NSTX-U

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Scientific Opportunities and Challenges in the Upgraded National Spherical Torus Experiment

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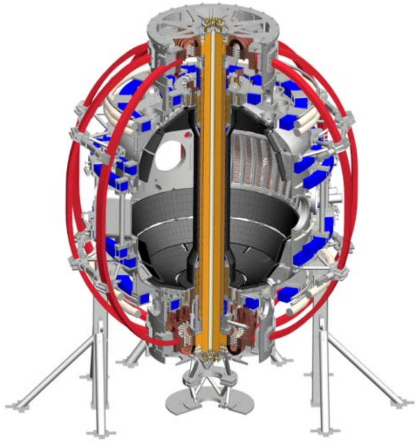
Jonathan Menard, PPPL
For the NSTX-U Team

PPPL
March 14, 2015



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Outline

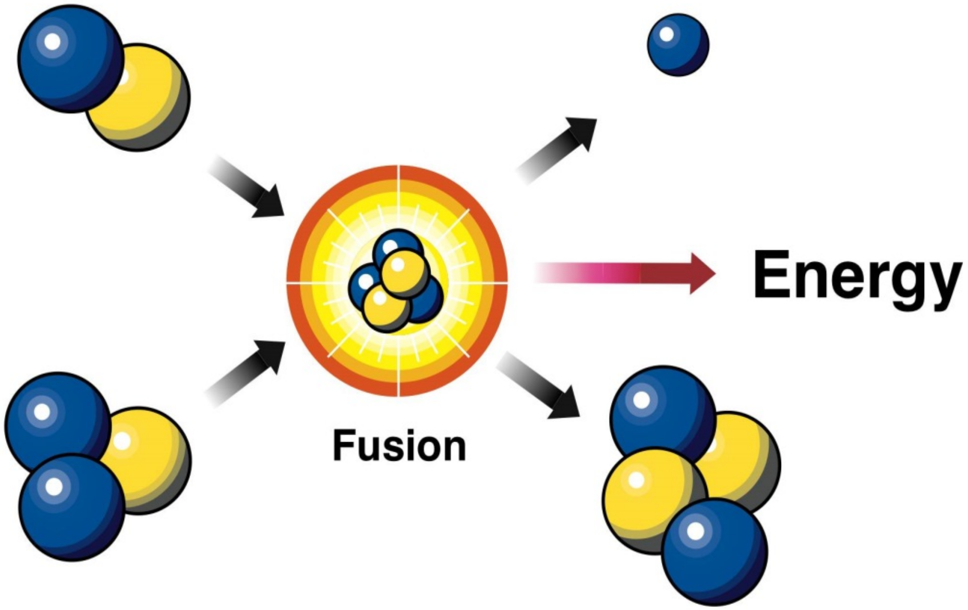
- Overview of fusion and plasma
- Plasma confinement methods
- What are tokamaks and stellarators?
- ITER burning plasmas
- Motivation for studying spherical tokamaks
- NSTX Upgrade scientific goals and questions
- NSTX Upgrade construction

What is fusion?

“D-T” fusion reaction:

D: deuterium

n: neutron



T: tritium

α: helium

High energy gain
≈ 1000 ×

$$E = mc^2$$

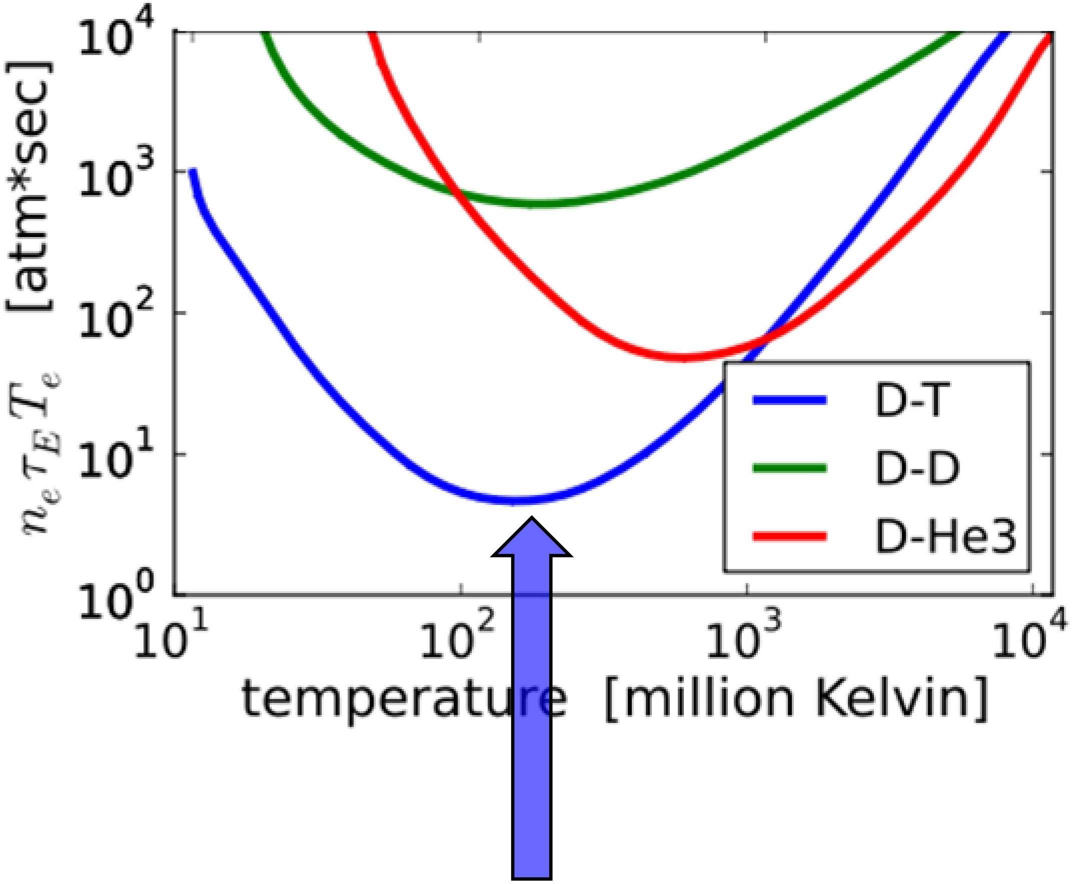
Advantages of fusion: safe, sustainable, high energy density, environmentally attractive

- **Cannot have runaway reaction**
  - –Only small amount of fuel present
  - –If particles cool, fusion stops
- **Abundant fuel supply**
  - –D from seawater: HDO, D/H = 1/6400
  - –T bred from lithium in earth's crust
- **High energy density**
  - –1 liter water = 500 liters gasoline
- **Waste short-lived, low-level**
- **No CO_2 production**

Fusion requires very high temperatures

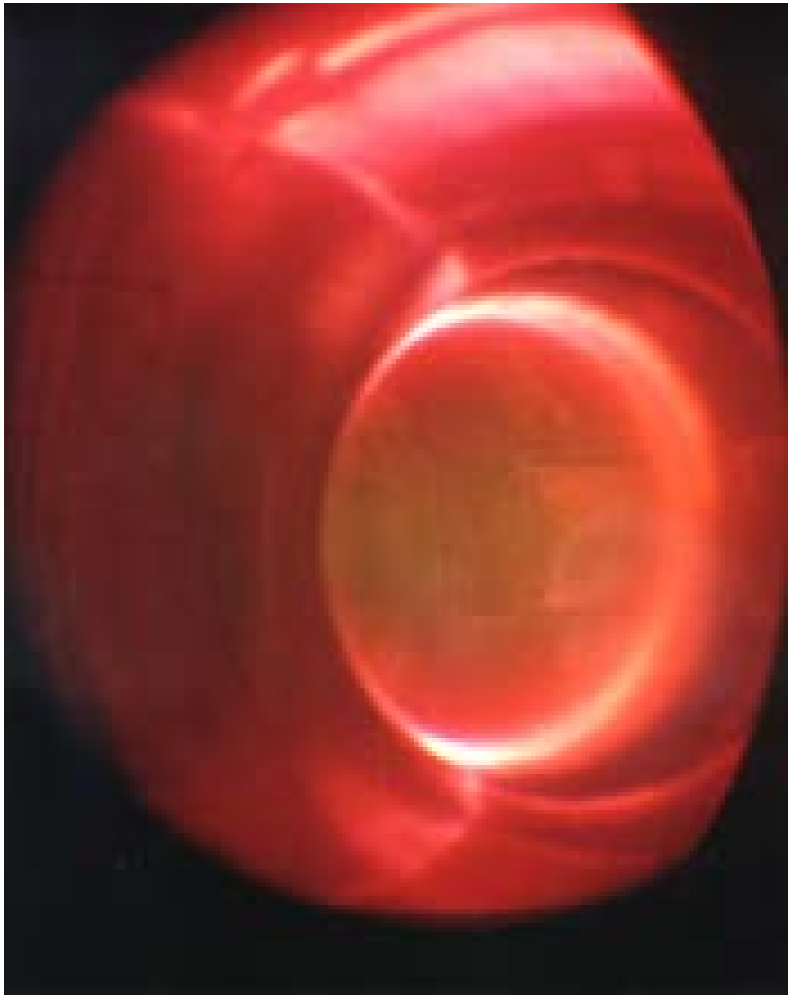
Fusion difficulty

(pressure × confinement)

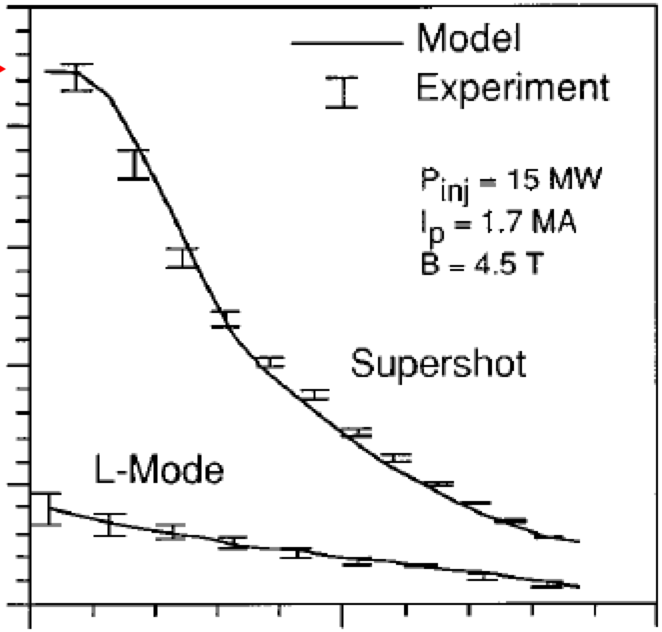


- Fusion is easiest here at **200 million °C (!!)** (350 million °F)
  - Requires lowest pressure *nT* and energy confinement time τ_E
  - Minimum fusion “triple-product” value: 8 atmosphere-seconds

Magnetic fusion has already achieved the necessary very high temperatures!



TFTR at PPPL (1990's)

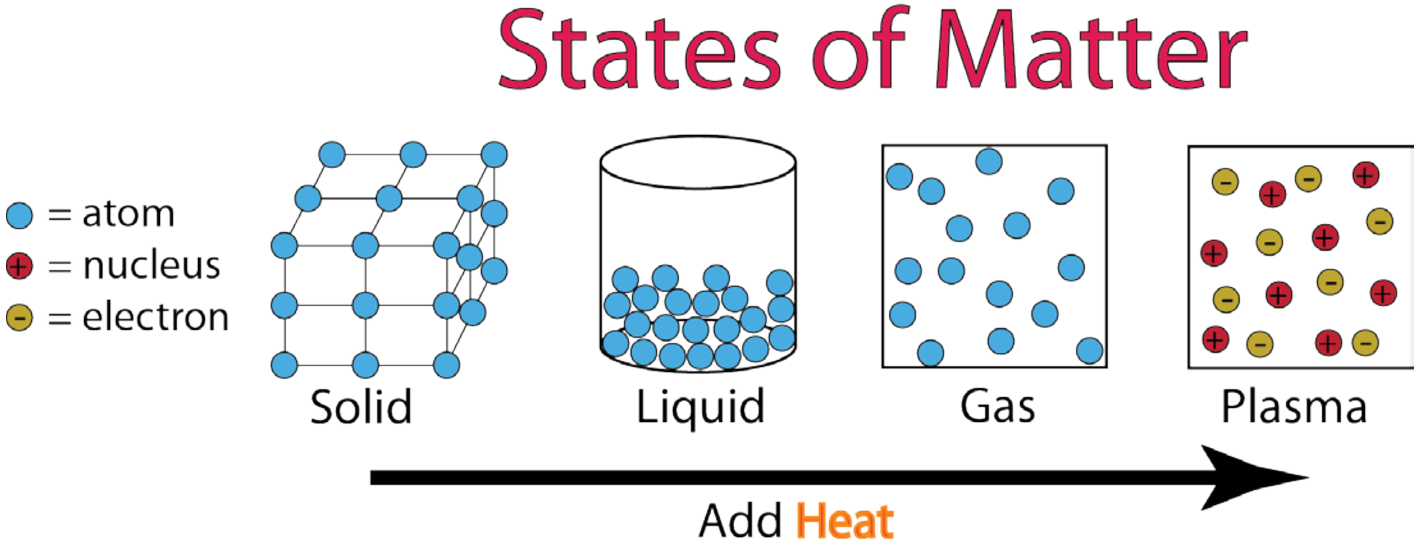


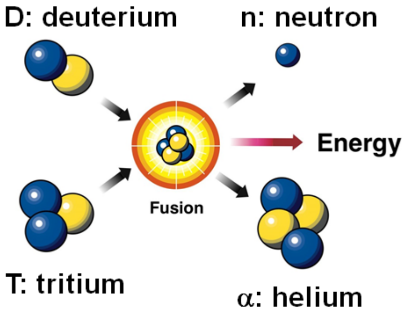
~250 million C →

core

boundary

Gas becomes plasma at fusion temperatures





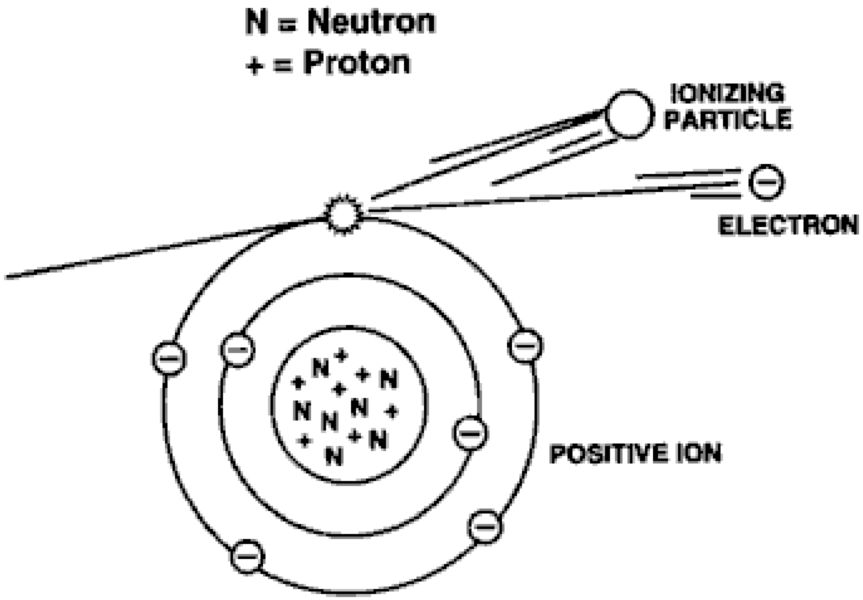
DT fusion:

- **Water: 0 °C 100 °C 160,000 °C 200,000,000 °C**

Plasma is a gas of charged particles:

"Soup" of negatively charged electrons, positive ions

- At fusion temperatures, particles are so energetic that negatively charged (-) electrons are stripped from neutral atom leaving positively charged (+) ions



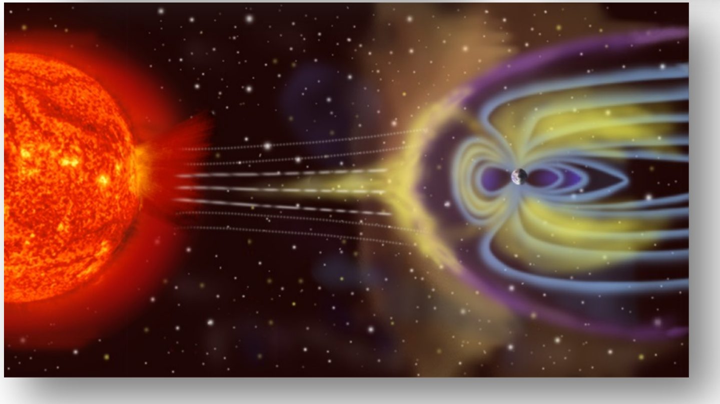
- **<u>One benefit of plasma state</u>: charged particle motion can be manipulated by electric and magnetic fields**

An estimated 99% of matter in the observable universe is in the plasma state

Plasma processing (semiconductors)

Neon signs

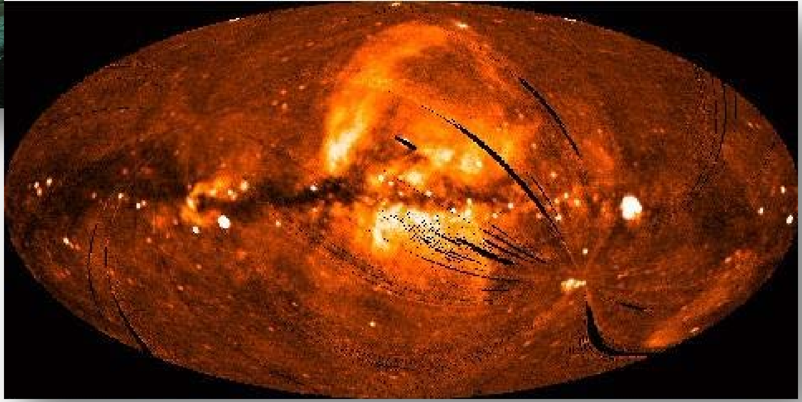
Lightening



Solar wind

Aurora Borealis

Interstellar medium

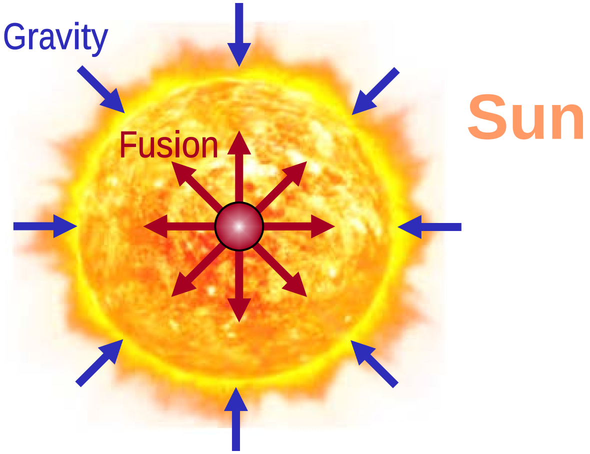


A few examples of plasma

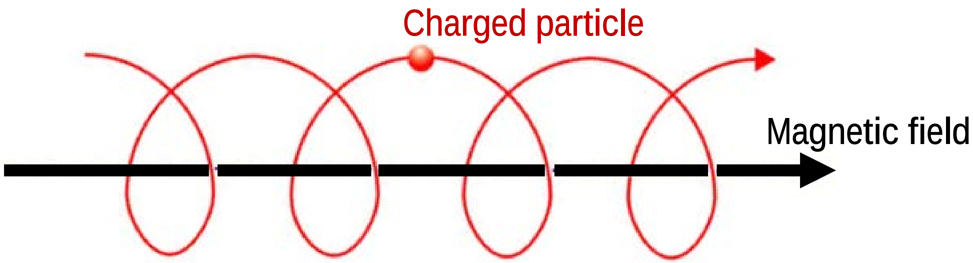
How do we confine plasma?

Confinement methods for controlled fusion

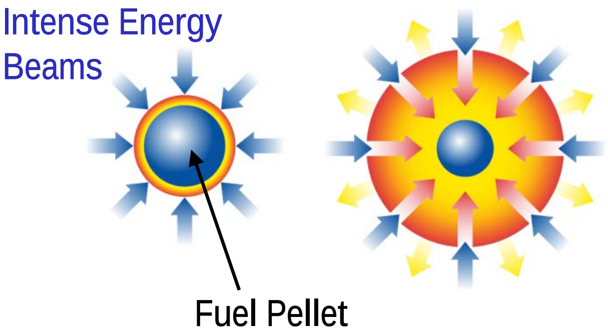
Gravitational Confinement



Magnetic Confinement

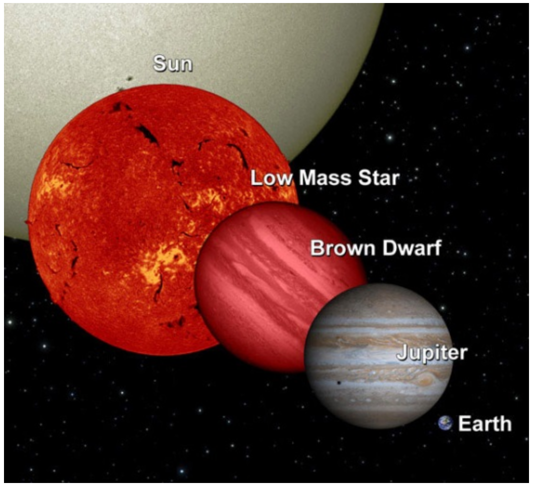


Inertial Confinement



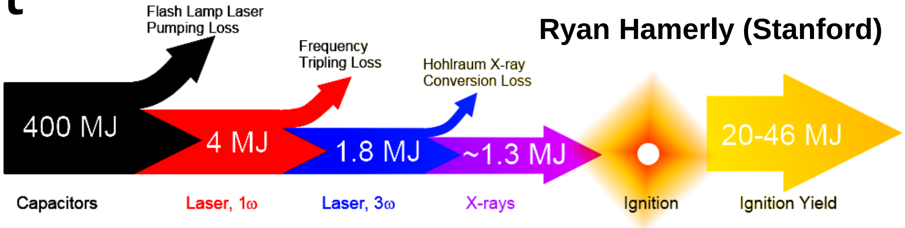
Magnetic fusion is arguably closest to ultimate goal of electricity generation

- Gravitational confinement fusion requires large device
  - Need 7-8% of mass of our sun
  - Approximately 10× diameter of Earth



- Laser fusion ala NIF at best has $E_{fusion} / E_{electrical}$ ~ 5%
  - So far, 0.004% efficient

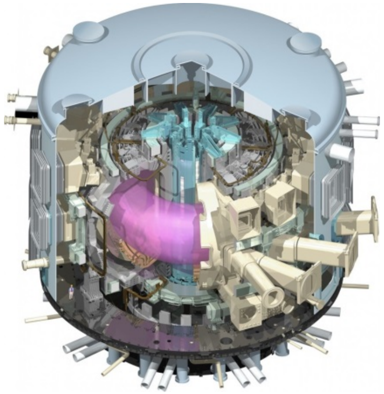
Ryan Hamerly (Stanford)



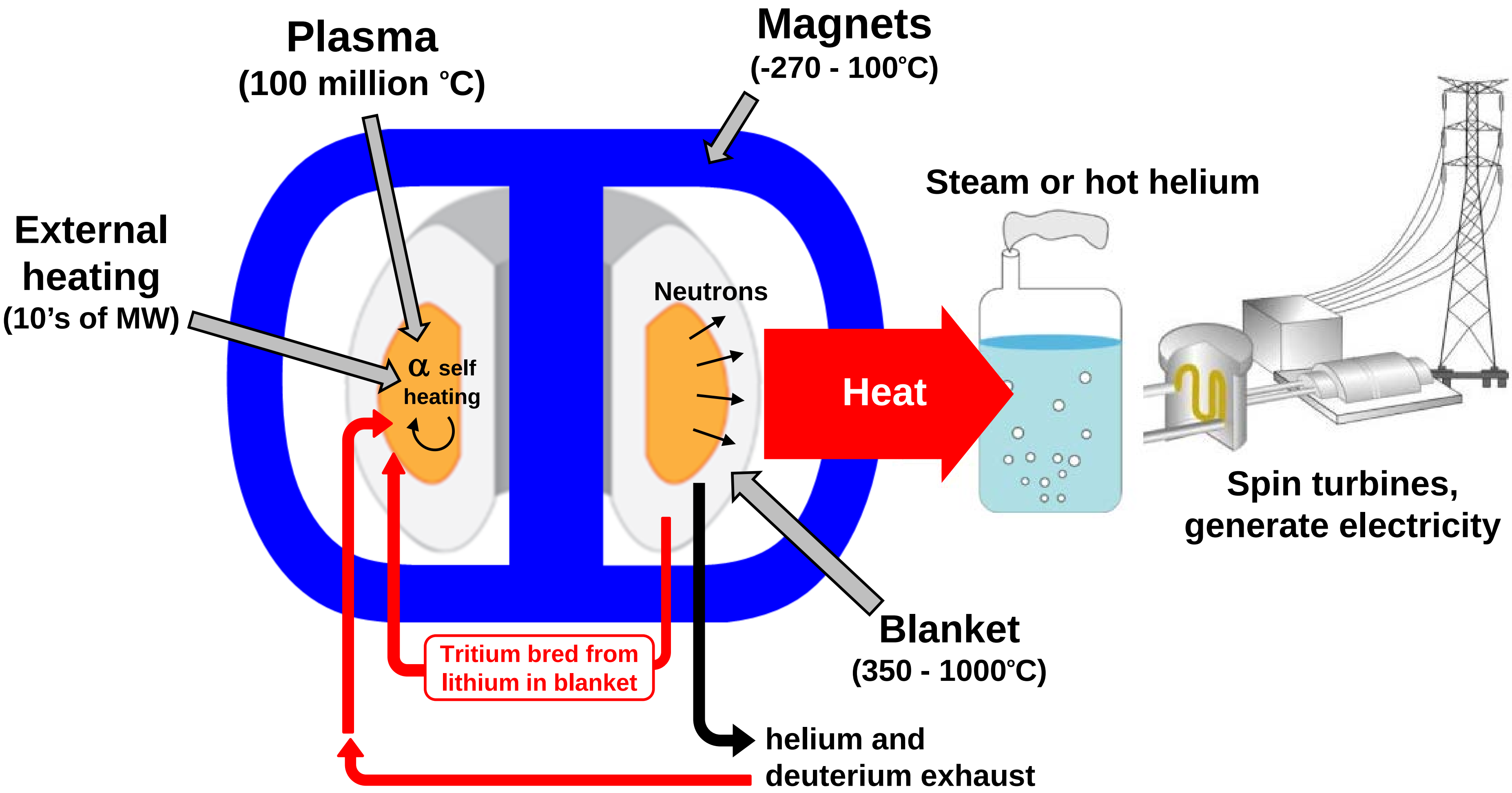
14kJ fusion yield achieved

- Magnetic fusion in ITER:
  - Goal: 500MW fusion power for ≤ 600MW electrical input for 400s

➢Industrial levels of fusion power



How would magnetic fusion make electricity?



