The Development of the Spherical Tokamak

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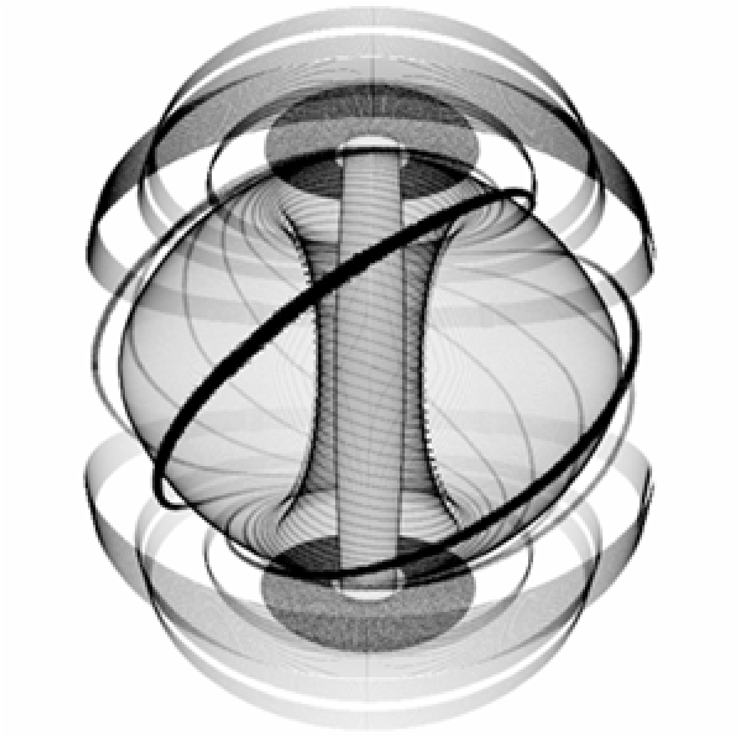


This work was funded jointly by the United Kingdom Engineering and Physical Sciences Research Council and by EURATOM.

OUTLINE OF TALK

Development of magnetic containment devices:
mirror → pinch → tokamak

What is a Spherical Tokamak / Torus (ST)?

History of the ST

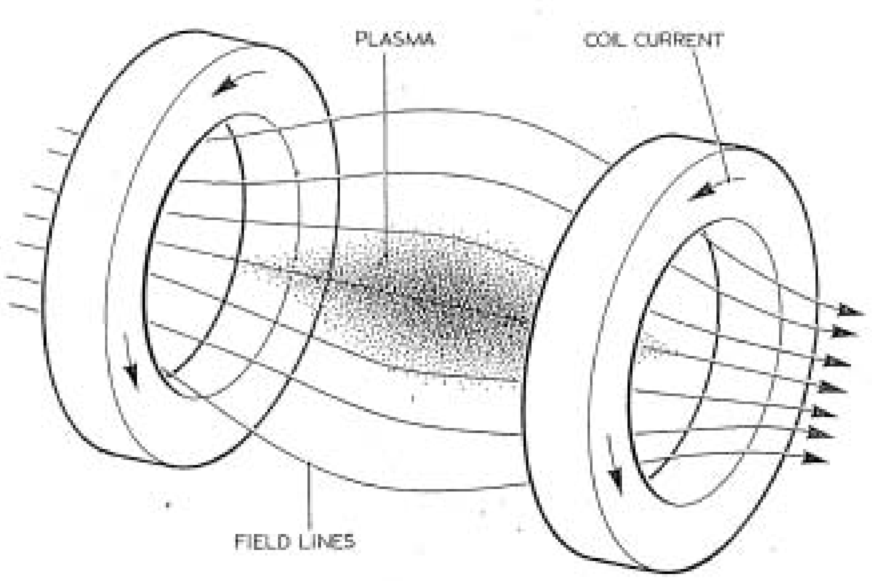
Present world STs

Some properties of the ST

Plans for future – and fusion! STs

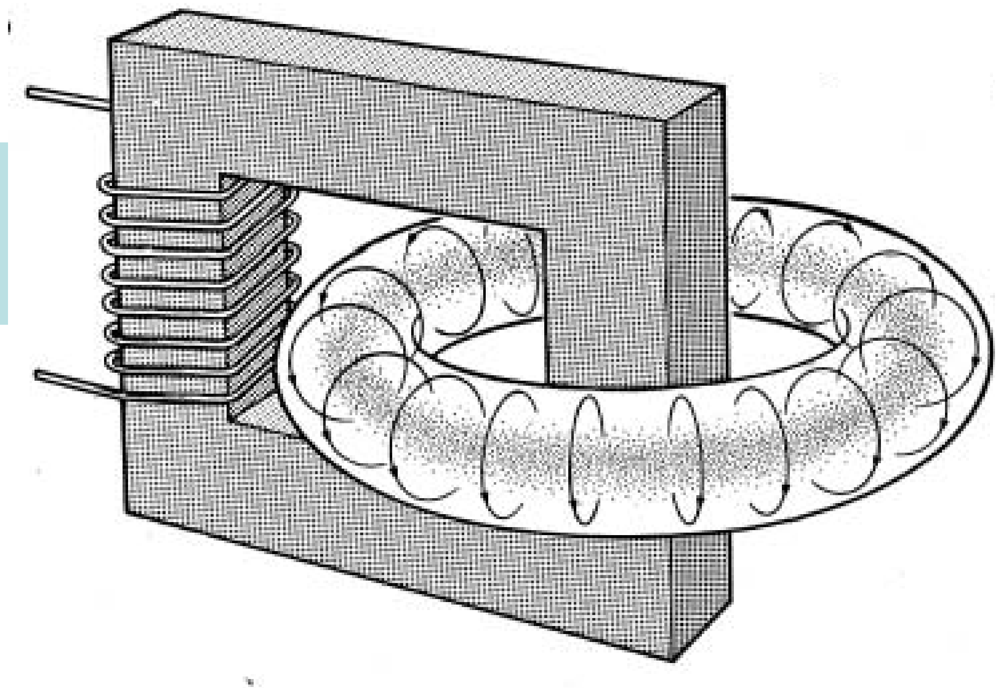
Acknowledgements: many ST enthusiasts world-wide, in particular: Mikhail Gryaznevich, Jon Menard, Martin Peng, Yuichi Takase, Andrew Kirk – and many colleagues at UKAEA Culham

It was first hoped that a Simple Magnetic Mirror would contain a plasma -



- but some plasma escapes from the ends

Hence the toroidal pinch:



Thompson, Blackman patent 1946

$R / a = 1.30m / 0.3m$

$I_p = 0.5MA$

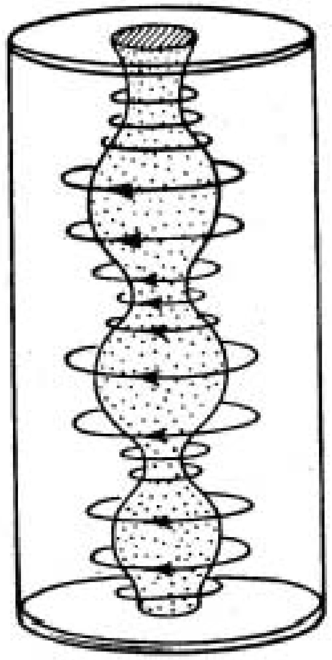
$\tau = 65s$

$T_i = 500keV$

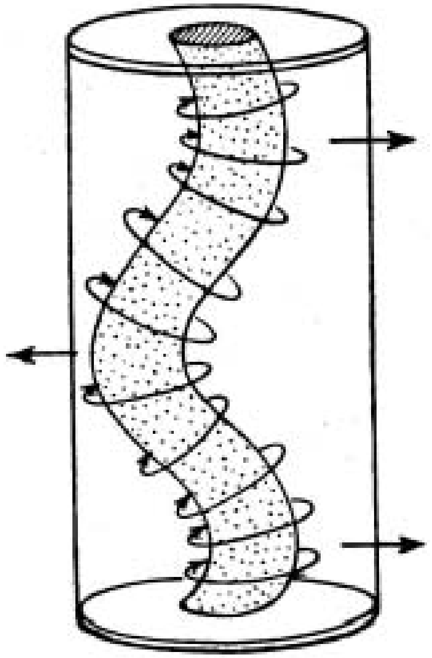
Fuel: D-D

Toroidal Pinch Studies - 1940’s and 1950’s

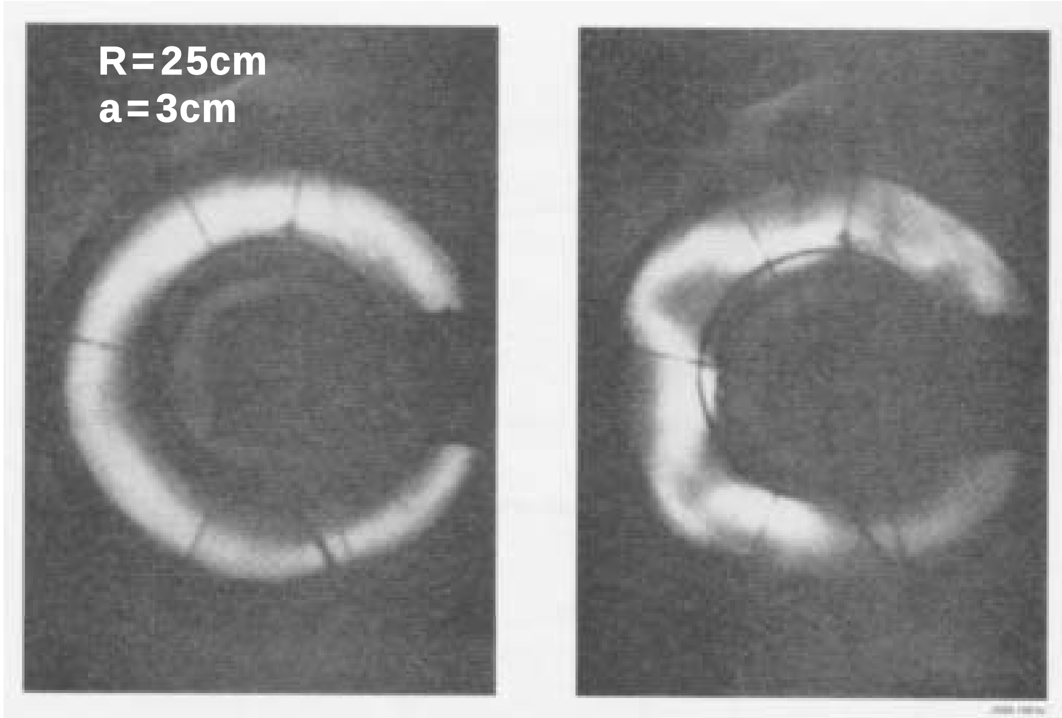
Alan Ware, Stanley Cousins at Imperial College & Aldermaston



Sausage Instability



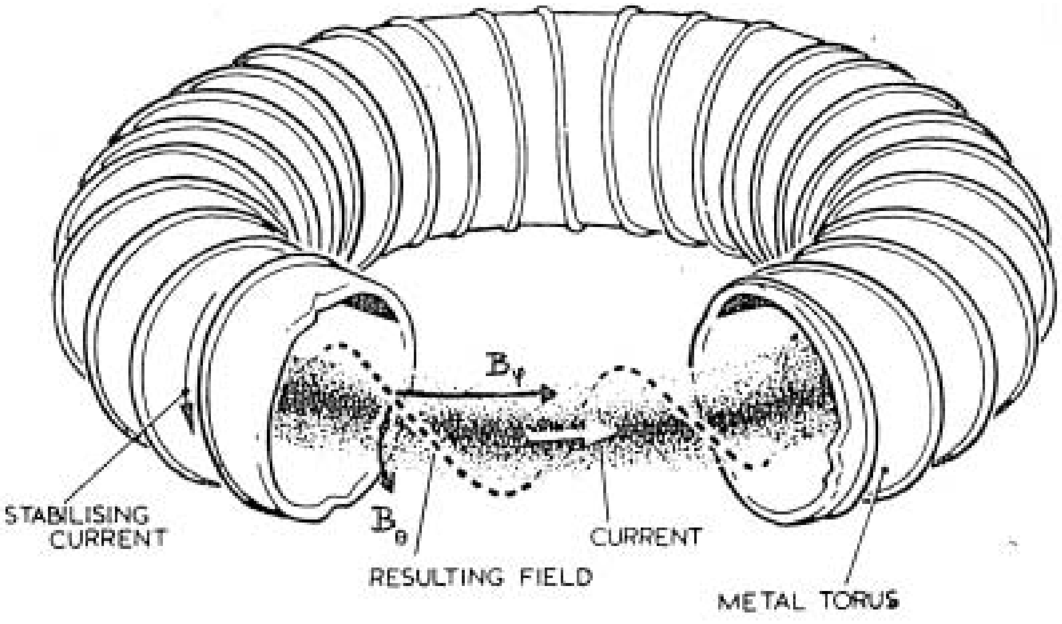
Kink instability



20 micro-sec

First observations of the KINK INSTABILITY

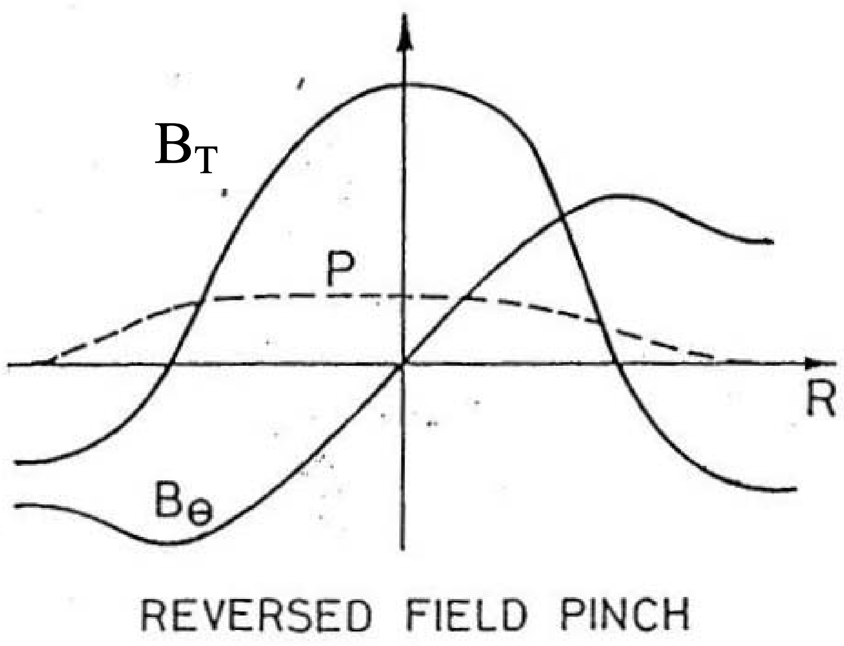
ZETA at Harwell - 1954-60s

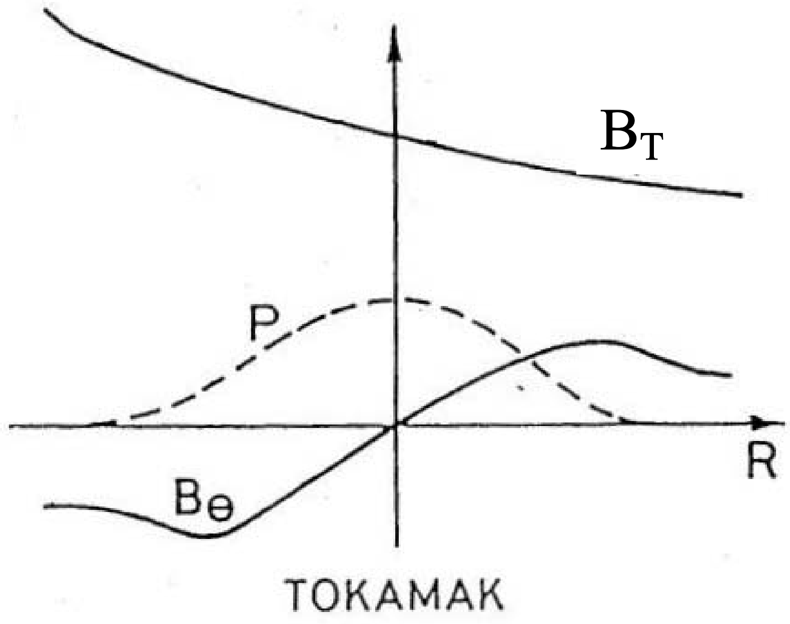


1954-1958 :
R / a = 1.5m / 0.48m
τ ~ 1ms
T_e = 0.15keV

Two types of toroidal pinch:

Addition of a small toroidal field in Zeta had improved stability. Sakharov & Tamm began magnetic confinement studies in 1950 [1] and suggested use of a much stronger toroidal field:





$|\mathbf{B}_T| \approx |\mathbf{B}_\theta|$

$|\mathbf{B}_T| \gg |\mathbf{B}_\theta|$

[1] Golovin & Shafranov PIroda 8 (1990) p25

THE TOKAMAK

toroidalnaya **ka**mera and **ma**gnitnaya **k**atushka

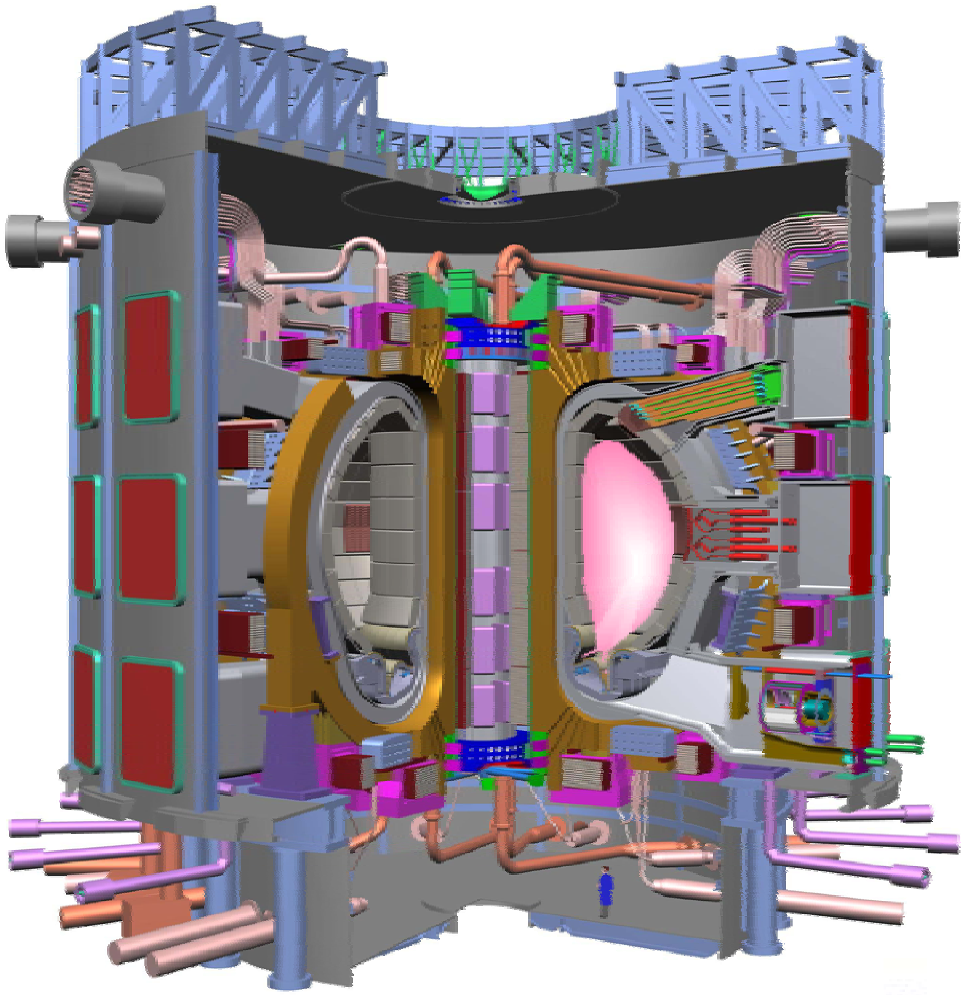
("Toroidal Chamber Magnetic Coils", in Russian!)

The first Tokamak was built in the late 1950's at the Kurchatov in Moscow. Claimed to be much hotter than pinches or other devices studied in the Western world.

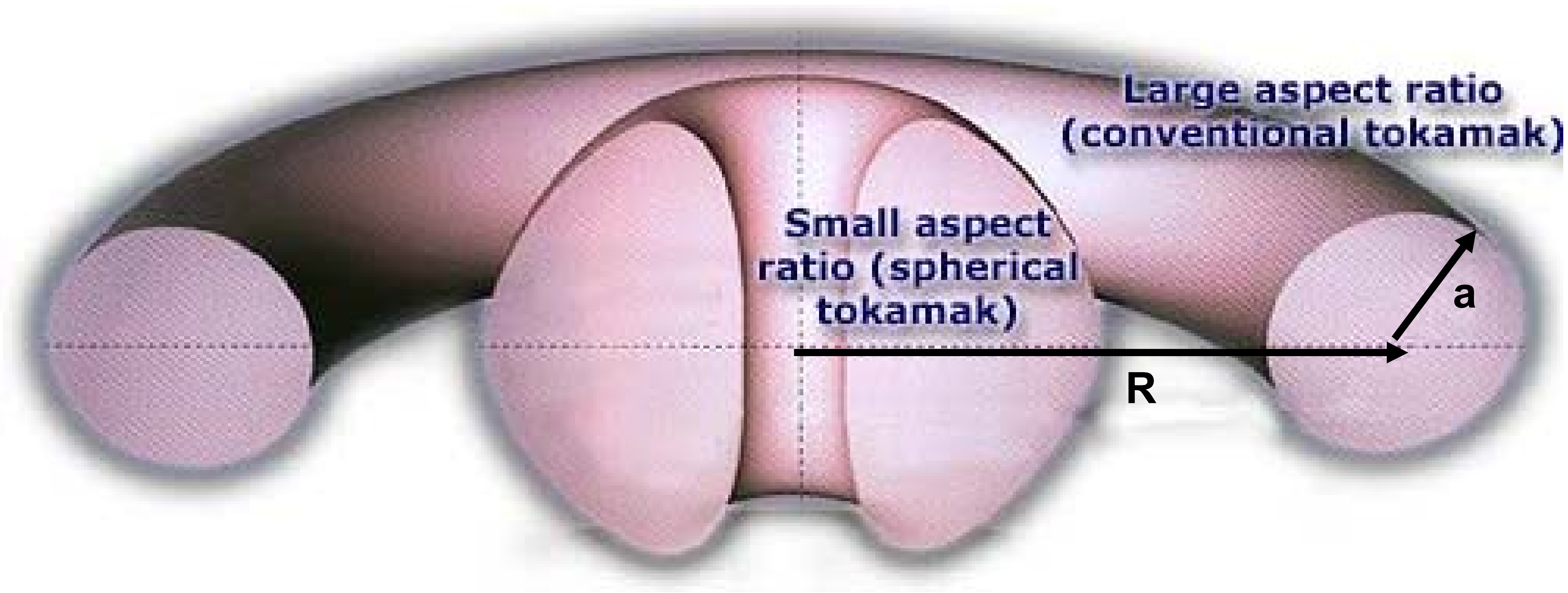
A team of Culham scientists spent a year in Russia, proving this was indeed the case on T3, using Thomson Scattering diagnostic.

The rest of the World began building tokamaks!

- Leading to ITER



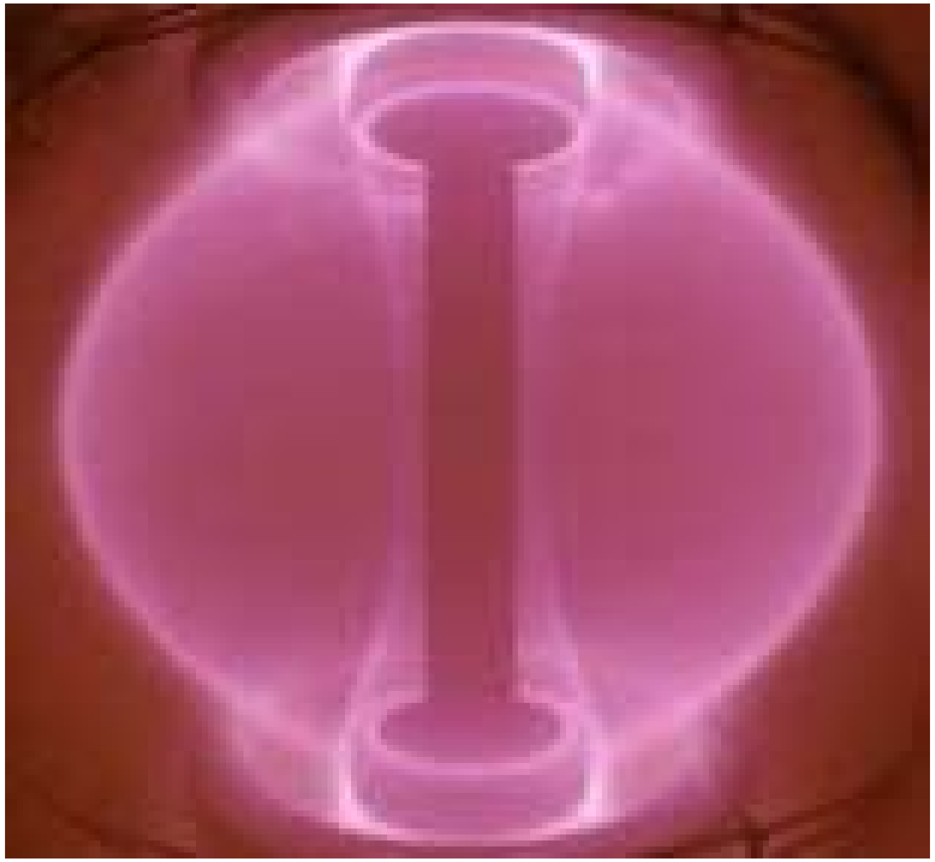
Can we improve the tokamak?



