Progress on developing the spherical tokamak for fusion applications

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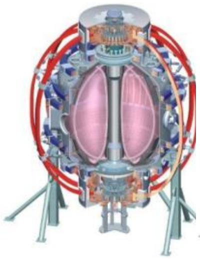
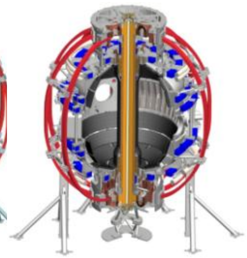
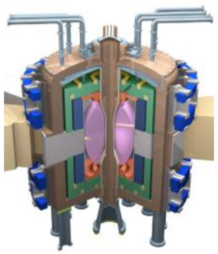
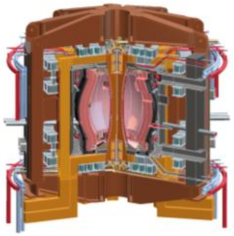
Fusion applications of low-A spherical tokamak (ST)

- Develop plasma-material-interface (PMI) solutions for next-steps
  - Exploit high divertor heat flux from lower-A/smaller major radius
- Fusion Nuclear Science/Component Test Facility (FNSF/CTF)
  - Exploit high neutron wall loading for material and component development
  - Utilize modular configuration of ST for improved accessibility, maintenance
- Extend toroidal confinement physics predictive capability
  - Access strong shaping, high β, $v_{fast} / v_{Alfvén}$, and rotation, to test physics models for ITER and next-steps (see NSTX, MAST, other ST presentations)
- Long-term: reduced-mass/waste low-A superconducting Demo

<u>This talk</u>:

- **Planned capabilities and construction progress of NSTX Upgrade**
- **Mission and configuration studies for ST-based FNSF/CTF**

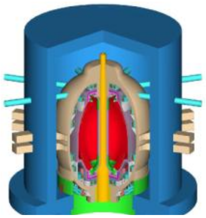
NSTX Upgrade will access next factor of two increase in performance to bridge gaps to next-step STs

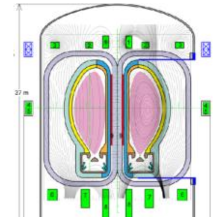
Parameter	NSTX	**NSTX Upgrade**	Fusion Nuclear Science Facility	Pilot Plant
Major Radius R_0 [m]	0.86	**0.94**	1.3	1.6 – 2.2
Aspect Ratio R_0 / a	≥ 1.3	**≥ 1.5**	≥ 1.5	≥ 1.7
Plasma Current [MA]	1	**2**	4 – 10	11 – 18
Toroidal Field [T]	0.5	**1**	2 – 3	2.4 – 3
Auxiliary Power [MW]	≤ 8	**≤ 19***	22 – 45	50 – 85
P/R [MW/m]	10	**20**	30 – 60	70 – 90
P/S [MW/m^2]	0.2	**0.4**	0.6 – 1.2	0.7 – 0.9
Fusion Gain Q			1 – 2	2 – 10

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

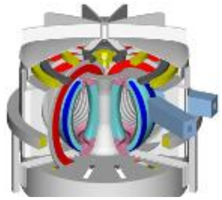
Low-A Power Plants



ARIES-ST (A=1.6)



JUST (A=1.8)



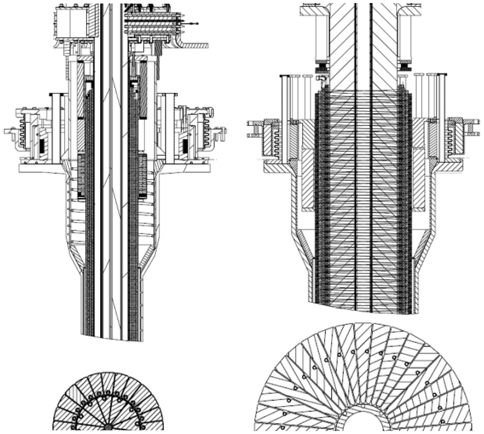
VECTOR (A=2.3)

Key issues to resolve for next-step STs

- **Confinement scaling (electron transport)**
- **Non-inductive ramp-up and sustainment**
- **Divertor solutions for mitigating high heat flux**
- **Radiation-tolerant magnets (for Cu TF STs)**

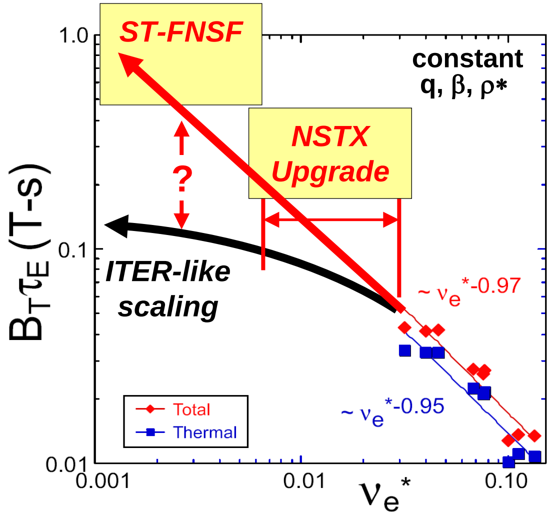
NSTX Upgrade will address critical plasma confinement and sustainment questions by exploiting 2 new capabilities

Previous center-stack | New center-stack

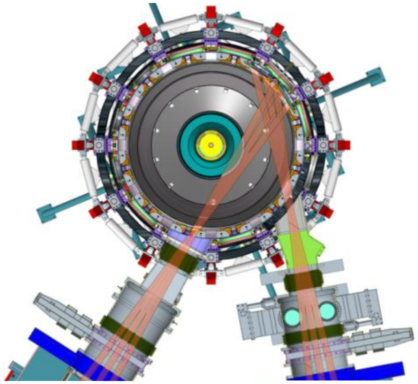


TF OD = 20cm | TF OD = 40cm

- 2x 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



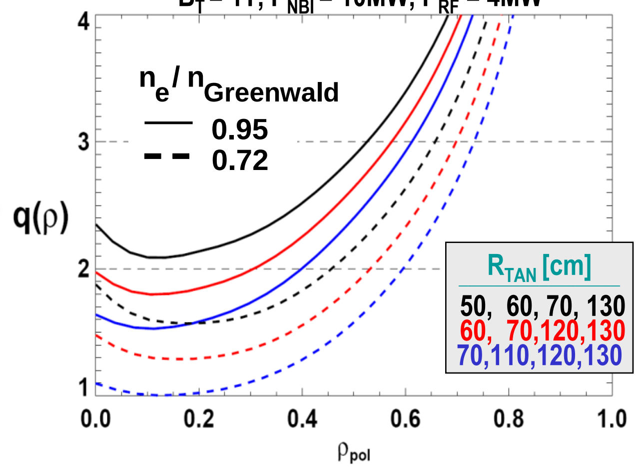
Normalized e-collisionality $\nu_e^* \propto n_e / T_e^2$



Present NBI | New 2nd 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

I_P=0.95MA, H_{98y2}=1.2, β_N=5, β_T = 10%
B_T = 1T, P_{NBI} = 10MW, P_{RF} = 4MW



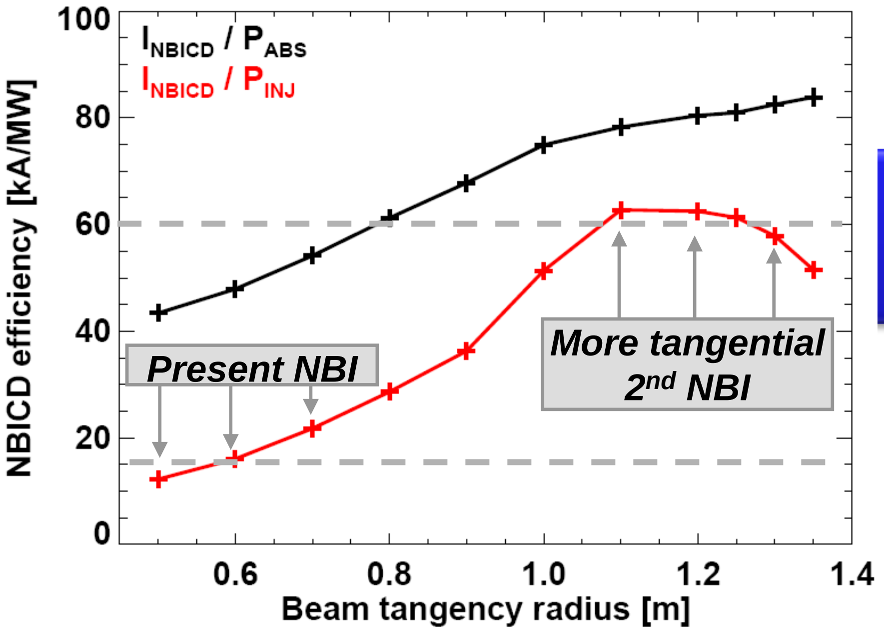
J. Menard, et al., Nucl. Fusion 52 (2012) 083015

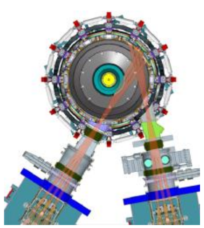
Non-inductive ramp-up from ~0.4MA to ~1MA projected to be possible with new centerstack (CS) + more tangential 2^{nd} NBI

- 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

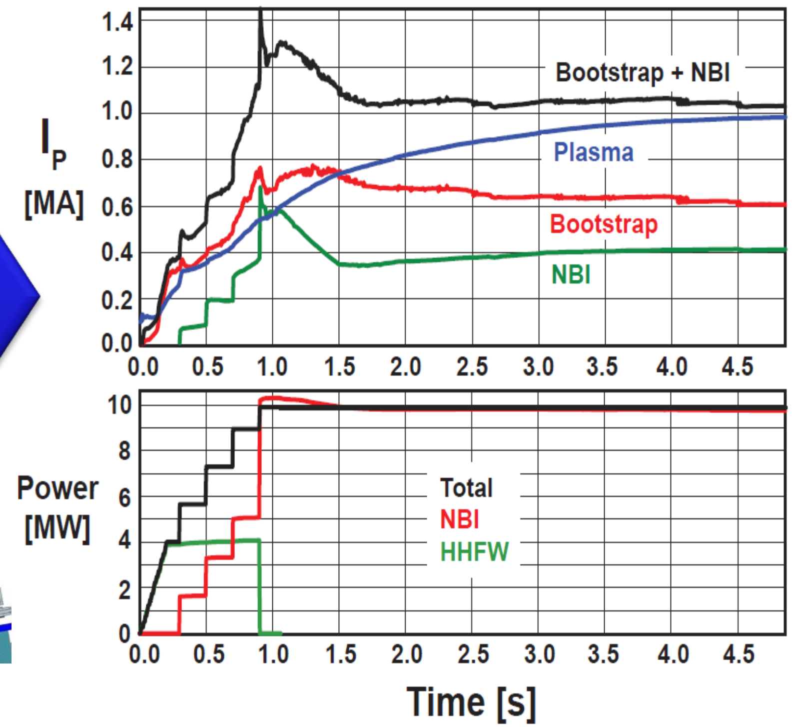
E_{NBI}=100keV, I_P=0.40MA, f_{GW}=0.62

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



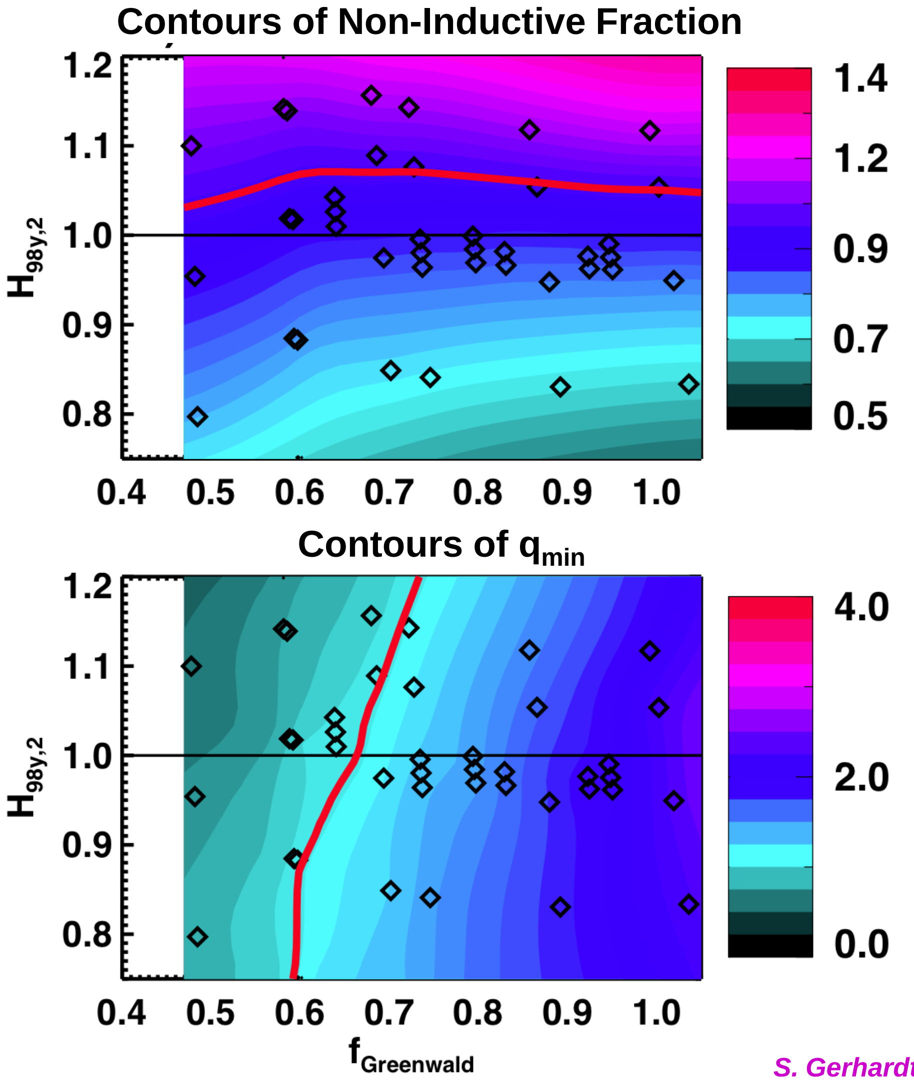


TSC simulation of non-inductive ramp-up from I_P = 0.1MA, T_e=0.5keV target at B_T=1T



100% non-inductive operating points projected for a range of toroidal fields, densities, and confinement levels

B_T = 1.0 T, I_P = 1MA, P_{inj} = 12.6MW



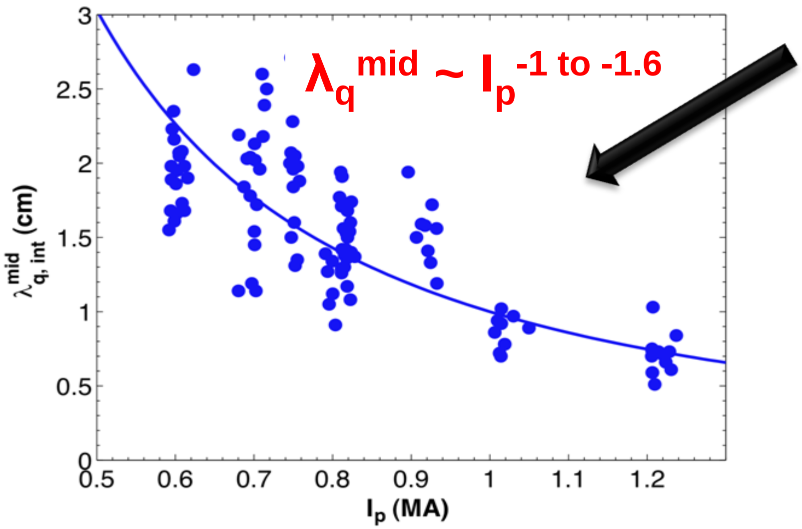
Projected Non-Inductive Current Levels for κ~2.85, A~1.75, f_{GW}=0.7

B_T [T]	P_{inj} [MW]	I_P [MA]
0.75	6.8	0.6-0.8
0.75	8.4	0.7-0.85
1.0	10.2	0.8-1.2
1.0	12.6	0.9-1.3
1.0	15.6	1.0-1.5

- From GTS (ITG) and GTC-Neo (neoclassical):
  - $\chi_{i,ITG}/\chi_{i,Neo} \sim 10^{-2}$
  - Assumption of neoclassical ion thermal transport should be valid

S. Gerhardt, et al., Nucl. Fusion 52 (2012) 083020

NSTX-U will investigate detachment and high-flux-expansion “snowflake” divertor for heat flux mitigation

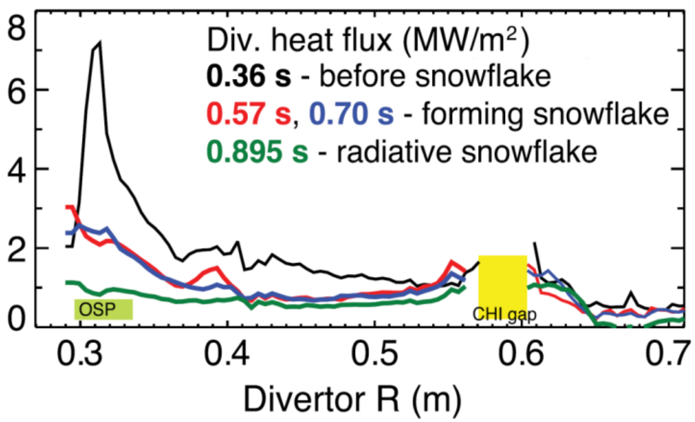


- Divertor heat flux width decreases with increased plasma current I_P

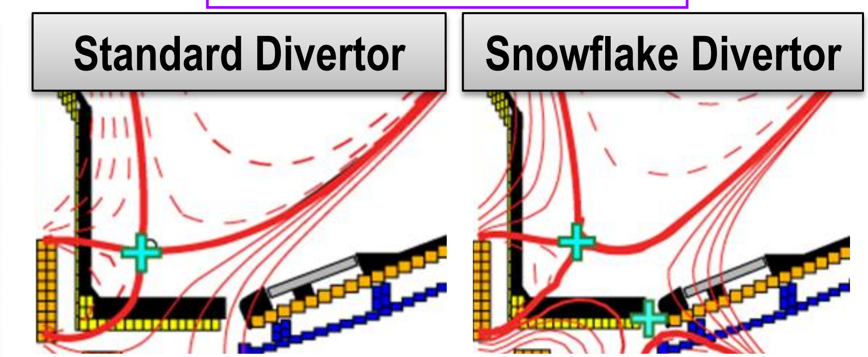
  → 30-45MW/m^2 in NSTX-U with conventional LSN divertor at full current and power

- Can reduce heat flux by 2-4 × in NSTX via partial detachment at sufficiently high f_{rad}

NSTX data



Soukhanovskii EX/P5-21



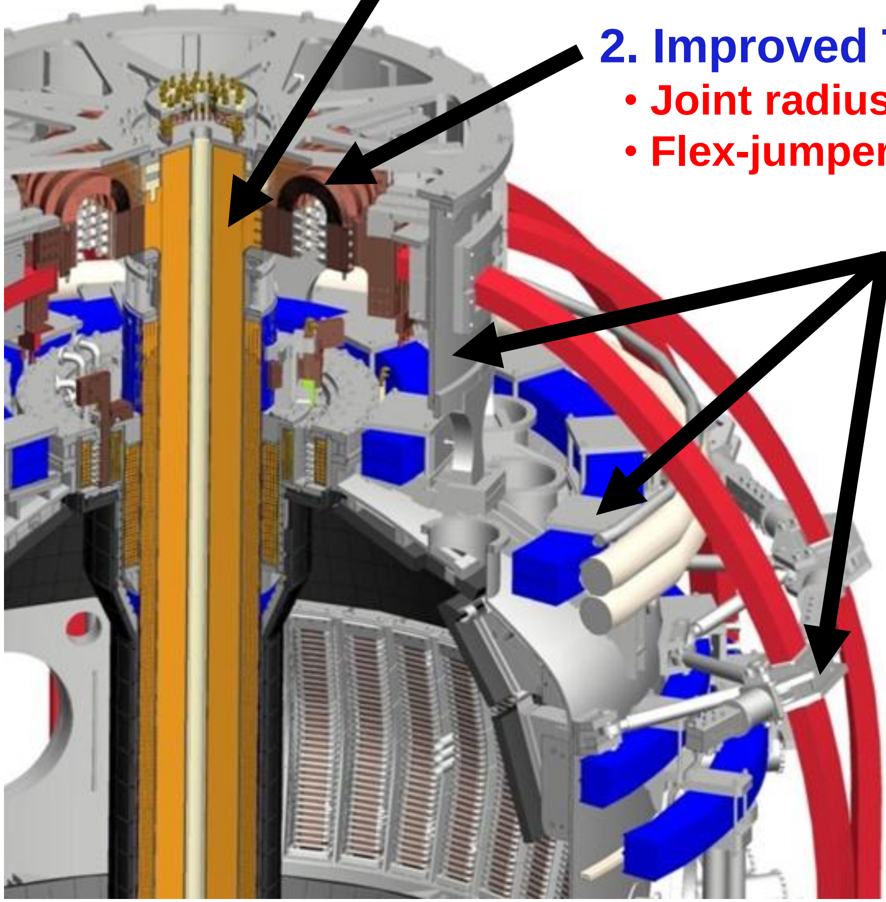
Snowflake→ high flux expansion = 40-60 lowers incident $q_{\perp}$, promotes detachment

NSTX-U: U/D balanced snowflake has < 10MW/m^2 at I_P = 2MA, P_{AUX}=10-15MW

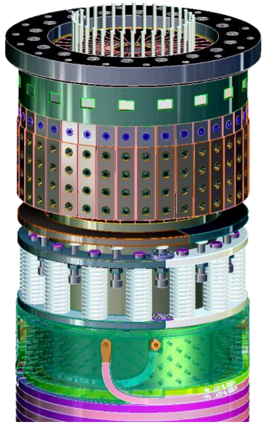
Major engineering challenge of NSTX Upgrade:
Field and current each increase 2× → E-M forces increase 4×

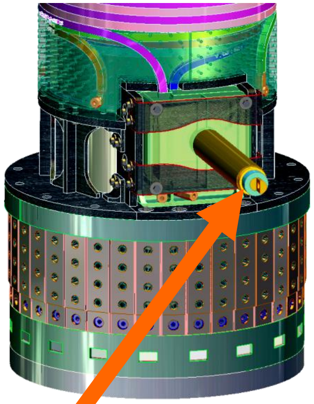
Design solutions for increased loads:

