

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

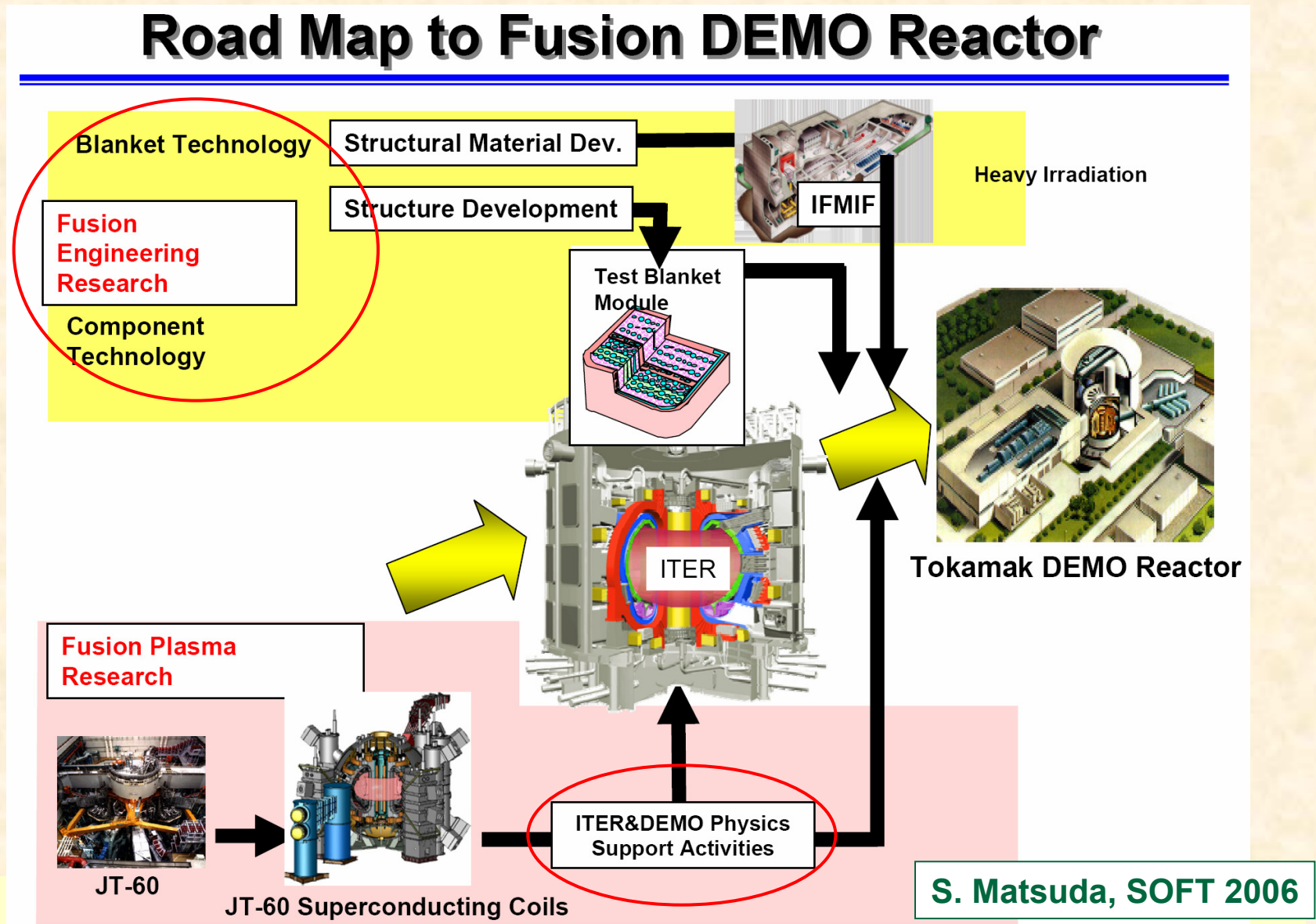
Martin Peng, NSTX Program Director

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

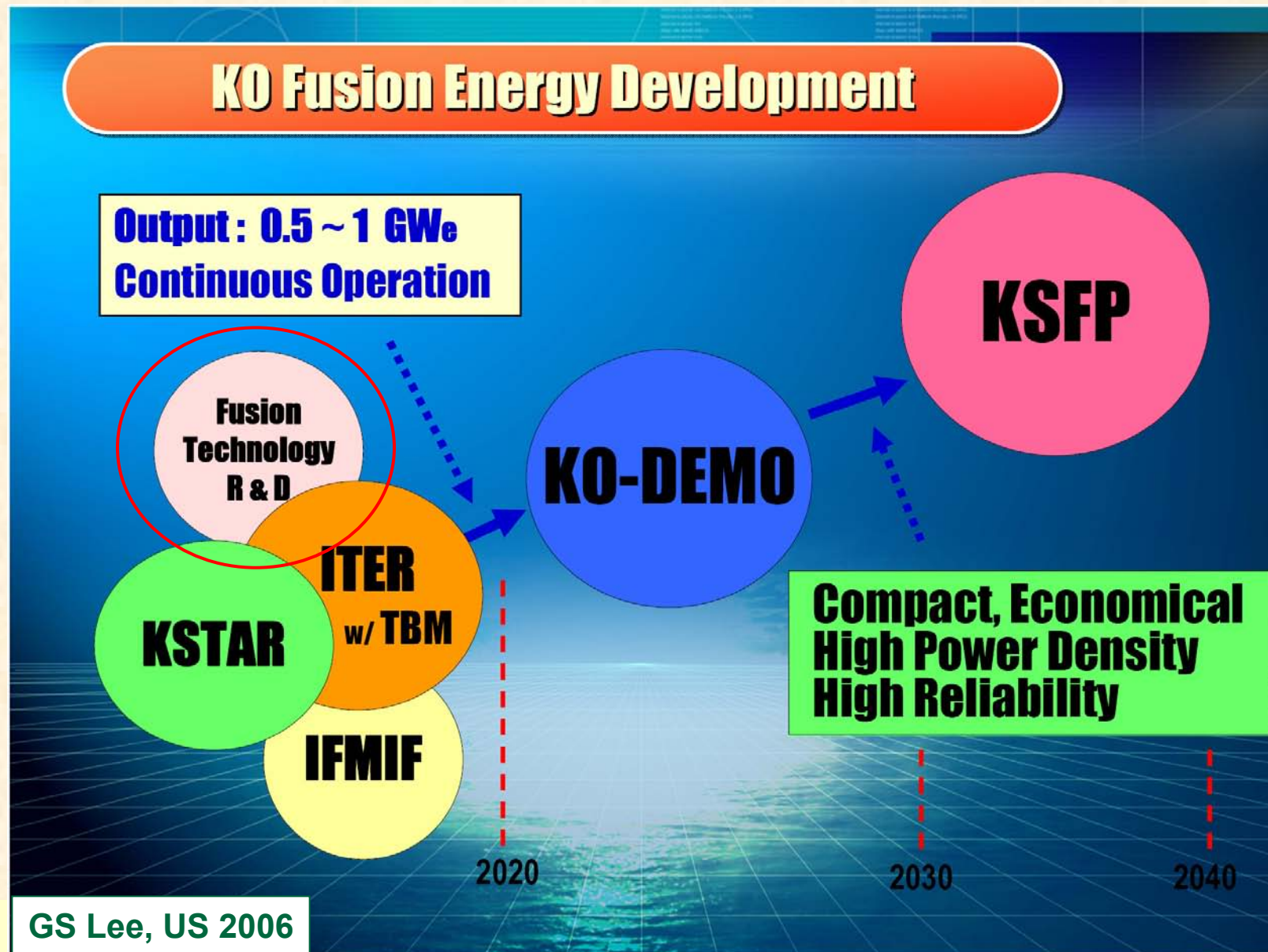
**OAK RIDGE NATIONAL LABORATORY
U. S. DEPARTMENT OF ENERGY**

