Physics Requirements for Fusion Technology Applications of the ST

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Is the ST a viable route to fusion?

The ST has a number of promising features, eg

simpler construction than conventional tokamaks
good confinement,
low halo currents,
high density operation,
good stability (particularly at high elongation).

But questions remain to be addressed:

How does confinement scale?
Are there options for handling the exhaust?
What is the pressure limit (β_N~6 achieved on START and NSTX)?
Can we demonstrate non-inductive current drive (and start-up)?
Fundamental plasma physics at high β~1

We are entering an exciting era

The role that the ST has to play in the development of fusion should become clear in the next few years

- **Two future ST devices to consider:**
  1 GW(e) ST Power plant (STPP)
  ST Component Test Facility (CTF) to complement IFMIF
- MAST Phase I mainly addressed **CTF** issues

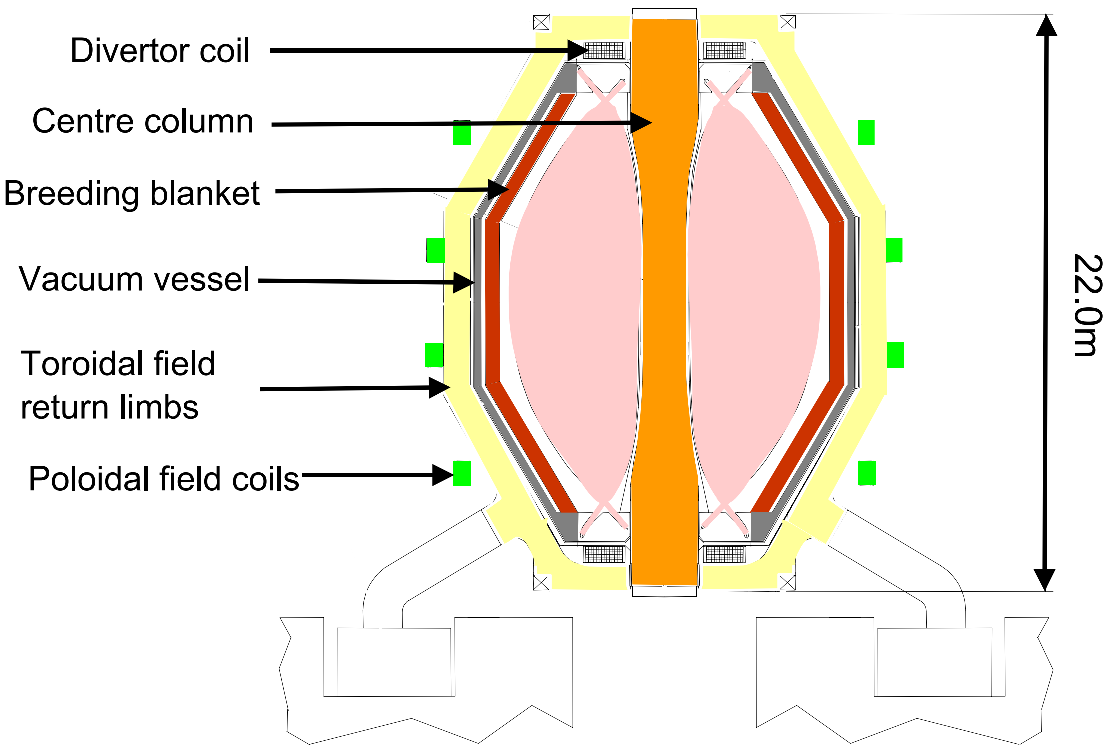
  - **STPP** requires $\kappa \sim 3.2$, $\beta_N \sim 8.2$, $I_p/I_{rod} \sim 1$

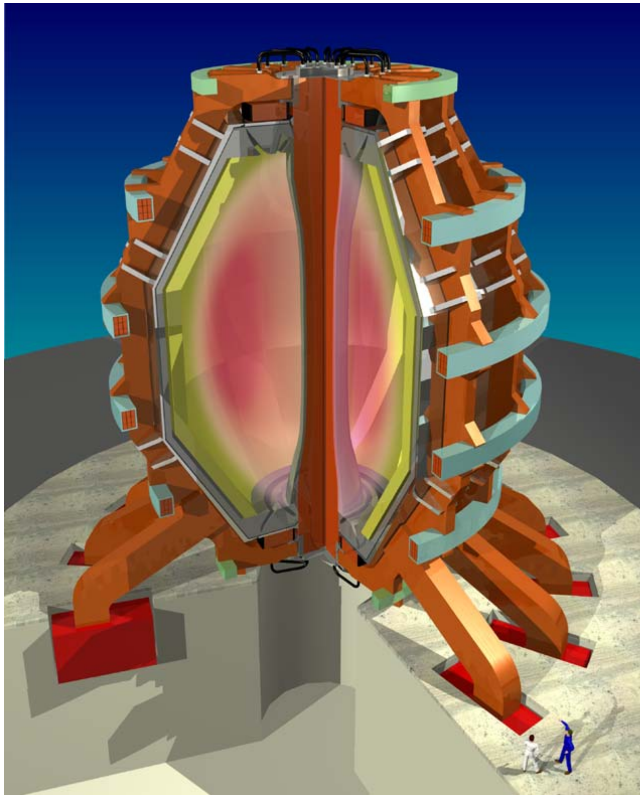
	STPP	CTF
Aspect ratio, A	1.4	1.6
Major/minor radius, R_0/a (m)	**3.42/2.44**	**0.7/0.44**
Elongation, κ	**3.2**	**2.5**
Triangularity, δ	0.55	0.4
Plasma current, I_p (MA)	**31**	**8**
Centre rod current, I_{rod} (MA)	30.2	12
q_0, q_a	2.9, 15	1.0, 6.0
Greenwald number	**0.65**	**0.26**
β_t β_N	**59, 8.2**	**15.3, 4**
Fusion Power	3.1 GW	26.1 MW
CD power (MW)	50	45
Auxiliary CD (MA)	**2.3**	**5.7**
Pressure driven current (MA)	**28.7**	**2.3**
Ohmic current (MA)	0	0
Confinement $H_{IPB98(y,2)}$	1.6	1.26

The Spherical Tokamak Power Plant

Design is strongly influenced by

- Desire for steady state operation
- Low toroidal field (high β), to keep design simple and minimise cost of electricity
- Neutron wall loading (determines device size)





Parameter Choice

Objective: to design a compact, steady state, ~1GW(e) ST power plant (aspect ratio A=1.4)

Neutron wall loading (3.5MWm^{-2}) drives the size: R=3.4m

Cost of electricity limits toroidal field, $I_{rod} \approx I_p$

MHD limits β_N=8.2

High elongation required for ~90% pressure-driven current; vertical instability $\Rightarrow$ κ=3.2

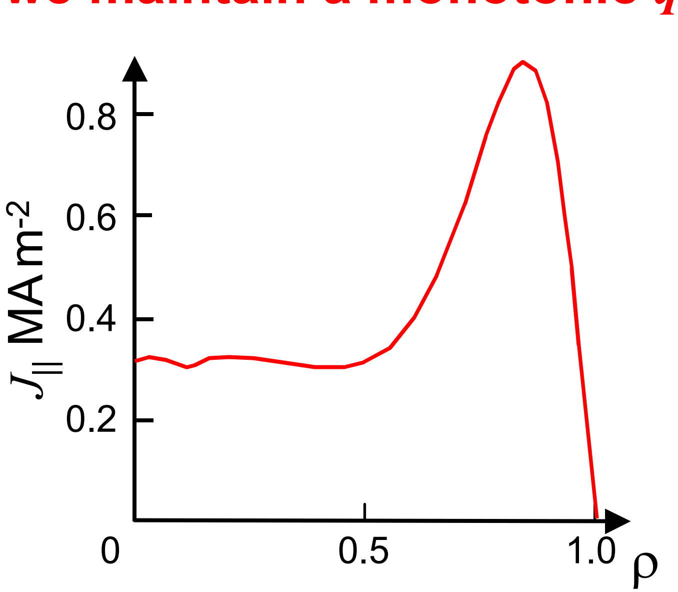
Required fusion power (~3GW) $\Rightarrow I_{rod}$=30.2MA ($\therefore$ I_p=31MA)

