

The National Spherical Torus Experiment Upgrade (NSTX-U)

xxx, PPPL
For the NSTX-U Team

xxx
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Remember to change the title in the slide master



Outline

- Overview of fusion and plasmas
- Fusion plasma confinement in a tokamak
- Motivation for studying spherical tokamaks
- NSTX Upgrade scientific goals and questions

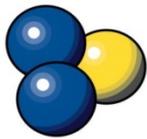
Advantages of fusion: safe, sustainable, high energy density, environmentally attractive

"D-T" fusion reaction:

D: deuterium n: neutron



Energy



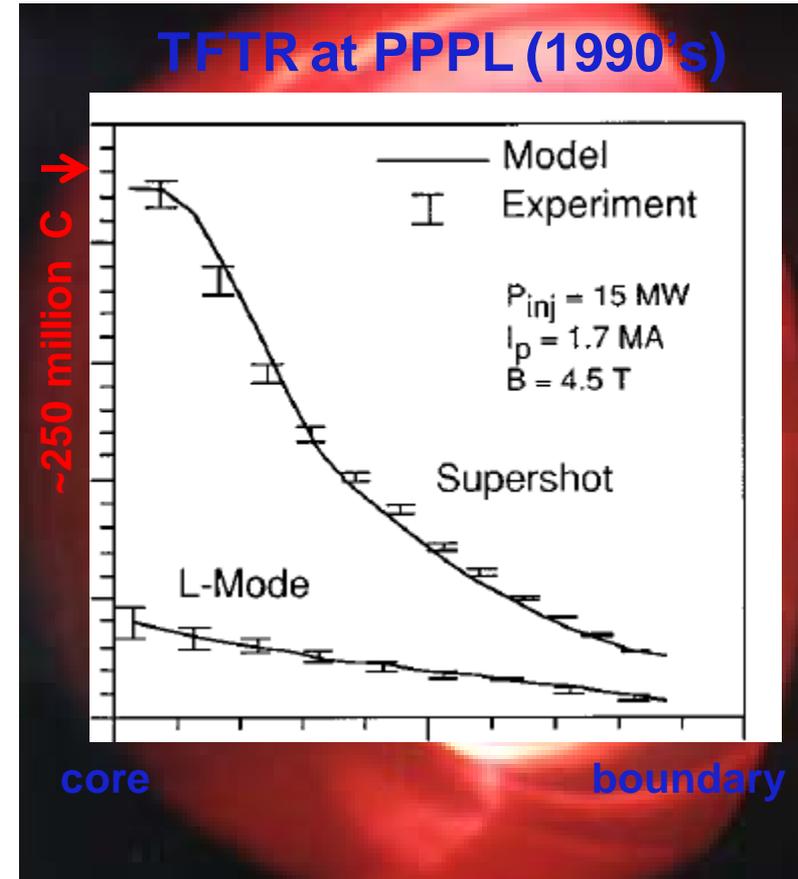
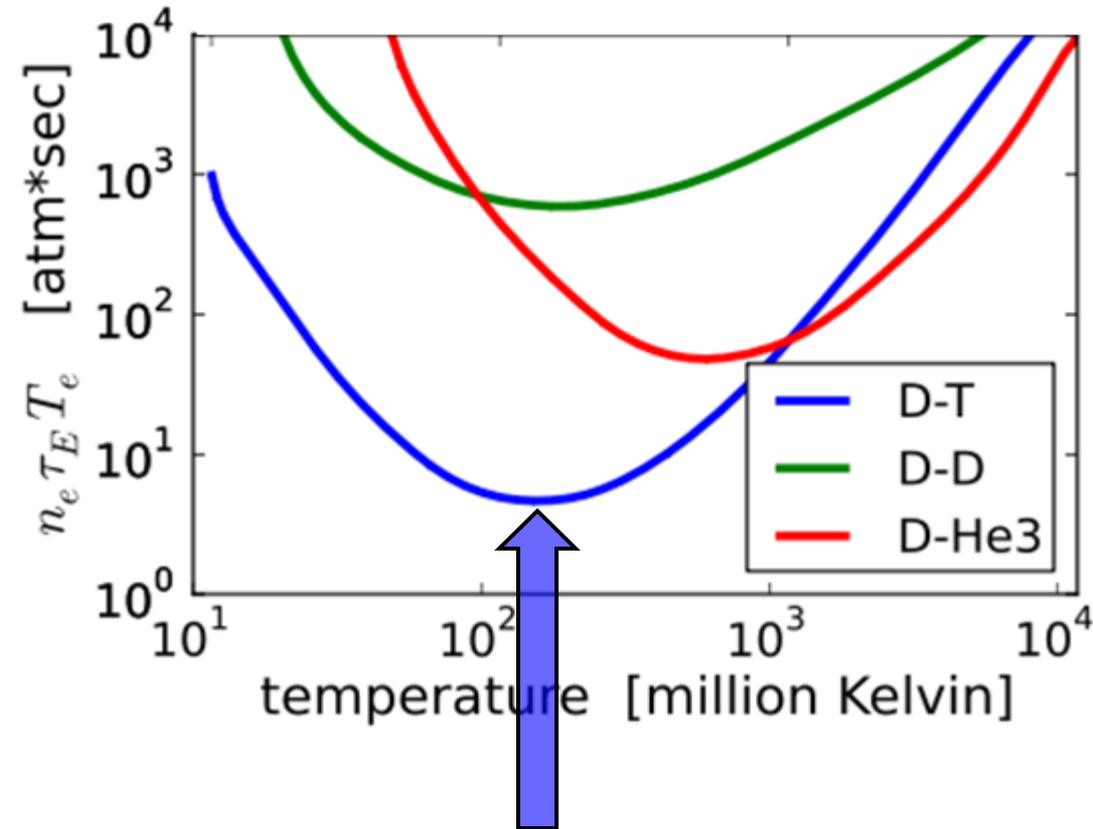
T: tritium

α : helium

High energy gain
 $\approx 1000 \times$

- **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 CO₂ production**

Fusion requires very high temperatures, but these have already been achieved

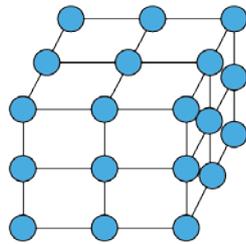


- Fusion is easiest here at **200 million °C (!!)** (350 million °F)
 - Minimum fusion “triple-product” (pressure nT and energy confinement time τ_E) value: 8 atmosphere-seconds

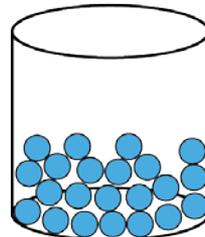
At fusion temperatures gas becomes plasma, a “soup” of electrons and positive ions

States of Matter

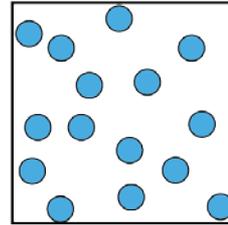
● = atom
⊕ = nucleus
⊖ = electron



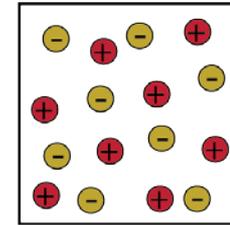
Solid



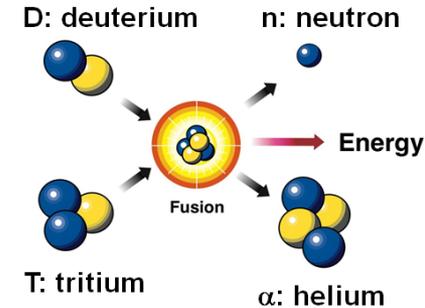
Liquid



Gas



Plasma



Water:

0 °C

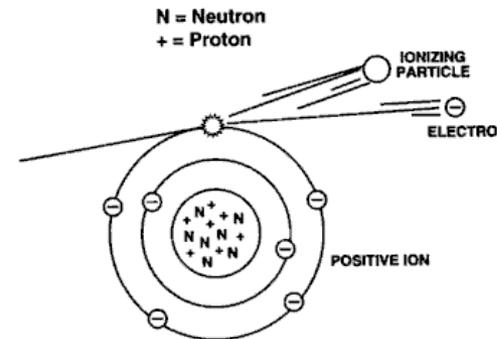
100 °C

160,000 °C

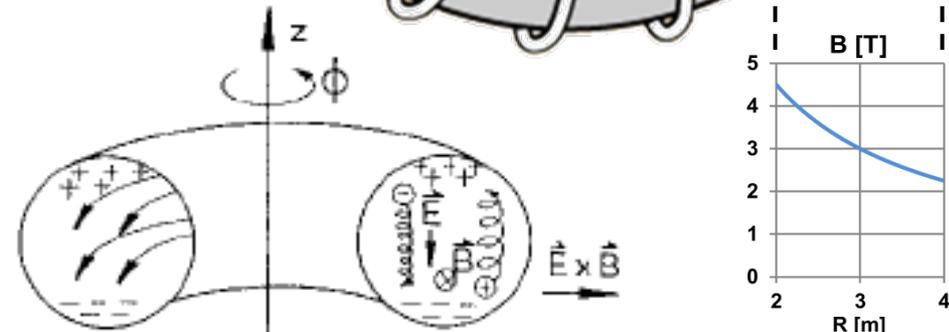
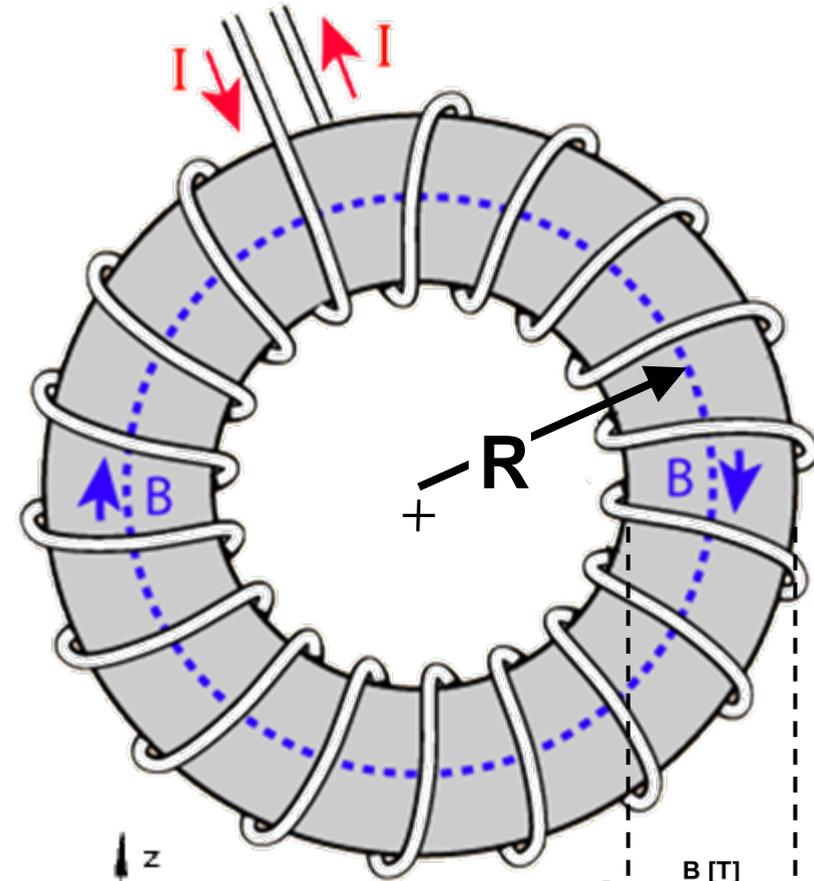
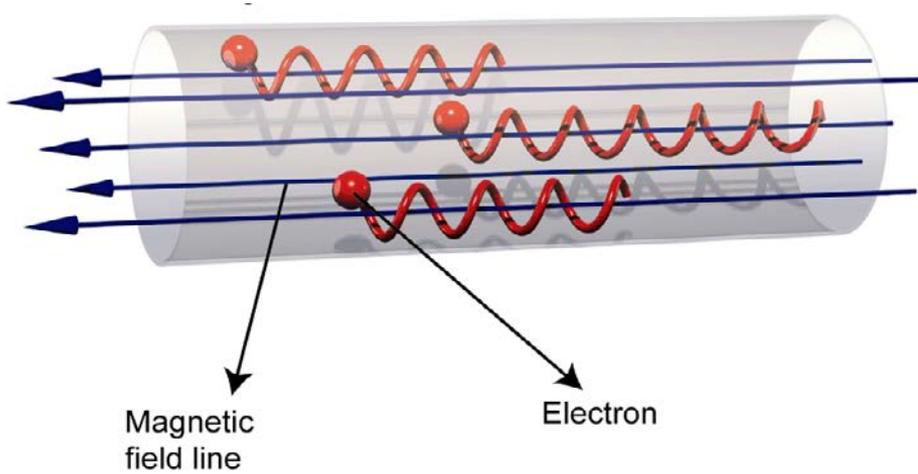
200,000,000 °C

DT fusion:

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

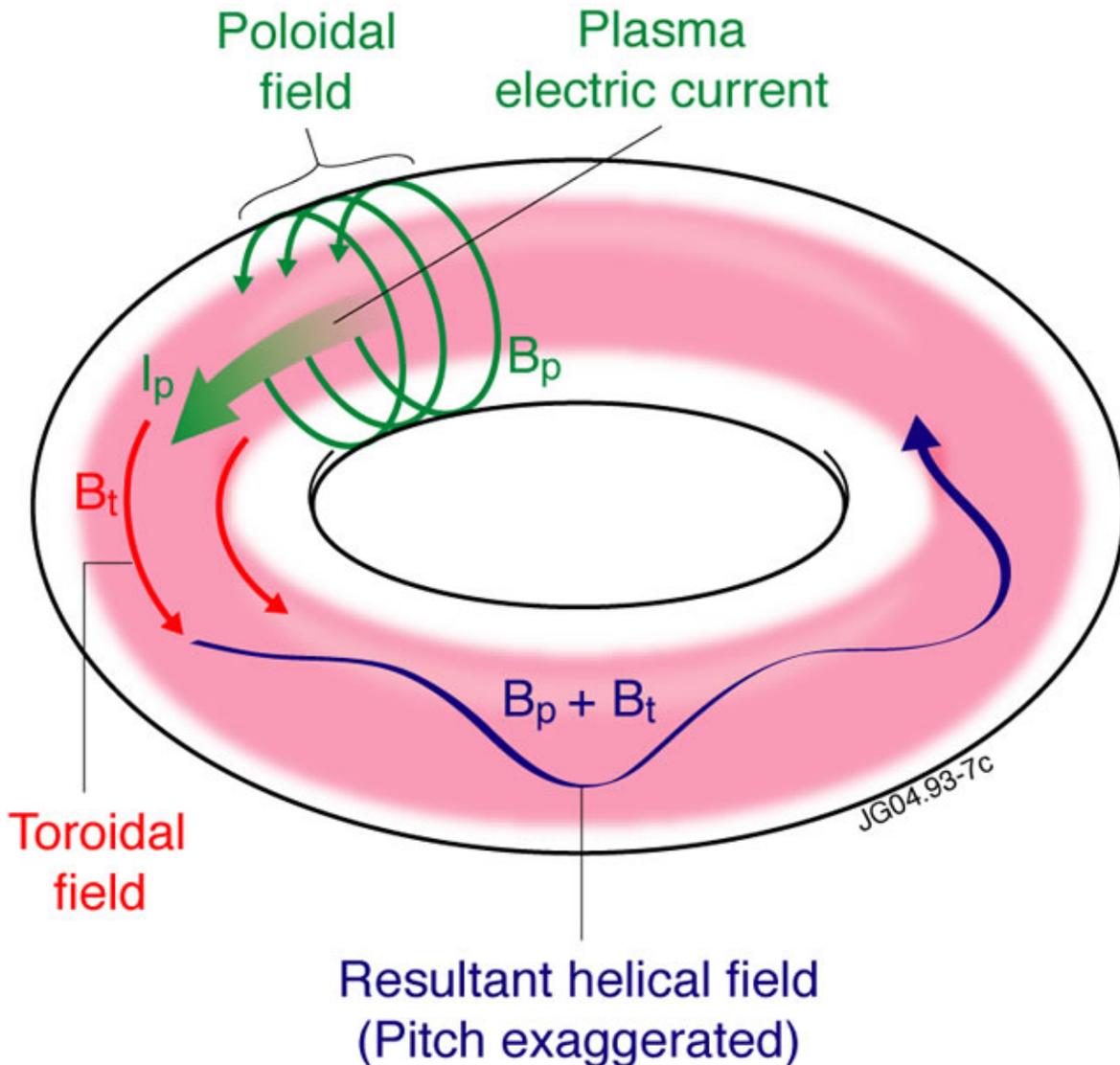


Charged particles confined by magnetic fields, but a simple toroidal field gives poor confinement



- Charged particles spiral around field lines, move freely along field
- Bending field into torus creates non-uniform $B \sim 1/R$
- Electrons and ions drift apart vertically \rightarrow charge separation
- Electric field created \rightarrow particles expelled outward

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



- Helical field = B-field covers toroidal surfaces
- Allows currents to flow along magnetic field
- Short-circuits electric fields that would otherwise expel plasma
- Particles tied to surface
- Improved confinement

“Tokamak”: Russian acronym for “toroidal chamber with magnetic coils”

Tokamaks are not presently steady-state, but in the future will operate without transformer

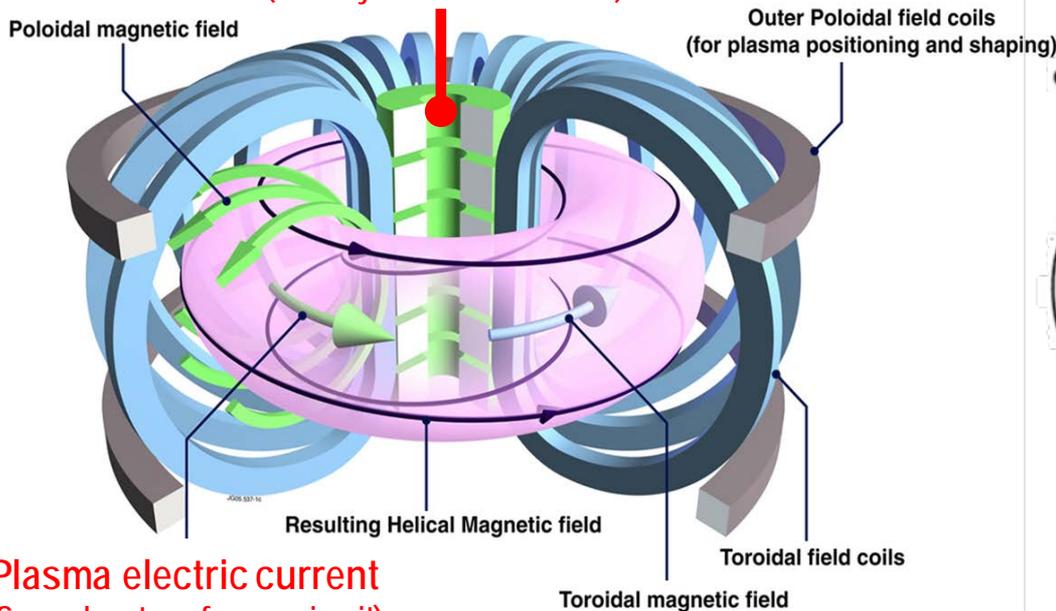
Conventional Tokamak

- Primary transformer coils drive toroidal current in plasma
- Must stop when transformer coils reach current/force limits

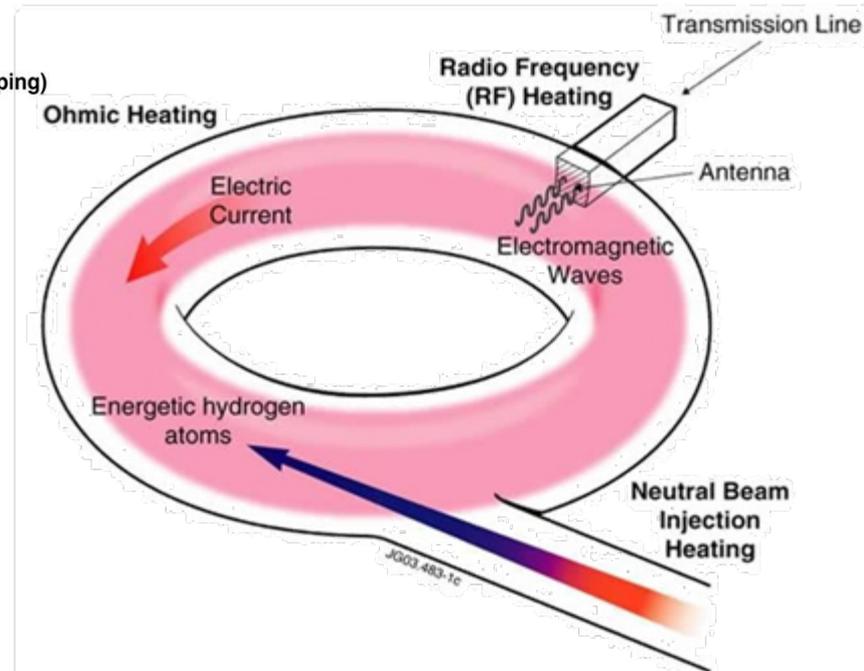
Advanced Tokamak

- 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

Inner Poloidal Field Coils (Primary transformer circuit)



Plasma electric current
(Secondary transformer circuit)



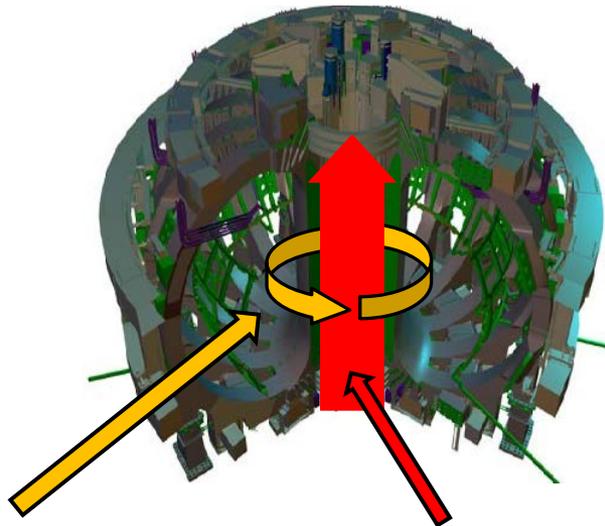
ITER will be first device to access “burning plasma”, is very large and expensive

ITER under construction in Cadarache, France



- 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 $Q = P_{\text{fusion}} / P_{\text{ext}} = 10 \rightarrow P_{\alpha} / P_{\text{ext}} = 2$
- $P_{\alpha} / P_{\alpha + \text{ext}} = 2/3$

- $A = 3.1$
- $R = 6.2\text{m}$
- $B_T = 5.3\text{T}$
- $I_p = 15\text{MA}$
- 6500 tons
- 80,000 km of Nb3Sn super-conducting strand



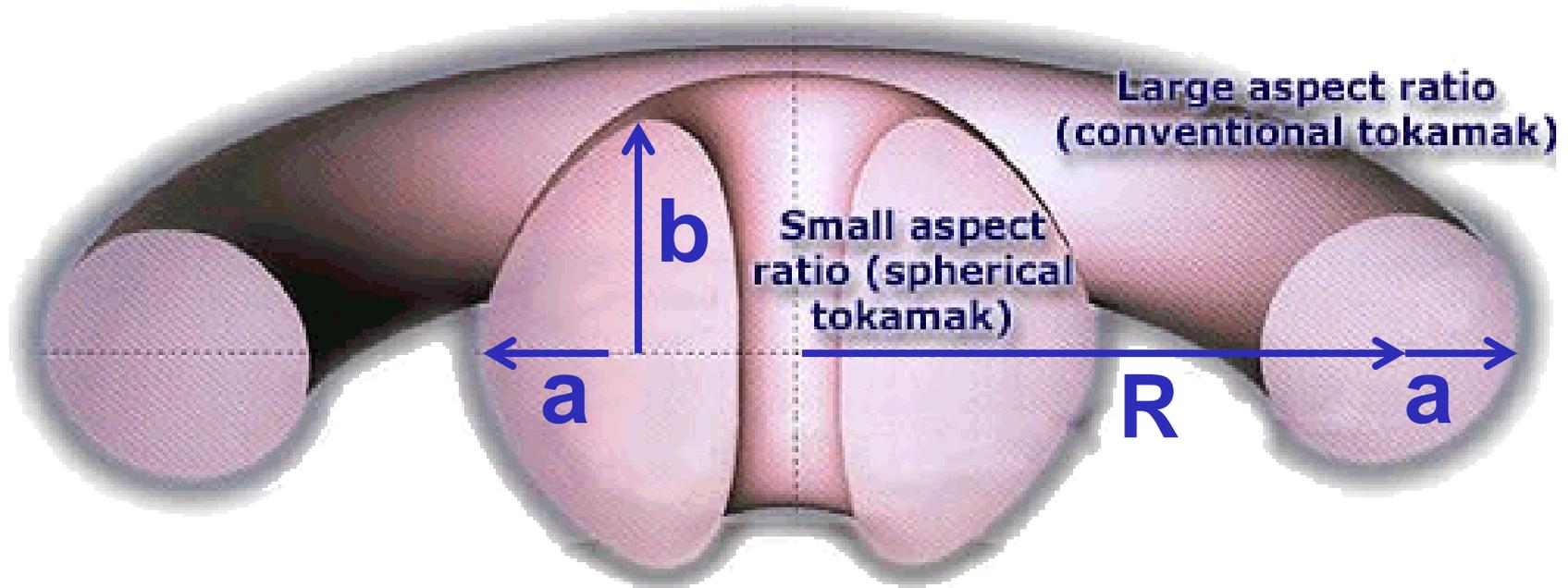
- Size of ITER driven largely by energy confinement which scales with plasma current, and therefore with size
- Can we make smaller devices with better confinement and smaller or cheaper magnets?

Plasma current: 15 MA Toroidal field current: 165 MA

Aspect ratio and elongation are important free parameters

Aspect ratio $A = R / a$, Elongation $\kappa = b / a$

$R =$ major radius, $a =$ minor radius, $b =$ vertical $\frac{1}{2}$ height



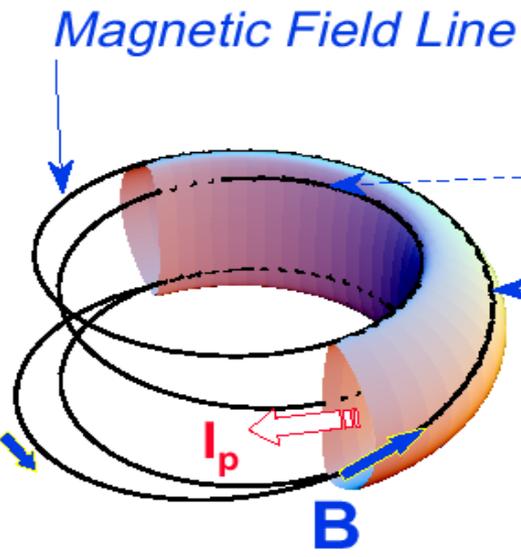
Spherical torus/tokamak (ST) has $A = 1.1-2$

Conventional tokamak typically $A = 3-4$

Higher elongation improves stability, confinement

ST can be compact, high beta, and high confinement and favorable average curvature improves stability

Tokamak

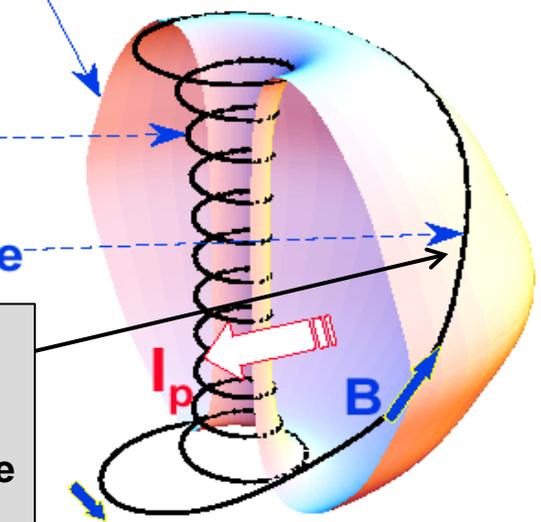


A ~ 3
 $\kappa = 1.5-2$
 $\beta_T = 3-10\%$

Toroidal beta $\beta_T = \langle p \rangle / (B_{T0}^2 / 2\mu_0)$
 $\beta_T [\%] \equiv \beta_N I_P / (aB_{T0})$

ST

Magnetic Surface



A ~ 1.5
 $\kappa = 2-3$
 $\beta_T = 10-40\%$

Higher elongation κ and low A lead to higher I_P , β_T and τ_E

- **ST has high I_P , achieves it cost effectively**



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

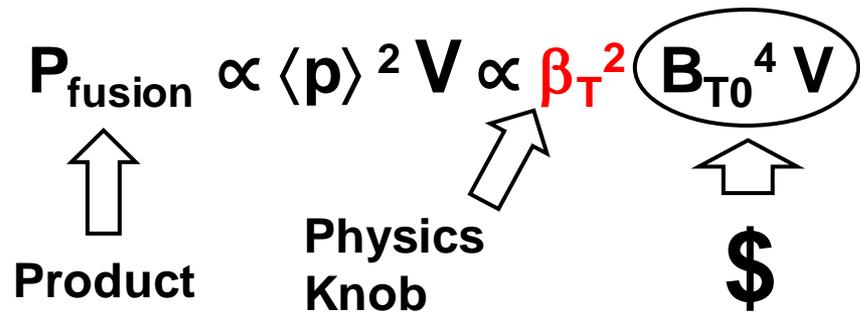
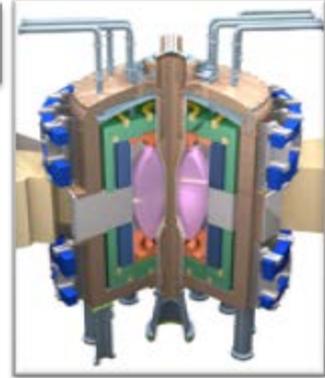
- **I_P increases performance**
- Energy confinement time $\tau_E \propto I_P$**

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

Fusion Nuclear Science Facility (FNSF) could aid technology development

- Plasma facing components for exhaust, vacuum vessel and shield, tritium fuel cycle...

