DEMO and the Route to Fusion Power

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Overview

- The Role of DEMO in the ‘Fast Track development of Fusion’
  - Roadmap to Fusion Power
  - Gap Analysis of development needs.
  - Fusion Development Issues.
  - Roles for ITER, IFMIF and DEMO.
- Targets and technical basis for DEMO:
  - Technical feasibility;
  - Economic and environmental acceptability of fusion;
  - Materials requirements;
  - Power exhaust handling;
  - Tritium self-sufficiency;
  - Physics issues;
  - Enabling technologies.
- Optimising a DEMO programme:
  - Strengthening the Programme/ Reducing risk – auxiliary facilities & supporting devices
  - Strategic developments to reduce risk to Fusion deploymnet
  - Accelerating the programme?

The Role of DEMO in the 'Fast Track' development of Fusion'

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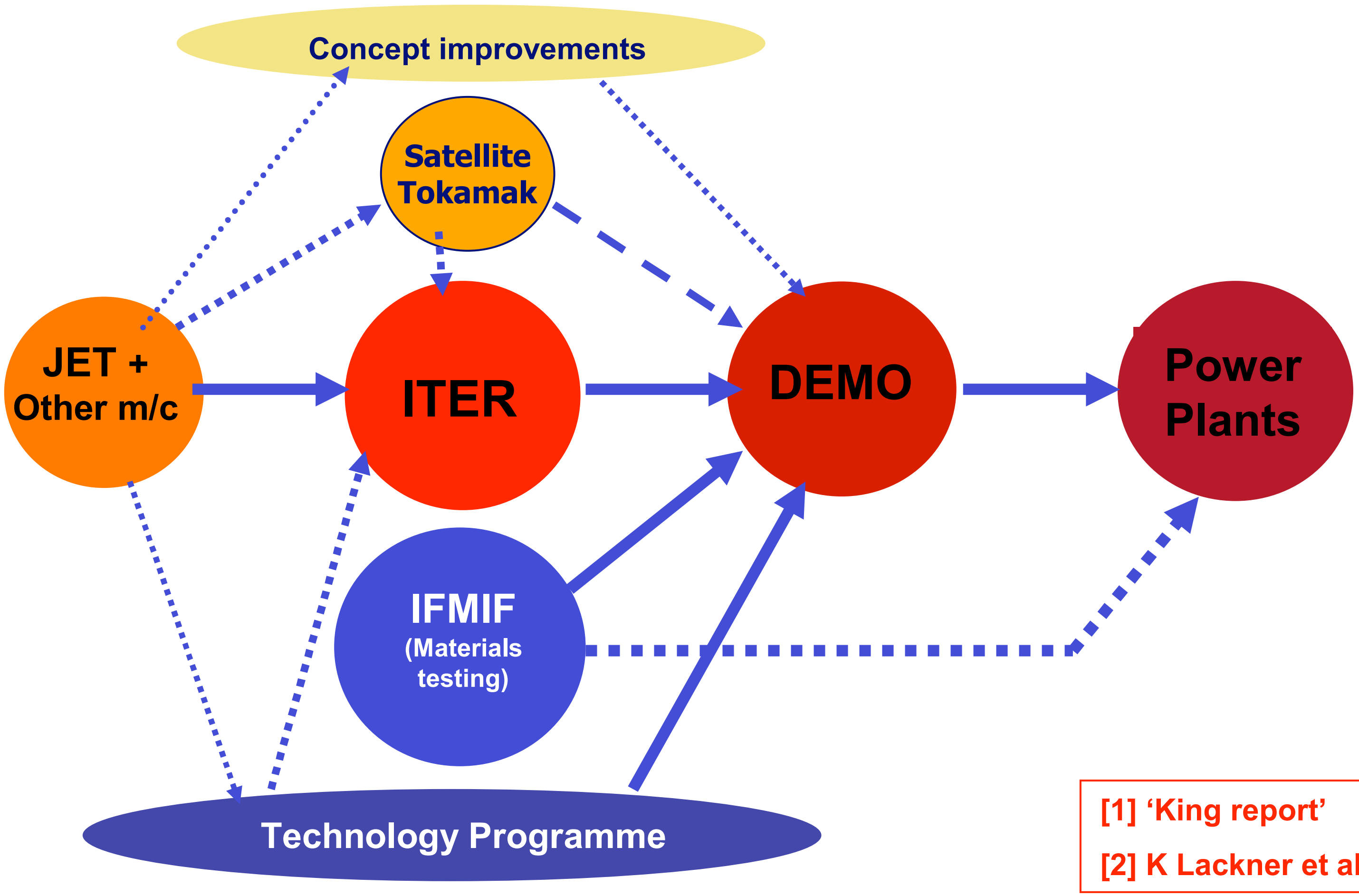
Roadmap to Fusion Power



[1] 'King report'

[2] K Lackner et al

- The 'Fast Track' to Fusion Power development is the Reference Strategy for EU, Japan and other countries. DEMO is the last 'research machine' before a commercial Fusion Power Plant (FPP). [timeline – see Appendix]

Roadmap to Fusion Power:

Device Roles

- **ITER – integrates plasma scenarios at 'reactor scale' to achieve extended 'plasma burn' and test or demonstrate 'reactor scale' technologies.**
- **IFMIF – an accelerator materials test facility with neutron spectrum close to that of a DT tokamak**
- **DEMO – integrates and demonstrates all relevant technology (including tritium breeding) in a 'prototype fusion power plant' + Grid connection to generate electricity.**

Gap Analysis of development needs.

- Defining the exact roadmap to DEMO needs developing in detail to avoid gaps in the physics and technology understanding and capabilities.
- What are the Fusion Development Issues?
- What devices or facilities are part of the Fast Track?
  – Existing;
  – approved for construction; and
  – foreseen in the strategy.
- For existing and approved devices/facilities:
  – how do they/will they contribute? → **development outputs**.
- For foreseen devices/facilities:
  – how will they contribute to development; and/or
  – for the later stages what are the pre-conditions for their success? →**development inputs.**
- **…and what else is needed for development of DEMO or Fusion?**

Gap Analysis of development needs (II).

- Gap Analysis
  - Do the existing and foreseen Roadmap devices answer all the questions required for successful construction/operation of an Fusion Power Plant (FPP) in a timescale consistent with the Fast Track?
  - If not, which support facilities would be needed? At which stage(s)?
  - Are there any fusion devices required to strengthen development or reduce risk - if so, at which stage(s)?

    → **development path change**
- Reality check
  - Could foreseeable ‘modest’ enhancements help to reduce the risk of failure to meet objectives?

    → **individual machine scope/design change**

Gap Analysis of Fusion Development Issues :

role of devices on the 'Fast Track'

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Plasma physics/ Plasma performance	Disruption avoidance	2	3		R	R	R
	Steady-state operation	2	3		r	r	r
	Divertor performance	1	3		R	R	R
	Burning plasma (Q>10)		3		R	R	R
	Start up	1	3		R	R	R
	Power plant plasma performance	1	3		r	R	R
Enabling technologies	Superconducting machine	2	3		R	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Heating, current drive and fuelling	1	2		r	R	R
	Remote handling	1	2		R	R	R
Materials & Component Nuclear performance & lifetime	Materials characterisation			3	R	R	R
	Plasma-facing surface	1	2		3	4	R
	Vessel/First Wall /blanket/divertor materials		1	1	3	4	R
	Vessel/ First Wall /blanket/divertor components		1	1	2	4	R
	T self sufficiency		1		3	R	R
Final System	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

Outputs:		
	1	Will **help to resolve** the issue
	2	**May resolve** the issue
	3	**Should resolve** the issue
	4	**Must resolve** the issue

Inputs:		
	r	**Pre-existing Solution is desirable**
	R	**Pre-existing Solution is a requirement**

UKAEA October 2007 (revised/improved version of original table in **UKAEA FUS 521, 2005**).

Ref [3] – I Cook et al

Note! – 'Approved devices' include JT60SA as a satellite tokamak

Fusion Development Issues :

role of ITER

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Plasma physics/ Plasma performance	Disruption avoidance	2	3		R	R	R
	Steady-state operation	2	3		r	r	r
	Divertor performance	1	3		R	R	R
	Burning plasma (Q>10)		3		R	R	R
	Start up	1	3		R	R	R
	Power plant plasma performance	1	3		r	R	R
Enabling technologies	Superconducting machine	2	3		R	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Heating, current drive and fuelling	1	2		r	R	R
	Remote handling	1	2		R	R	R
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	Plasma-facing surface	1	2		3	4	R
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	Vessel/ First Wall /blanket/divertor components		1	1	2	4	R
	T self sufficiency		1		3	R	R
Final System	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

ITER will play a crucial role in resolving the Plasma physics/performance issues and some of the Enabling Technology Issues

- …but ITER as currently conceived will not
  - totally resolve some Enabling Technology issues (especially those interacting with Plasma Physics)
  - resolve nuclear issues

Targets and technical basis for DEMO

Technical Feasibility Demonstration

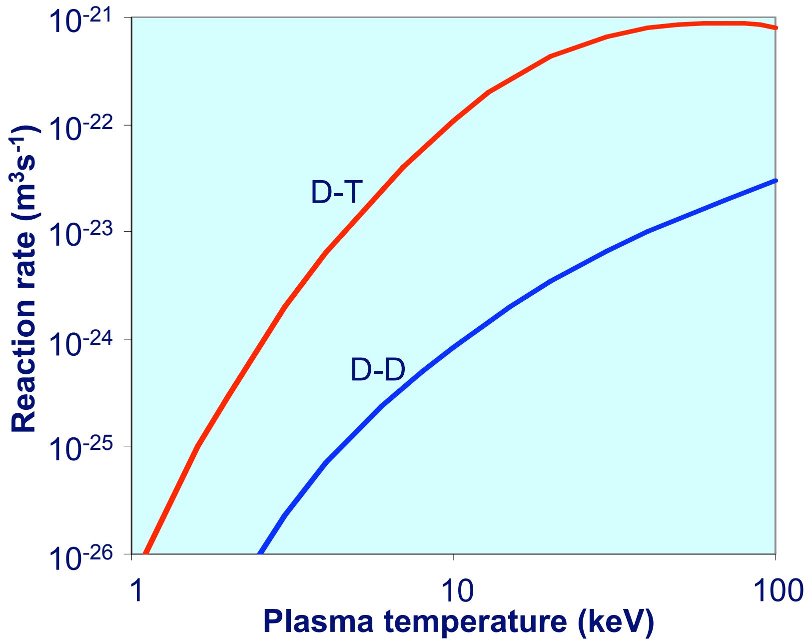
Tokamak basics:

β

- Plasma beta (β) is defined as:

  $\beta = $ (plasma pressure)/(confining magnetic field pressure) ~ nT/B^2

- β's importance comes from its relation to fusion power production:
  - fusion reaction rate ~T^2
  - fusion reaction rate ~ $(n_{Deuterium}).(n_{tritium})$~ density^2
  - so fusion power ~$(nT)^2$ ~ β^2



- β is limited to ~ 1-10% by instabilities.

- Fusion power ~ $\beta^2.B^4$ - clearly high magnetic field is attractive - but is costly! (economically and technically).

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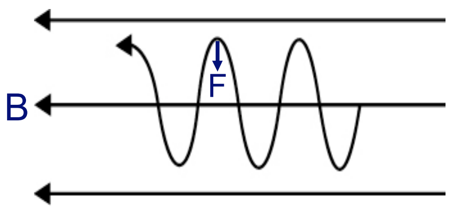
Tokamak particle orbits

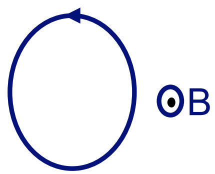
- Charged particles moving in a magnetic field feel a force:

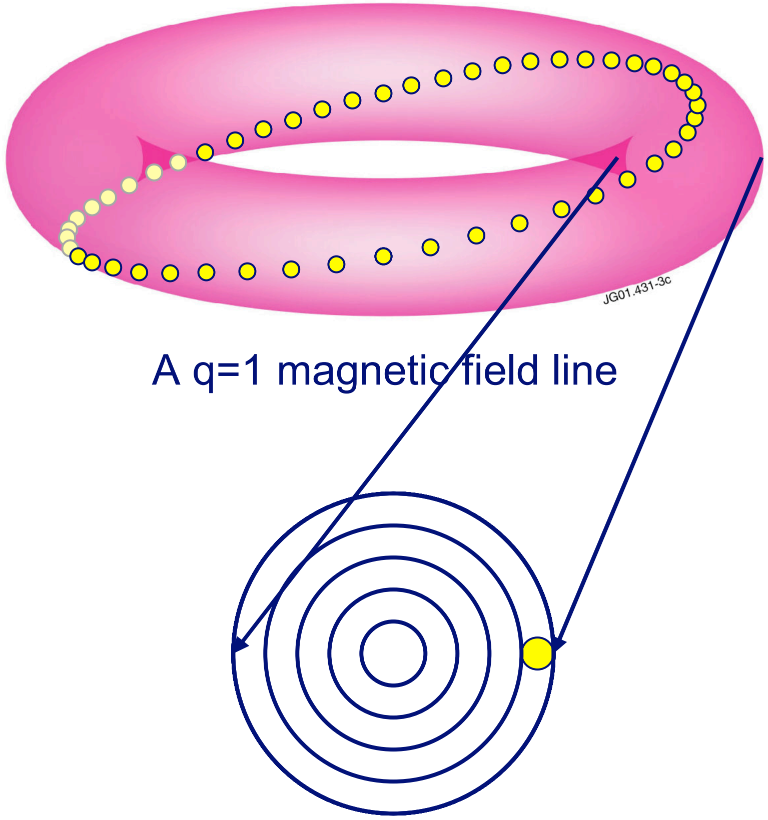
$$\mathbf{F} = q\mathbf{v} \times \mathbf{B}$$

- Ions and electrons move round the tokamak, orbiting around magnetic field lines

$$\text{radius} = \rho_{\text{Larmor}} = \frac{mv_{\perp}}{qB}$$





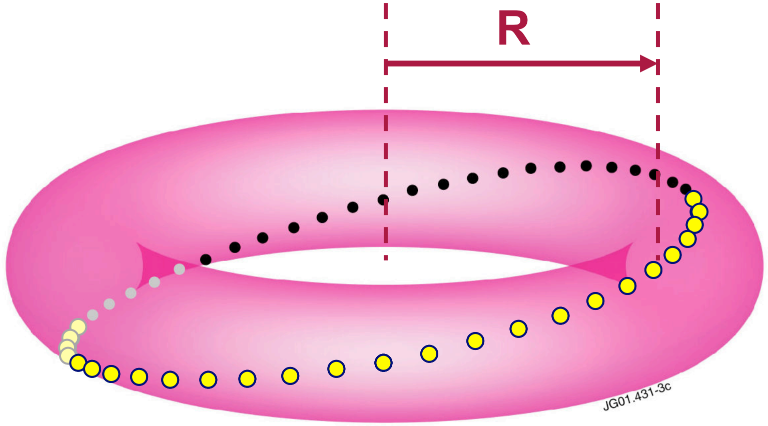


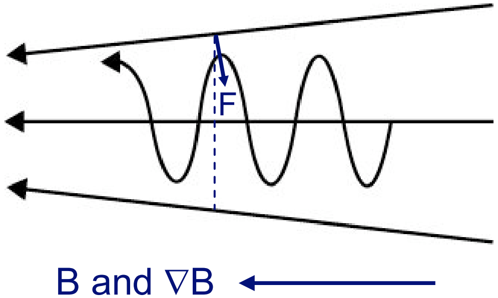
A q=1 magnetic field line

Poloidal cross-section

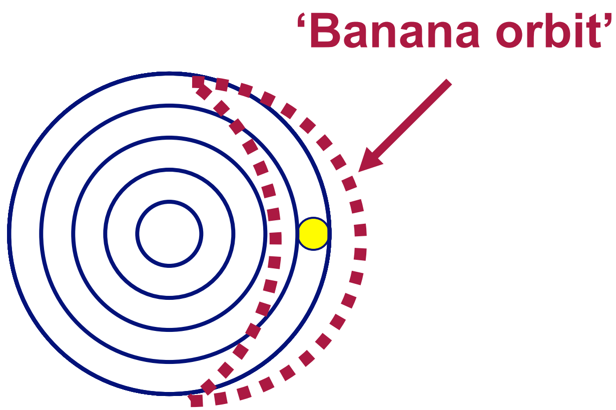
Trapped particles

- B in a tokamak varies as 1/R
- Particles feel a magnetic field gradient parallel to the field as they orbit around the plasma





- v×B force now provides deceleration parallel to the field at the centre of the Larmor orbit
- Particles can be reflected if:
  - particle parallel/perpendicular velocity ratio is small enough



Poloidal cross-section

Tokamak basics:

Vertical drifts

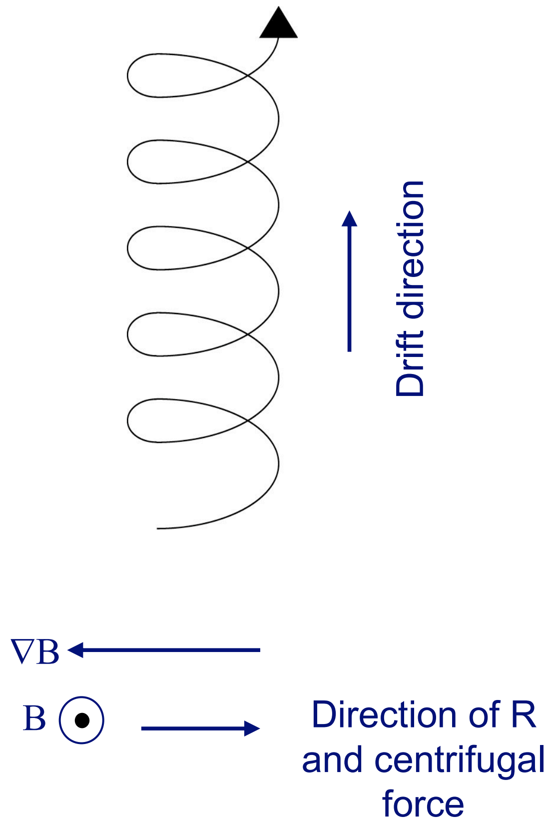
- Ions and electrons drift in the magnetic field gradient perpendicular to the field line
  - Larmor orbit non-circular in a non-uniform field

$$\mathbf{v}_{\text{gradB drift}} = \frac{v_\perp^2}{\omega_c 2} \cdot \frac{\mathbf{B} \times \nabla \mathbf{B}}{B^2}$$

- They also drift in a curved magnetic field due to centrifugal force
  - Larmor orbit non-circular with a non-uniform particle speed
- Giving a total drift velocity:

$$\mathbf{v}_{\text{drift}} = \frac{v_{\parallel}^2 + \frac{1}{2} v_\perp^2}{\omega_c R} \cdot \frac{\mathbf{i} \times \mathbf{B}}{B}$$

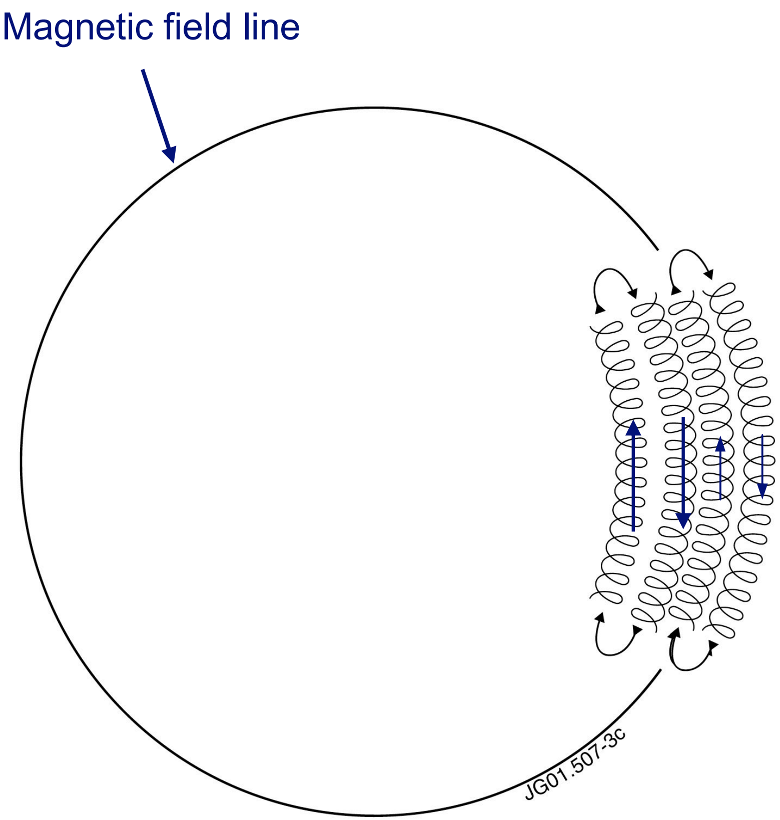
where **i** is the unit vector in the direction of R



Tokamak basics:

Banana orbits and bootstrap current

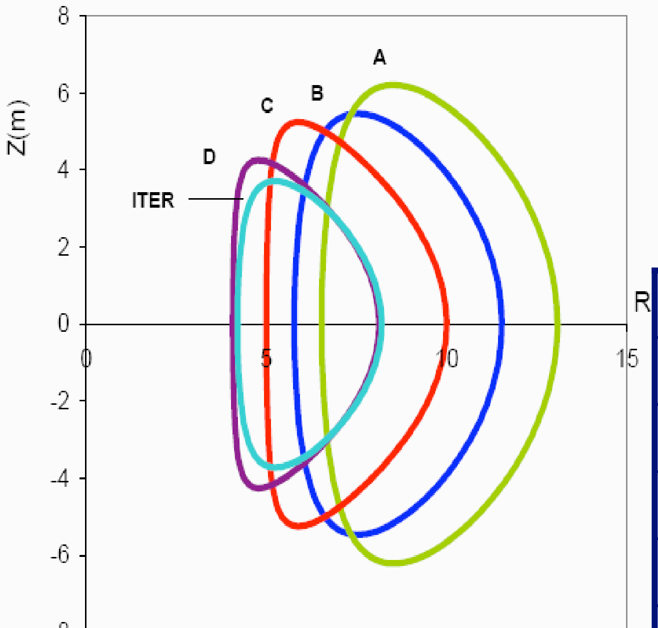
- Magnetic mirror in tokamak creates trapped particles.
- Drift velocities due to B-field gradients cause trapped ions to follow an orbit in the shape of a ‘banana’.
- The helical field stretches these orbits around the torus.
- Radial density gradients in the plasma generate an imbalance in the particle flow where adjacent orbits meet:
  - Net current (bootstrap*)
  - Drive is amplified by passing particles



*Ref [4] R J Bickerton *et al*

DEMO target parameters:

EU Power Plant Concept Studies (PPCS - 2005) gave a range of options



PPCS Models plasma cross-sections (& ITER for comparison)

- Power Plant Conceptual Studies (PPCS) invoke high density operation and enhanced energy confinement to achieve high β and high fusion yield – high β also gives high 'intrinsic' or 'Bootstrap' current drive – reducing external NICD needs

	PPCS A	PPCS B	PPCS C	PPCS D
I_p (MA)	30.5	28.0	20.1	14.1
P_{fus} (GW)	5.0	3.6	3.4	2.5
R (m)	9.55	8.6	7.5	6.1
B_T @ R (T)	7.0	6.9	6.0	5.6
Energy confinement enhancement	20% better than ITER		30% better than ITER	20% better than ITER
Density Limit	20% above ITER		50% above ITER	
β_N (thermal pressure)	2.8	2.7	3.4	3.7
P_{ADD} (MW)	246	270	112	71
Q	20	13.5	30	35
Bootstrap current fraction	0.45	0.43	0.63	0.76

Ref [5] D Maisonnier et al.
Ref [6] D J Ward,

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DEMO core technical parameters:

PPCS exercise uses progressively more aggressive technology

		Model A	Model B	Model AB	Model C	Model D
blanket	Structural material	Eurofer	Eurofer	Eurofer	Eurofer	SiC/SiC
blanket	Coolant	Water	Helium	Helium	LiPb/He	LiPb
blanket	Coolant T in/out (°C)	285 / 325	300 / 500	300 / 500	480 / 700 300 / 480	700 / 1100
blanket	Breeder	LiPb	Li_4SiO_4	LiPb	LiPb	LiPb
blanket	TBR	1.06	1.12	1.13	1.15	1.12
divertor	Structural material	CuCrZr	W alloy	W alloy	W alloy	SiC/SiC
divertor	Armour material	W	W	W	W	W
divertor	Coolant	Water	Helium	Helium	Helium	LiPb
divertor	Coolant T in/out (°C)	140 / 170	540 / 720	540 / 720	540 / 720	600 / 990

- **Model D is too advanced to be the basis of a Fast Track DEMO.**
- **A, AB, B and C* (a variant of C with near-term physics) are all attractive candidate first generation power plants, with excellent safety and environmental attributes and acceptably competitive economics.**

DEMO must integrate, demonstrate and validate all relevant technology (I)

- DEMO construction Materials must be robust against 14 MeV neutron damage.
  - Mechanical, thermal and structural quantities must show minimum change with high radiation dose.
  - Plasma facing materials must additionally resist erosion and sputtering and be compatible with high plasma performance.
  - Divertor materials must further be capable to take high-heat-flux and be compatible with joining to substructures with high flow active coolant capabilities.

DEMO must integrate, demonstrate and validate all relevant technology (II)

- Components must be robust against 14 MeV neutron damage in strong magnetic field environment with thermal cycling and occasional extreme forces.
- Joining and manufacturing techniques used in component fabrication must be validated and proven safe.
- Tritium self-sufficiency must be demonstrated via efficient Breeding Blanket systems and Tritium extraction cycles.
- Peripheral (Heating and Current Drive and Diagnostic) Systems and Balance of Plant systems must be compatible with a high power Nuclear Device.

Targets and technical basis for DEMO

Economic and Environmental Acceptability
of Fusion Power

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DEMO must show Fusion is Economically and Environmentally acceptable

- DEMO is:
  - the 'prototype Fusion Power Plant' and
  - the 'last Research Machine' before the Utilities take over Fusion development.
- It must achieve 'economic and environmental acceptability'.
- 'Acceptability' is a moving target, but whereas Technical Feasibility is about 'existence proof',
  - Economic Acceptability puts the emphasis on 'plasma and operational performance',and
  - Environmental Acceptability puts the emphasis on 'materials optimisation' and 'passive safety'.

DEMO Economics:

factors in Cost of Electricity

- **Fusion Power Plant studies (eg.EU PPCS) reveal a relatively simple scaling can be developed for Cost of Electricity (CoE).**
- **CoE depends on:**
  - **capital cost and hence size of 'nuclear island' (magnets,vacuum vessel, vessel contents)**
  - **Operational parameters:**

$$\mathrm{CoE} \propto \left(\frac{1}{A}\right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4}\, \beta_N^{0.4}\, N_{GW}^{0.3}}$$

Availability

Thermodynamic efficiency

Net electrical power

Physics - high β, high density

Ref [7] D J Ward

DEMO Economics:

Technology and Plasma Physics interaction

- The key factors, **in descending order of relative importance** are:
  - **Plant availability, A**:
    depends on TECHNOLOGY issues
  - **Thermodynamic efficiency**, η_{th} :
    depends on TECHNOLOGY issues
  - **Net electrical output of the plant, P_e** :
    depends on TECHNOLOGY & PLASMA PHYSICS issues
  - **Normalised plasma pressure, β_N** :
    depends on PLASMA PHYSICS issues
  - **Normalised (Greenwald) plasma density, N_{GW}** :
    depends on PLASMA PHYSICS issues
- Not explicitly brought out in this scaling is a dependence on the **divertor heat load limit (P_{div}).**
  - reducing P_{div} can be achieved by increasing machine size → increasing nuclear island capital cost, but also through reduction in β_N and N_{GW};
  - Divertor power handling solution involves integration of TECHNOLOGY and PLASMA PHYSICS issues

DEMO and Environmental Acceptability

- Environmental Acceptability comes not only from:
  - zero Greenhouse Gas and Acid Rain emissions (guaranteed);
  - small materials mining impact (very likely);
  - but also from:
  - the Waste Legacy, which is much smaller than Fission, with no long-lived Actinide products;
  - and from the Passive Safety of Fusion Devices removing the need for off-site evacuation even in the case of a Worst-case Design Basis Accident occurring.
- Materials development aims to reduce the waste burden by developing Reduced Activation and Low Activation materials → only Low-level (or better still hands-on recyclable) waste left 100 years after shut-down.
- Passive-safety depends strongly on
  - in-vessel inventorie… primary containment failure; &
  - desi… enable secondary containment to survive wi… active intervention systems.

ITER experience will be invaluable in licensing

Targets and technical basis for DEMO

Materials Requirements

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DEMO Materials issues

- Structural materials – **subjected to bombardment of 2 MW/m^2 from very energetic (14 MeV) neutrons**
- Plasma facing materials receive an additional average 500 kW/m^2 from hot particles and EM radiation **(up to 20 MW/m^2 on 'divertor')**
- Issues:
  - **Atoms knocked out of place several times a year (>100dpa over reactor life – 1MW.yr.m^{-2} of 14 MeV neutrons~ 10 dpa)**
    - **➔dislocation loops, other damage ➔swelling, hardening & embrittlement**
    - **➔enhanced diffusion ➔creep, rapid diffusion of impurities to grain boundaries, embrittlement etc.**
  - **Some elements transmute by nuclear reactions (this problem is much enhanced for high energy fusion neutrons).**
    - **➔Long-term radioactive products**
    - **➔Helium and hydrogen production in the lattice**
  - **He (the fusion "ash" from the plasma) and D/T get embedded in the lattice → nano-sized bubbles**

DEMO and Power Reactor beyond ITER in neutron damage - but heat flux issues are comparable.

	ITER	DEMO	Reactor
Fusion Power	0.5 GW	2.5 – 5 GW	2.5 - 5 GW
Heat flux (first wall) (divertor)	0.1-0.3 MW/m² ~ 10 MW/m²	0.5 MW/m² ~15-20 MW/m²	0.5 MW/m² ~20 MW/m²
Neutron Load (FirstWall)	0.78 MW/m²	< 2 MW/m²	~ 2 MW/m²
Integrated Neutron Load (First Wall)	0.07MW.year/m² (3 years operation)	5 - 8 MW.year/m²	10 - 15 MW.year/m²
Displacement per atom (dpa)	< 3 dpa	50 - 80 dpa	100 - 150 dpa
	Increasing Materials challenge →		
Transmutation product rates at first wall	~10 appm Helium / dpa ~45 appm H / dpa		

Fusion Development Issues :

role of IFMIF

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Plasma physics/ Plasma performance	Disruption avoidance	2	3		R	R	R
	Steady-state operation	2	3		r	r	r
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	T self sufficiency		1		3	R	R
Final System	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

- IFMIF will be the key device to characterise materials with Fusion neutron spectrum.
- Exact role of IFMIF in licensing is unclear until we know better the regulatory regime at the time/geographical location of DEMO – IFMIF will certainly help.
- IFMIF, because of its limited sample size (0.5 litre) can only give limited help to address the issues of component robustness.