Non-inductive current drive requires low density ~1.1$\times$10^{20}m^{-3} (~60% Greenwald)

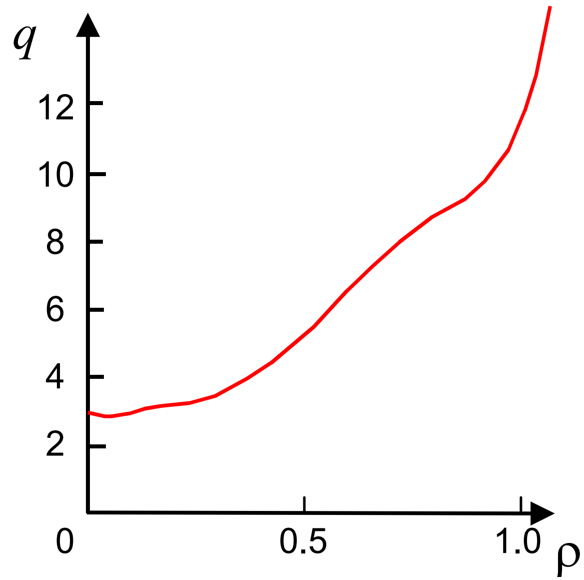
Confinement, τ_E=1.6$\tau_{IPB98(y,2)}$ or 1.4$\tau_{IPB98(y,1)}$

Pressure driven current: I_{bs} / I_{pl} = 28.7MA / 31MA

• **With 90% bootstrap current, the current profile is hollow, but we maintain a monotonic *q*-profile**



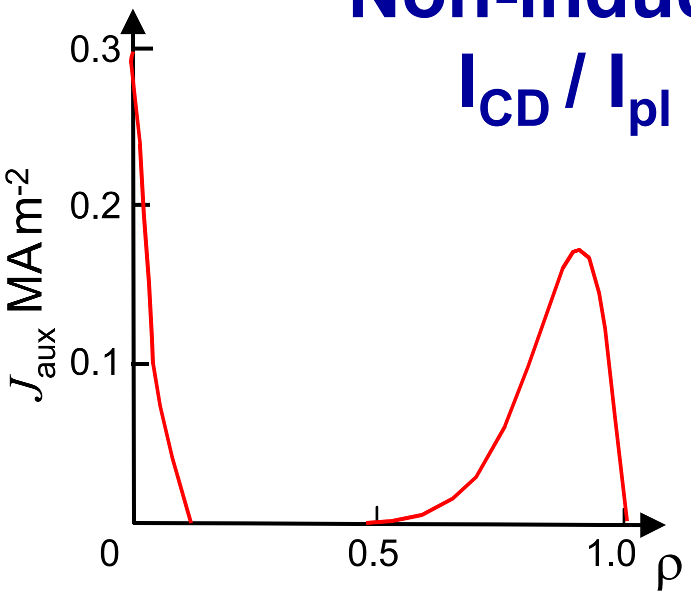
90% pressure driven current leads to non-monotonic current profile



But in an ST we can maintain a monotonic *q*-profile

Note exclusion of low order rational surfaces

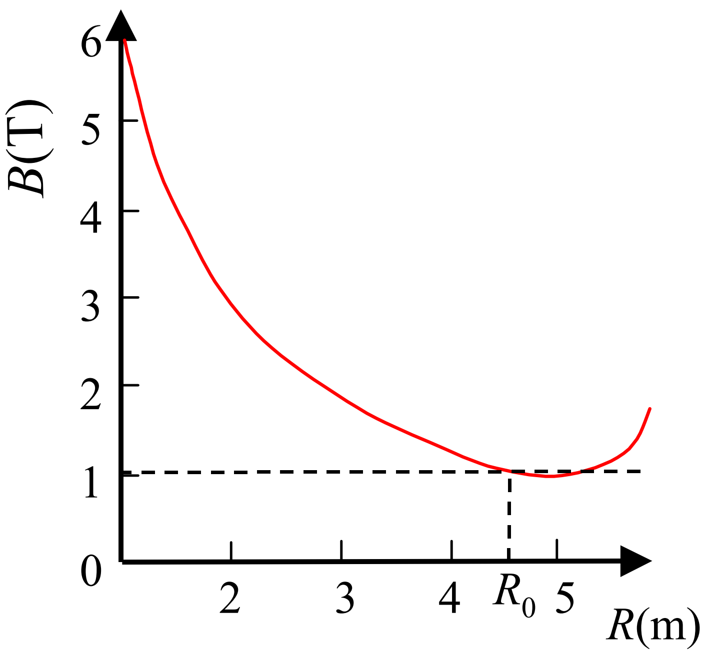
Non-inductive current drive: $I_{CD} / I_{pl} = 2.3MA / 31MA$



Options for 2.2MA off-axis CD:

40MW 80keV NBI, inclined beams

20-30MW LHCD (3.7GHz), but antenna?



Options for 0.14MA on-axis CD:

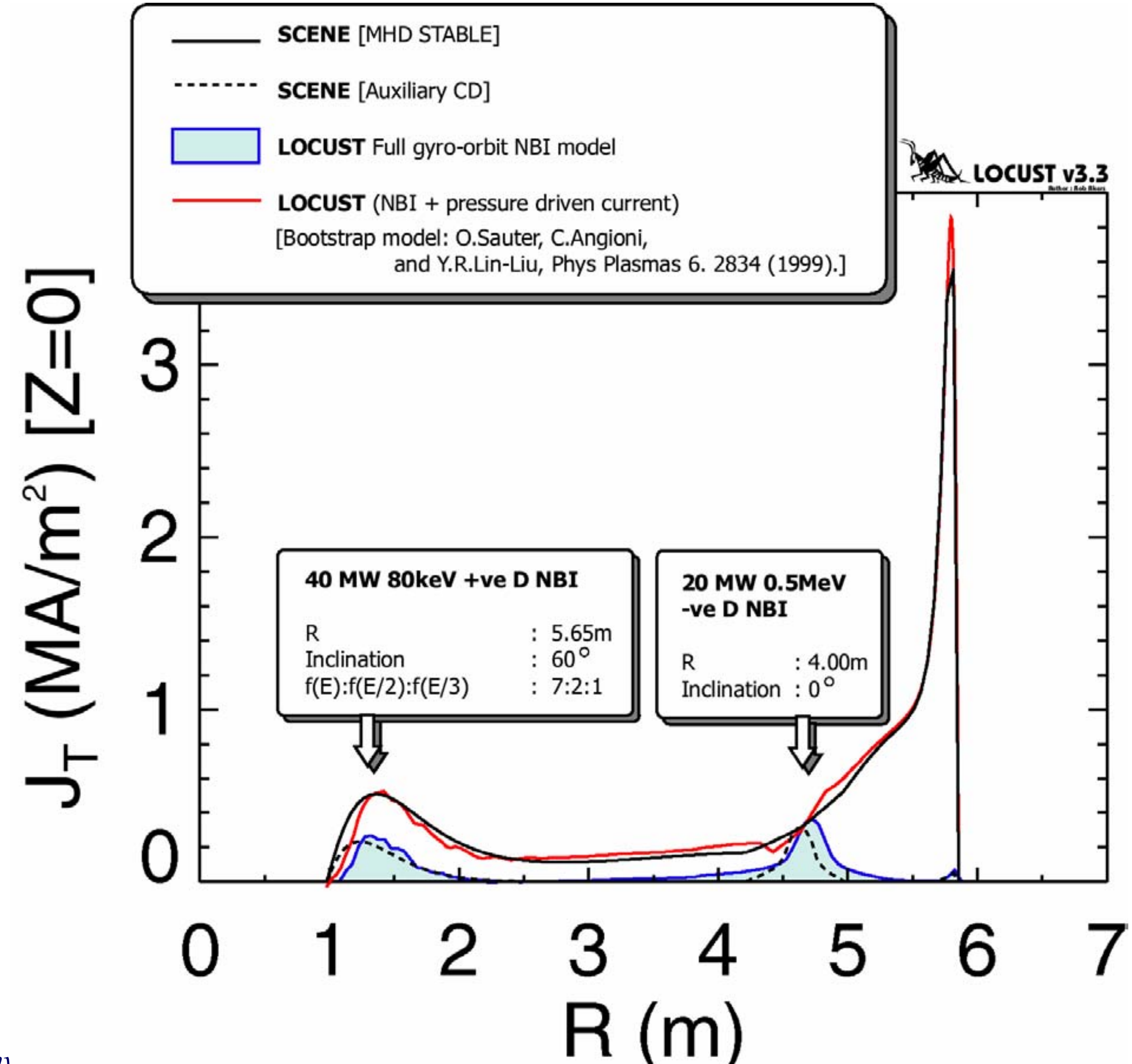
20MW 500keV NBI

15MW ECCD (130GHz, 4th harm.)