Plasma
(100 million °C)

Magnets
(-270 - 100°C)

External heating
(10’s of MW)

α self heating

Neutrons

Heat

Steam or hot helium

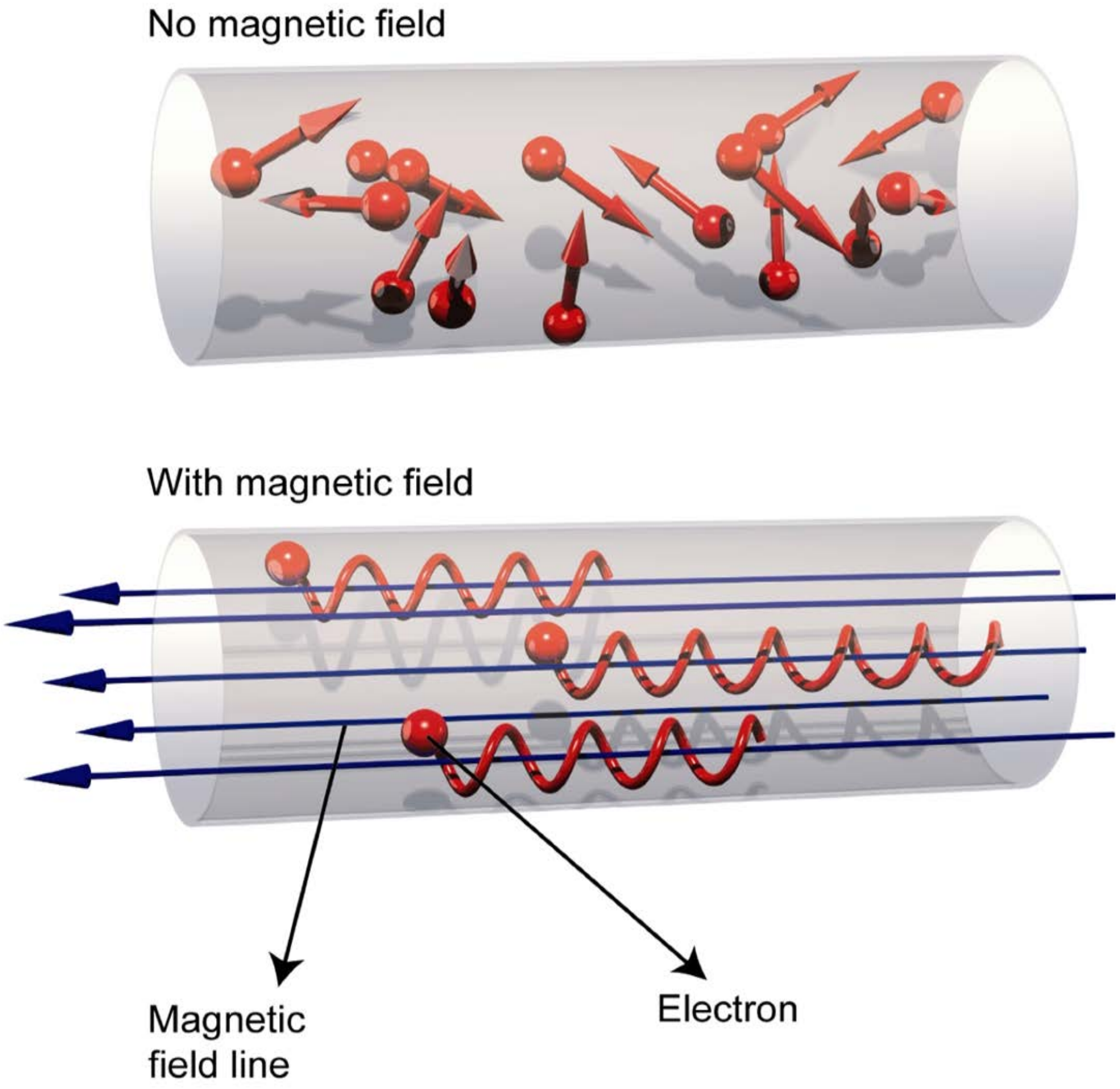
Spin turbines, generate electricity

Blanket
(350 - 1000°C)

Tritium bred from lithium in blanket

helium and deuterium exhaust

Charged particles confined by magnetic fields



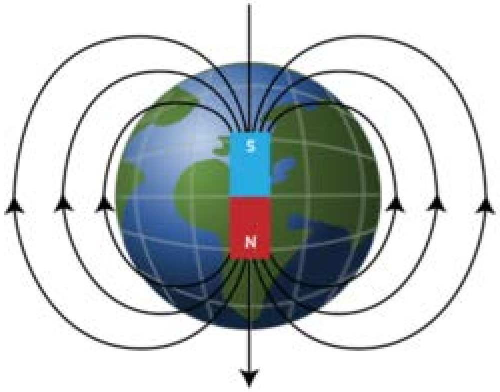
- No magnetic field: Charged particles move freely in all directions

- Charged particles spiral around magnetic field "arrows", but move freely along the field

- **Magnetic fusion goal**: make field so particles never touch walls

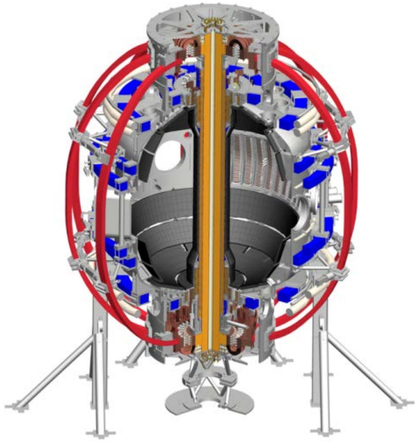
Example magnetic fields in units of Tesla [T]

Earth: ~5 × 10^{-5} T



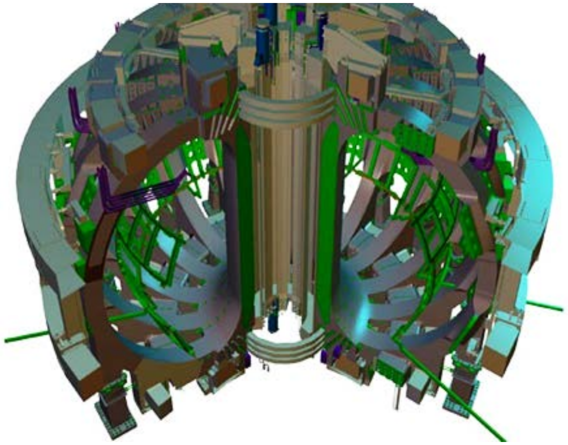
Refrigerator magnets
~1-5 × 10^{-3} T

PPPL's NSTX-U: 1 T



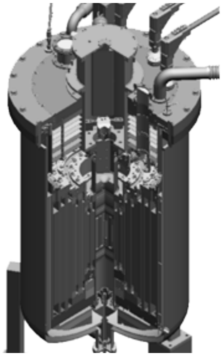
MRI: 0.5-3 T

ITER: 5.3 T



World Record: 100 T

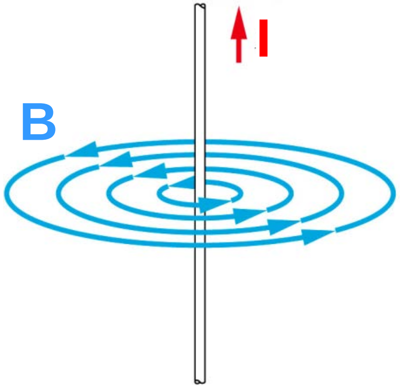
Non-destructive - for few milliseconds



National High Magnetic Field Laboratory

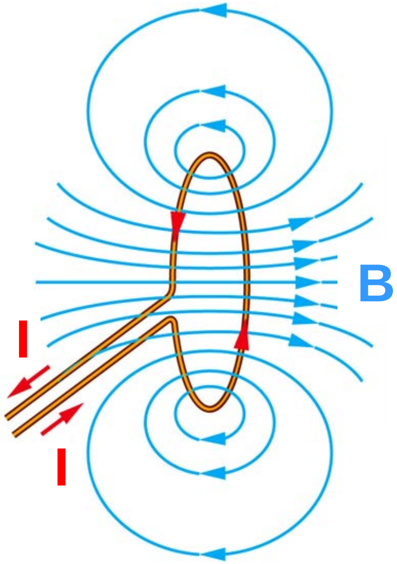
Magnetic fusion uses electromagnets

- Electromagnet: use electrical current "I" to create magnetic field "B"

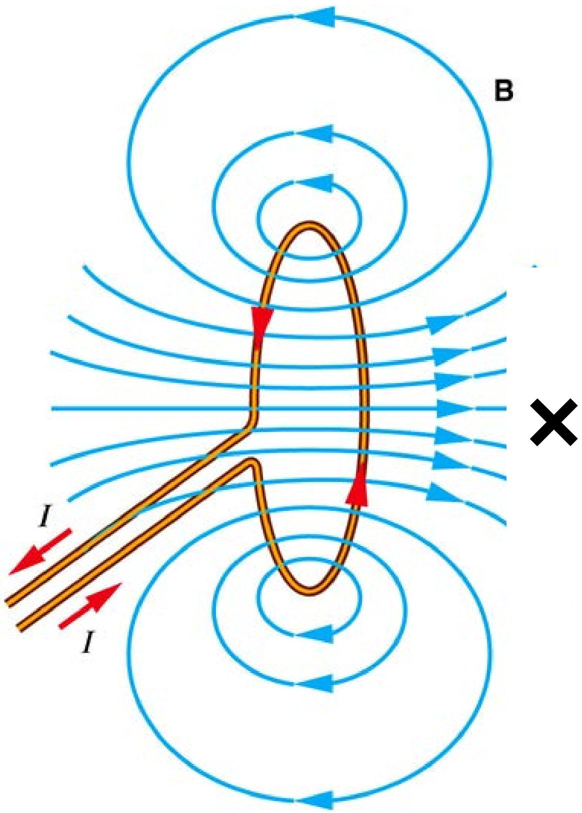


- "B" is <u>perpendicular</u> to "I"

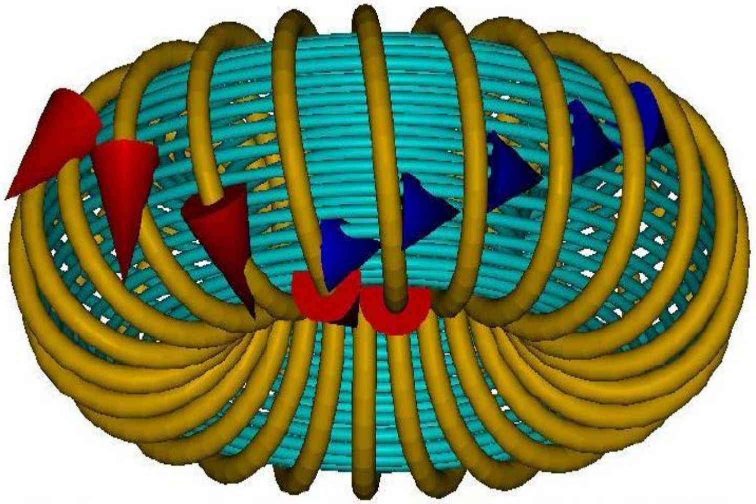
- Circular current "**coil**" makes strong nearly uniform B at center of coil
  - ➢ Want low/no resistance coil to reduce power: copper and/or superconductors



Fusion: Use multiple coils to create “toroidal” field

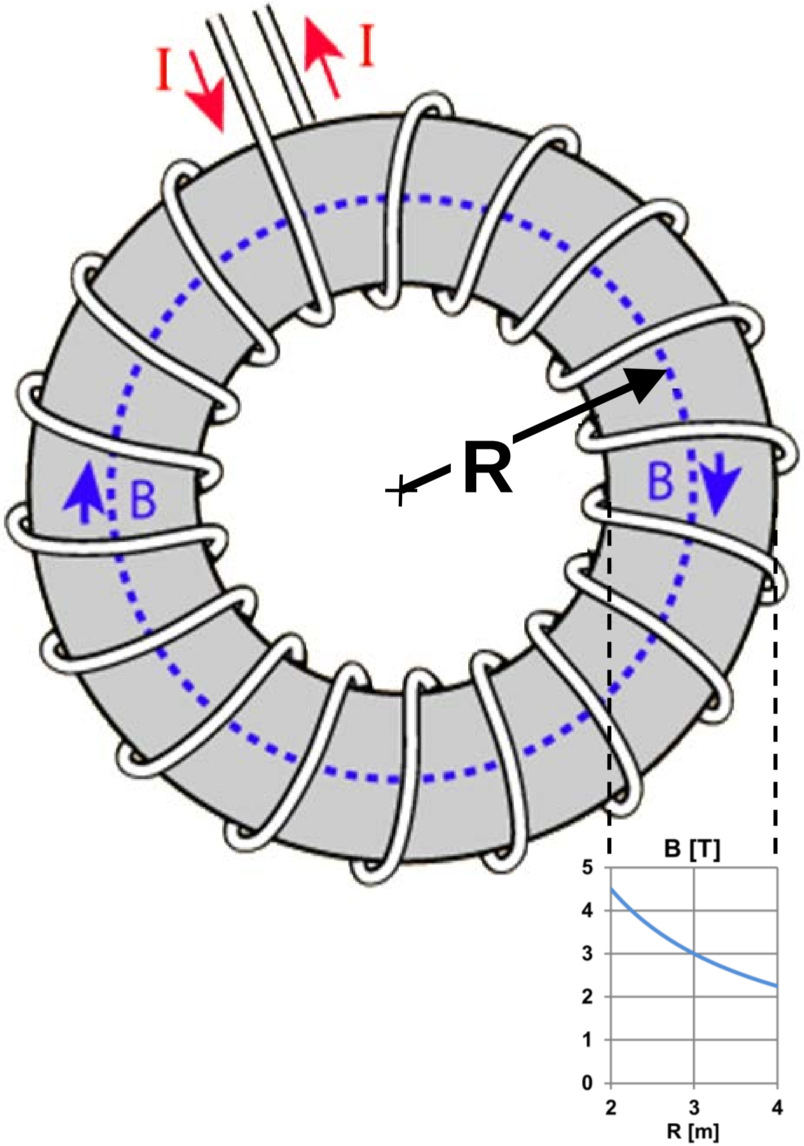


× 10-20



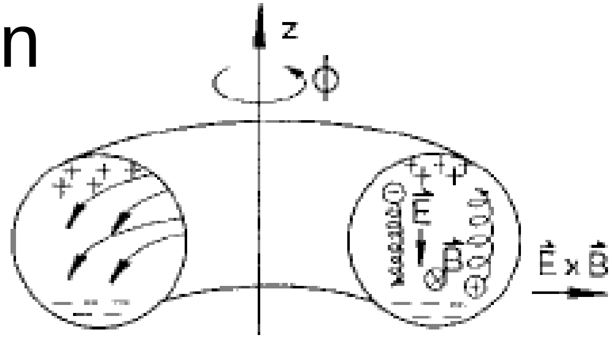
- Position many coils into donut or “torus” shape
- **Magnetic field never touches wall (!)**
- Do charged particles ever touch wall?

Simple toroidal field → poor confinement



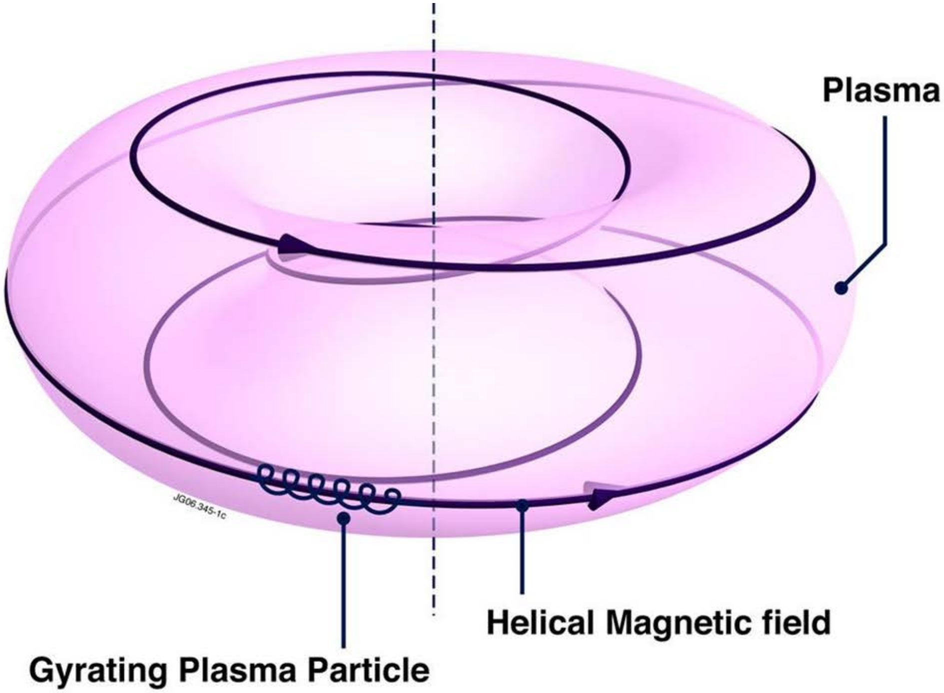
- Bending field into torus creates non-uniform B
  - B ~ 1 / R → factor of 2

- Electrons and ions drift apart vertically → charge separation



- Electric field created → particles expelled outward toward vessel wall

Helical magnetic field improves confinement



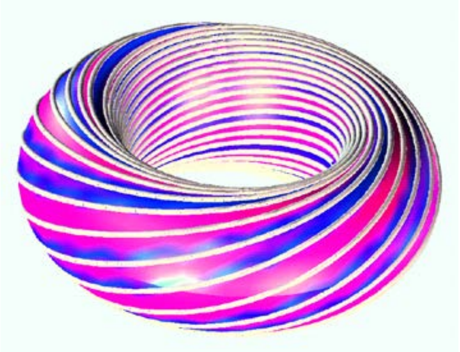
- "Helical" field = B-field covers toroidal surfaces
  - ➢ **B arrows never puncture donut**

- Allows currents to flow along magnetic field

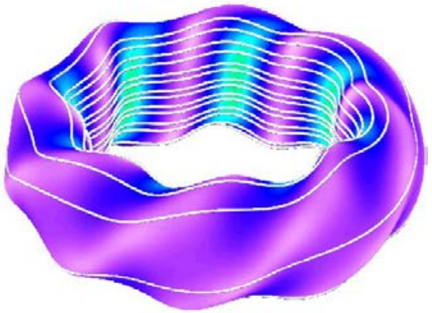
- Short-circuits electric fields that would otherwise expel plasma

➢ Particles tied to surface

➢ Improved confinement

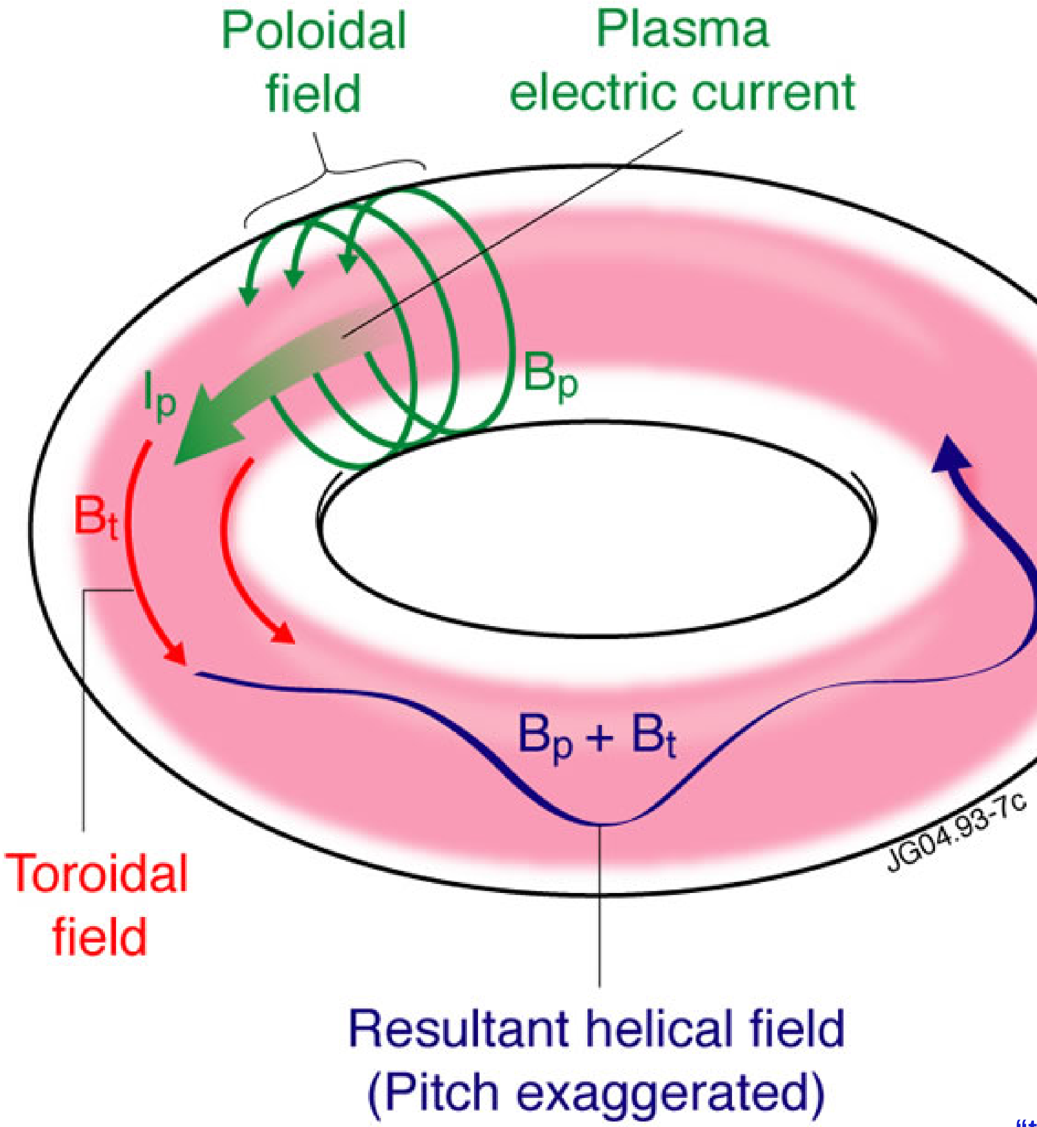


"Tokamak"



"Stellarator"

Tokamak: Poloidal field from plasma current + toroidal field from magnets create helical field

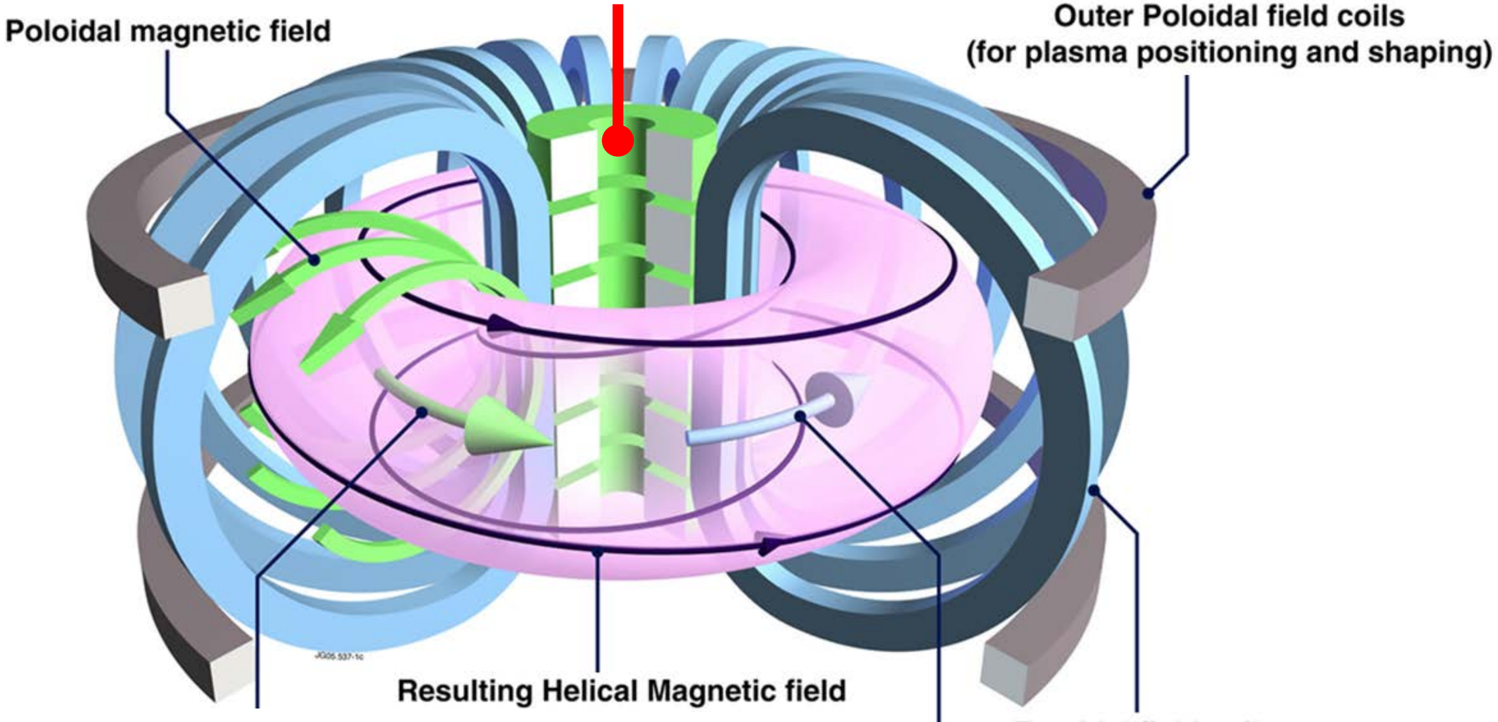


"Tokamak": Russian acronym for "toroidal chamber with magnetic coils"

