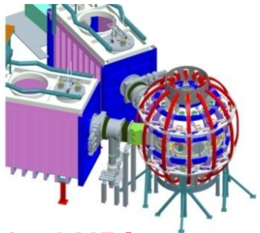
2014	2015	2016	2017	2018	2019	2020	2021	2022	2023

Upgrade Outage | **1.5 → 2 MA, 1s → 5s** | **Advanced PFCs, 5s → 10-20s**

New center-stack



2nd NBI



Start-up and ramp-up
- 0.3-0.5 MA CHI ●
- 0.5-1 MA CHI ●
- 0.2-0.4 MA plasma gun ●
- up to 1 MA plasma gun ●
- ● Extend NBI duration or implement 2-4 MW off-axis EBW H&CD
- ECH/EBW ● 1MW → ● 2 MW

Boundary physics
- Divertor cryo-pump ●
- Divertor Thomson ●
- ● Diagnostics for high-Z wall studies

Materials and PFCs
- U or L Mo divertor ●
- U + L Mo divertor ●
- All High-Z PFCs ●
- Hot High-Z FW PFCs ●

Lithium
- ● Li granule injector
- ● Upward LiTER
- ● Flowing Li divertor or limiter module
- ● Full toroidal flowing Li divertor

MHD
- ● MGI disruption mitigation tests
- ● Enhanced RFA/RWM sensors
- ● NCC coils
- ● NCC SPA upgrade

Transport & turbulence
- High k_θ ●
- δB ● polarimetry
- ● DBS, PCI or other intermediate-k

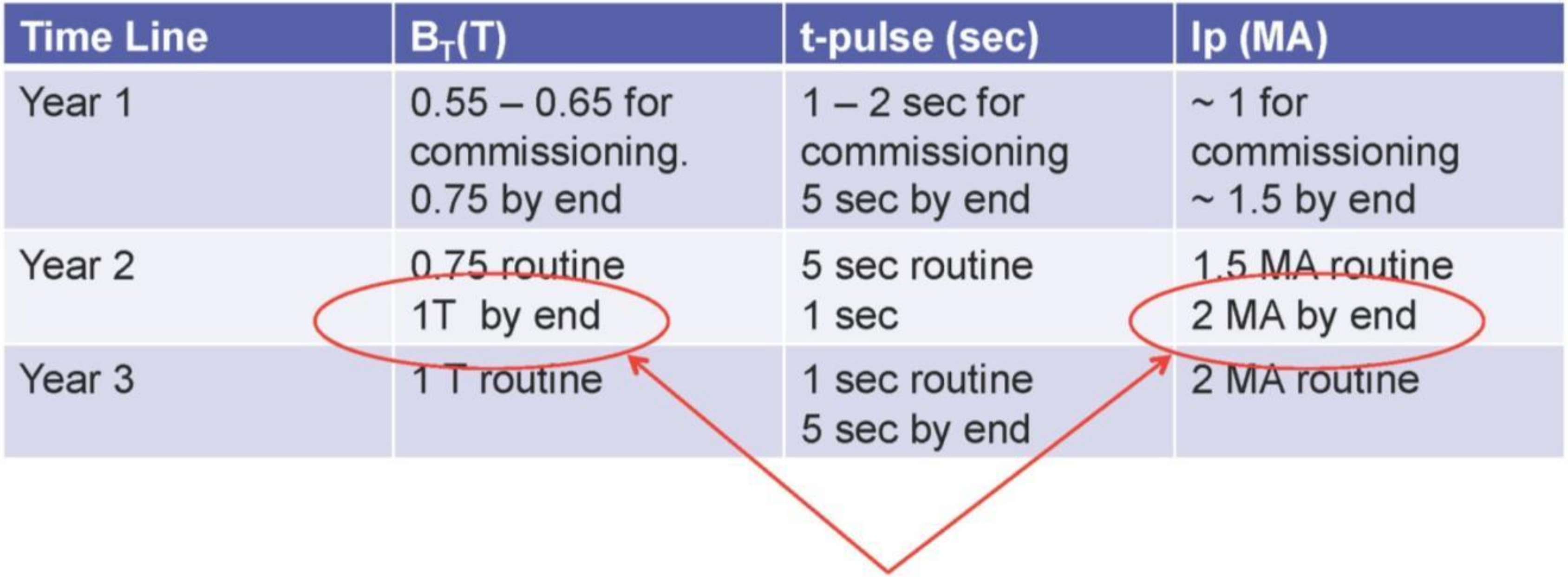
Wave-particle interactions
- ● HHFW feedthru & limiter upgrade
- ● HHFW straps for EHO, *AE
- ● Dedicated EHO or *AE antenna

Scenarios and control
- ● Rotation control
- ● Snowflake control
- ● Divertor radiation control
- ● q_{min} control
- ● Control integration

U.S. FNSF conceptual design including aspect ratio and divertor optimization

Tentative plans for initial operations

NSTX-U will be brought up methodically to full performance capabilities



Time Line	B_T(T)	t-pulse (sec)	Ip (MA)
Year 1	0.55 – 0.65 for commissioning. 0.75 by end	1 – 2 sec for commissioning 5 sec by end	~ 1 for commissioning ~ 1.5 by end
Year 2	0.75 routine 1T by end	5 sec routine 1 sec	1.5 MA routine 2 MA by end
Year 3	1 T routine	1 sec routine 5 sec by end	2 MA routine

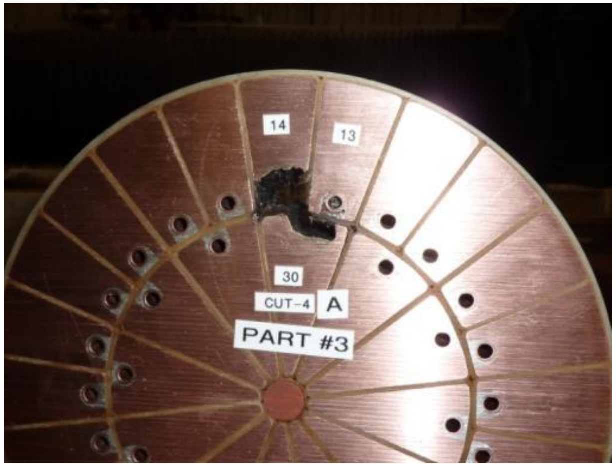
Full field and current by the end of year 2

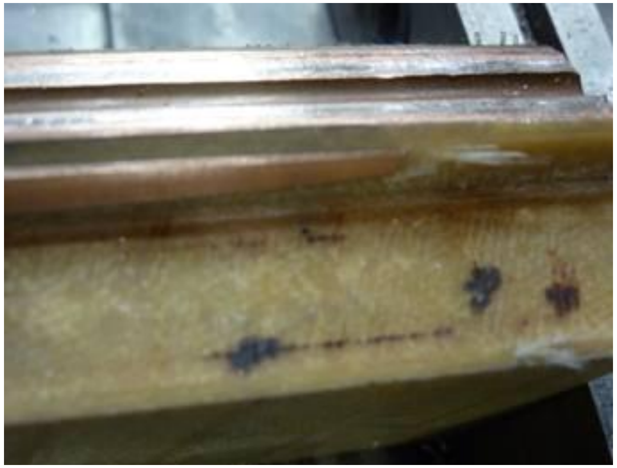
Backup slides

Upgrade Project Progress

NSTX operations was abruptly terminated in July 2011 due to a failure of the inner TF bundle

- **An autopsy/sectioning of the failed bundle was performed**

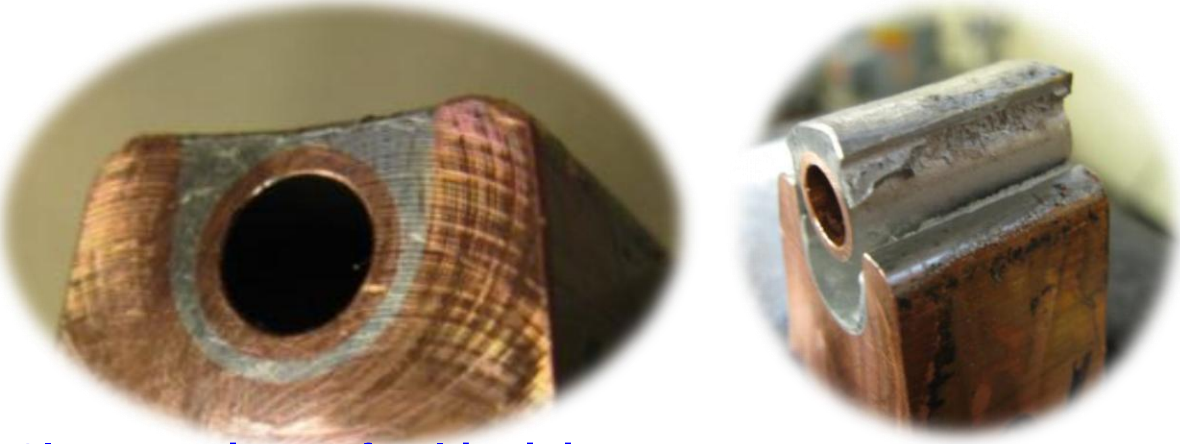




- Failure of conductors was noted to be located in 1 of 3 pairs that were sub-optimal during acceptance tests
  - 100-3000 M-ohm versus 30,000-50,000 M-ohm during 3 kV quadrant tests
- Forensics yielded convincing evidence with regard to both remaining sub-optimal pairs
  - Found 1-10 M-ohm conductive path in a distinct location measured with an ohmmeter
  - Discoloration of epoxy/glass insulation system
  - Localized resin-poor area
- **Increased conductivity traced to zinc chloride in residual solder flux**

→ Decided to start 2½ year Upgrade outage 6 months early

Developed new soldering technique with resin-based flux in response to TF bundle fault lesson-learned



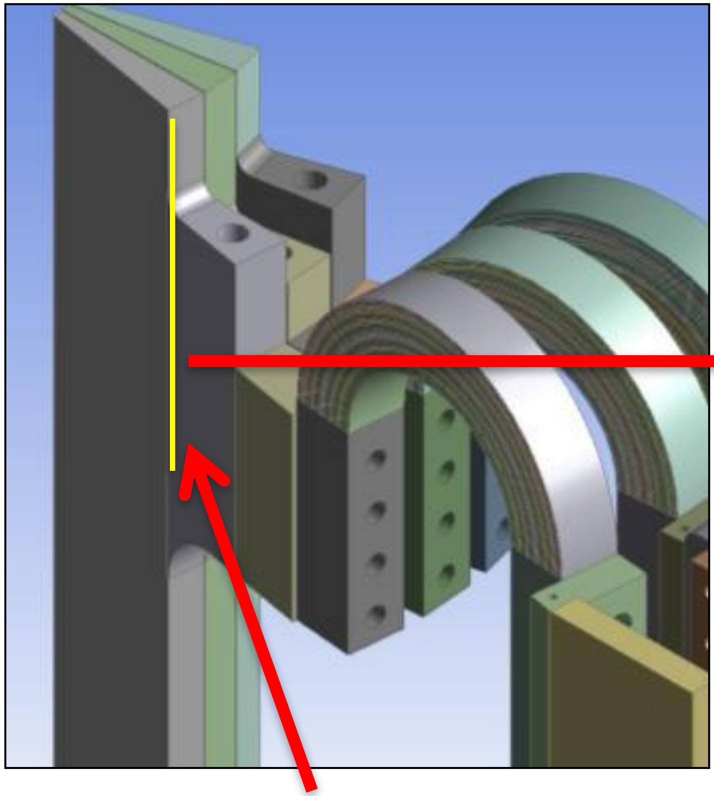
Images of tubes pull tested to ultimate strength of the solder . Note good wetting of both the tube and copper bar, indicating effectiveness of flux.

Close up views of solder joint

Solder paste injection over cooling tube

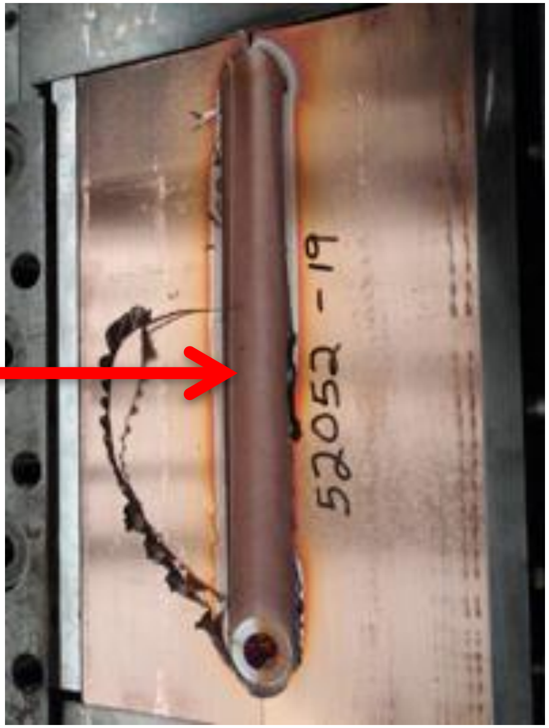
The manufacturing procedure and the soldering-line is nearing completion.