EBW very efficient, ~0.1AW^{-1} but premature absorption unresolved

Parasitic absorption on outboard mid-plane an issue: included in ECCD calc, but not EBW

Neutral beam injection modelling



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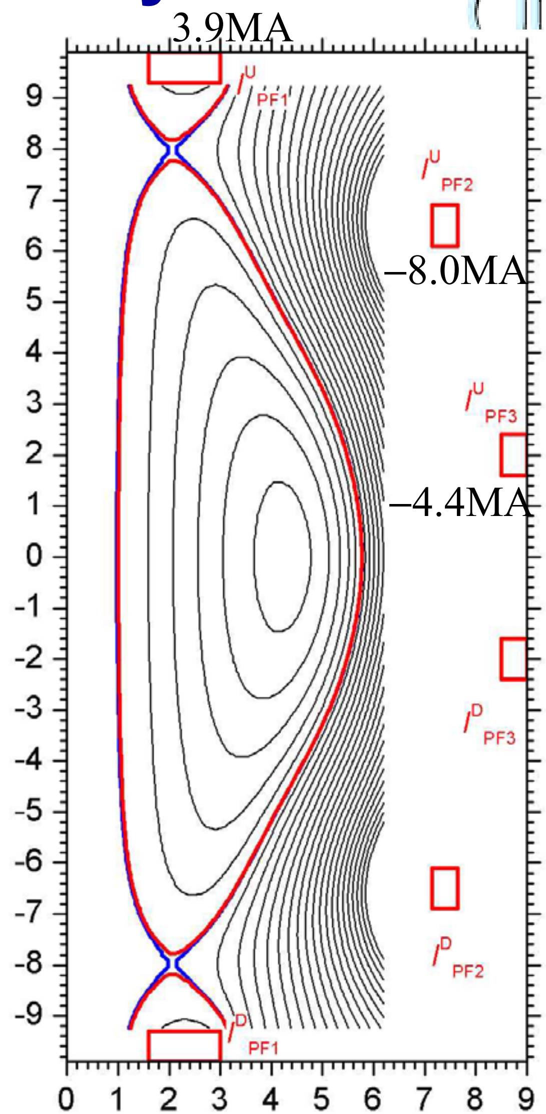
PF coil design and vertical stability

The desired plasma shape can be achieved with 3 pairs of PF coils



Vertical stability analysis indicates the equilibrium is close to marginal with no active feedback control

growth time =10ms

stability index, f_s=3.5

This can be achieved because of

tight aspect ratio

low internal inductance (hollow J profile, $l_i(2)$=0.21)

Prompt α-particle losses

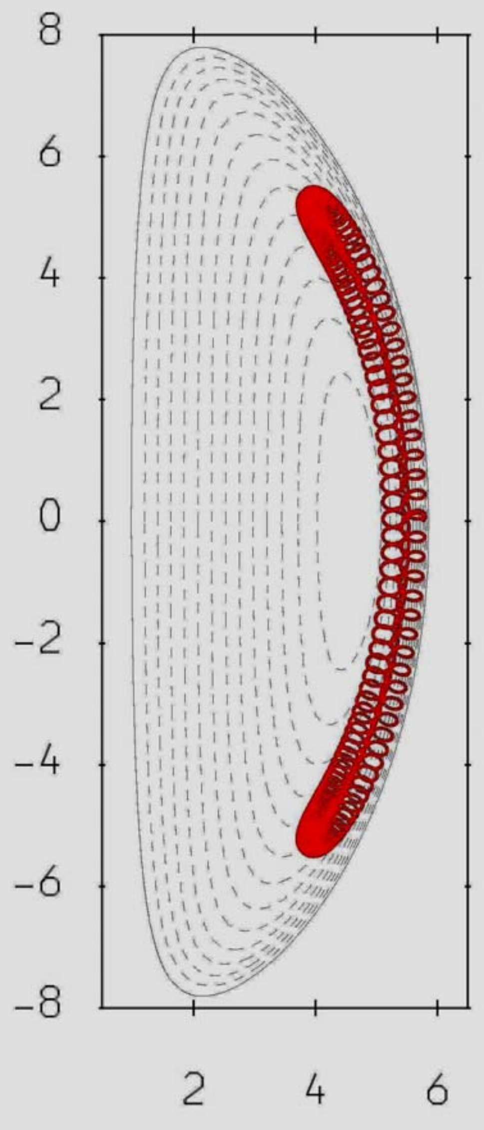
Toroidal and poloidal fields are comparable on the outboard side

a full orbit code is required to calculate prompt α losses

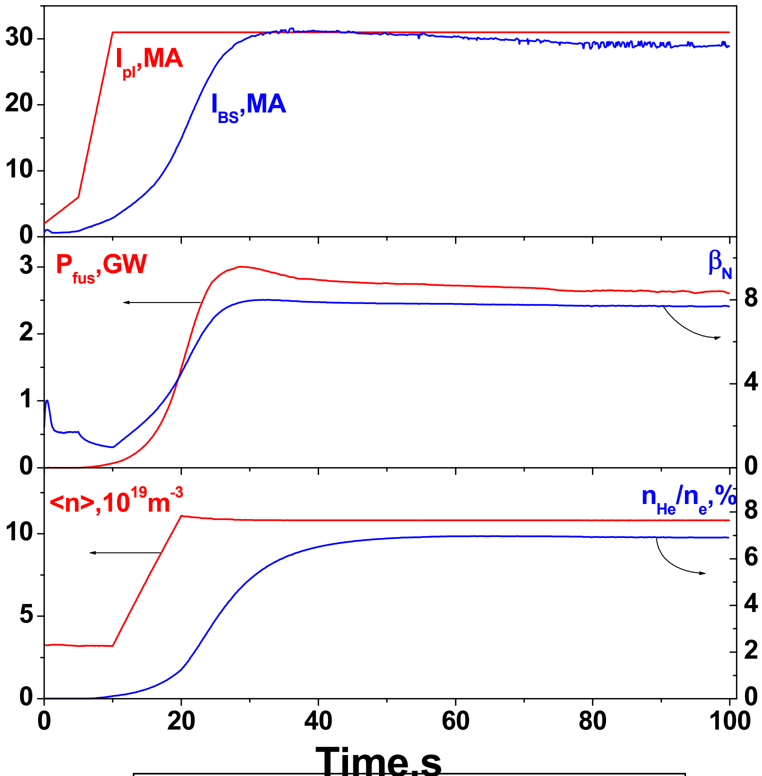
The increase in B with R has a beneficial 'pinching' effect on the orbits

