Confinement and stability discoveries from high-beta spherical tori

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Program Director for NSTX Upgrade

With contributions from the NSTX, MAST
Pegasus, and other ST research teams

Norman Rostoker Memorial Symposium
The Fairmont Newport Beach Hotel
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U.S. DEPARTMENT OF ENERGY

PRINCETON UNIVERSITY

PPPL
PRINCETON PLASMA PHYSICS LABORATORY

Outline

- **ST overview**
- **Global MHD Stability**
- **Energy Confinement**
- **Energetic Particles**
- **ST Upgrade Status**
- **Summary**

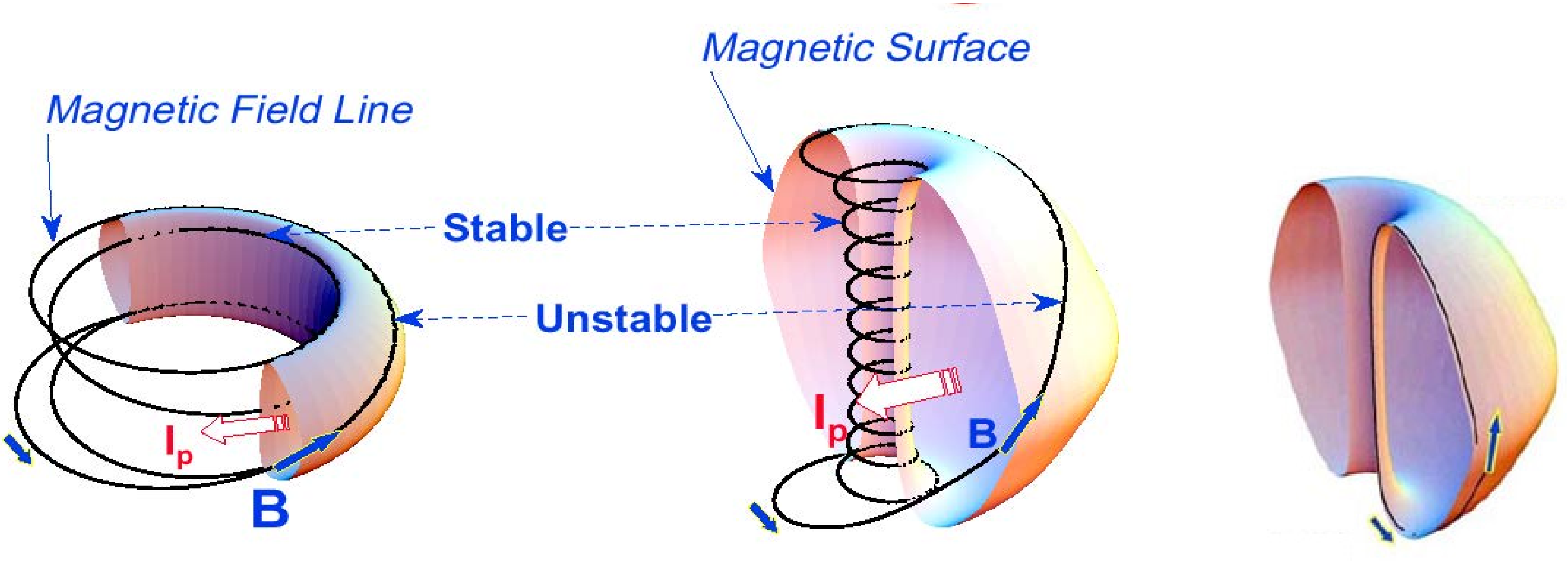
"Spherical" tokamak (ST) has aspect ratio A < 2

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



Tokamak

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

ST

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

FRC

$A \sim 1$,
$\kappa = 0.5\text{-}10$,
$q \sim 0$,
$\beta \sim 100\%$

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

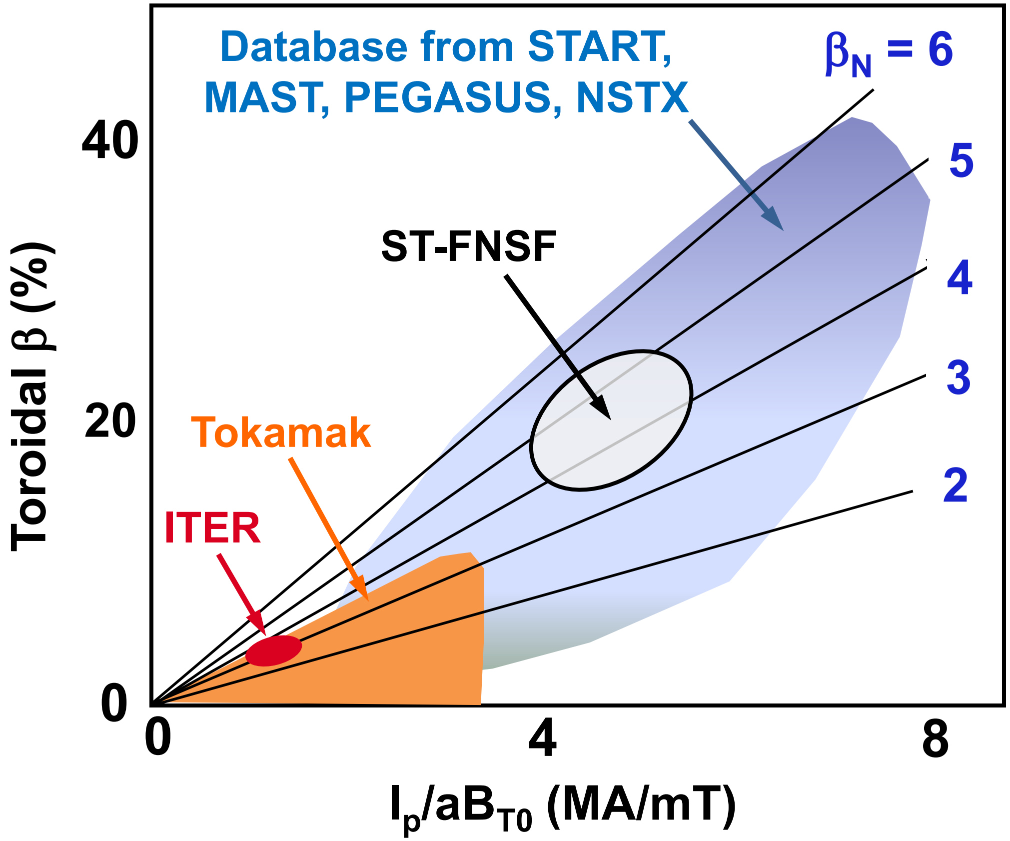
$$P_{fusion} \propto \langle p \rangle^2 \times Vol$$

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

Physics limited

$

Investment & engineering limited



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

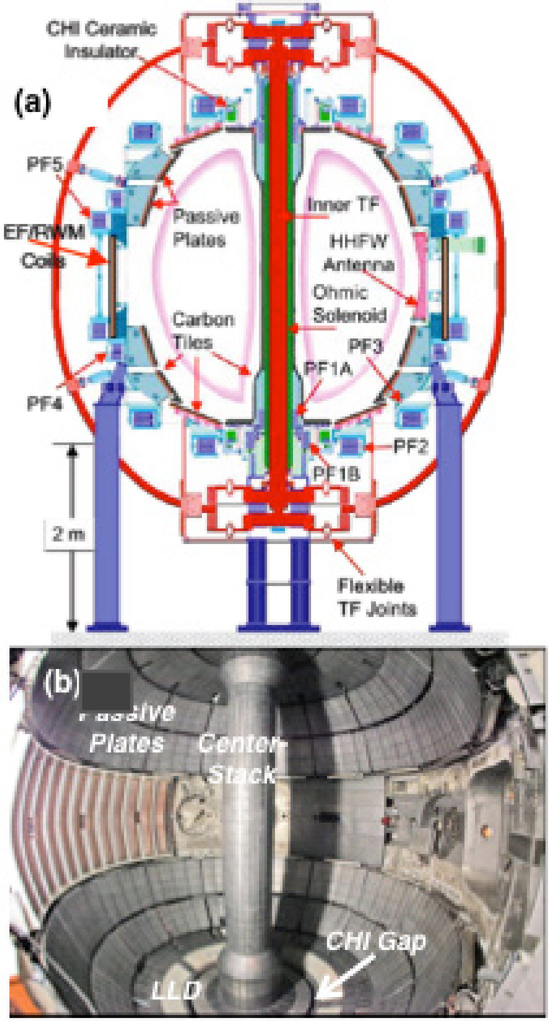
$$W_n \propto \beta_T^2 B_{T0}^4 a$$ (not strongly size dependent)

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

MA-Class ST Research Started ~2000

Complementary Physics Capabilities of NSTX and MAST

NSTX



M. Ono, IAEA 2000, NF 2001

Complementary Capabilities

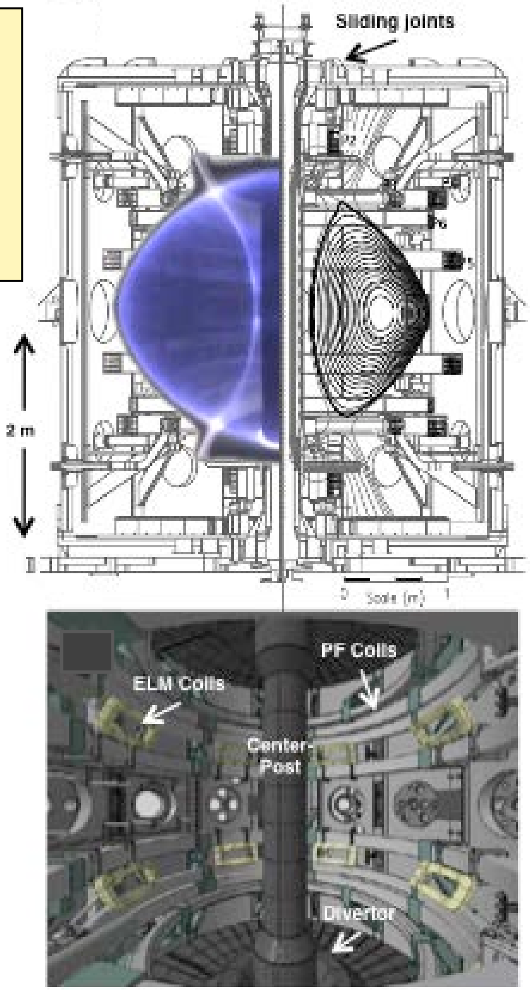
Passive Plates	Large divertor volume
Helicity Injection	Merging/Compression
Fast wave heating	Electron Cyclotron
1 x 6 RWM Coils	2 x 12 ELM coils

Similar Capabilities

NSTX	MAST
R = 85 cm	R = 80 cm
$A \geq 1.3$	$A \geq 1.3$
κ = 1.7 - 3.0	κ = 1.7 – 2.5
B_T = 5.5 kG	B_T ~ 5.0 kG
$I_p \leq 1.5$ MA	$I_p \leq 1.5$ MA
$V_p \leq 14$ m^3	$V_p \leq 10$ m^3
P_{NBI} = 7.4 MW	P_{NBI} = 4.0 MW

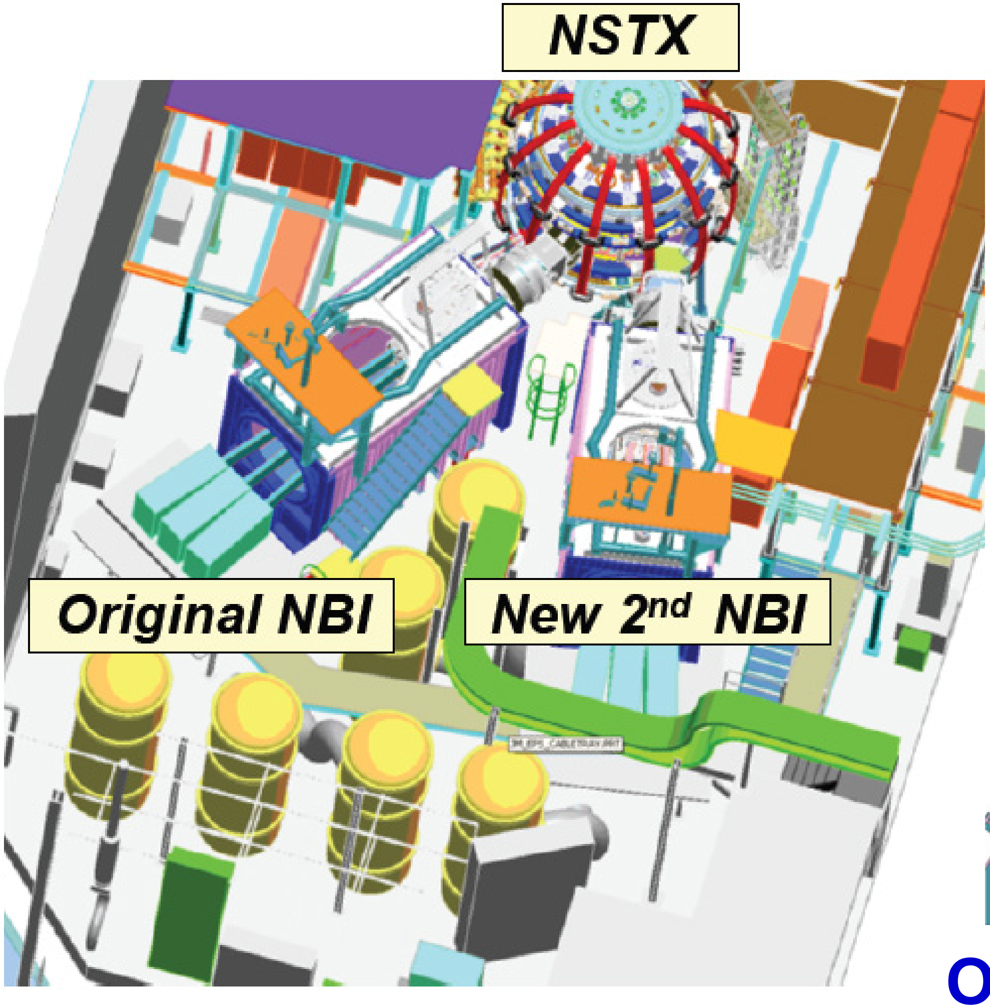
- Comprehensive diagnostics
- Physics integration
- Scenario development

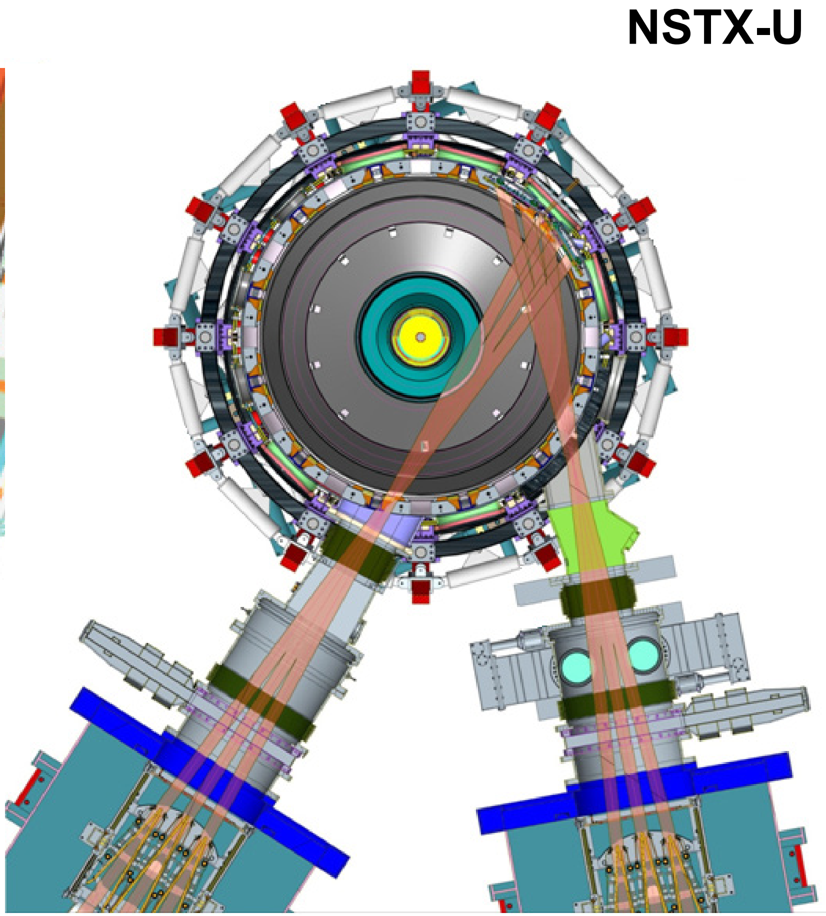
MAST



A. Sykes, IAEA 2000, NF 2001

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

New physics accessed in ST → enhanced understanding of toroidal confinement physics

- Lower A → higher β, strong shaping
- Higher β →
  - Electromagnetic effects in turbulence
  - More potential drive for fast-ion-driven instabilities
    - Simulate fast-ion transport of ITER / burning plasmas
  - Over-dense plasmas: RF heating, current drive
- Low-A / high-β broadly impact transport, stability:
  - Higher fraction of trapped particles (low A)
  - Increased normalized orbit size (high β)
  - Increase flow shear (due to low B, low A)
- Compact geometry (small R) → higher power and particle fluxes relevant to ITER, reactors

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Simulations find $\langle \beta_N \rangle$ is more aspect ratio invariant than β_N – both with & without wall stabilization

