Progress on developing the spherical tokamak for fusion applications

Jonathan Menard[1]

T. Brown[1], J. Canik[2], J. Chrzanowski[1], L. Dudek[1], L. El Guebaly[3], S. Gerhardt[1], S. Kaye[1], C. Kessel[1], E. Kolemen[1], M. Kotschenreuther[4], R. Maingi[2], C. Neumeyer[1], M. Ono[1], R. Raman[5], S. Sabbagh[6], V. Soukhanovskii[7], T. Stevenson[1], R. Strykowsky[1], P. Titus[1], P. Valanju[4], G. Voss[8], A. Zolfaghari[1], and the NSTX Upgrade Team

[1]Princeton Plasma Physics Laboratory, Princeton, NJ 08543
[2]Oak Ridge National Laboratory, Oak Ridge, TN, USA
[3]University of Wisconsin, Madison, WI, USA
[4]University of Texas, Austin, TX, USA
[5]University of Washington, Seattle, WA, USA
[6]Columbia University, New York, NY, USA
[7]Lawrence Livermore National Laboratory, Livermore, CA, USA
[8]Culham Centre for Fusion Energy, Abingdon, Oxfordshire, UK

24th IAEA Fusion Energy Conference
San Diego, USA
8-13 October 2012

Paper FTP/3-4

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

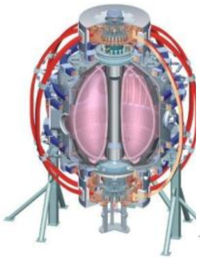
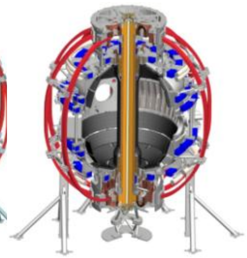
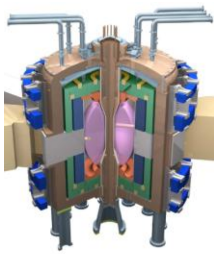
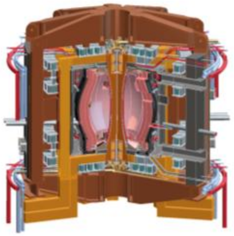
Fusion applications of low-A spherical tokamak (ST)

- 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

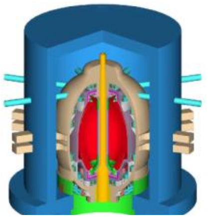
<u>This talk</u>:

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

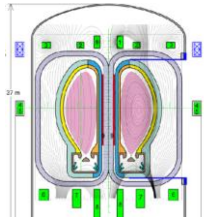
NSTX Upgrade will access next factor of two increase in performance to bridge gaps to next-step STs

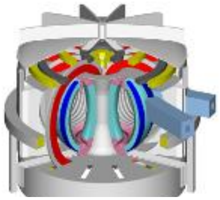
Low-A Power Plants



ARIES-ST (A=1.6)



JUST (A=1.8)



VECTOR (A=2.3)

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

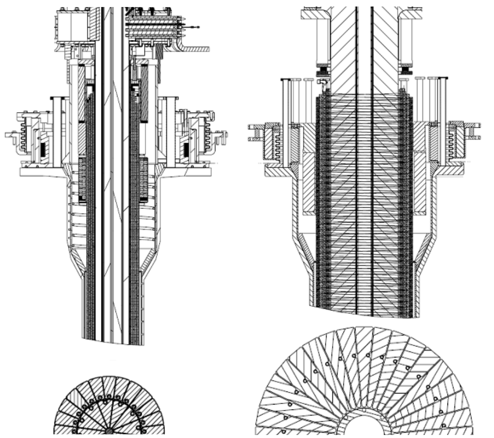
Includes 4MW of high-harmonic fast-wave (HHFW) heating power

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)**

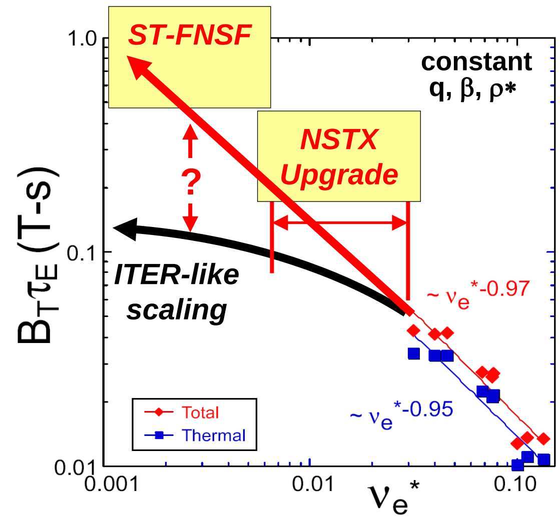
NSTX Upgrade will address critical plasma confinement and sustainment questions by exploiting 2 new capabilities

Previous center-stack | **New center-stack**

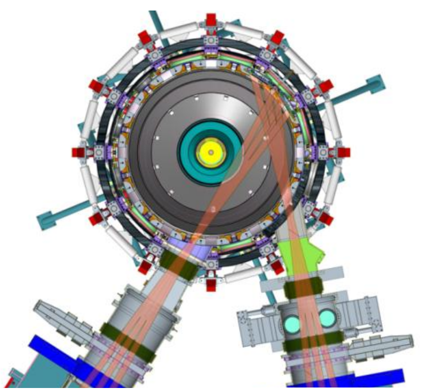


TF OD = 20cm | **TF OD = 40cm**

- 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



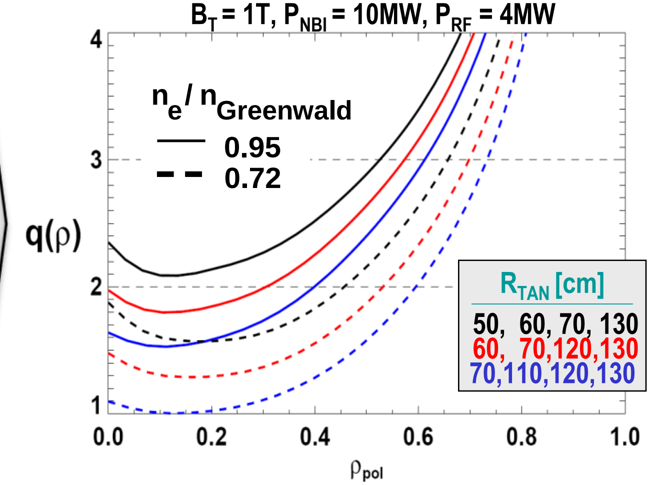
Normalized e-collisionality $\nu_e^* \propto n_e / T_e^2$