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

1. Opportunities in the "Demo after ITER" strategy



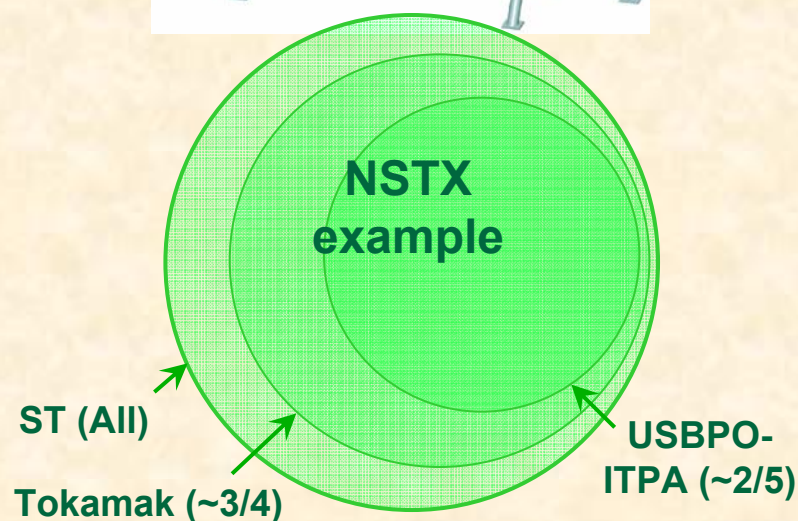
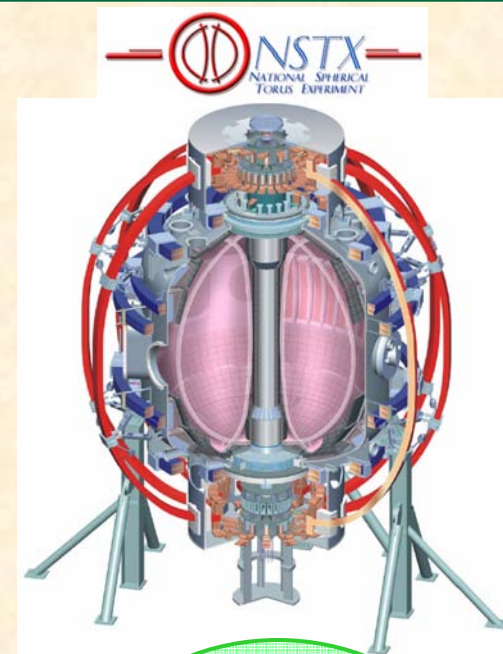
Korean fusion energy development plan introduces opportunities to accelerate fusion technology R&D



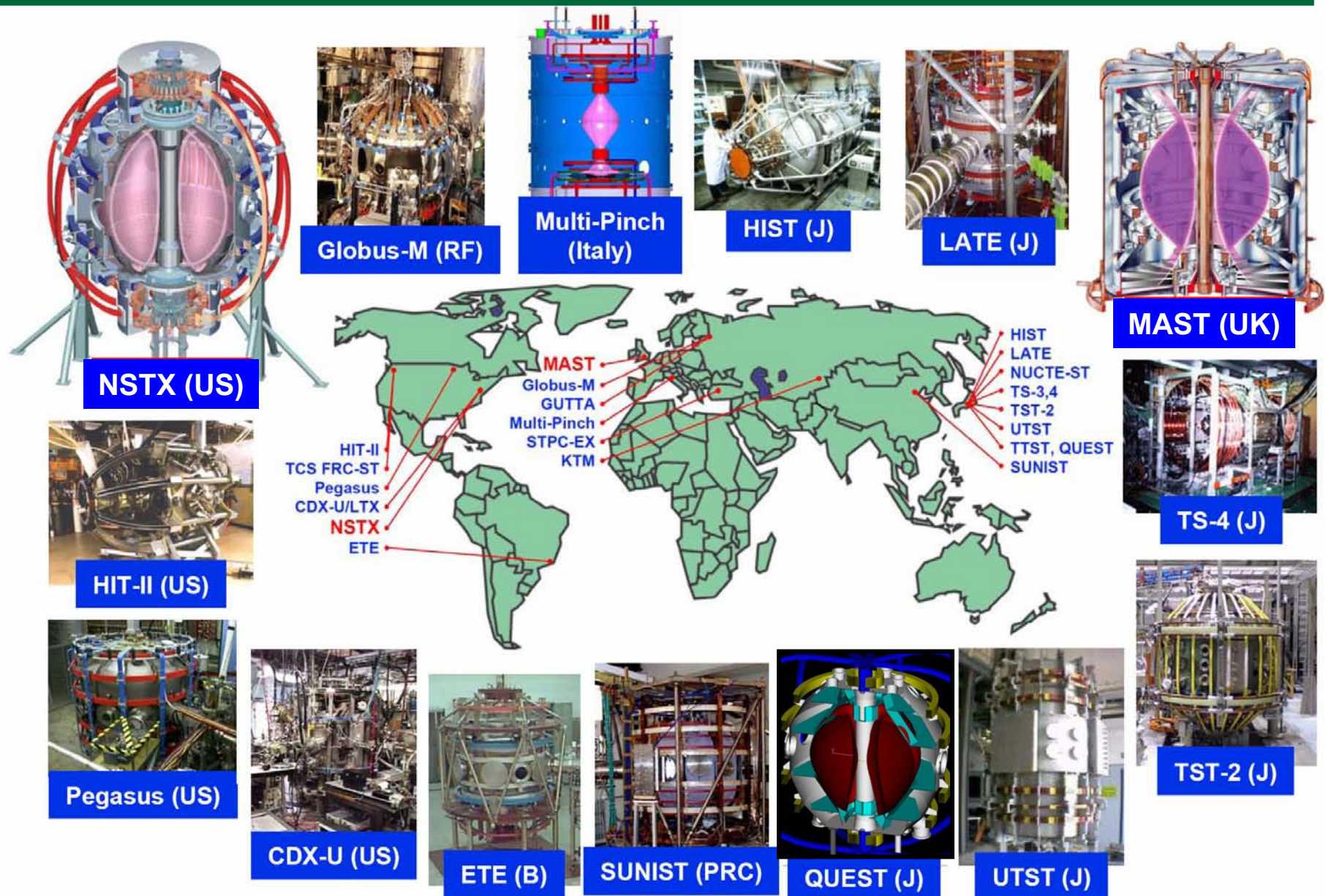
We encourage that the ST research community addresses issues in support of this strategy

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.



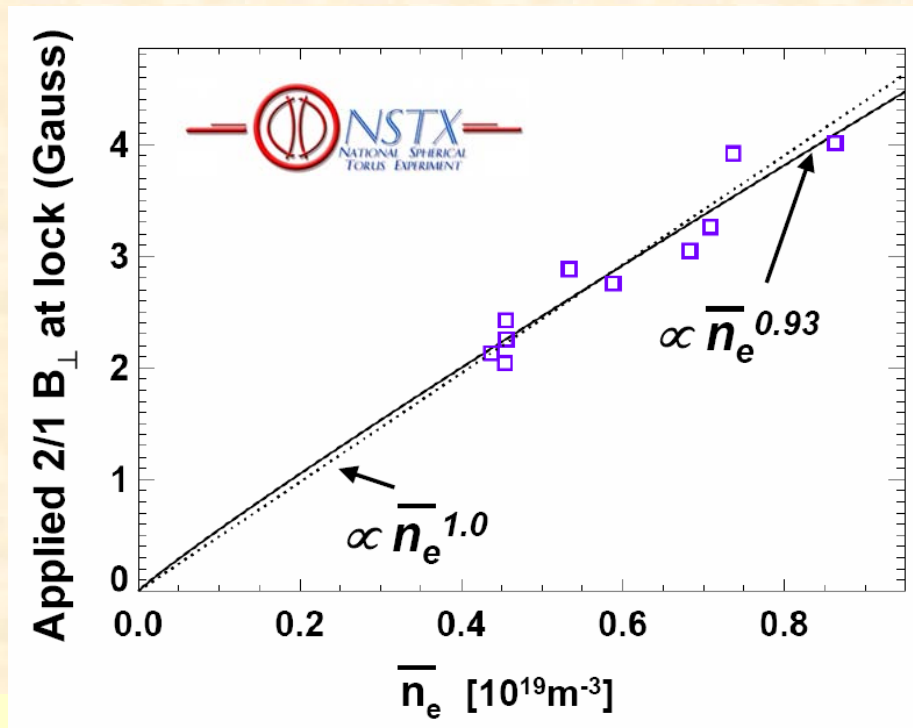
World Spherical Tokamak research has expanded to 22 experiments addressing all key physics issues