Conventional tokamak is not steady-state

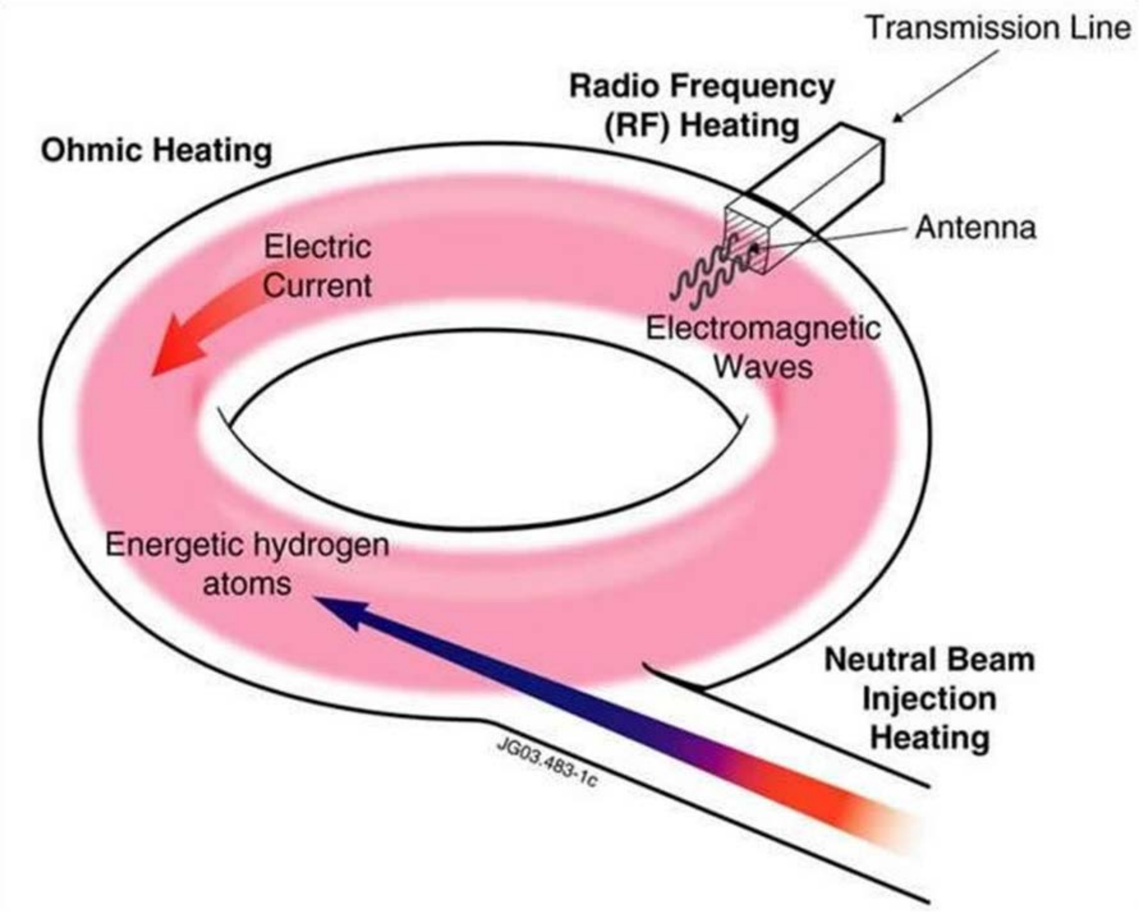
- Primary transformer coils drive toroidal electric current in plasma
- Must stop when transformer coils reach current and/or force limit

Inner Poloidal Field Coils
(Primary transformer circuit)



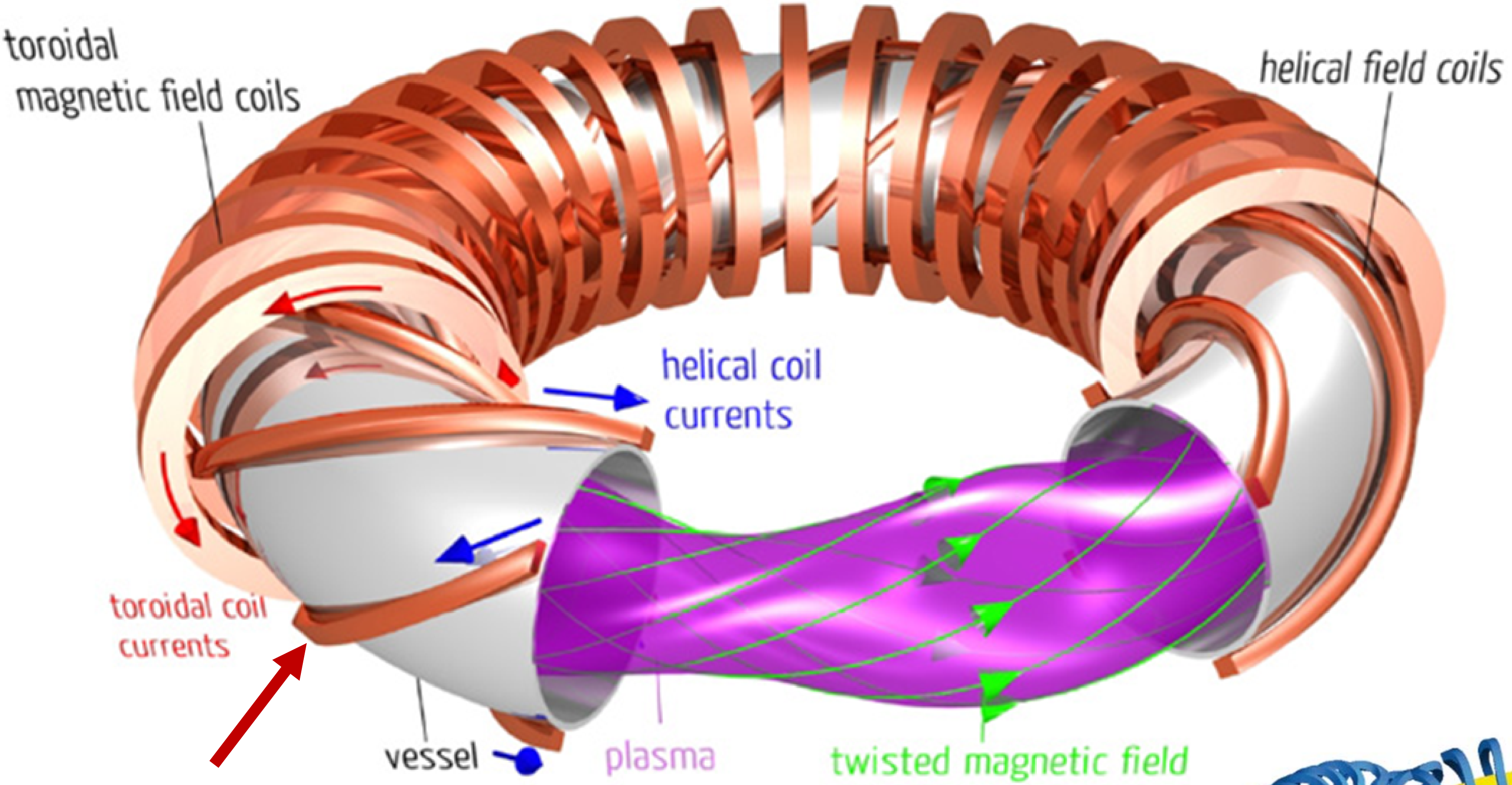
Plasma electric current
(Secondary transformer circuit)

"Advanced Tokamak" uses external + self-generated current to operate without transformer



- External: wave and beam current drive
- Self-generated: high pressure plasmas make "bootstrap" current
- Efficient tokamak power plant will need 70-90% self-generated current

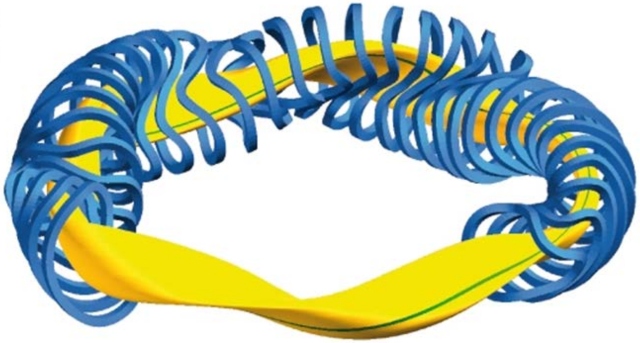
Stellarator: Toroidal / helical currents in coils + toroidal field from magnets create helical field



Helical coils

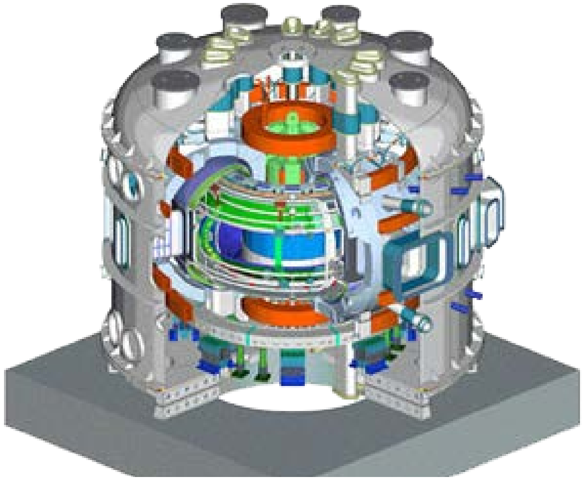
or

Discrete / modular coils



Tokamaks and stellarators are the leading configurations in magnetic fusion

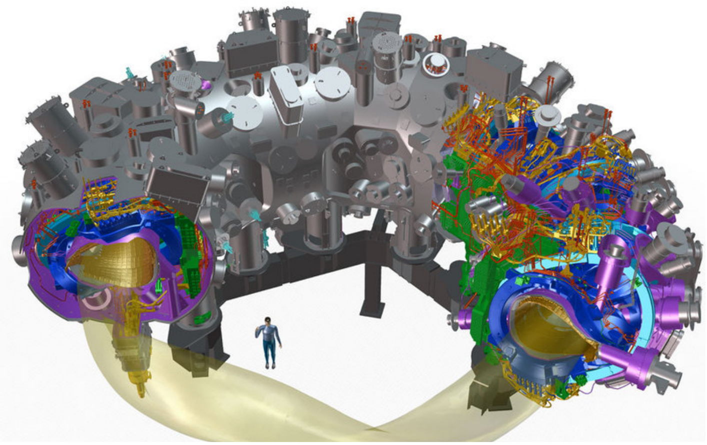
Superconducting tokamak



KSTAR (South Korea)

- Tokamak advantages:
  - Best confinement, closest to “breakeven”
  - Simpler planar coils and power/particle exhaust
- Disadvantages:
  - Must drive multi-mega-ampere plasma current
  - More prone to rapid loss of plasma = “disruption”

Superconducting stellarator



W7-X (Germany) – 1st plasma in late 2015

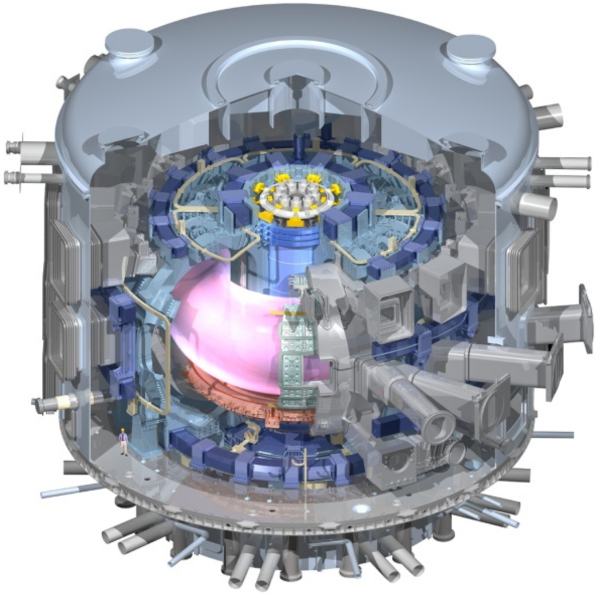
- Stellarator advantages:
  - No plasma current drive necessary
  - More stable, steady-state
- Disadvantages:
  - More complex coils and exhaust
  - Confinement < tokamaks (so far…)

ITER will be first device to access “burning plasma”

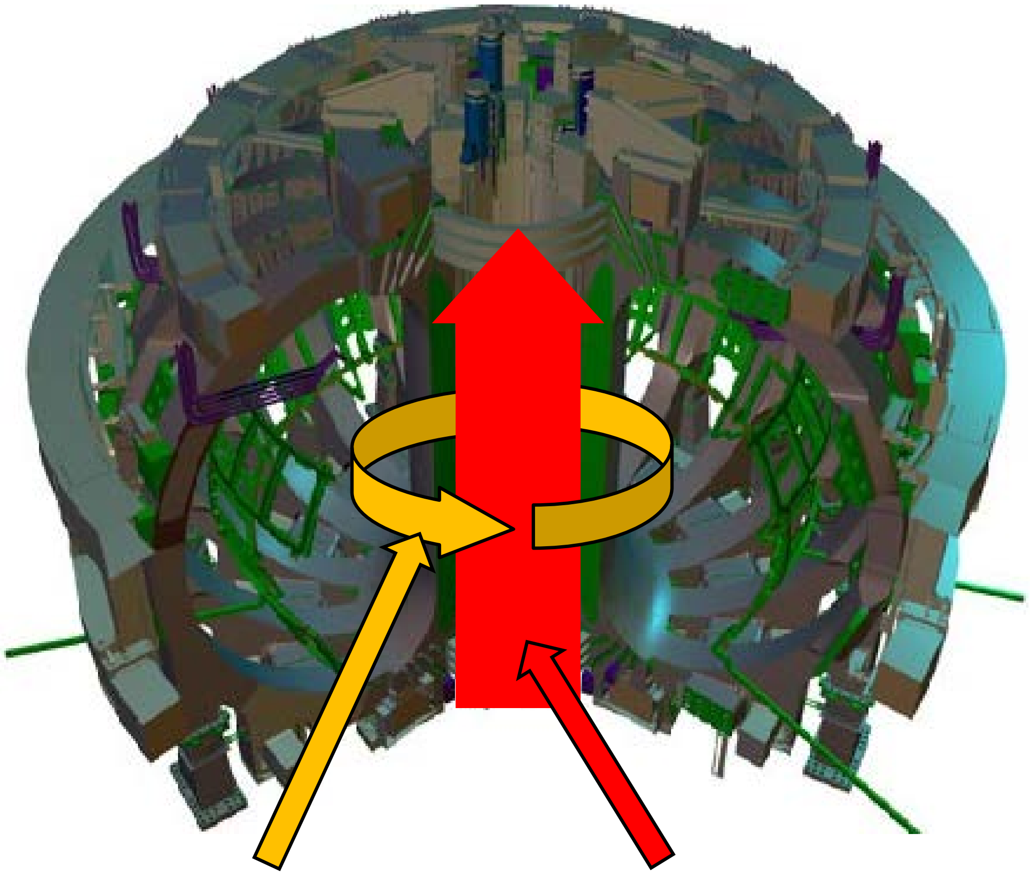
- Burning plasma: majority of plasma heating power comes from fusion alpha particles from DT reactions
  - DT reaction energy split: 1/5 in alphas, 4/5 in neutrons
- ITER goal $Q = P_{\text{fusion}} / P_{\text{external heating}} = 10$
- $Q = 10 \rightarrow P_{\text{alpha}} / P_{\text{external}} = 2$
- $P_{\text{alpha}} / P_{\text{alpha + external}} = 2 / 3 > 50\%$

ITER under construction in Cadarache, France

A=3.1, R=6.2m, B_T=5.3T, I_P=15MA



ITER magnets will be largest ever built



- 18 toroidal field magnets
- 12 Tesla at coil
- Weight: 6500 tons
- 80,000 km of Nb3Sn superconducting strand in total length

Plasma current:
15 million amps

Toroidal field current
165 million amps

15 amps
(1800W)

Size of ITER driven largely by plasma confinement

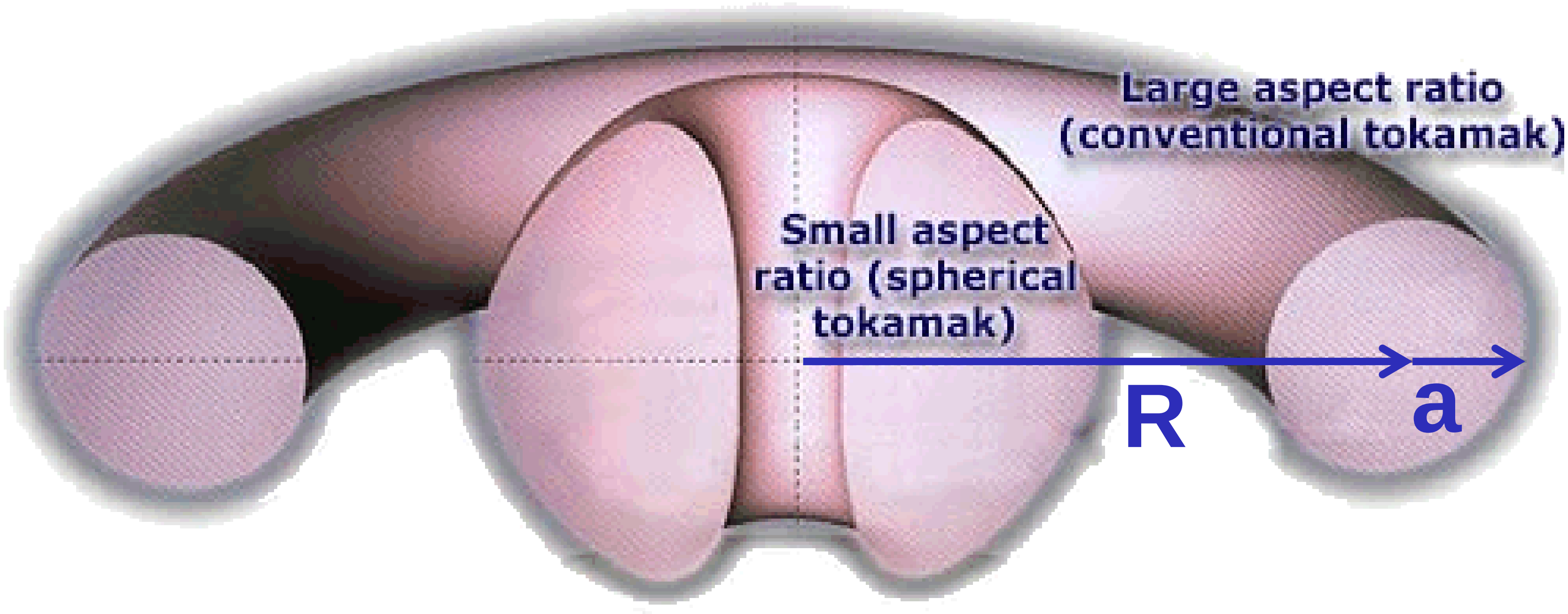
- Energy confinement scales with plasma current
- Large plasma current requires large toroidal field and/or plasma size for plasma to remain stable
- Current and confinement both scale with size
- **Can we make smaller devices with better confinement and smaller or cheaper magnets?**
- Such questions motivate exploring alternatives…

How might we possibly improve the conventional tokamak?

Aspect ratio is important free parameter

Aspect ratio A = R / a

R = major radius   a = minor radius

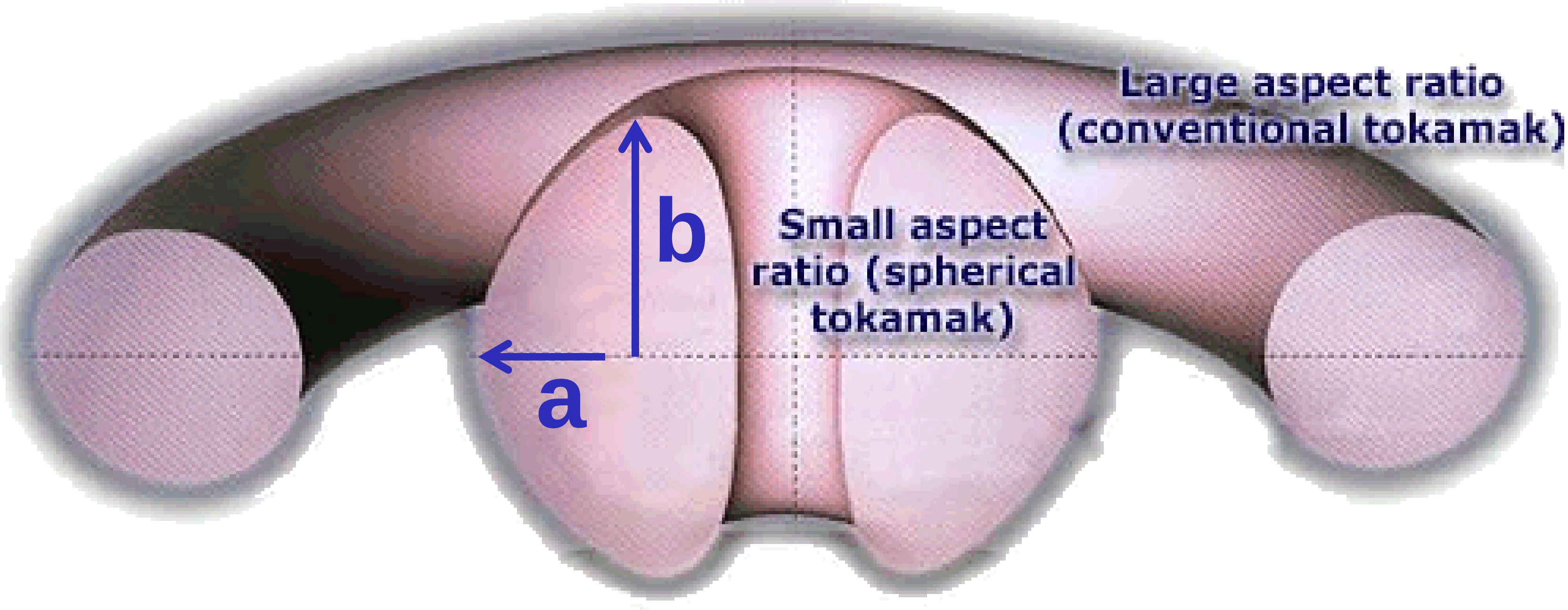


Spherical torus/tokamak (ST) has A = 1.1-2
Conventional tokamak typically A = 3-4

STs have higher natural elongation

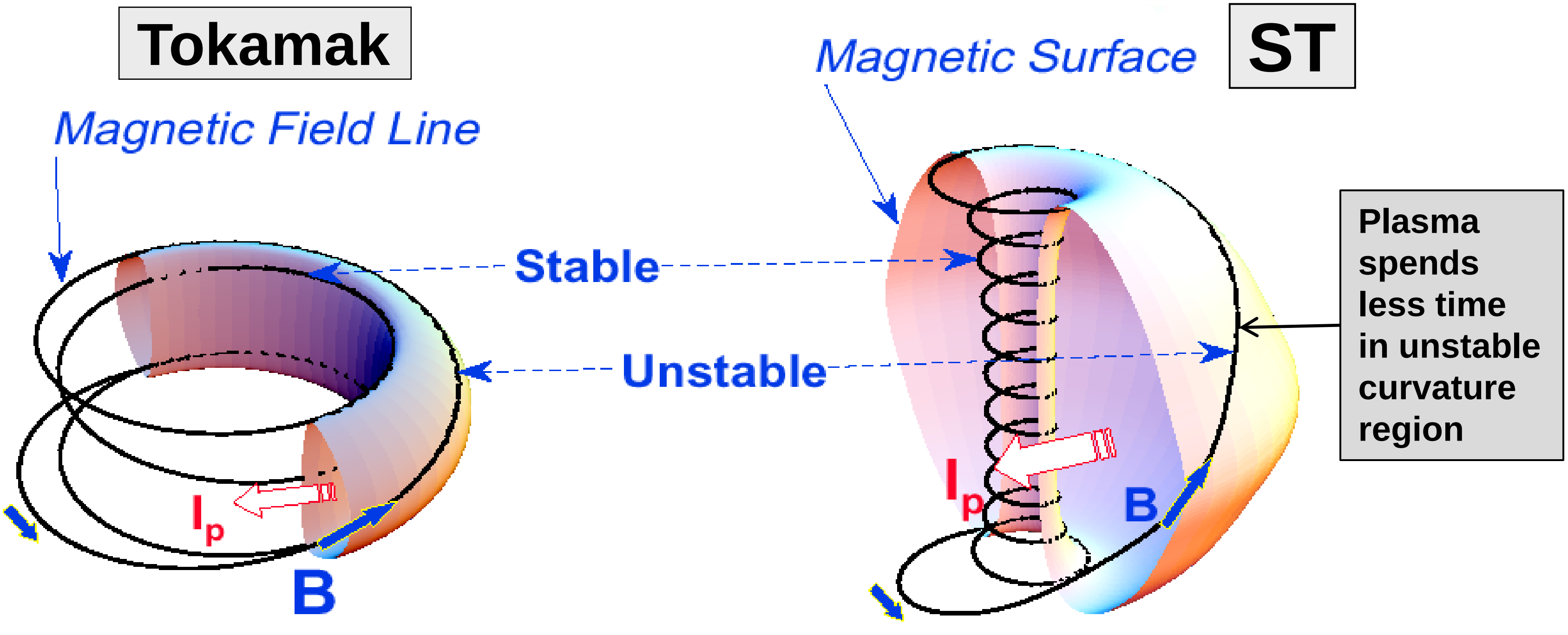
Elongation $\kappa = b / a$

b = vertical ½ height a = minor radius



Higher elongation improves stability, confinement

Favorable average curvature improves stability



Plasma spends less time in unstable curvature region

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

$A \sim 1.5$
$\kappa = 2\text{-}3$
$\beta_T = 10\text{-}40\%$

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

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 I_P due to high κ and low A

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

- I_P increases tokamak performance

  Energy confinement time $\tau_E \propto I_P$

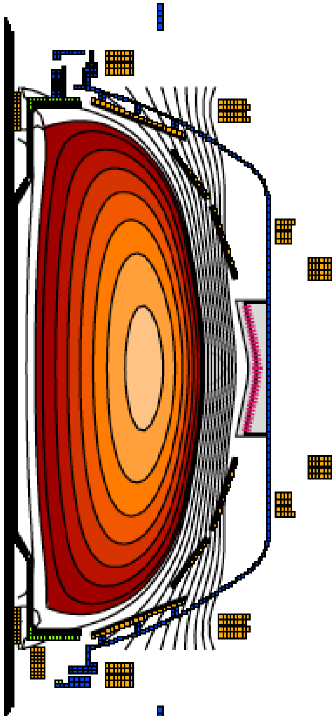
  Normalized pressure $\beta_T [\%] \equiv \beta_N I_P / (aB_{T0})$