Friction Stir Welding



Lead Extension to Inner TF Conductor

Development trials required to prove dissimilar material welding



Neutral Beam refurbishment proceeding well

☑ *Neutral beam refurbishment continuing*

☑ *Major components delivered (ie NB port cap, large VAT valves, power cables manufactured)*

☑ *Rectangular bellows fabricated and passed leak test.*

☑ *Six ion sources (for 2 beamlines) ready to go!*

☑ *Lintel removed*

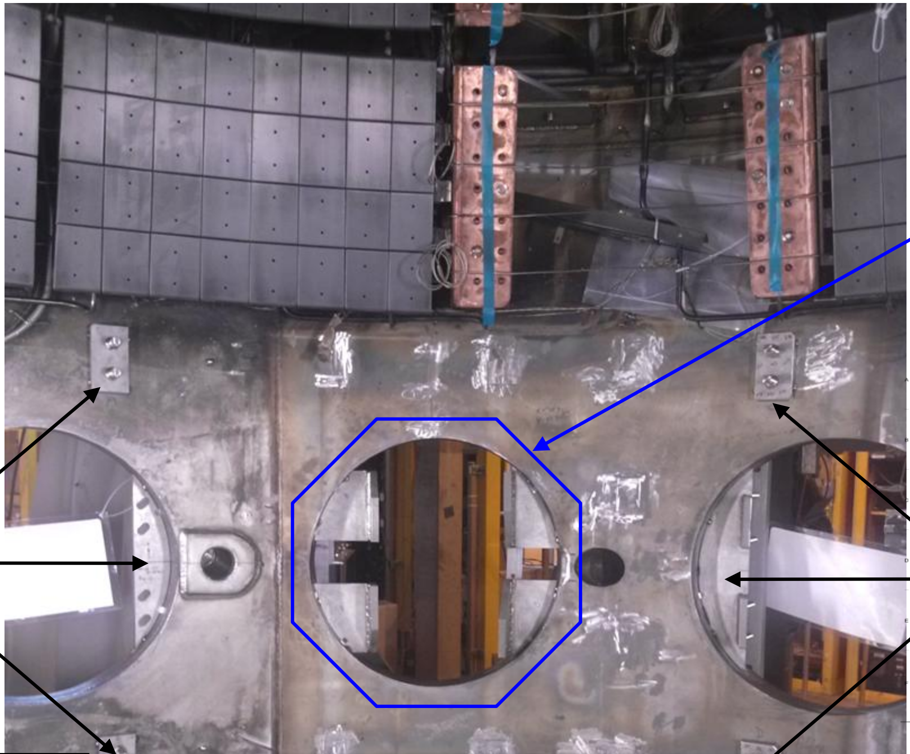
☑ *Penetrations complete.*

☑ *Relocation into NSTX TC planned for October 2012 but on track for September*

NBI Armor fabrication and installation in progress…

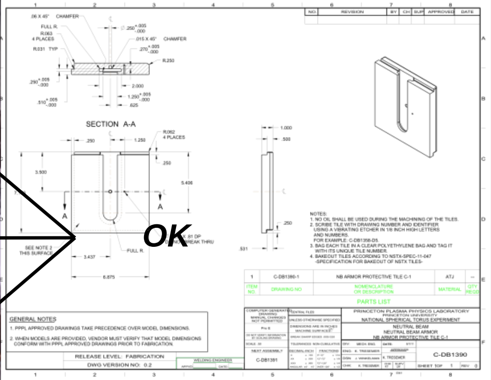
Supports aligned using metrology and tack welded in place to determine field fit changes…

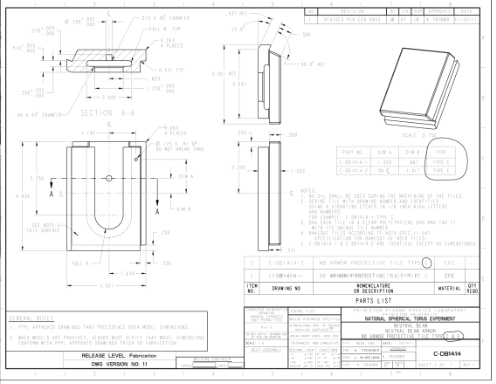
Bay H supports need to be field fit and aligned using backing plates before final welding…

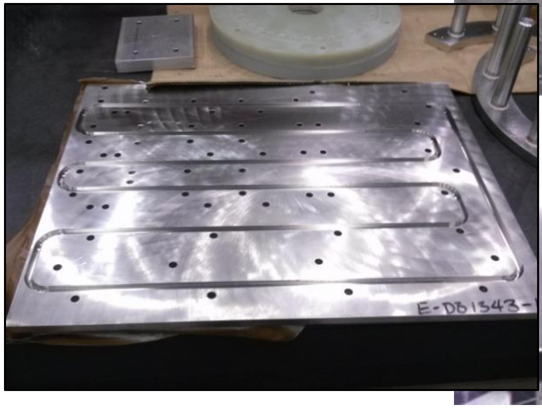


OK

OK

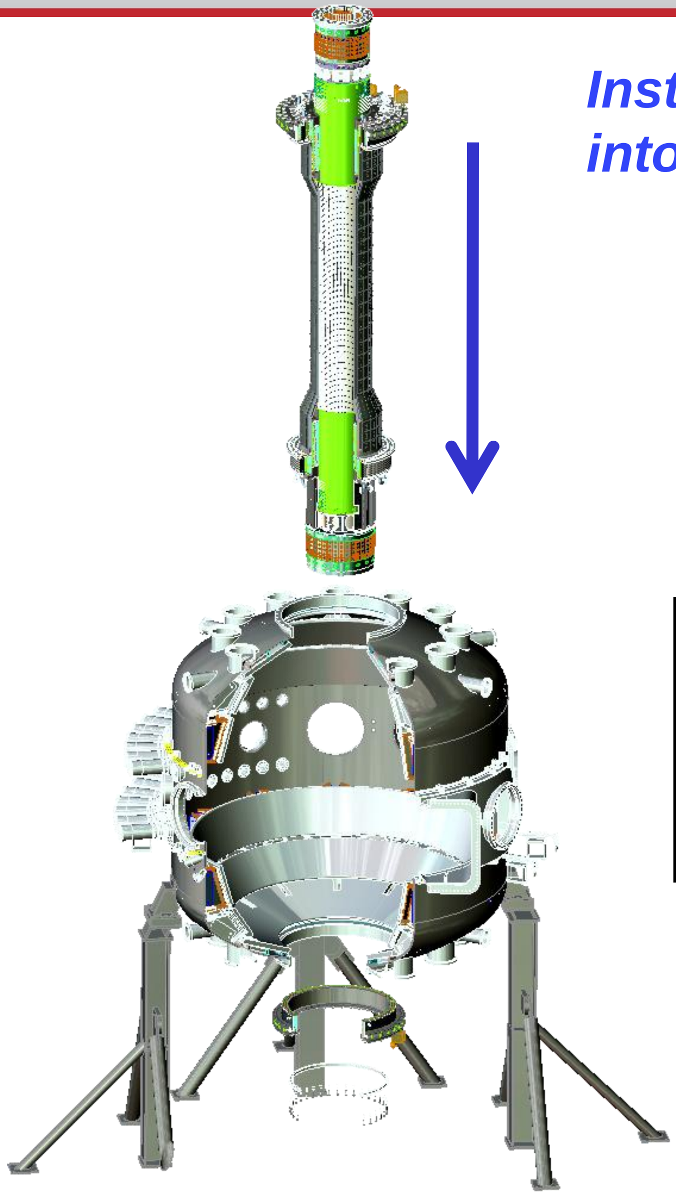






- *All 4 Backing plates fabricated; final machining for manifolds in progress…*
- *Manifolds done. Fitup on backing plates…*
- *Carbon tile order in progress*
- *Stiffener pieces fabrication done*
- *Braze/weld quadrants in shop next*
- *Align/assemble/install/leakcheck*

Install Fabricated CS Assembly into VV



Install the completed Center Stack Assembly into the Vacuum Vessel (Spring 2014)

Have removed and re-installed CS numerous times over the last 10 years.

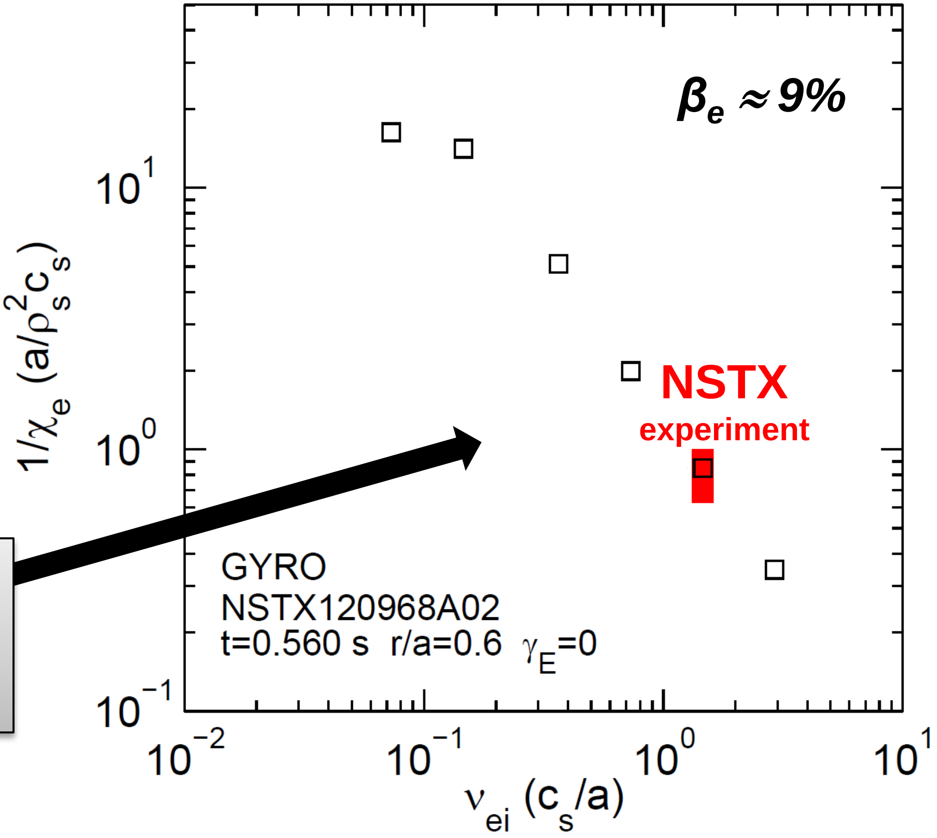
Backup slides

Transport research

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

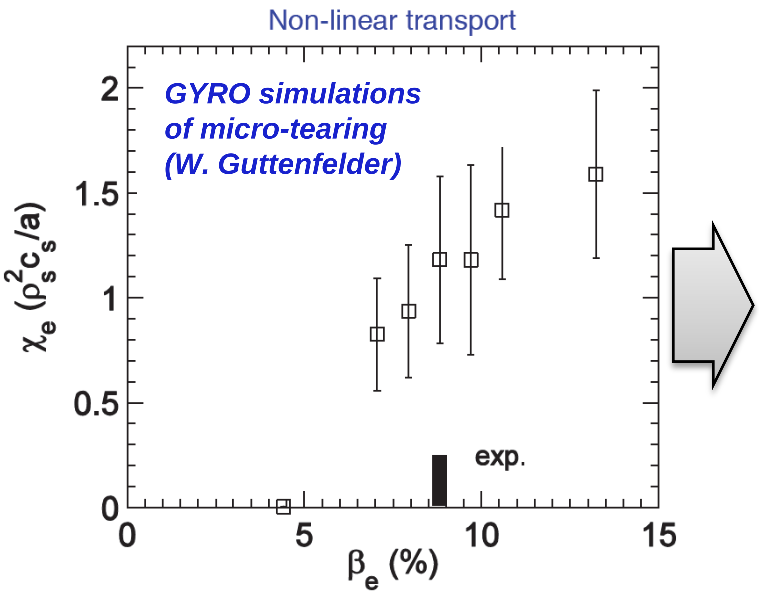
- Non-linear gyrokinetic turbulence simulations of micro-tearing instabilities predict **$\tau_E \propto 1/\chi_e \propto 1/\nu_e^*$**

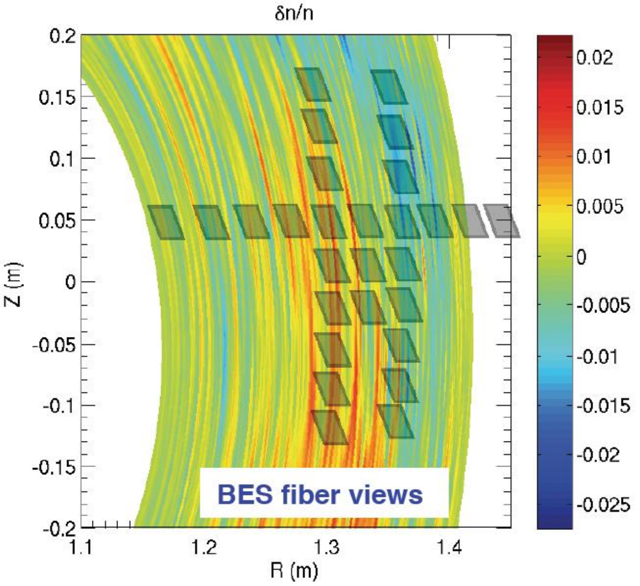
- Predominantly electromagnetic turbulence – result of high β
- Candidate explanation for ST confinement scaling observed on NSTX and MAST

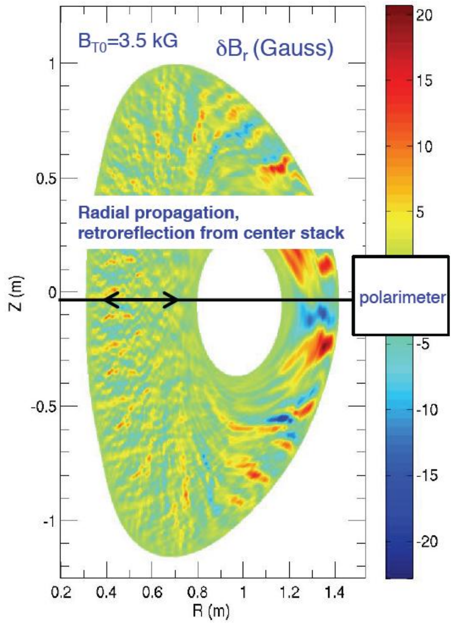


Lower ν* accessible in Upgrade will clarify roles of micro-tearing vs. ETG, TEM in ST e-transport

NSTX-Upgrade will extend diagnosis and understanding of micro-instabilities potentially responsible for anomalous transport in STs







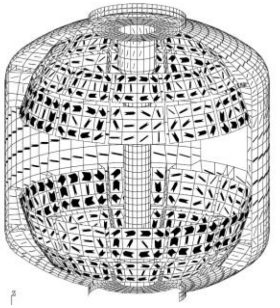
- Electrons dominant loss channel for ST thermal confinement
  - Micro-tearing strong candidate for anomalous thermal e-transport at higher β
  - ETG can also contribute to e-transport at lower β
  - Alfvénic instabilities (GAE/CAE) can also cause core electron transport
- NSTX-U goal is to study full turbulence wave-number spectrum:
  - low-k – ITG/TEM/AE/μ-tearing (BES, polarimetry) + high-k – ETG (μ-wave scattering)
- NSTX-U will access unique turbulence regime: high β + lower ν*

