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

Masayuki Ono

PPPL, Princeton University

On behalf on the world ST community!

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Acknowledgements

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- INL
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- ORNL
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- *PPPL
- Purdue U
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- Think Tank, Inc.
- UC Davis
- UC Irvine
- UCLA
- UCSD
- U Colorado
- U Illinois
- U Maryland
- U Rochester
- U Tennessee
- *U Washington
- *U Wisconsin

* With ST Facility

International Institutions:

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- U St. Andrews
- York U
- *Tokamak Energy
- Chubu U
- Fukui U
- Hiroshima U
- *Hyogo U
- *Kyoto U
- *Kyushu U
- Kyushu Tokai U
- NIFS
- Niigata U
- *U Tokyo
- Tsukuba U
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- Hebrew U
- *Ioffe Inst
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- TRINITI
- D.V.Efremov Inst
- INTEKHMASH
- NFRI
- KAIST
- POSTECH
- *Seoul National U
- *Beijing National L
- ASIPP
- ENEA, Frascati
- CEA, Cadarache
- IPP, Jülich
- 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

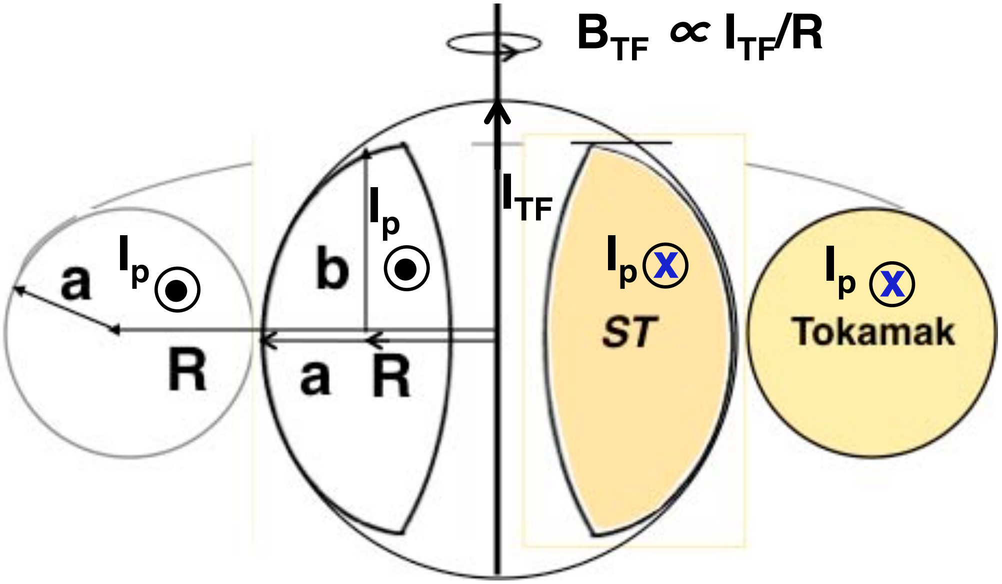
Talk Outline

- **Unique ST properties**
- ST Fusion Energy Development Path
- World ST Facilities
- Unique ST Physics Regimes
- ST-FNSF Relevant Experiments
- ST Facility Upgrade Status
- Summary

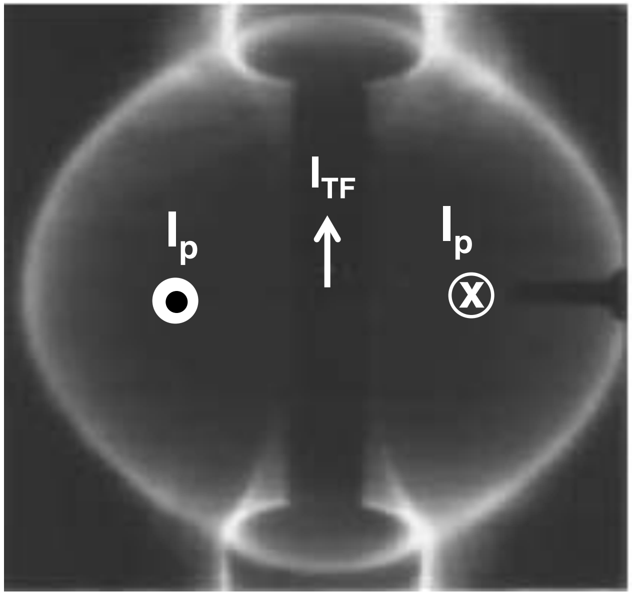
ST is a low aspect ratio tokamak with A < 2
Natural elongation makes its spherical appearance

Aspect Ratio A = R/a	*Elongation $\kappa = b/a$*

"natural" = "without active shaping"



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)

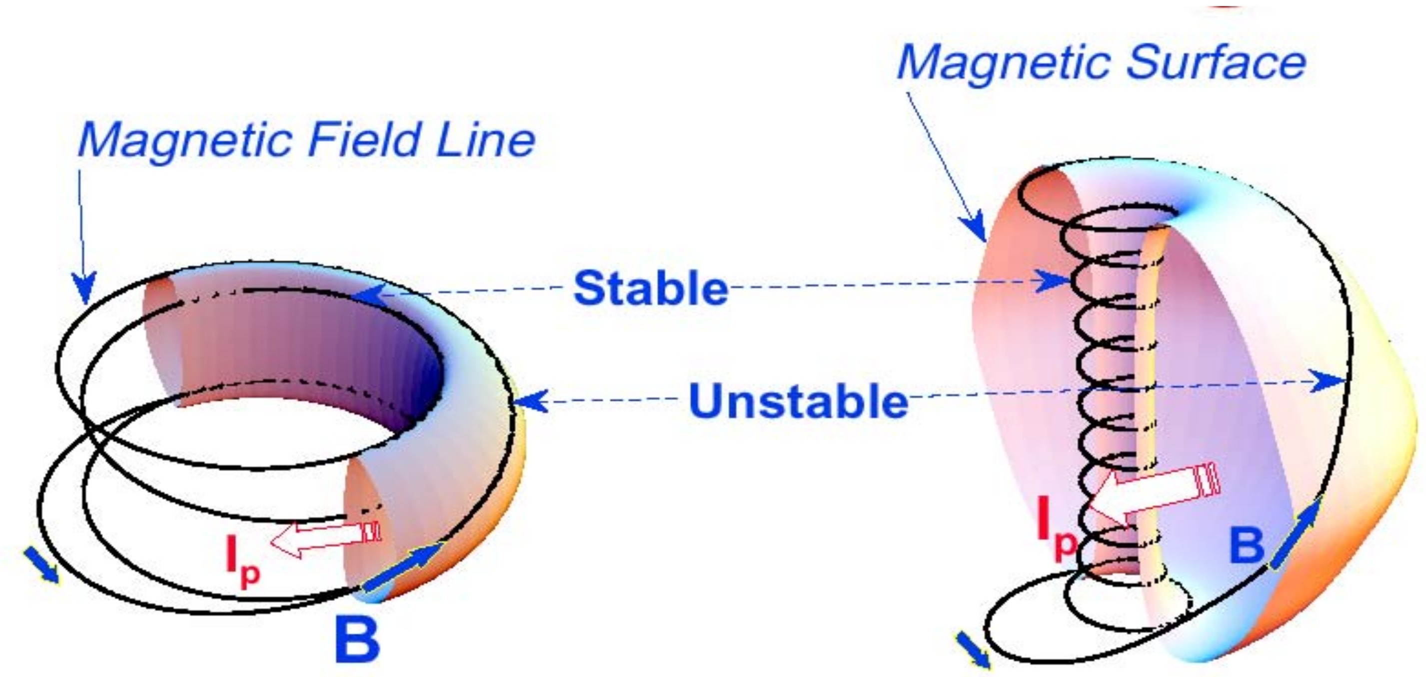
A spherical tokamak (ST) is a high beta tokamak

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

ST



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

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

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 $\kappa = b/a$

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**

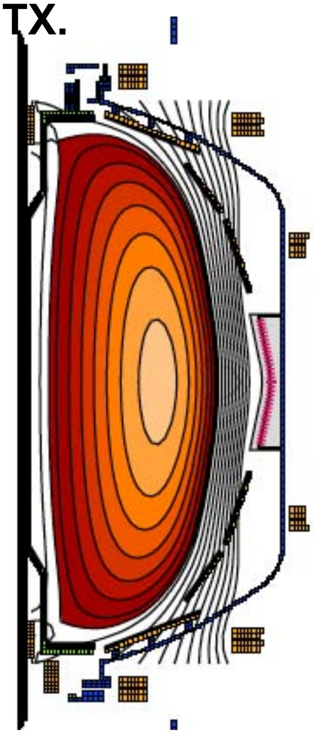
$$\tau_E \propto I_p$$

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

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

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

High *κ* ~ 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 toroidicity)→ 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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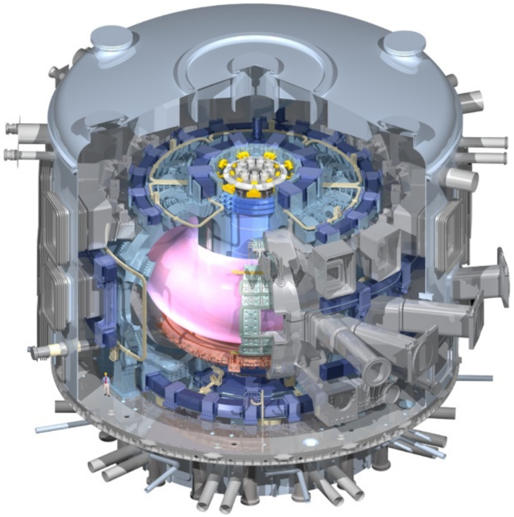
Unique ST properties support and accelerate a range of development paths toward fusion energy

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 → high P/R, P/S
- Novel solutions: snowflake, liquid metals, Super-X, hot high-Z walls

Enable Compact Fusion Nuclear Science Facility

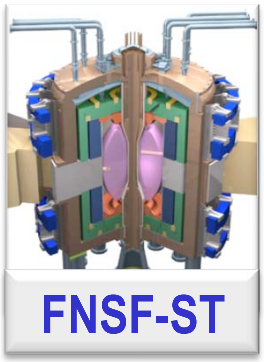
- High neutron wall loading
- Potentially smaller size, cost
- Smaller tritium (T) consumption at fixed neutron wall loading
- Accessible / maintainable

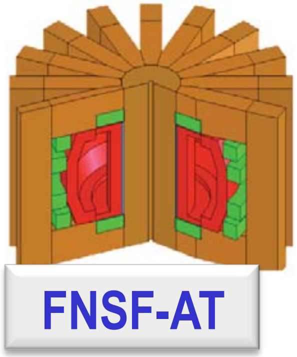
Fusion needs FNSF(s) (modest cost, low T, and reliable) to Test and Qualify Fusion Components

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





- **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

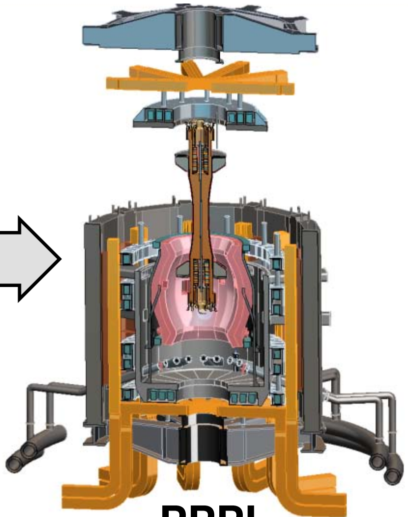
Tritium breeding ratio (TBR) near 1

- Requires sufficiently large R_0, careful design

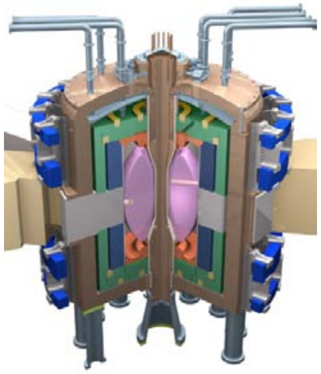
R&D Needs for an ST-FNSF

- 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

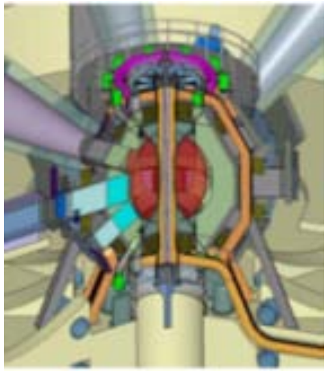
Example ST-FNSF concepts



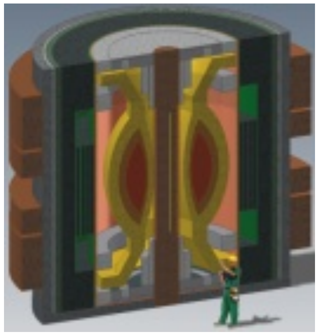
PPPL



ORNL



Culham (UK)



UT Austin

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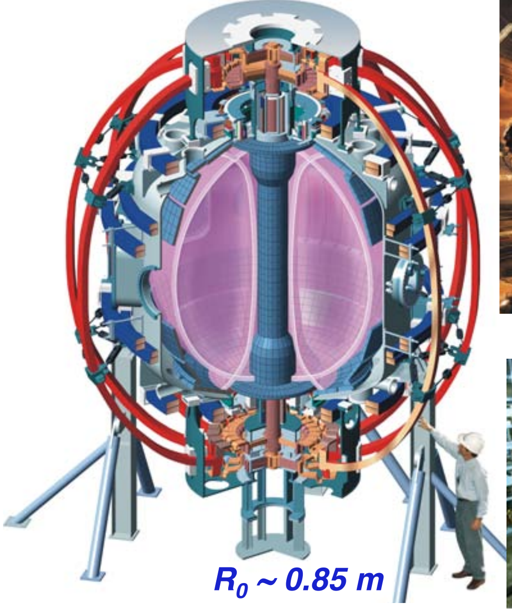
Operating ST Research Facilities Since 2000

