Superconducting Magnets for Nuclear Fusion

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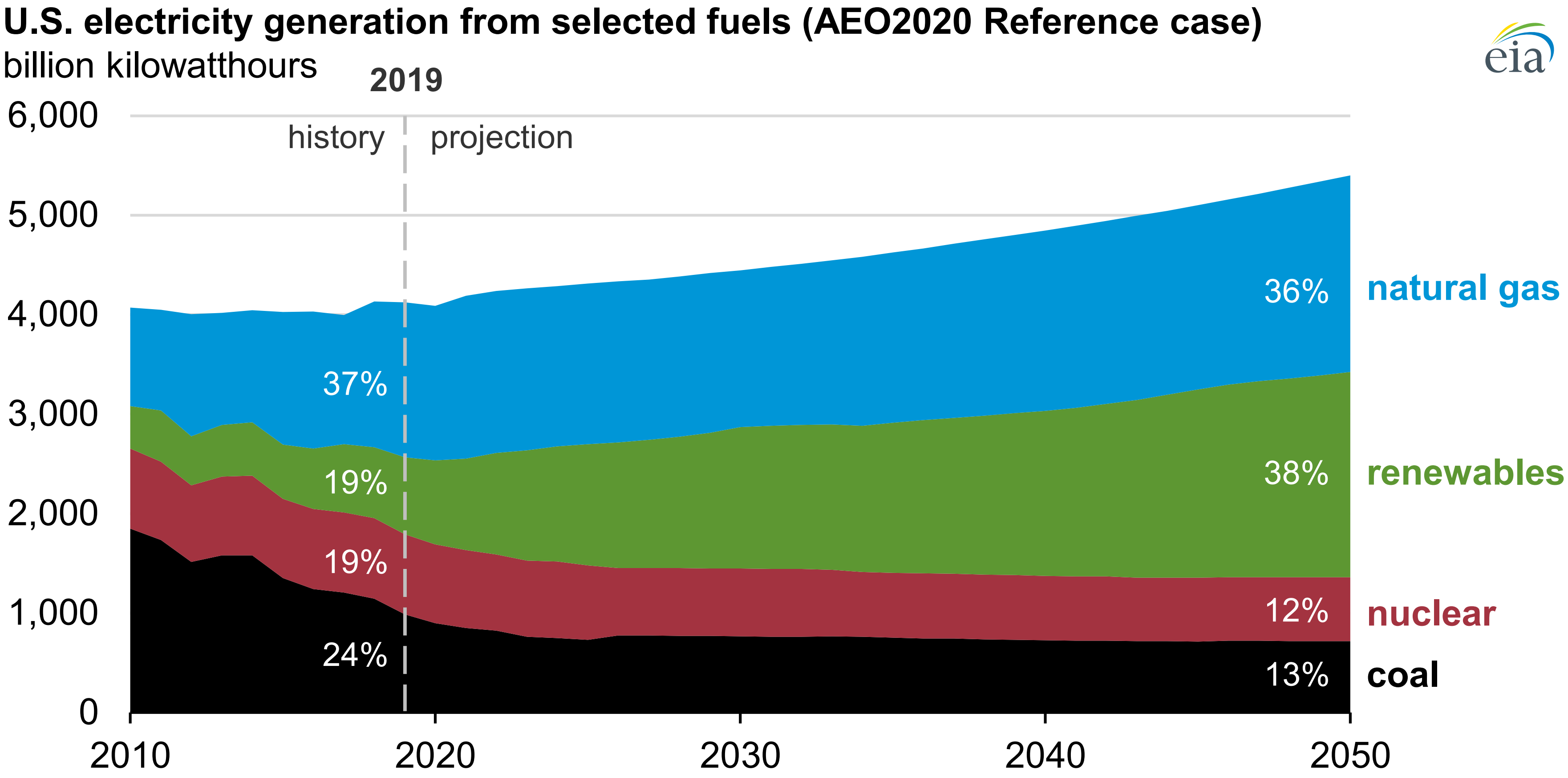
Climate Change and New Energy Source

The potential future effects of global climate change include more frequent **wildfires**, longer periods of **drought** in some regions, and an increase in the duration and intensity of **tropical storms**.
Credit: left - Mellimage/Shutterstock.com, center - Montree Hanlue/Shutterstock.com, right - NASA..

Climate Change and New Energy Source

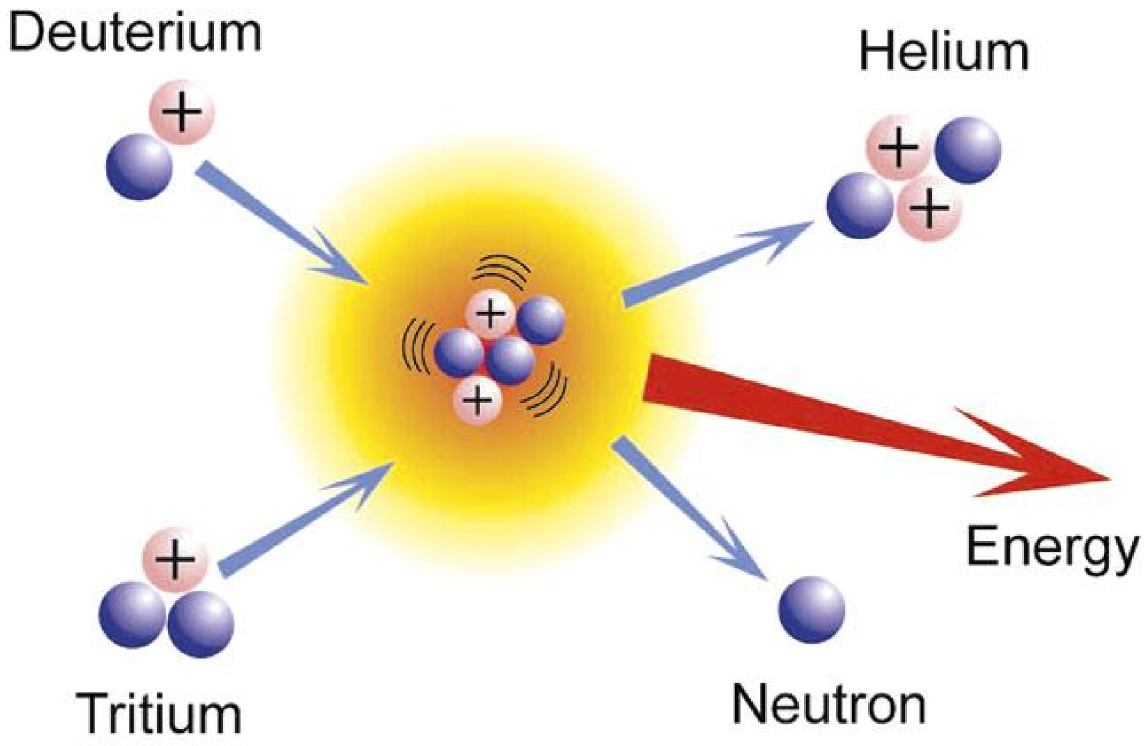


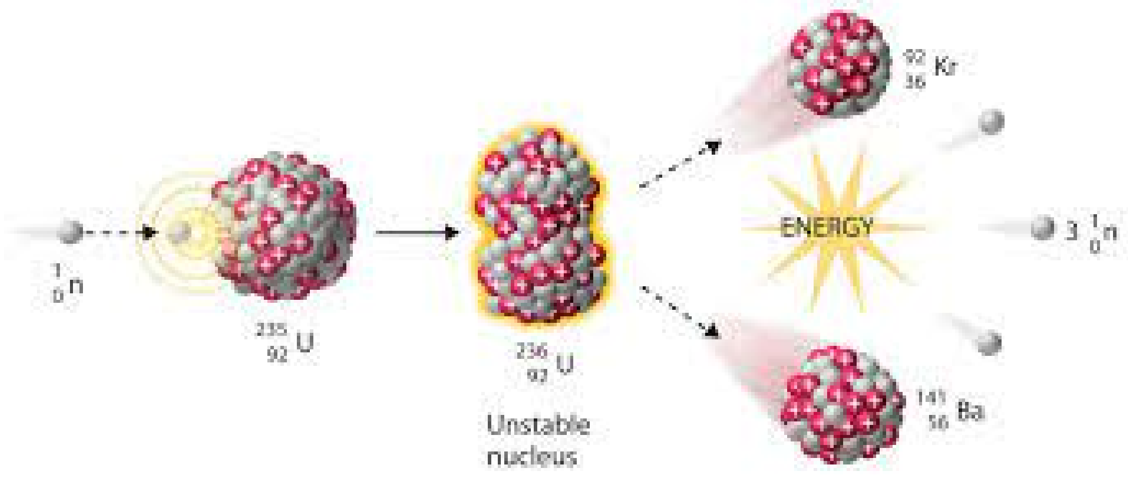
We need energy sources such as

- No emission of greenhouse gases
- Safe for environment and human
- Abundant fuel supply
- Capability for the base energy

Fusion and Fission Energy

· Fusion is the energy source of the Sun
· Reaction which reconstructs of nuclei towards stability

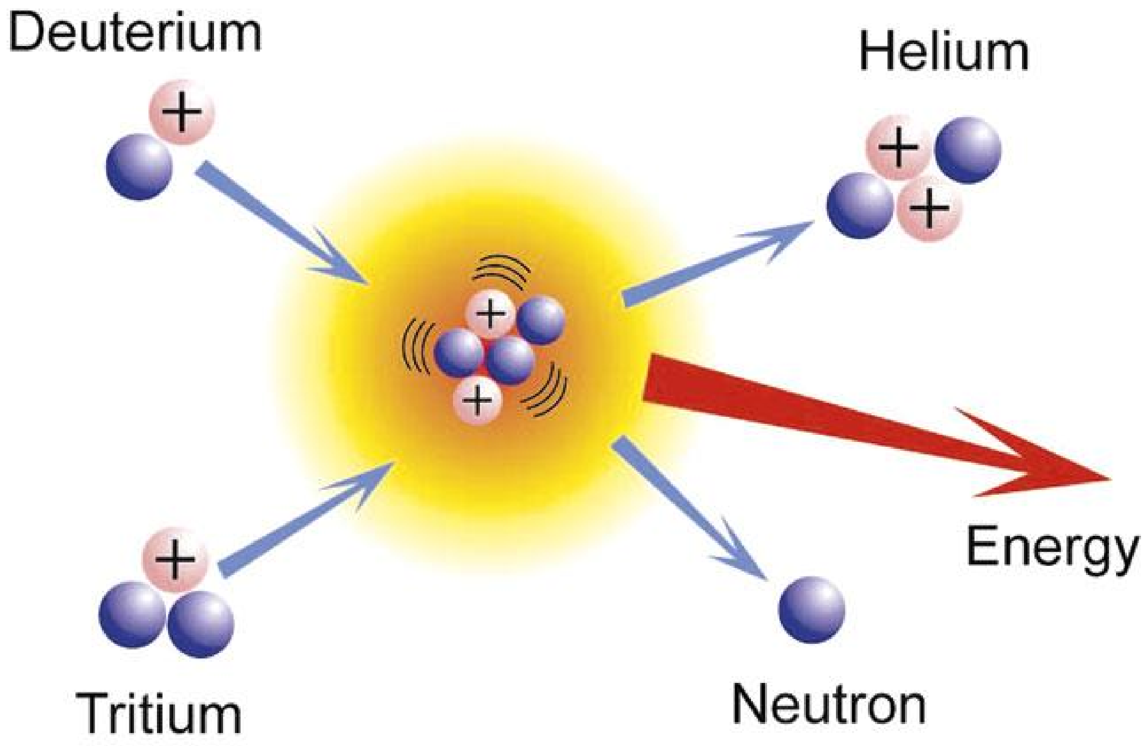


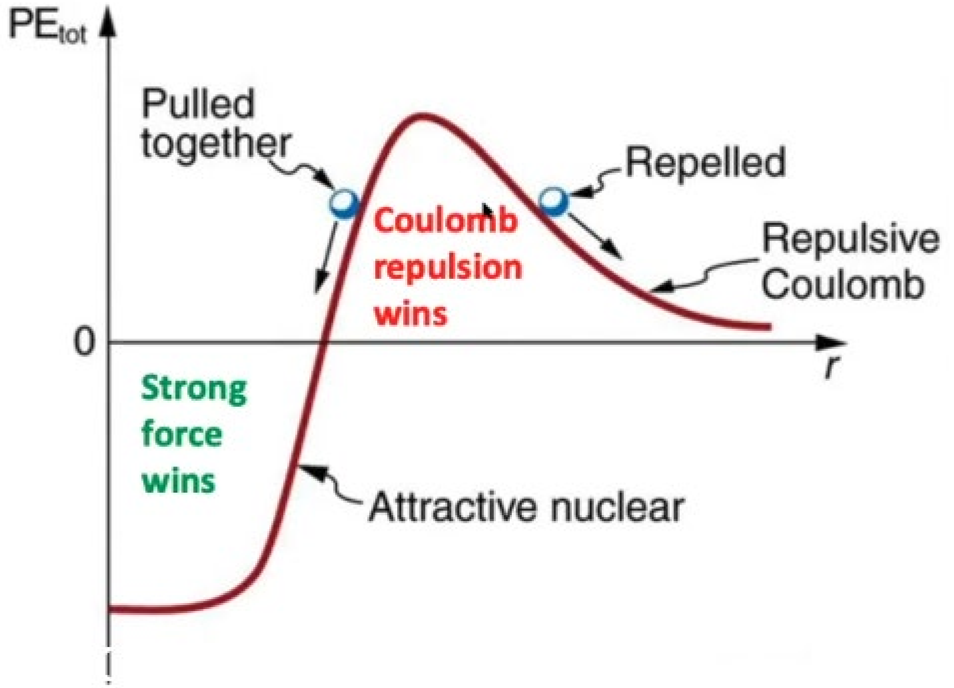


$^{2}H + ^{3}H \rightarrow ^{4}He$ **(3.5 MeV) + n (14.1 MeV) :** $E=\Delta mc^2$

Fusion Energy

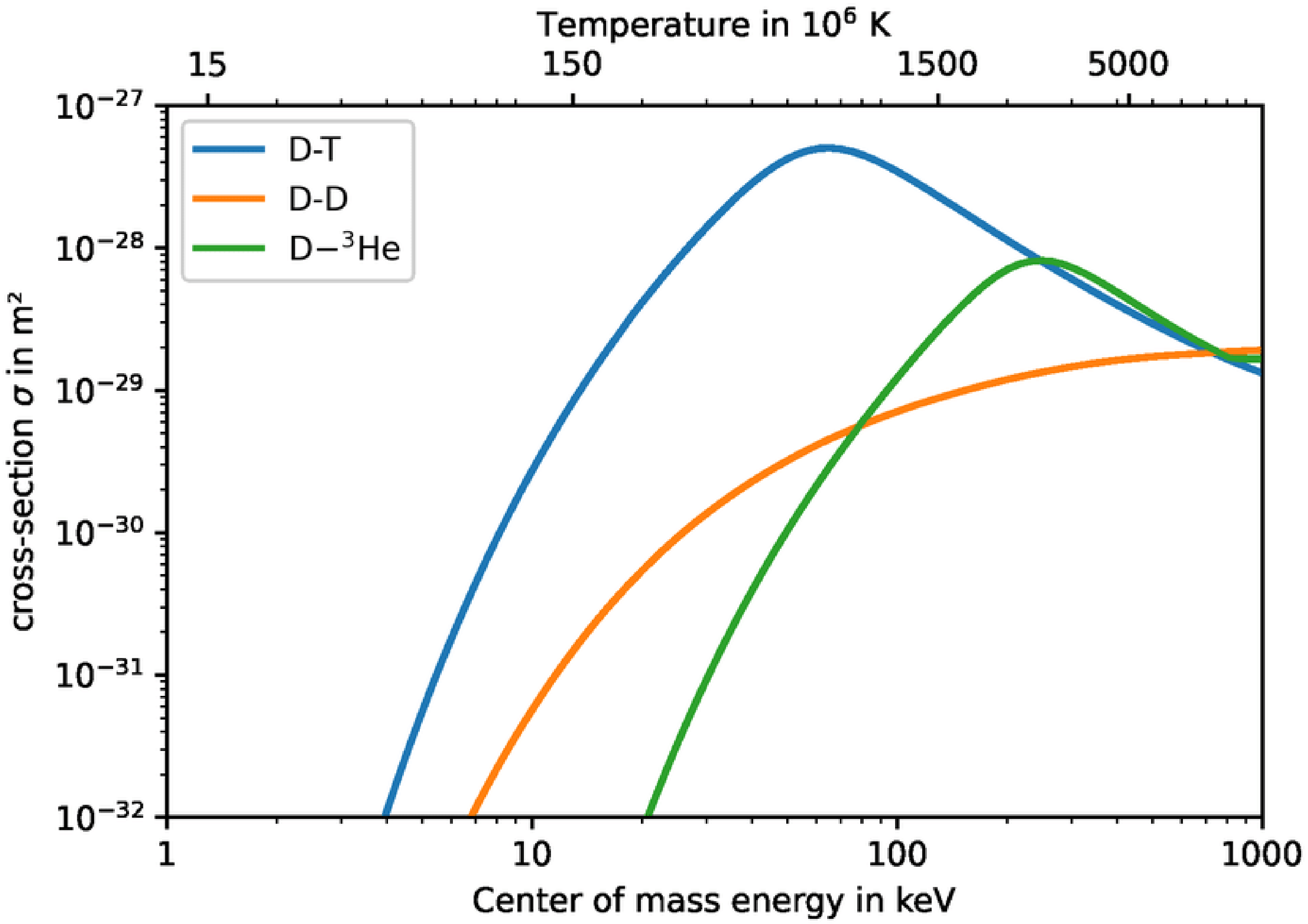
- Fusion can occur only at high temperature (~150 million C) due to the Coulomb barrier
- How can we contain the nuclei of that high temperature?



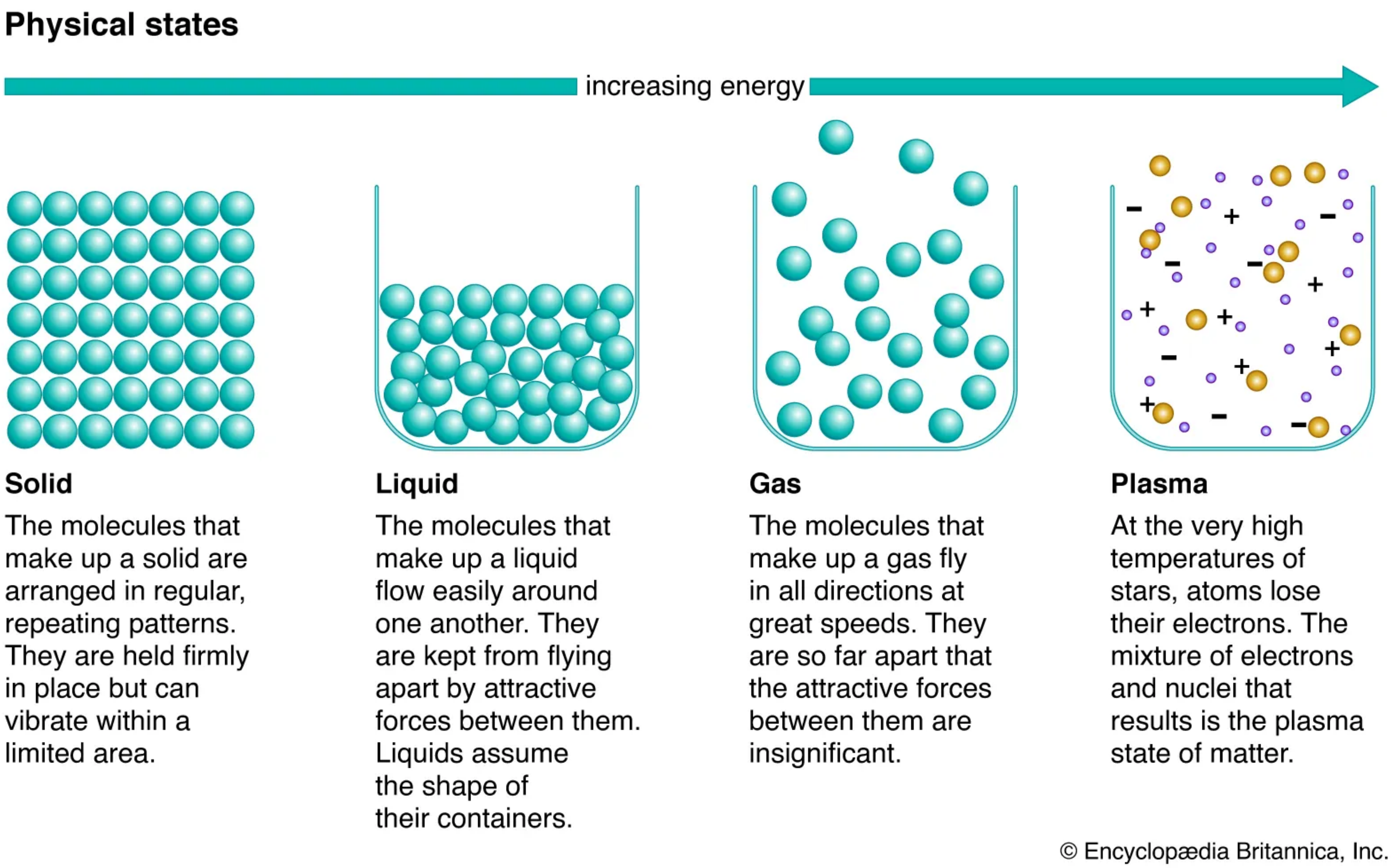


$^{2}H + {}^{3}H \rightarrow {}^{4}He$ (3.5 MeV) + n (14.1 MeV)

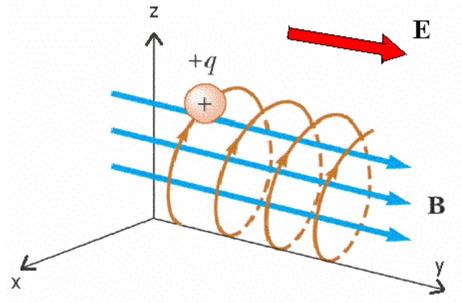
Cross Section of Nuclear Reaction as a function of energy (temperature)



States of Matter

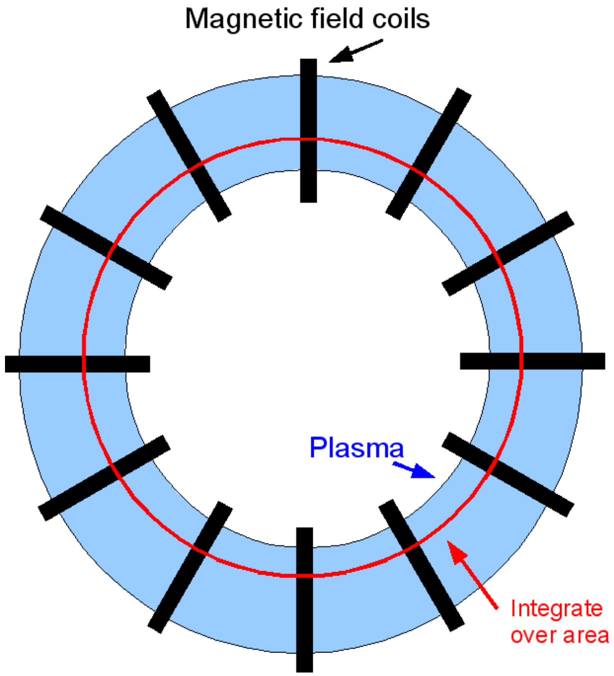


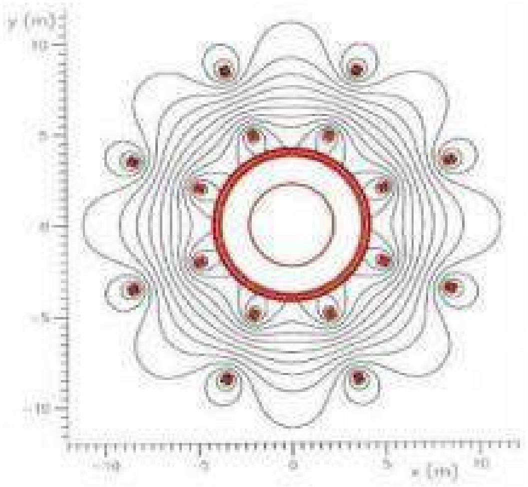
Confinement of Plasma by Magnetic Fields



Motion of charged particle
in magnetic field

- How can we contain the nuclei of that high temperature (~150 million C)?
- A key issue for nuclear fusion reactors is plasma confinement, which can be achieved by means of strong and properly shaped magnetic fields.





