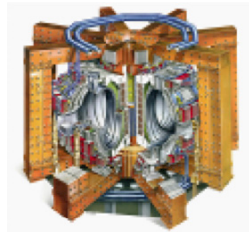


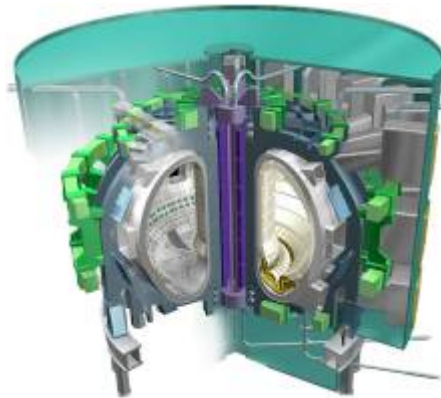
Magnetic Fusion Status and Outlook



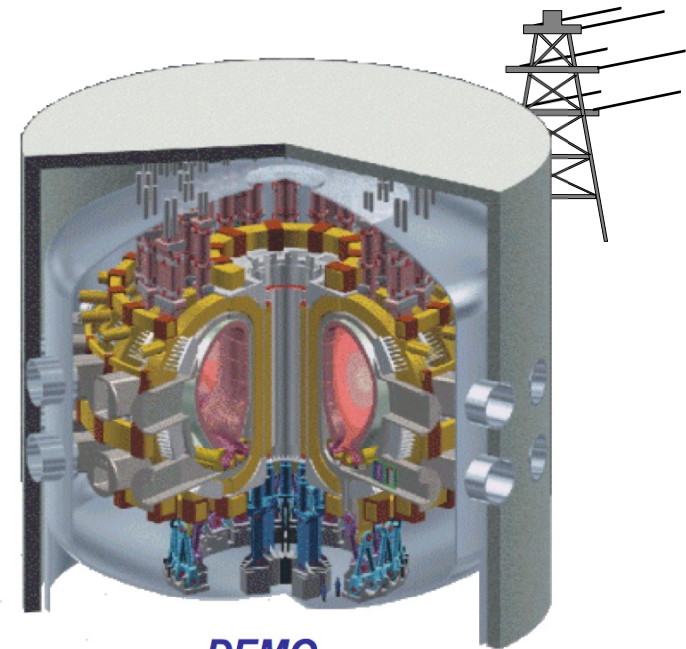
TEXTOR



JET



ITER



DEMO

Jef ONGENA
Royal Military Academy – Brussels

4th Joint EPS-SIF
International School on Energy
Villa Monastero
Varenna, Lago di Como
26 July 2017

J.Ongena

Fusion Status and Outlook

Varenna, Lago di Como, 26 July 2017

Outline

- Roadmap for the realization of fusion energy
- Largest tokamak in the world
 - Joint European Torus (JET)
 - Status of magnetic fusion research at JET
 - Comparison with inertial fusion at NIF
- Largest tokamak in construction :
 - International Thermonuclear Experimental Reactor (ITER)
- Plans for the future
 - Demonstrator Reactor (DEMO)
- Physics and Engineering developments needed
 - for the construction of DEMO

Joint European Torus (JET)

Largest tokamak in the world

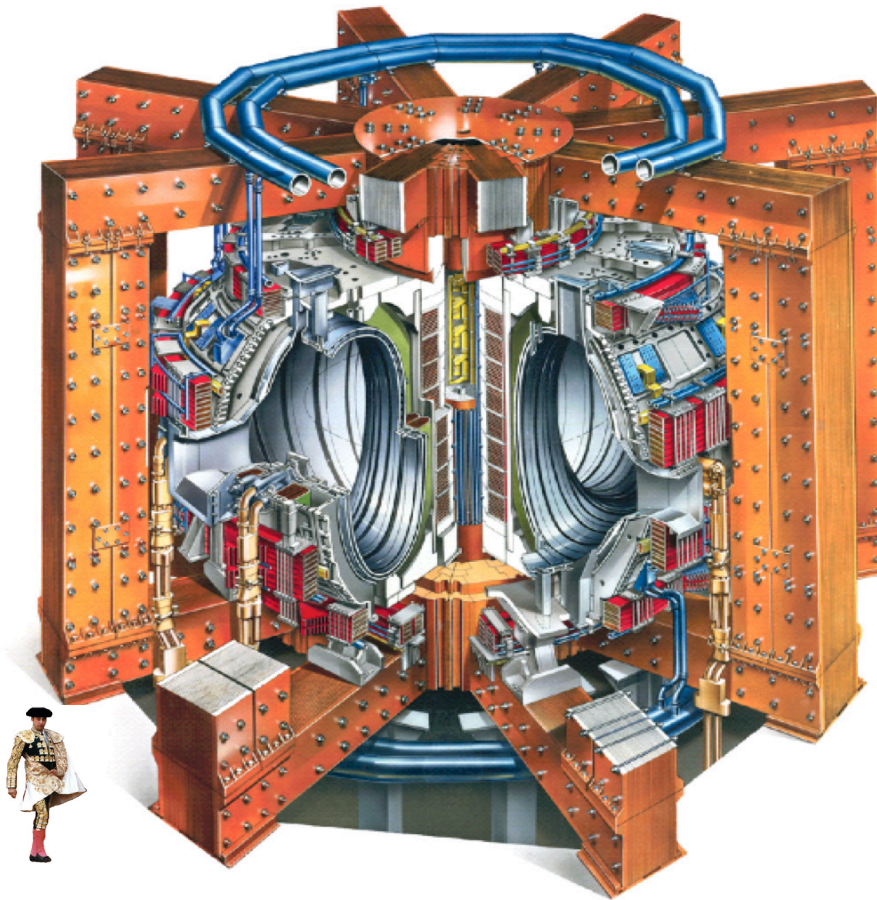
(in Culham, at 10km from Oxford)



www.jet.efda.org

Joint European Torus (JET)

Common European Facility (Oxfordshire, UK)
Largest tokamak worldwide



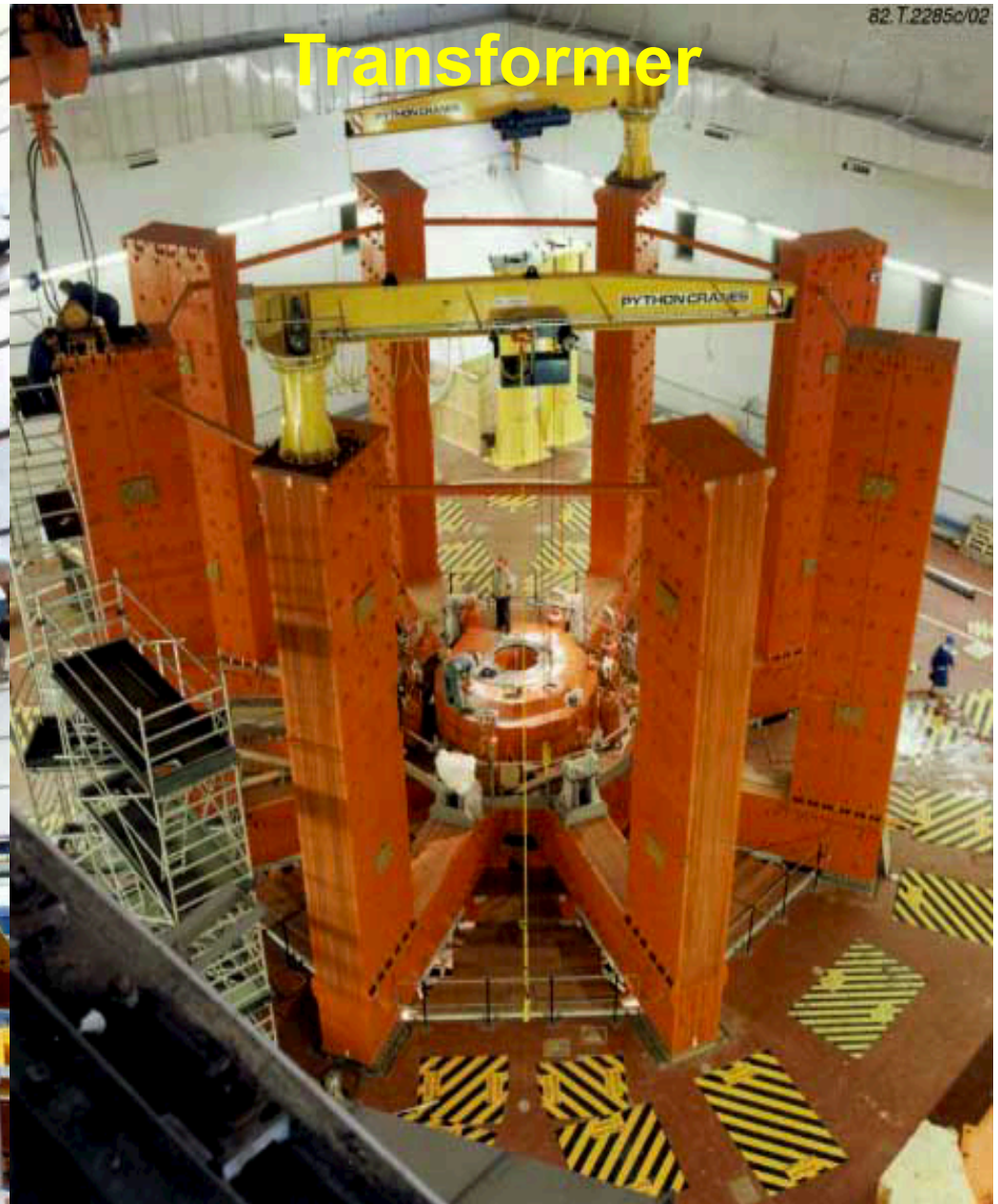
Vacuum vessel	3.96m high x 2.4m wide
Plasma volume	80 m³ - 100 m³
Plasma current	up to 5 MA in present (divertor) configurations
Toroidal magnetic field	up to 4 Tesla

82.348c

Construction of JET (1983)

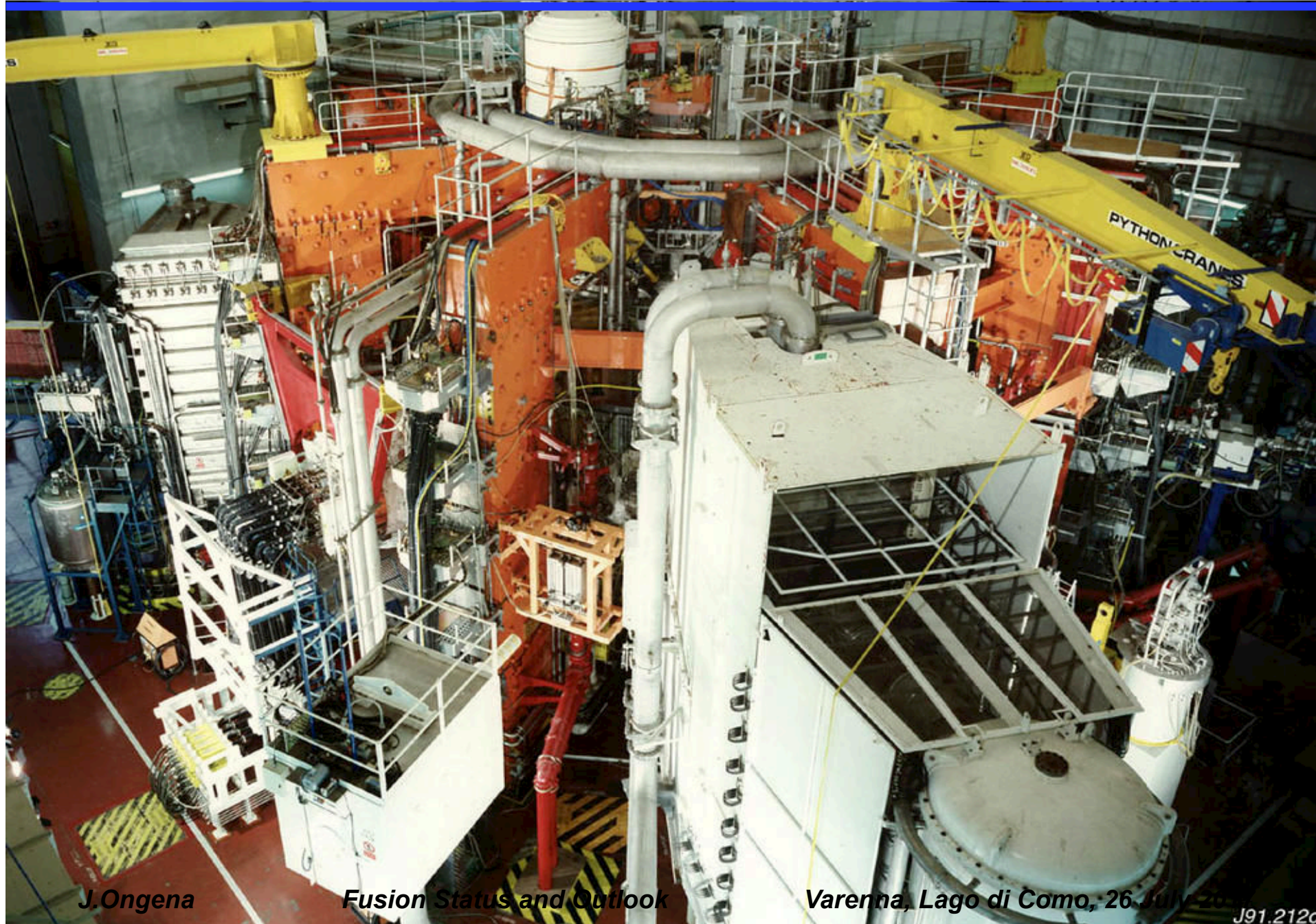


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Joint European Torus (JET)



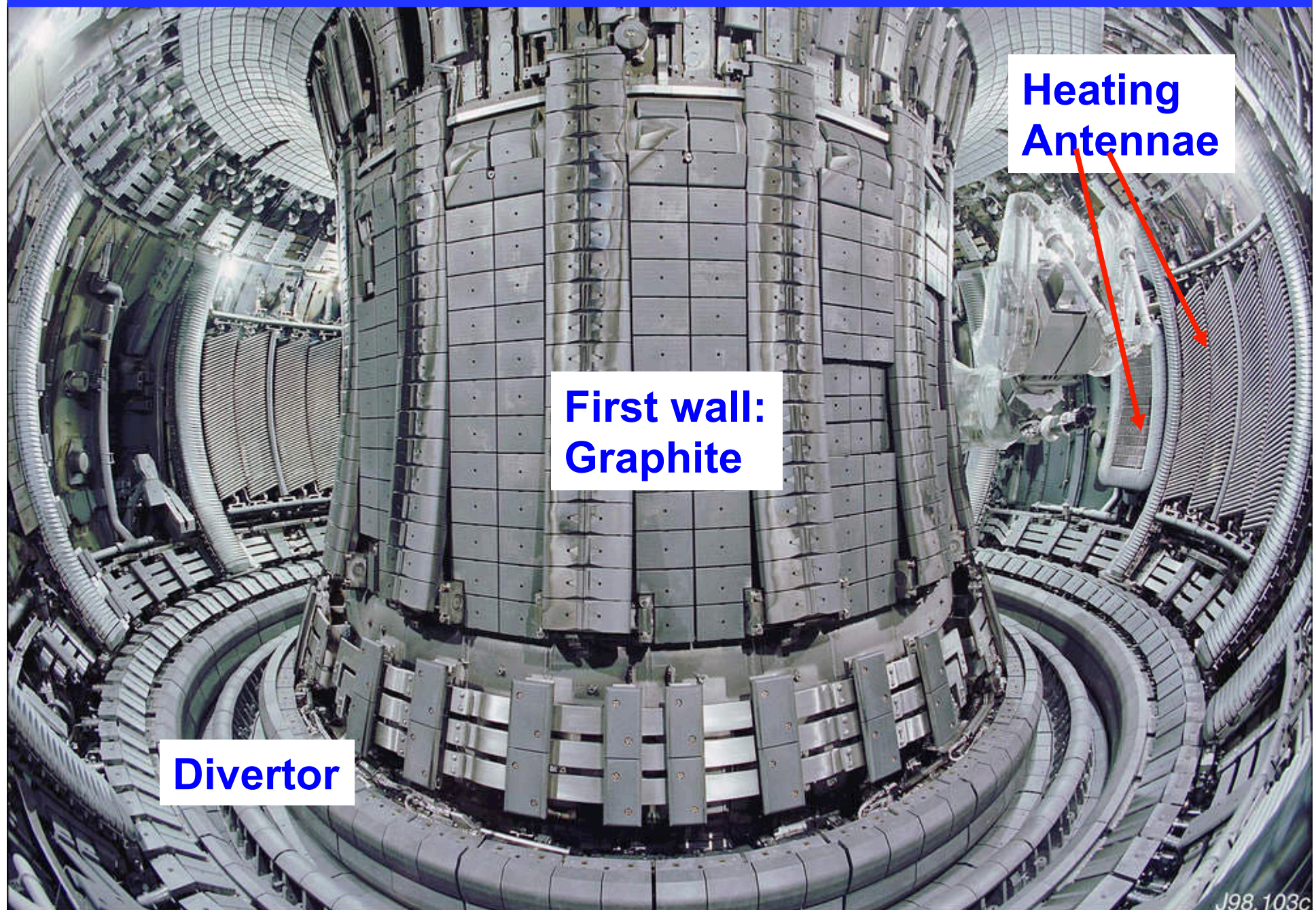
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Fusion Status and Outlook

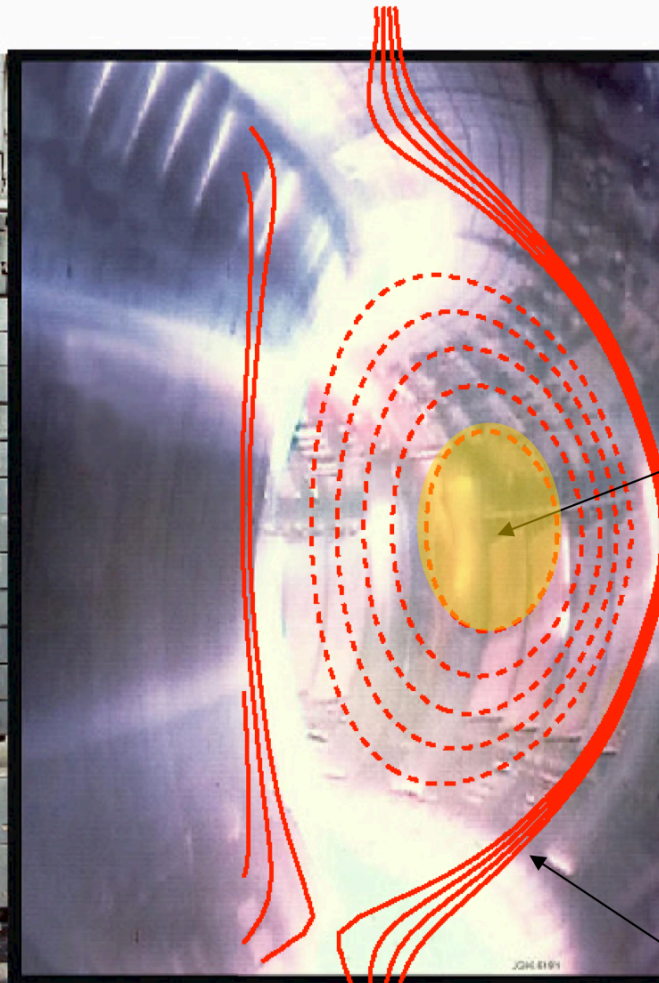
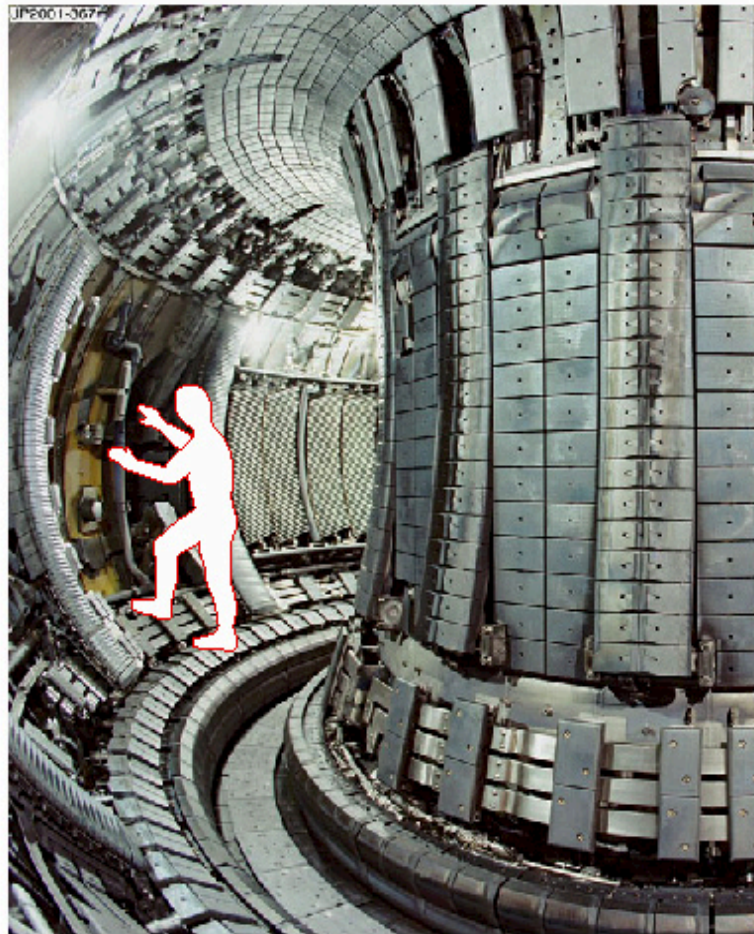
Varenn, Lago di Como, 26 July 20