Para / diamagnetic effects and B_P / B ratio important at low-A

$$\beta_T \equiv 2\mu_0 \langle p \rangle / B_{T0}^2$$

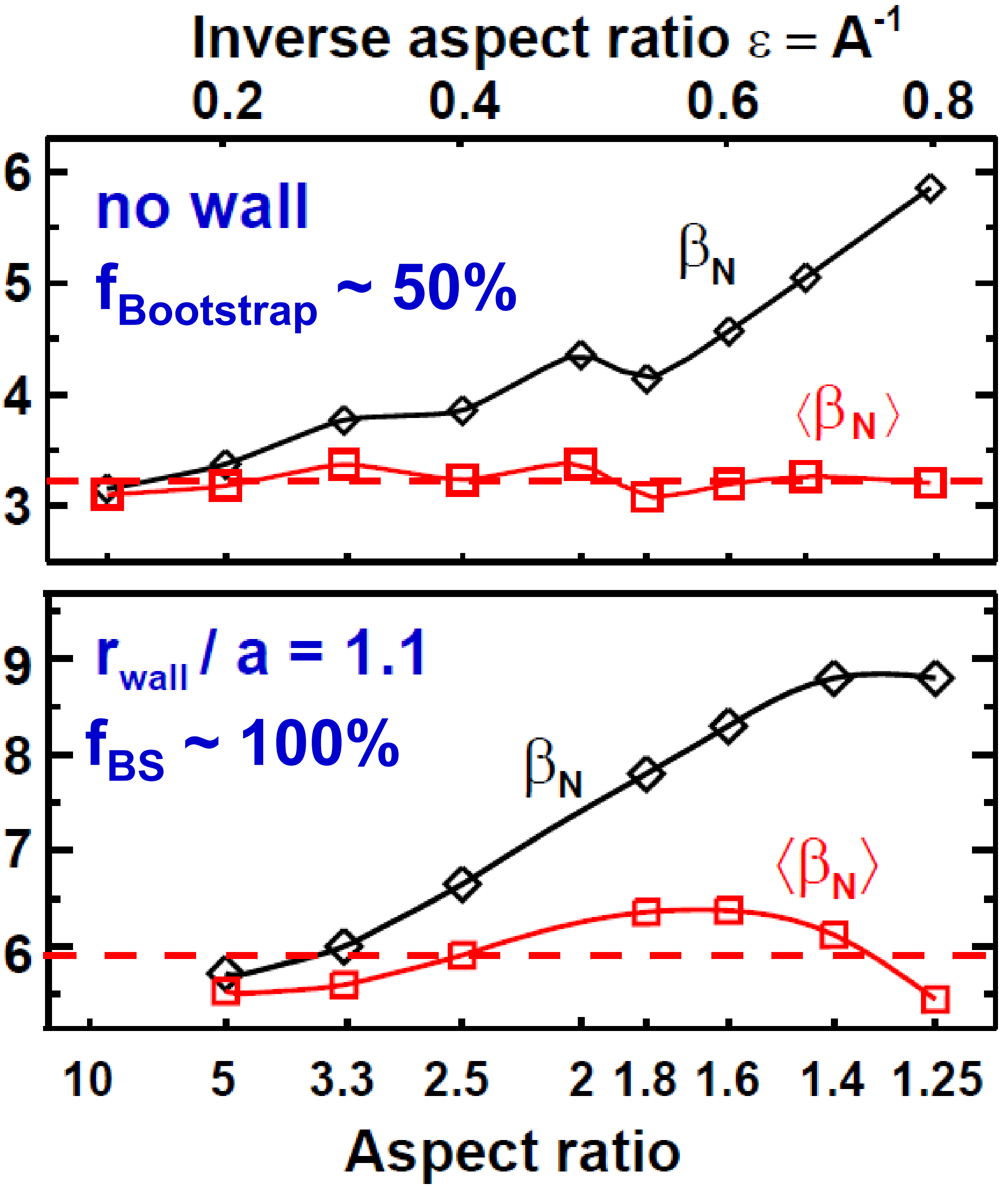
$$\langle \beta \rangle \equiv 2\mu_0 \langle p \rangle / \langle B^2 \rangle$$

$$I_N \equiv I_P / aB_{T0} \ [\mathrm{MA/mT}]$$

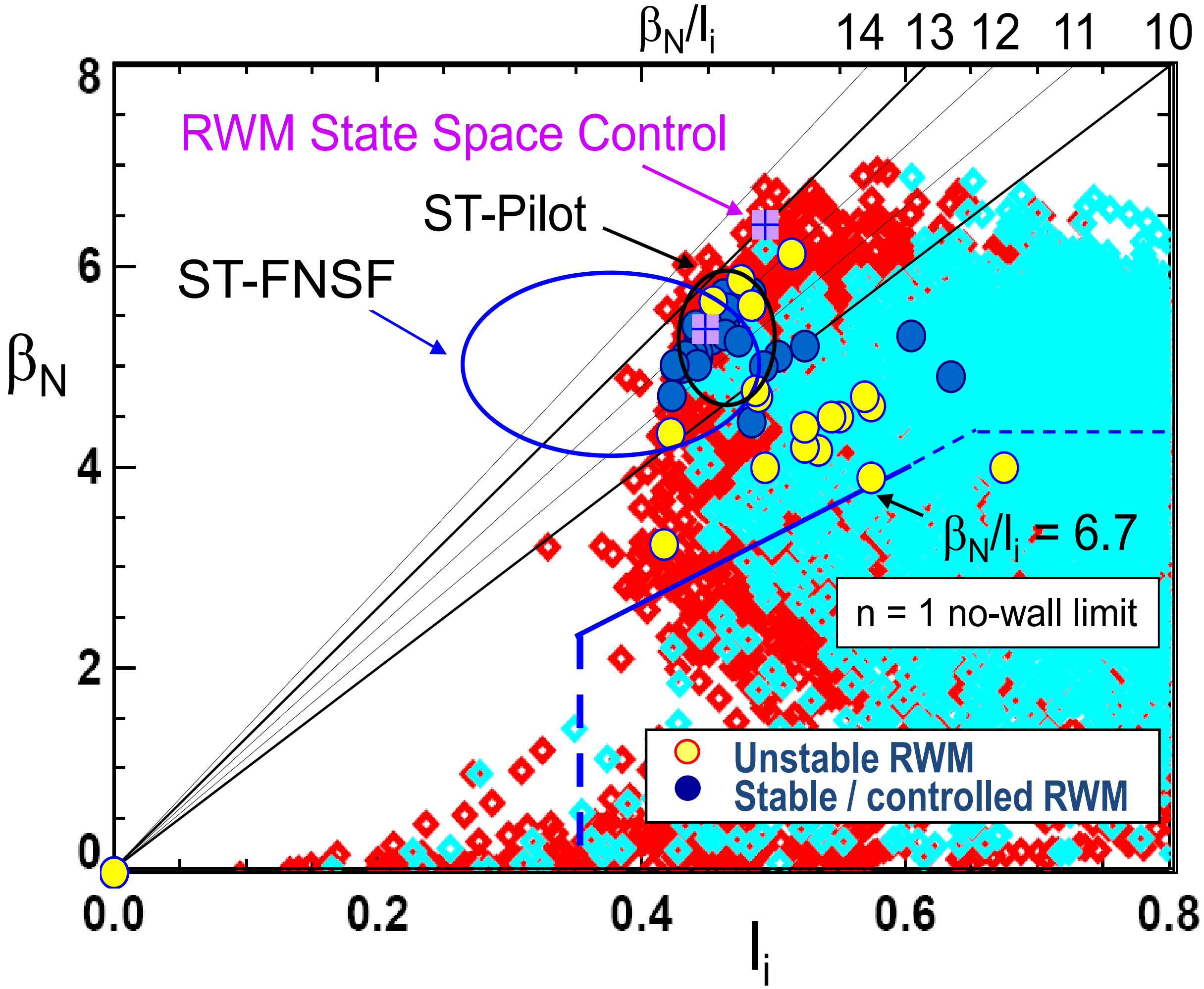
$$\beta_N \equiv \beta_T (\%) / I_N$$

$$\langle \beta_N \rangle \equiv \langle \beta \rangle (\%) / I_N$$

J. Menard, PoP 2004, PPPL-3779



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

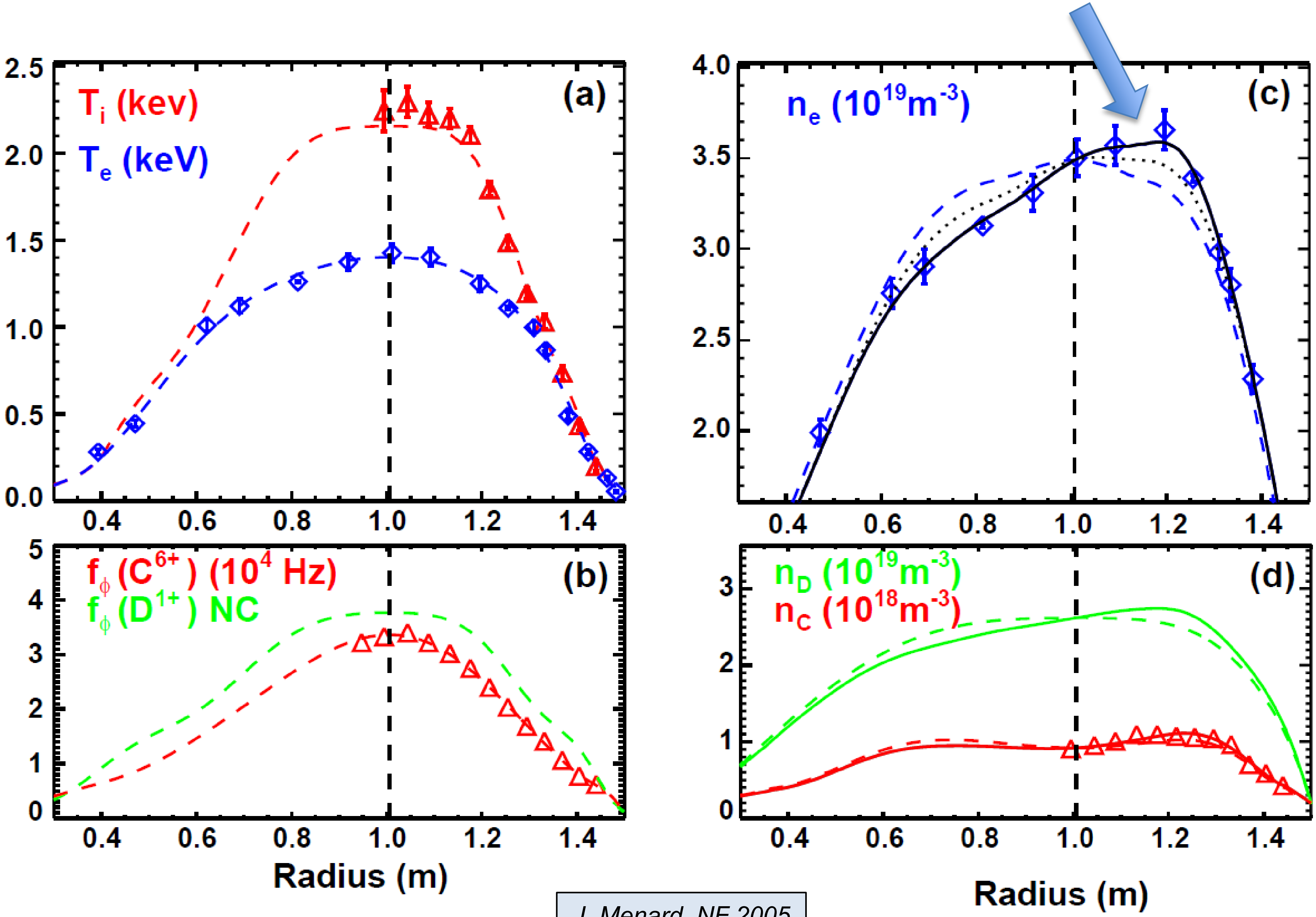


- High β_N regime is important for bootstrap current generation.
- High β_N / l_i regime important since high f_{BS} regime has low l_i.

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

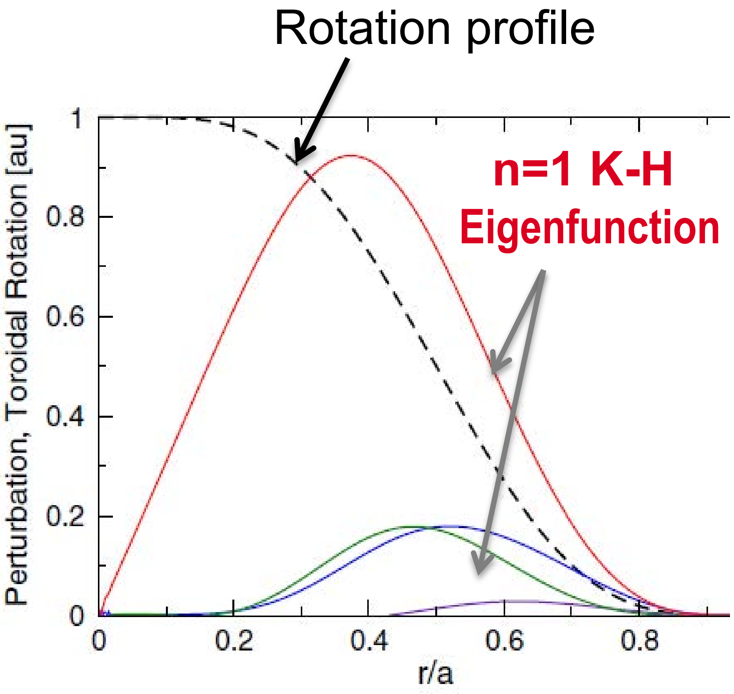
Major NSTX-U mission is to achieve fully non-inductive operation at high β

Rotation / centrifugal effects important and measureable in equilibrium



J. Menard, NF 2005

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



Rotation profile

n=1 K-H Eigenfunction

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

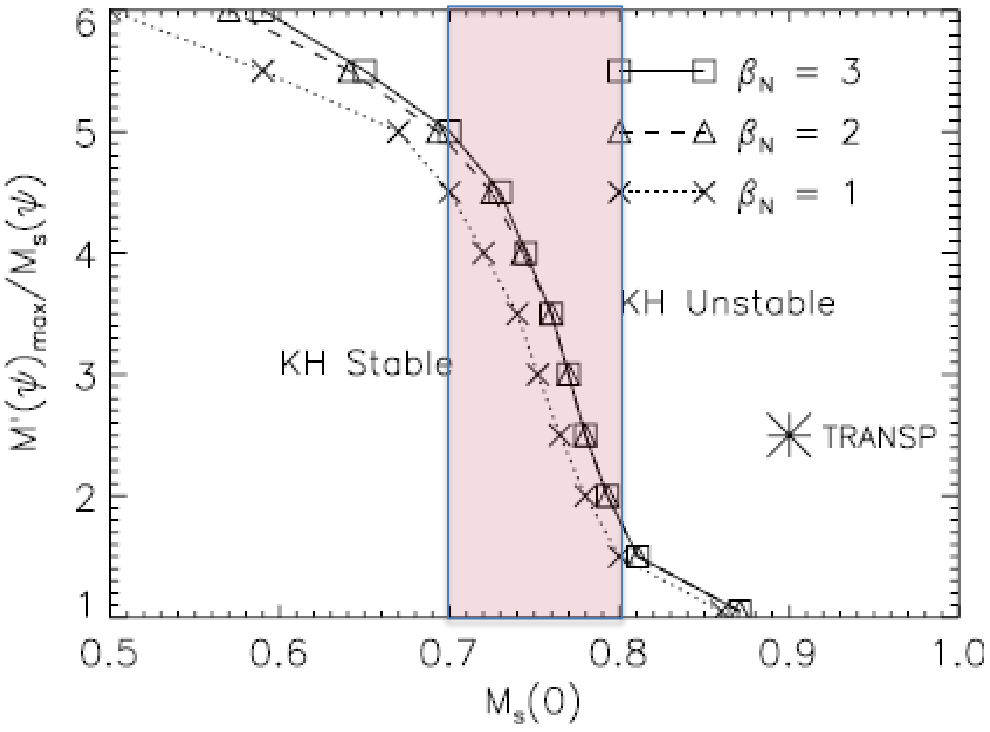
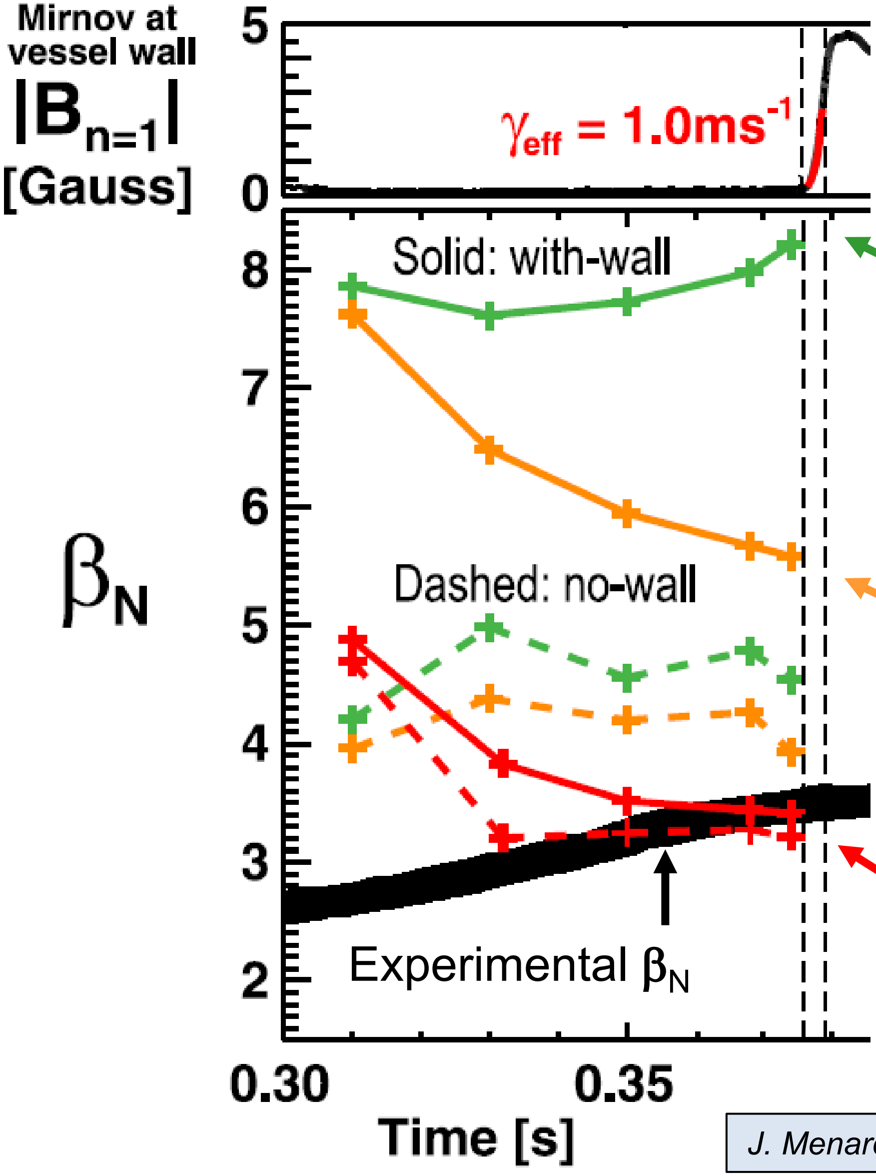