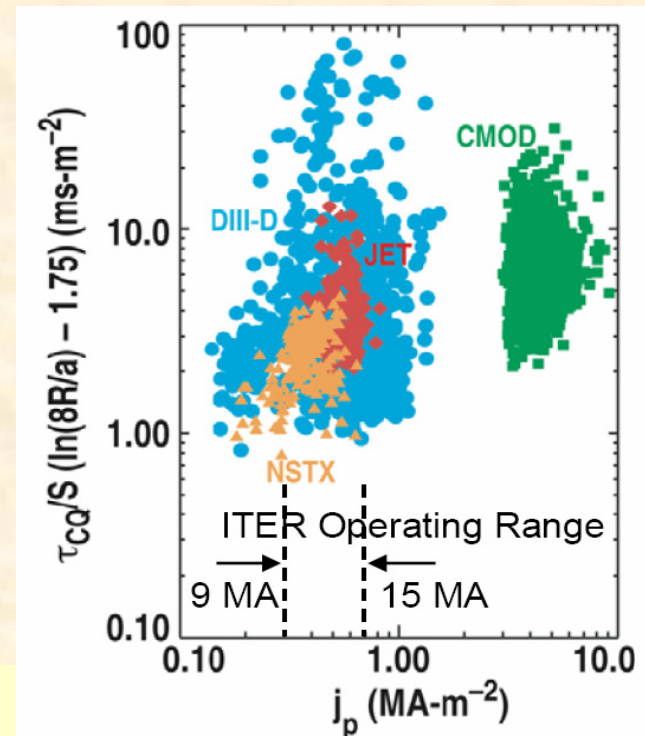
ST supports and benefits from USBPO-ITPA in preparing for burning plasma research on ITER

- 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



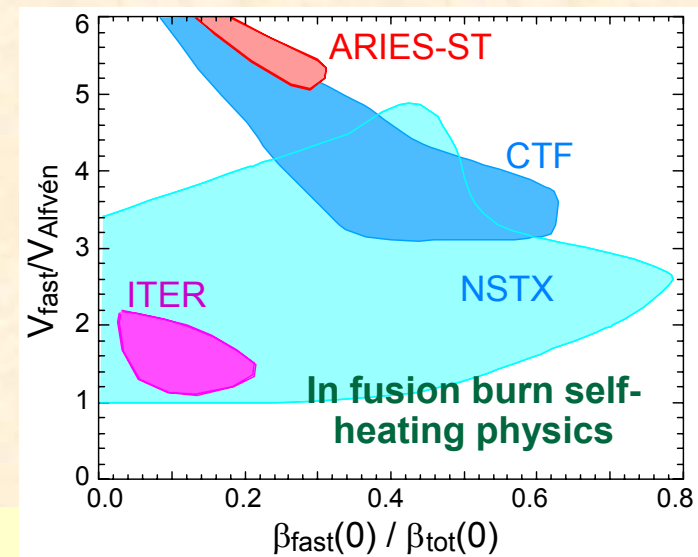
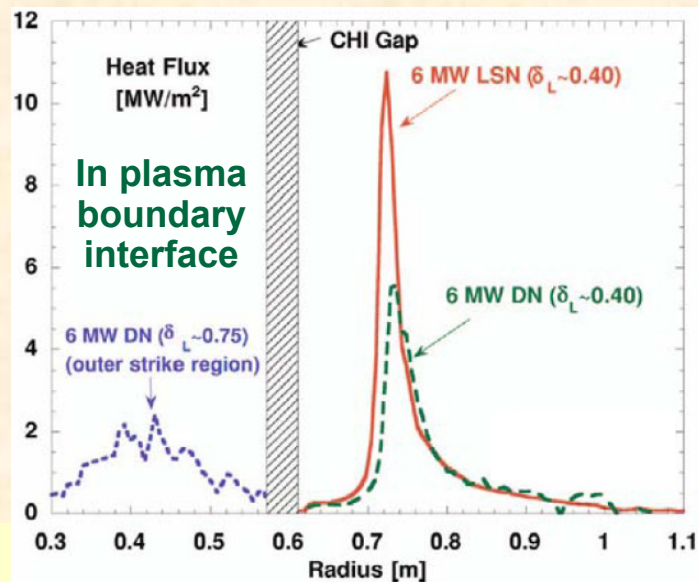
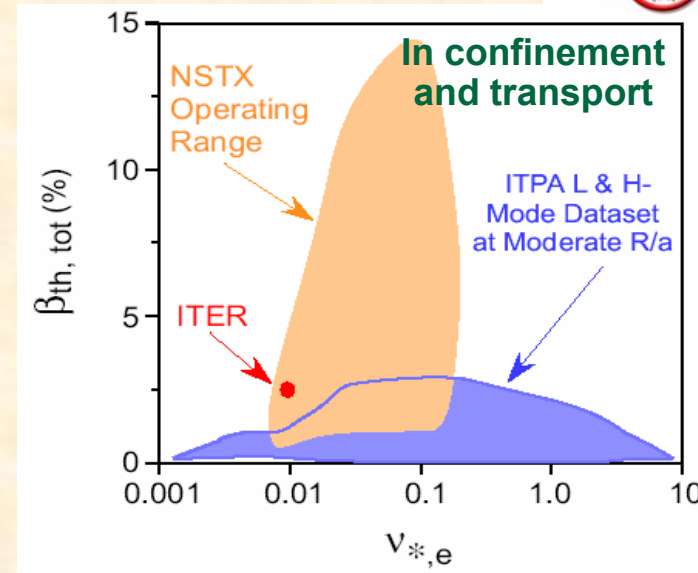
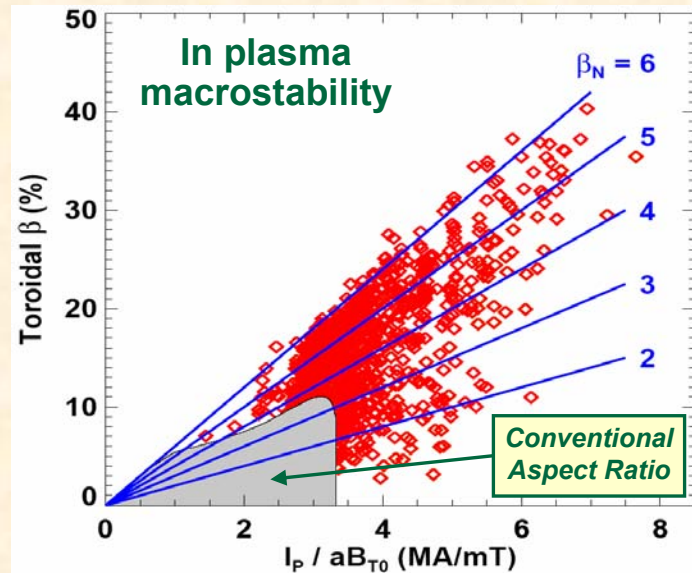
Normalizing disruption quench times (τ_{CQ}) to plasma inductance removes apparent j_p dependence



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	ρ^* 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

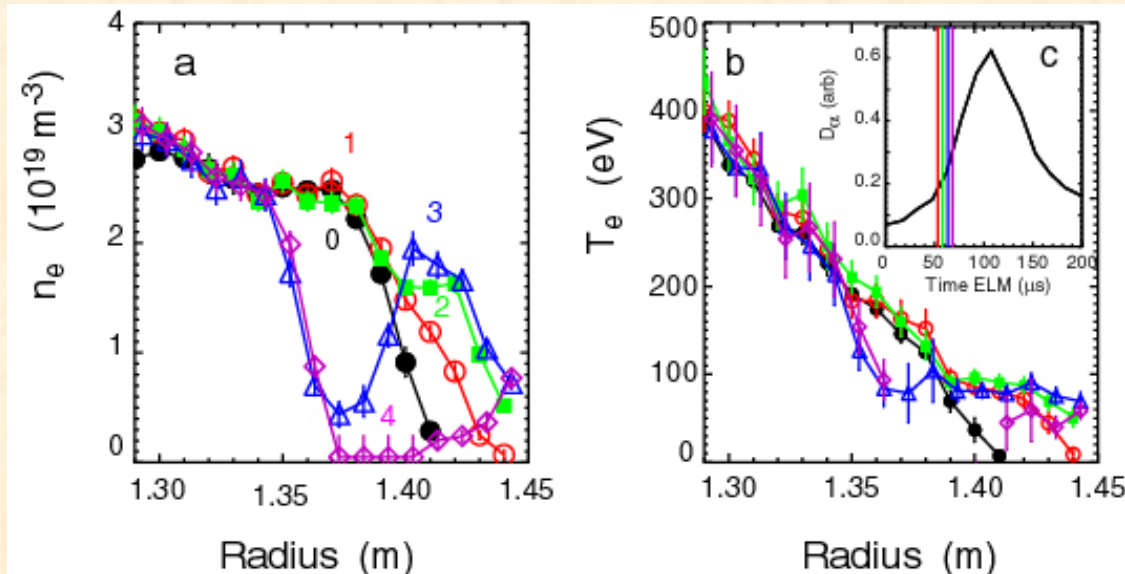
ST complements and extends Tokamak physics in investigating key fusion scientific issues for both