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

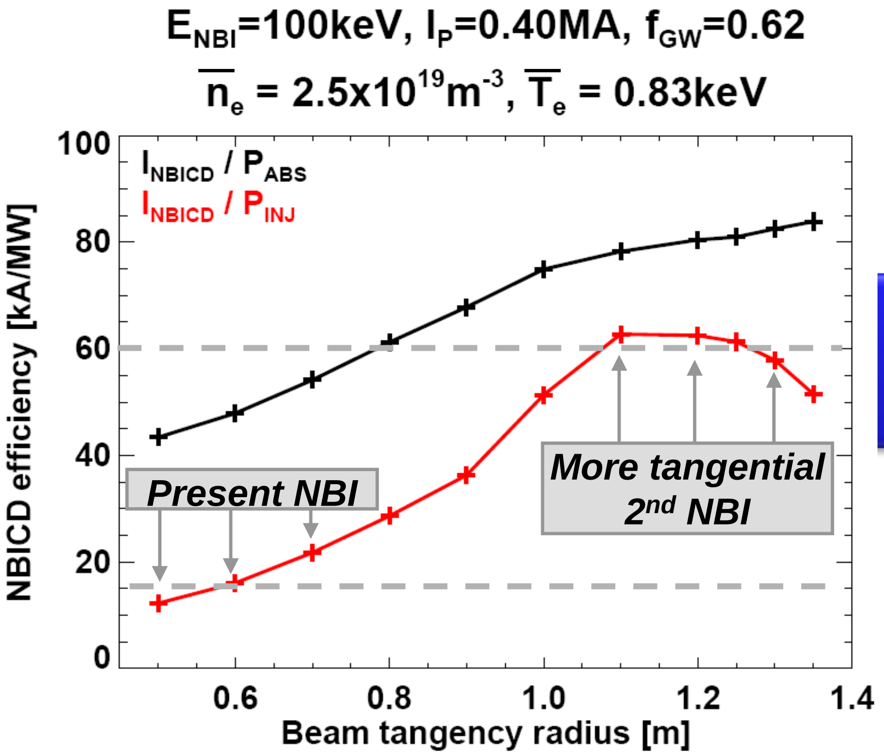
I_P=0.95MA, H_{98y2}=1.2, β_N=5, β_T = 10%
B_T = 1T, P_{NBI} = 10MW, P_{RF} = 4MW



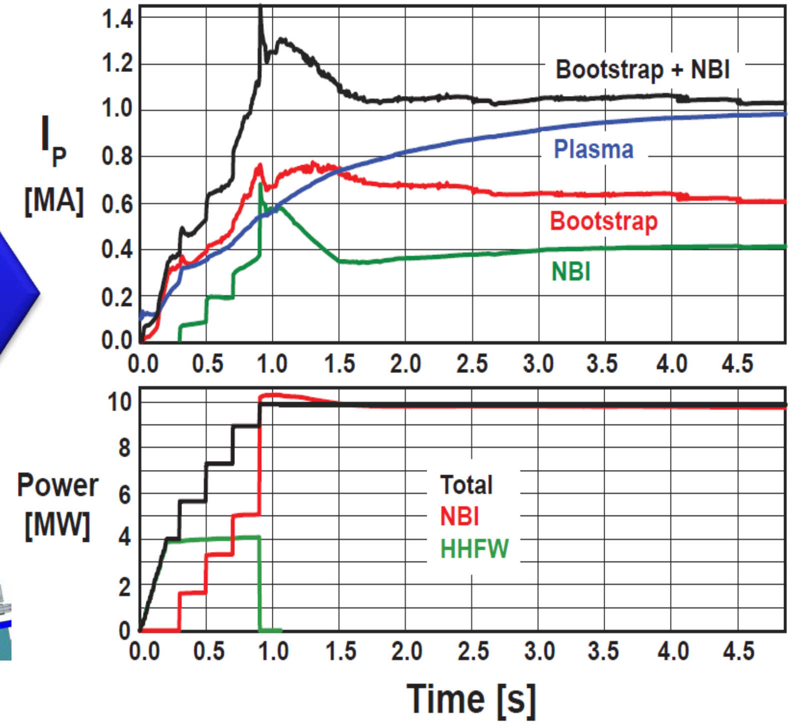
J. Menard, et al., Nucl. Fusion 52 (2012) 083015