- ST achieves high performance cost effectively

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

  $

High $\kappa \sim 3.0$ equilibrium in NSTX

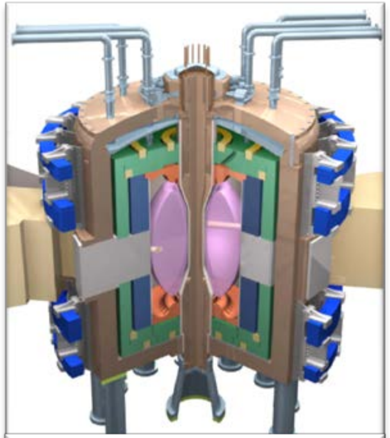


Fusion technology development is major challenge

Fusion Nuclear Science Facility (FNSF) could aid development

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

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



ST-FNSF

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

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

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

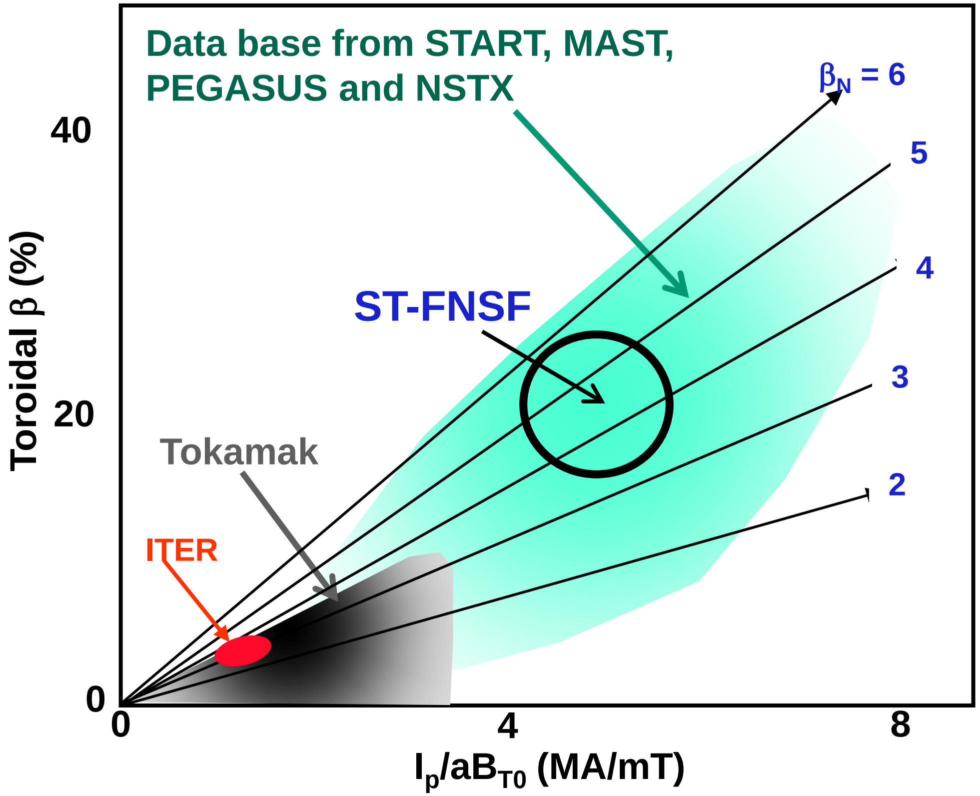
$P_{fusion} \propto \beta_T^2 \; B_{T0}^4 \times \text{Vol}$

Product

Physics Knob

$

Investment



High neutron wall loading W_n possible in a compact ST

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

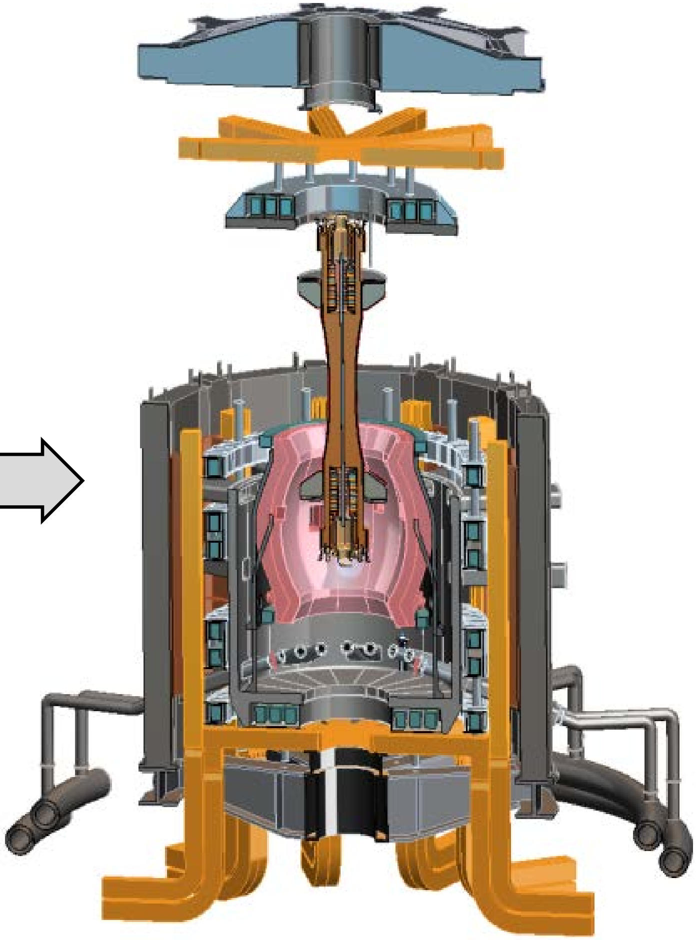
$W_n \propto \beta_T^2 B_{T0}^4 a$ (not strongly size dependent)

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

Design studies show ST potentially attractive as FNSF

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

PPPL ST-FNSF concept

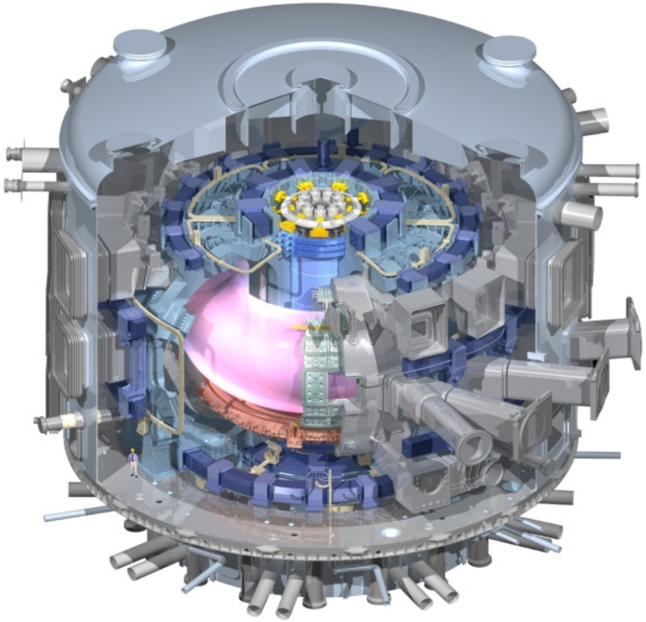


Unique ST properties also support ITER

ST Extends Predictive Capability for ITER and Toroidal Science

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

Burning Plasma Physics - ITER

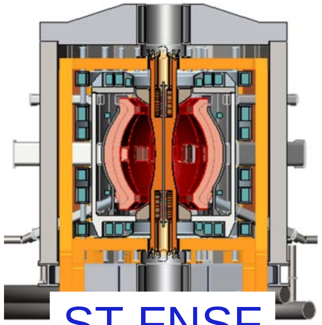


What are the goals of
NSTX Upgrade (NSTX-U)?

What key science questions
will NSTX-U address?

NSTX Upgrade mission elements

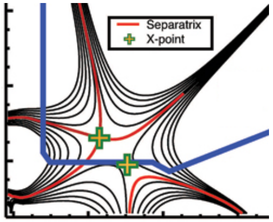
- Advance ST as candidate for Fusion Nuclear Science Facility (FNSF)



ST-FNSF

- Develop solutions for the plasma-material interface challenge

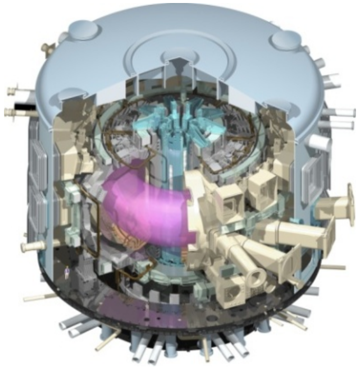
Liquid metals/Li



“Snowflake”

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

ITER

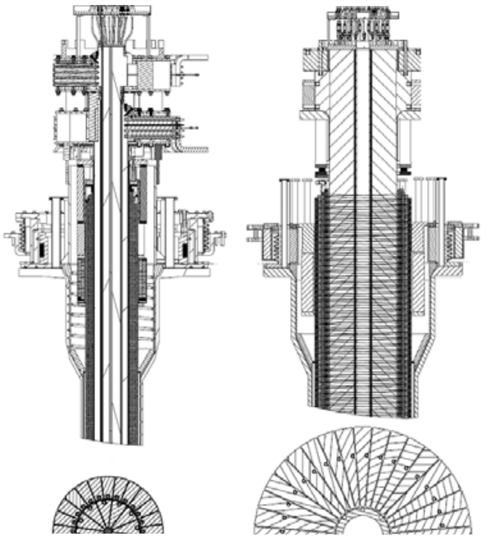


- Develop ST as fusion energy system

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

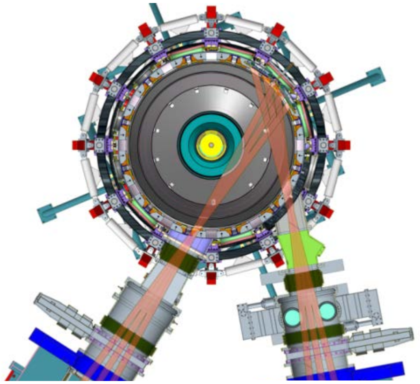
Previous center-stack

New center-stack



TF OD = 20cm

TF OD = 40cm



Present NBI

New 2nd NBI

- **New center-stack doubles toroidal field, plasma current, 5× pulse-length**
- **Tangential injection provides 3-4× higher current drive at low I_P:**
  - **2x higher absorption (40→80%) at low I_P**
  - **1.5-2x higher current drive efficiency**

~ 5-10x increase in nTτ from NSTX

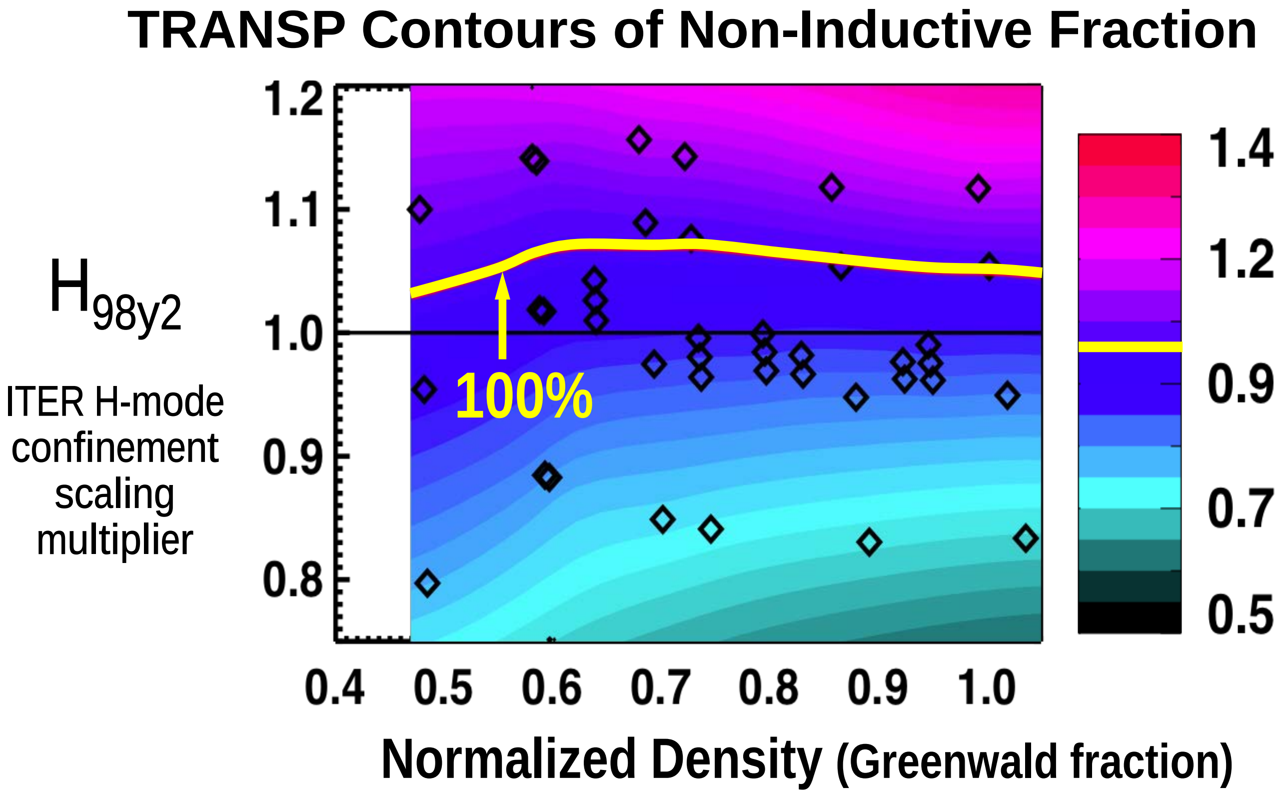
NSTX-U average plasma pressure ~ tokamaks

Key NSTX-U research topics for FNSF and ITER:

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

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

NSTX achieved 70% "transformer-less" current drive
Will NSTX-U achieve 100% as predicted by simulations?

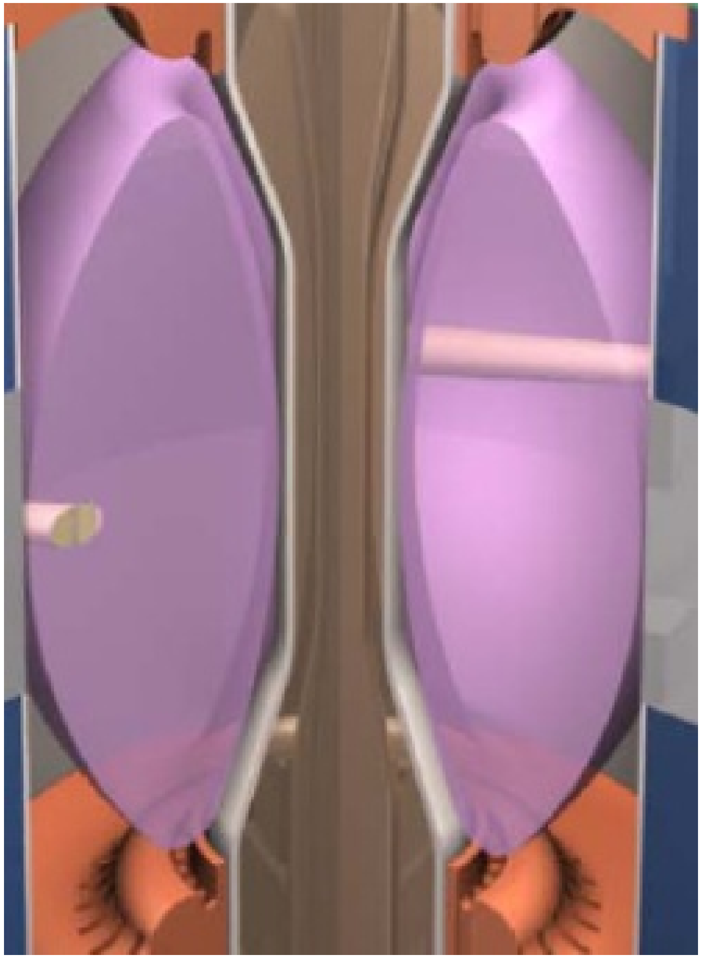


I_P=1 MA, B_T=1.0 T, P_{NBI}=12.6 MW

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

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

Compact ST-FNSF has no/small central solenoid

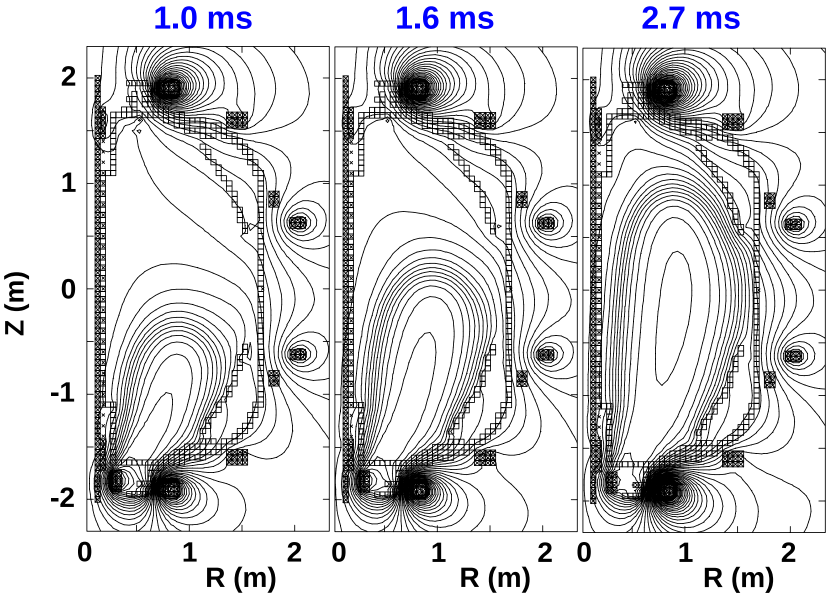


~ 1-2 MA of transformer-free start-up current needed for FNSF
→ 10-20% of total current

- **Long-term major goal of NSTX-U: generate and sustain a high-performance plasma without using any transformer** (this will not be easy…)

NSTX achieved 200kA (~20%) “transformer-less” start-up
Will NSTX-U achieve 400kA or more as per simulations?

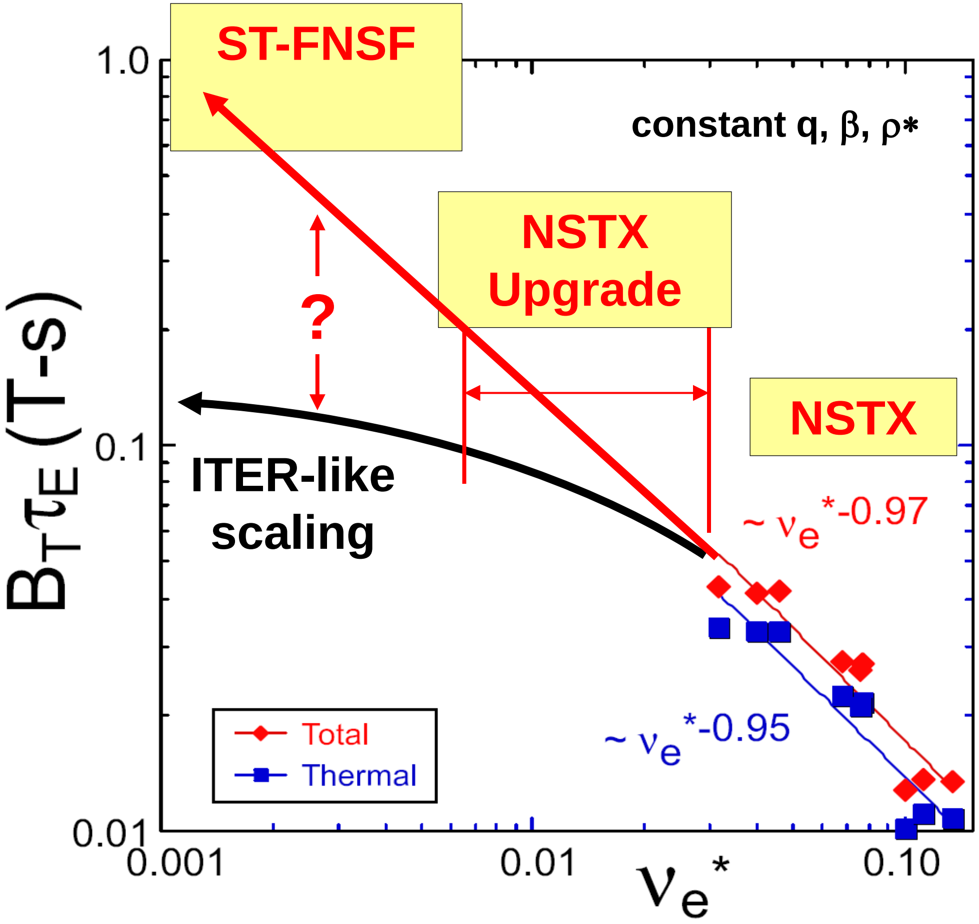
- TSC code (2D) successfully simulated helicity injection I_P ~200kA in NSTX



Additional electron heating likely required
Design of heating system is underway...

NSTX / MAST observed confinement increase at higher T_e (!)

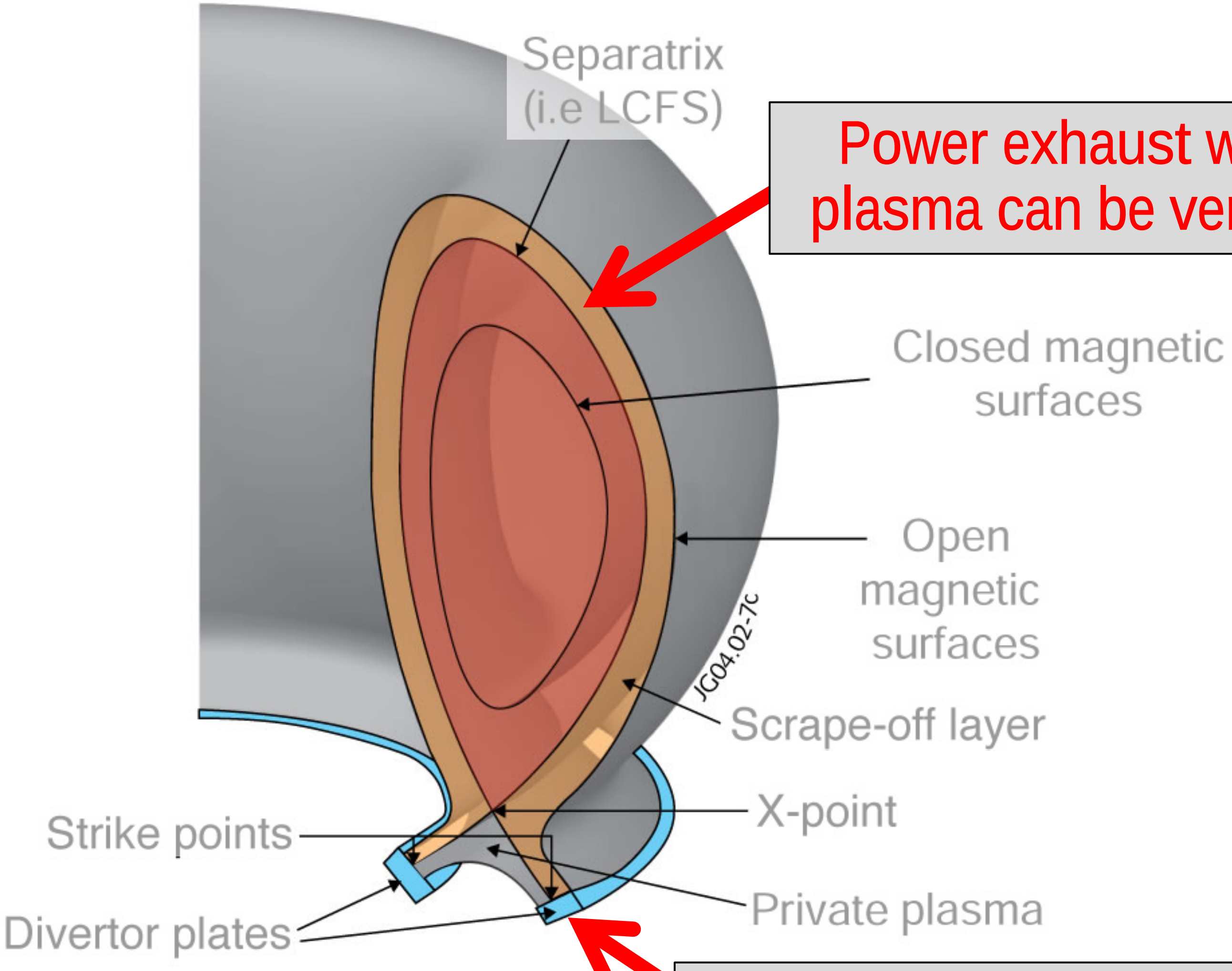
Will confinement trend continue, or look like conventional A?



