Progress toward Fusion Component Test Facility (CTF), an integrated test facility

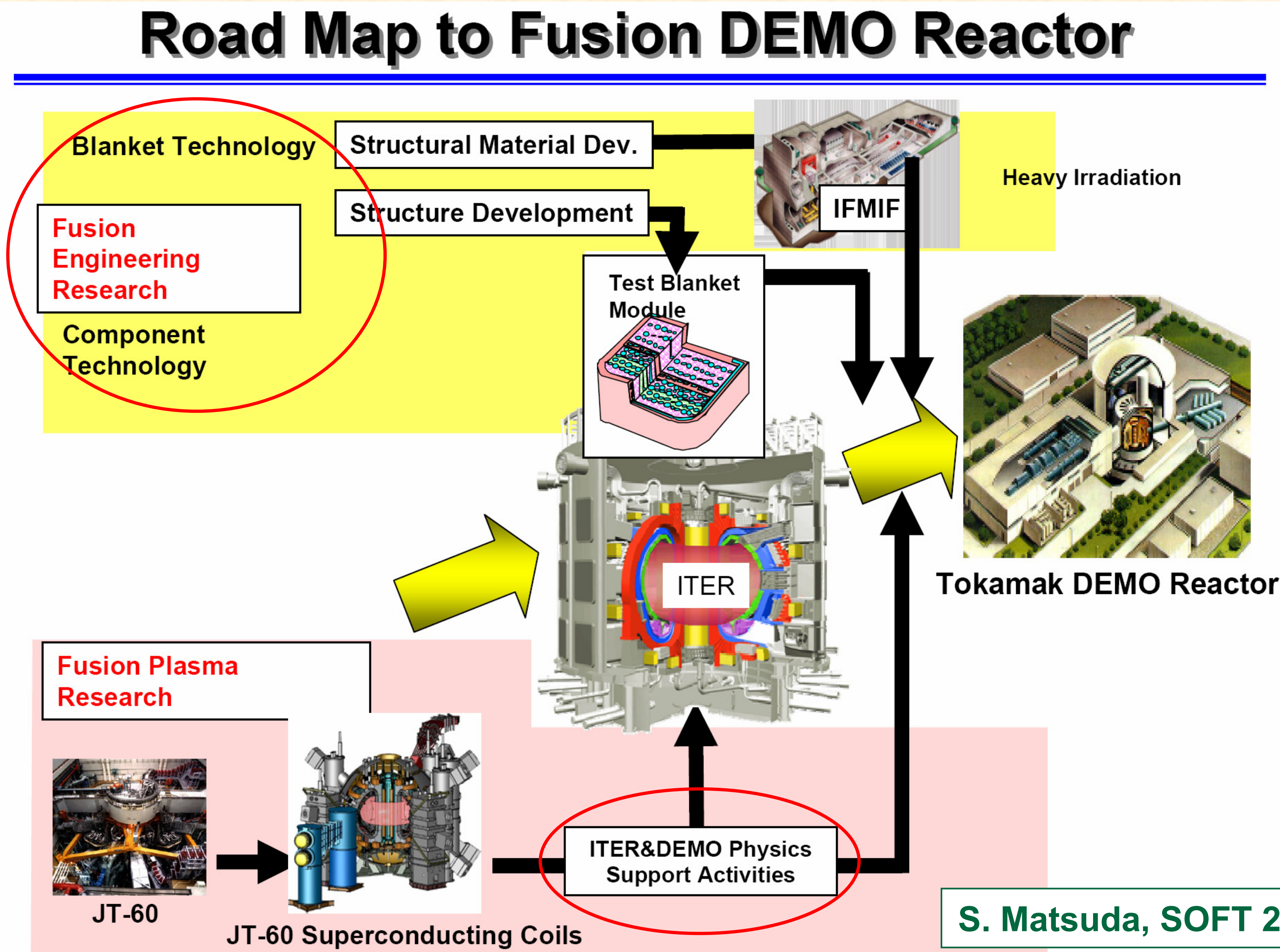
ST research is making rapid progress toward CTF, a fusion energy science & technology facility to enable fast implementation of Demo

1. Opportunities in the "Demo after ITER" strategy
2. How to support this fast Demo strategy?
3. Attractive CTF option for steady state integrated testing
4. Broad common progress – two CTF physics R&D needs

Martin Peng, NSTX Program Director

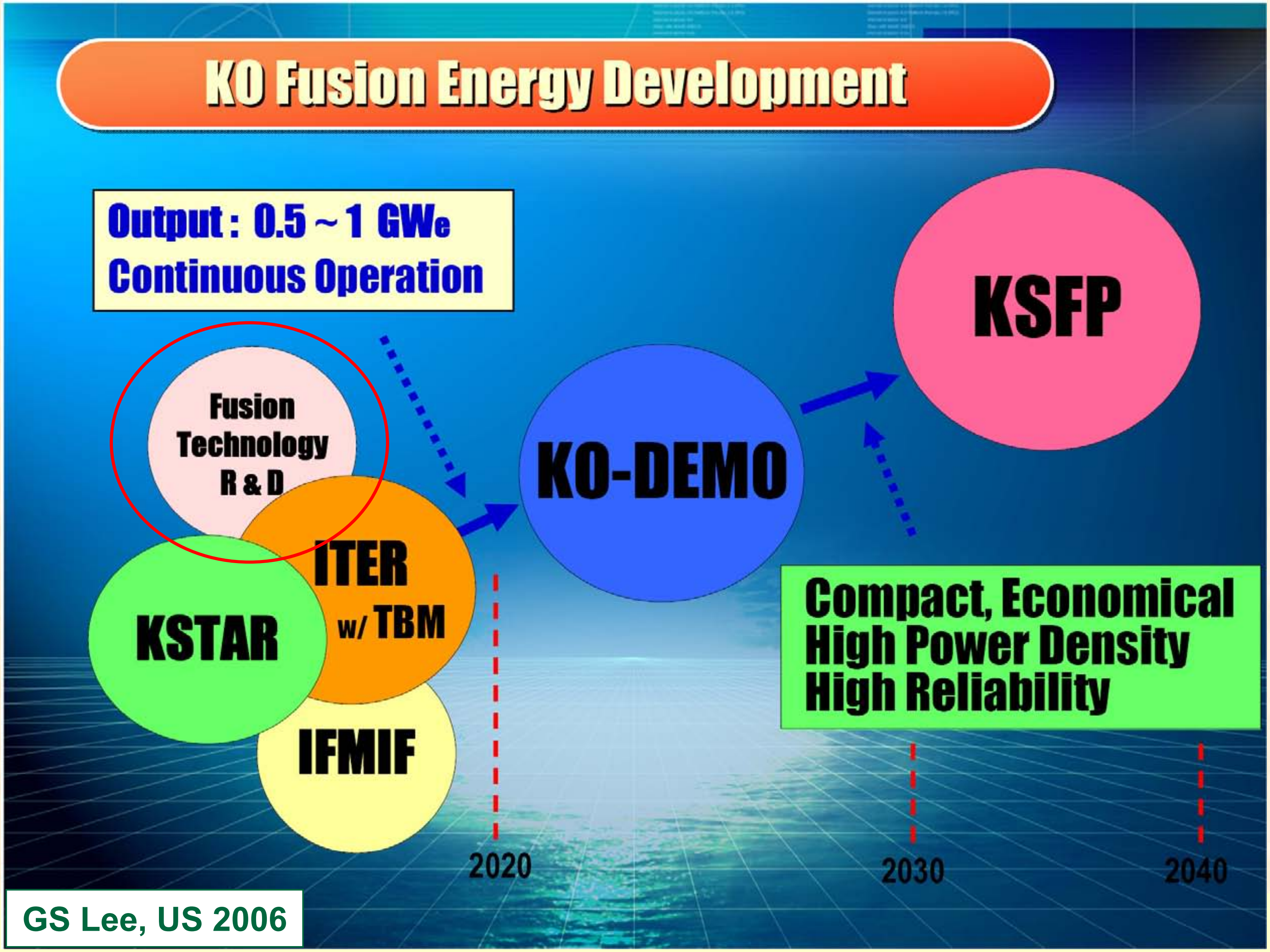
48th Annual Meeting of the Division of Plasma Physics
October 30-November 3, 2006, Philadelphia, PA

EU-Japan's Broader Approach to Demo introduces opportunities in fusion physics and components R&D



S. Matsuda, SOFT 2006

Korean fusion energy development plan introduces opportunities to accelerate fusion technology R&D



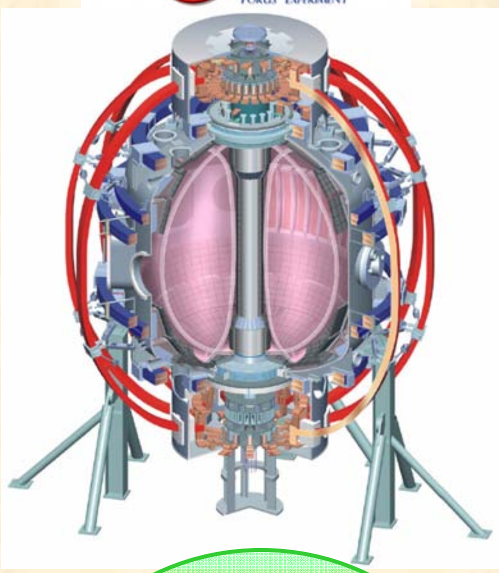
GS Lee, US 2006

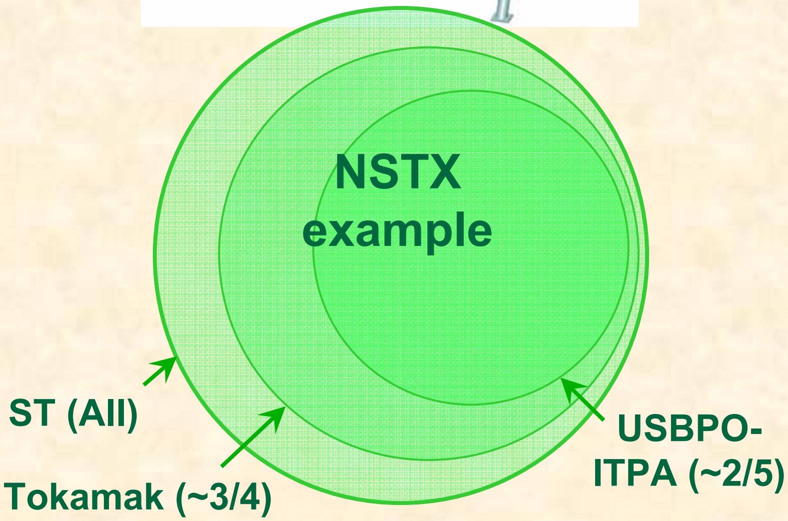
We encourage that the ST research community addresses issues in support of this strategy

