I3.006

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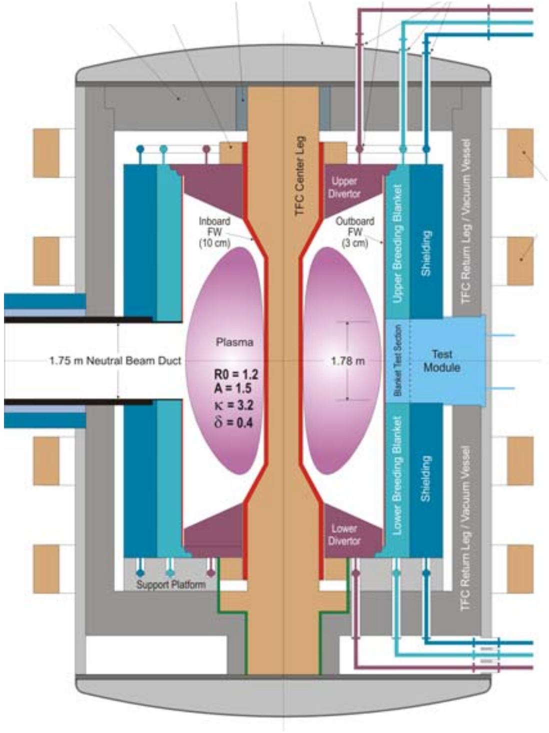
A Component Test Facility (CTF) Based on the Spherical Tokamak



Y-K Martin Peng,

PJ Fogarty, DJ Strickler, TW Burgess, BE Nelson, J Tsai
Oak Ridge National Laboratory, UT-Battelle

C Neumeyer, R Bell, C Kessel, J Menard, D Gates, B LeBlanc, D Mikkelsen, L Grisham, J Schmidt, P Rutherford
Princeton Plasma Physics Laboratory

A Field, A Sykes, I Cook
UKAEA Culham Science Center

S Sabbagh, Columbia University
O Mitarai, Kyushu Tokai University
Y Takase, University of Tokyo

32nd EPS Conference on Plasma Physics
Combined with the
8th International Workshop on Fast Ignition of Fusion Targets
27 June – 1 July 2005
Tarragona, Spain

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

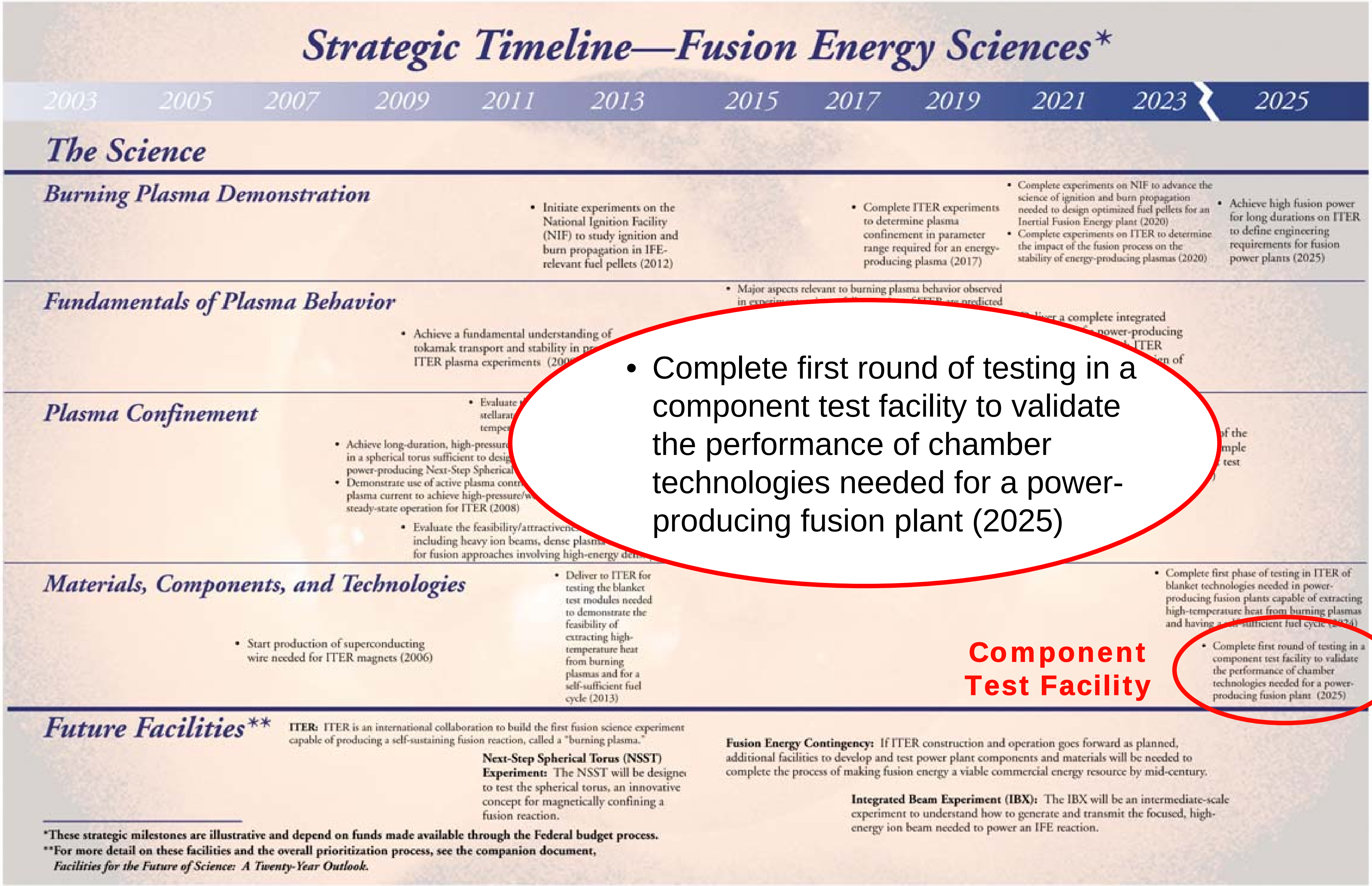
- **INL operated 45 small research fission facilities during 1951-69**
- **Necessary fusion Demo-relevant testing environment:**
  [M Abdou et al, Fusion Technology, **29** (1999) 1.]
  - **High 14 MeV neutron flux over large wall areas**
  - **High duty factor to achieve high neutron fluence per year**
  - **High accumulated fluence in facility lifetime**
- **Test tritium self-sufficiency – 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	>$10^{6\text{-}7}$	~$10^{7\text{-}8}$
14-MeV neutron flux on wall (MW/m^2)	~0.8	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	64	97
Total area of (test) blankets (m^2)	~12	~65	~670
Expected fusion power (MW)	~500	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**

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)
- 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 experim ITER are predicted
- liver a complete integrated power-producing ITER ion of

Plasma Confinement

- Evaluate stellarat temper
- Achieve long-duration, high-pressur in a spherical torus sufficient to desig power-producing Next-Step Spherical
- Demonstrate use of active plasma contr plasma current to achieve high-pressure/w steady-state operation for ITER (2008)
- Evaluate the feasibility/attractivene including heavy ion beams, dense plasm 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.