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

Favorable confinement results could lead to more compact ST reactors

All modern tokamaks / STs use a “divertor” to control where power and particles are exhausted

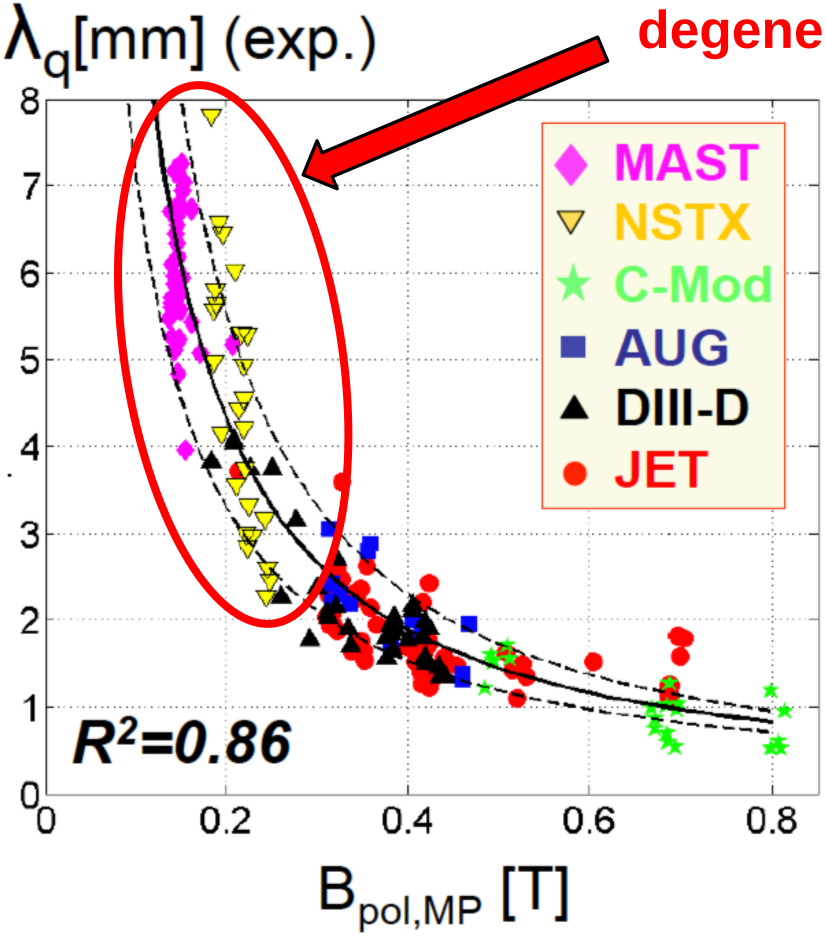


Power exhaust width outside main plasma can be very narrow (few mm)

Power / particles contact target plates here

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

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



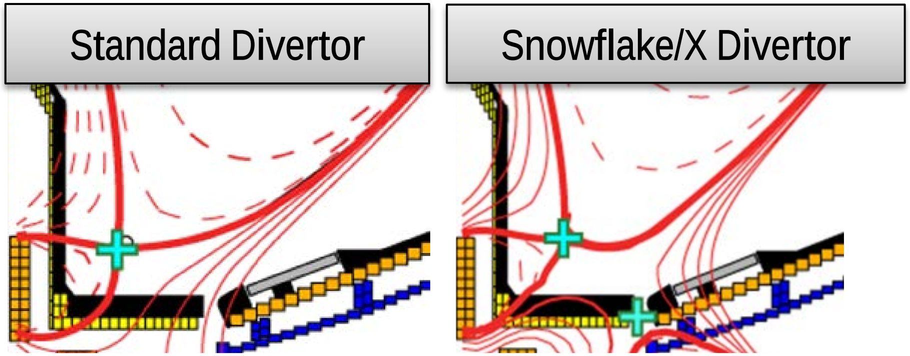
ST data breaks aspect ratio degeneracy of data set

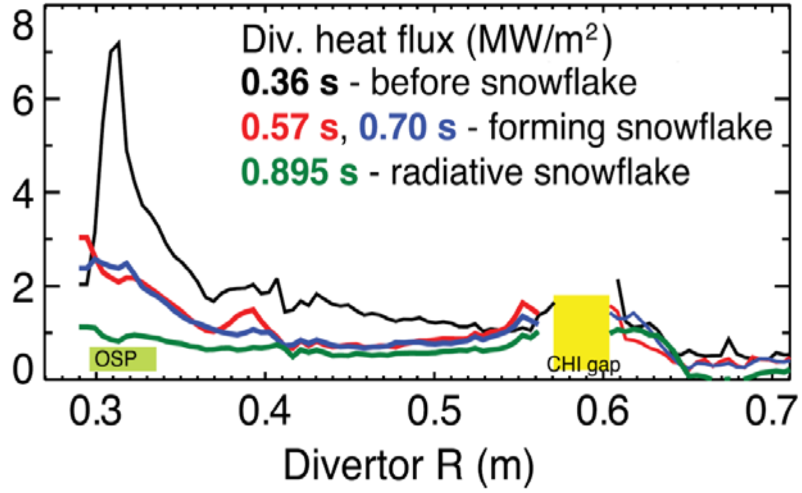
Issue: If peak heat flux in divertor region exceeds 10 MW/m^2 → material damage

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

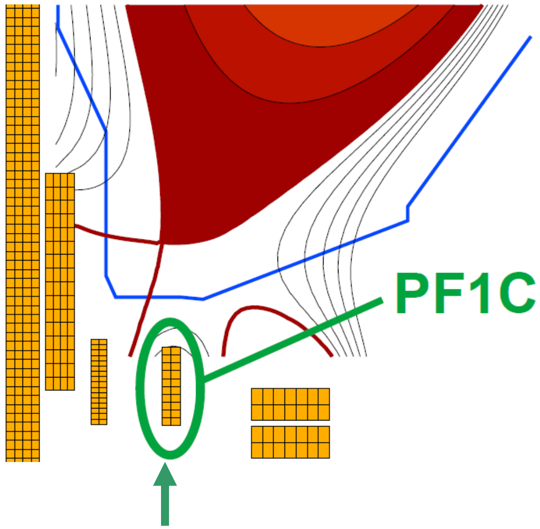
NSTX-U will test ability of radiation and advanced divertors to mitigate very high heat-fluxes

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





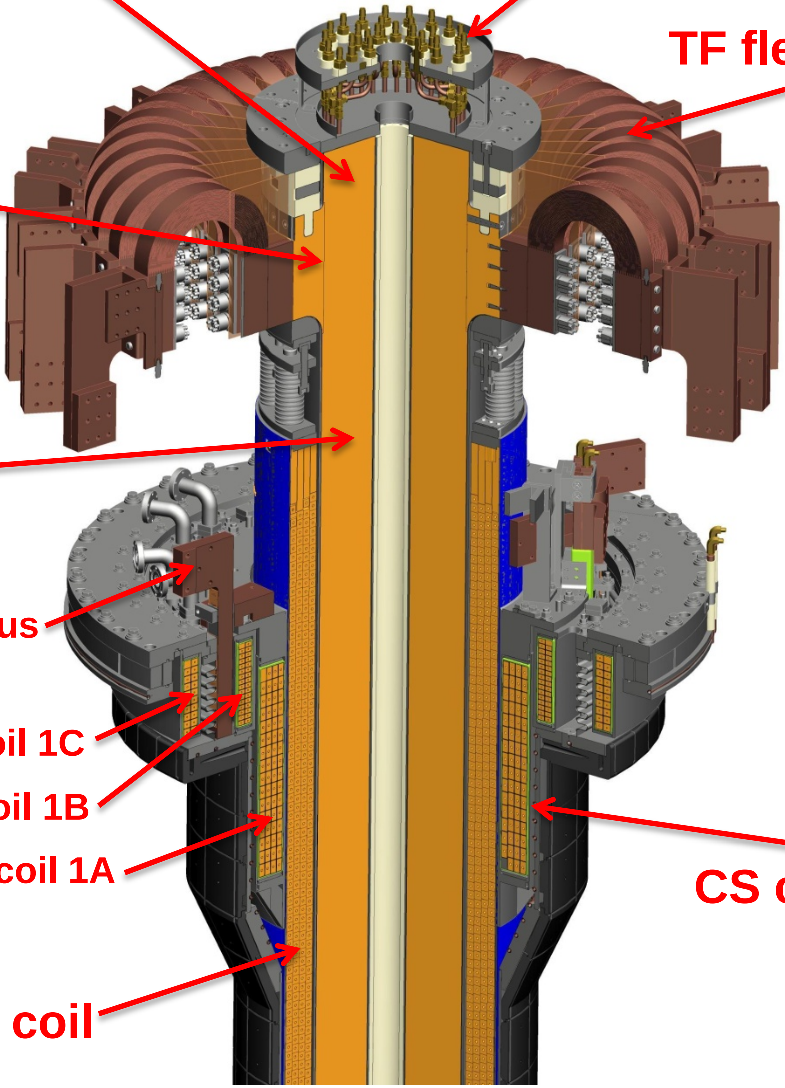
NSTX-U peak heat fluxes will be up to 4-8× higher than in NSTX



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

How was NSTX-U constructed?

New center-stack designed to handle increased forces
Identical 36 TF conductors and innovative flex-bus design



TF conductor

Friction-stir welded TF flag joints

CTD-425 cyanate ester epoxy blend insulation

CHI bus

PF coil 1C

PF coil 1B

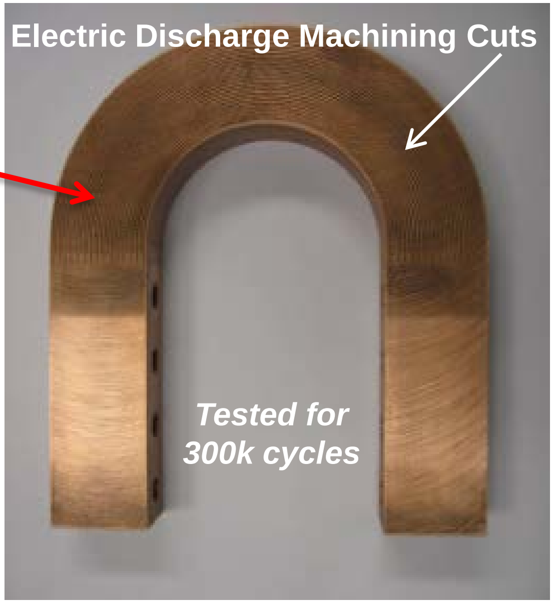
PF coil 1A

OH coil

TF cooling lines

TF flex-bus

CS casing



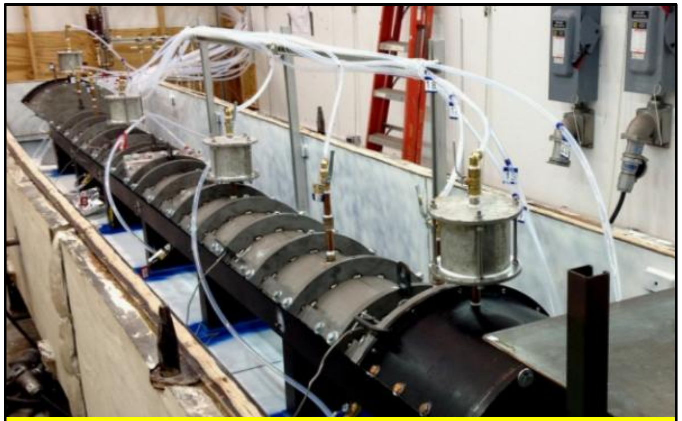
Center-stack fabrication and assembly (1)

Innovative manufacturing techniques developed

Cooling tube soldered with resin-based flux into inner TF conductor

Conductor being wrapped with fiberglass insulation

Insulated conductor being placed into mold



Assembled TF mold ready for Vacuum Pressure Impregnation (VPI) w/ CTD-425

Completion of first TF quadrant

Center-stack fabrication and assembly (2)

4 TF quadrants assembled

Full TF Bundle after VPI

Aquapour + glass layer applied

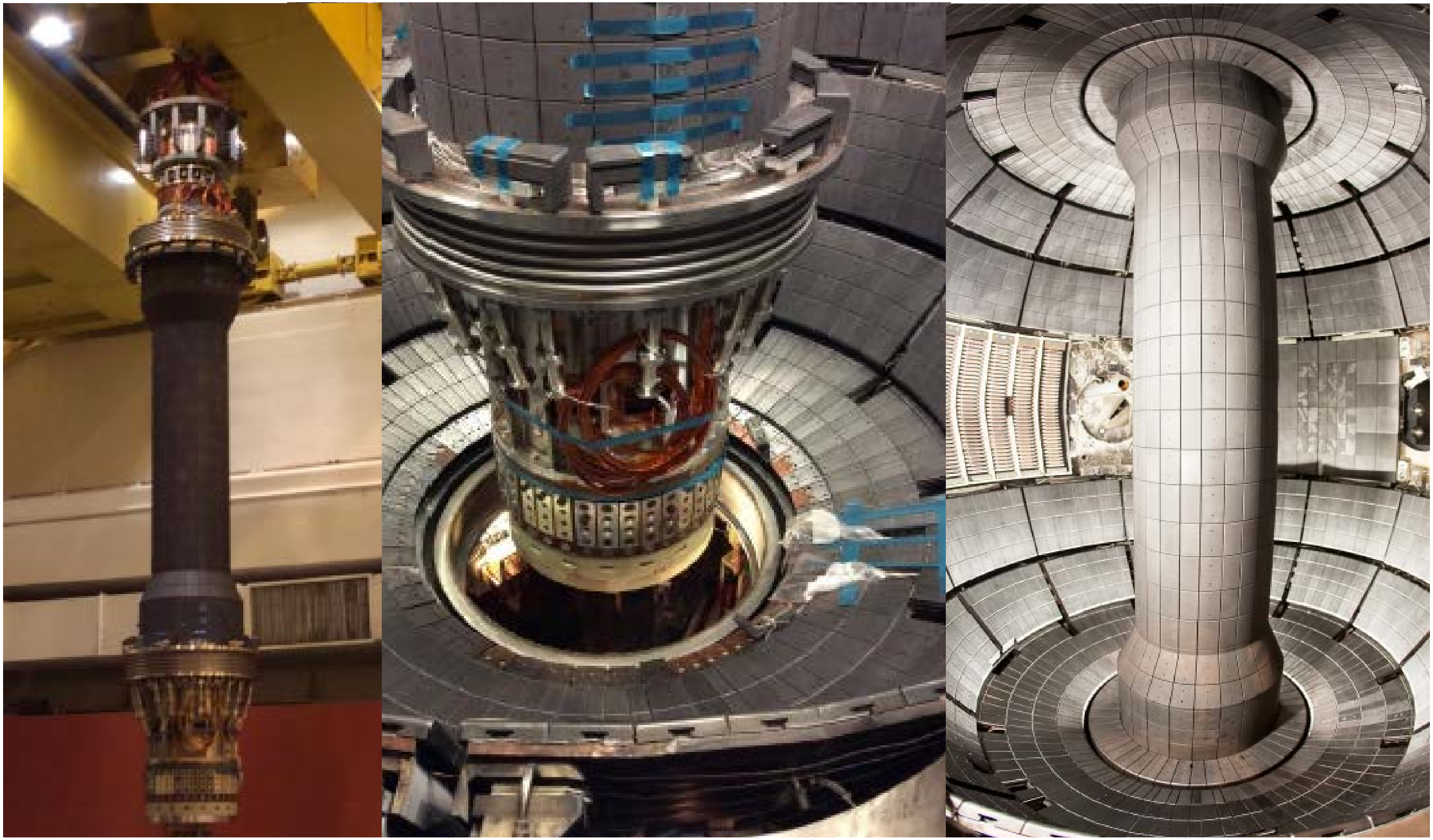
OH conductor wound over TF

Completion of transformer winding and VPI

Same-day delivery to D-Site costs extra

New Center-Stack installed in NSTX-U (!)

Vacuum pump-down achieved in January, 2015



Relocated 2nd NBI beam line box from the TFTR test cell into the NSTX-U test cell

TFTR NBI beam box and components successfully tritium decontaminated

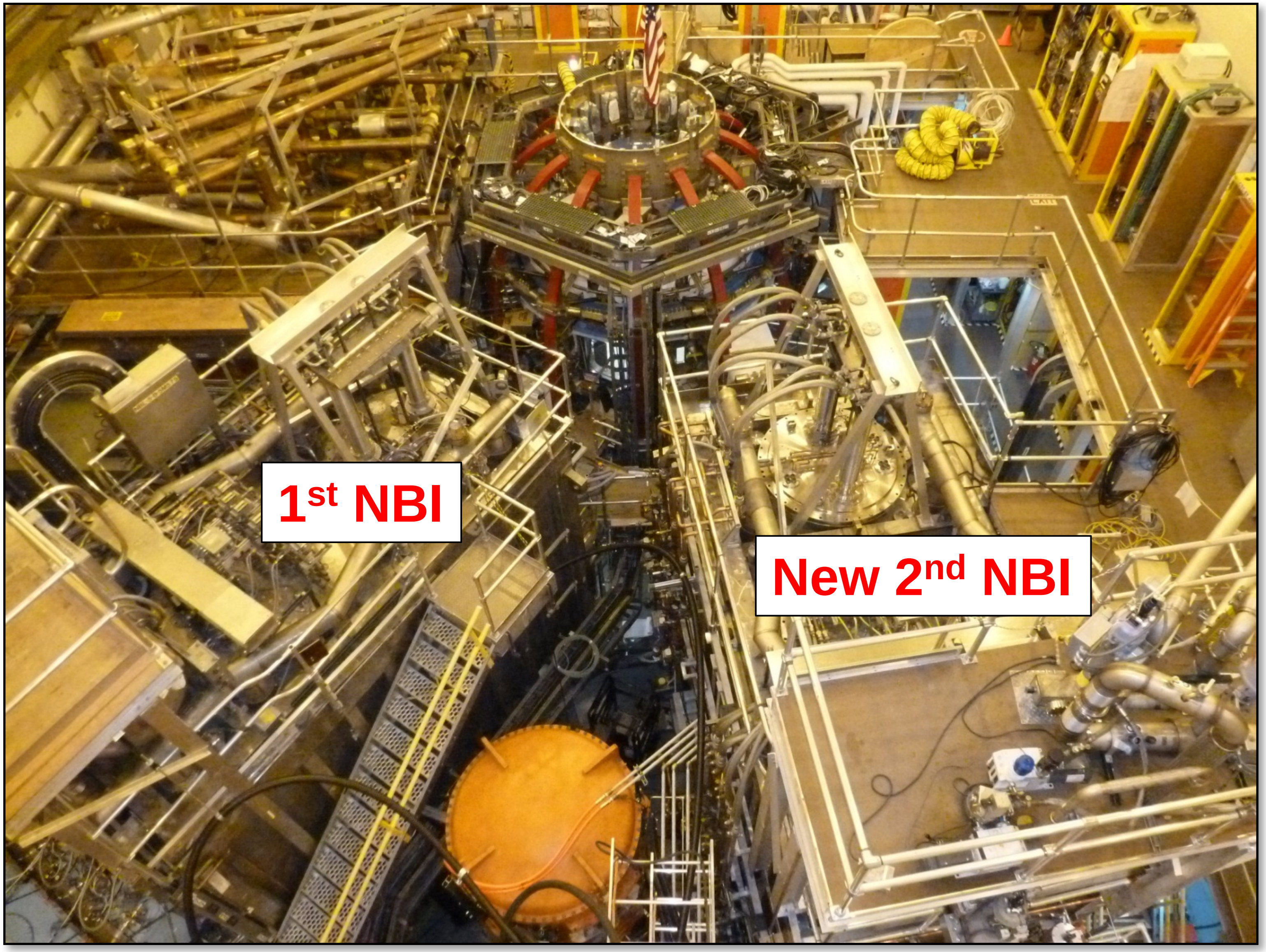
Beam Box being lifted over NSTX

Beam Box placed in its final location and aligned

Beam Box being populated with components

NSTX Upgrade Project nearing completion

Test plasmas expected in April, research plasmas in June

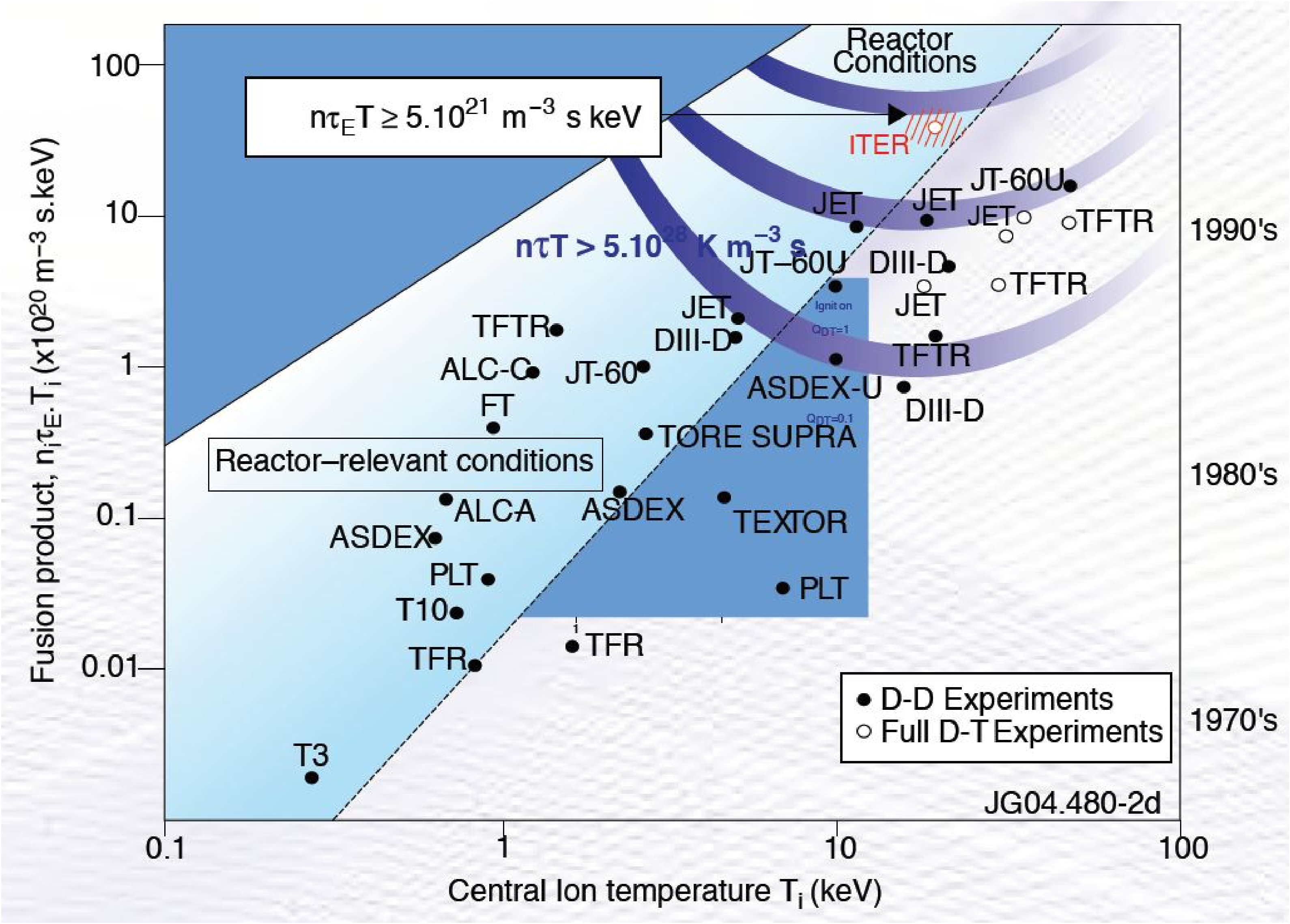


Movie time…

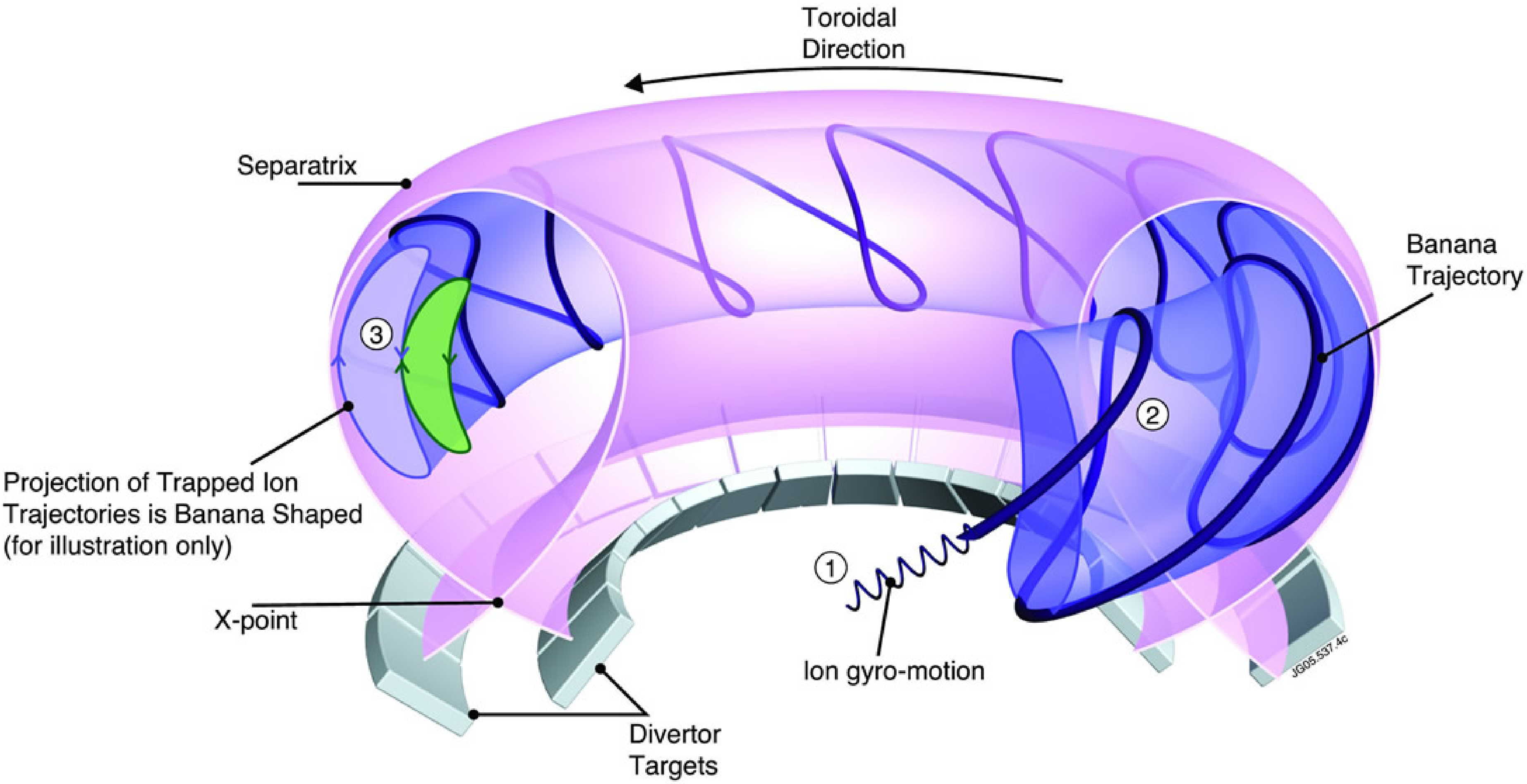
Thank you!

