NSTX-U

Supported by

U.S. DEPARTMENT OF ENERGY

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Scientific Opportunities and Challenges on the National Spherical Torus eXperiment Upgrade (NSTX-U)

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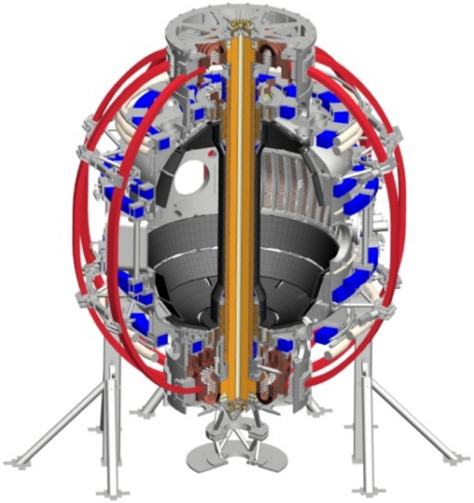
Jon Menard
NSTX-U Program Director

Graduate Student Seminar
PPPL theory conference room
February 2, 2015



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ASCR, Czech Rep

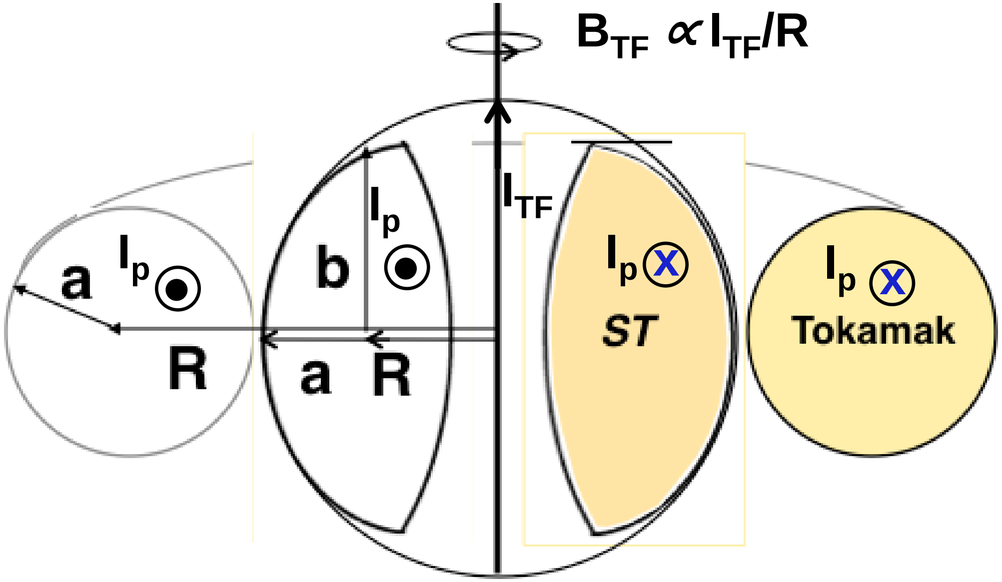
Talk Outline

- Introduction to Spherical Tokamaks/Tori (STs)
- Description of NSTX Upgrade
- Unique physics characteristics of ST and research opportunities
  - Macroscopic Stability
  - Transport and Turbulence
  - Boundary Physics
  - Energetic Particles
  - Non-inductive start-up, ramp-up, sustainment
- Summary

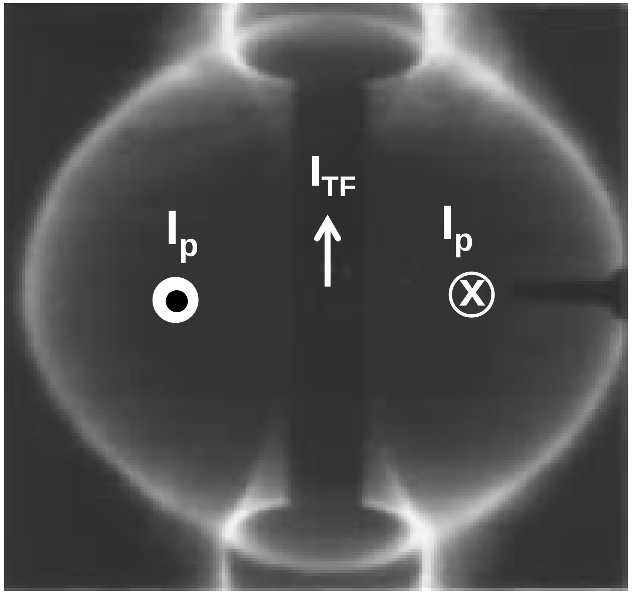
ST is a low aspect ratio tokamak with A < 2

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).

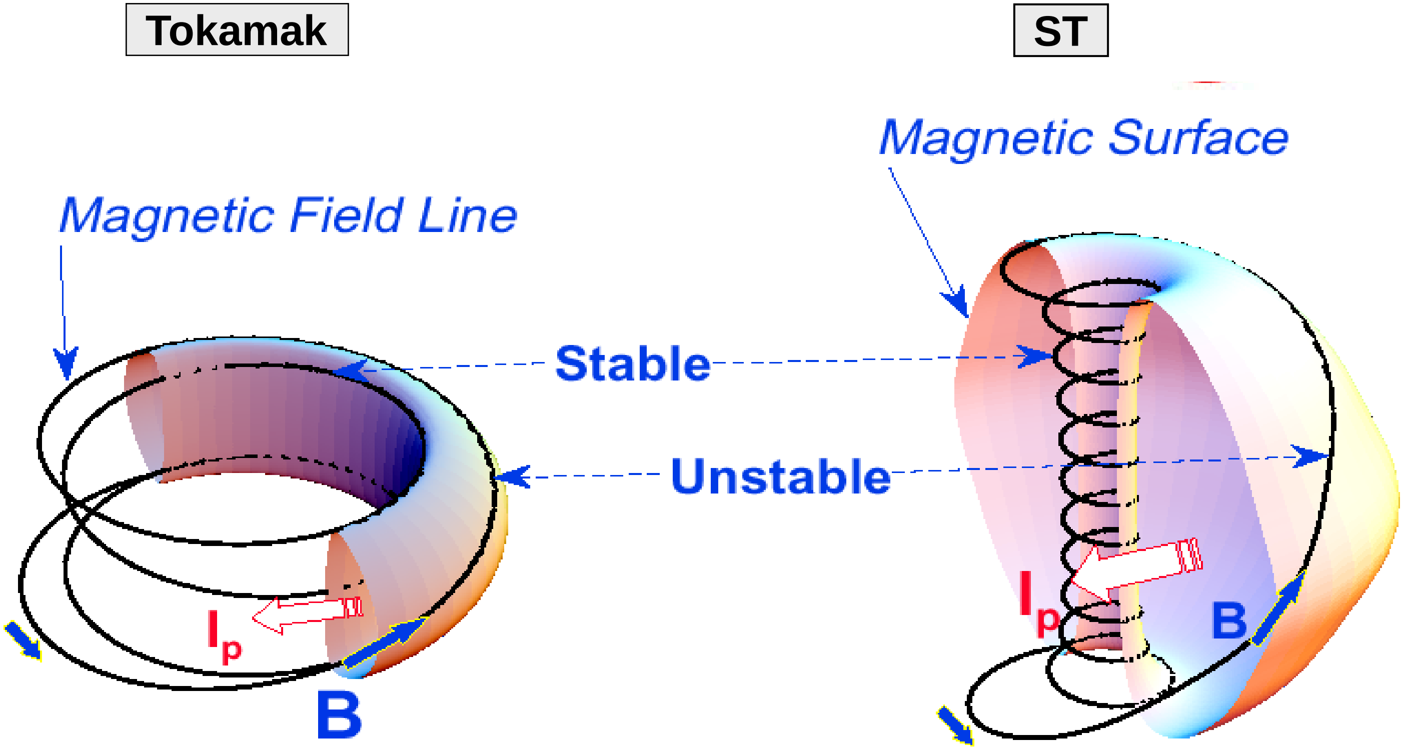
Note: ST differs from FRC, spheromak due to B_{TF}

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

A spherical tokamak (ST) is a high beta tokamak

Favorable average curvature improves stability at high beta

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



A ~ 3,
κ = 1.5-2,
q_{95} = 3-4,
β_T = 3-10%

A ~ 1.5,
κ = 2-3,
q_{95} = 8-12,
β_T = 10-40%

ST can be compact, high beta, 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 Ip due to high κ and low A**

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

S. Jardin et al., FS&T (2003)

- **Ip increases tokamak performance**

$$\tau_E \propto I_p$$

$$\beta_T [\%] \equiv \beta_N I_p / (aB_{T0})$$

- **ST can achieve high performance cost effectively**

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

⇧

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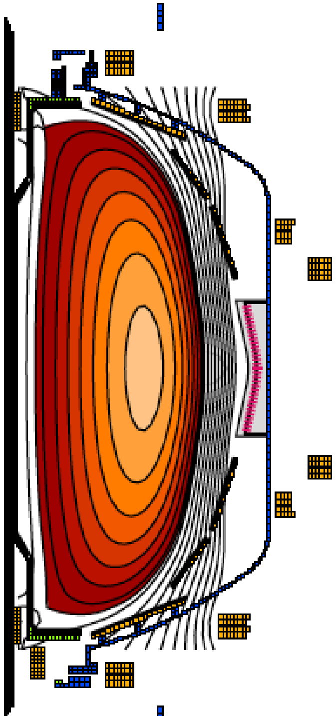
High $\kappa \sim 3.0$ equilibrium in NSTX.



D.A. Gates et al., NF (2007).