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

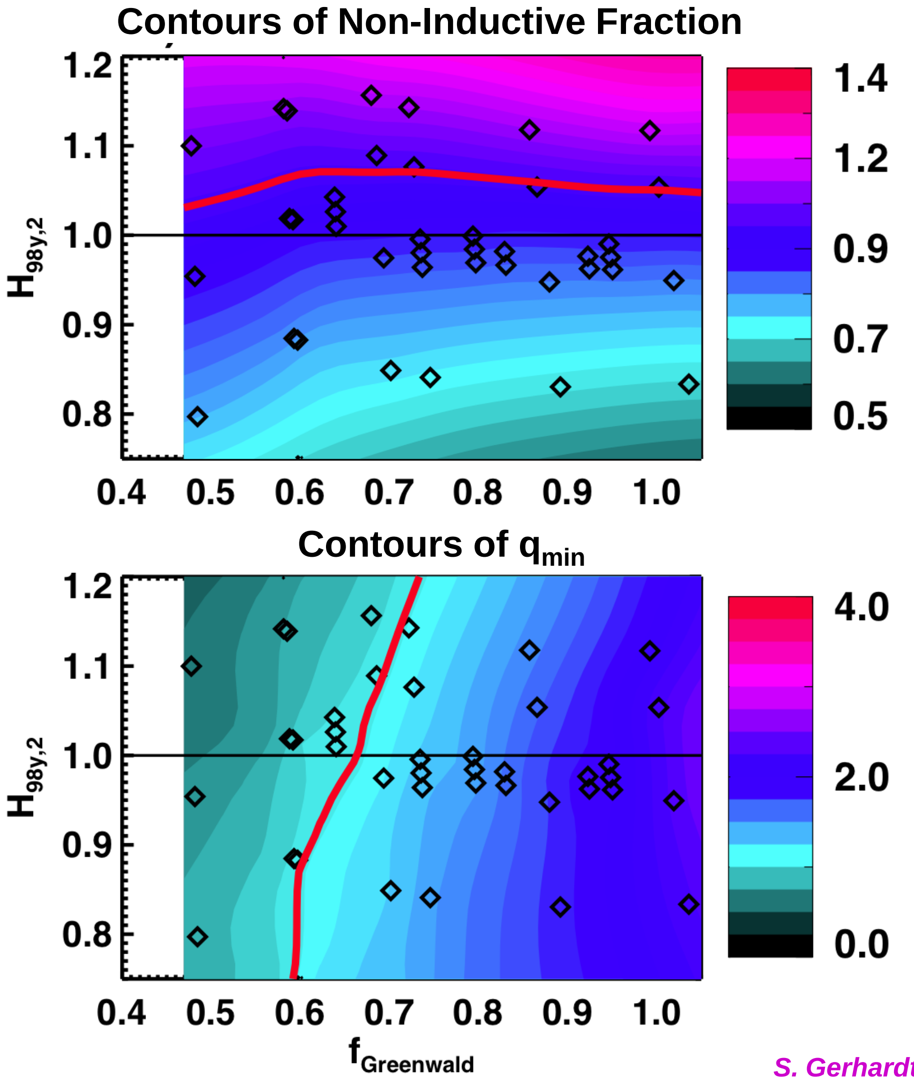


TSC simulation of non-inductive ramp-up from I_P = 0.1MA, T_e=0.5keV target at B_T=1T



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



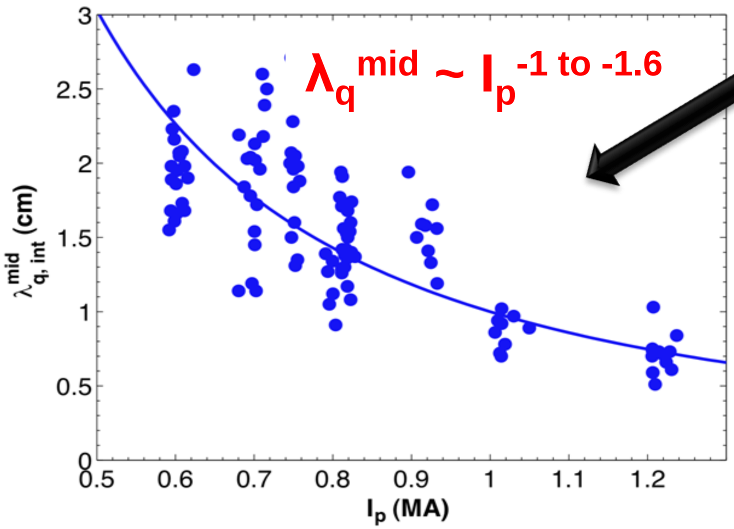
Projected Non-Inductive Current Levels for κ~2.85, A~1.75, f_{GW}=0.7

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):
  - $\chi_{i,ITG}/\chi_{i,Neo}$ ~ 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



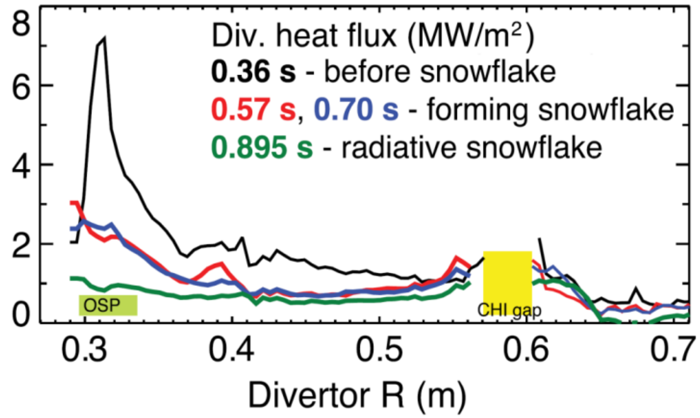
- Divertor heat flux width decreases with increased plasma current I_P

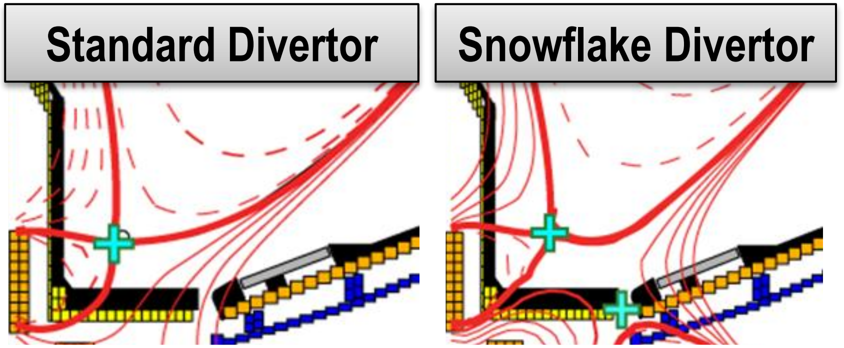
  → 30-45MW/m² in NSTX-U with conventional LSN divertor at full current and power

- 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/m² at I_P = 2MA, P_{AUX}=10-15MW

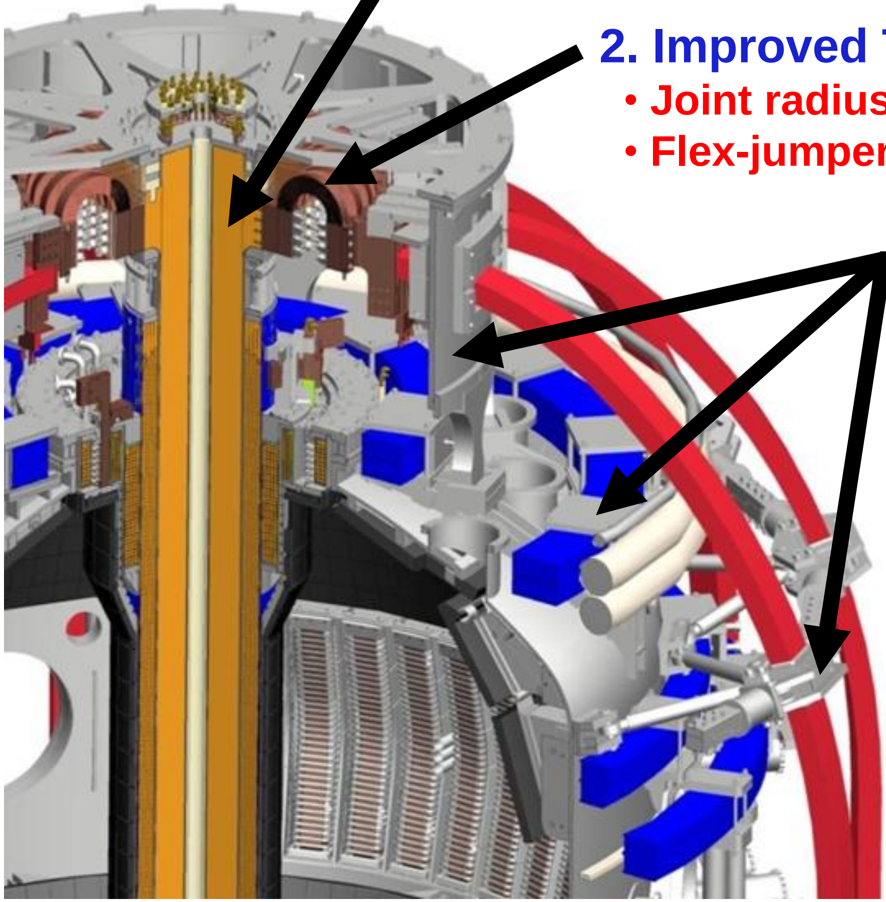
Major engineering challenge of NSTX Upgrade:
Field and current each increase 2× → E-M forces increase 4×

