U.S. DEPARTMENT OF ENERGY | Office of Science

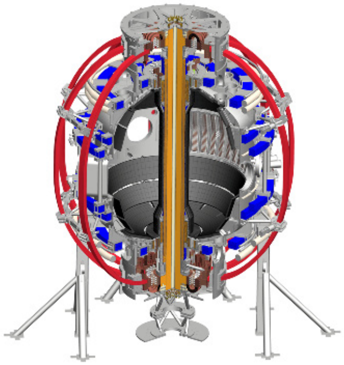
NSTX-U

Progress and plans for NSTX Upgrade and prospects for next-step spherical tori

Jonathan Menard, PPPL

Plasma Physics Seminar (Physics/ECE/NE 922)
Physics Building, Room 2241
September 14, 2015

PPPL



Outline

- ST Overview and Motivation
- NSTX-U Mission and Status
- NSTX/NSTX-U Research Highlights
  - Global MHD Stability
  - Transport and Turbulence
  - Energetic Particles
  - Power Exhaust
  - Plasma Start-up
- Brief Overview of NSTX-U Research Plans
- Summary

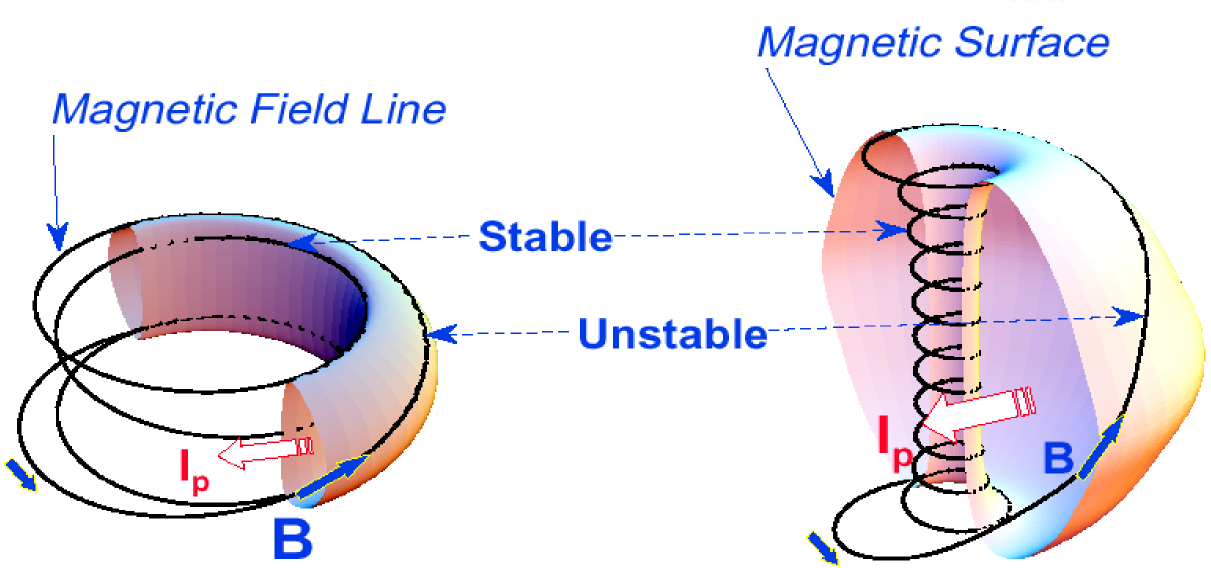
"Spherical" tokamak (ST) has aspect ratio A < 2

Aspect Ratio $A = R/a$	Elongation $\kappa = b/a$	Toroidal Beta $\beta_T = \langle p \rangle / (B_{T0}^2 / 2\mu_0)$

- Natural elongation makes its spherical appearance
- Favorable average curvature improves stability at high beta → access high normalized pressure, rapid rotation, low collisionality

Tokamak

$A \sim 3\text{-}4$,
$\kappa = 1.5\text{-}2$,
$q_{95} = 3\text{-}4$,
$\beta_T = 3\text{-}10\%$



ST

$A \sim 1.2\text{-}2$,
$\kappa = 2\text{-}3$,
$q_{95} = 6\text{-}20$,
$\beta_T = 10\text{-}40\%$

ST can be compact, high β, and high confinement

Higher elongation κ and low A lead to higher I_P , β_T and τ_E

Aspect Ratio A = R /a	**Elongation κ = b/a**	**Toroidal beta $\beta_T = \langle p \rangle / (B_{T0}^2/2\mu_0)$**

- ST has high I_P due to high κ and low A

  $I_P \sim I_{TF} (1 + \kappa^2) / (2 A^2 q^*)$

- I_P increases tokamak performance

  Energy confinement time $\tau_E \propto I_P$

  Normalized pressure $\beta_T [\%] \equiv \beta_N \; I_P / (aB_{T0})$

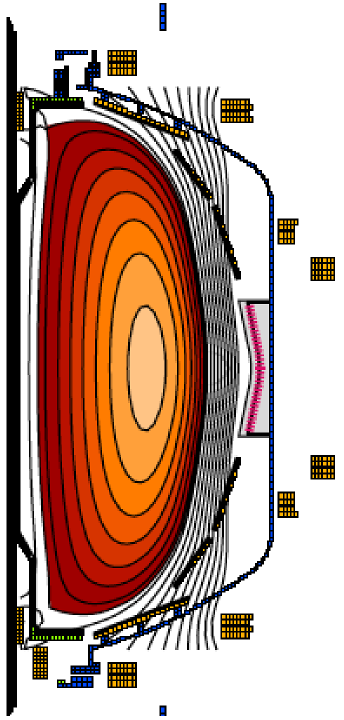
- ST achieves high performance cost effectively

  $I_P \sim I_{TF}$ for ST due to low A and high κ

  ↑

  $

High κ ~ 3.0 equilibrium in NSTX

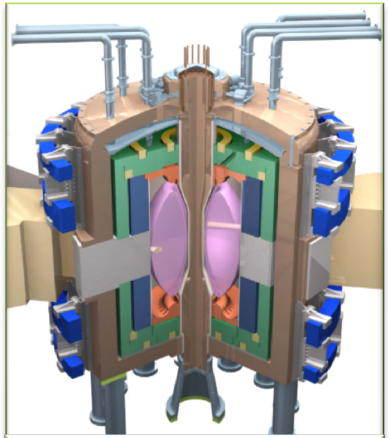


Fusion technology development is major challenge

Fusion Nuclear Science Facility (FNSF) could aid development

Need to develop reliable, qualified nuclear and other components unique to fusion:

- Divertor, plasma facing components
- Blanket and Integral First Wall
- Vacuum Vessel and Shield
- Tritium Fuel Cycle
- Remote Maintenance Components



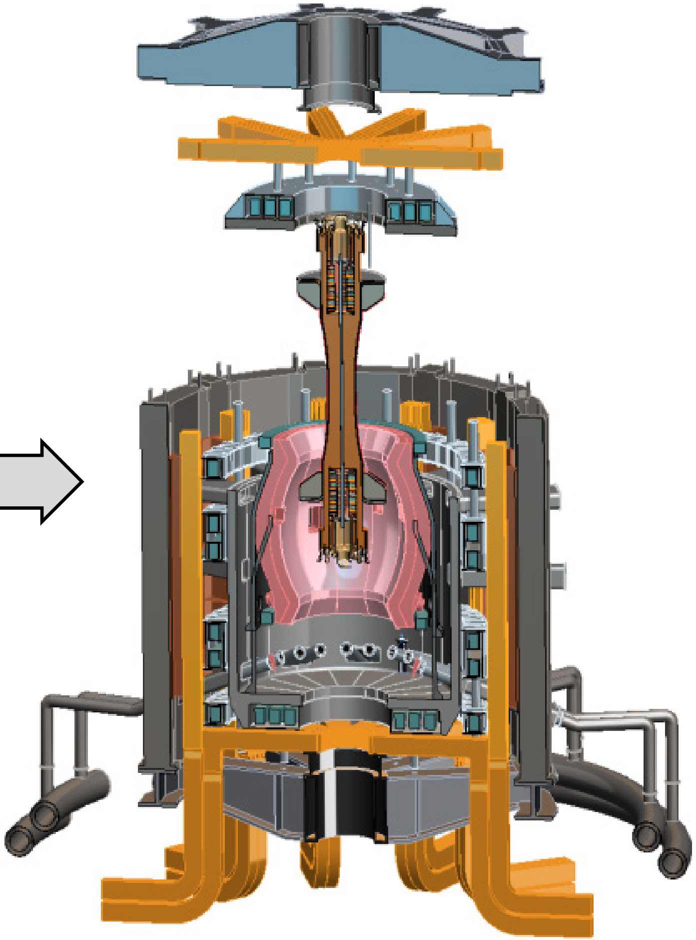
ST-FNSF

- Without R&D, fusion components could fail prematurely
- FNSF can help develop reliable fusion components
- FNSF facilities must be: modest cost, low T, and reliable

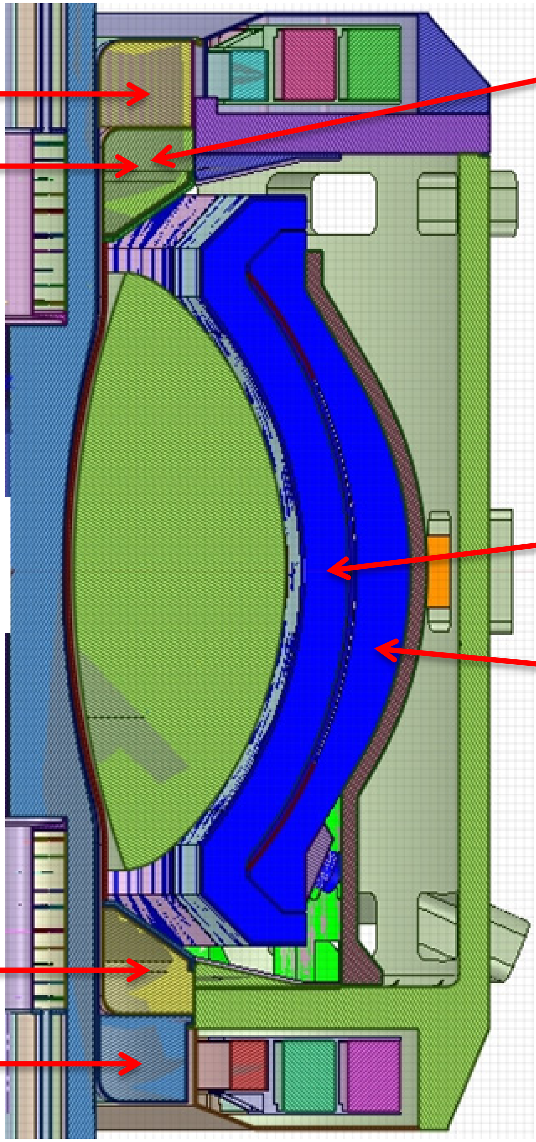
Design studies show ST potentially attractive as FNSF

PPPL ST-FNSF concept

- **Projected to access high neutron wall loading at moderate R, P_{fusion}**
  - W_n ~ 1-2 MW/m^2 , P_{fus} ~ 50-200MW, R ~ 0.8-1.8m
- **Modular, simplified maintenance**
- **Tritium breeding ratio (TBR) near 1**
  - Requires sufficiently large R, careful design



Tritium breeding at top/bottom of ST device important for achieving high Tritium Breeding Ratio (TBR) > 1



R=1.7m configuration

Breeding at CS ends important:
ΔTBR = +0.07

Inner Blanket Segment = **0.81**

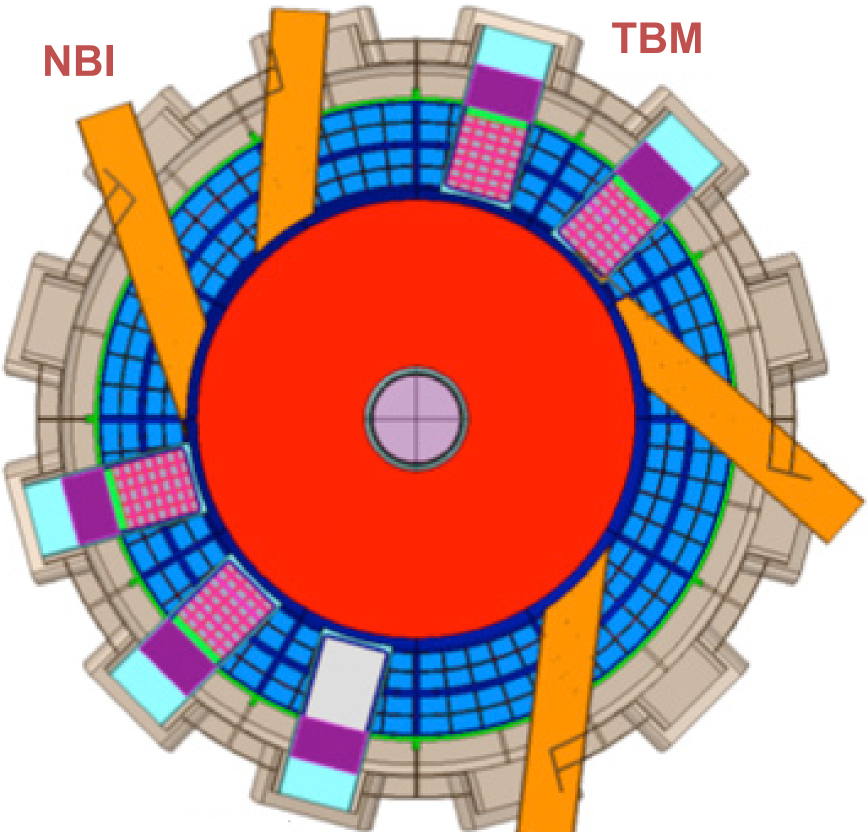
Outer Blanket Segment = **0.15**

Total TBR ~ 1.03 with no penetrations or ports
(heterogenous outboard blanket)

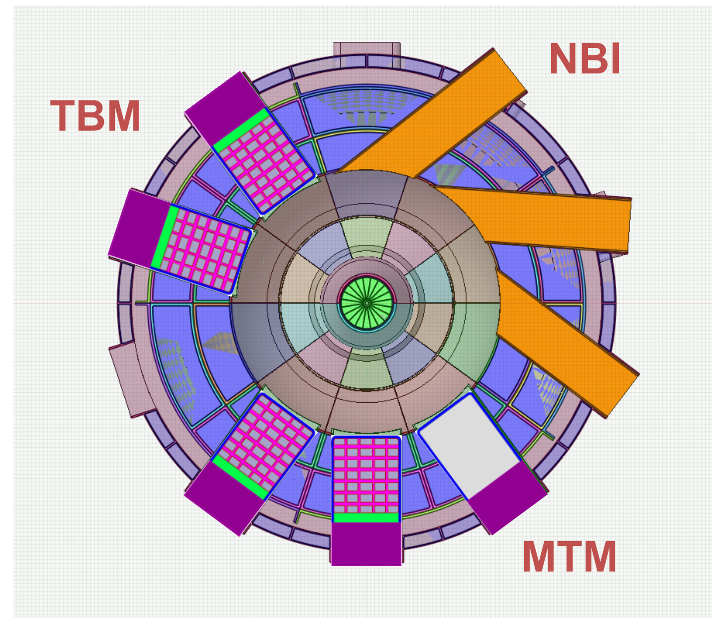
Compatible with long-leg / Super-X divertor

Recent studies show sufficiently large major radius ST-FNSF can be tritium self-sufficient

R=1.7m: **TBR ≥ 1**



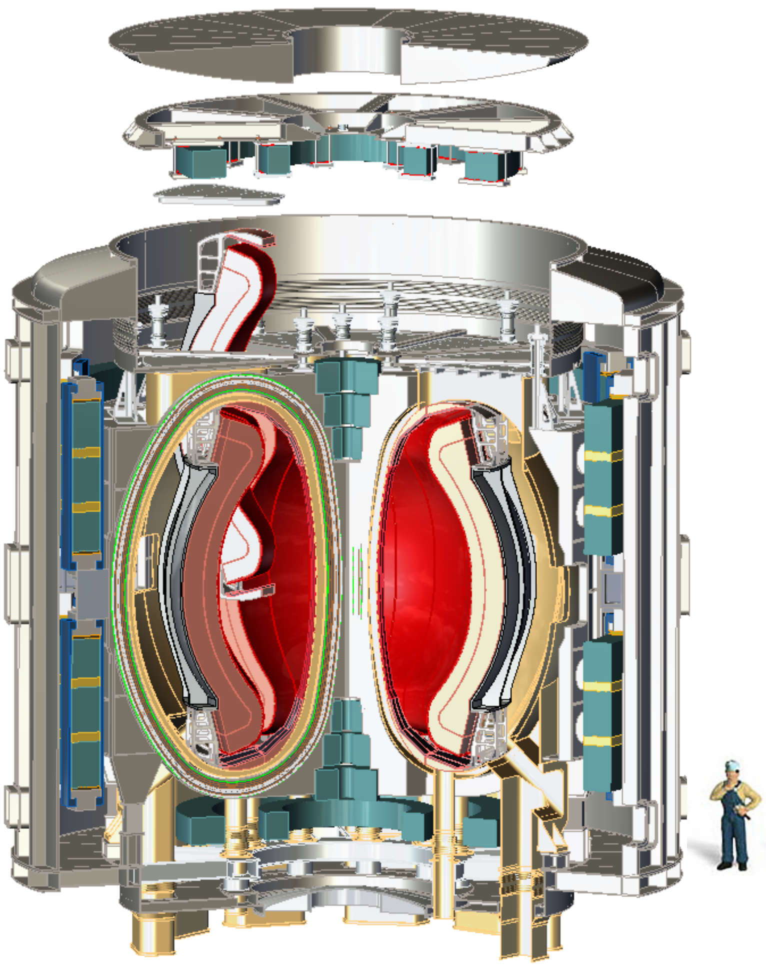
R=1.0m: **TBR < 1 (≈ 0.9)**



- **1m device cannot achieve TBR > 1 even with design changes**
- **Requirement**: purchase ~0.4-0.55kg of T/FPY from outside sources

NBI = Neutral Beam Injector,
TBM = Test Blanket Module, MTM = Materials Test Module

HTS potentially attractive for making electrically efficient ST* (~10× lower magnet cooling power vs. copper)



R_0 = 1.4m, B_T = 3.2T, I_P = 7-8MA, $P_{fusion\text{-}DT}$ = 100MW

**Work supported by Tokamak Energy (UK) - 2014*

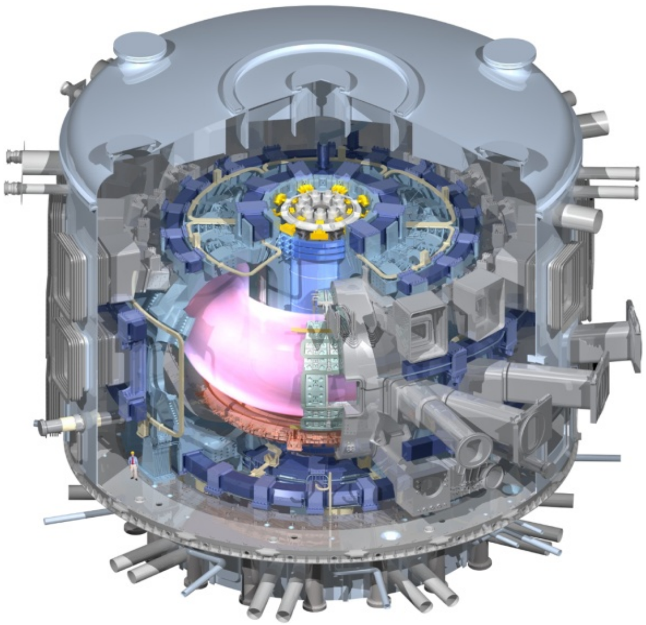
- Possible missions:
  - Steady-state toroidal PMI facility, FNSF
  - ST Pilot Plant (Q_{eng}~1 for weeks/months)
    - Requires high H_{98y2} = 1.7-2
- Initial configurations favorable:
  - A=1.8-2, strong shaping: κ~2.5-2.7, δ~0.5
  - All equilibrium PF coils outside TF
    - No joints needed for HTS TF coils
  - Long-legged divertor for $q_{div\text{-}pk}$ < 5MW/m^2
  - Vertical port-based maintenance
  - WC inboard thermal shield for TF
- Now actively investigating:
  - HTS lifetime in radiation environment
  - Blanket/shield thickness, location, TBR

Unique ST properties also support ITER

ST Extends Predictive Capability for ITER and Toroidal Science

- High β physics, rotation, shaping extend stability, transport knowledge
- NBI fast-ions in present STs mimic DT fusion product parameters in ITER → study burning plasma science
- STs can more easily study electron scale turbulence at low collisionality → important for all magnetic fusion

Burning Plasma Physics - ITER



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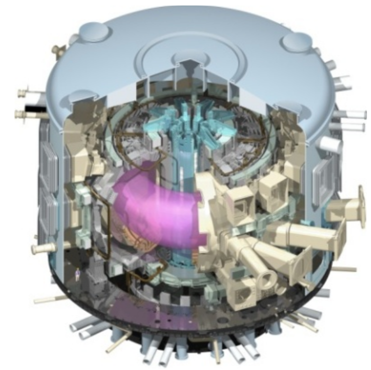
NSTX Upgrade mission elements

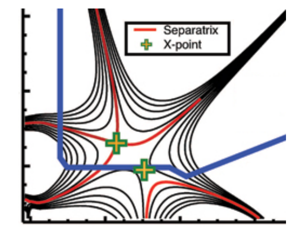
- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond



ITER

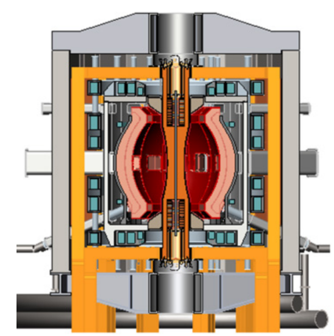
- Develop solutions for the plasma-material interface (PMI) challenge

Lithium

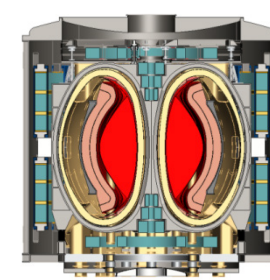


"Snowflake"

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



ST-FNSF / Pilot-Plant

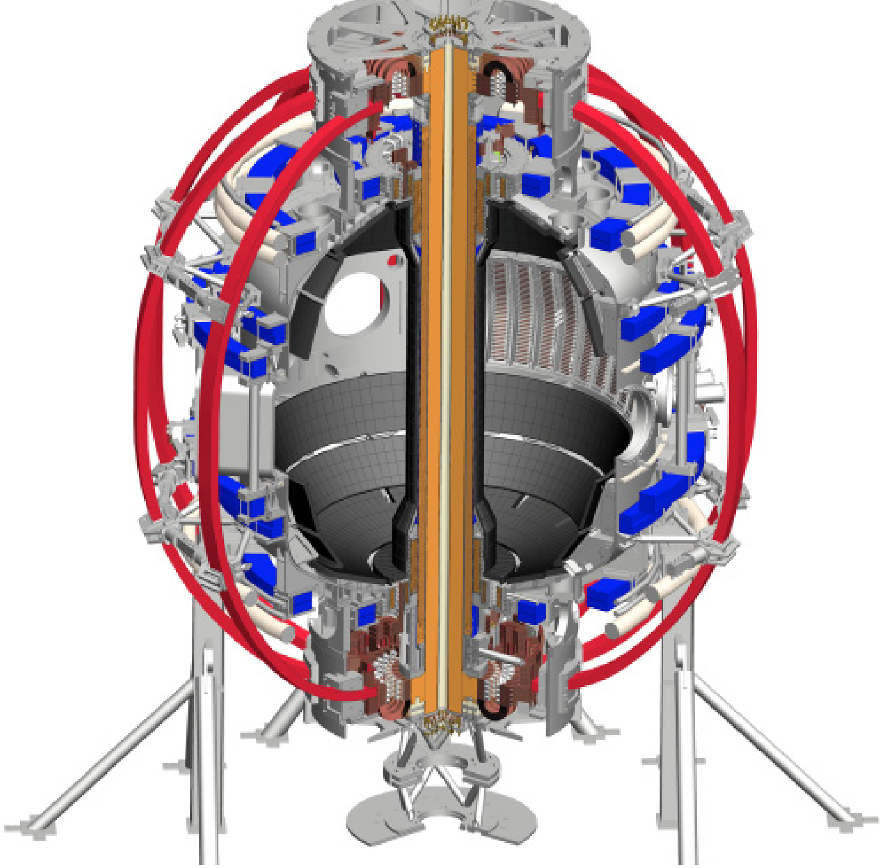


- Develop ST as fusion energy system

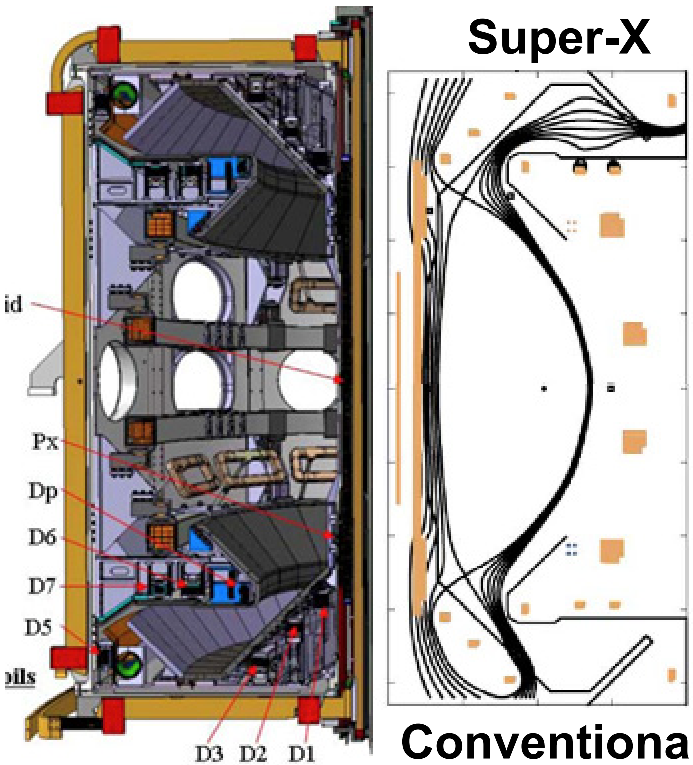
NSTX and MAST are undergoing major upgrades

~2x higher B_T, I_p, P_{NBI} and ~5x pulse length vs. NSTX / MAST

NSTX-U



MAST-U



Tangential 2nd NBI + high β_N and f_{BS} for full non-inductive sustainment, ramp-up

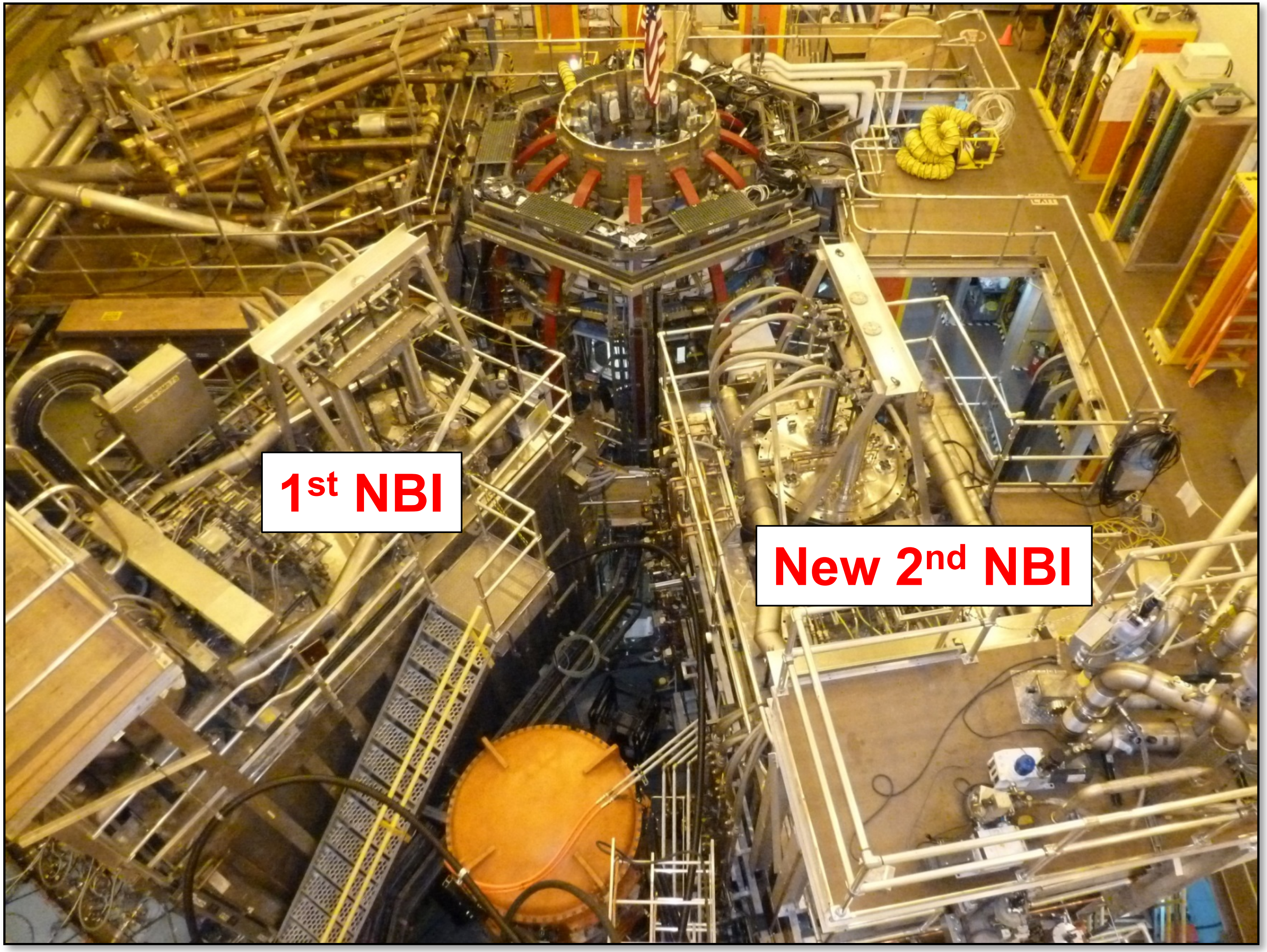
First test plasma ~1 month ago

Super-X divertor configuration for FNSF/DEMO divertor solution

First test plasma 2017

NSTX Upgrade project recently completed

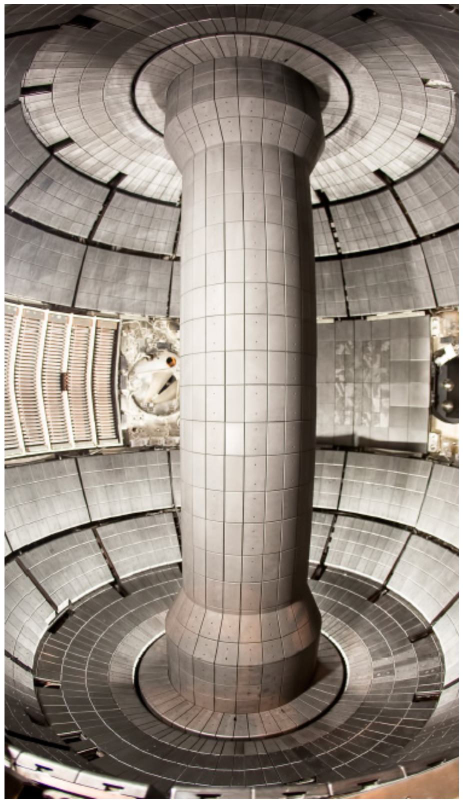
On cost and schedule, first test plasma ~100kA (Aug. 10, 2015)