Backup slides

High-beta research

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

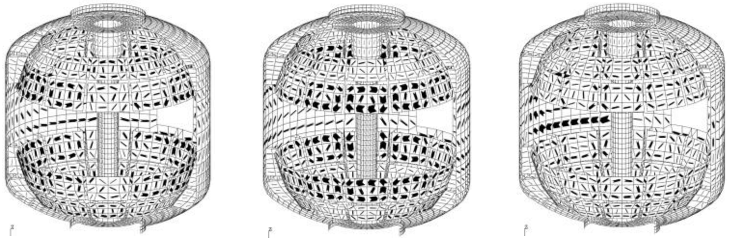
Full 3-D model



~3000+ states

State reduction (< 20 states)

RWM eigenfunction (2 phases, 2 states)

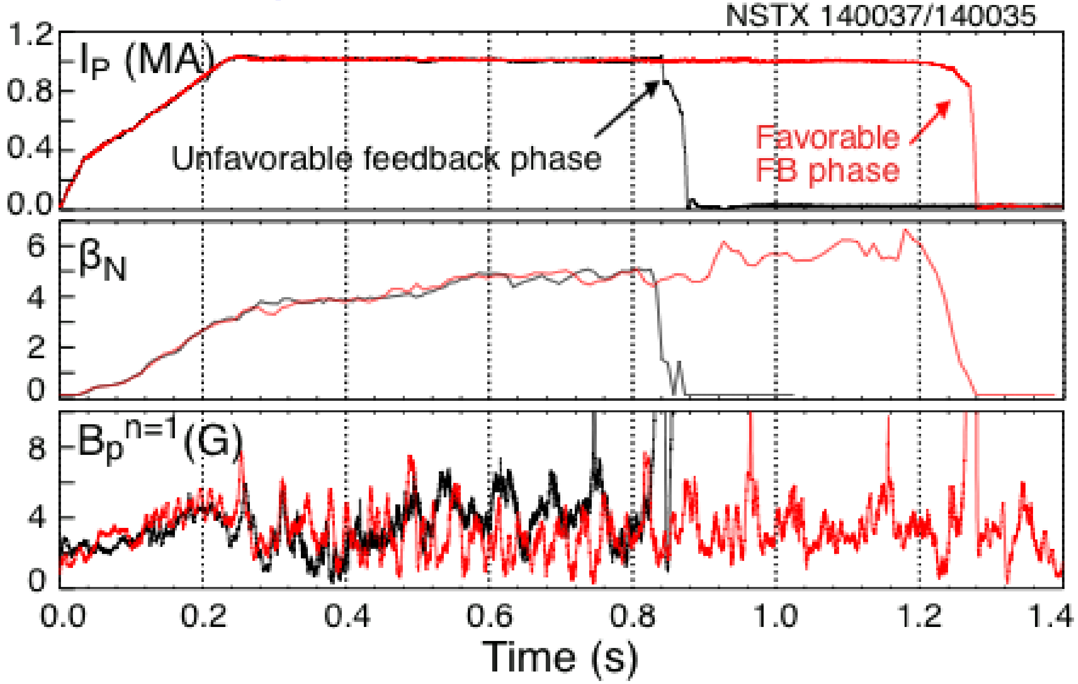


$(\hat{x}_1, \hat{x}_2)$ $\hat{x}_3$ $\hat{x}_4$ $\hat{x}_N$

truncate

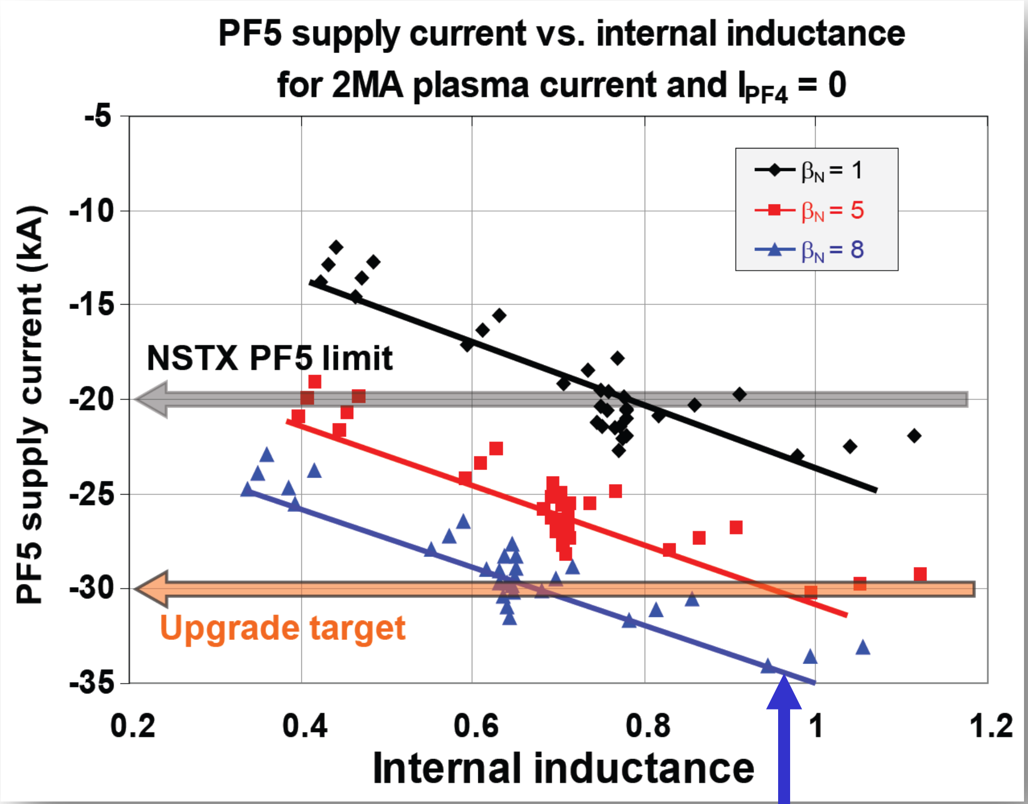
- *Device R, L, mutual inductances*
- *Instability B field / plasma response*
- *Modeled sensor response*

State space feedback with 12



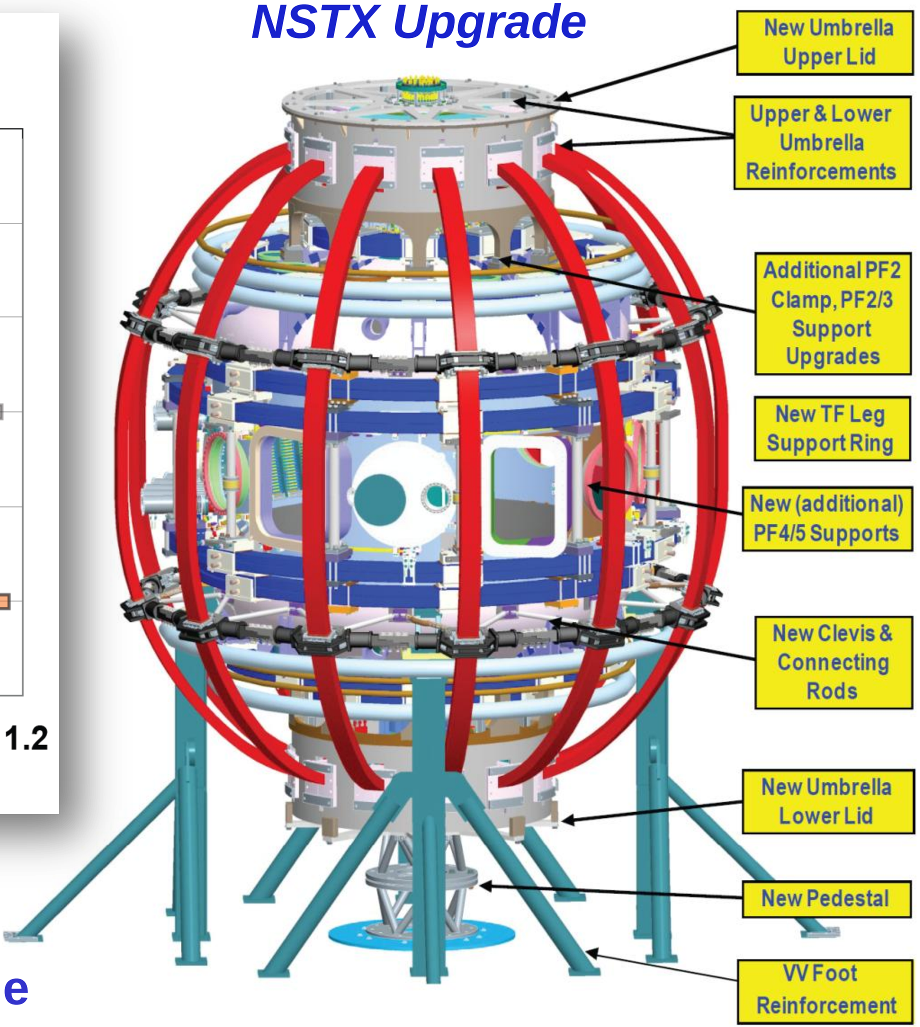
- Controller can compensate for wall currents
  - Including mode-induced current
  - Examined for ITER
- Successful initial experiments
  - Suppressed disruption due to n = 1 applied error field
  - Best feedback phase produced long pulse, $\beta_N = 6.4$, $\beta_N / l_i = 13$

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

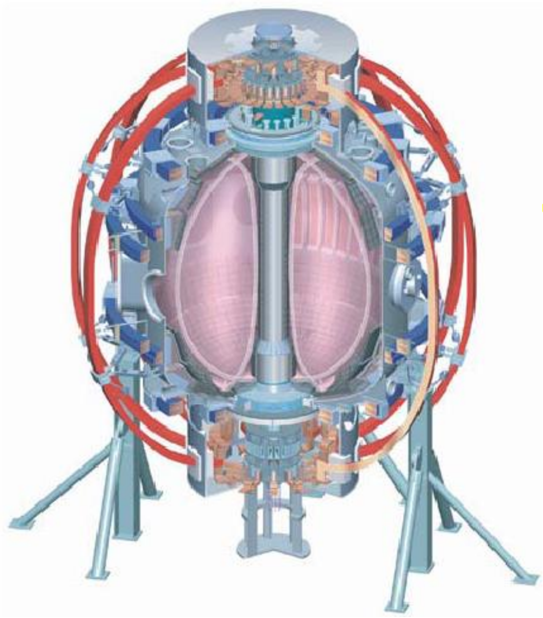
New Pedestal

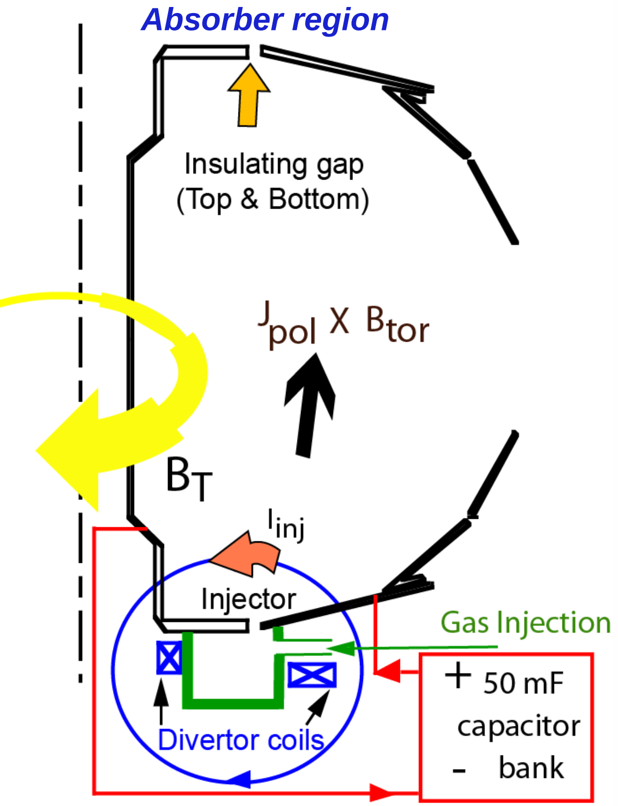
VV Foot Reinforcement

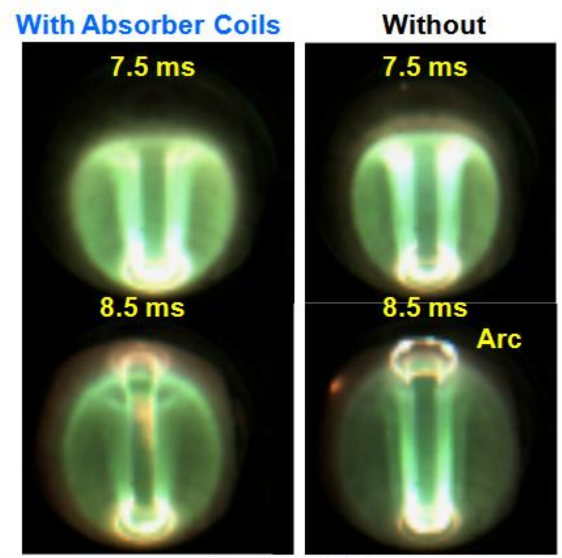
Backup slides

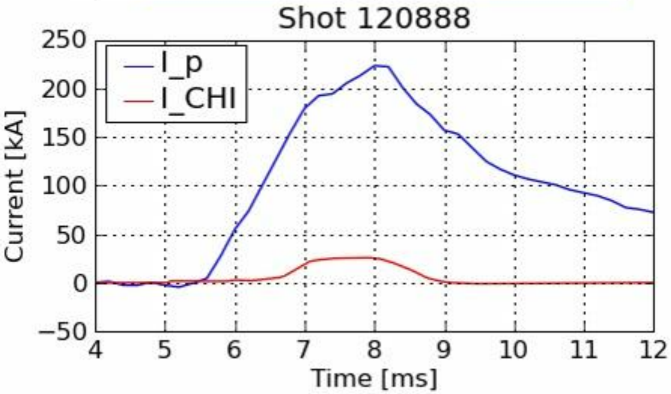
Start-up and sustainment research

Transient CHI: Axisymmetric Reconnection Leads to Formation of Closed Flux Surfaces