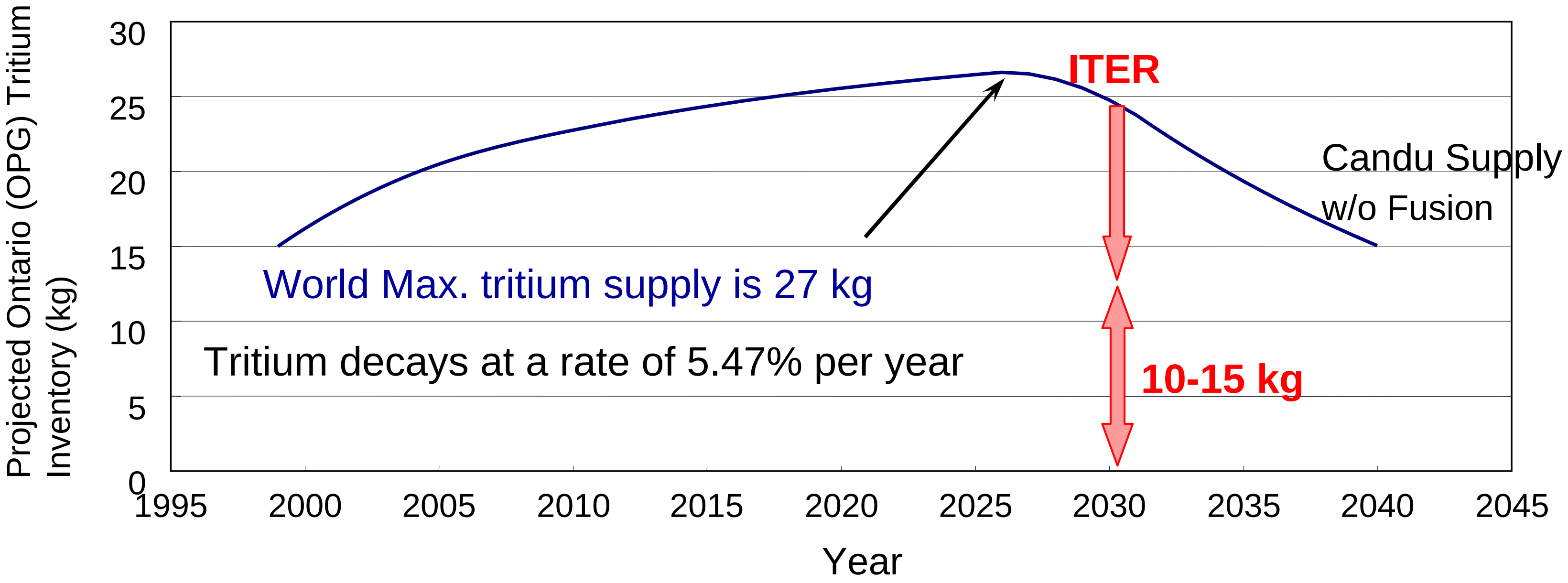
***These strategic milestones are illustrative and depend on funds made available through the Federal budget process.**