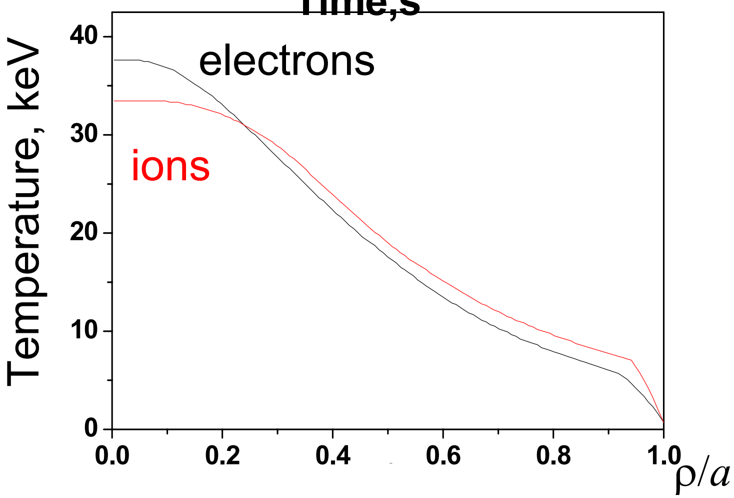
helps reduce prompt losses

Prompt losses, including TF ripple (<1% across plasma), are tolerable ~4–5%



Transport Scenario





We specify density profile and total current, and calculate evolution of: current profile, He ash, temperature and fusion power

Thermal diffusivity has a constant part, adjusted so that τ_E=1.4$\tau_{IPB98(y,1)}$

Transport equations solved using ASTRA:

Employs 50MW NBI

Confirms 3GW fusion power

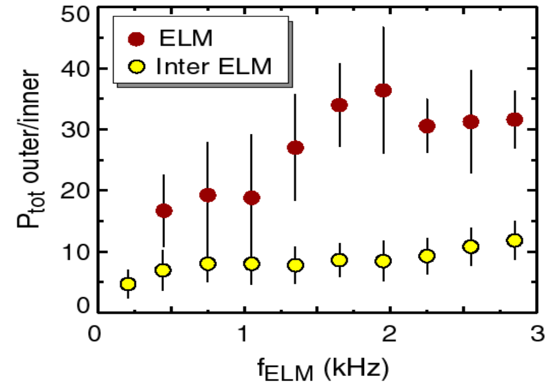
90% pressure driven current

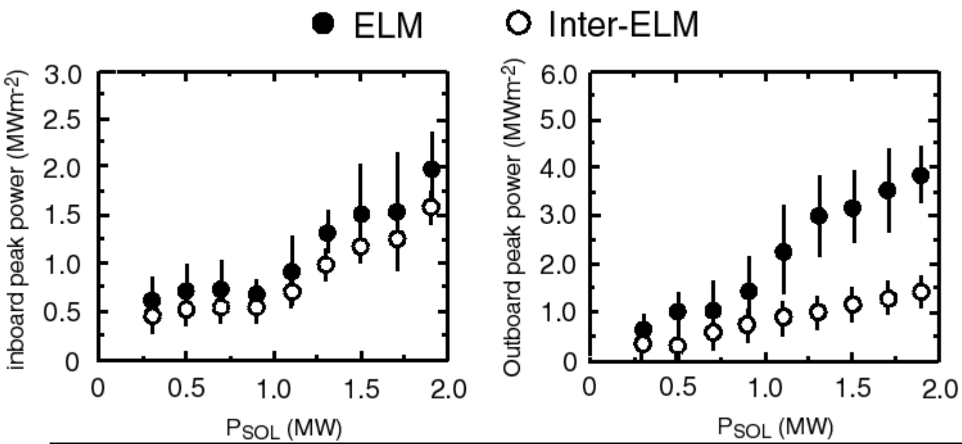
comparable electron and ion temperature profiles

Exhaust

ST geometry ⇒ most heat goes to the outboard divertor leg (95% in MAST L-mode)

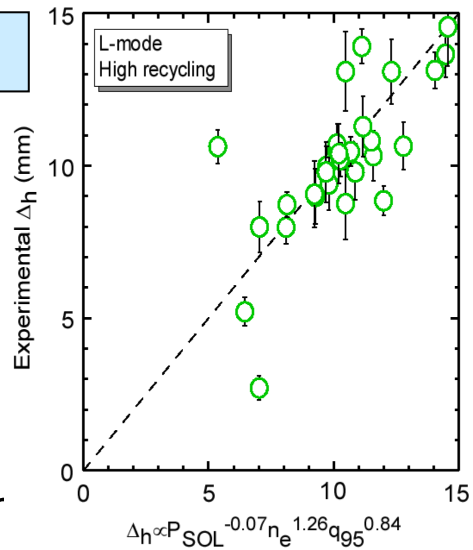
Extrapolate from MAST?





Power load linear with P_{SOL} (justification?)

SOL width



Or take a scaling for the SOL width based or interchange mode turbulence:

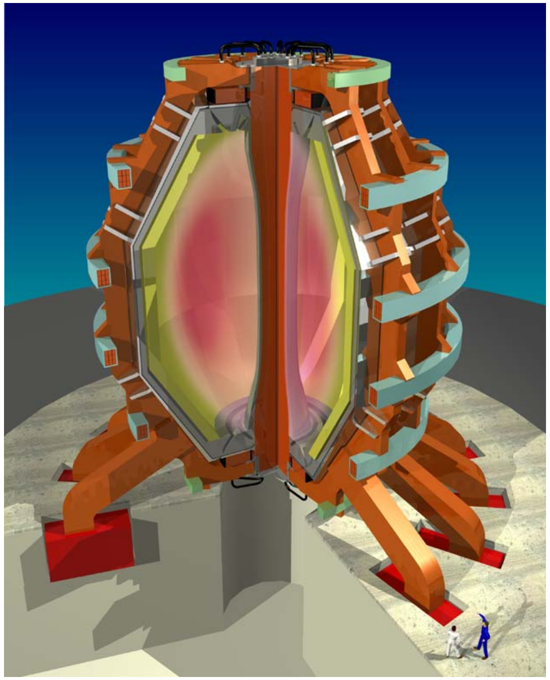
~28mm (3.3mm) SOL width at outboard (inboard) divertor

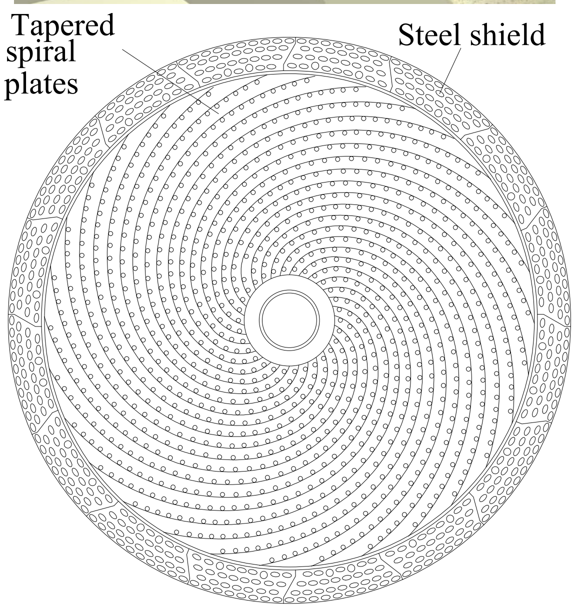
assume 50% radiated power

outboard loading: ~40MWm^{-2} (10^0 angling of divertor plate)

inboard loading: ~26MWm^{-2} (5^0 angling of divertor plate)

Engineering Design





Centre column:

650 tonnes, water-cooled

constructed from 30 tapered copper plates, wrapped around a central tube

thin steel shield provides effective protection

First wall and blanket:

martensitic steel first wall

lithium silicate breeding blanket, with Be multiplier, separated by He-cooled steel plates