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

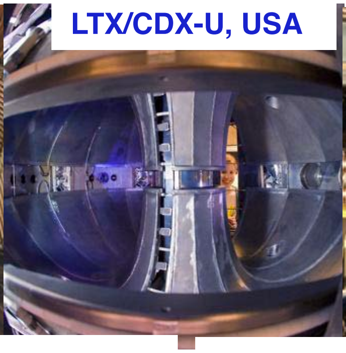
NSTX, USA



$R_0 \sim 0.85$ m

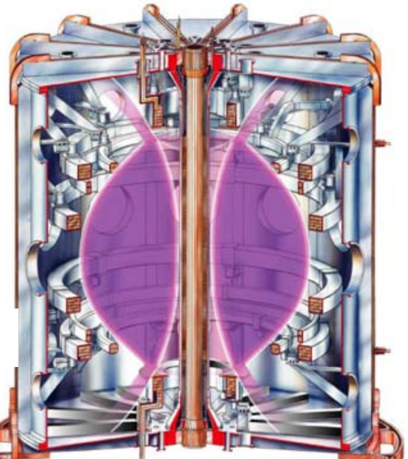
PEGASUS, USA

LTX/CDX-U, USA



GLOBUS-M, Russia

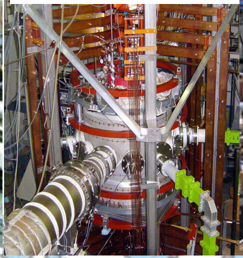
MAST, ***UK***



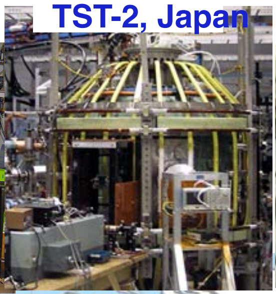
$R_0 \sim 0.75$ m

HIST, Japan

LATE, Japan

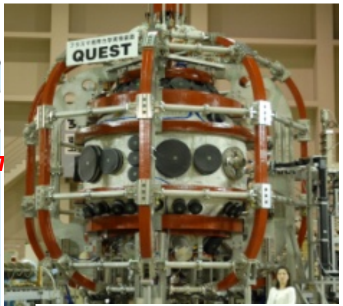


TST-2, Japan



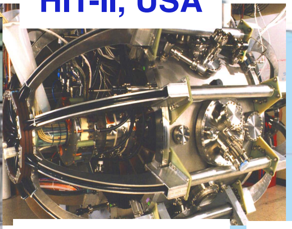
UTST, Japan

QUEST/CPD, Japan



VEST, Korea

HIT-II, USA



STs operated since 2000

SNIST, China

ETE, Brazil

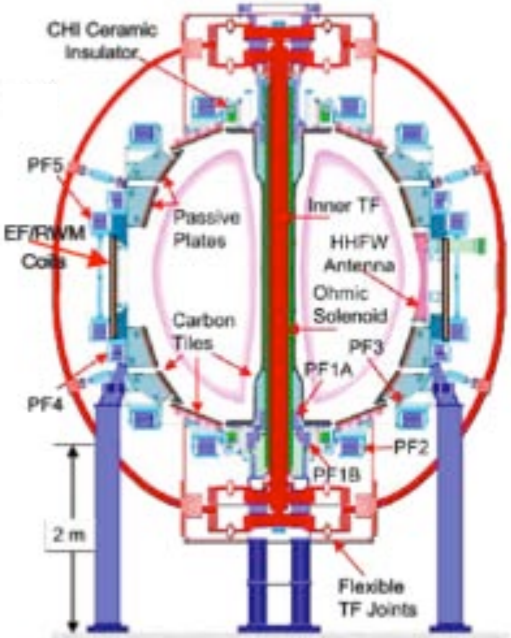
ST25, UK

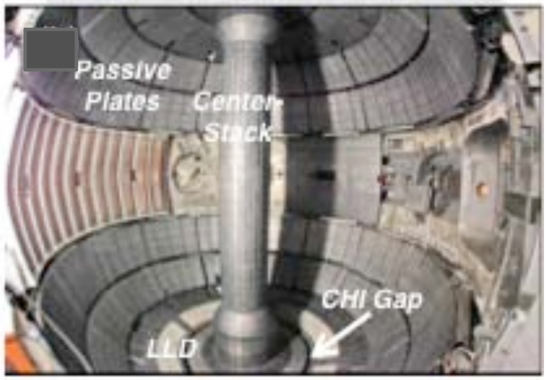
TS3/4, Japan

MA-Class ST Research Started in 2000

Complementary Physics Capabilities of NSTX and MAST

NSTX





M. Ono, et al., IAEA 2000, NF 2001

Complementary Capabilities

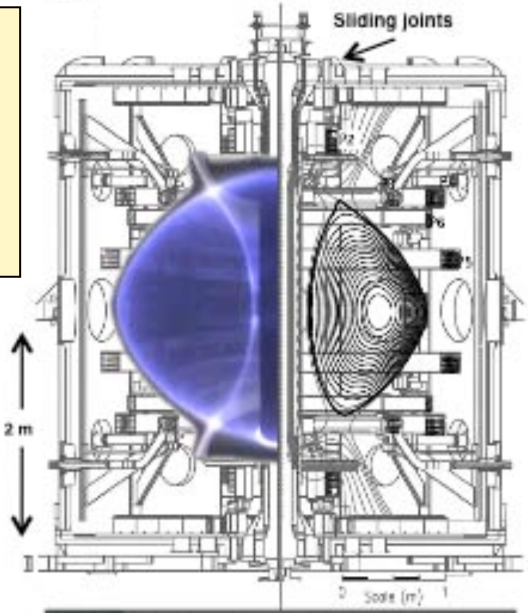
Passive Plates	Large divertor volume
Heilicity Injection	Merging/Compression
HHFW	ECH
1 x 6 RWM Coils	2 x 12 ELM coils

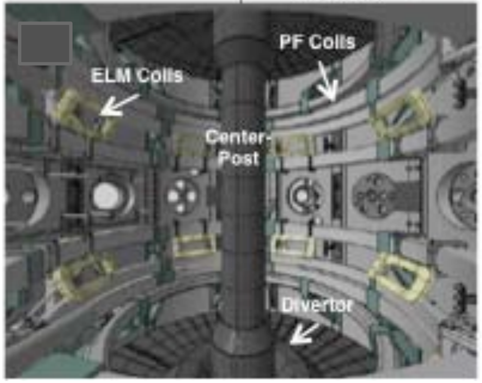
Similar Capabilities

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	$B_T \sim$ 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**

MAST





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

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

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

Product

Physics Knob

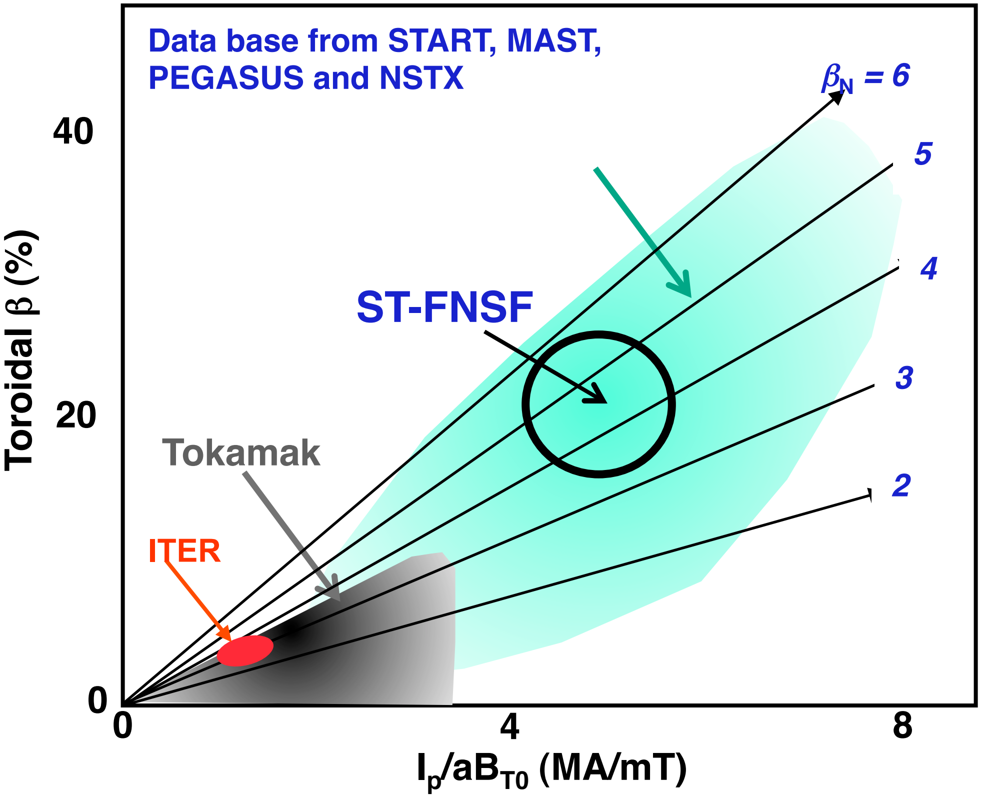
$ Investment

High neutron wall loading W_n possible in a compact ST

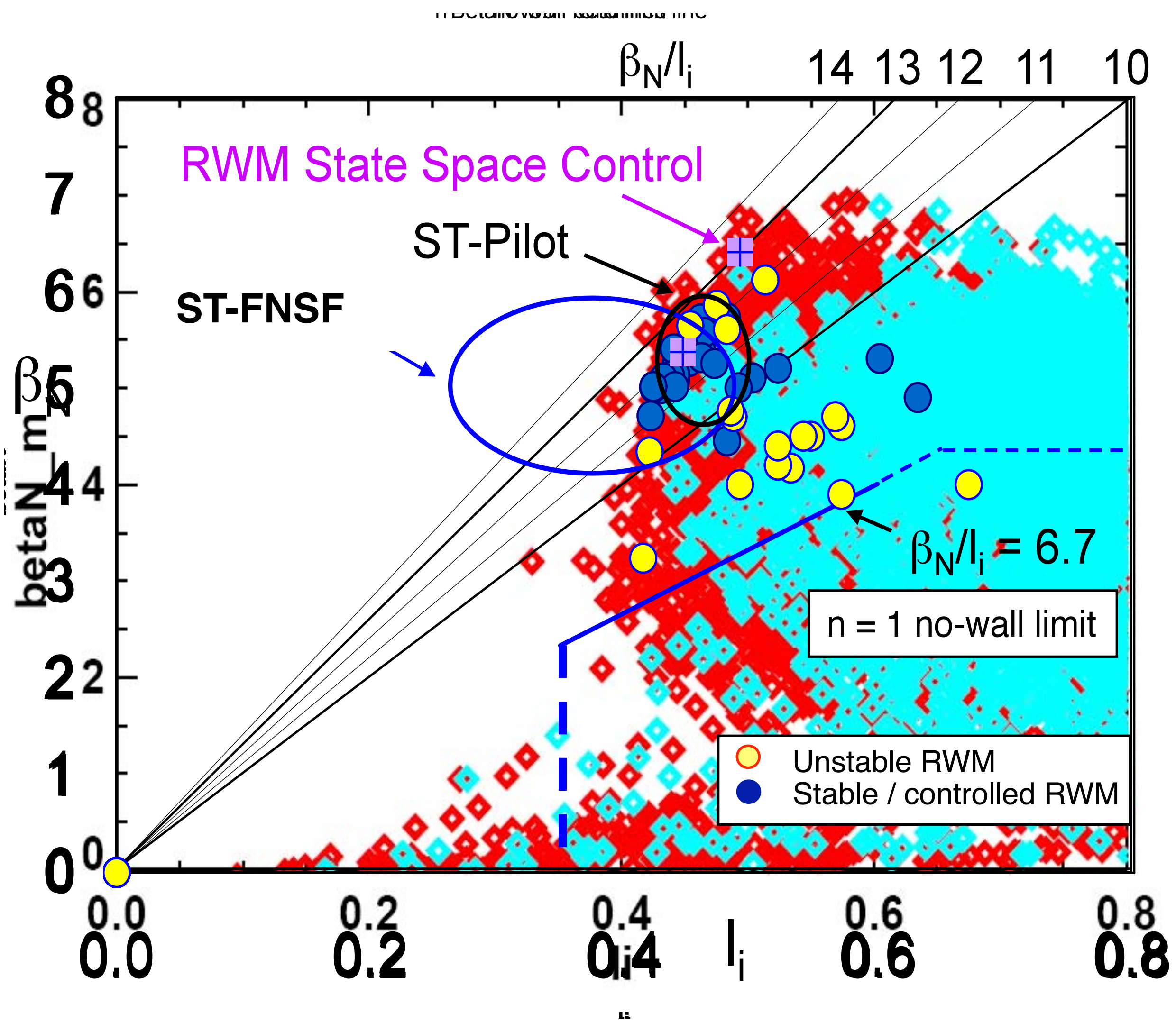
$$W_n \propto P_{fusion} / Area$$

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

⇨ ***W_n ~ 1 MW/m² with R ~ 1 m FNSF feasible!***



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

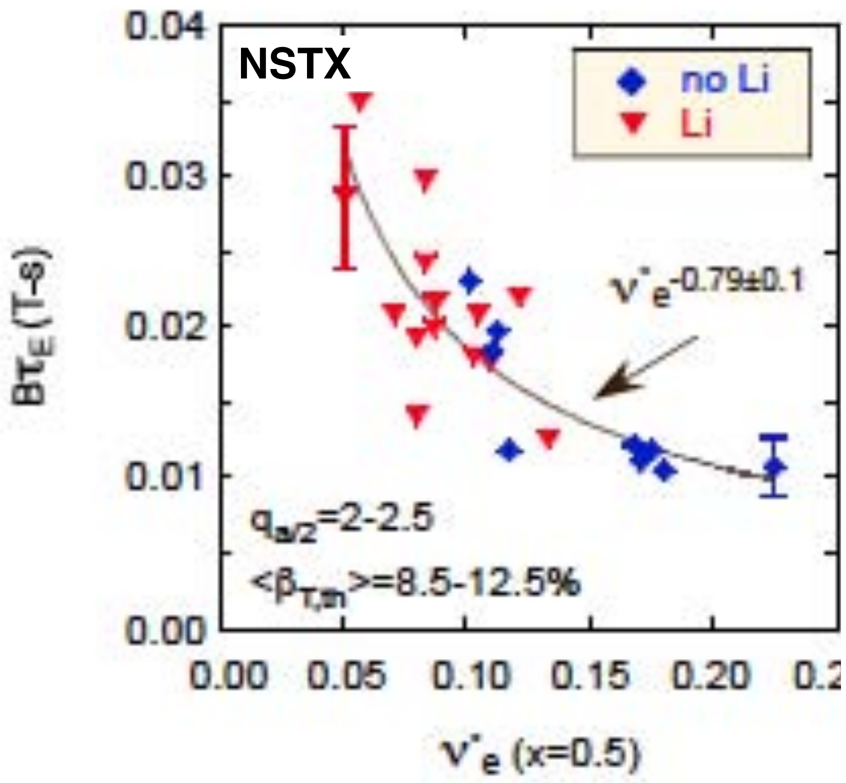
Major mission of NSTX-U is to achieve fully non-inductive operations at high β

Favorable Confinement Trend with Collisionality and β found

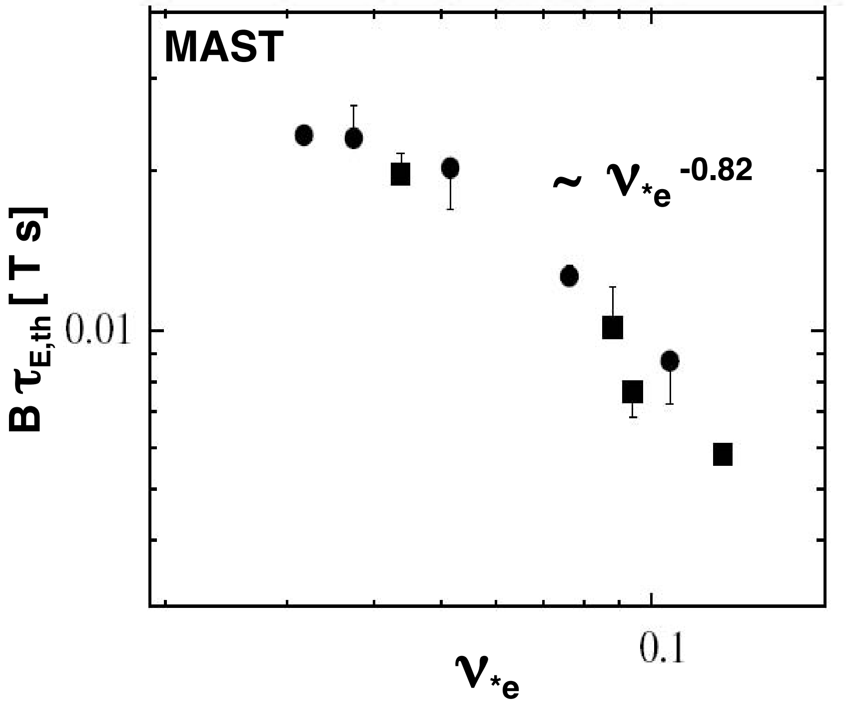
Important implications for future STs and Demo with much lower ν_*

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

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



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

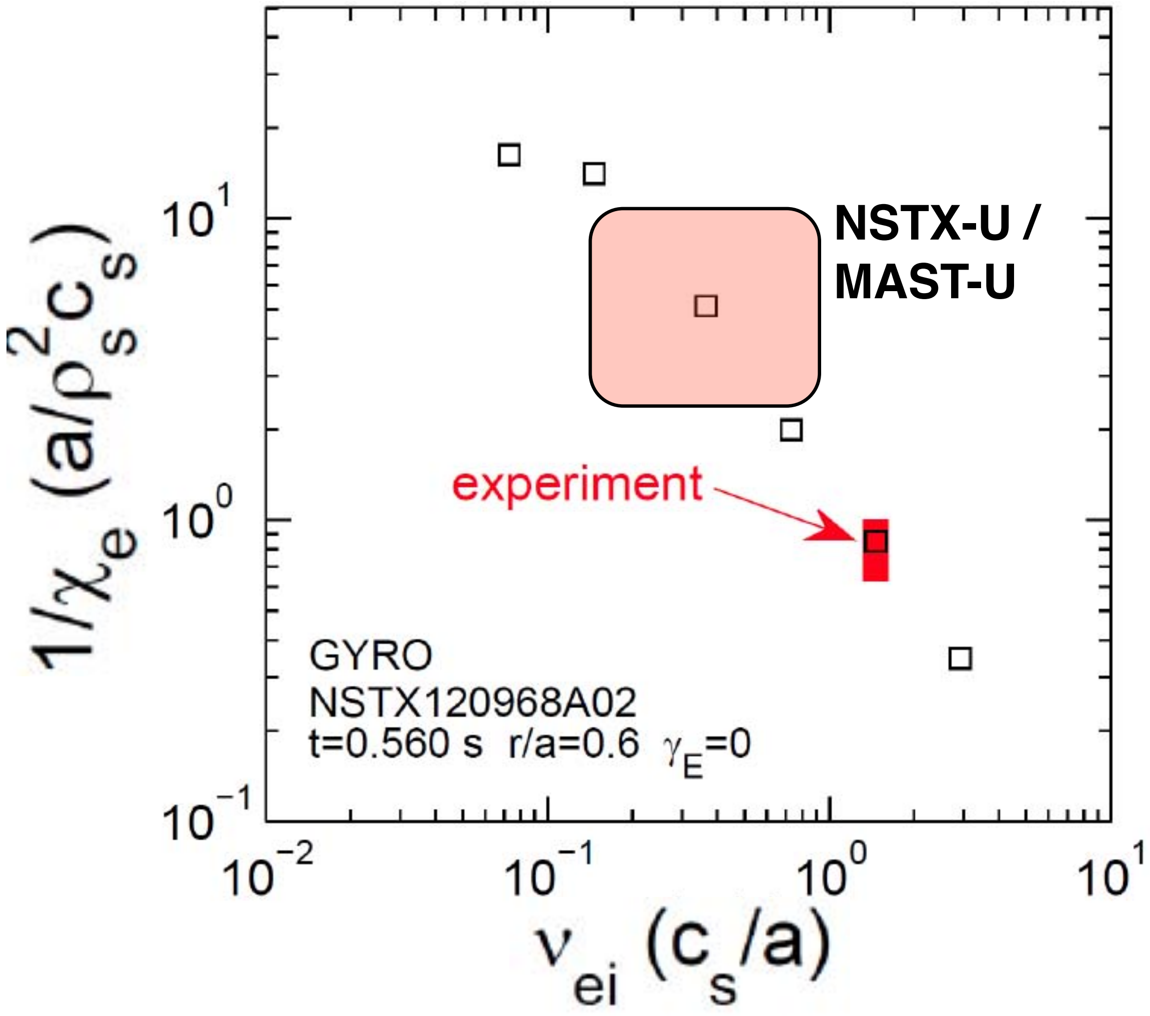


M. Valovic et al., NF (2011)

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.