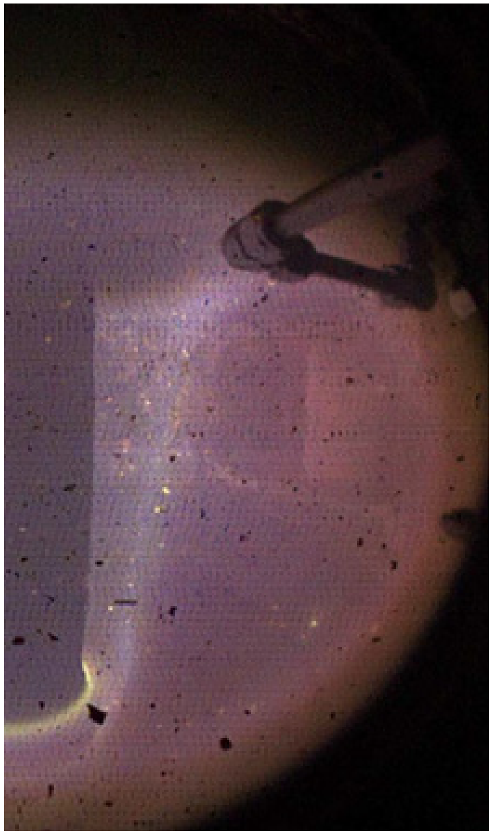
New centerstack (CS) highlights: Jan – Aug 2015

CS crane lift

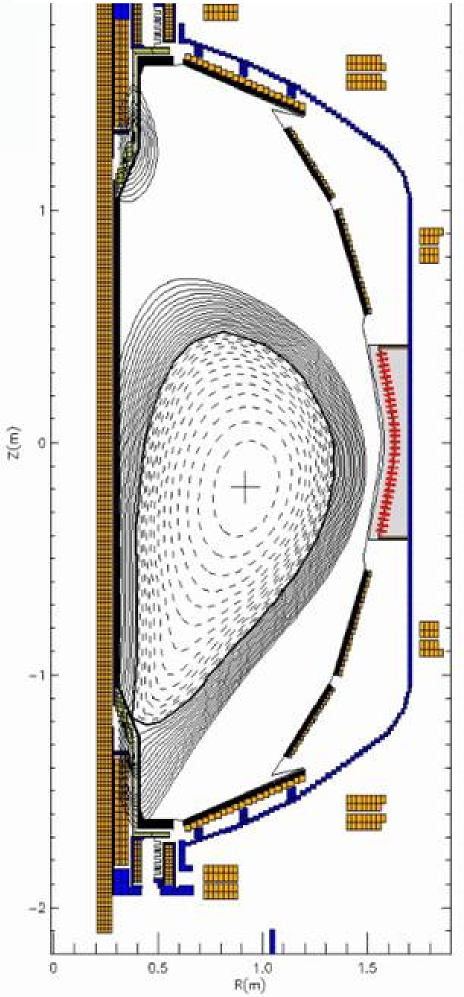
CS installed



First test plasma
(Ohmic heating only)



Magnetics functional → EFIT reconstructions



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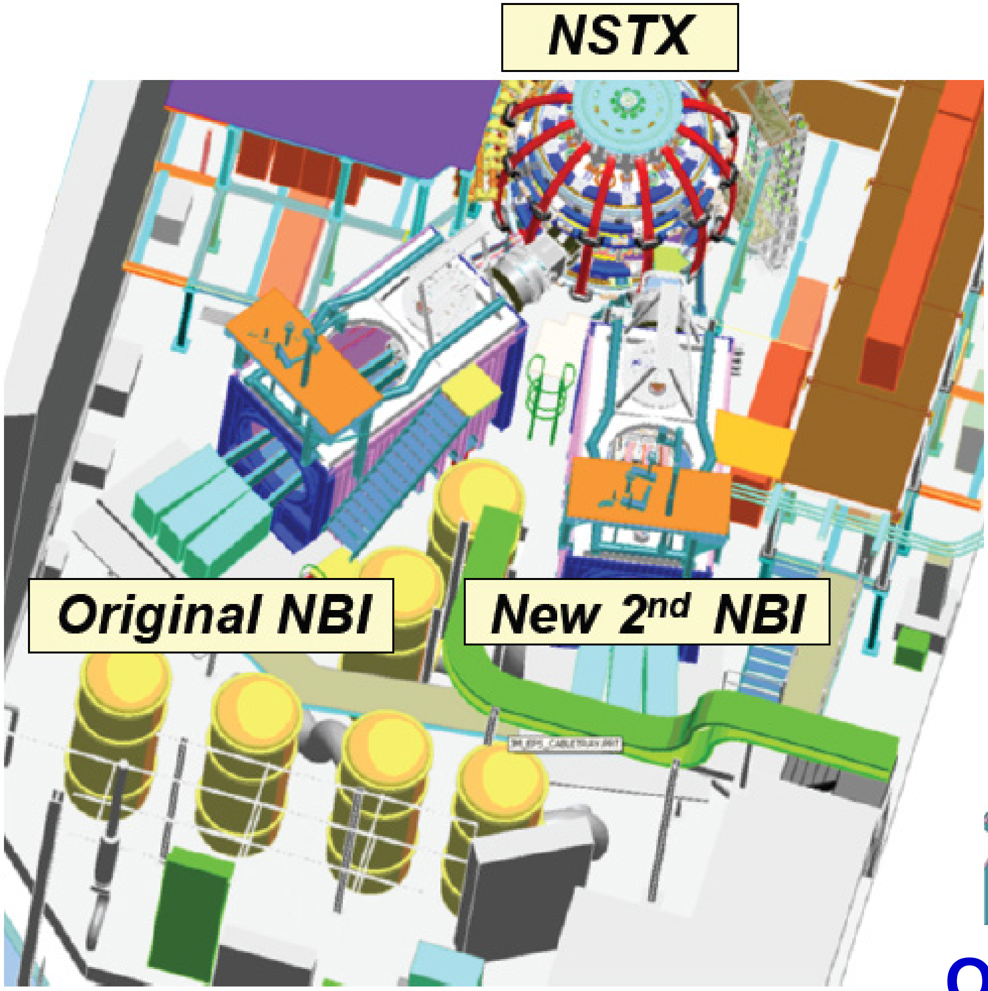
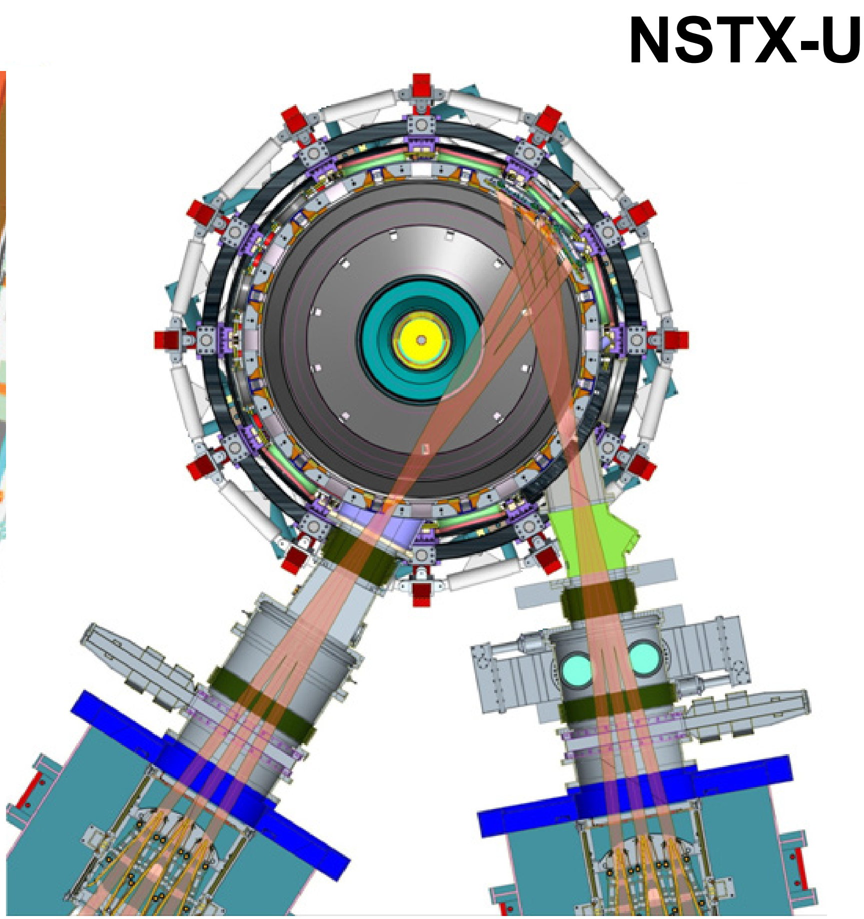
Mega-ampere-class STs rely heavily on co-injected neutral beams for heating, current drive





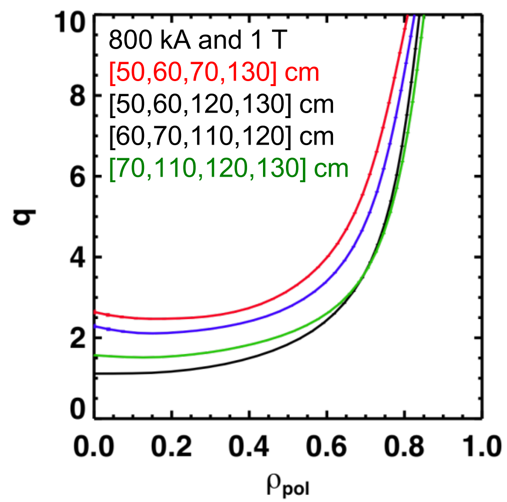
Original NBI
(R_{TAN} = 50, 60, 70cm)
5MW, 5s, 80keV

New 2nd NBI
(R_{TAN}=110, 120, 130cm)
5MW, 5s, 80keV

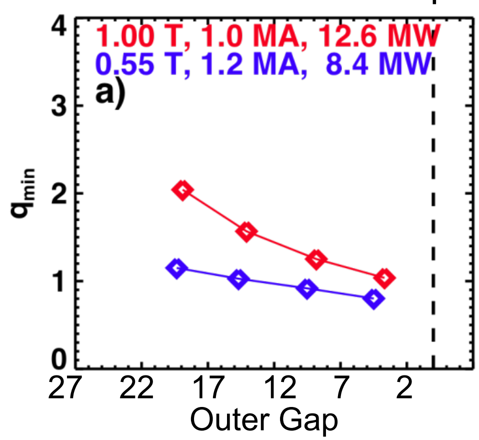
NSTX-U developing a range of profile control actuators for physics studies, scenario optimization for FNSF/Pilot

q-Profile Actuators

Variations in Beam Sources
800 kA Partial Inductive, 87% < f_{NI} < 100%

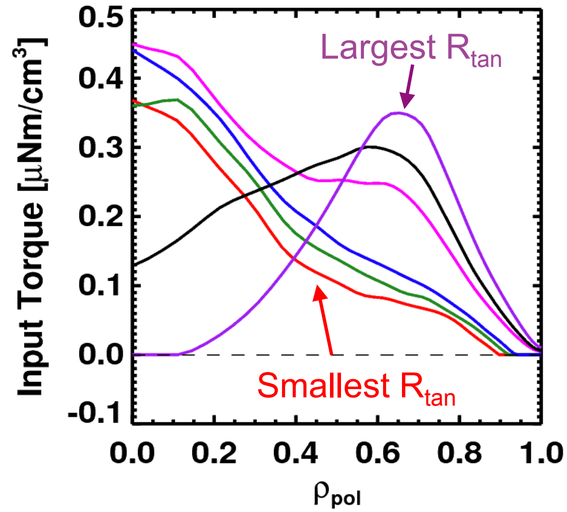


Variations in Outer Gap

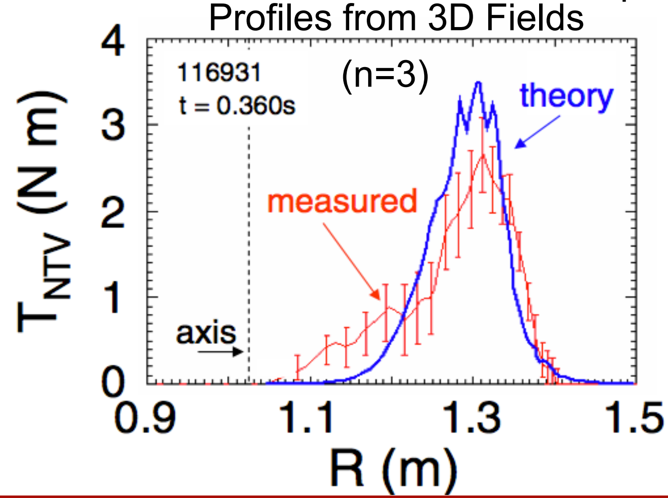


Rotation Profile Actuators

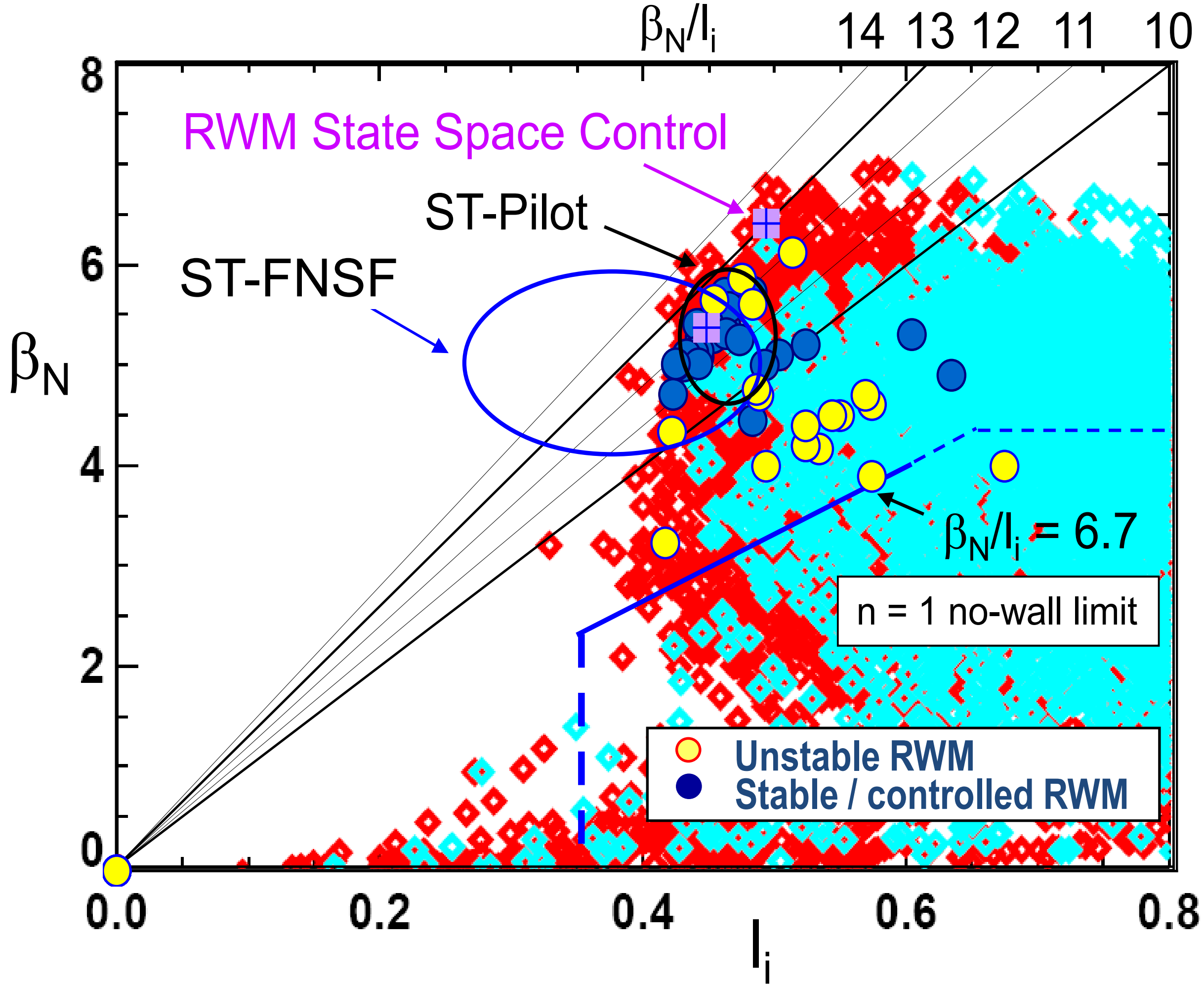
Torque Profiles From 6 Different NB Sources



Measured and Calculated Torque Profiles from 3D Fields



Record β_N and β_N / l_i accessed in NSTX using passive + active resistive wall mode stabilization



- High β_N regime is important for bootstrap current generation.
- High β_N / l_i regime important since high f_{BS} regime has low l_i.

S. Sabbagh, PRL 2006
J. Berkery, PRL 2011
W. Zhu, PRL 2006

Major NSTX-U mission is to achieve fully non-inductive operation at high β

Kelvin-Helmholtz (KH) instabilities predicted when central sound-speed Mach number $M_s \approx 0.7\text{-}0.8$

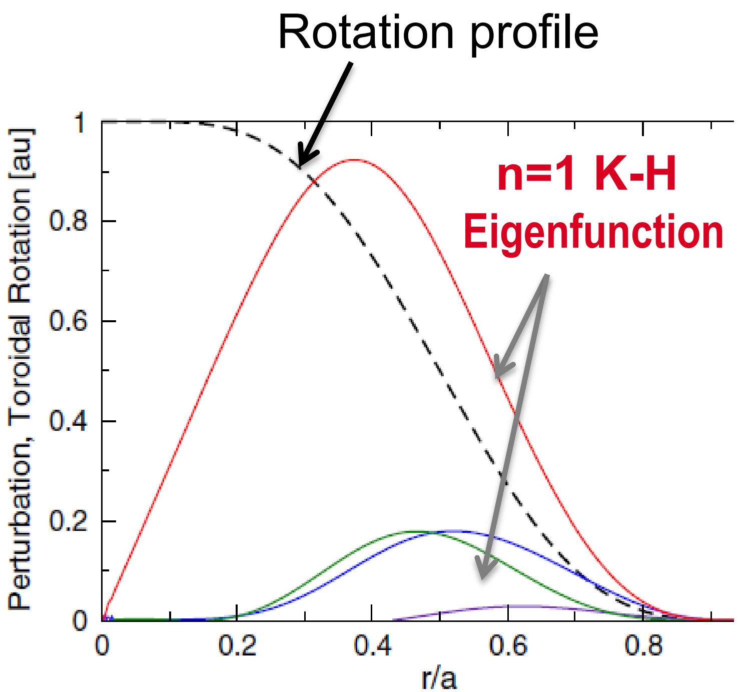


Figure 4. The $n = 1$ $KH_{\parallel}$ eigenfunction when the flow profile (dashed line) is centred at $r/a = 0.5$.

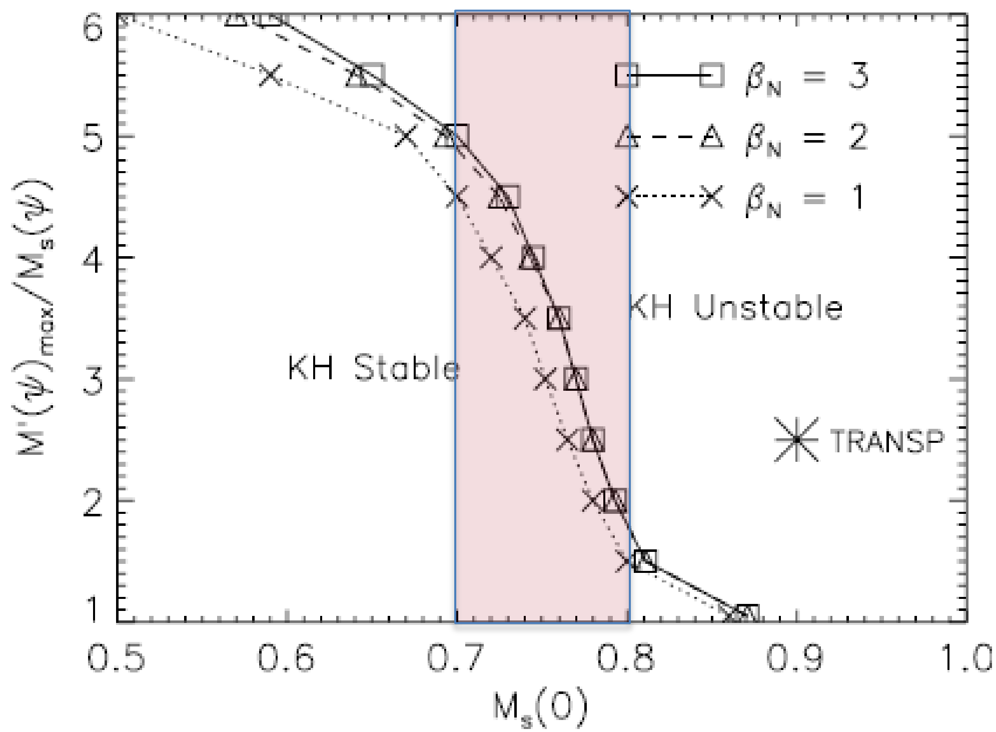
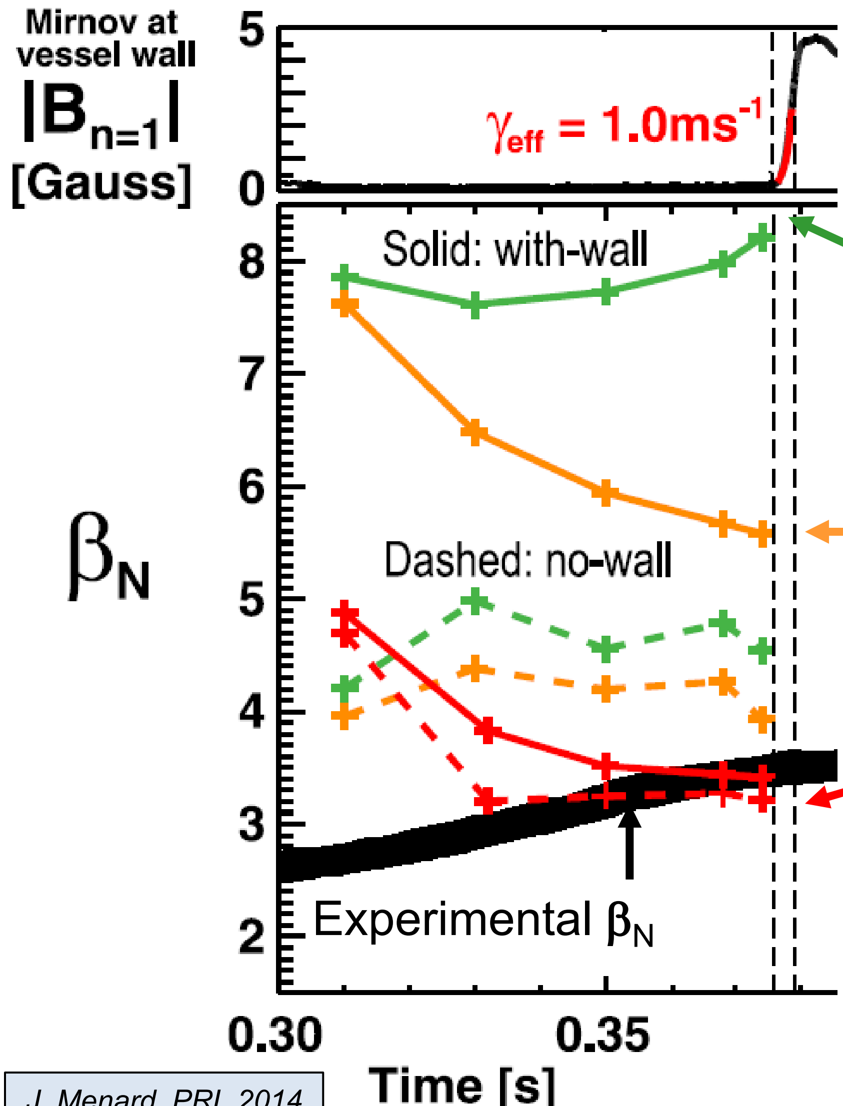


Figure 5. The $KH_{\parallel}$ stability boundary in terms of flow speed and gradient for three different plasma pressures when the safety factor is fixed at $q_{min} \simeq 1.3$. A typical rotation profile from TRANSP predictions is shown for reference, indicating that CTF with fully uni-directional beams is likely to be $KH_{\parallel}$ unstable.