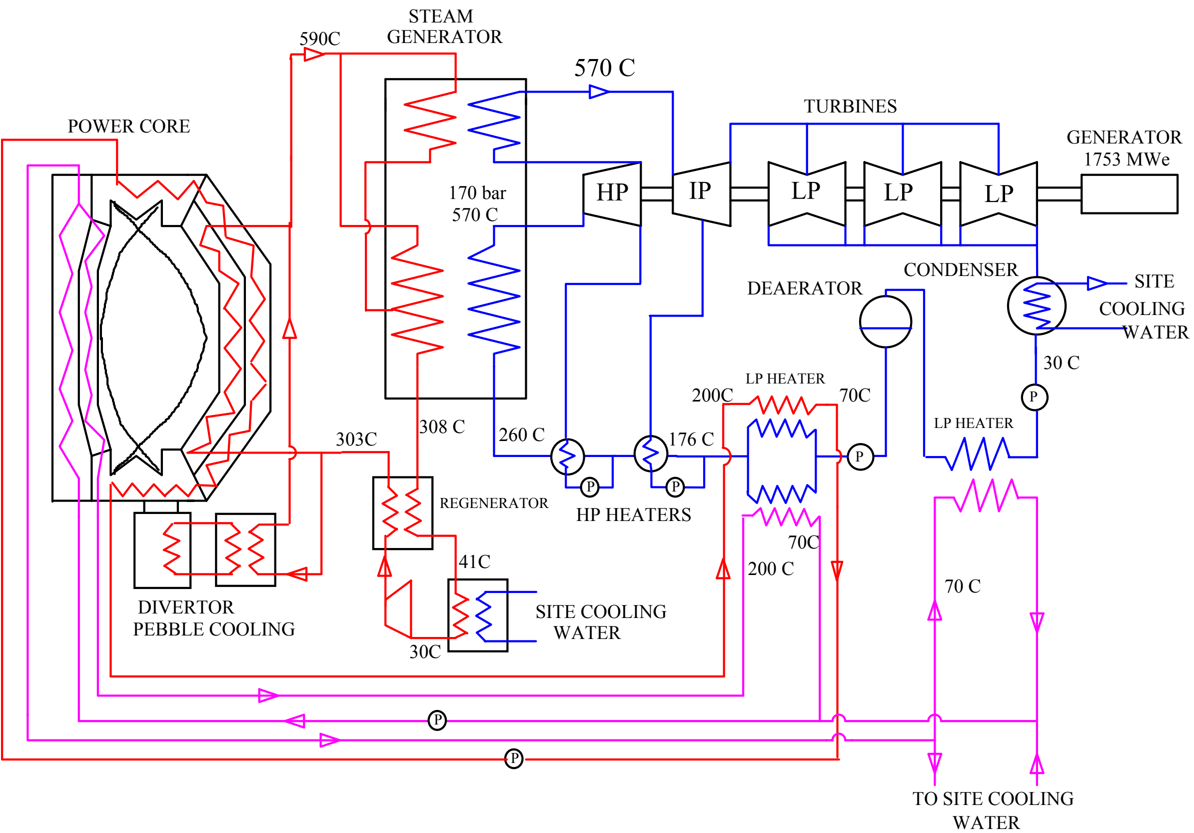
T breeding ratio ~1.1

PF coils:

normal-conducting Cu divertor coil

super-conductor for other 2 pairs (normal conducting Cu an option)

The power cycle



Design of power cycle takes account of wide spectrum of heat sources:

~80% at high T (~600°C)

~20% at low T (70-200°C)

Total fusion thermal input: 3.3GW

Gross electrical output: 1.75GW

30% of the output goes into driving the main electrical subsystems

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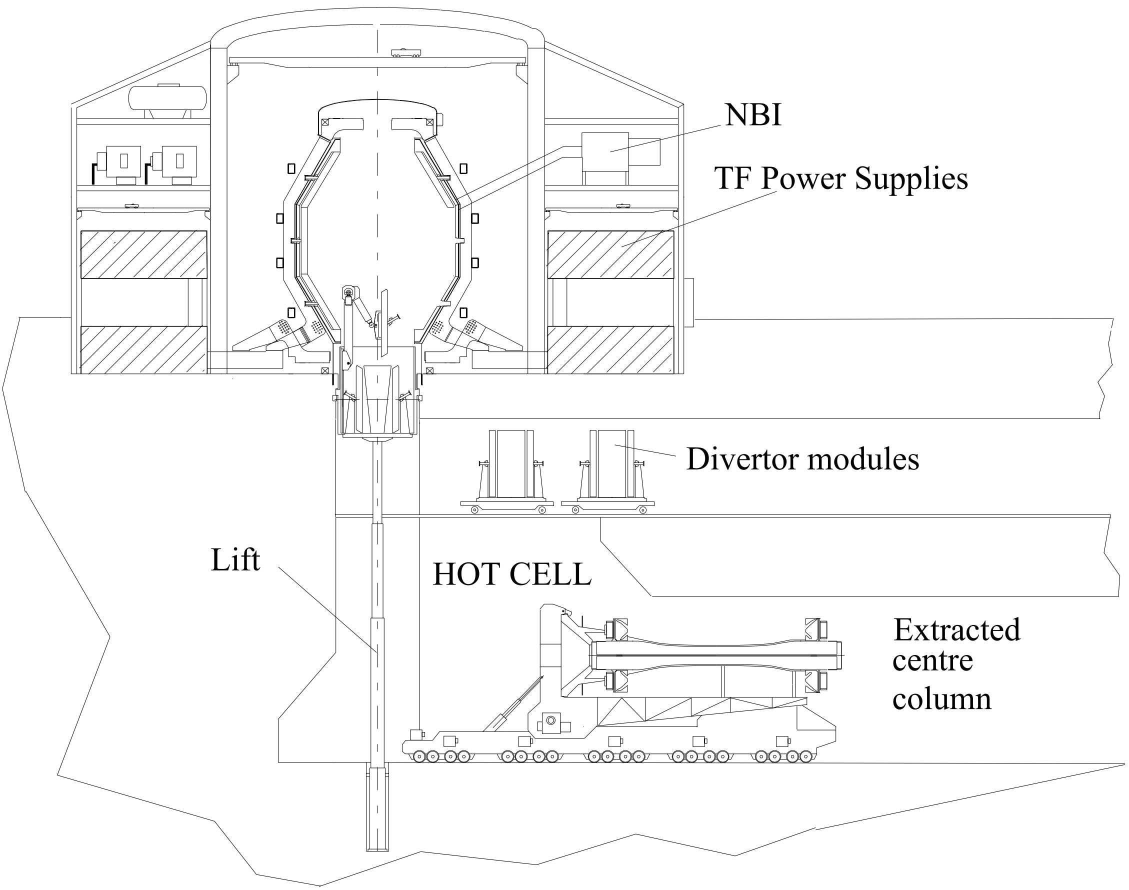
The design has been kept simple to ease maintenance



Maintenance schedule:

centre column replaced and refurbished every 2 yrs

removal of centre column allows easy access to blanket modules

mid-plane modules replaced every 2 yrs (others 4 yrs)

Components Test Facility (CTF)

