Spherical Torus (Spherical Tokamak) on the Path to Fusion Energy

ST can support fast implementation of fusion Demo in unique, important ways

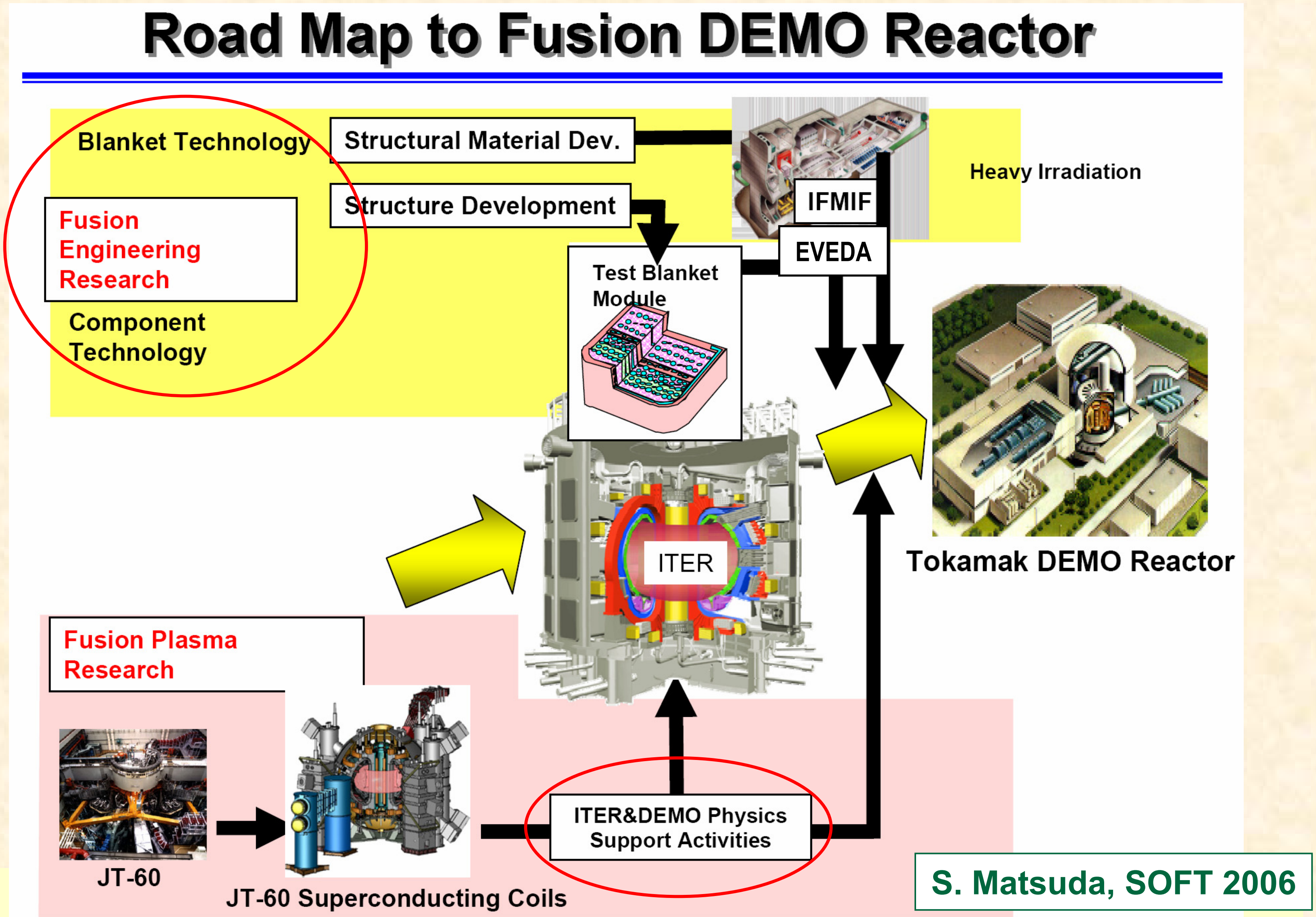
1) Opportunities to support the strategy of Demo after ITER
2) Important ways in which ST can do so
3) Component Test Facility for steady state integrated testing
4) Broad progress and the remaining CTF physics R&D needs

Martin Peng, NSTX Program Director

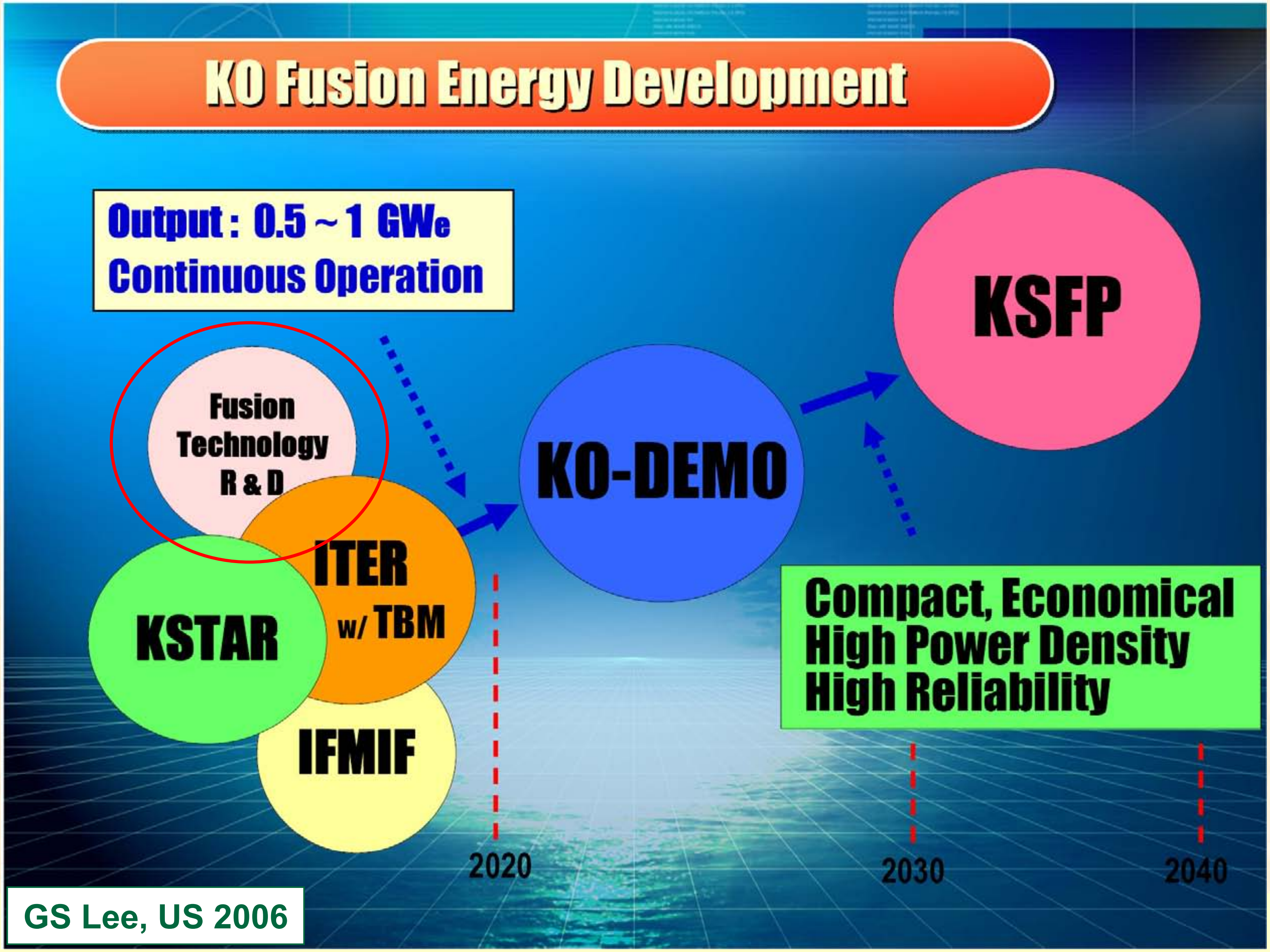
Fusion Power Associates Annual Meeting and Symposium
Fusion: Pathway to the Future
September 27-28, 2006, Washington D.C.