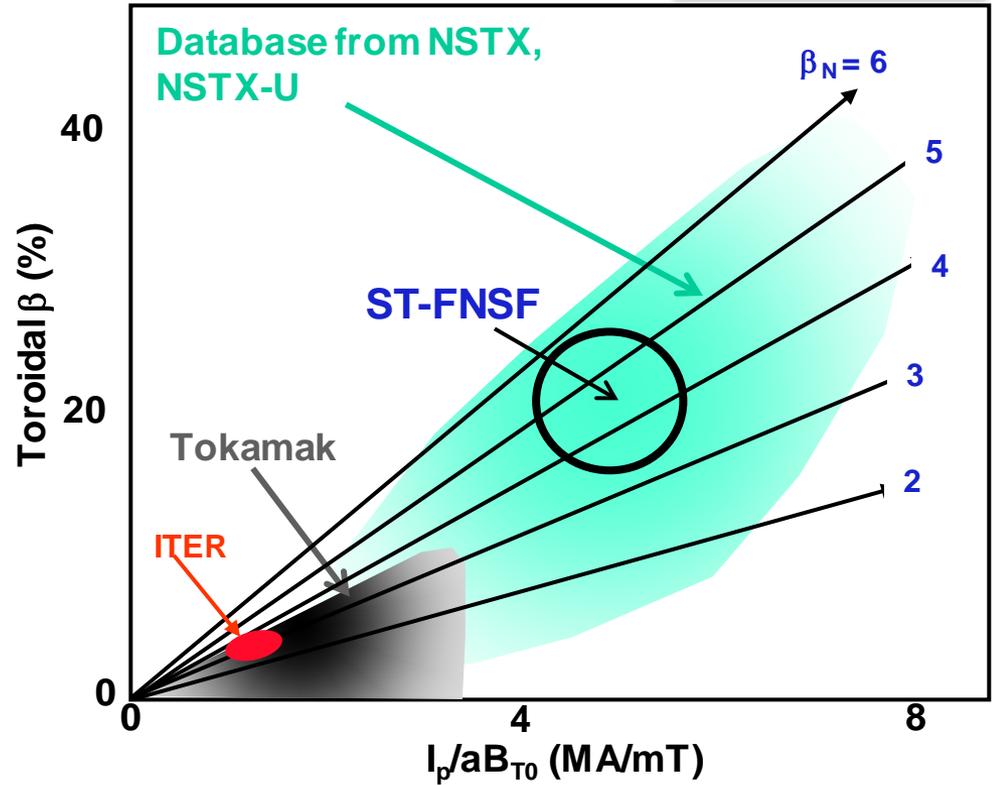
ST-FNSF



High neutron wall loading W_n possible in a compact ST

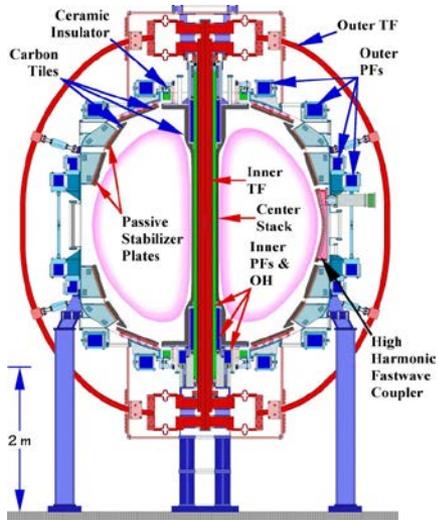
$W_n \propto P_{\text{fusion}} / A \propto \beta_T^2 B_{T0}^4 a$
 (not strongly size dependent)

$W_n \sim 1 \text{ MW/m}^2$ with $R \sim 1 \text{ m}$
FNSF feasible!



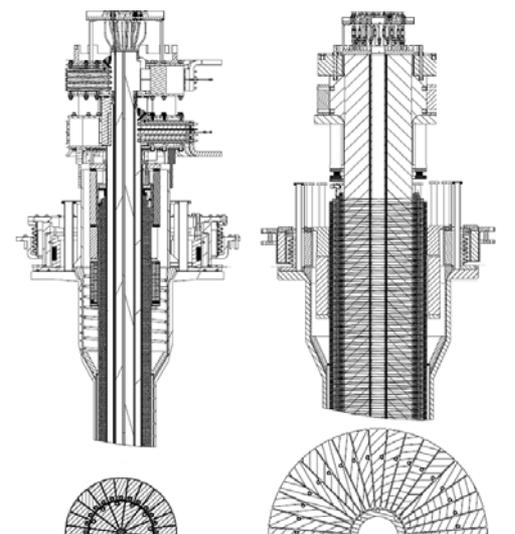
The National Spherical Torus Experiment (NSTX)

- Located at the Princeton Plasma Physics Laboratory in Princeton, NJ
- Operated from 1999-2010
- Major accomplishments:

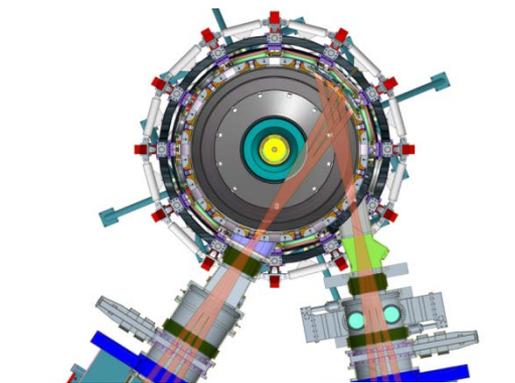


NSTX-U will provide new data to support ST-FNSF design, ITER operations, boundary solutions

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



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



Present NBI **New 2nd NBI**

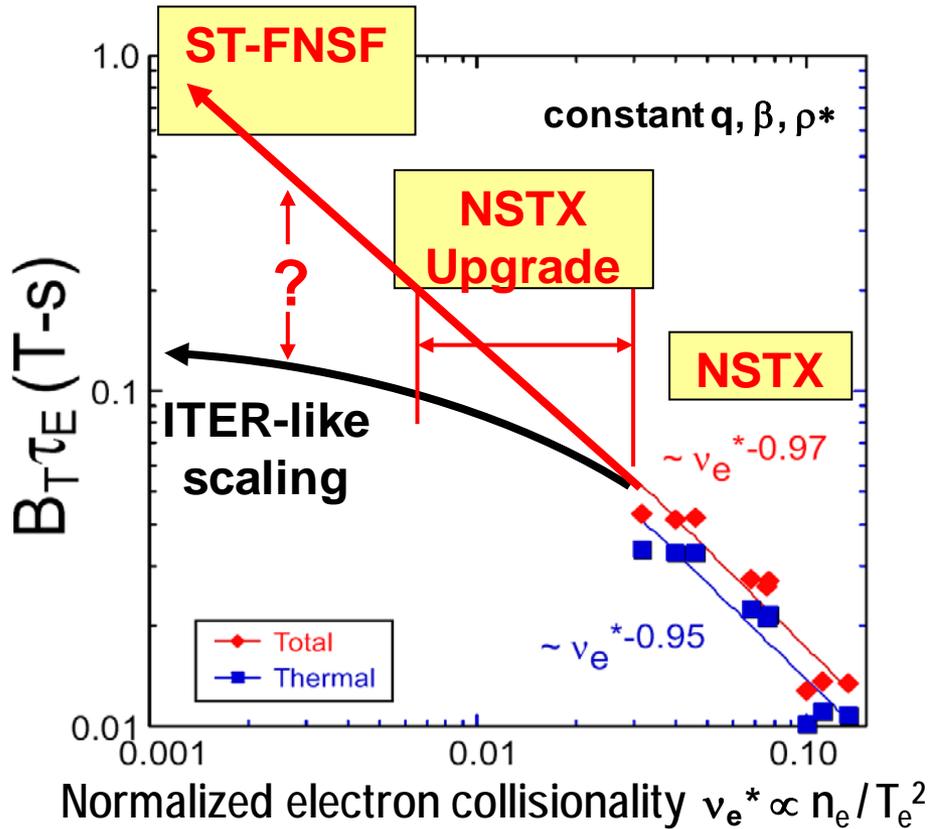
- **New center-stack doubles toroidal field, plasma current, 5 × pulse-length**
- **Tangential injection provides 3-4 × higher current drive at low I_p :**
 - **2x higher absorption (40 → 80%) at low I_p**
 - **1.5-2x higher current drive efficiency**

~ 5-10x increase in $nT\tau$ from NSTX
NSTX-U average plasma pressure ~ tokamaks

NSTX-U mission elements:

- **Advance ST as candidate for FNSF**
- **Develop solutions for the plasma-material interface challenge**
- **Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond**
- **Develop ST as fusion energy system**

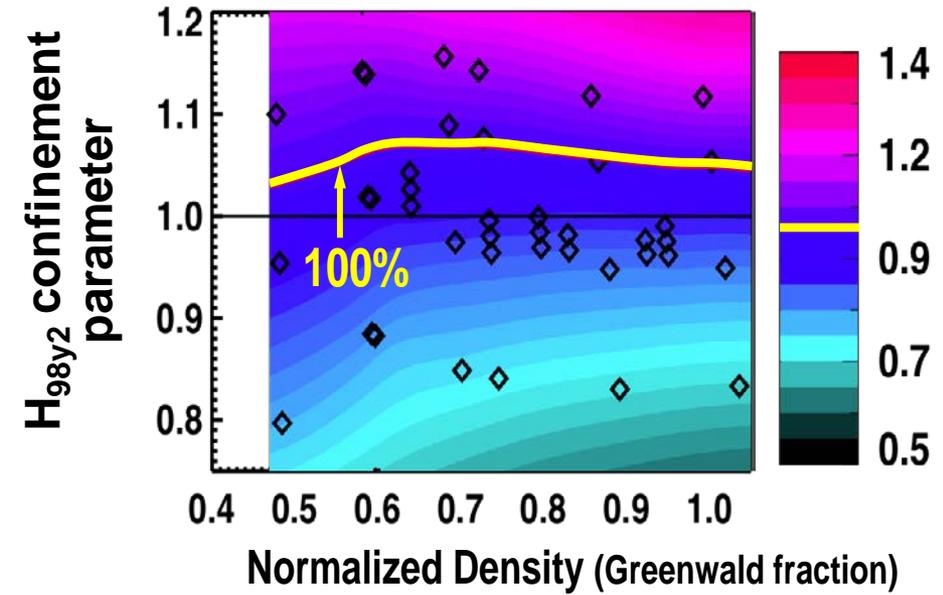
NSTX achieved favorable confinement and current drive results; Will trends continue in NSTX-U?



NSTX observed confinement increase at higher T_e !
Will trend continue, or look like conventional A?

Favorable confinement results could lead to more compact ST reactors

TRANSP Contours of Non-Inductive Fraction



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

NSTX achieved 70% “transformer-less” current drive
Will NSTX-U achieve 100% as predicted?

Steady-state operation required for FNSF

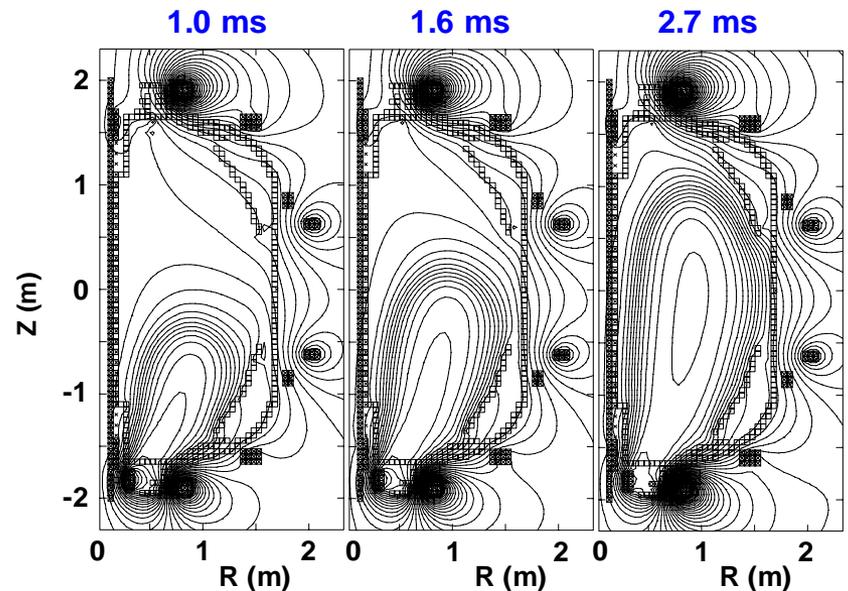
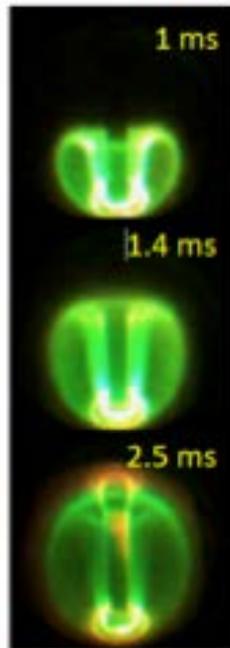
I_p Start-up/Ramp-up Critical Issue for ST-FNSF, NSTX achieved 200kA (~20%) “transformer-less” start-up

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

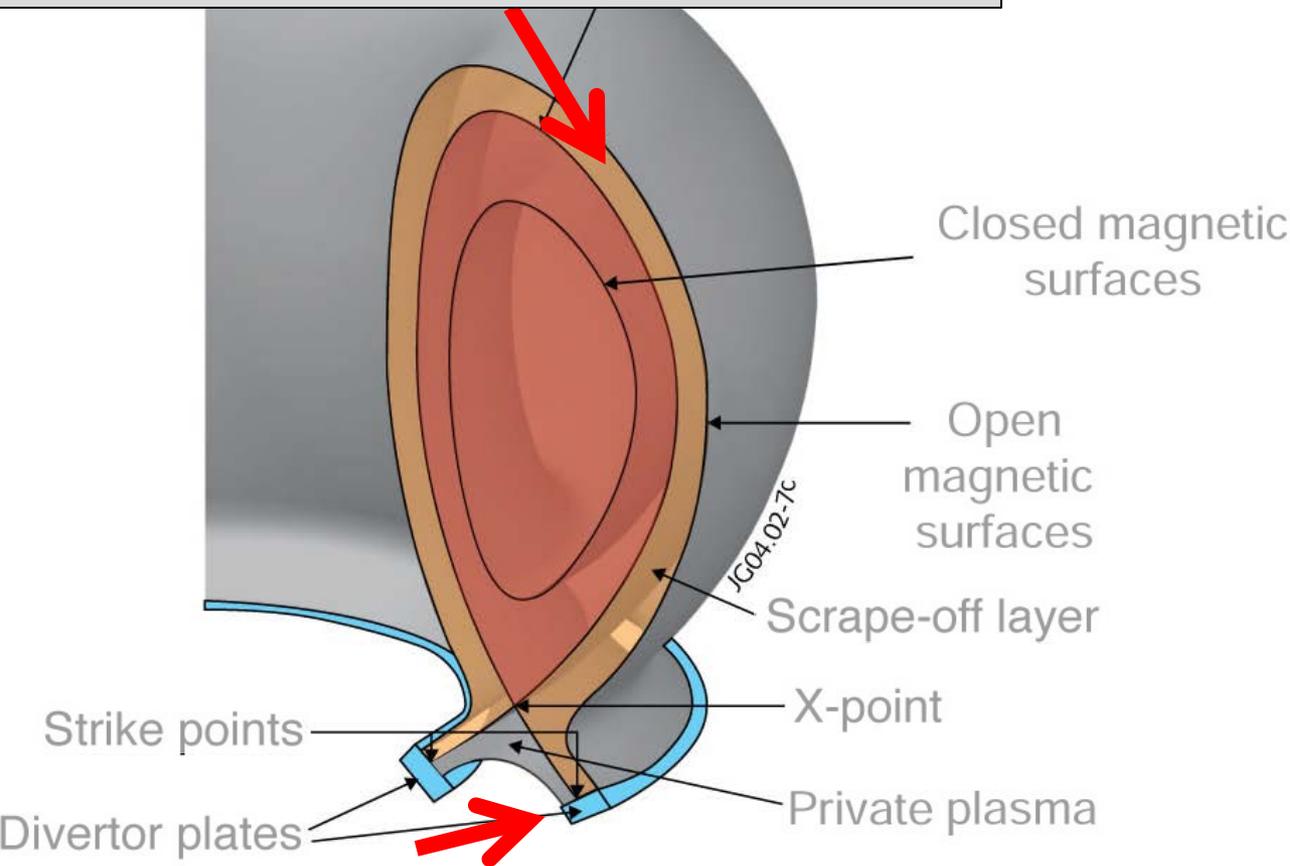
Long-term major goal of NSTX-U: generate and sustain a high-performance plasma **without using any transformer**



TSC code successfully simulated helicity injection $I_p \sim 200\text{kA}$ in NSTX
Will NSTX-U achieve 400kA or more as per simulations?

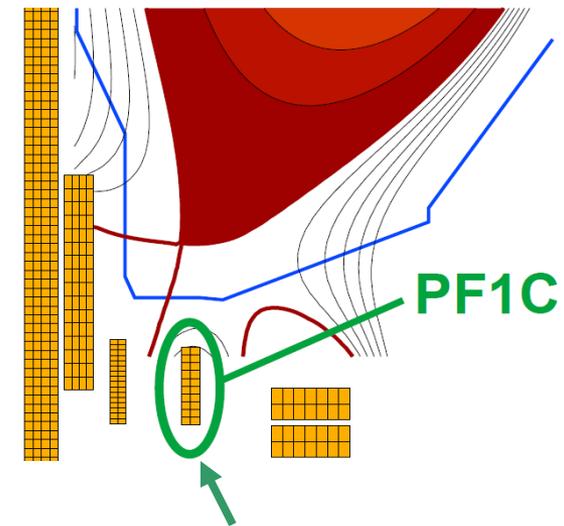
NSTX-U will test ability of advanced divertors to mitigate very high heat-fluxes

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



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

Additional null-point in "snowflake" divertor expands field, reduces heat flux



NSTX-U has additional coils improved control of snowflake

Power / particles contact target plates here

Your more specific slides here

Conclusions

- NSTX-U is advancing the spherical torus as a candidate for a Fusion Nuclear Science Facility

Opportunities to participate in NSTX-U research

- nstx-u.pppl.gov

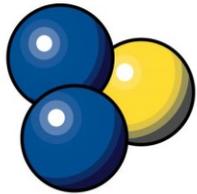
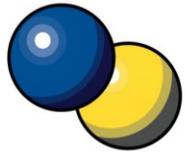
Back-up slides

What is fusion?

"D-T" fusion reaction:

D: deuterium

n: neutron



Energy



T: tritium

α : helium

**High energy gain
 $\approx 1000 \times$**

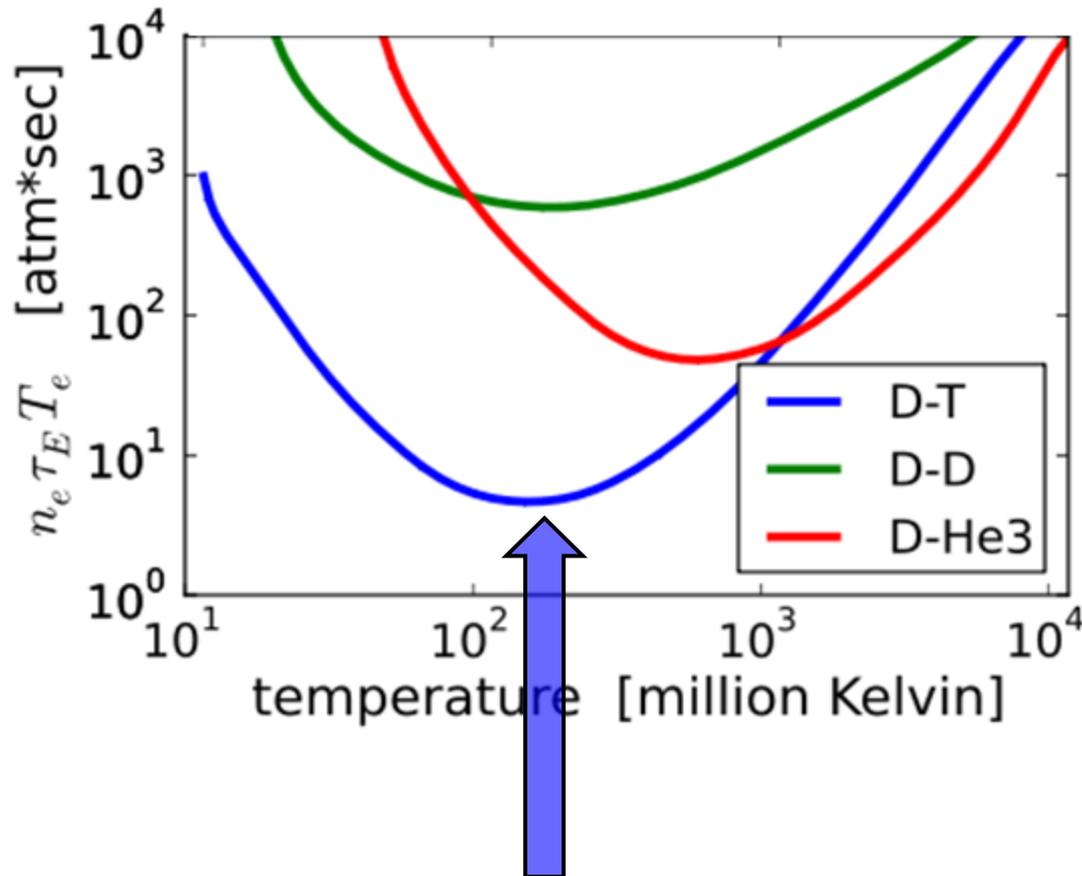
$$E = mc^2$$

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 CO₂ production**

Fusion requires very high temperatures

Fusion difficulty
(pressure \times confinement)



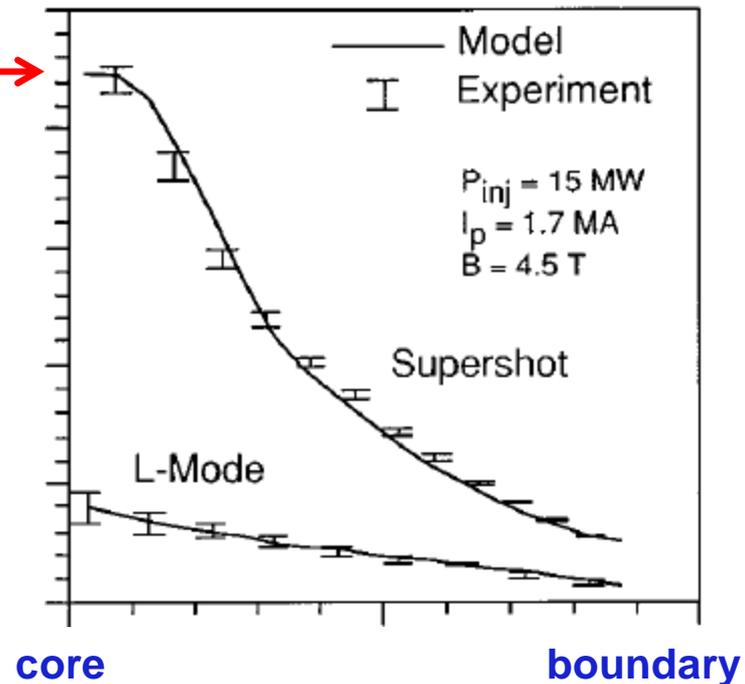
- Fusion is easiest here at **200 million °C (!)** (350 million °F)
 - Requires lowest pressure nT and energy confinement time τ_E
 - Minimum fusion “triple-product” value: 8 atmosphere-seconds

Magnetic fusion has already achieved the necessary very high temperatures!



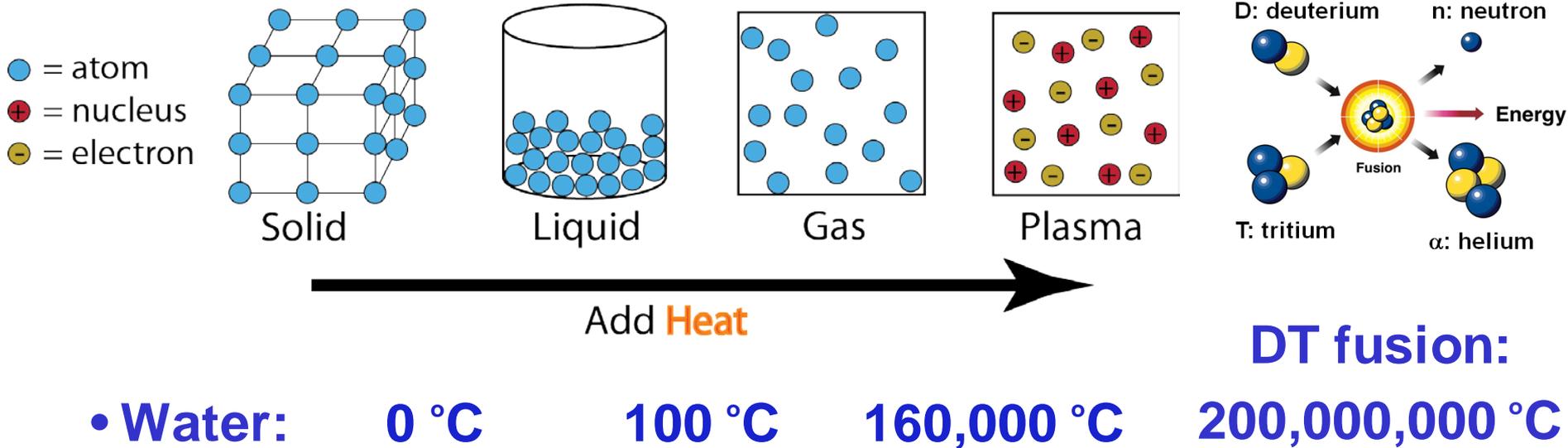
~250 million C →

TFTR at PPPL (1990's)



Gas becomes plasma at fusion temperatures

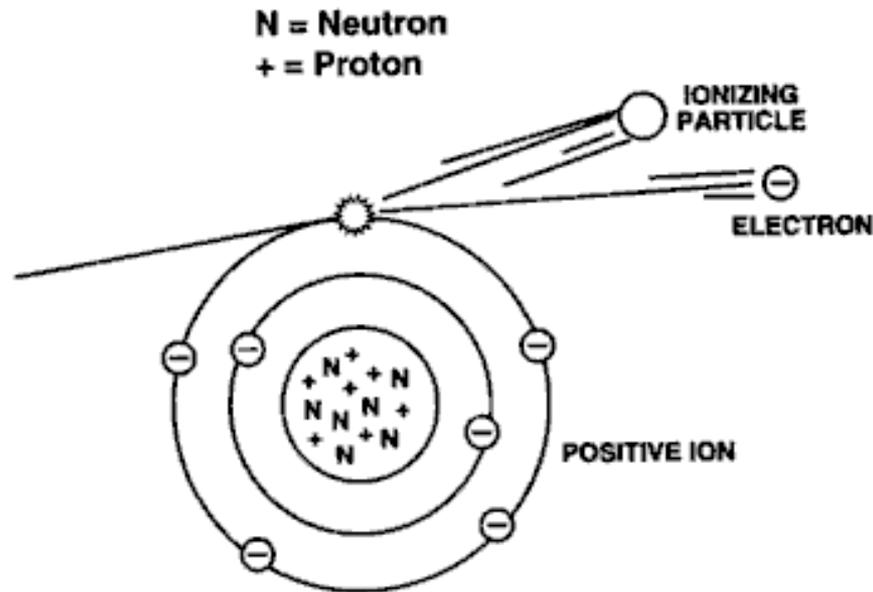
States of Matter



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



- **One benefit of plasma state: 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

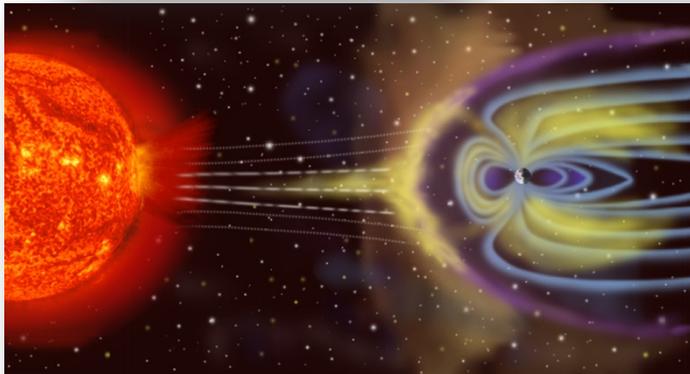
Plasma processing (semiconductors)



Neon signs



Lightening

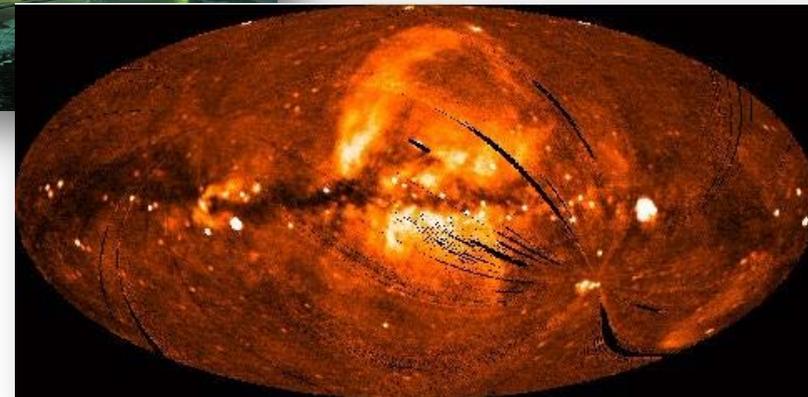


Solar wind



Aurora Borealis

Interstellar medium

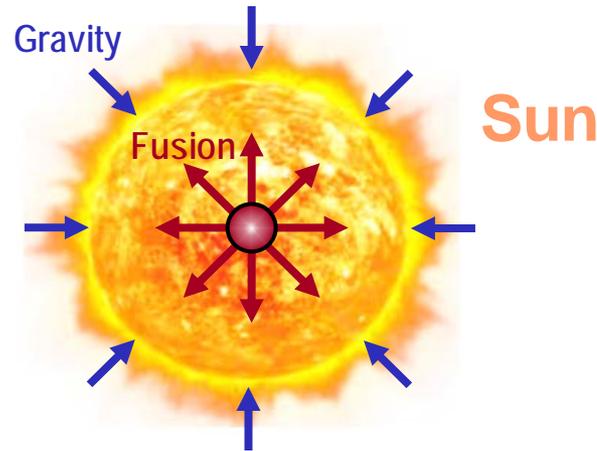


A few examples of plasma

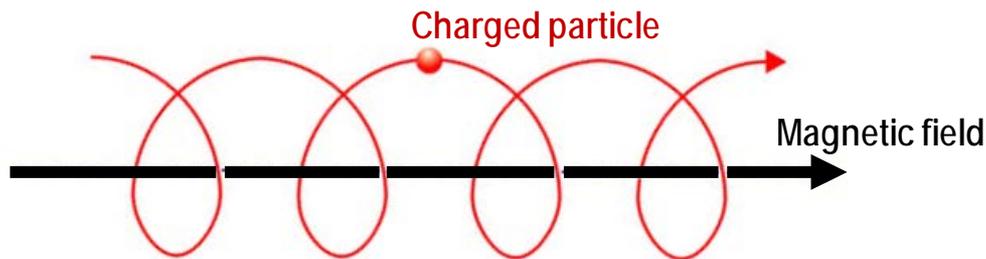
How do we confine plasma?