2. How to support this fast Demo strategy?

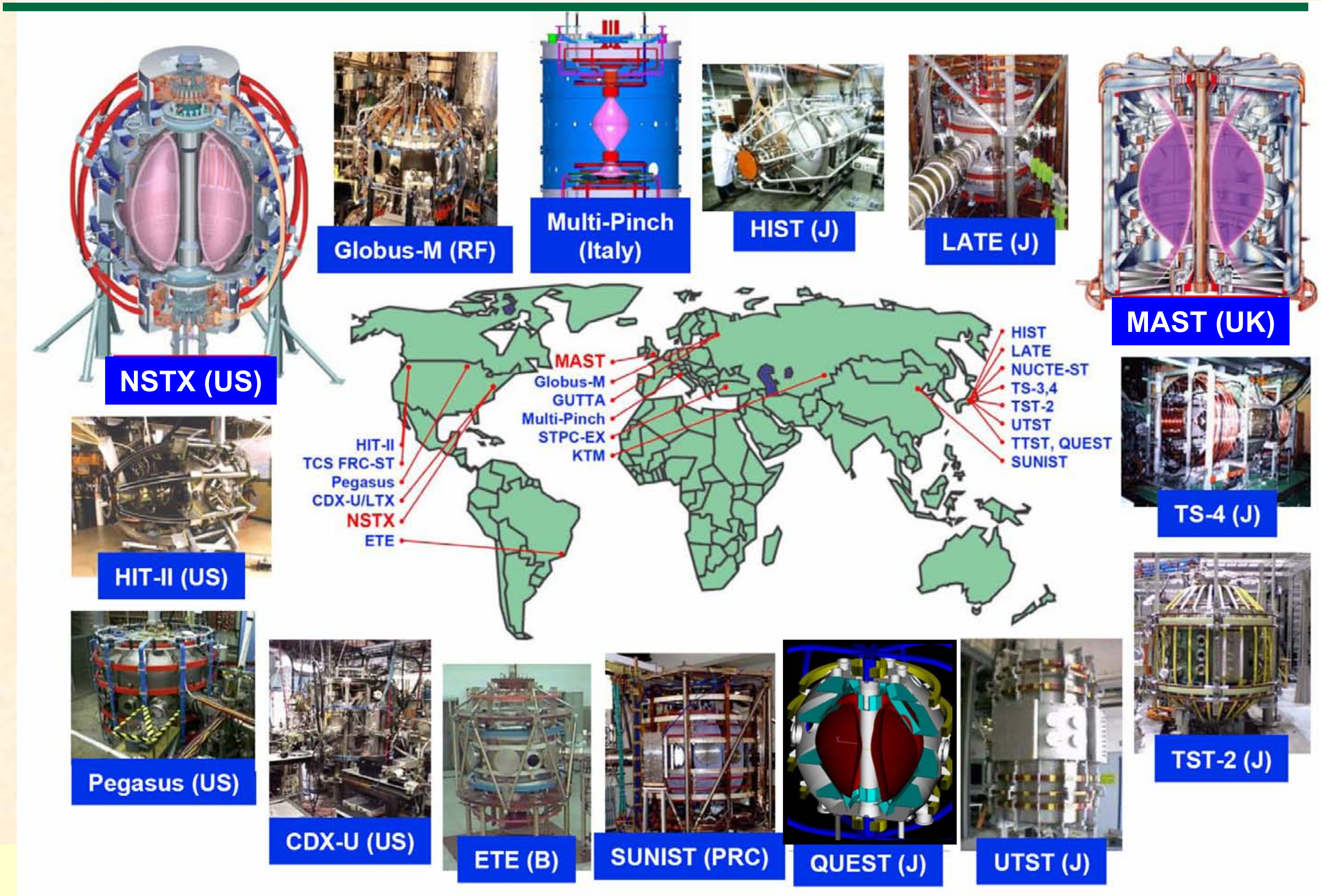
- **Support and benefit from USBPO-ITPA activities** using the physics breadth provided by ST in preparation for burning plasma research in ITER.
- **Complement and extend tokamak physics experiments,** by maximizing synergy in investigating key scientific issues of toroidal fusion plasmas
- **Enable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

NSTX NATIONAL SPHERICAL TORUS EXPERIMENT





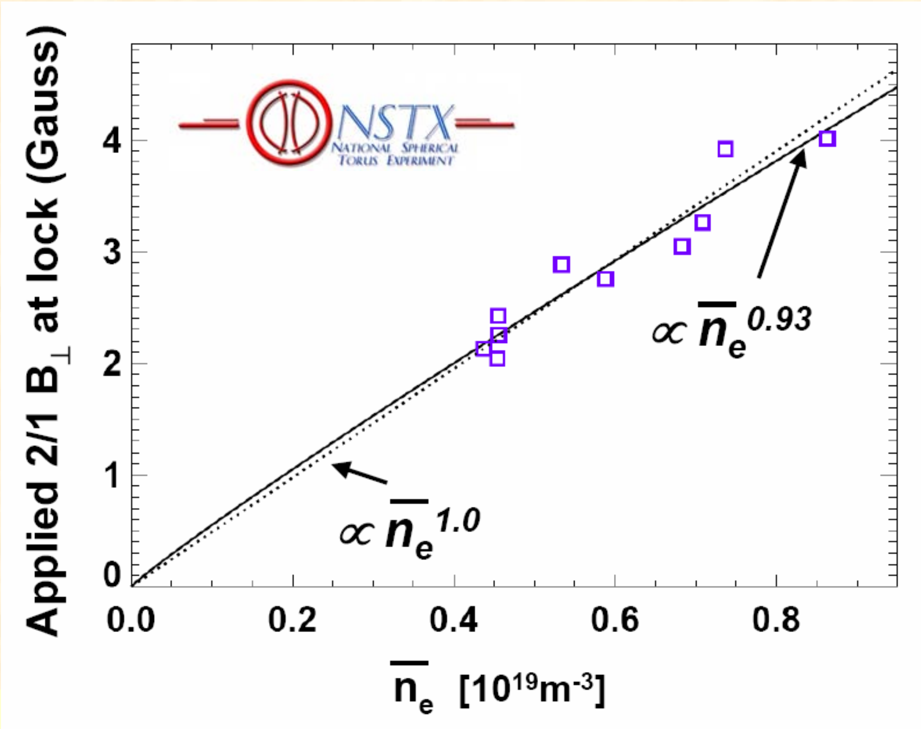
World Spherical Tokamak research has expanded to 22 experiments addressing all key physics issues



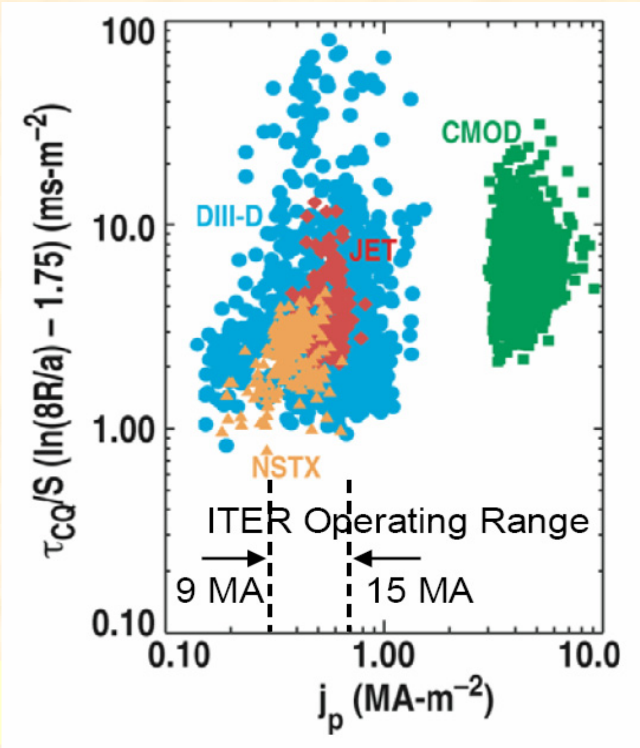
ST supports and benefits from USBPO-ITPA in preparing for burning plasma research on ITER

- NSTX completed in 2006 half of the 22 ITPA 2006-7 joint experiments
- ST "exceptions" prove the ITER "rules," test theory, enhance predictive capability, and verify commonalities

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



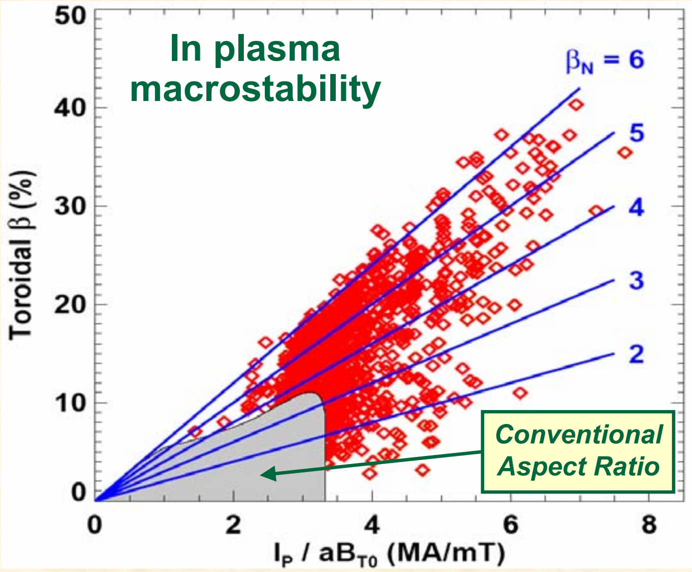
Normalizing disruption quench times (τ_{CQ}) to plasma inductance removes apparent j_p dependence