1. Simplified inner TF design
   - Single layer of TF conductors
2. Improved TF joint design
   - Joint radius increased → lower B
   - Flex-jumper improved
3. Reinforcements:
   - Umbrella structure
   - PF, TF coil supports



Upper TF/ OH Ends

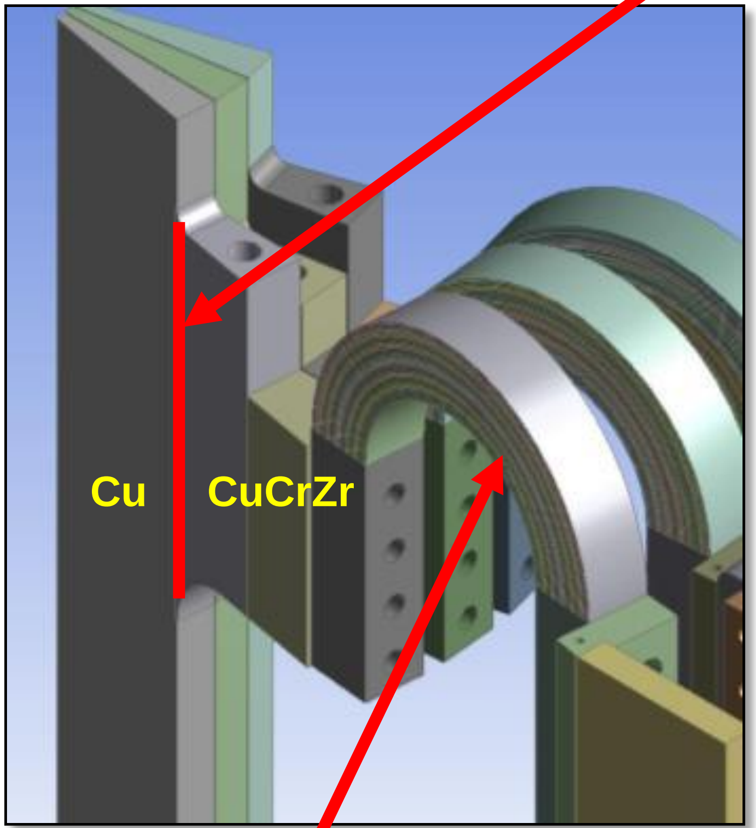


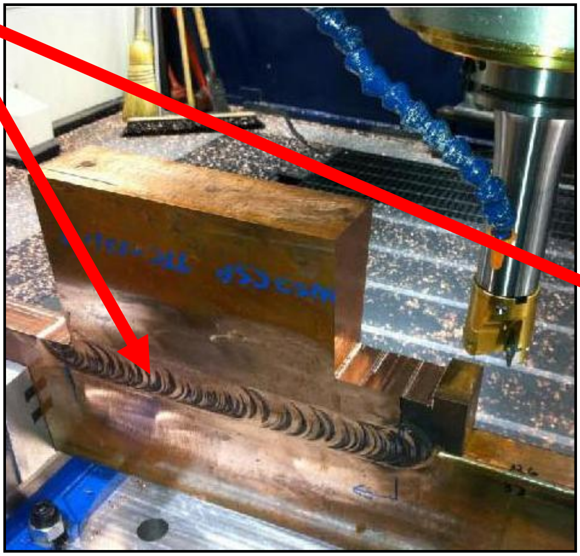


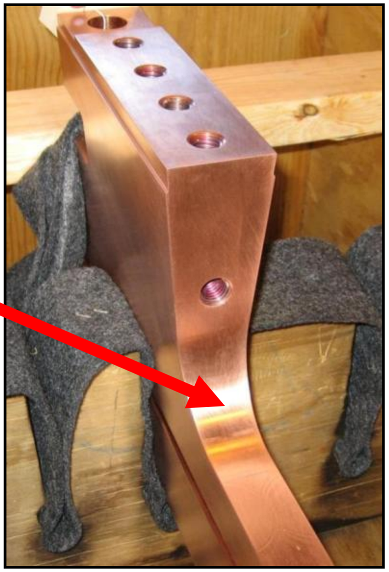
4. OH leads placed at bottom, made coaxial to minimize forces, error-fields

Substantial R&D completed to achieve higher toroidal field with new center-stack

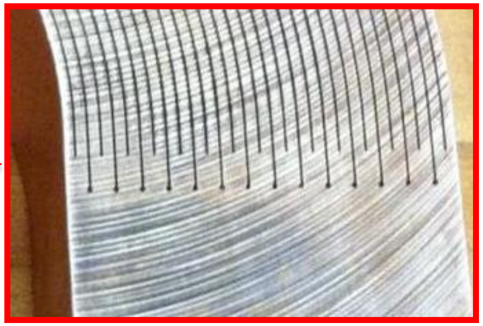
Friction-stir welded joint







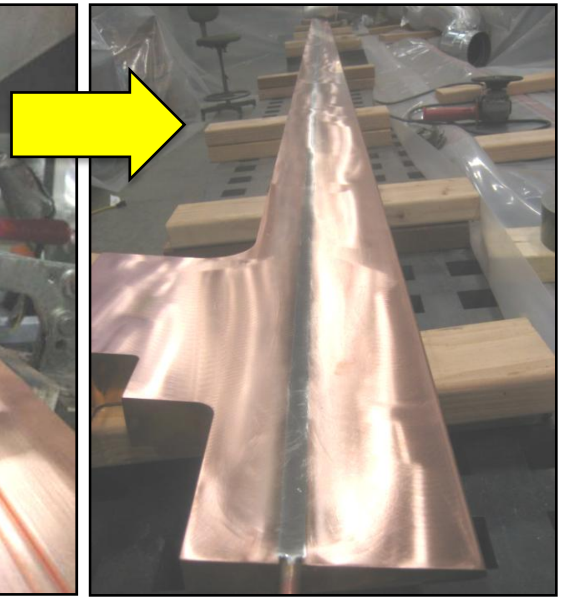
Flexible TF strap



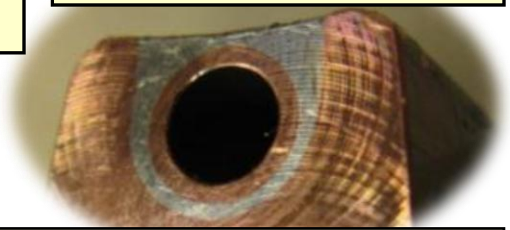
Wire EDM used instead of laminated build

TF cooling tube soldering & flux removal process improved, 1st quadrant of TF bundle to be completed November 2012

Bar placed on heat plate, cooling tube inserted into grove



Bar post-soldering and ground smooth



Close up view of solder joint on test conductor

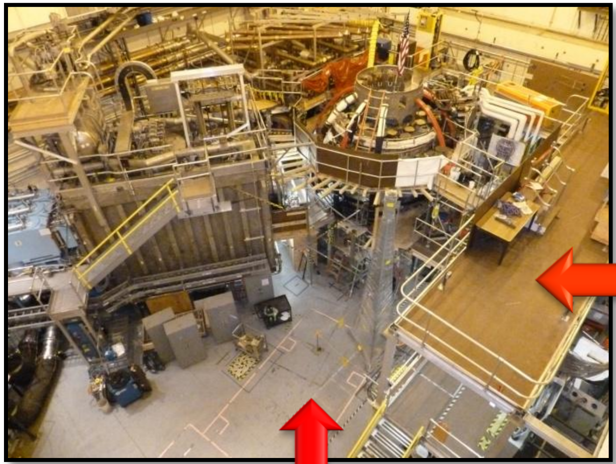
Vacuum-pressure impregnation (VPI) using special cyanate-ester epoxy blend (CTD-425) required for shear strength will be used for the inner TF assembly

Recent successful VPI trials

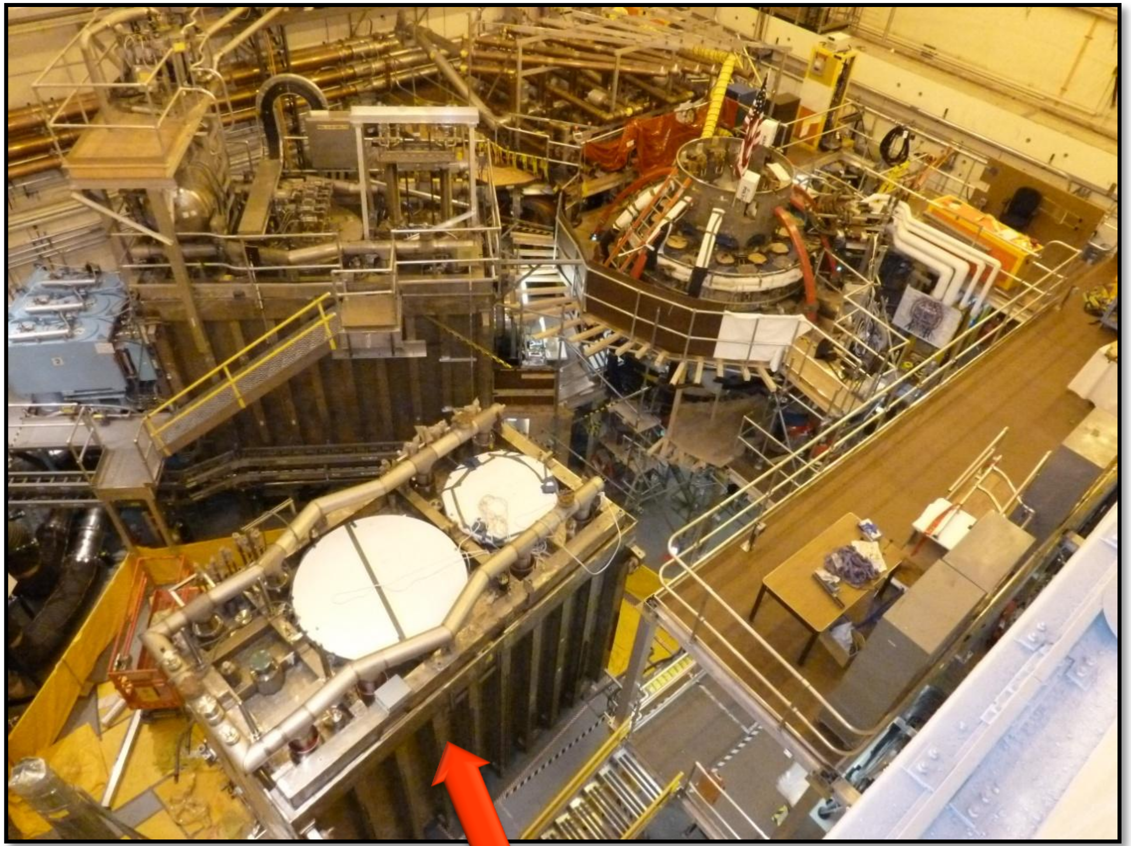
Quadrant mold for VPI nearly ready

Significant progress made during past year to prepare NSTX-U test-cell and 2nd NBI

Oct. 2011: Start of construction



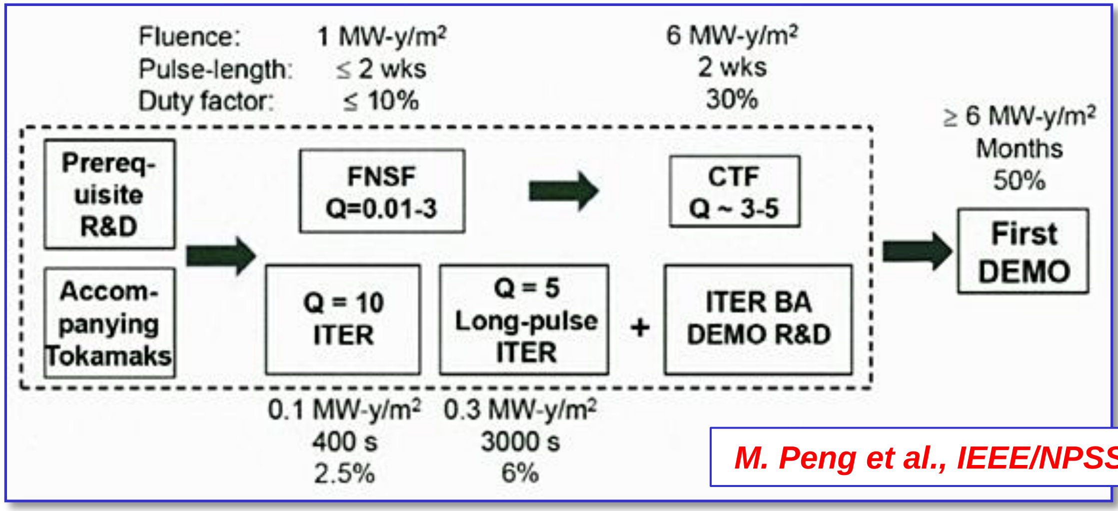
Sept. 2011: NBI space cleared
Upper diagnostic platform installed

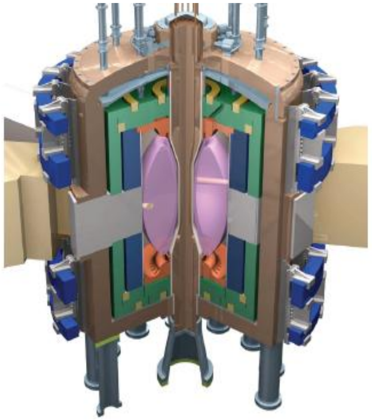


Oct. 2012: 2nd NBI box moved to test cell

Successful operation of NSTX-U (and MAST-U) would provide basis for design and operation of next-step ST

- Present next-step focus is on Fusion Nuclear Science Facility
  - Mission: provide continuous fusion neutron source to develop knowledge-base for materials and components, tritium fuel cycle, power extraction
- FNSF → CTF would complement ITER path to DEMO



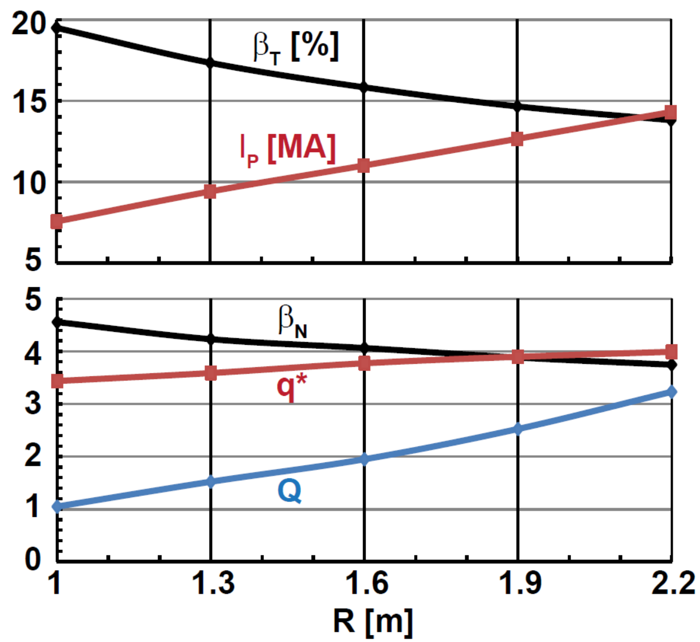


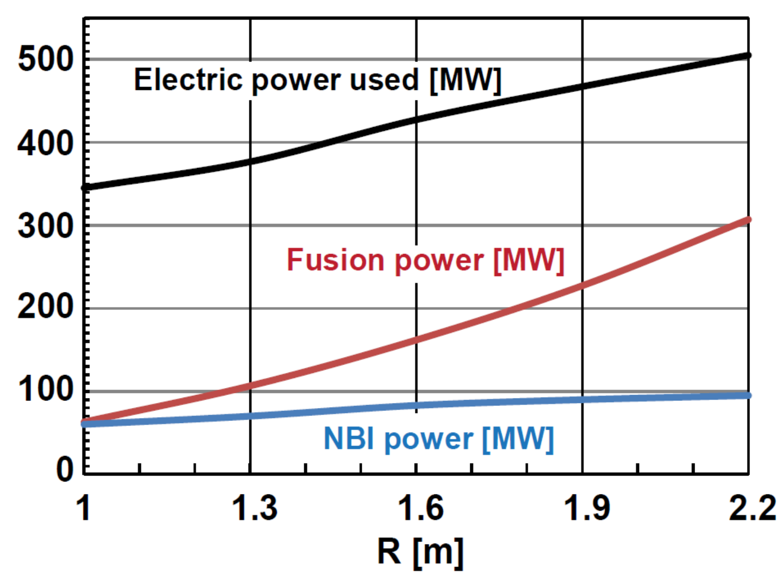
M. Peng et al., IEEE/NPSS Paper S04A-2 - 24th SOFE Conf. (2011)

- Studying wide range of ST-FNSF configurations to identify advantageous features, incorporate into improved ST design
- Investigating performance vs. device size since fusion power, gain, tritium consumption and breeding, … depend on size

Increased device size provides modest increase in stability, but significantly increases tritium consumption

- Scan R = 1m → 2.2m (smallest FNSF → pilot plant with Q_{eng} ~ 1)
- Fixed average neutron wall loading = $1MW/m^2$
- B_T = 3T, A=1.7, κ=3, H_{98} = 1.2, $f_{Greenwald}$ = 0.8
- 100% non-inductive: f_{BS} = 75-85% + NNBI-CD (E_{NBI}=0.5MeV JT60-SA design)

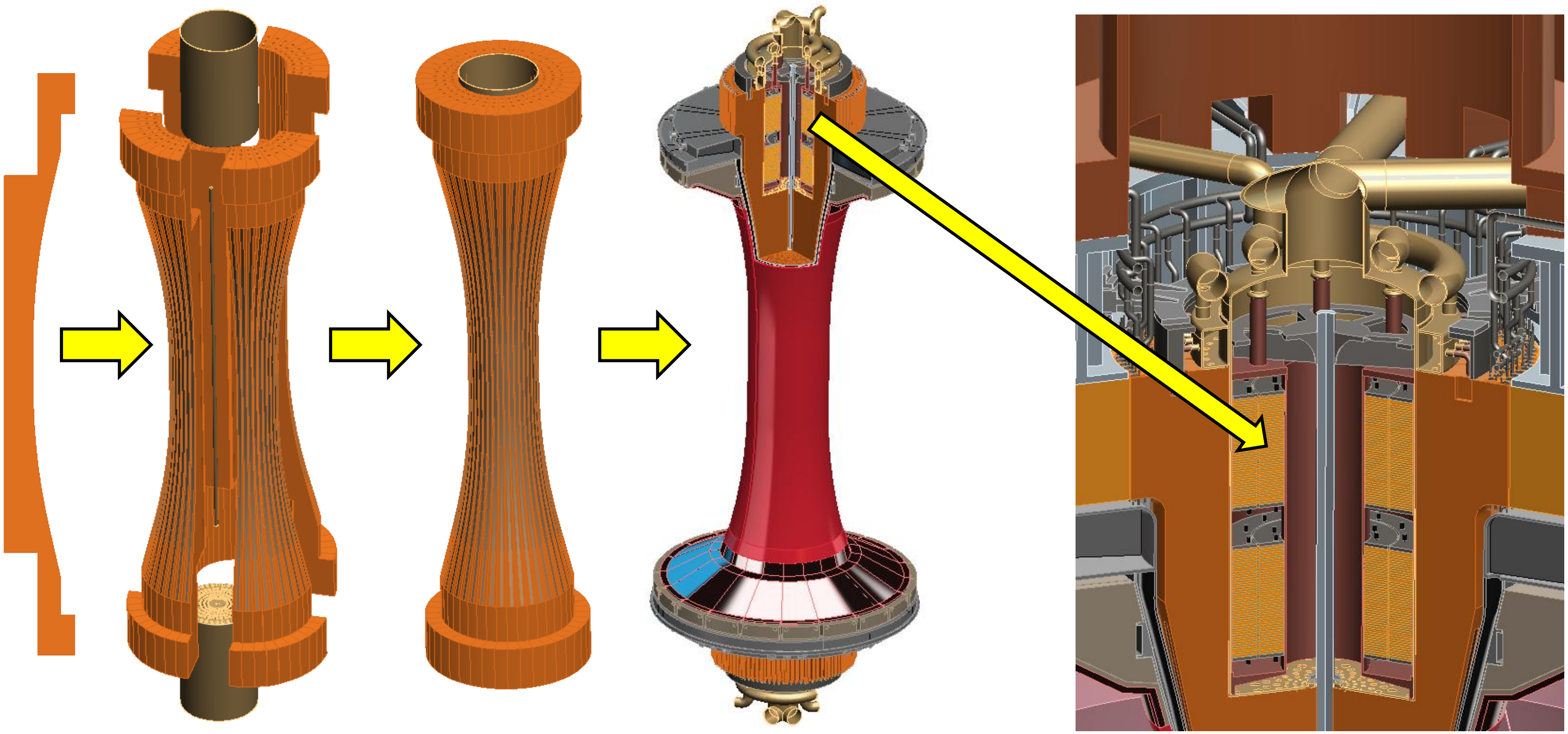




- Larger R lowers β_T & β_N, increases q*
- **Comparable/higher β_T and β_N values already sustained in NSTX**