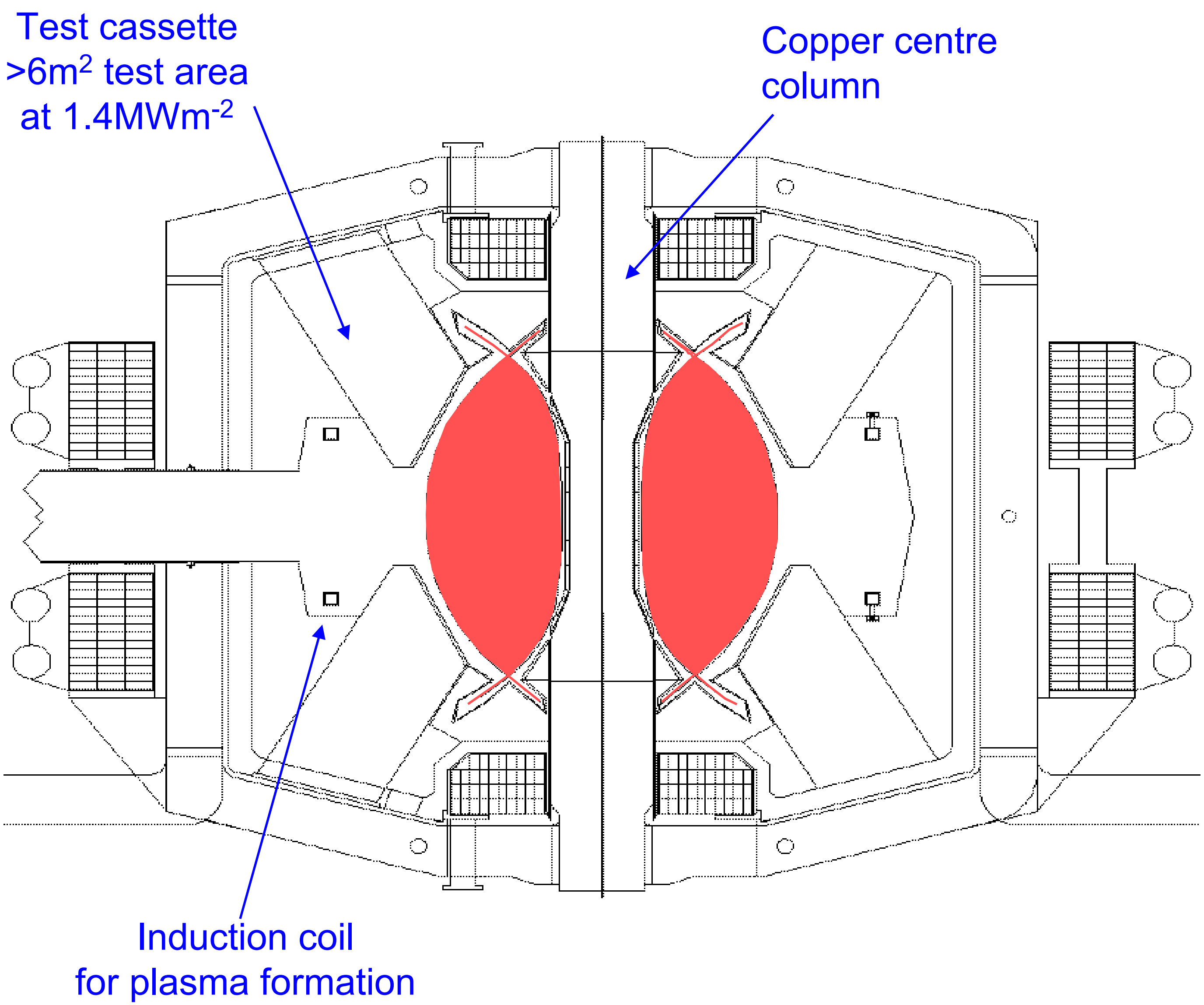
Many in the fusion community feel that there is a need for CTF

- IFMIF could test small material samples
- CTF could test larger scale components, and would complement IFMIF data

The requirements of such a device are:

- To provide sufficient neutron flux with limited T-consumption (no T breeding assumed)
  - Drives one to a compact device: the ST is suitable
- Must operate in steady state (eg ~40% availability)
- Ideally should be available on a nearer term time-scale than the power plant
  - Less aggressive physics assumptions

Components Test Facility design (example)



Theoretical feasibility studies (physics and engineering) have shown that the ST has a role to play in the development of fusion power

Many of the issues are the same as those for ITER, but in addition it is important that MAST and other STs:

confirm the encouraging theoretical predictions:

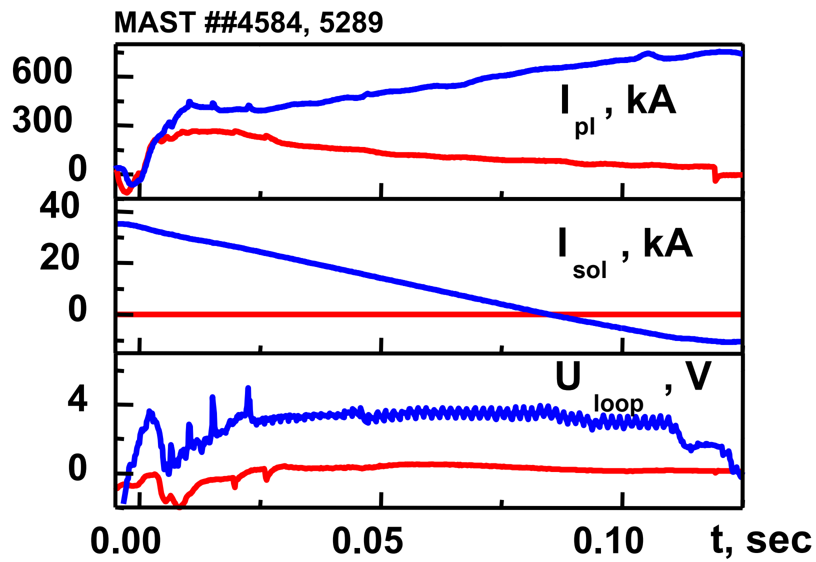
- high β operation
- high elongation, high bootstrap current scenarios
- good confinement
- current drive efficiency (NBI and RF)

improve confidence in areas of uncertainty:

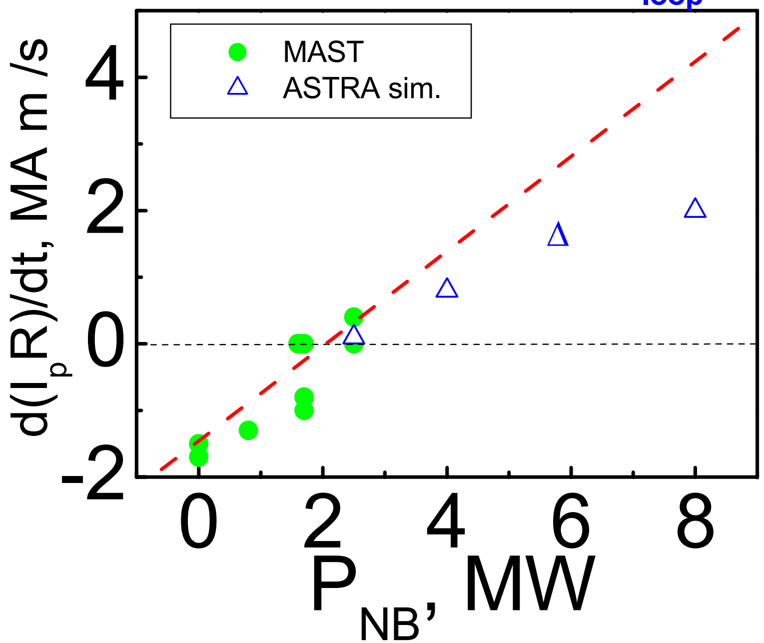
- exhaust and ELMs
- fast particle instabilities
- impact of sawteeth
- neoclassical tearing modes and resistive wall modes
- non-inductive start-up

