2nd ITER International Summer School
In conjunction with
the 47th Summer School of JSPF for Young Plasma Scientists
Kyushu Univ., 25 July, 2008

Summary

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Help of lecturers, K. Ikeda, F. Wagner, X. Garbet, S. Ishizaka, C. S. Chang, P. Diamond, T. Tsunematsu, S. Matsuda, T. Tanabe, C. Skinner, A. Fujisawa, D. Campbell, H. Yamada, A. Fukuyama, is highly acknowledged and I wish to thank M. Yagi and S. Inagaki for support in preparing it.

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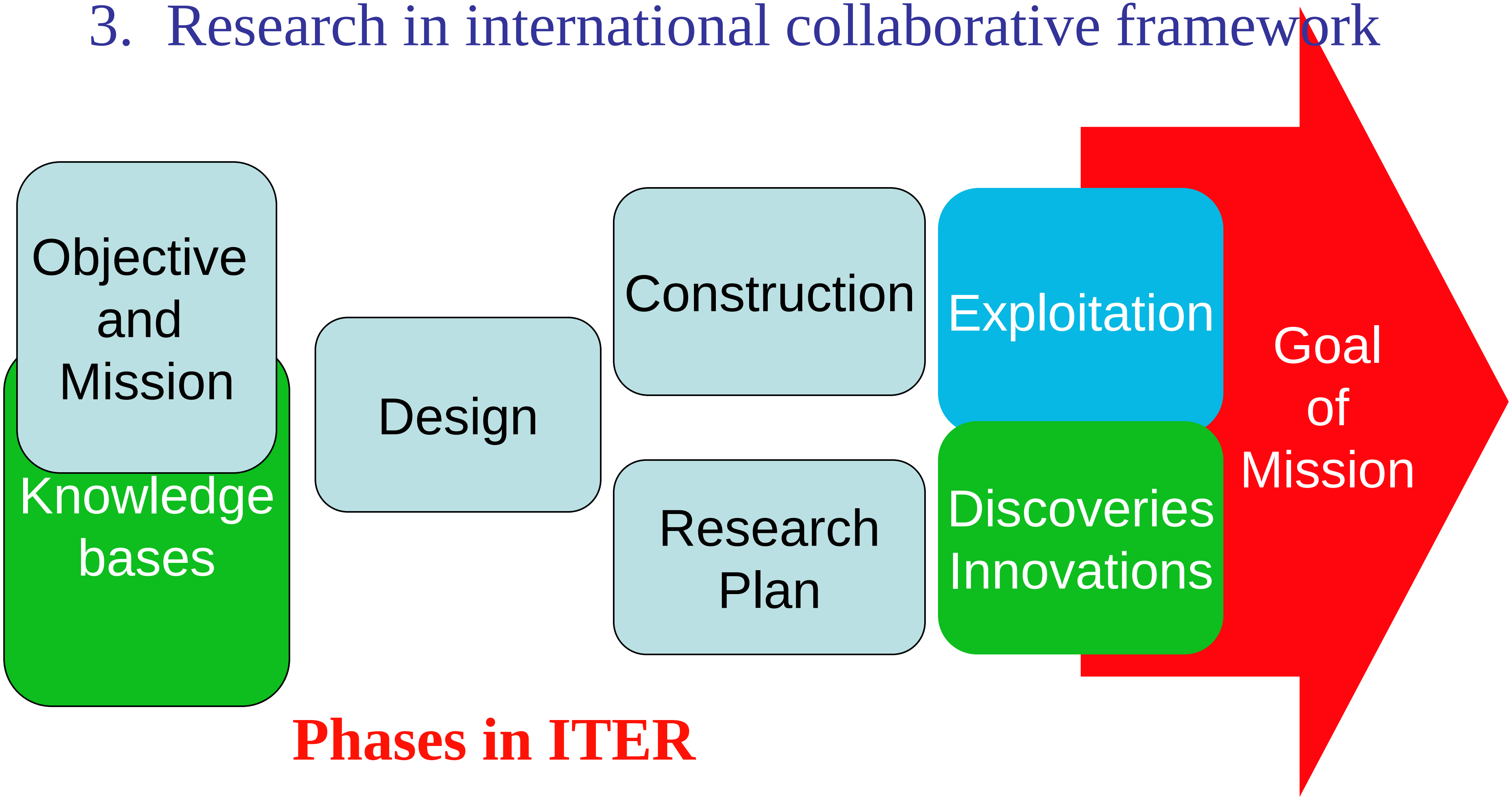
New Era of ITER Research

1. Future evolution of the research
2. Fusion research under the nuclear reaction
3. Research in international collaborative framework



Phases in ITER

Structure of lectures

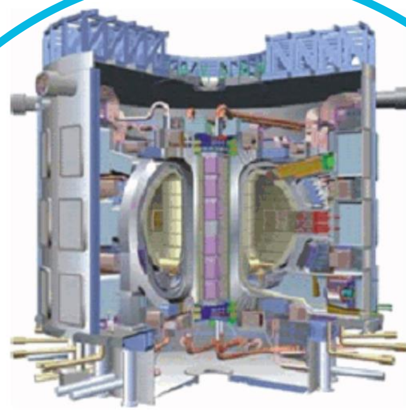
- Challenge for mission
- Fusion study in the nuclear systems
- International culture

missions

Booth

Student-Organized Session

working



at/for ITER

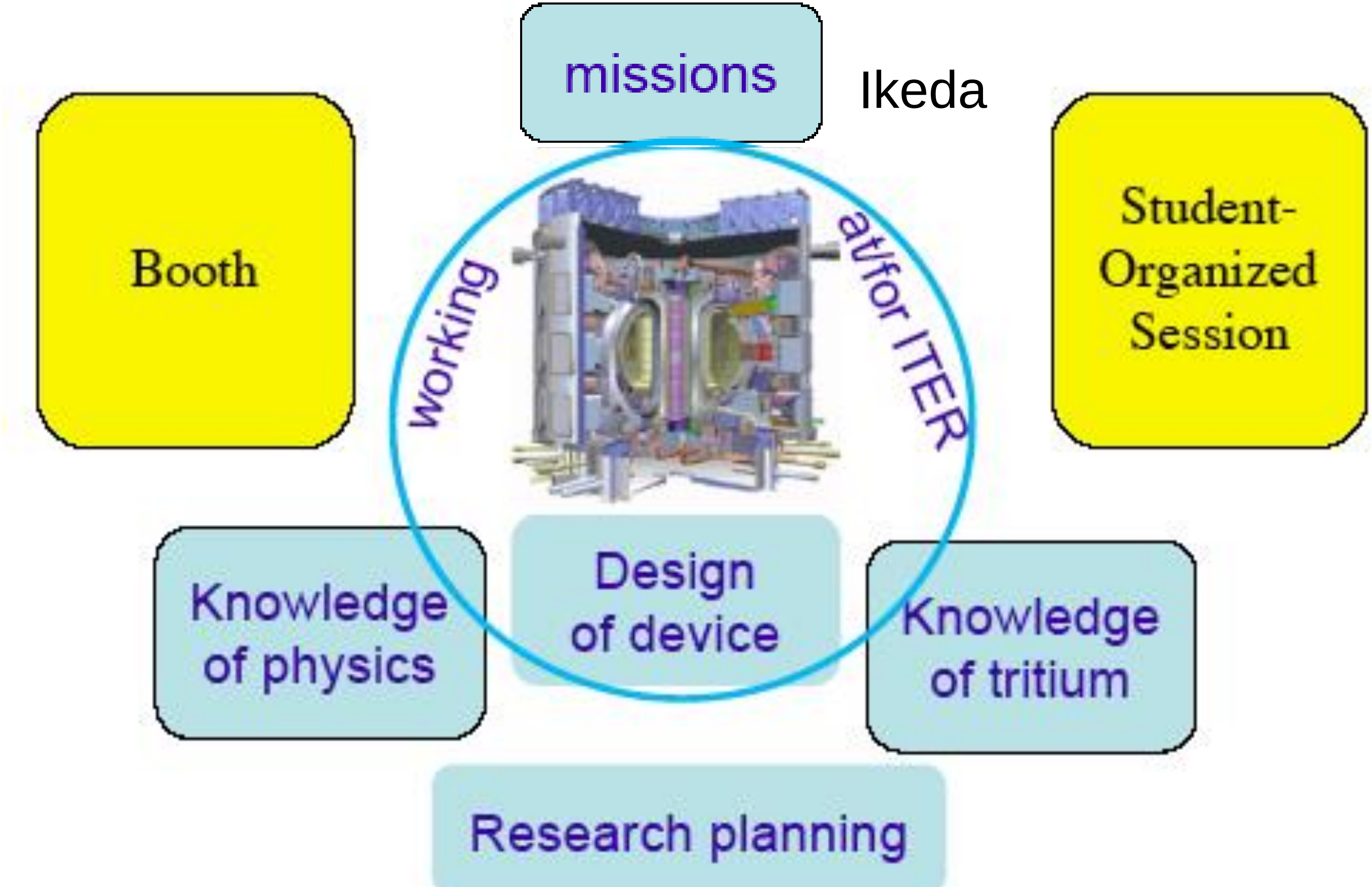
Design of device

Knowledge of physics

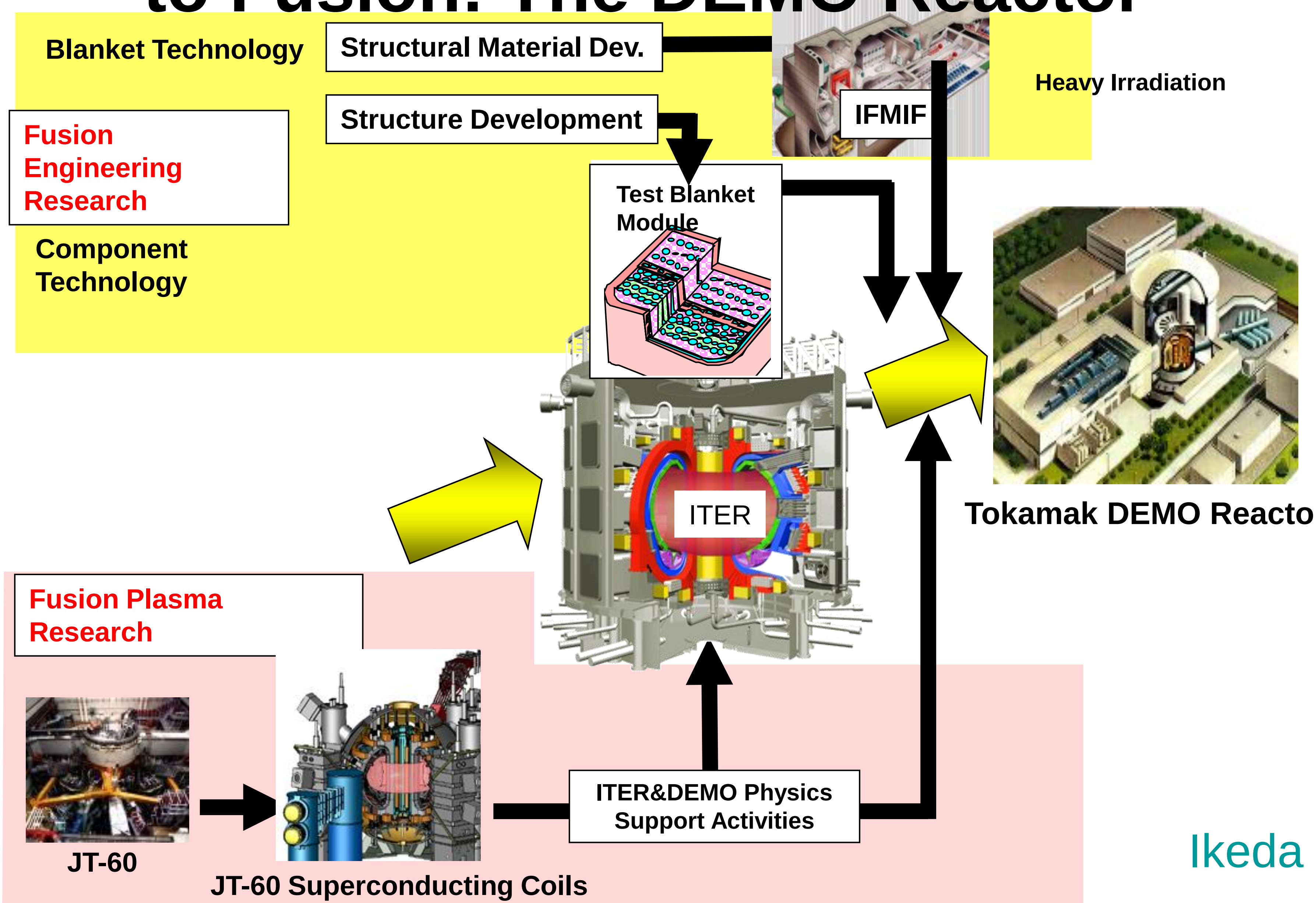
Knowledge of tritium

Research planning

Ikeda


missions
Booth
working
at/for ITER
Student-Organized Session
Knowledge of physics
Design of device
Knowledge of tritium
Research planning

The Present and the Future Road Map to Fusion: The DEMO Reactor



Technical Objectives of ITER

Plasma Performance:

- $Q \geq 10$ with a burn duration between 300 and 500 s,
- aim at demonstrating steady state operation with Q>5,
- possibility of controlled ignition.

Engineering Performance and Testing:

- demonstrate availability and integration of essential fusion technologies,
- test components for a future reactor,
- test tritium breeding module concepts; with a 14MeV neutron average power load on the first wall > 0.5 MW/m^2 and fluence 0.3 > MWa/m^2,
- the option for later installation of a tritium breeding blanket

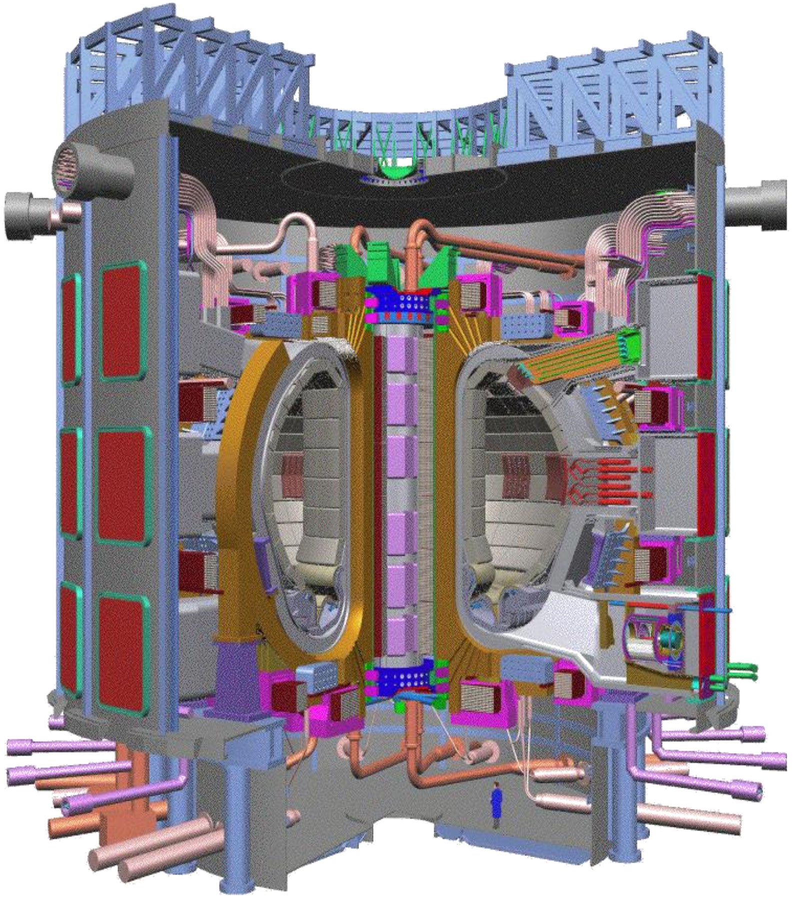
Tsunematsu

ITER Technical Objectives and Implementation

- **Engineering Design of ITER:**

Main Parameters of ITER

Total fusion power	**500 MW**
Additional heating power	**50 MW**
Q - fusion power/ additional heating power	**≥ 10**
Average 14MeV neutron wall loading	**≥ 0.5 MW/m^2**
Plasma inductive burn time	**300-500 s ***
Plasma major radius (R)	**6.2 m**
Plasma minor radius (a)	**2.0 m**
Plasma current (I_p)	**15 MA**
Toroidal field at 6.2 m radius (B_T)	**5.3 T**



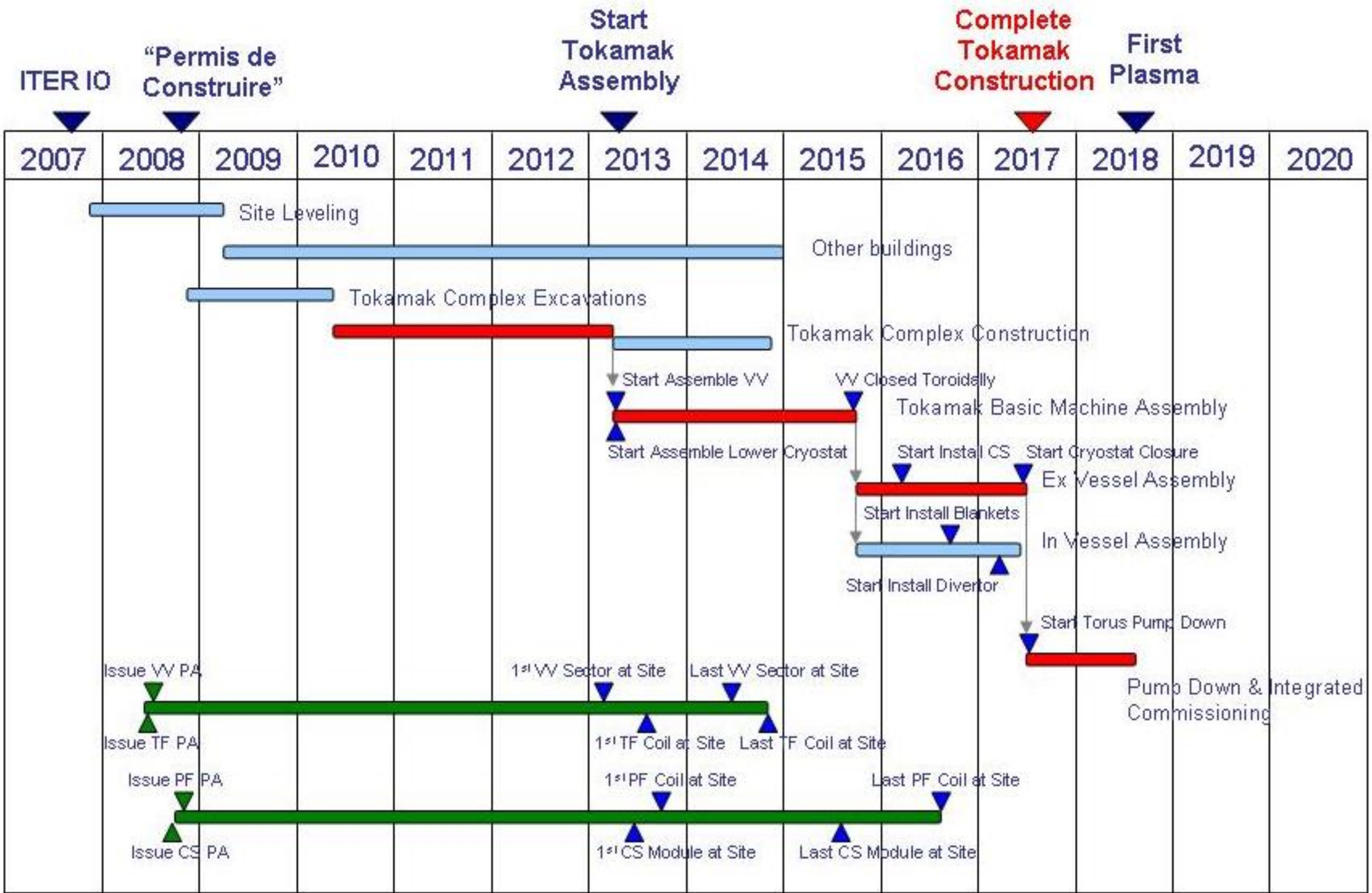
*** under nominal operating conditions**

The Costs: 5 billion € for ten years of construction and 5 billion € for 20 years of operation and decommissioning

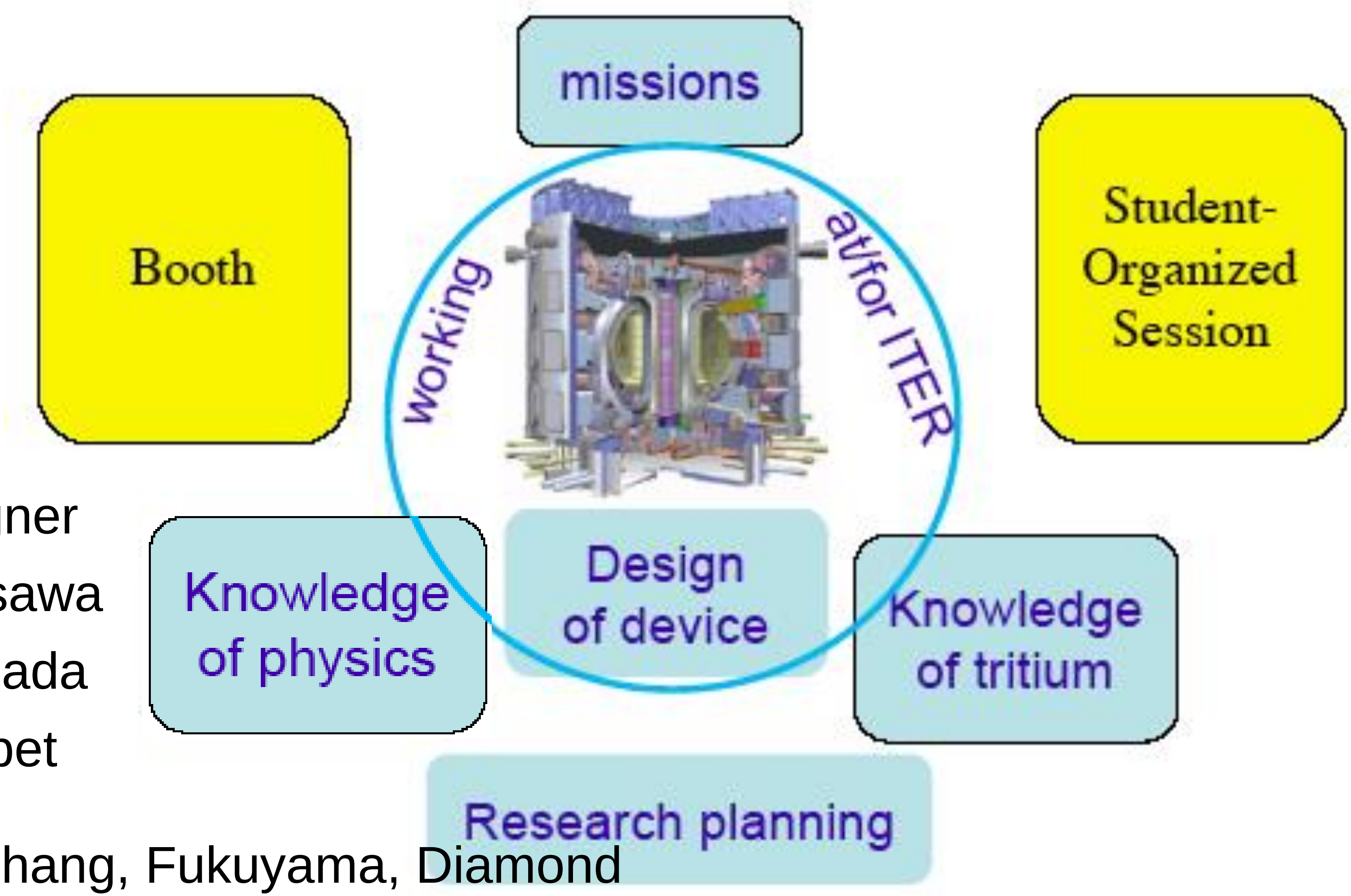
The execution: ~90% of the contributions are in kind

Ikeda

Resulting Reference IPS



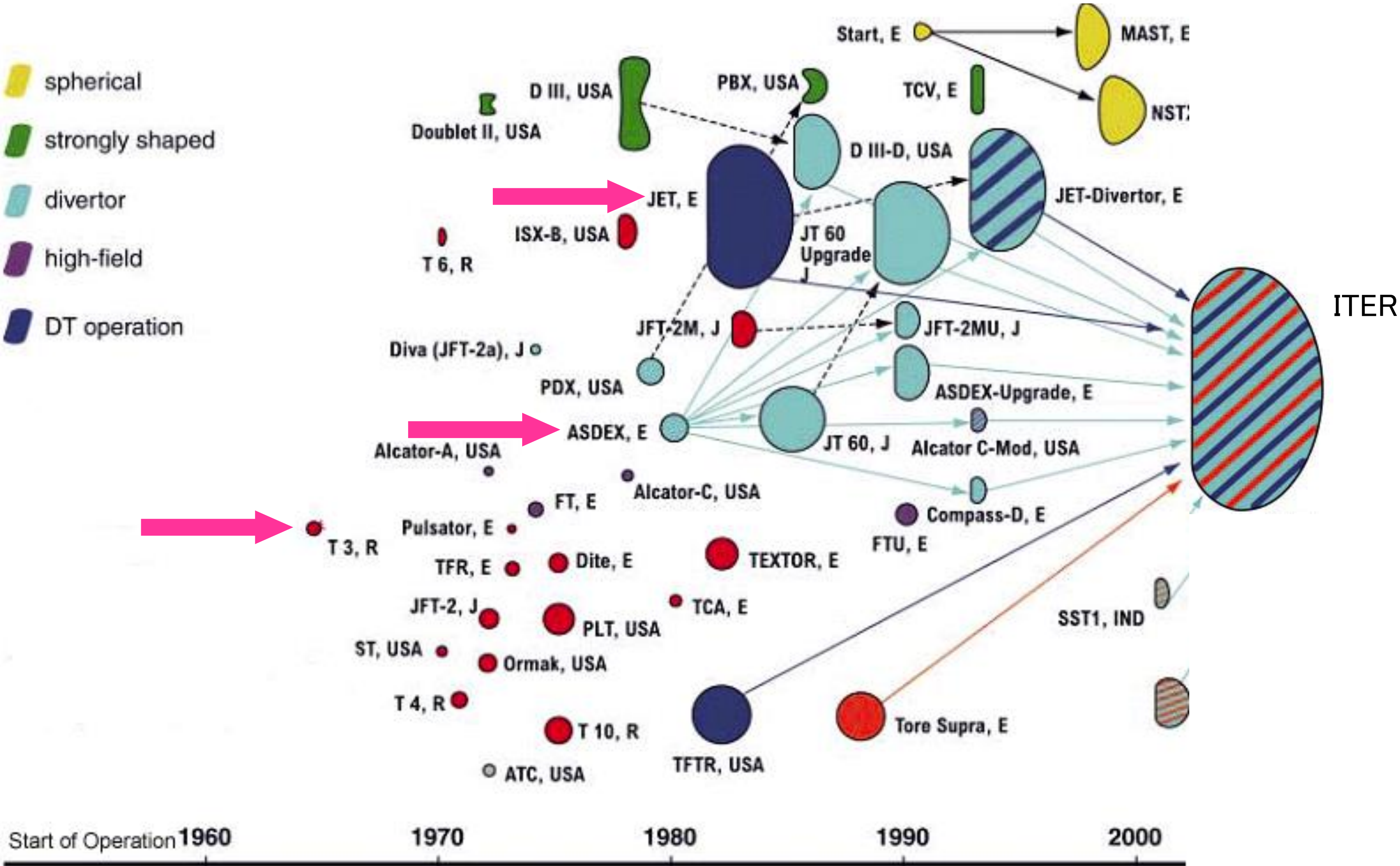
Ikeda



Wagner
Fujisawa
Yamada
Garbet

Chang, Fukuyama, Diamond

The pathfinders for ITER



Historical Result ofT-3 Tokamak

T_e ~ 100-2000eV, T_i ~ 300 eV,
n_e ~ 10^{12}-5×10^{13} cm^{-3}, τ_E ~ 10 ms.
From the presentation at the 3rd IAEA conference (1968, Novosibirsk)

This result was confirmed by the Thomson scattering measurement, which was performed by Culham group (1969).
Ref: M.J. Forrest, N.J. Peacock, D.C. Robinson, V.V. Sannikov, P.D. Wilcock;
"Mesurement of the Parameters in TOKAMAK T3-A by Thomson Scattering" CLM-R 107 (July, 1970)