J91.212c

Inside view of JET with Carbon first wall (up to 2010)



Inside of JET with and without plasma



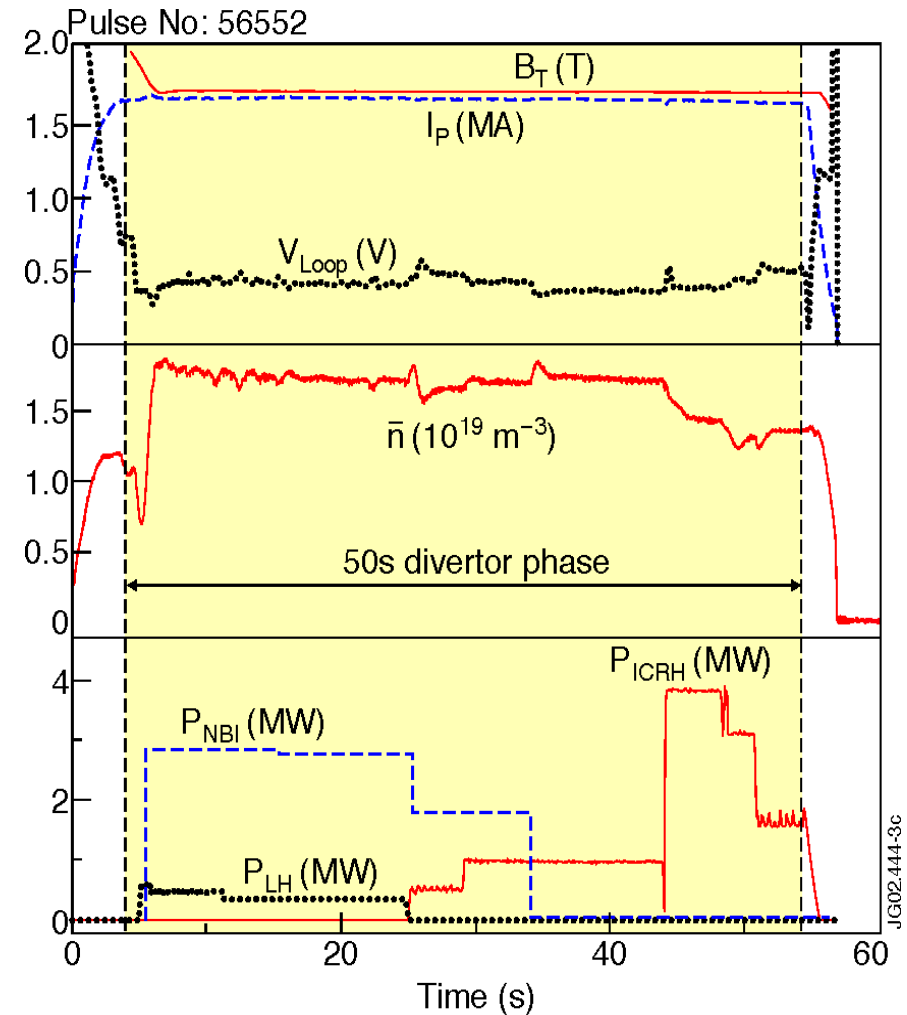
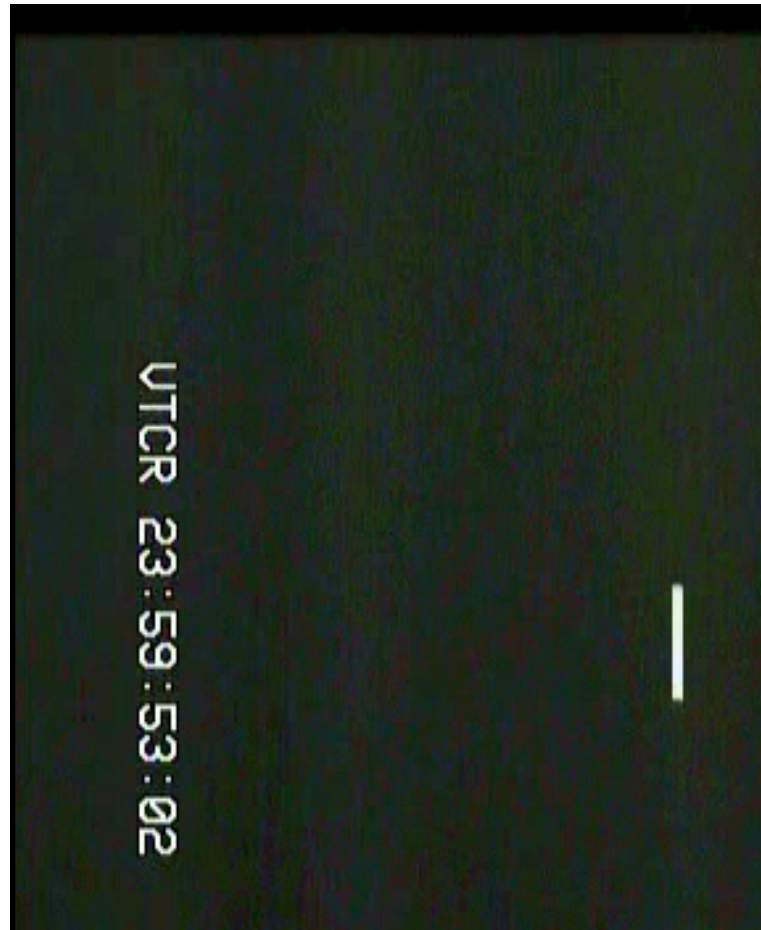
200 000 000 C

Magnetic surfaces

26 July 2017

Example of long pulse in JET

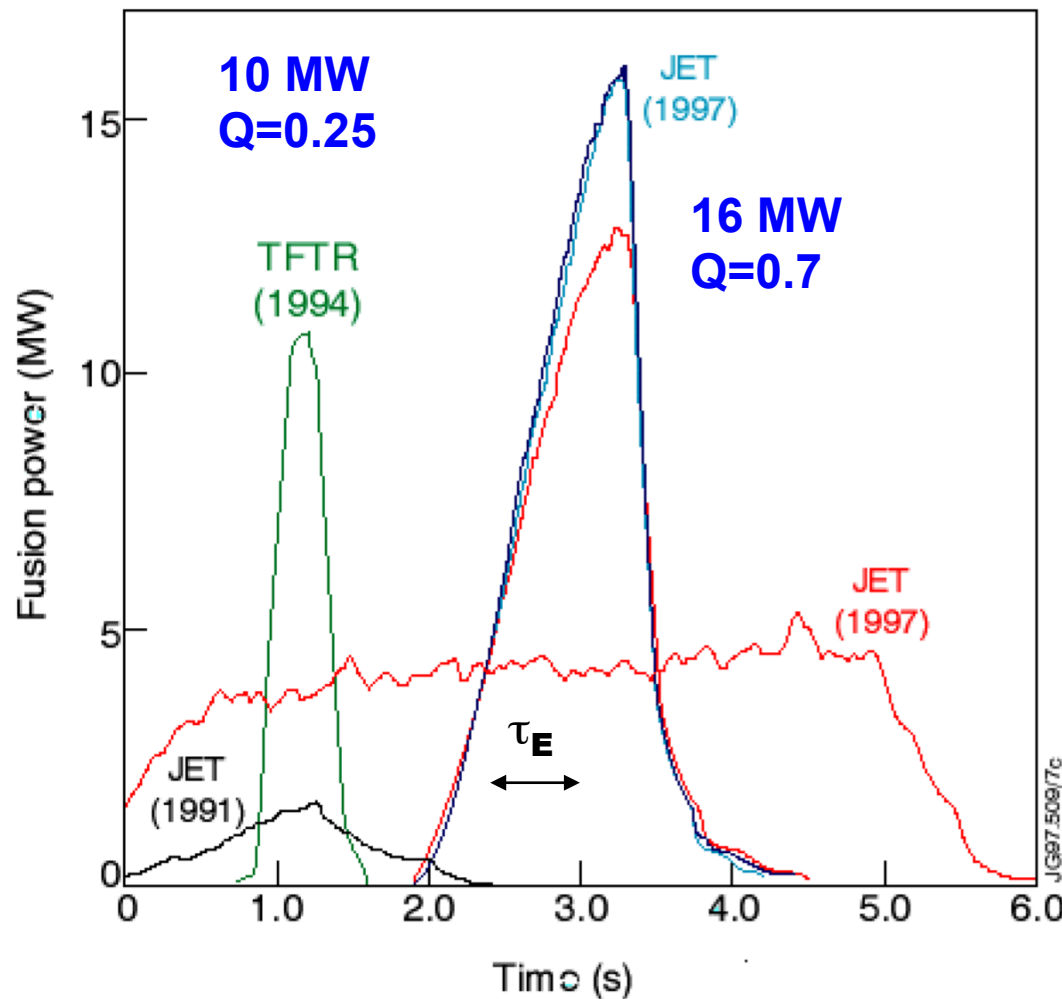
Long (1 min) JET Plasmas in ITER configuration



Also : Tore Supra (France): 6min30s, LHD (Japan): 30min

Status of Fusion Research

Record fusion energy production in JET
16MW (1997), $Q \sim 0.7$, very close to break-even



TFTR : Tokamak Fusion Test Reactor
Princeton University (1983-1997)

JET : Joint European Torus
Culham Labs (1983 – now)

Power amplification factor Q and fusion gain G

Magnetic fusion: operation in steady, long intervals

$$Q = \frac{P_{\text{fusion}}}{P_{\text{external heating}}} = \frac{E_{\text{fusion}} / \Delta t}{E_{\text{external heating}} / \Delta t}$$

Inertial fusion: pulsed operation

$$G = \frac{E_{\text{fusion}}}{E_{\text{external heating}}}$$

E_{fusion} : energy delivered to the pellet;

rather inefficient process: most of the energy is used for the ablation of the outer layers

Important difference between Q and G: time factor

If $G = 1$ in inertial fusion, $E_{\text{fusion}} = E_{\text{external heating}} = \text{few kJ}$

If $Q = 1$ in magnetic fusion

(e.g. JET, during 1s), $E_{\text{fusion}} = 16\text{MW} \times 1\text{s} = 16\text{MJ}$

(e.g. ITER, during 10 min), $E_{\text{fusion}} = 50\text{MW} \times 600\text{s} = 30\text{GJ}$

Latest results in Inertial Fusion Research

Fuel gain exceeding unity in an inertially confined fusion implosion

O.A.Hurricane et al., Nature, 20 february 2014, Vol 506, pp 343-348

Table 1 | Measured and derived implosion performance metrics

Quantity	N131119 ^{425 TW} 1.9 MJ	N130927 ^{390 TW} 1.8 MJ
Y_{13-15} (neutron)	$(5.2 \pm 0.097) \times 10^{15}$	$(4.4 \pm 0.11) \times 10^{15}$
T_{ion} (keV) D-T	5.0 ± 0.2	4.63 ± 0.31
T_{ion} (keV) D-D	4.3 ± 0.2	3.77 ± 0.2
DSR (%)	4.0 ± 0.4	3.85 ± 0.41
τ_x (ps)	152.0 ± 33.0	161.0 ± 33.0
$P0_x, P0_n$ (μm)	$35.8 \pm 1.0, 34 \pm 4$	$35.3 \pm 1.1, 32 \pm 4$
$P2/P0_x$	-0.34 ± 0.039	-0.143 ± 0.044
$P3/P0_x$	0.015 ± 0.027	-0.004 ± 0.023
$P4/P0_x$	-0.009 ± 0.039	-0.05 ± 0.023
Y_{total} (neutron)	6.1×10^{15}	5.1×10^{15}
E_{fusion} (kJ)	17.3	14.4
r_{hs} (μm)	36.6	35.5
$(\rho r)_{\text{hs}}$ (g cm^{-2})	0.12–0.15	0.12–0.18
E_{hs} (kJ)	3.9–4.4	3.5–4.2
E_{α} (kJ)	2.2–2.6	2.0–2.4
$E_{\text{DT,total}}$ (kJ)	8.5–9.4	10.2–12.0
G_{fuel}	1.8–2.0	1.2–1.4



JET – a testbed for ITER

Research topics in preparation of ITER

- Plasma facing materials: ITER like Wall
- Development of operational scenarios for ITER
- Heating systems
- Mitigation of Edge Localised Modes & avoidance of disruptions
- Preparation of diagnostics in a high neutron flux environment
- ...

Which first wall for ITER/DEMO ?

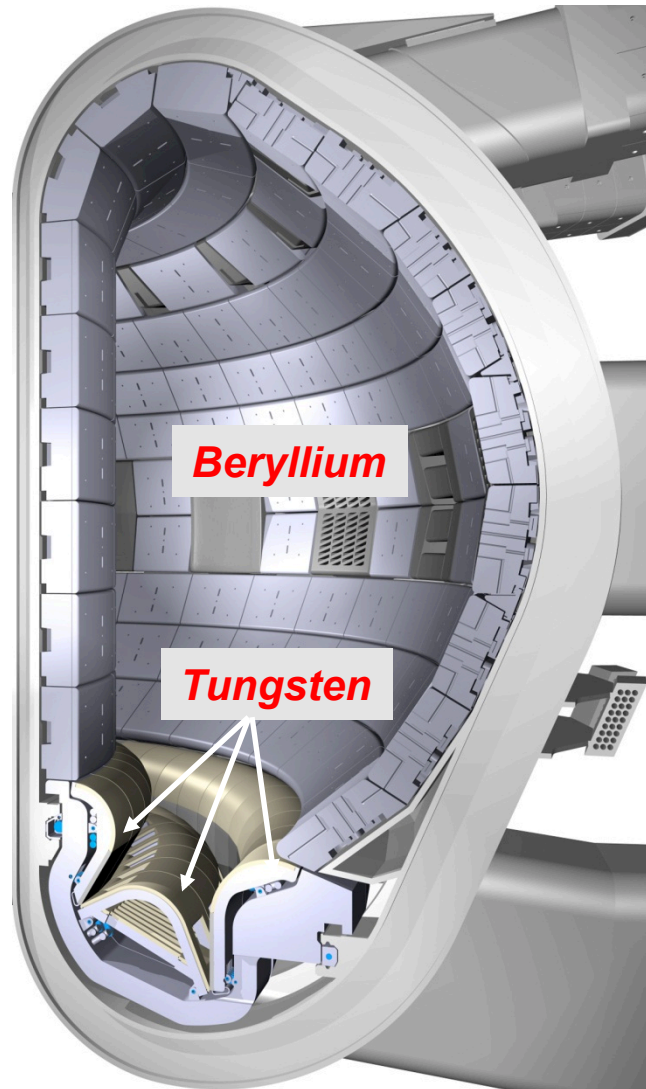
Graphite has been used in last 20 years to optimise plasma requirements

- High temperature : no melting, only sublimation at $T \sim 3000\text{C}$
- Easy plasma operation/performance
- Resistance against power transients and operational failures

Not suitable for power reactor because of additional requirements

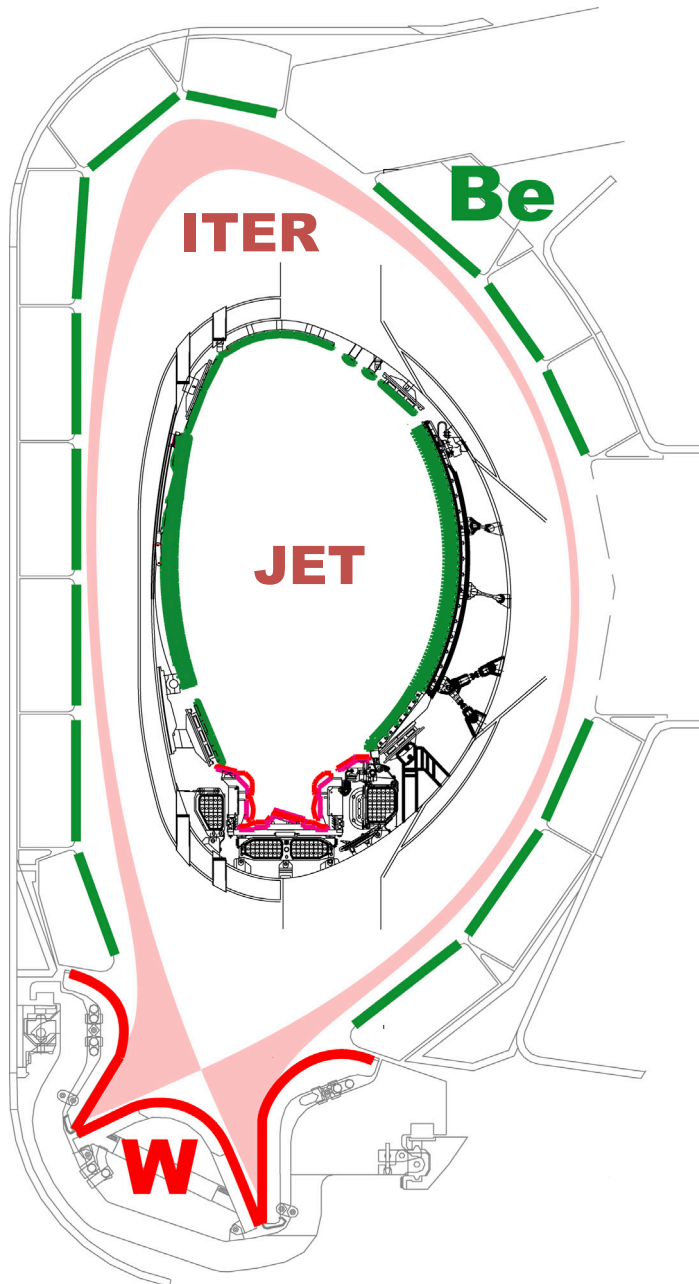
- Lifetime (low erosion) (DEMO)
- Low T uptake (ITER)
- Neutron compatibility (DEMO)

Which first wall for ITER/DEMO ?



- **Beryllium (Be) ?**
 - + Reduced T retention in vacuum chamber
 - + Low Z
 - But rather low melting point : 1287 C
- **Tungsten (W) ?**
 - + Strongly reduced retention
 - + High melting point : 3422 C
 - But high Z : very low concentration tolerable
- **Possible solution : combination Be + W ?**
 - Minimise use of W
 - Only there where it cannot be avoided : divertor

ITER-Like Wall (ILW) at JET (since 2011)



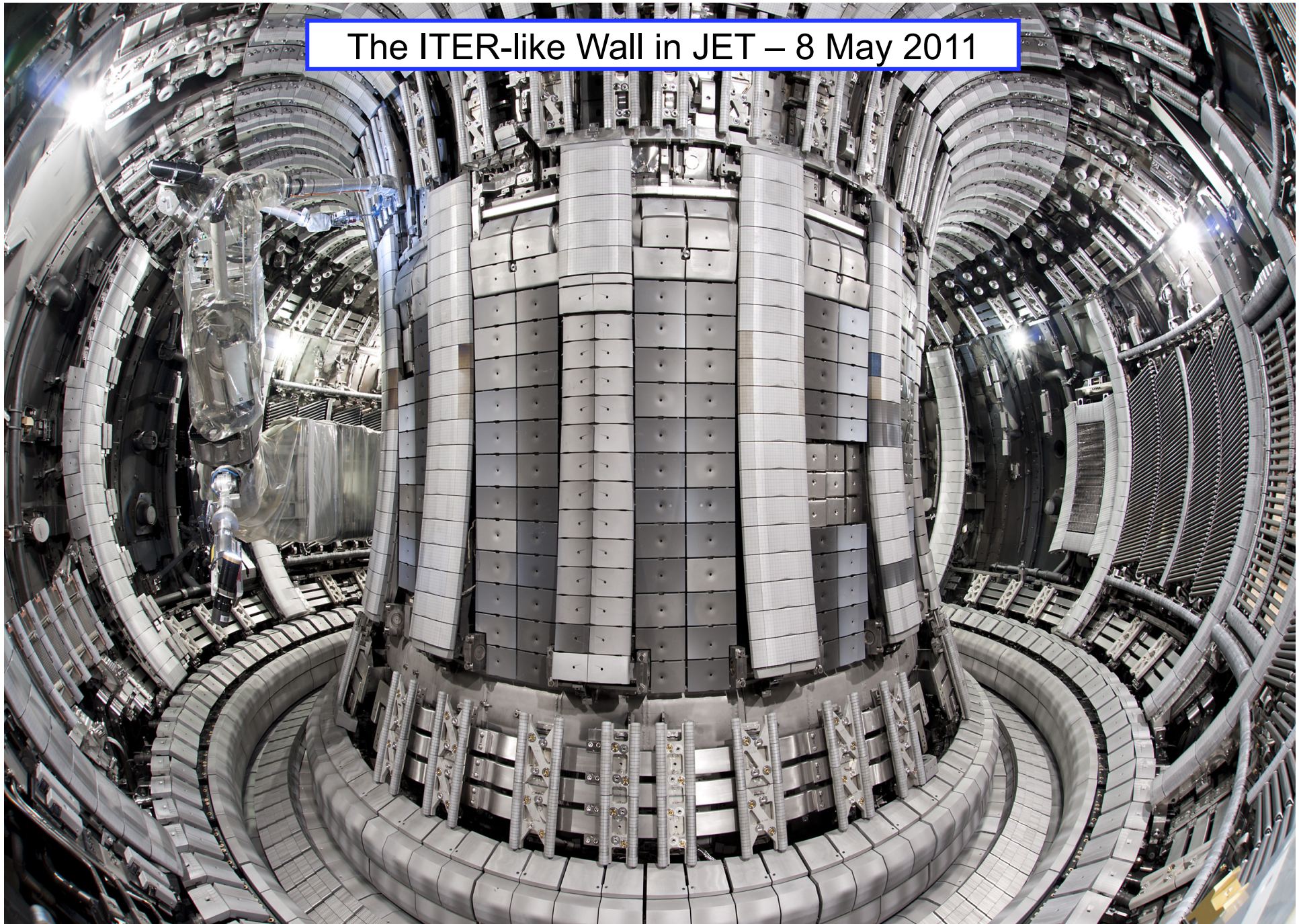
The combination of Beryllium (**Be**) and tungsten (**W**) is the first wall for ITER:

- “Carbon-free” environment
 - Reduced tritium retention
 - Loss of carbon as main radiator
 - Very low content of low Z impurities (O and C)
- Change in way of operating a tokamak
 - Need for better plasma control
 - Need for melt protection schemes of first wall elements

Main goals of the ILW experiment in JET

- I. Demonstrate low fuel retention, migration and possible fuel recovery*
- II. Demonstrate plasma compatibility with metallic walls*

The ITER-like Wall in JET – 8 May 2011



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ITER-like Wall Project Metrics

Number of installed items: 2,880

Number of individual tiles: 5,384 Be tiles (~2 tons Be / ~ 1m³)
1,288 W-coated CFC tiles
9,216 W-lamellas (~2 tons W / ~ 0.1m³)
15,828

Total number of parts: 82,273 counting bulk W modules as one part

Bulk W total parts: 191,664 including 100,080 shims

Duration of manned access: < 7% of in-vessel time
(Mostly infrastructure and welding/repairs)

A summary of results in JET from operation with the ILW

A wealth of new and very relevant results for ITER

- Cleaner plasmas: much less impurities (mainly C and O)
- Much easier plasma startup
- Different plasma conditions : lower edge densities (pumping of the wall much stronger); influence on confinement
- Disruption force stronger (lower natural radiation): need for mitigating the disruption by blowing high Z gases (Ar) in plasma
- Need to be careful with W accumulation
- Building up experience with IR observation/protection system for the Be wall
-

From JET to ITER



***ITER = International Thermonuclear Experimental Reactor**

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Fusion Status and Outlook

Varennna, Lago di Como, 26 July 2017

ITER – a unique project

- **More than half of world population is represented**
- **Dominant position of new economies (India, China)**

What is ITER?