Figure 5. The $KH_{\|}$ stability boundary in terms of flow speed and gradient for three different plasma pressures when the safety factor is fixed at $q_{min} \simeq 1.3$. A typical rotation profile from TRANSP predictions is shown for reference, indicating that CTF with fully uni-directional beams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

I. Chapman, NF 2012

Hybrid MHD-drift-kinetic stability calculations find rotation + fast-ions can weaken wall-stabilization of ∇p-driven kink



- MARS-K code needed / used to explain NSTX instability onset at highest rotation, β_{fast} fractions

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

J. Menard, PRL 2014

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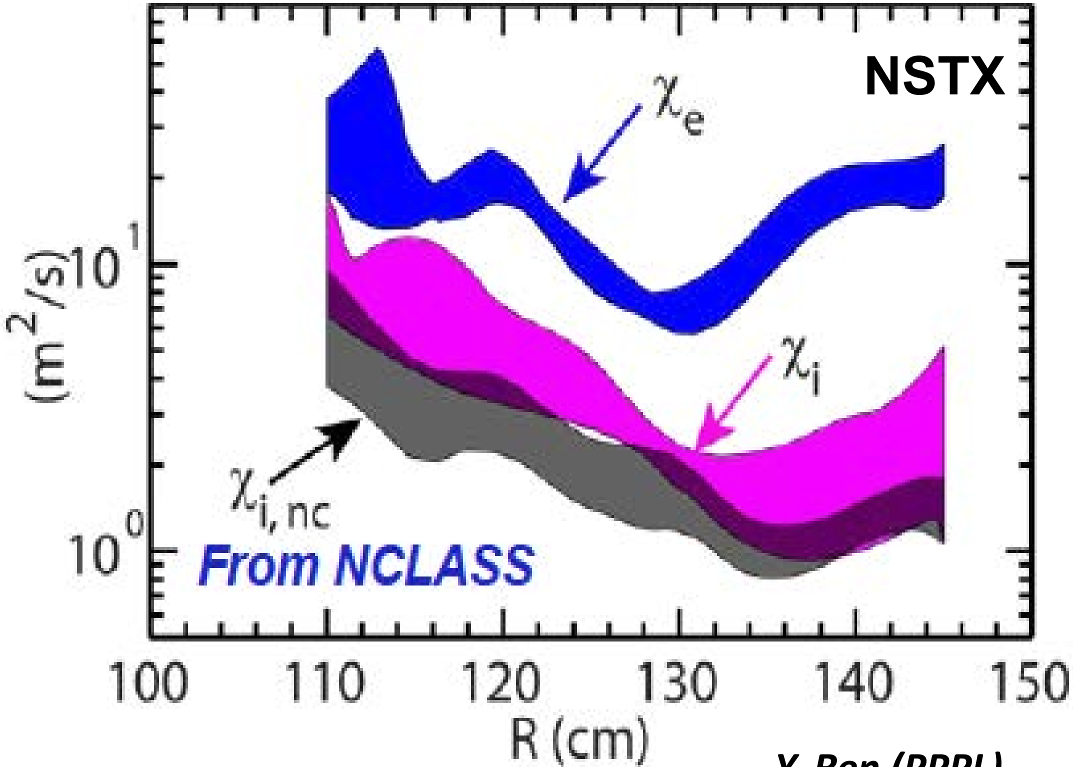
High confinement multiplier H needed for compact ST

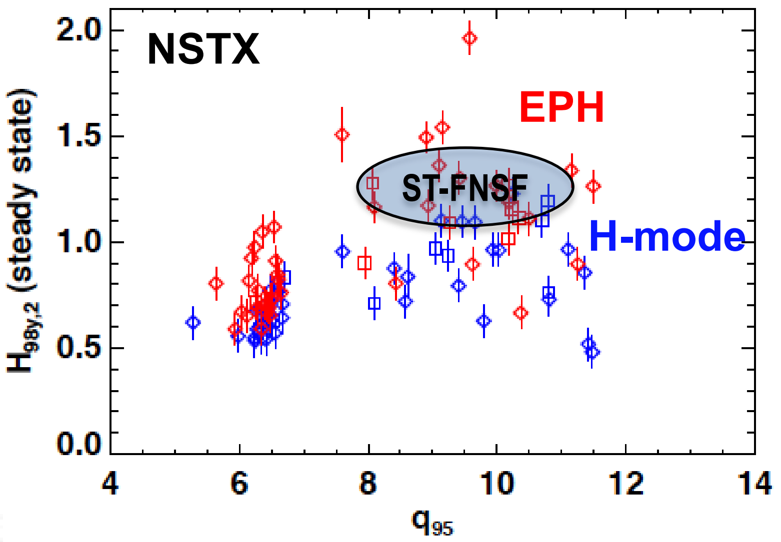
Fusion gain Q depends strongly on “H”, $Q \propto H^{5\text{-}7}$
H = 1.2 – 1.3 enables compact FNSF, design flexibility/margin

- 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)

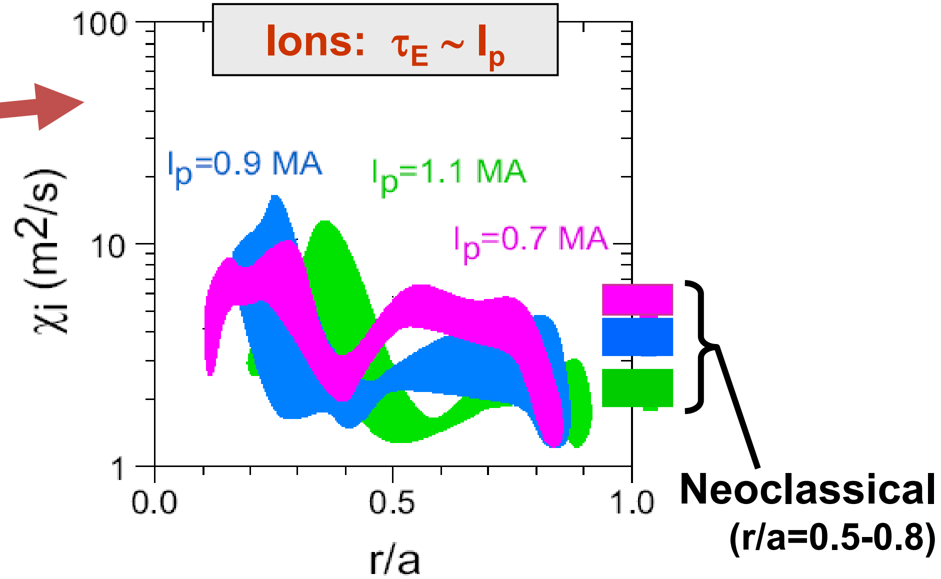
Enhanced pedestal H-mode (EPH) has H up to 1.5-2 → attractive for ST-FNSF



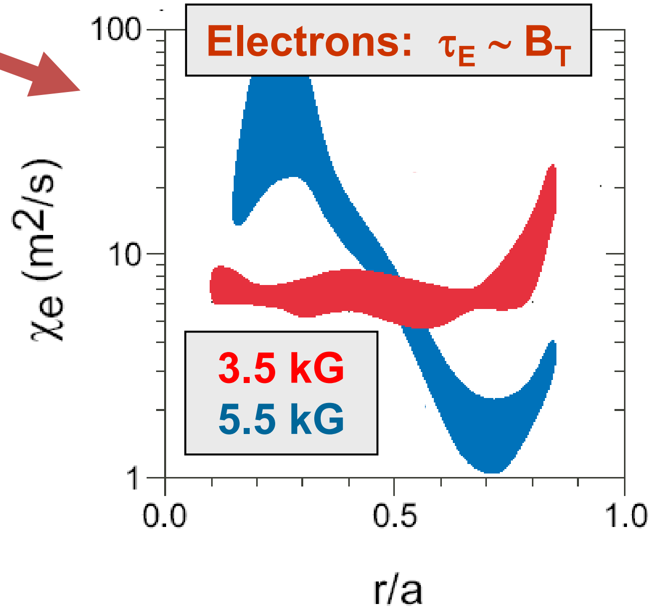
R. Maingi, PRL 2010
S. Gerhardt NF 2014

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



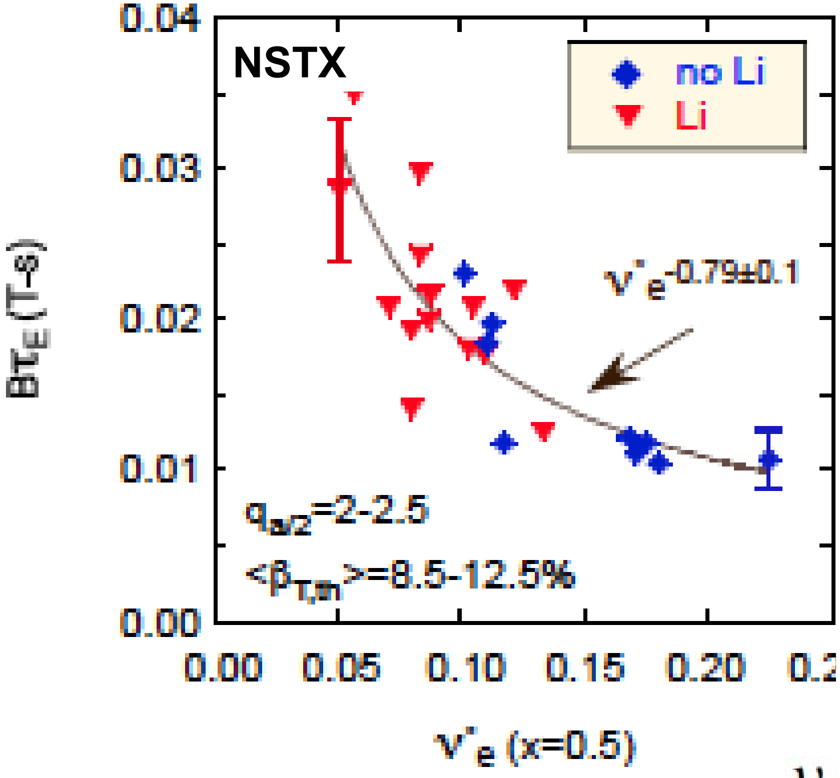
S.M. Kaye, PRL 2007

Favorable confinement trend with collisionality, β found

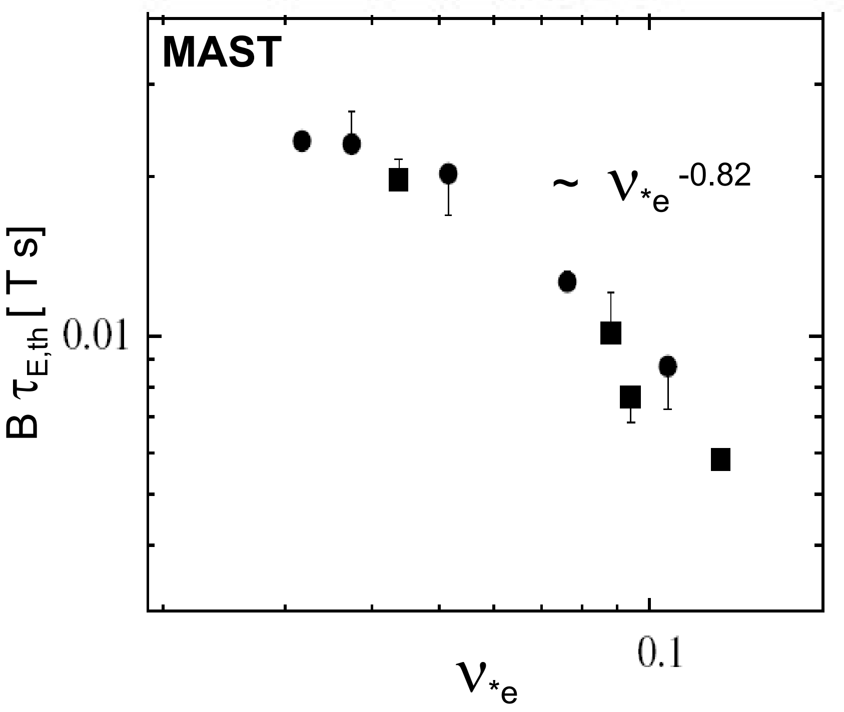
Important implications for future ST FNSF, Demo with lower ν_*

$\tau_{E,\,th} \propto \nu_{*e}^{-0.1}\ \beta^{-0.9}$ **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



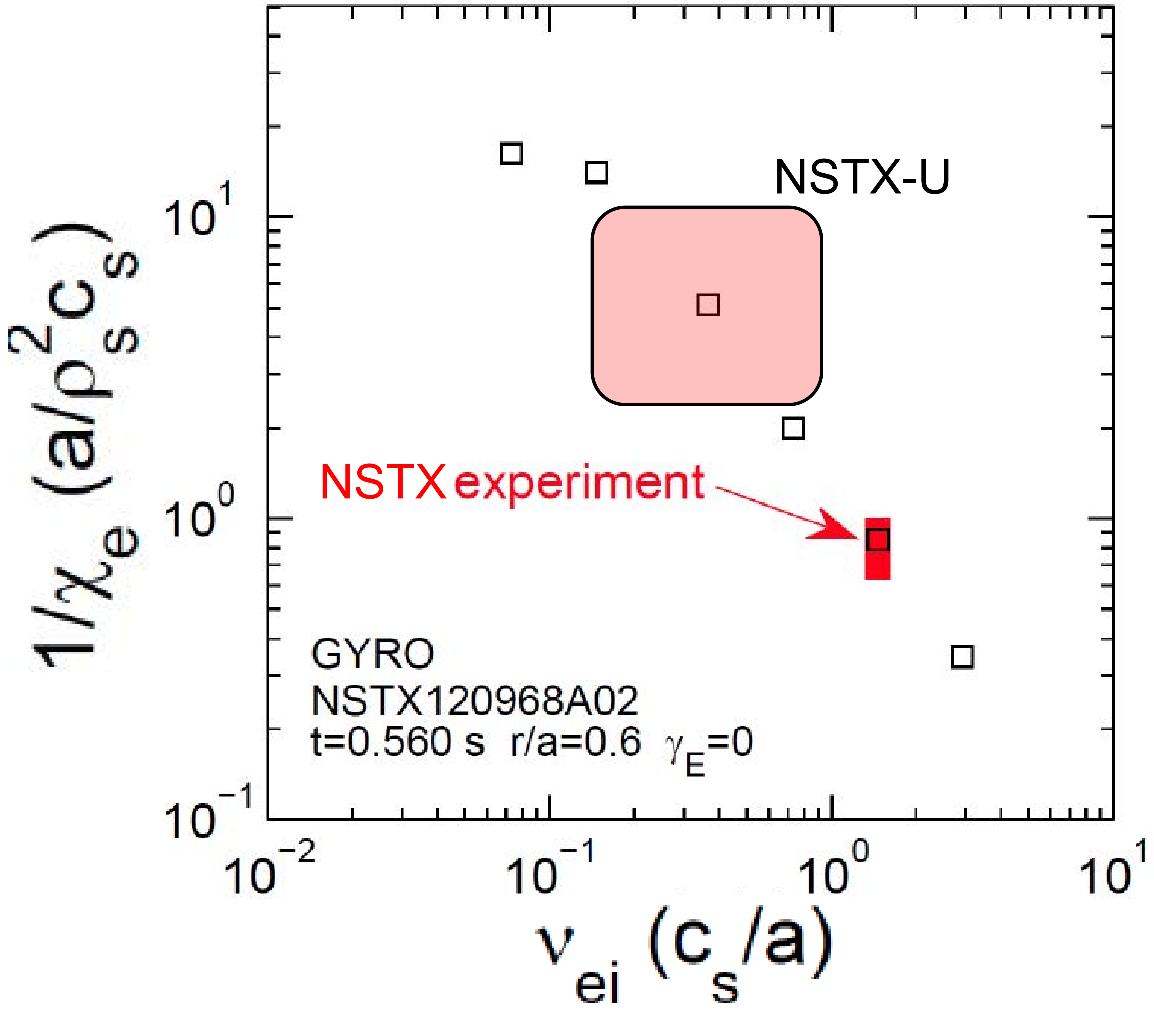
$$\nu_* \propto \bar{n}_e / T^2$$

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

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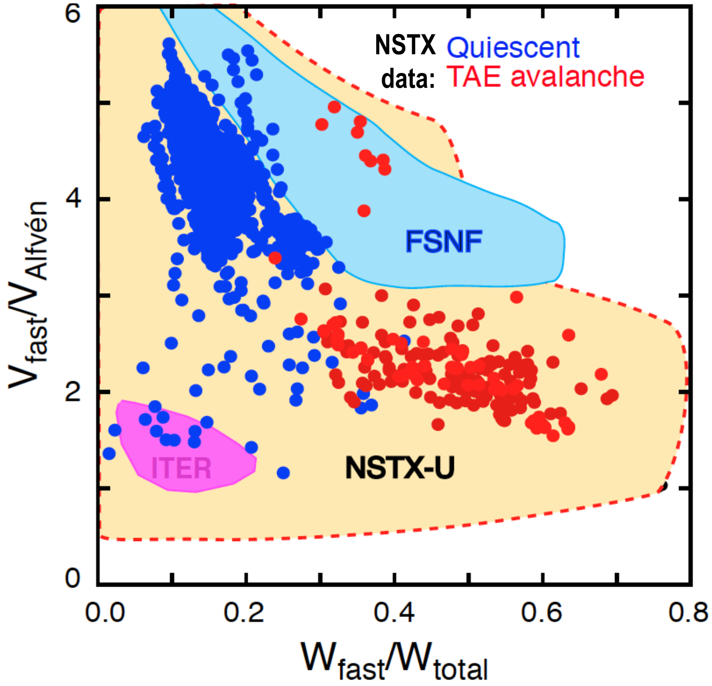
NBI-heated STs excellent testbed for α-particle physics