DEMO Structural steels

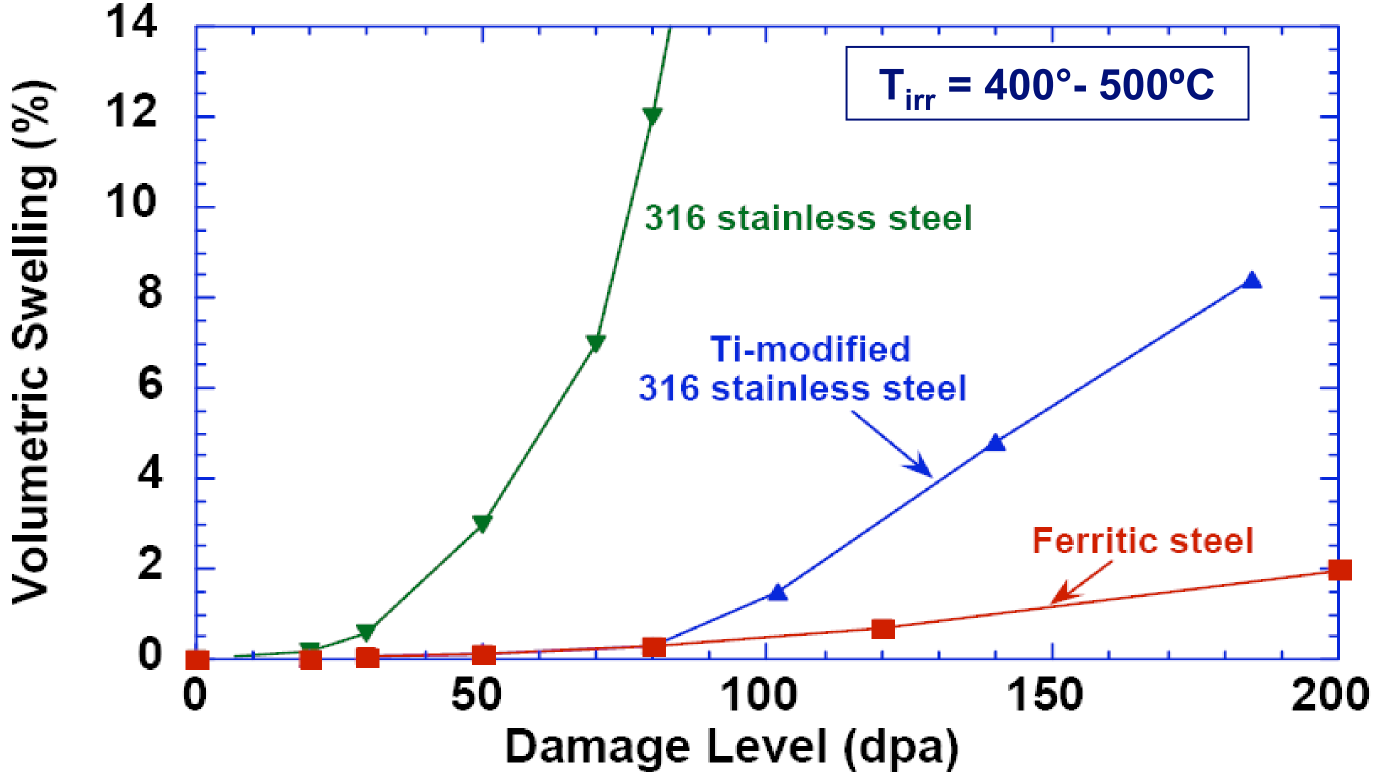
General framework:

- Materials have to have high damage resistance to **limit shutdowns for replacement of eg. blankets**.
- Operational cycles place stringent limts:
  - Materials must operate at **high-temperatures** and through many room-temperature shutdowns;
  - Temperature gradients exist in materials – poloidal variation in plasma load, gradient through to coolant;
  - → temperature fluctuations up to ~600°C range (in case of **high temperature gas cooling**) → mechanical properties must be maintained.
- Innovative engineering/materials solutions are needed.
  - Alloys, forming self-stabilizing phases and microstructures under irradiation in the **operating temperature and irradiation dose ranges,**
  - Tailored pre-fabricated microstructures (nanostructures) with **sufficient** long-term stability under fusion irradiation.

Economic Acceptability considerations

DEMO Structural materials:

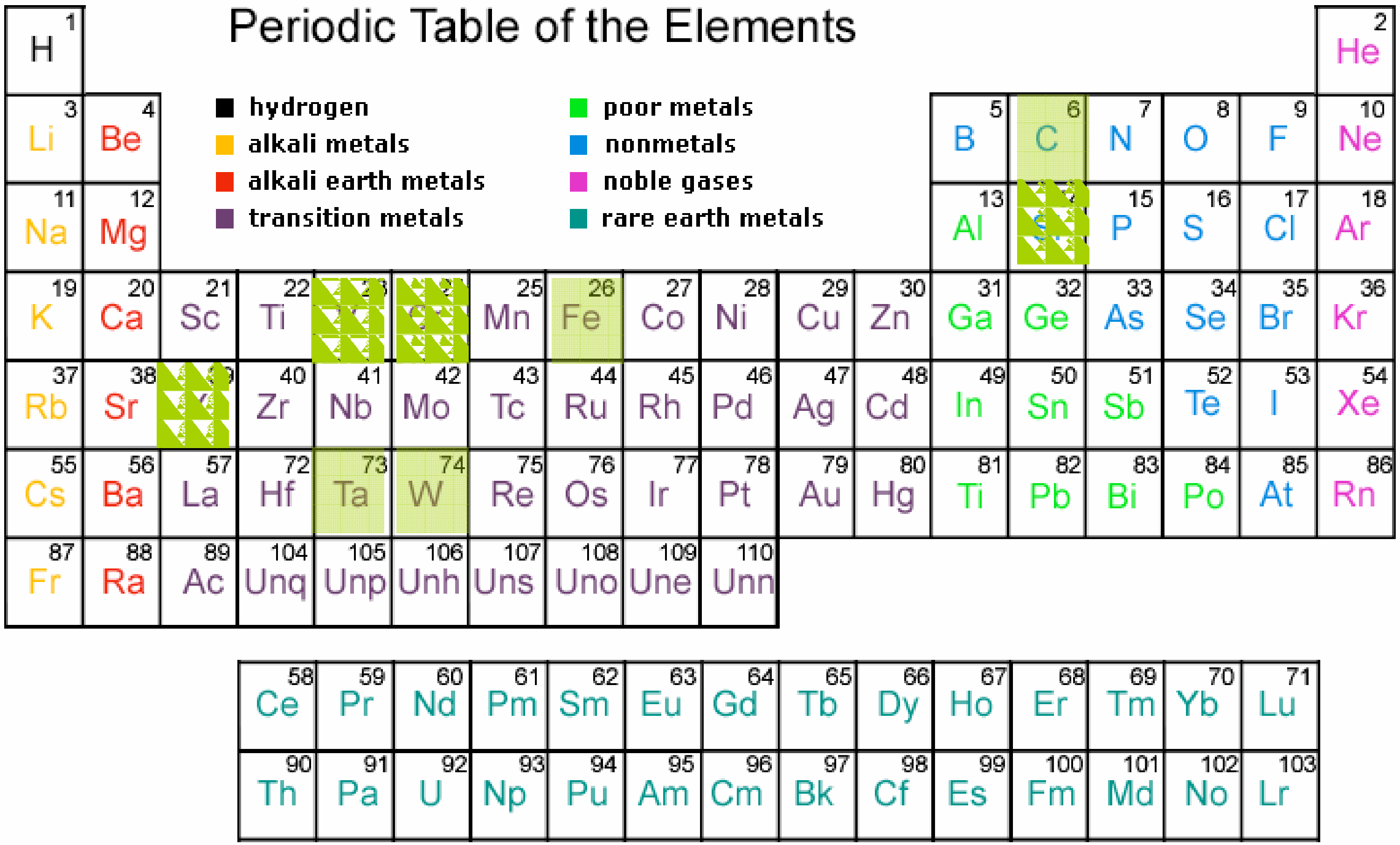
swelling of steels



- Lowest swelling occurs in body-centred-cubic (BCC) alloys (Ferritic steels, Vanadium alloys)
- BCC materials are however subject to radiation embrittlement.

DEMO Materials Environmental basis(I)

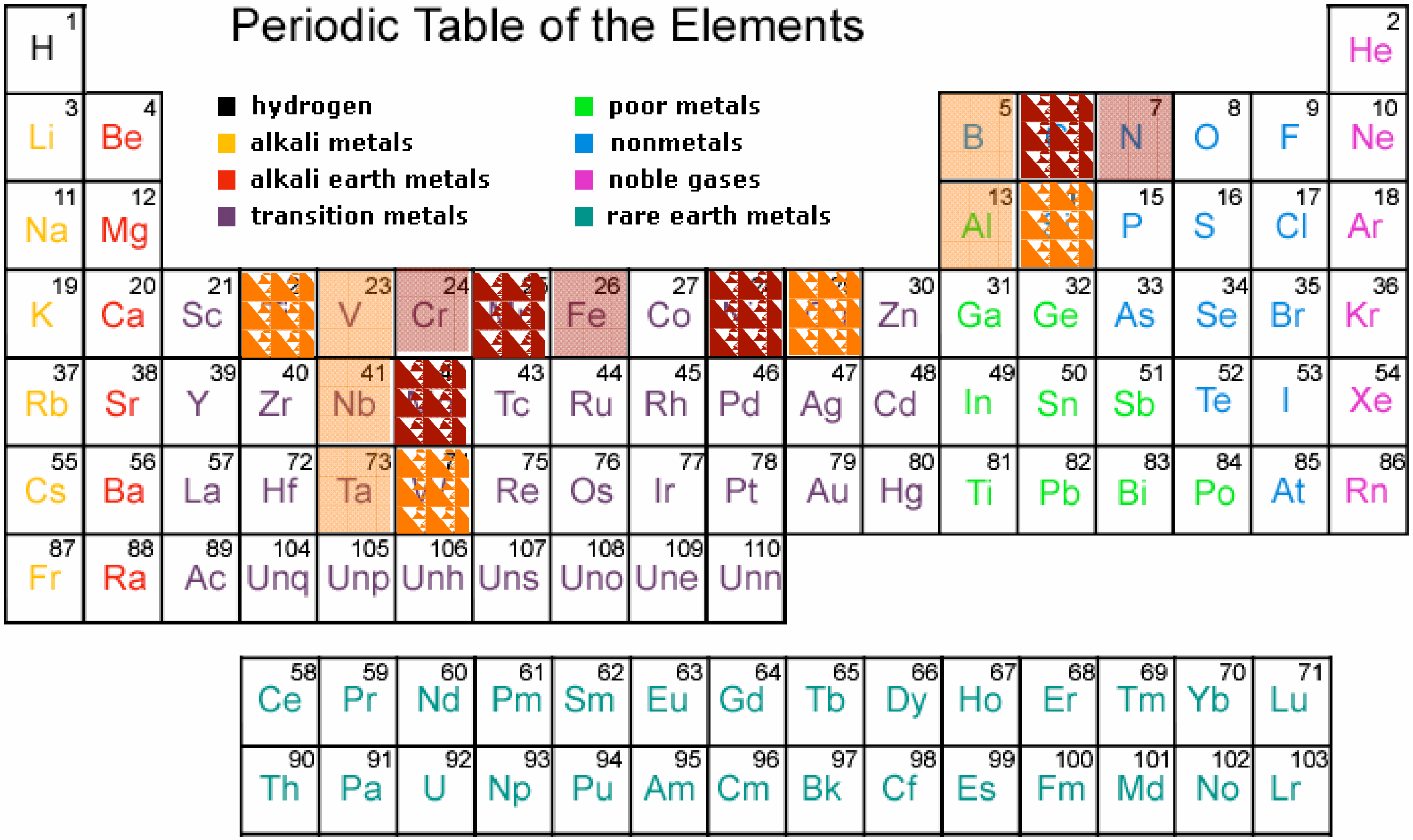
(courtesy Dr Robin Forrest –UKAEA, IAEA)



Only the elements in green can be used: anything else useful is transmuted by high-energy neutrons into VERY radioactive products

DEMO Materials Environmental basis(II)

(courtesy Dr Robin Forrest – UKAEA, IAEA)

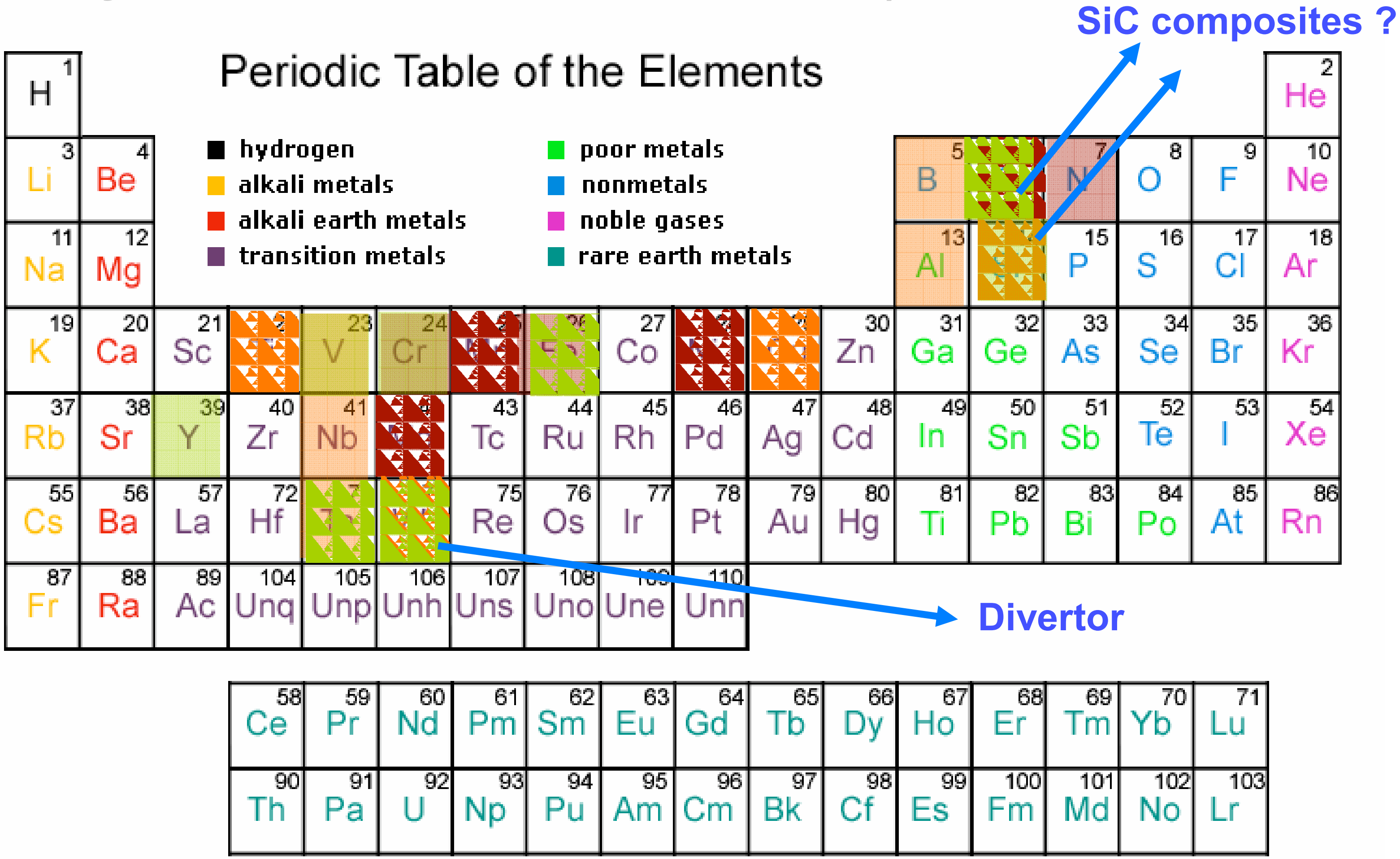


"vital" for steels

"useful" for steels

DEMO Materials Environmental basis(III)

(courtesy Dr Robin Forrest – UKAEA, IAEA)



“vital” for steels

“useful” for steels

Acceptable for a fusion reactor

“Reduced activation” steels

For the experimental ‘Reduced Activation Ferritic Martensitic’ steels: -
→Ta replaces Nb,
→ V replaces Ti
→Cr replaces Mn … up to a point… nothing much replaces Mo.

F82H (Japan): Fe - 7.7%Cr – 2%W - 0.2%V - 0.04%Ta - 0.09%C

Eurofer (EU): Fe - 8.9%Cr – 1%W - 0.2%V - 0.14%Ta - 0.12%C

There are also “Oxide Dispersion Strengthened” (ODS) variants -

Nanoscale Y_2O_3 particles:

- act as He, H sinks and improve defect rate,
- strengthen,
- reduce creep.

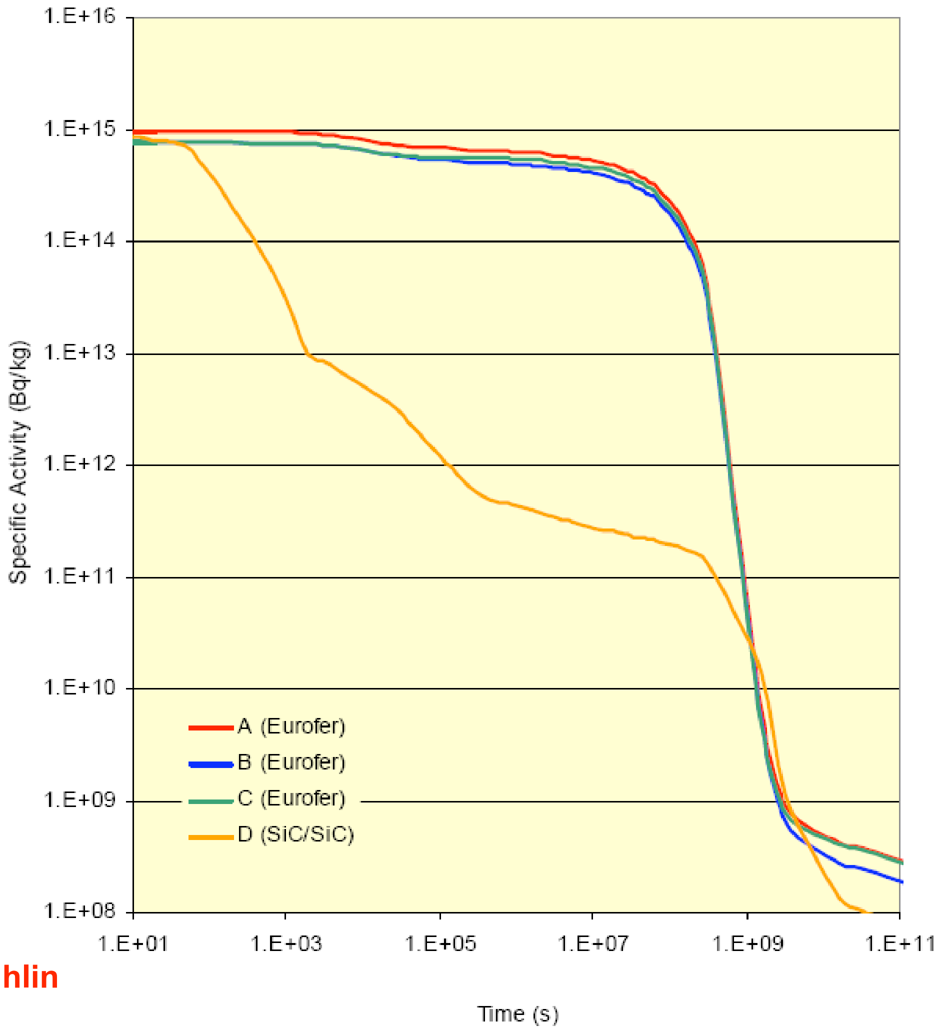
Currently only small experimental batches made

These will be “cool” enough to be recycled and re-used after about 50-100 years storage after 5 years service in the first wall.

DEMO Materials: – Environmental Waste burden

Specific activation of outboard midplane first wall materials after
25 years full power operation – blanket replacement every 5 years
for the EU PPCS model reactors.

Advanced fusion materials should decay in ~ 100 years.



Ref [8]: R J Pampin-Garcia and M J Loughlin

DEMO Materials environmental basis:

Manufacturability

Real materials have trace impurities
Eurofer Chemical composition (wt%):

1. **Pure - ideal**
2. **Real - present day**
3. **Achievable**

Reference Eurofer

EUROFER (data in wt%)

Element	Case 1 (specification, without impurities)	Case 2 (real material)	Case 3 (achievable material)
Al		0.008	0.0001
As		0.02	0.001
B		0.001	0.0001
C	**0.11**	**0.11**	**0.11**
Ca		0.0002	0.0001
Ce		0.003	0.0001
Co		0.005	0.001
Cr	**9.0**	**9.0**	**9.0**
Cu		0.0037	0.001
Fe	**bal**	**bal**	**bal**
Hf		0.0001	0.0001
K		0.0002	0.0001
Mn	**0.40**	**0.40**	**0.40**
Mo		0.0012	0.0001
N	**0.03**	**0.03**	**0.001**
Nb		0.001	0.00001
Nd		0.0002	0.0001
Ni		0.005	0.001
O		0.01	0.001
P		0.005	0.001
Re		0.0001	0.0001
Ru		0.001	0.001
S		0.003	0.001
Sb		0.01	0.001
Si	**0.05**	**0.05**	**0.05**
Sn		0.003	0.001
Ta	**0.07**	**0.07**	**0.07**
Ti	**0.01**	**0.01**	**0.01**
V	**0.20**	**0.20**	**0.20**
W	**1.1**	**1.1**	**1.1**
Zr		0.0001	0.0001

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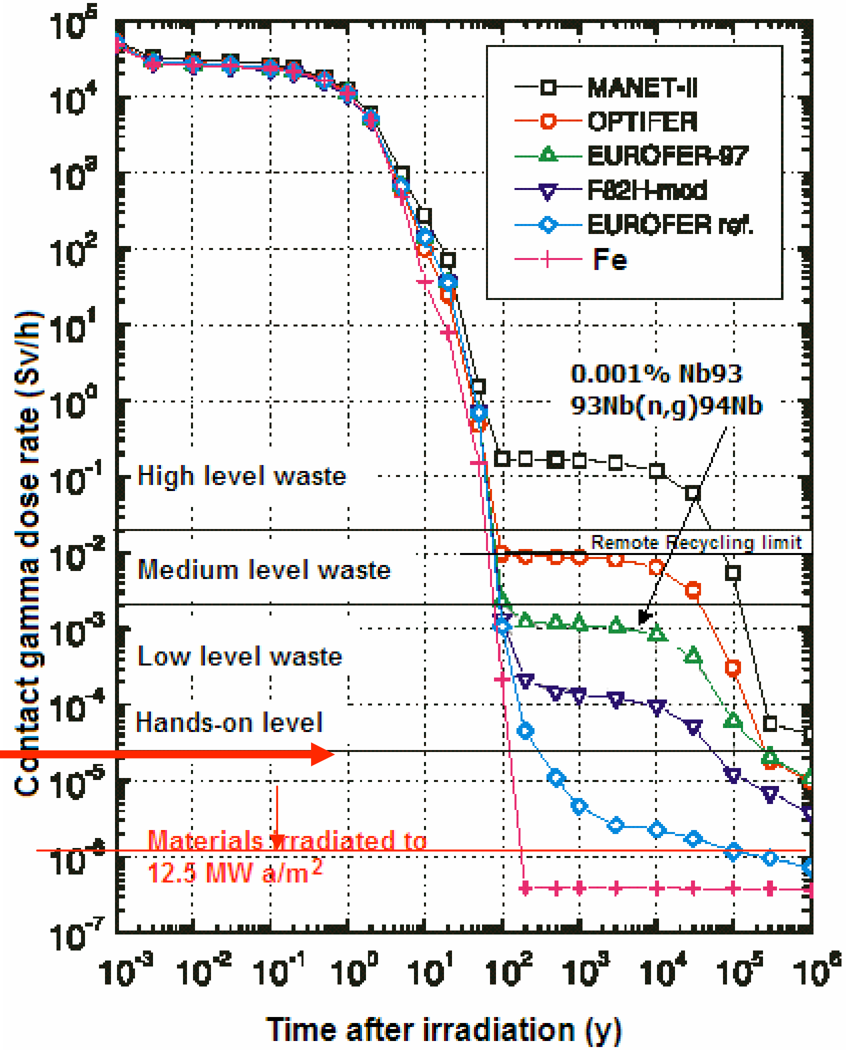
DEMO Materials environmental basis:

Manufacturability – effect on waste

EUROFER Blanket Material
- **replace every 5 years;**
- **P_{fus} = 3 GW;**
- **Neutron Wall Load = 2.3 MW.m^{-2} for 5 years**

For EUROFER to achieve Reference composition Nb impurity needs to be further decreased by two orders of magnitudes to 0.00001% (~0.1 ppm)

Hands-on recycling level



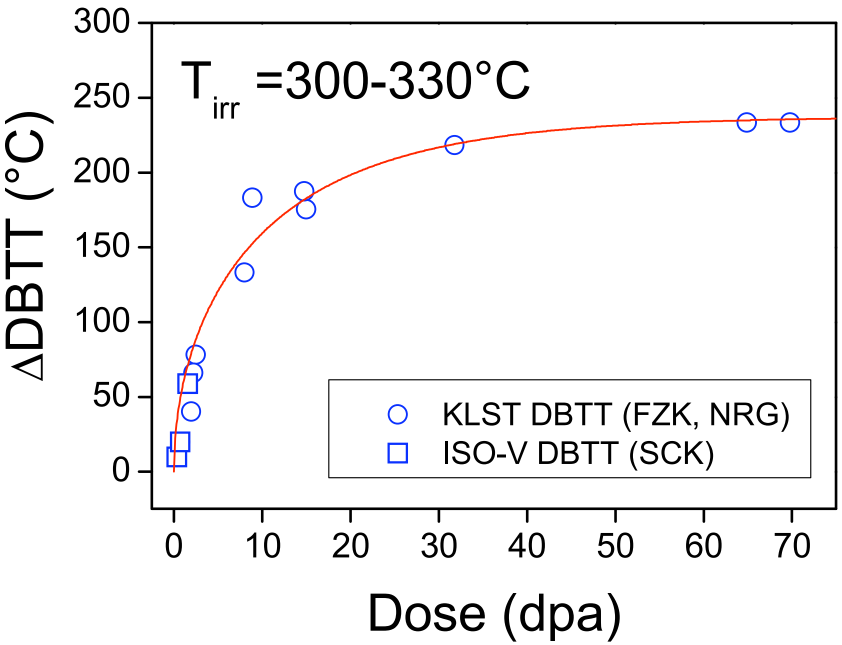
Ref [9]: P Batistoni et al.

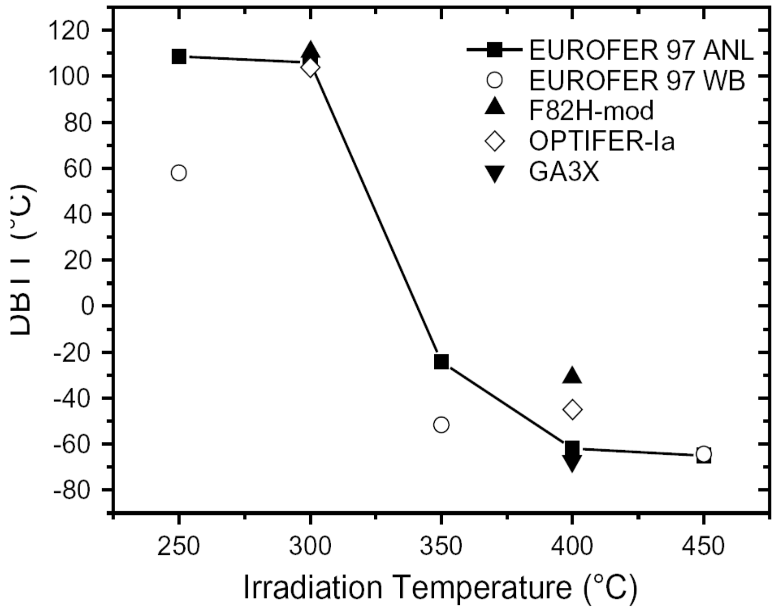
DEMO Structural steels:

Ferritic-martensitic steels embrittlement

- **Ferritic-martensitic steels developed for Fusion eg. EUROFER97 (EU)or F82H (Japan) have good long-term stability. However:**
  - **these steels become brittle if irradiated at room temperature or temperatures up to ~ 300ºC**

Ref [10]: E Gaganidze





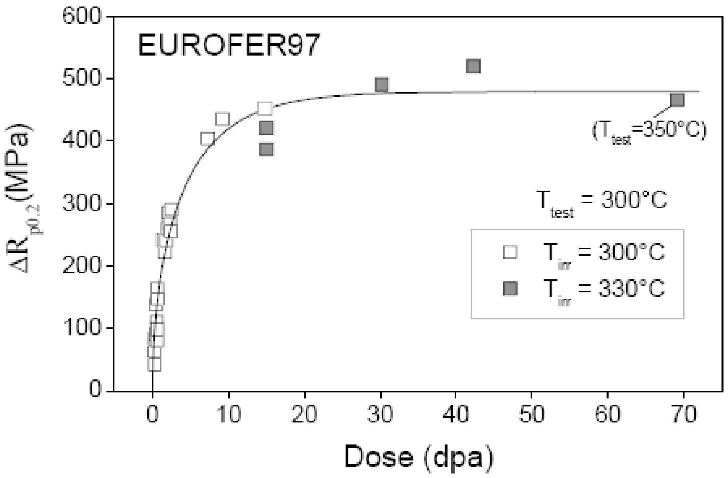
•***Embrittlement can be mitigated*** (annealed) using high operating temperatures above 350ºC → hard to achieve everywhere in a Tokamak reactor 'core'.

•Note ***transmutation helium embrittlement effect*** not included in experimental data, from low energy pile irradiation → expected to alter the behaviour.

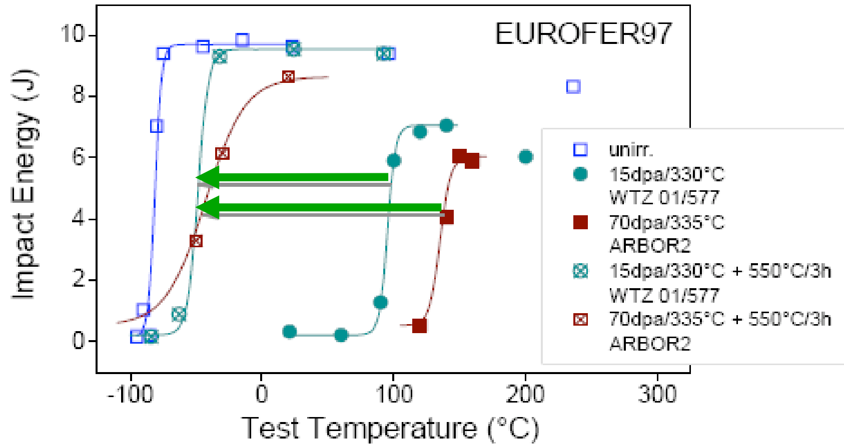
DEMO Structural steels:

Ferritic-martensitic steels embrittlement (II)

FzK / SCK-CEN



Ref [11]: E Gaganidze



- Radiation hardening occurs at DEMO relevant temperature (up to 300-350°C)
- DBTT ~ 150° - 200ºC →unusable for reactor

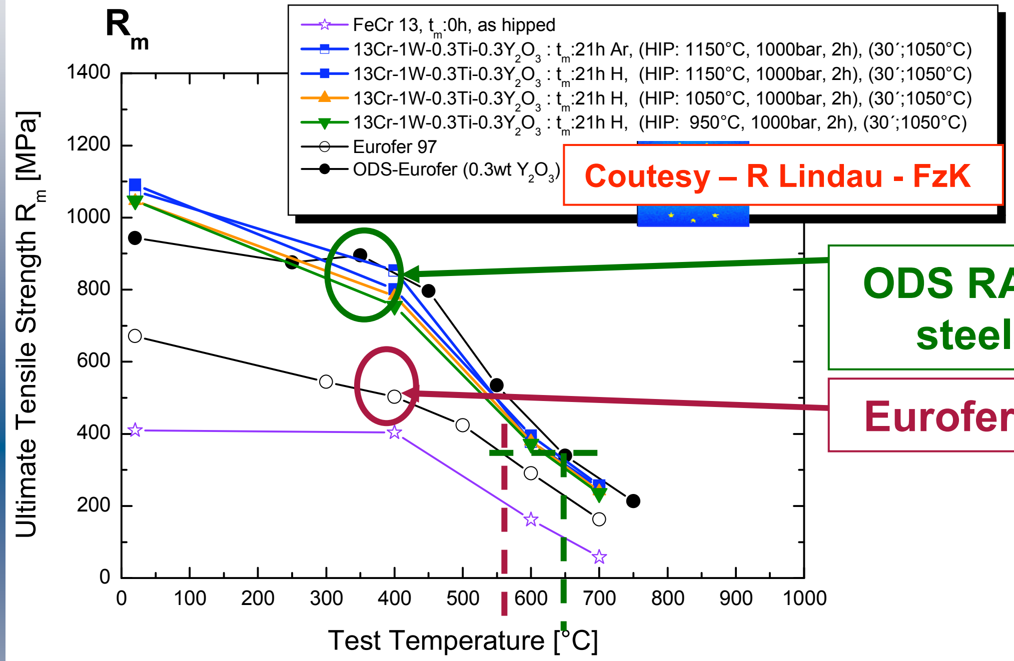
Good news --Annealing at 550°C before cooldown restores acceptable DBTT – can this be arranged operationally??