New physics regimes are accessed at low aspect ratio, enhancing the understanding of toroidal confinement physics

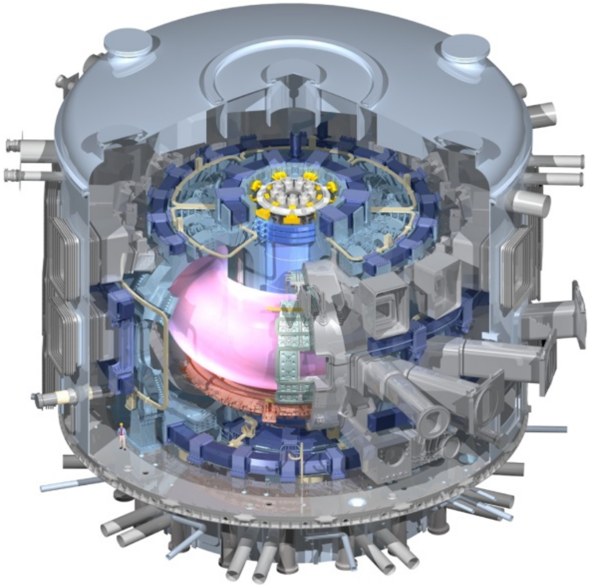
- Lower A → increased toroidicity → higher β, strong shaping
- Higher β → electromagnetic effects in turbulence, energetic-particle modes, RF heating and CD (over-dense plasmas)
- Higher fraction of trapped particles (low A), increased normalized orbit size (high β), and flow shear (due to low B, low A)→ broad range of effects on transport and stability
- Increased normalized fast-ion speed (high β) → simulate fast-ion transport/losses of ITER
- Compact geometry (small R) → high power/particle/neutron flux relevant to ITER, reactors

Unique ST properties support and accelerate a range of development paths toward fusion energy

Extend Predictive Capability for ITER and Toroidal Science

- High β physics, rotation, shaping for MHD, transport
- Non-linear Alfvén modes, fast-ion dynamics, Electron gyro-scale turbulence at low ν^*

Burning Plasma Physics - ITER

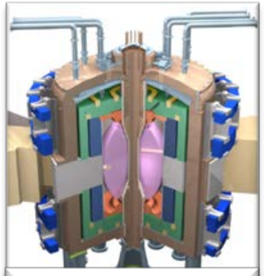


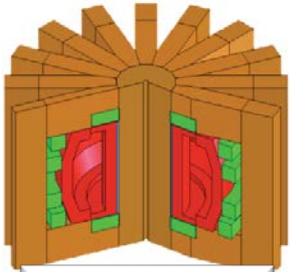
Fusion needs FNSF(s) (modest cost, low T, and reliable) to Test and Qualify Fusion Components

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

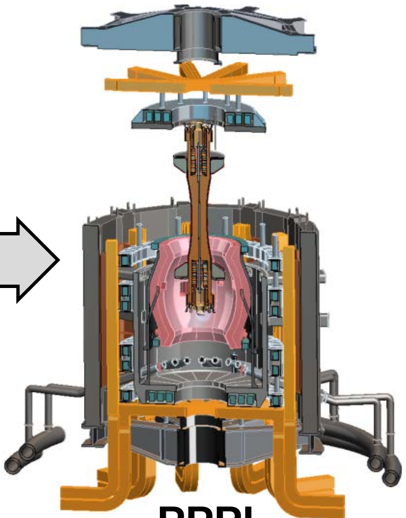
- **Without R&D, fusion components could fail prematurely which often requires long repair/down time. This would cripple the DEMO operation.**
- **FNSF can help develop reliable fusion components.**
- **Such FNSF facilities must be modest cost, low T, and reliable.**

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)

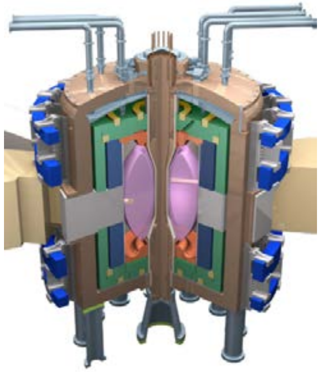
There have been several studies of ST-FNSF showing the potential attractiveness of this approach

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

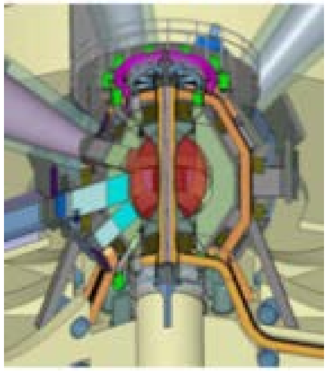
Example ST-FNSF concepts



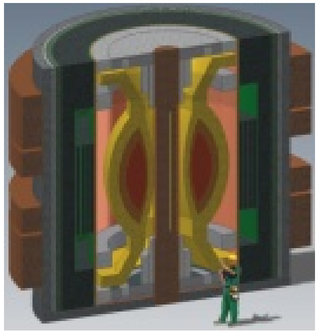
PPPL



ORNL



Culham (UK)

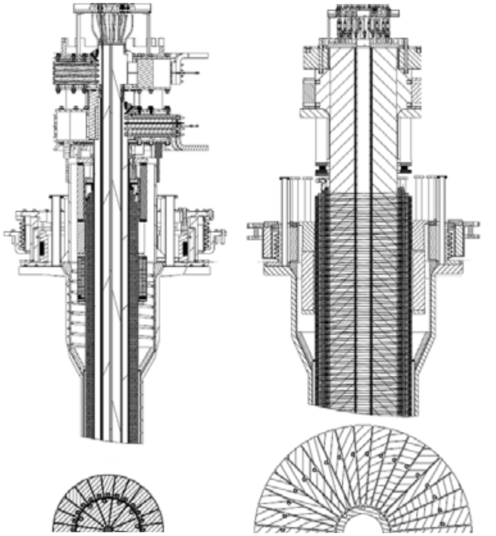


UT Austin

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

NSTX-U to provide data base to support ST-FNSF design, ITER operations, boundary solutions

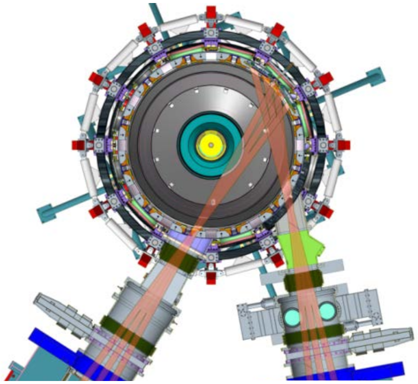


Previous center-stack

New center-stack

TF OD = 20cm

TF OD = 40cm



Present NBI

New 2nd NBI

- **New CS provides higher 2x 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**
  - **1.5-2x higher current drive efficiency**

~ 5-10x increase in $nT\tau$ from NSTX

NSTX-U average plasma pressure ~ Tokamaks

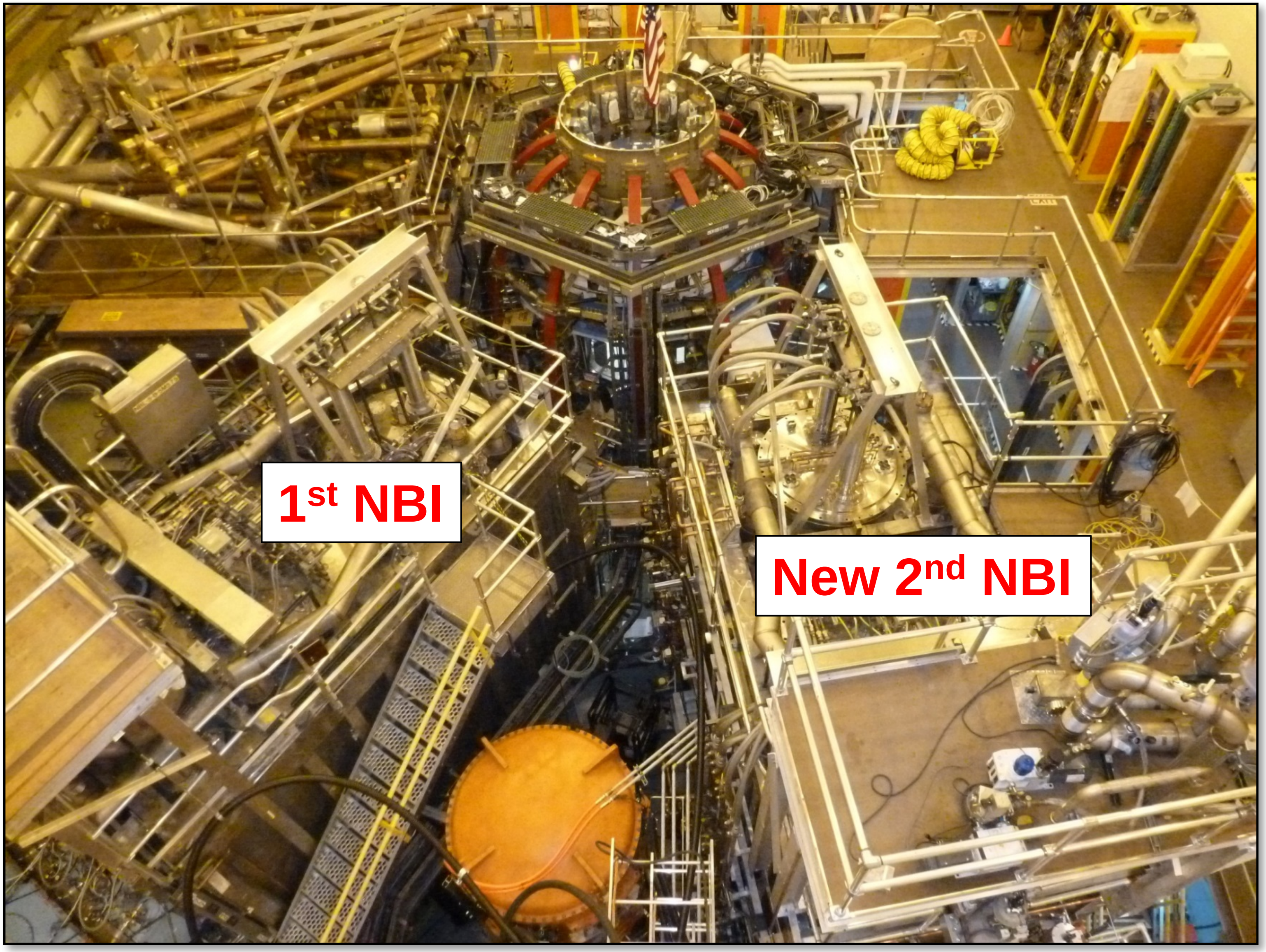
Key NSTX-U research topics for FNSF and ITER

- **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)