Aspect Ratio = Major radius / minor radius

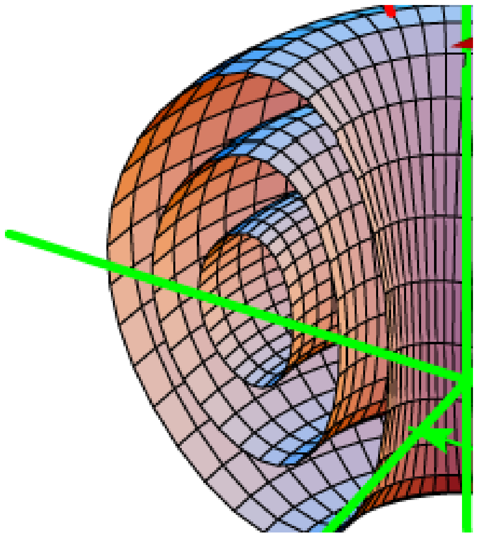
$$A = R / a$$

What is a Spherical Torus / Tokamak (ST)?



START

(1990-1998)



ST is a tokamak with a low aspect ratio (R/a ~ 1.1 - 2)
‘Spherical’ in appearance - but still topologically a Tokamak!

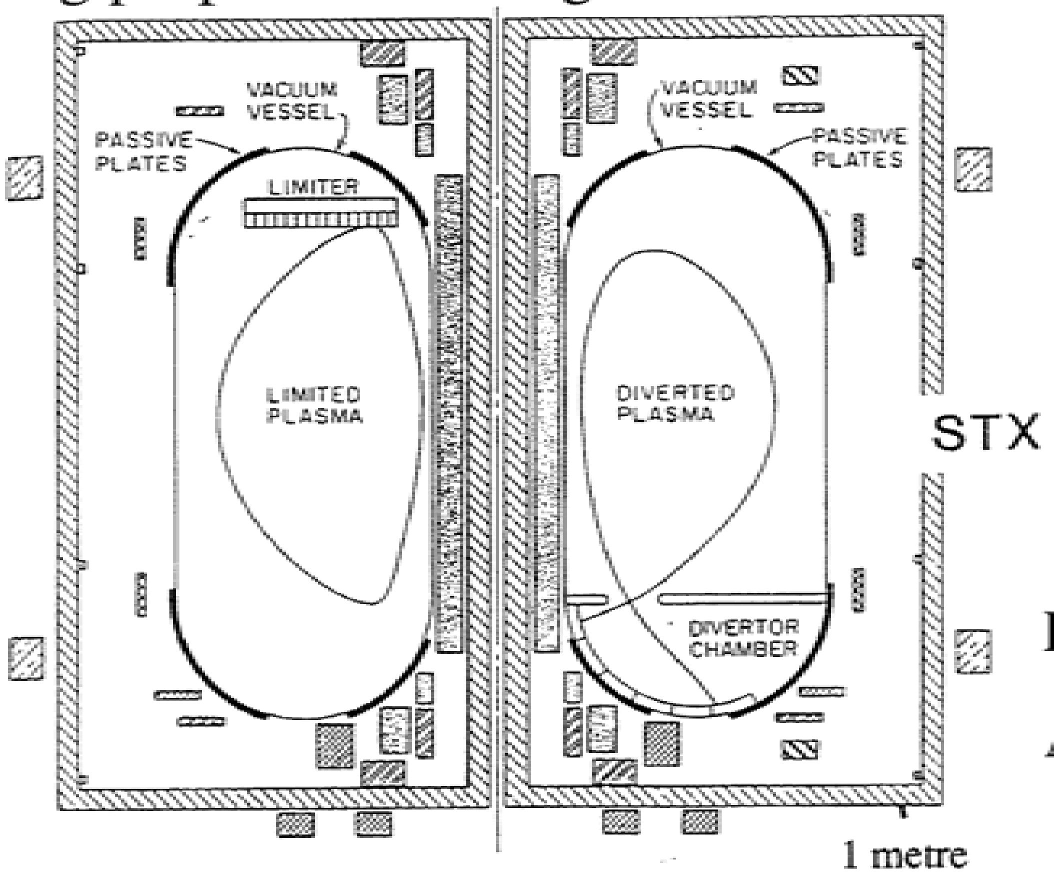
An ST exhibits tokamak properties – however
SOME TOKAMAK FEATURES BECOME HIGHLY EMPHASISED

History of the Spherical Tokamak (1)

1982 – 7;

- Realisation that low-A provided high-β (Sykes, Turner, Patel, Aachen EPS 1982, APS Review paper 1985)
- Talk by Martin Peng at Culham in 1984
- Peng & Strickler published a summary of the physics of low A [1]
- Robinson advocated low-A [2] as a means of combining RFP efficiency with the stability of a tokamak

•Peng proposed building STX at ORNL



Estimated cost \$6M

Aspect ratio A ~ 1.67

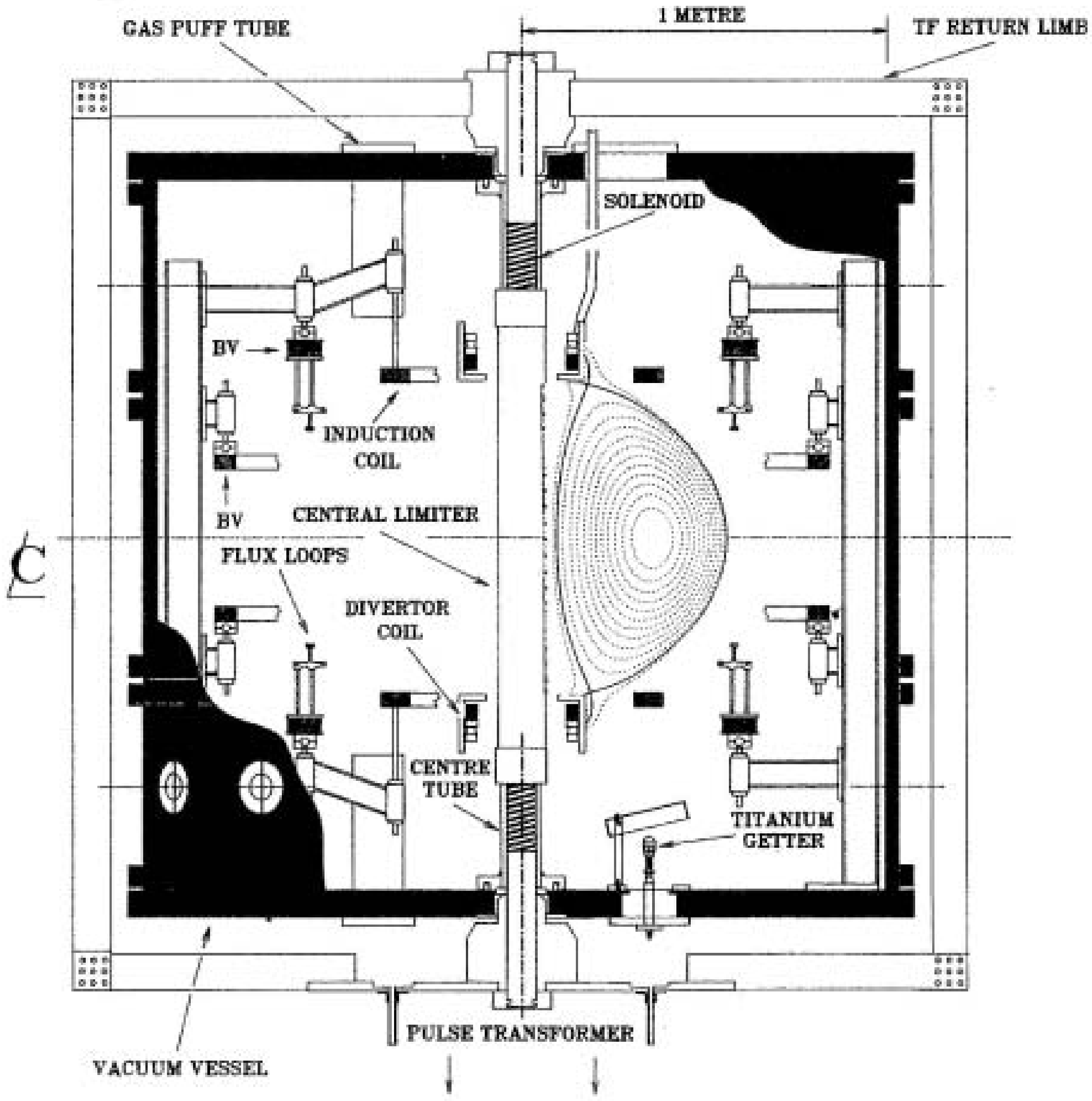
[1] Y-K M Peng & D Strickler, NF 26 769 (1986)]

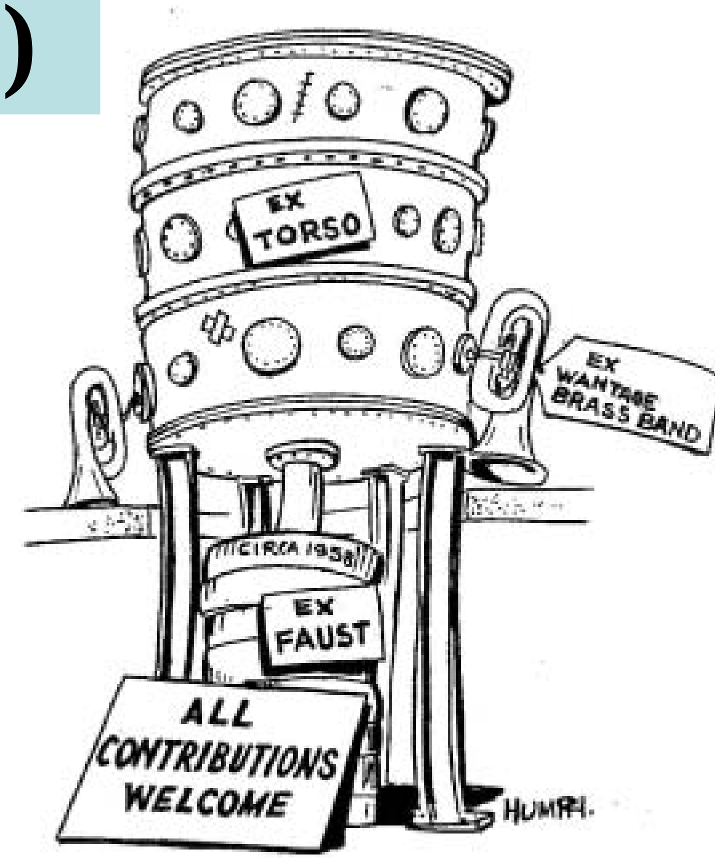
[2] D C Robinson, in Fusion Energy & Plasma Physics, World Scientific Press p601 (1987)

History of the Spherical Tokamak (2)

1988 - 91

- STX abandoned
- Robinson & Todd design low-budget START experiment at Culham



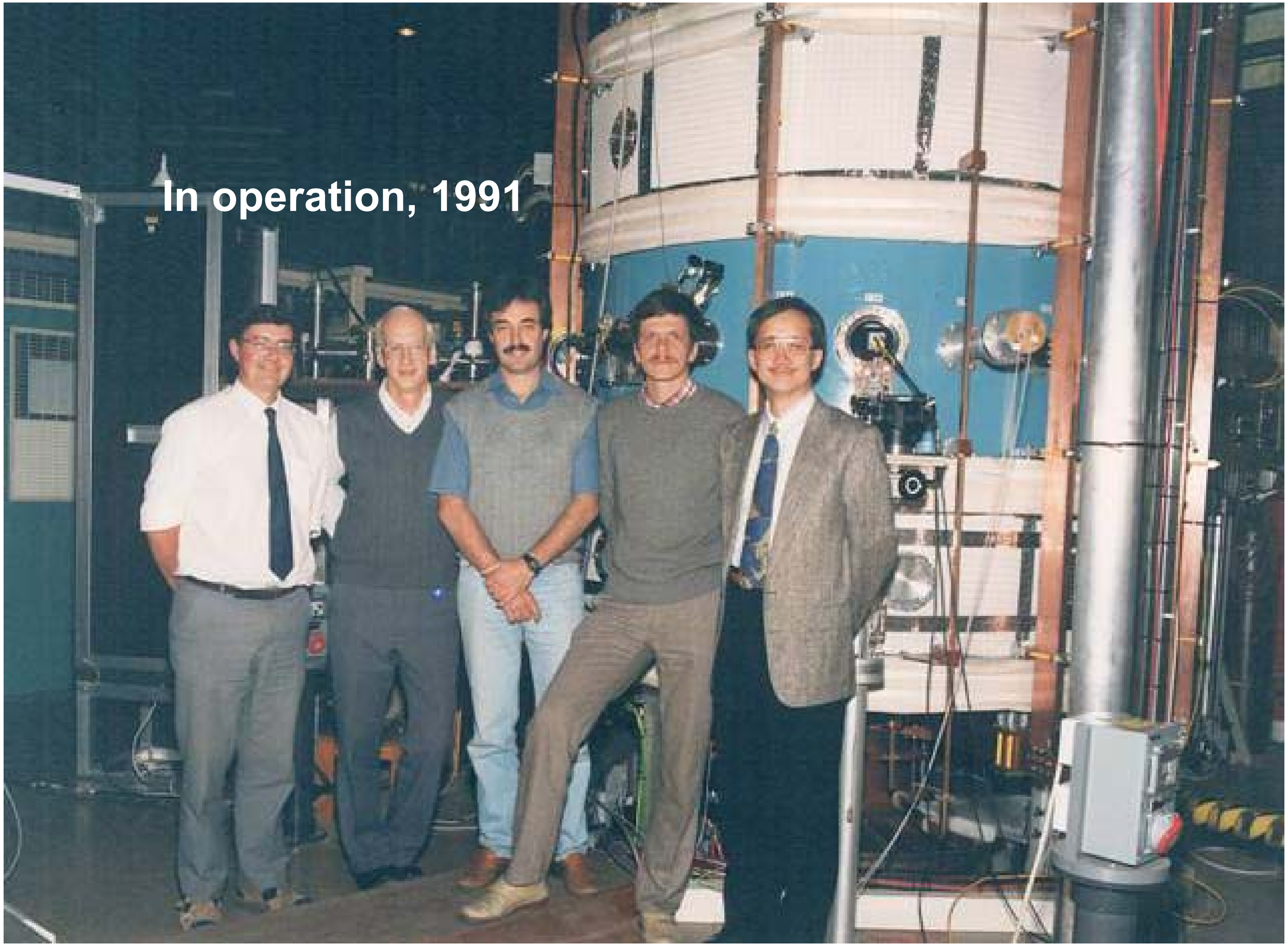


First plasma in START, Jan 1991

Initial build cost ~ £0.1M;

Aspect ratio A ~ 1.25

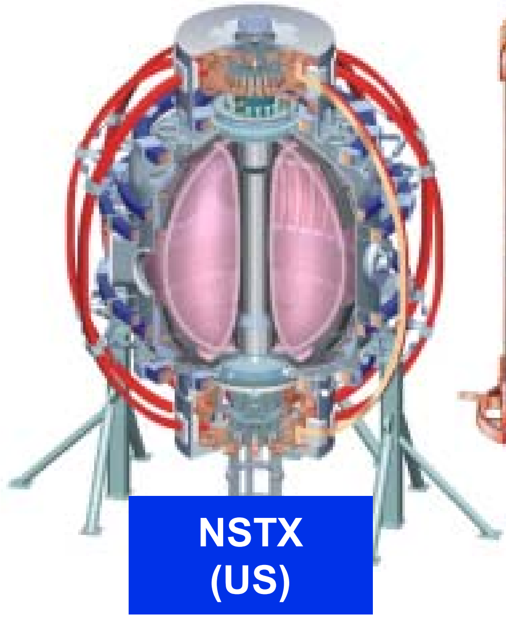
START, UKAEA (1990-1998)

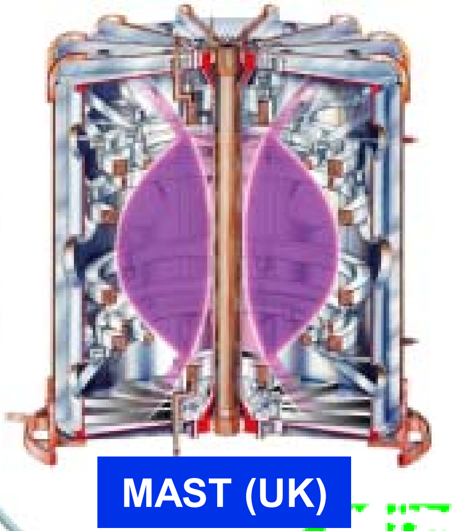


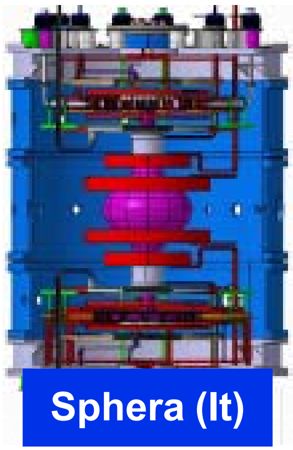
Alan Sykes   Dick Colchin   Edson Del Bosco   Mikhail Gryaznevich   Martin Peng

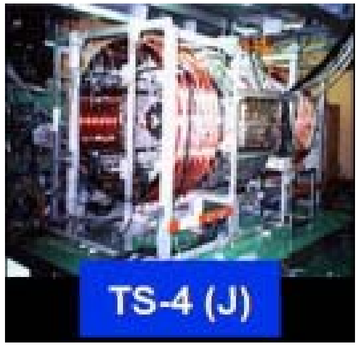
World ST Program is growing !

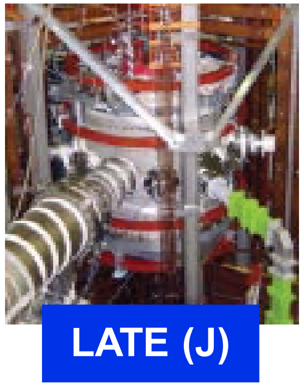
Slide courtesy Y-K M Peng

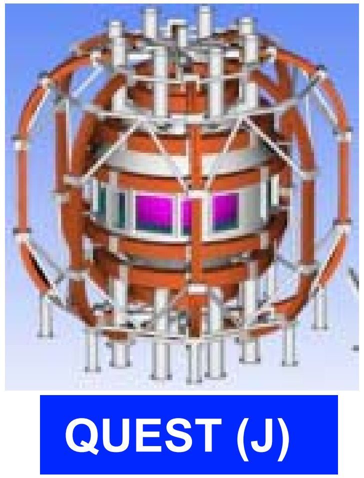






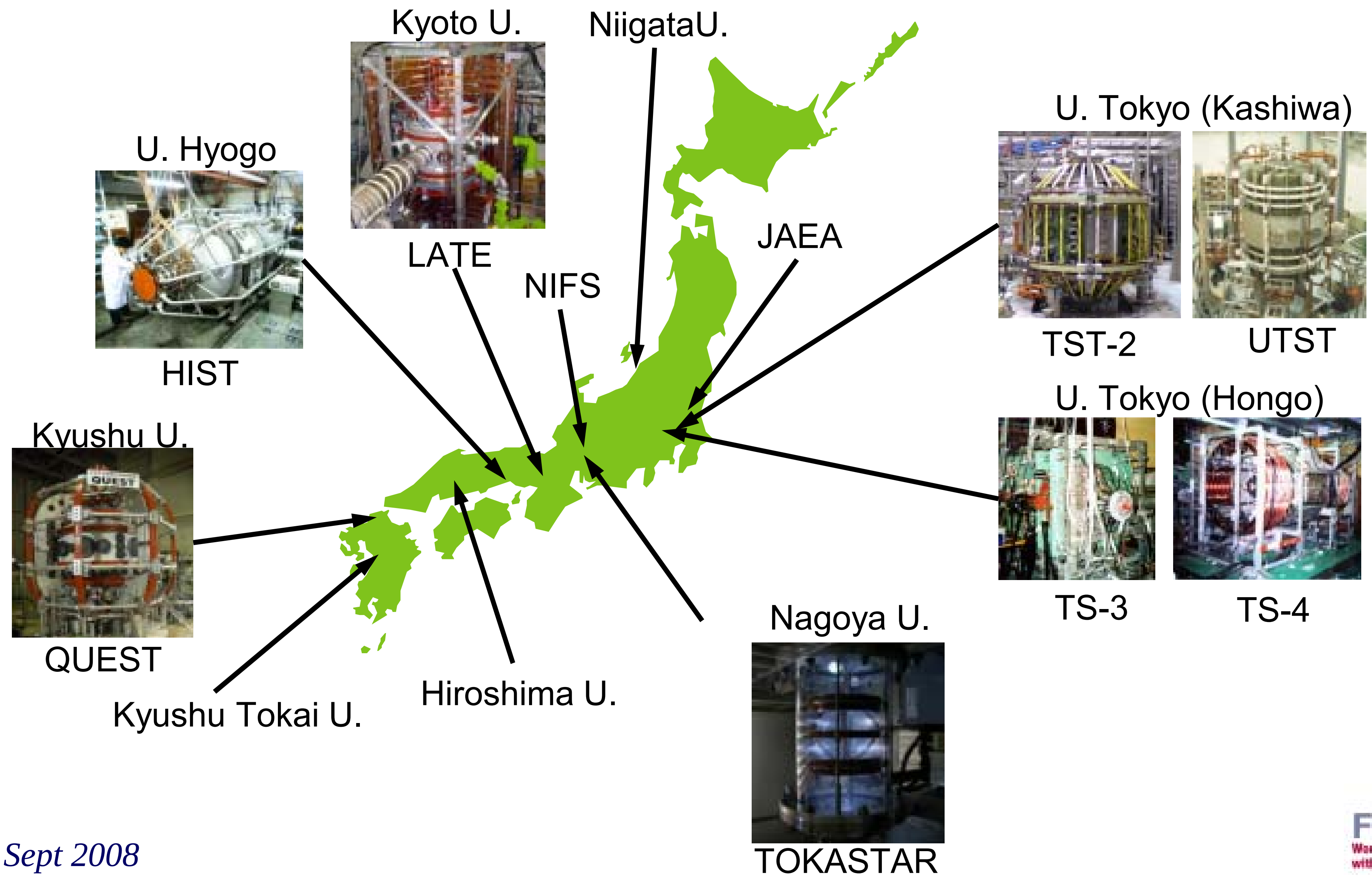






ST Experiments in Japan

Tokyo: ultra-high beta, plasma merging, current start-up, RF heating & current drive
Kyoto: current start-up, RF current drive
Hyogo: coaxial helicity injection, advanced fuelling
Kyushu: steady-state, plasma-wall interaction
Niigata, NIFS, JAEA: ST equilibrium and stability, 3-D MHD Simulation, reactor study