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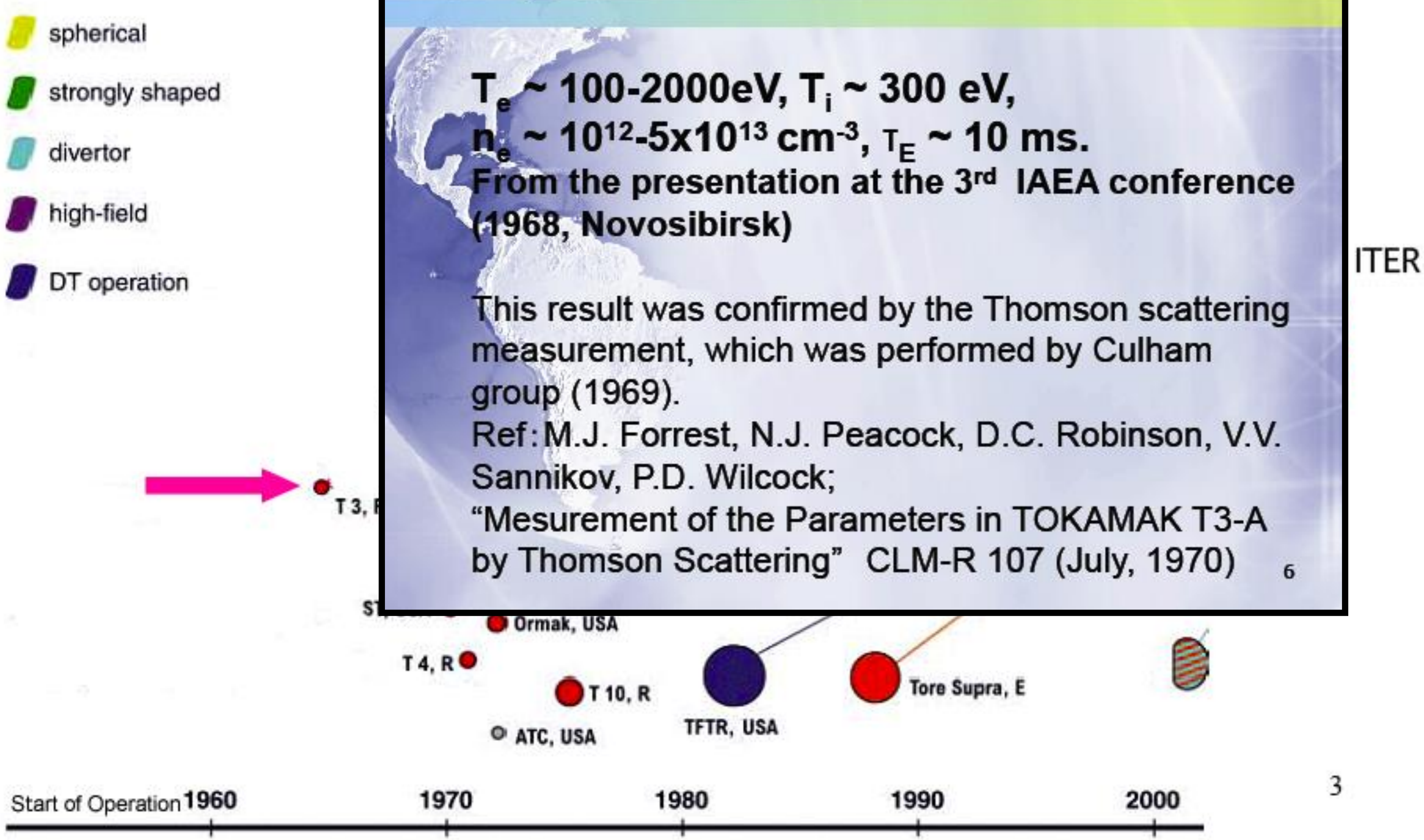
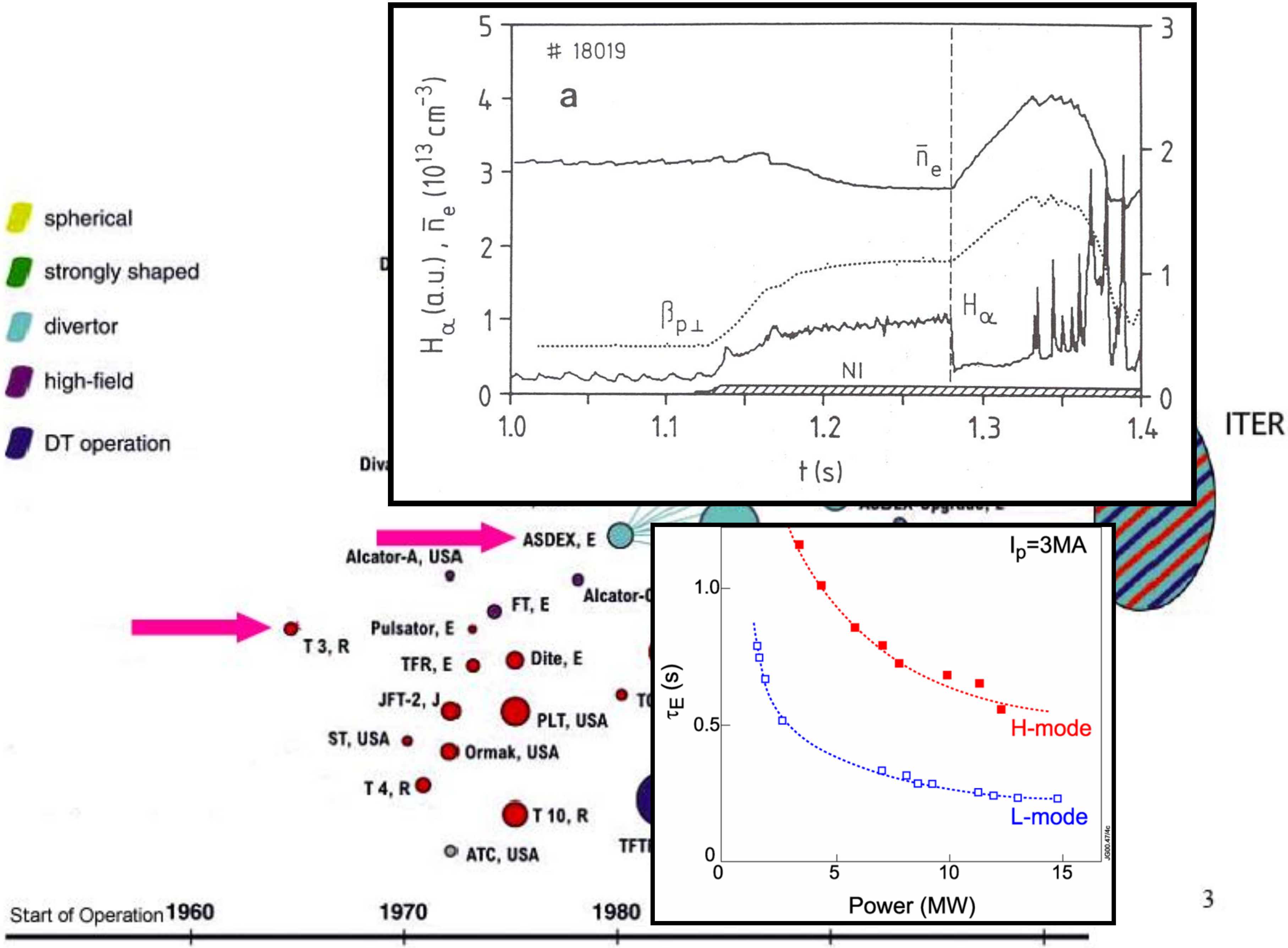
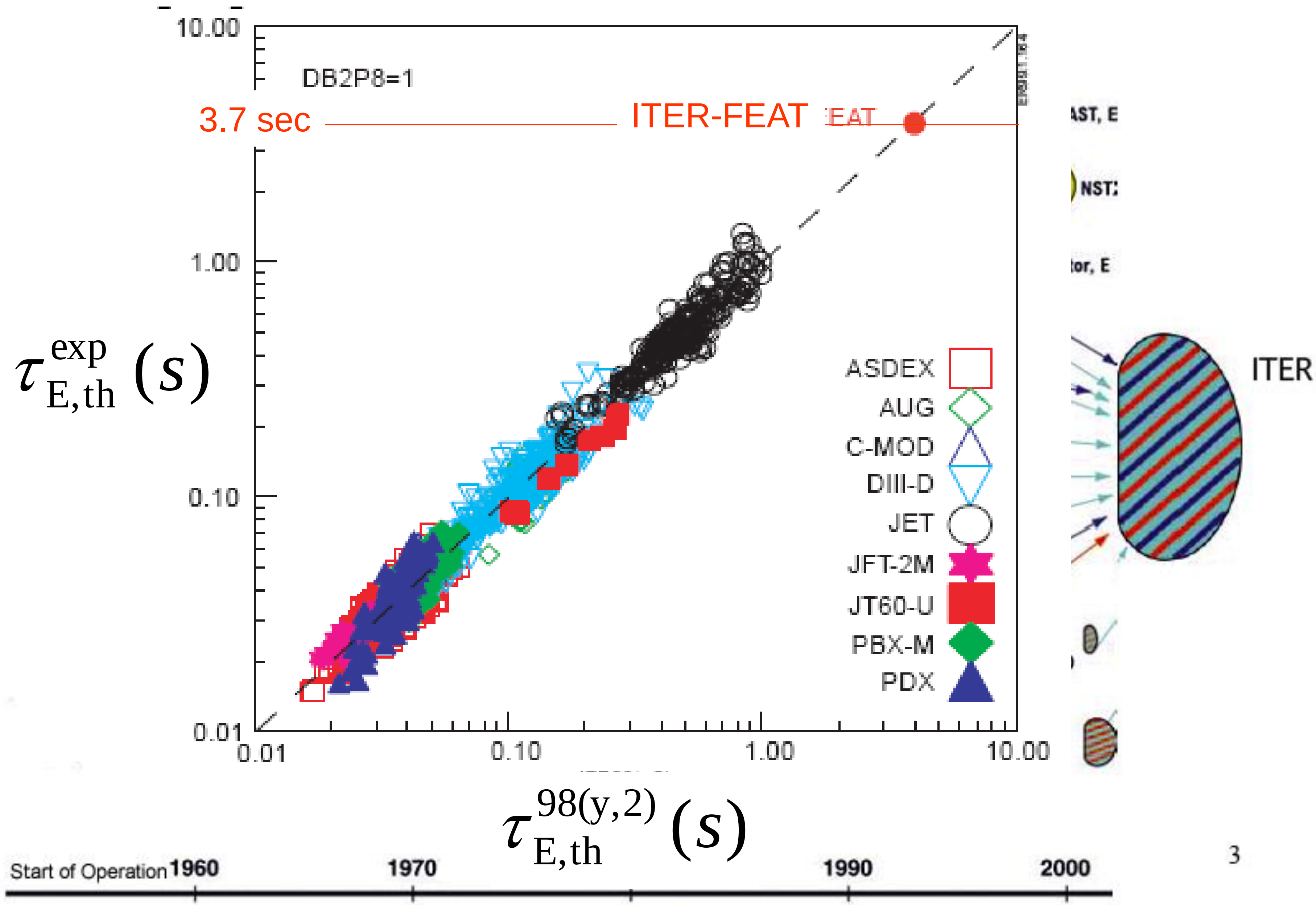
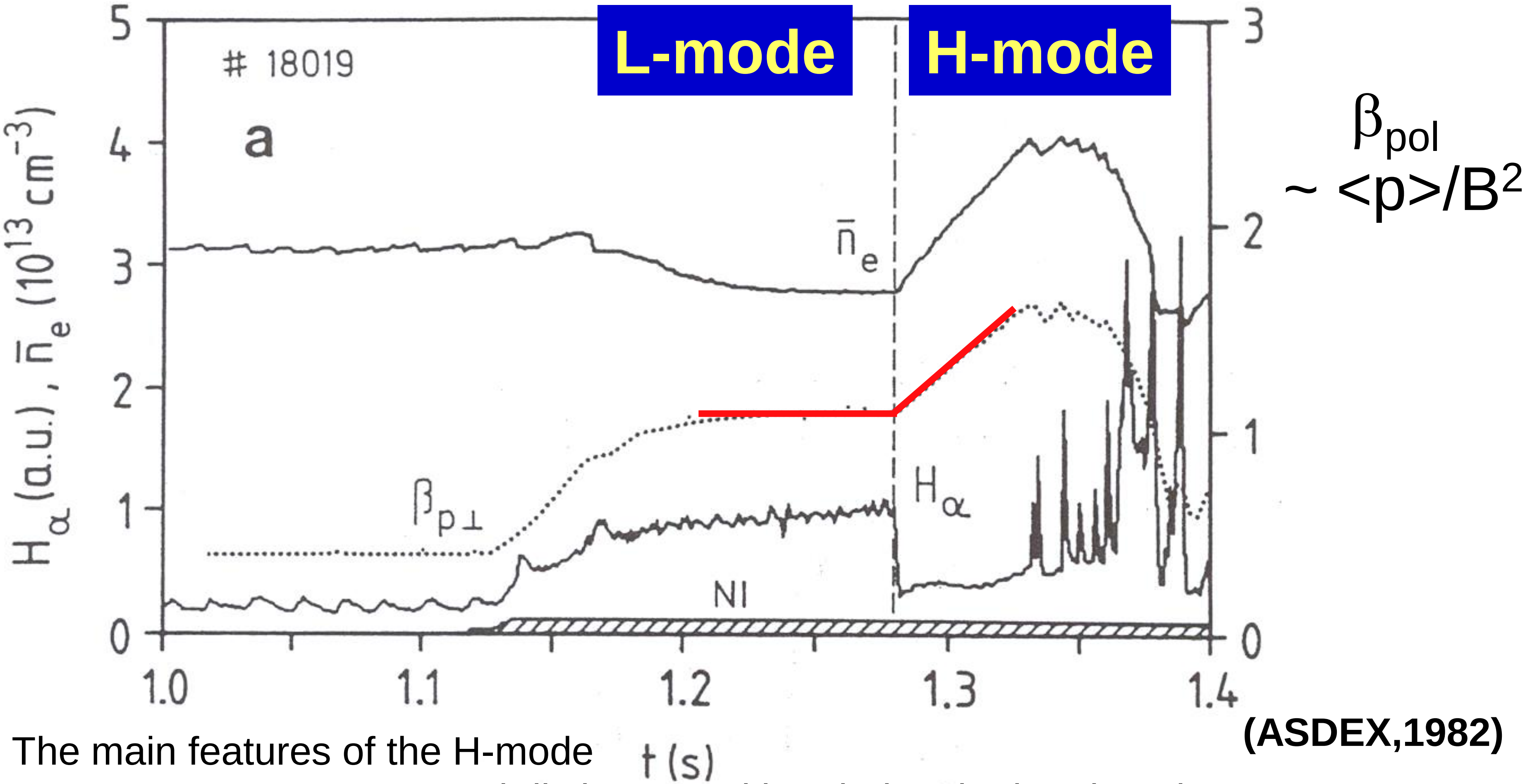



spherical
strongly shaped
divertor
high-field
DT operation
ITER
18019
a
H_α (a.u.), $\bar{n}_e$ (10^{13} cm^{-3})
$\bar{n}_e$
$\beta_{p\perp}$
H_α
NI
t (s)
ASDEX, E
Alcator-A, USA
FT, E
Pulsator, E
T 3, R
TFR, E
Dite, E
JFT-2, J
PLT, USA
ST, USA
Ormak, USA
T 4, R
T 10, R
ATC, USA
I_p=3MA
τ_E (s)
H-mode
L-mode
Power (MW)
Start of Operation 1960
1970
1980


DB2P8=1
3.7 sec
ITER-FEAT
ASDEX
AUG
C-MOD
DIII-D
JET
JFT-2M
JT60-U
PBX-M
PDX
10.00
1.00
0.10
0.01
ITER
$\tau_{\mathrm{E,th}}^{\mathrm{exp}}\ (s)$
$\tau_{\mathrm{E,th}}^{\mathrm{98(y,2)}}\ (s)$
Start of Operation
1960
1970
1990
2000

H-mode and edge transport barrier



(ASDEX,1982)

The main features of the H-mode

- a spontaneous and distinct transition during the heating phase
- both energy- and particle confinement time increase
- the tracer for the transition is the $H\alpha$-radiation
- new instabilities appear in the H-phase: ELMs, edge-localised modes

Characteristics of the H-mode

Confinement improved to the L-mode by factor 2 (H_{89} =2)

Edge pedestal

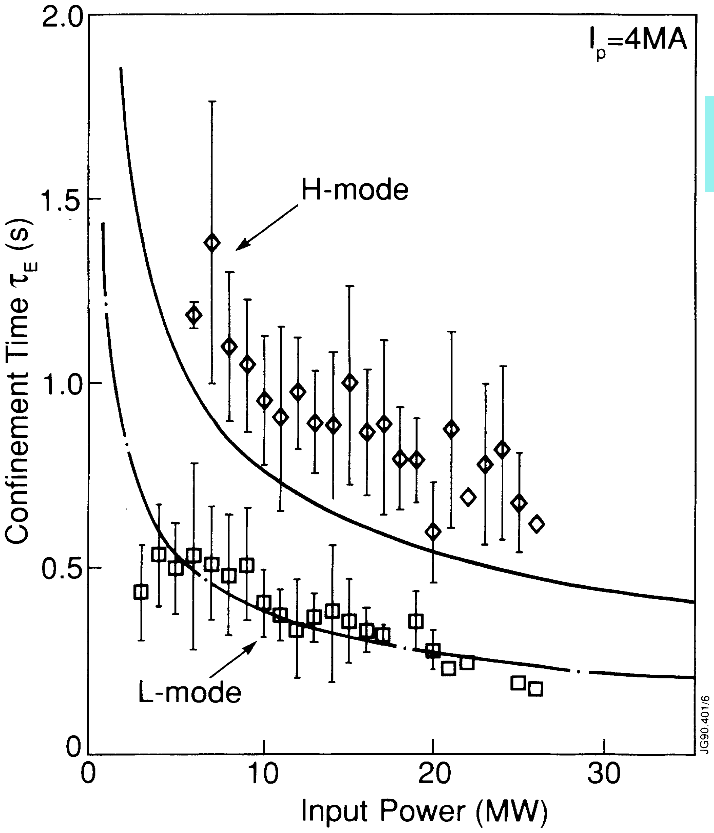
ELMs

Power threshold:
H-mode: $P > P_{LH}$

$$P_{LH} = 2.84 M^{-1} B^{0.82} \bar{n}_{20}^{0.58} R a^{0.81} \quad (\mathrm{MW})$$

Note the isotopic dependence

In Deuterium, $P_{LH}^{ITER} \sim 50$ MW

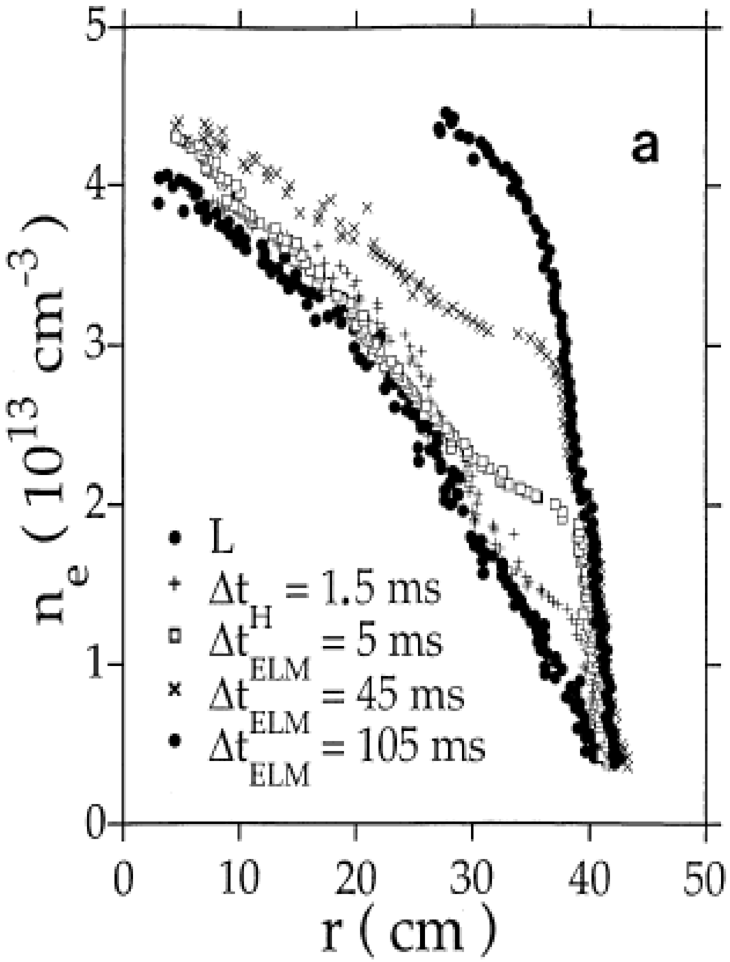


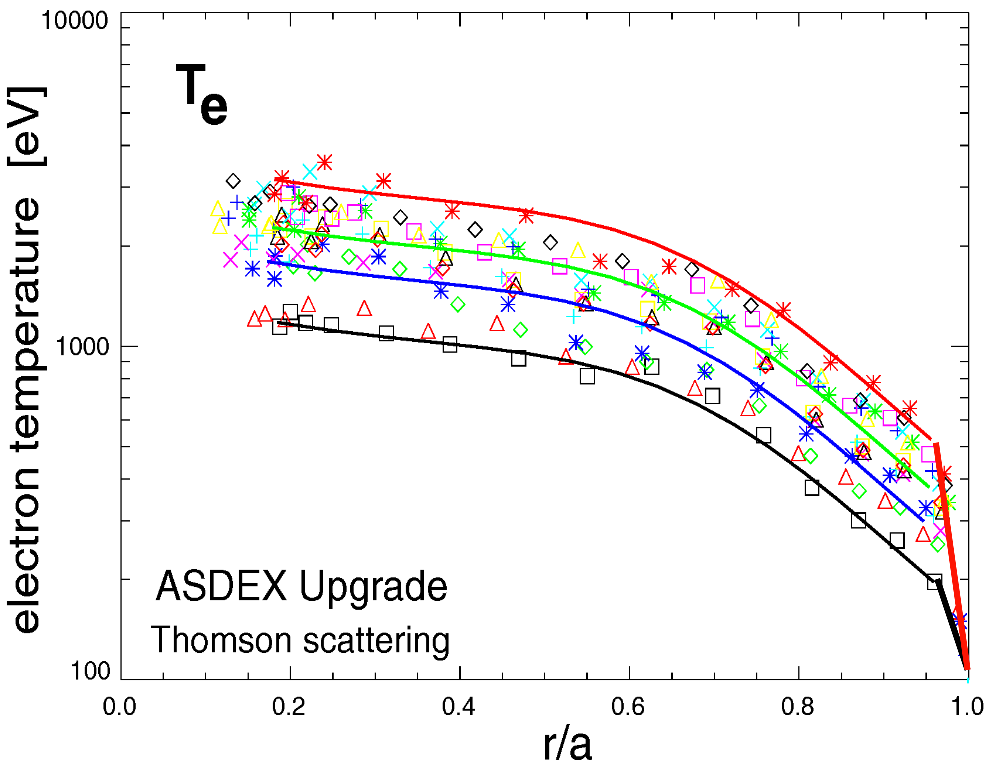
JET

Profile characteristics of the H-mode

Development of a pedestal

Edge transport barrier



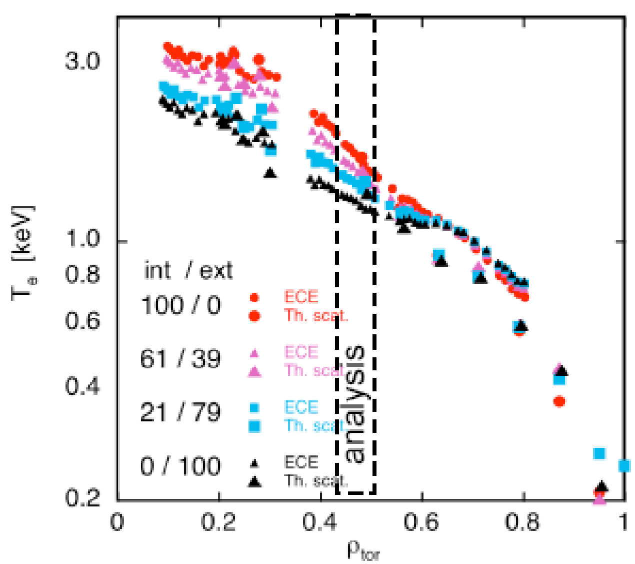


Note the similarity of the T_e profiles
“profile stiffness”

Electron temperature profile stiffness and TEM

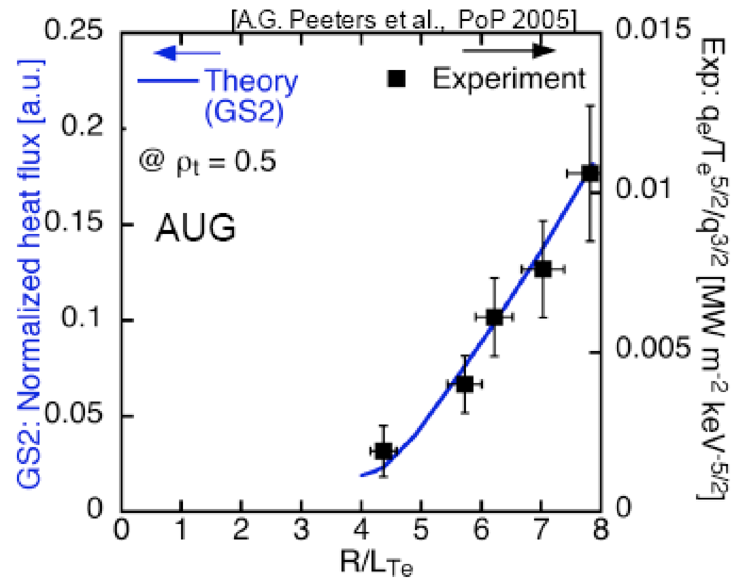
ASDEX-upgrade; F. Ryter

On-axis and off-axis



R/L_{Te} varies continuously
R/L_{Te} ≈ 4 with off-axis

Comparison of experimental results with gyro-kinetic calculations



TEM dominant modes

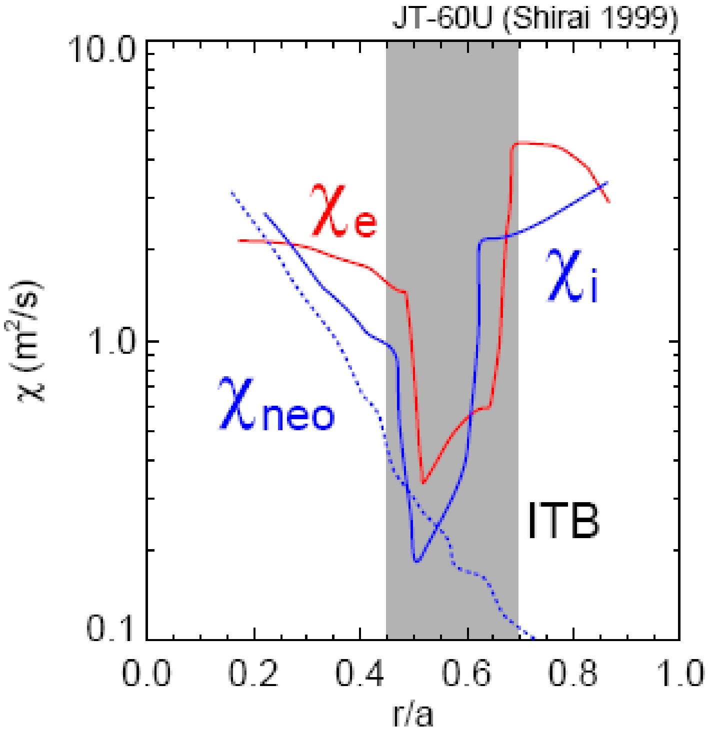
Threshold in R/L_{Te} agrees

Slope q_e = f(R/L_{Te}) agrees

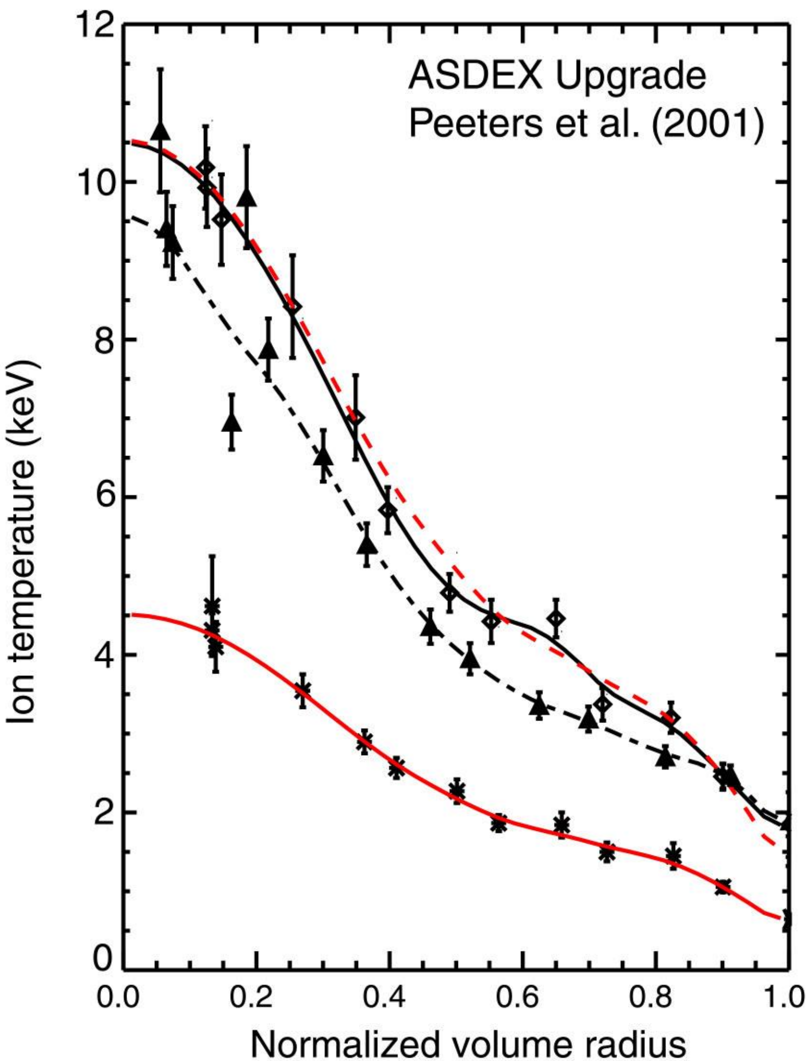
Similar results from T_i profile analysis and γ and R/L_{Ti} for ITGs

Nonlinear response of heat flux against gradient

Internal Transport Barriers

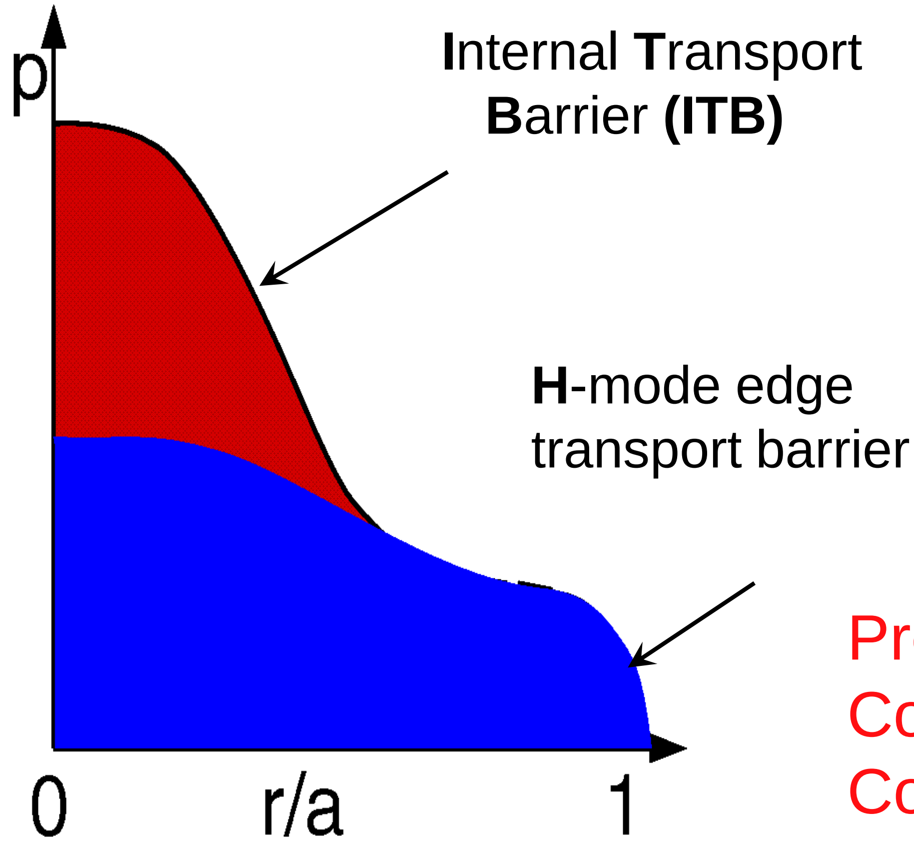


ITB



Heating
Magnetic shear
Etc.

The hope for ITER



Predictability?
Compatibility?
Controllability?

Understanding necessary

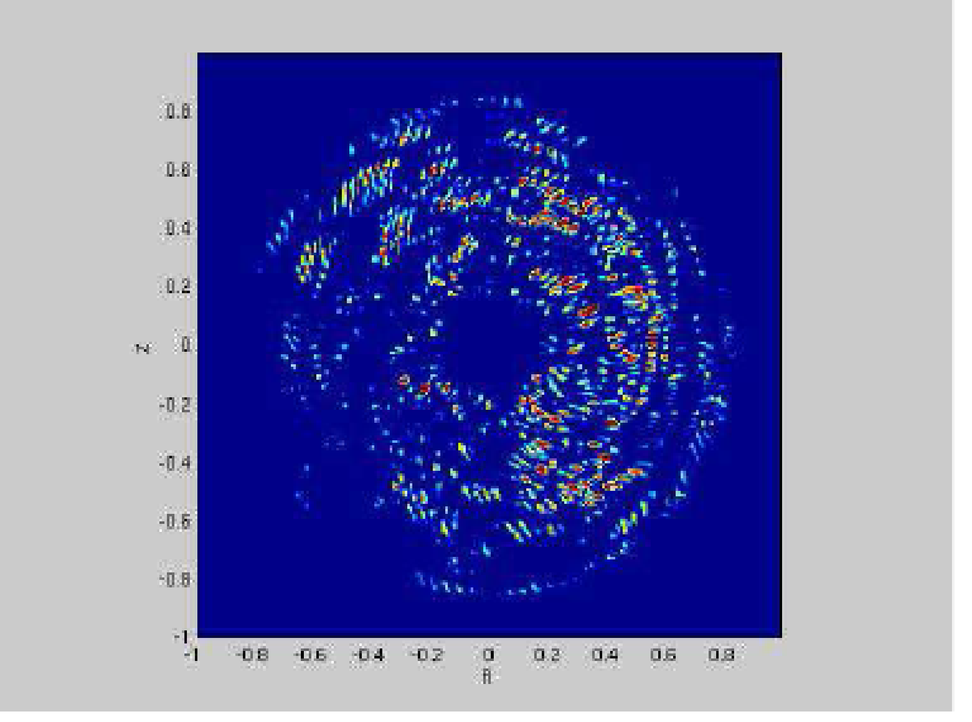
Turbulent transport is dominant

- Losses are mainly conductive

$$\tau_E \approx \frac{a^2}{\chi_{turb}}$$

→Turbulent diffusion χ_{turb} determines the confinement.

- However:
- parallel transport is nearly collisional,
- collisional transport can be dominant in transport barriers.



