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NSTX

Progress of NSTX Program in Physics Basis for 10-MA Devices

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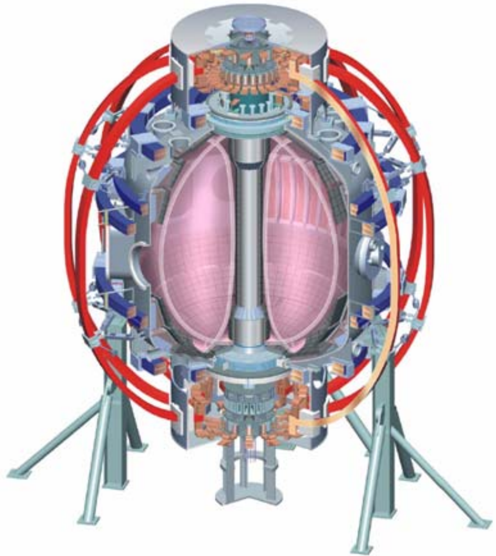
Martin Peng
And the NSTX Team

47th Annual Meeting of the Division of Plasma Physics

Denver, Colorado
October 24-28, 2005



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Progress of NSTX Program in Physics Basis for 10-MA Devices – Component Test Facility (CTF)

Y.-K. M. Peng (ORNL, UT-Battelle @ PPPL) & NSTX Team

Recent progress in Spherical Torus (ST) plasma science has indicated relatively robust and attractive physics conditions in a number of topical areas including shaping, stability limits, energy confinement, self-driven current, sustainment, and divertor heat flux. This progress has enabled an updated projection of the plasma conditions of a 10-MA ST such as the Component Test Facility (CTF), which is a necessary step in the development of practical fusion energy. The results indicated designs with R_0 = 1.2 m, A = 1.5, elongation $\kappa \sim 3$, $B_T \sim 2$ T, producing a fusion burn power of 140 MW, and a fusion neutron flux of 2 MW/m2, driven by 50 MW of combined neutral beam and RF heating and current drive power. The design uses a single-turn toroidal field coil center leg without a central solenoid, and will require physics data on solenoid-free plasma current initiation, ramp-up to, and sustainment at multiple MAs. An assessment of the ST physics basis to establish the design of such a 10-MA device and comparison with the present and planned achievements of the NSTX Program will be presented.

CTF – A Facility Required for Developing Engineering and Technology Basis for Fusion Energy

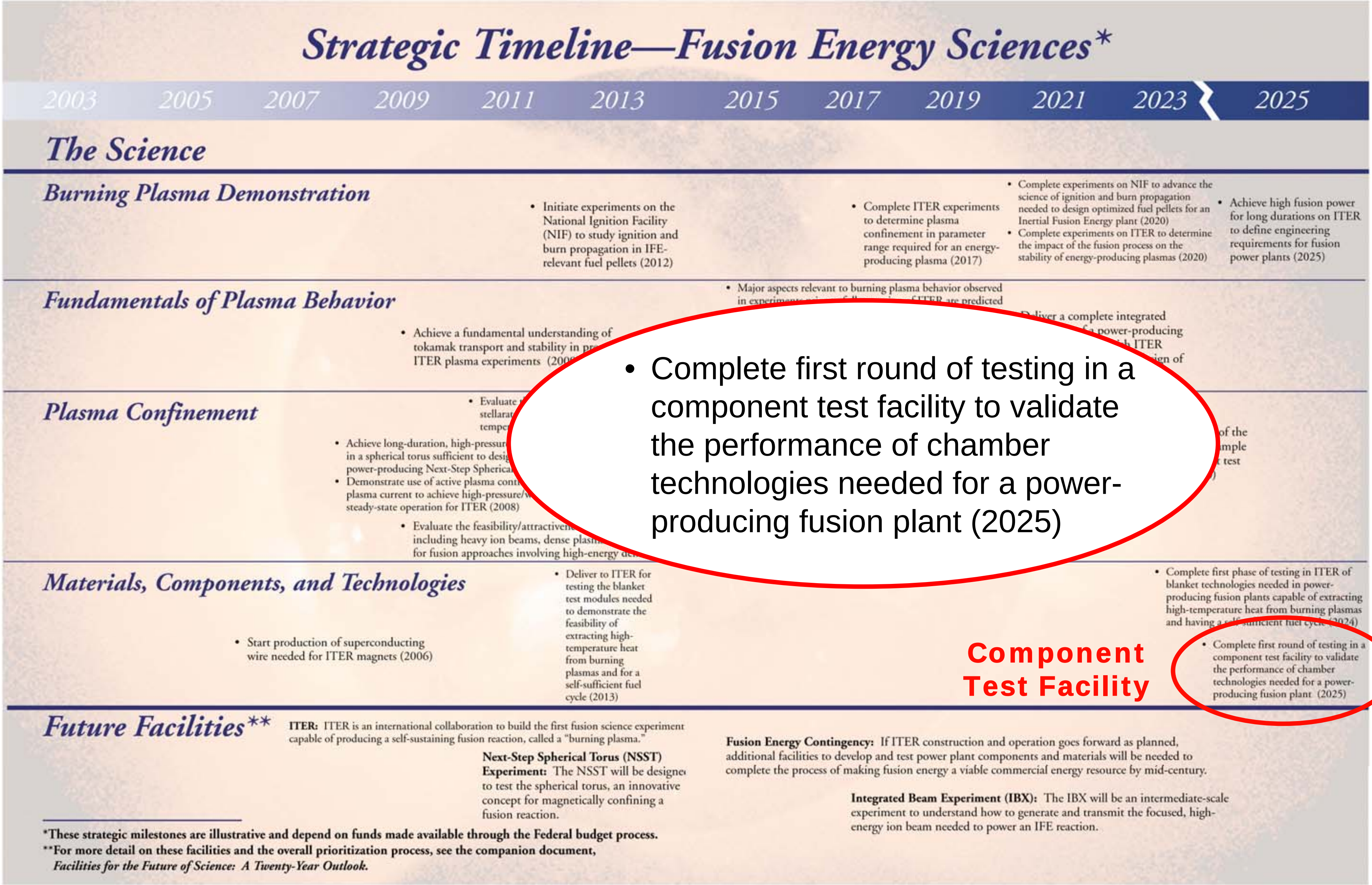
- **INL operated 45 small research fission facilities during 1951-69**
- **Necessary fusion Demo-relevant testing environment:**
  [M Abdou et al, Fusion Technology, **29** (1999) 1.]
  - **High 14 MeV neutron flux over large wall areas**
  - **High duty factor to achieve high neutron fluence per year**
  - **High accumulated fluence in facility lifetime**
- **Test tritium self-sufficiency – CTF goal: 80 – 100% recovery**
- **This presentation:**
  - **Programmatic importance**
  - **Desired engineering features**
  - **Plasma and device parameters based on latest physics understanding**
  - **Database needs in physics, engineering, & technology**

CTF Bridges Large Gaps between ITER and Demo in Tritium Self-Sufficiency, Duty Factor, Neutron Fluence, and Divertor Heat Flux

Fusion Power Conditions	**ITER**	**CTF**	**Demo**
Tritium self-sufficiency goal (%)		>80	>100
Sustained fusion burn duration (s)	~10^3	>10^{6-7}	~10^{7-8}
14-MeV neutron flux on wall (MW/m^2)	~0.8	1–2	~3
Duty factor (%)	~2	>30	75
Accumulated neutron fluence (MW-yr/m^2)	~0.3	>6	6–20
Divertor heat flux challenge, P/R (MW/m)	24	30–48	97
Total area of (test) blankets (m^2)	~12	~65	~670
Expected fusion power (MW)	~500	72–144	2500

- **CTF provides prototypical fusion power conditions at *reduced size and power***
- **Potential to "buttress" ITER & IFMIF in *accelerating development of fusion power*** [I Cook et al., UKAEA FUS 521 (Feb. 2005)]
- **DOE Office of Science 20-Year Strategic Plan for Fusion includes *CTF to succeed ITER construction***

DOE Office of Science 20-Year Strategic Plan for Fusion Includes CTF to Succeed ITER Construction



*Strategic Timeline—Fusion Energy Sciences**

The Science

Burning Plasma Demonstration

- Initiate experiments on the National Ignition Facility (NIF) to study ignition and burn propagation in IFE-relevant fuel pellets (2012)
- Complete ITER experiments to determine plasma confinement in parameter range required for an energy-producing plasma (2017)
- Complete experiments on NIF to advance the science of ignition and burn propagation needed to design optimized fuel pellets for an Inertial Fusion Energy plant (2020)
- Complete experiments on ITER to determine the impact of the fusion process on the stability of energy-producing plasmas (2020)
- Achieve high fusion power for long durations on ITER to define engineering requirements for fusion power plants (2025)