Operating window (FM steels) ~350ºC - 550ºC (upper limit set by strength)

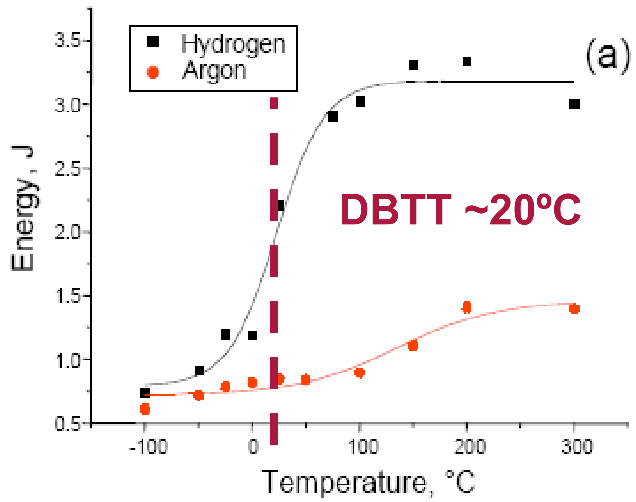
Bad News --He – damage, studied at low fission dpa by 'enhancement' (B-doped steels (EU) and He+ beam (JA) shows added hardening at concs > 400 appm (EU) and >1000 appm(JA) Ref [12] S Jitsukawa ; Ref [13] E Materna-Morris

DEMO Structural steels:

high temperature strength – ODS steels



Ref [14]: Z Oksiuta and N Baluc



- Conventional ferritic and ferritic-martensitic steels (EUROFER97 or F82H):
  - **lose mechanical strength at high operating temperatures (550ºC upper-limit).**
  - **suffer from thermal creep (accelerated by irradiation).**
- ODS steels have higher strength at high T and better resistance to thermal creep

but (even un-irradiated) are brittle at room temperature. Development needed

DEMO Structural steels:
modelling and experimental strategy

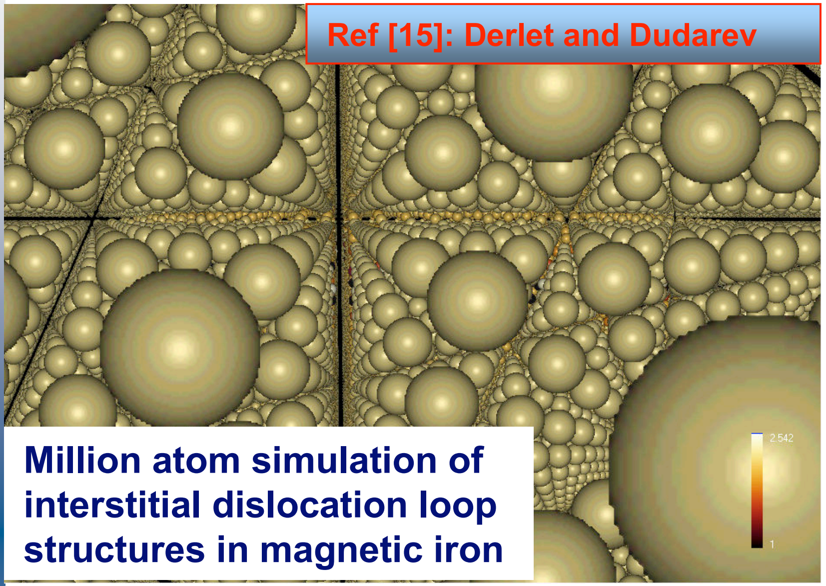
Modelling focusses on EUROFER-type RAFM steels and related model alloys under fusion relevant conditions

Objectives:

- **Extrapolate the results of existing experimental tests to the more complex and diverse range of conditions expected in a fusion power plant**
- **Develop close links with experimental work through the investigation of observation-focused models**
- **Guide and help optimise, the experimental programme on materials testing and development**
- **Acquire expertise and formulate principles required for developing innovative fusion materials with superior properties/performance**

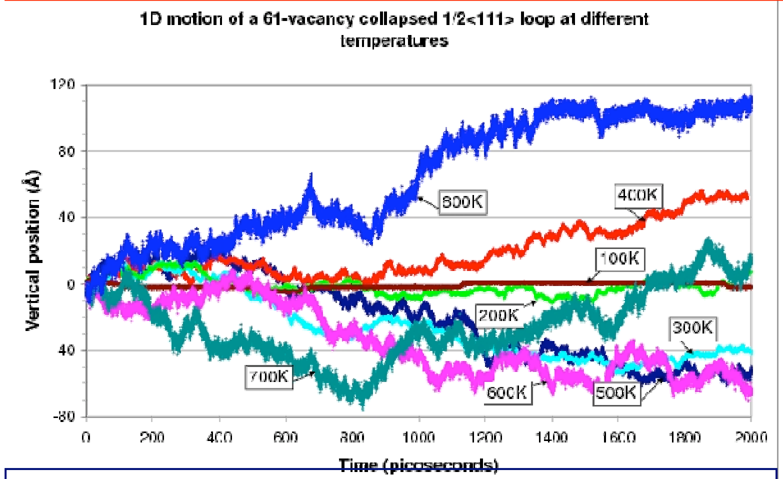
Modelling of dislocations

Ref [15]: Derlet and Dudarev



Million atom simulation of interstitial dislocation loop structures in magnetic iron

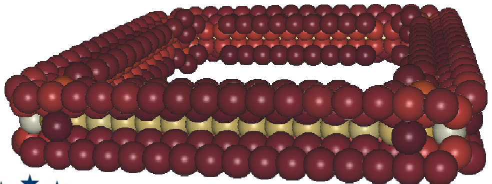
Courtesy M R Gilbert (Culham/ Univ of Oxford)

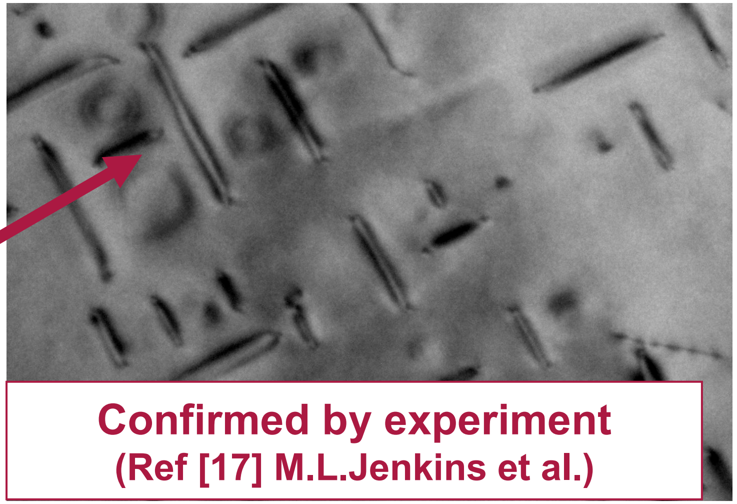


Thermally activated diffusion of defects

- Interaction between dislocations as a function of temperature affects strength of materials.
- Simulations predict ' <100> dislocation loops' should adopt an unusual square shape with their sides parallel to particular directions.

Ref [16]: Dudarev et al.





Confirmed by experiment
(Ref [17] M.L.Jenkins et al.)

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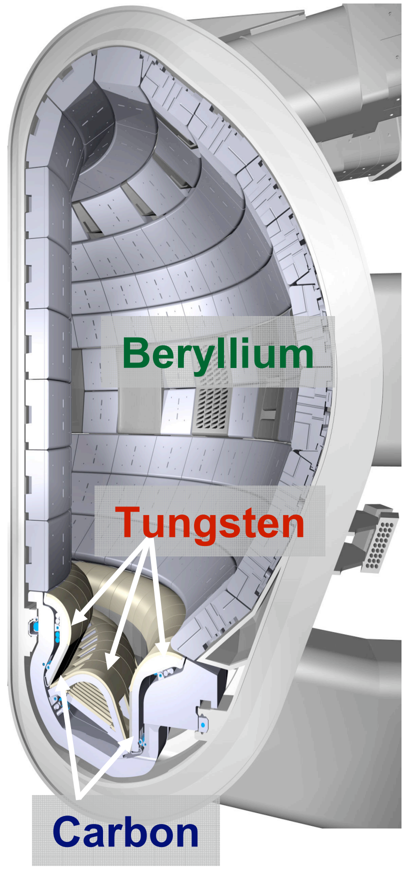
DEMO Plasma facing materials

- PFM challenges
  - Withstand high power fluxes, steady-state & transients
  - Minimise erosion – resist sputtering and chemical erosion
    - Keep the plasma pure
    - Component lifetime
  - High stability under neutron irradiation
  - Minimise retention of tritium (reactor inventory)

Carbon unusable in reactor

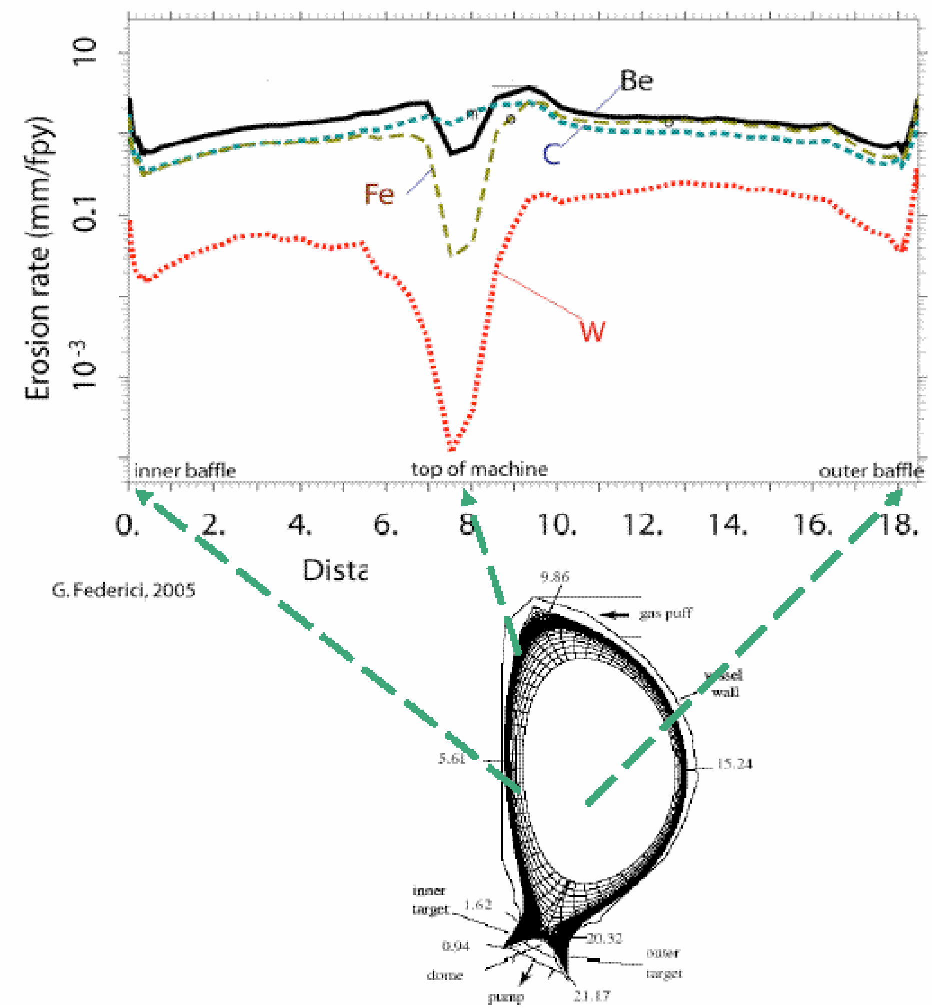
Beryllium unusable in reactor



ITER

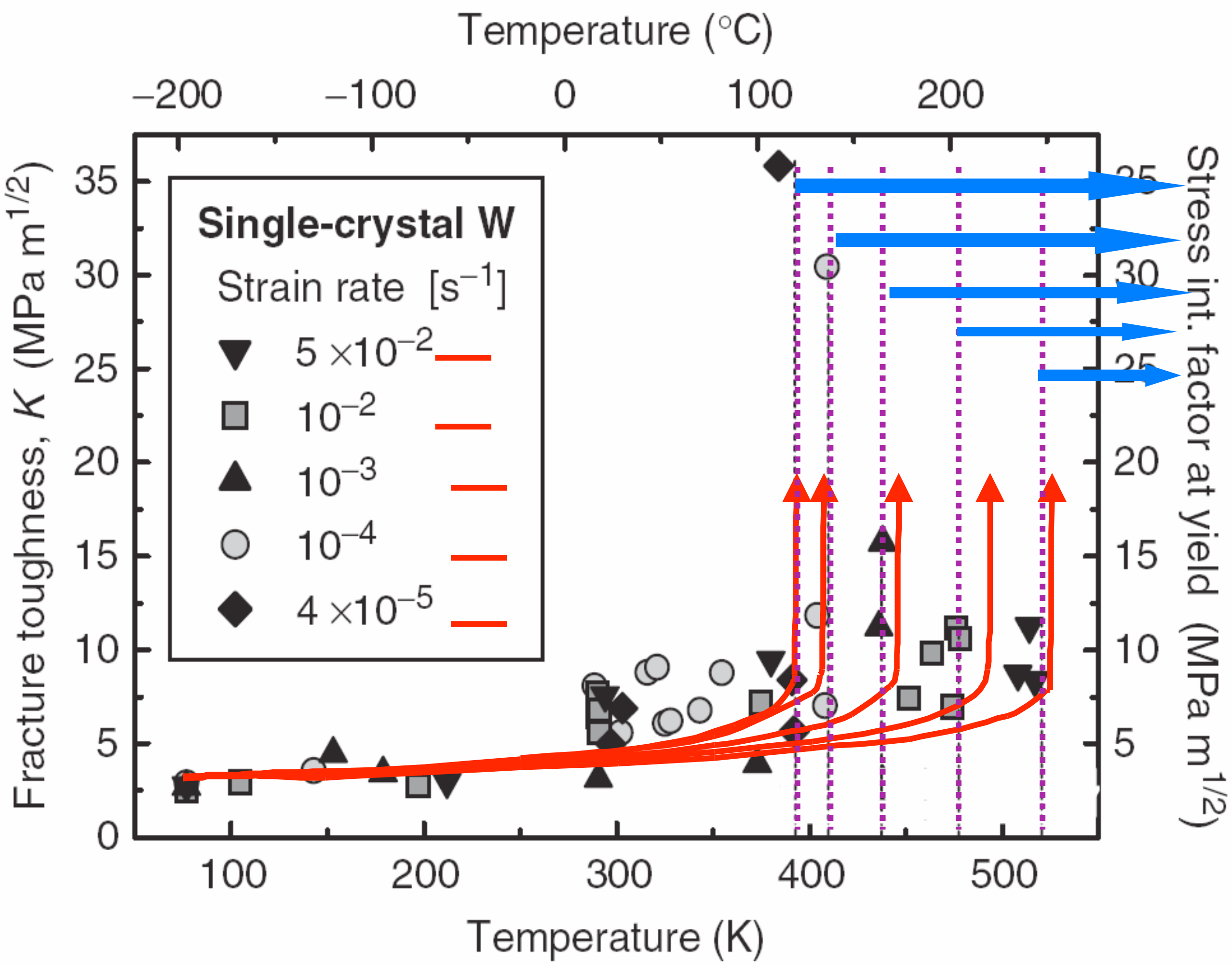
DEMO Divertor erosion
– advantage of Tungsten

- Present day divertors largely use CFC – but erosion rates in DEMO conditions (where heat loads could be > 20 MW/m^2) would be unacceptable.
- Current programmes feature investigation of Tungsten as PFC for ITER Divertor to prove for DEMO.
- **Challenges for Tungsten lies in fabricability of complex structures.**



DEMO Divertor – Tungsten is brittle!

Modelling Brittle-Ductile Transition in Tungsten



UKAEA-Univ of Oxford (SG Roberts et al)

Experimental Brittle-ductile transitions at different strain rates

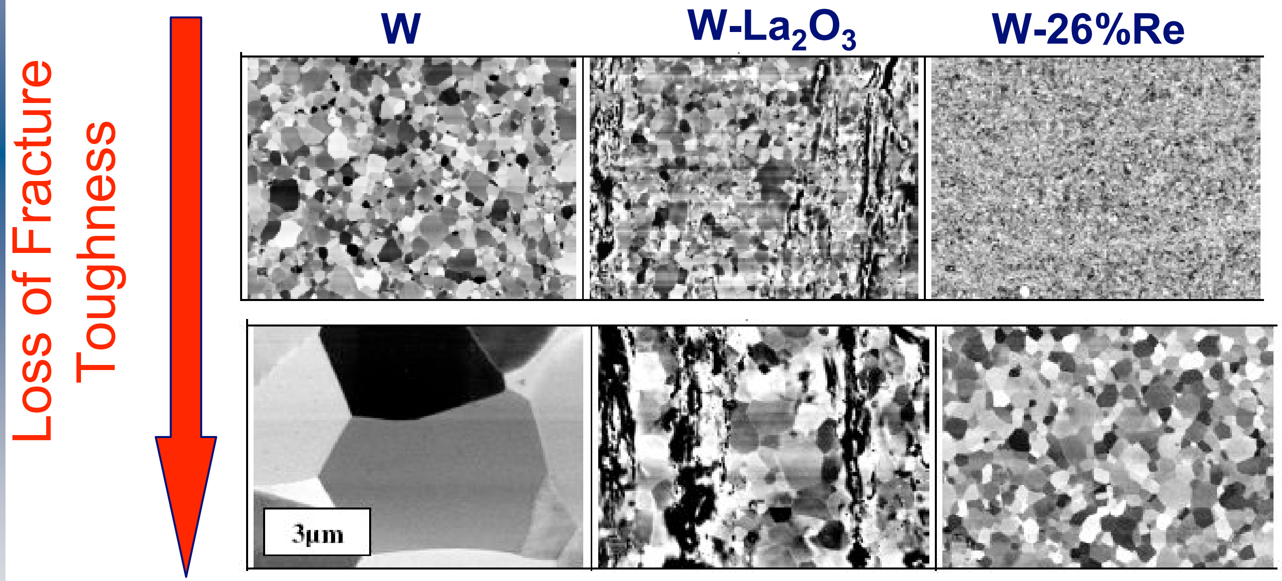
Modelled Brittle-ductile transitions

Severe lower operating temperature limit for high thermal stress/shock divertor regions.

DEMO Divertor:

tungsten upper temperature limit set by re-crystallisation

- Tungsten lower temperature limit in power loading conditions set by DBTT at ~700ºC.
- Below this (< 500ºC) embrittlement by radiation.
- Upper limit is set by re-crystallisation of W and W alloys ~ 1200ºC.



Annealing 1 hour

835 ^{0}C
Initial Microstructure unaffected

1200 ^{0}C
Unacceptable Re-crystallisation

New alloys under development : W-Ti, W-V for structural applications   W-Y2O3, W-Si-Cr, W-TiC for armour.   *R. Pippan et al.*

Targets and technical basis for DEMO

Power exhaust handling.

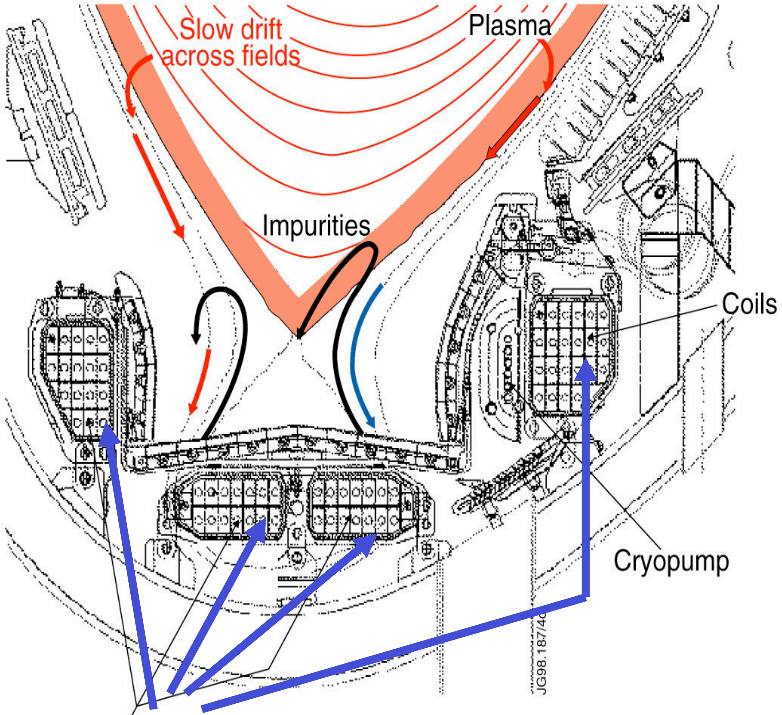
UKAEA Fusion
Working with Europe

Tokamak basics:

Divertors

- Fusion plasmas can become polluted by impurities from the vessel wall as it is heated up and bombarded.
- Helium 'ash' is also produced by the fusion reactions (slowed down α-particles).
- 'Divertors' are the solution
- In a 'divertor' the main plasma is separated from target tiles by a 'private' plasma.
- Flows in the 'private' region resist impurity influx.

JET Divertor region

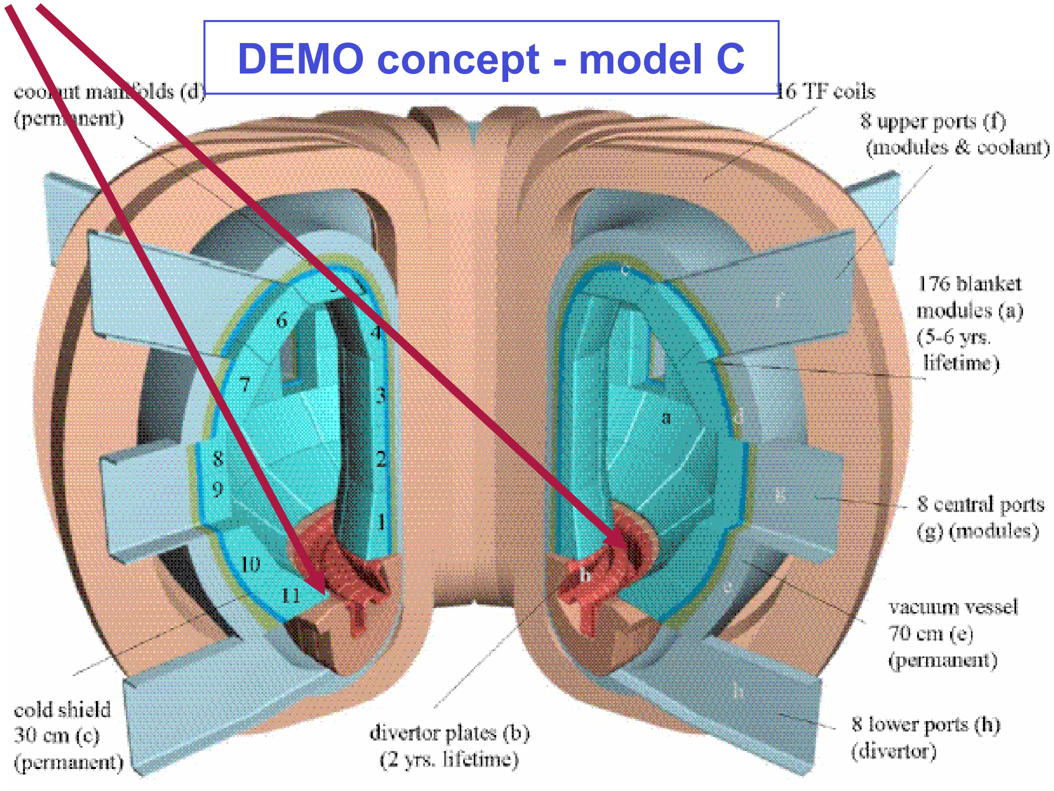


Coils for magnetic 'separatrix' - open field lines conduct plasma to target

DEMO power exhaust handling:

the Divertor and high-heat-flux components

- **Mission of a Tokamak Divertor:**
  - take the directed exhaust of heat and particles (including pumping Helium 'ash' from thermalised αs);
  - provide a barrier to keep sputtered impurities out of the plasma;
  - provide ~20% of the heat to the 'steam circuits'.



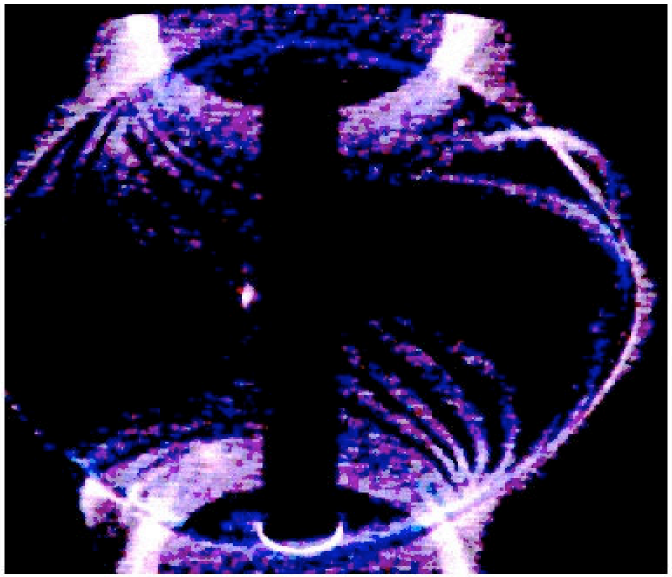
- the price of Divertor benefits (H-mode, density control, impurity control) is a very high power loading in the Divertor region > 20 $MW.m^{-2}$ possible in a DEMO/reactor – already near this in JET but pulsed mode.

Tokamak basics:

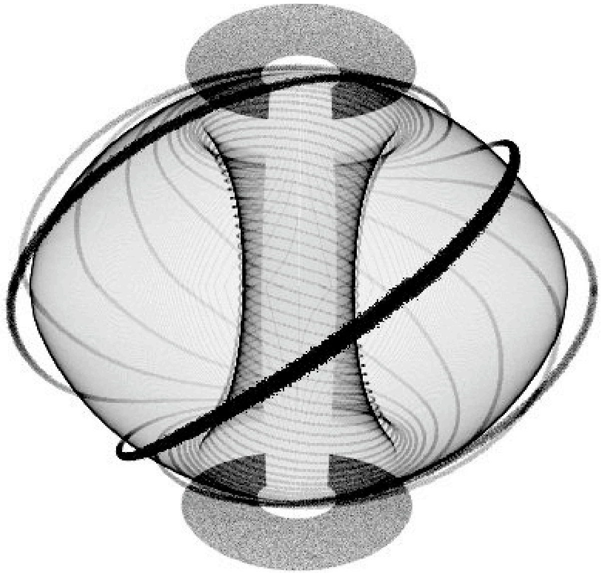
Edge Localised Modes (ELMs)

- **In high confinement modes Plasma turbulence is suppressed to form a Transport Barrier at the edge**
- Plasma pressure builds within the barrier, and periodically breaks down at high pressure ('ELM') **- turbulence degrades energy confinement – but also causes damage to the divertor - handling, and overcoming turbulence is a huge challenge for physics and engineering.**

Experiment MAST – UKAEA Culham



Theory - Imperial/ Culham



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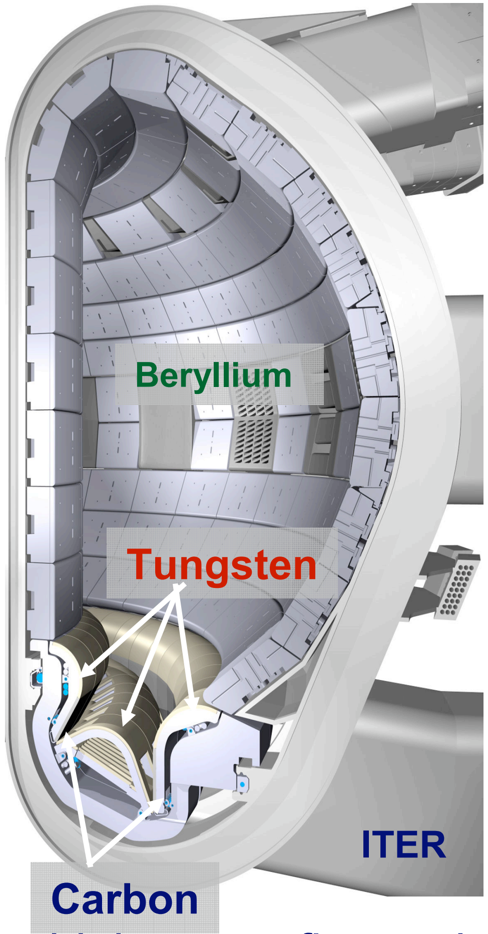
Divertor realisation links many engineering and materials challenges

- Engineering:
  - High heat flux technology – **fatigue, component lifetime**, CFD etc.;
  - Response to transients, EM loads;
  - **Maintainability –** Remote handling design.
  - Manufacturability in ITER or reactor – relevant materials (tungsten).
- Materials challenges, as previously stated
  - high particle fluxes, minimise erosion – resist sputtering and chemical erosion
  - high stability under neutron irradiation
  - minimise tritium retention (reactor inventory) and activation.