- The simplest configuration is to rely on a toroidal field.

$$\nabla \times \mathbf{B} = \mu_0 \mathbf{J}$$

$$2\pi R B_\phi = \mu_0 I$$

- But practical field strength varies as a function of plasma radius

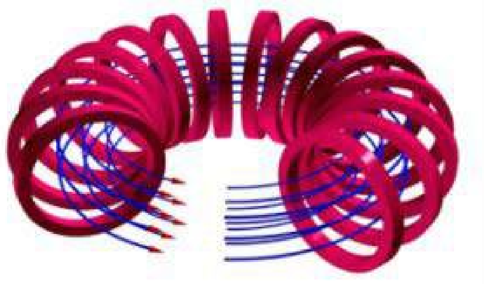
$$B_\phi = \frac{C}{R} \qquad \nabla B = \nabla\left(\frac{C}{R}\right) = -\frac{C}{R^2}\mathbf{e}_R = -\frac{B}{R}\mathbf{e}_R$$

$$F = -\mu \nabla B$$

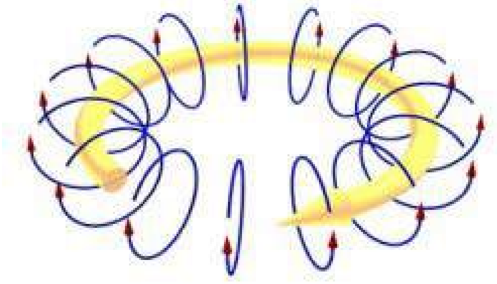
Which leads motion to the torus wall.

Tokamak (toroidal chamber with magnetic coils)

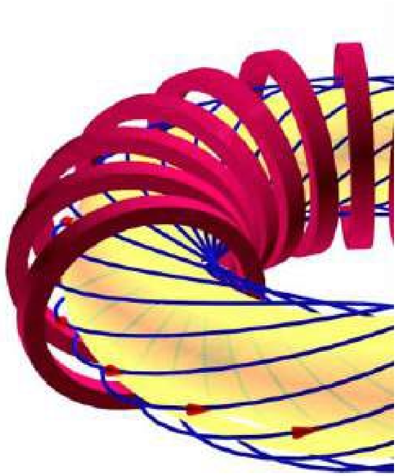
• A toroidal plasma current will generate a poloidal field, and the drift will be balanced by a return flow along the field.
• In a tokamak, the desired field lines are obtained by the combination of a toroidal magnet system and the circulation of a strong current in the plasma.
• The magnetic flux through the iron core is increase ($-\frac{\partial B}{\partial t} = \nabla \times E$) and it generates a toroidal current/poloidal field.
• The tokamak concept was developed by I. Tamm and A. Sakharov in the 1950's.



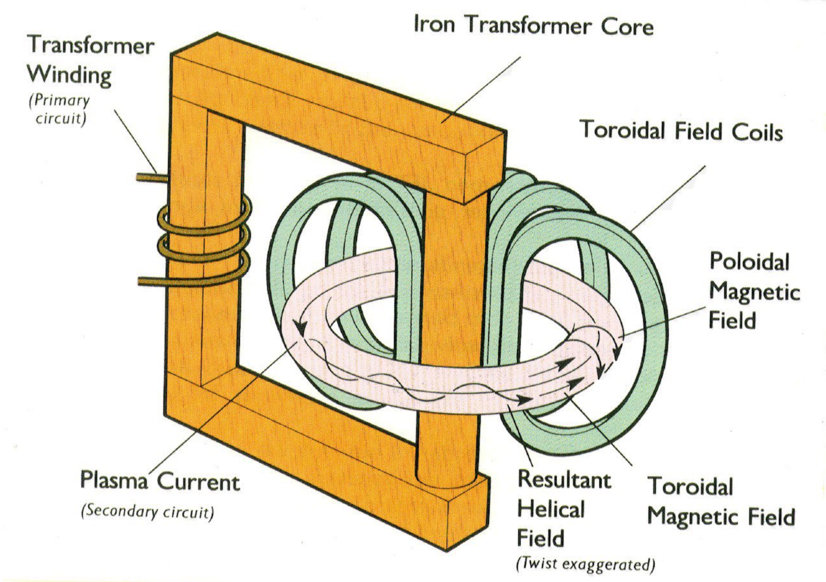
Toroidal Field



Poloidal Field Induced by Plasma Current

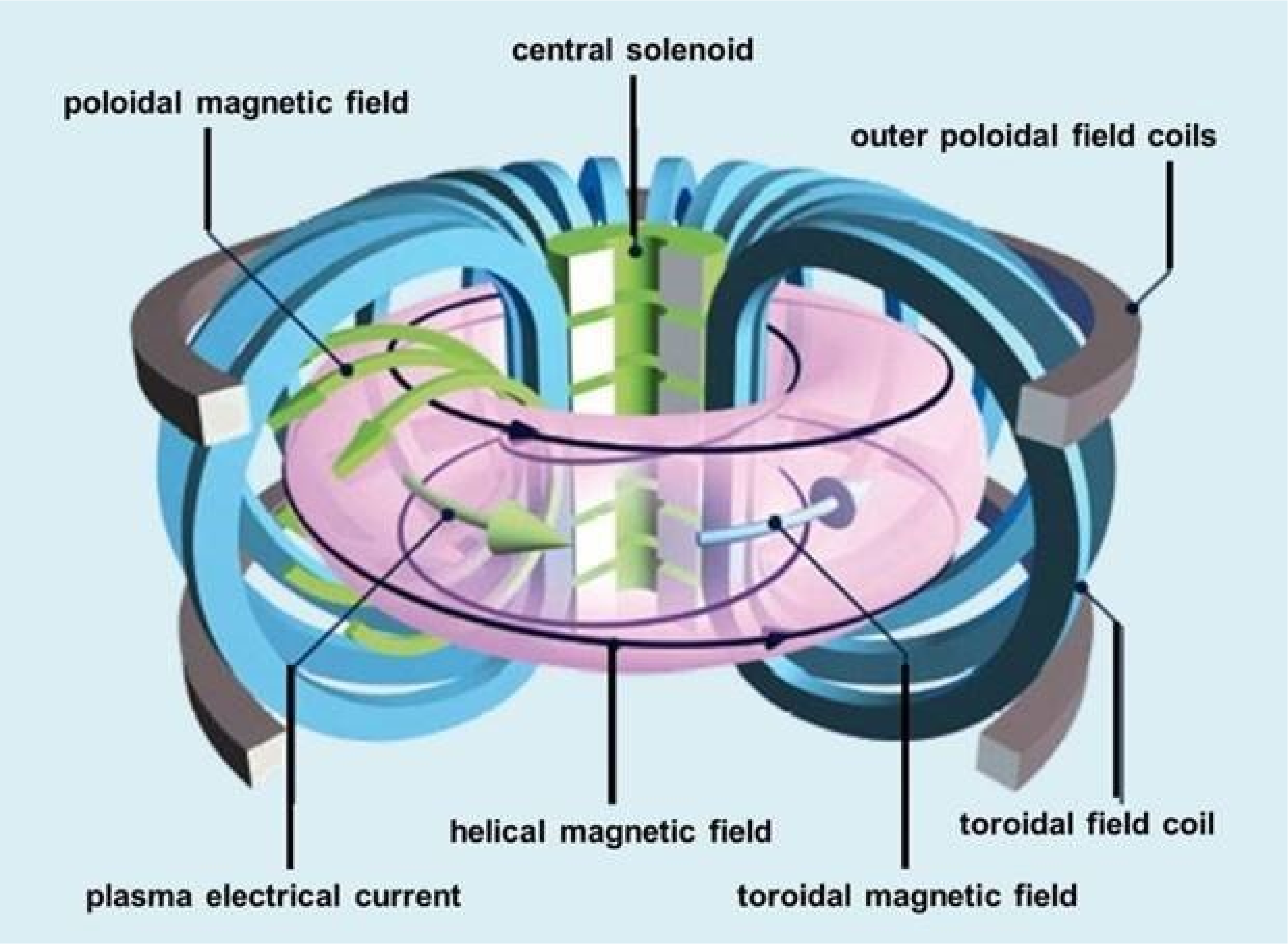


Toroidal & Poloidal Fields Combination



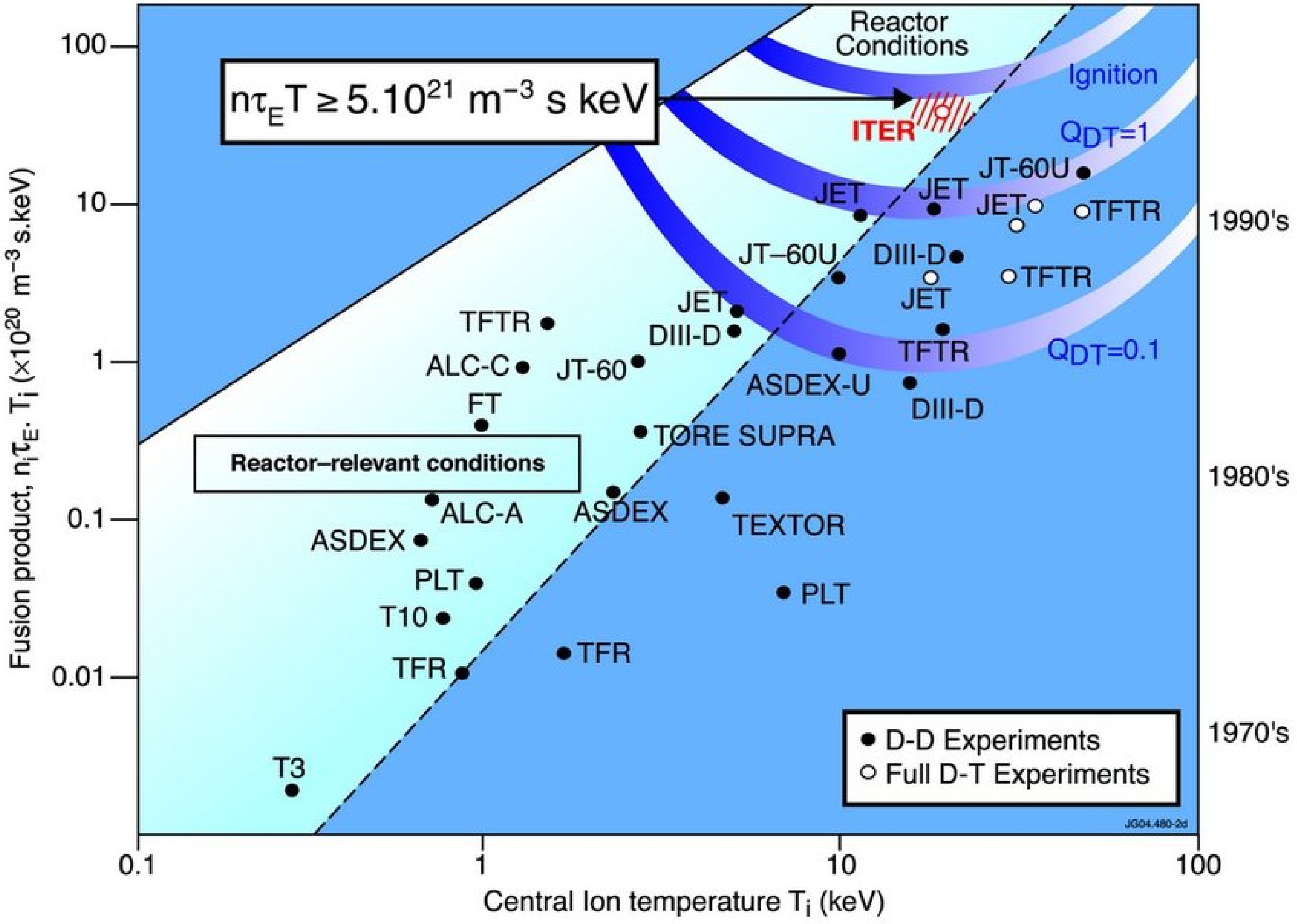
The plasma is the second winding of a transformer.

Magnets in a Tokamak



- **TF Coils** are used for charged particles' confinement in plasma.
- **CS coils** are used to produce inductive flux and to ramp up plasma current; they also play a role in plasma shaping and vertical stability.
- **PF coils** are used to control radial position of plasma, as well as for plasma shaping and vertical stability.

Can Tokamak be Energy Source?



- There are more than 100s of tokamaks in the world.
- Record of fusion was in JET in 1993.

$$Q_{DT}=\frac{P_{fus}}{P_{in}}<0.6$$

- For fusion as an energy source

$$Q_{DT} \gg 1$$

- Results of Nuclear physics is

$$P_{fus}/V \propto (\frac{\beta_N}{q^*})^2(\epsilon\frac{1+\kappa^2}{2})^2 B_T^4$$

$\frac{\beta_N}{q^*}$ *: safety factor,*

$\epsilon\frac{1+\kappa^2}{2}$*: geometrical factor*

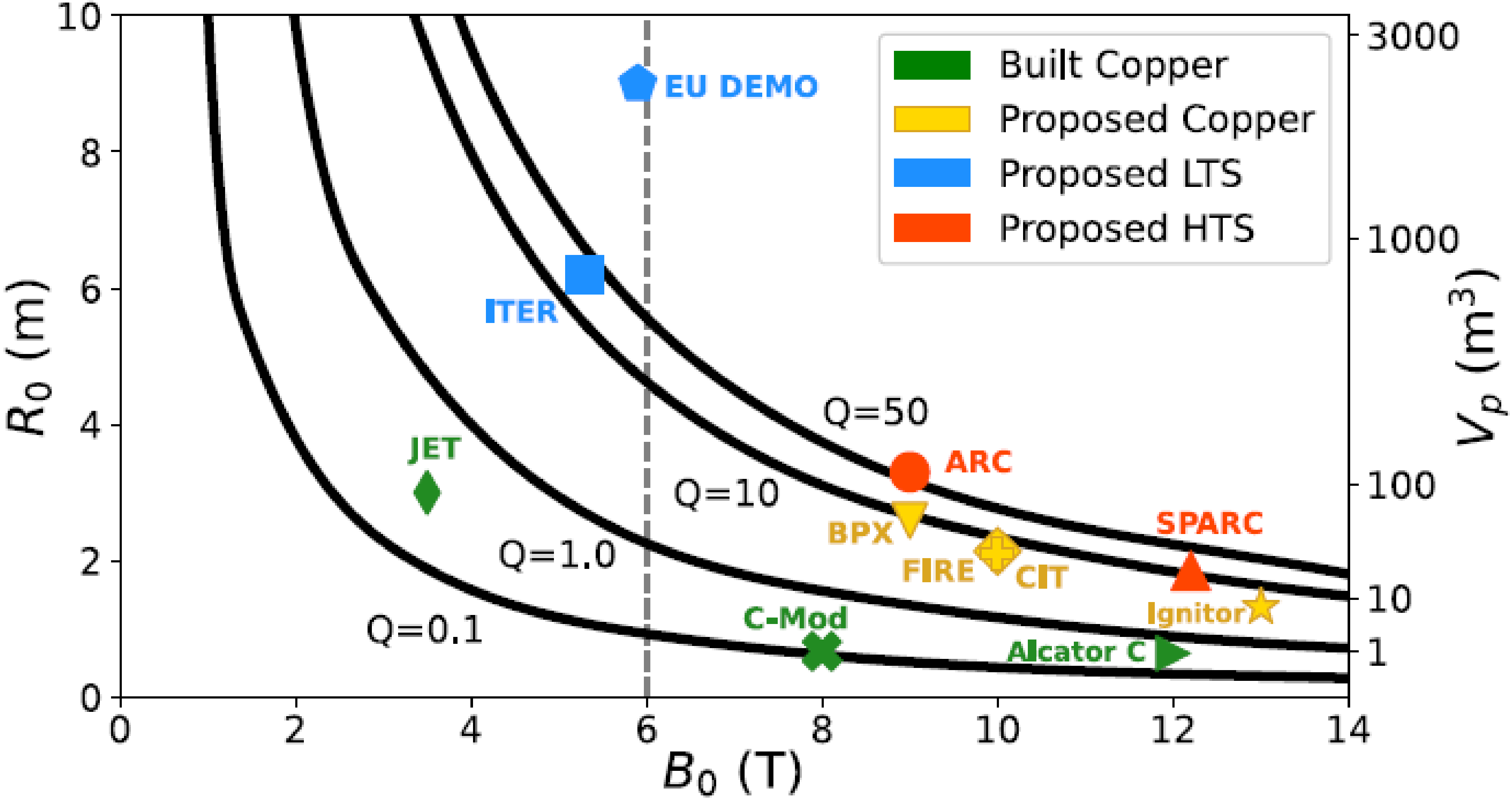
Do not vary significantly from tokamak to tokamak, so

$$\Rightarrow \boldsymbol{P_{fus} \propto R^3 B_T^4}$$

Size of the Tokamak

Creely, A. J., Greenwald, M. J., Ballinger, S. B. et al. (42 more authors) (2020) Overview of the SPARC tokamak. Journal of Plasma Physics. 865860502

$$P_{fus} \propto R^3 B_T^4$$



Parameters of Tokamaks

Parameter		SPARC (CFS)	C-Mod	AUG	DIII-D	EAST	KSTAR	Ignitor	CIT	FIRE	BPX	ITER
R	m	1.85	0.67	1.65	1.66	1.70	1.80	1.32	2.10	2.14	2.59	6.2
a	m	0.57	0.21	0.50	0.67	0.40	0.50	0.47	0.65	0.60	0.80	2.0
B_T	T	12.2	8.0	3.9	2.2	3.5	3.5	13.0	10.0	10.0	9.0	5.3
I_p	MA	8.7	2.0	1.6	2.0	1.0	2.0	11.0	11.0	7.7	11.8	15.0
$P_{aux,max}$	MW	25	6	30	27	28	16	24	20	20	20	73
$P_{flattop}$	s	10	1	10	6	1000	300	4	5	20	10	400
P_{fus}	MW	140						96	800	150	100	500
Q		11						9	∞	10	5	10

Planned and/or under construction

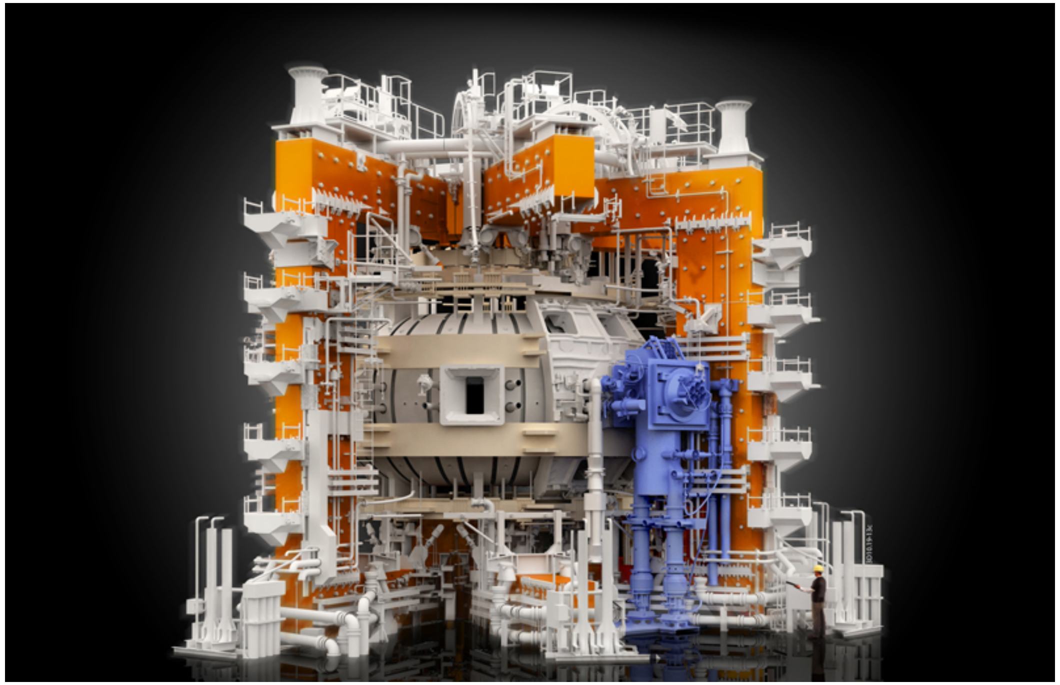
Existing

Conventional vs. Superconducting

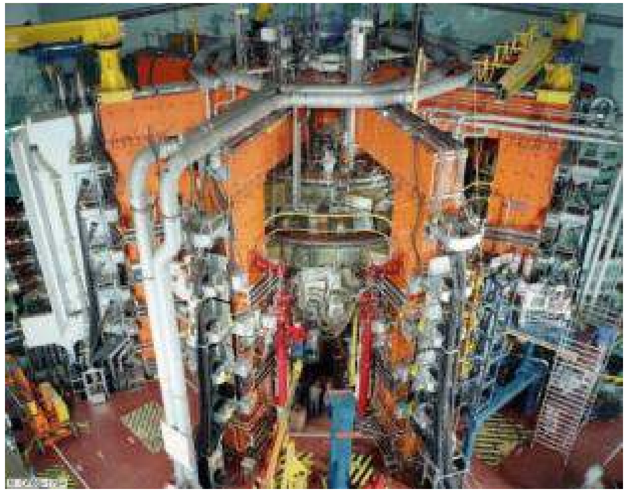
- At present, the largest fusion machine is the Joint European Torus (JET) Tokamak, located in Culham, UK
  - major plasma radius: ~3 m,
  - plasma volume: ~100 m^3,
  - magnetic field: ~3.45 T.

