DE LA RECHERCHE À L'INDUSTRIE

# *MAGNETIC CONFINEMENT PRINCIPLES IN TOKAMAK DEVICES AND ACTUAL CHALLENGES*


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# OUTLINE

**Part 1: Introduction**

- Tokamak principles
- Challenges for continuous tokamak operation
- Advanced scenarios and internal transport barriers

**Part 2: Towards Real time Control of advanced scenarios**

- Profiles reconstruction: the essential role of the diagnostics
- Real-time control of kinetic and magnetic profiles in present devices

**Part 3: Preparing the future**

- Controlling impurities to improve the control of advanced scenarios
- Real-time fusion DT burn control
- Core performance control with plasma facing components constraints

**Part 4: ITER status**

**Part 5: Conclusions and perspectives**

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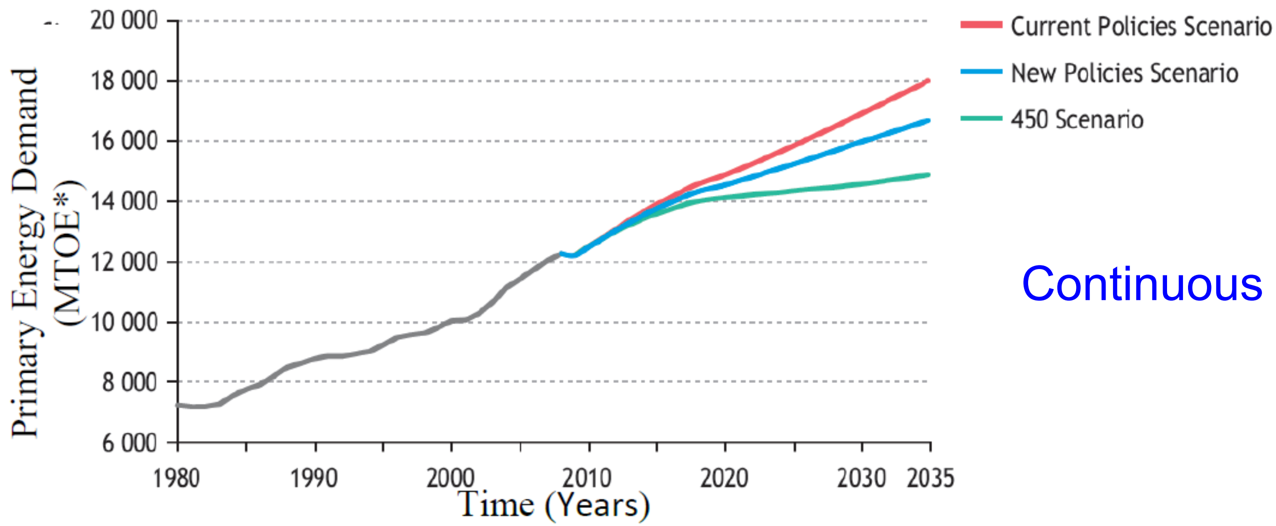
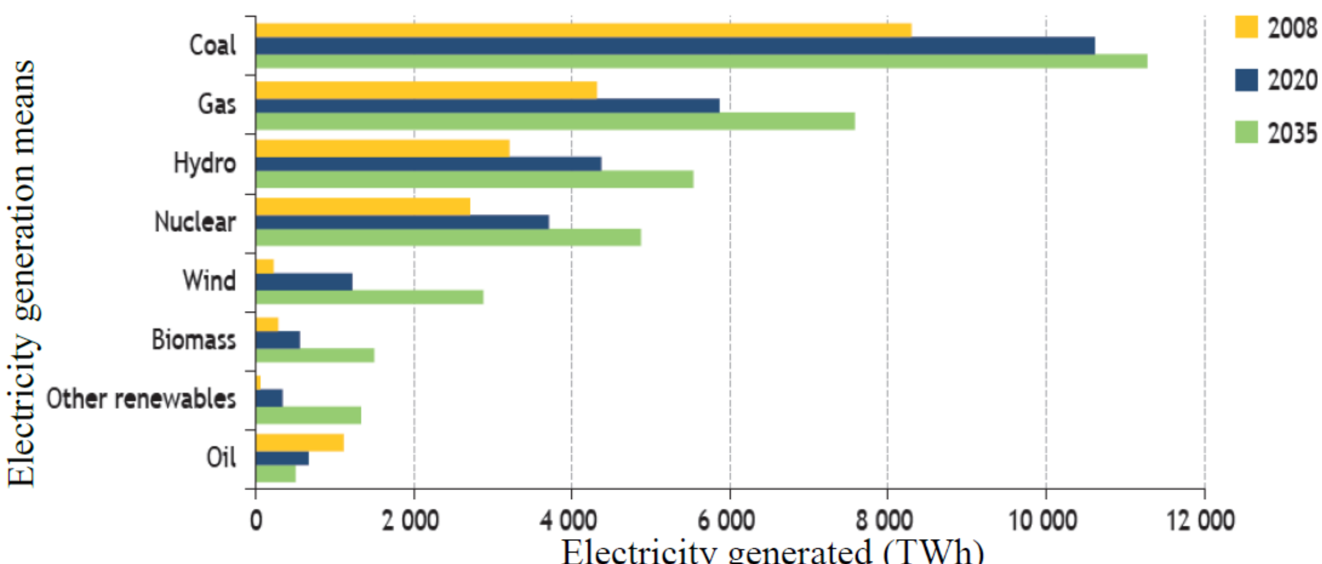
# PREAMBLE




*MTOE: Million Tons of Oil Equivalent

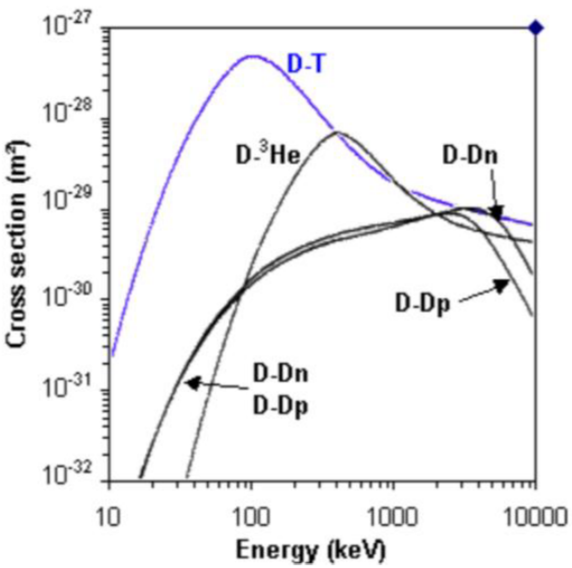
Continuous increasing demand



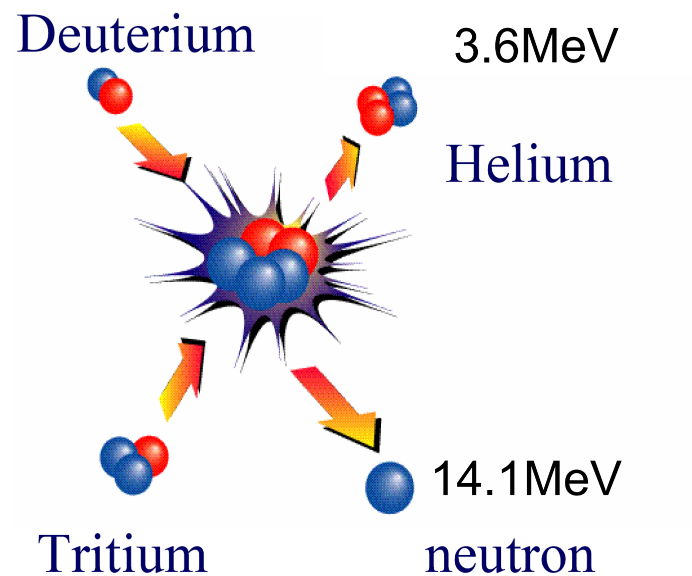

Greenhouse issues

# FUSION: A PROMISSING ALTERNATIVE

## Light nuclei fuse into heavier nuclei



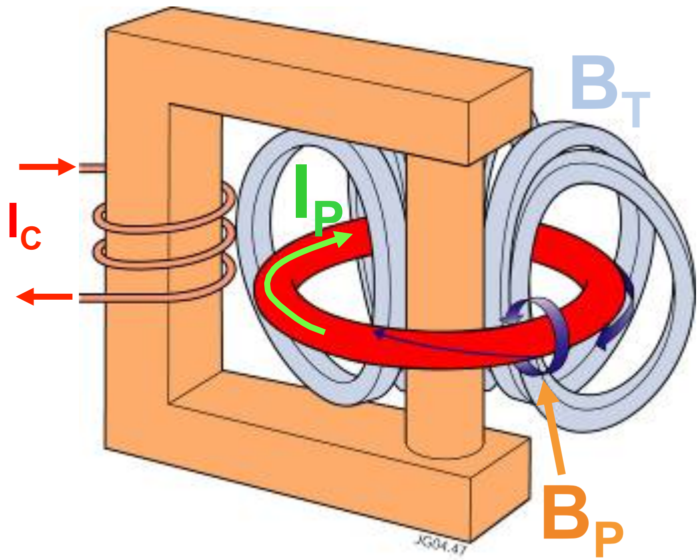

DT fusion reaction has a cross section (reactivity) about two orders of magnitude higher than the DD reaction




$\alpha$ particles contributes to plasma heating

**No greenhouse effects and large amount of released energy**

# TOKAMAK PRINCIPLE




$I_P$ = $I_{inductive}$ + $I_{Non\text{-}Inductive}$

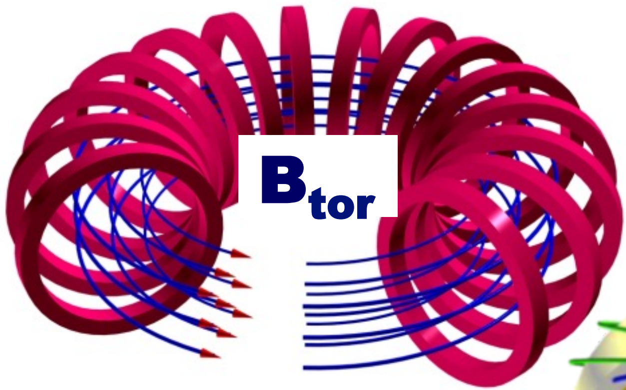
- **Long-Pulse Operation**

$\rightarrow$ $I_P = I_{Non\text{-}Inductive}$

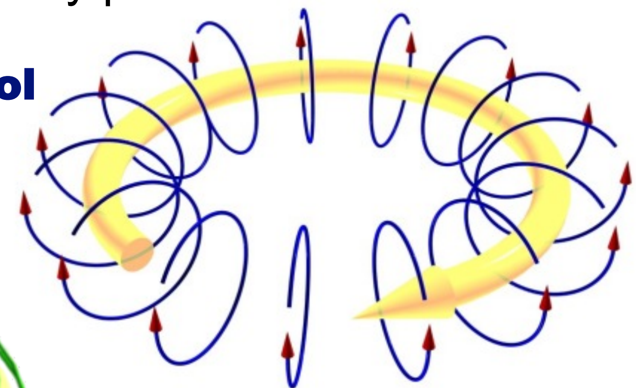
- **Non-Inductive Current Drive**
  - **Externally driven, e.g. waves injection**
    - To drive 15MA on ITER requires 150MW
    - 150MW coupled power requires ~ 1GW fusion
  - **Internally driven $\propto \nabla$Pression: bootstrap effect**
- **Efficient reactor at high fusion gain $Q = P_{fus}/P_{add}$ relies on the optimisation of bootstrap current**

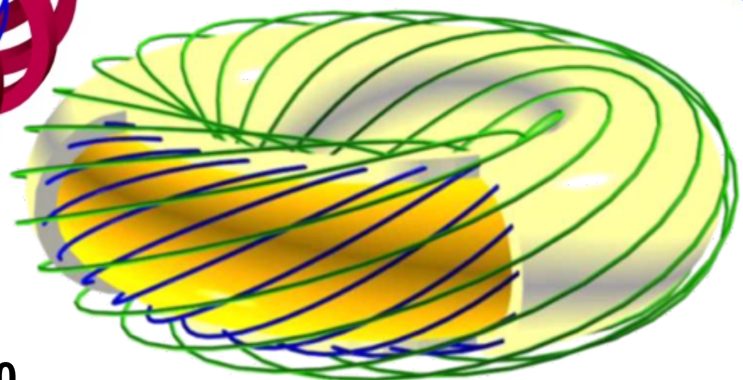
# SOME BASIC PARAMETERS IN TOKAMAKS

Efficiency of confinement mesured by $\beta$




$B_{pol}$





- **Toroidal**
  - $\beta_t = \langle P \rangle / p_{Btor}$
  - $p_{Btor} = B^2_{tor}/2\mu_0$

- **Poloidal**
  - $\beta_p = \langle P \rangle / p_{Bpol}$
  - $p_{Bpol} = B^2_{pol}/2\mu_0$

Safety factor q

number of toroidal turns for one poloidal turn
$q = d\varphi / d\theta$

# CHALLENGES FOR CONTINUOUS OPERATION

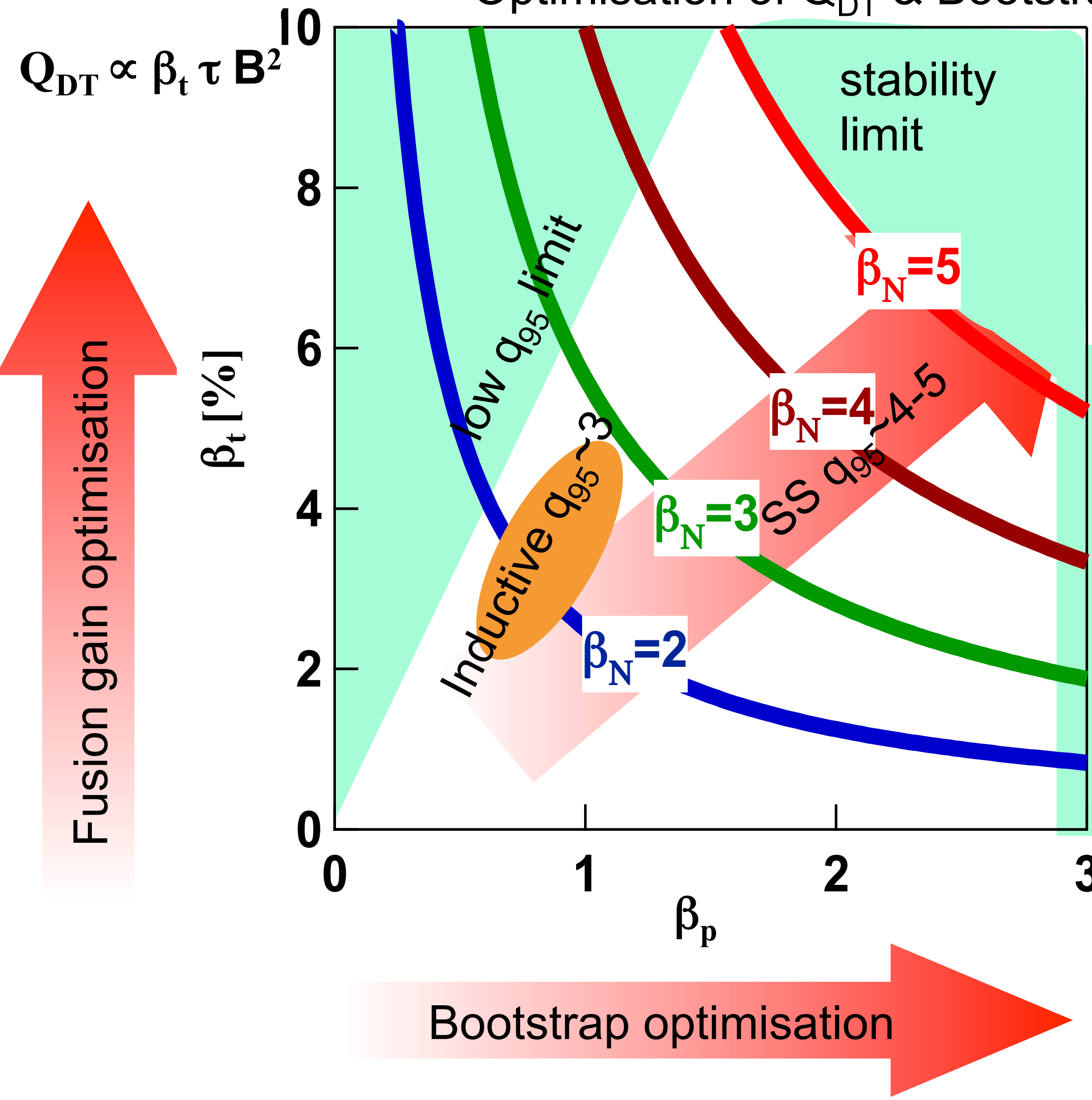
- **Fully non-inductive regime**
- **High confinement & bootstrap current**
- **Real time control of kinetic & magnetic configuration close to operational limits with a large fraction self $\alpha$-heating & bootstrap**
- **Issues about the technology of Long Pulse Operation**
  - **High temperature supra conducting coils, Plasma facing components for high heat and neutron fluxes, Structure materials, Reliable and efficient Heating and Current Drive systems, Diagnostics, data acquisition, fuel cycle**

**Worldwide research activity: physics, modelling, technology**

# CHALLENGES FOR CONTINUOUS OPERATION

Optimisation of $Q_{DT}$ & Bootstrap current

$Q_{DT} \propto \beta_t \tau B^2$

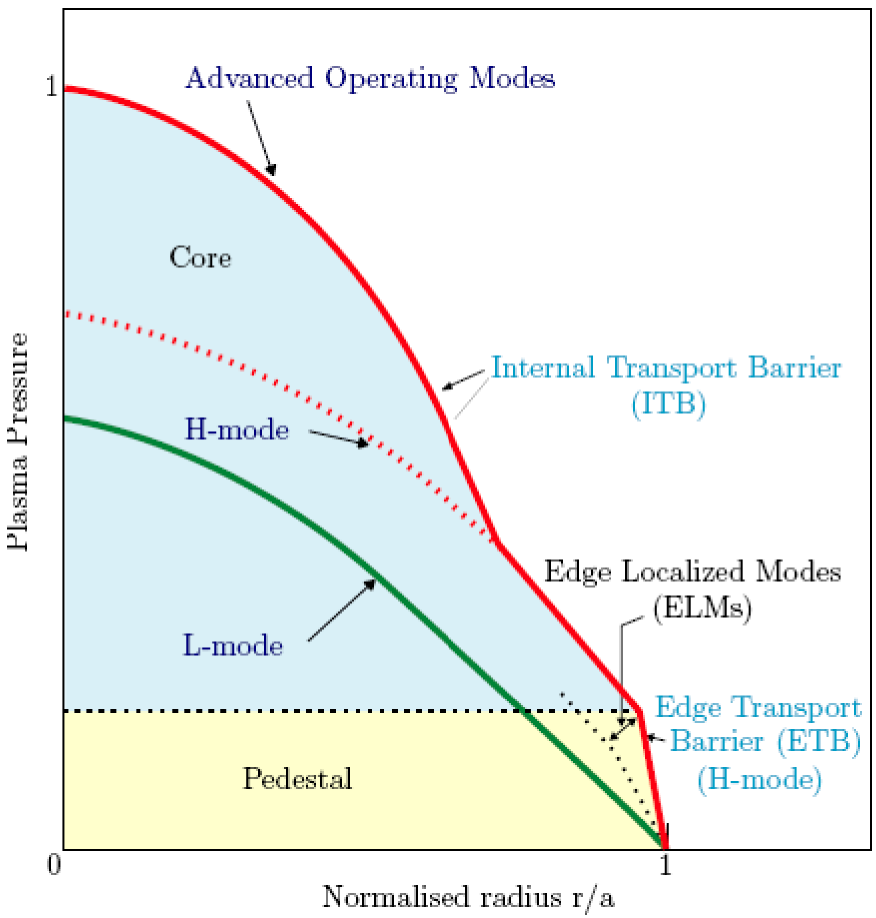



- ↑ fusion power + bootstrap
  → high $\beta_N$ (plasma P/magnetic P)
  **since** $\beta_t \beta_p \propto (1+\kappa^2)\, \beta^2_N$
  → Confinement ($\tau$)
  → B
- Optimise shaping ($\kappa$)
- Stability