Oak Ridge National Laboratory
U. S. Department of Energy

EU-Japan plan of Broader Approach toward Demo introduces opportunities in physics and component



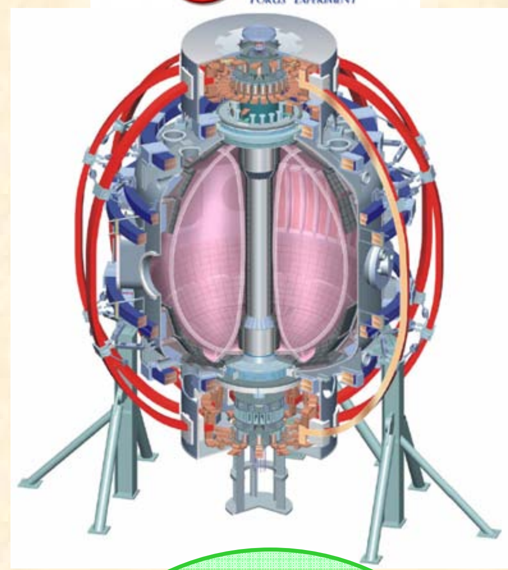
Korean fusion energy development plan introduces opportunities in accelerating fusion technology R&D

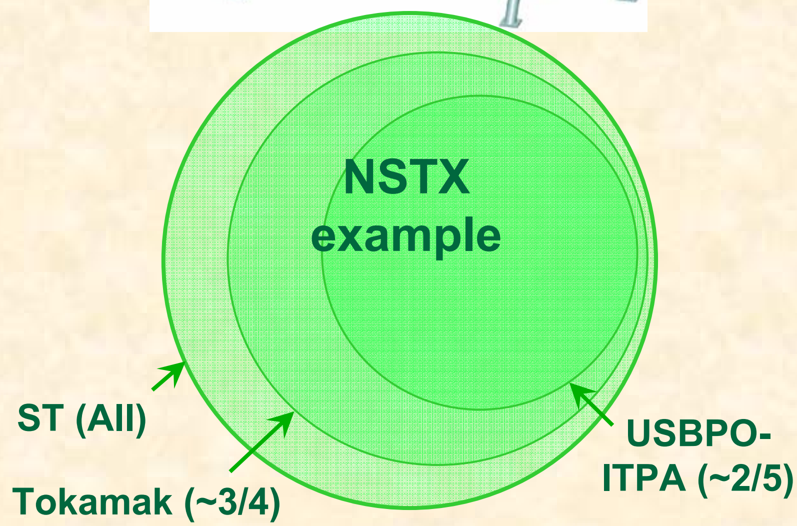


GS Lee, US 2006

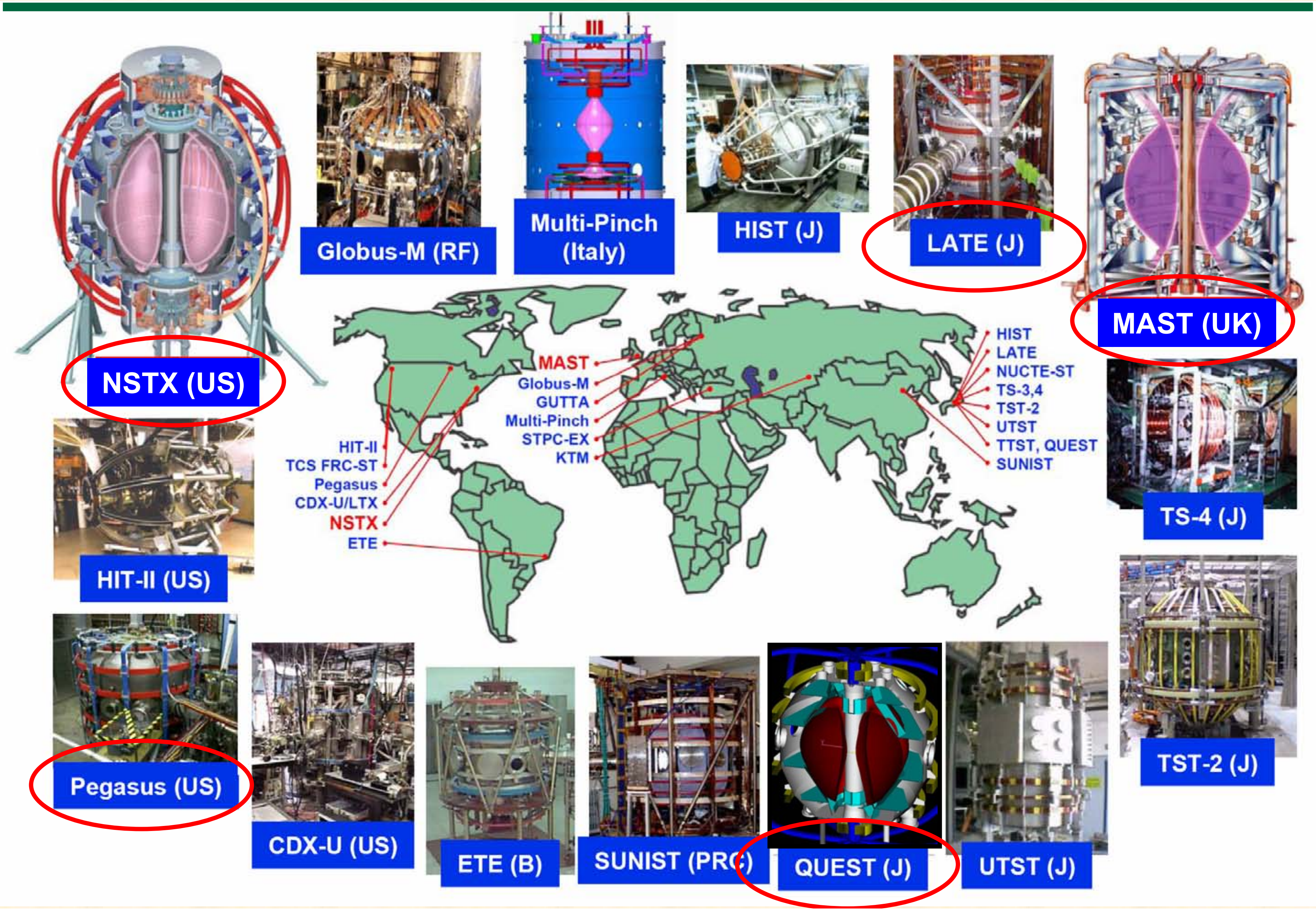
We propose that ST research addresses issues in support of this strategy

- **Support and benefit from USBPO-ITPA activities** in preparation for burning plasma research in ITER using physics breadth provided by ST.
- **Complement and extend tokamak physics experiments,** by maximizing synergy in investigating key scientific issues of tokamak fusion plasmas
- **Enable attractive integrated Component Test Facility (CTF)** to support Demo, by establishing ST database and leveraging the advancing tokamak database for ITER burning plasma operation and control.





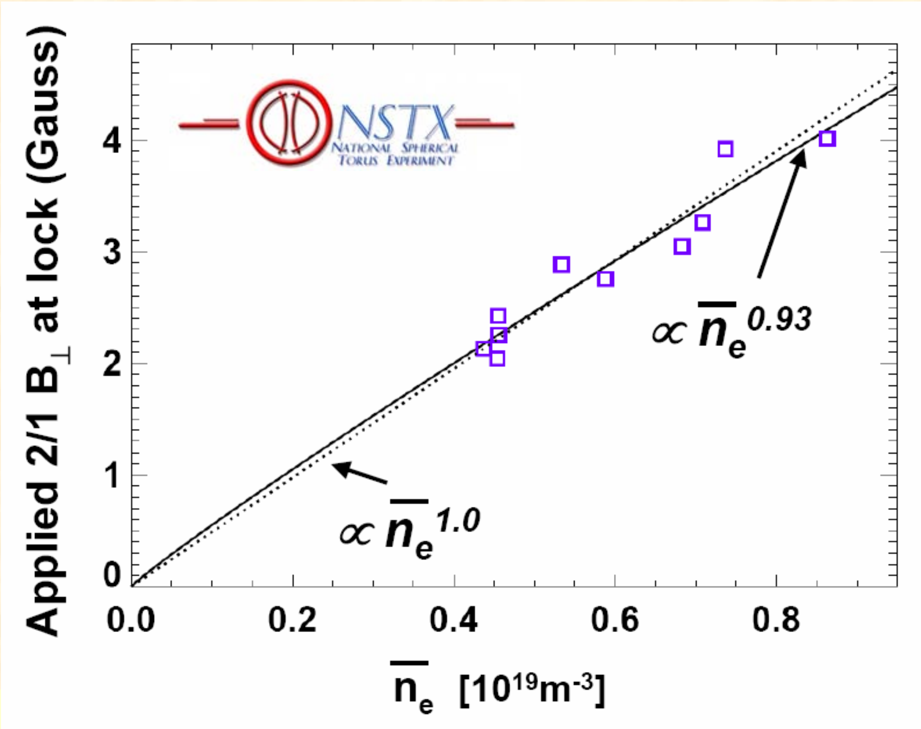
World Spherical Tokamak research has expanded to 22 experiments addressing key physics issues