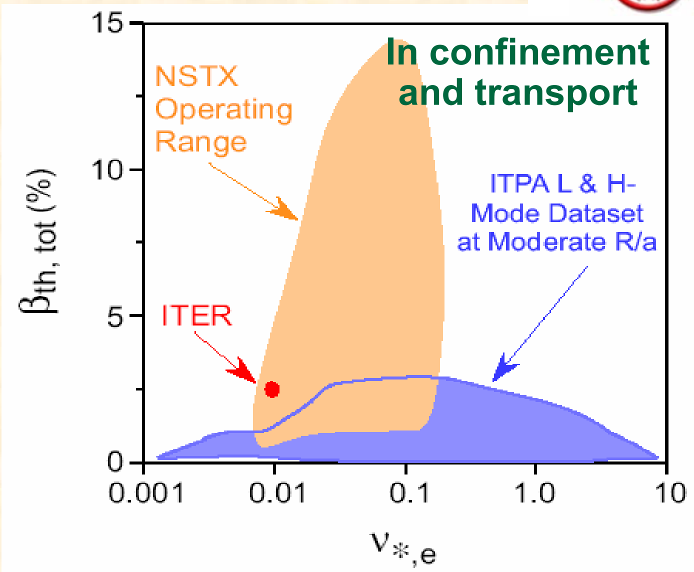


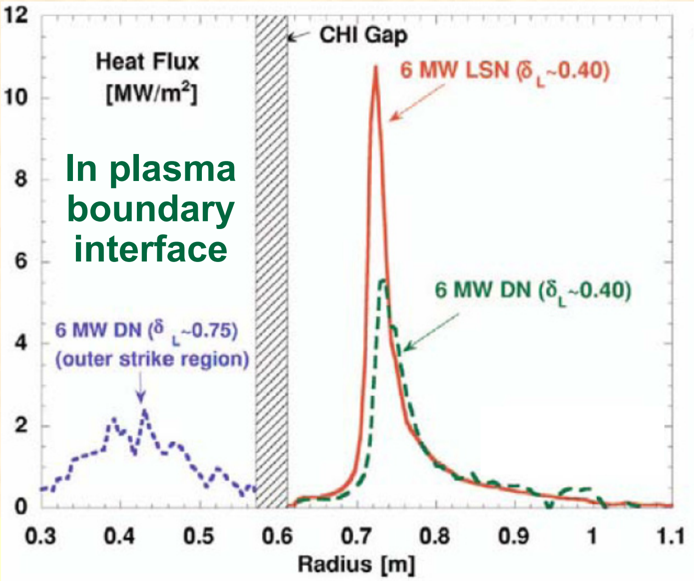
NSTX devoted 2/5 run time to, and completed half of, the ITPA 2006-2007 Joint Experiments commitment

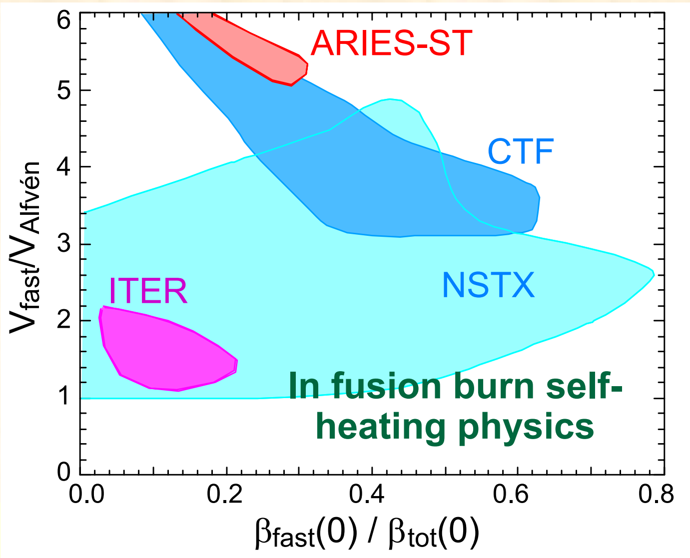
ID No	2006 Proposal Title	Devices	Status
CDB-2	Confinement scaling in ELMy H-modes: β degradation	AUG, DIII-D, JET, JT-60U, Tore-Supra(L), MAST, NSTX	complete
MDC-2	Joint experiments on resistive wall mode physics	DIII-D, JET, NSTX, JT-60U, AUG and TEXTOR	complete
MDC-9	Fast ion redistribution by beam driven Alfvén modes and excitation threshold for Alfvén cascades	JT-60U, JET, DIII-D, NSTX, MAST, AUG	complete
TP-8.1	ITB Similarity Experiments	MAST, NSTX	complete
PEP-9	Pedestal similarity study	DIII-D, MAST, NSTX	complete
DSOL-15	Inter-machine comparison of blob characteristics	C-Mod, NSTX, TJ-II,JET, TCV, HT-7, Tore-Supra, AUG, JT-60U	complete
DSOL-18	Impurity migration and deposition study	NSTX, AUG, JET	complete
SSO-2.2	MHD effects on q-profile and confinement for hybrid scenarios	AUG, JET, DIII-D, JT-60U, NSTX	complete
SSO-2.1	Complete mapping of hybrid scenario	JET, JT-60U, DIII-D, AUG, NSTX	complete
CDB-6	Improving the condition of Global ELMy H-mode and Pedestal databases: Low A	MAST, NSTX, DIII-D	partial
CDB-8	rho* scaling along an ITER relevant path at both high and low beta	JET, DIII-D, C-mod, AUG, NSTX	partial
CDB-9	Density profiles at low collisionality	JET, DIII-D, C-mod, AUG, JT-60U, TCV, Tore-Supra, MAST, FTU, NSTX,T-10	partial
MDC-5	Comparison of sawtooth control methods for neoclassical tearing mode suppression	AUG , DIII-D, JET, NSTX, TCV and HL2A, C-mod, FTU	partial
MDC-6	Low beta error field experiments	C-mod, TEXTOR, MAST, DIII-D, NSTX, JET(done)	partial
PEP-16	Small ELM regime comparison	NSTX, MAST, C-mod	partial
DIAG-1	Assessment of the effect of noise on vertical velocity measurement	JET, JT-60U, TCV, NSTX, AUG	partial
DIAG-2	Environmental tests on Diagnostic First Mirrors (FMs)	T-10, TEXTOR, Tore-Supra, JET, DIII-D, TCV, AUG, LHD, FTU, NSTX, C-mod, JT-60U	partial
MDC-4	Neoclassical tearing mode physics - aspect ratio comparison	AUG, MAST, NSTX, DIII-D	2007
TP-6.3	NBI-driven momentum transport study	DIII-D, JT-60U, NSTX, MAST, JET	2007
TP-9	H-mode aspect ratio comparison	NSTX, DIII-D, MAST,T-10	2007
PEP-10	The radial efflux at the mid-plane and the structure of ELMs	AUG, MAST, NSTX, C-mod	2007
PEP-13	Comparison of small ELM regimes in JT-60U and AUG and JET	AUG, JT-60U, JET, NSTX	2007
SSO-2.3	ρ* dependence on confinement, transport and stability in hybrid scenarios	DIII-D, JET, AUG, JT-60U, NSTX	2007

ST complements and extends Tokamak physics in investigating key fusion scientific issues for both



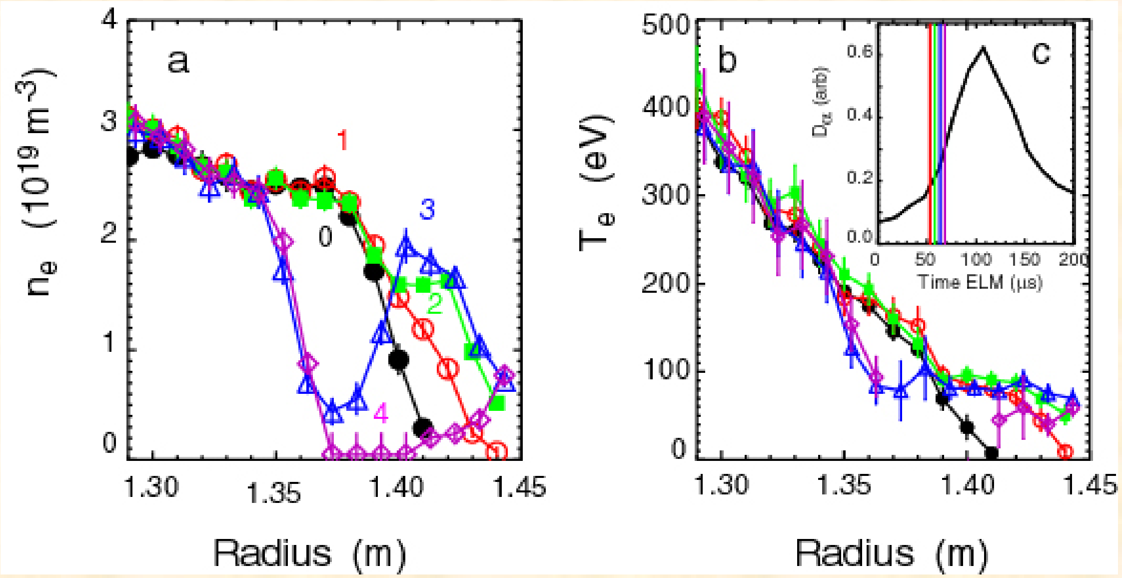




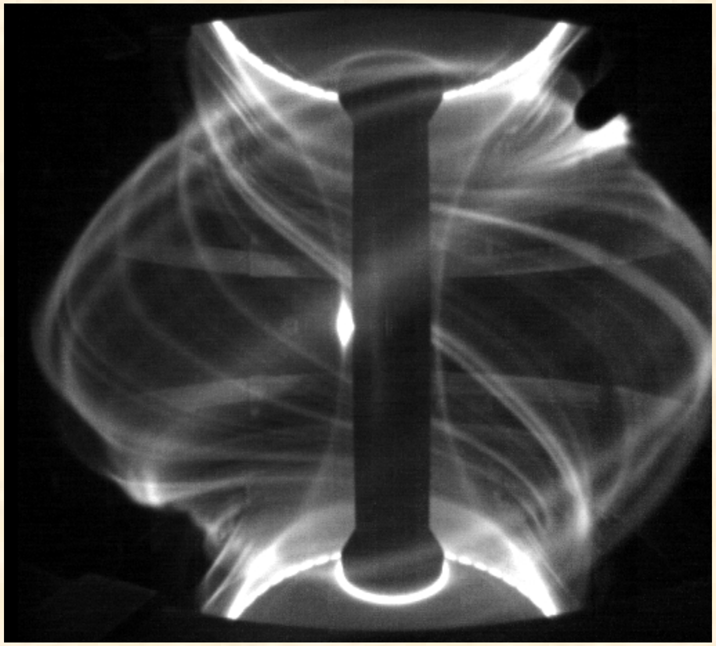


Unique ST features enable new insight into common critical ST & Tokamak physics issues

Evolution of energy & particle content of ELM filaments frozen in action for the first time on MAST