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

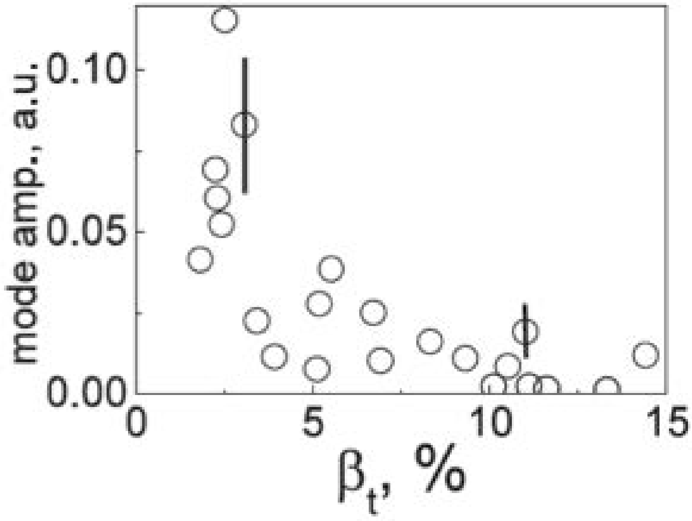
EP parameter space



E. Fredrickson, NF 2013

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

Stabilization of TAEs at high β in MAST

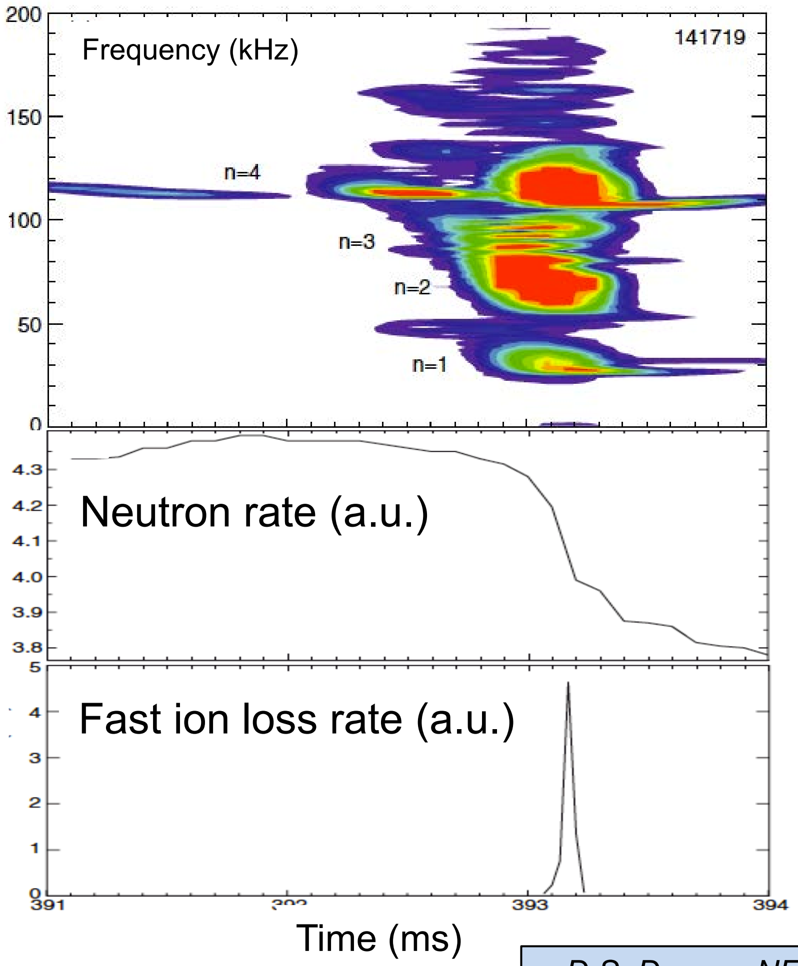


M.P. Gryaznevich, PPCF 2004

“TAE avalanche” shown to cause energetic particle loss

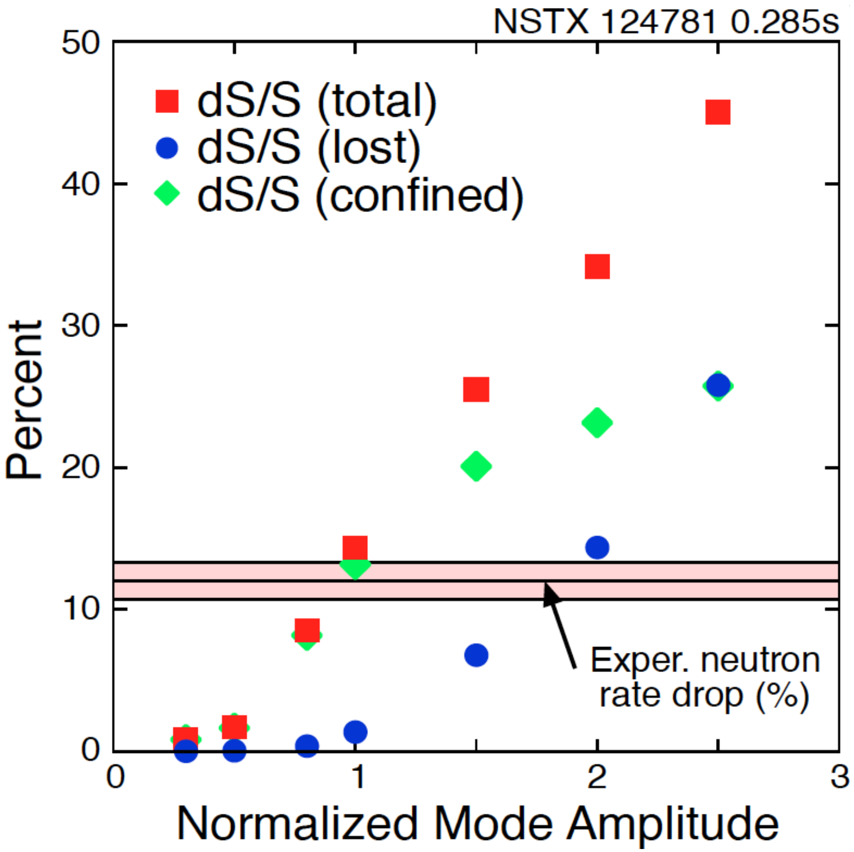
Uncontrolled α-particle loss could cause reactor first wall damage

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



D.S. Darrow, NF (2013)

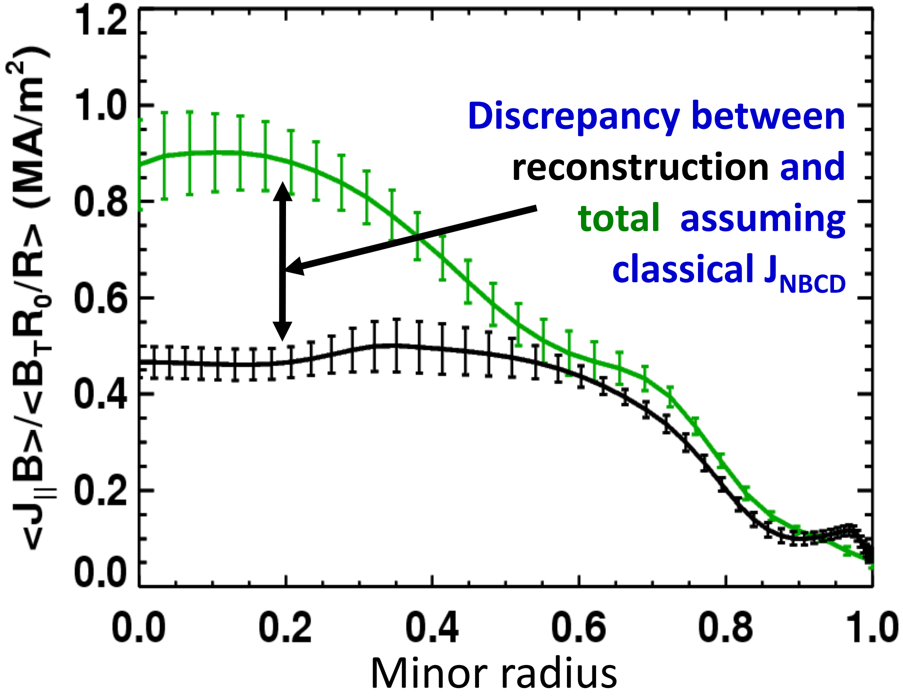
Progress in simulation of neutron rate drop due to TAE avalanche



E. Fredrickson, NF 2013

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

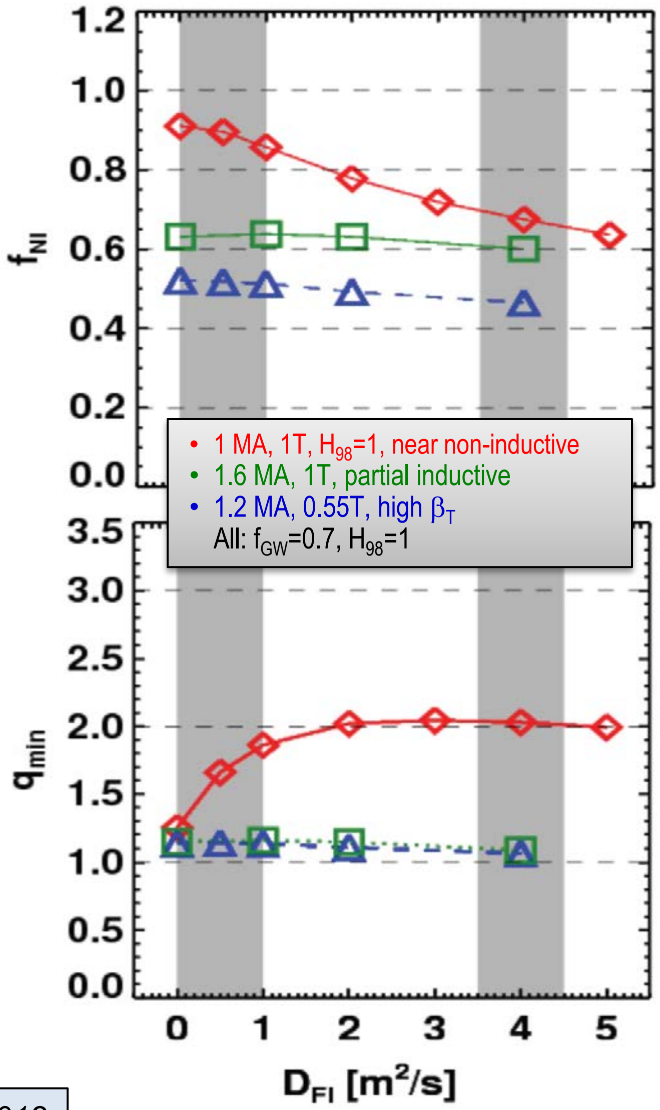
NSTX: rapid avalanches can lead to redistribution/loss of NBI current drive



Discrepancy between reconstruction and total assuming classical J_{NBCD}

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

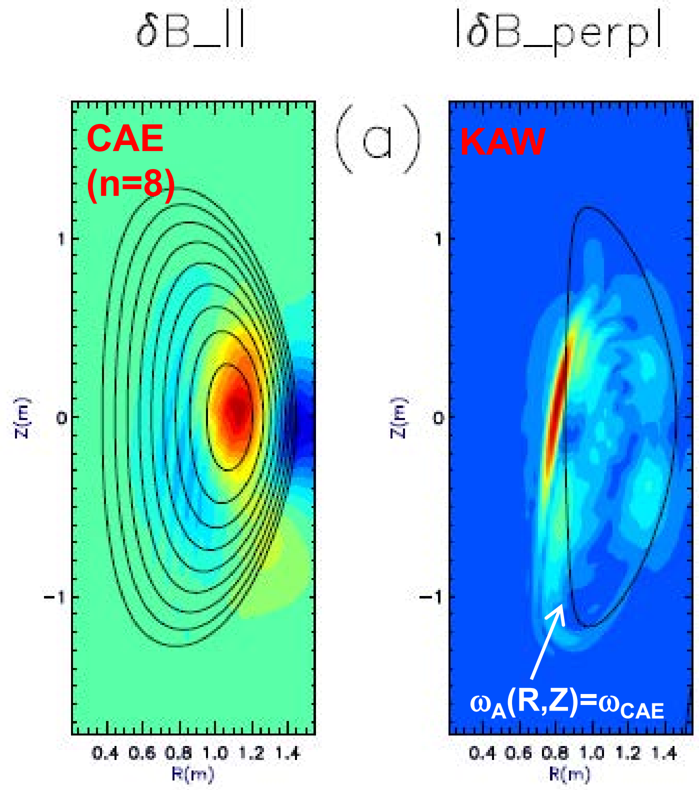
NSTX-U TRANSP simulations



S. Gerhardt NF 2011, NF 2012

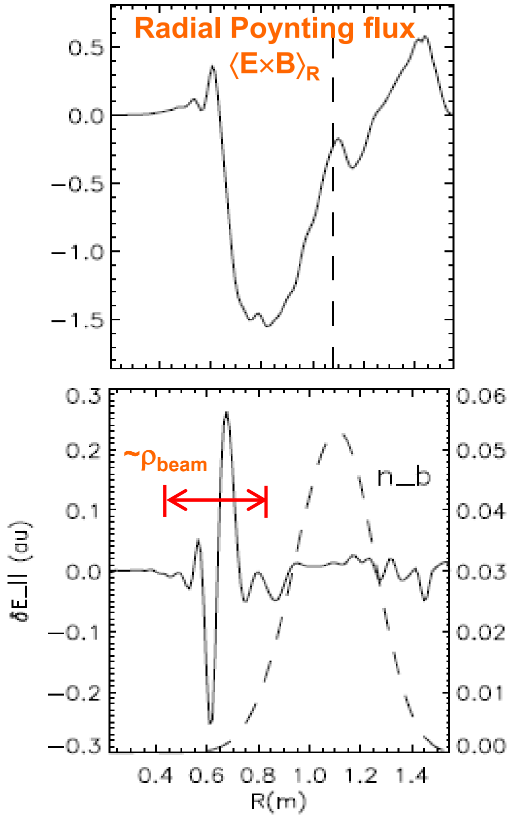
CAE mode-conversion to kinetic Alfvén waves (KAW) predicted to transfer core NBI power to mid-ρ electrons

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
   - $P_{CAE\rightarrow KAW}$ ~ **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



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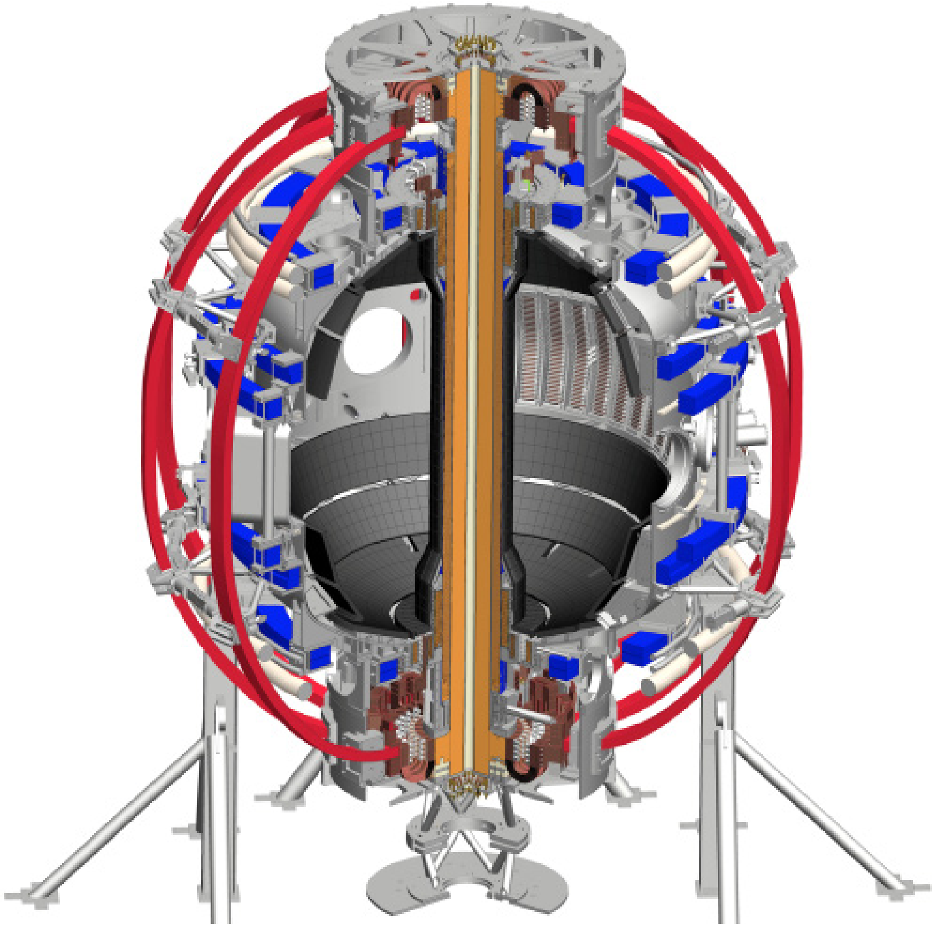
NSTX and MAST are undergoing major upgrades

