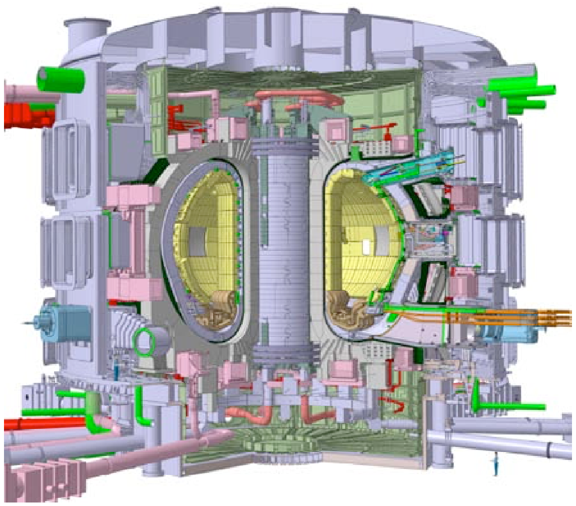
THE PHYSICS AND TECHNOLOGY BASIS OF ITER AND ITS MISSION ON THE PATH TO DEMO



SOFE - ICOPS

San Diego, 01 – 05 June 2009

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To the PDDG

ITER Organization

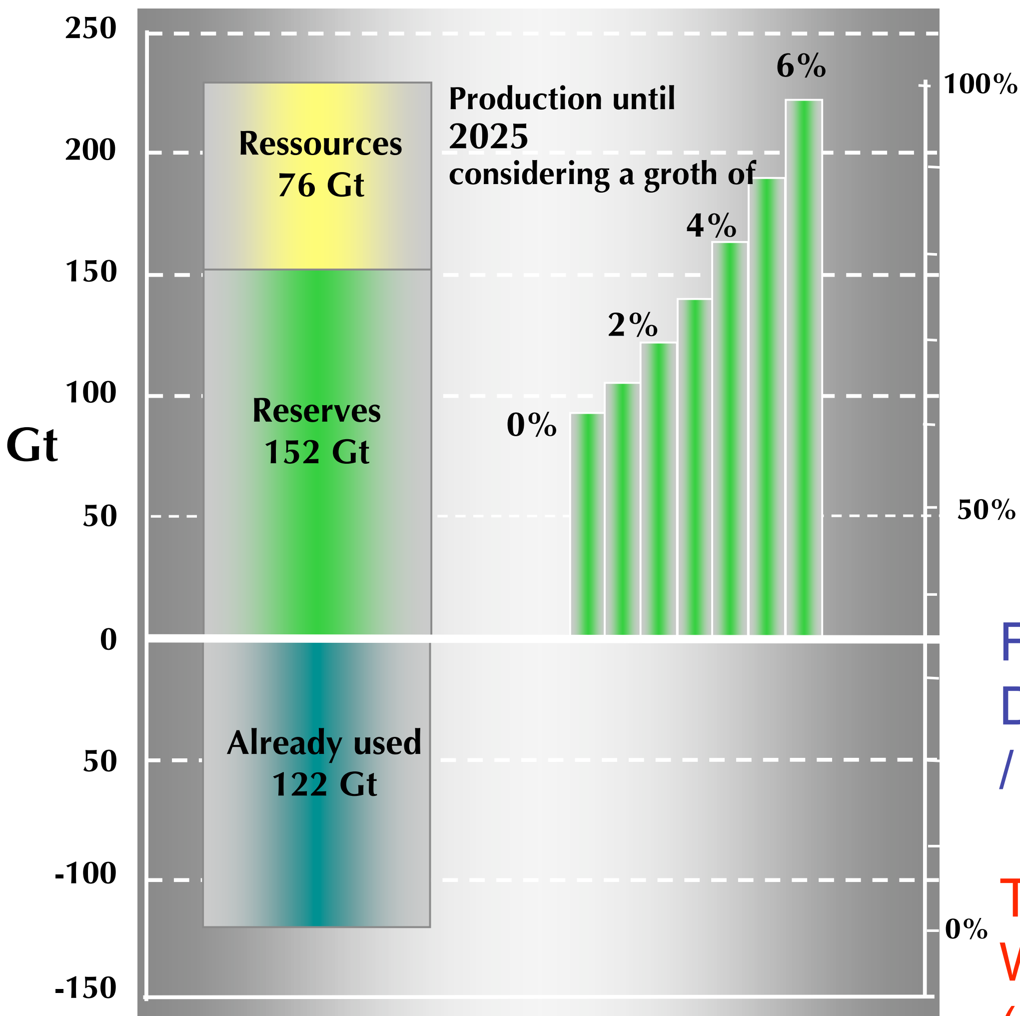
Acknoledgements: D. Campbell, M. Glugla, M. Merola N. Mitchell, A. Tanga, A. Tesini

Outline

- Motivation for fusion development (very brief):
  - Climate change, finite Oil and Natural Gas resources
  - Fusion as long-term solution for part of the problem
- Fusion basics – magnetic confinement
- ITER and its mission,
  - Physics Development towards ITER
  - Technology Development for ITER – the ITER design (see also S. Chiocchio)
  - Status of ITER
- Road-map and Technologies needed for DEMO (very brief)
- Conclusion

The climate change problem, the finite oil resources

World oil reserves / resources



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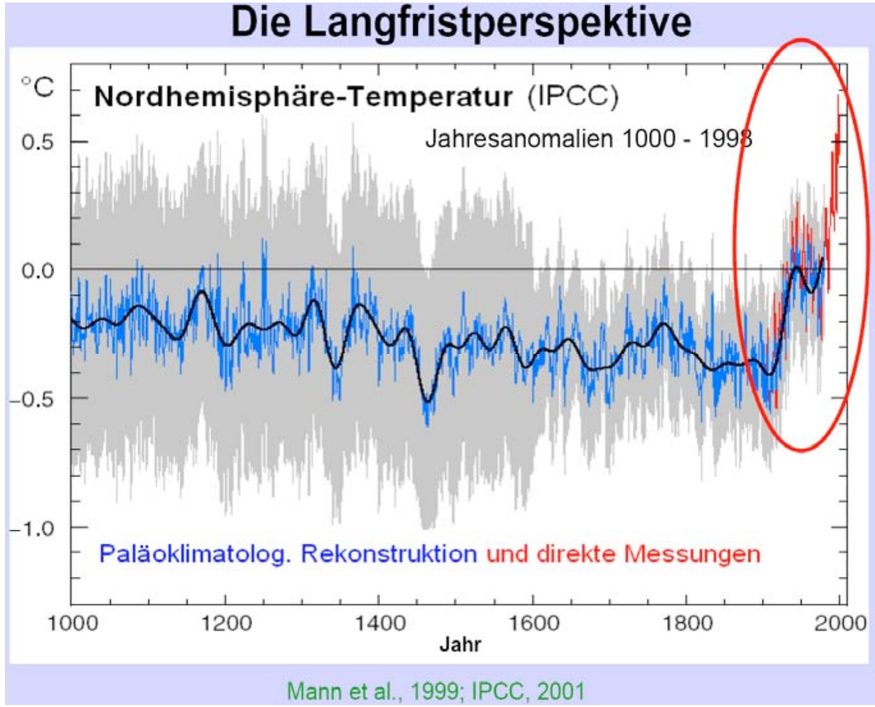
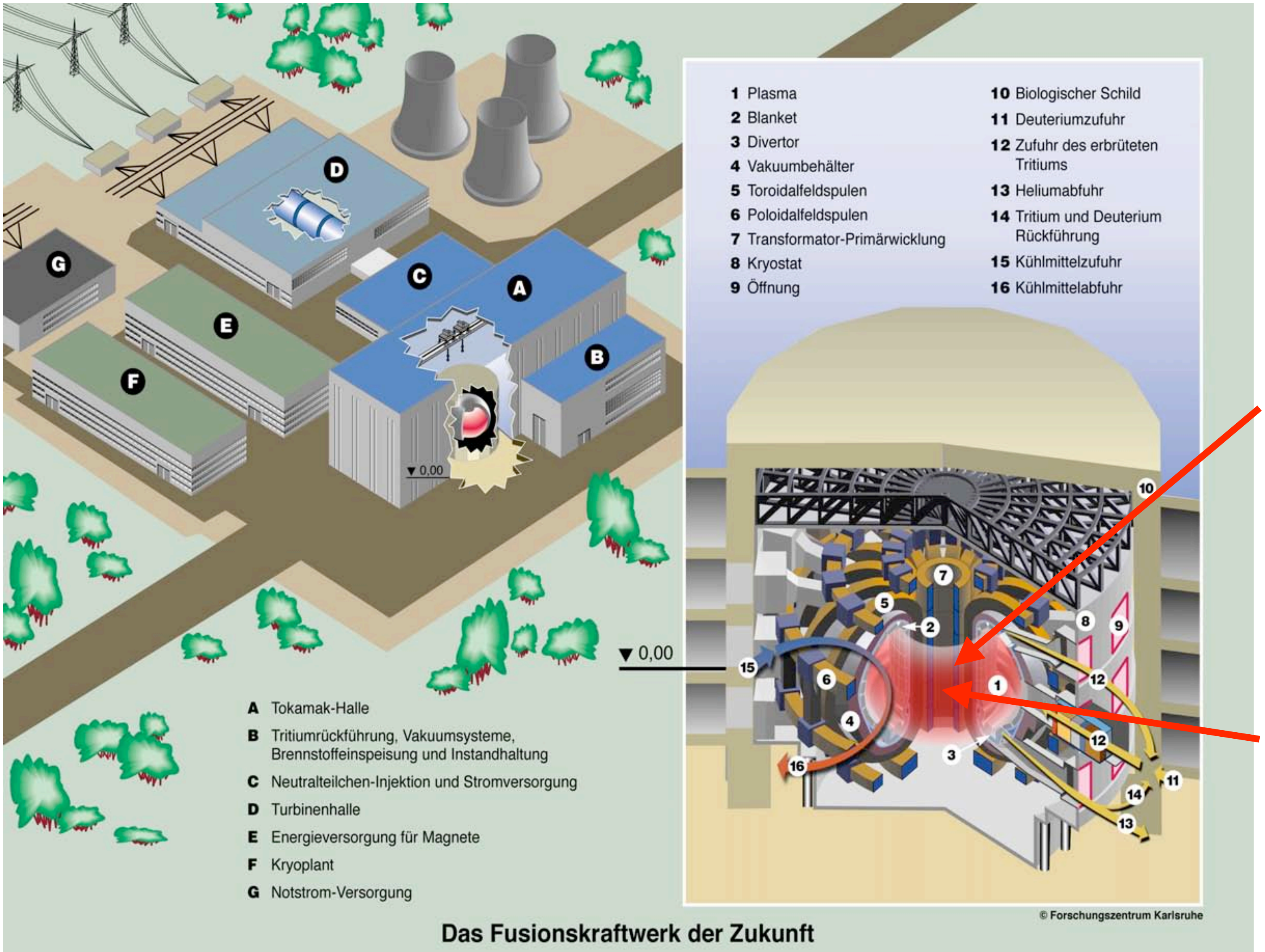


Finite oil resources and reserves
Depending on growth oil could run out / be expensive within 2 decades

To replace only half of it means Terra W of energy from other sources (nuclear, coal, renewables, fusion)

Schematic View of a future Fusion Power Reactor



Fusion can be a long term solution not a short term fix

Power generated by hot plasma

(20 keV = 200 Mio °C)

4/5 th of Power transported by 14 MeV Neutrons

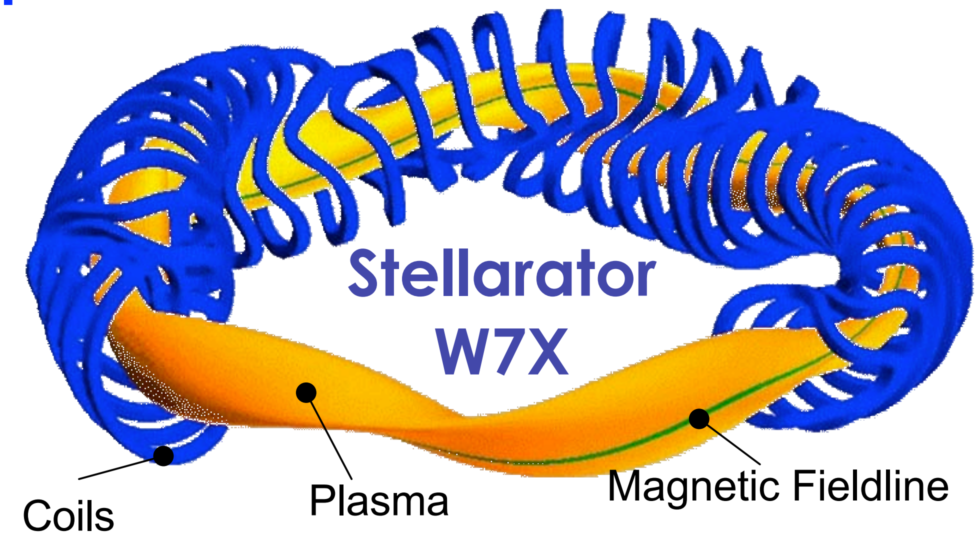
Magnetic Confinement of a plasma with 10 to 20 keV

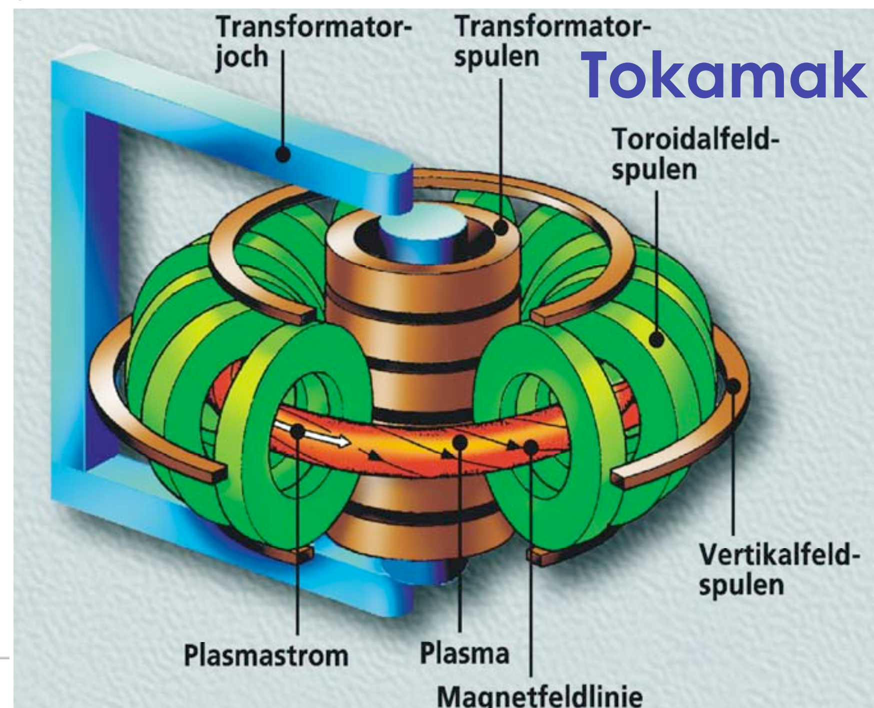
A toroidal magnetic system needs:

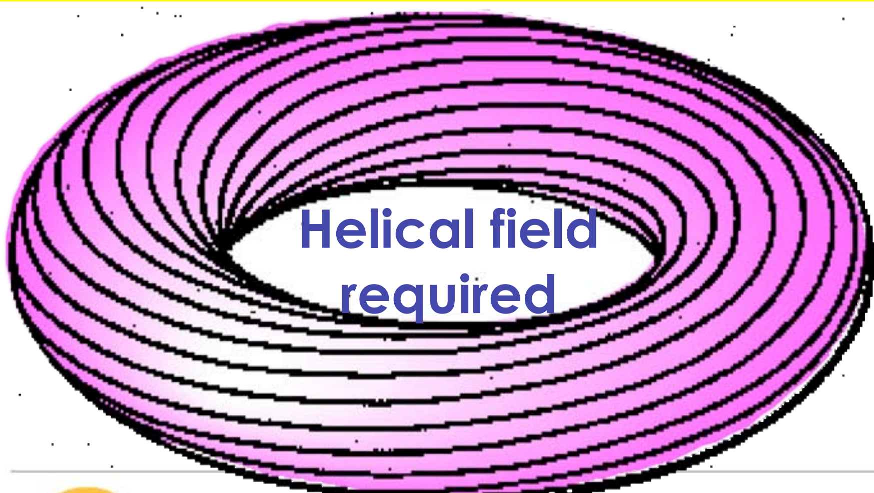
- a helical field configuration to compensate drifts
- a magnetic well

Two successful systems:

Stellarator / Tokamak - ITER







A modern Tokamak – Vertical-, Radial-, Divertor Fields

Poloidal field coils

Vertical Field

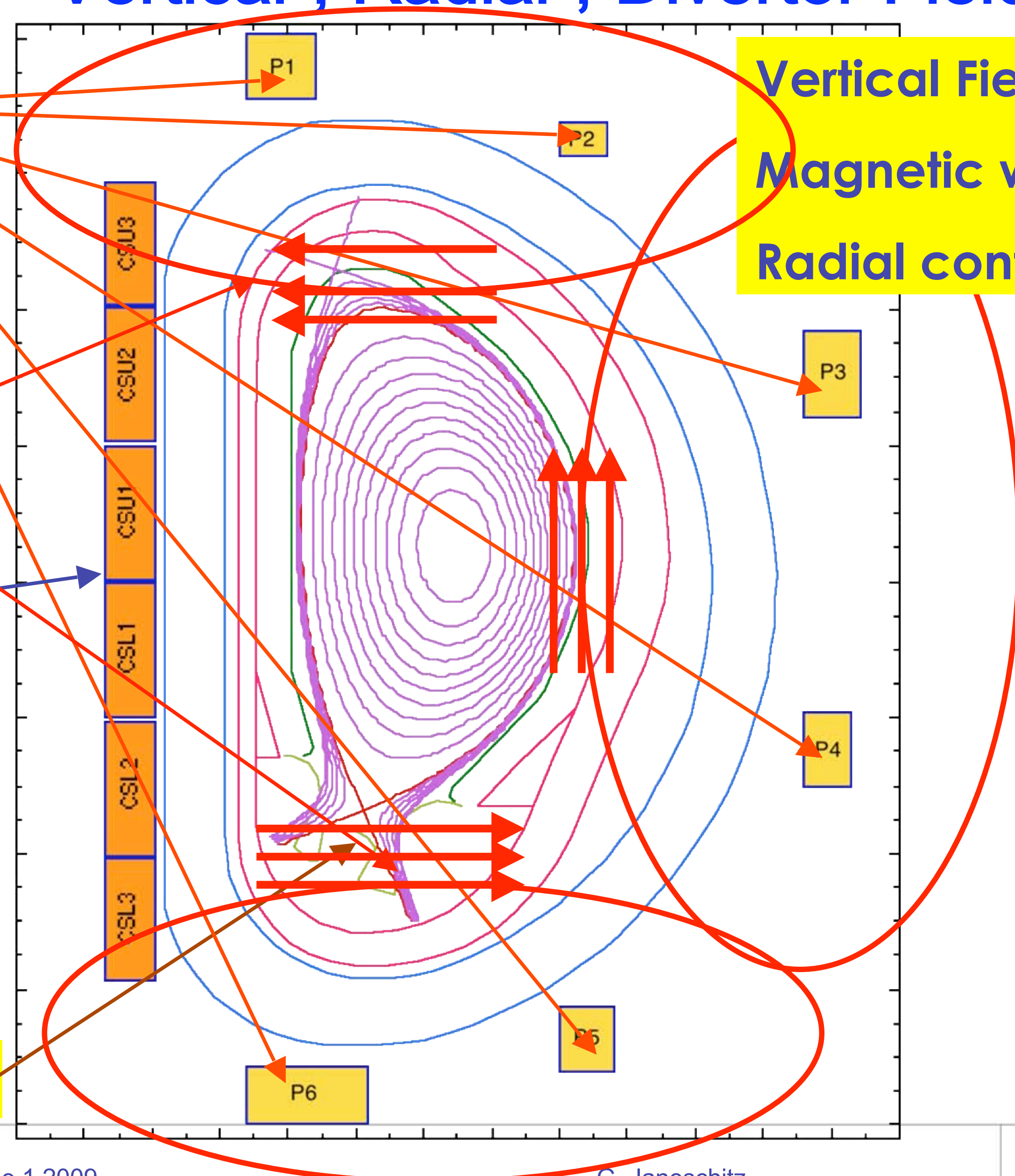
Magnetic well

Radial control

Radial Field

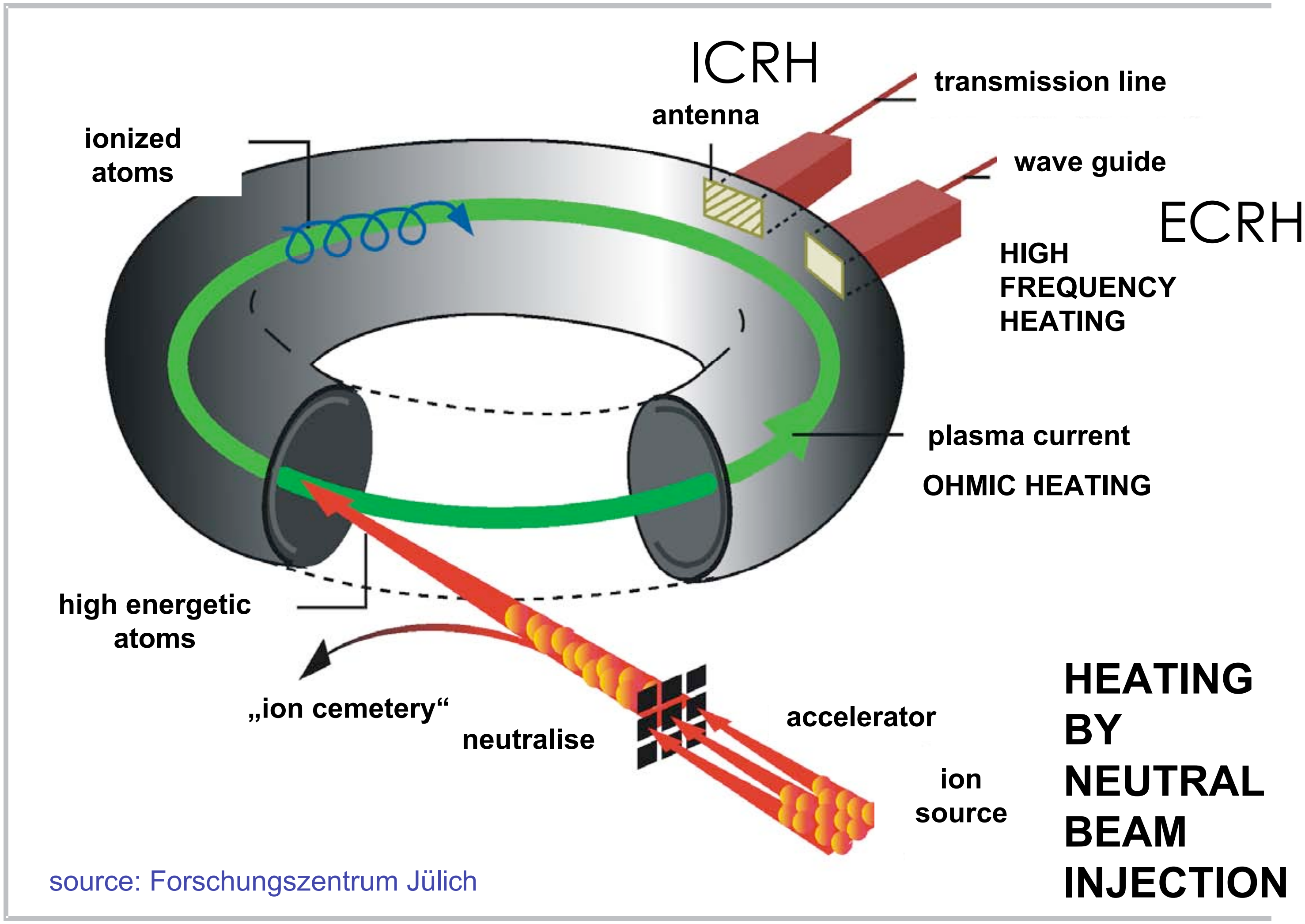
vertical control

Transformator



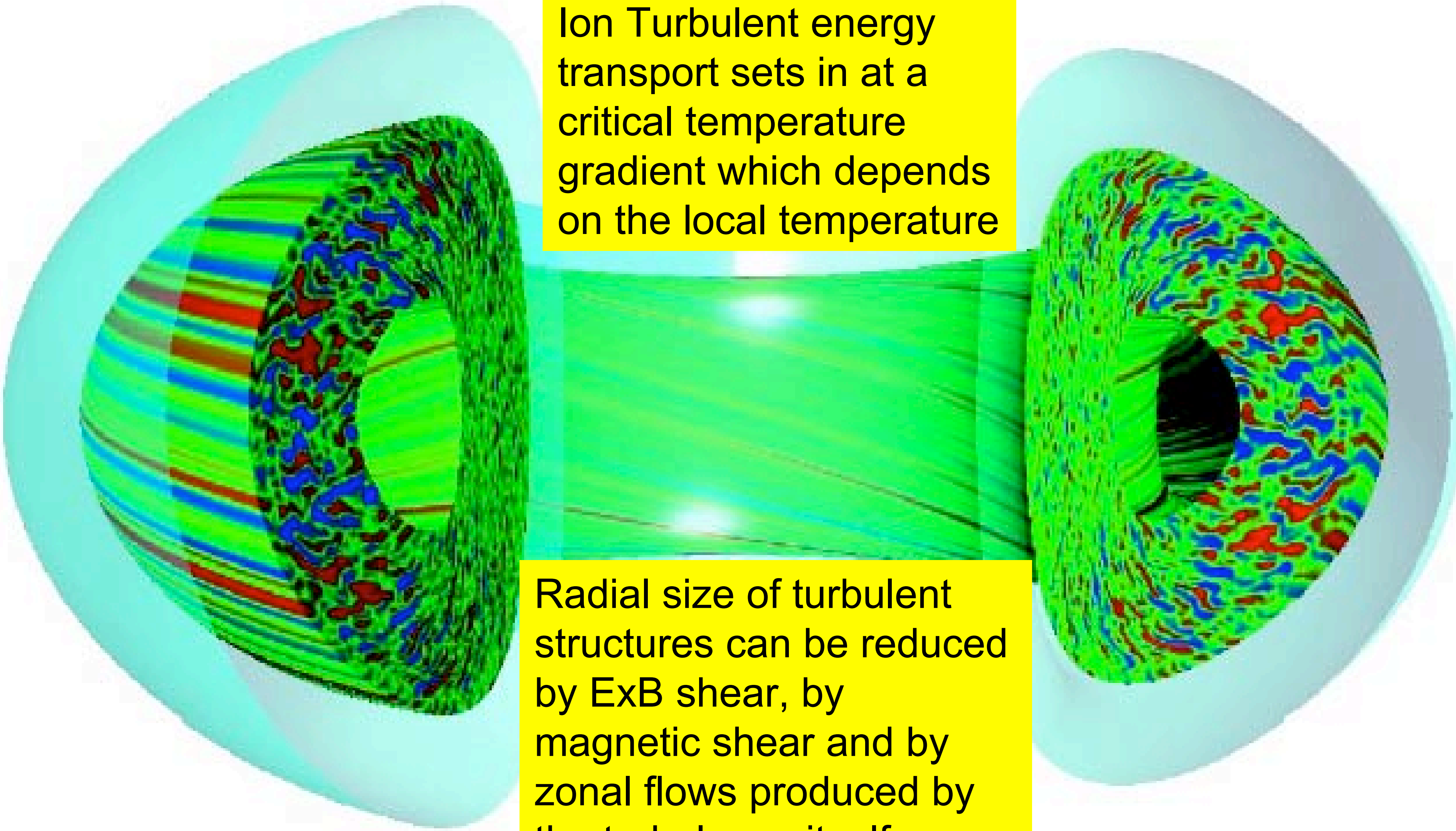
Divertor

Methods for the heating of a tokamak plasma



source: Forschungszentrum Jülich

Energy and particle Transport is governed by turbulence

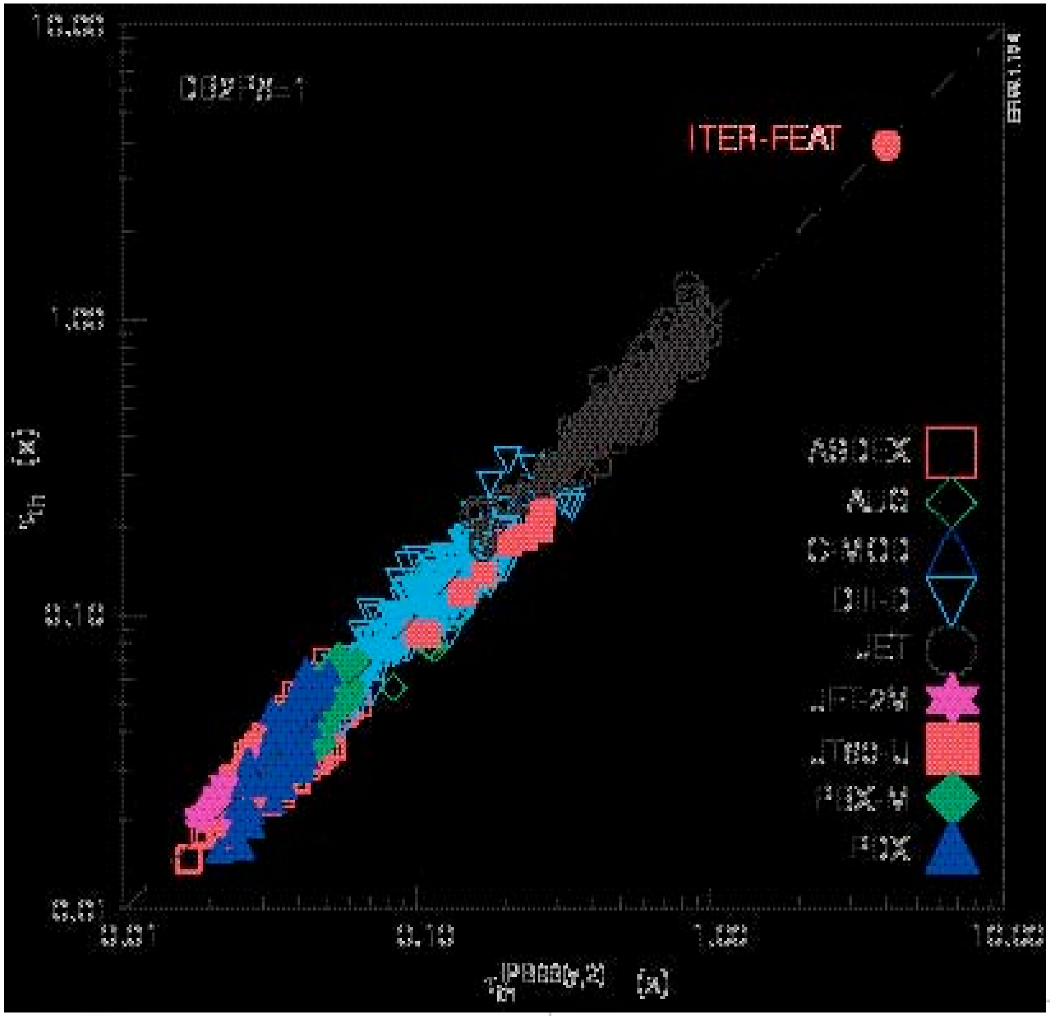


Ion Turbulent energy transport sets in at a critical temperature gradient which depends on the local temperature

Radial size of turbulent structures can be reduced by ExB shear, by magnetic shear and by zonal flows produced by the turbulence itself

Confinement Scaling Relation

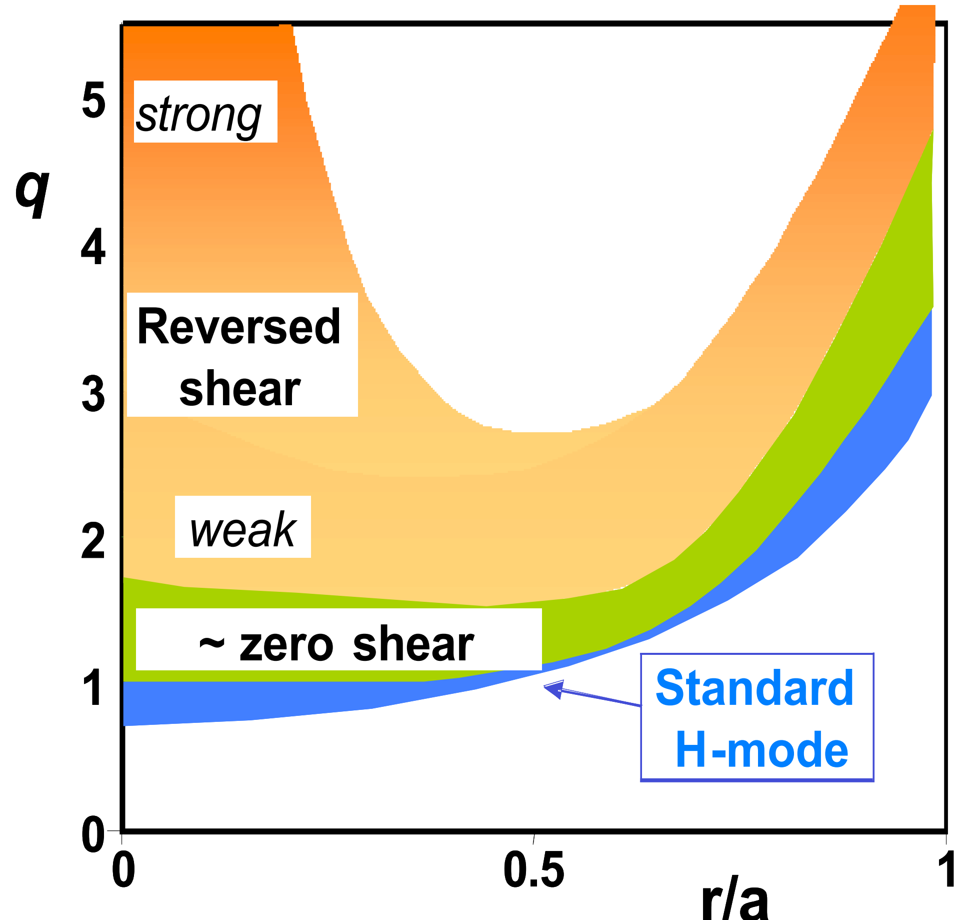
$$\tau_{E,th}^{ELMy} = 0.0562 \times I^{0.93} B^{0.15} P^{-0.69} n_{e,19}^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa^{0.78}$$



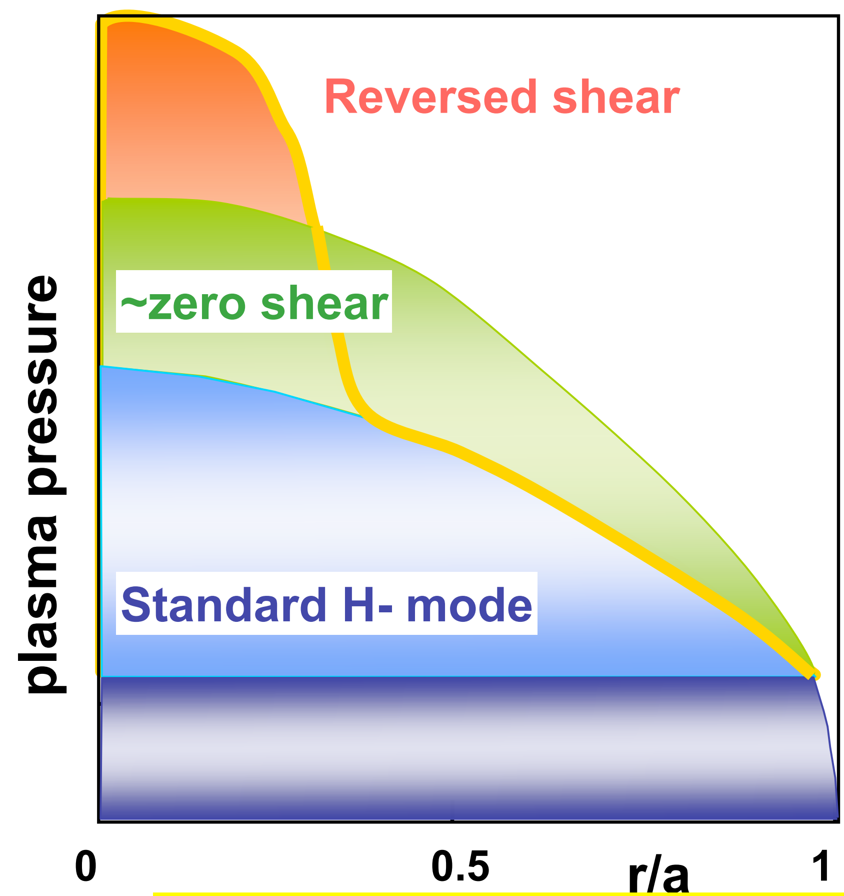
- Extrapolation of global thermal confinement time to ITER-FEAT (R = 6.2 m), using the IPB98(y,2) scaling (t_E=3.9s)
  - Only Q= 10 possible, no ignition
  - For ignition R~ 8 m and t_E >5.7 sec needed
- **ITER represents an extrapolation of a factor ~4 beyond existing database**

We start to explain the blue profile types by physic models !!

q profiles for standard and advanced scenarios



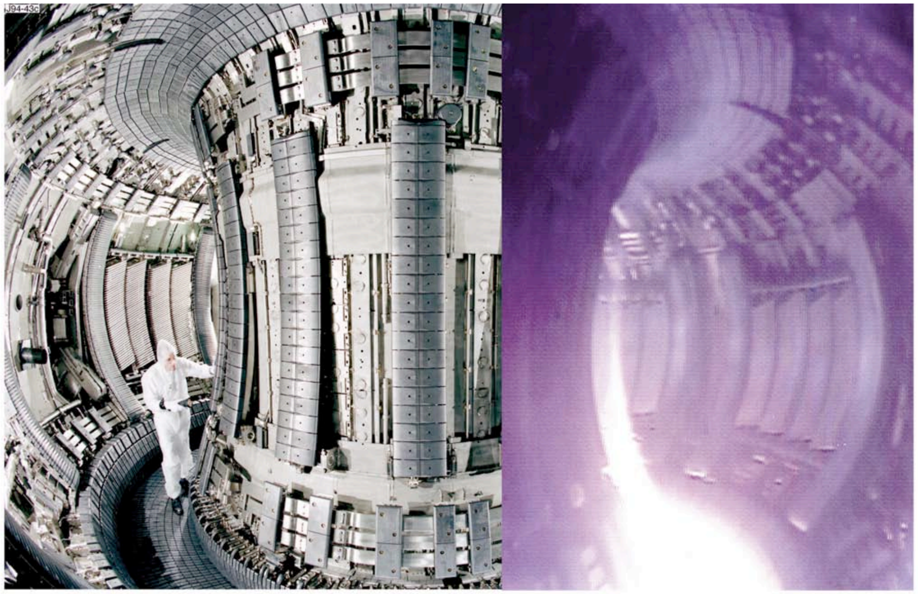
Pressure profiles for standard and advanced scenarios



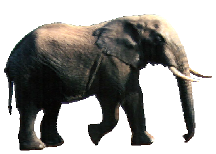
Pressure profiles produce Bootstrap Currents => important for Steady State Operation

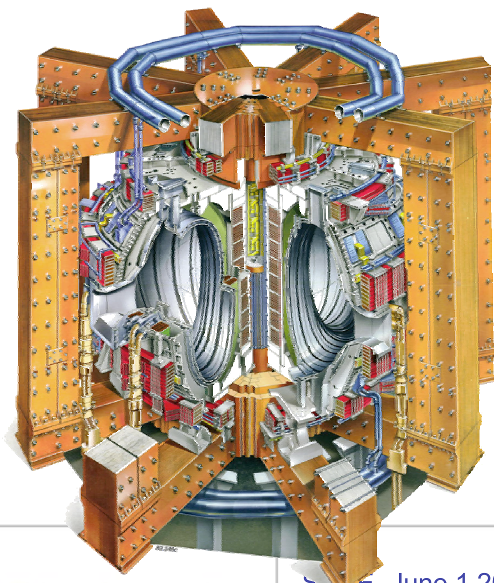
H-mode pedestal ballooning unstable => ELMs

Family of Tokamaks defined the ITER Physics Basis

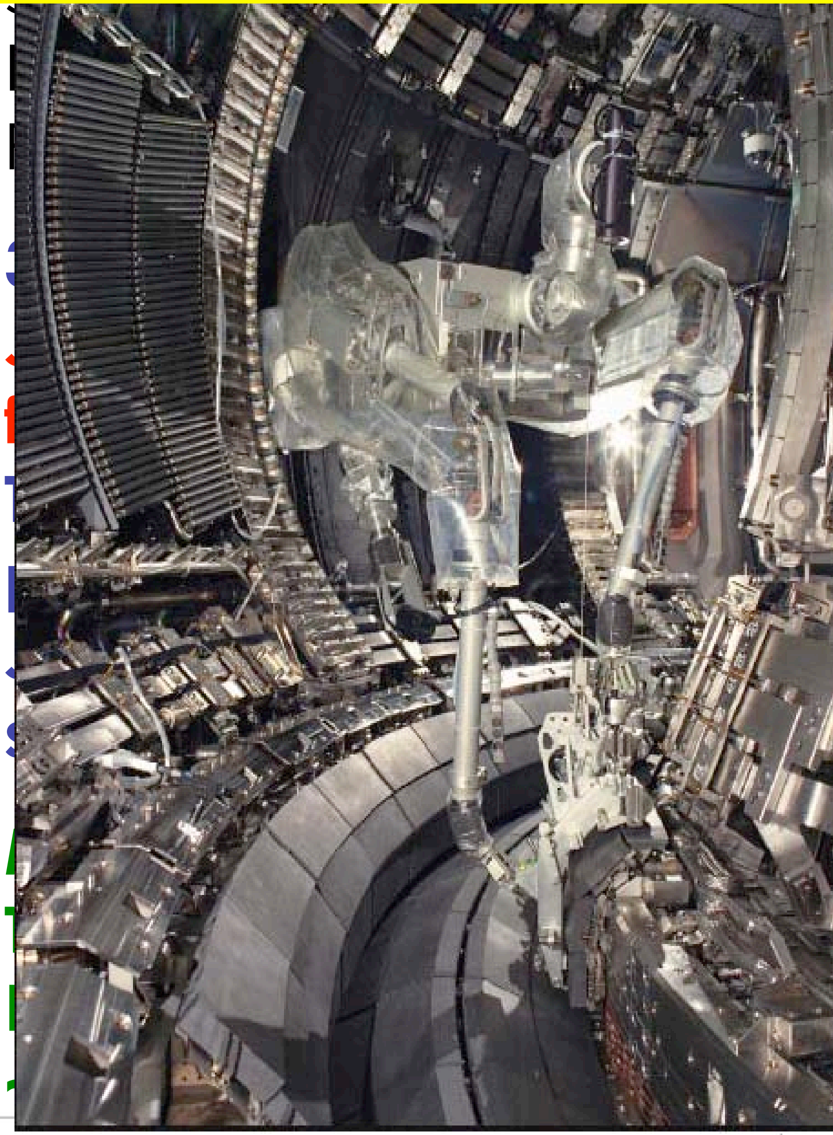


JET – Internals & Plasma

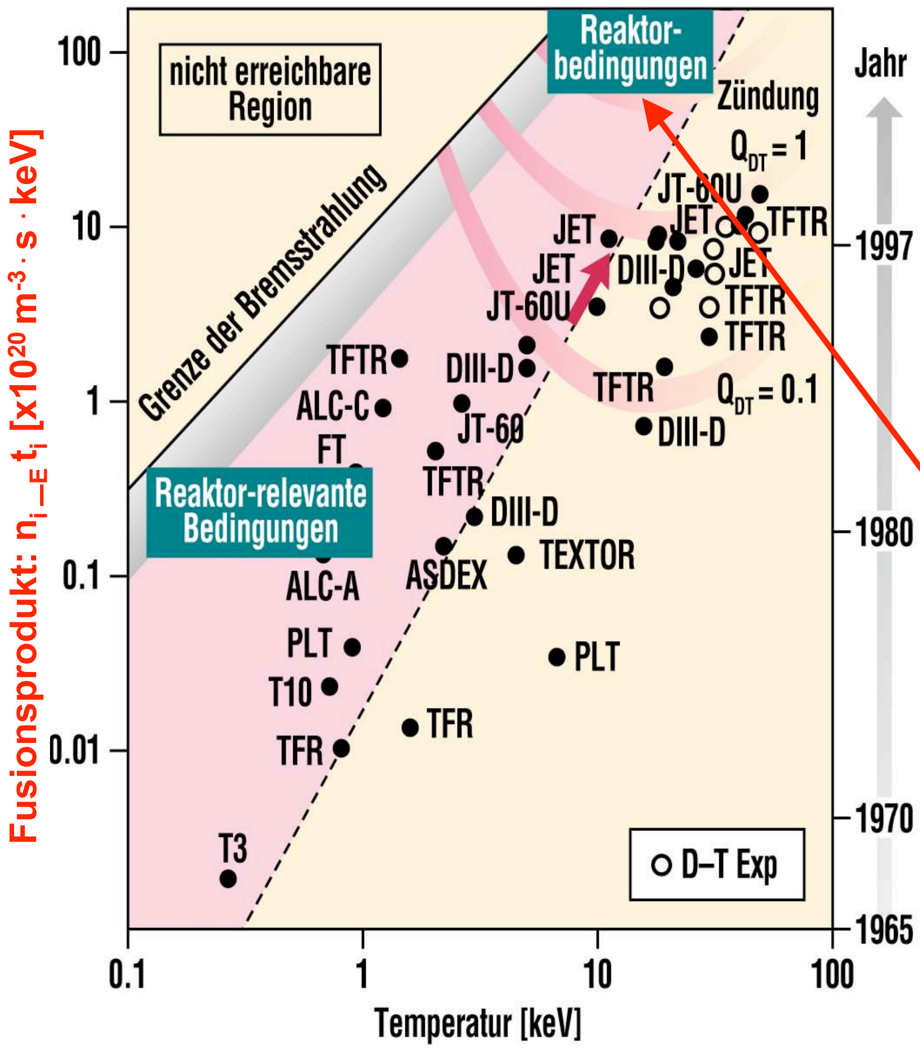


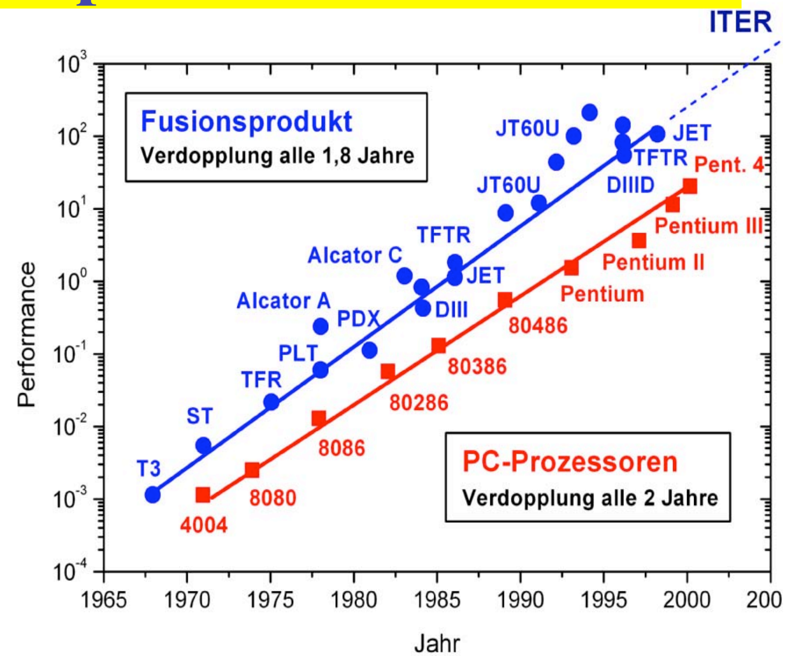


Remote Handling in JET



Progress in Fusion Tripple Product similar to Progress in Microprocessor Development





Fusion Reactor Class in the area of 3 to 6 10^{21}

- Ti > 10 keV

  => $n_e \times \tau_E \sim 6.0\ 10^{20}\ m^{-3}s$

  => $n_e \sim 1.0\ 10^{20}\ m^{-3}$

  => $\tau_E \sim 6.0$ s

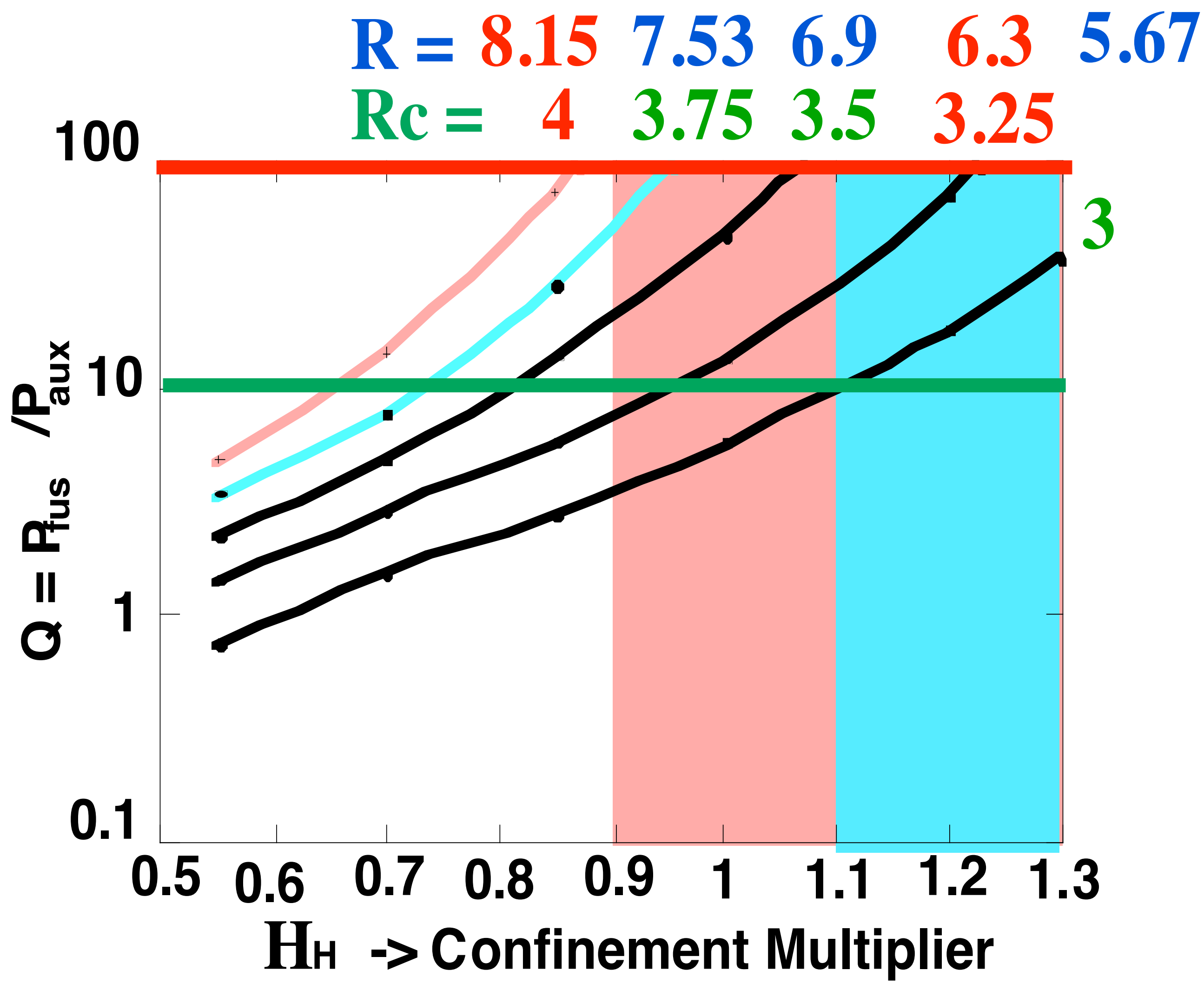
=> Energy Confinement time between 3 and 6 sec

Output of a system code for possible ITERs

physics and engineering constraints are combined in a

"System Code"

which is able to calculate a consistent parameter set.



