Motivations for Spherical Torus research and initial results from NSTX Upgrade

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Thanks to many contributions from ST community and NSTX-U Researchers

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Outline

• Introduction
• Why spherical tori / tokamaks (STs)
• NSTX-U initial results, long-term directions
• Summary
Why fusion?

“D-T” fusion reaction:

- High energy gain ≈ 1000 ×
- No runaway reactions
- Abundant fuel supply
- Waste short-lived, low-level
- No CO₂ production

D: deuterium  n: neutron
T: tritium  α: helium

E = mc²
Fusion requires very high temperatures

• Fusion is easiest here at 200 million °C (!!) (350 million °F)
  – Requires lowest pressure nT and energy confinement time $\tau_E$
  – 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)

Model

Experiment

$P_{\text{inj}} = 15 \text{ MW}$

$I_p = 1.7 \text{ MA}$

$B = 4.5 \text{ T}$

Supershot

L-Mode

core

boundary
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 \( \frac{E_{\text{fusion}}}{E_{\text{electrical}}} \sim 5\% \)
  - So far, 0.006% efficient

- Magnetic fusion in ITER:
  - Goal: 500MW fusion power for \( \leq 600\text{MW electrical input for 400s} \)
    - Industrial levels of fusion power
Tokamaks and stellarators are the leading configurations in magnetic fusion

Superconducting tokamak

KSTAR (South Korea)

Superconducting stellarator

W7-X (Germany) – 1st run campaign in 2016

• 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”
    • Potential show-stopper for tokamak power-plant

• 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: $\frac{1}{5}$ in alphas, $\frac{4}{5}$ in neutrons
- ITER goal $Q = \frac{P_{\text{fusion}}}{P_{\text{external heating}}} = 10$
- $Q = 10$ $\Rightarrow$ $\frac{P_{\alpha}}{P_{\text{external}}} = 2$
- $\frac{P_{\alpha}}{P_{\alpha + \text{external}}} = \frac{2}{3} > 50$

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

These are large numbers
Perspective

• Studying burning plasmas is essential to fusion development, and ITER is presently the best approach

• But as we look beyond plasma self-heating toward economical electricity production, how might we improve?
Fusion power density $\propto (\text{plasma pressure})^2$

$\beta \equiv \text{plasma pressure} / \text{magnetic pressure} = p/(B^2/2\mu_0)$

Maximum $\beta$ limited by MHD instabilities

B limited by magnet stress, cooling, quench

Fusion power density $\propto \beta^2 B^4$

Assuming cost $\propto$ size $\rightarrow$ need higher fusion power / volume = high fusion power density

Maximize / optimize this product
Aspect ratio is an important free parameter.

Aspect ratio $A = \frac{R}{a}$

- $R =$ major radius
- $a =$ minor radius

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

Conventional tokamak typically $A = 2.5-4$
STs have higher natural elongation

Elongation $\kappa = \frac{b}{a}$

$b = \text{vertical } \frac{1}{2} \text{ height}$ \hspace{1cm} $a = \text{minor radius}$

Higher elongation improves stability, confinement
Favorable average curvature improves stability

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

Tokamak

$A \sim 2.5-4$
$\kappa = 1.5-2$
$\beta_T = 3-10\%$

ST

$A \sim 1.3-2$
$\kappa = 2-3$
$\beta_T = 10-40\%$

Plasma spends less time in unstable curvature region
STs can access very wide range of $\beta_T$

Pegasus ST recently accessed $\beta_T \sim 50-100\%$

$\beta_T$ for sustained, low-$\ell_i$, high-$\kappa$, LHI-driven plasmas

$\beta_T$, $\beta_N$ for sustained, low-$\ell_i$, high-$\kappa$, LHI-driven plasmas
Why explore spherical torus/tokamak?

• Potentially attractive for electricity production
  – Requires accompanying magnet innovations

• High neutron wall loading in small device
  – Well suited for fusion nuclear component R&D

• Improve toroidal physics predictive capability
  – High $\beta$ and high temperature at low collisionality
  – Understand confinement, fast-ion physics for ITER
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How would magnetic fusion make electricity?