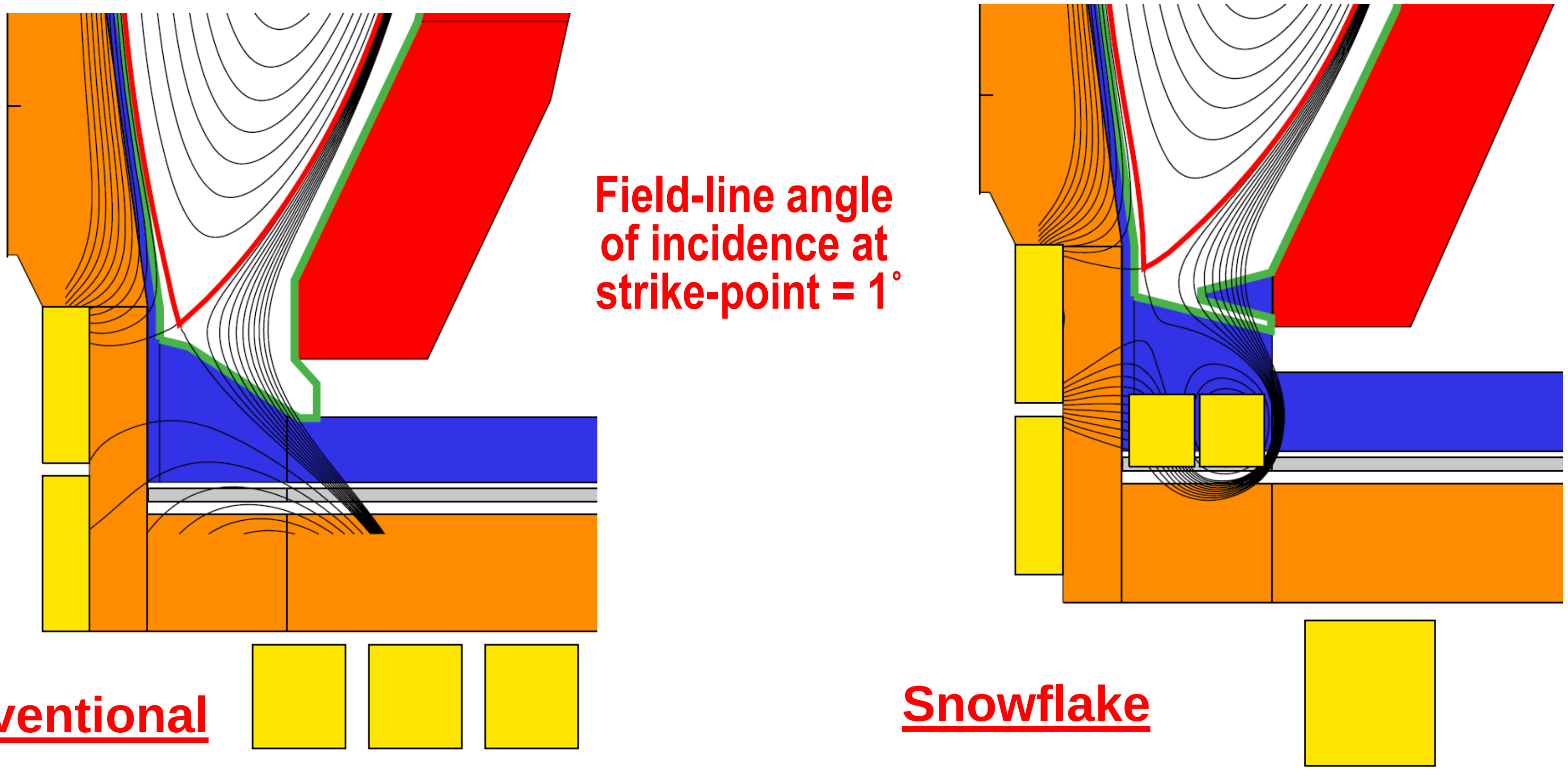
- Q = 1 → 3, P_{fusion} = 60MW → 300MW **→ 5× increase in T consumption**
- 2-3× higher wall loading for CTF/Pilot Plant if β_N → 6, H_{98} → 1.5 (not shown)

FNSF center-stack can build upon NSTX-U design and incorporate NSTX stability results



- **Like NSTX-U, use TF wedge segments (but brazed/pressed-fit together)**
  - Coolant paths: gun-drilled holes or NSTX-U-like grooves in wedge + welded tube
- **Bitter-plate divertor PF magnets in ends of TF enable high triangularity**
  - **NSTX data:** High $\delta > 0.55$ and shaping $S \equiv q_{95}I_P/aB_T > 25$ minimizes disruptivity
  - Neutronics: MgO insulation can withstand lifetime (6 FPY) radiation dose

Divertor PF coil configurations identified to achieve high δ while maintaining peak divertor heat flux < 10MW/m^2



Conventional

- **Flux expansion = 15-25, δ_x ~ 0.55**
- **1/sin(θ_{plate}) = 2-3**
- **Detachment, pumping questionable**
  - Future: assess long-leg, V-shape divertor (JA)

Snowflake

- **Flux expansion = 40-60, δ_x ~ 0.62**
- **1/sin(θ_{plate}) = 1-1.5**
- **Good detachment (NSTX data) and cryo-pumping (NSTX-U modeling)**

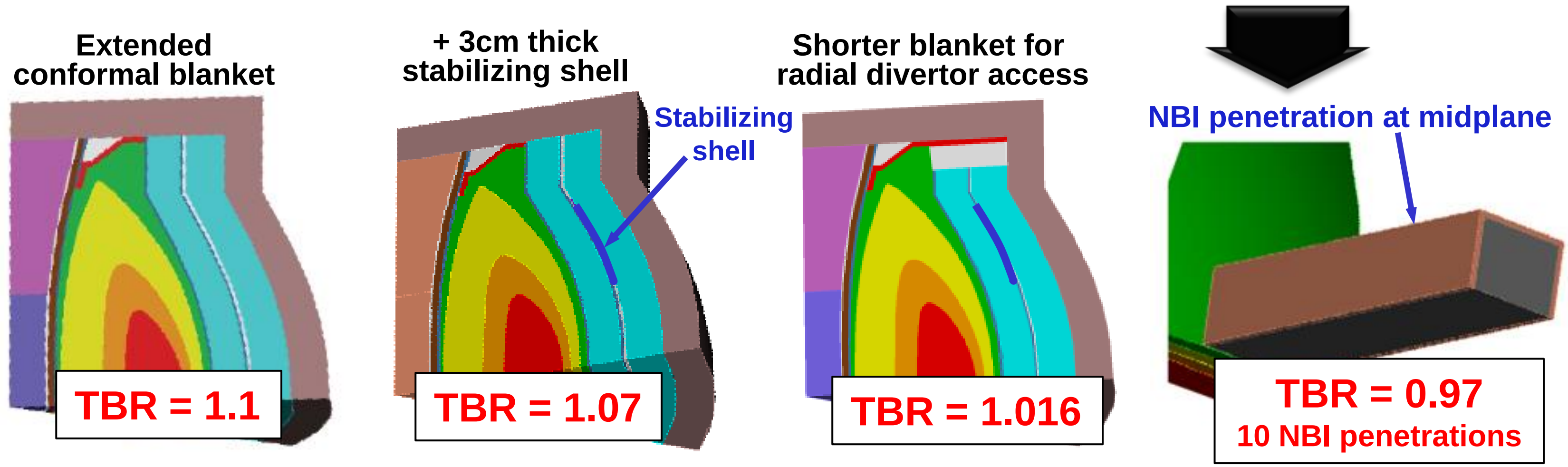
- **Will also test liquid metal PFCs in NSTX-U for power-handling, surface replenishment**

Jaworski EX/P5-31

Cost of tritium and need to demonstrate T self-sufficiency motivate analysis of tritium breeding ratio (TBR)

- Example costs of T w/o breeding at \$0.1B/kg for R=1 → 1.6m
  - FNS mission: 1MWy/m^2 — \$0.33B → \$0.9B
  - Component testing: 6MWy/m^2 — \$2B → \$5.4B
- Implications:
  - TBR << 1 likely affordable for FNS mission with R ~ 1m
  - Component testing arguably requires TBR approaching 1 for all R

- **Initial analysis: R=1.6m ST-FNSF can achieve TBR ~ 1**

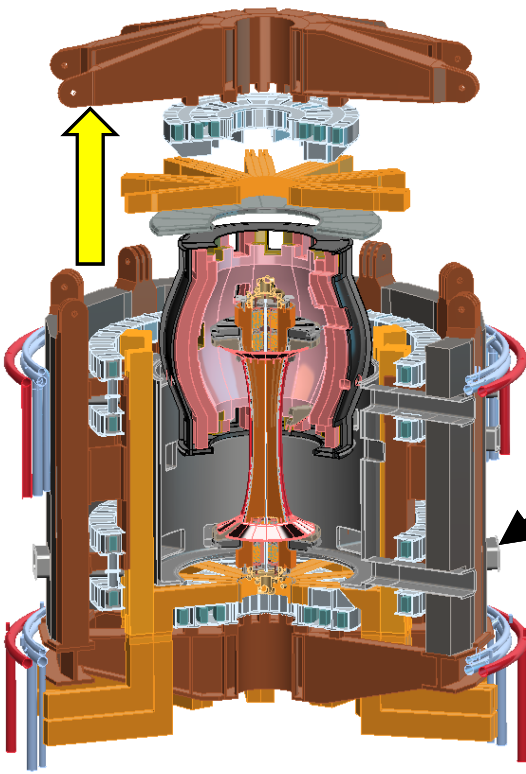


- Future work: assess smaller R, 3D effects (inter-blanket gaps, test-blankets)

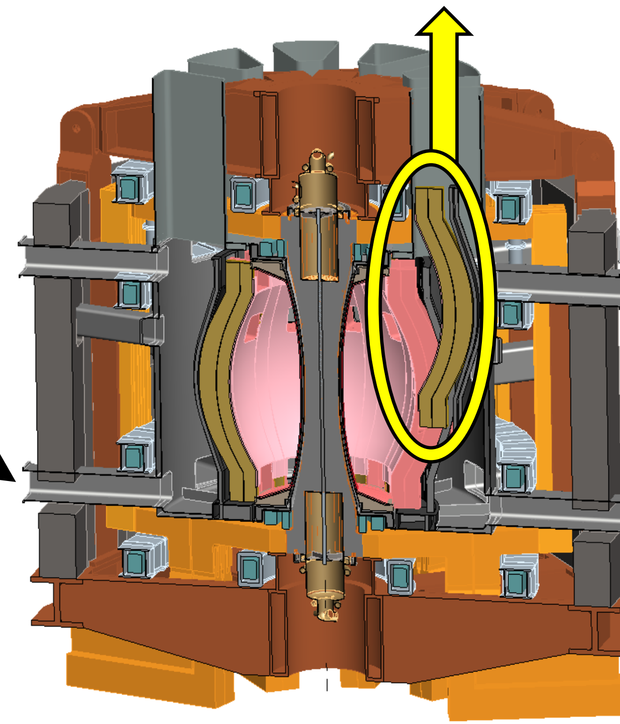
Flexible and efficient in-vessel access important for testing, replacing, improving components, maximizing availability

Several maintenance approaches under consideration:

- Vertically remove entire blanket and/or center-stack
  - **Better for full blanket replacement?**
- Translate blanket segments radially then vertically
  - **Better for more frequent blanket module replacement and/or repair?**



Radial ports for divertor maintenance or pumping



- **May be possible to combine features of both approaches**

Summary

- NSTX Upgrade device and research aim to narrow performance and understanding gaps to next-steps

- Upgrade Project has made good progress in overcoming key design challenges
  - Project on schedule and budget, ~45-50% complete
  - Aiming for project completion in summer 2014

- ST-FNSF development studies are quantifying performance dependence on size
  - Building on achieved/projected NSTX/NSTX-U performance and design
  - Incorporating high δ, advanced divertors, TBR ~ 1, good maintainability

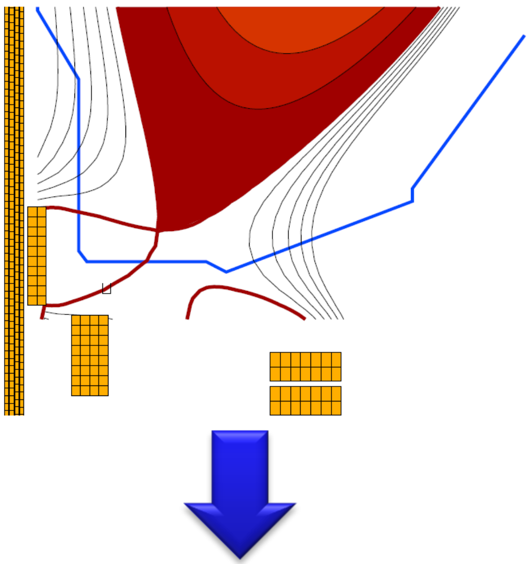
Backup slides

Backup slides

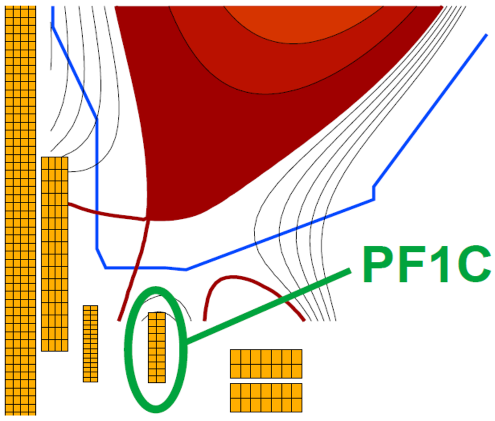
NSTX Upgrade Project

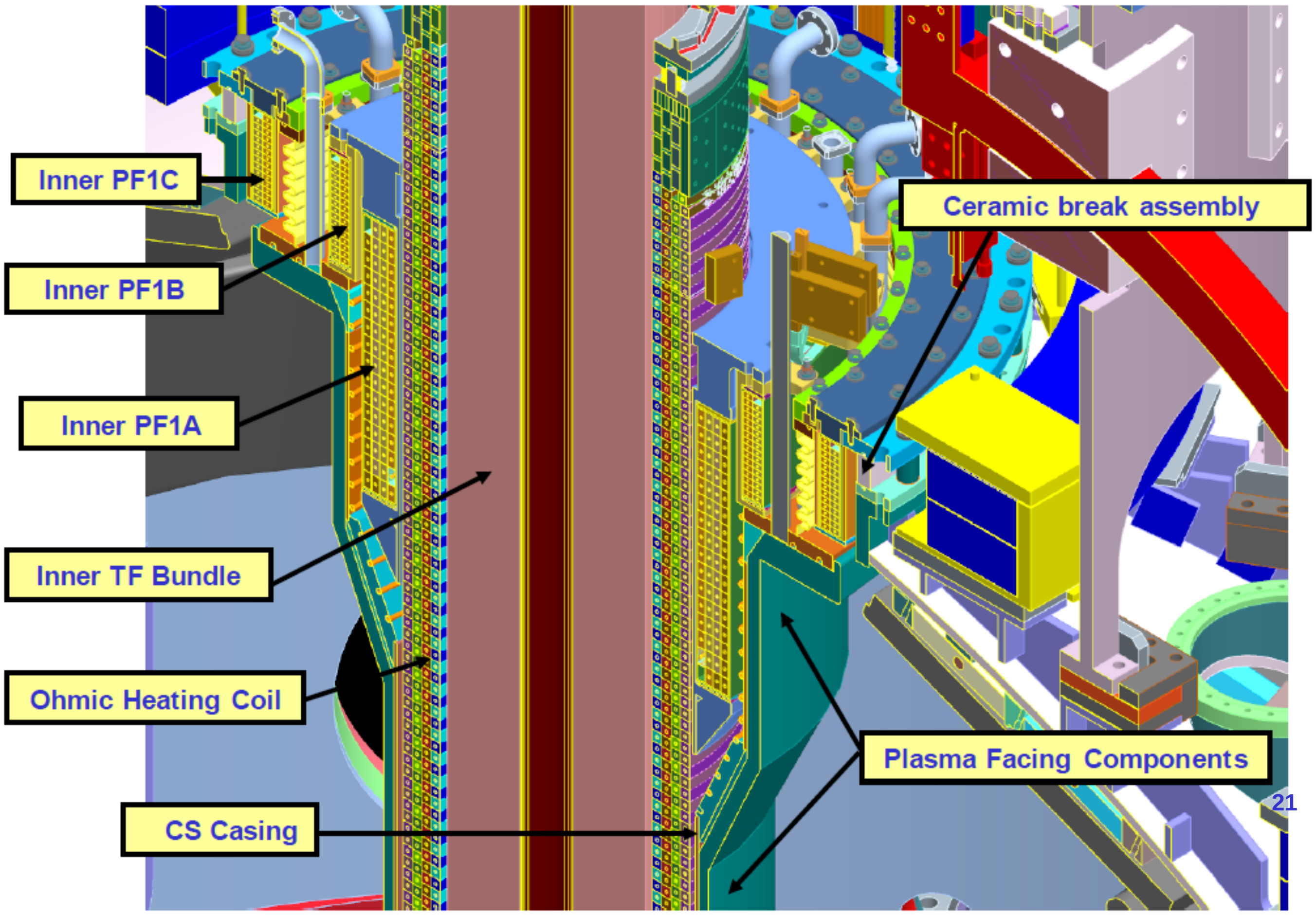
Upgrade CS design provides additional coils for flexible and controllable divertor including snowflake, and supports CHI

NSTX Snowflake



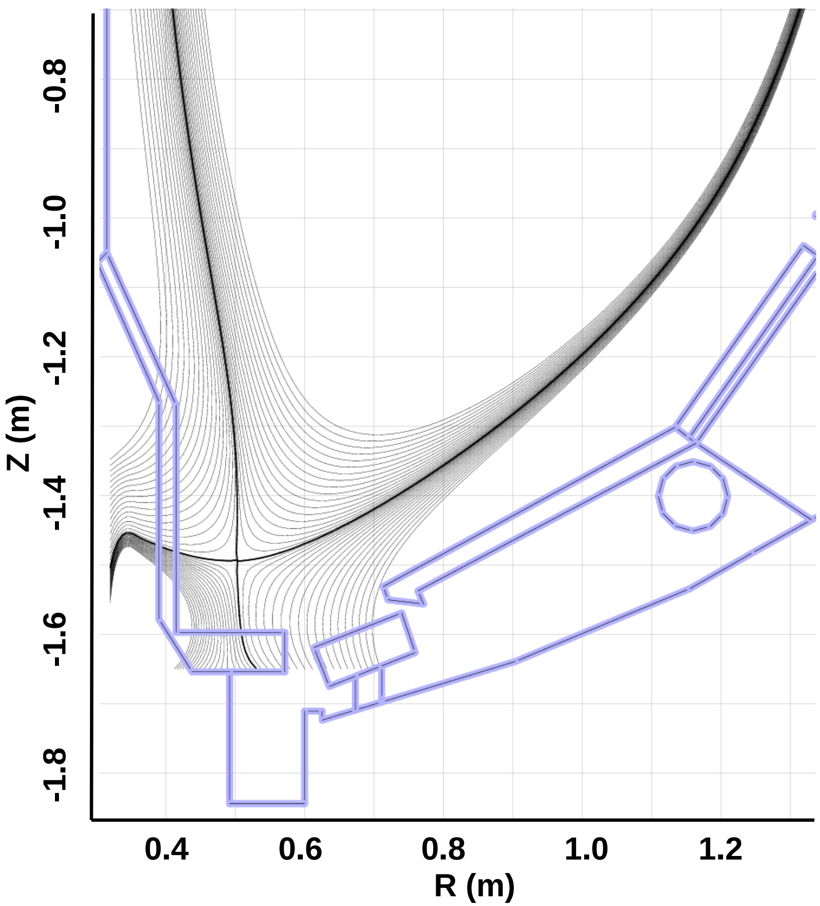
NSTX-U Snowflake





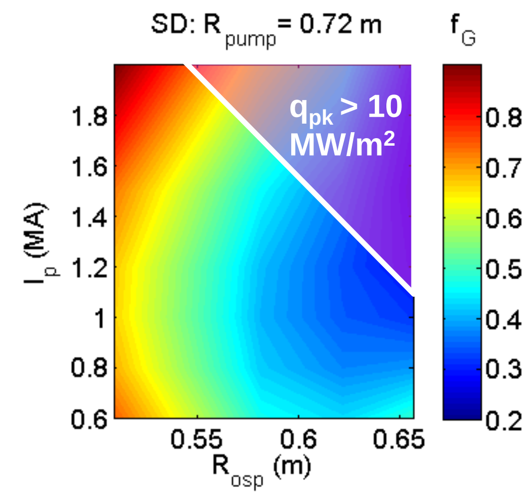
Optimized plenum geometry capable of pumping to low density for a range of R_{OSP}, I_P in NSTX Upgrade

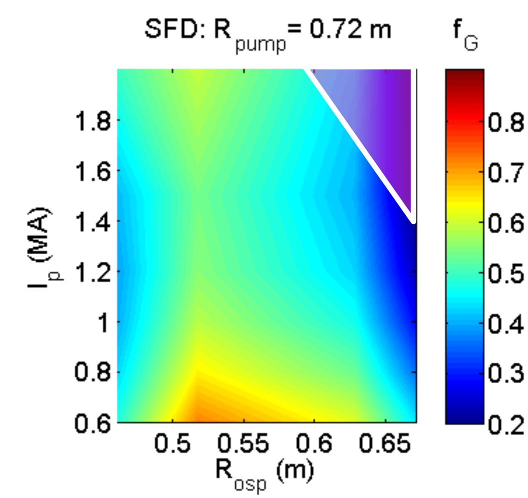
SD = Standard Divertor (shown)
SFD = Snowflake Divertor



SOLPS geometry to be used in future calculations

- Greenwald density fraction f_G 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