Turbulent flux

Fluctuations of ExB drift velocity produce turbulent transport

- ExB drift

$$v_E = \frac{B \times \nabla\phi}{B^2}$$

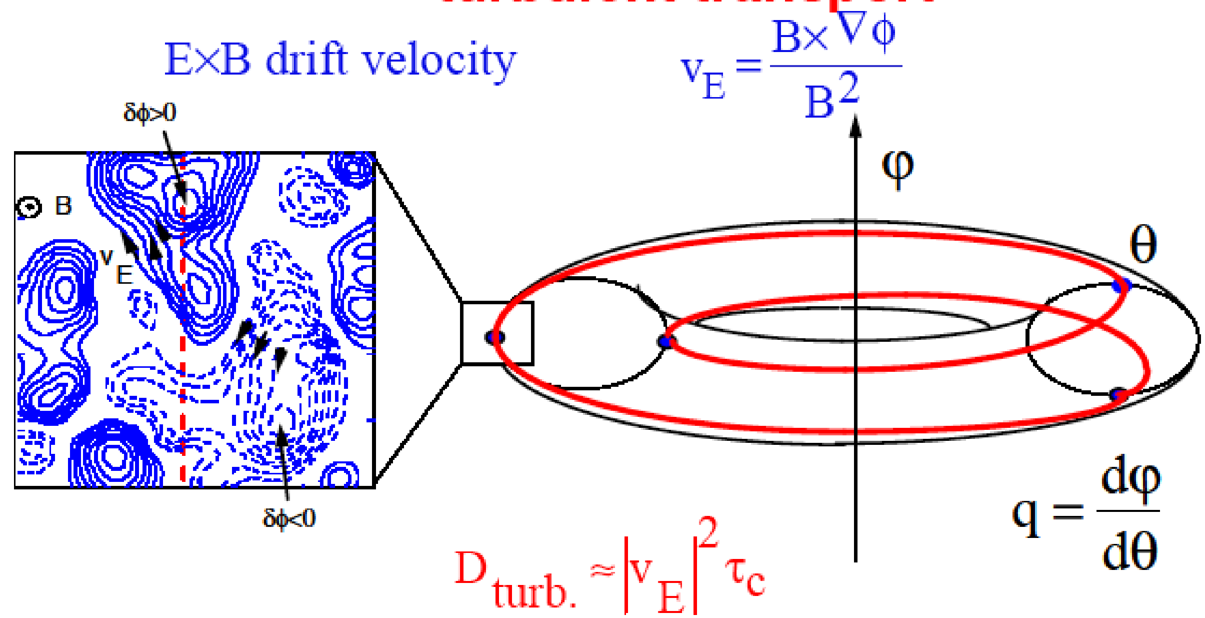
- Turbulent diffusion

$$D_{turb} \propto |v_E|^2 \tau_c$$

$$\propto L_c^2 / \tau_c$$

- Turbulent flux

$$\phi_E = \frac{3}{2} \langle p v_E \rangle$$

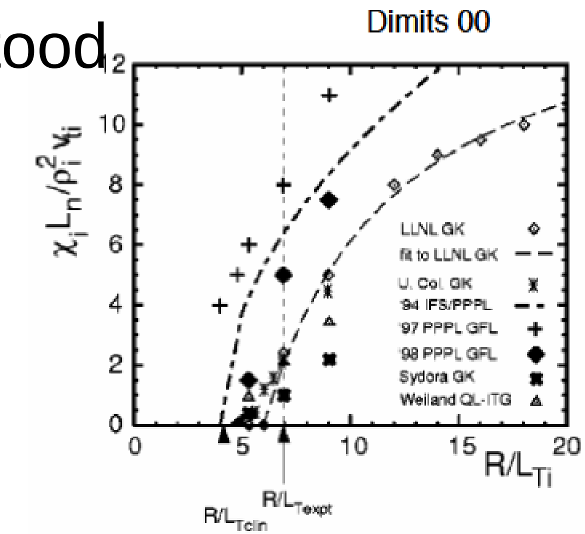


What determines length and time ?
Nonlinear theory and simulation are necessary.

Assessment

Ion heat transport is rather well understood

ITG dominated, inclusion of ZF effect.

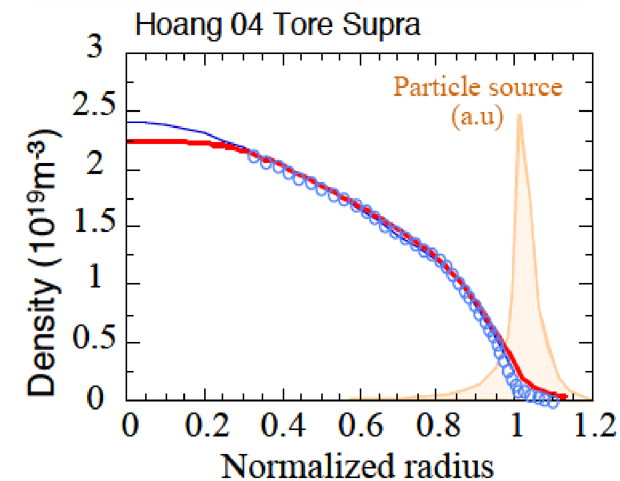


Electron heat transport

TEM important, ETG debated.

Particle transport

Turbulent pinch need further study.



Momentum transport

More puzzling.

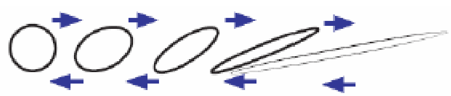
Structure of momentum radial flux Diamond 07

$$\Gamma_\Omega = -D\frac{d\Omega_\varphi}{dr} + V\Omega_\varphi + S$$

pinch residual stress

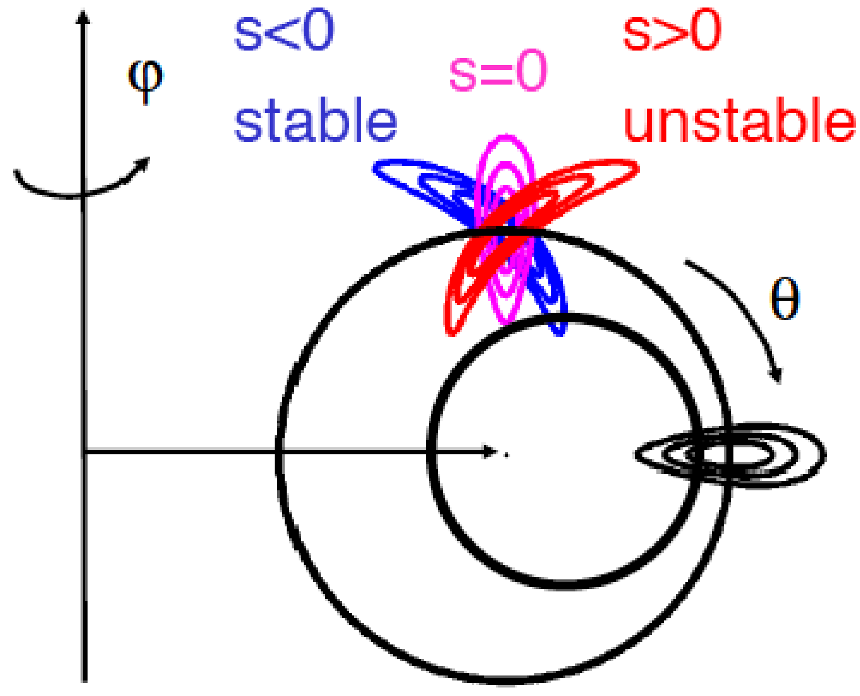
Several mechanisms may lead to improved confinement

- Flow shear: same effect as Zonal Flows



- Magnetic shear

- T_e/T_i, Z_{eff}, density gradient, fast particles… : not generic



Vortex distorsion

Development of reduced models: present status

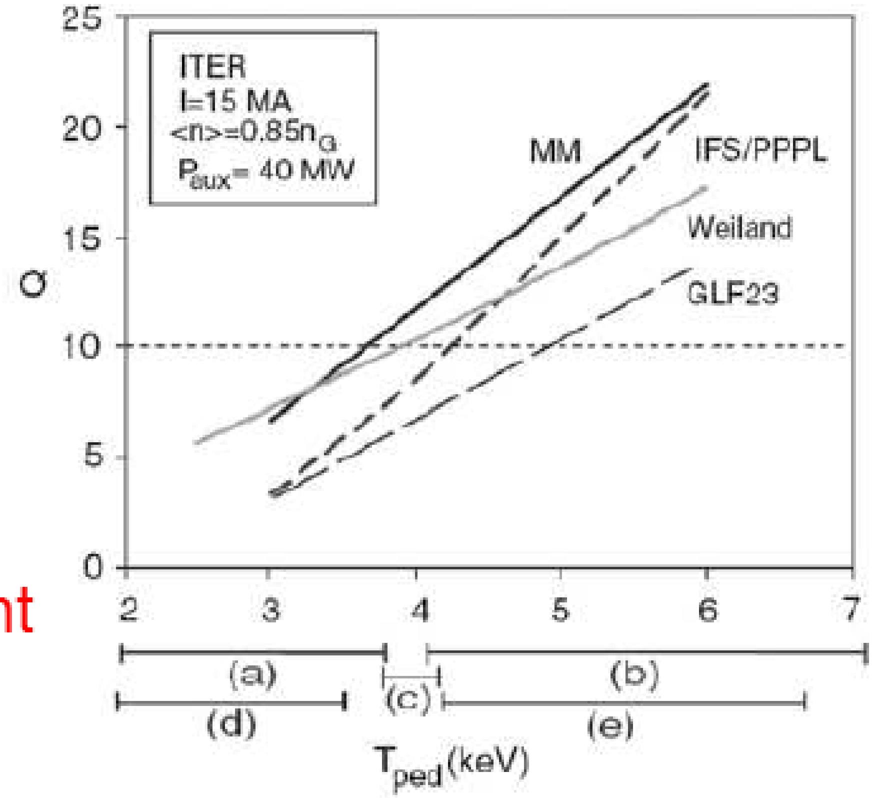
- Encouraging results

see lecture by Pr Fukuyama.

- However, still some uncertainty on the prediction of ITER performances.

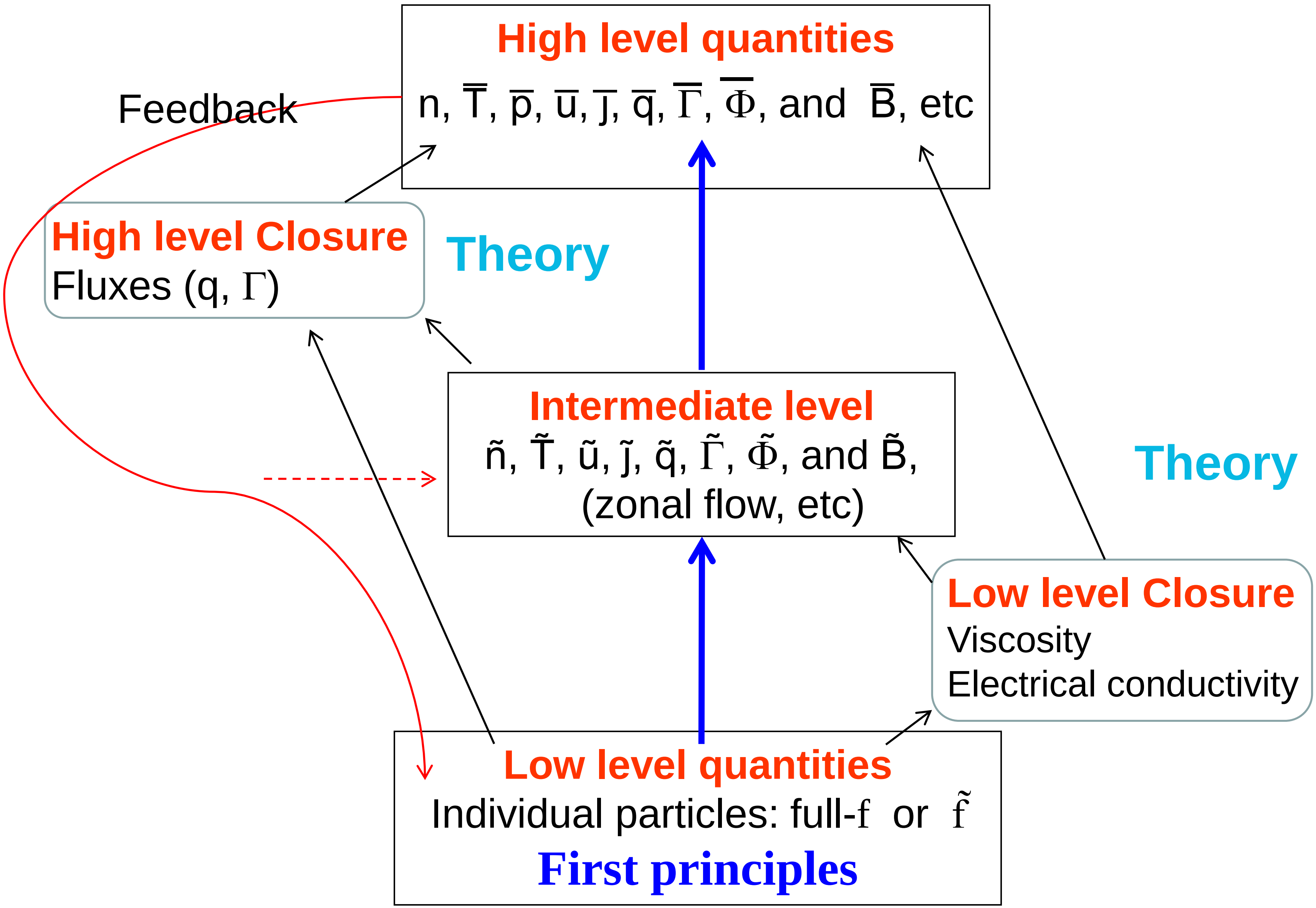
- Requires an improvement on transport models.

Iter Physics Basis



Statistical (primacy) hierarchy levels

What physical quantities are we trying to compute?



Rapid advancement was made in:

Basis of gyro-reduced kinetic equations

Library for PDE solvers

Architecture of parallelization

Shared memory, distributed memory, domain decomposition

Particle simulation

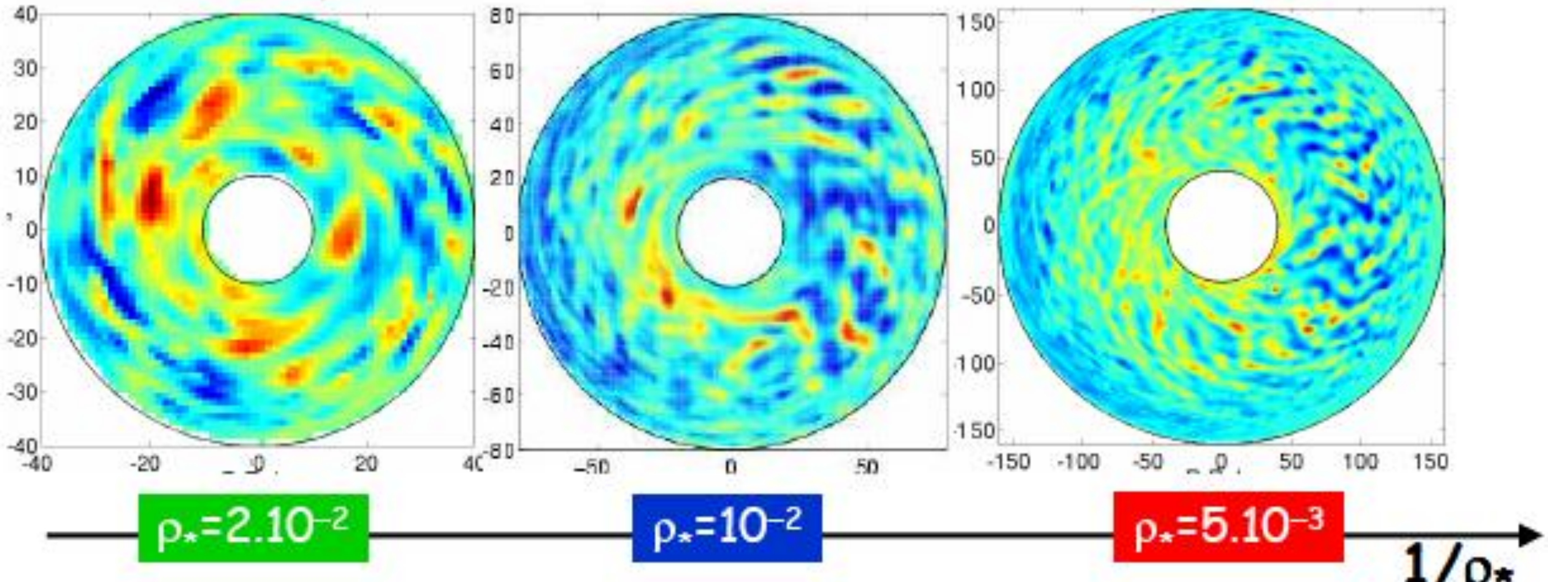
Pros and Cons of various gyrokinetic simulation types

Types	**Pros**	**Cons**
Radially and toroidally global	Large scale event Toroidal mode coupling	Computationally expensive
Radially global, toroidally wedge	Radial relations	Toroidal mode coupling? (Verifications exist)
Radially local ($\rho/a \to 0$)	Computationally cheaper	Large scale radial event?

An example of gyroBohm scaling

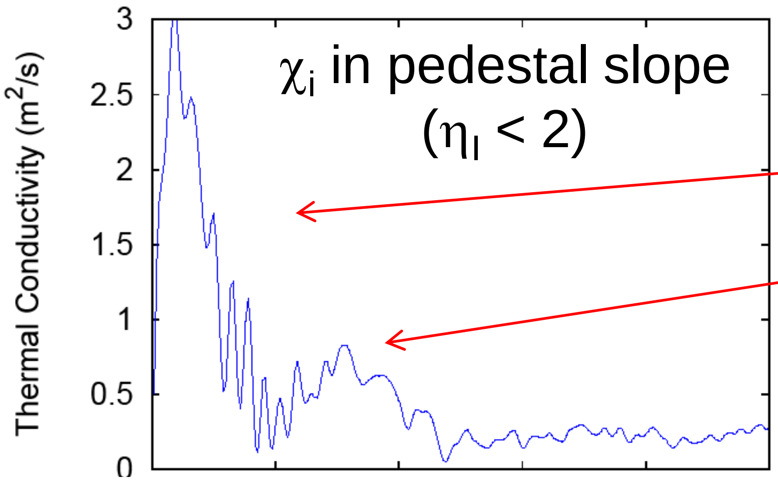
- Simulations where the scale ρ^* is changed by a factor 4
- Agree with $L_c \equiv \rho_c$ and $\chi \equiv (T/eB)\,\rho_c/a \rightarrow \omega_c \tau_E \equiv \rho_*^{-3} F(\beta, \nu_*)$

Sarazin 07



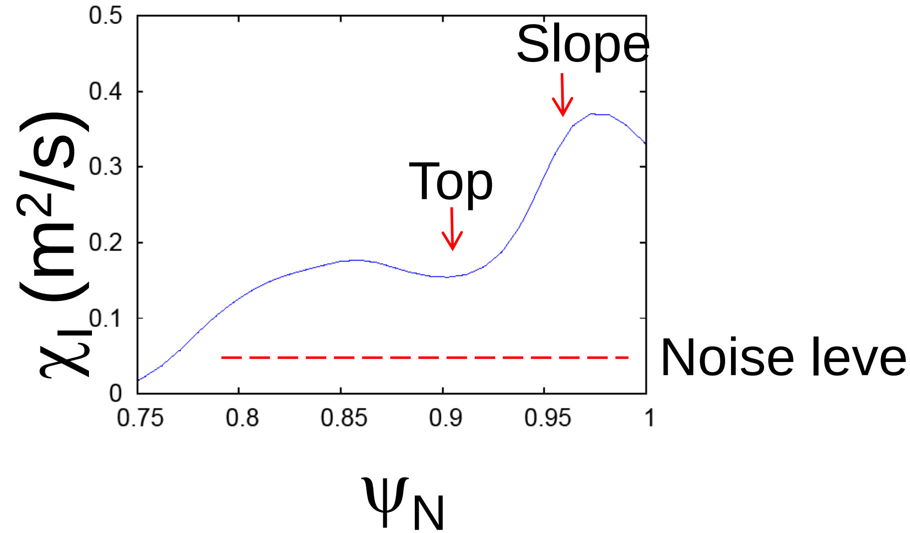
Ion thermal conductivity behavior in time

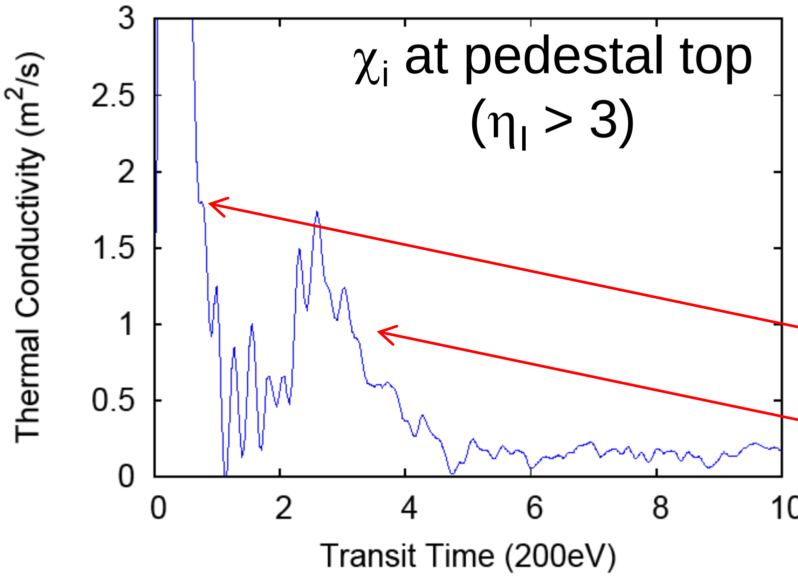
- Collisionless
- 3.2 billion particles
  (3,500 particles per node)



Neoclassical settling down

ITG growth and settling down





Neoclassical settling down

ITG growth and settling down

Code (solution) verification

Are the computational model equations solved correctly and accurately? Verification deals with mathematics.

1. Numerical studies of convergence rates
2. Monitoring of physically conserved quantities
3. Benchmarking with other codes
4. Comparing with analytical solutions
5. Method of manufactured solution

Code (model) validation

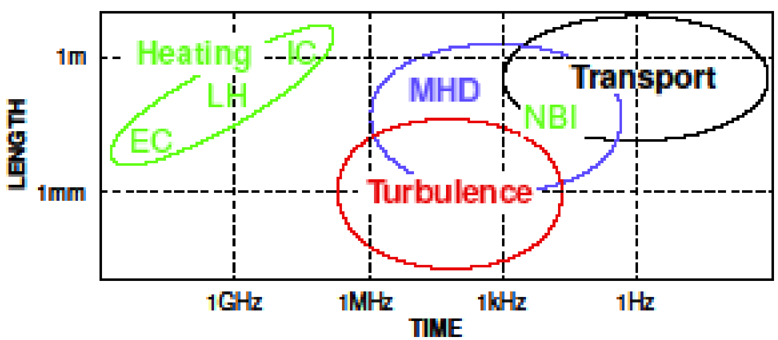
Are the "models" accurate representation of the real world? Validation deals with physics experiments.

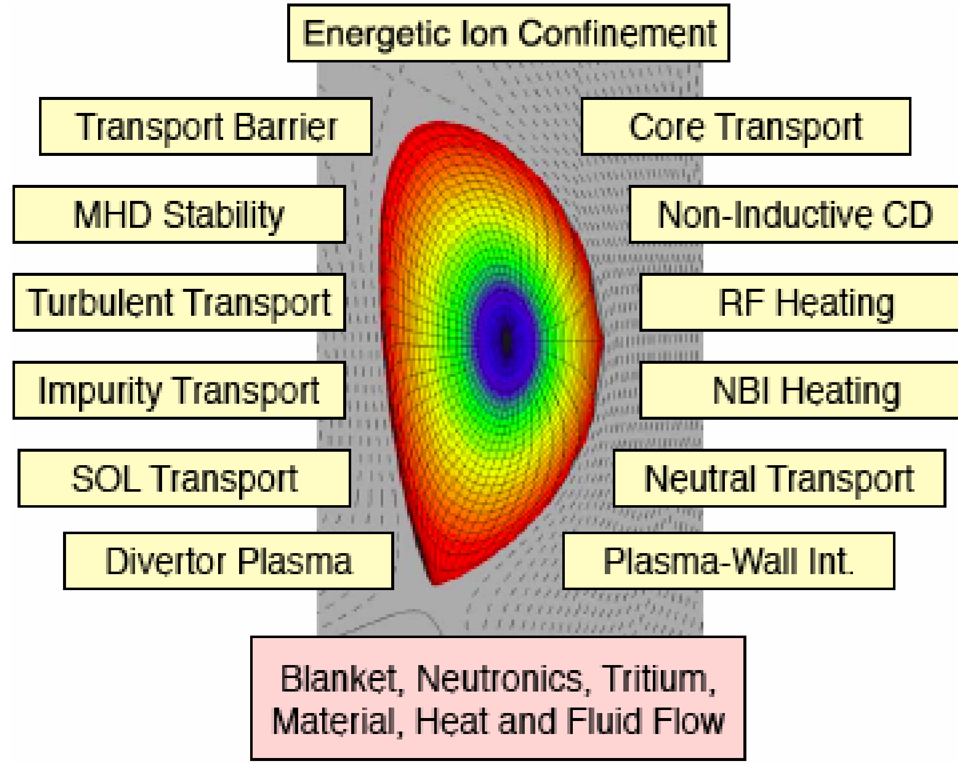
1. The "models" include the equations and the solving conditions.
2. More meaningful after verification
3. Should include the observables at all hierarchical level, if possible.
4. New experiments may need to be designed.
5. Synthetic diagnostics is another issue for meaningful validation

Simulation of Tokamak Plasmas

Broad range of time scale:
100GHz ~ 1000s

Broad range of Spatial scale:
10 μm ~ 10m

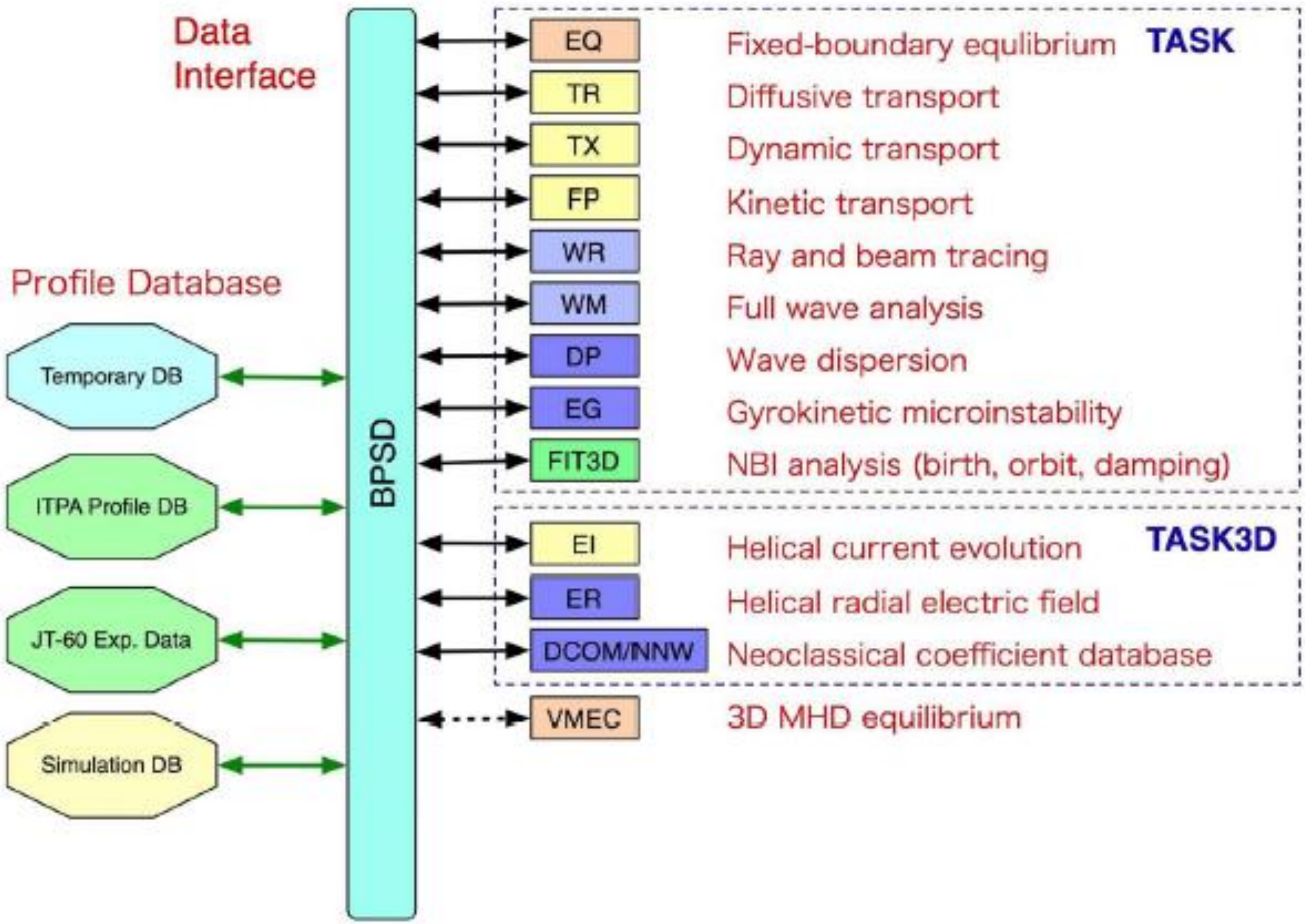




One simulation code never covers all range.

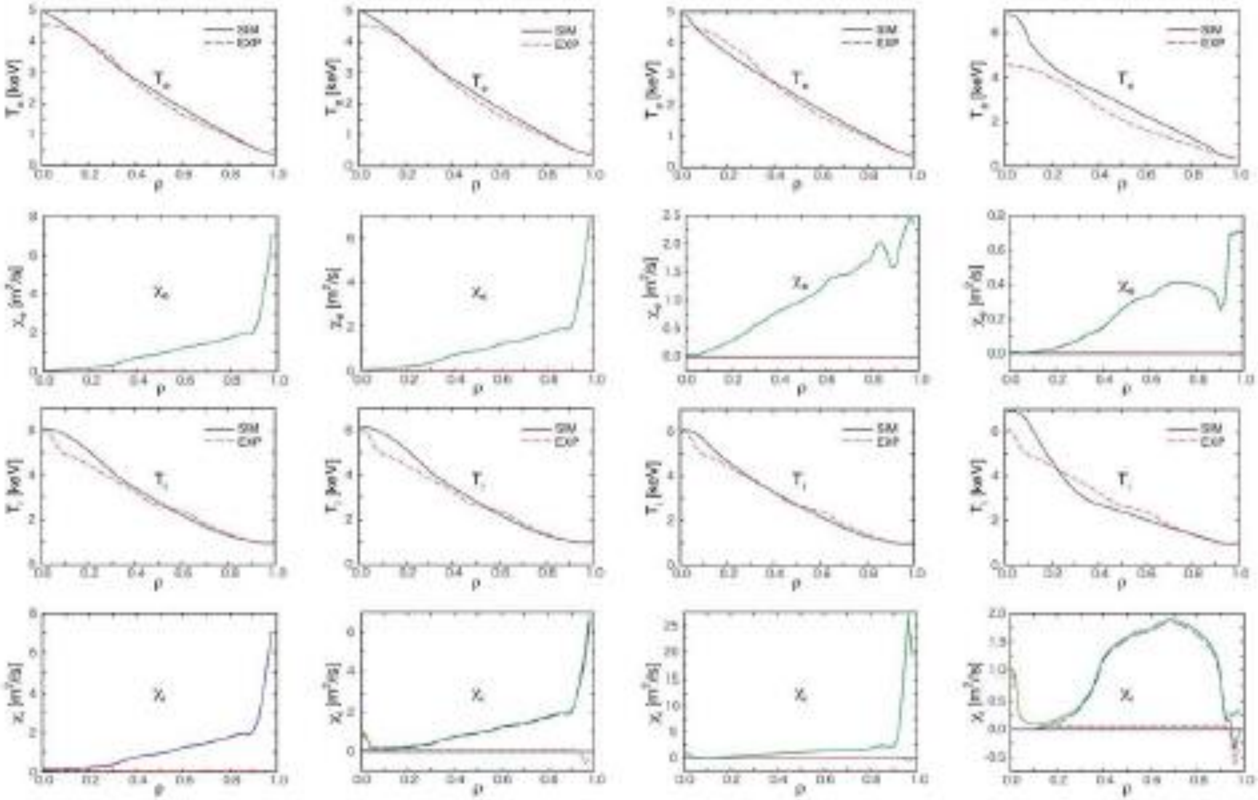
Integrated simulation combining modeling codes interacting each other

Present Structure of the TASK code

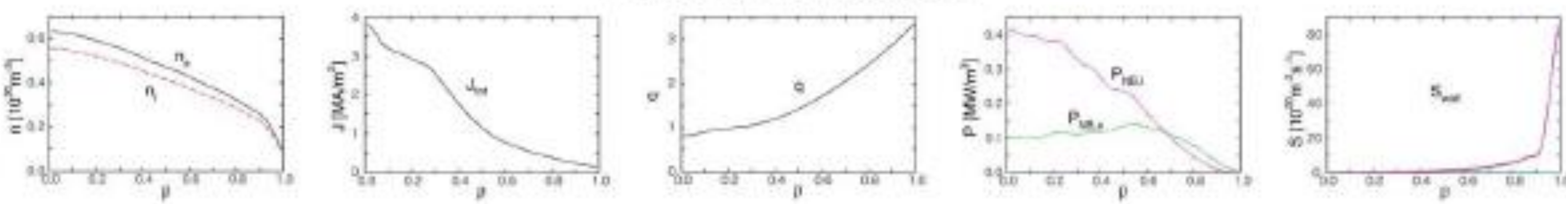


TFTR #88615 (L-mode, NBI heating)