Back-up slides

Magnetic Confinement Fusion Performance

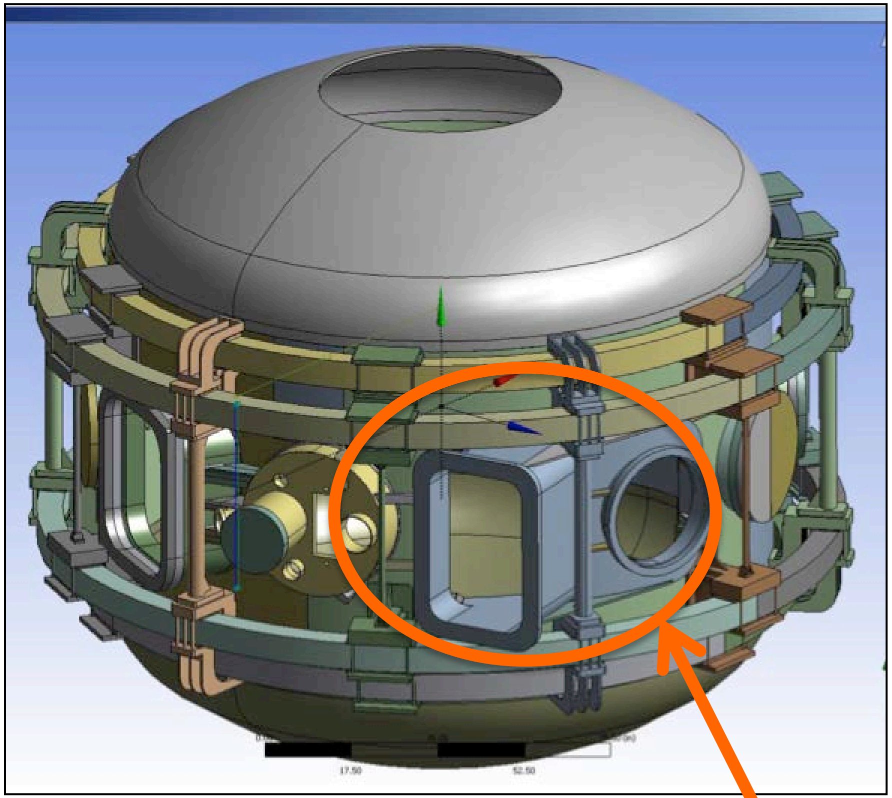


Particle trajectories in a tokamak



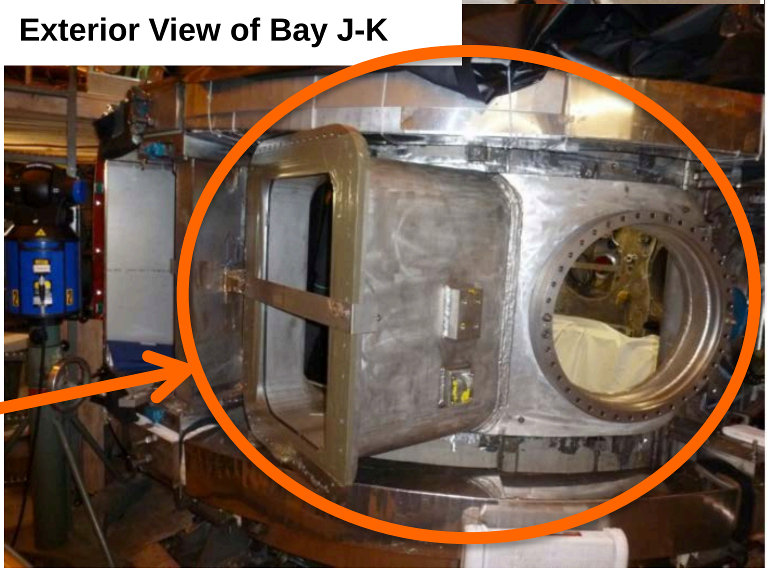
Highly tangential 2^{nd} NBI enabled by new port

Outer wall radius moved outward to avoid beam clipping



Interior View of Bay J-K

Exterior View of Bay J-K



JK cap

Substantial structural and vacuum vessel upgrades

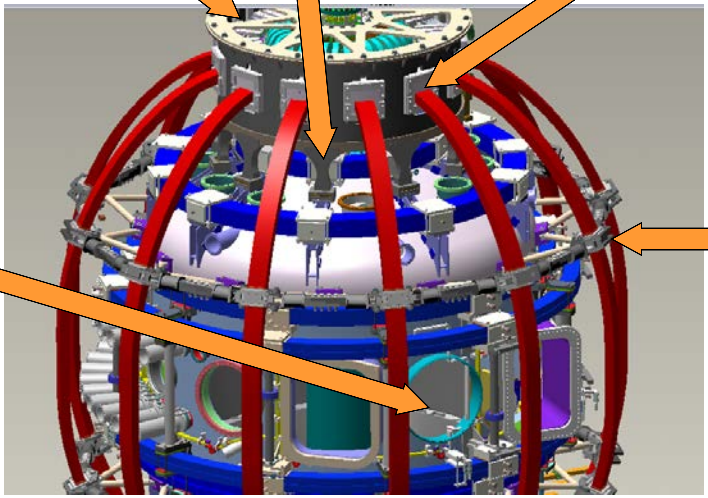
Must handle 4x higher electromagnetic loads

Upper Aluminum Block Internal Reinforcements

Upper Umbrella Arch Reinforcements

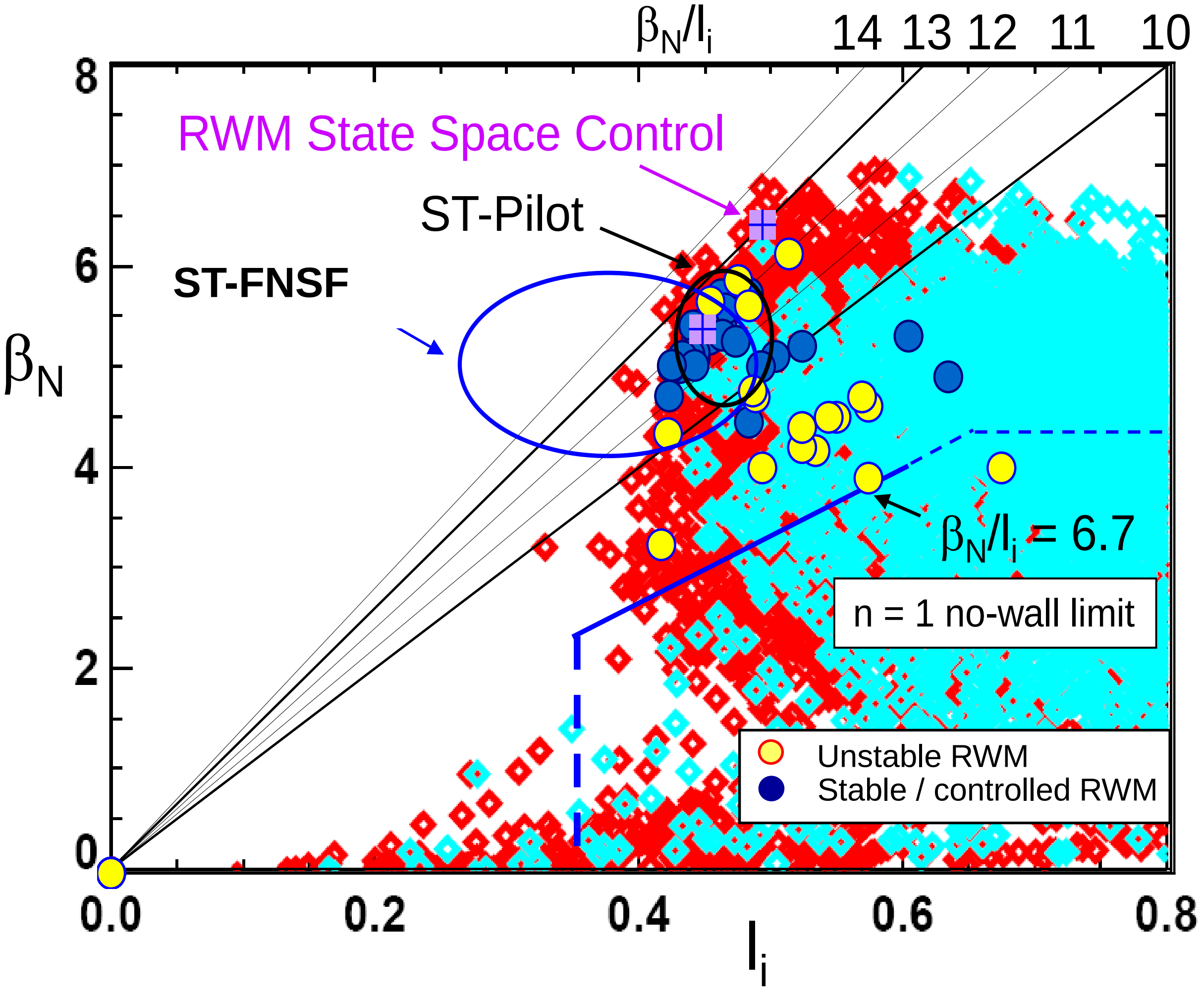
Upper Aluminum Block External Reinforcements

Bay-L Cap



TF-Vessel Clevis, TF Outer Leg Support

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)

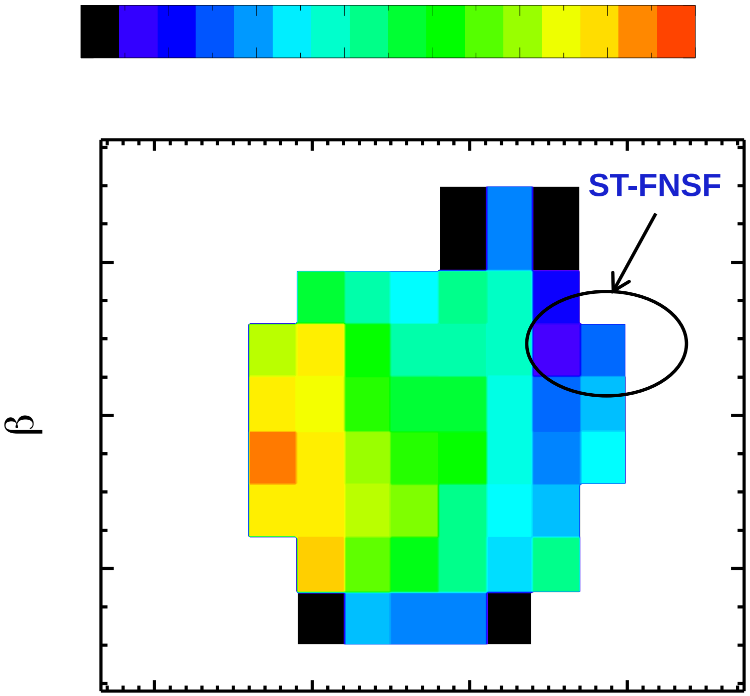
J. W. Berkery, PRL (2011)

W. Zhu, PRL (2006)

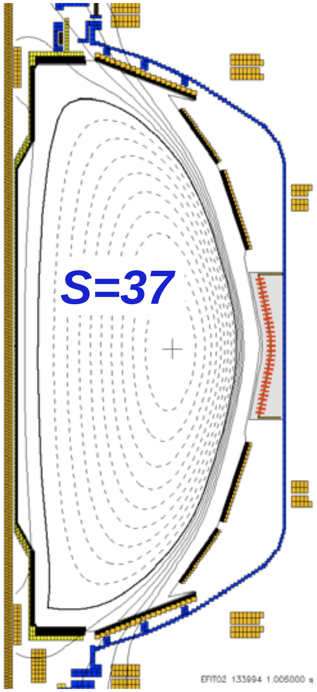
S.A. Sabbagh at this APS

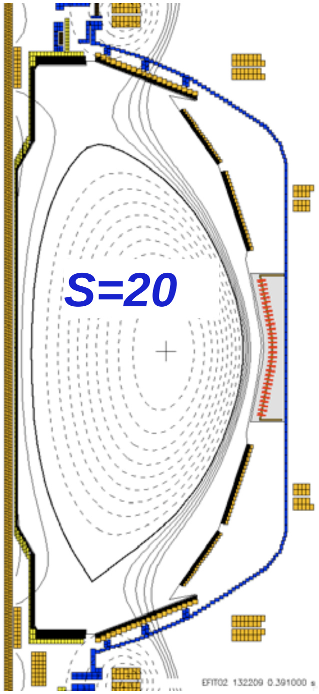
Major mission of NSTX-U is to achieve fully non-inductive operations at high β

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



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





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

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

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

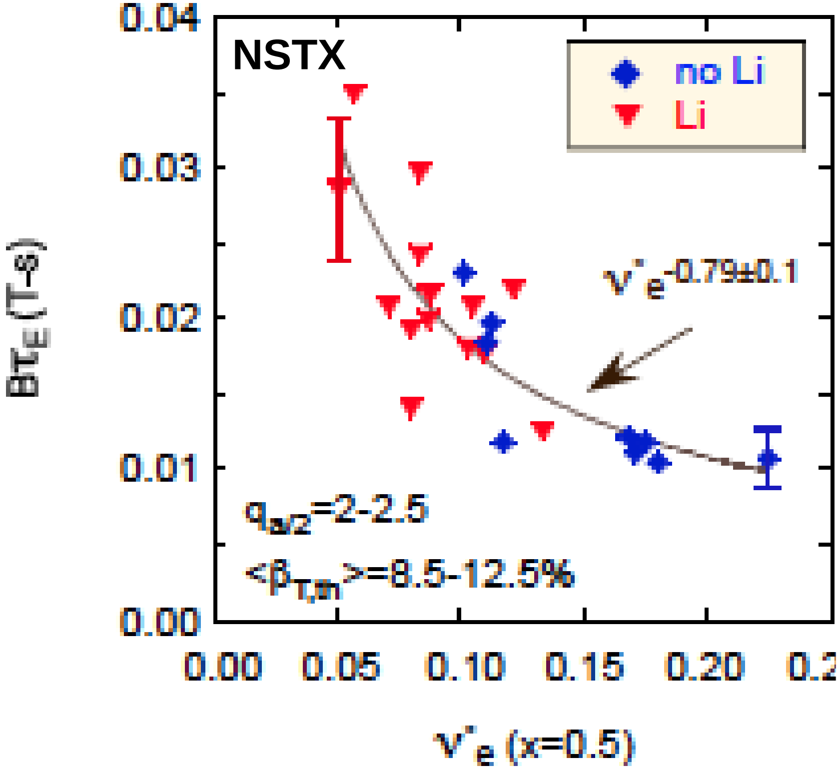
Upgrades will test and improve these favorable trends in a systematic way

Favorable Confinement Trend with Collisionality and β found

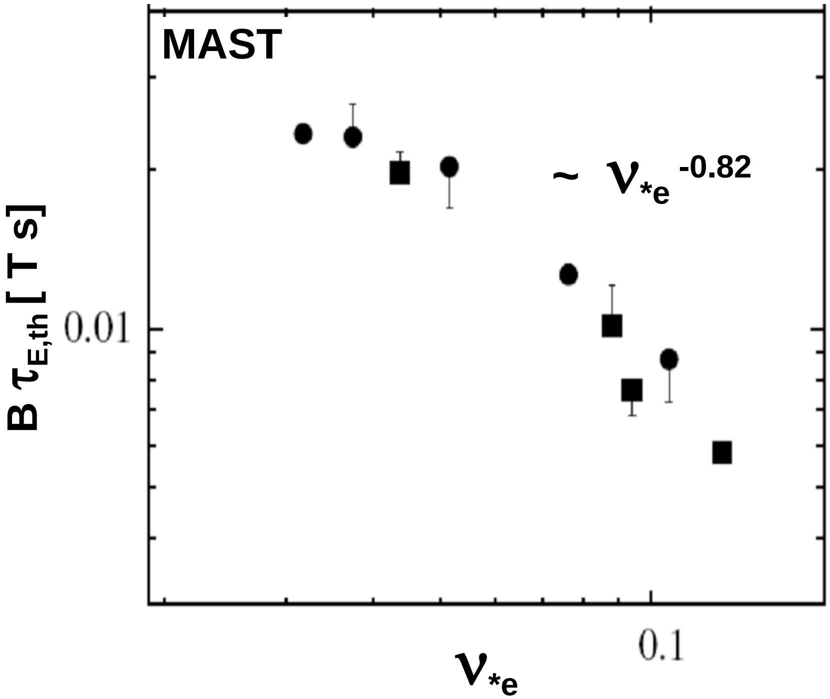
Important implications for future STs and Demo with much lower ν_*

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

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



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

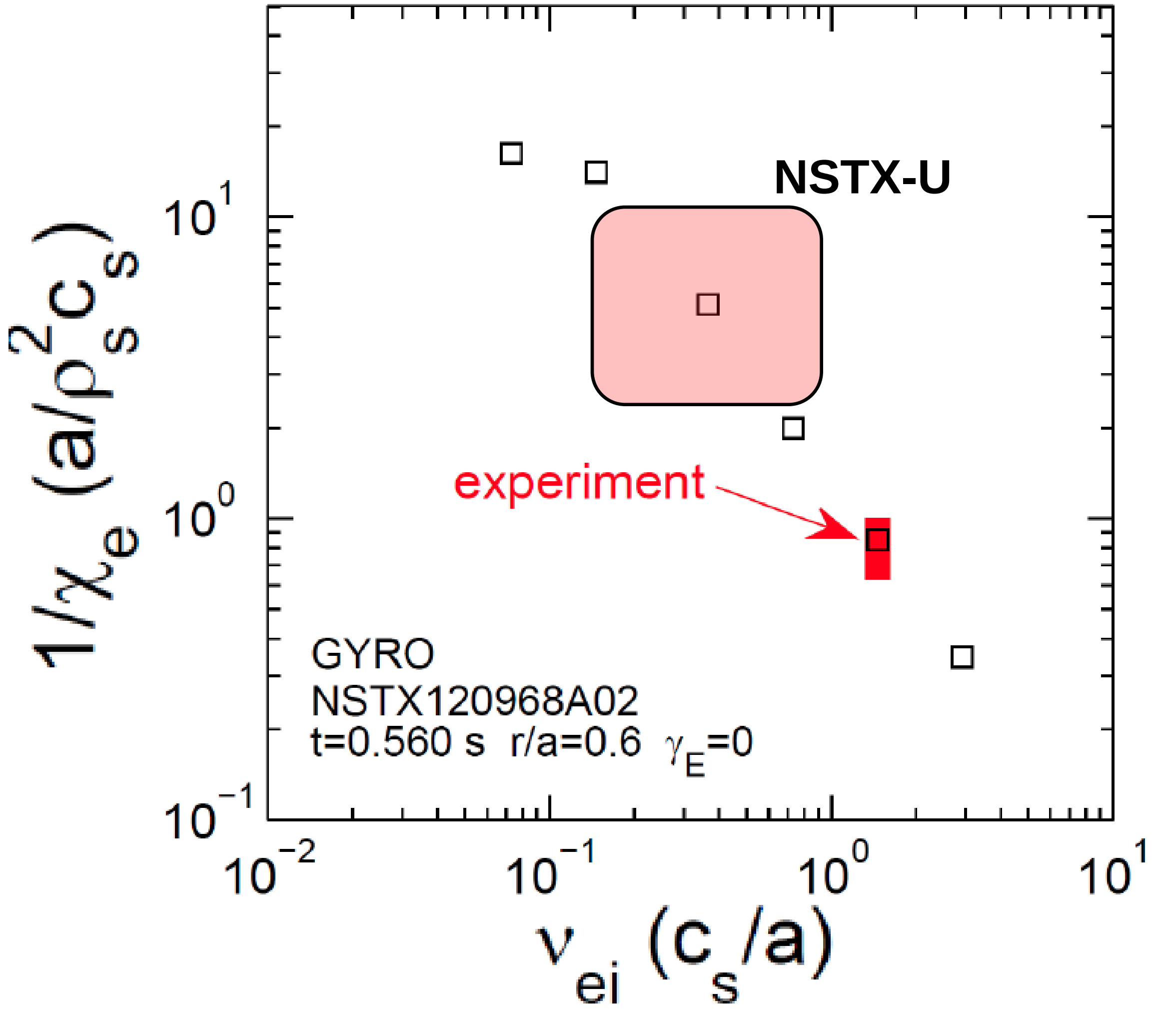


M. Valovic et al., NF (2011)

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.

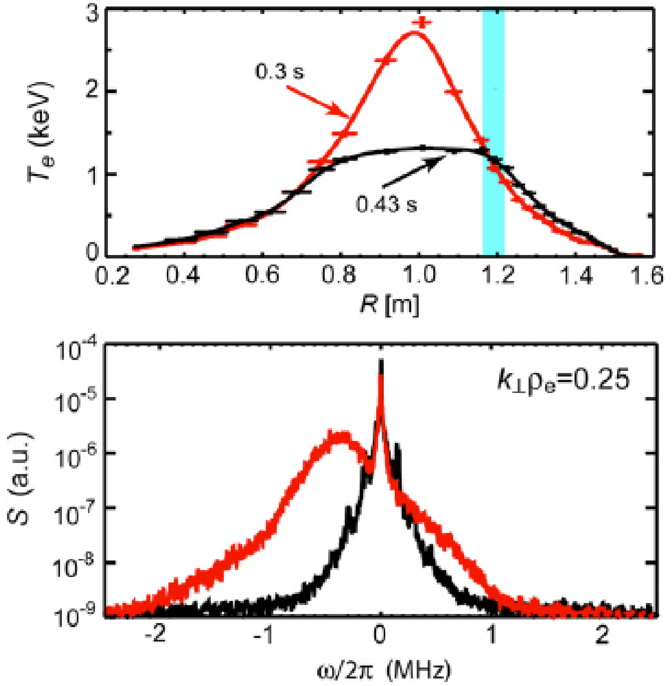
Further electron confinement improvement expected due to reduced collisionality.

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

ETGs measured for the first time with high-k scattering

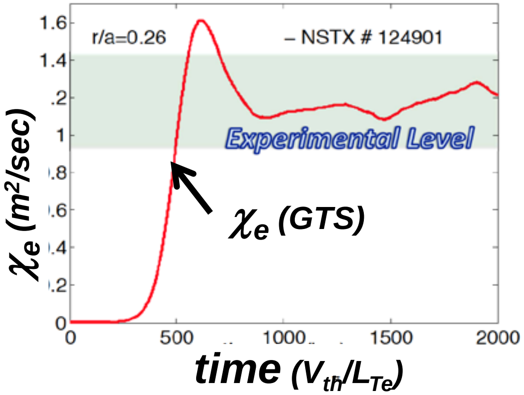
High β_e or larger $\rho_e \propto \beta_e^{0.5}$ of ST plasma enabled measurement of ETGs.

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



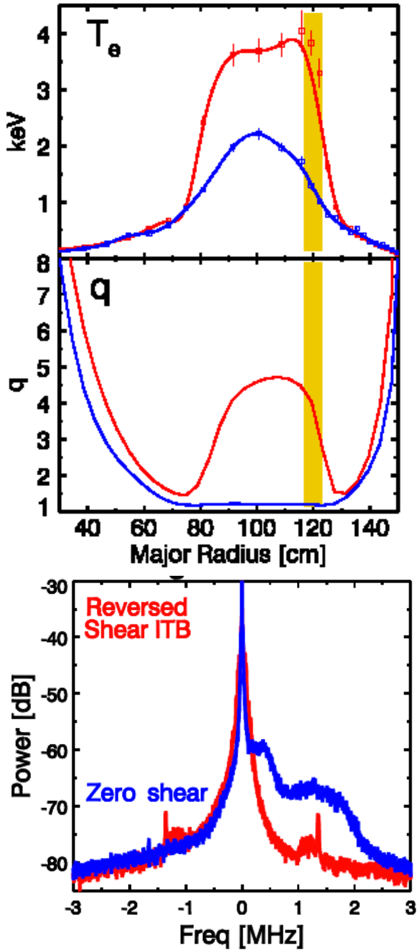
E. Mazzucato et al., NF (2009)

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

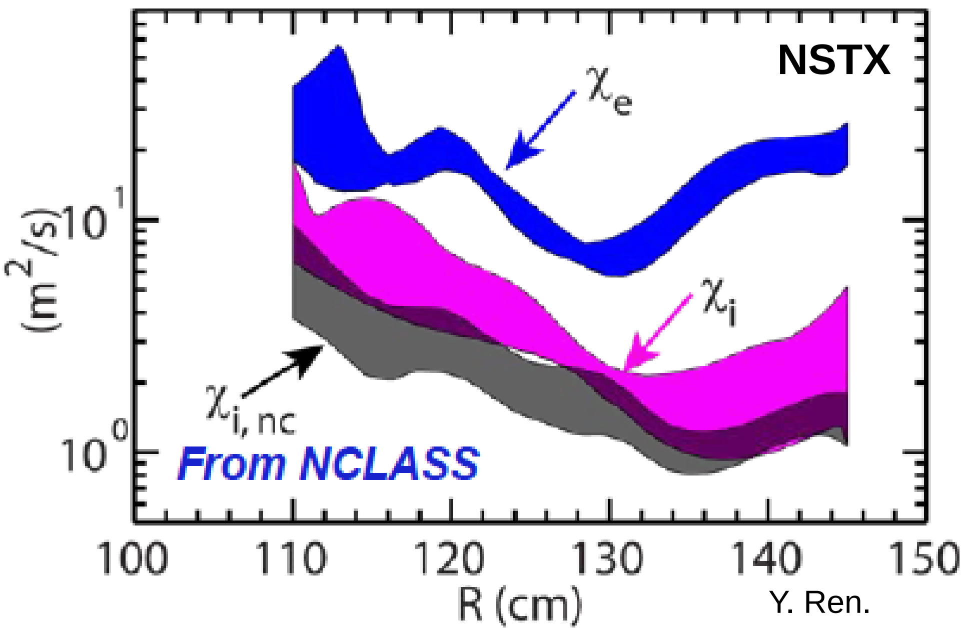
High Confinement Needed for Compact FNSF

High confinement H-mode in the range of FNSF obtained

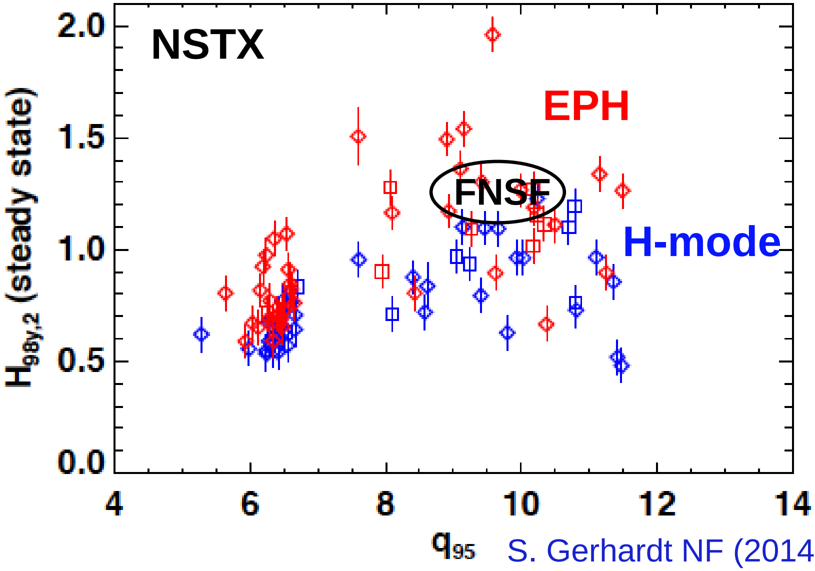
Fusion gain Q depends strongly on "*H*", Q $\propto H^{5-7}$
Higher H enables compact ST-FNSF H = 1.2 – 1.3
Higher H gives more reactor design flexibility and margins.