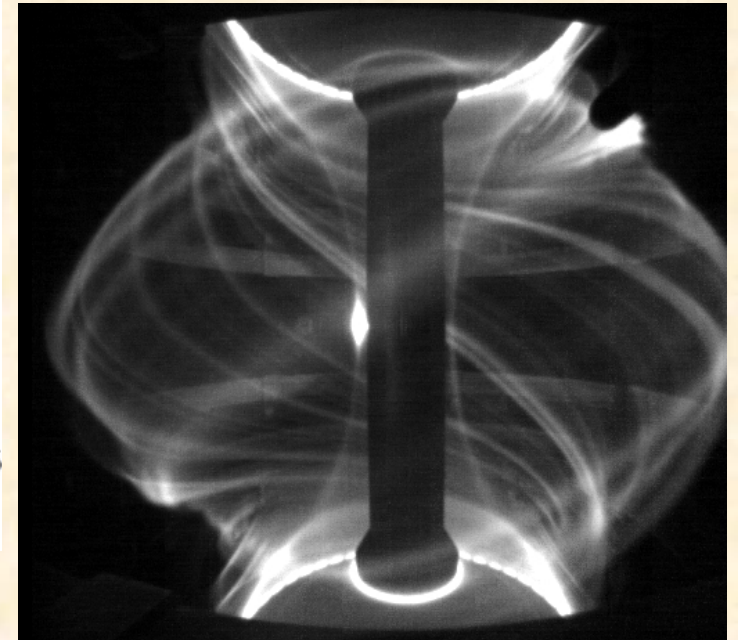
Unique ST features enable new insight into **common** critical ST & Tokamak physics issues



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

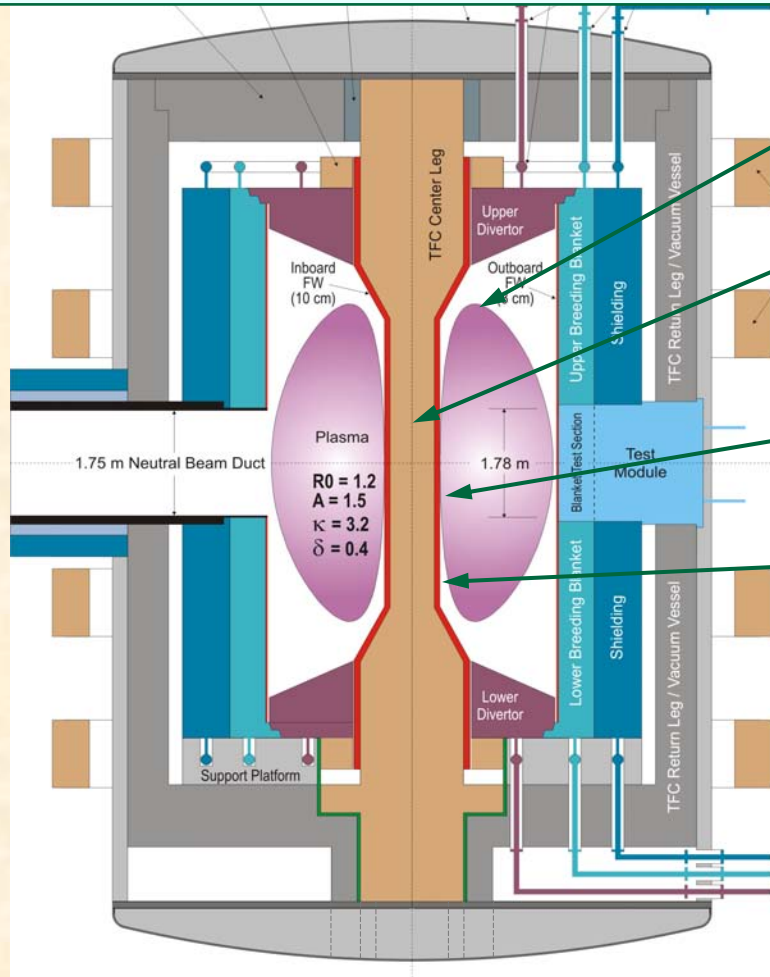
- 4 lasers fired with 5 μs separation
- 1-cm spatial resolution



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

3. An attractive CTF option

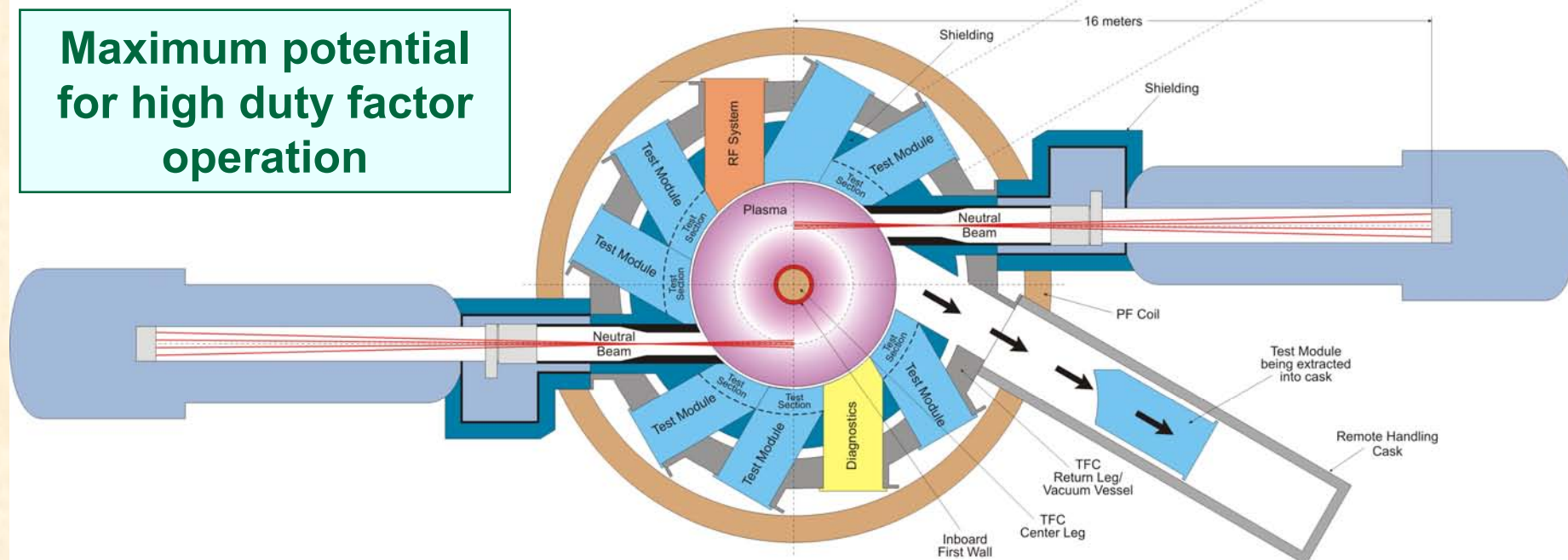
$R = 1.2 \text{ m}$, $a = 0.8 \text{ m}$, $I_p = 8 - 12 \text{ MA}$,
 $W_L = 1.0 - 3.0 \text{ MW/m}^2$