- ITER's overall programmatic objective:
 - to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes
 - to design, construct and operate a tokamak experiment at a scale which satisfies this objective

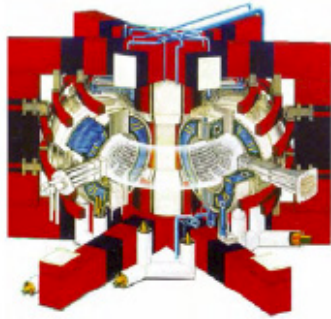
- ITER is a tokamak designed to confine a DT plasma in which α -particle heating dominates all other forms of plasma heating
⇒ **an experimental nuclear fusion reactor**
 - ✓ **Designed to achieve $P_{\text{fusion}} = 500$ MW with gain $Q \geq 10$ for 300-500 s**
 - ✓ **Aims to achieve $P_{\text{fusion}} \geq 350$ MW with $Q \geq 5$ for 1000-3000 s**
 - ✓ **Aims at exploring “controlled ignition” ($Q \geq 30$)**

- ITER is a **unique worldwide collaboration** in research involving the EU (plus Switzerland), China, India, Japan, Russian Federation, South Korea and United States

ITER Design Parameters

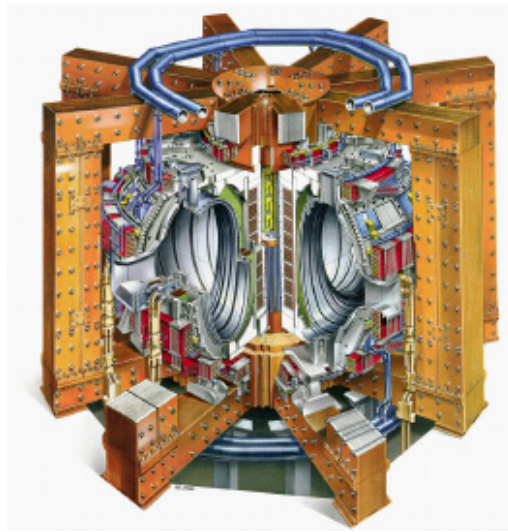
	ITER
Major radius	6.2 m
Minor radius	2.0 m
Plasma current	15 MA
Toroidal magnetic field	5.3T
Elongation / triangularity	1.85 / 0.49
Fusion power amplification	≥ 10
Fusion power	~500 MW
Plasma burn duration	300-500 s

ITER : ~ 2x larger than JET



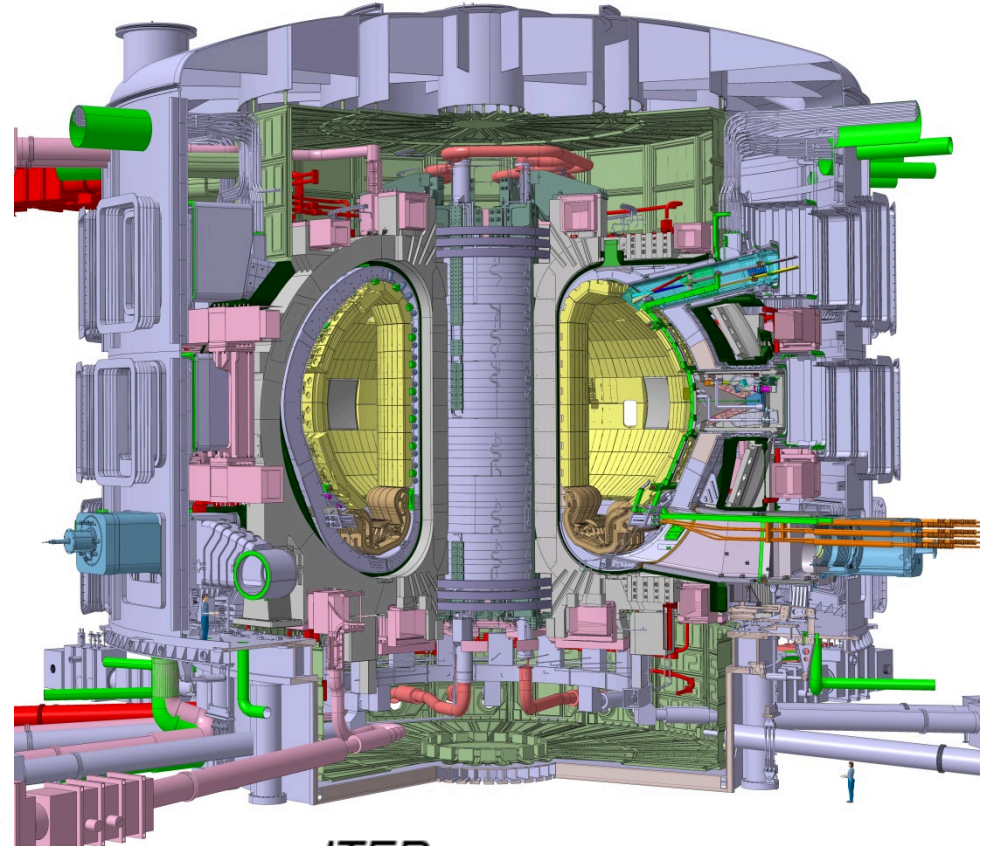
Tore Supra

V_{plasma} 25 m³
 P_{fusion} ~0
 t_{plasma} ~400 s



JET

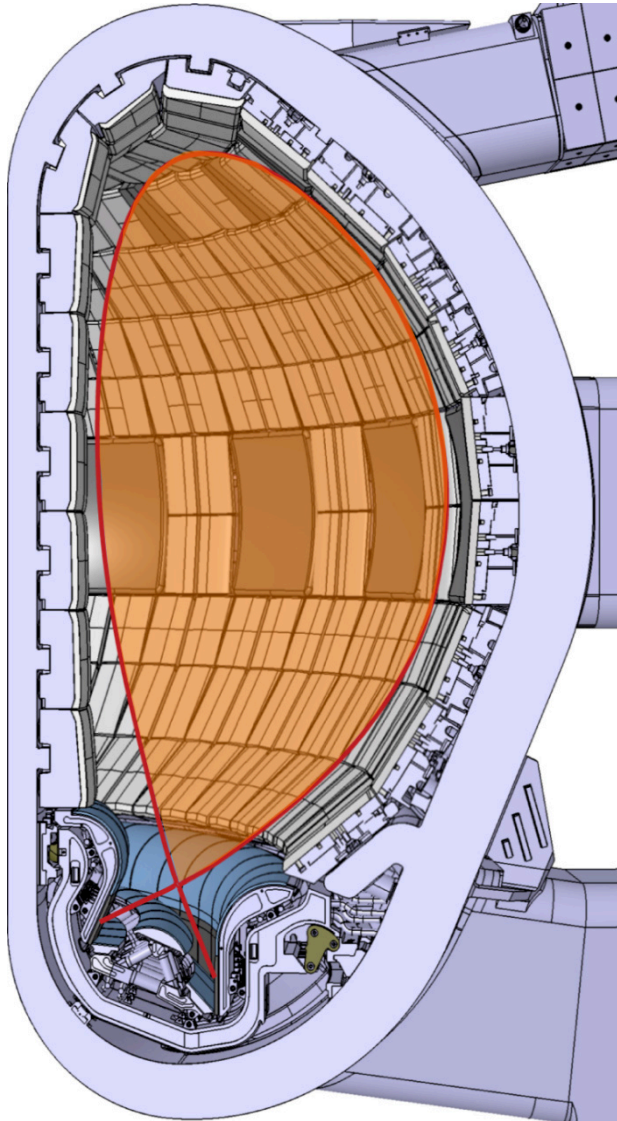
V_{plasma} 80 m³
 P_{fusion} ~16 MW 2s
 t_{plasma} ~30 s



ITER

V_{plasma} 830 m³
 P_{fusion} ~500 MW 500s
 t_{plasma} ~400 s

Comparison ITER / JET components



- dimensions: $R=6.2$ m, $a=2$ m
2 x JET
 - plasma heating 150 MW
5 x JET
 - discharge duration: 500s
50 x JET
 - plasma energy content: 300 MJ
30 x JET
 - fluency / discharge: $\sim 10^{27}/\text{m}^2$
 ~ 2000 x JET
- ⇒ **plasma surface interaction /
plasma facing materials
are central issues**

ITER - Main Features

$$I_p = 15 \text{ MA}$$

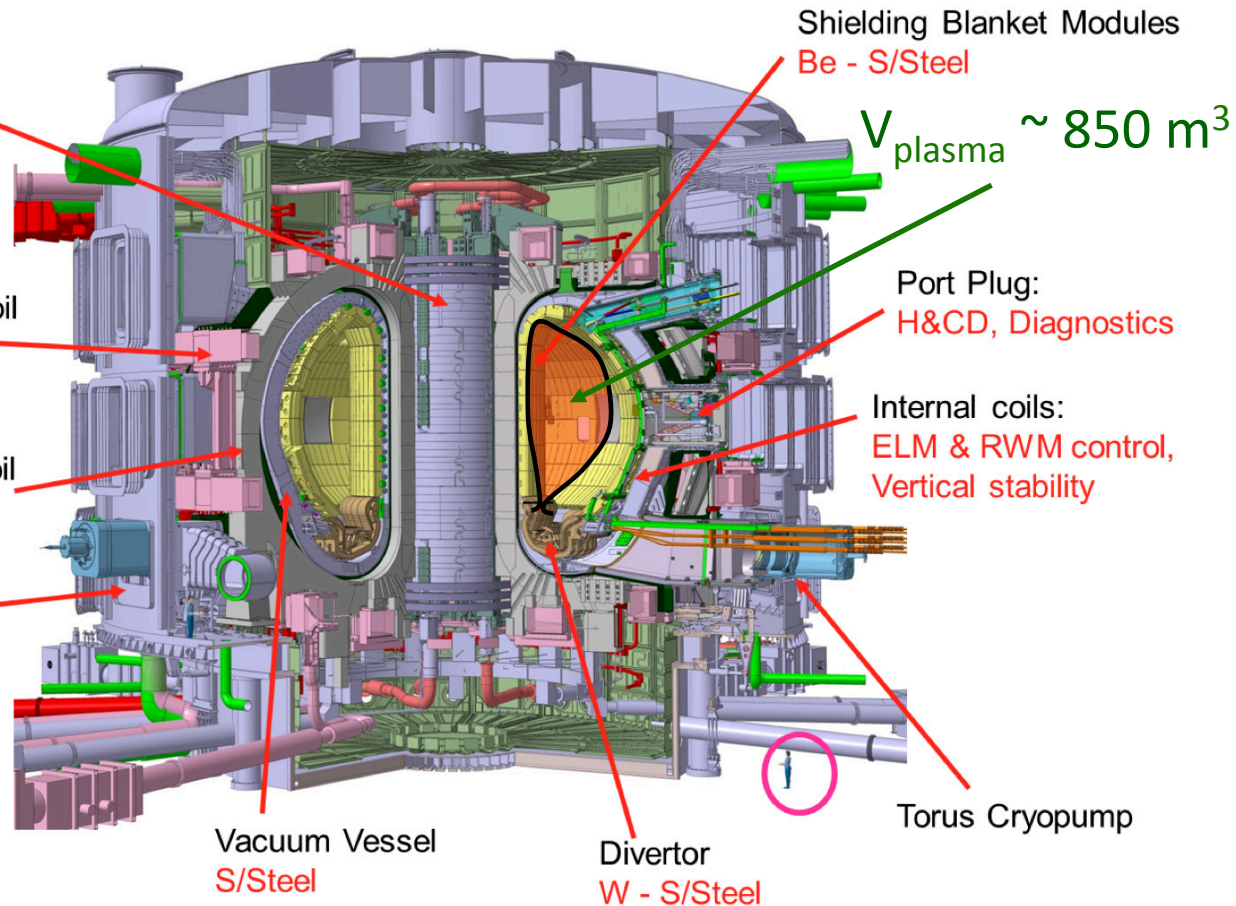
Central Solenoid
Nb₃Sn-SC

Poloidal Field Coil
NbTi-SC

Toroidal Field Coil
Nb₃Sn-SC

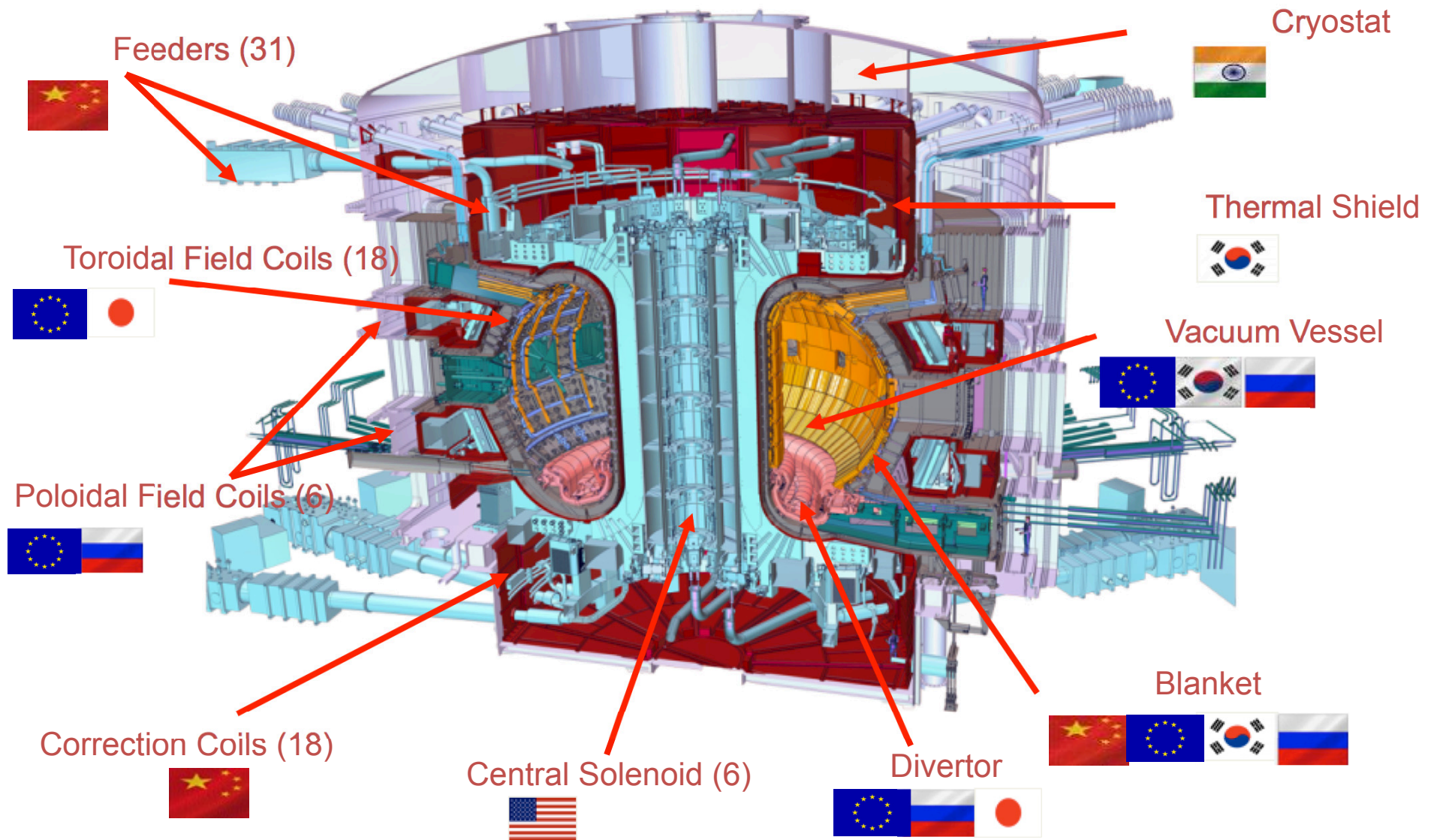
$$B_t = 5.3 \text{ T}$$

Cryostat
S/Steel



NBI (1 MeV)	ECH (170 GHz)	ICH (40-55 MHz)	LH (5 GHz)	Total
33 MW (+16.5 MW)	20 MW (20 MW)	20 MW (20 MW)	0 MW (20 MW)	73 MW (130 MW-110 MW simultaneous)

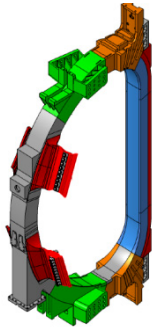
Construction of ITER – Who manufactures what?