Fundamentals of Plasma Behavior

- Achieve a fundamental understanding of tokamak transport and stability in pr ITER plasma experiments (200
- Major aspects relevant to burning plasma behavior observed in experiments pri ITER are predicted
- Deliver a complete integrated a power-producing ITER ign of

Plasma Confinement

- Evaluate stellarat temper
- Achieve long-duration, high-pressur in a spherical torus sufficient to desi power-producing Next-Step Spherical
- Demonstrate use of active plasma cont plasma current to achieve high-pressure/ steady-state operation for ITER (2008)
- Evaluate the feasibility/attractiven including heavy ion beams, dense plas for fusion approaches involving high-energy de

of the mple test

Materials, Components, and Technologies

- Start production of superconducting wire needed for ITER magnets (2006)
- Deliver to ITER for testing the blanket test modules needed to demonstrate the feasibility of extracting high-temperature heat from burning plasmas and for a self-sufficient fuel cycle (2013)
- Complete first phase of testing in ITER of blanket technologies needed in power-producing fusion plants capable of extracting high-temperature heat from burning plasmas and having a self-sufficient fuel cycle (2024)
- Complete first round of testing in a component test facility to validate the performance of chamber technologies needed for a power-producing fusion plant (2025)

*Future Facilities***

ITER: ITER is an international collaboration to build the first fusion science experiment capable of producing a self-sustaining fusion reaction, called a "burning plasma."

Next-Step Spherical Torus (NSST) Experiment: The NSST will be designe to test the spherical torus, an innovative concept for magnetically confining a fusion reaction.

Fusion Energy Contingency: If ITER construction and operation goes forward as planned, additional facilities to develop and test power plant components and materials will be needed to complete the process of making fusion energy a viable commercial energy resource by mid-century.

Integrated Beam Experiment (IBX): The IBX will be an intermediate-scale experiment to understand how to generate and transmit the focused, high-energy ion beam needed to power an IFE reaction.

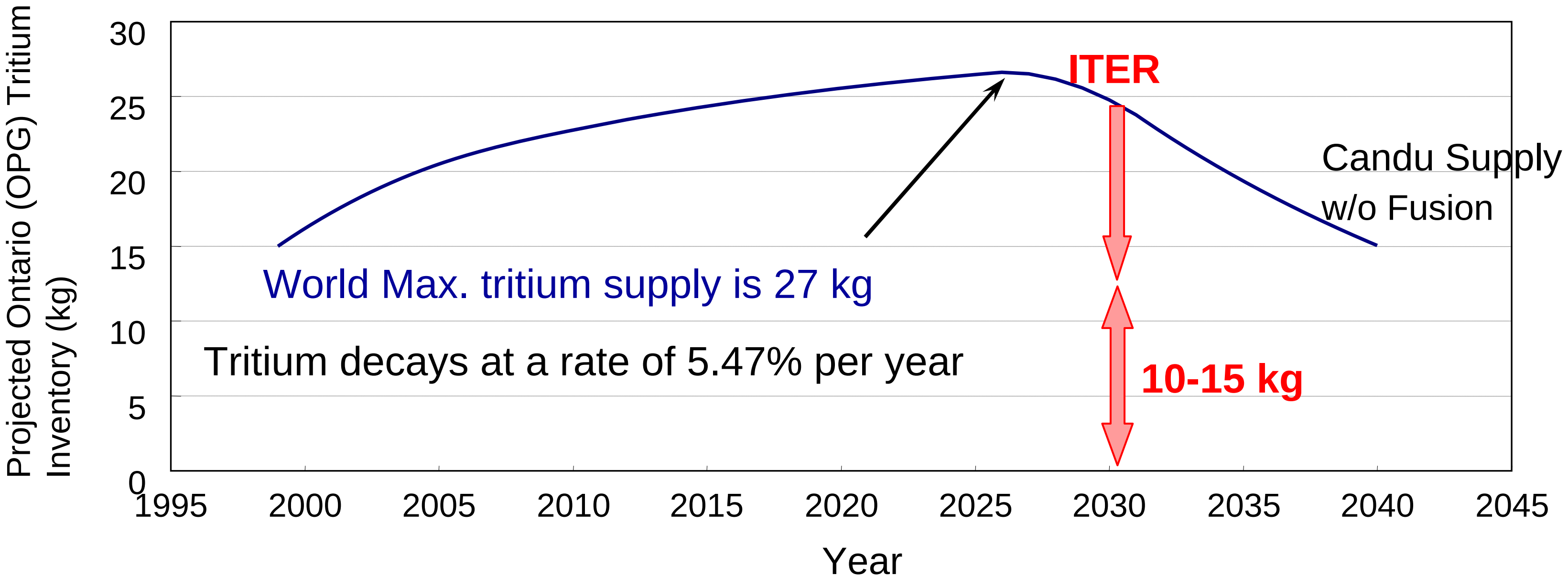
***These strategic milestones are illustrative and depend on funds made available through the Federal budget process.**

****For more detail on these facilities and the overall prioritization process, see the companion document, *Facilities for the Future of Science: A Twenty-Year Outlook.***

- Complete first round of testing in a component test facility to validate the performance of chamber technologies needed for a power-producing fusion plant (2025)

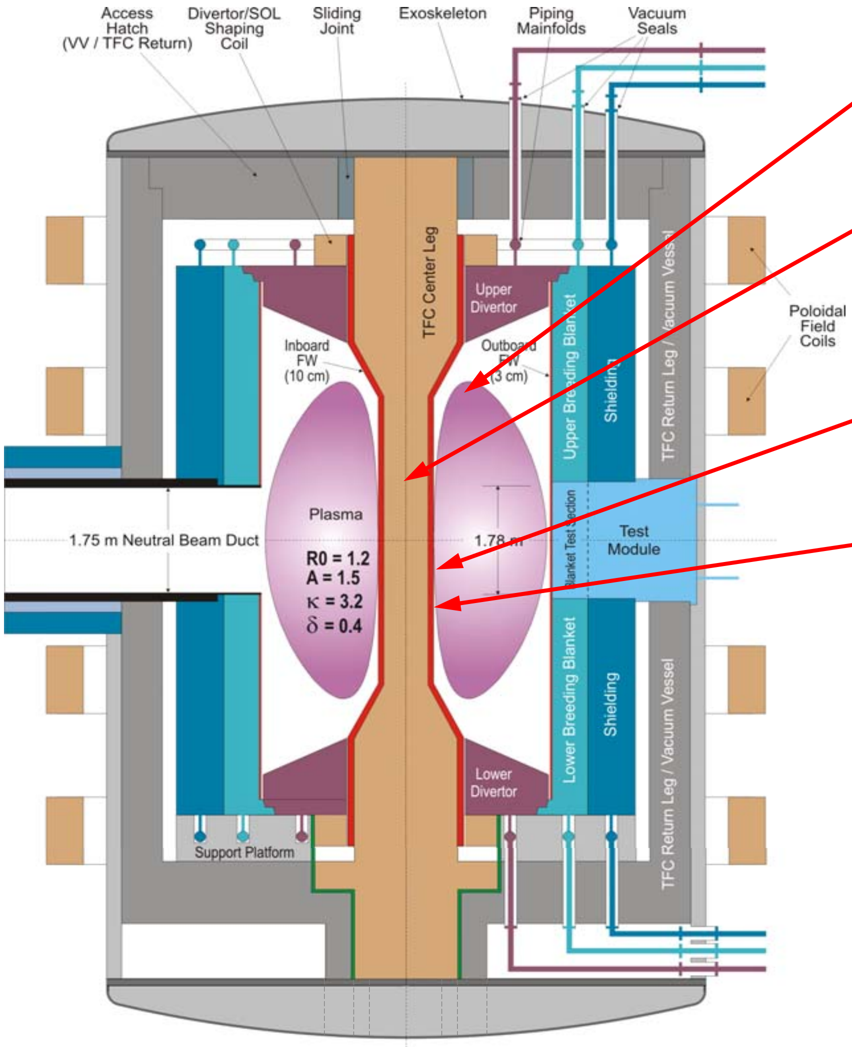
Component Test Facility

Projected World Tritium Supply Necessitates Testing in CTF Before Implementation in Demo



- ITER uses ~11 kg T to provide 0.3 MW-yr/m^2; 10-15 kg remains
- Demo burns tritium @ 2.7 kg/week to produce 2500 MW fusion power
- **To accumulate 6 MW-yr/m^2 (component testing mission), and assuming 80% breeding fraction,**
  - **Demo requires 56 kg**
  - **CTF requires 4.8 kg**