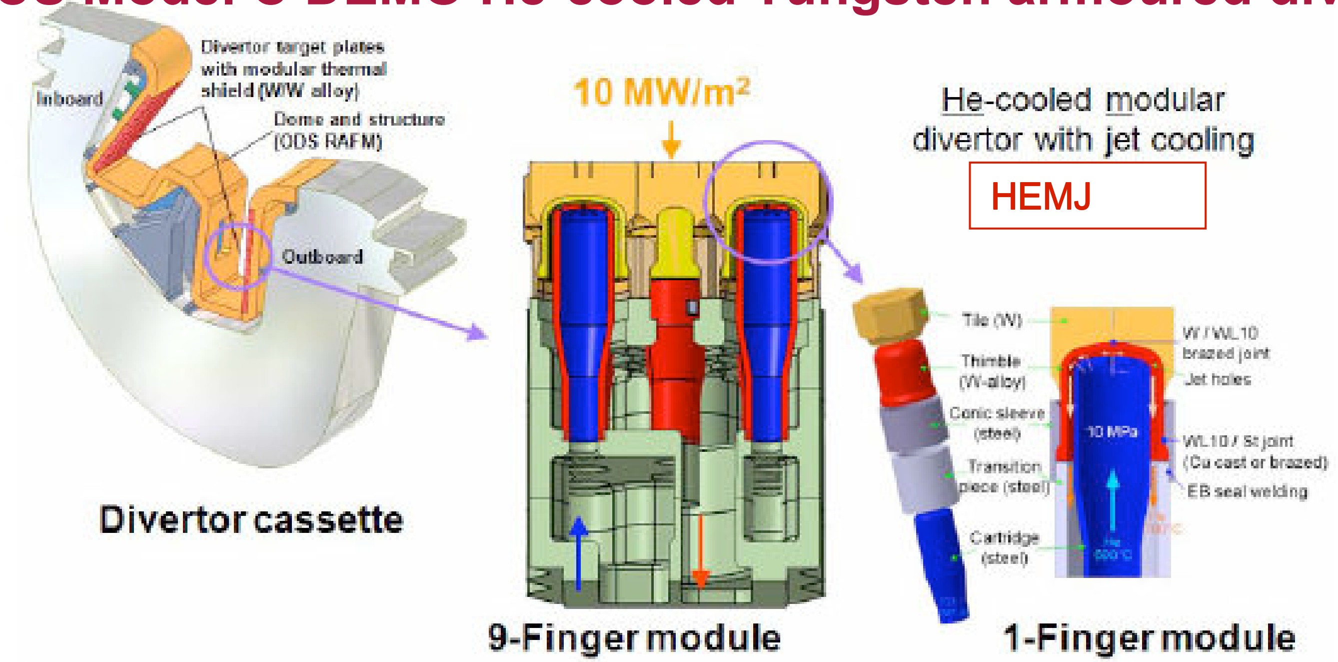


- ITER Phase 1 has a carbon-fibre divertor target in very-high power flux regions (biggest research base) - but CFC too easily eroded and has high T inventory **full Tungsten divertor to be tested in ITER Phase 2.**

DEMO tungsten divertor design:

structural and armour use of tungsten would enable high temperature He-cooling and hence increase reactor thermal efficiency

PPCS Model C DEMO He-cooled Tungsten armoured divertor



- **Tungsten ductile operating window ~ 700°C - 1200°C (lower level DBTT, upper level re-crystallisation) - OK for 14 MeV neutron damage 600°C ↑**
- **He - cooled at 600°C, 10 MPa** • **Braze to W-alloy substructure at 1050°C**
- **Thimbles tested at 10 MW.m^{-2} ≤ 200 cycles** • ~ 200 10^5 thimbles in a reactor!
- **Power performance target still some way to go**

Ref [18] P Norajitra et al

Targets and technical basis for DEMO

Tritium self-sufficiency

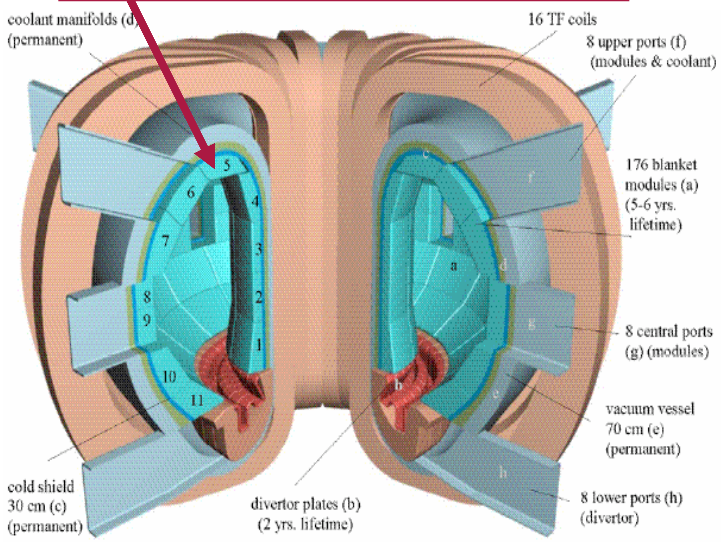
Breeding Blankets

- **Blankets - perhaps the most important and novel parts of the Fusion Power Plant (FPP)**
  - one of the few components which will be common to Magnetic and Inertial confinement Fusion reactors

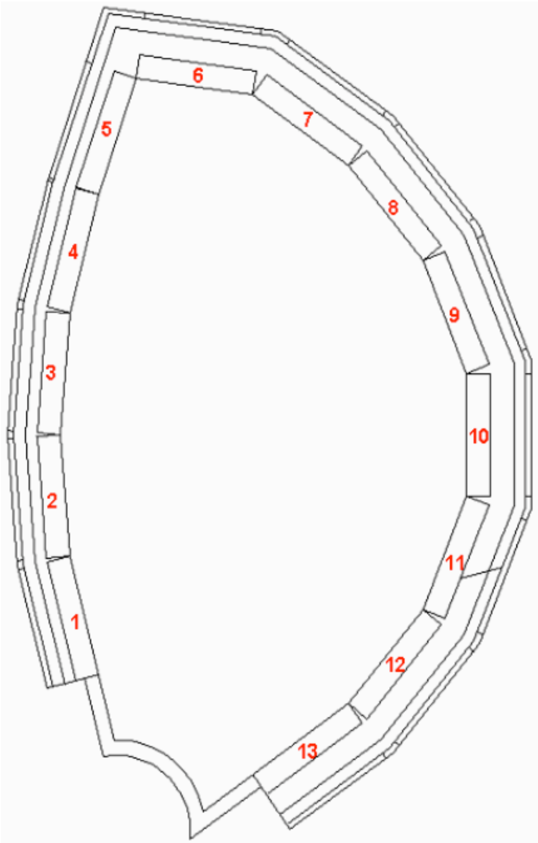
The missions of the FPP fusion blanket are:

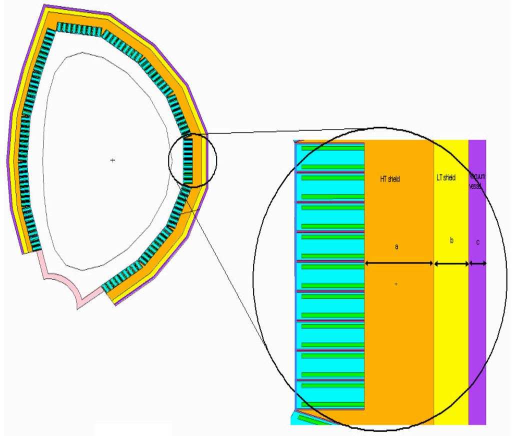
- **breed tritium fuel from the 14 MeV neutron flux, with sufficient efficiency;**
- **integrate a method to recover the bred tritium into the processing plant to produce pure tritium fuel for the power plant, and feed the tritium stocks of a fusion economy;**
- **absorb maximum possible fraction of the energy from the 14 MeV neutron flux;**
- **integrate primary circuit coolant systems to transport the heat produced in the blanket to the power plant 'steam generating circuits';**
- **survive several (5+) years in the intense neutron environment with high integrity and continuing tritium and heat generation efficiency**

DEMO concept - model C



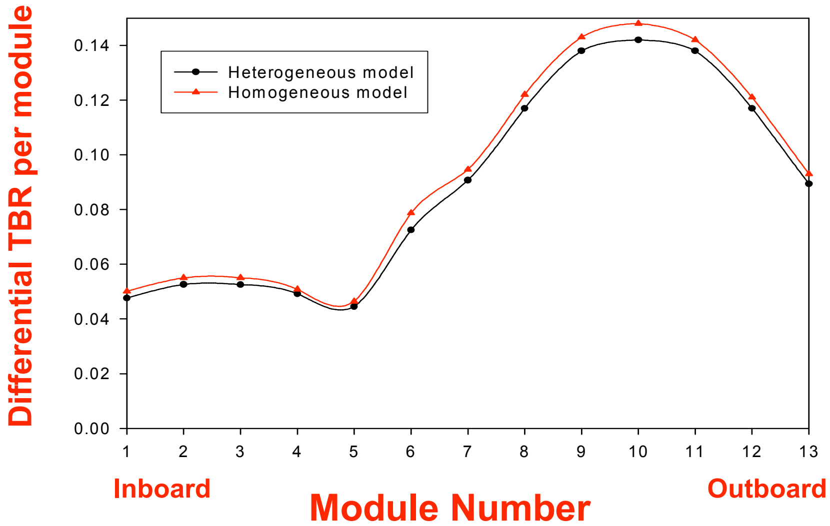
Blanket must cover maximum possible area to achieve Tritium breeding ratio above unity





Based on PPCS Model B with pebble bed blanket

Ref [19]; L W Packer



TBR(global) =1.19

→falls to <1 if inboard modules excluded

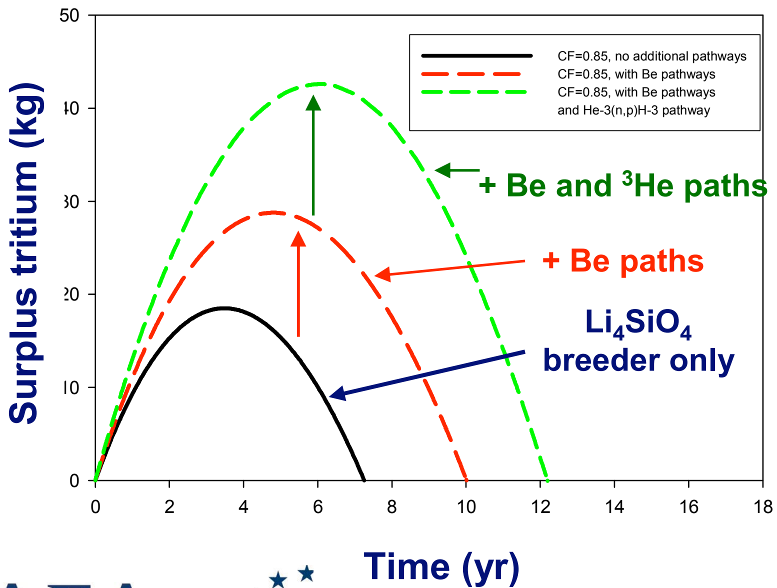
Key blanket design choices

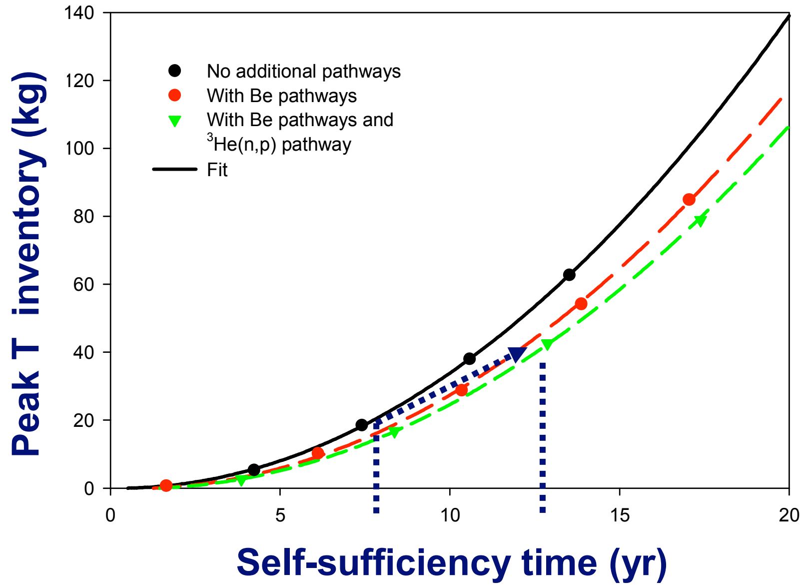
- **Primary coolant**
  – water, helium, liquid metal ($Li_{17}Pb_{83}$)
- **Tritium generating material**
  – lithium ceramic, e.g. lithium orthosilicate (Li_4SiO_4)
  – liquid lithium-lead eutectic ($Li_{17}Pb_{83}$)
- **Neutron multiplier using (n,2n)**
  – beryllium, lead
- **Shielding**
  – water/steel, tungsten carbide
- **Structural material**
  – low-activation ferritic steel, silicon carbide

Complex interaction of all blanket component materials/fluids affects tritium self-sufficiency

- eg. Tritium self-sufficiency of a lithium ceramic blanket with Beryllium multiplier (eg. in pebble form) is enhanced by
- the tritium-producing reactions in the **beryllium** (beryllium pathways):
  $^{9}Be(n,t\alpha)^{3}H$ and $^{9}Be(n,\alpha)\rightarrow{}^{6}He(\beta^{-})\rightarrow{}^{6}Li(n,\alpha)^{3}He$
- and even by the choice of **^{3}He purge gas** (**helium pathways**): **$^{3}He(n,p)^{3}H$**
- **this is especially useful for a low-enrichment blanket (30% ^{6}Li example shown)**

Ref [19]: L W Packer

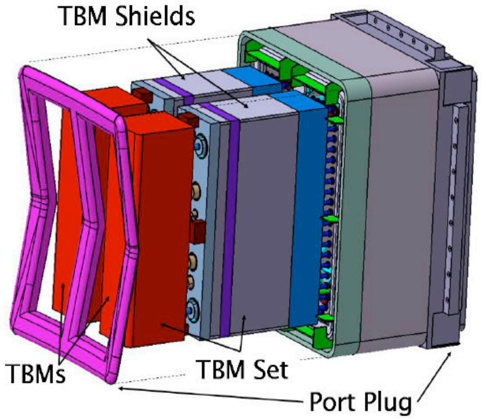


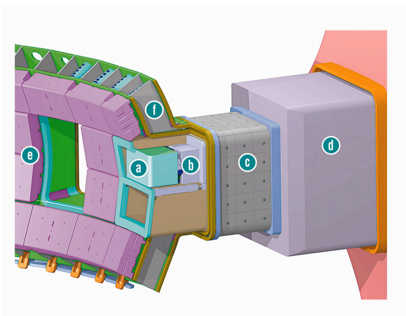


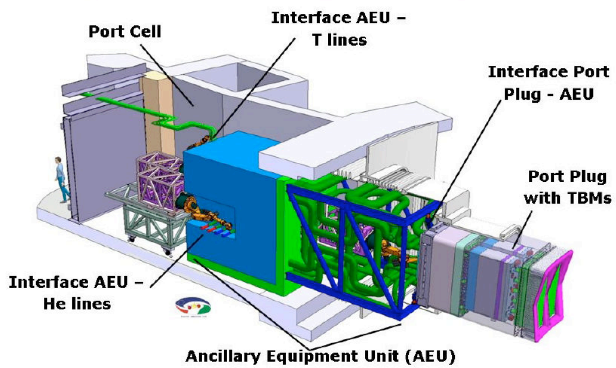
ITER 'Test Blanket Module' Programme is a key stage in DEMO blanket development

ITER phases	TBM types
H-H	"Electro-Magnetic" TBM (**EM-TBM**)
D-D + beg. D-T (low-duty)	"Neutronics" TBM (**NT-TBM**)
D-T (low-duty + beg. high-duty)	"Thermomechanics & tritium control" TBM (**TT-TBM**)
D-T (high duty)	"Integral/Plant Integration" TBM (**PI-TBM**)

Source – 2008 Ann Report of the Association FzK/Euratom – L Boccaccini et al

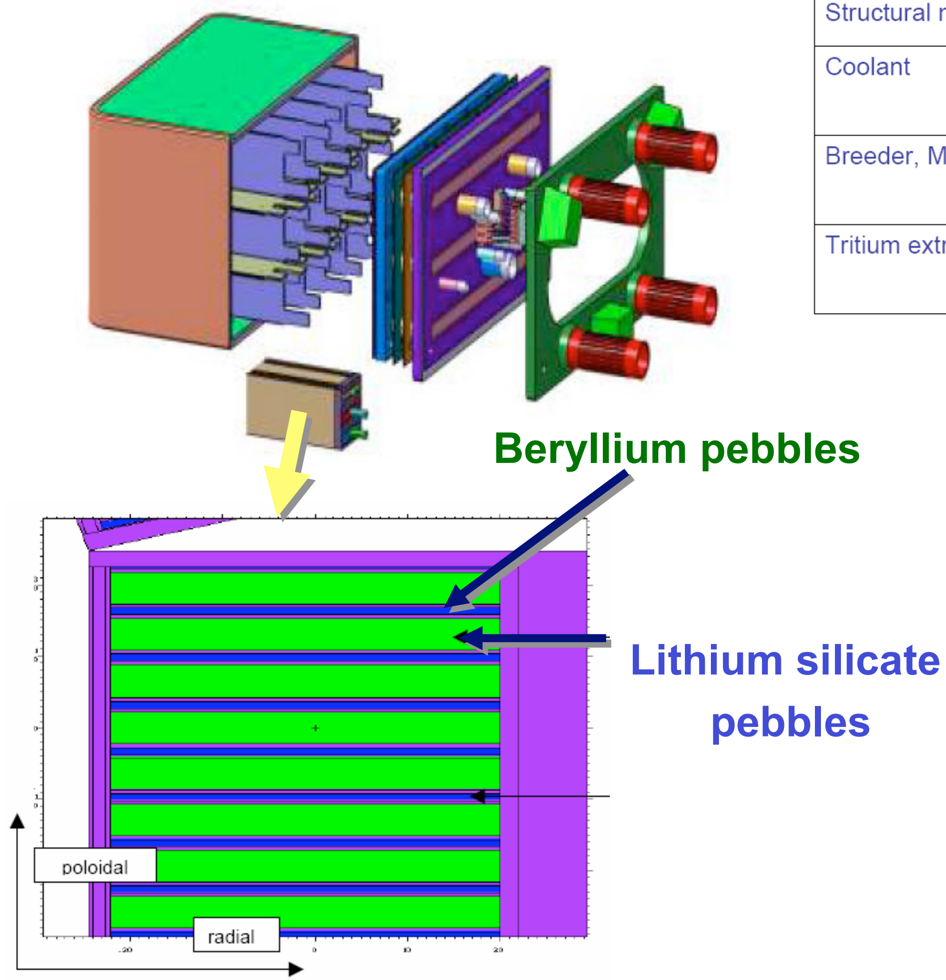




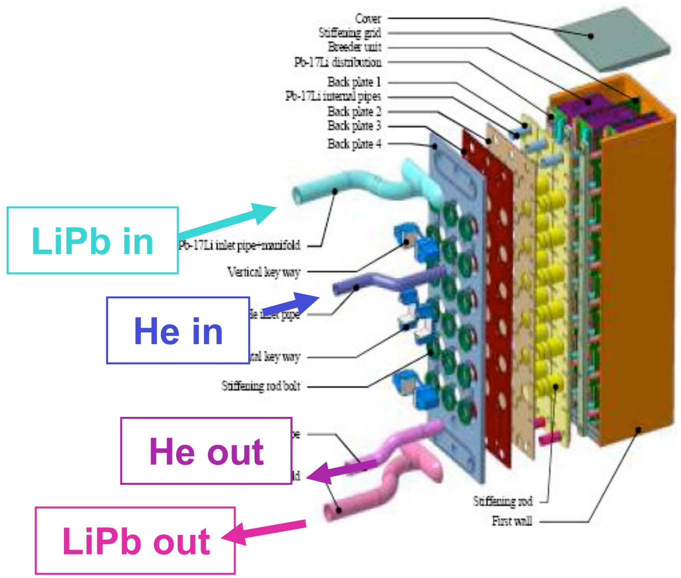


EU blanket concepts for ITER TBM

Helium-cooled Pebble-Bed (HCPB)

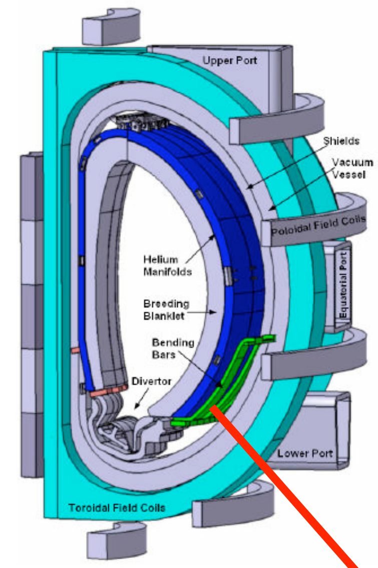


	HCPB	HCLL
Structural material	RAFM steel (EUROFER)	RAFM steel (EUROFER)
Coolant	Helium, 8 MPa, 300/500°C	Helium, 8 MPa, 300/500°C
Breeder, Multiplier	Solid breeder (pebble beds) Li_2TiO_3/Li_4SiO_4, Be/$Be_{12}Ti$	Liquid breeder Pb-15.7Li
Tritium extraction	Low pressure He loop (~1 bar)	Slowly re-circulating PbLi (geodesic pressure)

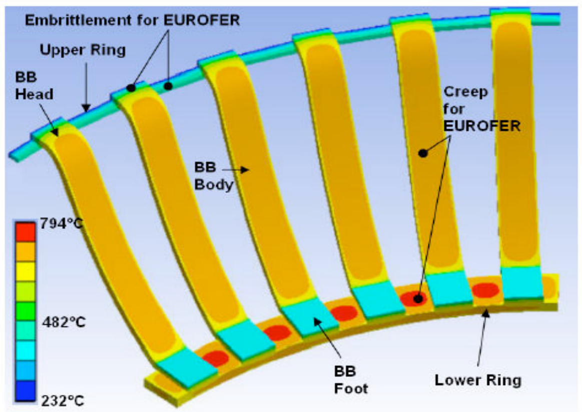


Helium-cooled Lithium Lead (HCLL)

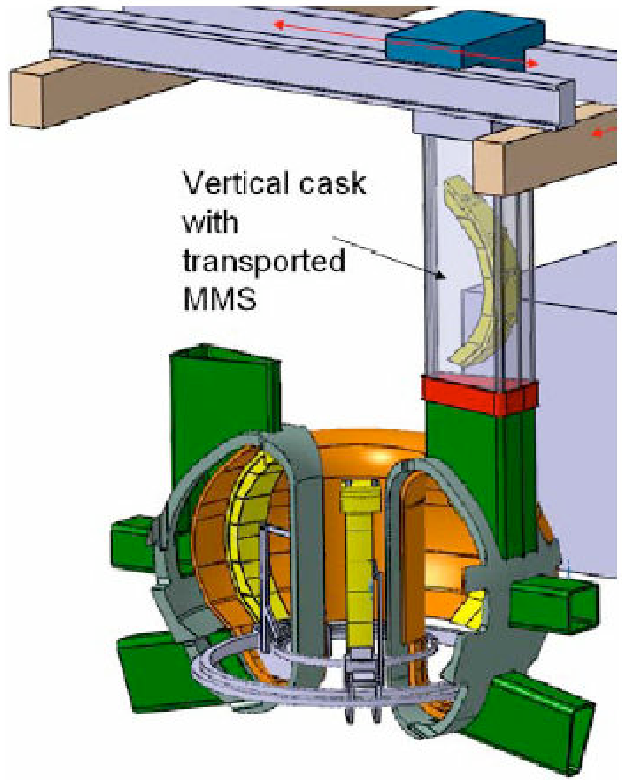
Blankets choices affect all aspects of a DEMO design



- In-vessel:
  - basic radial build;
  - first wall conductivity (breakdown), magnetic field ripple;
  - plasma shape
  - allowable temperatures



- Remote Handling concepts /area layout:
  - as a consumable item!



Source – 2008 Ann Report of the Association FzK/Euratom – EFDA/06-1454 study E Magnini et al

Blankets choices affect all aspects of a DEMO design

- Energy use of secondary circuits (and hence nett efficiency) – high pumping power required for:
  - MHD-induced pressure drops for Liquid-metal designs;
  - high-flow, high pressure Helium cooling (375 MW pump power for PPCS Model B).
- Character of 'Balance of Plant':
  - PWR-like primary circuits for a water-cooled blanket → piggy-back on Fission-plant engineering;
  - high-pressure helium cooling primary circuits – may or may-not be developed by Generation IV fission – separate development programme?
- In-vessel operational safety/availability:
  - hazards of interaction between coolant and blanket material (eg. H_2O – Li ceramics or H_2O – beryllium);
  - hazards from corrosion by coolant (Li molten salts, liquid LiPb);
  - rupture of high pressure coolant (water raises steam – rupture to vessel?; He ruptures module – regenerates cryopump?).
- Waste inventory – blanket change several times in reactor life → large waste inventory → need to minimise changes and induced activity.

Fusion Development Issues

- the key importance of the choice of Blanket concept points up the problems of the ‘standard’ DEMO Fast Track model

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Plasma performance	Disruption avoidance	2	3		R	R	R
	Steady-state operation	2	3		r	r	r
	Divertor performance	1	3		R	R	R
	Burning plasma (Q>10)		3		R	R	R
	Start up	1	3		R	R	R
	Power plant plasma performance	1	3		r	R	R
Enabling technologies	Superconducting machine	2	3		R	R	R
	Heating, current drive and fuelling	1	2		r	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Remote handling	1	2		R	R	R
Materials, Component performance & lifetime	Materials characterisation			3	R	R	R
	Plasma-facing surface	1	2		3	4	R
	FW/blanket/divertor materials		1	1	3	4	R
	FW/blanket/divertor components		1	1	2	4	R
	T self sufficiency		1		3	R	R
Final Goal	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

- Fast Track scenario implies changes between DEMO Phase 1 & Phase 2 should be limited; but
- …the ITER TBM programmes **as currently conceived** will not give EU full information on all possible technologies/problems for DEMO Phase 1 – possibility of major design changes; plus
- …. DEMO Phase 1 has to be designed whilst the TBM programme is still running.

EU ITER TBM testing limits
DEMO Phase 1 possibilities in Europe

Returning to PPCS table:

PPCS Model	Plasma physics	Structural material	Other blanket materials (coolant)	Other divertor materials (coolant)
A	'Near-term'	Eurofer	LiPb/water	W/Cu/water
AB	'Near-term'	Eurofer	LiPb/He	W/He
B	'Near term'	Eurofer	Li_4SiO_4/Be/He	W/He
C	'Medium term'	Eurofer/ODS	LiPb/SiC/He	W/He
D	'Medium term'	ODS/SiC	LiPb/SiC	W/LiPb

Only these Blanket concepts tested in ITER tokamak environment /integrated systems necessary input to DEMO Phase 1.
JA/CN will test water-cooled and CN will test Dual Coolant – but test results will not be available to EU without licensing – do we need to review EU strategy?

Targets and technical basis for DEMO

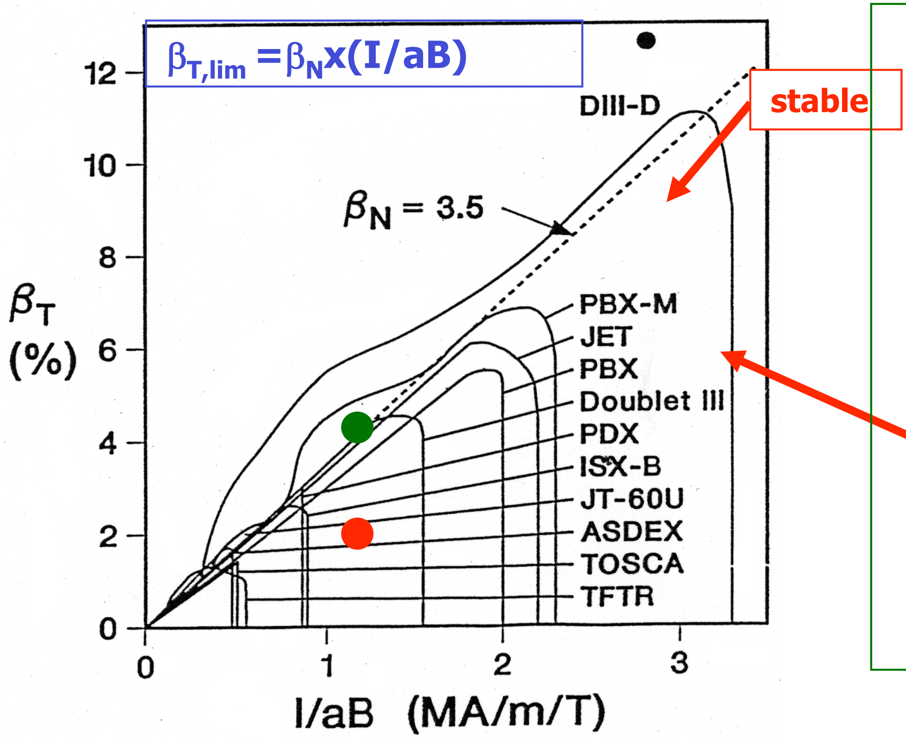
Physics issues

DEMO Physics Issues

- ITER is intended to answer all relevant Physics questions at 'reactor scale'.
- However, some physics issues should be emphasised as possible challenges pre-DEMO because:
  - the parameters in the DEMO physics scenarios are key to economic fusion – normalised plasma pressure (β) and normalised density ($N=n/n_G$) and 'standard' DEMO values lie beyond the ITER baseline scenario; or
  - The physics relates to technical feasibility where DEMO scenarios go beyond ITER – steady-state current drive, especially intrinsic 'Bootstrap' current fraction and highly-radiating discharges (f_{rad}>90%) to keep Divertor power loading within engineering limits.

DEMO operational β: –DEMO β is well beyond the ITER level, but just in stable envelope of present tokamaks.

- DEMO models require β_N in the range 2.7/2.8 (PPCS A/B) → 3.7 (PPCS D)
- High-β plasmas suffer from Magneto-hydrodynamic (MHD) instabilities – 'ballooning', 'kinking' and 'tearing' plasma field lines destroy confinement.

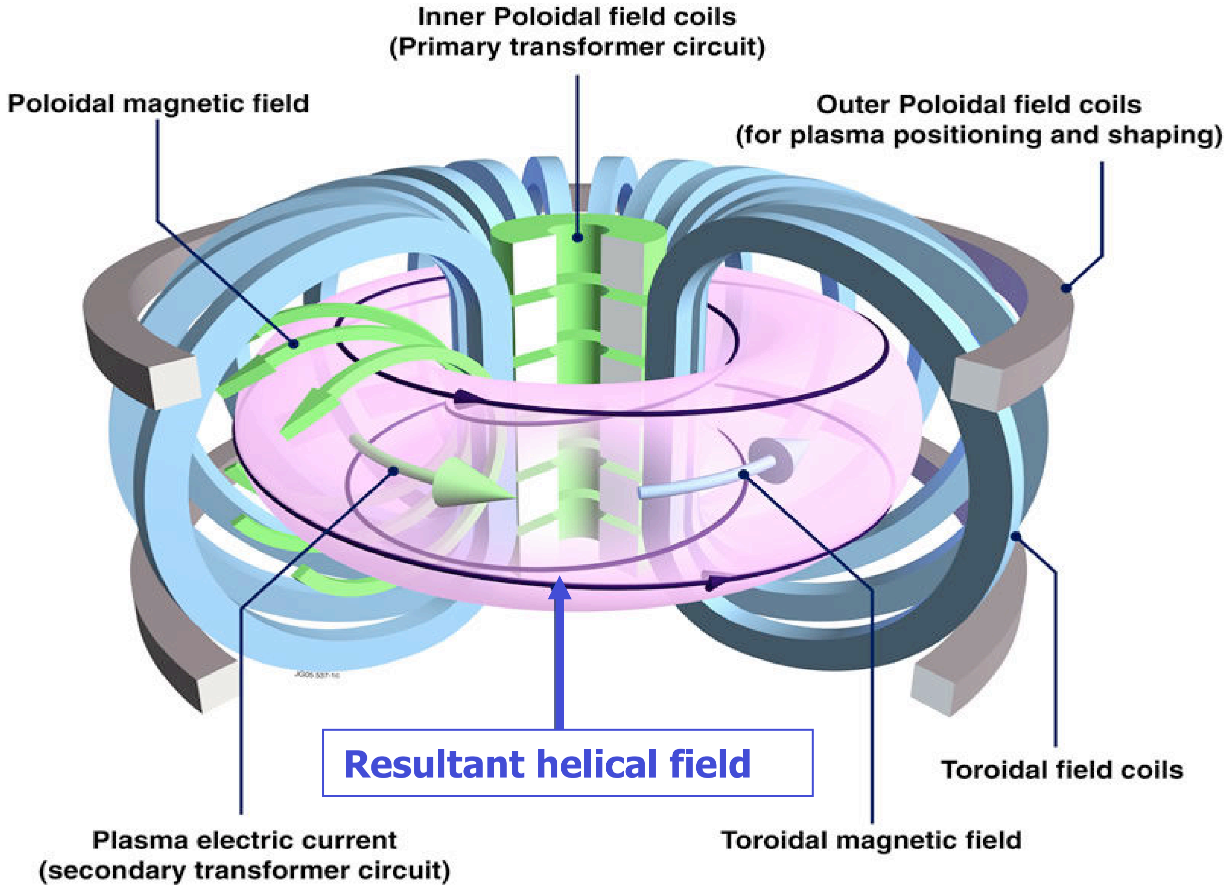


- Figure of merit β_N
  -- $\beta_N = \beta_T/(I_p/aB_T)$
- Stable region for Tokamaks:
  - increases with plasma current;
  - decreases with toroidal field
  but
  - falls catastrophically if I_p/B_T gets too high.
- $I_p/B_T \sim 1/q$ 'q' is the **Safety Factor** (no. of toroidal transits of a field line per poloidal transit)

● ITER Q=10 ELMy H-mode ● DEMO PPCS Mod C

Tokamak basics:

'Safety-factor' -- q

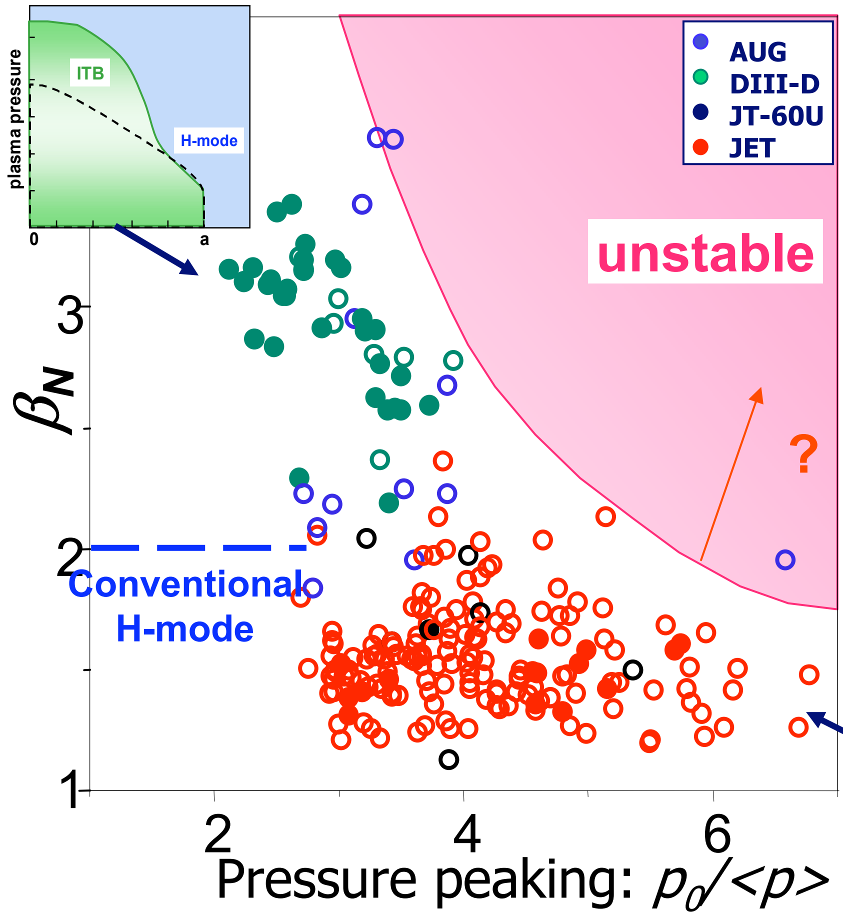


- On a surface in the plasma, 'q' =toroidal transits per poloidal transit
- q (edge) ~ 3 is approx the safe lower limit

‘Bootstrap’ current:
interaction with β–limits

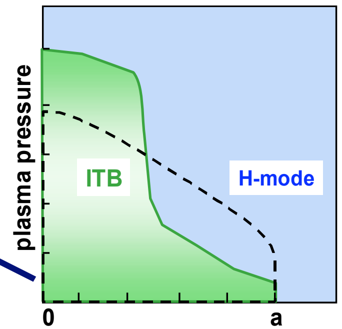
- Trapped particles in a tokamak field, combined with the pressure (density) gradients in the plasma generate an intrinsic current – the ‘Bootstrap current’.
- Steady-state operation of a fusion power plant requires external current drive – to minimise the power requirements, a high fraction of Bootstrap current is required.
- Conditions for high bootstrap current lead to reduction in attainable pressure (β) in two ways:
  - plasma pressure gradients, are strongest off-axis – currents lead to a reduction in plasma inductance, and hence to reduced β – limit; and
  - strong gradients lead to kink instabilities and reduce the stable domain.

