Recent Progress on Spherical Torus Research and Implications for Fusion Energy Development Path

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On behalf on the world ST community!

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Acknowledgements

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U Tennessee
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*U Wisconsin

* With ST Facility

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IPP, Garching
ASCR, Czech Rep

- At present, over 500 researchers and 140 graduate students are engaged in ST research worldwide.
- Over 1,000 ST related refereed publications since 2000.

For more detail, M. Ono and R. Kaita, ST review paper for PoP

Talk Outline

- **Unique ST properties**

- ST Fusion Energy Development Path

- World ST Facilities

- Unique ST Physics Regimes

- ST-FNSF Relevant Experiments
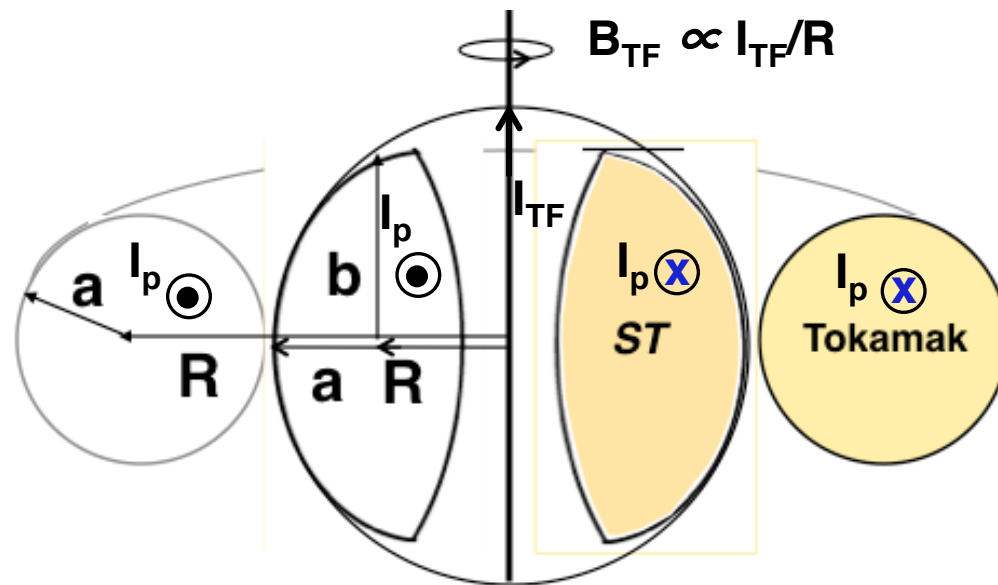
- ST Facility Upgrade Status

- Summary

ST is a low aspect ratio tokamak with A < 2
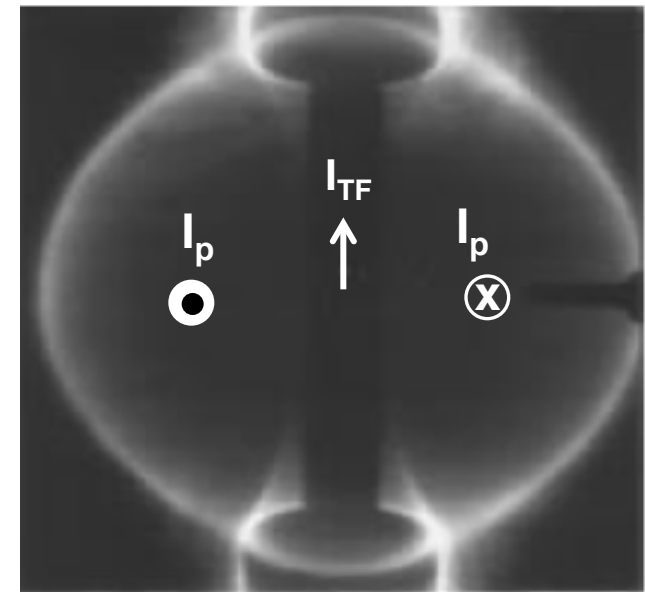## Natural elongation makes its spherical appearance

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

$B_{TF} \propto I_{TF}/R$

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
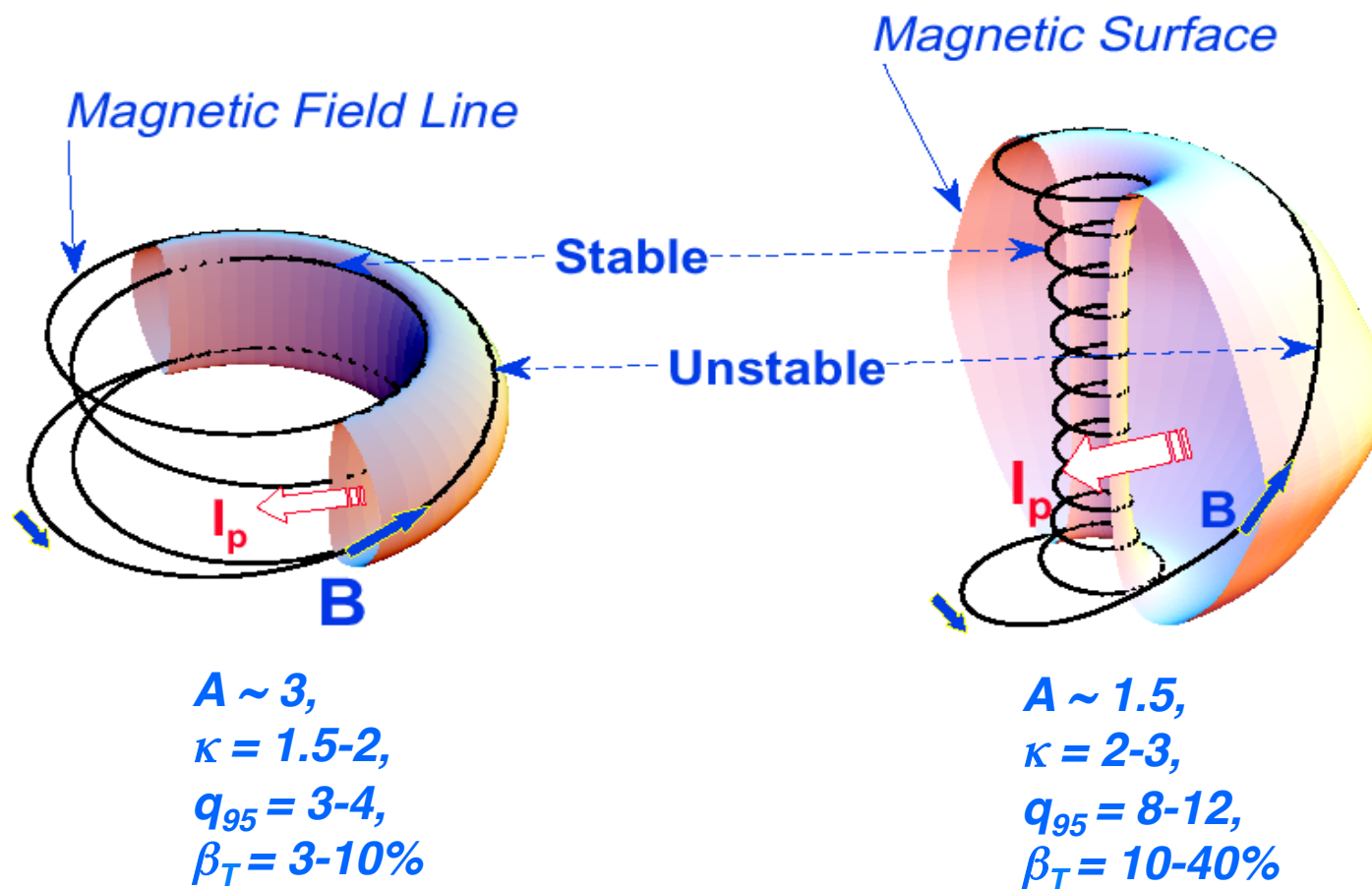## Favorable average curvature improves stability at high beta

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

Tokamak

Magnetic Field Line

Stable

Unstable

I_p

B

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

ST

Magnetic Surface

I_p

B

$A \sim 1.5,$
$\kappa = 2\text{-}3,$
$q_{95} = 8\text{-}12,$
$\beta_T = 10\text{-}40\%$

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)
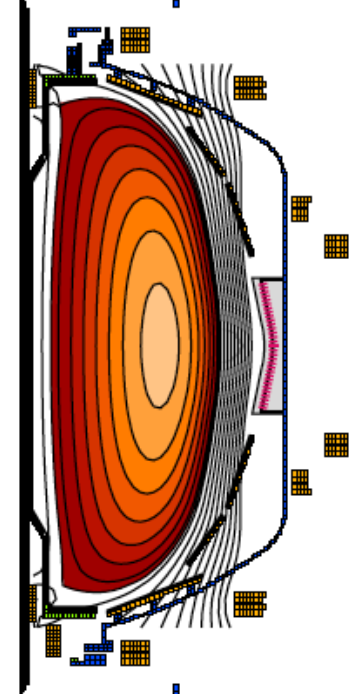
- Ip increases tokamak performance

$$\tau_E \propto I_p$$

$$\beta_T \equiv \beta_N \, I_p / (a B_{T0})$$

- ST can achieve high performance cost effectively

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

⬆

\$

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, EP-modes, RF heating and CD**

- **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**

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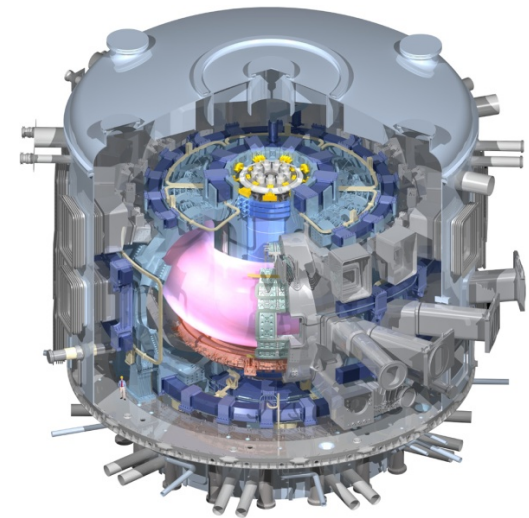
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Unique ST properties support and accelerate a range of development paths toward fusion energy

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

STs Narrow Gaps to FNSF/Pilot/DEMO:

Goal: 100% non-inductive + high β

Plasma-Material Interface Research

Strong heating + smaller R $\rightarrow$ high P/R, P/S

Novel solutions: snowflake, liquid metals, Super-X, hot high-Z walls

Enable Compact Fusion Nuclear Science Facility

High neutron wall loading

Potentially smaller size, cost

Smaller tritium (T) consumption at fixed neutron wall loading

Accessible / maintainable

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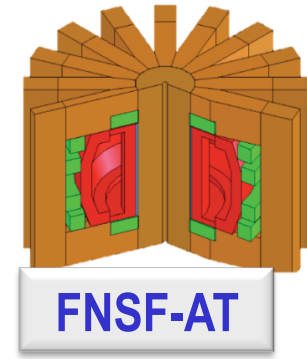
- **Without R&D, fusion components could fail prematurely which often requires long repair/down time. This would cripple the DEMO operation.**
- **FNSF can help develop reliable fusion components.**
- **Such FNSF facilities must be modest cost, low T, and reliable.**

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

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
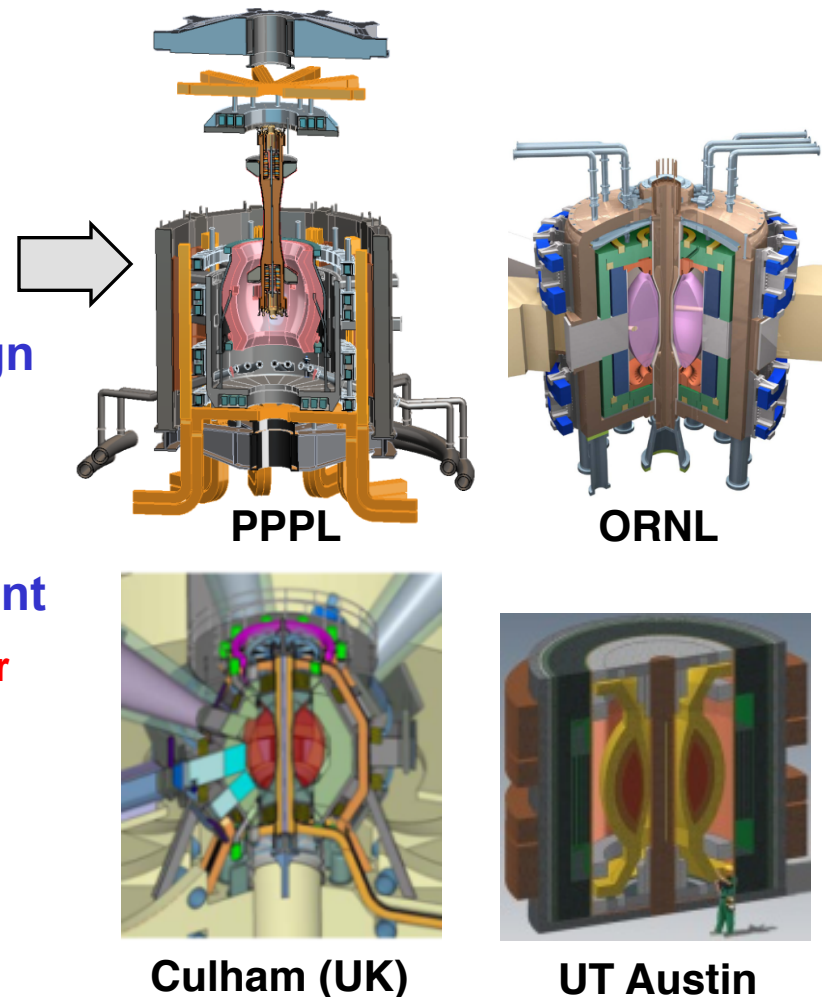
PPPL

ORNL

Culham (UK)

UT Austin

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- **Unique ST properties**

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- **Summary**

Operating ST Research Facilities Since 2000

## NSTX and MAST: MA-class STs, Smaller STs addressing topical issues

NSTX, USA

$R_0 \sim 0.85$ m

PEGASUS, USA

LTX/CDX-U, USA

GLOBUS-M, Russia

MAST, UK

$R_0 \sim 0.75$ m

HIST, Japan

LATE, Japan

TST-2, Japan

UTST, Japan

QUEST/CPD, Japan

VEST, Korea

HIT-II, USA

SNIST, China

ETE, Brazil

ST25, UK

TS3/4, Japan

HIT-II   PEGASUS

NSTX/LTX

MAST
ST25(HTS)

GLOBUS-M

KTM

SUNIST

QUEST
VESTHIST
LATE
TST-2
TS3/4
UTST

ETE

⭐ STs operated since 2000
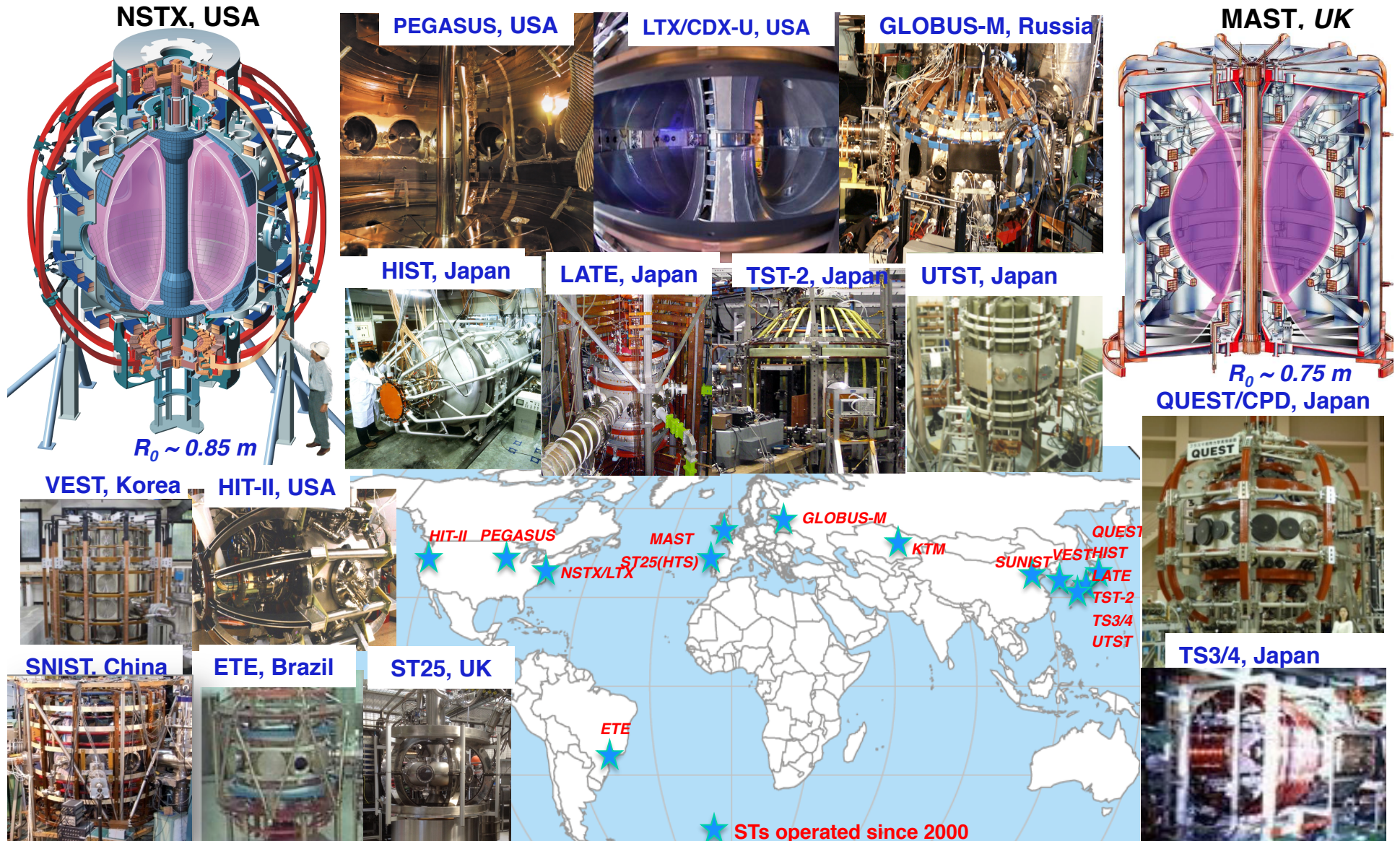
MA-Class ST Research Started in 2000
Complementary Physics Capabilities of NSTX and MAST