Plasma
(100 million °C)

Magnets
(-270 - 100°C)

Steam or hot helium

Tritium bred from lithium in blanket

Blanket
(350 - 1000°C)

helium and deuterium exhaust

External heating
(10's of MW)

α self heating

Heat

Spin turbines, generate electricity

Neutrons
Electricity gain $Q_{\text{eng}}$ determined primarily by engineering efficiencies and fusion gain

$$Q_{\text{eng}} \equiv \frac{\text{Electricity produced}}{\text{Electricity consumed}} = \frac{\eta_{\text{th}} \left( M_n P_n + P_\alpha + P_{\text{aux}} + P_{\text{pump}} \right)}{P_{\text{aux}} + P_{\text{pump}} + P_{\text{sub}} + P_{\text{coils}} + P_{\text{control}}}$$

$$Q_{\text{eng}} = \eta_{\text{th}} \eta_{\text{aux}} Q \times \frac{(4M_n + 1 + 5/Q + 5P_{\text{pump}}/P_{\text{fus}})}{5(1 + \eta_{\text{aux}} Q P_{\text{extra}}/P_{\text{fus}})}$$

$\eta_{\text{th}}$ = thermal power conversion efficiency
$\eta_{\text{aux}}$ = injected power wall plug efficiency
$Q \equiv P_{\text{fus}} / P_{\text{aux}}$ = fusion power / auxiliary power

Parameter Assumptions:
- $M_n = 1.1$, $P_{\text{pump}} = 0.03 \times P_{\text{th}}$
- $P_{\text{sub}} + P_{\text{control}} = 0.04 \times P_{\text{th}}$
- $\eta_{\text{aux}} = 0.3$
- $\eta_{\text{CD}} = I_{\text{CD}} R_0 n_e / P_{\text{CD}} \approx 0.3 \times 10^{20} \text{A/W/m}^2$

For more details see J. Menard, et al., Nucl. Fusion 51 (2011) 103014
Gain is very strong function of confinement: 
\[ Q_{DT} \propto H^{2 \rightarrow 5} \text{ from low } \rightarrow \text{ high gain} \]

Fusion power density \( \equiv \Gamma_{DT} = n_D n_T \langle \sigma \nu \rangle_{DT} E_{DT} \propto p^2 \)
\[ P_{fusion} \propto \left( P \tau_E \right)^2 / V \]
\[ \tau_E \propto H I_P^{\alpha_I} B_T^{\alpha_B} n_e^{\alpha_n} P^{-\alpha_P} R^{\alpha_R} \kappa^{\alpha_\kappa} \epsilon^{\alpha_\epsilon} \quad \epsilon \equiv A^{-1} \]
\[ P = P_{aux} (1 + \lambda_{DT} Q_{DT}) \quad Q_{DT} \equiv P_{fusion} / P_{aux} \quad \lambda_{DT} = 0.2 \]
\[ Q_{DT}^* \equiv Q_{DT} / (1 + \lambda_{DT} Q_{DT})^{2(1-\alpha_P)} \]
\[ \propto H^2 I_P^{2\alpha_I} B_T^{2\alpha_B} n_e^{2\alpha_n} P_{aux}^{1-2\alpha_P} R^{2\alpha_R-3} \kappa^{2\alpha_\kappa-1} \epsilon^{2\alpha_\epsilon-2} \]

Fix current, field, density, geometry, auxiliary power, \( \alpha_p = 0.7 \):
\[ Q_{DT} \leq 1 \rightarrow Q_{DT} \approx Q_{DT}^* \propto H^2 \quad Q_{DT} \gg 1 \rightarrow Q_{DT} \propto Q_{DT}^* 2.5 \propto H^5 \]
Steady-state tokamaks: current-driven kink limit less relevant

Normalized $\beta (\beta_N)$ and “bootstrap” fraction ($f_{BS}$) more important

Relevant variables are $\beta_N / f_{BS}$ and normalized density $f_{gw}$

Gain vs. physics & engineering constraints

$Q_{DT}^* \propto H^2 (\beta_N / f_{BS})^{C_\beta} B_T^{C_B} f_{gw}^{C_gw} P_{aux}^{C_p} R^{C_R} \kappa^{C_\kappa} \epsilon^{C_\epsilon}$