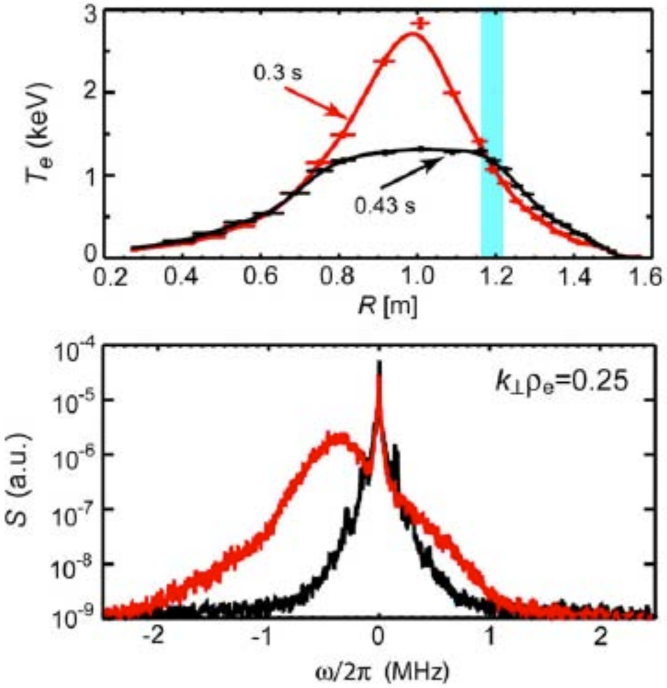
- **MT growth rate decreases with reduced collisionality in qualitative agreement with the NSTX experiment.**
- **Futher electron confinement improvement expected for NSTX-U and MAST-U due to reduced collisionality.**

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

ETGs measured for the first time with high-k scattering

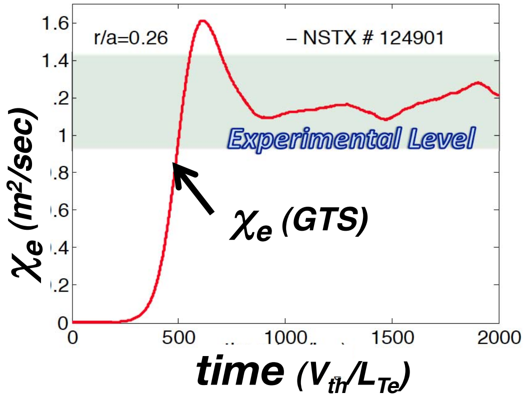
High β_e or larger $\rho_e \propto \beta_T^{0.5}$ of ST plasma enabled measurement of ETGs.

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



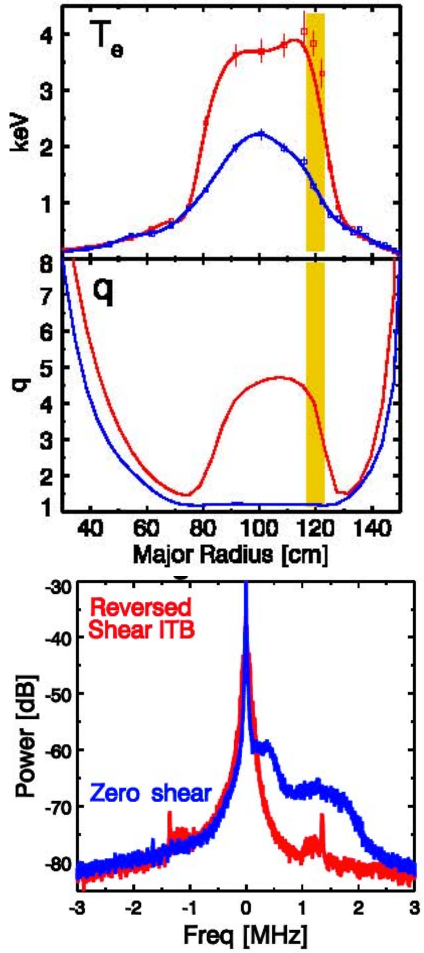
E. Mazzucato et al., NF (2009)

Calculated ETG χ_e by GTS code agrees with experiment



W. Wang et al.,

ETG Suppression in Reversed Shear



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

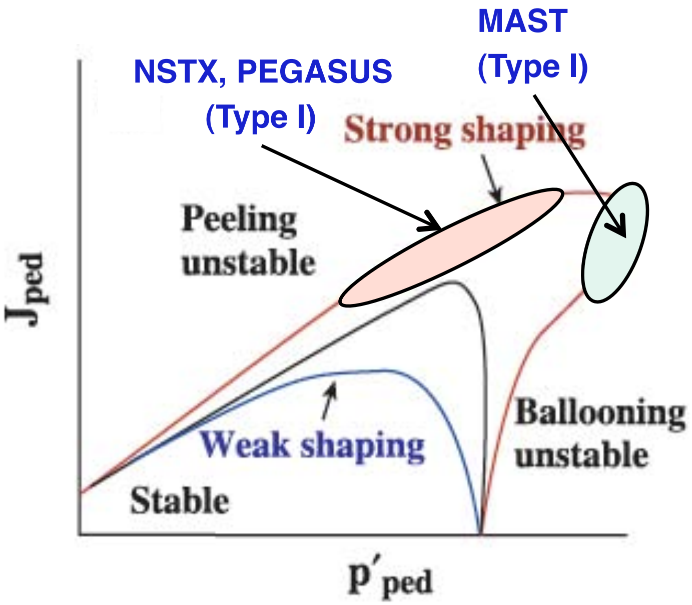
- **Shear stabilization of ETGs** D. R. Smith, et al., PRL (2009)
- **Density gradient stabilization of ETGs** Y. Ren, et al., PRL (2011)

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

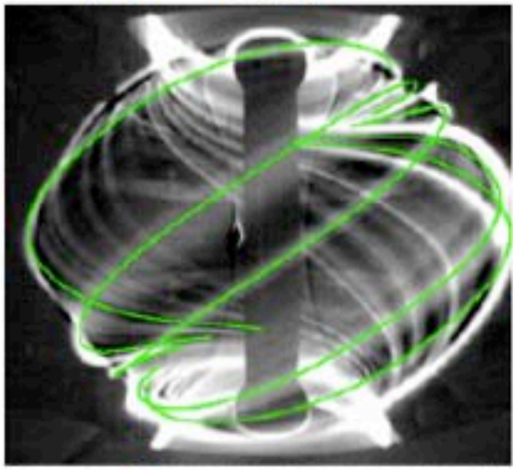
H-mode / ELM physics: High Priority Research Goal
Unmitigated ELMs could cause PFC damage in reactors

ST is in strongly shaped ELM regimes

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

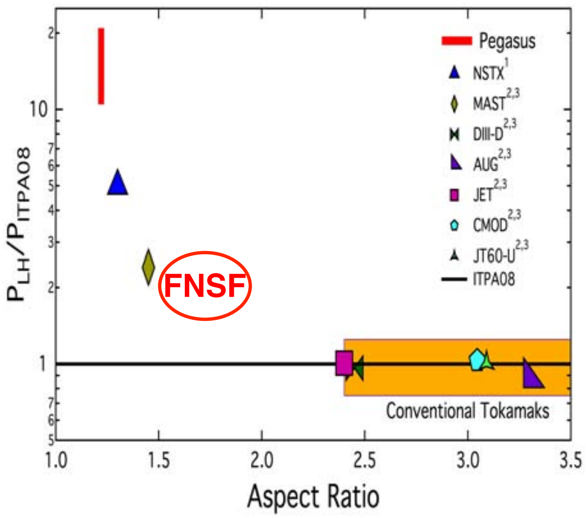


Video images of MAST plasmas showing a filamentary ELM structure.



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

L-H power threshold scaling extended for low A

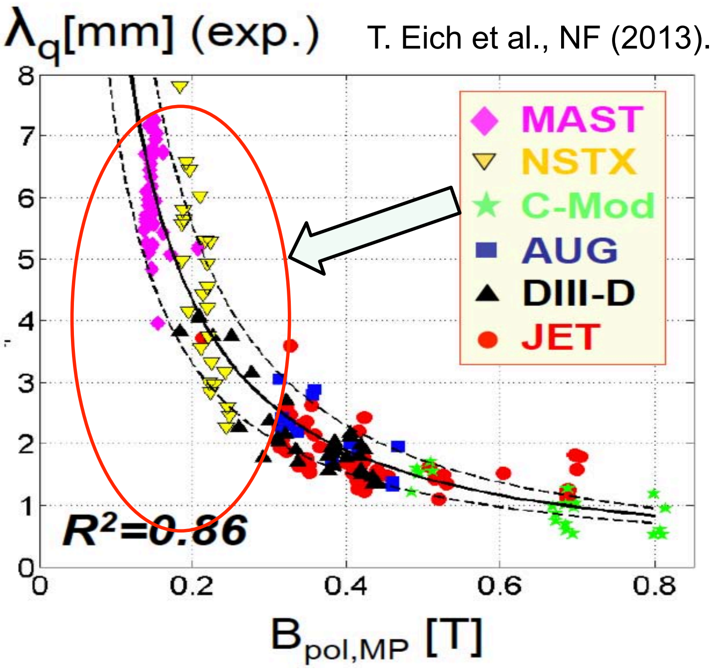


K.E. Thome et al., EPR (2014)

- **NSTX/MAST/PEAGASUS 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.

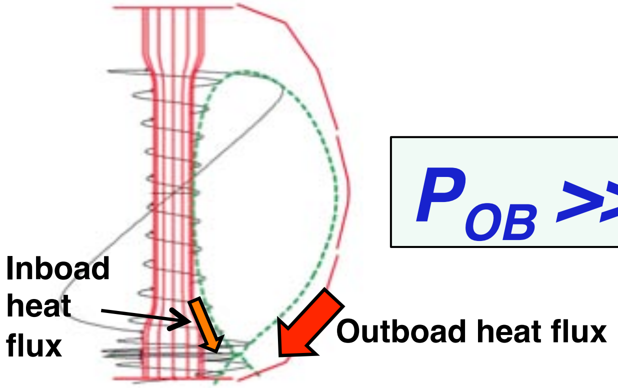


T. Eich et al., NF (2013).

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

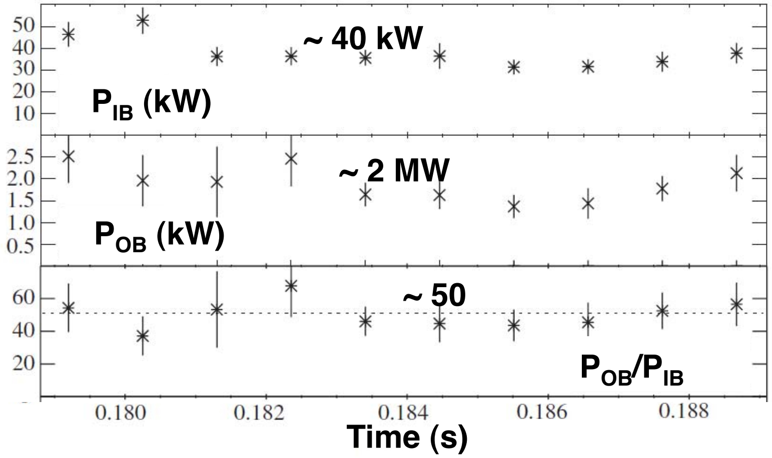
- * Unfavorable for large size, Ip devices such as ITER and Demo
- "P B / R" as the new heat flux metric which is favorable for STs

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



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

Ratio of outboard power flux vs. inboard in MAST



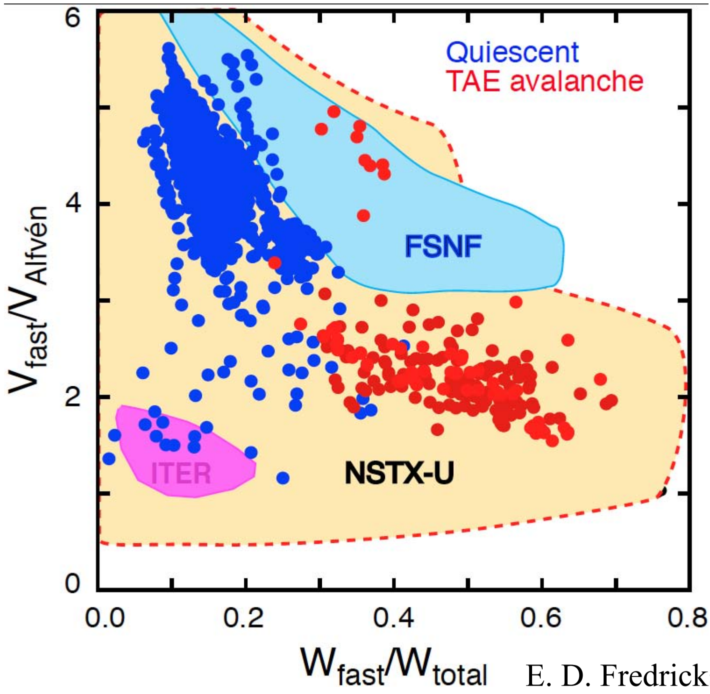
G.F. Counsell et al., PPCF (2002)

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/MAST-U will also explore $V\alpha < V_{Alf}$ regime giving more flexibility**

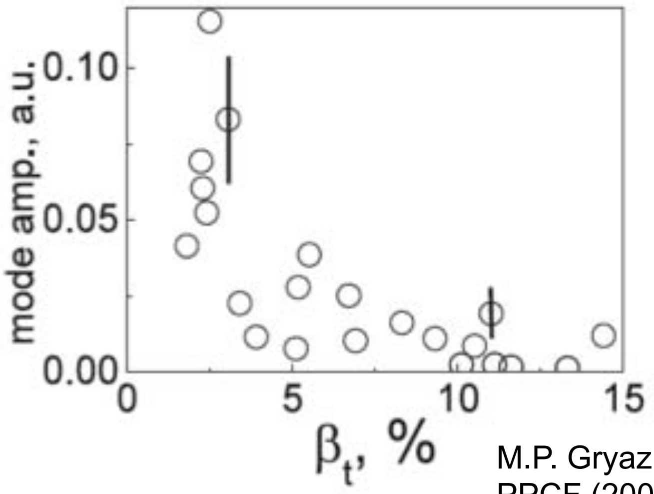
EP parameter space



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

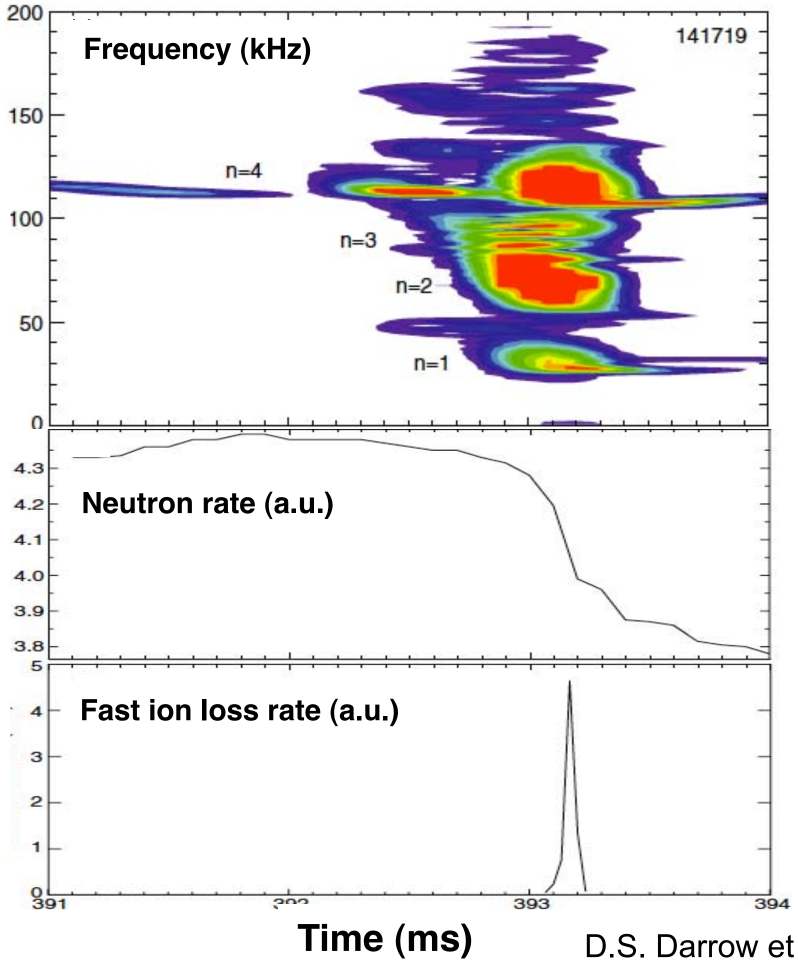


M.P. Gryaznevich et al. PPCF (2004)