****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)

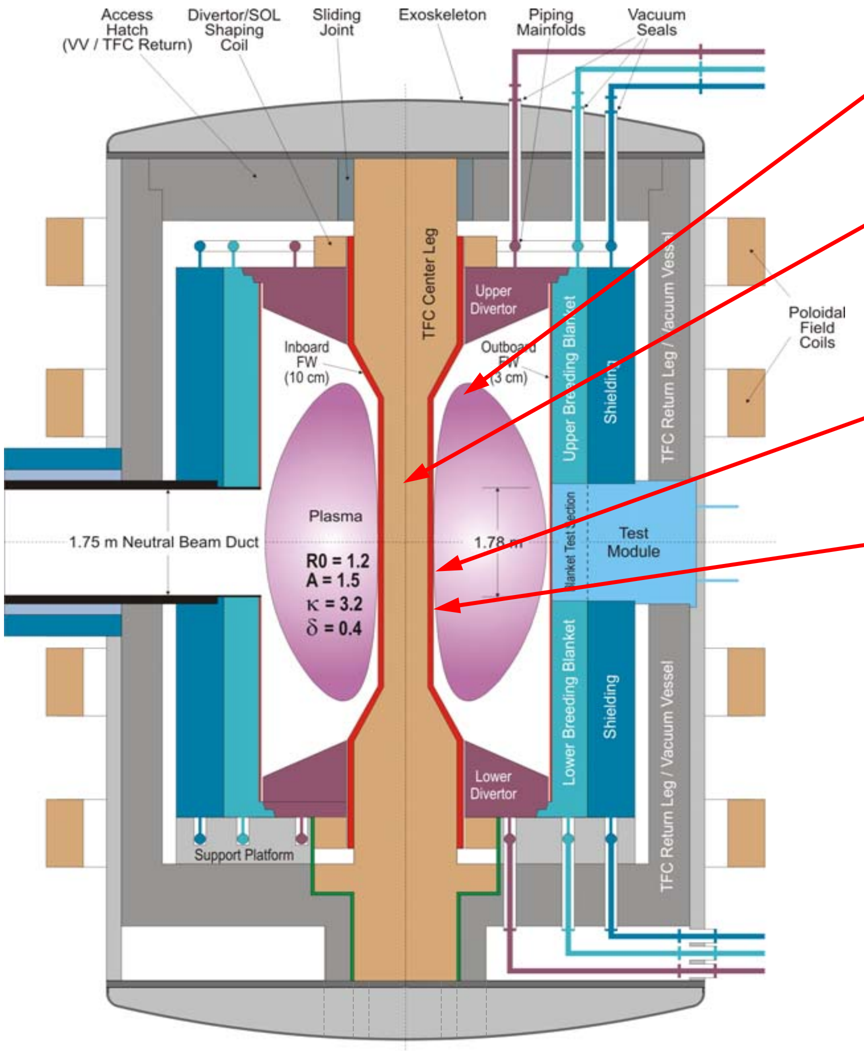
Component Test Facility

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

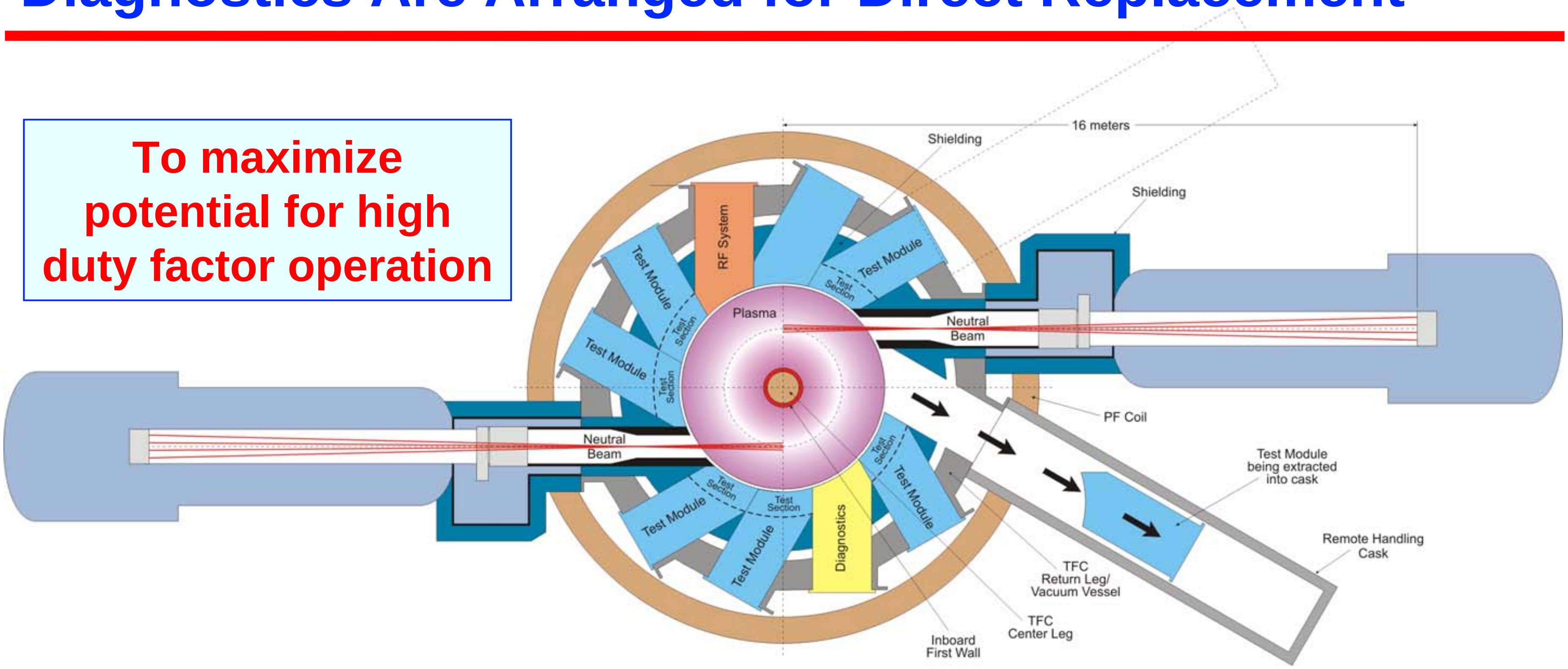


$R_0 = 1.2$ m, $a = 0.8$ m

- **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

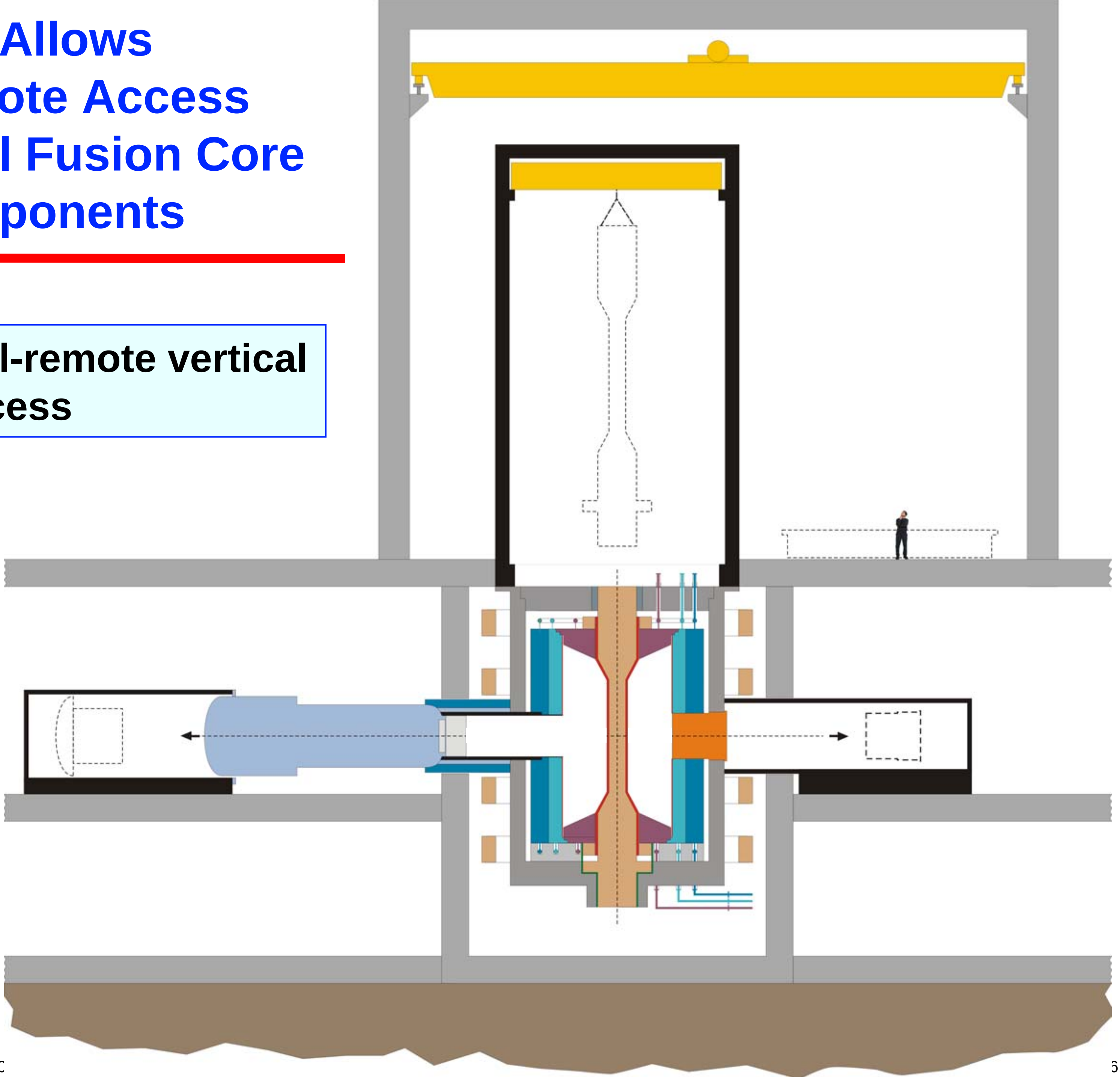
To maximize potential for high duty factor operation