Use electrostatic gyro-Bohm $\tau_E$ scaling with no $\beta$ degradation (NSTX, JET, DIII-D)

$Q_{DT}^* \propto R^2 H^2 (1 - f_{CD})^{-2} f_{gw}^{0.7} B_T^3 \kappa^{3-5} \beta_N^2 \epsilon^{1.6}$

External current drive fraction

Need to optimize this product vs. aspect ratio
High temperature superconductors (HTS) could substantially expand fusion magnet performance.

http://magnet.fsu.edu/~lee/plot/plot.htm
Cables formed from HTS tapes achieving high winding pack current density at high B

Conductor on Round Core Cables (CORC)

\[ J_{WP} \sim 70 \text{MA/m}^2 \text{ 19T} \]

Base cable: 50 tapes YBCO  Tapes with 38 mm substrate (Van Der Laan, HTS4Fusion, 2015)

Higher \( J_{cable} \) HTS cable concepts under development:

Base Conductor
He Gas Cooled

\[ 8 \text{kA}, \quad J_{WP} \sim 160 \text{MA/m}^2 \]
High current density HTS cable motivates consideration of lower-A tokamak pilot plants

Fix plasma major radius $R_0=3m$, heating power $P_{NNBI}=50MW$

- **ITER-like TF magnets:**
  - $J_{WP}=20MA/m^2$, $B_{max} \leq 12T$
  - $P_{fusion} \leq 130MW$, $P_{net} < -90MW$

- **$J_{WP} \sim 30MA/m^2$, $B_{max} \leq 19T$**
  - $P_{fusion} \sim 400MW$
  - Small $P_{net}$ at $A=2.2-3.5$

- **$J_{WP} \geq 70MA/m^2, B_{max} \leq 19T$**
  - $P_{fusion} \sim 500-600MW$
  - $P_{net} = 80-100MW$ at $A=1.9-2.3$

$A \sim 2$ attractive at high $J_{WP}$
\( A = 1.8-2.3 \) maximizes TF magnet utilization, and TF will be significant fraction of core cost

\[
\frac{\text{Fusion power}}{\text{TF coil volume}}
\]

\( J_{WP} = 70\text{MA/m}^2 \)

Eff. shield thickness:
- 0.3m
- 0.4m
- 0.5m
- 0.6m
- 0.7m

\( A = 4 \) utilization
- \( \sim 1/3 \text{-} 1/2 \) of max at low \( A \)
$A \geq 3$ maximizes blanket volume utilization

Fusion power / blanket volume

Which components - magnets, blankets, ... dominate cost ?
Costing of these (never-been-built) objects has large uncertainty

Eff. shield thickness:
- 0.3m
- 0.4m
- 0.5m
- 0.6m
- 0.7m

$J_{WP} = 70 \text{MA/m}^2$
A=2, $R_0 = 3m$ HTS-TF FNSF / Pilot Plant

- $B_T = 4T$, $I_P = 12.5MA$
- $\kappa = 2.5$, $\delta = 0.55$
- $\beta_N = 4.2$, $\beta_T = 9\%$
- $H_{98} = 1.8$, $H_{Petty-08} = 1.3$
- $f_{gw} = 0.80$, $f_{BS} = 0.76$
- Startup $I_P$ (OH) $\sim 2MA$
- $J_{WP} = 70MA/m^2$
- $B_{T-max} = 17.5T$
- No joints in TF
- Vertical maintenance

- $P_{fusion} = 520$ MW
- $P_{NBI} = 50$ MW, $E_{NBI} = 0.5$MeV
- $Q_{DT} = 10.4$
- $Q_{eng} = 1.35$
- $P_{net} = 73$ MW

- $\langle W_n \rangle = 1.3$ MW/m$^2$
- Peak n-flux = 2.4 MW/m$^2$
- Peak n-fluence = 7 MWy/m$^2$

Cryostat volume $\sim 1/3$ of ITER
Why explore spherical torus/tokamak?