High 'Bootstrap' current fraction:
interaction with β–limits



pressure and current profiles (low-l_i) unfavourable for stability

→ only weak barriers, at large radius *stable at high-β*



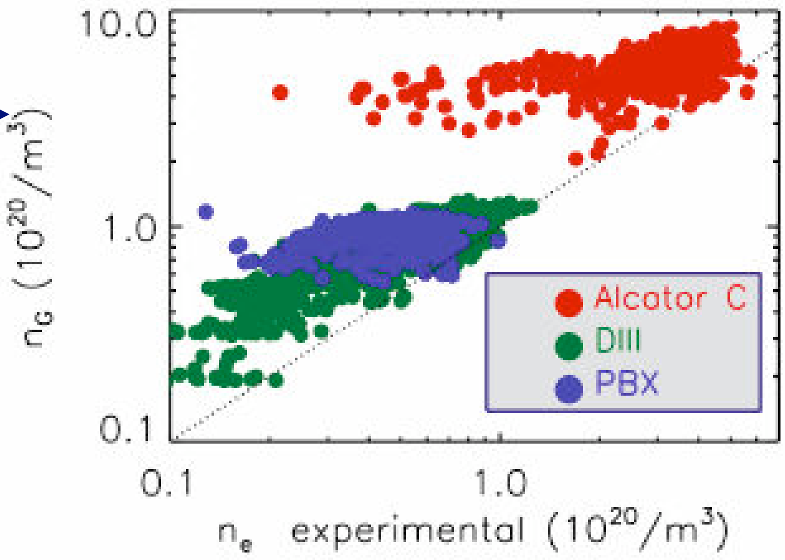
High density operation:

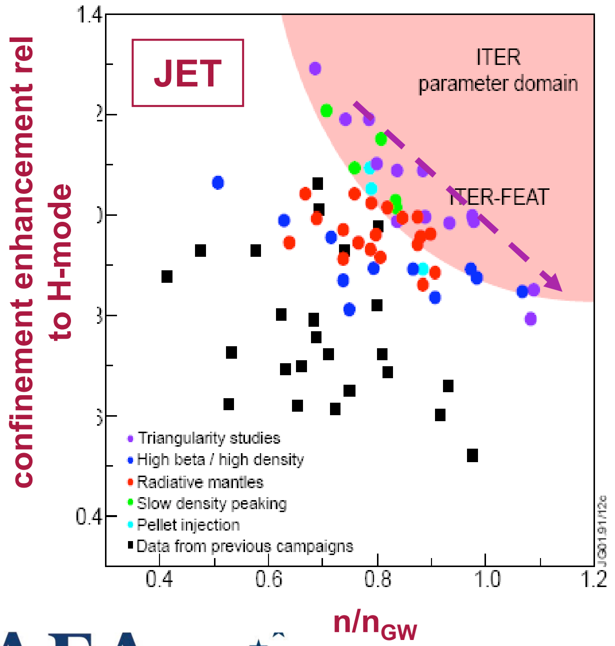
above the Greenwald limit

All the PPCS DEMO designs rely on operation above the empirical Greenwald density limit $N_{GW} = n/n_{GW} = 1.2$ (Mod A) – 1.5 (Mod C&D).

But the Greenwald density limit agrees with experimental data →

$$n_{GW} = I_p \big/ (\pi a^2) = 1.59\, g \frac{B_T}{q_{95} R}$$





← …also H-mode energy confinement quality tends to decline above Greenwald limit.

Radiation-dominated plasmas

- Plasmas with a high fraction of radiated power (f_{rad}) are essential to keep the power loading levels on the DEMO divertor within tolerable levels.
- All PPCS models assume 90% of power radiated in the plasma (mainly bulk). This is important against ELMs.

Parameter	Model A	Model B	Model C	Model D
Divertor Peak load (MW/m^{-2})	15	10	10	5
Z_{eff}	2.5	2.7	2.2	1.6

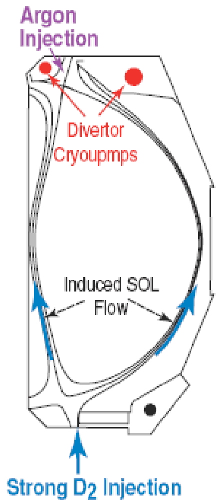
- Present database for high f_{rad} plasmas with high confinement and high β is almost non-existent.
- Such plasmas are required even for tungsten-armoured divertor, to keep not only power, but also erosion levels within limits.
- Plasmas have to be compatible with low impurity content.

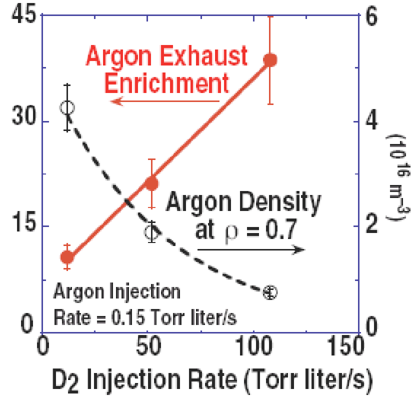
Radiation-dominated plasmas:
some way to go

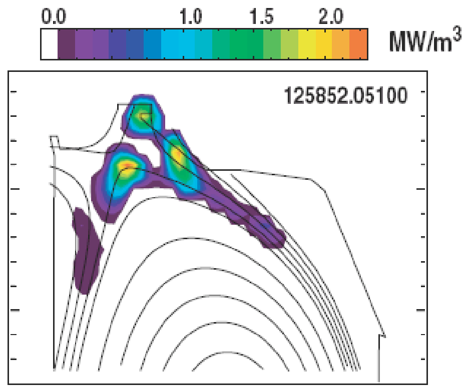
- Upstream gas puffing and divertor exhaust induce strong SOL flow ⇒
- Very high enrichment value obtained ⇒
- P_{rad}/P_{NBI} ~ 60% with $Z_{eff} \simeq 2.0$

DIII-D

[20] M R Wade et al, 21st IAEA FEC Chengdhu, 2006

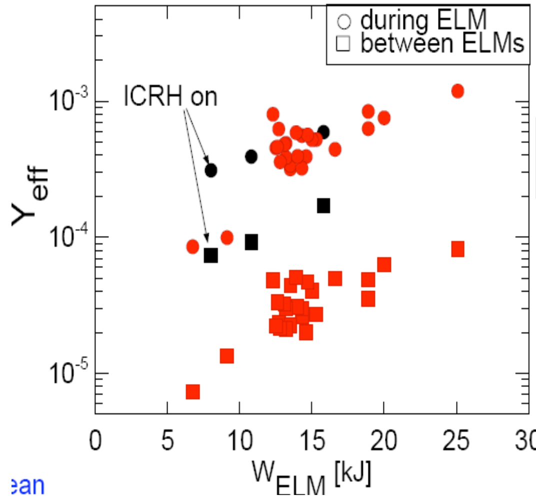






- $\beta_N = 2.6$, $H_{89} = 2.0$, $G = 0.4$ maintained

- DIII-D plasmas with impurity seeding reach 60% radiation and keep ‘Model A’ level β and purity.
- ELM-effect on impurities →ASDEX-U sputtering of tungsten is enhanced factor~10 in ELMs



ASDEX-U

[21]R Dux et al, 21st IAEA FEC Chengdhu, 2006

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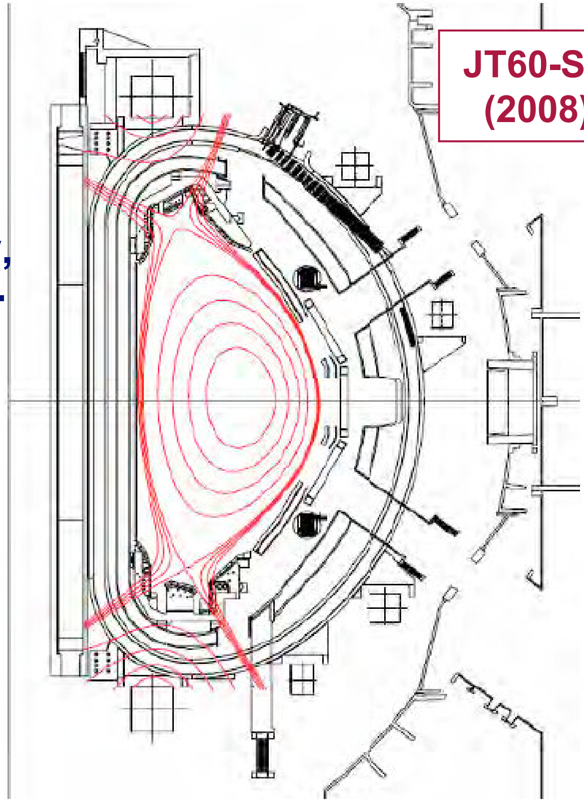
Solving physics issues in parallel to ITER?
– 'satellite' tokamak

- ITER's nuclear device status makes it fairly inflexible to try new concepts/solutions in 'mock-up' → concept of a 'satellite tokamak'
- Only presently approved satellite is JT60-SA (Japan/EU)-starts 2015!
- Main mission of JT60-SA is 'steady-state' advanced scenarios with ~ 100% Non-inductive current drive …but
  - JT-60SA will operate at N_{GW} ~ 0.5 -0.8
  - JT60-SA will have a carbon water-cooled divertor (up to 15 $MW.m^{-2}$)
  - …many DEMO physics issues eg.high density, high radiation and divertor technology power-loading interface are left unanswered.



JT60-SA (2008)

Nominal Parameter	Draft 2008
Plasma R major radius [m]	2.97
Plasma a minor radius [m]	1.18
Plasma Ip [MA]	5.5
Plasma A aspect ratio	2.5
Plasma κx	1.93
Plasma δx	0.57
Plasma q95 (within,betap li range)	~3
Toroidal Field Bt [T]	2.25
Plasma Volume [m³]	~140
Plasma n Greenwald [10^{20}/ m³]	1.24
Shape Parameter - S	6.1
Flattop flux @li=0.85 [Wb]	~8
TF Ripple at R+a	0.85%

Targets and technical basis for DEMO

Enabling technologies

UKAEA Fusion
Working with Europe

Enabling Technologies for DEMO/FPP

- Superconducting coils at large 'reactor scale' and Tritium Inventory control and processing of Tokamak T-loaded exhaust should be solved by ITER (the latter will be a licensing requirement)

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Enabling technologies	Superconducting machine	2	3		R	R	R
	Heating, current drive and fuelling	1	2		r	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Remote handling	1	2		R	R	R
Materials	Materials characterisation			3	R	R	R

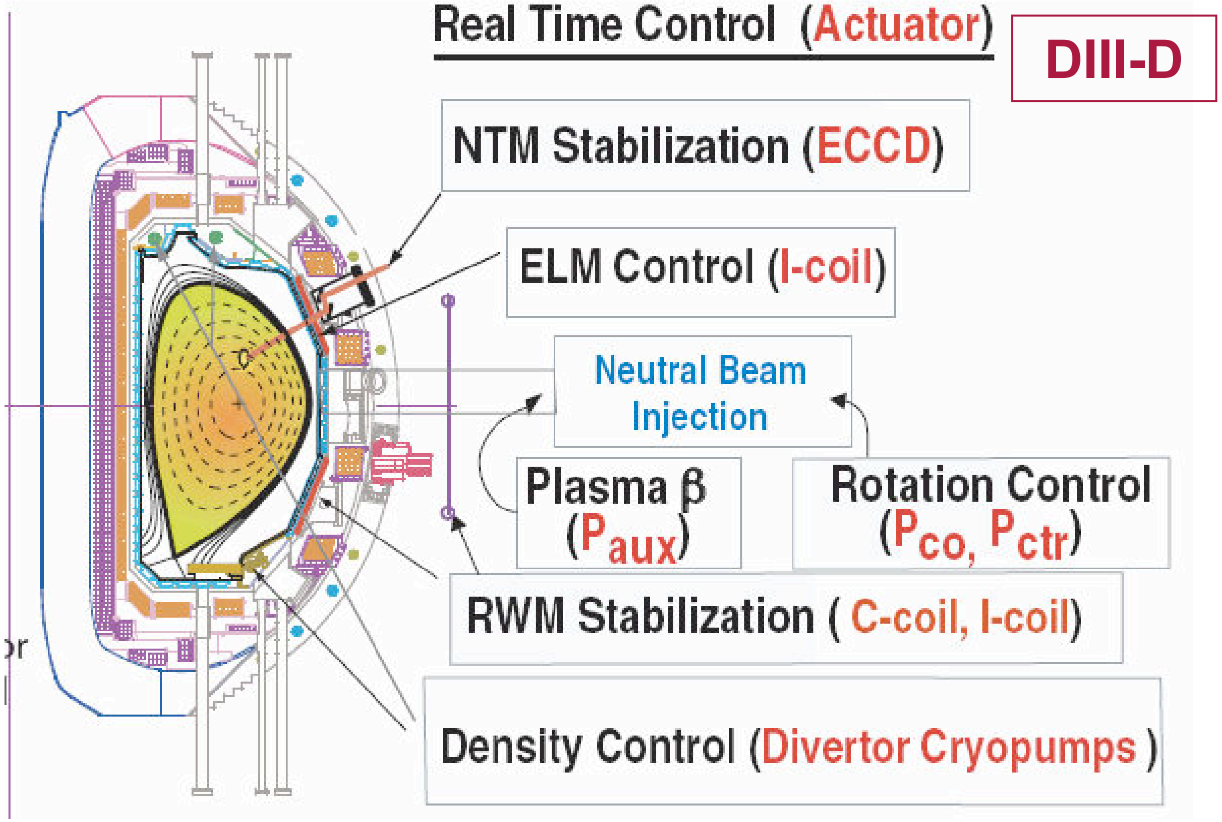
- Other 'Enabling technologies (H&CD, Diagnostics&Control and Remote Handling) will make great strides forward on ITER, but will not be brought to 'DEMO Phase 2' readiness – programme must consider how to bridge the Gap between ITER and DEMO (not the same solution for all three fields).

Output:		
	1	Will help to resolve the issue
	2	May resolve the issue
	3	Should resolve the issue
	4	Must resolve the issue

Input:		
	r	Solution is desirable
	R	Solution is a requirement

UKAEA September 2007 (revised/improved version of original table in UKAEA FUS 521, 2005).

Modern advanced tokamaks have plethora of feedback control systems to achieve high performance



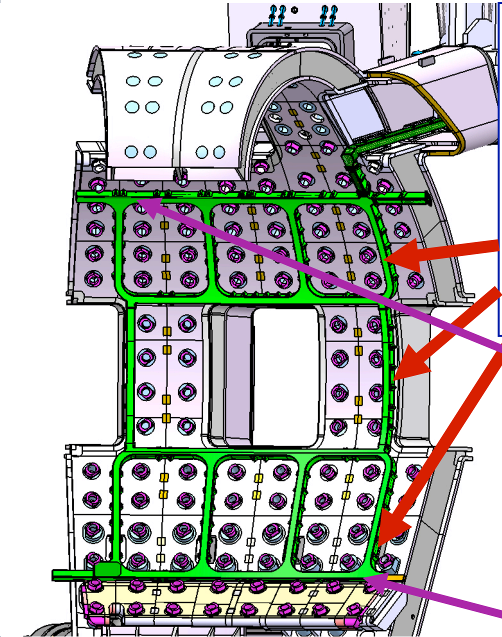
- All actuators require input plasma diagnostic measurement for feedback control.
- Some (eg. control coils) are in vacuo – impossible for a reactor.

DEMO development issues for Diagnostics and control

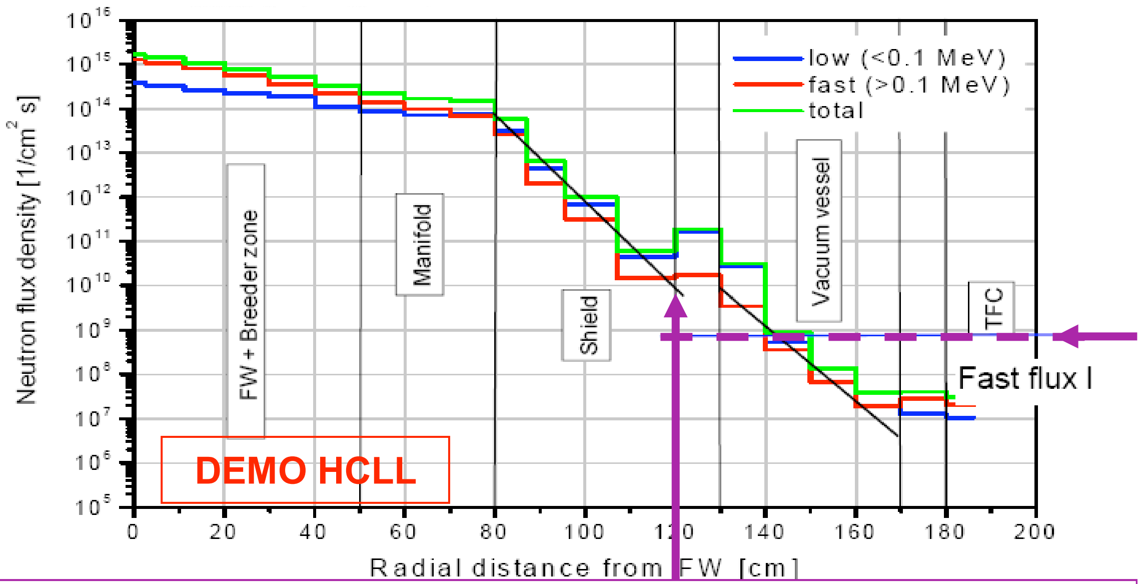
- Diagnostics:
  - survivability of windows in high radiation environment (ITER systems will only see <1dpa – DEMO up to 5-10 dpa per fpy);
  - availability of lines-of-sight through blankets (spectroscopic and optically-based measurements of plasma temperature, density, current profile etc).
- Control systems:
  - Inability of control coils to survive in-vessel radiation doses – poor controllability/response time using coils placed far from plasma;
  - Restriction of lines-of-sight, and limits to numbers of magnetic-coil measurements in-vessel (system complexity optimisation) → sparse dataset available → development of control algorithms based on sparse data.
- Control using sparse data and remote actuators needs piloting on flexible 'satellite tokamak'. Already ITER has compromised and plans in-vessel coil-set for ELM, vertical stability and Resistive Wall Mode control.

ITER In-vessel coils for vertical stability/ELM control/Resistive Wall Mode control:

coils are behind the Blanket modules



- ELMs (edge modes expelling particles and energy) are thermal pulses of 500μs duration, the peak energy density must be < 0.5MJ.m^{-2} to avoid excessive damage to walls and divertor.
- In-vessel coils → magnetic perturbations → destabilise edge modes whilst still small → small energy deposition



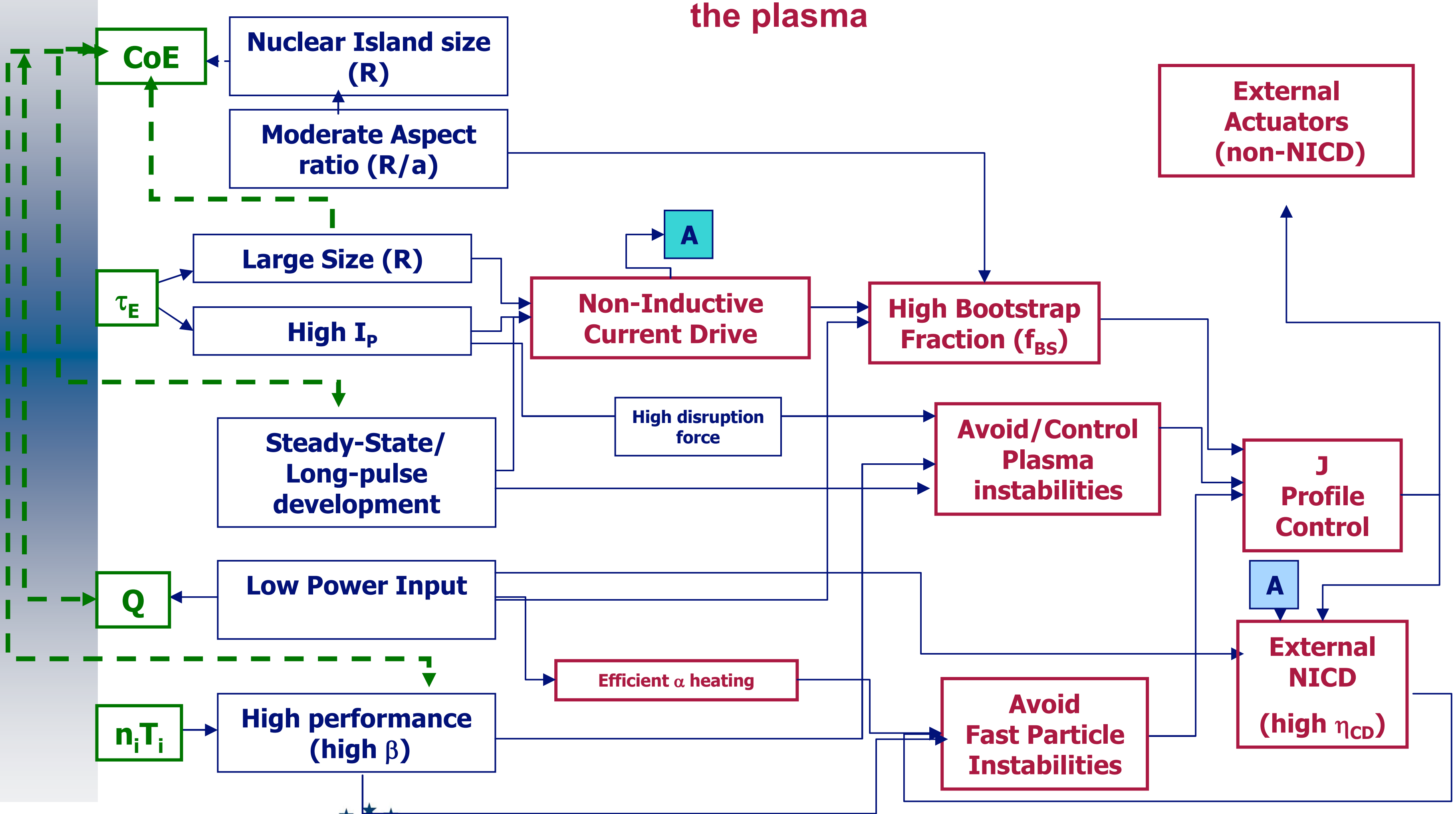
Equivalent position in DEMO – coil insulation sees ~ 100x lifetime limit.

- Other fast response coils correct vertical instabilities
- ..ex-vessel stabilisation systems not fast enough

Source -2008 Ann Report of the Assoc FzK/Euratom – U Fischer et al

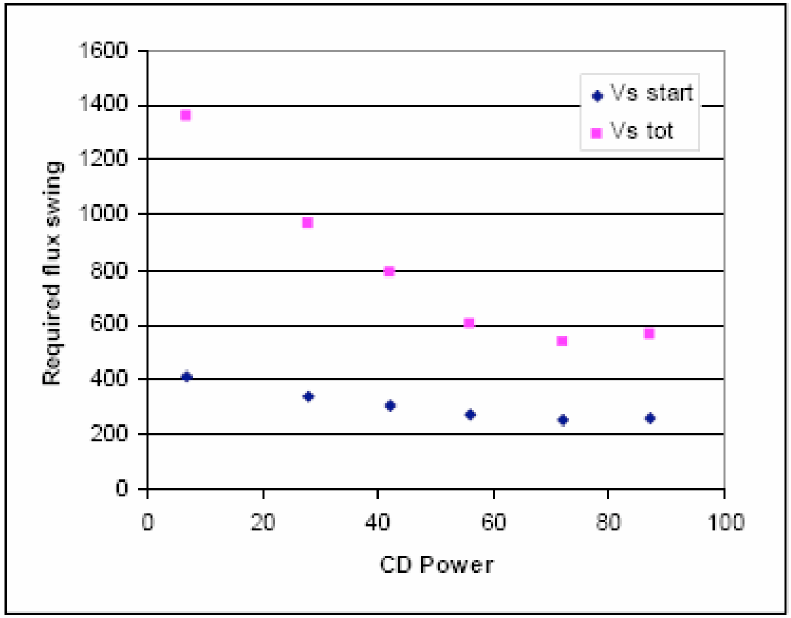
Current and Current Profile Relationships in a Reactor-sized experimental device

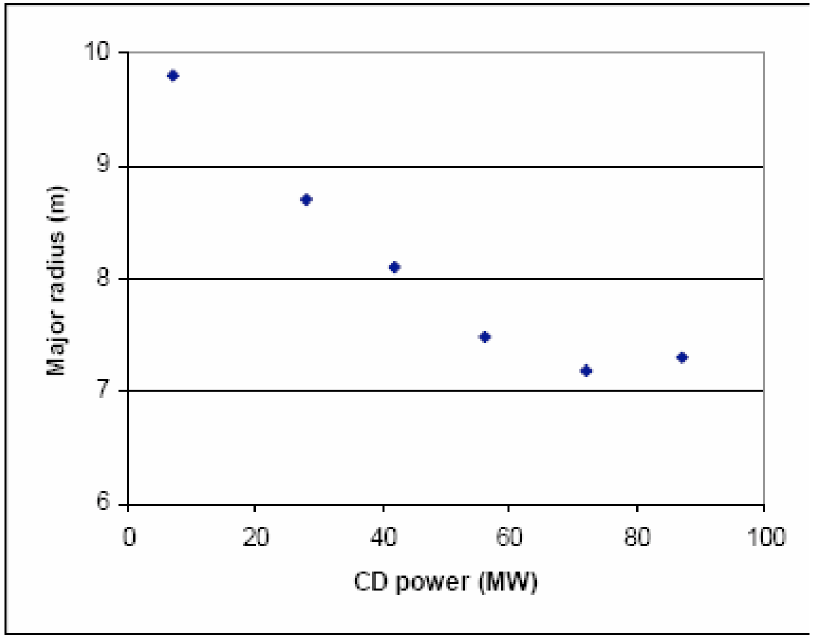
All Current Drive systems rely on creating or injecting a fast particle population in the plasma



Effect of Additional current drive on size of a Pulsed Device

Major radius and flux swing are substantially reduced by adding CD power (here 2 MeV NNBI)





Flux in solenoid ~ R^2 → Vs drive

Fixed pulse length – 8 hours

Courtesy of D. Ward (UKAEA)

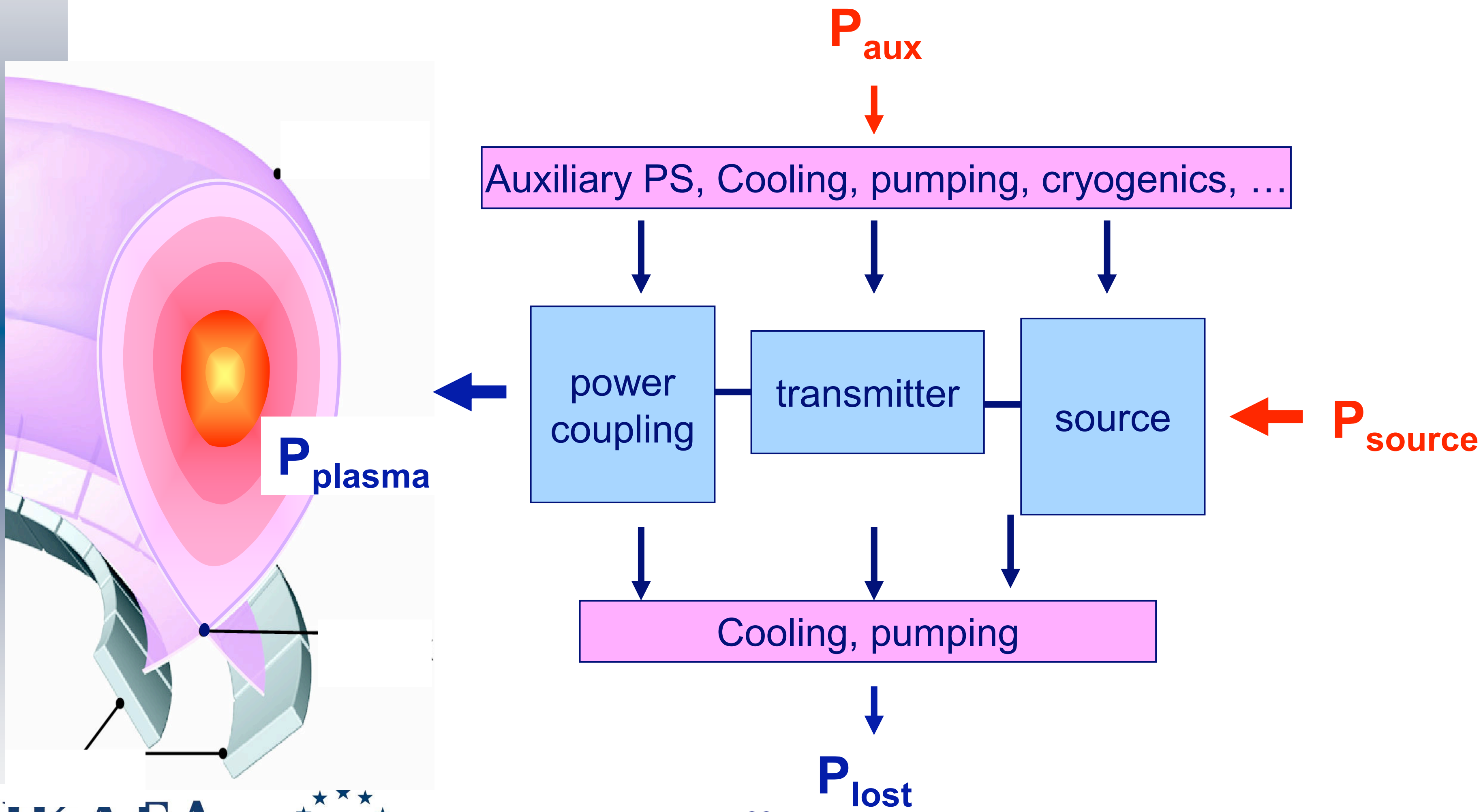
Cost of Electricity: headline consequences for H&CD (I)

$$\mathrm{CoE} \propto \left(\frac{1}{A}\right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4}\,\beta_N^{0.4}\,N_{GW}^{0.3}}$$

- **Availability** has the biggest leverage on cost – to achieve high availability:
  - Continuous operation – few outages for ‘replenishing actions’;
  - High Reliability – application of ‘industrial’ methods;
    – impact of choice of materials and fabrication techniques;
    – simplification of systems;
    – use of margins;
    - standby redundancy possibilities (last resort)
  - Maintenance infrequent and quick – use of margins;
    – simplification of systems.

H&CD Wall Plug Efficiency Issues

$$\eta_{WP} = P_{plasma} / (P_{source} + P_{aux})$$



Cost of Electricity: headline consequences for H&CD (II)

$$\mathbf{CoE} \propto \left(\frac{1}{A}\right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4} \beta_N^{0.4} N_{GW}^{0.3}}$$

- **Nett electrical output** depends on Heating and Current Drive Efficiency.
- P_e depends on Real Site Q (Q_{site}) – note **not Q_{fus} !!**.

$$P_e = (Q_{site} - 1) \cdot P_{in}$$

$$Q_{site} = P_{fus} / (P_{wp} / (\eta_{wp}) + P_{mag} + P_{BOP})$$

- P_{WP} is the 'wall plug' H&CD power entering the Tokamak;
  η_{WP} is 'wall plug efficiency' of H&CD system
  P_{mag} is the power used in magnetically containing the plasma;
  P_{BOP} is the 'balance of plant' power – cooling for divertor, blanket etc.