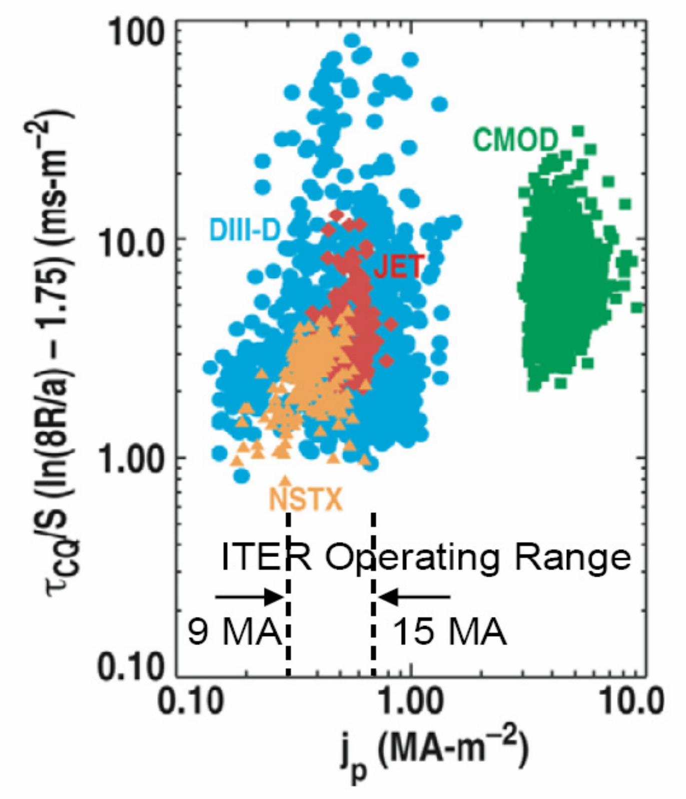
ST supports and benefits from USBPO-ITPA in preparing for burning plasma research on ITER

- NSTX completed in 2006 half of the 22 ITPA 2006-7 joint experiments
- ST "exceptions" prove ITER rules and test theory

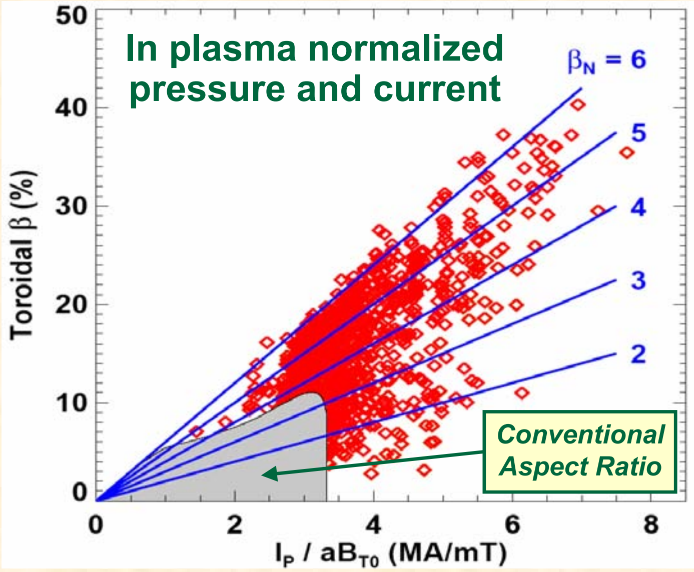
"Locked mode" threshold $\propto$ n, despite factor 10 lower field & factor 5 smaller size

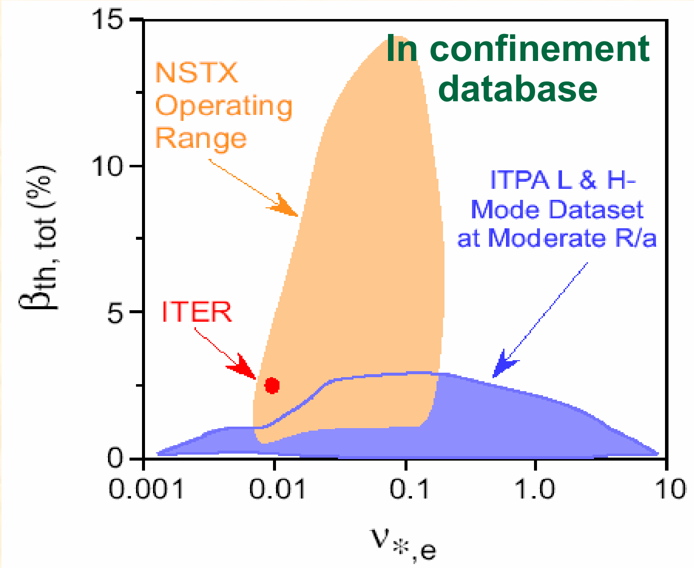


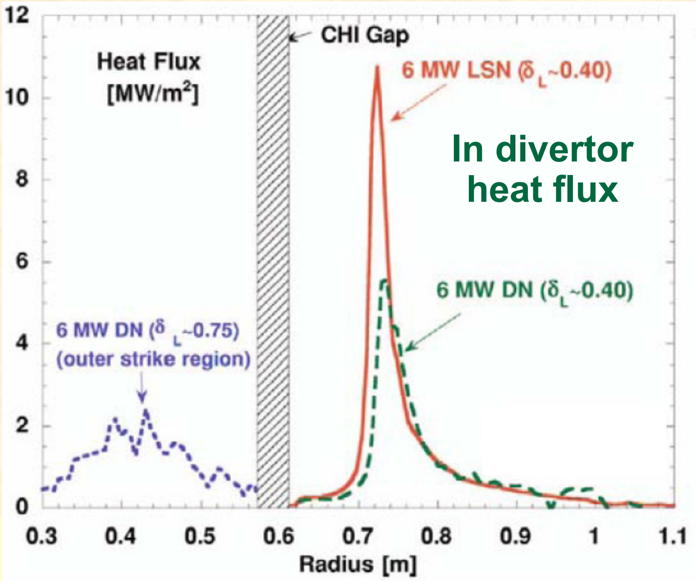
Normalizing disruption quench times (τ_{CQ}) to plasma inductance removes apparent j_p dependence

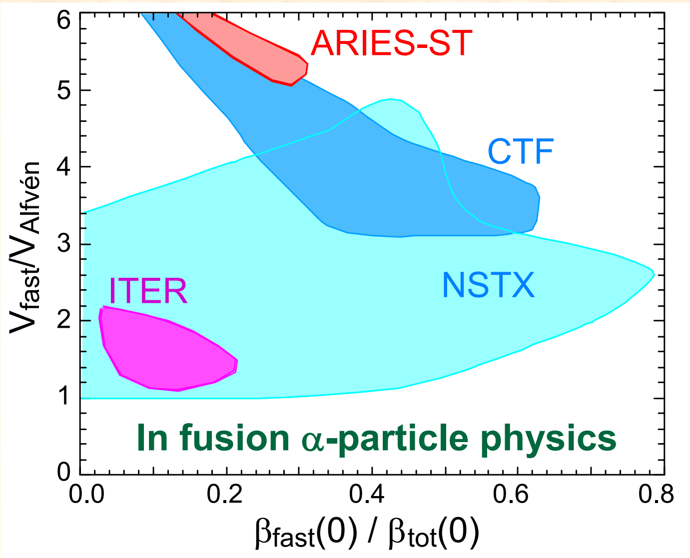


ST complements and extends tokamak physics in investigating key fusion scientific issues