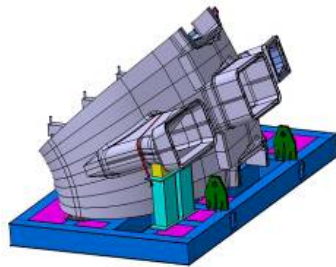
Delivering Components to ITER

Local Communities Provided Road Upgrades

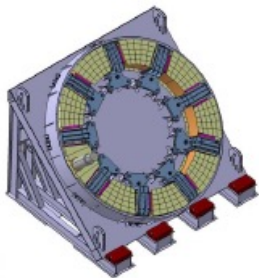
FRANCE



TF Coil ~360 t
16 m Tall x 9 m Wide



VV Sector ~400 t
12 m Tall x 9 m Wide



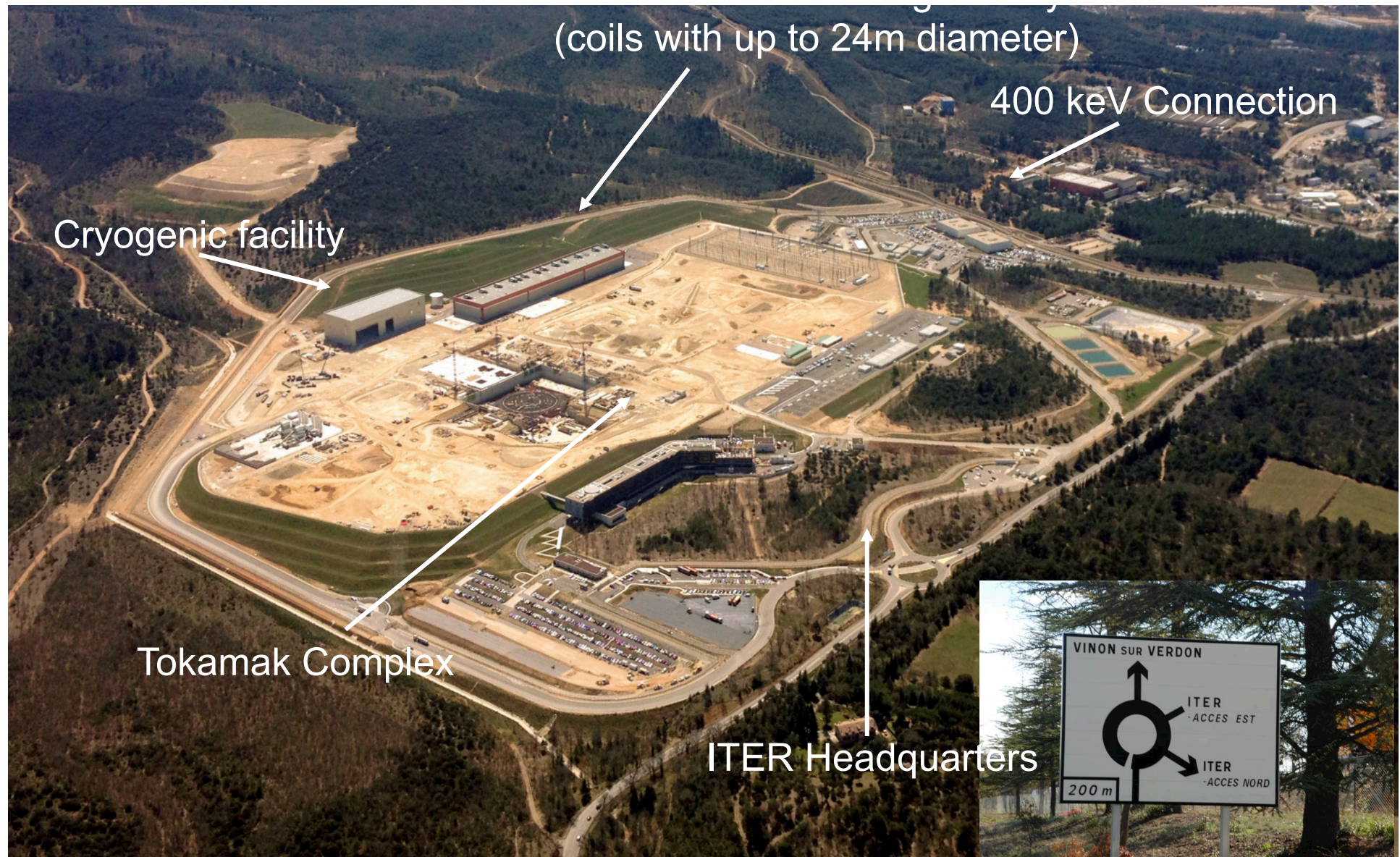
PF1 Coil ~200 t
9.4 m Diameter

Test Convoy arriving on the work site



Test Convoy for delivery of heavy components
(TF Coils, VV Sectors, and PF1 Coil)
successful (16th – 20th September)

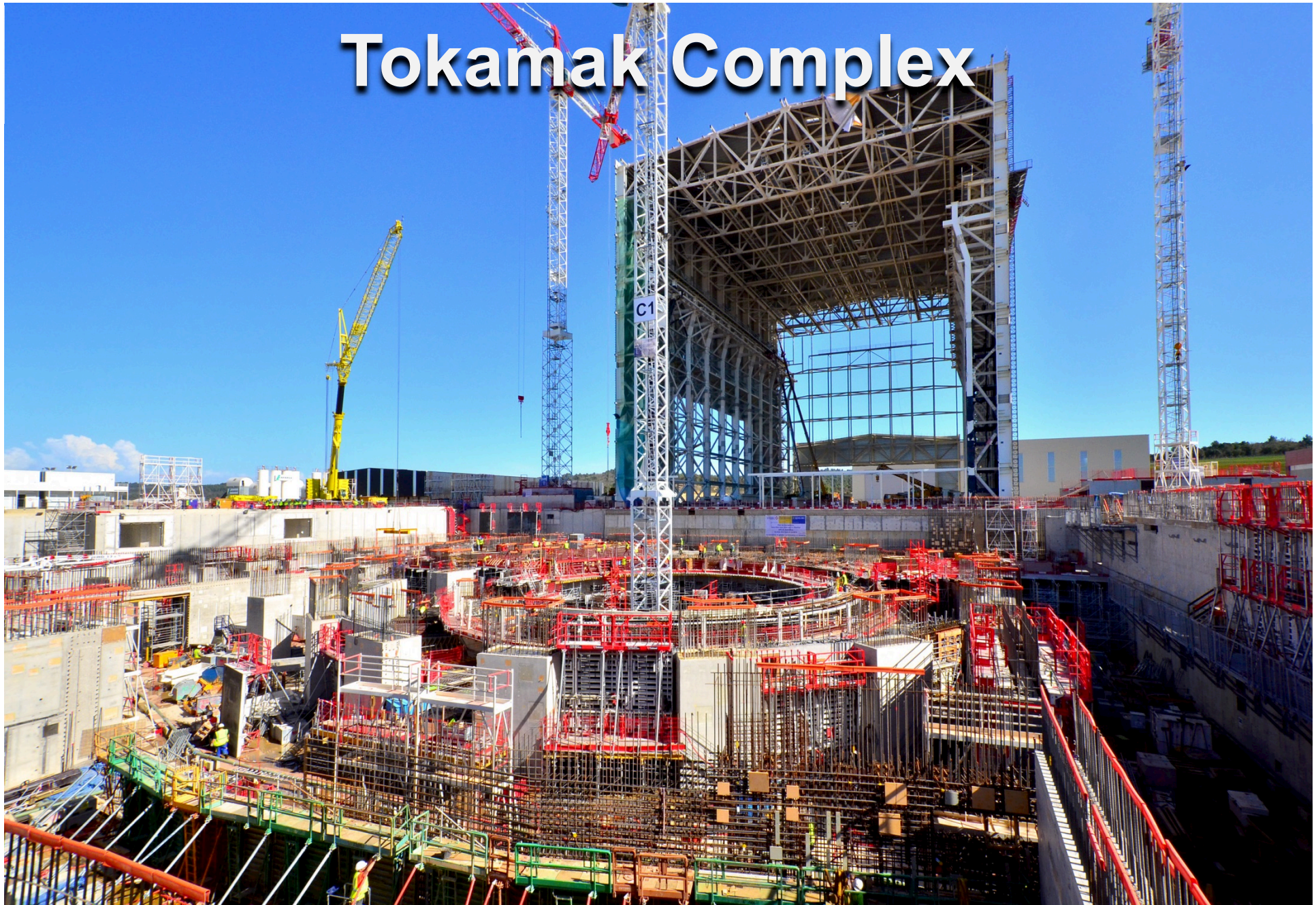
Overview of the laboratory (2010)



Overview of the laboratory (2016)



Tokamak Complex



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Tokamak Complex (April 2016)



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Assembly hal (60m high)

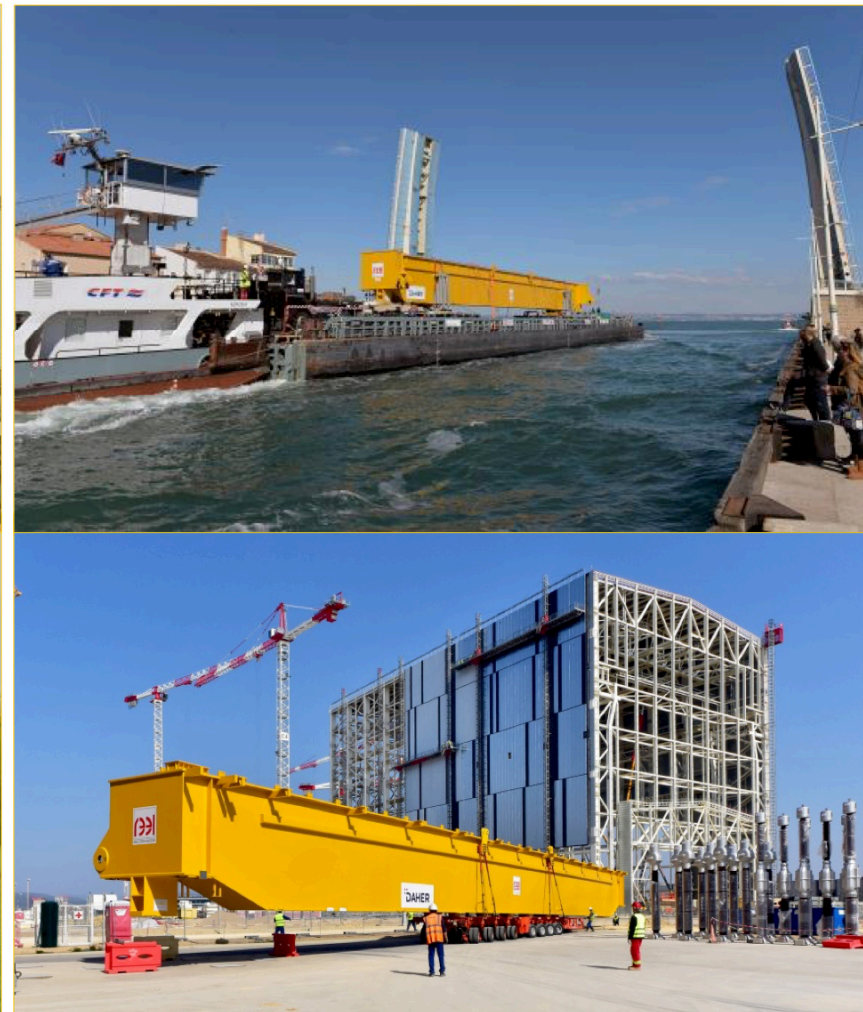


ITER May 2017

Tokamak Housing



Transport of huge beams for crane in assembly hall



Specially reinforced route between Marseille – Manosque

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Poloidal coil winding building (2016)



Cryolab



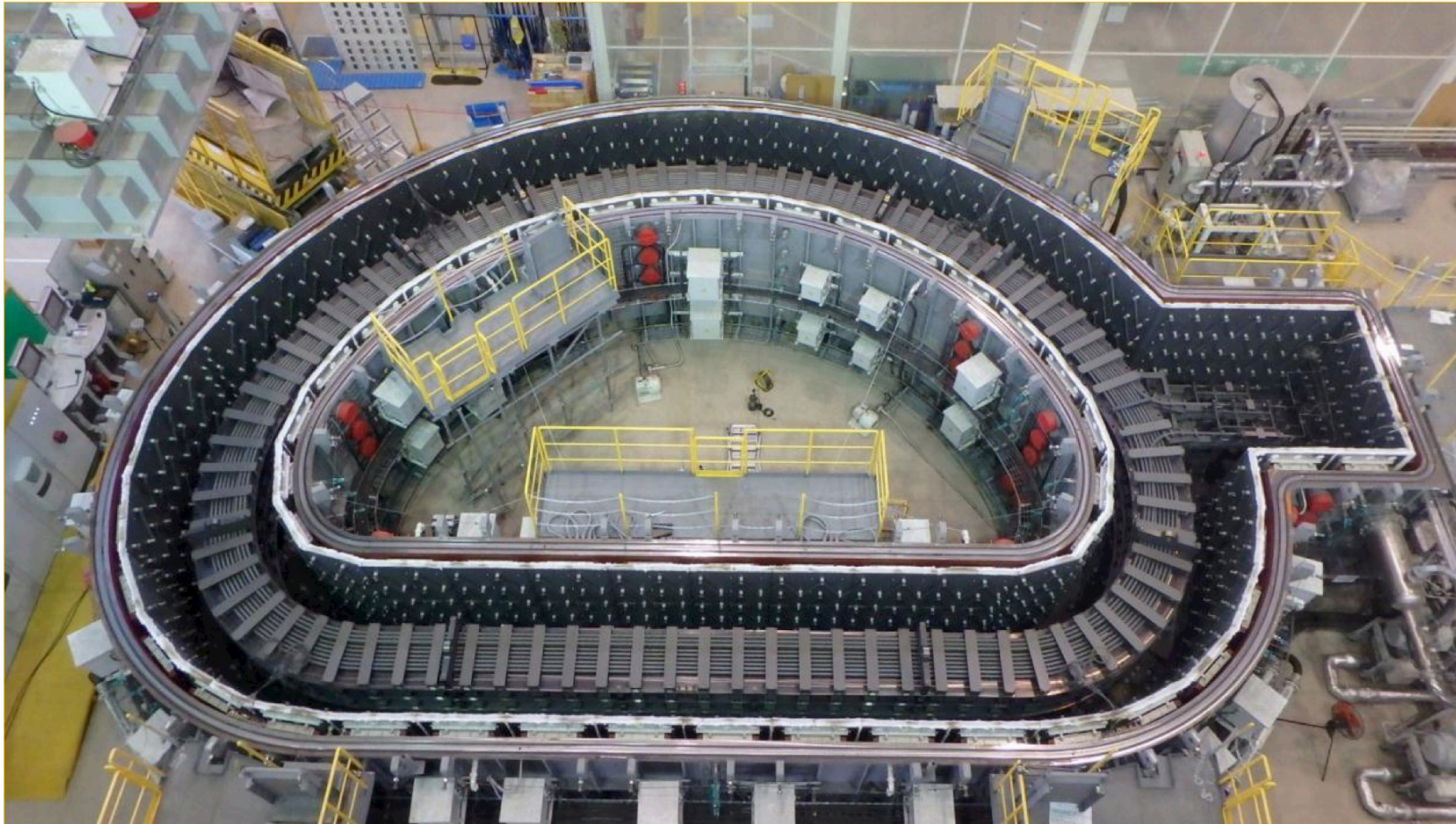
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Construction of toroidal coils (Japan)

Heat treatment of the superconducting coils



Traitement à haute température des bobinages des aimants de champ toroidal à l'usine Mitsubishi de Futami (Japan).

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Ground insulation for the toroidal coils winding pack (Europe)



En Italie, la fabrication de la première des 10 bobines de champ toroïdal (sur 18) que doit livrer l'Europe est en cours. Une fois finalisée, la bobine sera insérée dans une « cassette » d'acier – l'ensemble pèsera plus de 310 tonnes, le poids d'un Boeing 747 à pleine charge.

Base plate for the ITER cryostat (India)

30m diameter

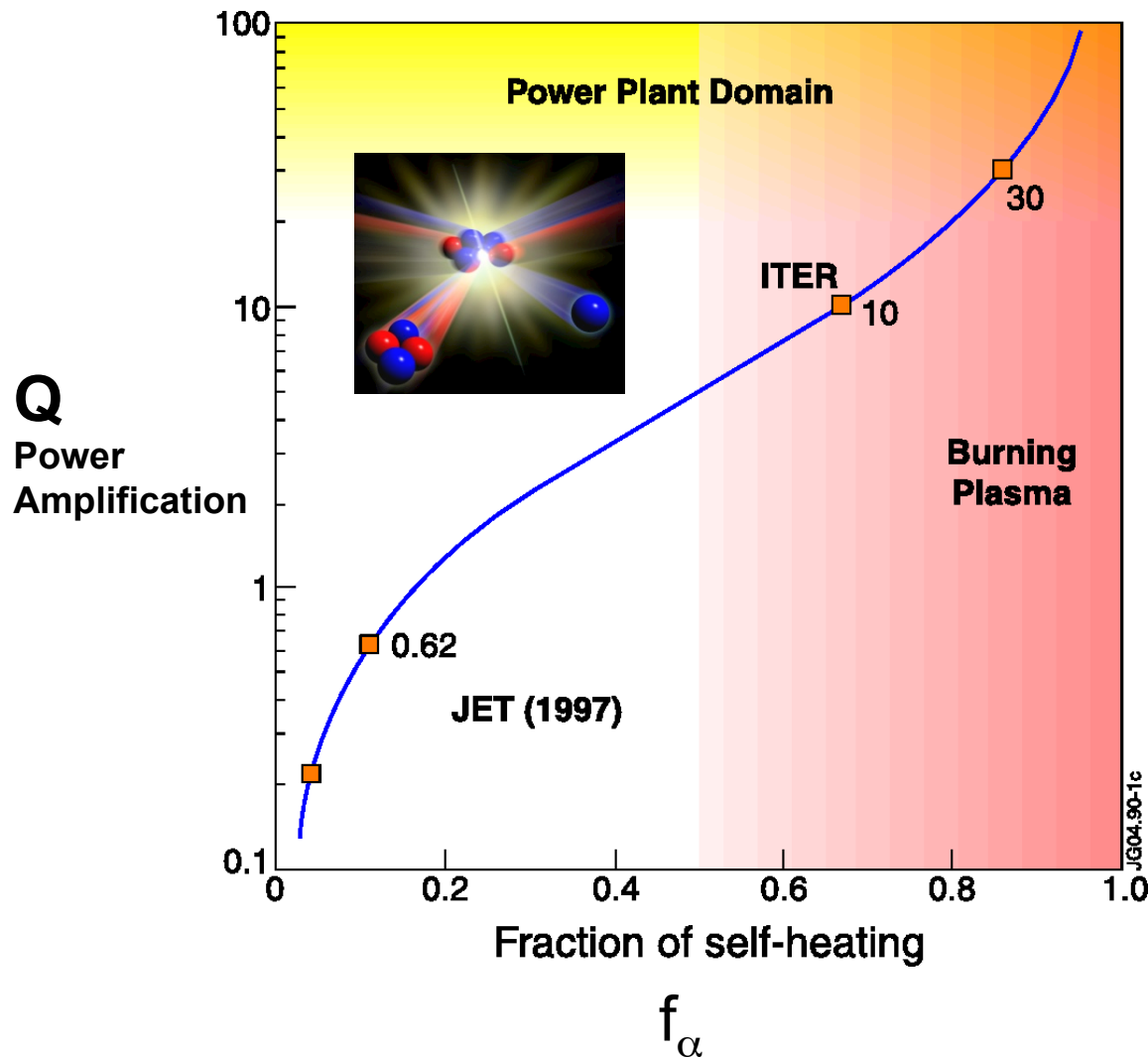


La base du cryostat a été provisoirement assemblée dans l'usine Larsen & Toubro Ltd en Inde. Les éléments ont été livrés les 10 et 17 décembre 2015.



La structure de transport et d'assemblage du cryostat (30 mètres de diamètre) est finalisée. Démontée, elle sera livrée à ITER dans les semaines qui viennent.

ITER is important for further progress in fusion



Power amplification
(engineering parameter,
related to plant efficiency)

$$Q = P_{\text{fusion}} / P_{\text{ext}}$$

Fraction of plasma self-heating by fusion born α -particles

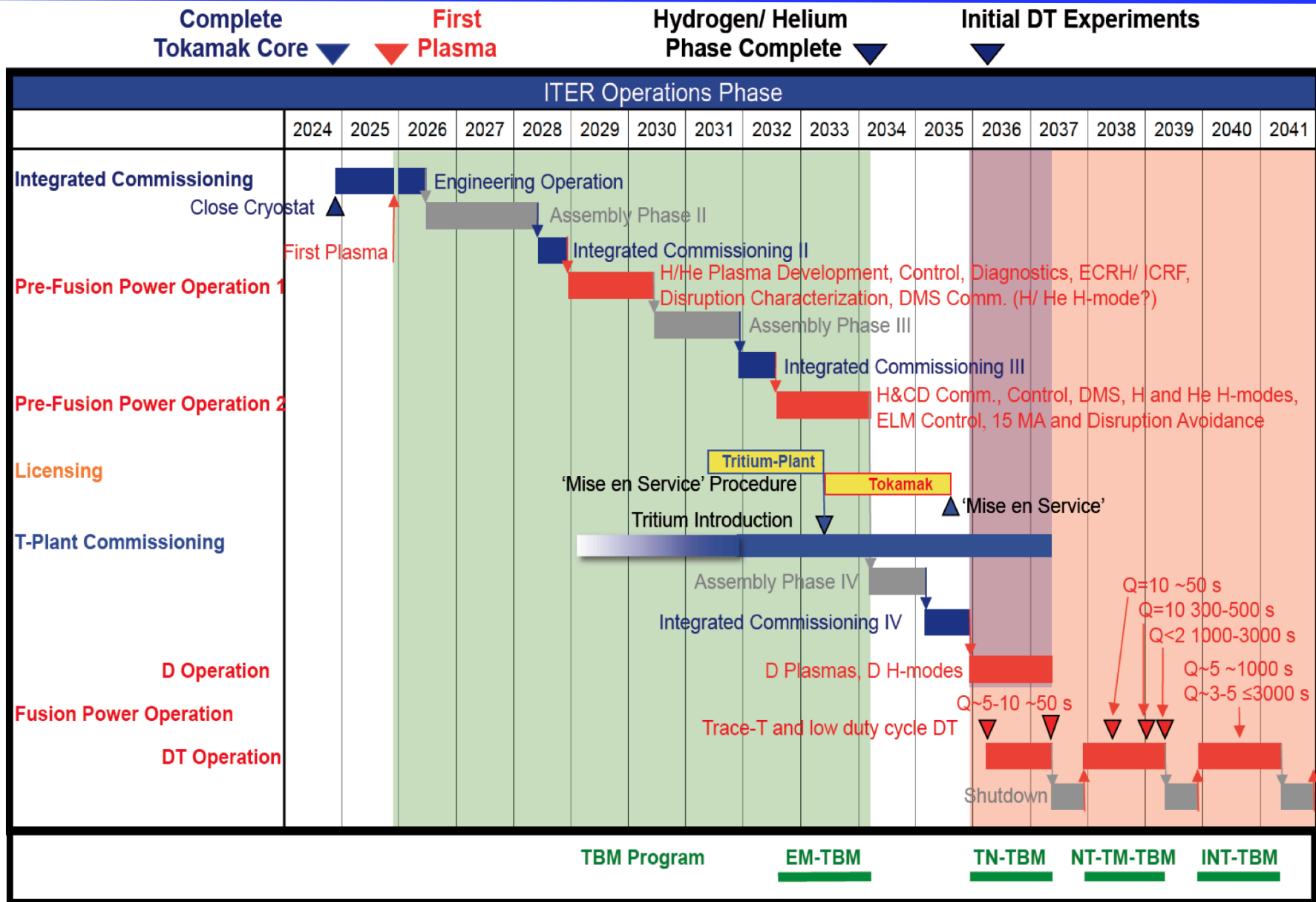
$$f_\alpha = Q / (Q+5)$$

with $Q > 10$, ITER will provide

****for the first time ****

access to plasmas with
adequate self heating
($f_\alpha > 2/3$)

ITER Research Plan



ITER needed to prepare the definition of DEMO

Data needed for the definition of a fusion reactor (DEMO)

Test on ITER all necessary ingredients and integrate them in one scenario

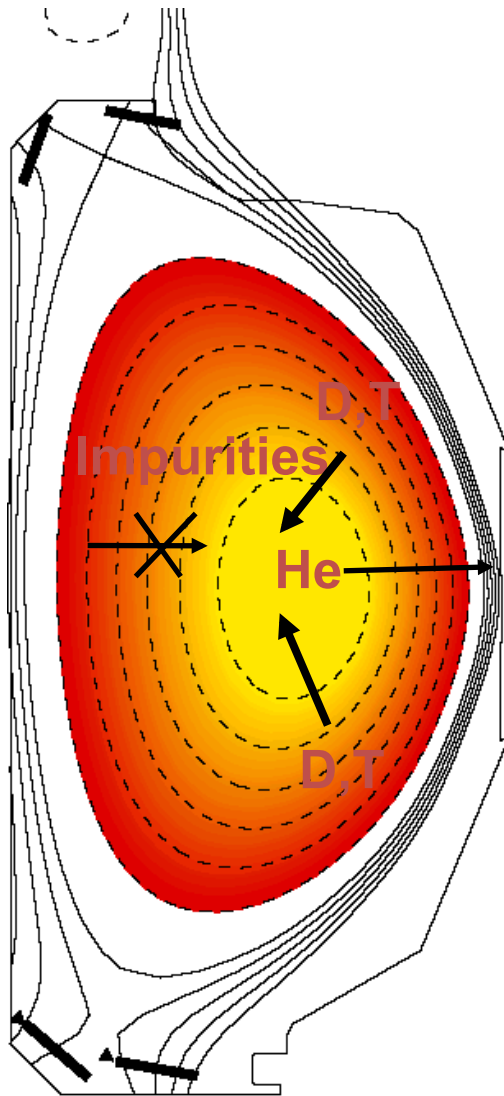
- Stable and stationary operating scenario
- Getting the necessary heat into (large) plasmas
- Controlling heat load to first wall
- Controlling disruptions
- Run a tokamak with an all metallic wall
- Prepare Tritium Breeding techniques
-

We have learned up to now ‘how to start the fusion fire’

What do we need to learn ? ‘how to maintain the fusion fire’

How to maintain the fusion 'fire' ?