“TAE avalanche” shown to cause energetic particle loss

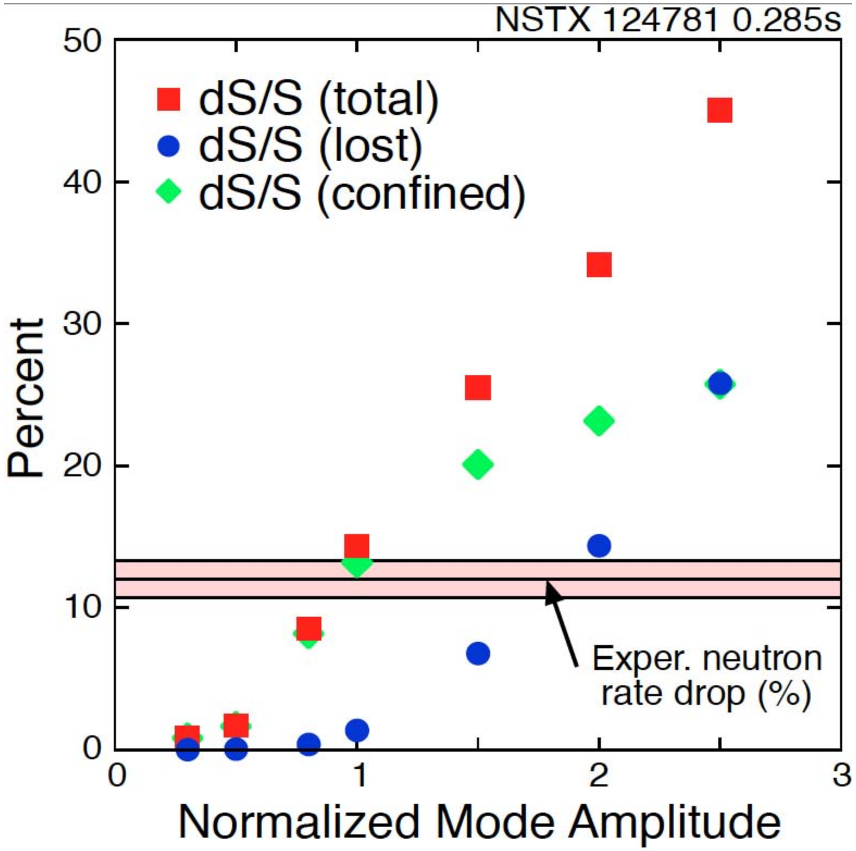
Uncontrolled α-particle loss could cause reactor first wall damage

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



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

Progress in simulation of neutron rate drop due to TAE avalanche



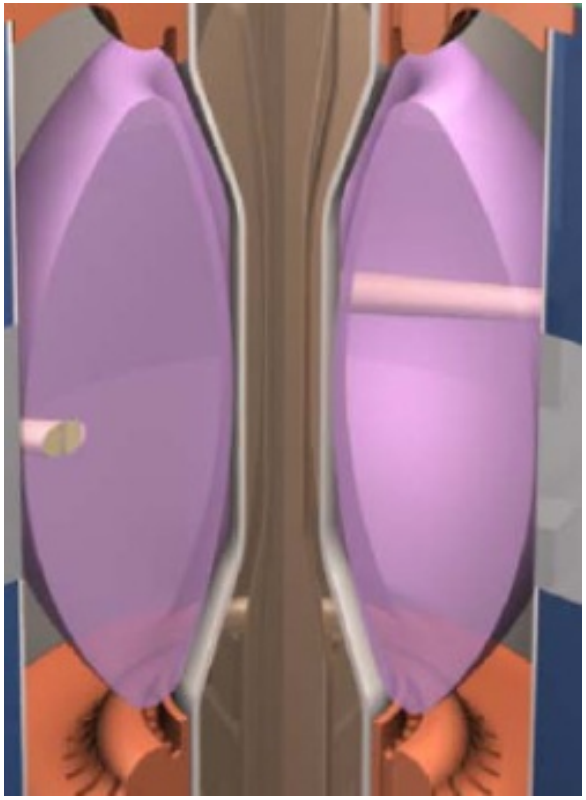
E. D. Fredrickson et al., NF (2013)

Talk Outline

- Unique ST properties
- ST Fusion Energy Development Path
- World ST Facilities
- Unique ST Physics Regimes
- **ST-FNSF Targeted Experiments**
- ST Facility Upgrade Status
- Summary

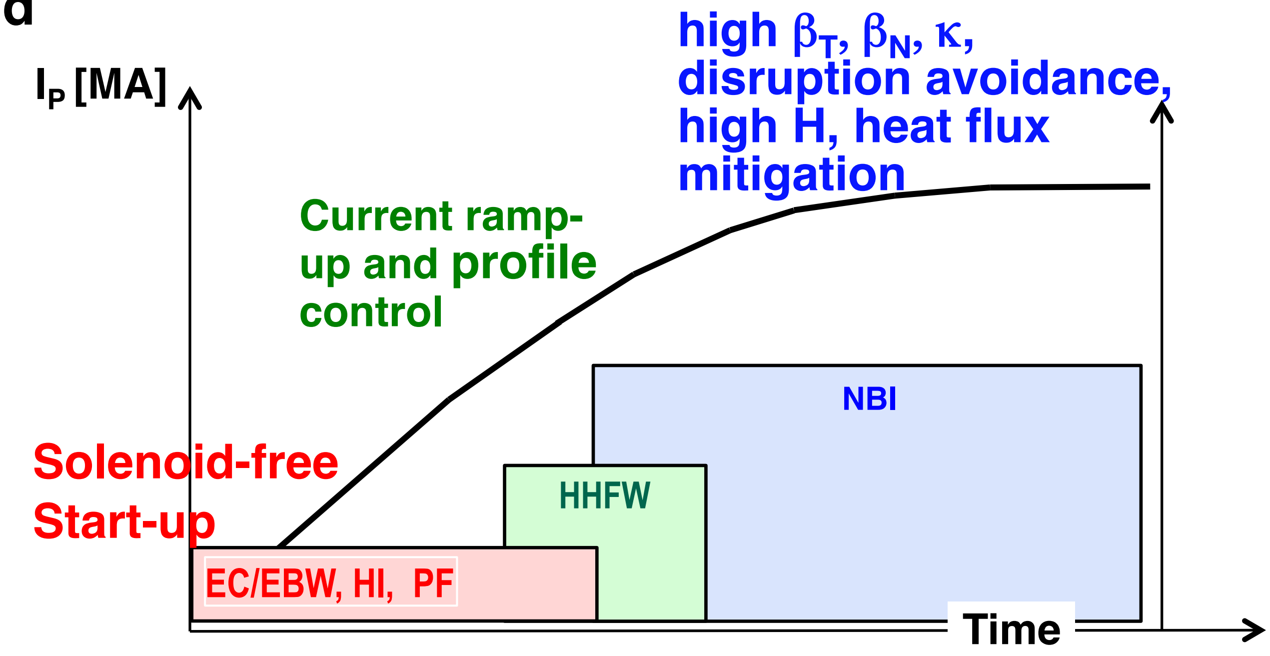
STs Addressing Critical Issues for FNSF and Demo

Compact ST-FNSF has no/small central solenoid



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

ST-FNSF Scenarios

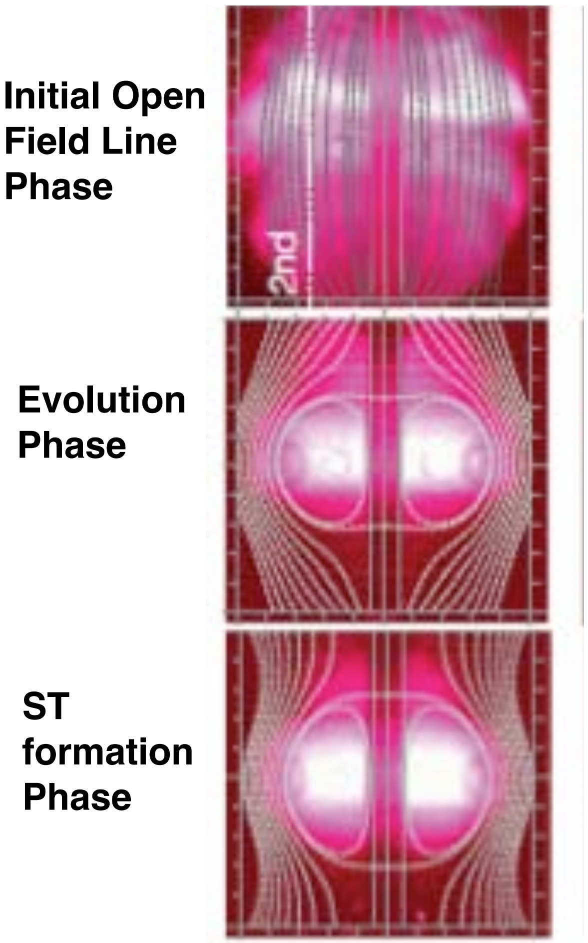


- **Three novel techniques for solenoid-free start-up and ramp-up investigated**
  - **ECH/EBW**
  - **Helicity Injection**
  - **Merging-compression**

Efficient ECH/EBW start-up and sustainment demonstrated

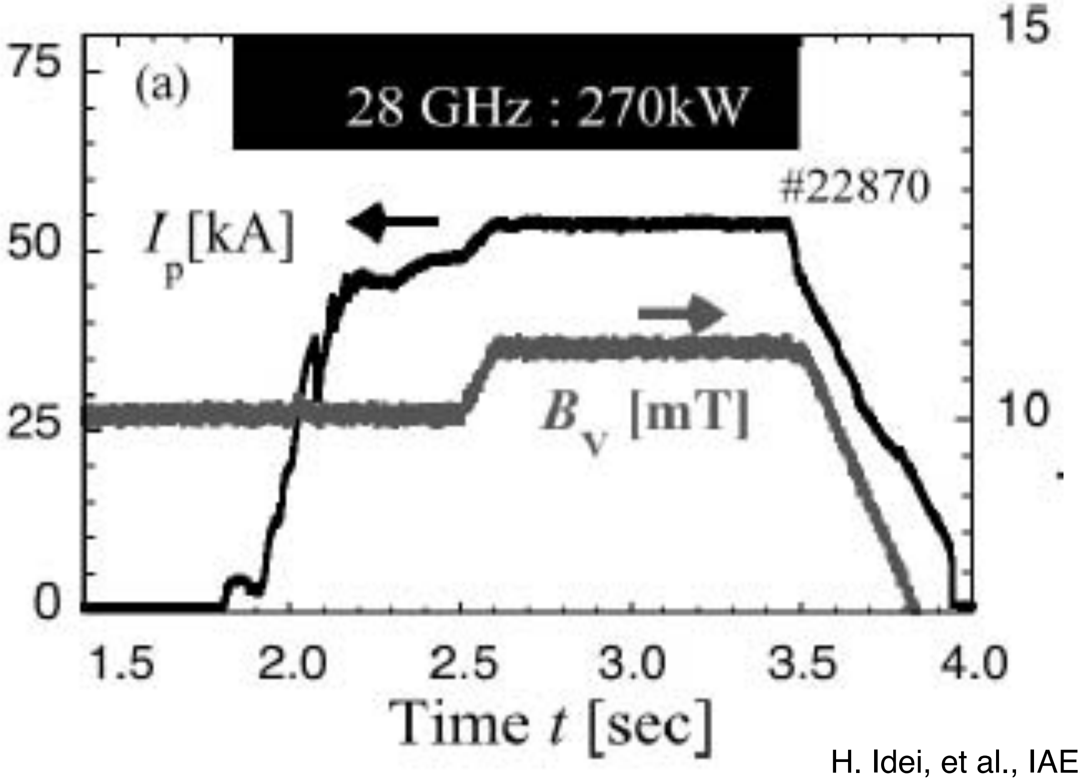
RF start-up investigated in CDX, TST-2, LATE, MAST, QUEST, VEST, SUNIST

LATE ECH/EBW



M. Uchida et al., PRL (2010).

55 kA sustained with 270 kW in QUEST, Japan



H. Idei, et al., IAEA 2014

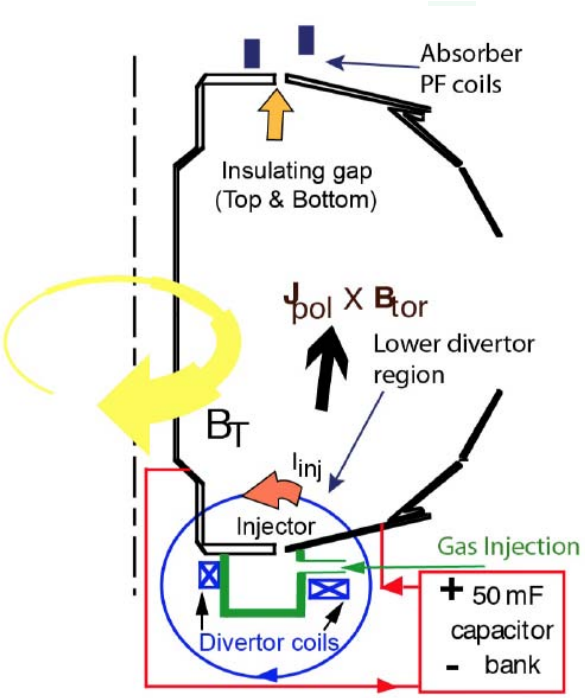
72 kA start-up current achieved with 75 kW in MAST, UK

T.S. Bigelow et al., ISTW 2013

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

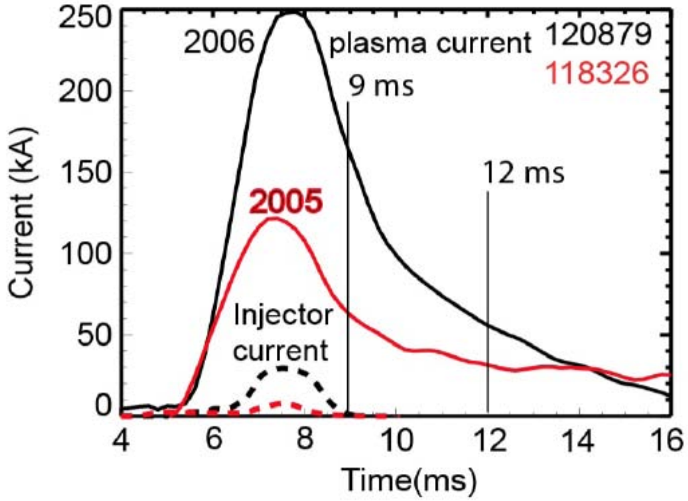
Helicity Injection Is an Efficient Method for Current Initiation
Coaxial Helicity Injection (CHI) Concepts Being Devloped

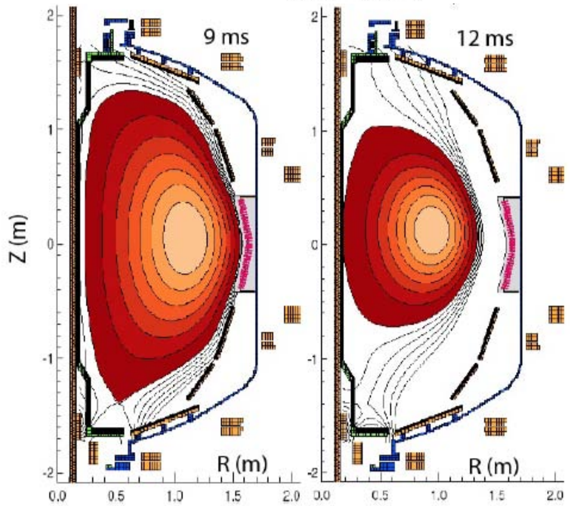
CHI developed on HIT and HIT-II and transferred to NSTX