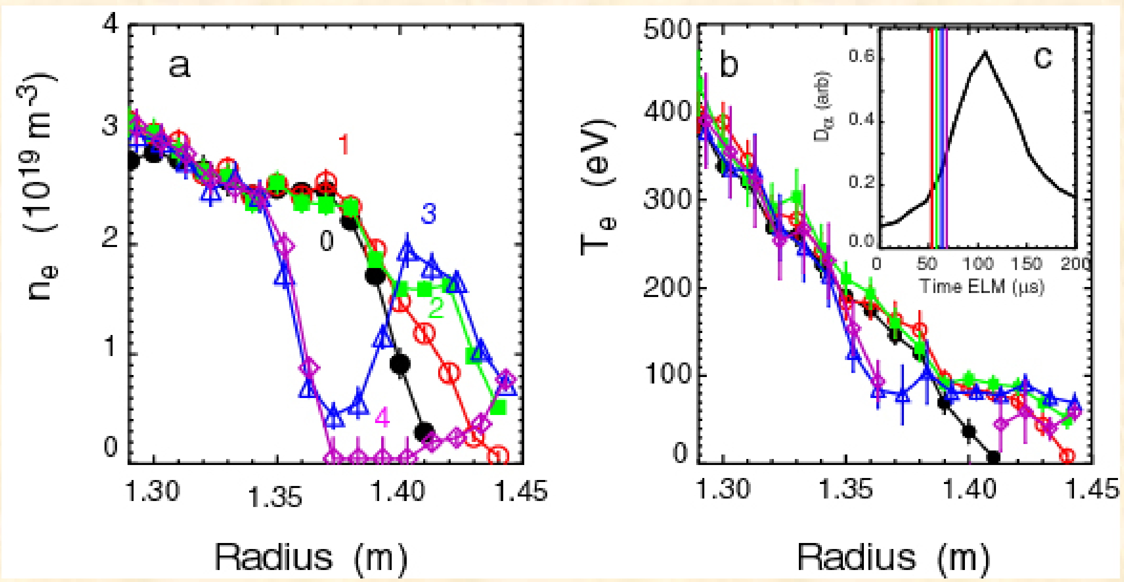




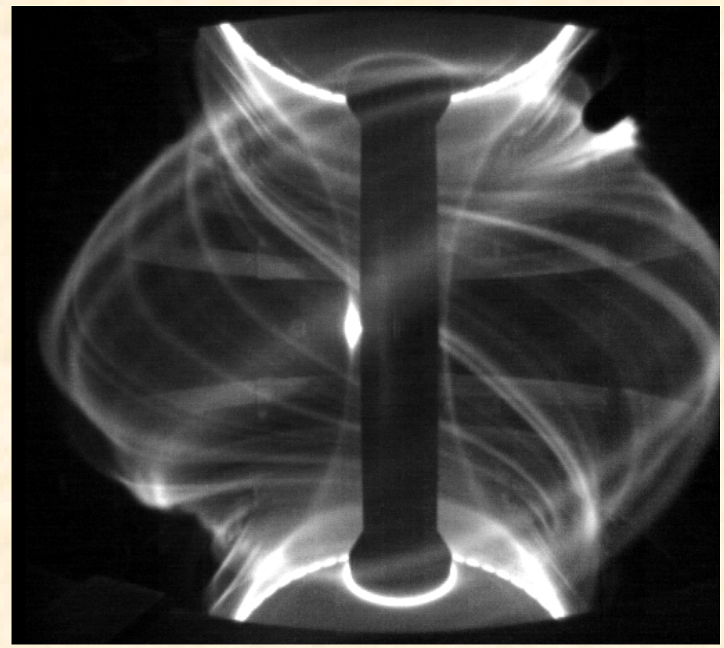


Unique ST features can be exploited to provided new insight into Tokamak physics

Fast evolution and energy & particle content of ELM filaments measured for the first time on MAST



ELM filaments in MAST

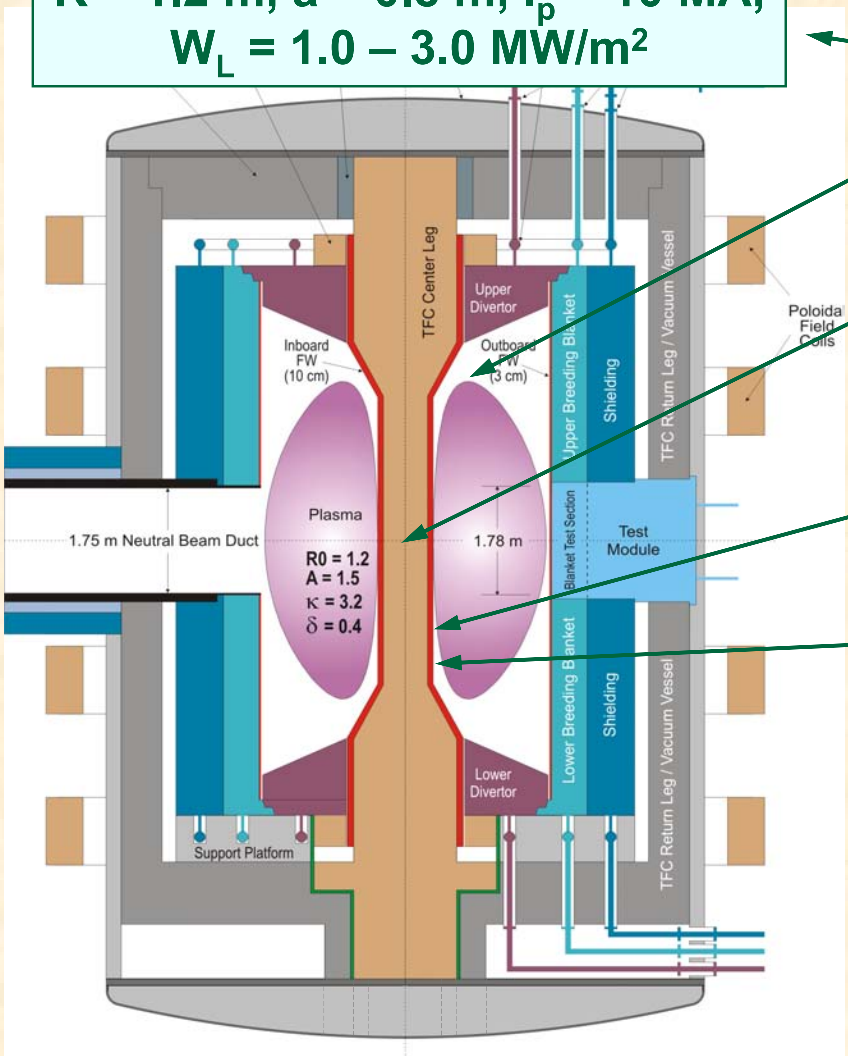


<u>**NdYag Thomson Scattering system**</u>

- **4 lasers fired with 5 μs separation**
- **1-cm spatial resolution**

ST research enables attractive integrated Component Test Facility (CTF) to support fast Demo realization