NSTX
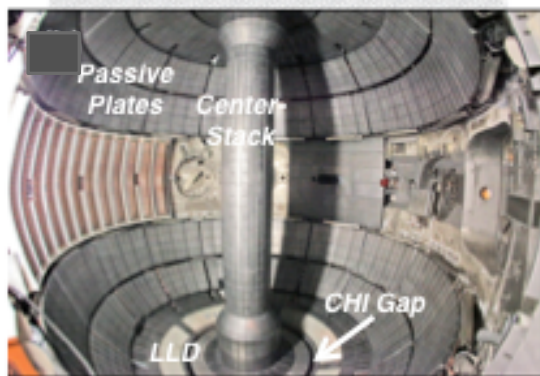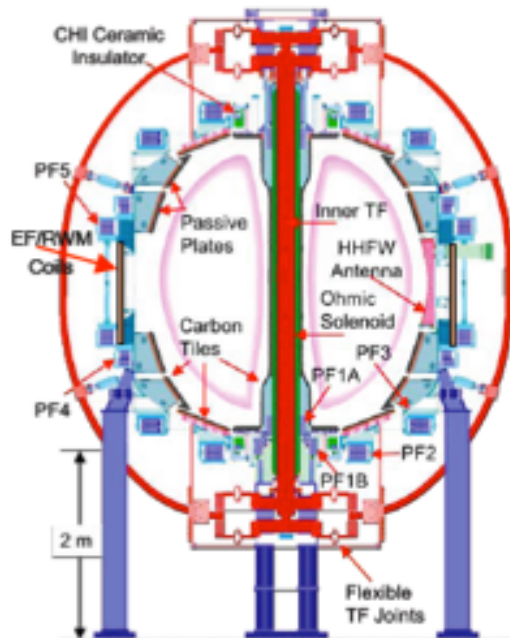
Complementary Capabilities

Passive Plates	Large divertor volume
Heilicity Injection	Merging/Compression
HHFW	ECH
1 x 6 RWM Coils	2 x 12 ELM coils

MAST

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	$B_T \sim$ 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**
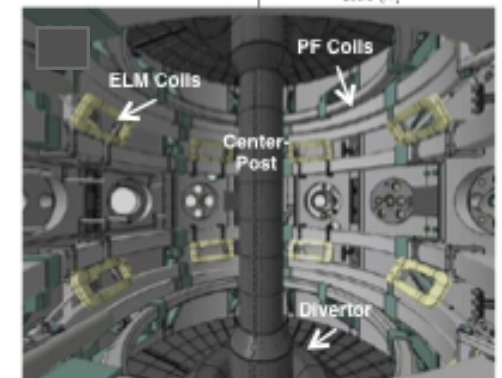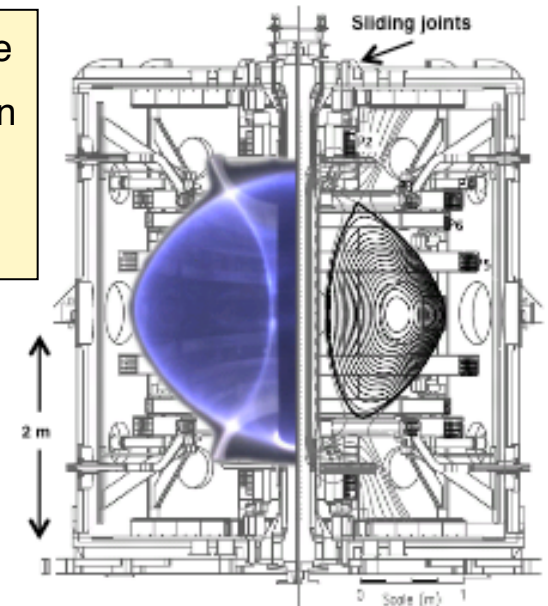
M. Ono, et al., IAEA 2000, NF 2001

A. Sykes, et al., IAEA 2000, NF 2001

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

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

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

Product **Physics Knob** **$ Investment**
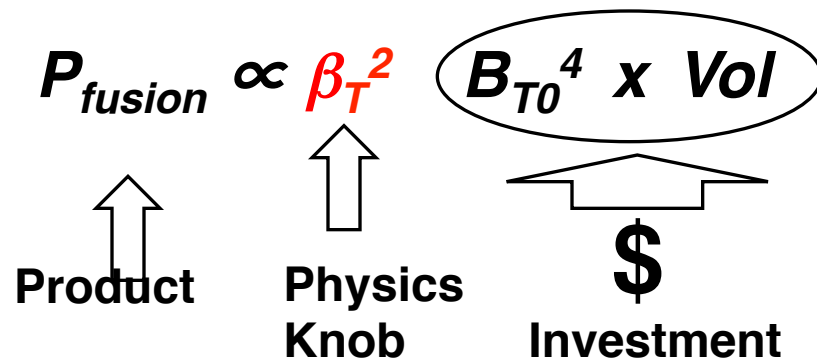
High neutron wall loading W_n possible in a compact ST

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

$$W_n \propto \beta_T^2 \, B_{T0}^4 \, a \quad \text{(not strongly size dependent)}$$
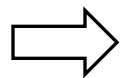
$\Rightarrow$ $W_n \sim 1 \, MW/m^2$ with $R \sim 1 \, m$ FNSF feasible!

Data base from START, MAST, PEGASUS and NSTX

$\beta_N = 6$

5

4

3

2

ST-FNSF

Tokamak

ITER

Toroidal β (%)

I_p/aB_{T0} (MA/mT)
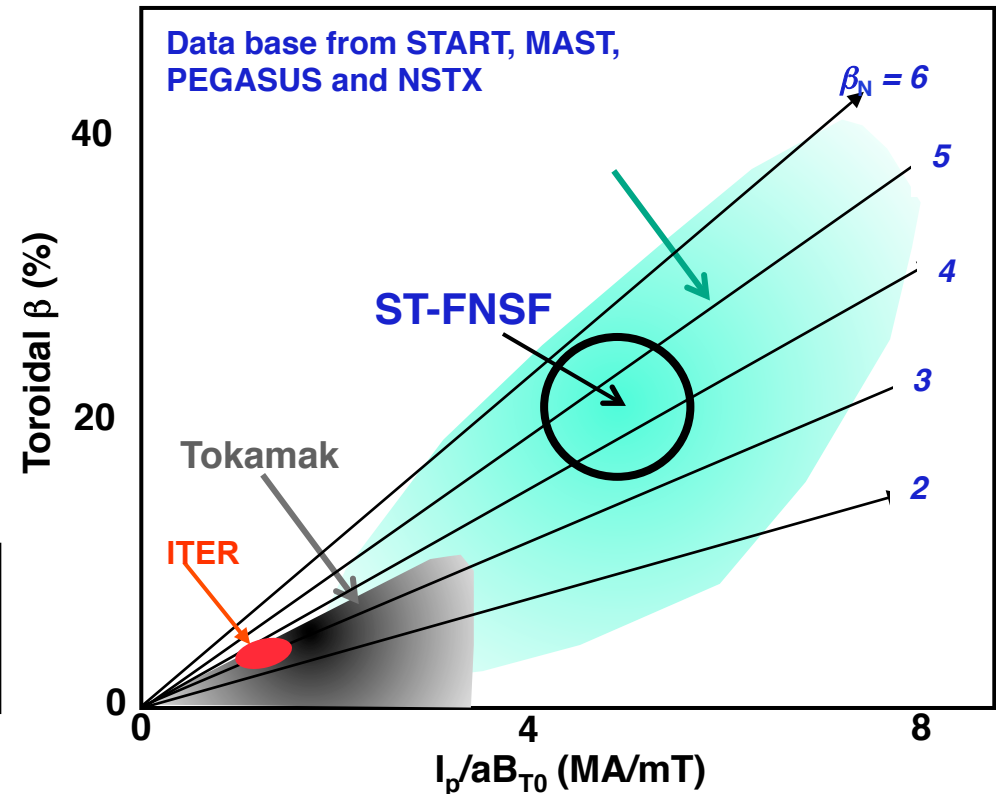
Record β_N and β_N/l_i accessed using resistive wall mode stabilization
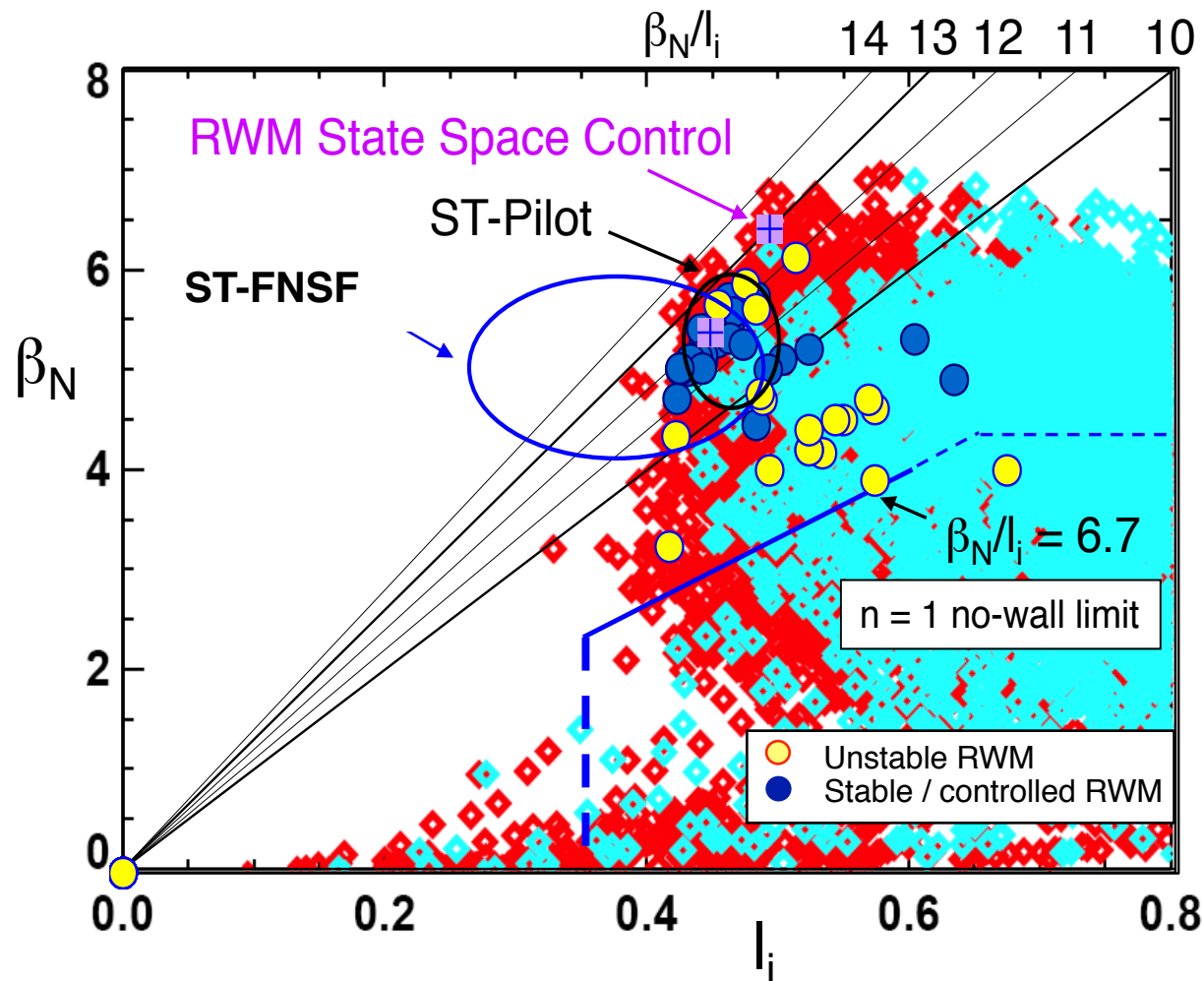
- *High β_N regime is important for bootstrap current generation.*

- *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

Favorable Confinement Trend with Collisionality and β found
Important implications for future STs and Demo with much lower ν_*

$$\tau_{E, th} \propto \nu_*^{-0.1} \beta^{-0.9} \quad \text{tokamak empirical scaling (ITER98}_{y,2})$$
$$\tau_{E, th} \propto \nu_*^{-0.8} \beta^{-0.0} \quad \text{ST scaling}$$

NSTX

no Li
Li

$\nu^*_e{}^{-0.79\pm0.1}$

$q_{a/2}=2-2.5$
$<\beta_{T,th}>=8.5-12.5\%$

$B\tau_E$ (T-s)

ν^*_e (x=0.5)

MAST

$\sim \nu_{*e}^{-0.82}$

$B\tau_{E,th}$ [T s]

ν_{*e}
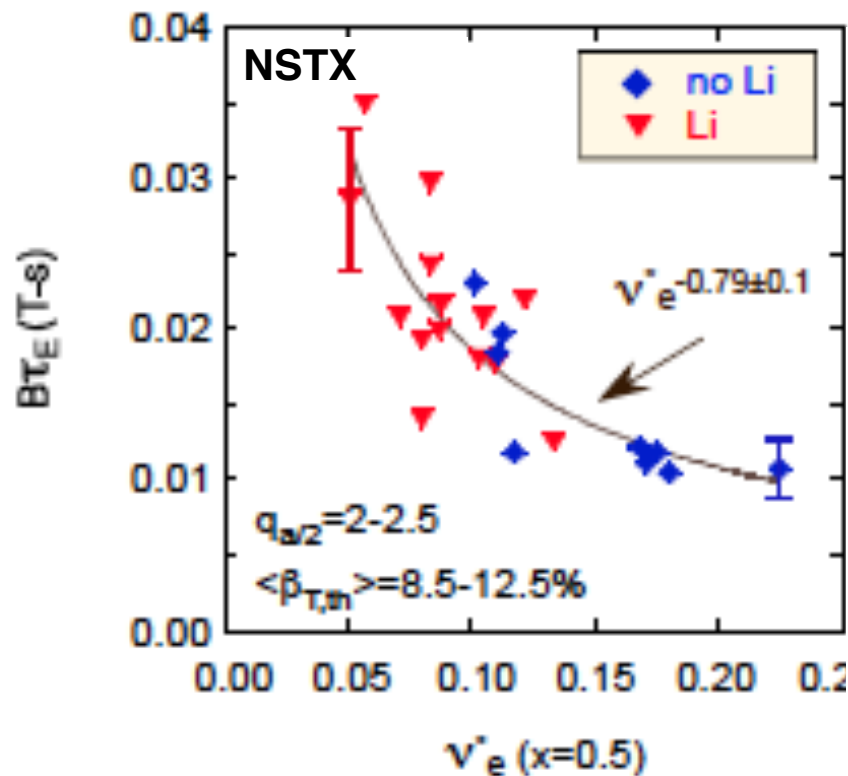
S.M. Kaye et al., NF (2007) (2013)
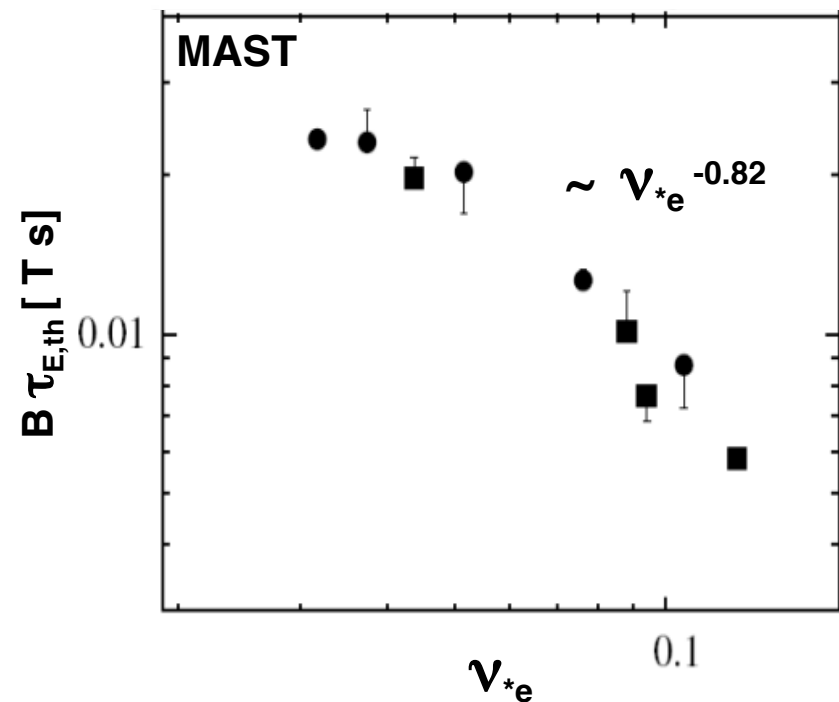
M. Valovic et al., NF (2011)