(Some) of the questions waiting for an answer



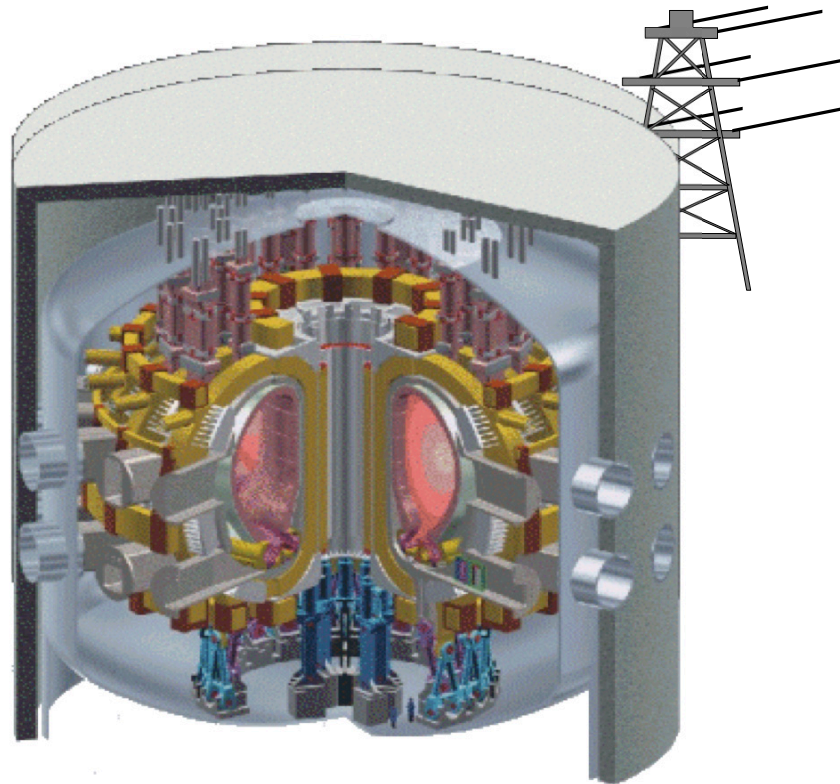
Physics

- Clean plasma centre needed
 - He must disappear quickly (...but not too quick...)
 - Low level of other impurities
- High fusion reactivity :
 - Ensure a good flow of D and T to the plasma center
- Stable plasma:
 - Suppress instabilities

Technological

- Check first wall properties
- Check T breeding techniques

Development needs for ITER / DEMO



Main Differences between ITER and DEMO

ITER	DEMO
Experimental device	Close to commercial plant
400s pulses / long interpulse time	Long pulses / if possible steady state
Many diagnostics	Diagnostics only for operations
Many H&CD systems	Optimized H&CD system
No T breeding required	Self sufficient T breeding
316 SS structure	Reduced activation material
Modest n-fluence, low dpa Low material damage	High n-fluence, high dpa Significant material damage
Nuclear Lab license	Nuclear plant license

Still a big step from ITER to DEMO !

DEMO concepts

- DEMO is currently based on the **tokamak concept**
- The stellarator line requires at least one intermediate step following the W7-X class of device
- Nevertheless, reactor concepts are being developed based on helical devices : e.g. LHD

Two DEMO versions currently analysed

Conservative or “Early” DEMO Construction in “20 years” from now ?

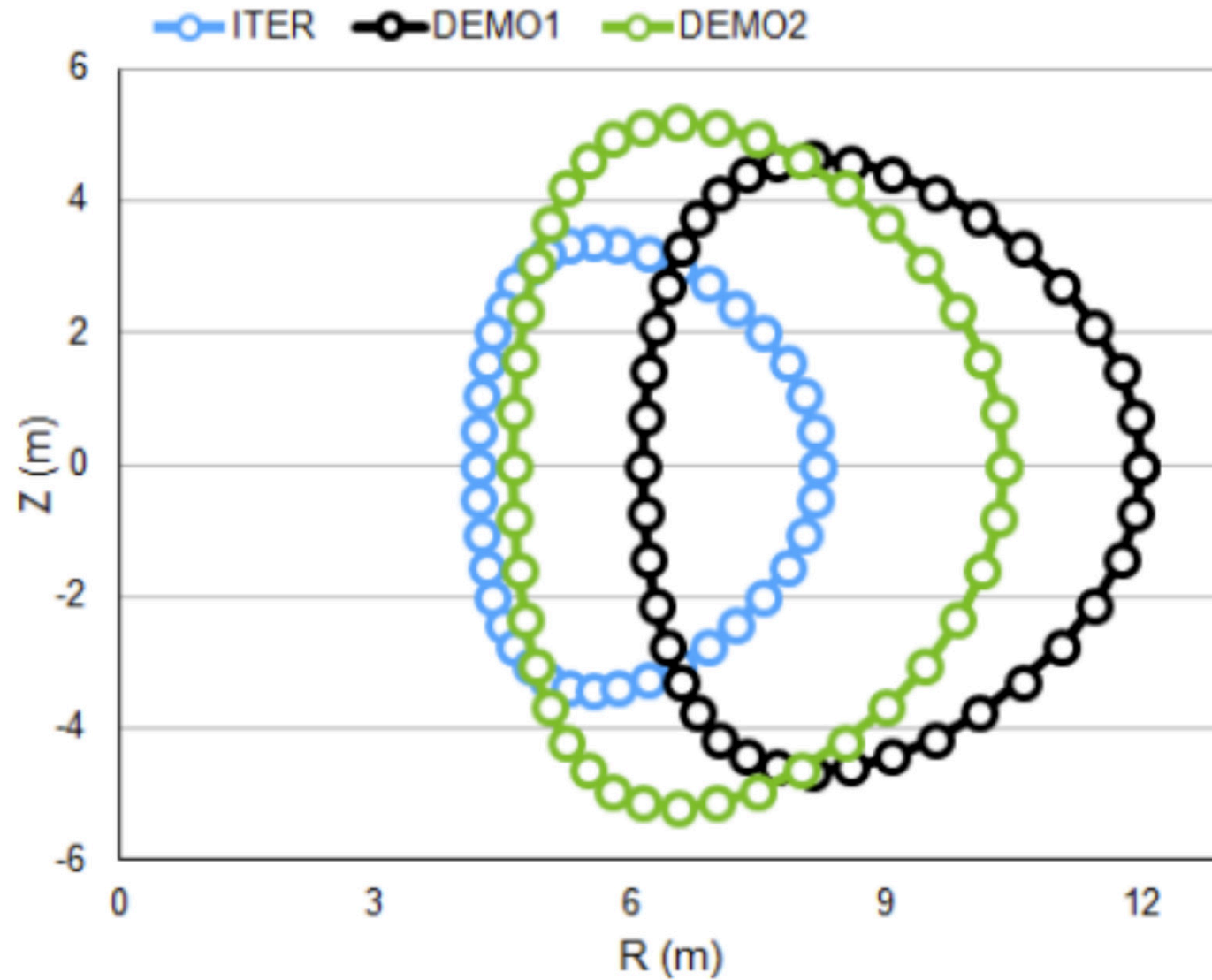
	DEMO 1 (2016)	ITER
Operation Mode	Pulsed	Pulsed
P_{th} (MW)	2200	500
P_{net} (MW)	500	-
R_0 (m)	9.1	6.2
a (m)	2.9	2.0
B_t (T)	5.7	5.3
I_p (MA)	20	15
f_{BS}	35%	-
P_{aux} (MW)	50	70
H_{98}	1.0	1
β_N ($\beta_{N,th}$)	2.6	2

Two DEMO versions currently analysed

Advanced (Optimistic) Design — Construction on a longer term

	DEMO 2 (2016)	ARIES AT (1998)
Operation Mode	Steady State	Steady State
P_{th} (MW)	3255	1900
P_{net} (MW)	950	-
R_0 (m)	7.5	5.2
a (m)	2.9	1.3
B_t (T)	5.6	5.9
I_p (MA)	22	12.8
f_{BS}	61%	91%
P_{aux} (MW)	135	35
H_{98}	1.2	1
β_N ($\beta_{N,th}$)	3.4 (2.8)	5.4

Cross sections DEMO 1&2 / ITER



Ongoing DEMO / ITER Research - 1
Heating Systems
Example of Neutral Beam Injection

Physics of neutral beam injection

- Ion Source produces D , D^+ , D_2^+ , D_3^+
- Neutralisation efficiency for positive ions is very low at high
- Presently in operation: sources of 1-2 MW, up to 150keV (JET) (penetration depth ~ 1 m at high density)

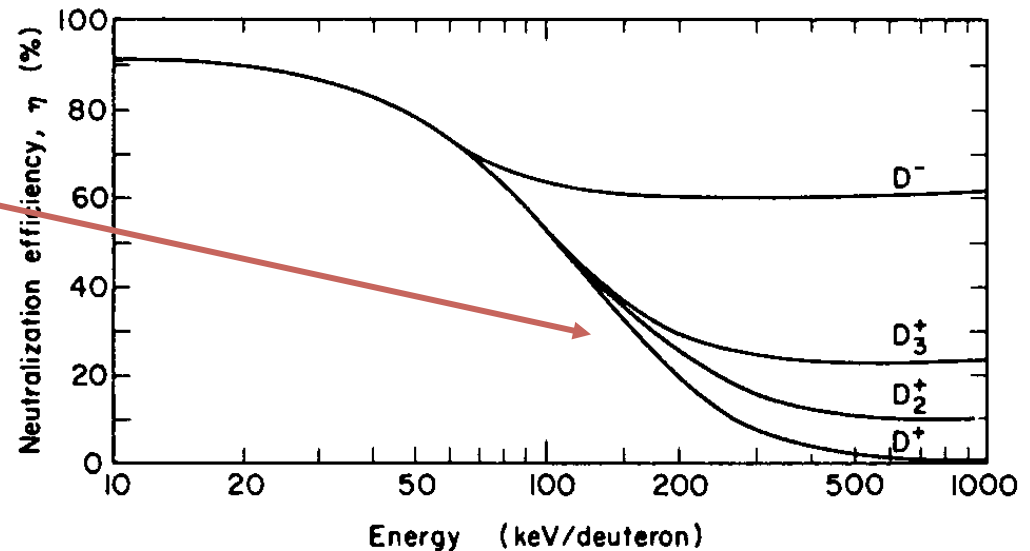


FIG.2. Maximum neutralization efficiency in D_2 vs beam energy, for each of the four beams, D^+ , D_2^+ , D_3^+ , and D^- .

[Berkner K.H., et al., Nuc. Fus. 15(1975)249-154.]

Voltage required for sufficiently deep deposition in ITER

- To heat the plasma: 0.5 MeV
- For current drive: 1. – 2. MeV
- For an efficient neutral beam system: need to neutralize NEGATIVE ions.

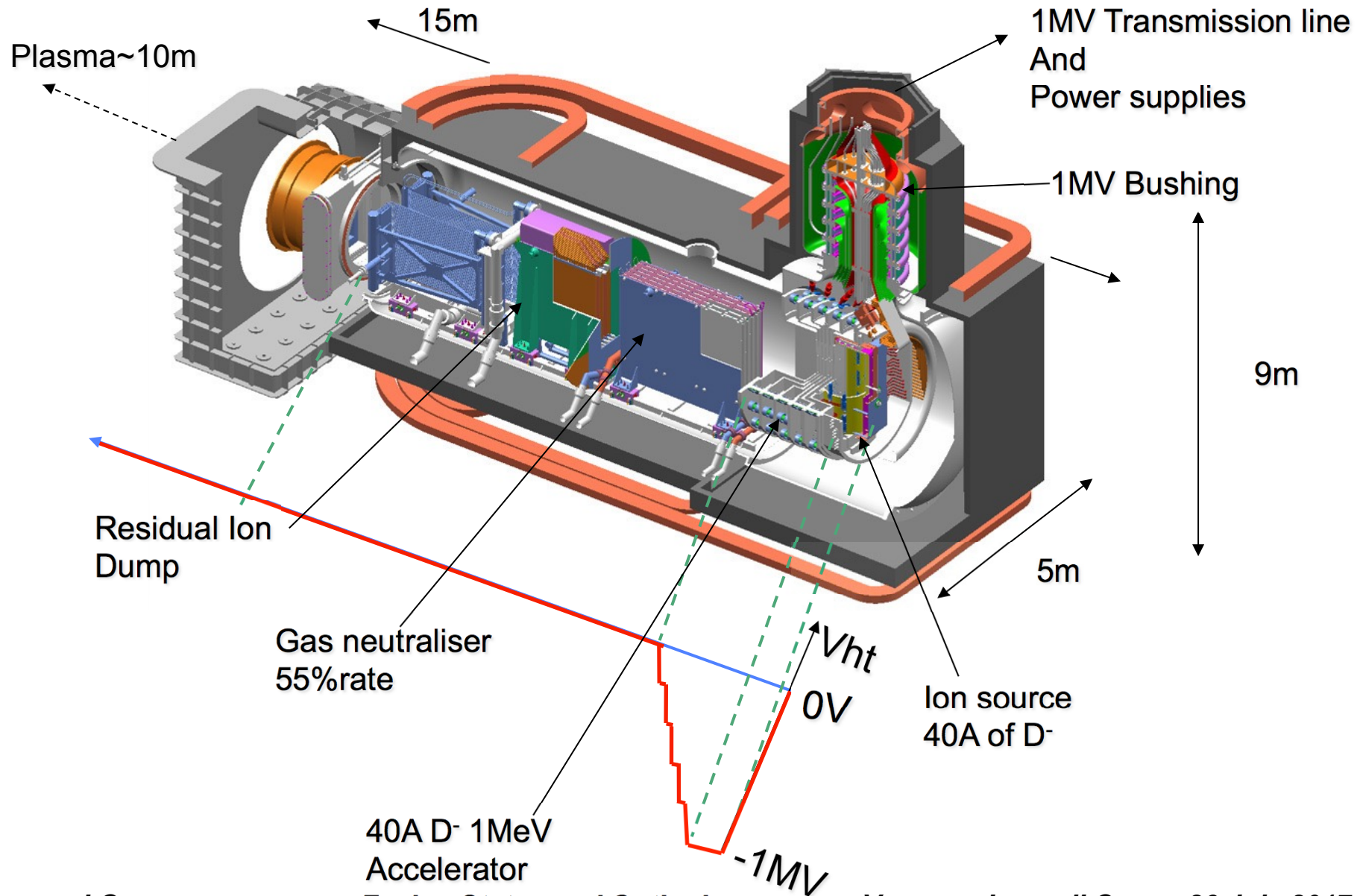
THIS IS A TECHNOLOGY THAT NEEDS A LOT OF DEVELOPMENT

N-NBI Characteristics : planned and achieved

		ITER (rf)	LHD (arc)	JAEA JT60U (arc)		JAEA MV TF (arc)	IPP (rf source)	
Species		D ⁻	H ⁻	D ⁻	H ⁻	H ⁻	H ⁻	D ⁻
Energy	keV	1000	180	400		937		
Voltage holding	kV	1000	190	500		1000		
Accelerated current	A	40	30	17		0.33	1.4	
Extracted current density	A/m ²	285	250			144		280
Pulse length	s	3600	2	2		2	3600	4

NBI developments for ITER (and DEMO)

ITER Neutral Beam Injection line: 17MW of D⁰ at 1MeV

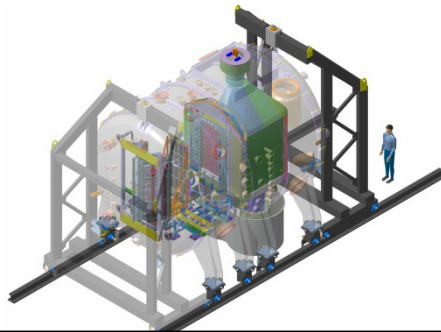


Negative NBI : New lab in construction in Padova

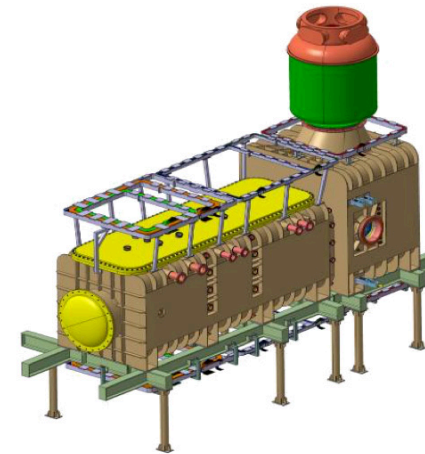
PRIMA Padova Research on ITER Megavolt Accelerator



- Mission of PRIMA SPIDER MITICA:
 - Achieve the ***nominal parameters***
 - ***Optimise*** the NBI operation
 - Maximize the ***reliability*** of the injectors
 - ***Develop technologies*** for the injectors



SPIDER Source for Production of Ion of Deuterium Extracted from Rf plasma



MITICA Megavolt ITER Injector & Concept Advancement

PRIMA lab (SPIDER and MITICA) in Padua (August 2013)



J.Ongena

Fusion Status and Outlook

Varennna, Lago di Como, 26 July 2017

Ongoing DEMO Research - 2

First wall material research

IFMIF – International Fusion Materials Irradiation Facility

Understanding material degradation due to 14.1 MeV n flux

IFMIF

- Neutron source with adequate flux and suitable energy
 - ✓ to simulate the neutronic conditions in a fusion power plant
- 40 MeV D⁺ on 25mm liquid Li sheet (5MW, 125 mA accelerator)