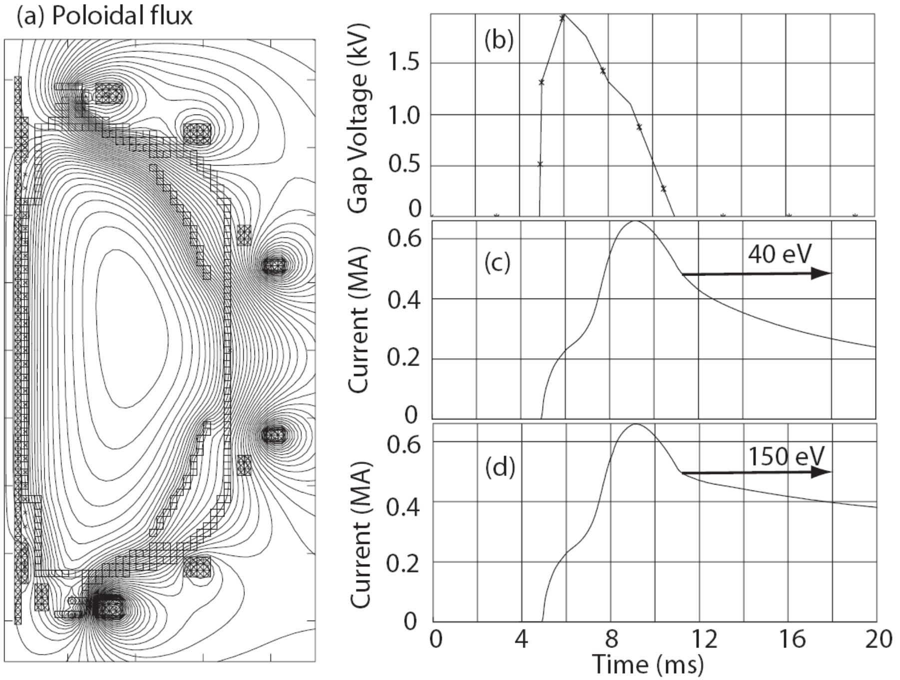


$$I_P = I_{inj}(\psi_T / \psi_{inj}) \qquad I_{inj} = 2\psi_{inj}^2 / (\mu_o^2 d^2 I_{TF})$$

- Current multiplication increases with toroidal field
  - Favorable scaling with machine size
  - High efficiency (10 Amps/Joule in NSTX)

Simulations of CHI project to increased start-up current in NSTX Upgrade, highlight need for additional electron heating

- TSC simulations of transient CHI consistent with NSTX trends
- Favorable projections for NSTX-U:
  - TF increased to 1T and injector flux increased to about 80% of max allowed → **can generate up to ~400kA closed-flux current**
  - **Figs (a-c): T_e = 40 eV, Z_{eff} = 2.5**
  - **Fig (d): T_e = 150 eV for t > 12 ms**

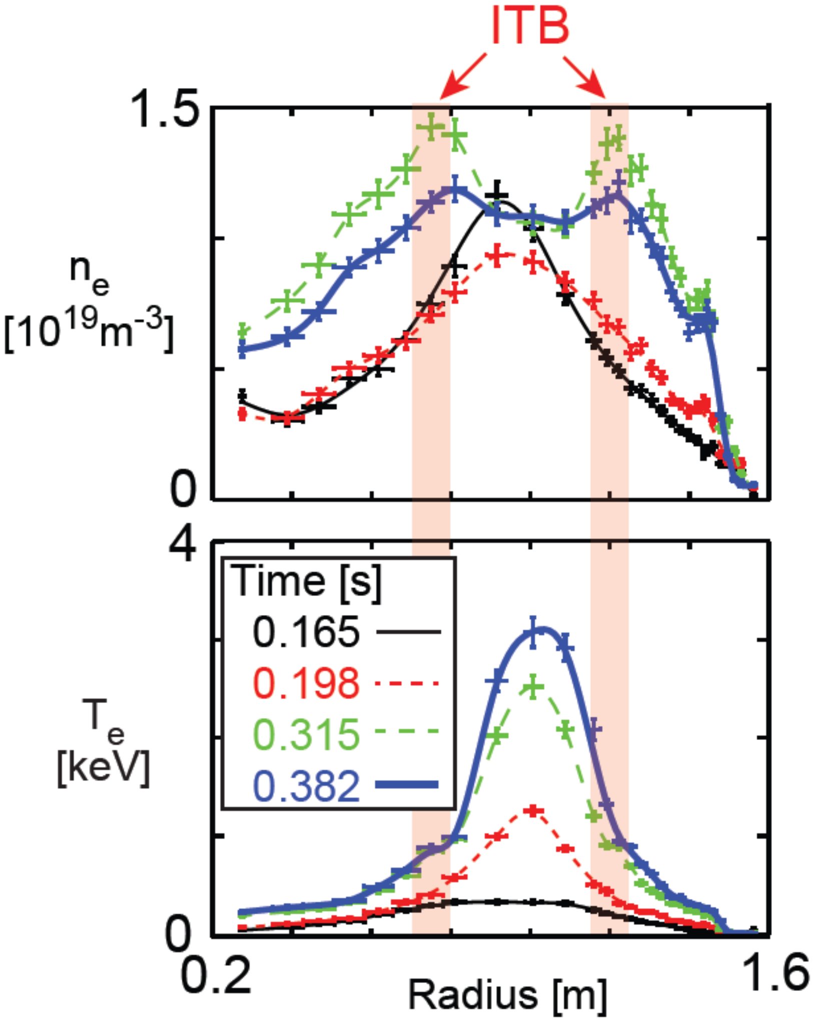


- T_e ~150-200eV needed to extend current decay time to several 10's of ms
- Low density and β of CHI plasma + transient position (i.e. outer gap) evolution → HHFW coupling and heating very challenging
- NSTX CHI plasmas not over-dense → 28GHz ECH heating of 1T CHI plasma likely best option for generating non-inductive ramp-up target

See presentations by R. Raman and G. Taylor for more details

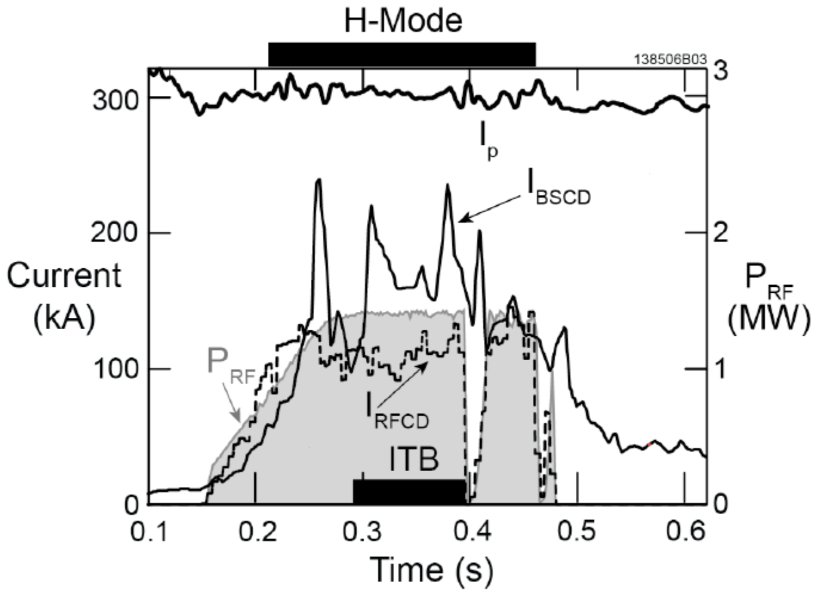
HHFW promising for heating low current target plasma for NBI non-inductive ramp-up

Can heat I_P ~ 300kA, 200eV plasma to T_e = 3keV w/ low P_{RF} ~1.4MW

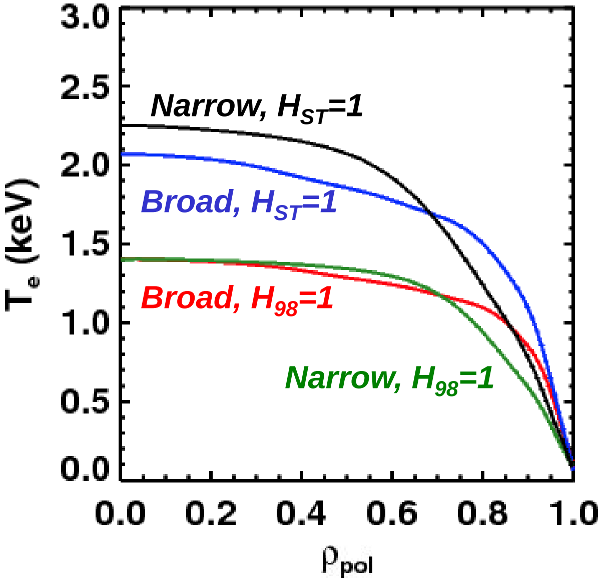


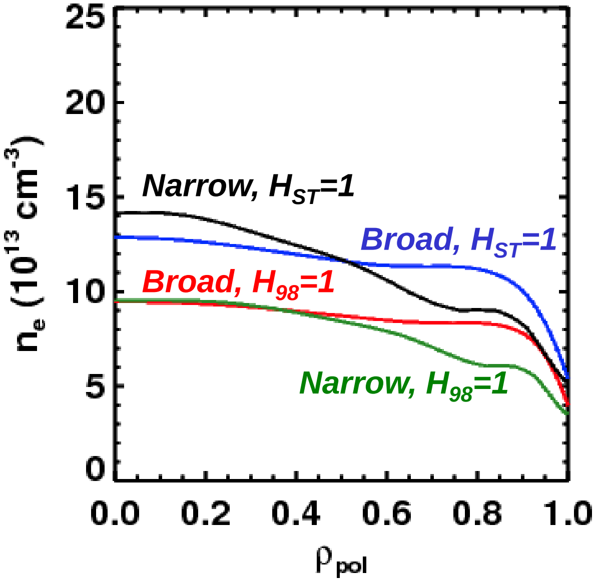
- Form core + edge transport barriers
- Non-inductive fraction of 65-85%
  - 40-50% bootstrap, 25-35% RF-CD
- Projects to 100% non-inductive at P_{RF} = 3-4MW in NSTX-U

  → Target for NBI I_P ramp-up



Scenario modeling using TRANSP projects to 100% non-inductive current at I_P = 0.9-1.3MA at B_T=1.0 T



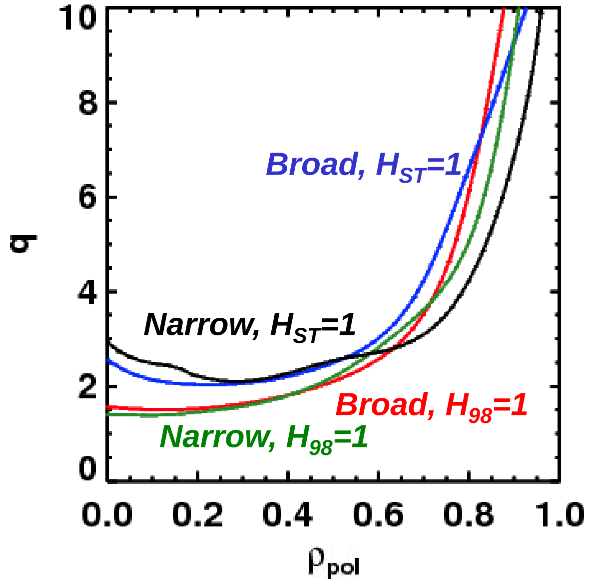


Dashed: ITER-98 confinement scaling

$$\tau_{98y,2} \propto I_P^{0.93} B_T^{0.15} \overline{n}_e^{0.41} P_{Loss}^{-0.69}$$

Solid: ST confinement scaling

$$\tau_{ST} \propto I_P^{0.57} B_T^{1.08} \overline{n}_e^{0.44} P_{Loss}^{-0.73}$$



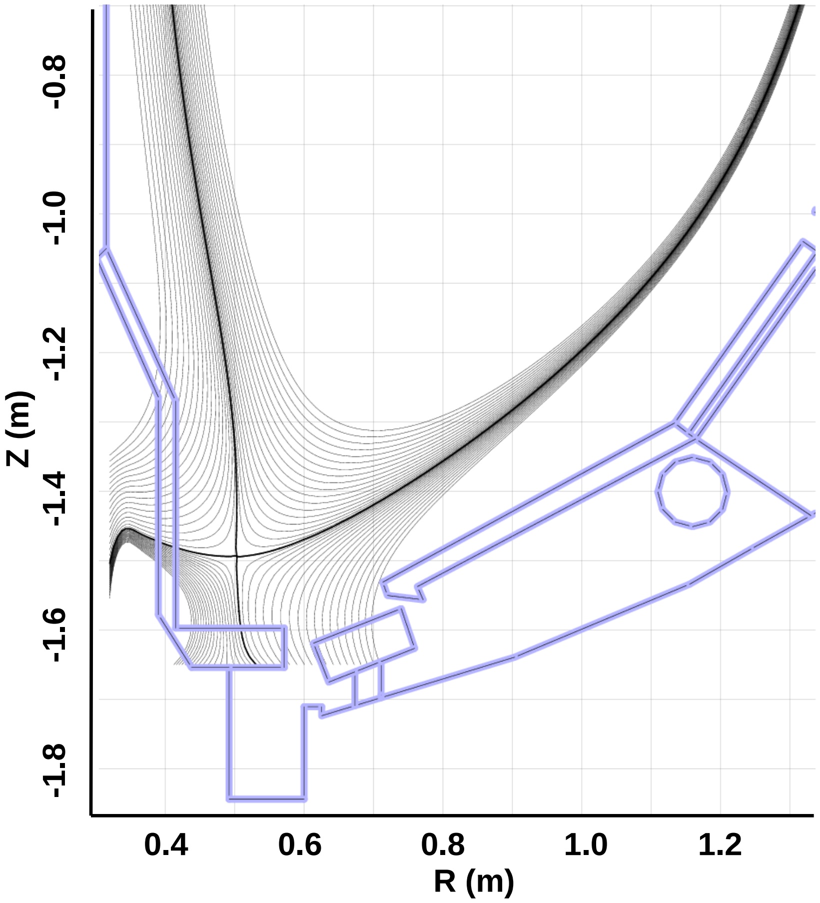
- **Fix: 1.0T, P_{inj}=12.6 MW, f_{GW}=0.72**
- **Fix: A=1.75, κ=2.8**
- Find the non-inductive current level for 2 confinement and 2 profile assumptions…*yields 4 different projections.*