Features of Optimized ST Fulfill the CTF Mission Effectively

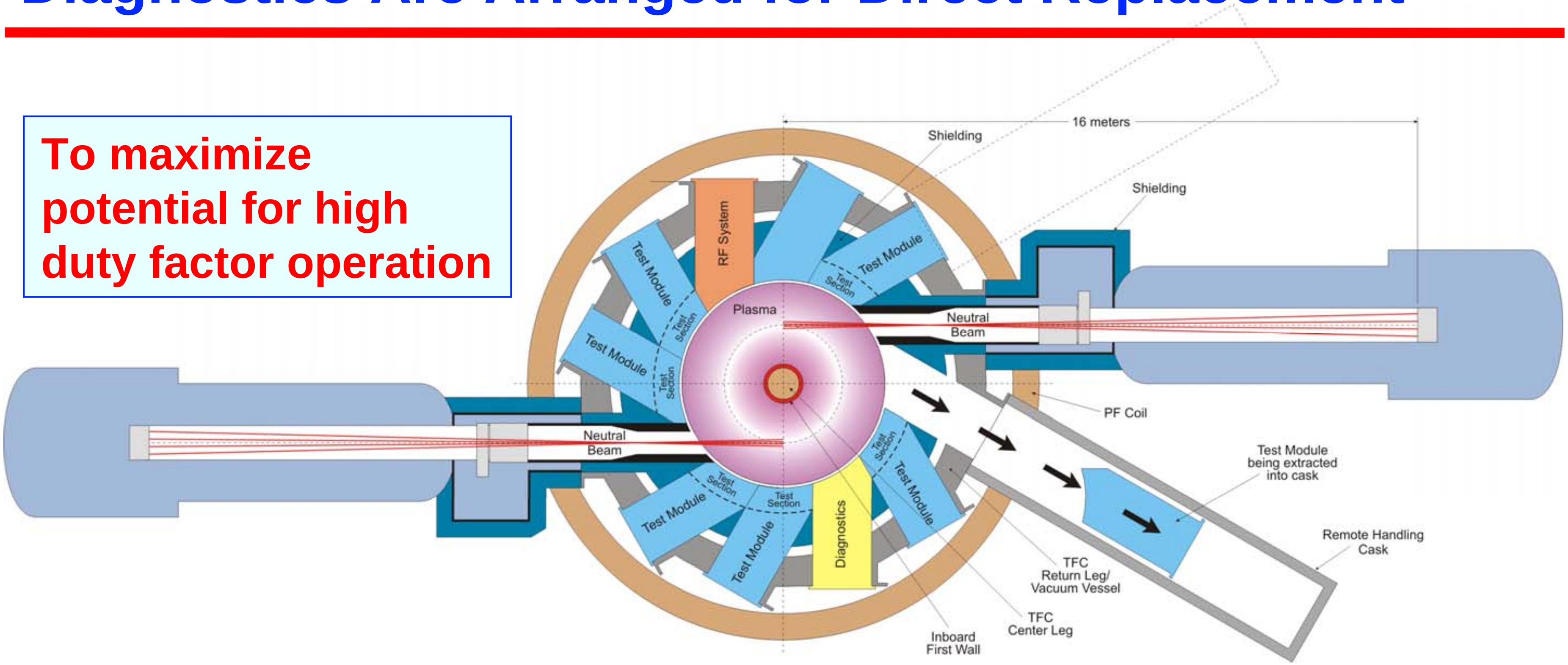


R_0 = 1.2 m, a = 0.8 m

- **Natural elongation at low ℓ_i** → simple shaping coils
- **I_{TF} ~ I_p; moderate B_T** → slender, demountable, single-turn TF center leg
- **No central solenoid** → no inboard nuclear shielding
- **No inboard blanket** → compact ST device with small radius & aspect ratio
- **~5% fusion neutrons lost to center leg** → high tritium breeding ratio
- **Culham CTF: more compact, less fusion power, same W_L**
  [H Wilson et al., IAEA FEC 2004, FT/3-1a.]

Mid-Plane Test Modules, Neutral Beam Injection, RF, Diagnostics Are Arranged for Direct Replacement

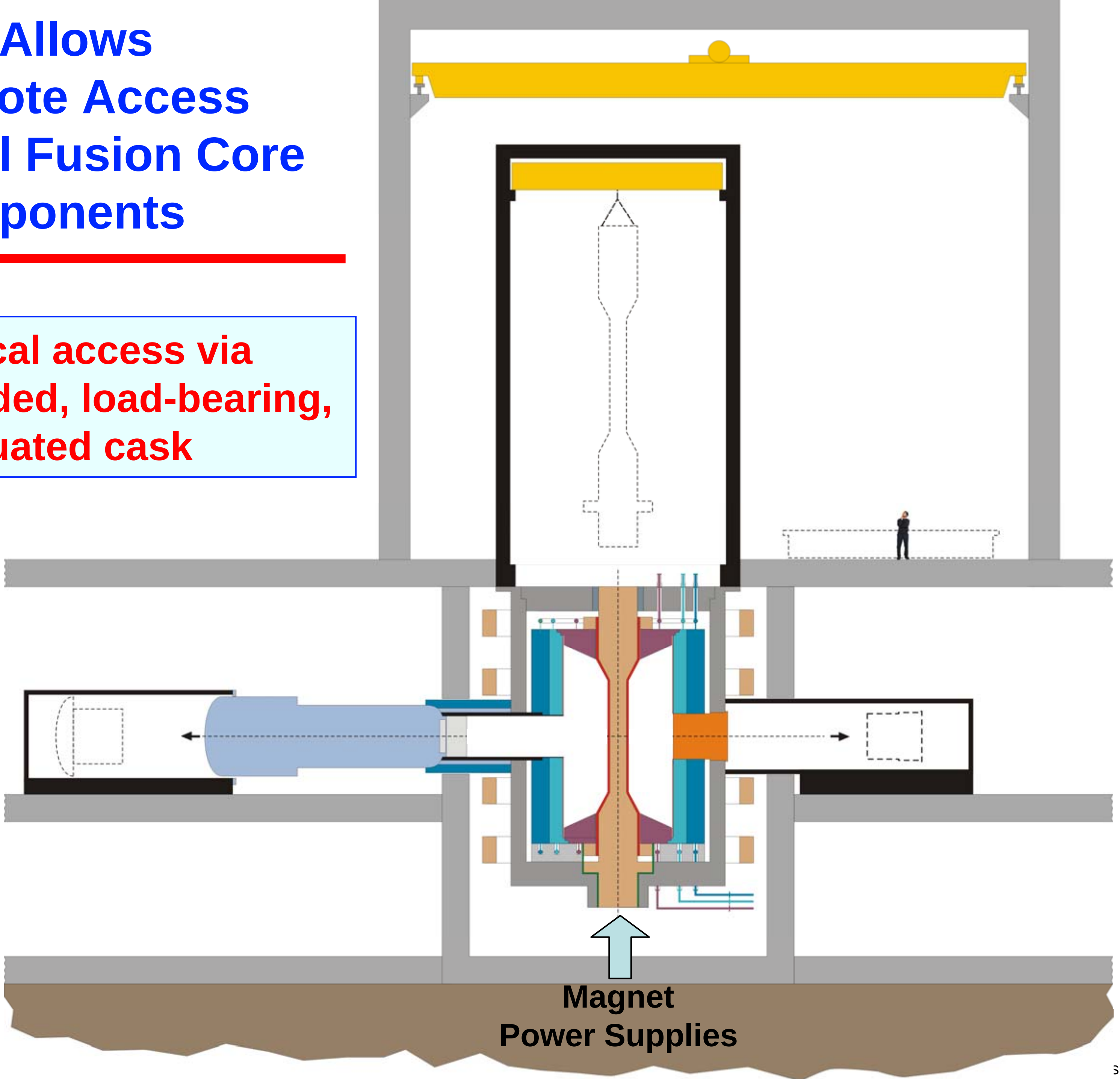
To maximize potential for high duty factor operation



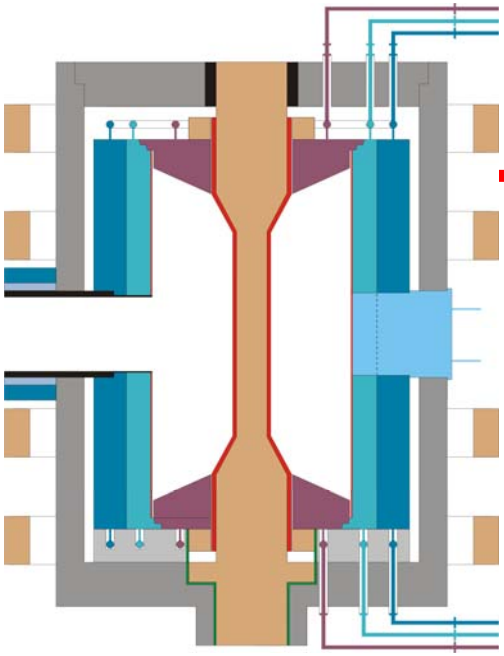
- 8 mid-plane blanket test modules provides ~ 15 m^2 at maximum flux
  - Additional cylindrical blanket test area > 50 m^2 at reduced flux
- 3 m^2 mid-plane access for neutral beam injection of 30 MW
- 2 m^2 mid-plane accesses for RF (10 MW) and diagnostics
- All modules accessible through remote handling casks (~ITER)