Confinement methods for controlled fusion

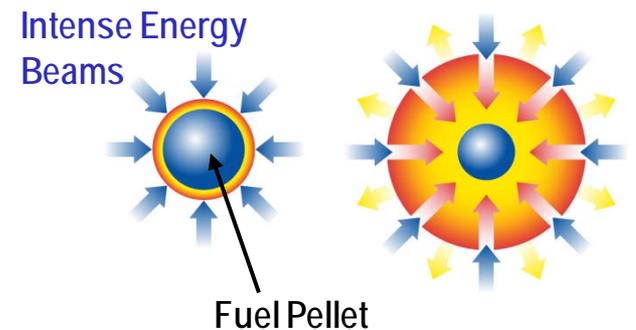
Gravitational Confinement



Magnetic Confinement

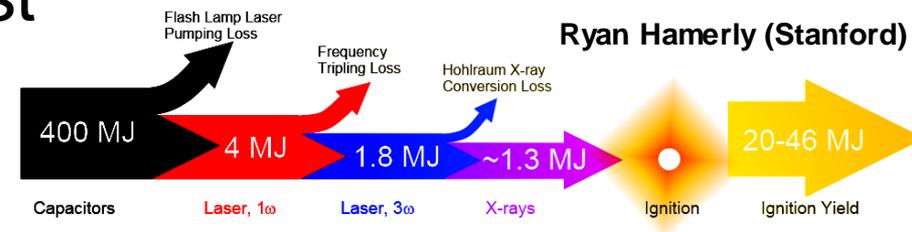
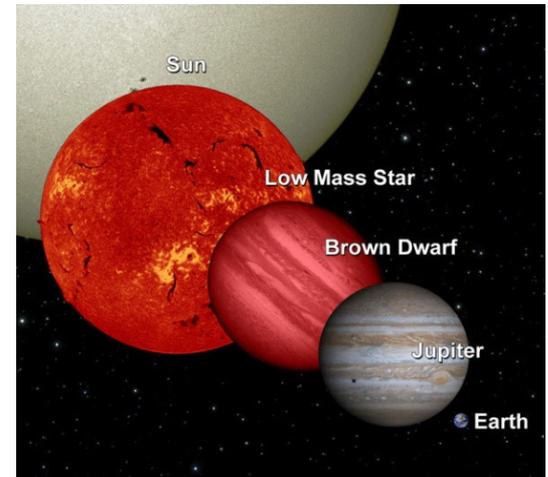


Inertial Confinement

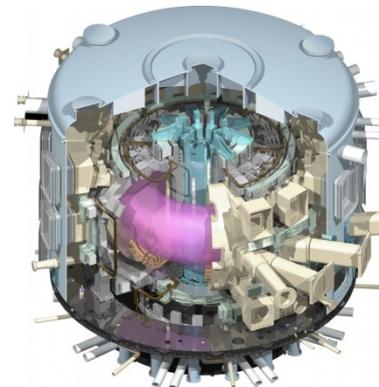


Magnetic fusion is arguably closest to ultimate goal of electricity generation

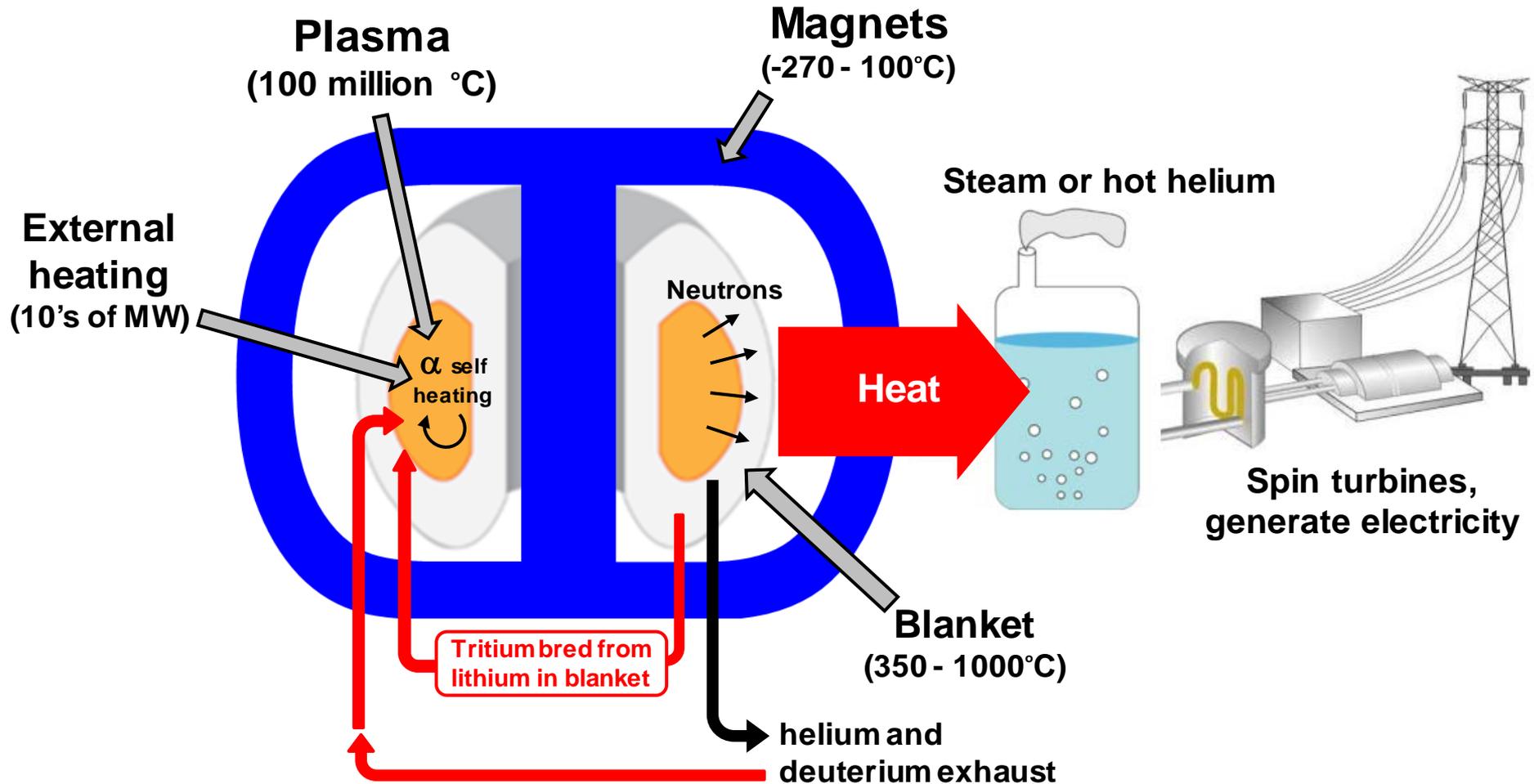
- Gravitational confinement fusion requires large device
 - Need 7-8% of mass of our sun
 - Approximately 10x diameter of Earth
- Laser fusion ala NIF at best has $E_{\text{fusion}} / E_{\text{electrical}} \sim 5\%$
 - So far, 0.004% efficient
- Magnetic fusion in ITER:
 - Goal: 500MW fusion power for $\leq 600\text{MW}$ electrical input for 400s
 - Industrial levels of fusion power



14kJ fusion yield achieved

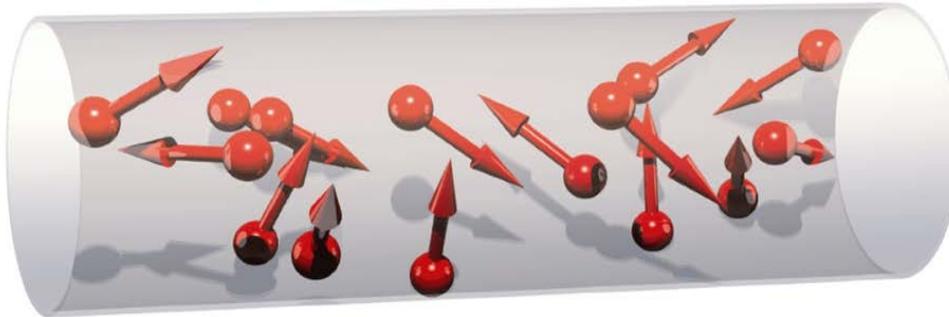


How would magnetic fusion make electricity?



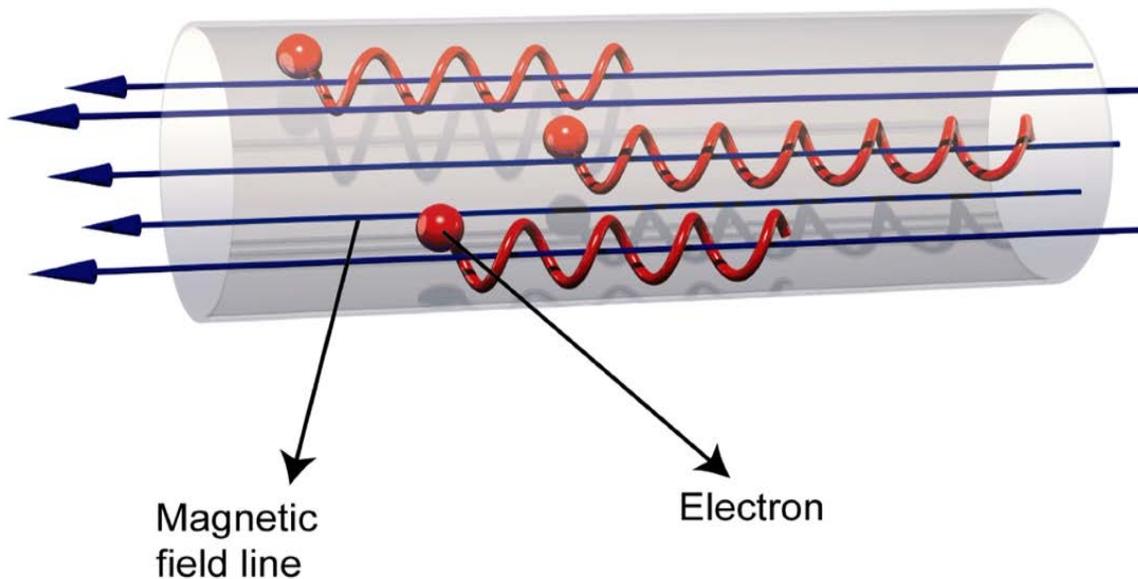
Charged particles confined by magnetic fields

No magnetic field



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

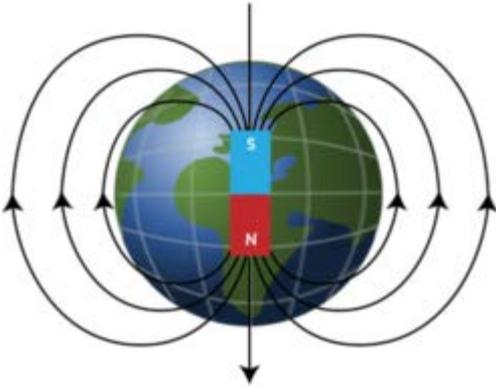
With magnetic field



- 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]

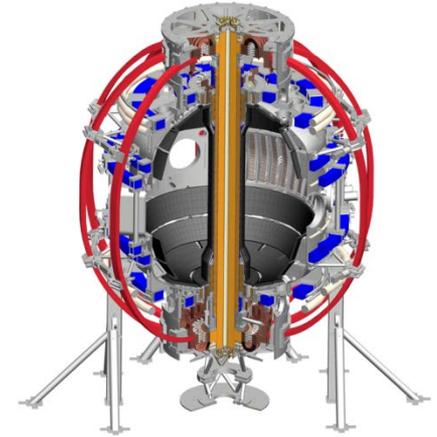
Earth: $\sim 5 \times 10^{-5}$ T



Refrigerator magnets
 $\sim 1-5 \times 10^{-3}$ T



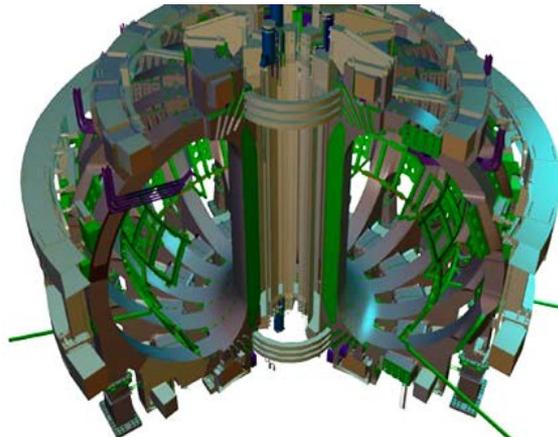
PPPL's NSTX-U: 1 T



MRI: 0.5-3 T

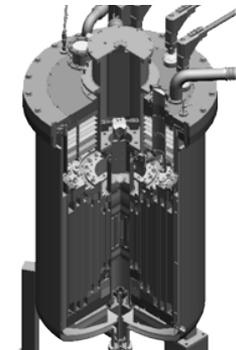


ITER: 5.3 T



World Record: 100 T

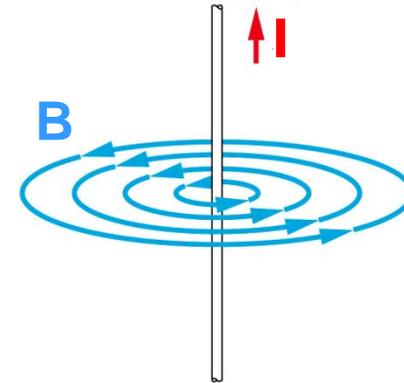
Non-destructive - for few milliseconds



National High Magnetic Field Laboratory

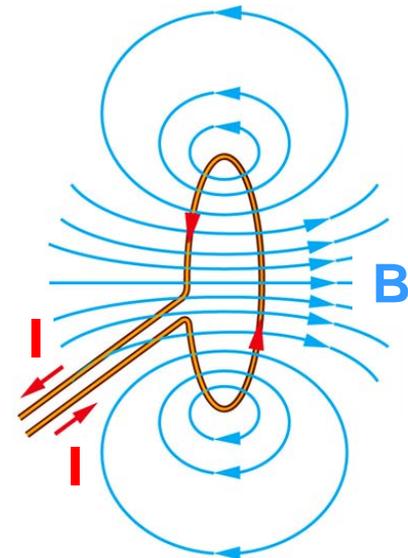
Magnetic fusion uses electromagnets

- Electromagnet: use **electrical current “I”** to create **magnetic field “B”**

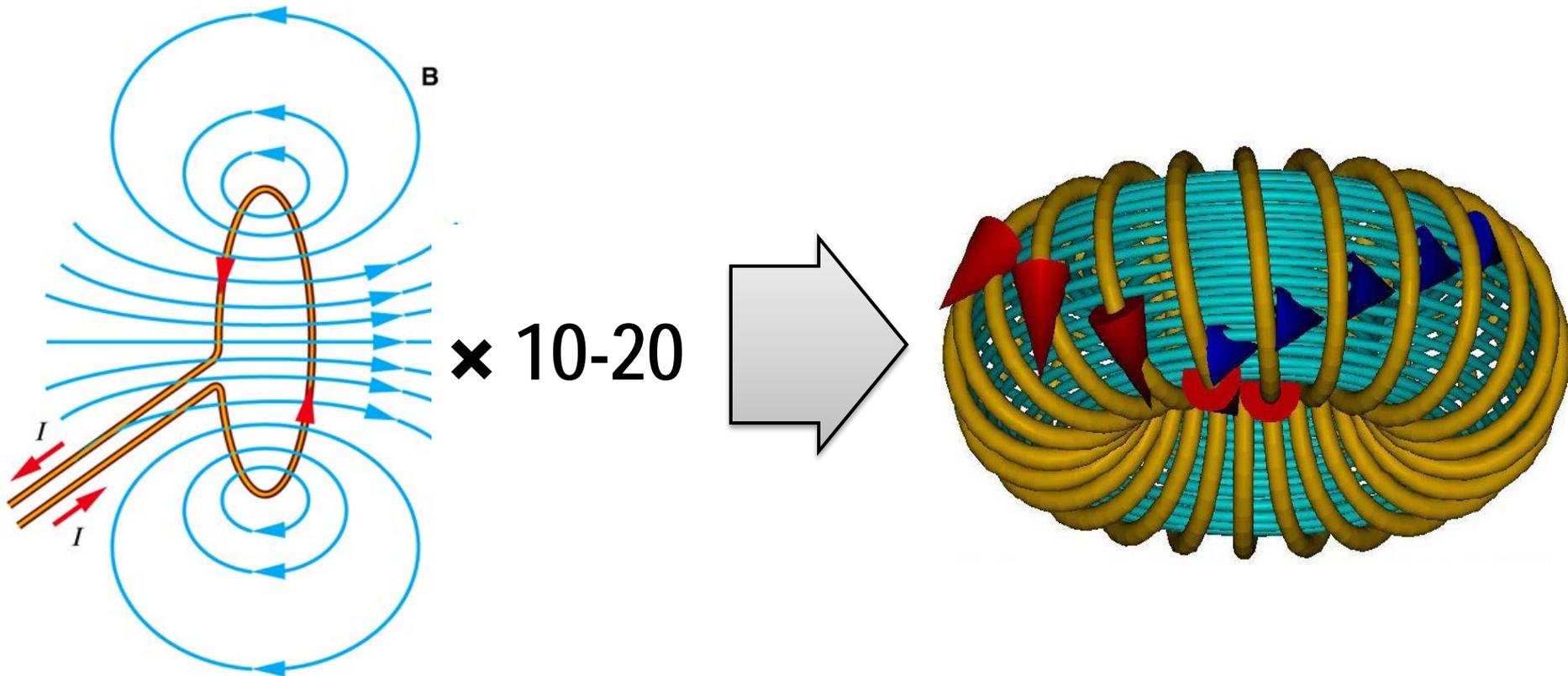


- “B” is perpendicular to “I”

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

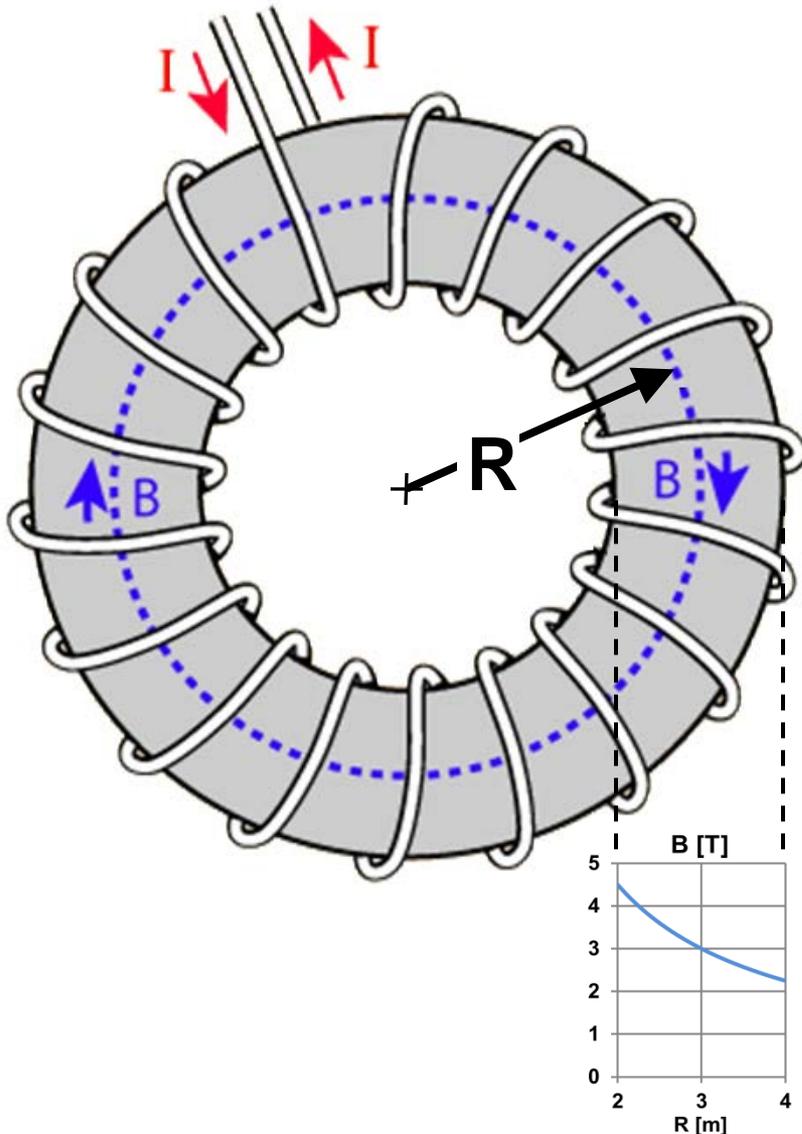


Fusion: Use multiple coils to create “toroidal” field

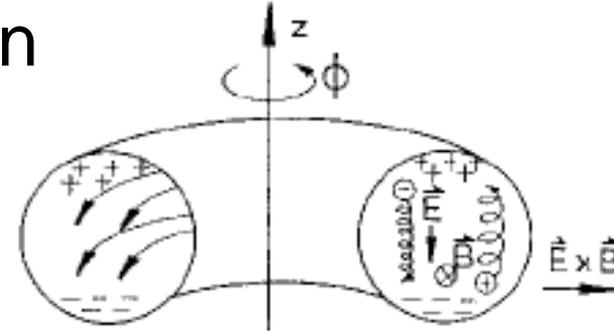


- Position many coils into donut or “torus” shape
- **Magnetic field never touches wall (!)**
- Do charged particles ever touch wall?

Simple toroidal field \rightarrow poor confinement

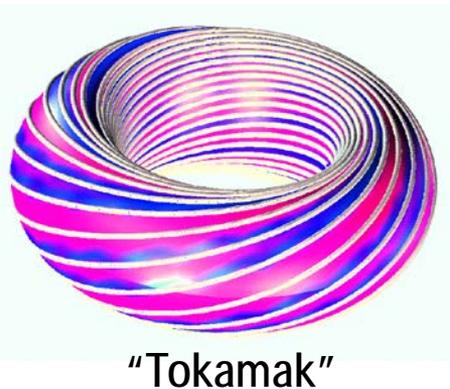
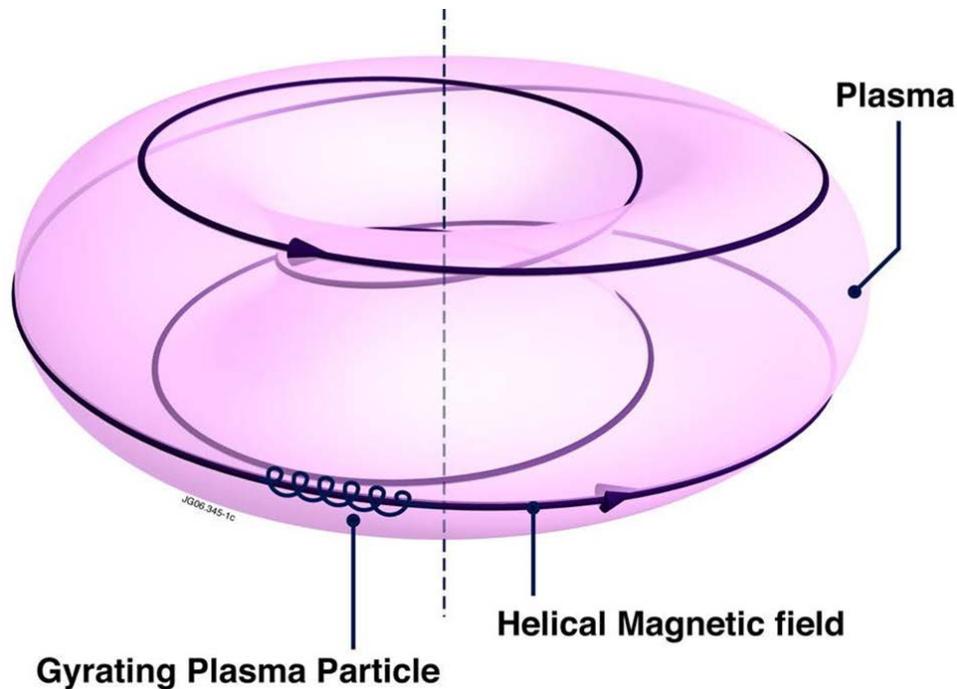


- Bending field into torus creates non-uniform B
 - $\triangleright B \sim 1 / R \rightarrow$ factor of 2
- Electrons and ions drift apart vertically \rightarrow charge separation



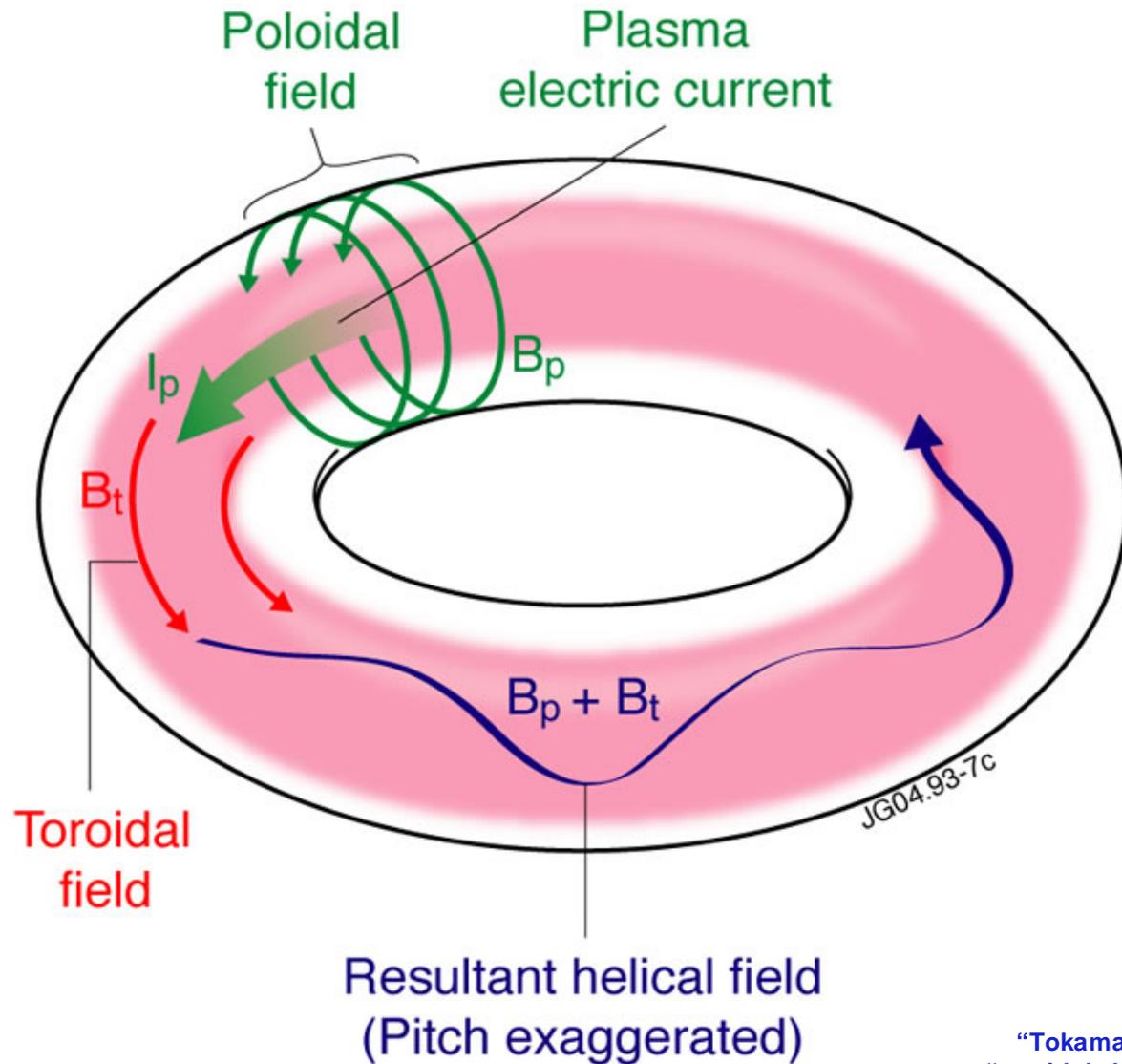
- Electric field created \rightarrow particles expelled outward toward vessel wall

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**

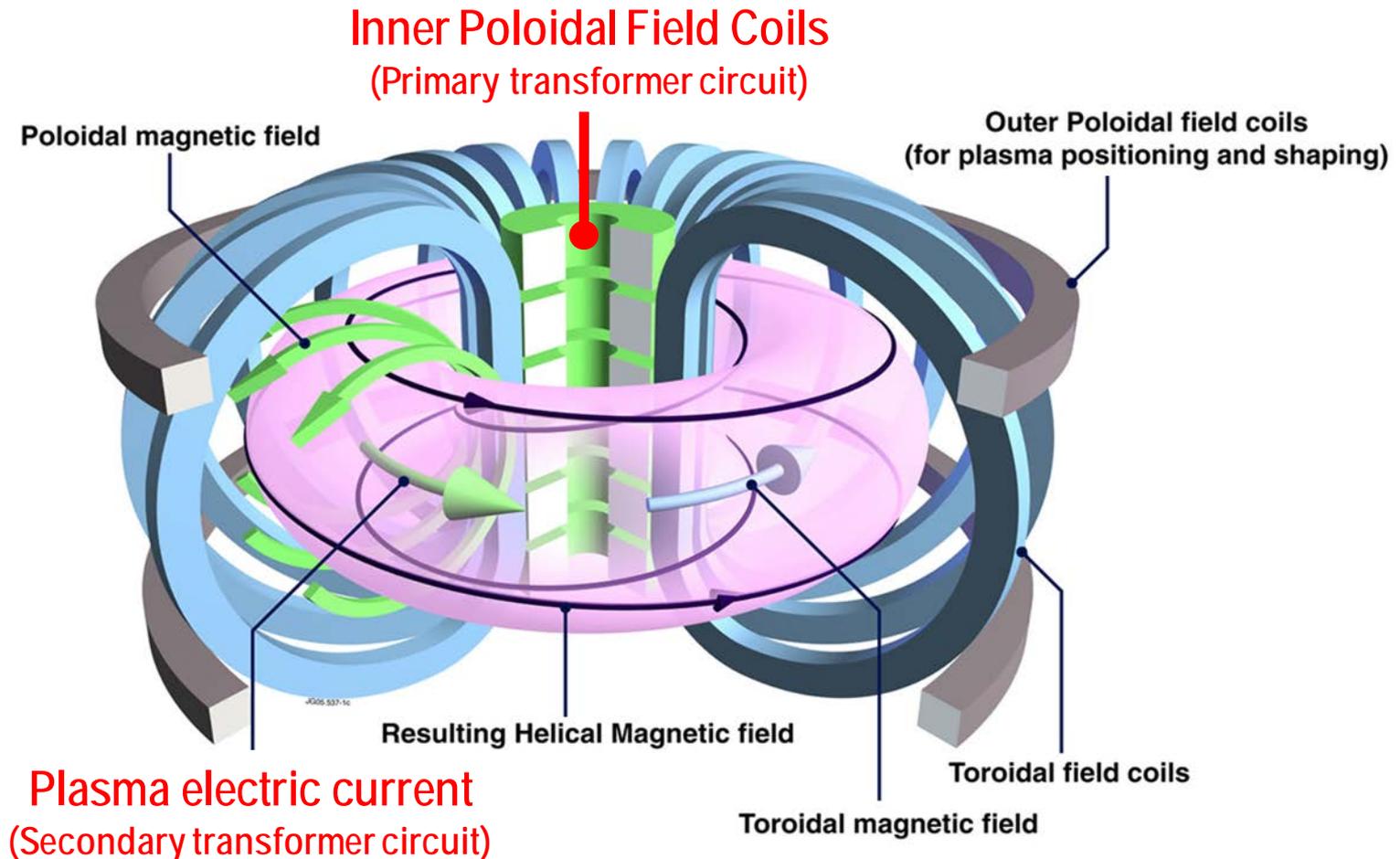
Tokamak: Poloidal field from plasma current + toroidal field from magnets create helical field