Cost of Electricity: headline consequences for H&CD (III)

- Using superconducting magnets P_{mag} is small (10s of MW for ITER).
- Unless blanket and divertor are helium cooled, P_{BOP} is <100 MW
- Hence for 'near-term physics DEMO options'
  → Q_{site} is dominated by H&CD system 'efficiencies'.

$$P_e \sim \left(P_{fus}\left(\eta_{wp}\right)/\, P_{wp} - 1\ \right)\cdot P_{in}$$

Current Drive figure of merit of efficiency

$$\gamma = \frac{RI_{CD}}{P}\frac{n}{10^{20}}(m^{-2}AW^{-1})$$

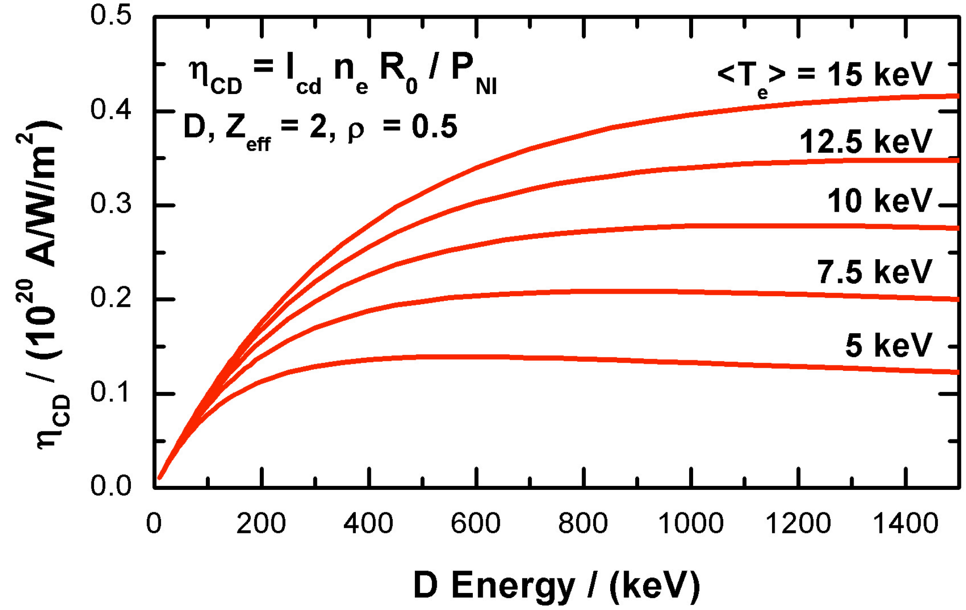
- Hence for a machine of **given size,** with low P_{BOP}, P_{mag}

$$\mathrm{CoE} \propto (\frac{1}{A})^{0.6}\frac{1}{\eta_{th}^{0.5}}\frac{1}{\left(\eta_{wp}\gamma_{CD}\right)^{0.4}\beta_N^{0.4}N_{GW}^{0.3}}$$

H&CD efficiency for DEMO:

assumptions vs reality

DEMO studies (eg. PPCS) assume High energy (1.5 MeV) NBI as reference Current Drive system At DEMO temperatures around 20 keV, expect γ ~ 0.4 - 0.5

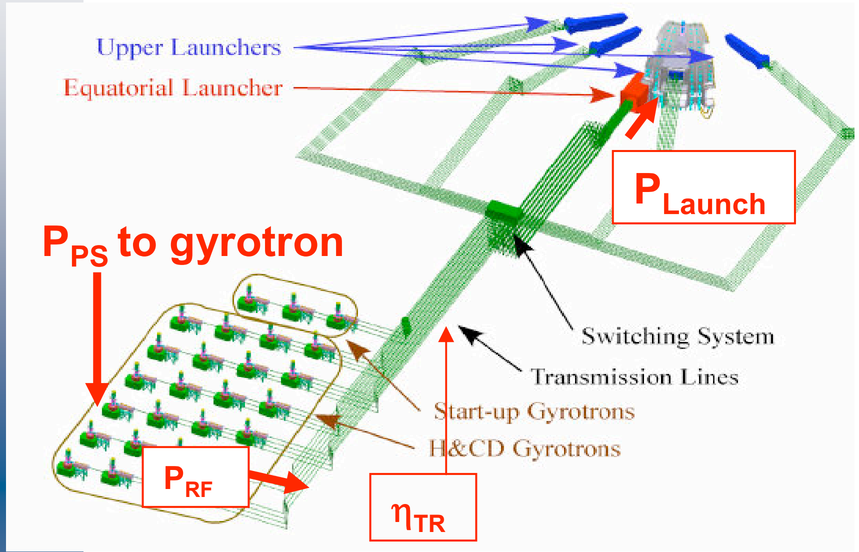


Predicted current drive efficiencies extrapolated to DEMO temperatures:

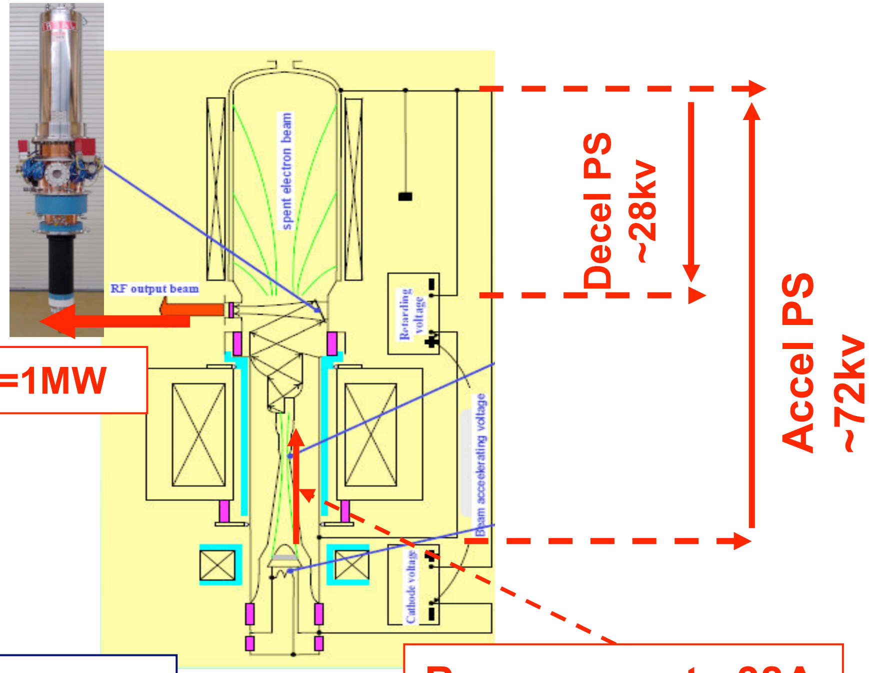
Neutral Beam (1.5 MeV)	**γ ~ 0.4 - 0.45**
Electron Cyclotron CD	**γ ~ 0.15**
Ion Cyclotron	**γ ~ 0.3 - 0.4**
Lower Hybrid CD	**γ ~ 0.3 - 0.35**

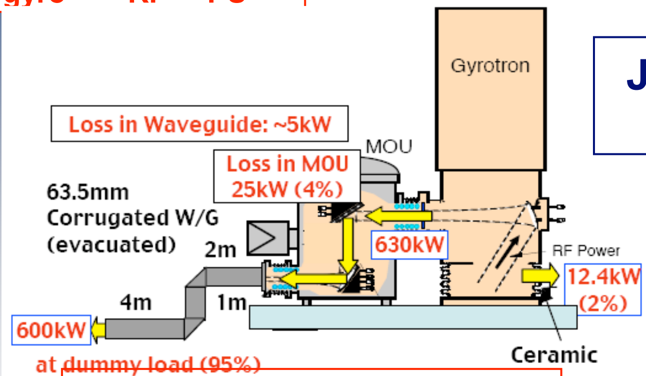
H&CD efficiency for DEMO:

assumptions vs reality (II) - ECRH system efficiency: ITER System



$\eta_{gyro} = P_{RF}/P_{PS}$





Japanese Gyrotron
η_{gyro} = 55%

η_{TR} ~ 600/630 ~ 0.95

For ECRH η_{WP} ~ 0.55x0.95 ~ 0.52

See eg. Ref [22] Kasugai et al., and refs therein

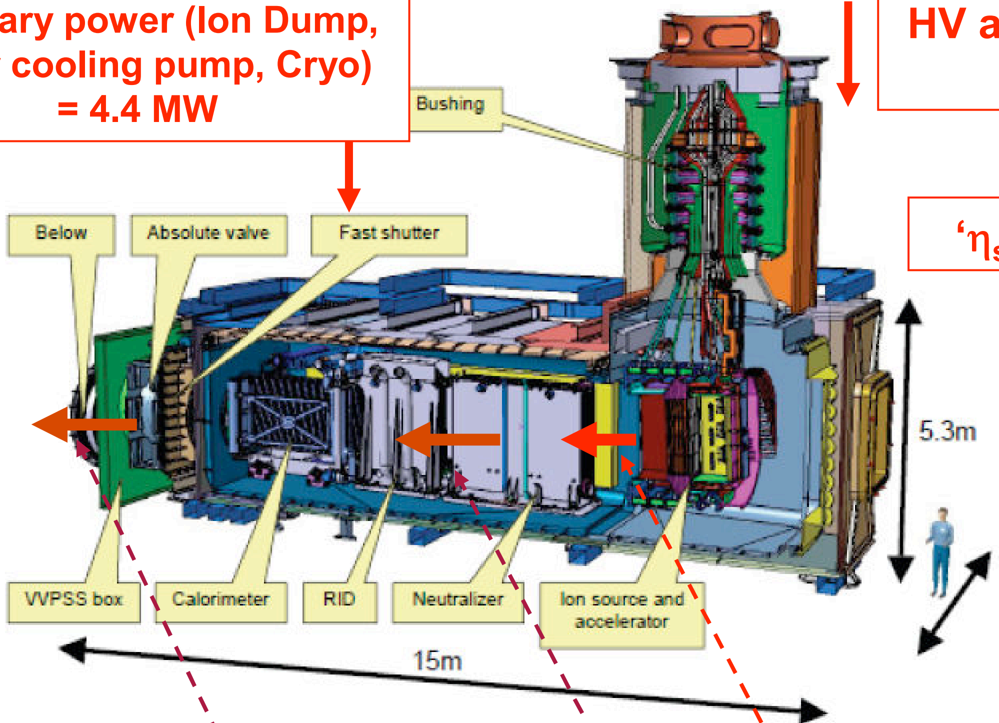
UKAEA Fusion Working with Europe

H&CD efficiency for DEMO:

assumptions vs reality (III) - NBI system efficiency: ITER System

Auxiliary power (Ion Dump, water cooling pump, Cryo) = 4.4 MW

HV and Source power = 58.2 MW



'η_{source}' ~ 40.8/(58.2+4.4) ~ 0.66

Source – courtesy R S Hemsworth - ITER

NB to plasma = 18.8 MW

Neutralised Beam = 23.2 MW

Accelerated beam =40.8 MW

η_{TR} ~ 18.8/40.8 ~ 0.46

For NBI η_{WP} ~ 0.66x0.46 ~ 0.30

H&CD efficiency for DEMO:

assumptions vs reality (IV)

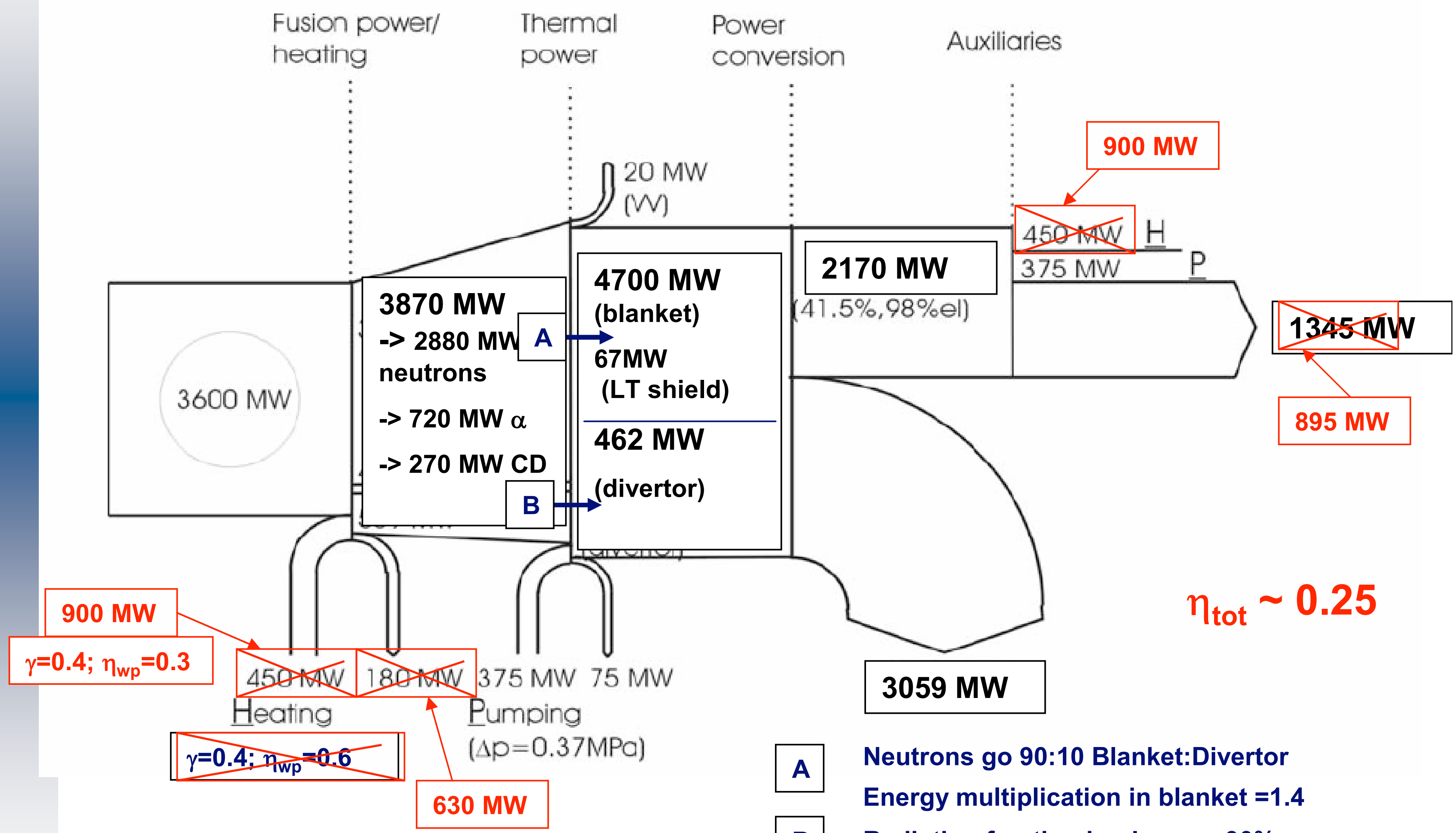
- DEMO assumptions:

  $\eta_{WP} \bullet \gamma_{CD} = 0.24 - 0.27$
- Negative NBI $\eta_{WP} \bullet \gamma_{CD} \sim 0.12 - 0.14$
- ECCD $\eta_{WP} \bullet \gamma_{CD} \sim 0.08$
- ICRF $\eta_{WP} \bullet \gamma_{CD} \sim [0.18 - 0.24] \bullet f_{coupled}$

  (where $f_{coupled}$ = fraction of generator power coupled at edge of plasma ~ 0.4 max H-mode – note no experiment has ever coupled >12MW ICRF power into an H-mode) ~0.07 – 0.095 for H-mode
- Lower Hybrid CD $\eta_{WP} \bullet \gamma_{CD} \sim [0.15 - 0.18] \bullet f_{coupled}$

  (LH klystrons are ~ 50% efficient – again $f_{coupled}$ is fraction of generator power coupled by grill to plasma – note, no experiment has ever coupled more than 4MW LH power into an H-mode)
- With these levels the installed CD powers on PPCS power plants go up considerably

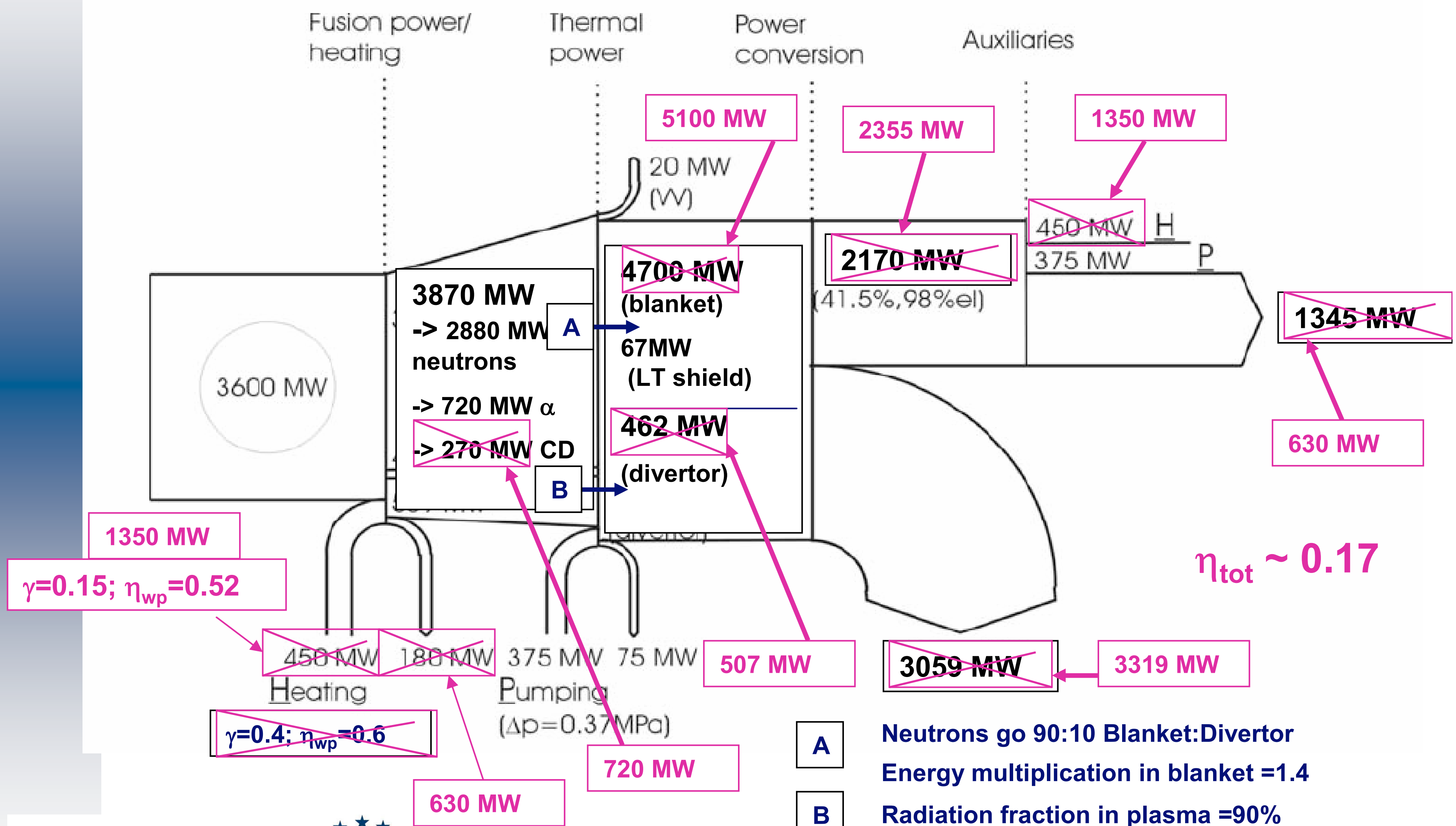
'Realistic H&CD' – Effect on power balance in PPCS Model B: (I) NBI



A — Neutrons go 90:10 Blanket:Divertor
Energy multiplication in blanket =1.4

B — Radiation fraction in plasma =90%
Pumping power goes 25:75 Divertor:Blanket

'Realistic H&CD' – Effect on power balance in PPCS Model B: (II) ECCD



A Neutrons go 90:10 Blanket:Divertor

Energy multiplication in blanket =1.4

B Radiation fraction in plasma =90%

Pumping power goes 25:75 Divertor:Blanket

Priorities H&CD development for DEMO/Reactors

- Wave systems:
  - ECCD about at the technology limit – need new physics!!
  - ICRF – coupling needs to go up significantly (tetrode sources already at technology limit) – experiments on tokamaks!!
  - LHCD – cannot penetrate high density plasmas – klystron sources near technology limit – probably not for development?
- Beam systems
  - Energies above 1 MeV – diminishing returns.
  - **Higher efficiency transmission or neutralisation**
  - **Higher brightness sources (smaller lower cost)**

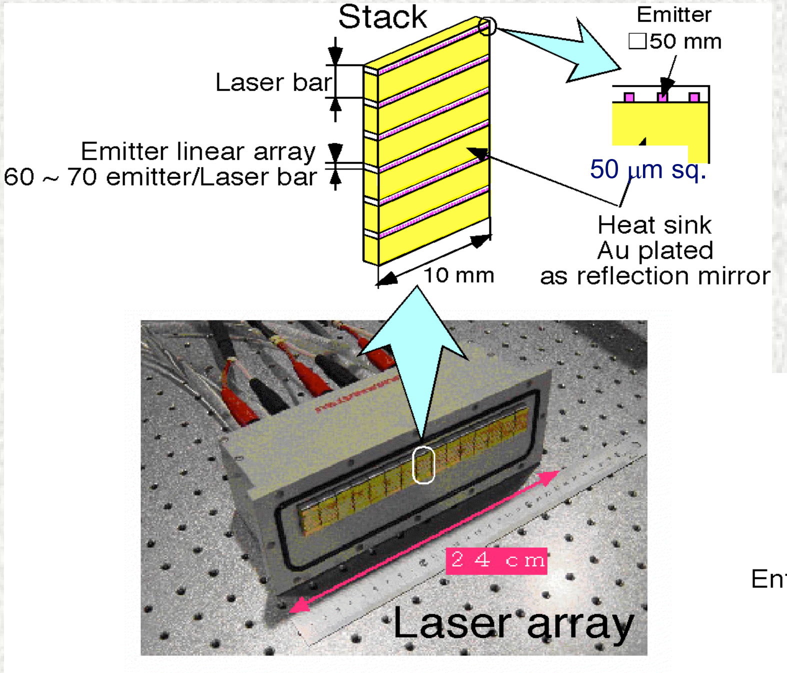
Priorities H&CD development for DEMO/Reactors – Negative NBI

- Improving neutralisation
  - Optical neutraliser;
  - Plasma neutraliser;
  - Li-vapour neutraliser???
    → also gives reduction of load on Ion Dumps – improvement of fatigue margins
- Improvement of power transmission
  - Reduction of stripping losses;
  - Reduction of co-electron extraction.
- Higher brightness sources (higher j^-)
  - Smaller size units;
  - Possibility of modularisation of power input.
- Low gas operation
  - Optical, Li-vapour or Plasma neutralisers;
  - Low-pressure source operation;
    →Removal of cryopumps.

Photon (laser) neutralizer

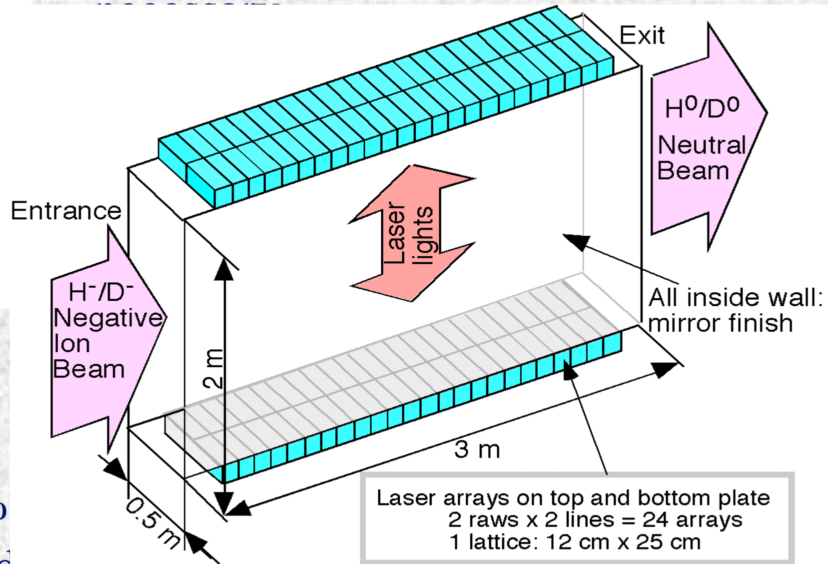
-would offer best combination of low gas flow and high power transmission

Challenging in an environment with high 14 MeV neutron fluxes



Application to neutralizer of ITER NBI size

- 96 arrays on top and bottom,
- Power required to drive laser: 620 kW
- Reflection more than 2000 times



2.7 kW cw Semiconductor laser array

- Light emission efficiency: 40%
- Au plated, reflection rate: 99.95 %
- HAMAMATSU Photonics Co. Ltd.

Slide courtesy of R S Hemsworth -ITER

Remote Handling drives Availability

Availability depends on the components and systems reliability and on the time required to replace them:

$$A = \frac{MTBF}{MTBF + MTTR}$$

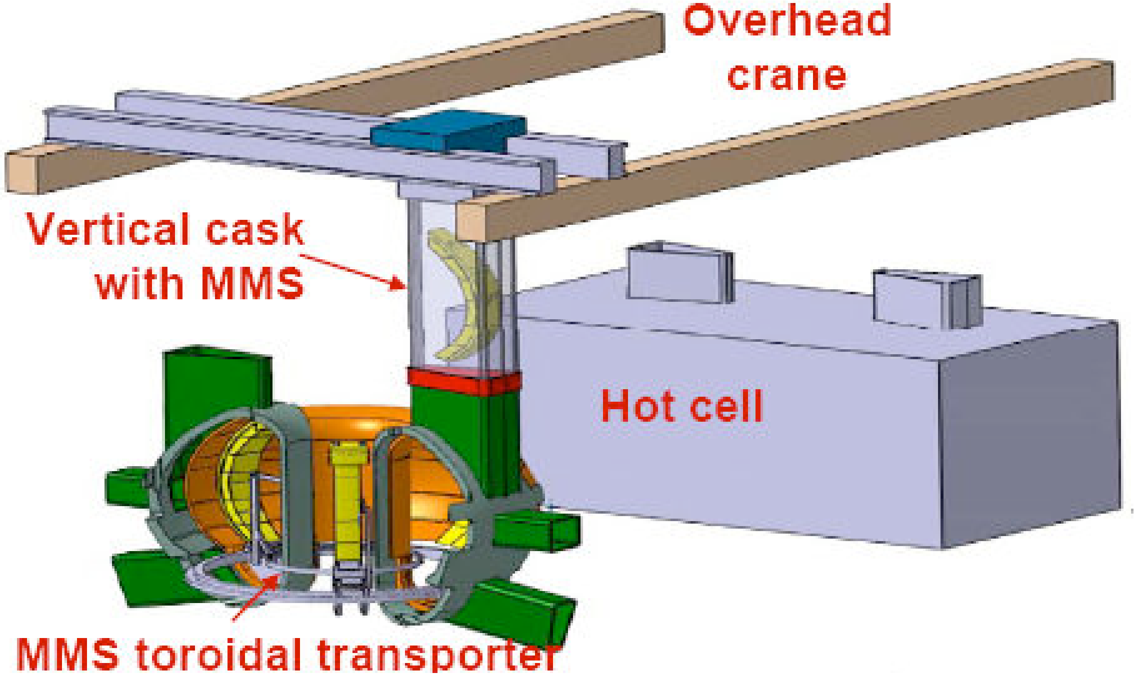
- MTBF: Mean Time Between Failure (or Mean Time Between Replacement)
- MTTR: Mean Time To Repair (or Mean Time To Replace)

Estimate of reactor scheduled availability:

- Scheduled outages: replacement of in-vessel components (divertor lifetime 2-2.5FPY, blanket lifetime 4-5FPY), assume all other scheduled operations to be carried out in parallel.
- Subtract an arbitrary figure for unscheduled outages.

Optimising MTTR is key for DEMO Remote Handling – requires large development programme – dedicated test stands

Remote Handling for DEMO far exceeds ITER requirements



- Much heavier components (blanket segments ~ 70-90 tonnes).
- High radiation environment in the machine (much higher than ITER – radiation-hard detection systems)
- Much stricter contamination control
- Higher reliability/availability – lower turn-round time

Optimising a DEMO programme

UKAEA Fusion
Working with Europe

Fusion Development Issues

- requirements of DEMO Phase 1

	Issue	Approved devices	ITER	IFMIF	DEMO Phase 1	DEMO Phase 2	Power Plant
Enabling technologies	Superconducting machine	2	3		R	R	R
	Heating, current drive and fuelling	1	2		3	R	R
	Power plant diagnostics & control	1	2		r	R	R
	Tritium inventory control & processing	1	3		R	R	R
	Remote handling	1	2		R	R	R
Materials, Component performance & lifetime	Materials characterisation			3	R	R	R
	Plasma-facing surface	1	2		3	4	R
	FW/blanket/divertor materials		1	1	3	4	R
	FW/blanket/divertor components		1	1	2	4	R
	T self sufficiency		1		3	R	R
Final Goal	Licensing for power plant	1	2	1	3	4	R
	Electricity generation at high availability				1	3	R

Output:		
	1	Will help to resolve the issue
	2	May resolve the issue
	3	Should resolve the issue
	4	Must resolve the issue

Input:		
	r	Solution is desirable
	R	Solution is a requirement

UKAEA September 2007 (revised/improved version of original table in UKAEA FUS 521, 2005).

Overload of DEMO Phase 1 issues to resolve in addition to crucial validation of Breeding Blankets and fuel cycle

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Strengthening the DEMO programme and reducing risk

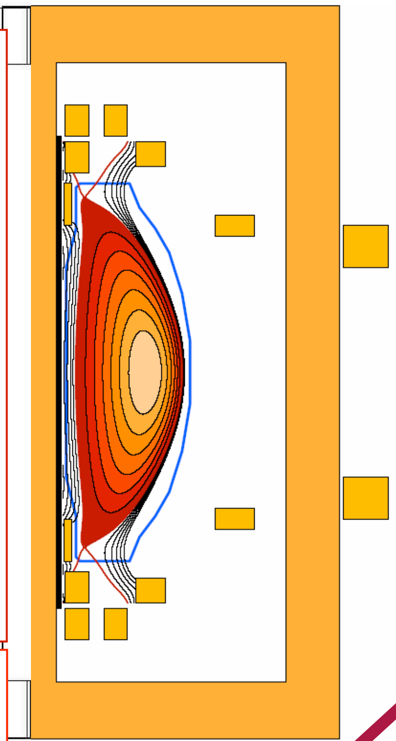
- In addition to the 'conventional Fast Track', desirable 'DEMO Priorities' should be addressed to strengthen the programme and reduce risk.
- **Materials**
  - development of low-activation ODS steels ductile at room temperature;
  - development of operational cycling scenarios to anneal radiation damage in Eurofer/RAFM;
  - manufacture of 'pure' RAFM steel varieties;
  - manufacturability of tungsten/tungsten alloy divertor structures.
- **Divertor**
  - **Proving a tungsten divertor at high power (ITER Phase 2);**
  - **Development and test of a helium-cooled full tungsten divertor at high power;**
  - **Investigation of alternative divertor concepts (use of plasma configurations to spread power loads) – eg. 'Super-X' divertor.**
- **Blankets technology**
  - Widening the EU blanket programme to include mock-ups and engineering prototypes of water-cooled blanket concepts.