- ◆ 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

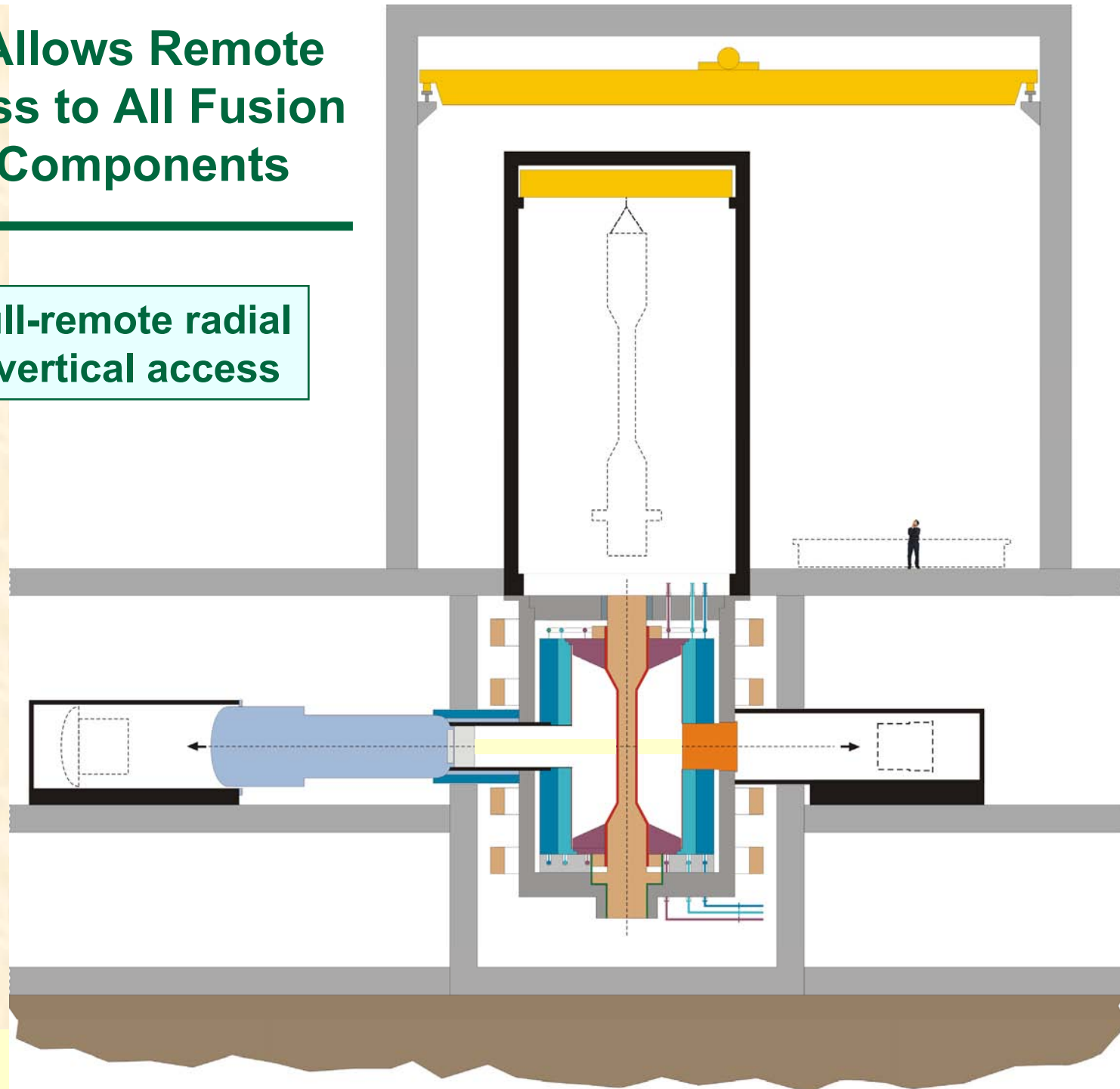
Mid-Plane Test Modules, Neutral Beam Injection, RF, Diagnostics Are Arranged for Direct Replacement



- 8 mid-plane blanket test modules provides $\sim 15 \text{ m}^2$ at maximum flux
 - Additional cylindrical blanket test area $> 50 \text{ m}^2$ 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 ($\sim$ ITER)

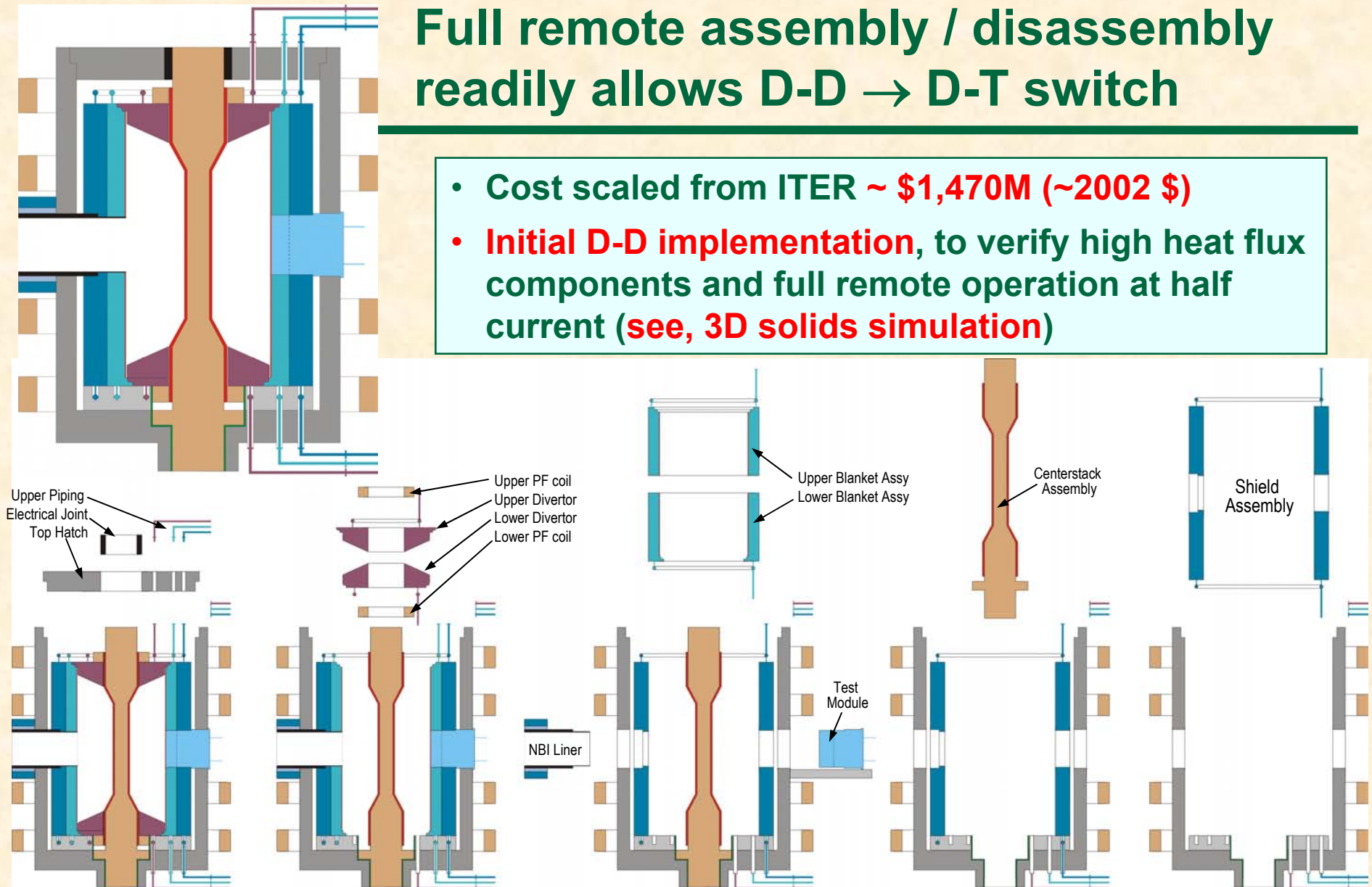
CTF Allows Remote Access to All Fusion Core Components

- 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/m²) – I (in 2002 M\$)

SuperCode Costing Components	$R_0=1.2m$	Comments
1. <u>Toroidal Device</u>	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-
– Device, penetration shielding	43	breeding blankets cost 1/3 of advanced test blankets**
– Divertor, PFCs	29	ITER-FEAT: 109; FIRE: 42; CTF: $U_{Div} = 1.61/m^2$
– Fueling	12	ITER-FEAT: 10; FIRE: 9
2. <u>Device Ancillary Systems</u>	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. <u>Tokamak Gas & Coolant Systems</u>	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/m²) – II (in 2002 M\$)

SuperCode Costing Components	$R_0=1.2m$	Comments
4. Power Supplies & Control	120	
– 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. Heating, Current Drive, Diagnostics	210	
– 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. Site, Facilities and Equipment	252	
– 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	1,050	
with 40% Contingency	1,470	

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

Strategic Timeline—Fusion Energy Sciences*

2003 2005 2007 2009 2011 2013 2015 2017 2019 2021 2023 2025

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)

ITER

- 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

Tokamaks

- Achieve a fundamental understanding of tokamak transport and stability in pre-ITER plasma experiments (2009)

- Major aspects relevant to burning plasma behavior observed in experiments prior to full operation of ITER are predicted with high accuracy and are understood (2015)
- Determine the physics limits that constrain the use of inertial fusion energy drivers in future key integrated experiments needed to resolve the scientific issues for inertial fusion energy and high-energy density physics (2015)

Fusion Simulation Project

- Deliver a complete integrated simulation of a power-producing plasma, validated with ITER results, that enables the design of fusion power plants (2020)

Plasma Confinement

- Evaluate the ability of the compact stellarator configuration to confine a high-temperature plasma (2012)

Stellarators

- Resolve key scientific issues and determine the confinement characteristics of a range of attractive confinement configurations (2015)

NSTX

- Achieve long-duration, high-pressure, well-confined plasmas in a spherical torus sufficient to design and build fusion power-producing Next-Step Spherical Torus (2008)
- Demonstrate the feasibility of plasma conditions and self-generated plasma current to achieve high-pressure/well-confined steady-state operation for ITER (2008)

- Evaluate the feasibility/attractiveness of potential drivers, including heavy ion beams, dense plasma beams, and laser for fusion approaches involving high-energy density (2009)

- Determine the potential of one or more of the promising plasma configurations (for example a spherical torus) for use as a component test facility or a fusion power source (2020)

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)

Proof of Principle's

Component Test Facility

- 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 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 designed 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 Blanket Experiment (IBEX): The IBEX 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*.