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- High neutron wall loading in small device
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- Improve toroidal physics predictive capability
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Fusion technology development is major challenge
Fusion Nuclear Science Facility (FNSF) could aid development

Need to develop reliable and qualified nuclear components unique to fusion:
- Divertor and plasma facing components
- Blanket and first wall
- Vacuum vessel and shielding
- Tritium fuel cycle
- Remote maintenance

- 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, reliable
Design studies show ST potentially attractive as FNSF

- Projected to access high neutron wall loading at moderate $R$, $P_{\text{fusion}}$
  - $W_n \sim 1\text{-}2\text{ MW/m}^2$
  - $P_{\text{fus}} \sim 50\text{-}200\text{MW}$
  - $R \sim 0.8\text{-}1.8\text{m}$
- Modular design, maintenance
- Tritium breeding ratio (TBR) near 1
  - Requires sufficiently large $R$
  - Careful layout / design
$R \geq 1.7\text{m}$ necessary for net breeding at $A=1.7$

- $R=1.7\text{m}: \quad \text{TBR} \geq 1$
- $R=1.0\text{m}: \quad \text{TBR} < 1 \ (\approx 0.9)$

- Need to purchase Tritium from outside sources:
  - $-\$12-55\text{M} / \text{full power year (FPY)}$

*TBM = Test Blanket Module*  
*MTM = Materials Testing Module*
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NSTX Upgrade Device and Test Cell – Aerial View
NSTX Upgrade will access new physics with 2 major new tools:

1. New Central Magnet
   - Higher $T$, low $\nu^*$ from low to high $\beta$
     - Unique regime, study new transport and stability physics

2. Tangential 2nd Neutral Beam
   - Full non-inductive current drive
     - Not demonstrated in ST at high-$\beta_T$
     - Essential for any future steady-state ST
NSTX / MAST confinement increased at higher $T_e$ (!)
Will confinement trend continue, or look like conventional A?

Normalized electron collisionality $\nu_e^* \propto \frac{n_e}{T_e^2}$

Favorable confinement results could lead to more compact ST reactors

Low $\nu^*$ $\Rightarrow$ need higher plasma current, toroidal field, heating power, density control
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.
NBI-heated STs excellent testbed for $\alpha$-particle physics

- NSTX-U: large fast-ion dynamic range spanning ST and conventional A
  - Toroidal field $2 \times$ NSTX $\rightarrow$ $V_{\text{fast}} < V_A$ $\rightarrow$ stabilize modes
  - Tangential 2$^{\text{nd}}$ NBI $\rightarrow$ very flexible fast-ion distribution
    - Vary pitch angle, pressure profile

Can we find TAE-quiescent, high-performance regimes in NSTX-U?  
And predict fast-ion confinement for ITER scenarios?
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)

- Separatrix (i.e LCFS)
- Closed magnetic surfaces
- Open magnetic surfaces
- Scrape-off layer
- Strike points
- X-point
- Private plasma
- Divertor plates

Power / particles contact target plates here
Tokamak + ST data: power exhaust width varies as $1 / B_{\text{poloidal}}$

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

ST data breaks aspect ratio degeneracy of data set

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

Issue: If peak heat flux in divertor region exceeds $\sim 10 \text{ MW/m}^2 \rightarrow$ material damage
NSTX-U will have major boost in performance

**1. New Central Magnet**
- 2× toroidal field (0.5 → 1T)
- 2× plasma current (1 → 2MA)
- 5× longer pulse (1 → 5s)

**2. Tangential 2nd Neutral Beam**
- 2× heating power (5 → 10MW)
  - Tangential NBI → 2× current drive efficiency
- 4× divertor heat flux (→ ITER levels)
- Up to 10× higher nTτ_E (~MJ plasmas)
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NSTX-U had scientifically productive 1st year

- Achieved H-mode on 8th day of 10 weeks of operation
- Surpassed magnetic field and pulse-duration of NSTX
- Matched best NSTX H-mode performance at ~1MA
- Identified and corrected dominant error fields
- Commissioned all magnetic and kinetic profile diagnostics
- Injected up to 12MW NBI power into armor by end of run
- Discovered new 2nd NBI modifies several fast-ion modes
- Implemented techniques for controlled plasma shut down, disruption detection, commissioned new tools for mitigation

- 2016 run ended prematurely due to fault in divertor PF coil
  - Coil forensics, Extent of Condition \(\rightarrow\) new coil fab, other repairs
  - Aim to resume plasma operation during 2018 – but timing still TBD
NSTX-U has surpassed maximum pulse duration and magnetic field of NSTX.