- **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



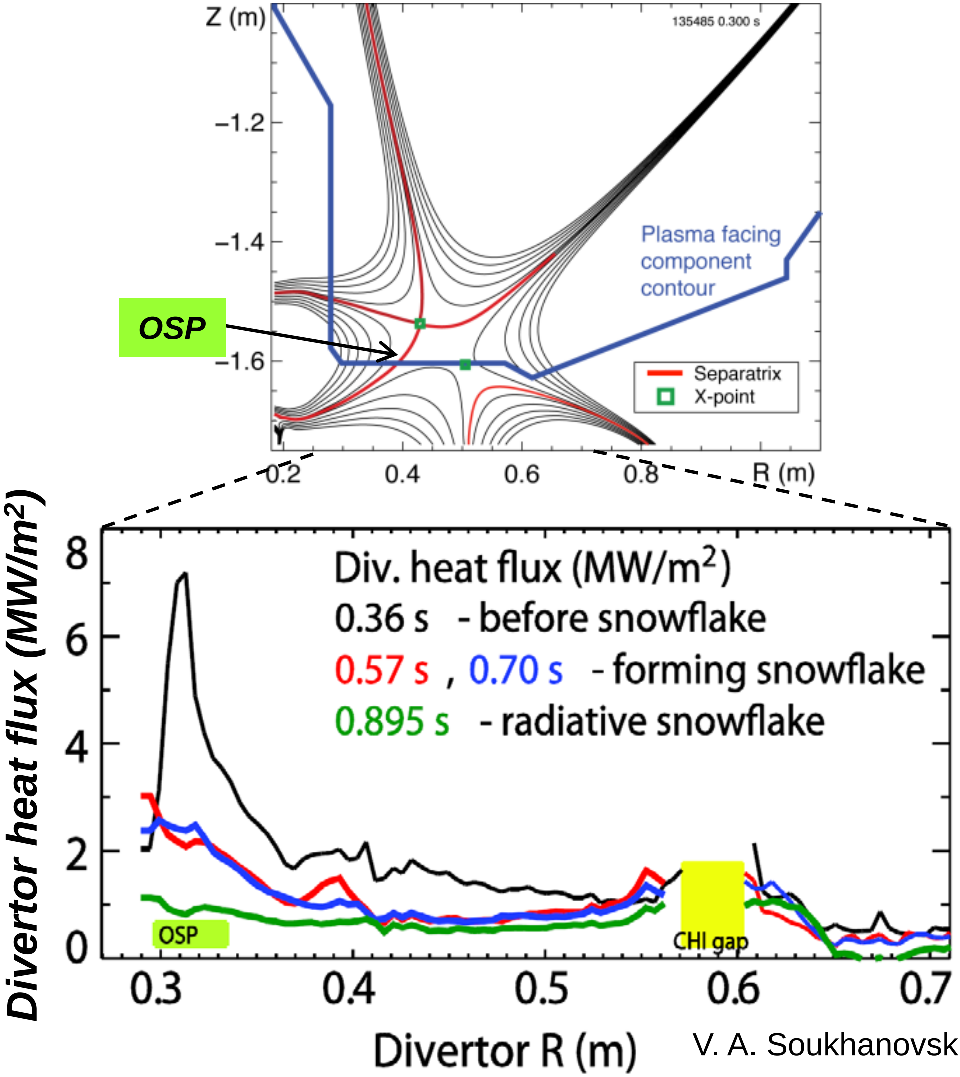
Y. Ren.



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

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.

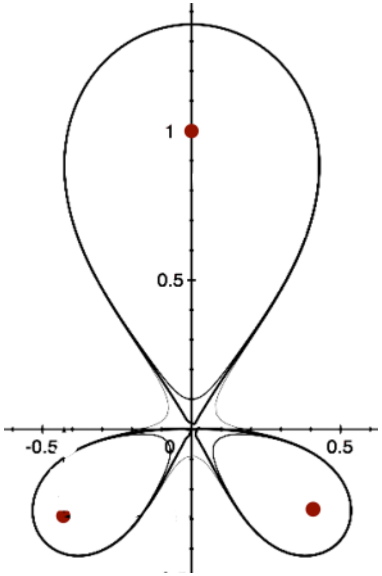
- **Up-and-down symmetric Snow Flake / X-divertors**
- **Lithium + high-Z metal PFCs**

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

Innovative Heat Flux Mitigation via Divertor Flux Expansion

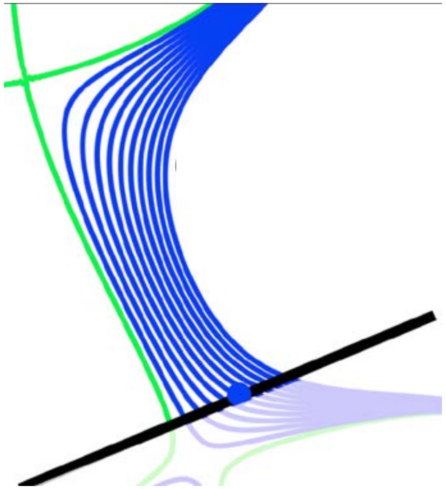
Lower toroidal field of outboard divertor leg of STs facilitates heat flux mitigation by divertor flux expansion solutions

Snow-flake



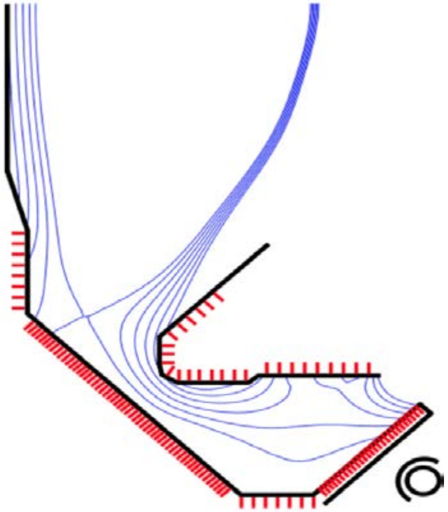
D. Ryutov, et al., PoP (2007)

X-Divertor: CREST



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

Super-X: MAST-U



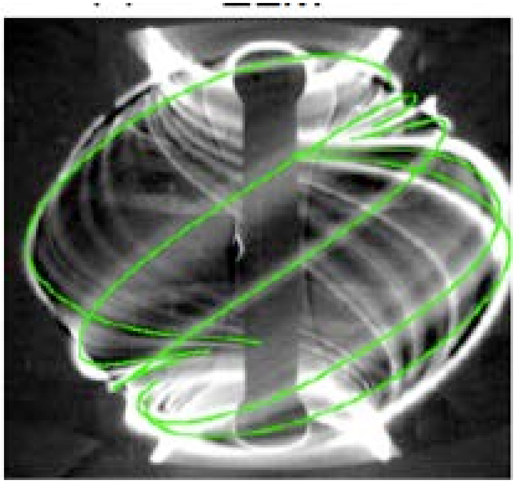
Kotschenreuther, et al., PoP (2007)

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

H-mode / ELM physics: High Priority Research Goal

Unmitigated ELMs could cause PFC damage in reactors

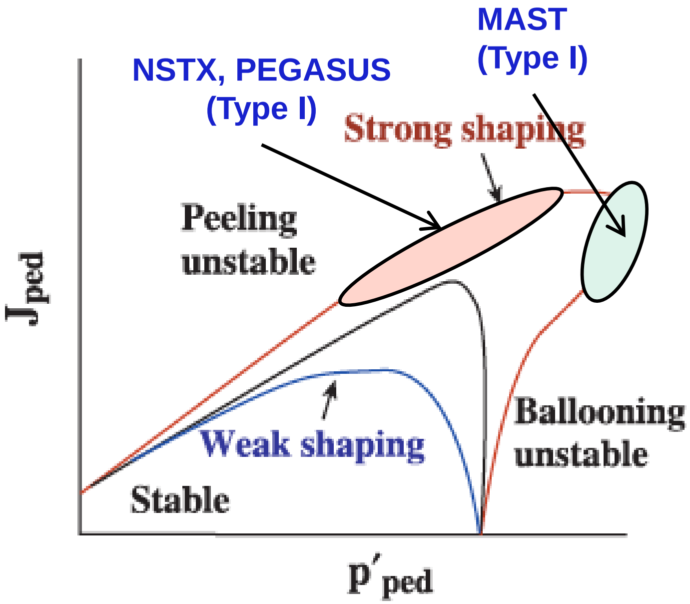
Video images of MAST plasmas showing a filamentary ELM structure.



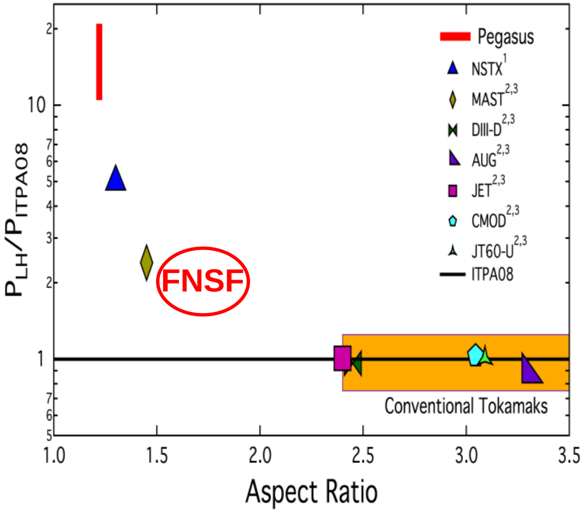
N. Ben Ayed et al., PPCF (2009).

ST is in strongly shaped ELM regimes

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



L-H power threshold scaling extended for low A

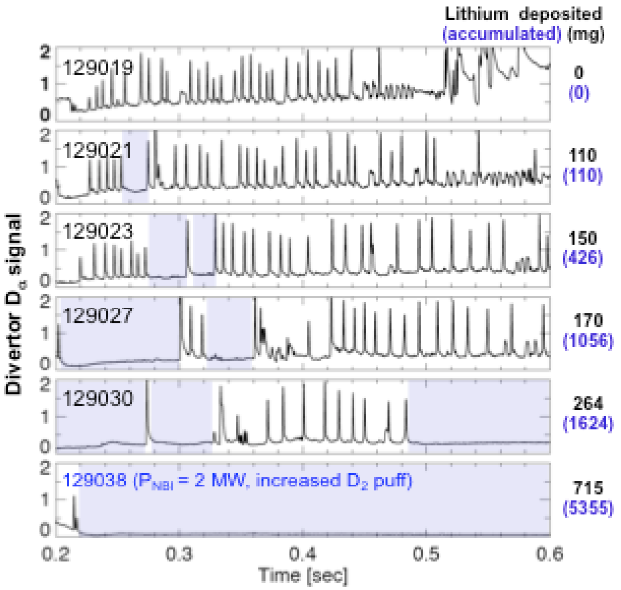


K.E. Thome et al., EPR (2014)

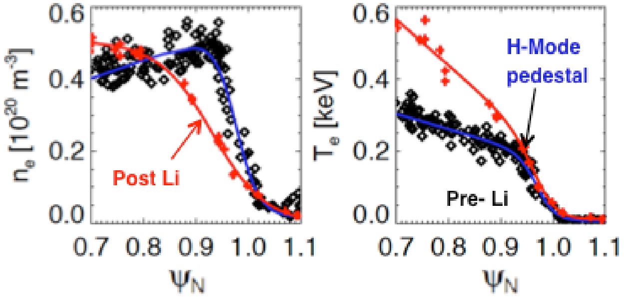
- NSTX/MAST/PEGASUS accessed H-mode at very low heating power < 1 MW and also in ohmic plasmas
- NSTX-U and MAST-U will provide H-mode access scaling for FNSF

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

ELMs stabilized with edge pressure modification with Li in NSTX

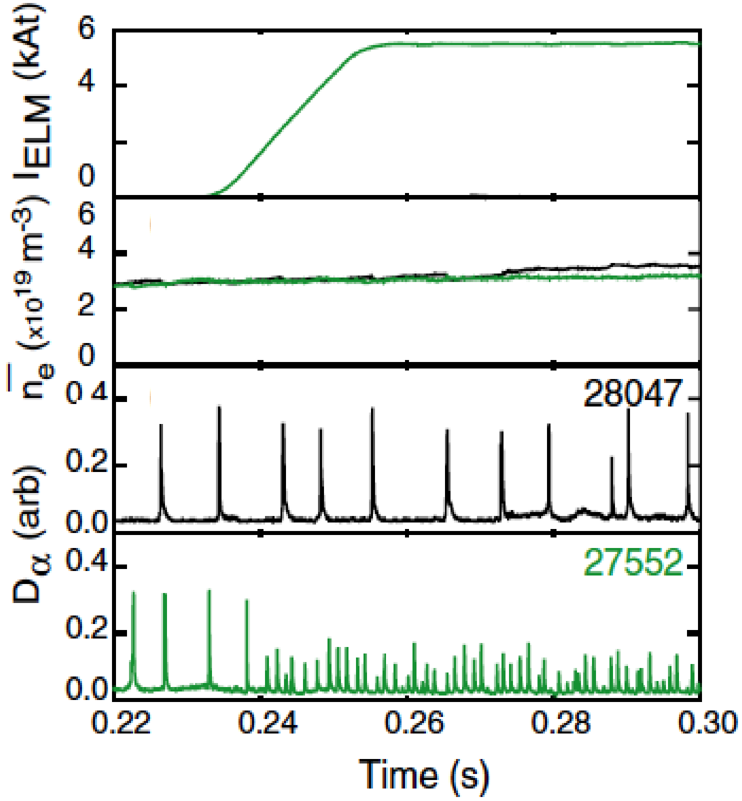


D.K. Mansfield, et al., JNM (2009)



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

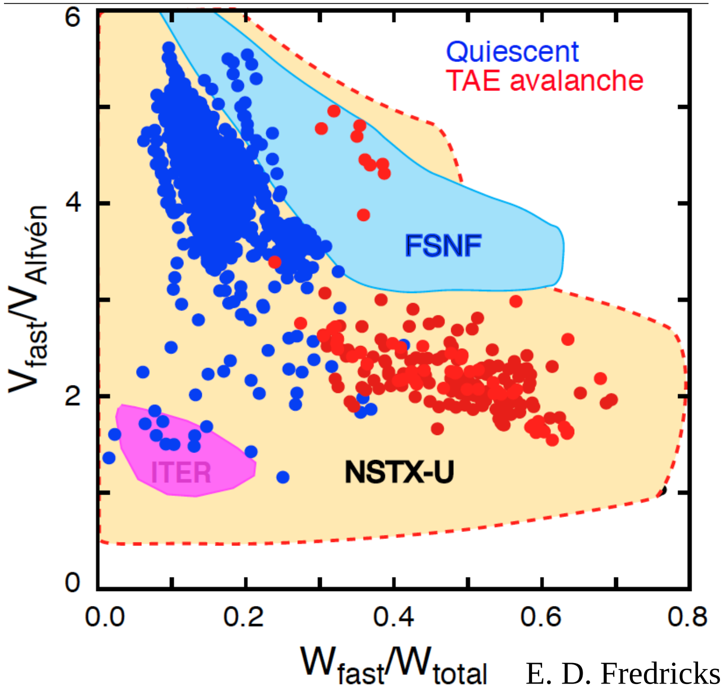
A. Kirk et al., NF (2013)

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

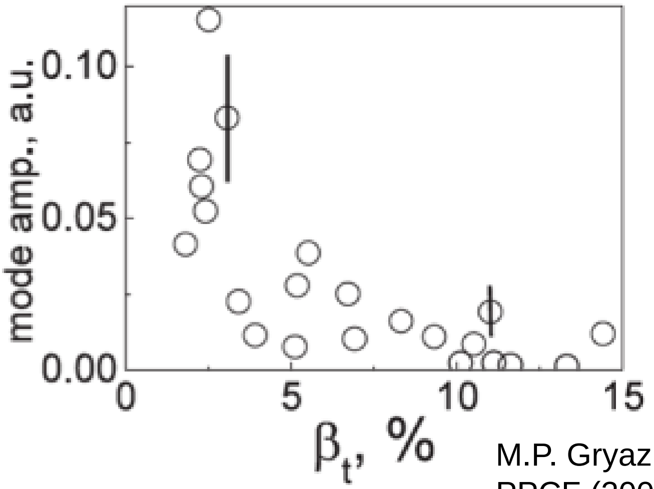
EP parameter space



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

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

Stabilization of TAEs at high β in MAST

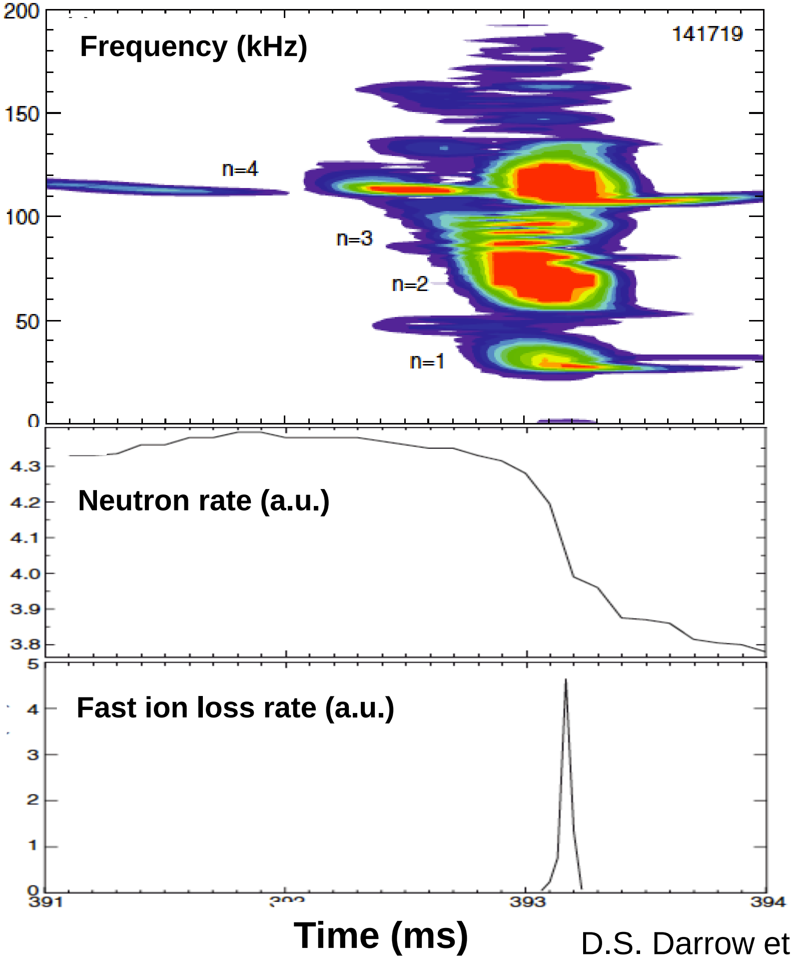


M.P. Gryaznevich et al.
PPCF (2004)

“TAE avalanche” shown to cause energetic particle loss

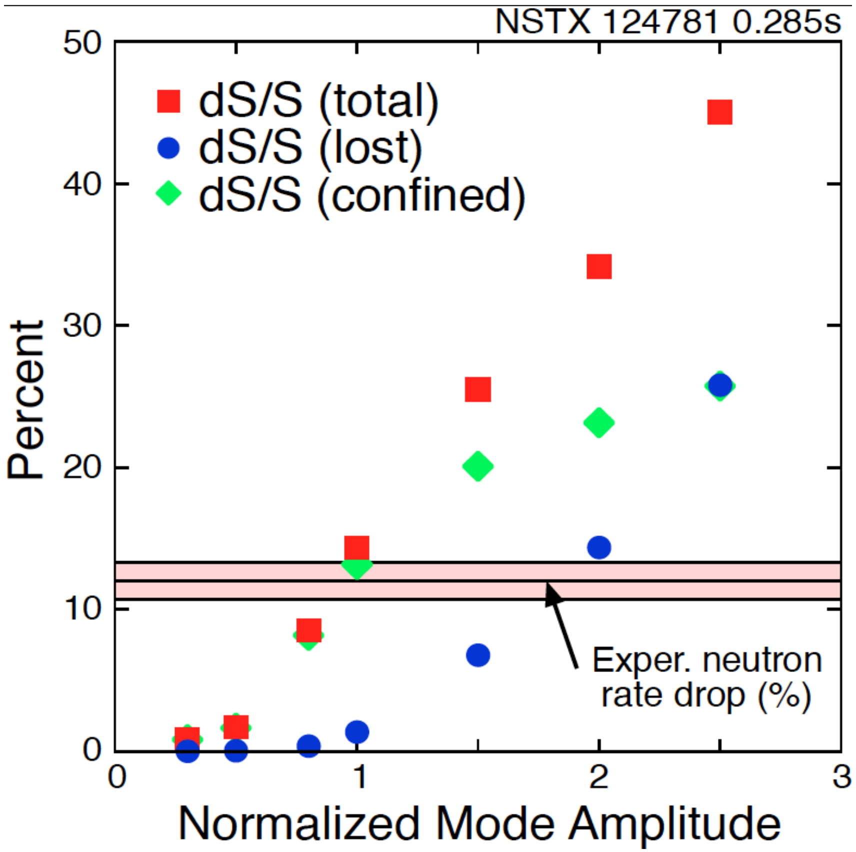
Uncontrolled α-particle loss could cause reactor first wall damage

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



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

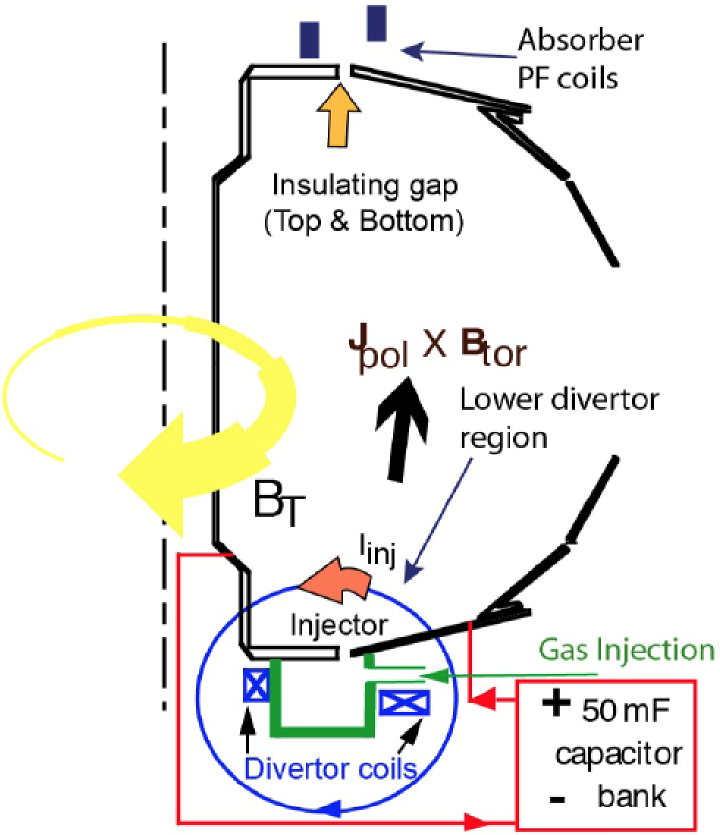
Progress in simulation of neutron rate drop due to TAE avalanche



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

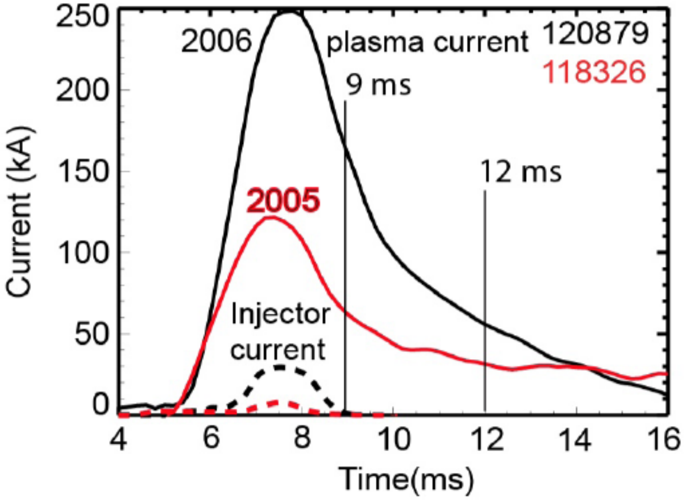
Helicity Injection Is an Efficient Method for Current Initiation
Coaxial Helicity Injection (CHI) Concepts Being Developed

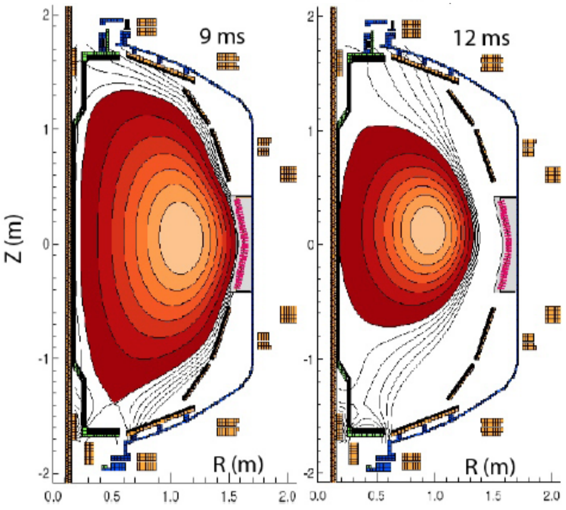
CHI developed on HIT and HIT-II and transferred to NSTX