CTF Allows Remote Access to All Fusion Core Components

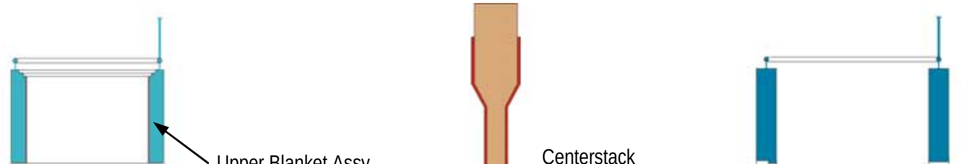
Vertical access via shielded, load-bearing, evacuated cask

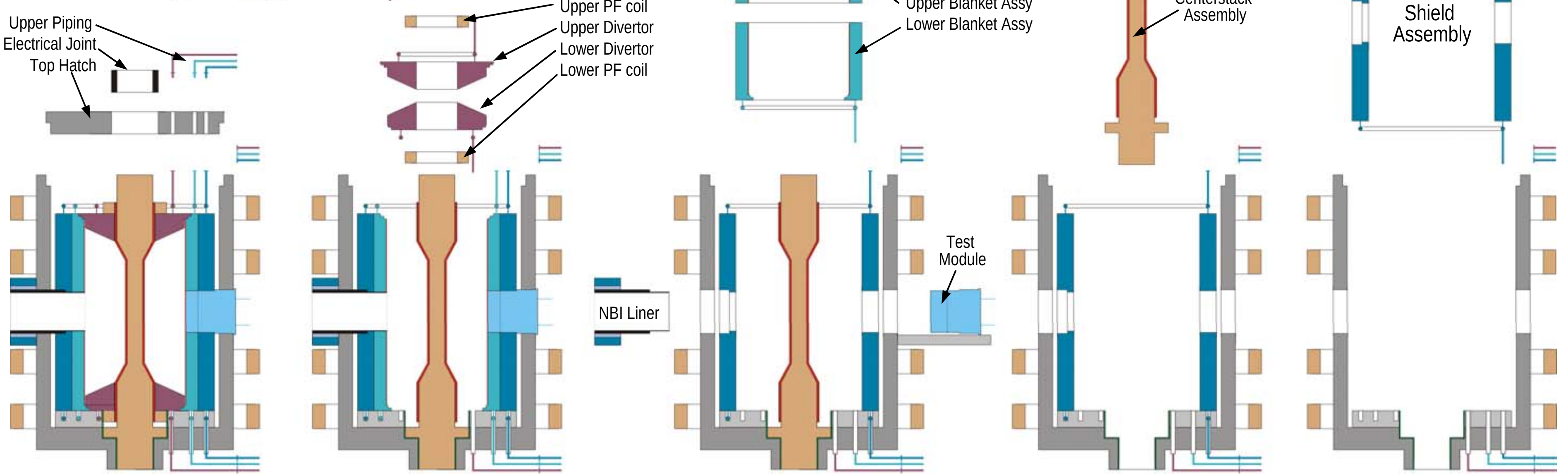


Machine Assembly/Disassembly Sequence Are Made Manageable



- **Hands-on connect and disconnect service lines outside of shielding and vacuum boundaries**
- **Divertor, cylindrical blanket, TF center leg, and shield assembly removed/installed vertically**





- Disconnect upper piping
- Remove sliding electrical joint
- Remove top hatch

- Remove upper PF coil
- Remove upper divertor
- Remove lower divertor
- Remove lower PF coil

- Extract NBI liner
- Extract test modules
- Remove upper blanket assembly
- Remove lower blanket assembly

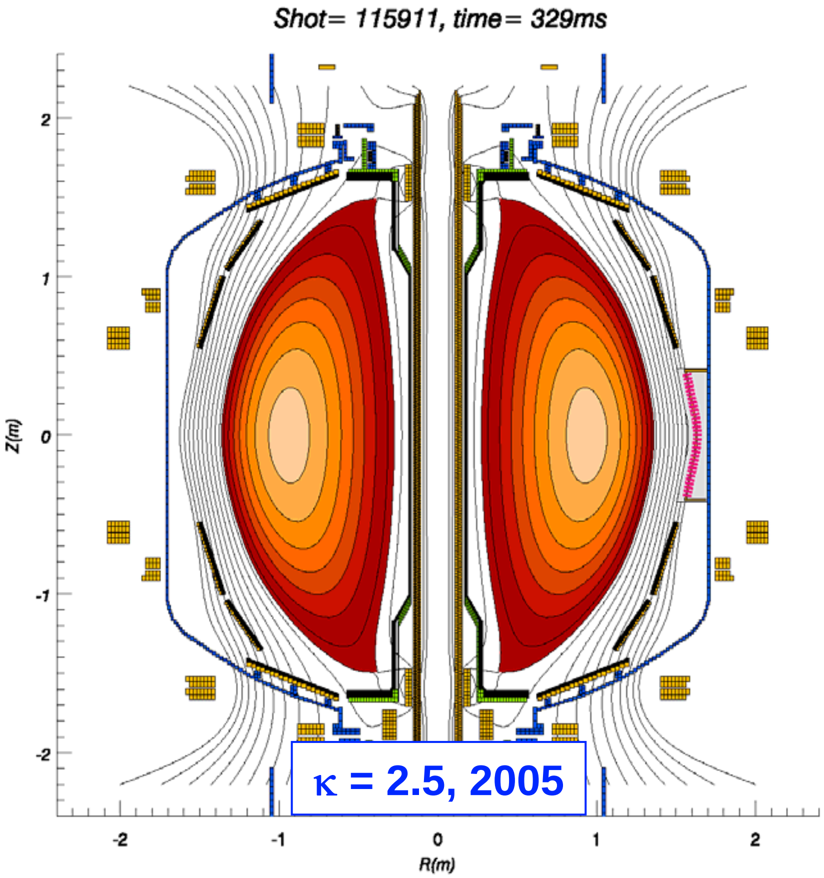
- Remove centerstack assembly

- Remove shield assembly

CTF Should Utilize Attractive ST Physics Properties

Proof of Principle:

- **Show CTF scientific feasibility**
- **Identify reliable operating regime**



Shot= 115911, time= 329ms

$\kappa = 2.5$, 2005

Utilizes applied field efficiently

- Strong plasma shaping & self fields (vertical elongation ~ 3, $B_p/B_t \sim 1$)
- Very high β_T (~ 40%) & bootstrap current

Contains plasma energy efficiently

- Small plasma size relative to gyro-radius ($a/\rho_i \sim 30$–50)
- Large plasma flow ($M_A = V_{rotation}/V_A \leq 0.4$)
- Large flow shearing rate ($\gamma_{ExB} \leq 10^6$/s)

Disperses plasma fluxes effectively

- Large mirror ratio in edge B field ($f_T \sim 1$)
- Strong SOL expansion

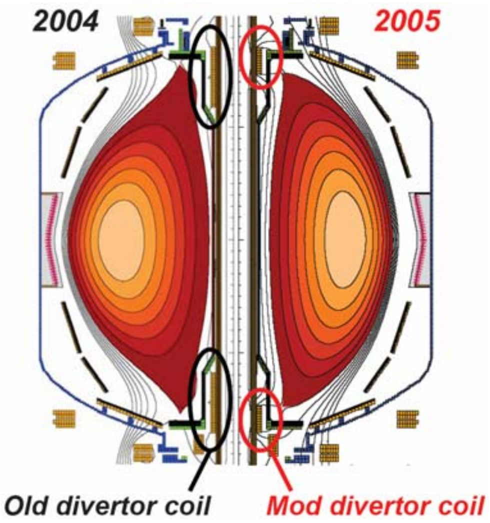
Allows easier solenoid-free operation

- Small magnetic flux content ($\sim \ell_i R_0 I_p$)