- **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)**

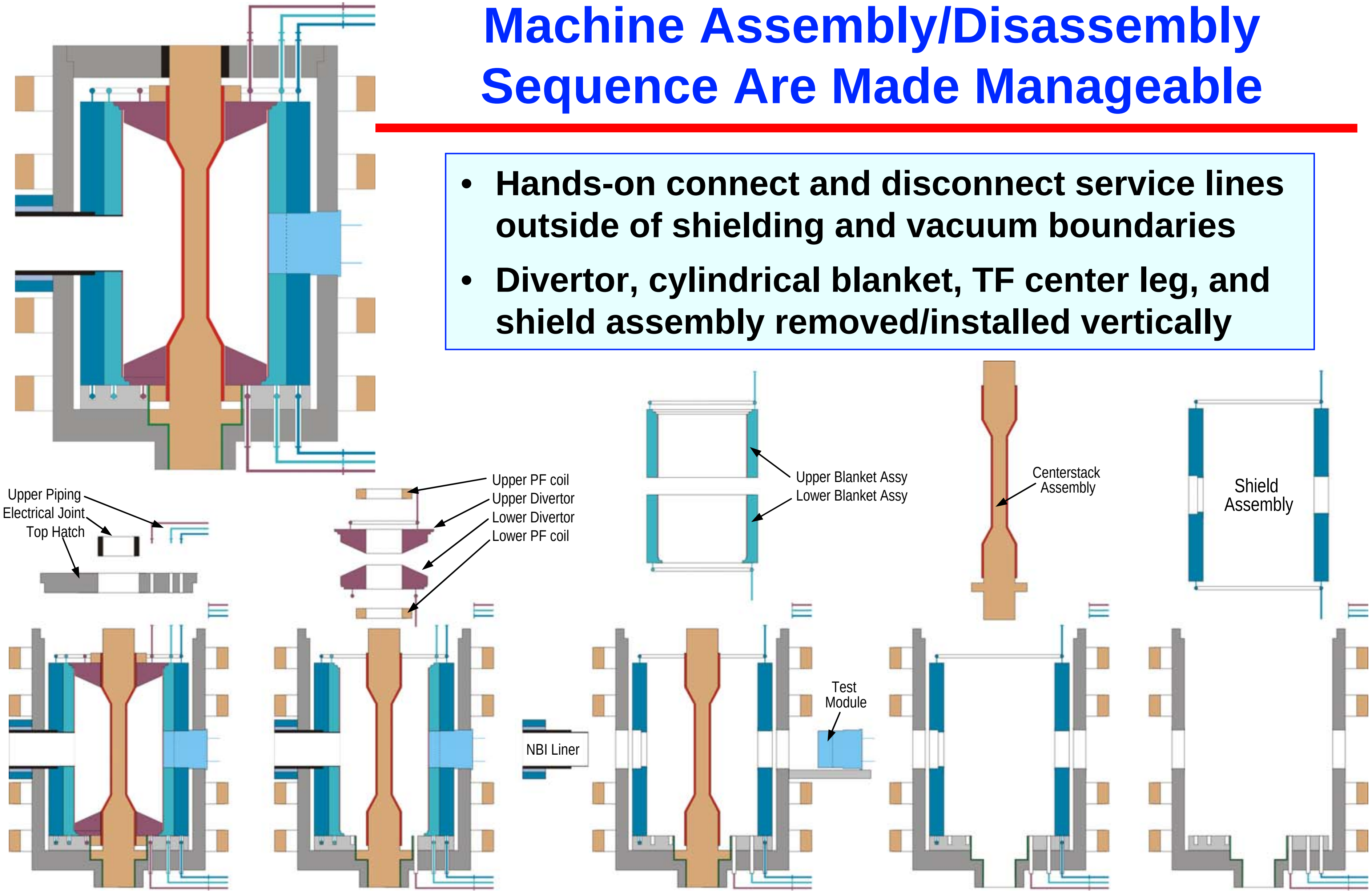
CTF Allows Remote Access to All Fusion Core Components

- **Full-remote vertical access**



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

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

Systems Code ⇒ R_0 = 1.2 m, a = 0.8 m, κ = 3.2, B_T = 2.5 T

14MeV neut. flux, MW/m^2	**2.0**	**4.0**
I_p, MA	12.8	16.1
Combined H_{98pby} factor	1.48	1.38
β_T, %	24	**39**
$\beta_N H_{89P}$	11.3	16
Safety factor, q_{cyl}	3.0	**2.4**
n/n_{GW}	0.17	0.21
I_{BS}/I_p	0.43	0.44
P_{fusion}, MW	144	288
P_{NBI+RF}, MW	40	65
Neutral beam energy, kV	160	250
f_{rad}, % (for P_{div} = 15 MW/m^2)	**75**	**90**
Net $T_{consumption}$/yr goal, gm	14	180

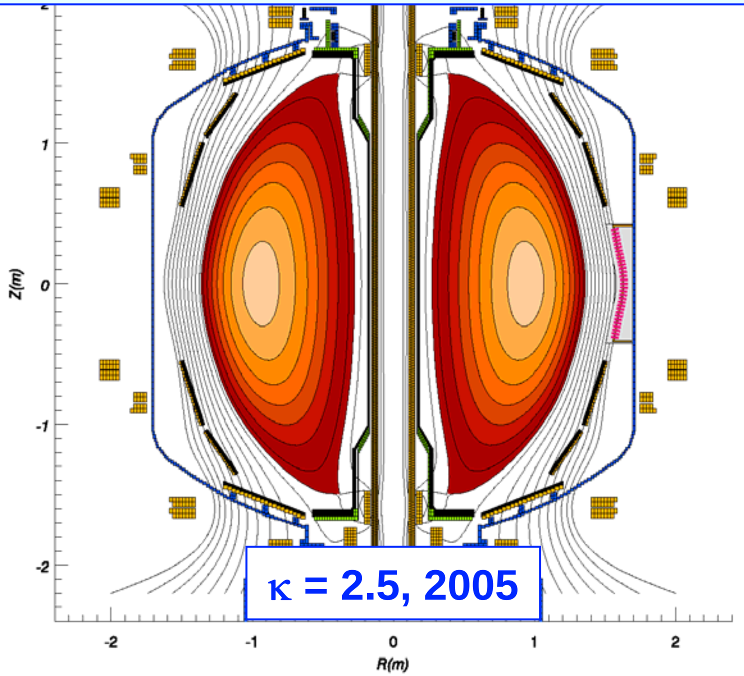
- **Baseline (2 W/m^2) parameters within ST plasma operation limits**
- **Higher neutron fluxes reach progressively more limits**
  - In β, q_{cyl}, and f_{rad}
  - Requires densities << limit
- **Technology & physics of CTF advances in synchrony**
  - 2 MW/m^2 – medium ST physics to test technologies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