FBX-II

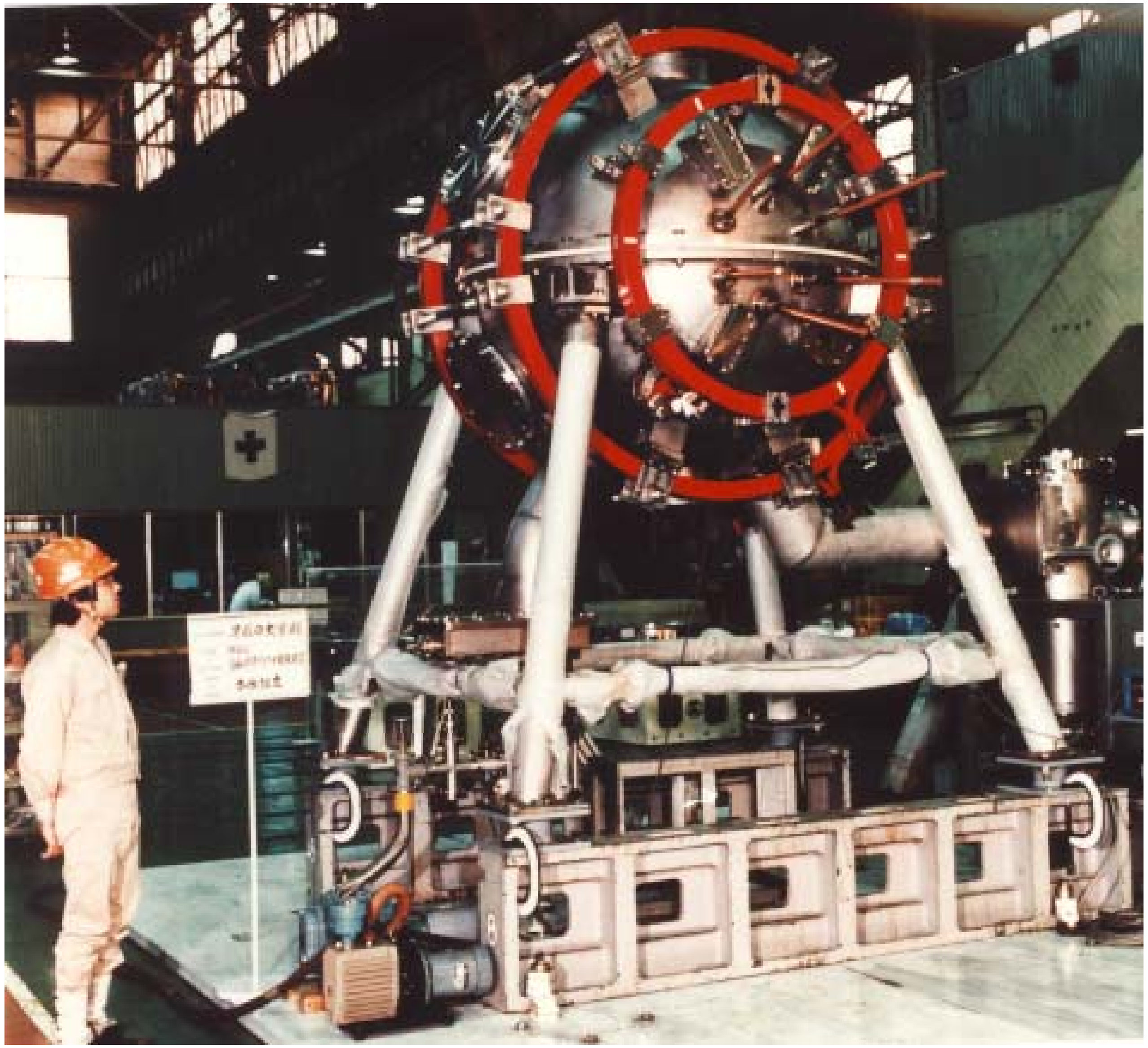
Prof Masaru Irie at Waseda University, Japan

First plasma May 1984

ST operation 1994:

Ro=0.47, a=0.33, A=1.4,

Ip ~ 100kA, B_{To} ~ 0.5T

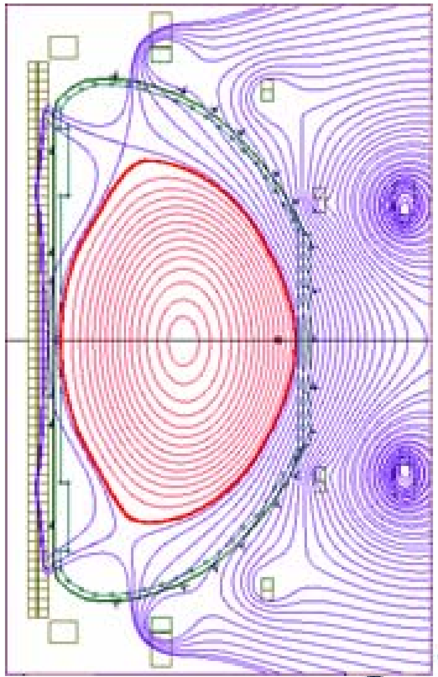


GLOBUS-M, St Petersburg

First plasma: November 2000.

	design	(achieved)
Major radius R (m)	0.36	(0.37)
Minor radius a (m)	0.24	(0.24)
Elongation	2.2	(1.8)
Aspect ratio (R/a)	1.5	(1.5)
Plasma current (MA)	0.5	(0.35)
Toroidal field at R (T)	0.6	(0.35)
Aux. Heating, MW	1	(1.2)
Pulse length (s)	0.3	(0.3)

Key research objectives:
- **Confinement studies**
- **NBH, ICRH, EBW, LH studies**
- **Diagnostics development**
- **Start-up using plasma gun**



A New ST for Studying Steady State Operation

Kyushu Univ. *QUEST*

Research Objectives:

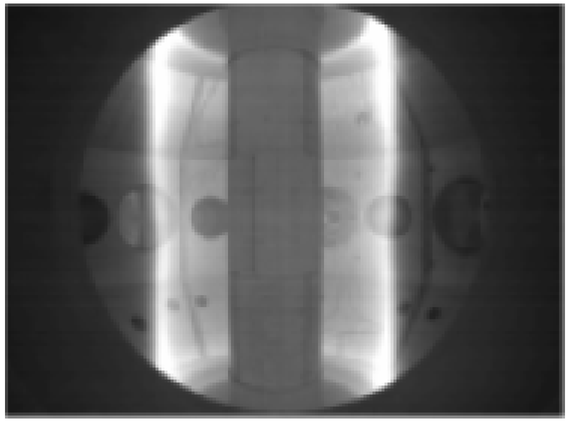
(1) Steady State Operation (SSO) of ST
>> L/R-time (RF driven)

(2) Physics and Engineering and PWI studies of SSO

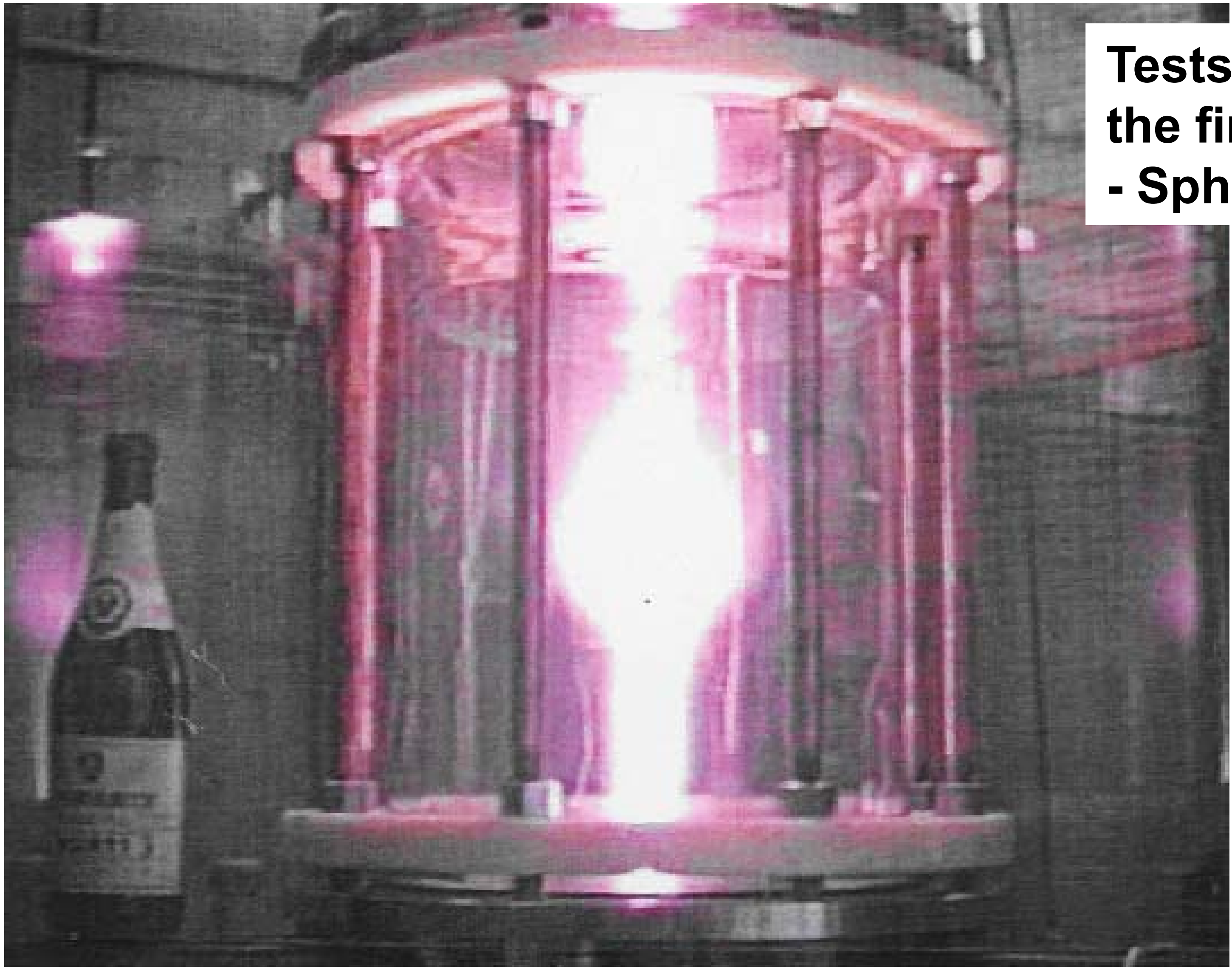
QUEST construction completed
March 2008

$R / a = 0.68 / 0.40$m
$B_t = 0.25$T (0.5T short pulse)
$I_p = 0.1$MA (0.3MA short pulse)

QUEST 1st plasma
(ECH produced)
June 2008



Can we omit the solid central column?



Tests on proto-pinch – the first stage of Proto - Sphera at Frascati

START vessel (for use by Proto-Sphera) at Frascati with new proprietor Franco Alladio

Small-scale STs - MEDUSA

Built by Ray Fonck and students at Madison

(MEDUSA= Madison **EDU**cational **S**mall **A**spect tokamak**)**

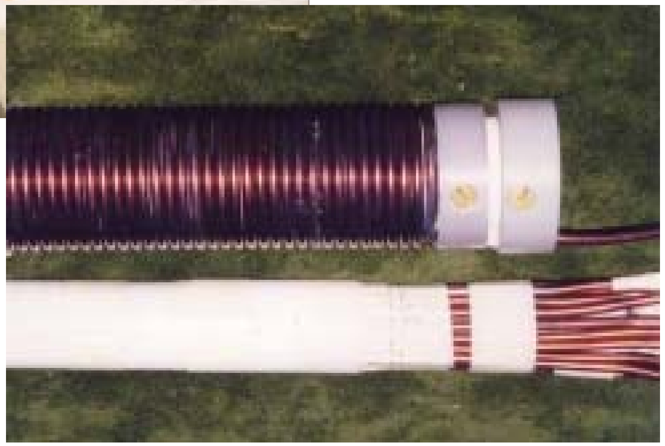
Operational 1993

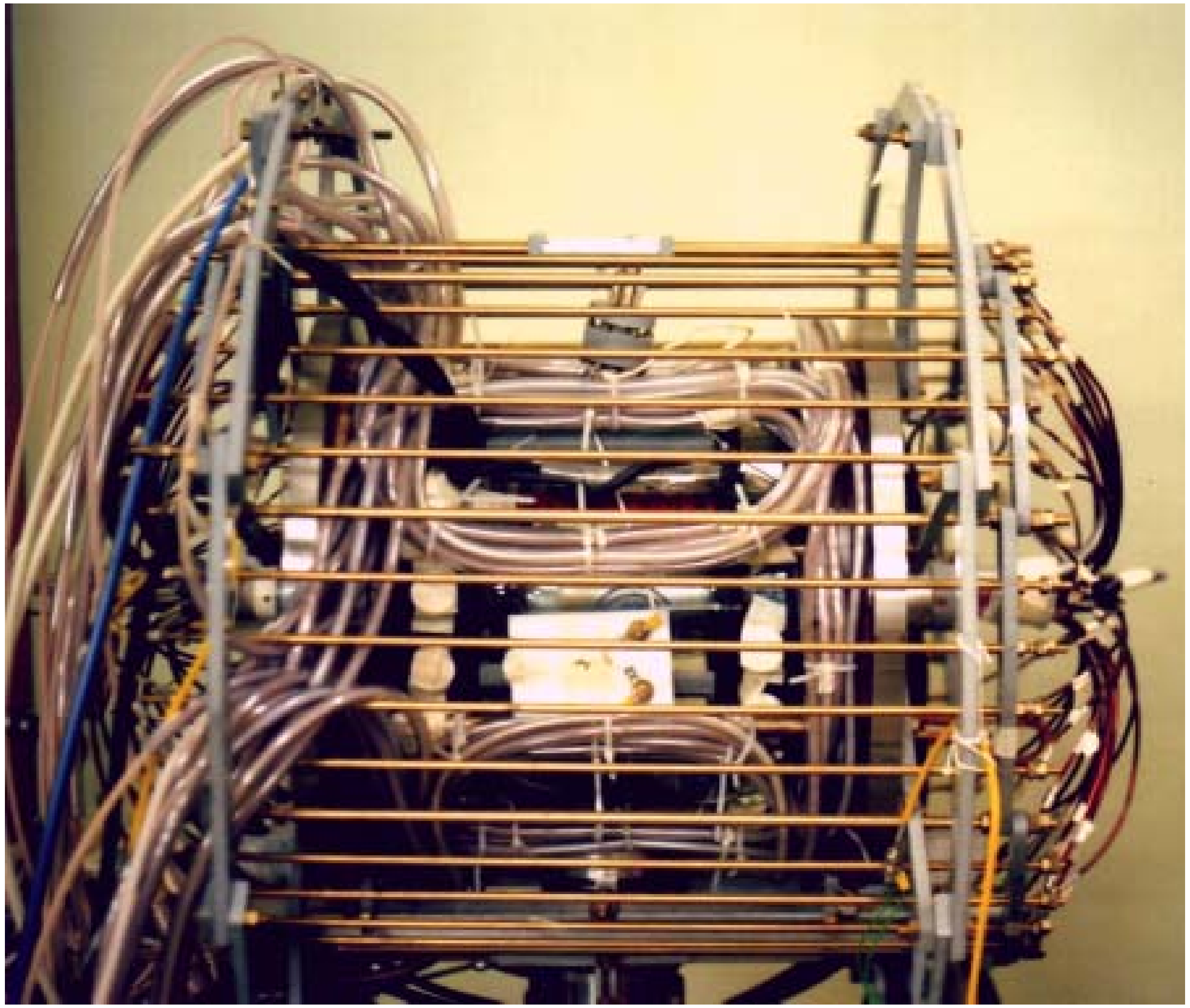
'MEDUSA' Wisconsin, USA

Glass vacuum vessel; R = 12cm A = 8cm Ip = 10 – 40kA Bt ~ 0.5T, tpulse ~ 5ms

Fore-runner of PEGASUS

The Flinders ST





Built by Lance McCarthy (Flinders); first plasma (2kA for 0.5ms) on 8th Oct 1998

Studied effect of RMF current drive on an existing ST plasma

(Lance McCarthy 2002 Nucl. Fusion 42 1304-1313)

NSTX, Princeton, US

NSTX team, June 2001

MAST – Mega Amp Spherical Tokamak

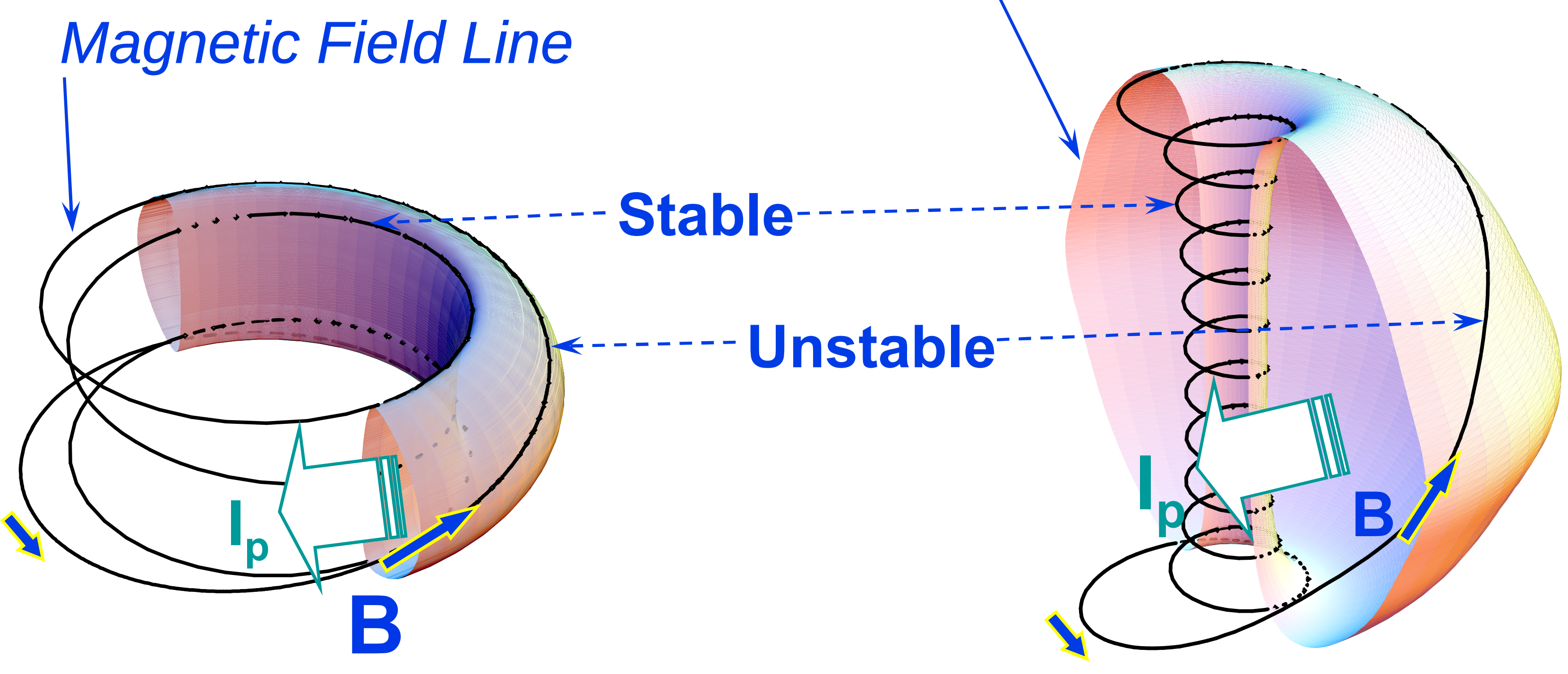
Features of the ST

- and -

Properties of low-A plasmas

In an ST, field lines spend most time in the high TF region near the centre…..

(cartoon Courtesy Martin Peng)



Conventional Tokamak

(safety factor q = 4)

Spherical Tokamak

(safety factor q = 12)

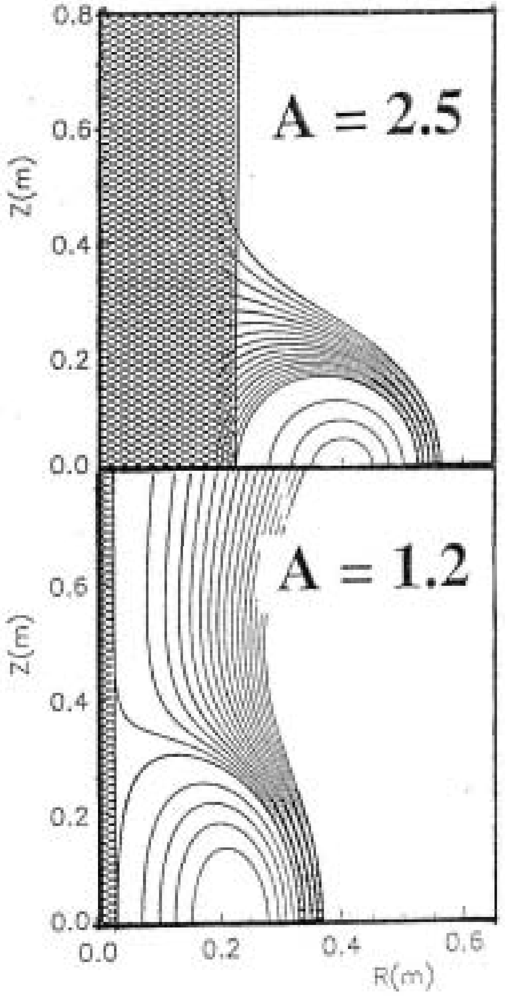
This leads to many differences in ST plasma parameters and properties

STs exhibit ‘natural’ elongation, and require much lower toroidal field, as shown in these equilibrium examples:

In each case, the minor radius = 15cm and I_p = 100kA.

A uniform vertical field is applied.

The plasma profiles and toroidal field are chosen to give q_o = 1, q_a = 8

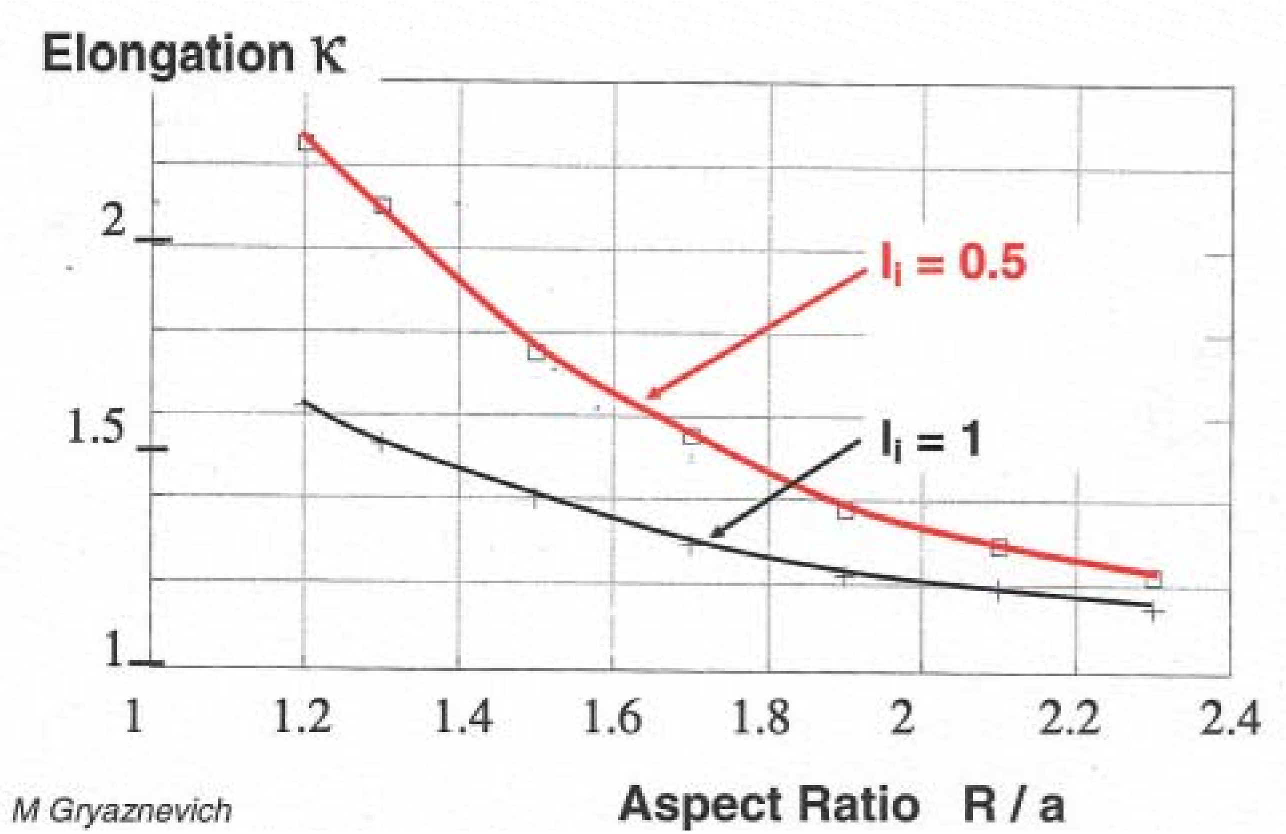


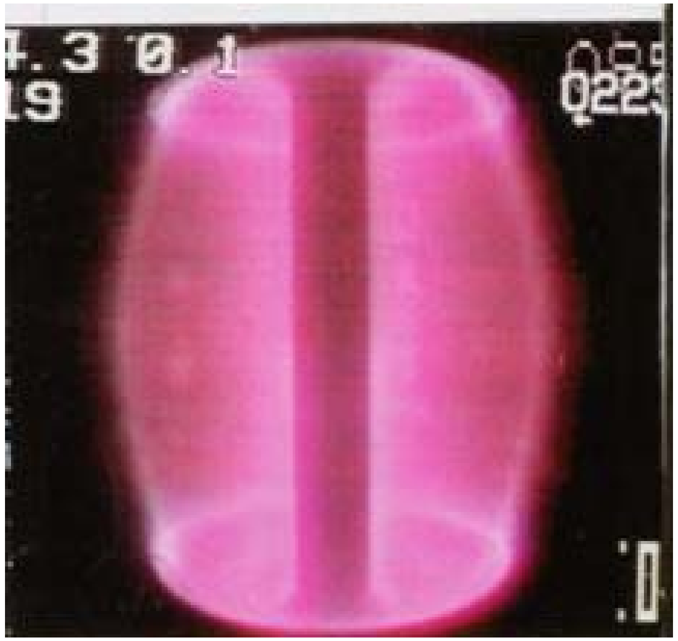
A	B_{T0}(T)	k	q_ψ
2.5	1.58	1.1	8.2
1.2	0.07	2.0	8.7

Note that, as aspect ratio is reduced from 2.5 to 1.2:

- elongation increases naturally from 1.1 to 2.
- Toroidal field required to achieve same q_a for given I_p falls by factor 20.