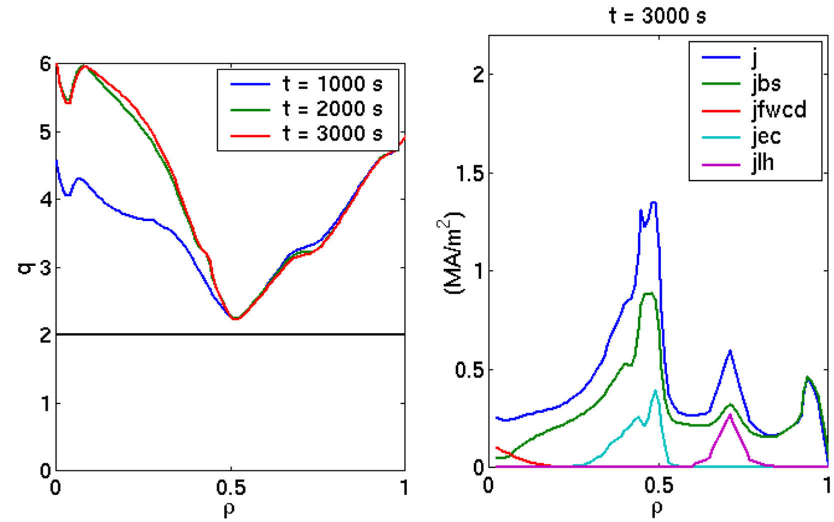
q & pressure profiles to be controlled

$I_{boot}/I_p \propto \varepsilon^{1/2} \beta_p = \varepsilon^{-1/2}\, q\, \beta_N$

# Advanced scenarios and internal transport barriers




## ITER steady-state scenario



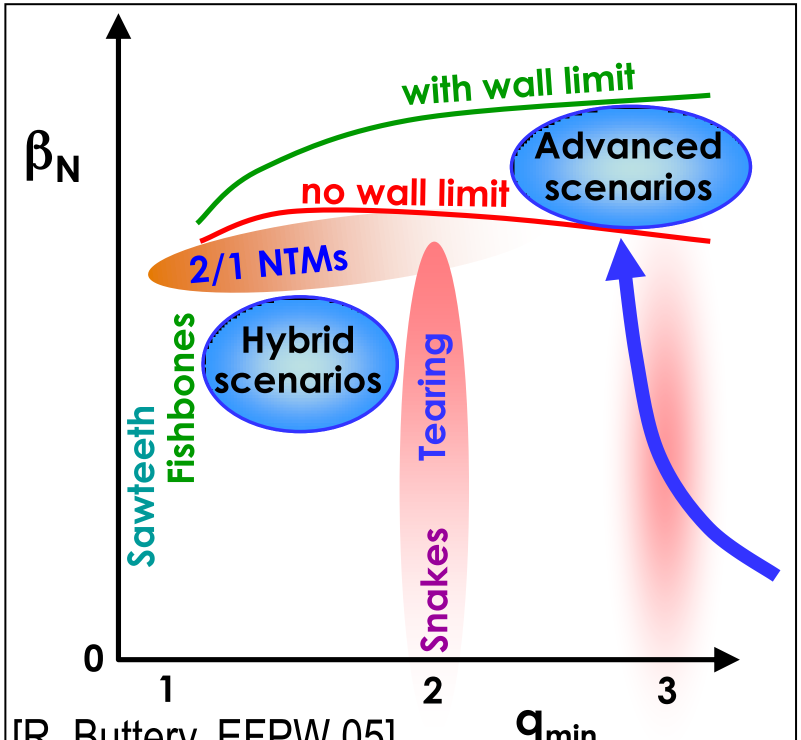

*J. Garcia et al., PRL* ***100*** *(2008) 255004*

Advanced scenarios seems to be a very good candidate for steady state operation

# MHD Stability issues & real time control

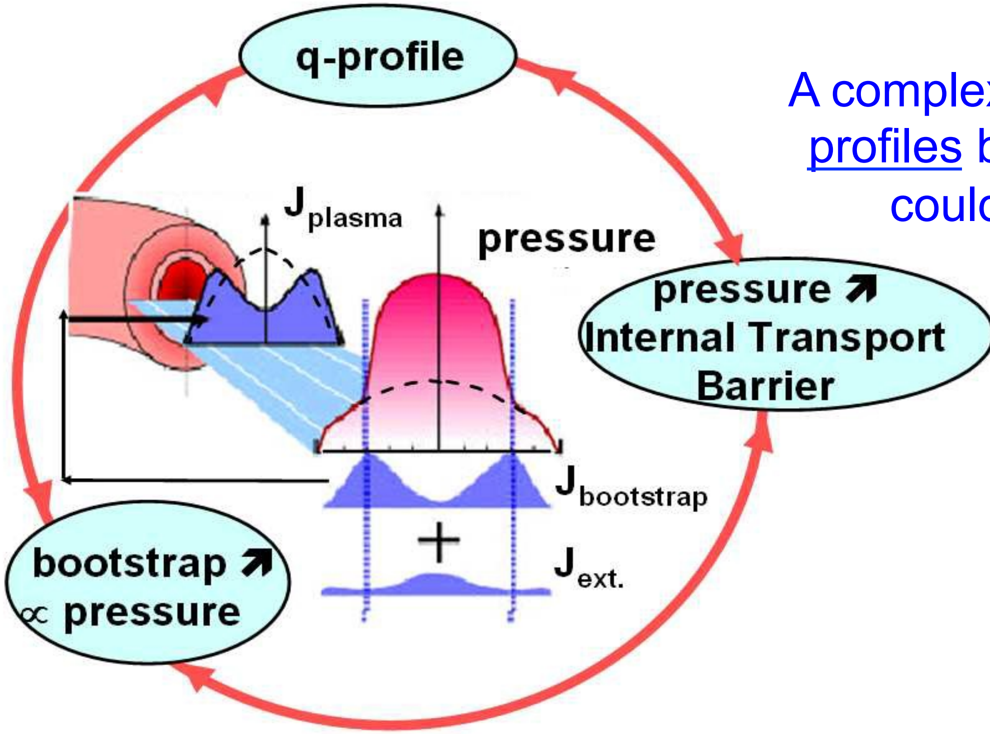
- Edge Localized Modes
  - Damage to Plasma Facing Components
- Neo-classical tearing modes
  - Limiting pressure, risk of disruption
- Resistive wall modes
  - Limiting pressure
- Disruptions
  - High damage
- Fast particle modes
  - Limiting $\alpha$-heating, CD

Necessity for Real Time feedback control
& localized current drive

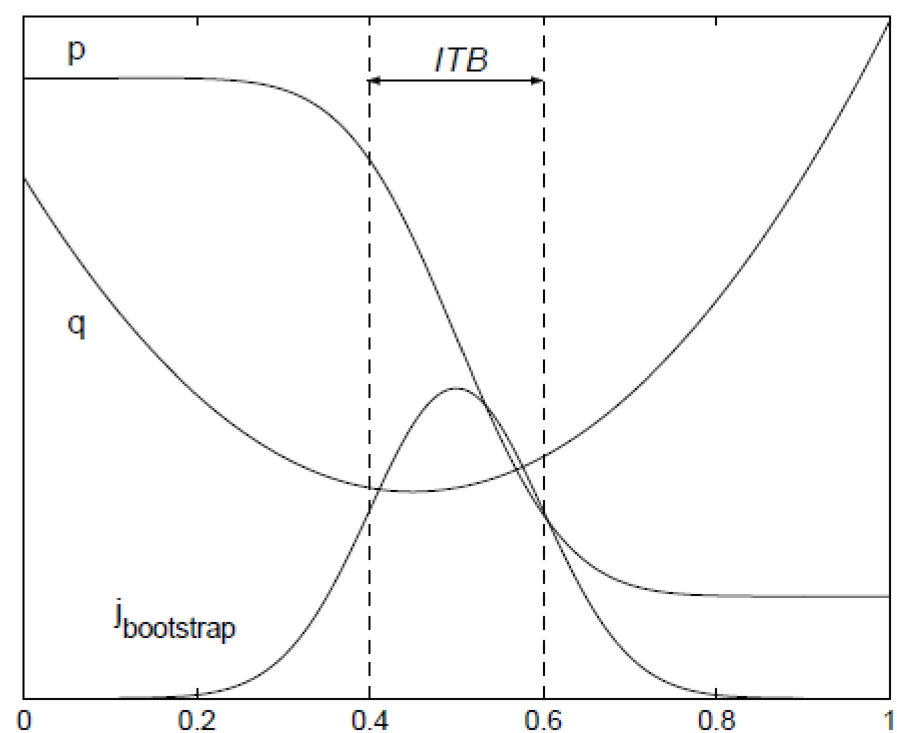


[R. Buttery, EFPW 05]

# Advanced scenarios and internal transport barriers




A complex interplay between current and pressure profiles but potential control of the current profile could lead to improved core confinement




**Challenge**: how to control the strong non linear coupling of the magnetic configuration (q profile), plasma confinement (energy, particle and momentum), bootstrap current, fusion power and MHD stability?

⇨ Real-time control

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# Towards Real time Control of advanced scenarios

The active control of the magnetic and kinetic configuration requires the use of a complete set of real-time:

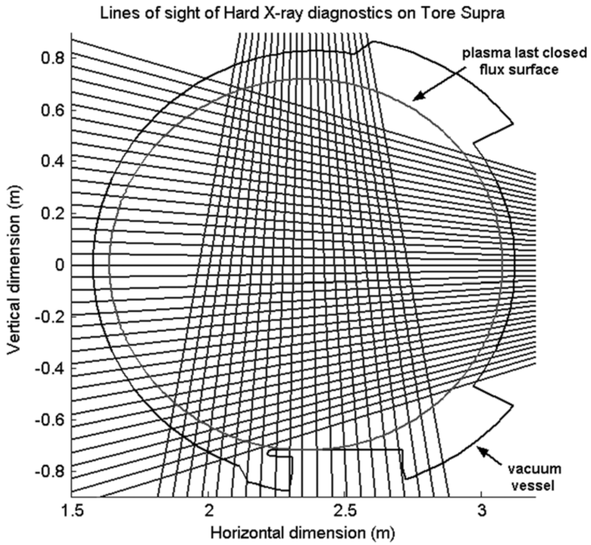
- Sensors  (accurate measurements, time and space resolved)
- Actuators (that can act on the plasma and on its profiles)
- Communication networks (fast enough to transfer the information to local PCs)
- Generic control algorithms (which assume that a plasma model has been identified)

Control algorithms should run during the all duration of the pulse (initiation, current ramp up phase, fusion burn phase during the flat top,burn termination and current ramp down

Initial challenge: Te and Ti measured directly but how to get current density profile in real-time?

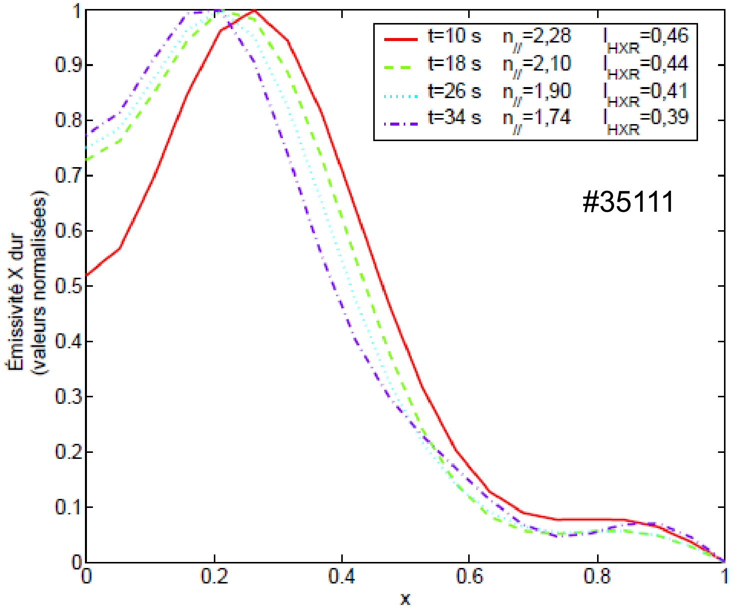
# PROFILES RECONSTRUCTION USING HXR

On Tore Supra, all the current can be driven non inductively by Lower Hybrid waves (LHCD power) during very long duration (several minutes). This wave generates suprathermal electrons that can be seen by Hard X-ray measurement.




59 lines of sights

Abel inversion




Measurements and Abel inversion have been made available in real-time (each 16 ms) authorizing current density profile control on Tore Supra.

# PROFILES RECONSTRUCTION USING POLARIMETRY

## 1. Flux surface reconstruction

$$\begin{cases} R = R_{axis} + \Delta(\rho) + \rho\cos(\vartheta + \gamma(\rho)\sin\vartheta) \\ Z = Z_{axis} + k(\rho)\rho\sin\vartheta \end{cases}$$

$$\Delta(\rho) = \left(\frac{\rho}{\rho_{edge}}\right)^{\alpha s} \Delta_{edge} \quad \text{Magnetic Shift}$$

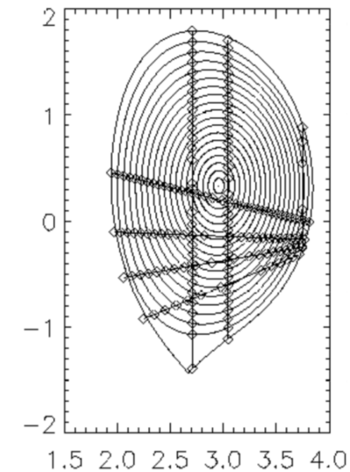
$$\gamma(\rho) = \left(\frac{\rho}{\rho_{edge}}\right)^{\alpha t} \gamma_{edge} \quad \text{Triangularity}$$

$$k(\rho) = \alpha_{1k}\rho^2 + \alpha_{2k}\rho^4 + k_{axis} \quad \text{Elongation}$$

Monotonic variation

$\alpha_s$, $\alpha_t$, $\alpha_{1k}$, $\alpha_{2k}$ determined using Boundary conditions

## 2. Intersection with lines of sights




## 3. Density reconstruction ($\chi^2$ resolution)

$$N_i = C_N \int_{l_i} n_e(l)dl \quad \text{Interferometer KG1V} \qquad i = 2\ldots8$$

$$n_e(x) = n_0(1-\rho^2)\cdot(1+p\rho^2+q\rho^4)+n_w \quad \text{Density formula}$$

## 4. Poloidal field reconstruction ($\chi^2$ resolution)

$$\Delta\Gamma_i = C_B \int_{l_i} n_e(l) B_{\vartheta,II} dl \quad \text{FIR-KG4} \qquad i = 2\ldots8$$

$$B_\vartheta^R = -\frac{1}{2\pi R}\frac{\partial\psi}{\partial\rho}\frac{\partial\rho}{\partial Z}$$

$$B_\vartheta^Z = \frac{1}{2\pi R}\frac{\partial\psi}{\partial\rho}\frac{\partial\rho}{\partial R}$$

Known from geometry

Poloidal field components

$$\frac{\partial\psi}{\partial\rho} = a_0\rho^3 + a_1\rho^2 + a_2\rho$$

Poloidal Field formula

# PROFILES RECONSTRUCTION USING POLARIMETRY

5. q profile calculation

$$q = \frac{\Delta\Phi}{\Delta\Psi}$$ Expression used for q_profile

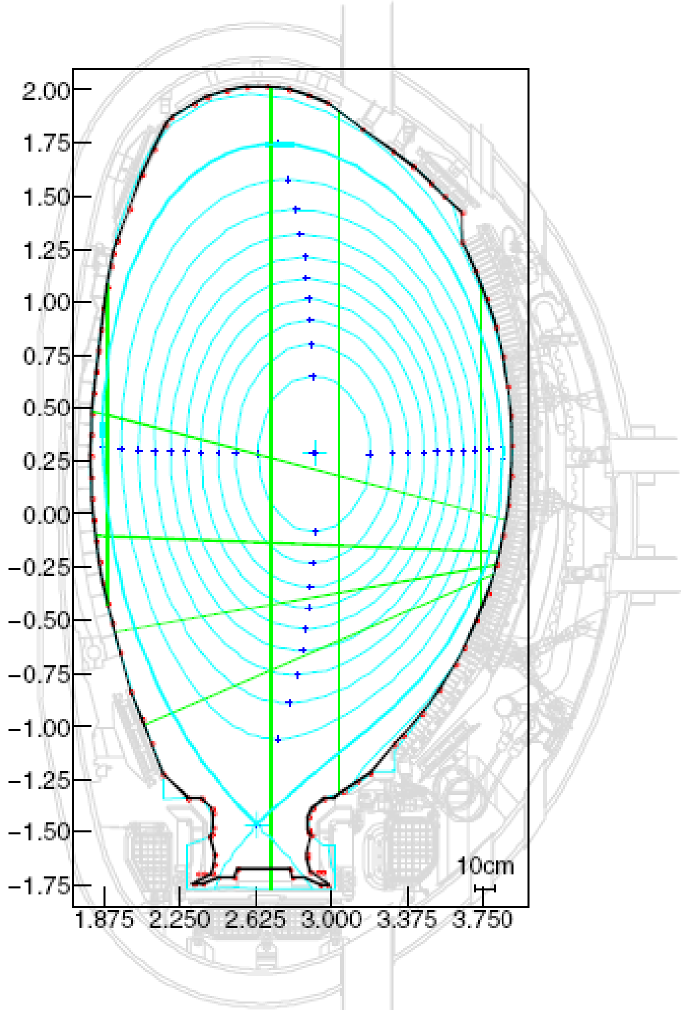
$$\Delta\Phi = \Phi_2 - \Phi_1 \qquad \Delta\Psi = \Psi_2 - \Psi_1$$

$$\Phi = \int_\Omega B_\varphi d\Omega$$ Toroidal flux

$$\Psi = \int_\Sigma B_\vartheta d\Sigma$$ Poloidal flux

Lots of efforts including validation of the results (density and q profiles).
Such method has been implemented in real-time at JET in 2002.
q profiles were reconstructed each 50ms.