Very fast camera image of full Edge Localized Mode (ELM) filaments in MAST



NdYag Thomson Scattering system

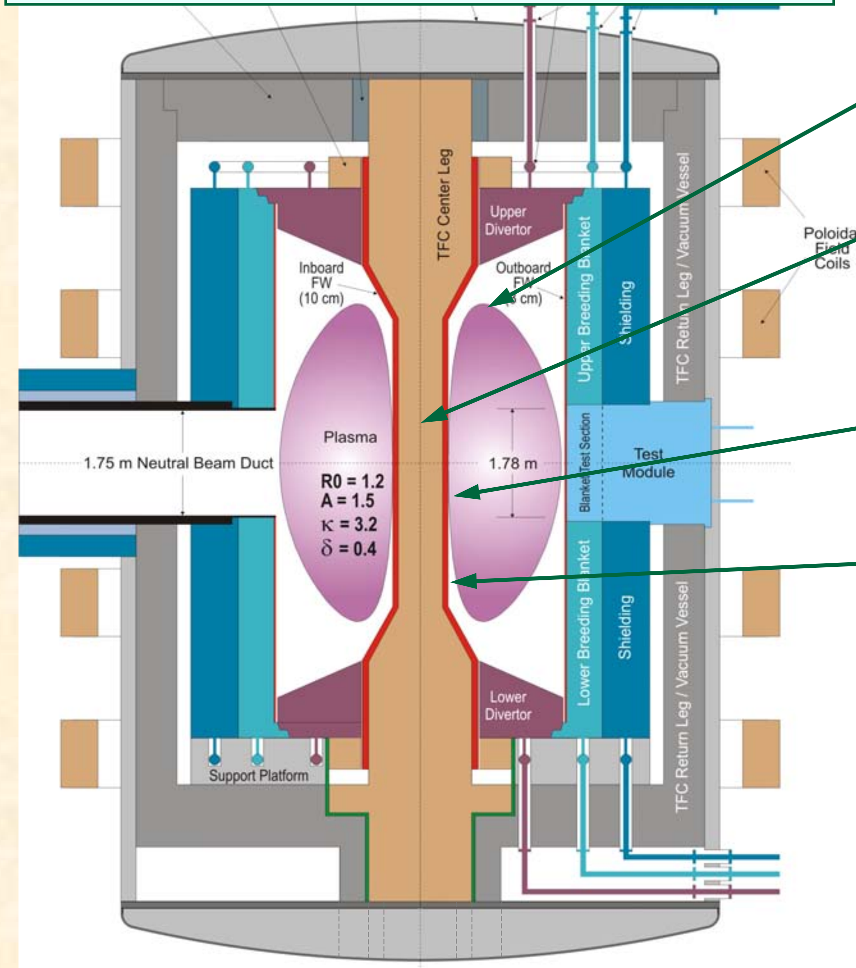
- **4 lasers fired with 5 μs separation**
- **1-cm spatial resolution**

UKAEA Fusion Working in Europe

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

3. An attractive CTF option

$R = 1.2$ m, $a = 0.8$ m, $I_p = 8 - 12$ MA,
$W_L = 1.0 – 3.0$ MW/m^2

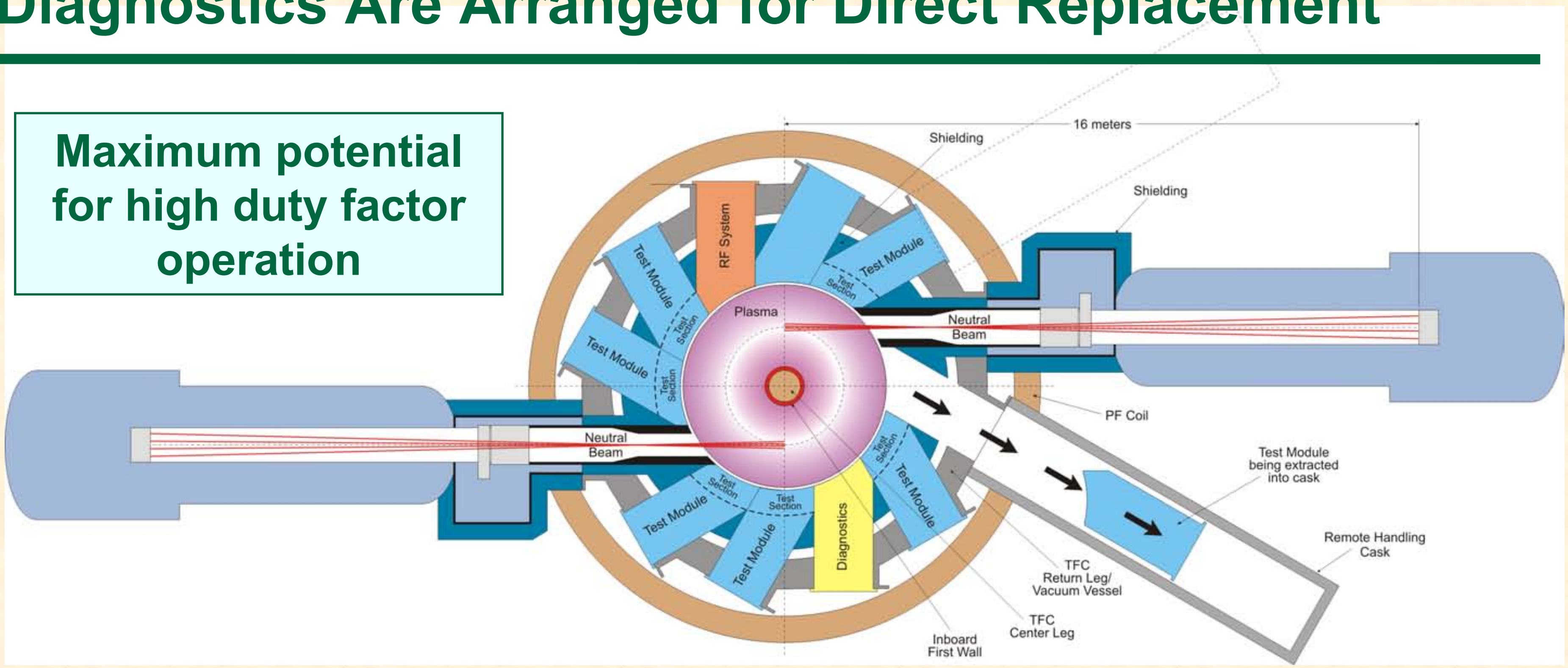


- Small unit size and high W_L
- Natural elongation → simple shaping coils
- $I_{TF} \sim I_p$; moderate B_T → slender, single-turn, demountable TF center leg
- No central solenoid → no inboard nuclear shielding
- No inboard blanket → smaller aspect ratio & size
- ~6-7% fusion neutrons lost to center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

Peng et al, PPCF 2005

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

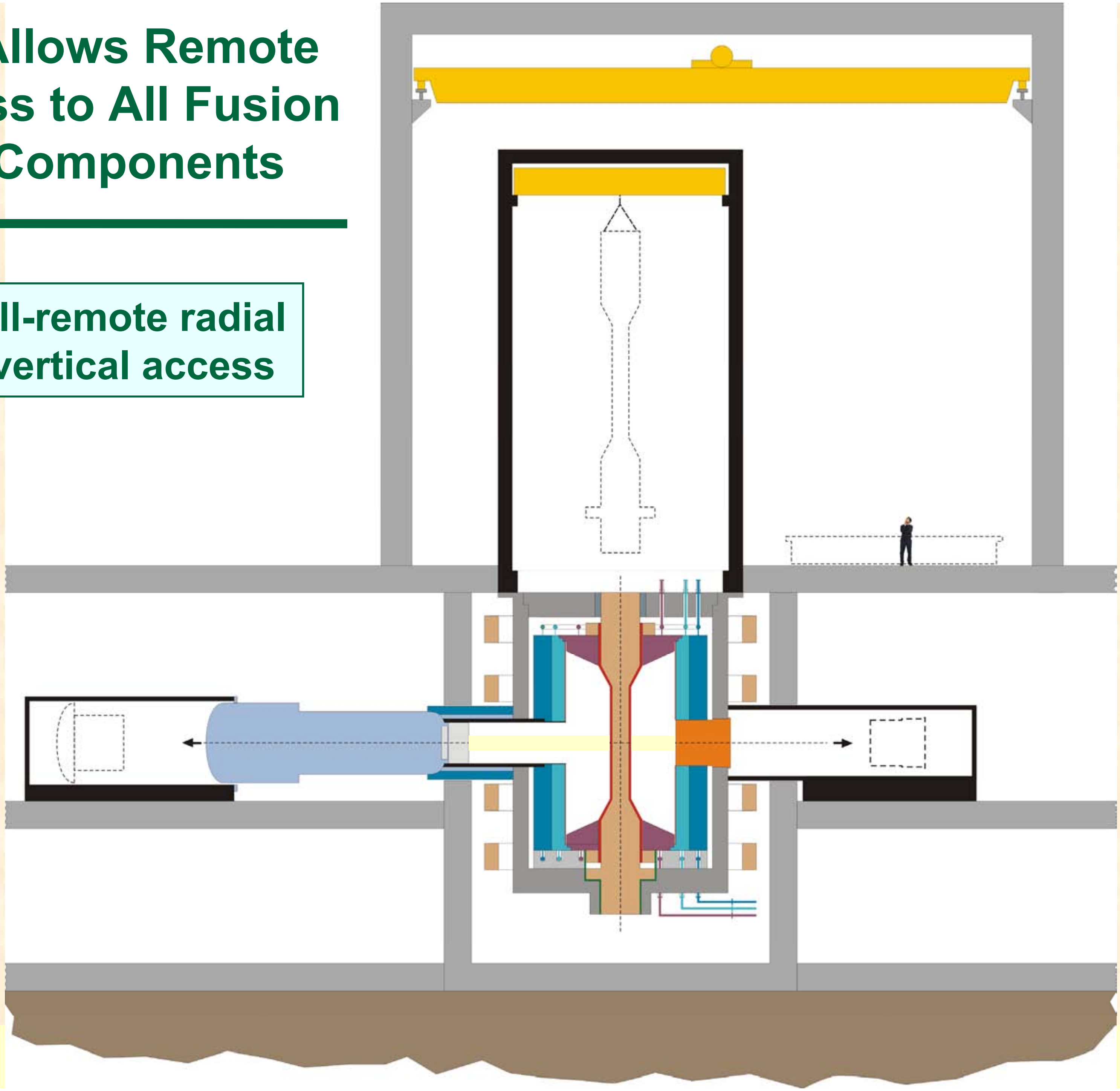
Maximum potential for high duty factor operation