Heating and Current Drive opportunities

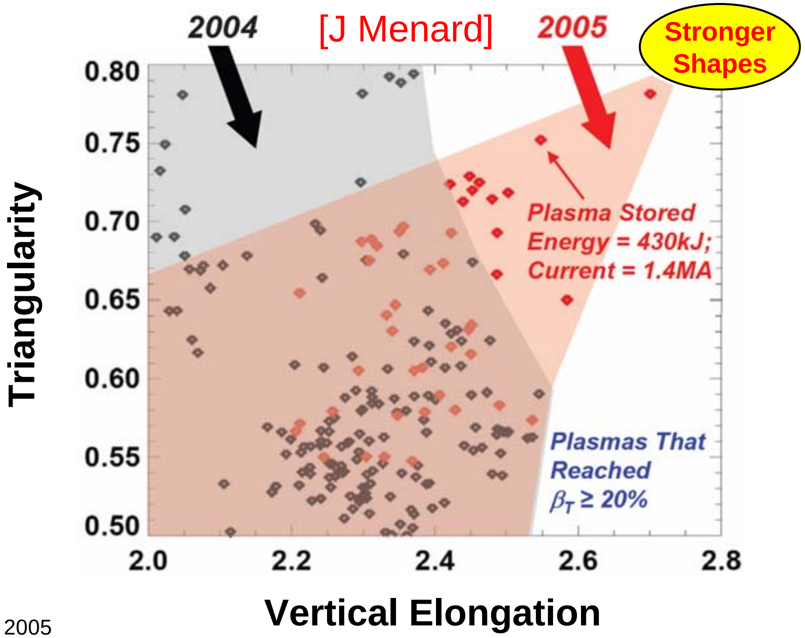
- Supra-Alfvénic fast ions ($V_{fast}/V_A \sim 1$–5)
- High dielectric constant ($\varepsilon = \omega_{pe}^2/\omega_{ce}^2 \sim 50$)

NSTX Dramatically Expanded the Spherical Torus Operating Space to Clarify Future ST Options

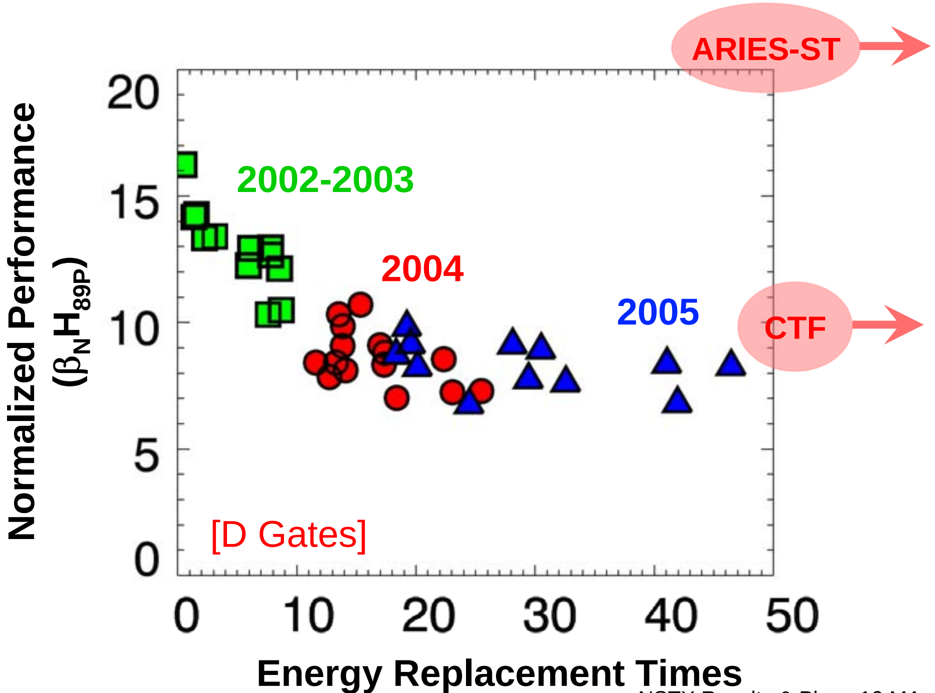


2005 Campaign

- Improved divertor coils
- Extended plasma to stronger shapes
- High triangularity at high elongation leads to quiescent core and edge conditions

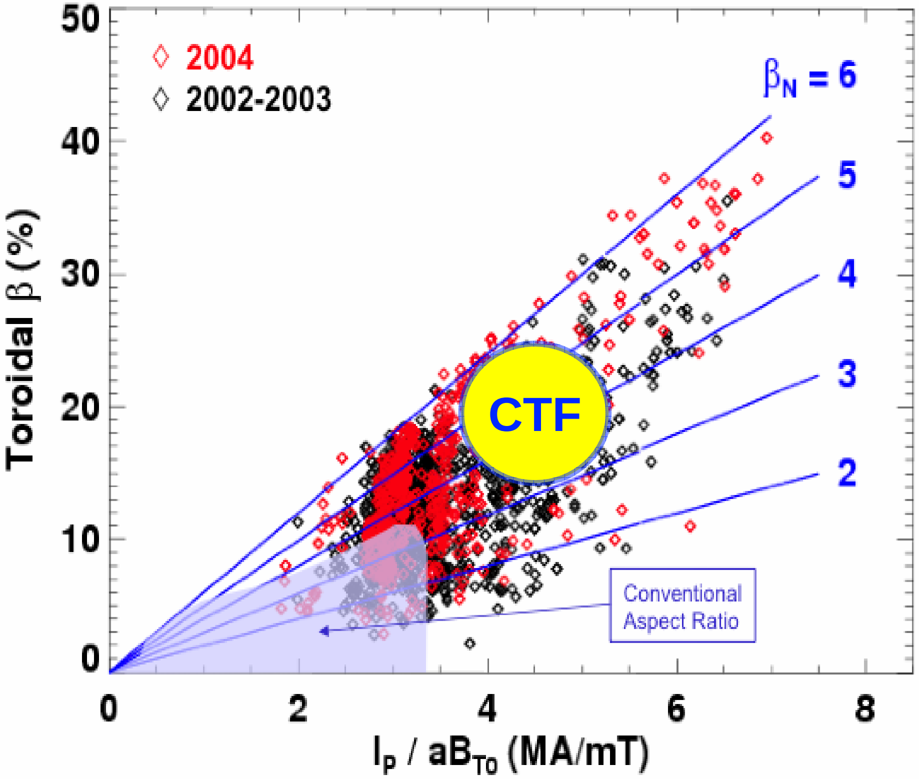


[J Menard]

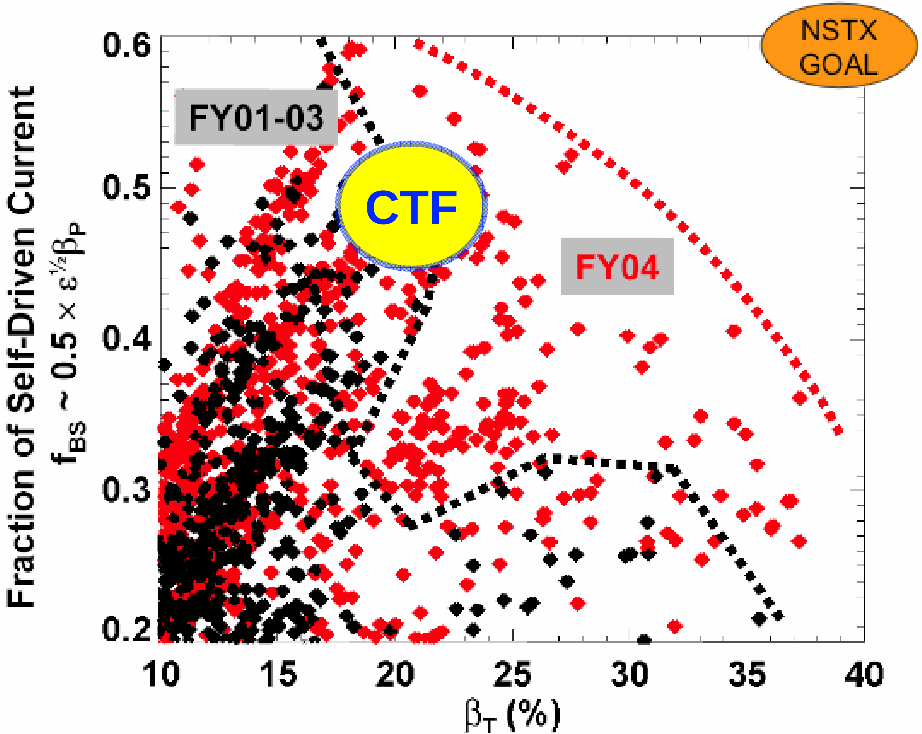


[D Gates]

NSTX Data Map a Large Domain in β_T and f_{BS} in Which to Design Reliable CTF Operation



Higher κ (= 3.2) designed for CTF would provide increased margin in (I_p/aB_{T0}), f_{BS}, and q_{cyl}



Initial CTF Parameters Are Estimated Based on the Design Concept & Present Physics Understanding

Systems Code $\Rightarrow R_0 = 1.2$ m, $a = 0.8$ m, $\kappa = 3.2$, $B_T = 2.5$ T