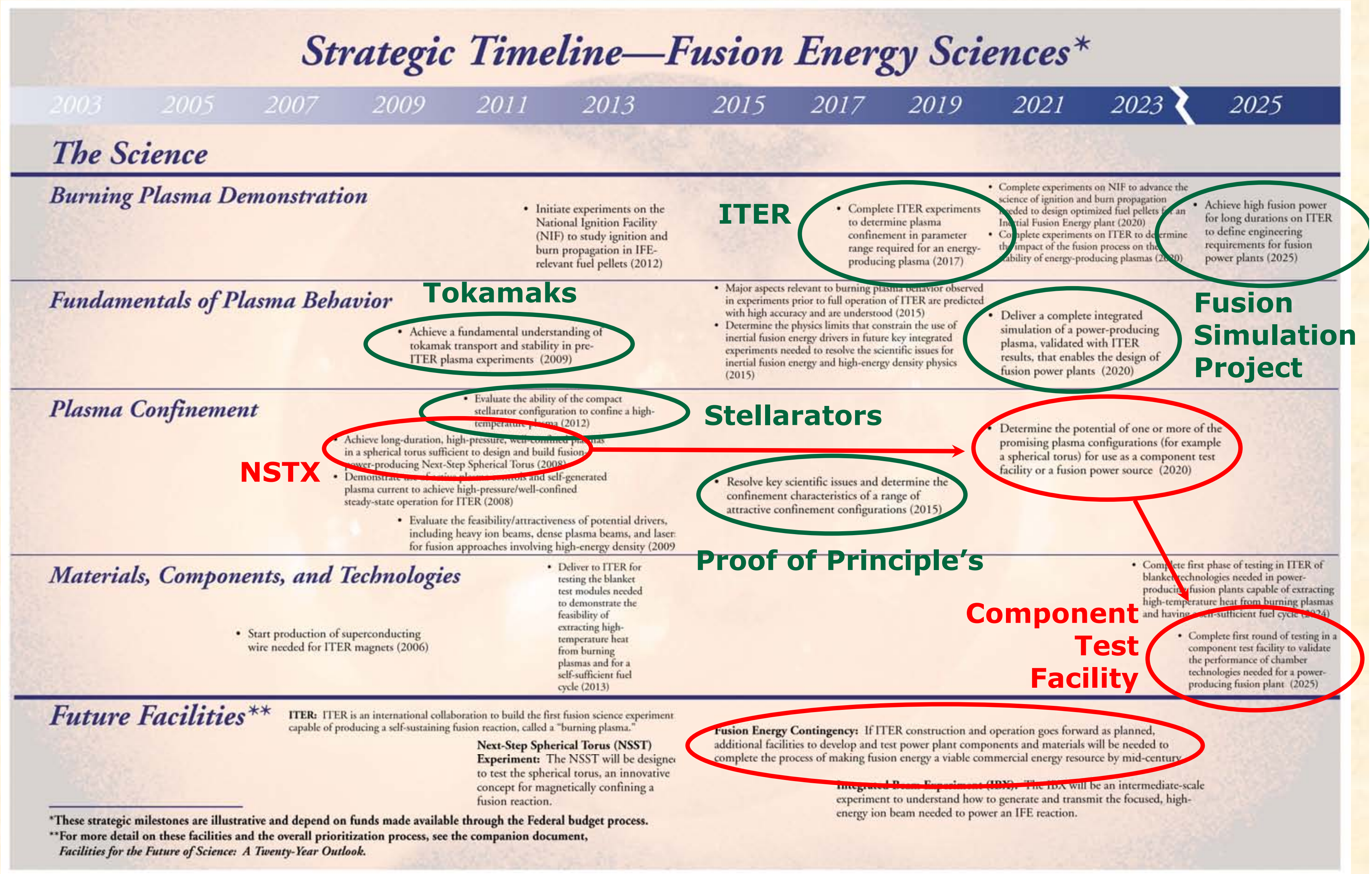
$R = 1.2$ m, $a = 0.8$ m, $I_p \sim 10$ MA, $W_L = 1.0 – 3.0$ MW/m^2



- Small unit size and high W_L
- Natural elongation at low ℓ_i → simple shaping coils
- $I_{TF} \sim I_p$; moderate B_T → slender, single-turn, demountable TF center leg
- No central solenoid → no inboard nuclear shielding
- No inboard blanket → smaller aspect ratio & size
- ~6-7% fusion neutrons lost to center leg → adequate tritium recovery

Peng et al, PPCF 2005

USDOE plan includes CTF to complete first round of testing in 2025 – consistent with the fast strategy



CTF will carry out steady state integrated science & technology testing to ensure Demo success

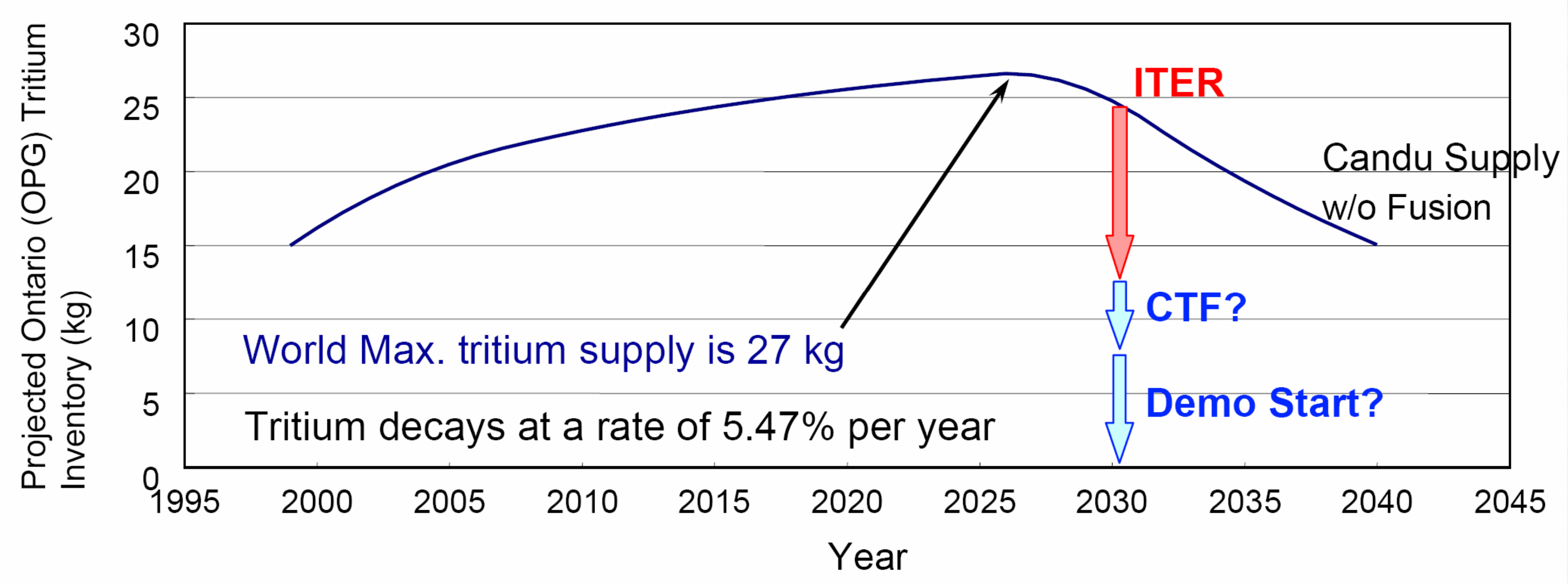
Abdou et al, Fusion Technology 1999

	ITER	CTF	Demo
Tritium self-sufficiency goal (%)	~0	80-100	>100
Burning plasma duration (s)	~10^3	>$10^{6\text{-}7}$	~$10^{7\text{-}8}$
Total 14-MeV neutron fluence (MW-yr/m^2)	~0.3	~6	6-20
14-MeV neutron flux on wall (MW/m^2)	~0.8	0.5-3	3-4
Expected fusion power (MW)	~500	35-210	2500
Total area of (test) blankets (m^2)	~12	~70	~670
Plasma-wall interface index, P/R (MW/m)	24-32	30-100	~100

CTF tests and develops "full-function" chamber components in small sizes

- Under integrated conditions with burning plasma, plasma-material interaction, fusion neutron & tritium physics, combined materials in fusion power environment, etc.
- Covering all chamber systems for power conversion, high heat flux, tritium recovery, toroidal field center leg, and safety & environment, etc.

Small CTF efficiently uses world's limited tritium supply before Demo starts to produce own tritium