Confinement	**Profiles**	**I_P [kA]**	**β_N**
H_{98}=1	Broad	975	4.34
H_{ST}=1	Broad	1325	5.32
H_{98}=1	Narrow	875	4.87
H_{ST}=1	Narrow	1300	5.97

Backup slides

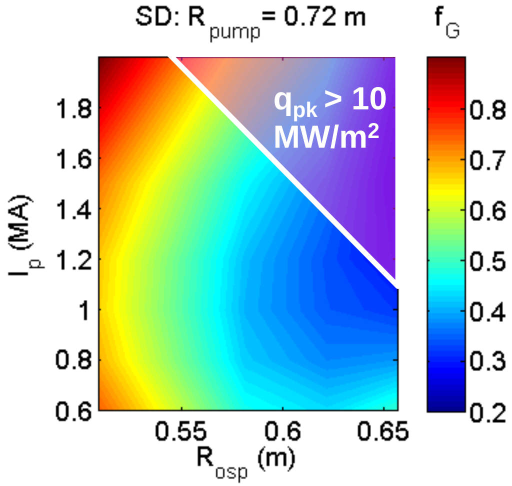
Cryo-pumping design

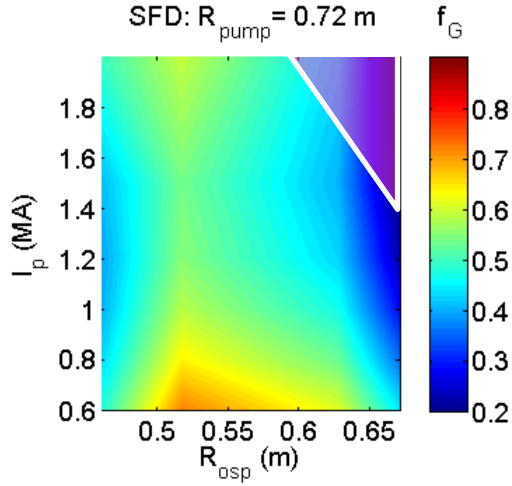
Optimized plenum geometry capable of pumping to low density for a range of R_{OSP}, I_p



SOLPS geometry to be used in future calculations

- Equilibrium f_G down to < 0.5
  - Moving R_{OSP} closer to pump allows lower n_e, but limited by power handling
- High flux expansion in SFD gives *better* pumping with SOL-side configuration
  - More plasma in far SOL near pump
  - More room to increase R_{OSP} at high I_p





Backup slides

Liquid metal and PFC research

Liquid metal PFCs should be pursued to mitigate risk of tungsten not extrapolating to fusion reactor.

- Recent FESAC report: "The uncertainty in establishing PFC solutions is high, as the environment is severe and the requirements for long lifetime are challenging."
  - Tungsten is leading candidate but has issues with neutron damage, erosion, melting, brittleness, thermal fatigue.
- ReNeW highlighted that DEMO PFCs are much more challenging than ITER's.
  - advocated substantial program to assess new ideas, incl. liquid metals (Li, Sn, Ga).
  - No neutron damage, erosion, thermal fatigue in liquids – but technical base less mature.
- Importantly, liquid flow over tungsten substrate may be unique way to eliminate net erosion and flaking to help make tungsten work
- **Liquid PFCs have potential to relieve over-constrained problem: they do not need to *simultaneously* satisfy plasma and nuclear loading constraints.**
- Significant uncertainties in both approaches suggest both W and liquids should be investigated
- ReNeW recommended: "*Liquid surface PFC operation in a tokamak environment…*"

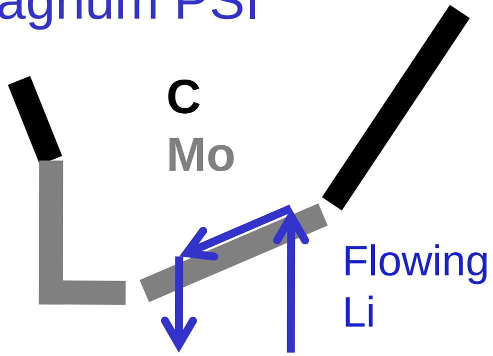
PPPL/PU/NSTX-U team pursuing multidisciplinary approach to developing liquid metal PFCs for NSTX-U, FNSF, beyond

Multi-scale R&D approach from atoms to PFCs:

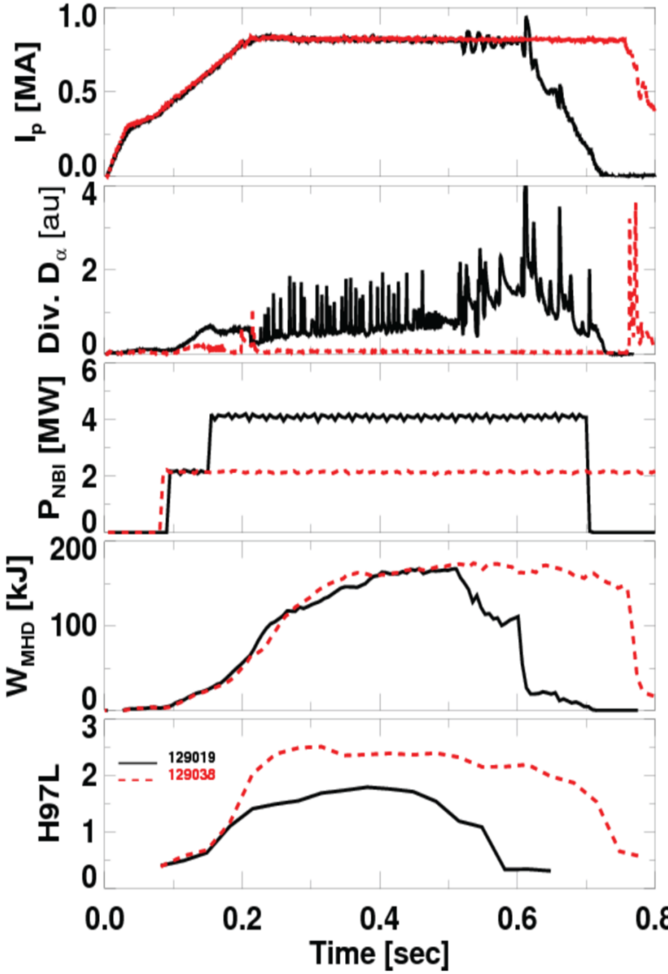
1. Understand impact of lithium on core and edge transport and stability.
2. Assess D pumping vs. surface conditions:
   - Atomistic MD modeling (ORNL)
   - Lab expt. on ideal systems e.g. single xtal Mo + monolayer Li + D^0, D^+ beam.
     detailed surface analysis via XPS, AES, TPD, SAM… (Purdue / PPPL Labs)

2. Assess Heat Flux handling in linear plasma facility:
   - PFC prototype tests with high power plasmas in Magnum PSI

3. Tokamak integration:
   - XGC Kinetic modeling, non-equilibrium Li radiation
   - LTX liquid Li studies, MAPP -> LTX then NSTX-U
   - Li granule injector tests on EAST, then NSTX-U
   - Divertor Li-PFC design, then testing in NSTX-U



Solid Li surface coatings: pump D, increase energy confinement, eliminate ELMs, but confine impurities too well



◀ **Without Li**

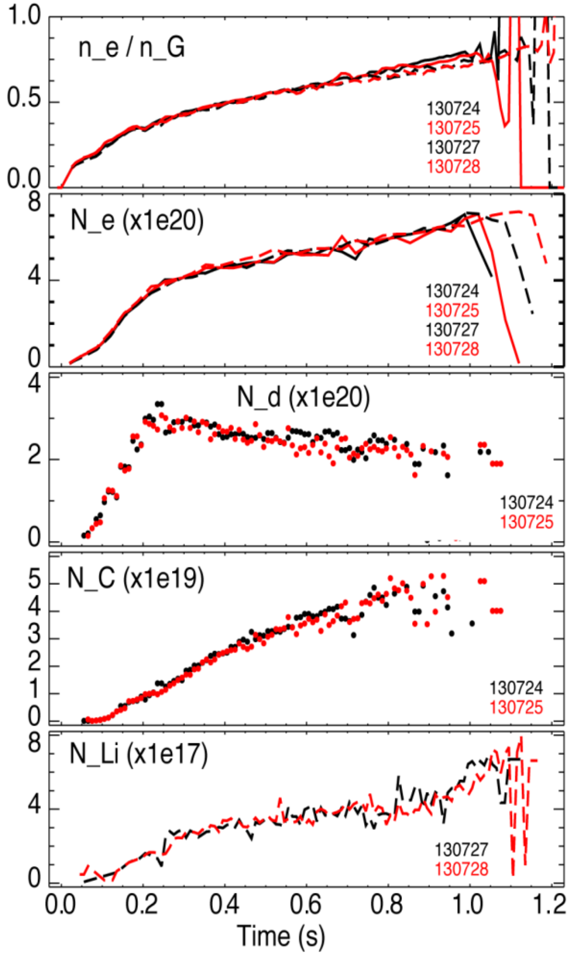
◀ **With Li**

◀ **ELM-free, reduced divertor recycling**

◀ **Lower NBI to avoid β limit**

◀ **Similar stored energy**

◀ **H-factor increases up to 40%**



◀ **Monotonic increase in electron density**

◀ **Deuterium inventory pumped out**

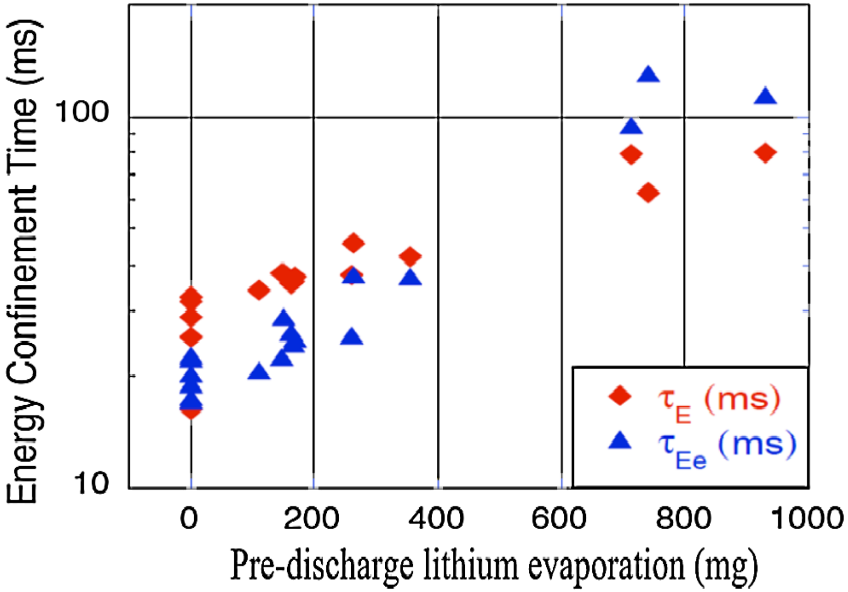
◀ **But carbon inventory increases (Z_{eff} ~ 3-4)**

◀ **Li inventory rises slowly, concentration is very low**

Based on these results, NSTX is shifting emphasis from D inventory control to C impurity reduction

Lithium coatings will continue to be an important research tool for NSTX-U

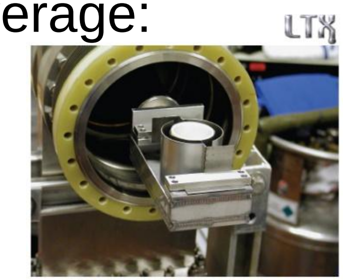
R. Maingi, et al., PRL 107, 145004 (2011)



- Energy confinement increases continuously with increased Li evaporation in NSTX
- High confinement very important for FNSF and other next-steps

what is τ_E upper bound?

- Work with LTX to understand Li chemistry, impact of wall temperature, Li coating thickness
- Assess D pumping vs. surface conditions (MAPP), lab-based surface studies, PFC spectroscopy
- Design/develop methods to increase Li coating coverage:
  - upward evaporation
  - evap into neutral gas
  - Li paint sprayer



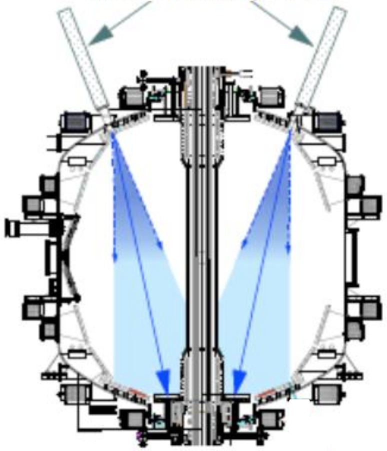
Y_2O_3 crucible, Ta heater
➤Tested to 700 °C

- Assess impact of full wall coverage on pumping, confinement
- Test Li coatings for pumping longer τ_{pulse} NSTX-U plasmas

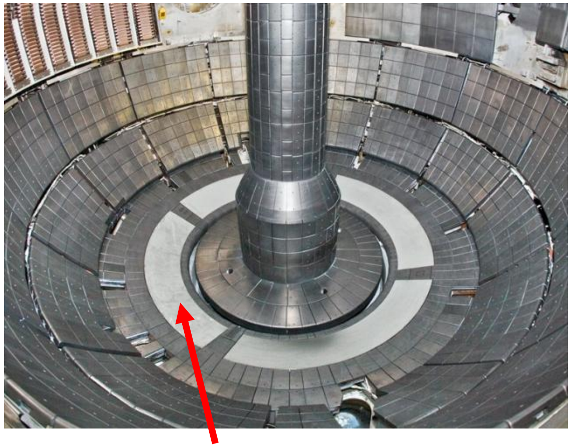
NSTX is a world leader in assessing lithium plasma facing components as a possible PMI solution for magnetic fusion

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

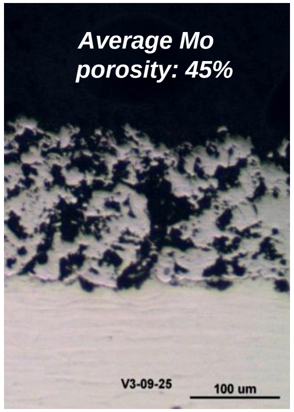
Dual Liquid Lithium Evaporator
For Li wall coatings
Now routinely used