14MeV neut. flux, MW/m^2	**1.0-2.0**	**4.0**
I_p, MA	9.1-12.8	16.1
Combined H_{98pby} factor	1.6-1.5	1.38
β_T, %	14-24	**39**
$\beta_N H_{89P}$	9.0-11.3	16
Safety factor, q_{cyl}	4.2-3.0	**2.4**
n/n_{GW}	0.16-0.17	0.21
I_{BS}/I_p	0.52-0.43	0.44
P_{fusion}, MW	72-144	288
P_{NBI+RF}, MW	36-47	65
Neutral beam energy, kV	112-160	250
f_{rad}, % (for P_{div} = 15 MW/m^2)	**65-79**	**90**
Net $T_{consumption}$/yr goal, gm	0-14	180

- **Baseline (1-2 W/m^2) parameters within ST plasma operation limits**
- **Higher neutron fluxes reach progressively more limits**
  - In β, q_{cyl}, and f_{rad}
  - Requires densities ~ 20% limit
- **Technology & physics of CTF advances in synchrony**
  - 1-2 MW/m^2 – medium ST physics to test technologies beyond ITER
  - 4 MW/m^2 – more advanced ST physics to test DEMO level technologies

CTF Stable β Values Rely on Continued Progress in ST Macro-Stability Research

Sustained Parameters	**CTF ($\tau \gg \tau_{skin}$)**	**Long Pulse Data ($\tau > \tau_{skin}$)**
I_p/aB_T (MA/m-T)	4.6 – 6.4	≤4.4
Safety factor, q_{cyl}	4.2 – 3.0	≥2.2
β_N (%-m-T/MA)	3.1 – 3.9	≤5
β_T (%)	14 – 24	≤23
Start-up to $\mu_0 \ell_i R I_p$ (Wb)	≥3.8	~0.13 (goal)

Required Investigations

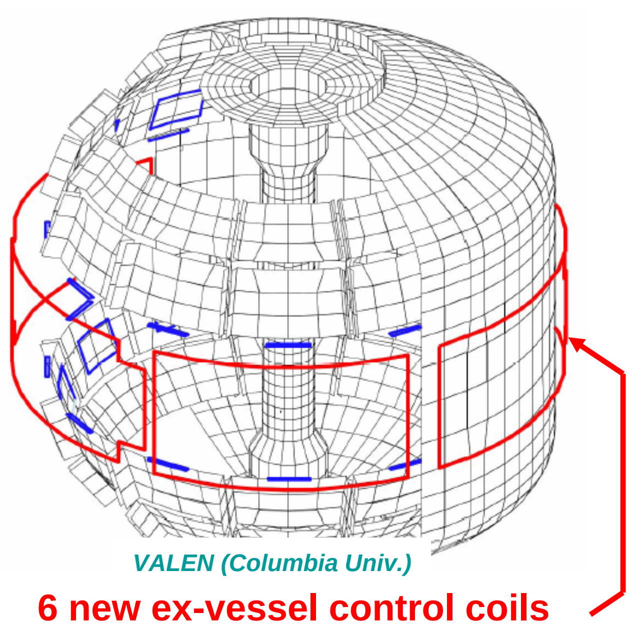
- Macro-stability near CTF conditions: $\kappa \leq 2.7$ and $\tau \gg \tau_{skin}$
- Error field & resistive wall mode, with strong plasma rotation, toward high reliability & higher β_N
- Solenoid-free start-up to ~ 0.5 MA plasma target for NBI and EBW

Issue: solenoid-free startup [R Raman]

Error Field Reduction Are Shown to Improve Plasma Sustainment at High β

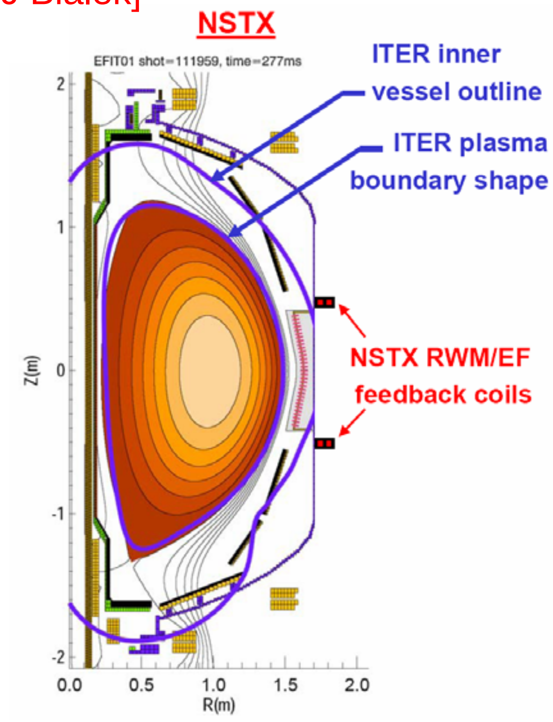
- **Passive plate and feedback coils influence modes in manners similar to ITER blankets and nearby control coils positions**
- **Is there a error field threshold, below which high β can be sustained indefinitely?**

[J Menard, S Sabbagh, J Bialek]



VALEN (Columbia Univ.)

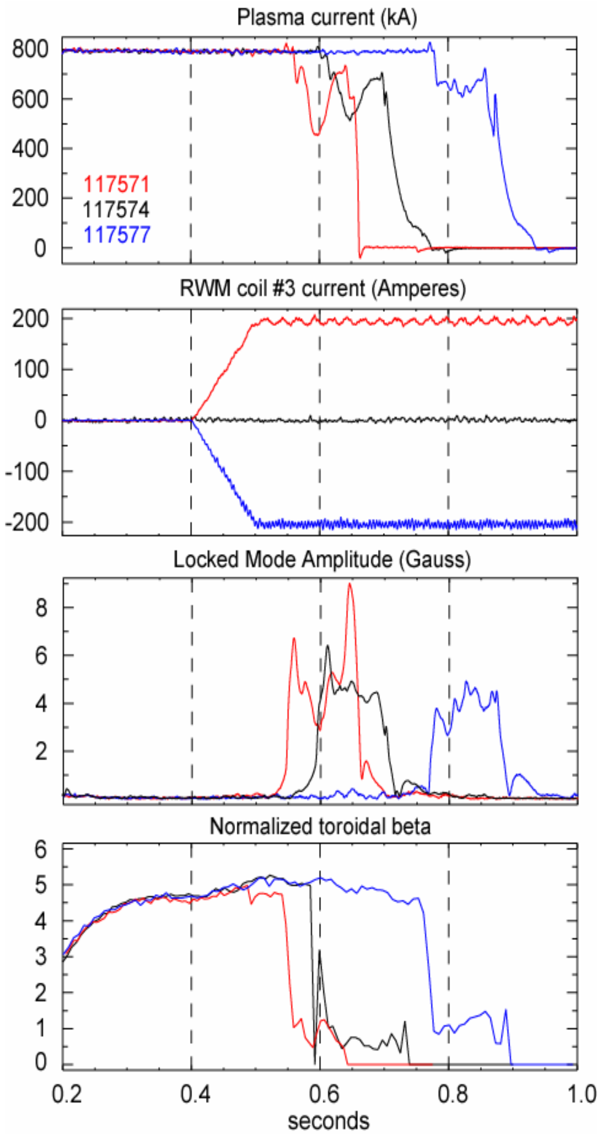
6 new ex-vessel control coils
+ 48 in-vessel sensors



ITER inner vessel outline

ITER plasma boundary shape

NSTX RWM/EF feedback coils



Plasma current (kA)

RWM coil #3 current (Amperes)

Locked Mode Amplitude (Gauss)

Normalized toroidal beta

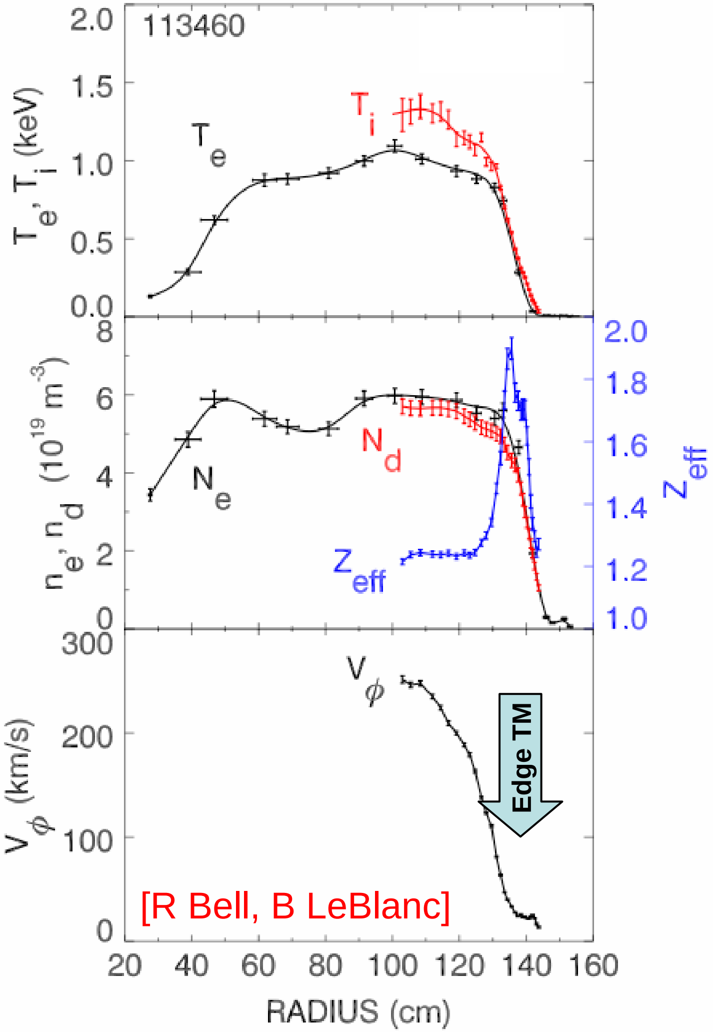
CTF Confinement Assumptions Are Suggested by Long-Pulse Plasmas in NSTX & MAST