Design solutions for increased loads:

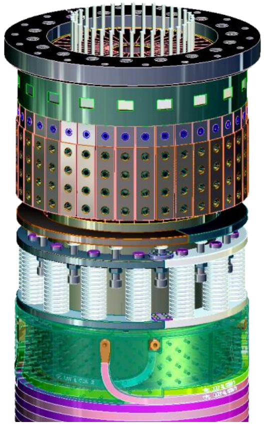
1. Simplified inner TF design
- Single layer of TF conductors

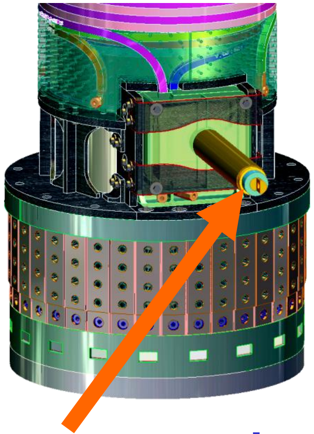
2. Improved TF joint design
- Joint radius increased → lower B
- Flex-jumper improved

3. Reinforcements:
- Umbrella structure
- PF, TF coil supports



Upper TF/ OH Ends

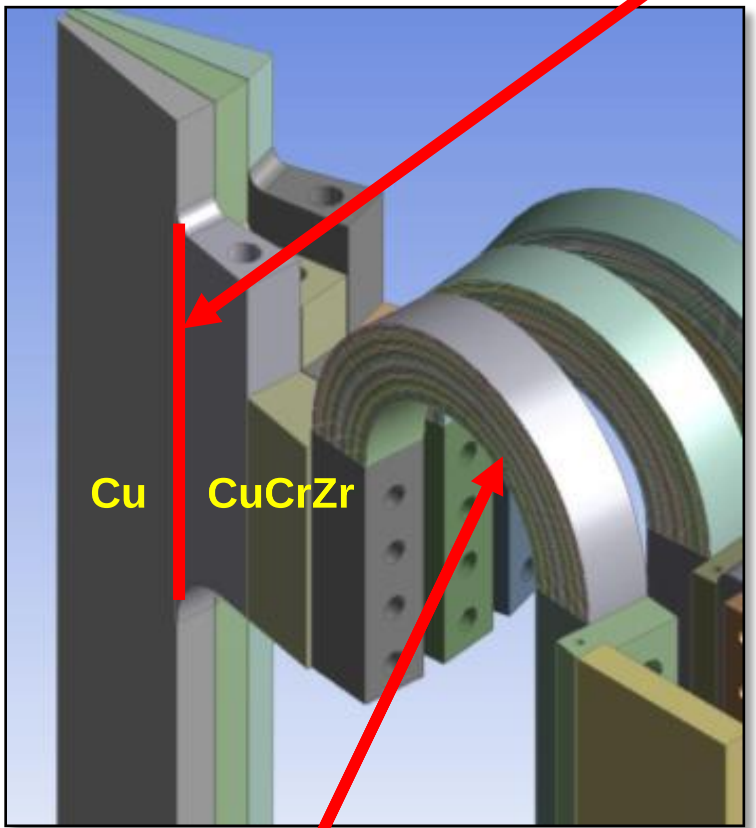


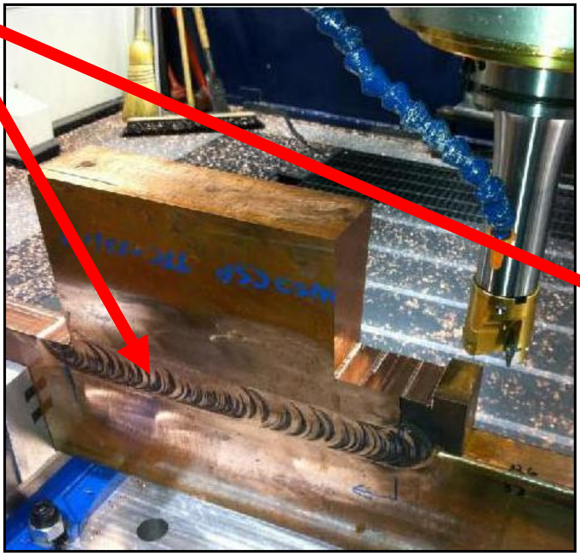


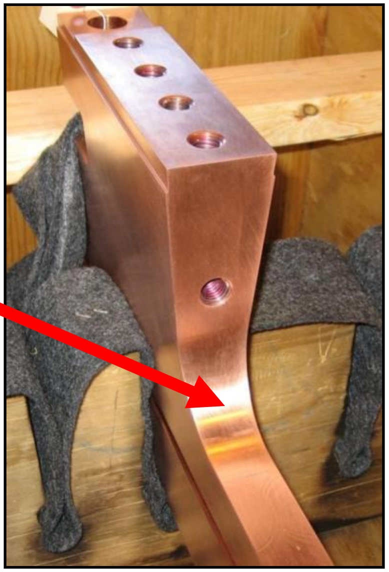
4. OH leads placed at bottom, made coaxial to minimize forces, error-fields

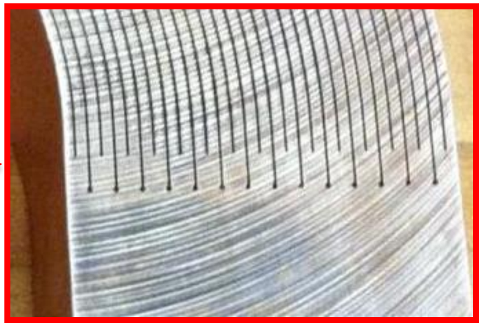
Substantial R&D completed to achieve higher toroidal field with new center-stack