I. Chapman, NF 2012

Hybrid MHD-drift-kinetic stability calculations find rotation + fast-ions reduce stability of ∇p-driven Ideal Wall Mode



J. Menard, PRL 2014

- MARS-K code needed to explain NSTX Ideal Wall Mode (IWM) onset at highest rotation, β_{fast}

Low-rotation fluid with-wall limit is very high → marginal β_N ~ 7-8

Increasing rotation lowers max β_N to ~5.5 at mode onset time

Full kinetic treatment including fast-ions → marginal β_N ~3.5 → most consistent with experiment

Rotation, rotation shear, fast-ion content must be optimized to maximize RWM and IWM limits

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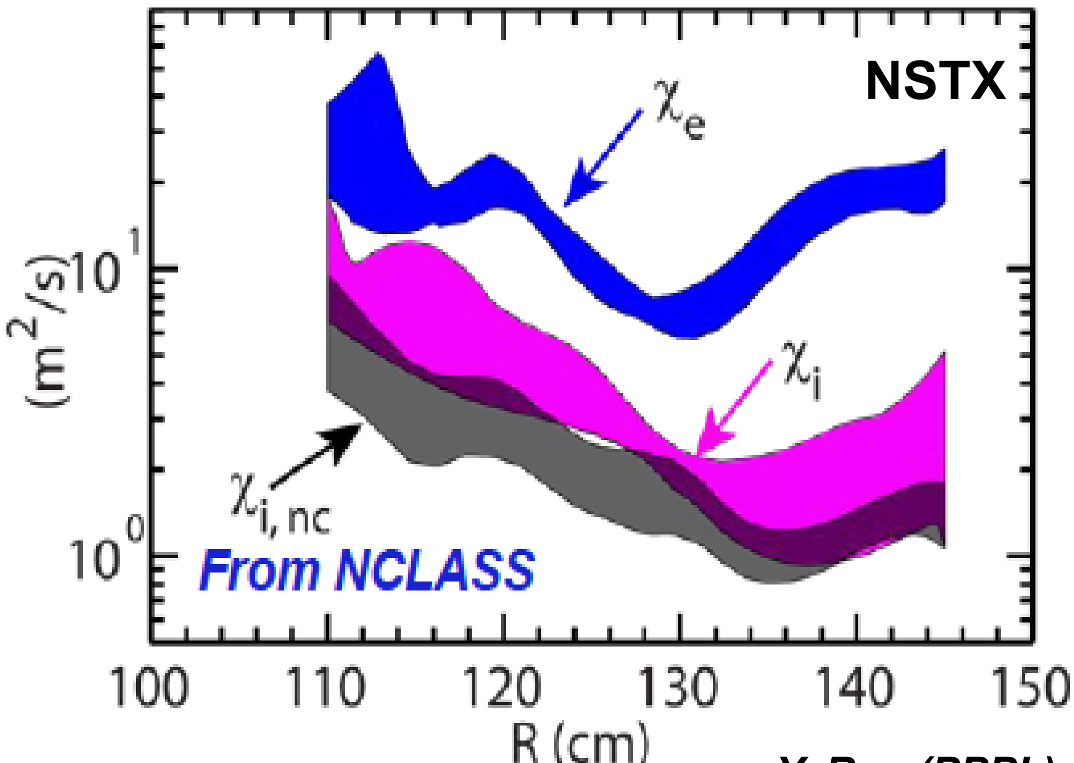
High confinement multiplier H needed for compact ST

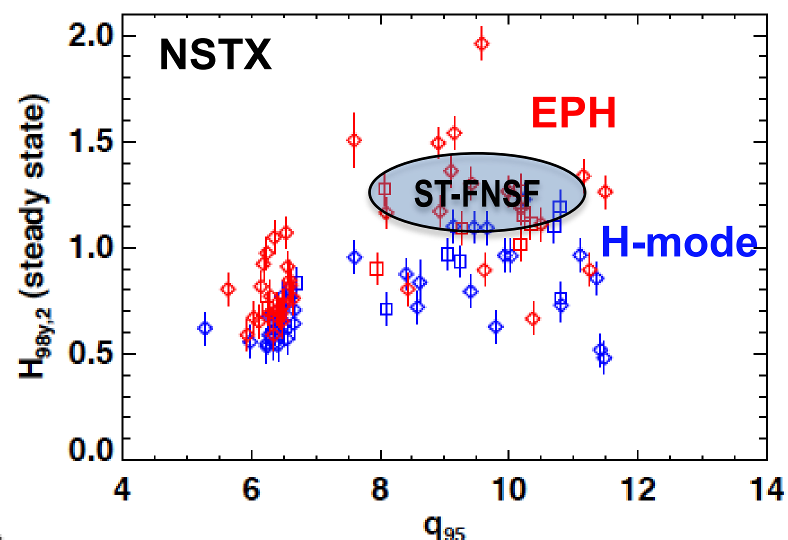
Fusion gain Q depends strongly on "H", $Q \propto H^{5\text{-}7}$

- Ion energy transport in H-mode ST plasmas near neoclassical level due to high shear flow and favorable curvature
- Electron energy transport anomalous (as for all tokamaks)



Y. Ren (PPPL)

Enhanced pedestal H-mode (EPH) has H up to 1.5-2 → attractive for ST-FNSF

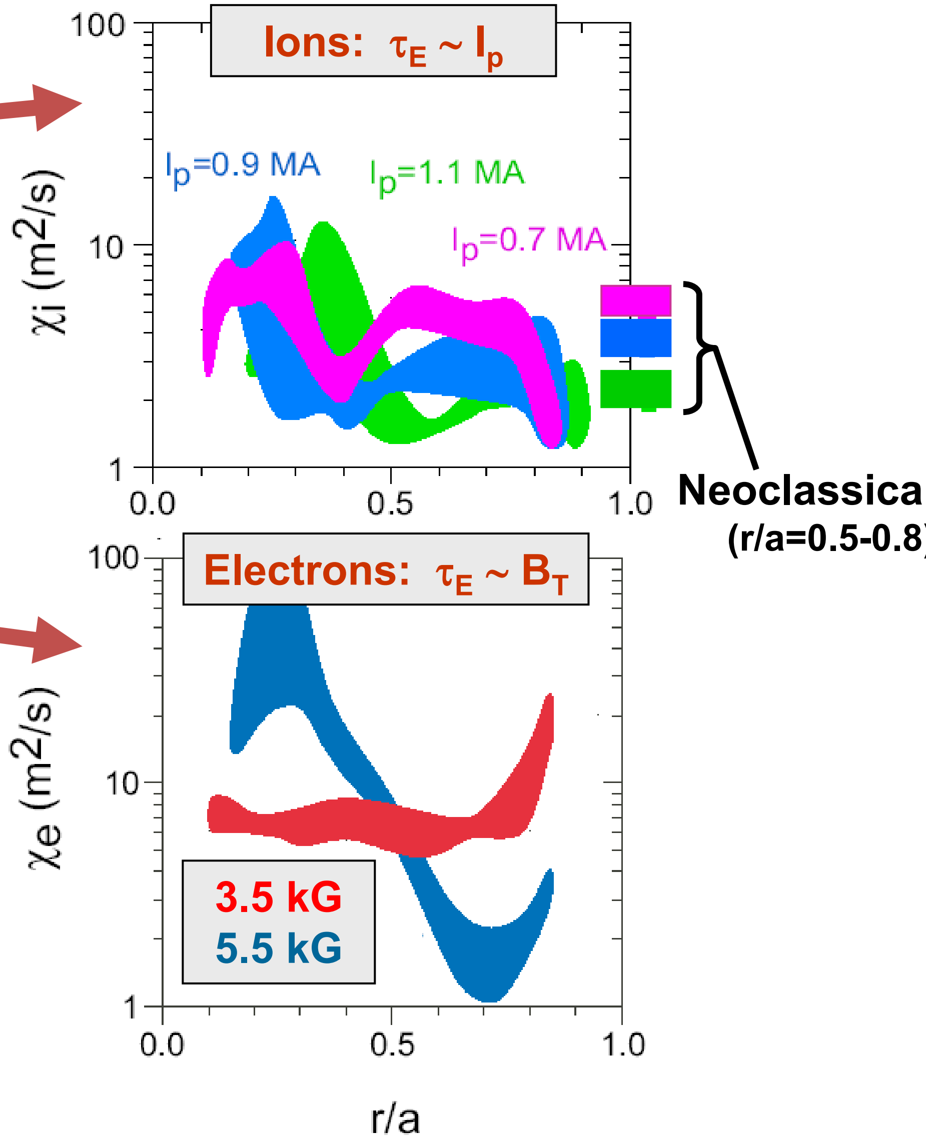


R. Maingi, PRL 2010
S. Gerhardt NF 2014

Electron and ion τ_E scale differently in ST, and different than at higher aspect ratio

- **Ion $\tau_E \sim I_P$, consistent with neoclassical ion transport**
  - Implies ion turb. suppressed by high E × B shear → possibility of isolating causes of e-transport
- **Electron $\tau_E \sim B_T$**
  - Could imply Electron Temperature Gradient (ETG) modes, and/or electromagnetic turbulence

S.M. Kaye, PRL 2007

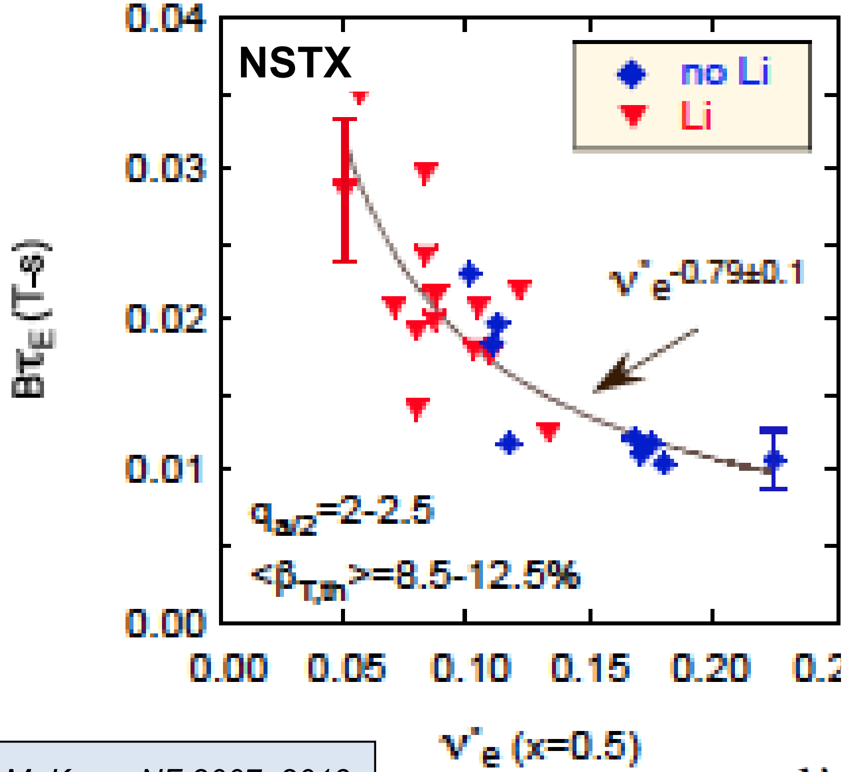


Favorable confinement trend with collisionality, β found

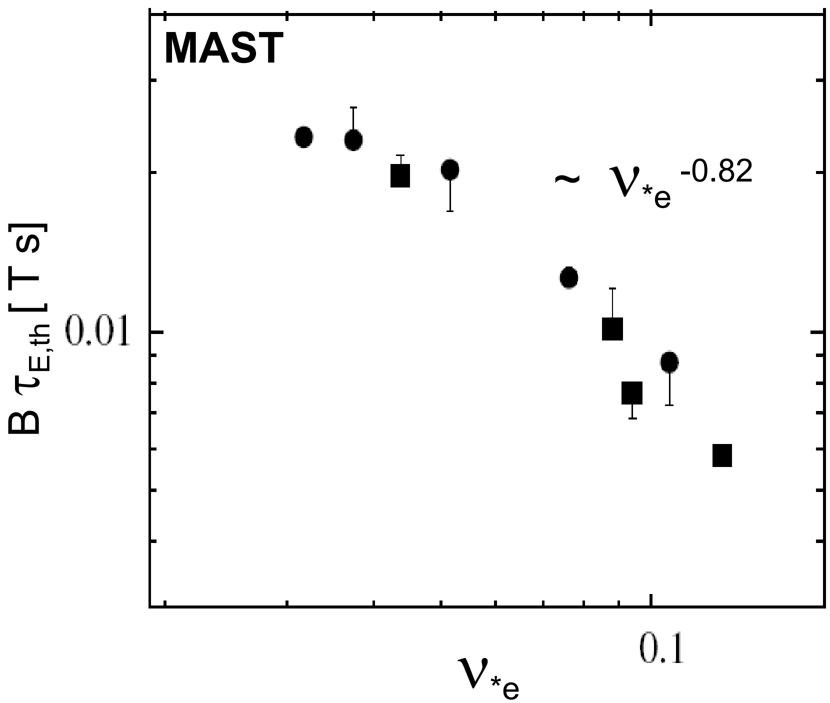
Important implications for future ST FNSF, Demo with lower ν_*

$\tau_{E,\,th} \propto \nu_{*e}^{-0.1}\ \beta^{-0.9}$ **tokamak empirical scaling (ITER 98$_{y,2}$)**

$\tau_{E,\,th} \propto \nu_{*e}^{-0.8}\ \beta^{-0.0}$ **ST scaling**



S.M. Kaye, NF 2007, 2013



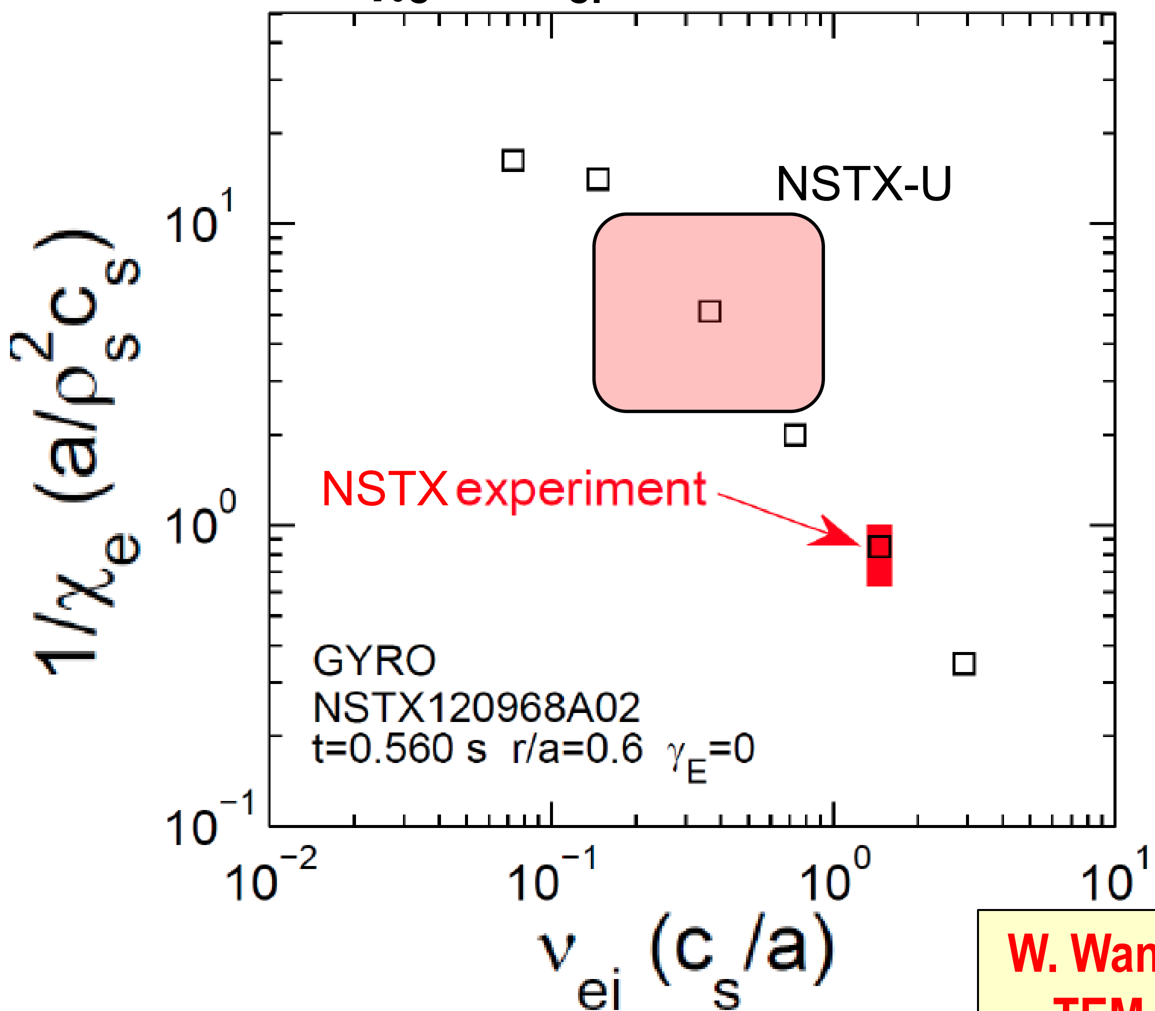
$$\nu_* \propto \bar{n}_e / T^2$$

M. Valovic, NF 2011

Very promising ST scaling to reactor condition, if continues on NSTX-U/MAST-U

Micro-tearing-driven (MT) transport may explain ST τ_E collisionality scaling

MT-driven χ_e vs. ν_{ei} using the GYRO code



- MT growth rate decreases with reduced collisionality in qualitative agreement with the NSTX experiment.
- Further electron confinement improvement expected due to reduced collisionality.

W. Guttenfelder, PoP 2013, PoP 2012, PRL 2011

W. Wang: Recently found dissipative TEM has similar ν* scaling (GTS)

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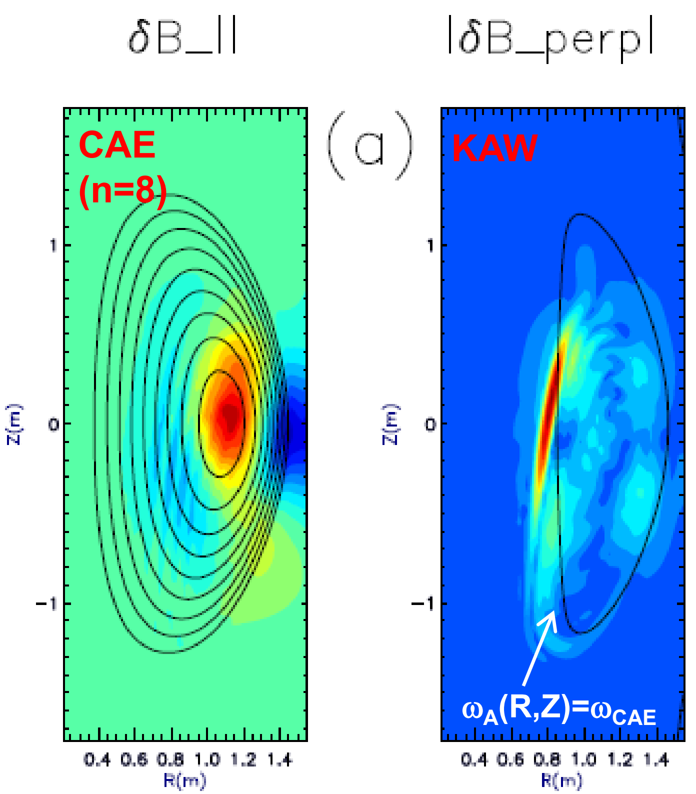
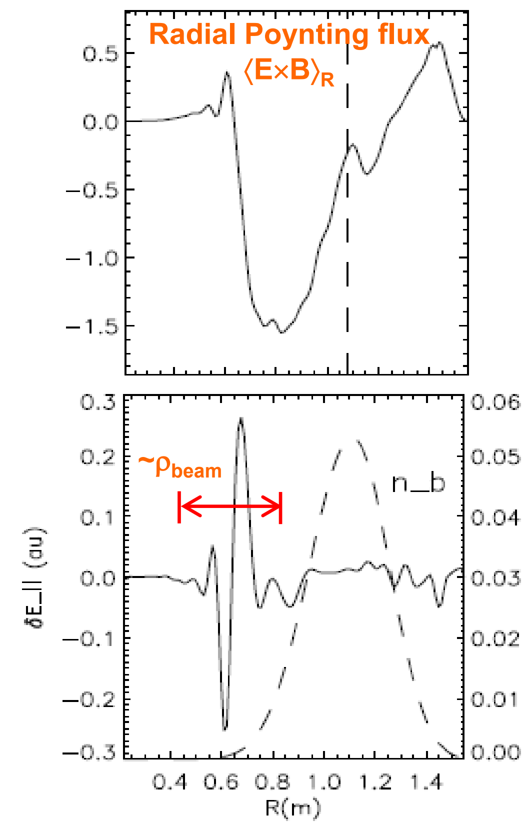
CAE mode-conversion to kinetic Alfvén waves (KAW) predicted to transfer core NBI power to mid-ρ electrons

1) GAE/CAEs cause large χ_e through stochastic orbits (N. Gorelenkov, NF 2010)
2) CAEs also couple to KAW - Poynting flux redistributes fast ion energy near mid-radius, $E_{||}$ resistively dissipates energy to thermal electrons
   – $P_{CAE \to KAW}$ ~ **0.4 MW** from QL estimate + experimental mode amplitudes
   – $P_{e,NBI}$ ~ **1.7 MW** for ρ <0.3, NBI power deposited on core electrons



Up to 25% of electron heating power transferred to KAW off-axis

HYM code
E. Belova, PRL 2015

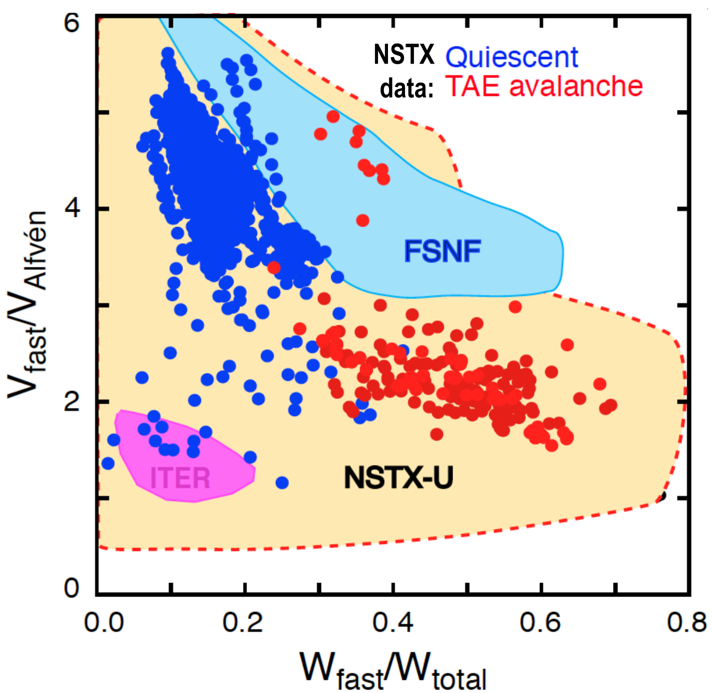


NBI-heated STs excellent testbed for α-particle physics

Alfvenic modes readily accessible due to high $V_{fast} > V_{Alfvén}$

- α-particles couple to Alfvénic modes strongly when $V_\alpha > V_A \sim \beta^{-0.5} C_s$
- $V_\alpha > V_A$ in ITER and reactors: condition easily satisfied in ST due to high β
- Fast-particle-driven Alfvén Eigenmodes: Toroidal, Global, Compressional
- NSTX-U will also explore $V_{fast} < V_A$ regime giving more flexibility

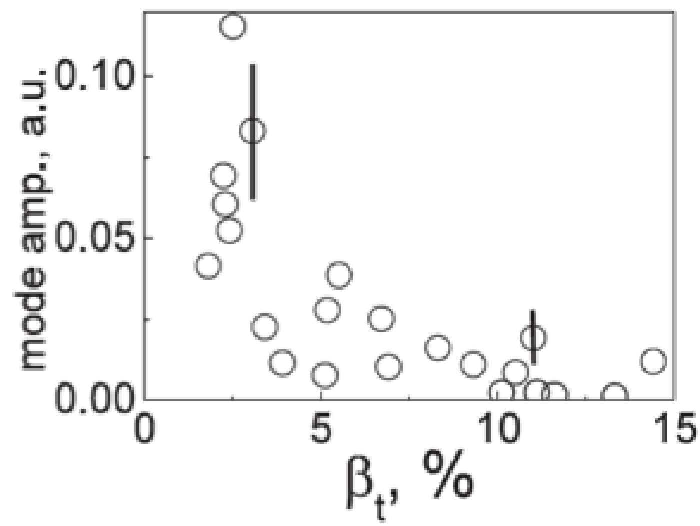
EP parameter space



E. Fredrickson, NF 2013

TAEs significantly modified at high β as $V_A \rightarrow C_s$

Stabilization of TAEs at high β in MAST

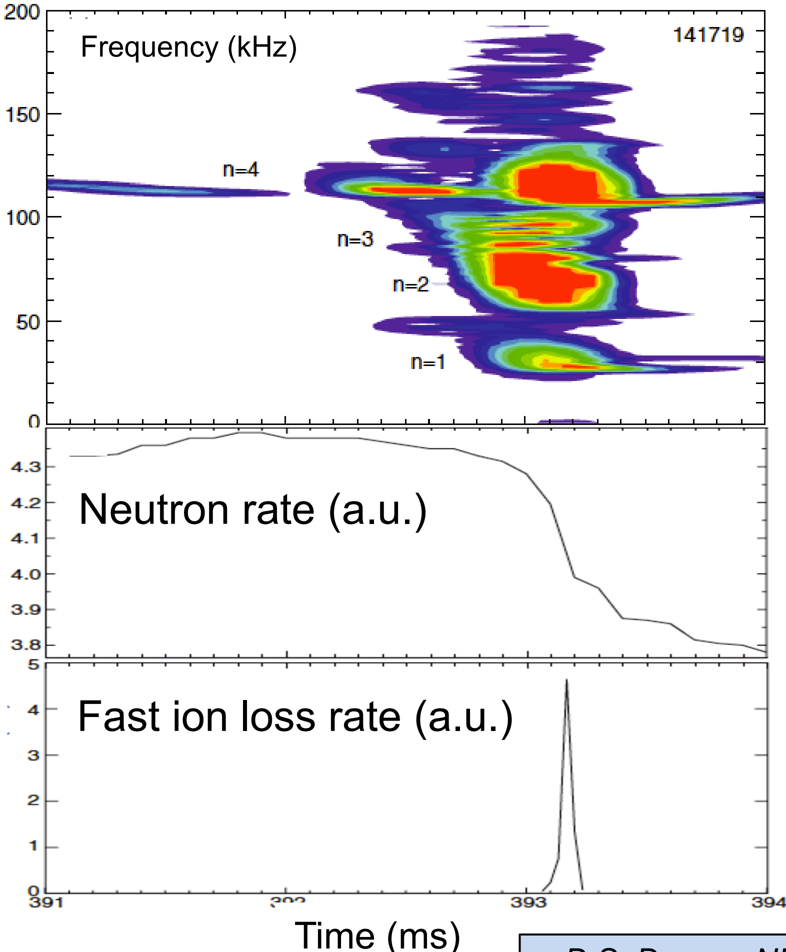


M.P. Gryaznevich, PPCF 2004

"TAE avalanche" shown to cause energetic particle loss

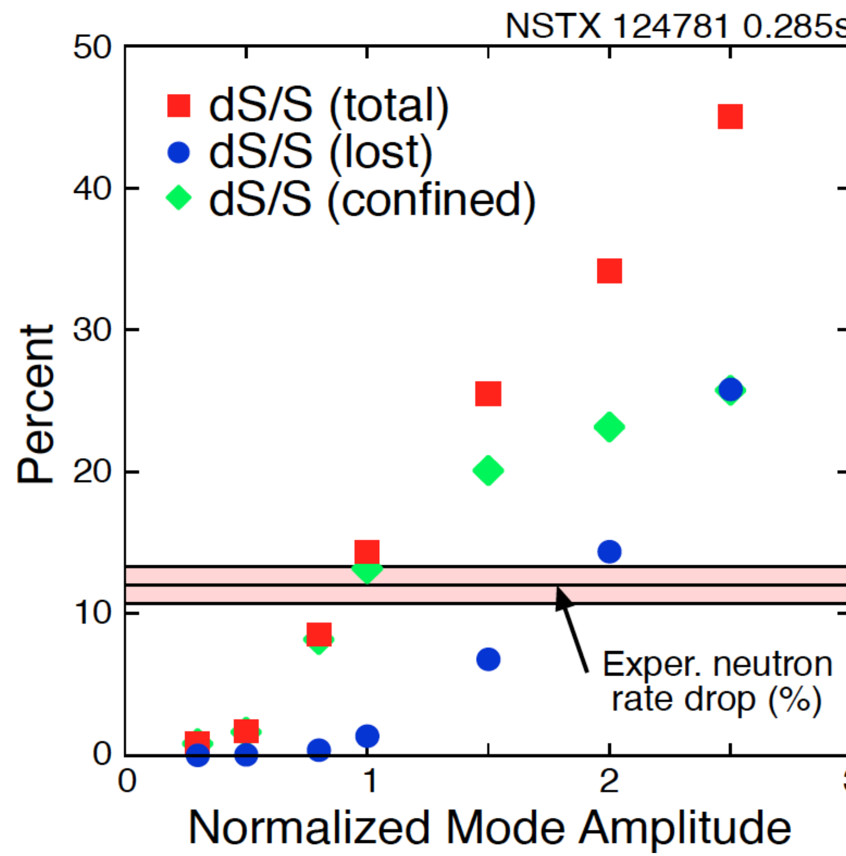
Uncontrolled α-particle loss could cause reactor first wall damage

Multi-mode TAE avalanche can cause significant EP losses as in "sea"of TAEs expected in ITER



D.S. Darrow, NF (2013)

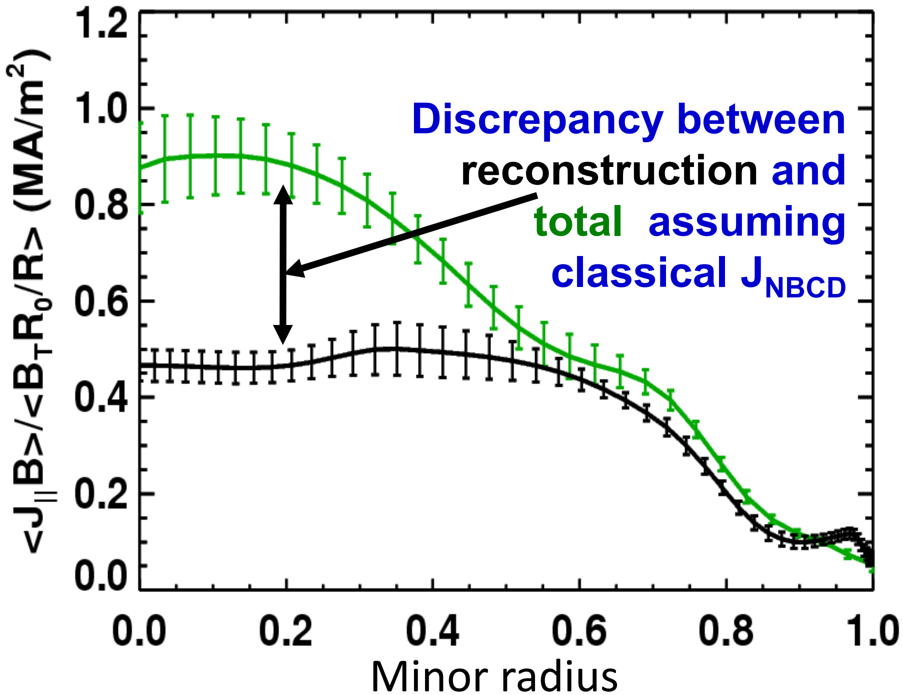
Progress in simulation of neutron rate drop due to TAE avalanche