‘Natural’ elongation increases for flatter current profiles





High elongation $\kappa \sim 3$ at low inductance ($l_i \sim 0.3$) in START

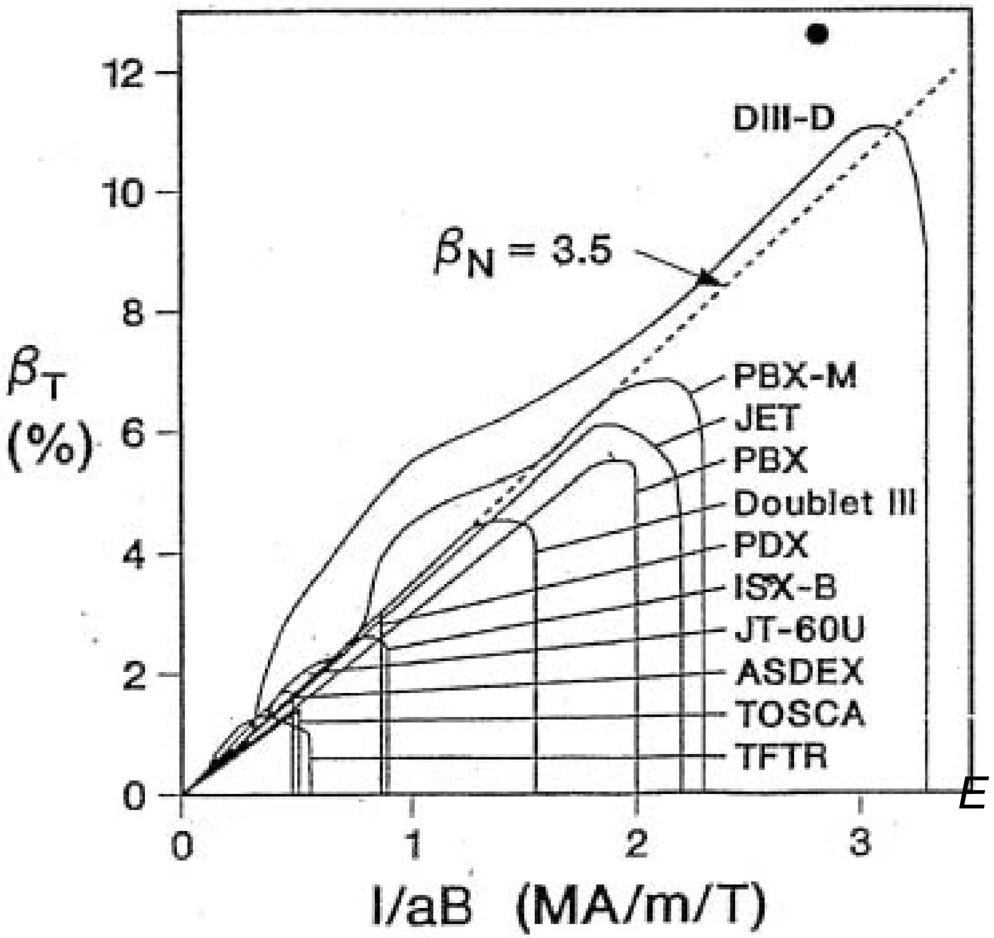
Features of the ST: some examples

- Beta limit
- Neo-classical effects
- ELM studies
- Confinement scalings
- Alfvén Mode studies

EXAMPLE 1: High beta is expected in an ST:

Theory: the Troyon limit $\beta_T = \beta_N I / aB$ can be written as $\beta_T = 5\beta_N \kappa / Aq_j$ and low A, high k are features of the ST

Experiment:

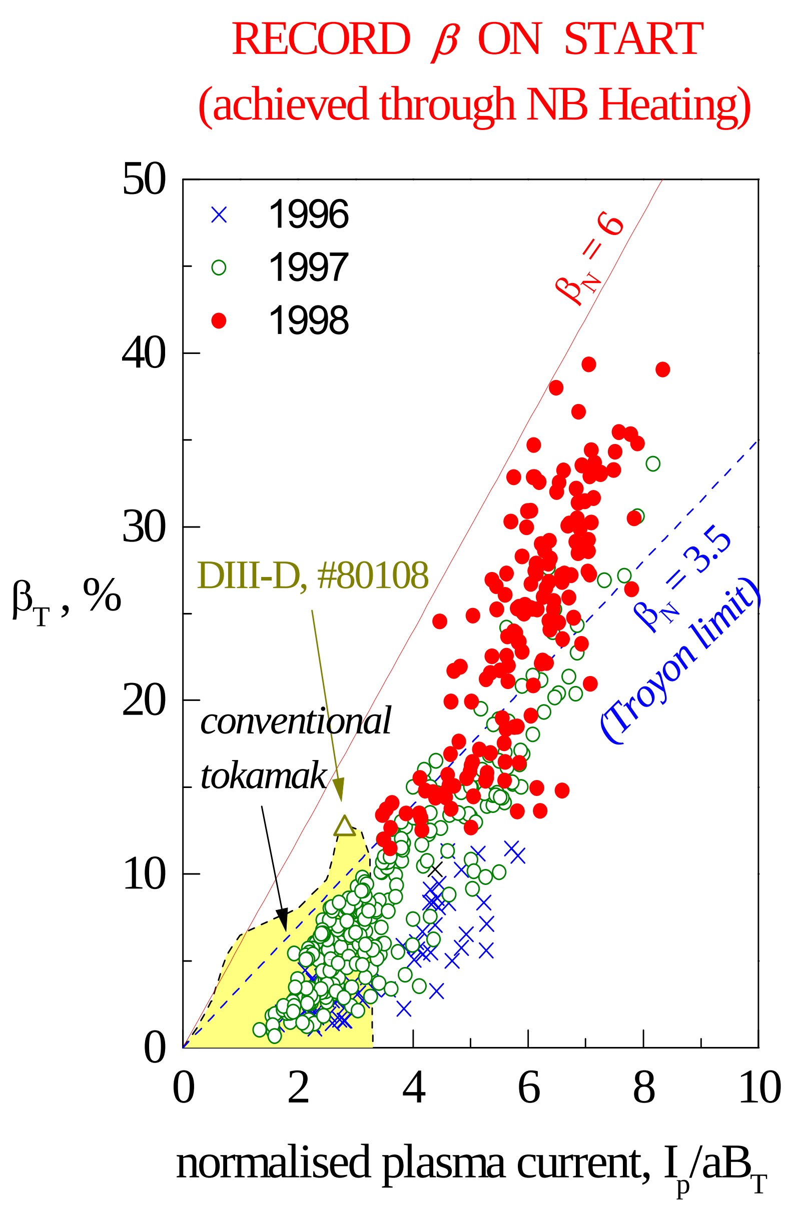


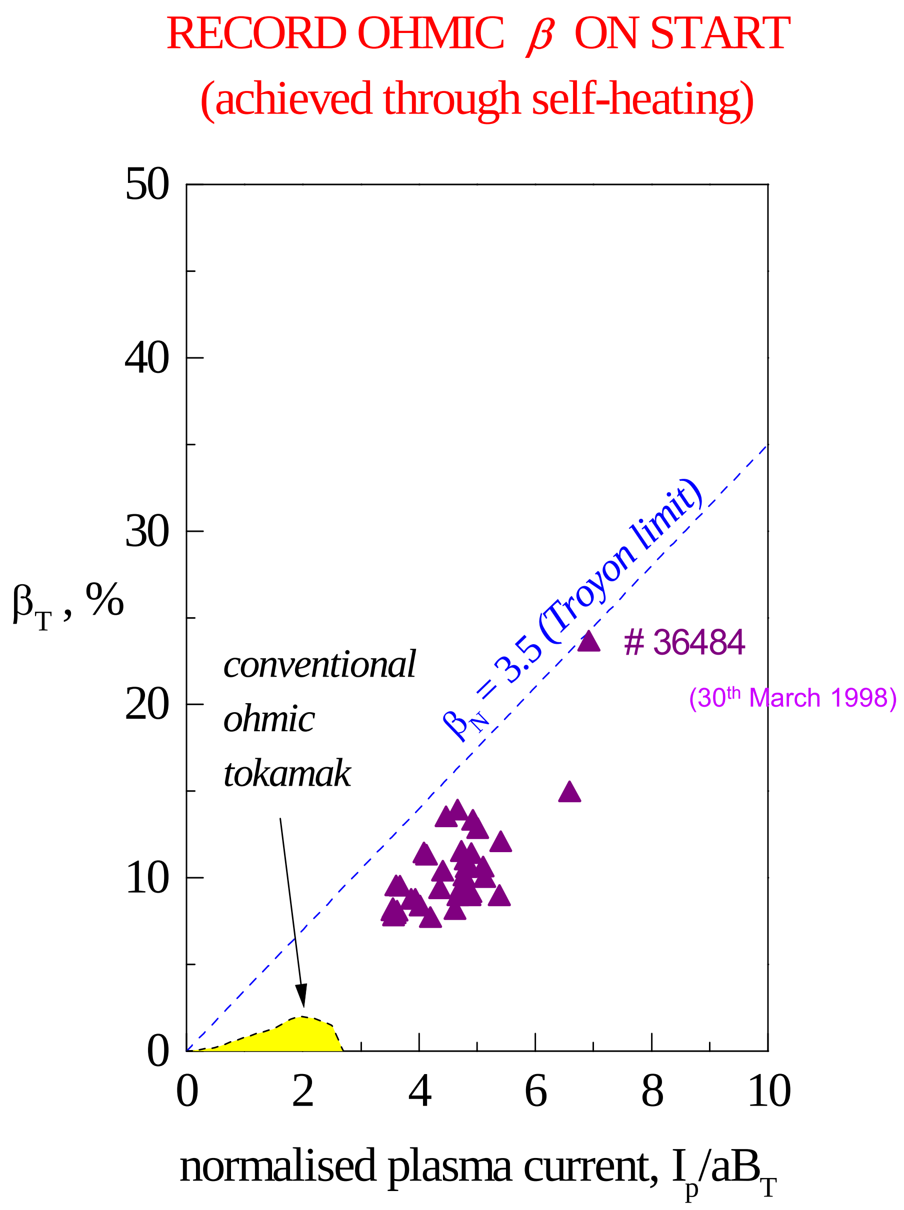
For each tokamak, the right-hand limit to operation is the onset of the low-q limit at $q_a \sim 2$

Large A, circular section machines (eg TFTR) meet this limit at low I/aB and so have low β

E J Strait, Phys. Plasmas I, 1415 (1994)

START achieved record beta values





note: START NBI injected 1MW into a 0.5m^3 plasma, i.e. 2MW/m^3; MAST injects 4MW into a 5m^3 plasma, i.e. 0.8MW/m^3 ; NSTX injects 7MW into a 5m^3 plasma, i.e. 1.4MW/m^3

Example 2: Neo-classical effects in STs

The large increase in B_T on the inboard side of an ST causes many lower energy particles to be trapped in 'banana orbits' – considerably increasing the resistivity, and demanding higher loop voltage to drive the current. However this brings advantages…

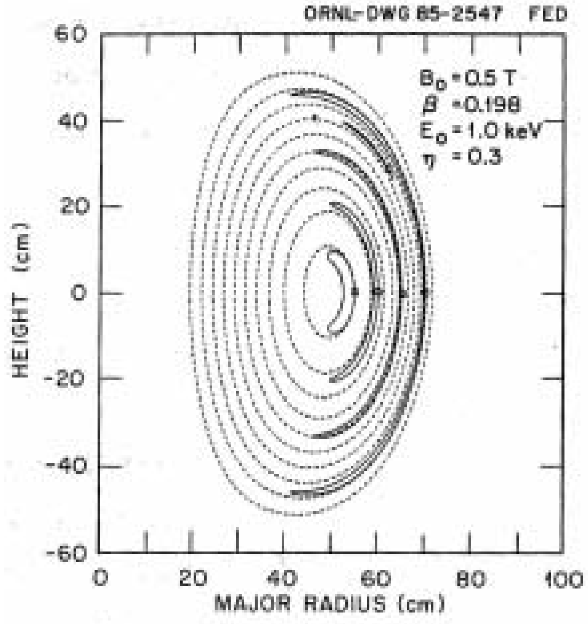
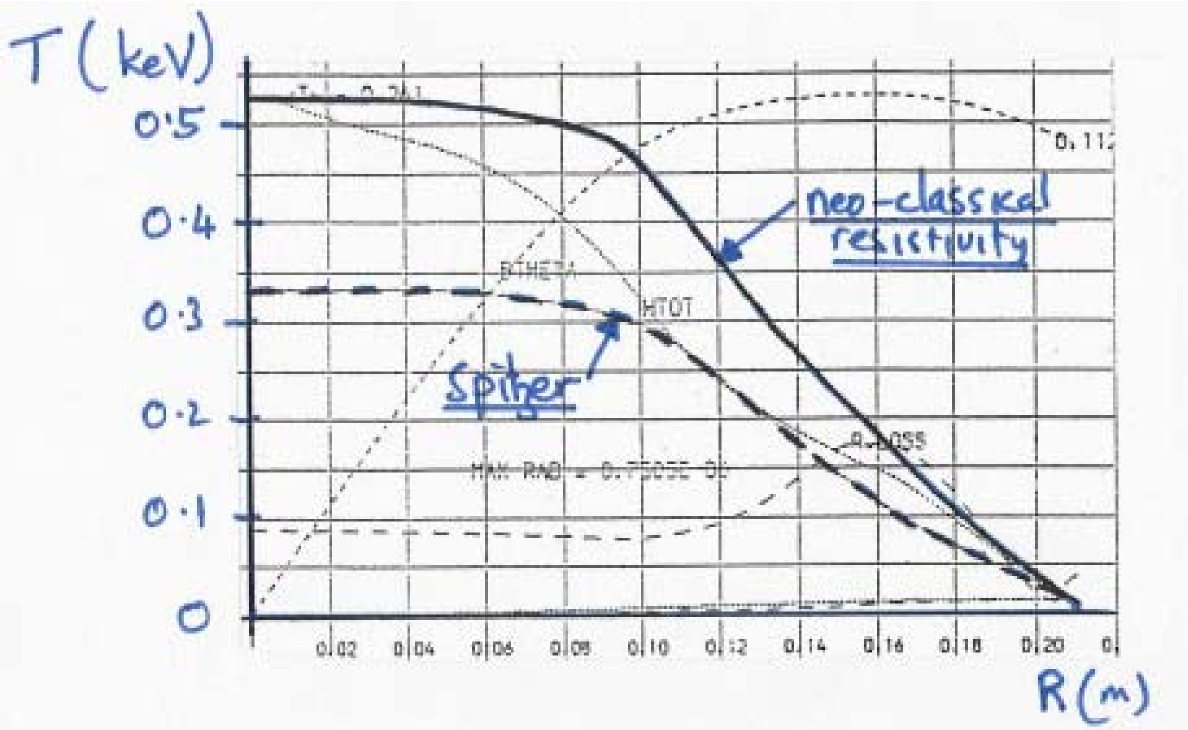


Fig. 2.7.5. 1-keV H^+ orbits in STX.

Consider 2 simulations of START, using the same (INTOR model) transport on a 118kA START plasma, with impurity fractions chosen to give Zeff=3 and the density profile scaled so that P_{RAD} / P_{Ω} = 50% :



(a) **With Spitzer resistivity** (no bootstrap):

Vloop = 1.1V; Jboot = 0; τ = 3.6ms;
neo = 3×10^{19}; **β = 4%**

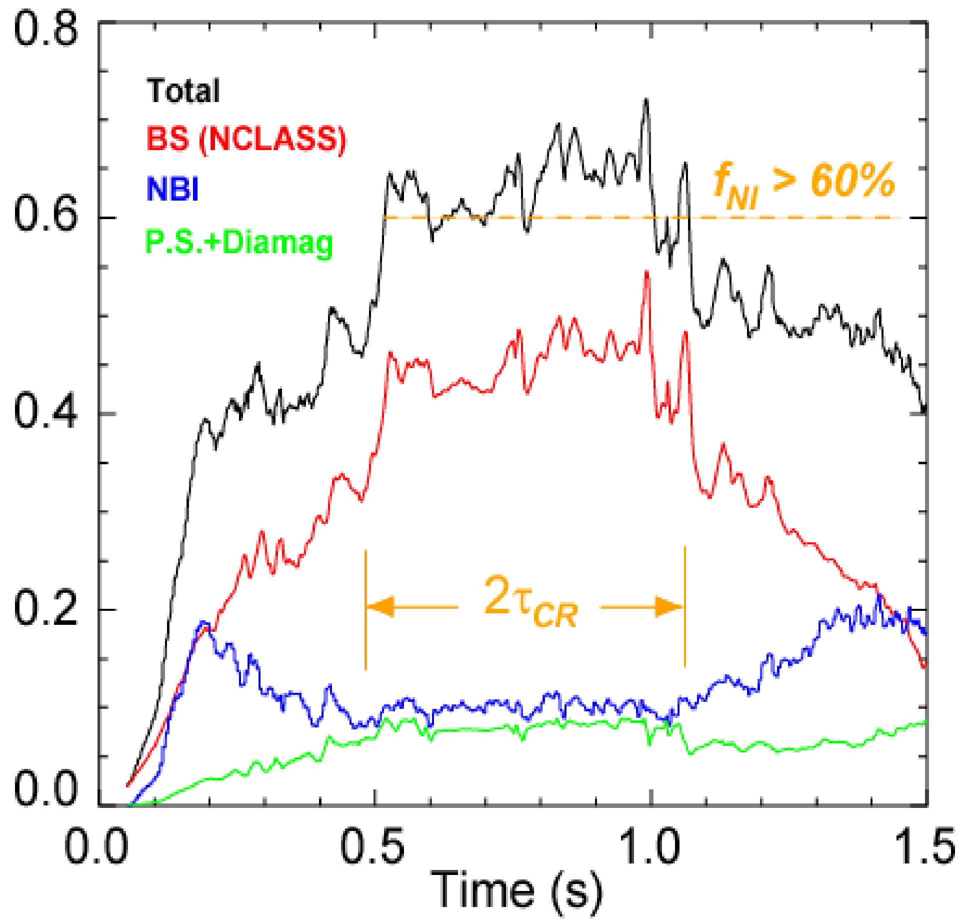
(b) **With neo-classical resistivity**:
Vloop = 1.9V; Jboot = 36kA; τ = 5.5ms;
neo = 4.4×10^{19}; **β = 9.1%**

Neo-classical effects (cont'd): 'bootstrap' current

$$j_B \sim \sqrt{A}\, \beta_N\, (1 + \kappa^2) / l_i^{1.2}$$

STs have low A – but high elongation κ
→ high bootstrap currents feasible?

TRANSP non-inductive current fractions for NSTX shot 116313G12

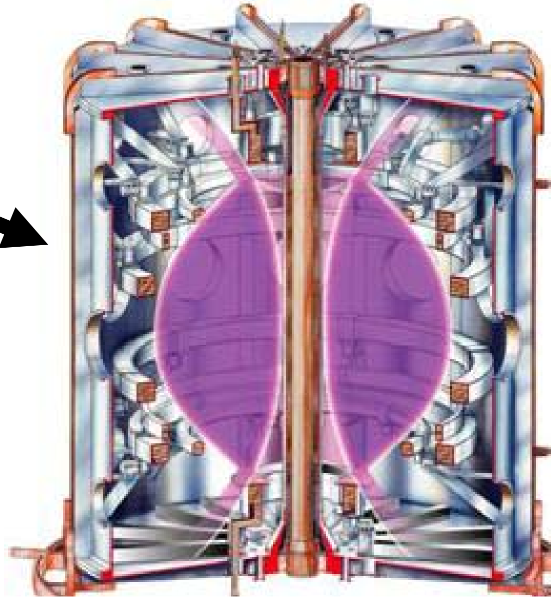


f_{NICD} = 65% on NSTX

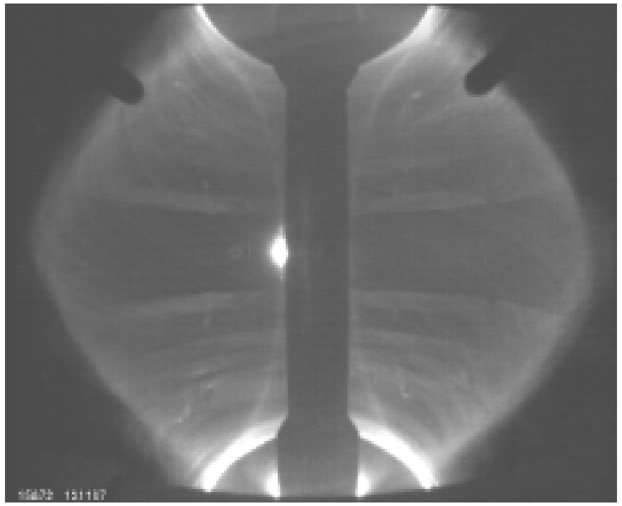
NSTX and MAST both plan upgrades with x2 NBI power, predicted to achieve full non-inductive current drive

Example 3: ELMs (Edge Localised Modes)

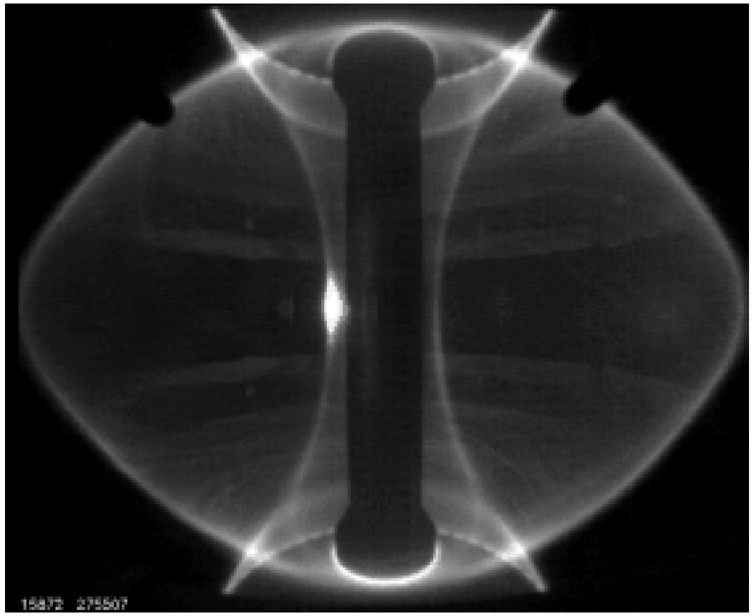
MAST is uniquely suited to ELM studies ….



Turbulence limits the confinement ('L'-mode)…..