Friction-stir welded joint





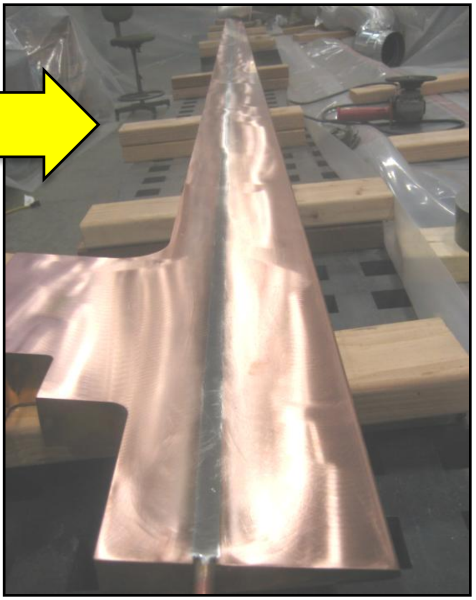




Flexible TF strap

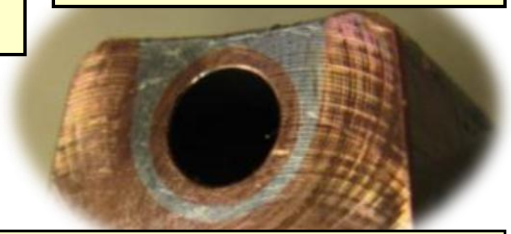
Wire EDM used instead of laminated build

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

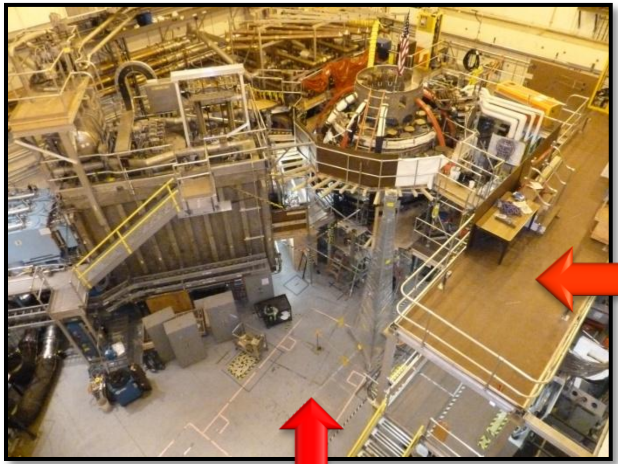
Vacuum-pressure impregnation (VPI) using special cyanate-ester epoxy blend (CTD-425) required for shear strength will be used for the inner TF assembly