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

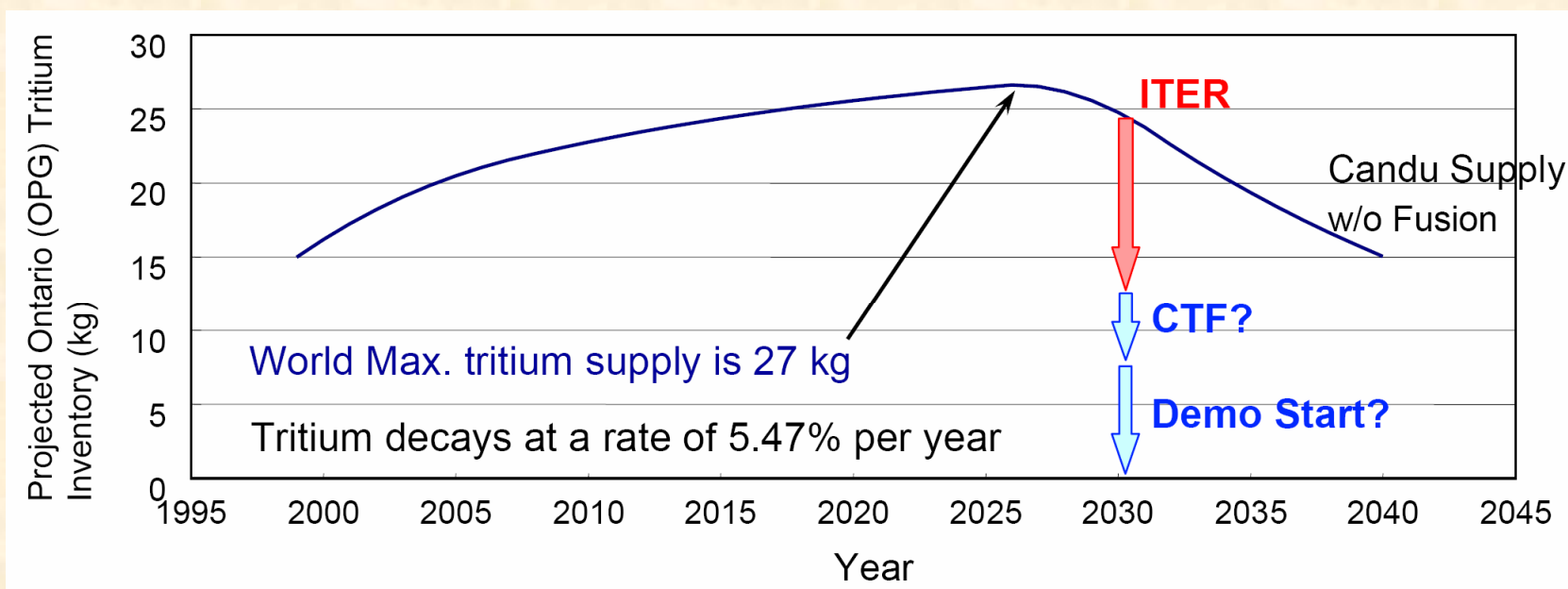
Abdou et al, Fusion Technology 1999

	ITER	CTF	Demo
Tritium self-sufficiency goal (%)	~0	80-100	>100
Burning plasma duration (s)	~10 ³	>10 ⁶⁻⁷	~10 ⁷⁻⁸
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



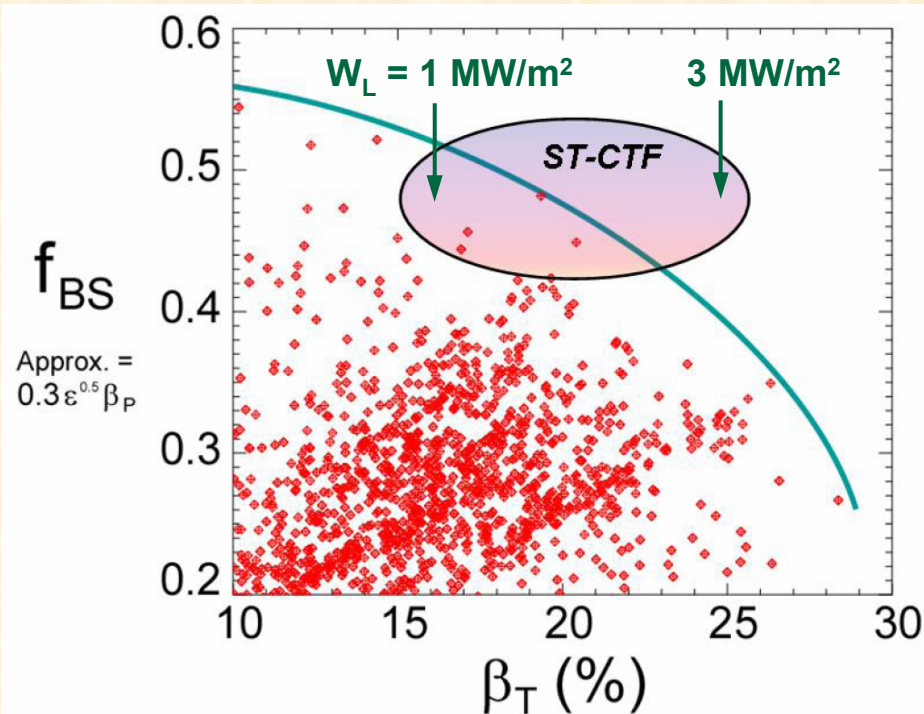
- ITER uses ~11 kg T to provide 0.3 MW-yr/m²
- Assuming 80% tritium recovery,
 - One CTF needs 5 kg to enable testing to 6 MW-yr/m²
 - Demo needs 3 kg/month to produce 2500 MW fusion power
- Avoid relying on more tritium from fission to sustain testing in Demo

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

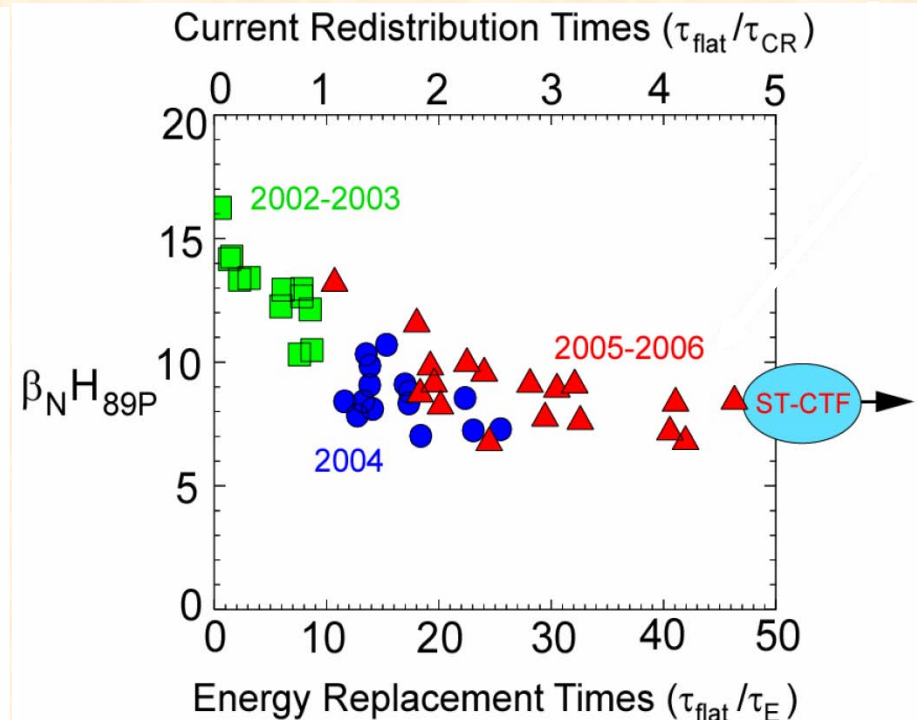


4. Common progress & remaining physics R&D

Normalized sustainment conditions were obtained for projected baseline W_L



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



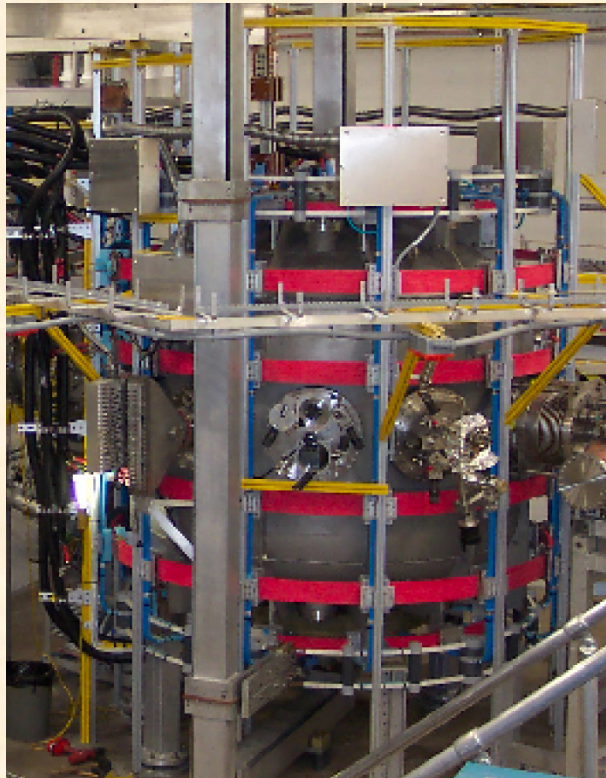
Need #1: more solenoid-free start-up data to enable projections to full plasma current of ~ 10 MA