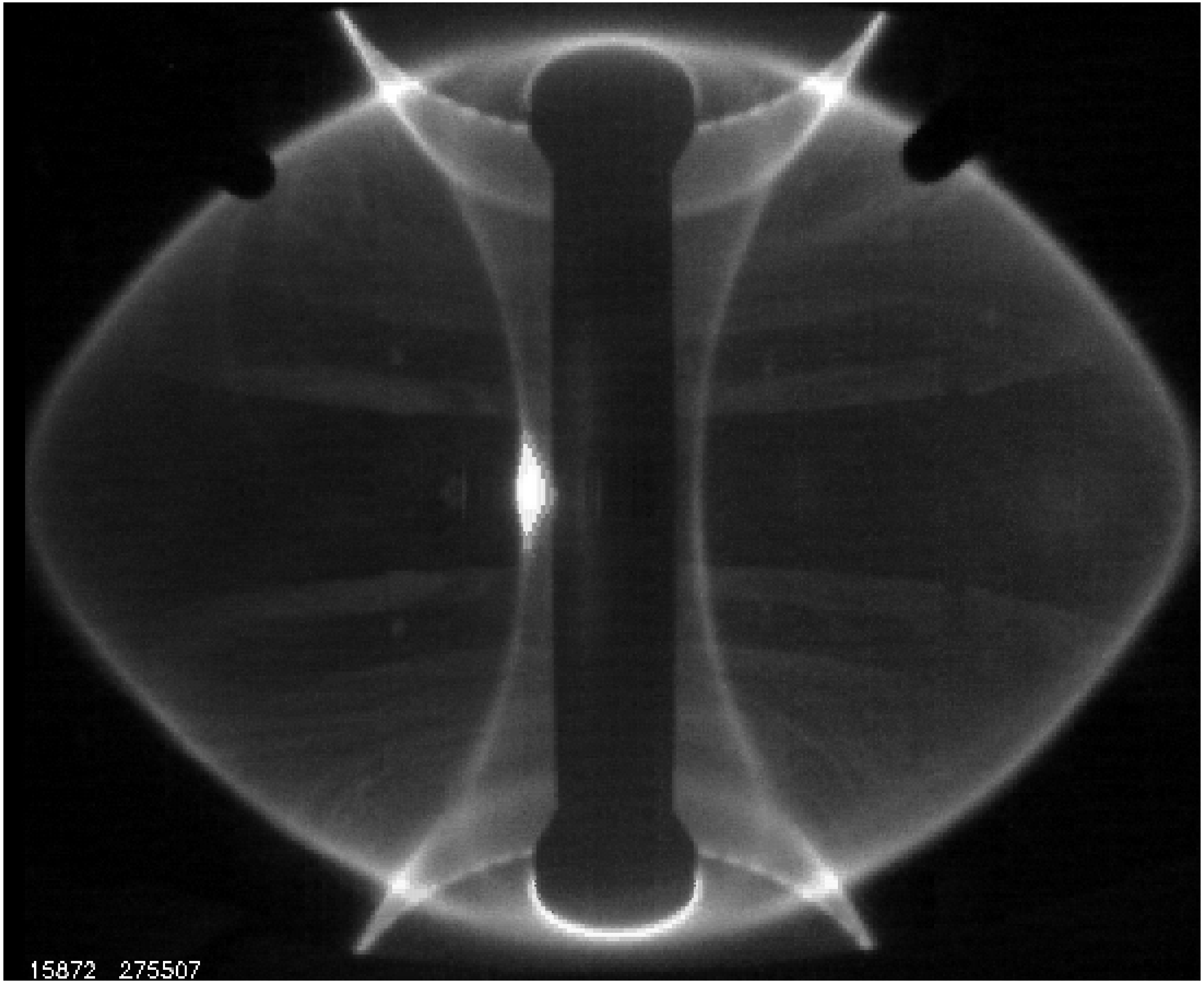


Under certain conditions the plasma can enter an improved confinement (H-mode) regime.

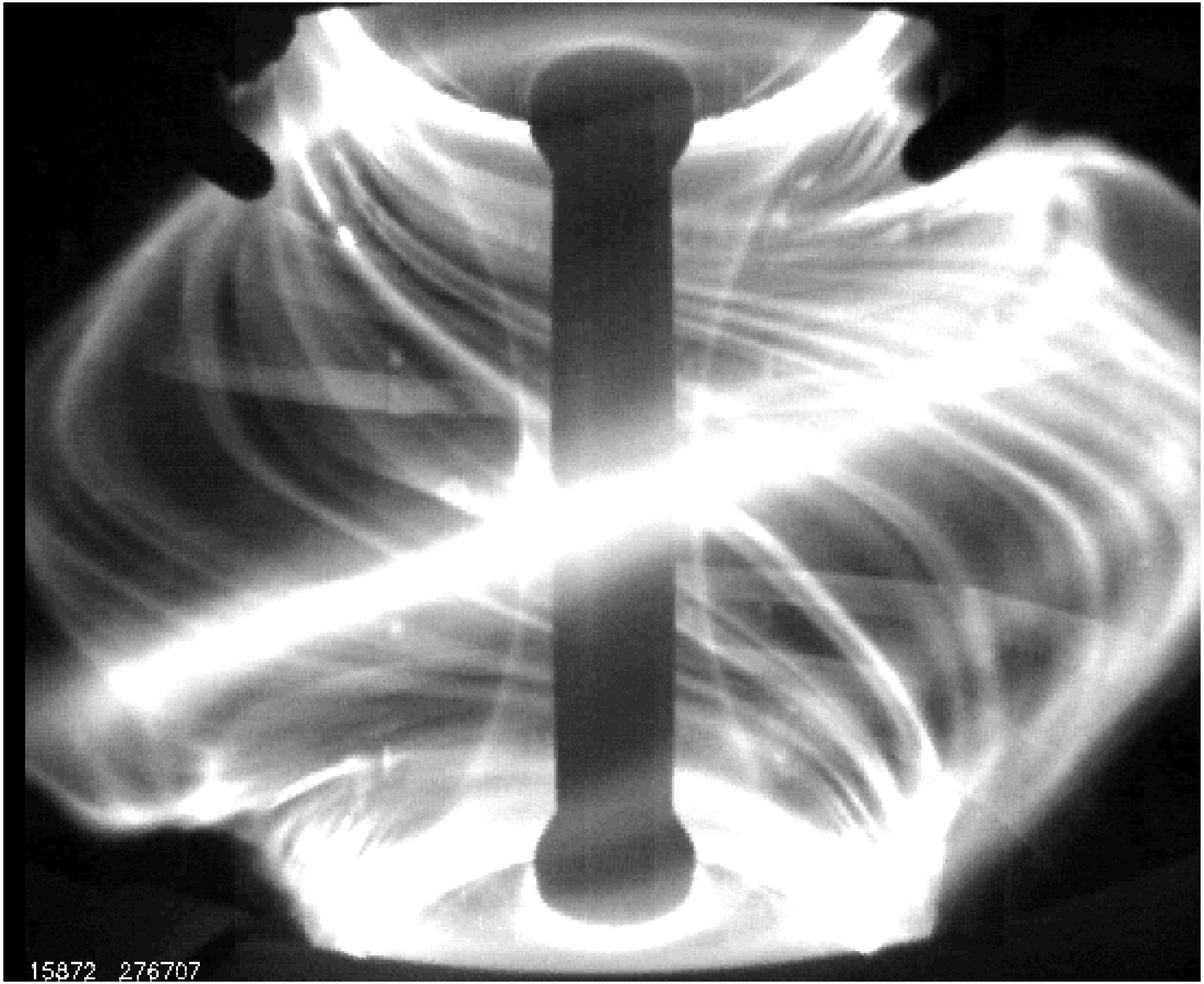


But the steep edge gradients produce instabilities called ELMs (Edge Localised Modes): can release several % of the plasma energy
On ITER: 20 MJ in 0.5ms

The price of H-mode - ELMs

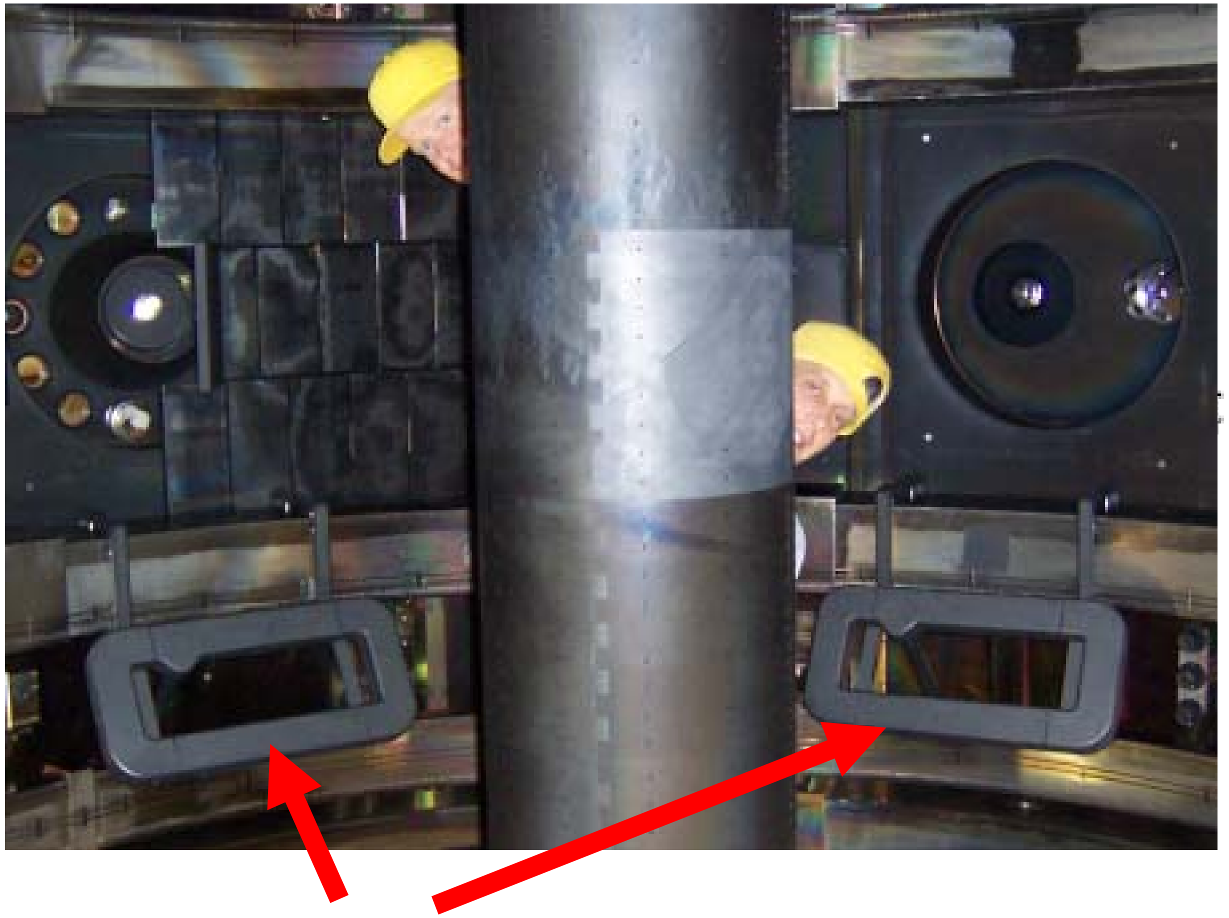


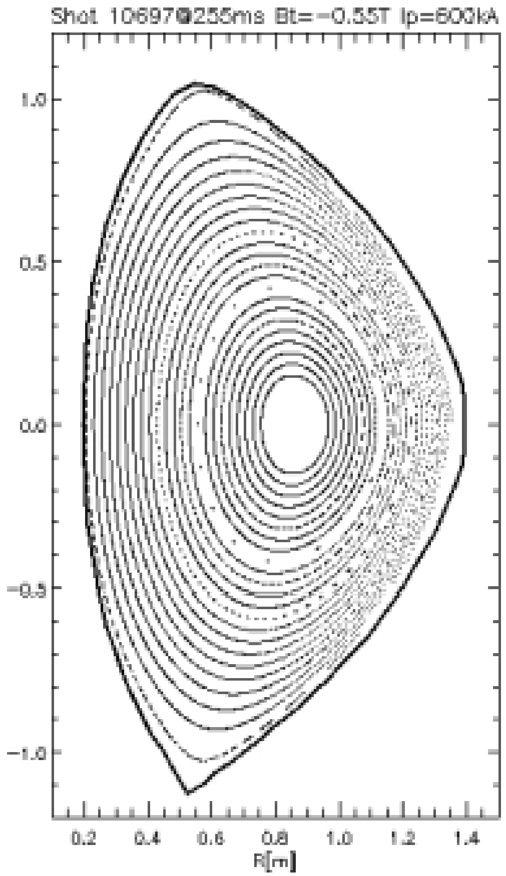
The price of H-mode - ELMs

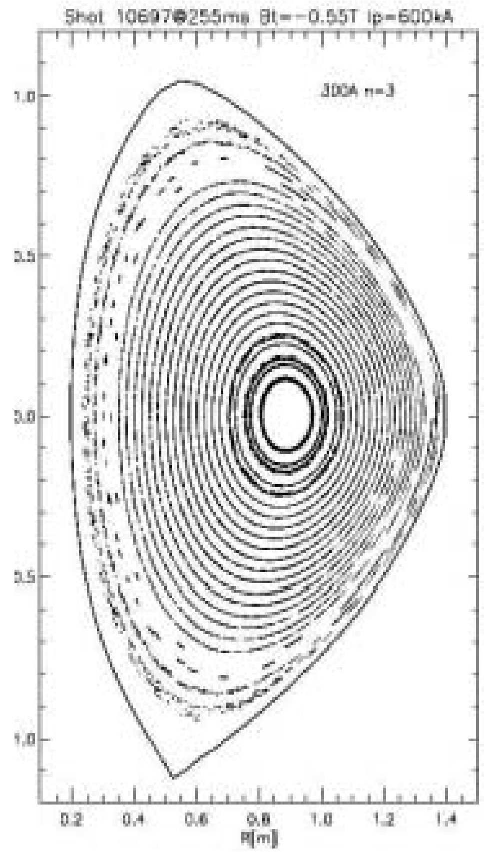


How to mitigate ELMs

Want to keep the good confinement due to H-mode but need to stop the instability growing
– ergodise the flux surfaces near the plasma edge ?







12 'ELM coils' now installed on MAST | Icoil = 0 | Icoil = 300A

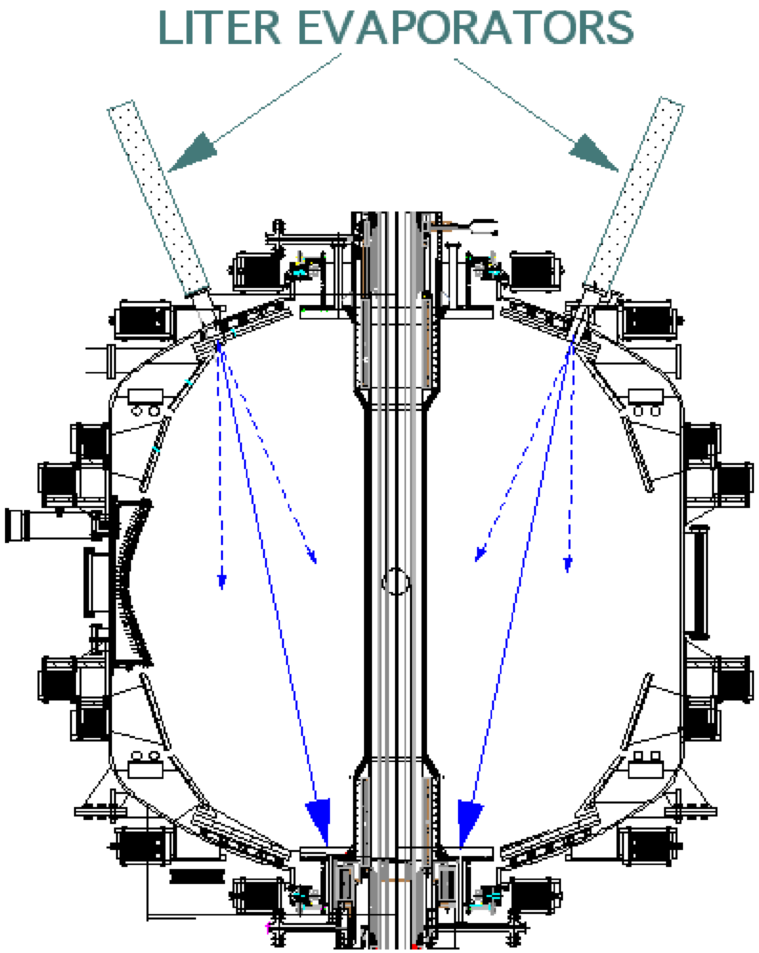
ELM coil studies now under-way on MAST

Fusion Working with Europe

NSTX is unique in exploring lithium in a diverted H-mode plasma – with remarkable results

Lithium evaporators (LITERs) provide complete coverage of lower divertor

NO between-shot He glow required

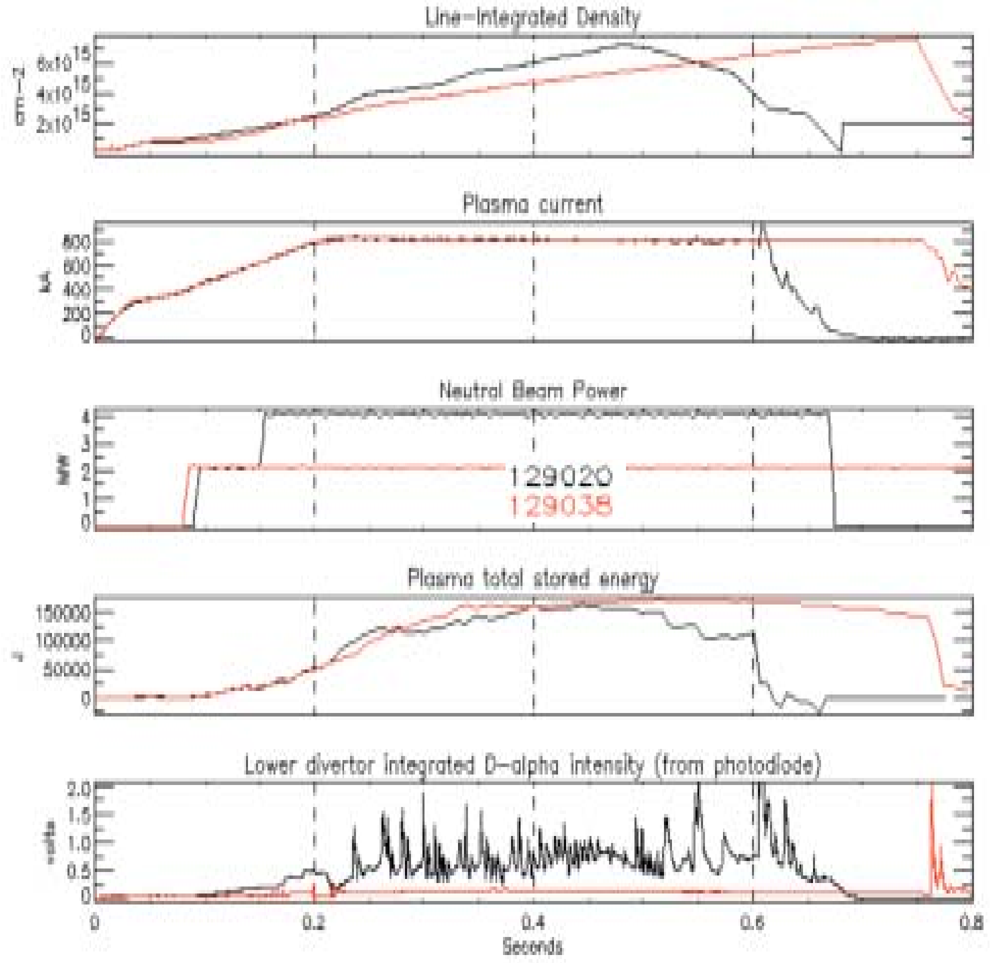


reduced recycling

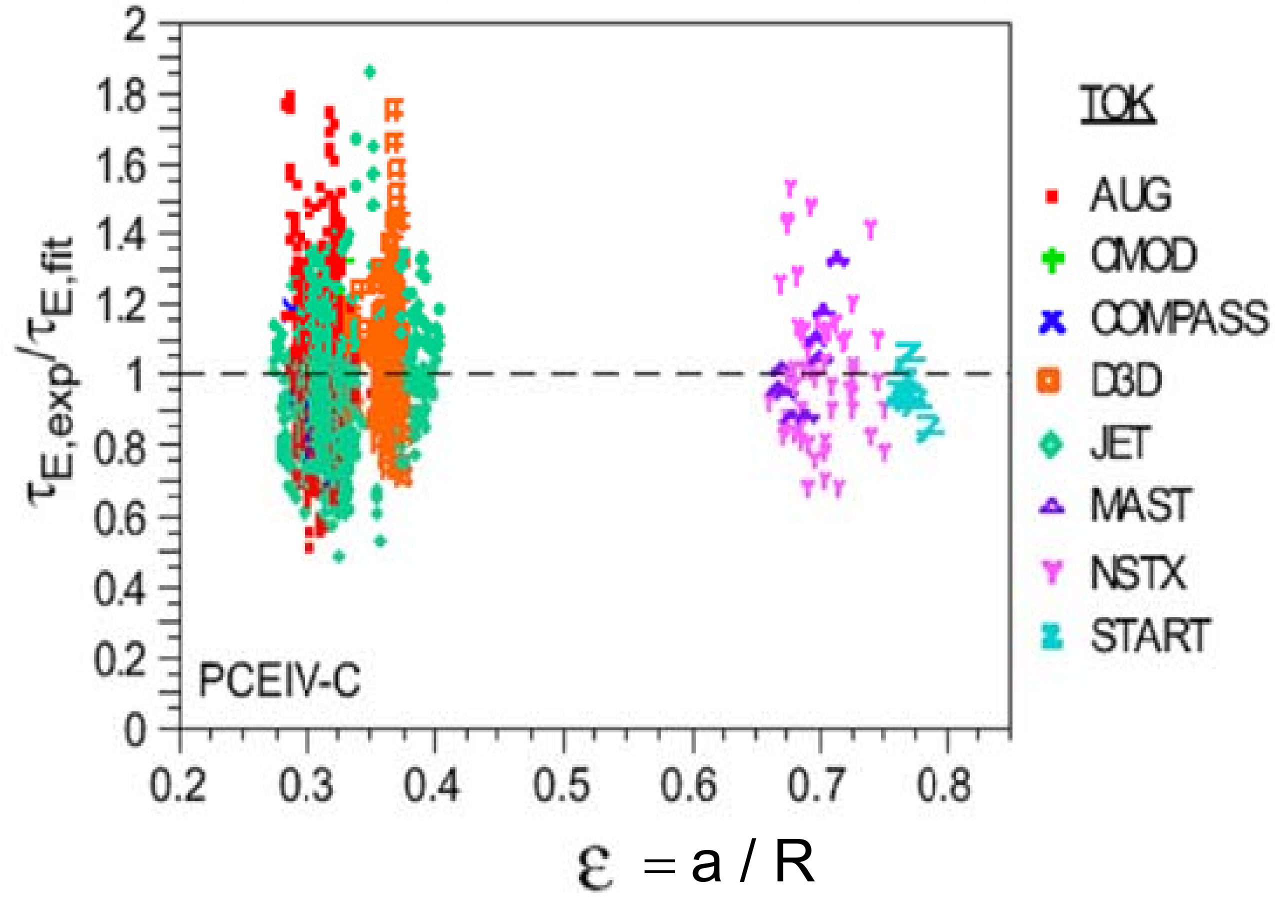
ELMs can be eliminated

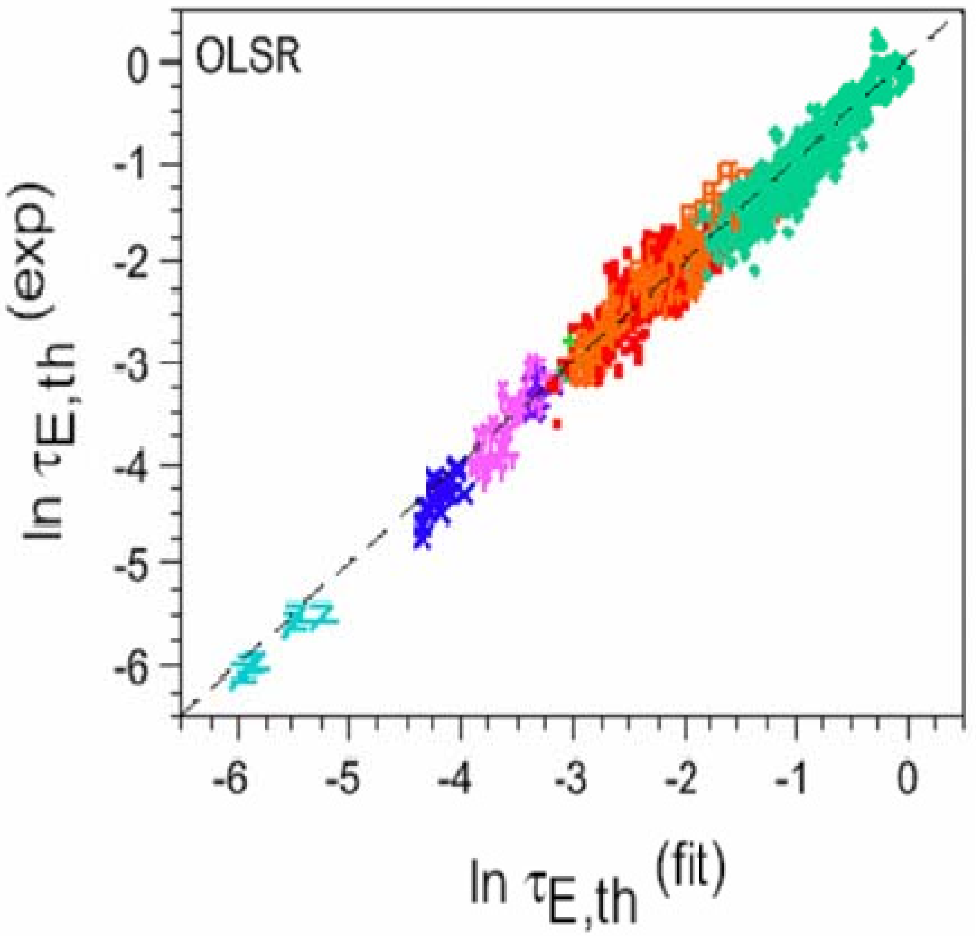
Confinement time doubled (up to 80ms)