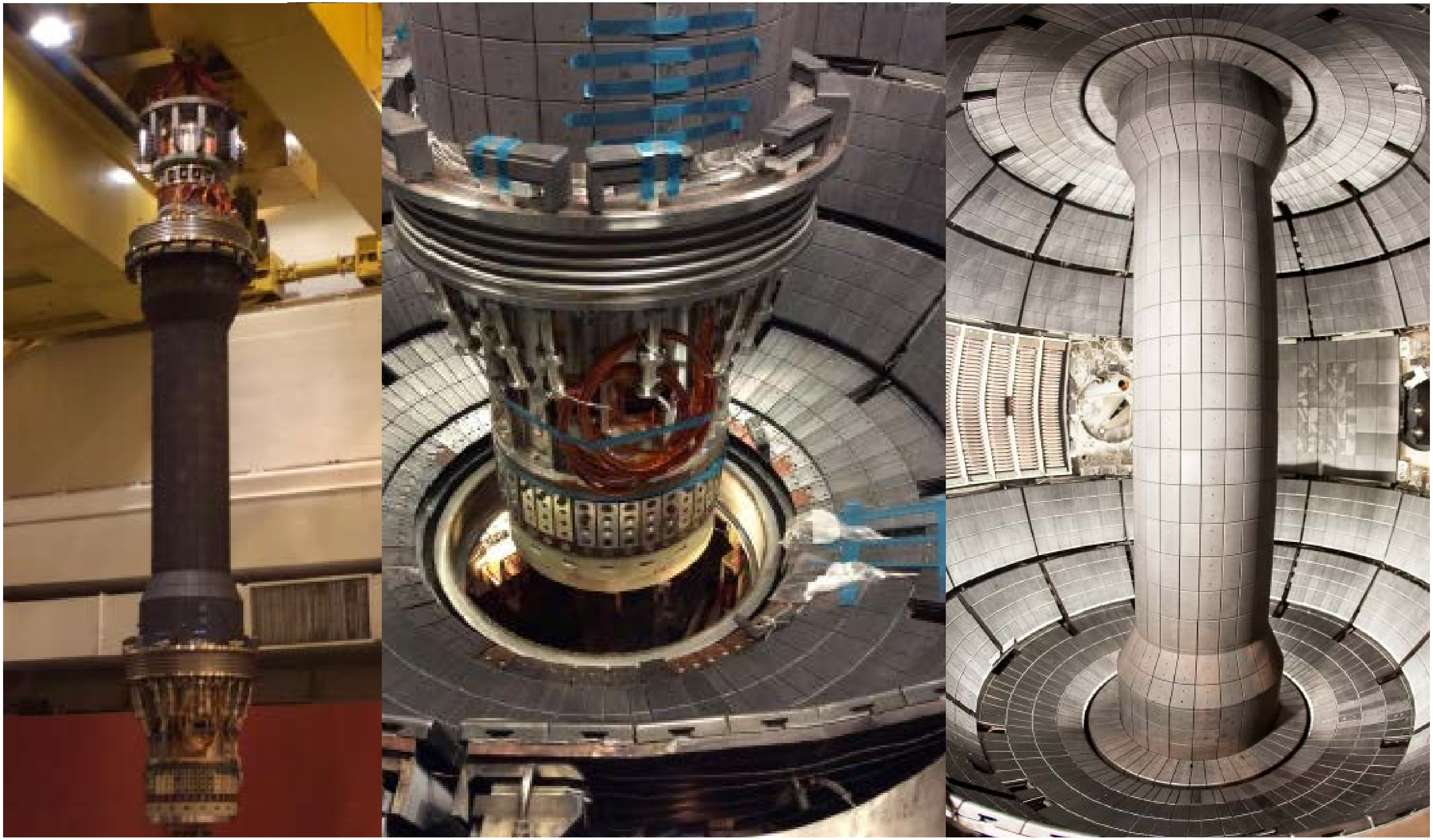
NSTX Upgrade Project nearing completion

First plasma expected Mar/Apr 2015, research in May/June



New Center-Stack Installed In NSTX-U

Vacuum pump-down achieved last month (January)



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Higher β_T enables higher fusion power and compact FNSF for required neutron wall loading

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

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

Product

Physics Knob

$

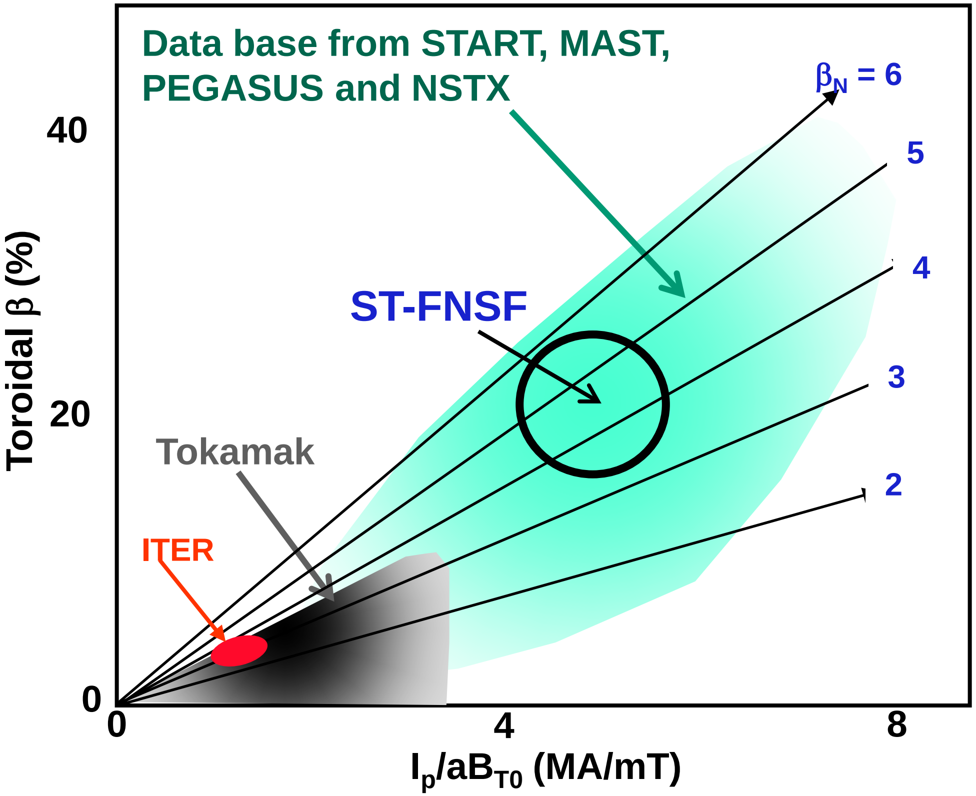
Investment

High neutron wall loading W_n possible in a compact ST

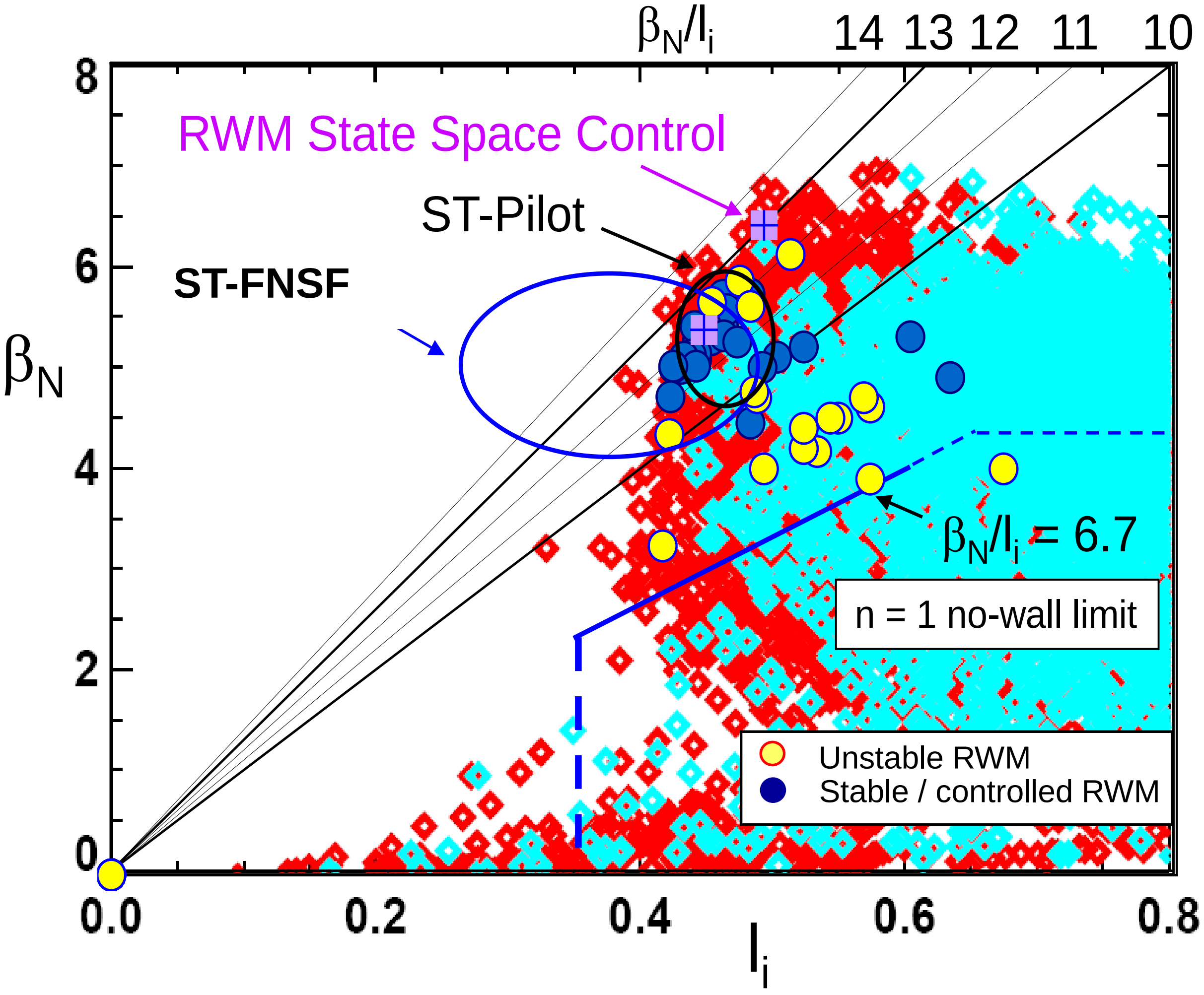
$$W_n \propto P_{fusion} / Area$$

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

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



Record β_N and β_N / l_i accessed in NSTX using 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.A. Sabbagh PRL(2006)

J. W. Berkery, PRL (2011)

W. Zhu, PRL (2006)

S.A. Sabbagh at this APS

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

Research opportunities

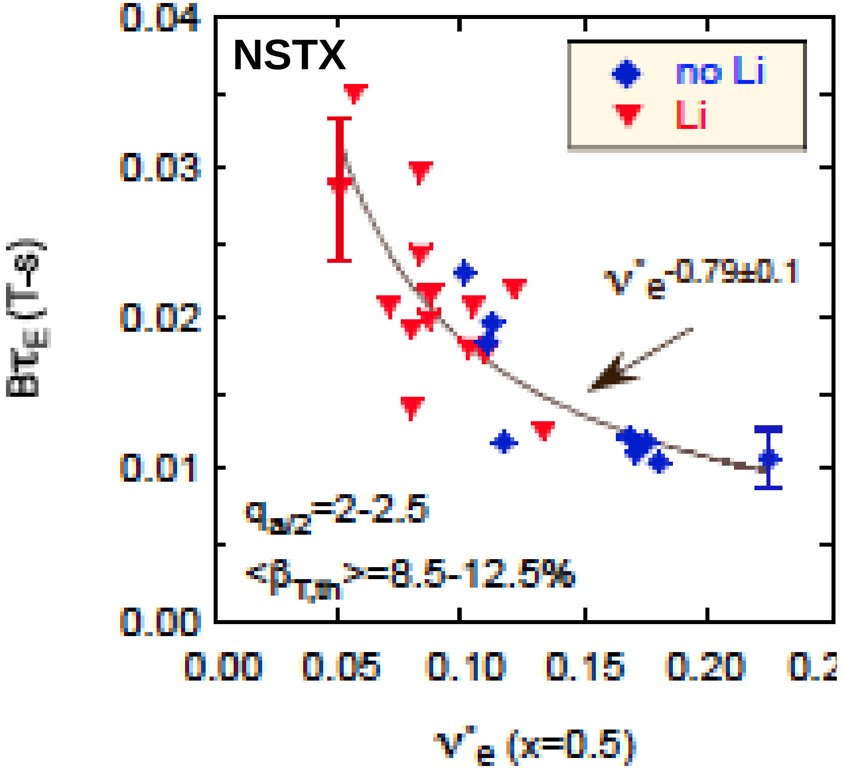
- How do RWM stability and rotation damping from neoclassical toroidal viscosity (NTV) change at the lower (ultimately up to 10x lower) collisionality values of NSTX-U?
  - Enhanced kinetic damping for RWM?
  - Enhanced rotation damping in $1/\nu$ regimes?
- What are leading causes of disruptions in ST? and what are implications for next steps?
  - NSTX-U will implement advanced profile and instability controls – how much does this reduce disruptivity?
  - NSTX-U will perform in-depth study/diagnosis of “halo/hiro” currents, i.e. edge plasma and wall currents and forces induced by disruptions