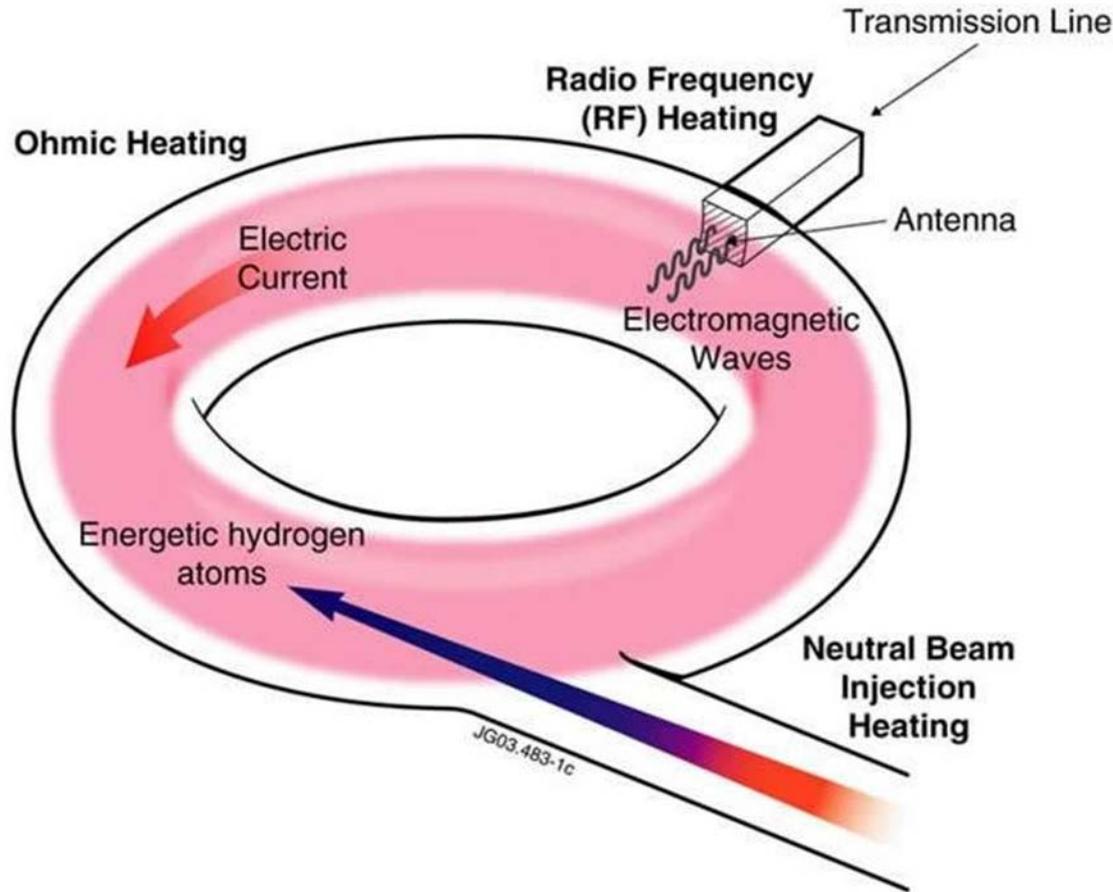
“Tokamak”: Russian acronym for “toroidal chamber with magnetic coils”

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

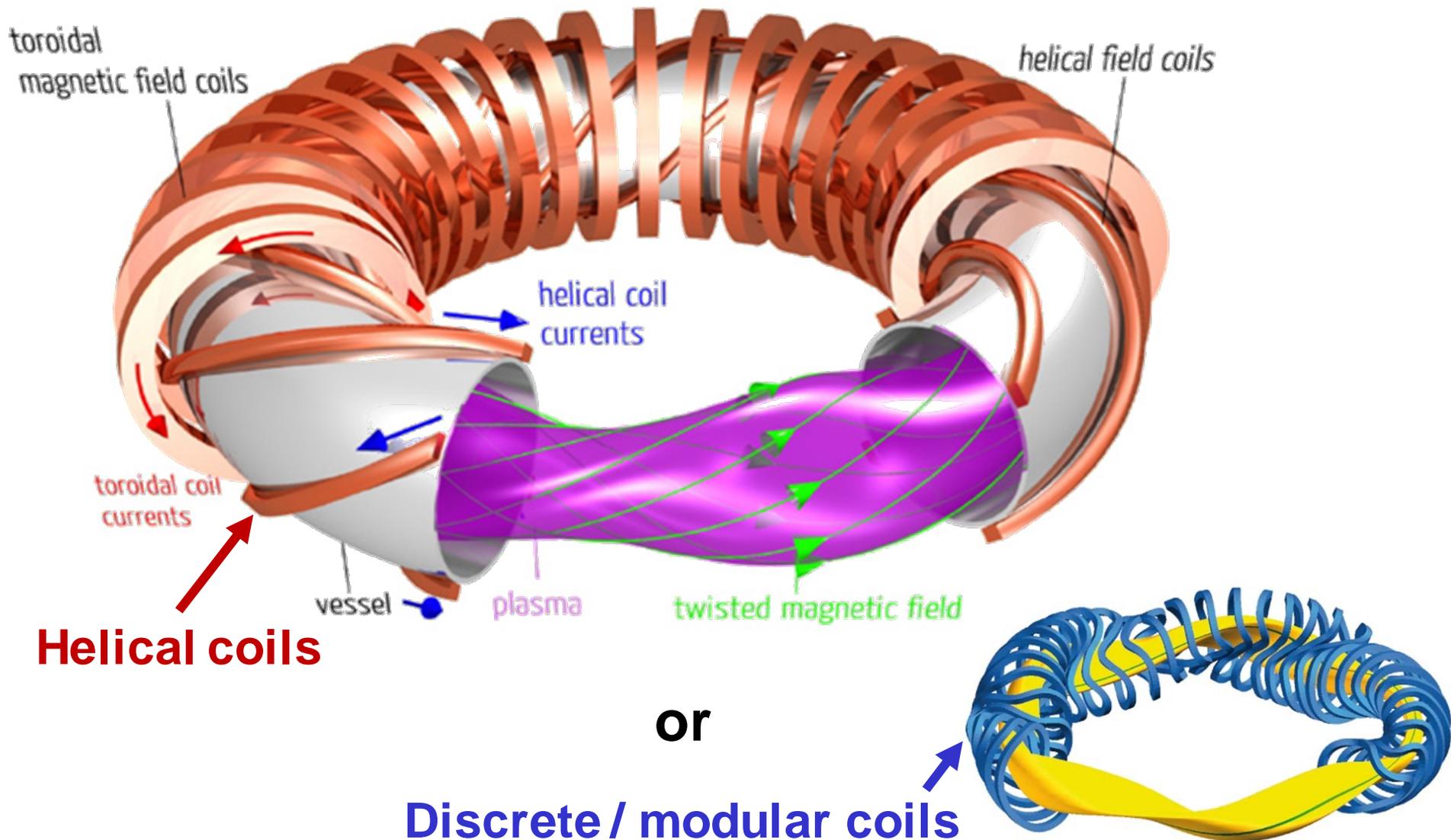


“Advanced Tokamak” uses external + self-generated current to operate without transformer



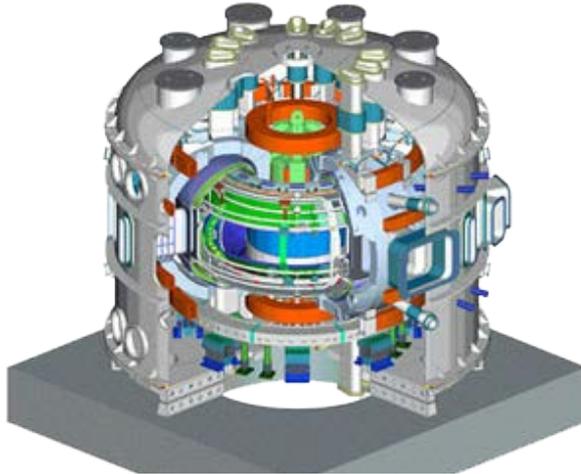
- External: wave and beam current drive
- Self-generated: high pressure plasmas make “bootstrap” current
- Efficient tokamak power plant will need 70-90% self-generated current

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



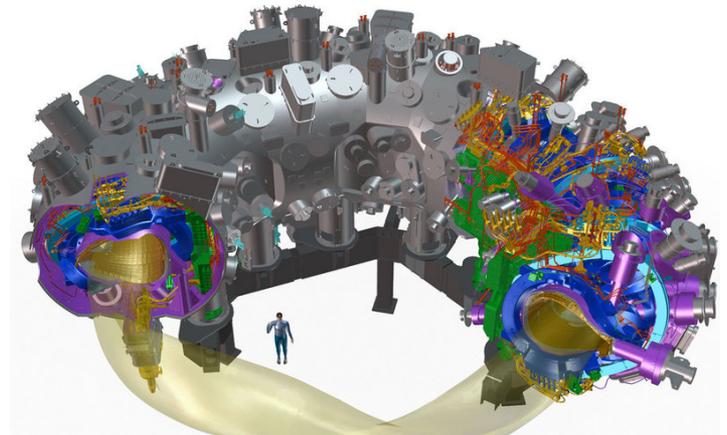
Tokamaks and stellarators are the leading configurations in magnetic fusion

Superconducting tokamak



KSTAR (South Korea)

Superconducting stellarator



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

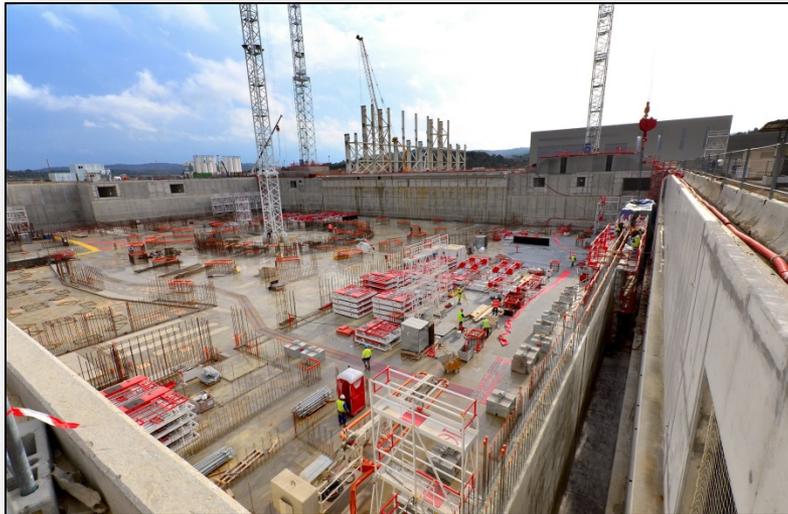
- Tokamak advantages:
 - Best confinement, closest to “breakeven”
 - Simpler planar coils and power/particle exhaust
- Disadvantages:
 - Must drive multi-mega-ampere plasma current
 - More prone to rapid loss of plasma = “disruption”

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

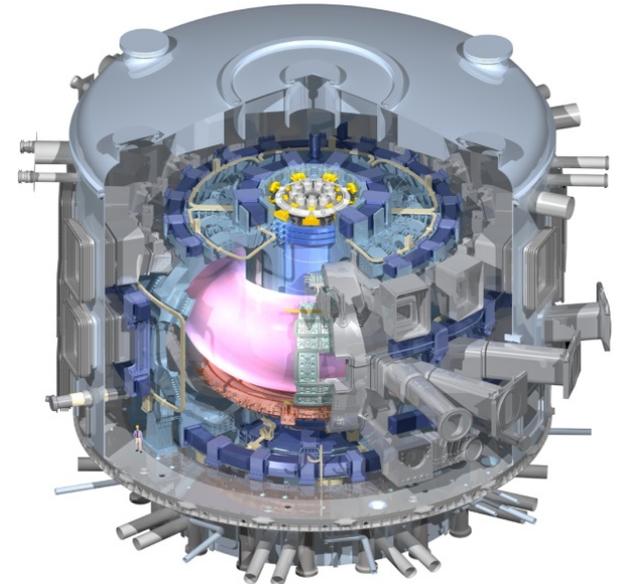
ITER will be first device to access “burning plasma”

- 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 $Q = P_{\text{fusion}} / P_{\text{external heating}} = 10$
- $Q = 10 \rightarrow P_{\text{alpha}} / P_{\text{external}} = 2$
- $P_{\text{alpha}} / P_{\text{alpha}} + \text{external}} = 2 / 3 > 50\%$

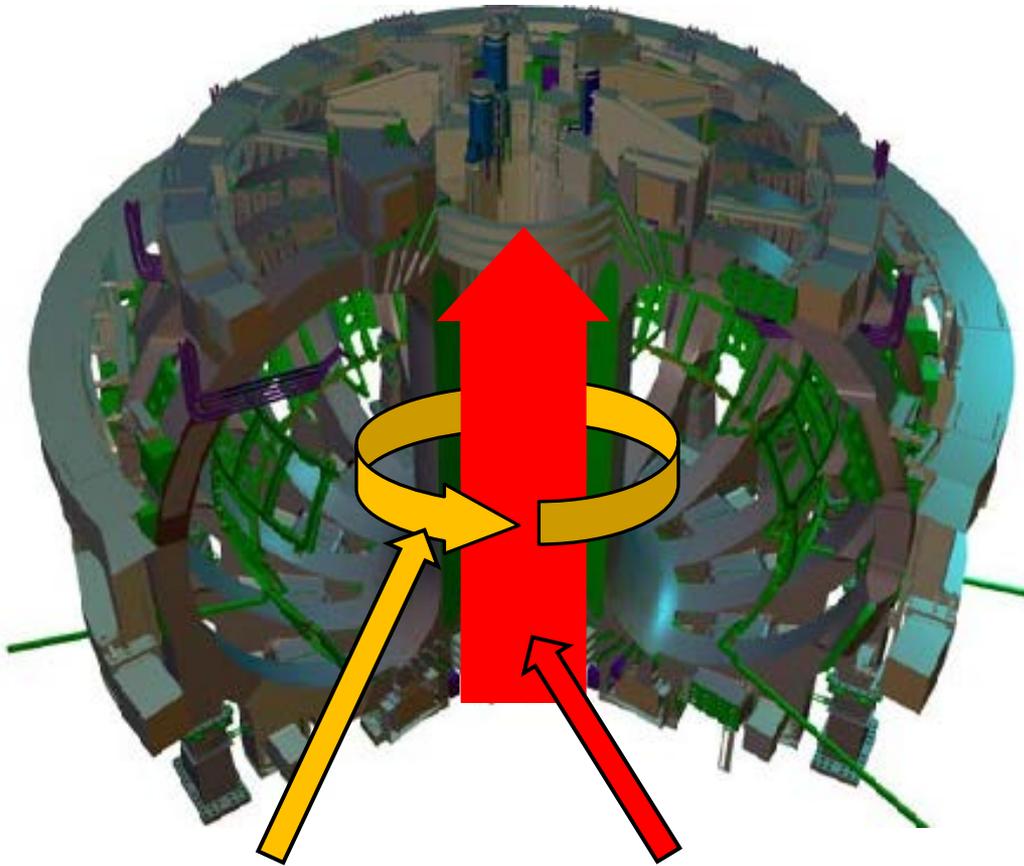
ITER under construction in Cadarache, France



$A=3.1, R=6.2\text{m}, B_T=5.3\text{T}, I_p=15\text{MA}$



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

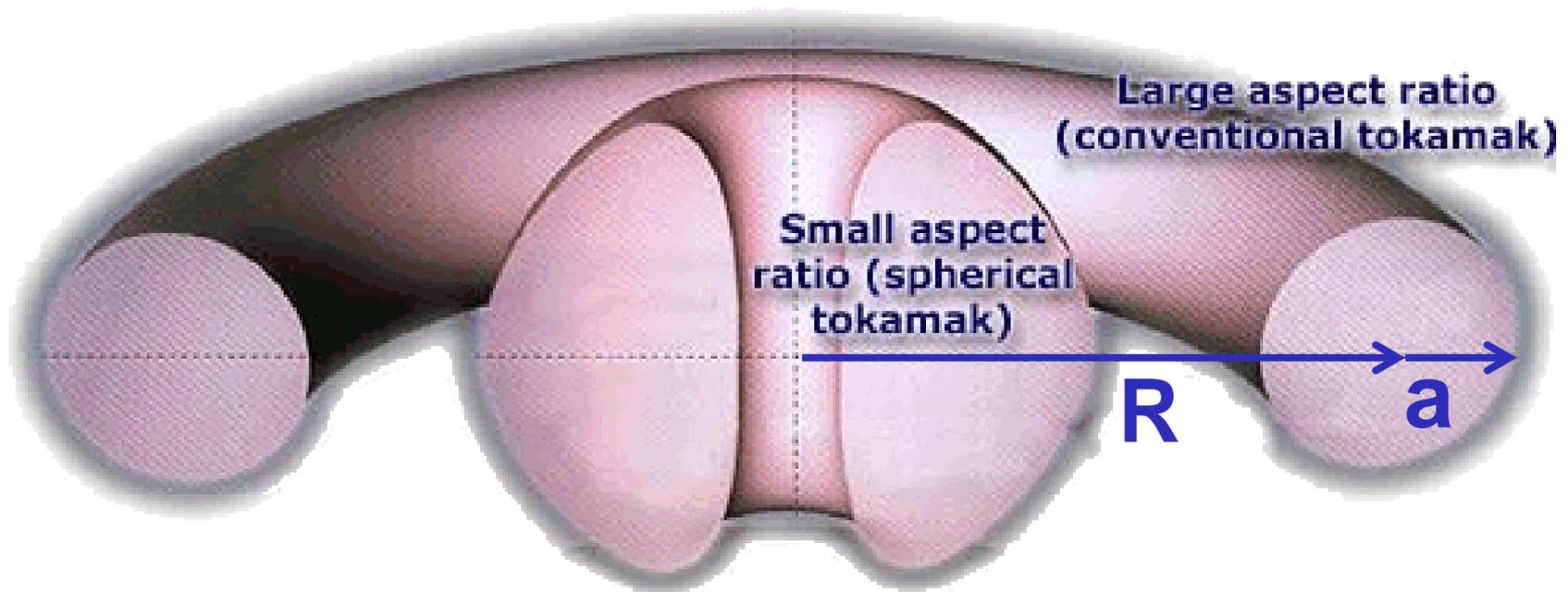
- Energy confinement scales with plasma current
- Large plasma current requires large toroidal field and/or plasma size for plasma to remain stable
- Current and confinement both scale with size
- **Can we make smaller devices with better confinement and smaller or cheaper magnets?**
- Such questions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

How might we possibly improve
the conventional tokamak?

Aspect ratio is important free parameter

$$\text{Aspect ratio } A = R / a$$

R = major radius a = minor radius



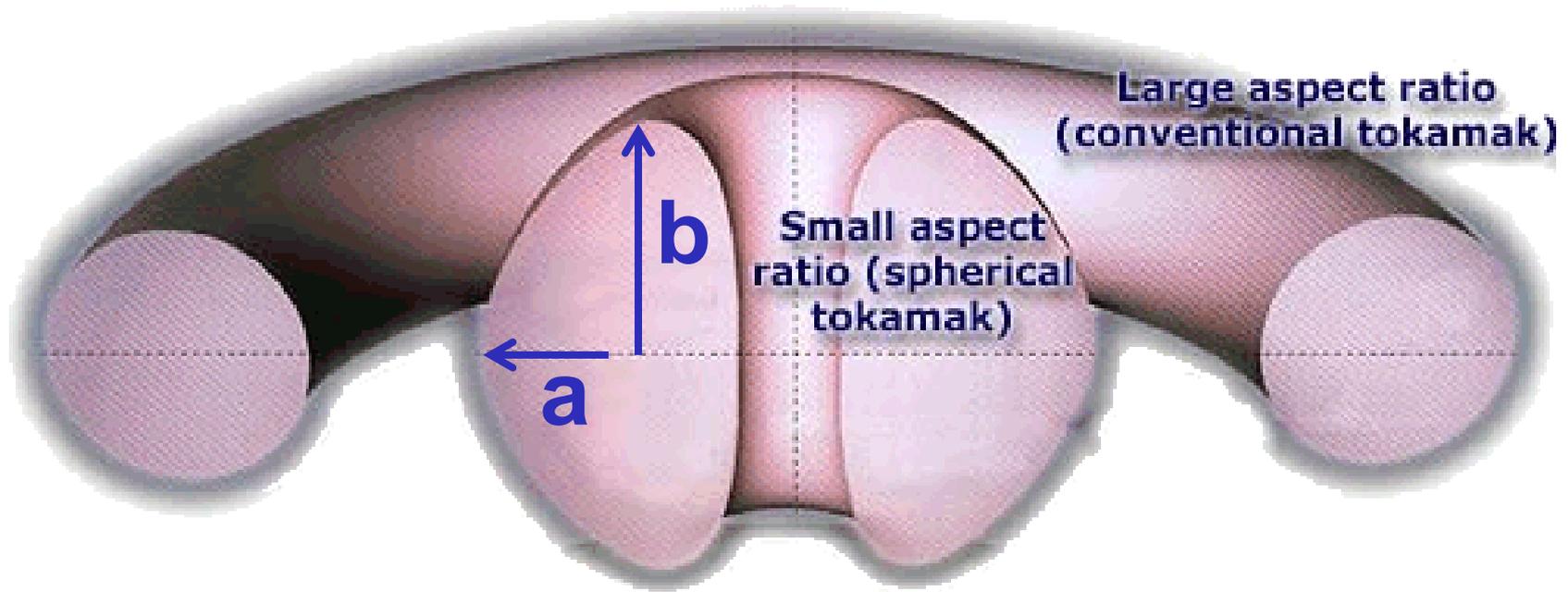
Spherical torus/tokamak (ST) has $A = 1.1-2$
Conventional tokamak typically $A = 3-4$

STs have higher natural elongation

Elongation $\kappa = b / a$

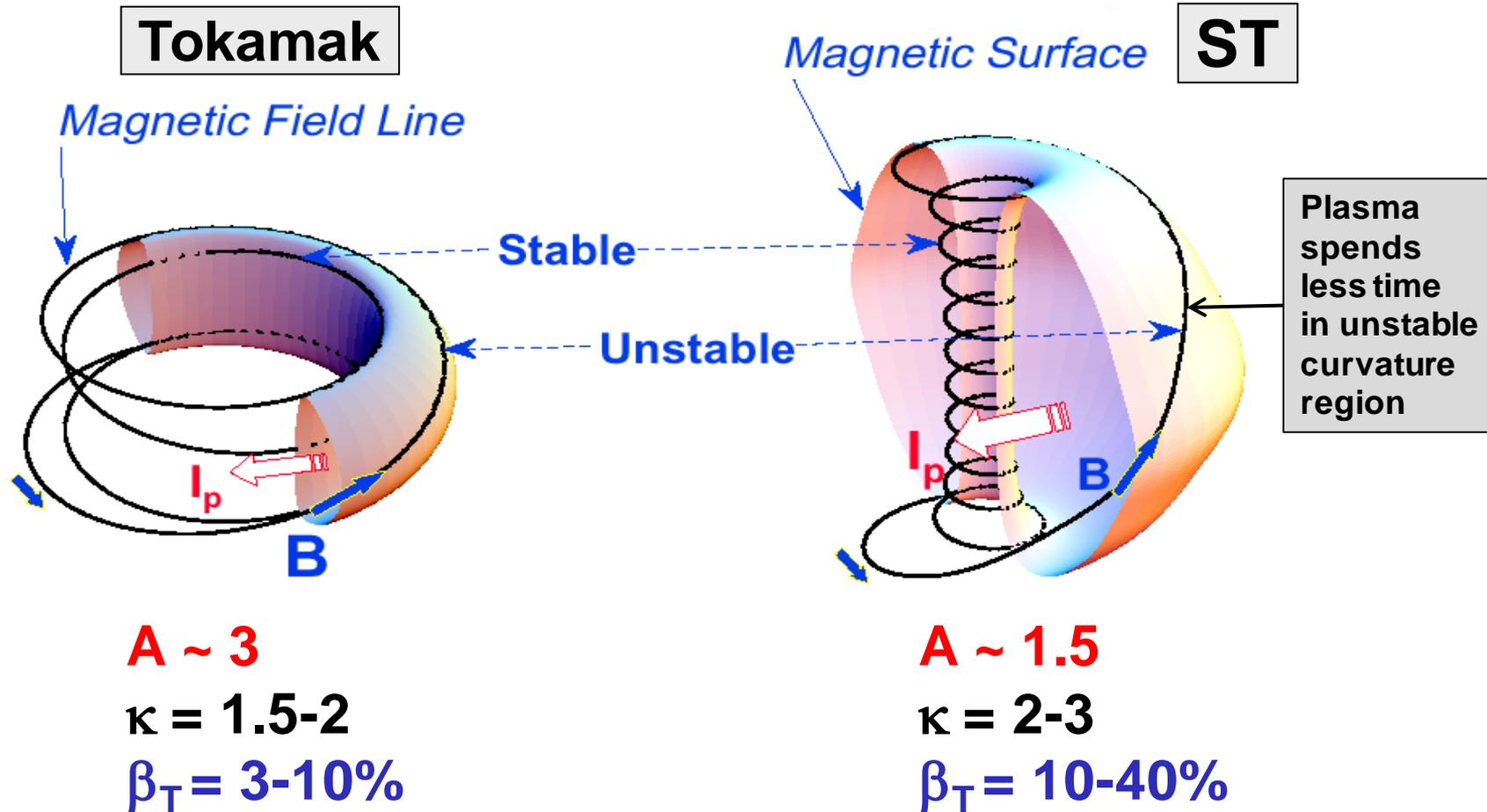
$b =$ vertical $\frac{1}{2}$ height

$a =$ minor radius



Higher elongation improves stability, confinement

Favorable average curvature improves stability



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 $\tau_E \propto I_p$

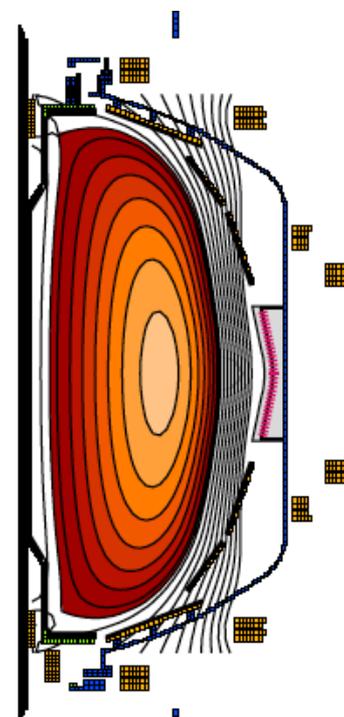
Normalized pressure $\beta_T [\%] \equiv \beta_N I_p / (aB_{T0})$

- 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

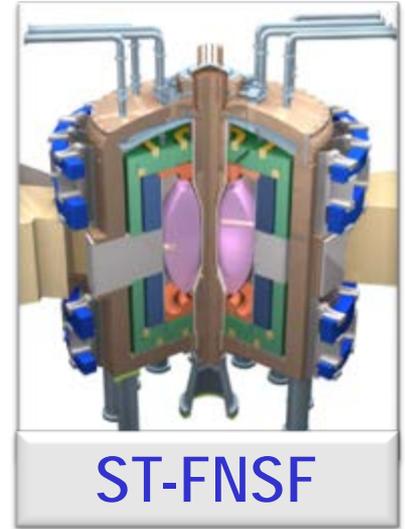


Fusion technology development is major challenge

Fusion Nuclear Science Facility (FNSF) could aid development

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

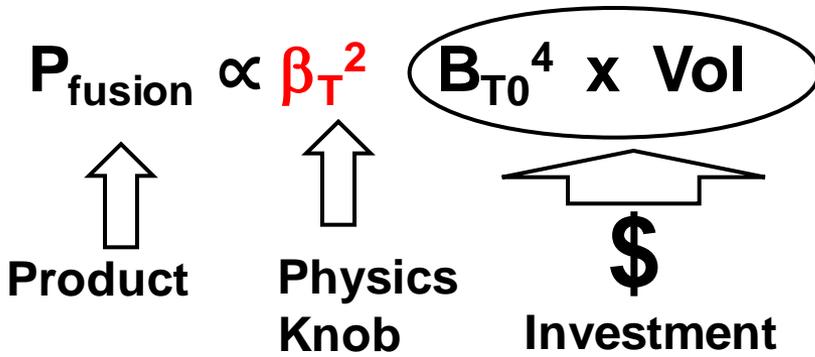
- Divertor, plasma facing components for exhaust
- Blanket and Integral First Wall
- Vacuum Vessel and Shield
- Tritium Fuel Cycle
- Remote Maintenance Components



- **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

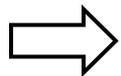
$$P_{\text{fusion}} \propto \langle p \rangle^2 \times \text{Volume}$$



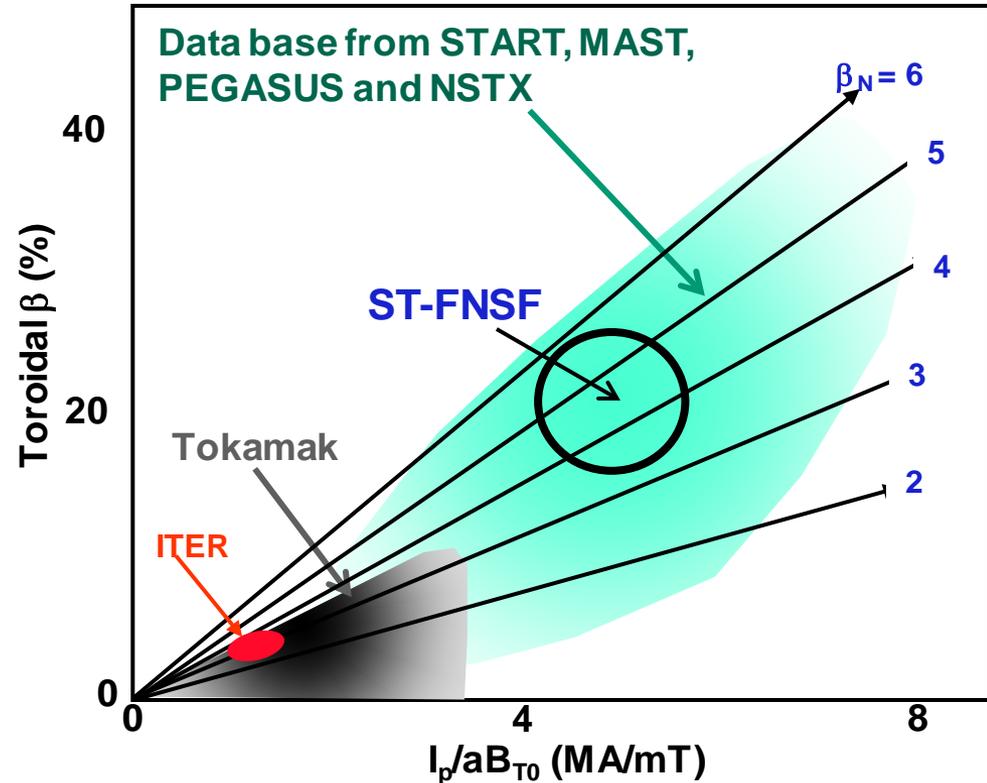
High neutron wall loading W_n possible in a compact ST

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

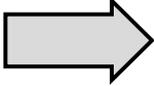
$$W_n \propto \beta_T^2 B_{T0}^4 a \quad (\text{not strongly size dependent})$$



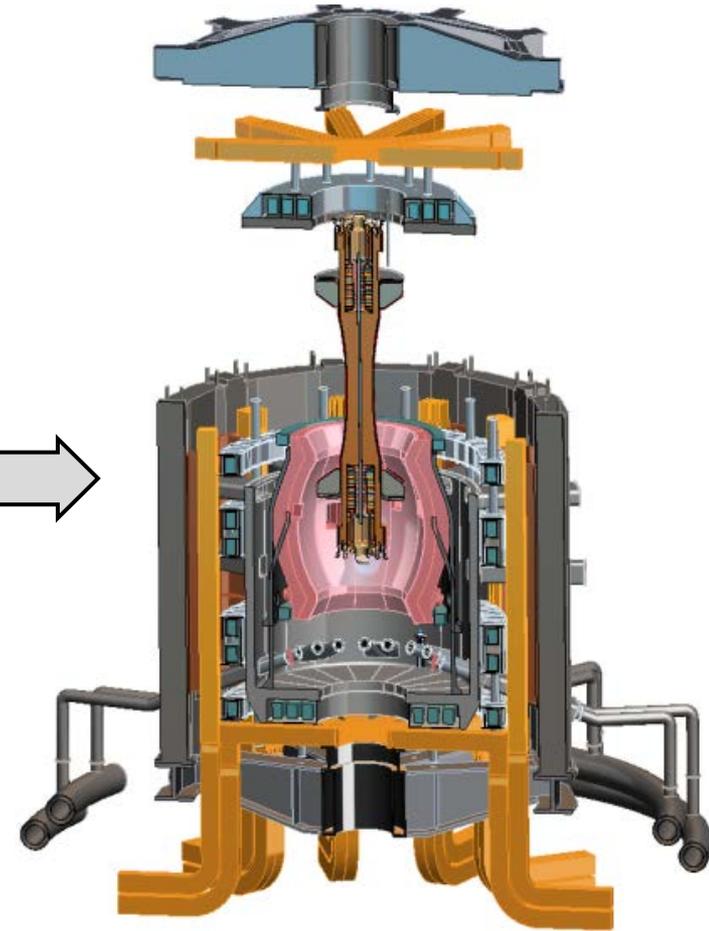
$W_n \sim 1 \text{ MW/m}^2$ with $R \sim 1 \text{ m}$ FNSF feasible!