CTF Can Utilize Attractive ST Physics Properties

Encouraging NSTX & MAST results

C Roach: I2.006,
J Menard: O4.007,
S Kaye: P5.042,
B Stratton: P1.060,
R Raman: P1.063,
I Chapman: P2.062,
V Soukhanoskii: P4.016,
B Dudson: P4.019,
E ElChambre: P5.015,
D Applegate: P5.101,
A Surkov: P5.103,

A Kirk: O4.001
P Helander: I5.003
A Sykes: P4.112
E Fredrickson: P1.061
V Rozhanski: P2.017
D Howell: P2.061
R Maingi: P4.017
M Wisse: P4.100
M Redi: P5.041
G Madison: P5.102
G Antar: D5.005



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/ρ_i~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 ($f_T \lesssim 1$)
- Strong SOL expansion

Allows easier solenoid-free operation

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

Heating and Current Drive opportunities

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

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

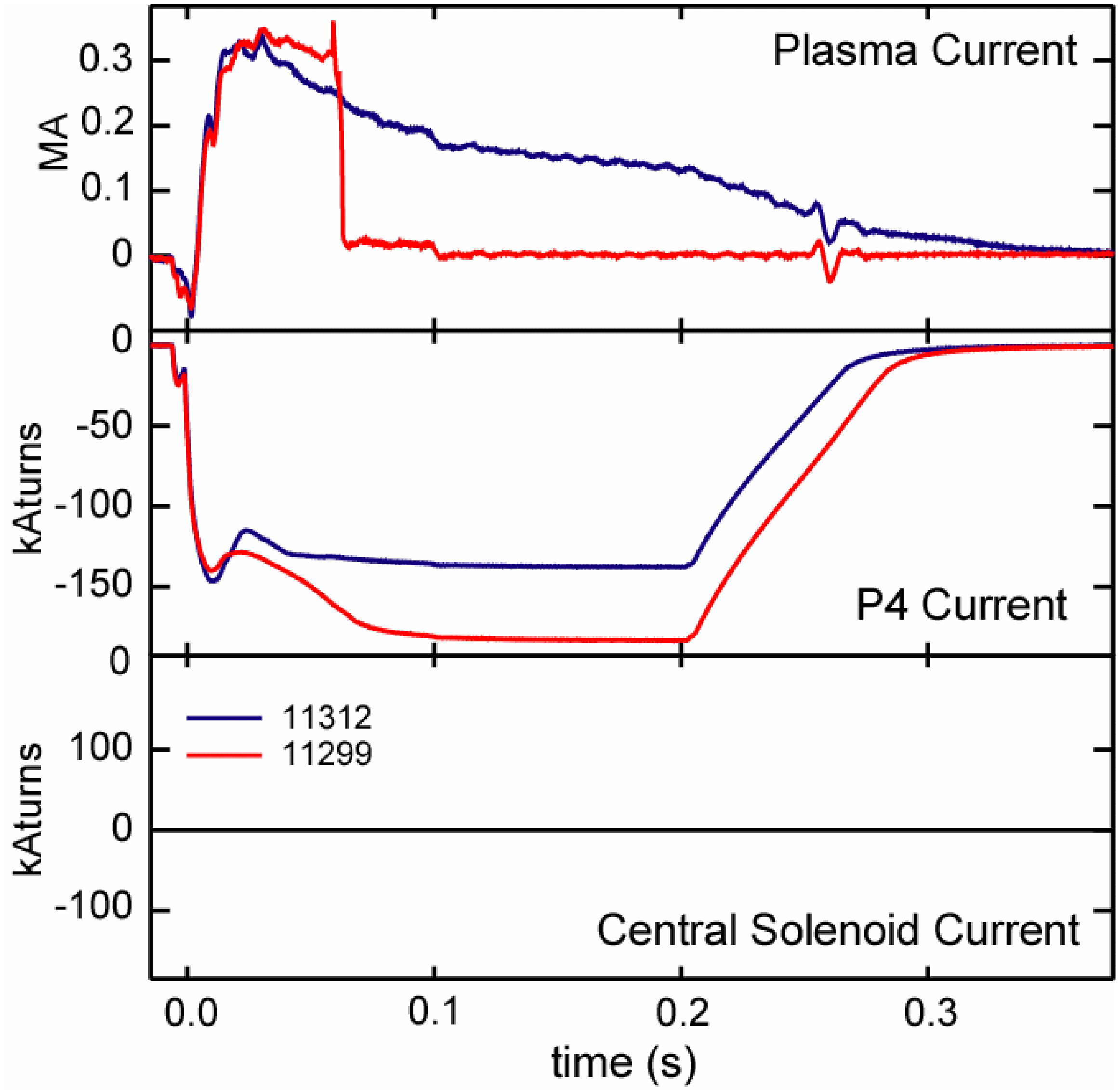
Sustained Parameters	**CTF** **($\tau >> \tau_{skin}$)**	**Long Pulse Data** **($\tau > \tau_{skin}$)**
I_p/aB_T (MA/m-T)	$\leq$6.4	$\leq$4.4
Safety factor, q_{cyl}	$\geq$3.0	$\geq$2.2
β_N (%-m-T/MA)	$\leq$3.9	$\leq$5
β_T (%)	$\leq$24	$\leq$23
Start-up to $\mu_0 \ell_i R I_p$ (Wb)	$\geq$3.8	~0.13 (goal)

