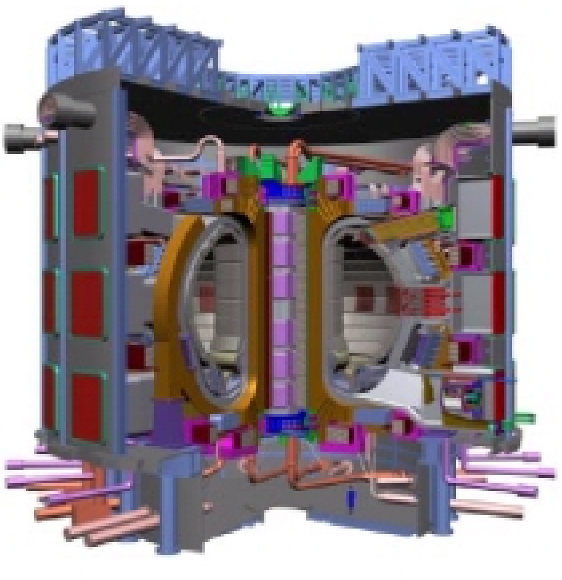
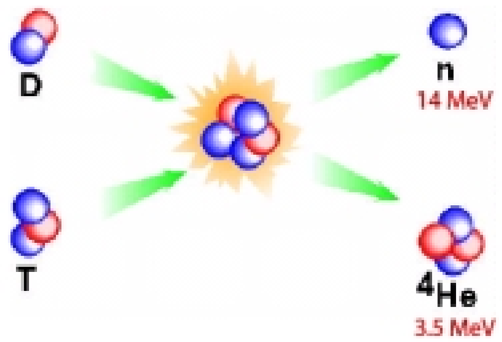




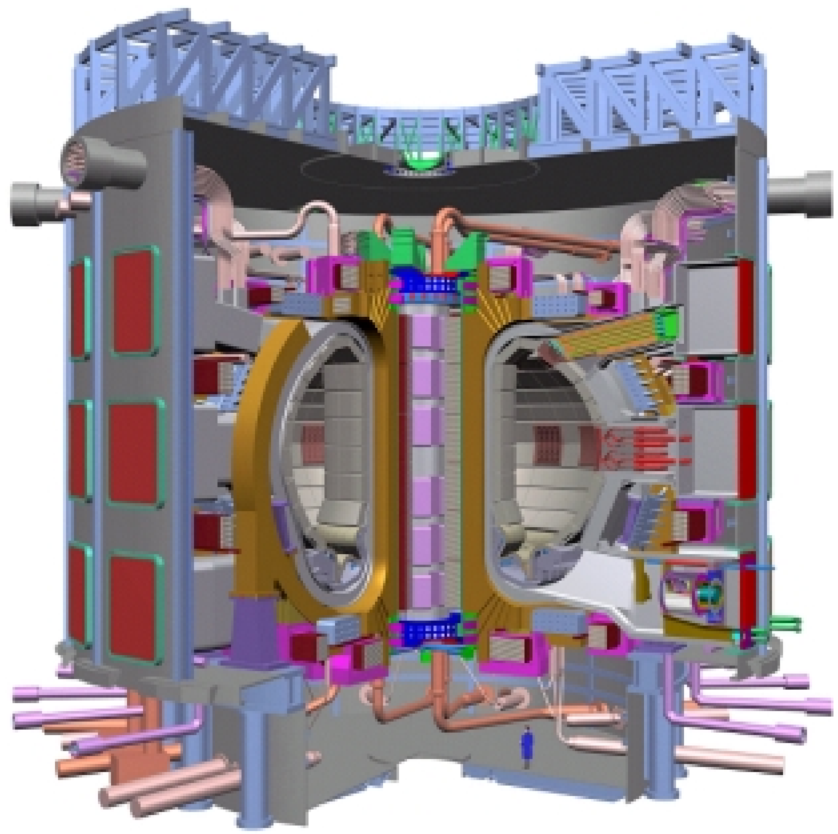
ITER: Realising the promise of Fusion Energy

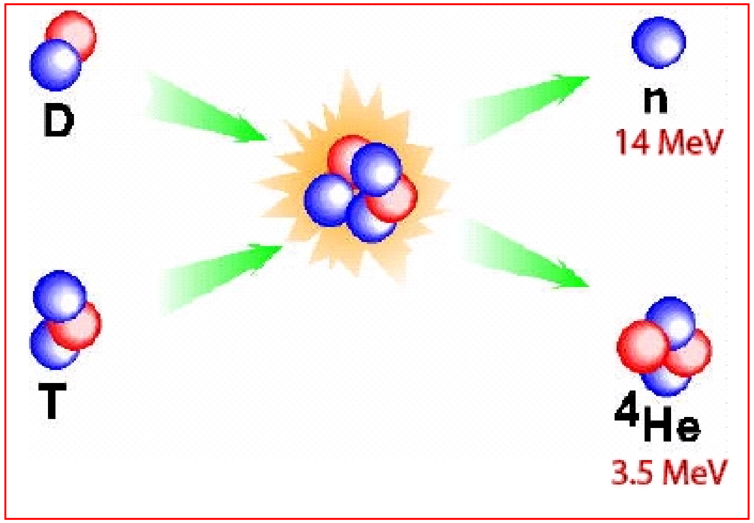
P. Barabaschi
for the ITER Team

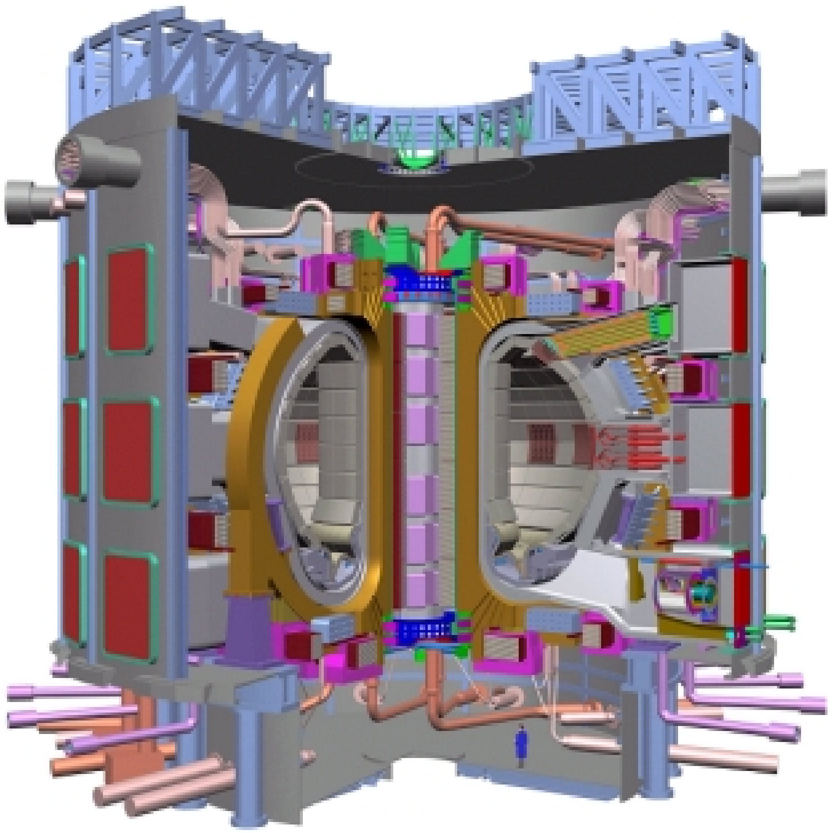
Budapest 28th August 2002
12th EPS General Conference

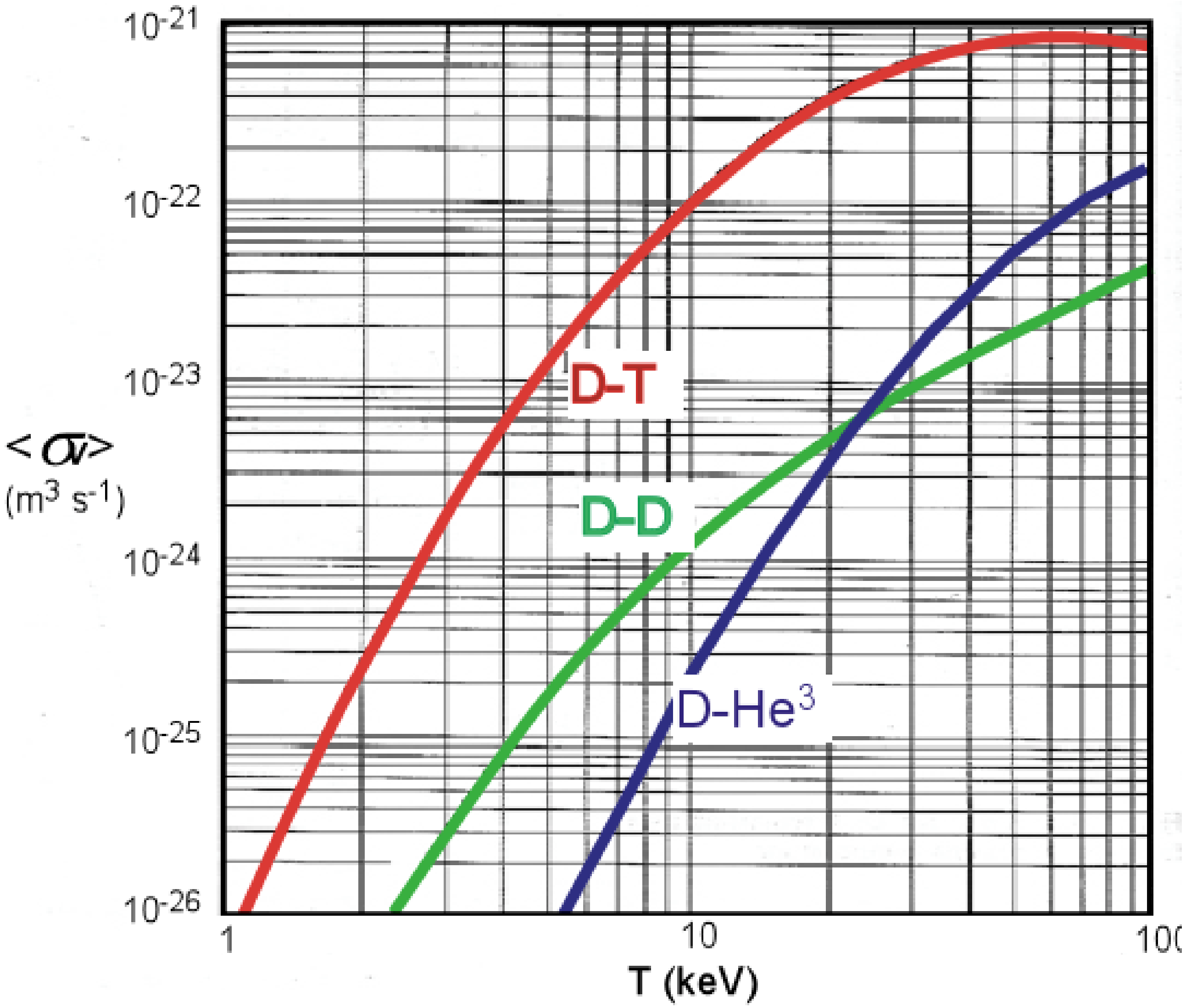
Synopsis

- **Introduction**
  - Nuclear Fusion
  - The Tokamak , ITER
- **Issues in Physics - Design**
  - Plasma shape - Magnet
  - Stability - Vessel and Blanket
  - Confinement
  - Alpha Particles
  - Diagnostics
- **Status of Negotiations, Sites**
- **Conclusions**







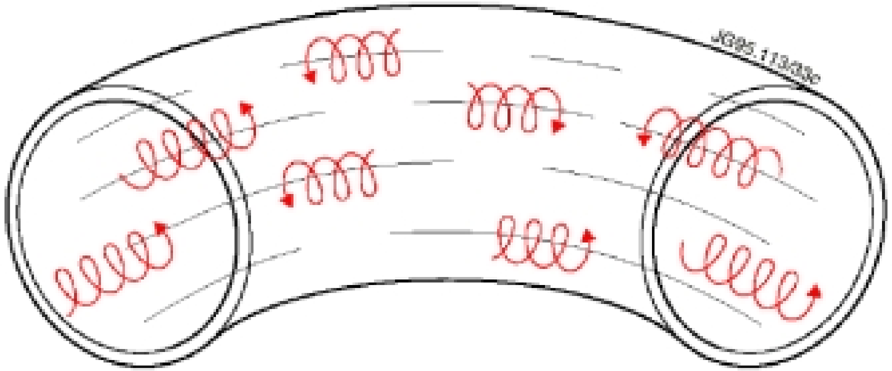


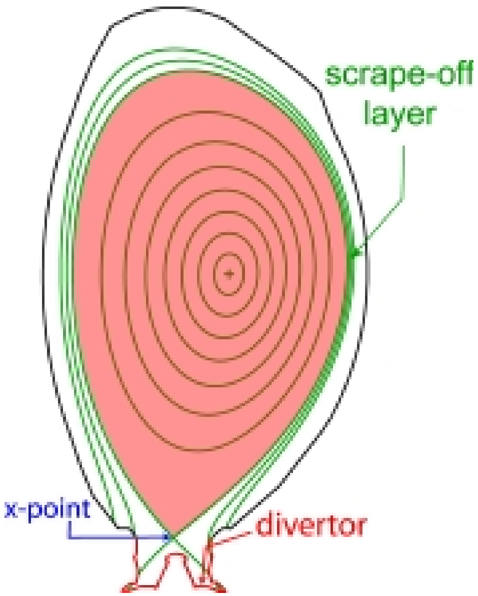
main goal:

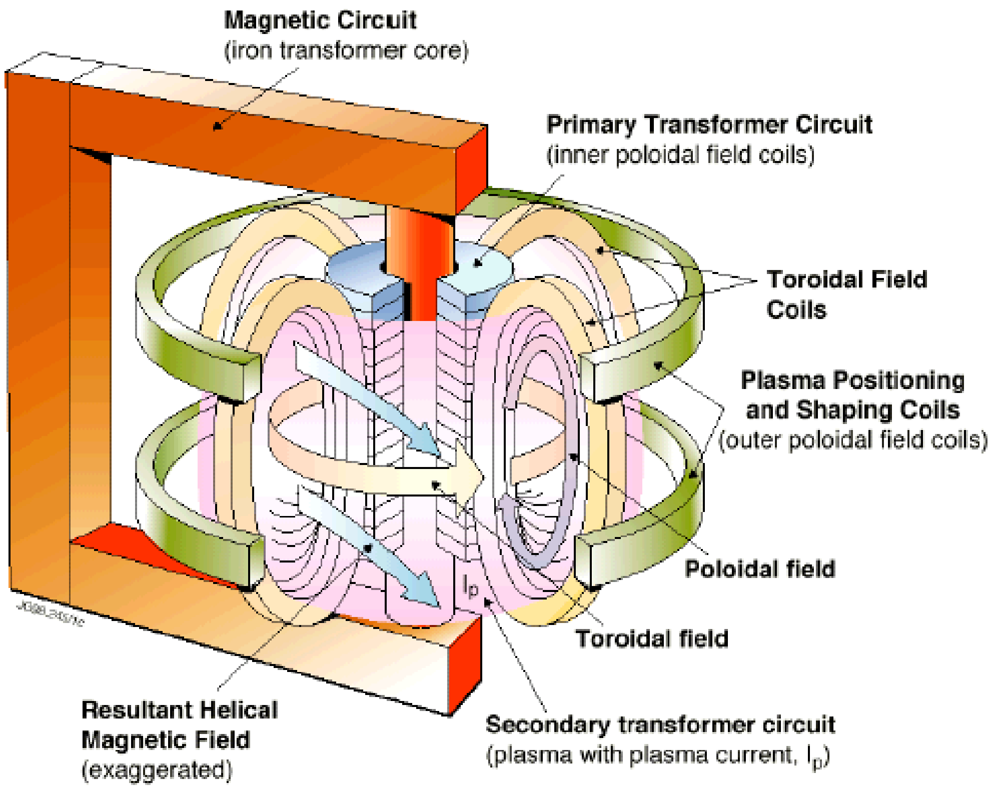
To demonstrate the feasibility of fusion energy for peaceful purposes

The Tokamak

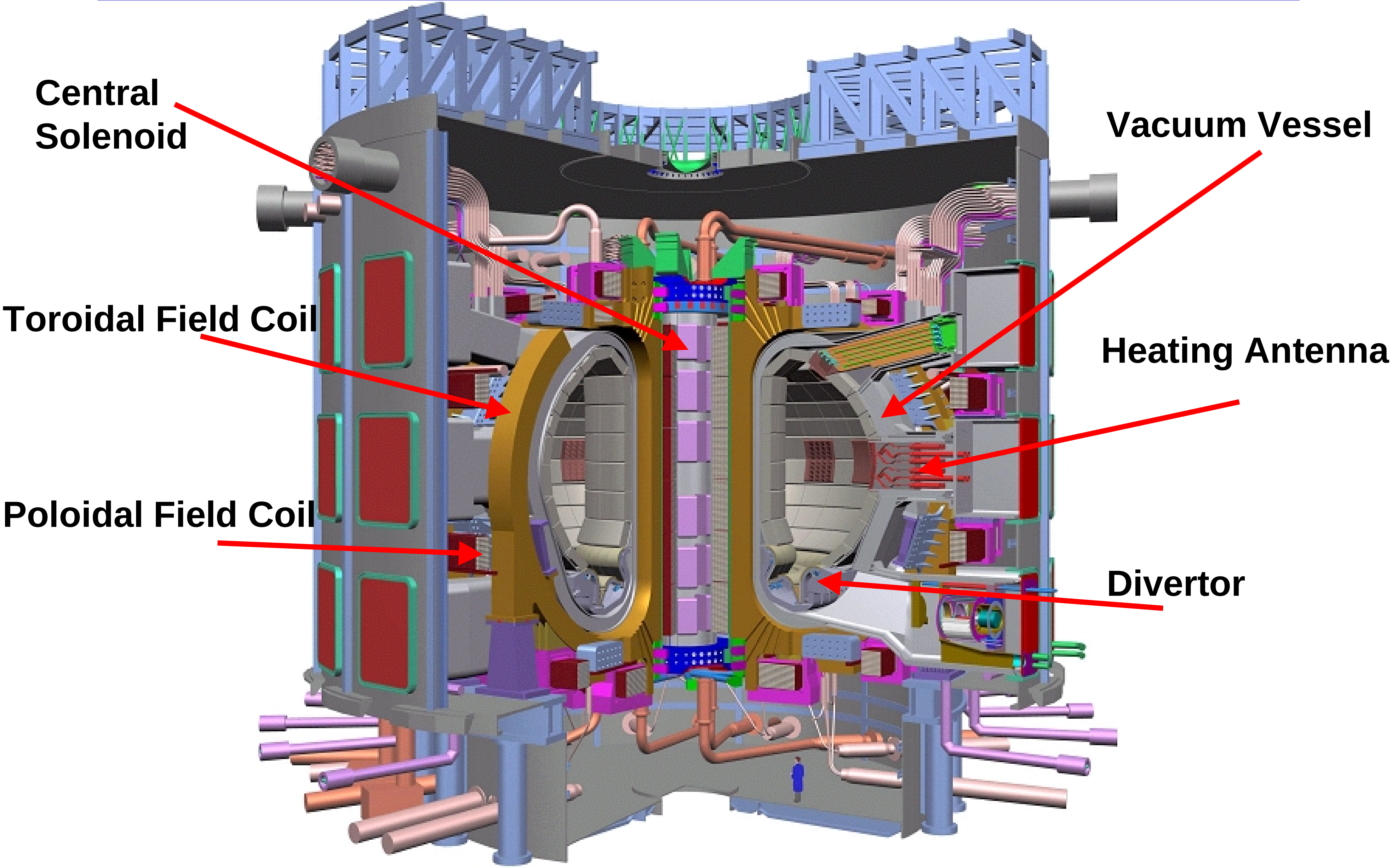
- Confinement: combination of an externally imposed **toroidal** field + **poloidal** field from plasma current.
- in a toroidal configuration plasma particles are lost to the vessel walls by relatively slow diffusion across the field lines.
- Magnetic forces = plasma pressure ($j \times B = \nabla P$)