Design studies show ST potentially attractive as FNSF

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

PPPL ST-FNSF concept

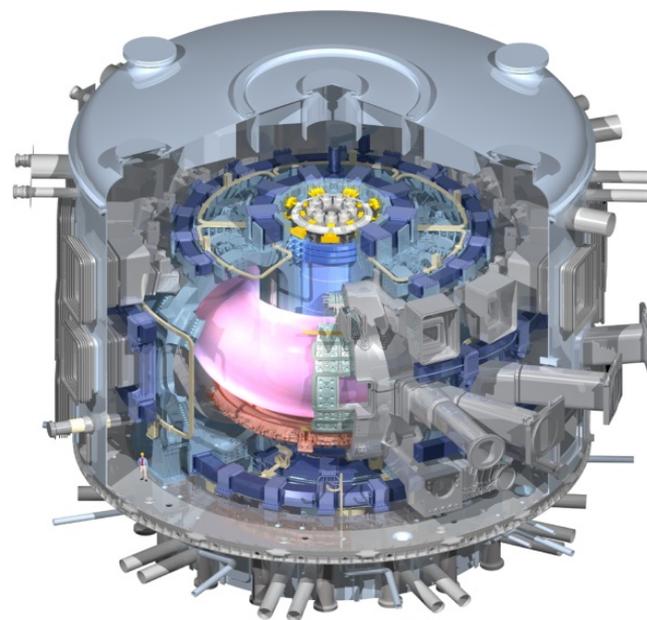


Unique ST properties also support ITER

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

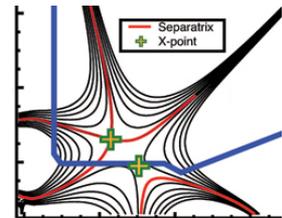
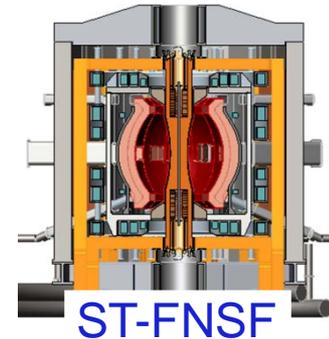


What are the goals of
NSTX Upgrade (NSTX-U)?

What key science questions
will NSTX-U address?

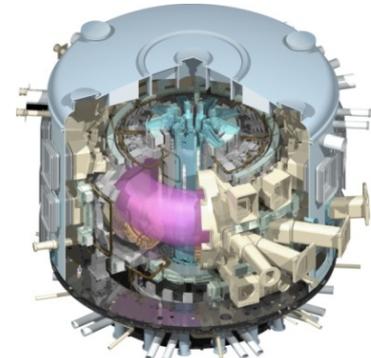
NSTX Upgrade mission elements

- Advance ST as candidate for Fusion Nuclear Science Facility (FNSF)
- Develop solutions for the plasma-material interface challenge
- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond
- Develop ST as fusion energy system



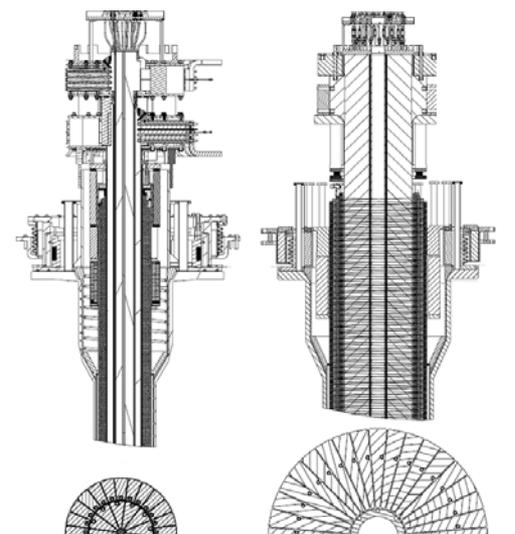
Liquid metals/Li “Snowflake”

ITER



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

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



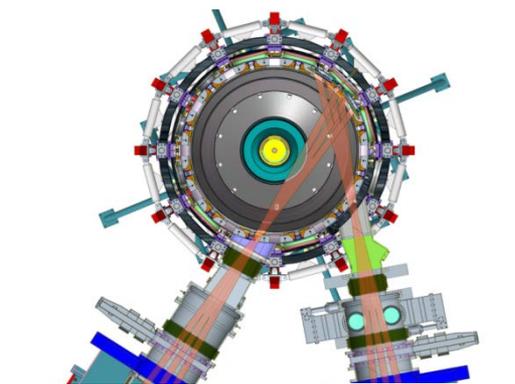
TF OD = 20cm **TF OD = 40cm**

- **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 $nT\tau$ from NSTX
NSTX-U average plasma pressure ~ tokamaks

Key NSTX-U research topics for FNSF and ITER:

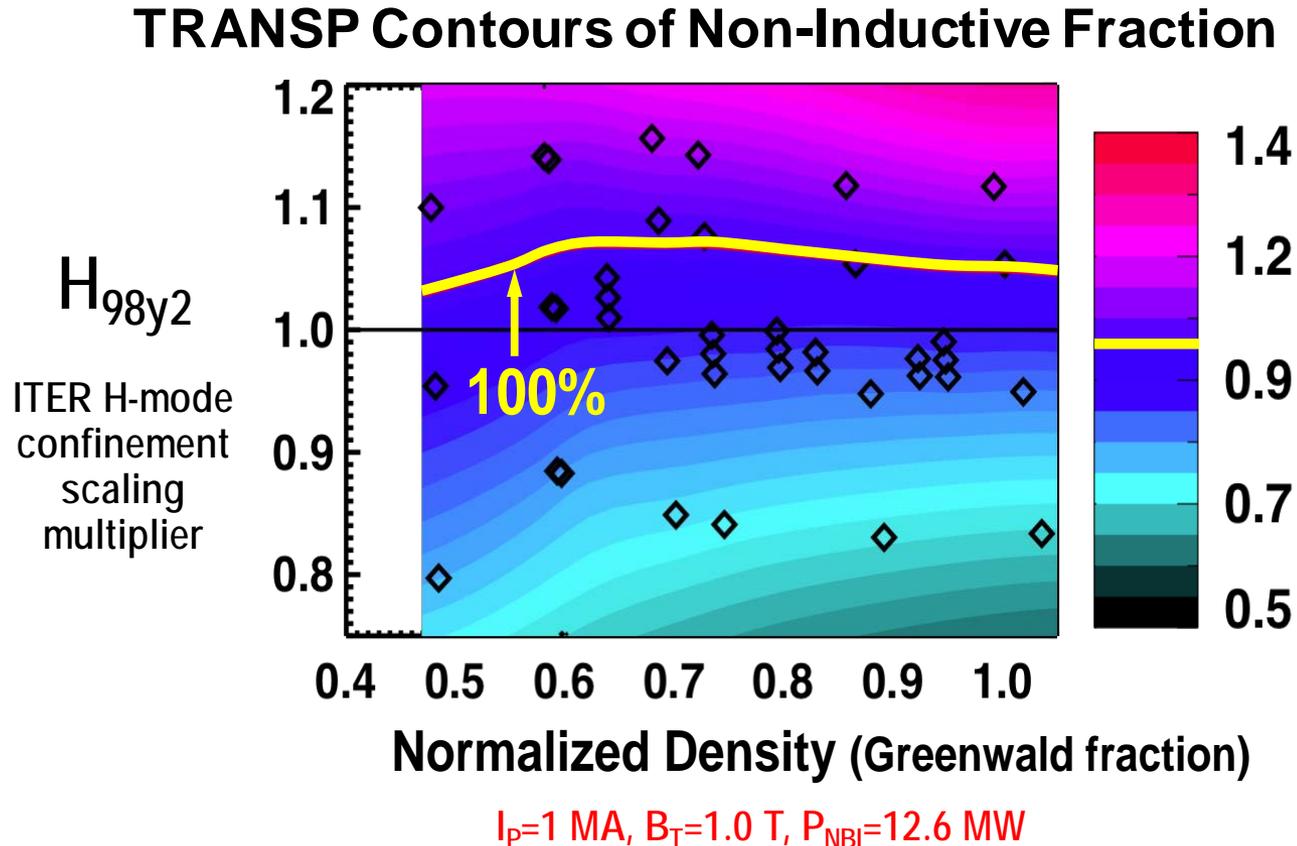
- **Stability and steady-state control at high β**
- **Confinement scaling (esp. electron transport)**
- **Non-inductive start-up, ramp-up, sustainment**
- **Divertor solutions for mitigating high heat flux**



Present NBI **New 2nd NBI**

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

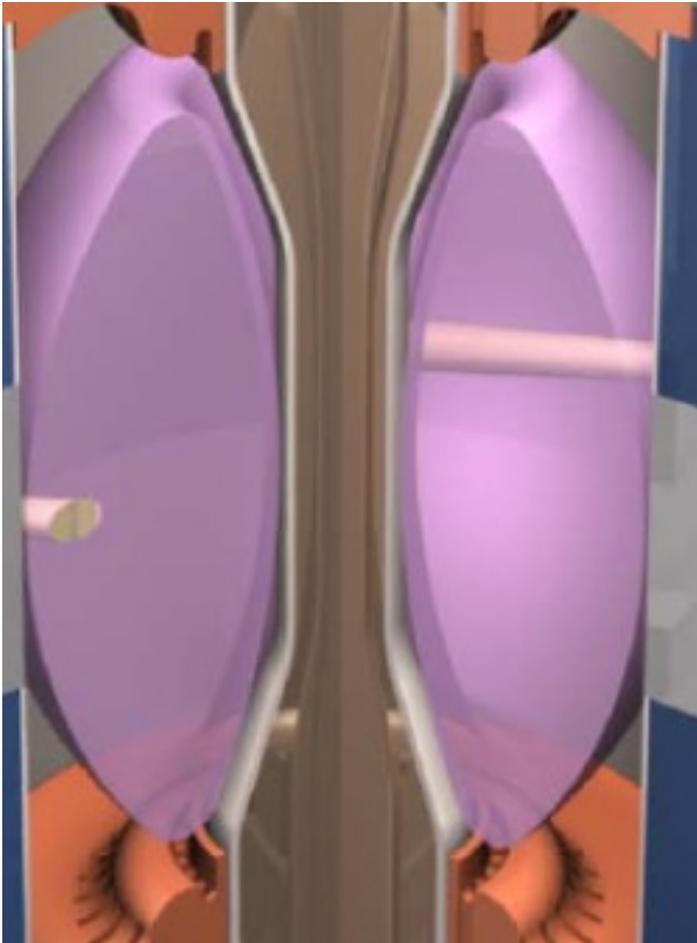
NSTX achieved 70% “transformer-less” current drive Will NSTX-U achieve 100% as predicted by simulations?



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**



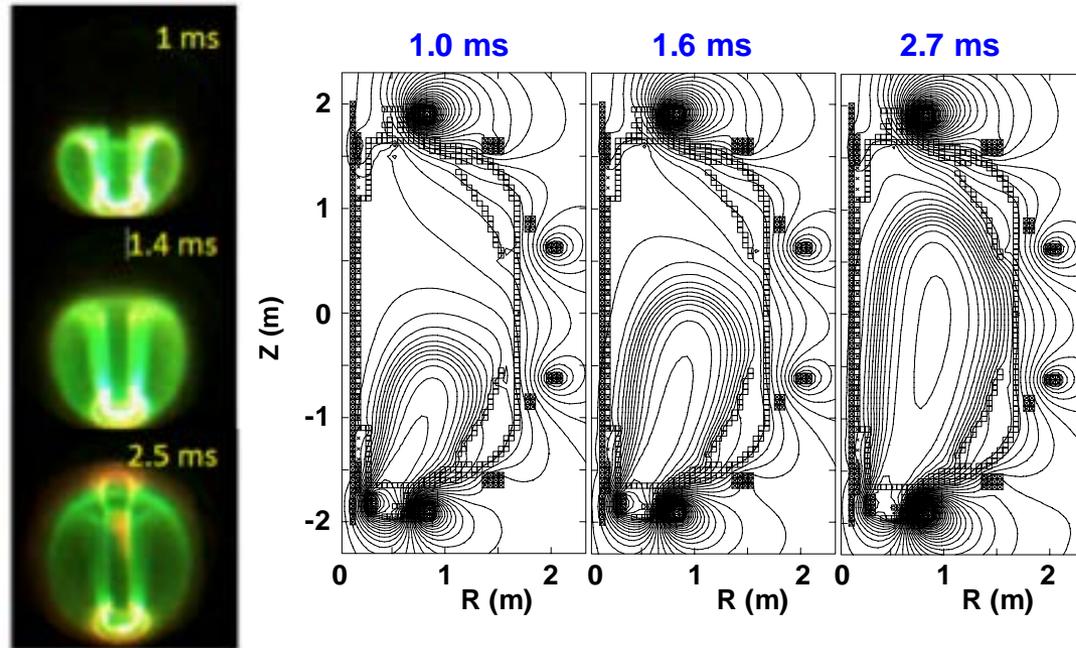
**~ 1-2 MA of transformer-free
start-up current needed for FNSF
→ 10-20% of total current**

- **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?

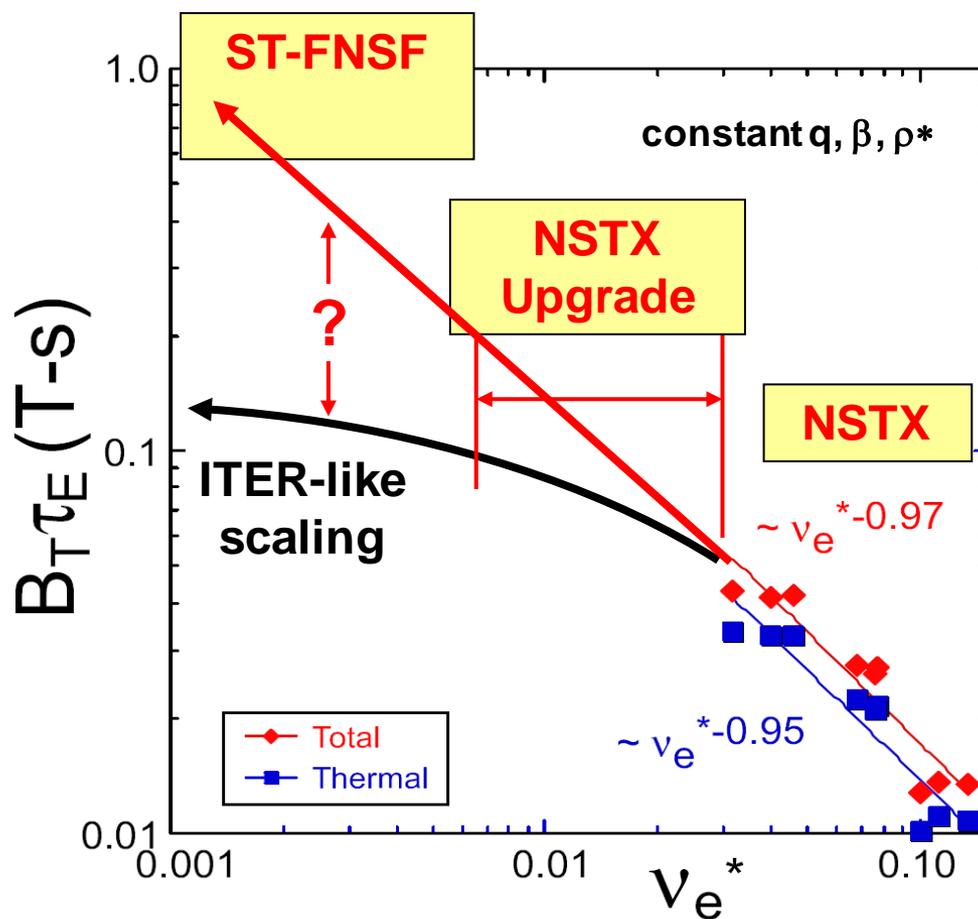
- TSC code (2D) successfully simulated helicity injection $I_p \sim 200\text{kA}$ in NSTX



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

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

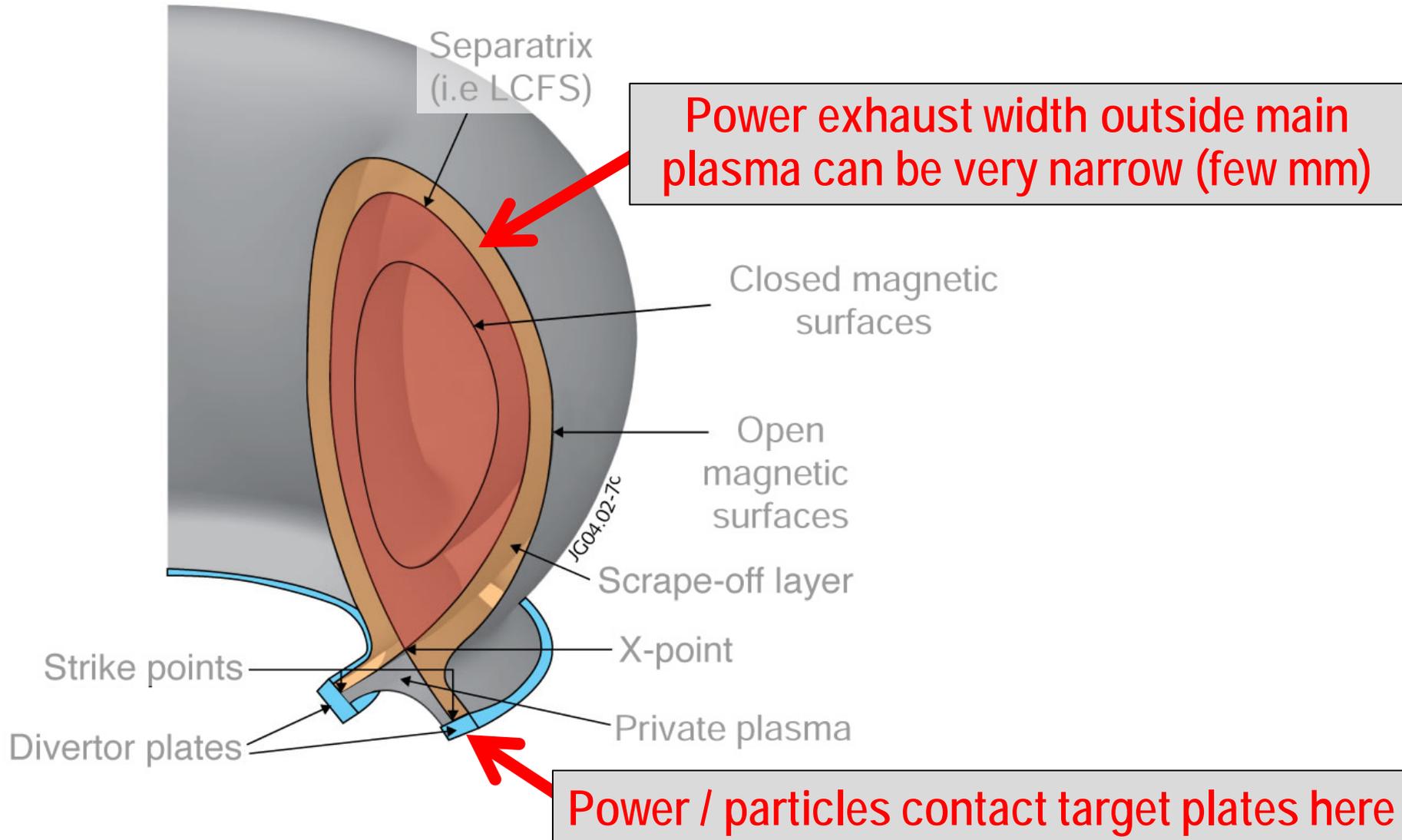
Will confinement trend continue, or look like conventional A?



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

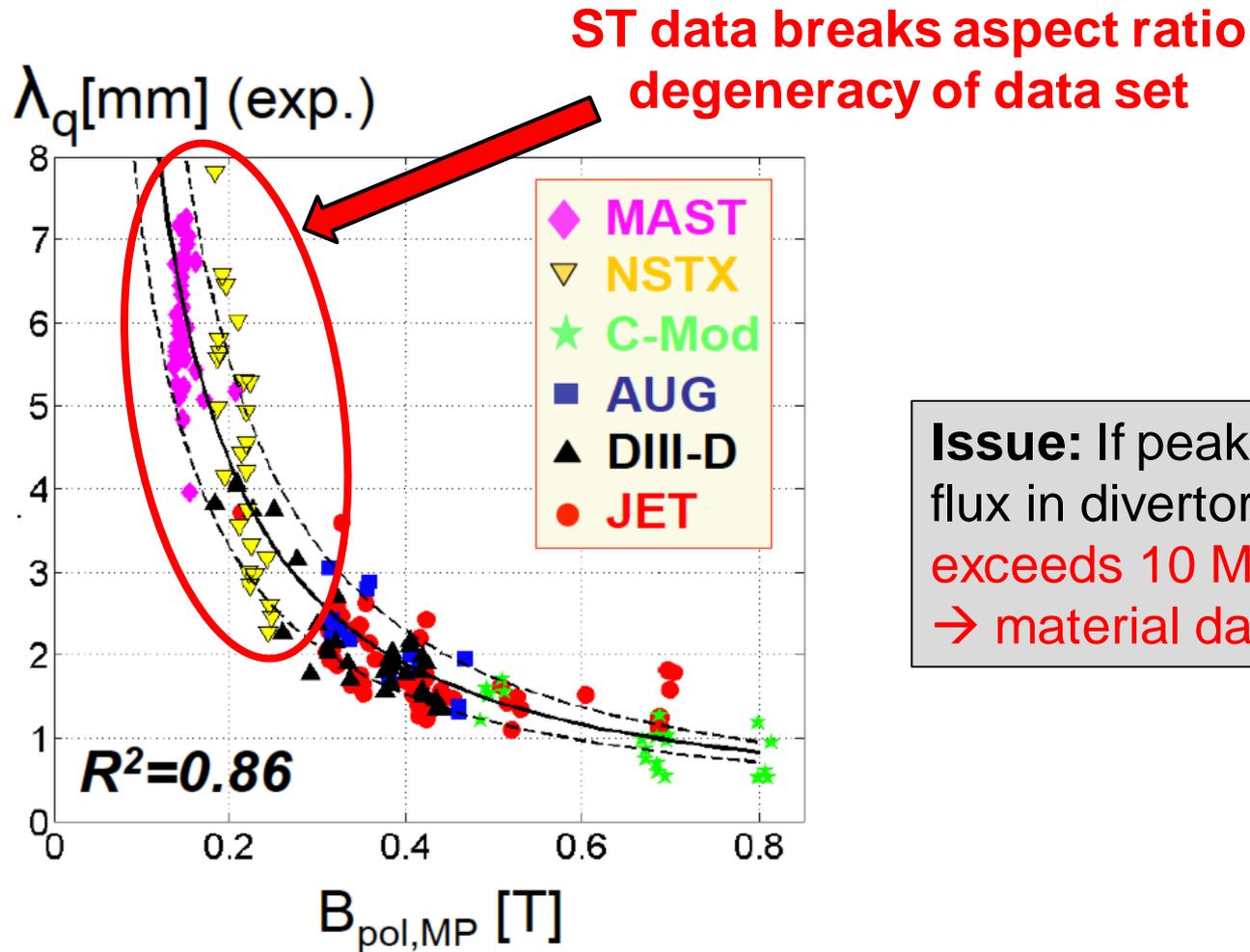
Favorable confinement results could lead to more compact ST reactors

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



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

Will previous ST trend continue at $2 \times I_P, B_P, B_T$, power?

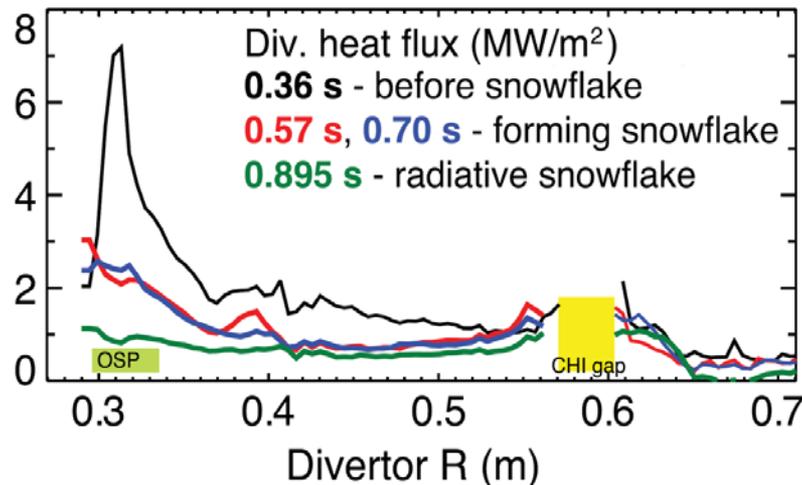
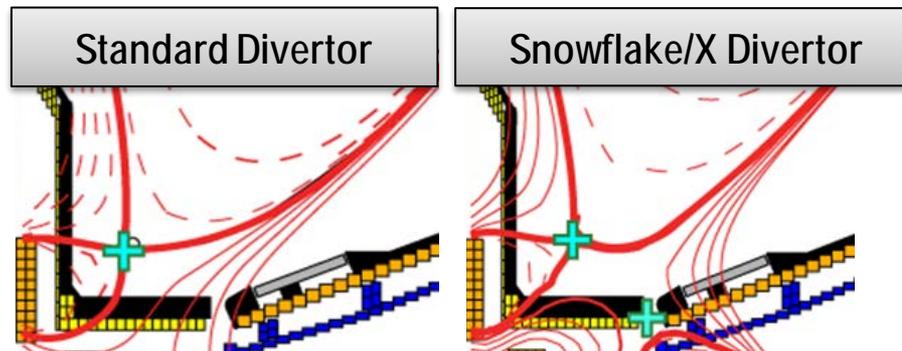


Issue: If peak heat flux in divertor region exceeds 10 MW/m^2 \rightarrow material damage

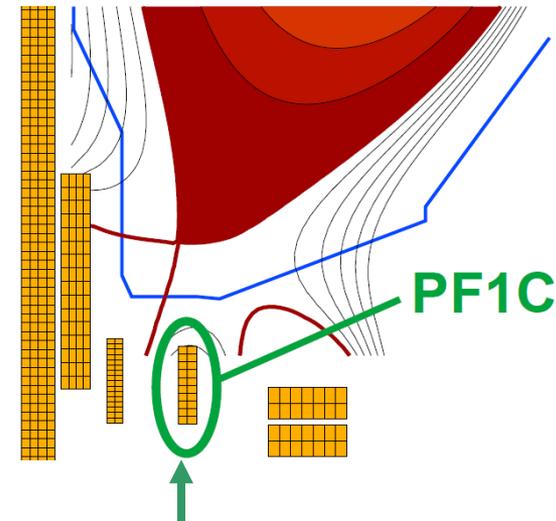
Wider heat-flux width may offset smaller R \rightarrow maybe better than tokamak

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

- NSTX: reduced heat flux 2-4 \times 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 \times higher than in NSTX

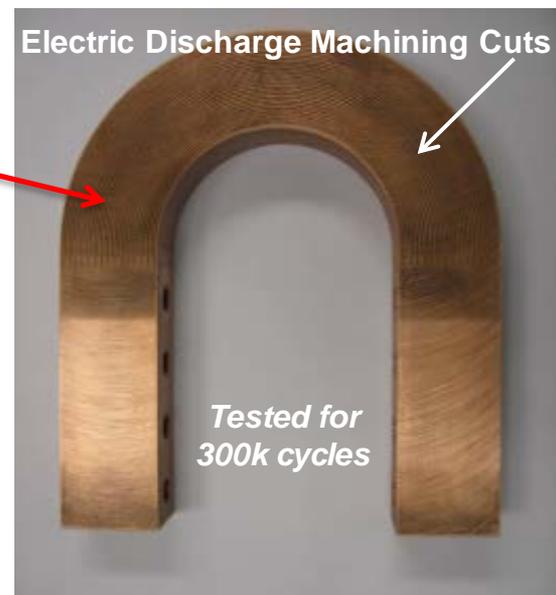
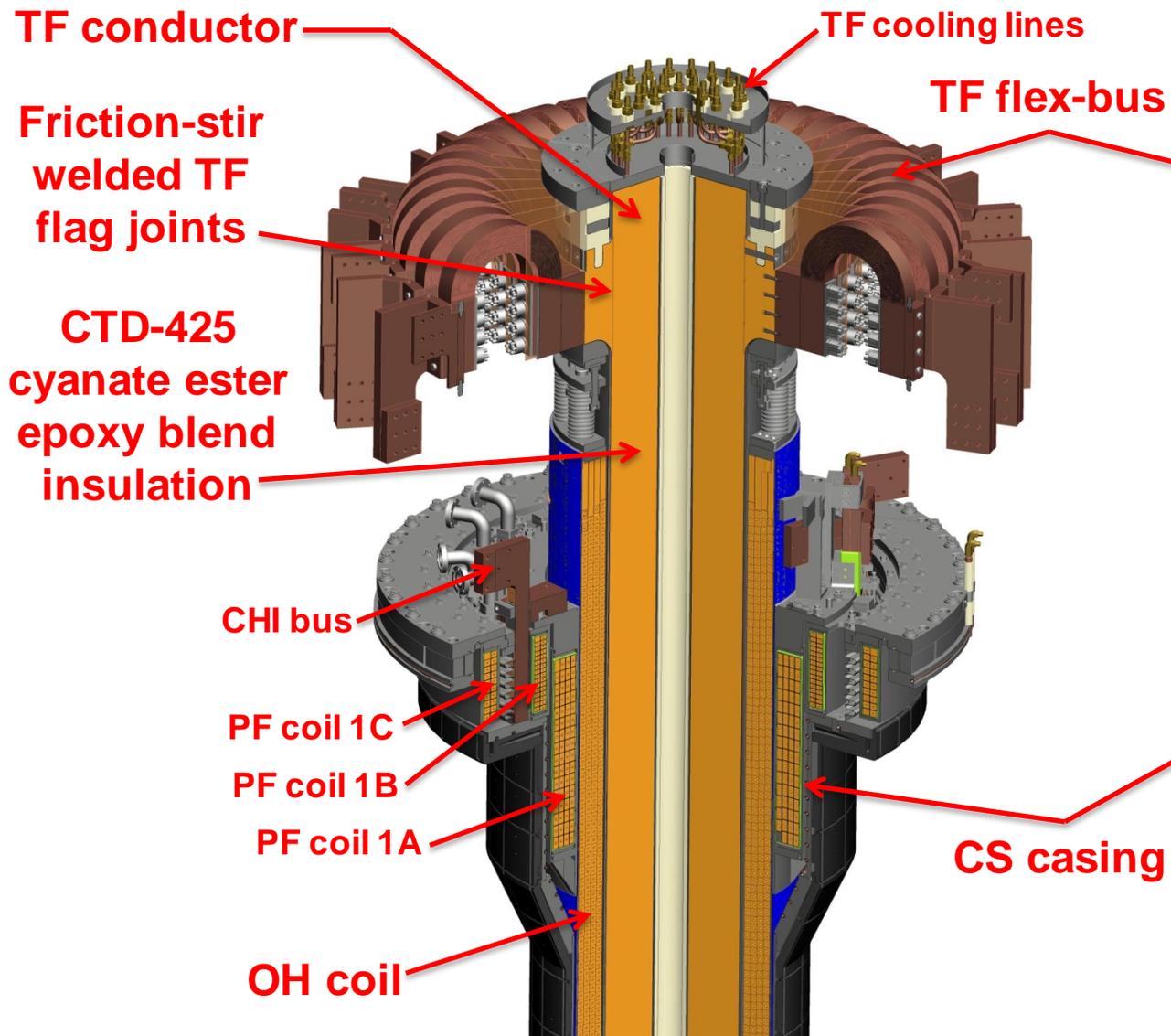


NSTX-U has additional coils for up-down symmetric snowflake/X, improved control

How was NSTX-U constructed?