Required Investigations

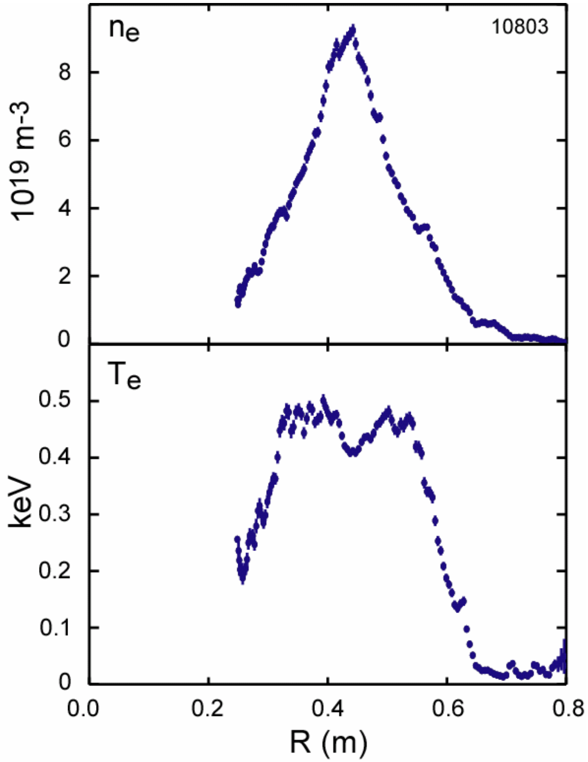
- Macro-stability near CTF conditions: $\kappa \leq 2.7$ and $\tau >> \tau_{skin}$
- Error field & resistive wall mode, with strong plasma rotation, toward high reliability & higher β_N
- Solenoid-free start-up to ~ 0.5 MA plasma target for NBI and EBW

Issue: solenoid-free startup [Raman: P1.063; Sykes: P4.112]

‘Double Null Merging’ Scheme on MAST: Plasma Current up to 340kA Formed and Plasma Sustained for 0.3sec with Zero Current in Central Solenoid (Sykes: P4.112)



UKAEA Fusion Working in Europe



Plasma is hot (~ 0.5keV) and dense (9x10^{19}m^{-3})

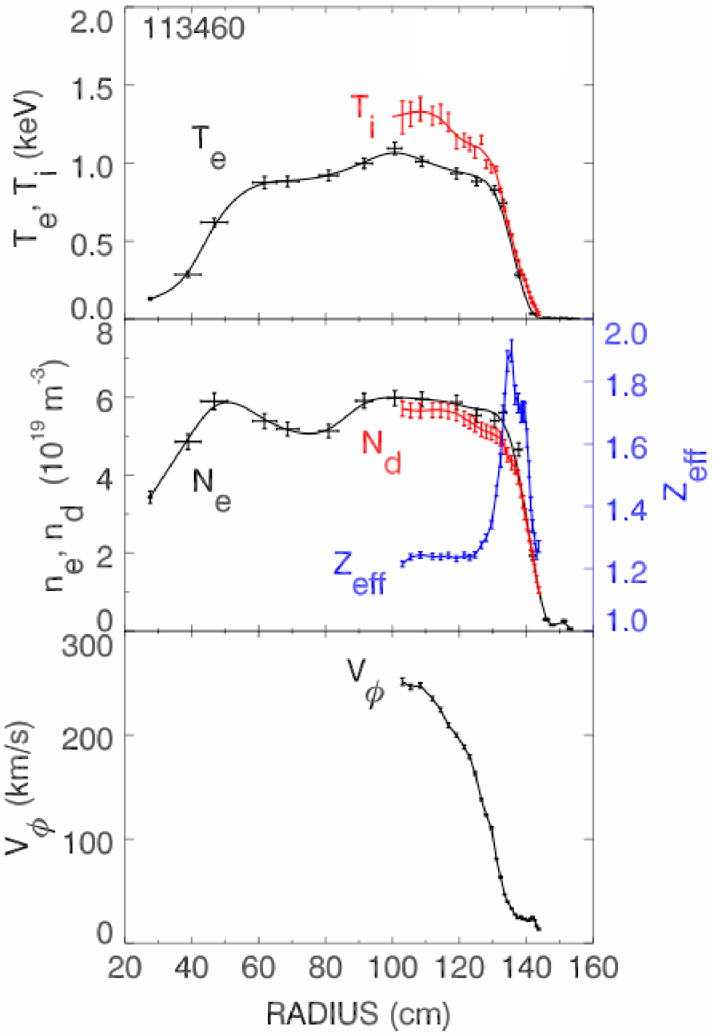
CTF Confinement Assumptions Are Suggested by Long-Pulse Plasmas in NSTX & MAST

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/ρ_i*)	~50	~30
H_{98pby2}	≤1.5	≤1.3 for >τ_{skin}

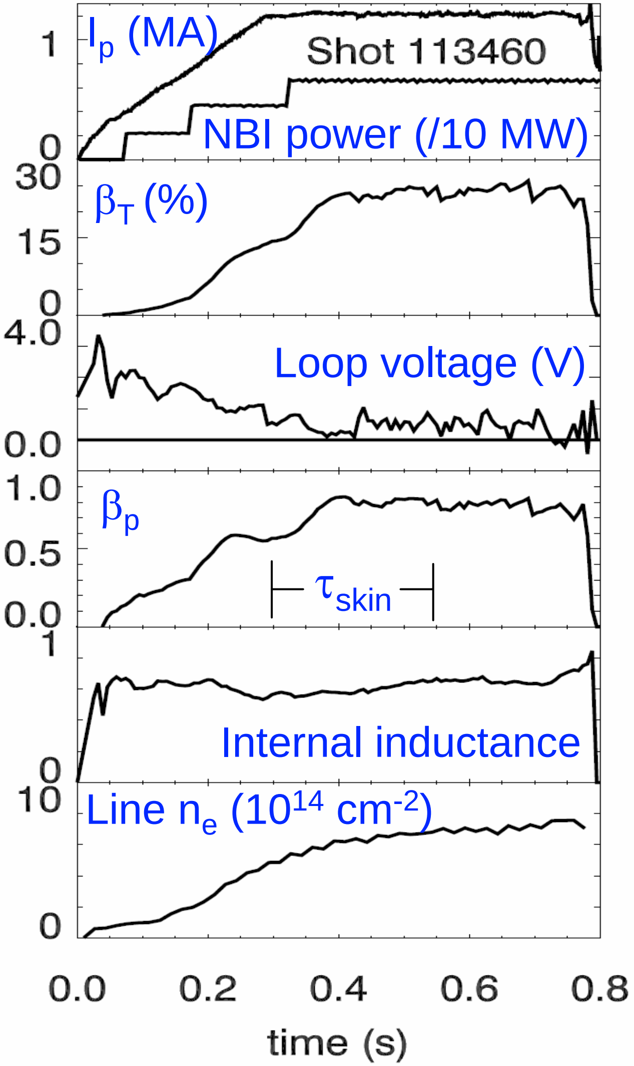
Required Investigations

- Strongly rotating plasma with ion "internal transport barrier" via co-NBI
- Beta-exponent in scaling
- Density control at low n_{GW}, such as via lithium
- Electron transport vs. β effects: τ_{Ee} [Kaye: P5.042]
- Ion transport vs. neoclassical: τ_{Ei} [Roach: I2.006]