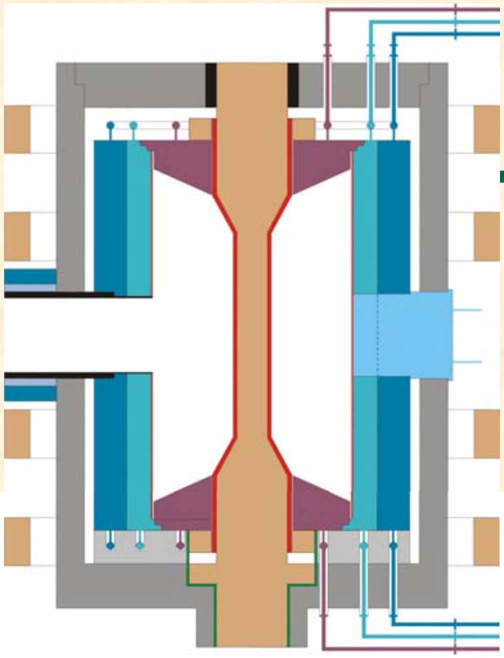
- 8 mid-plane blanket test modules provides ~ 15 m^2 at maximum flux
  - Additional cylindrical blanket test area > 50 m^2 at reduced flux
- 3 m^2 mid-plane access for neutral beam injection of 30 MW
- 2 m^2 mid-plane accesses for RF (10 MW) and diagnostics
- All modules accessible through remote handling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

CTF Allows Remote Access to All Fusion Core Components

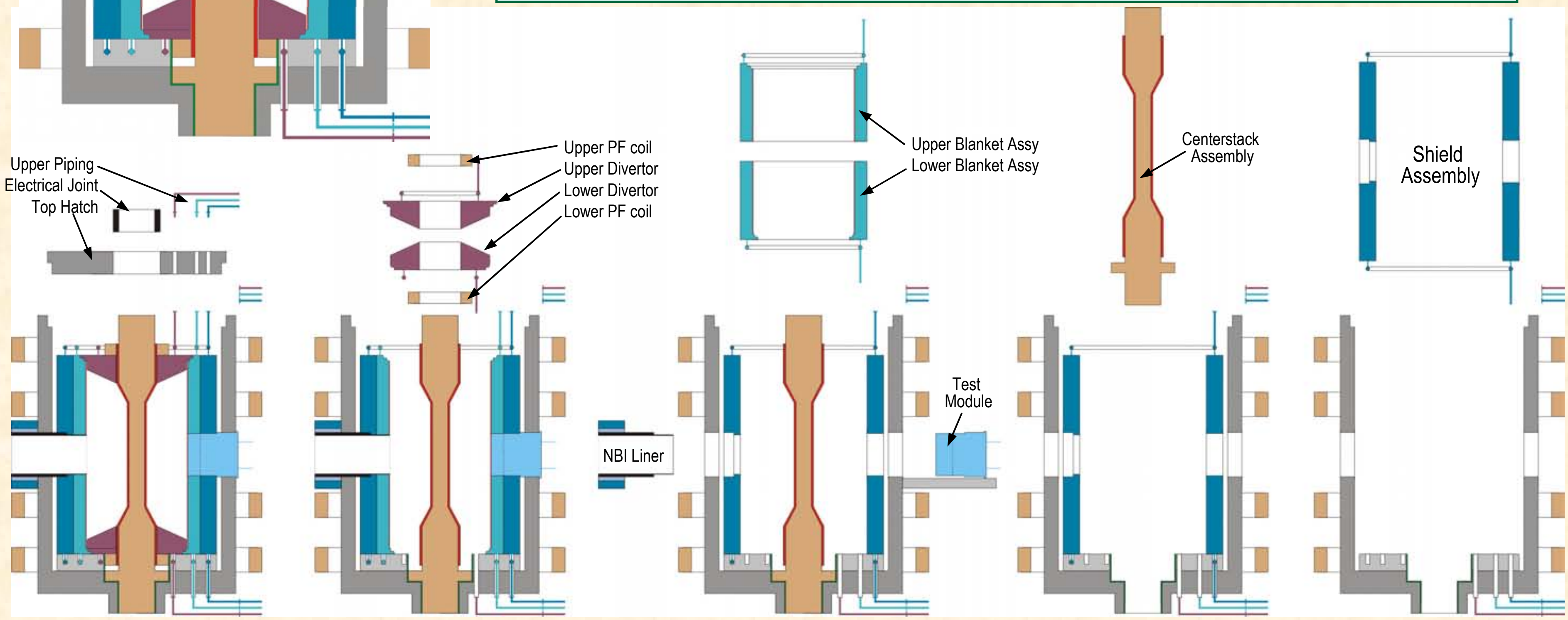
- Full-remote radial & vertical access



Full remote assembly / disassembly readily allows D-D → D-T switch



- Cost scaled from ITER ~ $1,470M (~2002 $)
- Initial D-D implementation, to verify high heat flux components and full remote operation at half current (see, 3D solids simulation)



- 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

Comparative costing of CTF (W_L=1 MW/m^2) – I (in 2002 M$)

SuperCode Costing Components	R_0=1.2m	Comments
1. Toroidal Device	193	
– TF magnets	38	
• *TFC center post*	*(12)*	$U_{TFcenter}$ = \$0.075M/ton (single-turn cooled GlidCop)
• *TFC outer magnet (VV)*	*(26)*	$U_{TFouter}$ = \$0.03M/ton (single-turn Al, combined with VV)
– PF magnets	50	U_{PF} = \$0.058M/ton (no OH solenoid)
– Device structure	11	U_{MS} = \$0.052M/ton
– Vacuum vessel	0	Combined with TFC outer conductor
– Blanket modules	10	ITER-FEAT: 220; FIRE (reflector): 19*; CTF: basic T-breeding blankets cost 1/3 of advanced test blankets**
– Device, penetration shielding	43	
– Divertor, PFCs	29	ITER-FEAT: 109; FIRE: 42; CTF: U_{Div} = 1.61/m^2
– Fueling	12	ITER-FEAT: 10; FIRE: 9
2. Device Ancillary Systems	187	
– Machine assembly tooling	29	ITER-FEAT: 72; FIRE: 0; CTF only: $\propto R^{3/4}$
– Remote handling equipment	152	ITER-FEAT: 145, FIRE: 101; CTF only: requires high duty factor RH operation, $\propto R^{1/2}$
– External cryostat	0	
– Primary heat transport	6	U_{PHT} = \$72.3/$W^{0.7}$
– Thermal shield	0	
3. Tokamak Gas & Coolant Systems	88	
– Vacuum	19	ITER-FEAT: 37; FIRE: 14; CTF only: $\propto R^{1/4}$
– Tritium (and fuel) handling	41	ITER-FEAT: 104; FIRE: 9; CTF only: $\propto P_F^{1/2}$
– Aux heat transport	8	U_{AHT} = \$33.9/$W^{0.7}$
– Cryogenic plant	0	
– Heat rejection	8	
– Chemical control	12	

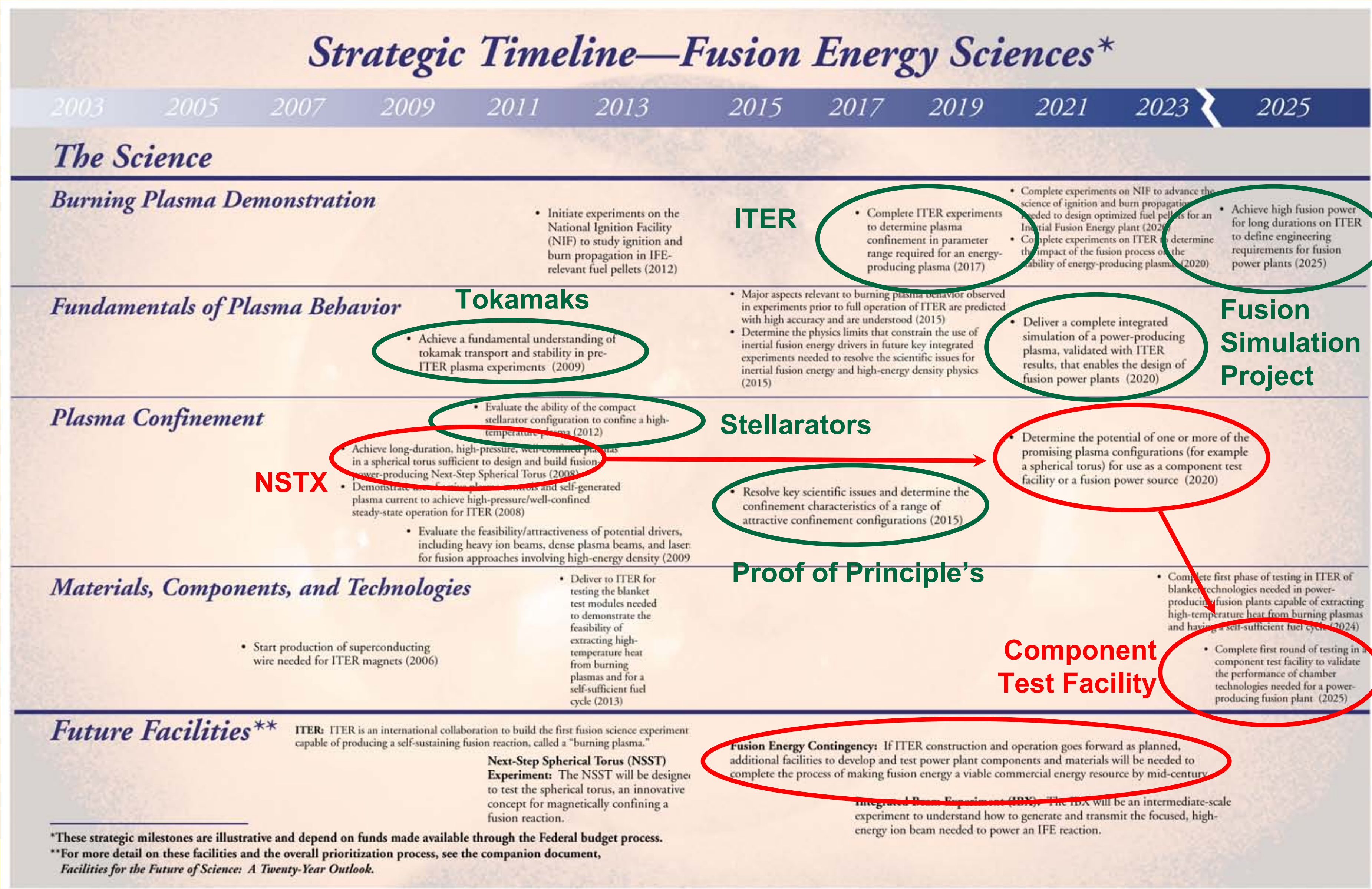
* ITER-FEAT-FIRE Cost Comparison, Fusion Study 2002, Snowmass; ** Comments by M. Abdou, B. Nelson

Comparative costing of CTF (W_L=1 MW/m^2) – II (in 2002 M$)