Sustained Parameters	CTF (1-2 MW/m ²) ($\tau \gg \tau_{\text{skin}}$)	NSTX long pulse ($\kappa \leq 2.5$, $\tau \sim \tau_{\text{skin}}$)
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
$H_{98\text{pby}2}$	1.3	≤ 1.3

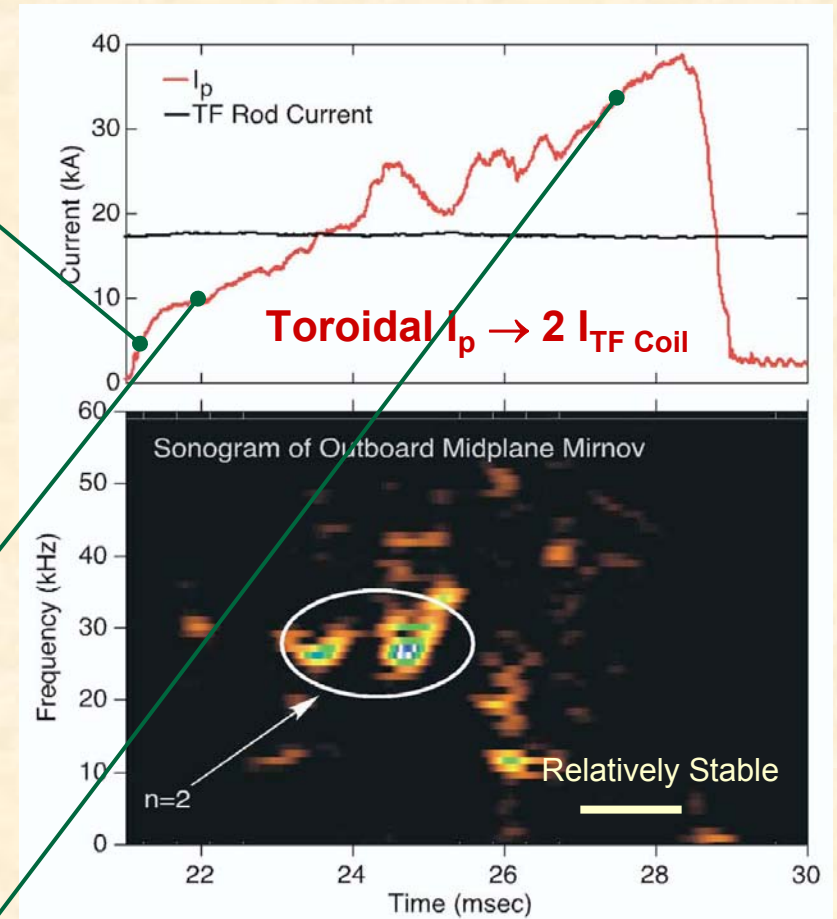
- Feasibility proven on NSTX (CHI), MAST (merging compression), Pegasus (plasma gun) & LATE, TST-2, JT-60U (RF+VF swing, then NBI up to 0.6MA)
- NSTX to investigate combining CHI, EBW, HHFW, NBI & VF swing
- Progress: basis for baseline CTF stability and confinement established

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

PEGASUS Toroidal Experiment
University of Wisconsin-Madison

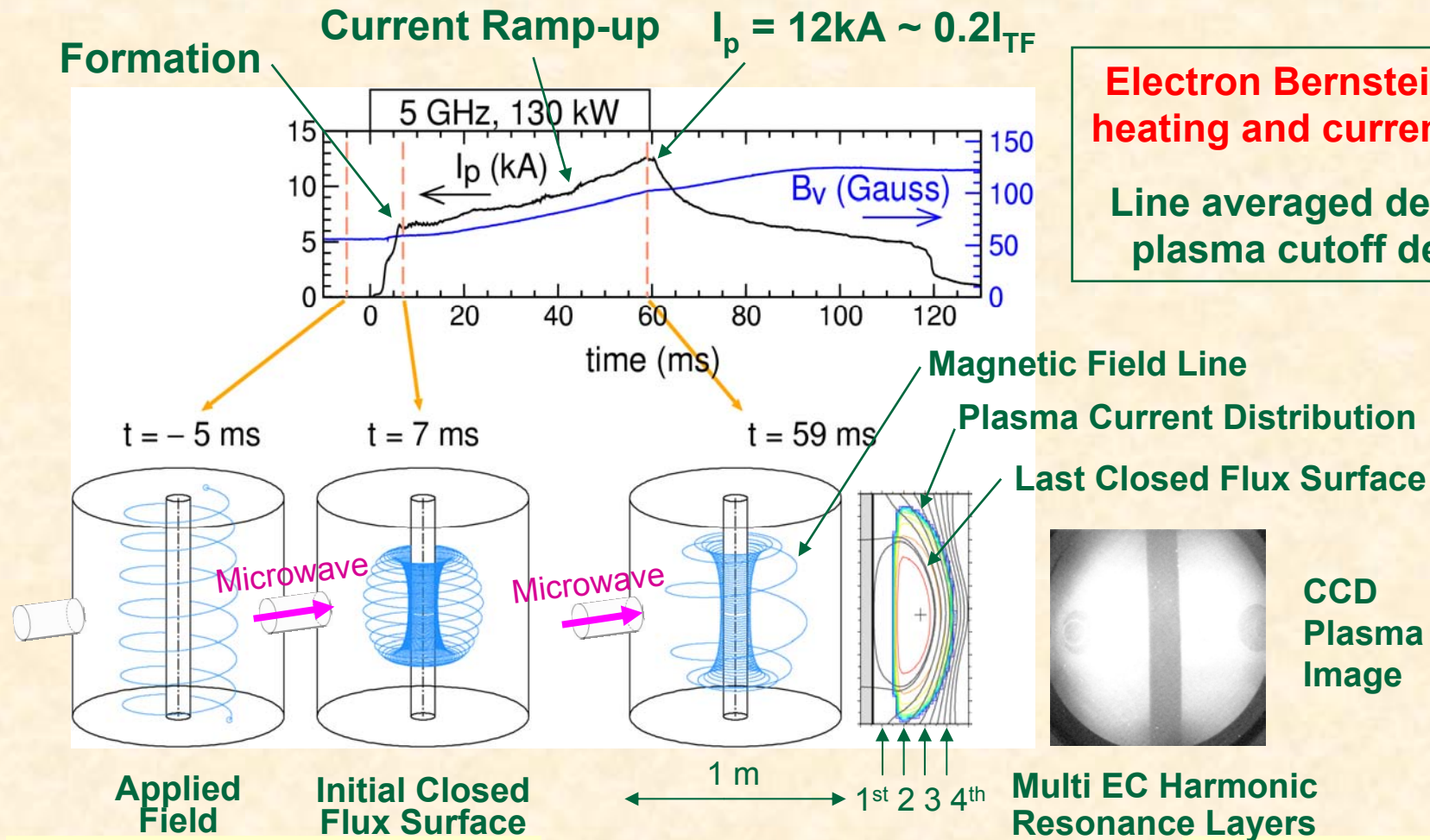


Plasma Gun



**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

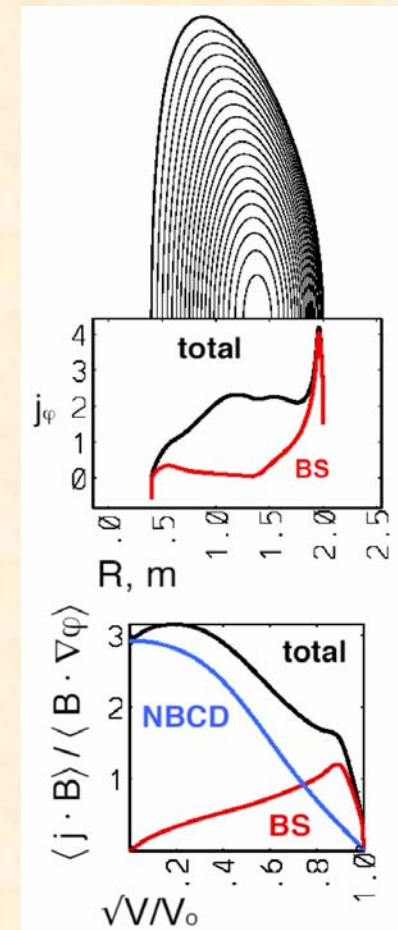


Need #2: steady state high-heat-flux divertor physics data to enable projections to CTF operations