E. Fredrickson, NF 2013

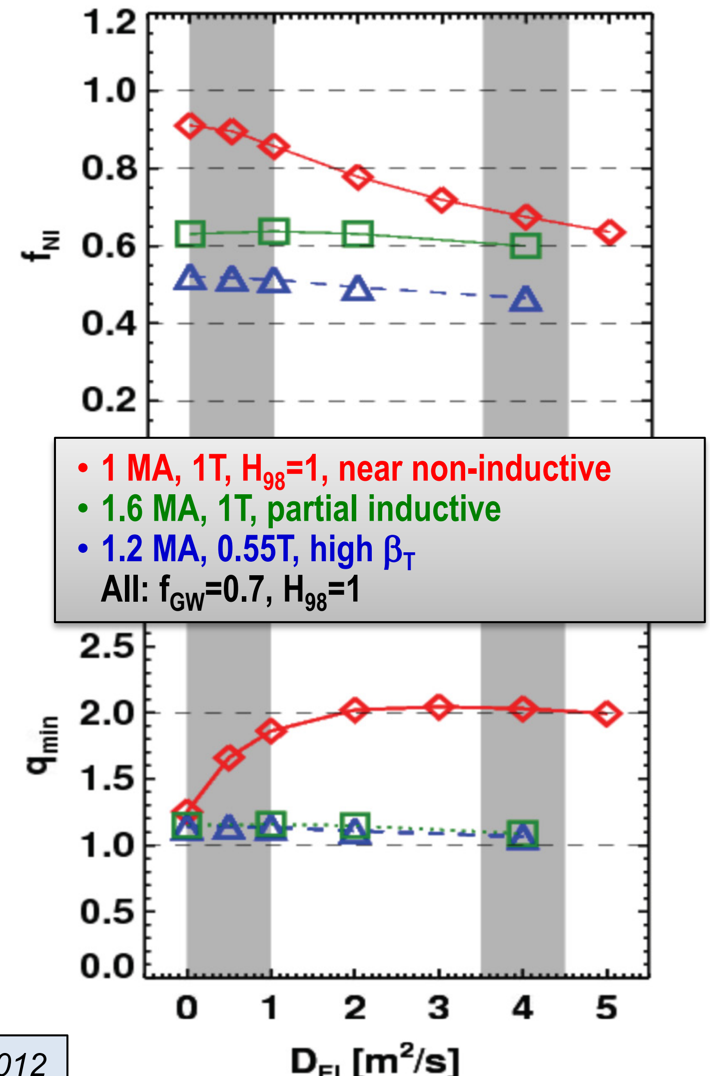
Rapid TAE avalanches could impact NBI current-drive in advanced scenarios for NSTX-U, FNSF, ITER AT

NSTX: rapid avalanches can lead to redistribution/loss of NBI current drive



700kA high-β_P plasma with rapid TAE avalanches has time-average D_{FI} = 2-4m^2/s

NSTX-U TRANSP simulations



S. Gerhardt NF 2011, NF 2012

Outline

- ST Overview and Motivation
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- **NSTX/NSTX-U Research Highlights**
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  - **Power Exhaust**
  - Plasma Start-up
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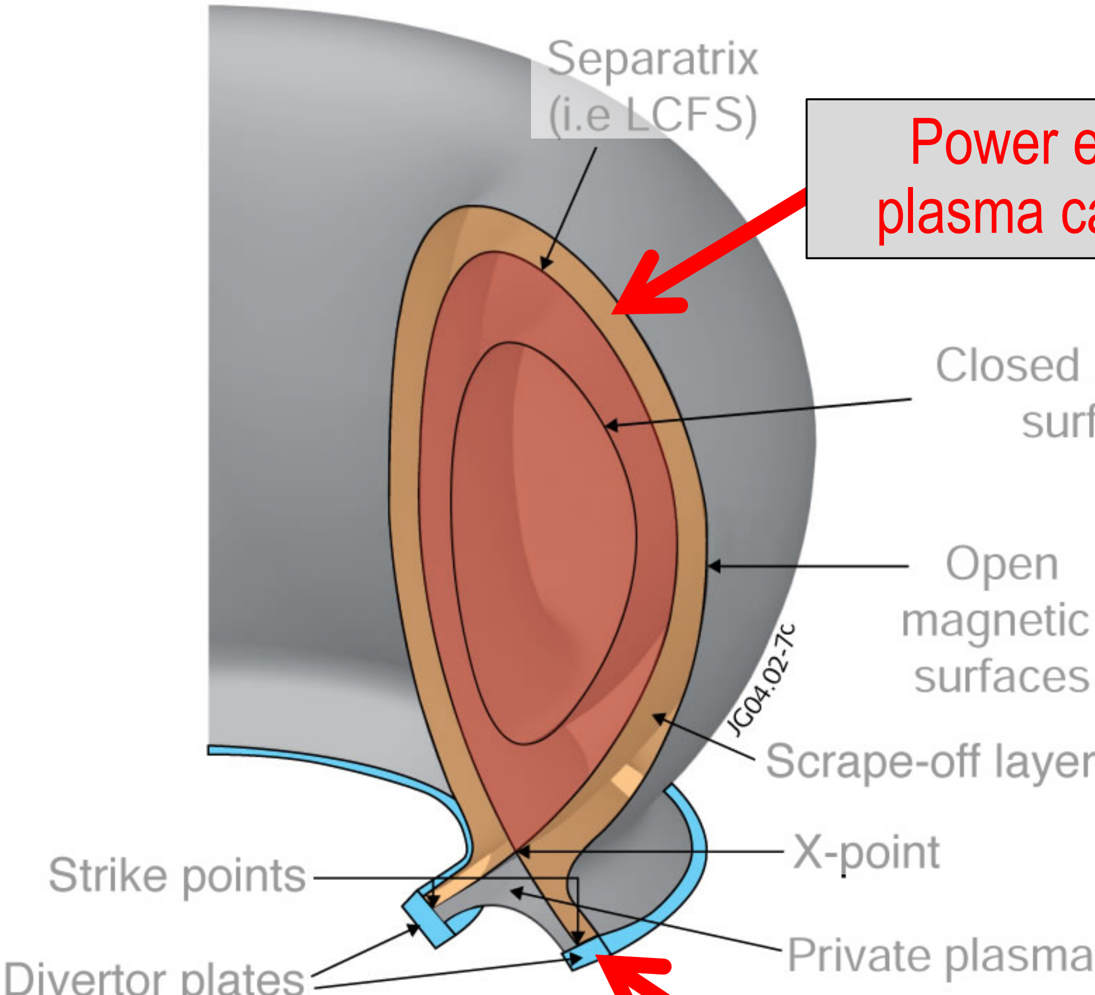
All modern tokamaks / STs use a “divertor” to control where power and particles are exhausted



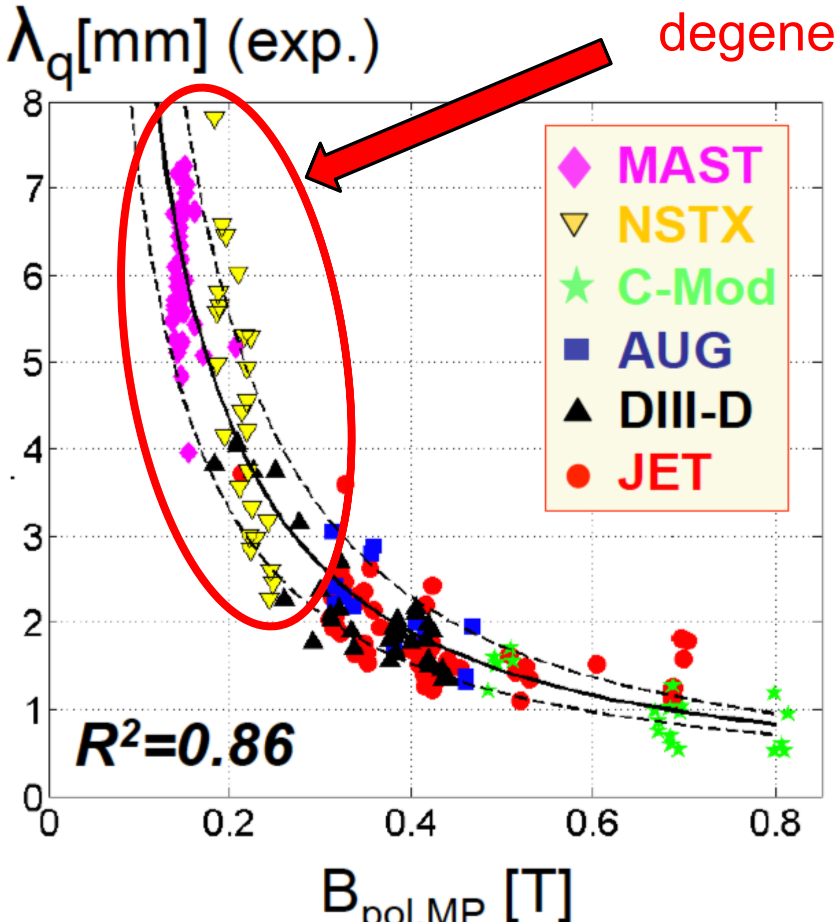
Power exhaust width outside main plasma can be very narrow (few mm)

Power / particles contact target plates here

Tokamak + ST data: power exhaust width varies as 1 / $B_{poloidal}$

Will previous ST trend continue at 2× I_P , B_P , B_T, power?

ST data breaks aspect ratio degeneracy of data set

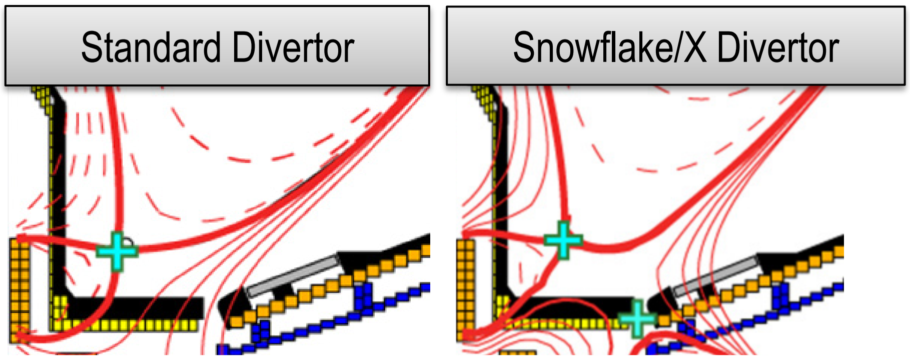


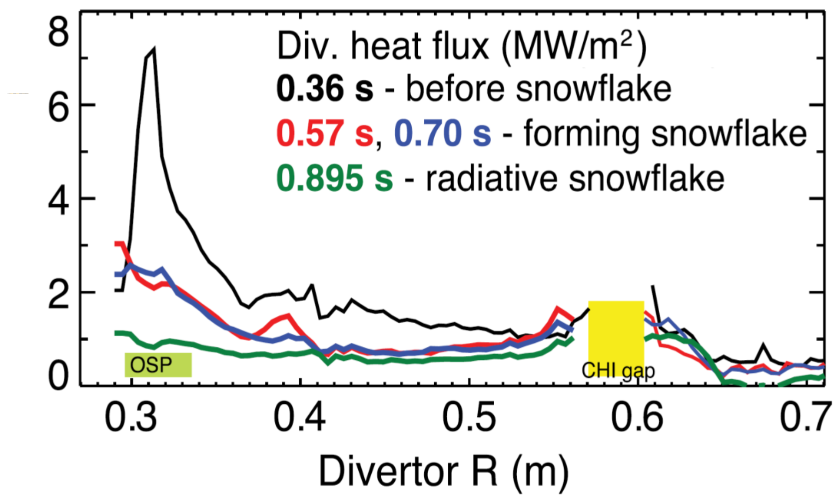
Issue: If peak heat flux in divertor region exceeds 10 MW/m^2 → material damage

Wider heat-flux width may offset smaller R → maybe better than tokamak?

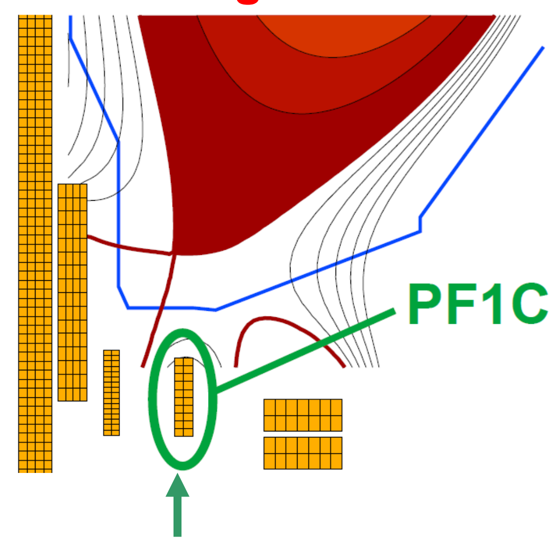
NSTX-U will test ability of radiation and advanced divertors to mitigate very high heat-fluxes

- NSTX: reduced heat flux 2-4 × via radiation (partial detachment)
- Additional null-point in divertor expands field, reduces heat flux





NSTX-U peak heat fluxes will be up to 4-8 × higher than in NSTX



NSTX-U has additional coils for up-down symmetric snowflake/X, improved control

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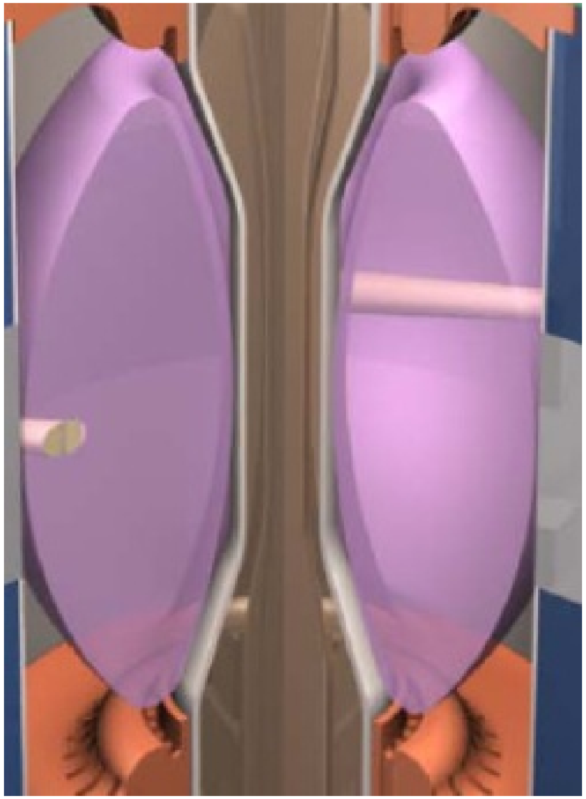
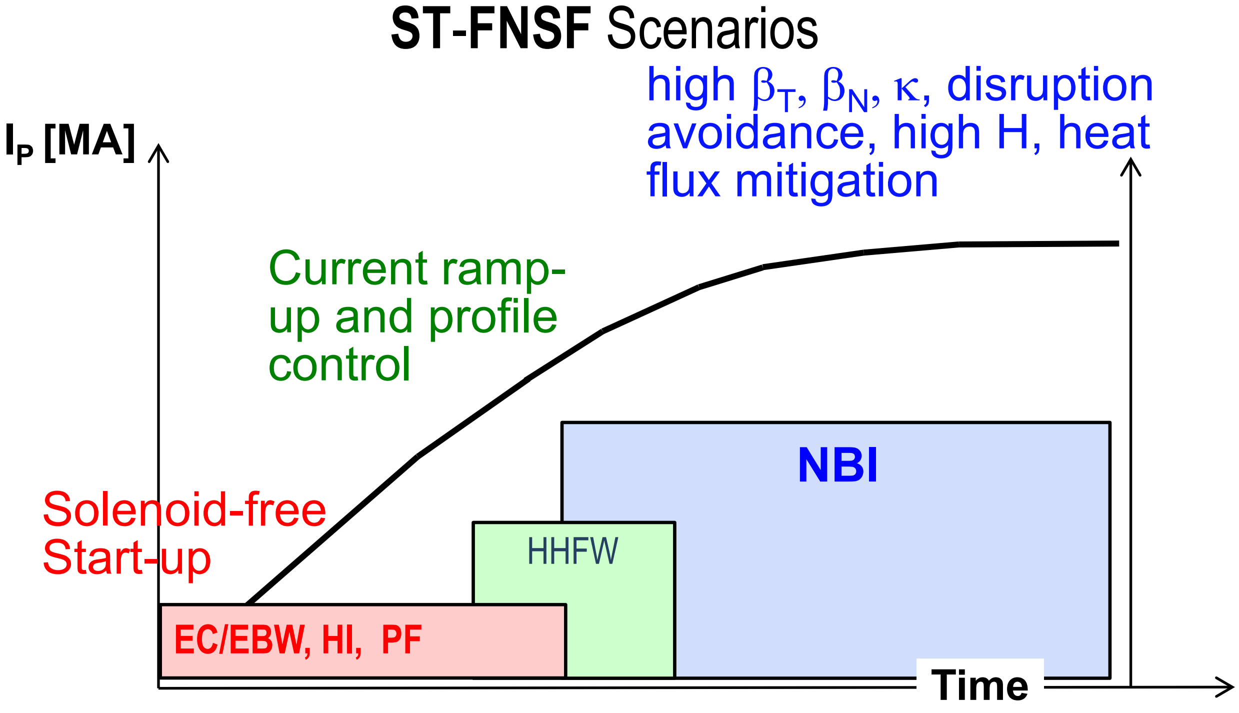
I_P Start-up/Ramp-up Critical Issue for ST-FNSF/Demo

Compact ST-FNSF has no/small central solenoid



~ 1-2 MA of solenoid-free start-up current needed for FNSF

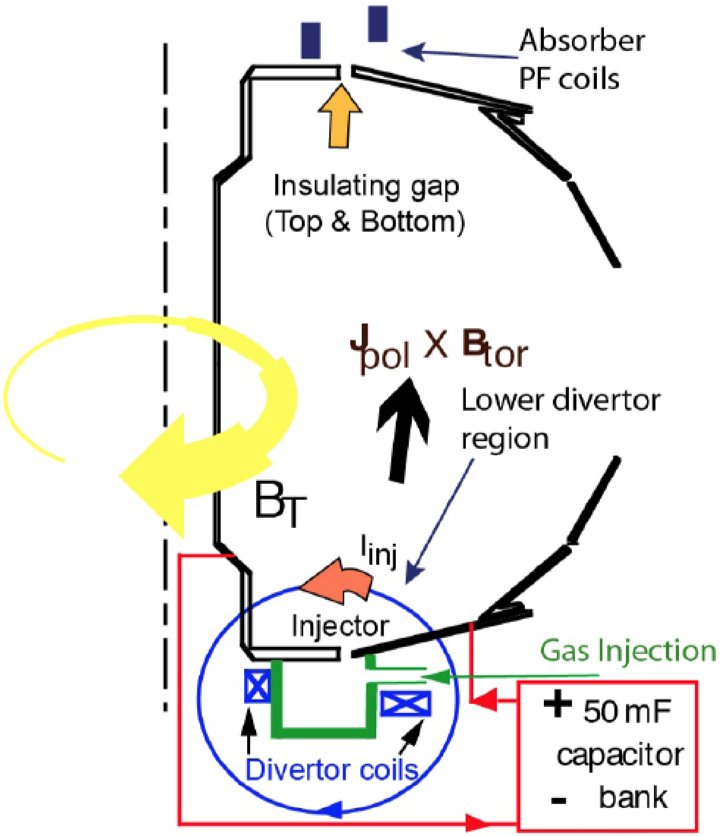


- Two novel techniques for solenoid-free start-up and ramp-up will be investigated
  - RF: ECH/EBW and HHFW
  - Helicity Injection

Helicity Injection is efficient method for current initiation

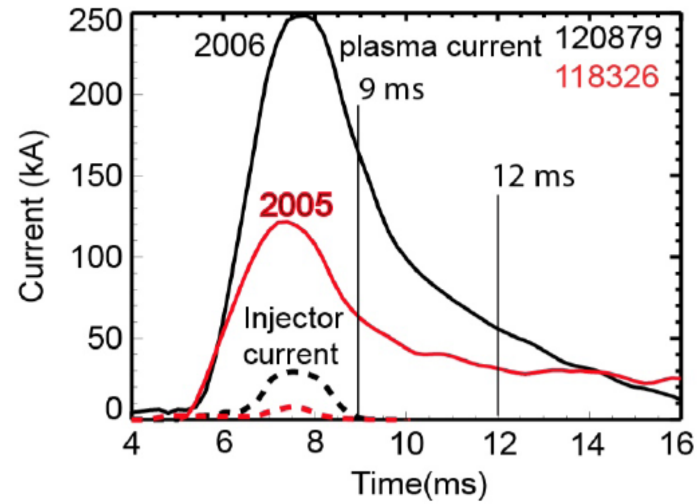
Coaxial Helicity Injection (CHI) concepts being developed

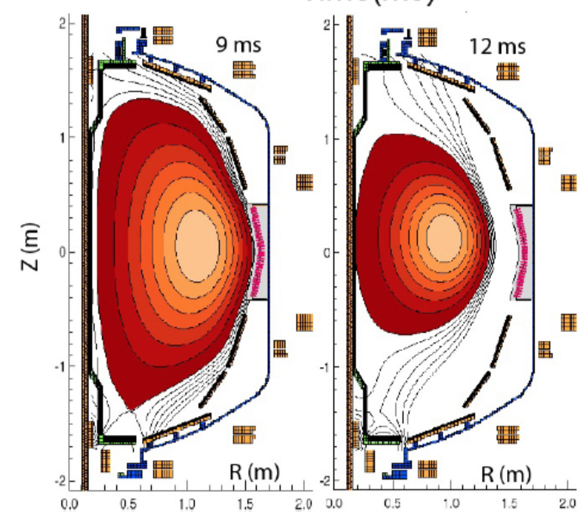
CHI developed on HIT, HIT-II
Transferred to NSTX / NSTX-U



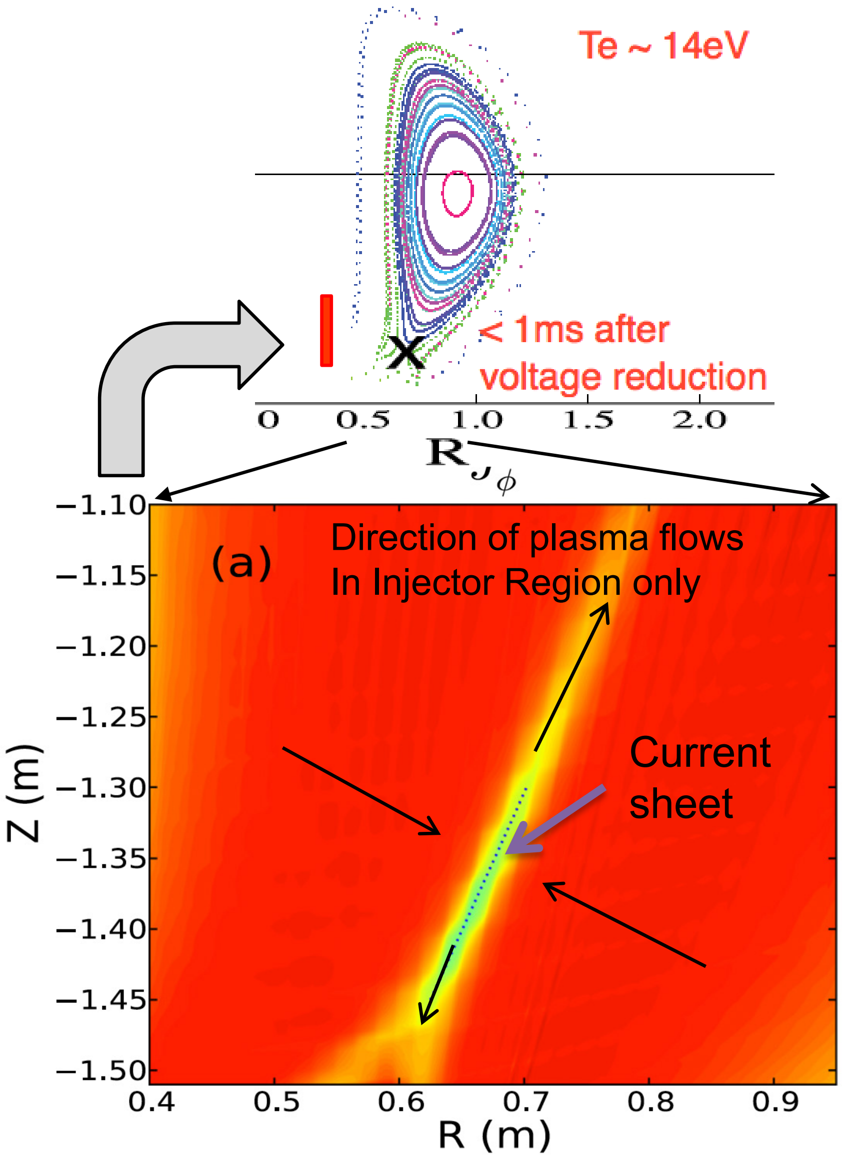
R. Raman et al., PRL 2006

Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX





NIMROD simulations → CHI in NSTX has resemblance to 2D Sweet-Parker reconnection



- Toroidal electric field generated in injector region by reduction of injector voltage and current
  - $E_{toroidal} \times B_{poloidal}$ drift brings oppositely directed field lines closer and causes reconnection, generating closed flux

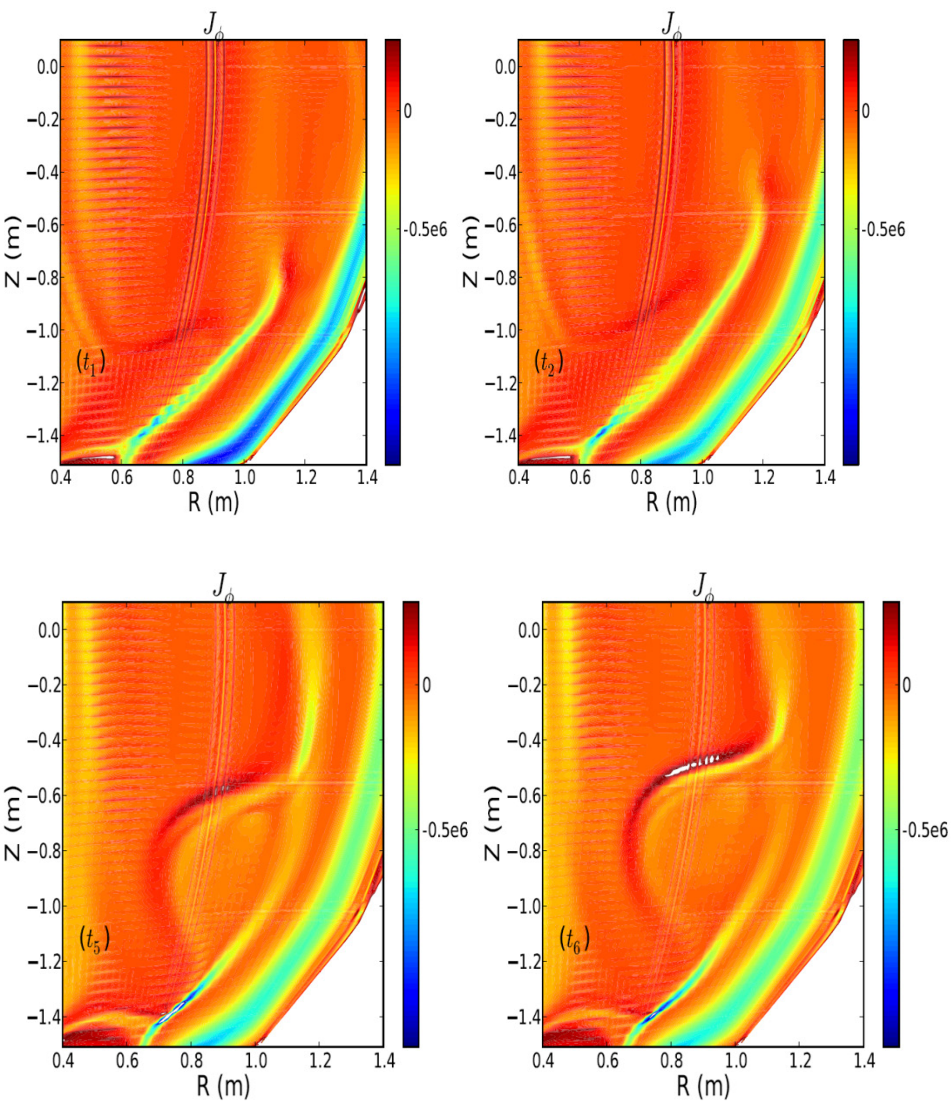
- Elongated Sweet-Parker-type current sheet
- n > 0 modes / MHD not strongly impacting 2D reconnection

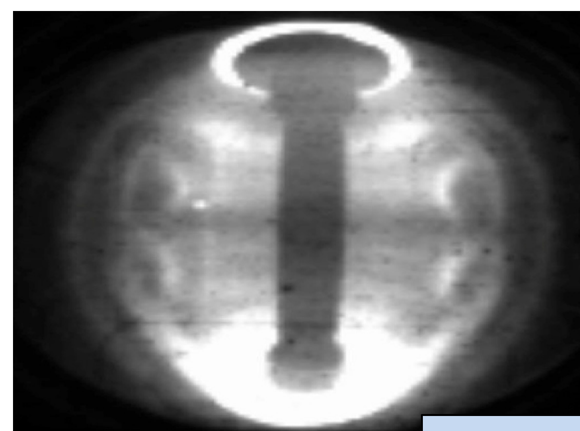
F. Ebrahimi, PoP 2013, PoP 2014

CHI current sheet unstable → plasmoids → merging

Possible lab observation of plasmoids → contribute to lab-astro

Current sheet shown in the lower half of the device.