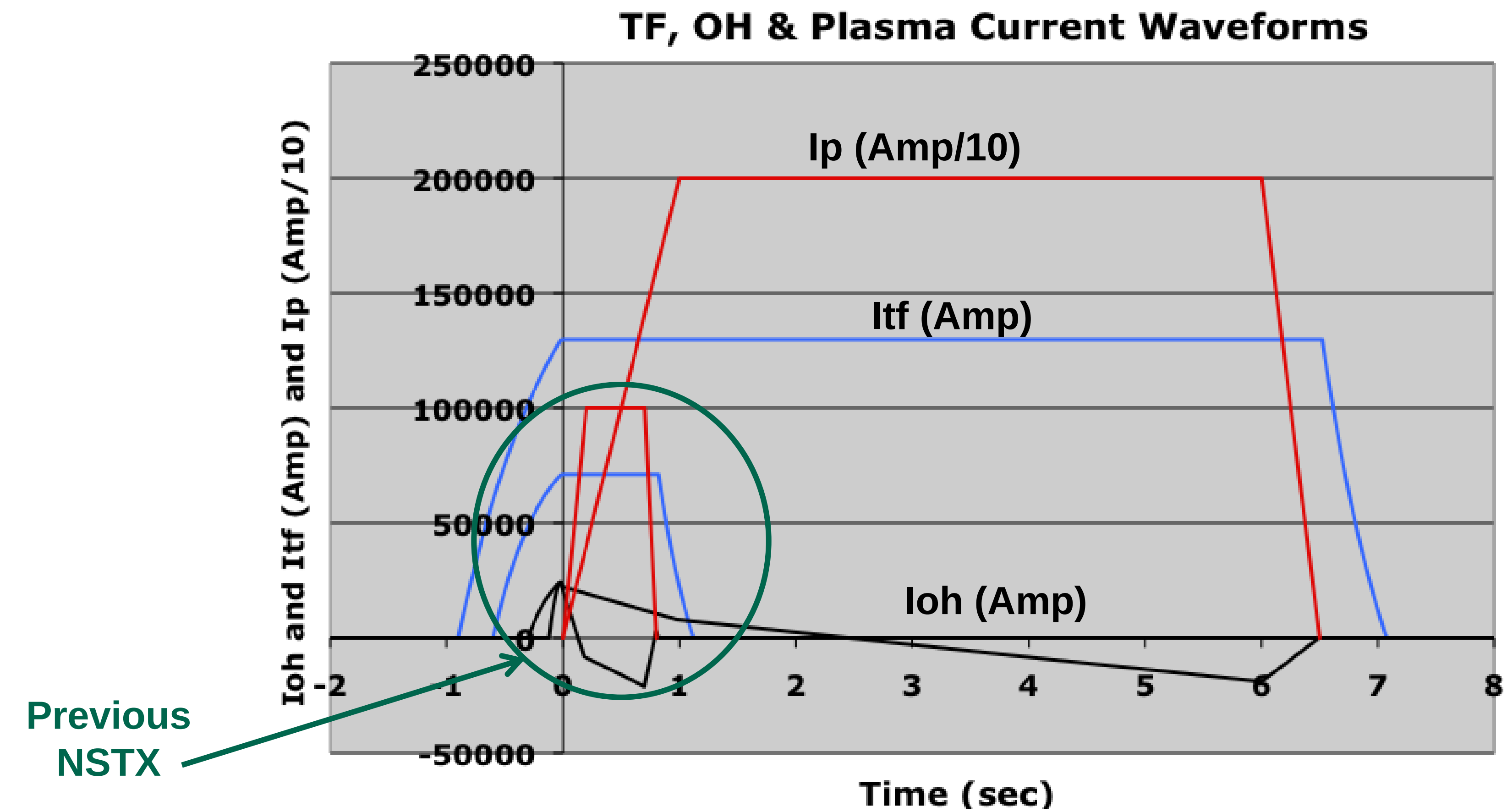


Upgrade substantially increases B_T, I_p, P_{NBI}, τ_{pulse}

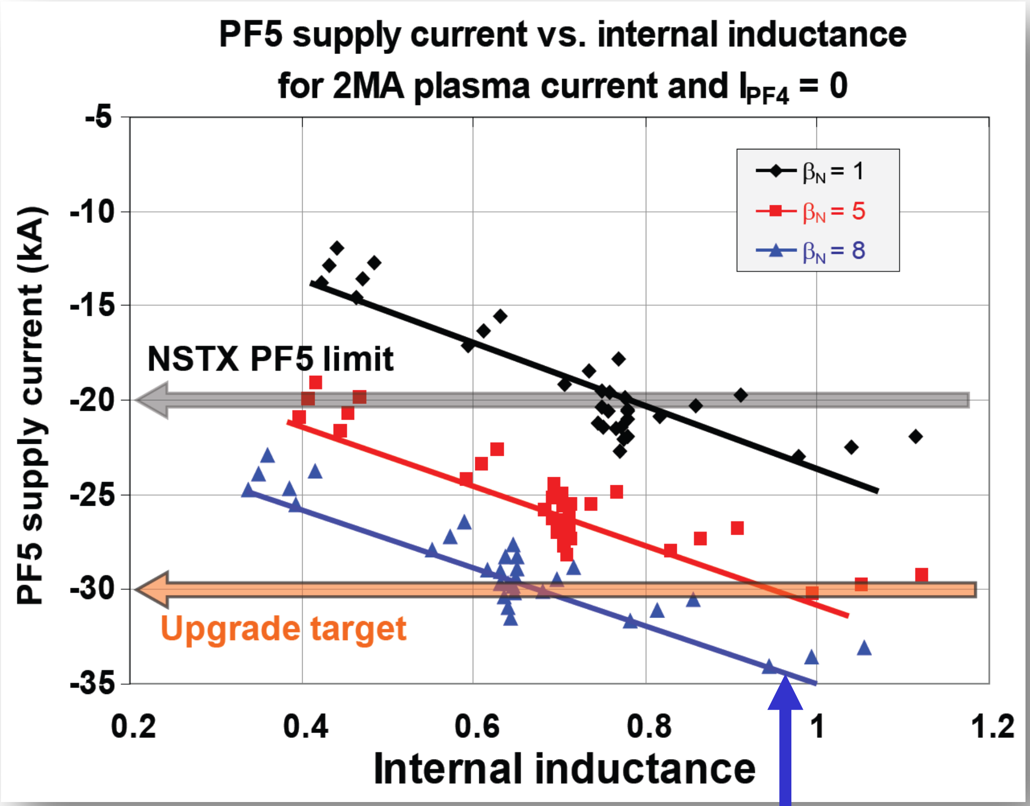
Field and current will be within factor of 2 of initial operation of ST-FNSF

Relative performance of Upgraded NSTX vs. Base:

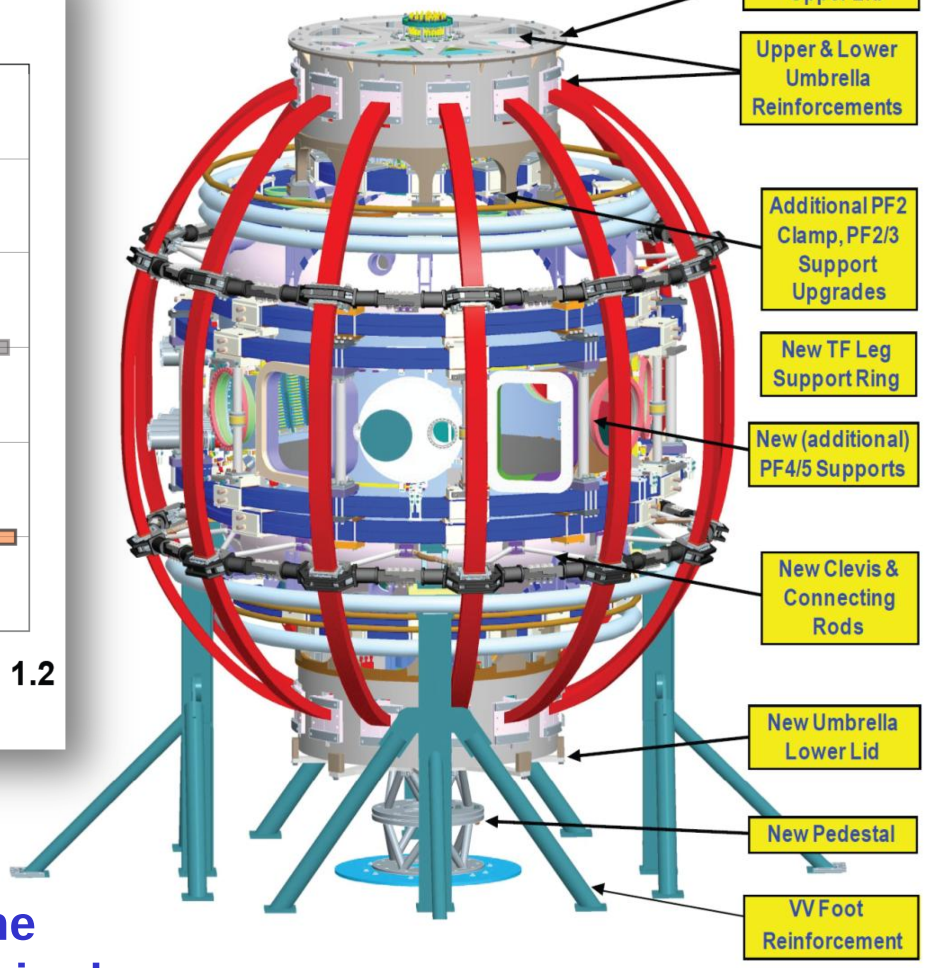
- I_P = 1 → 2MA, B_T = 0.5 → 1T (at same major radius)
- Available OH flux increased 3x, 3-5x longer flat-top
- NBI power increased 2x (5→10MW for 5s, 15MW 1.5s)
- Plasma stored energy increased up to 4x (0.25→ 1MJ)



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

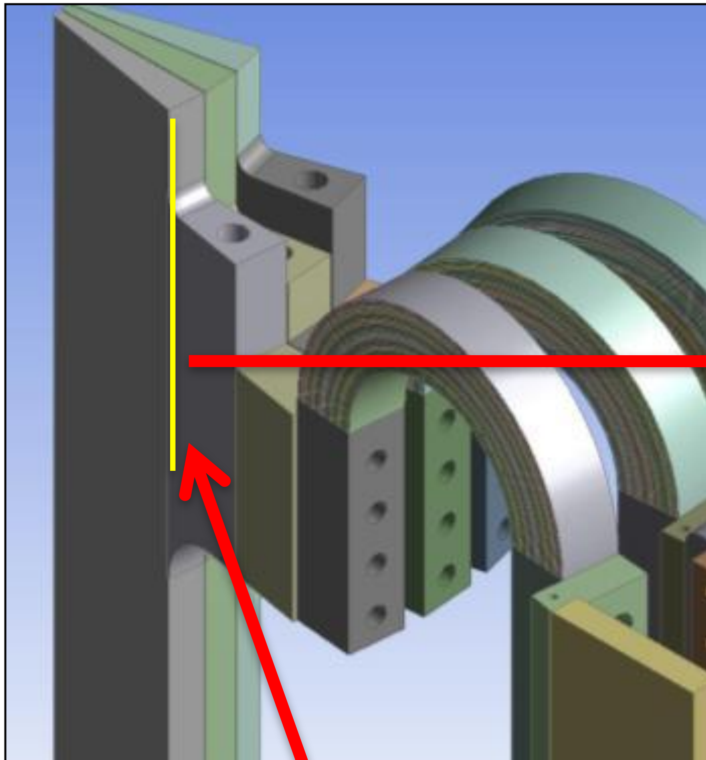


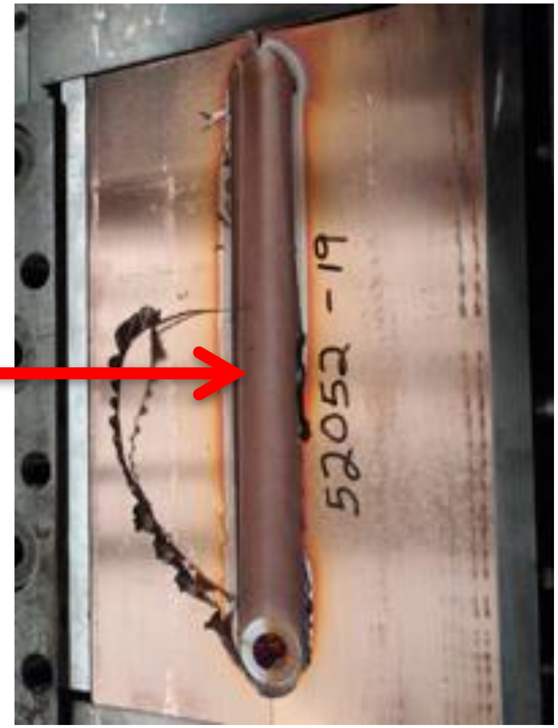
NSTX Upgrade



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

Friction Stir Welding

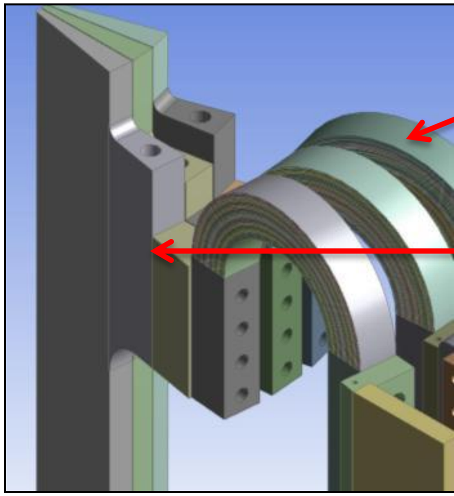




Lead Extension to Inner TF Conductor

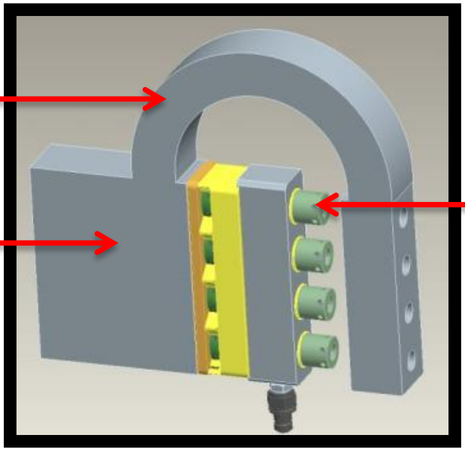
Development trials required to prove dissimilar material welding

Features of TF inner/outer flex strap connector

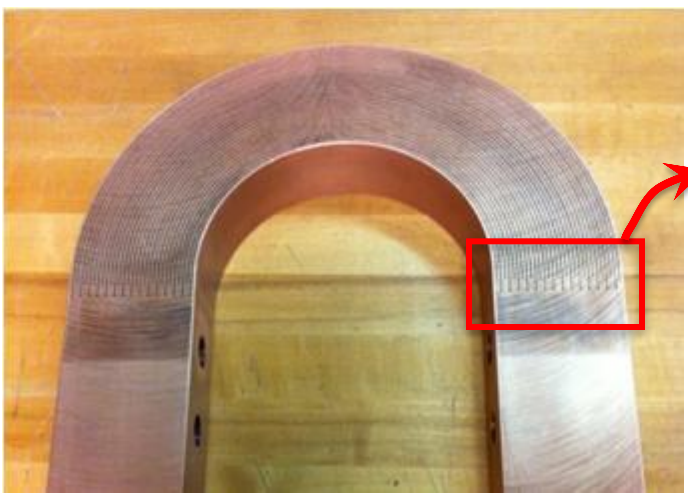


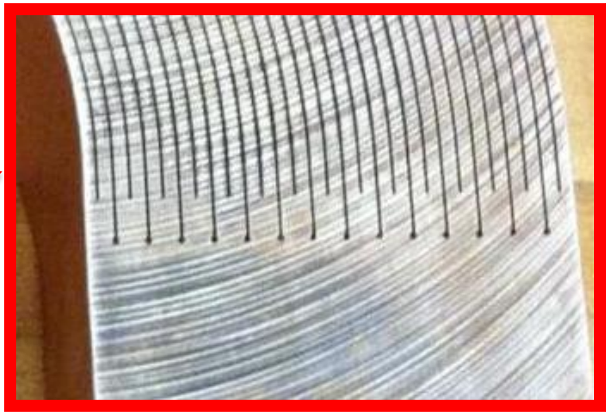
Flex strap

Inner TF

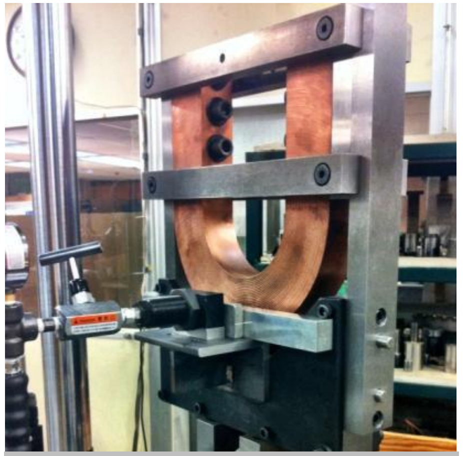


Supernuts® to be used to facilitate assembly





Wire EDM instead of laminated build

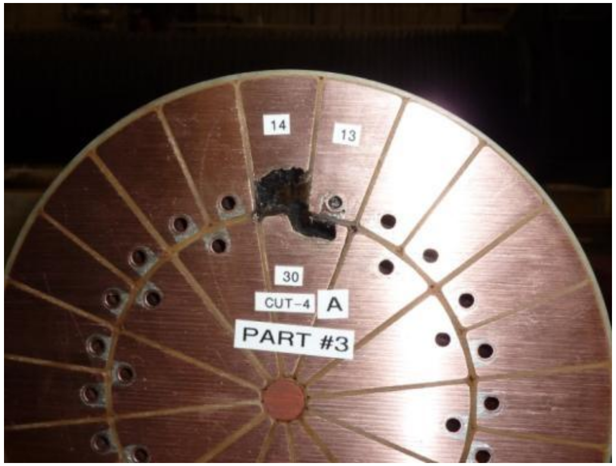


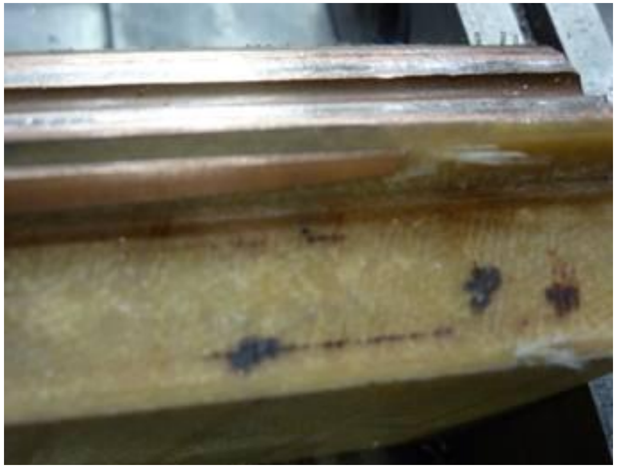
Testing (60,000 cycles)

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**

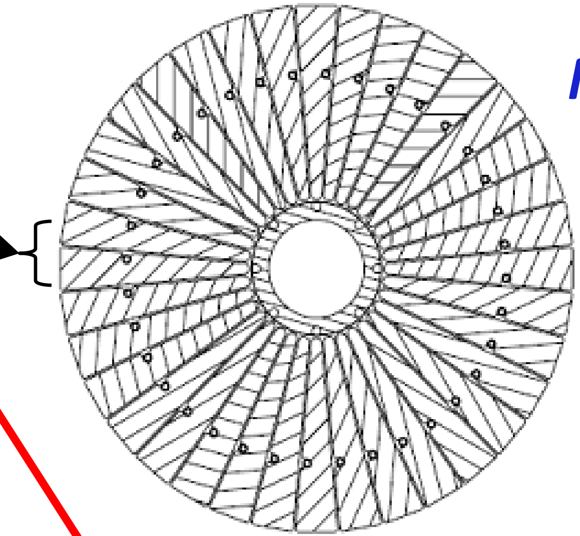




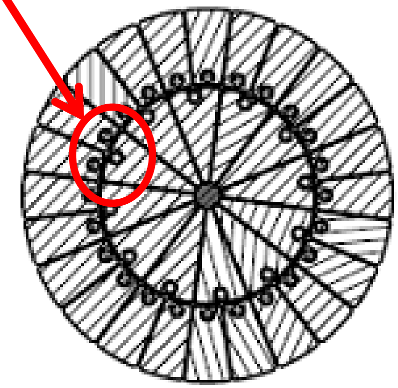
→ Decided to start 2½ year Upgrade outage 6 months early

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



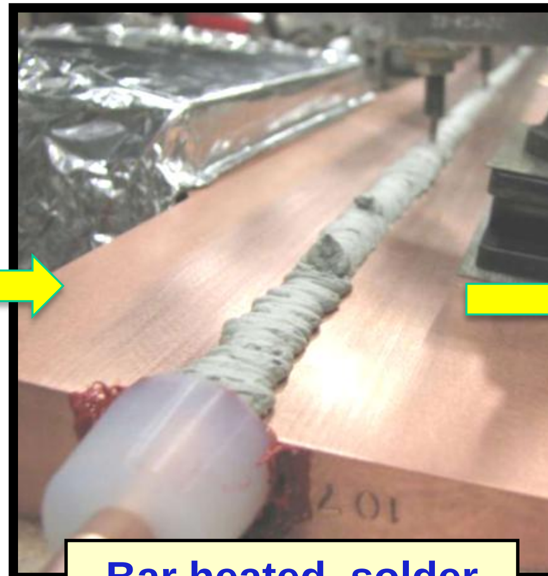
New



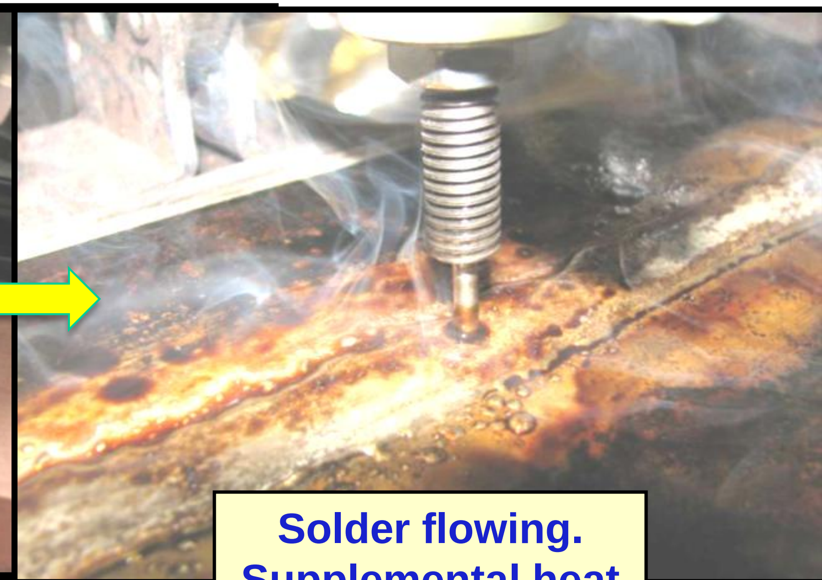
Old

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

Bar placed on heat plate, cooling tube inserted into grove

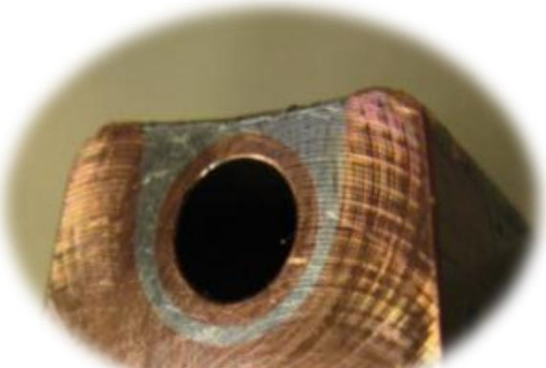


Bar heated, solder paste added

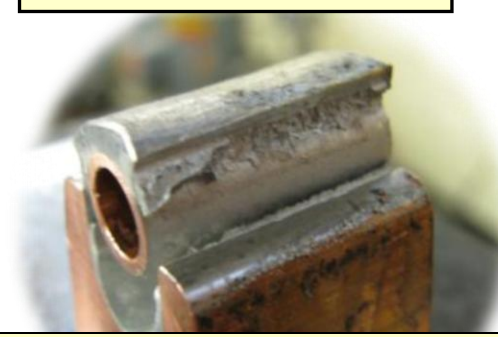


Solder flowing. Supplemental heat applied by torch

Bar ground smooth



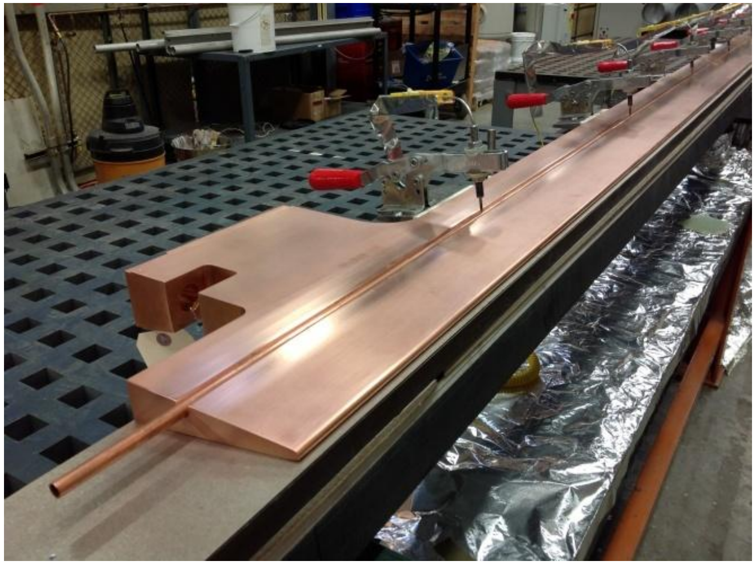
Close up views of solder joint



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

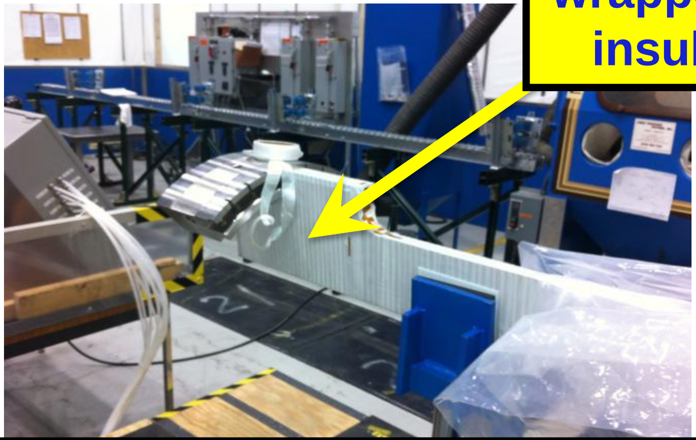
TF cooling tube soldering technique

- **Solder Trials:**
  - Trials have been performed with the assistance of Solder Consultant to verify materials and heating processes
  - Successful heat runs w/actual TF bar
- **Materials:**
  - Solder paste- 96.5 Sn /3.5 Ag w/ GMS based “R” flux [Glyceryl Mono-stearate, Terigitol (a detergent) and Cyclohexlamine Hydro-bromide]
- **Solder Temperatures:**
  - Liquidus ~221º C (430º F)
  - Soludus~ 246º C (475ºF)
  - TF solder temperature~ 270 degrees C to ensure wetting
- **Heating Method:**
  - Power supply w/heating plate
  - Torch heat to complete process