Sustained Parameters	**CTF ($\tau >> \tau_{skin}$)**	**Long Pulse Data ($\tau > \tau_{skin}$)**
$\langle T_i \rangle / \langle T_e \rangle$	~2	≤1.5 via co-NBI
n_e/n_{GW}	~0.2	0.2 – 0.8, rising in pulse
a/ρ_i ($=1/\rho_i^*$)	~50	~30
H_{98pby2}	≤1.5	≤1.3 for $>\tau_{skin}$

Required Investigations

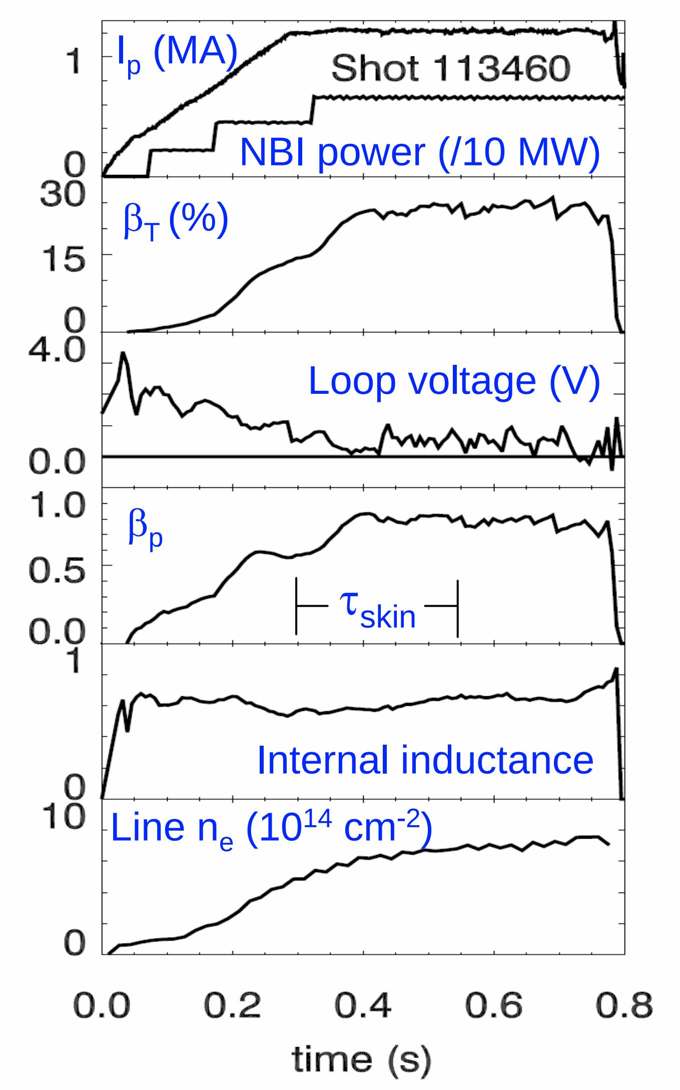
- Strongly rotating plasma with ion "internal transport barrier" via co-NBI
- Density control at low n_{GW}, such as via lithium
- **Electron transport vs. β effects: τ_{Ee}** [S Kaye]
- **Ion transport vs. neoclassical: τ_{Ei}**

Long-pulse H-mode



[R Bell, B LeBlanc]

NSTX Has Made Significant Progress Towards Goal of High-β_T, Non-Inductive Operation



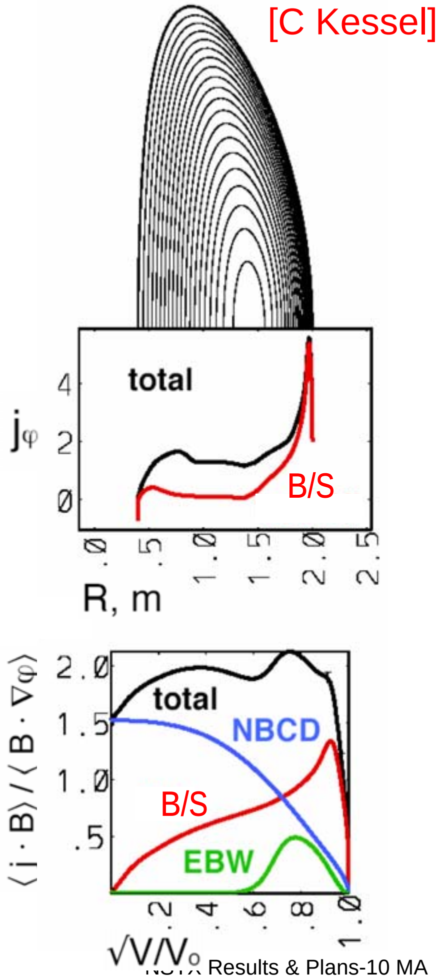
- $\tau_{Ip\ flattop} \sim 2\ \tau_{skin}$
- $\tau_{W\ flattop} \sim 9\ \tau_E$
- $\beta_T > 23\%$, $\beta_N > 5.3$
- $H_{89P} \sim 2$
- Internal inductance ~ 0.6
- $n_e \sim 0.6 \times 10^{13}\ /cm^3$
- 1.5-s pulses in 2005

[J Menard, D Gates]

ST Research Addresses CTF Heating & Current Drive Physics in the Same Regime

Sustained Parameters	CTF ($\tau >> \tau_{skin}$)	Long Pulse Data ($\tau > \tau_{skin}$)
$V_{Fast}/V_{Alfvén}$	3 – 6	1 – 4
I_{CD}/I_p	~0.5	≤0.3
$I_{BS+diam+PS}/I_p$	~0.5	≤0.6
P/R (MW/m)	30 – 48	≤9
SOL area expansion	10 – 20	~5
Radiation fraction (%)	65 – 79	≤30