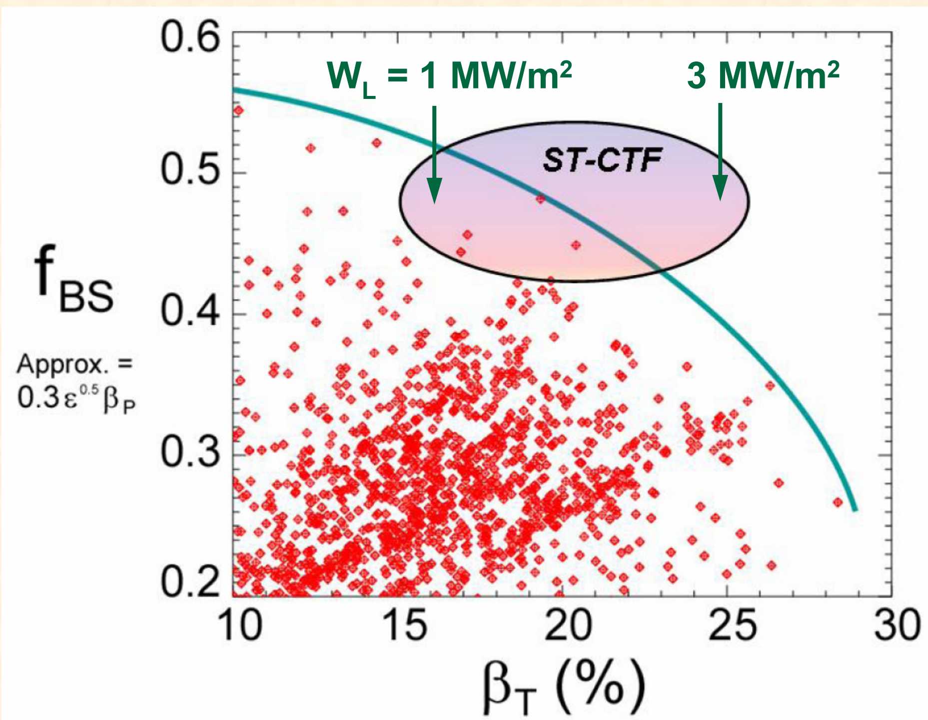


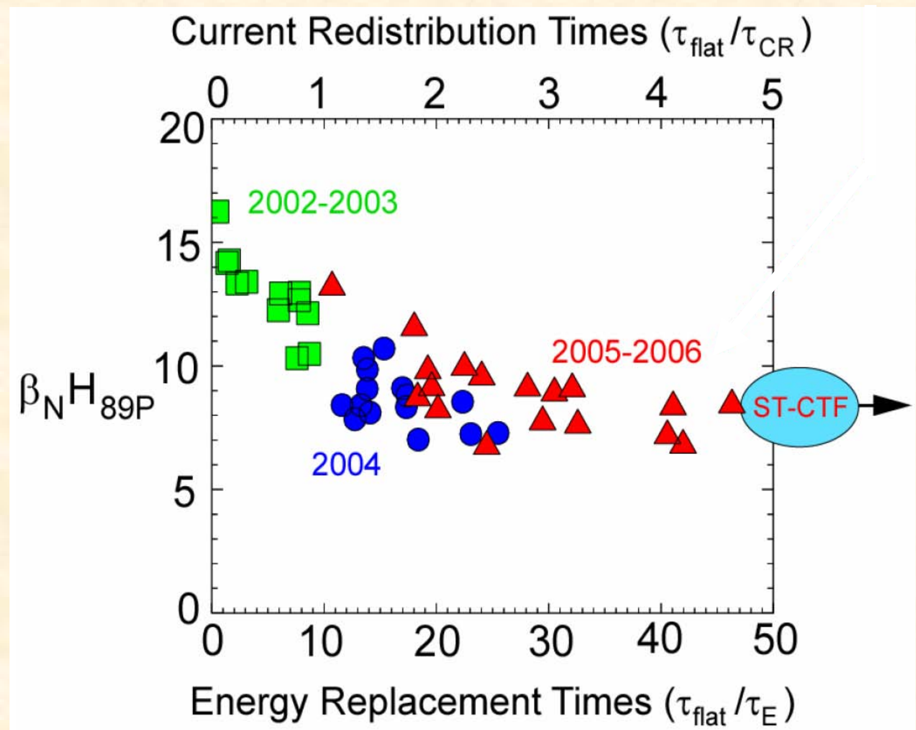
- ITER uses ~11 kg T to provide 0.3 MW-yr/m^2
- Assuming 80% tritium recovery,
  - One CTF needs 5 kg to enable testing to 6 MW-yr/m^2
  - Demo needs 3 kg/month to produce 2500 MW fusion power
- Avoid relying on more fission tritium to sustain testing in Demo

Progress: NSTX obtained physics database for the normalized conditions of basic CTF performance

Normalized sustainment conditions were obtained for projected base CTF W_L

Normalized performance of CTF was sustained for ~50 energy replacement times



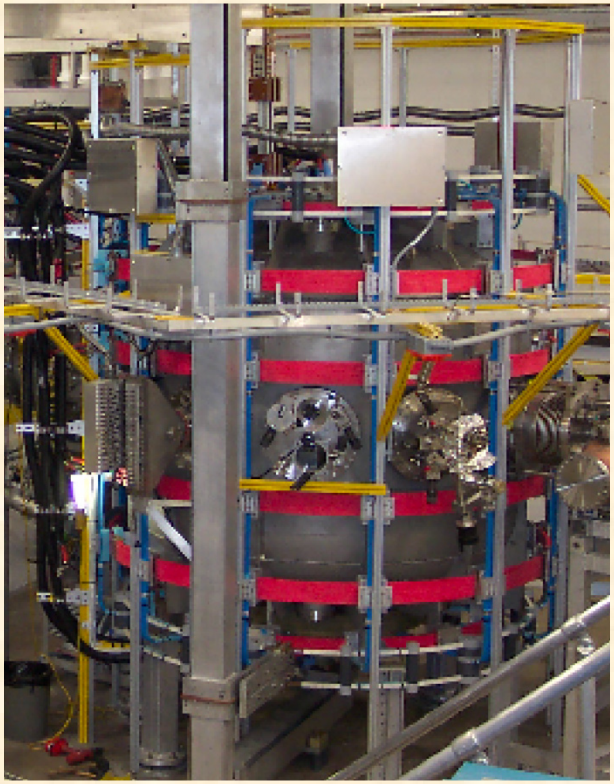


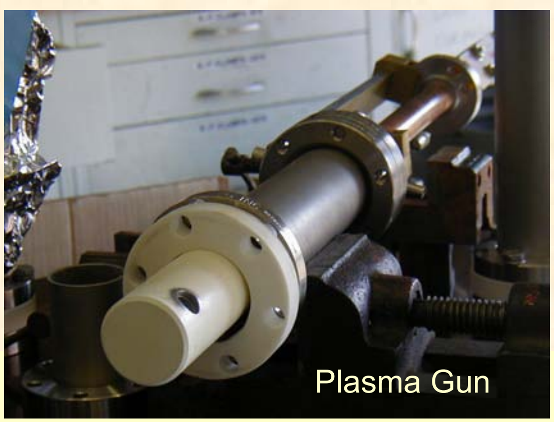
CTF need: more solenoid-free start-up data to enable projections to full current

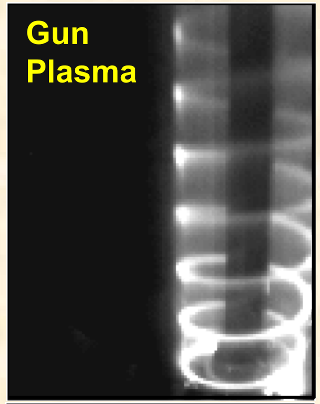
Sustained Parameters	CTF ($\tau >> \tau_{skin}$)	NSTX long pulse ($\kappa \leq 2.5$, $\tau \sim \tau_{skin}$)
I_p/aB_T (MA/m-T)	**4.6 – 6.0**	**3.8**
Safety factor, q_{cyl}	**4.0 – 3.1**	**3.3**
β_N (%-m-T/MA)	**3.0 – 3.9**	**5.1**
β_T (%)	**15 – 25**	**19**
a/ρ_i ($=1/\rho_i^*$)	**~ 50**	**~ 30**
H_{98pby2}	**1.3**	**$\leq$ 1.3**
Start-up to $\mu_0 \ell_i R I_p$ (Wb)	**2.7 – 3.6**	**~ 0.13 (goal)**

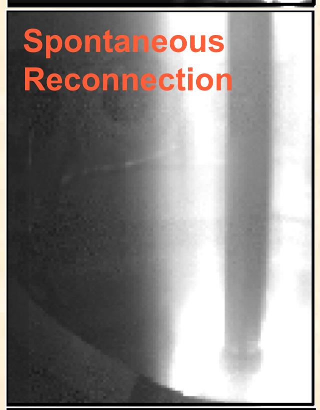
- **Feasibility proven on NSTX (CHI), MAST (merging compression), Pegasus (plasma gun) & LATE, TST-2, JT-60U (RF+VF swing, then NBI up to 0.6MA)**
- **NSTX to investigate combining CHI, EBW, HHFW, NBI & VF swing**
- **Progress: physics basis for CTF stability and confinement established**

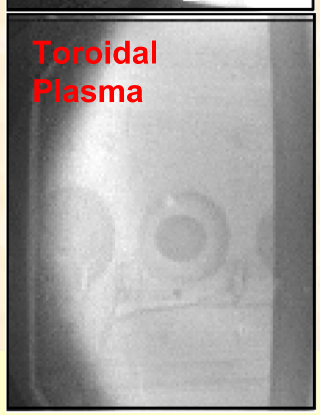
Scientific feasibility of start-up using plasma guns was shown on Pegasus