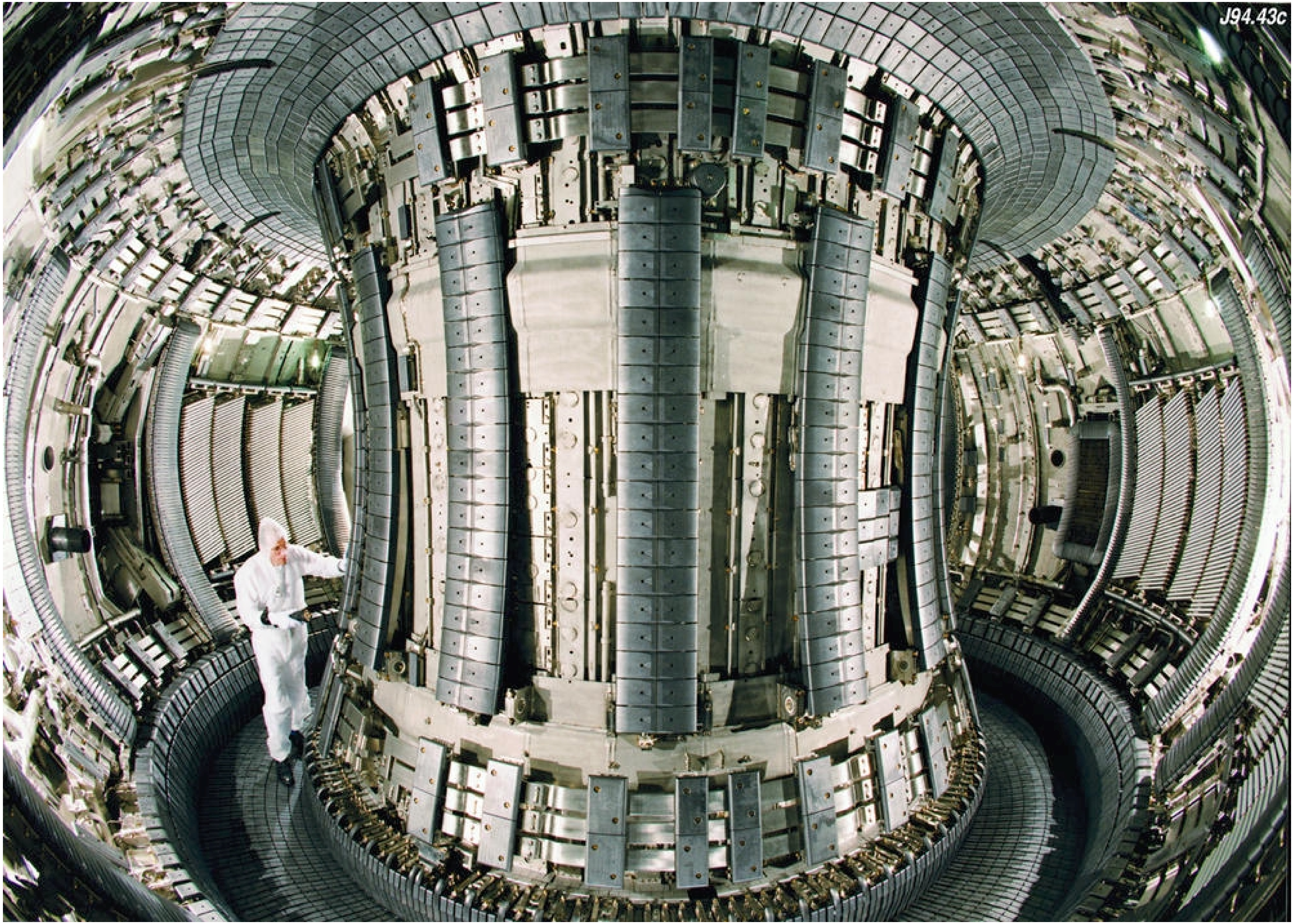


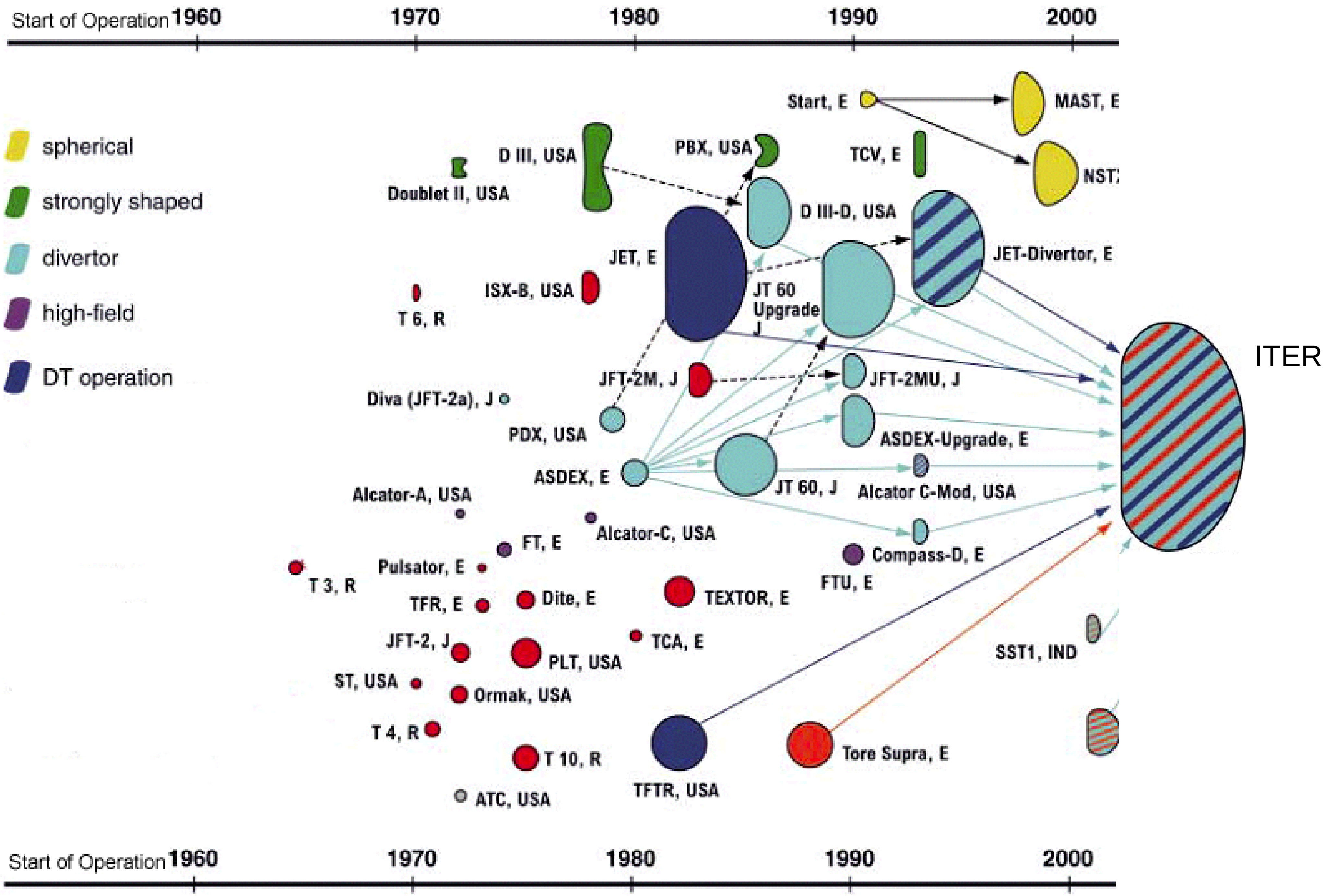




Overall view of ITER



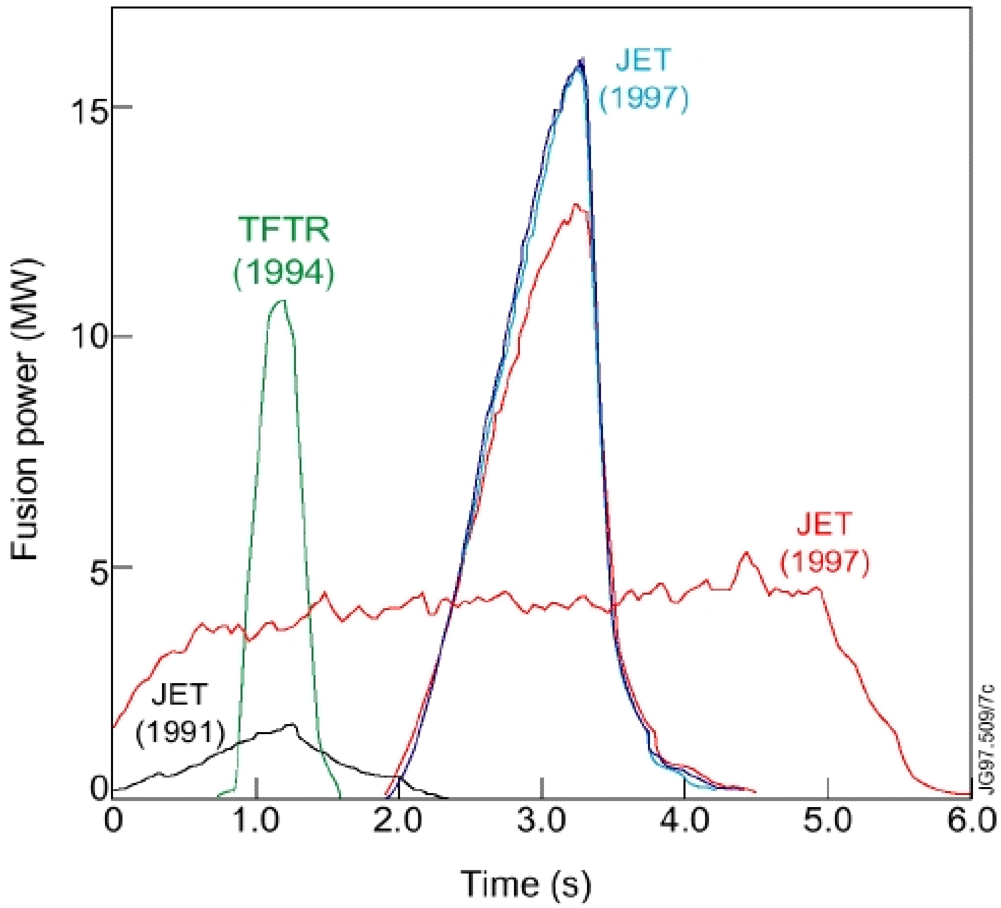

J94.43c


Start of Operation 1960
1970
1980
1990
2000
spherical
strongly shaped
divertor
high-field
DT operation
Start, E
MAST, E
D III, USA
PBX, USA
TCV, E
Doublet II, USA
D III-D, USA
JET, E
JET-Divertor, E
ISX-B, USA
JT 60 Upgrade J
T 6, R
ITER
JFT-2M, J
JFT-2MU, J
Diva (JFT-2a), J
PDX, USA
ASDEX-Upgrade, E
ASDEX, E
JT 60, J
Alcator C-Mod, USA
Alcator-A, USA
Alcator-C, USA
FT, E
Compass-D, E
T 3, R
Pulsator, E
FTU, E
TFR, E
Dite, E
TEXTOR, E
JFT-2, J
TCA, E
SST1, IND
PLT, USA
ST, USA
Ormak, USA
T 4, R
T 10, R
Tore Supra, E
TFTR, USA
ATC, USA
Start of Operation 1960
1970
1980
1990
2000

Fusion Power Production

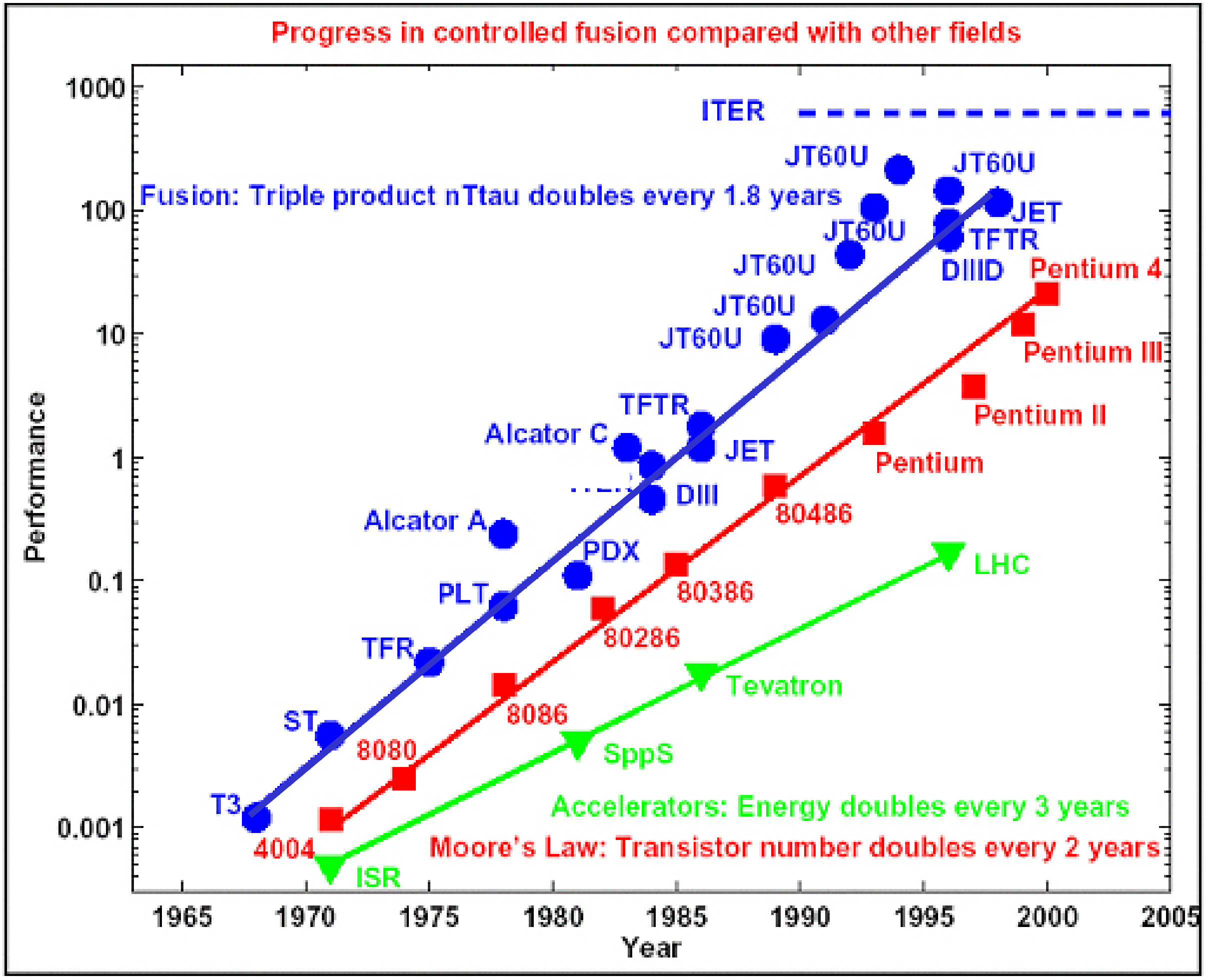
Experiments in JET and TFTR have initiated the study of DT plasmas with significant fusion power: best JET results correspond to a fusion power production of 16MW

α-particle heating amounted to ~15% of the input power to the plasma



Fusion Performance, $nT\tau$

- **Temperature (T_i):** **10-20 keV** (~10 × temperature of sun's core)
- **Density (n_i):** **$\sim 10 \times 10^{19}$ m^{-3}** (~10-6 of atmospheric particle density)
- **W confinement time (τ):** **few seconds** (plasma pulse duration ~500s)

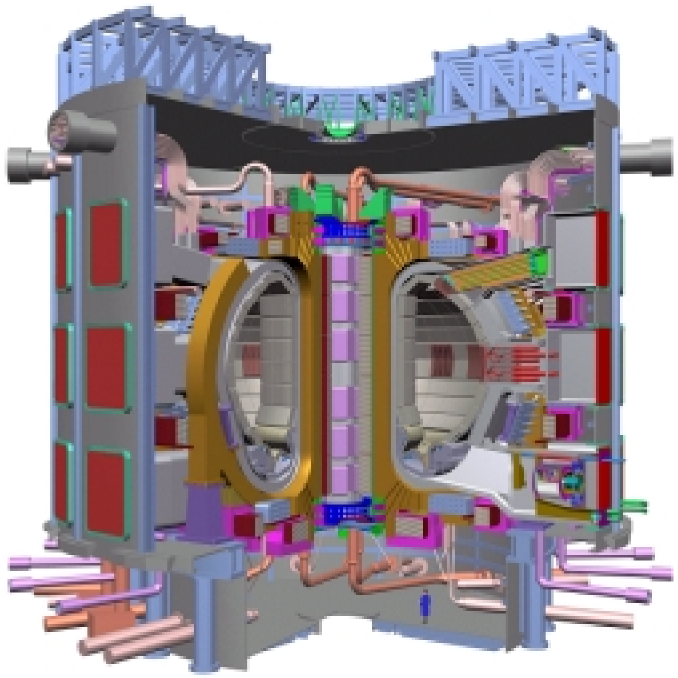


J Lister 2001 CRPP

$$\frac{P\alpha}{Pheat} \sim nT\tau$$

ITER background

- ITER is a major international collaboration involving the EU (plus Hungary, Switzerland,…), Canada, Japan and the Russian Federation.
- The overall programmatic objective is to “demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes”