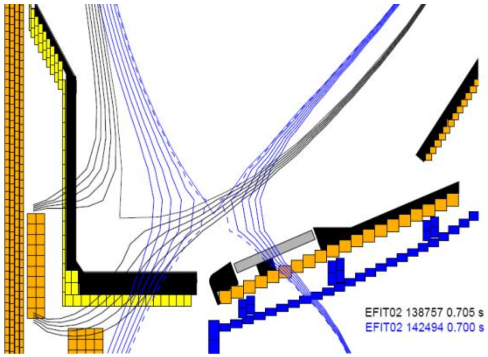
- **Liquid Lithium Divertor (LLD) motivation:**
  - Provide volume D pumping capacity (> solid Li coatings) for increased pumping and duration
  - Potential for handling high heat flux (longer term)



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

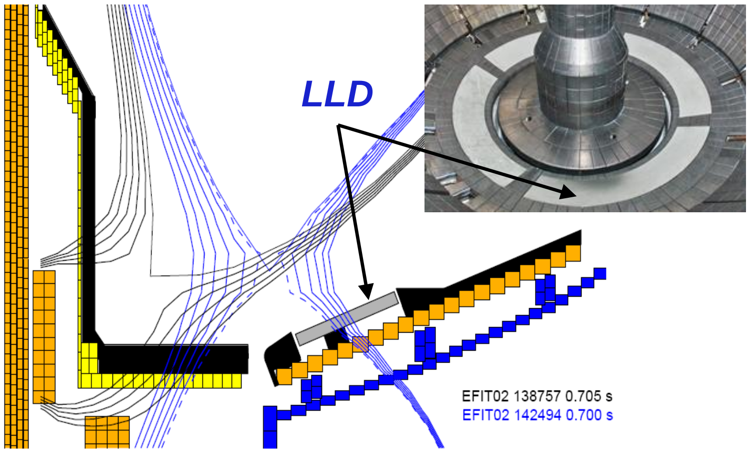


LLD surface cross section: plasma sprayed porous Mo



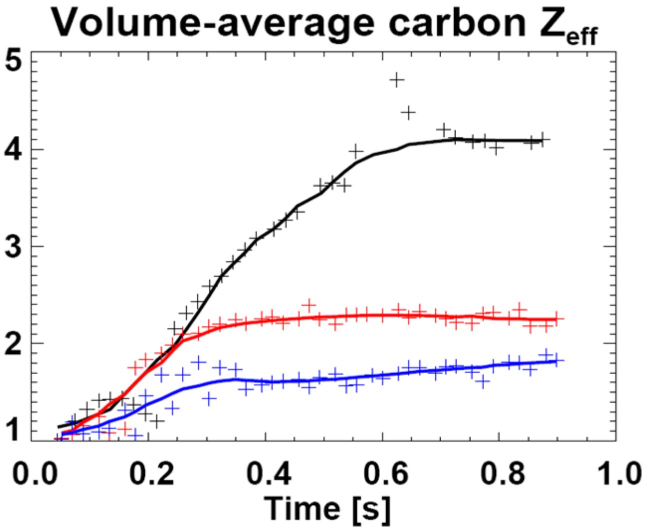
Controlled scans of strike-point location:
On inboard divertor
On LLD (outboard divertor)

Operation with outer strike-point on liquid lithium divertor (LLD) (porous Mo coated w/ Li) compatible w/ high plasma performance



LLD FY2010 results:

- **LLD did not increase D pumping beyond that achieved with LiTER**
- **No evidence of Mo from LLD in plasma during normal operation**
- **Operation with strike-point on LLD can yield <u>reduced</u> core impurities**
- **Row of inboard Mo tiles installed for FY11-12 run, can re-use in NSTX-U**



◄Strike-point on inner C divertor (no ELMs)

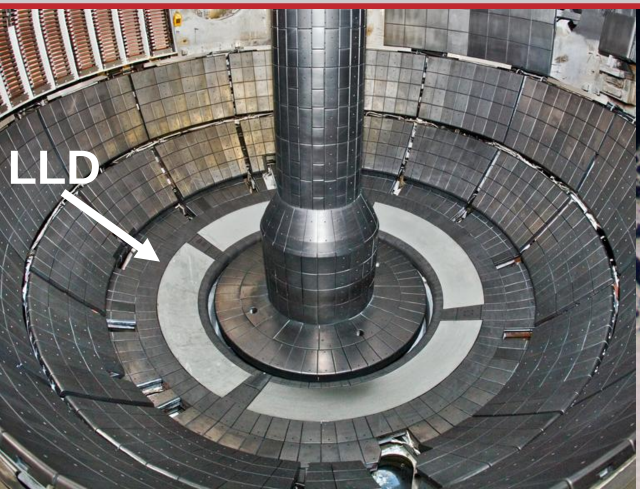
◄Strike-point on LLD, $T_{LLD} < T_{Li\text{-}melt}$

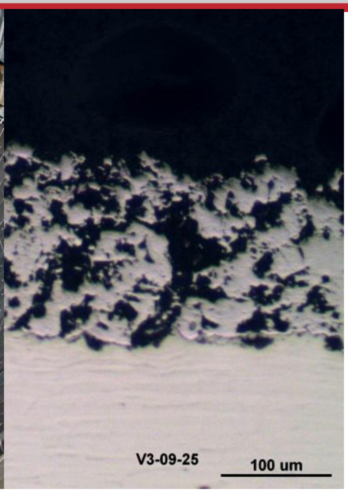
◄Strike-point on LLD, $T_{LLD} > T_{Li\text{-}melt}$ (+ fueling differences)

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

Li + plasma-facing component research will be continued, extended in NSTX-U

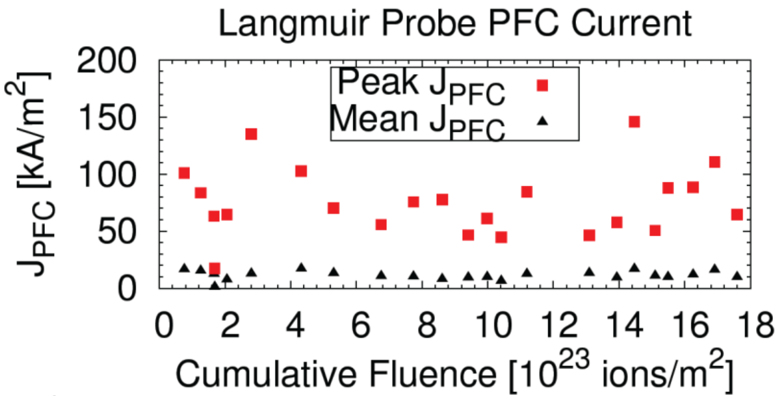
LLD with optimized pore size and layer thickness can provide stable lithium surface

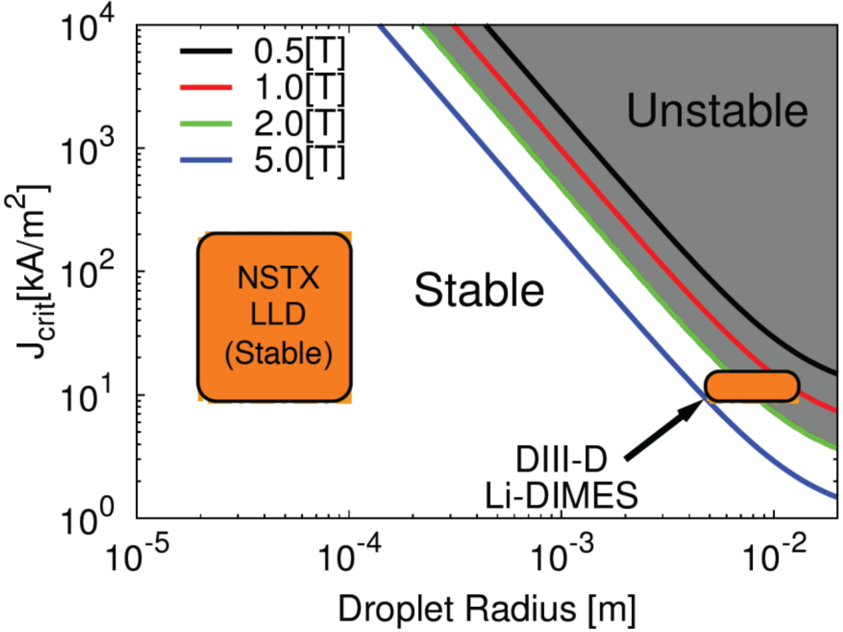




LLD surface cross section: plasma sprayed porous Mo

- LLD filled with 67 g-Li by evaporation, (twice that needed to fill the porosity).
- No major Mo or macroscopic Li influx observed even with strike point on LLD.
- No lithium ejection events from LLD observed during NSTX transients > 100 kA/m^2
  - Thin layers and small pore diameters increase critical current (J_{crit}) for ejection.
  - Modelling consistent with DIII-D Li-DIMES ejection at 10kA/m^2 and NSTX experience.





M.A. Jaworski, et al., J. Nucl. Mater. 415 (2011) S985.
D. Whyte, et al., Fusion Eng. Des. 72 (2004) 133.

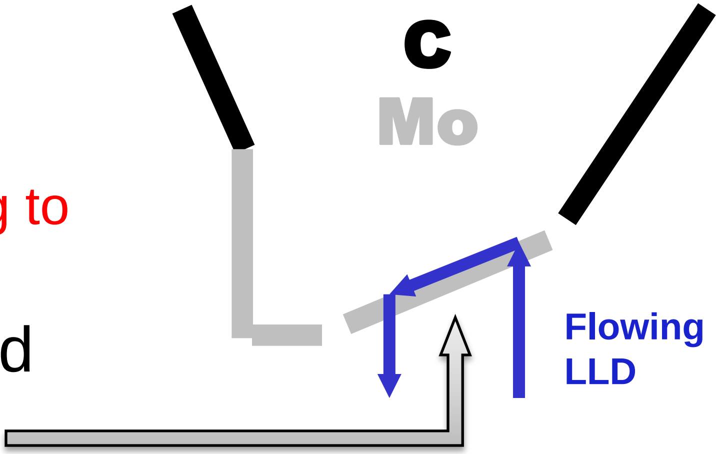
Flowing LLD will be studied as alternative means of particle and power exhaust, access to low recycling

- LLD, LTX → liquid Li required to achieve pumping persistence
  - Flowing Li required to remove by-products of reactions with background gases
- Substantial R&D needed for flowing Li
- Need to identify optimal choice of concept for pumping, power handling:
  - Slow-flowing thin film (FLiLi)
  - Capillary porous system (CPS)
  - Lithium infused trenches (LiMIT)

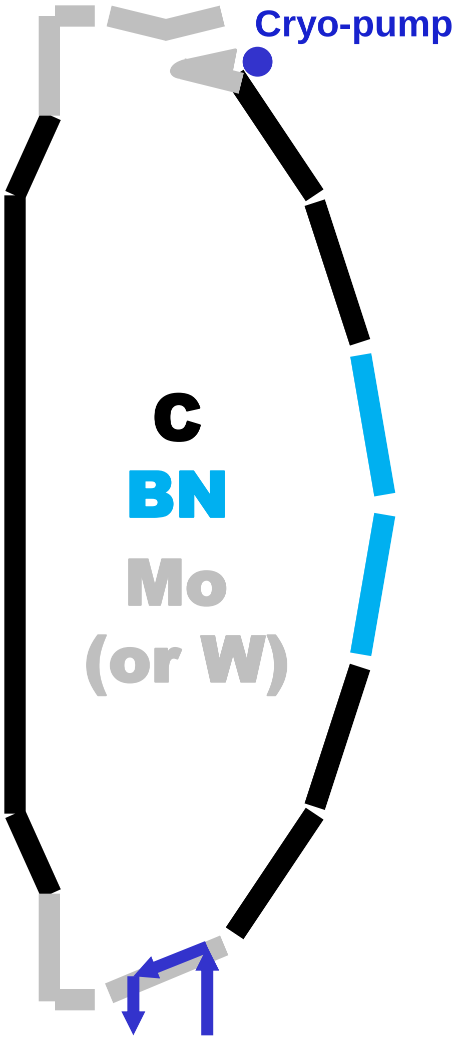
  All systems above require active cooling to mitigate highest heat fluxes of NSTX-U
- Elimination of C from divertor needed for "clean" test of LLD D pumping
  - May need to remove all C PFCs?

Possible approach:

- Dedicate 1-2 toroidal sectors (30-60˚ each) to LLD testing (and/or integrate with RDM?)
- Test several concepts simultaneously
- Full toroidal coverage after best concept is identified



Direct comparison of cryo-pumping and flowing LLD by end of next 5 yr plan would inform FNSF divertor decisions