# PROFILE RECONSTRUCTION USING GRAD SHAFRANOV SOLVER




In the presence of a magnetic field *B* the equations governing the plasma equilibrium are

- the magnetostatic Maxwell's equations

$$\begin{cases} \nabla . B = 0 \\ \nabla x\left(\dfrac{B}{\mu}\right) = j \end{cases}$$

- the equilibrium for the plasma itself which can be written as follow:

$$\boxed{\nabla p = jxB}$$

- It is clear from this last equation that the plasma is in equilibrium when the force $\nabla p$ due to the kinetic pressure is equal to the Lorentz force of the magnetic pressure *jxB*.

# PROFILE RECONSTRUCTION USING GRAD SHAFRANOV SOLVER

$$-\Delta^*\psi = rp'(\psi) + \frac{1}{\mu_0 r}(ff')(\psi)$$

Defining the poloidal flux by

$$\boxed{\begin{cases} B_r = -\dfrac{1}{r}\dfrac{\partial\psi}{\partial z} \\ B_z = \dfrac{1}{r}\dfrac{\partial\psi}{\partial r} \end{cases}}$$

Outside the plasma region $\Delta^*\psi = 0$

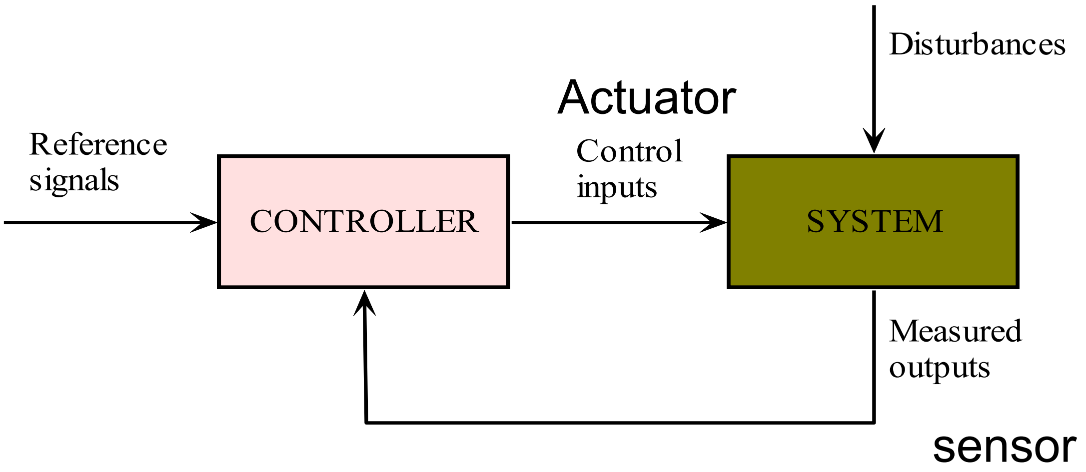
$$\Delta^* = \frac{\partial}{\partial r}\left(\frac{1}{\mu_0 r}\frac{\partial}{\partial r}\right) + \frac{\partial}{\partial z}\left(\frac{1}{\mu_0 r}\frac{\partial}{\partial z}\right)$$

The problem of the equilibrium of a plasma in a Tokamak is a free boundary problem in which the plasma boundary is defined as the last closed magnetic flux surface. Inside the plasma, the equilibrium equation in an axisymmetric configuration is called the Grad-Shafranov equation. The right hand side of this equation is a non-linear source which represents the toroidal component of the plasma current density.

A new real-time equilibrium code (EQUINOX) has been installed at JET and Tore Supra

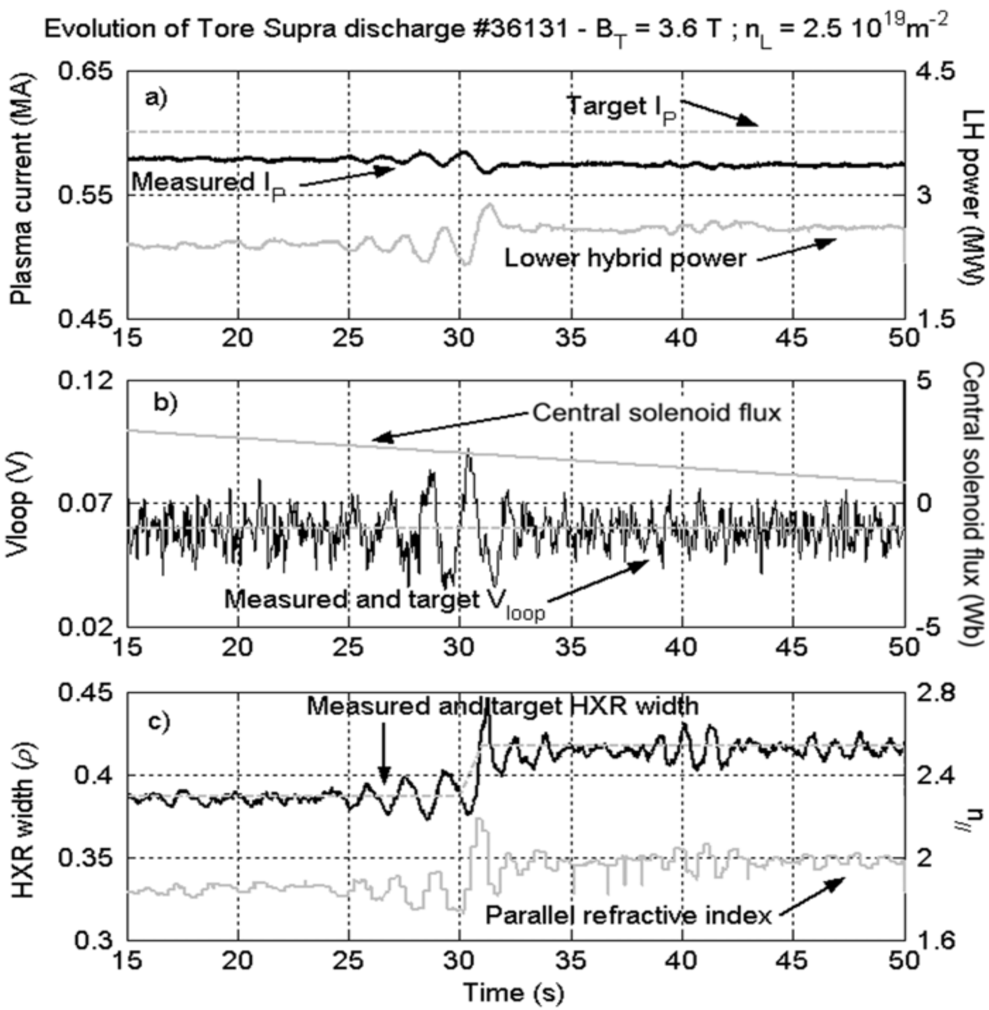
# CONTROLLING GLOBAL PARAMETERS

One parameter is controlled by One actuator.
Simple Proportional Integration Derivative (PID) controller is used
Fast and easy to implement
Gains are either adjusted empirically or estimates by open loops
Several examples can be found in different present tokamaks
This kind of control is routinely applied, and several decoupled PID can be applied



# CONTROLLING GLOBAL PARAMETERS
# HXR WIDTH

Global current profiles control using HXR reconstruction on Tore Supra

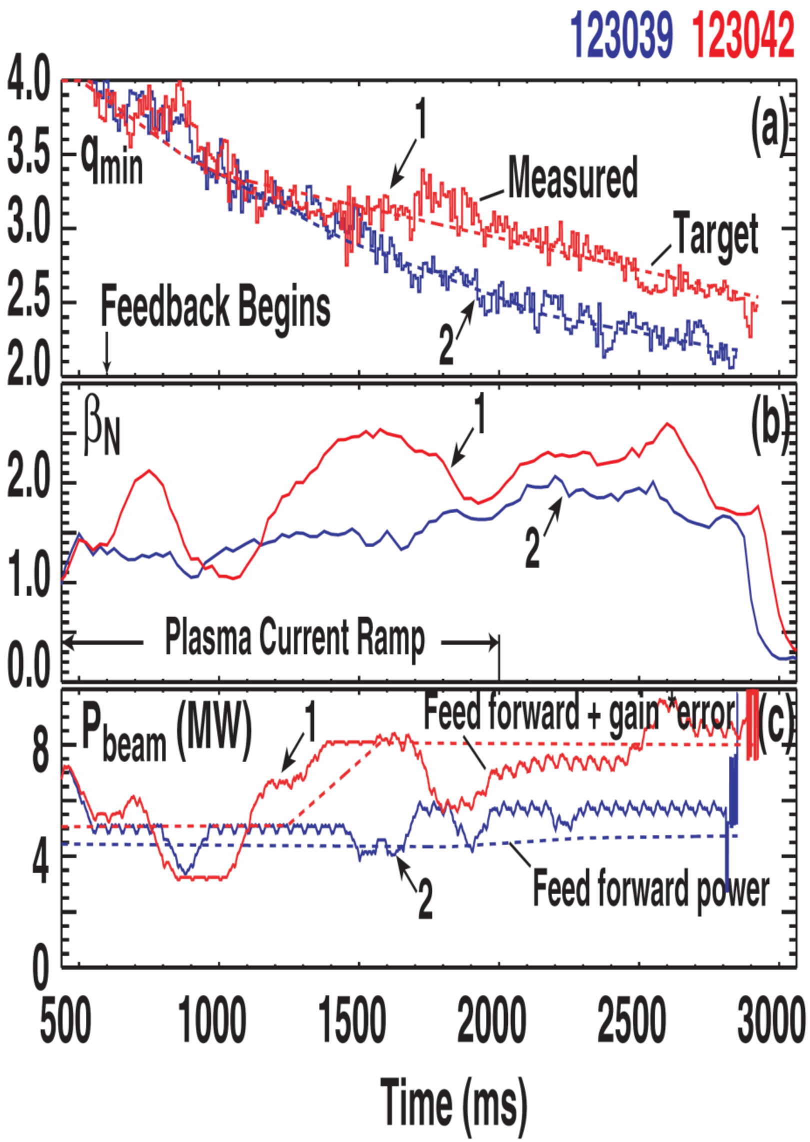



Integrated control
Control of plasma current with Lower Hybrid power

Control of the loop voltage by the central solenoid flux

Control of HXR width by refractive parallel index

# CONTROLLING GLOBAL PARAMETERS
# $Q_{MIN}$



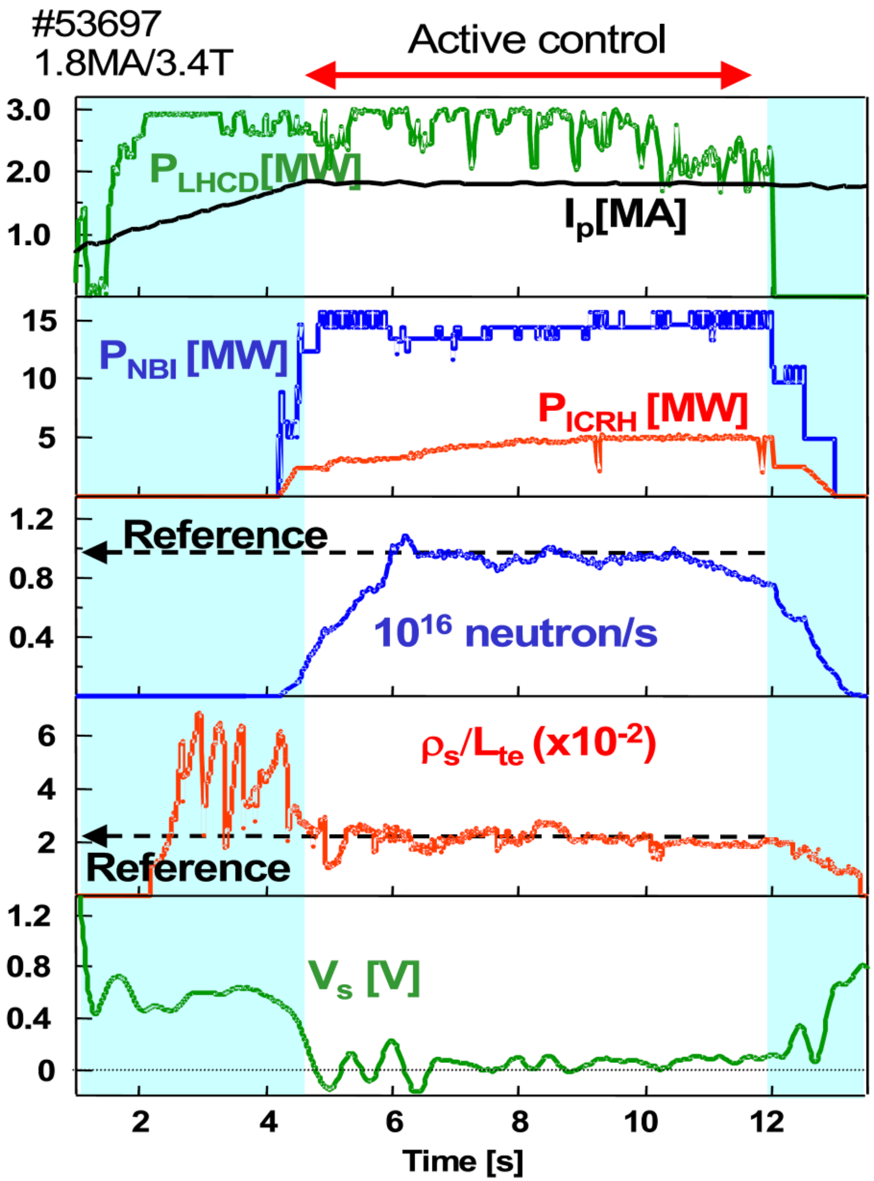

Feedback control of $q_O$ or $q_{min}$ during the plasma current ramp-up phase

Change of plasma conductivity through electron heating with NBI

RT q-profile using MSE data (Real time equilibrium code EFIT)

[J. Ferron et al Nuc Fus 2006]

# CONTROLLING GLOBAL PARAMETERS ITB STRENGH




$P_{NBI}$ RT controlled by neutron

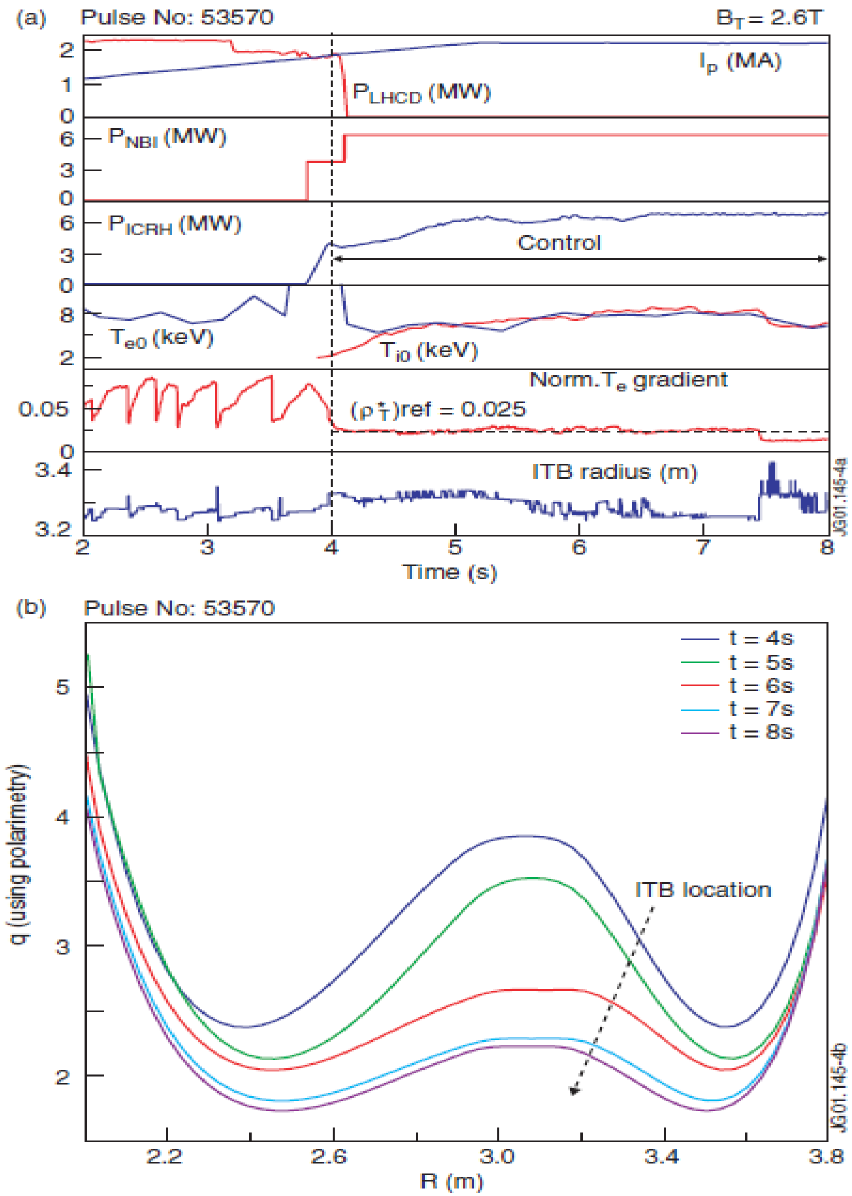
$P_{ICRH}$ RT controlled by $\rho_s/L_{Te}$ **where $L_T = \nabla T/T$**

proportional-integral

$$P(t)[MW] = P(t_0) + G_p \Delta X(t) + G_I \int_{t_0}^{t} \Delta X(u)du,$$

[Mazon, Litaudon, Moreau et al PPCF 02]

# CONTROLLING GLOBAL PARAMETERS AT JET USING POLARIMETRY RECONSTRUCTION




ITB strength perfectly controlled

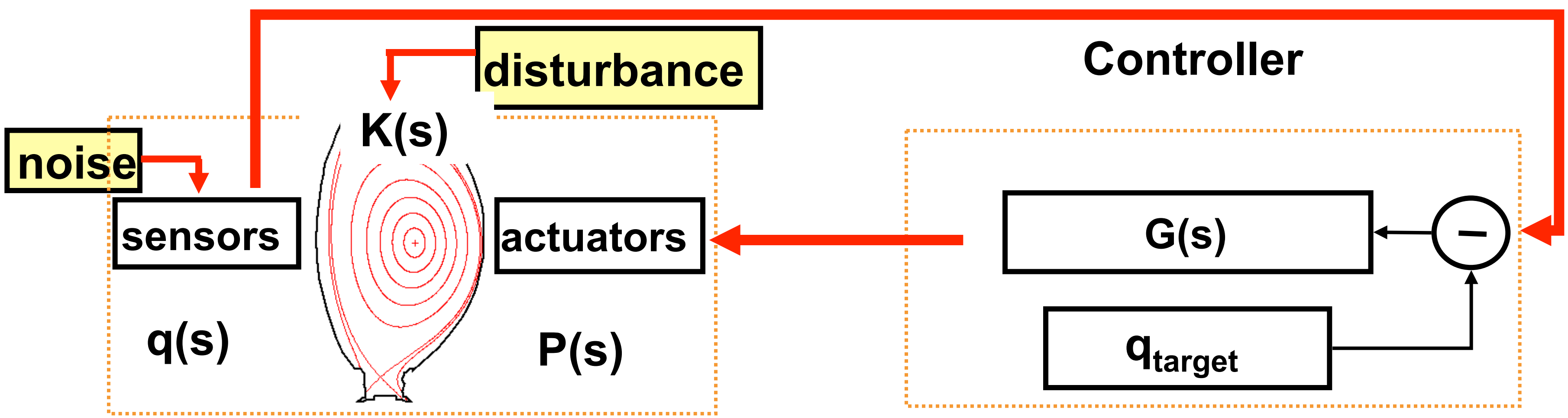
But q profile evolves and the ITB position schrinks