- The JET Tokamak relies on conventional magnets and requires up to 500 MW of electrical power!

- Larger machines (in size and/or magnetic field) call for the use of superconducting magnets.



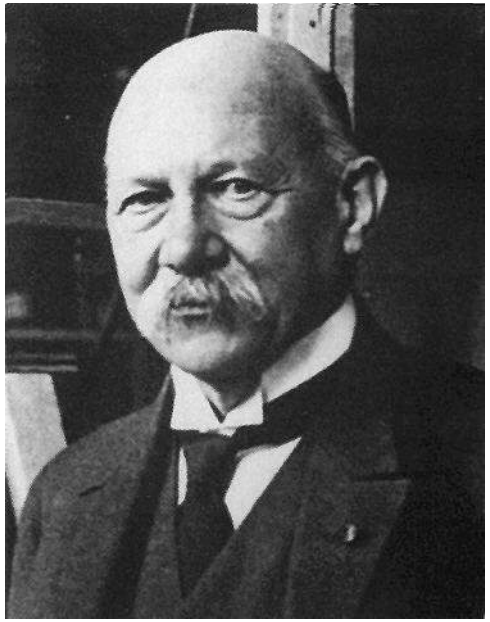
Views of JET Tokamak (2.96 m; 1983)



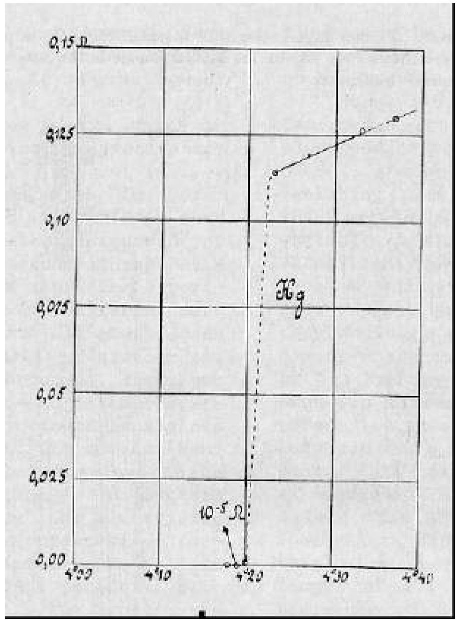
What is Superconductivity?

• Superconductivity is a unique property exhibited by some materials, whose resistivity drops to zero when cooled down below a threshold temperature called the critical temperature, T_C.

• As a result, materials in the superconducting state can transport current without power dissipation by the Joule effect.



H. Kammerling-Onnes (1853–1926)



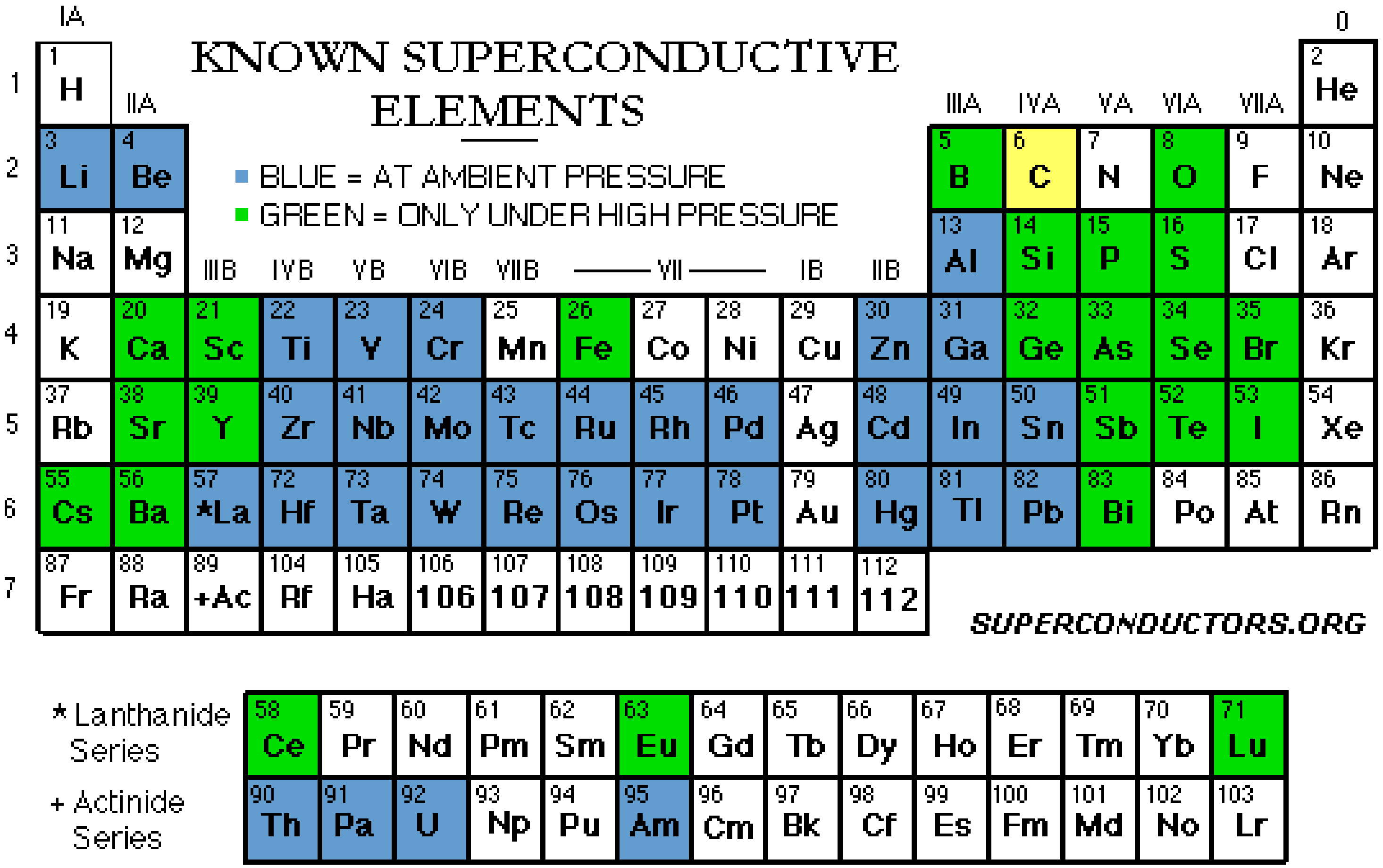
Discovery of Superconductivity On a Hg Sample (1911)

• Superconductivity was discovered in 1911 in a Laboratory of Leiden University, in the Netherlands, headed by Heike Kamerling-Onnes.

What is Superconductivity?

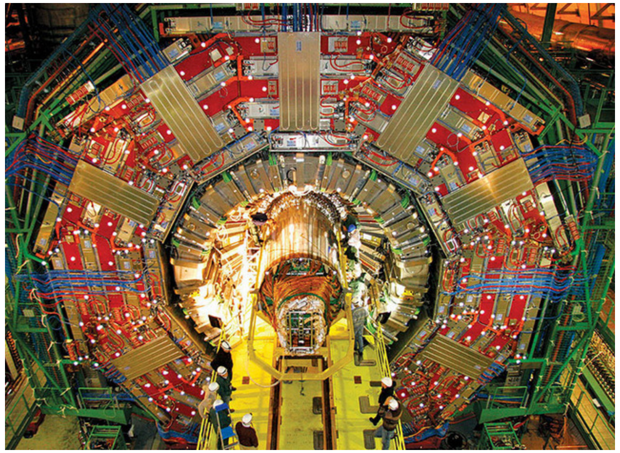
- Superconductivity is a rule rather than an exception.
- BCS (Bardeen, Cooper, Shrieffer) Theory gives Tc as follows

$$k_B T_c = 1.134 E_D e^{-1/N(0)V}$$

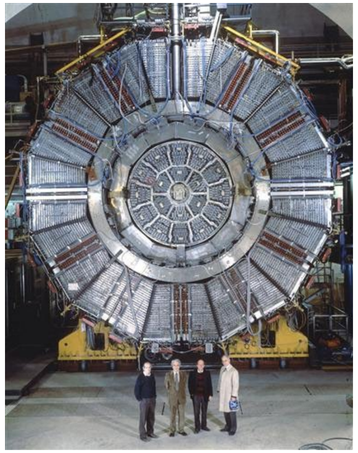


Pros of Superconductivity

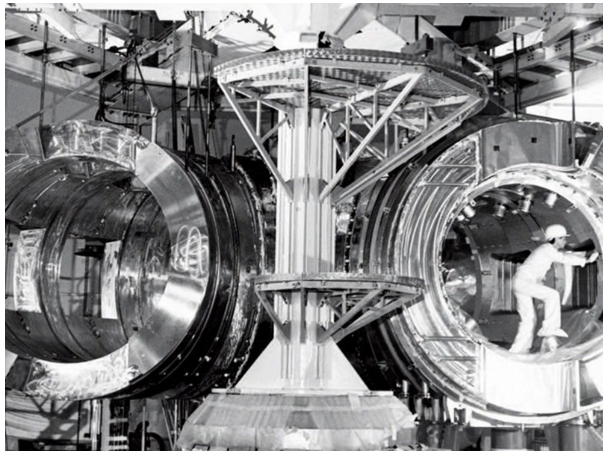
- Superconductivity offers at least two advantages
  - a significant reduction in electrical power consumption,
  - the possibility of relying on much higher overall current densities in the magnets coils (thereby resulting in more compact windings).
- This enables the design and manufacture of large and powerful magnet systems.



CMS @CERN



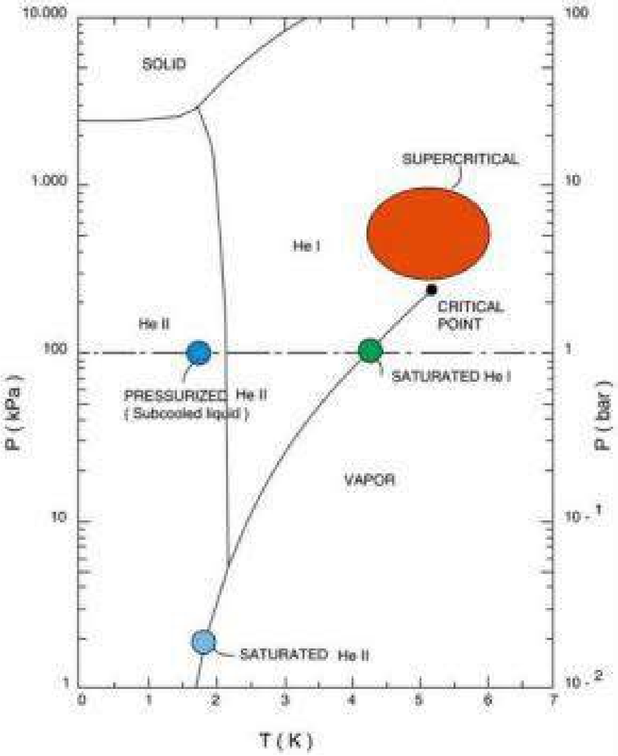
ALEPH Experiment @CERN (1989)



Tore Supra @CEA (1988)

Cons of Superconductivity

- The main drawback of superconductivity is that the magnet coil must be cooled down and kept below T_C throughout operation.
- The T_C of practical superconductors (NbTi and Nb_3Sn) calls for the use of liquid helium (LHe) whose boiling temperature is 4.2 K (–269℃) at 1 atm.
- Production and handling of LHe requires dedicated infrastructures and the magnet must be mounted inside a cryostat.

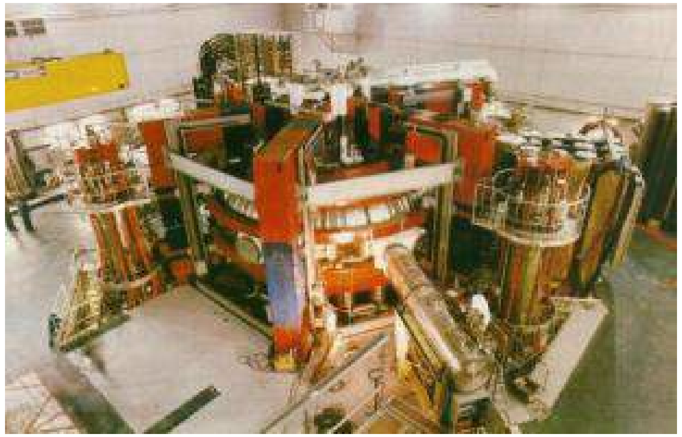


He Phase Diagram

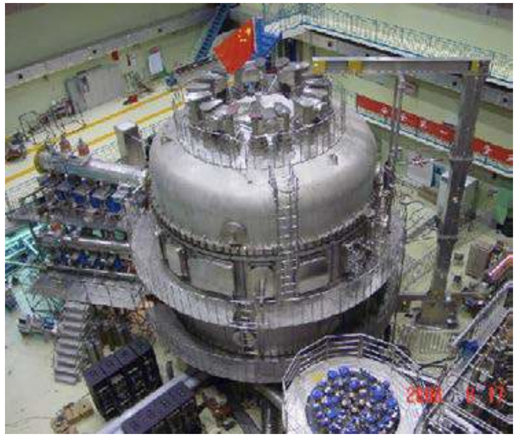
Cryo Plant of CERN

Cryo Test Facility (CEA/Saclay)

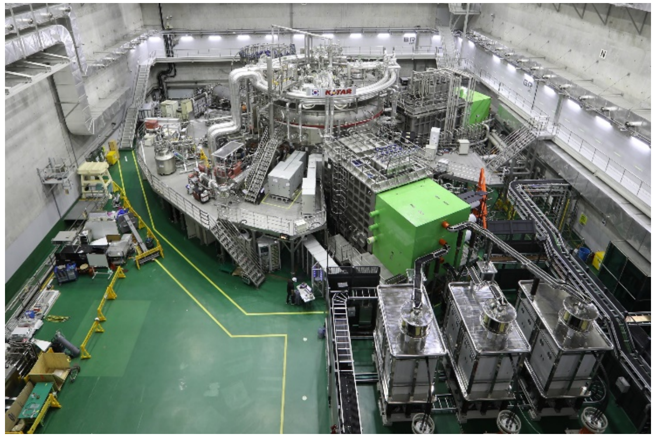
Superconducting Tokamak in the World



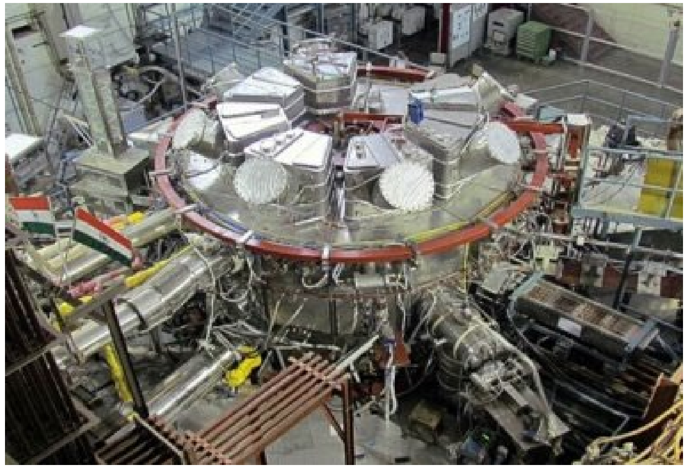
Tore Supra, CEA, France
(2.4 m; 1988)



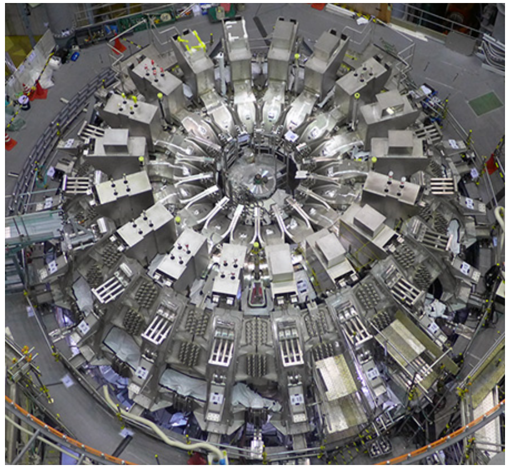
EAST, ASIPP, China
(1.8 m; 2006)



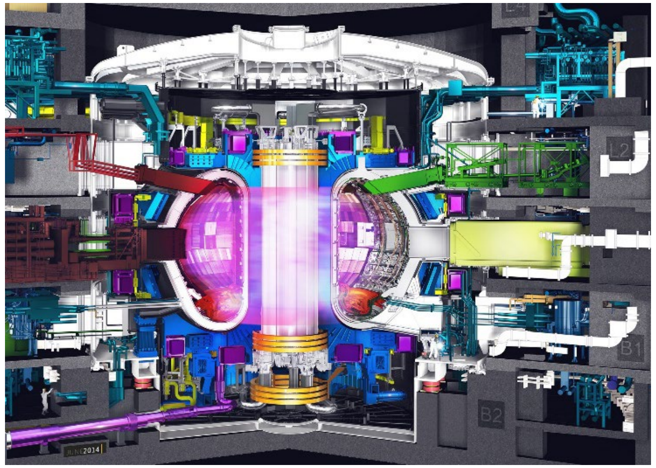
KSTAR, KFE, Korea
(1.8 m; 2008)



SST-1, IPR, India
(1.1 m; 2012)



JT-60SA, QST, Japan
(3 m; 2023?)

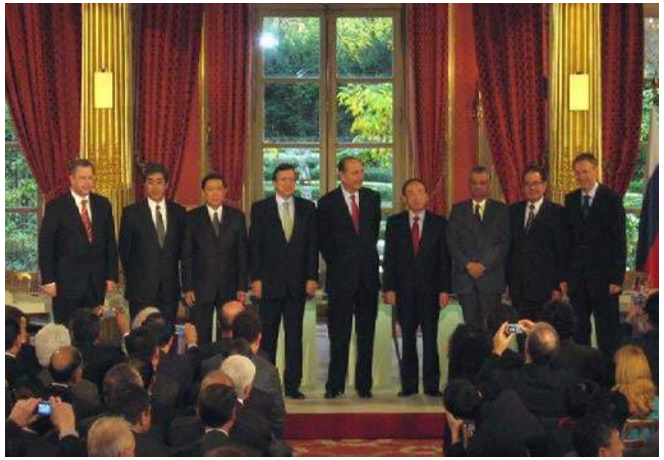


ITER, France
(6.2 m; 2025?)

ITER

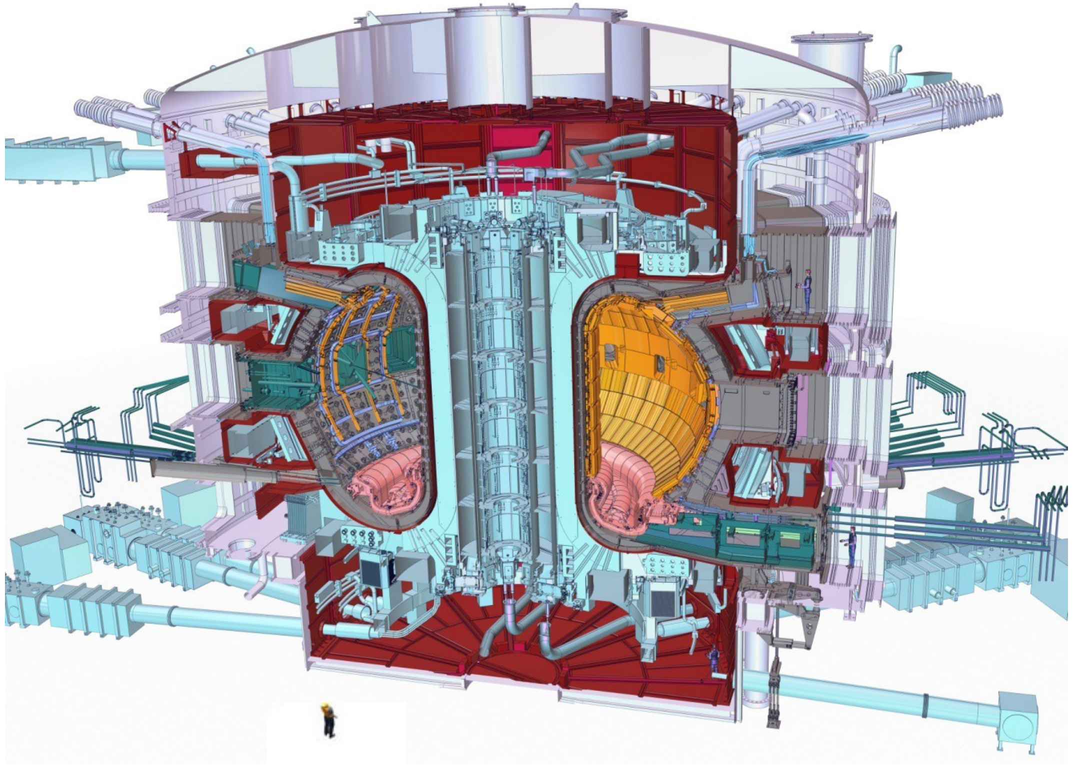
• ITER ("The way" in Latin) is an acronym of International Thermonuclear Experimental Reactor.

• The ITER project was born in 1985 at a superpower summit meeting in Geneva between R. Reagan and M. Gorbachev.



• 7 partners (China, EU, India, Japan, Korea, Russia and US) have been participating in ITER since 2006.