F. Ebrahimi , PRL 2015

Possible plasmoids during NSTX CHI

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Brief Overview of FY2016-18 NSTX-U Goals

- **FY2016**
  - Obtain first data at 60% higher field/current, 2-3× longer pulse:
    - Re-establish sustained low l_i / high-κ operation above no-wall limit
    - Study thermal confinement, pedestal structure, SOL widths
    - Assess current-drive, fast-ion instabilities from new 2nd NBI
- **FY2017**
  - Extend NSTX-U performance to full field, current (1T, 2MA)
    - Assess divertor heat flux mitigation, confinement at full parameters
  - Access full non-inductive, test small current over-drive
  - First data with 2D high-k scattering, prototype high-Z tiles
- **FY2018**
  - Assess causes of core electron thermal transport
  - Test advanced q profile and rotation profile control
  - Assess CHI plasma current start-up performance
  - Study low-Z and high-Z impurity transport
    - Possibly test/compare pre-filled liquid-Li tiles/PFCs vs. high-Z solid

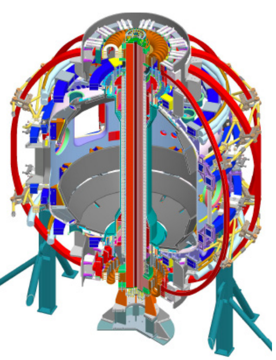
Five Year Facility Enhancement Plan (green = ongoing)

2015: Engineering design for high-Z tiles, Cryo-Pump, NCC, ECH

Fiscal Year:	2015	2016	2017	2018	2019
	Upgrade Outage	1.5 → 2 MA, 1s → 5s			
Run Weeks:		14-16	16-18	12-16	10-12

Major enhancements:
- ● Base funding
- ○ +15% incremental

New center-stack



Boundary Science + Particle Control

Pulse-burst MPTS

Boronization

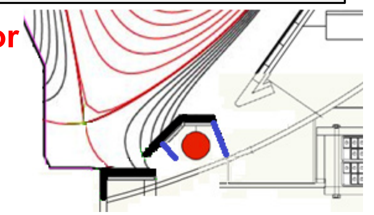
High-Z tile row on lower OBD

Li granule injector

Upward LiTER

MAPP

Lower divertor cryo-pump



High-Z PFC diagnostics

LLD using bakeable cryo-baffle

Core Science

MGI disruption mitigation

Upgraded halo sensors

Laser blow-off

42 ch MPTS

48 ch BES

MSE/LIF

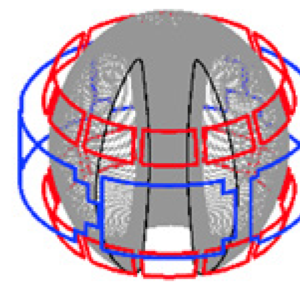
High k_θ

Charged fusion product

4 coil AE antenna

δB diag. - incremental

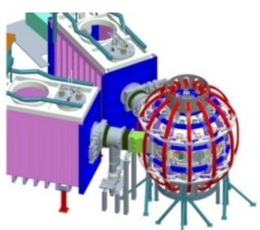
Off-midplane 3D coils (NCC)



Enhanced MHD sensors

DBS, PCI, or other intermediate-k

Neutron collimator



2nd NBI

Integrated Scenarios

Establish control of:

Snowflake

FIReTIP

Rotation

$\bar{n}_e$

q_{min}

Divertor P_{rad}

HHFW limiter upgrade

1 MW ECH/EBW

Upgraded CHI for ~0.5MA

0.5-1 MA CHI

up to 1 MA plasma gun

Summary

- NSTX-U will provide many opportunities to study toroidal confinement physics in novel regimes:
  - Low aspect ratio, strong shaping, high β, low collisionality
  - Access strong fast-ion instability drive, high rotation
  - Advanced divertors, high-Z PFCs, lithium walls
- NSTX-U/ST results will inform optimal configuration, aspect ratio for next-step PMI, FNSF, Pilot devices
- Physics research to begin in Nov/Dec 2015
  - Many UW researchers already collaborating on NSTX-U!

Thank you!

Backup

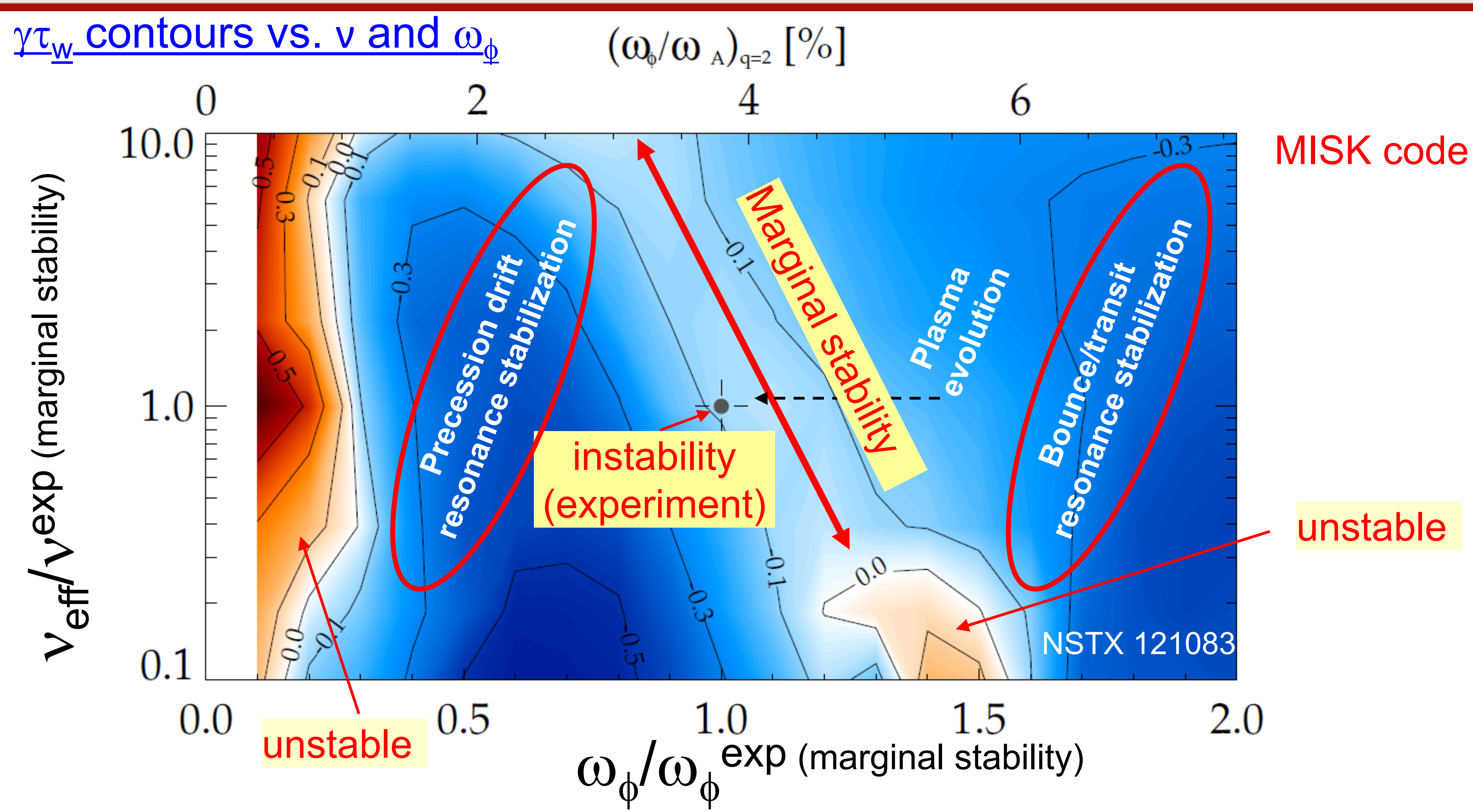
NSTX-U Milestone Schedule for FY2016-18

	FY2016	FY2017	FY2018
Run Weeks: Incremental	14 / 16	16 / 18	12 / 16
Boundary Science + Particle Control	R16-1: Assess H-mode confinement, pedestal, SOL characteristics at higher B_T, I_P, P_{NBI}	R17-1: Assess scaling, mitigation of steady-state, transient heat-fluxes w/ advanced divertor operation at high power density R17-2: Assess high-Z divertor PFC performance and impact on operating scenarios	R18-1: Assess impurity sources and edge and core impurity transport IR18-1: Investigation of power and momentum balance for high density and impurity fraction divertor operation
Core Science	R16-2: Assess effects of NBI injection on fast-ion f(v) and NBI-CD profile	R17-3: Assess τ_E and local transport and turbulence at low ν^* with full confinement and diagnostic capabilities	IR18-2: Assess role of fast-ion driven instabilities versus micro-turbulence in plasma thermal energy transport Begin ~1 year outage for major facility enhancement(s) sometime during FY2018
Integrated Scenarios	R16-3: Develop physics + operational tools for high-performance: κ, δ, β, EF/RWM	IR17-1: Assess fast-wave SOL losses, core thermal and fast ion interactions at increased field and current R17-4: Develop high-non-inductive fraction NBI H-modes for sustainment and ramp-up	R18-2: Control of current and rotation profiles to improve global stability limits and extend high performance operation R18-3: Assess transient CHI current start-up potential in NSTX-U
FES 3 Facility Joint Research Target (JRT)	C-Mod leads JRT: Assess disruption mitigation, initial tests of real-time warning, prediction	DIII-D leads JRT: TBD… possibly something on energetic particles	NSTX-U leads JRT: TBD

Backup

NSTX Physics Results

Kinetic RWM theory consistent with RWM destabilization at intermediate plasma rotation; stability altered by collisionality



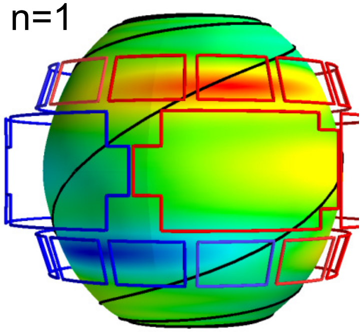
- Destabilization appears between precession drift resonance at low ω_ϕ, bounce/transit resonance at high ω_ϕ
- Destabilization moves to increased ω_ϕ as ν decreases

S. Sabbagh, NF 2010
J. Berkery, PRL 2010

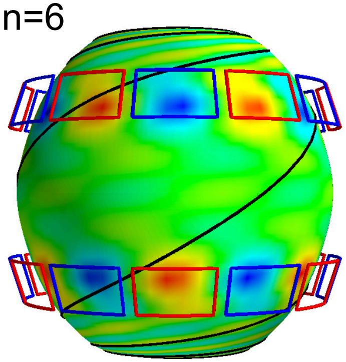
NCC Physics Design completed: Optimization for NTV braking performed with IPEC coupling matrix

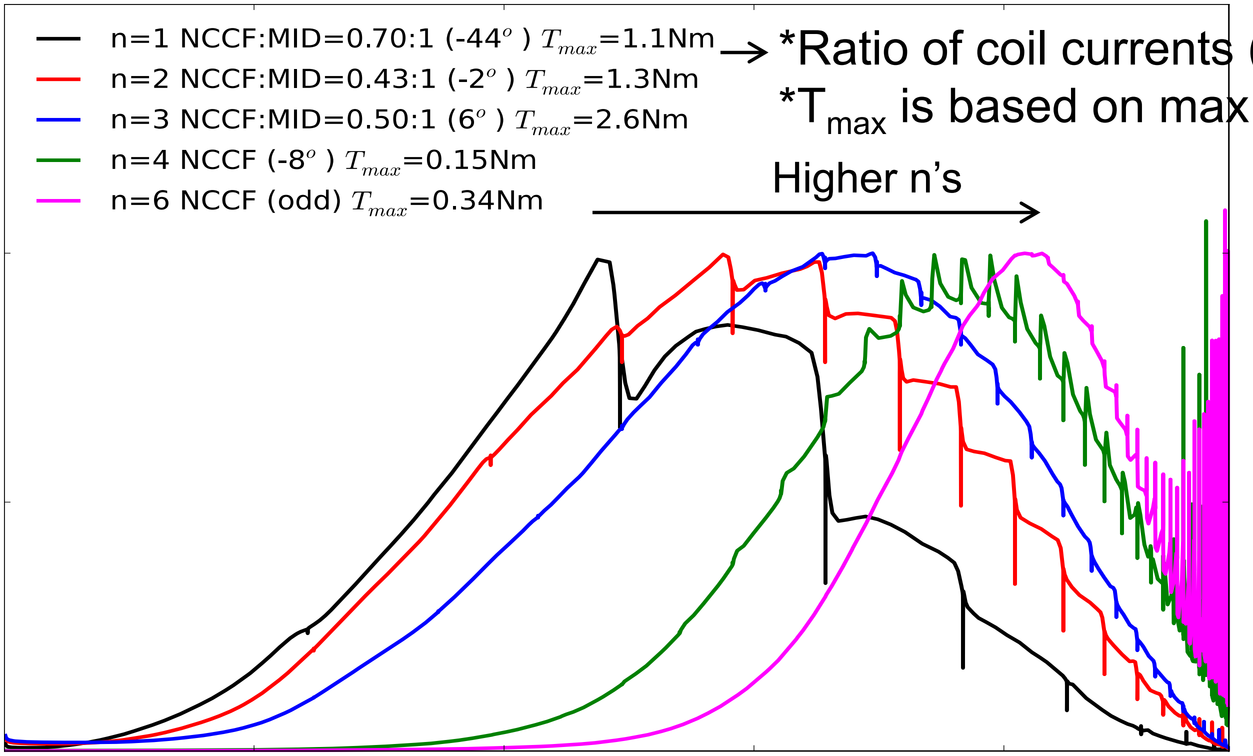
- NCC and midplane coils can be combined to remove the dominant resonant modes up to the second, giving the optimized NTV for core
  - NCC 2x12 provides n=1,2,3,4,6 optimized NTV, and 2x6 provides n=1,2,6
  - Optimized NTV can be used to control local torque with minimized resonance

n=1



n=6



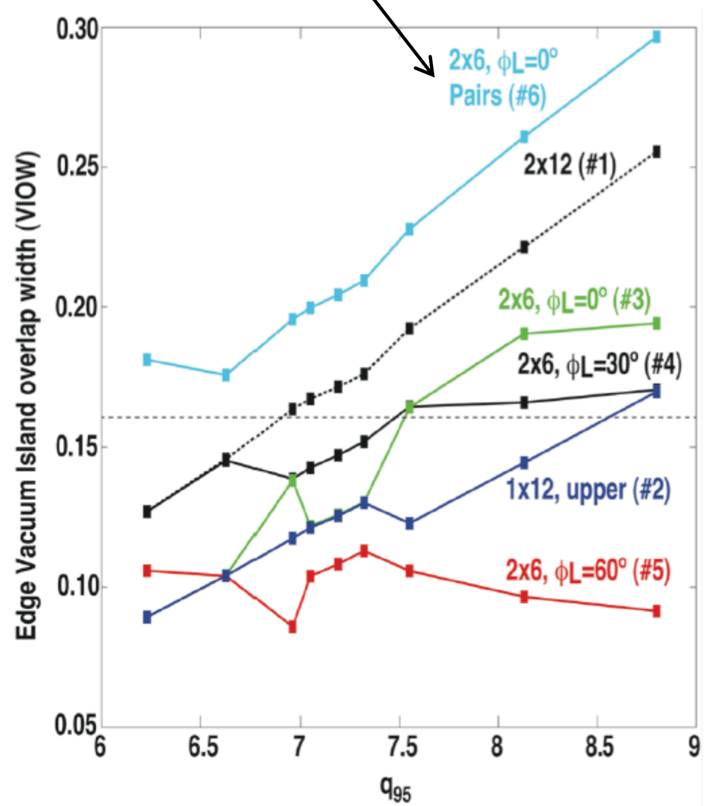


Study of RMP characteristics with NCC extended with TRIP3D (T. Evans, GA) – 2x12 NCC (and 2x7) favorable for RMP

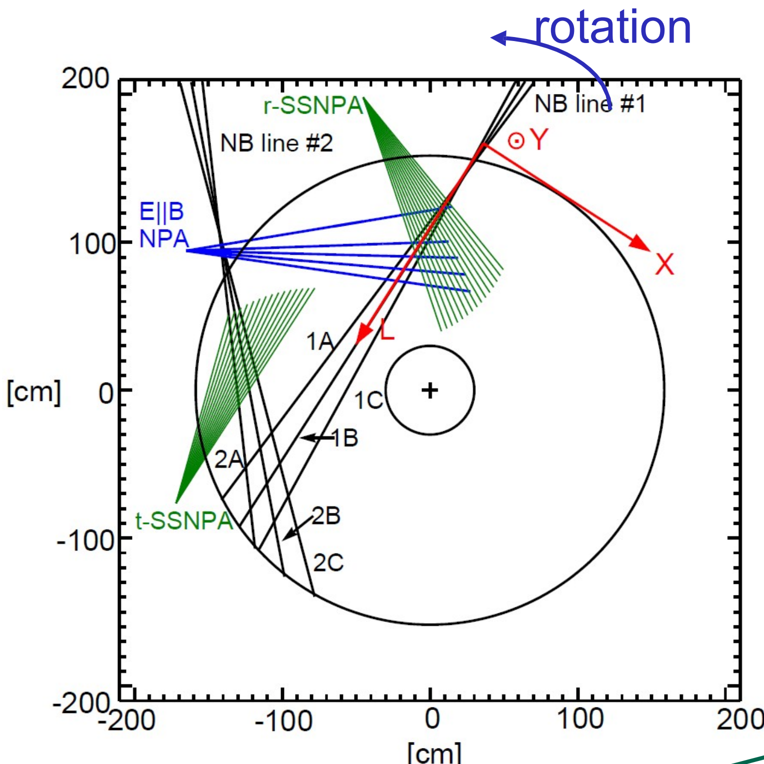
- Vacuum Island Overlap Width (VIOW) analysis shows full NCC 1kAt can produce sufficient VIOW in a wide range of q_{95}, but partial NCC needs more currents with low q_{95} targets
  - Also shows 2x7, with "one" more additional array upon partial NCC can provide the greater VIOW by toroidal coupling (n=2,4,9)

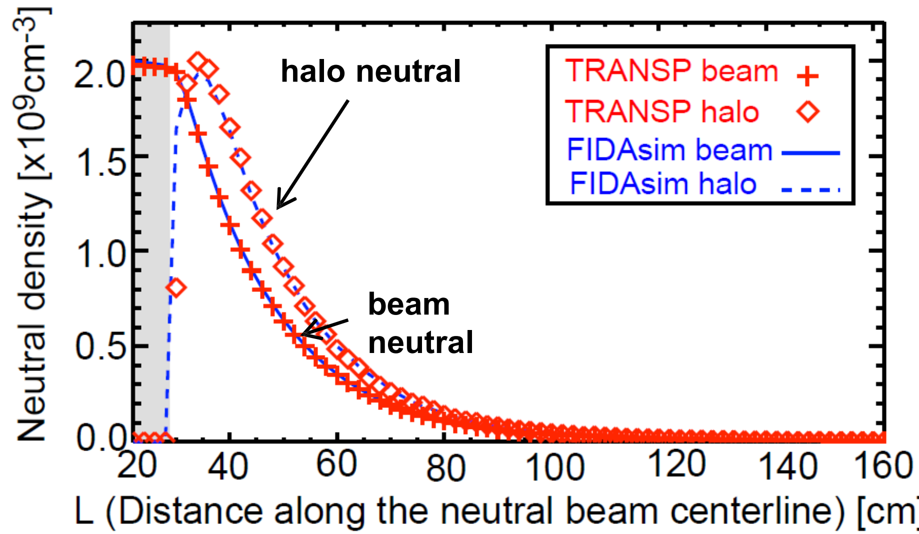
NCC Configurations Used for the plots Shown in Figures 1 and 2

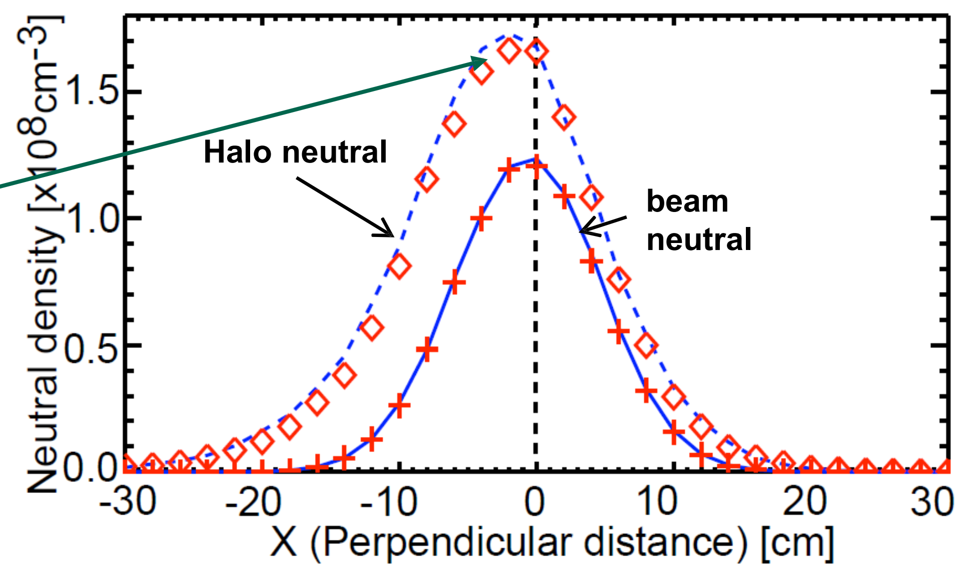
#	Description, Color	NCC Configuration Layout
1	2x12 **dashed black line**	+ + - - + + - - + + - - + + - - + + - - + + - -
2	1x12, upper **solid blue line**	+ + - - + + - - + + - - □ □ □ □ □ □ □ □ □ □ □ □
3	2x6, ϕ_L=0º **solid green line**	+ □ - □ + □ - □ + □ - □ + □ - □ + □ - □ + □ - □
4	2x6, ϕ_L=30º **solid black line**	+ □ - □ + □ - □ + □ - □ □ + □ - □ + □ - □ + □ -
5	2x6, ϕ_L=60º **dashed red line**	+ □ - □ + □ - □ + □ - □ - □ + □ - □ + □ - □ + □
6	2x7, ϕ_L=0º Pairs **solid light blue line**	+ □ - □ + - □ + □ - □ □ - □ + □ - + □ - □ + □ □



Excellent Agreement between TRANSP and FIDAsim when Using the Same ADAS Ground State Cross Section Tables (2)

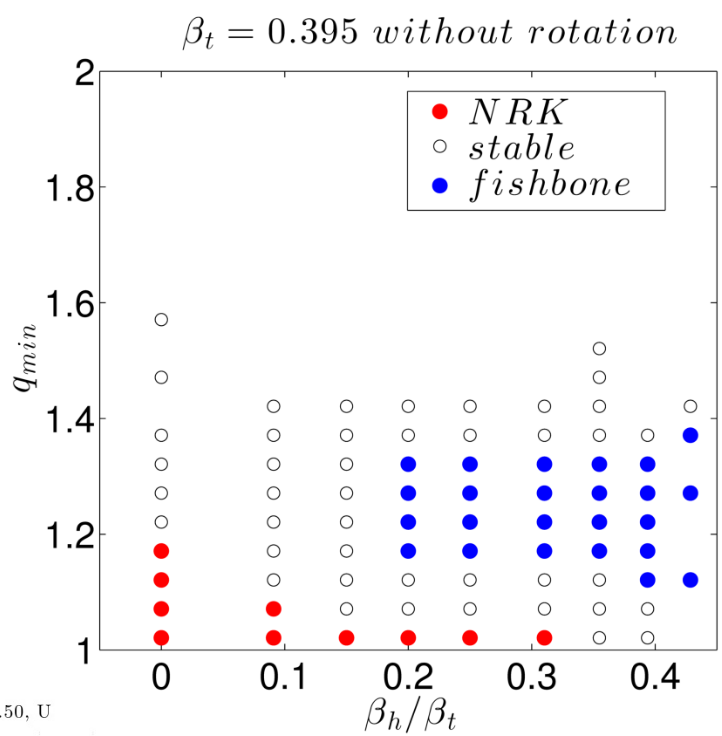


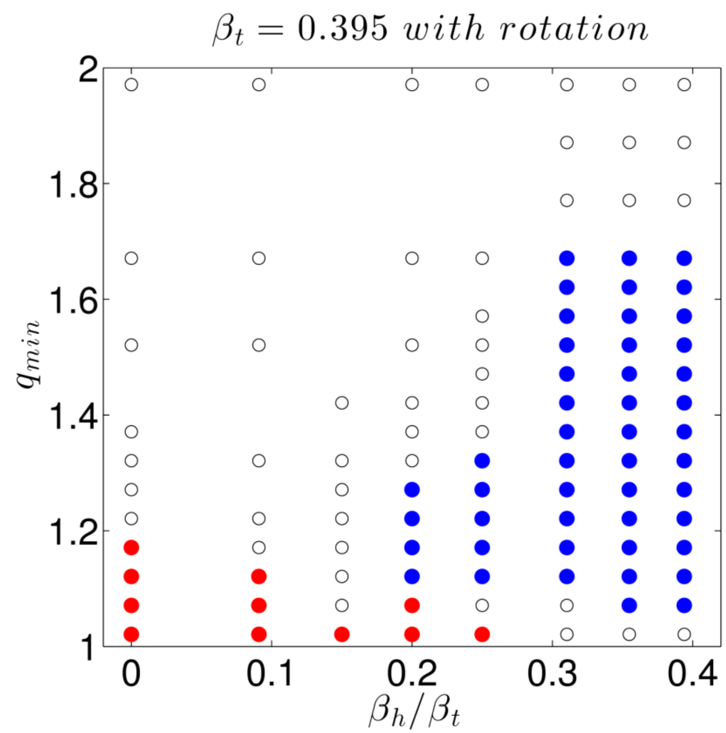




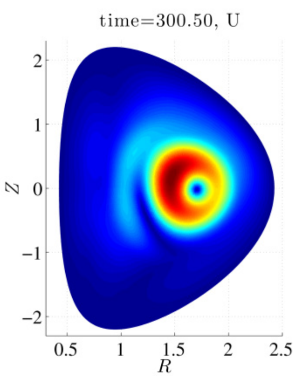
The peak of halo neutrals shifts in the same direction as toroidal rotation, as expected

Linear stability analysis of fishbones with M3D-K finds rotation / rotation shear destabilizing at elevated q

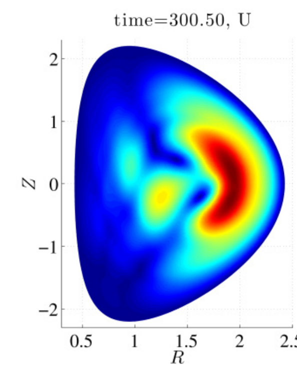




Guoyong Fu, Feng Wang
NSTX-U / theory partnership



- Eigenfunction broadens with rotation
  - Enhanced fast-ion transport or triggering / seeding of tearing modes?
  - Next step: investigate with non-linear runs.