New center-stack designed to handle increased forces

Identical 36 TF conductors and innovative flex-bus design



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



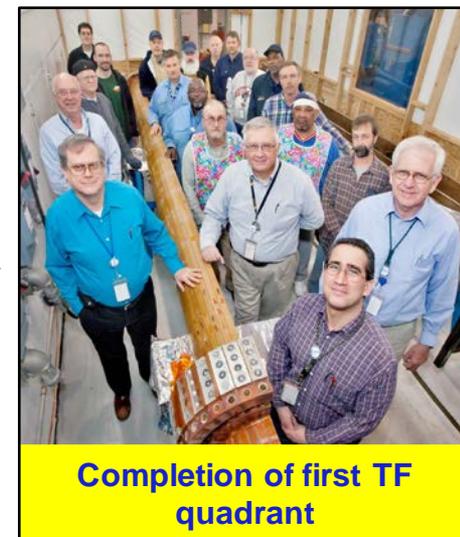
Insulated conductor being placed into mold



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



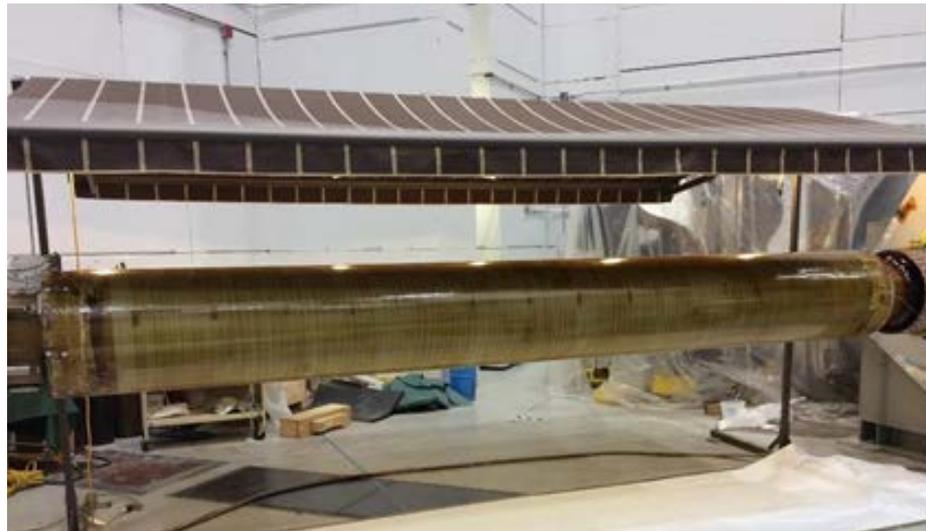
Aquapour + glass layer applied



OH conductor wound over TF



Completion of transformer winding and VPI

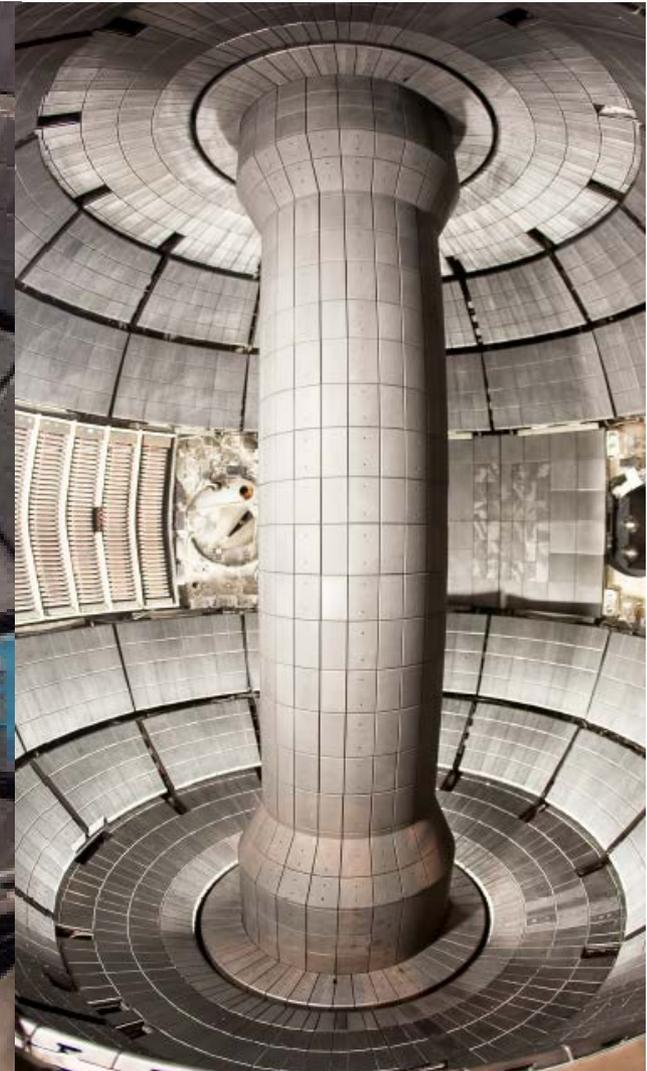


Same-day delivery to D-Site costs extra



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



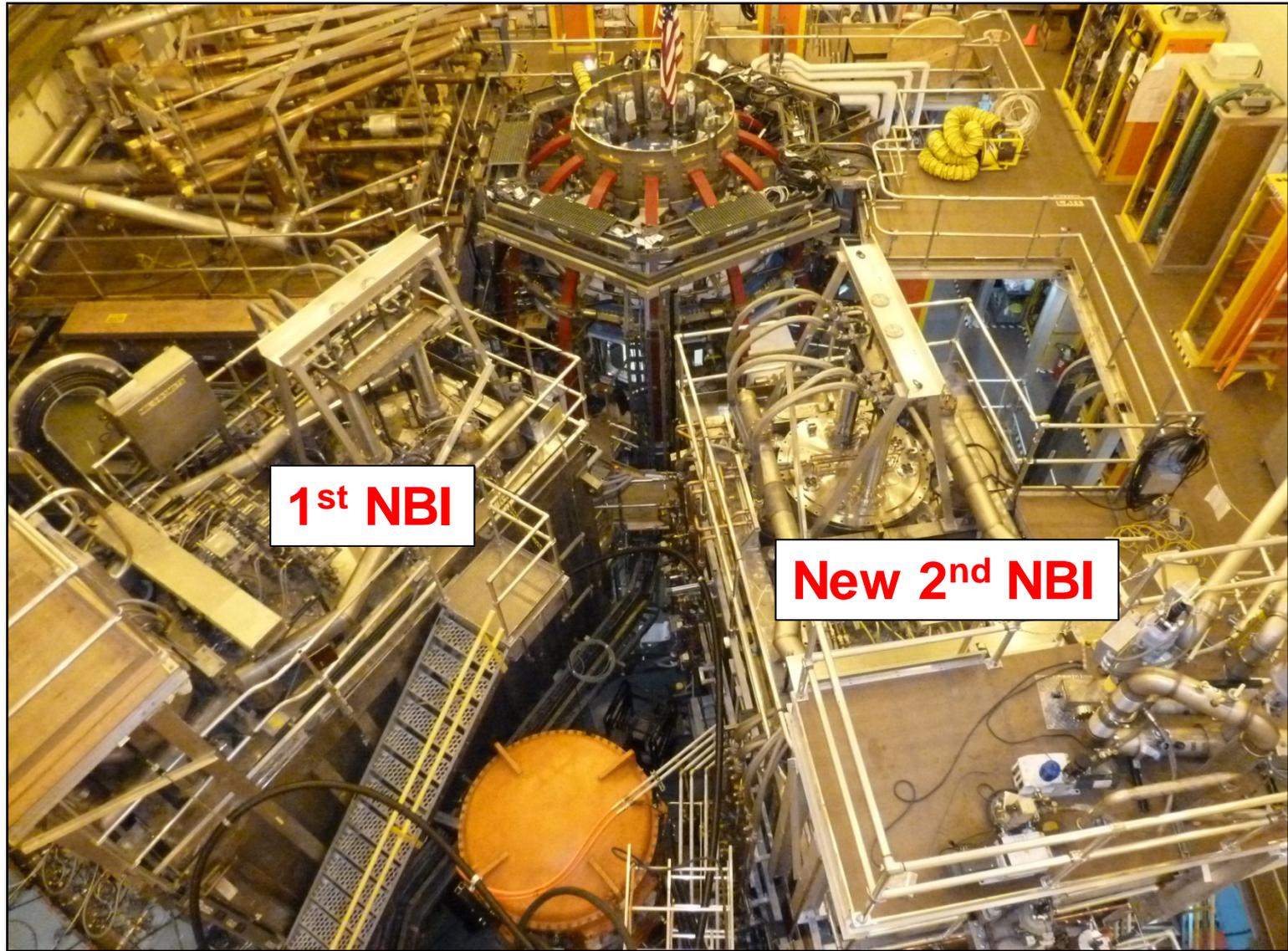
Beam Box placed in its final location and aligned



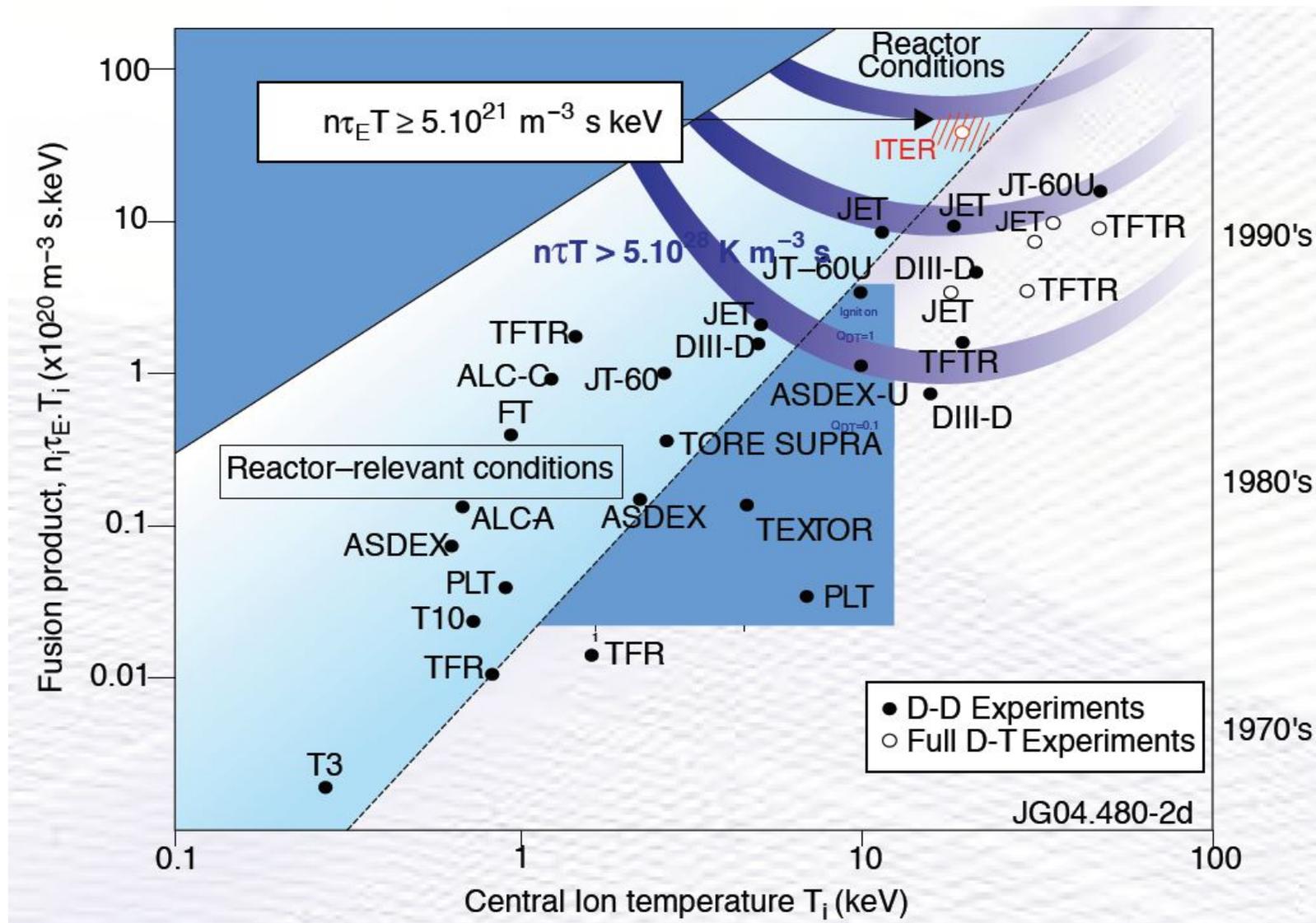
Beam Box being populated with components

NSTX Upgrade Project nearing completion

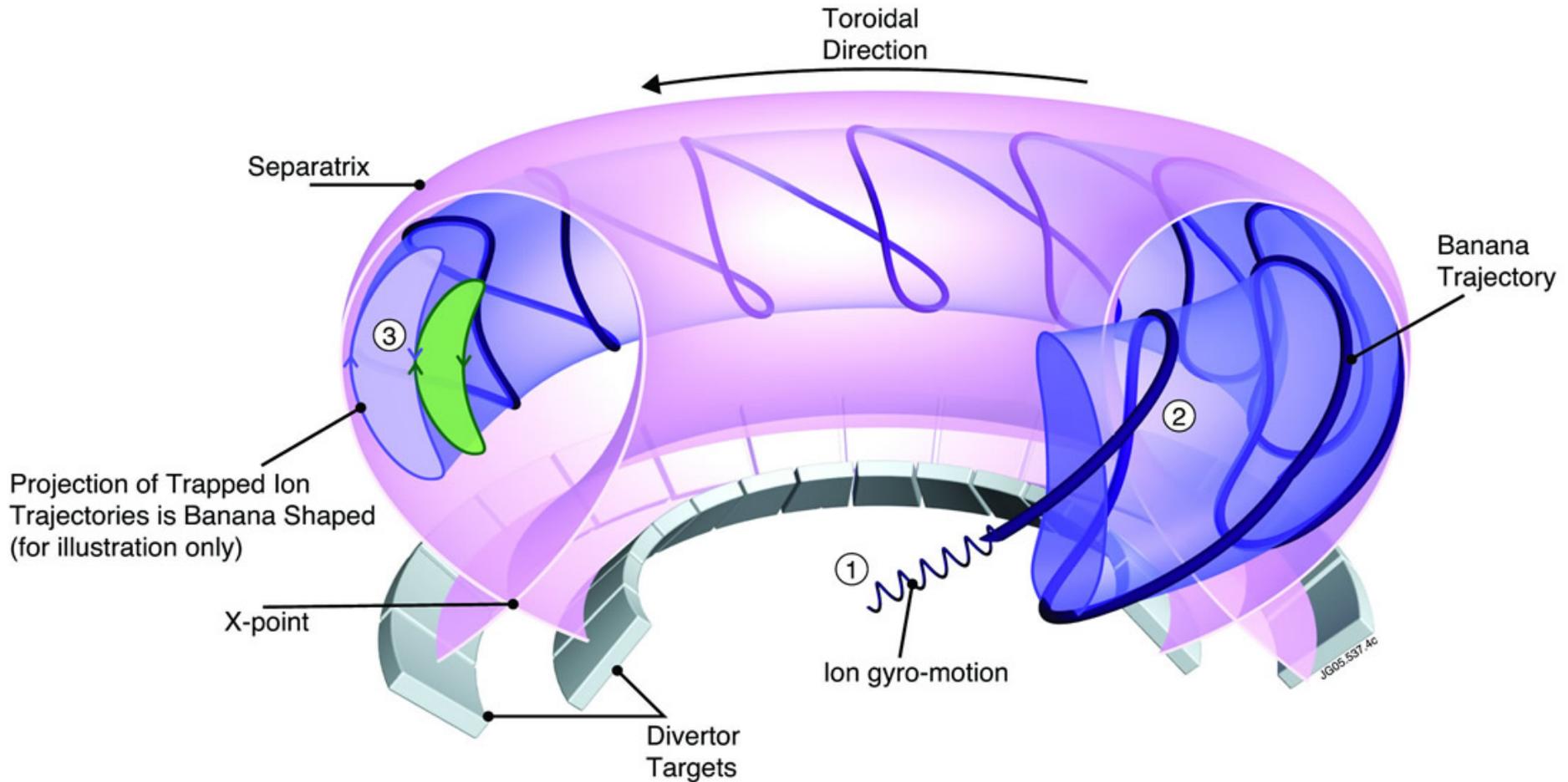
Test plasmas expected in April, research plasmas in June



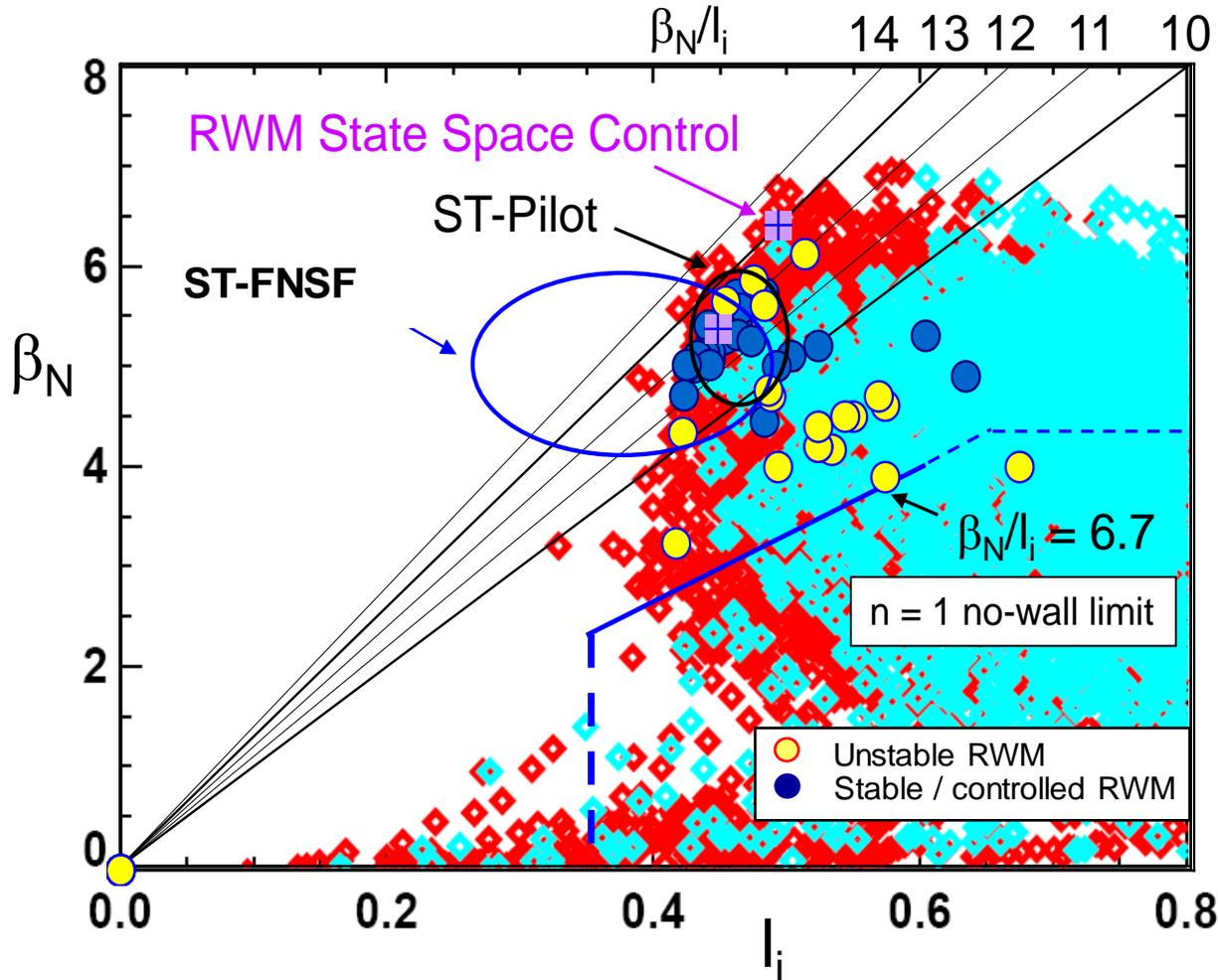
Magnetic Confinement Fusion Performance



Particle trajectories in a tokamak



Record β_N and β_N / I_i accessed in NSTX using resistive wall mode stabilization

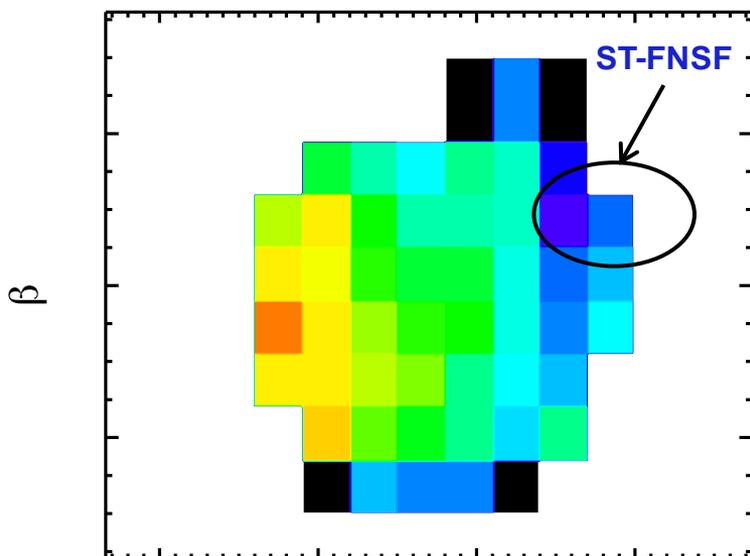


High β_N regime is important for bootstrap current generation.
 High β_N/I_i regime important since high f_{BS} regime has low I_i .

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

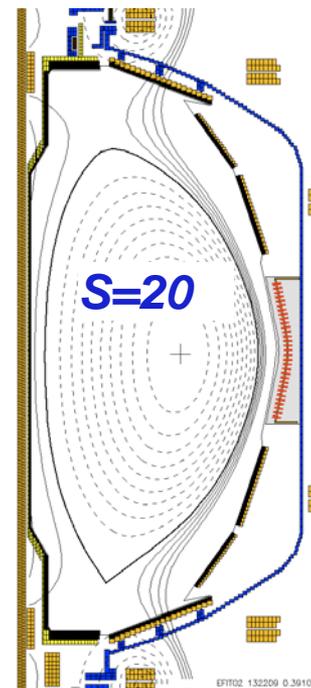
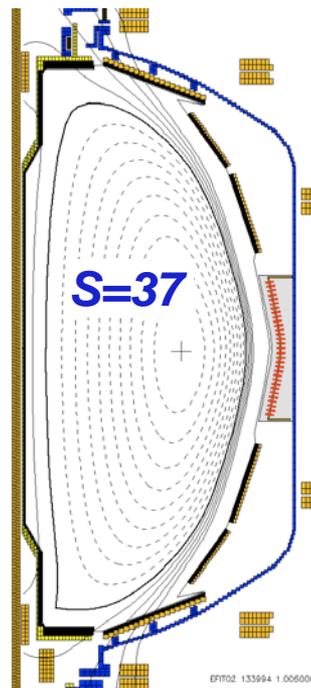
NSTX Data Demonstrates a Favorable Operations Window For Reduced Disruptivity in an ST-FNSF



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

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 I_i (broader current profile)
 - Decreasing pressure peaking

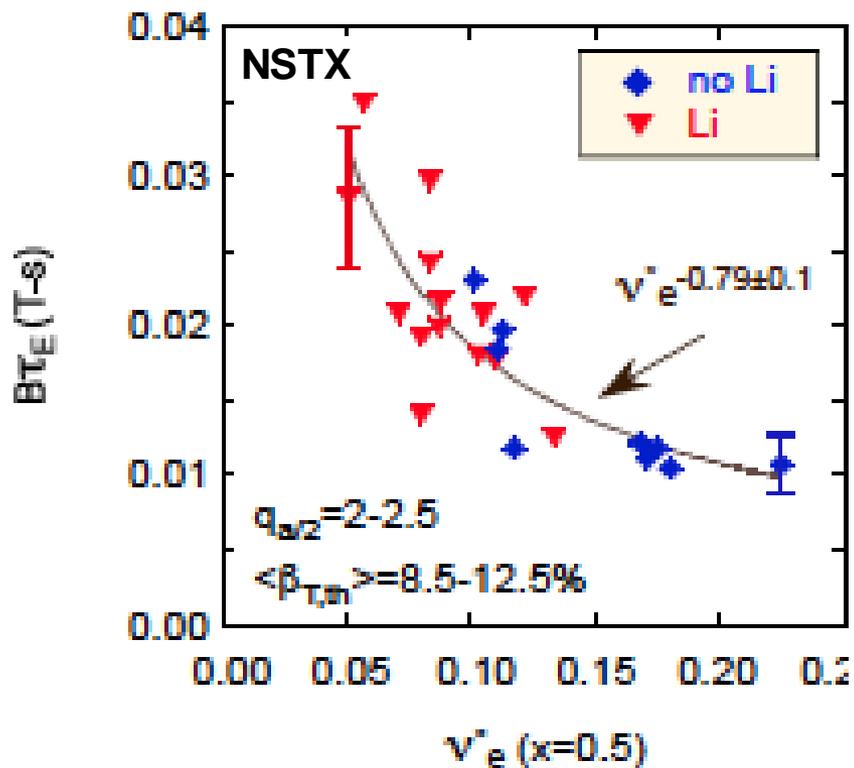
Upgrades will test and improve these favorable trends in a systematic way

Favorable Confinement Trend with Collisionality and β found

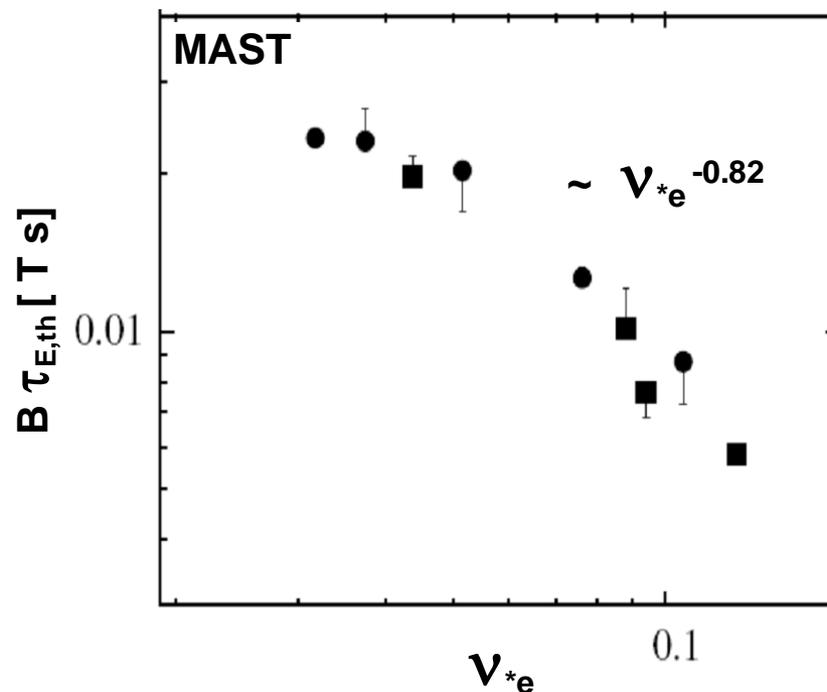
Important implications for future STs and Demo with much lower ν^*

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

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



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

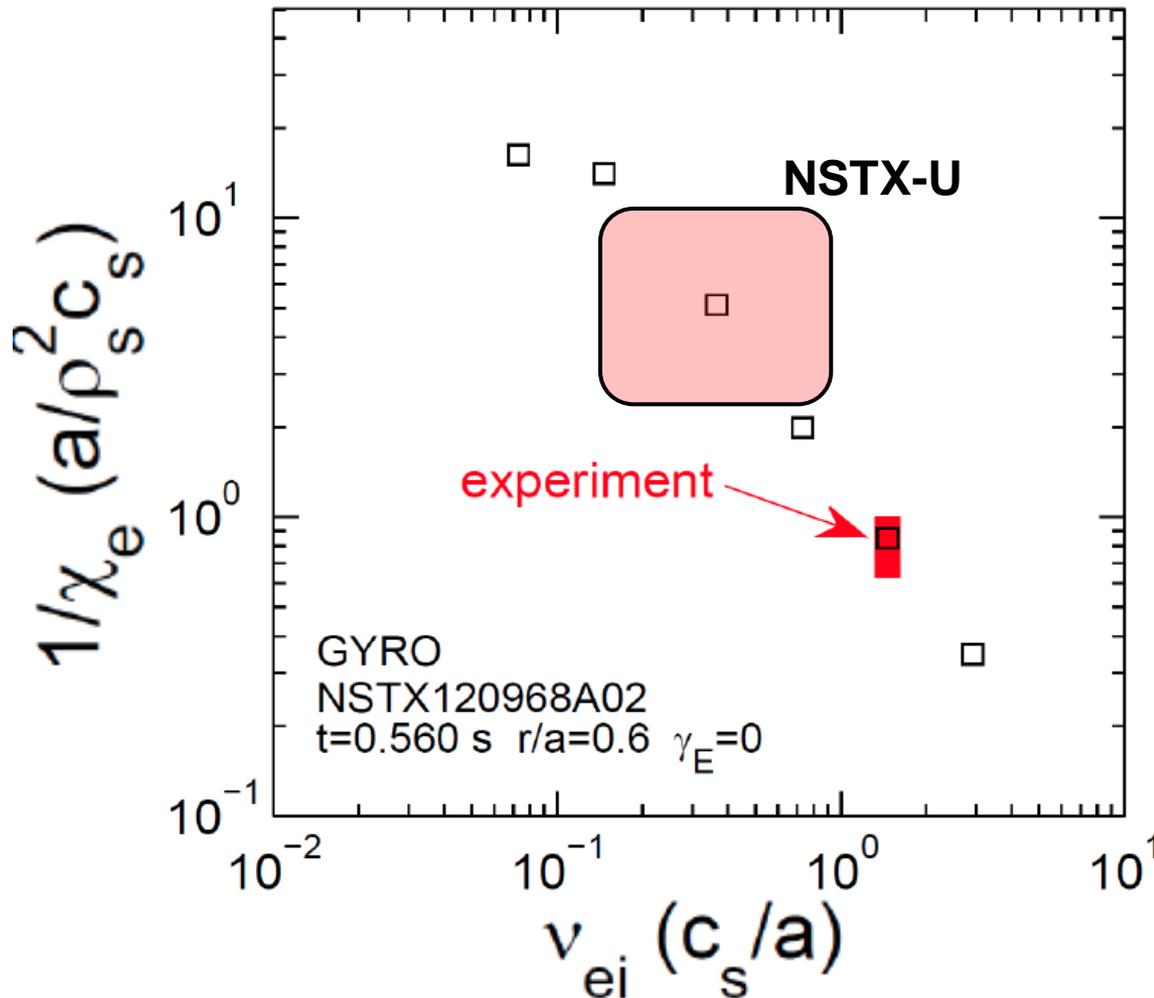


M. Valovic et al., NF (2011)

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. v_{ei} using the GYRO code.