Favorable Confinement Trend with Collisionality and β found

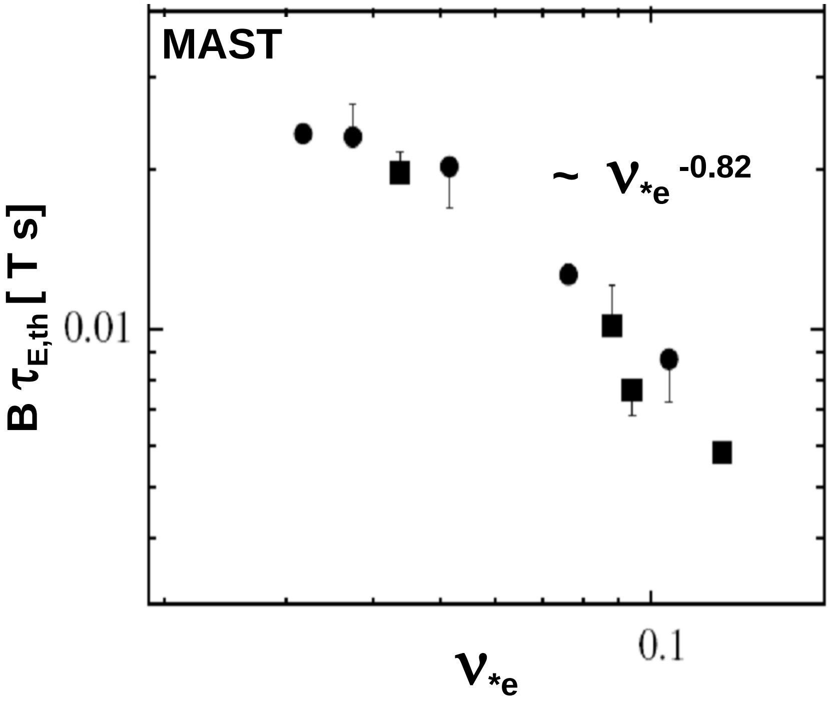
Important implications for future STs and Demo with much lower ν_*

$\tau_{E,\,th} \propto \nu_{*e}^{-0.1}\ \beta^{-0.9}$ **tokamak empirical scaling (ITER98$_{y,2}$)**

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



S.M. Kaye et al., NF (2007) (2013)

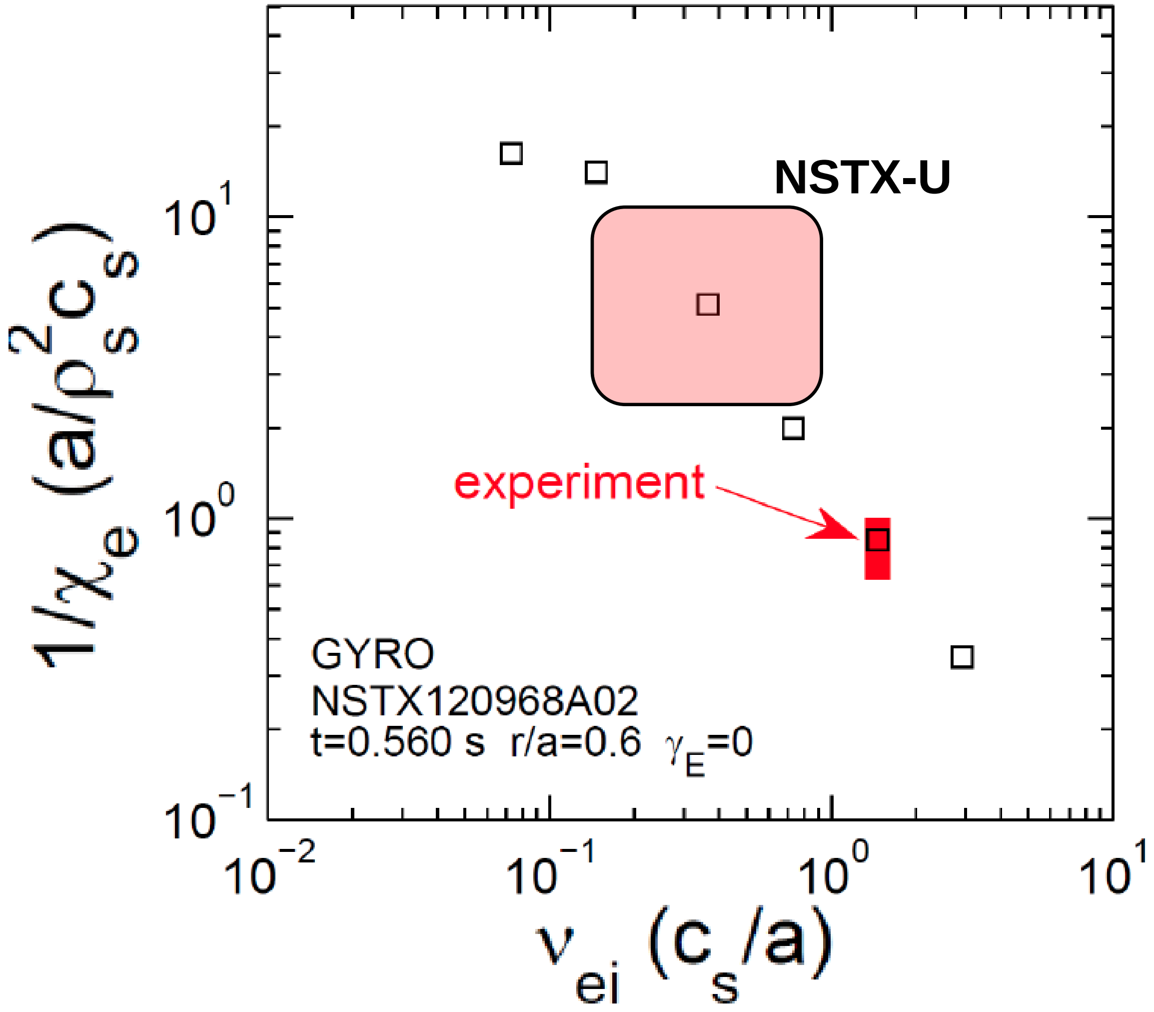


M. Valovic et al., NF (2011)

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

Microtearing-driven (MT) transport may explain ST collisionality scaling

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



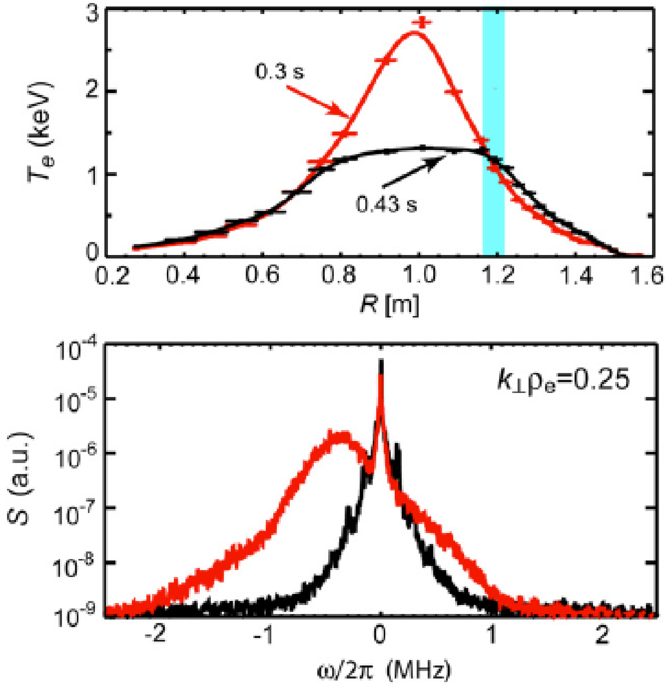
- **MT growth rate decreases with reduced collisionality in qualitative agreement with the NSTX experiment.**
- **Further electron confinement improvement expected due to reduced collisionality.**

W. Guttenfelder, et al., PoP(2012)

ETGs measured for the first time with high-k scattering

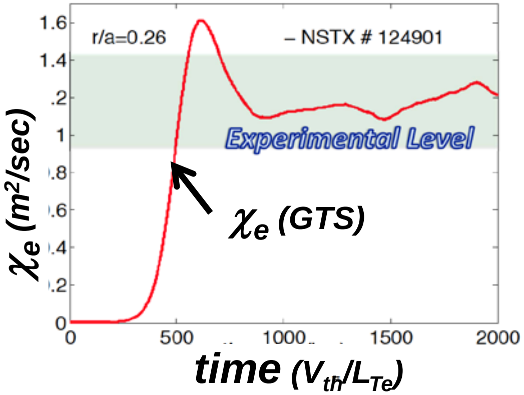
High β_e or larger $\rho_e \propto \beta_e^{0.5}$ of ST plasma enabled measurement of ETGs.