Recent successful VPI trials

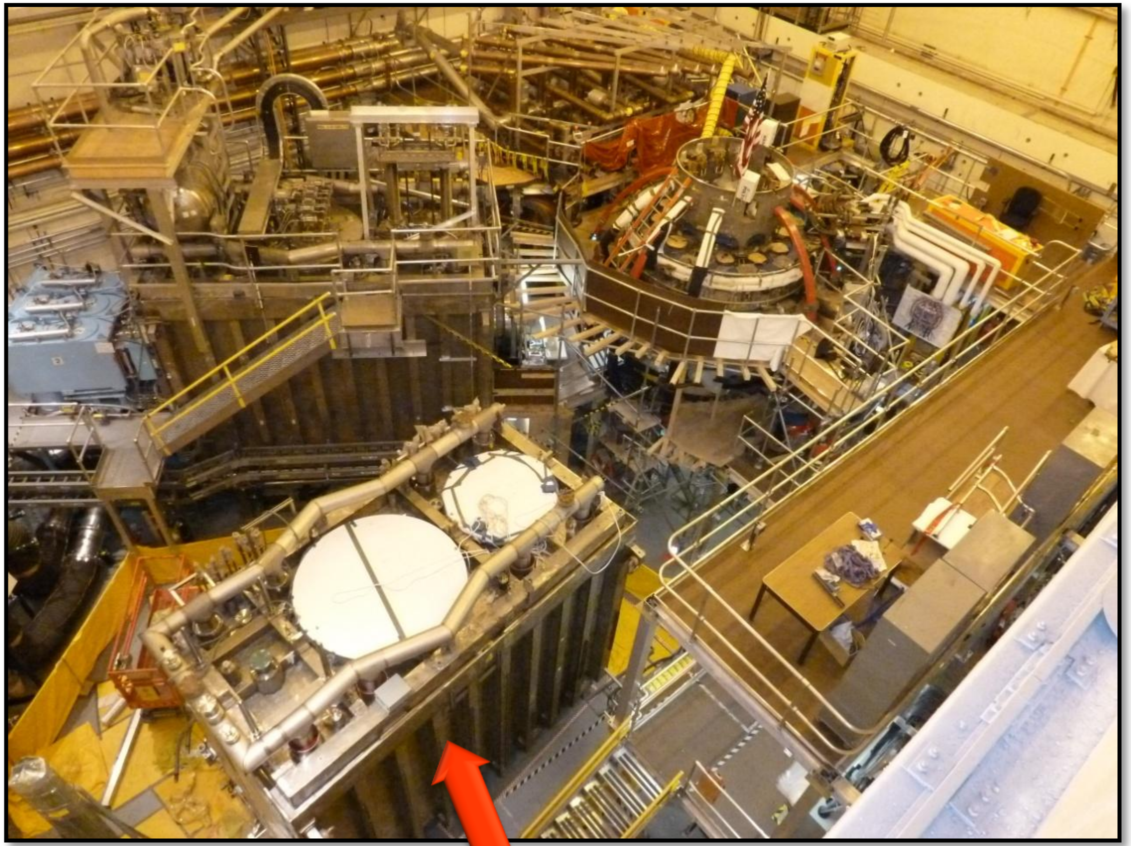
Quadrant mold for VPI nearly ready

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

Oct. 2011: Start of construction



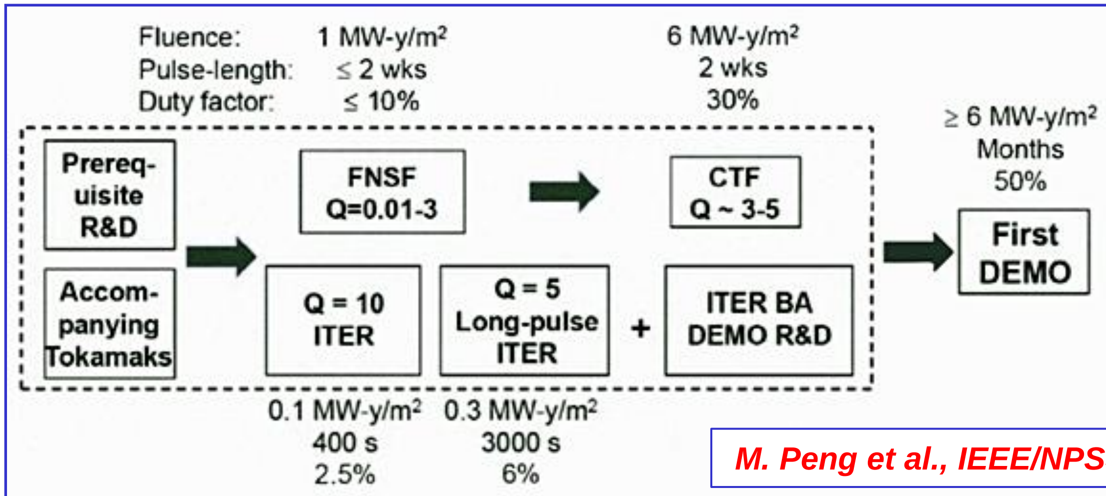
Sept. 2011: NBI space cleared
Upper diagnostic platform installed

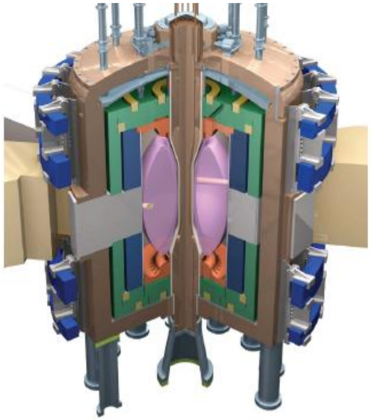


Oct. 2012: 2nd NBI box moved to test cell

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



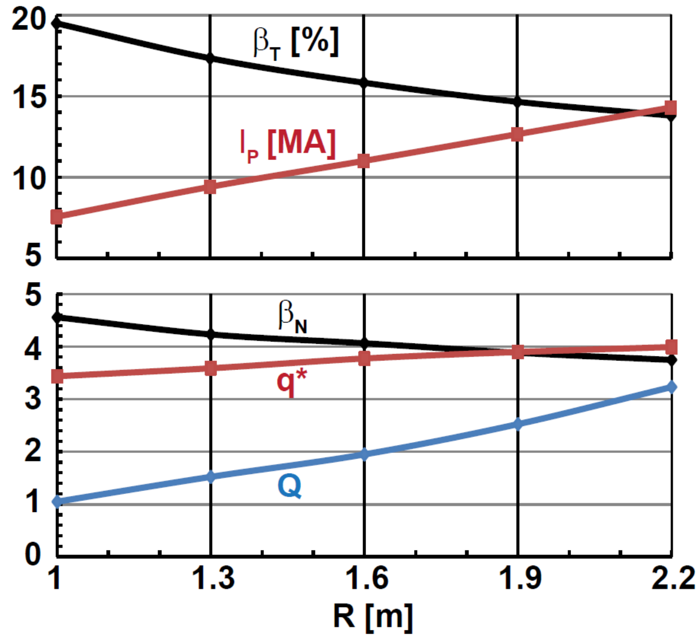


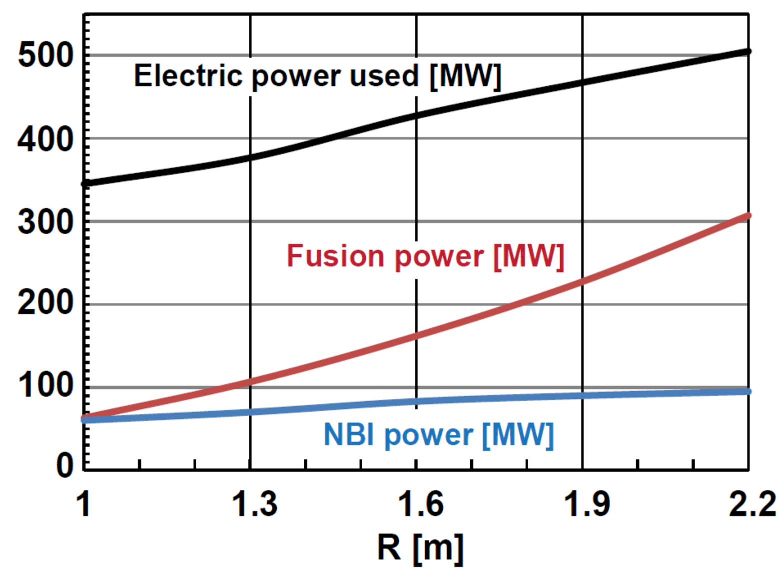
M. Peng et al., IEEE/NPSS Paper S04A-2 - 24th SOFE Conf. (2011)

- 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)