R. Raman et al., PRL (2006)

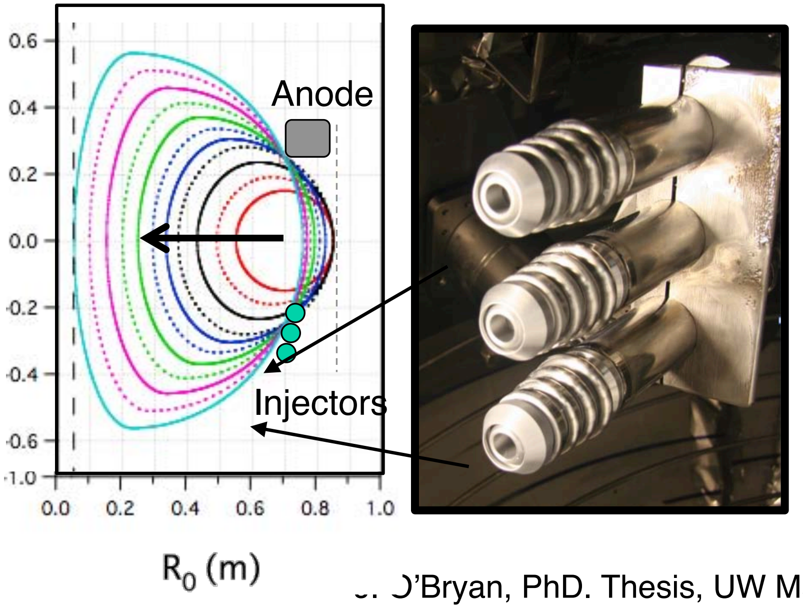
Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX





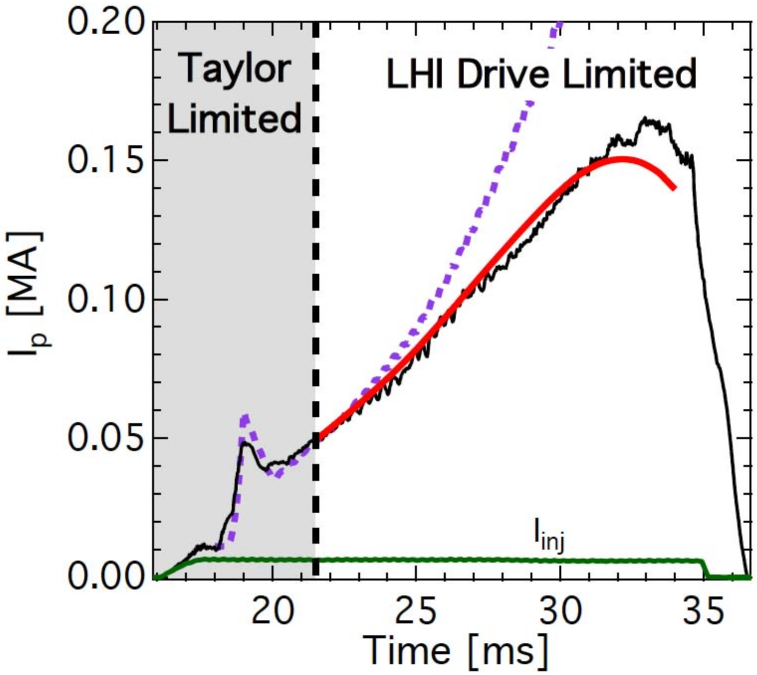
Helicity Injection Is an Efficient Method for Current Initiation
Lpocal Helicity Injection (LHI) Concepts Being Seveloped

3-6 kA current injector array in plasma SOL



O’Bryan, PhD. Thesis, UW Madiso

Long-Pulse Startup Demonstrated in PEGASUS with LHI



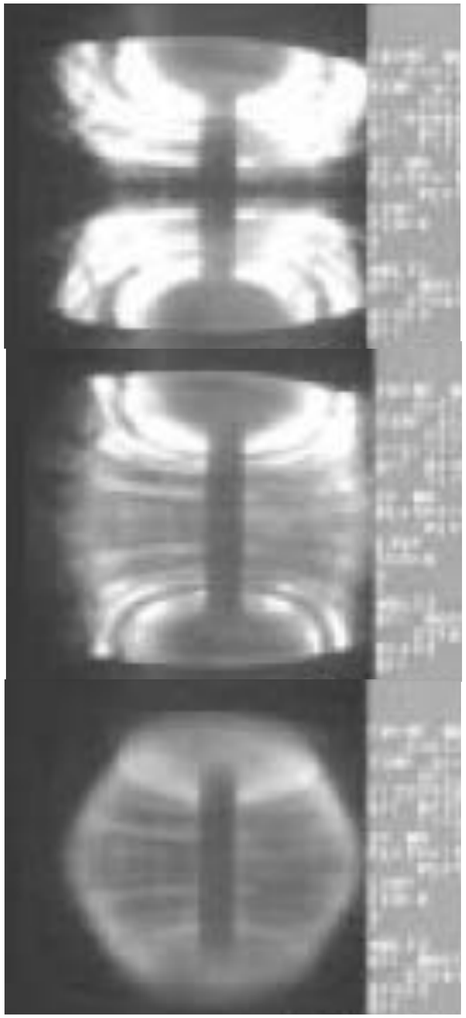
Improving predictive capability for both CHI and LHI

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

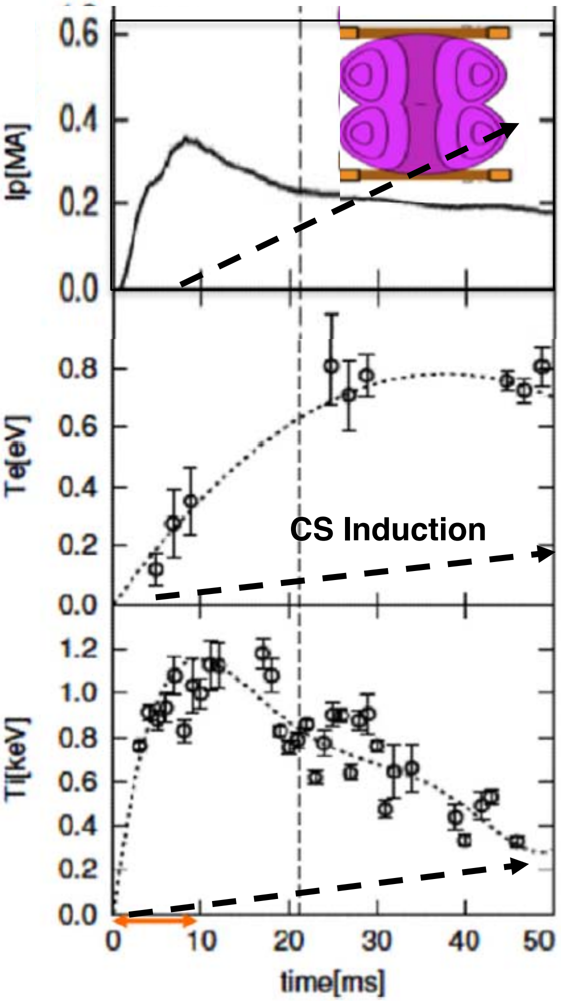
Merging Start-Up Yielded High Current STs

Rapid ion heating observed from magnetic reconnection

Merging-compression start-up in MAST



A. Sykes et al., NF (2001)



Y. Ono, et al., this APS

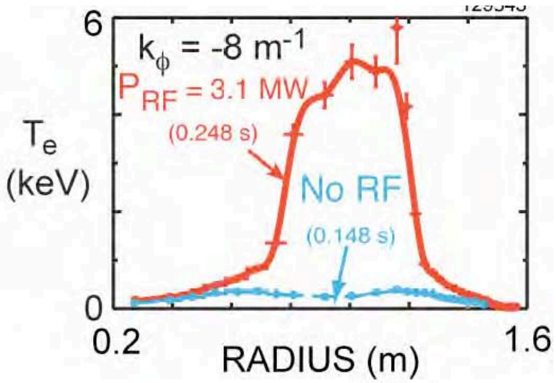
Ultra high β STs produced by mergings in TS-3 Device

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

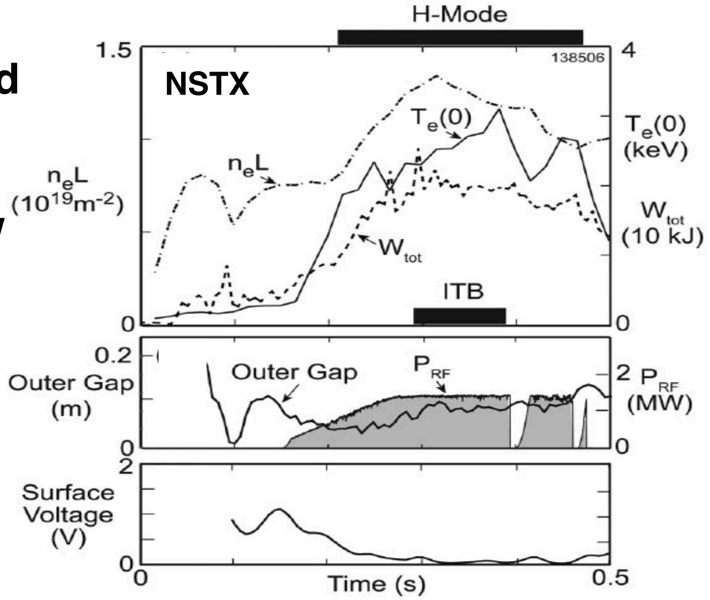
Fig. 22. Schematic of UTST. *Reprinted with permission from Y. Ono.*

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.

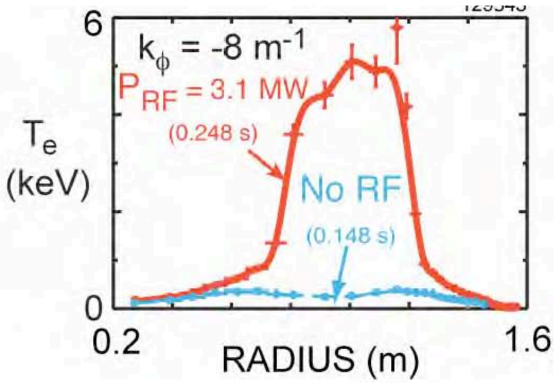


G. Taylor et al., PoP (2010), (2012)

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.



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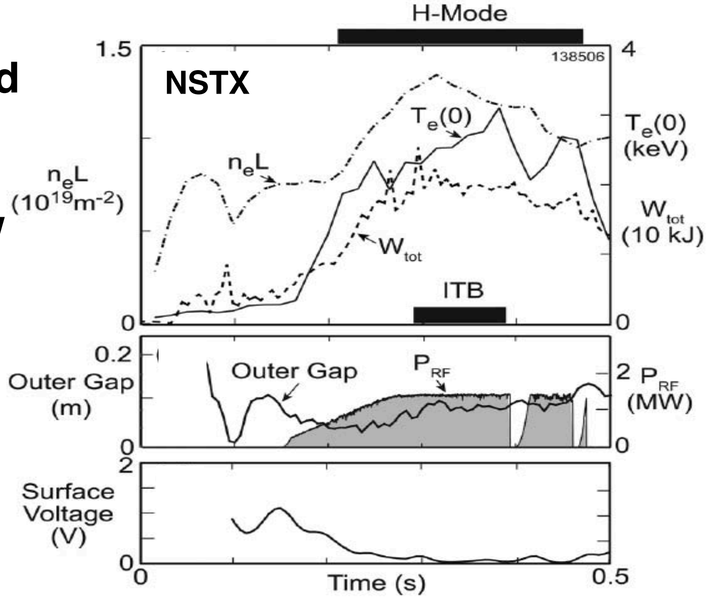
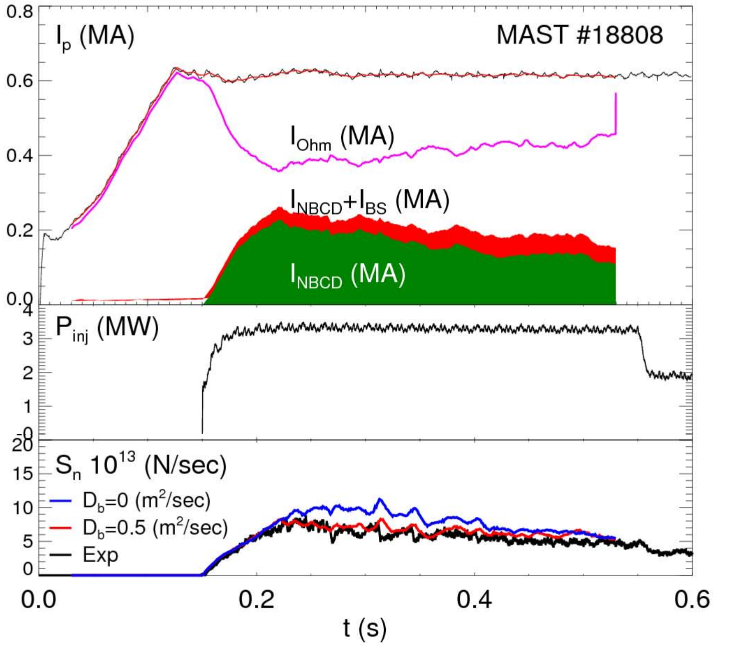


G. Taylor et al., PoP (2010), (2012)

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

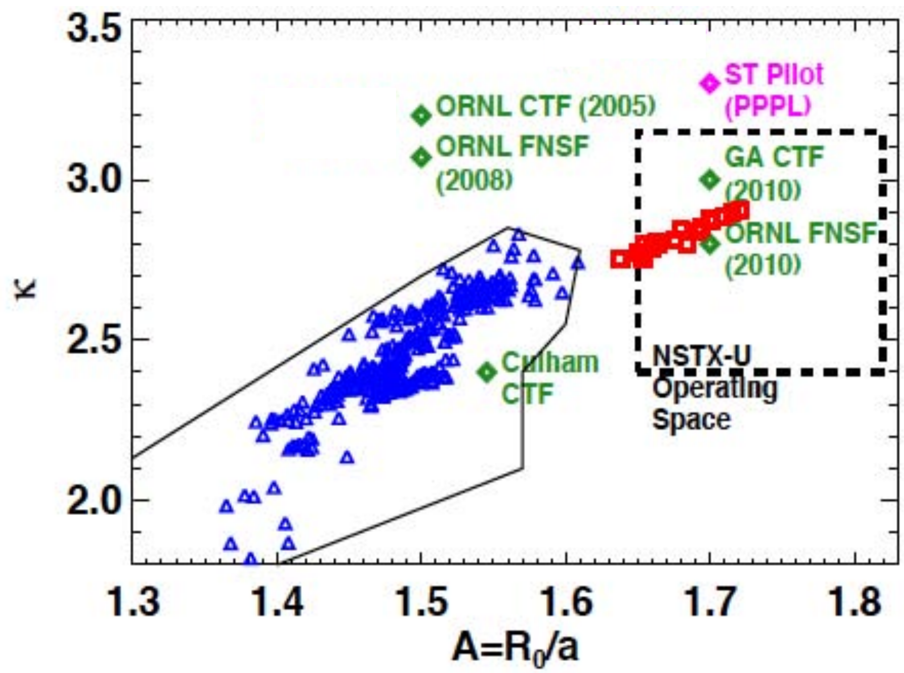


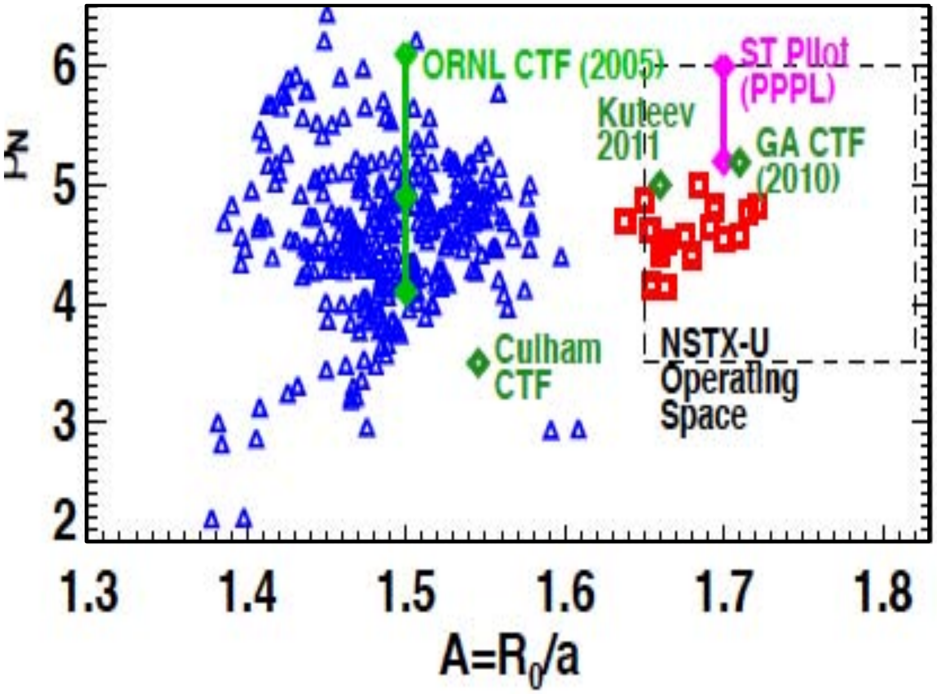
M. Turnyanskiy et al., NF (2009)

Off-axis current drive for profile control will be tested in both MAST-U and NSTX-U with major NBI upgrades.