Center-stack fabrication is now underway

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



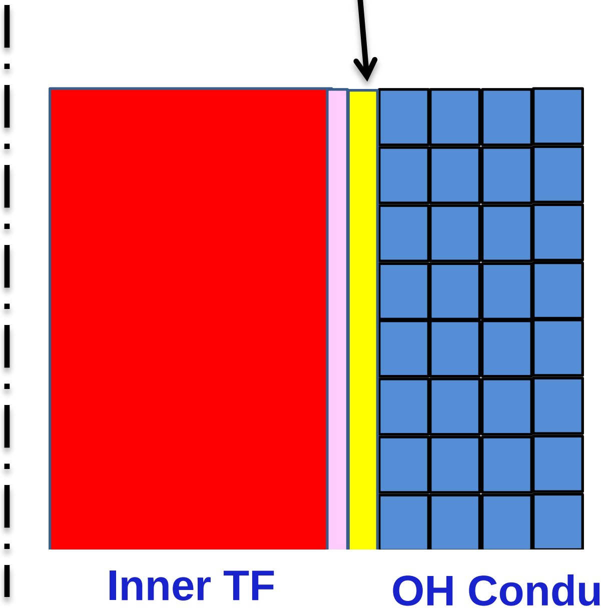
Bar being wrapped with insulation

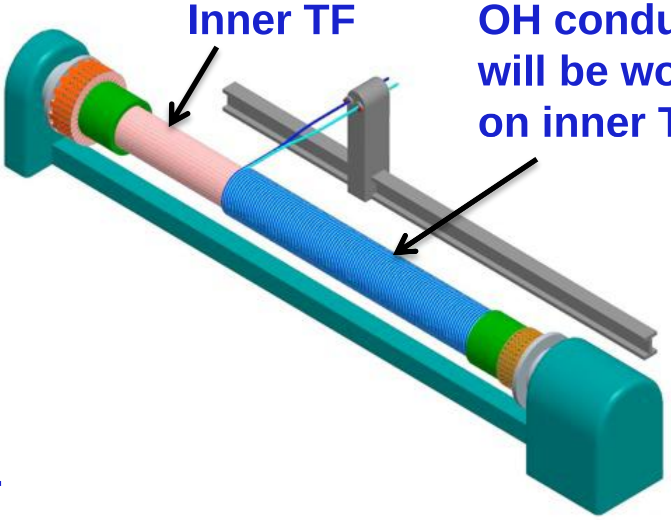
Now entering riskiest stage of project → inner TF and OH fabrication and VPI – will VPI 1st quadrant by end of 2012

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

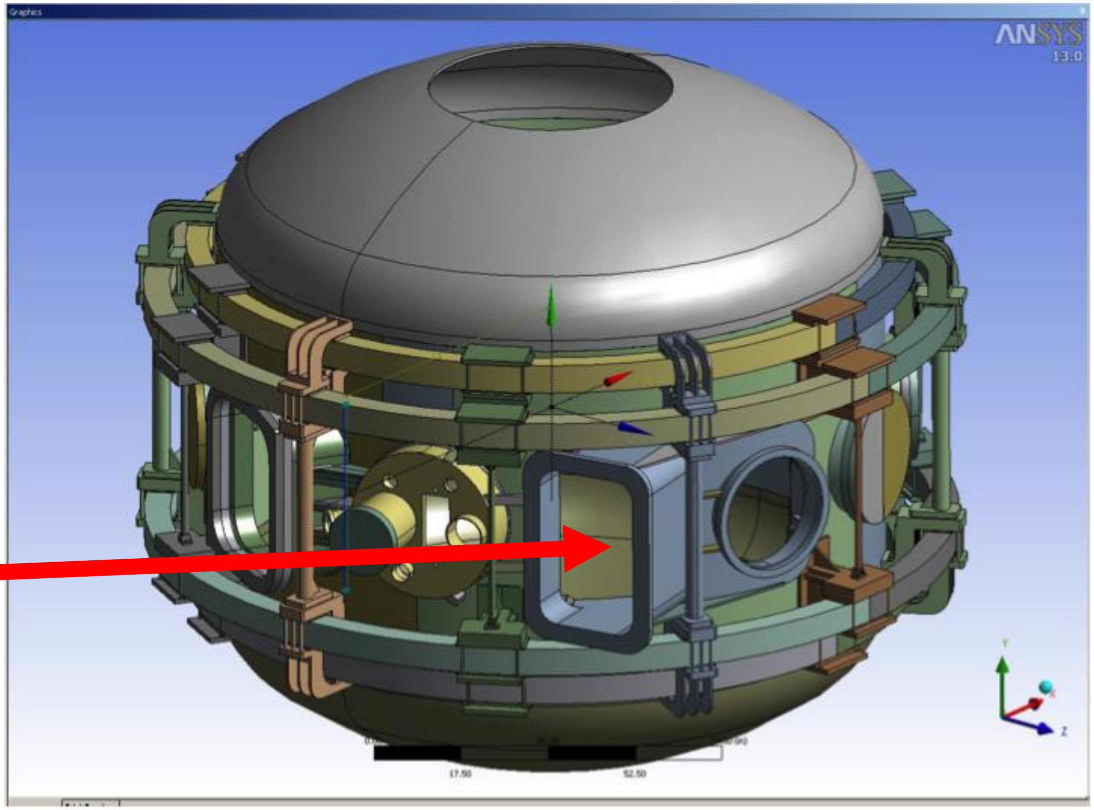
Recent successful VPI trials





New NBI port-cap being readied for installation

- **Preparing to plasma-cut hole in vessel for cap installation**



Tentative plans for initial operations

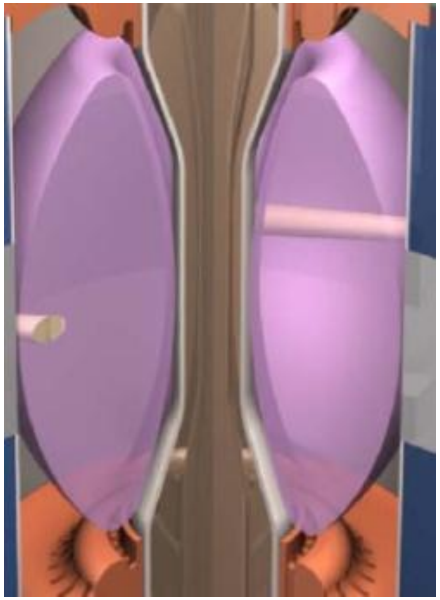
NSTX-U will be brought up methodically to full performance capabilities

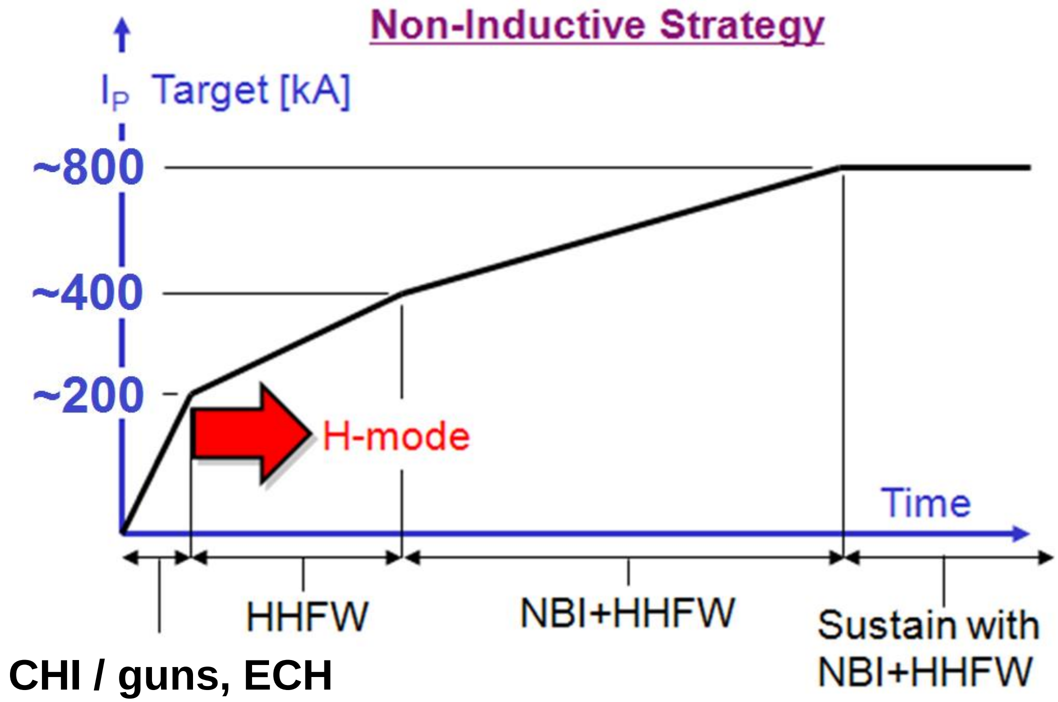
Time Line	B_T(T)	t-pulse (sec)	Ip (MA)
Year 1	0.55 – 0.65 for commissioning. 0.75 by end	1 – 2 sec for commissioning 5 sec by end	~ 1 for commissioning ~ 1.5 by end
Year 2	0.75 routine 1T by end	5 sec routine 1 sec	1.5 MA routine 2 MA by end
Year 3	1 T routine	1 sec routine 5 sec by end	2 MA routine

Full field and current by the end of year 2

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

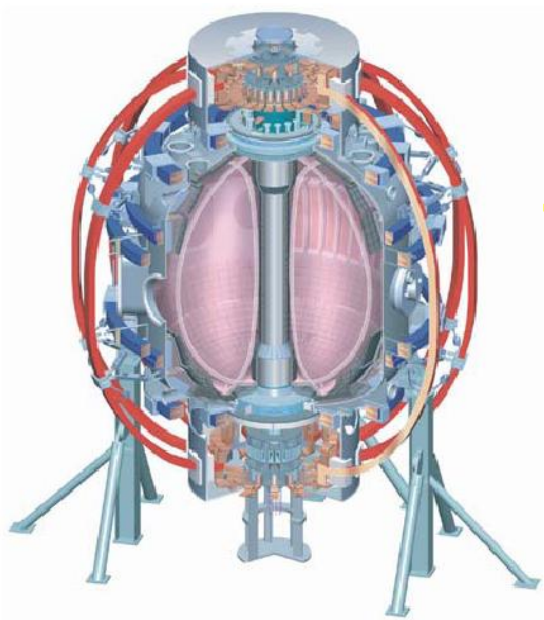
ST-FNSF has no/small central solenoid

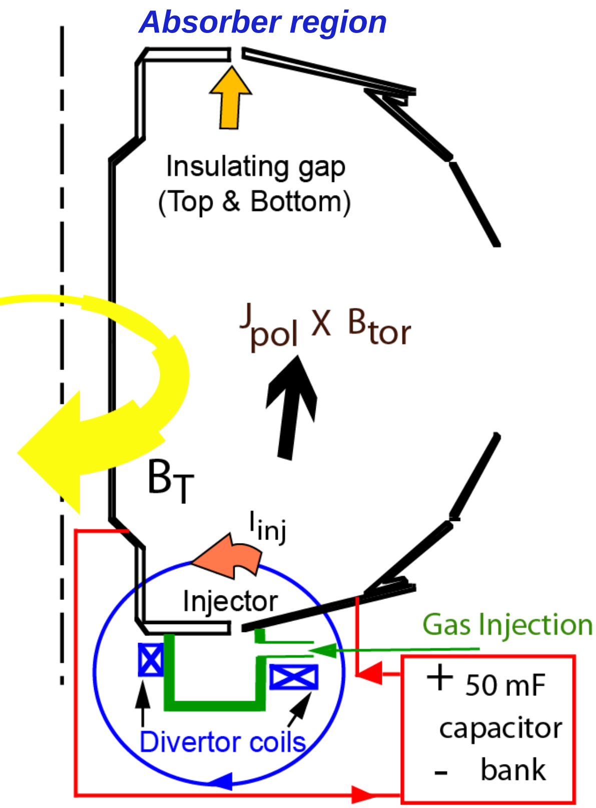


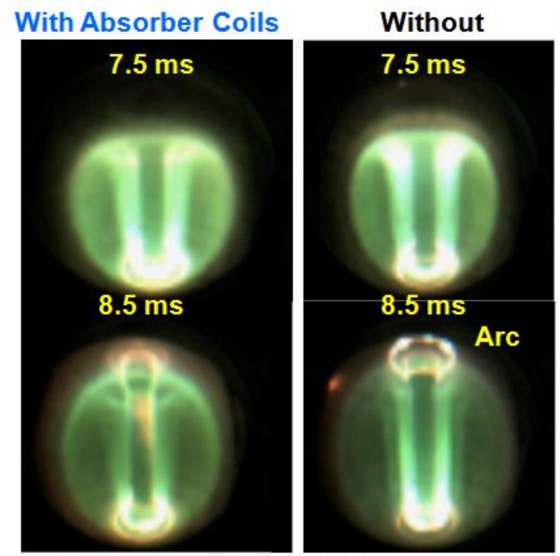


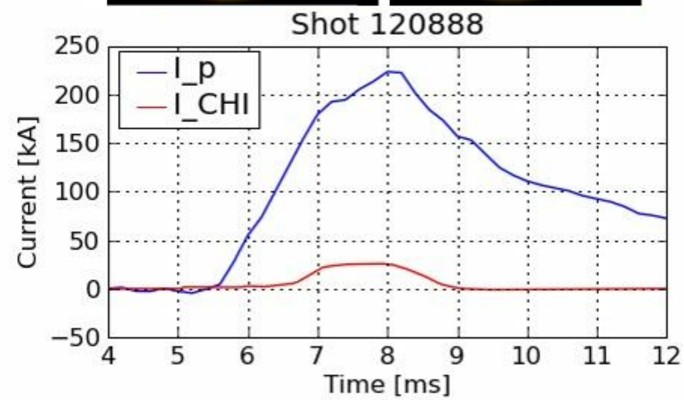
- **NSTX-U goals:**
  - Generate ~0.3-0.4MA full non-inductive start-up with helicity injection + ECH and/or fast wave heating, then ramp to ~0.8-1MA with NBI
  - Develop predictive capability for non-inductive ramp-up to high performance 100% non-inductive ST plasma → prototype FNSF

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







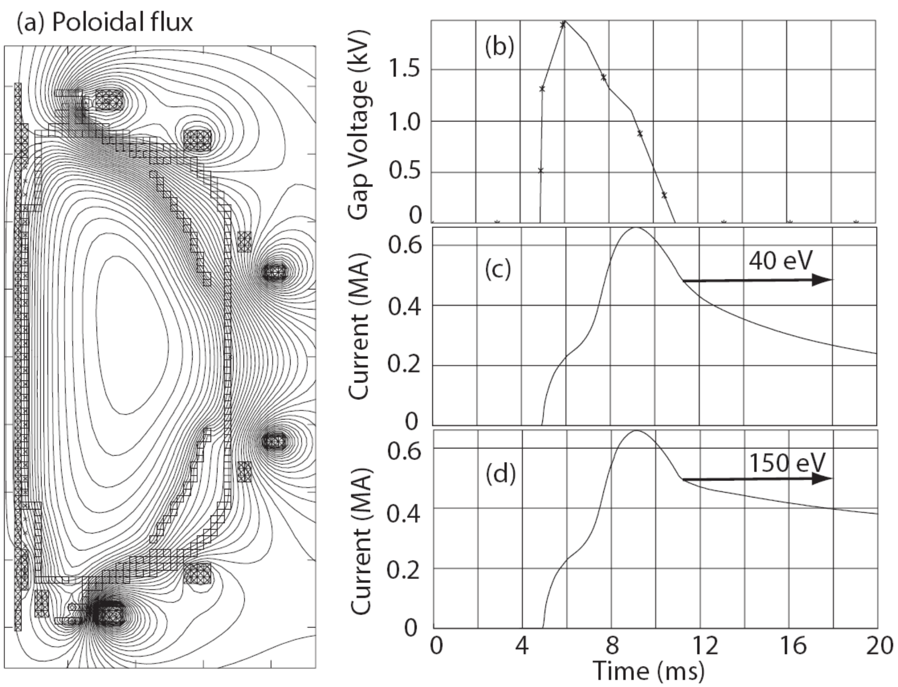


$$I_P = I_{inj}(\psi_T / \psi_{inj}) \qquad I_{inj} = 2\psi_{inj}^2 / (\mu_o^2 d^2 I_{TF})$$

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

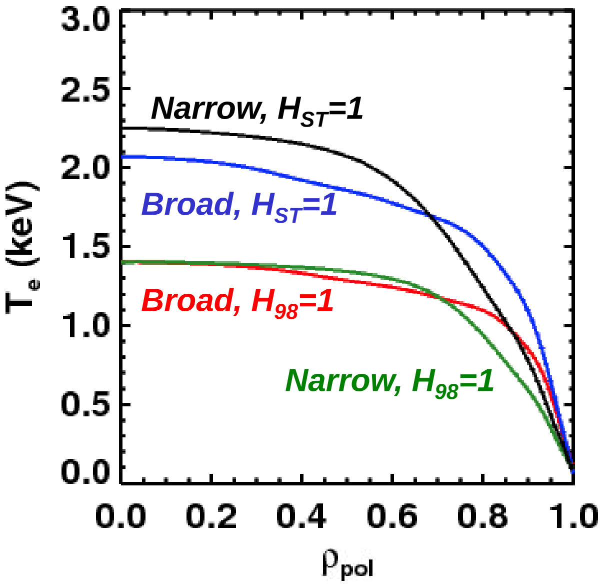
Simulations of CHI project to increased start-up current in NSTX Upgrade, highlight need for additional electron heating

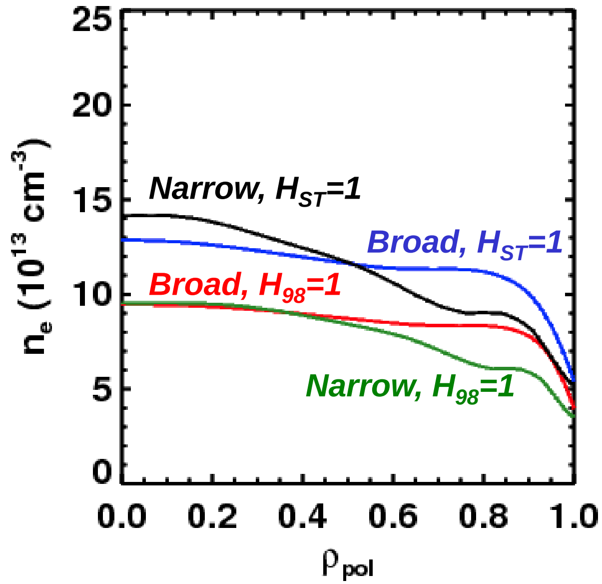
- TSC simulations of transient CHI consistent with NSTX trends
- Favorable projections for NSTX-U:
  - TF increased to 1T and injector flux increased to about 80% of max allowed → **can generate up to ~400kA closed-flux current**
  - **Figs (a-c): T_e = 40 eV, Z_{eff} = 2.5**
  - **Fig (d): T_e = 150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