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.
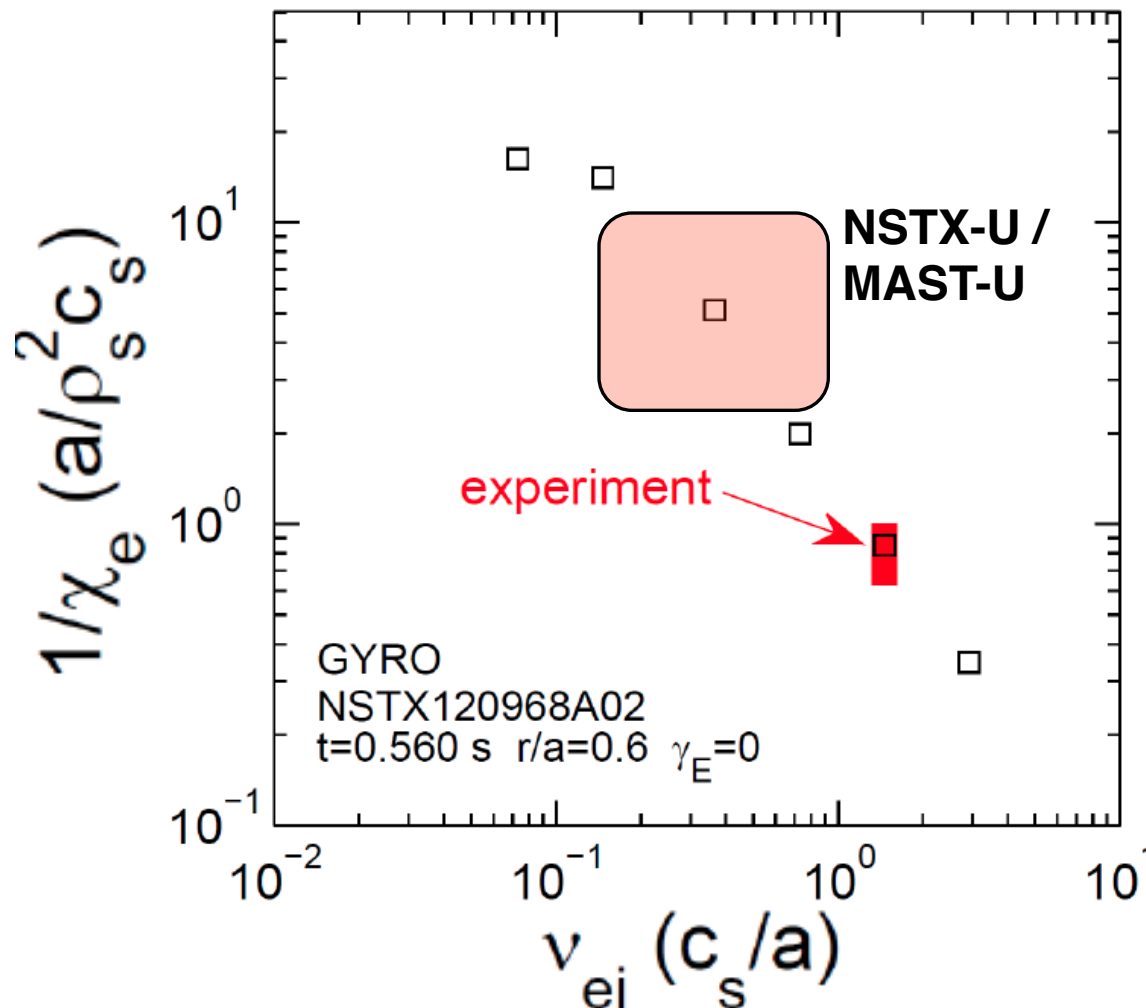
- **MT growth rate decreases with reduced collisionality in qualitative agreement with the NSTX experiment.**

- **Futher electron confinement improvement expected for NSTX-U and MAST-U 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
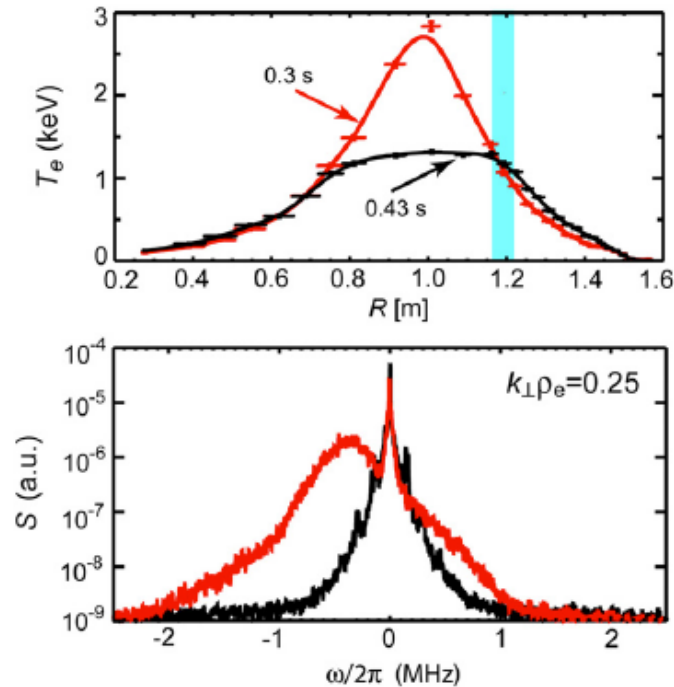
E. Mazzucato et al., NF (2009)

Calculated ETG χ_e by GTS code agrees with experiment
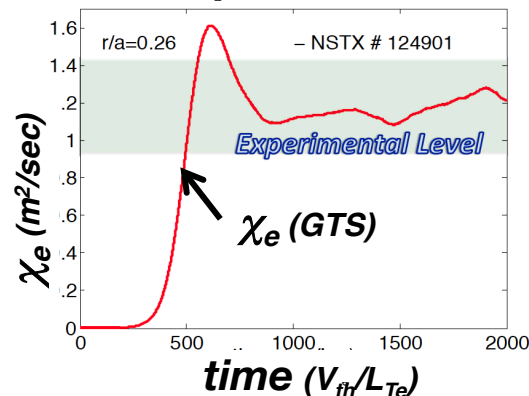
S. Ethier et al., IAEA (2010)

ETG Suppression in Reversed Shear
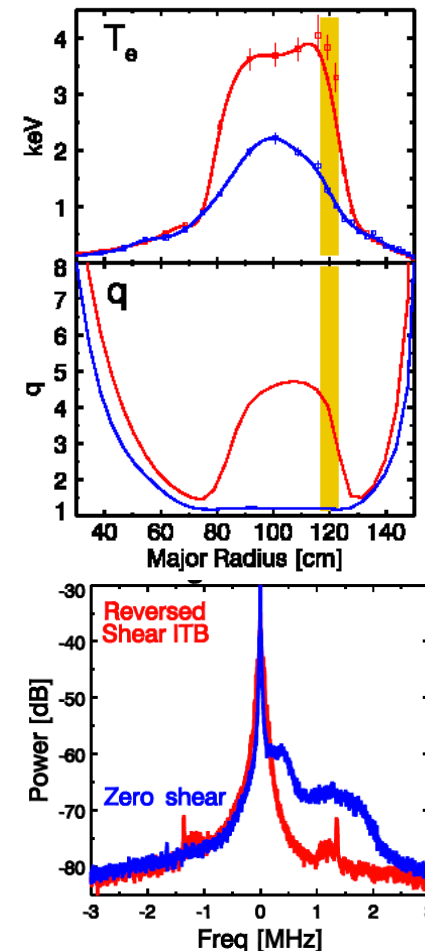
H.Y. Yuh et al., PoP (2009)
J.L. Peterson, et al., PoP(2011).

· **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

H-mode / ELM physics: High Priority Research Goal
Unmitigated ELMs could cause PFC damage in reactors

ST is in strongly shaped ELM regimes

P.B. Snyder et al., PoP (2002).
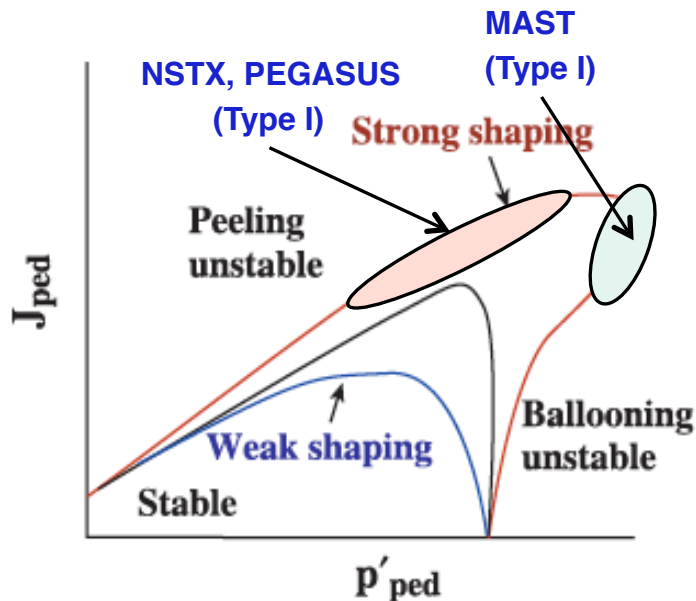
Video images of MAST plasmas showing a filamentary ELM structure.
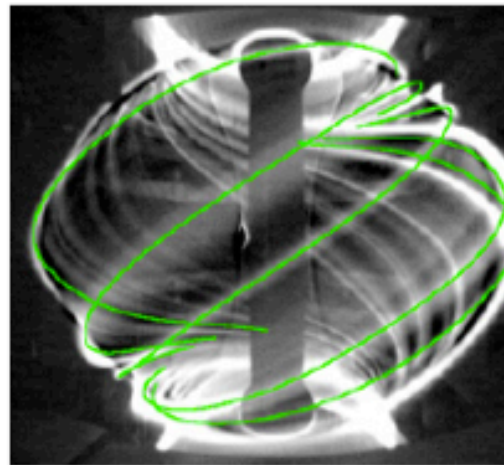
N. Ben Ayed et al., PPCF (2009).

L-H power threshold scaling extended for low A
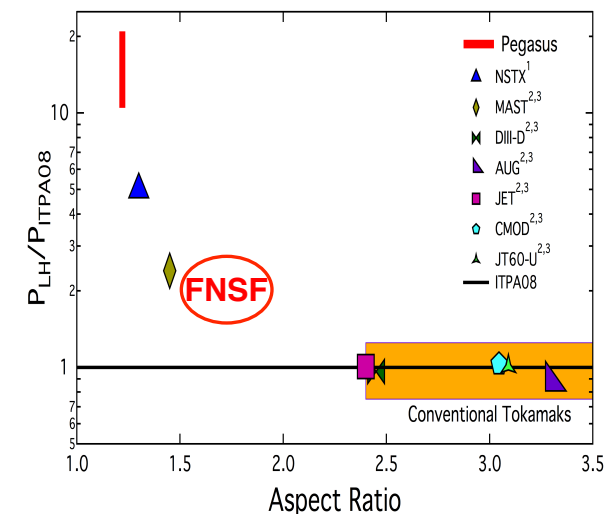
K.E. Thome et al., EPR (2014)

- **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**

Divertor heat flux in Low-A regime
ST power flux width clearly shows 1/Bp variation

STs data breaks A degeneracy of power flux width study.

λ_q[mm] (exp.) T. Eich et al., NF (2013).

Legend:
- ◆ MAST
- ▽ NSTX
- ★ C-Mod
- ■ AUG
- ▲ DIII-D
- ● JET

$R^2 = 0.86$

$B_{pol,MP}$ [T]
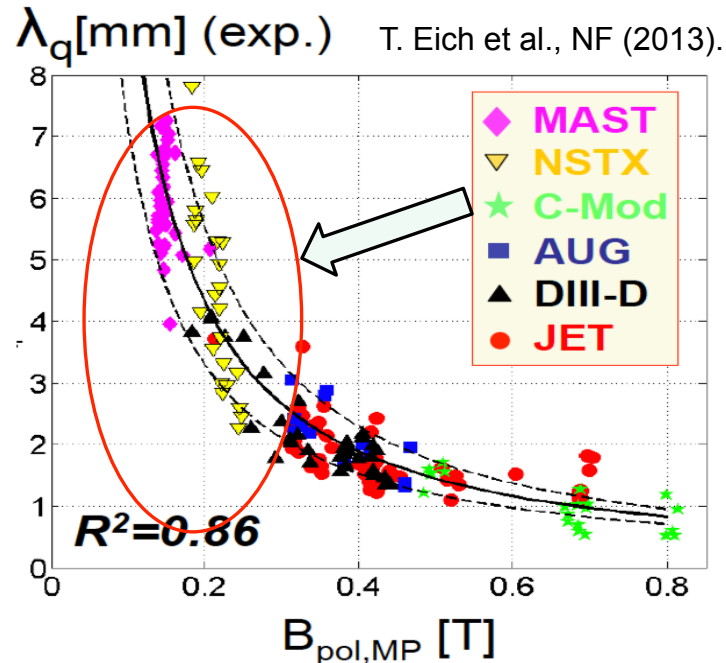
Heuristic model by R.J. Goldston, NF (2012).

* **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!

$$P_{OB} \gg P_{IB}$$

Inboad heat flux

Outboad heat flux
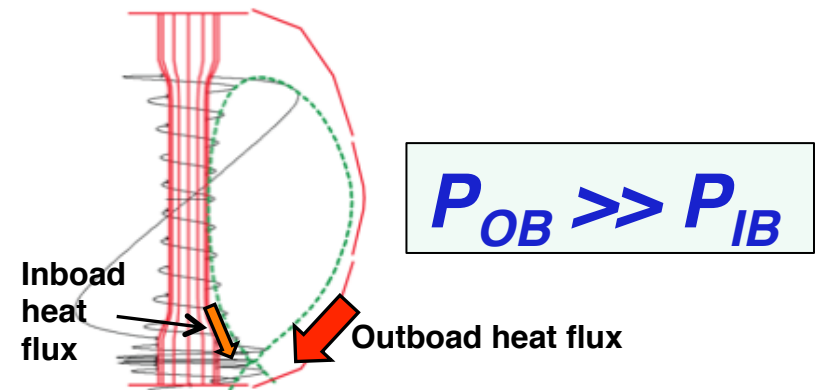
Ratio of outboard power flux vs. inboard in MAST

~ 40 kW

P_{IB} (kW)

~ 2 MW

P_{OB} (kW)

~ 50

P_{OB}/P_{IB}

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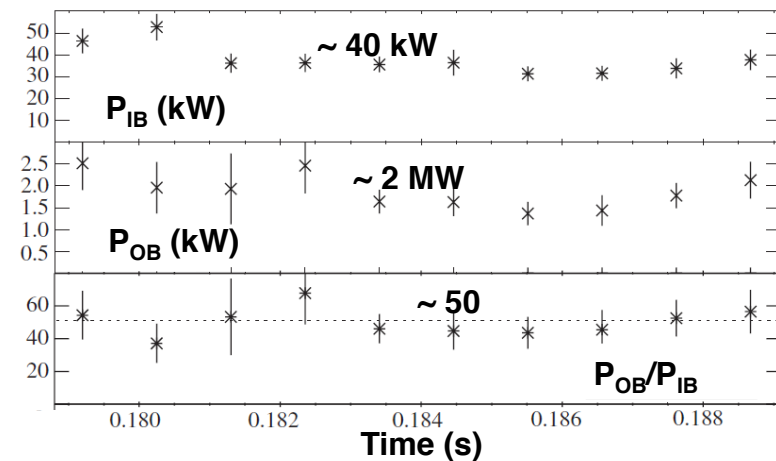
G.F. Counsell et al., PPCF (2002)

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 $V_\alpha > V_{Alf} \sim \beta^{-0.5} C_s$
- $V_\alpha > V_{Alf}$ 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/MAST-U will also explore $V_\alpha < V_{Alf}$ regime giving more flexibility

EP parameter space

E. D. Fredrickson et al., NF (2013)
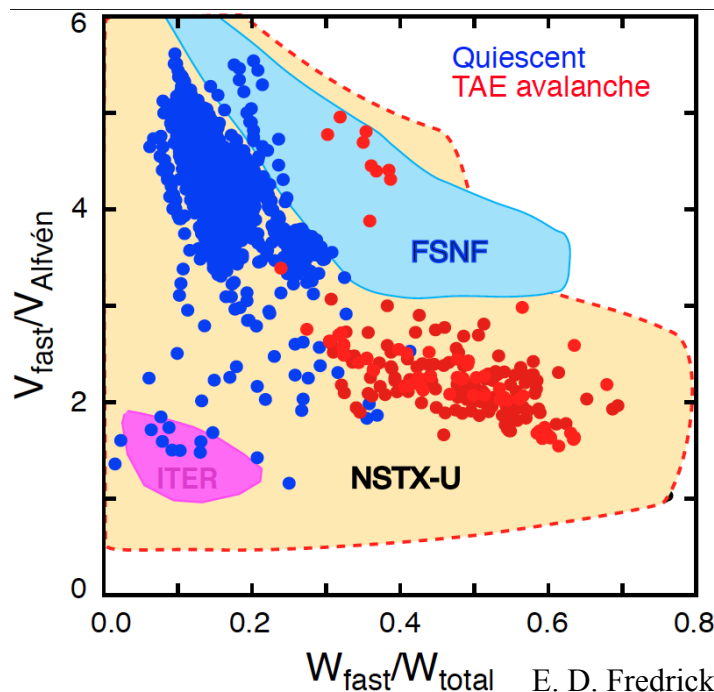
TAEs significantly modified at high β as $V_{Alf} \longrightarrow C_s$

Stabilization of TAEs at high β in MAST

M.P. Gryaznevich et al. PPCF (2004)
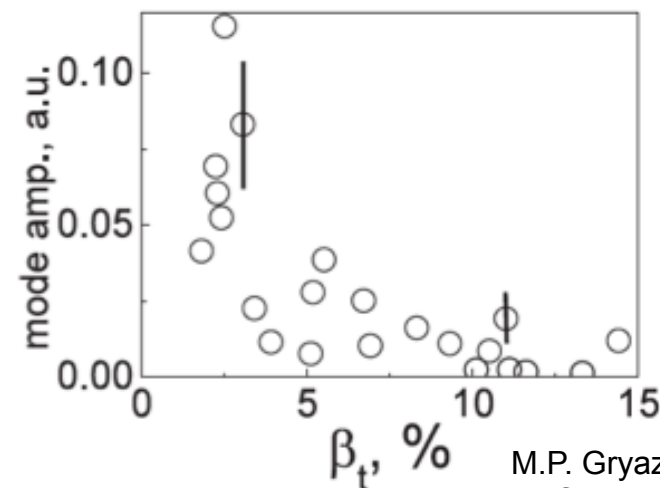
"TAE avalanche" shown to cause energetic particle loss
Uncontrolled α-particle loss could cause reactor first wall damage