New "kick" model for fast-ion transport predicts different J_{NB}, ion/electron power split, inferred thermal transport

Nucl. Fusion 55 (2015) 053018 M. Podestà *et al*

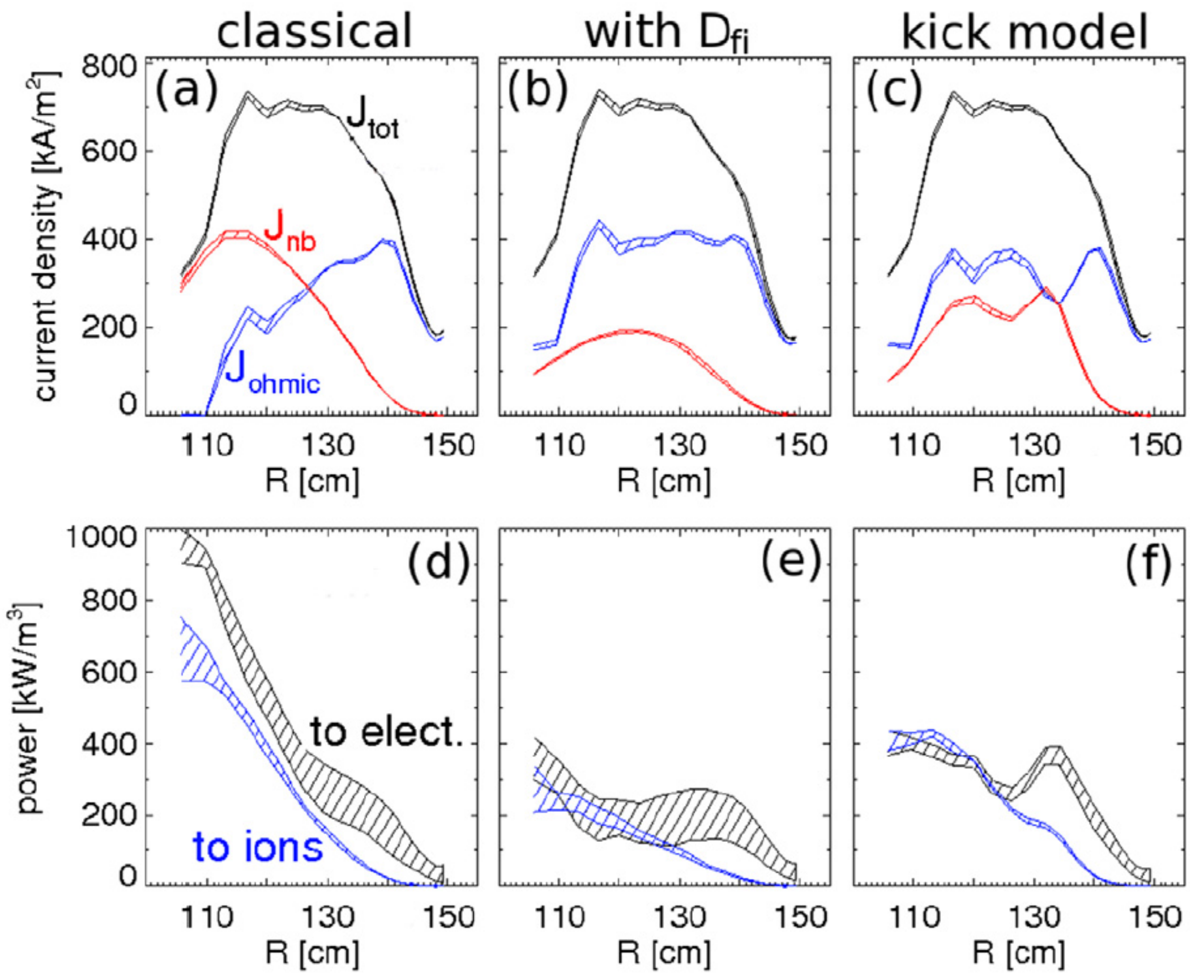
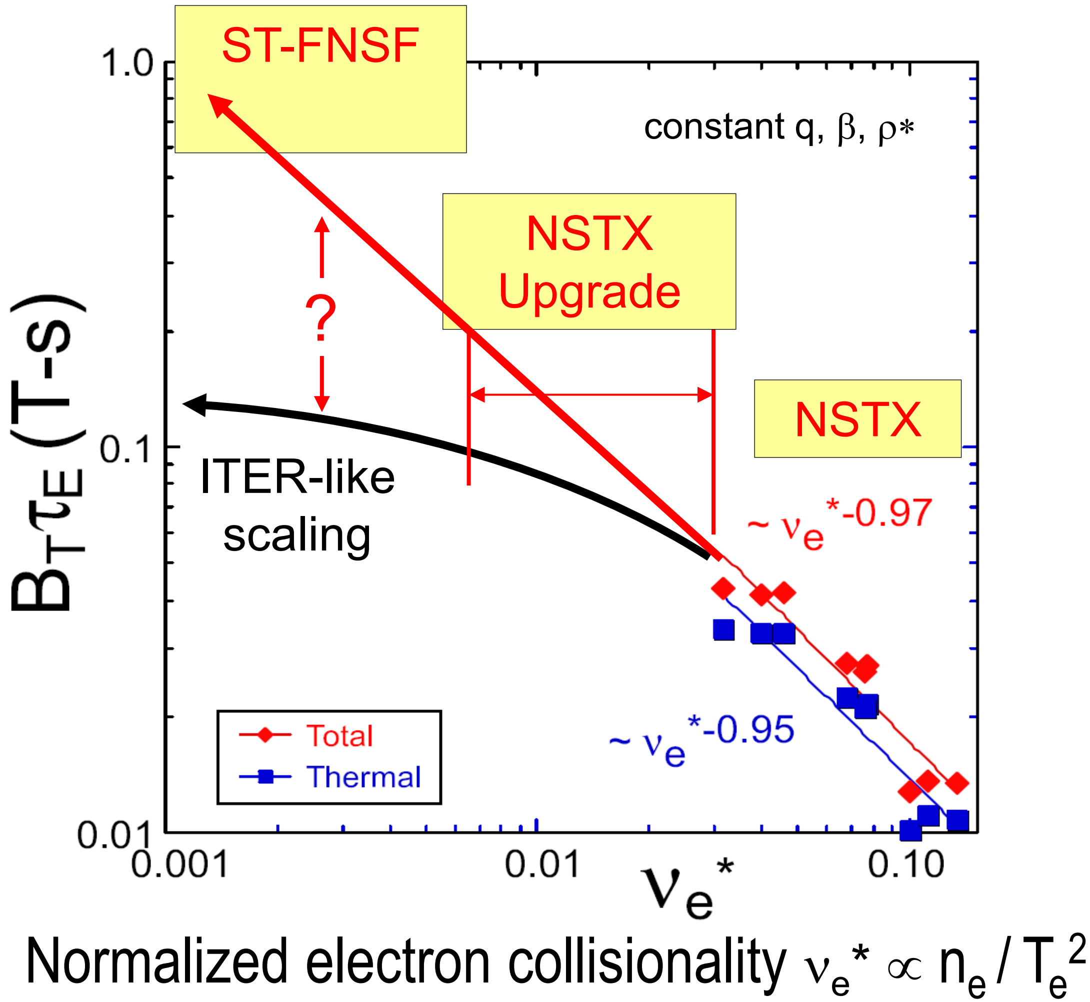


Figure 11. (*a*)–(*c*) Current density profiles calculated by TRANSP with different assumptions for fast ion transport (NSTX #139048). TRANSP results are averaged over $t = 300$–305 ms. (*d*)–(*f*) Total heating power transferred to electrons and ions for the three cases.

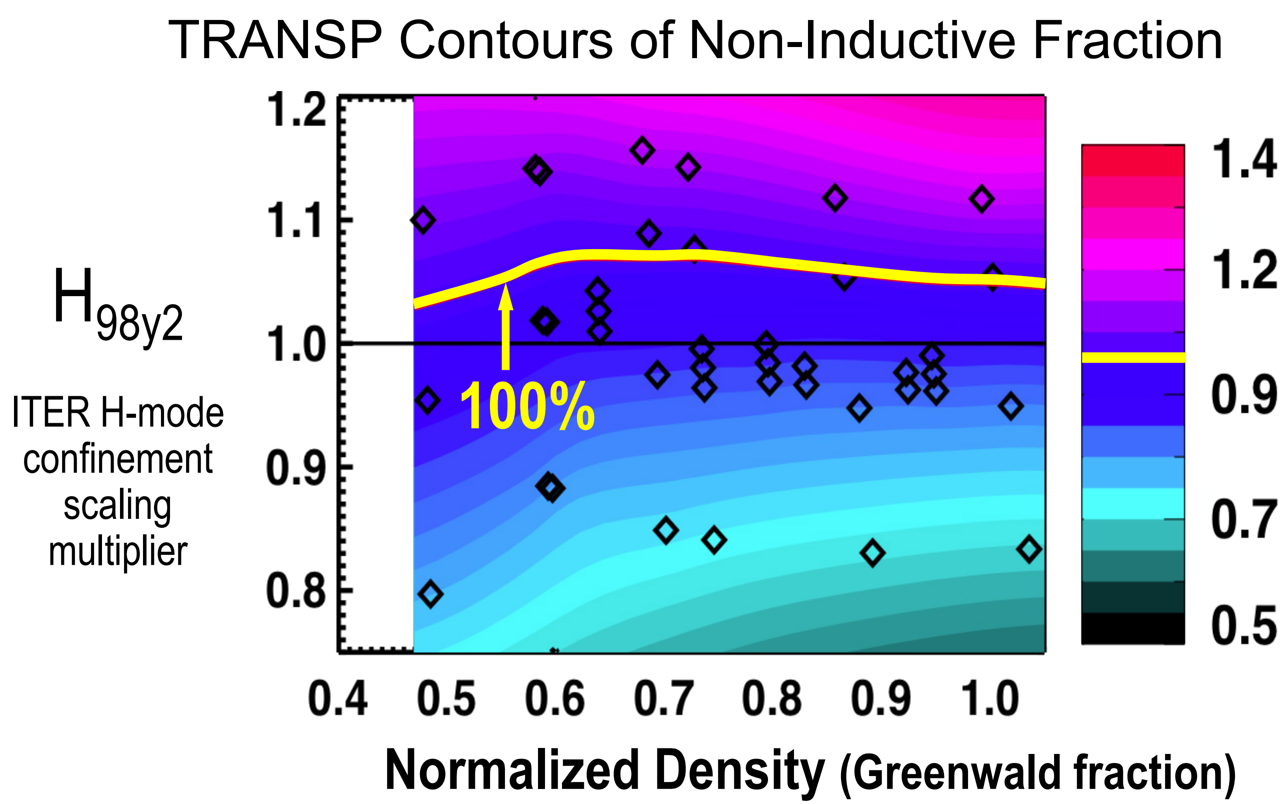
Major motivation for NSTX/MAST Upgrades: Determine if confinement trend continues, or is like conventional A



Favorable confinement results could lead to more compact ST reactors

NSTX achieved 70% "transformer-less" current drive

Will NSTX-U achieve 100% as predicted by simulations?

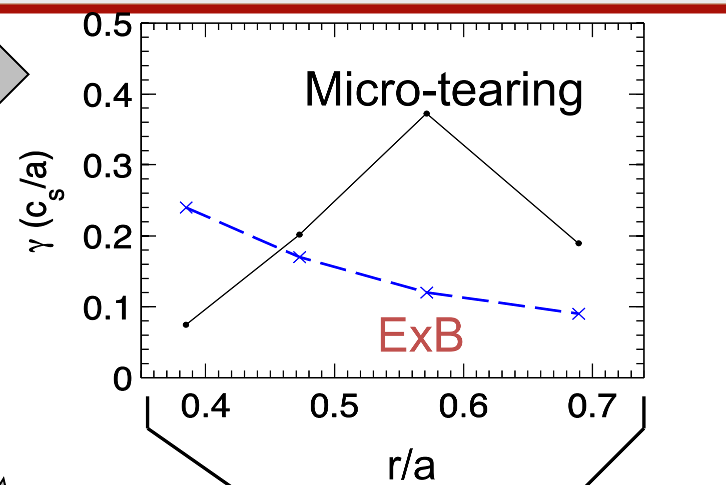


I_P=1 MA, B_T=1.0 T, P_{NBI}=12.6 MW

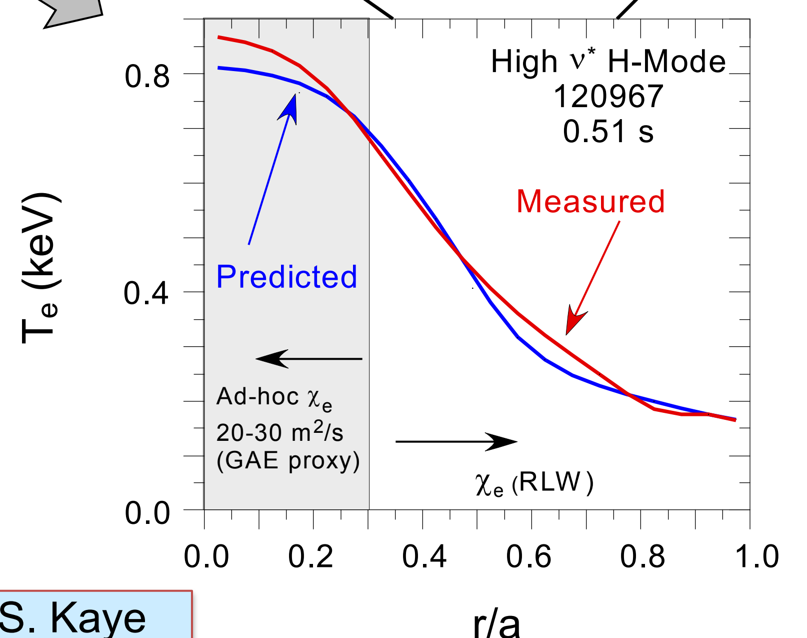
Steady-state operation required for ST, tokamak, or stellarator FNSF

Progress in predicting T_e using reduced χ_e models in regimes where single micro-instability is dominant

- Linear gyrokinetic simulations find microtearing unstable in mid-radius region of high-collisionality H-modes
  – Other micro-instabilities subdominant at this location for this class of discharge



- Reduced model for micro-tearing χ_e *(Rebut-Lallia-Watkins (RLW) - 1988)* shows reasonable agreement between predicted & measured T_e for r/a > 0.3
  – χ_e >> RLW must be used in core to match central T_e - may be due to GAE/CAE

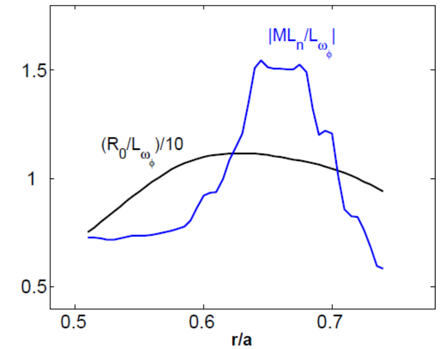


- Reduced ETG models in low-β L-modes also show reasonable T_e agreement for r/a > 0.3 (not shown)

S. Kaye

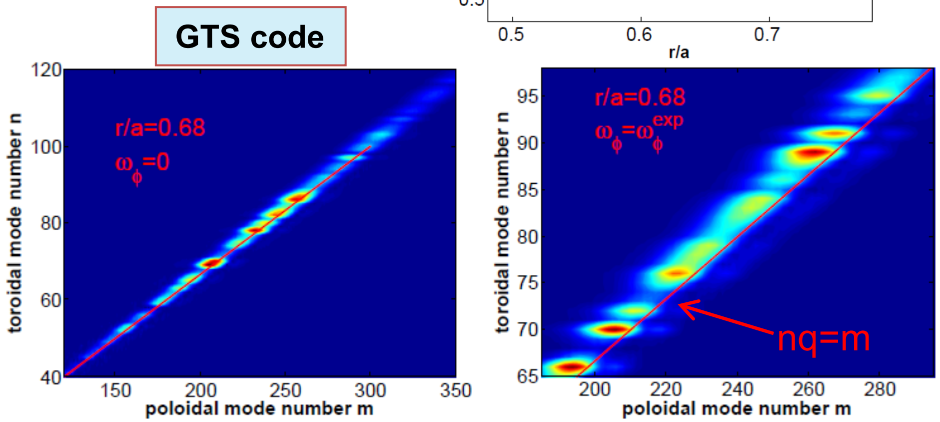
Strong flow shear can destabilize Kelvin-Helmholtz (K-H) instability in NSTX (Wang, TTF 2014)

- K-H instability expected for $|ML_n/L_\omega|>1$ from linear analytic theory (Catto, 1973; Garbet, 1999)

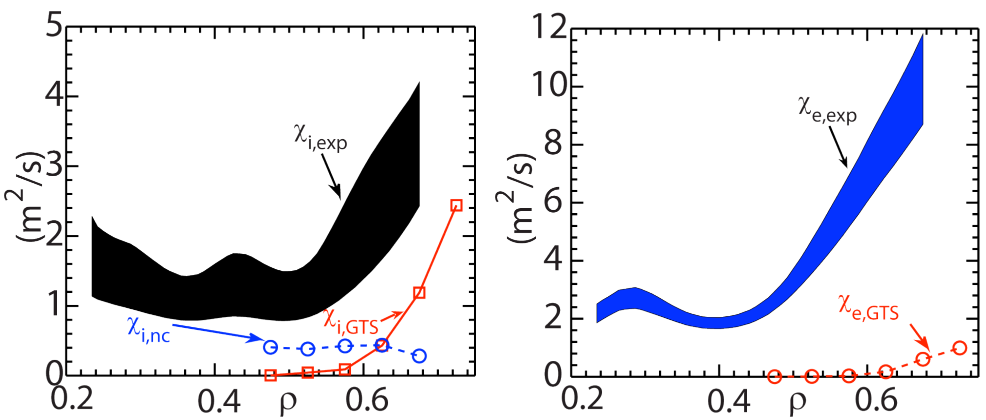


- K-H identified in global electrostatic ITG simulations for NSTX low β L-mode (Ren, 2013)
  - Mostly unstable K-H modes have finite $k_{\|}$ (shifted away from nq=m)

$$k_{\|} \sim \frac{k_\theta \rho_s}{2c_s}\frac{1}{n}\frac{d(nV_{\|})}{dr} \sim \frac{d\omega_\phi}{dr}$$



- K-H/ITG + neoclassical close to experiment χ_i
  - χ_e underpredicted
  - ETG possibly important

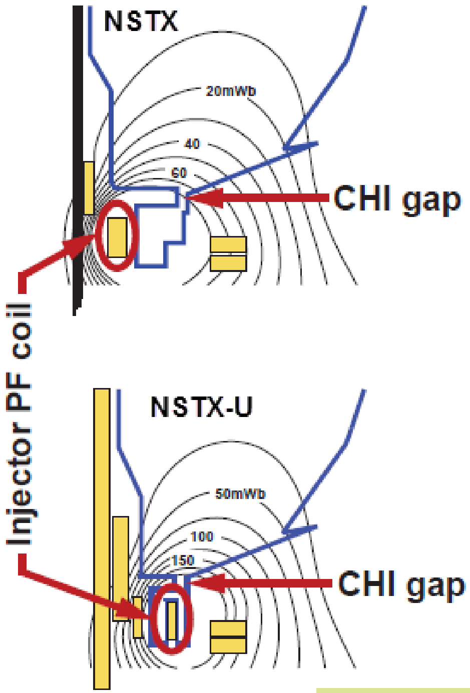


Nonlinear ETG simulations to be run in FY14-15 to investigate contribution in NSTX L-modes

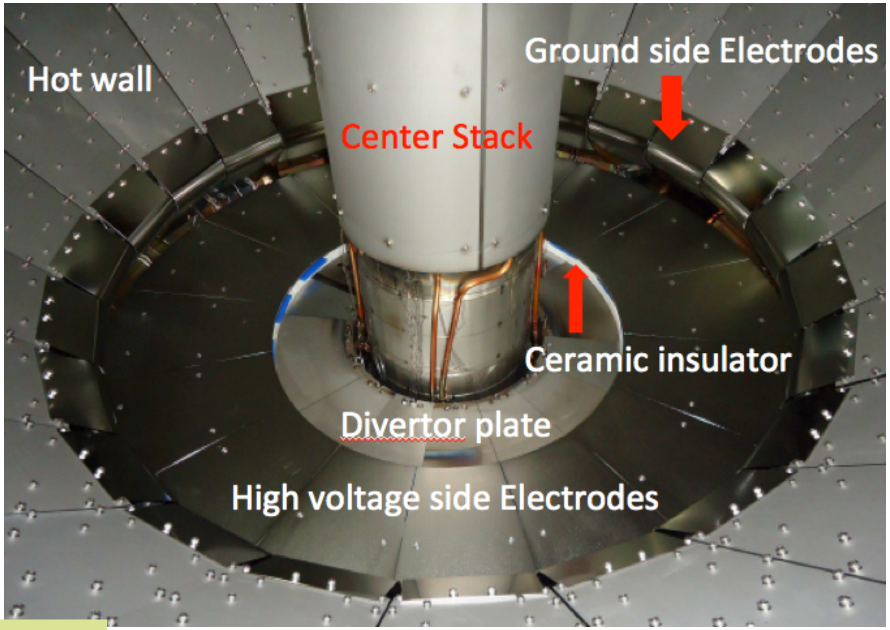
Collaborating with QUEST to explore CHI + ECH solenoid-free start-up in support of ST-FNSF

CHI Start-Up

- **Inj. Flux in NSTX-U is about 2.5 times higher than in NSTX**
- **NSTX-U coil insulation greatly enhanced for higher voltage ~ 3 kV operation**



CHI Implementation on QUEST showing installed electrodes



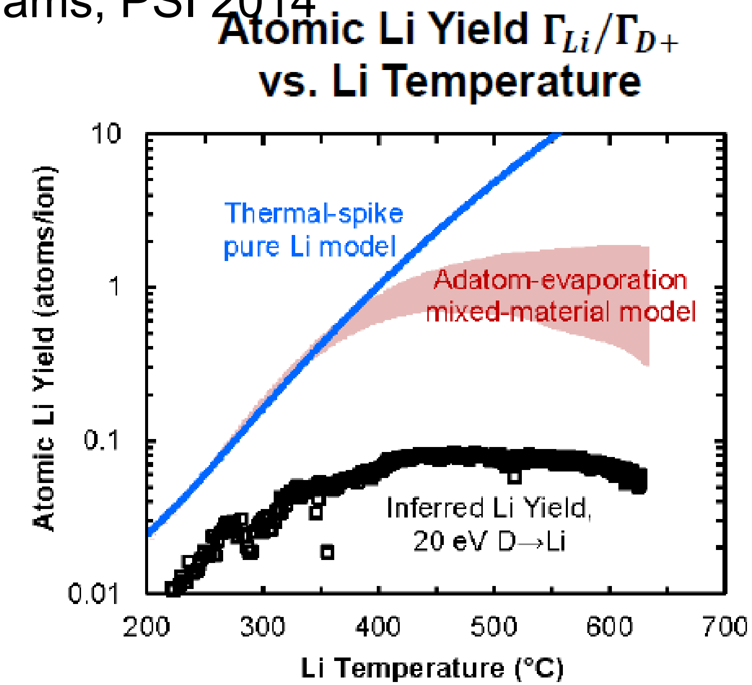
U. Washington

- Refurbishment of CHI Cap Bank completed.
- Fabrication of the CHI gas injection system and operation procedure for the QUEST ST experiment in Japan completed.
- Fabrication of the CHI capacitor bank for QUEST is nearing completion.

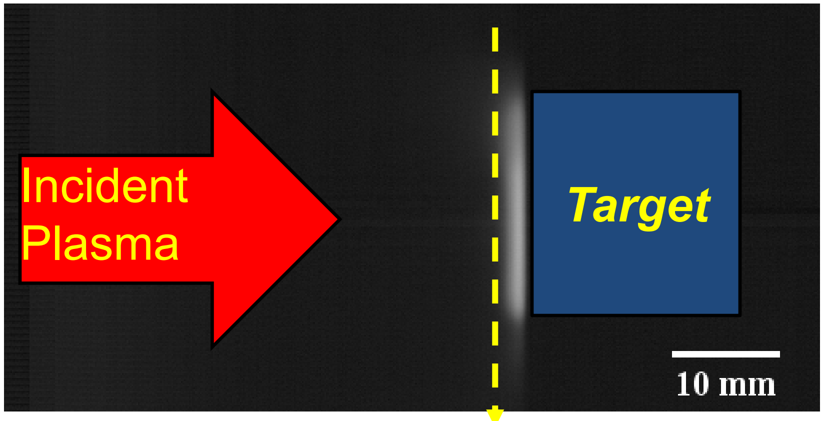
Magnum-PSI experiments on high-temperature Li show strongly reduced erosion and stable cloud production

- Gross erosion measured spectro-scopically in divertor-like plasma
  - Neon plasma reproduces Langmuir Law
  - Deuterium *suppresses* erosion
- Reduced gross erosion and strong re-deposition result in 10× longer lifetime of 1 micron coating
  - Consistent with Li trapping in pre-sheath
  - Pre-sheath scale length consistent with neutral Li emission region (~3mm)

Abrams, PSI 2014



Neutral Li emission, t=2.5s



Jaworski, PSI 2014

Backup

Copper TF ST-FNSF

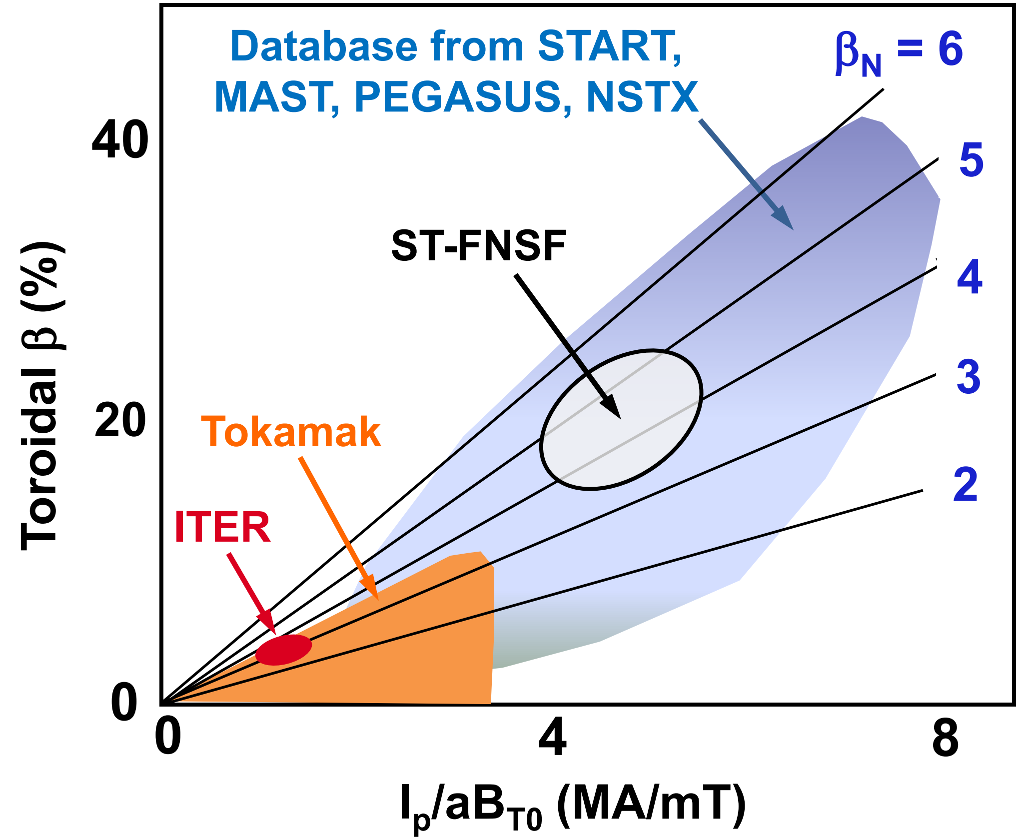
High β_T enables compact Fusion Nuclear Science Facility (FNSF) with high neutron wall loading

$$P_{fusion} \propto \langle p \rangle^2 \times Vol$$

$$P_{fusion} \propto \beta_T^2 \; B_{T0}^4 \times Vol$$

Physics limited (β_T^2)

$ — Investment & engineering limited ($B_{T0}^4 \times Vol$)



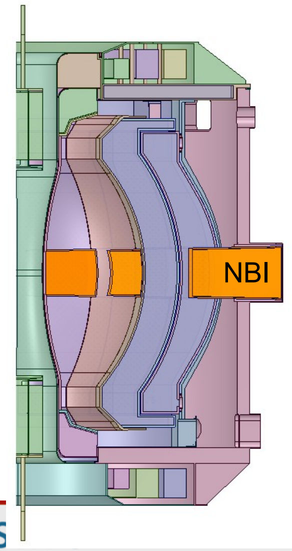
$$W_n \propto P_{fusion} / Area$$

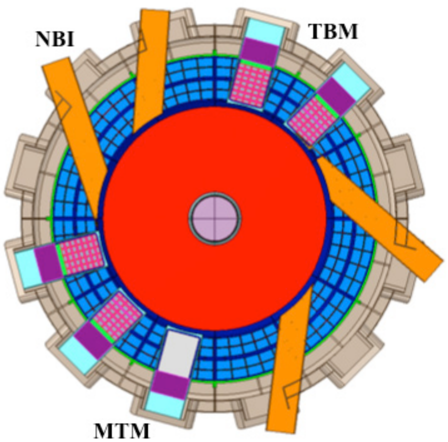
$$W_n \propto \beta_T^2 \; B_{T0}^4 \; a \quad \text{(not strongly size dependent)}$$

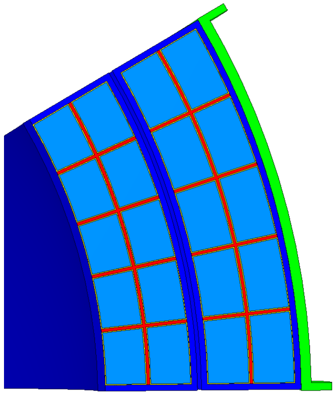
W_n ~ 1-2 MW/m² with R ~ 1-2m FNSF feasible!