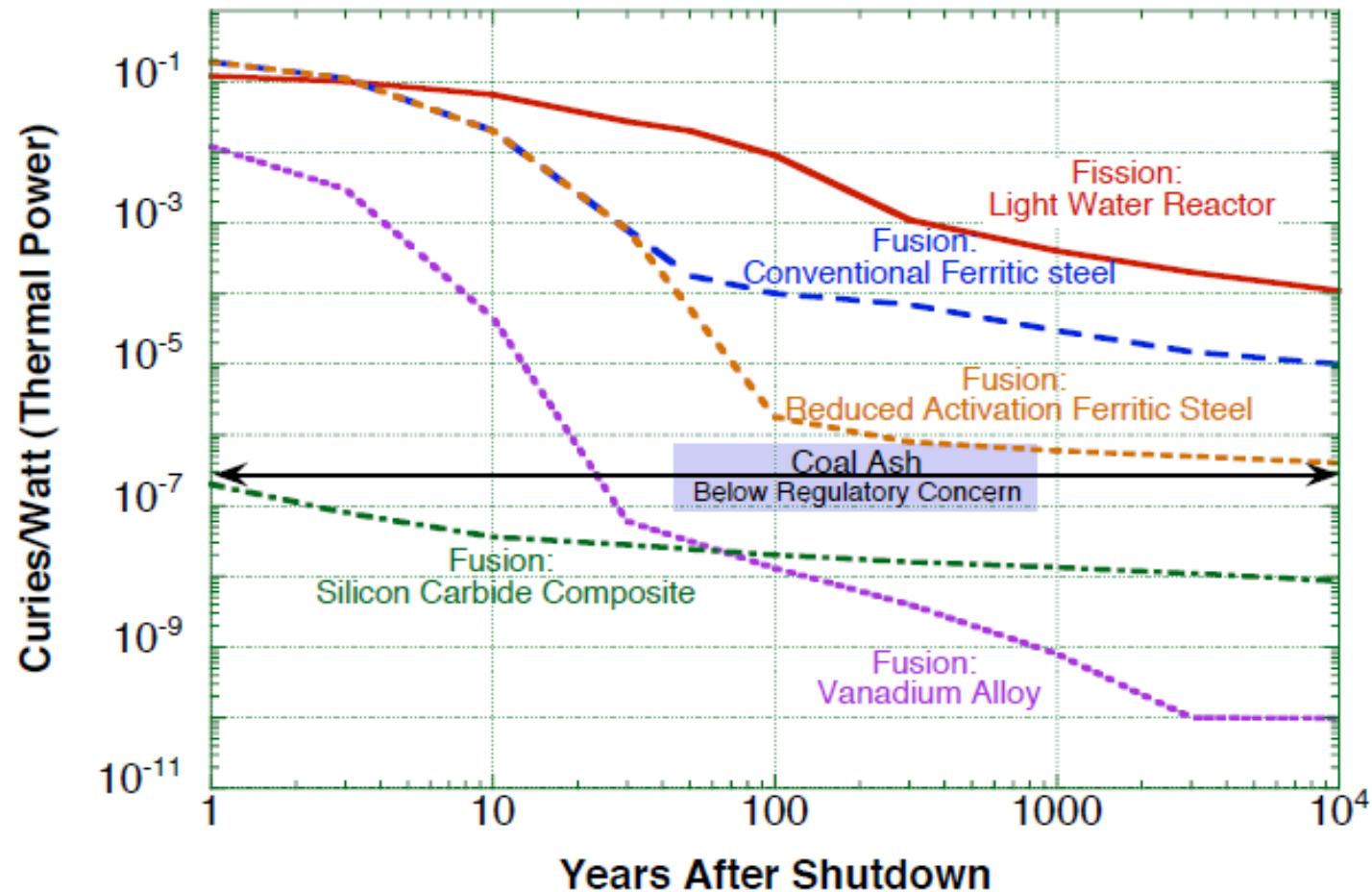
MISSION OF IFMIF

- Qualification of candidate materials
- Engineering data for design, licensing and safe operation of DEMO
- Completion, calibration and validation of databases
- Enlarge fundamental understanding of response of materials interacting with high energy neutrons

DEMO Reduced Activation : Hands on after ~ 100y?

DEMO : Vanadium Alloys or SiC ?

Comparison of Fission and Fusion Radioactivity after Shutdown



Material research for future Fusion Power Plants

- ITER: < 2 dpa (at the end of its operational life)
- Fusion power plant ~150 dpa (within 5 years)
- Transmutation of Fe → creation of p and α-particles in the material matrix:

$^{56}\text{Fe}(n, \alpha)^{53}\text{Cr}$ (incident n threshold at 2.9 MeV)

$^{56}\text{Fe}(n, p)^{56}\text{Mn}$ (incident n threshold at 0.9 MeV)

→ Swelling and embrittlement of materials in fusion reactor

Which Irradiation Facilities to use for Fusion Materials R&D?

Existing neutron sources cannot provide the answers

1. Fission reactors

- average neutron energy ~ 2 MeV
- *No efficient p^+ or α -particle generation*

2. Spallation sources

- wide spectrum with tails \sim hundreds of MeV
- *Neutron energy much too high*

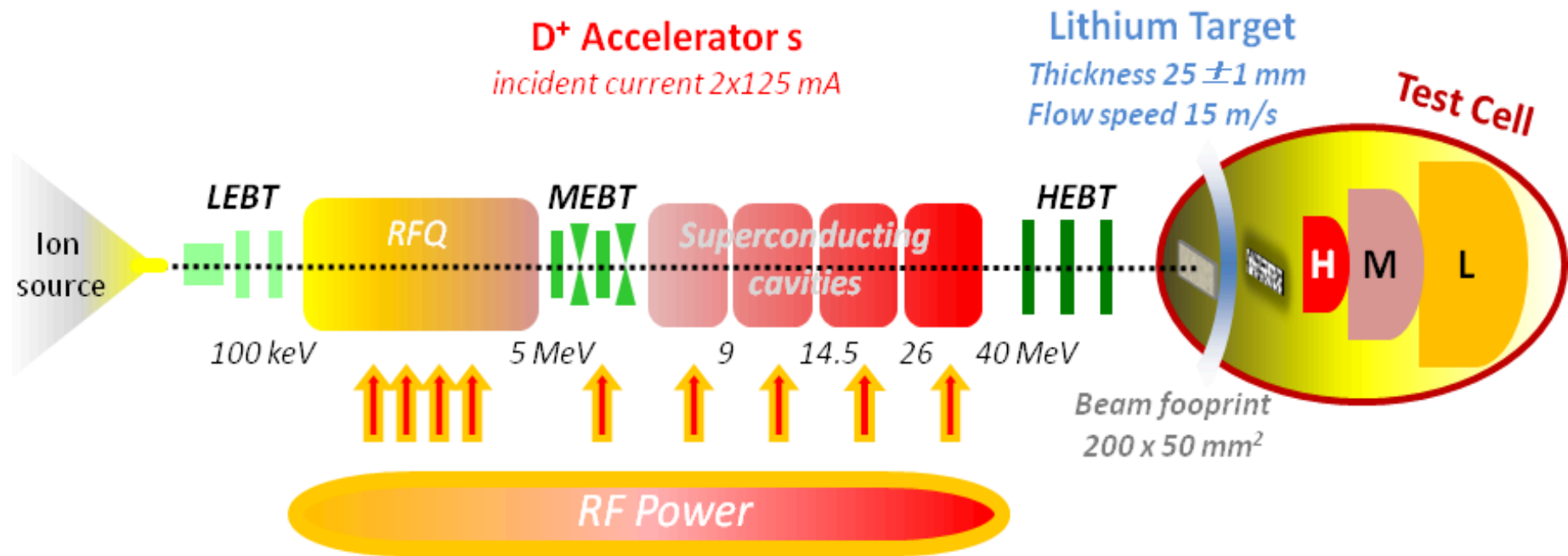
3. Ion implantation facilities

- *Insufficient volume and insufficient displacements per atom (dpa)*

A dedicated facility is needed:

International Fusion Materials Irradiation Facility (IFMIF)

IFMIF: Principle



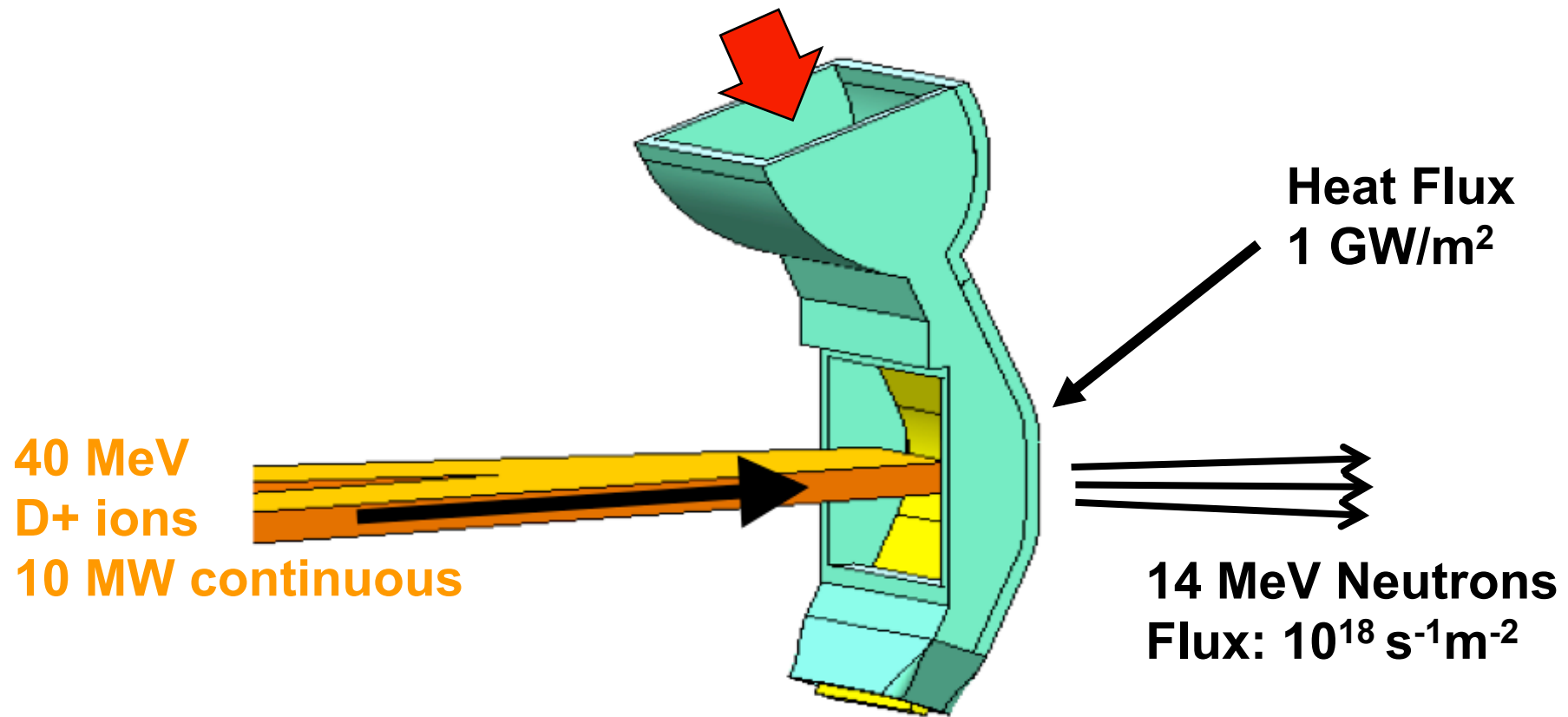
Part 1
Prototype accelerator in construction
In Rokkasho, Japan (LIPAc)

Part 2
Lithium target
tests in
Italy
Japan

Part 3
Test cell tests
In Japan
Germany
Belgium

IFMIF: Principle

Liquid Lithium
(15m/s, 25mm thickness, 250 °C)



IFMIF: a huge challenge

- Accelerator driven source of neutrons
- Neutrons from $^{\text{nat}}\text{Li}(\text{d},\text{xn})$ reactions
 - \searrow $^7\text{Li}(\text{d},\text{n})^8\text{Be}$, $^6\text{Li}(\text{d},\text{n})^7\text{Be}$, $^7\text{Li}(\text{d},\text{n}\alpha\alpha)$,
 $^7\text{Li}(\text{d},\text{np})^7\text{Li}$, $^7\text{Li}(\text{d},\text{nn})^7\text{Be}$, $^7\text{Li}(\text{d},\text{nd})^6\text{Li}$,...
- 2 accelerators 40MeV, 125mA, D^+ ions \rightarrow 2 x 5MW
- 10^{18} neutrons/m²/s with peak at 14 MeV
- Target heat load: 1GW/m² \rightarrow liquid target needed
15m/s, 250 °C, total 10m³ of liquid Li
- Function of liquid Li target:
 - \rightarrow generate sufficiently high neutron flux
 - \rightarrow dissipate 10 MW beam power

Testing the prototype for the Accelerator for IFMIF

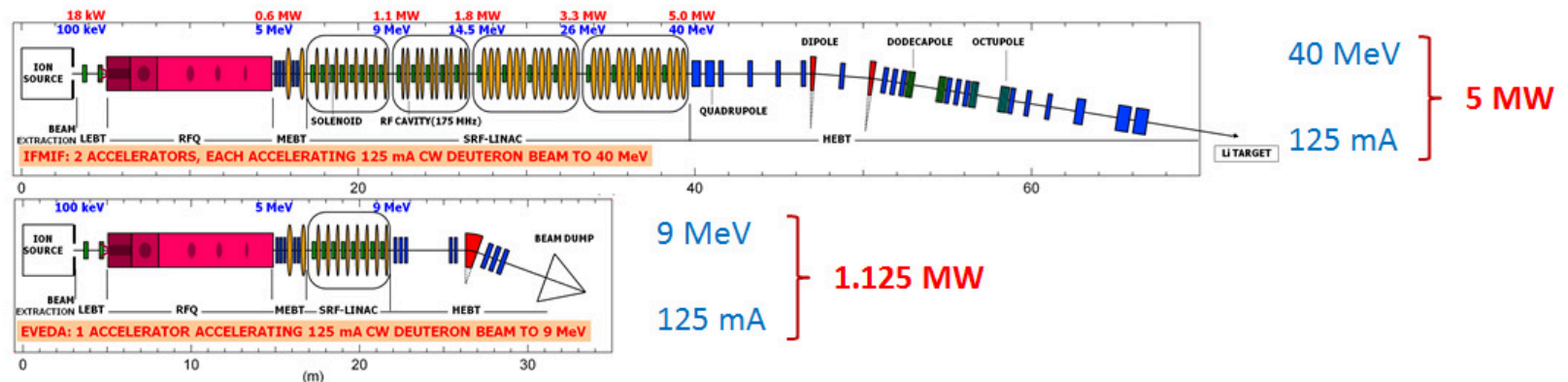
Linear IFMIF Prototype Accelerator : LIPAc

Scaled down version of IFMIF accelerator,
in construction in Rokkasho (Japan)

	IFMIF	EVEDA (LIPAc)
Beam Current	125 mA	125mA
Beam Power	5MW	1.125MW
Beam Energy	40 MeV	9 MeV

IFMIF

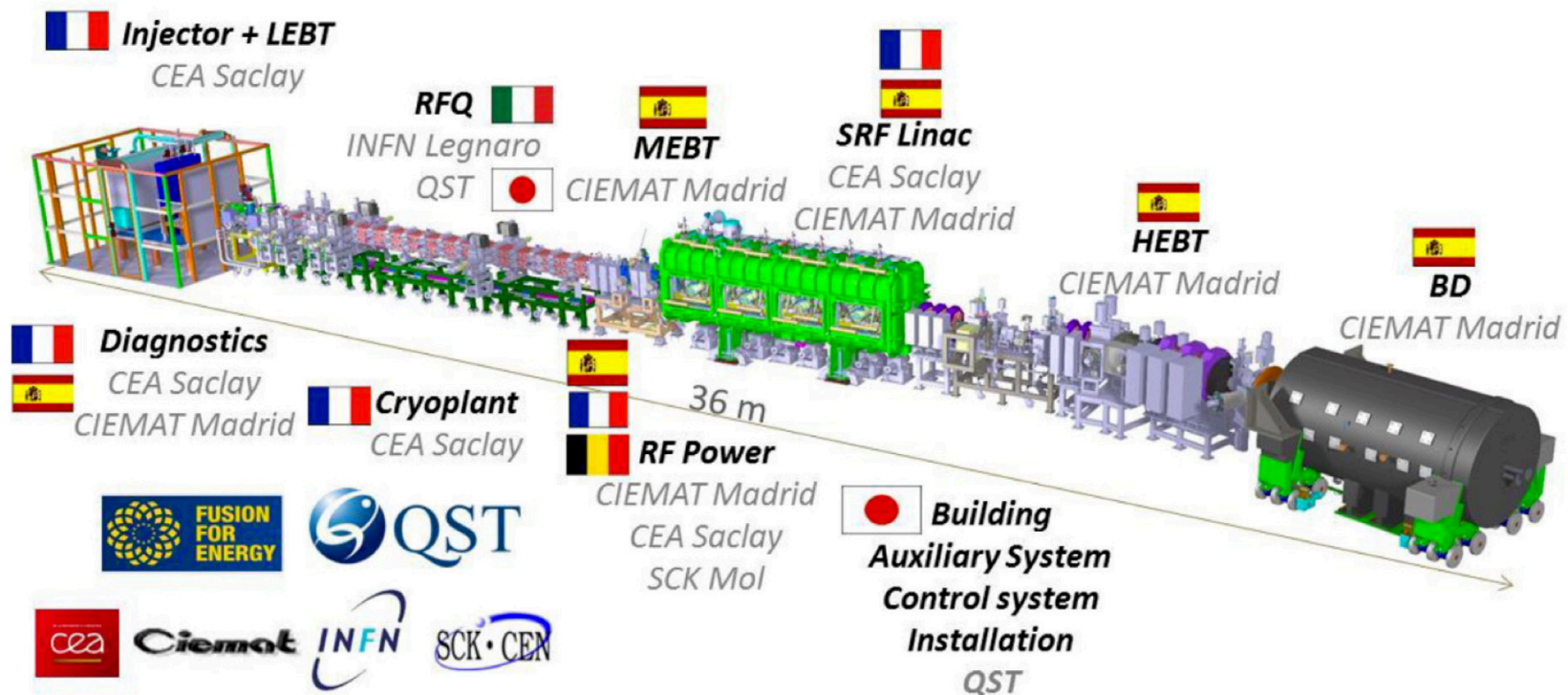
LIPAc



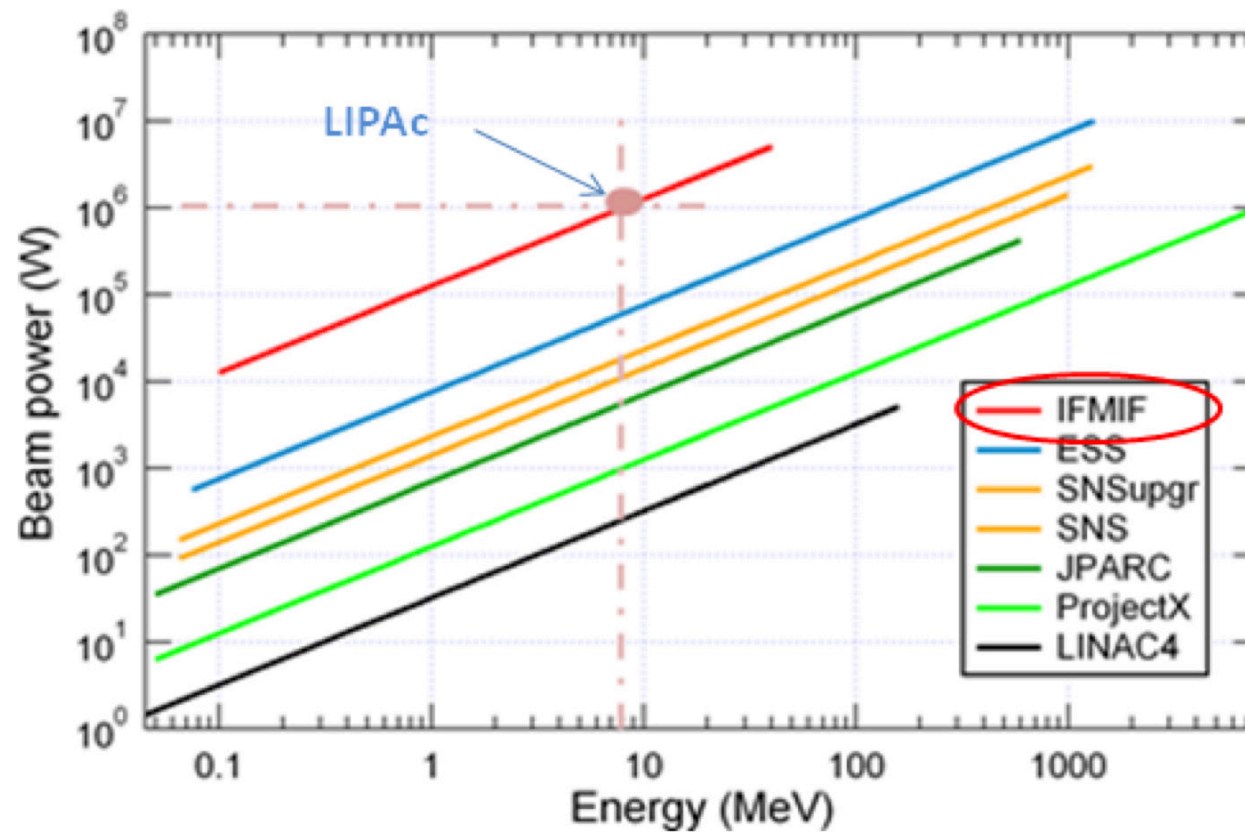
Testing the prototype for the Accelerator for IFMIF

Linear IFMIF Prototype Accelerator : LIPAc

Scaled down version of IFMIF accelerator,
in construction in Rokkasho (Japan)