The ITER Machine

- V: 840m^3

R/a: 6.2m /2m

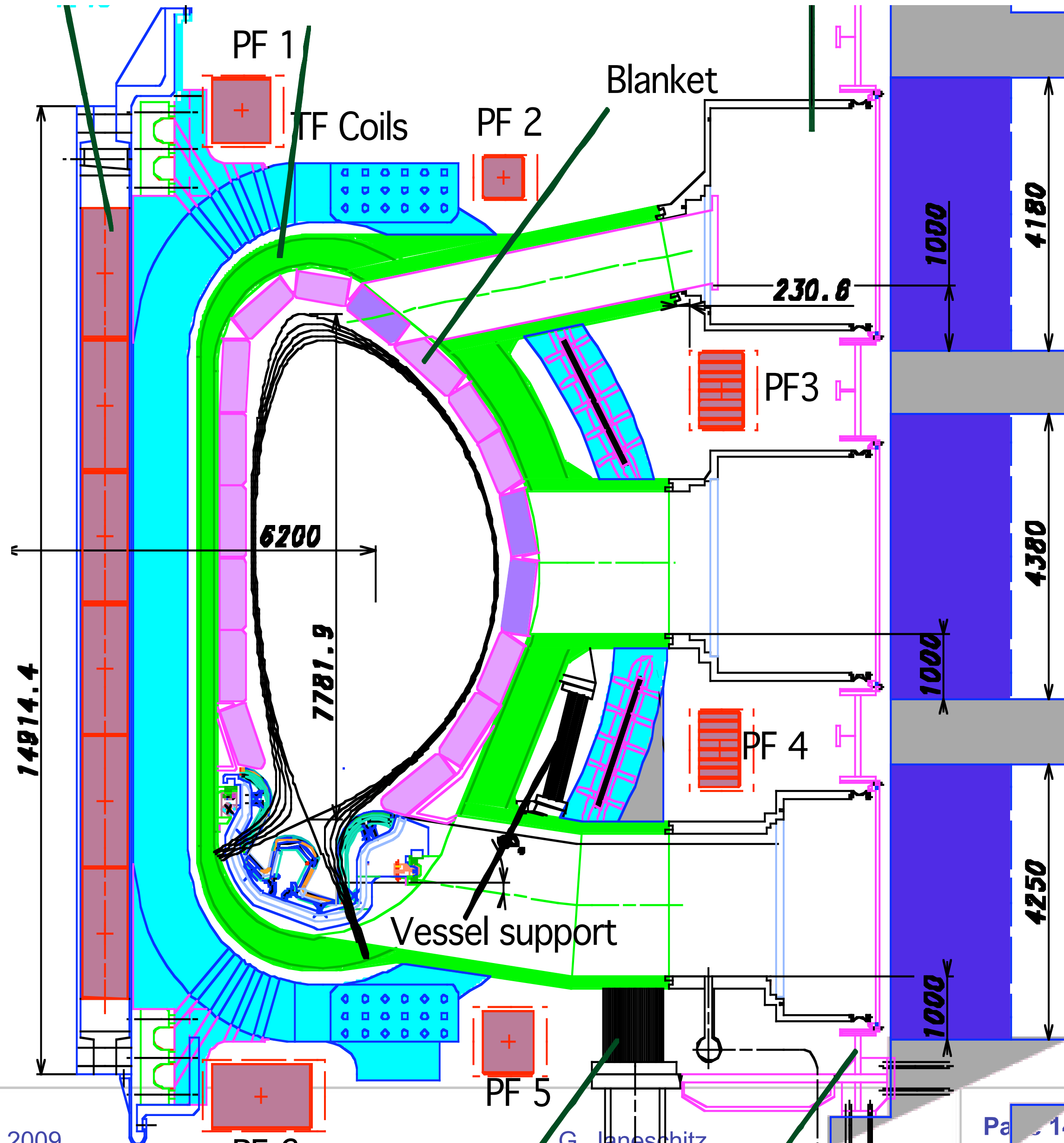
Vertical elongation: 1.85
Triangularity: 0.45

- Density: $10^{20} m^{-3}$
- PeakTemperature:17keV

-Fusion gain Q = 10
-Fusion Power: ~500MW
-Ohmic burn 400 sec

-Goal Q=5 for 3000 sec

- Plasma Current : 15MA
- Toroidal field: 5.4T



What is ITER ?

ITER is a major international collaboration in fusion energy research established in the 80th by Reagan - Gorbachev involving

the EU (plus Switzerland), China, India, Japan, the Russian Federation, South Korea and the United States

Physics Goals:

- ITER is designed to produce a plasma dominated by α-particle heating
- produce a significant fusion power amplification factor ($Q \geq 10$) in long-pulse operation
- aim to achieve steady-state operation of a tokamak ($Q = 5$)
- retain the possibility of exploring 'controlled ignition' ($Q \geq 30$)

Technology Goals:

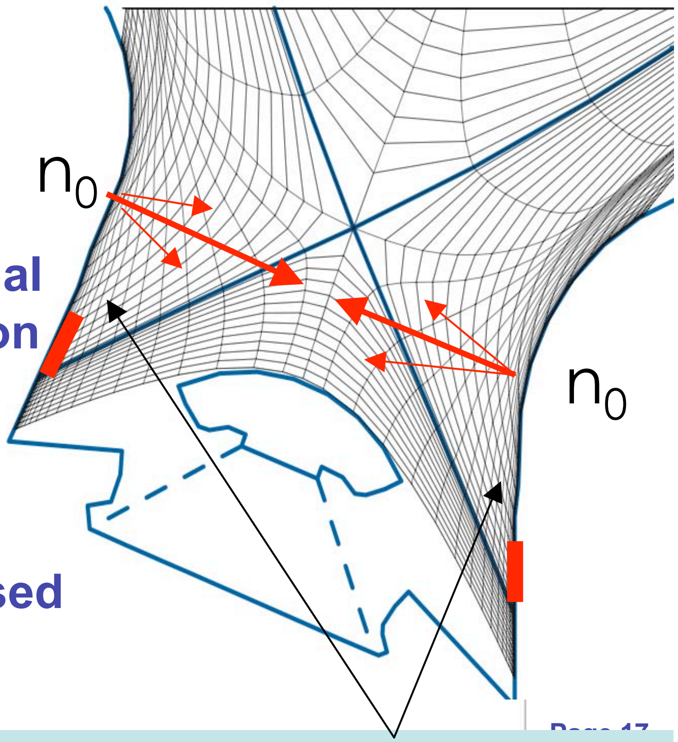
- demonstrate integrated operation of technologies for a fusion power plant
- test components required for a fusion power plant
- test concepts for a tritium breeding blanket

What were / are the major Challenges in Physics ?

- The solution to the divertor peak heatflux problem (solved / ongoing)
  - The development of the radiative divertor
- The prediction of the ITER Energy Confinement and thus the definition of its size (solved / ongoing)
  - Shortfalls of scaling as a predictive tool => Physics understanding needed !
- The impact of magnetic ripple on energy confinement (solved / ongoing)
  - Impact low at ripple < 0.5% at separatrix, however, not understood !!
- The developing understanding of ELMs and RWM and their stabilisation (solved / ongoing)
  - In vessel coils foreseen in ITER – physics not understood – risk !
- The definition of a credible steady state scenario (solved / ongoing)
  - Development needs new generation of Tokamaks (EAST, KSTAR, JT60SA, ITER)

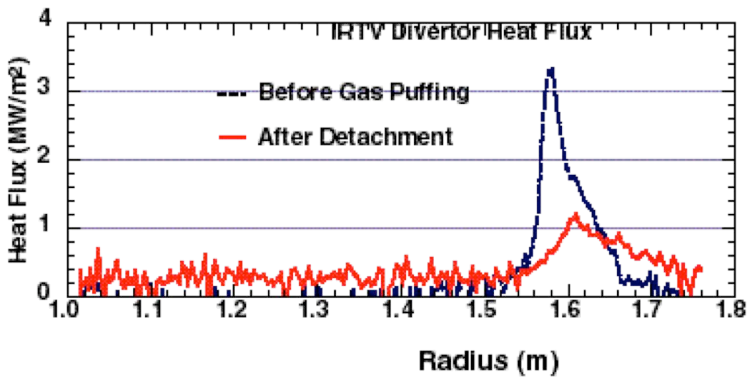
The Development of a Radiative Divertor

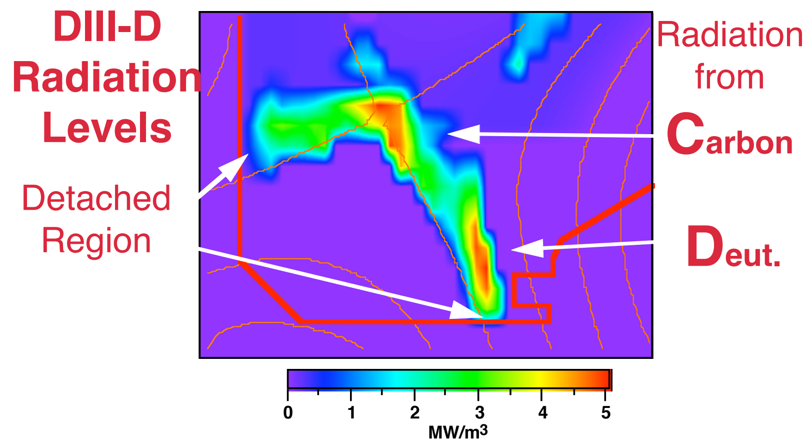
- A large fraction of the α - and additional heating power has to be dissipated inside the divertor – in ITER ~100 MW
- Due to fieldline geometry this power is concentrated in a toroidal band of a few cm width => large peak power flux (can be 40 MW/m^2)
  - Mitigation achieved by angling targets (limited) and by radiating power in the divertor (increase surface for power deposition) => peak power ~10 MW/m^2
  - However, radiation is limited due to pressure constancy along fieldlines in normal divertor operation
  - Loss of momentum lateral to plasma flux essential to increase radiation => neutral plasma interaction transports momentum laterally => needs low temperature divertor => high density
  - New physics was developed during 90th and validated on existing tokamaks => 2 D models used for prediction to ITER



Total fieldline angle >1 degree; => Poloidal Angle ~15 to 20 degree

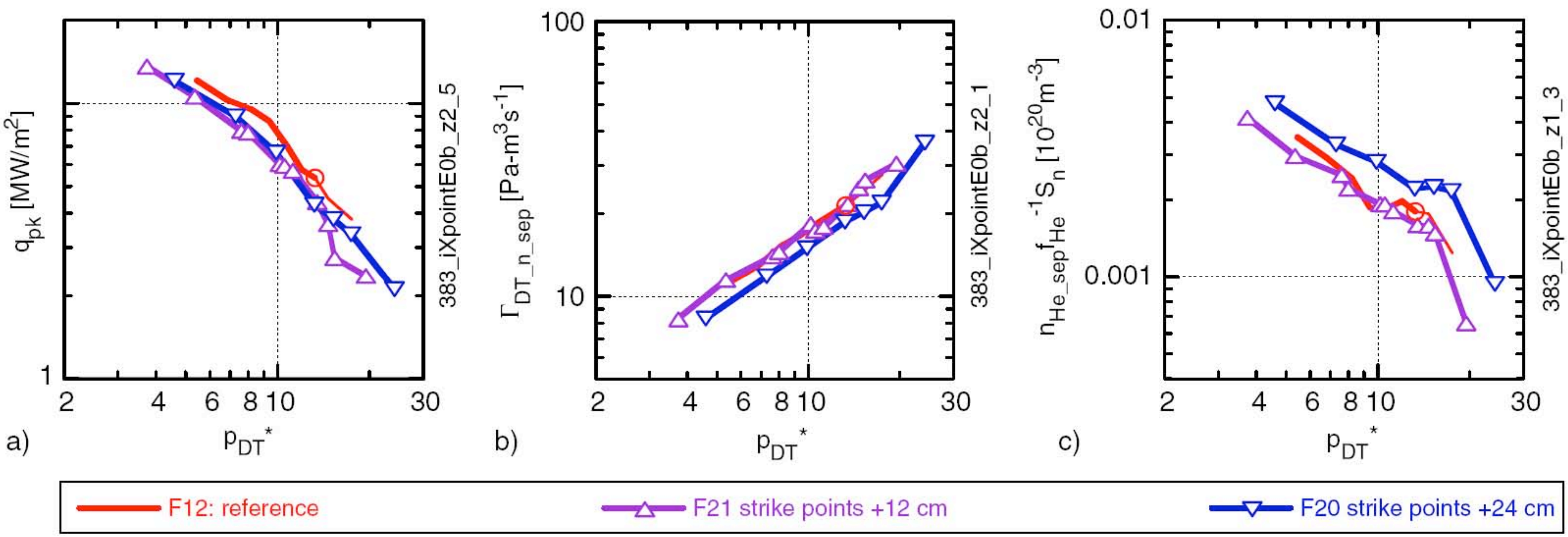
Experiments and Predictions for the ITER Divertor





Peak power flux in D-IIID significantly reduced by radiative divertor operation

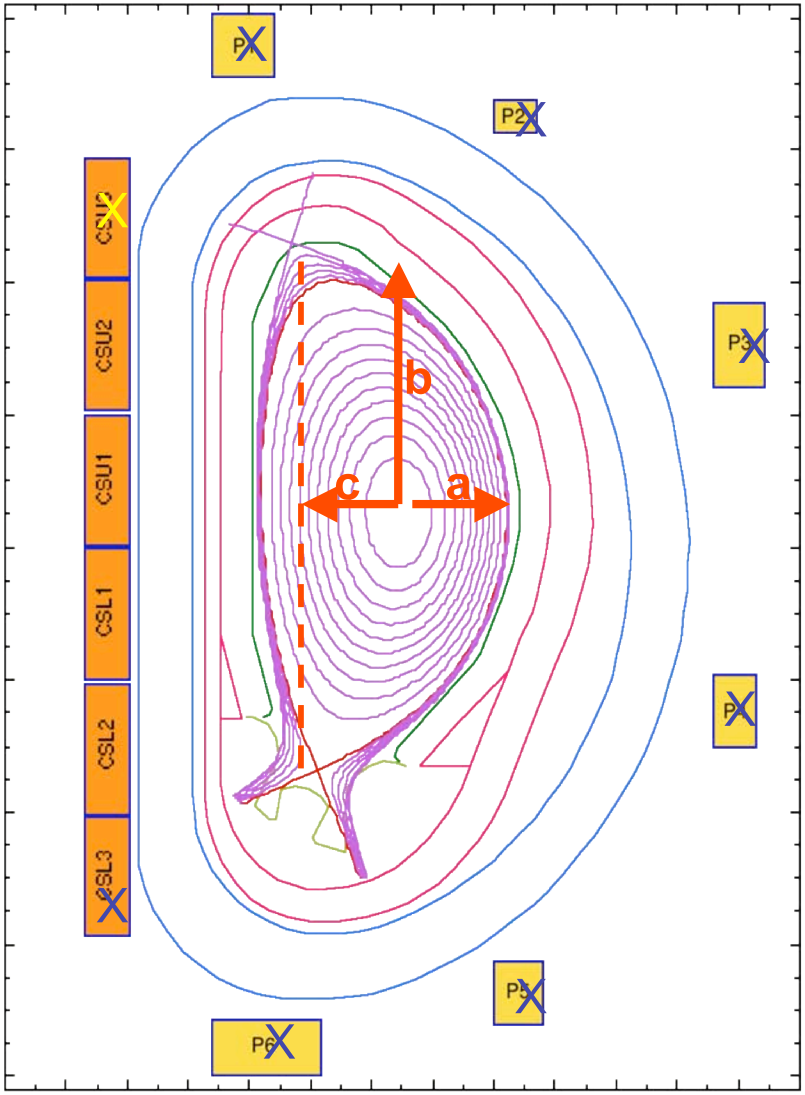
Power and He exhaust as well as fuel exhaust within achievable engineering parameter



2 important Confinement Physics issues understood

- Core confinement governed by two transport channels with different turbulence behaviour – dominant regime depends on H-mode pedestal temperature
  - Ion Temperatur GradientTransport (ITG): critical gradient proportional T
  - Electron Transport – less stiff, critical gradient seems prop 1 / SQR (T)
  - => A correct scaling depends on all machines being in the same regime - Ion or Electron dominated – was not the case end of 90s !! => error
  - Scaling was corrected 99 => bigger ITER machine – i.e. R = 6.0 to 6.2 m
- H-mode confinement regime depends on pedestal temperature
  - Scaling performed at medium density => it missed the degradation at high density – understanding emerged end of 90th => ITER design changed
  - Pedestal pressure ~ constant for given shaping (triangularity, elongation)
  - Change from large ITER to present machine triangularity increased from 0.2 to 0.35 at q95%, elongation 1.8
- Needed a different CS and TF coil design – impact on divertor !!
- **The above understanding allowed an optimisation of the ITER machine with less confinement margin => lower cost**

Plasma Elongation and Triangularity –> produced by currents in the poloidal field coils



Elongation = b / a:

- Ratio of "b" axis to plasma radius "a"
    - Larger Elongation increases plasma cross section and thus allows larger current for the same global edge "q"
    - Larger Elongation increases edge magnetic shear – confinement , β !!

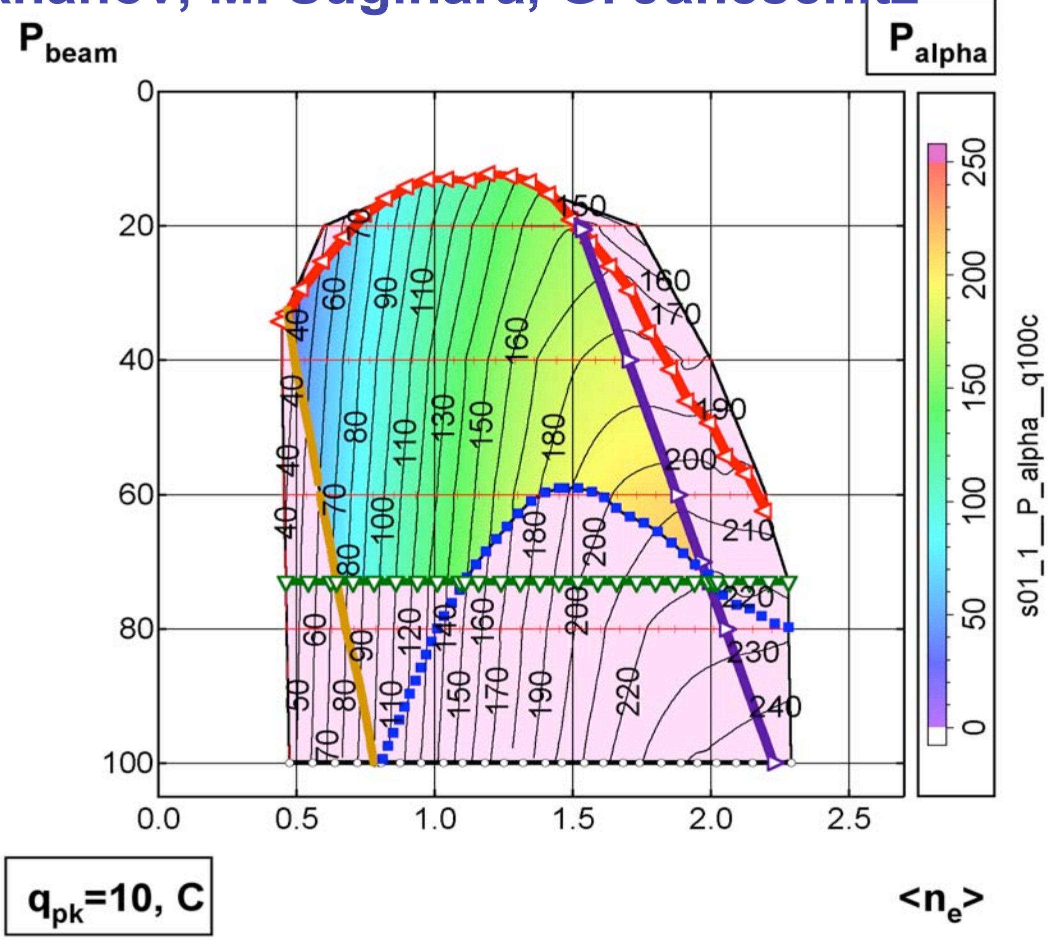
Triangularity = c / a:

- Ratio of distance from plasma center to upper and lower X-points (or turning points) "c" to plasma radius "a"
    - Higher triangularity increases edge magnetic shear – confinement, β !!

ITER Operation Space in H-mode predicted by an Integrated Plasma Model

G. Pacher, H. Pacher, Y. Igitkhanov, M. Sugihara, G. Janeschitz

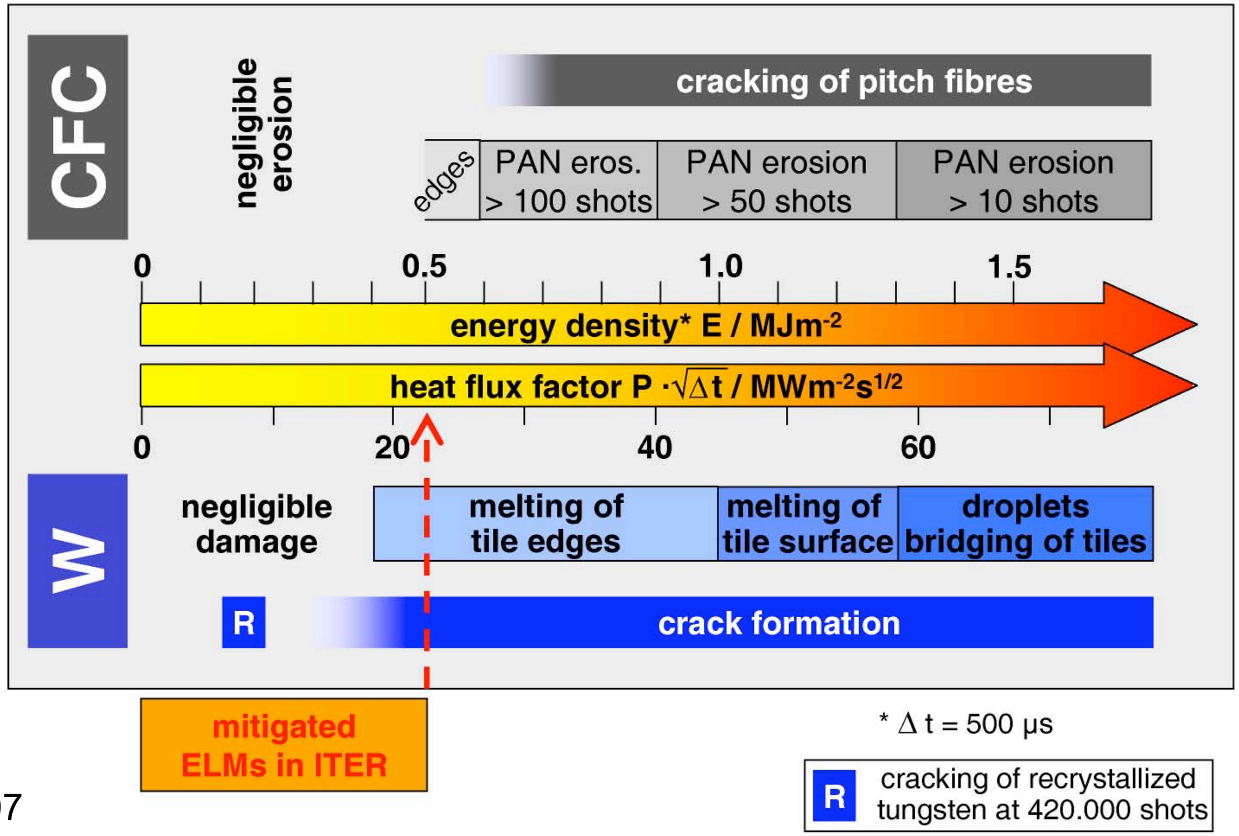
- One-dimensional modelling of the plasma core:
- Two dimensional modelling of the SOL and Divertor
- Physics based empirical model for the Pedestal
- Turbulence fluid model for the core (Multimode – Lehigh – University - Bethman)
- Multimode a bit optimistic !