ITER is designed to confine a DT plasma in which α-particle heating dominates all other forms of plasma heating:

a burning plasma experiment

Summary of ITER Challenges

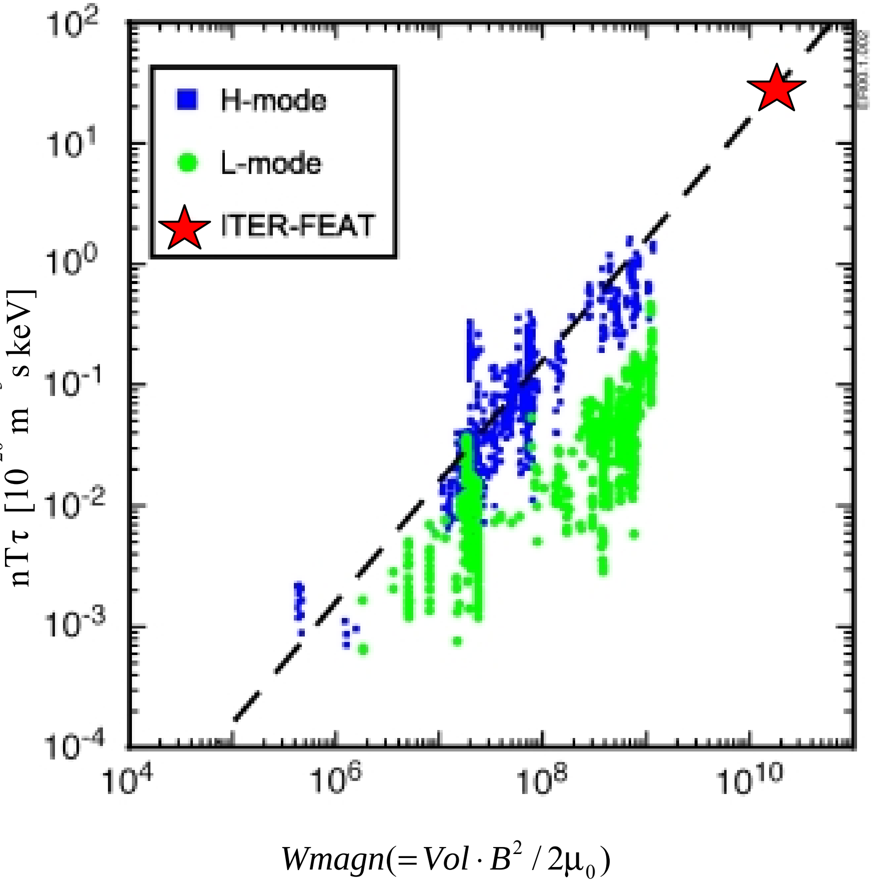
- **Long pulse burning plasma with Q>10 (Pα/Pheat>66%)**
  - Study alpha particle physics
  - Study controllability of fusion reaction
  - Study advanced regimes aiming at steady state operation
  - Study divertor operation (impurity control) in reactor conditions

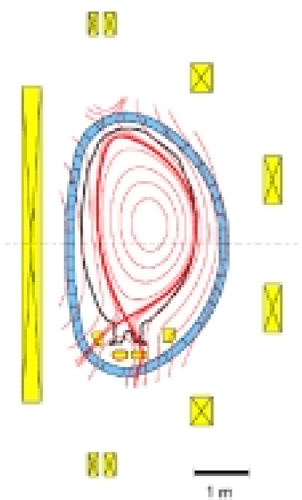
……In an integrated way

- **Demonstrate physics-technology integration and technologies essential to a fusion reactor in an integrated system**

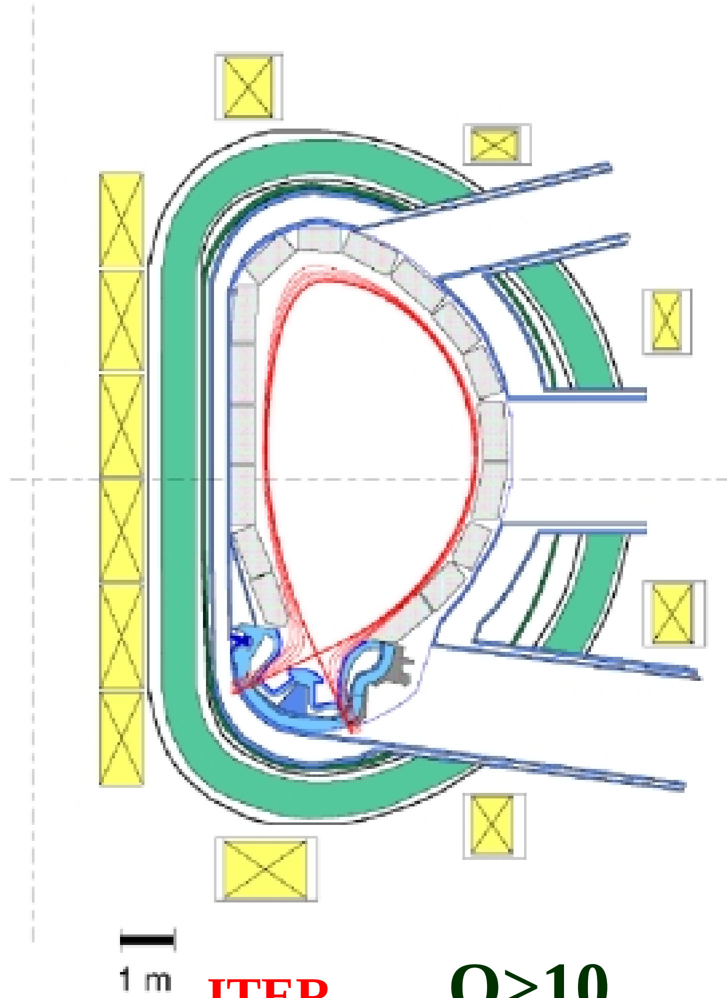
What size is right??

- $nT\tau \propto (H\ Ip)^3$
- Ip must be 3 times larger
- Size also because of new requirements





JET
R=3m
Ip=4MA

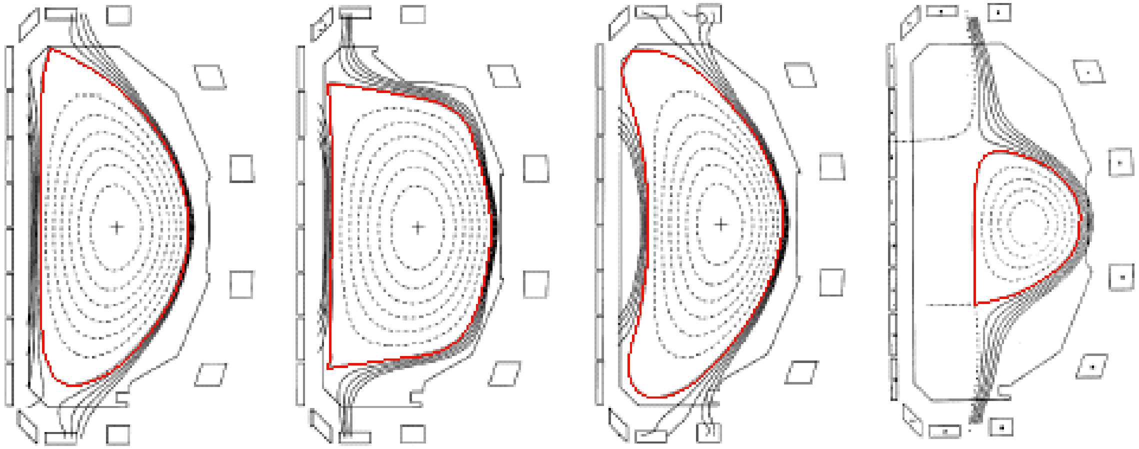


ITER
R=6.2m
Ip=15MA

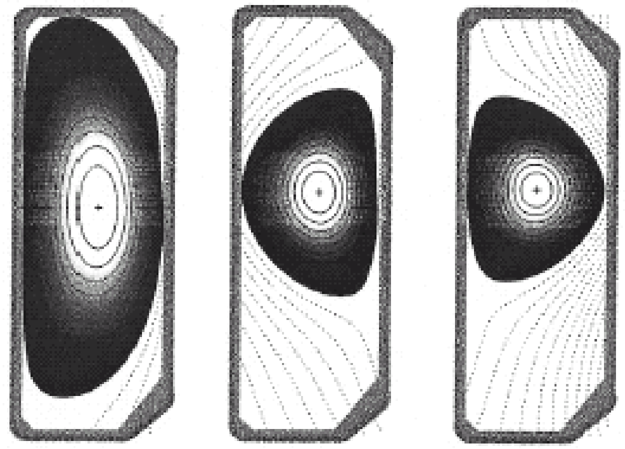
Q>10

Pfus~500 MW

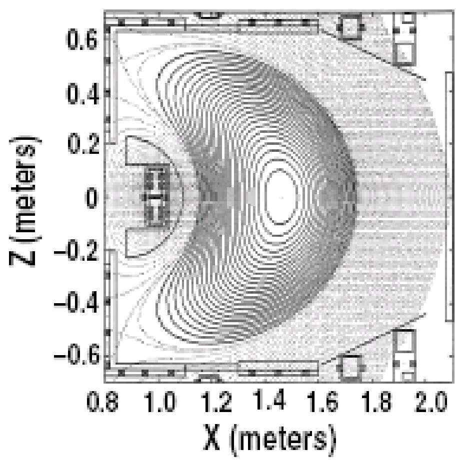
Plasma Equilibrium



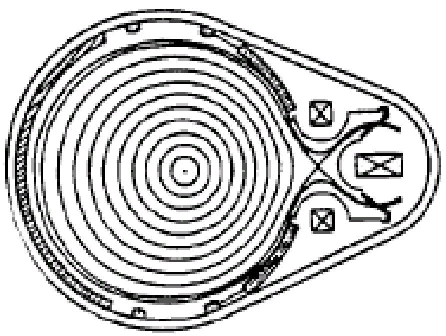
DIII–D



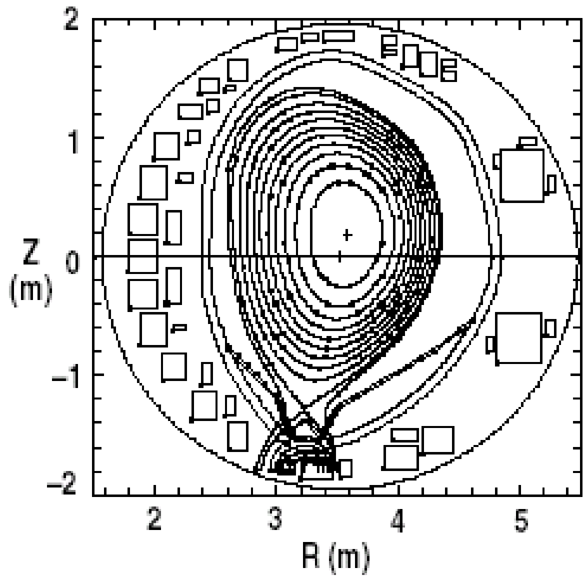
TCV



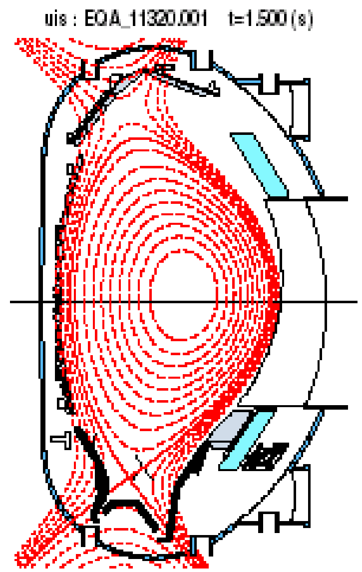
PBX–M



JT-60



JT–60U



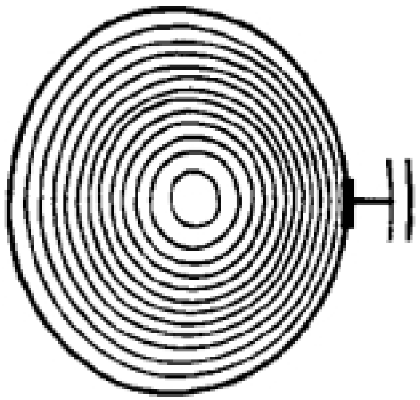
ASDEX UPGRADE

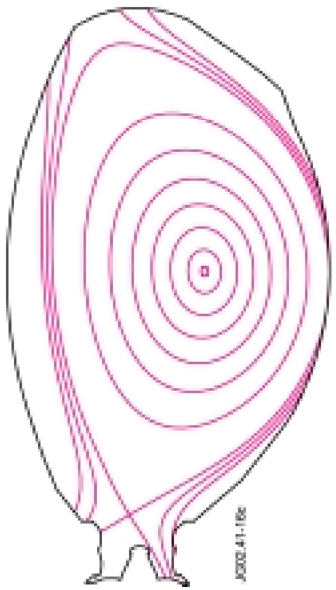
R. Stambaugh

Plasma "Shaping" driver of performance

Plasma shaping: Elongation, Triangularity, X-Point

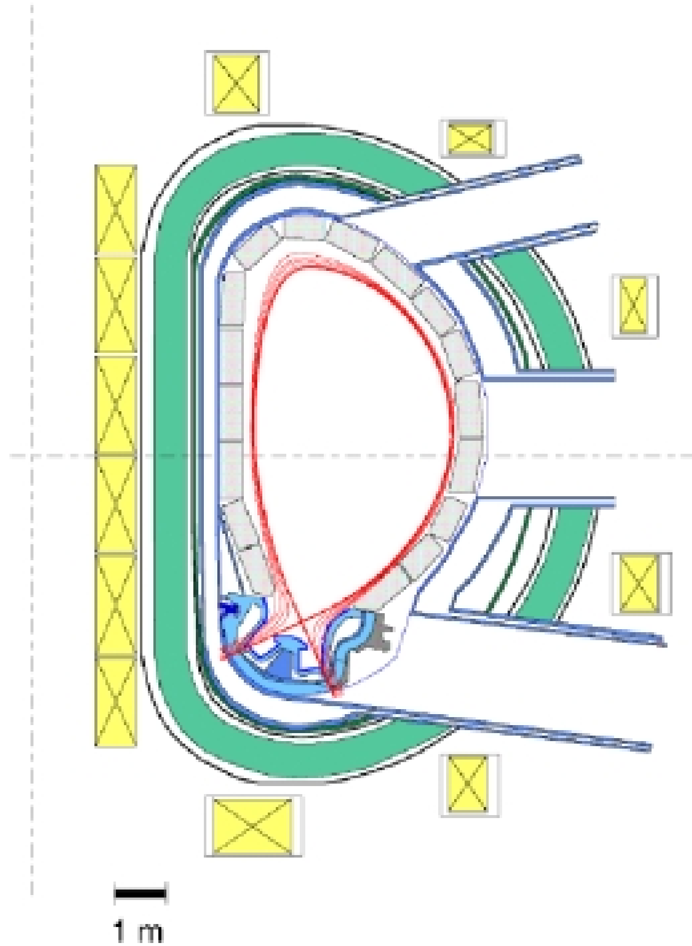
- => Confinement, H-Mode
- => improved stability
- => higher edge density
- => improved impurity control
- => Design of structure (e.g. TF magnet)