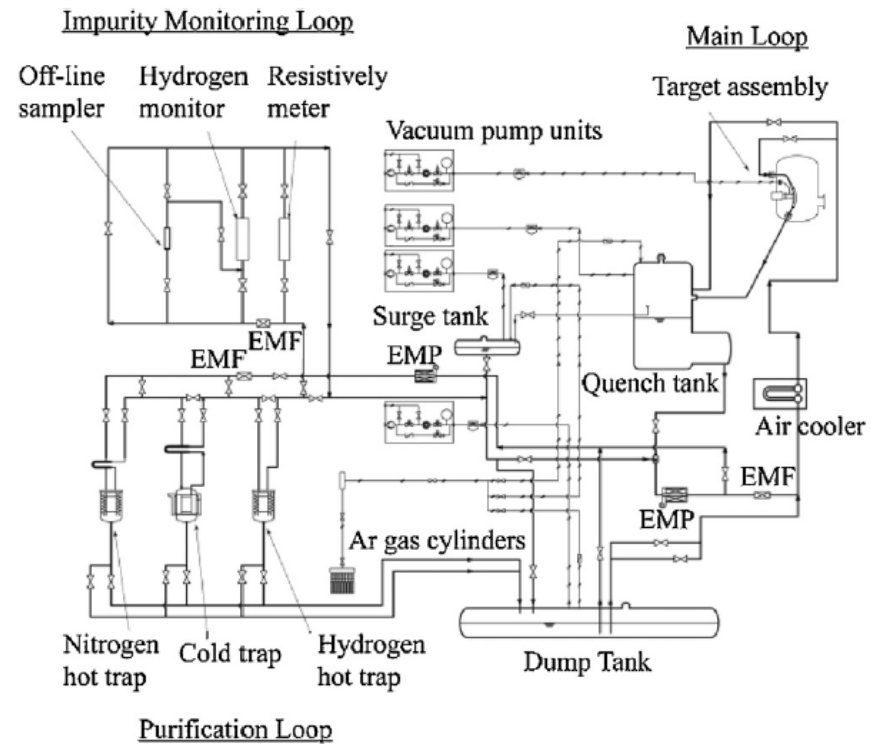


LIPAC (and IFMIF) : powerful accelerators



2. Testing the liquid Lithium loop in Oarai (Japan)

Largest Li loop in the world (2250 liter Li / min)
Successful production of stable 15m/s Li screen (25mm thick)



2. Lithium test loops in Brasimone (Italy) and Osaka (Japan)

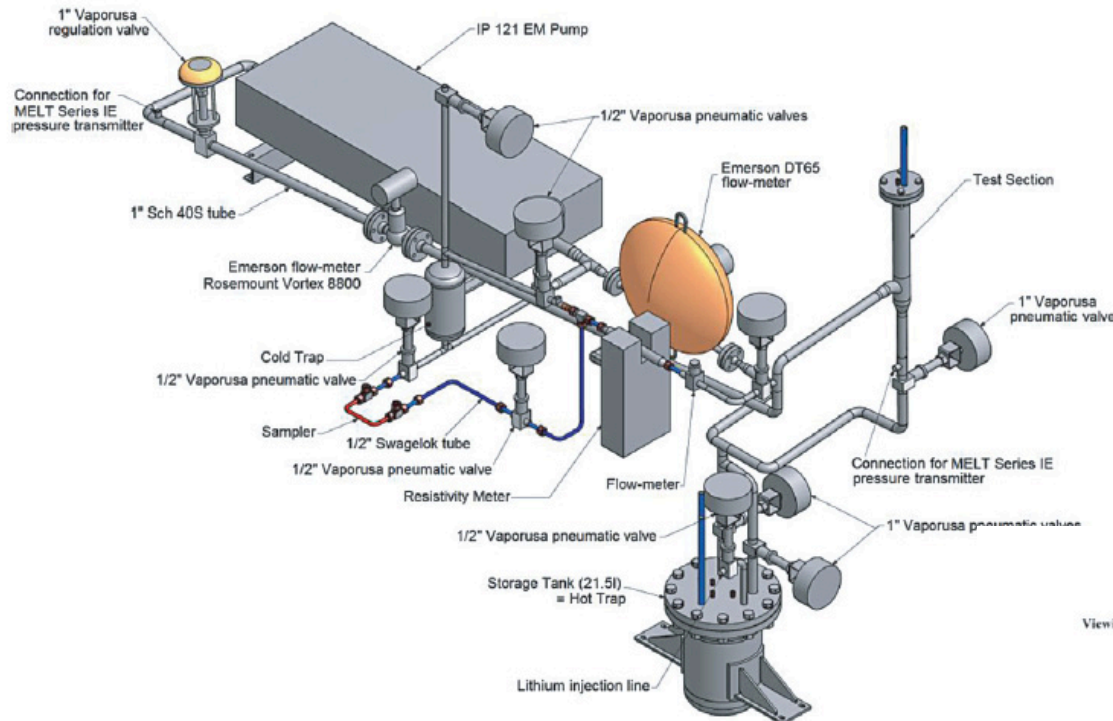


Figure 23. Layout of the LIFUS6 loop in Brasimone (ENEA).

Lifus6 Loop in Brasimone

Addressing erosion/
corrosion phenomena

Surface stability in Osaka loop

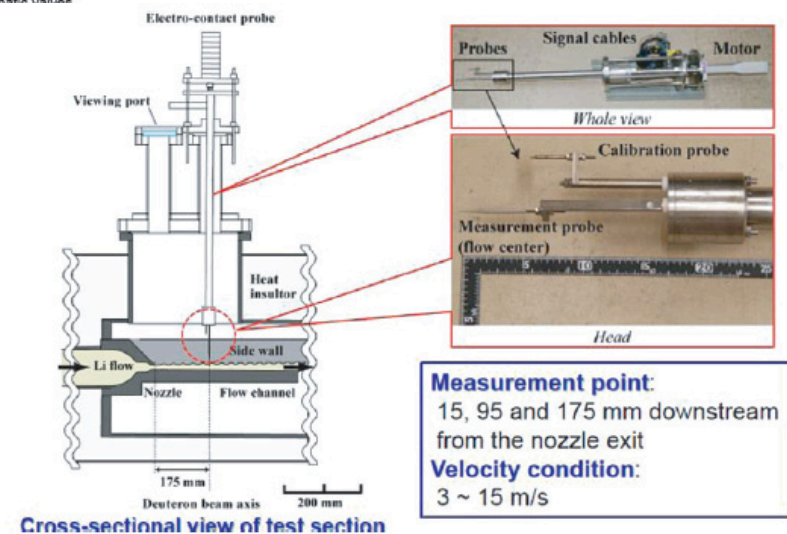
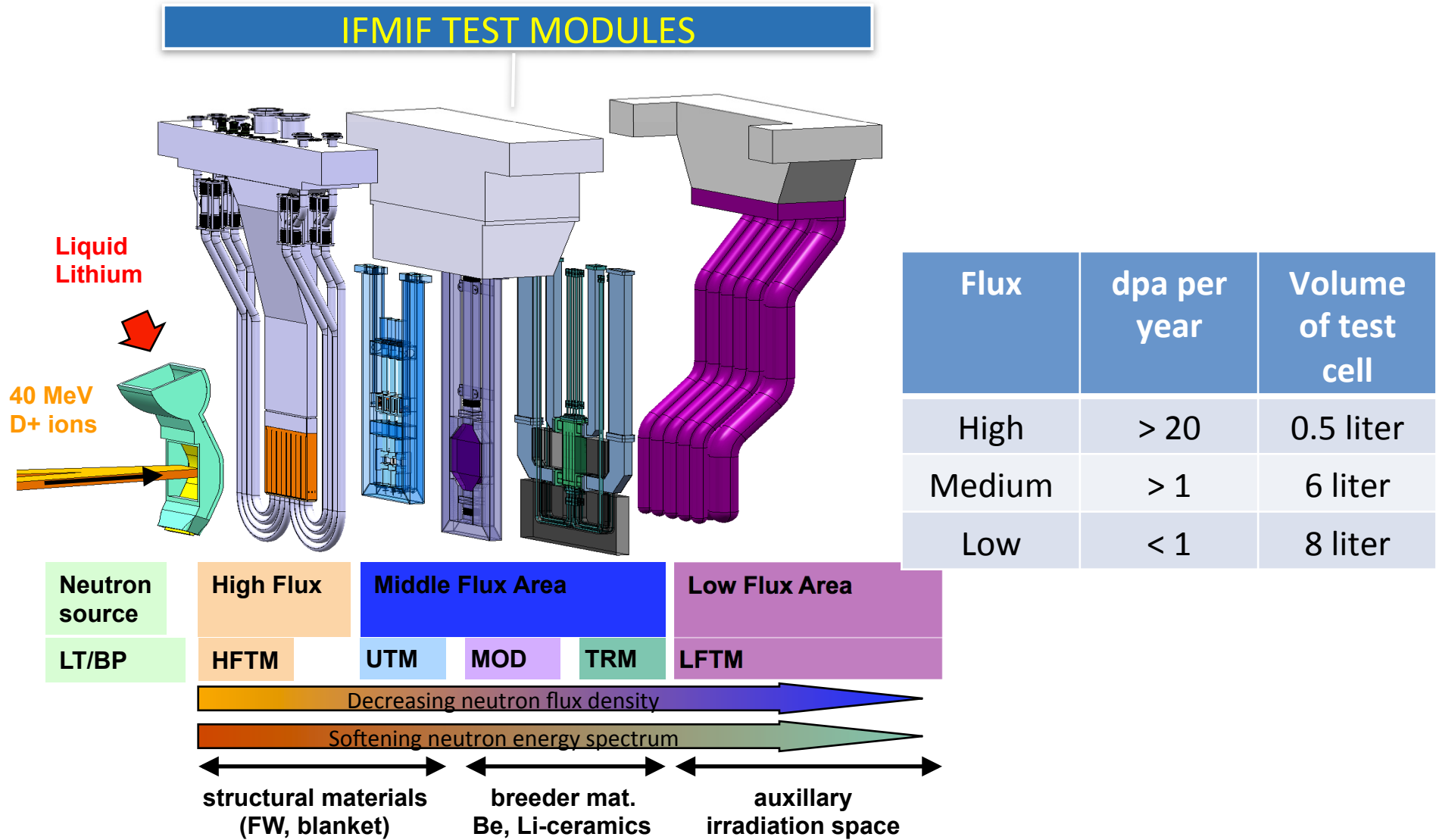


Figure 21. Set up of the contact probe measurement in the Osaka Li loop.

3. Validation of Specimen Test Facility in Karlsruhe (Germany)



All details on IFMIF

IOP PUBLISHING and INTERNATIONAL ATOMIC ENERGY AGENCY
Nucl. Fusion 53 (2013) 116001 (18pp)

NUCLEAR FUSION
doi:10.1088/0029-5515/53/11/116001

SPECIAL TOPIC

IFMIF: overview of the validation activities

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⁵ JAEA, Oarai, Japan

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Abstract

The Engineering Validation and Engineering Design Activities (EVEDA) for the International Fusion Materials Irradiation Facility (IFMIF), an international collaboration under the Broader Approach Agreement between Japan Government and EURATOM, aims at allowing a rapid construction phase of IFMIF in due time with an understanding of the cost involved. The three main facilities of IFMIF (1) the Accelerator Facility, (2) the Target Facility and (3) the Test Facility are the subject of validation activities that include the construction of either full scale prototypes or smartly devised scaled down facilities that will allow a straightforward extrapolation to IFMIF needs. By July 2013, the engineering design activities of IFMIF matured with the delivery of an Intermediate IFMIF Engineering Design Report (IEDR) supported by experimental results. The installation of a Linac of 1.125 MW (125 mA and 9 MeV) of deuterons started in March 2013 in Rokkasho (Japan). The world's largest liquid Li test loop is running in Oarai (Japan) with an ambitious experimental programme for the years ahead. A full scale high flux test module that will house ~1000 small specimens developed jointly in Europe and Japan for the Fusion programme has been constructed by KIT (Karlsruhe) together with its He gas cooling loop. A full scale medium flux test module to carry out on-line creep measurement has been validated by CRPP (Villigen).

(Some figures may appear in colour only in the online journal)

1. Introduction

In DEMO like in future fusion power plants, the deuterium-tritium nuclear fusion reactions will generate a large quantity of 14.1 MeV neutrons that will collide with the materials of the reactor vessel. The first wall, a combination of layers of different materials that aims to maximize the conversion of neutrons into thermal energy and breed tritium will be critically exposed. Understanding the degradation of the mechanical properties throughout the reactor's operational life is a key parameter to allow the design and eventual facility licensing by the corresponding nuclear authorities.

Inelastic collisions of neutrons with the nuclei in the structural materials over the threshold incident energy of around 3 MeV will transmute heavy nuclei, which can decay releasing p^+ - and α -particles. In turn, the elastic collisions are measured by NRT displacements per atom (NRT dpa) [1] with a cross section inversely proportional to the average displacement energy threshold for production of a Frenkel vacancy-interstitial atom defect pair in the material. Not all of the materials will present the same NRT dpa under the same neutron bombardment, neither will all areas inside the reactor vessel undergo the same flux and spectrum of neutrons. In addition, NRT dpa do not take into account the time-evolution

0029-5515/13/116001+18\$33.00

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“IFMIF: Overview of the validation Activities”

Juan Knaster et al.,
Nuclear Fusion 53 (2013) 116001

Latest news on EVEDA phase

Engineering Validation and Engineering Design Activities For IFMIF

OPEN ACCESS

IOP Publishing | International Atomic Energy Agency

Nuclear Fusion

Nucl. Fusion **57** (2017) 102016 (25pp)

<https://doi.org/10.1088/1741-4326/aa6a6a>

Overview of the IFMIF/EVEDA project^a

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F. Arbeiter⁶, P. Cara⁷, S. Chel⁸, A. Facco⁹, P. Favuzza¹⁰, T. Furukawa¹¹,
R. Heidinger⁷, A. Ibarra¹², T. Kanemura^{5,b}, A. Kasugai⁵, H. Kondo⁵,
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T. Yokomine¹⁵, E. Wakai¹⁶ and the IFMIF/EVEDA Integrated Project Team

Juan Knaster et al., Nuclear Fusion 57 (2017) 102016

LIPAc – Nov 2016

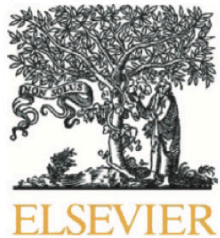


Ongoing DEMO Research - 3

Fusion Fuel from seawater

Latest news on EVEDA phase

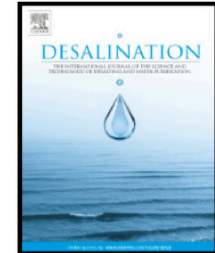
Desalination 359 (2015) 59–63



Contents lists available at ScienceDirect

Desalination

journal homepage: www.elsevier.com/locate/desal



Innovative lithium recovery technique from seawater by using world-first dialysis with a lithium ionic superconductor



Tsuyoshi Hoshino

Breeding Functional Materials Development Group, Department of Blanket Systems Research, Rokkasho Fusion Institute, Sector of Fusion Research and Development, Japan Atomic Energy Agency, 2-166 Obuchi, Omotedate, Rokkasho-mura, Kamikita-gun, Aomori 039-3212, Japan

T.Hoshino, Desalination 359 (2015) 59-63

Ongoing DEMO Research - 4 Tritium Blankets

(Acknowledgement : Nicolas Bekris)

Tritium Consumption in a future fusion reactor

Tritium CONSUMPTION of a
2700 MW Fusion, ~1000 MW electrical Power Plant :

Note: $1\text{eV} = \frac{1.602 \cdot 10^{-19} \text{ As} \cdot \text{V}}{\text{charge}} = 1.602 \cdot 10^{-19} \text{ Joule}$

1.) Energy per fused tritium atom (17.6 MeV in Joule):
 $17.6 \cdot 10^6 \cdot 1.602 \cdot 10^{-19} = 2.82 \cdot 10^{-12} \text{ Joule};$

2.) Fusion frequency = $P/E = 2700 \cdot 10^6 \text{ J/s} / 2.82 \cdot 10^{-12} \text{ J} = 9.57 \cdot 10^{20} \text{ 1/s};$

3.) Tritium mass flow = $3 \cdot \text{mass of proton (neutron)} \cdot \text{frequency} =$
 $3 \cdot 1.67 \cdot 10^{-27} \text{ kg} \cdot 9.57 \cdot 10^{20} \cdot 24 \cdot 60 \cdot 60 \cdot 1/\text{day} = 0.41 \text{ kg/day}$

Tritium T ~ 0.41 kg/day ← breeding in the reactor itself
with Lithium – neutron reaction

Deuterium D ~ 0.27 kg/day ← from sea water

Current world production of Tritium: 1 kg/year (!)

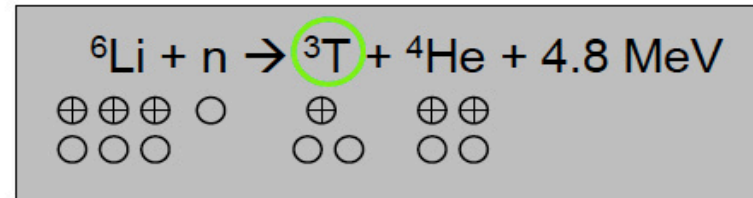
Tritium Consumption in a future fusion reactor

Needed resources for a D-T-Fusion Power Plant,
~1000 MW electrical :

Deuterium D ~ 100 kg/a → in $5 \cdot 10^{16}$ kg Oceans
Sufficient for 30 billion years !!

Tritium T: 55 kg/y

breeding with Lithium reaction →
Only 300 kg Li6 needed per year



About 10^{11} kg Lithium in landmass
Sufficient for 30'000 years

About 10^{14} kg Lithium in oceans
Sufficient for 30 million years !!

ITER will consume most of the world T supply

Consumption:
55.6 kg per 1000 MW fusion/yr

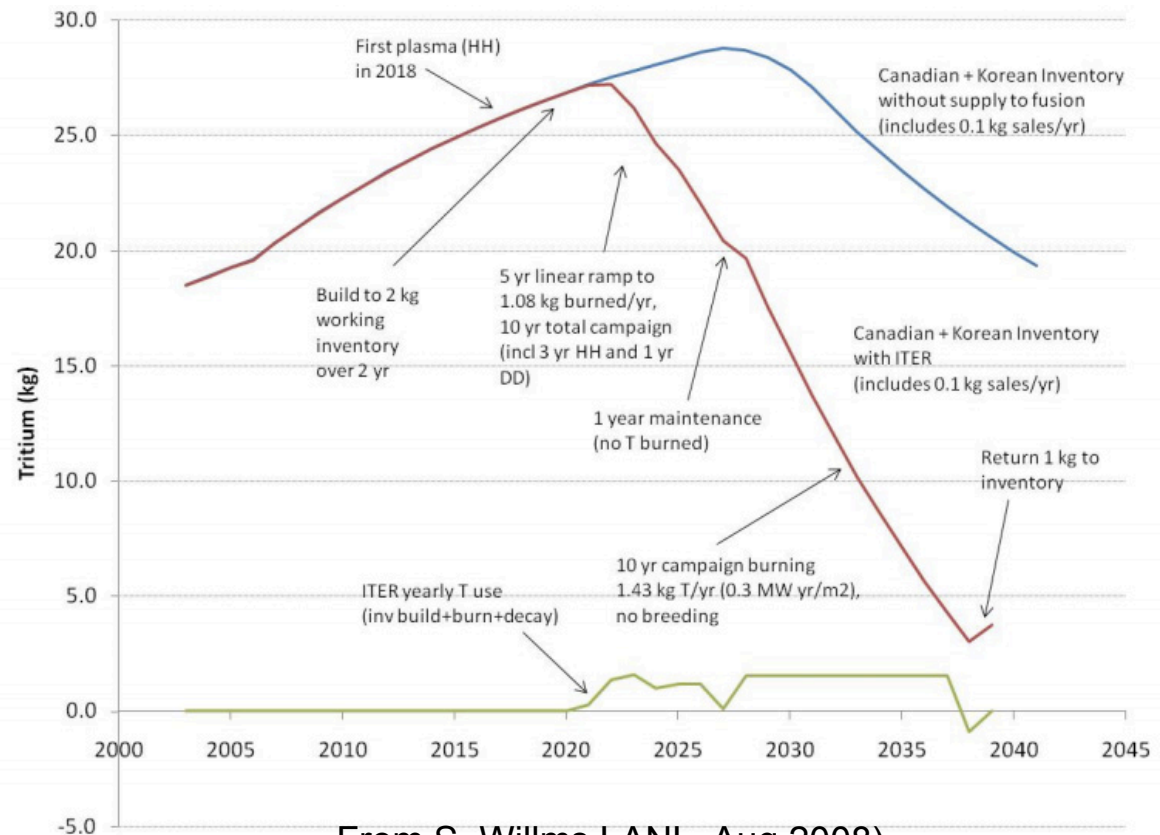
Production from fission:
2-3 kg/yr @\$84-130M/kg
CANDU reactor: 27 kg from over 40 years @\$30M/kg (current)

Tritium decays @ 5.4% per yr.