Multi-mode TAE avalanche can cause significant EP losses as in "sea" of TAEs expected in ITER
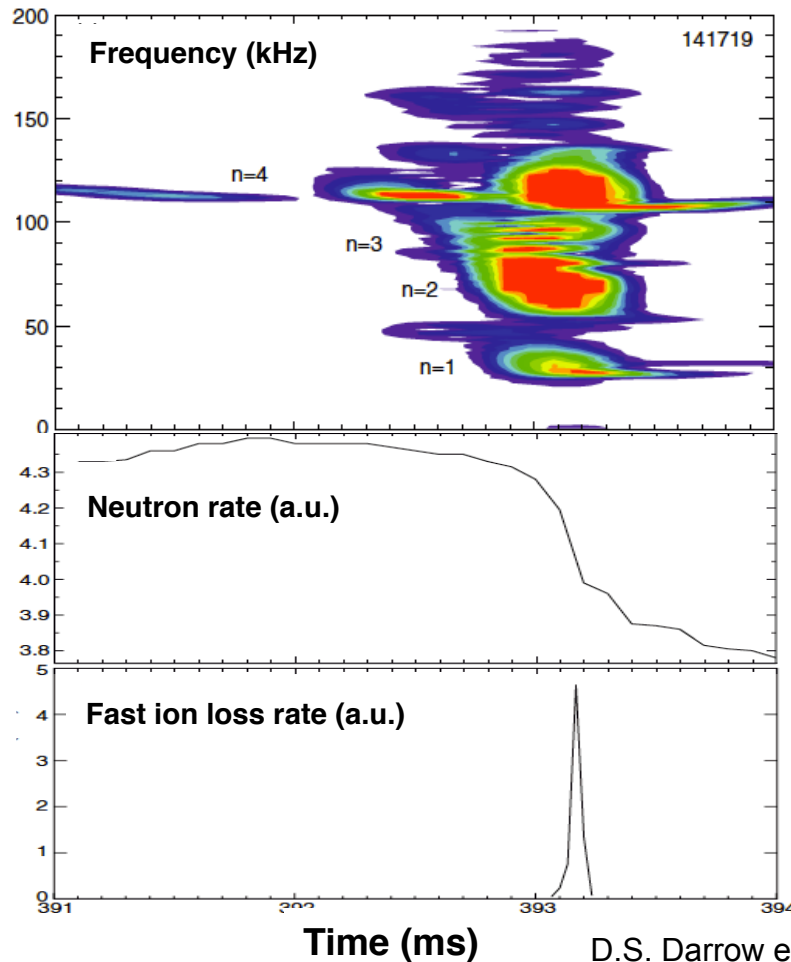
Progress in simulation of neutron rate drop due to TAE avalanche

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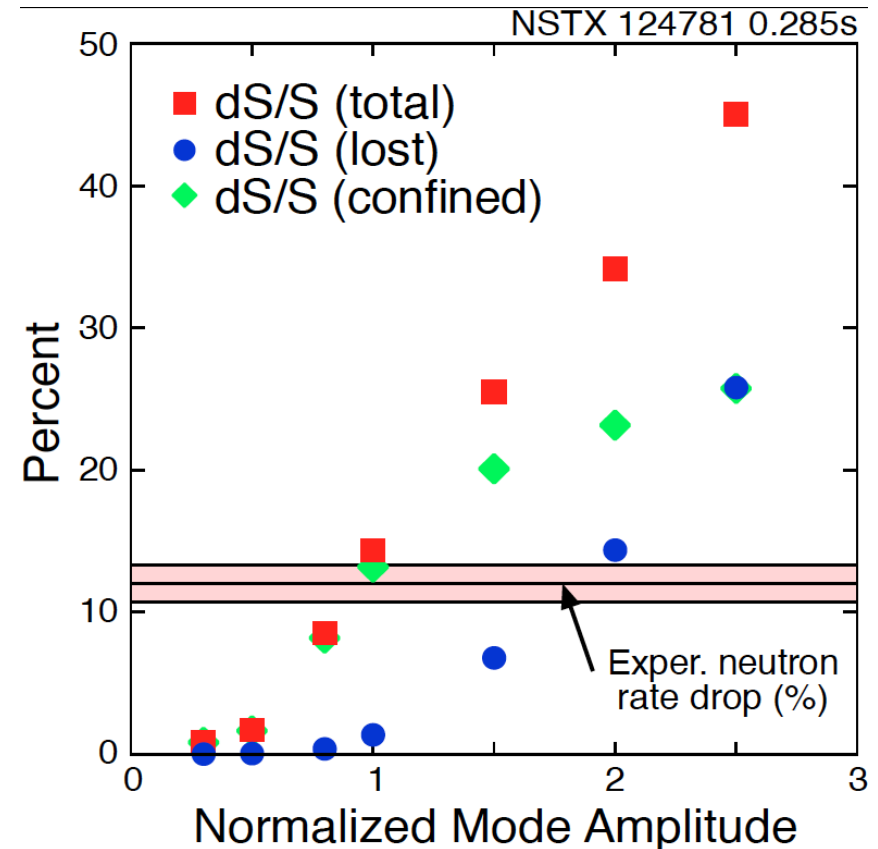
E. D. Fredrickson et al., NF (2013)

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- **ST-FNSF Targeted Experiments**

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STs Addressing Critical Issues for FNSF and Demo

Compact ST-FNSF has no/small central solenoid

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

ST-FNSF Scenarios

high β_T, β_N, κ, disruption avoidance, high H, heat flux mitigation

Current ramp-up and profile control

Solenoid-free Start-up

I_P [MA]

NBI

HHFW

EC/EBW, HI, PF

Time
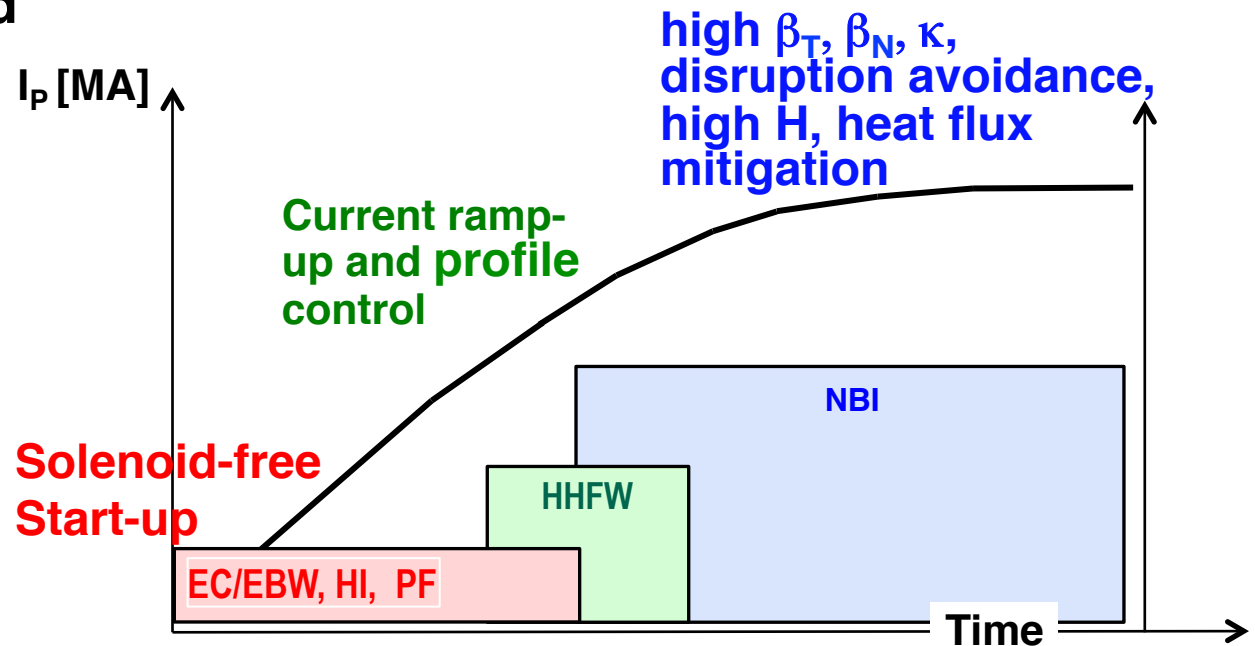
- **Three novel techniques for solenoid-free start-up and ramp-up investigated**
  - **ECH/EBW**
  - **Helicity Injection**
  - **Merging-compression**

Efficient ECH/EBW start-up and sustainment demonstrated

RF start-up investigated in CDX, TST-2, LATE, MAST, QUEST, VEST, SUNIST

LATE ECH/EBW

Initial Open Field Line Phase

Evolution Phase

ST formation Phase
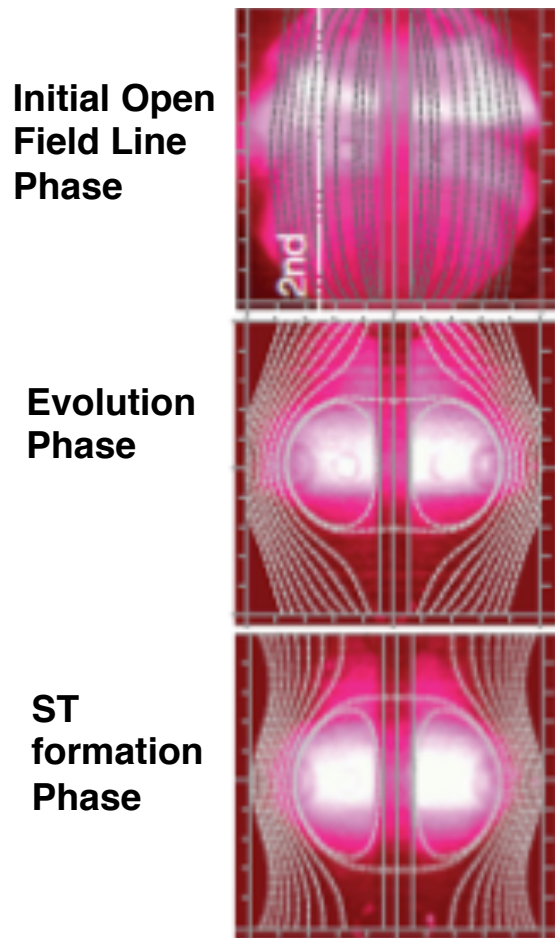
M. Uchida et al., PRL (2010).

55 kA sustained with 270 kW in QUEST, Japan
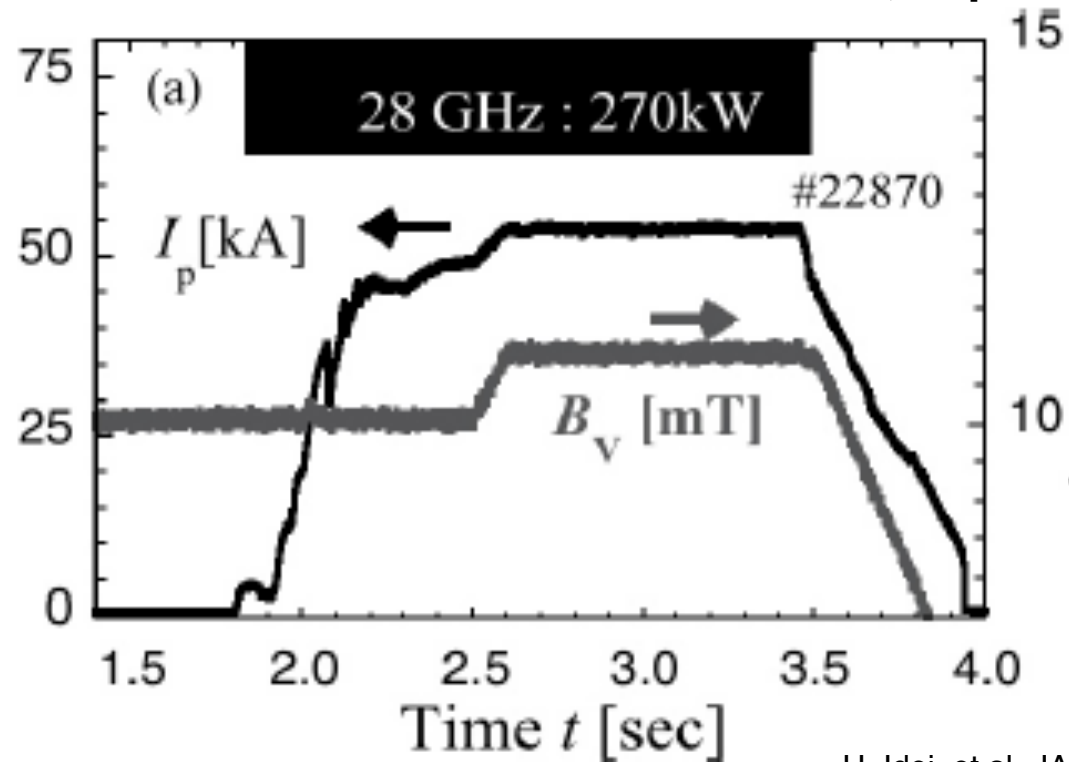
H. Idei, et al., IAEA 2014

72 kA start-up current achieved with 75 kW in MAST, UK

T.S. Bigelow et al., ISTW 2013

ECH/EBW start-up to be tested at MW level in QUEST, NSTX-U and MAST-U.

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

CHI developed on HIT and HIT-II and transferred to NSTX
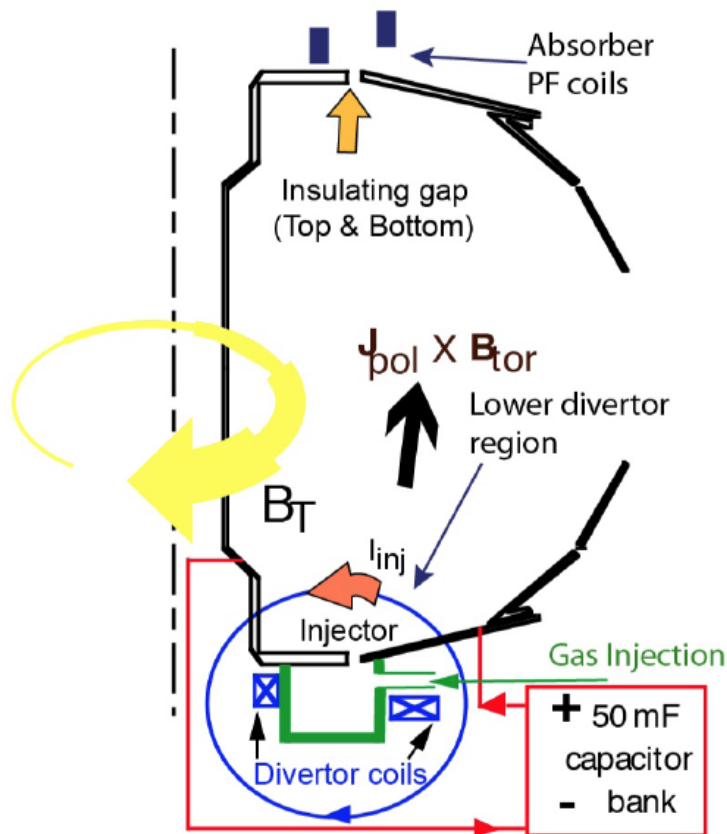
R. Raman et al., PRL (2006)

Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX
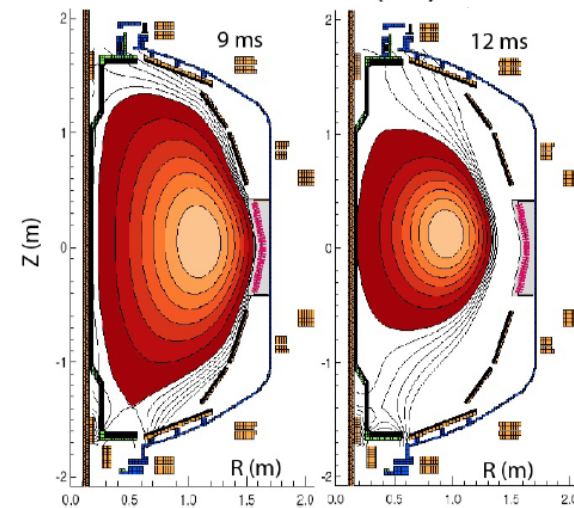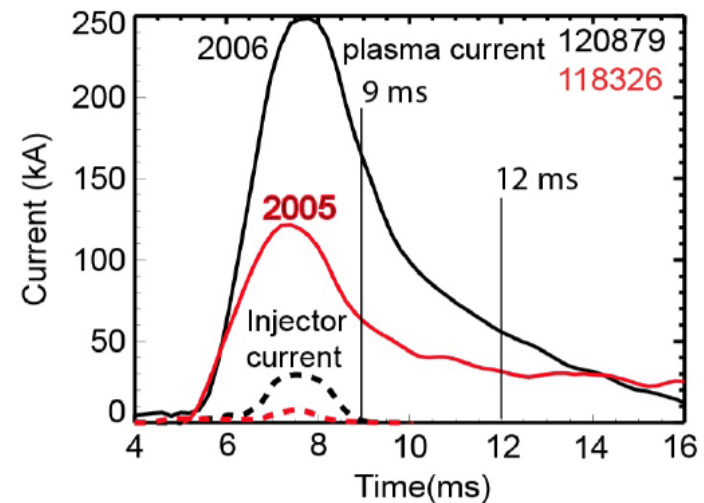
Helicity Injection Is an Efficient Method for Current Initiation
Local Helicity Injection (LHI) Concepts Being Seveloped