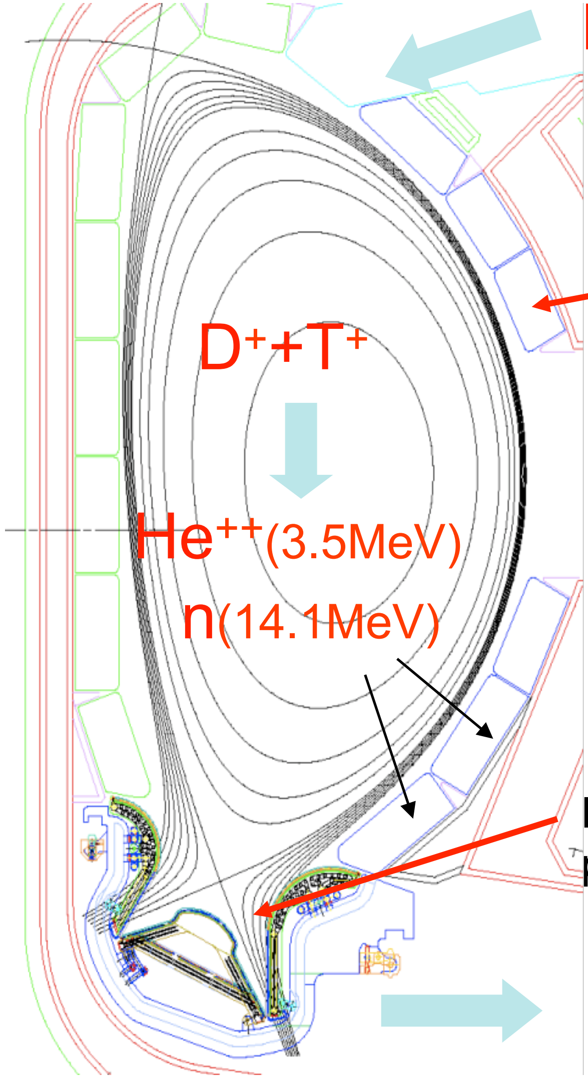
• ITER is designed to produce 500 MW of fusion power (Q=10) for extended periods of time (several 100s).



ITER

ITER Plasma:

R/a: Vo	6.2 m /2 m
lume:	830 m^3
Plasma Current:	15 MA
Toroidal field:	5.3 T
Density:	10^{20} m^{-3}
Peak Temperature:	$2''10^{8}$ K
Fusion Power:	500 MW
Plasma Burn	300 - 500 s
(“Steady-state”	~3000 s)



D_2,T_2 Fuel

Blanket:
neutron absorber

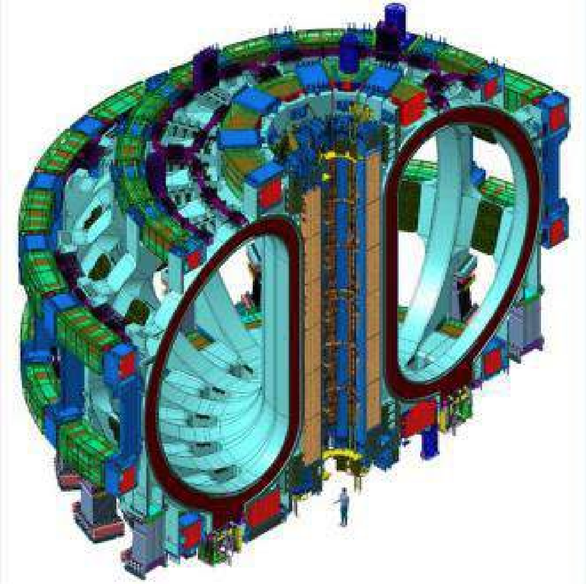
Power Plant
Li-->T
High temperature

Divertor:
particle and heat exhaust

He, D_2,T_2, impurities

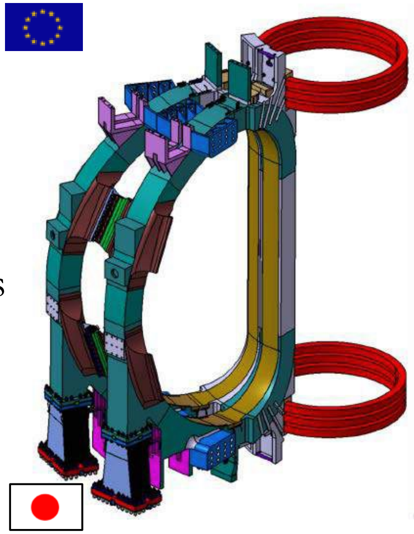
D.J. Campbell, Fusion Plasma Physics and ITER: An Introduction, CERN Academic Training Programme, 12-13 April 2011

ITER Magnet System



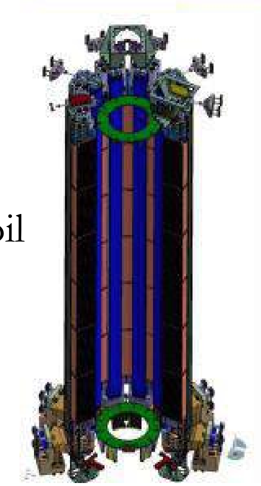
ITER Magnet System ~10,000 t

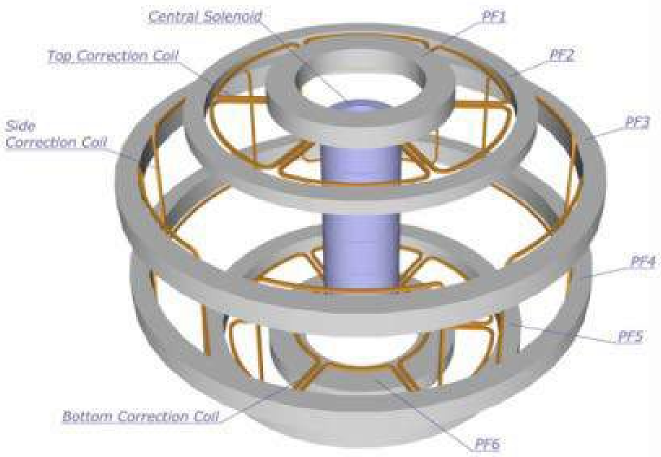
- The ITER magnet system is made up of
  - 18 Nb_3Sn Toroidal Field (TF) Coils,
  - a 6-module Nb_3Sn Central Solenoid (CS),
  - 6 NbTi Poloidal Field (PF) Coils,
  - 6 NbTi pairs of Correction Coils (CCs).



Pair of TF Coils

CS Coil



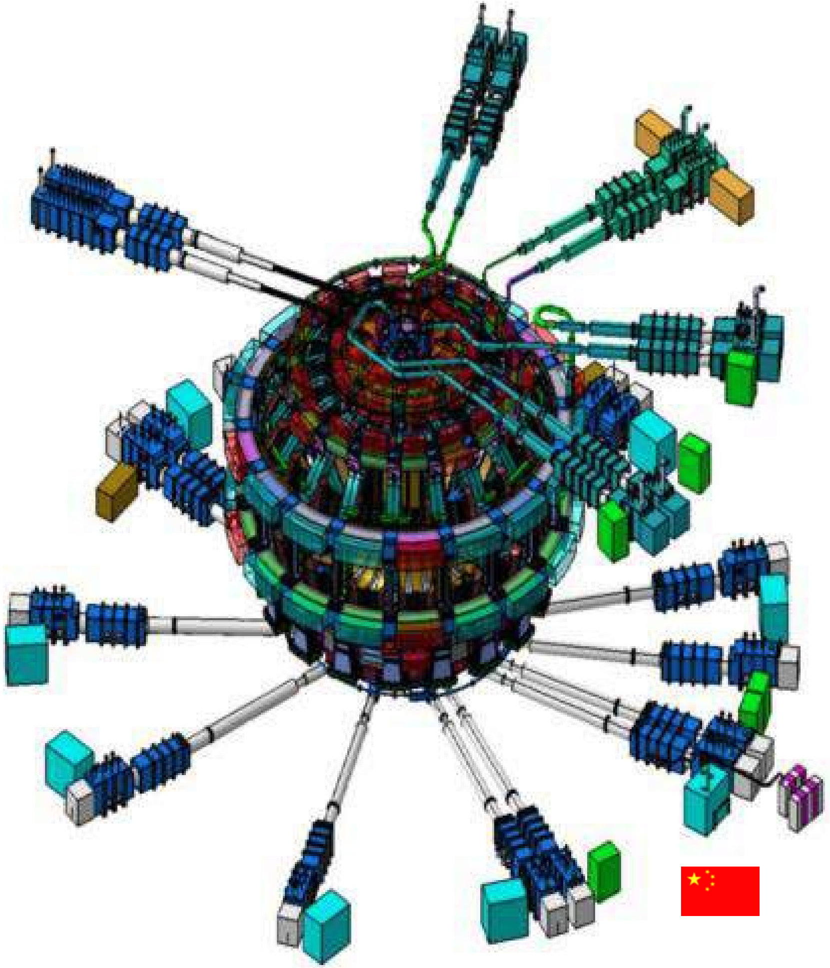


PF Coils

CCs

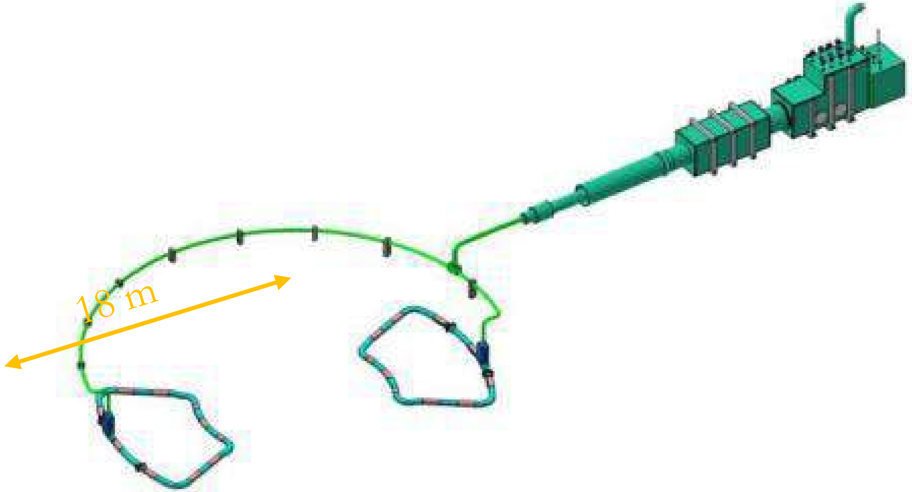
Feeders

- ITER magnets are supplied with current/cryogenic fluids by 31 Feeders.



ITER Feeder System

- The magnet Feeders include
  - NbTi CICC busbars (MB & CB),
  - Ag-Au(5.4%) BiSCCO 2223 HTS current leads.

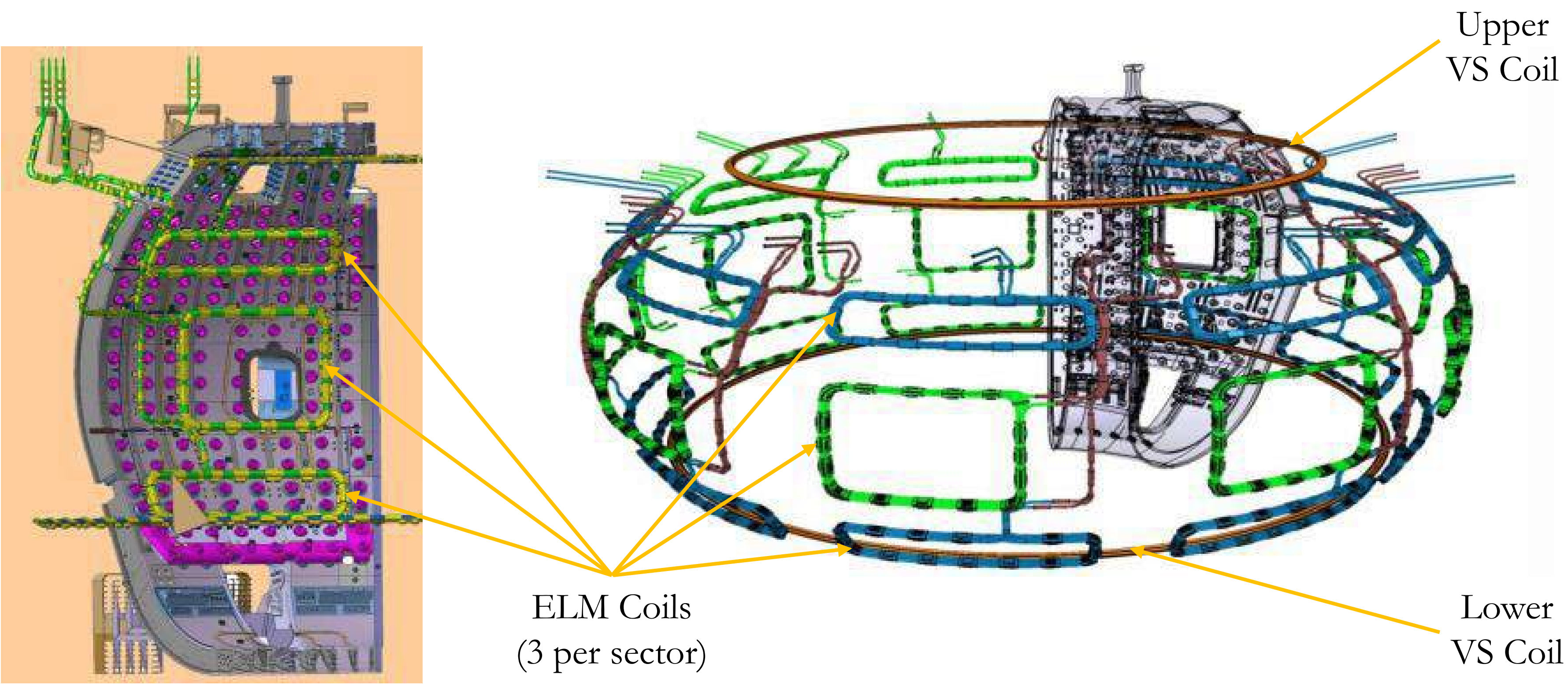


Detail of CC Feeder

68 kA Trial Lead Developed by ASIPP

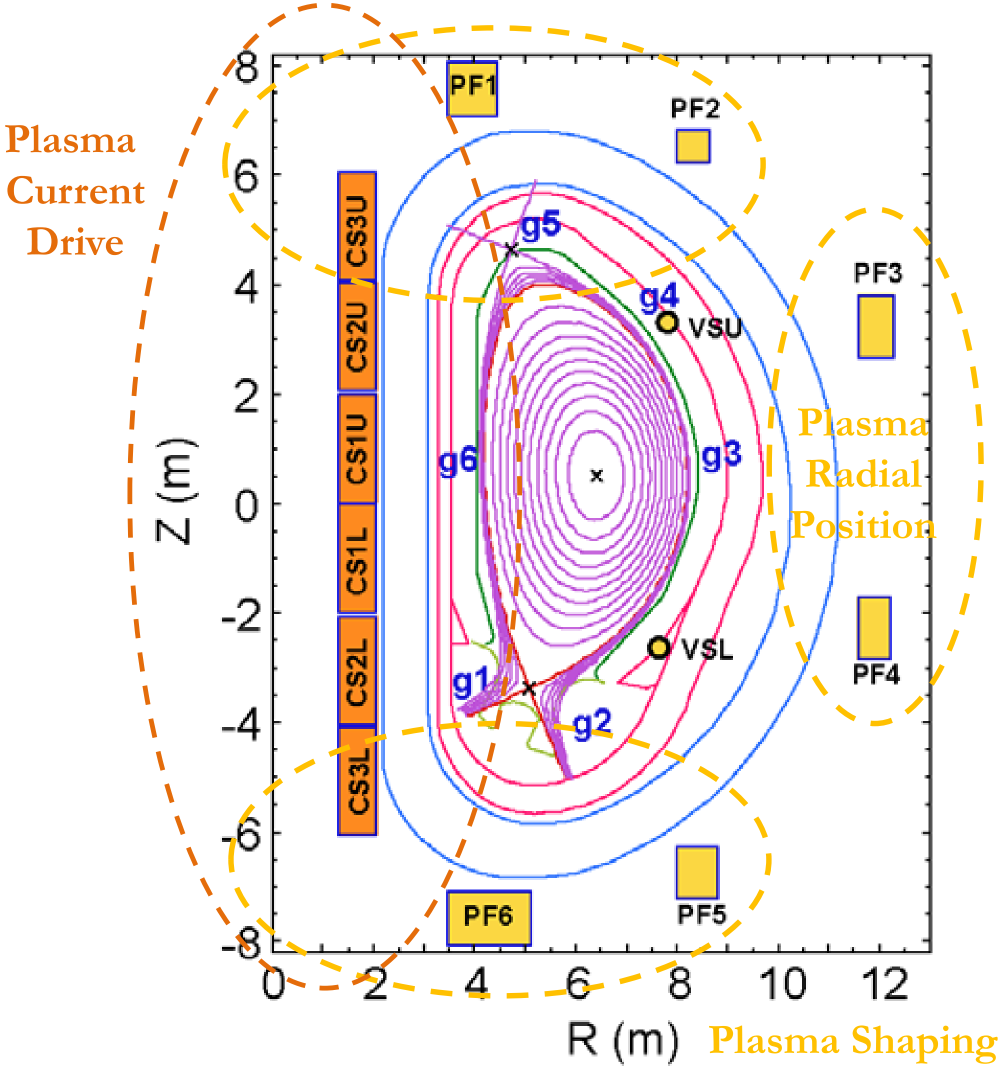
In-Vessel Coils

• In addition, the ITER tokamak is equipped with conventional, water-cooled, In-Vessel Coils, attached via rails to the outboard part of the inner wall of the vacuum vessel (underneath the blanket modules).



• The In-Vessel coils comprise: 3 x 9 picture-framed, ELM coils (aimed at suppressing Edge Localized Modes) and 2 ring, VS coils (aimed at better controlling Vertical Stability).

Roles of ITER Magnet Coils



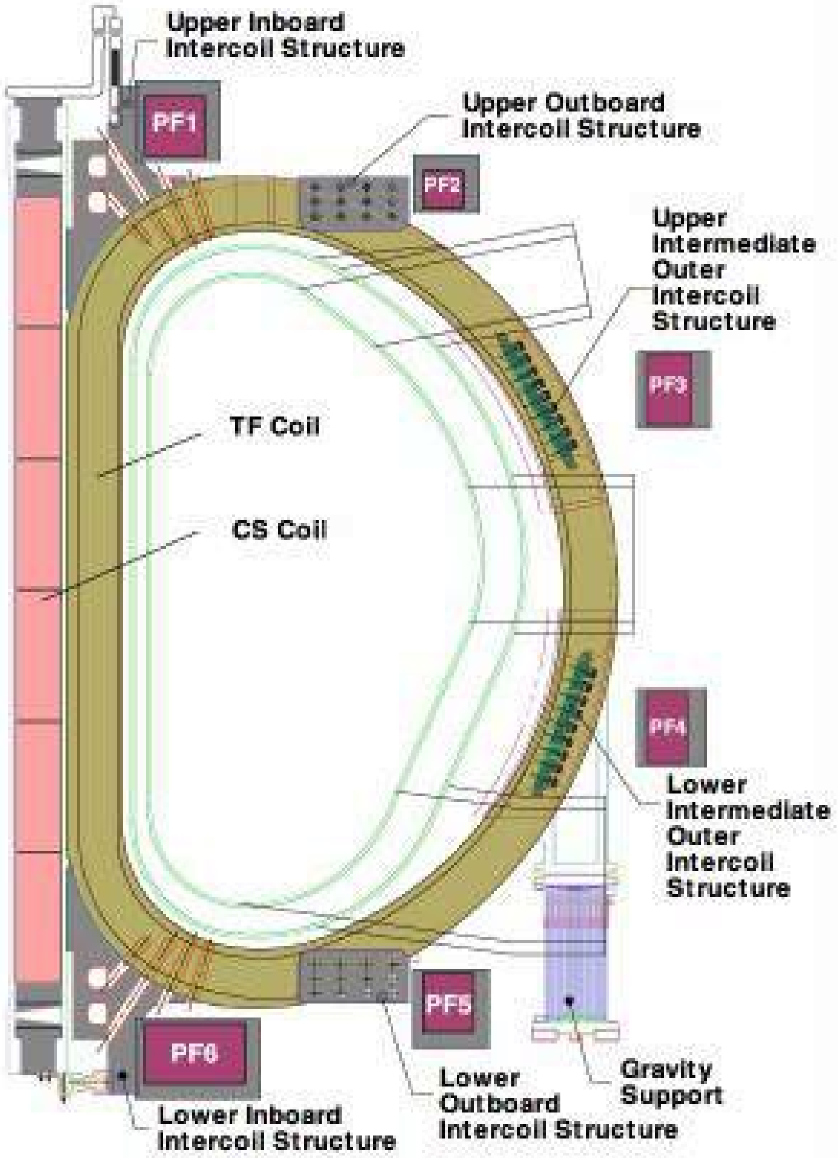
- **TF Coils** are used for charged particles' confinement in plasma.

- **CS coils** are used to produce inductive flux and to ramp up plasma current; they also play a role in plasma shaping and vertical stability.

- **PF coils** are used to control radial position f plasma, as well as for plasma shaping and vertical stability.