Long-pulse H-mode



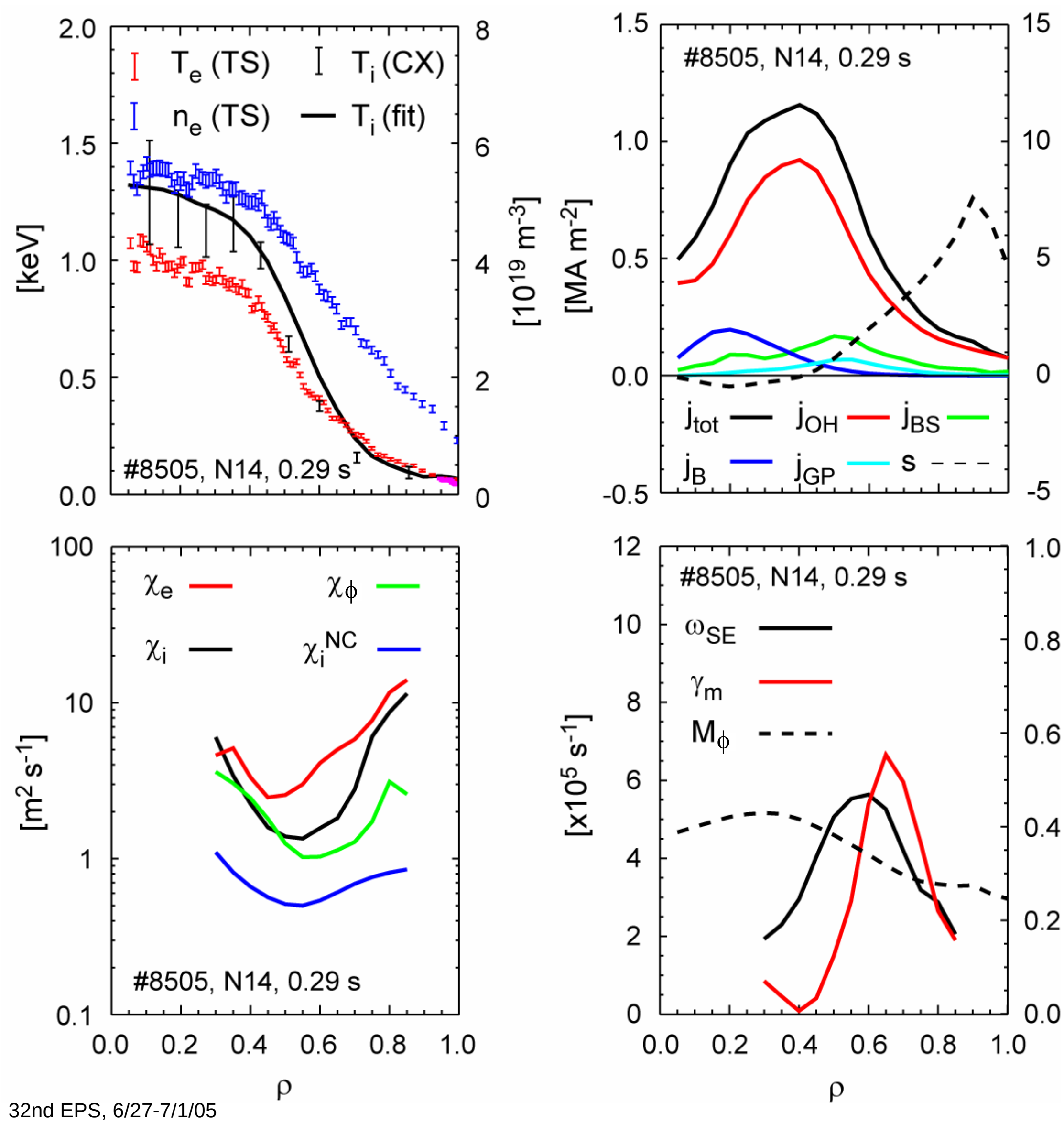
NSTX Has Made Significant Progress Towards Goal of High-β_T, Non-Inductive Operation



- $\tau_{Ip\ flattop} \sim 2\ \tau_{skin}$
- $\tau_{W\ flattop} \sim 9\ \tau_E$
- $\beta_T > 23\%$, $\beta_N > 5.3$
- $H_{89P} \sim 2$
- Internal inductance ~ 0.6
- $n_e \sim 0.5\times10^{13}$ /cm^3
- 1.5-s pulses in 2005

[J Menard: O4.007 – NSTX progress]

MAST Measured Sawtooth-Free L-Mode Plasma with Improved Core Confinement and Weak Central Shear, Potentially Suitable for CTF



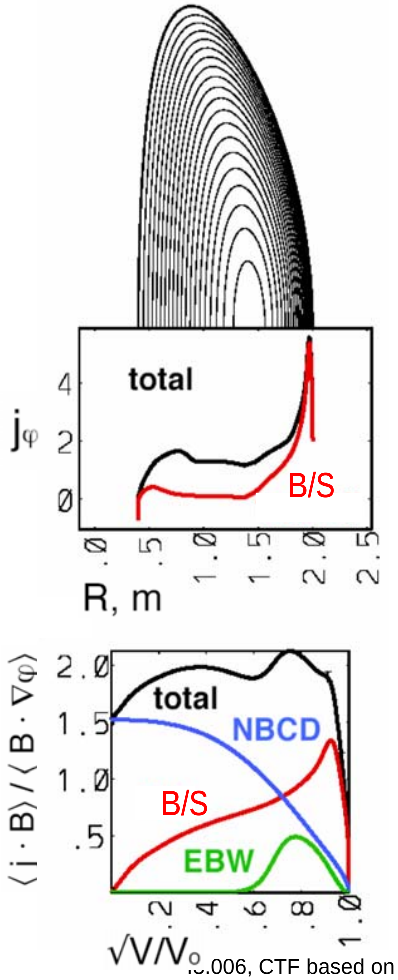
Transport analysis:

- $n_e/n_G \sim 0.7$; $P_{NBI} \sim 1.8$ MW
- $Q_i \sim Q_e$; $T_i \geq T_e \sim 1.0$ keV
- Hollow j(r) profile
- $\chi_i \sim 2\text{-}3\ \chi_i^{NC}$ at $\rho \sim 0.4\text{-}0.6$
- $\chi_e \sim 1\text{-}2\ \chi_i$
- ExB shear $\omega_{ExB} > \gamma^{ITG}$ at $\rho < 0.6$

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

Sustained Parameters	CTF ($\tau \gg \tau_{skin}$)	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)	64	≤9
SOL area expansion	10 – 20	~5
Radiation fraction (%)	~75	≤30