Operational and objective limits:

Q=5, LH transition, low temperature limit on alpha power, auxiliary power, edge density limit

ELM Control / Mitigation



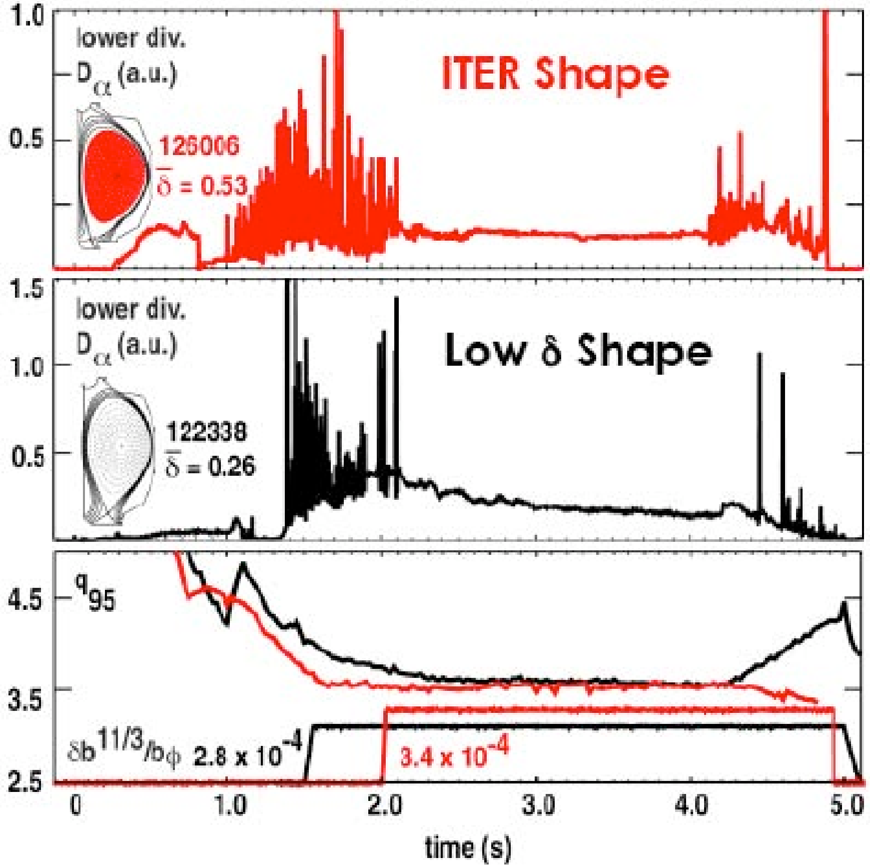
J Linke et al, 2007

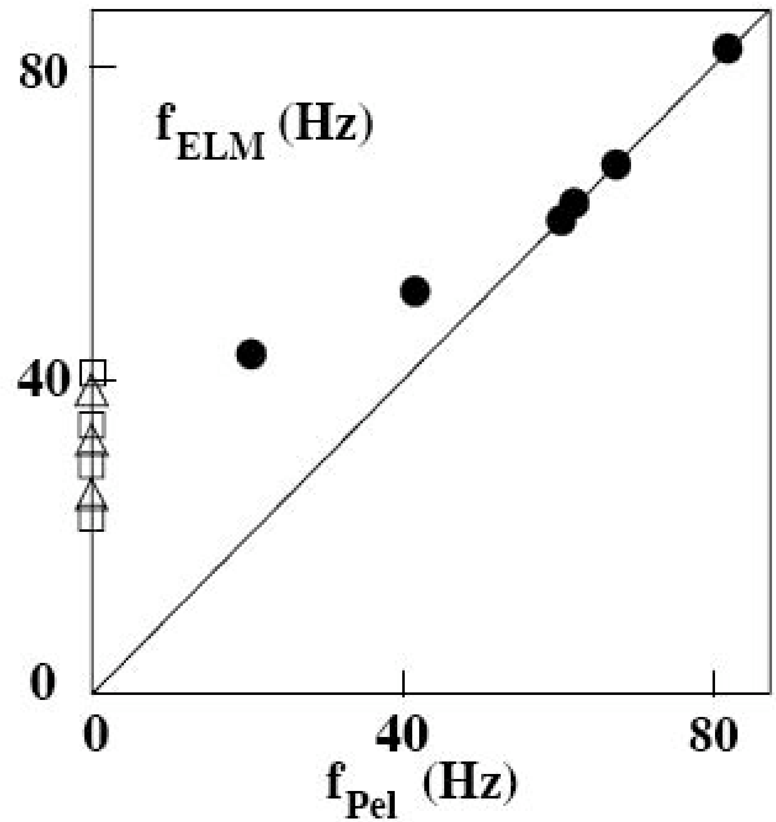
- **Recent predictions indicate that uncontrolled ELM heat pulse amplitude in ITER will produce energy densities at the divertor target of ~10MJm^{-2} - an order of magnitude above tolerable level for divertor PFCs:**
  - techniques for ELM suppression or mitigation essential

ELM Control/ Mitigation Techniques

DIII-D Magnetic Control

AUG Pellet Pacemaking

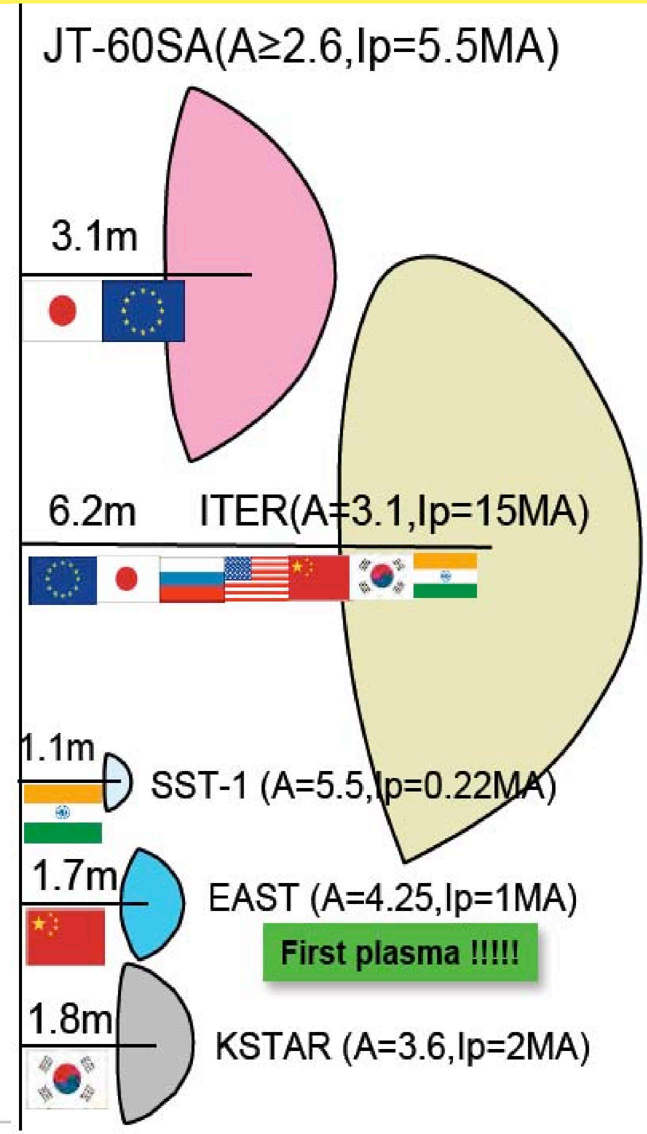




- **Two principal approaches are currently under development for ITER:**
  – edge ergodization by RMP coils – see foreseen design below
  – pellet pacemaking

Family of Superconducting Tokamaks in the World needed for development of Steady State Scenarios

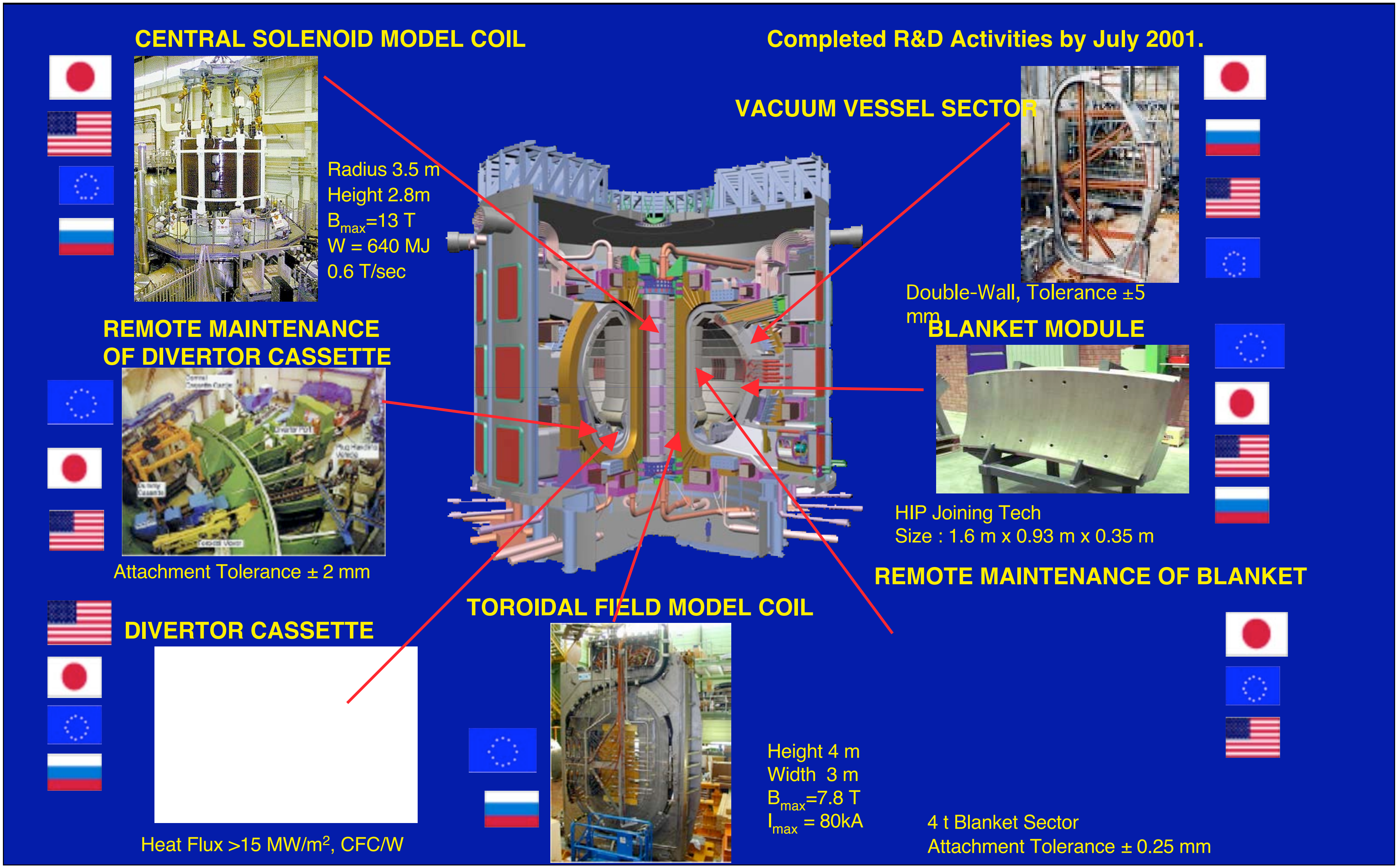
- The small Tokamaks are starting to operate shortly or have already started to operate
- They are all in countries (India, China Korea) who are practically Fusion newcomers
- The small machines will allow to push fusion research towards steady state operation and will be essential to prepare the ground for the two large projects
  - Operation - ITER - 2018 and JT60SA - 2014
- The US is also discussing at the moment to construct a large experiment as a satellite to ITER but no concrete designs have yet emerged



ITER Technology Challenges / Developments

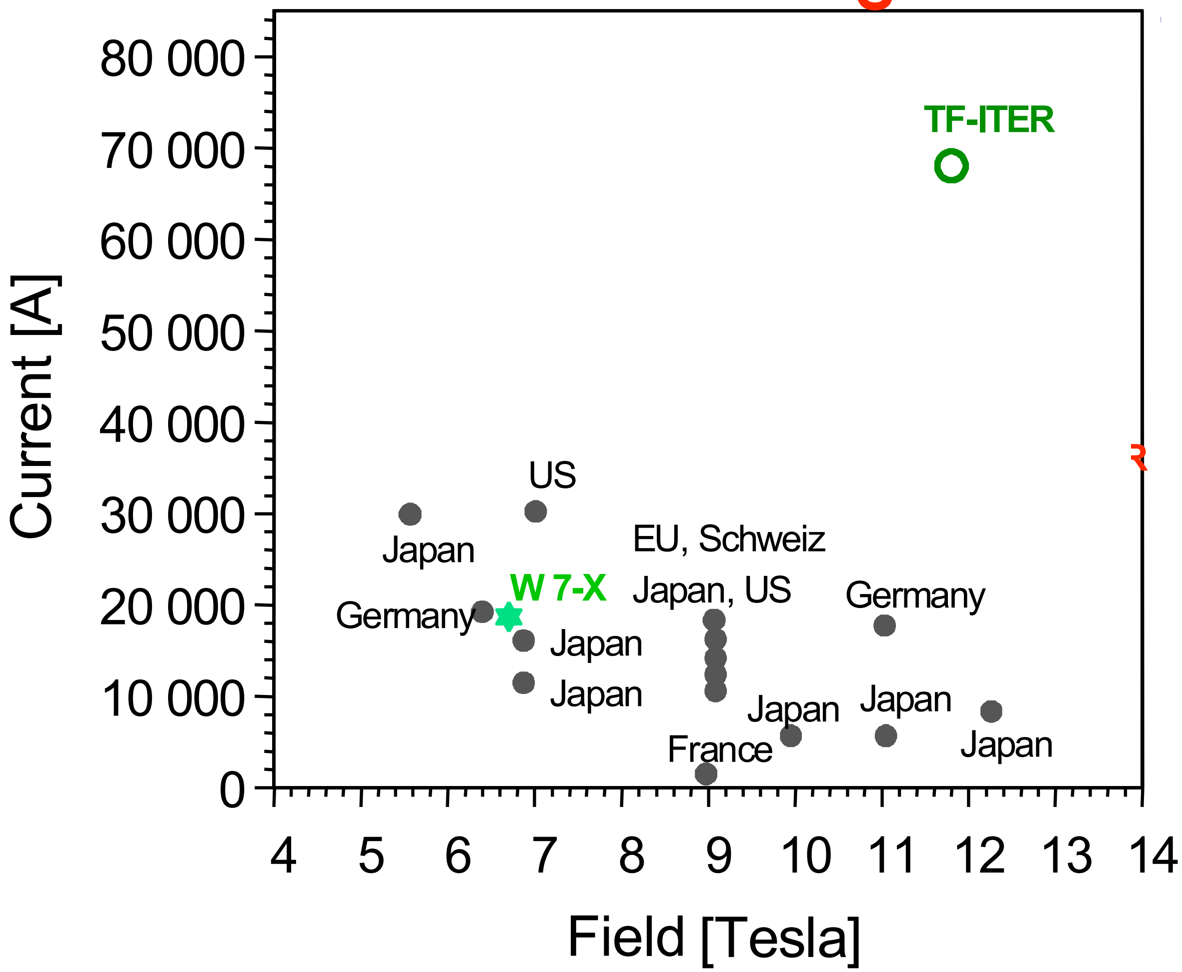
- The development of high field large superconducting magnets
  - Requirements seemed unfeasible 20 years ago
- The development of Divertor High Heatflux Components
  - A large step in development to achieve 20 MW/m_
- The development of Remote Maintenance
  - Thought to be impossible by engineers from nuclear industry
- The development of the DT fuel Cycle
  - A key for ITER and DEMO – a challenge up to today
- The development of Heating and Current Drive systems
  - A challenge for all systems envisaged, not fully solved today
- The development of ITER compatible Diagnostics
  - A step from laboratory type systems to a reliability similar to space

The ITER Design and Technology has been underpinned by R&D



Magnet Development: Current-Field-Chart (Lorentz-Force)

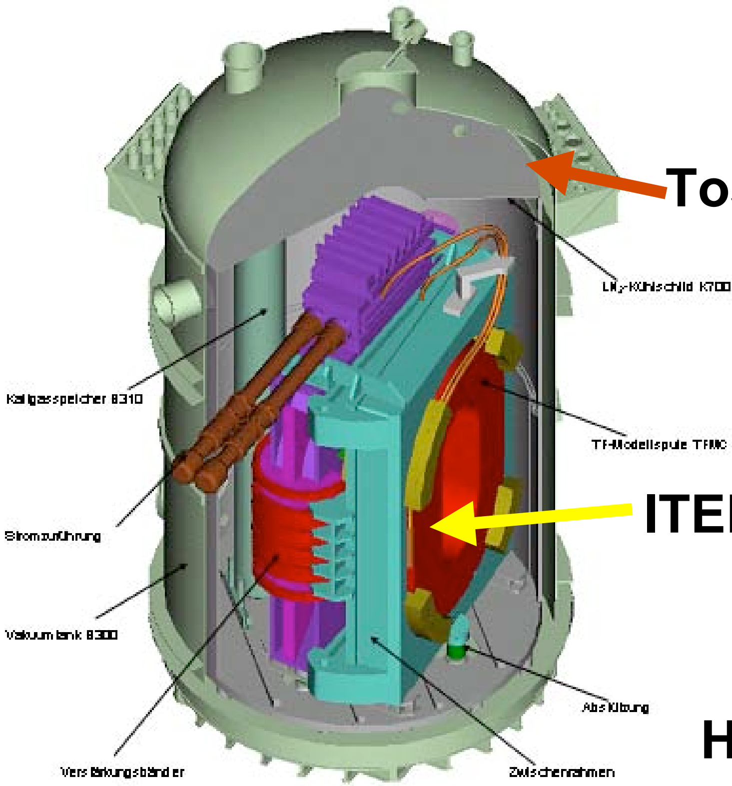
Situation today – gives confidence to be able to built the ITER magnets



Situation at the beginning of the 90th

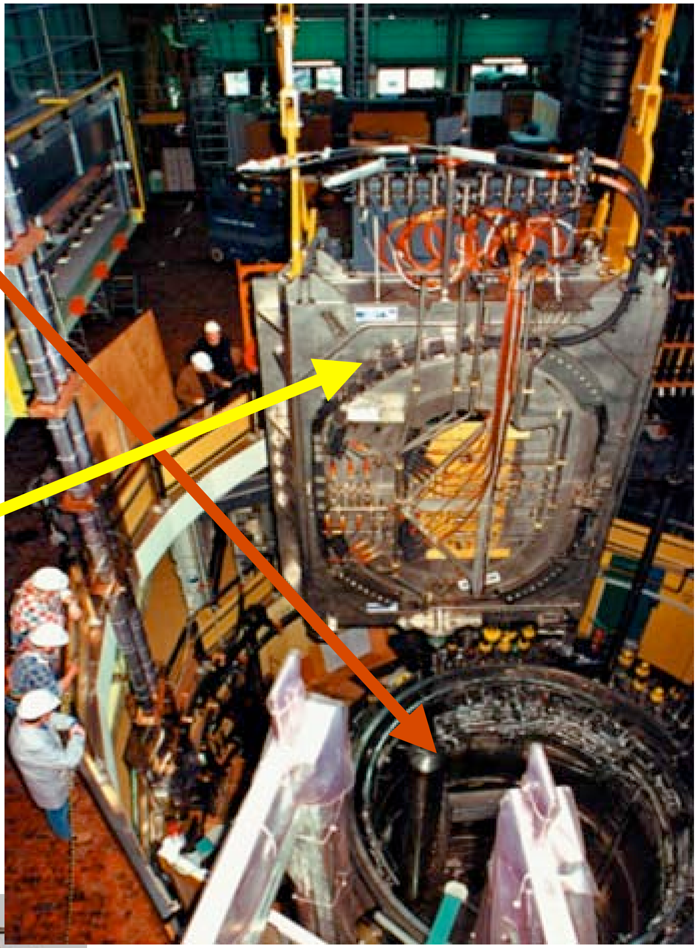
Superconducting coils from earlier experiments

ITER Model Coil, High Temperatur SC Current Leads

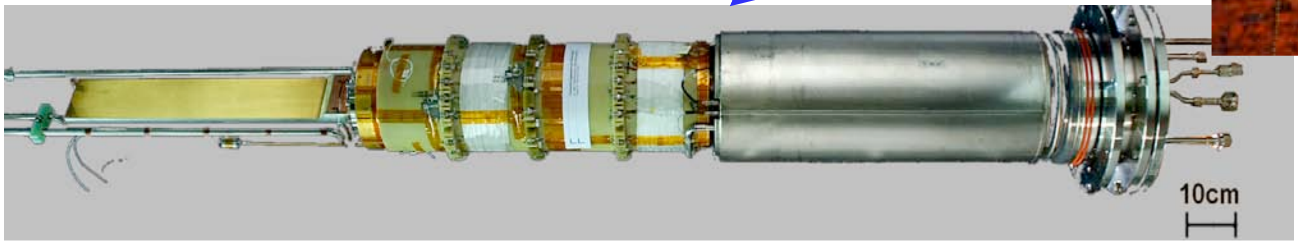


Toska Facility

ITER Model Coil



HTS Current Lead



FZK - Germany

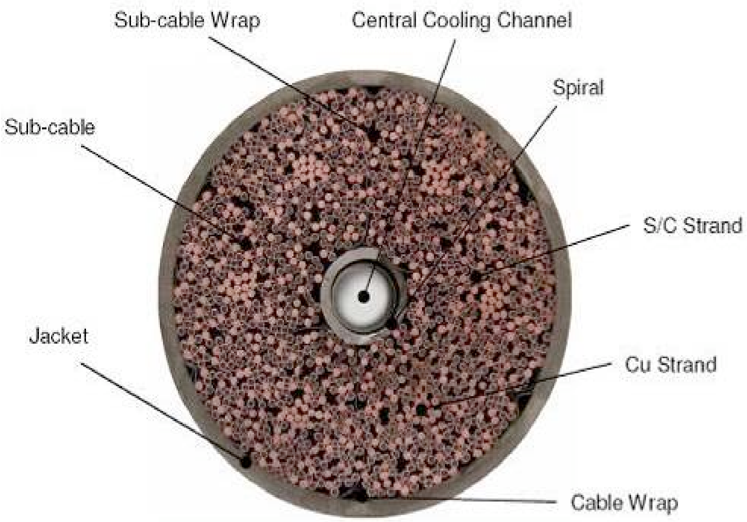
CS Model Coil R&D

Closing of the Test Cryostat (JA)

Overall Features of ITER Magnets
4 Main Systems, all superconducting

System	Energy GJ	Peak Field	Total MAT	Cond length km	Total weight t
Toroidal Field TF	41	11.8	164	82.2	6540
Central Solenoid	6.4	13.0	147	35.6	974
Poloidal Field PF	4	6.0	58.2	61.4	2163
Correction Coils CC	-	4.2	3.6	8.2	85

ITER Conductors



X-section of 70kA ITER TF Conductor (CEA)

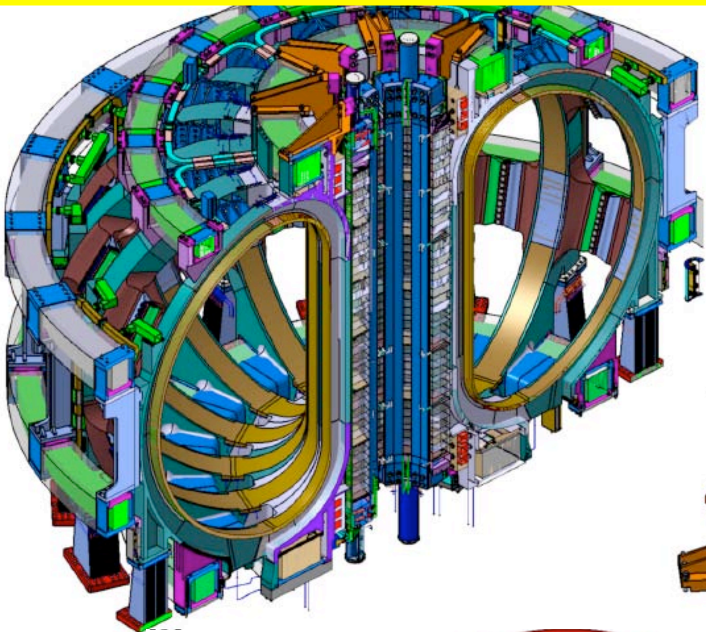
- ITER coils are wound from Cable-In-Conduit Conductors (CICC's), relying on superconducting multifilament composite strands mixed with pure Cu strands/cores.
- The strands are assembled in a multistage rope-type cable around an open central cooling spiral.
- The cable and its spiral are inserted inside a stainless steel conduit which provides helium confinement.