A successful ITER will exhaust most of the world supply of tritium

FDF has to breed all of its own tritium consumption and with the goal of providing start up tritium for DEMO

(From M. Abdou, UCLA, August 2008)

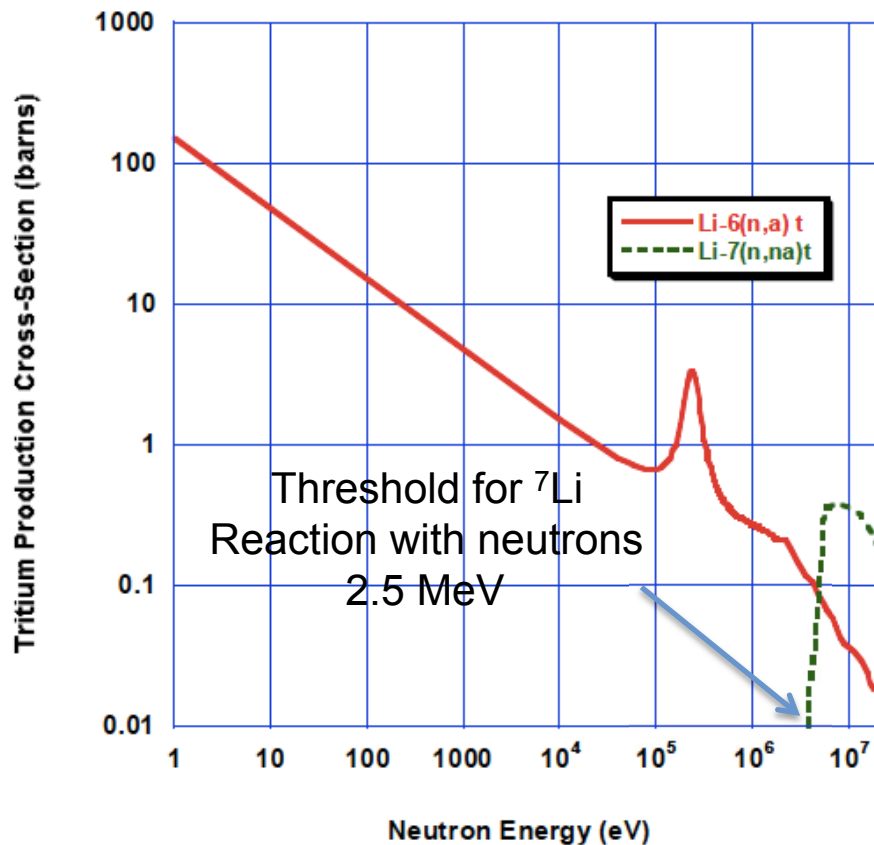


From S. Willms LANL, Aug 2008)

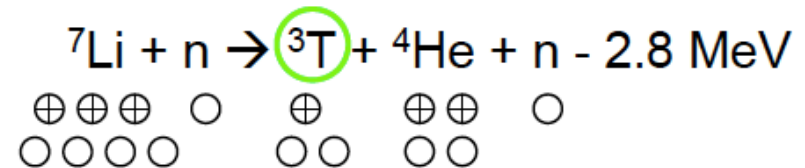
Fusion Facility must have a Tritium Breeding Ratio > 1

Physics of breeding T from Li

Li-6(n,alpha)t and Li-7(n,n,alpha)t Cross-Section

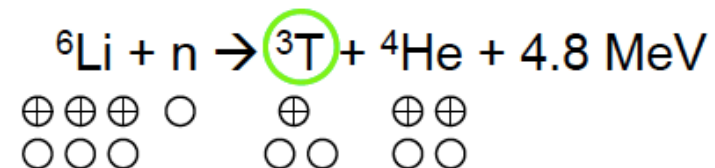


The ⁷Li reaction works with the high energy neutrons



The decelerated neutron can still make a Tritium atom by the ⁶Li reaction.

“Two Tritium atoms with one neutron”



Natural mixture:
92.5 % ⁷Li, only 7.5 % ⁶Li

⊕ Proton
○ Neutron

Because ⁶Li has no threshold, to increase the tritium production we need to enrich the Li compounds with ⁶Li.

Breeding Blanket Materials

Why don't we just use liquid Lithium?

(would allow for sufficient breeding)

- 1) Lithium is dangerous, it burns and ignites at air, strong reaction with water.
- 2) High solubility of Tritium → hard to extract

How else could we design Lithium blankets?

- A) Increase the number of neutrons
(using neutron multipliers)
- B) Use Lithium 6 enrichment in combination with high number
of slower neutrons

Breeding Blanket Materials

How can we use Lithium in blankets?

A) Ceramic materials (Oxides, used in solid state)
e.g. LiO_2 , LiAlO_2 , Li_2SiO_3 , Li_4SiO_4 , Li_8ZrO_6 , Li_2TiO_3

→ Tritium is collected by a Helium purge gas

B) Eutectic alloy (liquid state use) $\text{Li}_{17}\text{Pb}_{83}$ (17 atom% Lithium)

→ Tritium is extracted from the liquid metal outside the vacuum vessel

Ceramic Breeding Blanket Design

Blanket has to optimized: That neutrons from the plasma

- 1.) are causing neutron multiplication with Be and
- 2.) are absorbed in ^6Li

- Avoid inelastic scattering with and absorption in iron
- Small amount of steel structure, especially thin first wall
- Enrichment with ^6Li (e.g. 30%), especially where slow neutrons are present

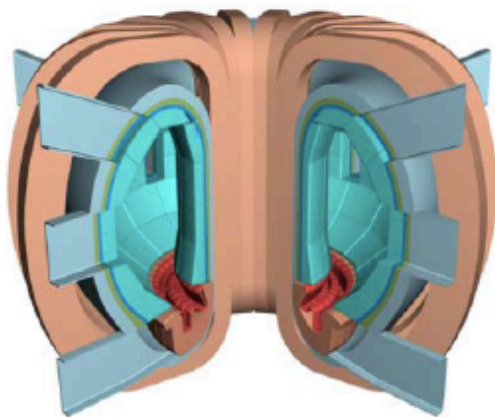


But safety and from this strength of the blanket has to be optimized, too

Breeding Blanket Modules for test in ITER

- **Helium-Cooled Ceramic Breeder (HCCB)** concepts using Ferritic/Martensitic Steel (FMS) structures, Be-multiplier, and Li_2TiO_3 or Li_4SiO_4 or Li_2O ceramic breeder: *China, EU, Japan, RF, Korea, USA*
- **Water-Cooled Ceramic Breeder (WCCB)** concept using FMS structures, Be-multiplier, and Li_2TiO_3 or other ceramic breeder: *Japan*

**SOLID
Breeder**



**LIQUID
Breeder**

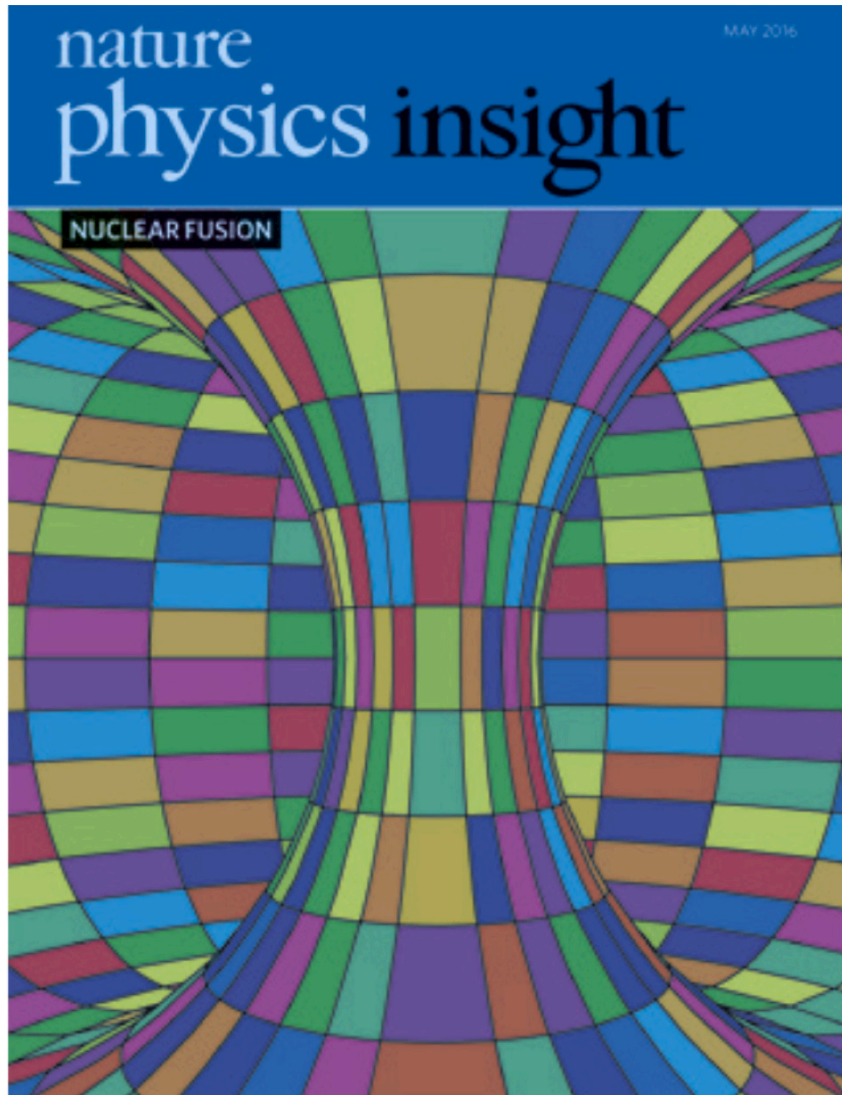
- **Helium-Cooled Lithium-Lead (HCLL)** concepts using liquid eutectic Pb-17Li, FMS structures: *EU, China*
- **Dual Coolant "He/Pb17Li" (DCLL)** concepts using liquid eutectic Pb-17Li, FMS structures: *US, China*
- **Dual Coolant "He/Molten Salt" (DCMS)** concepts using liquid FLiBe or FLiNaBe, FMS structures: *USA, Japan*
- **Self-Cooled liquid Lithium (SCL)** concept using Be-multiplier & Vanadium Alloy structures: *RF, Japan*
- **Helium-Cooled liquid Lithium (HCLi)** concept using FMS structures: *Korea*

Fusion: a necessary option for the future



Further Info

Latest status in fusion research



Nature Physics,
May 2016

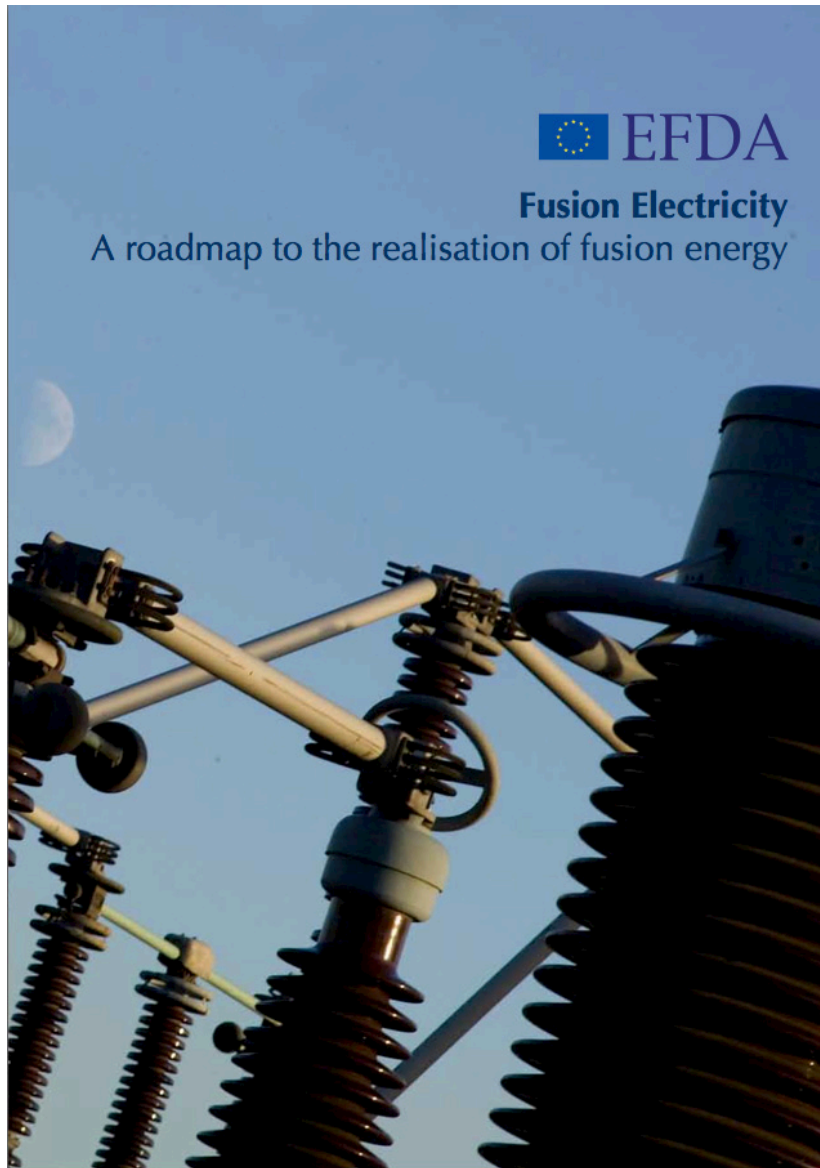
“Insight Section”
On nuclear fusion

66 pages of
last minute info on:

- Magnetic fusion
- Inertial fusion
- Fusion materials research
- Computational advances

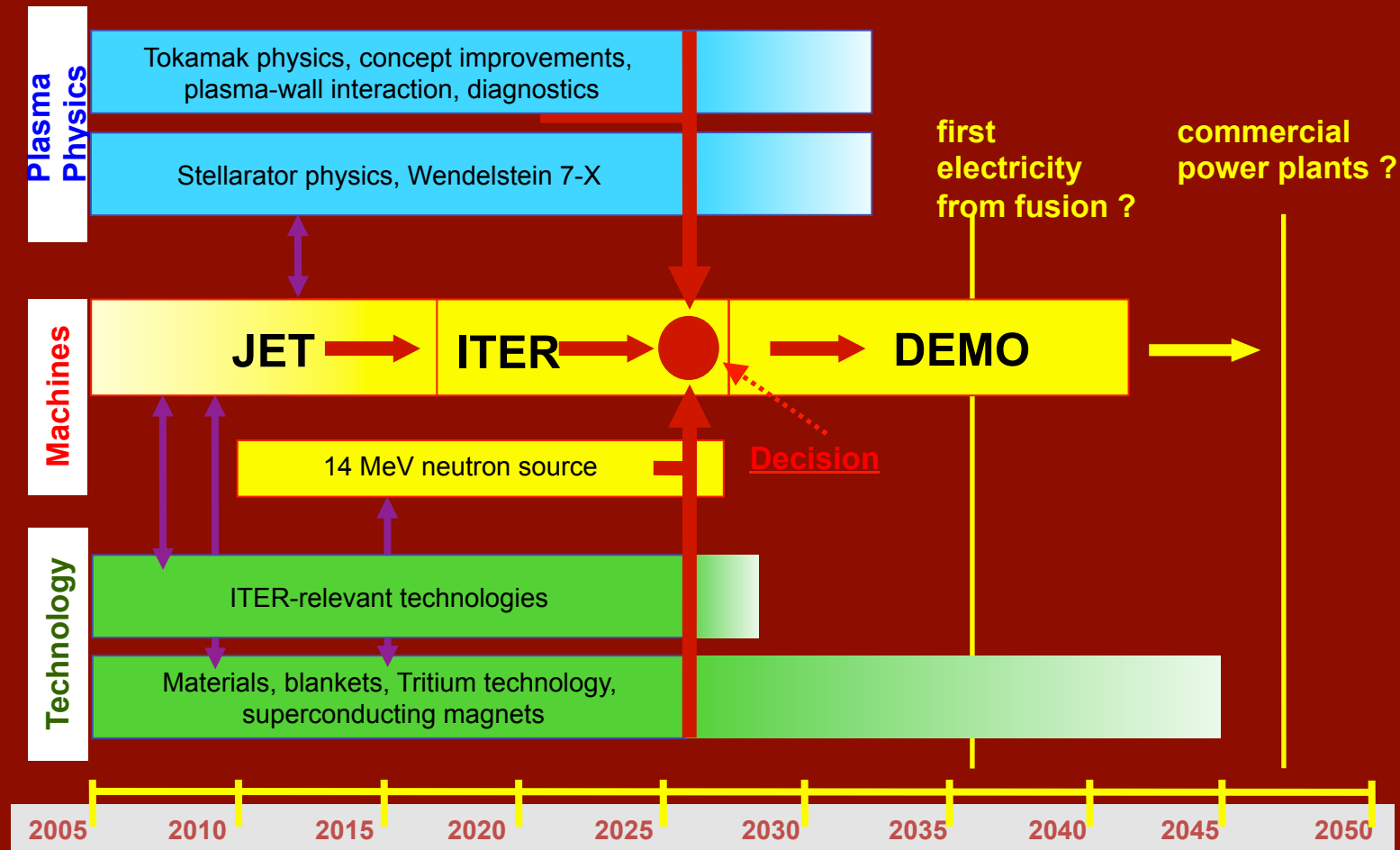
<http://www.nature.com/nphys/journal/v12/n5/index.html>

Roadmap for the realization of fusion energy – June 2013

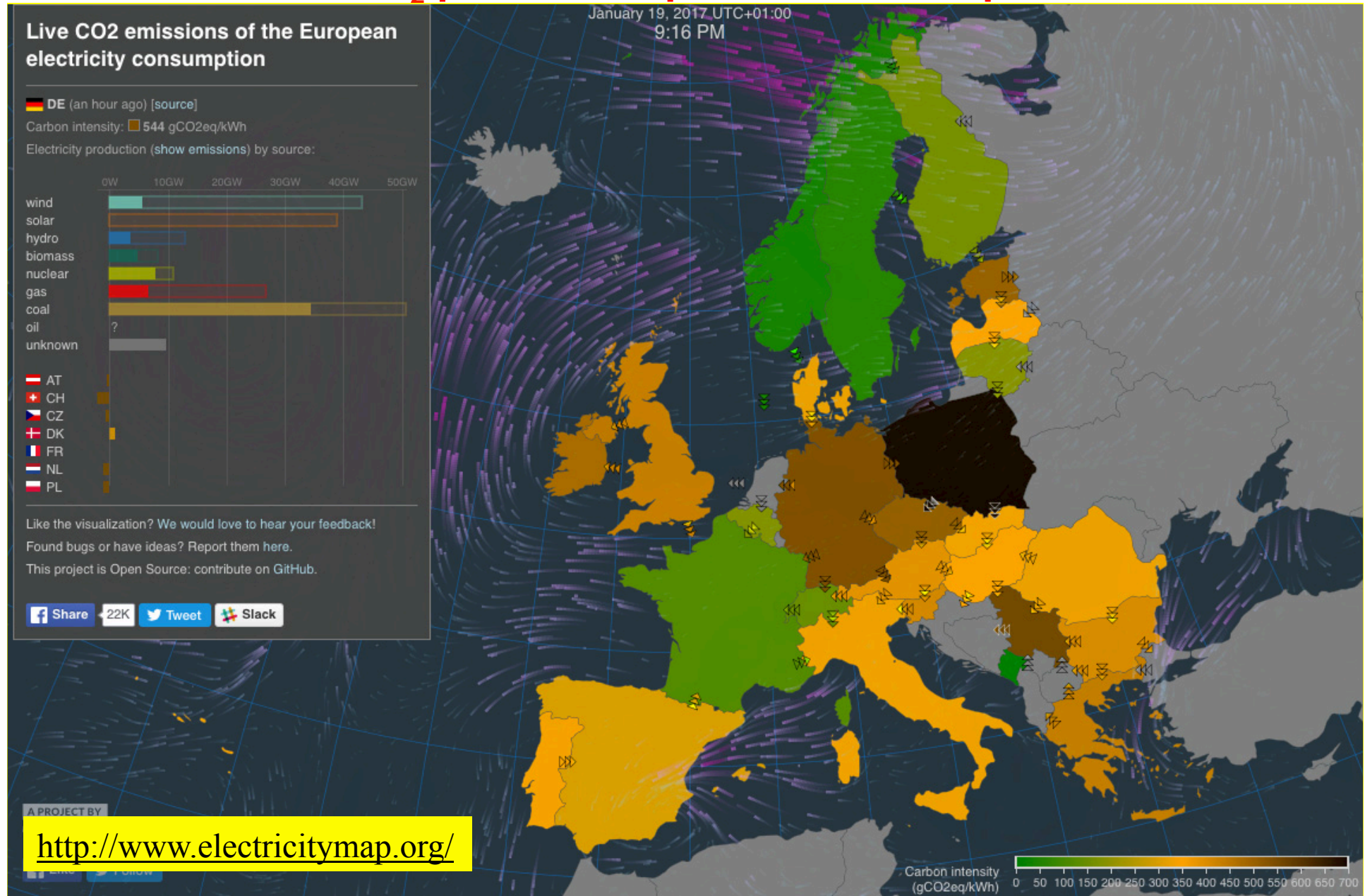


- ITER is the key facility
- DEMO – single step between ITER and commercial fusion power plant
- Need for IFMIF highlighted (14 MeV neutron source)
→ material qualification

'Roadmap' for Magnetic Fusion Research



Grammes of CO₂ production per kWh in European countries



Come and join fusion research !



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European Master of Science in
Nuclear Fusion and Engineering Physics

<http://www.em-master-fusion.org/#!program/c6xc>