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



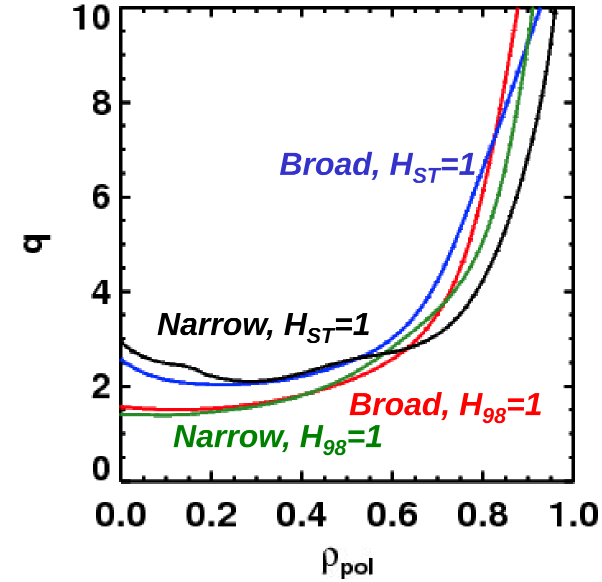


Dashed: ITER-98 confinement scaling

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

Solid: ST confinement scaling

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



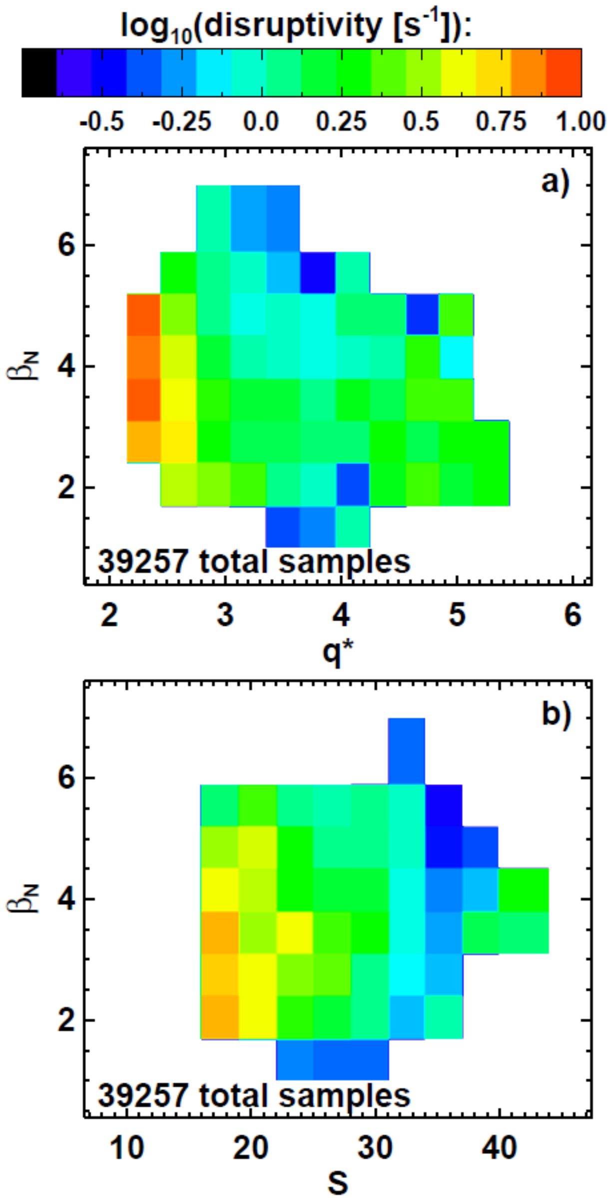
- **Fix: 1.0T, P_{inj}=12.6 MW, f_{GW}=0.72**
- **Fix: A=1.75, κ=2.8**
- Find the non-inductive current level for 2 confinement and 2 profile assumptions…*yields 4 different projections.*

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

ST Development

NSTX disruptivity data informs FNSF operating parameters



- Increased disruptivity for q* < 2.7
  - Significantly increased for q* < 2.5
- Lower disruptivity for β_N = 4-6 compared to lower β_N
  - Higher β_N increases f_{BS}, broadens J profile, elevates q_{min}
  - Operation above no-wall limit aided by:
    - NBI co-rotation
    - Close-fitting conducting wall
    - Active error-field and RWM control
- Strong shaping also important
  - $S \equiv q_{95} I_P/aB_T$
  - S > 30 provides strongest stabilization
  - S > 22-25 good stability
  - S < 22 unfavorable

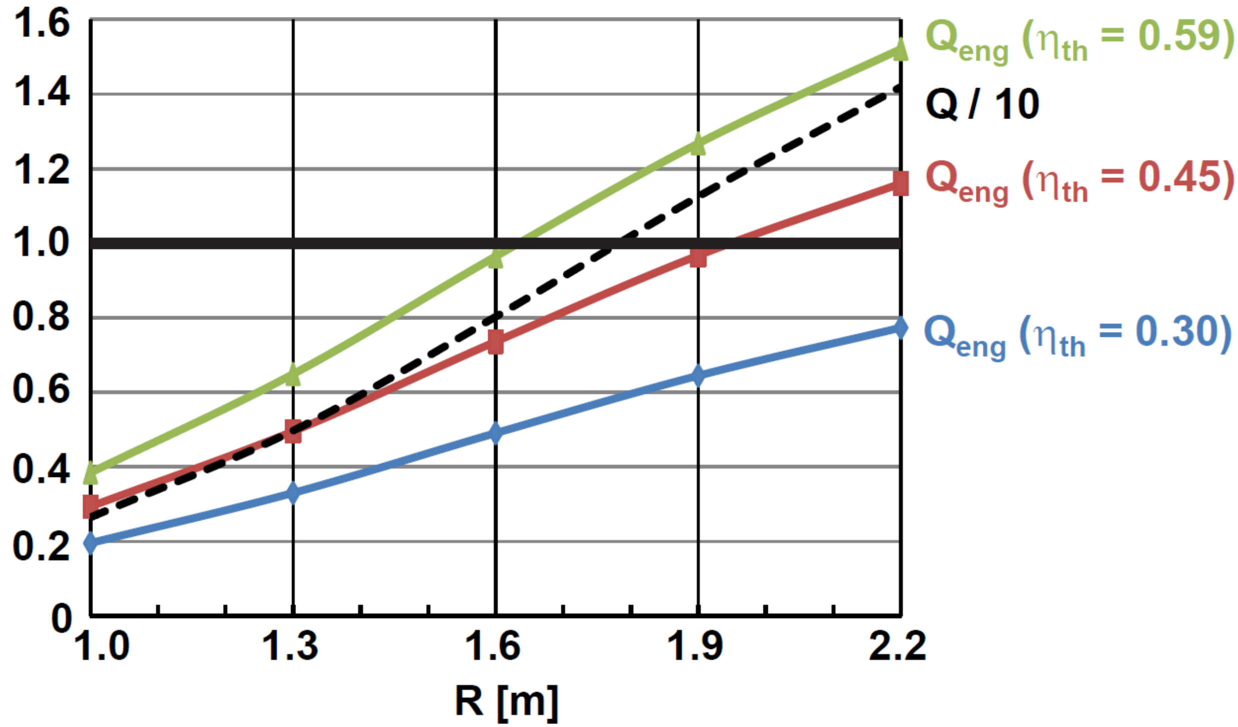
Gerhardt EX/9-3

Investigating high performance scenarios for accessing increased neutron wall loading and engineering gain

- Decrease B_T = 3T → 2.6T, increase H_{98} = 1.2 → 1.5
- Fix β_N = 6, β_T = 35%, q* = 2.5, $f_{Greenwald}$ varies: 0.66 to 0.47

$$Q_{eng} \equiv \frac{P_{electric}\ \text{produced}}{P_{electric}\ \text{consumed}}$$

See J. Menard, et al., Nucl. Fusion 51 (2011) 103014 for detail on definitions/parameters



Note: Outboard PF coils are superconducting

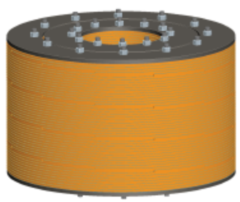
- **Size scan: Q increases from 3 (R=1m) to 14 (R=2.2m)**
- **Smallest ST for Q_{eng} ~ 1 is R=1.6m → requires very efficient blankets**
- **Average neutron wall loading increases from 1.8 to 3 MW/m^2 (not shown)**

Neutronics analysis indicates organic insulator for divertor PF coils unacceptable

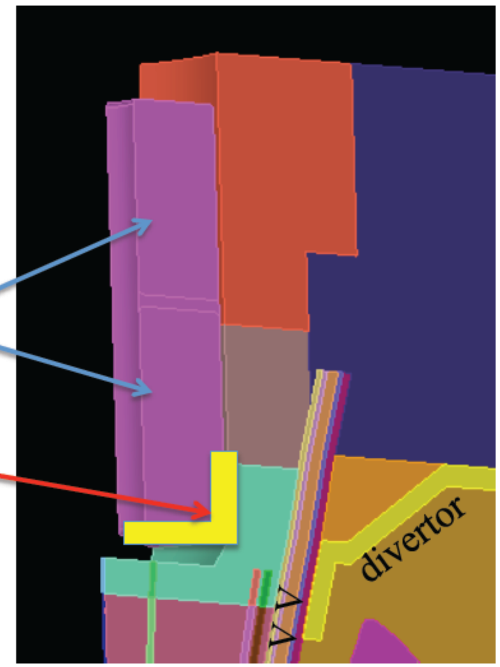
Dose Results

PF Coil **Composition**, per T. Brown

	Volume (in^3)	%
water	5017	20
copper	17059	68
insulator	1254	5
hardware	1756	7
Total	25087	100



PF Coils



3-D Model

Insulator (cyanate ester / epoxy blend)
placed at corner of PF coil to calculate peak dose

Peak dose to insulator $= 3 \times 10^{11}$ **rad @ 1 FPY**
$= 1.8 \times 10^{12}$ **rad @ 6 FPY**
>> 2×10^{10} rad limit

Replace inner PF coil every 24 days (not practical).

Is ceramic insulator more radiation resistant?

MgO insulation appears to have good radiation resistance for divertor PF coils

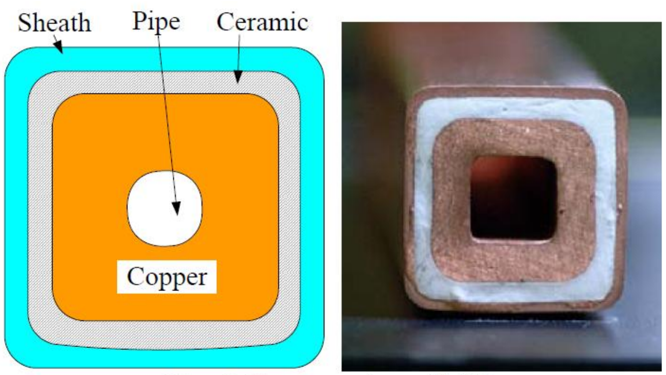


Fig. 3 Cross section of MIC

Table 1: Comparison of radiation resistant

	Organic		Inorganic
Insulation	Epoxy	Polyimide	MgO
Resistant	$>10^7$ Gy	$>10^9$ Gy	$>10^{11}$ Gy

R&D of a Septum Magnet Using MIC coil

Kuanjun Fan [1,A)], Hiroshi Matsumoto [A)], Koji Ishii [A)], Noriyuki Matsumoto [B)]

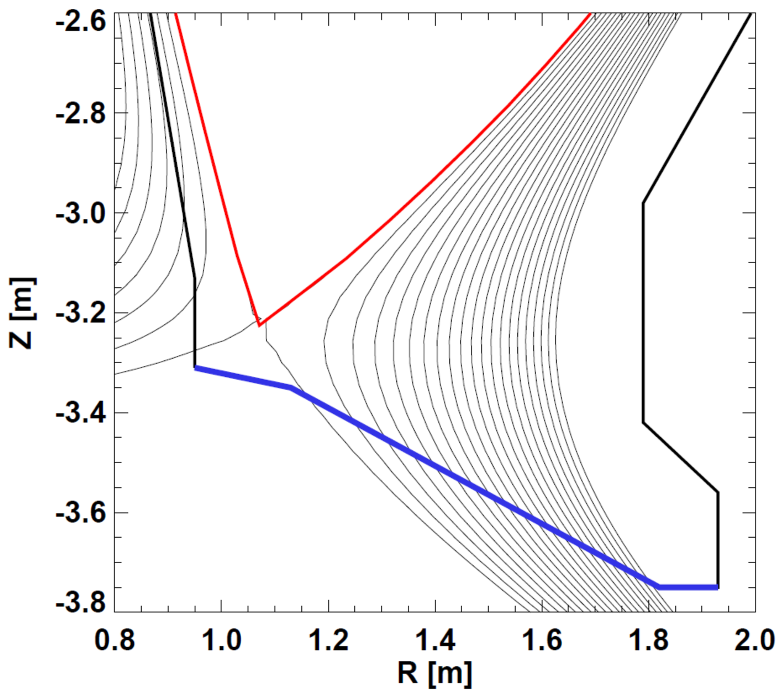
[A)] High Energy Accelerator Research Organization (KEK)
1-1 OHO, Tsukuba, Ibaraki, 305-0801, Japan

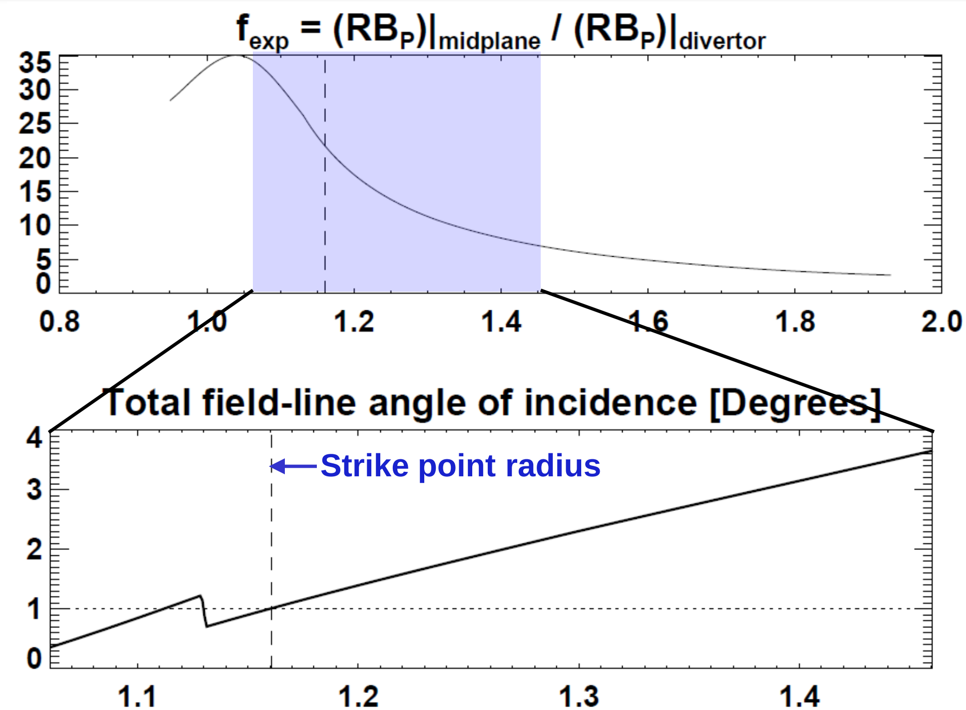
[B)] 2NEC/Token

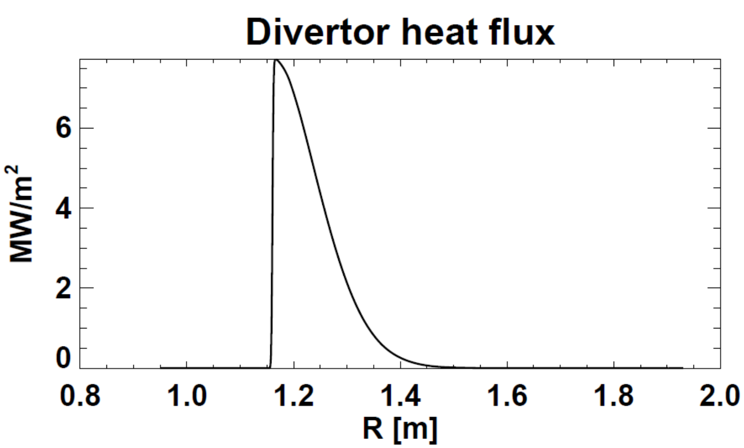
Proceedings of the 5th Annual Meeting of Particle Accelerator Society of Japan and the 33rd Linear Accelerator Meeting in Japan (August 6-8, 2008, Higashihiroshima, Japan)

- UW analysis of divertor PFs
  - 1.8×10^{12} rad = 1.8×10^{10} Gy at 6FPY for P_{fus} = 160MW
- Pilot mission for R=1.6m:
  - P_{fus} = 420MW vs. 160MW → 2.6x higher → 4.7×10^{10} Gy
  - Even for Pilot mission, dose is < limit of 10^{11} Gy
- Limiting factor may be Cu
- Need to analyze CS lifetime
- Revisit option for multi-turn TF and small OH solenoid

Parameters for R=1.6m ST-FNSF conventional divertor



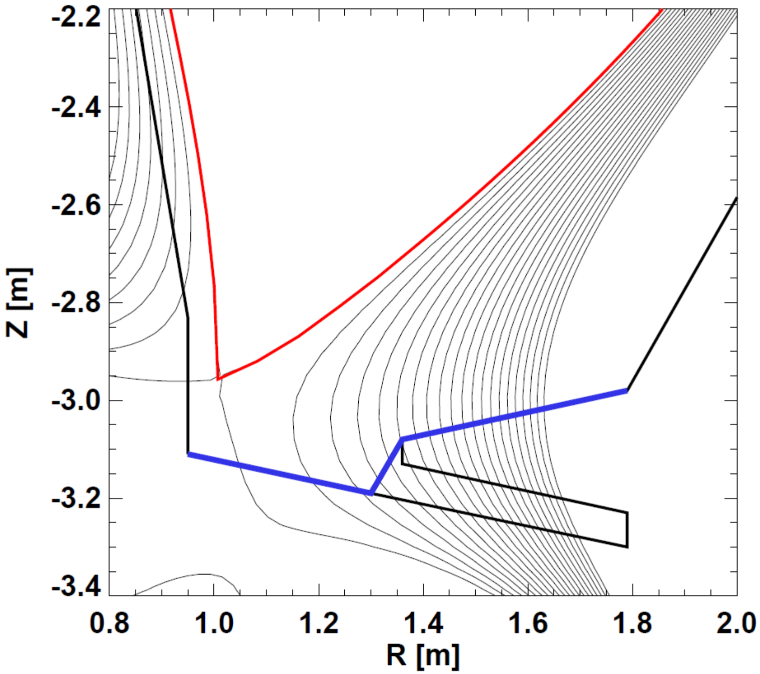


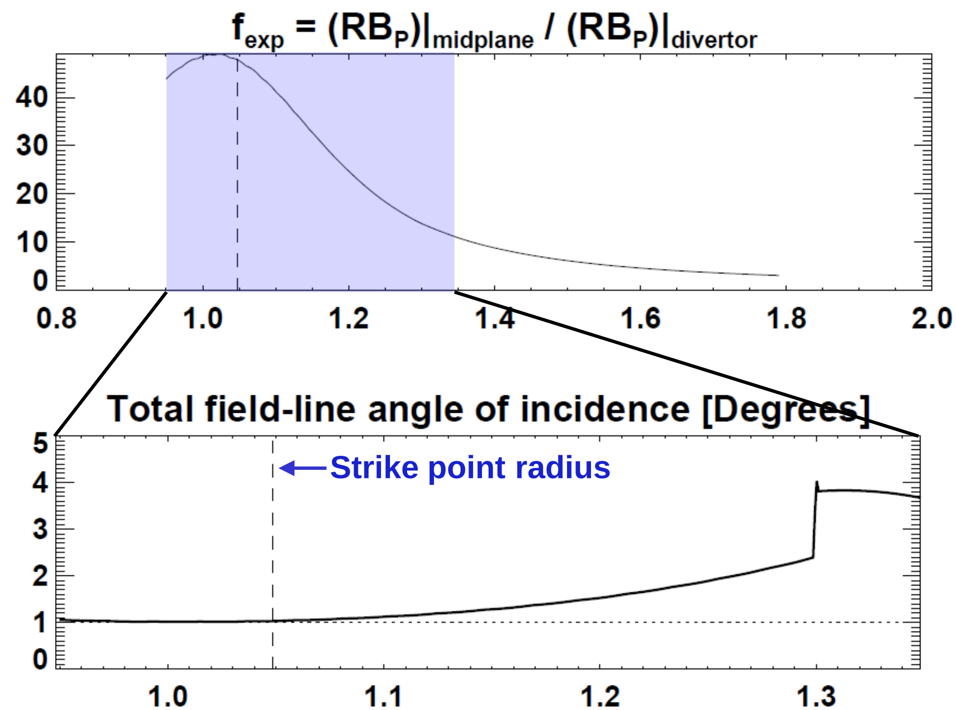


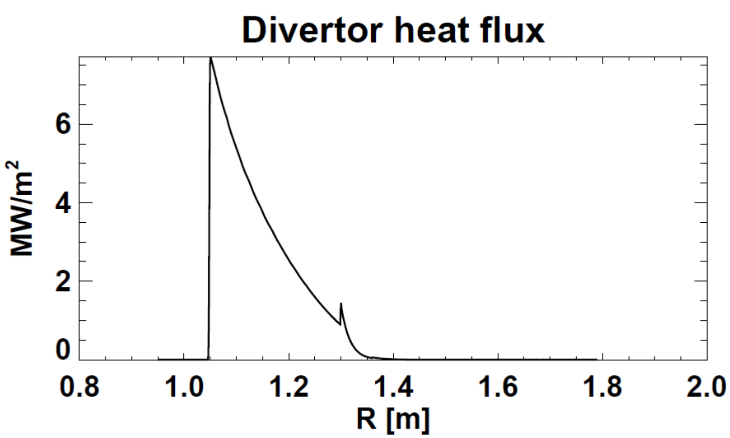
$$\text{Peak } q_{\perp\text{-div}} \approx \frac{P_{heat}\,(1-f_{rad})\,f_{obd}\,\sin(\theta_{pol})}{2\pi R_{strike}\,f_{exp}\,\lambda_{q\text{-mid}}\,N_{div}}$$