Demonstrated clearly the need to control simultaneously <u>kinetic and magnetic profiles.</u>

[Mazon, Litaudon, Moreau et al PPCF 02]

# MULTI-INPUT-MULTI-OUTPUT (MIMO) MODEL BASED PROFILE CONTROL

## All transfer functions are in matrix form




sensors plasma actuator

Controller

$\delta q(s)=K(s)\,\delta P(s)$

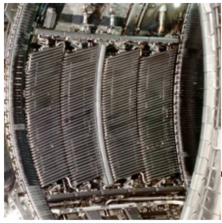
$\delta P(s)= G(s)\,\delta q(s)$

First approach: control based on pseudo-inverse of the steady-state gain matrix

$G(s)= g_c[1 + 1/(\tau_i s)]\,K(0)_{inv}$

[D. Moreau et al Nucl Fus 2003, D. Moreau et al Nucl Fus 2008]

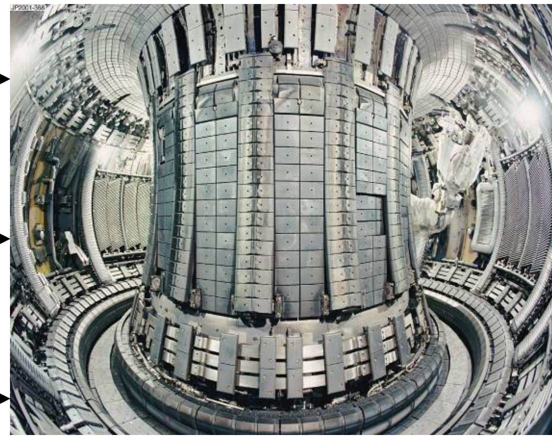
# MULTI-INPUT-MULTI-OUTPUT (MIMO) MODEL BASED PROFILE CONTROL

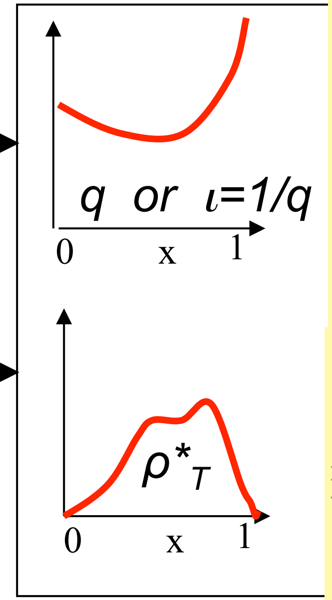


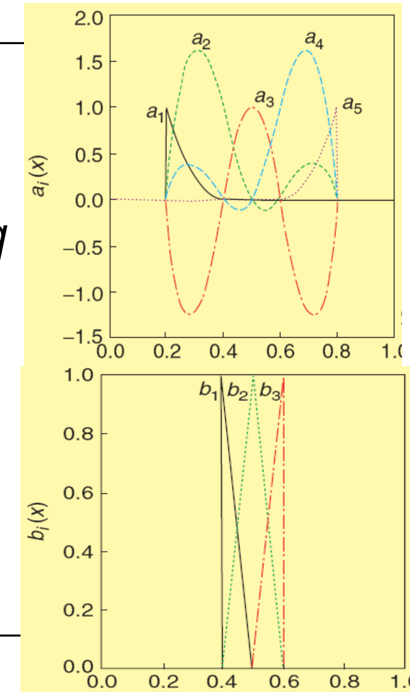
ICRH

LHCD

NBI





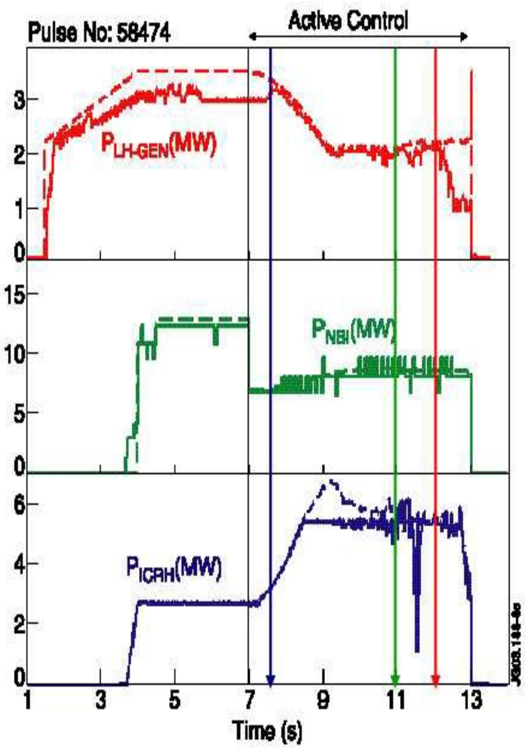

Galerkin's projection of 1/q and $\rho^*_T$

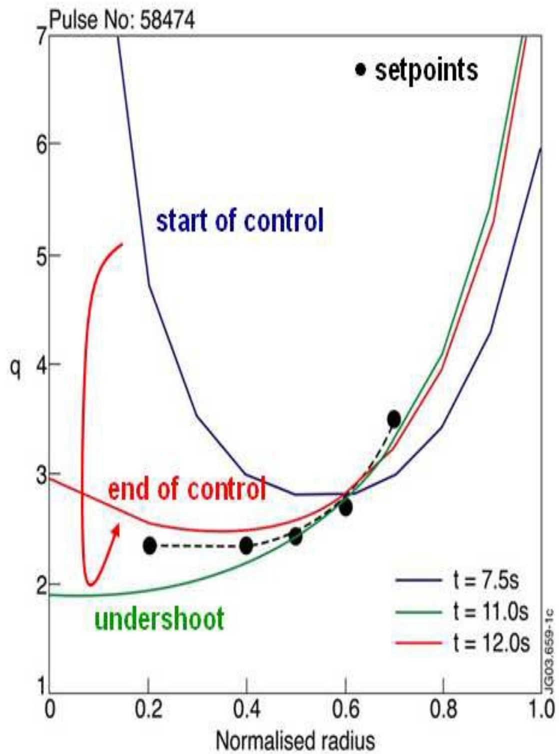
modulations around a reference steady-state:

$$\begin{bmatrix} K_{11} & K_{12} & K_{13} \\ K_{21} & K_{22} & K_{23} \\ K_{31} & K_{32} & K_{33} \\ K_{41} & K_{42} & K_{43} \\ K_{51} & K_{52} & K_{53} \\ K_{61} & K_{62} & K_{63} \\ K_{71} & K_{72} & K_{73} \\ K_{81} & K_{82} & K_{83} \end{bmatrix} \cdot \begin{bmatrix} \delta P_{ICRH} \\ \delta P_{LH} \\ \delta P_{NBI} \end{bmatrix} = \begin{bmatrix} \delta G\iota 1 \\ \delta G\iota 2 \\ \delta G\iota 3 \\ \delta G\iota 4 \\ \delta G\iota 5 \\ \delta G\rho_T^* 1 \\ \delta G\rho_T^* 2 \\ \delta G\rho_T^* 3 \end{bmatrix}$$

[D. Moreau et al Nuc Fus 2008, T. Tala et al Nuc. Fus 2005]

## MIMO Control (Multi-Input, multi-output)




"Model Based" control on 5 q-values

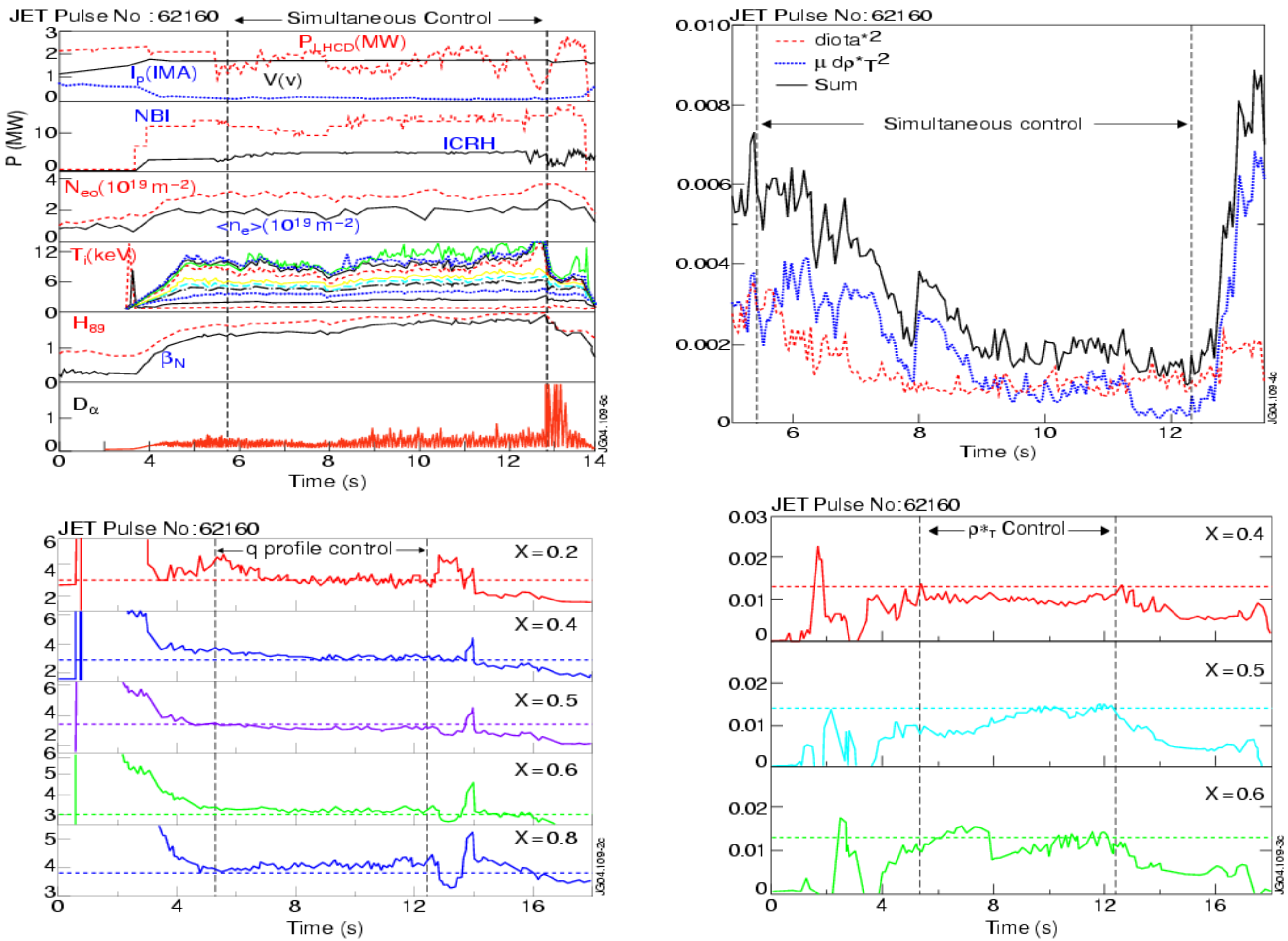
Three actuators used: LHCD, NBI and ICRH Powers

**Model based SVD control: steady-state gain matrix deduced from open loop experiments**

[D. Moreau, D. Mazon et al Nucl Fus 2003]

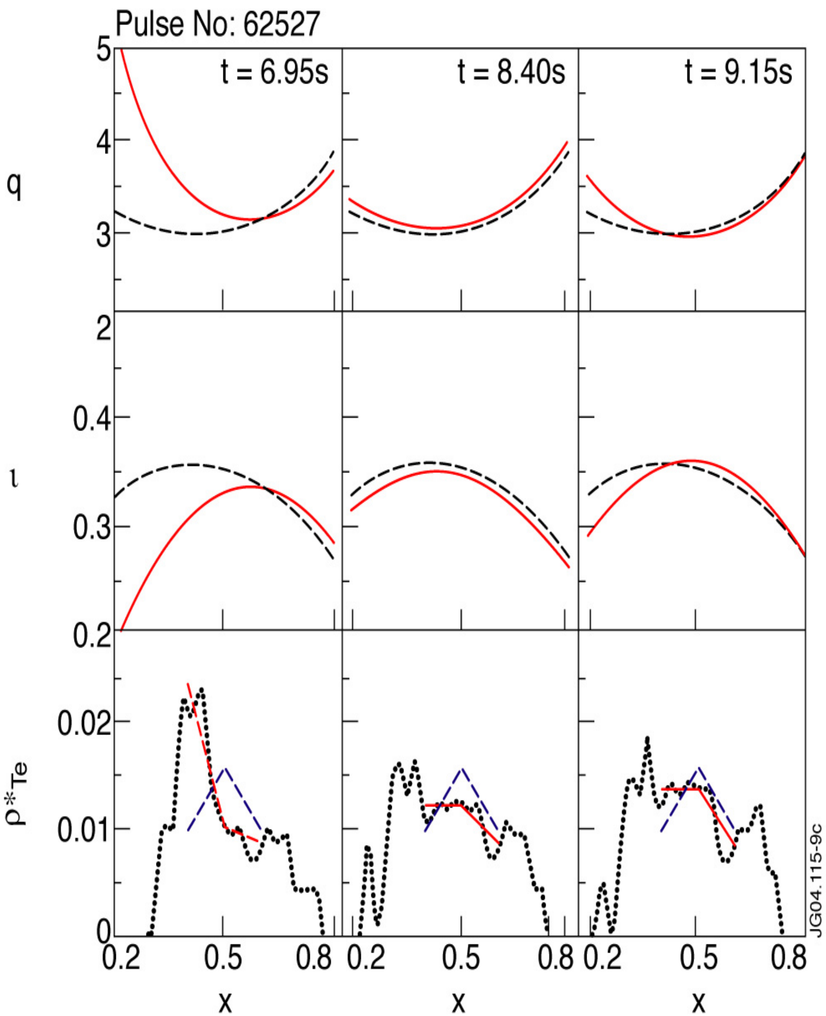
# SIMULTANEOUS CONTROL OF q AND PRESSURE PROFILES (STATIC MODEL)

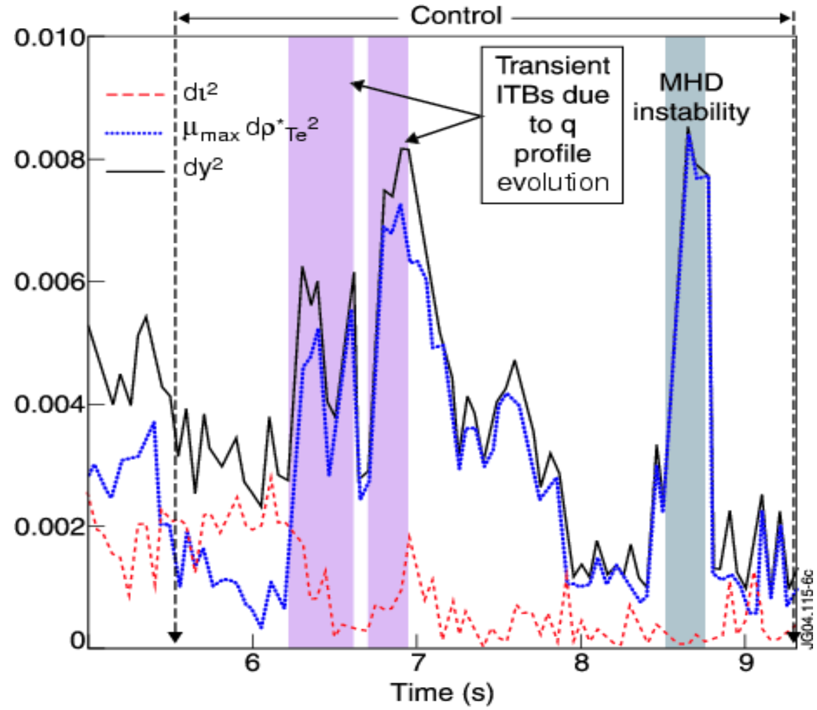
## Target profiles perfectly reached




[D. Moreau et al Nucl Fus 2003]

# MHD LIMITATIONS AND FAST TRANSIENTS




Limitation of steady-state gain matrix approach: controller response slow during fast events

Development of an optimal control (time dependent model)

[Laborde, Mazon, Moreau et al., PPCF (2005), Tala et al Nuc. Fus 2005, Moreau et al Nuc Fus 2008]

# TWO TIME SCALE MODEL

[Moreau, Mazon et al Nuc Fus 2008]

**Slow model: resistive time scale**

$$\frac{d\Psi}{dt} = \mathcal{A}_{slow} \cdot \Psi(t) + \mathcal{B}_{slow}\, U_{slow}(t)$$

$$\begin{bmatrix} V_\Phi(t) \\ T_i(t) \end{bmatrix}_{slow} = \mathcal{C}_{slow} \cdot \Psi(t) + \mathcal{D}_{slow}\, U_{slow}(t)$$

**Slow controller :** $U_{slow}(t)$

*opt. feedback*

**Fast model: momentum / thermal confinement time scale, $\tau = \varepsilon\, t \quad \varepsilon \ll 1$**

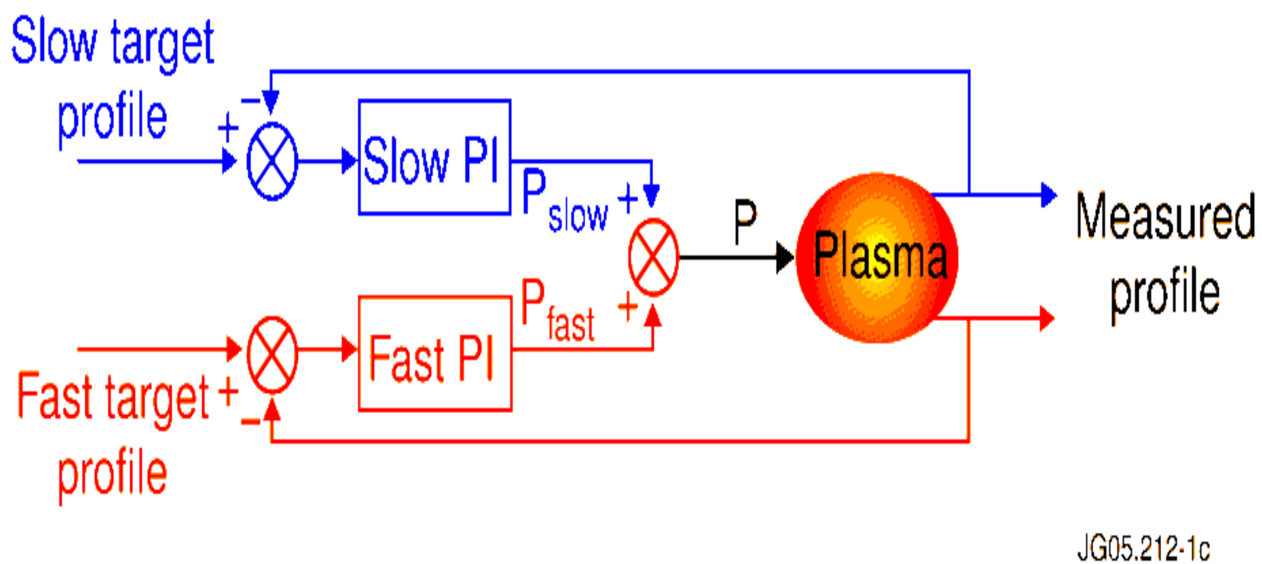
$$\frac{d}{d\tau}\begin{bmatrix} V_\Phi(\tau) \\ T_i(\tau) \end{bmatrix}_{fast} = \mathcal{A}_{fast} \cdot \begin{bmatrix} V_\Phi(\tau) \\ T_i(\tau) \end{bmatrix}_{fast} + \mathcal{B}_{fast}\, U_{fast}(\tau)$$