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

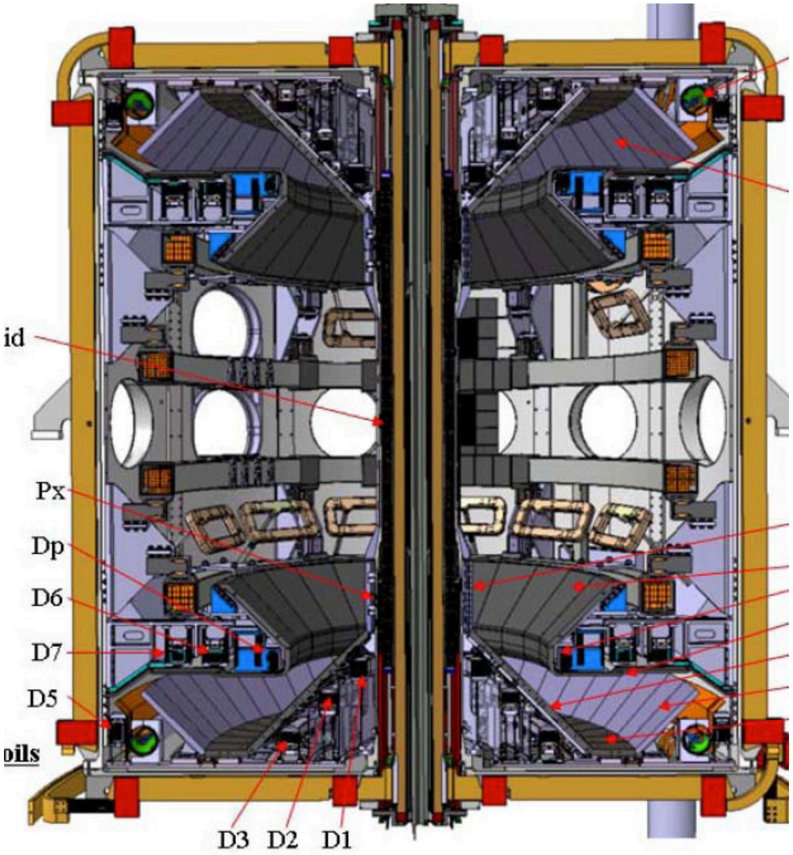
NSTX-U



Highly tangential 2nd NBI for non-inductive sustainment, profile control

First test plasma few weeks ago

MAST-U

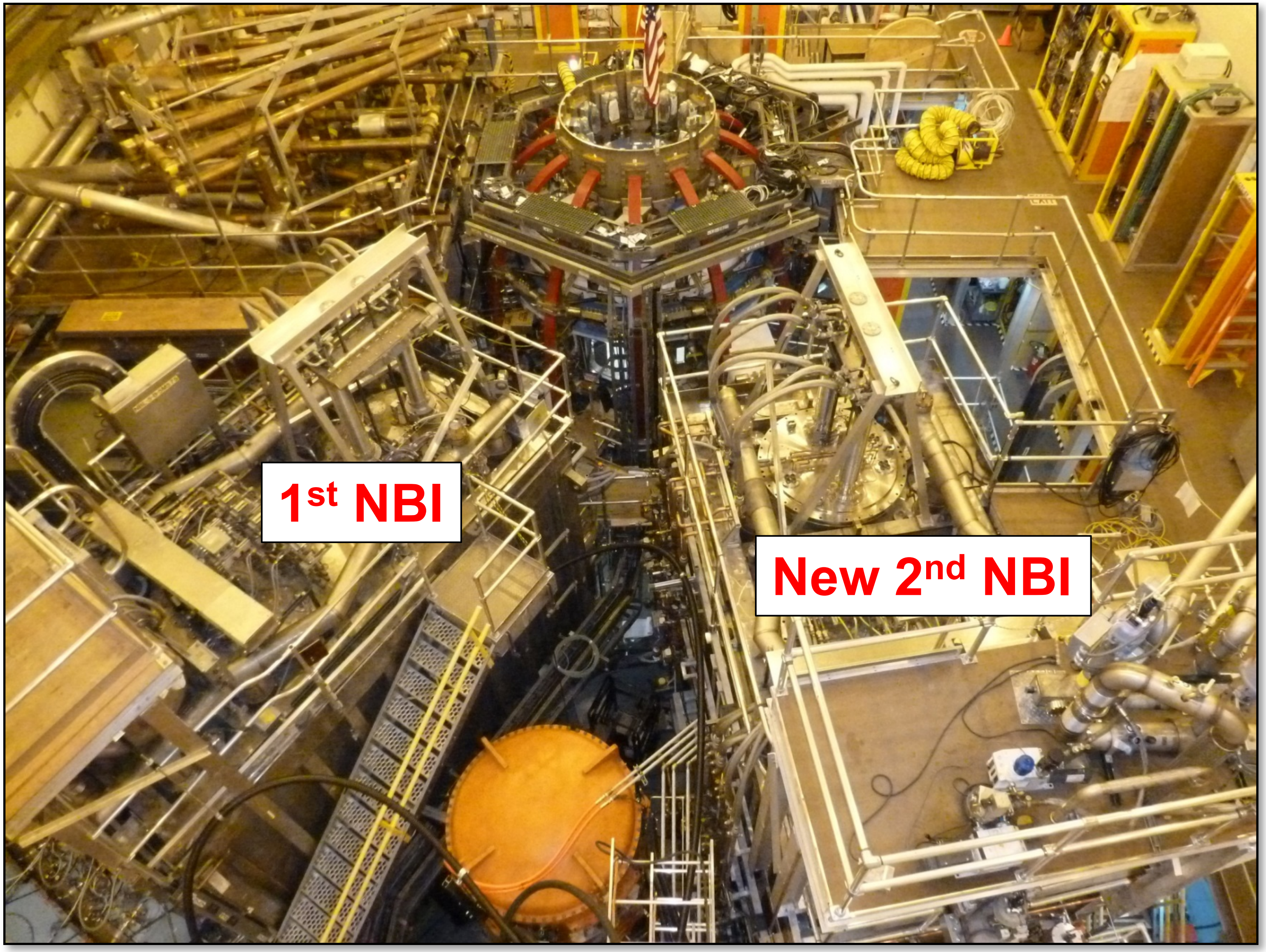


Super-X divertor configuration for FNSF/DEMO divertor solution

First test plasma 2017

NSTX Upgrade project recently completed

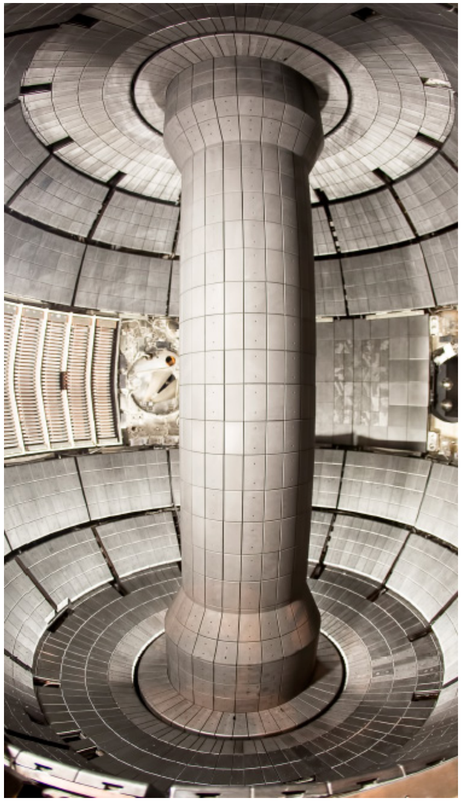
On cost and schedule, first test plasma ~100kA (Aug. 10, 2015)



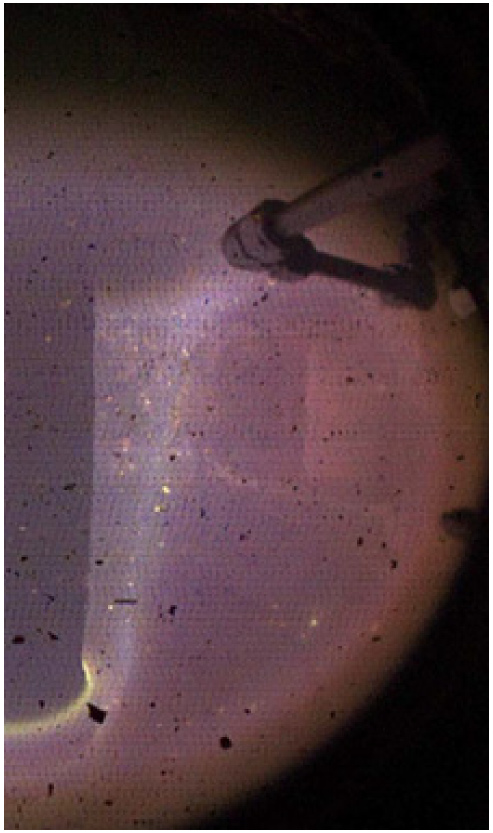
New centerstack (CS) highlights: Jan – Aug 2015

CS crane lift

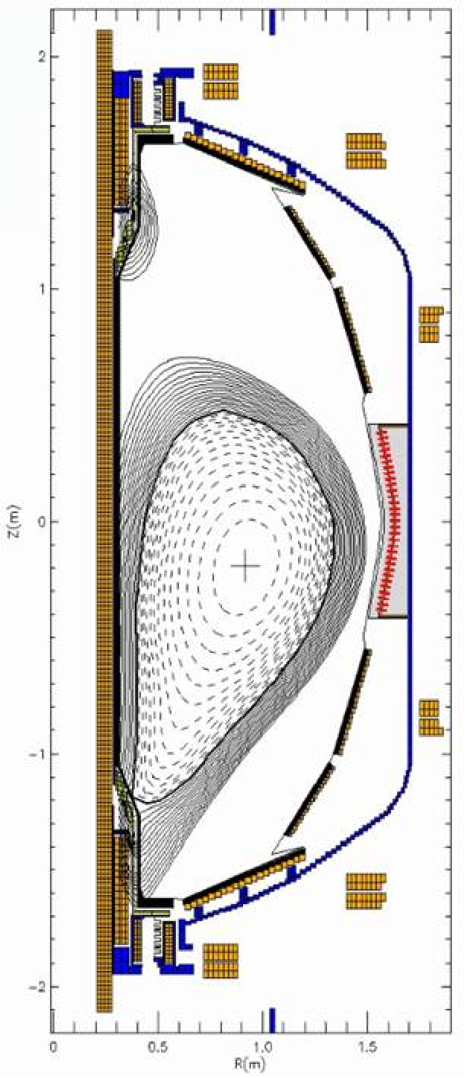
CS installed



First test plasma
(Ohmic heating only)



Magnetics functional → EFIT reconstructions



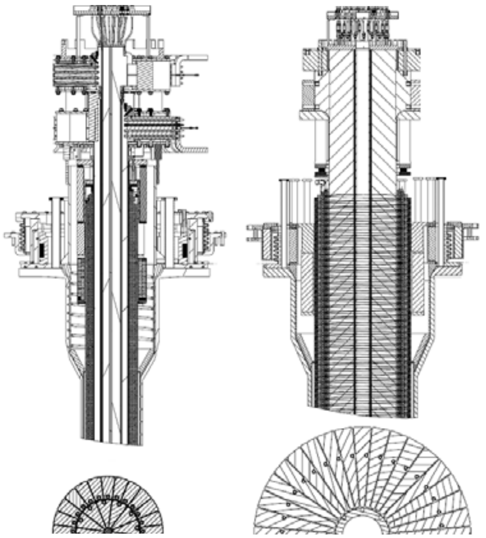
Summary

- Upgraded STs will provide many opportunities to study toroidal confinement physics in new regimes:
  - Low aspect ratio, strong shaping, high β, low collisionality
  - Access to strong fast-ion instability drive, high rotation
  - Advanced divertors, lithium walls, high-Z PFCs
- There are potentially interesting linkages between ST and CT / FRC physics that could be explored further:
  - Role of rapid rotation, strong beams, kinetic effects, …
- Thank you!

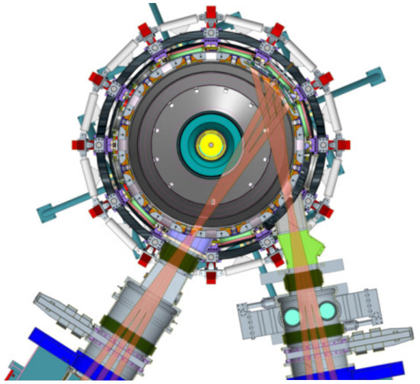
Backup

NSTX Upgrade will provide key data for ST-FNSF, ITER physics, boundary solutions, low-A Pilot/DEMO

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



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



Present NBI **New 2nd NBI**

- New CS: 2x higher B_T improves stability, access lower ν^*, 3-5s τ_{pulse} for J(r) equilibration
- 2nd more tangential neutral beam injector (NBI): 3-4x higher external current drive (CD)
  - 1.5-2x higher CD efficiency due to larger R_{tan}
  - 2x higher absorption (40→80%) at low I_P

~ 5-10x increase in nTτ from NSTX

NSTX-U average plasma pressure ~ Tokamaks

Key NSTX-U research topics for FNSF and ITER:

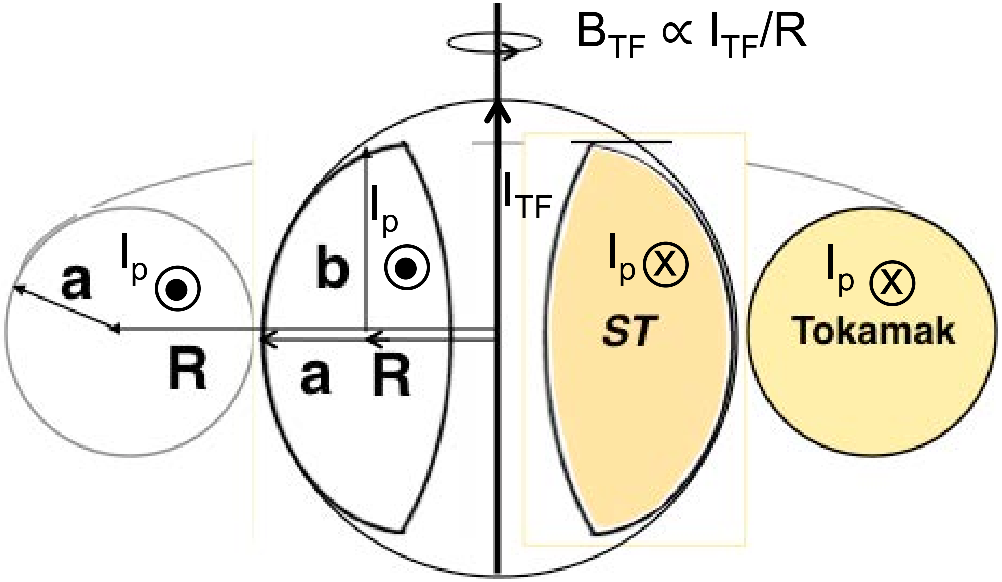
- **Stability and steady-state control at high β**
- **Confinement scaling (esp. electron transport)**
- **Non-inductive start-up, ramp-up, sustainment**
- **Divertor solutions for mitigating high heat flux**

J. Menard, et al., NF 2012

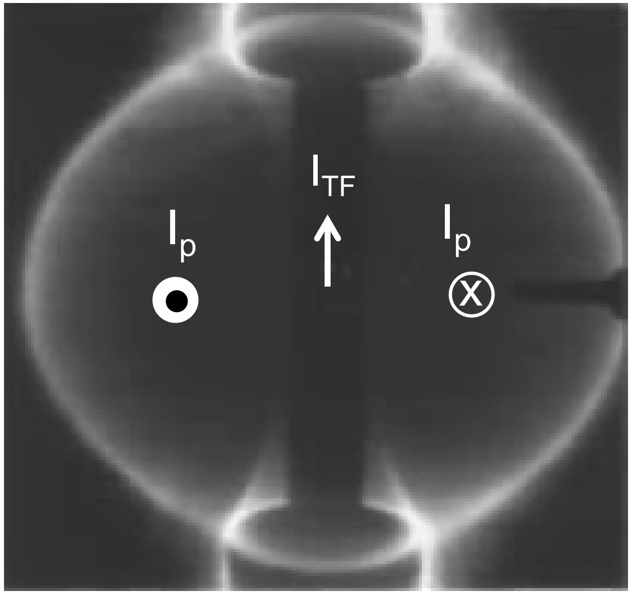
ST is a low aspect ratio tokamak with A < 2

Natural elongation makes its spherical appearance

Aspect Ratio $A = R/a$ | **Elongation $\kappa = b/a$** | "natural" = "without active shaping"



Camera image from START



A. Sykes, et al., Nucl. Fusion (1999).

Note: ST differs from FRC, spheromak due to B_{TF}

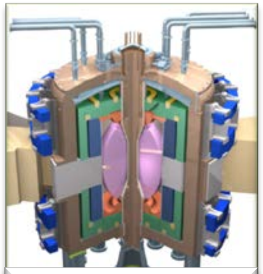
Y-K.M. Peng, D.J. Strickler, NF (1986)

Fusion needs FNSF(s) (modest cost, low T, and reliable) to Test and Qualify Fusion Components

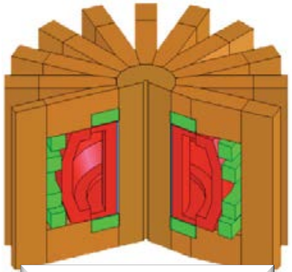
Fusion needs to develop reliable/qualified components which are unique to fusion:

- Divertor / PFC
- Blanket and Integral First Wall
- Vacuum Vessel and Shield
- Tritium Fuel Cycle
- Remote Maintenance Components

FNSFs



FNSF-ST



FNSF-AT

- Without R&D, fusion components could fail prematurely which often requires long repair/down time.
- FNSF can help develop reliable fusion components.
- Such FNSF facilities must be modest cost, low T, and reliable.