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

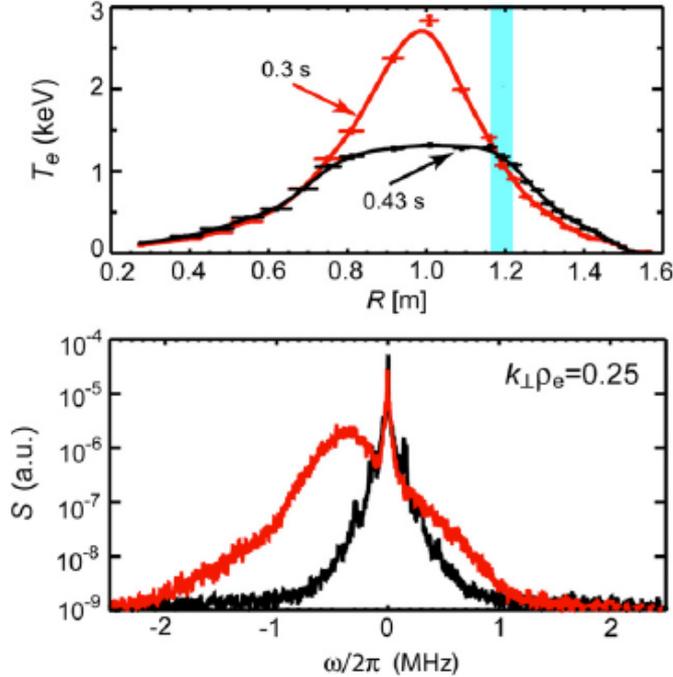
Further electron confinement improvement expected due to reduced collisionality.

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

ETGs measured for the first time with high-k scattering

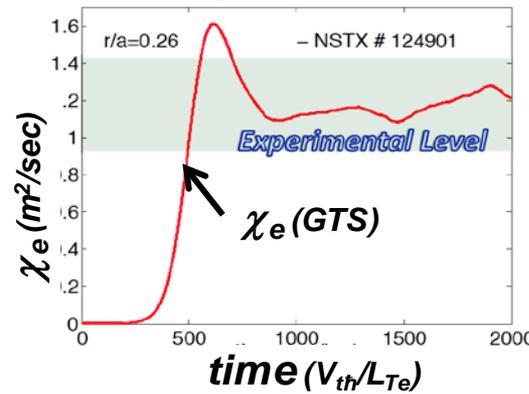
High β_e or larger $\rho_e \propto \beta_e^{0.5}$ of ST plasma enabled measurement of ETGs.

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



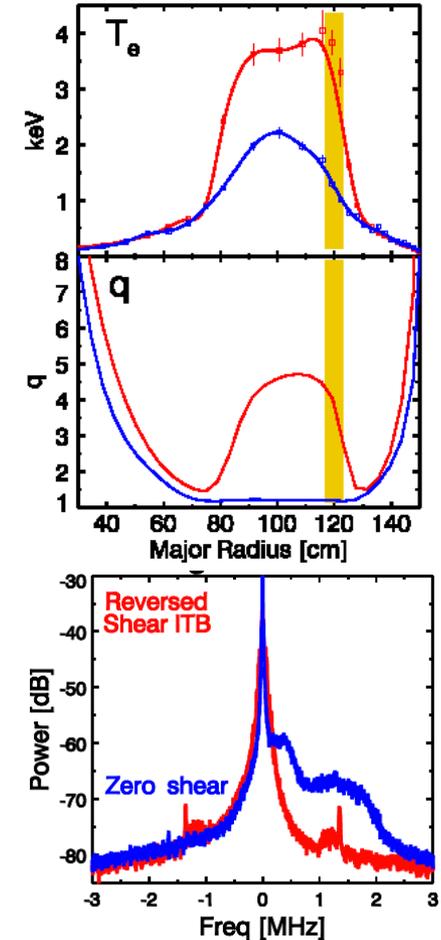
E. Mazzucato et al., NF (2009)

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

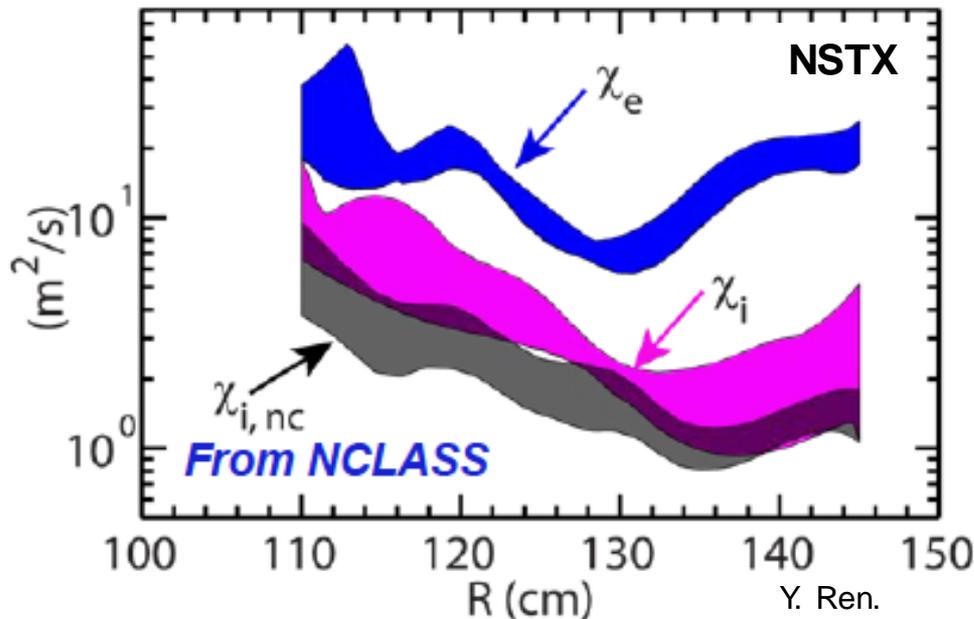
High confinement H-mode in the range of FNSF obtained

Fusion gain Q depends strongly on “ H ”, $Q \propto H^{5-7}$

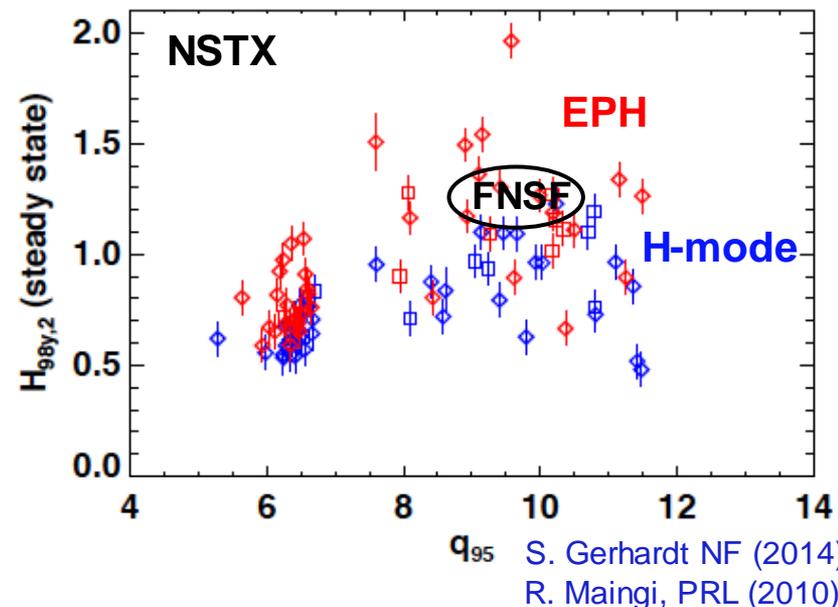
Higher H enables compact ST-FNSF $H = 1.2 - 1.3$

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

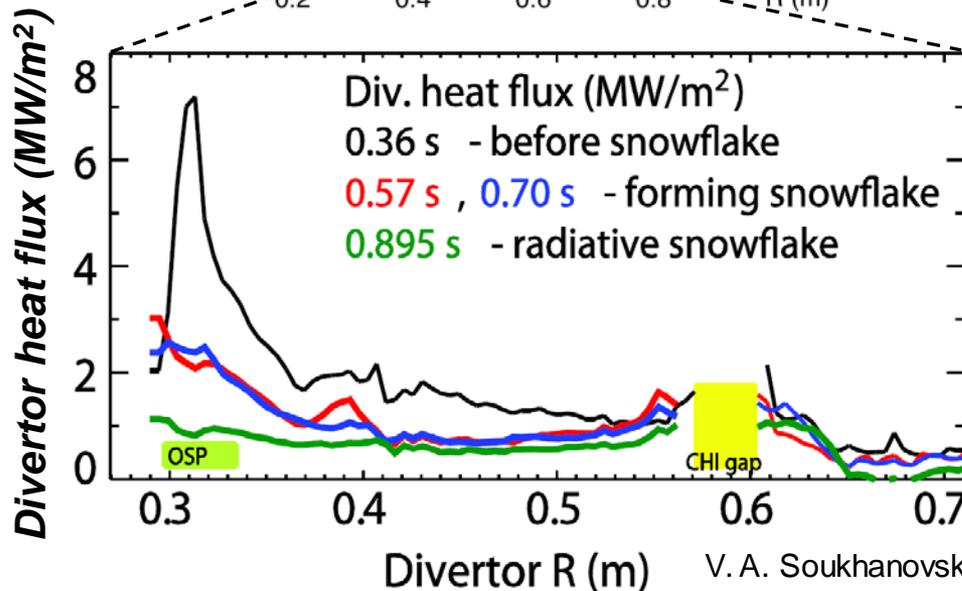
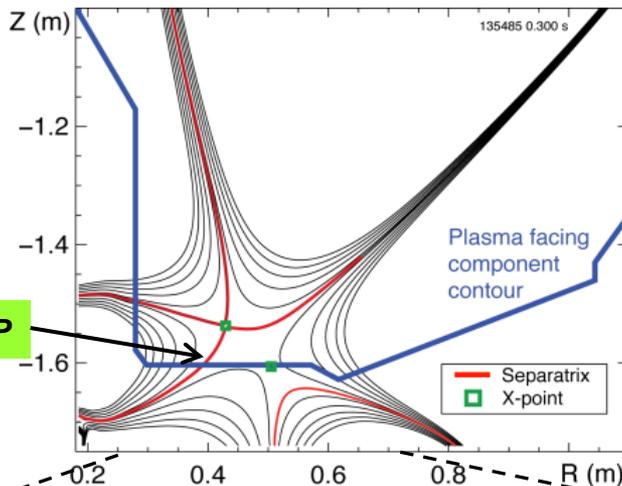


H-mode confinement in STs $H \sim 1$ (ITER98_{y,2}) but enhanced pedestal H-mode (EPH) has 50% higher H up to $H \sim 2$



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

Snowflake divertor in NSTX



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

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

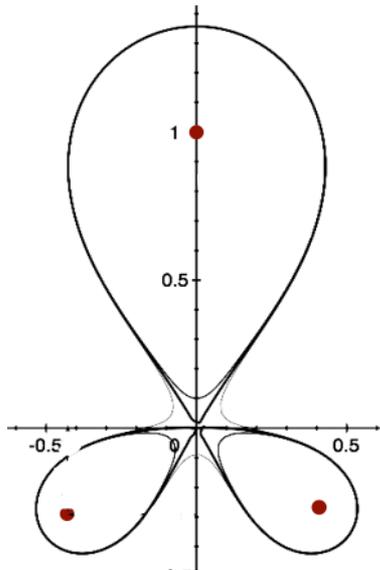
V. A. Soukhanovskii et al., PoP (2012)

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

Innovative Heat Flux Mitigation via Divertor Flux Expansion

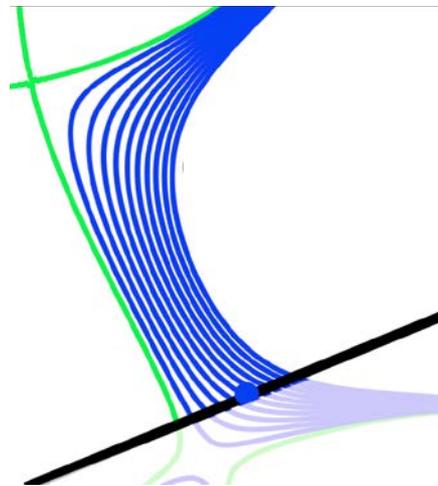
Lower toroidal field of outboard divertor leg of STs facilitates heat flux mitigation by divertor flux expansion solutions

Snow-flake



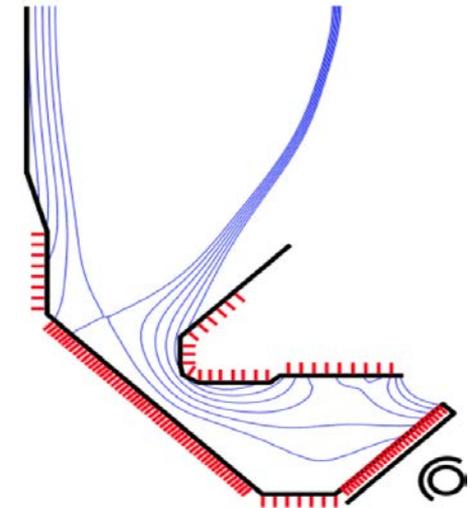
D. Ryutov, et al., PoP (2007)

X-Divertor: CREST



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

Super-X: MAST-U



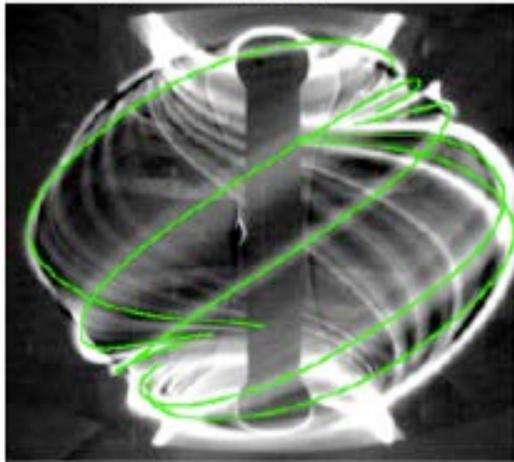
Kotschenreuther, et al., PoP (2007)

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

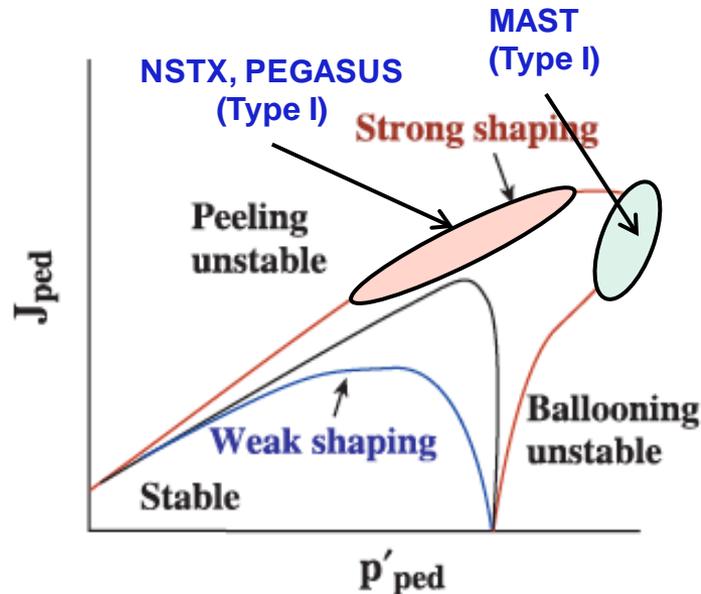
Video images of MAST plasmas showing a filamentary ELM structure.



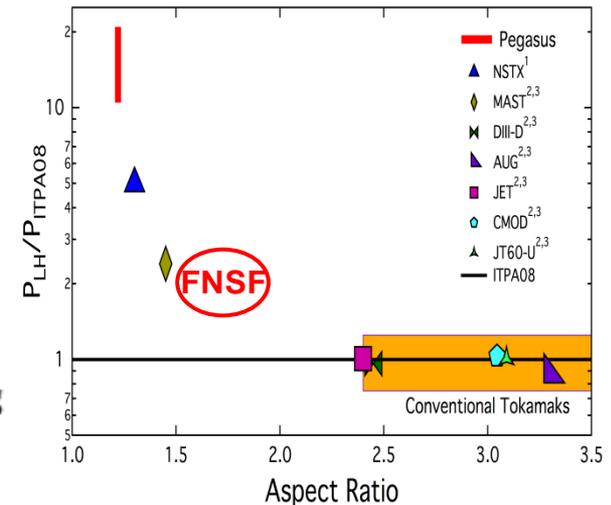
N. Ben Ayed et al., PPCF (2009).

ST is in strongly shaped ELM regimes

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



L-H power threshold scaling extended for low A



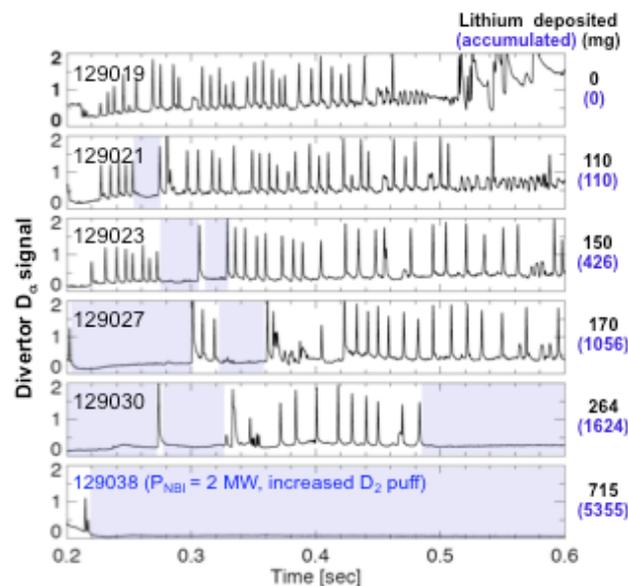
K.E. Thome et al., EPR (2014)

- 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

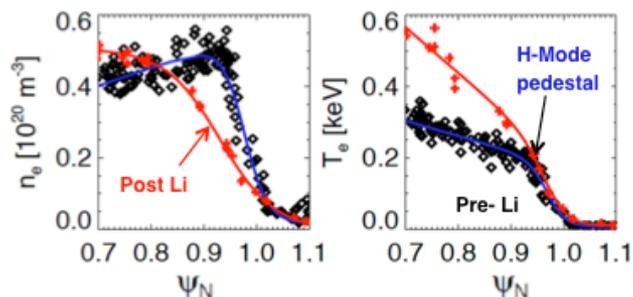
ELM Stabilization and Mitigation

Through application of lithium and 3-D fields

ELMs stabilized with edge pressure modification with Li in NSTX

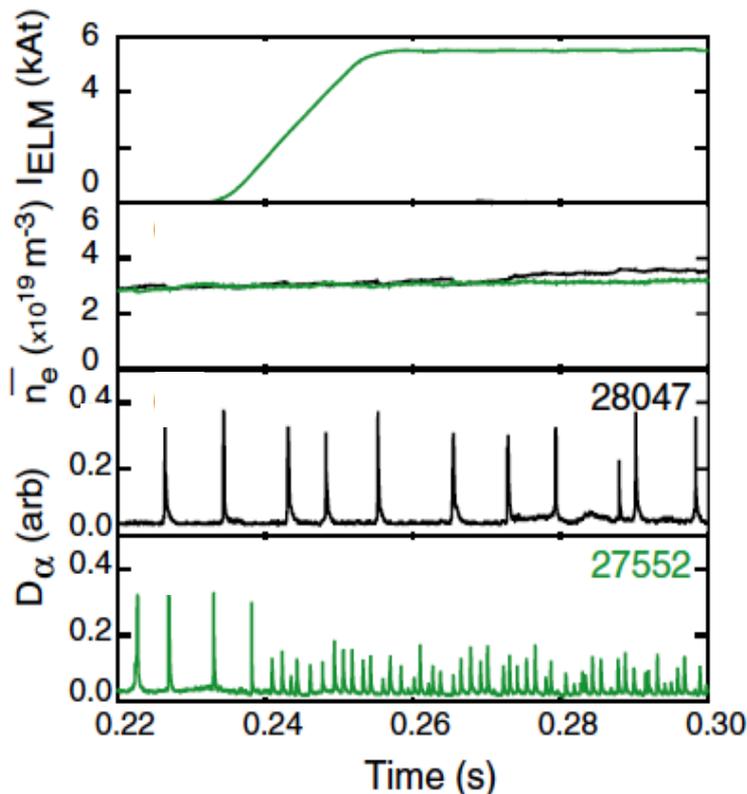


D.K. Mansfield, et al., JNM (2009)



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

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



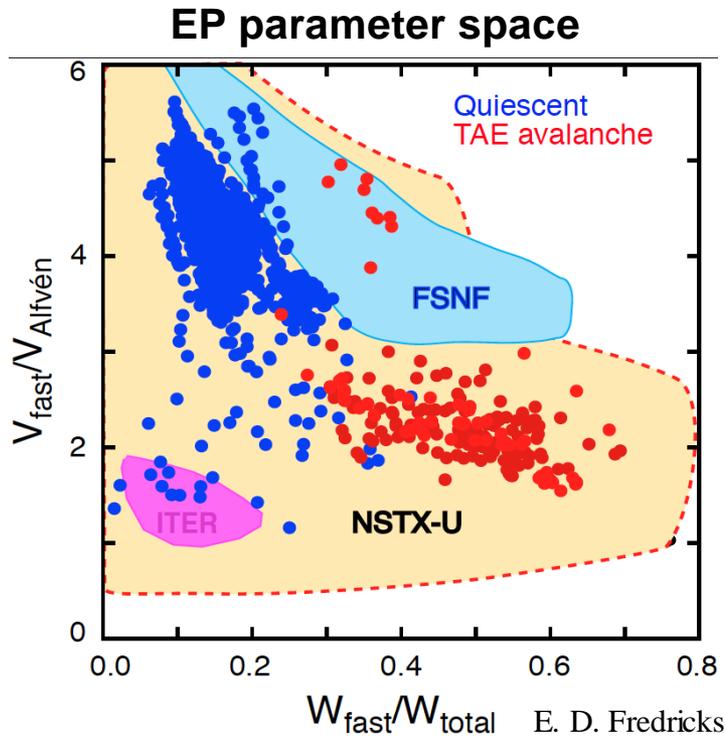
Increasing Type I ELM freq. by x 8 (900 Hz) has reduced heat flux

A. Kirk et al., NF (2013)

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

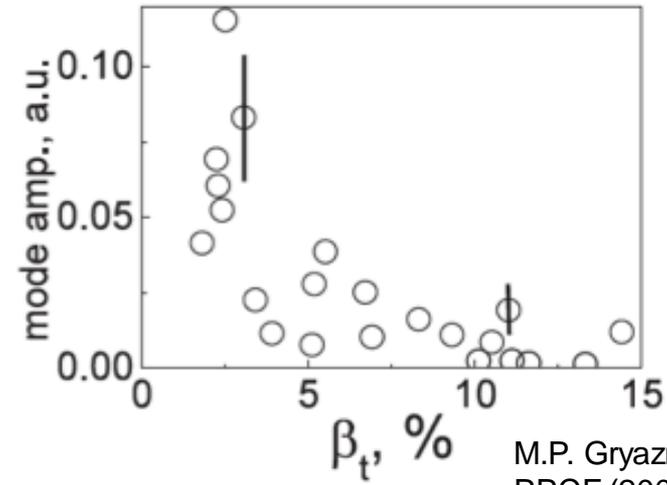
Alfvenic modes readily accessed due to high $V_\alpha > V_{Alf}$

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



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

Stabilization of TAEs at high β in MAST

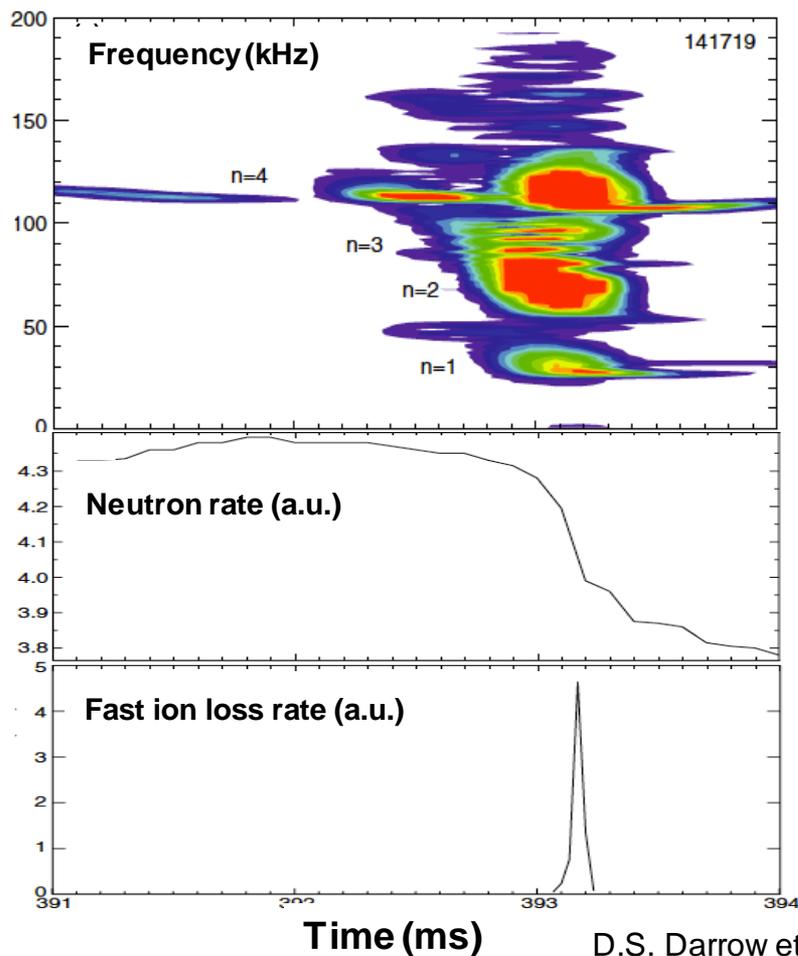


M.P. Gryaznevich et al. PPCF (2004)

“TAE avalanche” shown to cause energetic particle loss

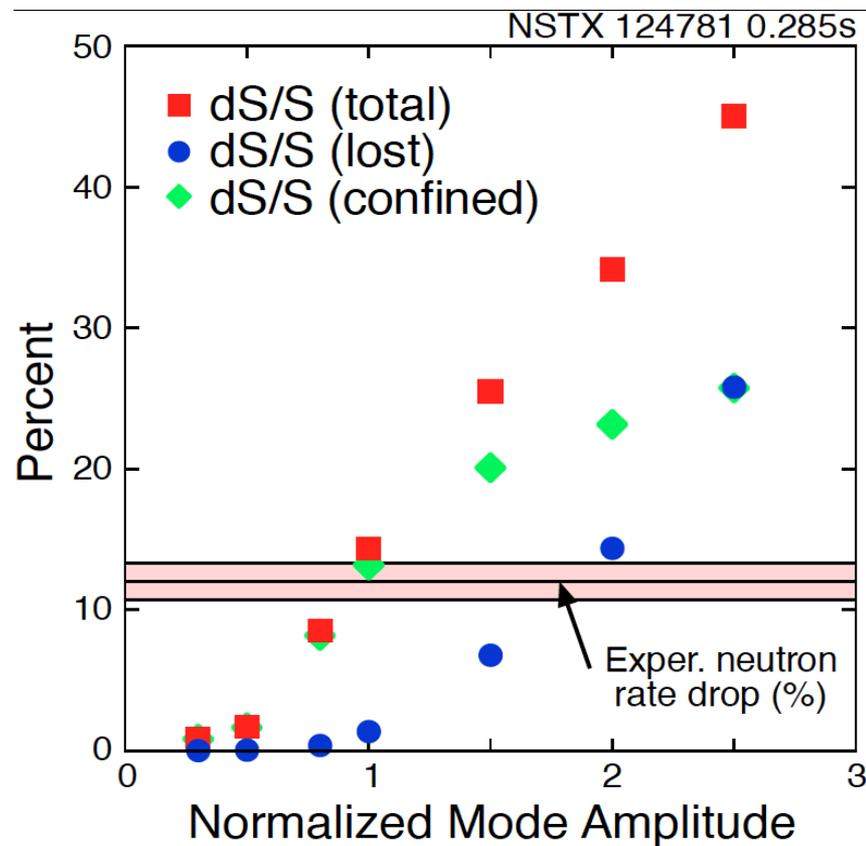
Uncontrolled α -particle loss could cause reactor first wall damage