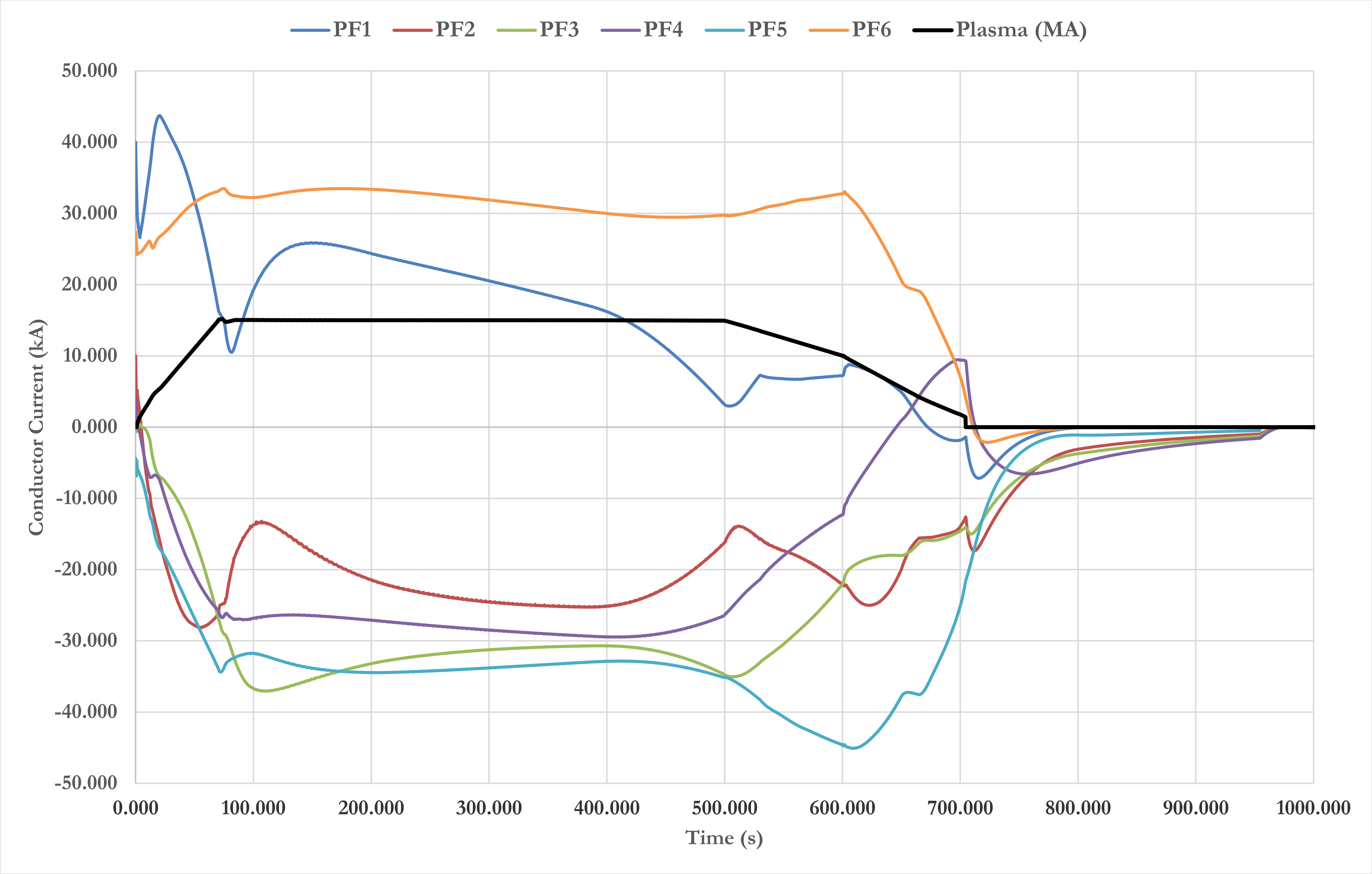
- CCs are used to correct field harmonics errors.

Parameters of ITER Magnets

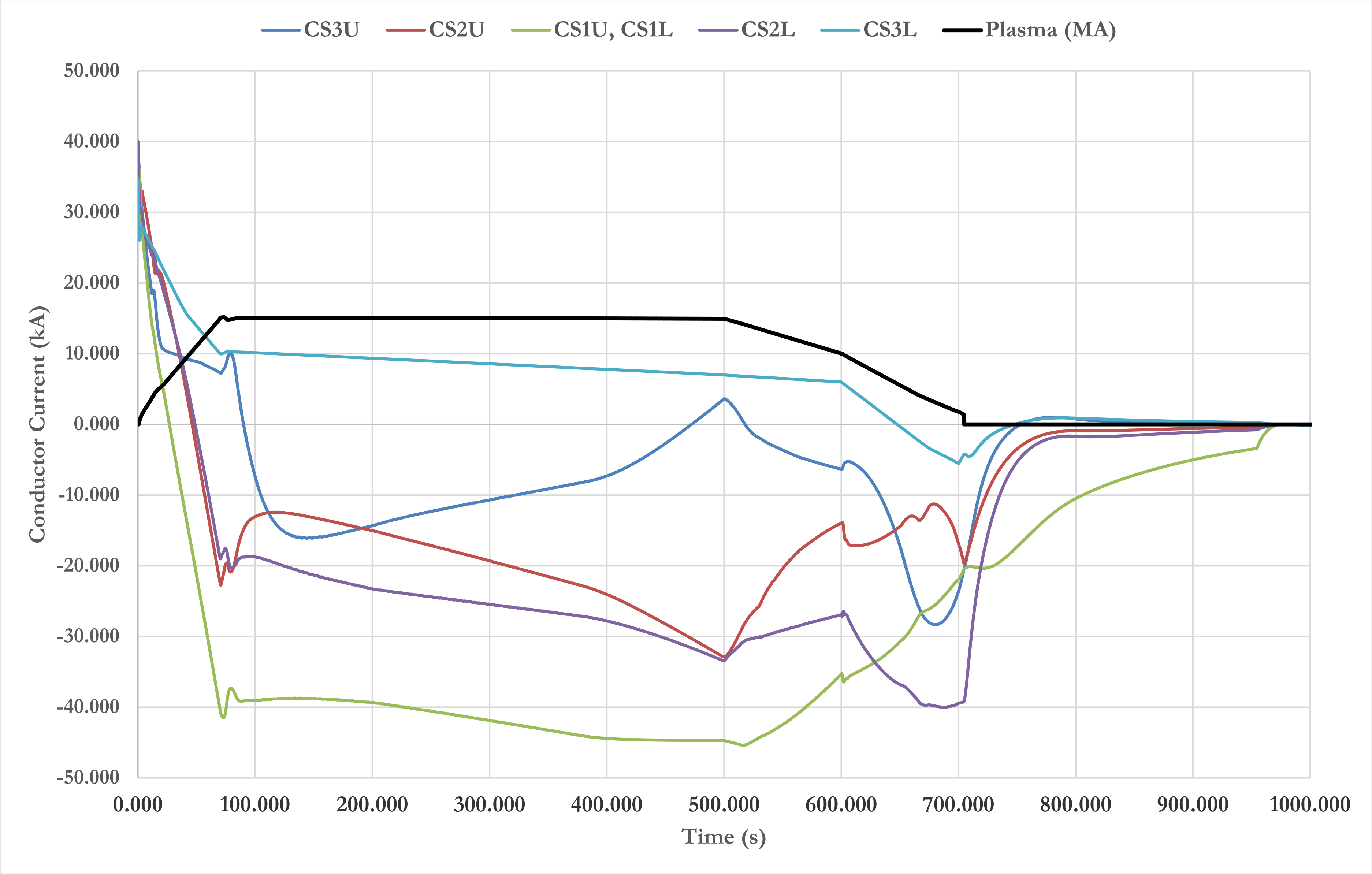


	TF	CS	PF
Number of Coils	18	1	6
Dimension	14x9 m	12x4 m	8 to 24 m
Conductor Type Quantity	Nb_3Sn CIC 88 km	Nb_3Sn CIC 42 km	NbTi CIC 65 km
Operating Current	68 kA	46 kA	52 kA
Operating Temperature	5 K	4.5 K	4.5 K
Peak Field	11.8 T	13.0 T	Up to 6.0 T
Stored Energy	41 GJ	6.4 GJ	4 GJ
Total Weight	6540 t	974 t	2163 t

PF Coil Reference Scenario (15 MA Plasma Current)



CS Coil Reference Scenario (15 MA Plasma Current)



Operational Requirements of ITER Magnets

• The basic design requirements are **30,000 plasma pulses** of 15MA current with a burn length of 400 s, and a total fusion power of 500 MW.

• The magnets will be designed for **100 cooldown and warm-up cycles, 1,000 TF charging cycles, 50 TF fast discharges and 10 quenche**s. Theses operations are all big stress to TF Coils.

• A fast discharge of the TF system will require a fast discharge of all coils (TF, CS, PF and CC). Fast discharge of a CS or PF coil will require an associated fast discharge only of the CS and PF system. A quench of a CC will require a fast discharge only of the CC system.

• The magnets need not be ready for further operation (i.e. completely re-cooled) until 96 hrs after a fast discharge of the TF system. In the event of a fast discharge of only CS, PF or CC, the magnets will be re-cooled ready for further operation within 2 hrs.

Magnet Design Criteria

- The magnets operate 4K-77K, mostly 4K. Compared to high temperature, yield and ultimate strength are increased but fracture toughness similar. **Fracture is relatively much more important as a failure mode than plastic yielding**.

- **Extensive use of non-metallic materials** especially for bonding and compressive load transmission.

- The **loads** have strong **cyclic** components.

- **Pressure is not significant design factor.** Electromagnetic loads cause 3D stress systems, generally need FE analysis, different to pressure vessels.

- Combination of structural support functions in a High Voltage environment; **the structural limits could be defined by the electrical functionality.**

- **No previous experience with such a design**

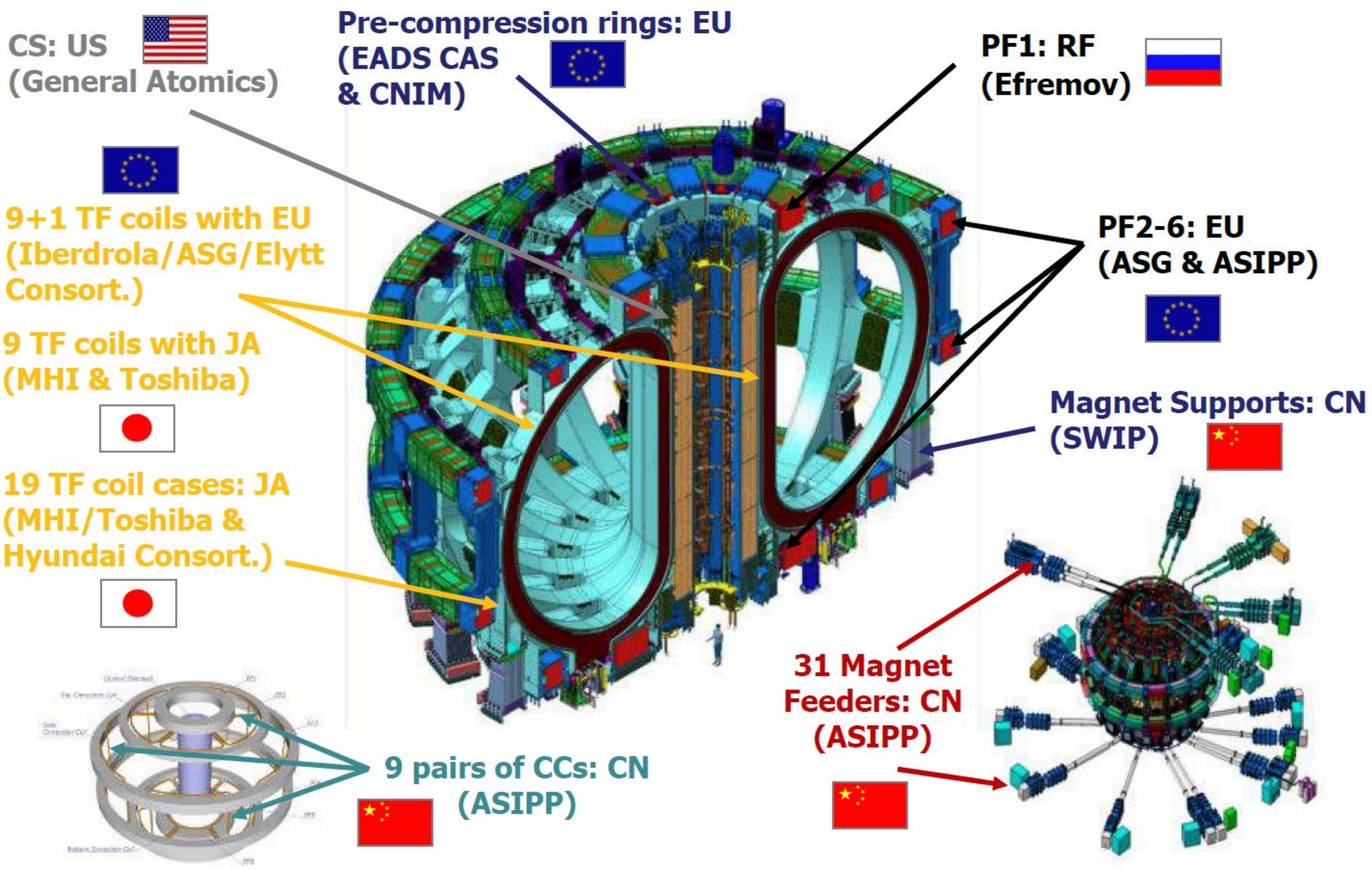
- **Needs for specific set of design criteria**

Procurement Allocation of Magnet System

- Artificial Breakdown of Procurement Package

Package			kIUA	CN	EU	IN	JA	KO	RF	US	FUND
1.1 Magnets	Toroidal Field Magnet Windings	1A	85.20		100%						
		1B	82.30				100%				
	Toroidal Field Magnet Structures	2A	51.40		10%		90%				
		2B	47.70				100%				
	Magnet Supports	2C	22.85	100%							
	Poloidal Field Magnet 1 & 6	3A	13.60		50%				50%		
	Poloidal Field Magnets 2, 3, 4, 5	3B	33.60		100%						
	Correction Coils	3C	2.60	100%							
	Central Solenoid Magnet	4A & 4B	39.60							100%	
	Feeders	5A	26.15	100%							
	Feeder Sensors	5B	18.05								100%
	Toroidal Field Magnet Conductors	6A	215.00	7%	20%		25%	20%	20%	8%	
	Central Solenoid Magnet Conductors	6B	90.00				100%				
	Poloidal Field Magnet Conductors	6C	74.25	69%	13%				18%		

ITER Magnet Supply: 10 PAs/5 DAs

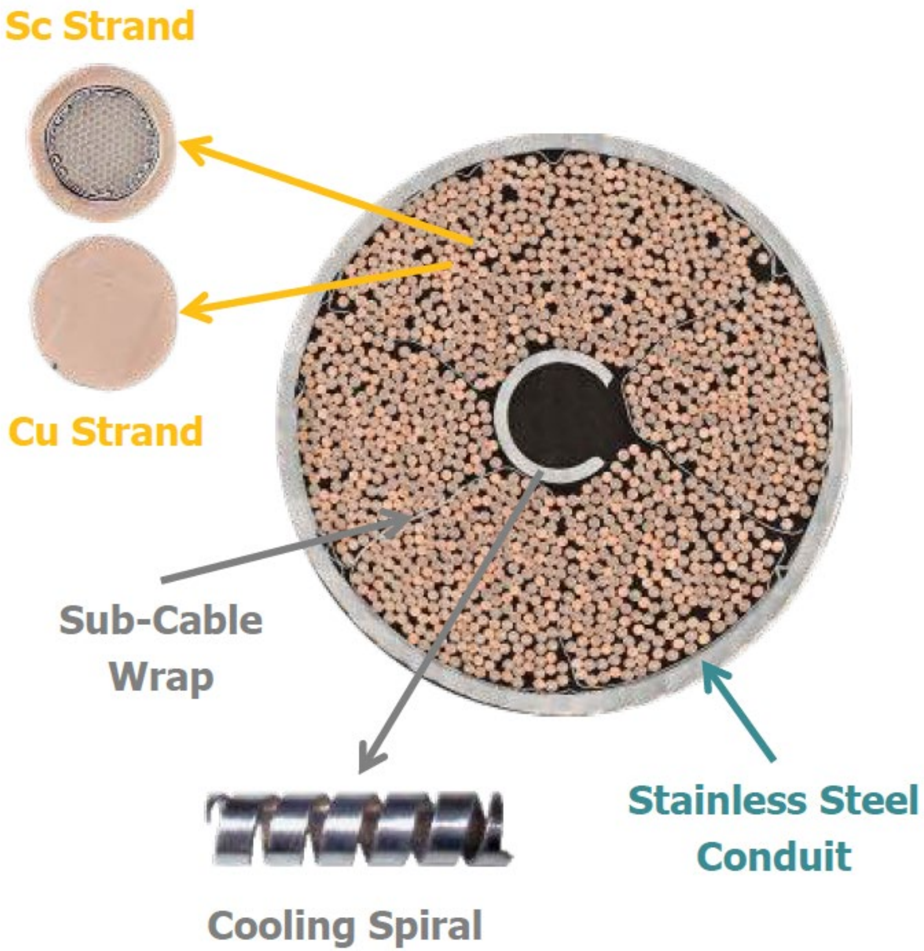


Cable-In-Conduit Conductors

• ITER magnets rely on Cable-In-Conduit Conductors (CICCs).

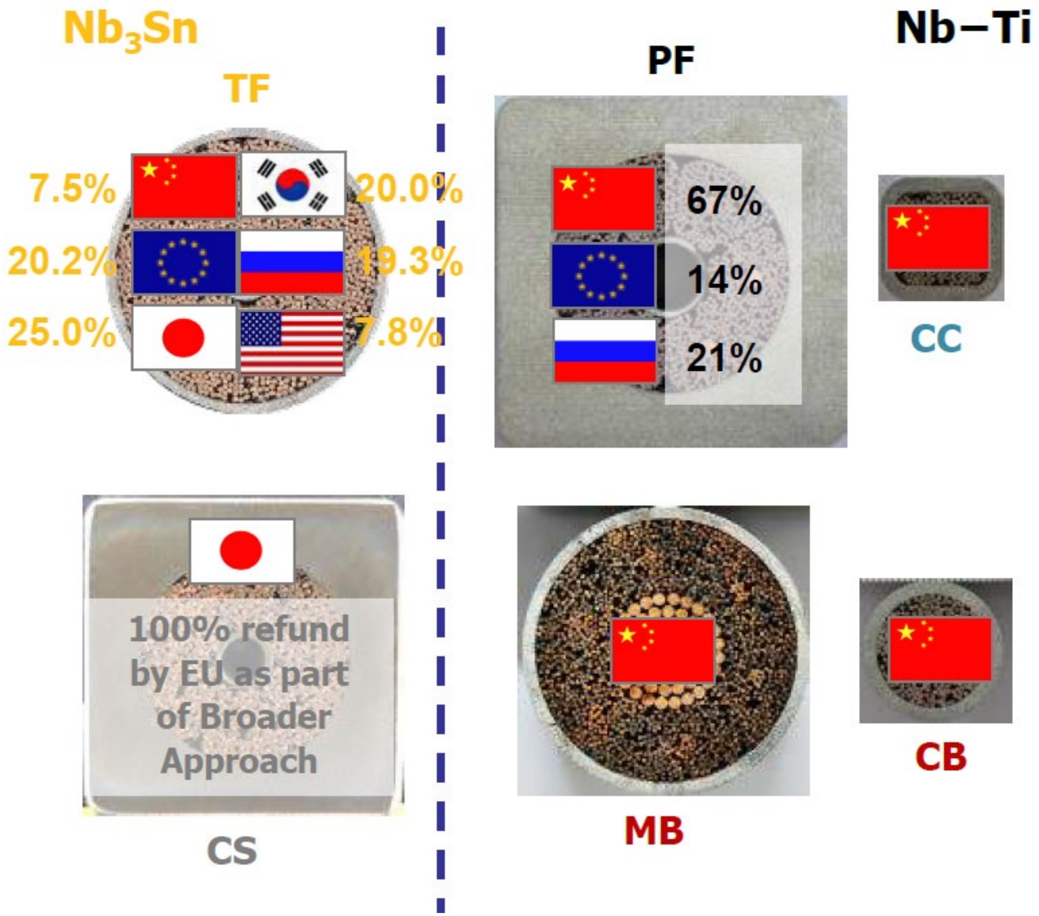
• Main features are
– Nb_3Sn or NbTi superconducting (sc) strands mixed with Cu strands,
– multi-stage, rope-type cable with stainless steel cable/subcable wraps and a central cooling spiral (save for CC/MB conductors),
– circular, rectangular or circle-in-square, austenitic stainless steel conduit made up of butt-welded jacket sections.

• Cooling is provided by forced flow of supercritical helium.



Procurement Sharing of Conductors

- ITER conductors represent about half of the cost of the ITER magnet system
- They are procured in-kind and shared among 6 ITER Partners: CN, EU, JA, KO, RF and US.

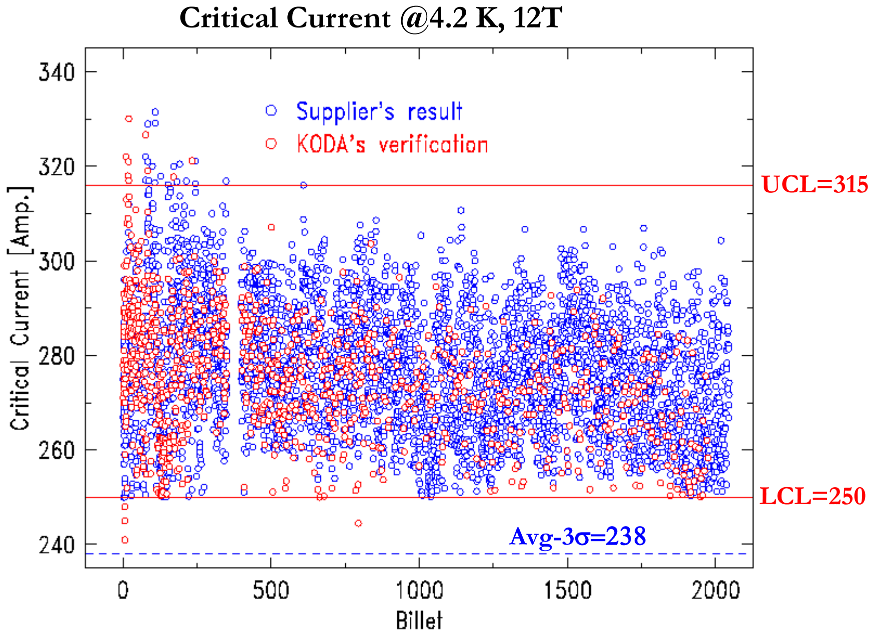


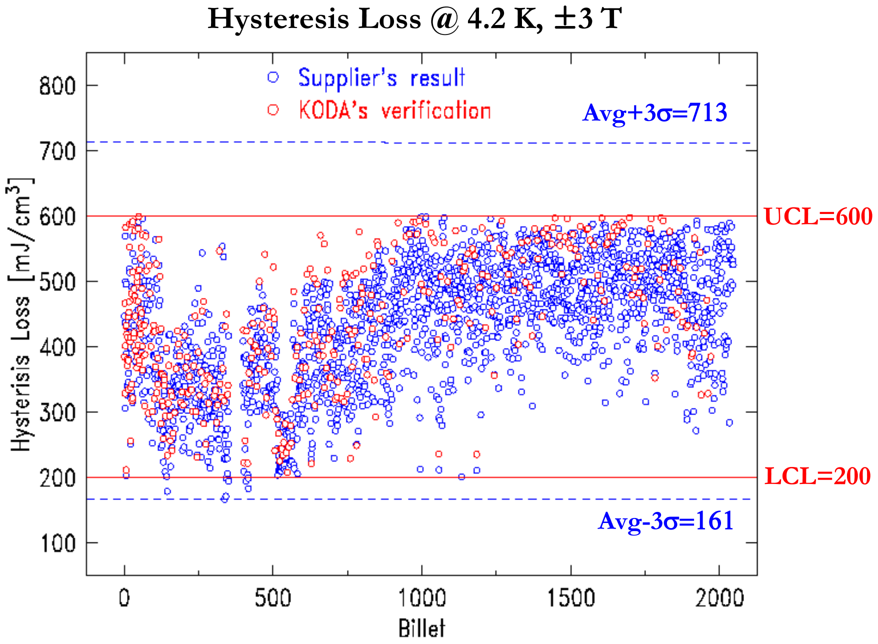
Specifications and Characteristics of Nb_3Sn Strand

- Nb_3Sn strand specifications are functional and call for
  - Diameter (un-reacted, Cr-plated) 0.820 ± 0.005 mm
  - Cu-to-Non-Cu volume ratio 1.00 ± 0.10
  - I_C at 4.22 K and 12 T (on ITER Barrel) > 250 A
  - Hysteresis loss per strand unit volume < 600 mJ/cm^3

    (± 3 T at 4.2 K cycle on a sample longer than 100 mm)
  - RRR (after Cr-plating & heat treatment) > 100

- Statistical Process Control is applied for monitoring the strand performance.