Non-solenoid start-up and current ramp

Plasma formation without use of solenoid flux demonstrated



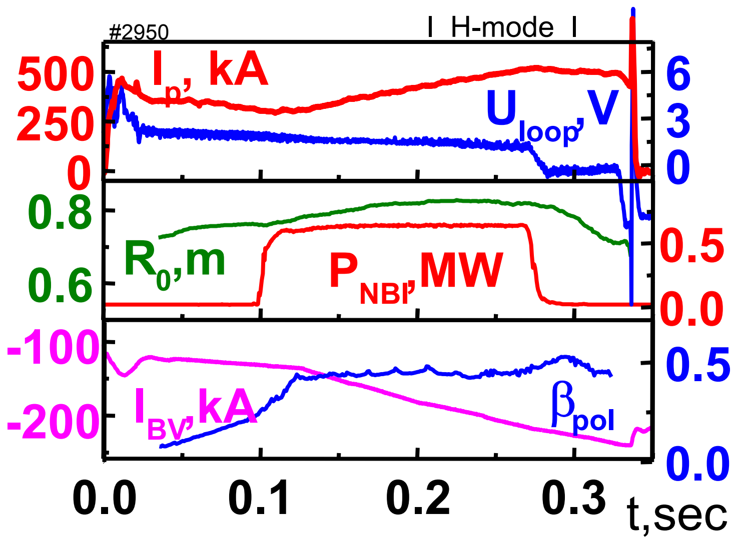
CURRENT RAMP at U_{loop} ~ 0

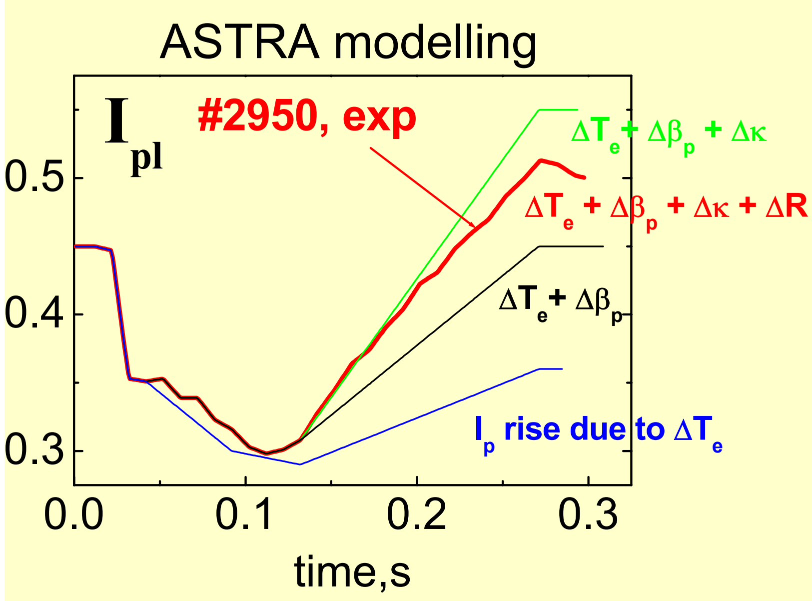


Plasma current can be ramped-up without use of the solenoid flux (ASTRA simulations)

Non-solenoid start-up and current ramp

Plasma current can be ramped-up using only flux from BV coils during NB heating and sustained for τ > resistive time



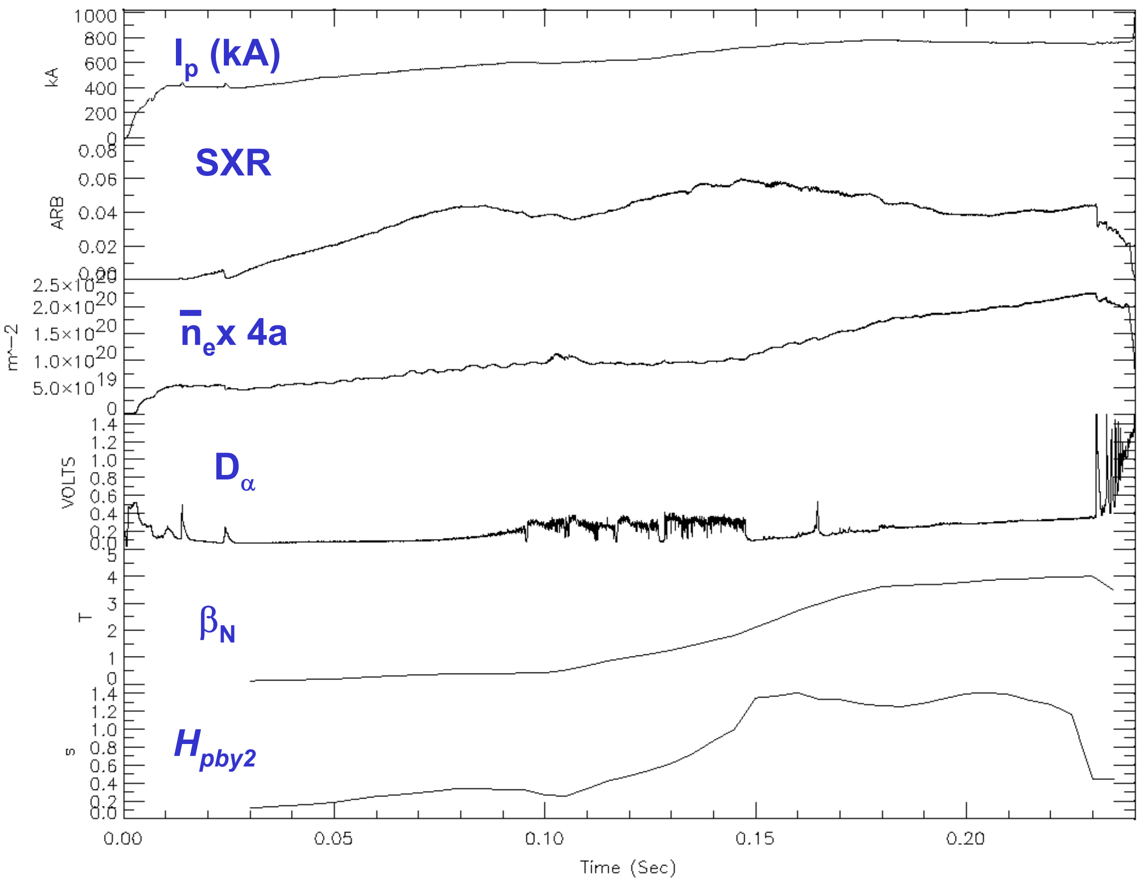


Plasma current doubled at constant U_{loop} during NBH
only 20% assigned to change in resistivity

Plasma current sustained at zero U_{loop} for 0.2s, which is ~ resistive time

Contribution to the baseline ST scenario

MAST #6326, HH98y2 ~ 1.4, β_N ~ 4, G ~ 0.6



High-beta ELM-free H-mode with q(0) > 1

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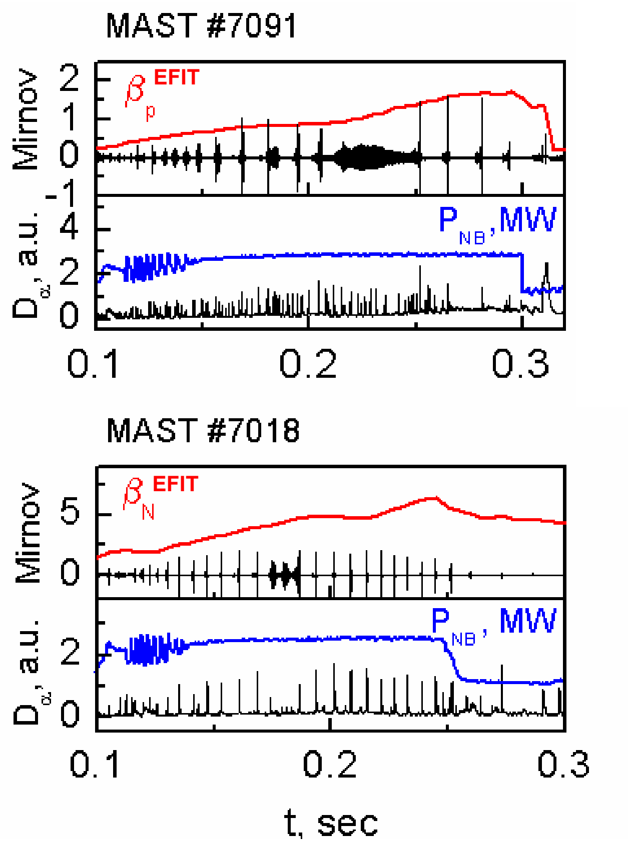
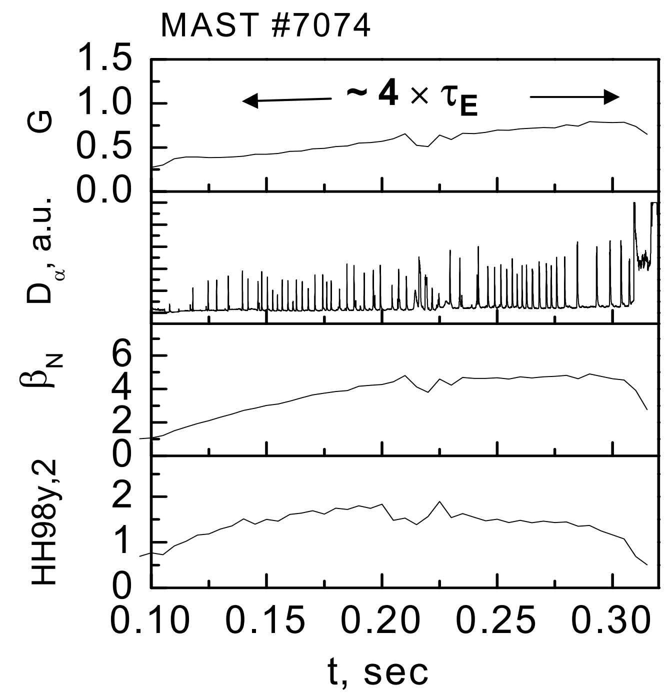
MAST