CTF Plasma Shape & Stable Current Profile



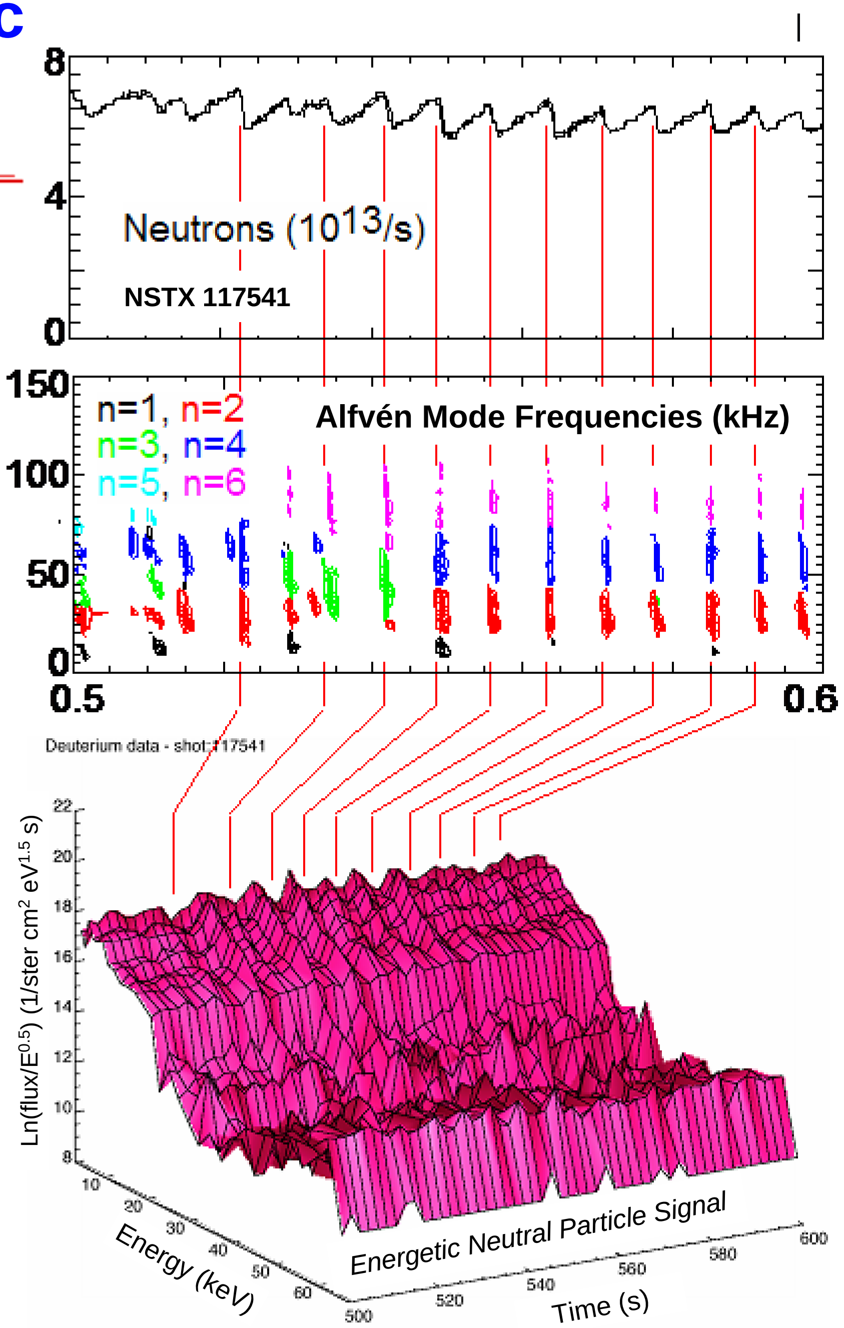
[C Kessel]

Required Investigations

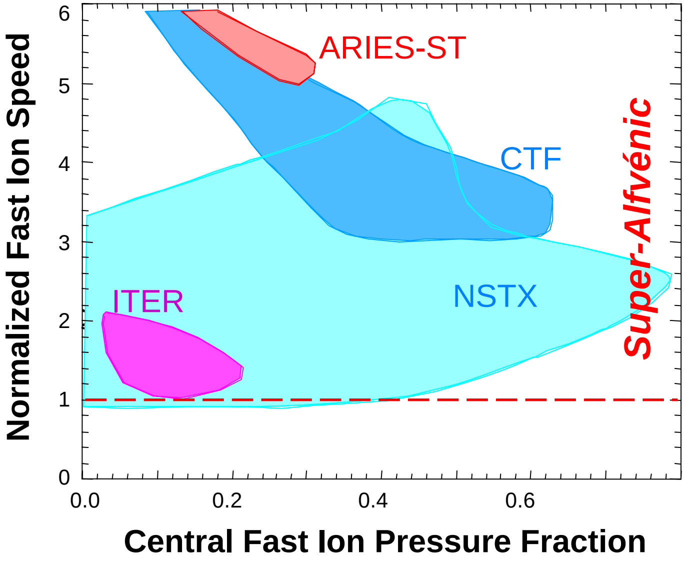
- Supra-Alfvénic ion driven modes, transport, and current
- Combined NBI-EBW, stable long-pulse operation with good confinement and substantial B/S and driven currents
- **Innovative divertor physics solutions**
  - **lithium divertor (NSTX); divertor biasing (MAST)**

NSTX Is Studying Super-Alfvénic Ion Heating for ITER and CTF

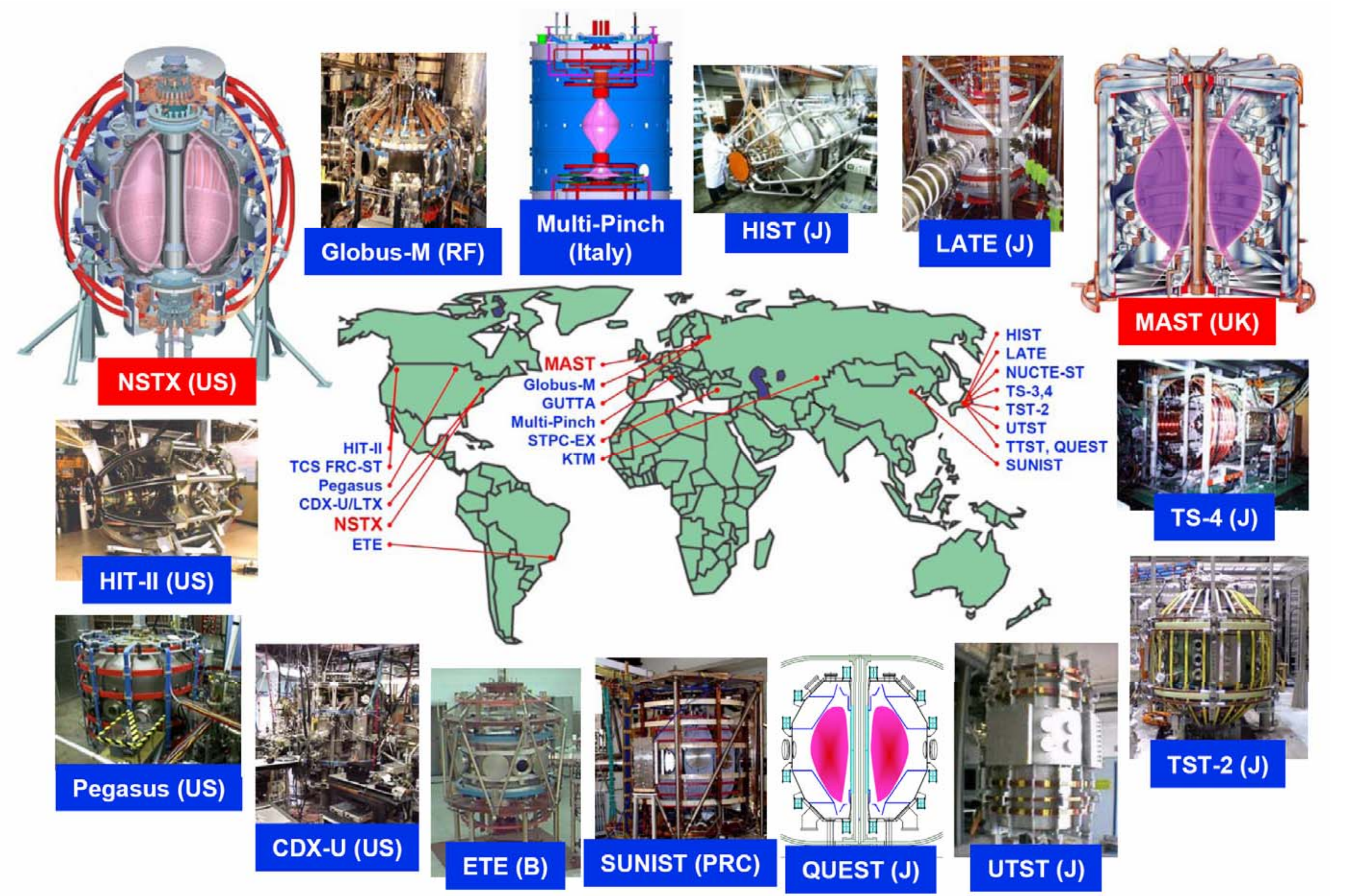
- **NSTX has, and ITER will have, Super-Alfvénic ions**
- **NSTX measured: instabilities driven by such fast ions & coincidental fast ion losses, but persistent losses not yet understood**
- **Interactions encouraged by small ρ_{fast}* (ITER), copious fast ions (both), and Doppler shifted resonance with Alfvén instabilities (both)**
- **Will fusion α's in ITER & CTF suffer similar losses?**
- **Only NSTX also has current profile measurement (MSE), important in determining mode number, n**



[E Fredrickson, S Medley]



22 “Concept Exploration” and “Proof of Principle” STs Are Working Together toward Common Goals



ST CTF Has Attractive Physics and Engineering Features to Fulfill a Critical Fusion Development Need

- CTF required for developing engineering and technology basis to accelerate fusion energy development
  - Bridges large development gaps between ITER and Demo
  - Limited tritium supply necessitates CTF testing before Demo
- ST features fulfill the CTF mission effectively
  - Fast replacement of test modules
  - Remote access to all fusion core components
- ST promises good physics basis for CTF
  - NSTX & MAST results encouraging
  - Reliable physics regime identified, away from known limits
- Additional ST physics data needs are identified

URL: http://nstx.pppl.gov/DragNDrop/APS-DPP_05/Posters/