Electron Temperature Gradient Mode (ETG) Excitation with Core Electron Heating



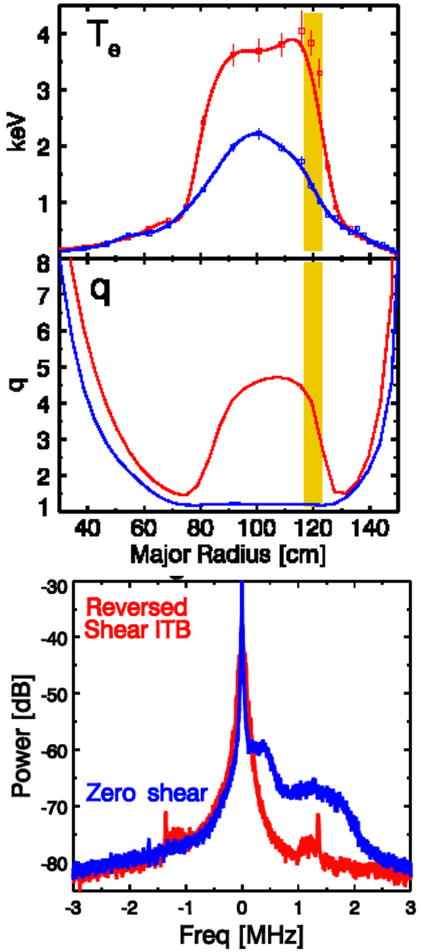
E. Mazzucato et al., NF (2009)

Calculated ETG χ_e by GTS code agrees with experiment



S. Ethier et al., IAEA (2010)

ETG Suppression in Reversed Shear



H.Y. Yuh et al., PoP (2009)
J.L. Peterson, et al., PoP(2011).

- **Shear stabilization of ETGs** D. R. Smith, et al., PRL (2009)
- **Density gradient stabilization of ETGs** Y. Ren, et al., PRL (2011)

Note: Here we call electron gyro-scale turbulence as ETGs

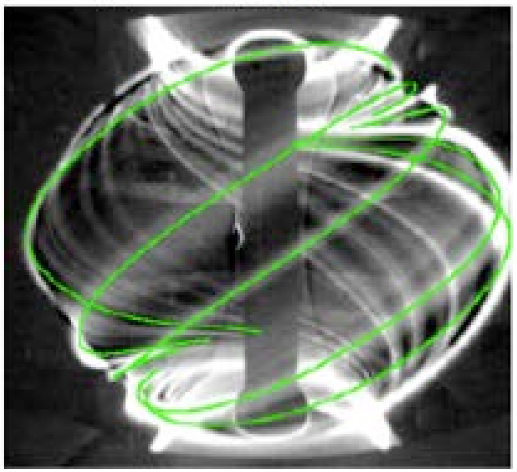
Research opportunities

- What are the leading causes of anomalous electron transport in the ST, and tokamaks generally?
  - Will electrostatic turbulence (ETG, TEM) dominate over micro-tearing ($\chi_e \sim \nu$) at lower collisionality values of NSTX-U?
- Will ion thermal and impurity transport remain predominantly neoclassical?
  - Neoclassical diffusivities also scale $\sim \nu$

H-mode / ELM physics: High Priority Research Goal

Unmitigated ELMs could cause PFC damage in reactors

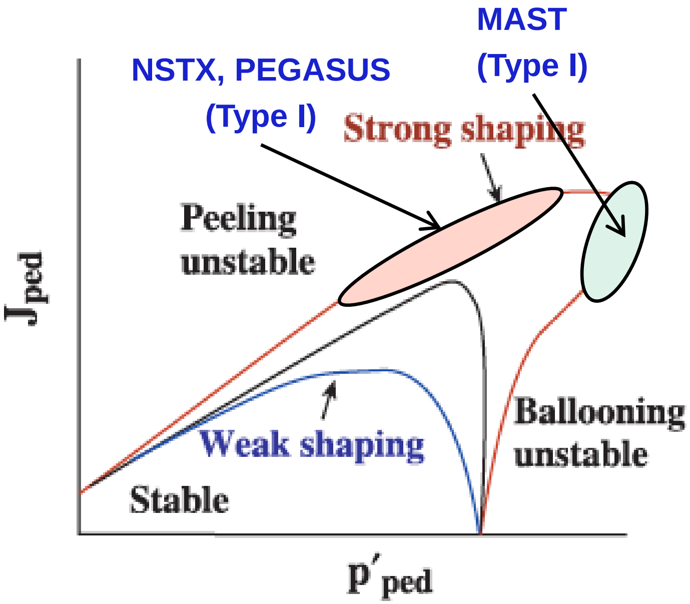
Video images of MAST plasmas showing a filamentary ELM structure.



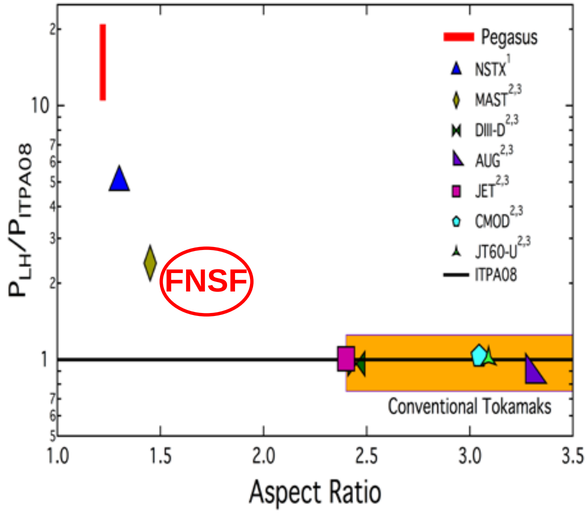
N. Ben Ayed et al., PPCF (2009).

ST is in strongly shaped ELM regimes

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



L-H power threshold scaling extended for low A



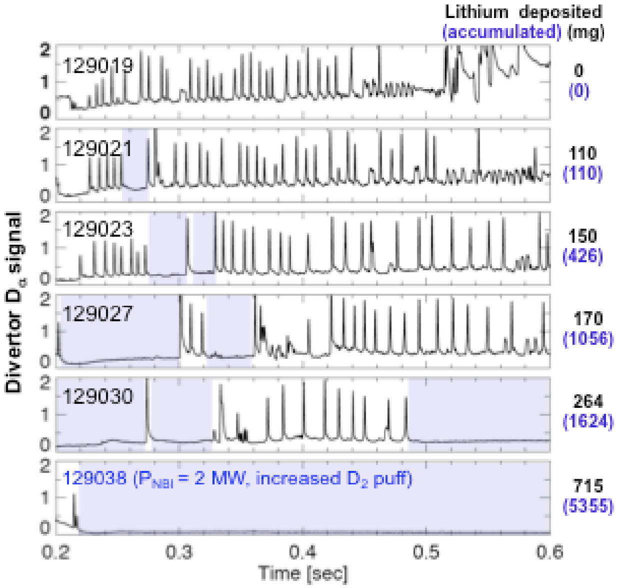
K.E. Thome et al., EPR (2014)

- **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**

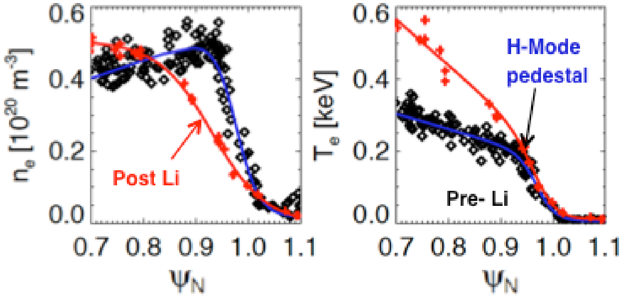
ELM Stabilization and Mitigation

Through application of lithium and 3-D fields

ELMs stabilized with edge pressure modification with Li in NSTX

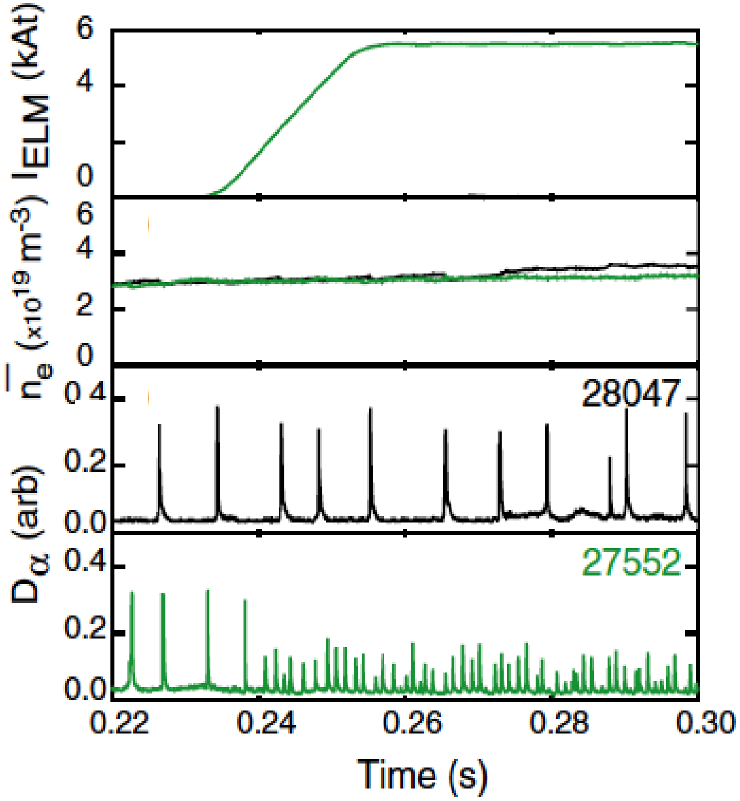


D.K. Mansfield, et al., JNM (2009)



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)

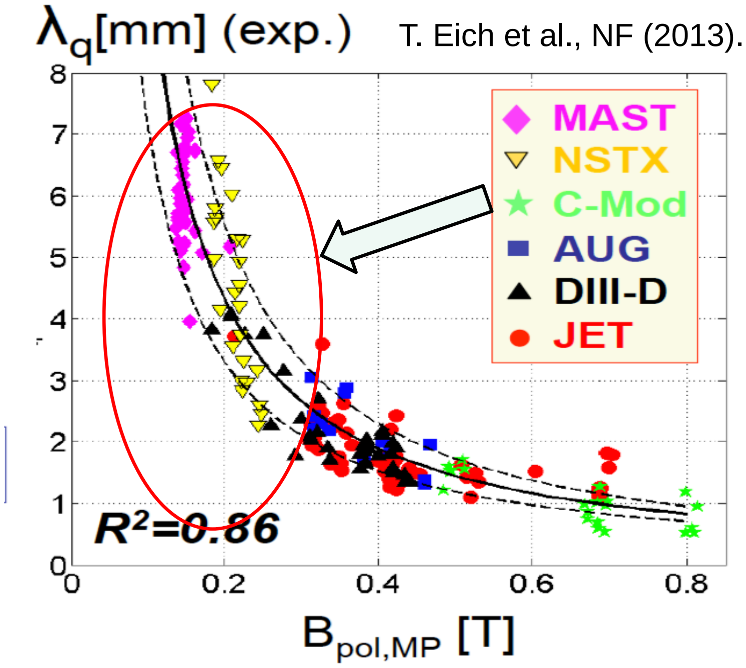
Research opportunities

- At higher aspect ratio and lower collisionality of NSTX-U, will ELMs remain predominantly $J_{||}$-driven?
- H-mode pedestal profiles determine overall fusion performance…. But what are dominant transport mechanism(s) in the H-mode pedestal region?
  - Neoclassical, kinetic ballooning, electron ∇T (ETG), other?
  - Ahmed Diallo’s Early Career Award with “burst” Thomson Scattering will help determine fast time evolution, transport
- Lithium coatings substantially increased NSTX global confinement (1.4x ITER H) – important for next-steps
  - How high can we make energy confinement in NSTX-U?
  - 2x ITER H-mode scaling? What sets the limit?