SuperCode Costing Components	R_0=1.2m	Comments
4. <u>Power Supplies & Control</u>	<u>120</u>	
– Magnet power supplies	63	
• *Resistive TFC*	*(52)*	U_{TFC} = $0.4M/MW (4X conventional power supply)
• *Resistive PFC*	*(11)*	U_{PFC} = $0.13M/MVA
– Heating system power supplies	0	Included in heating systems costs
– Site electric plant, transformers, etc.	21	ITER-FEAT: 38; FIRE: 18
– Device operational I&C	36	ITER-FEAT: 72; FIRE: 23
5. <u>Heating, Current Drive, Diagnostics</u>	<u>210</u>	
– ECW-EBW	40	CTF (8 MW @ 100 GHz), ITER-FEAT (12 MW @ 200 GHz: $111M)*
– NBI	125	CTF (34 MW at ~ 100-300 kV), ITER-FEAT (40 MW at 1 MV: $138M)
– LH	0	
– Plasma operational I&C	45	ITER-FEAT: $214M; FIRE: $29M
6. <u>Site, Facilities and Equipment</u>	<u>252</u>	
– Land, site improvement	0	Government site
– Buildings	180	ITER-FEAT: $546M; FIRE: $126M
– Hot cell	0	Included in Buildings
– Radwaste management	38	ITER-FEAT: $12M; FIRE: $11M (FNT testing at high duty factors on CTF substantially increases radwaste)
– Coolant supply and disposal	18	ITER-FEAT: $30M; FIRE: $18M
– General test and qualification	16	(CTF requires acceptance verification of all incoming test components.)
– Magnet fabrication tools	0	
Total Construction Cost	<u>1,050</u>	
with 40% Contingency	<u>1,470</u>	

* Comments by D. Rasmussen, R. Temkin

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



CTF would carry out integrated science-technology testing at Demo conditions in small size to ensure Demo success

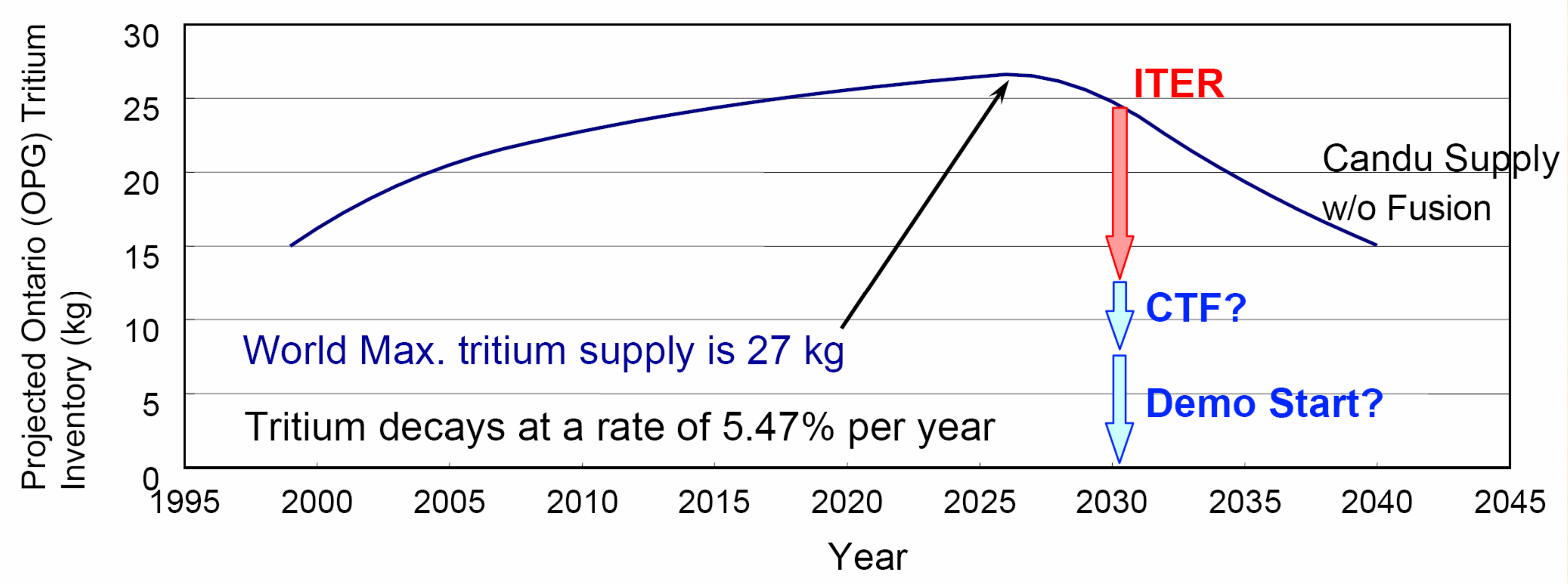
Abdou et al, Fusion Technology 1999

	ITER	CTF	Demo
Tritium self-sufficiency goal (%)	~0	80-100	>100
Burning plasma duration (s)	$\sim 10^3$	$>10^{6\text{-}7}$	$\sim 10^{7\text{-}8}$
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
P/R (MW/m)	~30	30-100	~100

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

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

CTF efficiently uses world's limited tritium supply before Demo becomes self-sufficient

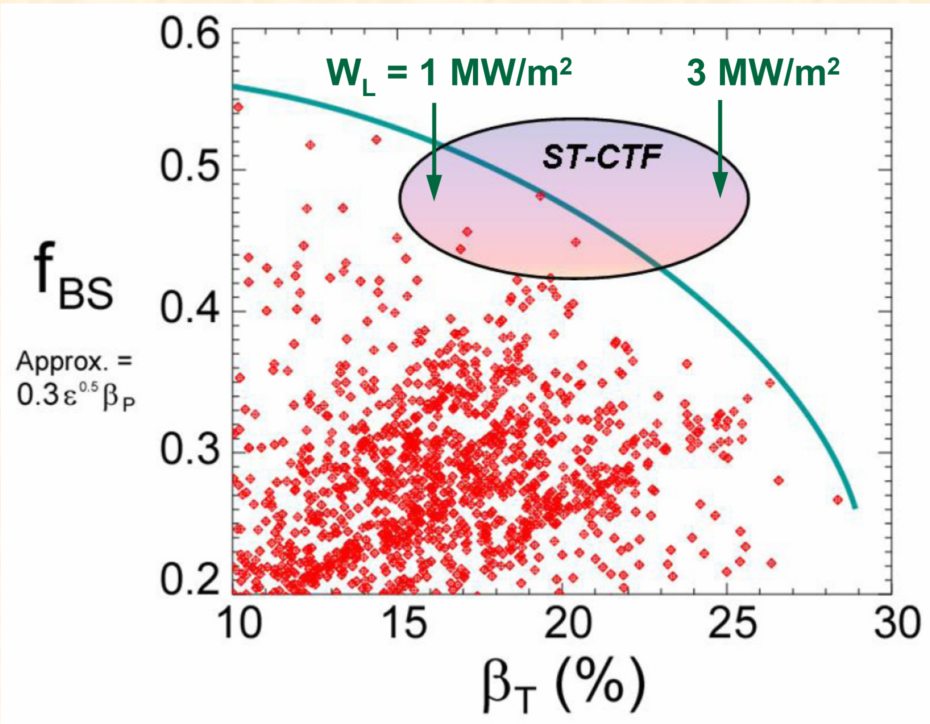


- ITER uses ~11 kg T to provide 0.3 MW-yr/m^2
- Assuming 80% tritium recovery,
  - One CTF needs 5 kg to enable testing to 6 MW-yr/m^2
  - Demo needs 3 kg/month to produce 2500 MW fusion power
- Avoid relying on more tritium from fission to sustain testing in Demo

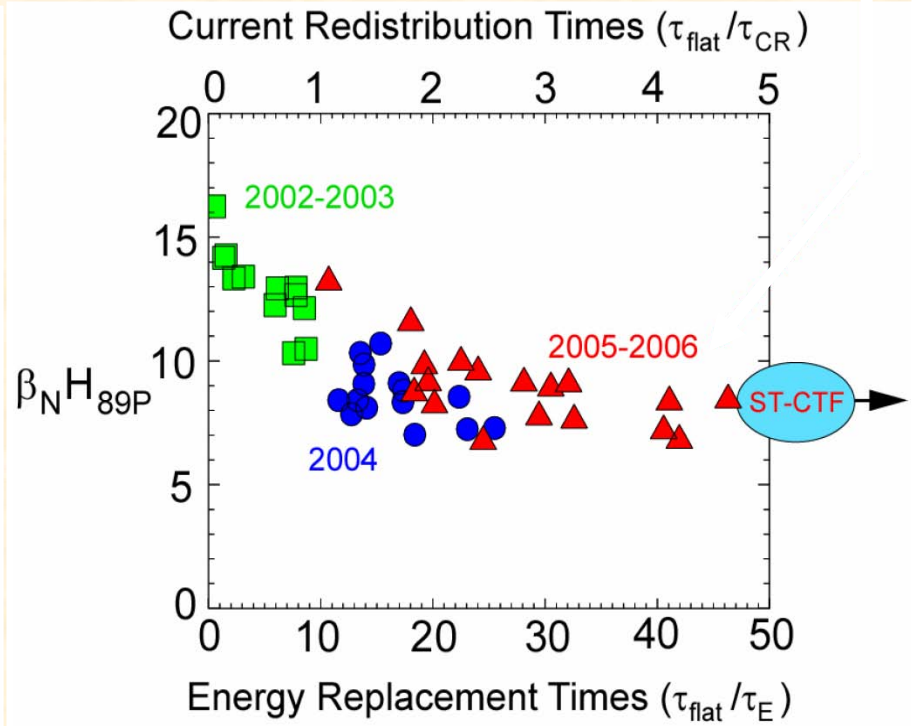
Progress: NSTX obtained physics database for the normalized conditions of basic CTF performance