NSTX has accessed A, β_N, κ needed for ST-based FNSF
Requires $f_{BS} \geq 50\%$ for plasma sustainment

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

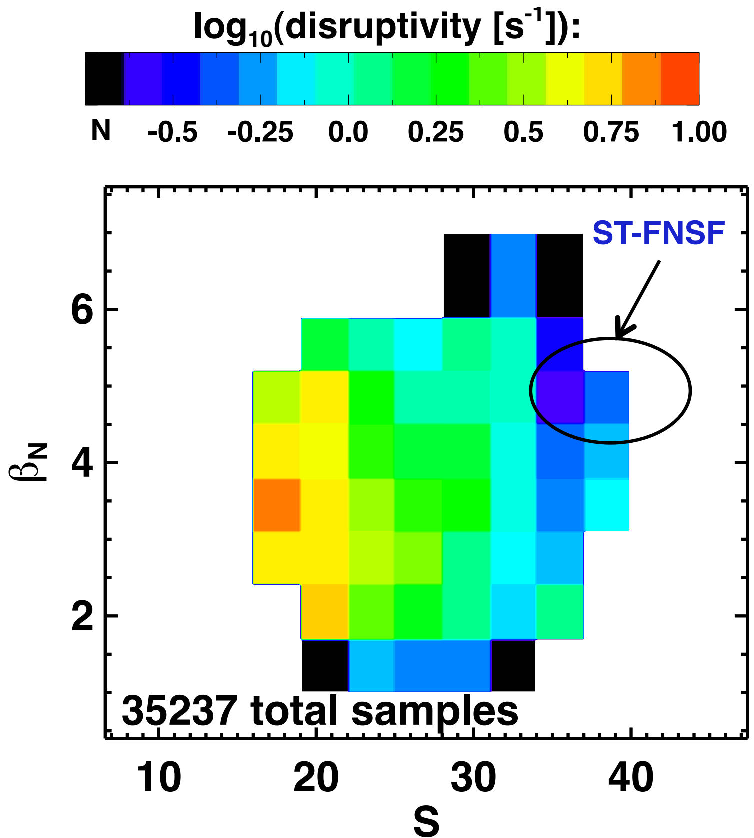




S.P. Gerhardt et al., NF (2011)

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

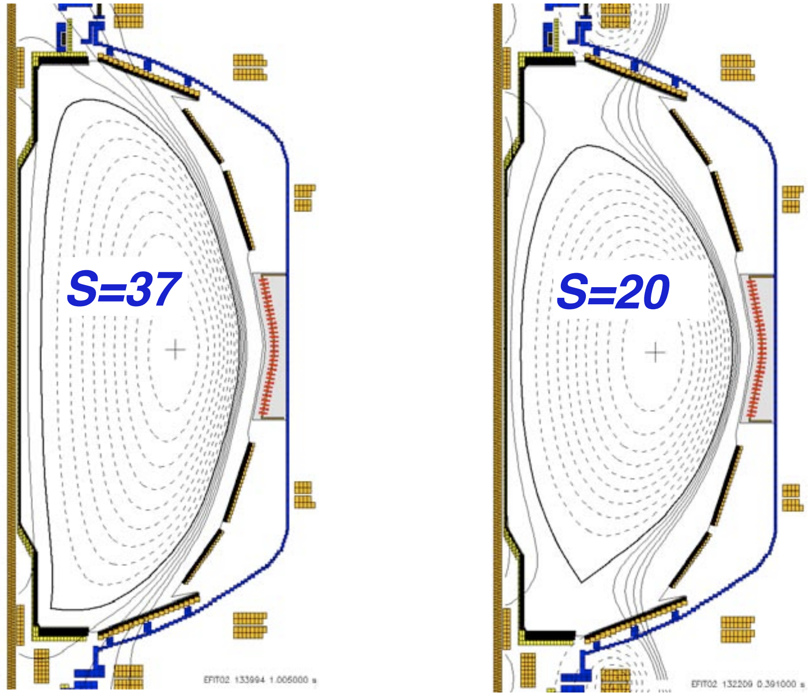
NSTX Data Demonstrates a Favorable Operations Window For Reduced Disruptivity in an ST-FNSF



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

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

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



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

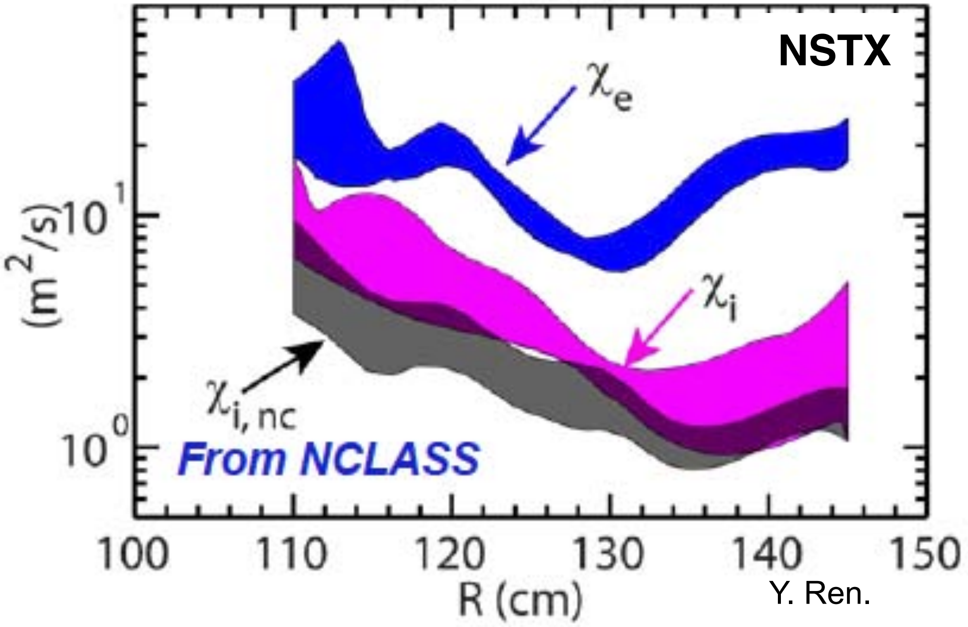
Upgrades will test and improve these favorable trends in a systematic way

High Confinement Needed for Compact FNSF

High confinement H-mode in the range of FNSF obtained

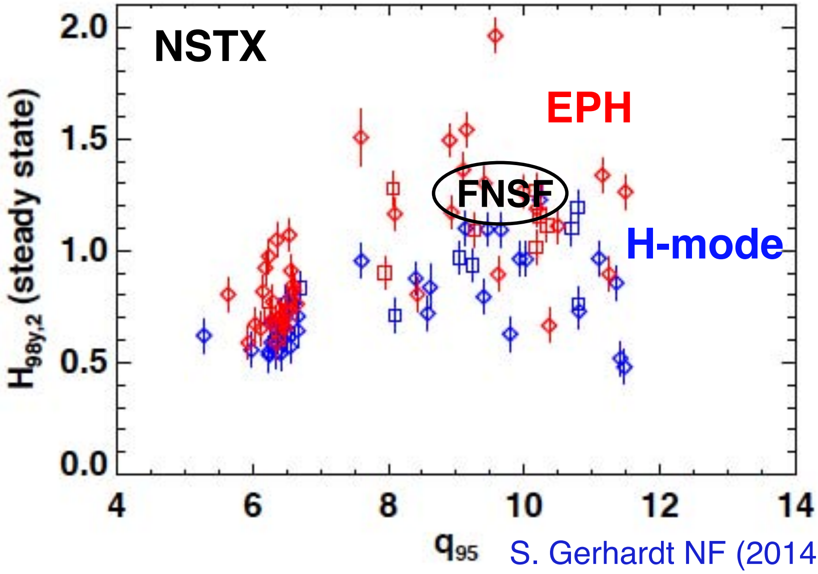
- **Fusion gain Q depends strongly on "*H*", $Q \propto H^{5-7}$**
- **Higher H enables compact ST-FNSF H = 1.2 – 1.3**
- **Higher H gives more reactor design flexibility and margins.**

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



Y. Ren.

H-mode confinement in STs H ~ 1 but enhanced pedestal H-mode (EPH) has 50% higher H up to H ~ 2



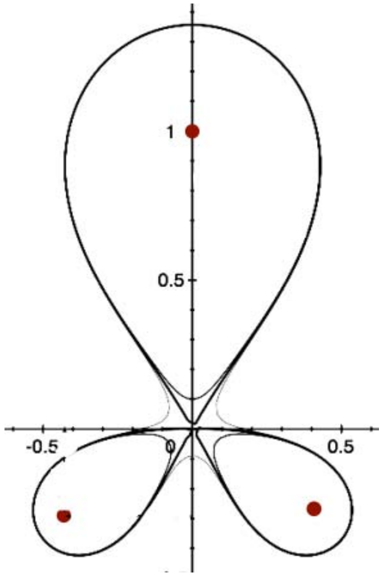
S. Gerhardt NF (2014)
R. Maingi, PRL (2010)

ST-FNSF has high P/R due to small R

Innovative Heat Flux Mitigation via Divertor Flux Expansion

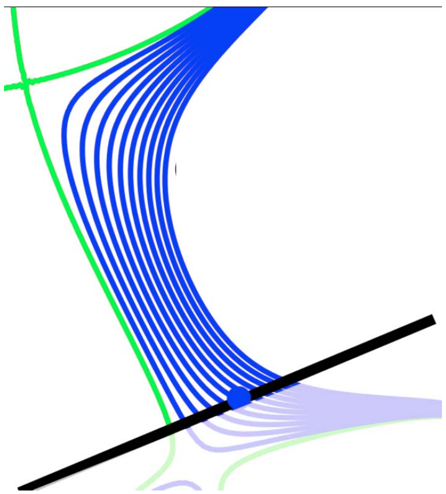
Lower toroidal field of outboard divertor leg of STs facilitates heat flux mitigation by divertor flux expansion solutions

Snow-flake



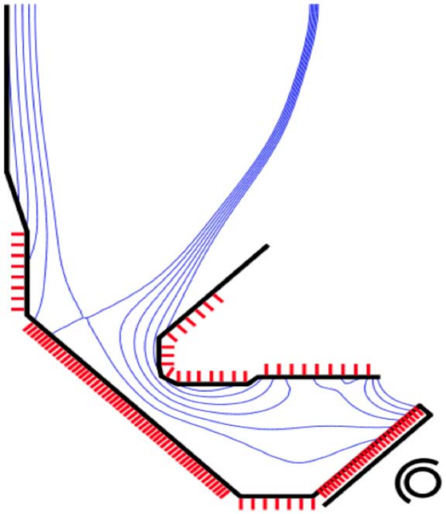
D. Ryutov, et al., PoP (2007)

X-Divertor: CREST



P.M. Valanju, et al., PoP (2009).

Super-X: MAST-U

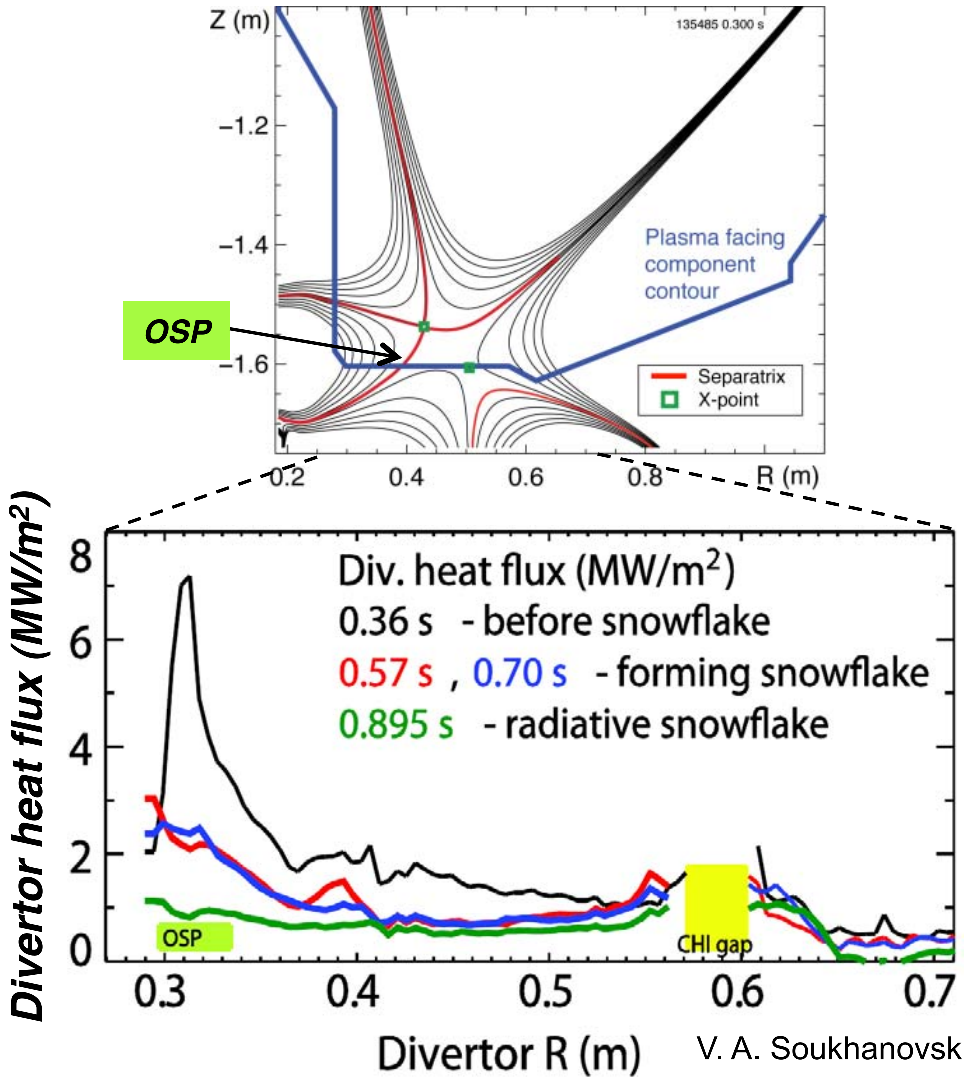


Kotschenreuther, et al., PoP (2007)

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



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 Divertors**
- **Lithium + high-z metal PFCs**

Talk Outline

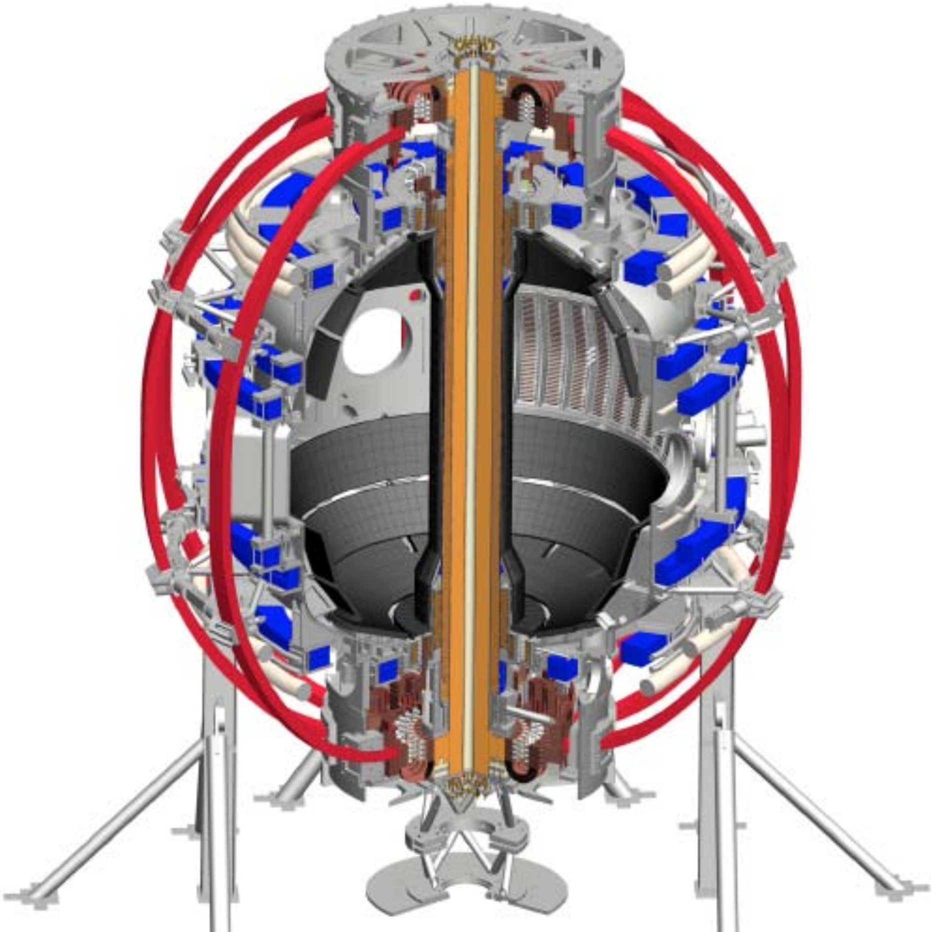
- Unique ST properties
- ST Fusion Energy Development Path
- World ST Facilities
- Unique ST Physics Regimes
- ST-FNSF Relevant Experiments
- **ST Facility Upgrade Status**
- Summary