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.

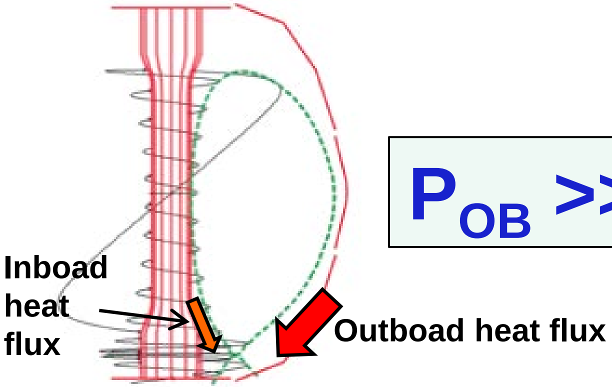


T. Eich et al., NF (2013).

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

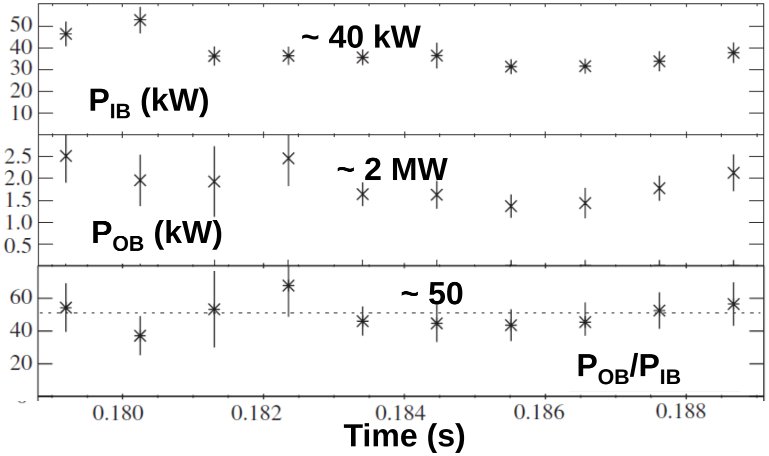
- * Unfavorable for large size, Ip devices such as ITER and Demo
- "P B / R" as the new heat flux metric which is favorable for STs

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



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

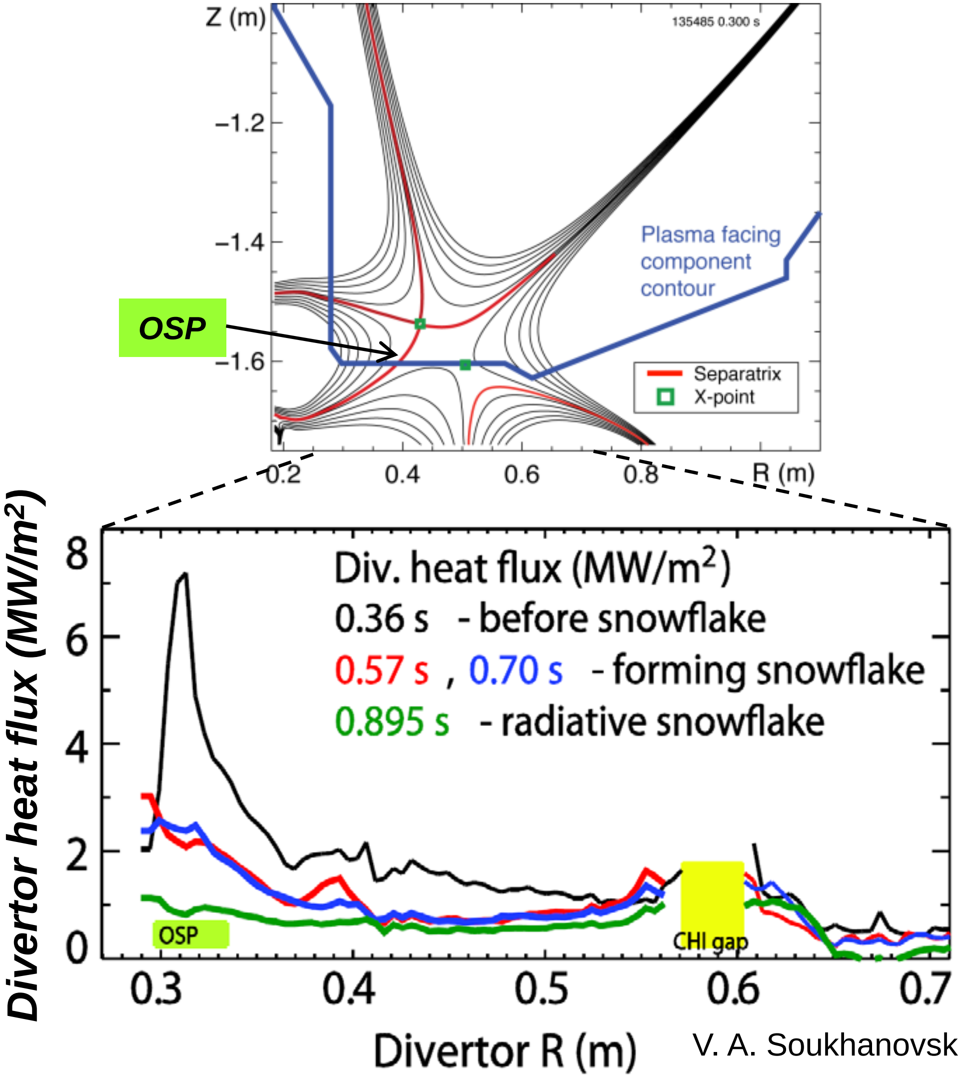
Ratio of outboard power flux vs. inboard in MAST



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**

Research opportunities

- What sets the heat flux width in the open field line region?
  - Mostly drifts + collisions (i.e. neoclassical) or does turbulence play a role?

- How does divertor geometry (such as snowflake) influence the plasma edge properties?

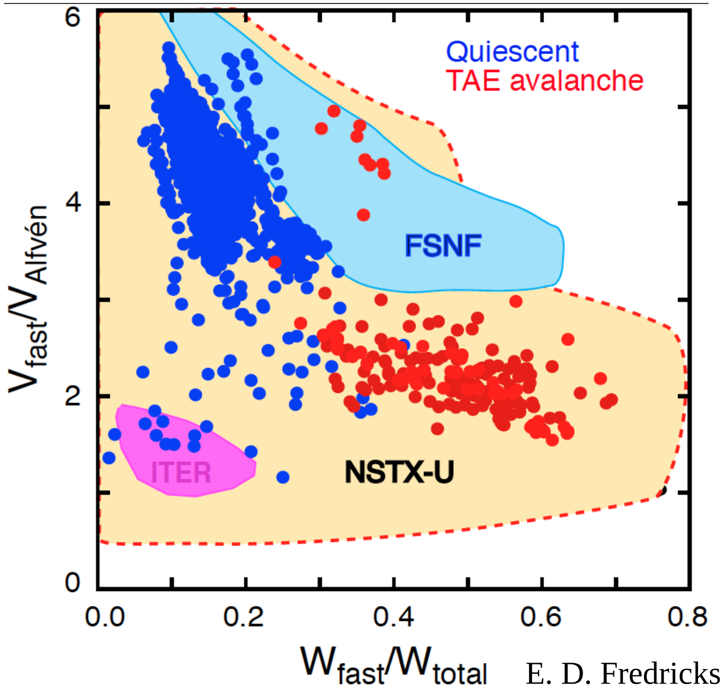
- How do Li and high-Z PFCs modify the edge region, and also the core performance?
  - NSTX-U will study “vapor shielding” regime to improve understanding of Li evaporation + radiation and impact on power exhaust / heat-flux mitigation / core performance

NBI heated ST plasmas provide an excellent testbed for α-particle physics

Alfvenic modes readily accessed due to high $V_\alpha > V_{Alf}$

- α-particles couples to Alfven-type mode strongly when $V_\alpha > V_{Alf} \sim \beta^{-0.5}$ Cs
- $V_\alpha > V_{Alf}$ in ITER and reactors
- In STs, the condition is easily satisfied due to high beta
- A prominent instabilities driven by fast particles are global and called toroidal Alfven eigenmodes (TAE).
- NSTX-U will also explore $V_\alpha < V_{Alf}$ regime giving more flexibility

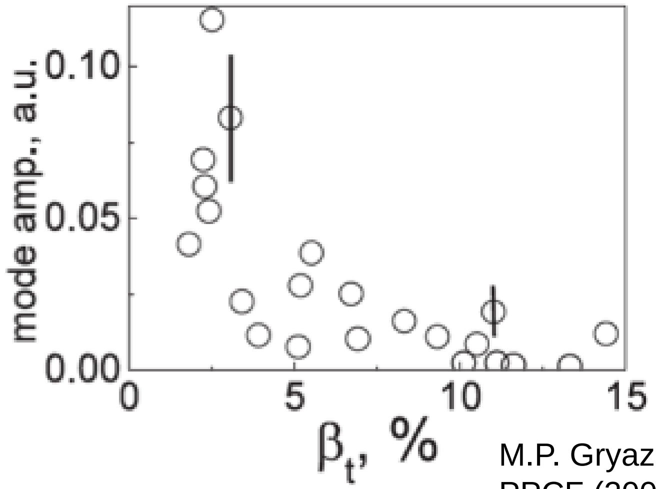
EP parameter space



E. D. Fredrickson et al., NF (2013)

TAEs significantly modified at high β as $V_{Alf} \rightarrow C_s$

Stabilization of TAEs at high β in MAST

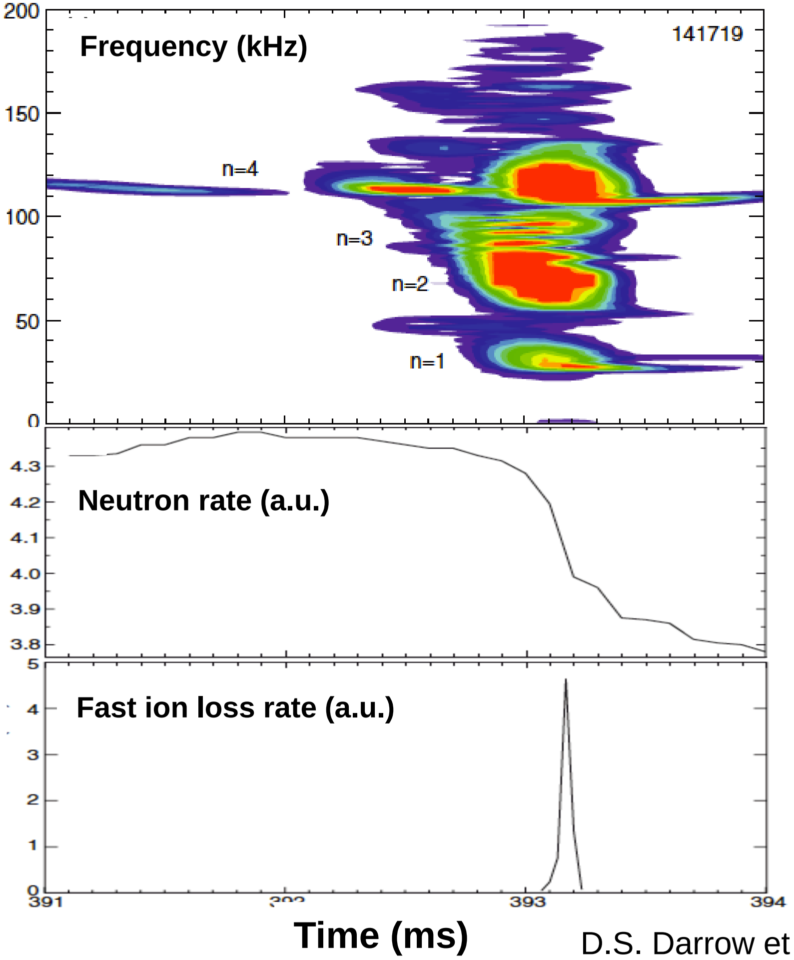


M.P. Gryaznevich et al. PPCF (2004)