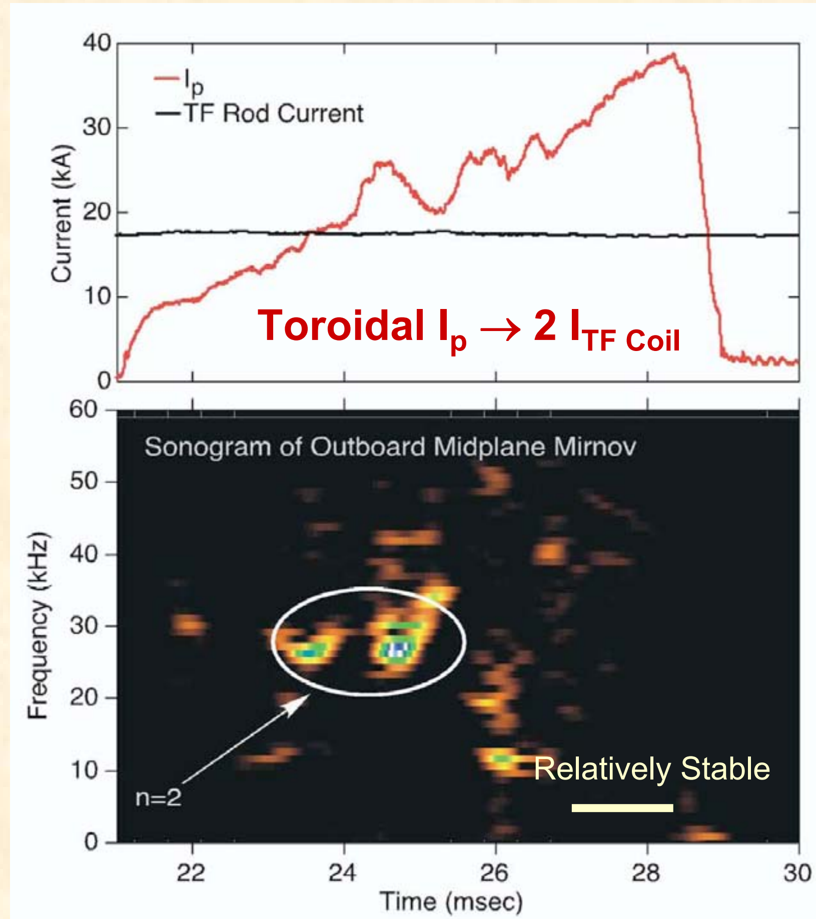












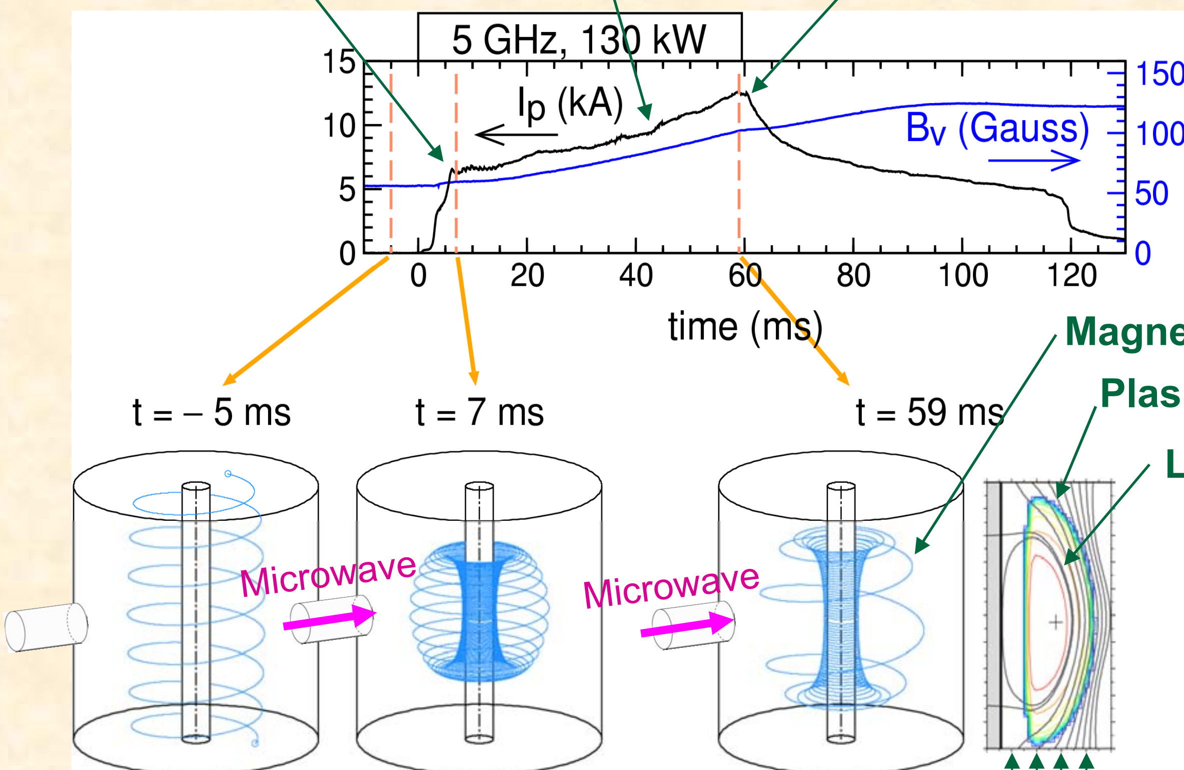
Scale up gun-driven current to ~1 MA (20 times), or ~10 MA (200 times)?

Feasibility of start-up by ECW/EBW + VF was shown in the LATE device at Kyoto U

Formation

Current Ramp-up

I_p = 12kA ~ $0.2I_{TF}$



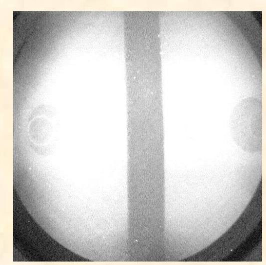
Electron Bernstein wave heating and current drive

Line averaged density > Plasma cutoff density

Magnetic Field Line

Plasma Current Distribution

Last Closed Flux Surface



CCD Plasma Image

Applied Field

Initial Closed Flux Surface

1 m

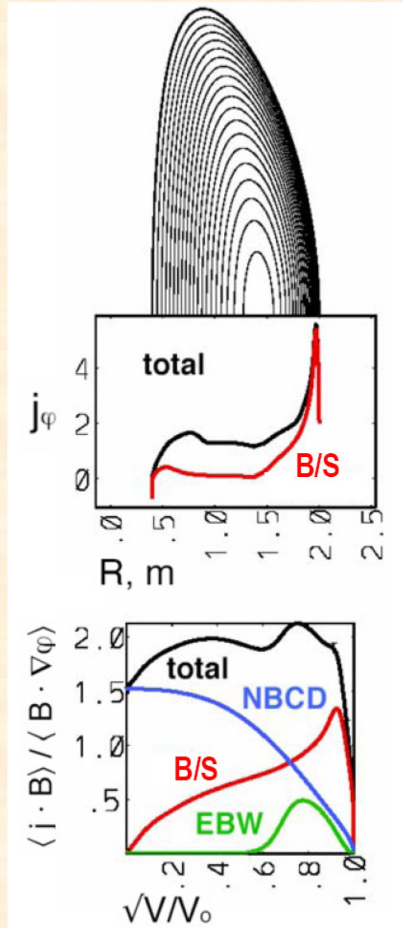
1st 2 3 4th

Multi EC Harmonic Resonance Layers

CTF need: steady state high heat flux divertor physics data to enable projections to high W_L

Sustained Parameters	CTF ($\tau >> \tau_{skin}$)	NSTX so far ($\kappa \leq 2.5$, $\tau \sim \tau_{skin}$)
$V_\alpha/V_{Alfvén}$	3 – 6	1 – 4 ($V_{NB}/V_{Alfvén}$)
I_{EBW}/I_p	~ 0.1	~ 0.1 (goal)
I_{BS}/I_{CD} fractions	0.45/0.55	0.3-0.4/0.2-0.1
P/R (MW/m)	30 – 100	$\leq$ 9
SOL area expansion	10 – 20	~ 5
Radiation fraction (%)	50 – 70%	$\leq$ 30