- P_{heat} = 115MW, f_{rad}=0.8, f_{obd}=0.8, $\sin(\theta_p)$ = 0.39
- R_{strike} = 1.16m, f_{exp} = 22, $\lambda_{q\text{-mid}}$ =2.7mm, N_{div} = 2

Parameters for R=1.6m ST-FNSF snowflake divertor





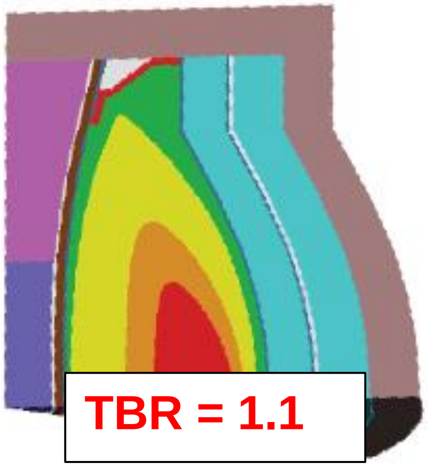


$$\text{Peak } q_{\perp\text{-div}} \approx \frac{P_{heat}\,(1-f_{rad})\, f_{obd}\, \sin(\theta_{pol})}{2\pi R_{strike}\, f_{exp}\, \lambda_{q\text{-mid}}\, N_{div}}$$

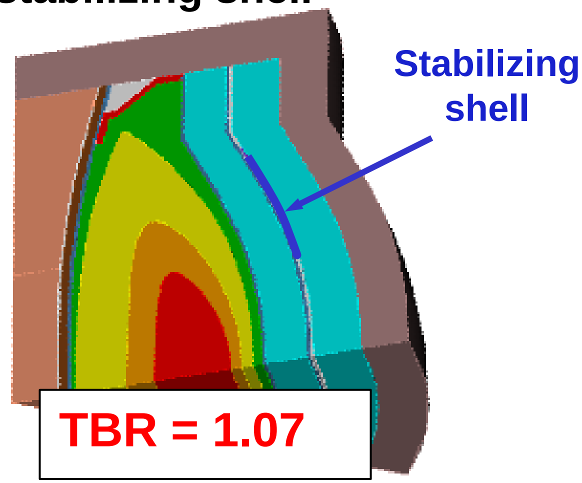
- P_{heat} = 115MW, f_{rad}=0.8, f_{obd}=0.8, $\sin(\theta_p)$ = 0.87
- R_{strike} = 1.05m, f_{exp} = 50, $\lambda_{q\text{-mid}}$ =2.7mm, N_{div} = 2

R=1.6m ST-FNSF TBR calculations

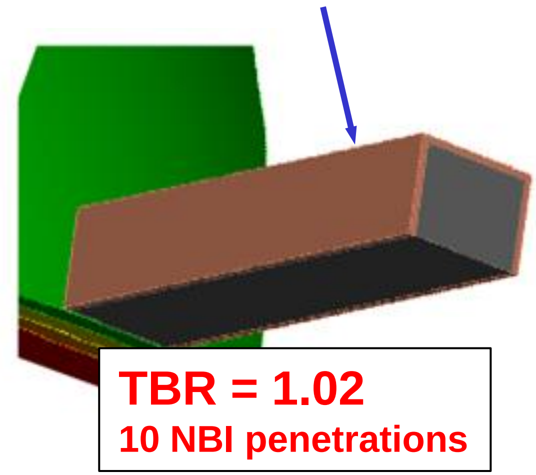
Extended conformal blanket



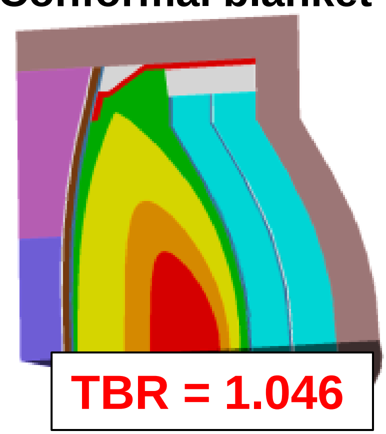
+ 3cm thick stabilizing shell



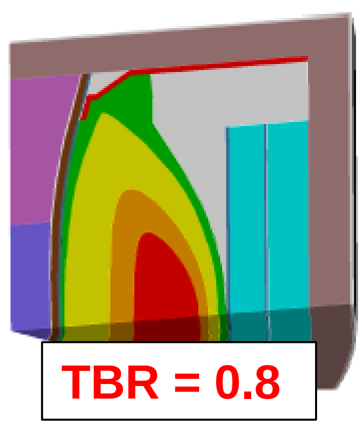
NBI penetration at midplane



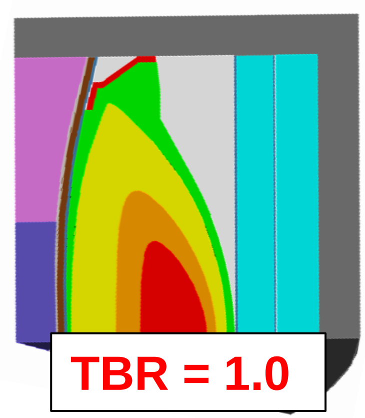
Conformal blanket



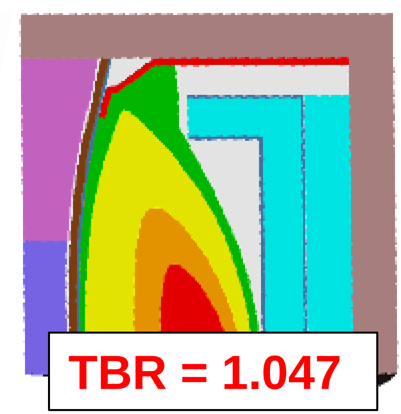
Straight blanket



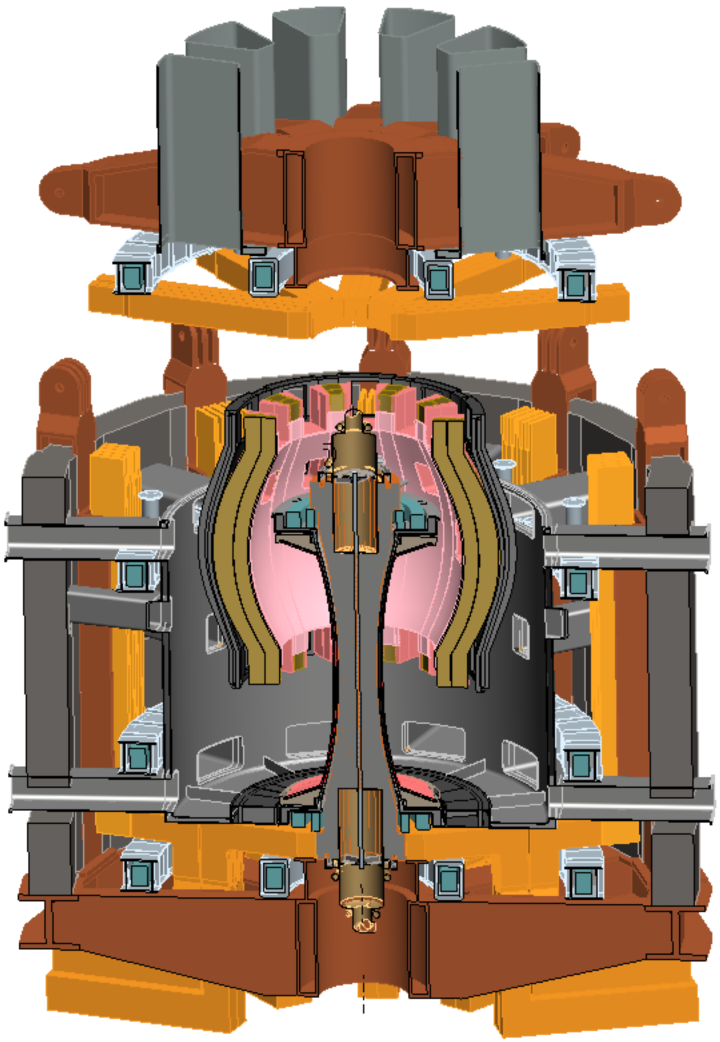
Extended straight blanket



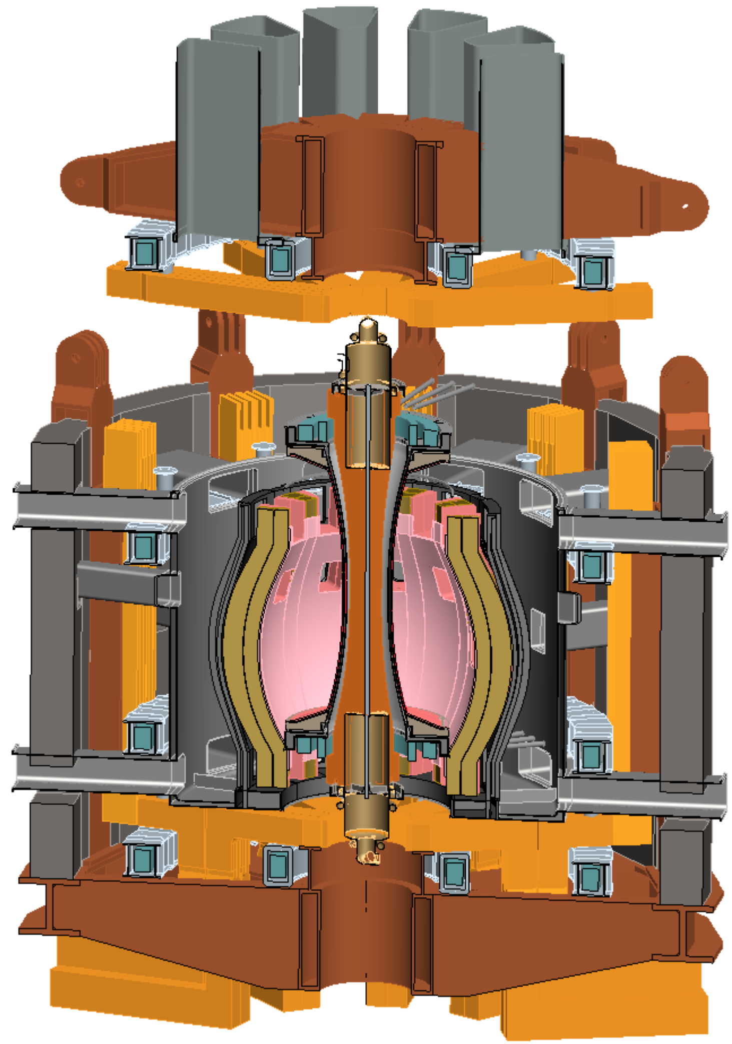
Straight blanket with flat top



Assembly and maintenance schemes with snowflake divertor and vertical ports

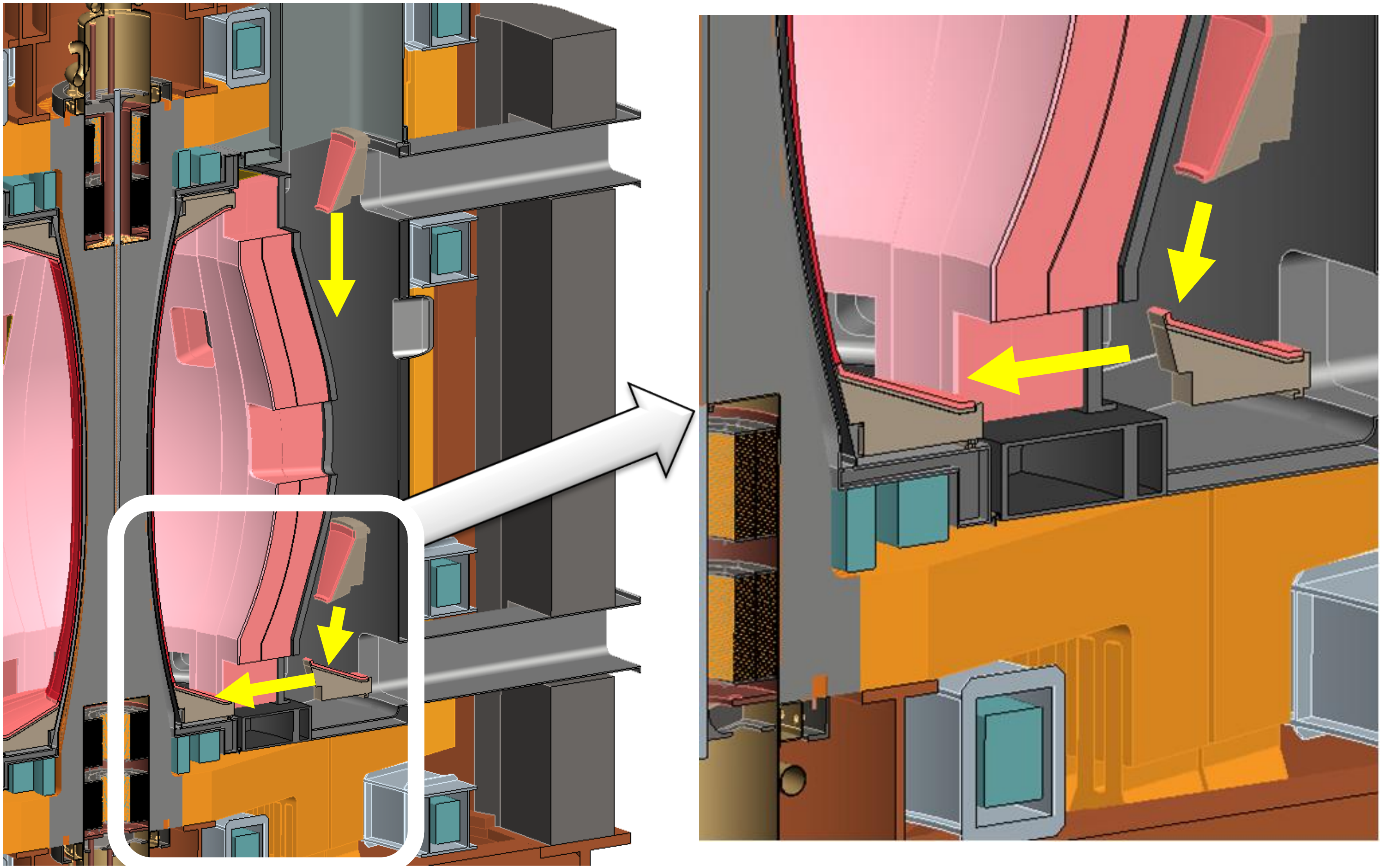


Full Blanket Assembly Removed

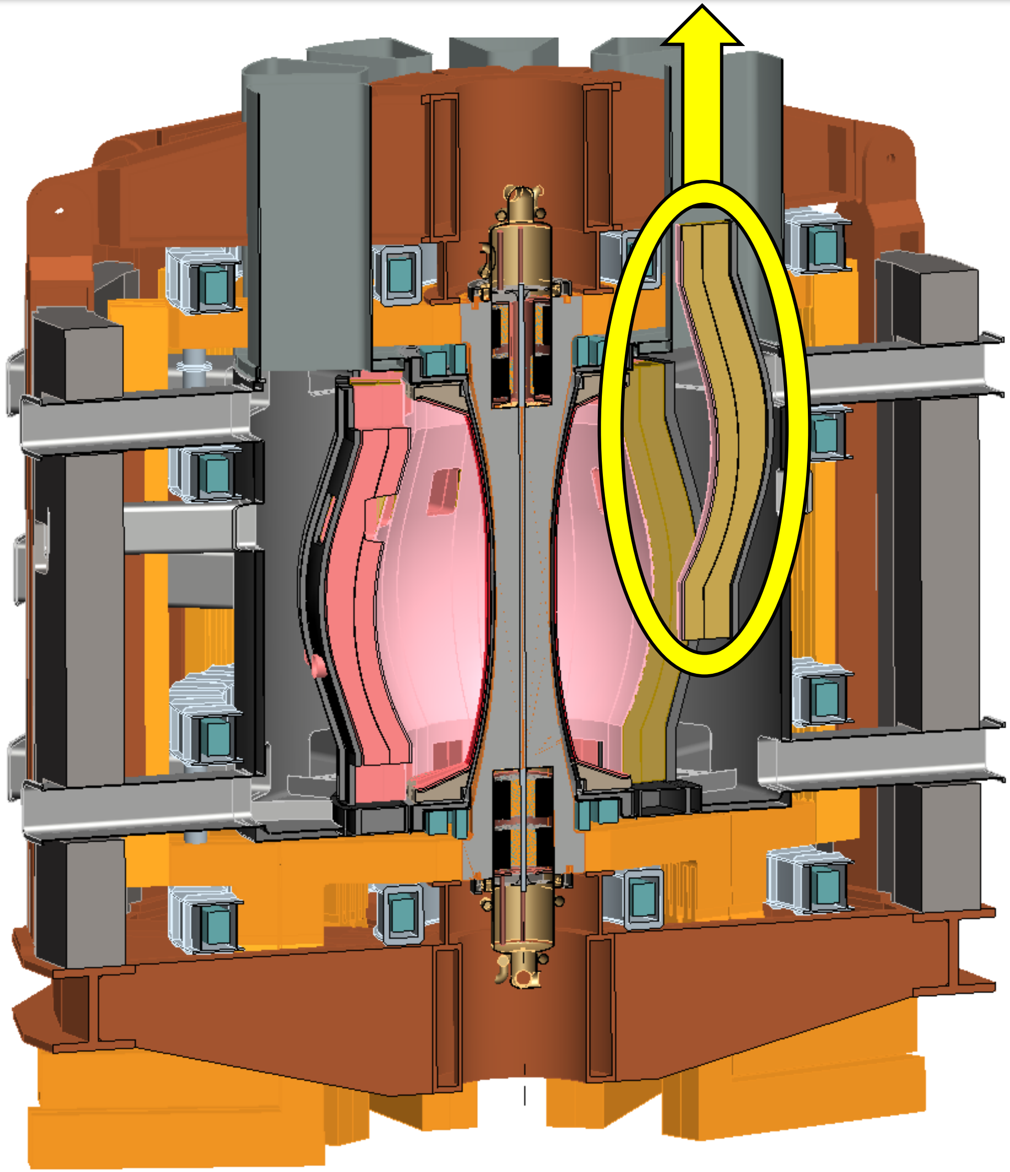


Centerstack Assembly Removed

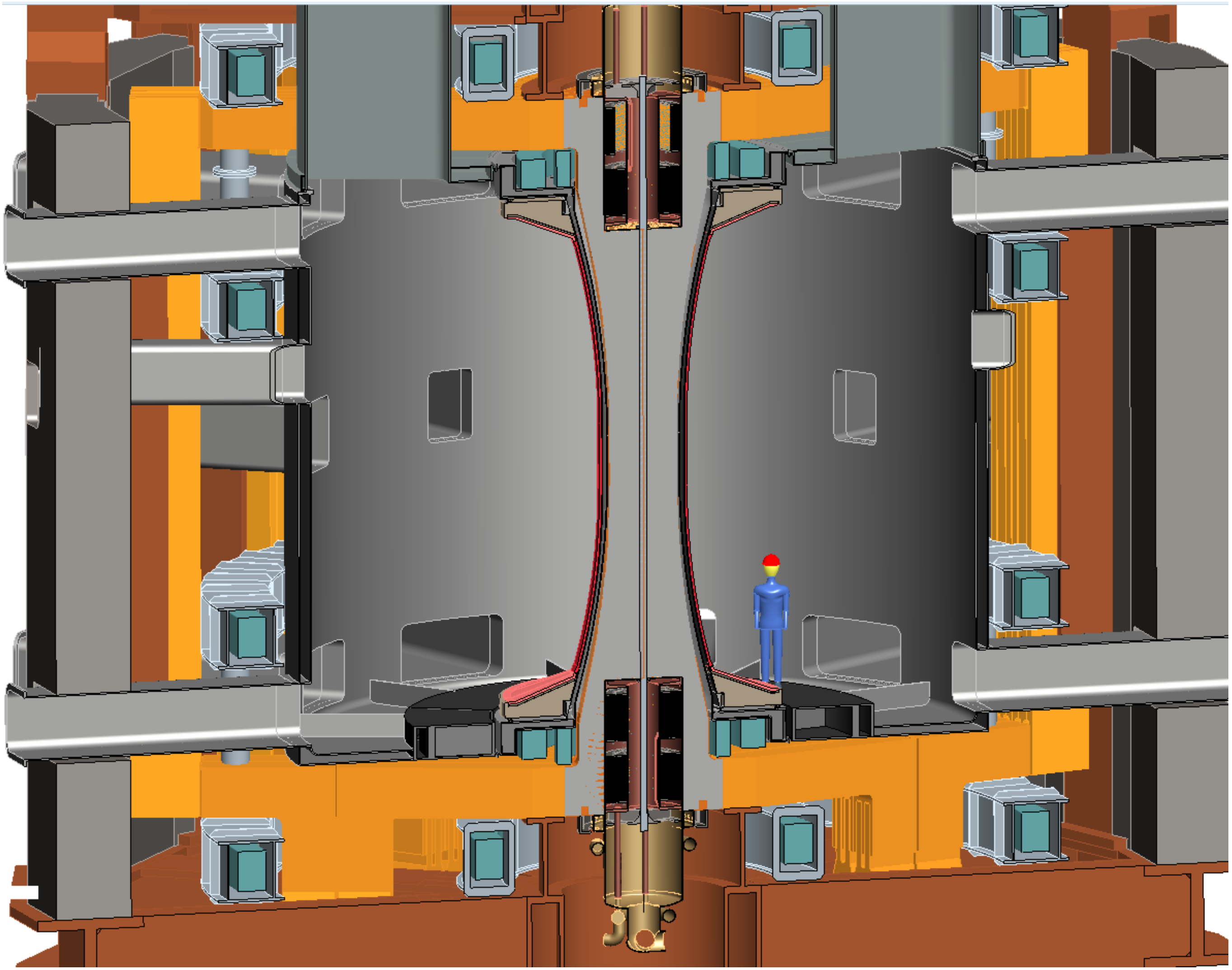
Possible divertor module maintenance scheme using radial installation and vertical translation through vertical ports



R=1.6m ST-FNSF cutaway view showing blanket module maintenance

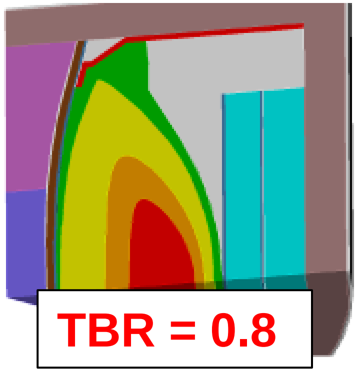


Large cylindrical vessel of R=1.6m FNSF could be used for PMI R&D (hot walls, Super-X?), other blanket configurations