**Fast controller :**

$$U(t) = U_{slow}(t) + U_{fast}(\varepsilon t) \qquad \begin{bmatrix} V_\Phi(t) \\ T_i(t) \end{bmatrix} = \begin{bmatrix} V_\Phi(t) \\ T_i(t) \end{bmatrix}_{slow} + \begin{bmatrix} V_\Phi(\varepsilon t) \\ T_i(\varepsilon t) \end{bmatrix}_{fast}$$

*opt. feedback*

# HOW IT WORKS DURING THE CONTROL

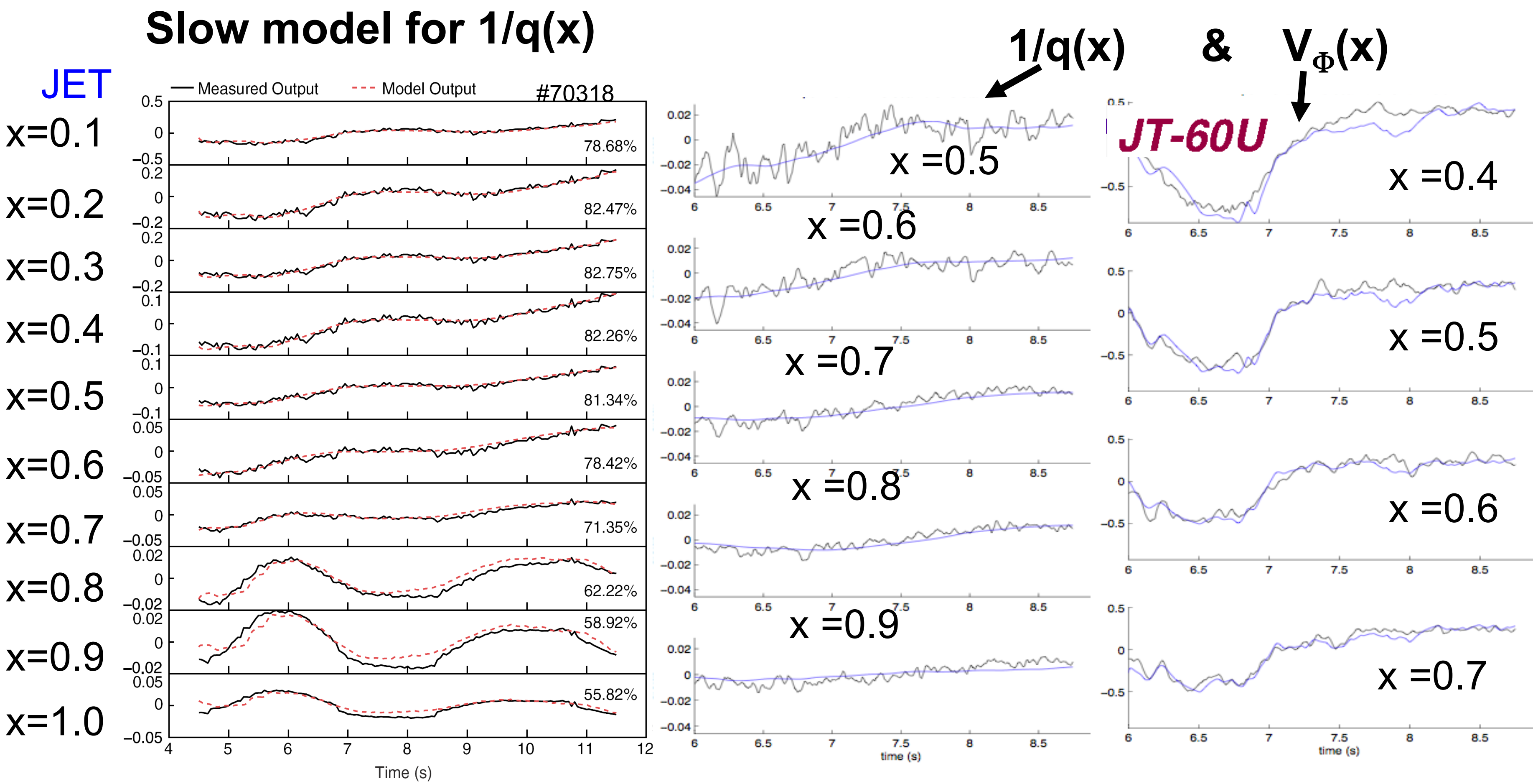



Targets are defined (for q and $\rho$*T)
Fast and slow targets are computed from the original targets
Pslow and Pfast are calculated by the controller

This methodology is not machine depending as soon as data are available in real time and has been applied to many tomamaks (JET, JT60U, DIII-D)

# JET AND JT-60U APPLICATION (MODEL IDENTIFICATION)

Very good agreement found between model and measured data on different quantities
Presently application to DIII-D data are on going.




[Moreau et al Proc. 48th IEEE Conf. on Decision/Control China 2008, IAEA 2010]

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**Part5:Conclusions and perspectives**

# CONTROLLING IMPURITY DISTRIBUTION IN ORDER TO IMPROVE THE CONTROL OF ADVANCED SCENARIOS

With the new metallic walls (W and/or Be) in JET, ASDEX, WEST and ITER, the impurities are now playing a dominant role and their effect could affect strongly the control of advanced scenarios.

For example it is well known that impurities accumulation can be the cause

- enhance fuel dilution in thermonuclear plasmas
- radiate significant amounts of energy out of them
- Trigger disruptions
- reveal abnormal erosion of PFCs

Major challenge: How to measure impurity distribution?

# CONTROLLING IMPURITY DISTRIBUTION IN ORDER TO IMPROVE THE CONTROL OF ADVANCED SCENARIOS

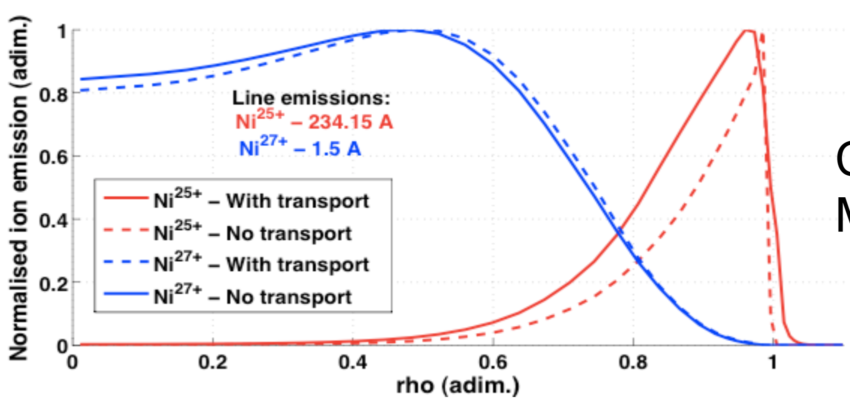
SXR measurement can provide very nice information about impurity distribution.

$$\varepsilon = n_e \times n_s \times \sum_{Z=0}^{Zs} \left( f_Z^s(T_e, n_e) \times K_Z^s(T_e, n_e) \right)$$

Recent studies have showed that at high temperature and for high Z impurities transport plays little role in the determination of fractional abundances which can be thus tabulated.

Discrimination in energy seems to play a role in the domain of validity of this hypothesis (develop new SXR detector (GEM) with capacity to discriminate in energy)

ITER case

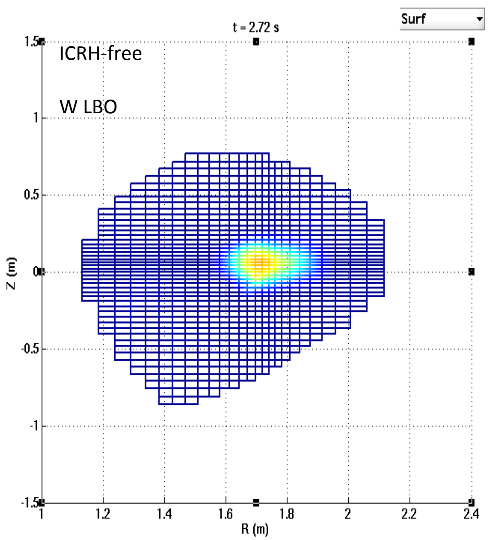


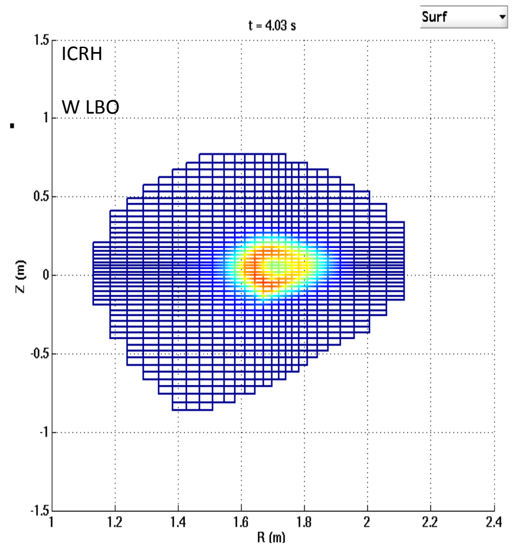

Courtesy M. O' Mulanne

More in D. Vezinet, D. Mazon et al. FST 2013

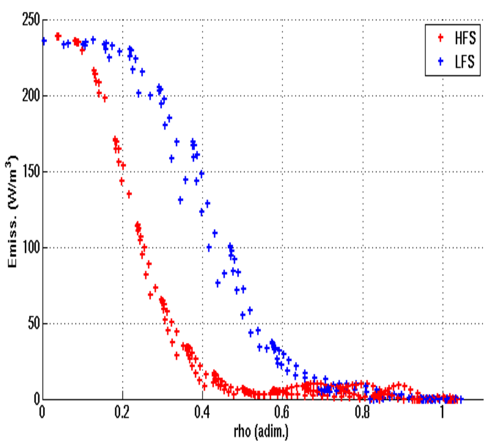
# CONTROLLING IMPURITY DISTRIBUTION IN ORDER TO IMPROVE THE CONTROL OF ADVANCED SCENARIOS

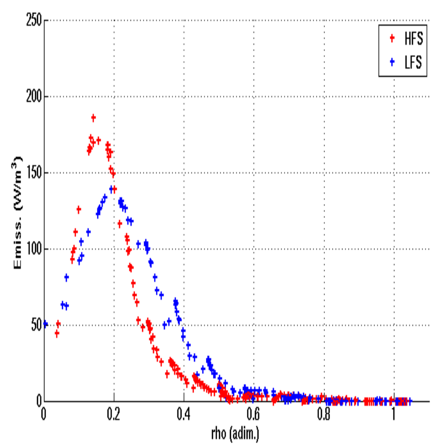
## Recent preliminary attempts to identify actuators (ASDEX U)




Tomographic reconstruction during W injection with and without ICRH




Related LFS/HFS asymmetries on the equatoria plane

# Real-time fusion DT burn control

In present day machines, **developing real-time control techniques in high bootstrap regimes in condition where the effects of the α particles heating source could be mimicked experimentally is a real challenge.**

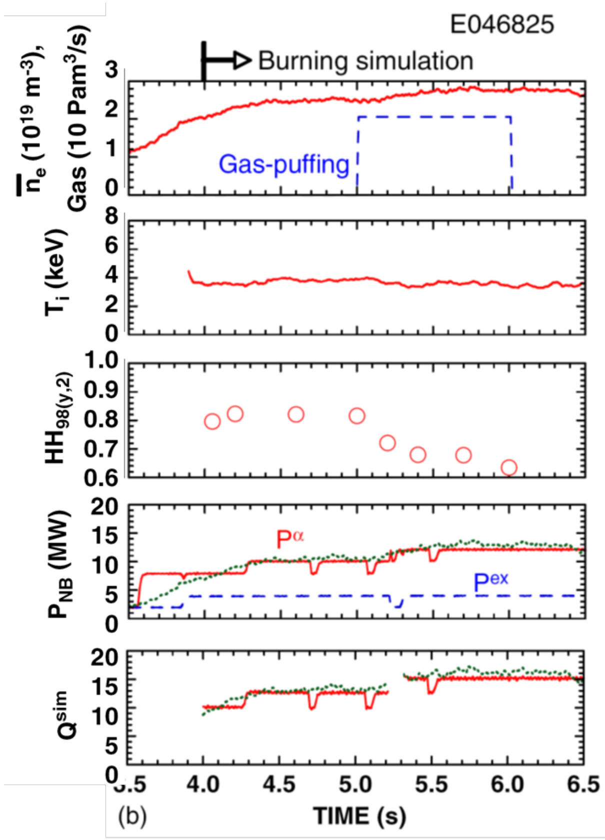
The fusion α particles effects add extra complexity in the control of the fusion performances. Indeed the **α particles heating is determined by the plasma profiles but at the same time the plasma kinetic and magnetic profiles are taylored by the α heating.**

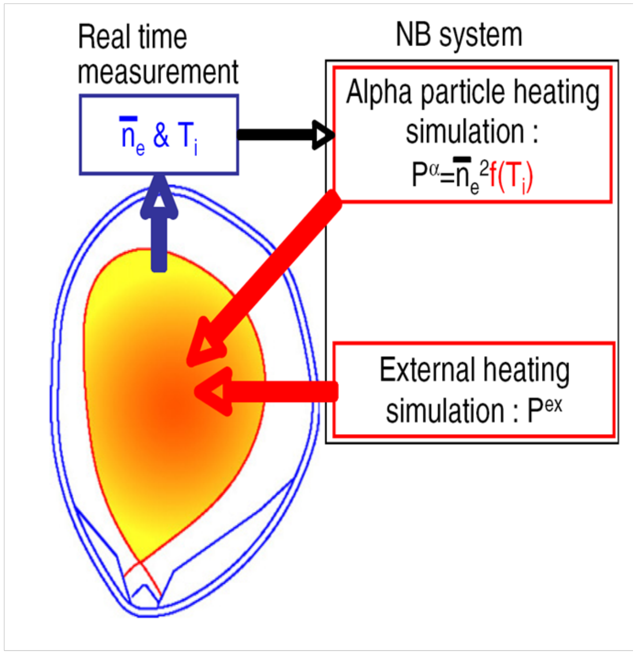
**This is an open field of research and lots of efforts of simulation and experiments are required** if we want one day to develop relevant control techniques for burning plasmas conditions minimizing the requirements in term of external heating powers while maximizing the fusion production

# Real-time fusion DT burn control

**Modelisation and experiments are nevertheless conducted.**

In experiments, $\alpha$ particle self heating power is mimicked using an external heating source.




[Takenaga et N Fus 2008]

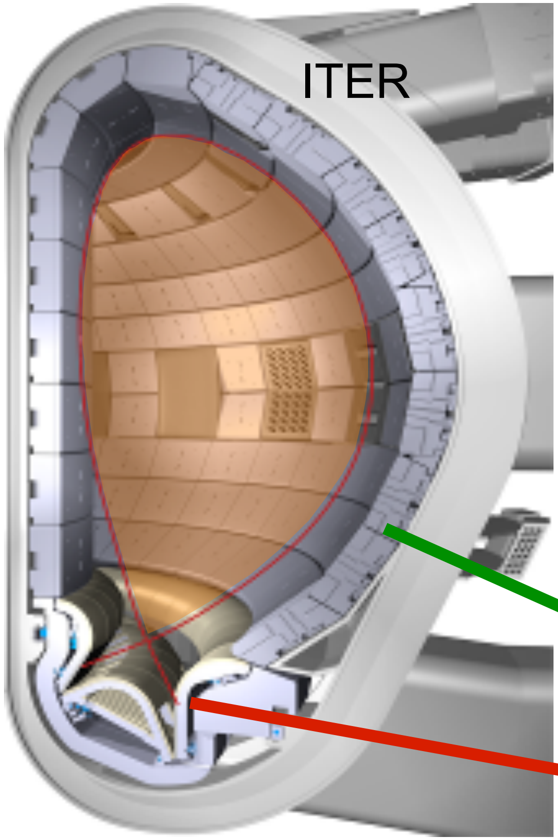
H-mode plasma

Larger dependence than expected of the fusion gain $Q_{DT}$ on $n_e$ found

Key features of the dynamics of a burning plasma are clearly reproduced

Address future control aspects

# Core performance control with plasma facing components constraints




Wall Scenarios Compatibility:

- maximum performance
- minimum T-retention
- minimum erosion
- maximum life-time of PFC

ITER plasma facing components

- Be wall
- Divertor: W-baffle

➢ **Effort in EU tokamaks to investigate PFC-scenario issues**

- Tore Supra (WEST): long pulse operation with actively cooled CFC components
- ASDEX Upgrade: conversion to all tungsten PFCs complete
- JET: installation of beryllium wall and tungsten divertor in 2010

# Core performance control with plasma facing components constraints

➢ **Control steady-state fusion performance**

– **control of kinetic & magnetic energy**

boundary co

dition

Conflicting actions for example

Reduction of the stationary power loads requires to inject high Z impurity to radiate the heat load on the PFCs, while optimisation of the fusion performance requires an increase of the additional power and a reduction of the core impurity content.

The controller should optimize in the least square sense the various edge and core constrains with various weight on each control (edge and core) parameters depending on the required optimisation.