No Li (black) With Li (red)



Example 4: Confinement studies





Studies on STs have increased confidence in ITER scalings….. □□□□

ONGOING! Present ITER scaling has $\tau \sim B_T^{\ 0.15}$ whereas latest results from NSTX, MAST indicate much stronger dependence $\tau \sim B_T^{\ 1.3}$ (MAST), $B_T^{0.91}$ (NSTX)

Courtesy Stan Kaye, PPPL, Martin Valovic UKAEA

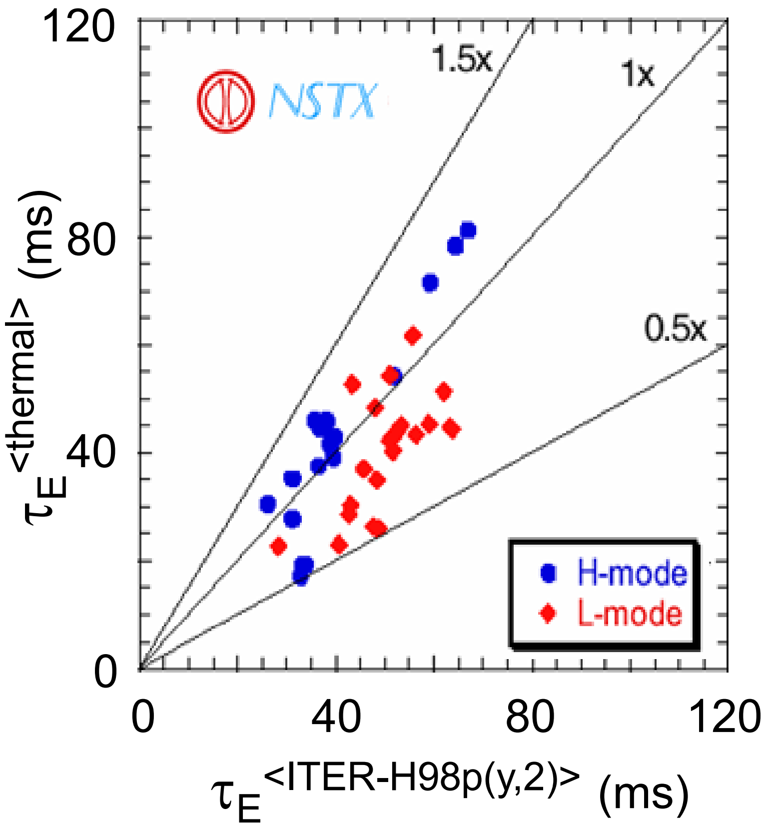
Plasma Phys. Control. Fusion ***48*** *(2006) A429–A438*

Fusion Working with Europe

Confinement studies (cont'd)

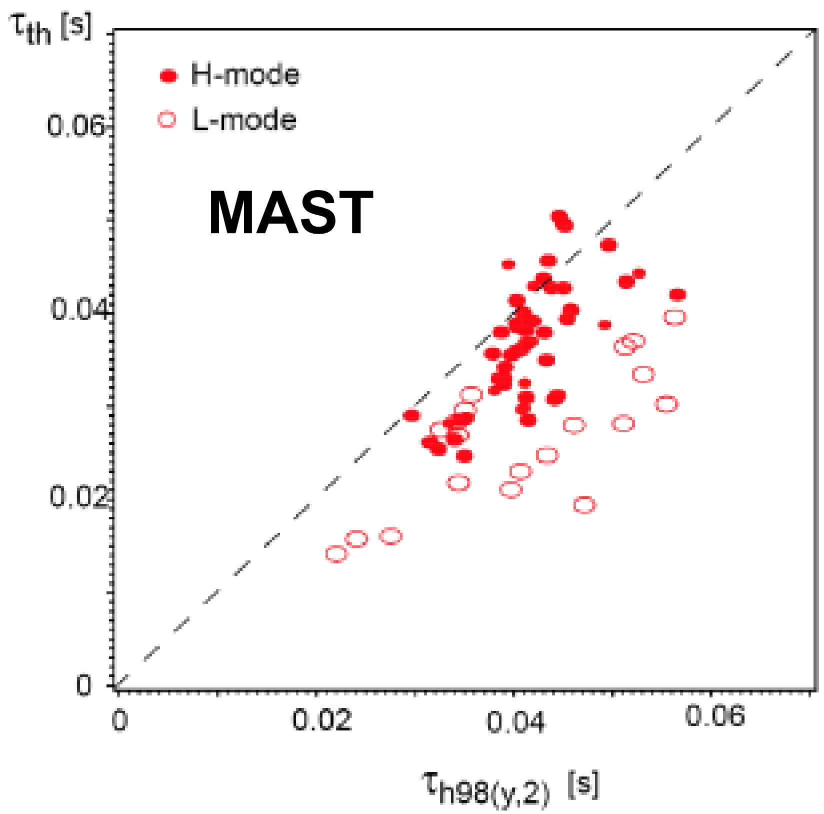
There is an increase in confinement time τ_E on going from L-mode to H-mode.

Studies show that the increase is small in STs (typically 1.20 in MAST or NSTX), ~2 in JET, and expected > 2 in ITER.



H-mode:
~agrees with IPB98y2 H-mode scaling

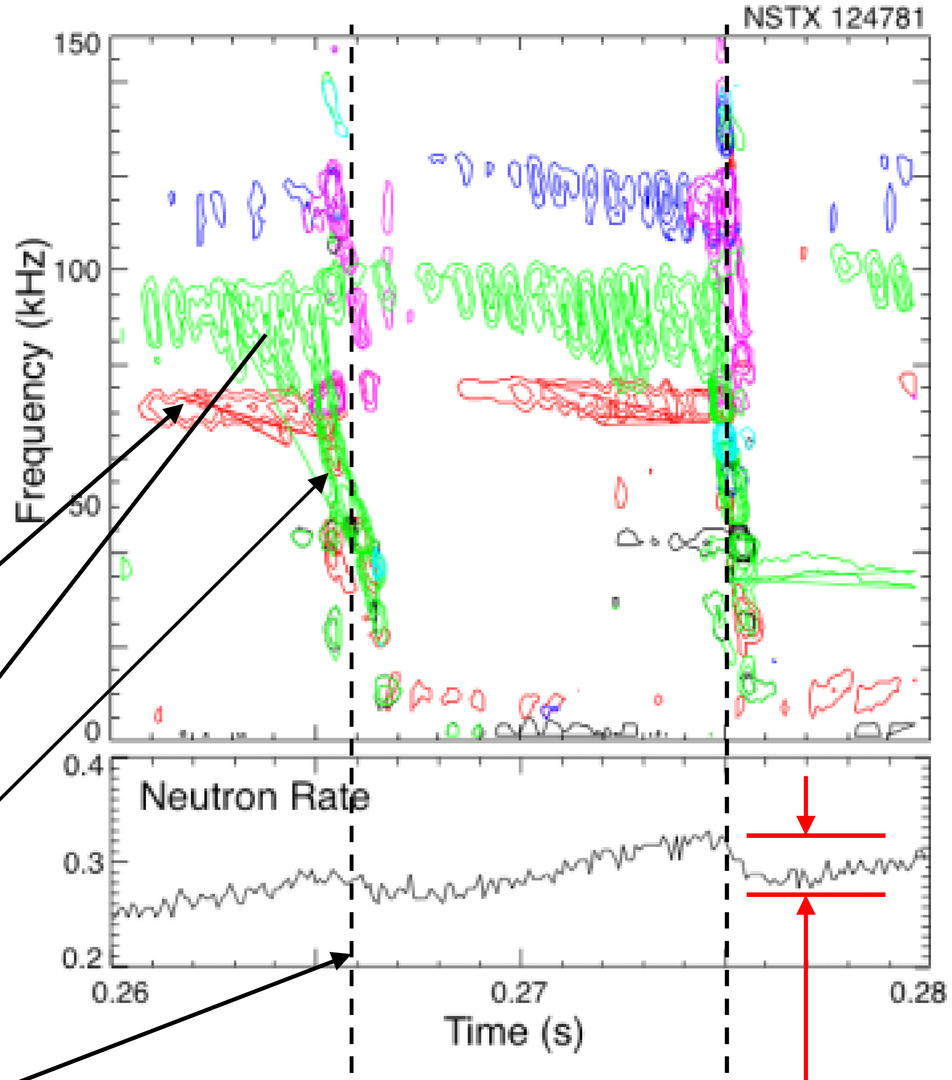
L-mode:
lower than H-mode data by ~ 20%



H- mode operation appears very important for ITER, not so important for STs

Example 5: NBI-heated STs can study fast particle effects - potentially important for ITER

NSTX and MAST use high-energy NB injection into low field plasmas (hence fast ptcle speed can exceed Alfven speed), giving unique opportunity to study fast particle effects



NSTX results:

- first see Alfven Eigenmodes (AE)
  - then chirping AE
  - then avalanches

- Avalanches correlate with drop in neutron emission

15% drop in neutron rate

Advantages of Spherical Tokamak research

Simple construction

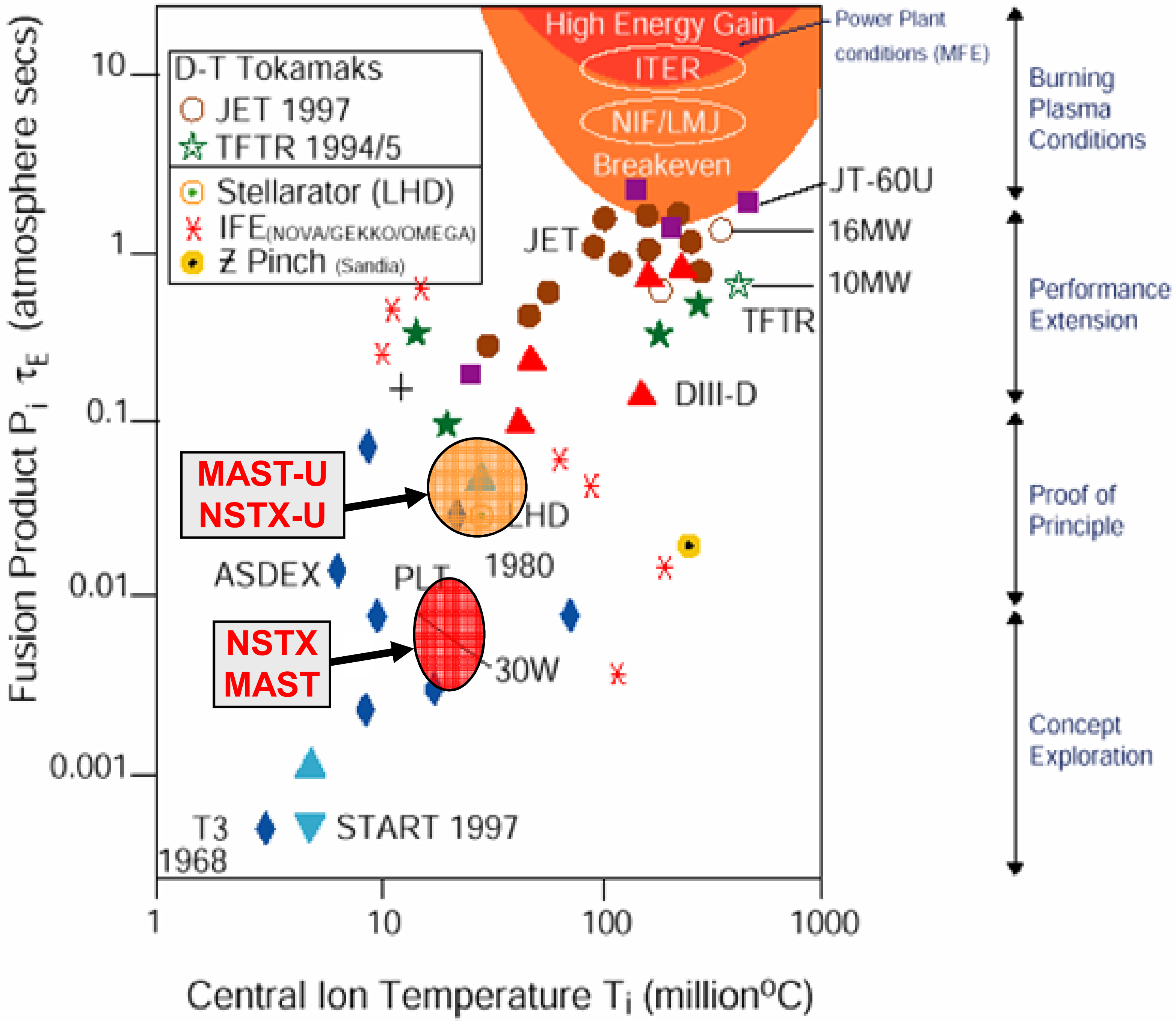
A lot of plasma for the money!

Provide valuable info on scaling laws

Provide insight into tokamak behaviour

BUT can STs make actual FUSION devices?

Progress Towards Fusion Power



Conditions for Fusion – problem 1

For ignition, the triple product $n_i\, T_i\, \tau_E$ must exceed

$$n_i\, T_i\, \tau_E > 3 \times 10^{21}\ m^{-3}\ keV\ s$$

present STs have low B_T, quite sufficient for stability and physics studies – BUT :

$n_{max} \sim I_p$ (Greenwald scaling) – but max. $I_p \sim B_T$

T approx $\sim B_T$ ($\sim B_T^{0.8}$ for Ohmic plasmas)

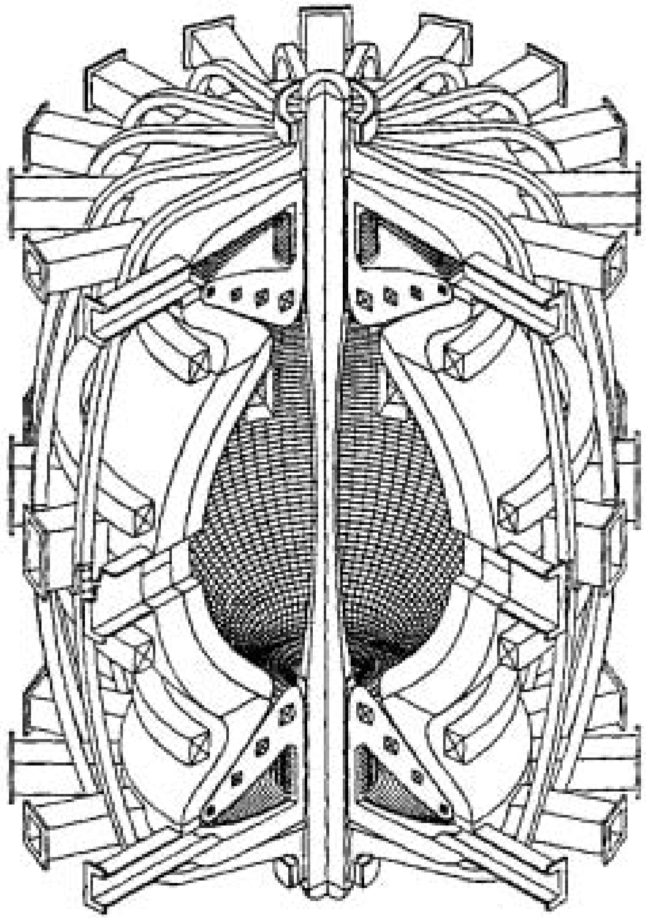
τ_E approx $\sim B_T$

Hence the present trend in STs to higher toroidal field!

Conditions for Fusion – Problem 2

There is no space for a full neutron shield around the c/col in any reasonable-sized ST!

The Peng-Hicks ST reactor concept offers a possible solution:
Copper centre column
No blanket (not needed at low A)
No shield (damage rate low, replace c/col every year or two)



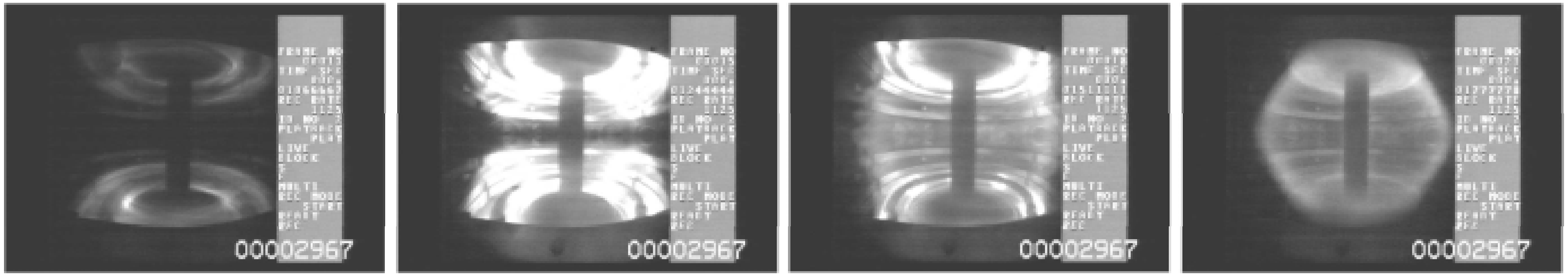
THIS GIVES THE POSSIBILITY OF COMPACT FUSION DEVICES

A family of such ST fusion devices was considered in
'The ST path to fusion power', R Stambaugh, V Chan, R Miller, M Schaffer Fusion Technology 33 1998 p1

M. Peng, J. B. Hicks, "Engineering Feasibility of Tight Aspect Ratio Tokamak (Spherical Torus) Reactors," Fusion Technology 1990

But with no shield, cannot have a central solenoid!

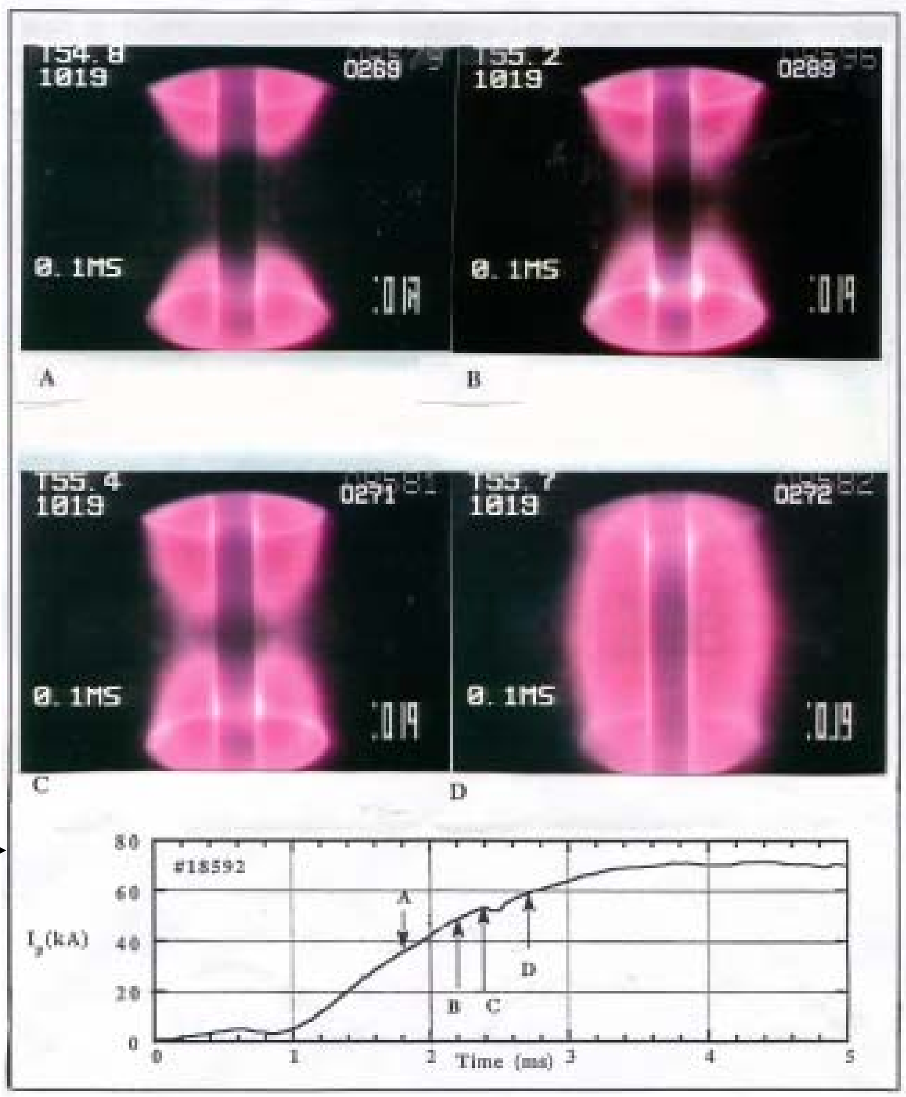
However, there are several possibilities for start-up….