ST-FNSF shielding and TBR analyzed with sophisticated 3-D neutronics codes

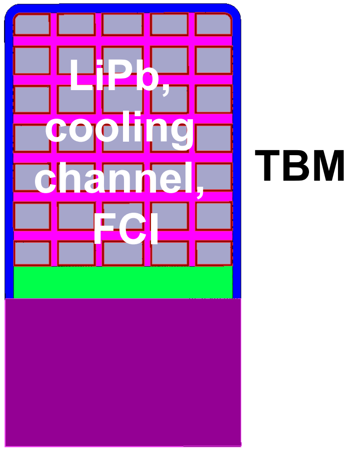
- CAD coupled with MCNP using UW DAGMC code
- Fully accurate representation of entire torus
- No approximation/simplification involved at any step:
  - Internals of two OB DCLL blanket segments modeled in great detail, including:
    - FW, side, top/bottom, and back walls, cooling channels, SiC FCI
  - 2 cm wide assembly gaps between toroidal sectors
  - 2 cm thick W vertical stabilizing shell between OB blanket segments
  - Ports and FS walls for test blanket / materials test modules (TBM/MTM) and NNBI







Heterogeneous OB Blanket Model, including FW, side/back/top/bottom walls, cooling channels, and SiC FCI



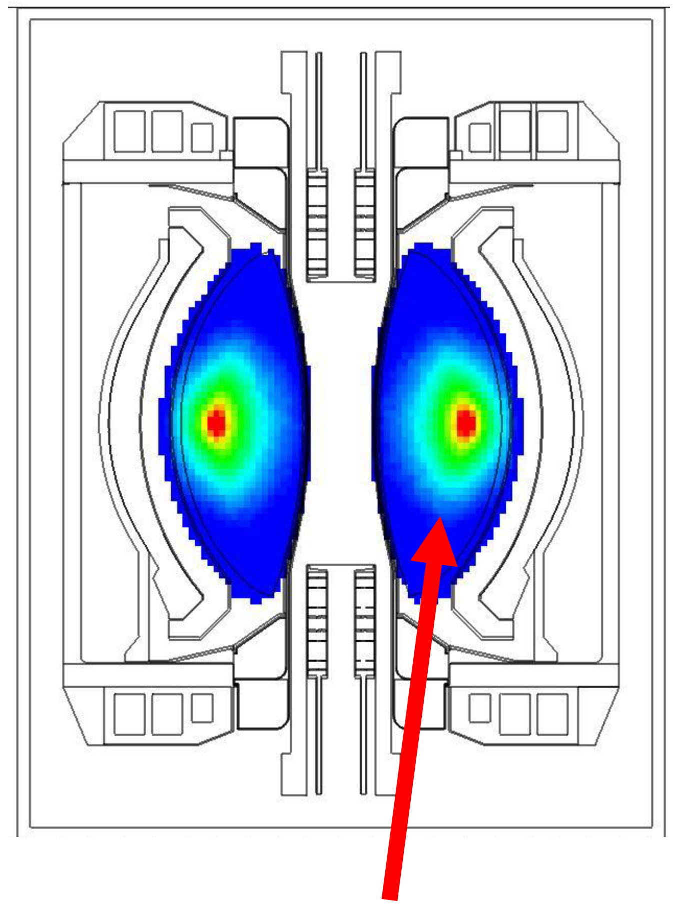
Two sizes (R=1.7m, 1m) assessed for shielding, TBR

<u>Parameter:</u>

Parameter		
Major Radius	1.68m	1.0m
Minor Radius	0.95m	0.6m
Fusion Power	162MW	62MW
Wall loading (avg)	$1MW/m^2$	$1MW/m^2$
TF coils	12	10
TBM ports	4	4
MTM ports	1	1
NBI ports	4	3

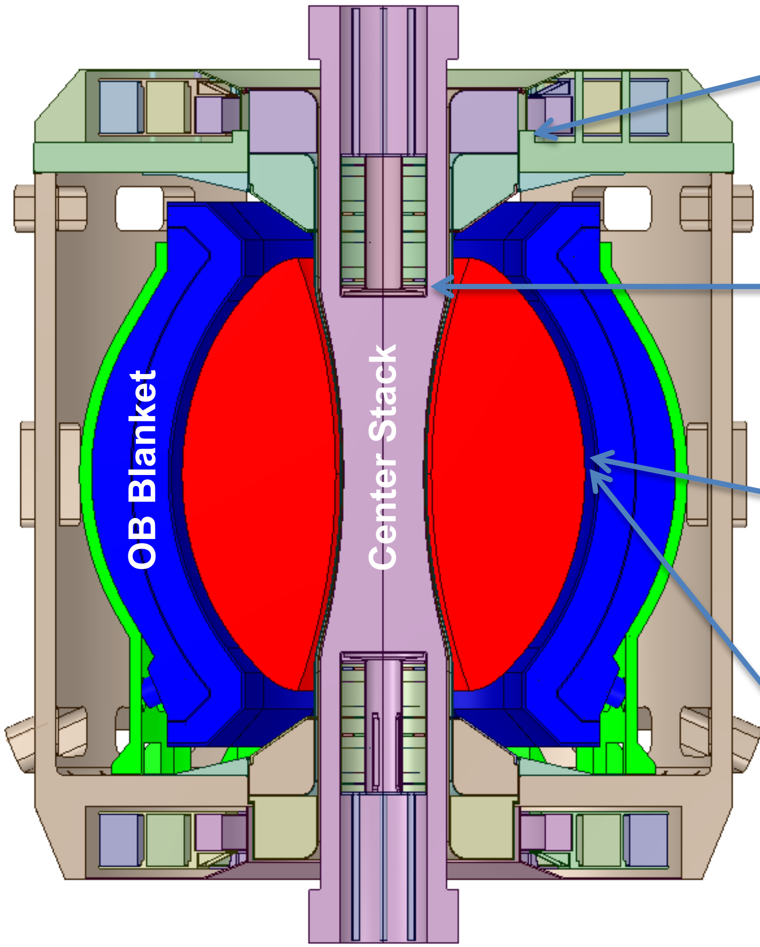
Plant Lifetime ~20 years

Availability 10-50%, 30% avg } **6 Full Power Years (FPY)**



Peak Damage at OB FW and Insulator of Cu Magnets

R=1.7m configuration



3-D Neutronics Model of Entire Torus

Dose to MgO insulator = 2×10^8 Gy @ 6 FPY < 10^{11} Gy limit

Dose to MgO insulator = 6×10^9 Gy @ 6 FPY < 10^{11} Gy limit

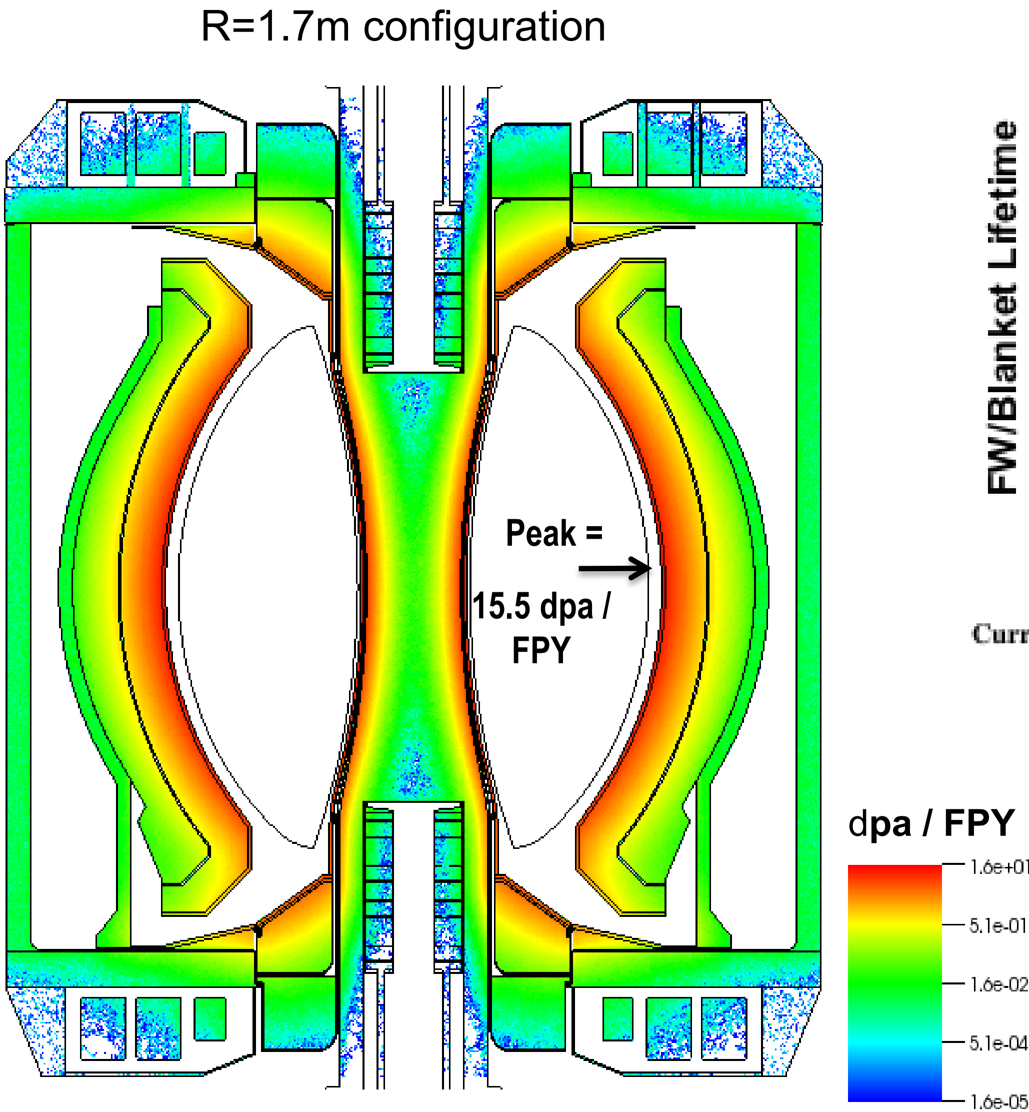
Peak dpa at OB midplane = 15.5 dpa / FPY

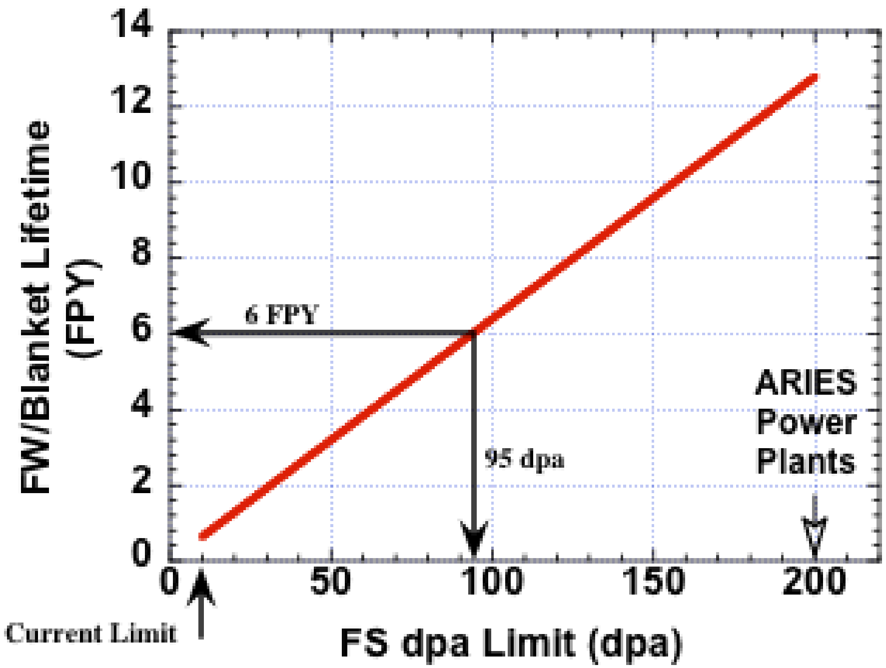
Peak He production at OB midplane = 174 appm/FPY

$\Rightarrow$ He/dpa ratio = 11.2

Mapping of dpa and FW/blanket lifetime
(R=1.7 m Device)

R=1.7m configuration

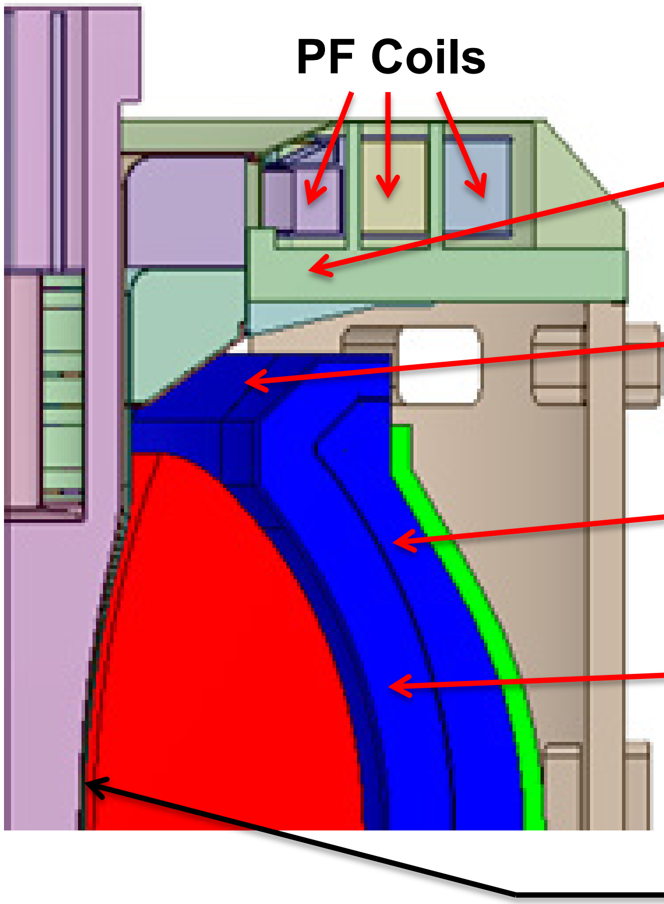




FW/blanket could operate for 6 FPY if allowable damage limit is 95 dpa

→ Peak EOL Fluence = 11 MWy/m^2

Options to increase TBR > 1

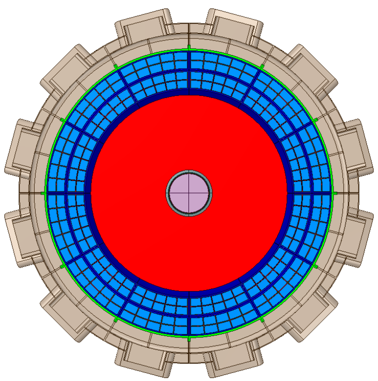


- Add to PF coil shield a thin breeding blanket (ΔTBR ~ +3%)
- Smaller opening to divertor to reduce neutron leakage
- Uniform OB blanket (1m thick everywhere; no thinning)
- Reduce cooling channels and FCIs within blanket (need thermal analysis to confirm)
- Thicker IB VV with breeding

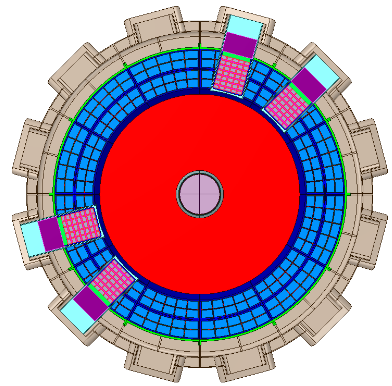
Potential for **TBR > 1 at R=1.7m**

UW has carried out most advanced and detailed analysis of Dual-Coolant Lead-Lithium (DCLL) blanket for ST-FNSF

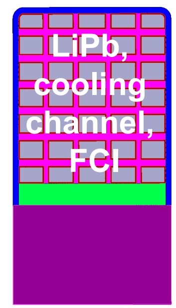
No ports or penetrations, homogeneous breeding zones:
TBR = 1.03



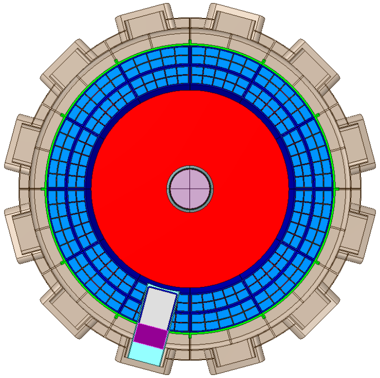
Add 4 Test Blanket Modules (TBMs)
TBR = 1.02 (ΔTBR = -0.01)



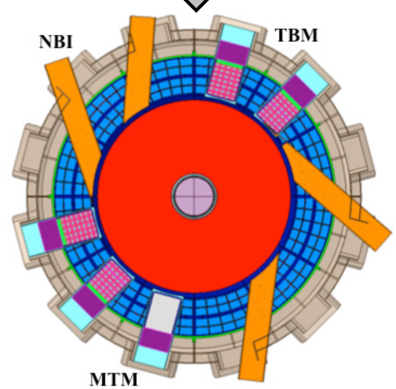
TBM



MTM



1 Materials Test Module (MTM)
TBR = 1.01 (ΔTBR = -0.02)



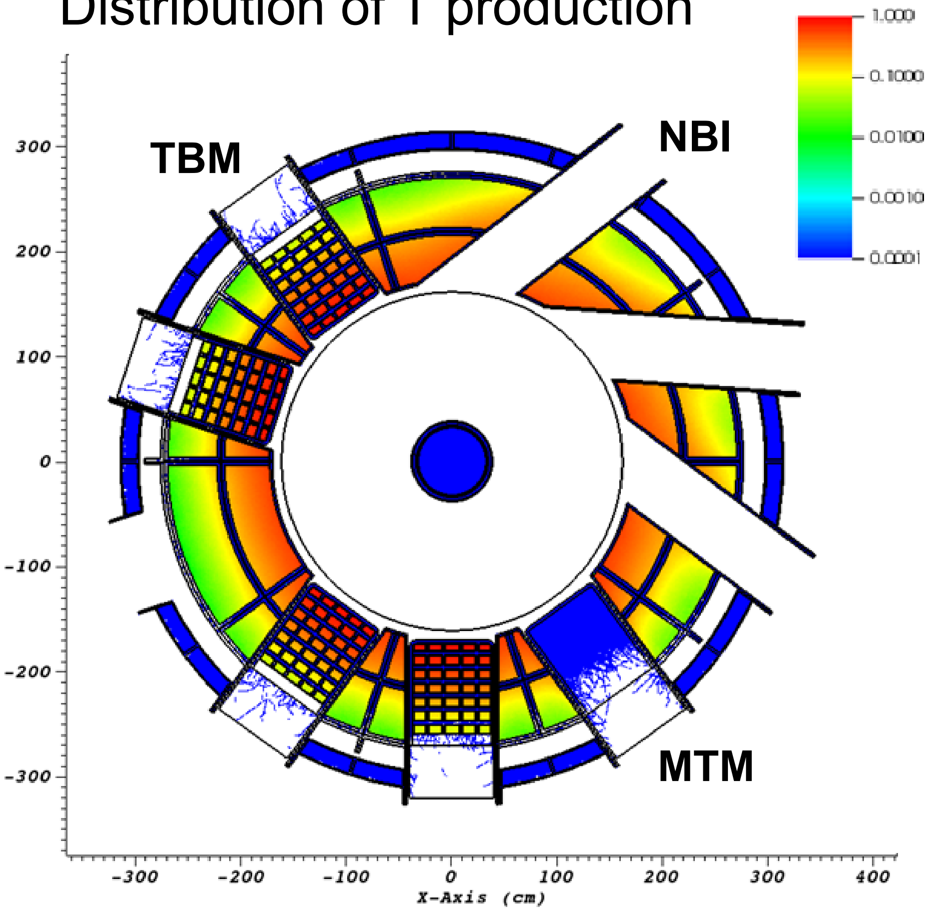
4 TBM + 1 MTM + 4 NBI
TBR = 0.97

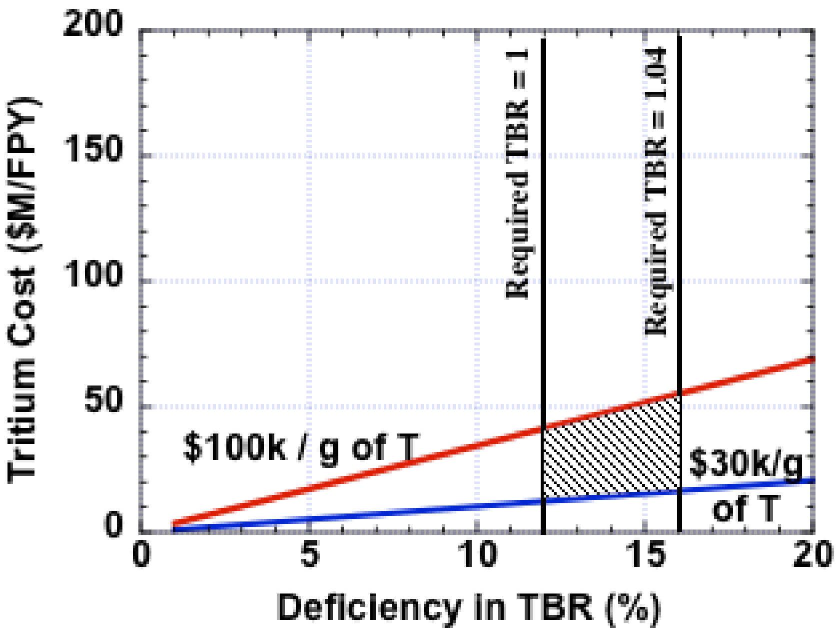
Approx. ΔTBR per port:

- TBM: -0.25%
- MTM: -2.0%
- NBI: -0.75%

R_0 = 1m ST-FNSF achieves TBR = 0.88

Distribution of T production





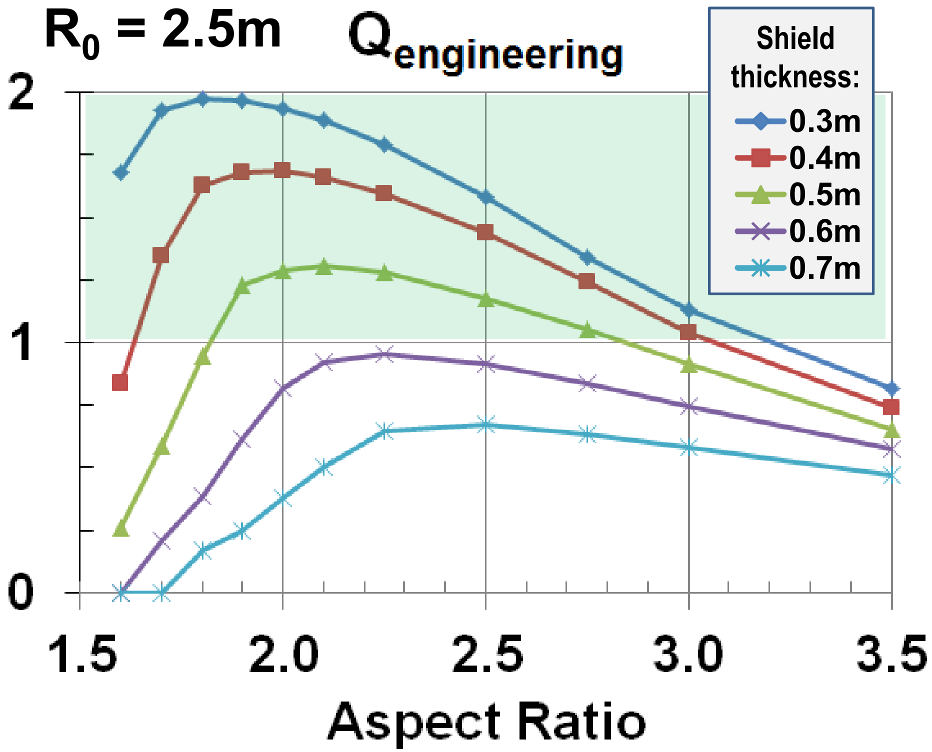
- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

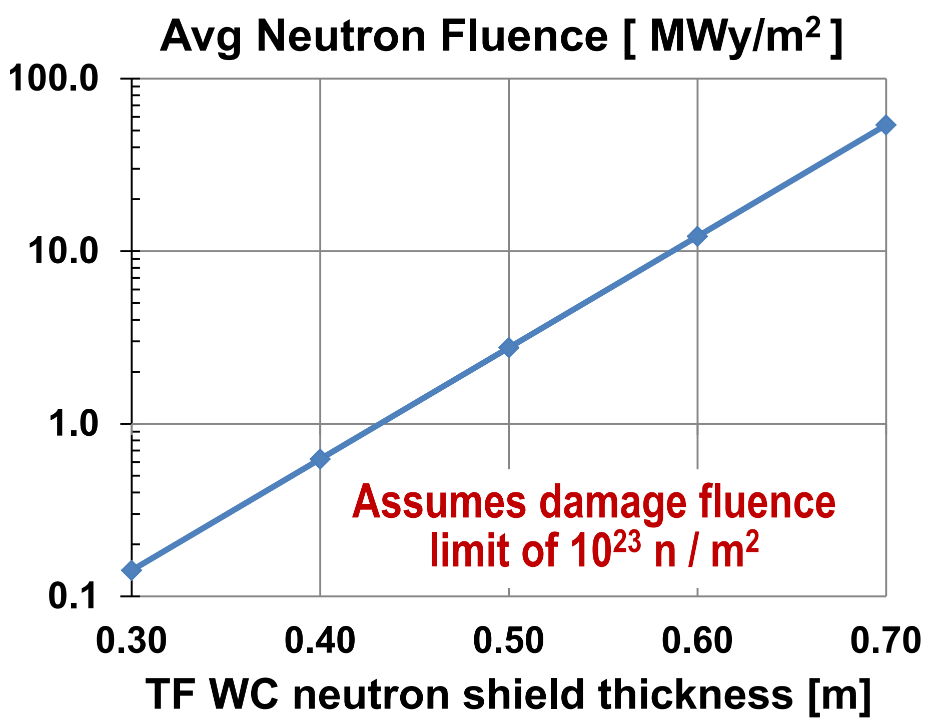
Backup

HTS-ST Pilot Plant Study

$Q_{eng} = P_{electric} / P_{consumed}$ is maximized between A = 1.8-2.5 for tokamak using HTS magnets

- Assumes fixed major radius R_0
- Elongation κ and β_N increase with increasing $\varepsilon = A^{-1}$
- Assumes no inboard tritium breeding (WC shield only)



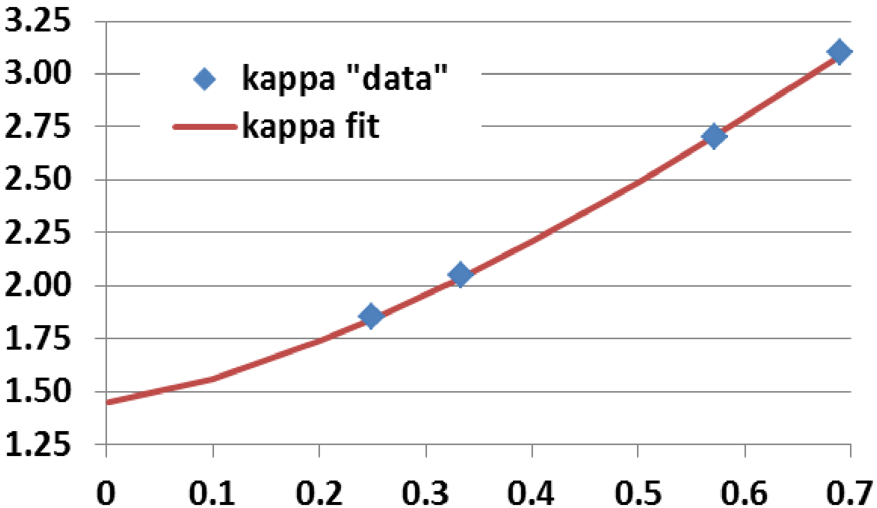


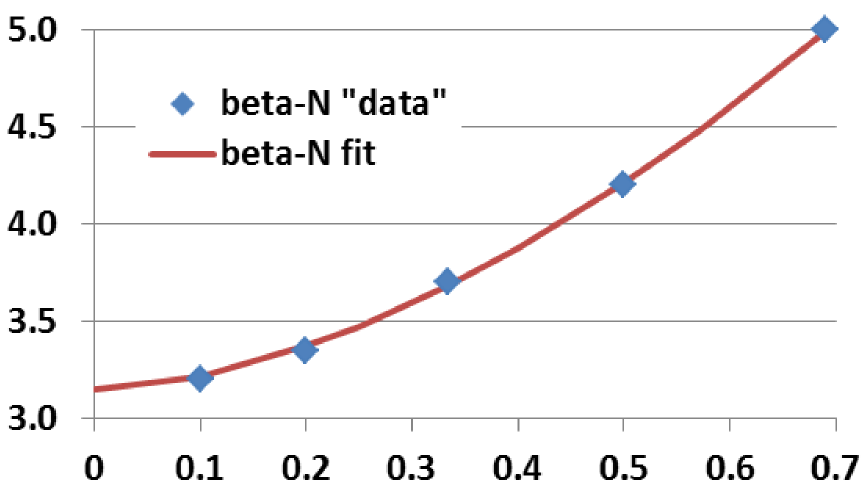
Result: Want Δ_{shield} / R_0 < ~20% for Q_{eng} > 1

HTS-ST Plasma constraints

- Fix plasma major radius at R_0 = 2.5-3m
  - Chosen to be large enough to allow space for HTS neutron shield and access Q_{eng} > 1
- Inboard plasma/FW gap = 4cm
- Use ε dependent $\kappa(\varepsilon)$, β_N (ε) (see next slide)
- Greenwald fraction = 0.8
- q* not constrained
  - q* is better ε-invariant than q_{95} for current limit
  - Want to operate with q* > 3 to reduce disruptivity
- 0.5MeV NNBI for heating/CD – fixed P_{NBI} = 50MW
- H_{98y2} adjusted to achieve full non-inductive CD

Aspect ratio dependence of limits: κ(ε), β_N(ε)





ε = A^{-1}

- NSTX data at low-A (+ NSTX-U and ST-FNSF modelling)
- DIII-D and EAST for higher-A
- κ → 1.4 for A → ∞
- Profile-optimized **no-wall** stability limit at f_{BS} ≈ 50%
  - Menard PoP 2004
- β_N → 3.1 for A → ∞

$$\beta_T \sim A^{-1/2}(1+\kappa^2)\beta_N^2/f_{BS}$$

$$\Rightarrow P_f \propto \epsilon(\kappa\beta_N B_T)^4$$

HTS-ST engineering constraints

- Magnet constraints (T. Brown ST-HTS Pilot, K-DEMO)
  - Maximum stress at TF magnet = 0.67-0.9GPa
  - Maximum effective TF current density = 65MA/m^2
  - OH at small R → higher OH solenoid flux swing for higher A
- Shielding / blankets
  - Assume HTS fluence limit of 3×10^{22} - 10^{23} n/m^2
  - No/thin inboard blanket, ~1+ m thick outboard blanket
    - 10x n-shielding factor per 15-16cm WC for HTS TF
    - Also 8cm inboard thermal shield + other standard radial builds
- Electrical system efficiency assumptions:
  - 30% wall plug efficiency for H&CD - typical of NNBI
  - 45% thermal conversion efficiency - typical of DCLL
    - Also include pumping, controls, other sub-systems
    - See Pilot Plant NF 2011 paper for more details

Selection of HTS-ST device performance goals

- Decided w/ CCFE + UW for HTS-ST to have goals of:
  - 6MWy/m^2 neutron wall loading (peak) at outboard midplane AND Q_{eng} ~ 1 – similar to previous PPPL Pilot Plant Study
  - Assume n-radiation damage limit of 3-4 × 10^{22}/m^2
    - HTS already tested to this damage fluence range (see next slide)

  → WC shield thickness ~ 60cm, Δ/R = 0.2 → R_0 = 3m
- With only outboard breeding, optimal A ~ 2.1-2.4
- But, for TBR ~ 1 probably need A ≤ 2 → chose A=2
- Chosen design point (so far):
  - R=3m, B_T = 4T, A=2, κ=2.5, β_N = 4.2 (~no-wall limit)
  - H_{98y2} ~ 1.7, H_{Petty} ~ 1.2-1.3, H_{ST} ~ 0.7, P_{fusion} ~ 500-600MW
  - 80% Greenwald fraction, 50MW of 0.5-0.7 MeV NNBI
  - I_P = 12MA, double-swing of small OH provides ~ 2-3MA

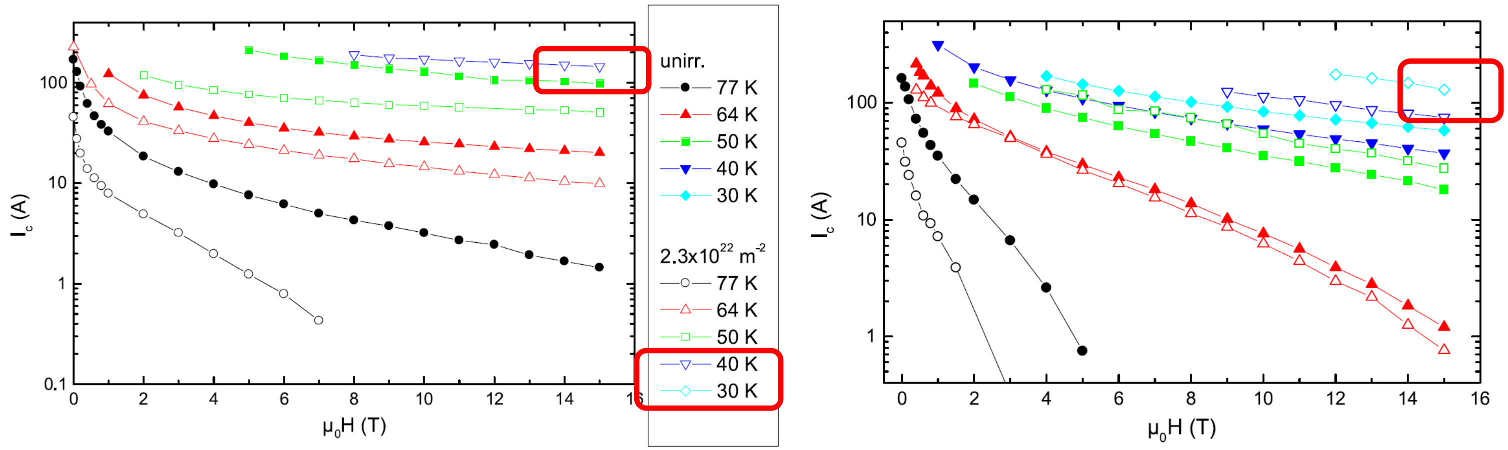


Figure 6. Critical currents (ASC-40) in magnetic fields applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) before and after irradiation to a fast neutron fluence of $2.3 \cdot 10^{22}$ m^{-2}.