- If the cost of volume neutron source (FNSF) facility is “modest” << ITER, DEMO, it becomes highly attractive development step in fusion energy research. *M.A. Abdou, et al., FTS (1996)*

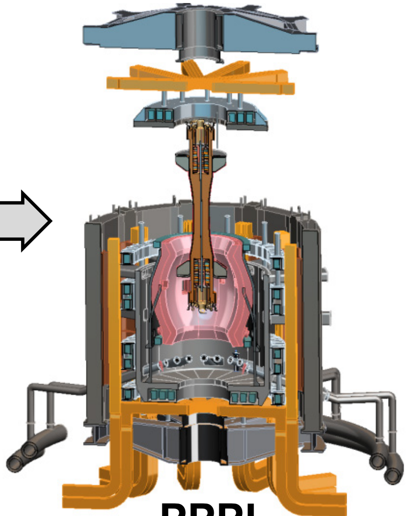
ST approach to FNSF potentially attractive

- **Projected to access high neutron wall loading at moderate R_0, P_{fusion}**
  - $W_n \sim 1\text{-}2\ \mathrm{MW/m^2}$, $P_{fus} \sim 50\text{-}200\mathrm{MW}$, $R_0 \sim 0.8\text{-}1.8\mathrm{m}$
- **Modular, simplified maintenance**
- **Tritium breeding ratio (TBR) near 1**
  - Requires sufficiently large R_0, careful design

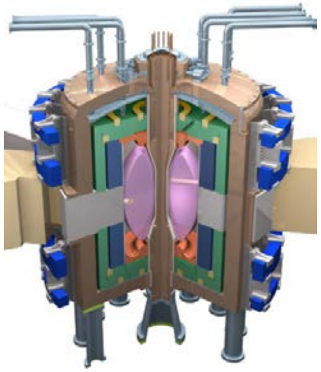
R&D Needs for an ST-FNSF

- Non-inductive start-up, ramp-up, sustainment
  - **Low-A → minimal inboard shield → no/small transformer**
- Confinement scaling (especially electrons)
- Stability and steady-state control
- Divertor solutions for high heat flux
- Radiation-tolerant magnets, design

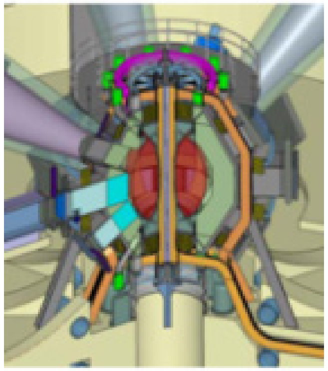
Example ST-FNSF concepts



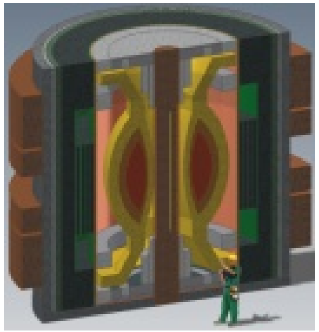
PPPL



ORNL

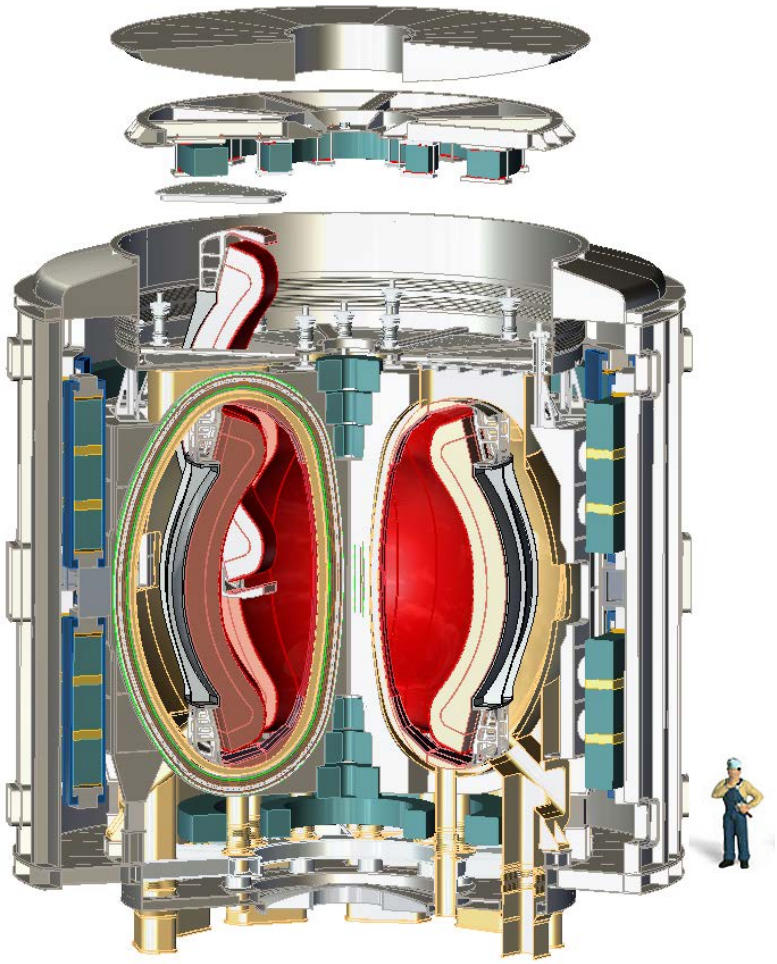


Culham (UK)



UT Austin

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



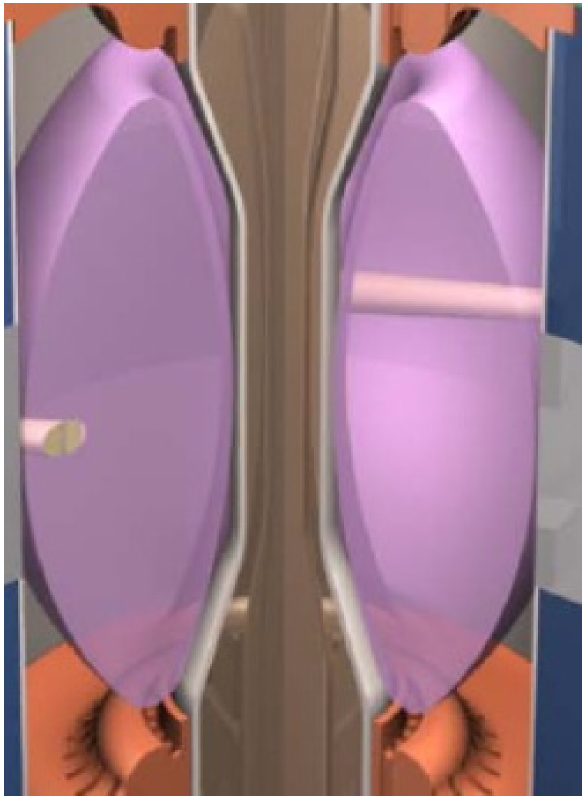
R_0 = 1.4m, B_T = 3.2T, I_P = 7-8MA, $P_{fusion\text{-}DT}$ = 100MW

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

- Possible missions:
  - Steady-state toroidal PMI facility
  - 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
- Many remaining issues:
  - HTS lifetime in radiation environment
  - Blanket/shield thickness, location, TBR

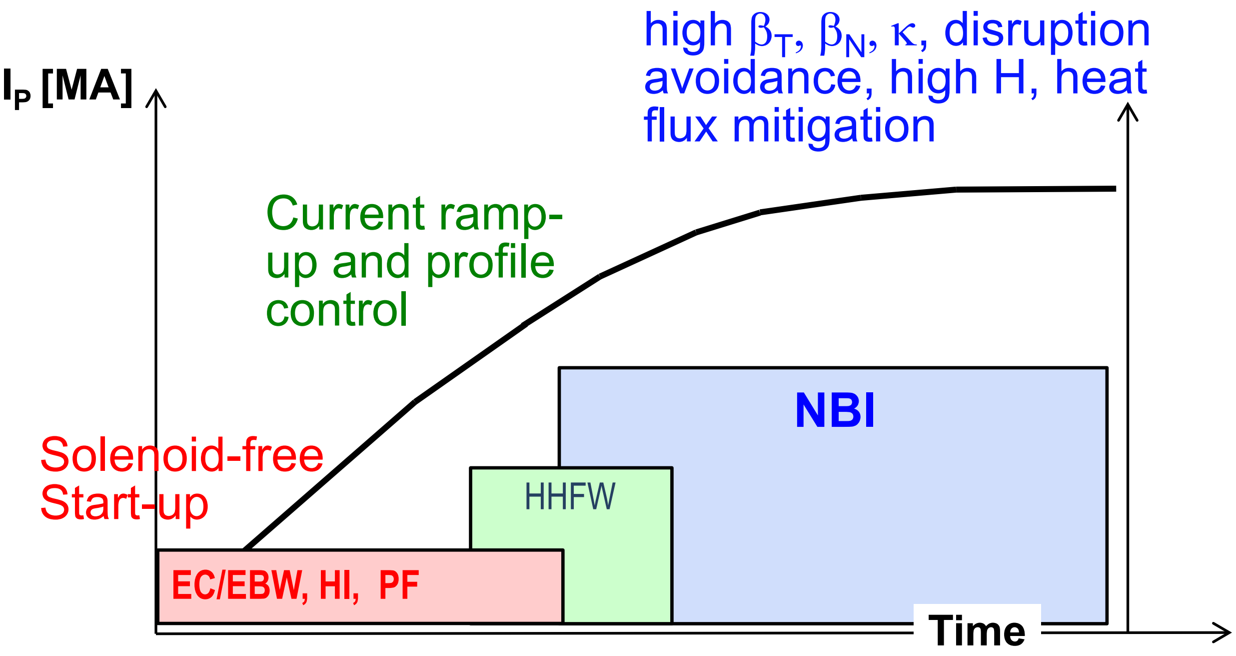
I_P Start-up/Ramp-up Critical Issue for ST-FNSF/Demo

Compact ST-FNSF has no/small central solenoid



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

ST-FNSF Scenarios

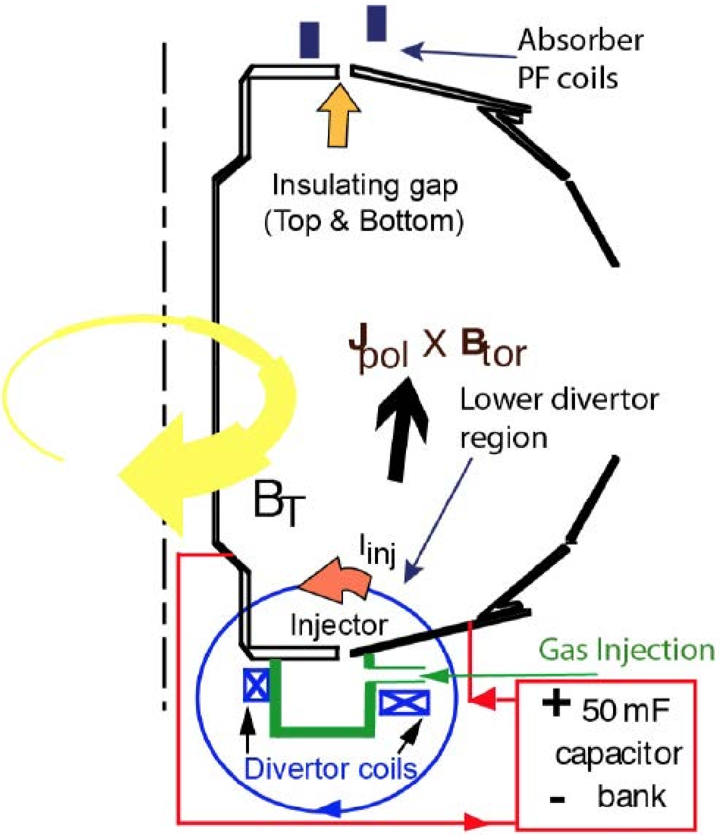


- Two novel techniques for solenoid-free start-up and ramp-up will be investigated
  - RF: ECH/EBW and HHFW
  - Helicity Injection

Helicity Injection is efficient method for current initiation

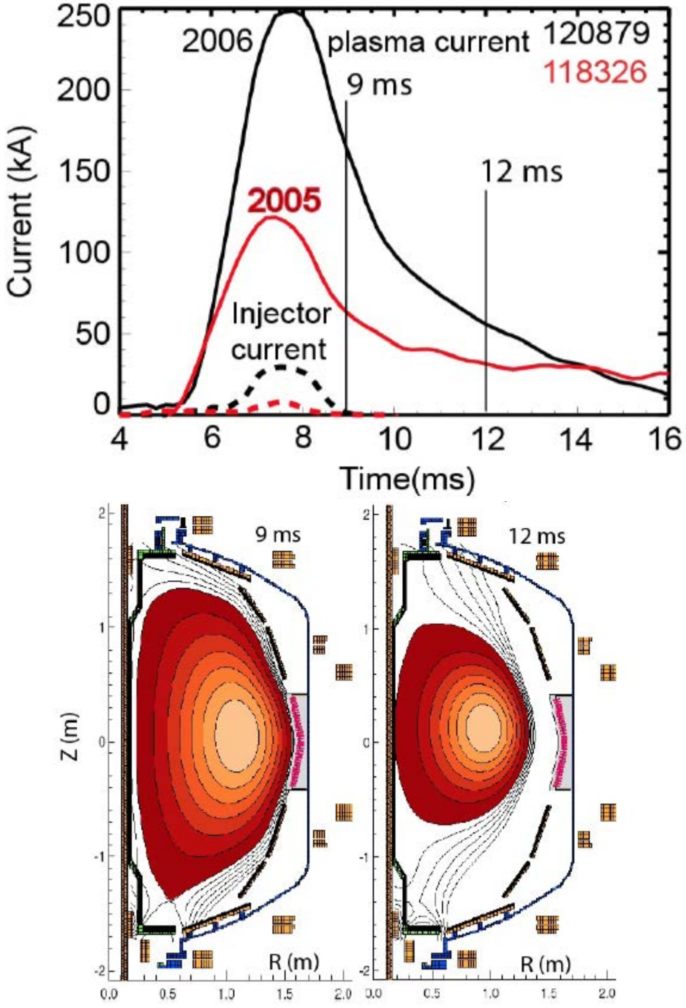
Coaxial Helicity Injection (CHI) concepts being developed

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

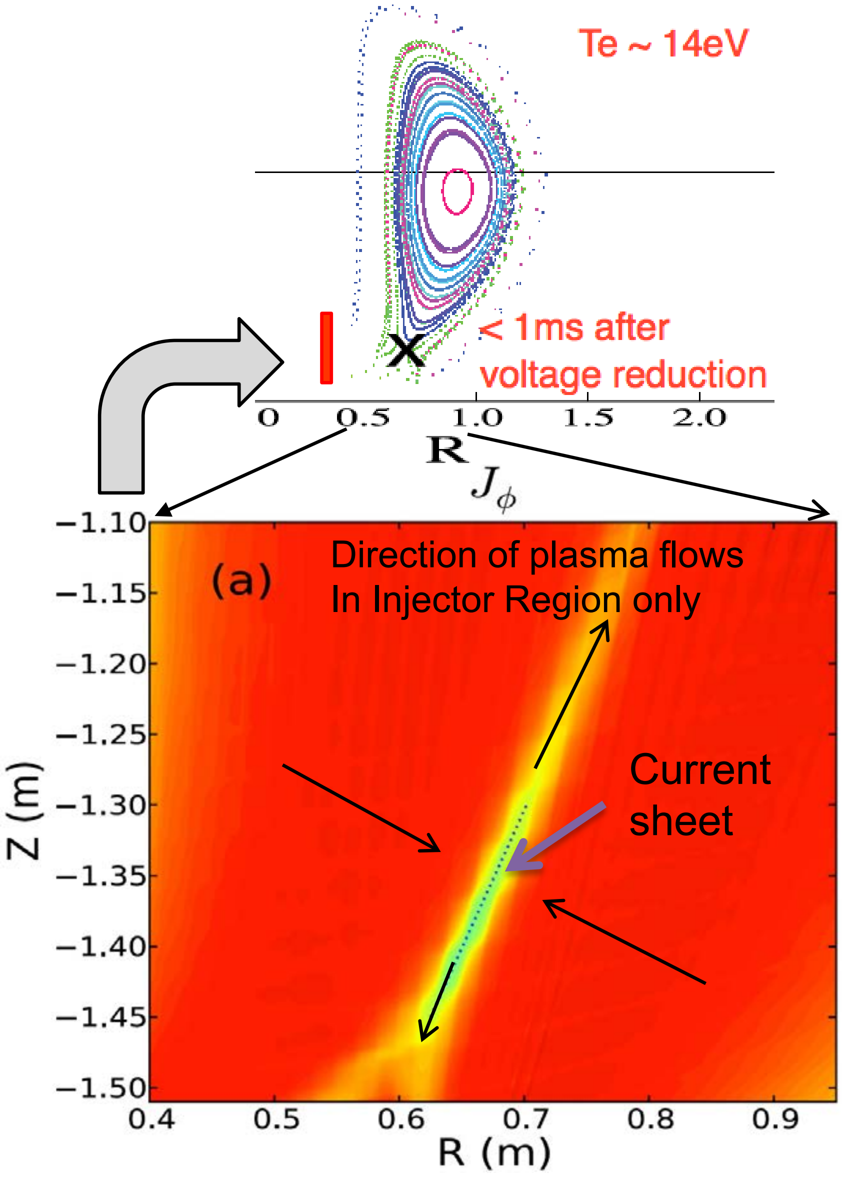


R. Raman et al., PRL 2006

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


Te ~ 14eV

< 1ms after voltage reduction

Direction of plasma flows In Injector Region only

Current sheet

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

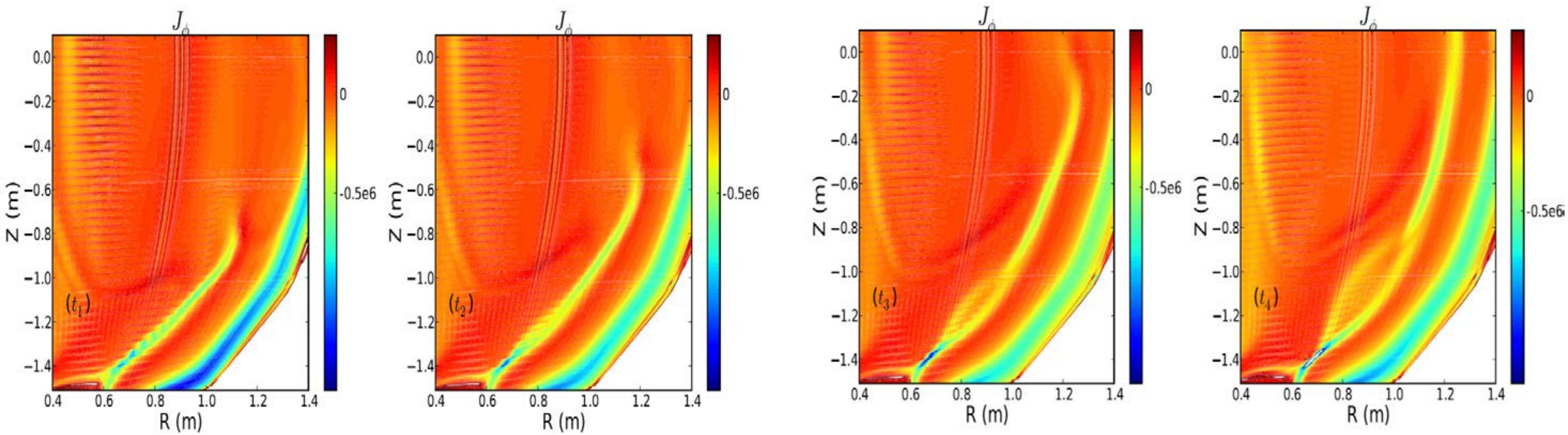
- Elongated Sweet-Parker-type current sheet
- n > 0 modes/MHD not strongly impacting 2D reconnection