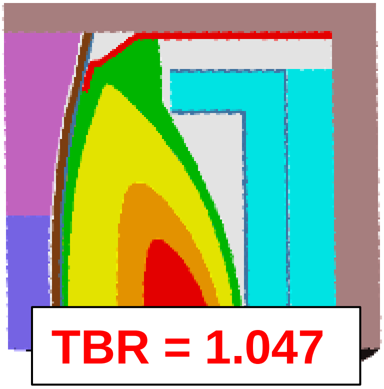


NOTE: TBR values do not include stabilizing shells or penetrations

Straight blanket



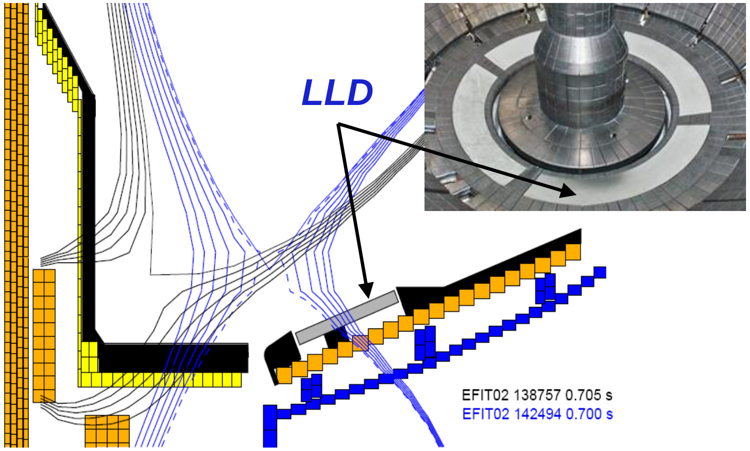
Straight blanket with flat top

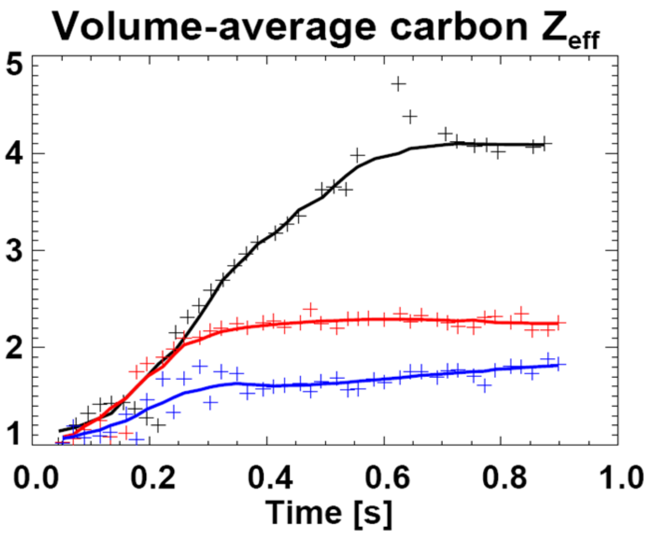


Backup slides

Liquid metal and PFC research

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 reduced 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)

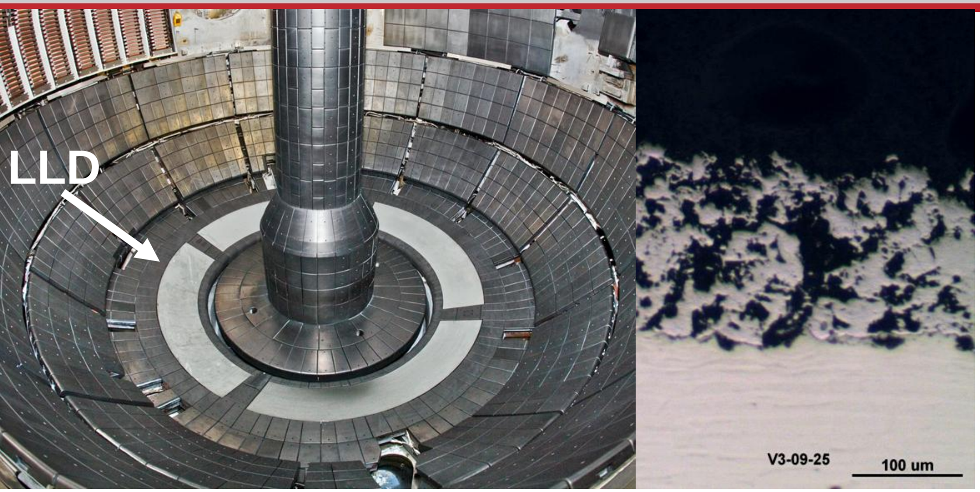
◄Strike-point on LLD, $T_{LLD} < T_{Li\text{-}melt}$

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

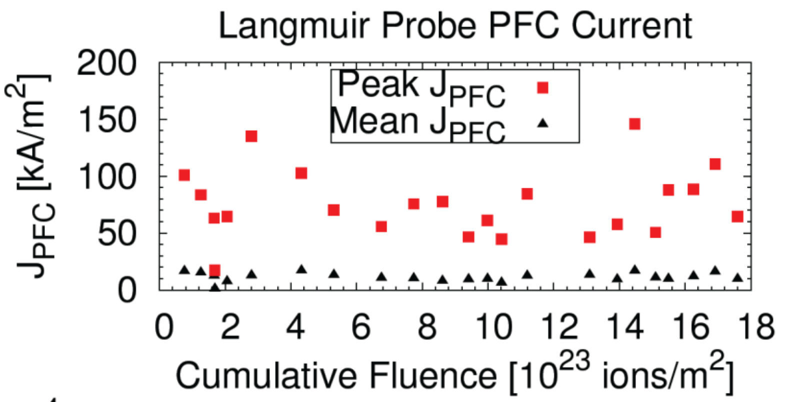
- No ELMs, no → small, small → larger

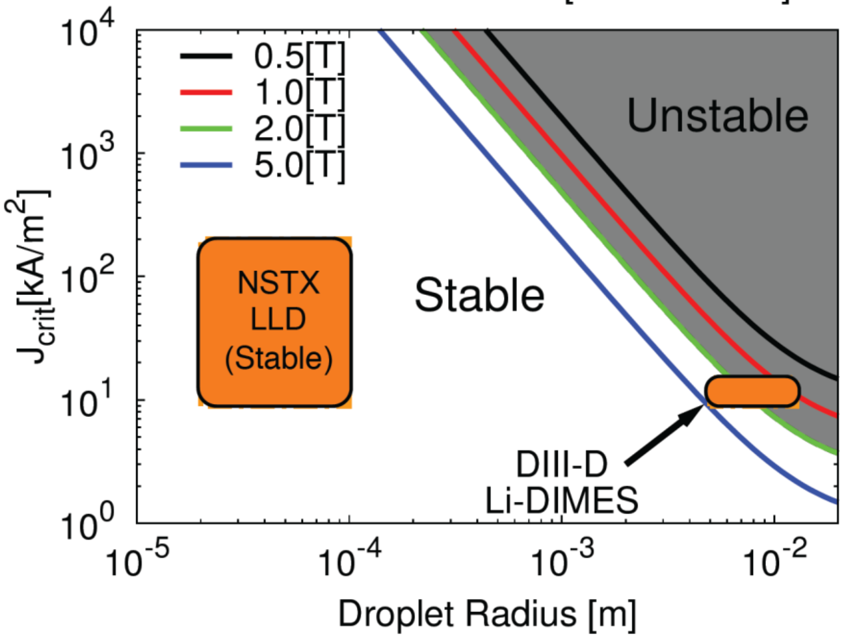
Li + plasma-facing component research will be continued, extended in NSTX-U

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.

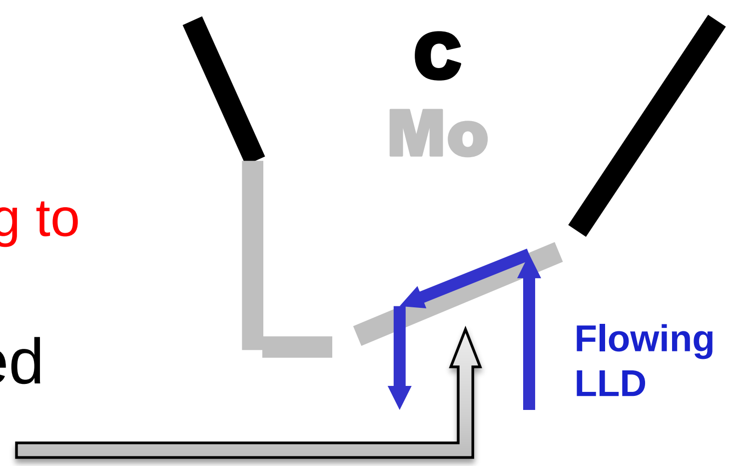
Flowing LLD will be studied as alternative means of particle and power exhaust, access to low recycling

- 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)

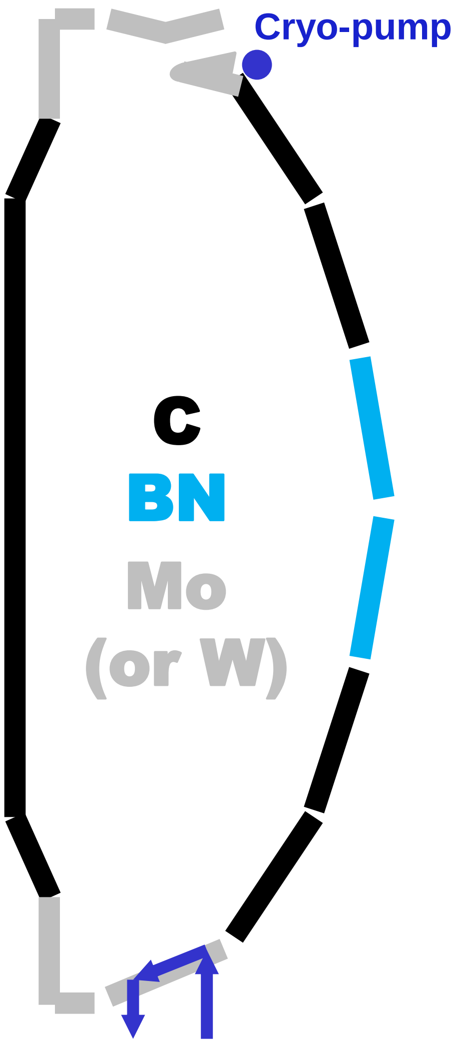
  All systems above require active cooling to mitigate highest heat fluxes of NSTX-U
- Elimination of C from divertor needed for "clean" test of LLD D pumping
  - May need to remove all C PFCs?

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

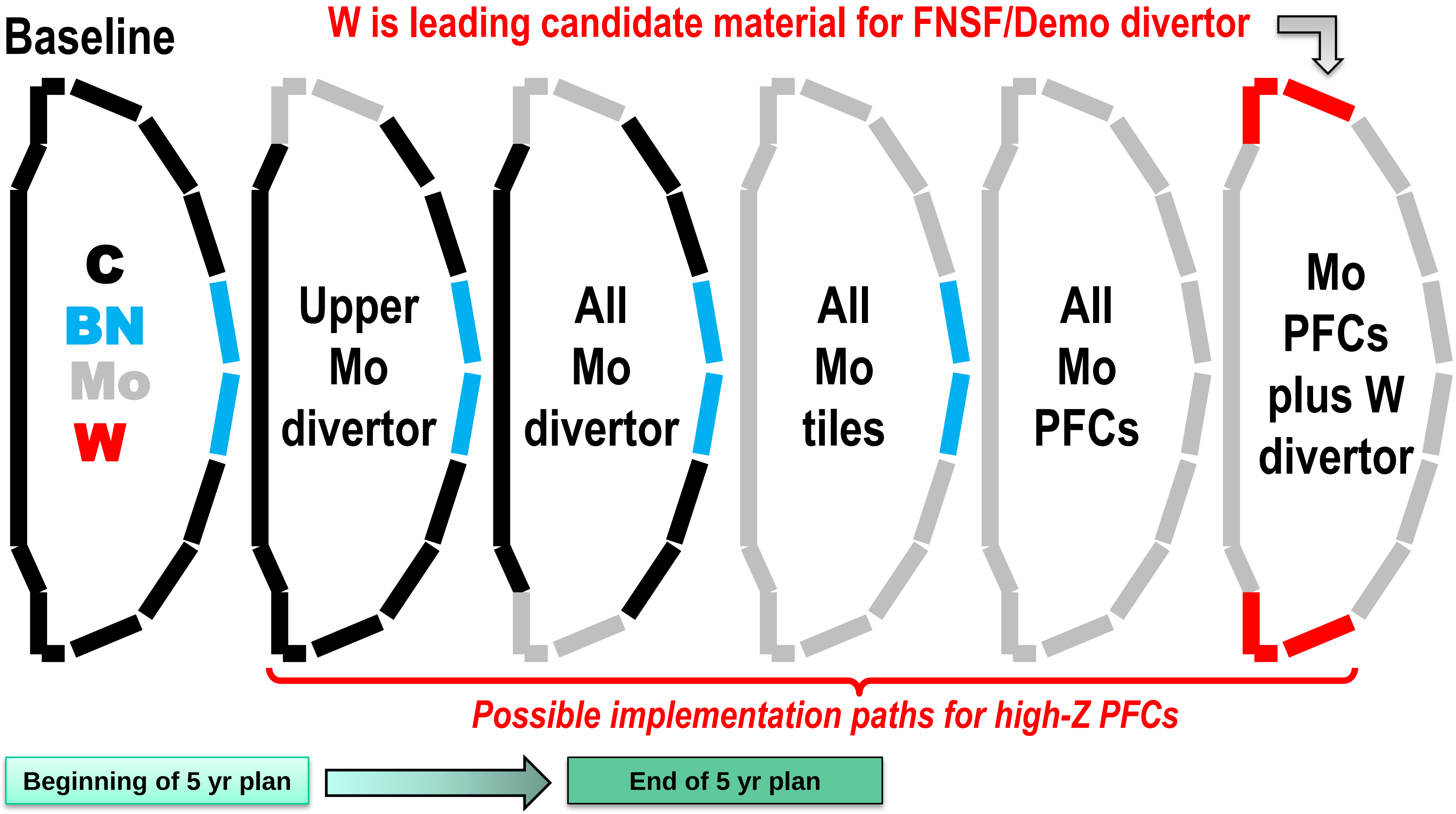


Flowing LLD, MAPP probe, possible replaceable divertor module (RDM)

- 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)

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**



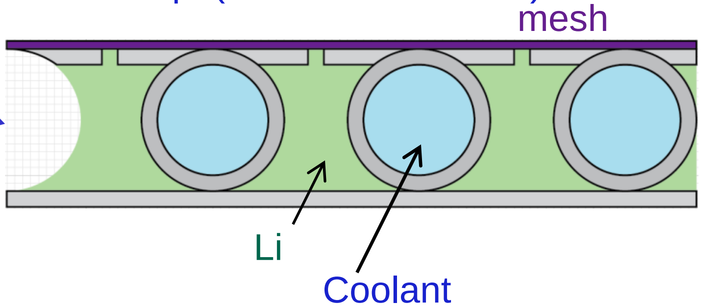
Lab-based R&D on liquid metal technology will inform long term PFC decisions:

Pre-NSTX-U 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 begun June 2012

Thin flowing Li film in FLiLi (Zakharov)

Soaker hose capillary porous system concept (Goldston/Jaworski)



(He, supercritical CO_2, Na heatpipe…)

PPPL Lithium Granule Injector Tested on EAST

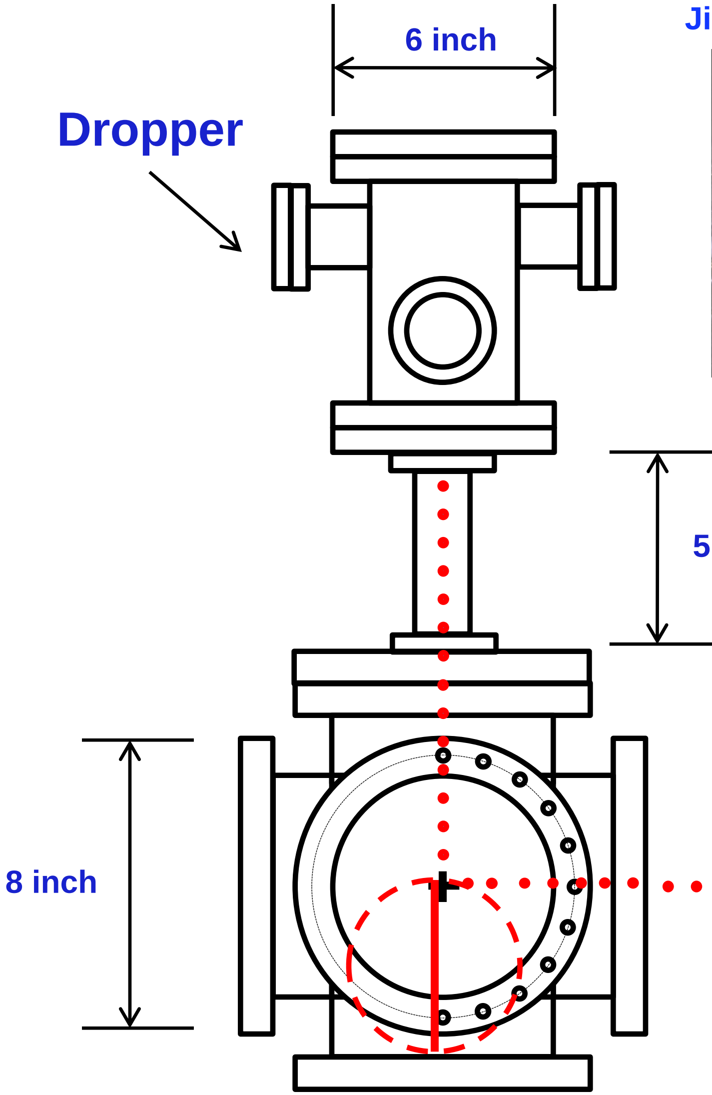
Dennis Mansfield (PPPL, retired)
Lane Roquemore (PPPL)

Independent Control:

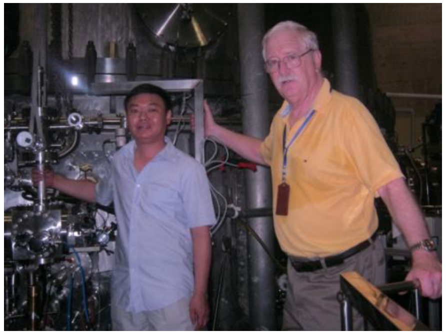
Granule Size
(change between shots)

Injection Speed
(ramp during shots)

Pacing Frequency
(ramp during shots)

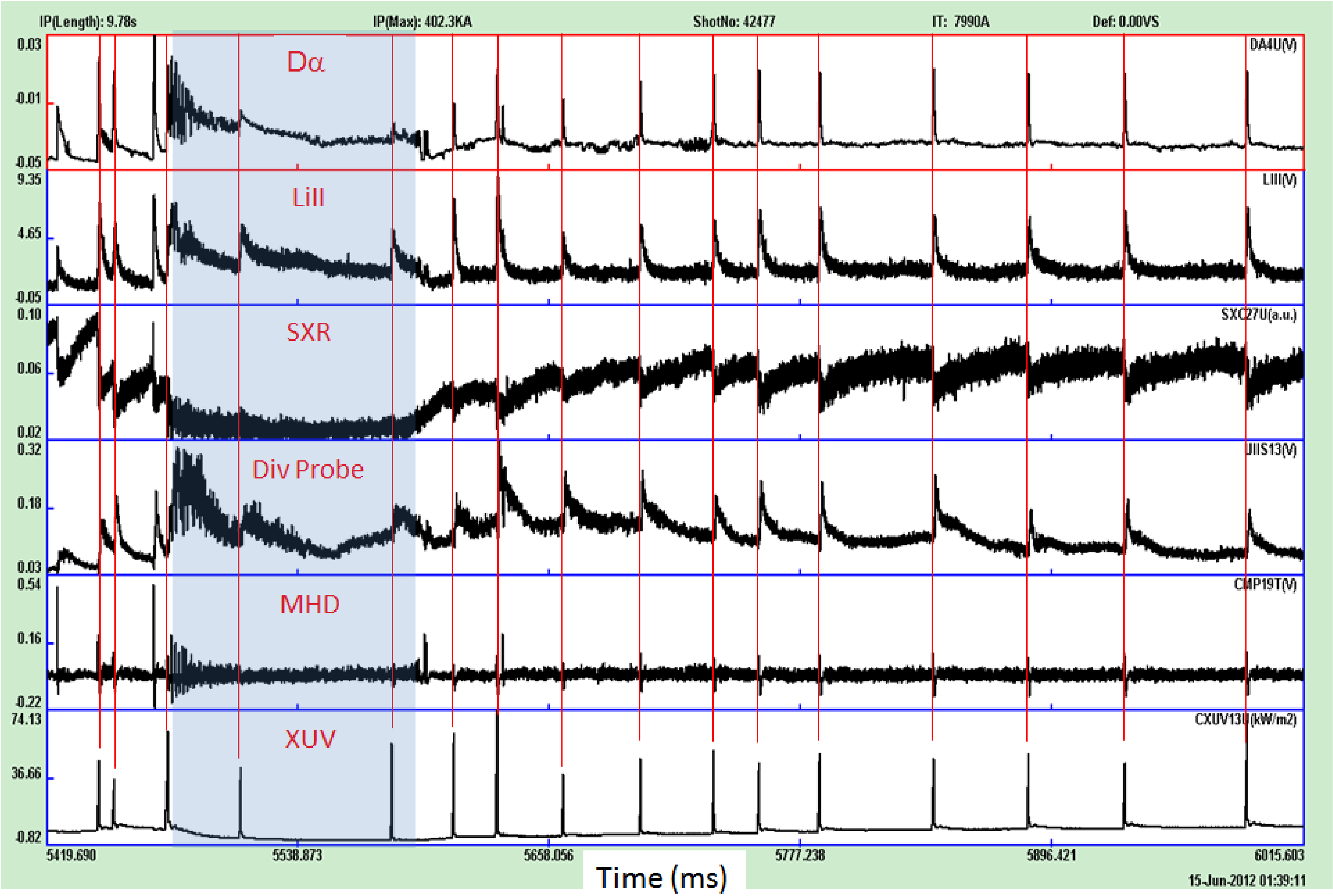


Jiansheng Hu and D. Mansfield



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