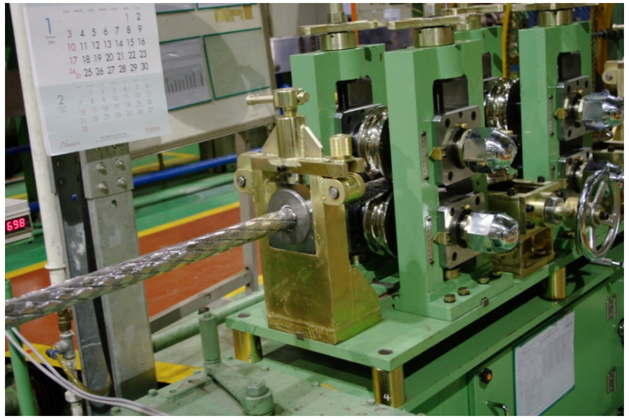


Cable

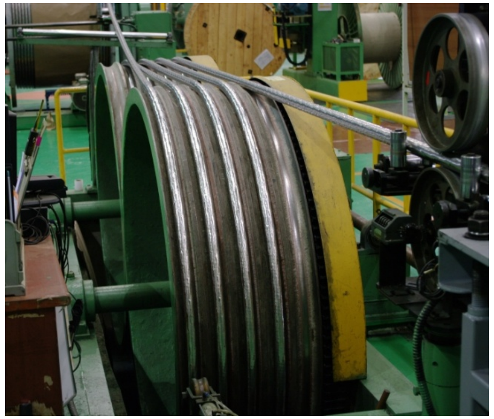
- Cable specifications are built-to-process and call for
  - Cable pattern ((2sc+1Cu)x3x5x5+core)x6
  - Twist pitches 80/140/190/300/420 mm
- 29 Cables have been manufactured, and the final ATPP was cleared in May 2014.
- All the final stage cablings and destructive tests have been witnessed by KODA.

Planetary Cabling Machine

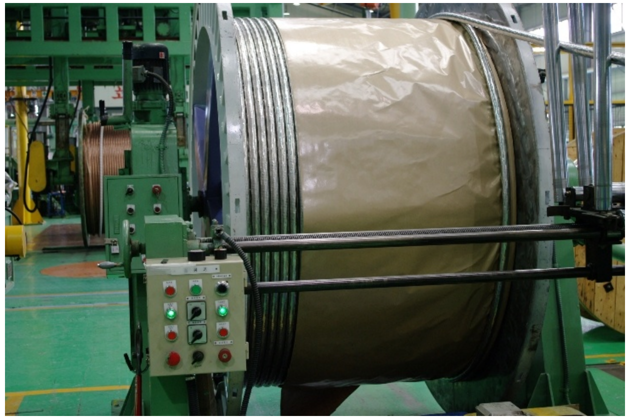
Compaction Rollers



Pulling Capstan



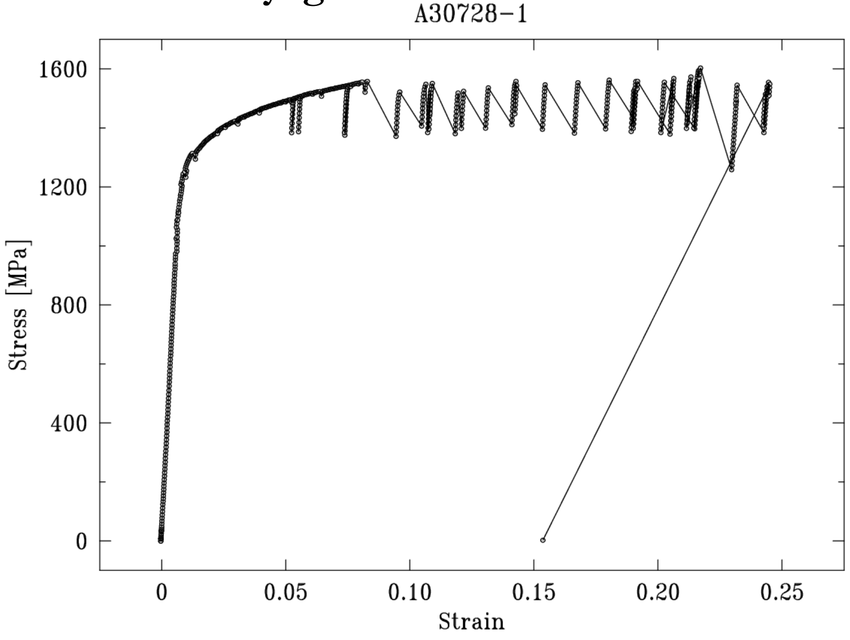
Take-up Spool



Jacket Sections

- Seamless Modified 316LN Tubes have been produced from EAF/ESR melting.
- Low Carbon/Cobalt and High Nitrogen contents were strictly controlled during production.
- The requirement for the cryogenic tensile tests (elongation > 20 %) is tough to satisfy.
- POSCO Specialty Steel completed the production of ~20 km length Jacket Sections successfully and the final ATPPs were cleared in June, 2012.

Cryogenic Tensile Test



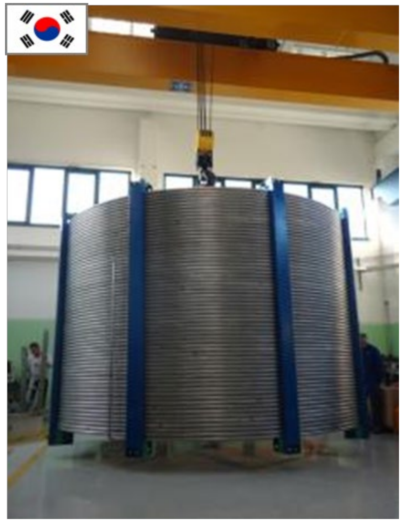
Hot Extrusion

Final Product

Highlights of Jacketing

- Conductor specifications are built-to-process and call for
  - Seamless Jacket Sections, butt-welded together
  - Cable pulled through jacket assembly + compaction
  - Conductor outside/inside diameter 43.7/39.7 mm
- 29 conductors were fabricated successfully and the final Hold Point was cleared in September, 2014.

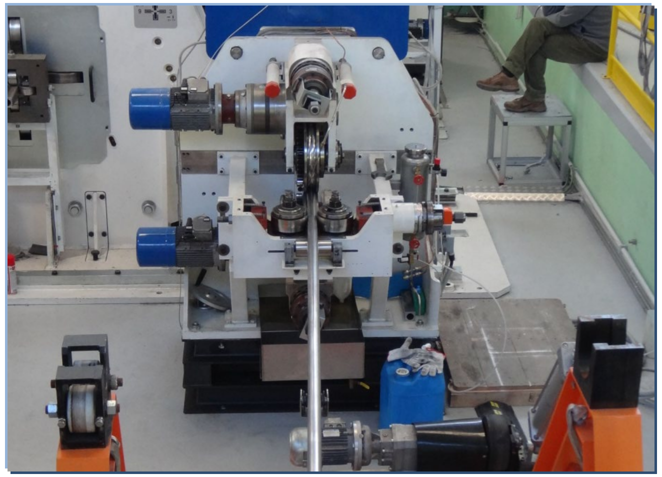
Spooled Conductor



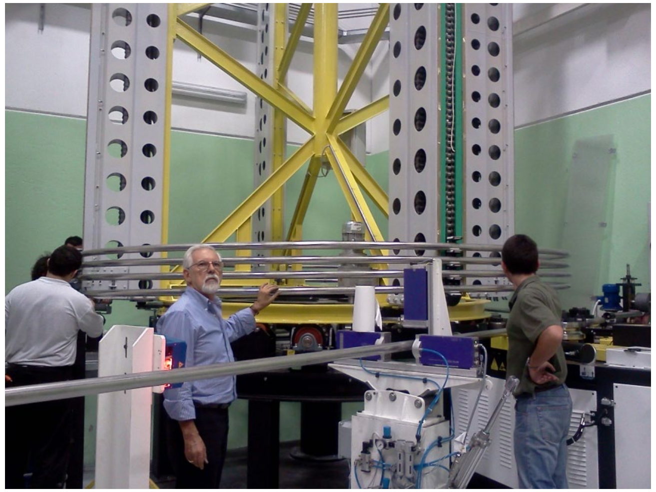
Cross Section

Orbital Butt Welding

Compaction

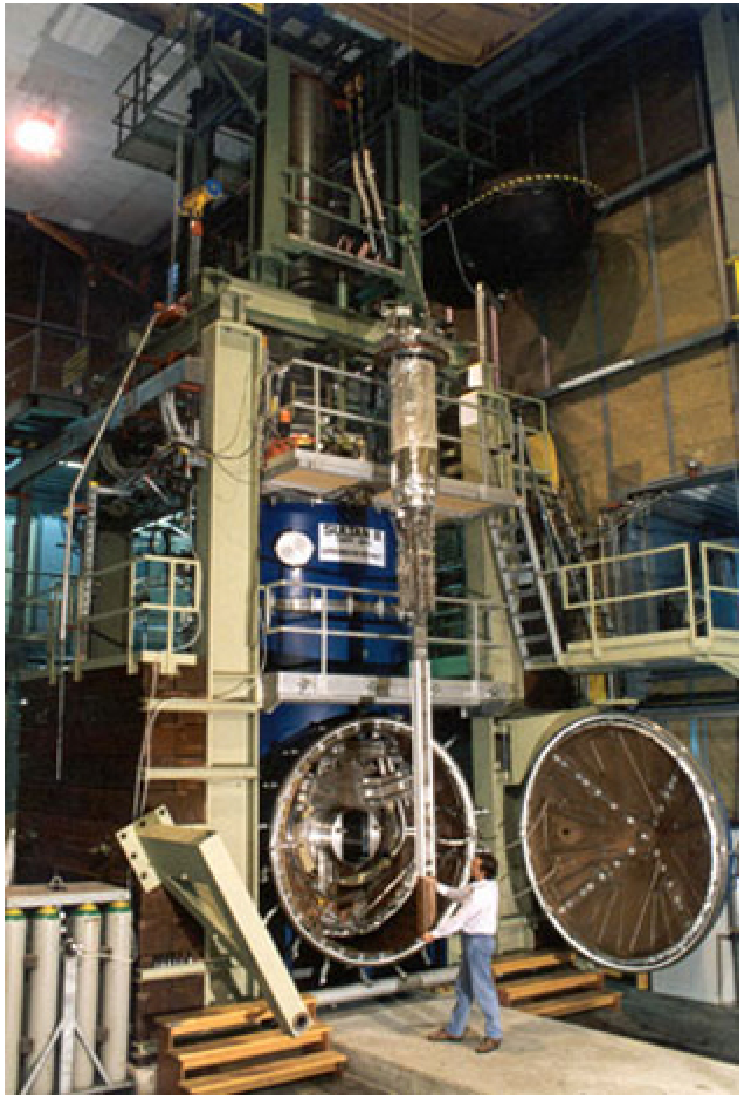


Spooling



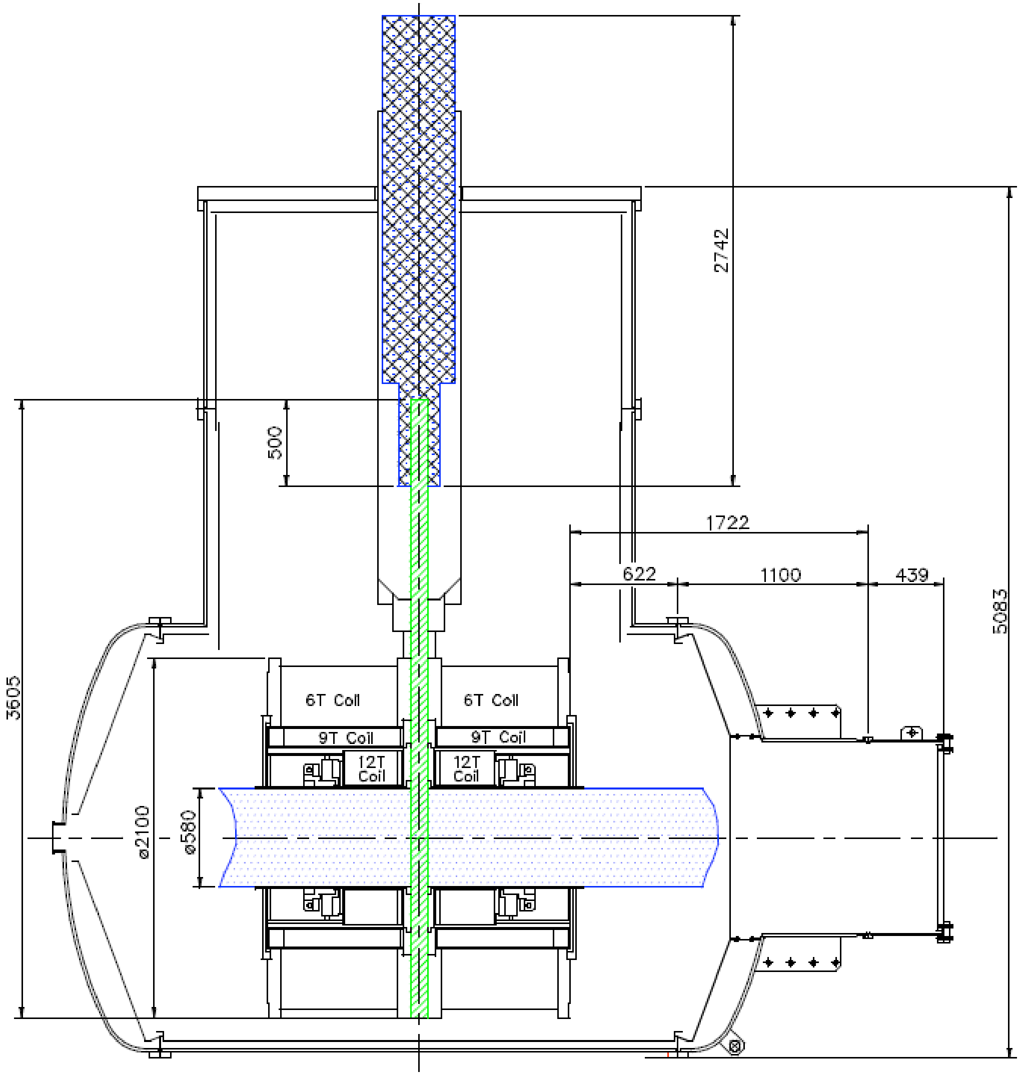
Qualification and Final Acceptance of the Conductor (SULTAN Test)

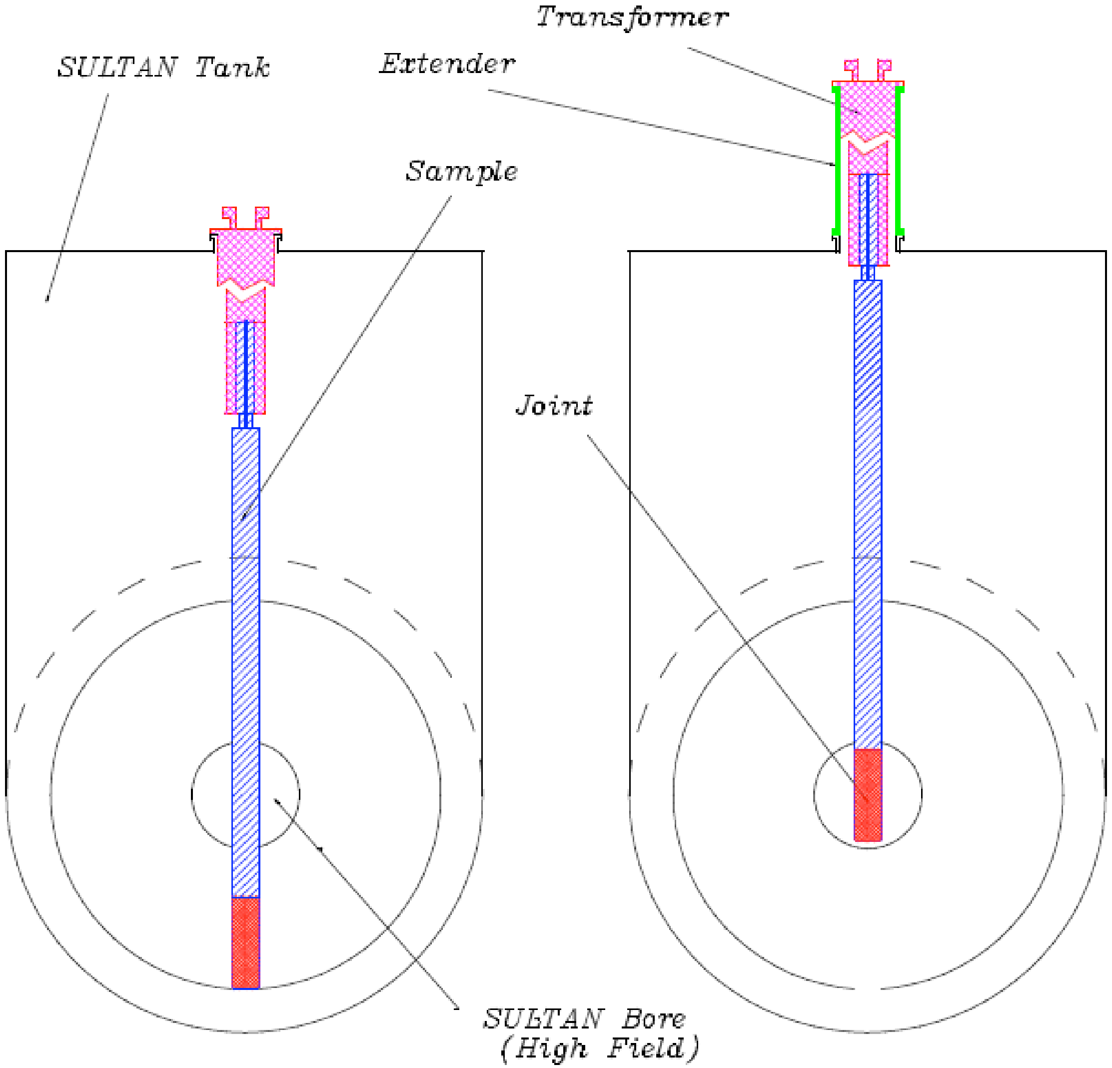
- SULTAN is the abbreviation of "SUpraLeiter Test Anlage"



Background Magnet

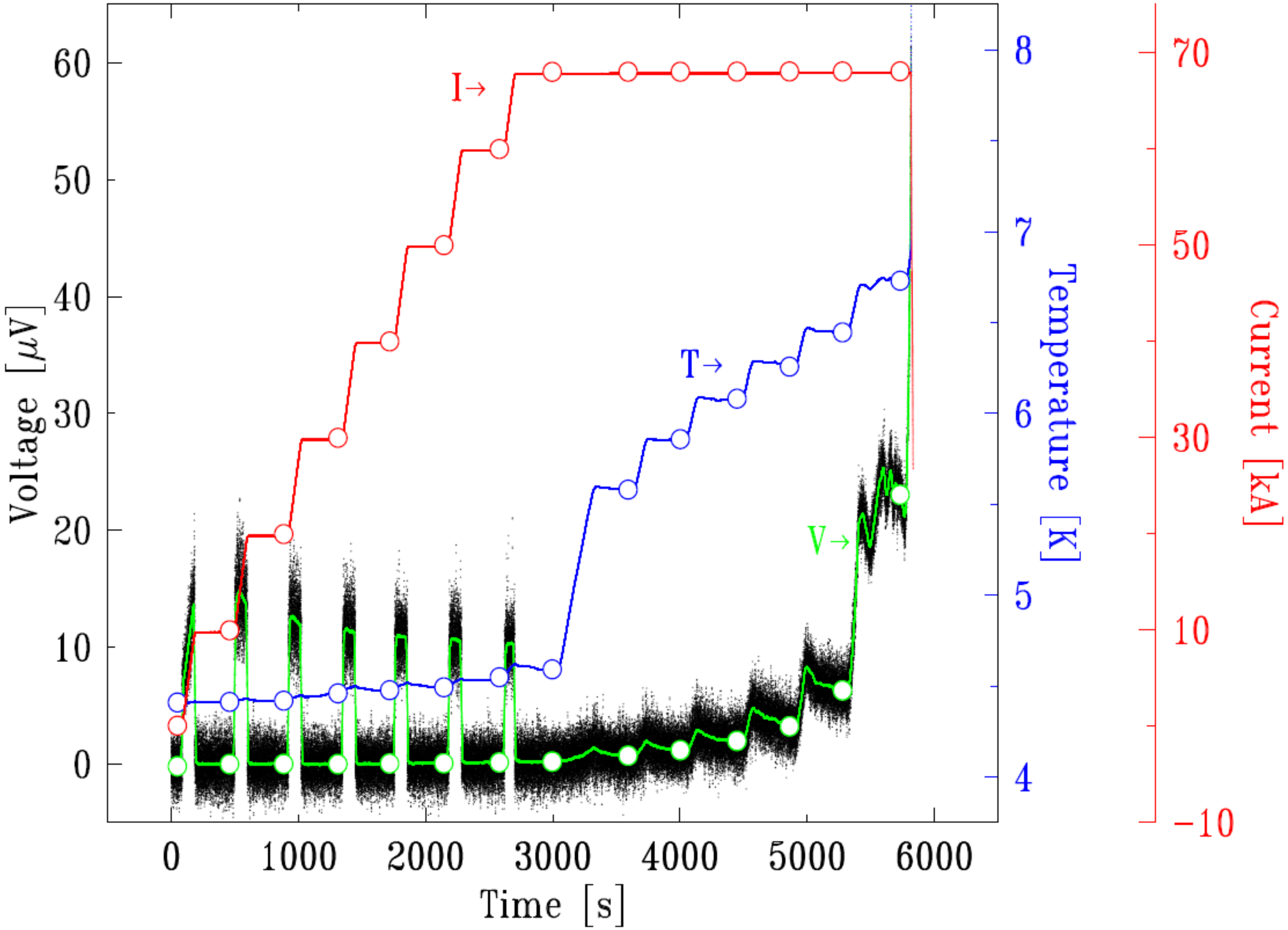
- Magnetic field is generated by three concentric pairs of superconducting split coils