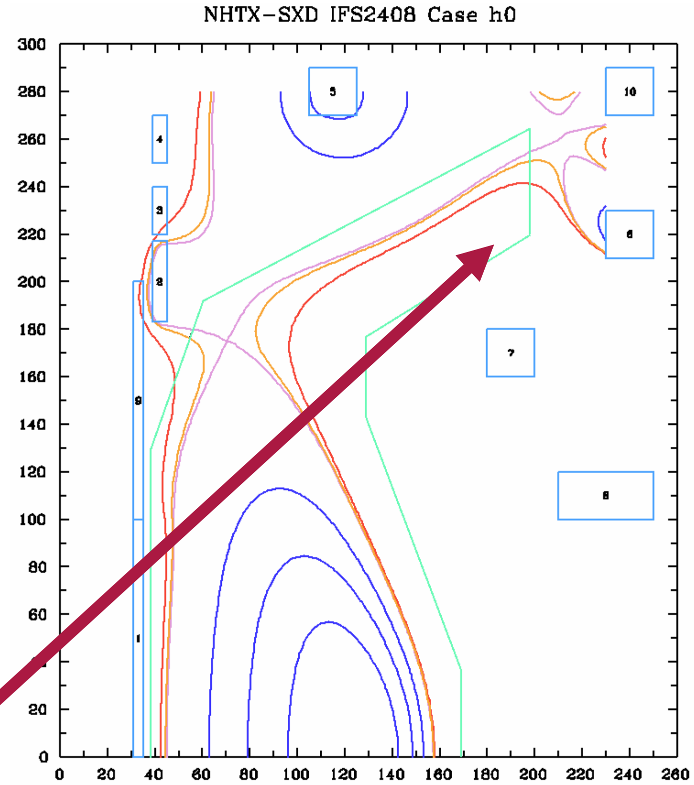
Could we use a novel plasma configuration instead of technology?

'Super-X' Divertor

- $R = 1.0$m
- $A = 1.8$
- $I_P = 3.5$ MA
- $B_T = 2.0$ T
- $P_{aux} = 50$ MW
- $P/R = 50$ MW/m
- $\beta_N = 4.5$
- $P_{aux}/P_{LH} \sim 8$

NHTX





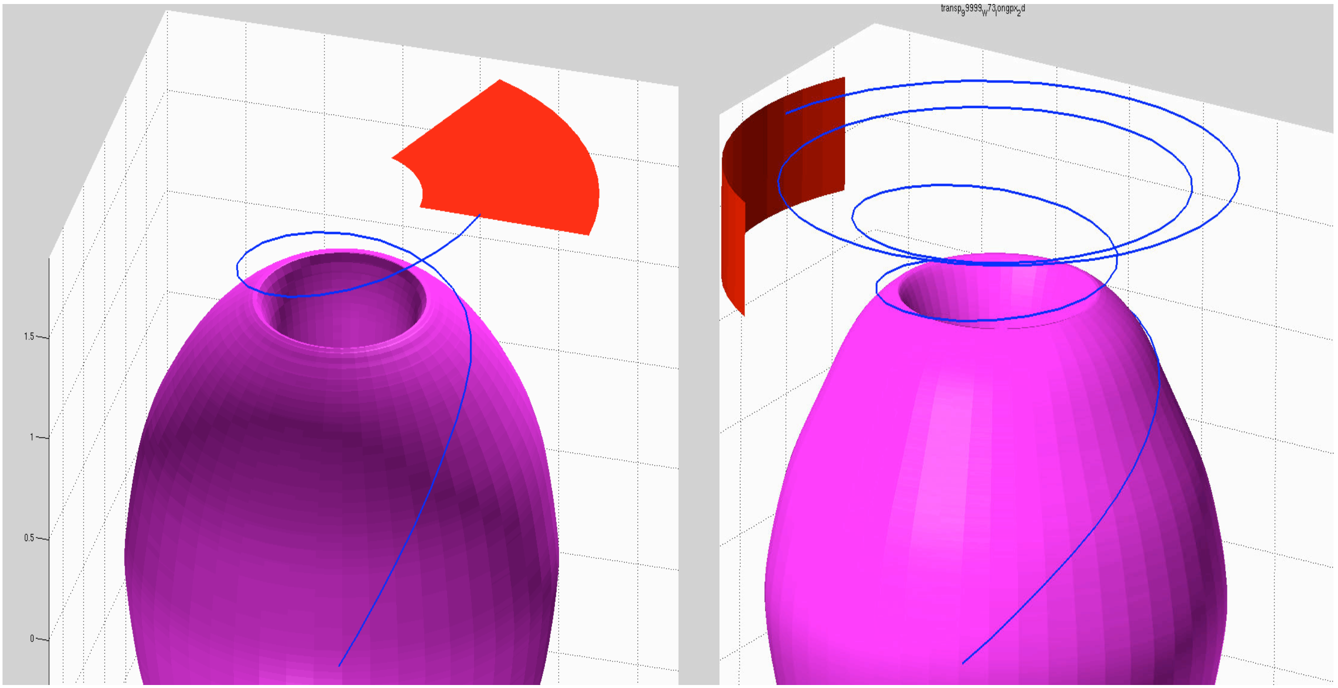
Ref [23]; R J Goldston et al

Ref [24]; P Valanju et al

As flux lines go to larger major radius the transverse power 'scrape-off layer' expands → lower power density at target. Target area could even be better shielded against neutrons??

Super-X configuration coupled with low poloidal field.

- A Super-X coupled with a low poloidal field gives a very long divertor 'leg' before the target is reached – room to puff gas and reduce particle energies and total power reaching divertor.
- Uniquely amongst Tokamaks, MAST (Culham) could accommodate the Super-X configuration and coils.



Strengthening the programme towards DEMO and reducing risk(II)

- **Enabling Technology programme and facilities:**
  - **Remote handling facility - between ITER and DEMO Phase 1 aimed at industrial reliability reducing/optimising maintenance times.**
  - Develop heavy-duty and Radiation Hard Remote Handling systems (DEMO activation levels will be ~ 2 orders of magnitude above ITER)
  - **Industrial-level reliability programmes for Heating and Current drive and Diagnostic systems.**
  - **Test facilities - between ITER and DEMO Phase 1 – to develop high wall-plug-efficiency for Neutral Beam** (and possibly Electron Cyclotron) **current drive systems**. (test on ITER Phase 2?).
  - Radiation-hardened diagnostic development.
- **An accompanying Tokamak programme:**
  - An enhanced satellite programme with a 'JET-class' device to complement JT-60SA concentrating on:
    - **plasma control with remote actutators and sparse diagnostics;**
    - **high heat flux divertors/ innovative divertor plasma configurations.**
    - **high density tokamak operation with high-radiation fraction plasmas**

Strategic developments to reduce risk to Fusion Power deployment

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Development of high-Temperature superconducting magnets

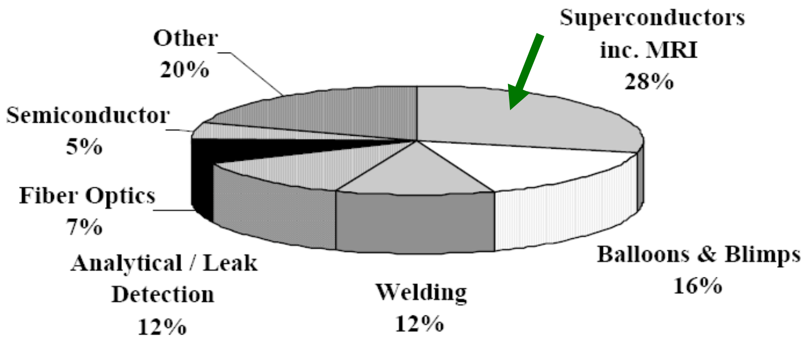
- High-temperature superconductors have already been shown in this course (lectures of M. Noe and S. Schlacter) to lead to:
  - power savings;
  - simplification of cryogenic plant;
  - simplification of shields etc.
- …but strategically, high-T superconductors are urgent for development in Fusion Technology because of the Helium resource problem.

World Helium reserves –
the problem (I)

- Terrestrial ^{4}He comes from radioactive decay of U+Th – Helium underground tends to collect in impermeable rock strata
- Thus - virtually all known sources of terrestrial helium are associated with natural gas (finite hydrocarbon resource!)
- The concentration of helium in Natural Gas is variable, and not *a priori* easy to predict.
- US Geological Survey (**USGS**) is the reference for all He reserve data.
  - 'Reserves' (Defined by US Bureau of Land Management) →supplies with>0.3% He - known to contain exploitable Helium concentrations.
  - 'Reserve Base' is a total of the predicted possible Reserves in known Natural Gas fields with > 0.5% possible He concentration.

World Helium reserves - the problem (II)

- Worldwide demand for helium in 2008 was --170 10^6 n.m^3
- Growing demand, ~ +5% pa
- Cryogenic ~25-30 % share mainly for superconductors



- If consumption increases by ~ +5%/yr
- Calculation of future availability suggests:
  - Reserves only
    - 30-35 years availability (with consumption growth)
  - 'Reserve base'
    - ~ 100 years availability (with consumption growth)
- If fusion continues to depend on helium for its future, these figures are not consistent with an 'unlimited energy source' !

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World Helium reserves

Atmospheric extraction?

- ^4He from α–decay of U+Th percolates through rock to atmosphere.
  - only the high energy tail of the Maxwellian distribution has enough velocity to escape the Earth's gravity.
- Atmosphere has ~ 3.7 10^9 Tonnes ^4He - residence time in atmosphere ~10^6y -- near surface [He] conc^n is ~ 5.22ppm
- Atmospheric extraction of noble gases has been done, usually with Neon (18ppm) – via a distillation/absorber unit attached to a large cryogenic Air Separation Plant (ASU)
- Costs are high Ne £2.50/litre(gas)(cf LHe cost ~ £3/litre liquid)
- Possible drivers which will advance the technology/ reduce costs
  - wide-scale use of ASU O_2 for Clean Coal, to enable CO_2 to be sequestered and H_2 to be recovered for transport
- Problems are the huge scale of task for helium
  - **eg. – if half the current global ASU plant capacity (~10^6 tonnes/day) were retrofitted with He recovery this would only satisfy ~ 1% of world He market!**

Helium use in Fusion

present and future (I)

- At present Helium use in fusion research is relatively limited.
  - Cryo-pumping of fusion species (H, D, T…).
  - Cryo-cooling required for diagnostics (low noise LIDAR etc) and gyrotrons.
- In future, use will escalate strongly:
  - Large volume cryo-pumps (with necessity to pump He 'ash')
  - Cooling required for giant superconducting 6T magnets - ITER has an inventory of ~ 24 Tonnes LHe, mainly for SC magnet and cryostat cooling.
  - Helium 'nearly perfect gas' for heat transfer and so is foreseen in some of the Blanket and Divertor designs.
- Long term, fusion materials have to come from sustainable sources… and this includes helium
- Fusion does produce helium but only in relatively small quantities →power plants ~170 kg/yr per GWe.

Helium use in Fusion

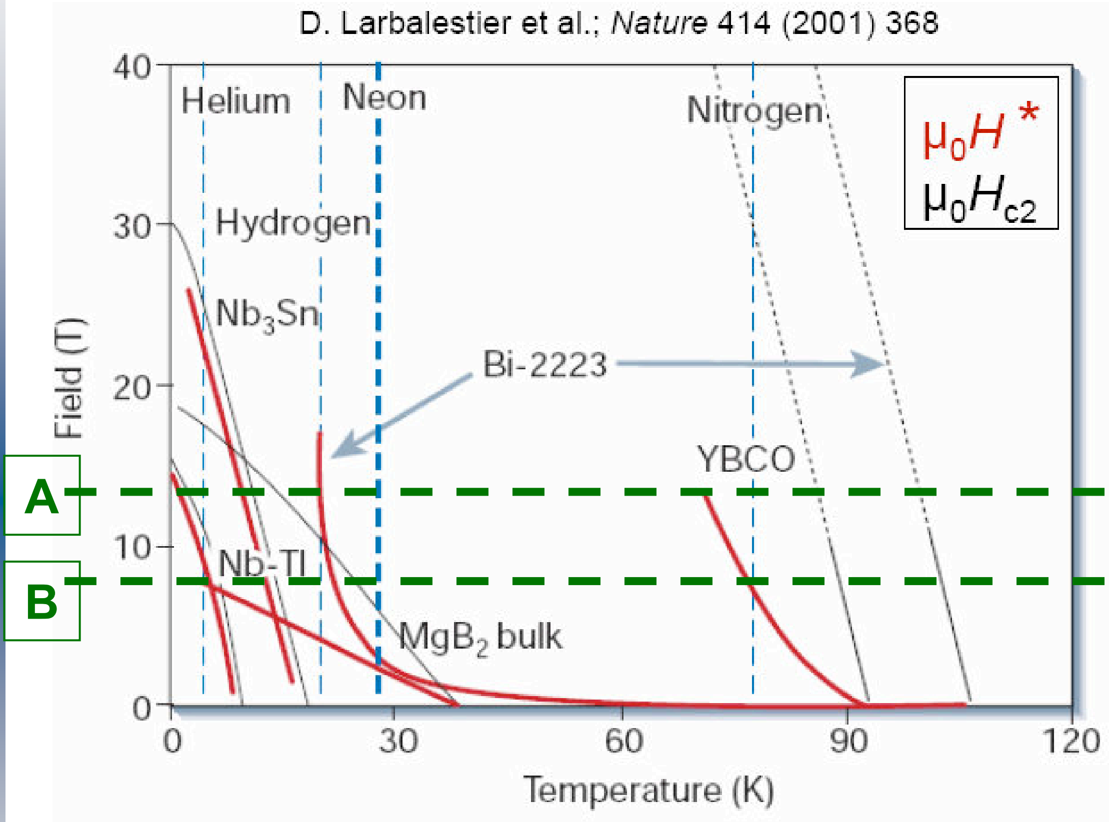
present and future (II) - cryogenics

- **Cryogenic loss -- some calculations:**
  JET cryoplant (a 'sealed' system) has inventory ~ 20000 litres LHe and loses ~ 800 l/wk (liquid loss ~ 2x inventory per year).
  Magnets 'quench', eg. CERN loses 1-2x inventory/yr
  Scale from JET; include Cryo, Magnets
  → ITER losses ~ 48 Tonnes/yr unless technology improved
  (~ 263 10^3 n.m3/yr ~ 0.15% of world consumption)
- For Power Plant take unit size as 1 GWe (2.5 GWTh):
  - **found scale factors go as ~ $(\text{power})^{0.75}$ (power goes as ~ volume, cooling goes ~ surface area → $(\text{volume})^{2/3}$**
  - **cryo (magnets + pumps) scaled from ITER (500 MWTh) ~ 80 T He inventory**
- By 2075, extrapolating US DOE – EIA figures for 2010-2030, the world electricity demand will be ~ 7.5 TWe.
- For 33% market share for Fusion (reasonable long-term goal) we would need ~ 2.5 TWe generating capacity. To charge this up with helium (Just for LHe-cooled magnets/cryostat, would take ~ the entire world 2003 Helium production, even 10% p.a. losses would then constitute ~ 2800 tonnes/yr at a time when He supplies from Natural Gas would be dwindling fast!

Helium is a vulnerability for fusion

- Tactically, strategically and economically it is unwise for Fusion to depend on a resource limited by other markets.
- Helium is an ‘associated product stream’ tied to the economics of the Natural Gas extraction – it is unlikely that the market will develop in a way favourable to fusion until it becomes a big player and then -- too late!
- Even advanced production such as air separation (which would free us of the physical limitations) is driven by development of other technologies/markets.
- To avoid this potential problem, it seems sensible to:
  - pursue a serious development of high-T superconductors which do not require helium cooling;
  - pursue serious development of non-helium-cooled Blanket/DIvertor designs.

Superconductors – helium free?



- H_{c2}: bulk superconductivity is destroyed
- H^*: bulk critical current density goes to zero
- $H^*(T)$ is
  - close to $H_{c2}(T)$ for Nb-Ti and Nb_3Sn,
  - about half of $H_{c2}(T)$ for MgB_2
  - much lower than $H^*(T)$ for YBCO and Bi-2223.

- **'A' is the approx field at the ITER TF conductor surface**
  'B' is the approx field at the ITER PF conductor surface

- It seems that only YBCO-type HTS can get SC performance above Liquid Neon temperatures – developments are clearly needed.

Strategic risk reduction:

A Component Testing programme?

- DEMO is envisaged to test components (eg. the blanket and divertor are key elements), but:
  - DEMO as a reactor demonstration, has to be a large, (P_{fus} >2.5GW) device - it requires to breed tritium, relying for high availability operation on some of the components it is supposed to test;
  - DEMO is a large and complex superconducting machine. The Mean-Time-To-Replace (MTTR) a test component will thus be large – leading to possible significant delays in a test programme.
  - DEMO has other missions requiring high-availability such as demonstrating electricity generation;
- As a strategic risk reduction exercise, the goals of a Components testing programme and the feasibility of a Components Test Facility (CTF) should be examined.

CTF

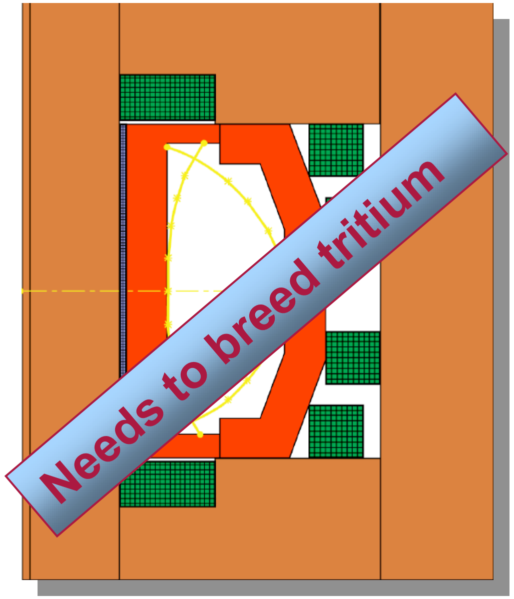
- 'CTF' is generally taken to indicate a relatively small size, low total fusion power device in which fusion technology component testing can take place in a tokamak environment:
  - at the smallest relevant scale;
  - with a true fusion neutron spectrum;
  - with a practical strategy for solving the tritium supply and consumption issues.

CTF

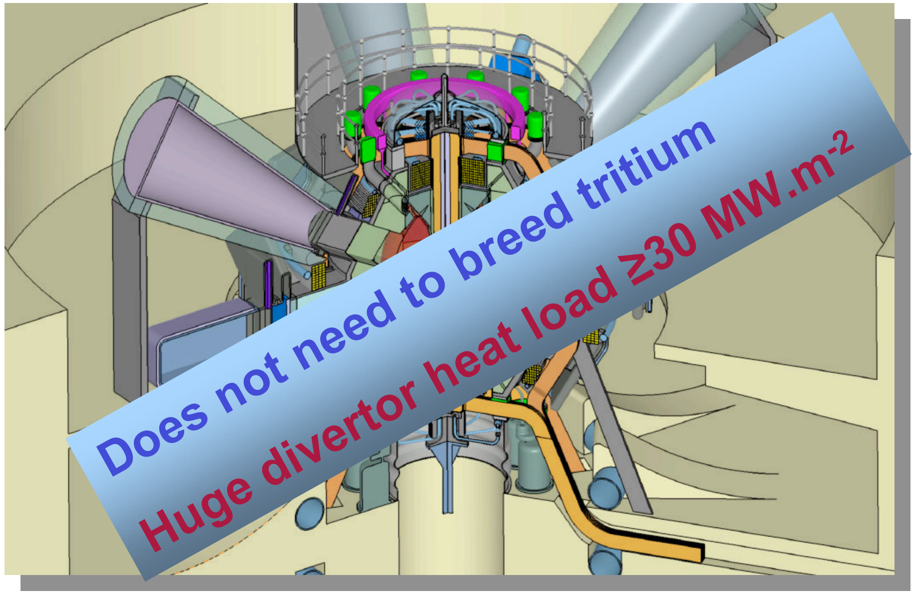
- A CTF must be able to:
  - produce long periods of low Q driven plasma burn to achieve the required integrated neutron yield;
  - accommodate fully functional test components on the scale of ~ 1 m^2 (relevant scale for component issues);
  - have a significant area, over 10 m^2, available to test several scaled components in parallel(e.g. blanket modules).
- Candidate tokamak designs exist for a CTF - all designs have specific major potential problems:
  - Conventional (A~3) designs require to breed tritium to carry out their programme;
  - Spherical Tokamak (A~1.5) designs need divertors capable of taking huge power loads (P >30 $MW.m^2$)

Two proposed Tokamak CTF designs

FDF (GA, USA)



ST-CTF (Culham, EU)



Steady State advanced tokamak
Fusion power ~ 250 MW
Neutron wall – load ~ 1.5 MW.m^{-2}
Tritium consumption ~ 13 kg/fpy

Compact Spherical Tokamak
Fusion power ~ 36 MW
Neutron wall- load ~ 1.0 MW.m^{-2}
Tritium consumption ~ 1.8 kg/fpy

If tritium consumption were below ~1-2 kg/yr sufficient Tritium would be available from Candu programme for both ITER and a CTF, within a time window– if not, breeding would be required

Role of CTF in a DEMO/FPP programme

- CTFs could fit into a Fusion Development schedule in two ways:
  - (I) As a risk reduction exercise aimed at 'weeding out' the initial poor reliability blanket designs and allowing DEMO to enter a more reliable phase more quickly. (equivalent to a 'burn-in' bench test for components) - generally achieved by testing at 1-2 MW/m^2 for 20% of lifetime.
  - (II) As part of a component engineering programme aimed at reliability growth and development of reliable concepts for an FPP - a long-term programme - but how long?
- Essentially a choice between
  (I) early CTF deployment and
  (II) longer term integration

Accelerating the DEMO programme

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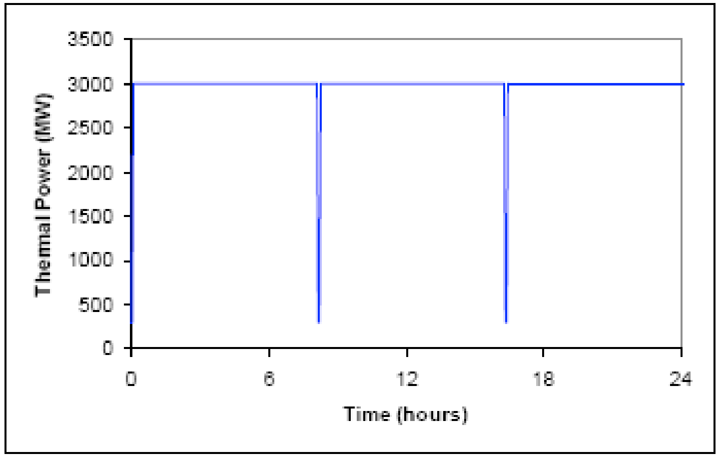
Accelerating a DEMO programme-

Pulsed operation?

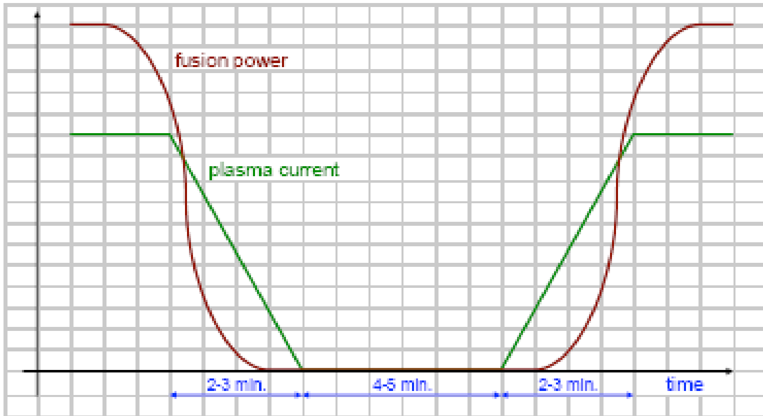
- **New facilities and/or machines** (more than one satellite tokamak, CTF) would not only reduce risk, but also accelerate the completion date of DEMO Phase 2.
- **An Alternative strategy for first DEMO could also be employed:**
  - **Pulsed operation** - would relieve urgency of milestones on heating & current drive systems and steady-state scenarios.
    - Pulse length ~ 8-10 hours (for 30000 pulse lifetime)
    - Proven at JET scale (1991)
    - Estimates show CoE ~ 20% higher (bigger Central Solenoid, hence bigger Nuclear Island)
  - **Need to evaluate in DEMO study – connection to the Grid must be constant – energy storage system development?**

Pulsed Operation



Evolution of the thermal power generated by a FPP operating in quasi-continuous mode with pulses 8 hours long.

Evolution of the fusion power and of the plasma current during the dwell time in case of quasi-continuous plasma operation.



Slide courtesy D Maissonier – EU Commission

Conclusions (I)

- The Gap Analysis using the 'Fusion Issues table' is a useful tool to analyse the priorities for DEMO development:
  - Confirming the prioritisation of the issues which we need to ensure ITER is equipped to resolve;
  - giving indications of the content of phases of the ITER exploitation;
  - showing desirable characteristics of a satellite programme to ITER;
  - identifying gaps in the technology programme supporting DEMO
- The crucial role of ITER in physics and some aspects of technology is evident from this analysis.
- The crucial role of IFMIF in characterising materials for DEMO (hence necessity of an early decision to construct) is emphasised.
- The analysis shows that auxiliary facilities for eg, Heating and Current Drive system development and Remote Handling development will help optimise the programme and reduce risk.

Conclusions (II)

- The analysis points out the risks in prematurely closing off certain Blanket designs for DEMO may not be wise – broaden the Blanket programme?
- The analysis shows a significant number of unresolved issues at DEMO Ph 1 start, resolving these on DEMO itself risks delays to the Fast Track strategy.
- The Analysis identifies strategies and possible ‘accompanying programme’ machines/facilities to ITER and DEMO Ph 1 to reduce risk and strengthen the programme.
- Certtain strategic decisions – constructing a Component Test Facility, developing Helium-free Superconductor magnets, and considering replacement of helium as a DIvertor/Blanket coolant are recommended.

Appendices

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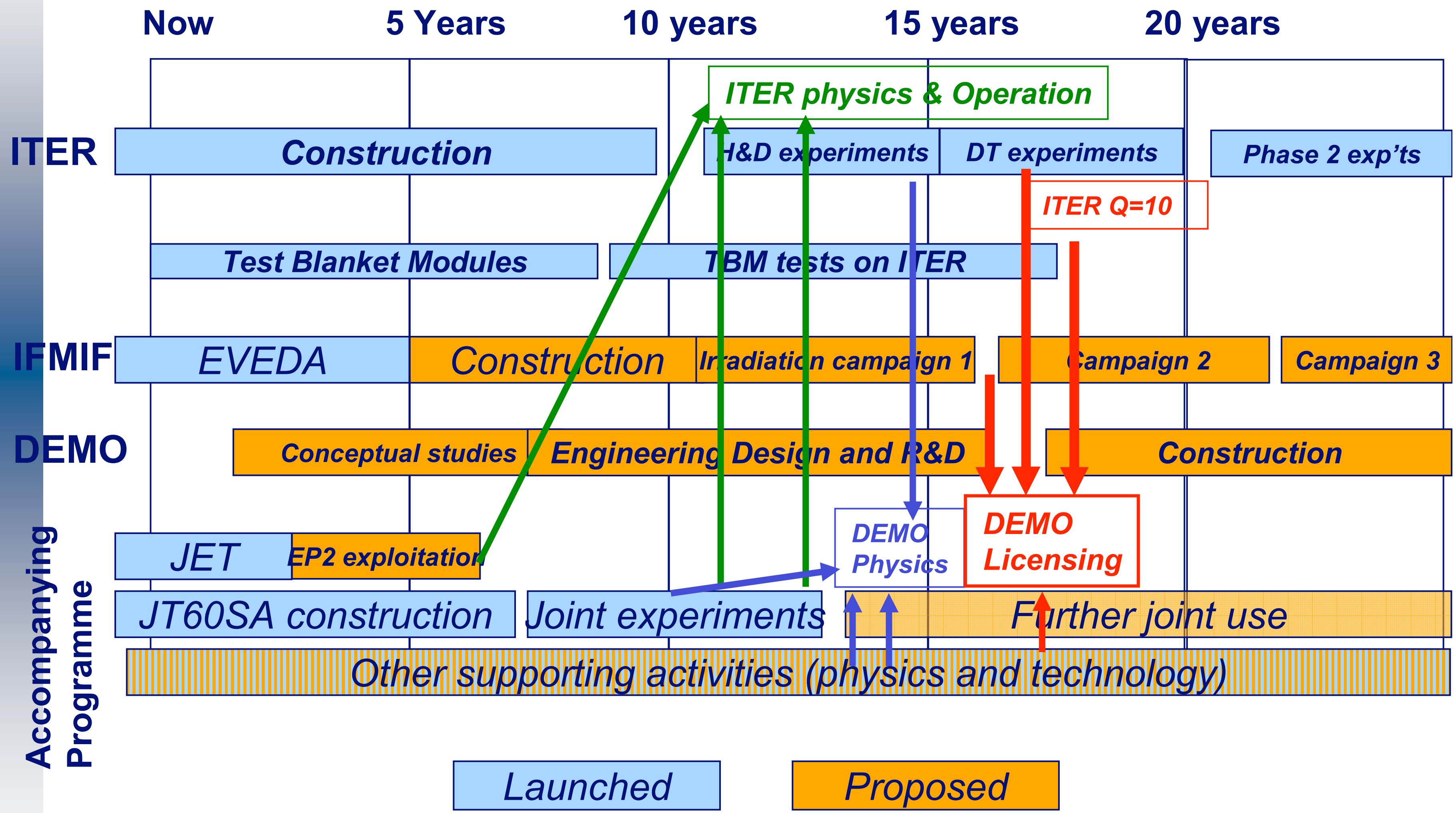
Fusion Roadmap timeline based on Fast Track Strategy [note pre-ITER delays!]