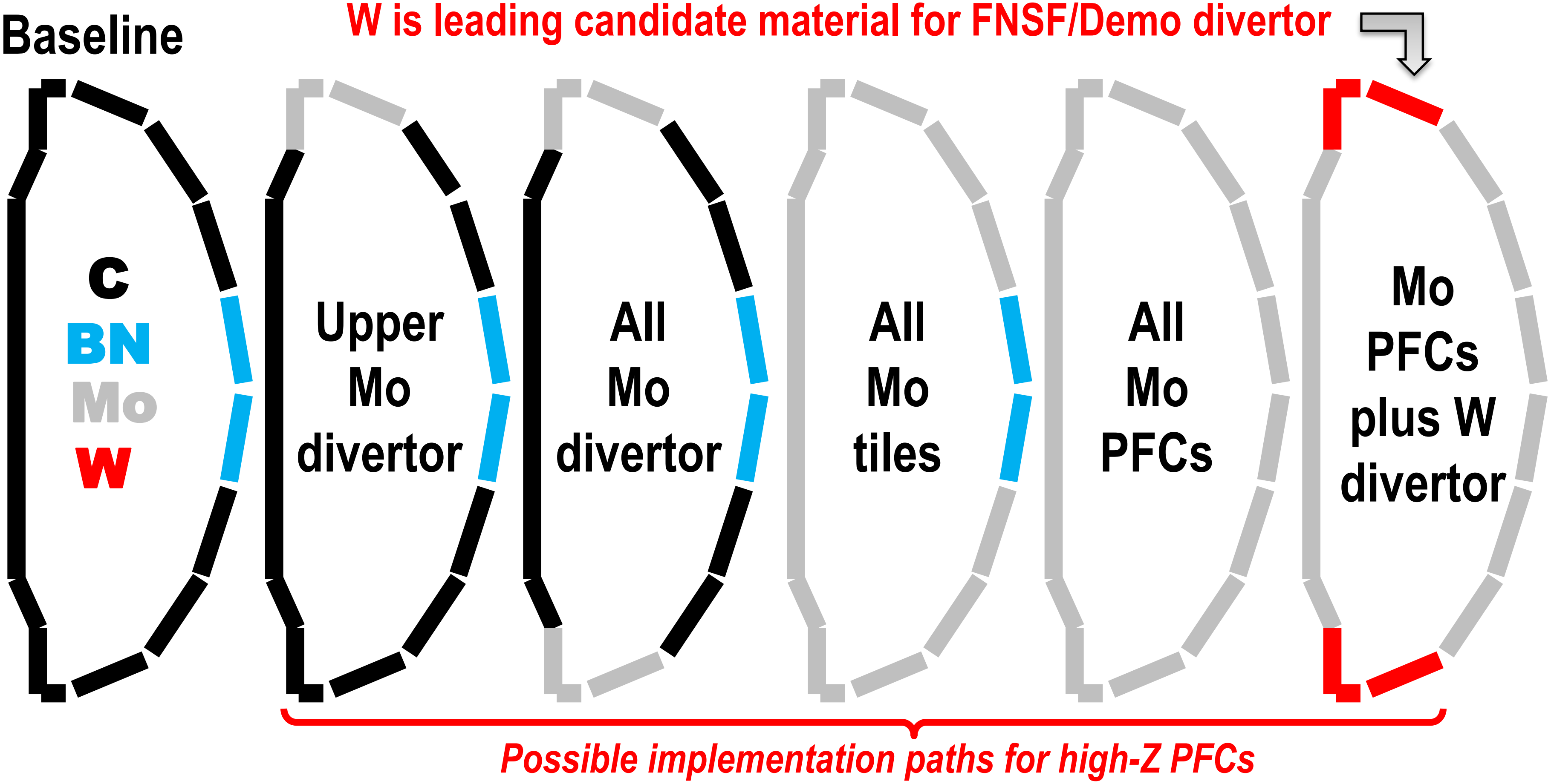


- Partially-detached snowflake + cryo-pump may provide sufficient heat-flux mitigation and particle control for NSTX-U, FNSF
- However, erosion of solid PFCs could pollute plasma, damage FNSF divertor/FW
  - FNSF at 30% duty factor → ~10^2 - 10^3 kg net erosion / year for typical FNSF size & power
  - Further motivates research in flowing liquid metals
- 5 year plan for divertors (present thinking):
  - Dedicate upper divertor to cryo-pump
  - Dedicate lower divertor to flowing liquid Li tests, materials analysis particle probe (MAPP)

Flowing LLD, MAPP probe, possible replaceable divertor module (RDM)

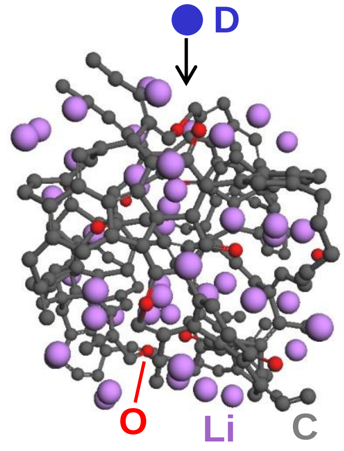
NSTX-U 5 year plan goal: transition to (nearly) complete wall coverage w/ metallic PFCs to support FNSF PMI studies

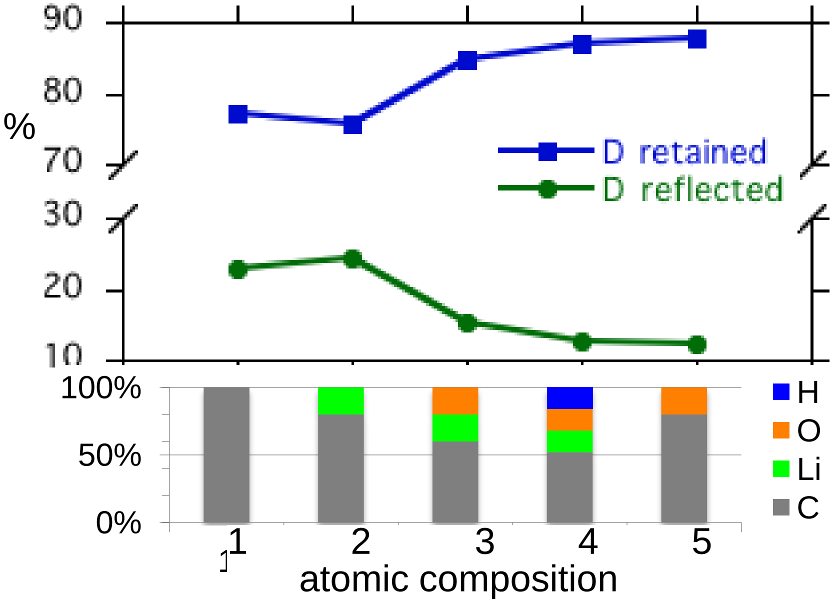
- **Assess compatibility of high τ_E and β + 100% NICD with metallic PFCs**



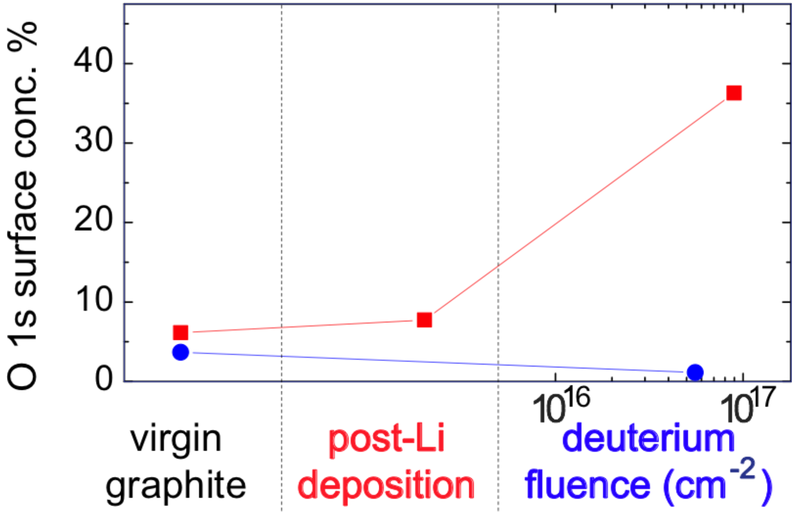
Simulations + lab results show importance of O in Li PMI

Quantum-classical atomistic simulations show that surface oxygen plays a key role in the deuterium retention in graphite. [ORNL, submitted to Nature Comm.]





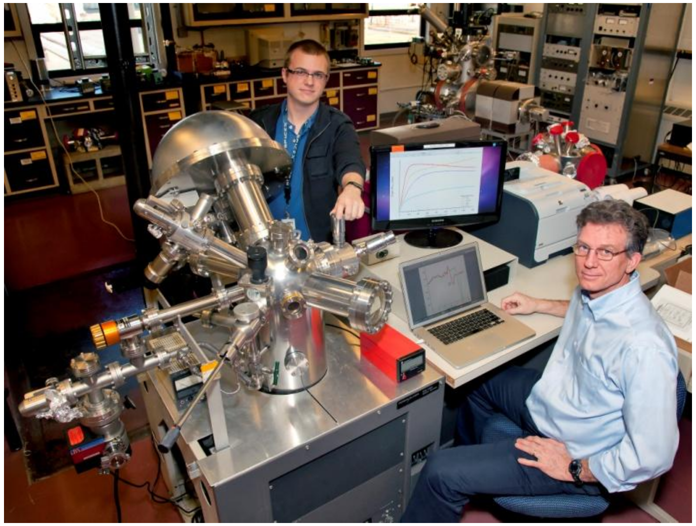
- XPS measurements *(Purdue)* show that 2 µm lithium increases the surface oxygen content of lithiated graphite to about 10%.
- Deuterium ion irradiation of lithiated graphite greatly enhances the oxygen content to 20% - 40%.
- In stark contrast, D irradiation of a graphite sample without lithium actually *decreases* the amount of O on the surface.



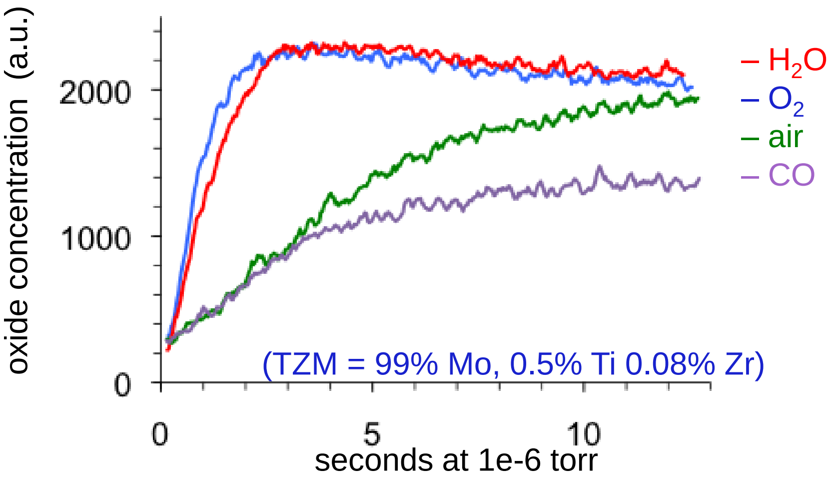
- Result explains why Li on C pumps D so effectively

PPPL/PU collaboration shows lithium reacts quickly with residual gases

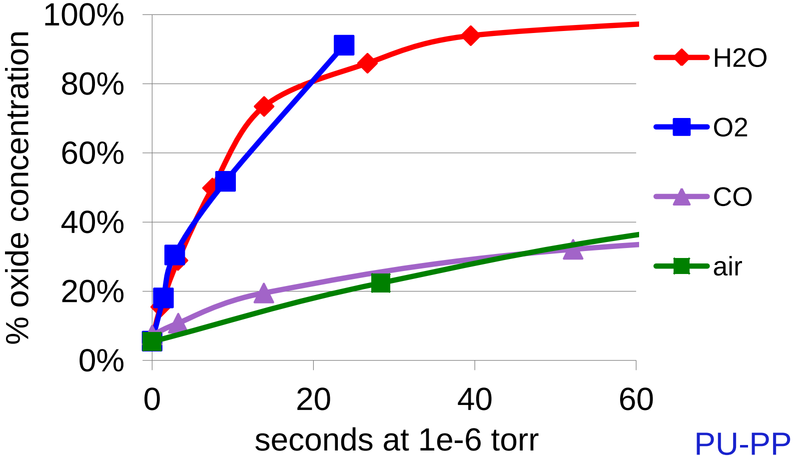
New Surface Analysis Labs at PPPL



Li surface oxidation time



1 monolayer of lithium on TZM Mo (AES O 1s)



Solid lithium (XPS Li 1s))

PU-PPPL

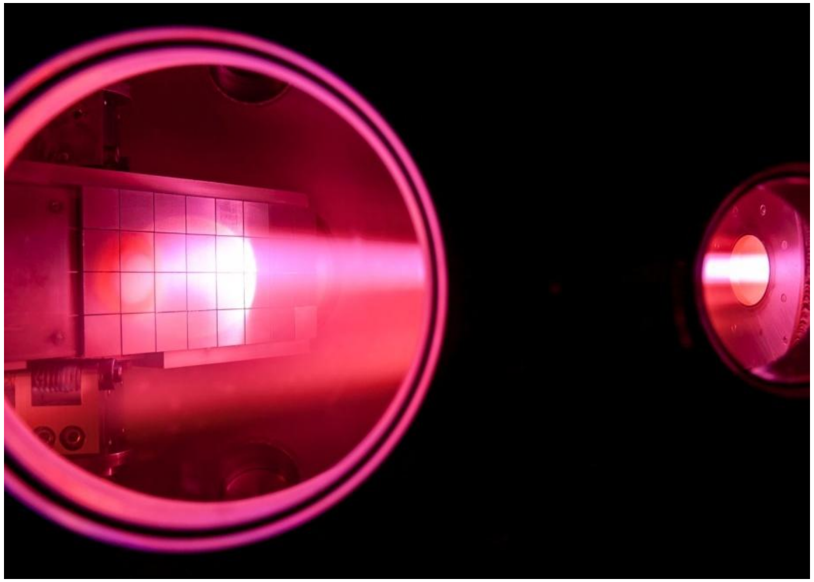
- Surface analysis experiments show PFC oxide coverage is expected in 10s of seconds from residual H_2O at typical NSTX intershot pressures ~1e-7 torr.
- Plasma facing surface after Li evaporation is a mixed material rather than ‘lithium coating’.
- **Short reaction times motivate flowing Li PFCs**

Prototype Li-PFC materials testing at Magnum-PSI

NSTX-U PFCs (ATJ, TZM (Mo), W) will be tested with and without Li coatings at NSTX-U pulse lengths and power levels with extensive diagnostics

Planned investigations:

- Li coating lifetime
- Hydrogenic recycling/retention as a function of exposure time & temperature.
- Erosion, migration, impurity production with and without lithium.