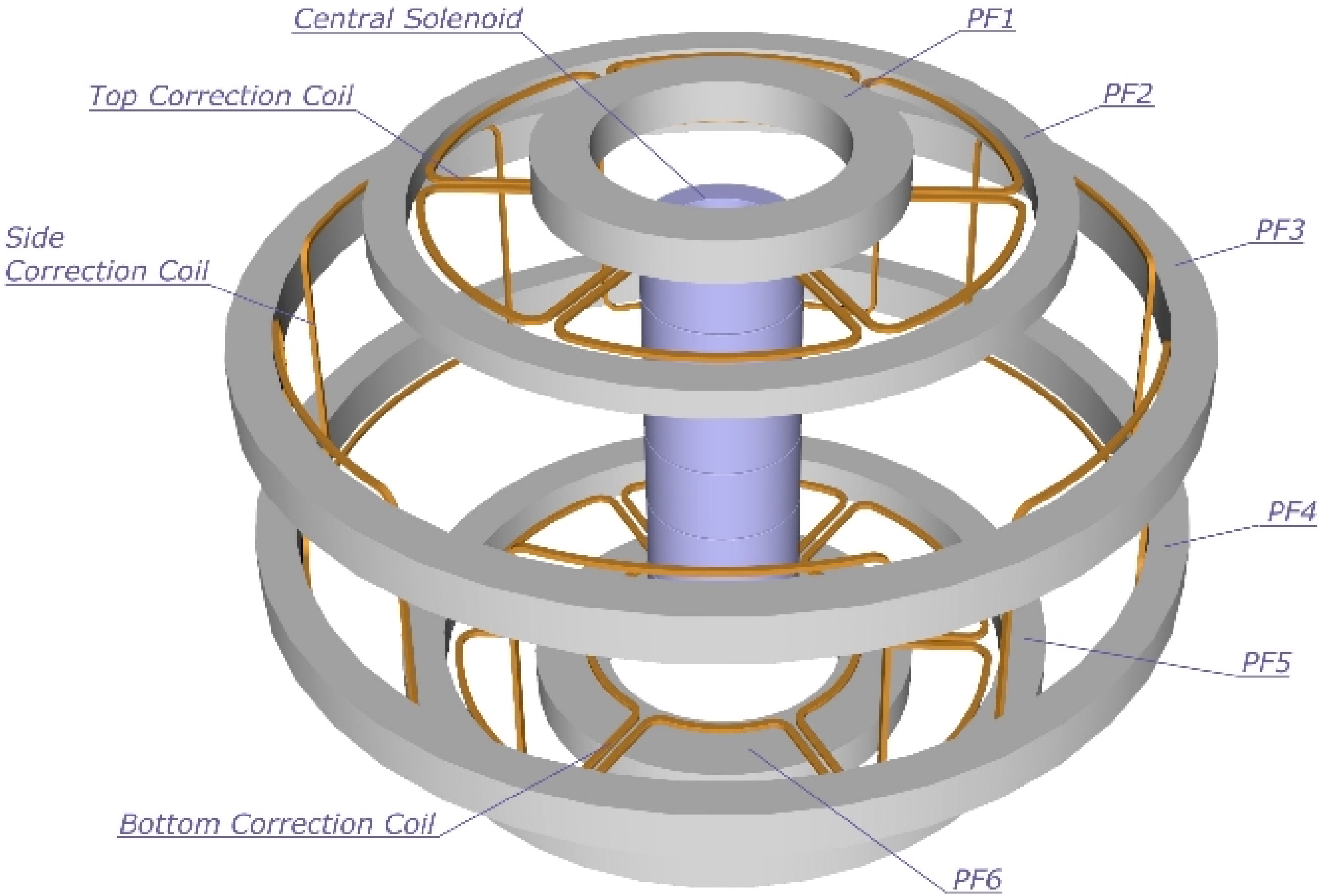


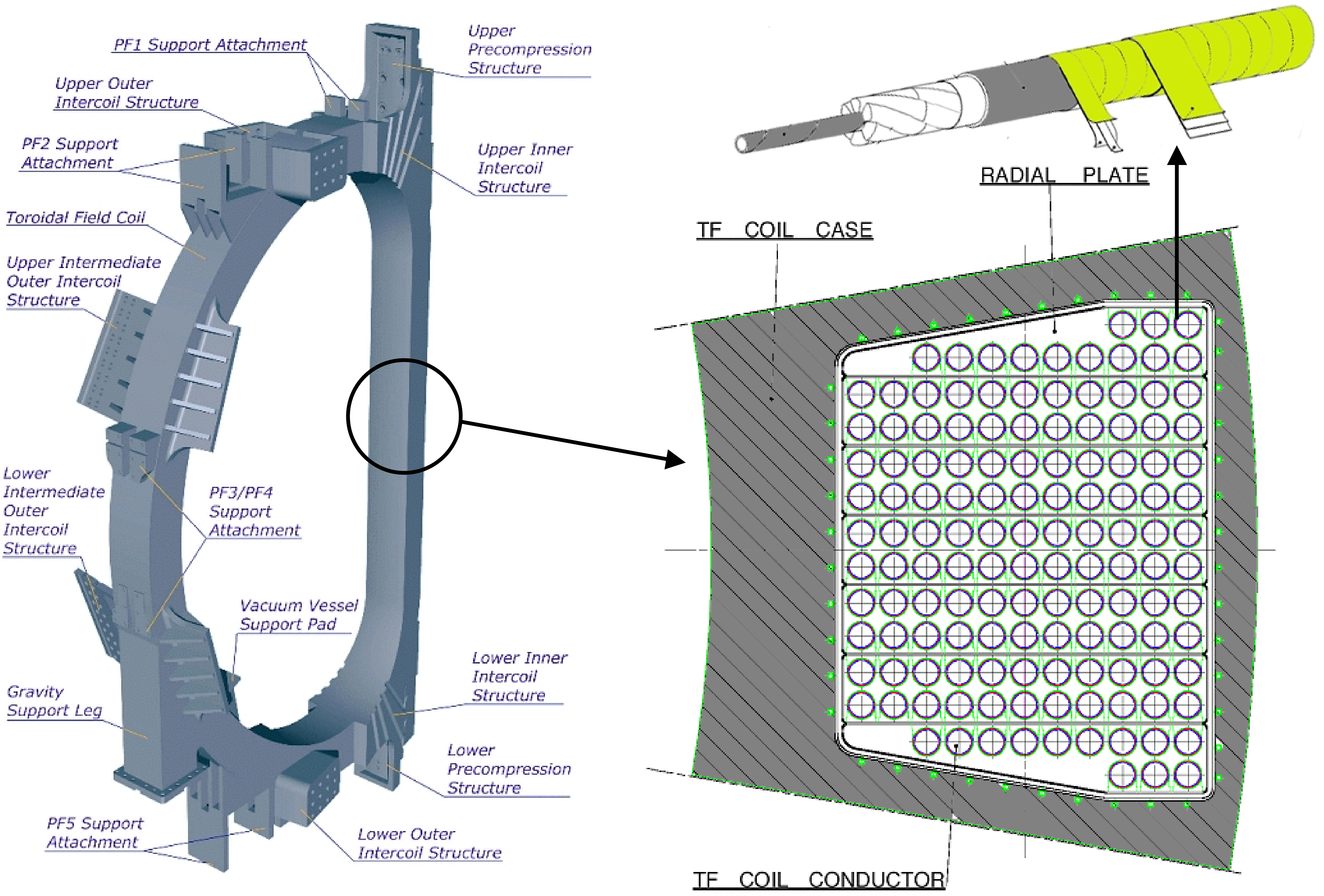
JET

High δ

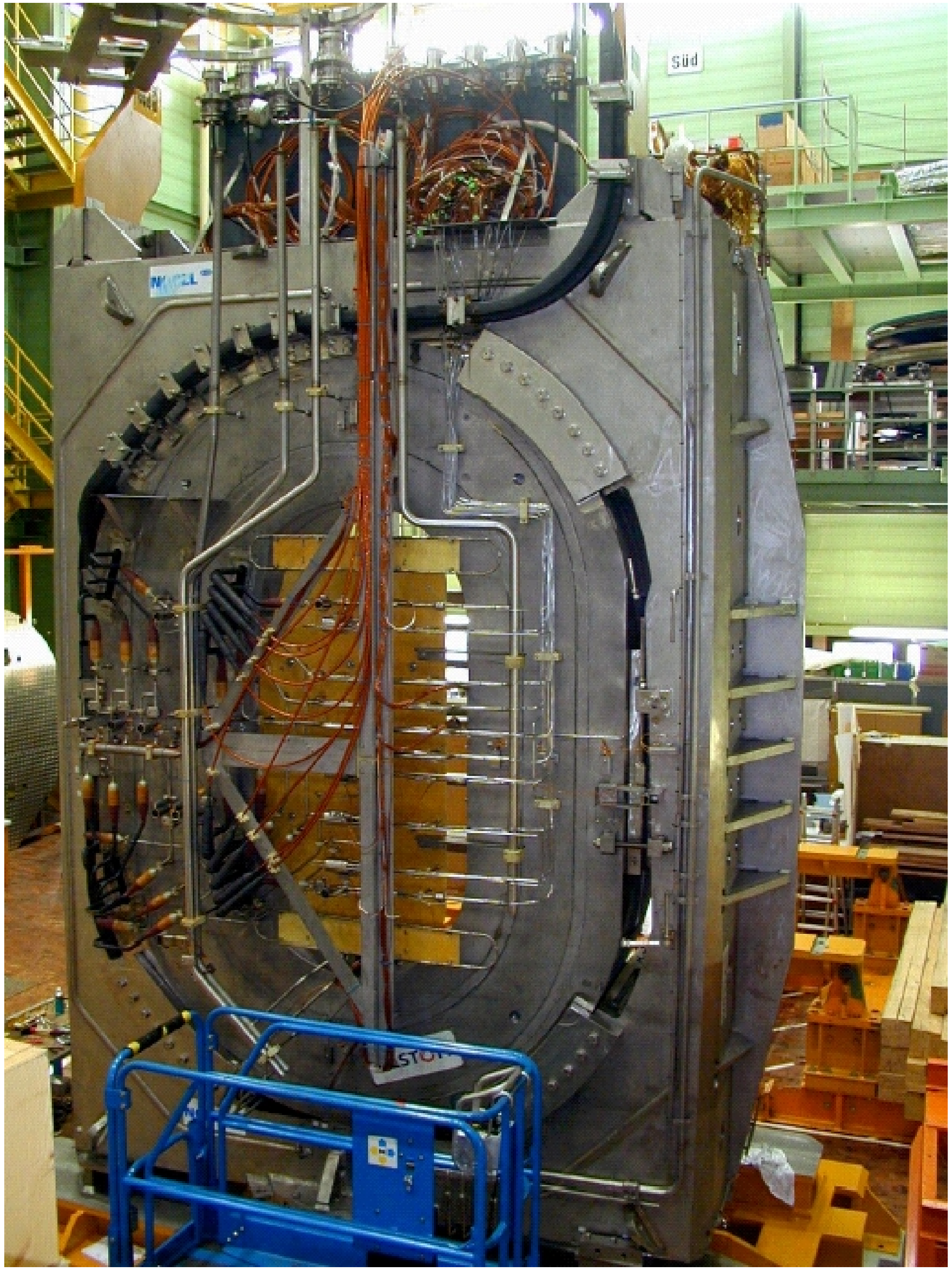


PF system


Central Solenoid
PF1
Top Correction Coil
PF2
Side
Correction Coil
PF3
PF4
PF5
Bottom Correction Coil
PF6



PF1 Support Attachment
Upper Precompression Structure
Upper Outer Intercoil Structure
PF2 Support Attachment
Upper Inner Intercoil Structure
Toroidal Field Coil
Upper Intermediate Outer Intercoil Structure
Lower Intermediate Outer Intercoil Structure
PF3/PF4 Support Attachment
Vacuum Vessel Support Pad
Lower Inner Intercoil Structure
Gravity Support Leg
Lower Precompression Structure
PF5 Support Attachment
Lower Outer Intercoil Structure
RADIAL PLATE
TF COIL CASE
TF COIL CONDUCTOR



TF Model Coil

CS Model coil

Plasma Instabilities

◆ **Limit current** $q_c = \dfrac{aB_\phi}{RB_p}$ - safety factor (q ~ 3)

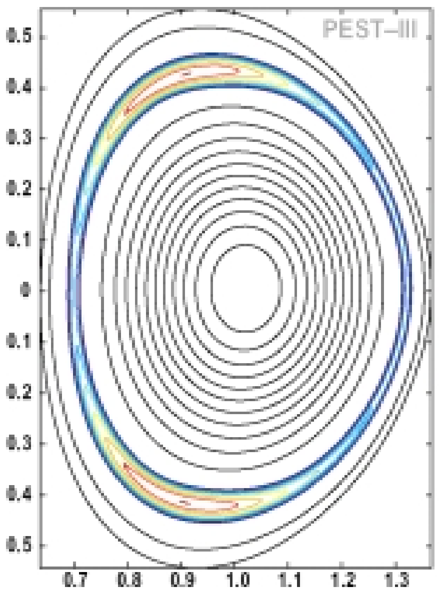
◆ **Limit pressure (β)**

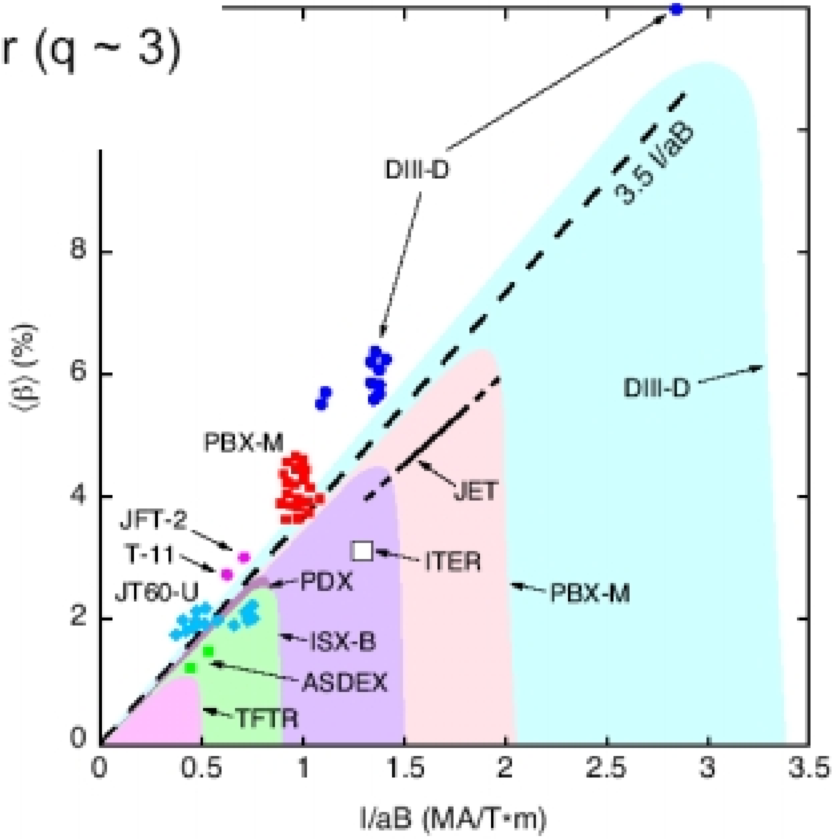
$$\beta(\%) = 100\frac{<p>}{B^2/2\mu_o} \qquad \beta_N = \frac{\beta(\%)}{I_p(MA)/aB}$$

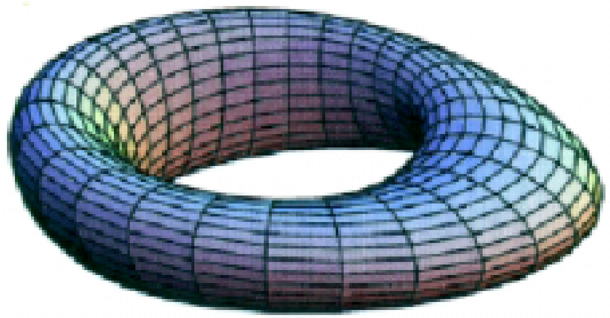
($\beta_N \leq$ 2.8-3.5 - "Troyon" limit)