Cooling Spiral

Final-Stage Cable (NFRI)

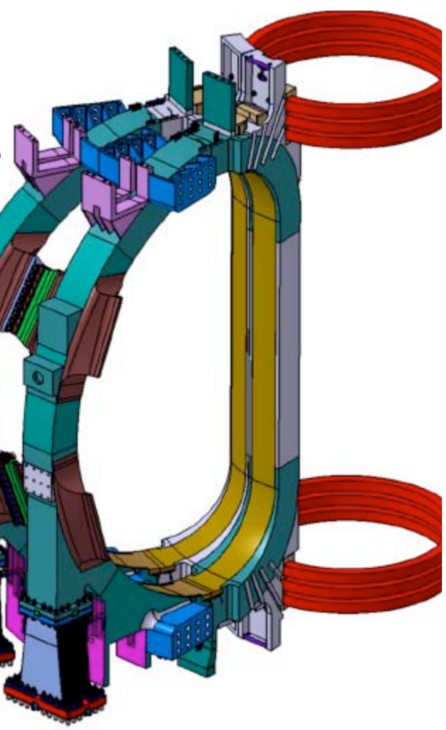
Stainless Steel Conduit (ASIPP)

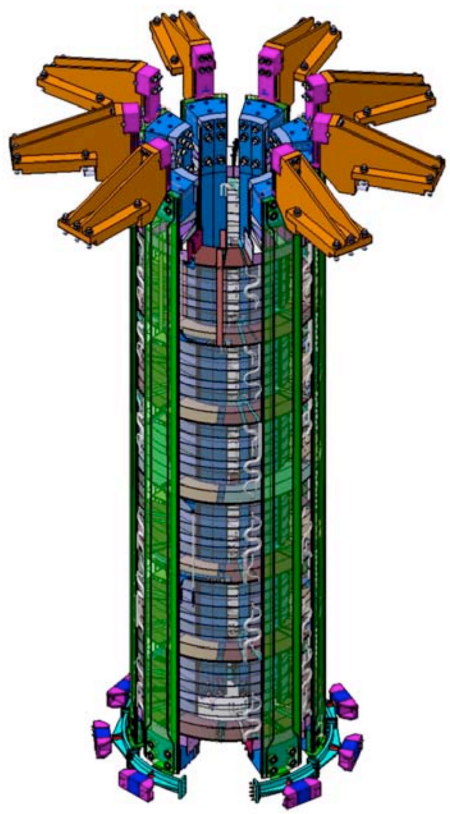
ITER Magnets are Produced in a World wide Collaboration



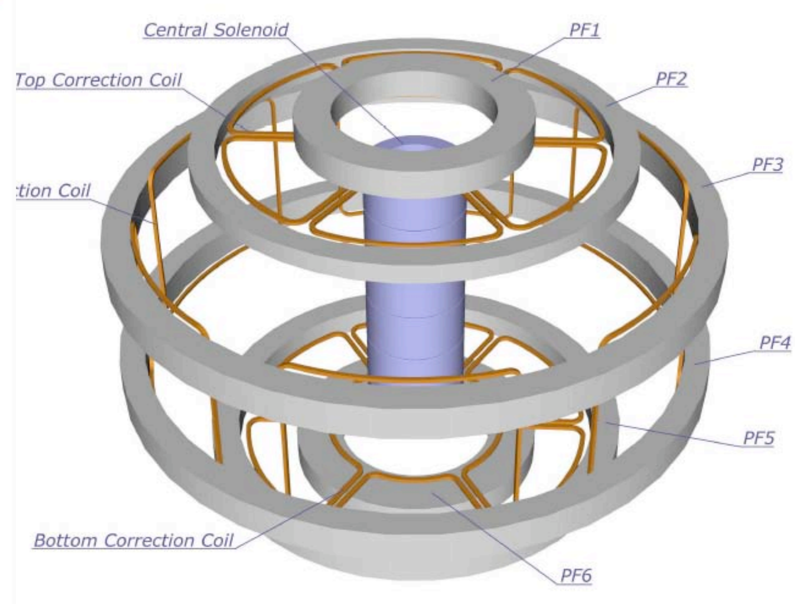
- The ITER magnet system is made up of
  - _ 18 Toroidal Field (TF) Coils,
  - _ a 6-module Central Solenoid (CS),
  - _ 6 Poloidal Field (PF) Coils,
  - 9 pairs of Correction Coils (CC).

Pair of TF Coils





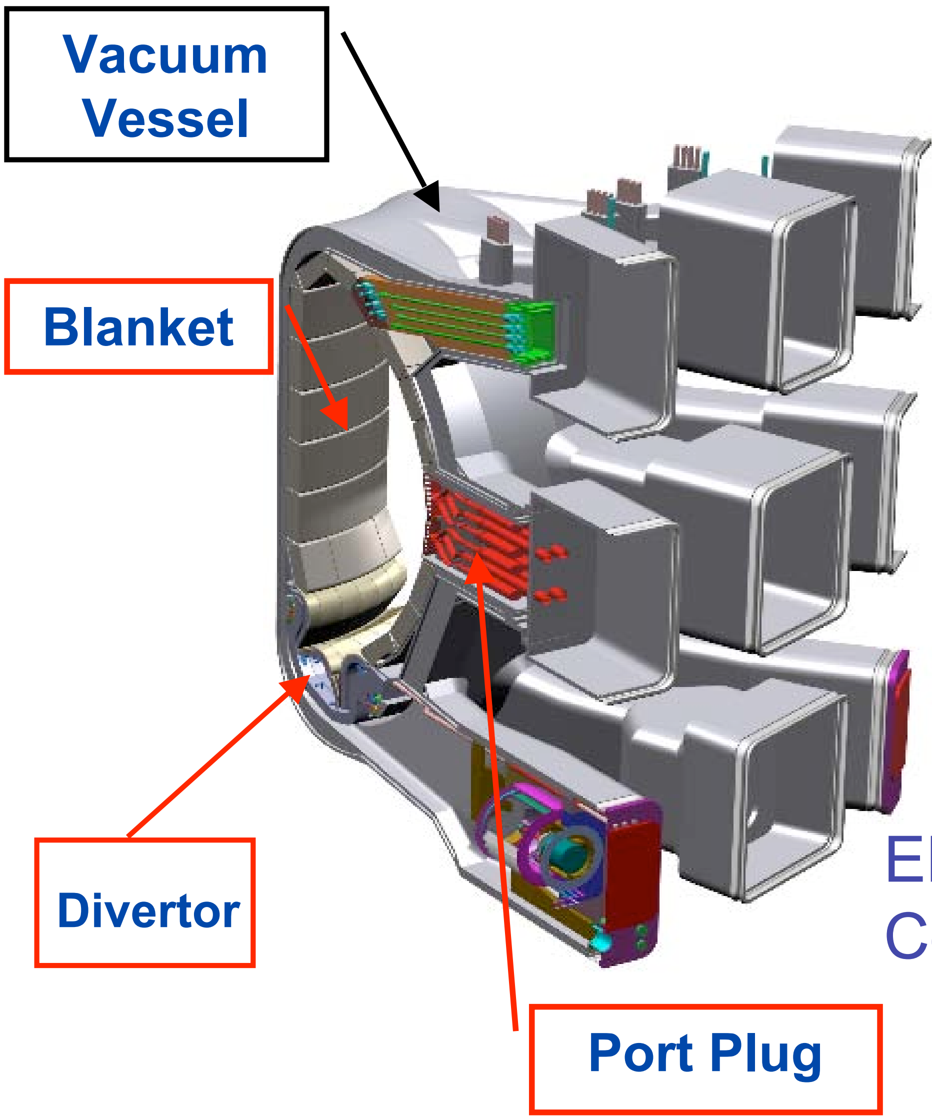
CS Coil

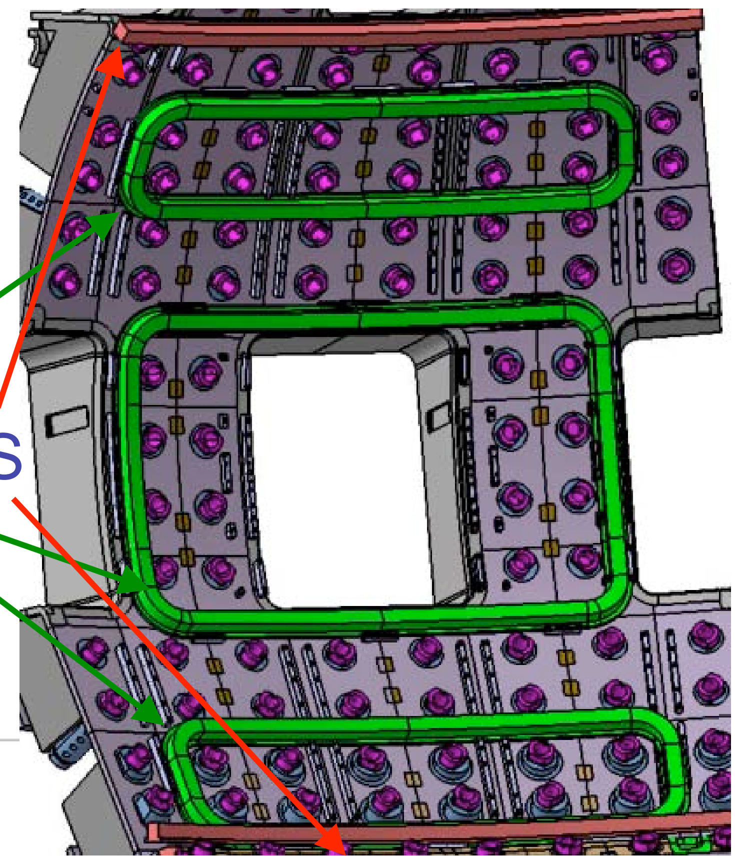


PF & CC Coils

Vessel / In-Vessel ITER Components

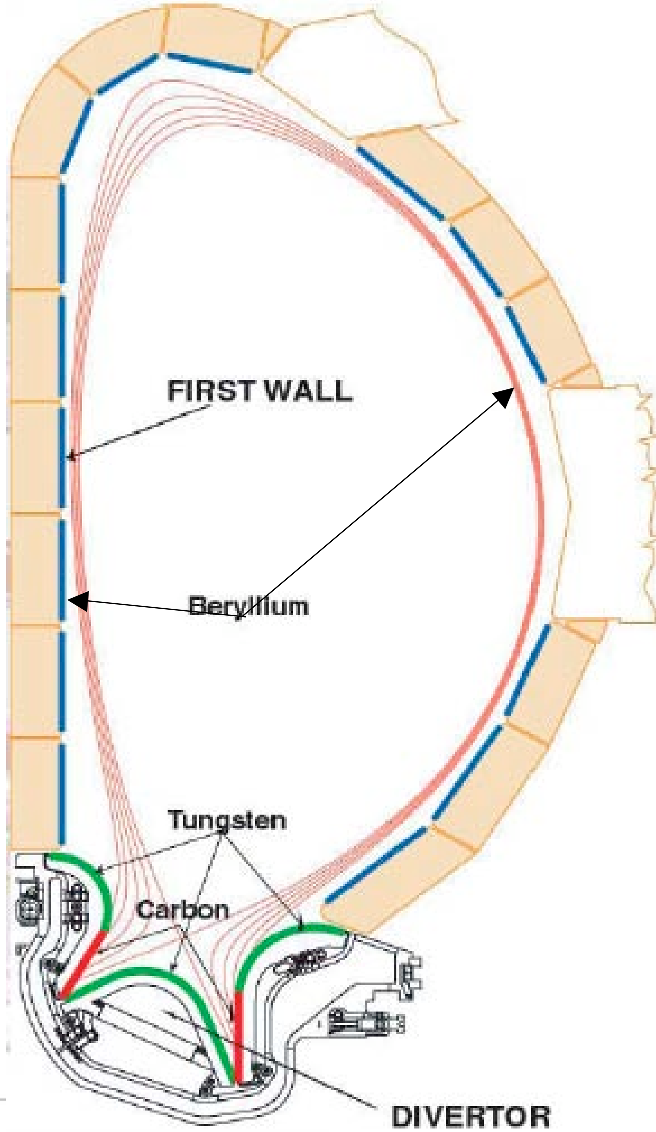
- The vacuum vessel is lined by modular removable components: : **blanket modules, divertor cassettes, ELM / VS coils** and **port plugs** (heating antennae, diagnostics and test blanket modules) All these removable components are mechanically attached to the VV.





Plasma Facing Components - Challenges

- CFC divertor targets (~50m^2):
  - erosion lifetime (ELMs!) and tritium codeposition
  - dust production
- Be first wall (~700m^2):
  - dust production and hydrogen production in off-normal events
  - melting during VDEs
- W-clad divertor elements (~100m^2):
  - melt layer loss at ELMs and disruptions
  - W dust production - radiological hazard in by-pass event



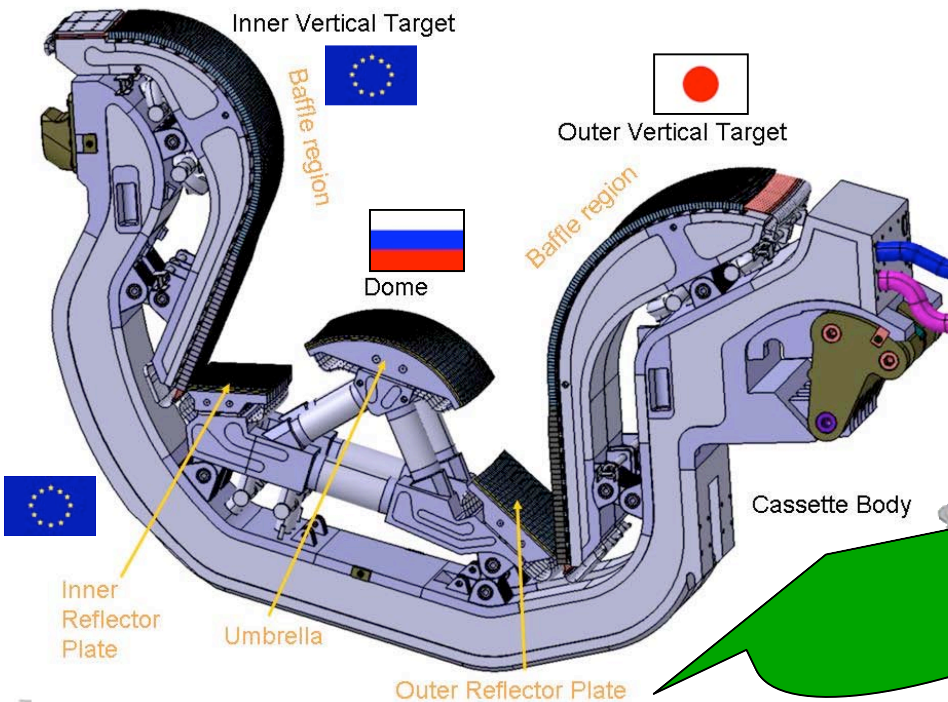
Divertor

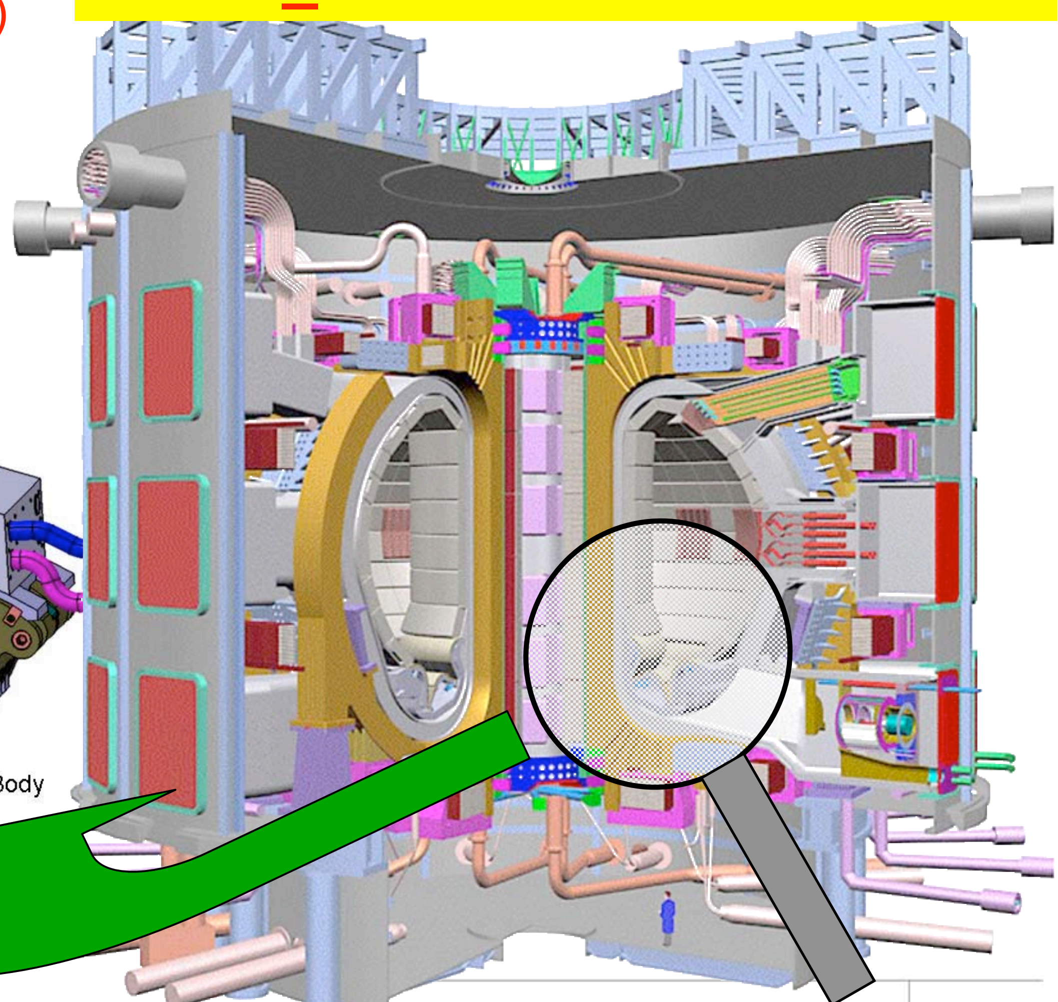
Divertor system main functions :

• Exhaust the major part of the plasma thermal power (including alpha power)

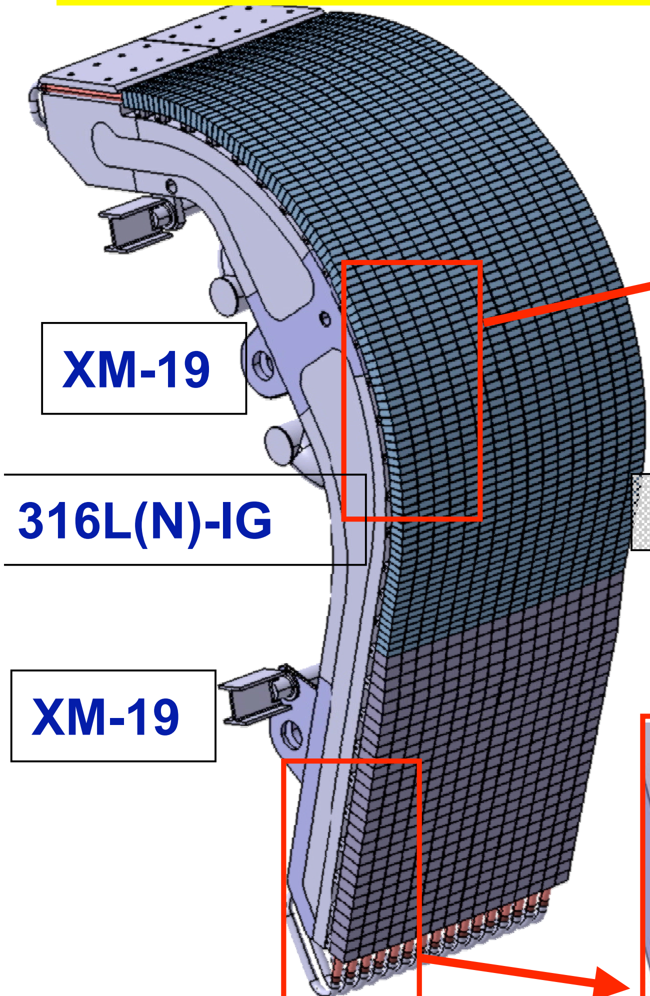
• Minimize the helium and impurities content in the plasma

Challenge to develop HHF Componets capable of 20 MW/m_





Plasma-Facing Components Vertical Targets



XM-19

316L(N)-IG

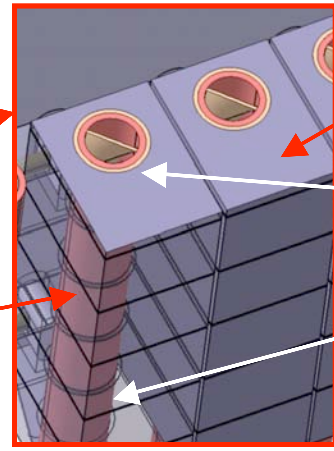
XM-19

Twisted tape: To increase the margins against the Critical Heat Flux

W monoblock

5 MW / m²

Smooth Tube



W monoblock

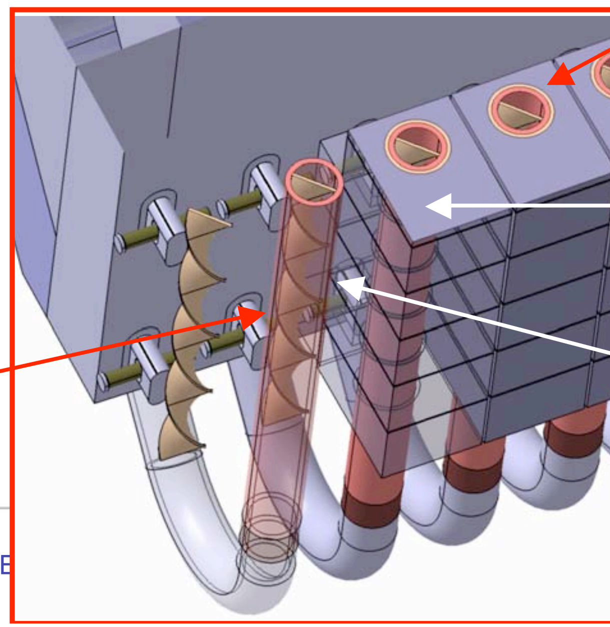
Copper Interlayer

CuCrZr Heat Sink

CFC monoblock

10 MW / m²

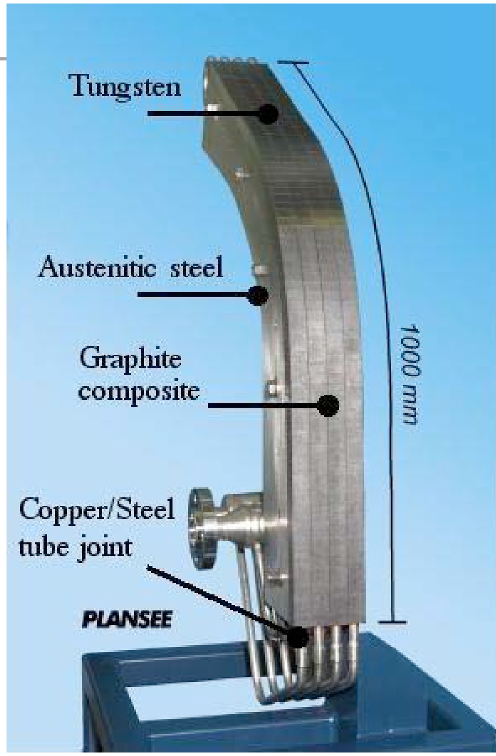
20 MW / m² 10 sec



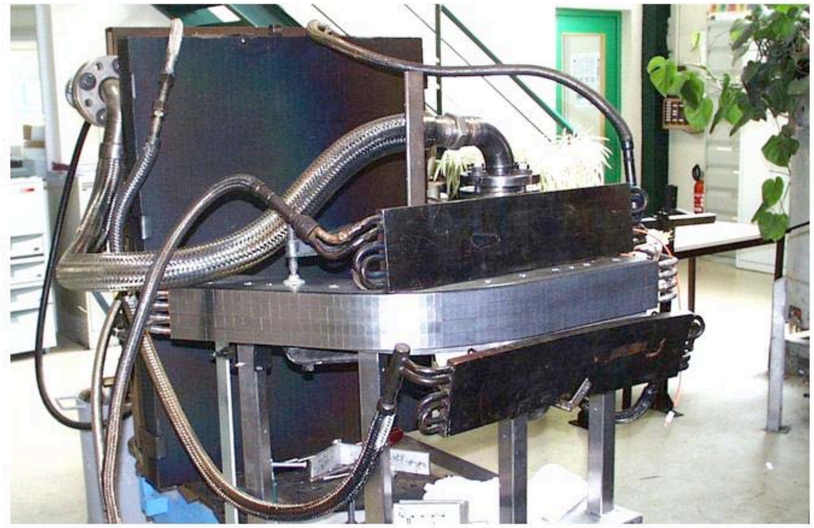
CFC monoblock
For first divertor set

Copper Interlayer

CuCrZr Heat Sink



Vertical Target
Full-Scale Prototype



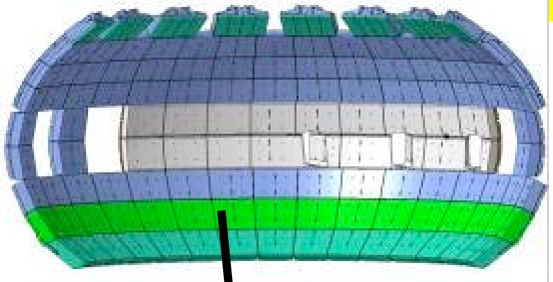
AREVA

- **W monoblocks:**
  10 MW/m² x 1000 cycles
  on other mock-ups:
  20 MW for 1000 cycles