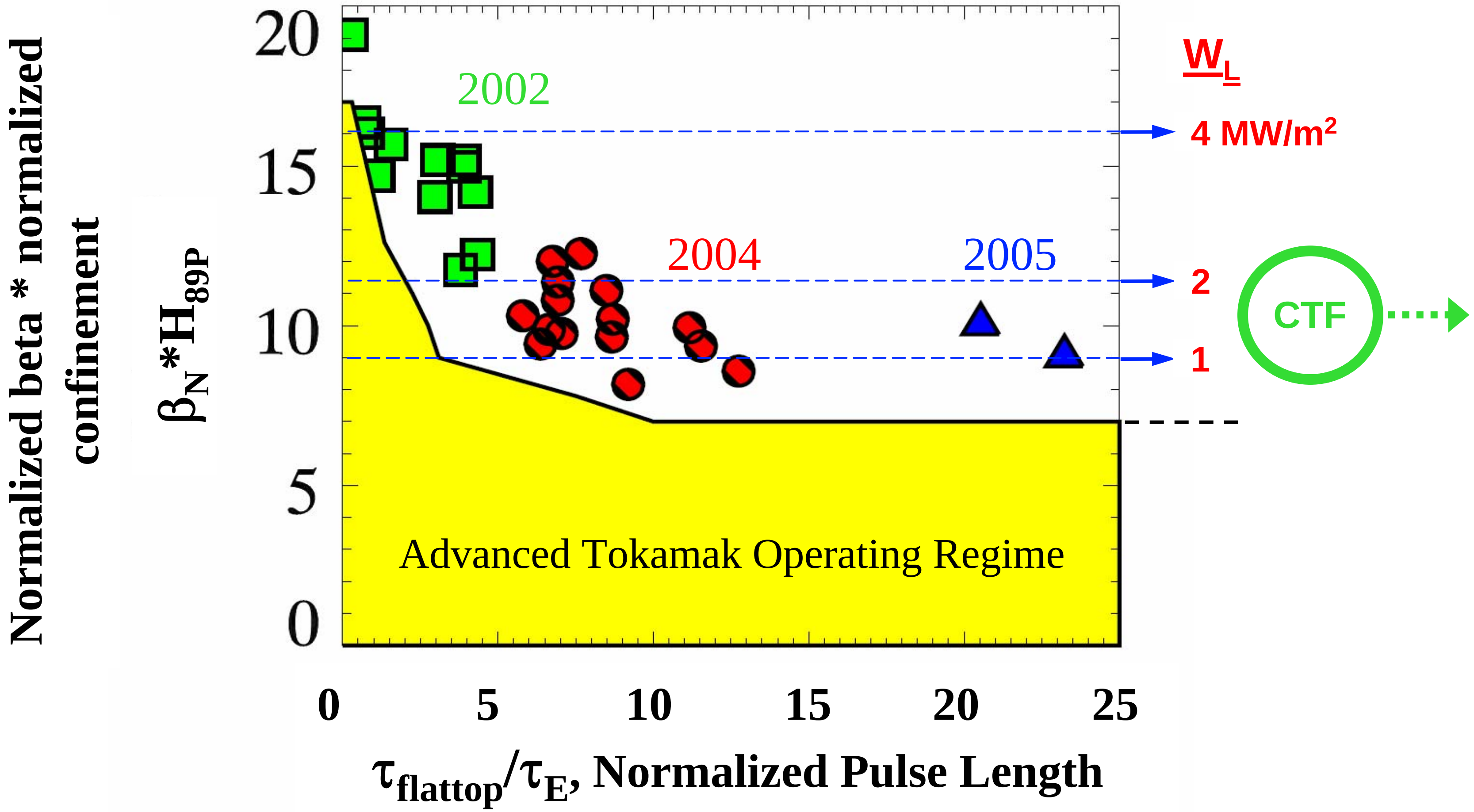
CTF Plasma Shape & Stable Current Profile



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)**

Normalized Plasma Performance (β_N*H_{89P}) with Long Pulse Lengths on NSTX Reached the CTF Level



CTF Technology Draws from and Extends Present Fusion Program Plans

To Achieve Baseline Performance (2 MW/m^2)

- **Plasma facing components – twice ITER fluxes**
  - Take advantage of DEMO-relevant ITER designs
  - Needs highly reliable and remotely replaceable divertor components; explore lithium options
- **Heating, current drive, and fueling – similar to ITER**
  - Positive & negative ion beam under development by LHD, JT60U; ITER NBI R&D
  - MW-level EBW at ~70 or 140 GHz being developed and used
  - Highly reliable and remotely replaceable RF launchers
- ***Requires database from long-pulse high performance tests (Tore Supra, KStar, LHD, ITER, test stands, etc.)***

New: TF system engineering – single turn copper

- TF center leg optimization and fabrication technology
- Multi-MA, low-voltage TF power supply

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**
- **Additional ST physics data needs are identified**
- **CTF technology draws from and extends present fusion program plans; single-turn toroidal field coil is new**

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

SuperCode Costing Components	R_0=1.2m A= 1.5	Comments
1. Toroidal Device	**193**	
– TF magnets	38	
• *TFC center post*	*(12)*	$U_{TFcenter} = \$0.075$M/ton (single-turn cooled GlidCop)
• *TFC outer magnet (VV)*	*(26)*	$U_{TFouter} = \$0.03$M/ton (single-turn Al, combined with VV)
– PF magnets	50	$U_{PF} = \$0.058$M/ton (no OH solenoid)
– Device structure	11	$U_{MS} = \$0.052$M/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 A=1.5**	**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**	
– ECH-EBW	40	8, 10 MW @ 100 GHz, 12 MW @ 200 GHz (ITER-FEAT: 111)*
– NBI	125	30, 33, 34 MW at ~ 400 kV (ITER-FEAT: 138)
– LH	0	
– Plasma operational I&C	45	ITER-FEAT: 214; FIRE: 29
6. Site, Facilities and Equipment	**252**	
– Land, site improvement	0	Government site
– Buildings	180	ITER-FEAT: 546; FIRE: 126
– Hot cell	0	Included in Buildings
– Radwaste management	38	ITER-FEAT:12; FIRE: 11 (CTF requires FNT testing at high duty factors, substantially increasing radwaste)
– Coolant supply and disposal	18	ITER-FEAT: ?; FIRE: 18
– General test and qualification	16	(CTF requires acceptance verification of all incoming test components.)
– Magnet fabrication tools	0	
Total Construction Cost, no Contingency	**1,050**	
with 40% Contingency	**1,470**	Included in the ST development cost

* Comments by D. Rasmussen, R. Temkin