"TAE avalanche" shown to cause energetic particle loss

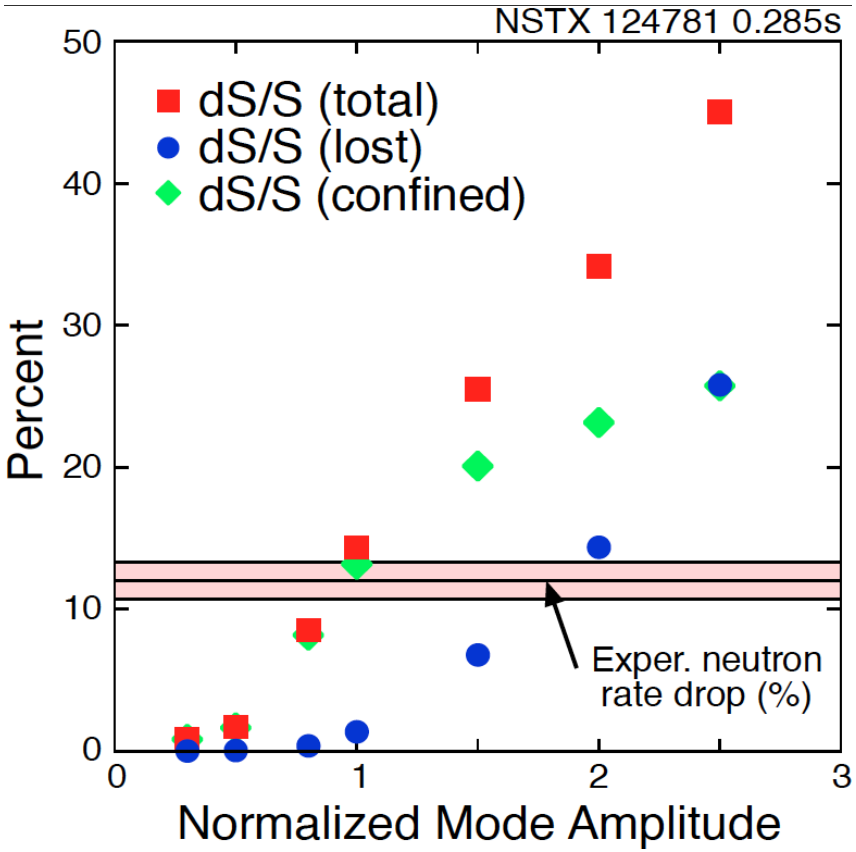
Uncontrolled α-particle loss could cause reactor first wall damage

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



D.S. Darrow et al., NF (2013).

Progress in simulation of neutron rate drop due to TAE avalanche



E. D. Fredrickson et al., NF (2013)

Research opportunities

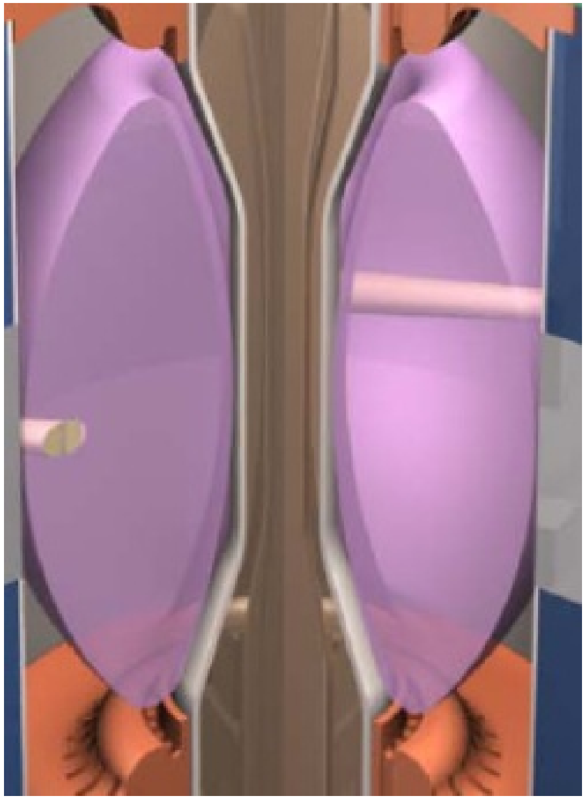
- Will reduced drive for fast-ion instabilities in NSTX-U reduce anomalous fast-ion transport?

- How will NSTX-U plasmas respond to more tangential neutral-beam / fast-ion injection?
  - Are current and momentum drive consistent with theory?

- What is role of high-f fast-ion instabilities (GAE/CAE) in core electron transport?
  - Is this physics unique to high-beta/ST?

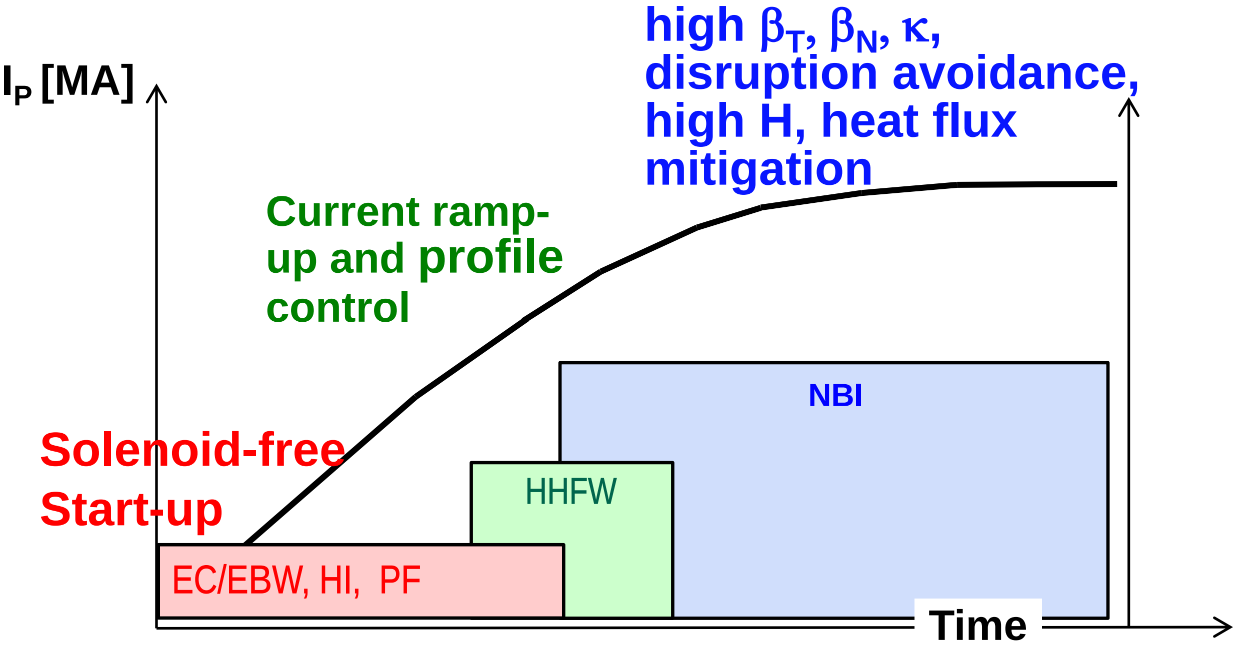
I_P Start-up/Ramp-up Critical Issue for ST-FNSF/Demo

Compact ST-FNSF has no/small central solenoid



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

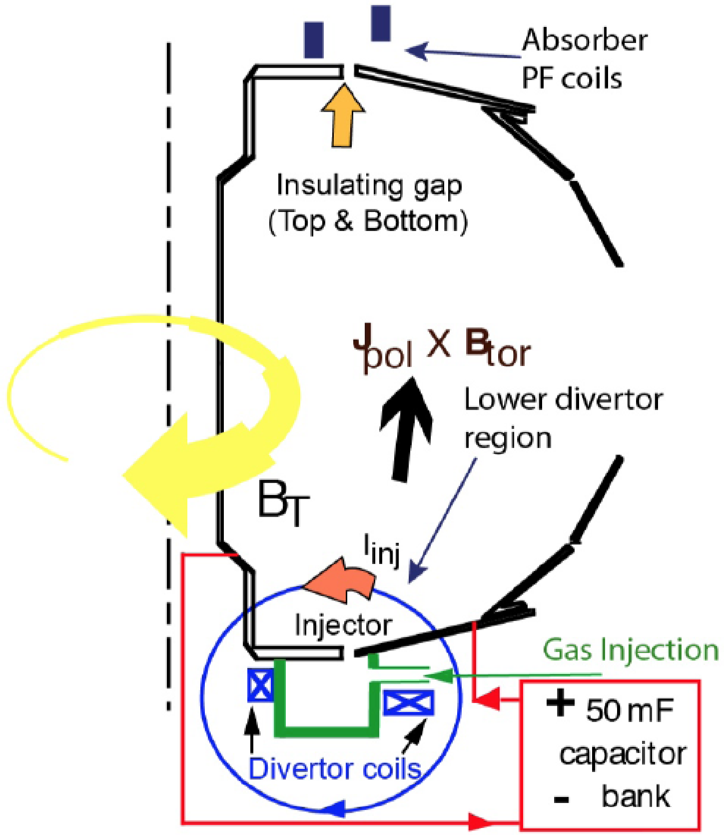
ST-FNSF Scenarios



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

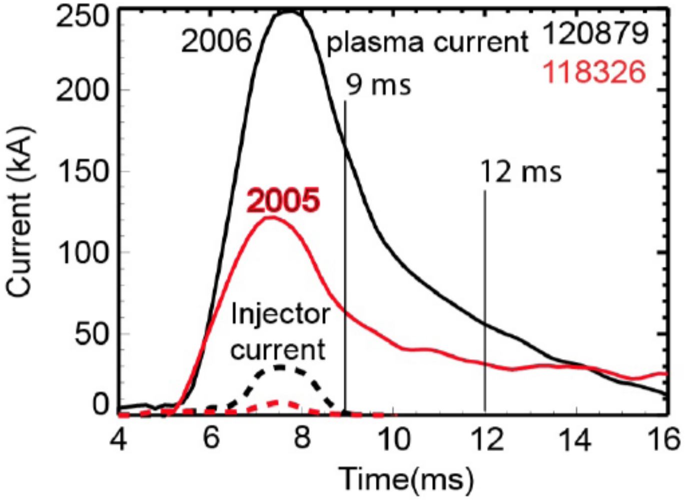
Helicity Injection Is an Efficient Method for Current Initiation
Coaxial Helicity Injection (CHI) Concepts Being Developed