- **CFC monoblock**
  10 MW/m² x 1000 cycles
  20 MW/m² x 1000 cycles
  23 MW/m² x 1000 cycles

2 decades of development to achieve these parameters also after neutron irradiation

Blanket System



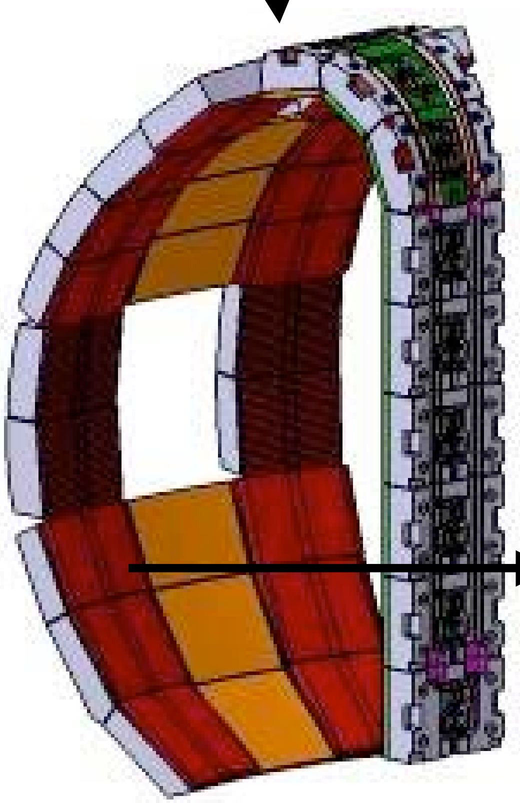
Challenge EM Forces

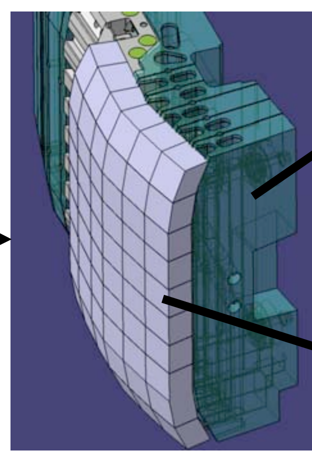
Scope

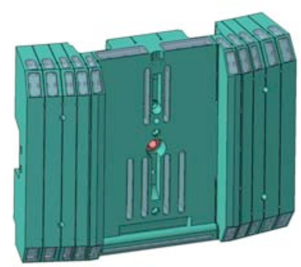
- 440 blanket modules at ~4 ton each
- ~40 different blanket modules

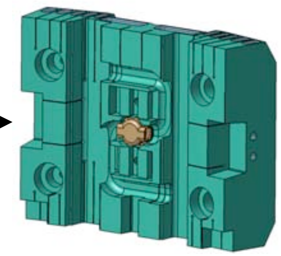
Blanket system main functions :

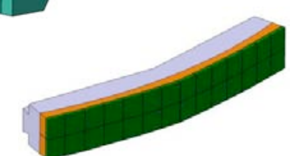
- Exhaust the majority of the fusion power
- Reduce the nuclear responses in the vacuum vessel and superconducting coils

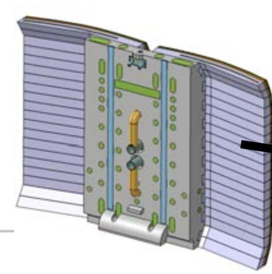


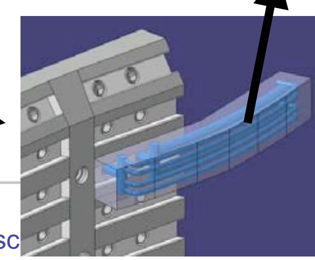




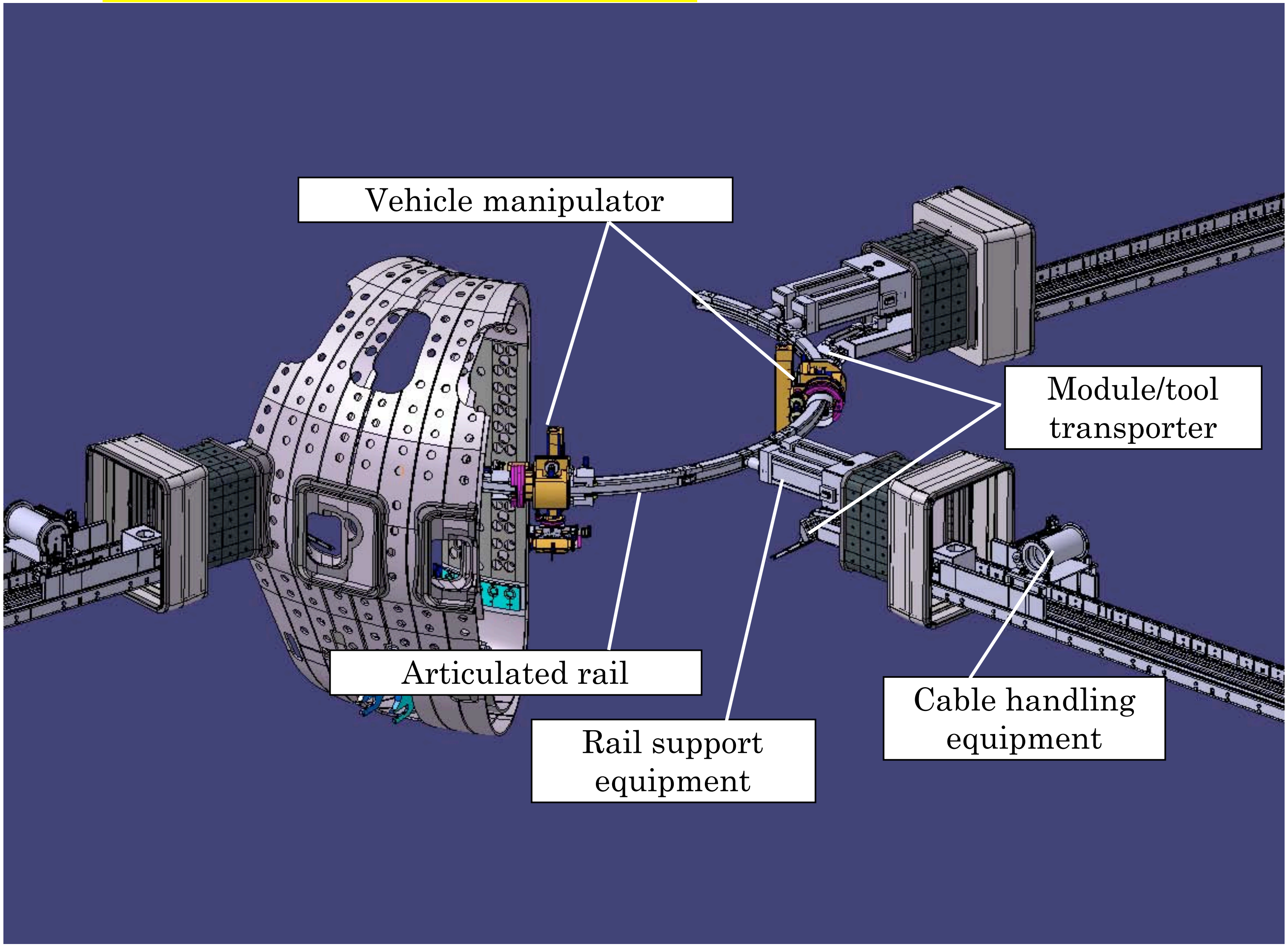








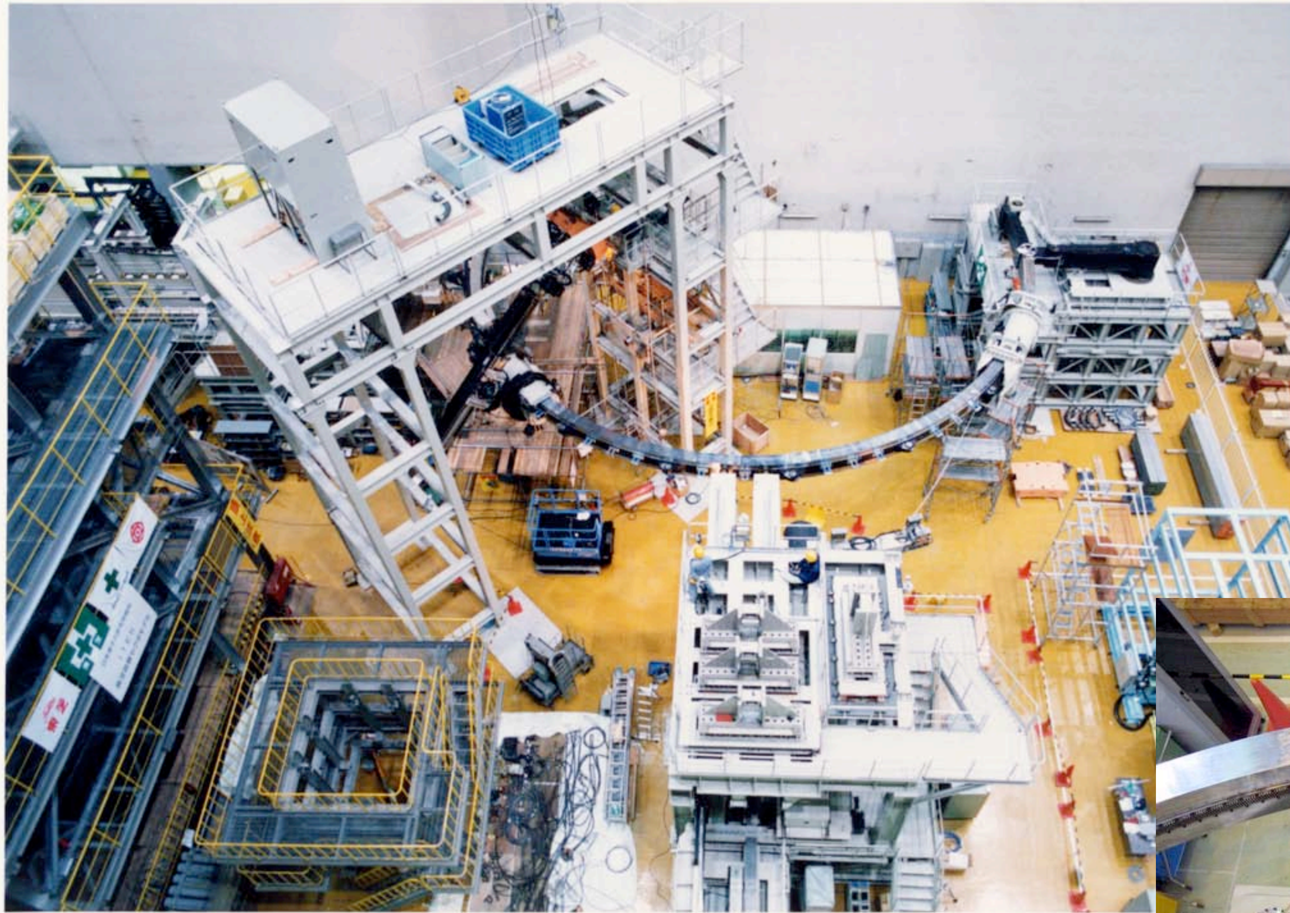
Blanket RH System



Payload: 4ton - (max. 4.5 ton at limited location)

Reach: 1.3 – 3.8m from the rail center - Rail location: R6.2m from machine center

Full scale 180° rail deployment test (1998)



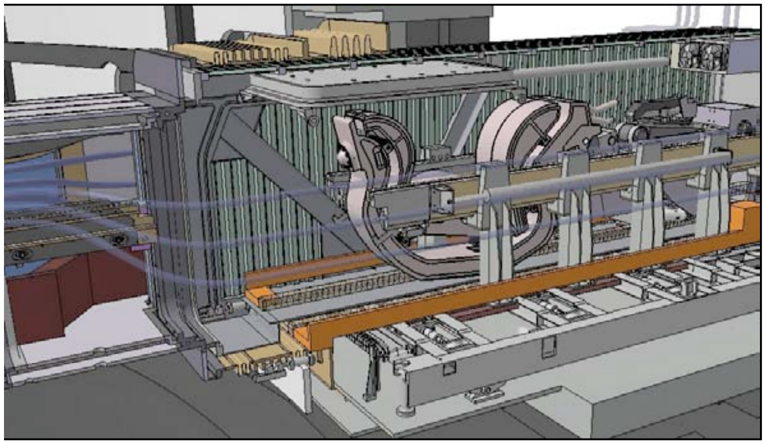
RH system for blanket has shown its ability to perform the job

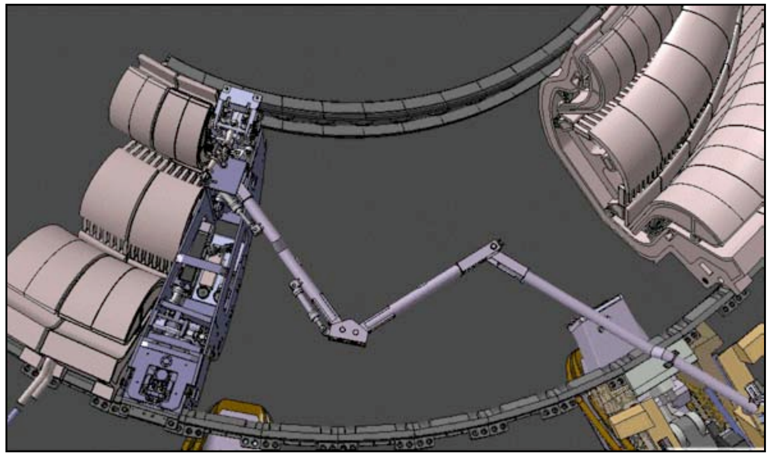
Some issues remain

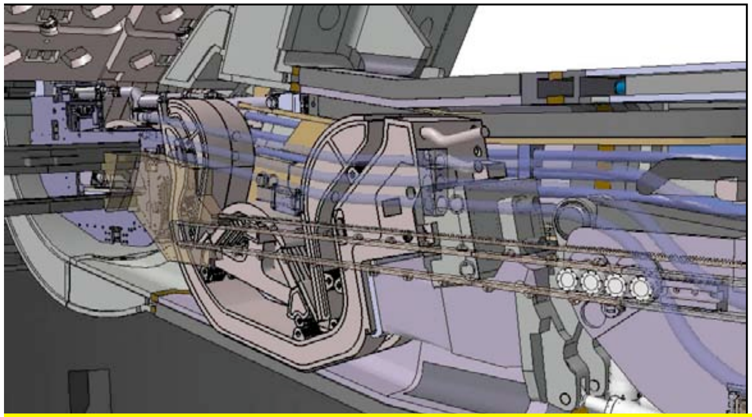
2 decades of development to provide demonstration of feasibility and reliability

In-port rail connection 2008-2009

Divertor RH Equipment demonstrated to work







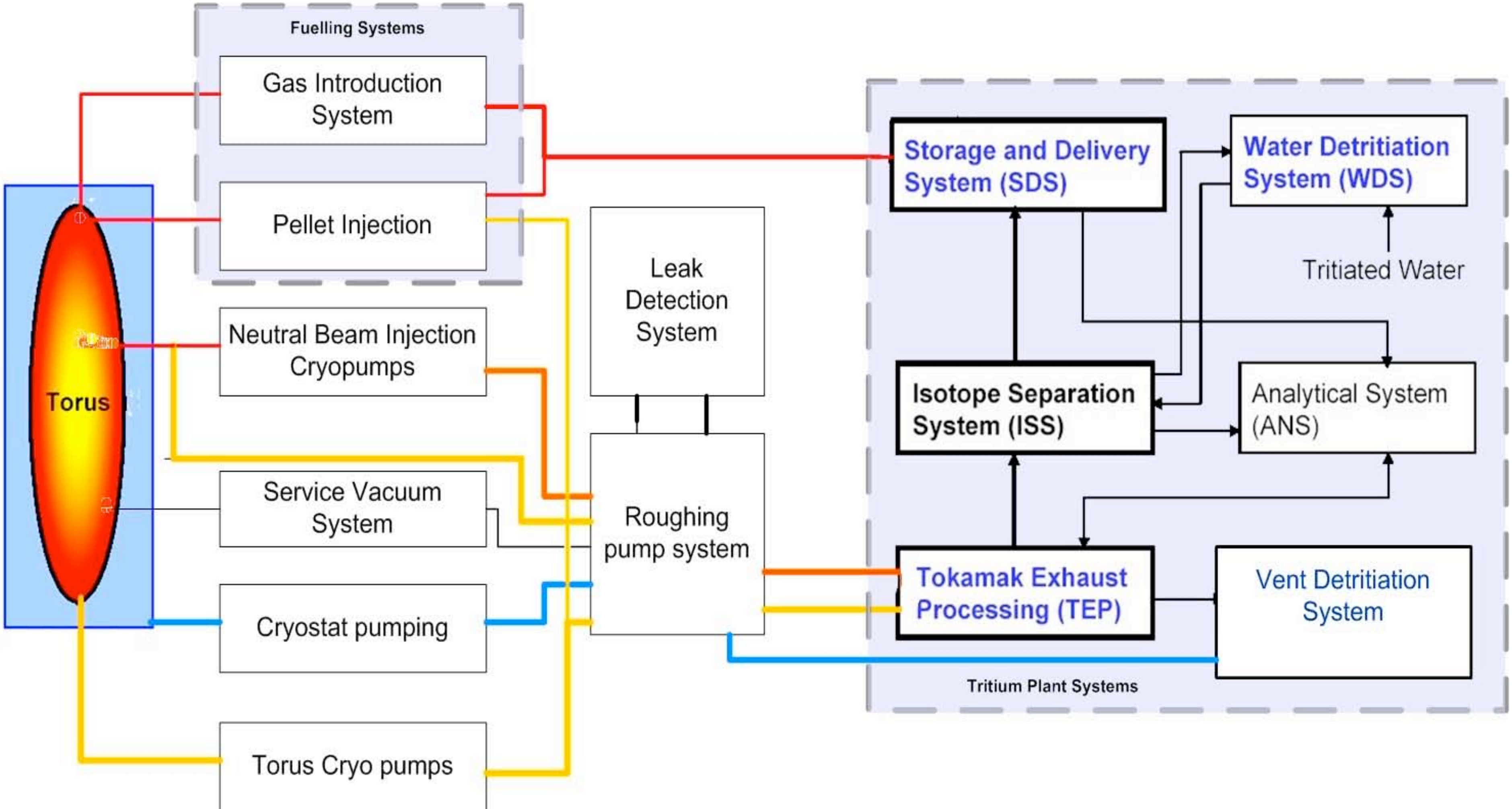
Divertor RH equipment is comprised of
Two main types of “cassette mover”:

- **Cassette Multi-functionl Mover (CMM)**
- **Cassette Toroidal Mover (CTM)**

Each are to be equipped with a dexterous manipulator arm and RH tooling.

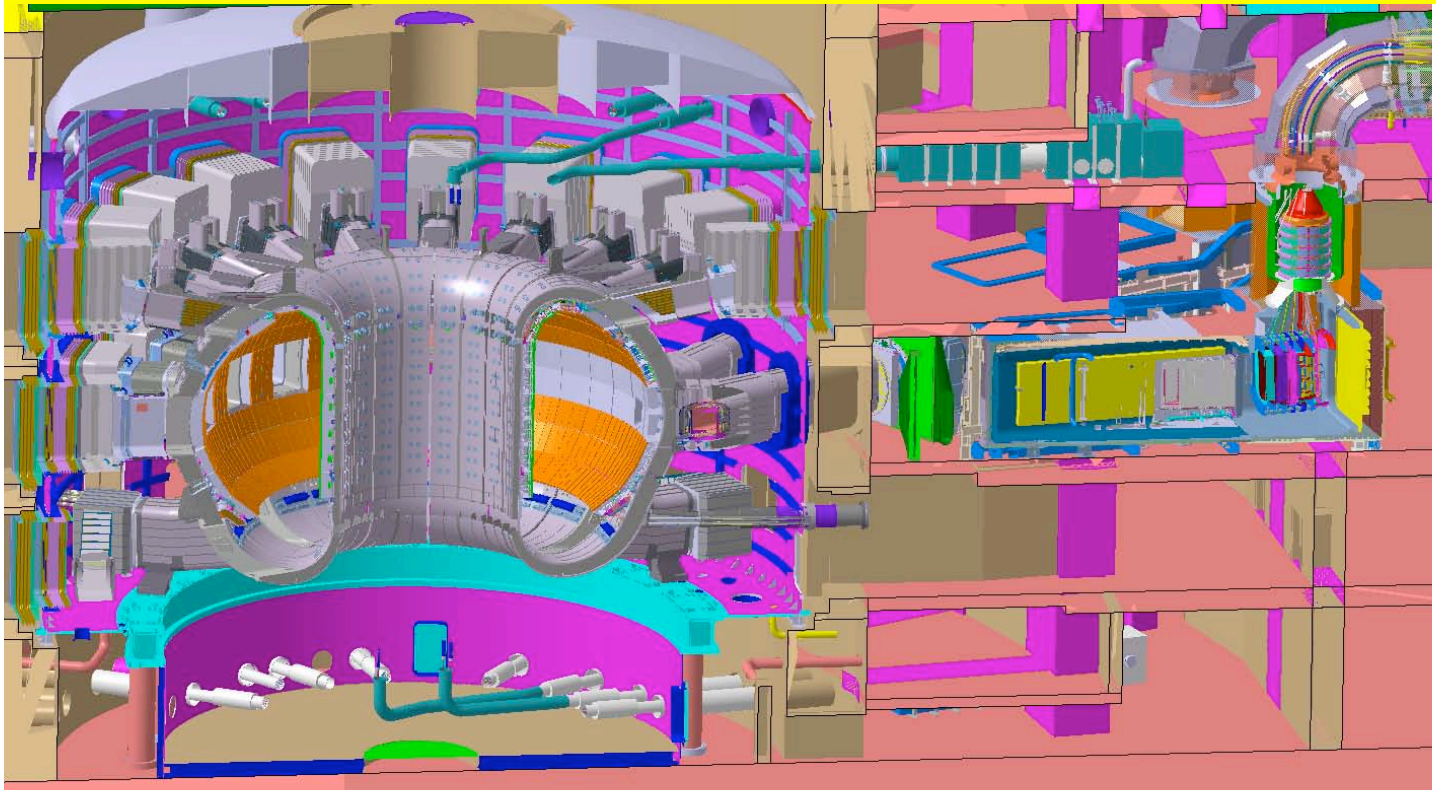
ITER Fuel Cycle

T-Plant handles all exhaust gases – release limits !!