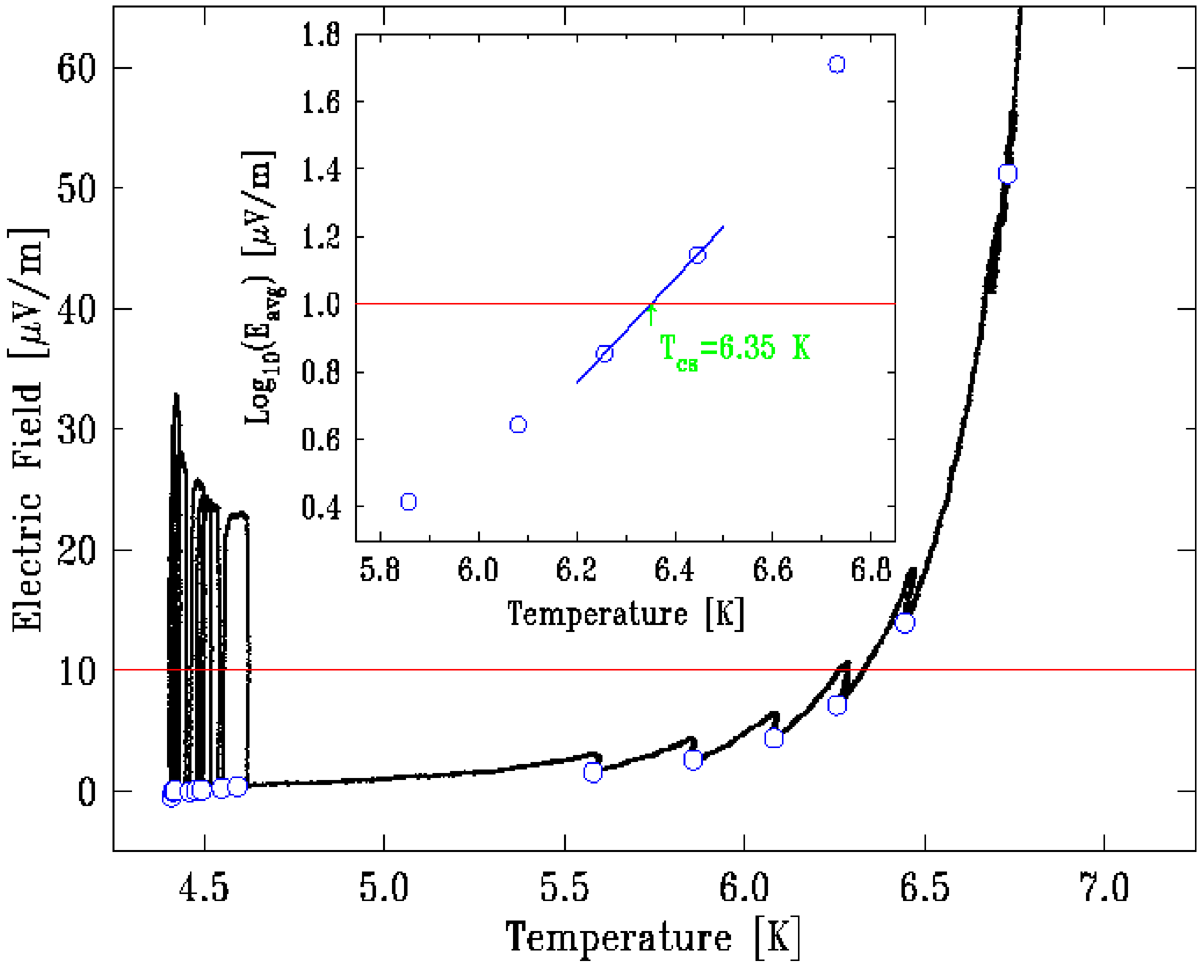
SULTAN Test (for ITER TF Conductor)

- Measurement of Current Sharing Temperature (T_{cs}) of short (~**3.6 m**) , full sized conductor sample @**68 kA, 10.78 T**.
- Ramping up the current, step by step up to **68 kA**, followed by ramping up the temperature, step by step till the conductor quenches (T_{cs} is defined as the temperature @ E = 10 μV/m.).



T_{cs} Assessment

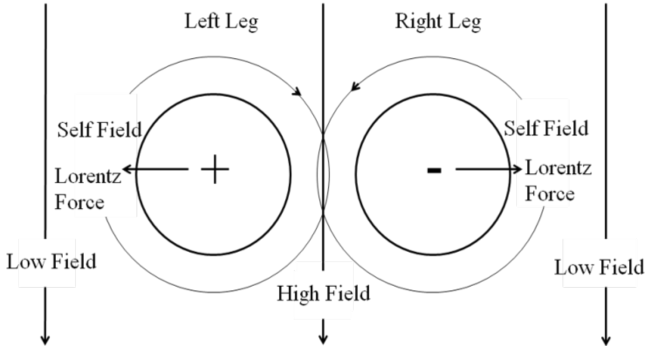
• Open circles are average point used for the calculation of T_{cs}. Black points are centered moving average of the electric field. Inset is the illustration of the T_{cs} assessment for Right Leg at N=1. Red lines are criteria for the critical electric field.

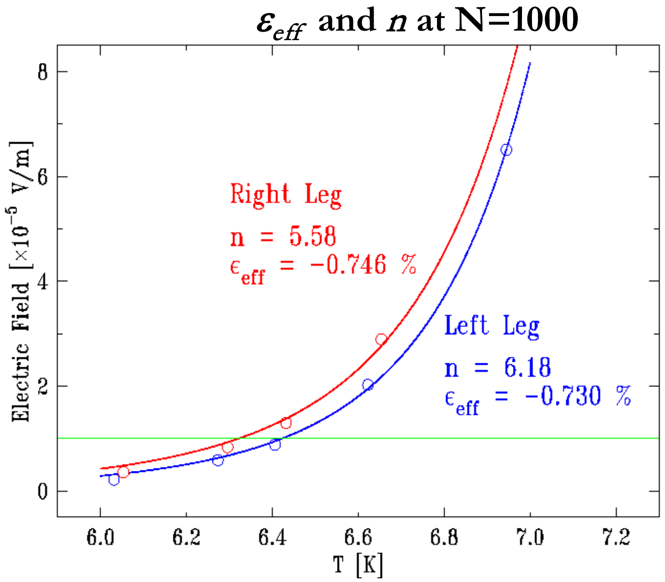


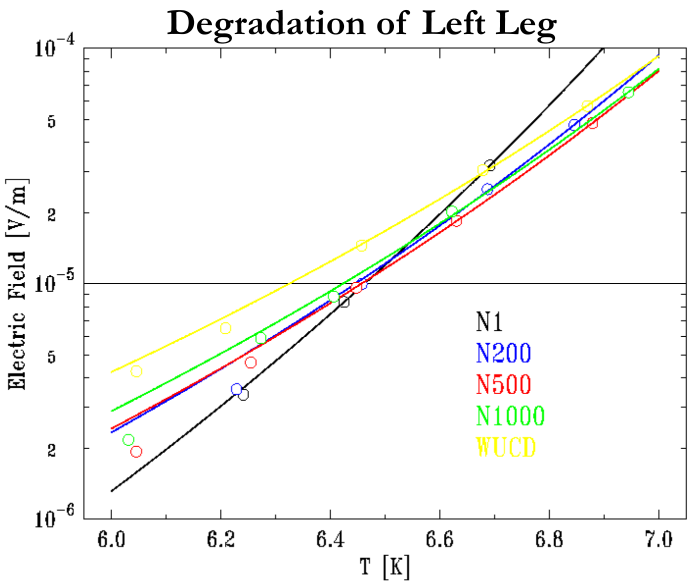
ε_{eff} and n value (TFKO8)

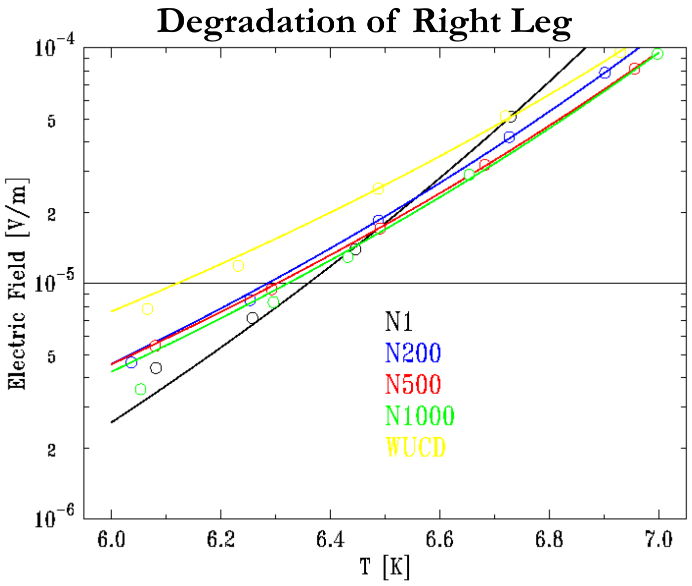
TFJA5

$$E(T) = \frac{E_0}{LA} \iint_{LA} \left[\frac{J_{op}}{J_C(B,T,\varepsilon_{eff})} \right]^n dAdz$$









$J_C(B,T,\varepsilon)$ for Nb_3Sn Strand

- See more details in IEEE Trans. Appl. Supercond., vol 19, no.3, pp. 1521-1524, Jun. 2009.

$$J_C = \frac{C}{B} s(\varepsilon)(1-t^{1.52})(1-t^2)b^p(1-b)^q \qquad t = \frac{T}{T^*_{C0}(\varepsilon)} \qquad b = \frac{B}{B^*_{C2}(T,\varepsilon)}$$

$$B^*_{C2}(T,\varepsilon) = B^*_{C20\,\max} s(\varepsilon)(1-t^{1.52})$$

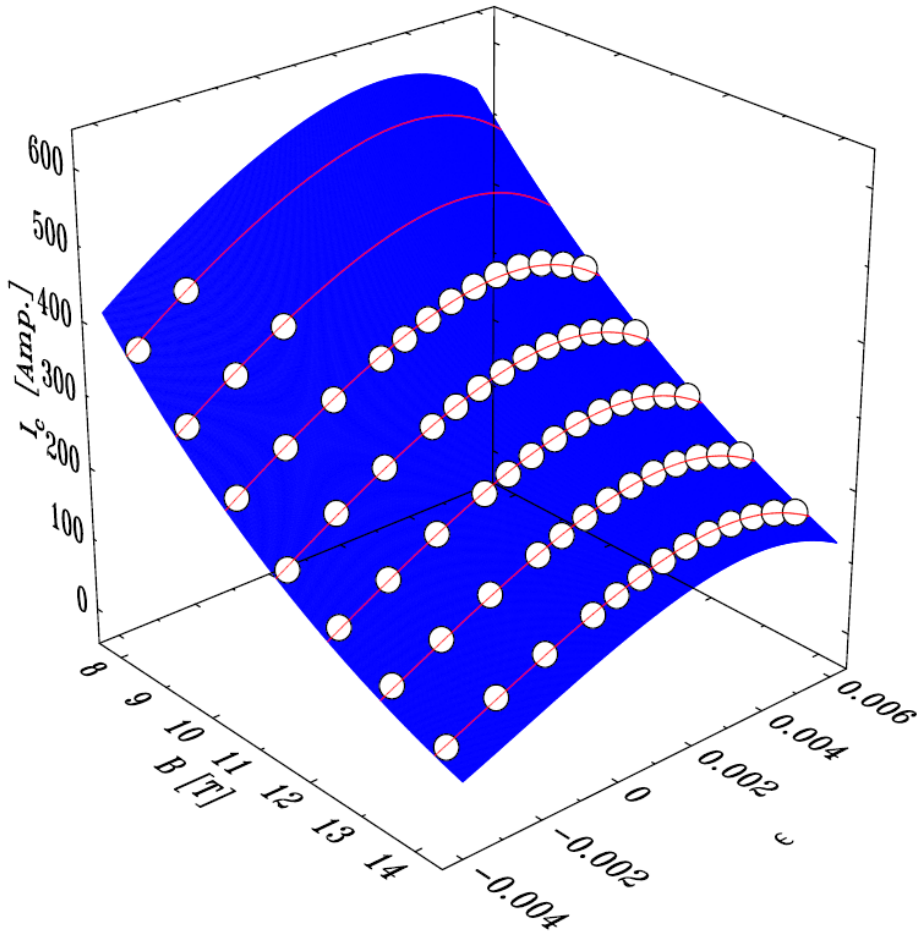
$$T^*_C(B,\varepsilon) = T^*_{C0\,\max}[s(\varepsilon)]^{\frac{1}{3}}\left(1-\frac{B}{B^*_{C2}(0,\varepsilon)}\right)^{\frac{1}{1.52}}$$

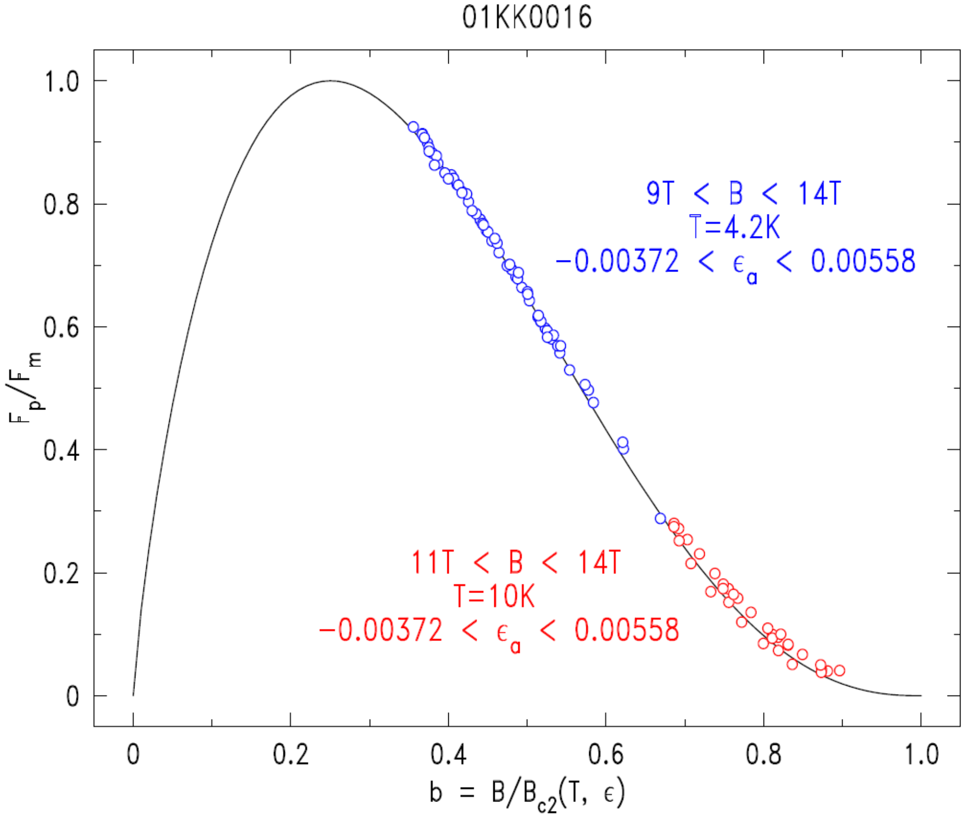
$$s(\varepsilon) = 1 + \frac{C_{a1}\left(\sqrt{\varepsilon_{sh}^2+\varepsilon_{0,a}^2}-\sqrt{(\varepsilon-\varepsilon_{sh})^2+\varepsilon_{0,a}^2}\right)-C_{a2}\varepsilon}{1-C_{a1}\varepsilon_{0,a}}$$

$$\varepsilon_{sh} = \frac{C_{a2}\varepsilon_{0,a}}{\sqrt{C_{a1}^2-C_{a2}^2}}$$

$$F_P = J_C(B,T,\varepsilon)\,B$$

$J_C(B,T,\varepsilon)$ for 01KK0016 (Representative of Korean Strand)

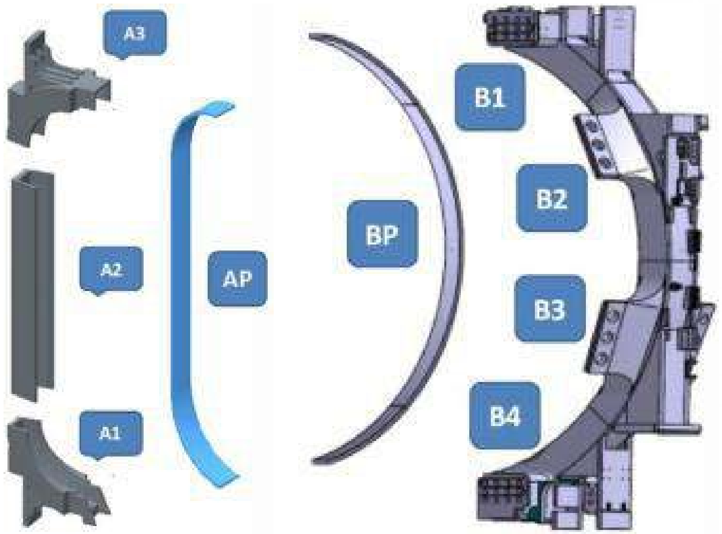




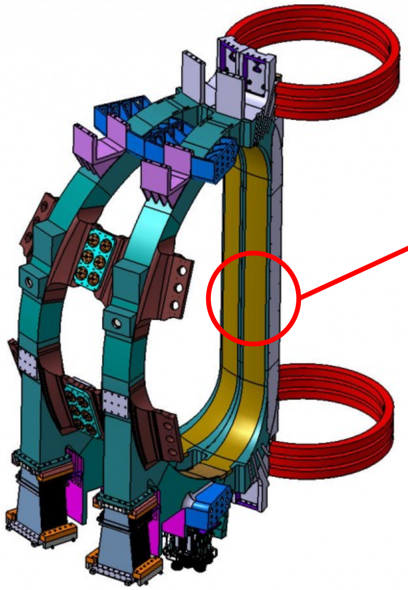
C	B_{c20m} (T)	T_{c0m} (K)	Ca_1	Ca_2	$\varepsilon_{0,a}$ (%)	ε_m (%)	p	q
35986	31.58	15.95	70.28	32.91	0.44	0.34	0.83	2.49

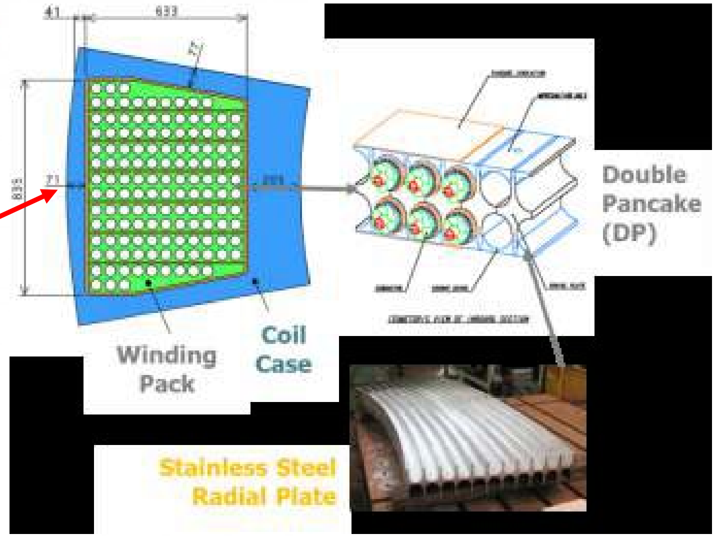
ITER TF Coils

• Each winding pack (WP) comprises 7 double pancakes (DPs), made up of a radial plate with precisely machined grooves into which the CICC is transferred upon heat treatment completion.