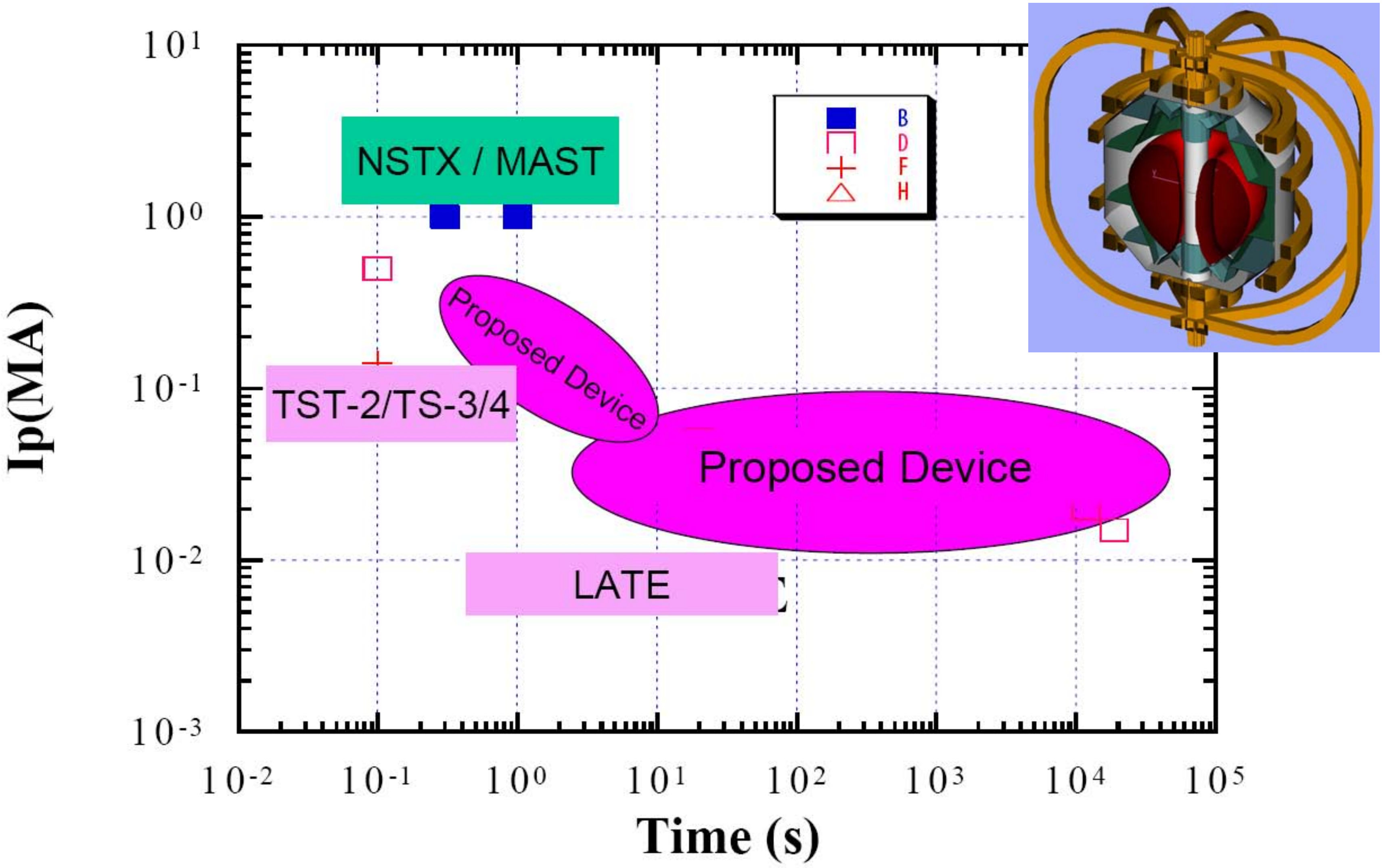
CTF plasma shape & stable current profile



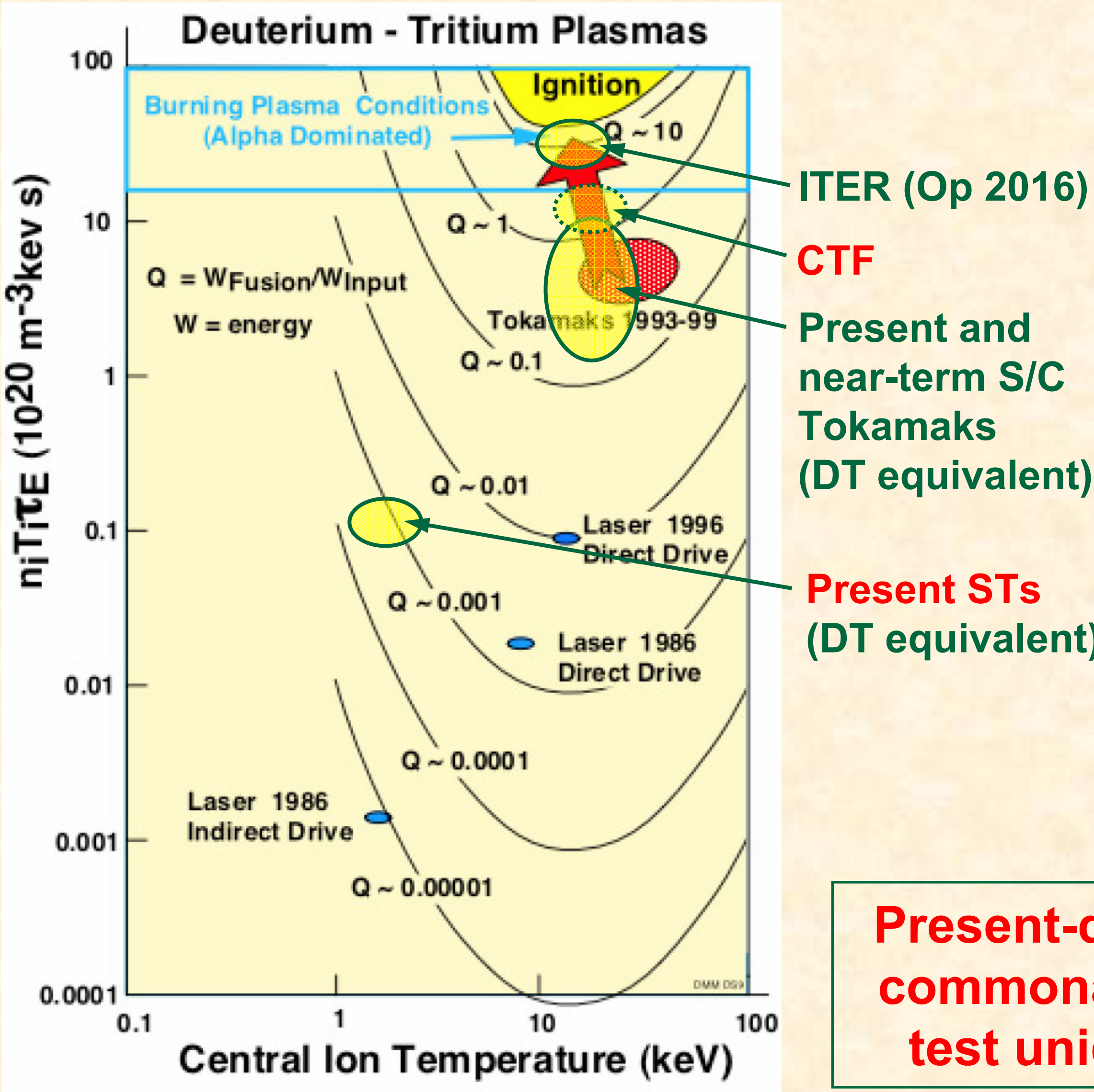
- **High priority for ITER, large tokamaks, S/C tokamaks, and new ST in Japan**
- **NSTX to investigate liquid lithium divertor target physics solution, and EBW heating & current drive**
- **Progress: basis for NBI & bootstrap current drive physics established**

New experiment at Kyushu U will begin in 2008 research on sustained ST plasma-wall physics and engineering

QUEST



CTF with Q ~ 1-3 can take advantage of the full physics database anticipated for ITER start



Fusion Demo Requires

- Ion temperature ~ 10keV
- Density x energy containment time, $n\tau_E \geq 10^{20}\ m^{-3}s$
- Pressure ~ 1 atm
- Very high max plasma facing surface heat flux (divertor) $\leq$ 20 MW/m^2
- Fusion energy gain, Q ~ 20

Present-day ST experiments verify commonalities with tokamaks and test unique CTF physics needs.

ST can support fast implementation of fusion Demo in unique, important ways

- EU, Japan, Korea plan to implement Demo following ITER
  - Opportunities for ST to support this strategy
- Propose that ST research community
  - Support and benefit from USBPO-ITPA activities
  - Complement and extend tokamak experiments
  - Enable CTF to carry out integrated testing to ensure Demo success
- Attractive CTF option identified
  - Included in the USDOE fusion plan
  - Physics basis already broadly established
  - Can leverage tokamak advances toward ITER
  - Except for solenoid-free current start-up & high heat flux divertor – being pursued worldwide; very-high heat flux, D-D device?