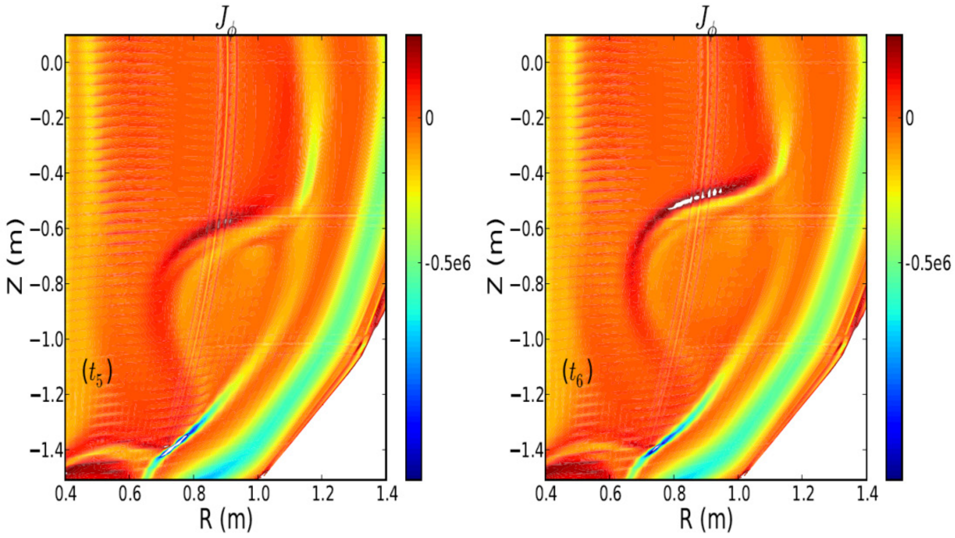
F. Ebrahimi, PoP 2013, PoP 2014

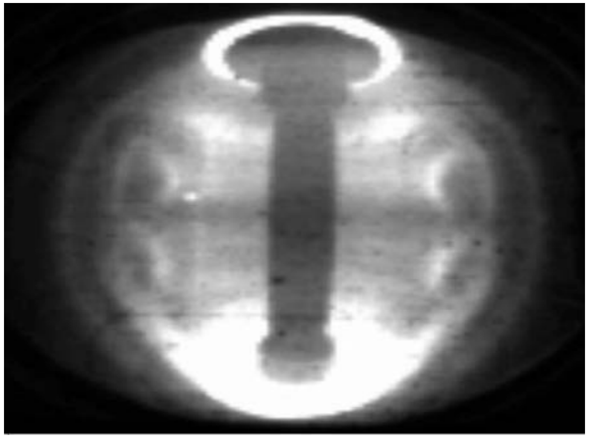
CHI current sheet unstable → plasmoids → merging

Possible lab observation of plasmoids - relevant to astrophysics

Current sheet shown in the lower half of the device.





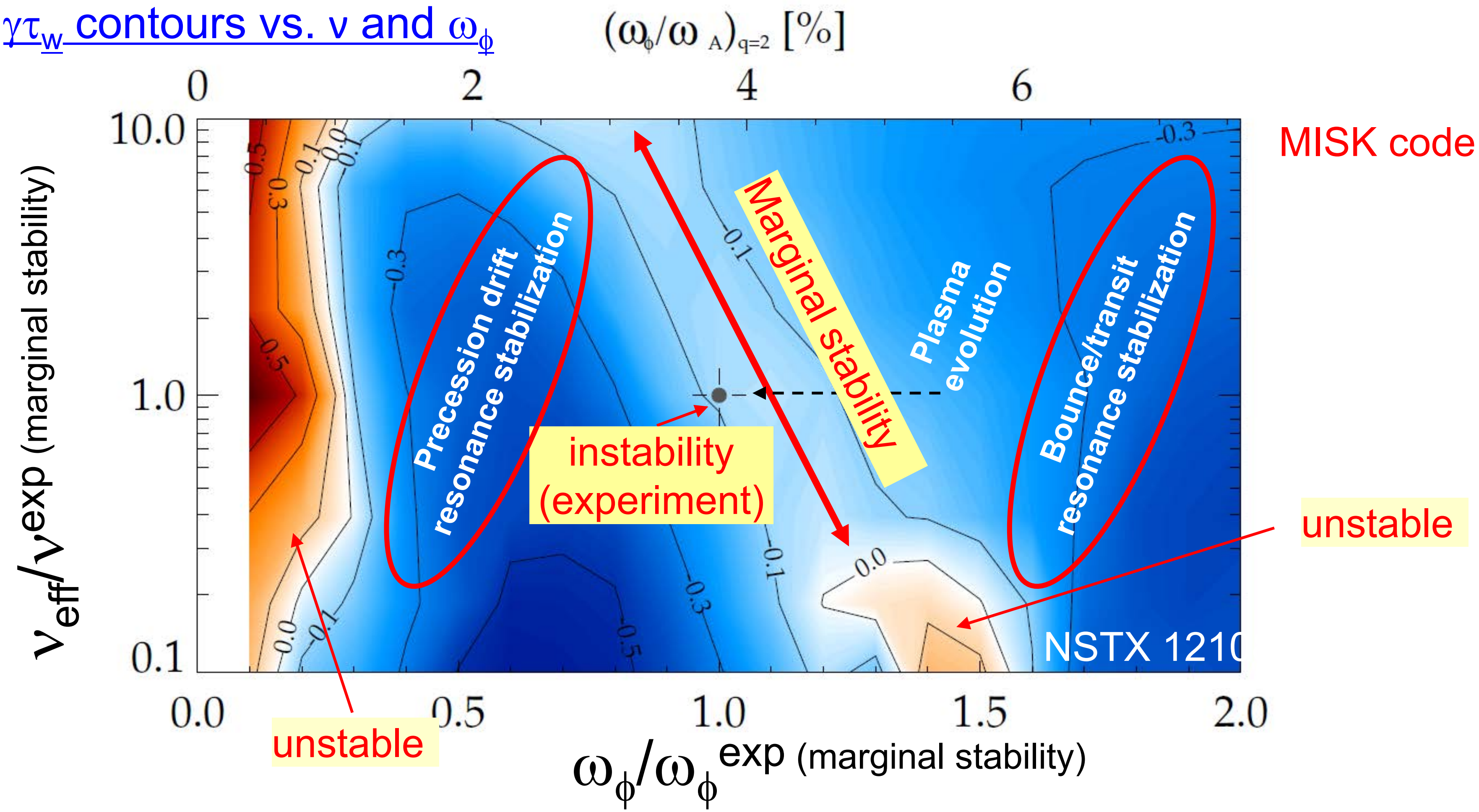


Possible plasmoids during NSTX CHI

Ebrahimi and Ramam, PRL 2015

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

$\gamma\tau_w$ contours vs. ν and ω_ϕ



MISK code

Precession drift resonance stabilization

Marginal stability

Plasma evolution

Bounce/transit resonance stabilization

instability (experiment)

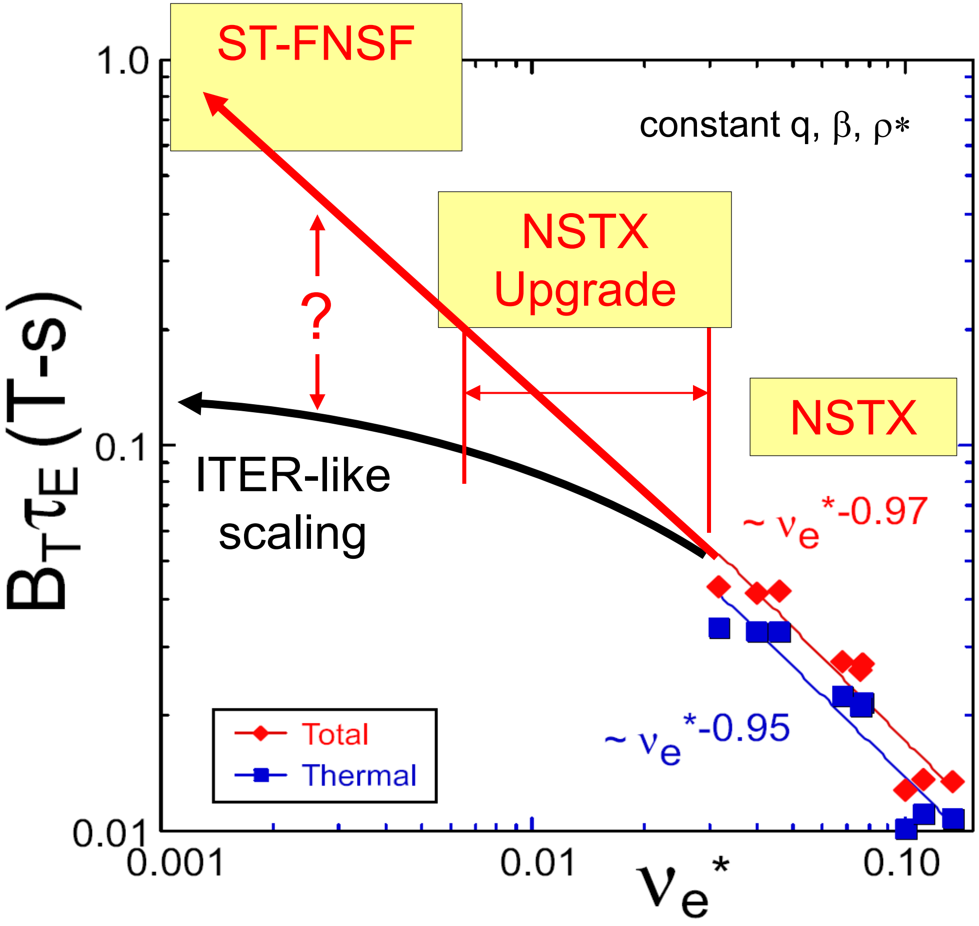
unstable

unstable

- 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

J.W. Berkery, et al., PRL **104** (2010) 035003
S.A. Sabbagh, et al., NF **50** (2010) 025020

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



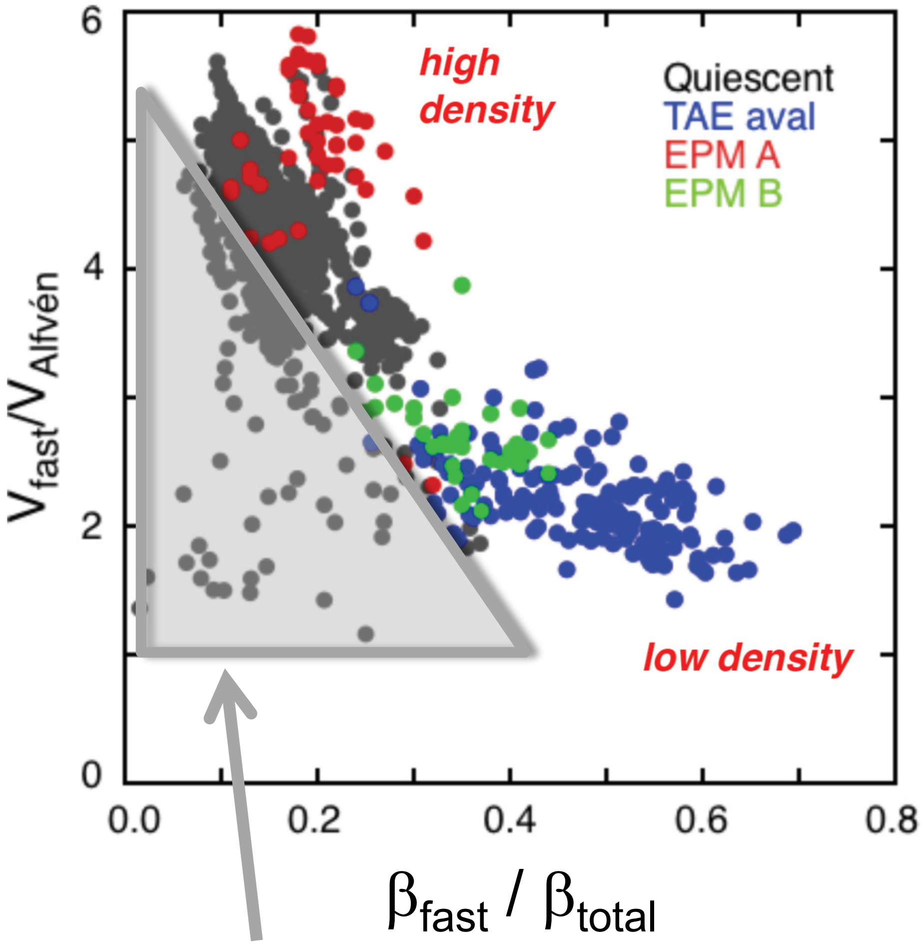
Normalized electron collisionality $\nu_e^* \propto n_e / T_e^2$

Favorable confinement results could lead to more compact ST reactors

Assessed parametric dependence of TAE avalanches and energetic particle modes (EPMs) in NSTX

Identified regimes w/ small fast-ion loss: important for NSTX-U, FNSF, ITER

- Modes lead to neutron rate decrements up to 30%
- TAE avalanches only occur for $\beta_{fast} > 0.3\ \beta_{total}$
- Conversely, quiescent plasmas were only seen where $\beta_{fast} < 0.3\ \beta_{total}$
- Two types of EPM (A&B)
  - A: Lower $q_{min} \rightarrow 1$ (later in shot), more bursty and fishbone-like, n=1-3
  - B: Higher q_{min}~2-3 (earlier in shot), more continuous, transitions to long-lived n=1

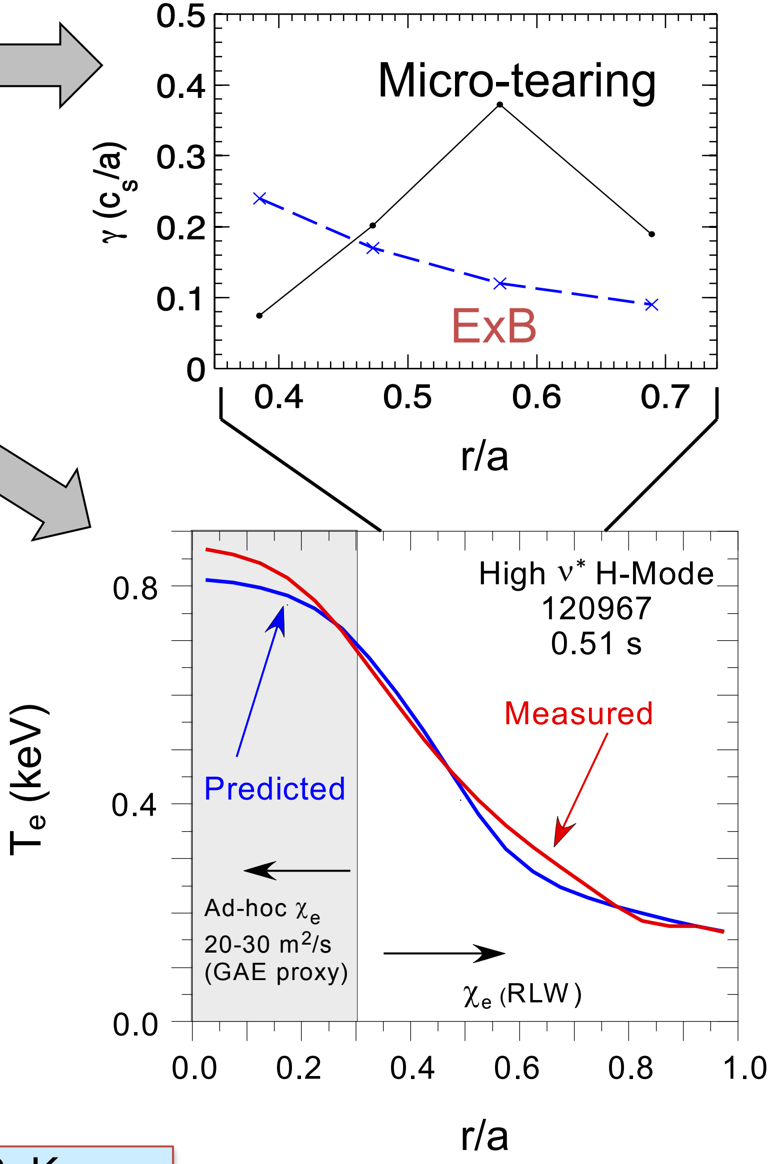


Quiescent operating range

E. Fredrickson

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)

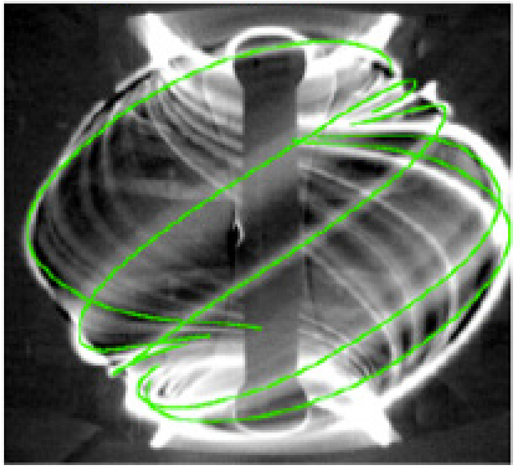


S. Kaye

H-mode / ELM physics: High Priority Research Goal

Unmitigated ELMs could cause PFC damage in reactors

Video images of MAST plasmas showing a filamentary ELM structure



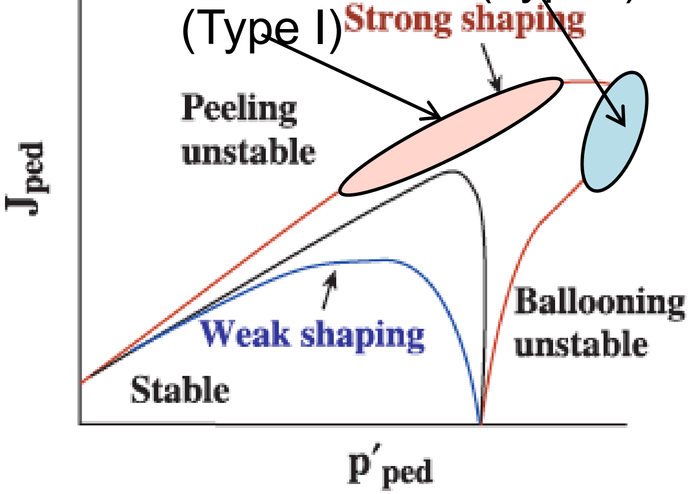
N. Ben Ayed et al., PPCF (2009).