4. Common progress & remaining physics R&D

Normalized sustainment conditions were obtained for projected baseline W_L



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

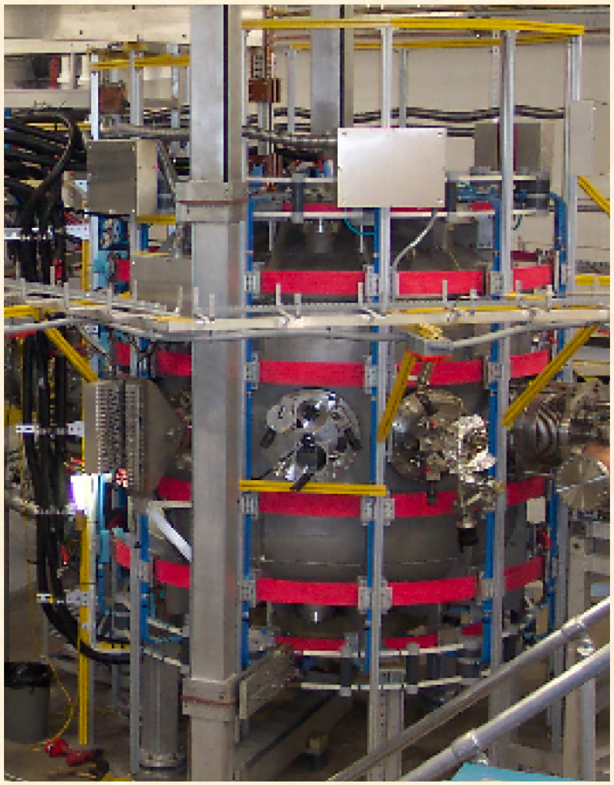


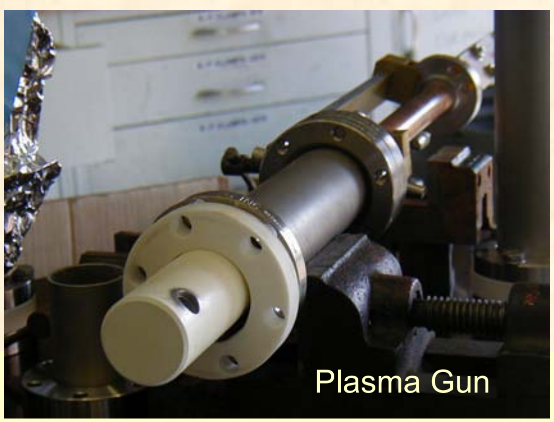
Need #1: more solenoid-free start-up data to enable projections to full plasma current of ~ 10 MA

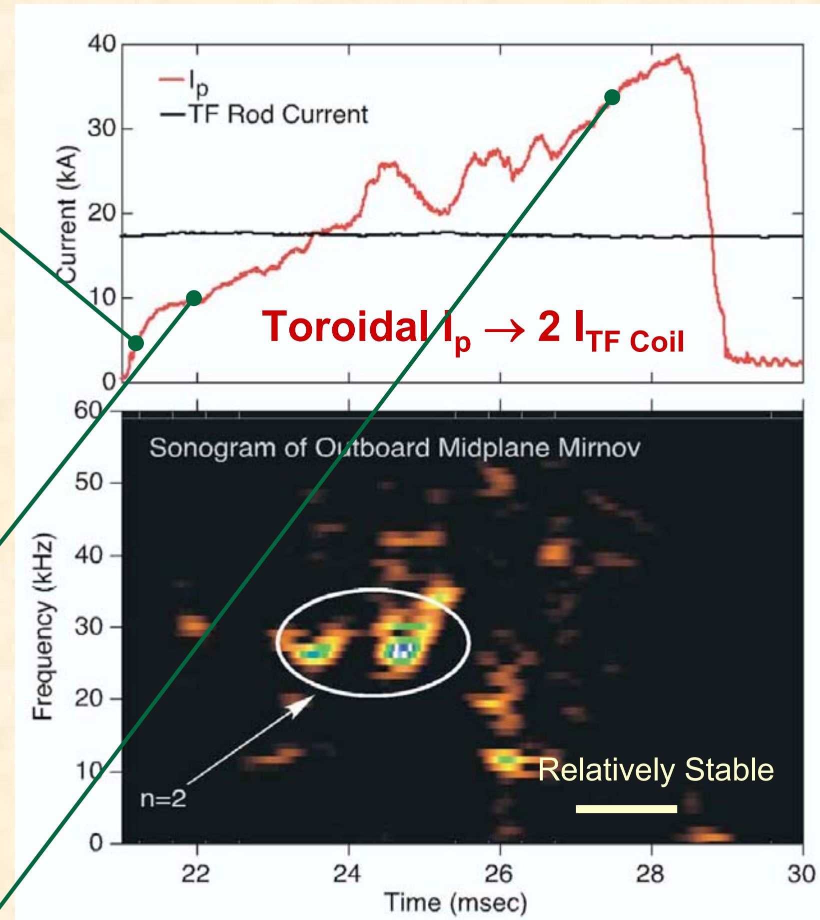
Sustained Parameters	CTF (1-2 MW/m^2) ($\tau >> \tau_{skin}$)	NSTX long pulse ($\kappa \leq 2.5$, $\tau \sim \tau_{skin}$)
Start-up $\mu_0 \ell_i R I_p$ (Wb)	2.3 – 3.6	~ 0.13 (goal)
I_p/aB_T (MA/m-T)	4.0 – 6.0	3.8
Safety factor, q_{cyl}	4.6 – 3.1	3.3
β_N (%-m-T/MA)	3.0 – 4.3	5.1
β_T (%)	15 – 25	19
a/ρ_i (= $1/\rho_i^*$)	~ 50	~ 30
H_{98pby2}	1.3	≤ 1.3

- 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: basis for baseline CTF stability and confinement established

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

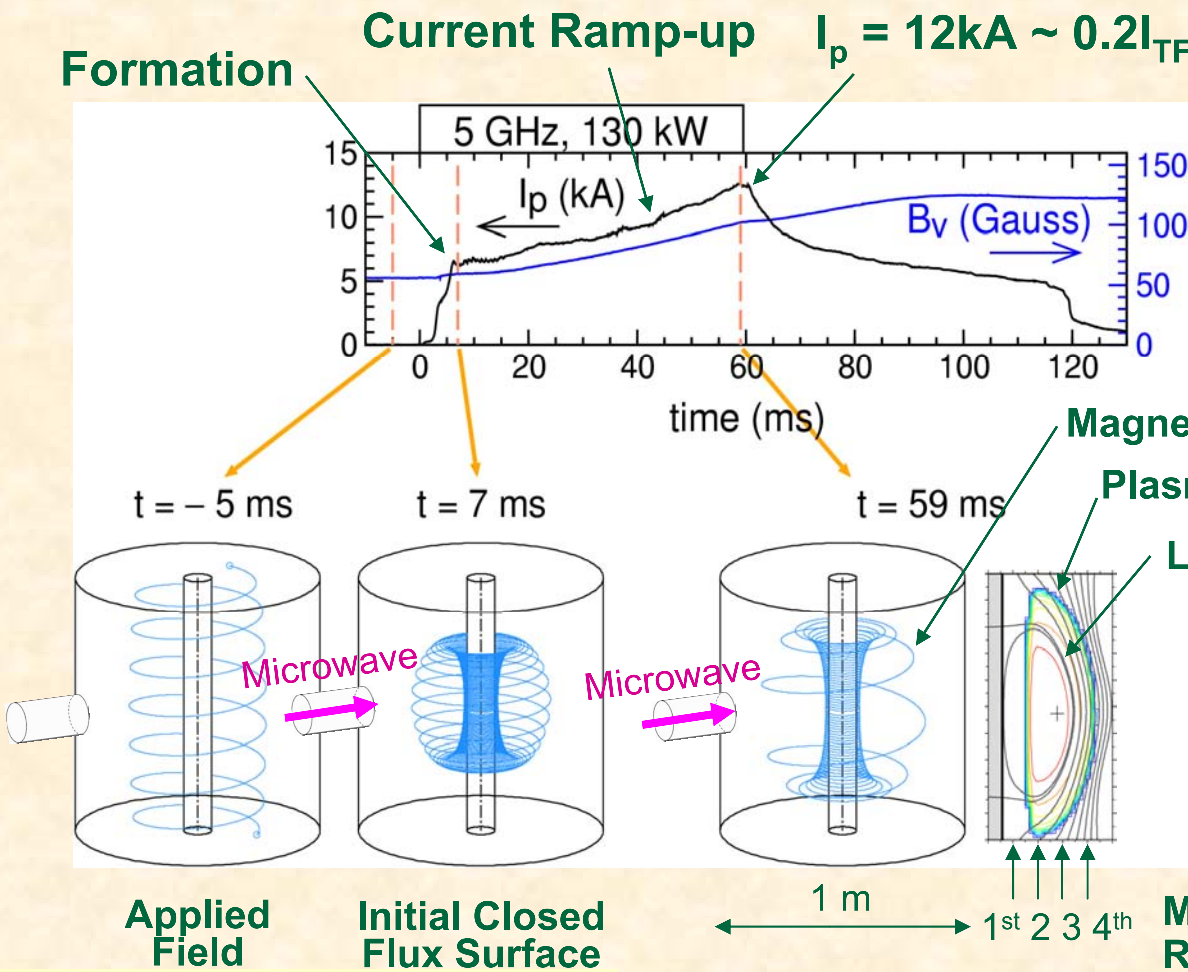






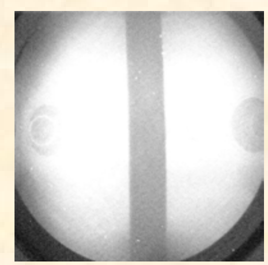
Scale up gun-driven current to ~1 MA (20 times), or ~10 MA (200 times)?

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



Electron Bernstein wave heating and current drive:

Line averaged density > plasma cutoff density



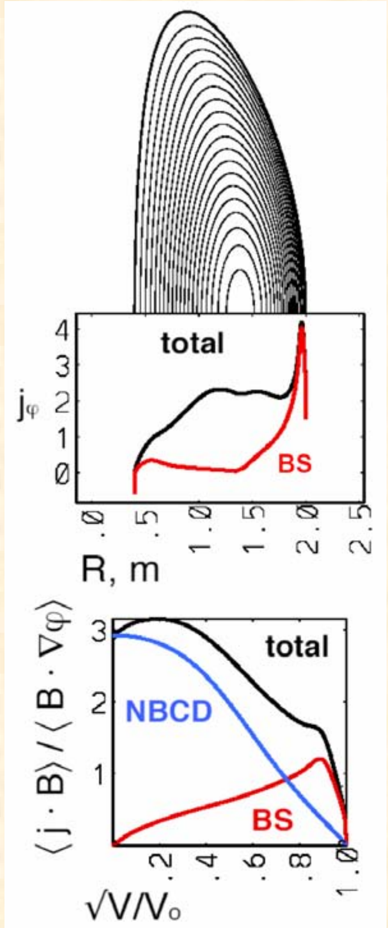
CCD Plasma Image

Need #2: steady state high-heat-flux divertor physics data to enable projections to CTF operations