◆ **Limit pressure gradient**

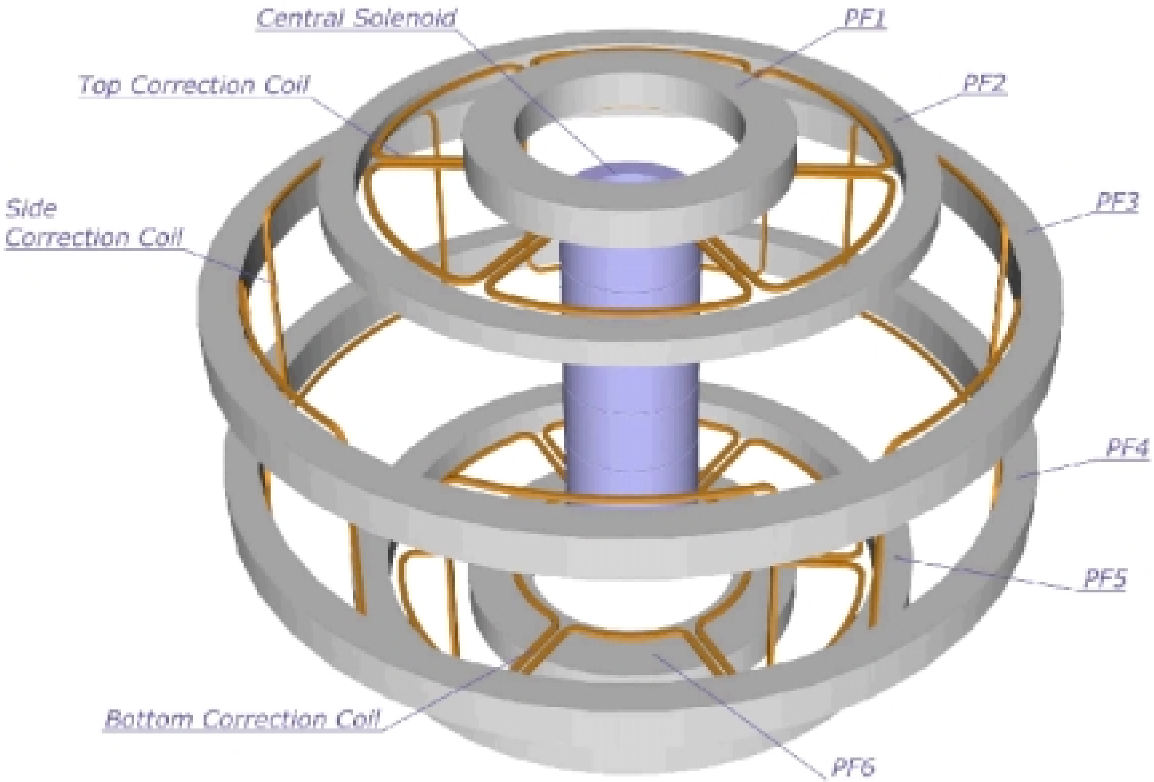
NTM

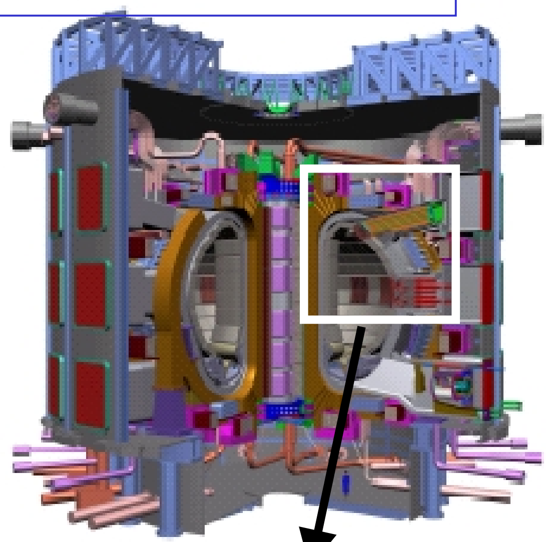


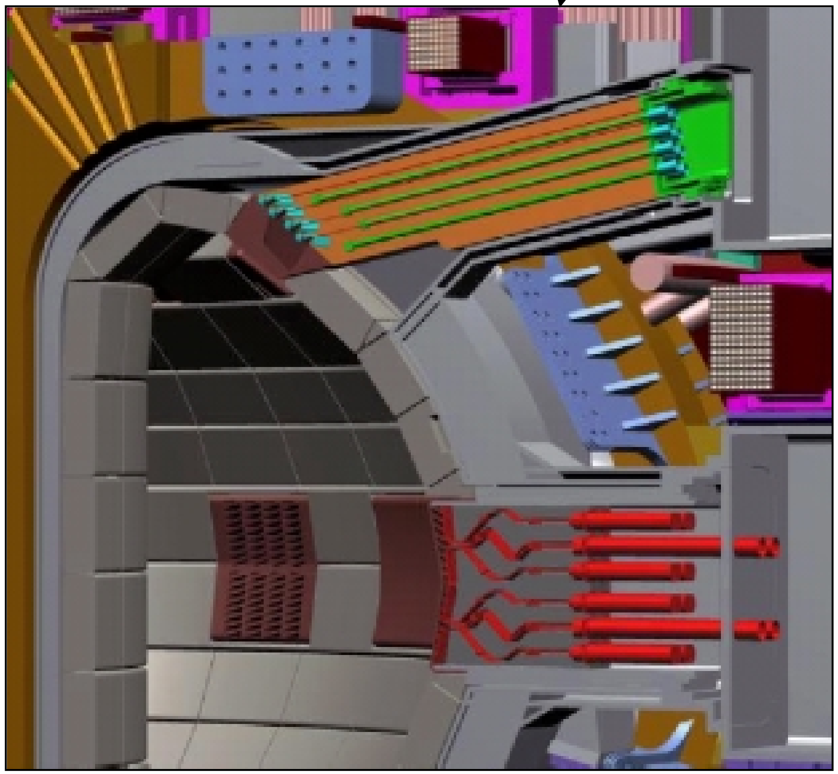




Tools for control

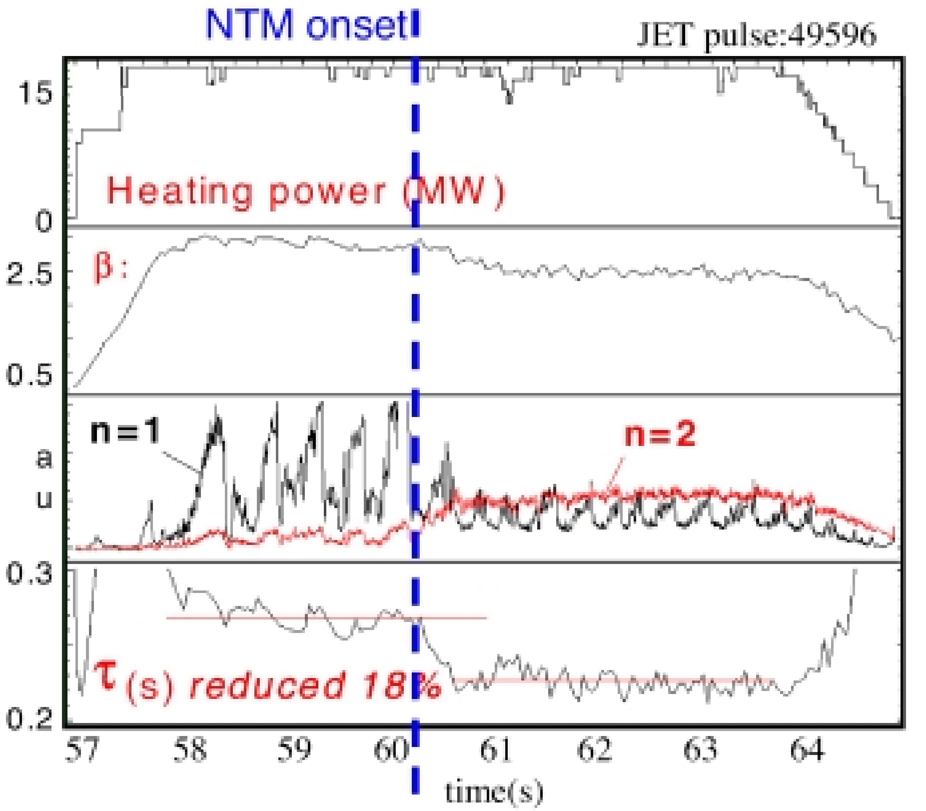


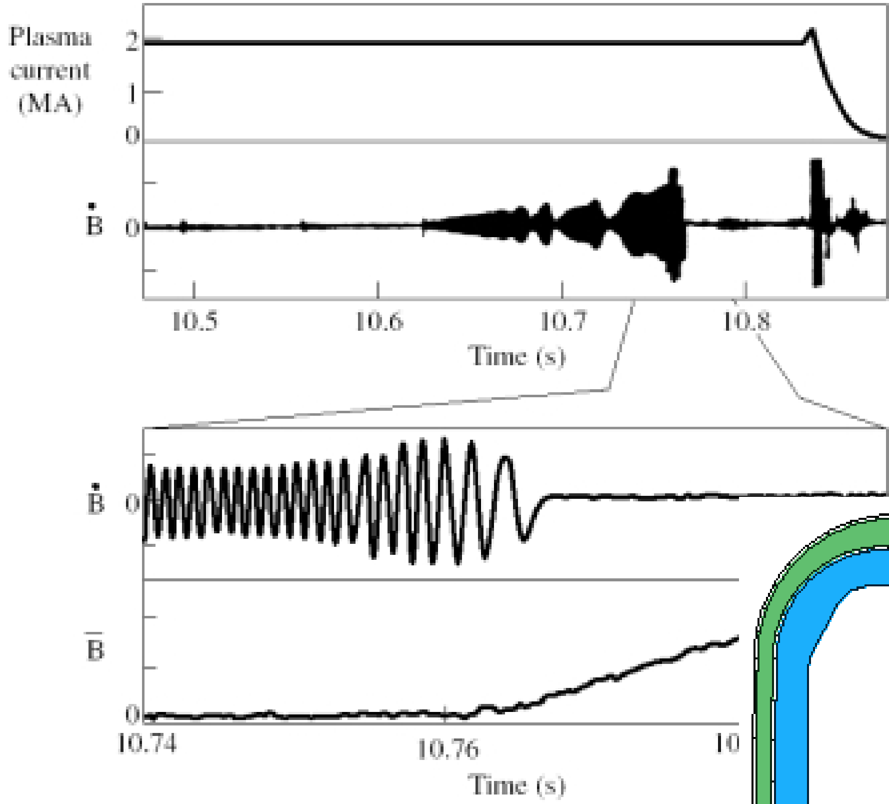




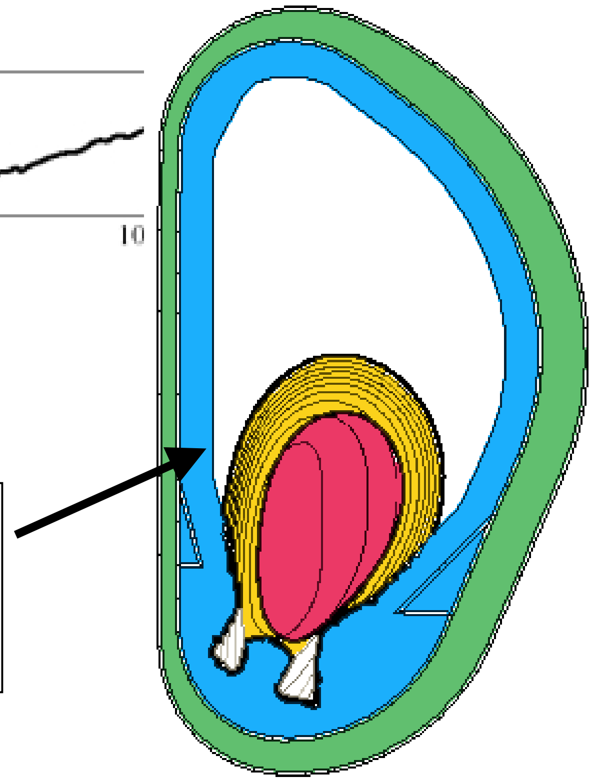
Consequences of instabilities

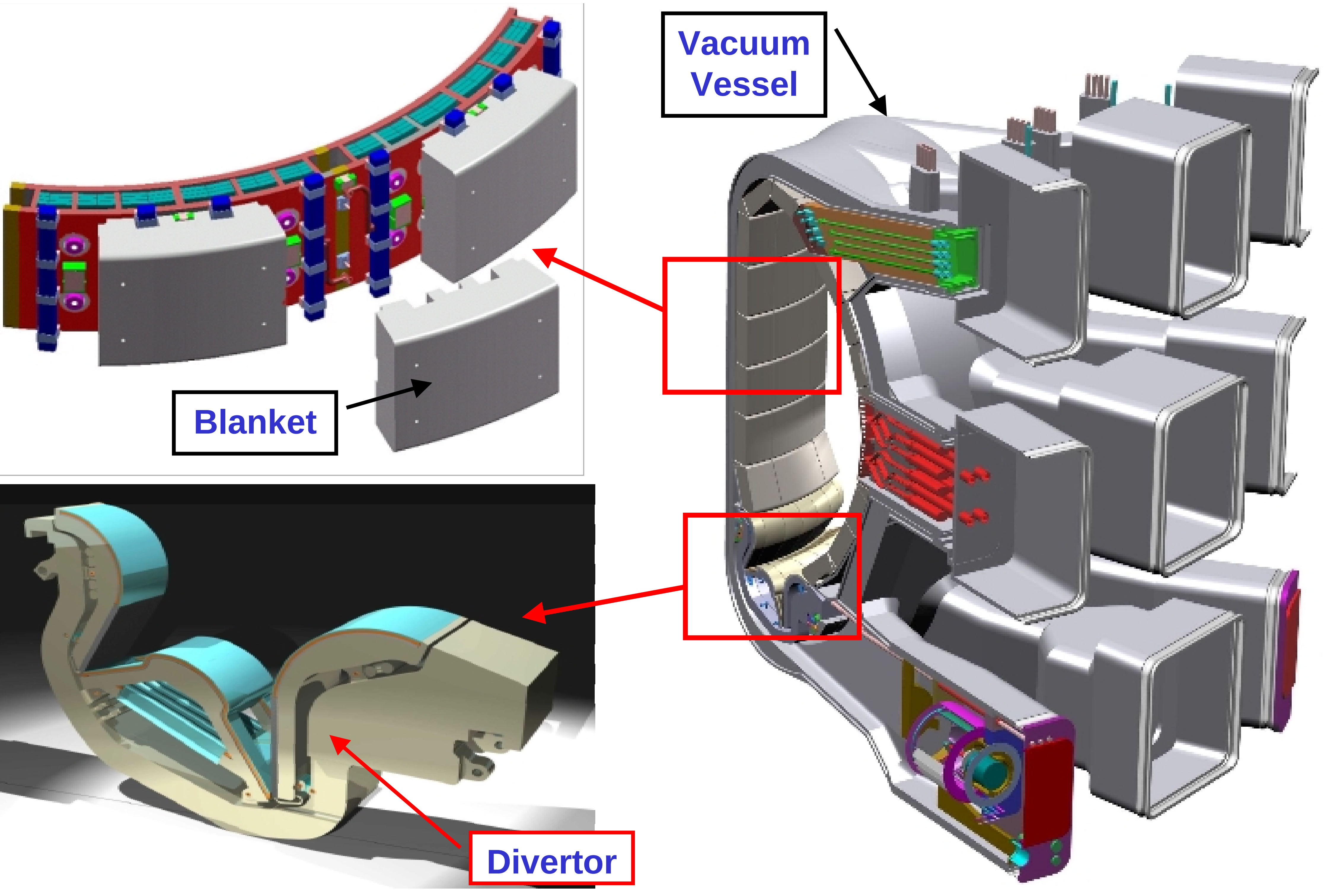
NTM ⇒ Poor confinement

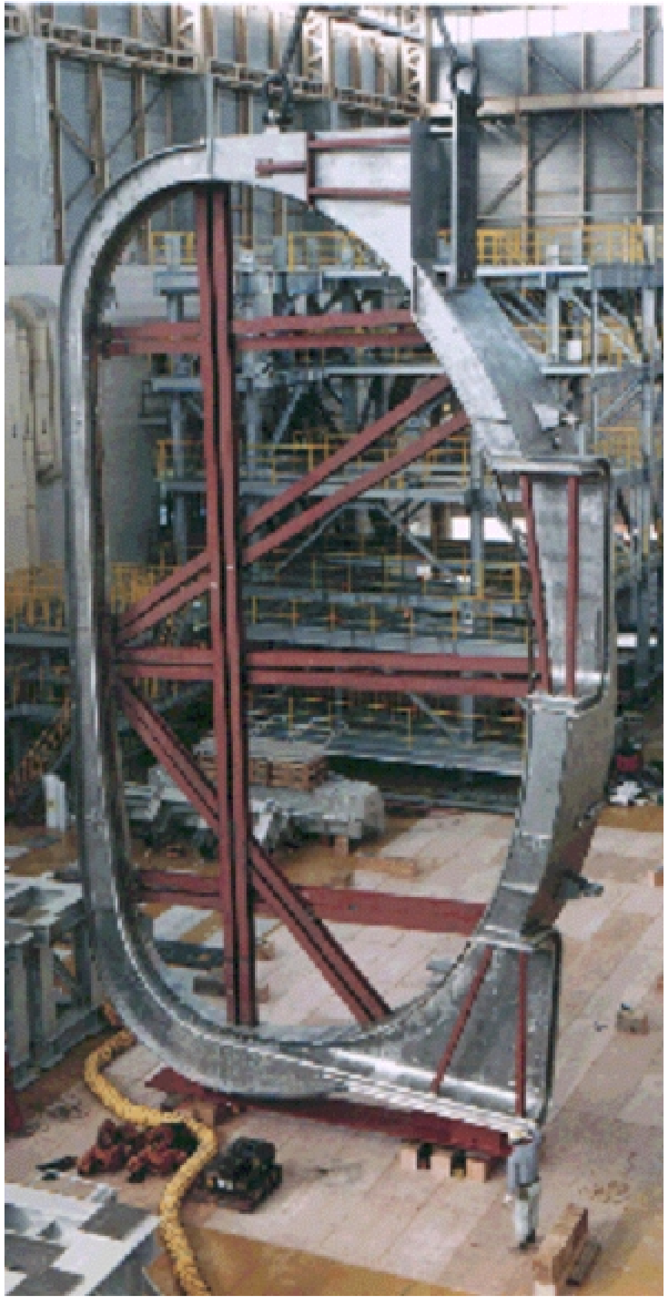




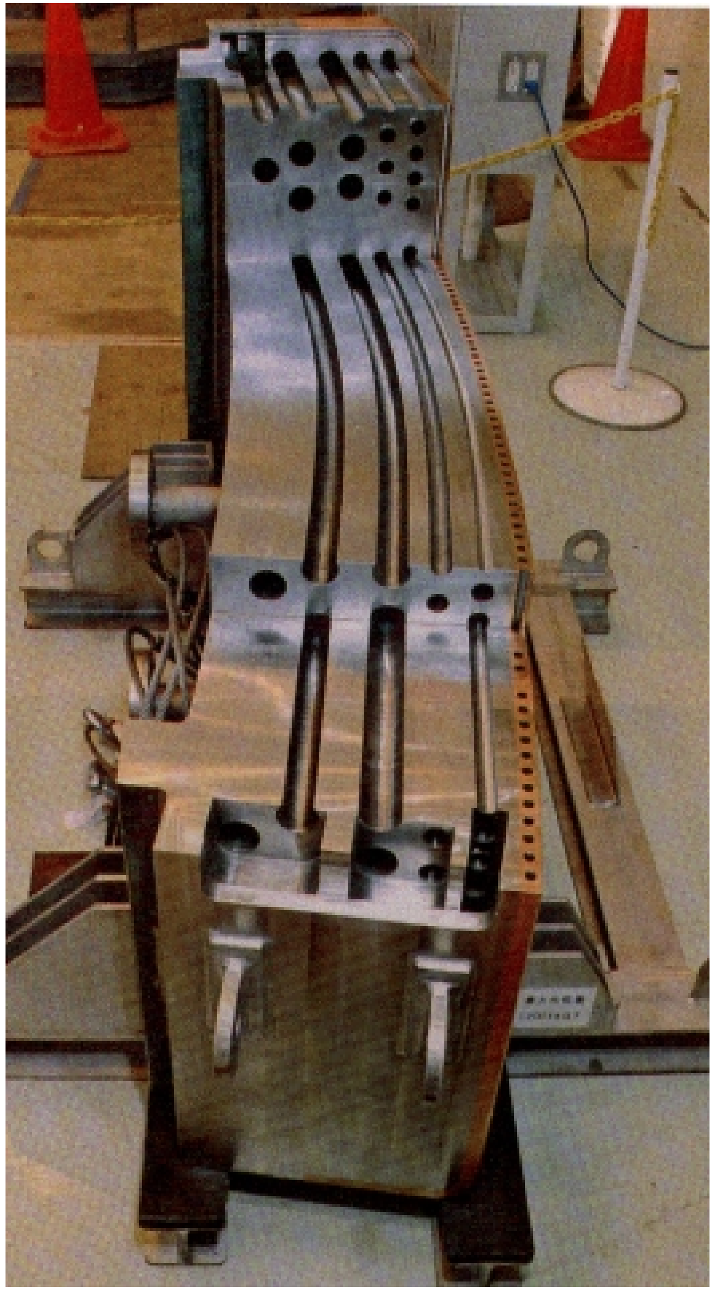
Vertical Instabilities give rise to significant mechanical loads