ST is in strongly shaped ELM regimes

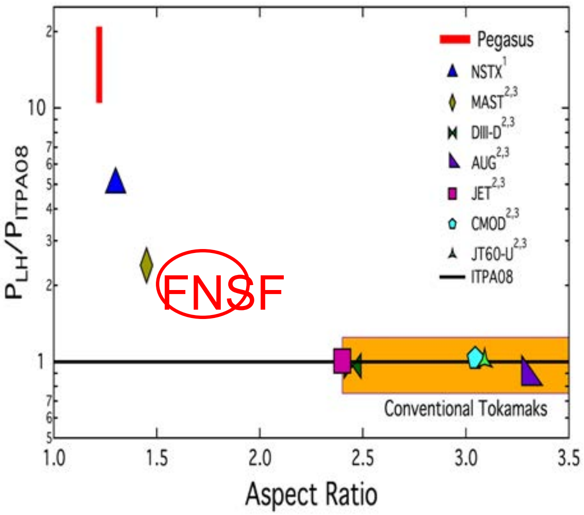
P.B. Snyder et al., PoP (2002).

MAST (Type I)

NSTX, PEGASUS (Type I)



L-H power threshold scaling extended for low A



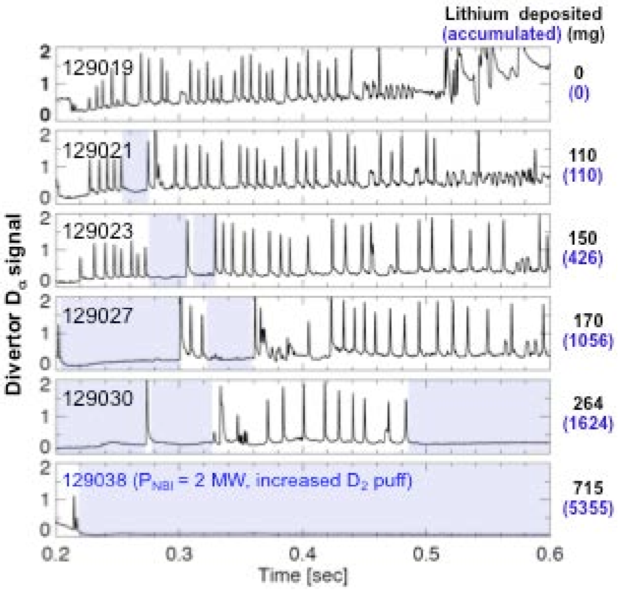
K.E. Thome et al., EPR (2014)

- NSTX/MAST/PEGASUS accessed H-mode at very low heating power < 1 MW and also in ohmic plasmas
- NSTX-U and MAST-U will provide H-mode access scaling for FNSF

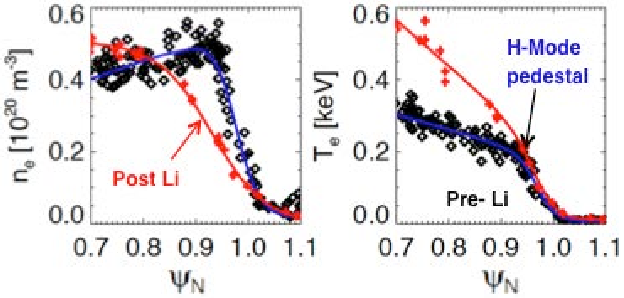
ELM Stabilization and Mitigation

Through application of lithium and 3-D fields

ELMs stabilized with edge pressure modification with Li in NSTX

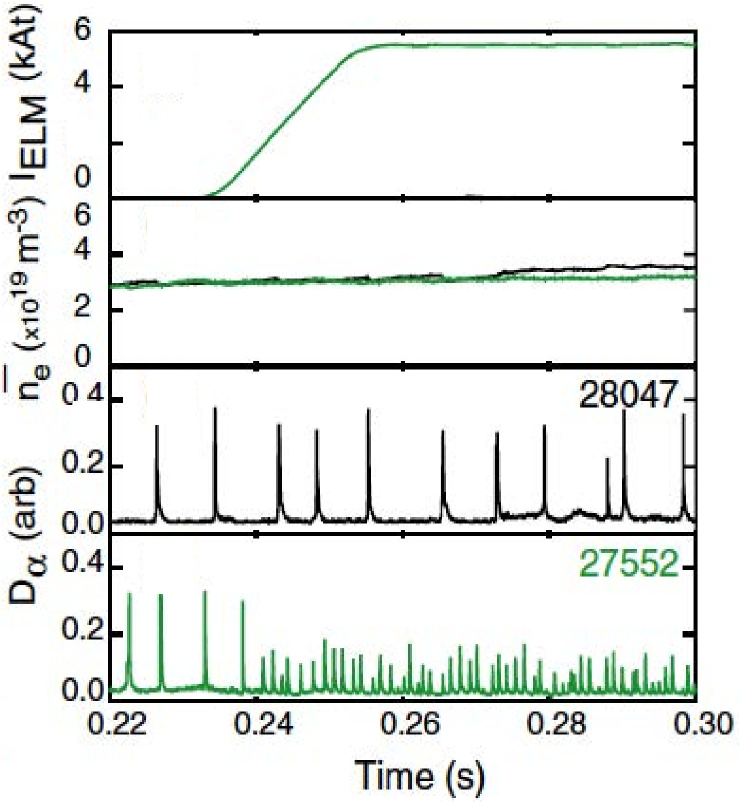


D.K. Mansfield, et al., JNM (2009)



R. Maingi, et al., PRL (2009).

ELM mitigation with n=3
3-D fields (ELM Coils) in MAST



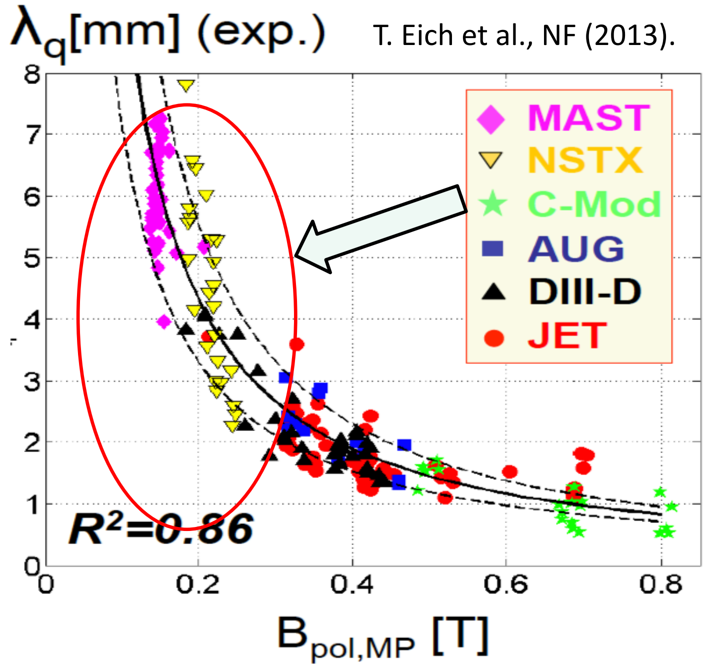
Increasing Type I ELM freq. by x 8 (900 Hz) has reduced heat flux

A. Kirk et al., NF (2013)

Divertor heat flux in Low-A regime

ST power flux width clearly shows 1/$B_{poloidal}$ variation

STs data breaks A degeneracy of power flux width study.



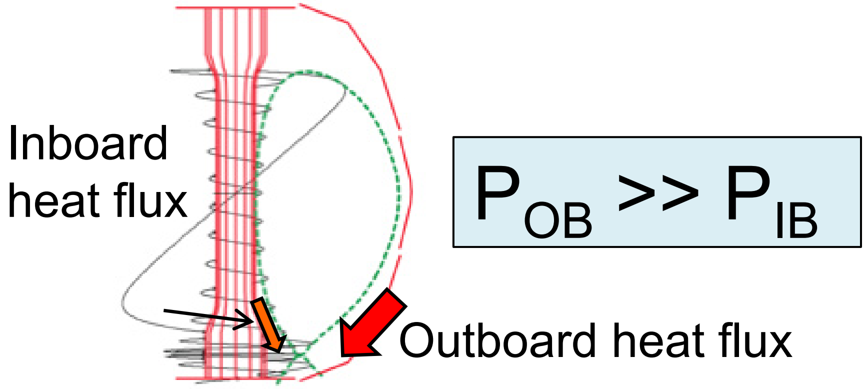
T. Eich et al., NF (2013).

Heuristic model by R.J. Goldston, NF (2012).

* Unfavorable for large size, Ip devices such as ITER and Demo

"P B / R" as the new heat flux metric which is favorable for STs

Most divertor power arrives at outboard side in MAST and NSTX!

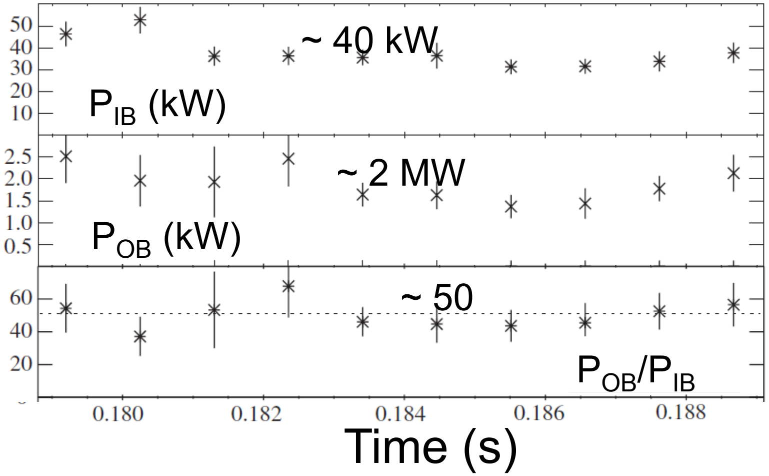


Inboard heat flux

$P_{OB} >> P_{IB}$

Outboard heat flux

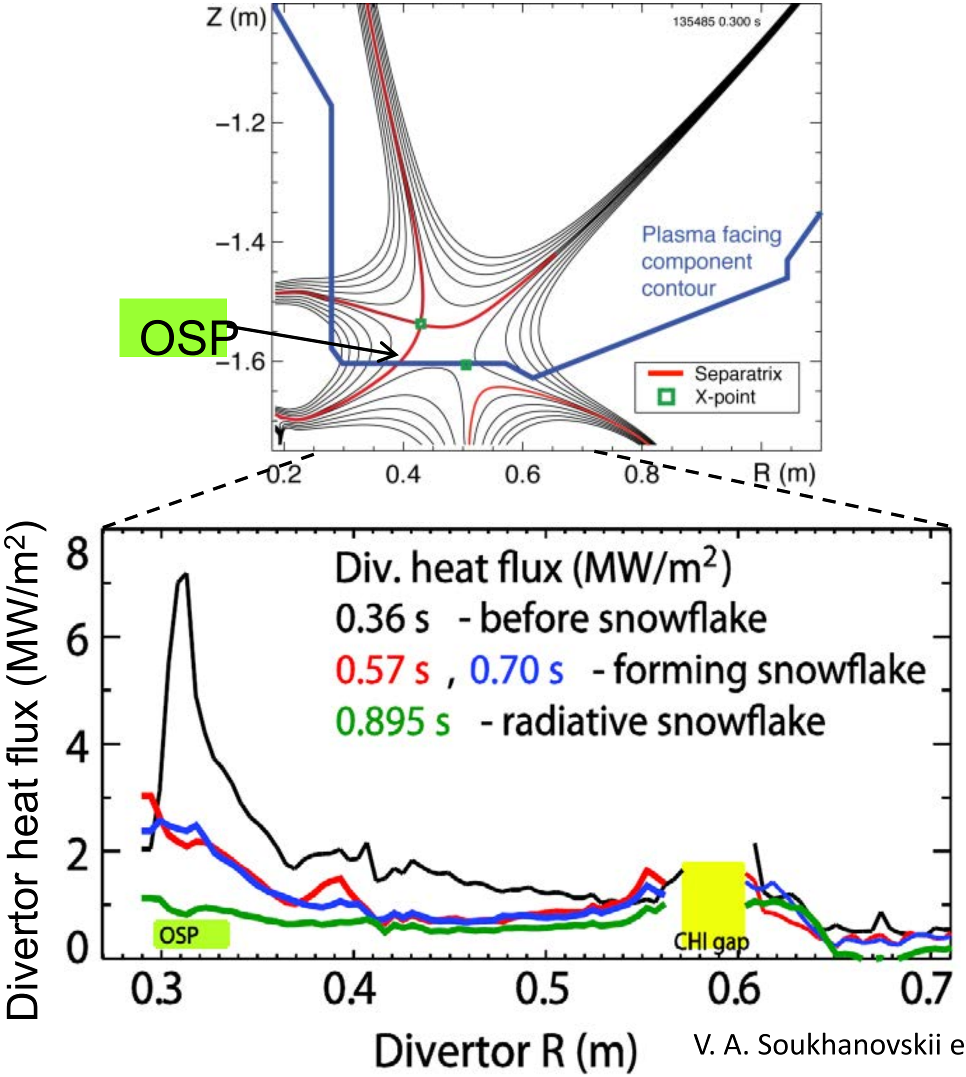
Ratio of outboard power flux vs. inboard in MAST



G.F. Counsell et al., PPCF (2002)

Divertor flux expansion of ~ 50 achieved with Snow Flake Divertor with large heat flux reduction in NSTX

Snowflake divertor in NSTX



V. A. Soukhanovskii et al., PoP (2012)

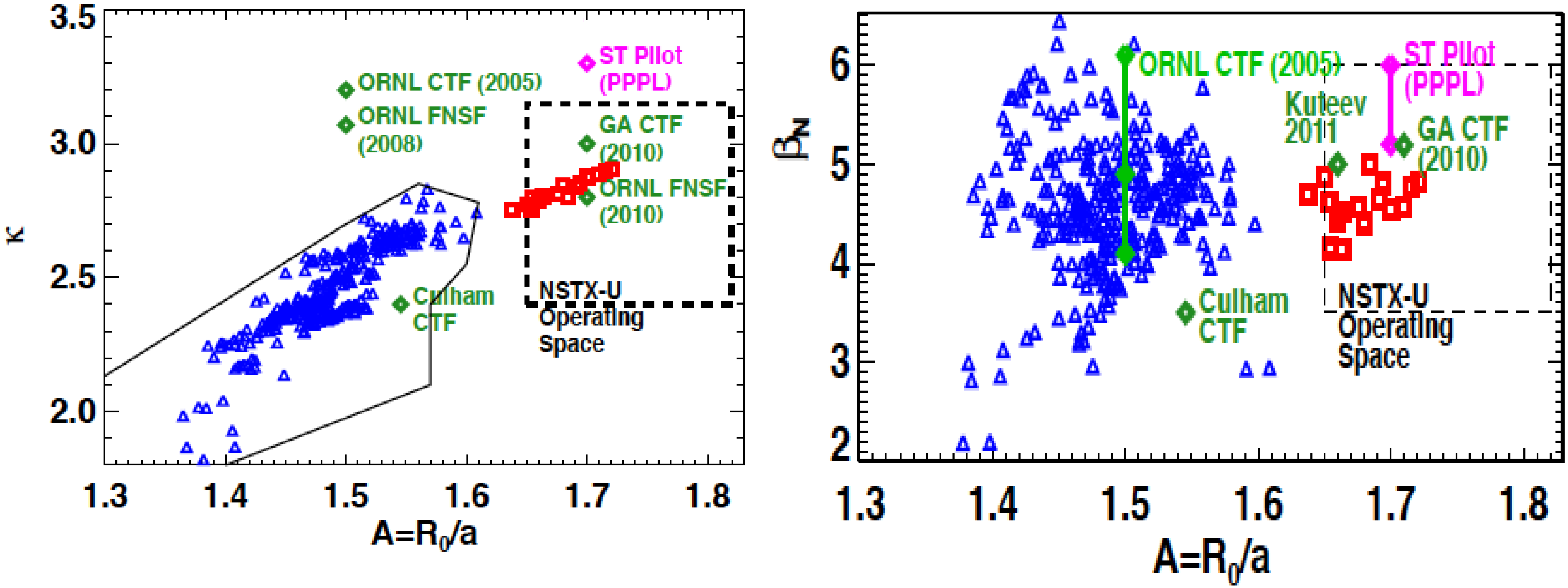
NSTX-U will investigate novel divertor heat flux mitigation concepts needed for FNSF and Demo.

- Up-and-down symmetric Snow Flake / X-divertors
- Lithium + high-Z metal PFCs

NSTX has accessed A, β_N, κ needed for ST-based FNSF

Requires $f_{BS} \geq 50\%$ for plasma sustainment

$$f_{BS} \equiv I_{BS} / I_p = C_{BS}\, \beta_p / A^{0.5} = (C_{BS}/20)\, A^{0.5}\, q^*\, \beta_N \propto A^{-0.5} (1+\kappa^2)\, \beta_N^2 / \beta_T$$



S.P. Gerhardt et al., NF (2011)

NSTX achieved f_{BS} ~ 50% and f_{NI} ~ 65-70% with beams
NSTX-U expects to achieve f_{NI} ~100% with the more tangential NBI (~ 1.5- 2x higher current drive efficiency)