'Merging/Compression' on START (breakdown around P3 coils):

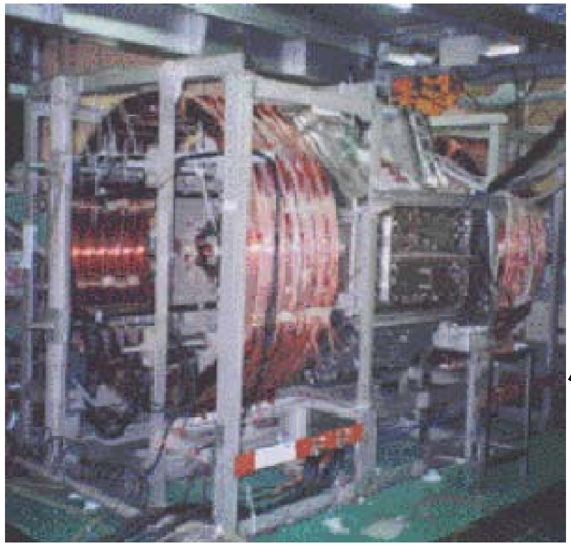
150kA obtained WITHOUT solenoid (500kA on MAST)

'Double Null Merging' produced 60kA on START (but with solenoid)



A New ST Aimed at Formation and Sustainment of Ultra-High β Plasma; UTST at Univ. Tokyo

TS-3 / TS-4

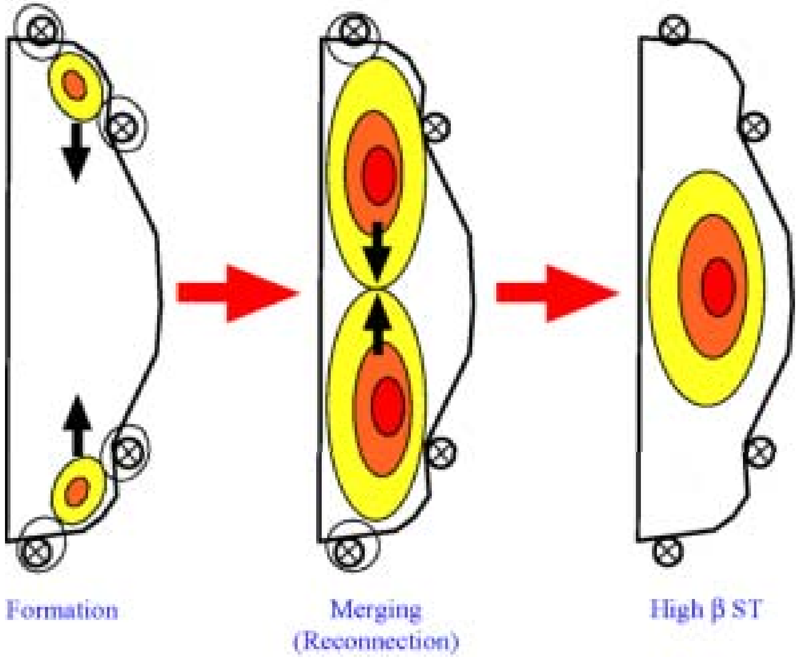


Formation of ultra-high β ST plasma using plasma merging

Supported by Grant-in-Aid for Scientific Research

UTST

Merging scenario



TST-2

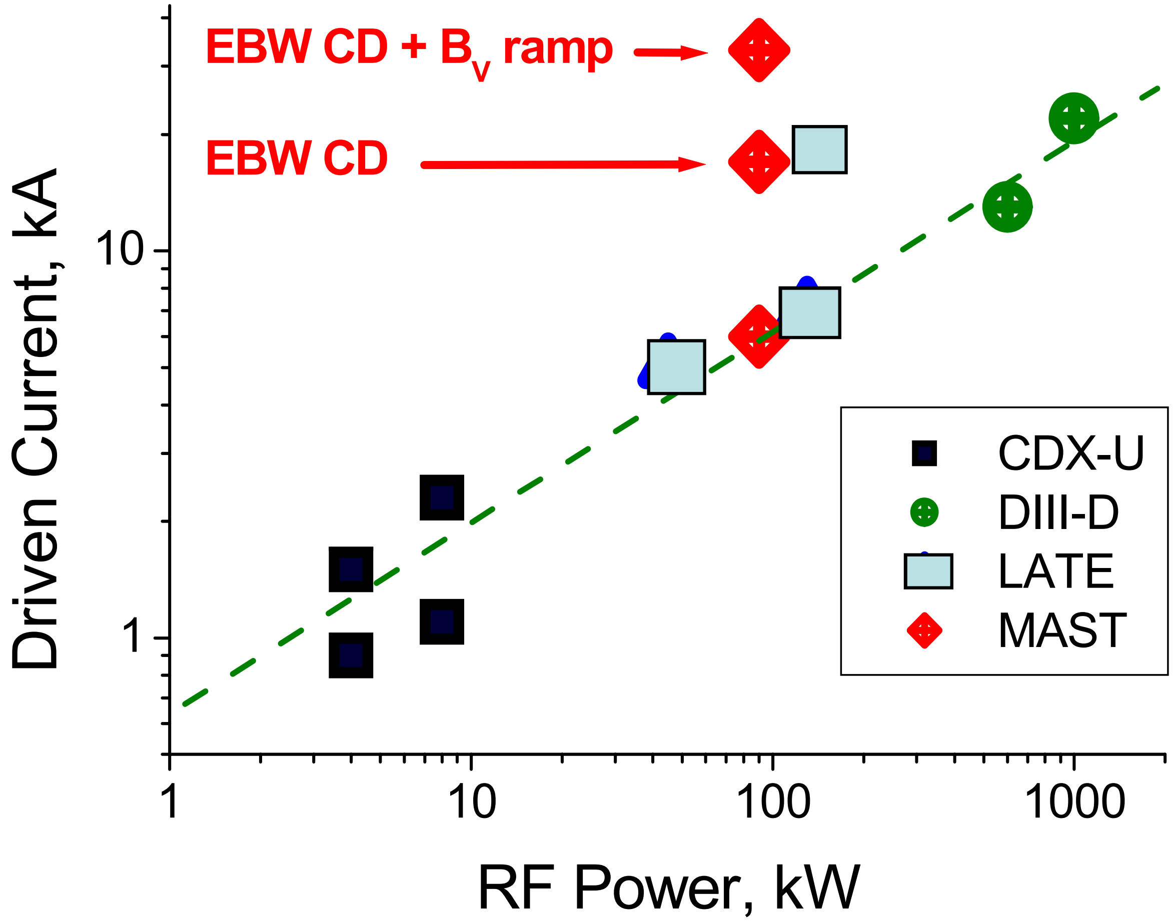
Sustainment using innovative RF methods

- HHFW (~ 20 MHz)
- LHFW (~ 200 MHz)

Induction coils OUTSIDE the vacuum vessel

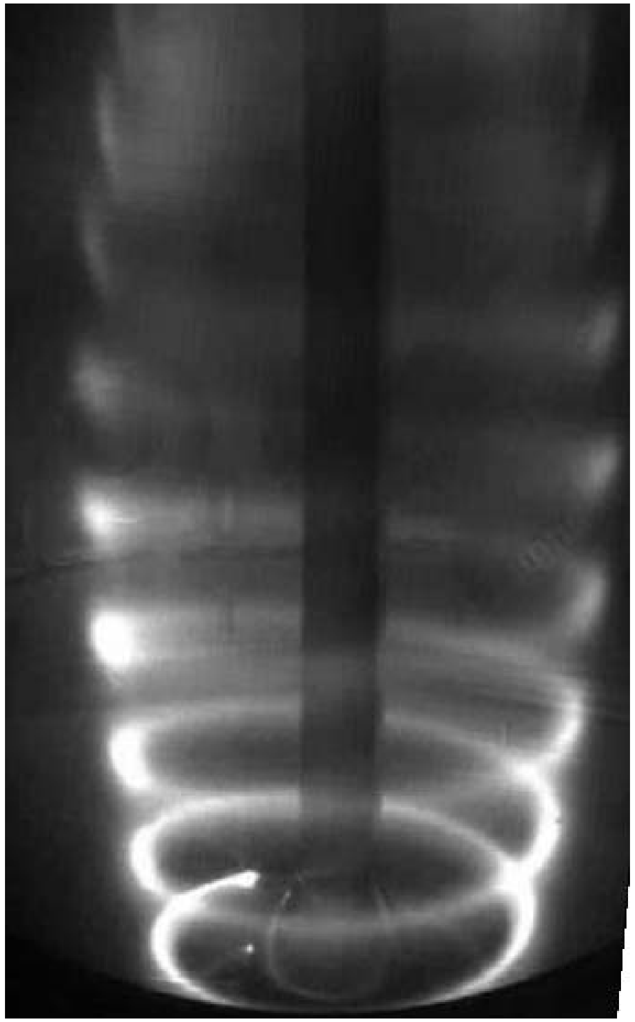
RF methods of plasma current initiation and ramp-up

A 28GHz system on MAST (Vladimir Shevchenko) has demonstrated efficient plasma current production, via Electron Bernstein Wave (EBW) current drive. A higher power long pulse tube is being loaned by ORNL to determine the scaling of these promising results

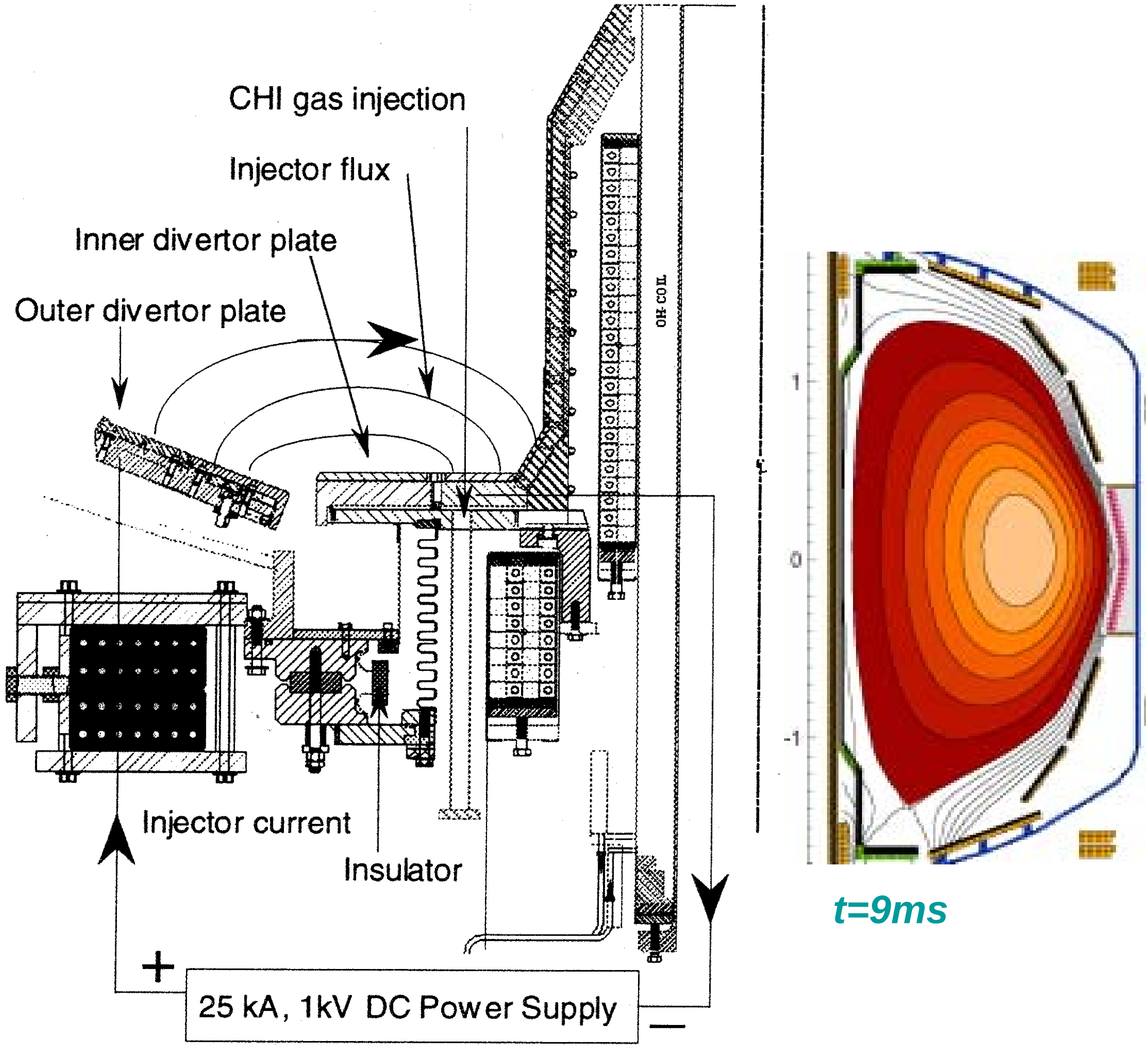


A 5GHz ECRH system on LATE (Kyoto Univ) has demonstrated plasma current ramp to 20kA using 150kW

Other methods: Coaxial Helicity Injection, arcs, plasma guns, etc



Plasma formation with helical arc
(G Garstka, Pegasus)



CHI on HIT and NSTX :
produced 160kA plasma

START-UP and RAMP-UP (2)

There are many schemes for producing the initial plasma current without a central solenoid.

Once a suitable target for NBI is obtained, ST plasmas have big advantages which makes ramp-up to the MA level very effective……

Some advantages of NBI

1. **Heats plasma, reducing resistive losses**
2. **Heats plasma – requiring increase in Bv, which is very effective in an ST**
3. **Produces direct current drive (If co-injection!)**
4. **Can produce hollow profiles – low li, high k, low flux consumption, high bootstrap, q_o >1 so no sawteeth, q_o > 2 so no bad NTMs….!**

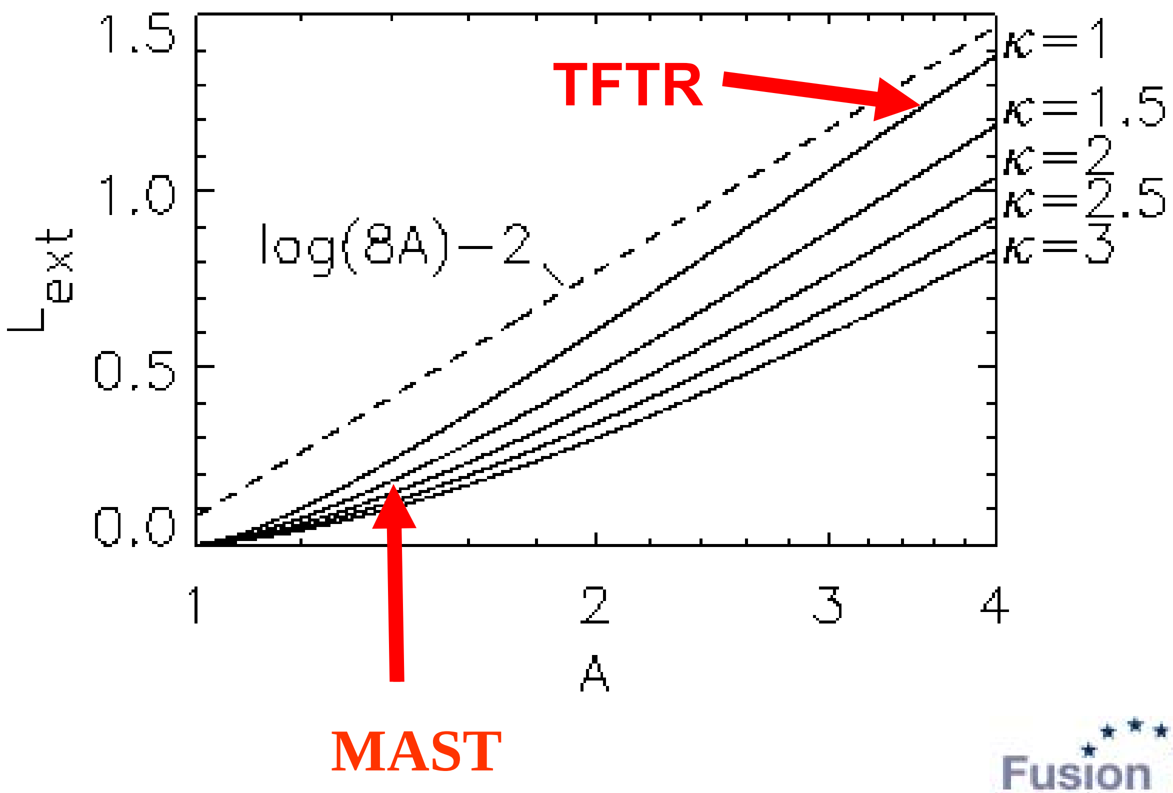
$$B_V = \mu_o I/(4\pi R)\ [\ L_{ext} + l_i/2 + \beta_p + 0.5]$$

Flux from Bv coils is especially important at low A

Compare two tokamaks of similar plasma area and current, assuming typical values li=1, $\beta_p = 1$:

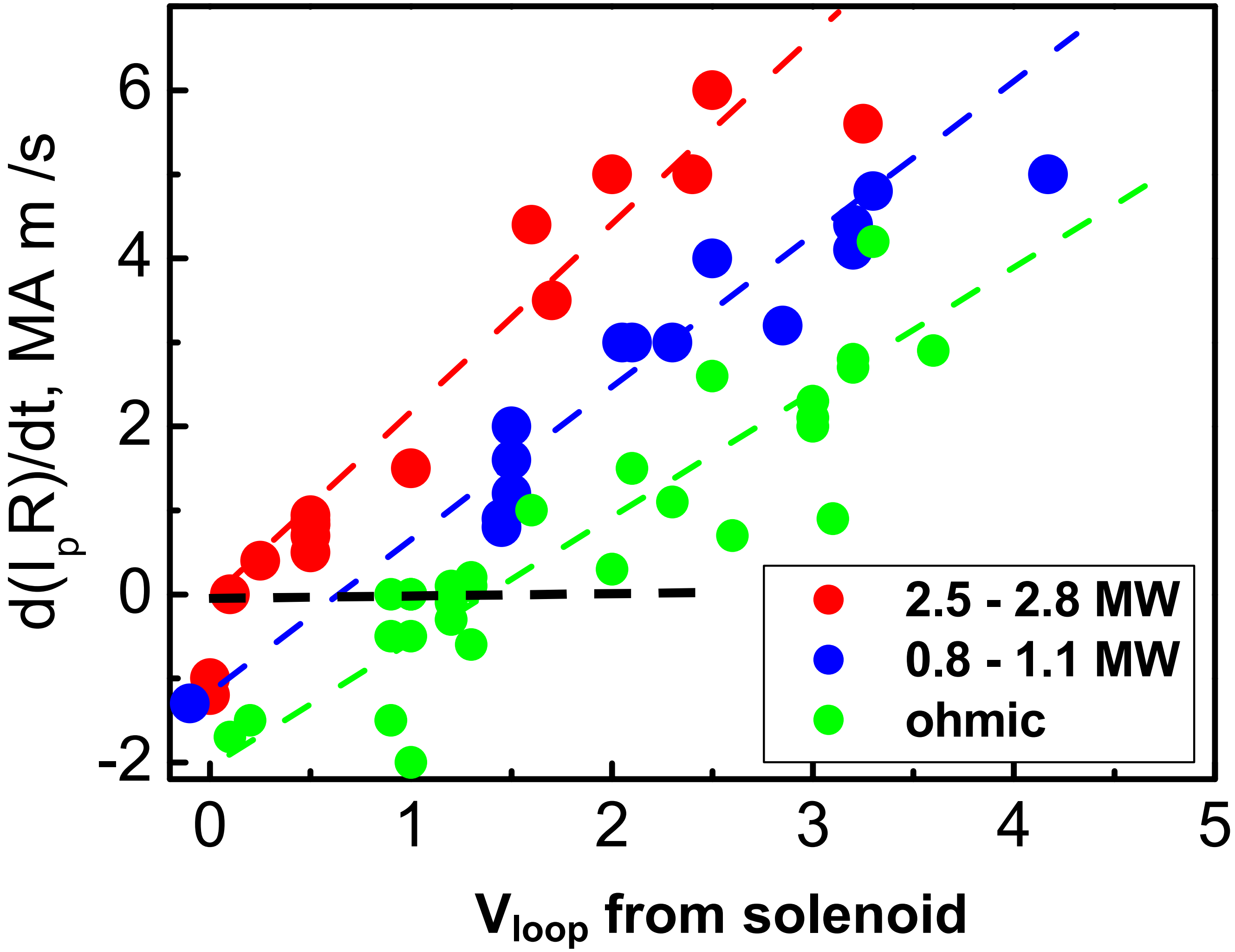
	A	k	R(m)	Ip (MA)	LI (Vs) Associated (1)	Vs Provided by Bv (2)	Vs available from solenoid
MAST	1.4	2	0.85	1.0	0.68	0.47	1
TFTR	3.5	1	2.5	1.0	5.5	2.5	12.5

For the ST, Bv flux is a big fraction **because the inductance of an ST plasma is so low**



(1) S P Hirshman & G H Nielson, Phys Fluids **29** (1986) p790
(2) O. Mitarai & Y Takase, Fusion Science & Technology, Nov 2002

Promising current ramp-up results from MAST



Plot shows current ramp rate in MAST produced by a range of V_{loop} from solenoid.

To maintain discharge:

Ohmic: V_{loop} ~ 1.4

Low power NBI: V_{loop} ~ 0.8

High power NBI: V_{loop} ~ 0.1

Implication is that for higher power NBI, MAST plasmas can be ramped up without central solenoid

Plans for the future

Major upgrades

An ST-based Component Test Facility (CTF)?

An ST Power Plant ?

A Q ~ 1 Physics Expt?

MAST, NSTX Upgrades plan to resolve main uncertainties of Fusion STs

Confinement scaling in STs may improve significantly due to:

(1) Increased B_T

(2) addition of density control (cryoplant, or Lithium)

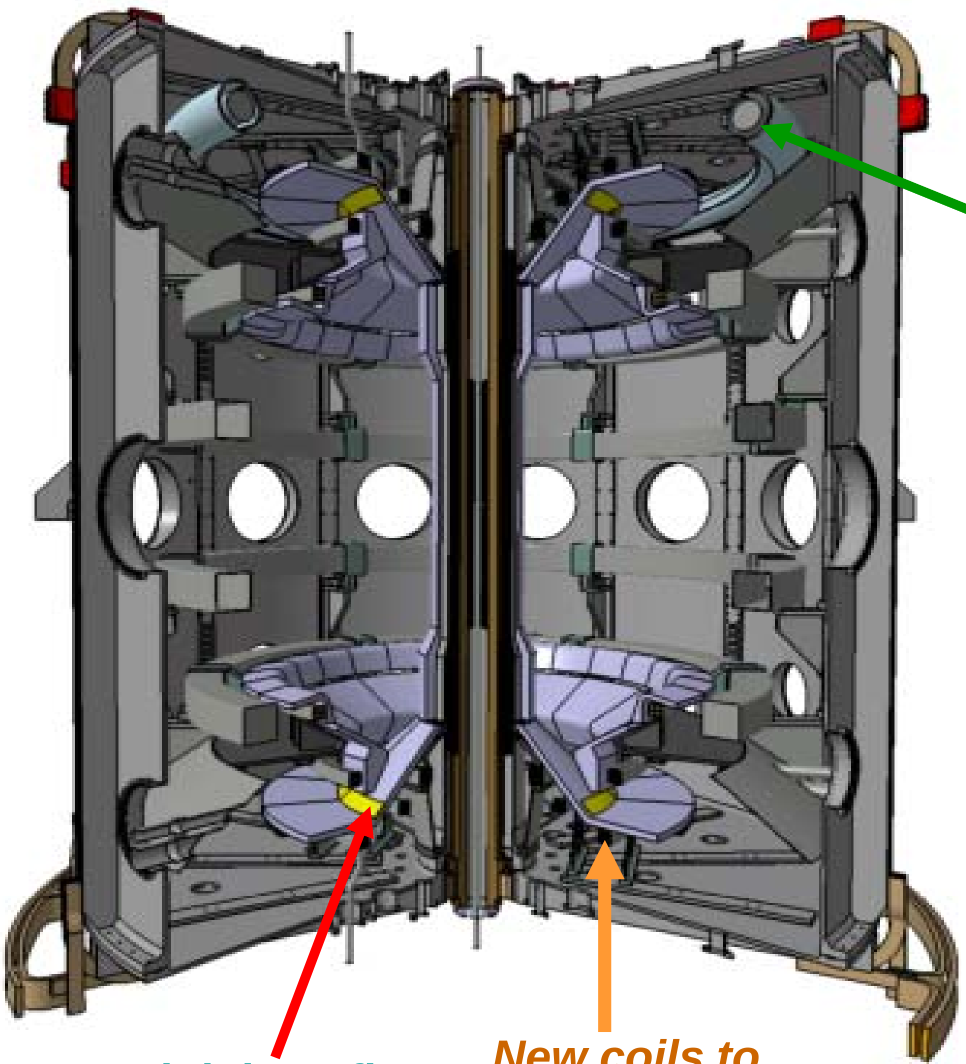
(3) ability to operate at $q_0 > 1$ (or 2) – hence eliminate sawteeth (NTMs)

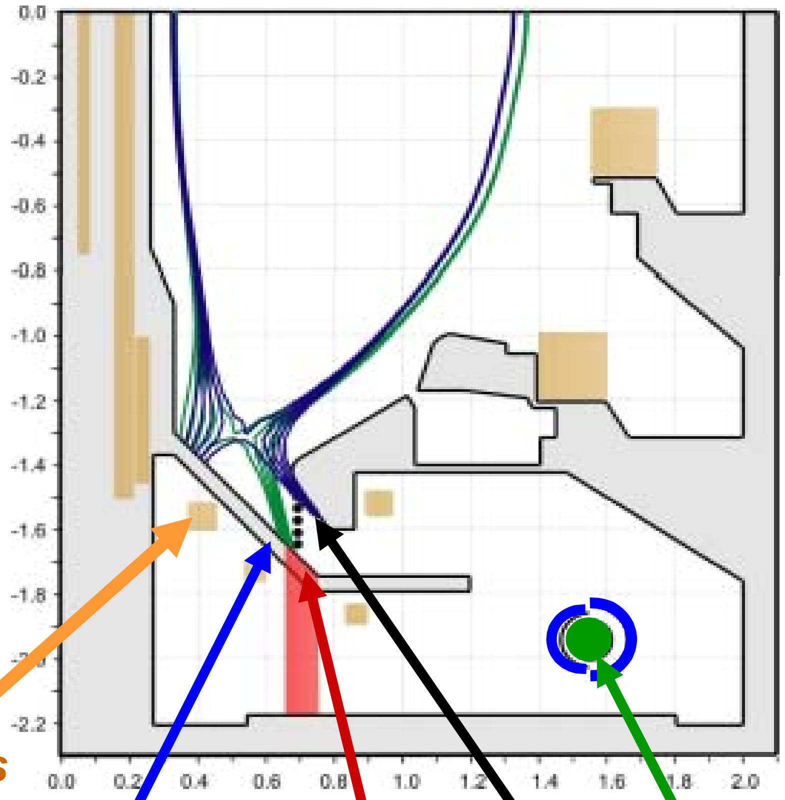
- and what is the H / L improvement?