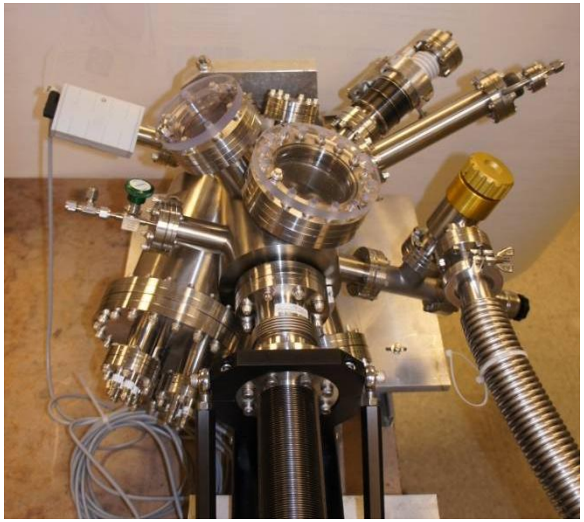
Magnum-PSI parameters relevant to NSTX-U

- 1.4 T for 12 s
- 10 MW/m^2
- $N_e \sim 1.2 \times 10^{20}$ m^{-3}
- $T_e \sim 3$ eV
- Bias ≤ 100 V
- Extensive diagnostics

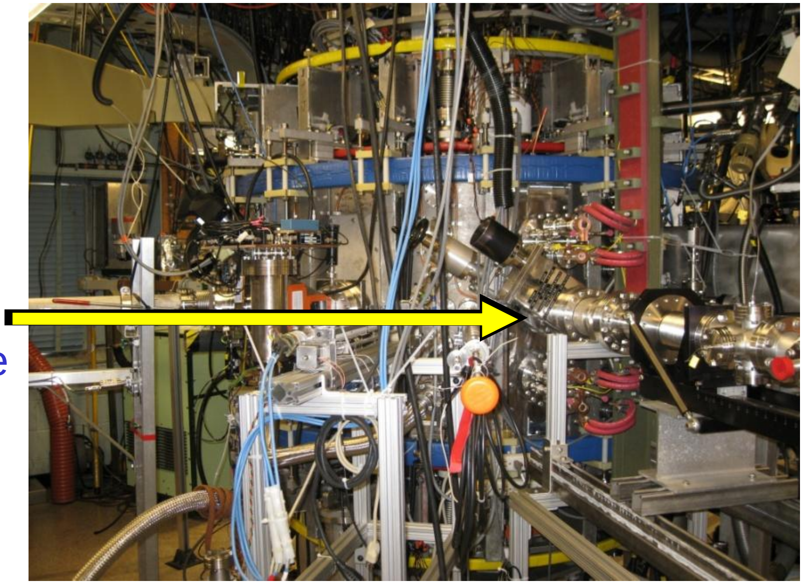
LTX is providing all-metal-wall tokamak investigating Li chemistry, temperature, thickness…

- Lithium Tokamak Experiment has: 1. 120 cm^2 Li-filled dendritic W limiter heatable ≤ 500 C
  2. Thick (>100 micron) evaporated Li films on 3,000 – 5,000 cm^2 upper heated liner
  3. Few hundred cm^3 pool of liquid Li in the lower shells (total ≤ 85% of plasma surface)
- Will investigate plasma-surface interactions, Li influx vs. temp., confinement, Te profile, liquid metal flows in B fields up to 0.3T
- Materials Analysis and Particle Probe (MAPP) will be used first on LTX in support of NSTX milestone R(13-2): "*Investigate relationship between lithium-conditioned surface composition and plasma behavior*" and transferred to NSTX-U later.
- MAPP's innovative design enables sample exposure to plasma and inter-shot surface analysis.

MAPP



MAPP will be installed on midplane LTX port



Lab-based R&D on liquid metal technology will inform long term PFC decisions:

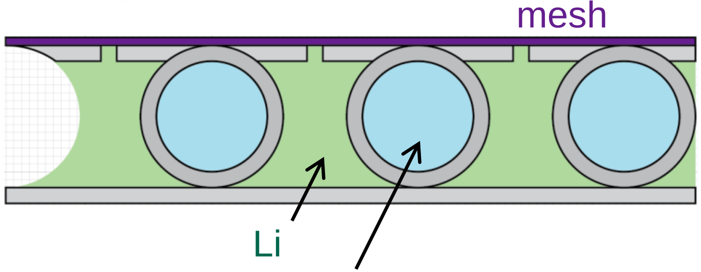
Pre-NSTX-U restart R&D initiated by PPPL:

1. Laboratory studies of D uptake as a function of Li dose, C/Mo substrate, surface oxidation, wetting…
2. Tests of prototype of scalable flowing liquid lithium system (FliLi) at PPPL and on HT7 →
3. Basic liquid lithium flow loop on textured surfaces
4. Analysis and design of actively-cooled PFCs with Li flows due to capillary action and thermoelectric MHD
5. Magnum-PSI tests begin June 2012

- Four proposals on Li-PFCs submitted to OFES Materials Solicitation to extend above work.
- Preparing for upcoming international collaboration solicitation, which will include possible tests of Li PFCs on HT-7 and EAST

Thin flowing Li film in FLiLi (Zakharov)

Soaker hose capillary porous system concept (Goldston)



Coolant
(He, supercritical CO_2, Na heatpipe…)

NSTX/EAST lithium collaborations

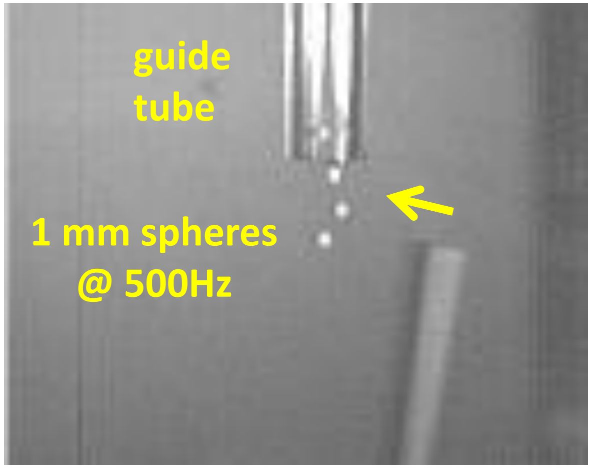
EAST is only other divertor H-mode facility using Li

- NSTX Li powder dropper achieved 1st H mode on EAST and drastically reduced MHD (in backup).
- 2nd dropper being built by ASIPP.
- Li granule injector to be installed on EAST midplane - will be used to trigger ELMs and control MHD

Plans:

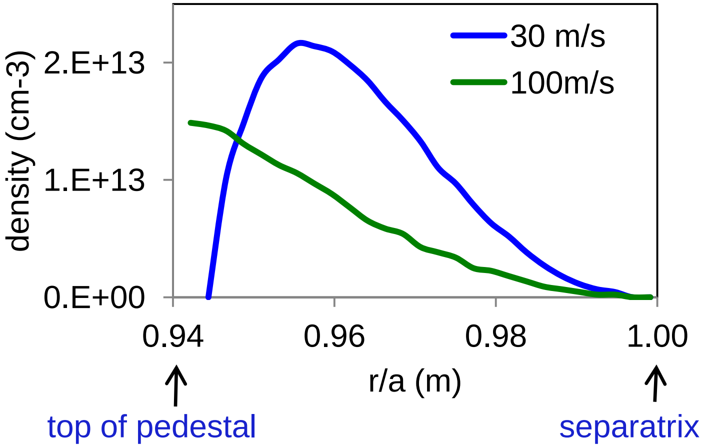
- Assess interplay between cryo-pumping and lithiumization, and high- Z PFC interactions/synergies with lithium
- Study effects of Li on thermal and particle transport, further develop sustained/long-pulse lithium delivery systems (Li injector, dropper)

Continuous Li delivery may be essential for long pulses.



Lithium granules injected using 95 m/s “impeller”

Li deposited between pedestal and separatrix



PPPL Lithium Granule Injector Tested on EAST

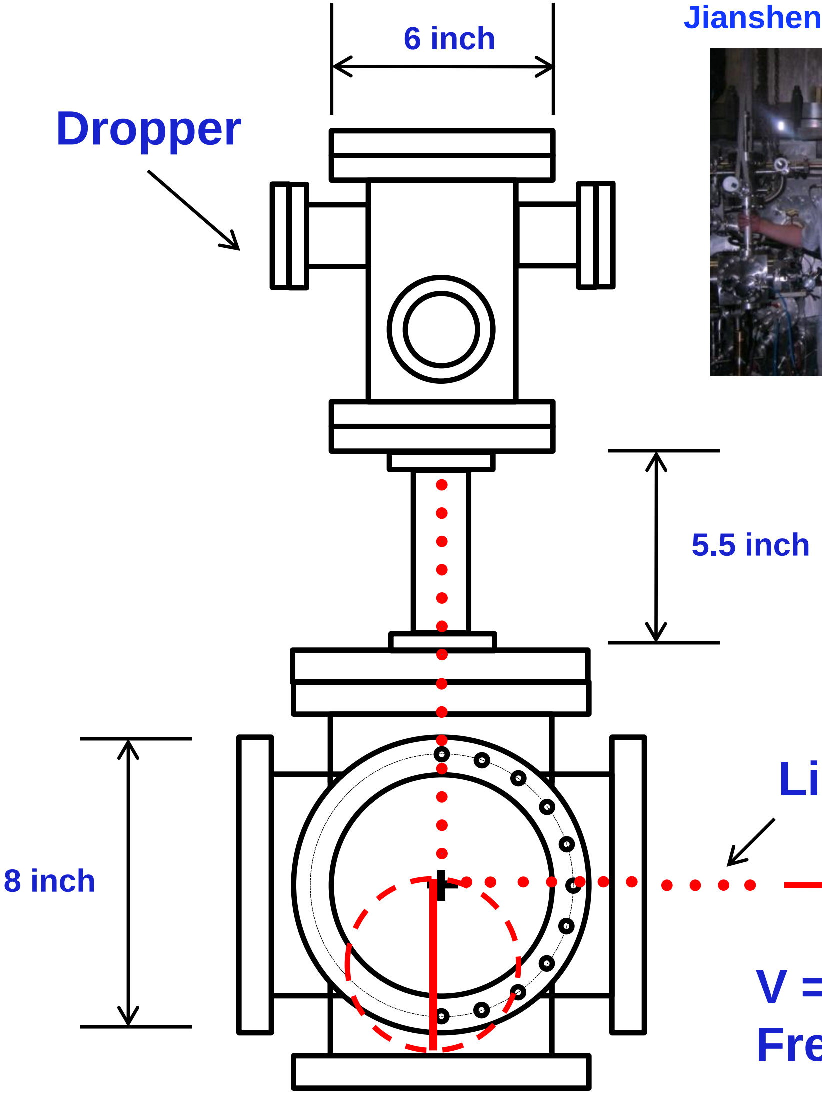
Dennis Mansfield (PPPL, retired)
Lane Roquemore (PPPL)

Independent Control:

Granule Size
(change between shots)

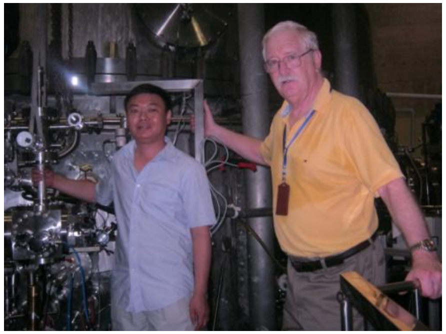
Injection Speed
(ramp during shots)

Pacing Frequency
(ramp during shots)



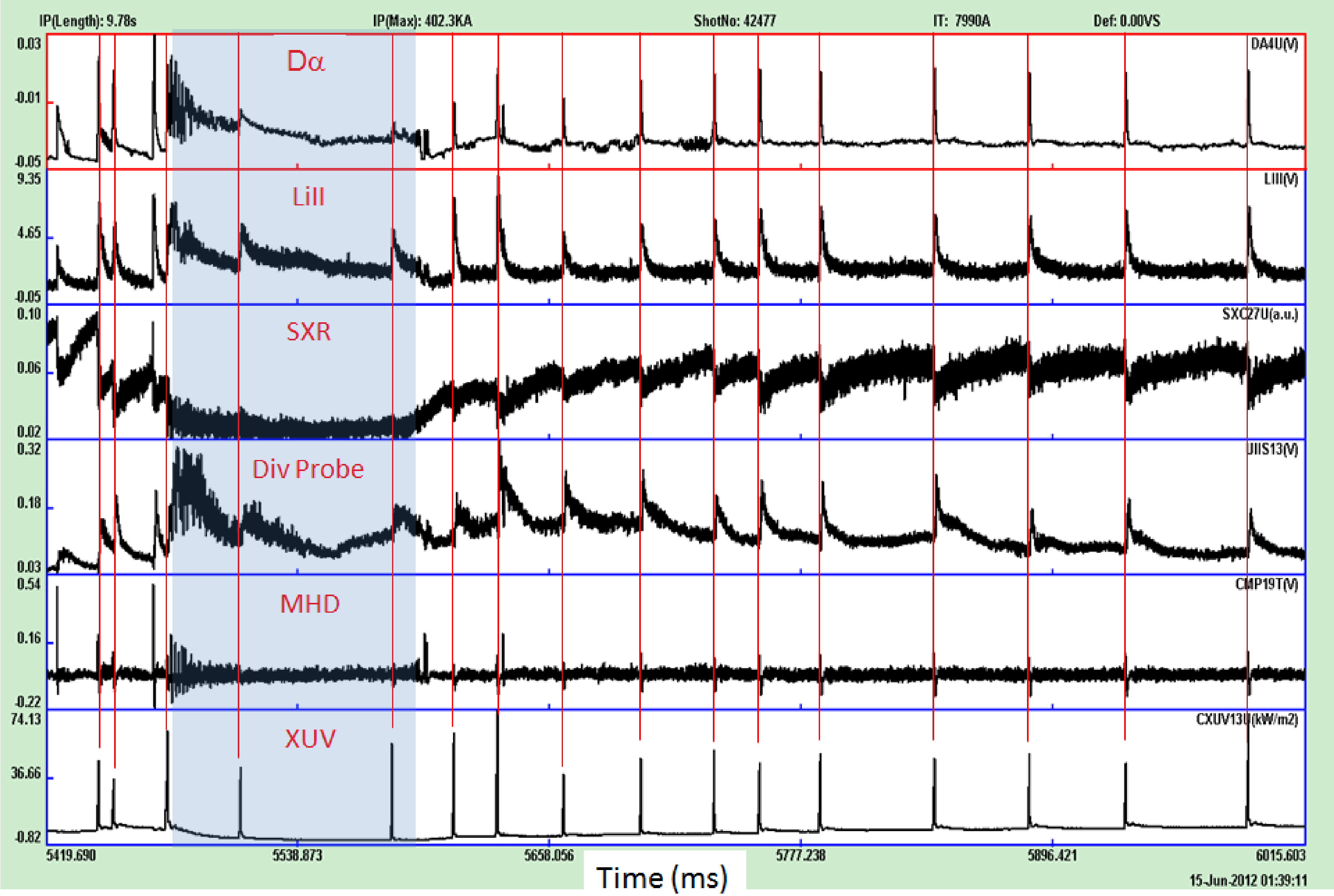
V = 0 - 100 m/s
Freq = 0 - 100 Hz

Jiansheng Hu and D. Mansfield



Triggered ELMs (~ 25 Hz) with 0.7 mm Li Granules @ ~ 45 m/s

→ could be very useful for triggering ELMs in Li-ELM free H-modes in NSTX-U



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