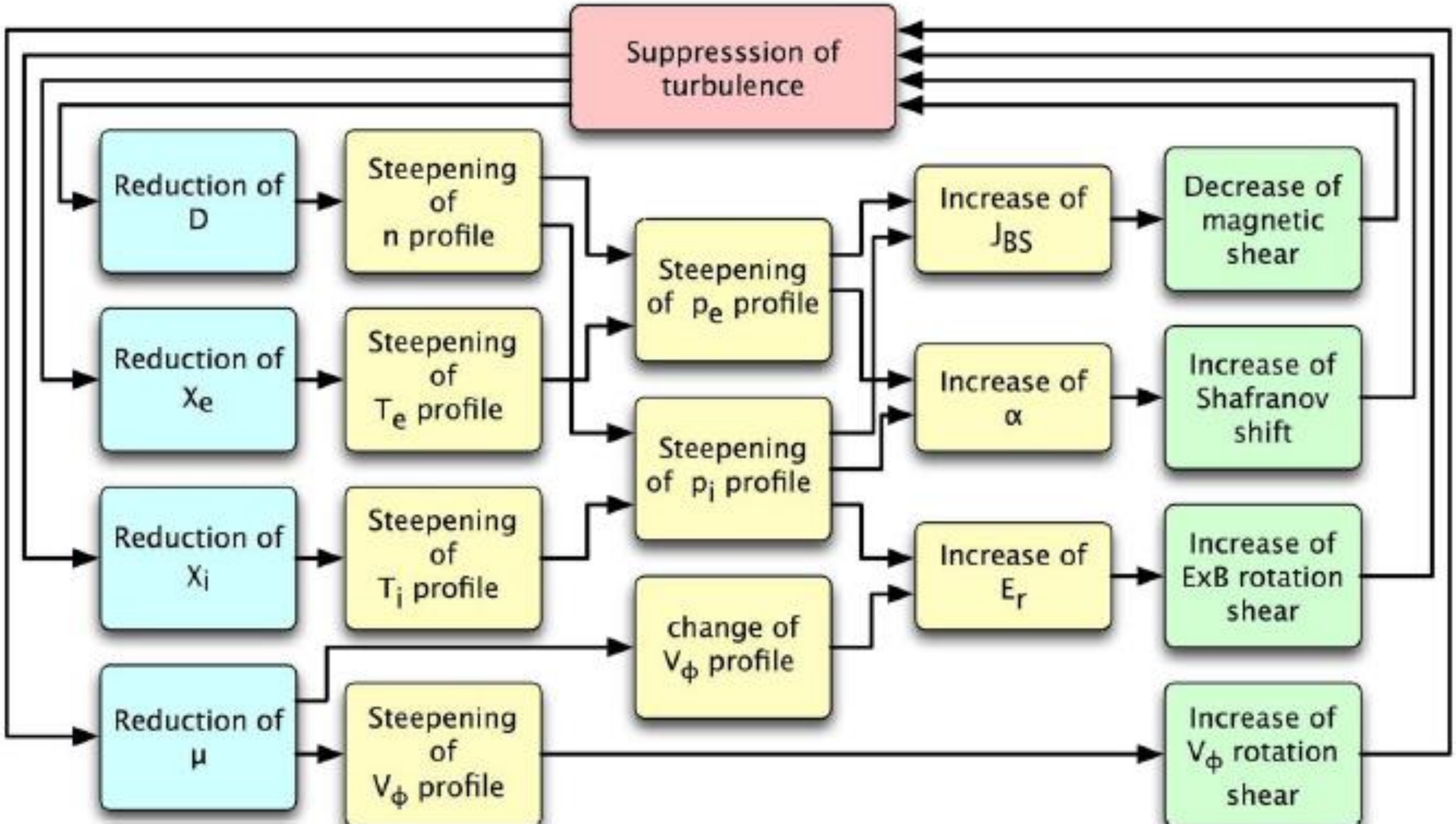
CDBM | **CDBM05** | **GLF23** | **Weiland**



Common Profiles



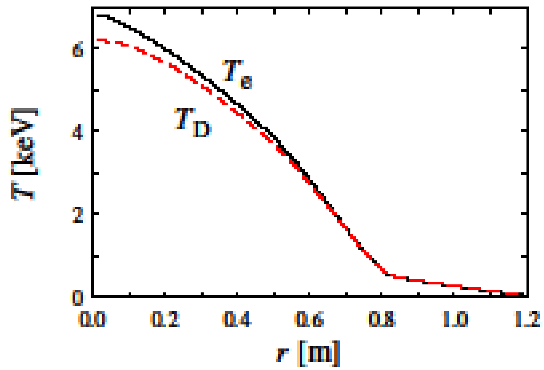
Modeling of Transport Barrier Formation



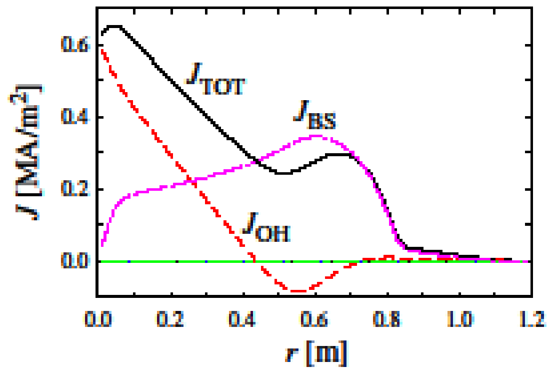
High β_p mode

- $R = 3\,\mathrm{m}$, $a = 1.2\,\mathrm{m}$, $\kappa = 1.5$, $B_0 = 3\,\mathrm{T}$, $I_p = 1\,\mathrm{MA}$
- one second after heating power of $P_H = 20\,\mathrm{MW}$ was switched on

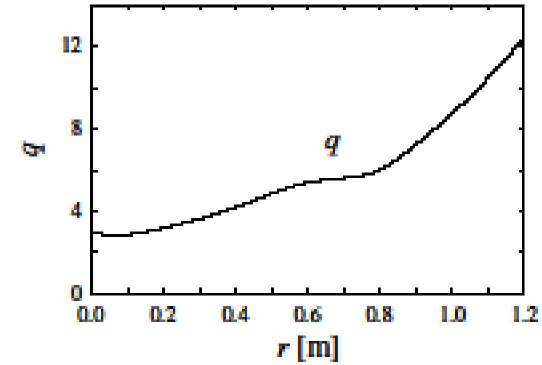
Temperater profile



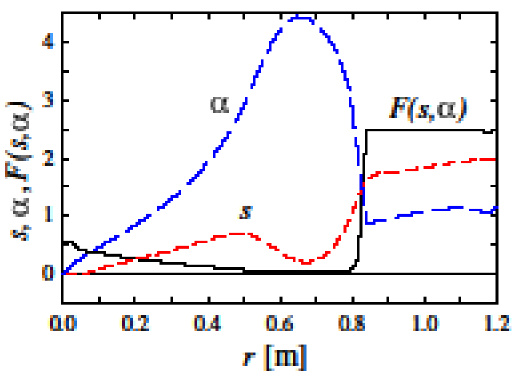
Current profile



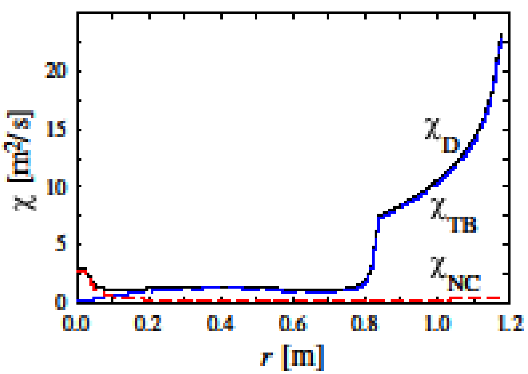
Safety factor



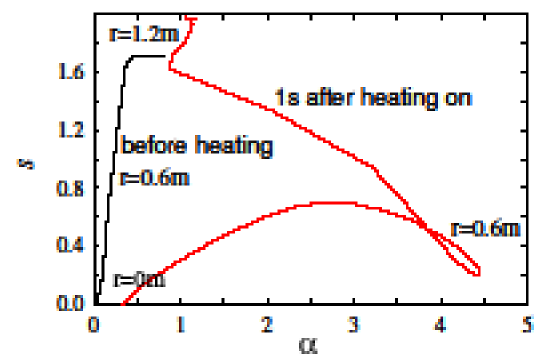
Shear and pressure



Thermal diffusivity

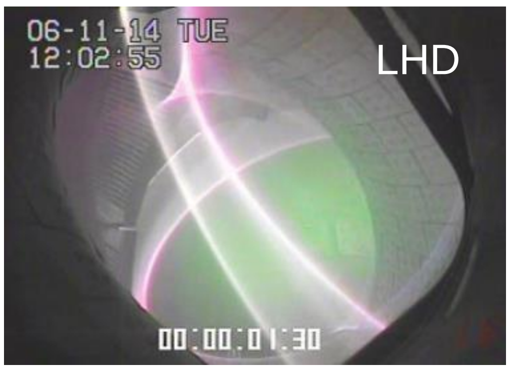


$s - \alpha$ diagram

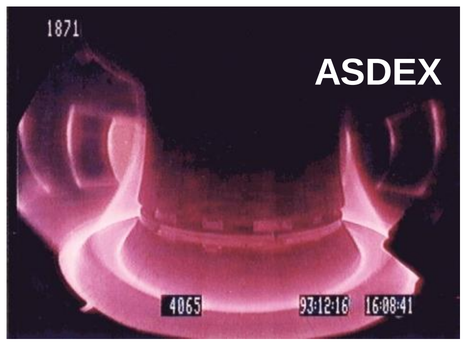


The portfolio of plasma confinement

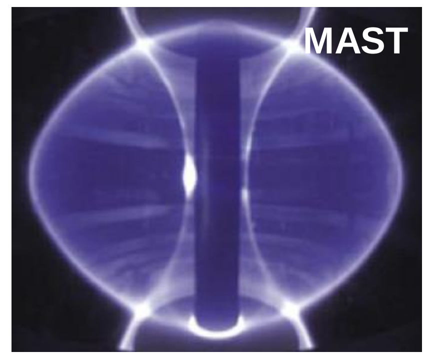
Externally controlled ◀ · ▶ **MHD Self-organized**



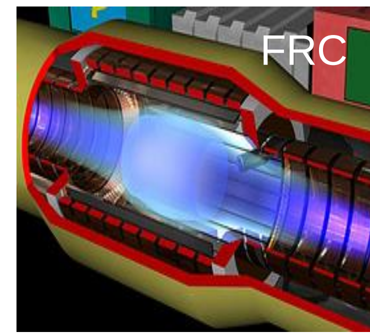
Helical system



Tokamak



Spherical Torus



Field Reversed Configuration

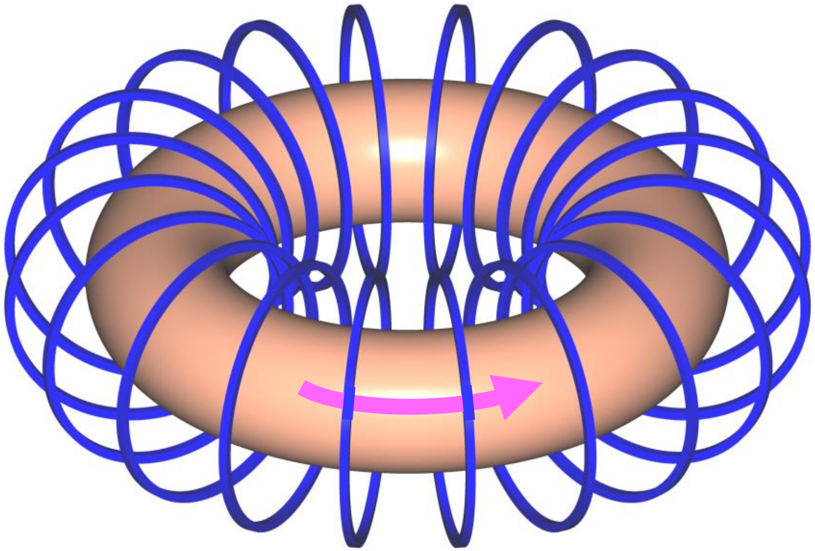
Comprehensive understanding and exact knowledge

Confinement of toroidal plasmas

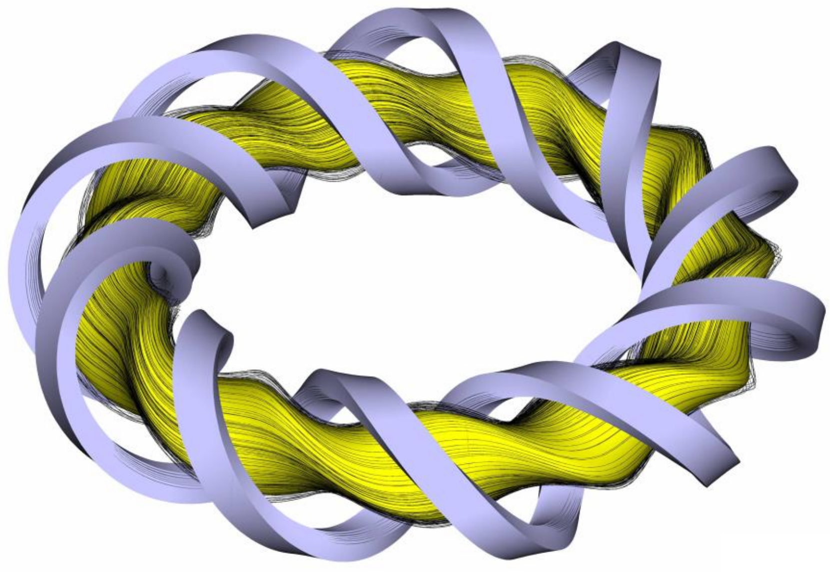
Requirement of rotational transform =
Circum-navigating magnetic field in torus (Toroidal)
+ Circum-navigating in the short way around closed surfaces (Poloidal)

➔ Two ways to generate poloidal field
1) Large net toroidal current in plasma : tokamak
2) Twisted coils : helical system (stellarator, heliotron, heliac,)

A pair of helical coil (double helix) : Heliotron



Tokamak (approximately 2-D)



Helical system (intrinsically 3-D)

3-D MHD equilibrium

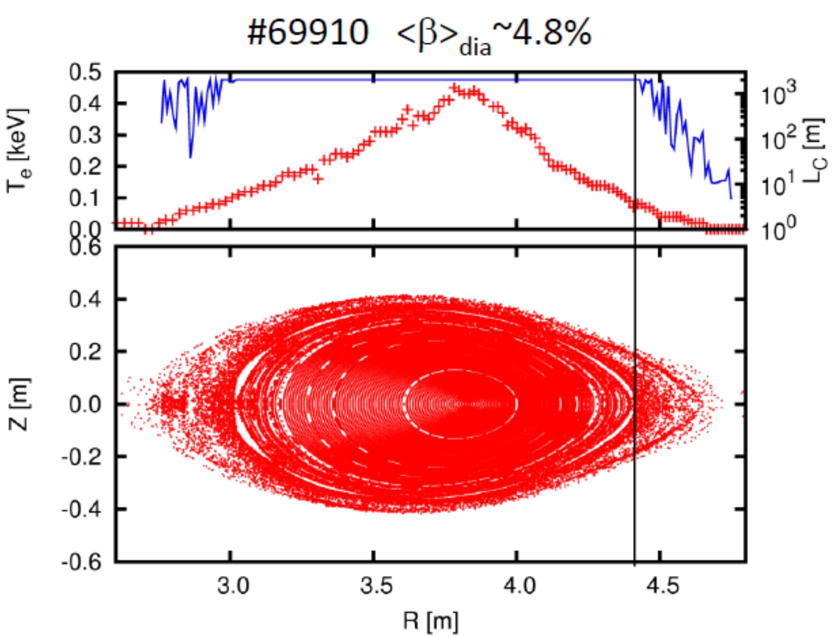
MHD equilibrium

$$\left.\begin{aligned}\mathbf{J}\times\mathbf{B} &= \nabla p\\ \nabla\times\mathbf{B} &= \mu_0\mathbf{J}\\ \nabla\cdot\mathbf{B} &= 0\end{aligned}\right\}$$

Grad-Shafranov eq. for axi-symmetry

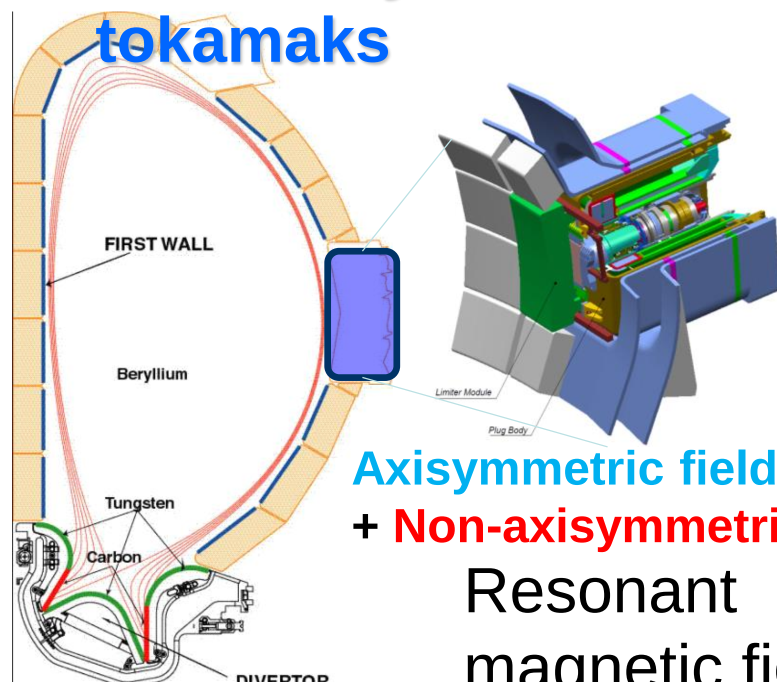
Distinguished feature of 3-D equilibrium : magnetic island, stochastic field

HINT code : calculate 3-D MHD equilibrium with time-dependent relaxation scheme



LHD

Edge plasma is essentially non-axi-symmetric even in tokamaks

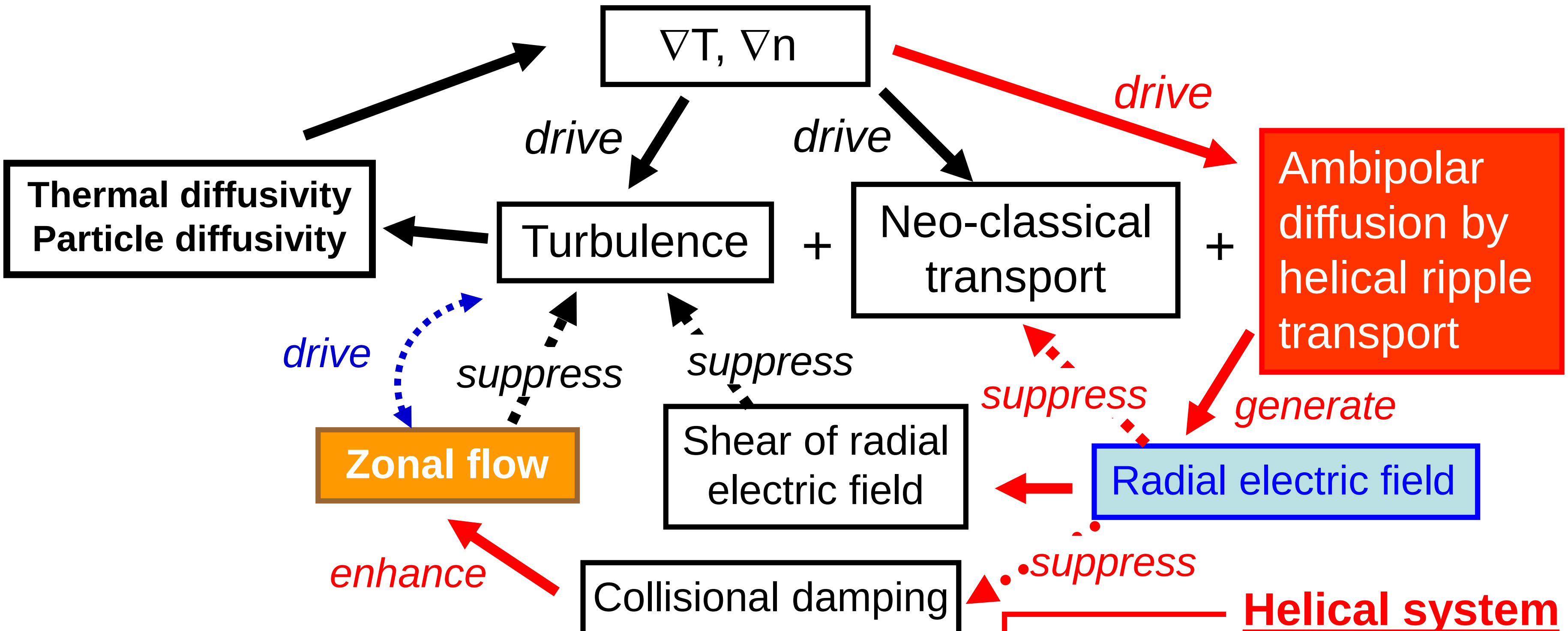


Axisymmetric field

+ Non-axisymmetric wall

Resonant magnetic field coil

Linkage of physical mechanisms to determine transport in toroidal plasmas



Tokamaks

Radial distribution of T&n is determined by self-consistent turbulence with
1) ∇T and ∇n
2) dE_r/dr
3) Zonal flow

Commonality
Contrast

Elevate understanding from correlation to causality

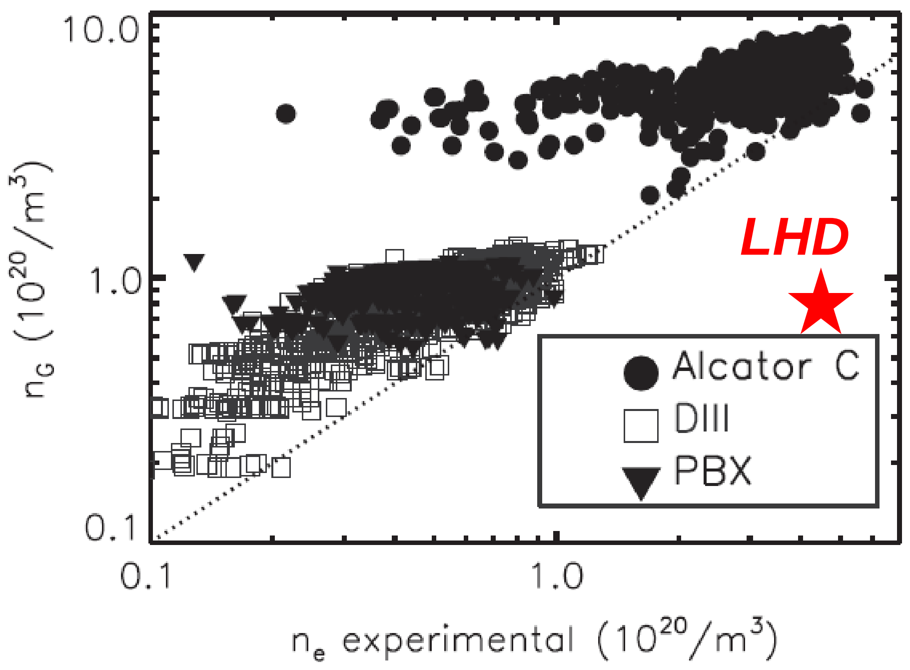
Helical system

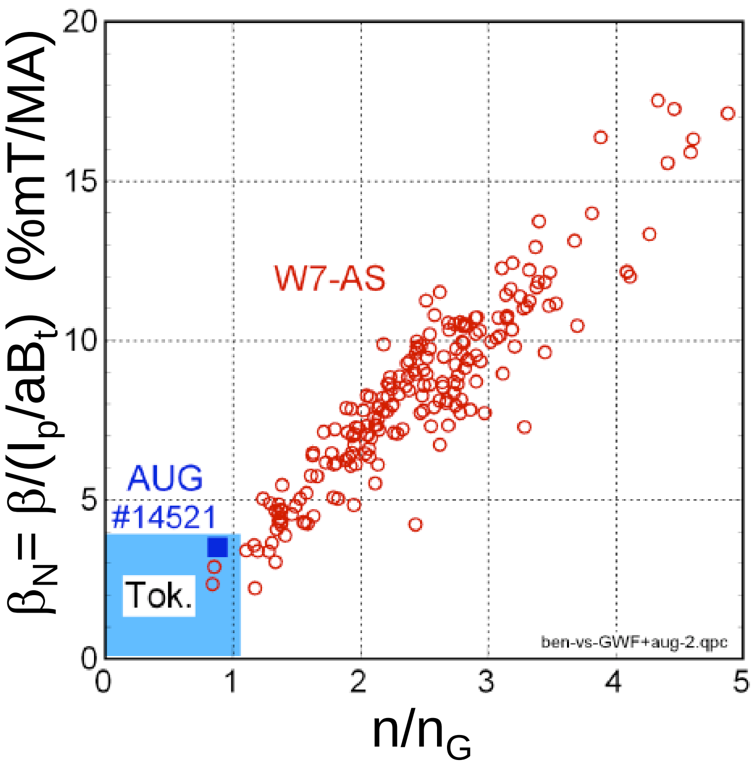
E_r due to helical ripple diffusion can control
1) dE_r/dr
2) Zonal flow via collisional damping
3) Neoclassical transport

➔ a new knob for ∇T and ∇n

Helical systems can be operated in much higher density regime than tokamaks

Greenwald density limit $n_G = \kappa J = I_p / \pi a^2$





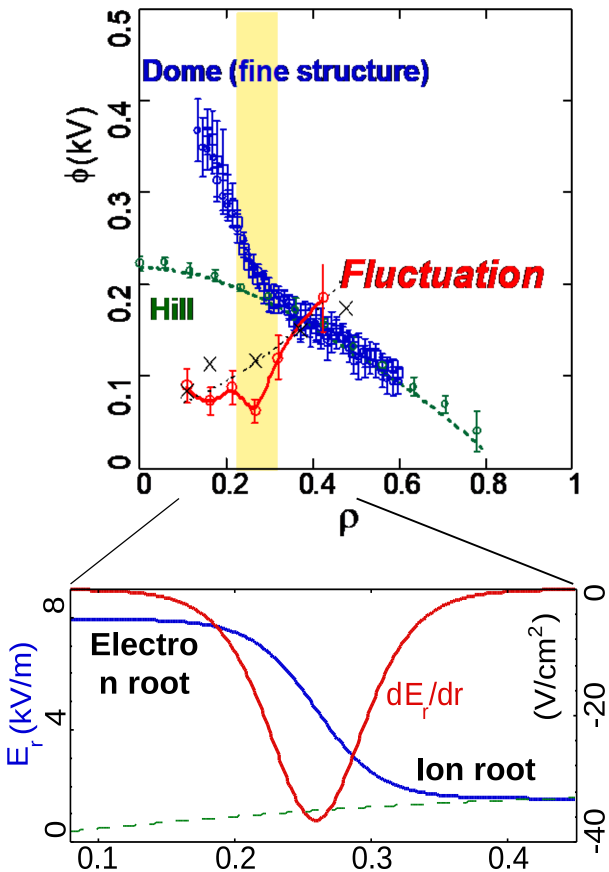
M.Greenwald, PPCF 2002

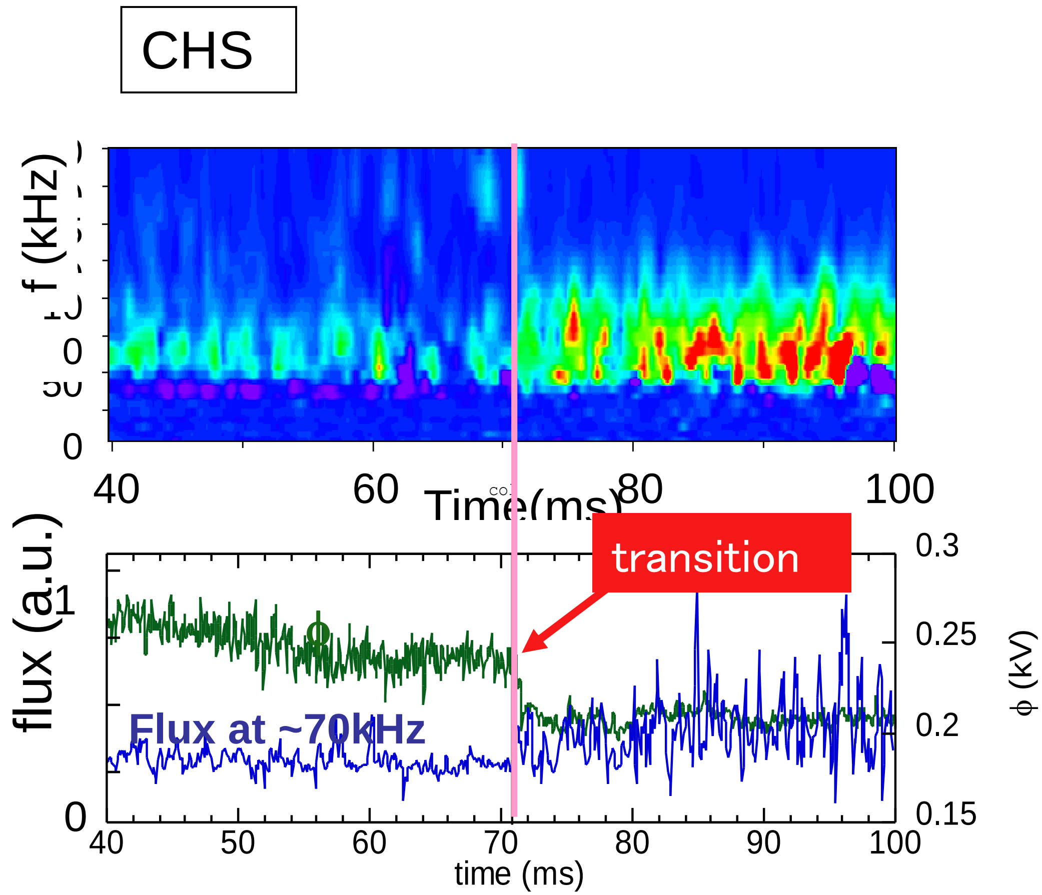
Courtesy of A.Weller

Clarification of underlying physics of density limit

Spontaneous Transition and Suppression

Simultaneous measurement of potential, density and fluctuations





Er-shear really suppresses the turbulence

Why is the power modulated？

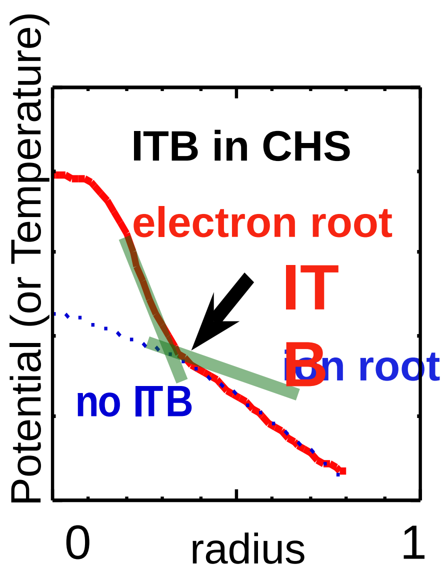
A. Fujisawa et al., PPCF **48** S205 (2006).

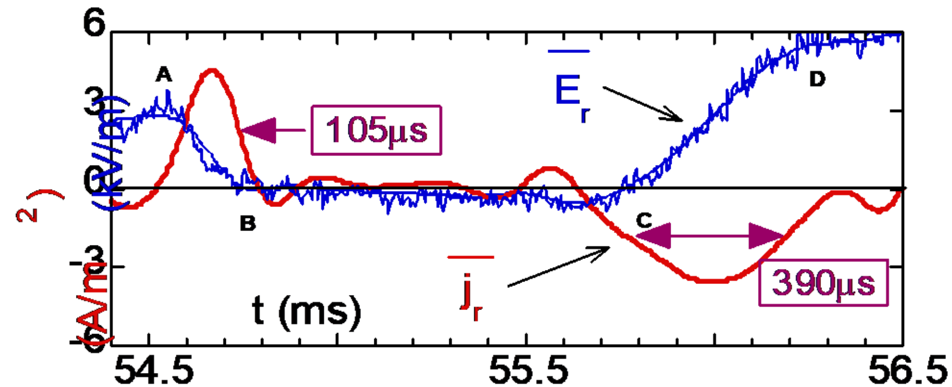
Physics of Bifurcation

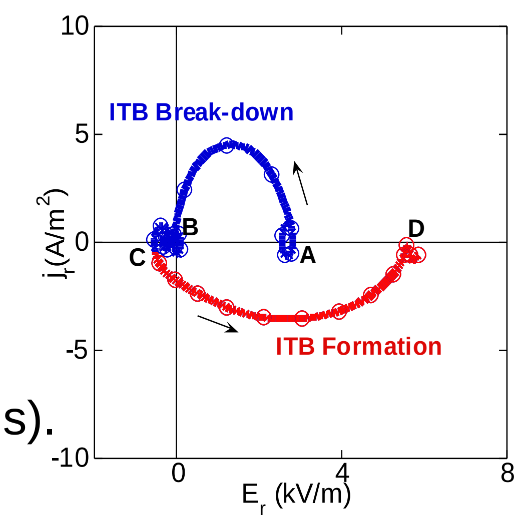
S-I. Itoh, K. Itoh, PRL **60** 2276 (1988)

Equation of E_r developments $\varepsilon_0 \varepsilon_p \dfrac{\partial E_r}{\partial t} = -\Gamma_i^{Neo}(E_r) + \Gamma_e^{Neo}(E_r) + \cdots$

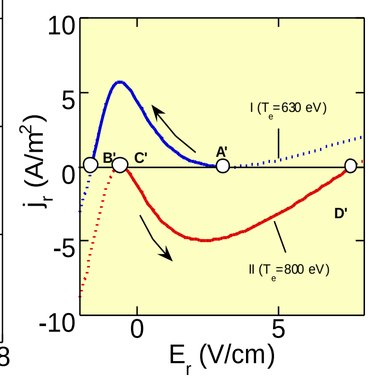
CH
S







Neoclassical calculation



Electric field transition happens
in much faster time scale (~100us)
than confinement timescale (~ a few ms).