3-6 kA current injector array in plasma SOL

Long-Pulse Startup Demonstrated in PEGASUS with LHI
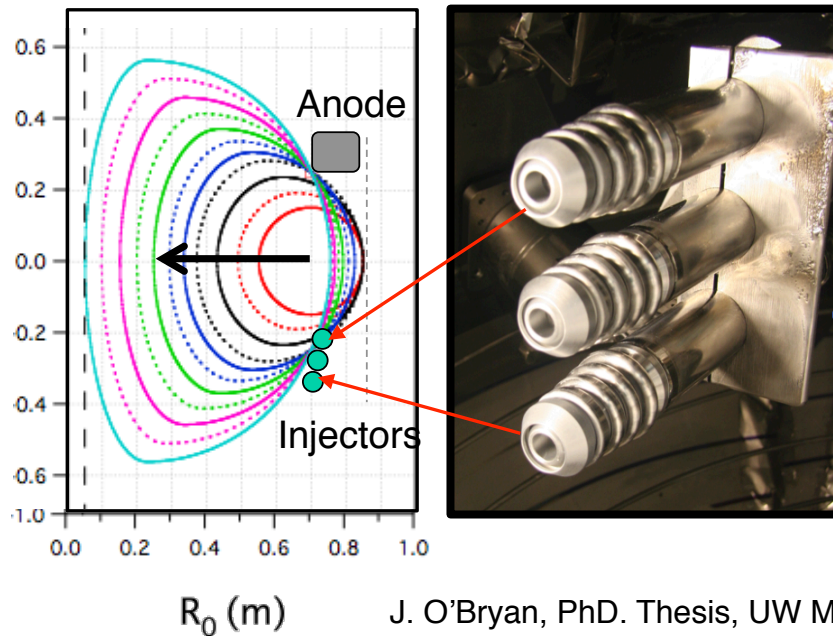
J. O'Bryan, PhD. Thesis, UW Madison 2014
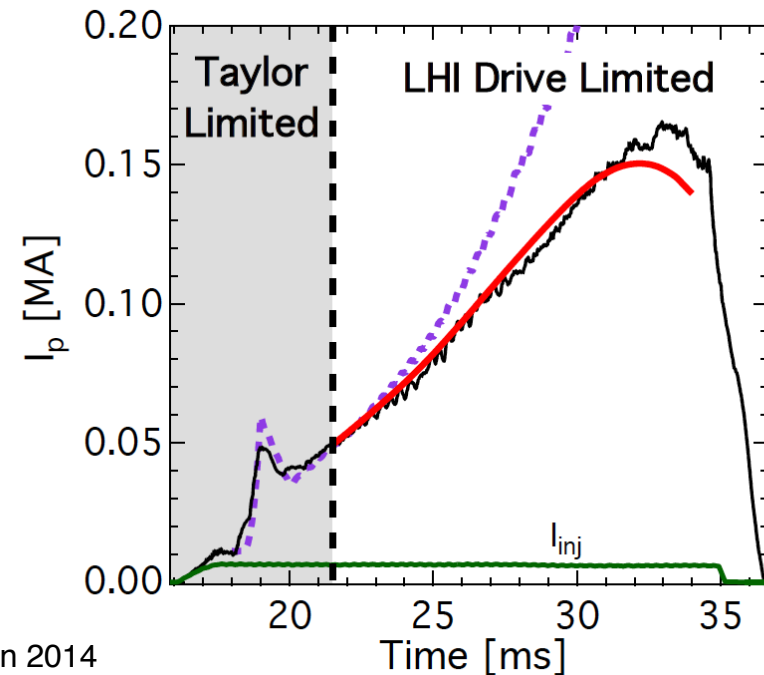
Improving predictive capability for both CHI and LHI

CHI and LHI startup to be tested at higher current ~ 0.5-1.0 MA in NSTX-U.

Merging Start-Up Yielded High Current STs
Rapid ion heating observed from magnetic reconnection

Merging-compression start-up in MAST
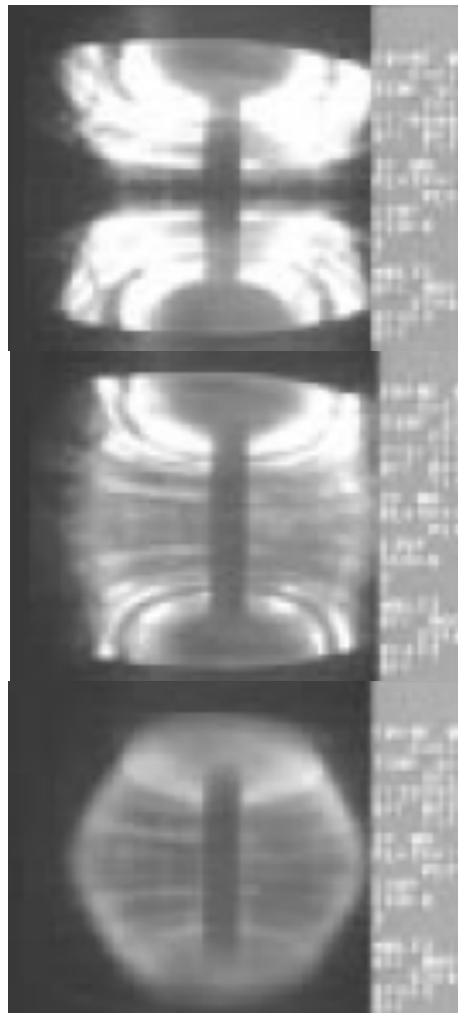
A. Sykes et al., NF (2001)

CS Induction
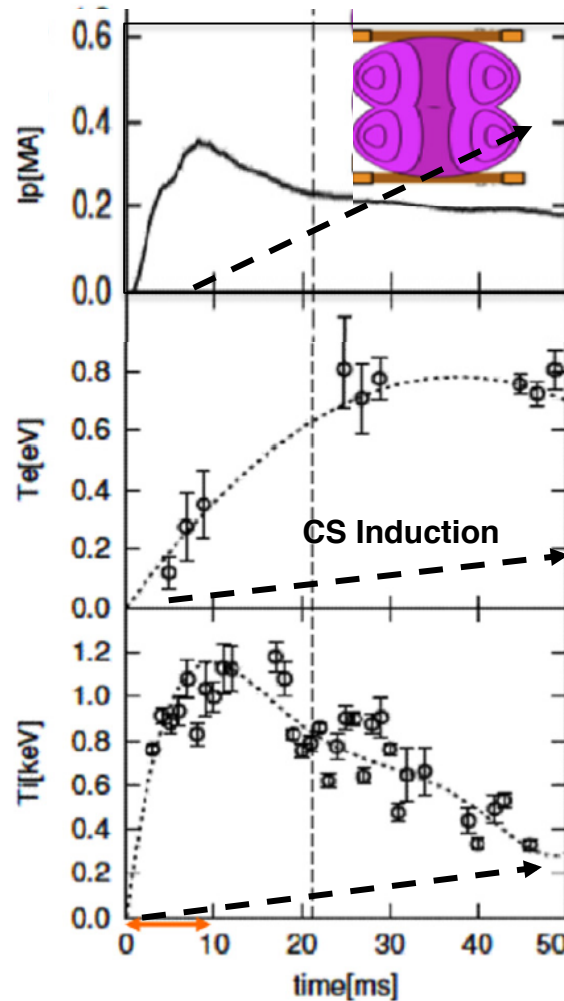
Y. Ono, et al., this APS

Ultra high β STs produced by mergings in TS-3 Device

Y. Ono, et al., NF (2003)

Fig 42. Reprinted UTST. Reprinted with permission from Y. Ono.

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

**Efficient HHFW electron heating due to high β_e achieved in NSTX.**
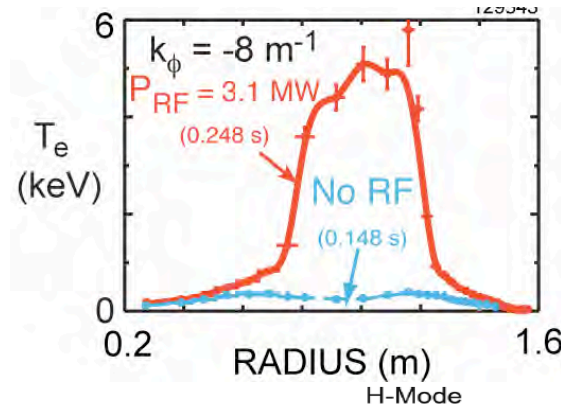
Near sustained discharges obtained with modest HHFW power.
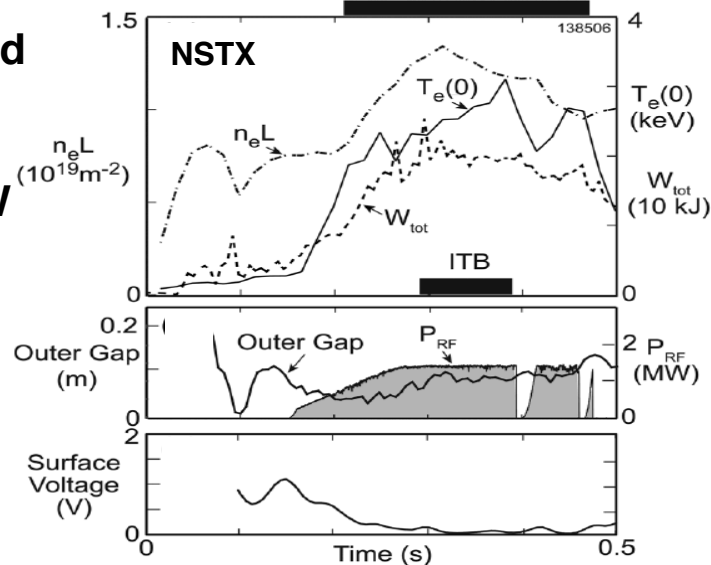
G. Taylor et al., PoP (2010), (2012)

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

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

Efficient HHFW electron heating due to high β_e achieved in NSTX.
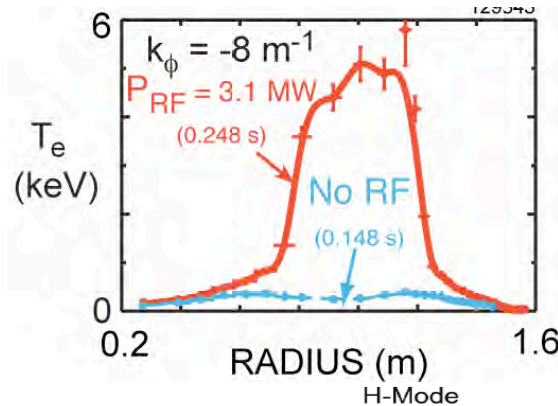
Near sustained discharges obtained with modest HHFW power.
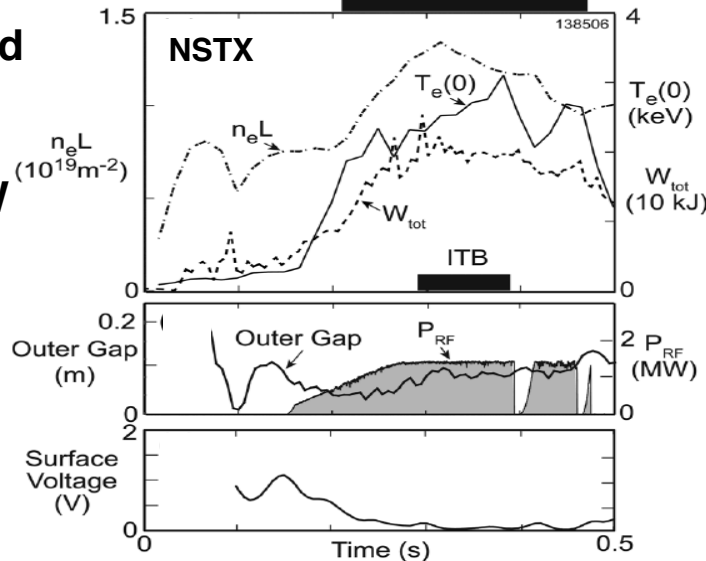
G. Taylor et al., PoP (2010), (2012)

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

Off-axis NBI CD Required for Profile Control Demonstrated in
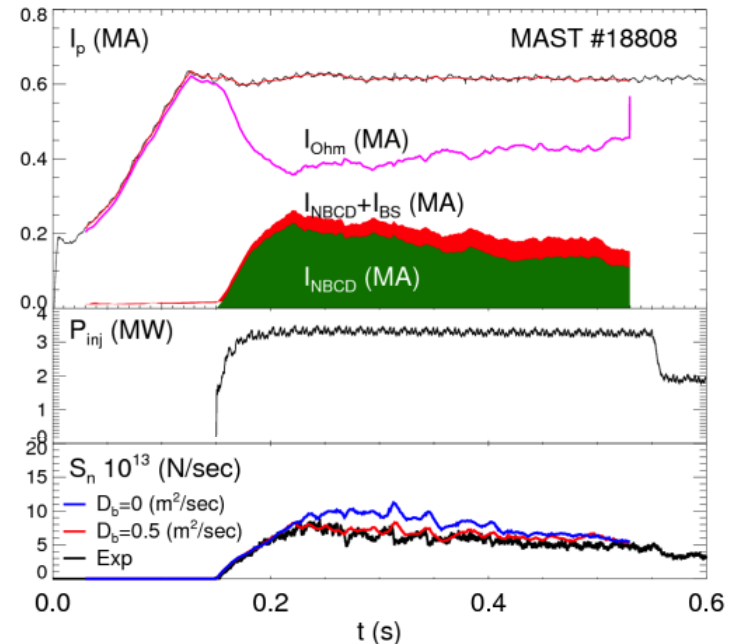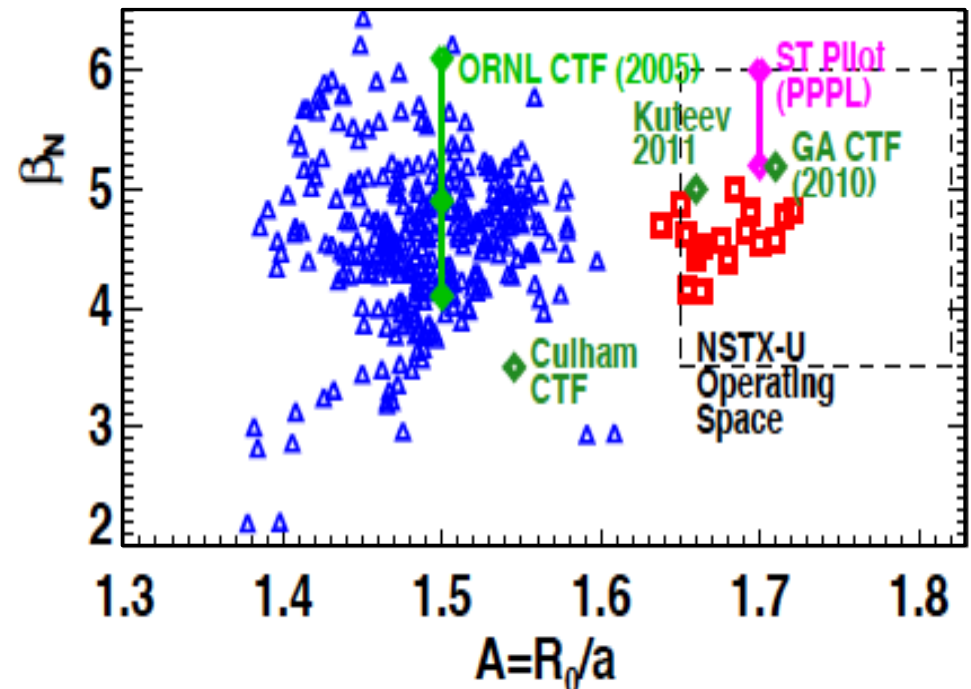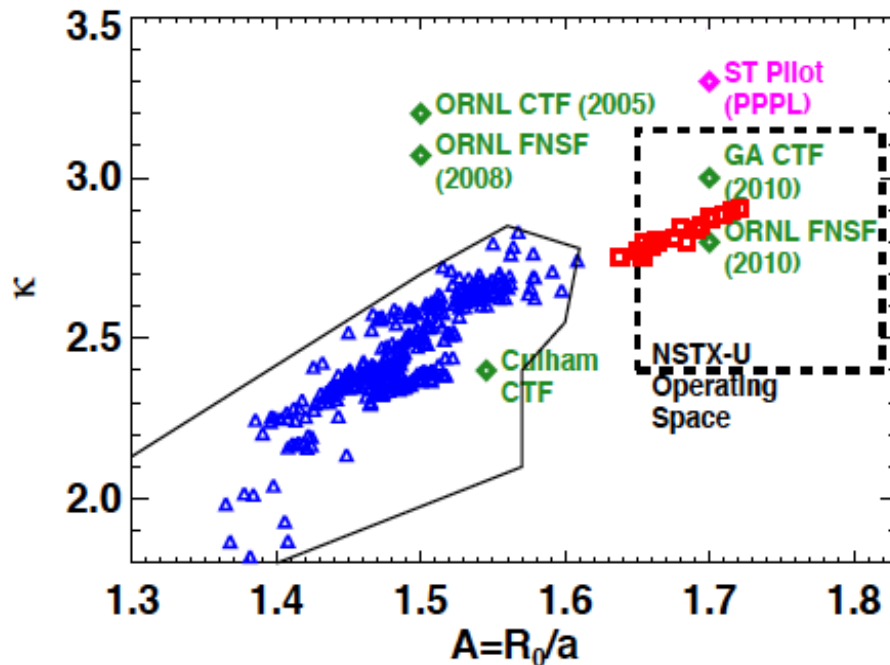

M. Turnyanskiy et al., NF (2009)

Off-axis current drive for profile control will be tested in both MAST-U and NSTX-U with major NBI upgrades.
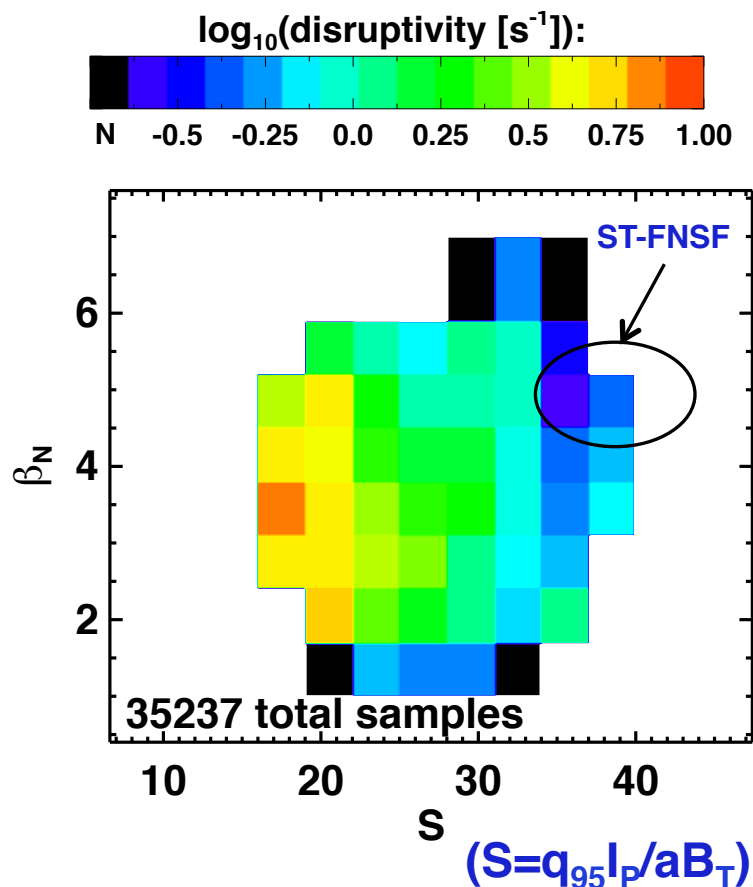
$$f_{BS} \equiv I_{BS} / I_p = C_{BS}\, \beta_p / A^{0.5} = (C_{BS}/20)\, A^{0.5}\, q^*\, \beta_N \propto A^{-0.5}\, (1+\kappa^2)\, \beta_N^2 / \beta_T$$


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

- **NSTX achieved $f_{BS} \sim 50\%$ and $f_{NI} \sim 65\text{-}70\%$ with beams.**
- **NSTX-U expects to achieve $f_{NI} \sim 100\%$ with the more tangential ($\sim$ x1.5- 2 more current drive efficient) NBI.**

NSTX Data Demonstrates a Favorable Operations Window For Reduced Disruptivity in an ST-FNSF

$\log_{10}$(disruptivity [s^{-1}]):

N -0.5 -0.25 0.0 0.25 0.5 0.75 1.00

ST-FNSF

β_N

35237 total samples

S

$(S=q_{95}I_P/aB_T)$

Example: Disruptivity is reduced with strong shaping of the plasma boundary.