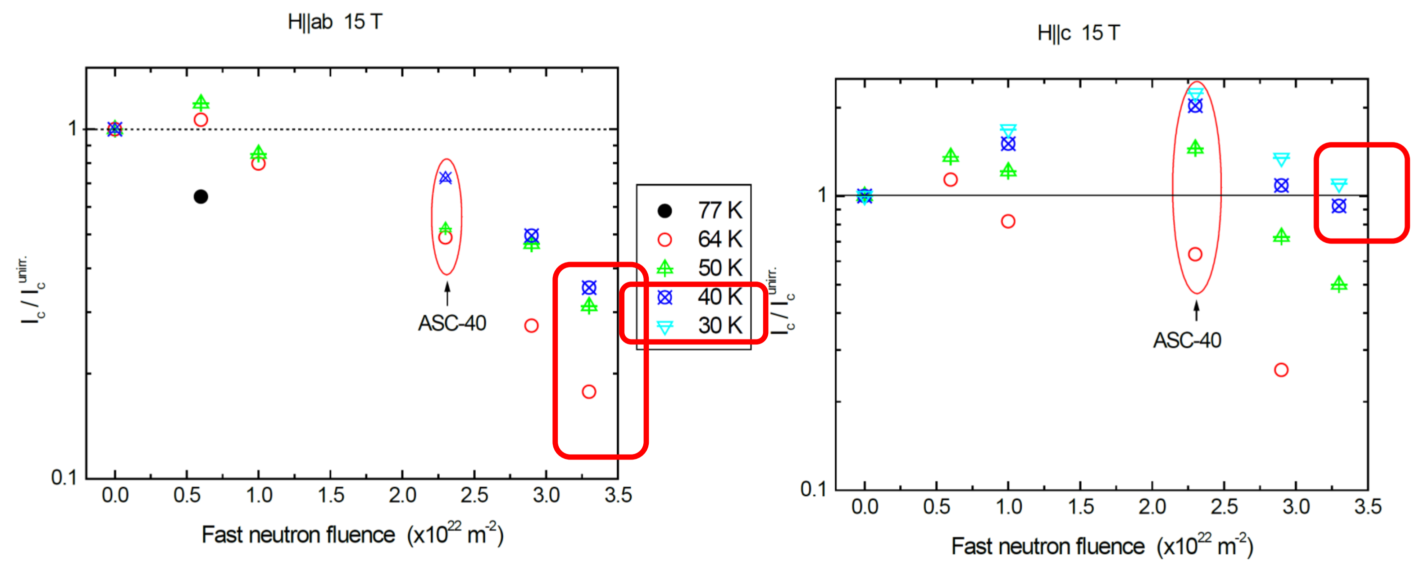
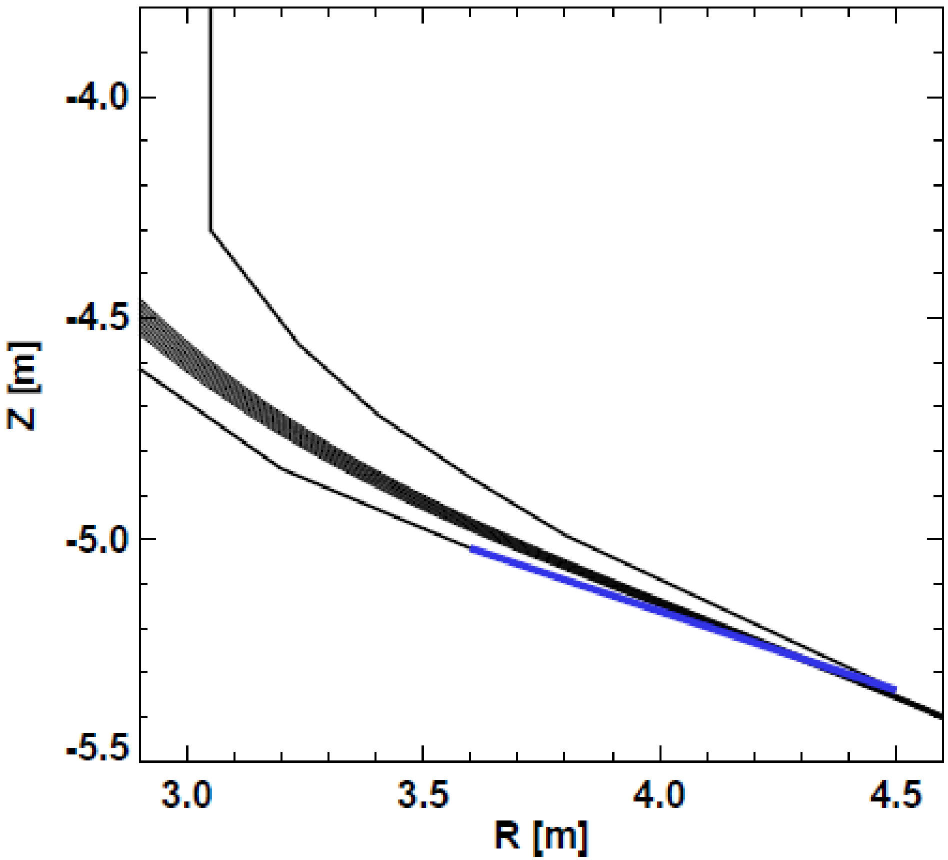
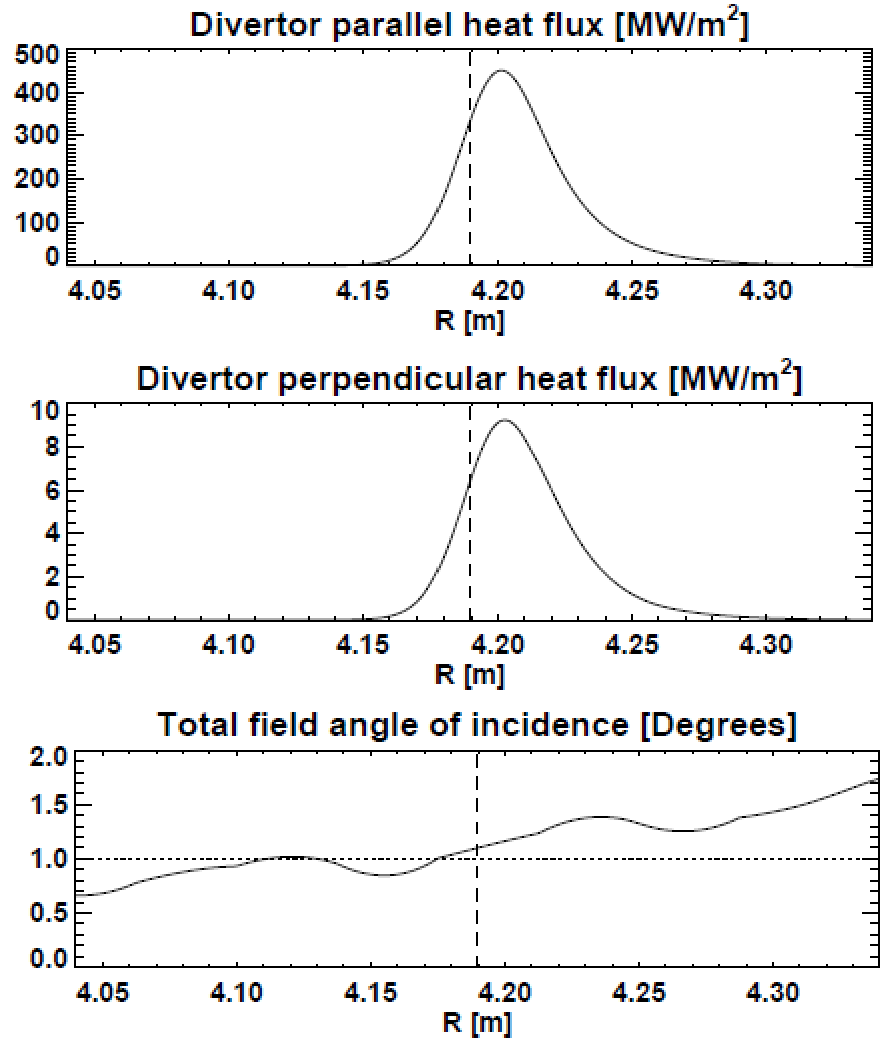


Figure 8. Normalized critical currents in a magnetic field of 15 T applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) as a function of neutron fluence.

Long-leg divertor aids heat flux reduction in HTS-ST

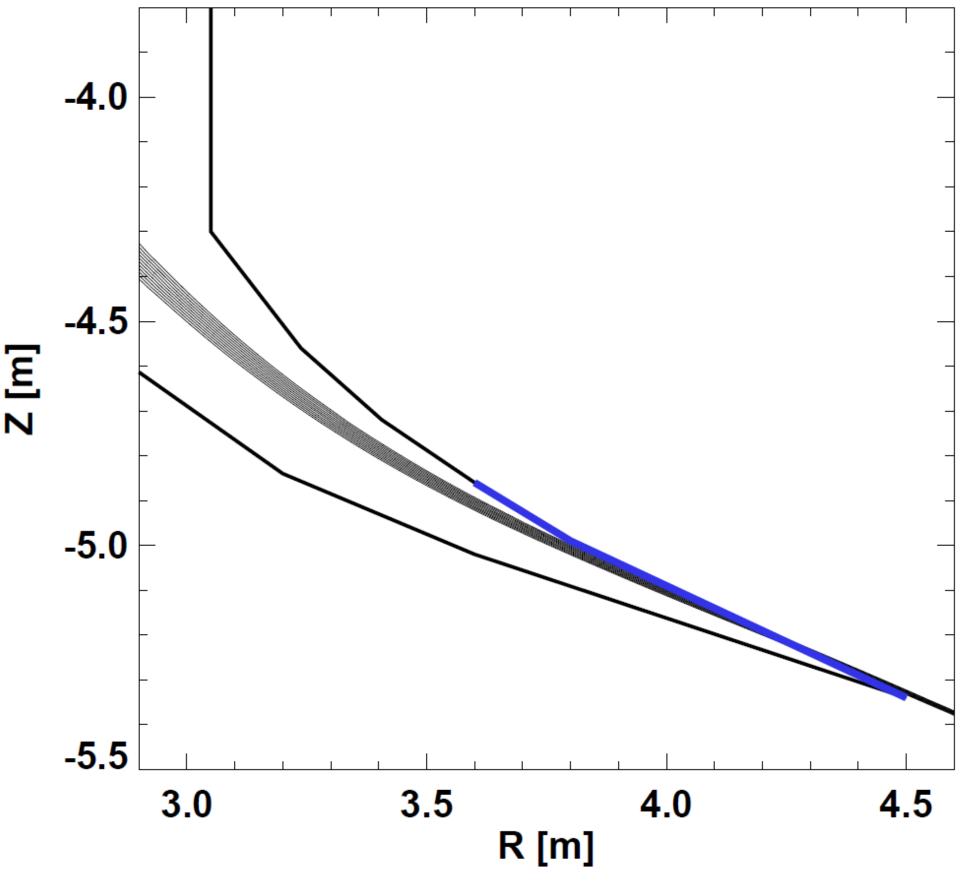


$\lambda_q \sim$ 1mm, assume $S \approx \lambda_q$ (closed divertor)
(T. Eich NF 2013)

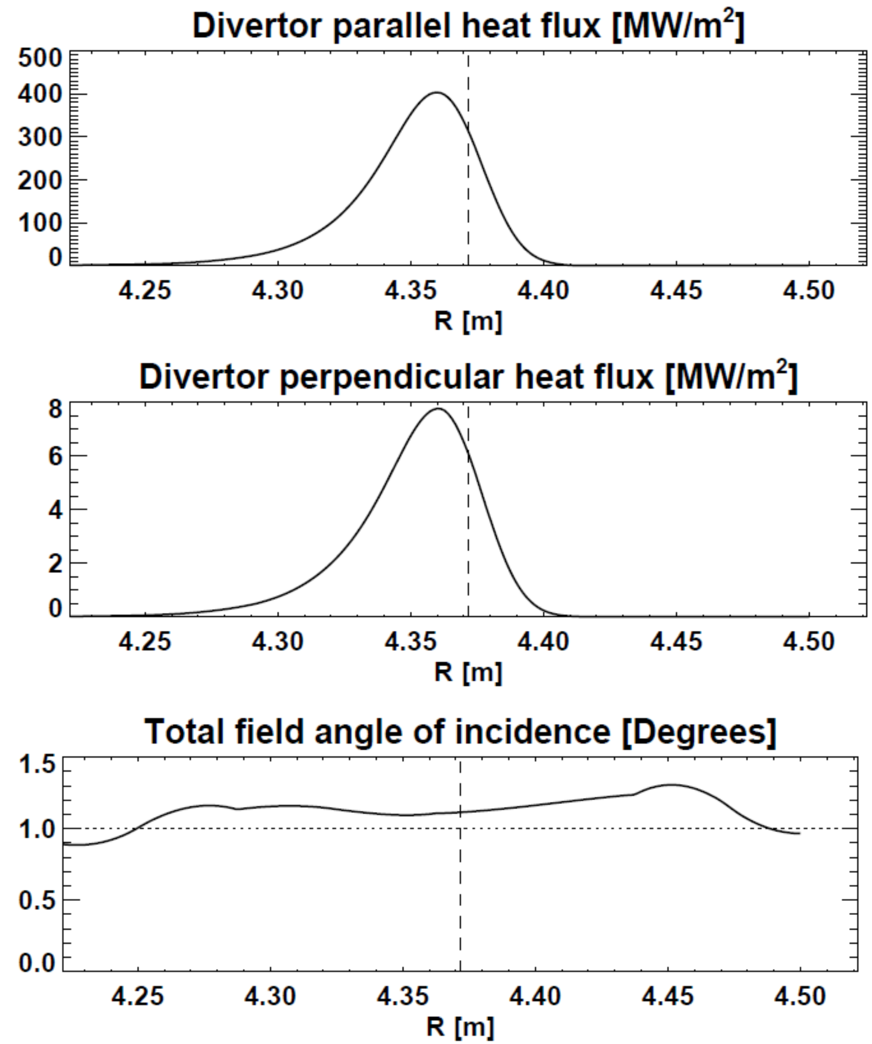


(Partial) detachment likely reduces peak $q_\perp$ by further factor of 2-4

Can also exhaust onto back of OB blanket
(like vertical target in conventional divertor)



λ_q ~ 1mm, assume S ≈ λ_q (closed divertor)
(T. Eich NF 2013)



(Partial) detachment likely reduces peak $q_\perp$ by further factor of 2-4

T. Brown radial build (includes small OH)

3.00 m R0 CCFE HTS ST radial build

10 TF coils | 2.00 AR

	COMP BUILD, Z=0 (in)	(mm)	(in)	(mm	TOTAL (mm)	TOTAL (in)	
	Machine Center				0		180
	TF center bore	68	2.677		68.0		
	OH coil	225.0	8.858	225	293.0	11.535	
	OH - TF gap	10	0.394		303.0		
TF inbd leg	Ext structure	190.0	7.480		493.0		0 needed to add to nose stu
	Clearance	2.00	0.079			230.000	
	ground wrap	4.00	0.157		499.0	19.646	
	Winding pack thk	240	9.449		729.0	28.701	
	ground wrap	4.00	0.157		733.0	28.858	
	Clearance	2.00	0.079				
	Ext structure	35.0	1.378	477	780.0	30.709	
	TF-OH TPT	2.0	0.079				
	VV TPT	5.0	0.197				
	wedge coil asmbly fit up	1.0	0.039				
Thermal Insu	Thermal Shield	8.0	0.315				
	Min TF/VV Gap	5.0	0.197	21	801	31.535	
inbd VV	VV shell thk	12	0.472				
	borated water/W shield	100	3.937				
	VV shell thk	12	0.472	124	925	36.417	
	WC inboard shield	500	19.685		1425	56.102	
	VV TPT	5.0	0.197				
FW	FW	30	1.181		1460	57.480	
	Plasma SO	40	1.575		1500	59.055	
	Plasma minor radii	1500	59.055				
Plasma R0					3000		

4.00	B0
16.5	Bmax
854	Ave Tresca stress (Mpa)

CALCULATIONS for OH at inner bore

$B_{oh} = \mu_o J_o (Ro - Ri)$ where $\mu_o = 4\pi \times 10^{-7}$

Assume J_o (MA/m^2) 70

B (T) = 19.8

OH stress = $B_i R_i J_o / 2$

stress = 47.11 Mpa

OH flux = B x mean area of xolenoid

flux = 2.03 volt seconds (weber)

Added space between OH and TF

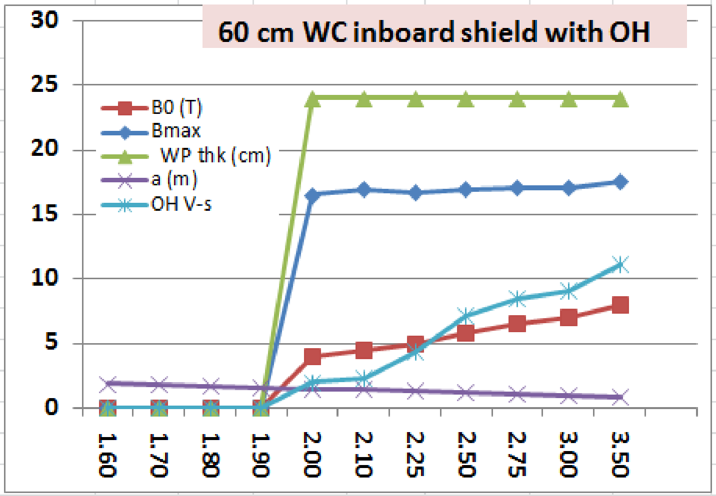
60 cm WC uses VV shield and shield in front of it.

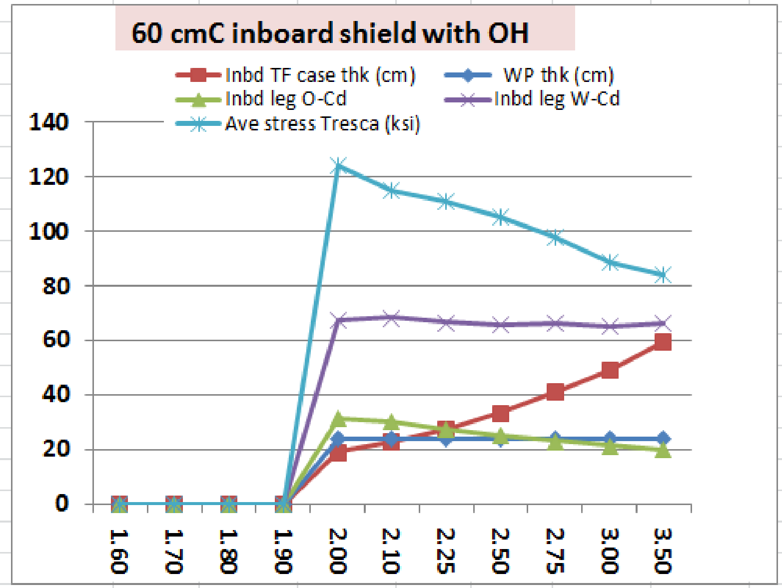
For AR 2 device:

OH located inside of TF bore: 854 Mpa Tresca stress with 4.0 B0 and 176.5 TF Bmax

OH flux: 2 Vs with 70 Jc solenoid and 19.8 T

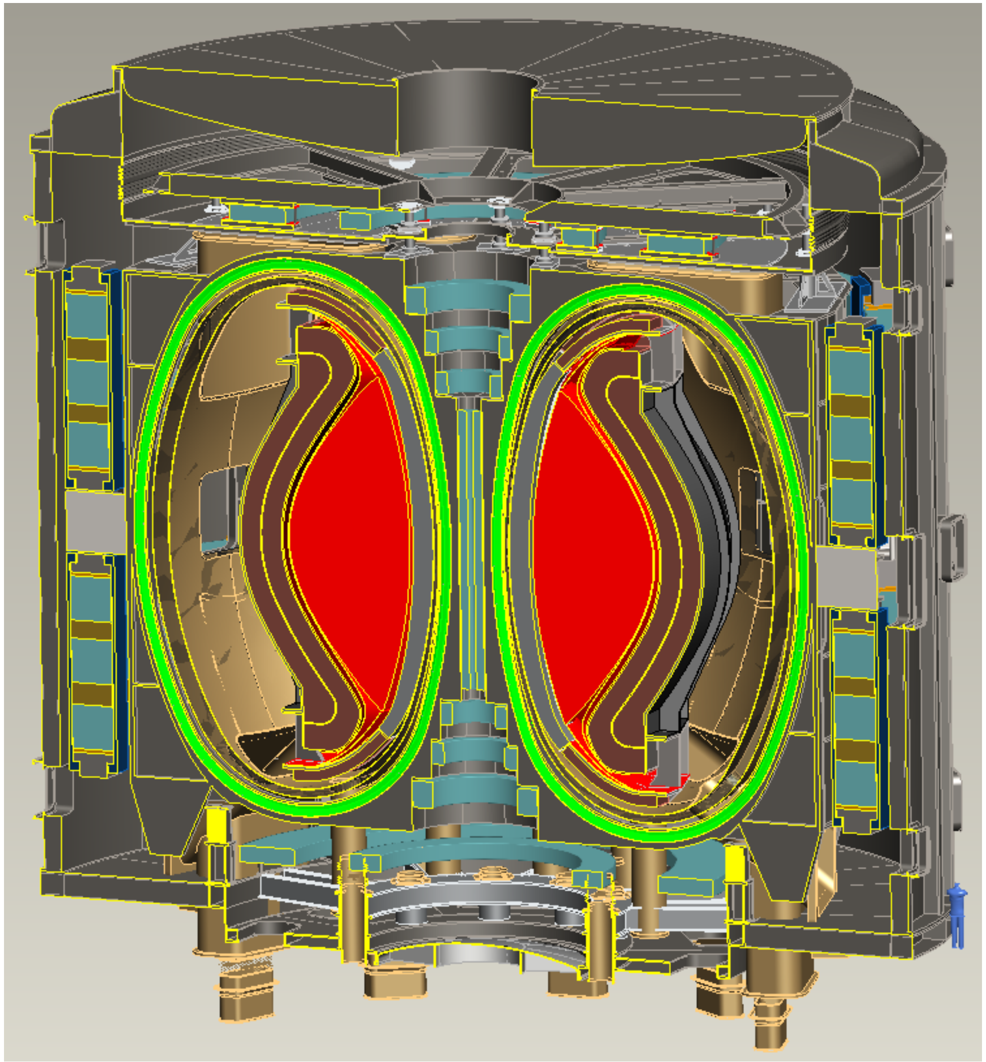
TF and OH magnet parameters vs. aspect ratio using models from Brown and Zhai



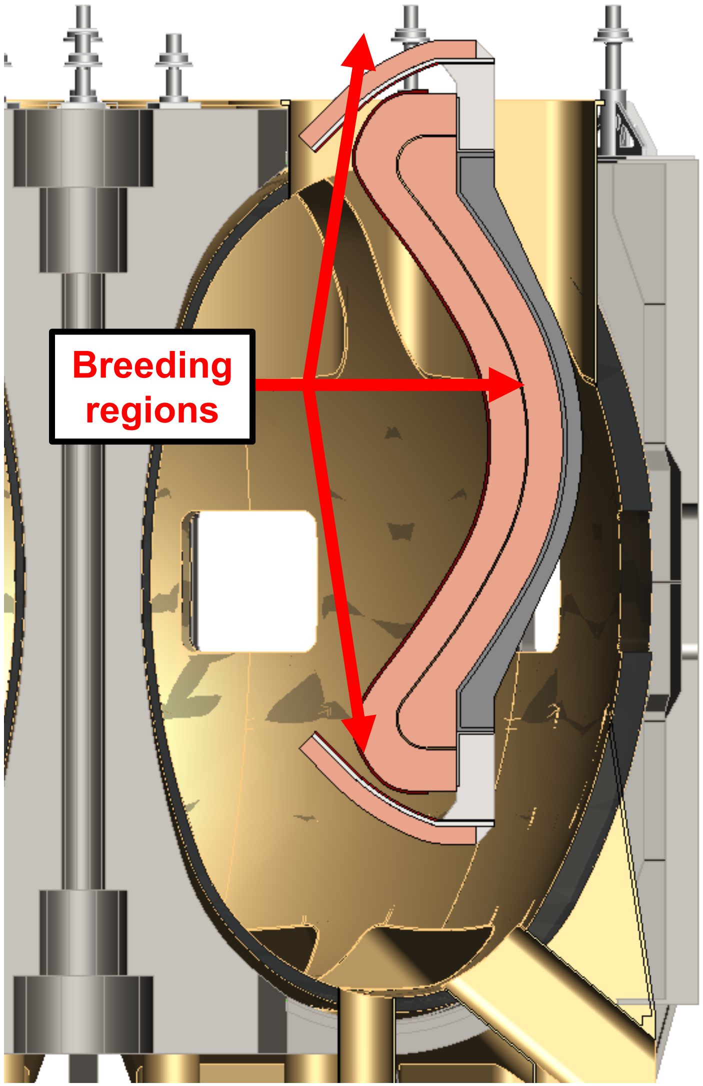


WC thk (cm)	R0 (m)	AR	ε= 1/AR	a (m)	B0 (T)	WP thk (cm)	Inbd TF case thk (cm)	Inbd leg O-Cd (MA/m^2)	Inbd leg W-Cd (MA/m^2	Bmax	Ave stress Tresca (Mpa)	Ave stress Tresca (ksi)	OH IR (cm)	OH thk (cm)	OH B (T)	OH V-s
60.00	**3.0**	1.60	0.63	1.875												
		1.70	0.59	1.765												
		1.80	0.56	1.667												
		1.90	0.53	1.579												
		2.00	0.50	1.500	4.00	24	19.0	31.5	67.6	16.50	854	124	6.8	22.5	19.8	2.0
		2.10	0.48	1.429	4.50	24	22.5	29.9	68.2	16.90	791	115	7.9	22.5	19.8	2.3
		2.25	0.44	1.333	5.00	24	27.5	27.3	66.5	16.70	764	111	15.0	22.5	19.8	4.3
		2.50	0.40	1.200	5.80	24	33.3	24.8	65.9	16.90	725	105	22.5	22.5	19.8	7.1
		2.75	0.36	1.091	6.50	24	41.0	23	66.0	17.10	672	97	25.7	22.5	19.8	8.5
		3.00	0.33	1.000	7.00	24	48.8	21.3	65.2	17.10	609	88	27.0	22.5	19.8	9.1
		3.50	0.29	0.857	8.00	24	59.1	19.8	66.1	17.50	580	84	31.0	22.5	19.8	11.1

Latest HTS-ST: R=3m, A=2, P_{fusion}~ 500MW, Q_{eng}~1-1.5



OB blanket module with divertor + extended blanket at top/bottom fits through vertical ports



- Potential advantages of this low-A configuration:
  - Reduced part count + no inboard breeding → simplified maintenance (?)
- But, very likely need to breed at top + bottom (included)
  - UW will assess TBR for DCLL
  - Analysis similar to Cu ST-FNSF
- Initiating study of LM/Li wall and divertor compatibility with this HTS configuration