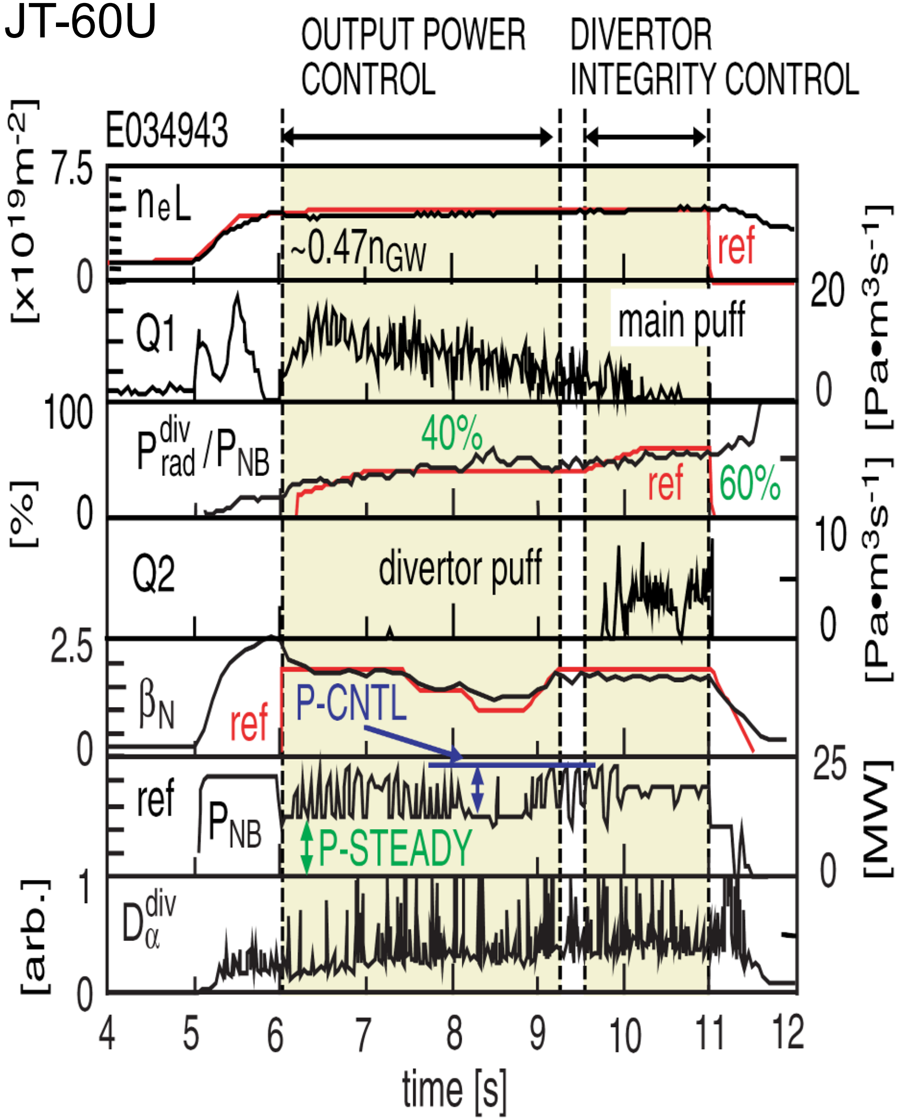
New field of research aiming at controlling both edge and core constraints

fuelling requirements, capacity of Tritium extraction plant & necessary Helium removal

[Litaudon et al  EPS PPCF 2007]

# Core performance control with plasma facing components constraints

JT-60U




[Fukuda et al FST 2002]

Simultaneous control of

- Density by gas puffing near the top of the vessel
- Divertor radiation by gas puffing in divertor
- Energy content by NBI power

Non-diagonal matrix control between actuators & sensors deduced from open loop experiments

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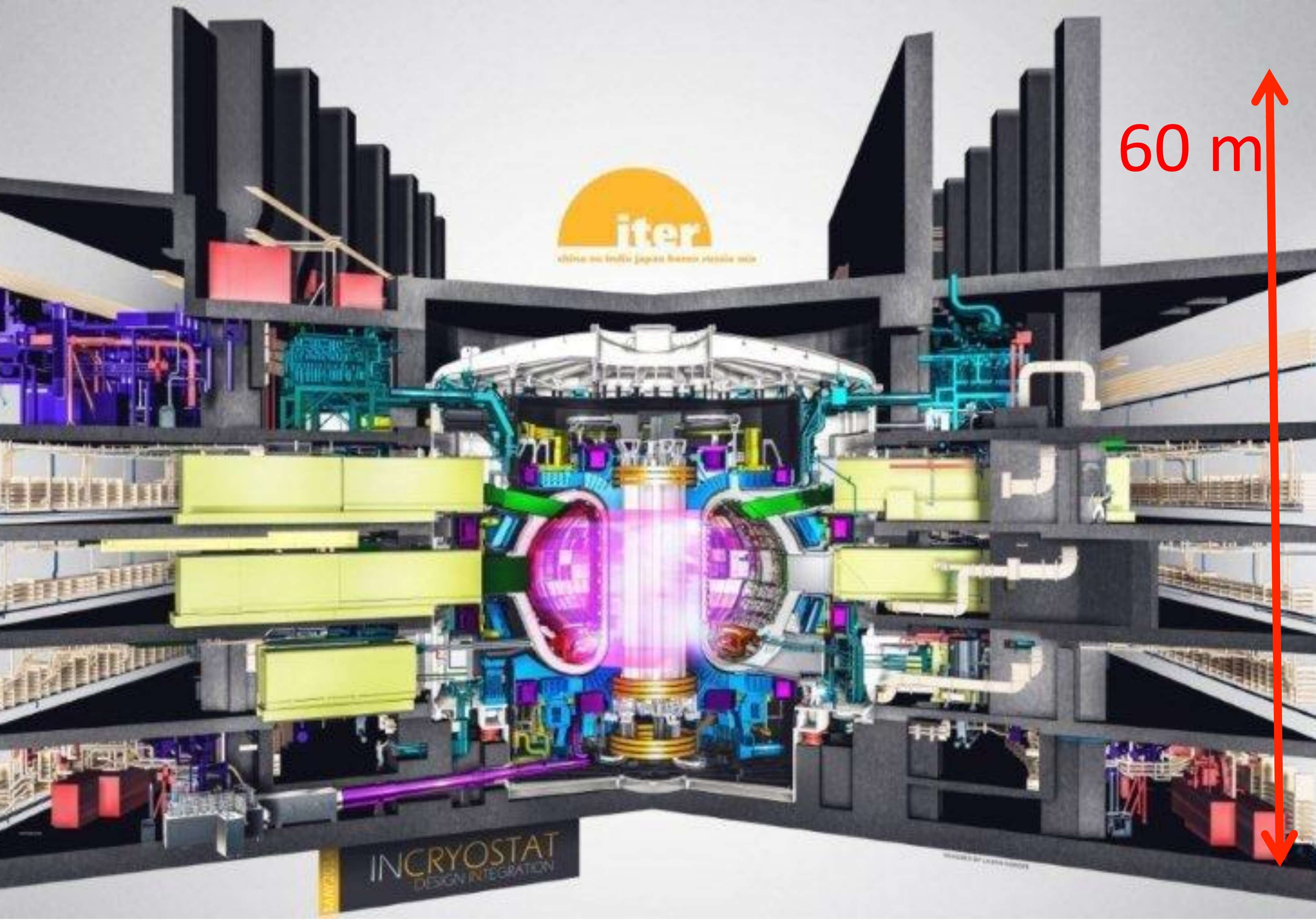
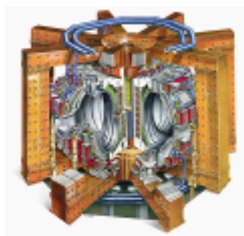
# ITER

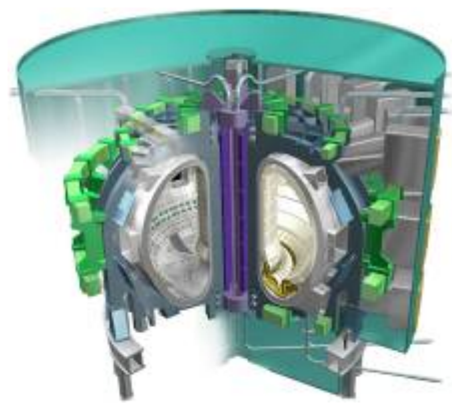
ITER = International Thermonuclear Experimental Reactor

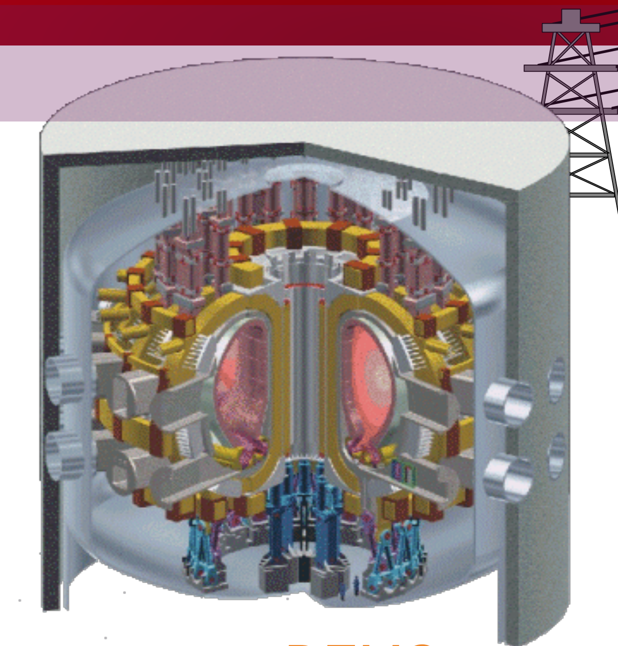

60 m
iter
china eu india japan korea russia usa
INCRYOSTAT
DESIGN INTEGRATION

# ITER: a key step towards fusion energy

## Aim: produce 10 times the energy input



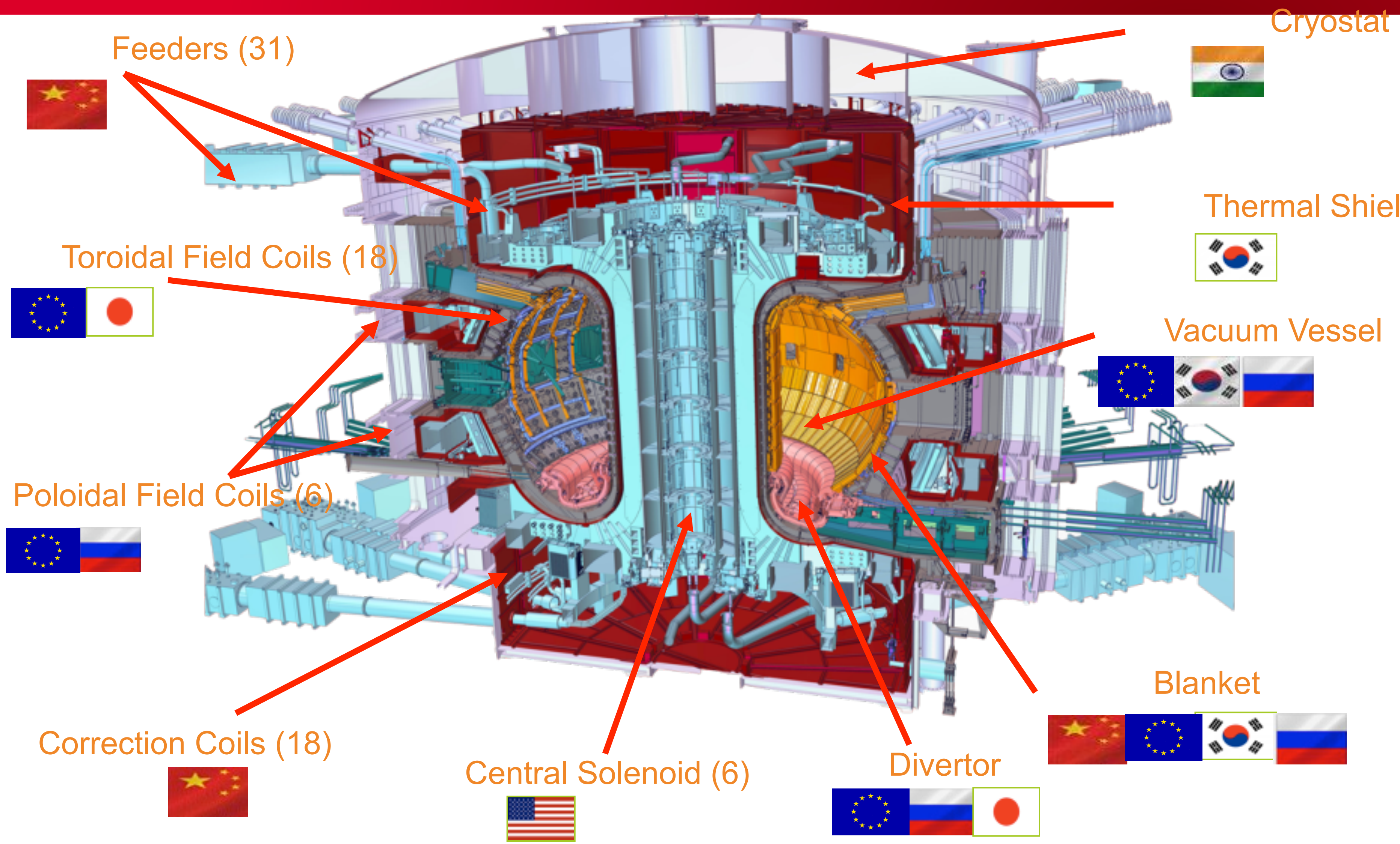




| *Tore Supra* | *JET* | *ITER* | *DEMO* |
|---|---|---|---|
| *25 $m^3$* | *80 $m^3$* | *800 $m^3$* | *~ 1000 - 3500 $m^3$* |
| *~ 0* | *~ 16 $MW_{th}$* | *~ 500 $MW_{th}$* | *~ 2000 - 4000 $MW_{th}$* |
| Q ~ 0 | Q ~ 1 | Q ~ 10 | Q ~ 30 |
| 0% | 10 % | 70 % | 80 to 90 % |

*Self heating*

# WHO MANUFACTURES WHAT?

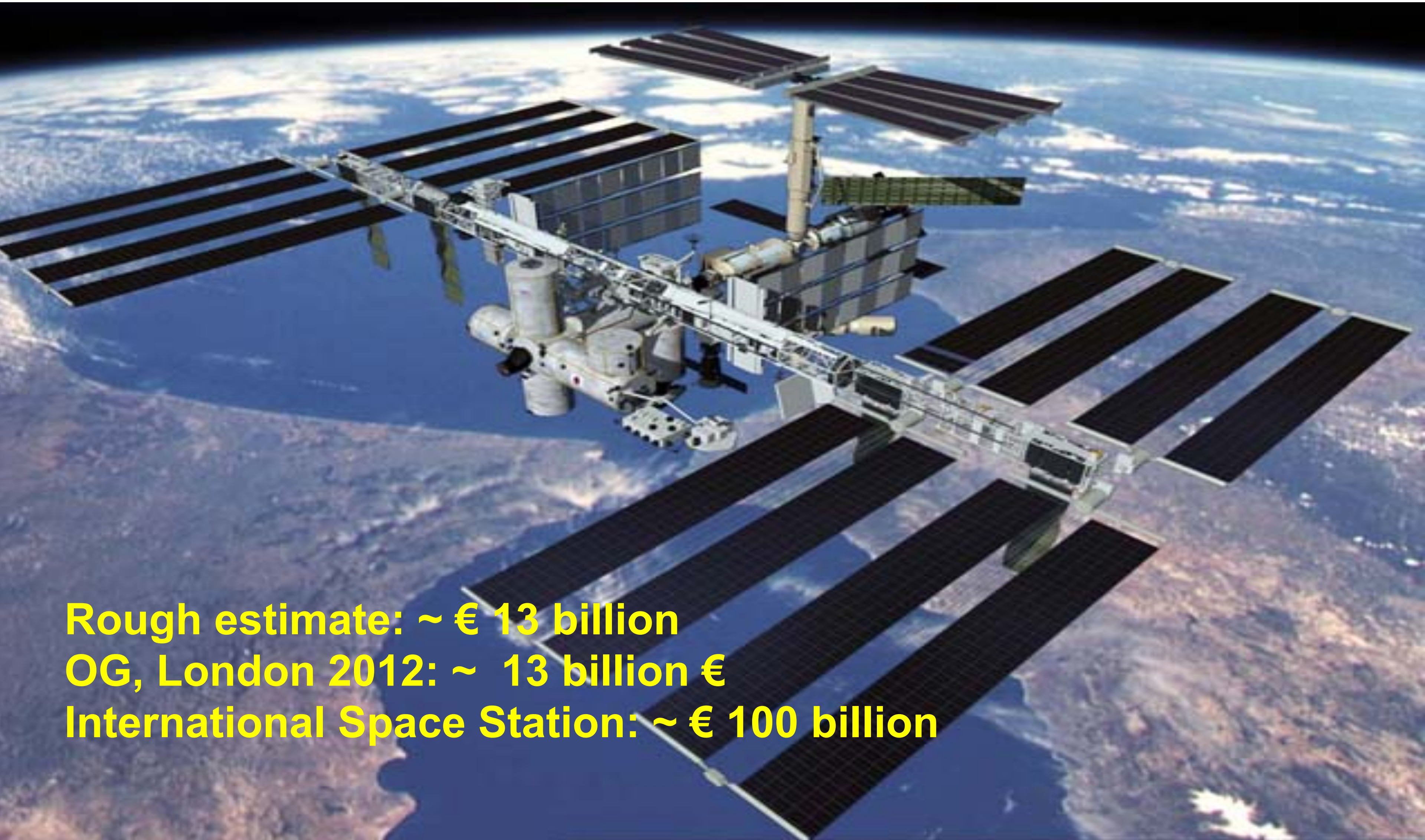


# THE ITER STAFF

In total 525 staff members (summer 2014) + 350 contractors
Currently ~ 1,200 people work at ITER

# How much does ITER cost?

More than 80% contributions are in kind

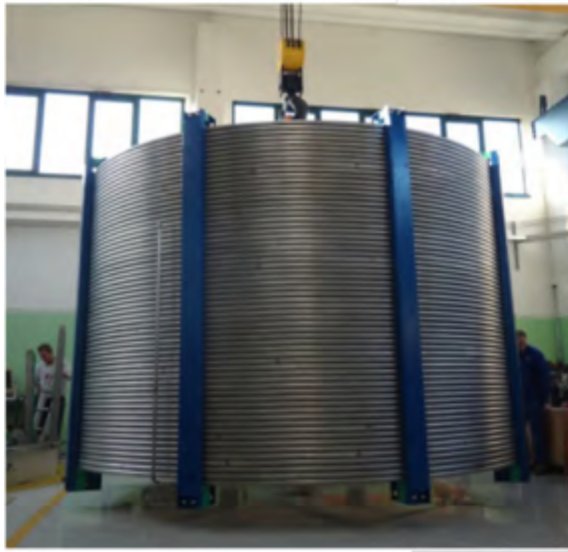


# Production is launched…

The regular radial plate prototype was completed by SIMIC in Camerana, Italy.
Photo courtesy of SIMIC