Electric field can trigger the bifurcation

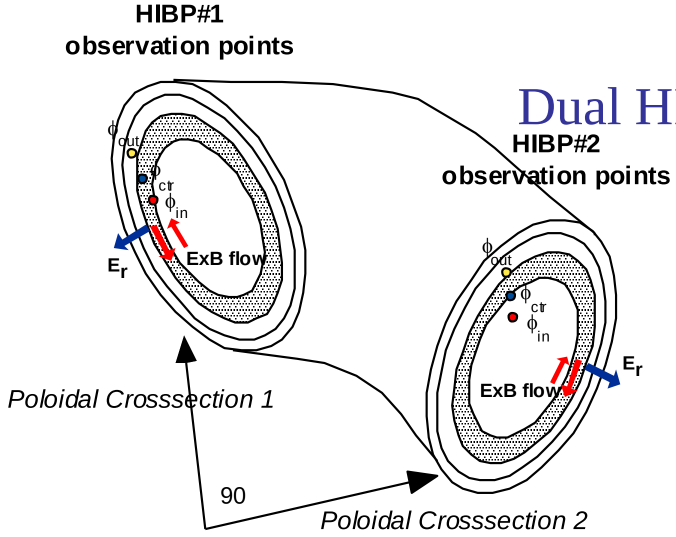
A. Fujisawa et al. PRL 79 1054 (1997)

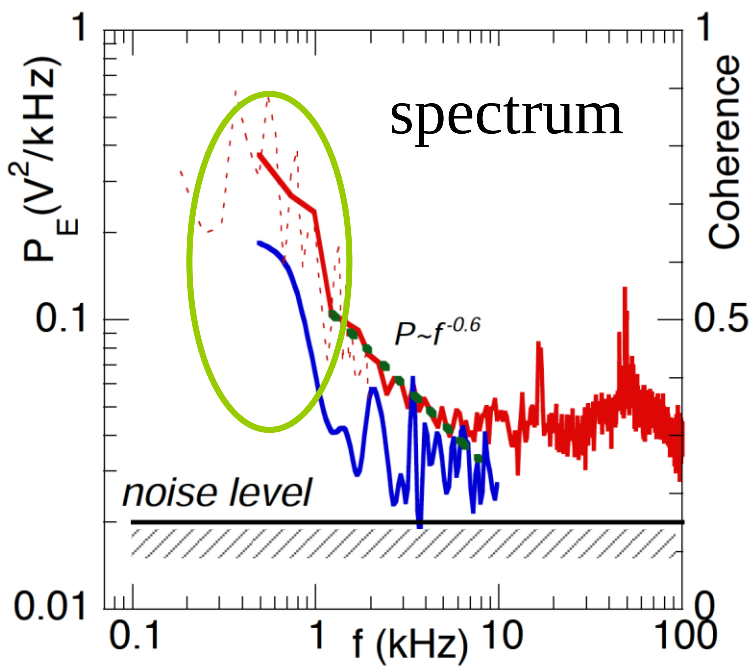
Discovery of Zonal Flows on CHS

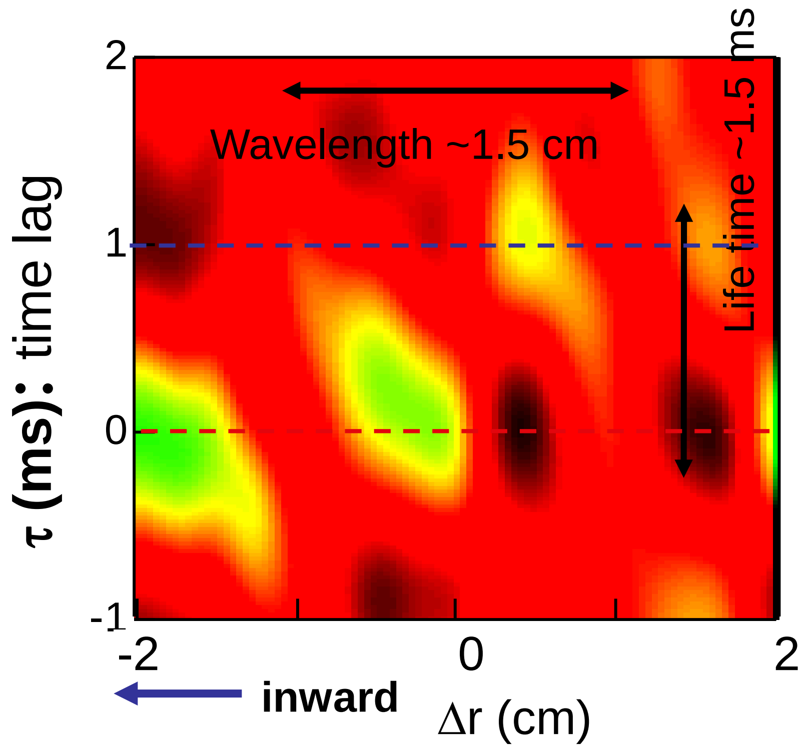
(Fujisawa, PRL 2004)

Plasma Turbulence - Itoh Project -

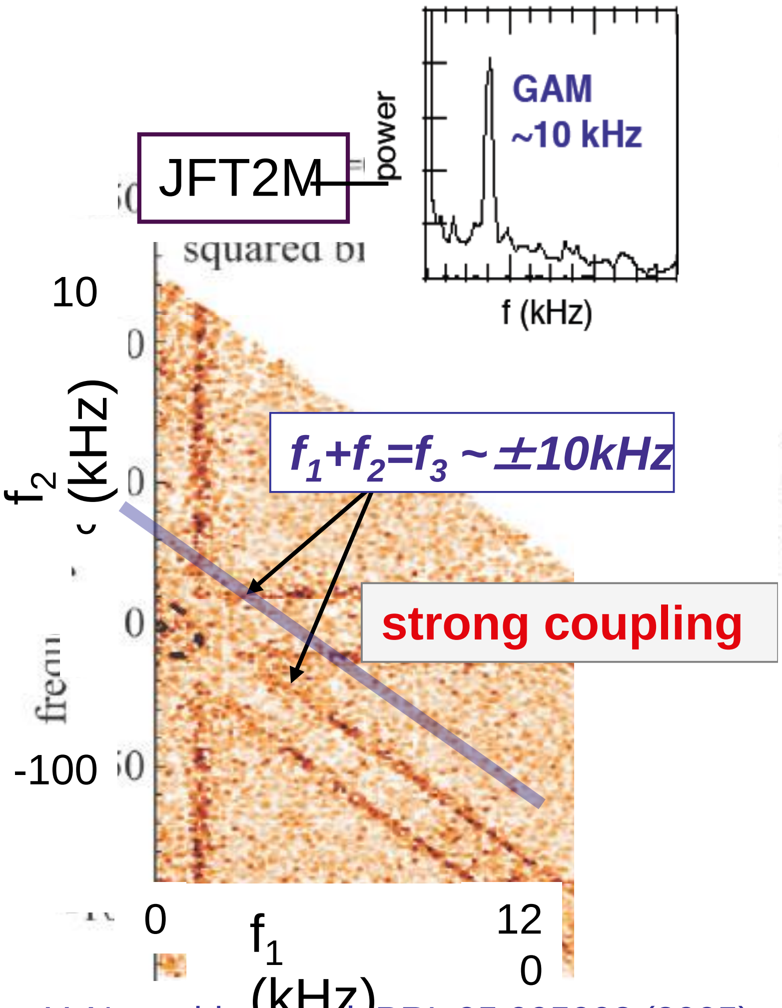
Dual HIBP





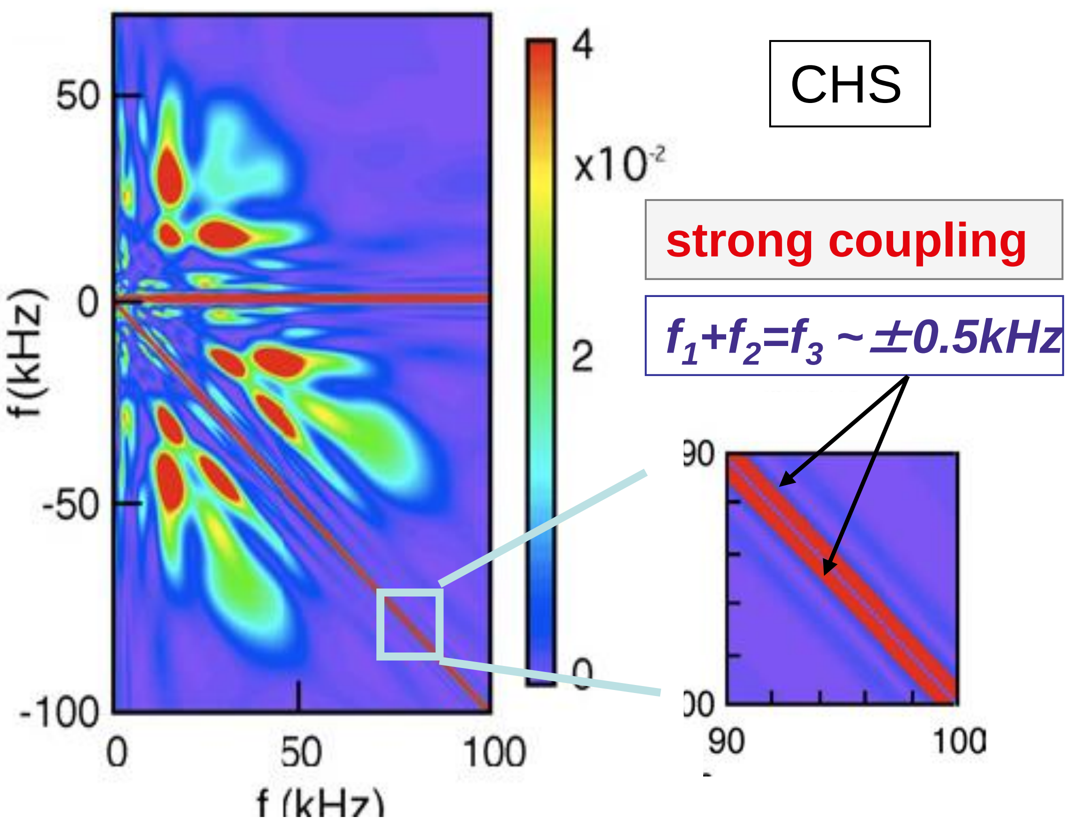


Bicoherence Analysis



Y. Nagashima et al. PRL 95 095002 (2005)

Bicoherence based on the Moleret wavelet

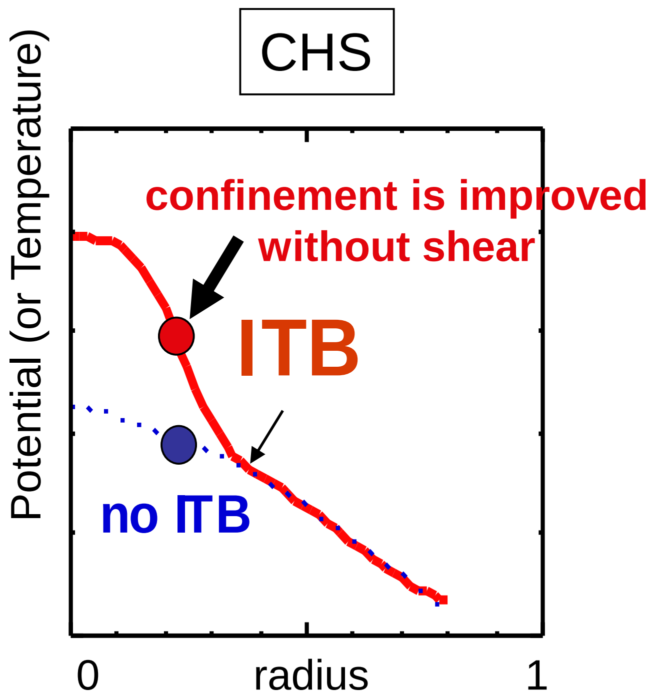


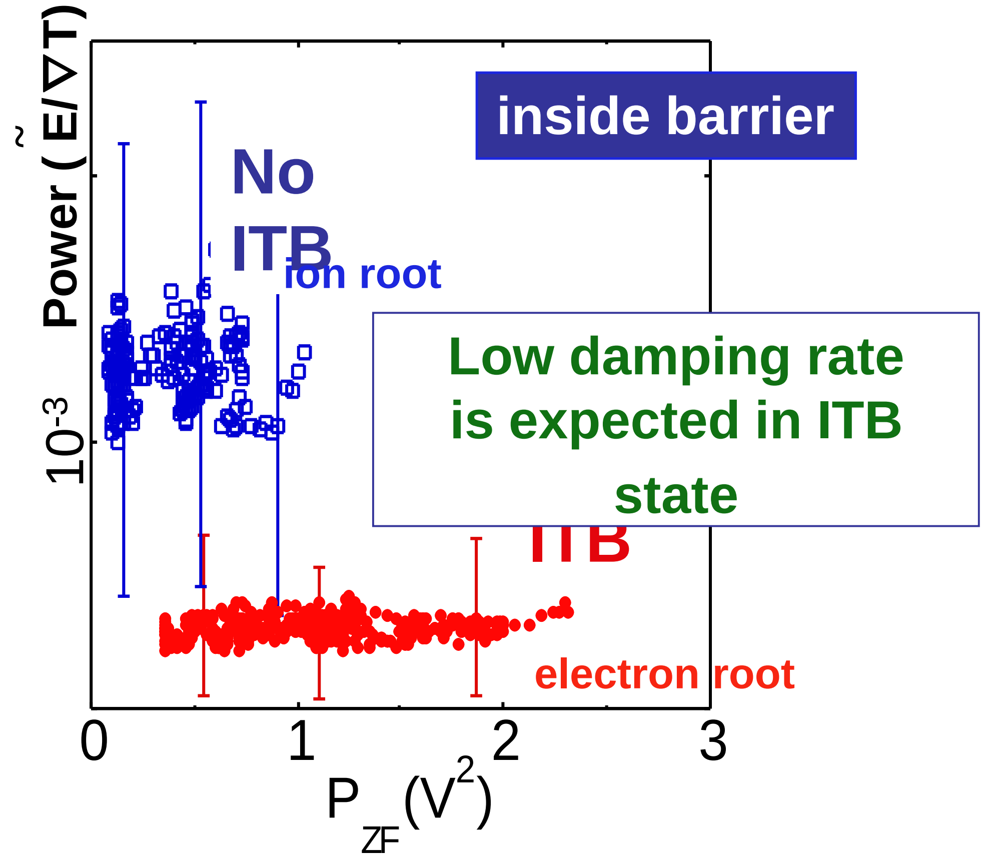
A. Fujisawa et al., PPCF 49 211 (2007)

Coupling between zonal flows and background turbulence is confirmed using bicoherence analysis

Improvement inside Barrier

A mystery: why is the confinement improved inside the barrier?





K. Itoh et al., Phys. Plasmas 14 20702 (2007)

Energy partition between zonal flows and turbulence is a key to determine the transport

New knob for control

Tutorial for unifying understanding

Mysteries and challenges

QuickTimeý Ç²
TIFFÅiLZWÅj êLí£ÉvÉçÉOÉâÉÄ
Ç™Ç±ÇÃÉsÉNÉ`ÉÉÇ¾å©ÇÈÇžÇ½Ç…ÇÕïKóvÇ-ÇÅB

Pedestal width?

D/T density ratio?

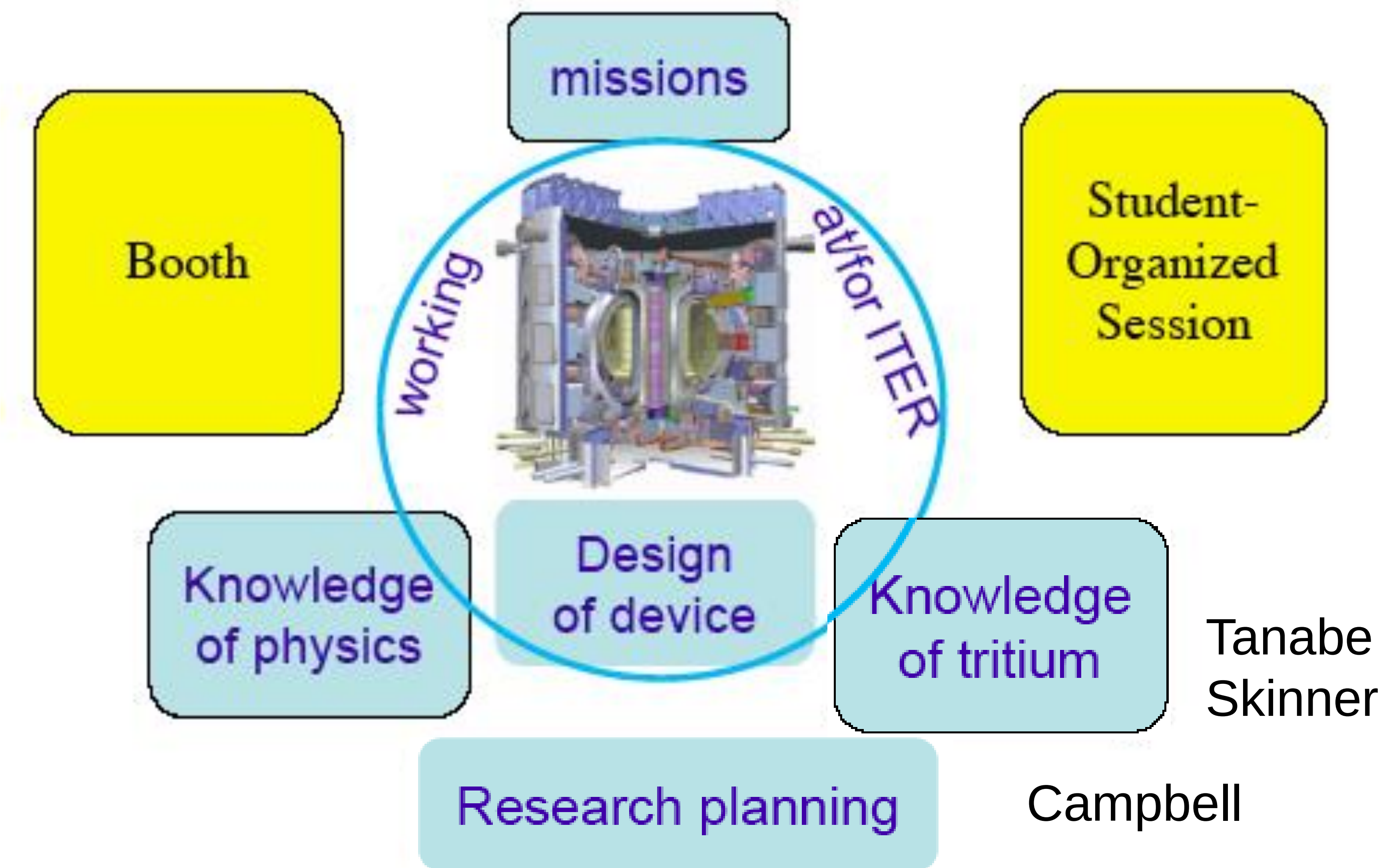
Transient response ?

Flow generation ?

Alfvenic and drift turbulence ?

Qualitative change in burning plasma ?

…….



missions
Booth
working
at/for ITER
Student-Organized Session
Knowledge of physics
Design of device
Knowledge of tritium
Tanabe
Skinner
Research planning
Campbell

Demand of understanding of Tritium in fusion

We do not have enough T; need T breeding

$D + T \rightarrow {}^4He\ (3.5MeV) + n(14.1MeV)$

plasma heating Energy and T breeding

- Overall breeding ratio is expected to be above ~1.1 (must)

Very hard to attain

Fusion Safety Issues (General)

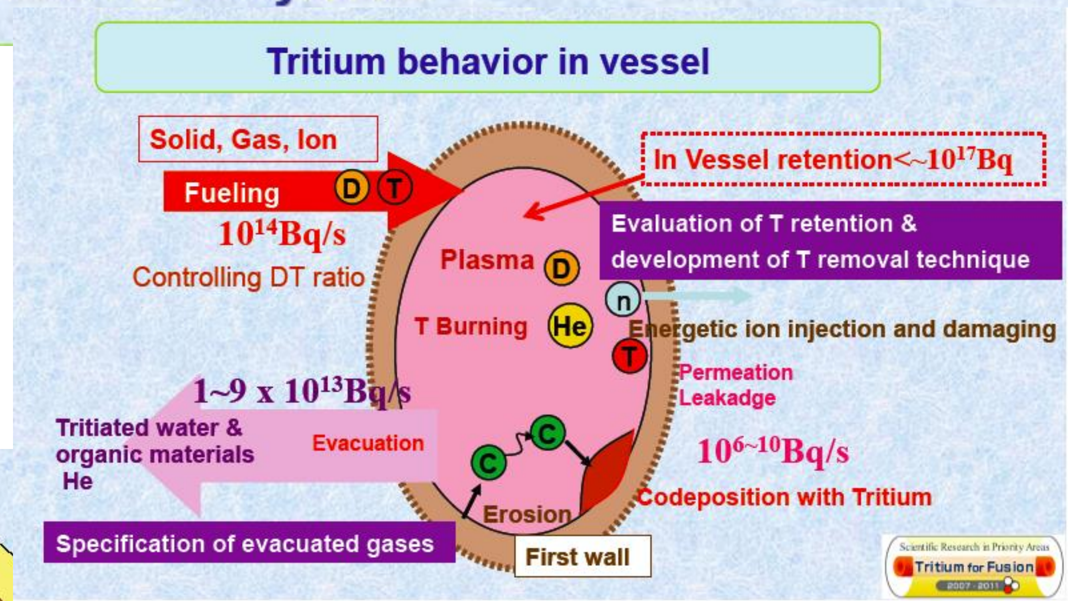
•Most serious hazard involve the tritium fuel and activated dust from erosion of plasma facing components

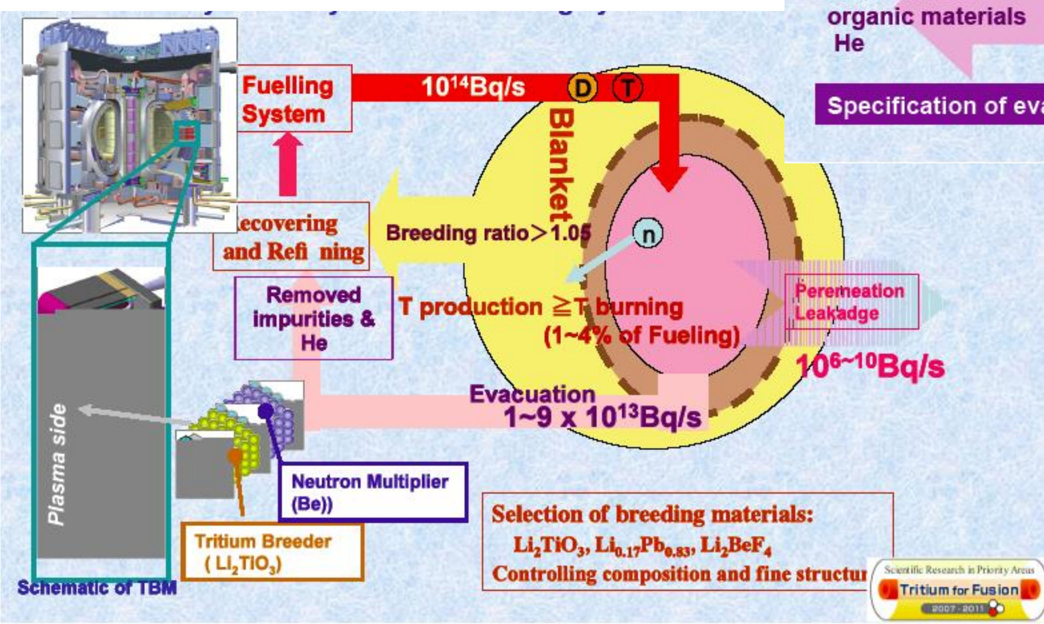
Where and in what form does Tritium go?

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

Issues and problems to be solved relating tritium – I

- Controlled fueling to keep continuous DT burning
- Tritium removal from in vessel components
- Tritium accountancy in tokamak system





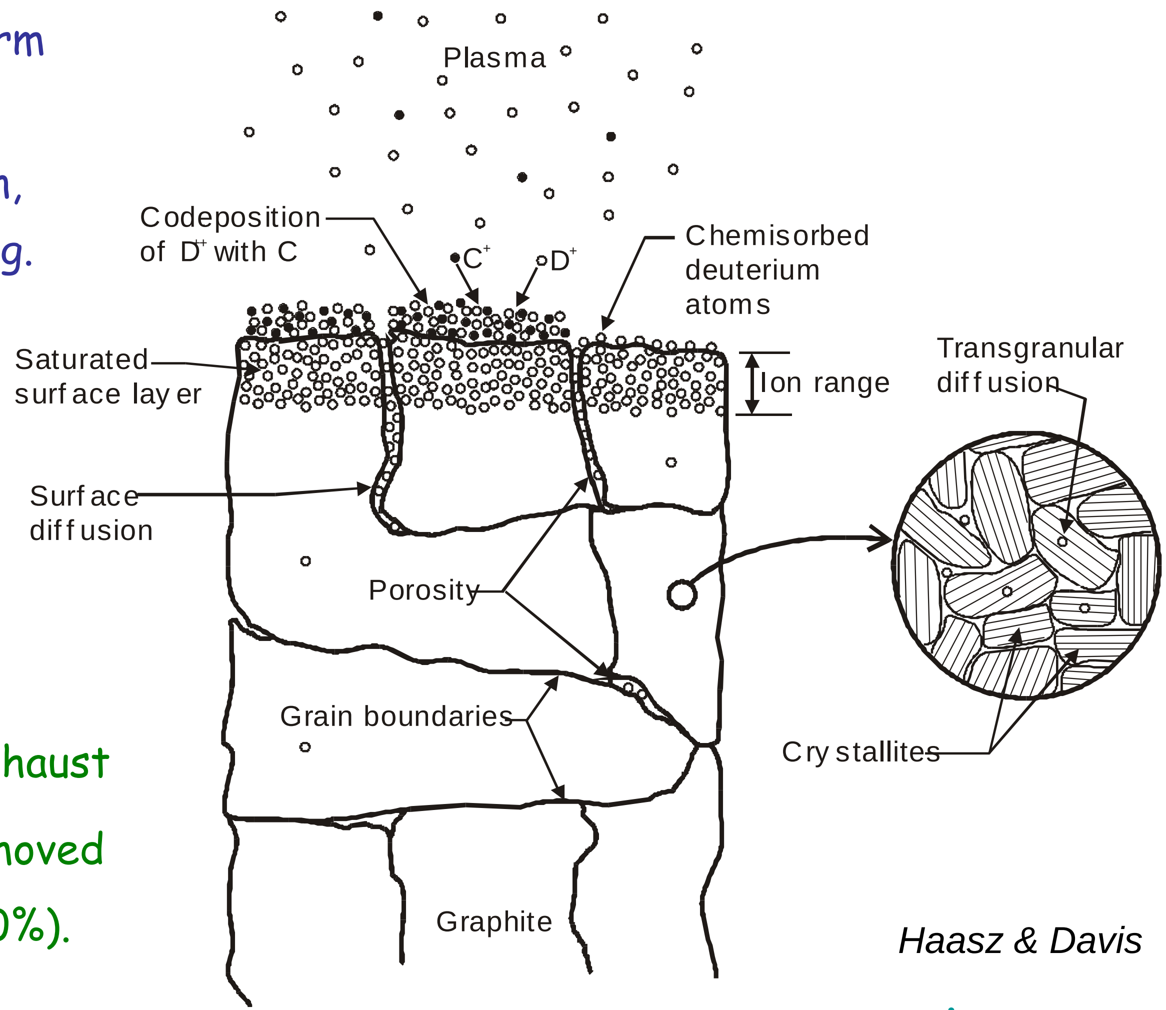
Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

Tritium retention

Basic mechanisms for retention

1. Short-term adsorption followed by outgassing (not a long-term problem).
2. Long-term deep implantation, diffusion, migration, trapping.
3. Long-term codeposition of tritium with plasma eroded materials e.g. C, Be.

Retention in graphite



Haasz & Davis

Two complementary methods to measure retention (R).

1. Gas balance, or fueling - exhaust (typically R≈ 10%-20%)
2. Analysis of components removed from vessel (typically R≈ 3% - 50%).

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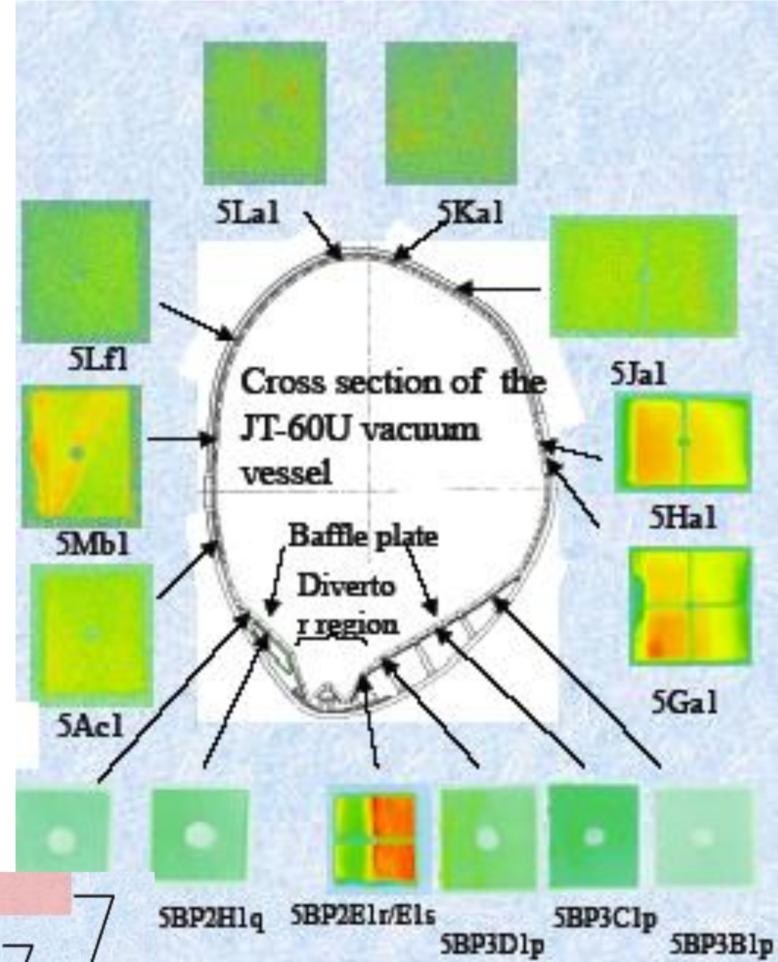
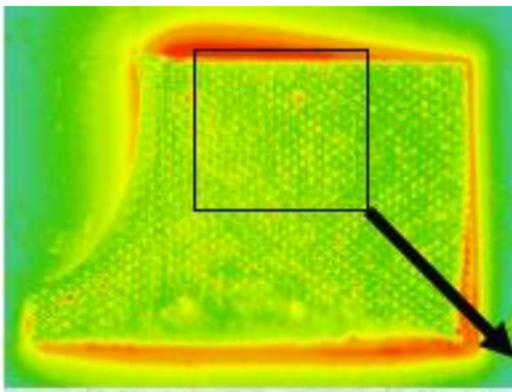
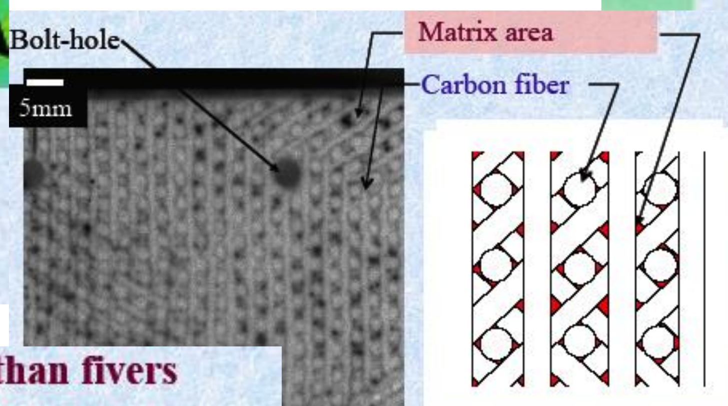
Multiple-scale structures of Tritium retention



…itium distribution by IP



Tritium image
(KC3 CFC tile)



(4-D CFC)

Matrix has higher retention than fivers

Different erosion yield between matrix and fibers produced non flat surface and codeposition with T on shadowed area

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

Estimation of in-vessel tritium retention includes very large error and uncertainty

- Evaluation of **hydrogenic retention** in present tokamaks is of high priority to establish a **database** and **a reference for ITER** (400 s …usually 10-20 s today).

- T retention constitutes an **outstanding** problem for ITER operation particularly for the **choice of the materials** (carbon ?)

- A **retention rate of 10%** of the T injected in ITER would lead to the in-vessel T-limit (350/700g) in **~35/70 pulses**. (every ~ **35/70** shots require removing of in vessel T?)

- Retention rates of this order **or higher** (~20%) are regularly found using **gas balance**.