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



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

Progress in simulation of neutron rate drop due to TAE avalanche

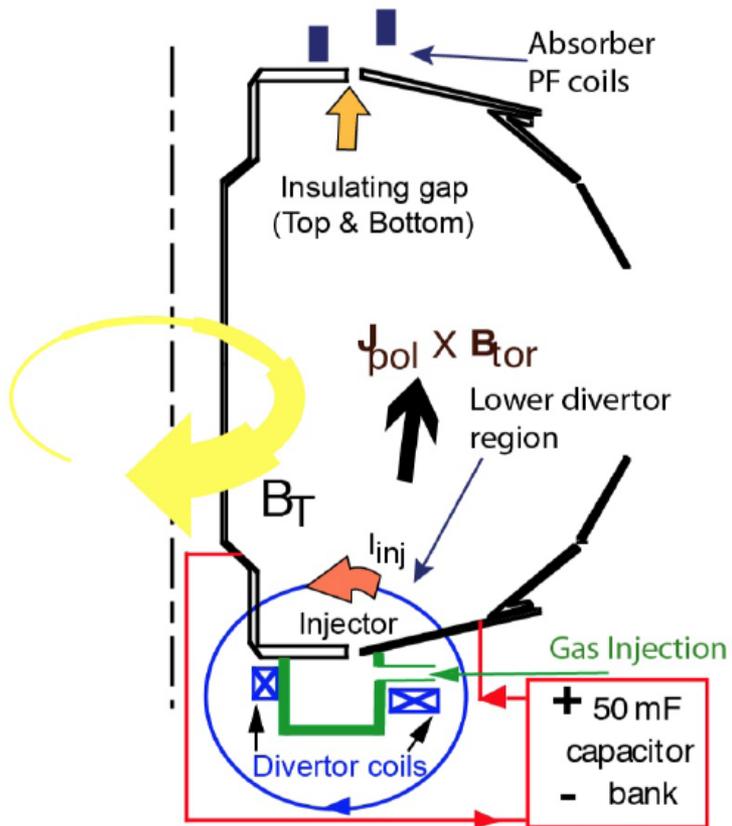


E. D. Fredrickson et al., NF (2013)

Helicity Injection Is an Efficient Method for Current Initiation

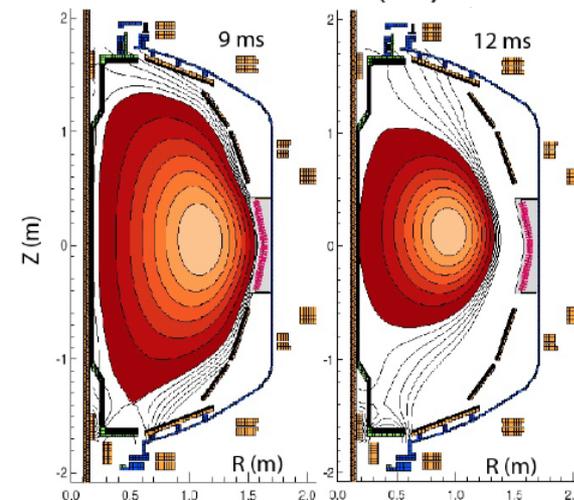
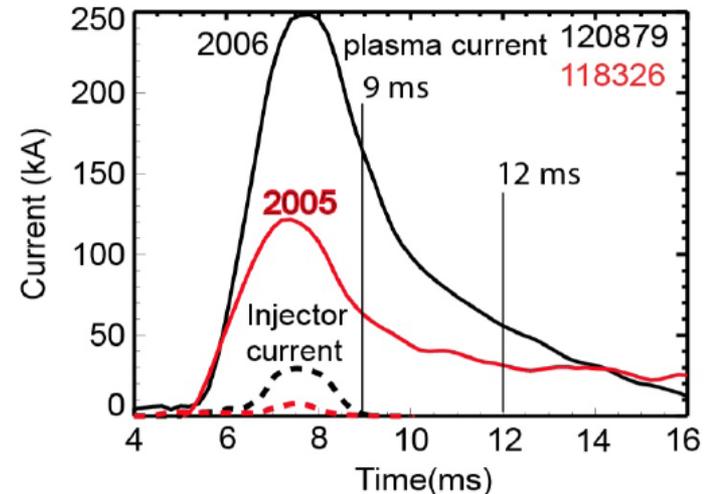
Coaxial Helicity Injection (CHI) Concepts Being Developed

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



R. Raman et al., PRL (2006)

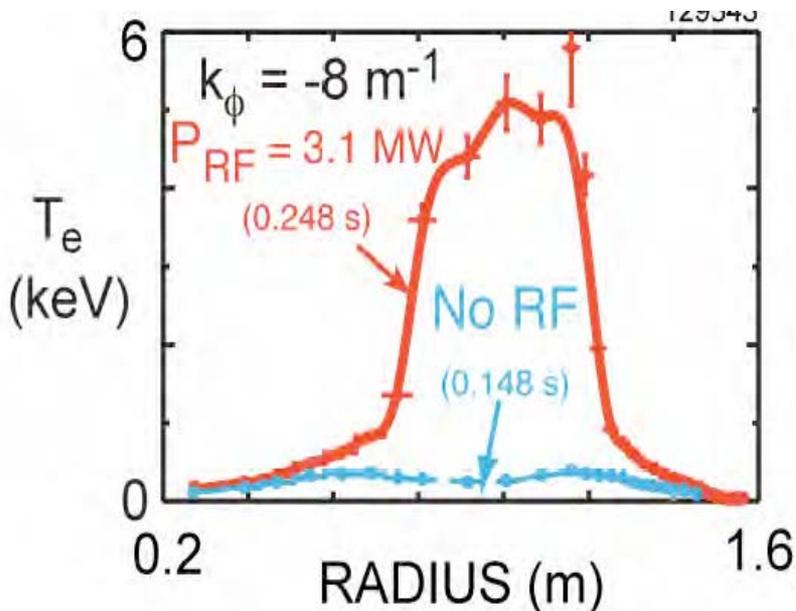
Discharge evolution of 160 kA closed flux current produced by CHI alone in NSTX



Current Ramp-Up and Profile Control Crucial for FNSF

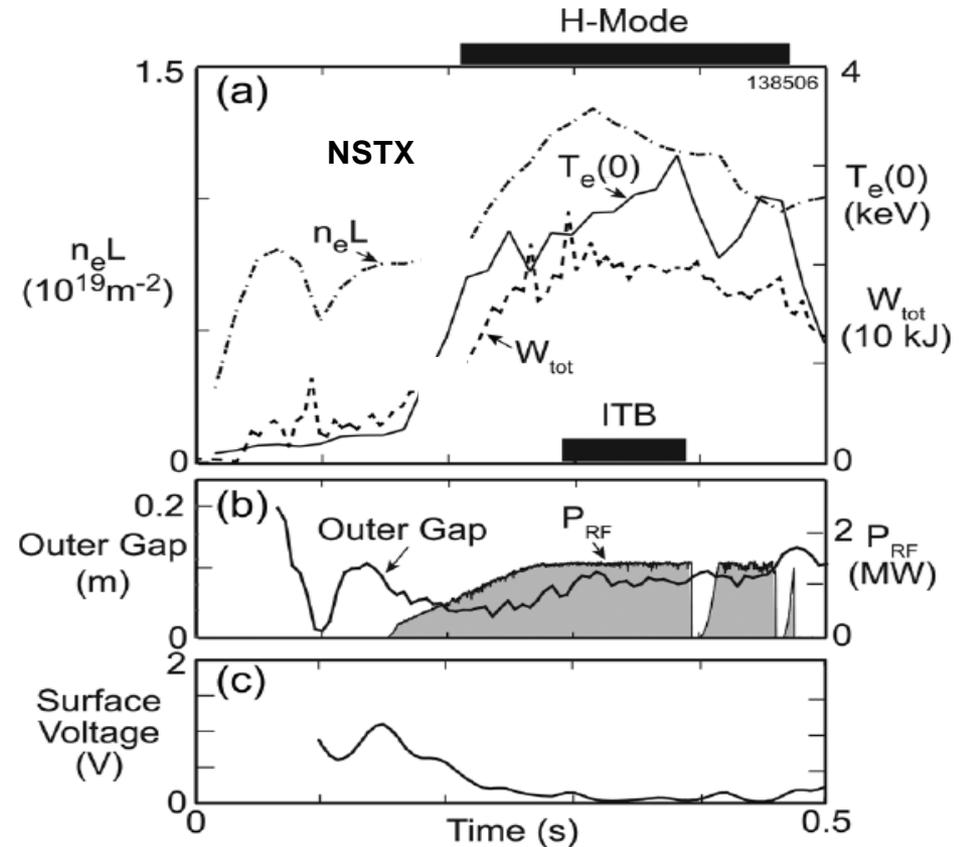
Major Research Topics for NSTX-U

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

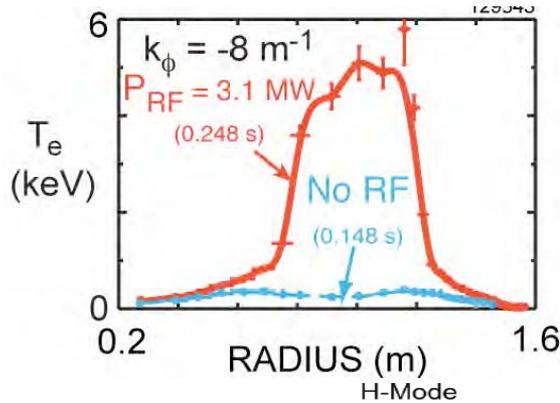


HHFW current ramp-up will be tested in NSTX-U at higher power ~ 4 MW.

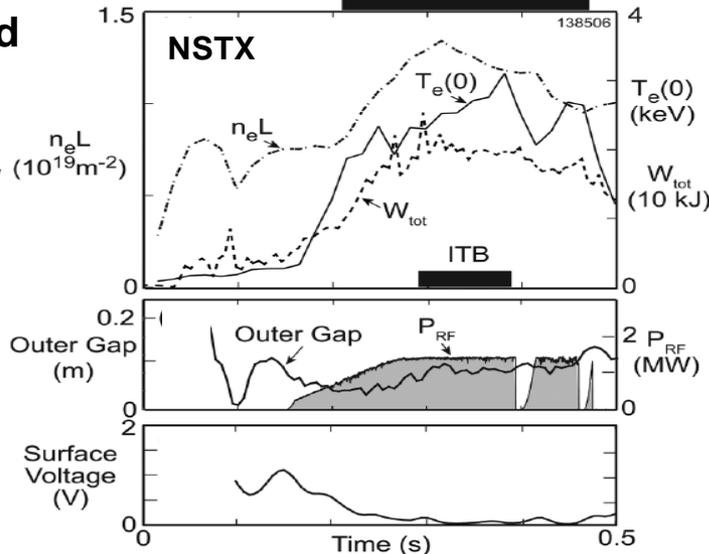
Current Ramp-Up and Profile Control Crucial for FNSF

Major Research Topics for MAST-U and NSTX-U

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



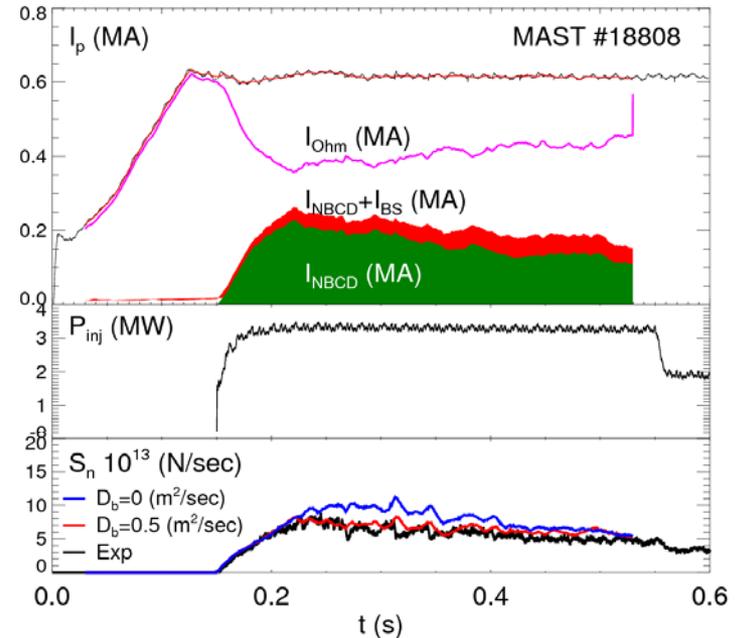
Near sustained discharges obtained with modest HHFW power.



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



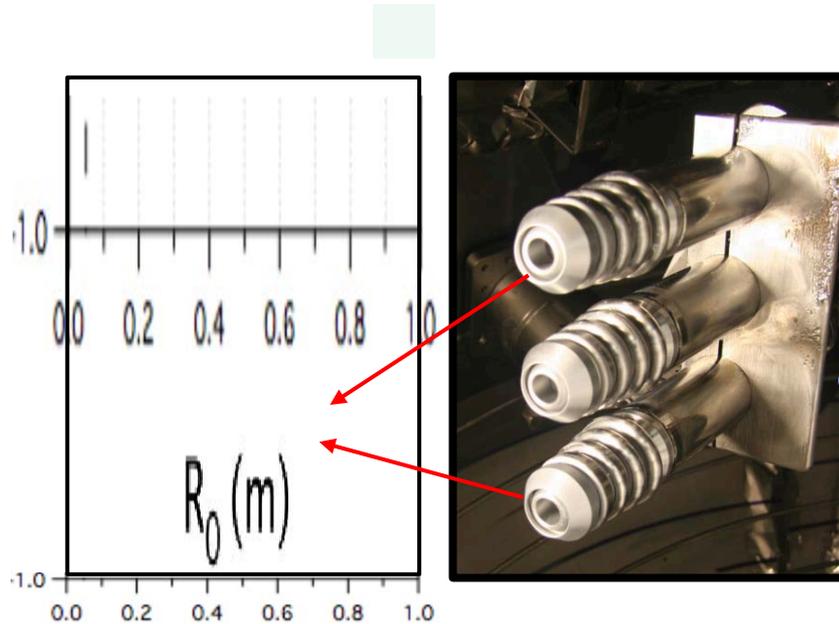
M. Turnyanskiy et al., NF (2009)

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

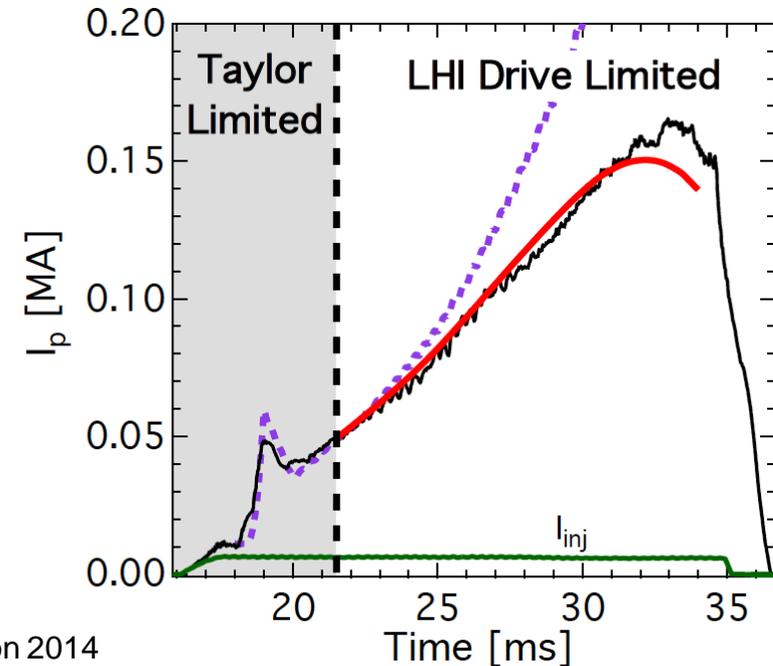
3-6 kA current injector array in plasma SOL



R_0 (m)

J. O'Bryan, PhD. Thesis, UW Madison 2014

Long-Pulse Startup Demonstrated in PEGASUS with LHI



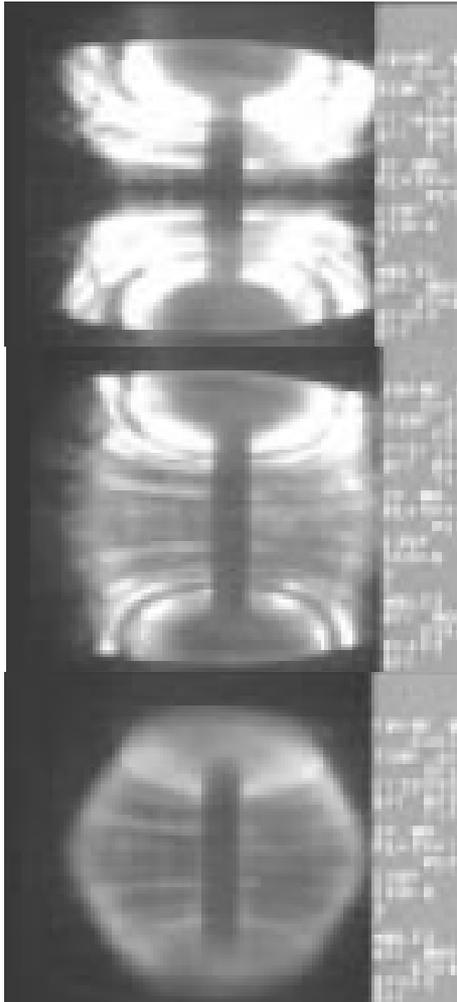
Improving predictive capability for both CHI and LHI

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

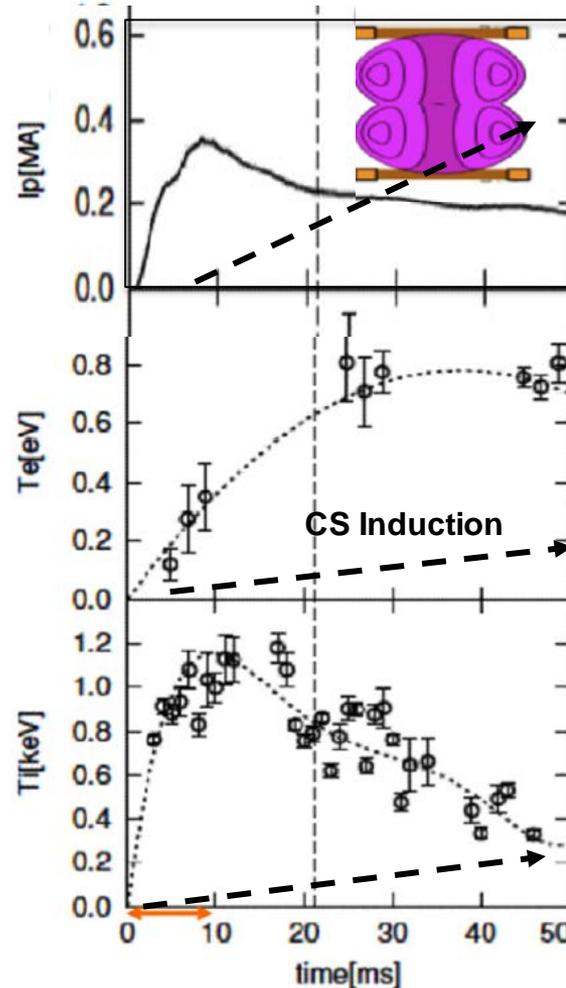
Merging Start-Up Yielded High Current STs

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

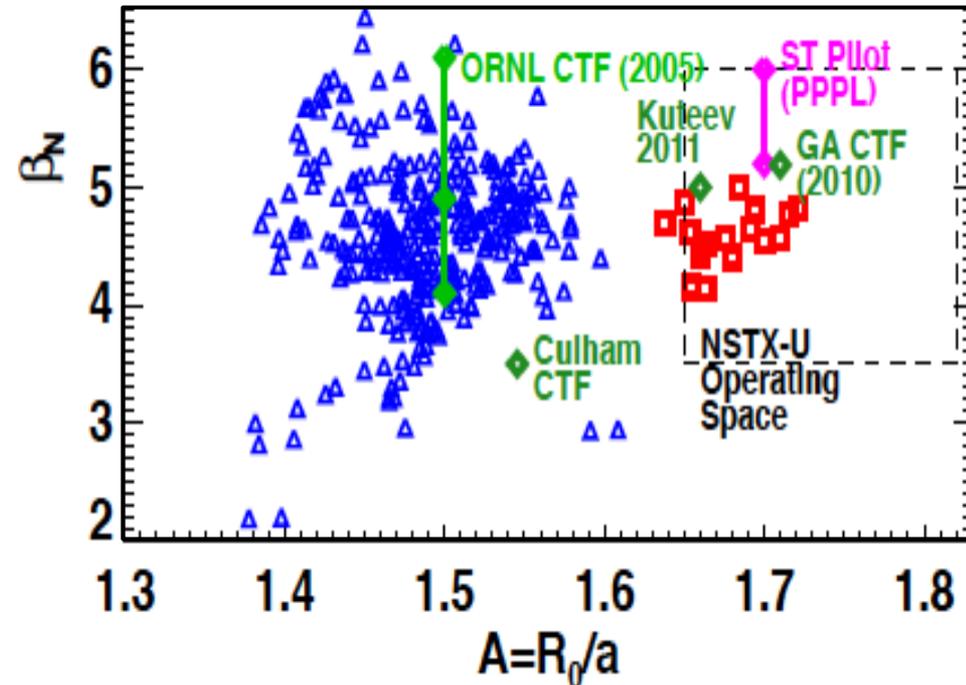
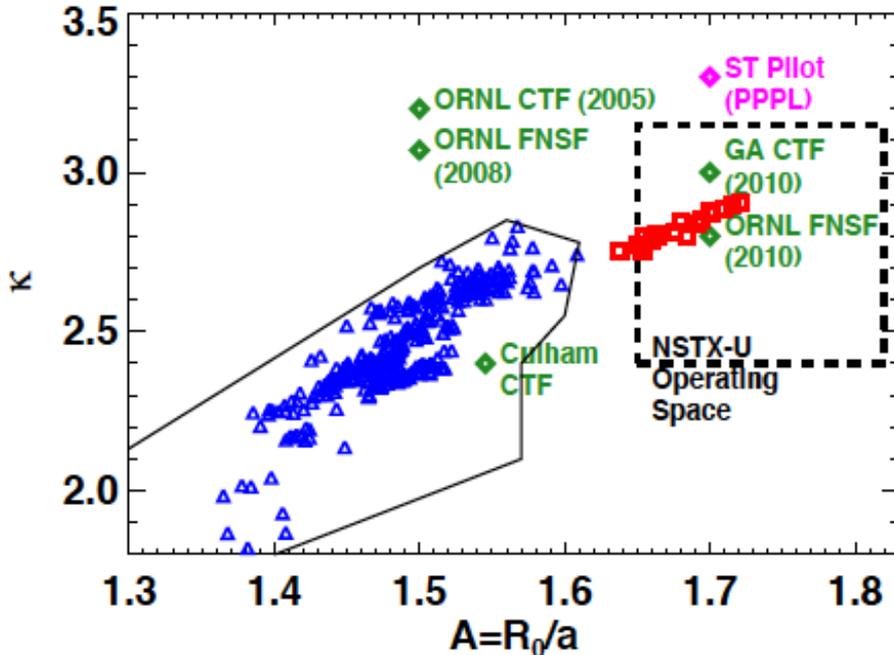
Ultra high β STs
produced by
mergings in TS-3
Device

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

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

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

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



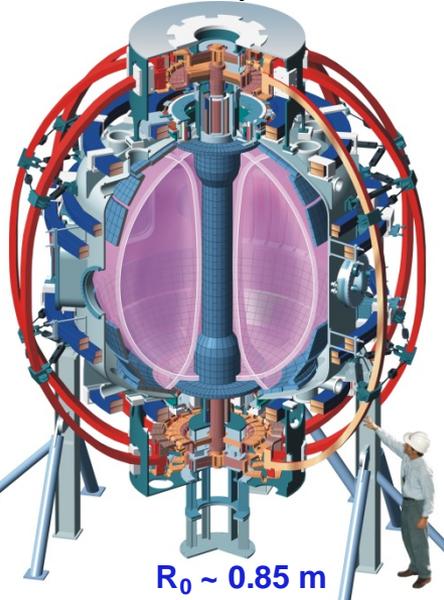
S.P. Gerhardt et al., NF (2011)

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

Operating ST Research Facilities Since 2000

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

NSTX, USA

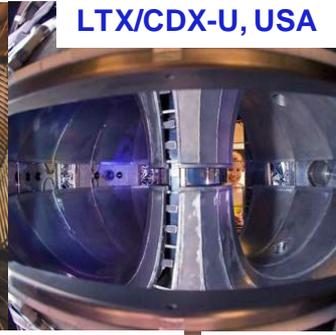


$R_0 \sim 0.85$ m

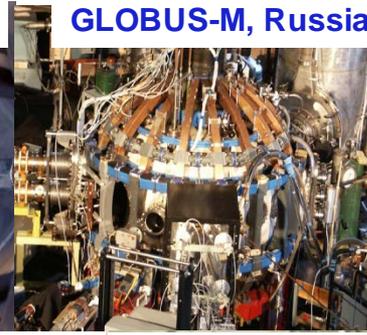
VEST, Korea HIT-II, USA



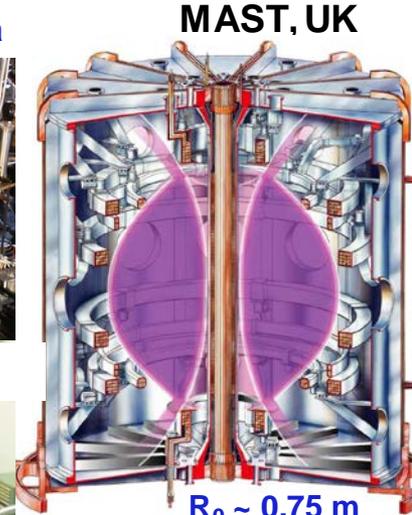
PEGASUS, USA



LTX/CDX-U, USA



GLOBUS-M, Russia

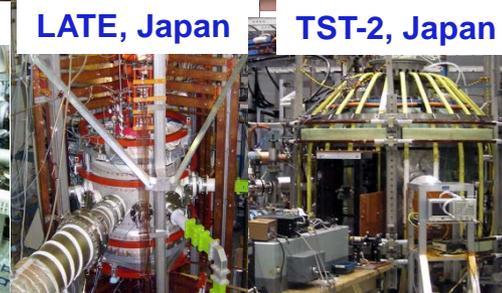


MAST, UK

$R_0 \sim 0.75$ m
QUEST/CPD, Japan

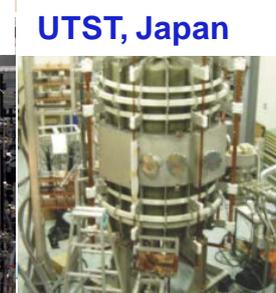


HIST, Japan

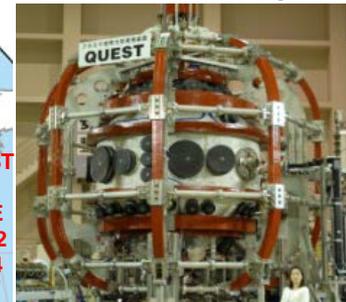


LATE, Japan

TST-2, Japan



UTST, Japan



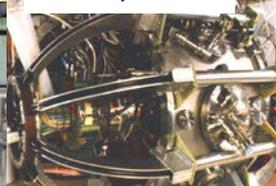
TS3/4, Japan



★ STs operated since 2000



SUNIST, China



ETE, Brazil

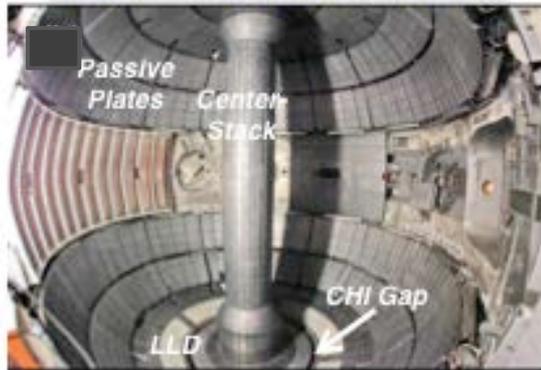
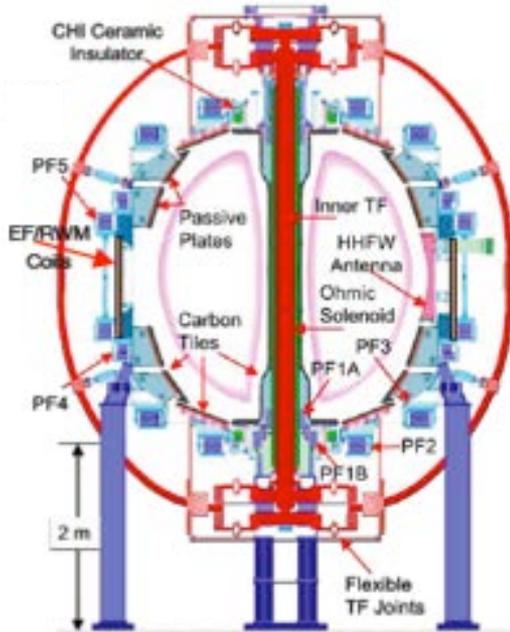


ST25, UK

MA-Class ST Research Started in 2000

Complementary Physics Capabilities of NSTX and MAST

NSTX



Complementary Capabilities

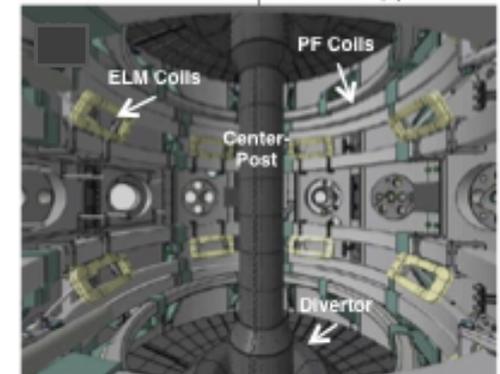
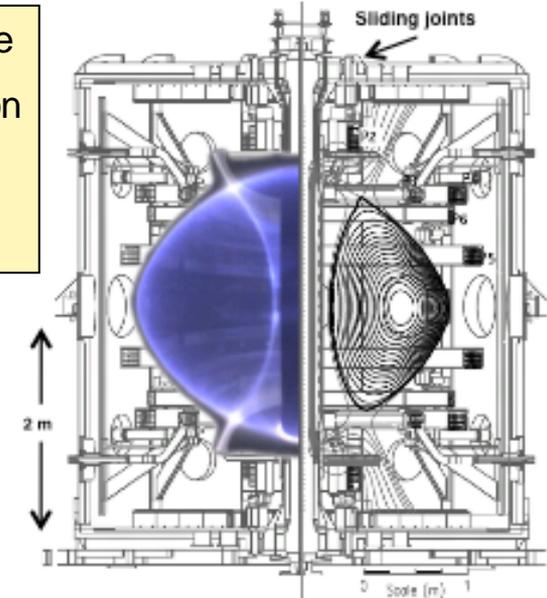
Passive Plates	Large divertor volume
Helicity Injection	Merging/Compression
HHFW	ECH
1 x 6 RWM Coils	2 x 12 ELM coils

Similar Capabilities

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

Comprehensive diagnostics
Physics integration
Scenario development

MAST



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

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