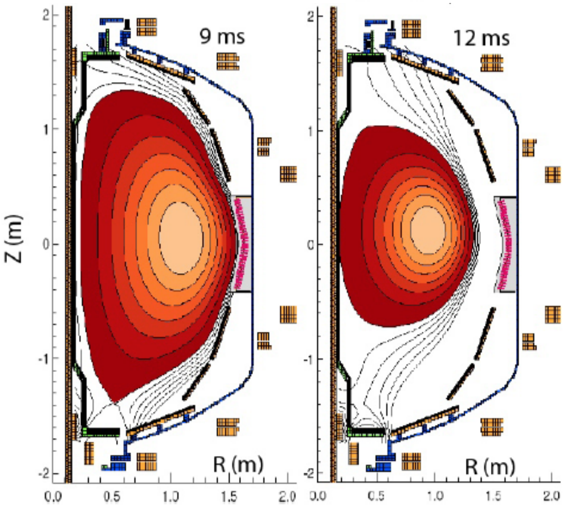
CHI developed on HIT and HIT-II and transferred to NSTX



R. Raman et al., PRL (2006)

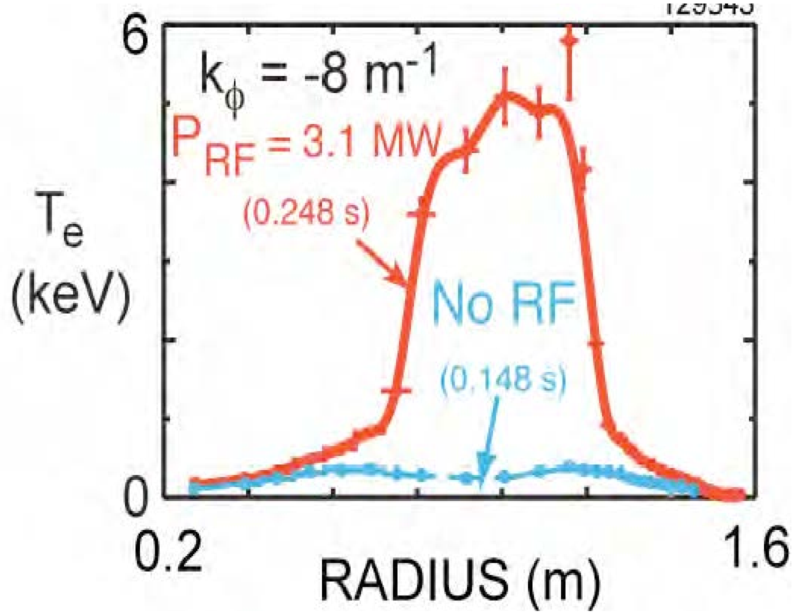
Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX





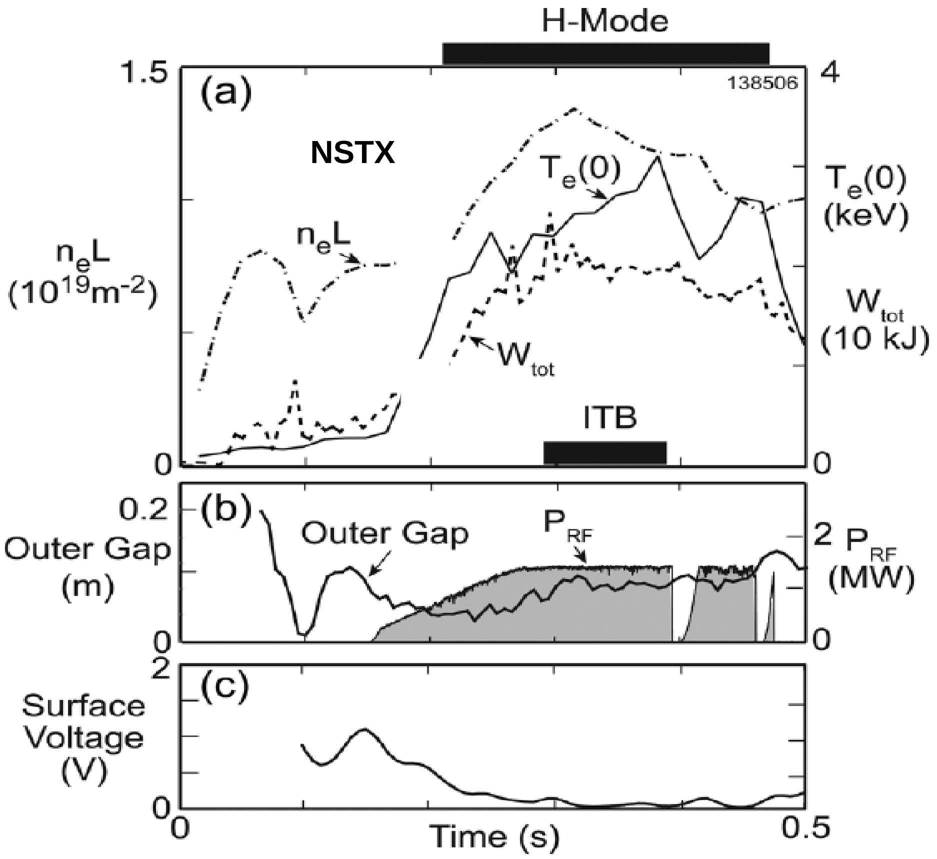
Current Ramp-Up and Profile Control Crucial for FNSF
Major Research Topics for NSTX-U

Efficient HHFW electron heating due to high β_e achieved in NSTX.



G. Taylor et al., PoP (2010), (2012)

Near sustained discharges obtained with modest HHFW power

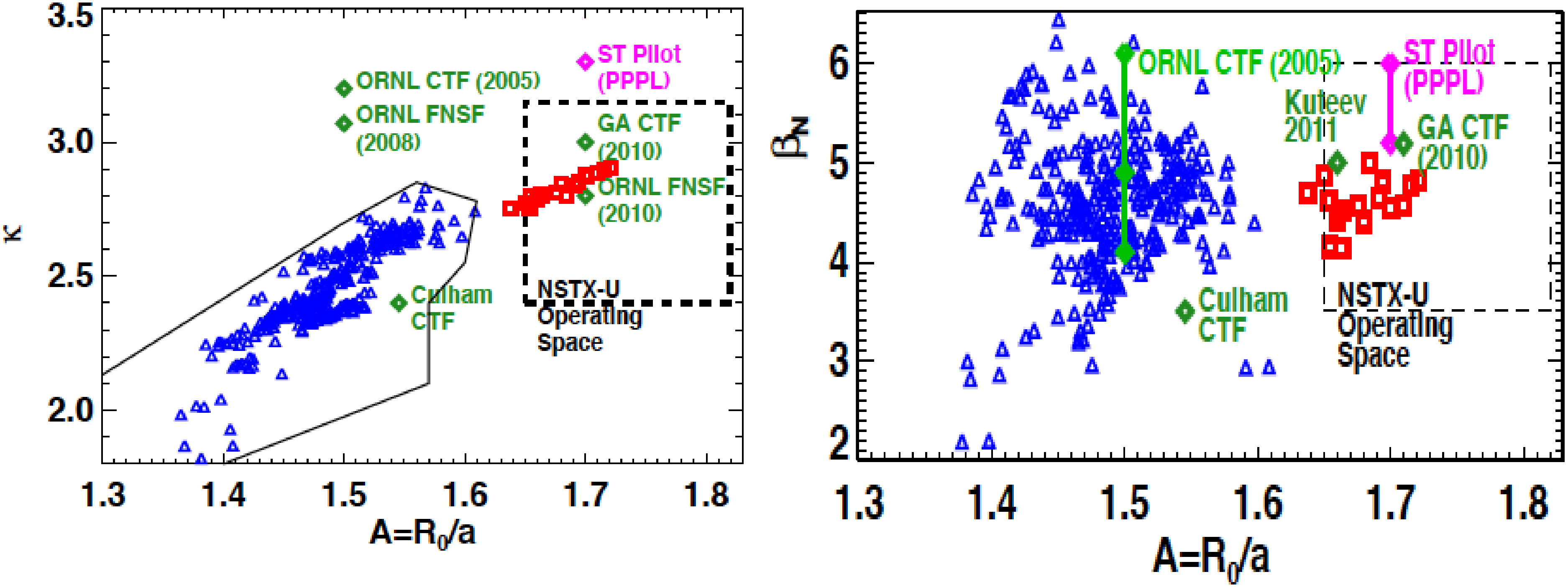


HHFW current ramp-up will be tested in NSTX-U at higher power ~ 4 MW.

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)**

Research opportunities

- Can NSTX-U really sustain all of its current without a transformer? If so, at what performance level?

- If NSTX-U succeeds at non-inductive sustainment, what level of non-inductive start-up and ramp-up is achievable?
  - Is it really possible to have a tokamak with NO solenoid?

- Can we develop simple yet powerful enough models to simulate all this accurately?
  - Complex combination of current drive, transport, stability (TRANSP, TSC, + other models)

Summary

- NSTX-U 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
- The opportunities described here are just a small fraction of the research possibilities available!
- Please see next slide for people to contact for more

Contacts for 1st/2nd year projects, thesis ideas

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Solenoid-free Start-up & Ramp-up
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Wave Heating and Current Drive
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Particle Control Task Force (FY2015-16)
Leader: R. Maingi, Deputy: J. Canik[1]

Goal: Develop pumping and fueling tools, operating scenarios, and control systems to achieve main-ion and impurity density control for long-pulse

Collaborators in Science Program Leadership:

1	ORNL
2	University of Wisconsin
3	LLNL
4	UT Knoxville
5	University of Illinois
6	Columbia University
7	Princeton University
8	Johns Hopkins University
9	UC Irvine
10	UC Los Angeles
11	University of Washington
12	MIT