# Toroidal field (TF) conductor fabrication by 6 domestic agency contributors

Korea



Russia

China

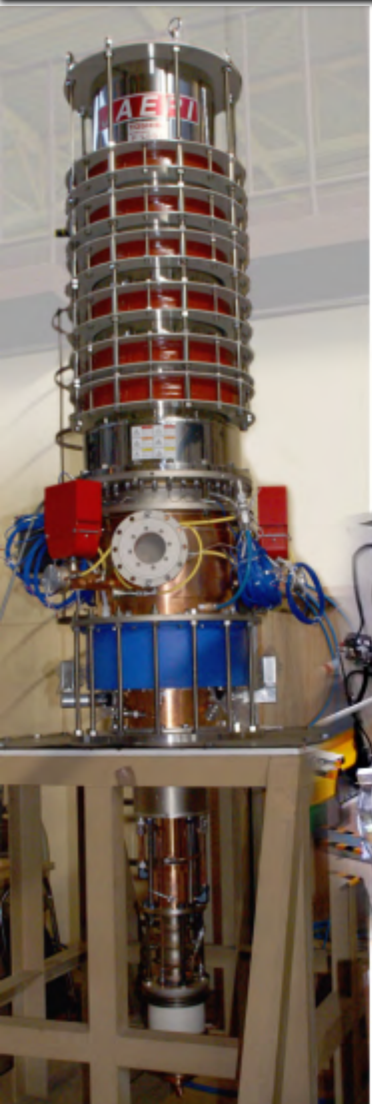
Japan

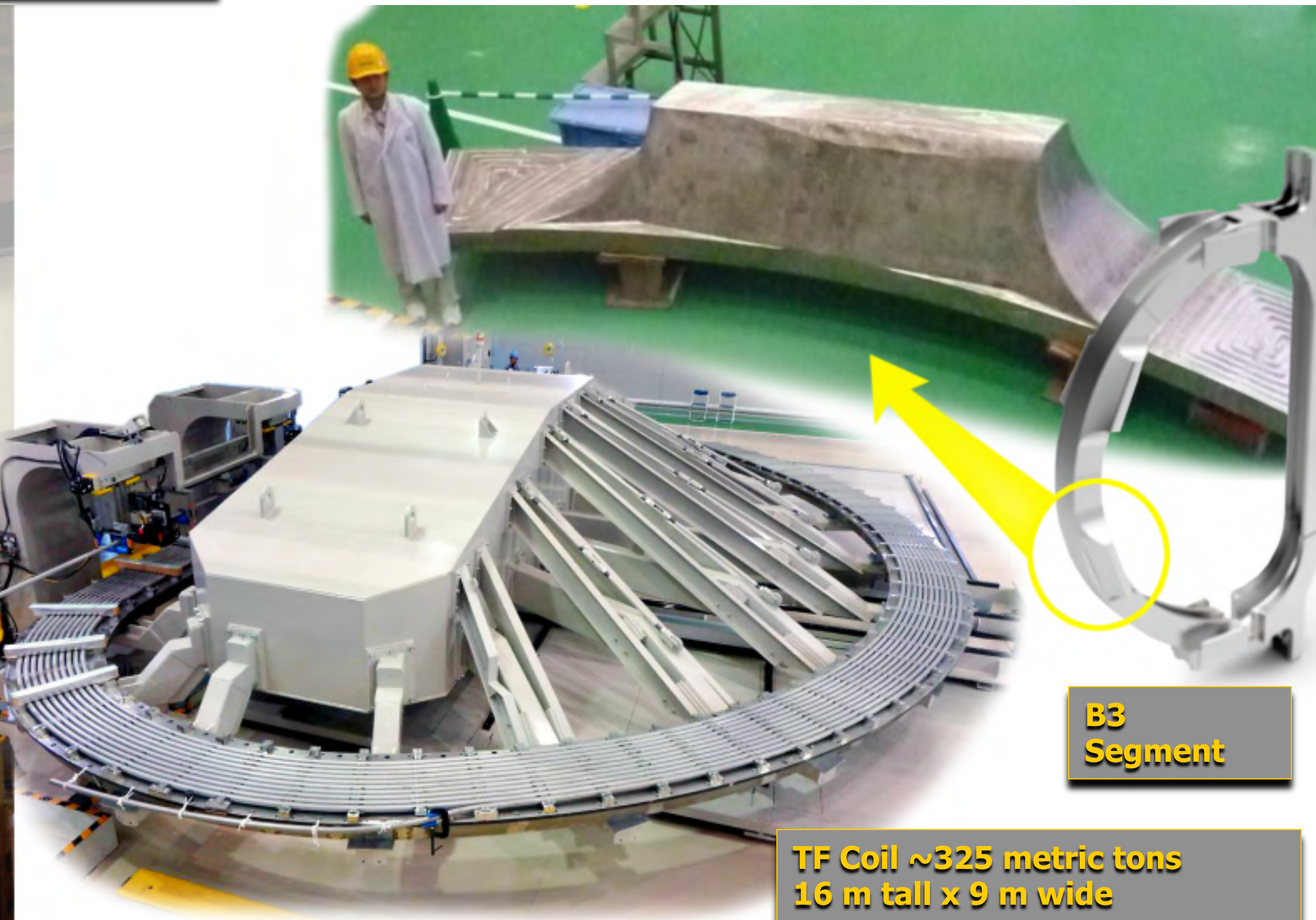
EU

US

# Japan – 170 GHz gyrotron and toroidal coil primary structure segment

**1MW gyrotron for plasma heating**



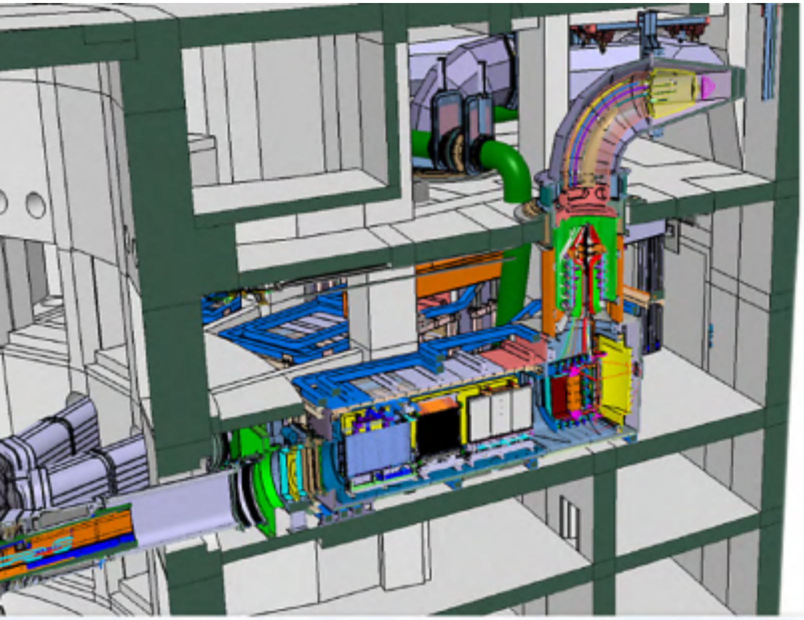


**B3 Segment**

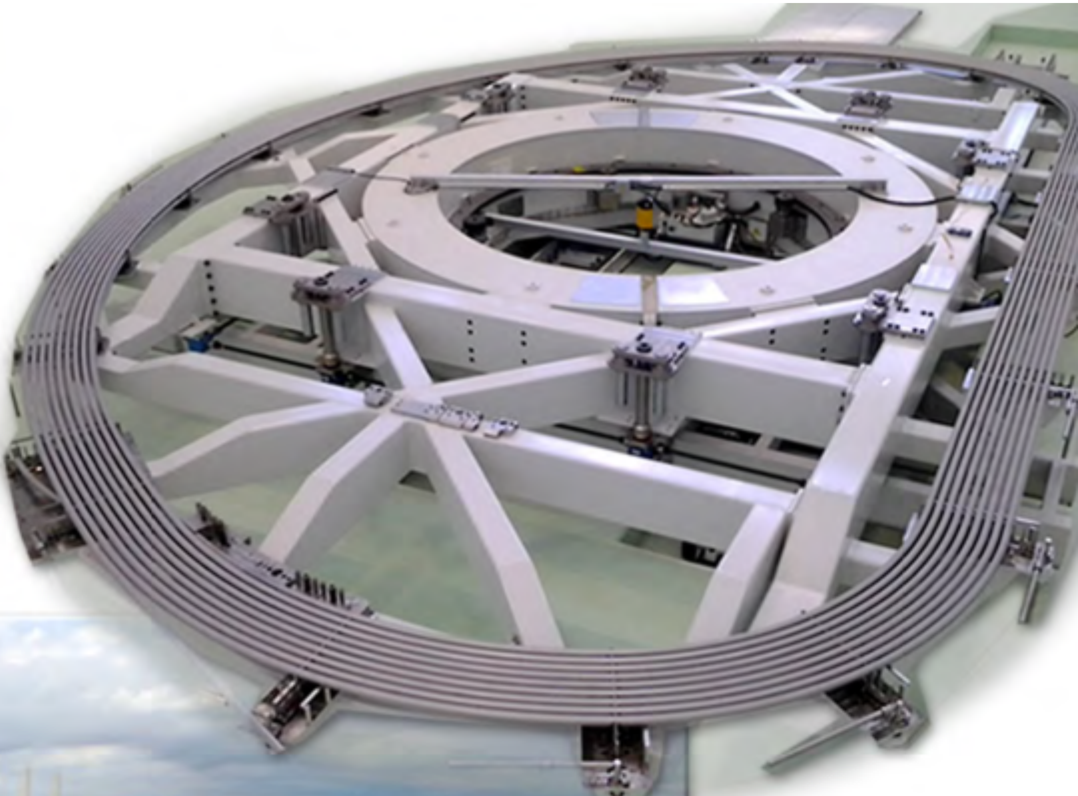
**TF Coil ~325 metric tons**
**16 m tall x 9 m wide**

# EU – Neutral beam injection test facilities and toroidal coil production

**ITER neutral beam injector**

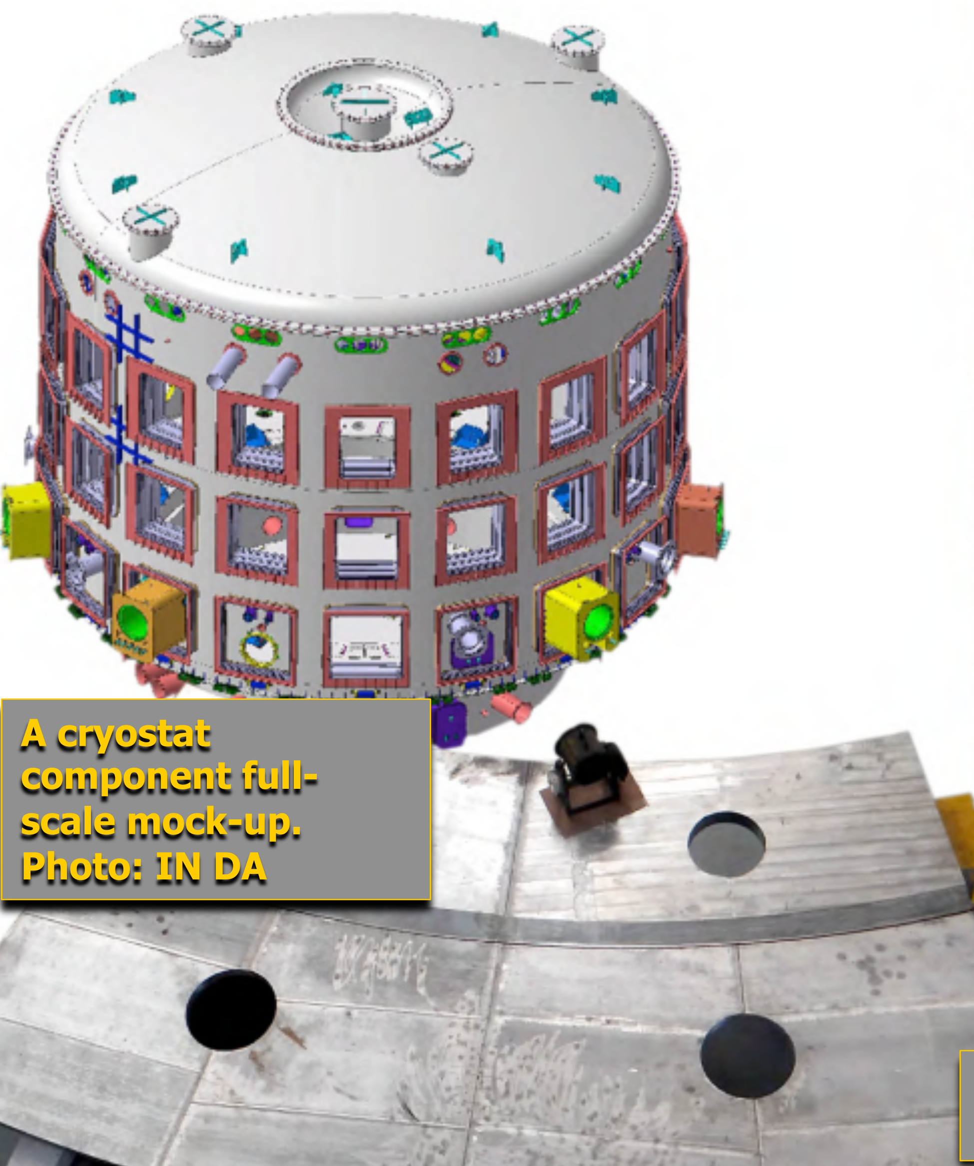


**First double pancake prototype for toroidal field conductor**



**Full scale NBI tests will be conducted at Padua site now under construction**

# India – Cryostat and diagnostic neutral beam

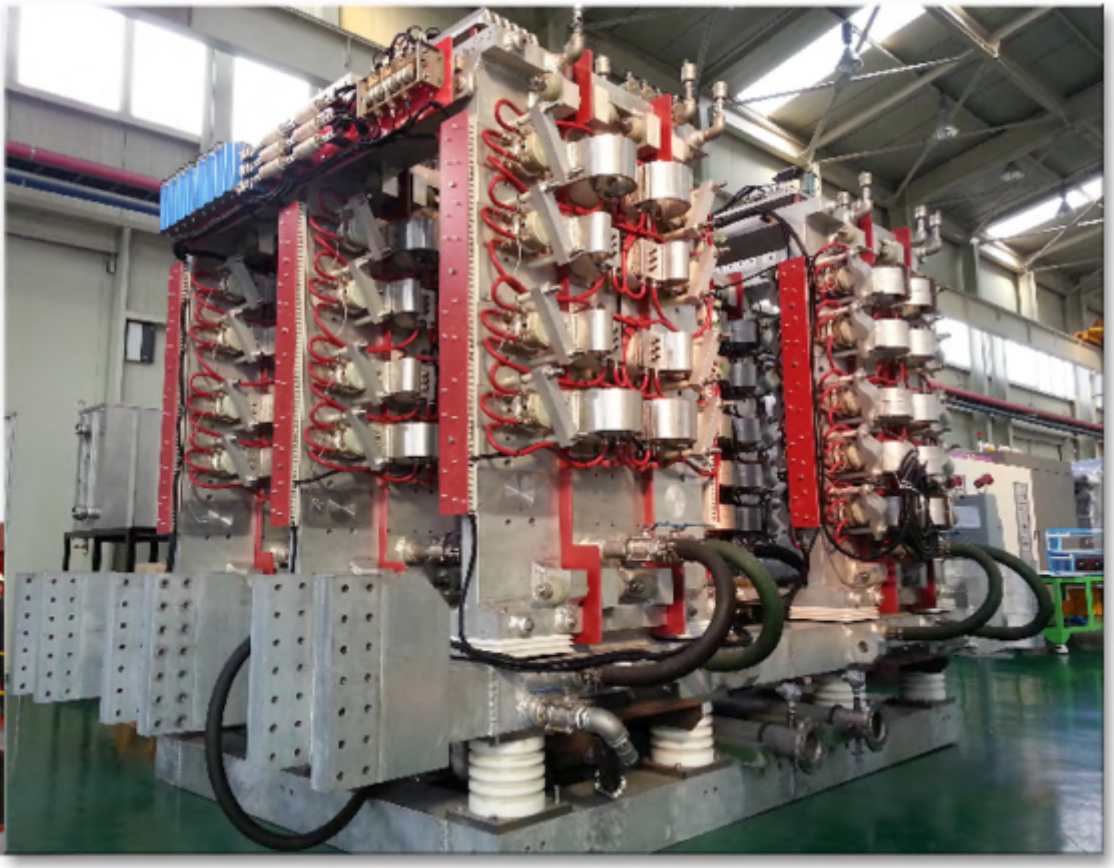


**A cryostat component full-scale mock-up. Photo: IN DA**

**High voltage power supply for Diagnostic Neutral Beam. Photo: IN DA**

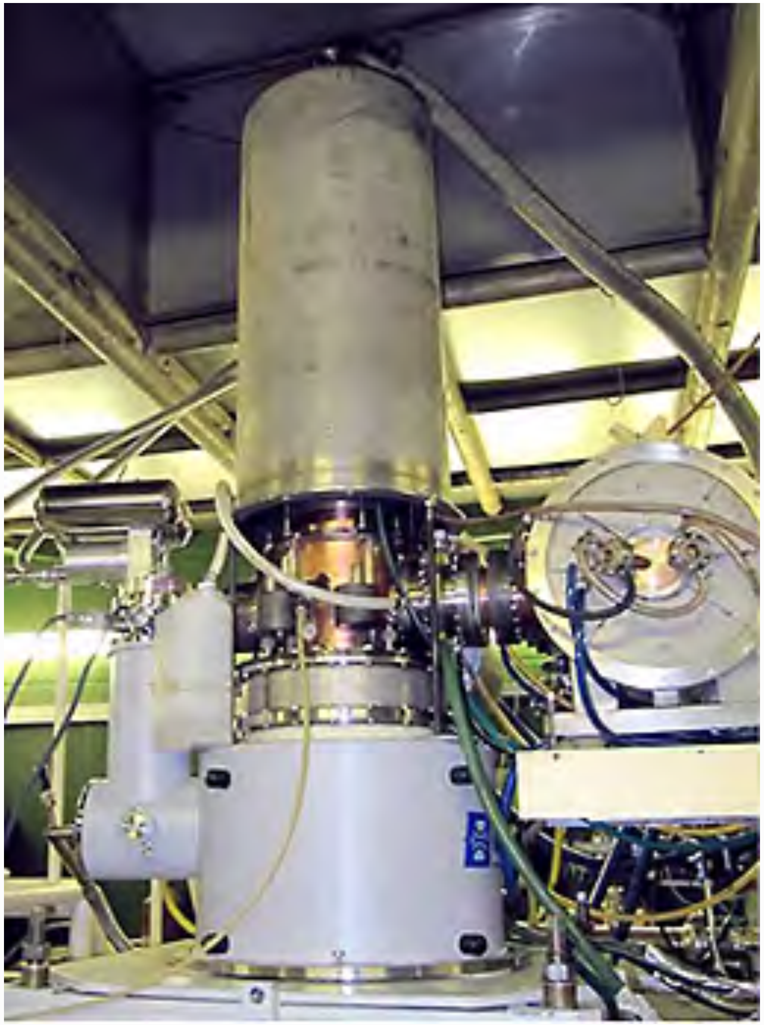
# Korea – Vacuum vessel segments and power convertor systems