R. Raman et al., PRL (2006)

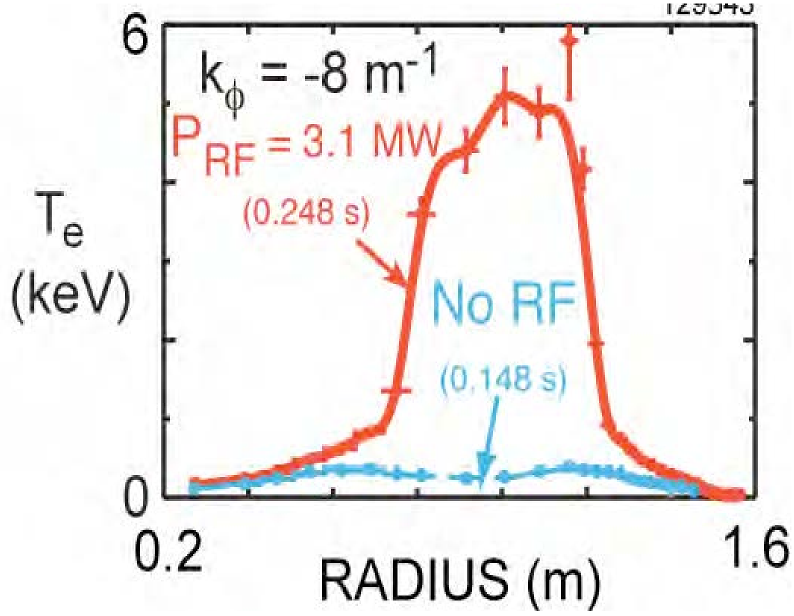
Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX





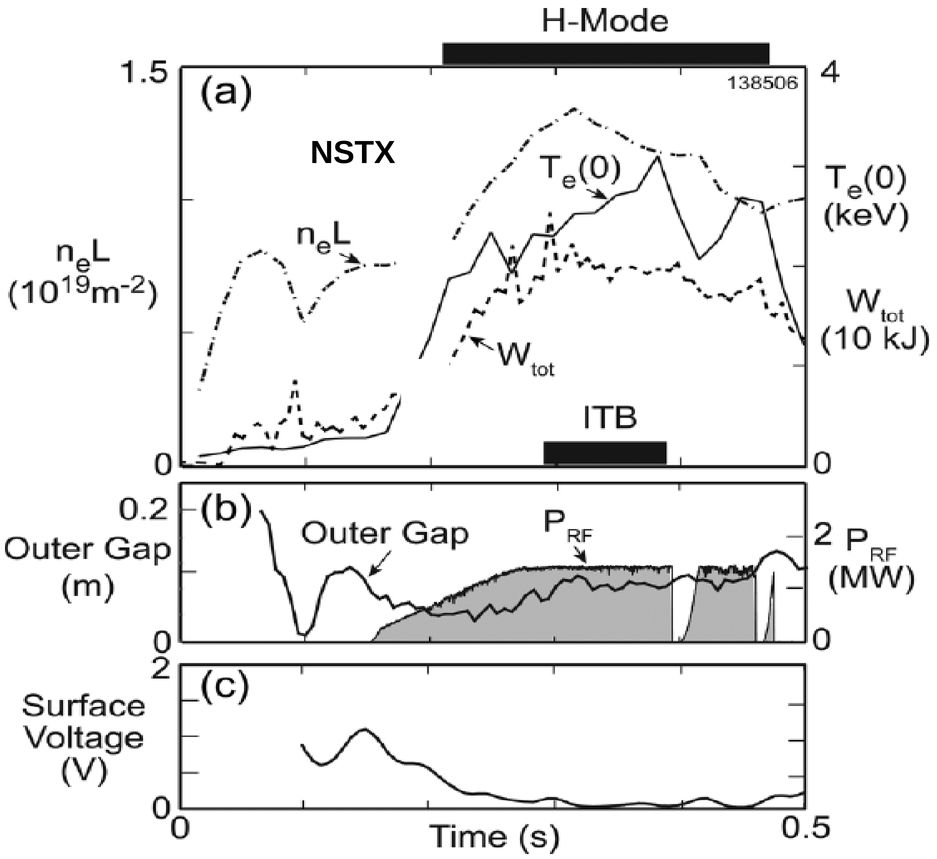
Current Ramp-Up and Profile Control Crucial for FNSF
Major Research Topics for NSTX-U

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



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

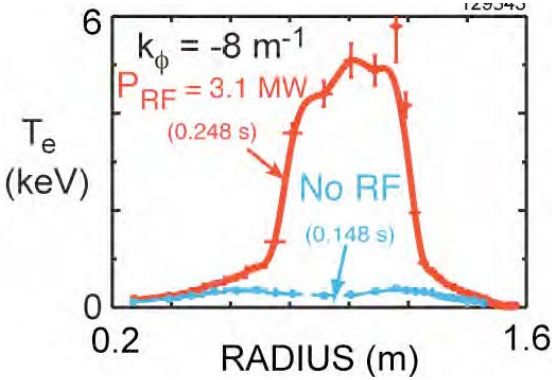
Near sustained discharges obtained with modest HHFW power



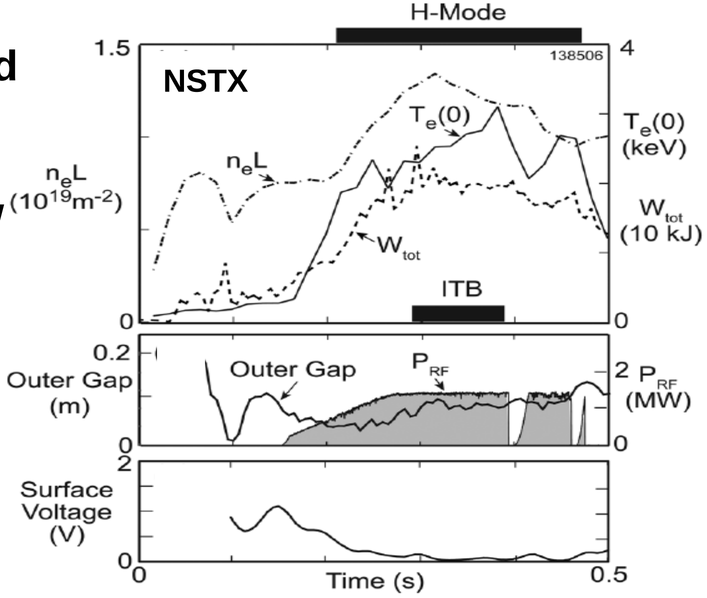
HHFW current ramp-up will be tested in NSTX-U at higher power ~ 4 MW.

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.



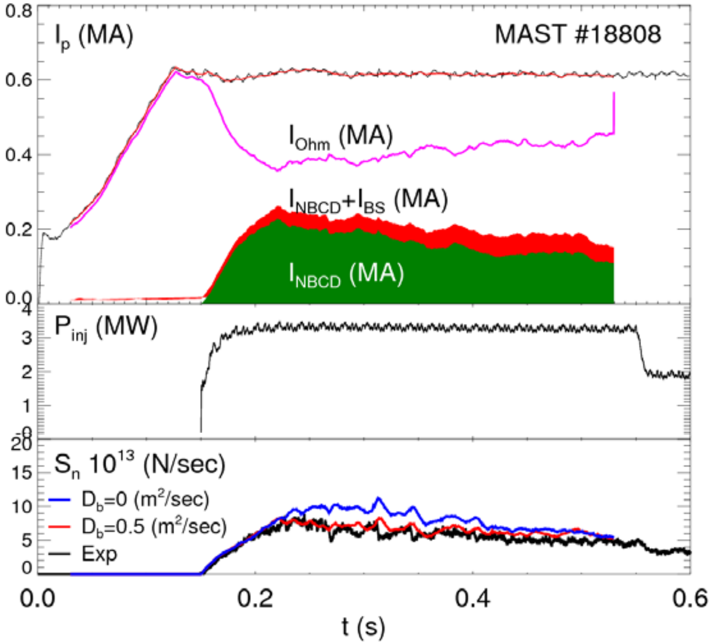
Near sustained discharges obtained with modest HHFW power.



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



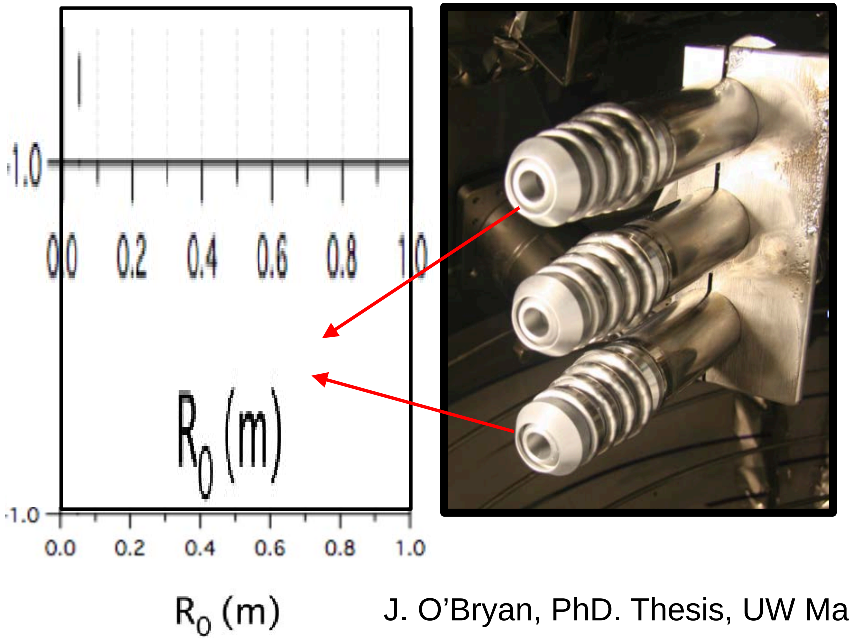
M. Turnyanskiy et al., NF (2009)

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

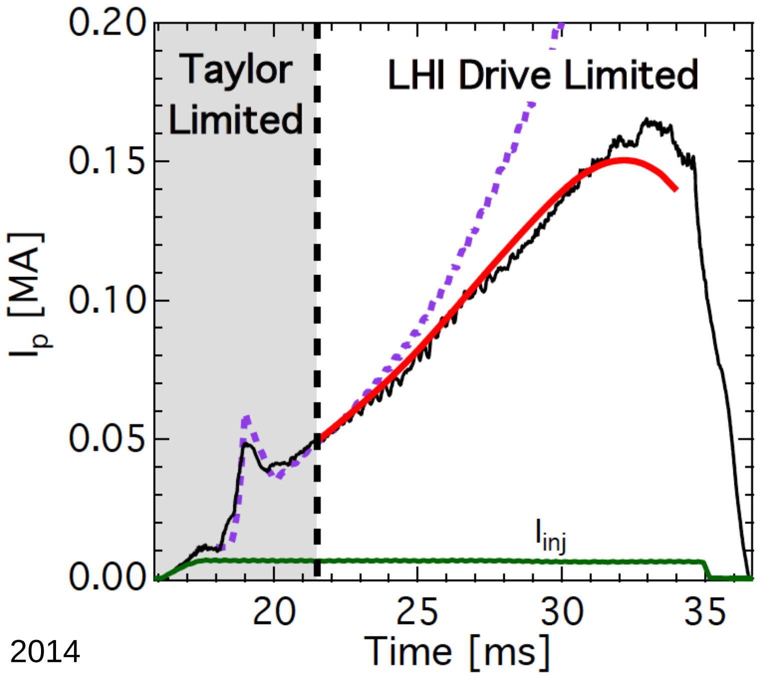
Helicity Injection Is an Efficient Method for Current Initiation
Local Helicity Injection (LHI) Concepts Being Seveloped

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



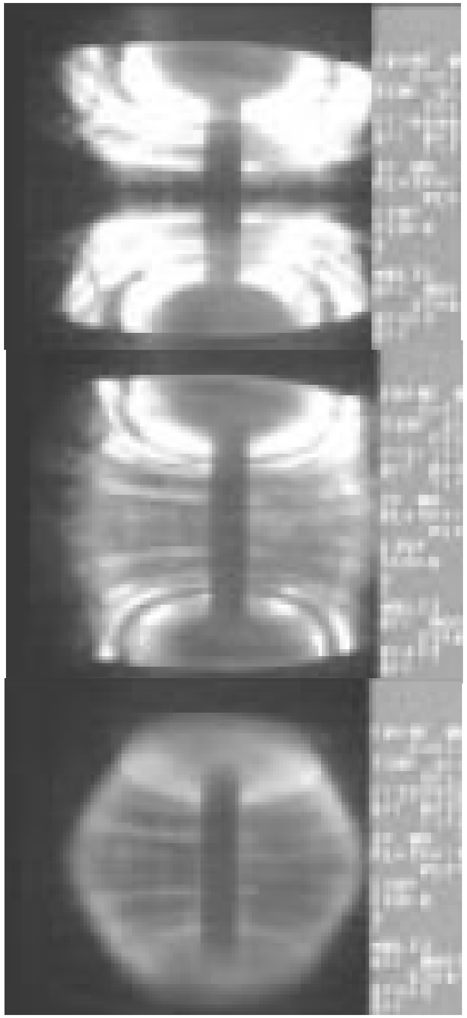
Improving predictive capability for both CHI and LHI

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

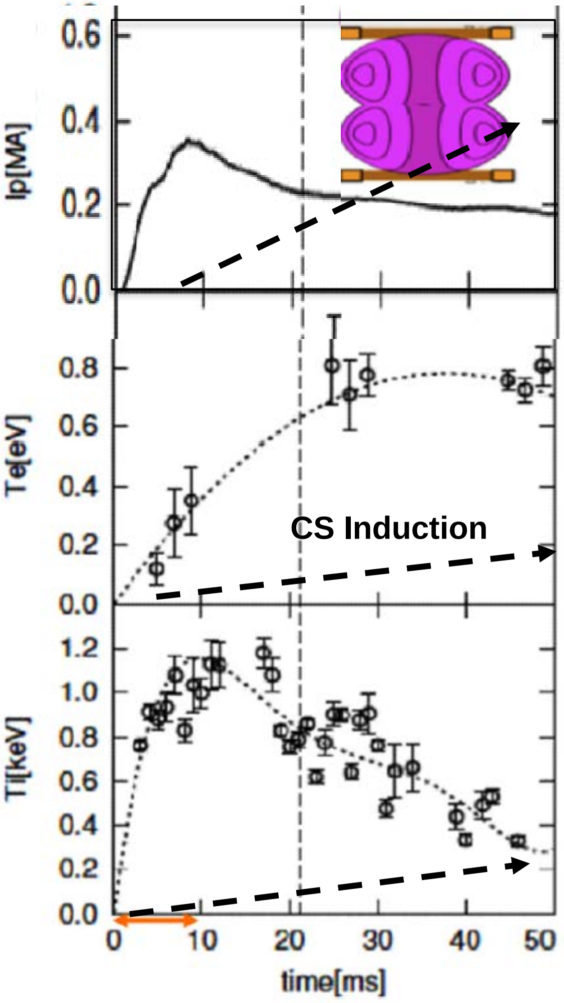
Merging Start-Up Yielded High Current STs

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)

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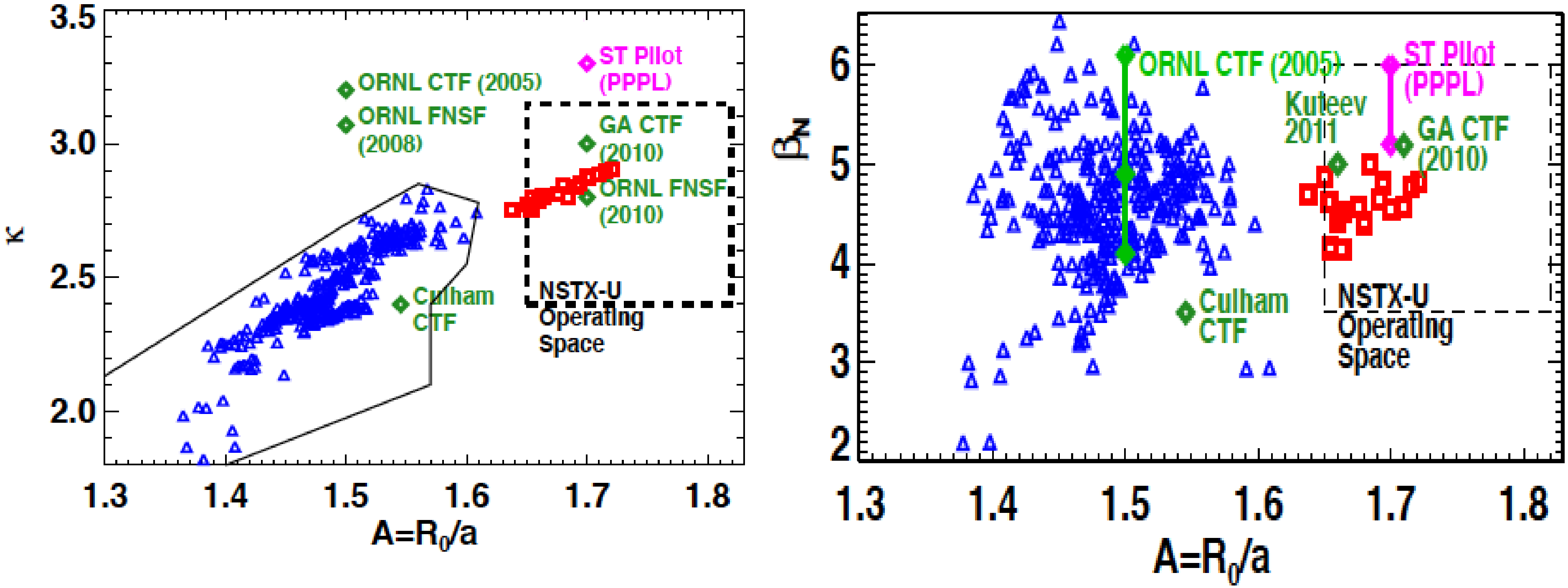
Fig. X2 Schematic of UTST. *Reprinted with permission from Y. Ono.*

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

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

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



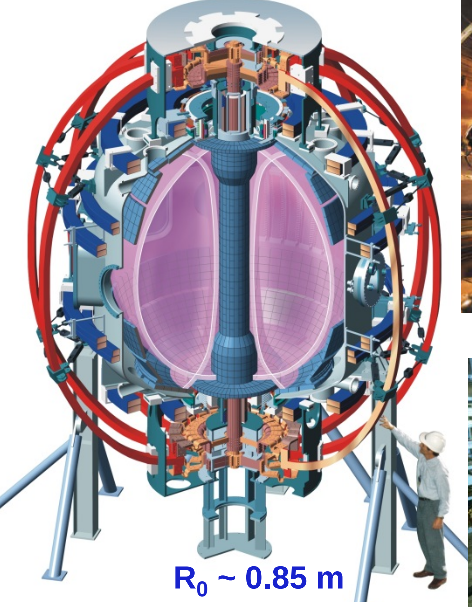
S.P. Gerhardt et al., NF (2011)

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)

Operating ST Research Facilities Since 2000

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

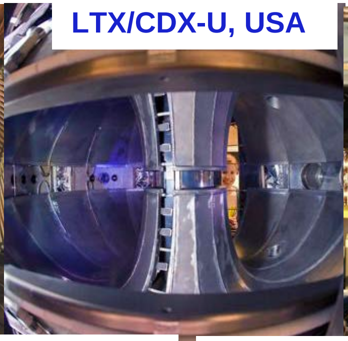
NSTX, USA



R_0 ~ 0.85 m

PEGASUS, USA

LTX/CDX-U, USA

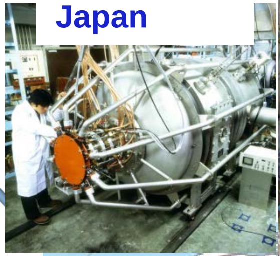


GLOBUS-M, Russia

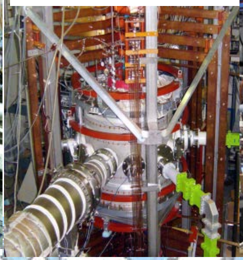
MAST, UK

R_0 ~ 0.75 m

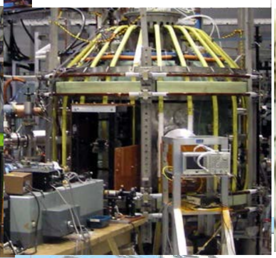
HIST, Japan



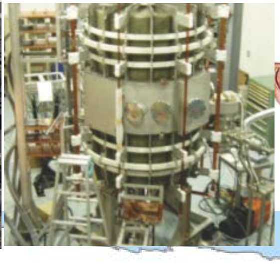
LATE, Japan



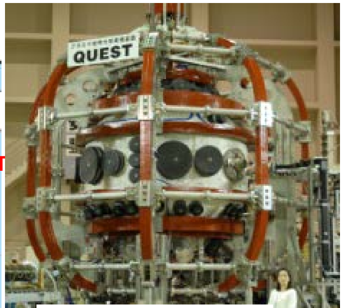
TST-2, Japan



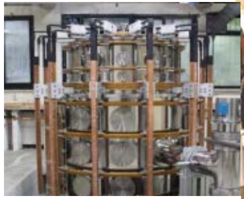
UTST, Japan



QUEST/CPD, Japan

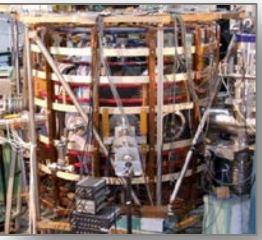


VEST, Korea



HIT-II, USA

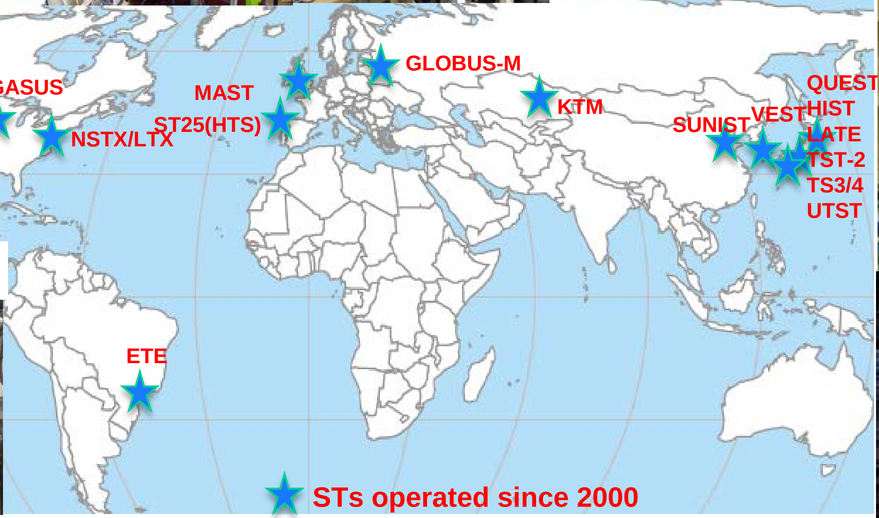
SUNIST, China



ETE, Brazil

ST25, UK

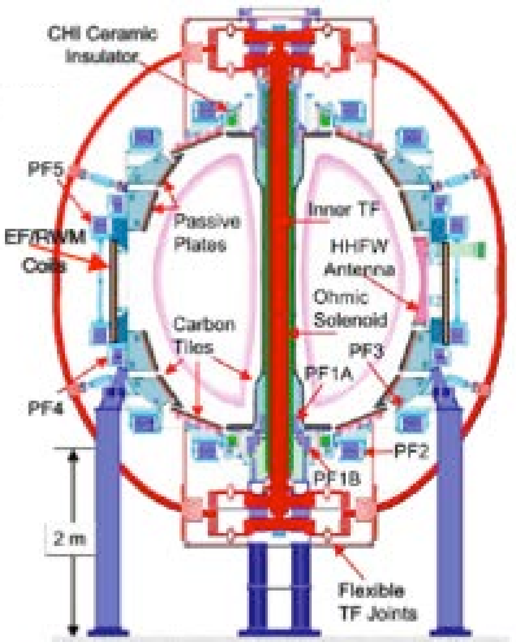
TS3/4, Japan

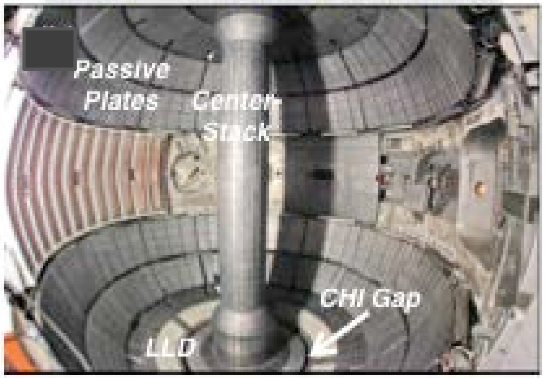


MA-Class ST Research Started in 2000

Complementary Physics Capabilities of NSTX and MAST

NSTX





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

Complementary Capabilities

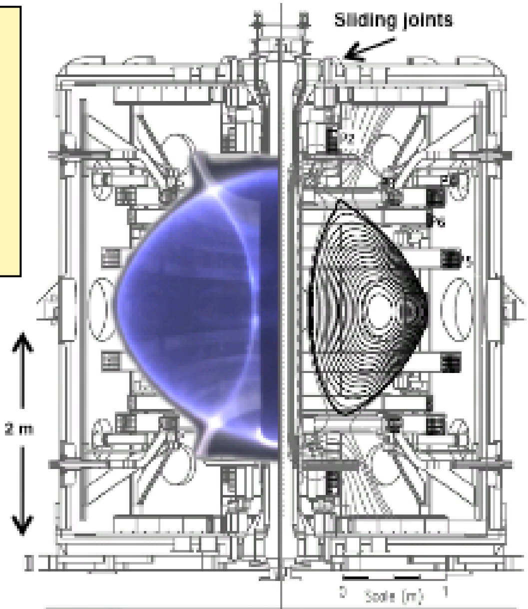
Passive Plates	Large divertor volume
Heilicity Injection	Merging/Compression
HHFW	ECH
1 x 6 RWM Coils	2 x 12 ELM coils

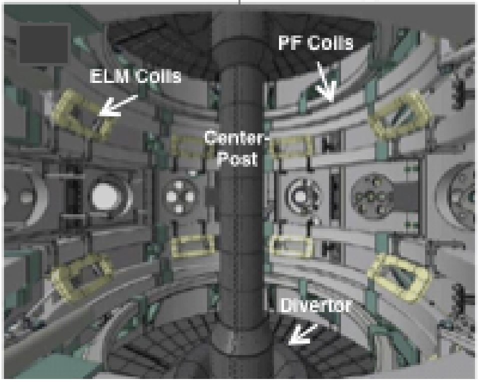
Similar Capabilities

NSTX	MAST
R = 85 cm	R = 80 cm
A ≥ 1.3	A ≥ 1.3
κ = 1.7 - 3.0	κ = 1.7 – 2.5
B_T = 5.5 kG	B_T ~ 5.0 kG
I_p ≤ 1.5 MA	I_p ≤ 1.5 MA
V_p ≤ 14 m^3	V_p ≤ 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