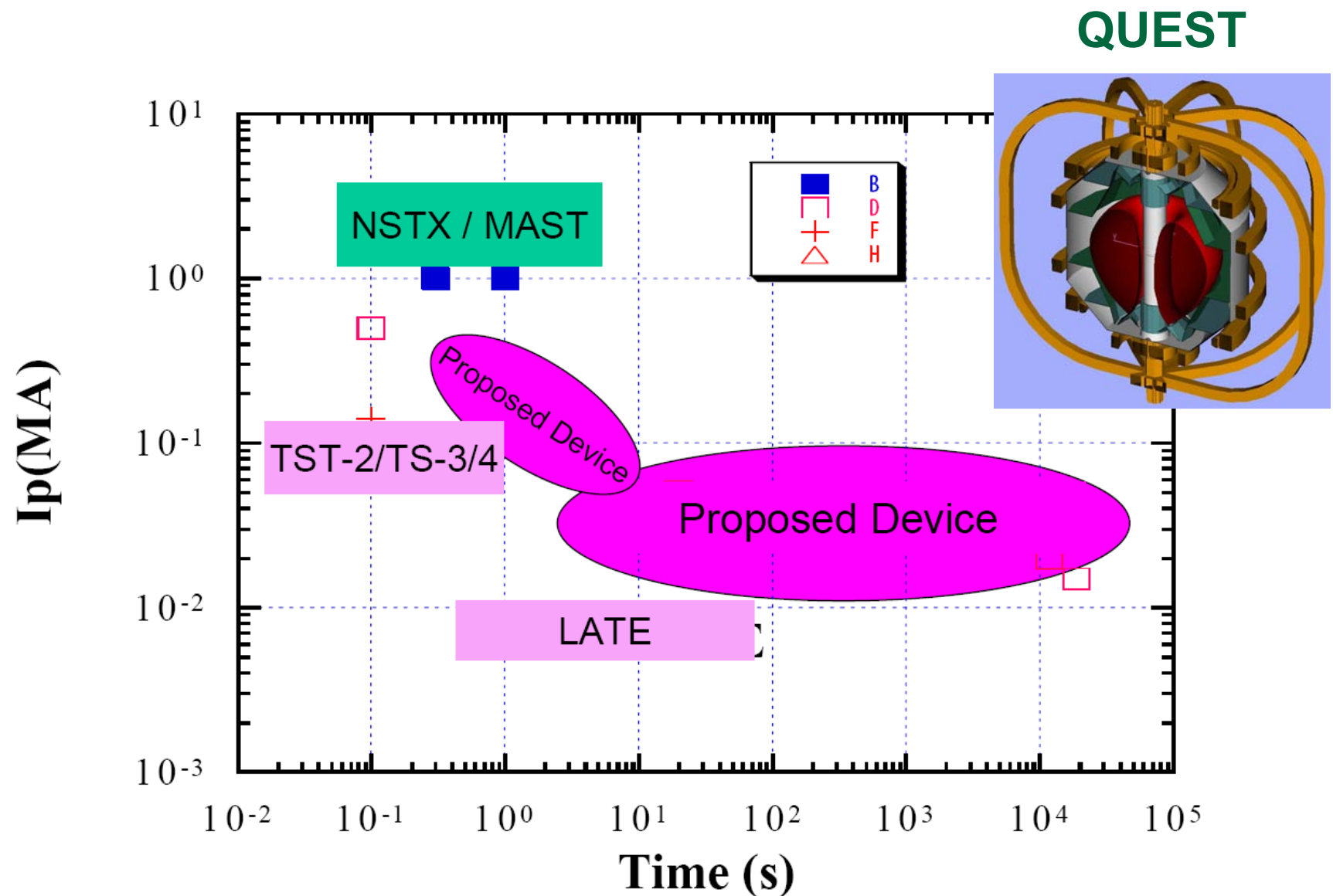
Sustained Parameters	CTF (1-2 MW/m ²) ($\tau \gg \tau_{\text{skin}}$)	NSTX long pulse ($\kappa \leq 2.5, \tau \sim \tau_{\text{skin}}$)
P/R (MW/m)	45 – 75	≤ 9
SOL expansion factor	10 – 20	~ 5
$V_{\alpha}/V_{\text{Alfvén}}$	3 – 6	1 – 4 ($V_{\text{NB}}/V_{\text{Alfvén}}$)
$I_{\text{BS}}/I_{\text{CD}}$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

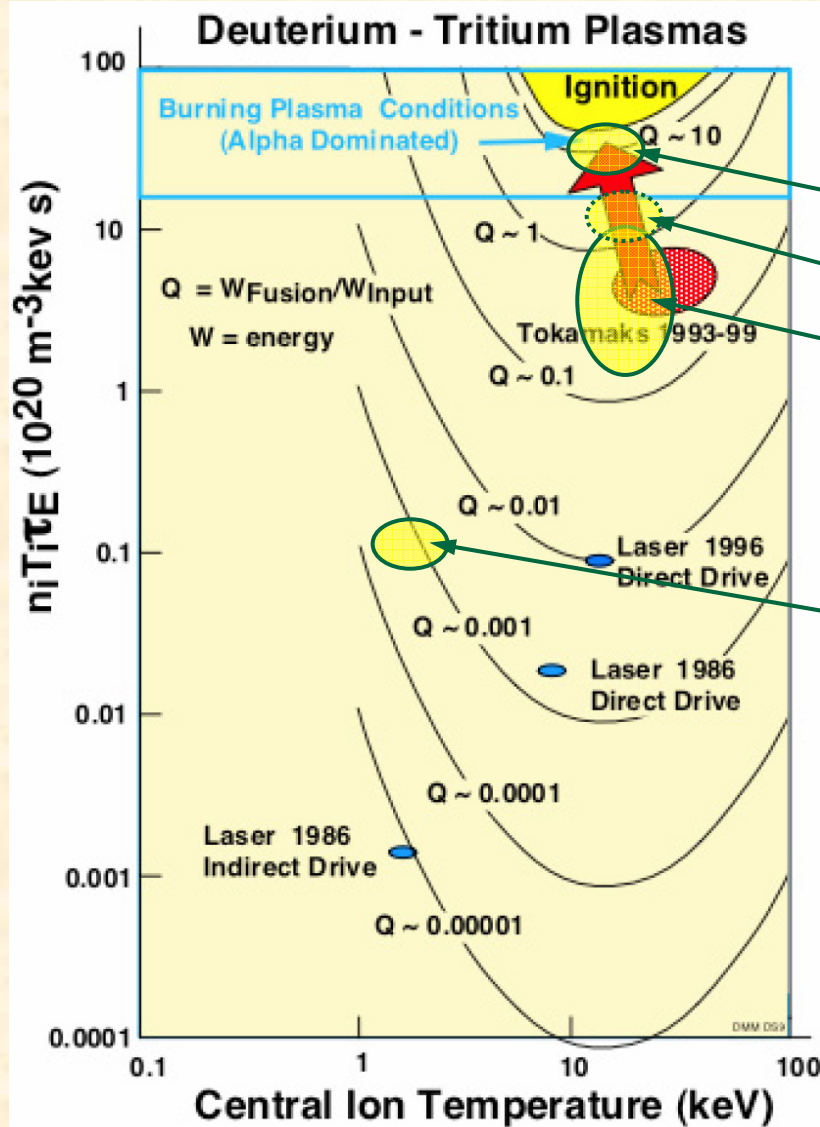
ITF plasma shape & stable current profile using NBI & Bootstrap



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



ITER (Op 2016)

CTF

- Present and near-term S/C Tokamaks (DT equivalent)

- **Present STs**
(DT equivalent)

Fusion Demo Requires

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

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; $\beta_N/\beta_{N-limit}$, %	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**