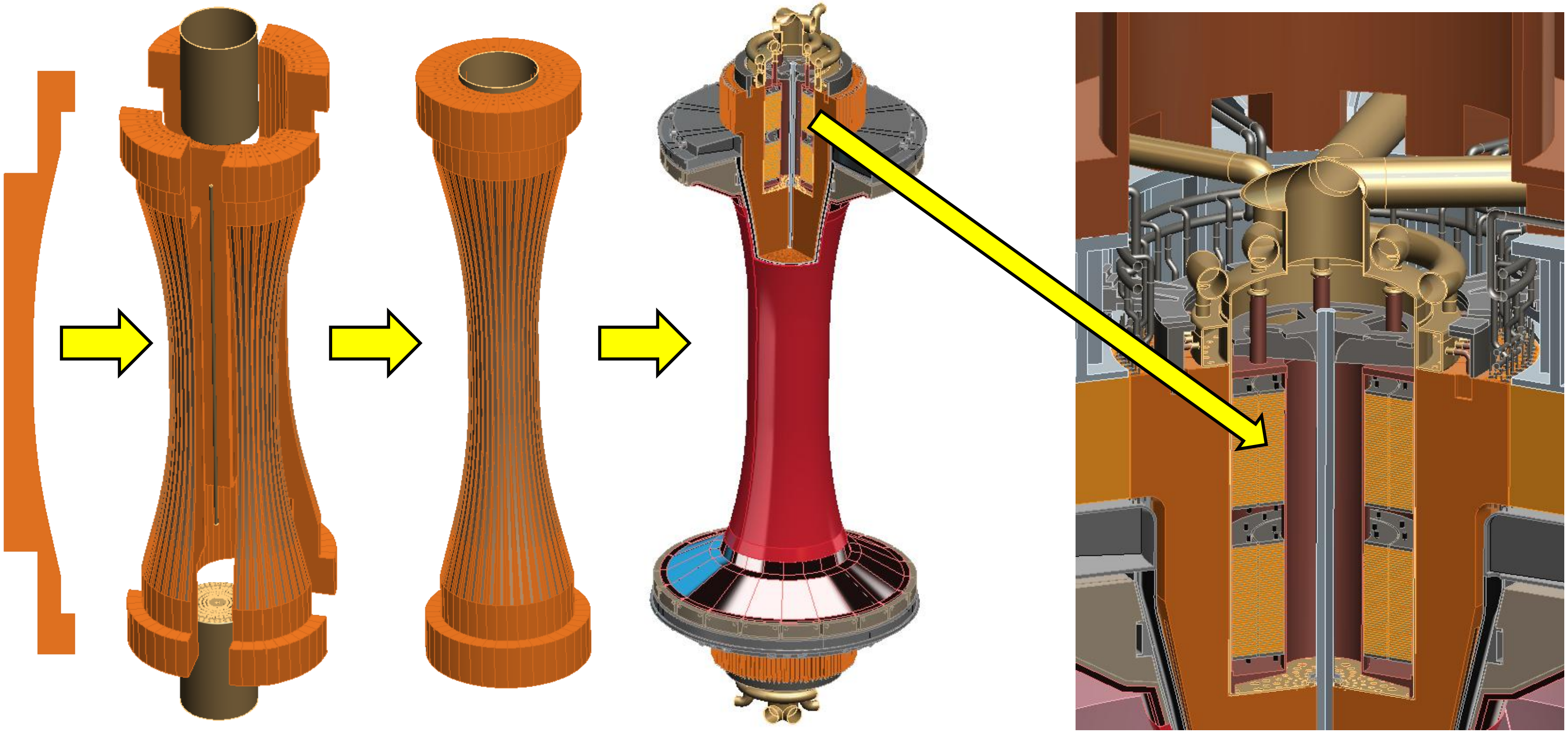




- Larger R lowers β_T & β_N, increases q*
- **Comparable/higher β_T and β_N values already sustained in NSTX**

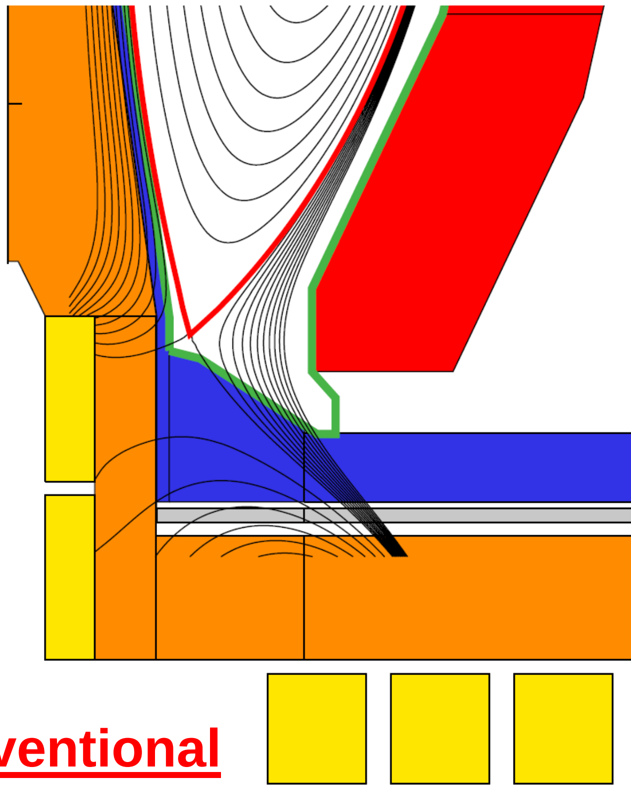
- Q = 1 → 3, P_{fusion} = 60MW → 300MW
  → 5× increase in T consumption
- 2-3× higher wall loading for CTF/Pilot Plant if β_N → 6, H_{98} → 1.5 (not shown)

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

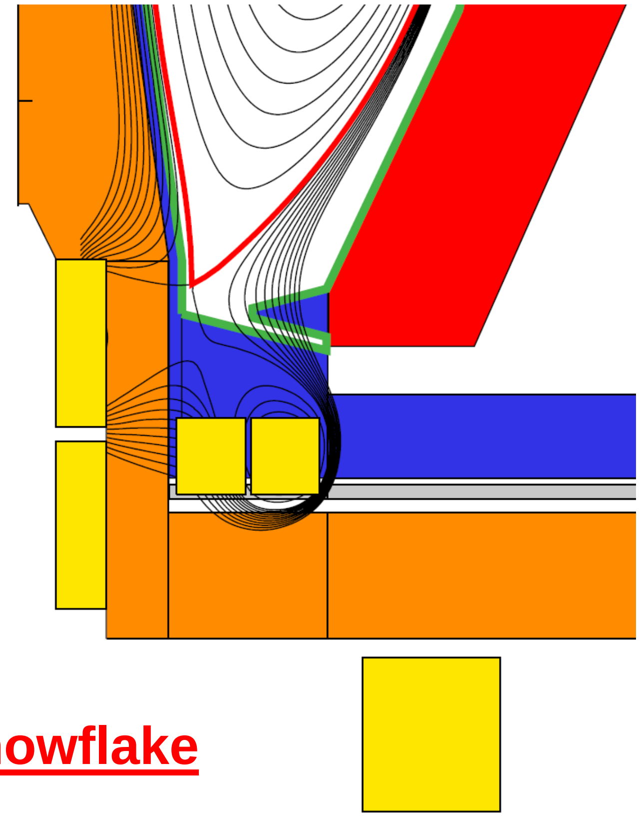


- **Like NSTX-U, use TF wedge segments (but brazed/pressed-fit together)**
  – Coolant paths: gun-drilled holes or NSTX-U-like grooves in wedge + welded tube
- **Bitter-plate divertor PF magnets in ends of TF enable high triangularity**
  – **NSTX data:** High δ > 0.55 and shaping $S \equiv q_{95}I_P/aB_T$ > 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