High beta discharges have low MHD activity:



MHD activity is low at high beta for broader (**H-mode**) profiles

High beta sustained for several confinement times:



High-beta ELMy H-mode with q(0) > 1

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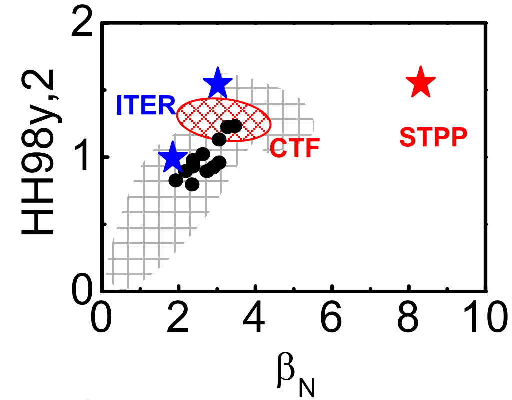
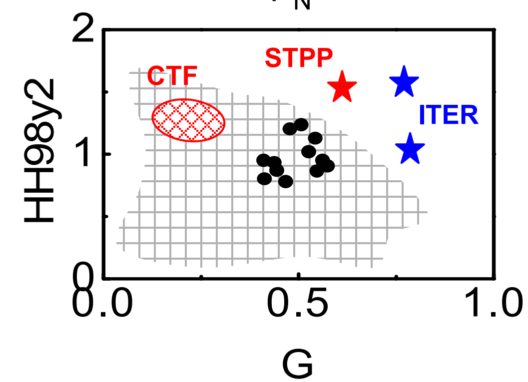
Contribution to the baseline ST scenario

CTF ops point	Achieved	simultaneous?
A = 1.6	A = 1.35 -1.6	
$\kappa \sim 2.5$	$\kappa \sim 2.45$	not with A
$l_i \sim 0.54$	$l_i \sim 0.5$	not with β_N
$\beta_N \sim 4.0$	$\beta_N > 5$	not with l_i
$I_p/I_{rod} \sim 0.67$	$I_p/I_{rod} \sim 0.84$	
HHpby2 ~ 1.26	HHpby2 >1.5	
$I_{non\text{-}ind}/I_p \sim 30\%$	$I_{non\text{-}ind}/I_p \sim 50\%$	not with A, κ, I_p/I_{rod}
G = 0.27	G > 1.5	
$\tau_{He}/\tau_E \sim 6 - 10$	?	check





Many of parameters required for ST Component Test Facility have been achieved simultaneously

However, access to operating point of the ST Power Plant is a challenge for future experiments

MAST operating space with **future ST** and **ITER** parameters (**dots** - kinetically validated data with low FP component and $-0.05 \leq (dW/dt)/P \leq 0.35$)

H Wilson, M Valovic

Experiments needed to advance CTF operation point studies:

CTF outstanding issues:

- OPS at $\mathbf{I_p/I_{rod} \sim 0.7}$ with high β_N and κ, low l_i and at **A = 1.6**
- Transport studies at high I_p/I_{rod} and high β (i.e. I_p>1MA, T_e>1keV)
- Particle transport, ash removal
- Confirm NBI CD, explore RF alternatives (eg EBW)

Other Next Step ST-relevant studies

- More high β studies, **limits,** low l_i
- Plasma control at high elongation
- B_V ramp studies and **overdrive** demonstration
- Non-solenoid **start-up** (including ECRH/**EBW** start-up) and integrated non-solenoid scenario

Conclusions

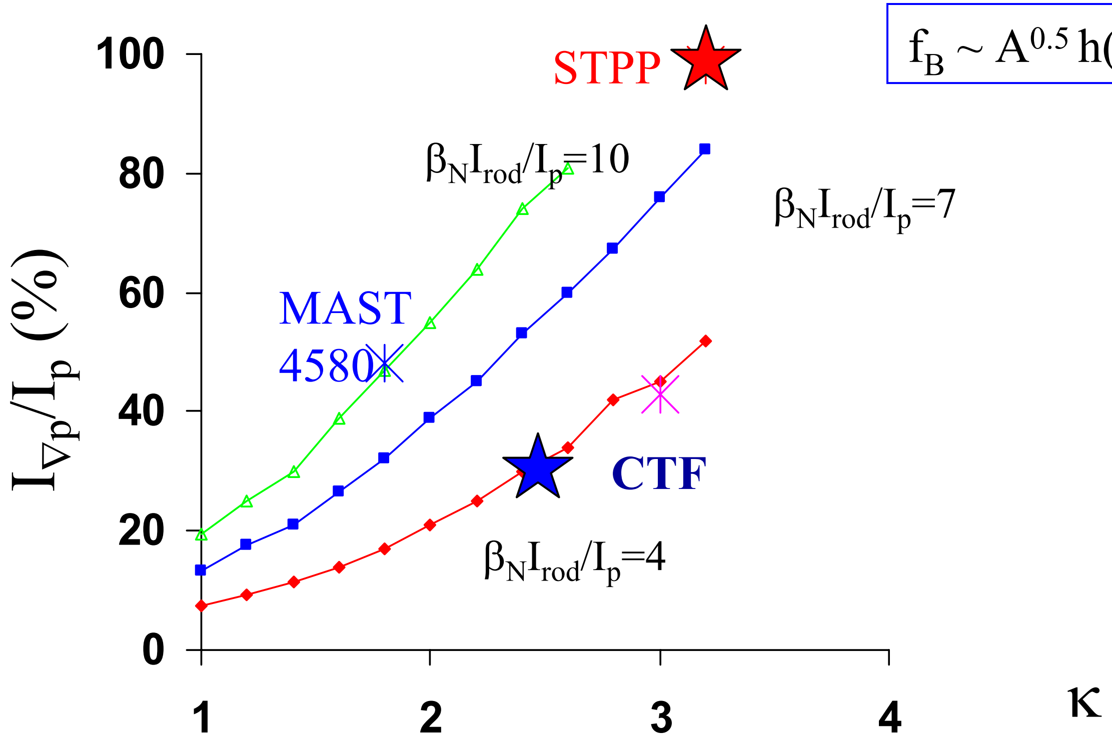
Theoretical feasibility studies (physics and engineering) have shown that the ST has a role to play in the development of fusion power

Considerable advances in areas relevant to the physics basis for operations in next-step STs *(start-up, current ramp, stability, confinement, current sustainment and exhaust issues)*

Together with the extensive array of high quality diagnostics on MAST, these results provide an excellent platform for further input to key physics studies and issues of specific relevance to the viability of the ST concept

High pressure-driven currents in STs

$f_B \sim A^{0.5}\, h(\kappa)\, \beta_N\, q_{circ}$



MAST #4580 has A =1.33, I_p = 0.675MA, I_{rod} = 1.87MA, and β_N= 4.5

CTF has A =1.6, I_p=8MA, I_{rod}=12MA, and β_N=4.0

ST power plant has A =1.4, I_p= 31MA, I_{rod}= 30.2MA, and β_N= 8.2