Compare similar NSTX / NSTX-U Boronized L-modes, $P_{\text{NBI}}=1\text{MW}$

Plasma current [MA]

Toroidal magnetic field [T]

NSTX-U L-mode duration exceeds longest NSTX H-mode

NSTX-U $B_T >$ highest NSTX $B_T$
Accessed high elongation $\kappa$ using progressively earlier H-mode and heating + optimized EFC.

• **Goal:** Internal inductance $I_i = 0.5-0.7 \Rightarrow \kappa = 2.4-2.7$
Recovered ~1MA H-modes with performance comparable to best NSTX plasmas at similar current

\[ H_{98} \geq 1, \beta_N \sim 3.5-4 \geq n=1 \text{ no-wall limit} \]
H-mode confinement > ITER scaling, consistent with ST scaling (so far) – need higher $I_p$, $B_T$ to test
Fast-ion confinement measured to be at / near predicted values at low total NBI power ~1-2MW

\[ E_{\text{NBI}} = 85 \text{keV} \]

- Good agreement between neutron measurement and TRANSP prediction

\[ E_{\text{NBI}} = 65 \text{keV} \]

- Need small anomalous fast ion diffusivity \( D_{af}=0.3\text{m}^2/\text{s} \) for agreement
New: Most tangential NBI generates counter-propagating Toroidal Alfvén Eigenmodes (TAEs)

- Counter-propagating TAE predicted for hollow fast-ion profiles
  

- TRANSP: As current builds up beam fast-ion beta profile predicted to become hollow

- 1st evidence of off-axis NBI in NSTX-U
New: Tangential 2nd neutral beam suppresses Global Alfven Eigenmode (GAE) – consistent with simulation

HYM code simulation of #204707, n=10

- HYM code: growth of n=10 counter-GAE from 1st NBI
- HYM: suppression of n=10 counter-GAE by 2nd NBI
- Most unstable $n$-number, mode $\omega$ consistent with HYM

New 2nd NBI already powerful tool for fast-ion mode physics
Goals for future NSTX-U operation

• Increase field to 0.8-1T, current to 1.6-2MA, extend flat-top duration (H-mode) to 2-5s

• Assess global stability, energy confinement, pedestal height/structure, edge heat-flux width

• Characterize 2nd beam: heating, current drive, torque / rotation profiles, fast-ion instabilities

• Push toward full non-inductive current drive

• Test advanced divertor heat flux mitigation
NSTX-U: Flared divertor using “snowflake/X” + radiation

MAST-U will test range of divertors:
- Conventional, snowflake (not shown)
- Long-leg “Super-X” with variable flaring

New PF coils in NSTX-U central magnet

STs investigating lithium (solid and liquid) walls to significantly increase energy confinement.

NSTX (wider $\rightarrow$ higher pedestals)

$H = 0.8 \rightarrow 1.4 \times \text{ITER98P}(y,2)$

LTX (flatter $\rightarrow$ higher $T$ profiles)

$H = 2-4 \times \text{ITER98P}(y,2)$


NSTX-U long-term goals

- 5 year: Integrate high confinement + $\beta_T$ + full non-inductive
- 10 year: Assess compatibility with high-Z & liquid Li PFCs

Possible progression of in-vessel / plasma-facing components:

- Near-term: C walls + Li coatings on bottom $\frac{1}{2}$ of wall
- Lower $\nu^*$: Add divertor cryo-pump + full-wall Li coatings
- LTX-like: Heatable C $\rightarrow$ high-Z for liquid Li divertor
  - Flowing LM module? + heatable (?) high-Z walls
Summary

• STs broaden our understanding of toroidal physics and enhance predictive capability for ITER & beyond

• ST potentially attractive as FNSF and Pilot Plant

• NSTX-U designed to be highest performance ST in world program - this is why the project is challenging

• Team is working very hard to ensure NSTX-U can run reliably at high performance & get back into operation!
Thank you!

Any questions?