TF Coil Structure

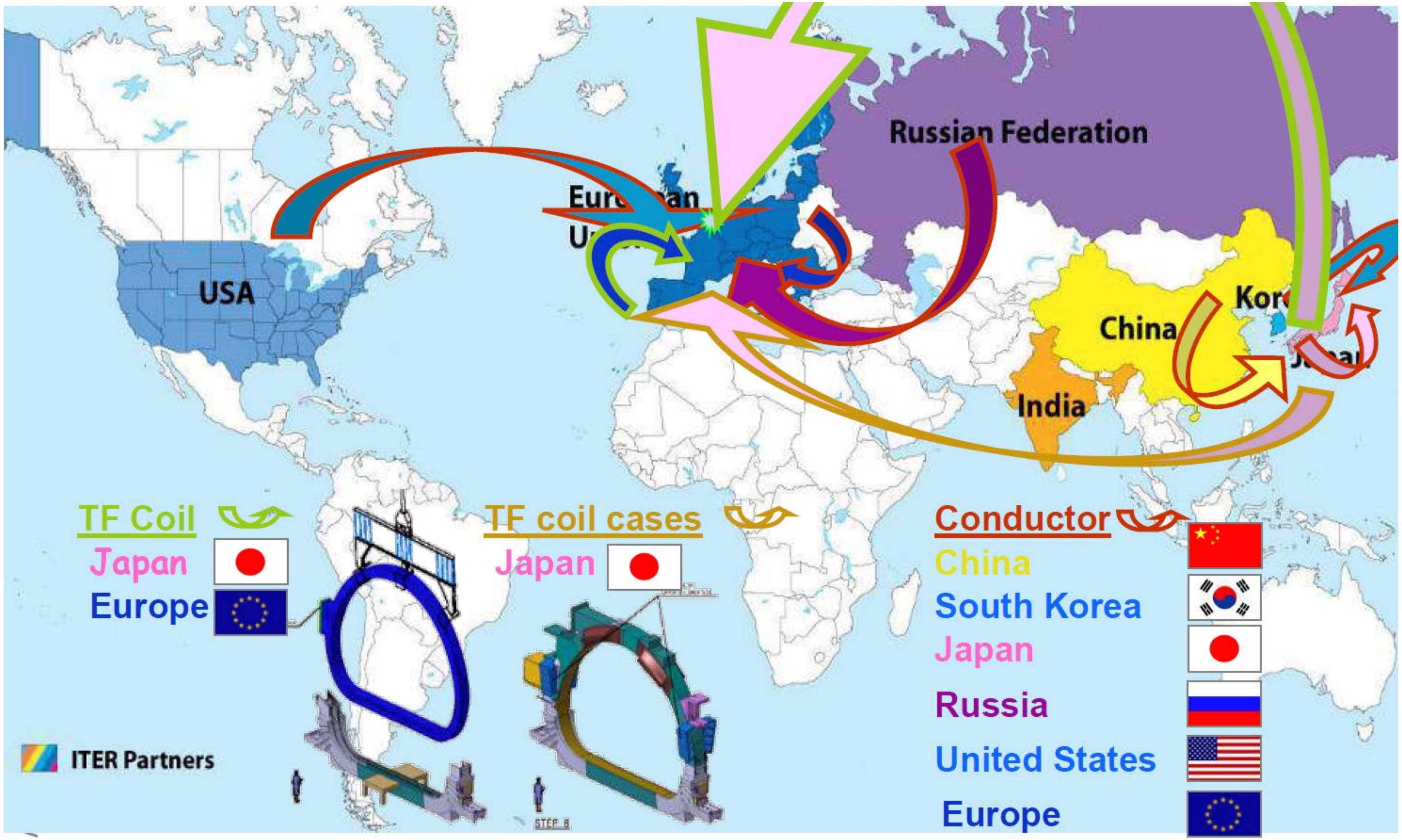

Pair of TF Coils



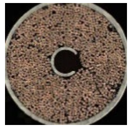
TF Coil Winding Pack

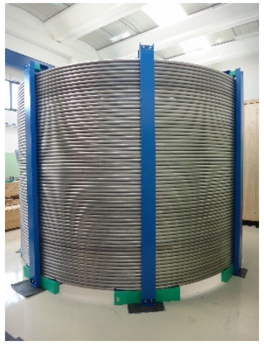
• The TF coil is made up of a winding pack (WP) inserted inside a thick coil case consisting of welded, stainless steel segments.

Manufacturing TF Coils: Complicated Interfaces



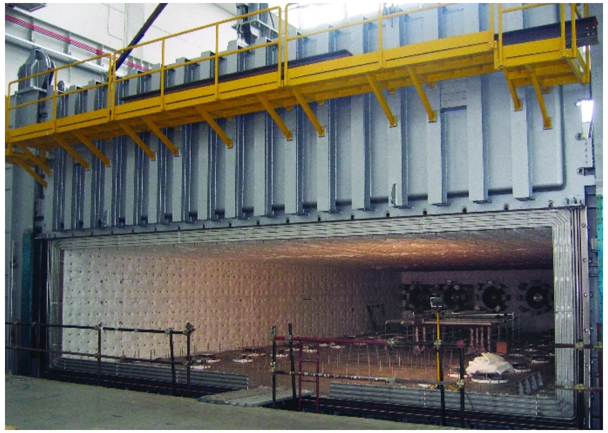
ITER TF DP Manufacture – 1





Conductor Spool

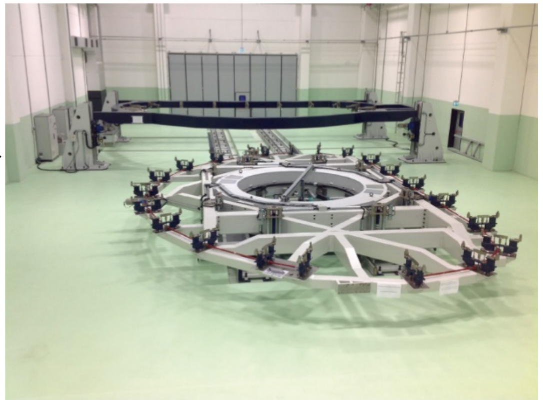
DP Winding



DP Heat Treatment

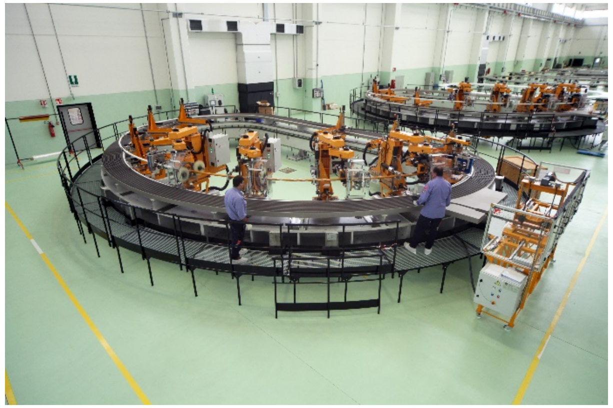
Radial Plate
Machining/Welding

Radial Plate Assembly



Transfer of DP into
Radial Plate

ITER TF DP Manufacture – 2



DP Turn Insulation
inside Radial Plate

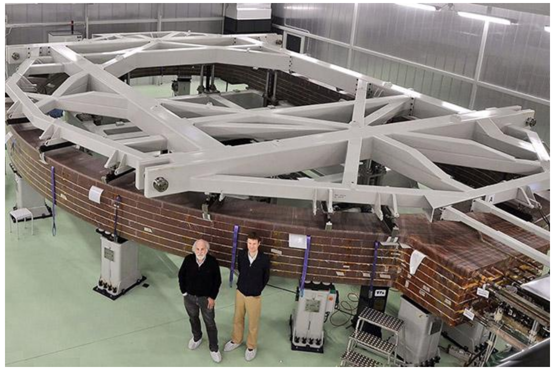
Cover Plate
Welding

DP Ground Insulation

DP Loading into Vacuum
Impregnation Mold

Impregnated DP

ITER TF WP Manufacture



7 DP Stacking

WP Insulation

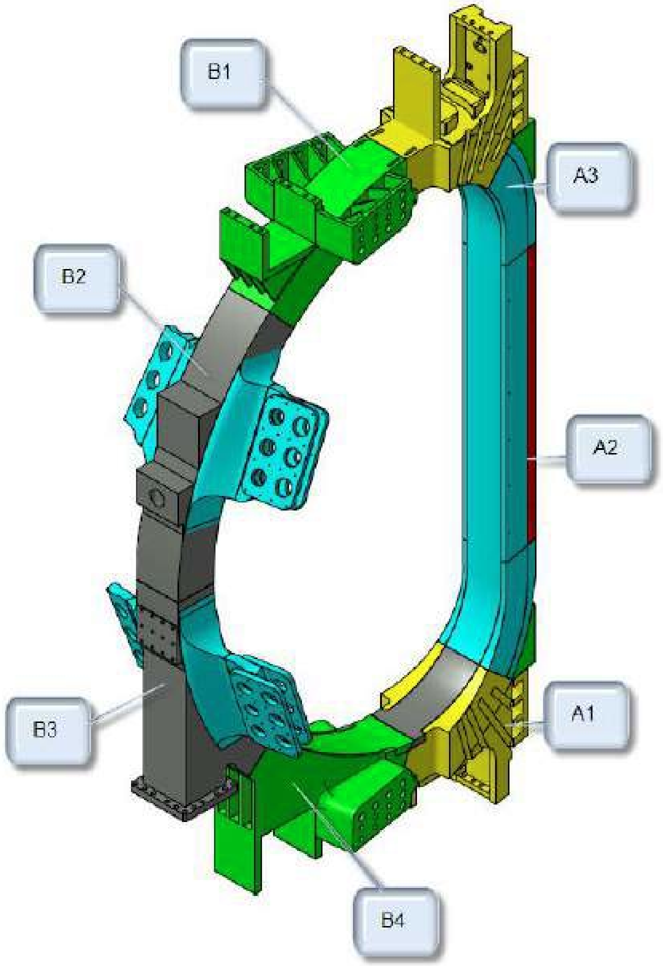
Terminal Preparation

Vacuum Pressure Impregnation

Completed WP

TF Coil Structure – 1

TF Coil Structure Breakdown



• TF structure calls for mass production of ～ 4500 tons of large, high-strength, 316LN steel components.

• Components are made of TIG welded assemblies of forged plates up to 200 mm and require tight deformation control to achieve final shape.

• Three suppliers have been selected: MHI and Toshiba (Japan) and HHI (Korea).

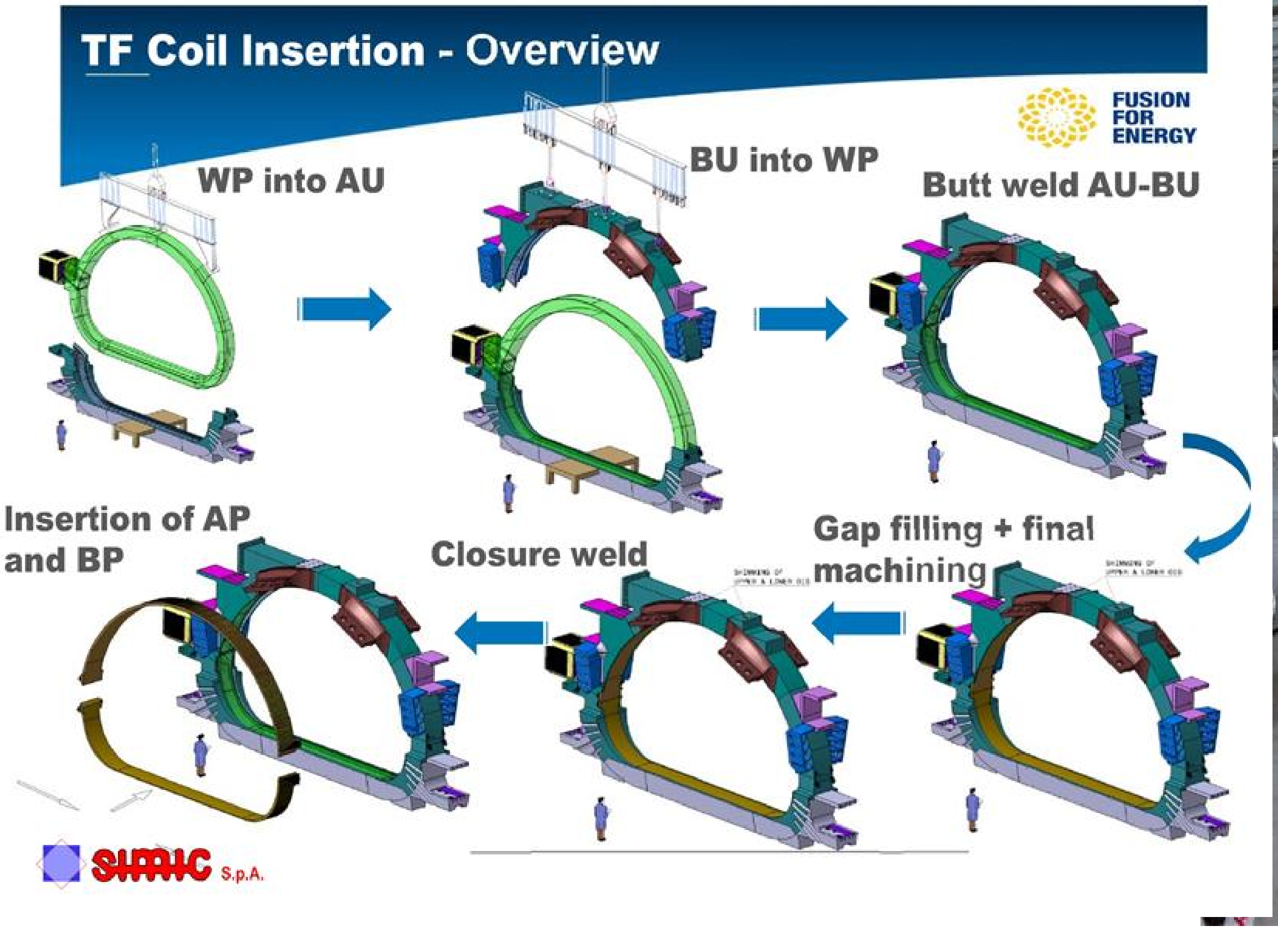
• Series production of AU, AP, BU and BP sub-assemblies are underway at all 3 suppliers.

TF Coil Structure – 2

At Hyundai Heavy Industries in Ulsan, Korea, the outboard leg sub-assemblies were first fitted together to verify manufacturing precision

The two legs were successfully matched in fitting tests at Hyundai Heavy Industries in December

ITER TF Coil

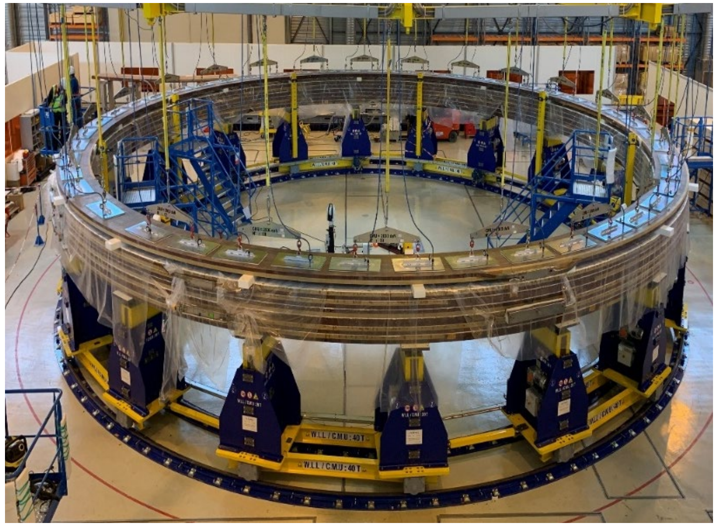


Europe's first Toroidal Field coil has been inserted in its case

Manufacturing ITER PF Coil

Technicians preparing for Vacuum Pressure Impregnation on PF6, ASIPP, Hefei, China

Completion of Vacuum Pressure Impregnation on 8 DP for PF1, Efremove, Russia



7 DPs of PF5 are stacked in the stacking area

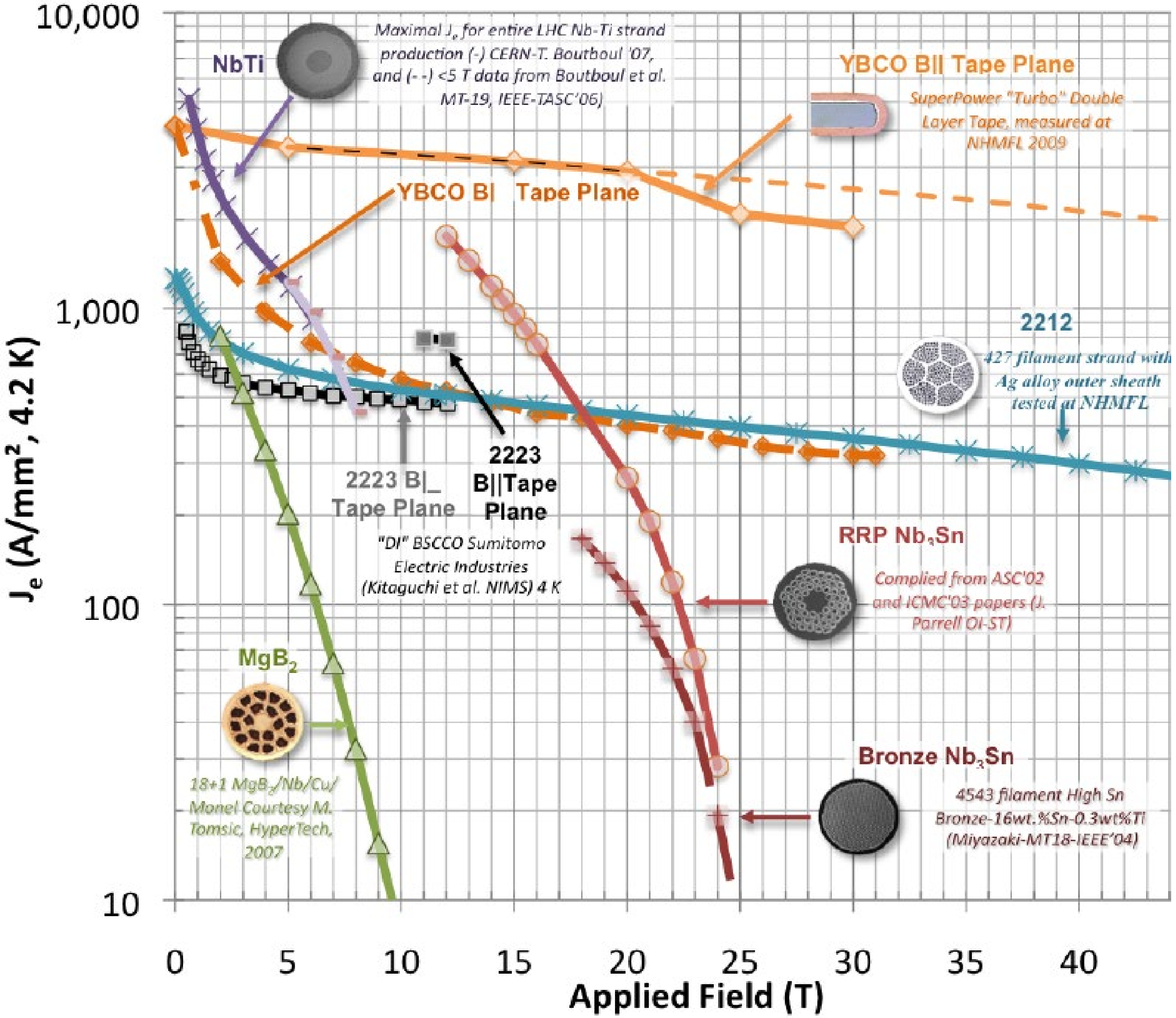
Manufacturing ITER CS Coil

Winding of CS Coil, GA, USA

The module during completion of helium piping

Transfer of central solenoid module coil to the heat treatment furnace

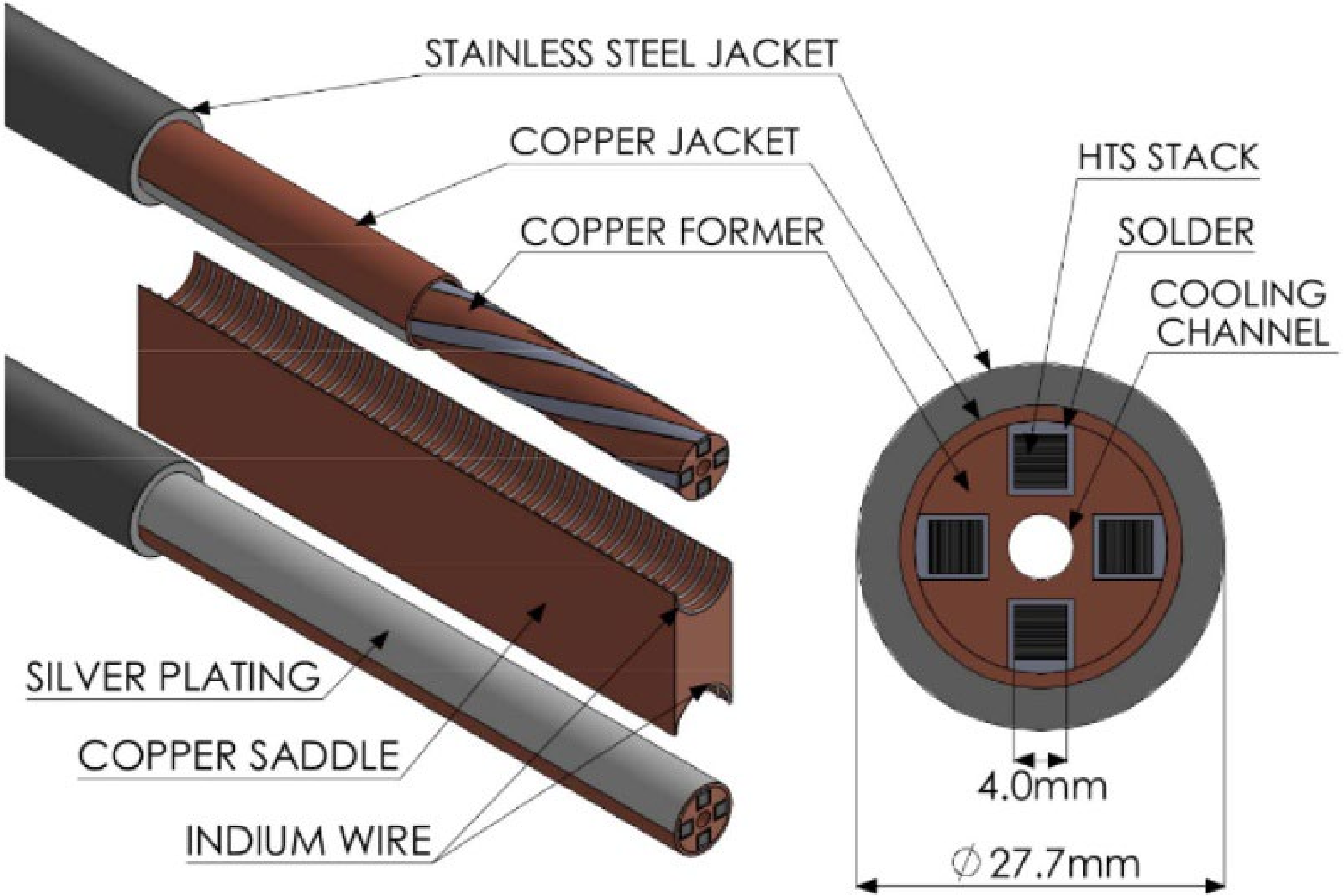
Digression: High Tc Superconductor



Superconducting Tokamaks

Parameter		SPARC	EAST	KSTAR
R	m	1.85	1.70	1.80
a	m	0.57	0.40	0.50
B_T	T	12.2	3.5	3.5
I_p	MA	8.7	1.0	2.0
$P_{aux,max}$	MW	25	28	16
$P_{flattop}$	s	10	1000	300
P_{fus}	MW	140		
Q		11		

Digression: High Tc Superconductor



VIPER cables for high-field superconducting fusion magnets.
(Right) HTS stacks are VPI soldered into channels in a twisted copper former containing a central cooling channel and surrounded by roll-formed copper and optional stainless-steel jacket.
(Left) A cut-away showing that joints are easily created by silver plating the cable ends and compressing them into a copper saddle.

Digression: High Tc Superconductor (SPARC)