S.P. Gerhardt et al., NF (2013)

S=37   S=20
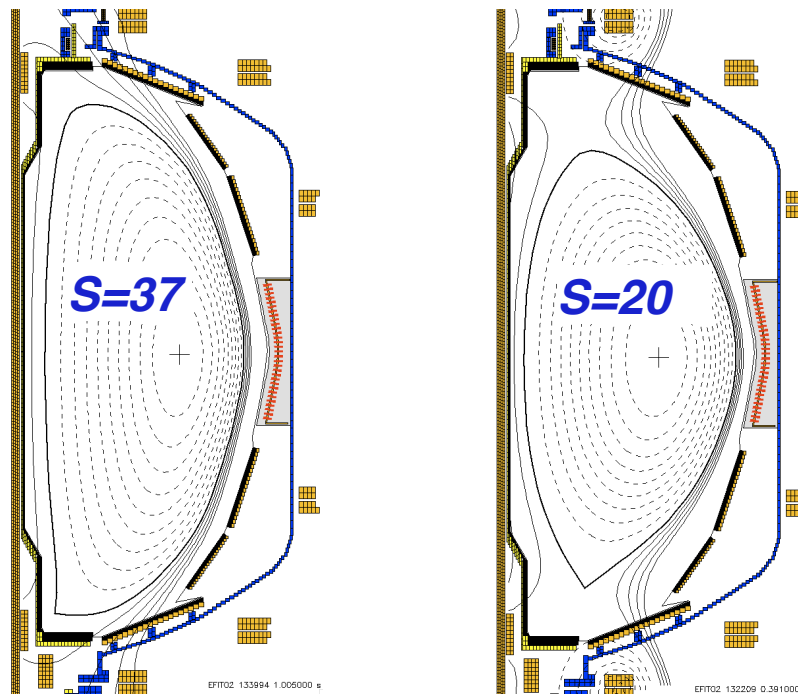
- **No strong increase in disruptivity as β_N increases**
- **Reduction in disruptivity also with:**
  - **Decreasing l_i (broader current profile)**
  - **Decreasing pressure peaking**

Upgrades will test and improve these favorable trends in a systematic way

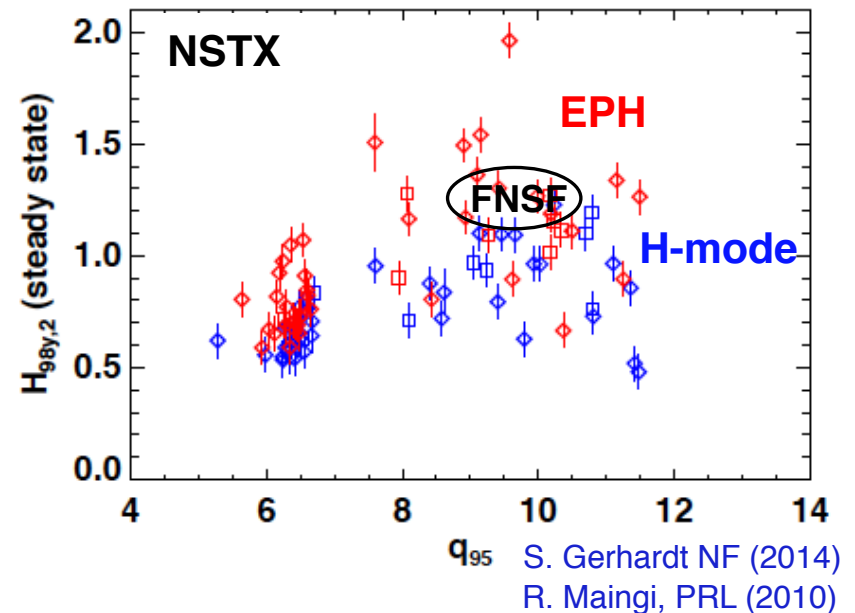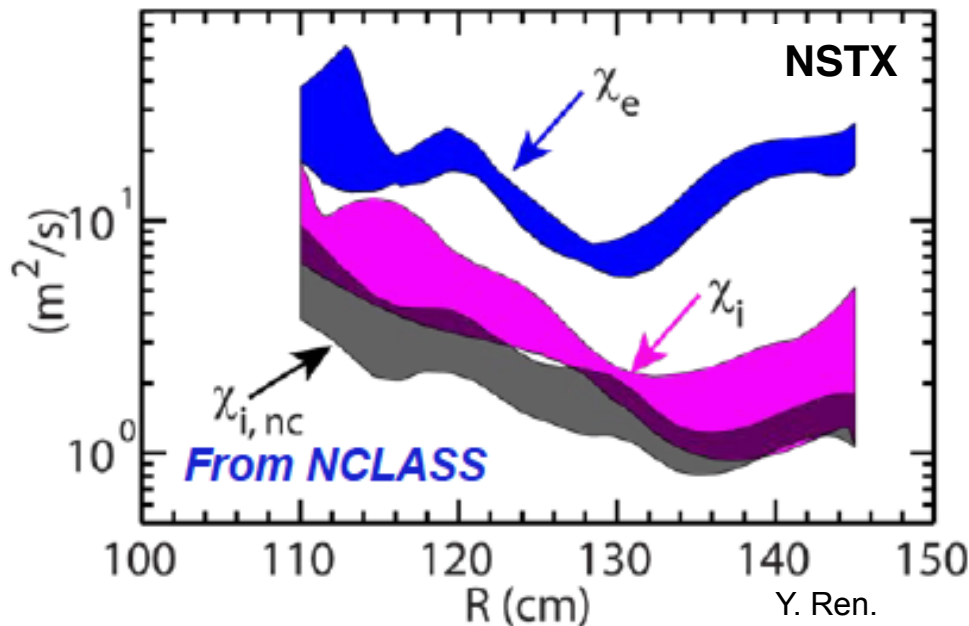# High Confinement Needed for Compact FNSF
High confinement H-mode in the range of FNSF obtained

- Fusion gain Q depends strongly on "H", $Q \propto H^{5-7}$
- Higher H enables compact ST-FNSF H = 1.2 – 1.3
- Higher H gives more reactor design flexibility and margins.

- Ion energy transport in H-mode ST plasmas near neoclassical level due to high shear flow and favorable curvature.
- Electron energy transport anomalous

H-mode confinement in STs H ~ 1 (ITER98$_{y,2}$) but enhanced pedestal H-mode (EPH) has 50% higher H up to H ~ 2

NSTX

χ_e

χ_i

$\chi_{i,nc}$

From NCLASS

(m^2/s)

R (cm)

Y. Ren.

NSTX

EPH

FNSF

H-mode

H$_{98y,2}$ (steady state)

q$_{95}$

S. Gerhardt NF (2014)
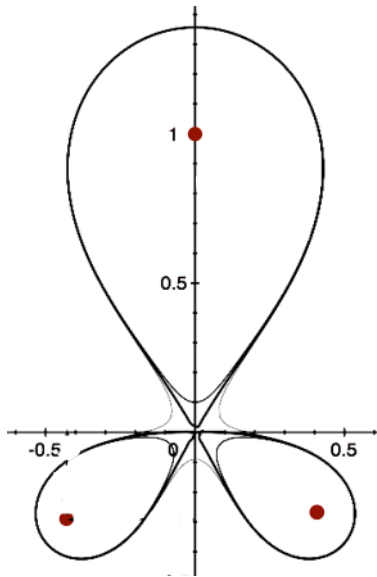R. Maingi, PRL (2010)

ST-FNSF has high P/R due to small R
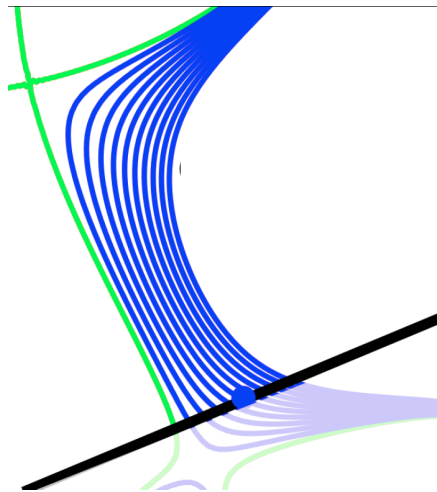
Innovative Heat Flux Mitigation via Divertor Flux Expansion

Lower toroidal field of outboard divertor leg of STs facilitates heat flux mitigation by divertor flux expansion solutions
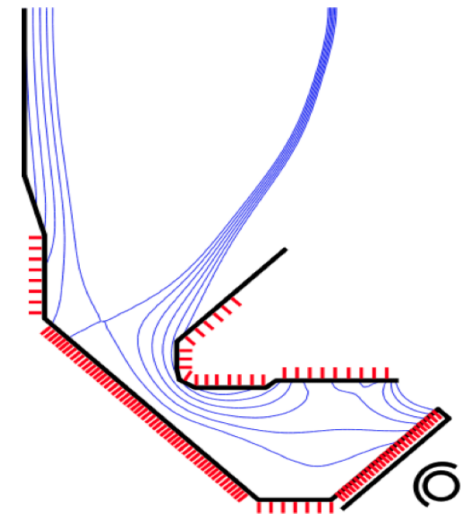
Snow-flake	X-Divertor: CREST	Super-X: MAST-U

D. Ryutov, et al., PoP (2007)   P.M. Valanju, et al., PoP (2009).   Kotschenreuther, et al., PoP (2007)

Major mission of MAST-U is to investigate up-down symmetric Super-X configuration. NSTX explored Snow-flake / X-divertor.

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

Snowflake divertor in NSTX

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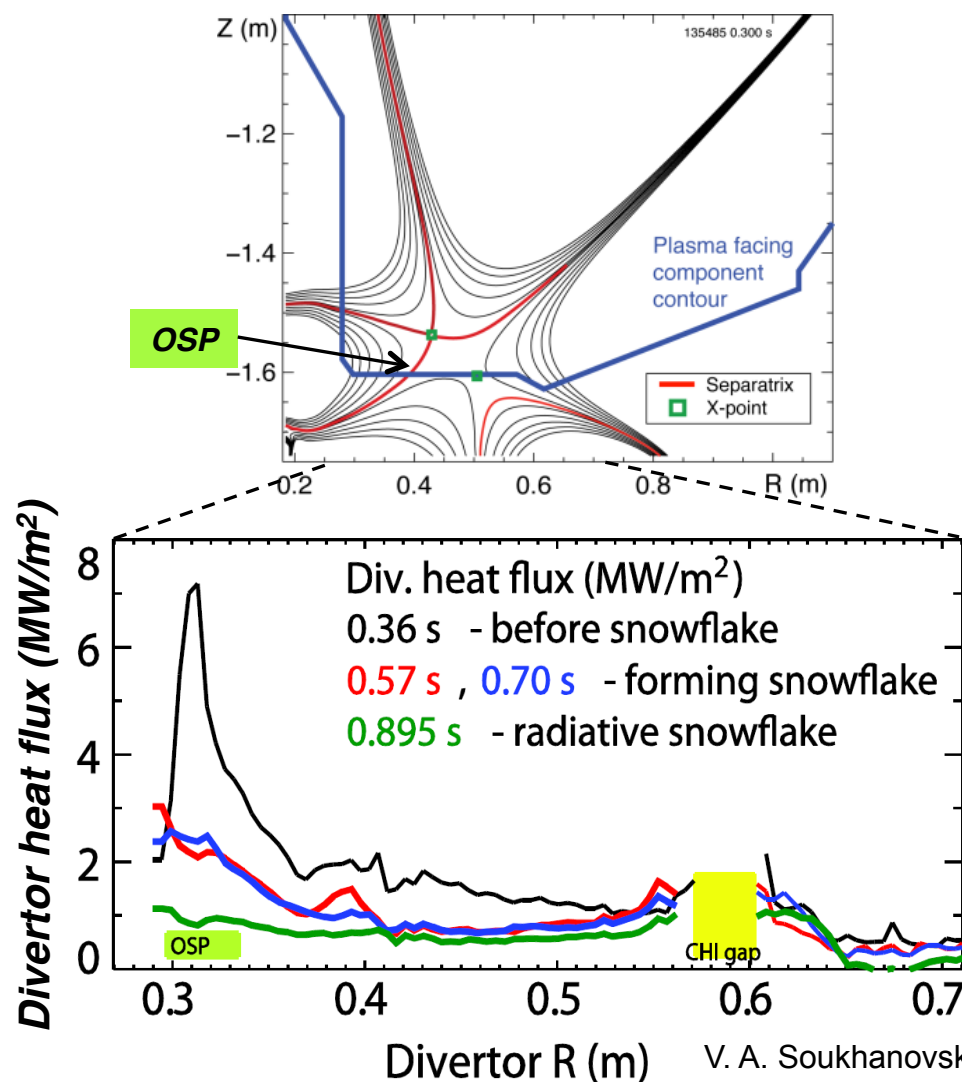
V. A. Soukhanovskii et al., PoP (2012)

Talk Outline

- Unique ST properties

- ST Fusion Energy Development Path

- World ST Facilities

- Unique ST Physics Regimes

- ST-FNSF Relevant Experiments

- **ST Facility Upgrade Status**

- Summary
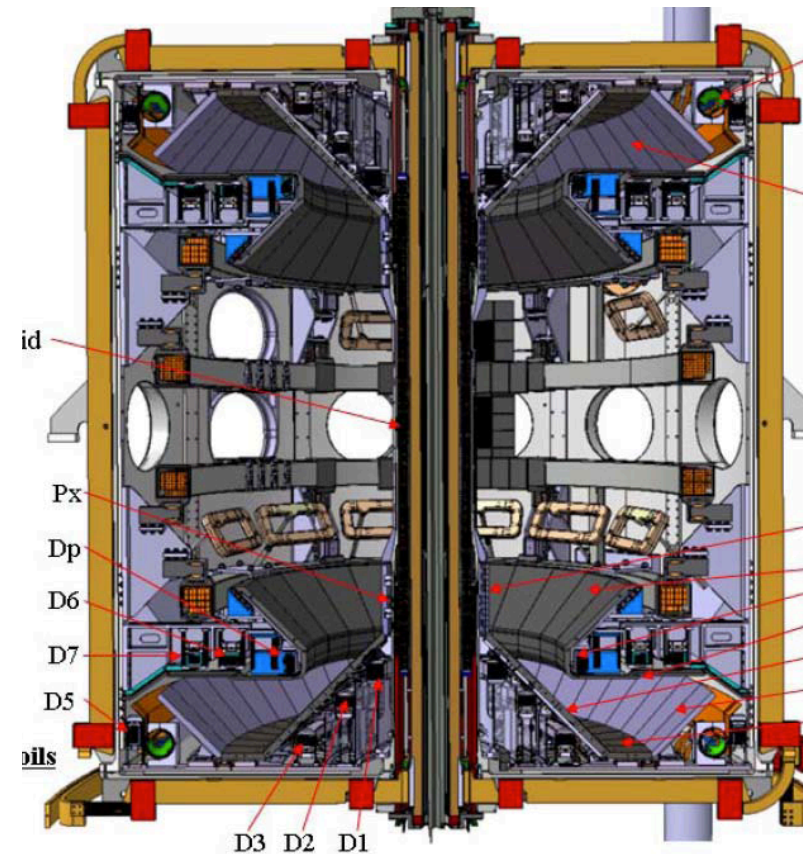
NSTX and MAST are undergoing major upgrades
~ x 2 B$_T$, I$_p$, P$_{NBI}$ and ~ x5 pulse length from NSTX/MAST