NSTX and MAST are undergoing major upgrades

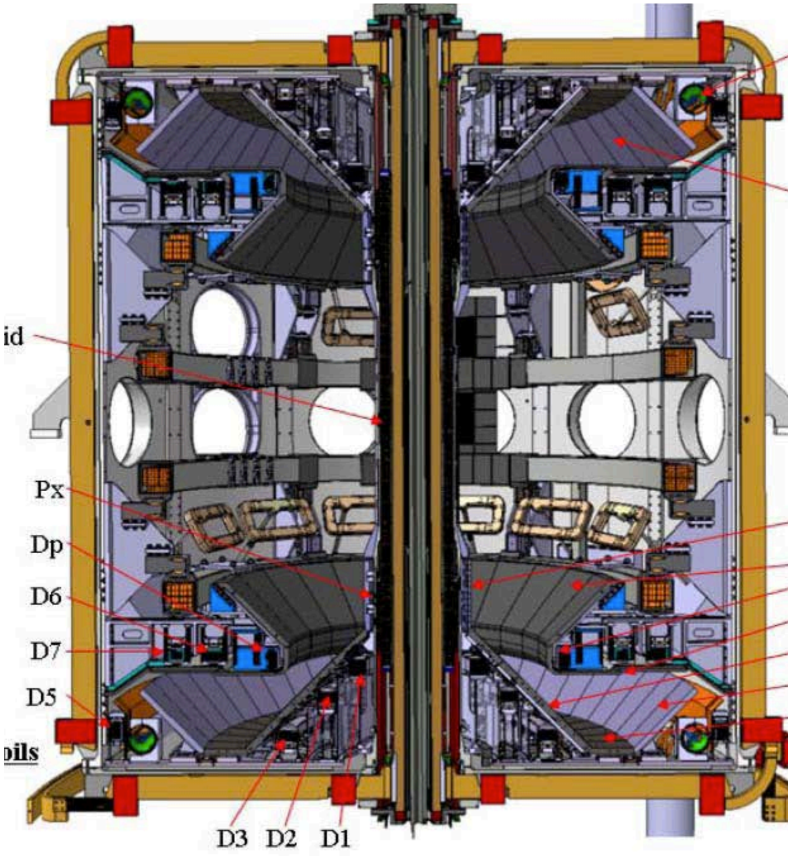
~ x 2 B_T, I_p, P_{NBI} and ~ x5 pulse length from NSTX/MAST

NSTX-U

MAST-U



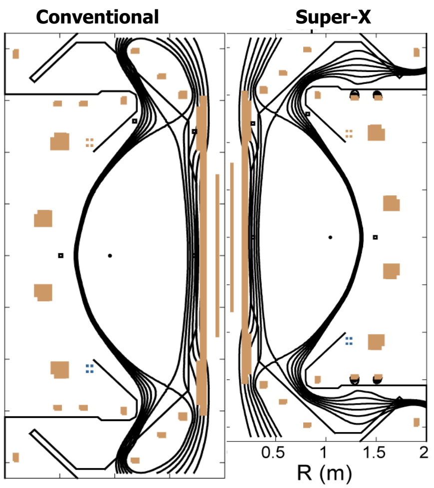
Highly tangential 2nd NBI for non-inductive operations



Super X-divertor configuration for FNSF divertor solution

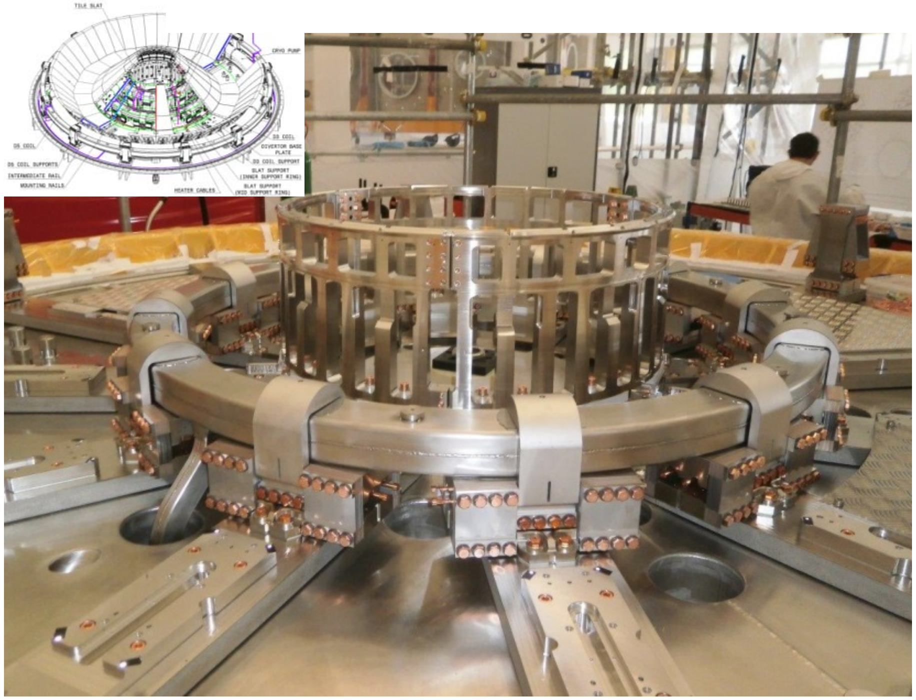
MAST-U to support novel exhaust concepts, ITER, and FNSF
Completed MAST operation in 2013 and began construction

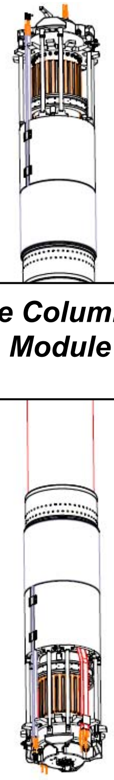
Super X Divertors



- 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.

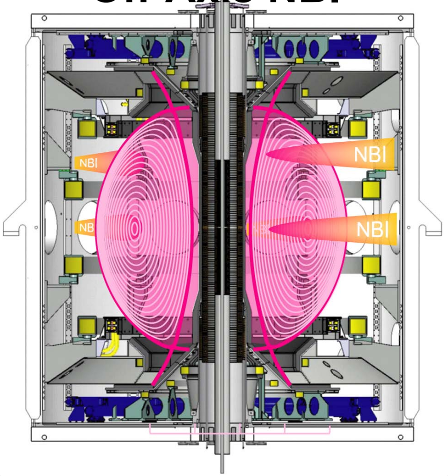
End Plate Modules (LEP)





Centre Column (CC) Module

Off-Axis NBI



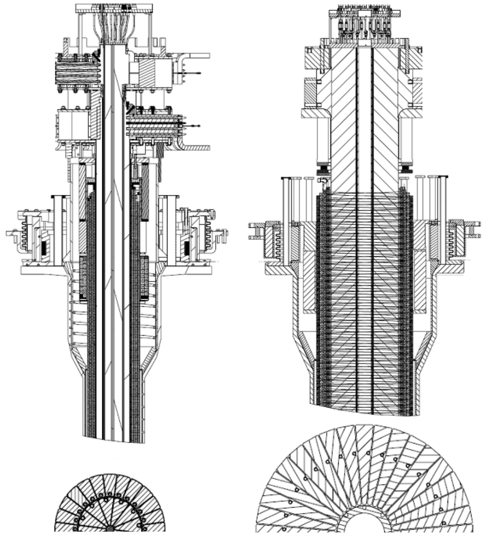
First plasma scheduled in 2016

J. Milnes SOFT 2014

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

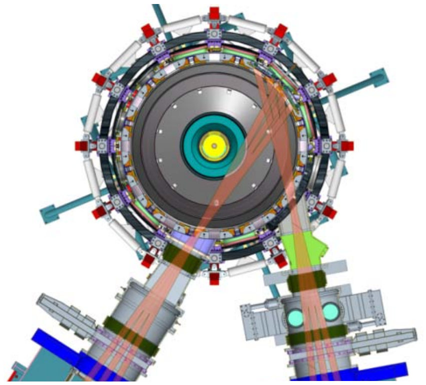
Previous center-stack

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 $n\tau T$ from NSTX

NSTX-U average plasma pressure ~ Tokamaks

Key NSTX-U research topics for FNSF and ITER

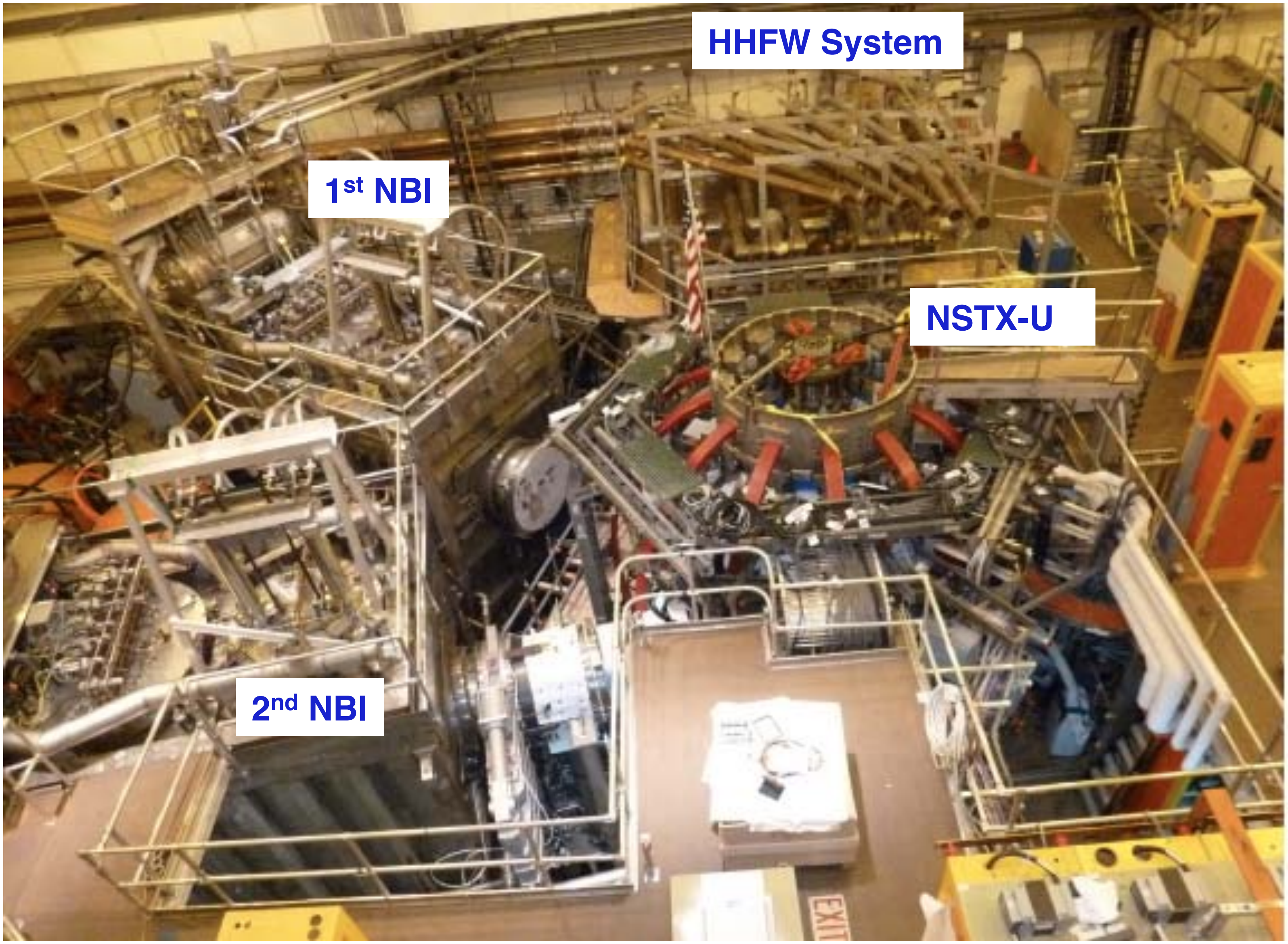
- **Stability and steady-state control at high β**
- **Confinement scaling (esp. electron transport)**
- **Non-inductive start-up, ramp-up, sustainment**
- **Divertor solutions for mitigating high heat flux**

J. Menard, et al., NF (2012)

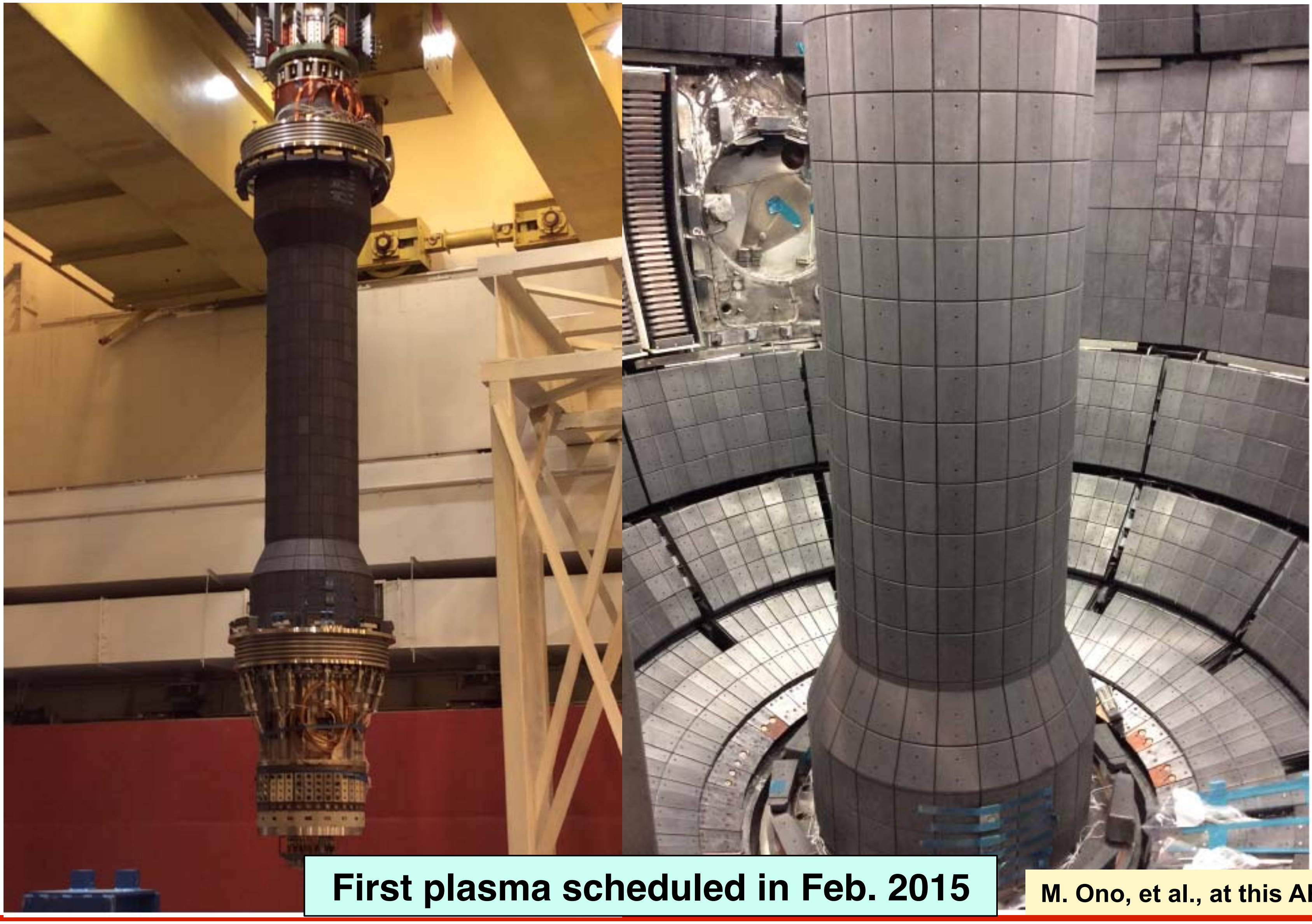
Research operation to resume in Apr. 2015