DEMO Materials: Realising the SiC/SiC goal

SiC subjected to FW neutron spectrum
P_{fus} = 3 GW
NWL = 2.3 MW/m^2
For 5 years

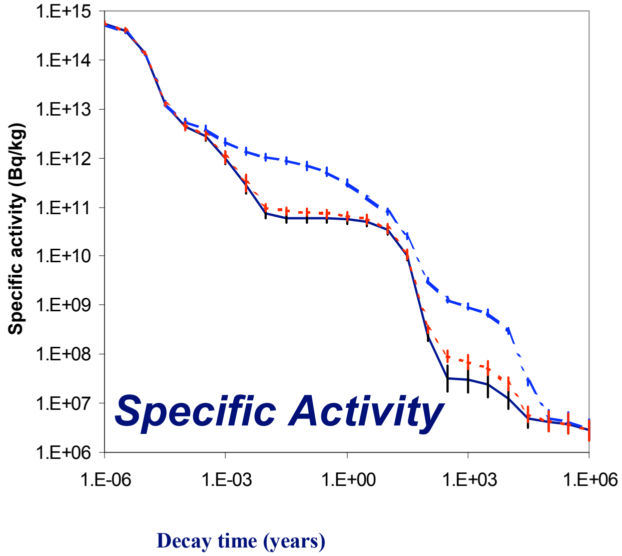
Chemical composition (wt%):

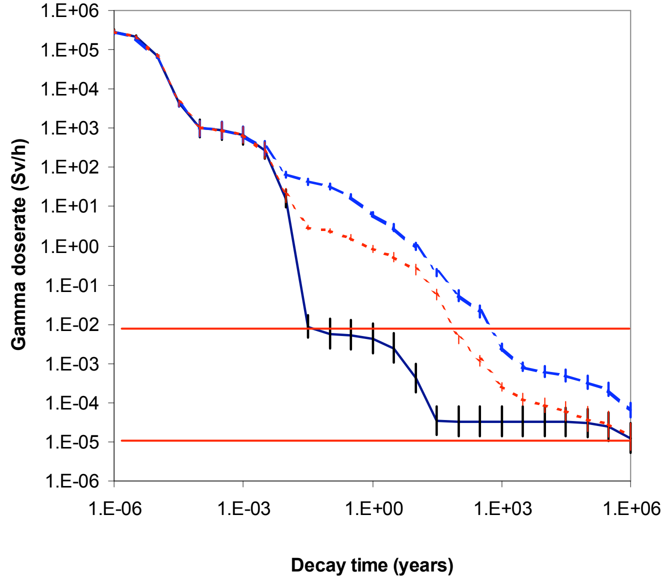
1. **Pure**
2. **Real (present day)**
3. **Achievable**

Element	Case 1 (specification without impurities)	Case 2 (real material)	Case 3 (achievable material)
Si	**bal**	**bal**	**bal**
C	**30 wt% (SiC/SiC) 46 wt% (C/SiC)**	**30 wt% (SiC/SiC) 46 wt% (C/SiC)**	**30 wt% (SiC/SiC) 46 wt% (C/SiC)**
Al		25	1
Ba		2.5	1
Bi		50	1
Cd		20	1
Ce		20	1
Co		10	1
Cr		8	1
Cs		10	1
Eu		2	1
Fe		130	10
Gd		5	1
Hf		10	1
Ho		5	1
Ir		50	1
K		8	1
Mo		50	1
N		1140	50
Nb		5	1
Nd		10	1
Ni		18	1
O		40000	1000
Os		30	1
Sn		1300	1
Tb		20	1
Tl		8	1
Y		1	1

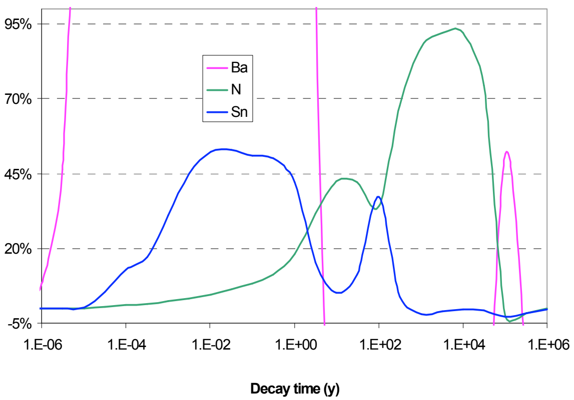
DEMO Materials: Realising the SiC/SiC goal

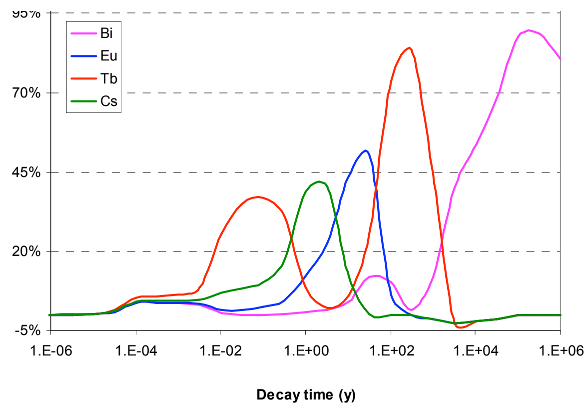
Specific activity and contact gamma dose rate in
SiC/SiC Pure (———), Real (today) (- - - - -), Achievable (- - - - -)





P Batistoni et al

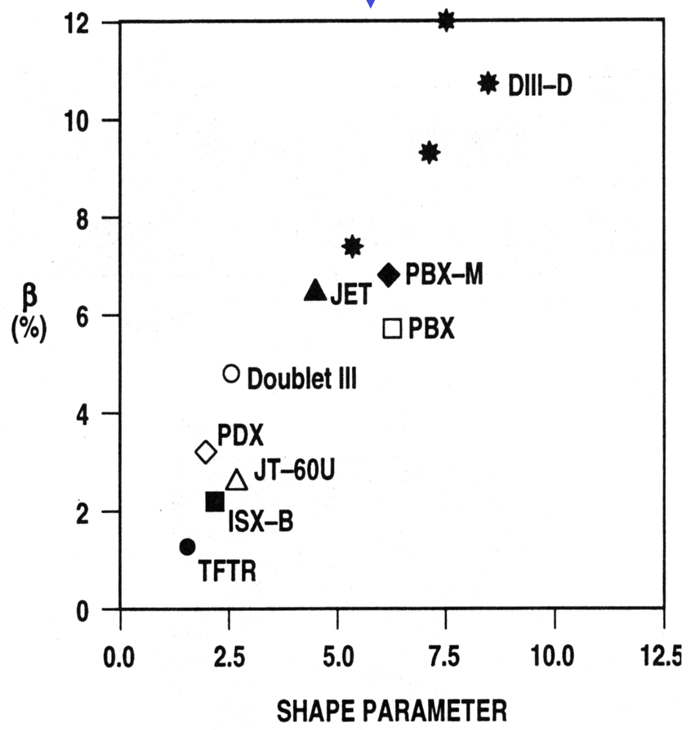




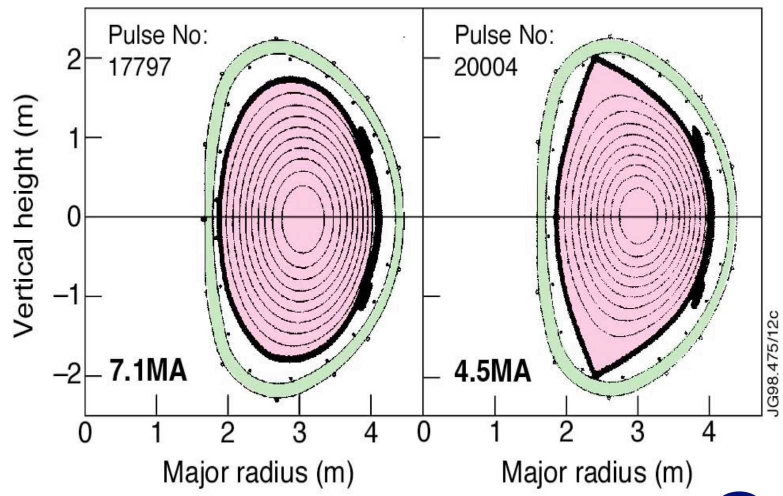
β-limits:

shaping the plasma

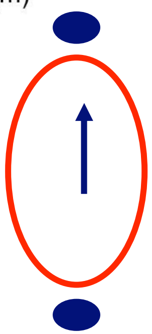
- Pressure-driven instabilities ('Ballooning') can be stabilised by shaping the plasma – uses extra coils - results in higher β.



- … but some of the plasma current capability is lost in shaping (eg. JET)



- …also elongated plasmas are more vertically-unstable
- …and shaped plasmas lead to higher disruption forces

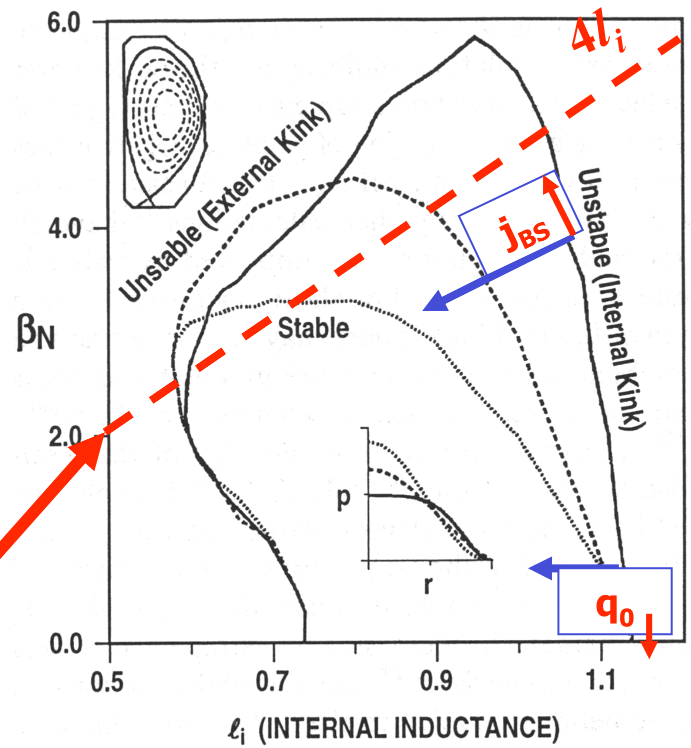


β- limits:

inductance-dependent kink-limit at high-β.

- Even with Ballooning instabilities it is possible to reach β_N~3.5.
- However we have to optimise plasma inductance (l_i) to give highest β
- The optimised (high) (l_i) do not correspond an optimised reactor current profile because:
  - avoidance of sawteeth (q_0 > 1 required)(lower- l_i);
  - Bootstrap current located at high grad p and hence off-axis (low- l_i).
- Numerically an ~ linear β_N- (l_i) dependence is obtained for broad P(r) quoted as '4l_i ' but actually varies with profiles).

Ref [25] Howl et al.,

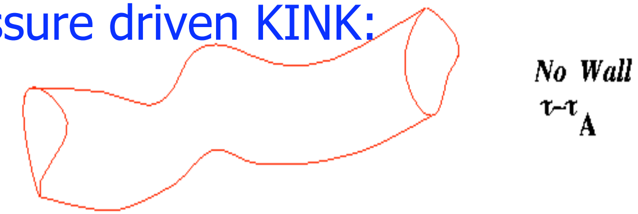


Indicates of the conflict between perfomance and S/S optimisation

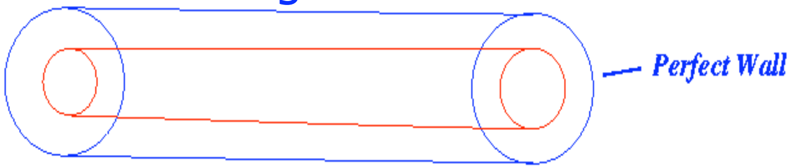
β -limits:

Kink stability – role of the wall and plasma rotation

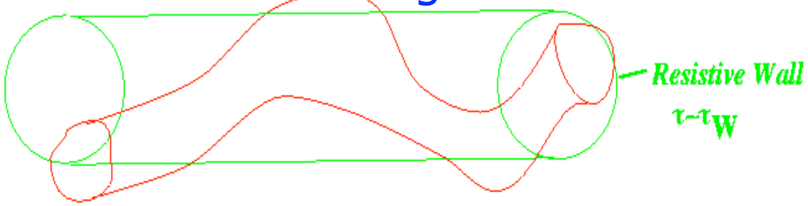
Pressure driven KINK:

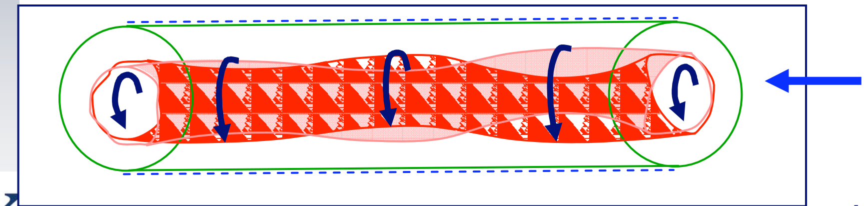


A close enough wall can stabilise it:



Real wall slows it's growth:

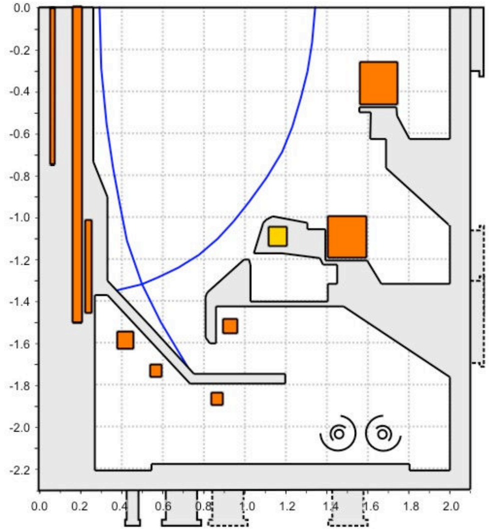


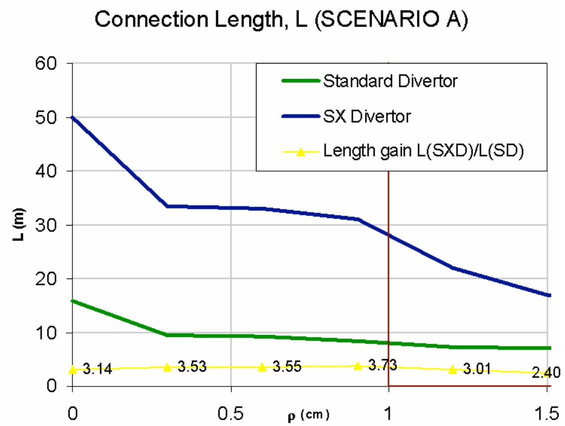


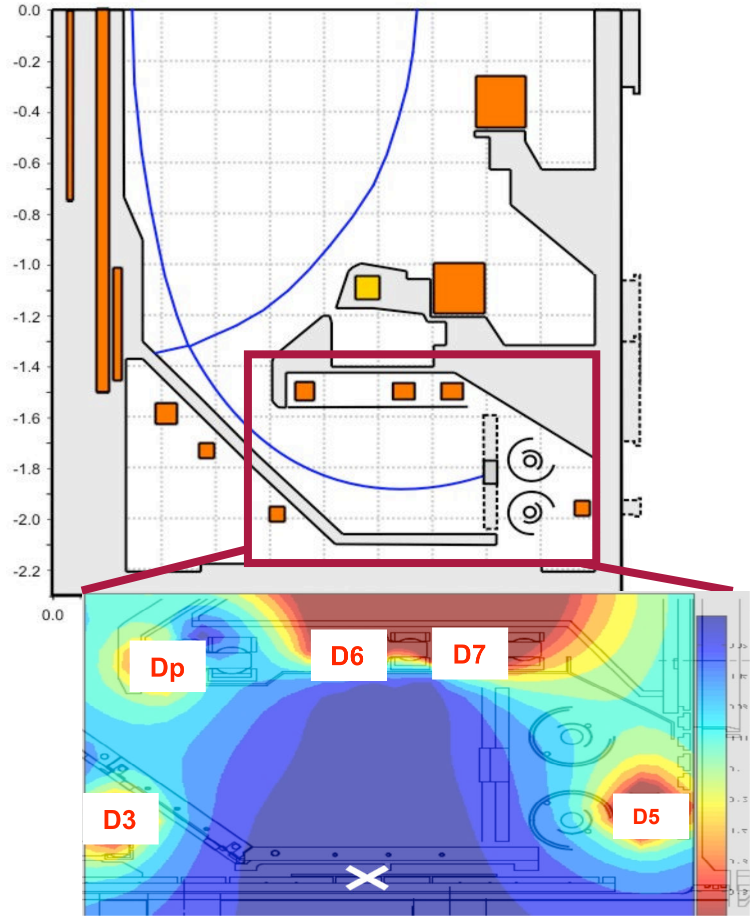
- The tokamak's conducting wall affects Kink stability:
  - Kinks occur and send flux through wall.
  - Timescale for penetration $\tau_W \sim \mu_0 \sigma r_W \delta_W/2$
  - Perfect conductor (infinite σ) → infinite stability
- Rotation: if the plasma rotates (eg.driven by high velocity beam injection)fraction of flux which penetrates wall $f \sim 1/\omega\tau_W$ where ω is rotation frequency f→ to zero at high ω.
  Rotation prevents wall penetration - mode sees perfect wall.

MAST-Upgrade proposal:

Expanded (super-X like) divertor - additional divertor coils create near poloidal null in sub-divertor region





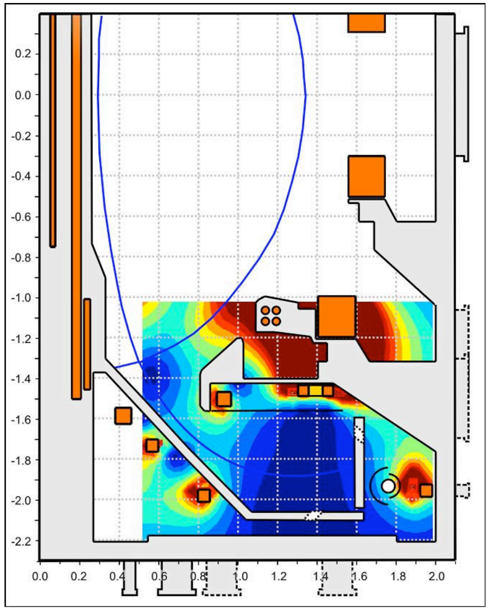


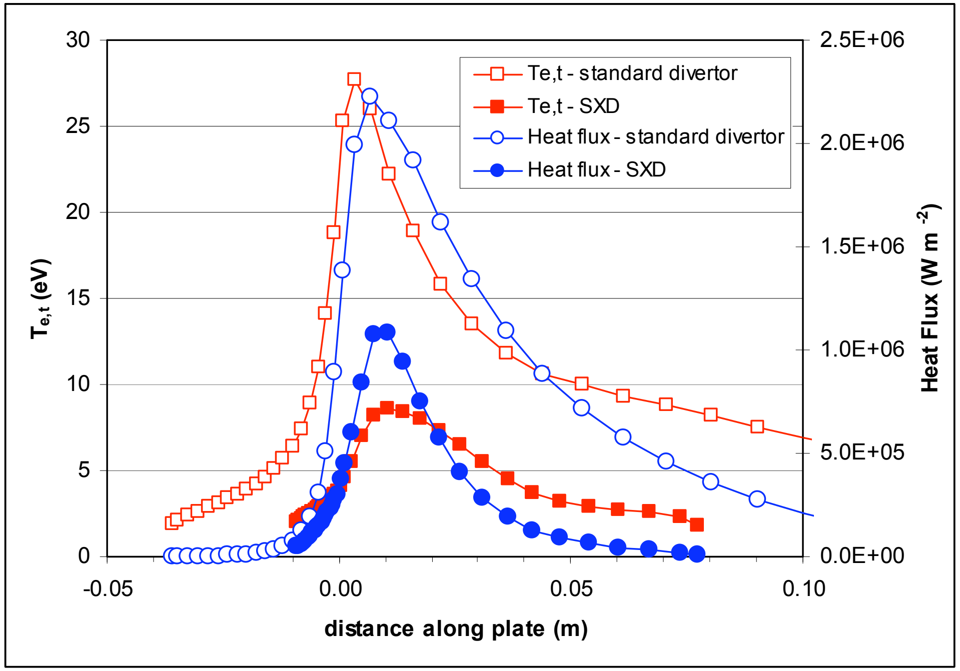
Ref [26]: S Lisgo et al

Length gain for midplane SOL lines is > factor 3 for $0<\rho<1$cm.

Preliminary modelling shows plasma stream temperatures and heat flux are reduced at the Super-X target

Source – courtesy J Canik - ORNL

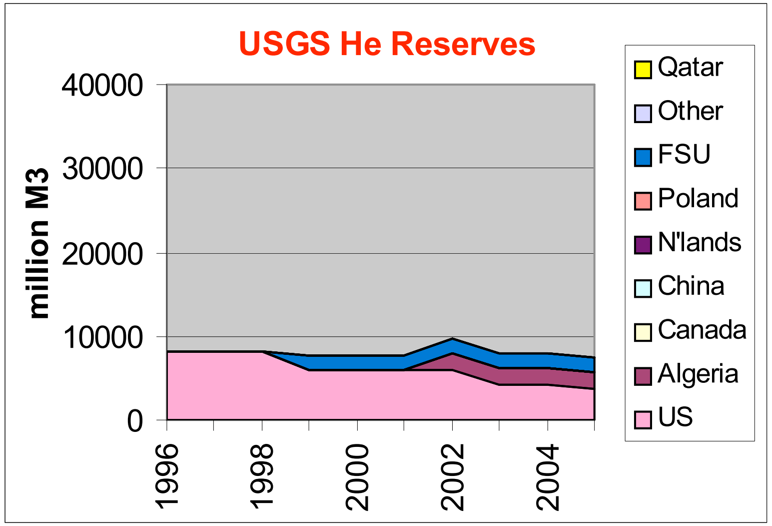


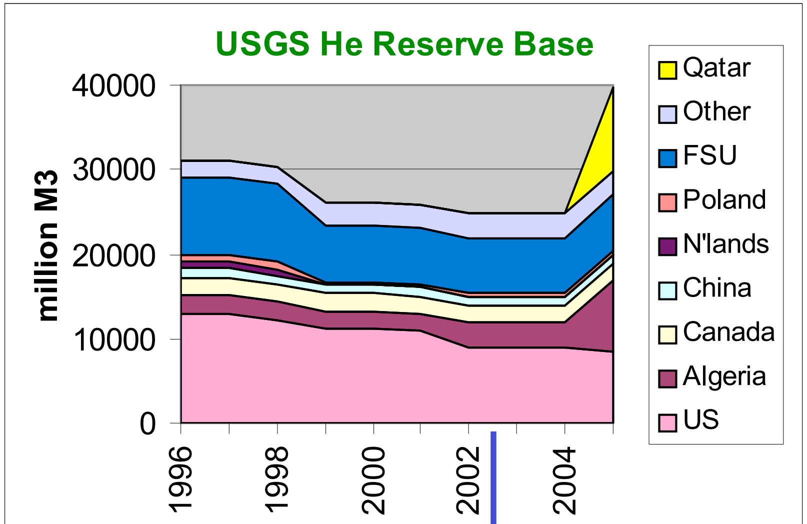


$$D_{perp} = 0.3m^2s^{-1}; \chi_{ei} = 1.0m^2s^{-1}; \Gamma_{core} = 3.3\cdot10^{20}D^+s^{-1}; R = 0.997$$

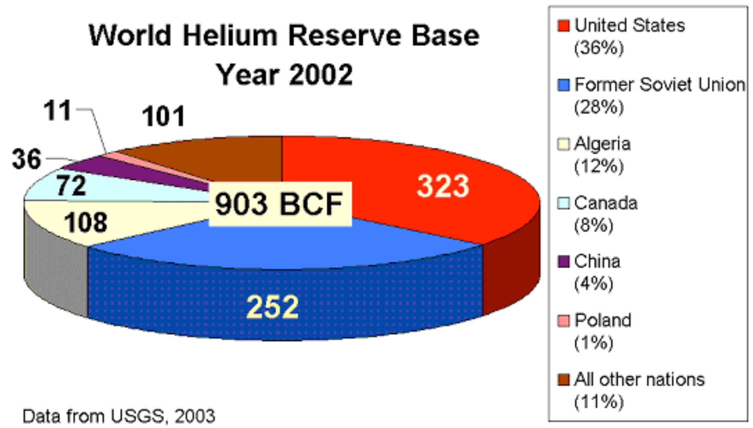
- SOLPS 2D modelling (ORNL).
- **Heat flux reduced by >2; plasma temp reduced by factor >3**
- Parameters set from comparisons between model and MAST ELM-free H-mode experimental data. Relate to relatively low-power H-mode (P_{NBI}=1.8MW)

World Helium reserves
USGS data





- US Geological Survey (USGS) data (30000 samples)
- 'BCF' = Billion cubic feet (~ 28.3 10^6 cu. Metres)
- Reserve Base ~ Reserves x3
- Note how massive amount from Qatar added in 2003-4!



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In-vessel components in a Fusion Environment

- The in-vessel Fusion Environment features multi-variate ‘fields’ and gradients:
  - Neutron effects (bulk heating; tritium production; helium production; activation)
  - Other heat sources (plasma surface bombardment, neutral particles)
  - Particle flux (energy & density gradients)
  - Magnetic field stresses/ eddy currents
  - Thermomechanical forces
  - Synergistic effects (difficult to predict from the simulations of the separate effects)
- Determining the effects on complex components and developing technologies to minimise adverse effects is a key to the development of safe and reliable fusion reactors.

Availability considerations

- Availability normally quoted as - for a group of n non-maintainable systems:

  $A=1/(1 + \Sigma_n(MTTR_i/MTTF_i))$

  $MTTR_i$= Mean Time to Repair/replace system i

  $MTBF_i$ = Mean Time To Failure system i

- Thus if A=0.33 (as DEMO needs in Phase 1) , $\Sigma(MTTR/MTBF)=2$.

- In general, for complex systems, minor failures are minimised by a regular maintenance cycle, so that MTBF refers only to major failures.

  $A=(1-f_p)/(1-f_p +\Sigma_n(MTTR_i^{maj}/MTBF_i^{maj}))$

  if one month of the year is devoted to maintenance, $\Sigma(MTTR^{maj}/MTBF^{maj})=1.8$

Availability considerations (II)

- For 13 systems (TF coils, PF coils, Blanket, Divertor, H&CD, Vessel, Coil PS, Cryogenics, H&CD PS, Fuelling,Tritium plant, Vacuum system, Conventional BoP)

  --> average <MTTR/MTBF> per system~ 0.14 with planned maintenance

  --> a complete Divertor replacement in <3.5 months;

  --> a Blanket module replacement in < 8.5 months.

- This shows the absolute necessity of getting:
  - very reliable ex-vessel systems, as every gain here eases the Divertor and Blanket replacement problems;
  - very efficient and reliable Remote Handling systems

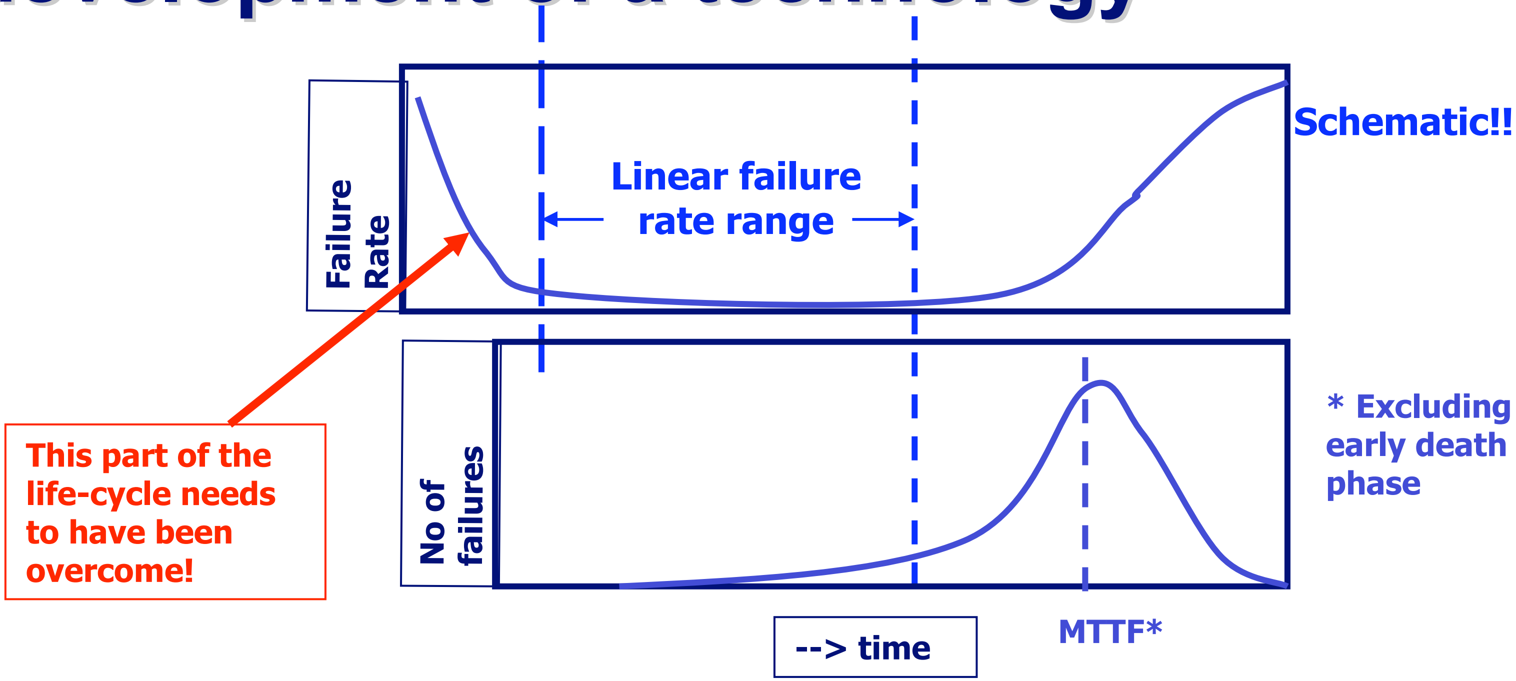
- The above figures are very challenging for Remote Handling systems, especially at the start of the DEMO stage.

- ...this highlights the problem of using a large DEMO as a general CTF.

Component testing programme goals

- Studies of a Component Testing Programme (eg. Abdou – 1994) show two distinct goals:
  - Engineering Feasibility and Performance verification;
  - Component Engineering Development and Reliability Growth.
- Engineering Feasiblity and Performance Verification
  - Uncover the synergistic effects in the Tokamak environment
  - Verify performance beyond beginning of life until changes in properties become small (changes substantial up to ~ 1-2 MW.y/m^2)
  - Initial Failure Mode Effects data
  - Establish engineering feasibility (basic functions and performance – according to Abdou - up to 10-20% of lifetime).
  - Select concepts for further testing
- Component Engineering Development & Reliability Growth
  - Identify lifetime limiting failure modes and effects in full coupled environment – failure rate data
  - Iterative redesign/test programmes aimed at reliability improvements

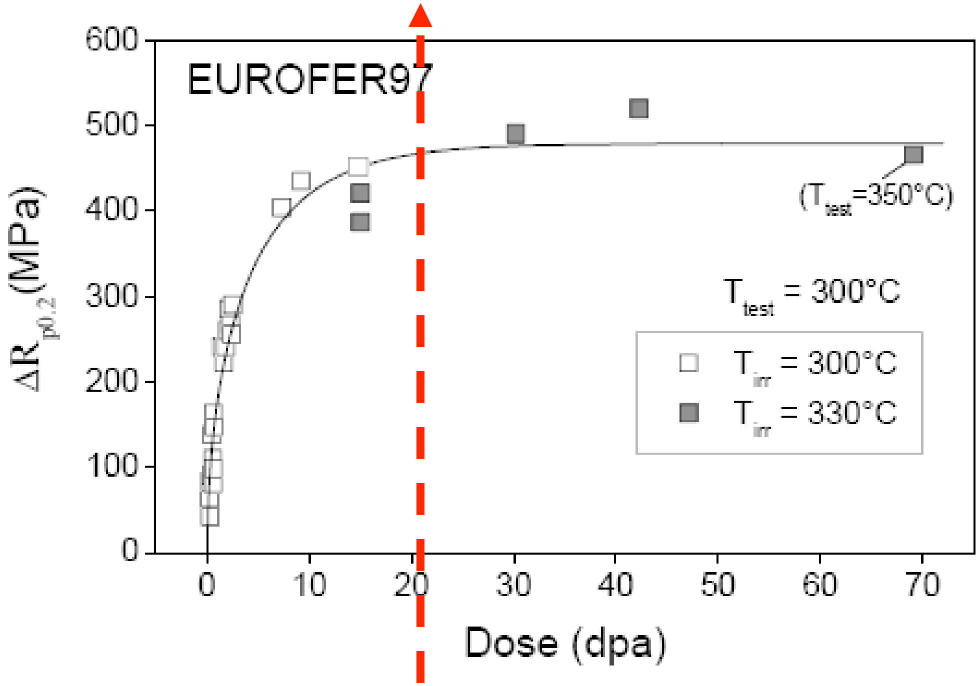
Considerations during development of a technology



- Strictly speaking the simple formulae only apply in the linear (low failure rate) range of the 'bathtub curve'
  --> low failure rates --> Poisson statistics.

- During a technology development, Test Facilities are needed to deal efficiently with the potential 'early death' phase (the 1-2 MW.yr.m^{-2} phase for neutron damage) --> CTF role (ITER to lesser extent).

Engineering feasibility and performance validation



- Testing up to ~ 2 MW.yr.m^{-2} (~20 dpa) at 1 MW.m^{-2} and 33% availability would take ~ 6 years

Comparison of Irradiation capability and Tritium consumption

Device	Major Radius [m]	inverse aspect ratio	Fusion Power [MW]*	Averaged Neutron Wall Loading [MW]	[dpa/fpy]	Tritium Consumption [Kg/fpy]
ST CTF	**0.85**	**0.65**	**35**	**1**	**10**	**1.8**
FDF	**3.5**	**0.2**	**246**	**1.5**	**15**	**13**
ITER	**6.32**	**0.3**	**500**	**0.5**	**5**	**26**
JET (DT)	3	0.33	16	0.07	0.7	0.8
DEMO(B)	8.6	0.33	3600	2.5	25	180

$$\frac{dpa}{fpy} \propto \frac{P_{Fus}}{\sqrt{2\left(1+\kappa^2\right)}\pi^2 \varepsilon R_0^{\ 2}} \qquad Burnup_T \propto \frac{m_T P_{Fus}}{E_{fus}}$$

Significant irradiation capability requires long pulse or steady state operation with significant fusion power production.

Reliability Growth

- Utilities will not rush out to build FPPs on the sole evidence of an established breeding cycle and an example of electricity generation.
- Industry needs economic models with established reliability and hence MTBF (and σ_{MTBF}) and MTTR of in-vessel components.
- Establishing these figures- the prelude to the 'Reliability Growth' phase needs a CTF programme in parallel to and beyond DEMO.
- A CTF will be able to test many 'identical' (same concept) small blanket modules in parallel.

Reliability Growth (II)

- How long is this programme?
- Depends on assumptions - involves a complex system of coupled probability equations. The most comprehensive US study (Abdou) quotes:
  - testing for ~ 6 MW.y/m^2
  - with 6-12 test modules
  - --> to get 90% confidence of reactor availabilities of 50% for the concept tested.
  - At 33% availability of the CTF and 1 MW.y/m^2 this would take 18 years.
- This answer is very model-dependent and hides a significant mathematical model with probability and confidence-level algorithms - EU programme should analyse some sample scenarios in order to fully define a CTF strategy.

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