- Retention rate **often lower** (3-4%) are obtained using **post mortem analysis**

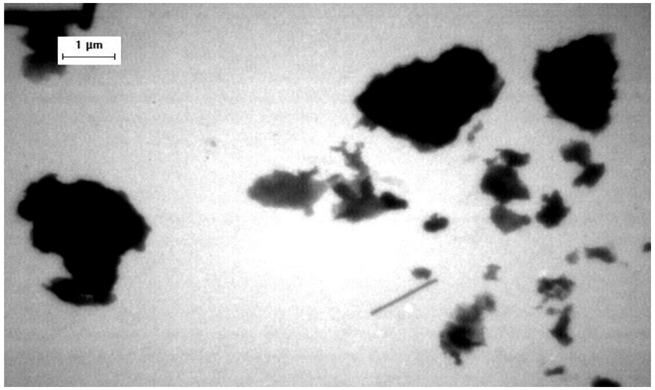
Tritiated dust more hazardous than HTO

- Tritiated dust obtained from TFTR
- Size analysis showed it is respirable
  - diameter = 1.2 μm
  - *In-vitro* dissolution rate measured
    in simulated lung fluid.

Result:

- Only 8% of carbon tritide was dissolved after 110 days.
- Low solubility means tritium will remain for long time increasing radiation dose to lung.

- Data needed on a:BeT dust to determine allowable exposure !



Tritiated dust levitation by beta induced static charge

Tritium activity

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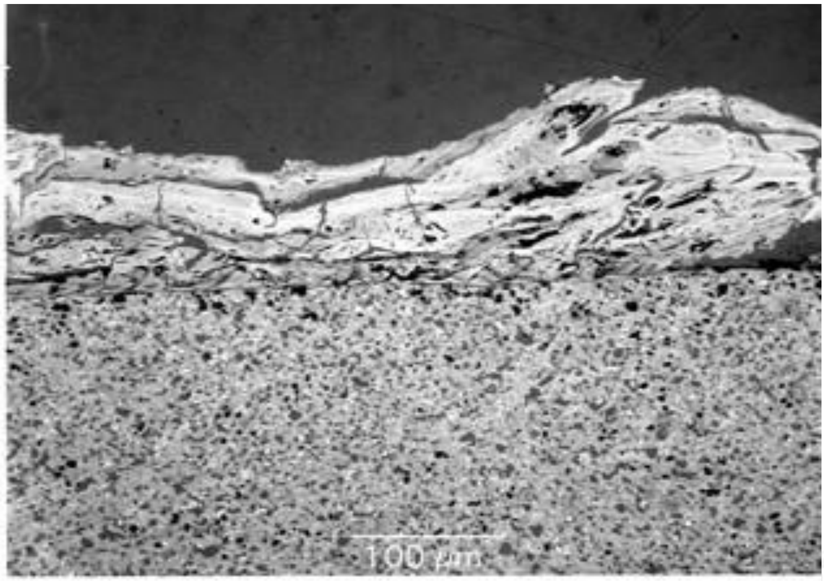
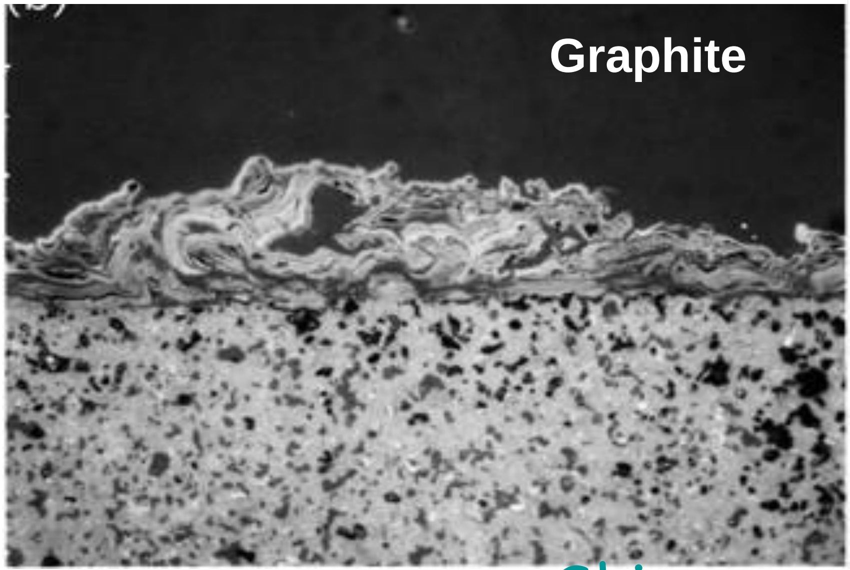
Cheng et al., Fus. Technol., 41 (2002) 867

TFTR codeposits containing tritium





TFTR tile samples impregnated with epoxy, polished and viewed in a metallurgical microscope.

Remarkably convoluted structure with distinct strata and voids that reflect the discharge history.

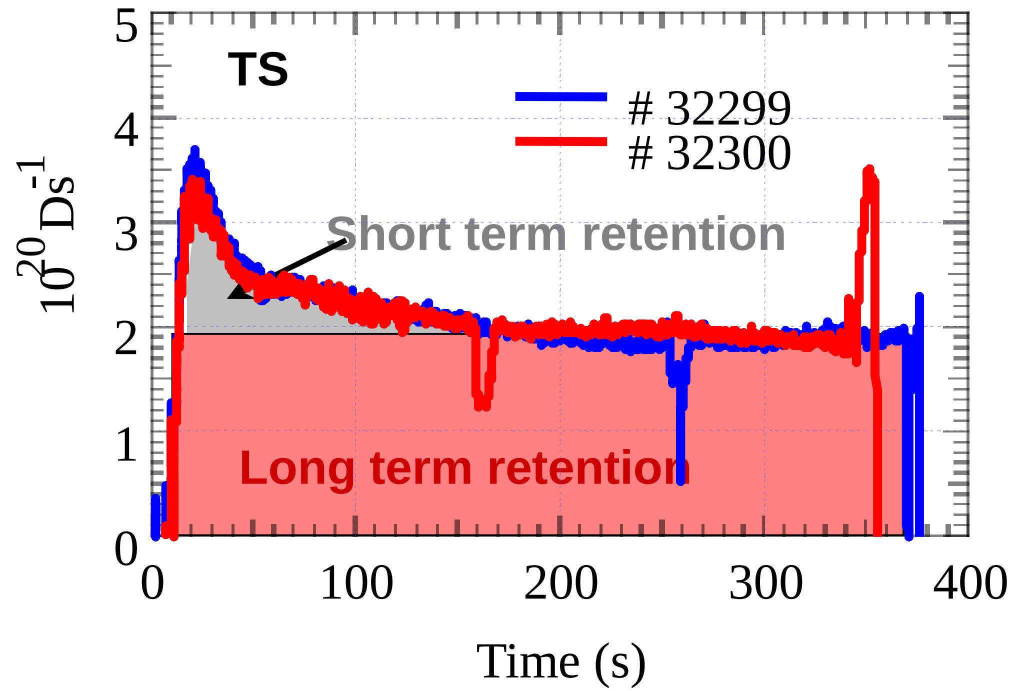
C. H. Skinner et al., Phys. Scripta T 103, 34-37 (2003)

R. Reiswig, S. Willms LANL

Skinner

Tore Supra experience

Long pulse, actively cooled circular tokamak with carbon tiles



Carbon plasma facing components -> continuous increase of tritium inventory with plasma duration via codeposition

Encouraging results with metals at Asdex-U

Surprising results from C-mod with Mo, W

JET ITER-like wall will get experience with Be/W tiles as envisaged for ITER DT experiments

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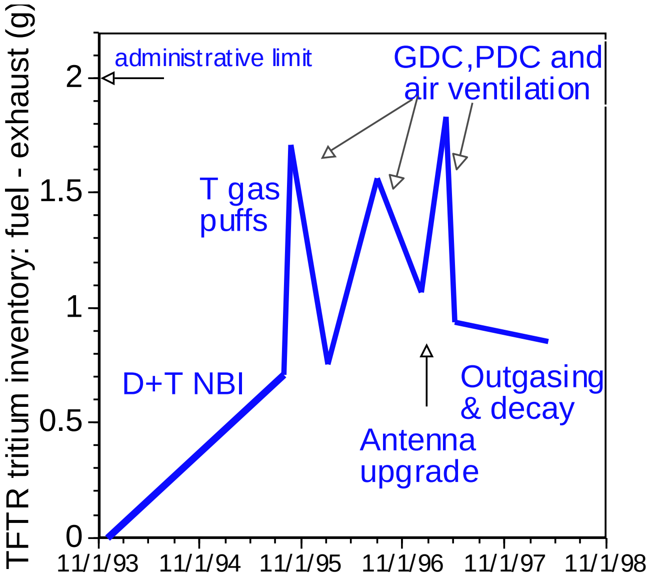
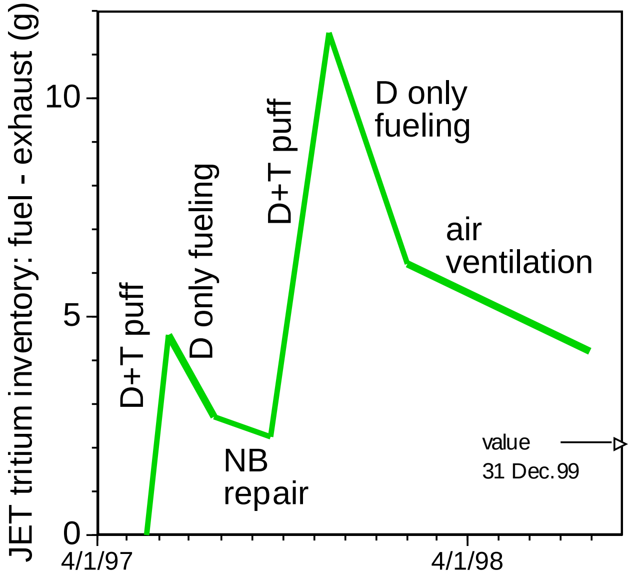
Tritium retention high in TFTR and JET

In TFTR 5 g of tritium were injected into circular plasmas over a 3.5 year period, mostly by neutral beam injection.





In JET 35 g of tritium were injected mostly by gas puffing over a 6 month campaign.

Global Retention:	TFTR:	JET:
Total tritium injected, NBI	3.1 g	0.6 g
gas puff	2.1 g	34.4 g
Total tritium retained during DT operations	2.6 g	11.5 g
Initial % retention during T puff fueling (wall saturation + isotope exchange)	- 90%	- 40%
Longer term % retention including D only fueling (mostly co-deposition)	51%	17%
Tritium remaining in torus	0.85 g (4/98)	4.2 g (7/98)
Long term retention	16% (4/98)	12% (7/98) 6% (12/99)

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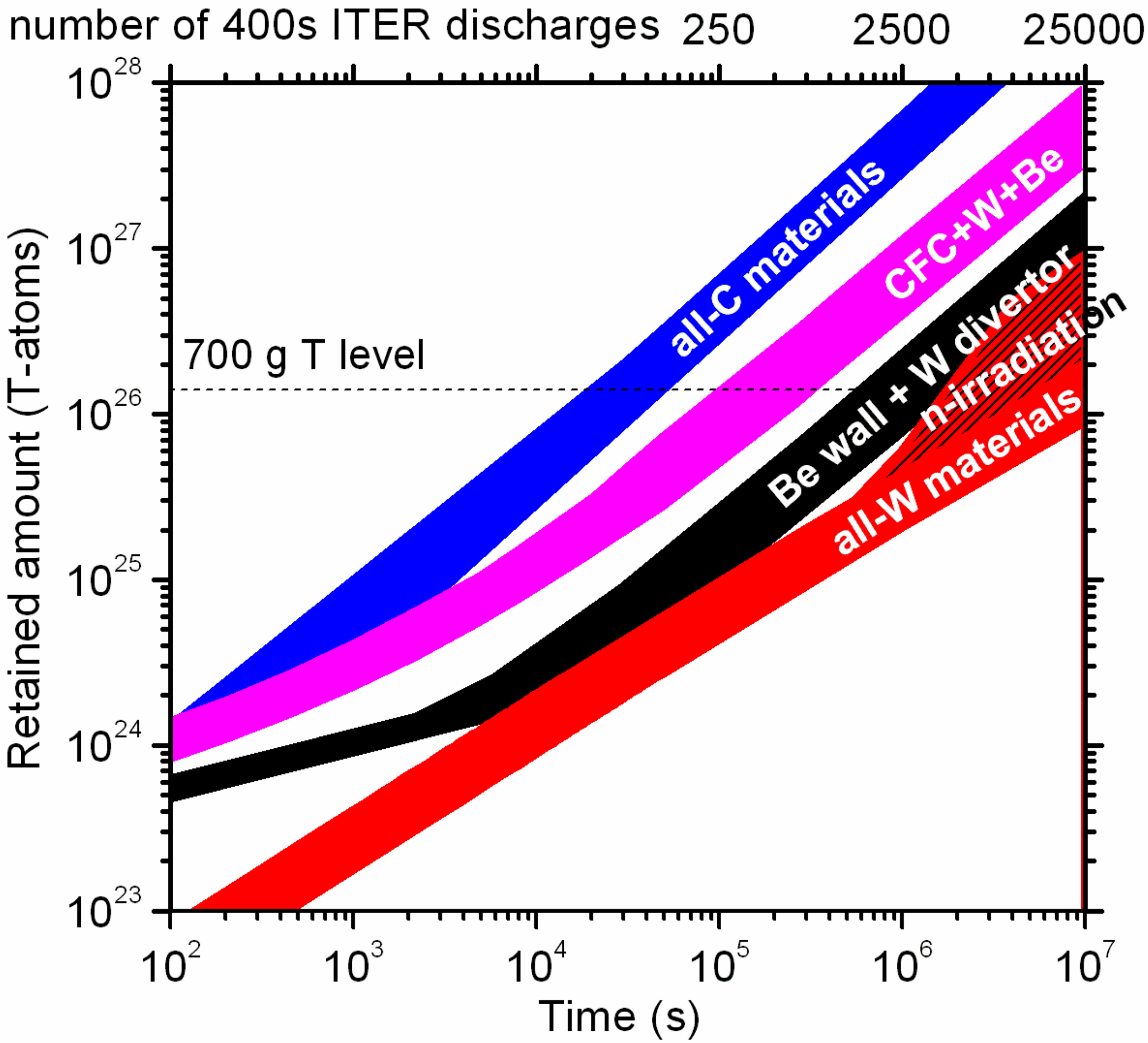
Tritium retention and removal rate in TFTR and JET unacceptable for ITER

Implantation + codeposition

Recent EU assessment of tritium inventory in ITER for various PFC material options
(to appear in PPFC)

Similar, independent plot by ITPA SOL/Div group (to appear in 2008 IAEA proceedings).



Roth PSI18

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Planning DT Experiments

Progress from DD to DT experiments is a major (and exciting!) step for the magnetic fusion programme

- DT fuel brings a new approach to the organization of the tokamak experimental programme:
  - Tritium is itself radioactive
  - Limited amounts of tritium are stored on-site to limit licensing requirements
  - Amount of tritium trapped inside vacuum vessel must be limited
  - DT fusion reactivity factor of >100 greater than DD reactivity
  - 14 MeV neutrons vs 2.4 MeV neutrons $\Rightarrow$ additional activation products

$\Rightarrow$ experimental programme must be planned with great care to minimize use of tritium and activation of the device structure

$\Rightarrow$ rehearsal of plasma scenarios in deuterium and careful development to optimize use of tritium

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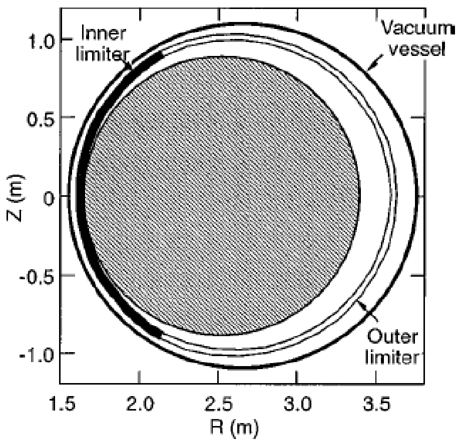
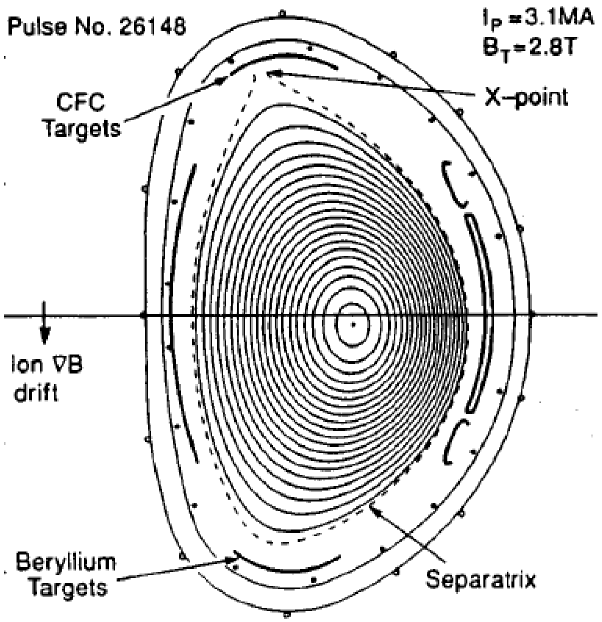
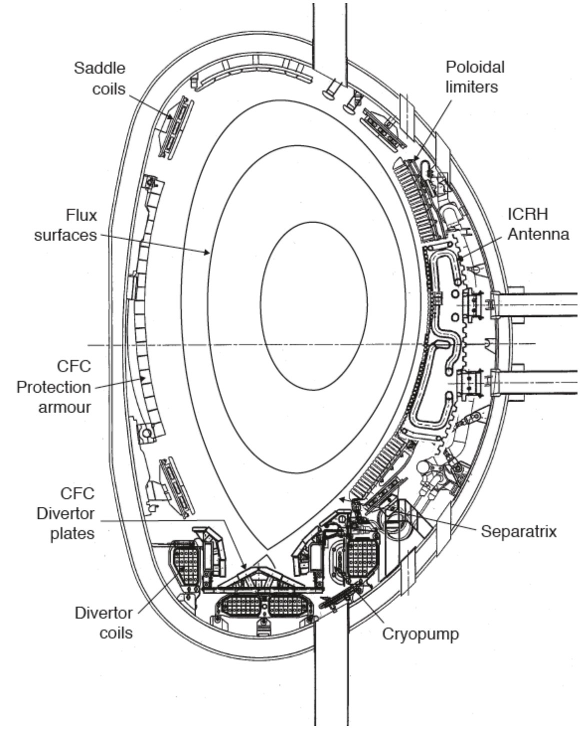
Experimental Configurations







TFTR 1994-1997 JET 1991 JET 1997, 2003

- TFTR and JET use different magnetic configurations:
  - TFTR DT experiments in limiter plasmas: L-mode, "supershot", ITB scenarios
  - JET DT experiments in diverted plasmas: L-mode, H-mode, ITB scenarios

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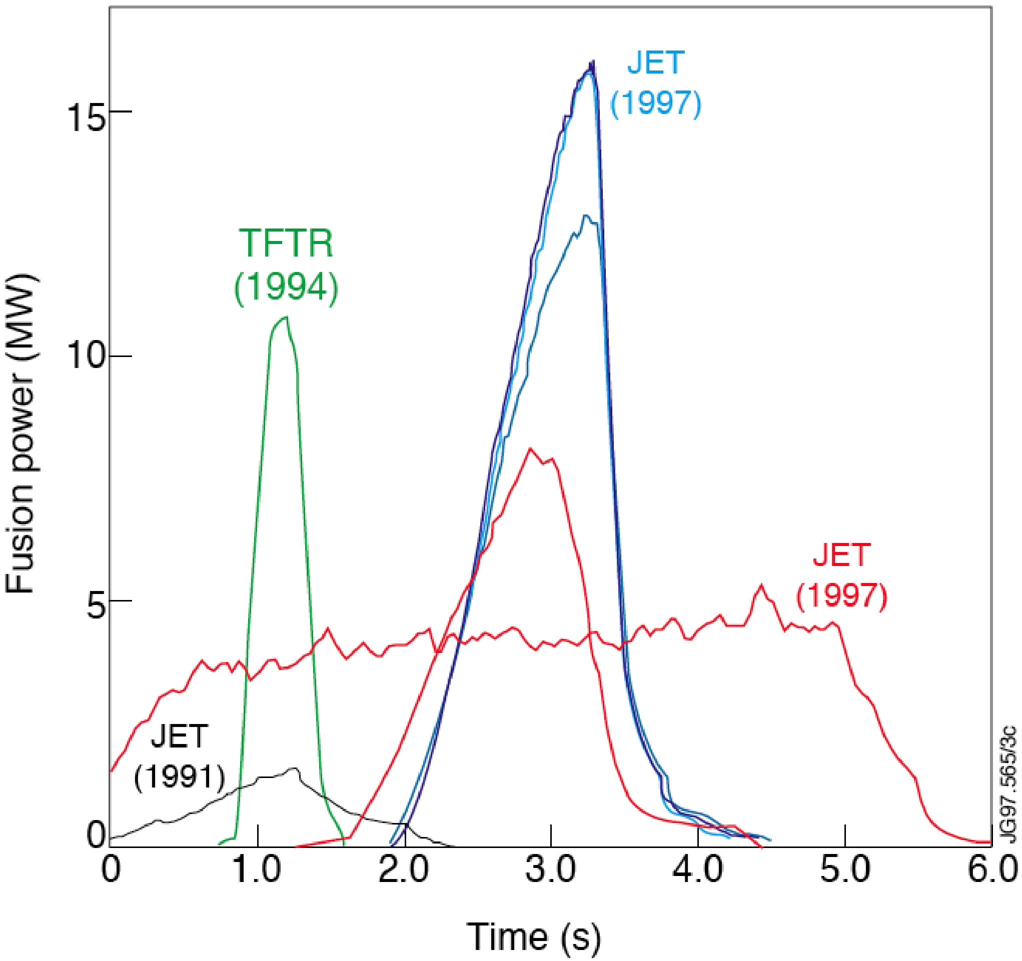
Fusion Power Production: Overview

- Summary of best fusion power performance achieved in DT experiments in JET and TFTR



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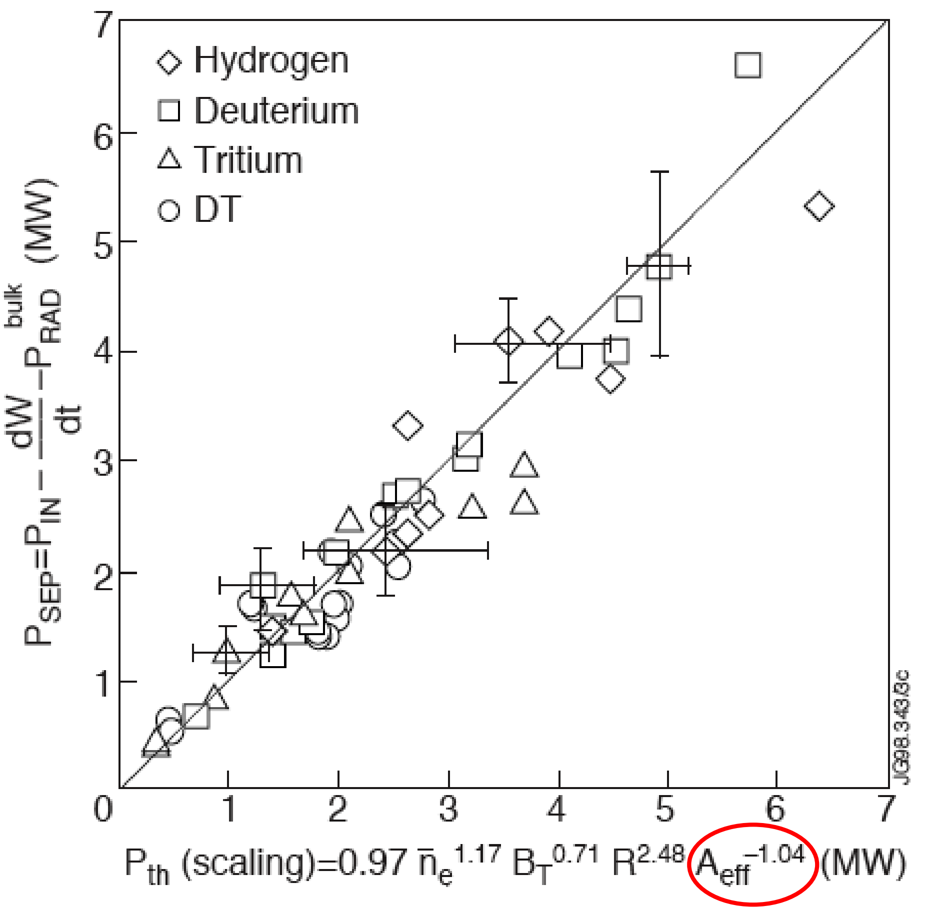
H-mode Power Threshold

- JET analysis of the power required to access the H-mode confirmed that:

$$P_{th} \propto \langle A \rangle^{-1}$$

- This result is important for ITER in that it indicates that access to the H-mode will be easiest in DT operation



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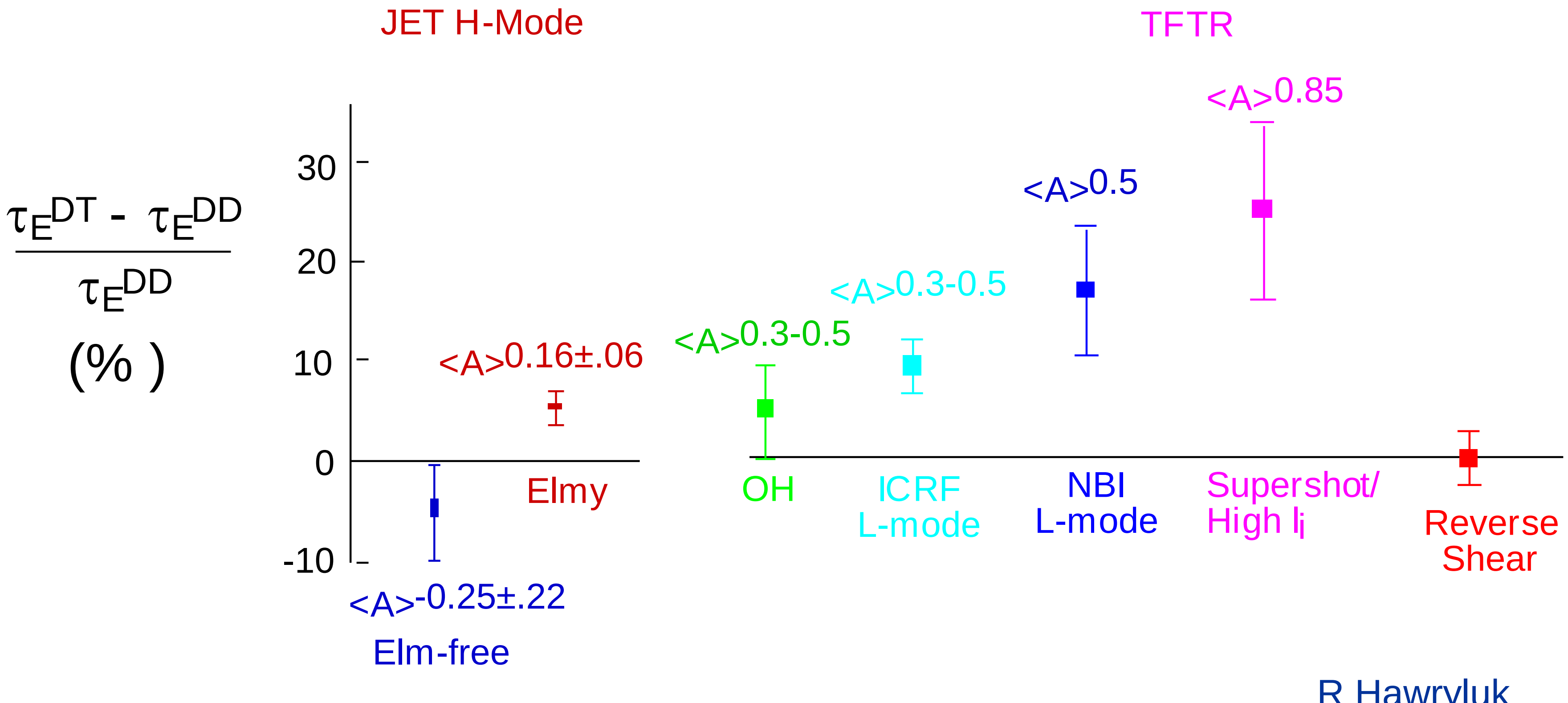
Plasma Energy Confinement: Overview



- **Overall, the isotope dependence of confinement has been found to vary widely, depending on plasma operating regime:**
  - indicates different processes influencing confinement and their varying importance in different plasma regimes