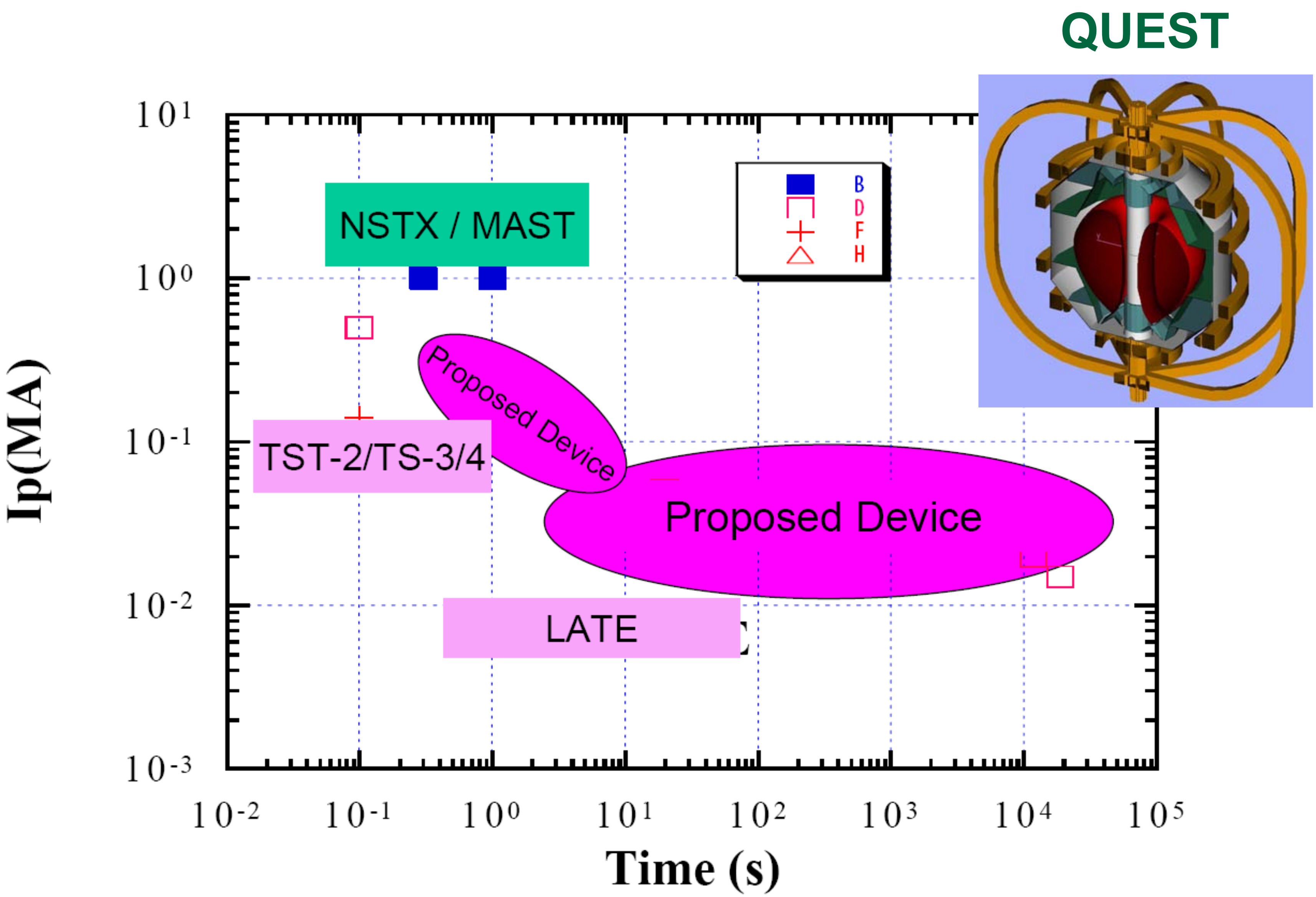
Sustained Parameters	CTF (1-2 MW/m^2) ($\tau >> \tau_{skin}$)	NSTX long pulse ($\kappa \leq 2.5$, $\tau \sim \tau_{skin}$)
P/R (MW/m)	45 – 75	≤ 9
SOL expansion factor	10 – 20	~ 5
$V_\alpha/V_{Alfvén}$	3 – 6	1 – 4 ($V_{NB}/V_{Alfvén}$)
I_{BS}/I_{CD} fractions	0.45/0.55	0.3-0.5/0.3-0.1

- High priority for ITER (P/R ~ 30 MW/m), large tokamaks, S/C tokamaks (KSTAR: P/R ~ 16 MW/m), and new ST in Japan
- NSTX to investigate physics solution of liquid lithium divertor targets
- Progress: basis for NBI & bootstrap current drive physics established

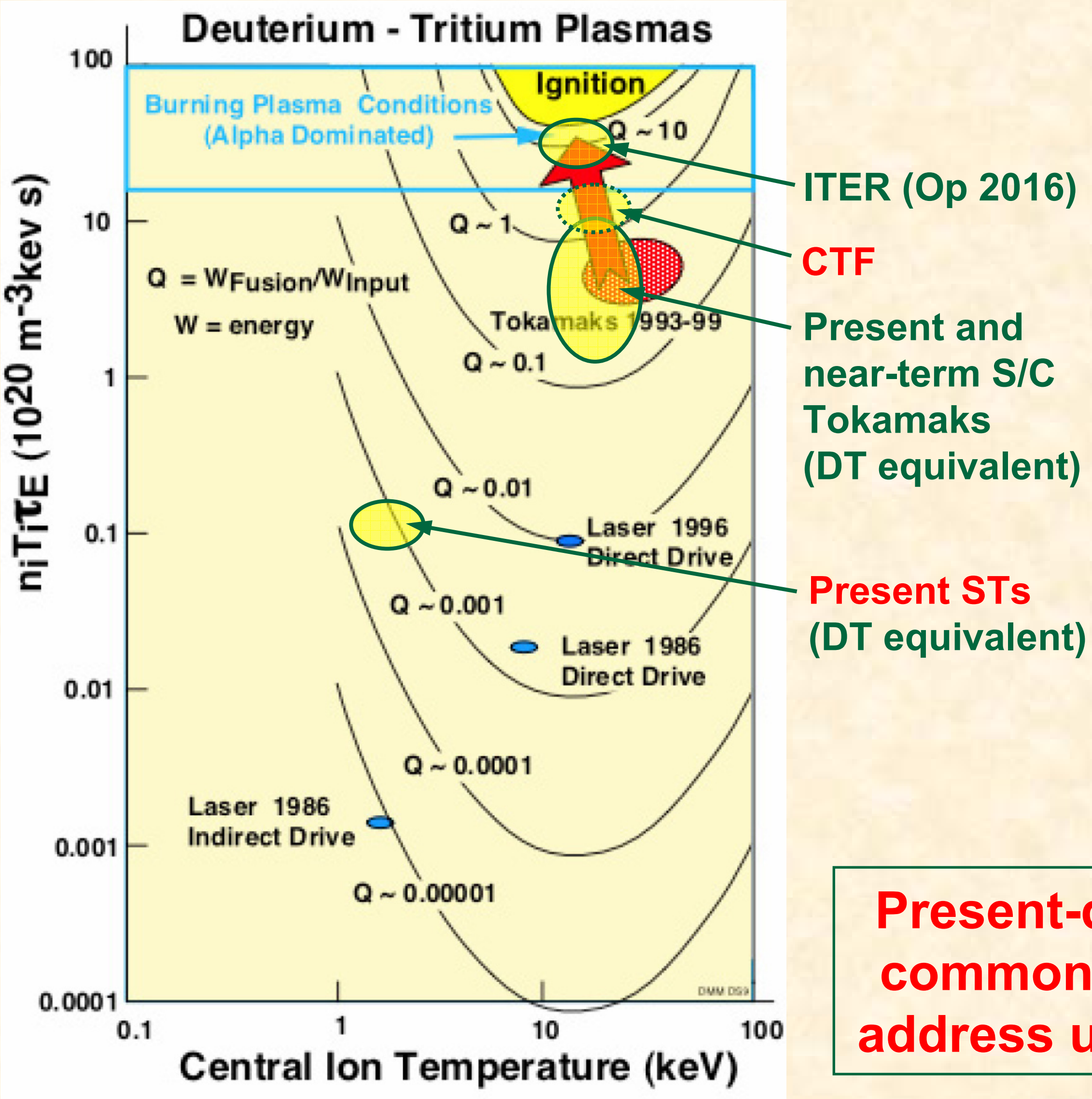
ITF plasma shape & stable current profile using NBI & BootStrap



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



CTF requiring only Q ~ 1-3 can take advantage of the full physics database being developed 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 address unique CTF physics needs.

CTF is chosen to permit high W_L while limiting new R&D and maximizing plasma reliability

- Takes advantage of s.s. divertor development / test for ITER and KSTAR
- Needs additional data on solenoid-free start-up to project to full current
- Very high β_N and I_{BS}/I_p operation projected for ST Demo

	Data+(ITER)		Increasing W_L →			ST-Demo
Fus. neut. flux, W_L, MW/m^2	0.01 (0.78)	**0.5**	**1.0**	**2.0**	**3.0**	3 – 4
P/R, MW/m	10 (30)	30	45	75	100	~200
Solenoid-free flux, $\mu_0 \ell_i R I_p$, Wb; I_p/I_{TFC}	1 (JT-60U); 2 (Pegasus)	3.0; 0.82	3.8; 0.82	4.9; 0.82	5.1; 0.80	23; 1
I_p, MA	6 (15)	6.7	8.4	10.9	11.4	20
β_N, %Tm/MA; $\beta_N/\beta_{N\text{-limit}}$, %	5.5; 65	4.7; 57	4.6; 56	4.5; 55	5.0; 62	8.5; 90
Safety factor, q_{cyl}	>2	2.7	2.7	2.7	2.8	2
n/n_{GW}	0.2 – 0.9	0.27	0.27	0.28	0.32	0.4
H_{98H}	<1.5	1.3	1.3	1.3	1.3	1
$\langle T_i \rangle / \langle T_e \rangle$	3 (~1)	1.6	1.8	1.7	1.6	1
I_{NB+RF}/I_p; I_{BS}/I_p	0.3; 0.7	0.51; 0.49	0.52; 0.48	0.52; 0.48	0.46; 0.54	0.1; 0.9
Neutral beam energy, kV	360 (1000)	135	171	229	280	1000
P_{fusion}, MW	15 (500)	37	74	148	222	1600
P_{NB+RF}, MW	40 (100)	37	47	57	70	100
I_{TFC}, MA	(100)	8.2	10.2	13.2	14.3	20

ST research is making rapid progress toward CTF, a fusion energy science & technology facility to enable fast implementation of Demo

- EU, Japan, Korea plan fast Demo implementation as early as 2020's
  - Opportunities for ST to support this strategy in science & technology
- Encourage that the growing ST research community
  - Support and benefit from USBPO-ITPA activities on common physics
  - Complement and extend Tokamak (and ST) experiments
  - Enable attractive CTF to carry integrated testing at Demo conditions in small size to ensure fast Demo success
- Attractive CTF option identified
  - Included in the USDOE fusion plan
- Progress & physics R&D needs
  - Physics basis already broadly established and further leverages common Tokamak advances toward ITER
  - Need additional data in 2 areas to enable projections to CTF conditions
    - Solenoid-free current start-up
    - Steady state high heat flux divertor