NSTX-U
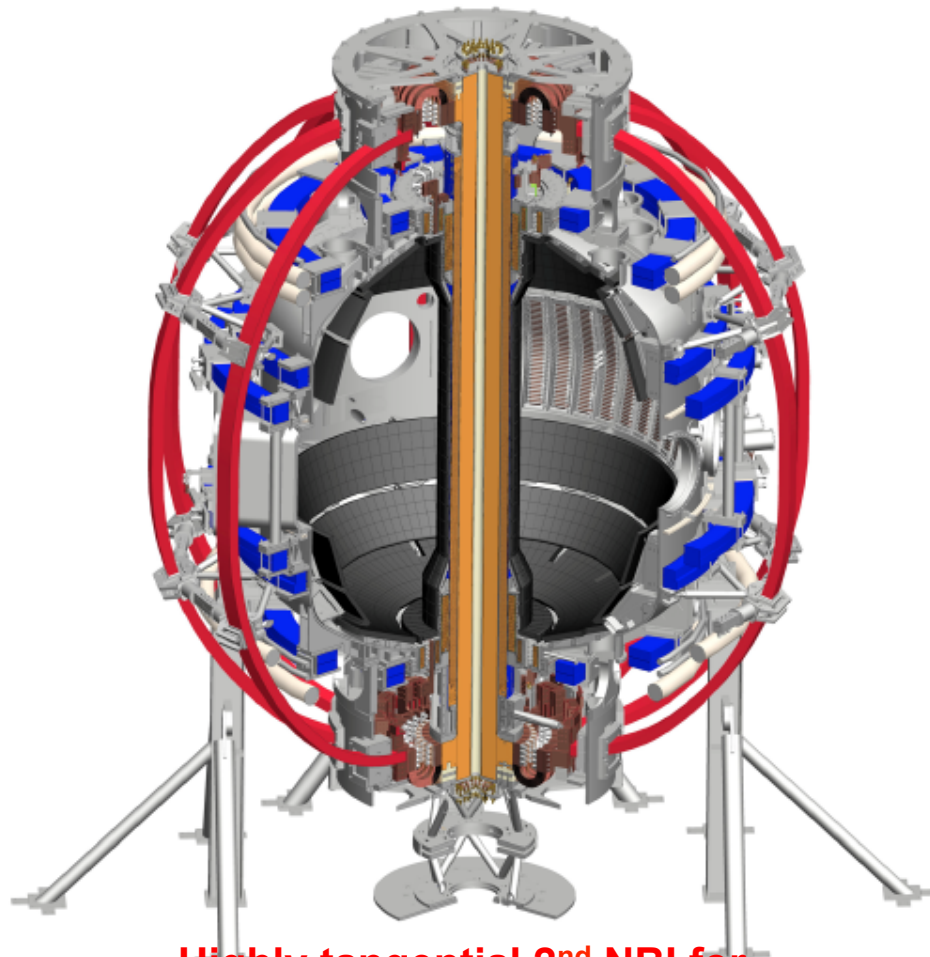
MAST-U

Highly tangential 2nd NBI for non-inductive operations

Super X-divertor configuration for FNSF divertor solution

MAST-U to support novel exhaust concepts, ITER, and FNSF
Completed MAST operation in 2013 and began construction

Super X Divertors

Conventional   Super-X

- **New Center Column for higher B_T and I_p**
- **Super-X Divertor for divertor heat flux mitigation**
- **Vertically off-axis NBI for current profile control.**

End Plate Modules (LEP)

Off-Axis NBI

Centre Column (CC) Module

First plasma scheduled in 2016

J. Milnes SOFT 2014

NSTX-U to provide data base to support ST-FNSF designs and ITER operations

Previous center-stack   **New center-stack**
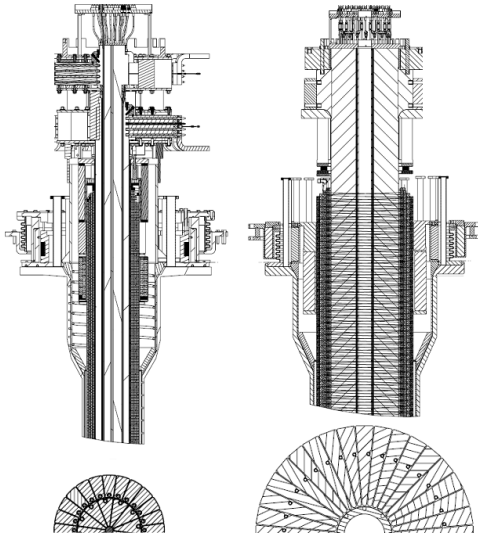
TF OD = 20cm   TF OD = 40cm
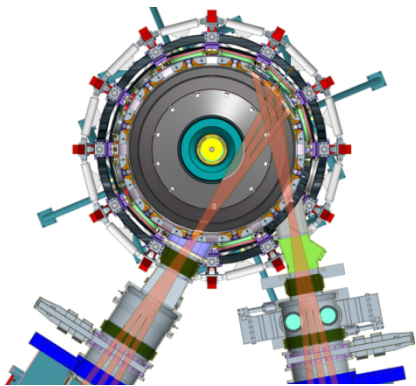
Present NBI   New 2nd NBI

- **New CS provides higher x2 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**

~ X 5 - 10 increase in nτT 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)
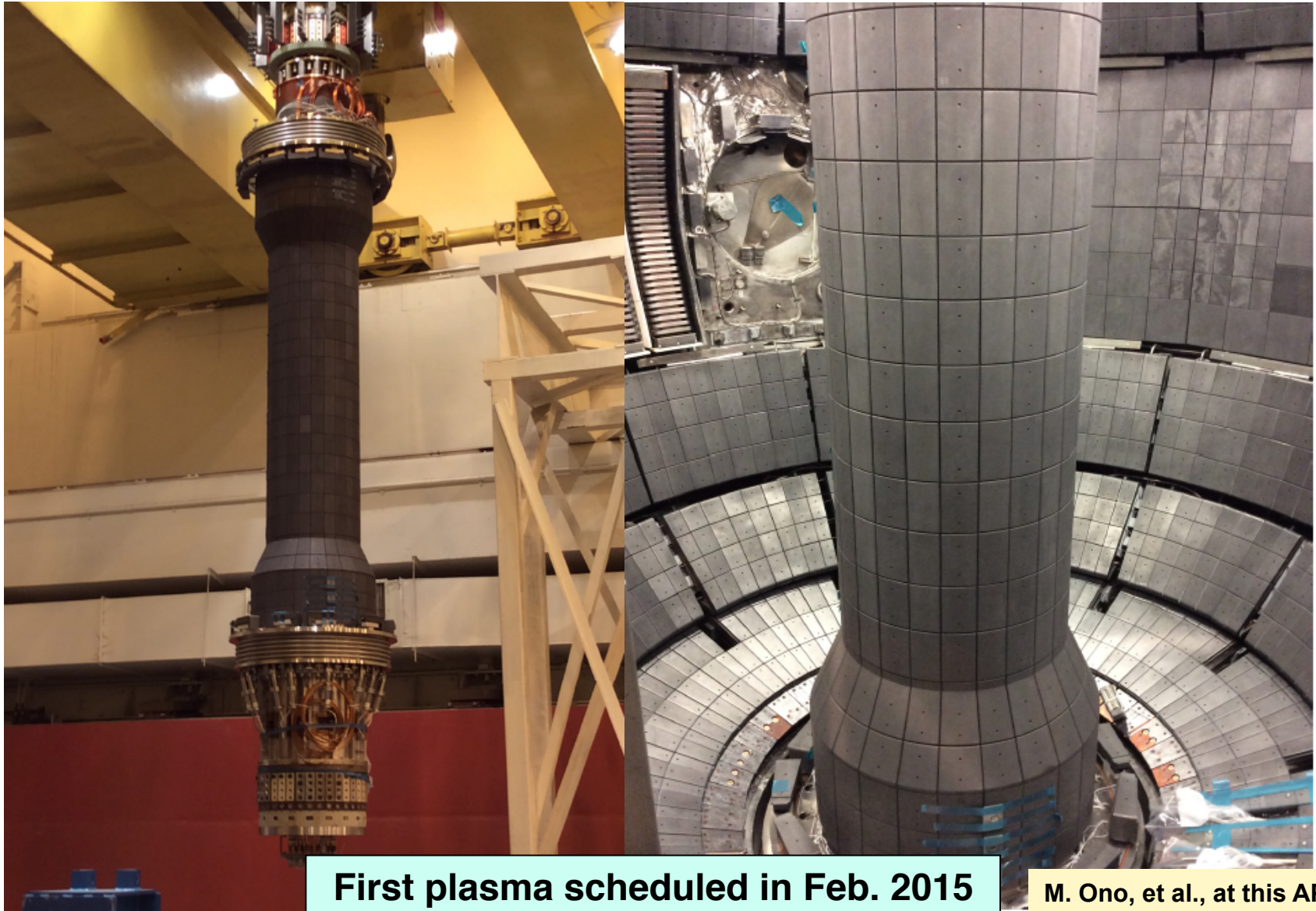
Research operation to resume in Apr. 2015

NSTX Upgrade Project Is Nearly Complete
Recent aerial view of NSTX-U Test Cell (Oct. 27, 2014)
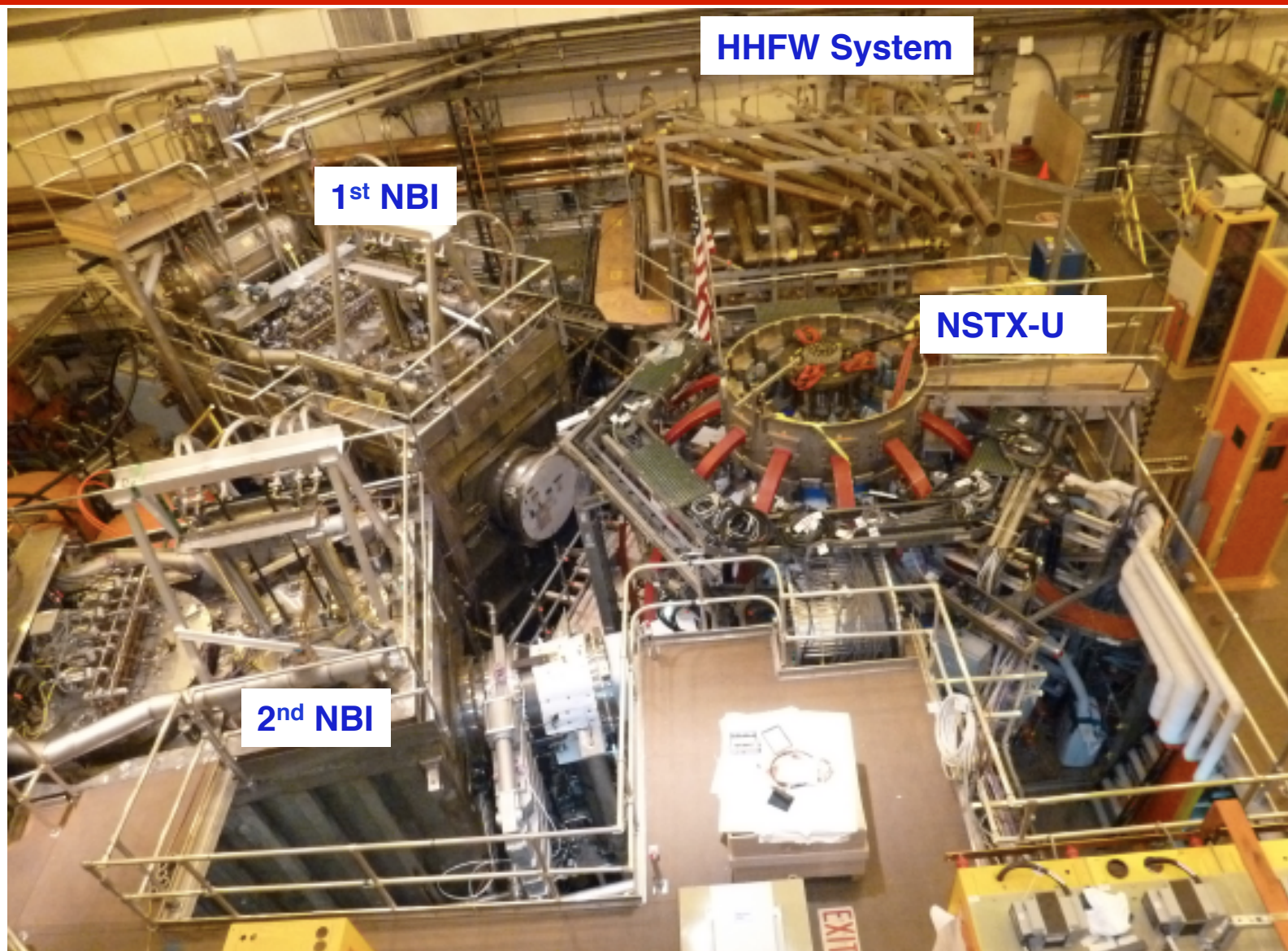
New Center-Stack Installed In NSTX-U (October 24, 2014)

First plasma scheduled in Feb. 2015

M. Ono, et al., at this APS

Summary of Spherical Torus Research
World-wide effort with over 500 researchers and 140 students

- ST is a member of tokamak family with aspect ratio ≤ 2.0

- Unique ST features include natural elongation, compact geometry, and high beta which would be suitable for compact FNSF and PMI solutions.

- Extreme ST physics parameters exercise tokamak theory/modeling to validate and improve predictive capability needed for ITER and beyond where large extrapolations are needed.

- MA-class ST research began in 2000 with NSTX and MAST together with smaller ST facilities worldwide. Today, 16 ST facilities are operational.

- STs contributed strongly to fusion research program in all fusion energy science areas

- STs performed FNSF relevant experiments achieving many of the key plasma parameters and research objectives.

- Next phase of ST research begins shortly as 1T-2MA-class MAST-U and NSTX-U Facilities are coming on line for FNSF and ITER

ST Related Presentations at this APS Meeting

Invited Talks

BI1.00001 (Mon): The effects of impurities and core pressure on pedestal stability in JET and MAST, Samuli Saarelma

GI1.00003 (Tue): Broadening of the divertor heat flux footprint with increasing number of ELM filaments in NSTX, Joon-Wook Ahn

NI2.00003 (Wed): High Power Heating of Magnetic Reconnection in Tokamak Merging Experiments, Yasushi Ono

TI1.00001 (Thu): Simulation of 3D effects on partially detached divertor conditions in NSTX and Alcator C-Mod. Jeremy Lore

TI1.00003 (Thu): Drift Kinetic Effects on 3D Plasma Response in High-beta Tokamak Resonant Field Amplification Experiments, Z.R. Wang

VI2.00002 (Thu): Unification of Kinetic Resistive Wall Mode Stabilization Physics in Tokamaks. S.A. Sabbagh

YI1.00006 (Fri): Energy Channeling and Coupling of Neutral-beam-driven Compressional Alfv'en Eigenmodes to Kinetic Alfv'en Waves in NSTX, Elena Belova

YI2.00003 (Fri) : High Performance Discharges in the Lithium Tokamak eXperiment (LTX) with Liquid Lithium Walls, John Schmitt

Oral Session GO3 (Tue): MAST-U, PEGASUS, NSTX-U, LTX

Poster Session PP8 (Wed): NSTX-U, LTX, PEGASUS, MAST-U, QUEST, TS-4

Back-up slides

Nearly self-sustained ST-Demo regimes identified
q-profile appears to be a differentiating feature for ST and AT