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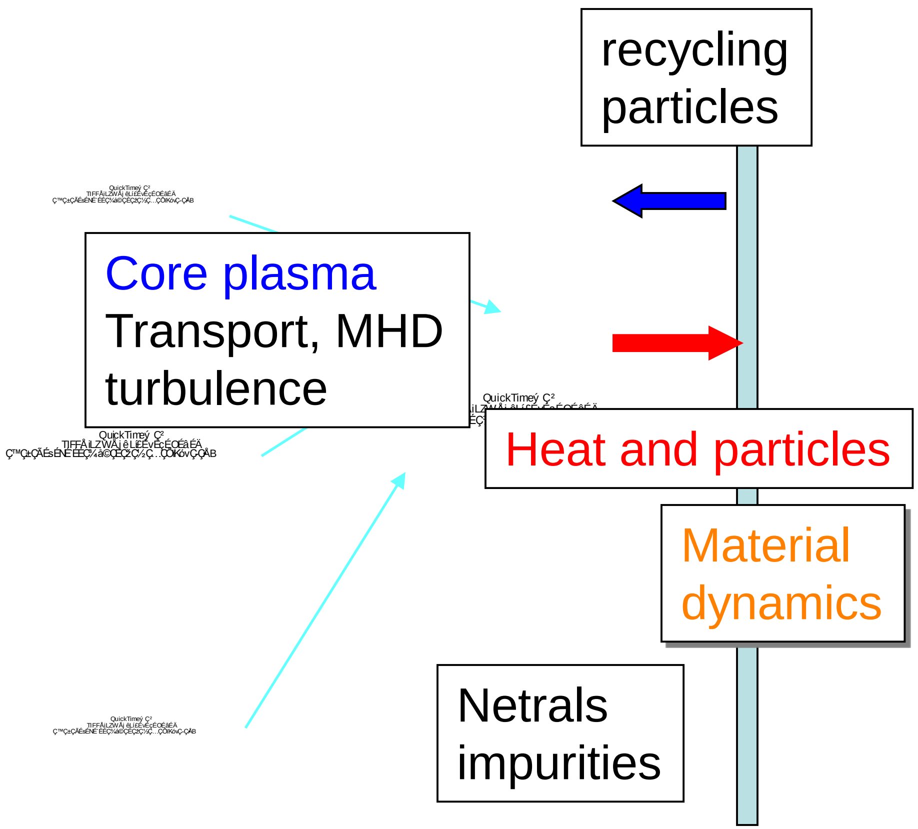
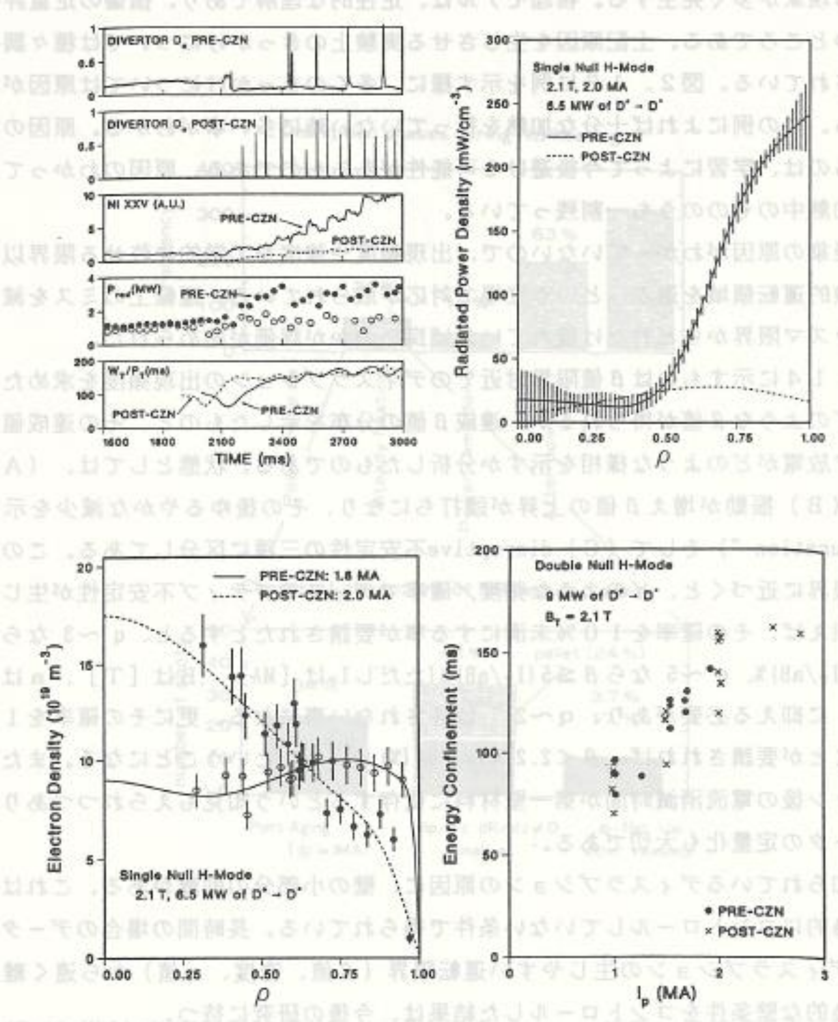
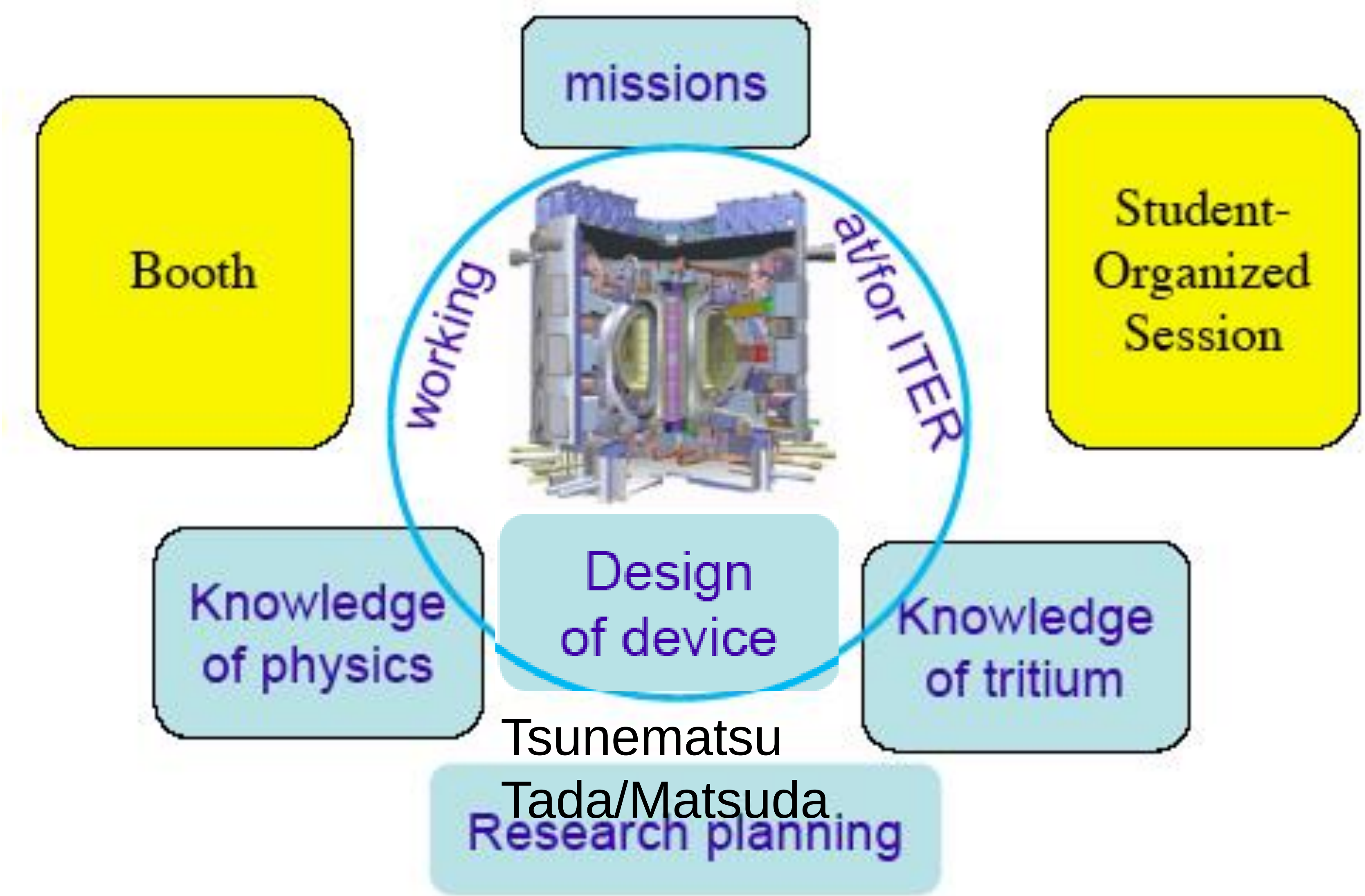
Collision of New Frontiers: Opportunities for Innovations



H-modes in C-wall and that in metal wall are different

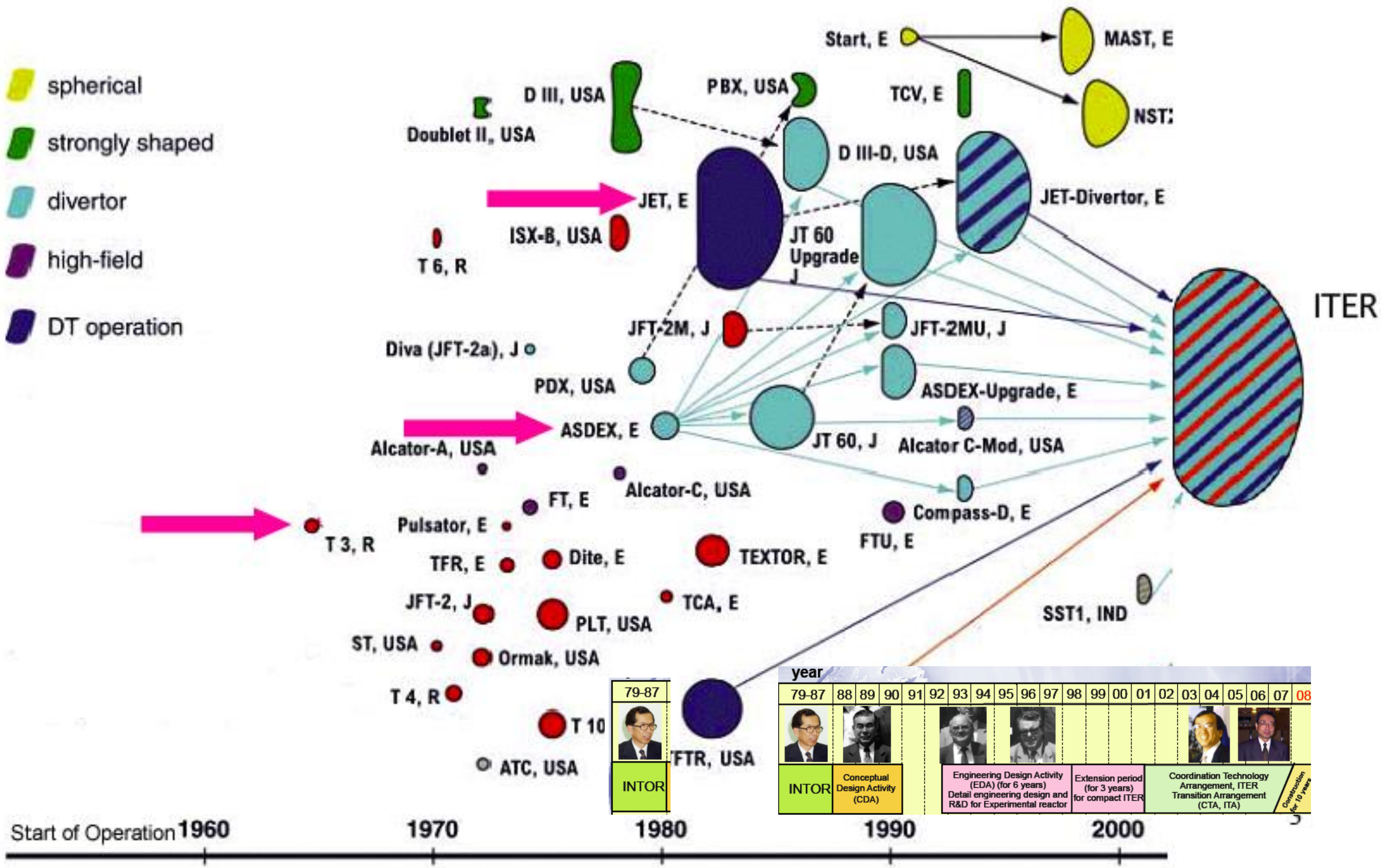


Courtesy: M. Yagi

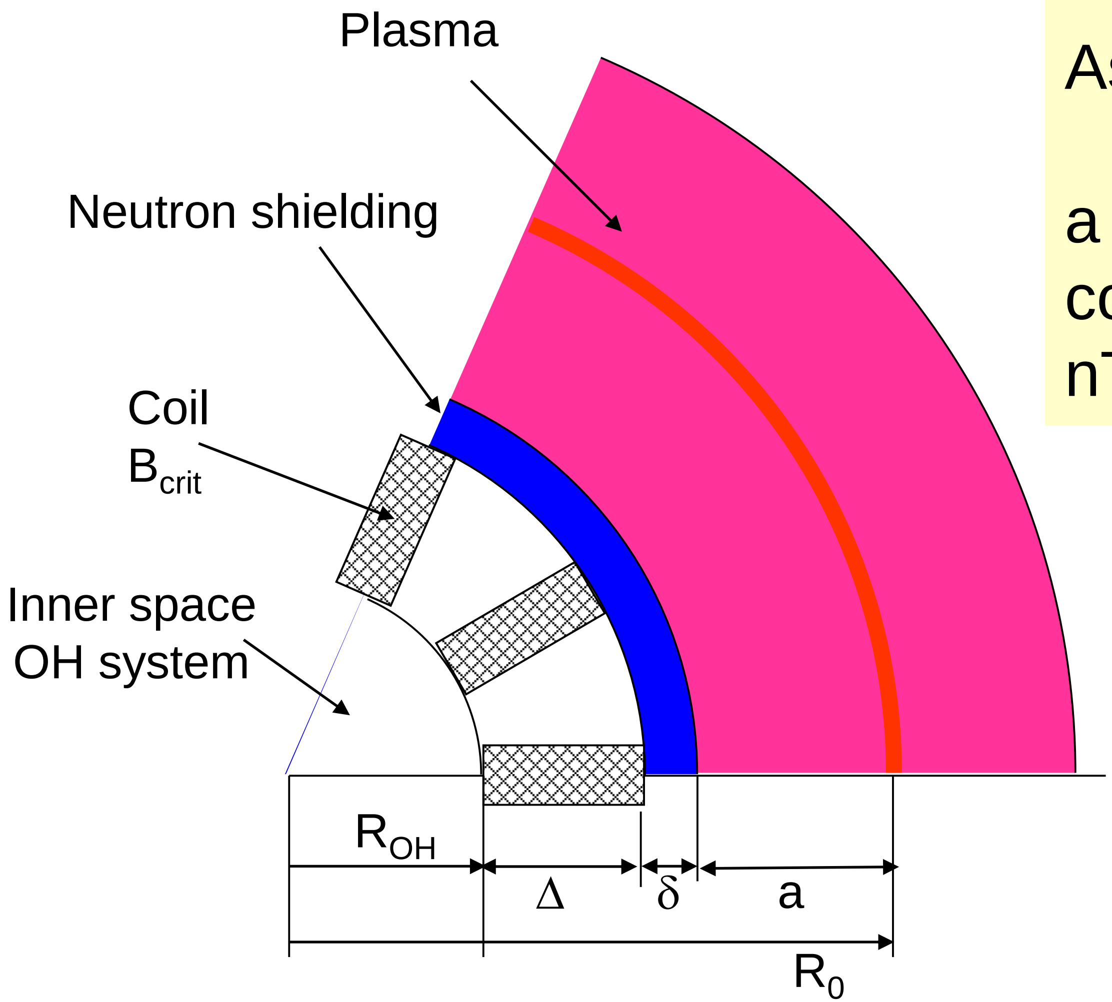


missions
Booth
Student-Organized Session
working
at/for ITER
Design of device
Knowledge of physics
Knowledge of tritium
Tsunematsu
Tada/Matsuda
Research planning

Interaction between physics basis and design



The size of ITER



Δ ~ 2 m; δ ~ 1.3 m

Aspect ratio: $A = R_0/a$

a determined by confinement to meet $nT\tau_E$ goal

Benefit of improved confinement

The importance of improved confinement:

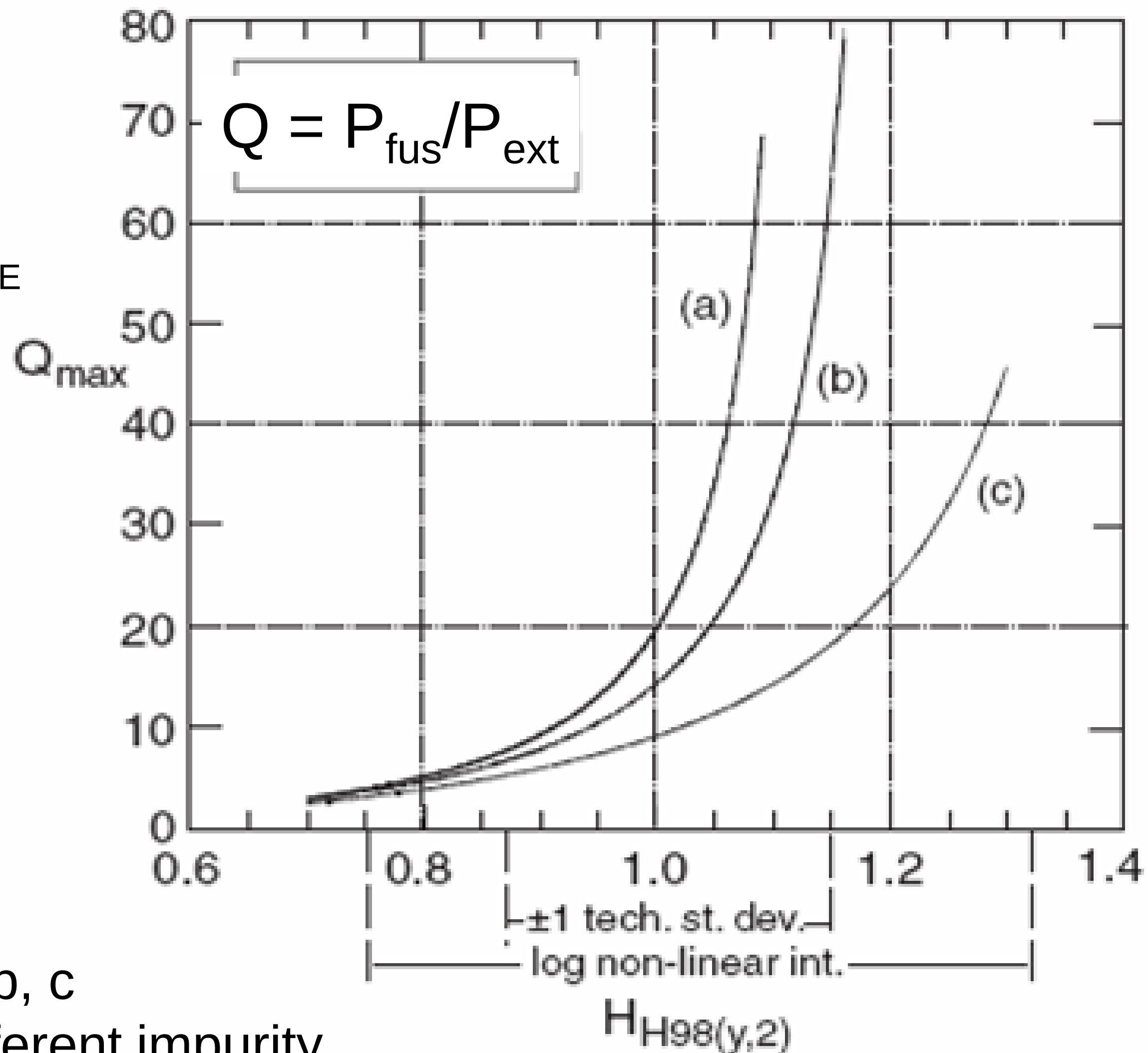
Improvement factor: τ_E => $H\tau_E$

Ignition:

$$\frac{\langle p \rangle \tau_E}{a^2 B_t^2} \sim H^2$$

Triple product:

$$nT\tau_E \propto H^2$$

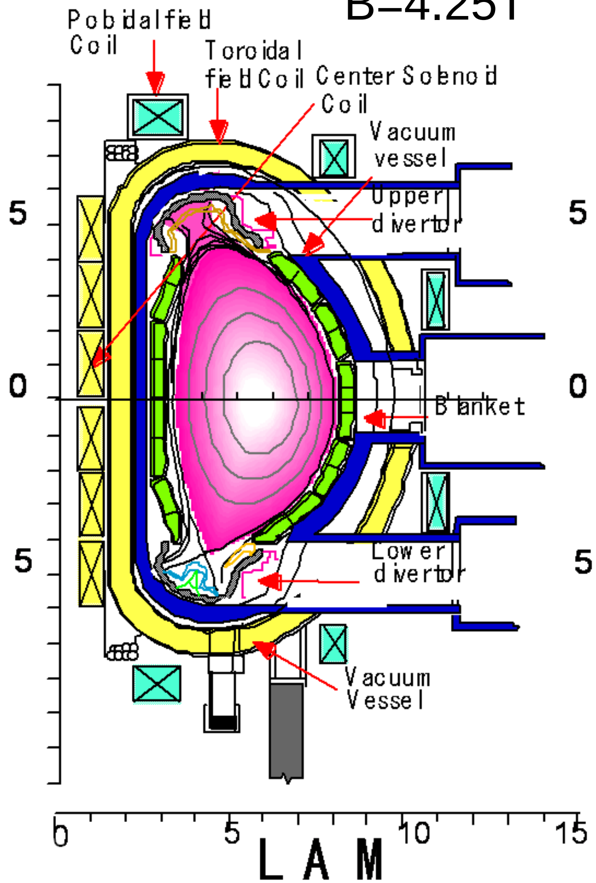


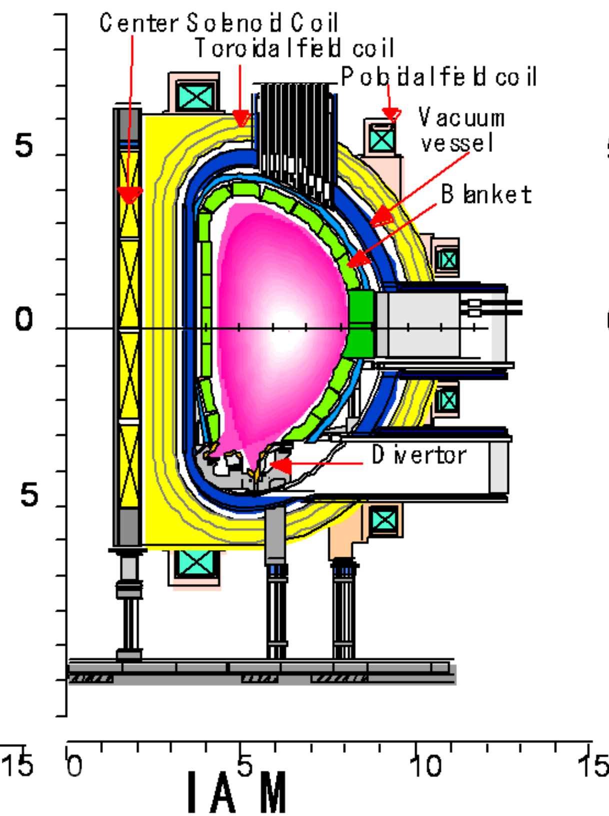
a, b, c
different impurity
confinement

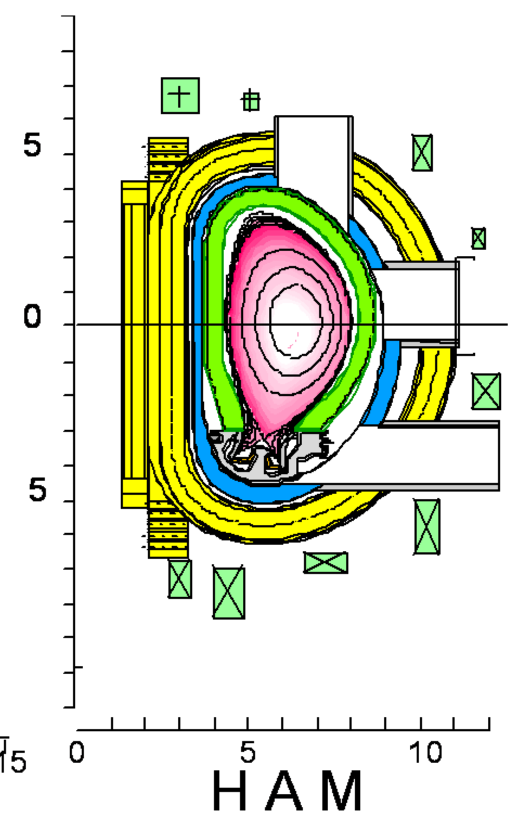
V. Mukhovatov

Selection of ITER Design

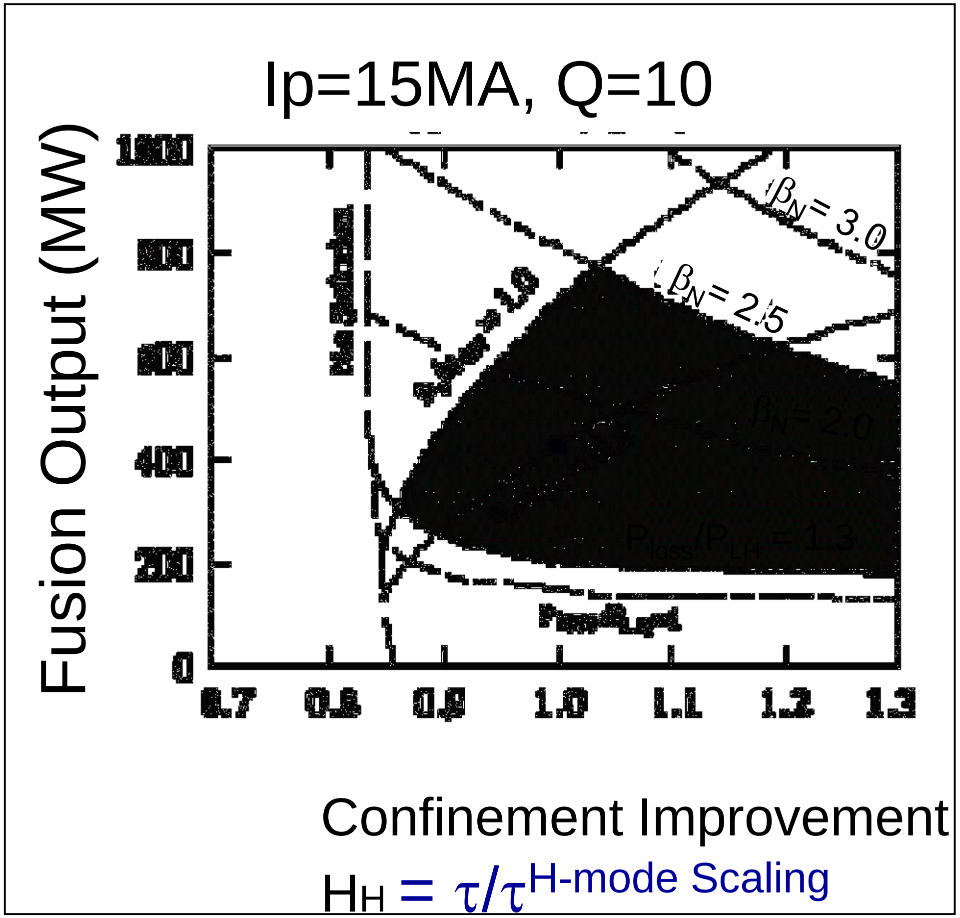
Major Radius : 6.45 m Minor Radius : 2.33 m	Major Radius : 6.2 m Minor Radius : 1.9 m	Major Radius : 6.3 m Minor Radius : 1.8 m
B=4.25T	B=5.4T	B=6.58T







Operation Space for Q=10



-density limit
<Greenwald density

-normalized β
<2.5

-access to ELMy H-mode
$P_{loss} > P_{LH}$ threshold power

$$P_{LH} = 0.042 n_{20}^{0.73} B_t^{0.74} S^{0.98} (MW)$$

~10% margin in confinement improvement

ITER retention depends on material choice

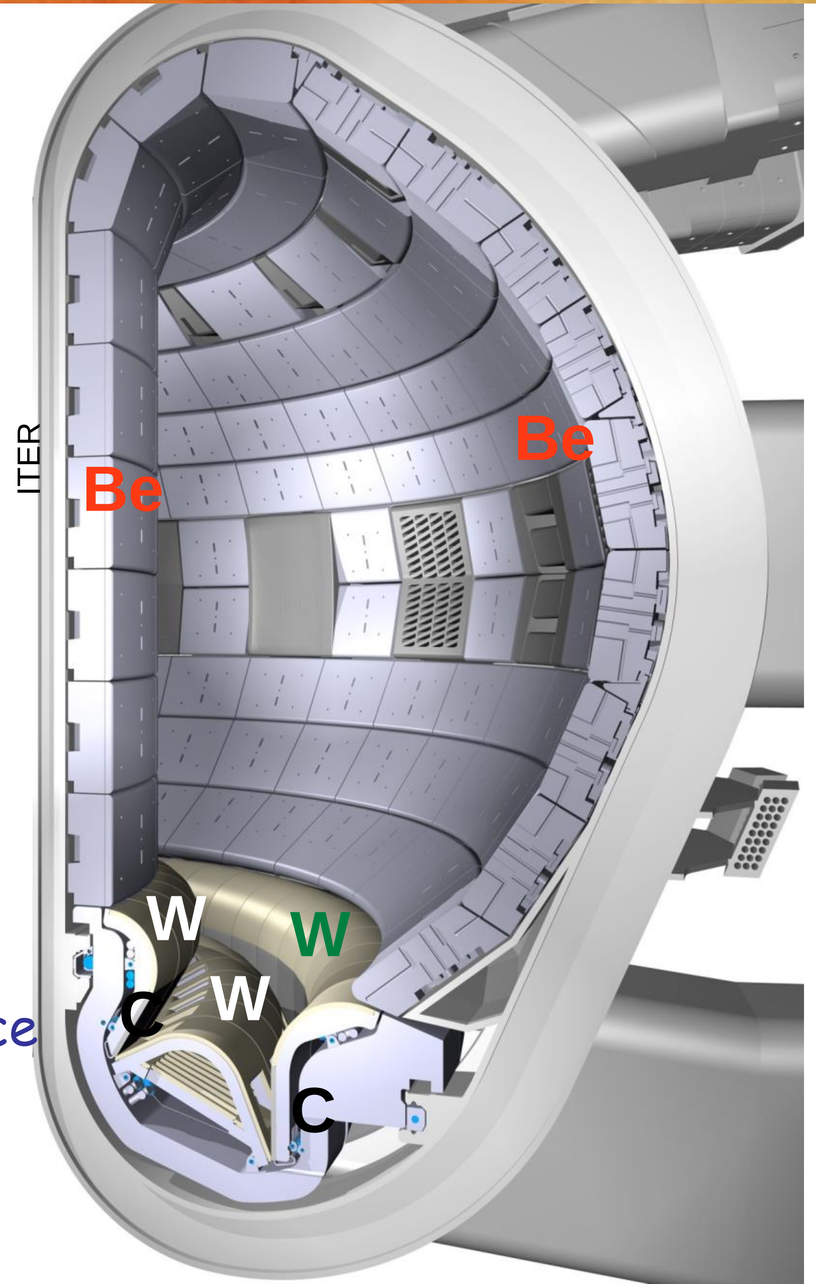
Present ITER strategy:

Initial hydrogen/deuterium phase:

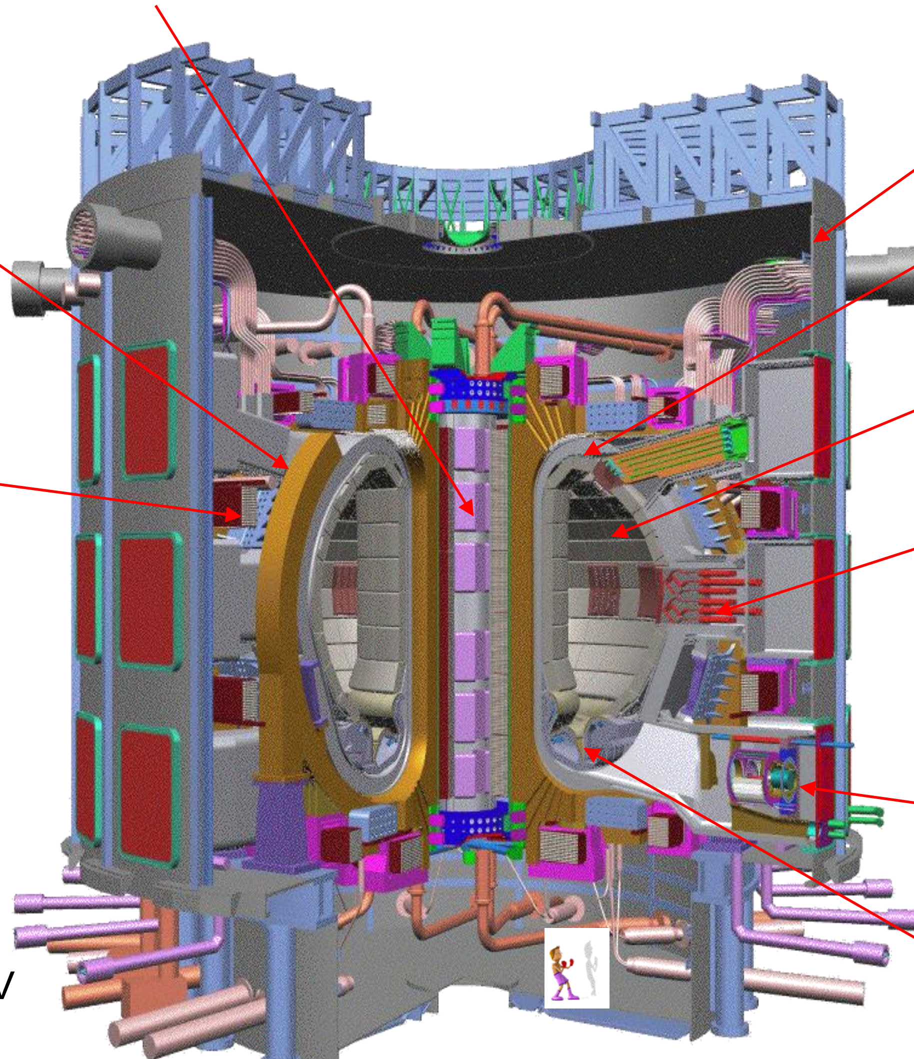
- Beryllium wall, 700 m^2
  (low Z = low radiation losses, oxygen getter,
  but low melt temperature)
- Tungsten baffle and dome, 100 m^2
  (high melt temp, low erosion, low T retention, but high rad. losses)
- Carbon divertor target 50 m^2
  (does not melt, good radiator for plasma detachment, but T retention is major issue)

Before DT operation

- Change to full tungsten divertor.
- Timing depends on experience with H retention and dust
- All-W as future DEMO relevant choice