Apart from the cryogenic guard vacuum – exhausts are centralized and controlled

ITER Vacuum Systems



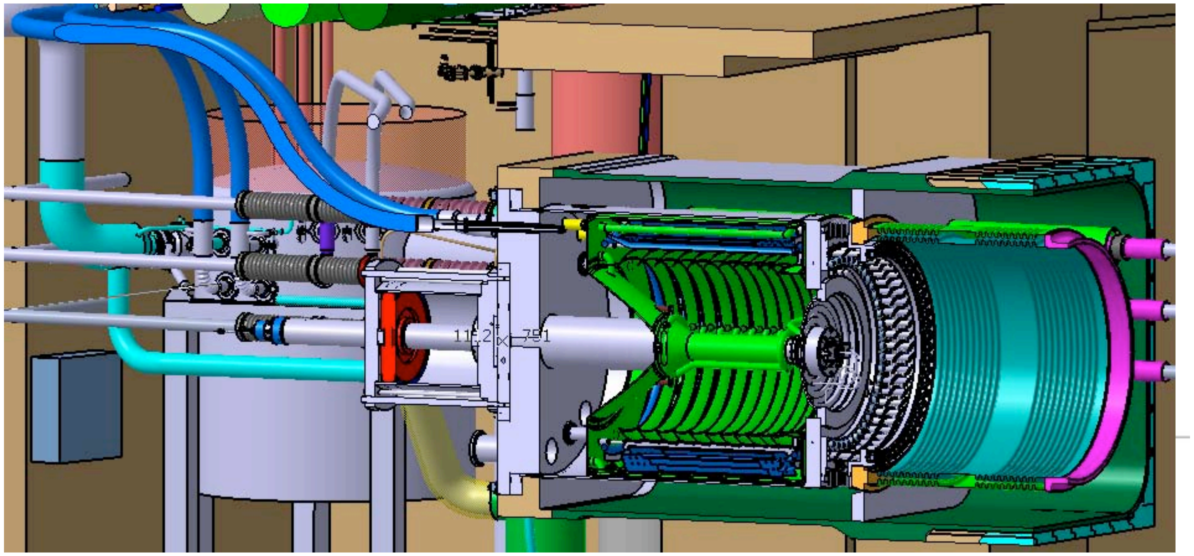
Cryostat vacuum(<10-4 Pa) 8500 m3

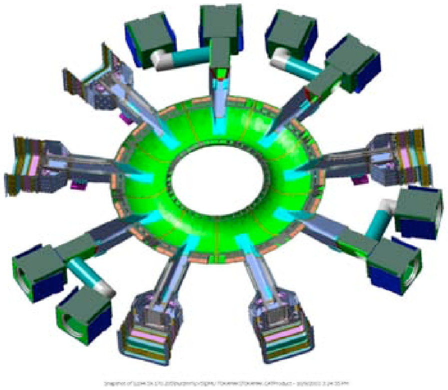
Torus vacuum(~10-6 Pa) 1400 m3

Neutral Beam vacuums(~10-7 Pa) 630 m3 (for 4)

Cryogenic Guard Vacuum

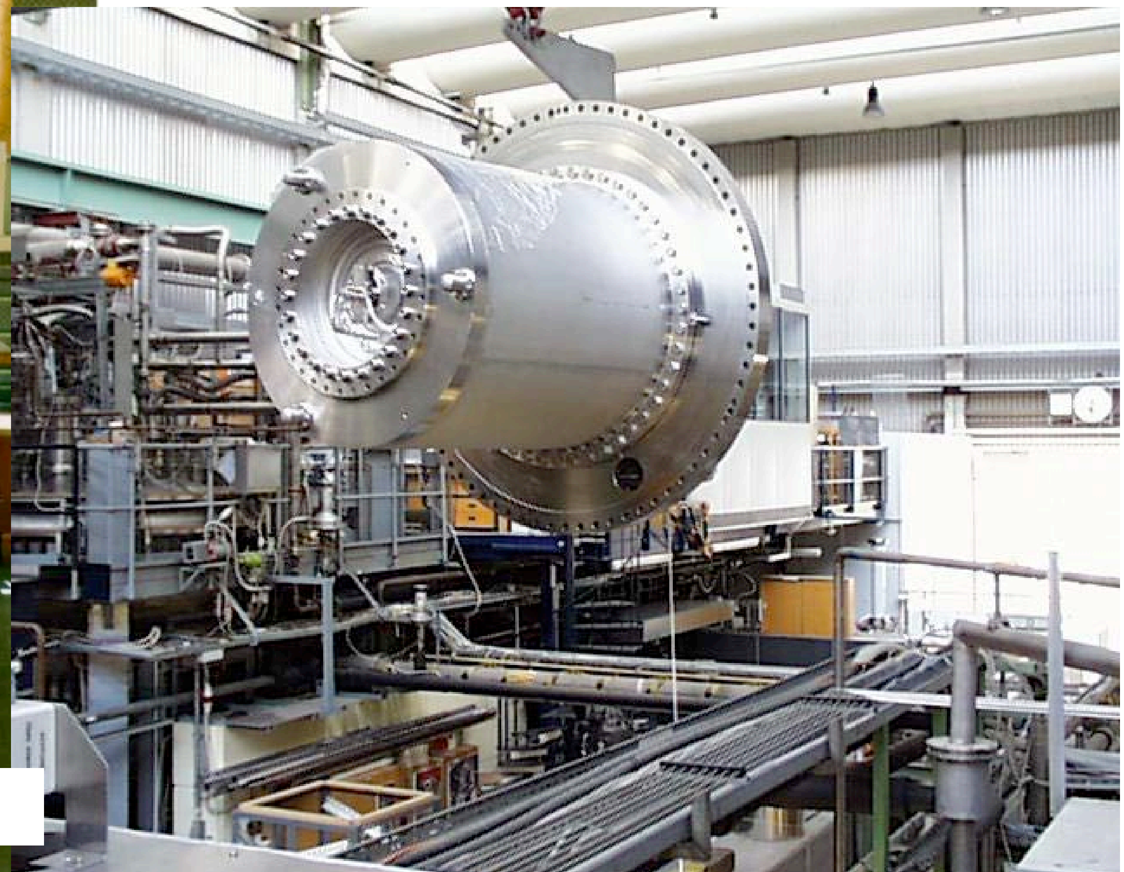
Service Vacuum System (Inc diagnostics)
ICRH and ECRH Vacuums





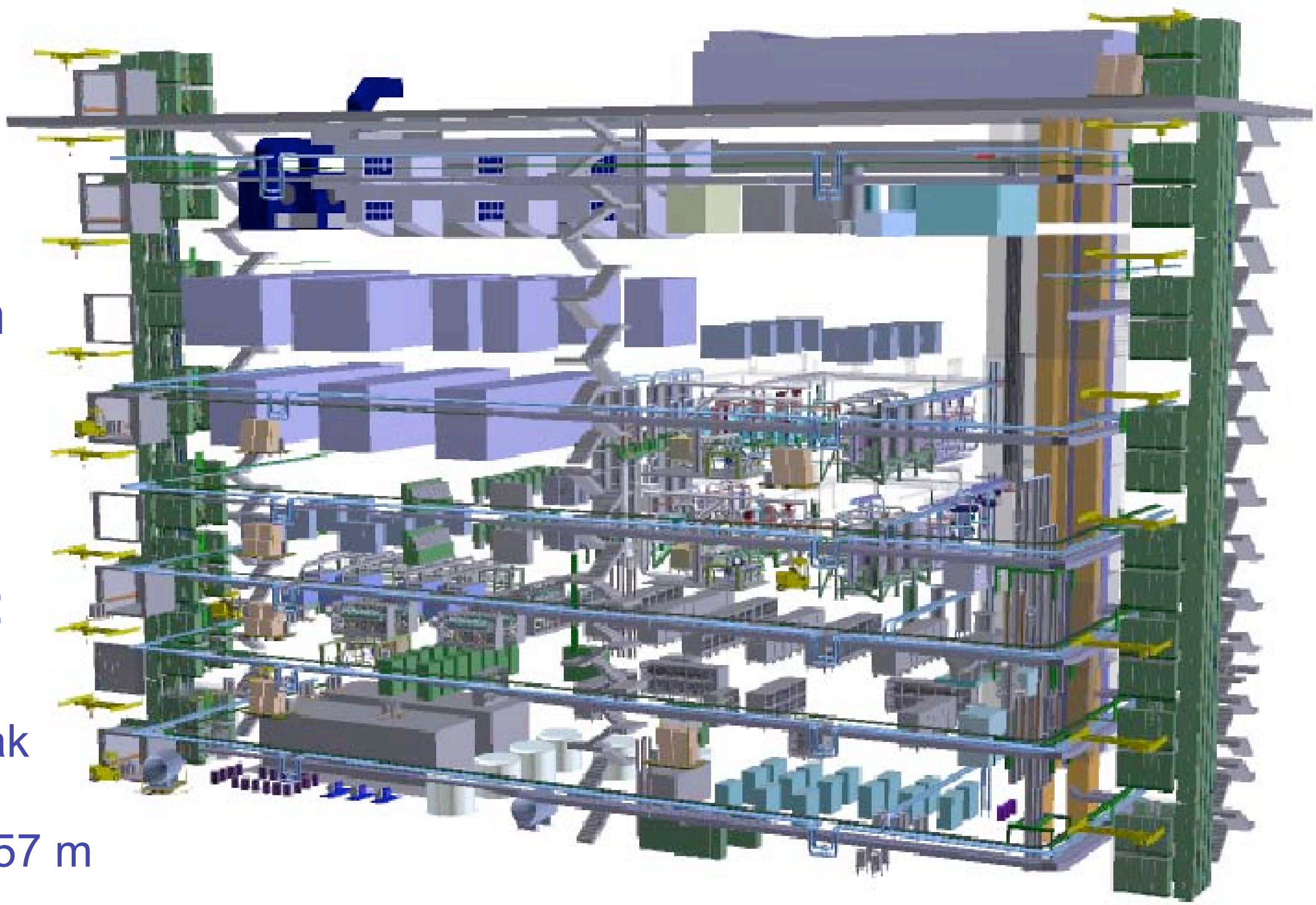
ITER Torus Cryopump Prototype tested in FZK

Successful development of charcoal coating for He pumping in FZK- Germany

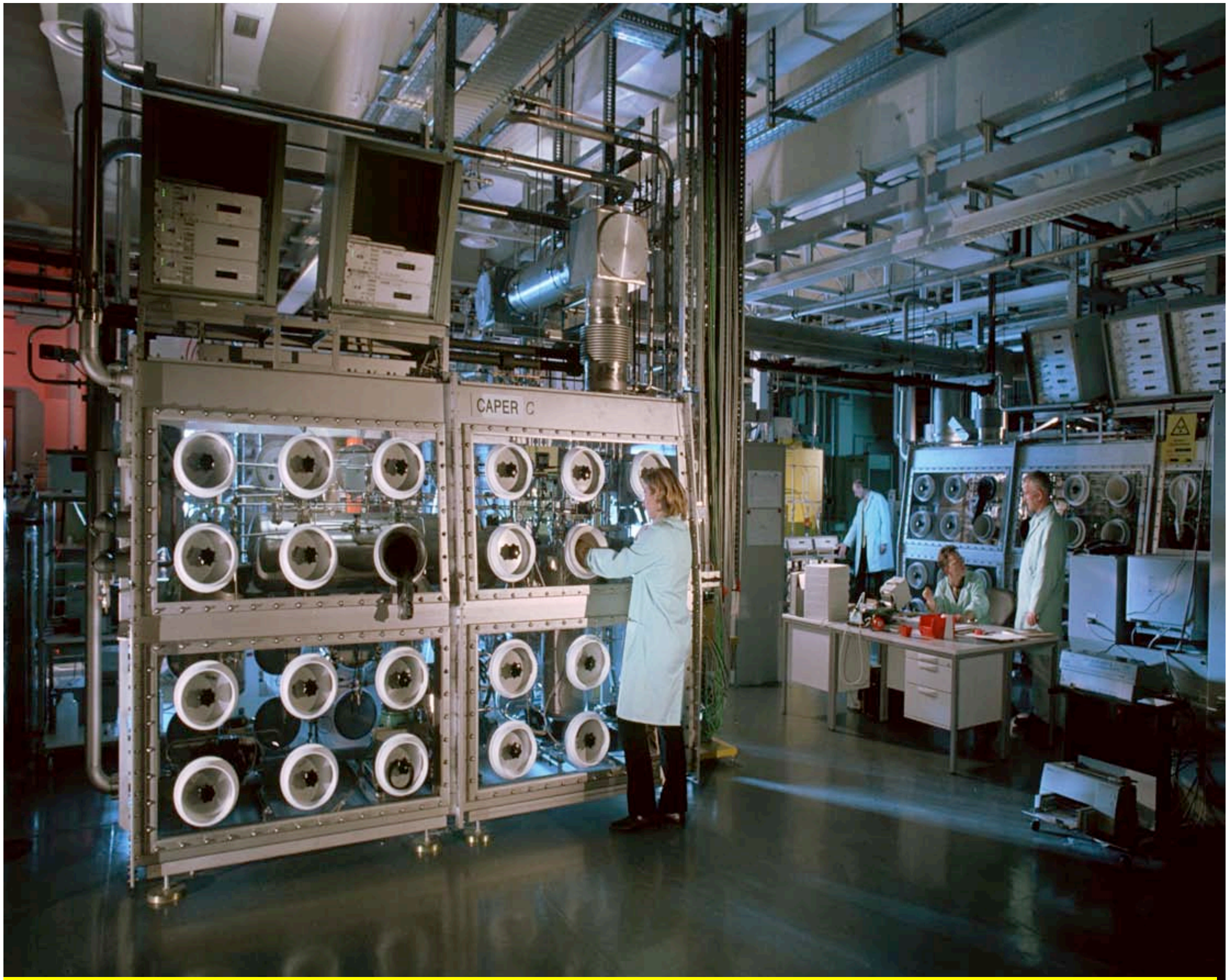


Tritium Plant Building Systems Layout

- 7 Floors
  - 2 below grade
- L = 80 m
- W = 25 m
- H = 35 m
- Release point elevation: 60 m
  - Tokamak building height: 57 m



Tritium Plant R&D in FZK



All Systems validated by R&D in FZK- Germany

T-plant systems

TEP – tokamak exhaust processing

ISS- Isotope separation

WDS- Water Detriation

Analytical System

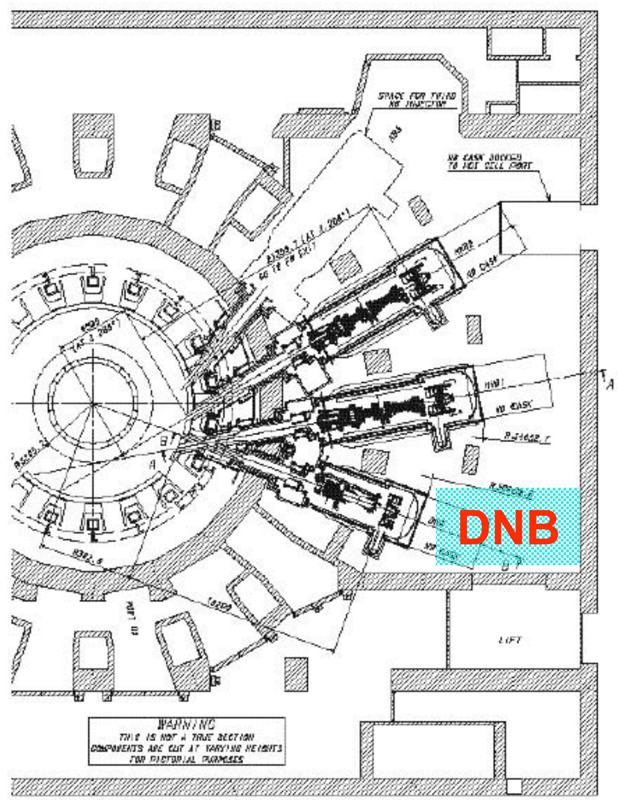
Storage system

ITER Heating and Current Drive

P_{aux} for Q=10 nominal scenario: 40-50MW

Heating System	Stage 1	Possible Upgrade	Remarks
NBI (1MeV –ive ion)	33	16.5	Vertically steerable (z at Rtan -0.42m to +0.16m)
ECH&CD (170GHz)	20	20	Equatorial and upper port launchers steerable
ICH&CD (40-55MHz)	20		$2\Omega_T$ (50% power to ions Ω_{He3} (70% power to ions, FWCD)
LHH&CD (5GHz)		20	$1.8<n_{par}<2.2$
Total	73	130 (110 simultan)	Upgrade in different RF combinations possible
ECRH Startup	2		126 or 170GHz
Diagnostic Beam (100keV, H^-)	>2		

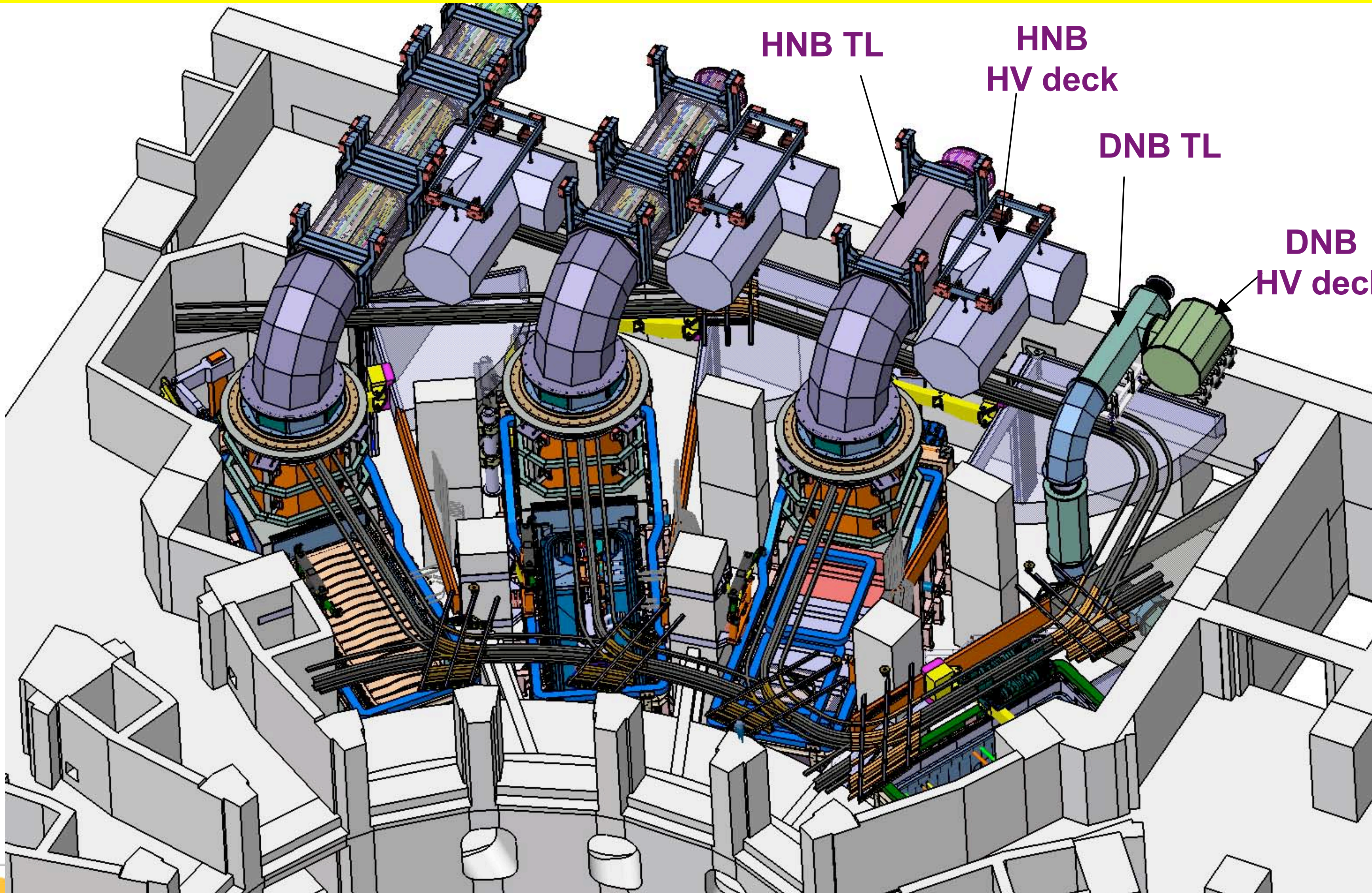
NBI Layout



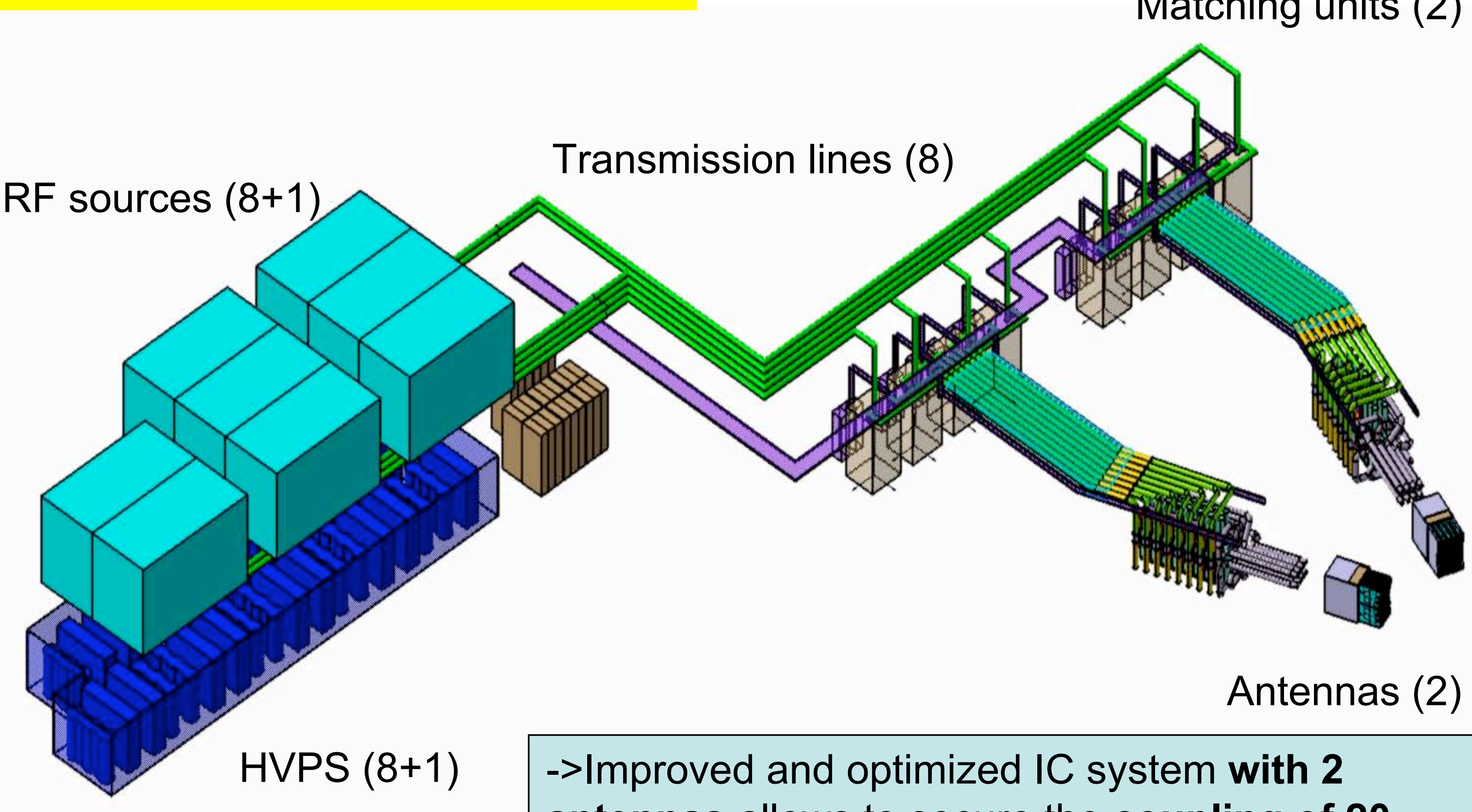
NBI: 1 MeV beams – 16 MW power deposited in the plasma by each

Development is a challenge – test bed under construction in Padua

first heating and the diagnostic beam installed in 2020 second in 2022



IC H&CD system



Challenge is antenna design and coupling to the plasma - TBD

->Improved and optimized IC system **with 2 antennas** allows to secure the **coupling of 20 MW** in all envisaged scenarios (large scrape off, short decay length, 40 to 55 MHz, dipole phasing) with 40 kV max in the circuits
->Efficient resilient matching limit ELM effect: reflected power is kept below 1% of forward power

ECRH - Progress on Gyrotrons

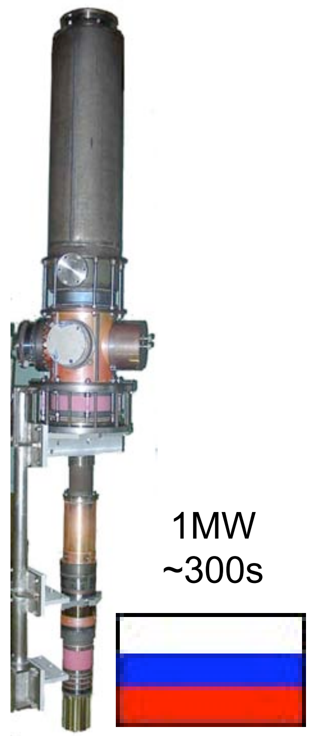
2 MW gyrotrons from EU, 1 MW from JA and RF.
• Diode type gun gyrotrons from EU and RF, triode type from JA (requiring an additional PS for the anode).