$$f_{BS} \equiv I_{BS} / I_p = C_{BS}\, \beta_p / A^{0.5} \propto A^{-0.5} (1+\kappa^2)\, \beta_N^2 / \beta_T$$
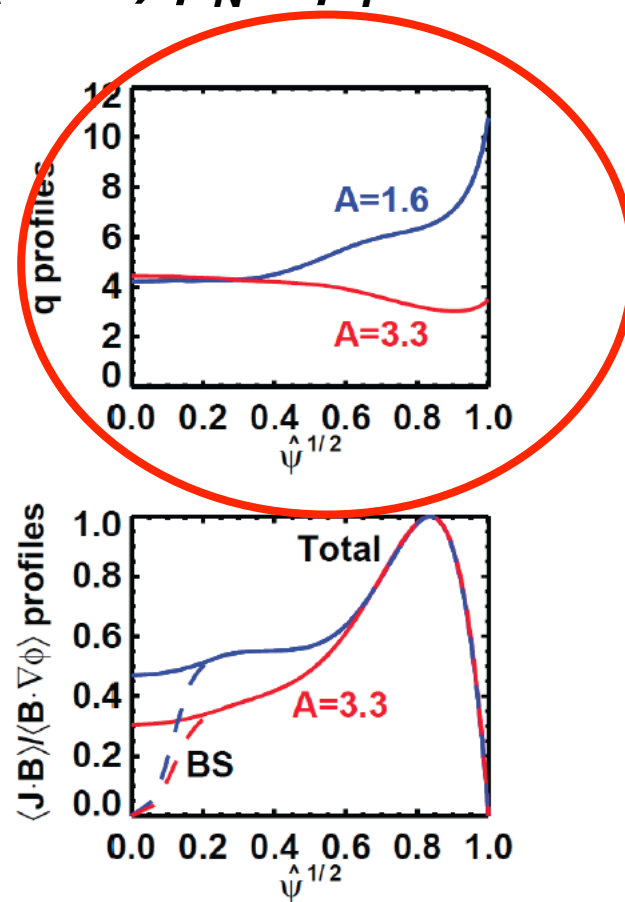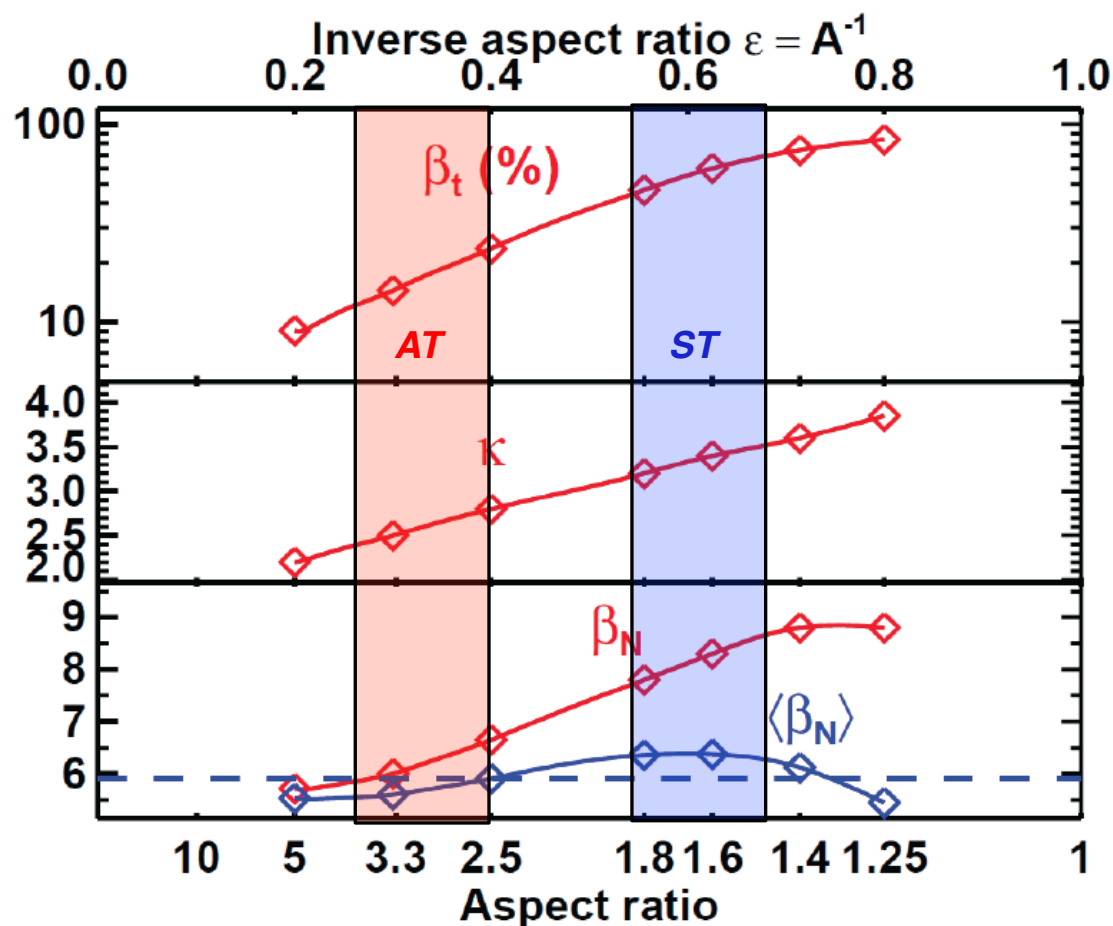
J. Menard et al., PPPL Report, 2003

Reversed shear AT likely suffers from infernal modes and/or double tearing at high beta, and we can potential reduce or eliminate reverse shear in ST due to higher edge q-shear from low-A

ST research program supports and accelerates a range of development paths toward fusion energy

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

ST Research →

STs Narrow Gaps to Pilot/DEMO:

Goal: 100% non-inductive + high β

Plasma-Material Interface Research
- Strong heating + smaller R → high P/R, P/S
- Novel solutions: snowflake, liquid metals, Super-X, hot high-Z walls

Fusion Nuclear Science Facility
- High neutron wall loading
- Potentially smaller size, cost
- Smaller T consumption
- Accessible / maintainable

Burning Plasma Physics - ITER

Fusion Nuclear Science Facility

FNSF-ST

FNSF-AT

Pilot Plant or DEMO

ARIES-ST

ARIES-AT

ARIES-CS
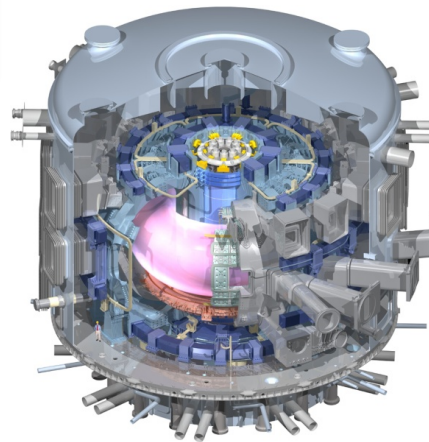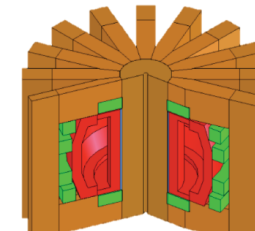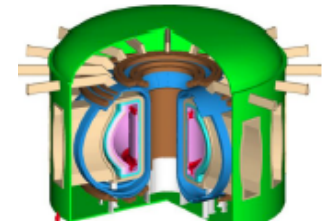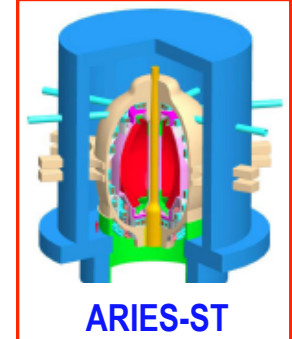
Steady-State, Plasma-Material Interface R&D

EAST   JT-60SA   NHTX   VULCAN

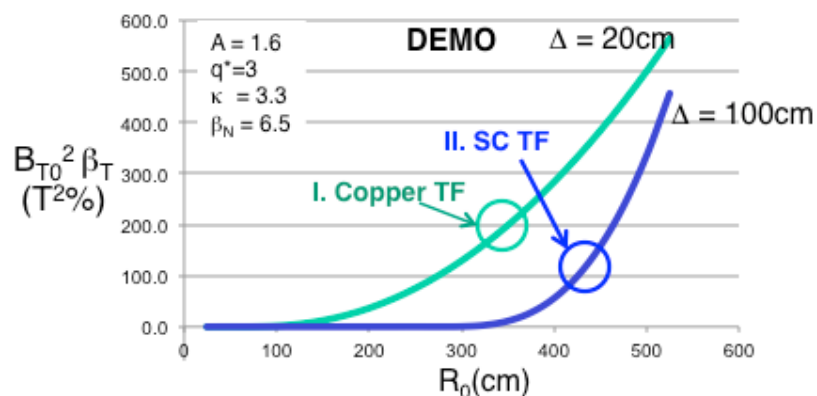KSTAR   W7-X, LHD   QUASAR
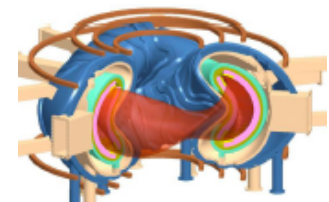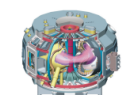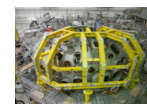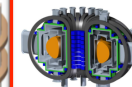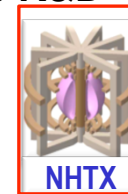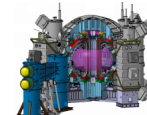
J. Menard

ST Fusion Power Plants
Copper vs. Superconducting Coils

Nearly fully self-sustained ST/Tokamal reactor requires high κ and β_N
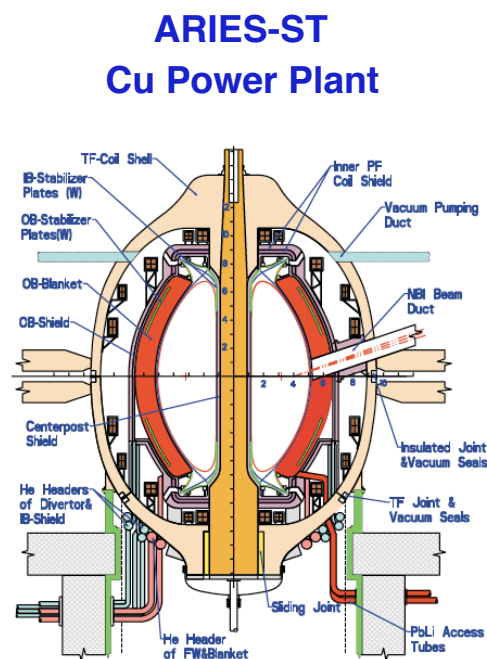
$$f_{BS} \equiv I_{BS} / I_p = C_{BS} \beta_p / A^{0.5} = (C_{BS}/20) A^{0.5} q^* \beta_N \propto A^{-0.5} (1+\kappa^2) \beta_N^2 / \beta_T$$

All of the ST power plant designs have $q_{95} \sim 10$ which could give needed MHD stability

ARIES-ST
Cu Power Plant

$R_0 \sim 3.2$ m

F. Najmabadi et al., FED (2003)
H. R. Wilson, et al., NF (2004)

JUST
SC ST Power Plant

$R_0 \sim 4.5m$

Y. Nagayama et al., IEEJ (2012)
B.G. Hong, Yet al.,NF (2011)
K. Gi IAEA(2014)

Copper design – Compact but due to larger recirculating power leading to higher fusion power needing aggressive β_T, β_N, κ. designs..
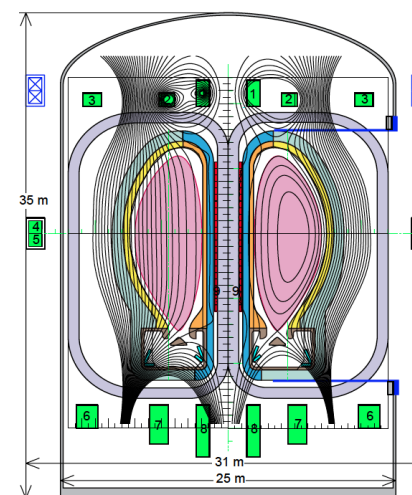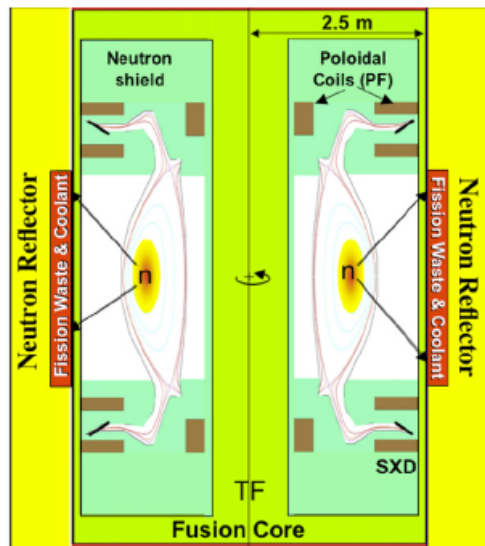
SC design – Larger size due to SC shielding requirement. But smaller recirculating power provides more flexibility in design such as operating at lower fusion power, more moderate β_T, β_N, κ, etc.

Non-conventional ST Fusion Power Reactors
Taking advantage of compact and light weight ST fusion core
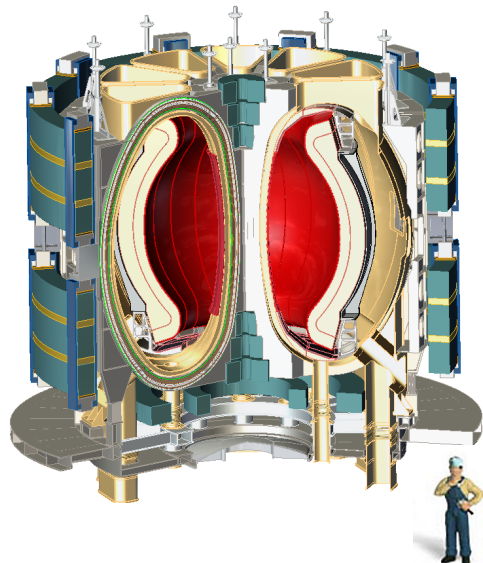
ST Fusion-Fission Hybrid

ST-FNSF-like Q ~ 1 facility producing net energy by "burning" highly toxic long-live nuclear waste

M. Kotschenreuther et al., FE&D (2009).

ST135
R = 1.35m
Q_{fus} ~ 5 Pilot Plants

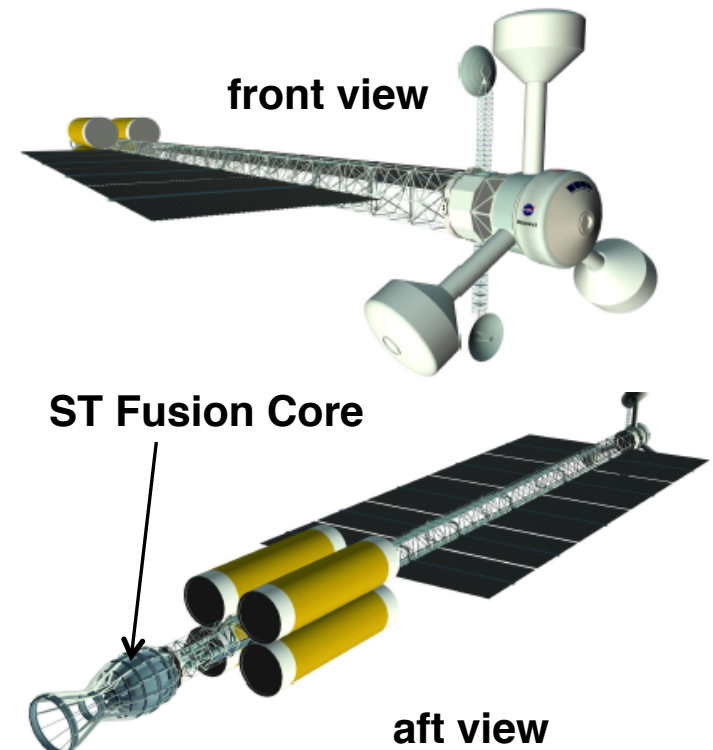**With HTS magnet operating at 3.7T/7 MA P_{fusion} = 185 MW**

A. Sykes, SOFT 2014

The Discovery II

Realizing "2001: A Space Odyssey": Piloted Spherical Torus Nuclear Fusion Propulsion
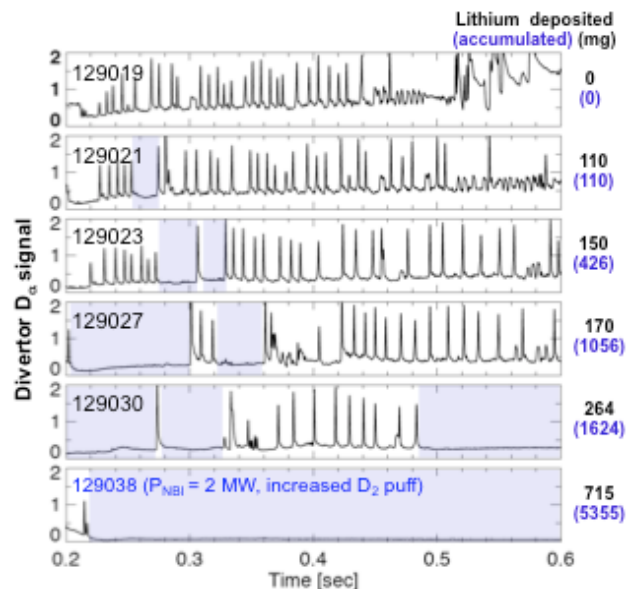

front view

ST Fusion Core
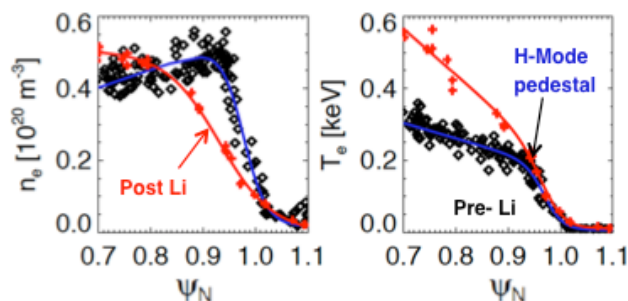
aft view

C.H. Williams, et al., NASA/TM-2005-213559

ELM Stabilization and Mitigation
Through application of lithium and 3-D fields

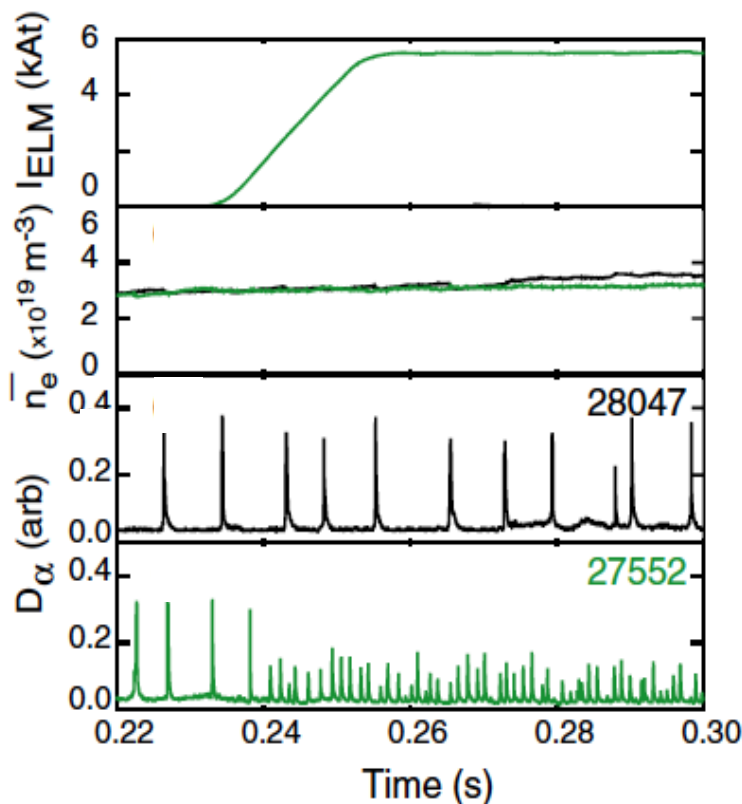**ELMs stabilized with edge pressure modification with Li in NSTX**

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)