Non-solenoid operation: demonstrate start-up, ramp-up and full non-solenoid current drive

MAST upgrade:

TF increased from 0.55T to 0.84T NBI power increased from 5MW to 12.5MW, giving full non-inductive current drive

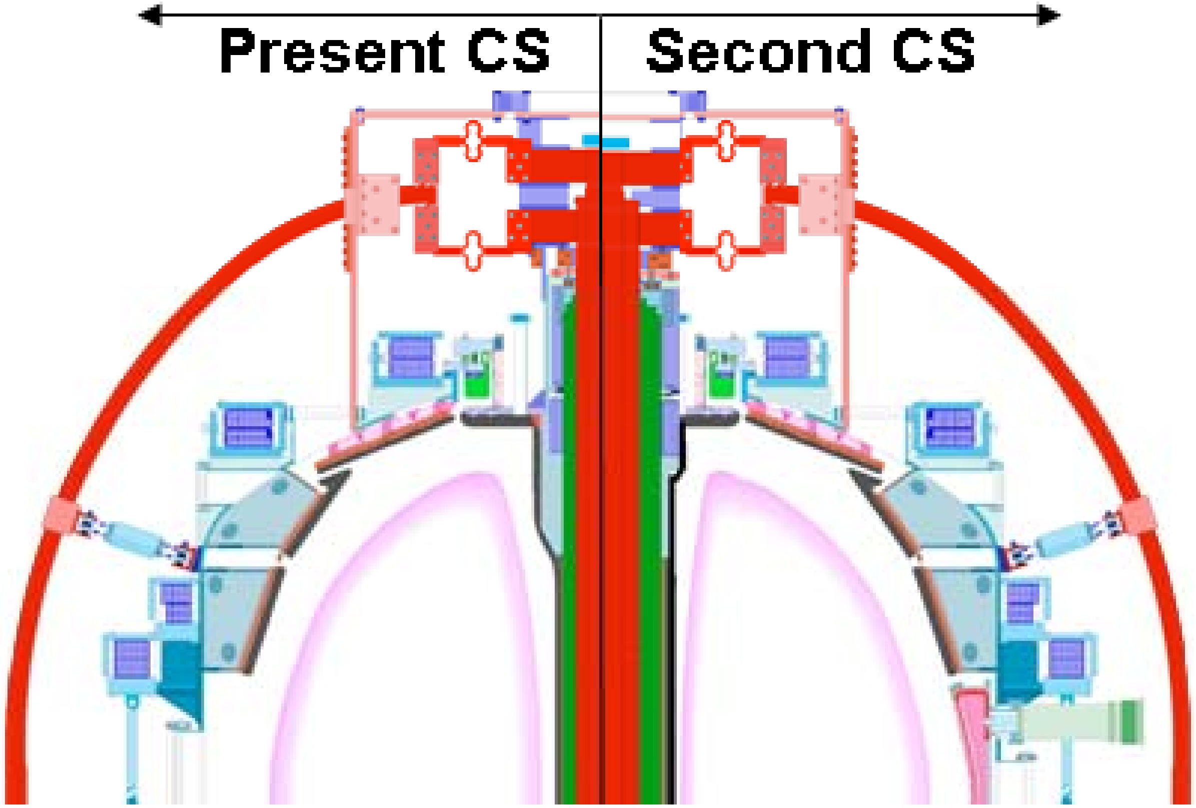


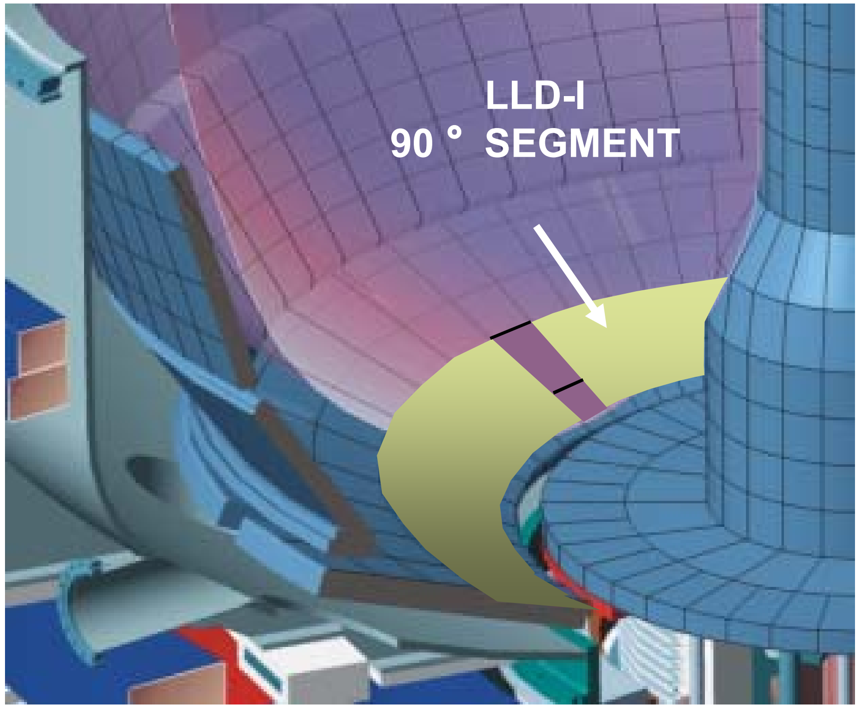


Capable of operation at $q_o > 2$ (so no sawteeth, no damaging NTMs)

Fusion Working with Europe

NSTX Upgrade : double B_T to 1T, add 2nd NBI





2nd NBI, tangential (should achieve 100% non-inductive current drive, and maintain qo > 1)

Lithium studies extended by LLD = Liquid Lithium Divertor

An ST-based Component Test Facility (CTF)?

There is an urgent need to test components for Fusion Power Plants under correct neutron load and thermal flux…

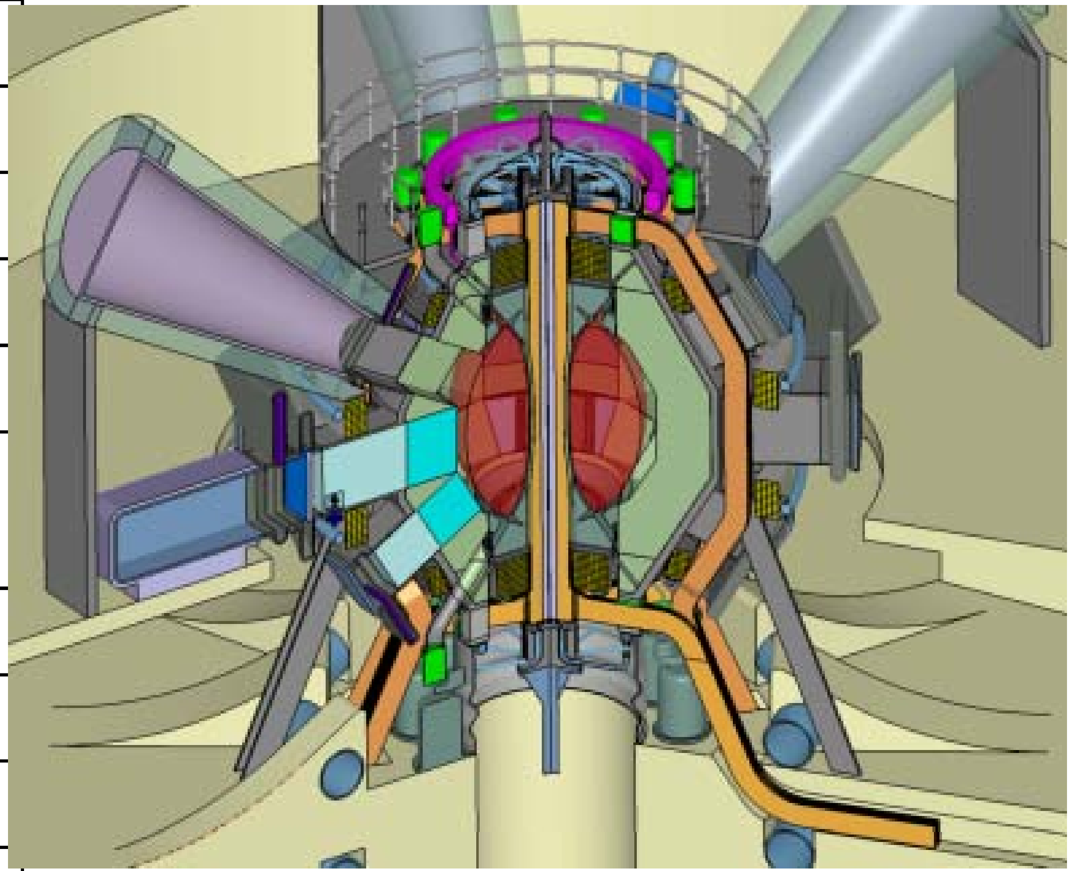
The conventional tokamak approach would require a JET-sized facility, complex and consuming much tritium, of which there is a world shortage…

A compact ST with simple centre column could provide good performance at minimum build cost with minimum requirement for tritium

Culham CTF design parameters

Parameter	ST-CTF
Major / minor radius	85/55cm
Elongation / triangularity	2.4/0.4
Plasma current/rod current	6.5/10.5MA
β_N	3.5
Average density	1.8×10^{20} m^{-3}
Average temperature	Te=6.5keV Ti=8keV
Confinement H98(y,2)	1.3
Auxiliary power	40MW
Fusion power (thermal + b-p)	35MW
Neutron wall loading	1MWm^{-2}

Ref: G Voss et al, ISFNT8 conf 2007, to appear in FED



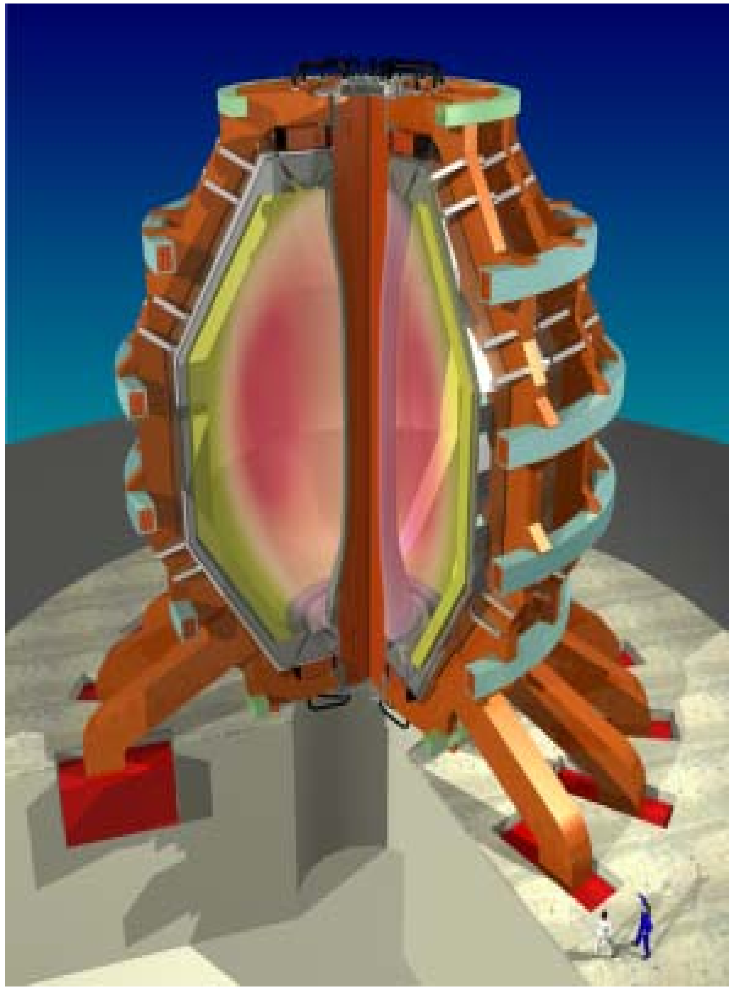
Tritium consumption ~ 0.6 kg/y

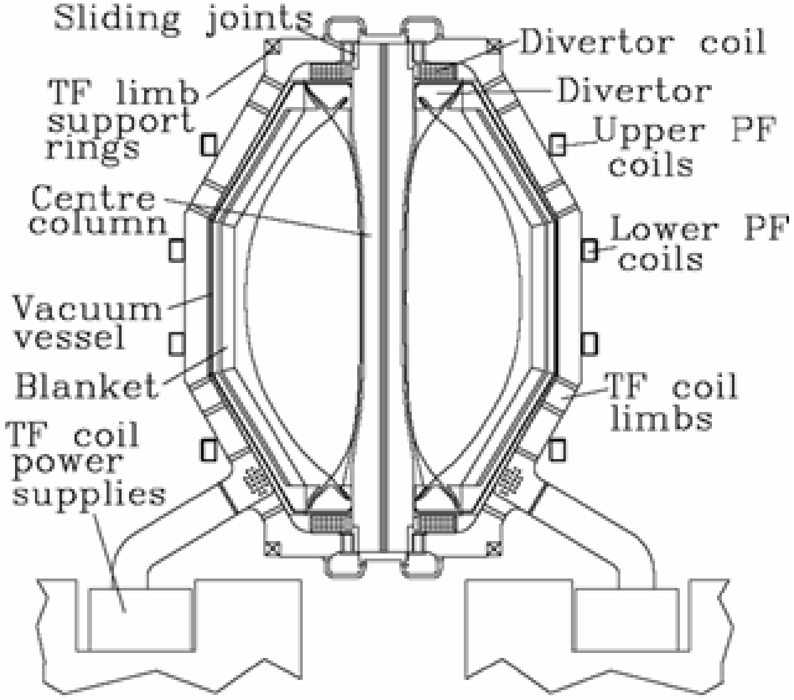
Features:

- ***Does not need to breed tritium***
- ***retractable small solenoid for start-up***
- ***2.5cm steel shield for c/col***

PPPL plan a larger CTF, with blanket for breeding tritium

ST Power plant - general features





$R/a = 3.4/2.4m$; $k = 3.2$

$I_p = 31MA$, $B_t = 1.8T$

$\beta_N = 8.2$, $P_{fusu} = 3.5$ GW

$Q = 50$, $P_{wall} = 3.5MW/m^2$

$f_{non\text{-}ind} = 0.95$

- Strong bootstrap current to minimise recirculating power
- copper single-turn TF, superconducting or cryogenic PF
- More advanced physics (wall mode control)

Physics: *Wilson et al, NF 2004*
Design, *Voss et al ISFNT 2000, 2002*

A Q ~ 1 ST Physics Experiment ?
(Q is the ratio: power output / power input)

The ST Power Plant is very demanding - due to the advanced physics required (on present scalings) to ensure that the Fusion power produced is economic

An ST Component Test Facility is a near-term possibility – but still a challenge – we need to provide steady-state operation at high wall load at minimum Tritium consumption.

For a medium-sized ST to merely demonstrate Q ~ 1 appears much easier…..

A Q ~ 1 ST ?

Consider an ST 50% larger than MAST, having R=1.2m, a=0.75m, with higher toroidal field 3T and current 4MA. From ITER pby2 scaling, the confinement factor will be x 36 at the same input power:

	B_T	Ip	R	$n_e e^{19}$	k	total
MAST	0.45	0.7	0.8	4	2.2	
STEP	3	4	1.2	20	3	
τgain	x1.3	x5.1	x2.2	x1.9	x1.27	x36

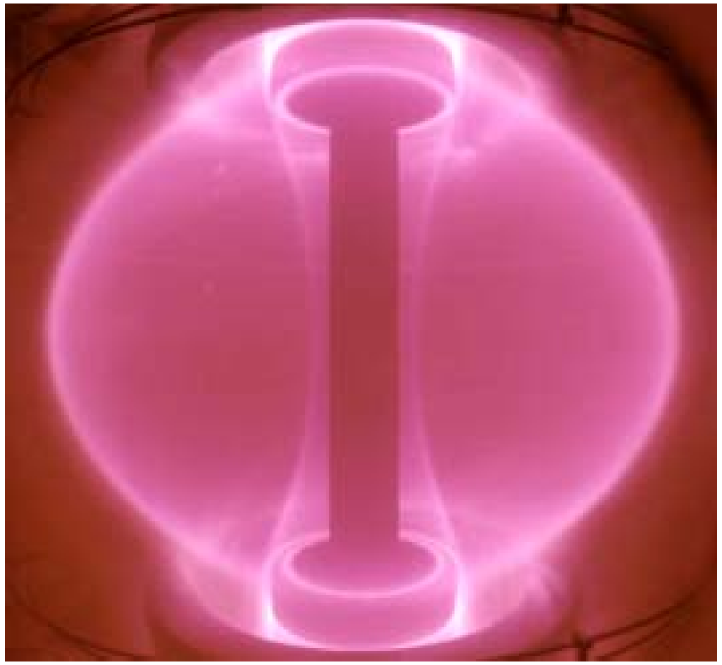
Modelling* shows such a device should achieve Q ~1 in L-mode, with a fusion power PF ~ 20MW giving a low average wall loading of 0.3MW/m^2

This device offers demonstration of Q ~ 1 with no ELMs, low wall loading, very low Tritium consumption *[so could JET!! – but at x10 volume]*

** R Galvao et al, to be presented at 2008 IAEA conf*

SUMMARY

STs have made a big impact on the Magnetic Fusion Programme

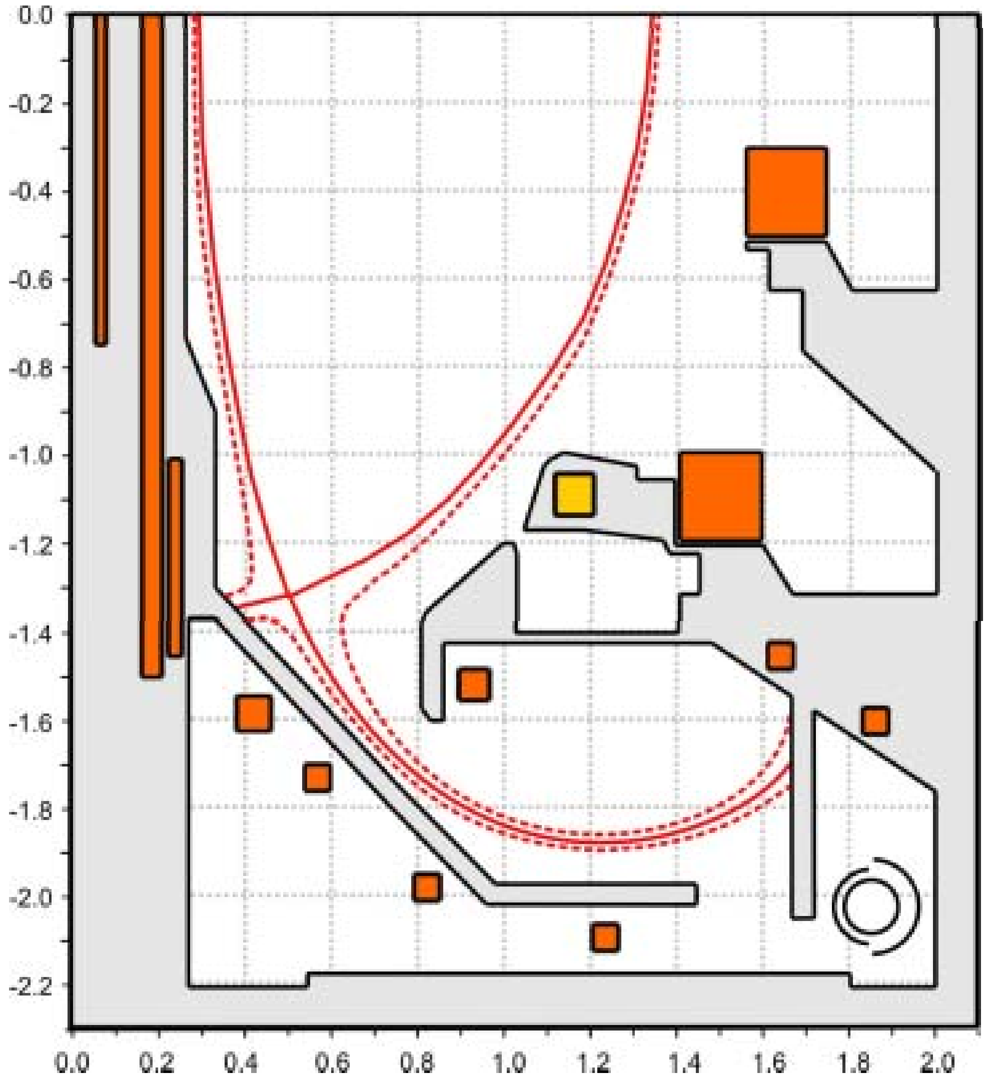


- Training researchers
- New insights into tokamak physics
- Development of scaling laws

They offer exciting possibilities for relatively economical SMALL Fusion Devices, e.g. Component Test Facility, a Q~1 physics expt…….

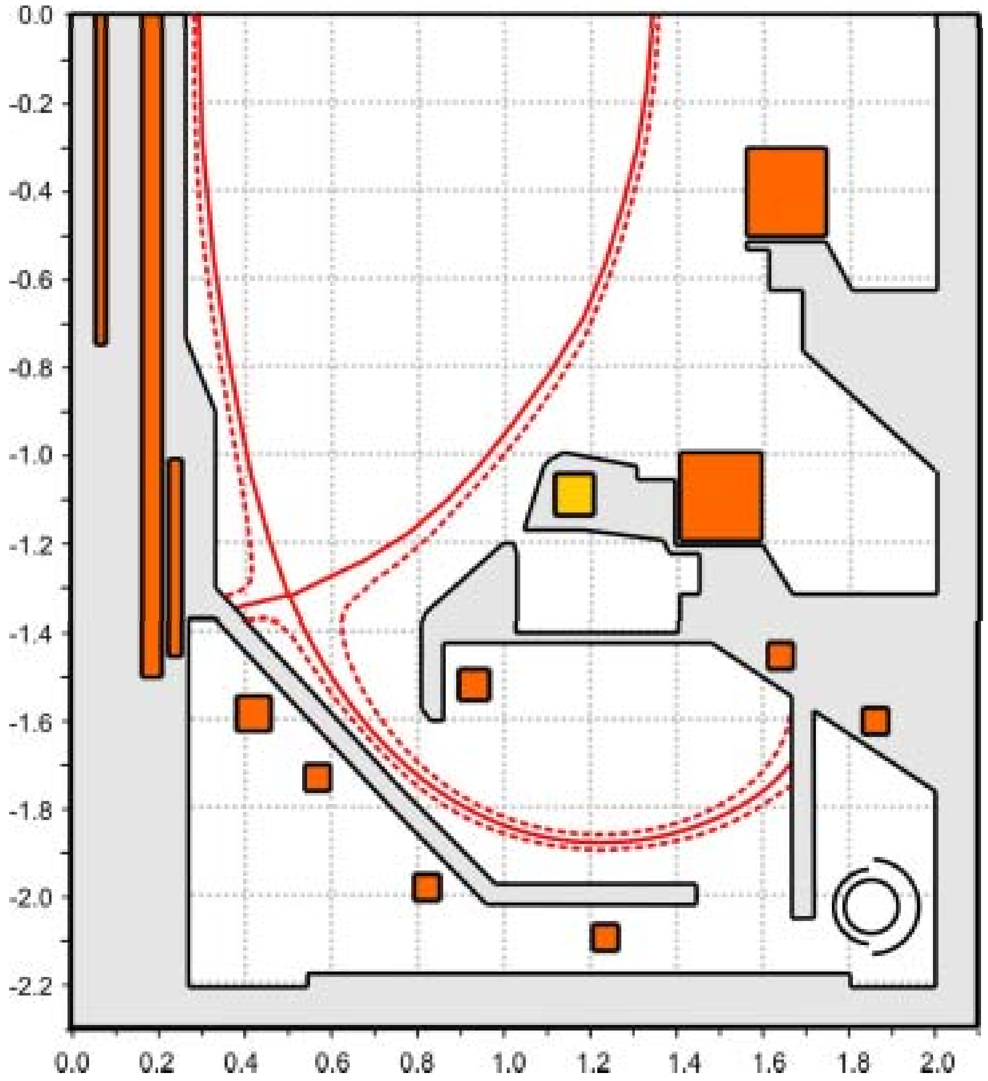
MAST upgrade: **divertor**

Option: space in MAST permits test of divertor-spreading concept, important for DEMO

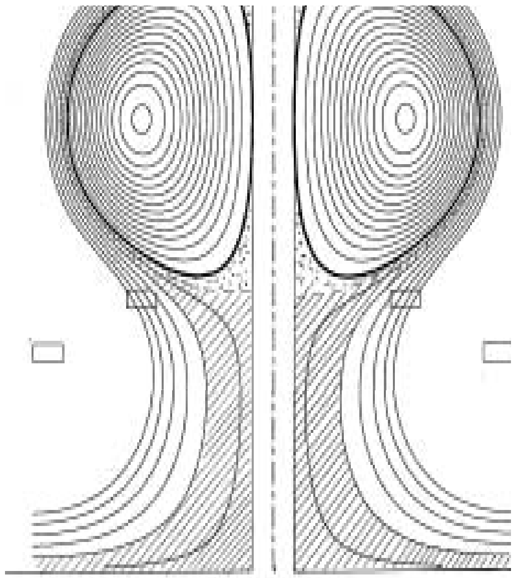


MAST upgrade: **divertor**

Option: space in MAST permits test of divertor-spreading concept, important for DEMO



Exploits the 'natural divertor' feature of STs !



As shown on START !