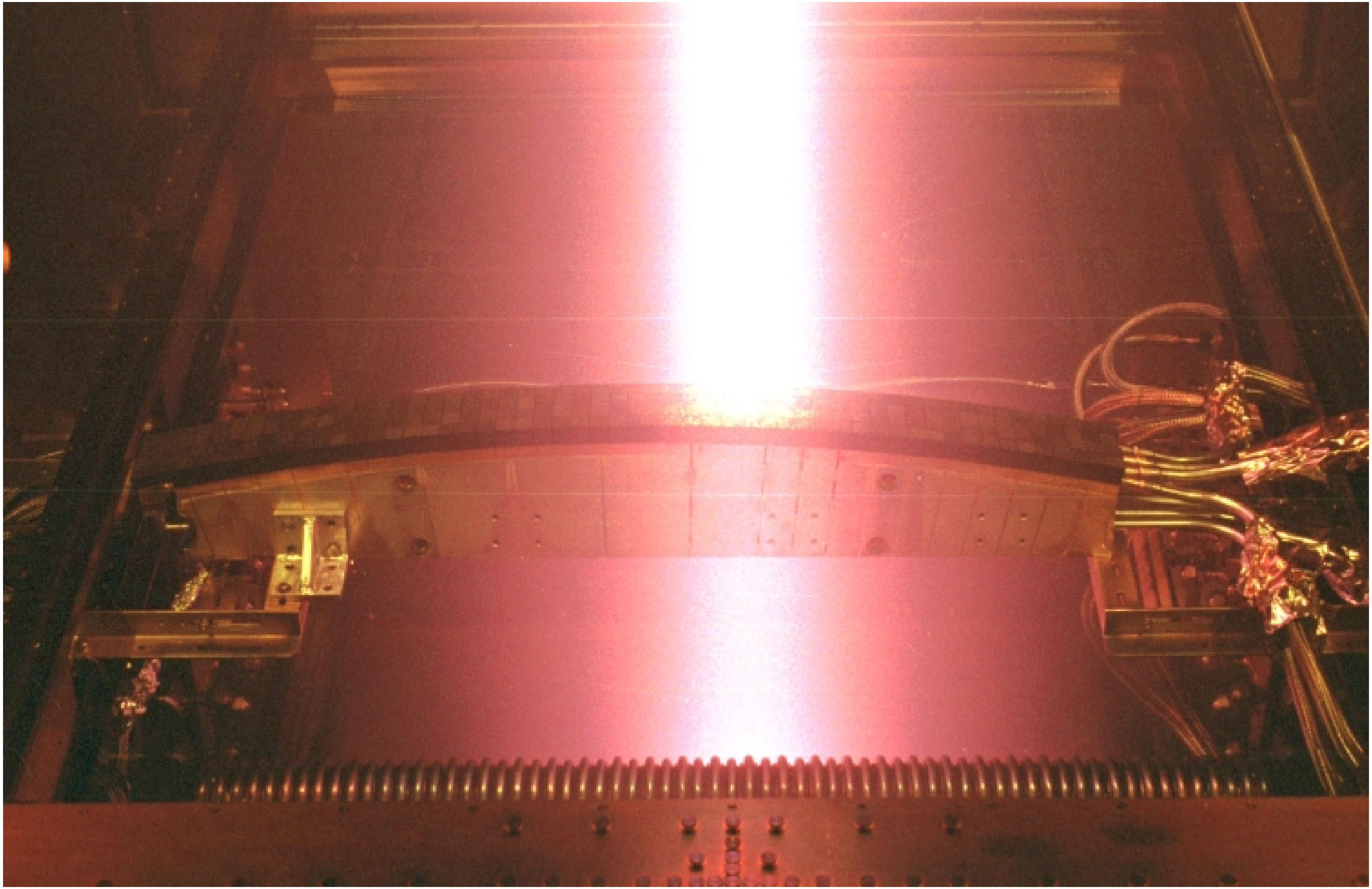

Vacuum Vessel
Blanket
Divertor



VV Sector



Blanket module



Synopsis

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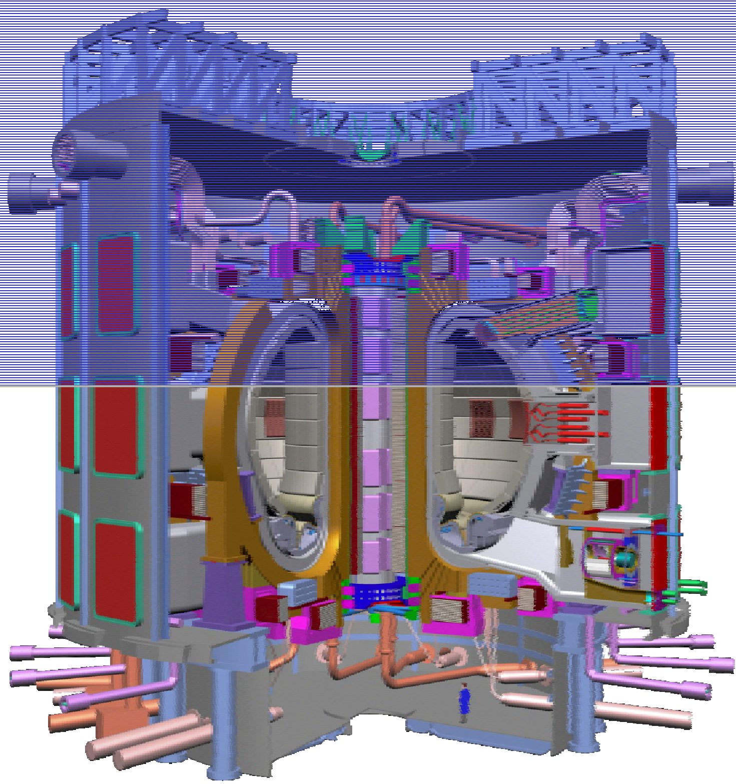
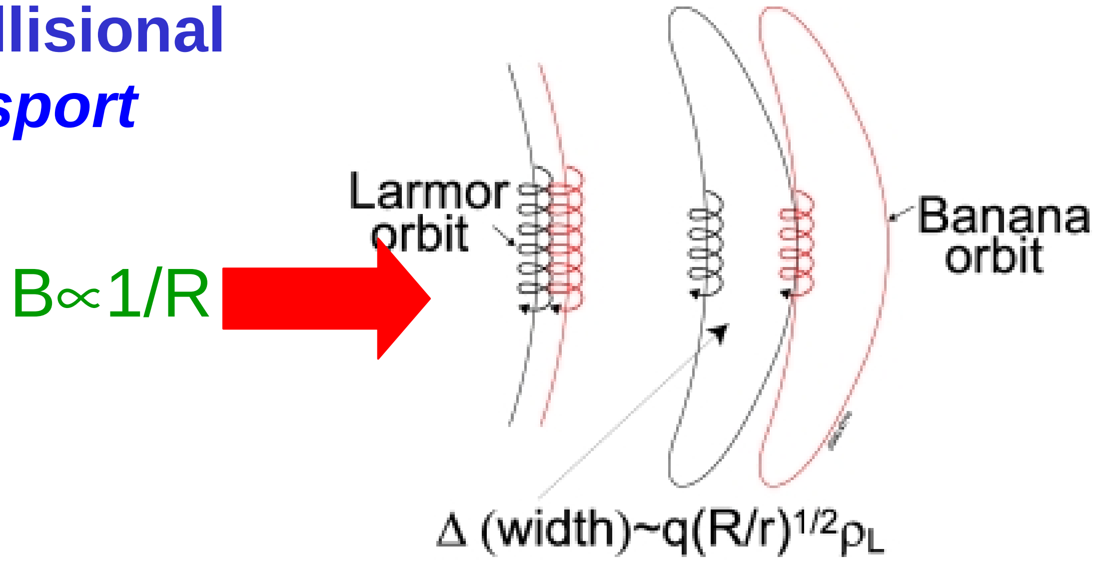
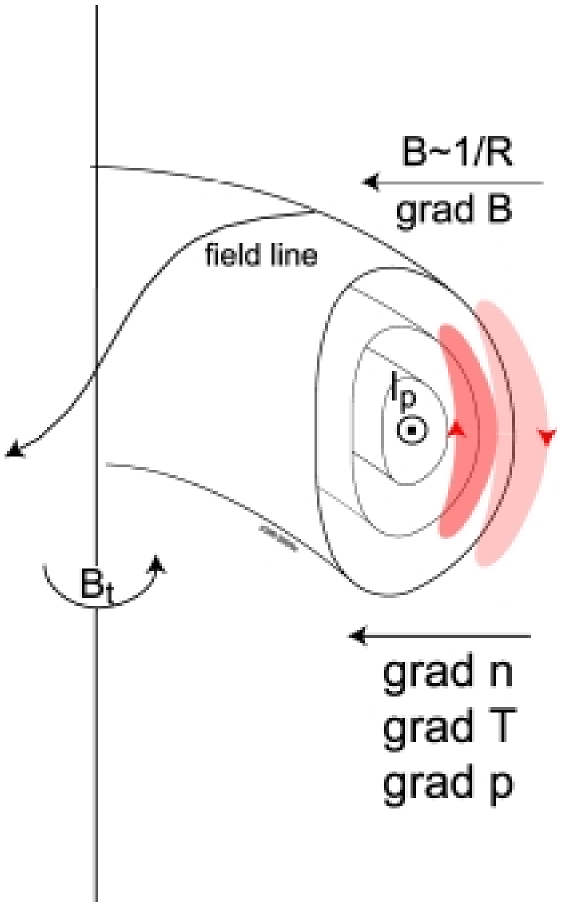
- **Conclusions**



Trapped particles transport & *//* currents

Trapped particles dominate collisional transport: *neoclassical transport*





- Banana orbits + ∇ in density and temperature profiles → bootstrap current I_{bs}
  - Driven by ∇, **damped by collisions**
  - I_{bs} up to ~70% I_p with strong ∇P_{core}
  - I_{bs} may dominate edge currents
  - I_{bs} affects MHD stability, core & edge

Turbulence and transport

⊥Transport in plasmas is >> neoclassical

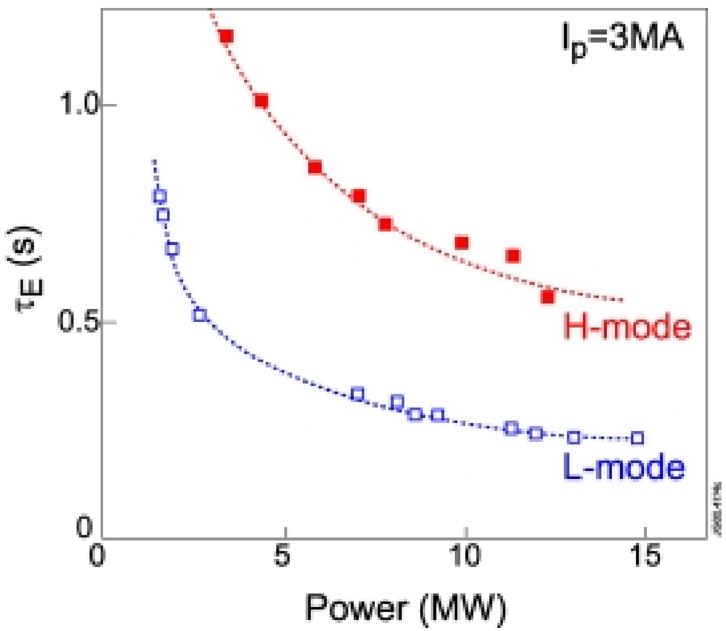
⊥electron and ion transport is anomalous

$(\chi_e \sim 100\chi_{e,neo}$ - $\chi_i \sim 3\text{-}4\chi_{i,neo})$

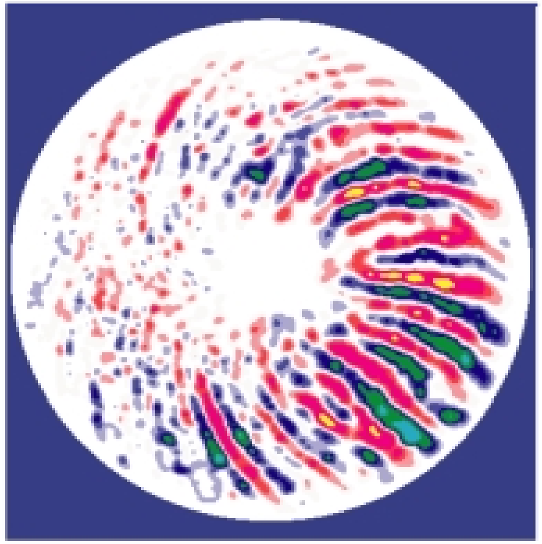
Turbulence produces radial 'streamers' which represent regions of rapid transport

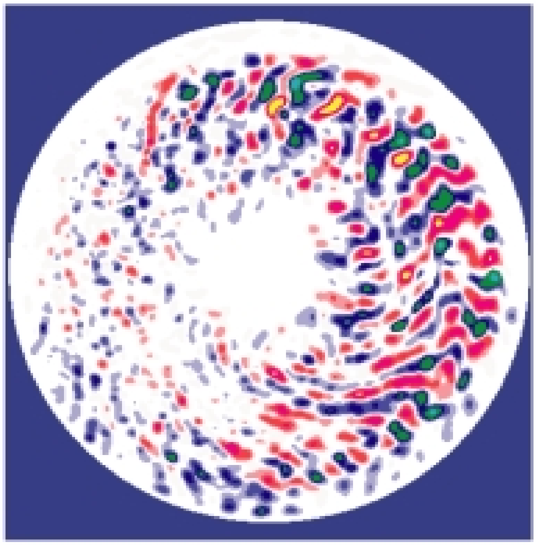
Sheared flows produce disintegration of 'streamers' and reduce transport

$\tau_E = W_p/P_{in} \downarrow$ with P_{in}



no EXB

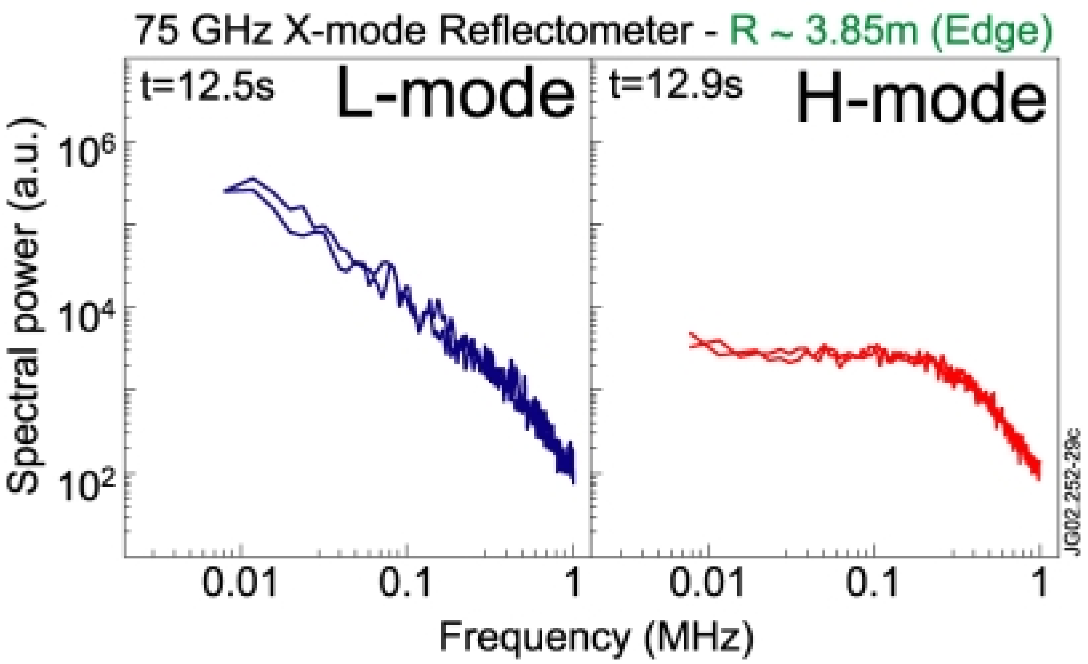


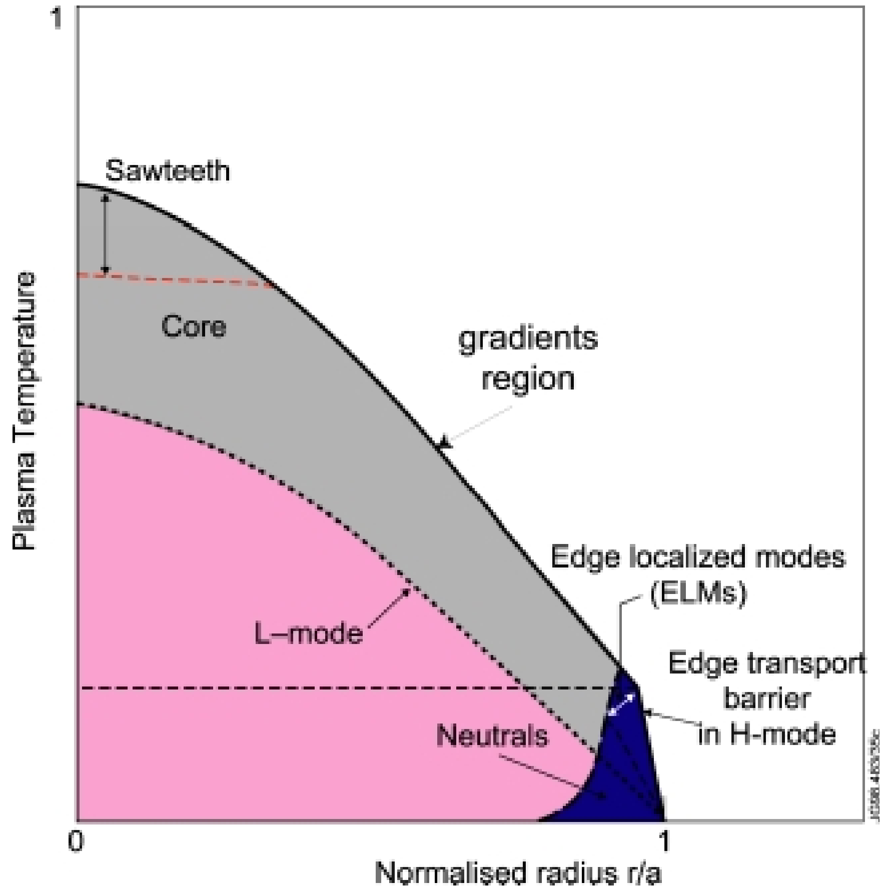


with EXB

H Mode: edge turbulence suppression

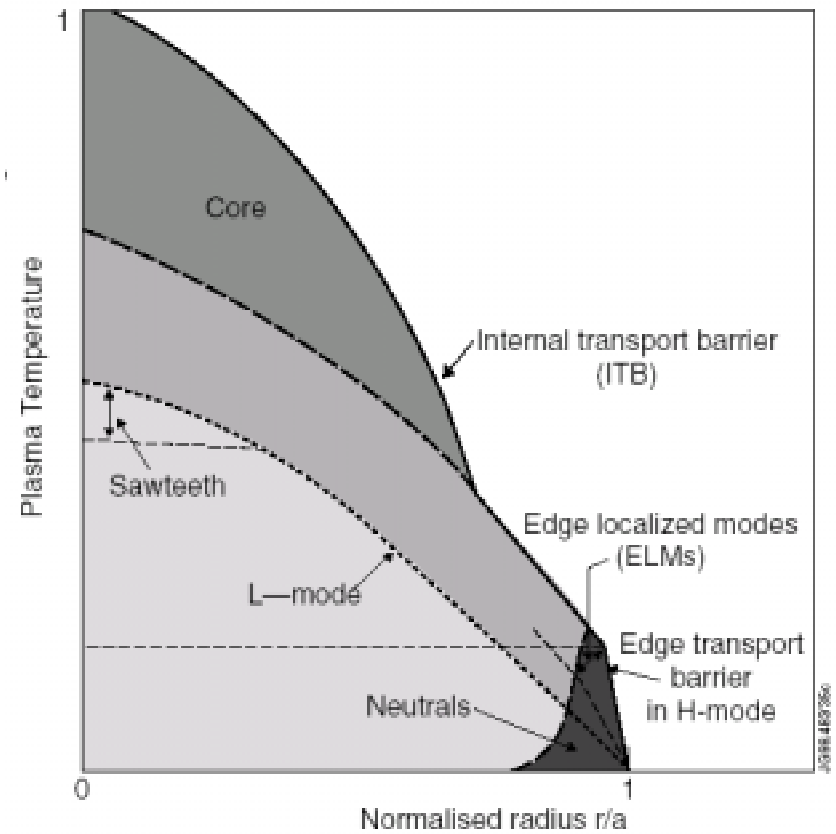
- **H mode discovered in Asdex in 1982**
- H mode$\rightarrow$ $\tau_{E,H} \sim 2\tau_{E,L}$ **– Edge Transport Barrier + ELMs: periodic fast relaxation of the ETB (expel energy & particles)**





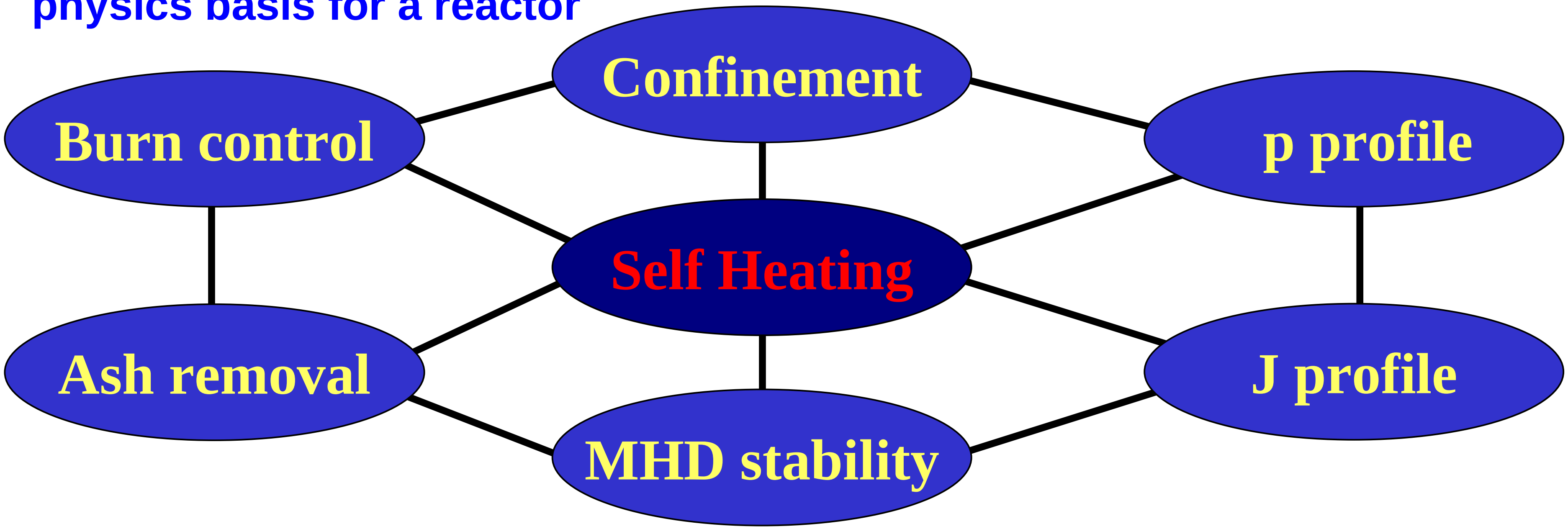
ITB: Internal Transport Barriers

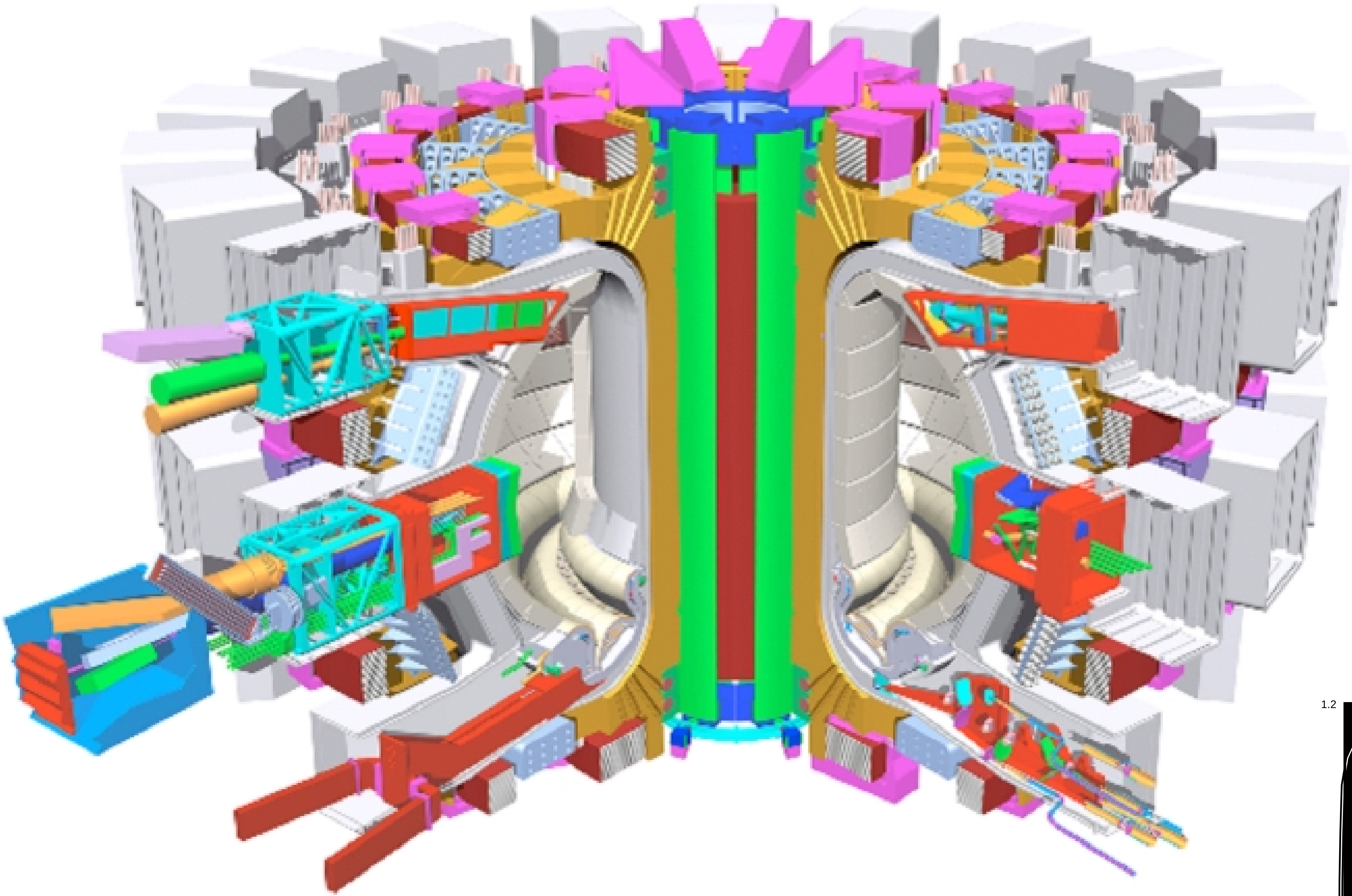
- Main challenge is to improve confinement
- Most promising for reactor
- Less current required, more "bootstrap current" available
- Challenge in:
  - Stability
  - α particle confinement
  - Consistency of profiles
  - Impurity accumulation