Field-line angle of incidence at strike-point = 1°



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

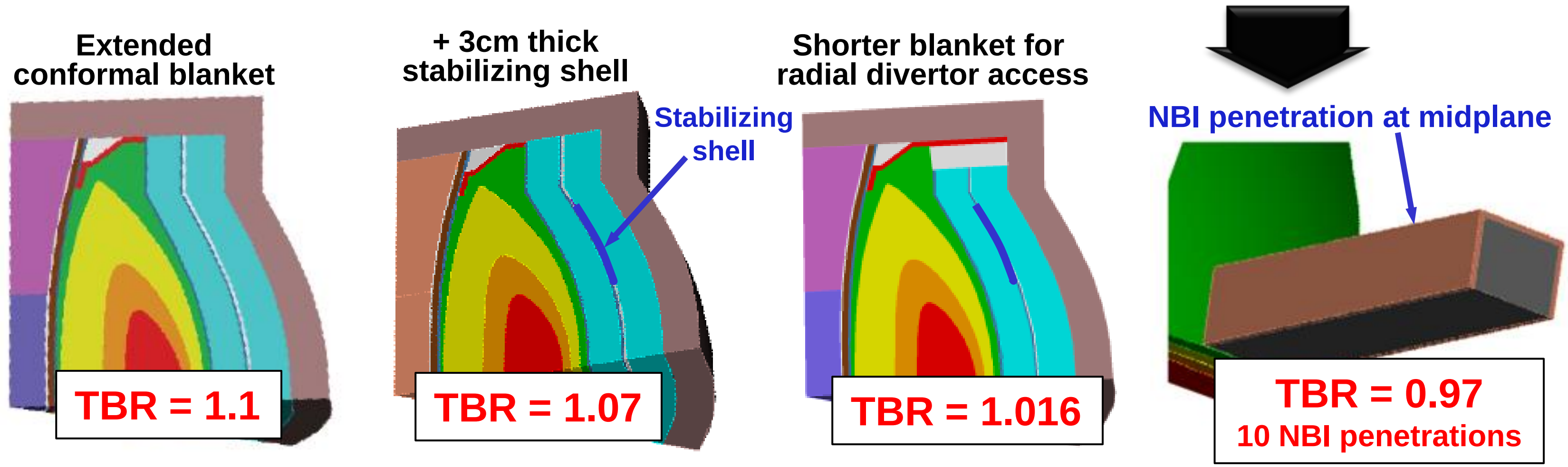
- **Flux expansion = 40-60, δ_x ~ 0.62**
- **1/sin(θ_{plate}) = 1-1.5**
- **Good detachment (NSTX data) and cryo-pumping (NSTX-U modeling)**

- **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**

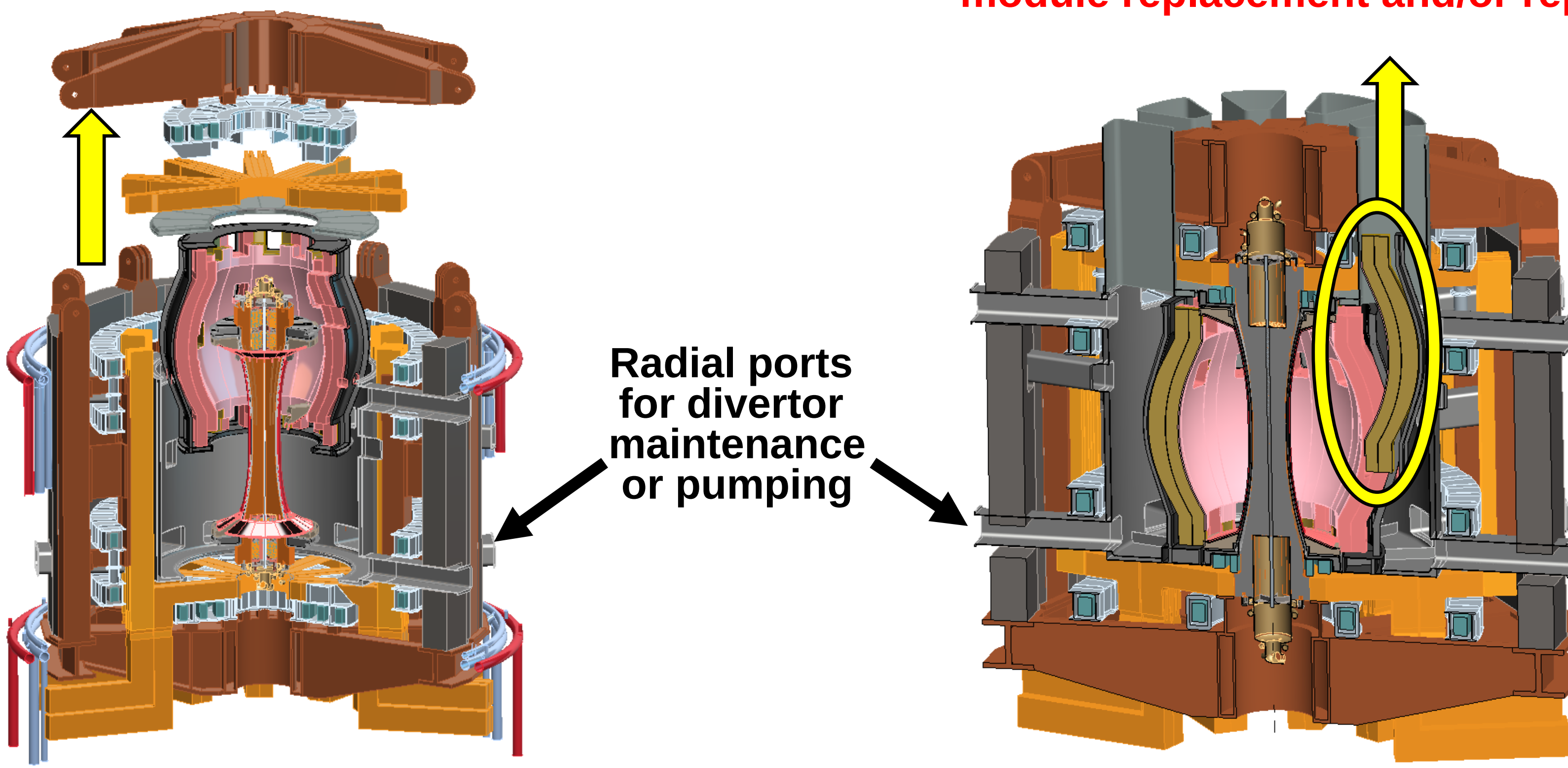


- Future work: assess smaller R, 3D effects (inter-blanket gaps, test-blankets)

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?**



- **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