The core of ITER



Major plasma radius 6.2 m

Plasma Volume: 840 m^3

Plasma Current: 15 MA

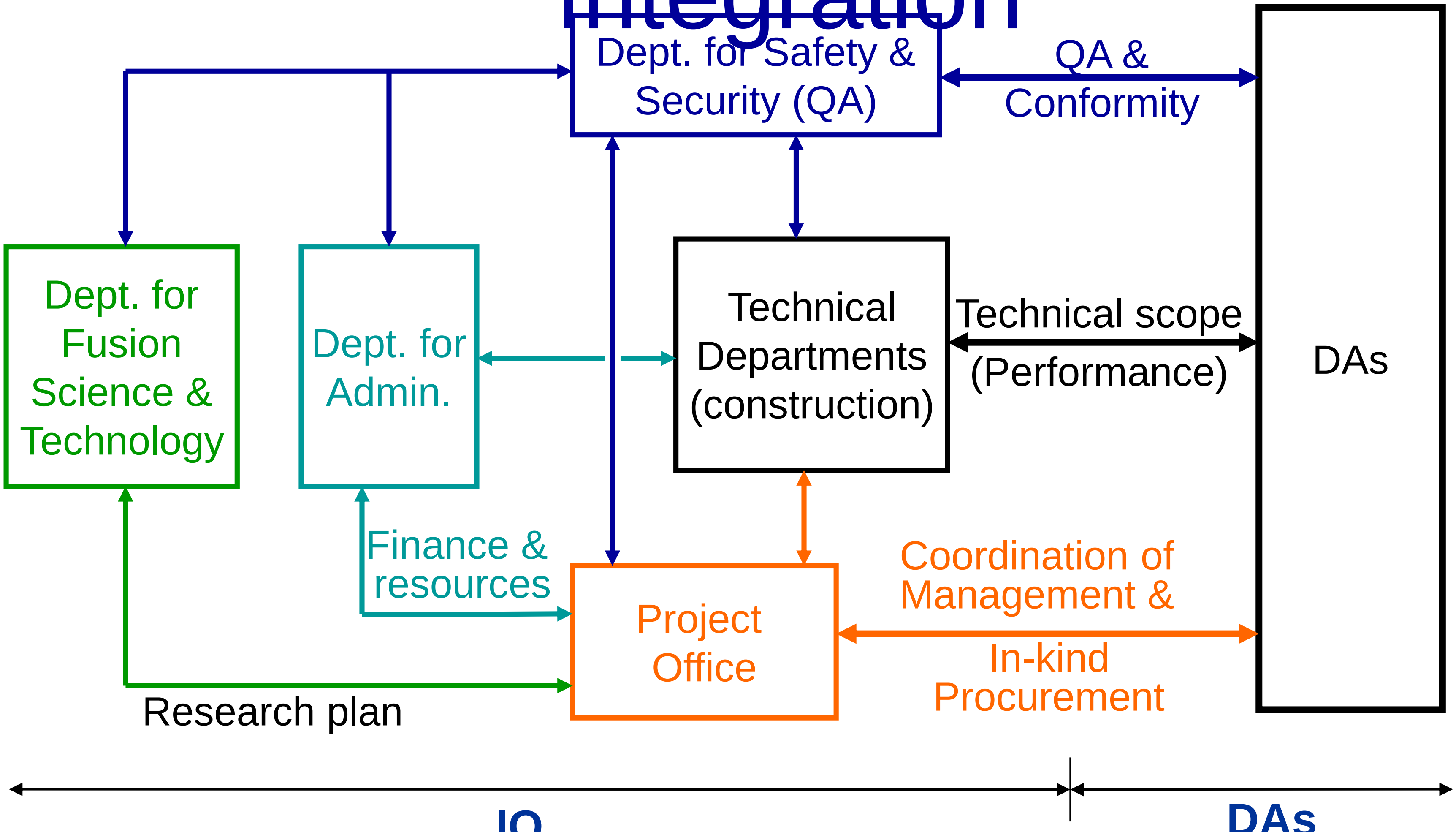
Typical Density: 10^{20} m^{-3}

Typical Temperature: 20 keV

Fusion Power: 500 MW

Machine mass: 23350 t (cryostat + VV + magnets)
- shielding, divertor and manifolds: 7945 t + 1060 port plugs
- magnet systems: 10150 t; cryostat: 820 t

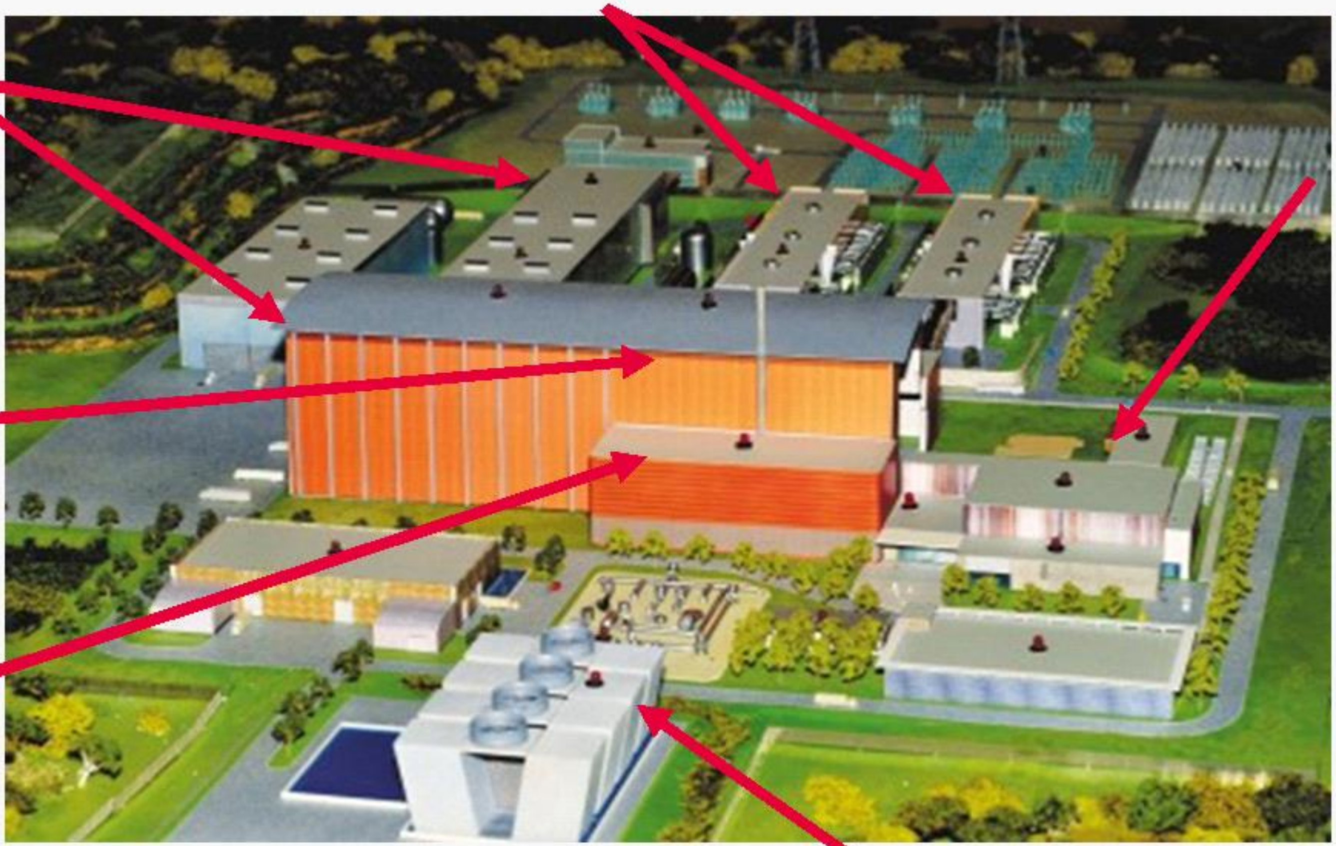
Overall Work Relationship for System Integration



System Interface Control among systems
Procurement Interface Control among DAs

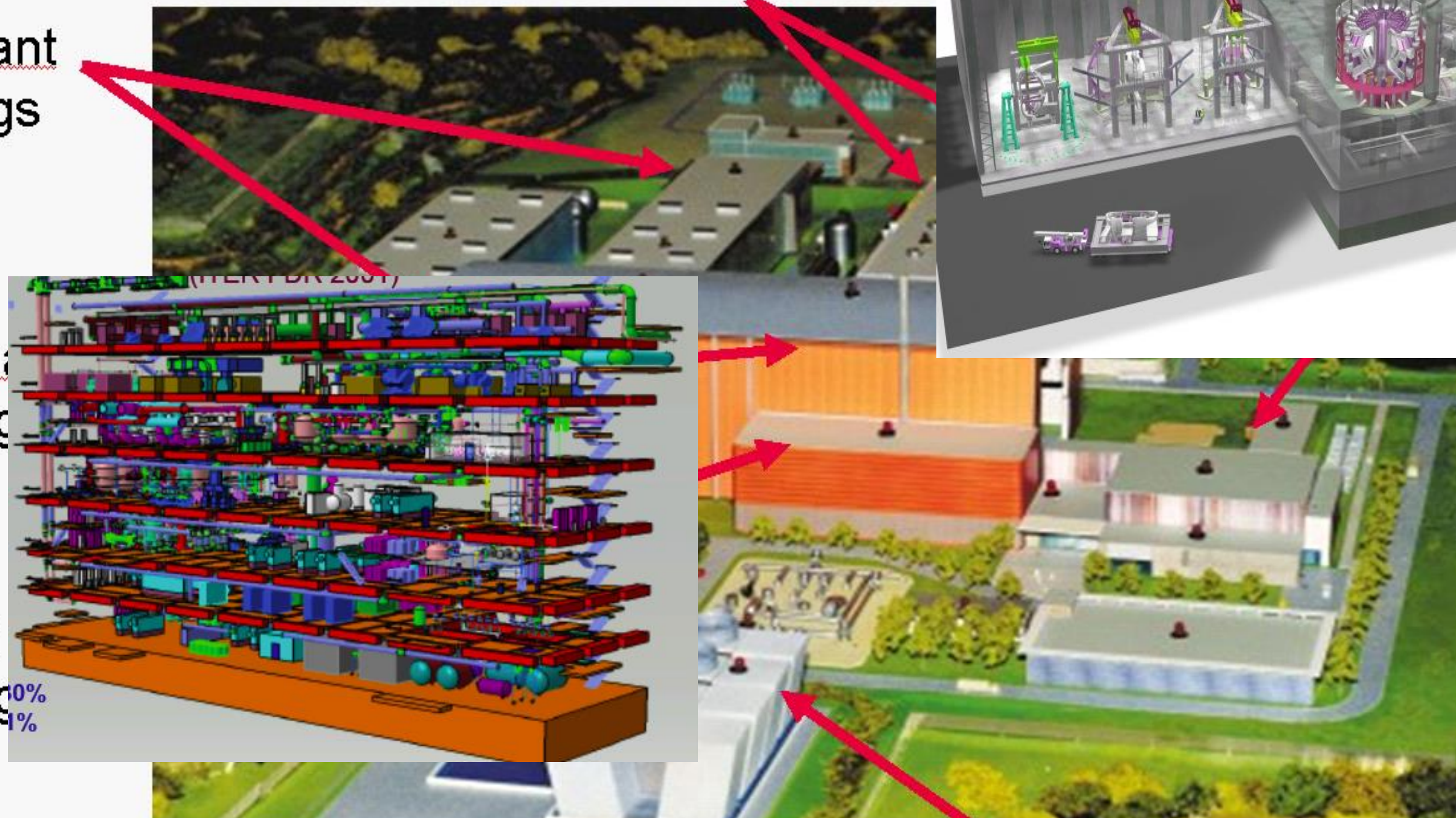
PBS		11	15	16	17	18	22	23	24	26	27	31	32	34	41	43	45	46	51	52	53	54	55	56	61	62	63	64	65
Magnets	11		•	•	•		•	•	•		[illegible]	•	•	•	•		•	•					•		•	•			
Vacuum Vessel	15	•		•	•	•	•	•	•	•	•	•	•	•			•		•	•	•	•	•	•	•	•		•	
Blanket systems	16	•	•		•	•	•	•		•		•	•						•	•	•	•	•	•		•		•	
Divertor	17	•	•	•		•	•	•		•		•	•										•			•		•	
Fuelling & wall conditioning	18		•	•	•		•	•	•	•	•	•	•	•		•	•	•			•					•			•
Machine Assembly & tooling & installation	22	•	•	•	•	•		•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•		•
Remote Handling equipment	23	•	•	•	•	•	•		•	•	•	•	•			•	•	•	•	•	•	•	•	•	•	•		•	•
Cryostat	24	•	•			•	•	•		•	•	•	•	•		•	•	•					•			•			
Cooling water system	26		•	•	•	•	•	•	•			•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•		•
Thermal shield	27	•	•			•	•	•	•			•		•			•	•					•			•			
Vacuum	31	•	•	•	•	•	•	•	•	•	•		•	•		•	•	•	•	•	•	•	•	•		•			•
Tritium plant	32	•	•	•	•	•	•	•	•	•		•				•	•	•					•			•		•	•
Cryoplant & cryodistribution	34	•	•			•	•		•	•	•	•			•		•	•							•	•	•		
Coil power supplies & distribution	41	•					•			•				•		•	•	•							•	•	•		
Steady state electrical power network	43					•	•	•	•	•		•	•		•		•	•	•	•	•	•	•	•	•	•	•	•	•
CODAC	45	•	•			•	•	•	•	•	•	•	•	•	•	•			•	•	•	•	•	•	•	•	•	•	•
Safety & interlock systems	46	•				•	•	•	•	•	•	•	•	•	•	•			•	•	•	•	•	•	•	•	•	•	•
Ion cyclotron H&CD system	51		•	•			•	•		•		•				•	•	•							•	•			
Electron cyclotron H&CD system	52		•	•			•	•		•		•				•	•	•							•	•			
Neutral Beam H&CD system	53		•	•		•	•	•		•		•				•	•	•					•		•	•	•		
Lower Hybrid H&CD system*	54		•	•			•	•		•		•				•	•	•							•	•			
Diagnostics	55	•	•	•	•		•	•	•	•	•	•	•			•	•	•			•					•			
Test blankets	56		•	•			•	•		•		•				•	•	•								•			•
Site	61	•	•				•	•		•				•	•	•	•	•	•	•	•	•				•	•		•
Reinforced concrete buildings	62	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•	•			•	•
Steel frame buildings	63						•			•				•	•	•	•	•			•				•				•
Radiological protections	64		•	•	•			•					•			•	•	•								•			
Liquid and gas distribution	65					•	•	•		•		•	•			•	•	•						•	•	•	•		

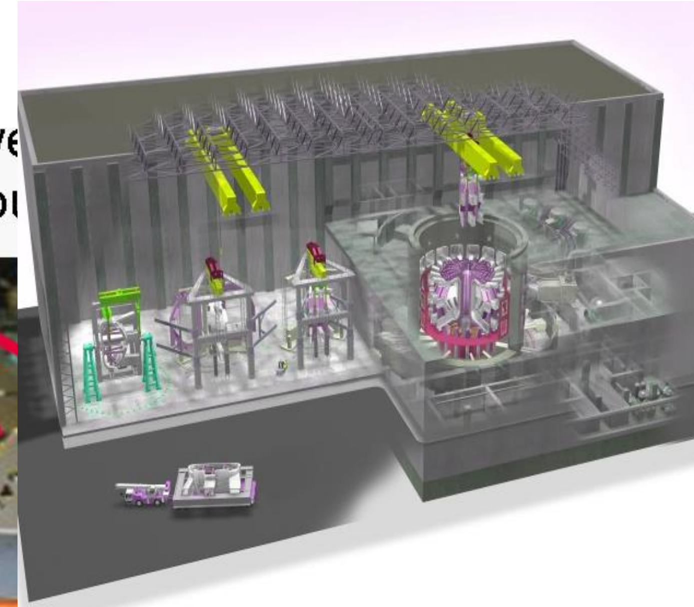
Plan of ITER Site Layout



- Will cover an area of about 60 ha
- Large buildings up to 170 m long
- Large number of systems

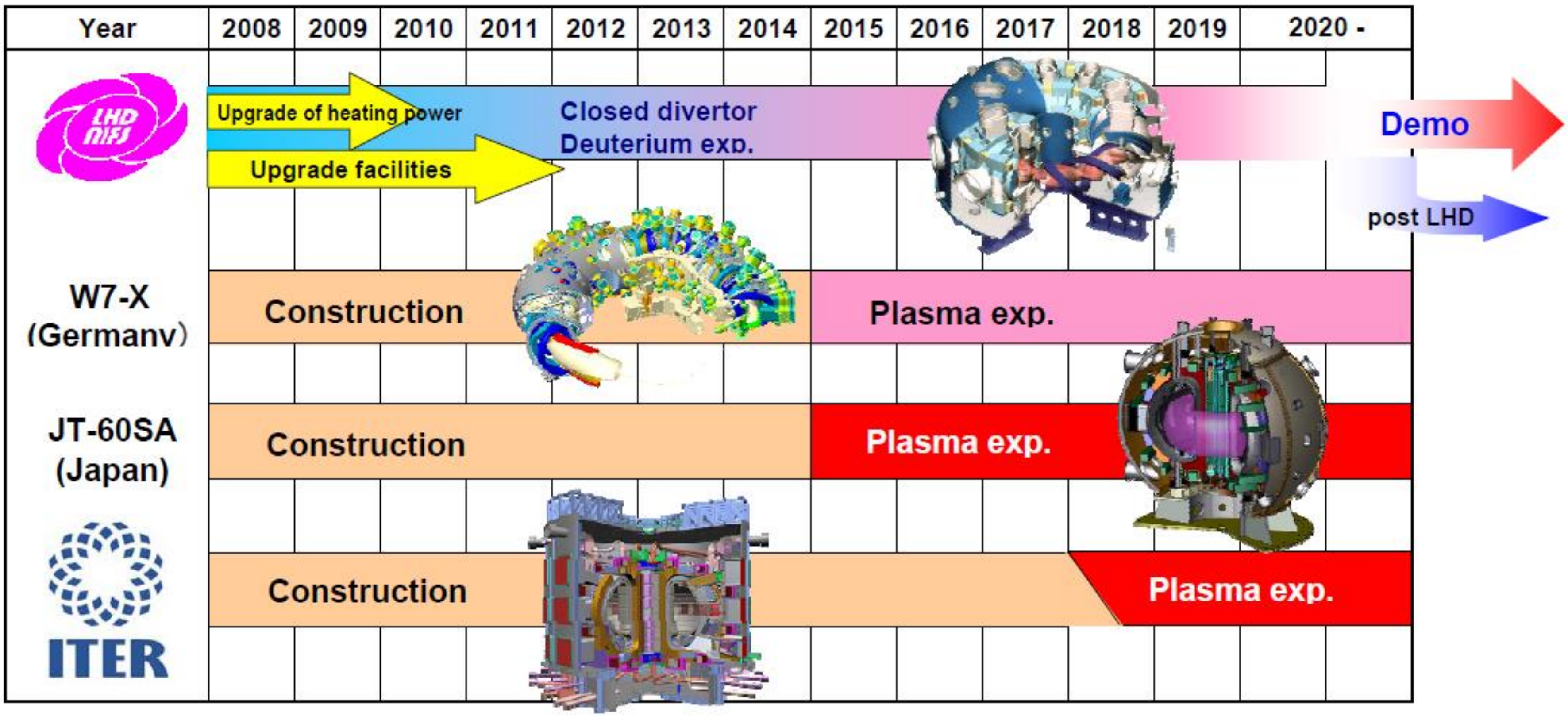
Plan of ITER Site Layout



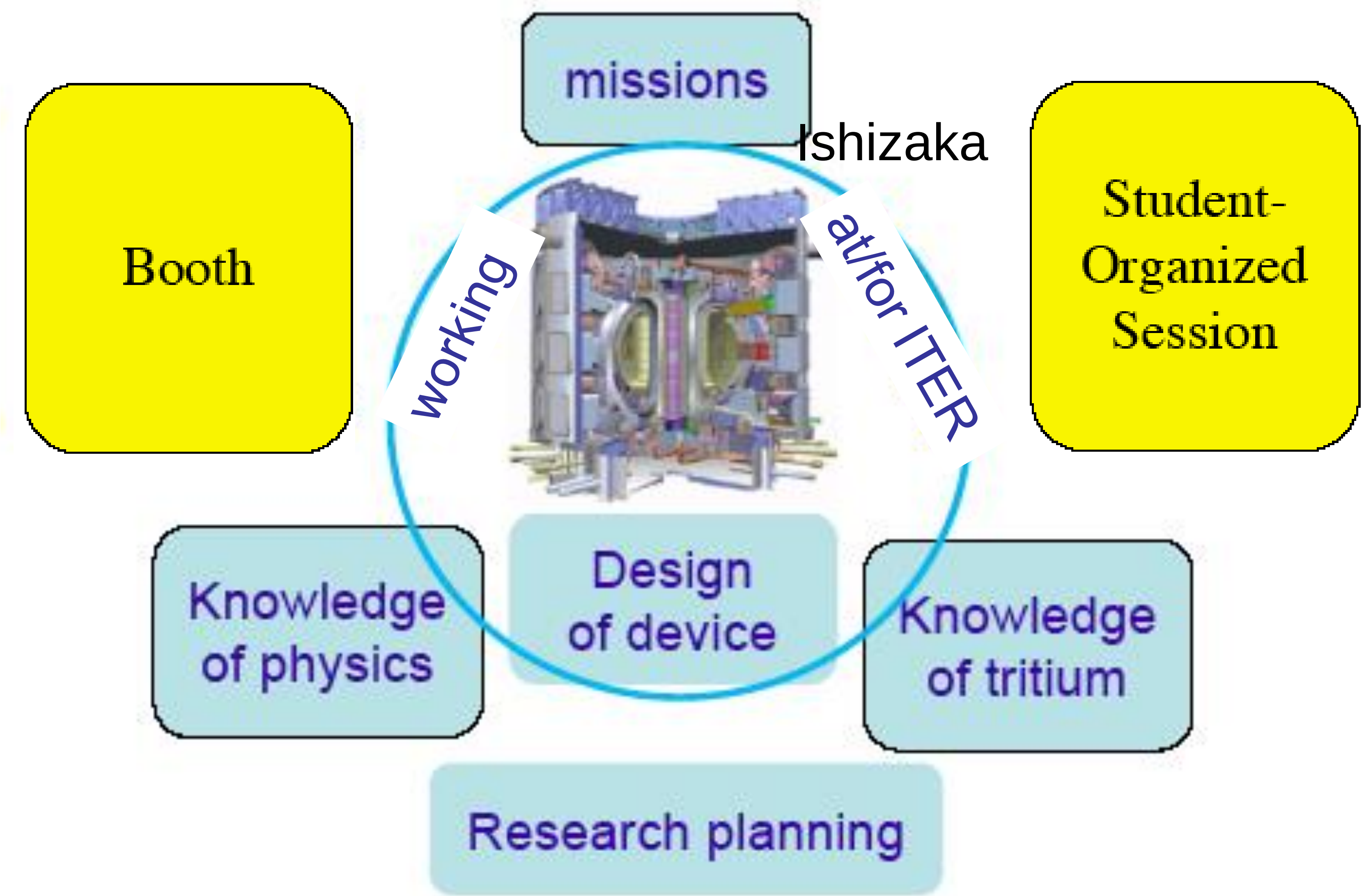


- Will cover an area of about 60 ha
- Large buildings up to 170 m long
- Large number of systems

Effective use of facility for bidirectional benefits
- Strategy in this decade -

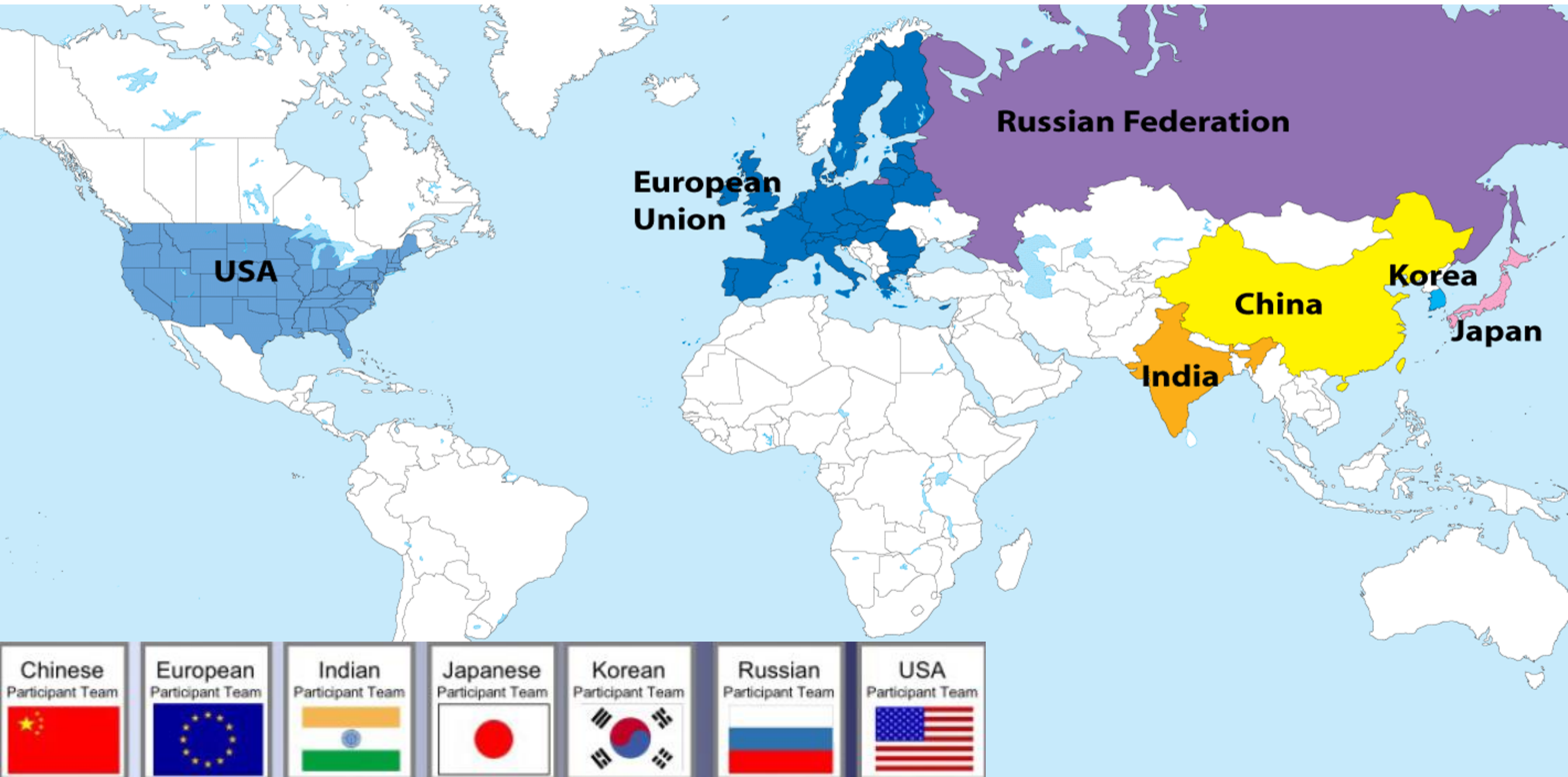


1. Two time scales; in these 10 years & next decade
2. Provision against risks and alternative plan (Portfolio)
3. Enhancement of collaboration, Human resource development
4. NIFS offers collaboration for public subscription


missions
Ishizaka
Booth
working
at/for ITER
Student-Organized Session
Knowledge of physics
Design of device
Knowledge of tritium
Research planning

ITER – a truly international cooperation….

Seven Members, representing more than half the world's population, are involved in the construction.



Working for ITER: General Roles & Responsibilities

ITER Organization and the Fusion Community in Members work together on ITER.

- ITER Organization (IO)
  - Planning/Design
  - Integration / QA / Safety / Licensing / Schedule
  - Installation
  - Testing + Commissioning
  - Operation

- Members – Domestic Agencies (DAs)
  - Detailing / Designing
  - Procuring
  - Delivering
  - Support installation

- Members -Scientific Community

  The IO is to assume responsibilities for coordinating physics research plans for ITER of the Members, e.g. using existing framework of the IPTA (International Tokamak Physics Activities).

Working at the ITER Organization

- Staff (normally 5 yrs contract)

  Professionals & Supporting Staff

- Visiting Researchers

- Post-doctoral Researchers

Principality of Monaco Post-doctoral Research Fellowship

The principal objective

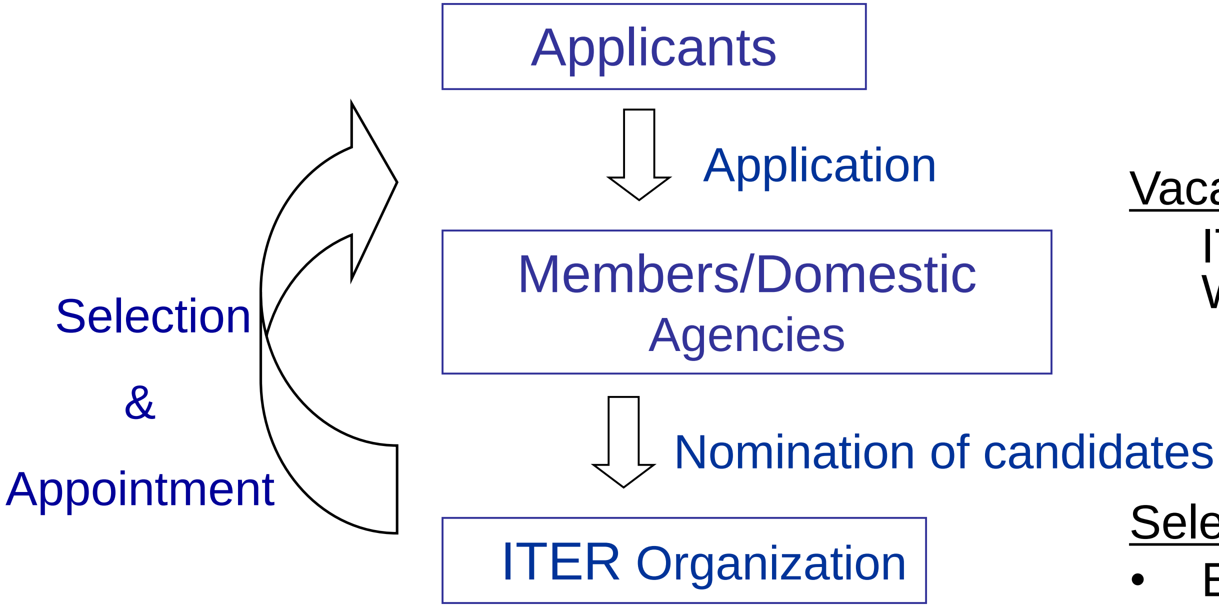
Development of excellence in research in fusion science and technology within the ITER framework. Brilliance and creativity, together with an understanding of the relevance of individual's research interests to the ITER project are required

Possible Candidate for the Programme 2008

- Nationality of the ITER Members or Principality of Monaco
- Awarded PhD after 1 January 2005

Next Opening December 2009 (tbc)

ITER Organization Recruitment Process



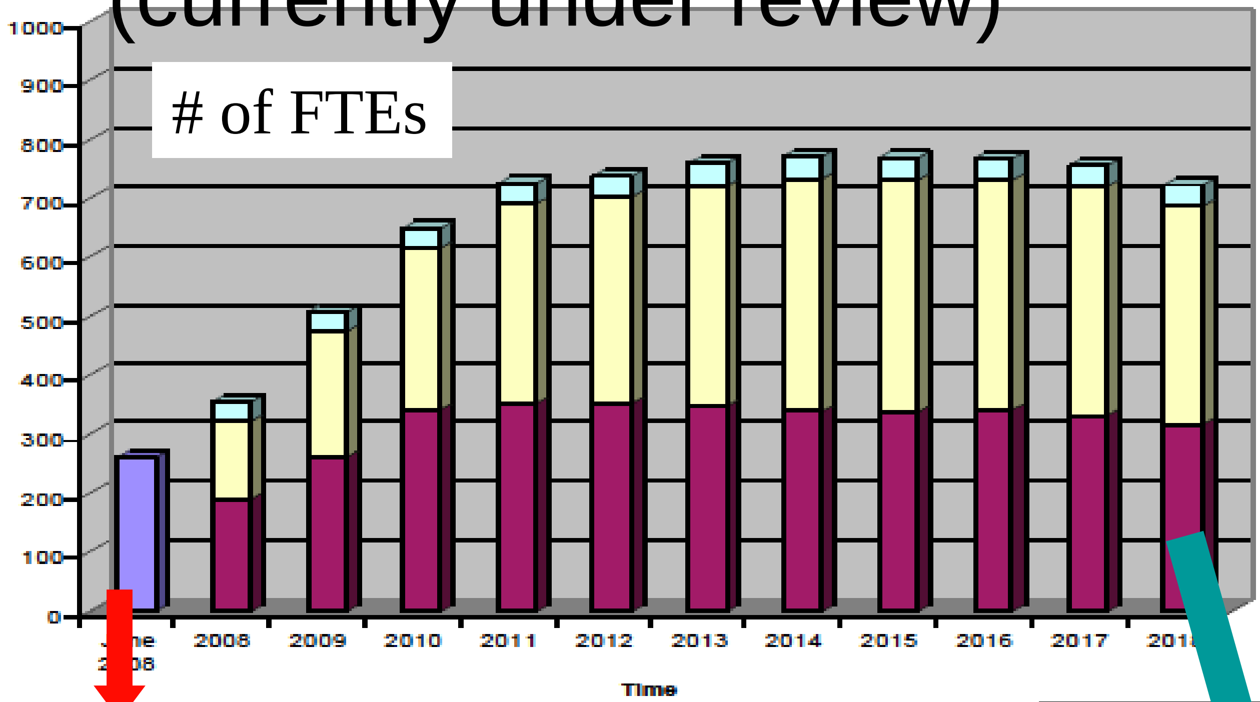
<u>Vacancy Notice</u>

ITER Organization Website: www.iter.org

<u>Selection Criteria</u>

- Educational background
- Professional experience
- Ability to work in a multinational team
- Coordination and communication skills
- Fluency in English etc.

Long-Term Manpower Resource Estimates
(currently under review)



February 2006

2018

From the Drawing Board to Reality

ITER booth and BA booth

Panel Discussion

Experiment

Panel Discussion

Theory

Panel Discussion

Simulation

Panel Discussion

ITER

Talk-in with DG

Message for the Youngsters

- Although the present focus of the project is construction activities, ITER is also a major scientific and technological research programme, for which the best of world intellectual resources is needed.
- Challenges for the young, necessary for fulfilment of the objective of the ITER will be identified.
- It is important that young students and researchers in the world recognize the rapid development of the project, and the fundamental issue that must be overcome in ITER.

The 2nd ITER International Summer School 2008 provides

Accurate knowledge for solving problems

Global view to identify *raison d'etre* of one's research

Structuring knowledge for problem definition

Learning from First-class researchers for innovations

Initiatives through student-organized sessions

Experiences of participation in ITER culture

"Prost 乾杯" for the Future

Solution for the energy problem will be more and more demanding