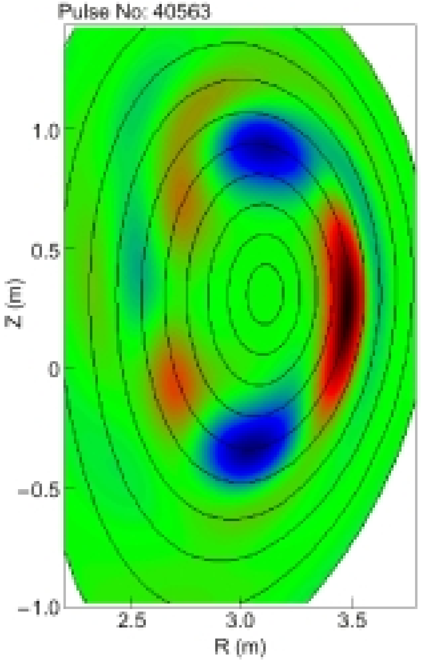


The Central Goal : Self Heating Effects

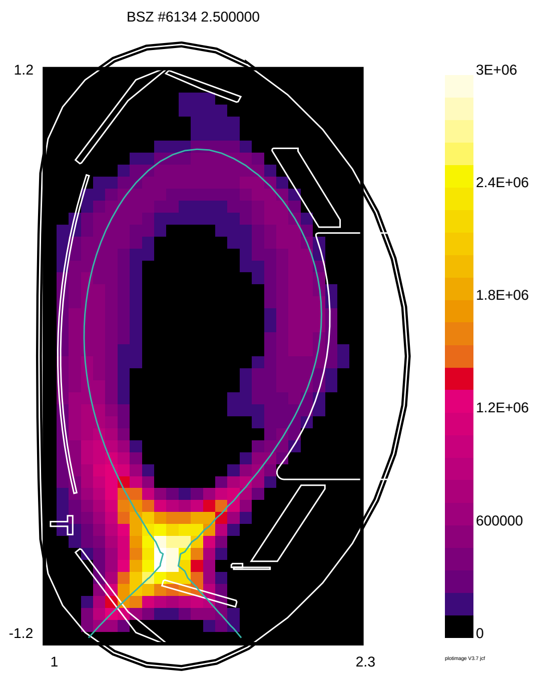
- Exploration of the new physics associated with a significant population of α-particles (self heating/organization) will be a central aspect of ITER
- We expect confinement, slowing down and heating to be acceptable
- Influence of α-particles on mhd stability is likely to provide a rich subject for study:
    - there have been remarkable advances in theory, but the uncertainties are such that we need to operate ITER to quantify the impact of α-particles on mhd stability and, ultimately, on plasma performance
    - Alfven Eigenmode instability in steady-state scenarios is a key challenge for physics basis for a reactor



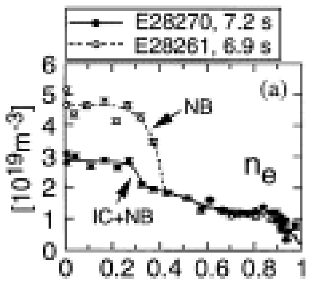




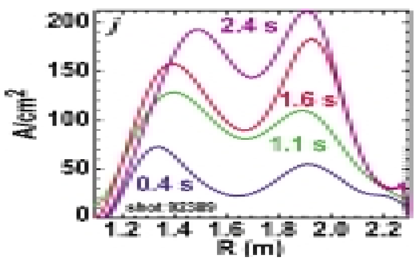
Pulse No: 40563
Z (m)
R (m)



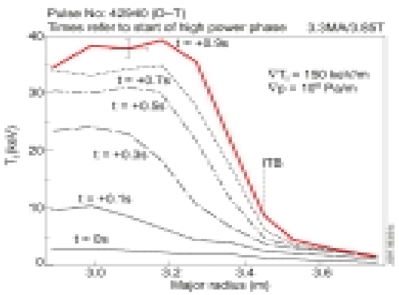
BSZ #6134 2.500000
3E+06
2.4E+06
1.8E+06
1.2E+06
600000
0



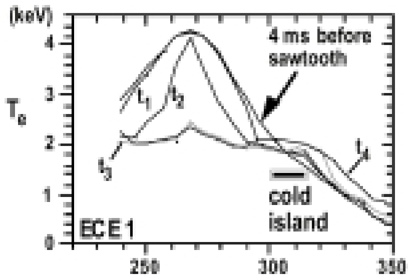
E28270, 7.2 s
E28261, 6.9 s
(a)
NB
IC+NB
n_e
$[10^{19}m^{-3}]$



j
2.4 s
1.6 s
1.1 s
0.4 s
A/cm²
R (m)



Times refer to start of high power phase
ITB



(keV)
T_e
4 ms before sawtooth
cold island
ECE 1

Negotiations

- ITER Final Design Report provides all necessary technical data for a decision on construction

- Quadripartite meetings on Negotiations began in June 2001
  - involving Canada, Euratom, Japan, Russian Federation

- The tasks of the Negotiators include
  - drafting the ITER Joint Implementation Agreement;
  - selecting the ITER construction site;
  - defining procurement rules
  - identifying the Senior Management

- The timescale for the negotiations foresees in mid-2003:
  - Consensus on site preference and,
  - Joint Implementation Agreement available for signature
  - a transition phase to maintain design continuity up to ratification.

- A siting decision can only be decided at the top political level, as all sites are technically acceptable.

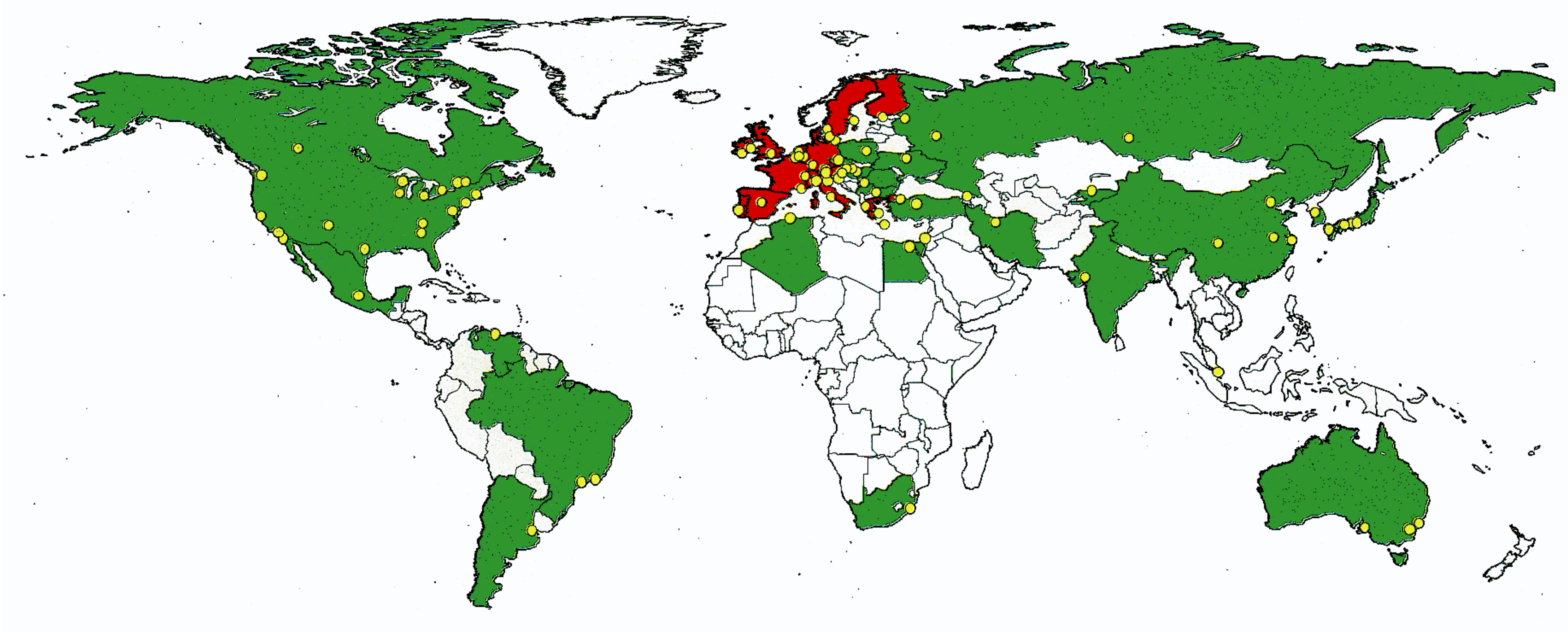

France


Spain


ITER SITE
Darlington
NGS
Canada


Japan

How ITER will be Operated



A Power Reactor

- **Tokamak may not be 'best' but:**
  - **Maybe good enough**
  - **not so far from any other system**

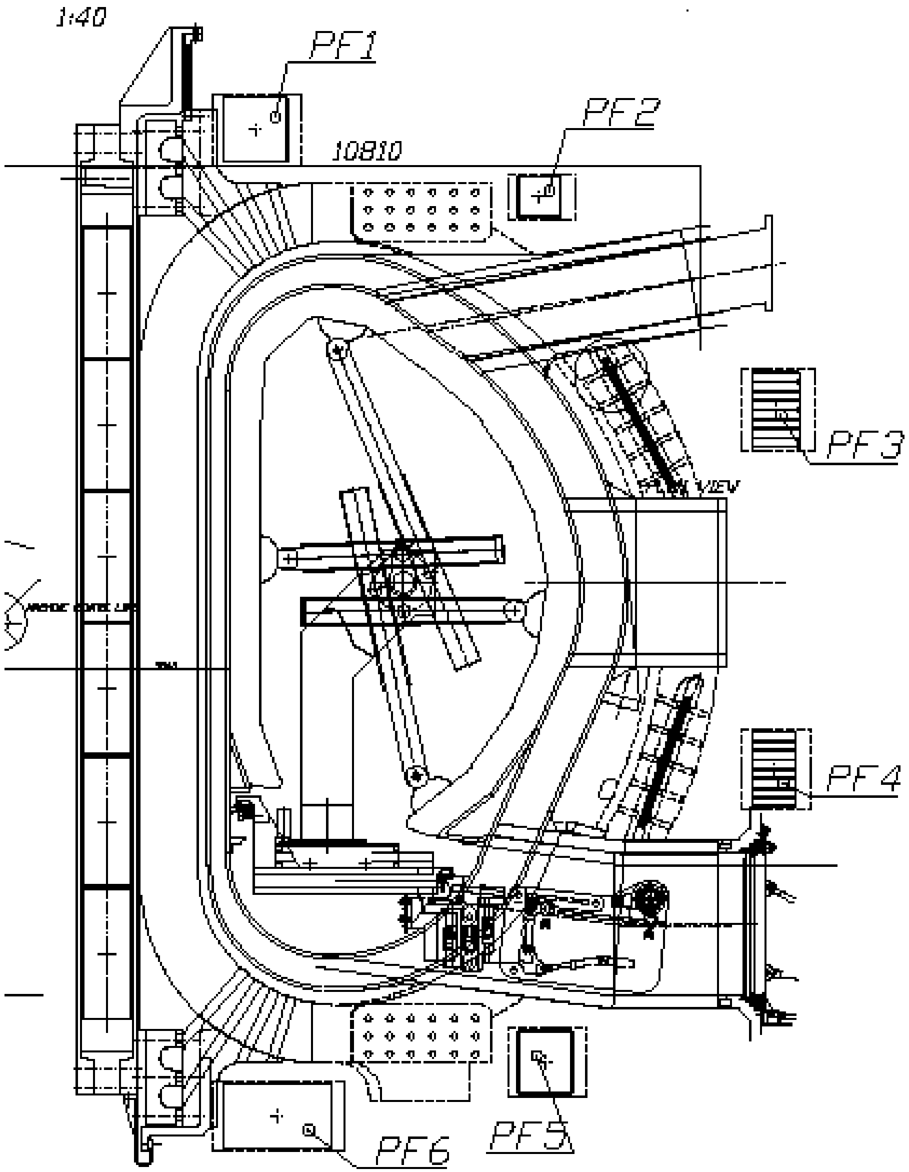
- **Where is a Reactor is different from an experiment?**

- **Additional problems**
  - **In physics :**
    - Beta(density, peaking),
    - Steady State
  - **In engineering :**
    - Remote Maintenance
    - Current drive efficiency
    - Reliability
    - Availability
    - Power exhaust
    - Materials

- **With some simplifications**
  - **In physics :**
    - Experimental Flexibility
  - **In engineering:**
    - Disruptions/VDEs?
    - Diagnostics
    - Heating methods
    - Fatigue

How far is reactor from ITER?

Major radius (m)	6.2	
Minor radius (m)	2	
Elongation (95% flux)	1.7	
Triangularity (95%)	0.33	
Toroidal field on axis (T)	**6.4**	**5.3**
Plasma Current (MA)	15	
Safety factor, q_{95}	3.8	3.0
Normalised β_N (including fast α's)	**3.8**	**1.9**
Bootstrap fraction, f_{bs}	0.53	
Confinement coefficient, H_H	**1.2**	**0.9**
n_{line} / n_{GW}	1	1
n_0 / n_{ped}	**1.6**	**1.0**
Av. Neutron wall loading (MW/m^2)	2.2	.45
Aux. heating power (MW)	80	40
Fusion power (MW)	2000	400
Q	25	10
Blanket energy gain	1.35	
Total thermal power (MW)	2600	
Thermal to electrical efficiency (%)	38	
Gross electrical power (MW)	1000	



Conclusions

- Fusion is a energy option worth pursuing and an exciting field of physics research.
- The need for a burning plasma experiment at the centre of the fusion development strategy is clear. A machine integrating appropriate physics and technology is the right next step, and ITER fulfils this role.
- The fusion programme in the Parties is scientifically and technically ready to take the important ITER step.
- The EDA demonstrated the desirability of jointly implementing ITER in a broad-based international collaborative frame.
- The start of negotiations on an agreement for joint construction and operation is a very positive step in the commitment of the Parties for ITER.

- ITER will develop a large part of the knowledge for the achievement of fusion energy. It will do so aided by smaller scale experiments which will be needed to deepen the understanding of fusion plasma physics.

Acknowledgements

- Thanks to

The ITER International and Participant's Teams

- Special thanks to:

D. Campbell

K. Lackner

F. Wagner

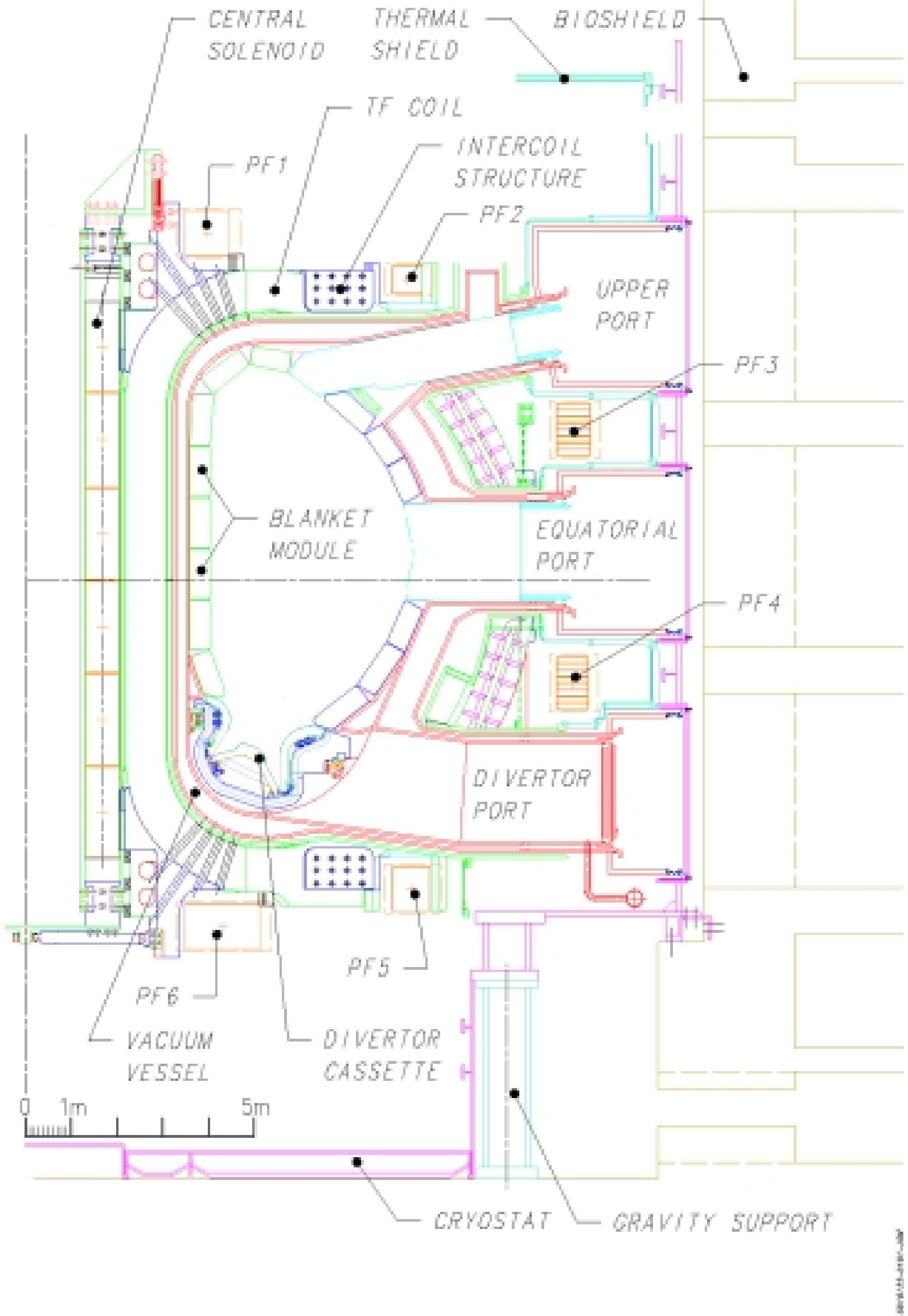
H. Bruhns

G. Saibene

G. Federici

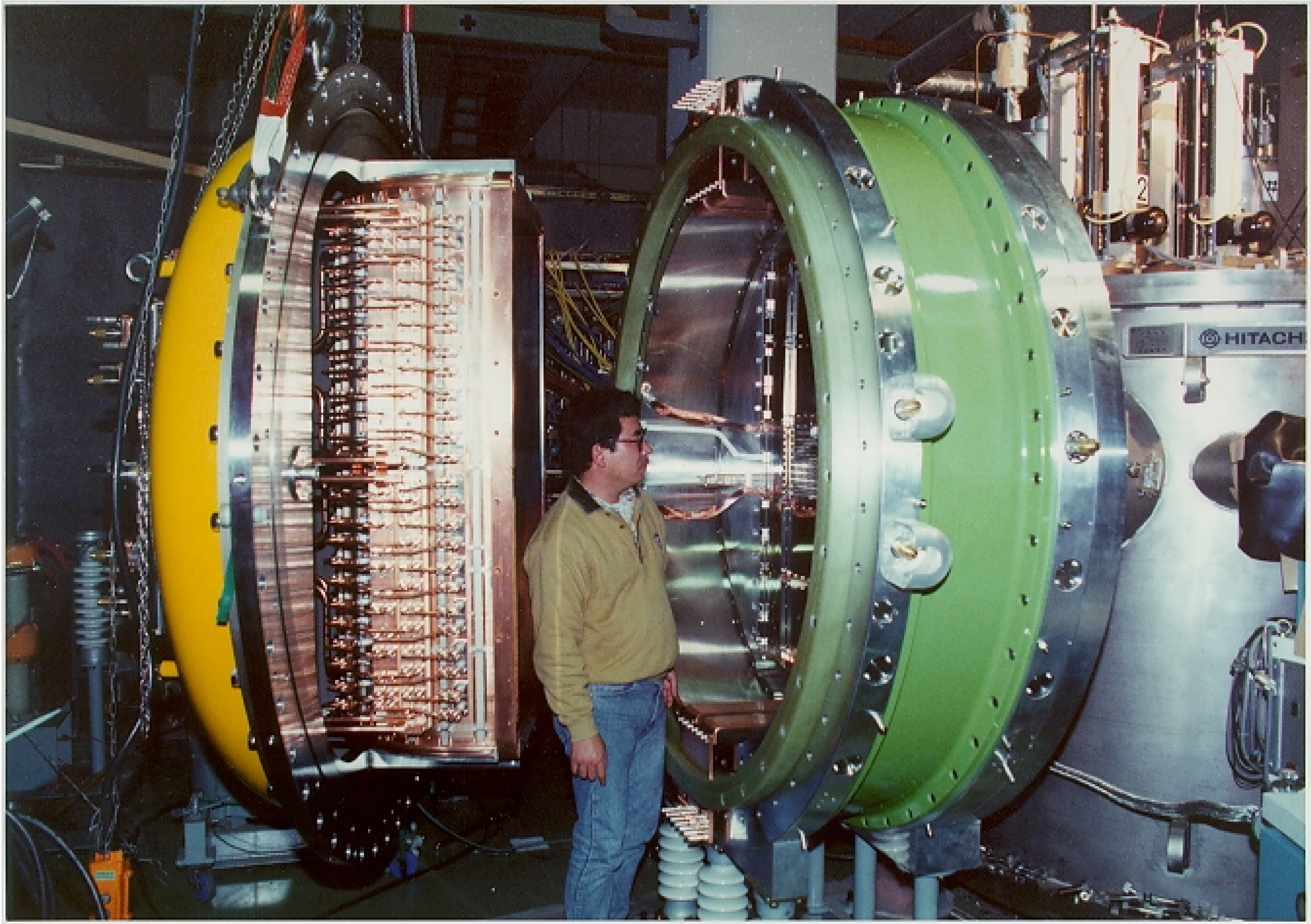
Y. Shimomura

R. Aymar

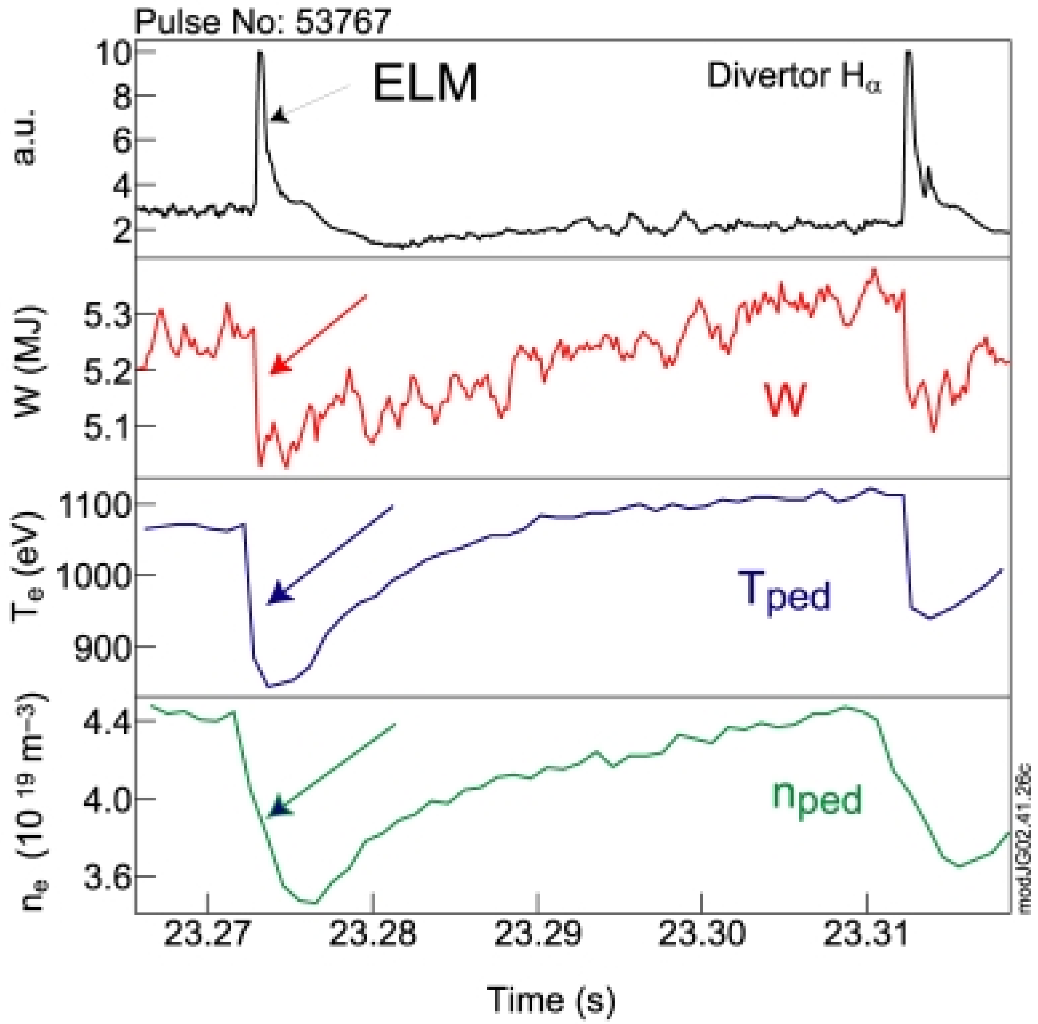

CENTRAL SOLENOID
THERMAL SHIELD
BIOSHIELD
TF COIL
INTERCOIL STRUCTURE
PF1
PF2
UPPER PORT
PF3
BLANKET MODULE
EQUATORIAL PORT
PF4
DIVERTOR PORT
PF5
PF6
VACUUM VESSEL
DIVERTOR CASSETTE
0
1m
5m
CRYOSTAT
GRAVITY SUPPORT

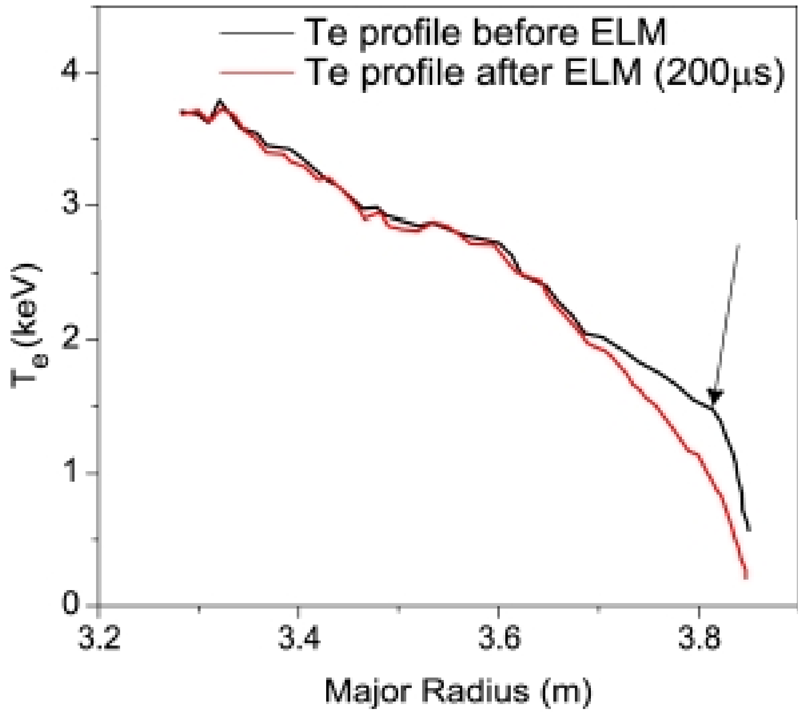
Fusion Safety

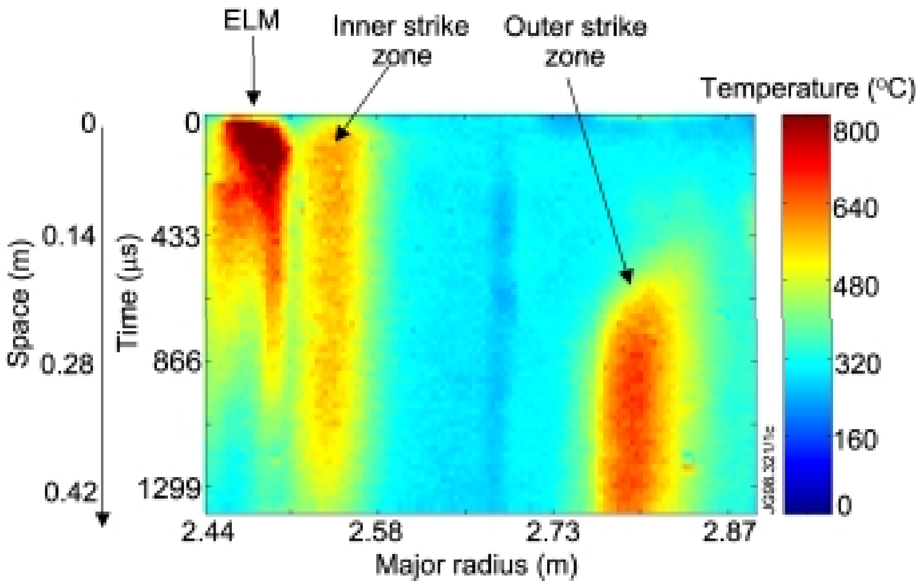
- ITER will be a precedent for future fusion licensing
- Work towards internationally accepted basic principles and safety criteria for fusion energy
- Interact with regulatory experts to ensure ITER options can be licensed in any Party
- Maximise use of inherent fusion safety characteristics.
    - intrinsic passive shutdown and fail safe termination of plasma
    - low decay heat density - no meltdown even with loss of all cooling systems
- Apply well-established safety approaches.
    - establish and comply with rigorous general safety and environmental design criteria based on established limits — ALARA principle, ICRP, IAEA
    - robust structure, maximising safety contributions from systems needed for operation
    - defence in depth (prevention, protection, mitigation), passive heat removal system
- Ultimate safety margin to tolerate 100% release of “at risk” inventories and still meet **non-evacuation criteria.**


HITACH

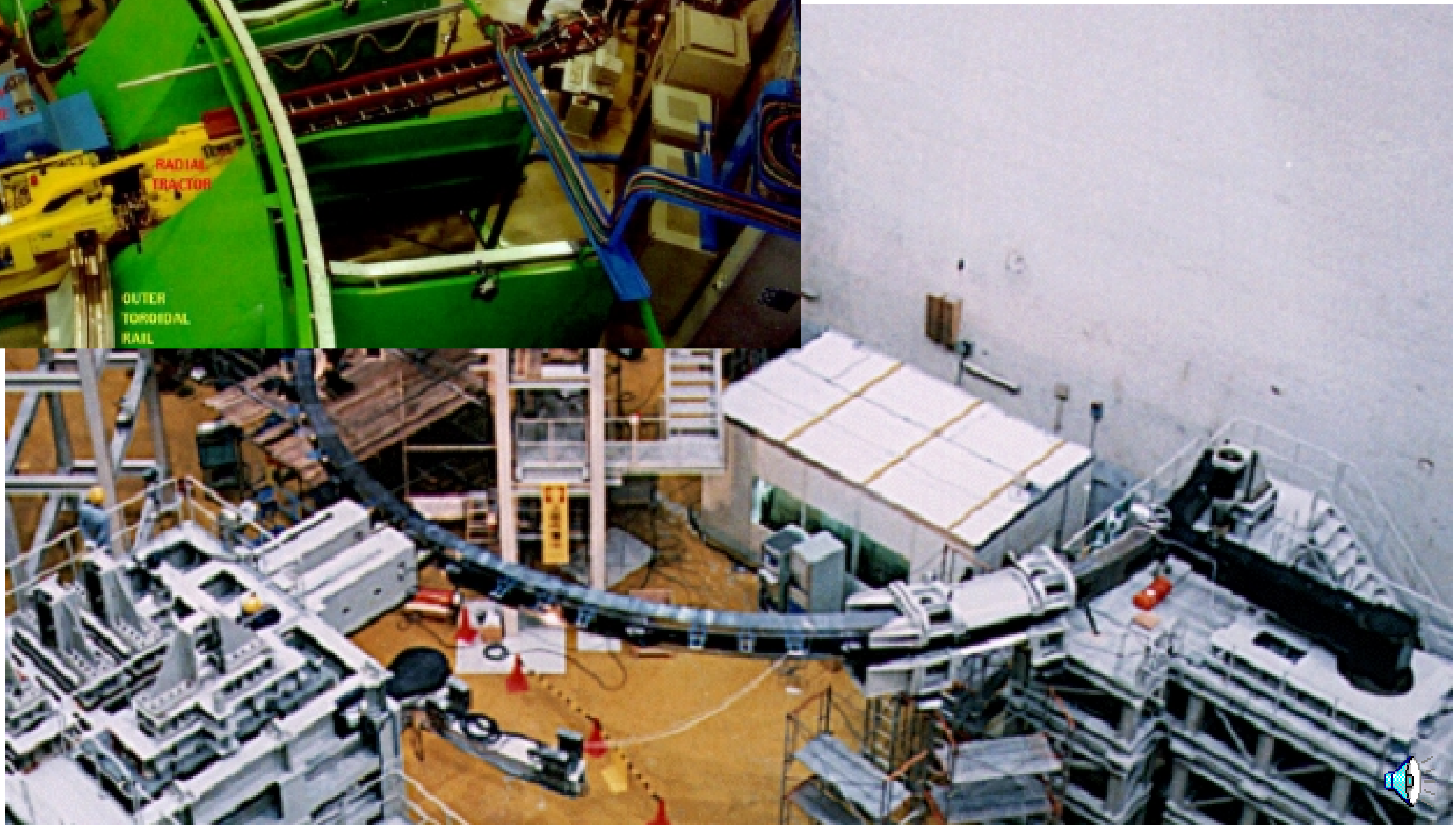
ELMs → high power density to the divertor








BIO-SHIELD
CENTRAL CASSETTE CARRIER
PLUG HANDLING VEHICLE
DUMMY CASSETTE
RADIAL TRACTOR
OUTER TOROIDAL RAIL

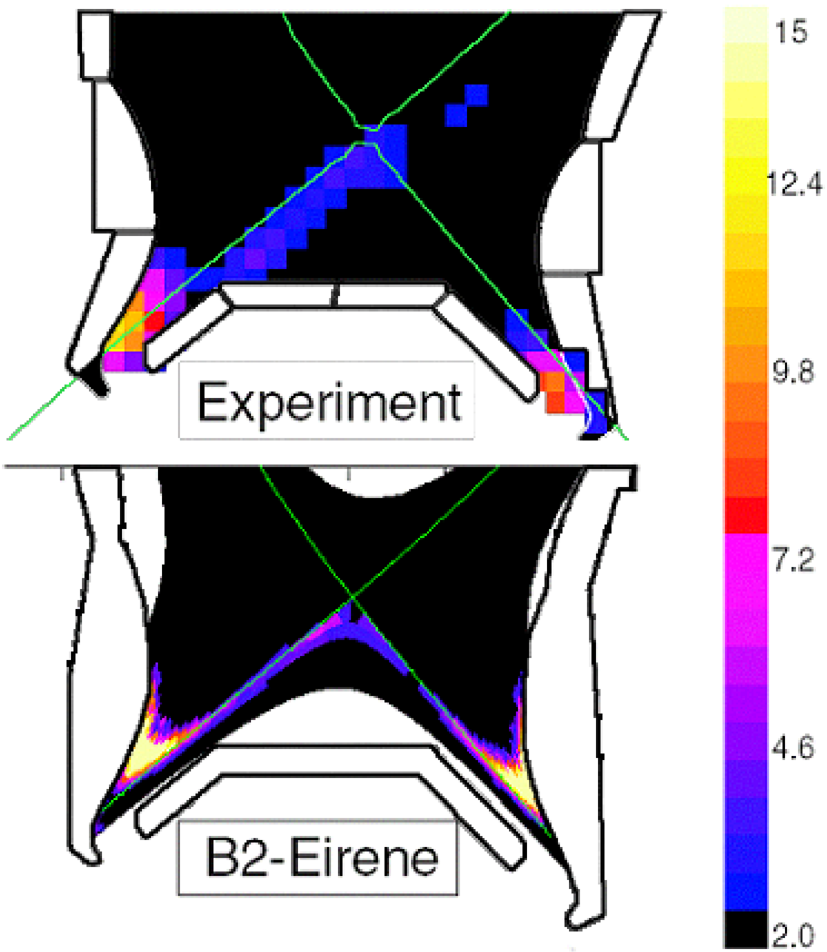


◆ Power exhaust:

~100MW must be exhausted within engineering limits on walls: < 10 MWm^{-2}

◆ Helium exhaust:

Helium "ash" must be pumped away from plasma edge to keep helium concentration in plasma below ~10%



Radiation losses