Challenge is the development of the sources – good progress

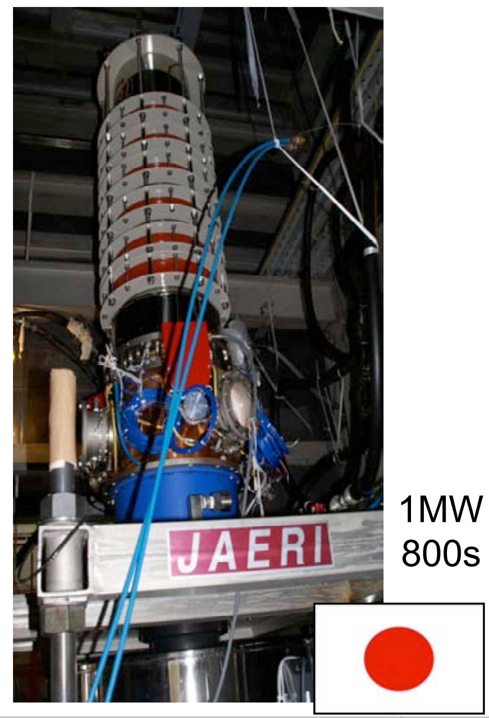
And the development of the lounchers – very good progress

Short pulse prototype

2MW short pulses

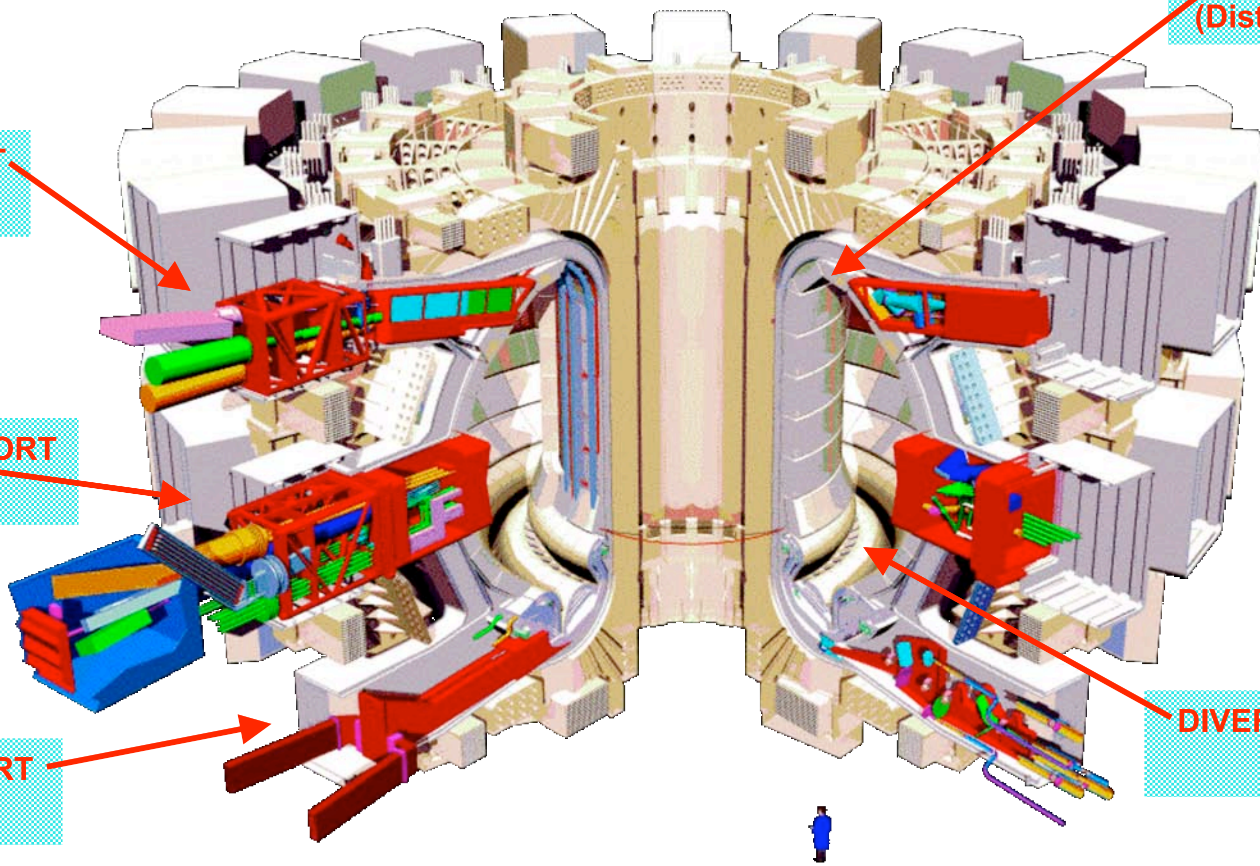


1MW
~300s



1MW
800s

ITER Diagnostics



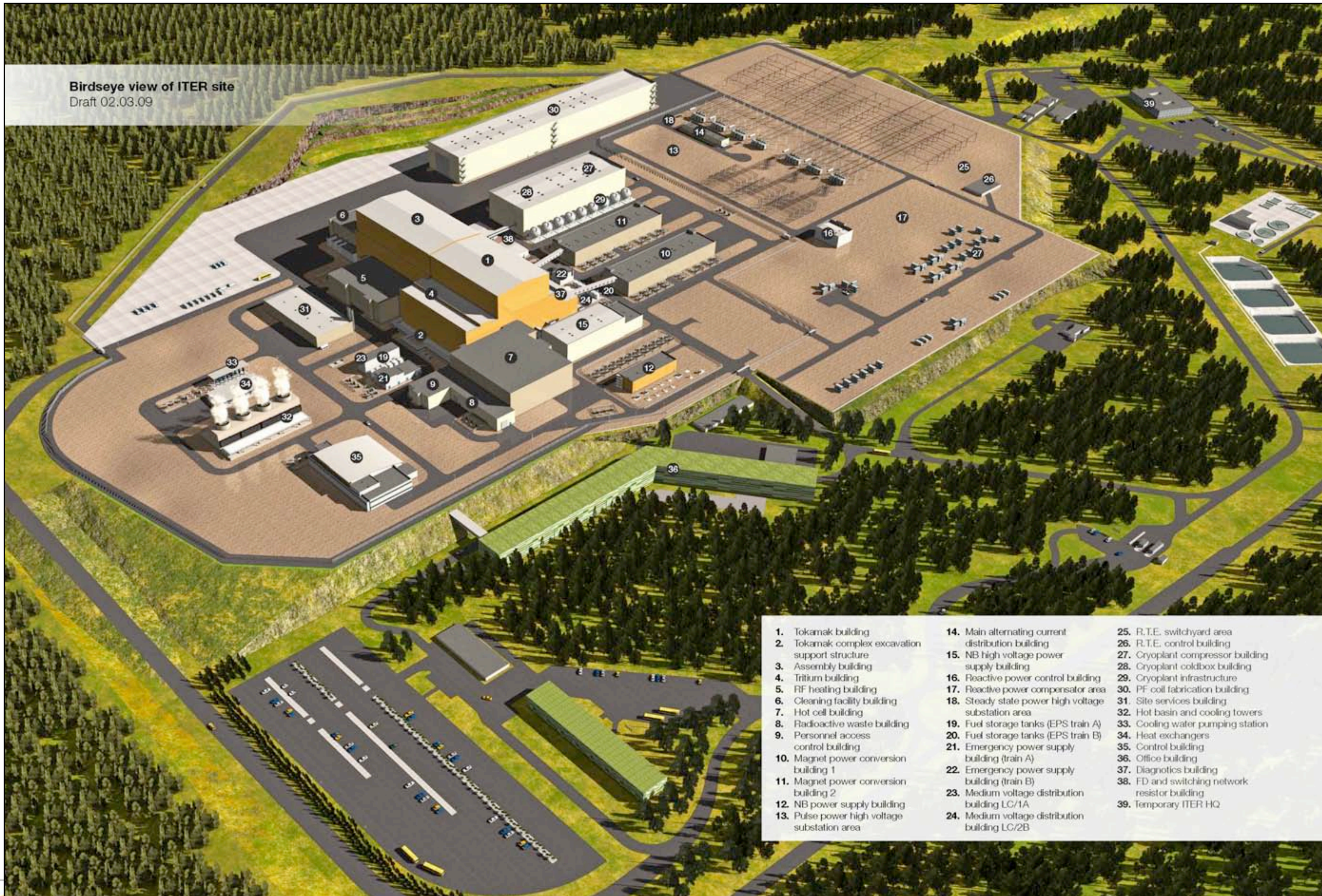
- **About 40 large scale diagnostic systems are foreseen:**
  - Diagnostics required for protection, control and physics studies
  - Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  - Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ….)

ITER - Status

- The ITER Team has been established on the Cadarache site for ~2 years
  - now ~280 team staff on site
- ITER Organization formally established in October 2007
- Design Review carried out to revise Baseline by mid-2008
- On 31 January 2008, the files for DAC (Demande d'Autorisation de Création), including the Preliminary Safety Report, in application of the TSN law, completed and sent to the French Nuclear Authorities **=> construction license**
- The main platform levelling works have finished on the ITER site

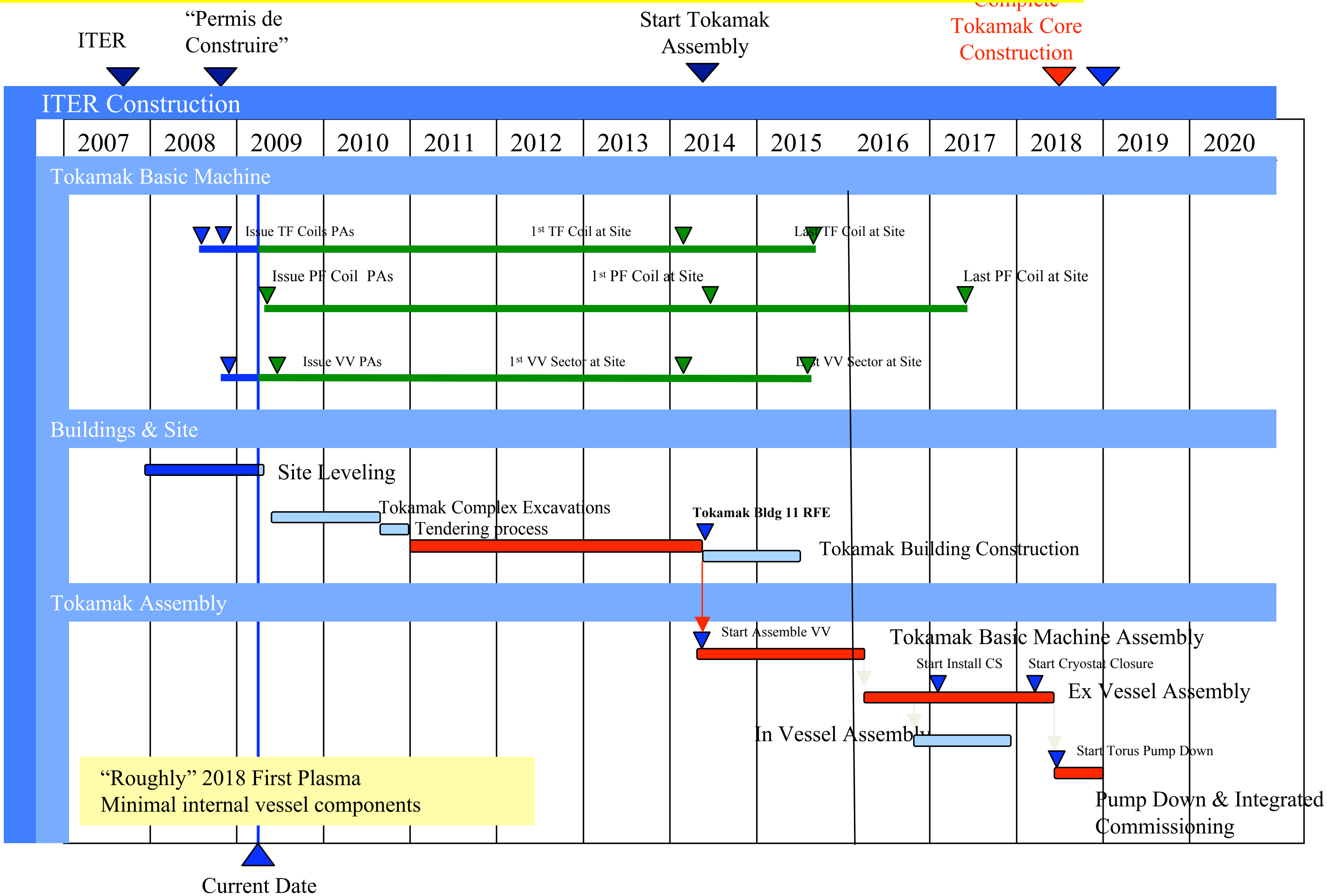
Cadarache Site

ITER Project Site Layout: 3-D graphics view



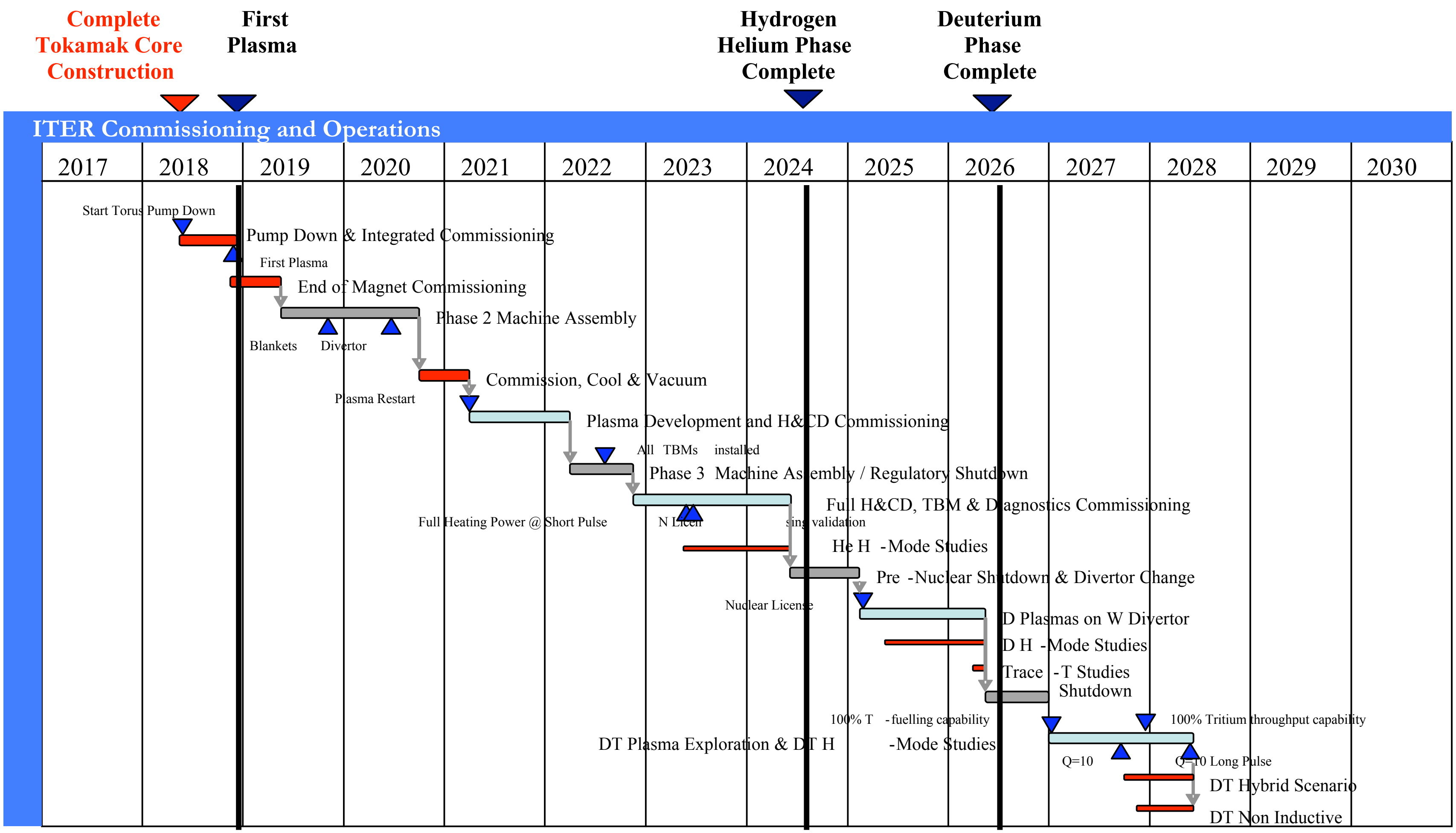

ITER SITE View West – March 09

ITER Construction – Updated Schedule

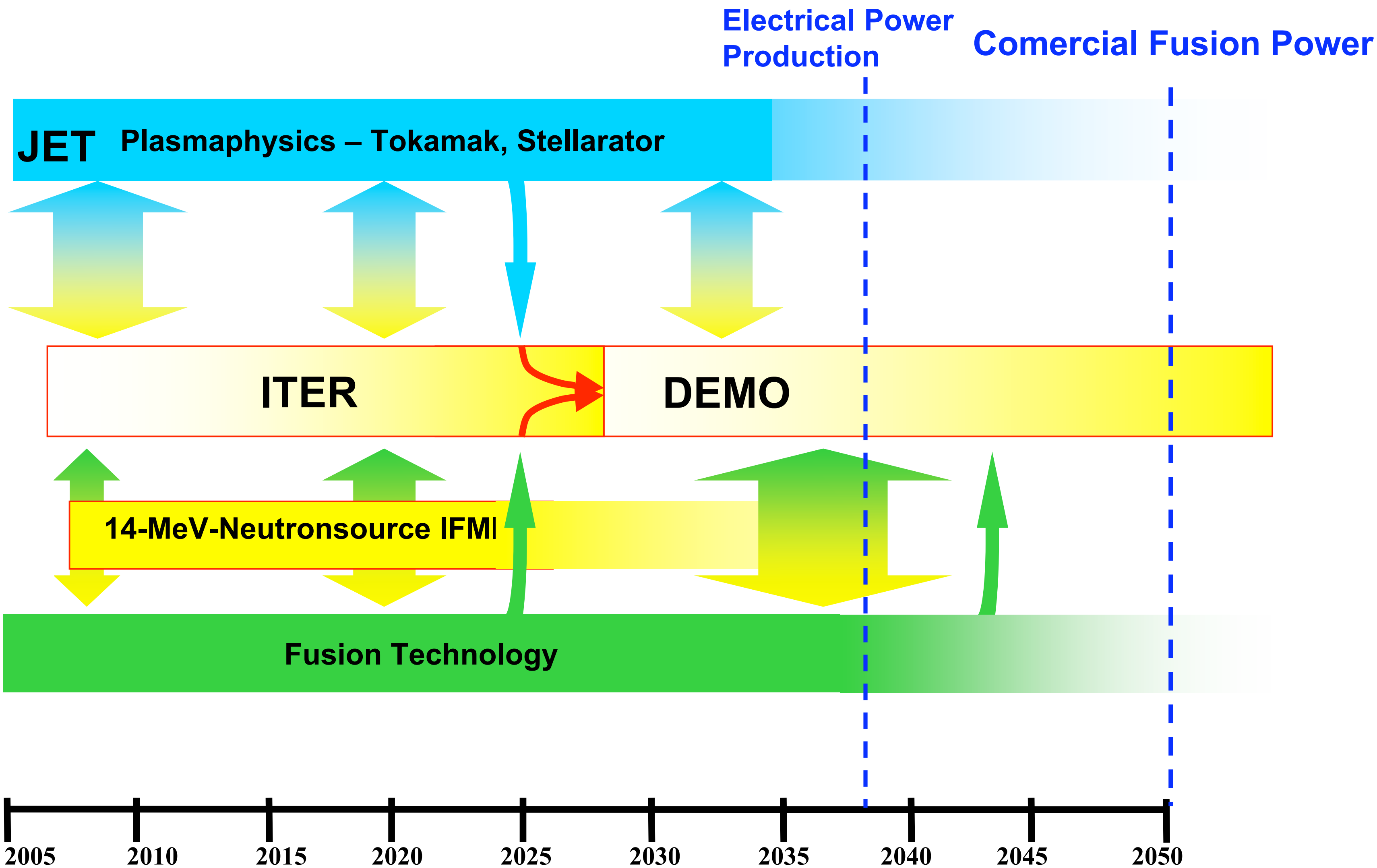


Updated Schedule: Experimental Schedule

- First D-T Plasma foreseen at the end of 2026 or beginning of 2027
- Effective use of Shutdown: Phase 2 assembly, W-divertor exchange



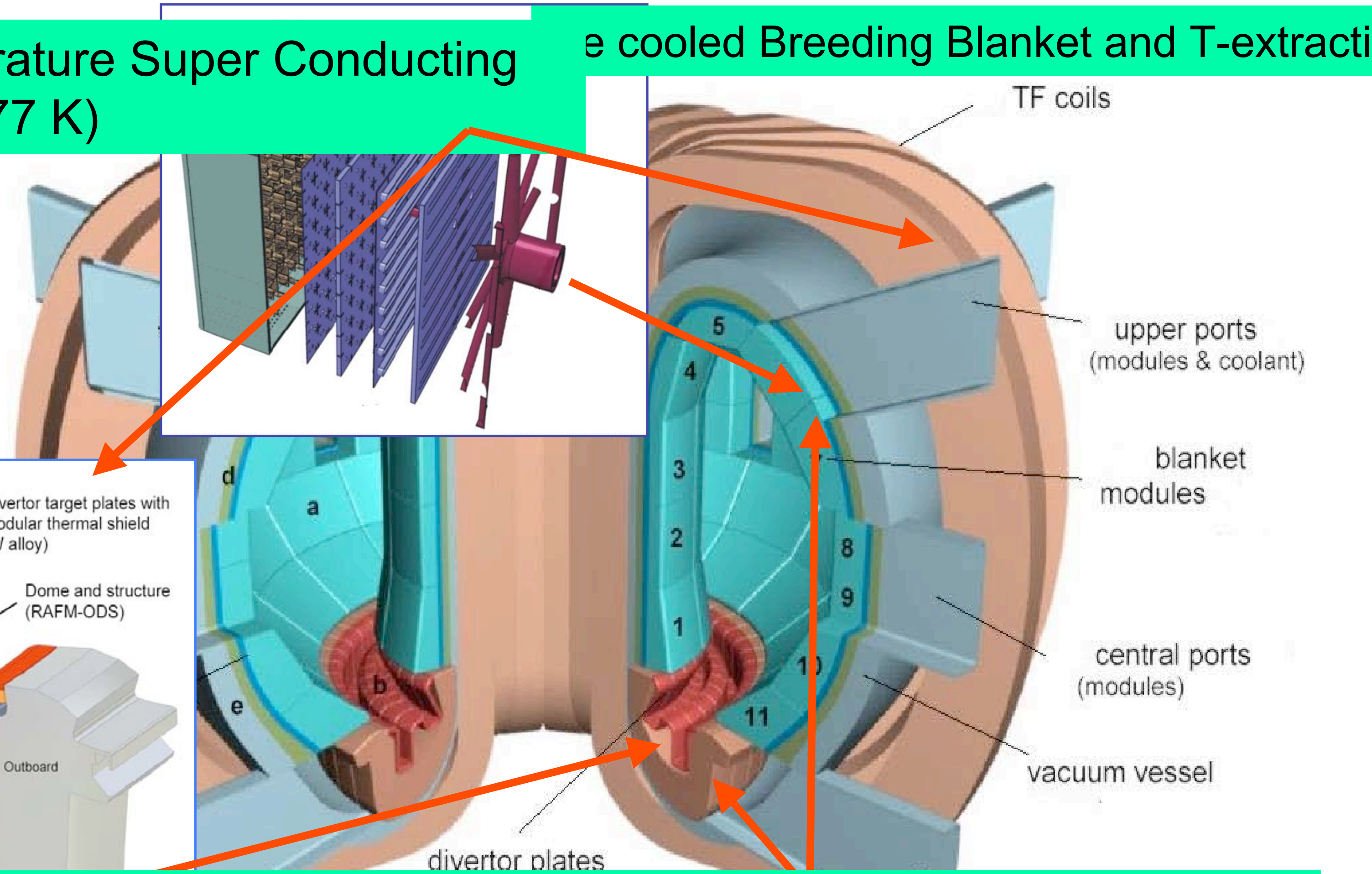
Road Map to the Fusion Reactor (Fast Track - EU)

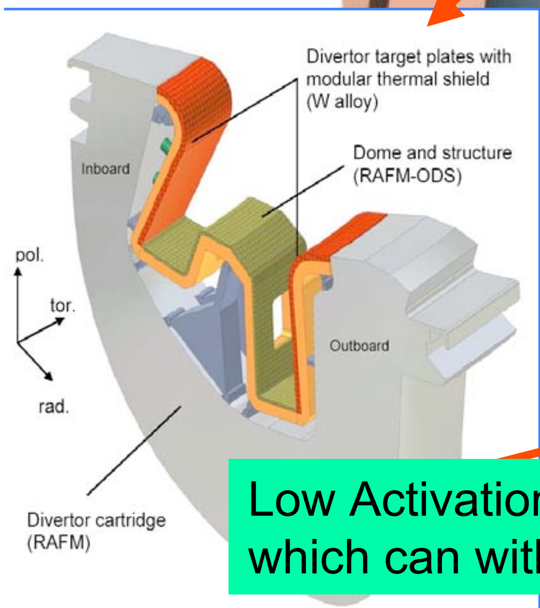


Main Technology Developments for DEMO

High Temperature Super Conducting Coils (40 to 77 K)

e cooled Breeding Blanket and T-extraction





Low Activation Structural Material for the "In Vessel" Components which can withstand the large neutron fluence (150 dpa end of life)

Conclusions

ITER has many assets as a burning plasma experiment and is the key step towards the realization of fusion energy

- The Physics and Technology Development of the last two decades was very impressive and made the realisation of ITER possible (was not the case beginning of the 90^{th})
- Based only on the physics and technology advances we could have started to construct ITER ~5 to 8 years ago
- However, to set up an international project like ITER takes time and also to establish the teams to built it
- ITER construction is now well on its way with many procurements being started even if in some cases remaining design issues have to be tackled

- **Please see for further information on ITER the talks of S. Chiocchio and G. Johnson as well as talks and posters from many other colleagues from ITER and the DAs**