Manufacturing of an ITER vacuum vessel segment in Korea



Prototype of the AC/DC power converter for the ITER vertical stabilization coils

# Russia – 170 GHz gyrotron and poloidal field magnet conductor development



1MW gyrotron for plasma heating and current drive plus mode suppression

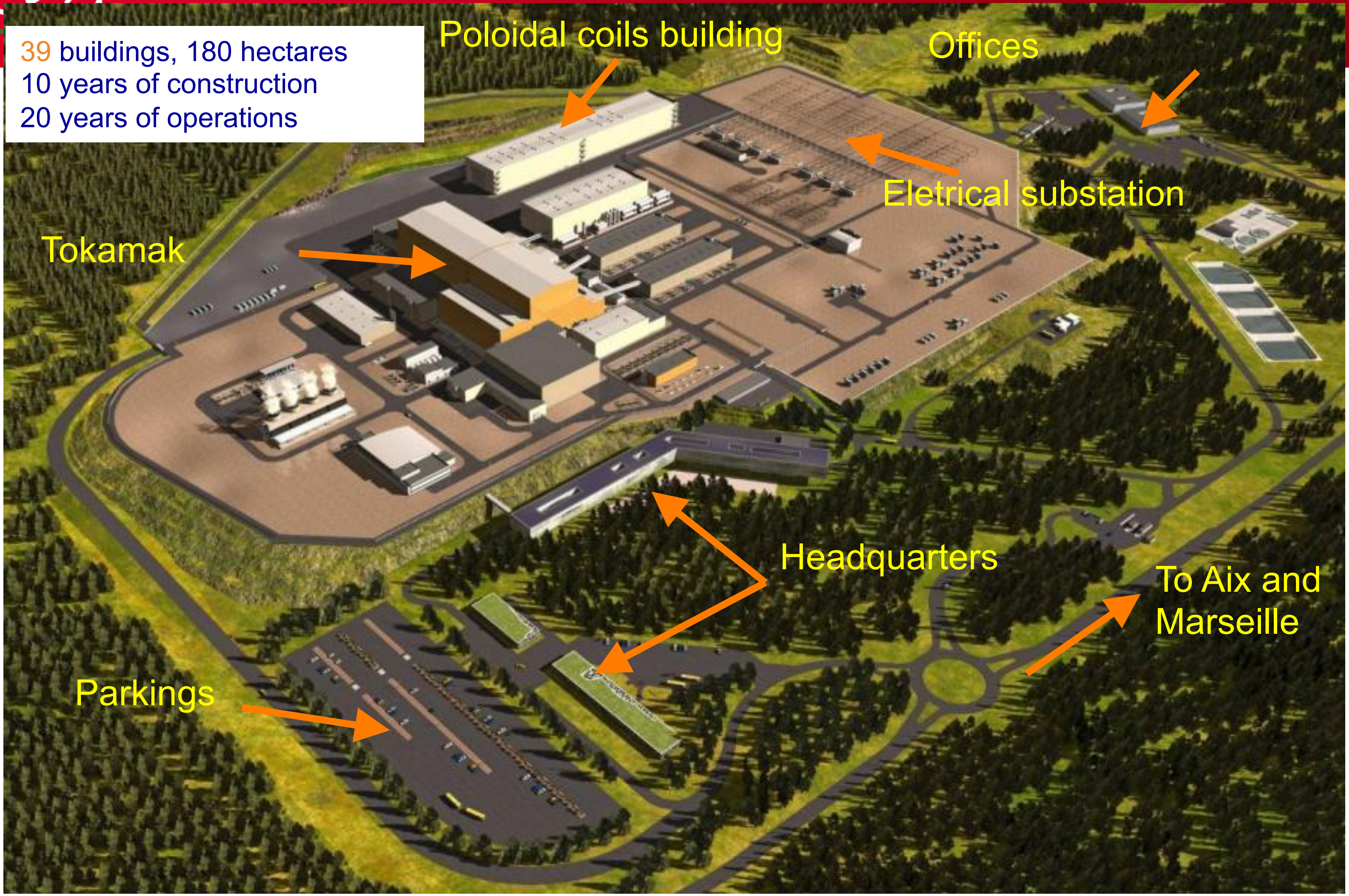
A copper dummy conductor fabricated for the PF1coil

# China – Poloidal field conductor


Poloidal field dummy conductor was loaded on a ship bound for France in April 2013. Photo: CN DA

Poloidal field dummy conductor was delivered from China to the ITER site in June 2013. Photo: ITER Organization

# The ITER site after construction



# Progress on the ITER worksite

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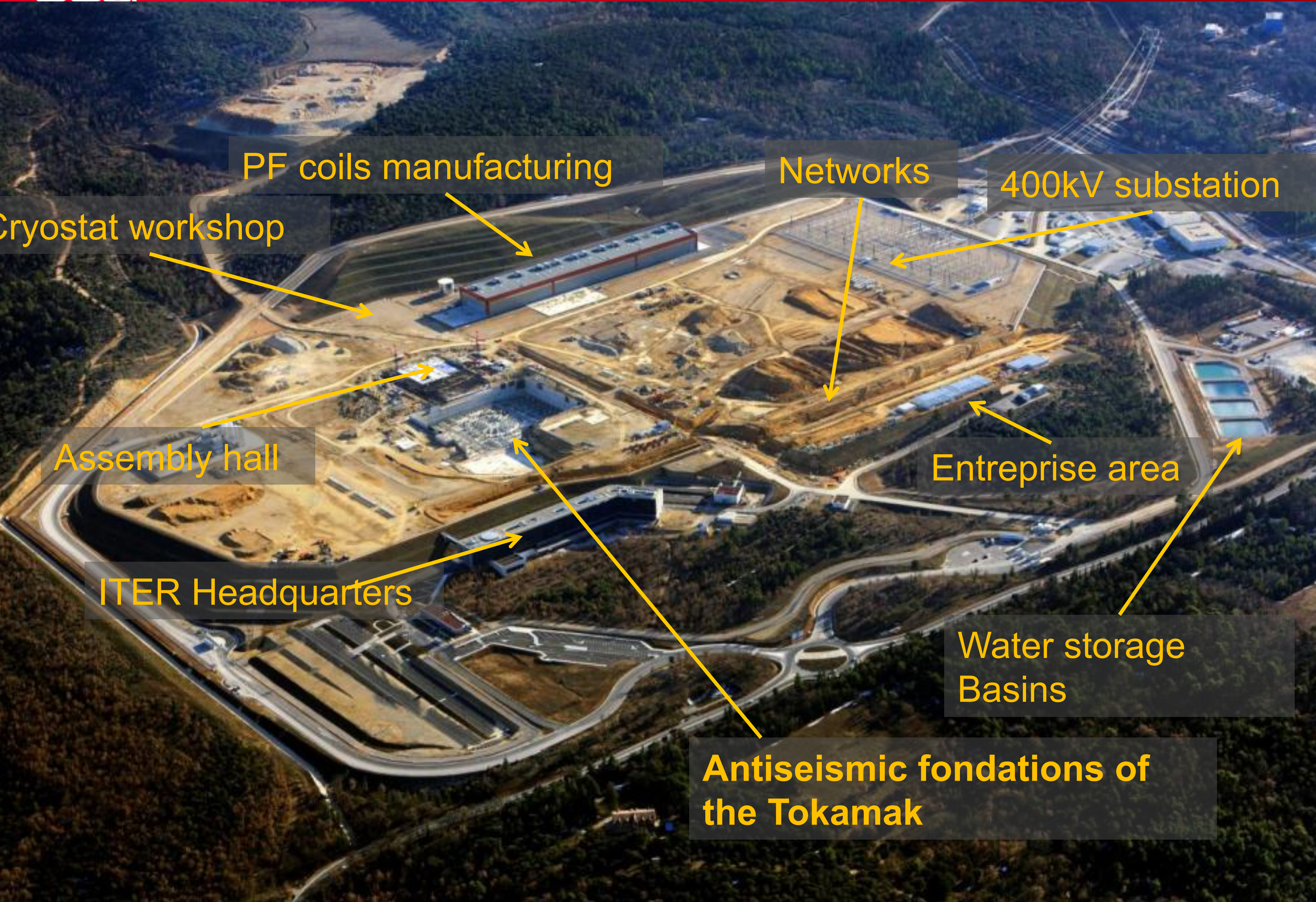
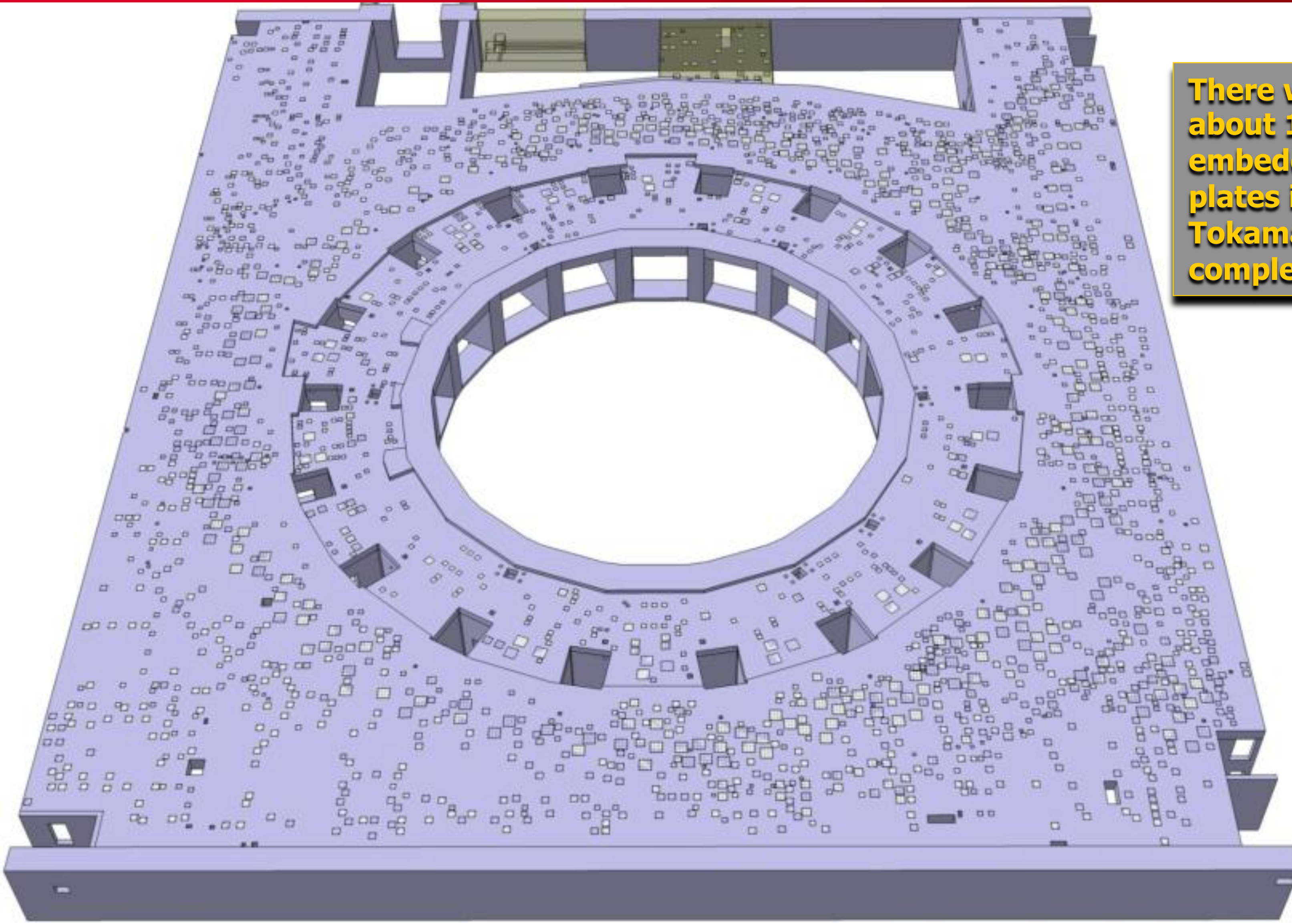


# B2 ceiling Embedded plates layout



**There will be about 100,000 embedded plates in the Tokamak complex**

# The entreprise area

# Progress on the ITER worksite

# Progress on the ITER worksite

**Resting on the 493 columns of the Tokamak Pit seismic system, a second basemat (1.5-m. thick) has been completed 2 weeks ago and will support the 360 000-ton Tokamak Complex buildings.**

# Progress on the ITER worksite

Inside the Poloidal Coils building: this large spreader beam (200 tons) will handle heavy loads during the poloidal coils winding and assembly process.

# Projet schedule

d by the ITER Council in July 2010

| | |
|---|---|
| Assembly: | 2014-2018 |
| First plasma: | Nov 2020 |
| D-T operations: | 2027 |

DE LA RECHERCHE À L'INDUSTRIE

# Transports

- Heaviest reactor components: > 750 tonnes
- Special itinerary 104 km from harbor to St Paul-lez-Durance
- Test convoy on 16-20 September 2013
- First component to be delivered in June 2014
- ~ 230 Heavy and Exceptional Loads 2014-2018, very slow (average speed 5 km/h)


# Delivering the reactor components


**The largest and heaviest loads will arrive at the French harbour of Fos-sur-Mer. Exceptional convoys will then transport the loads along the dedicated 104 kilometre-long ITER Itinerary to Cadarache.**
**The ITER Itinerary was financed by French government and CG 13.**

# Succesful test convoy (16-20 September 2013)

**The test convoy simulated the transport of a component of 600 tons 9 meters high. Leaving from Fos-sur-mer (Mediterranean Sea) on the 16th, it arrived on the ITER site on 20 September at 4:45 am.**

# The test convoy


Caracteristics of the trailer + mock-up :
M = 775 t, H = 10,60 m, L = 33 m, l = 9 m

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# CONCLUSIONS AND PERSPECTIVES

Advanced scenarios are quite attractive but challenging!

Major & recent experimental progress to tackle real time control issues for steady-state tokamak operation have been made worldwide

But more is needed and integration of the different attempts should be made:

**Challenging issues for future research direction**

- **Integrated control of edge and core physics?**
  - **Develop generic methods, RT diagnostics, control loops, tokamak control system, etc**
- **Demonstration of the controllability of bootstrap-dominated regime with dominant $\alpha$-heating ?**
  - **experiments & modelling**

# THANK YOU FOR YOUR ATTENTION

Commissariat à l'énergie atomique et aux énergies alternatives
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DSM
WEST

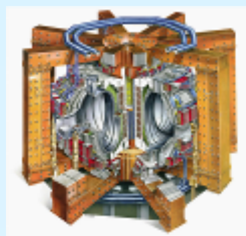
# CHALLENGES FOR CONTINUOUS OPERATION

irfm
cea
cadarache

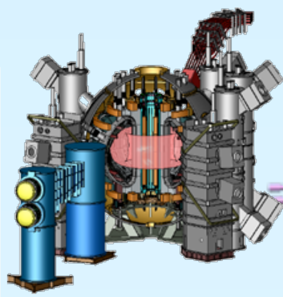
## A scientific and technical challenge

Tore Supra

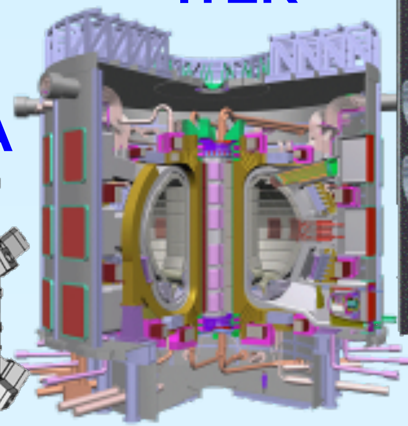
JET



JT60-SA



ITER

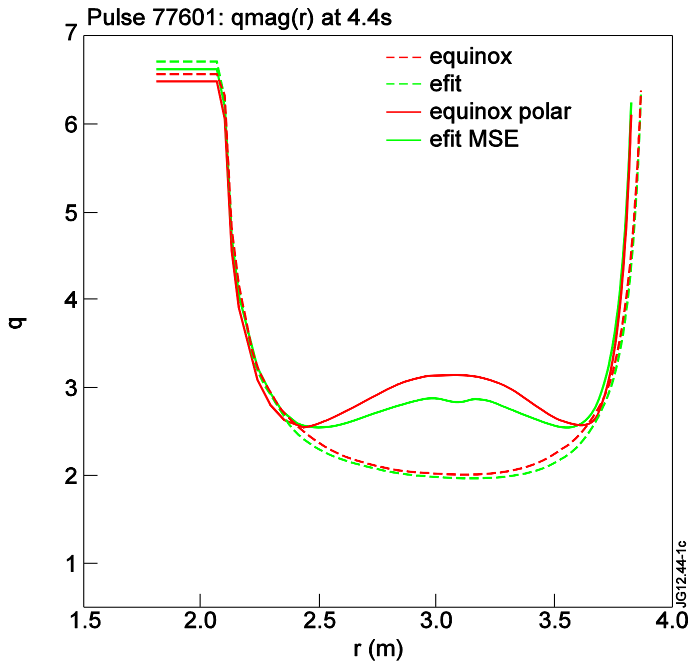


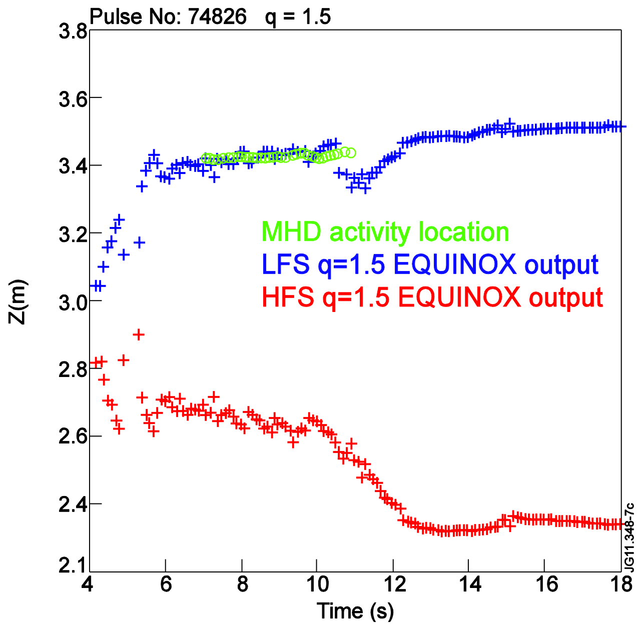
DEMO

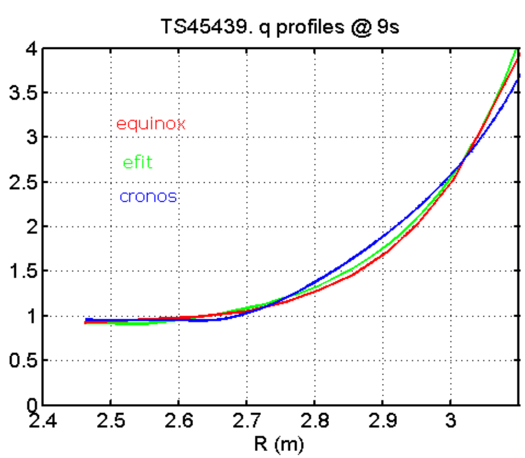
| | Tore Supra | JET | JT60-SA | ITER | DEMO |
|---|---|---|---|---|---|
| | DD | Q ~ 1 | DD | Q ~ 10 | Q ~ 30 |
| duration | ~400s | 2s | ~100s | 400-3600s | Continuous |
| self-heating | 0% | 10% | 0% | 70% | 80 to 90% |
| bootstrap | 20% | 20% | >60% | 10-50% | 60-80% |

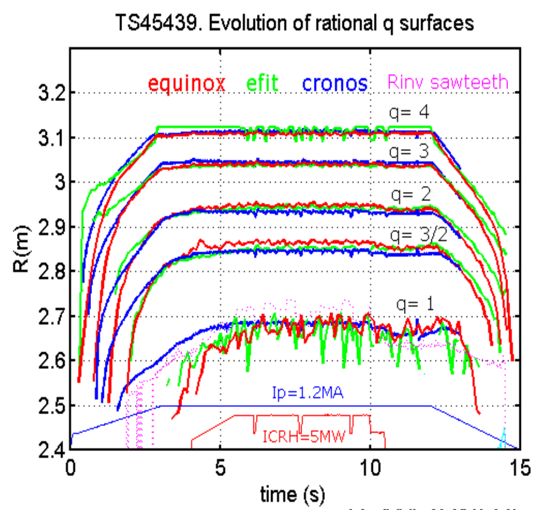
**Existence and control of a <u>self-organised</u> plasma state for continuous tokamak operation ? (non inductive current self-generated by plasma pressure gradients through the bootstrap effect and most of the heating sources are self-generated by the $\alpha$ particles produced by the fusion reaction)**

# PROFILE RECONSTRUCTION USING GRAD SHAFRANOV SOLVER



# BOOTSTRAP CURRENT