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

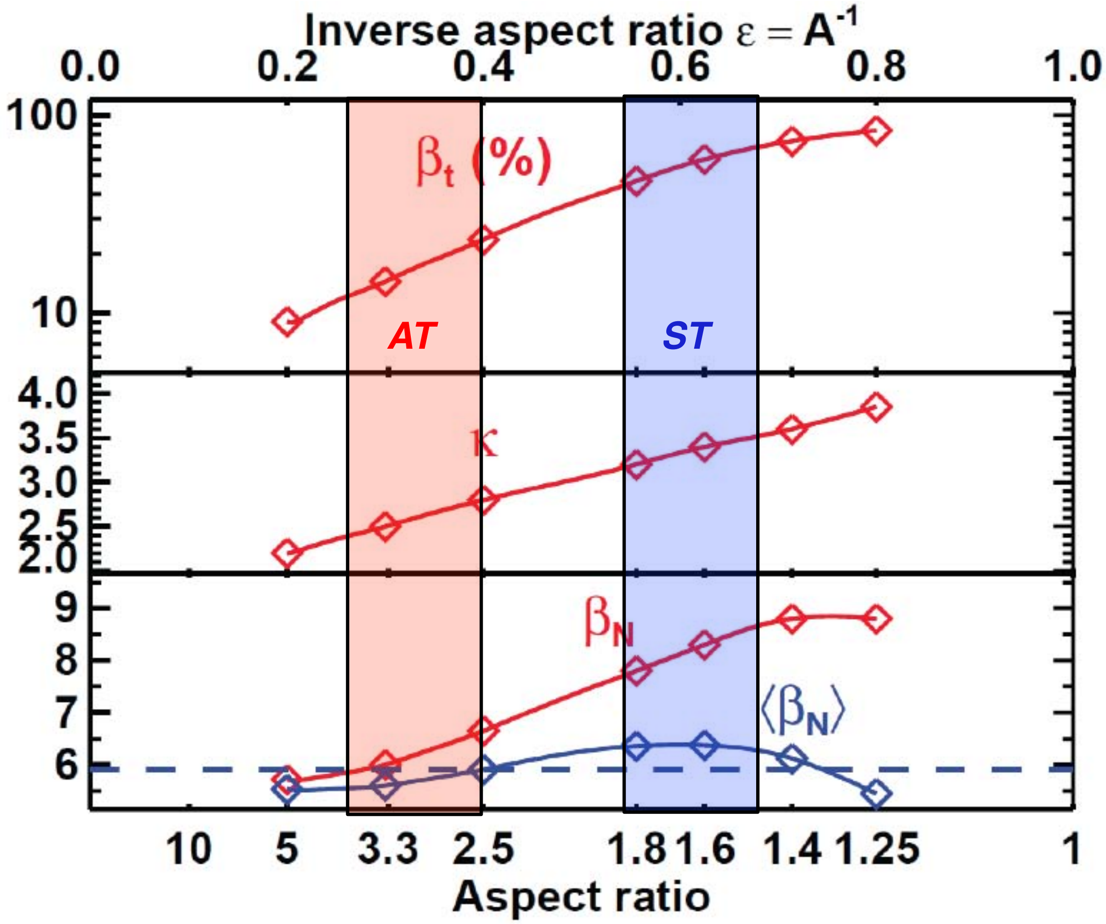
Poster Session PP8 (Wed): NSTX-U, LTX, PEGASUS, MAST-U, QUEST, TS-4

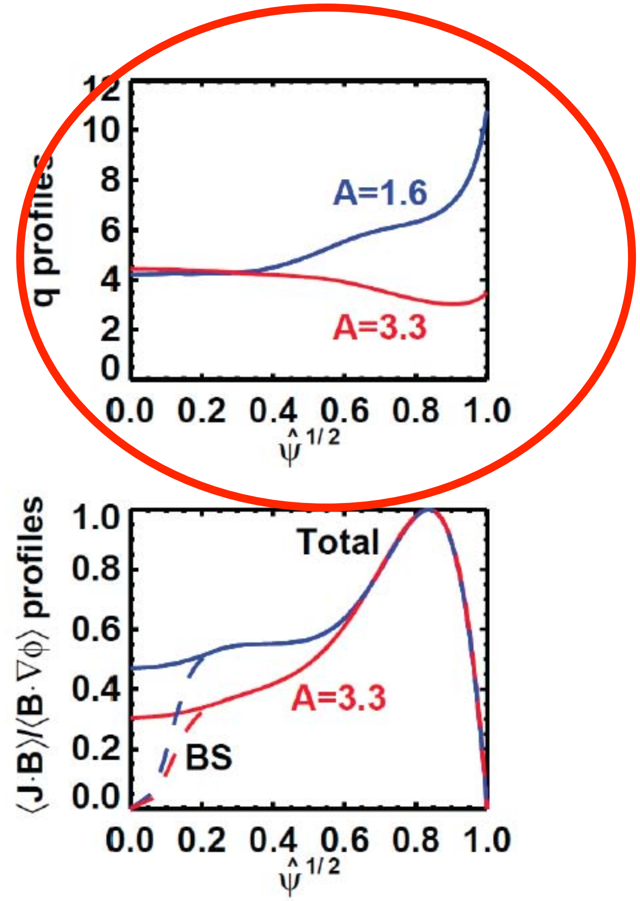
Back-up slides

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

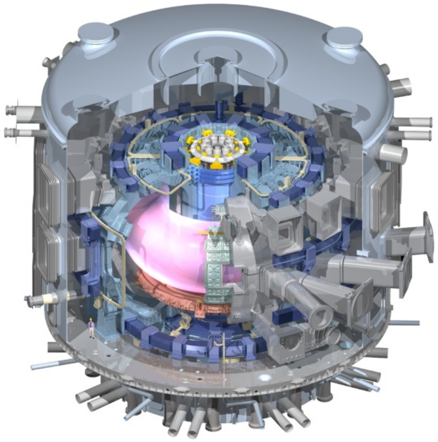
STs Narrow Gaps to Pilot/DEMO:

Goal: 100% non-inductive + high β

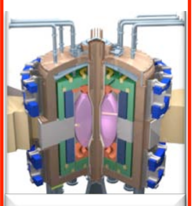
Plasma-Material Interface Research
- Strong heating + smaller R → high P/R, P/S
- Novel solutions: snowflake, liquid metals, Super-X, hot high-Z walls

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

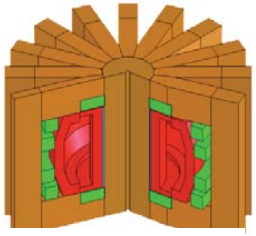
Burning Plasma Physics - ITER



Fusion Nuclear Science Facility

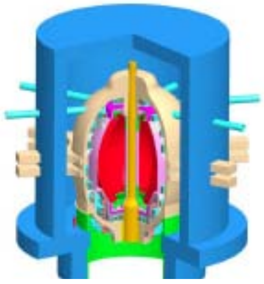


FNSF-ST

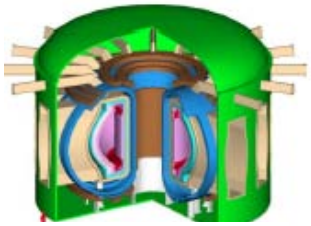


FNSF-AT

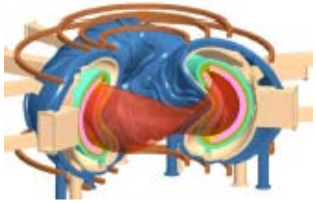
Pilot Plant or DEMO



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

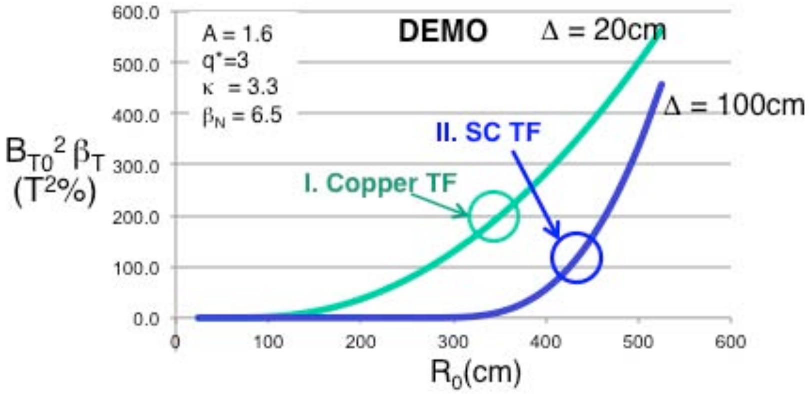
ST Fusion Power Plants

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

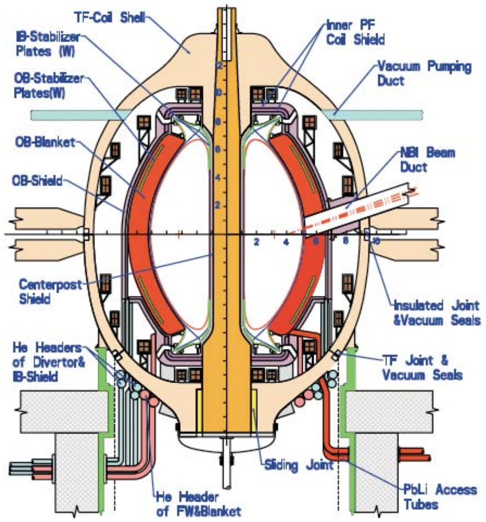


Copper design – Compact but due to larger recirculating power leading to higher fusion power needing aggressive β_T, β_N, κ. designs..

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.

ARIES-ST

Cu Power Plant

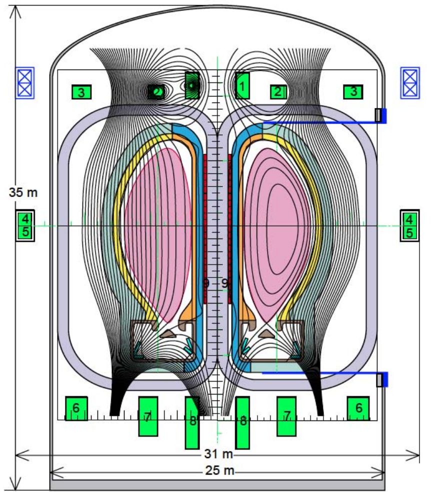


$R_0 \sim 3.2$ m

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

JUST

SC ST Power Plant



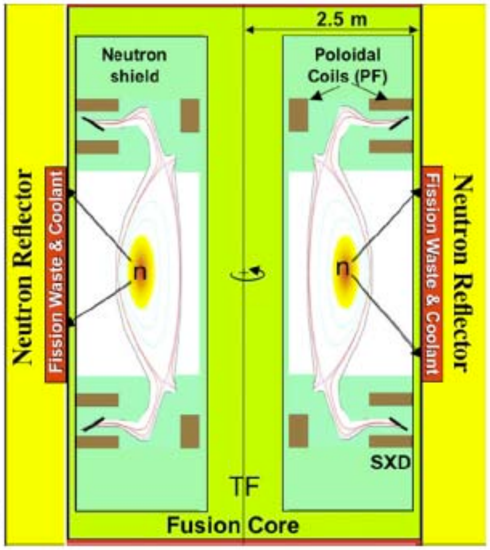
$R_0 \sim 4.5$m

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

Non-conventional ST Fusion Power Reactors

Taking advantage of compact and light weight ST fusion core

ST Fusion-Fission Hybrid



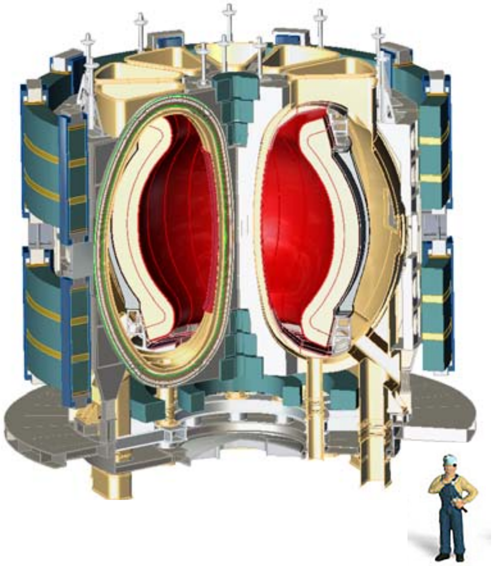
ST-FNSF-like Q ~ 1 facility producing net energy by “burning” highly toxic long-live nuclear waste

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

ST135

R = 1.35m

Q_{fus} ~ 5 Pilot Plants

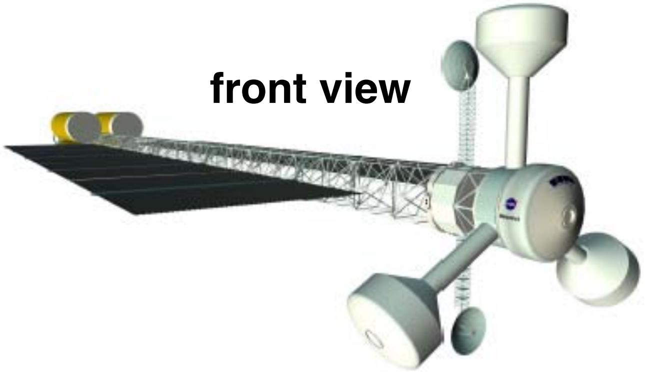


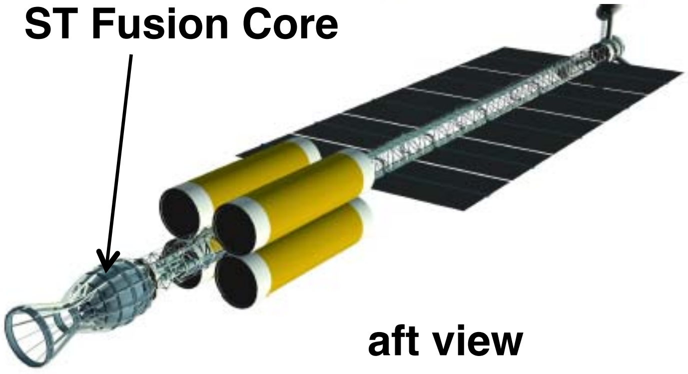
With HTS magnet operating at 3.7T/7 MA
P_{fusion} = 185 MW

A. Sykes, SOFT 2014

The Discovery II

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

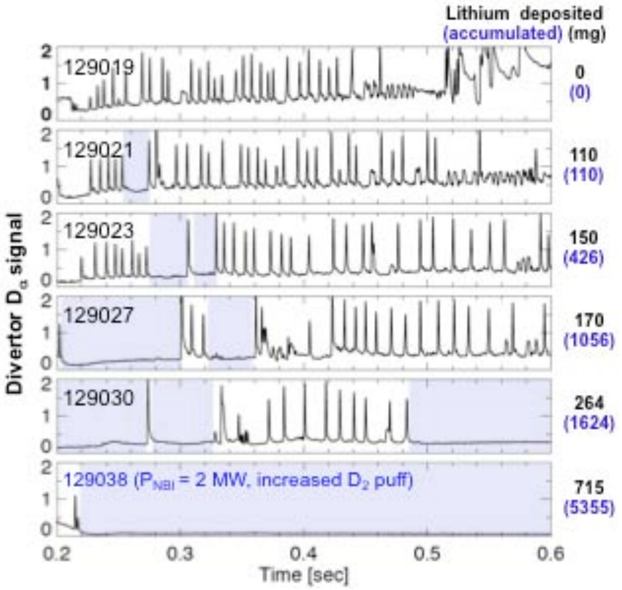




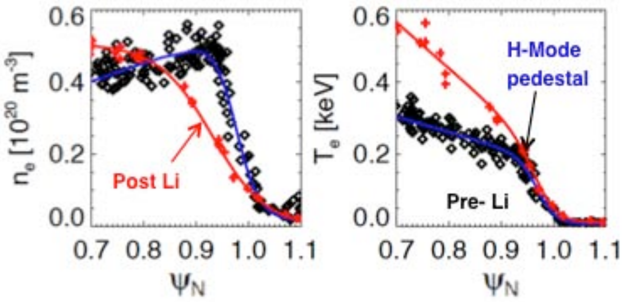
C.H. Williams, et al., NASA/TM-2005-213559

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

ELMs stabilized with edge pressure modification with Li in NSTX

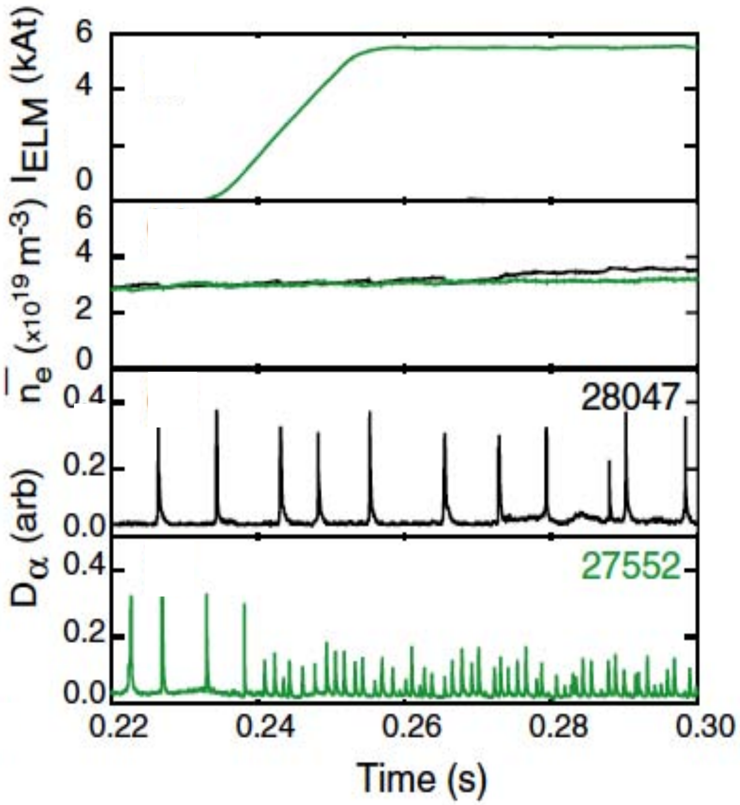


D.K. Mansfield, et al., JNM (2009)



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)