The Neutral Beam Test Facility for the ITER project

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ITER

Mega-Science Project among 7 Members:

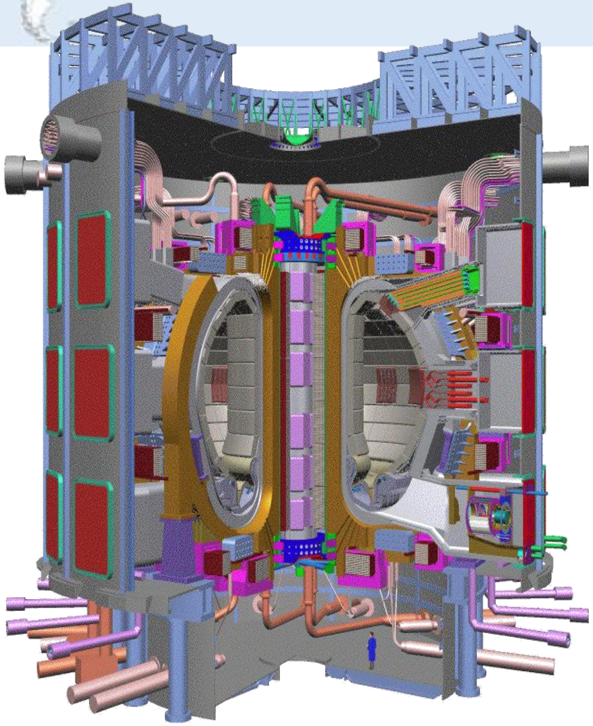
China, EU, India, Japan, Korea, Russia & US

Designed to produce 500 MW of fusion power for an extended period of time

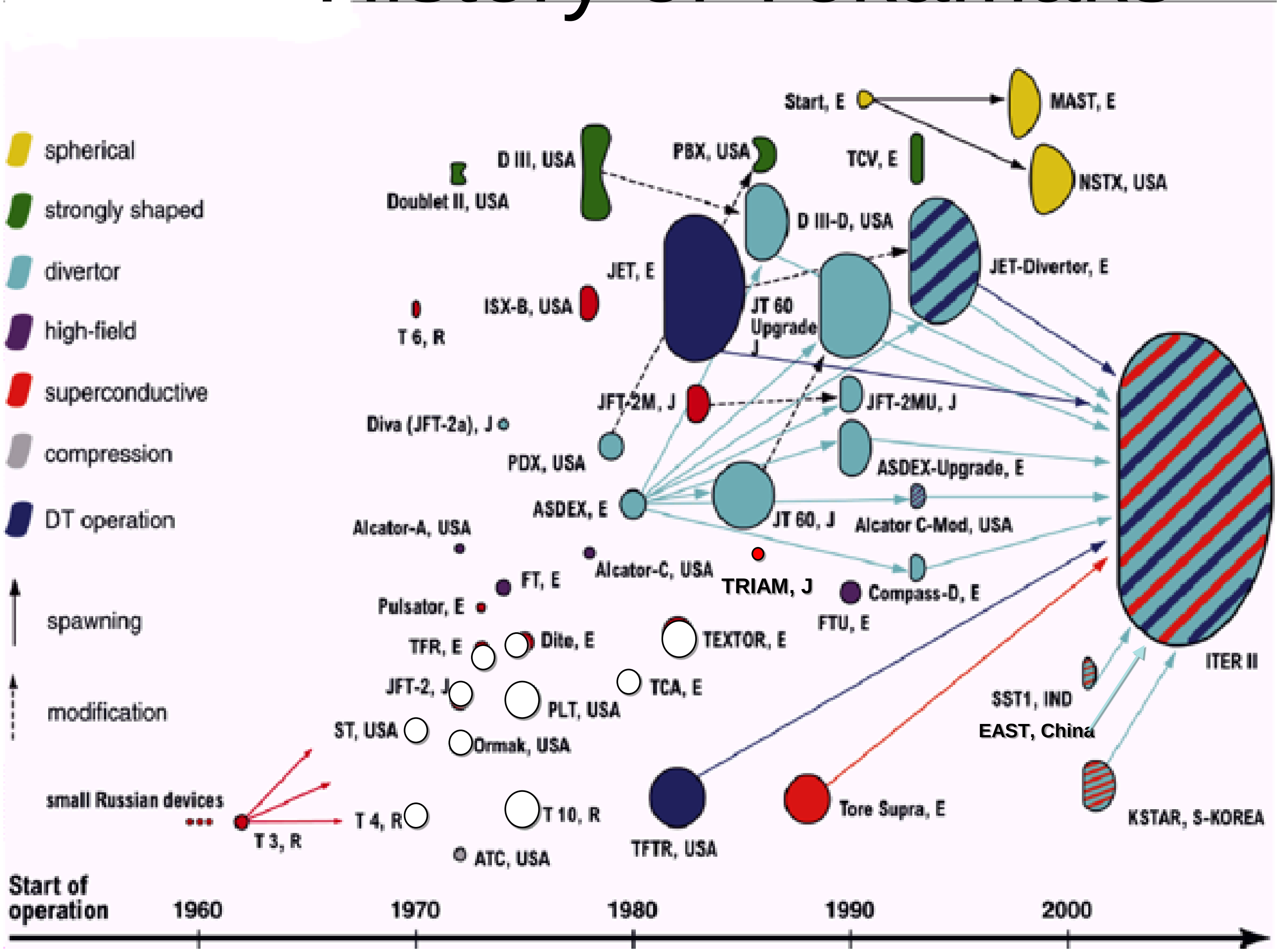
Will bring together most key technologies needed for future fusion power plants

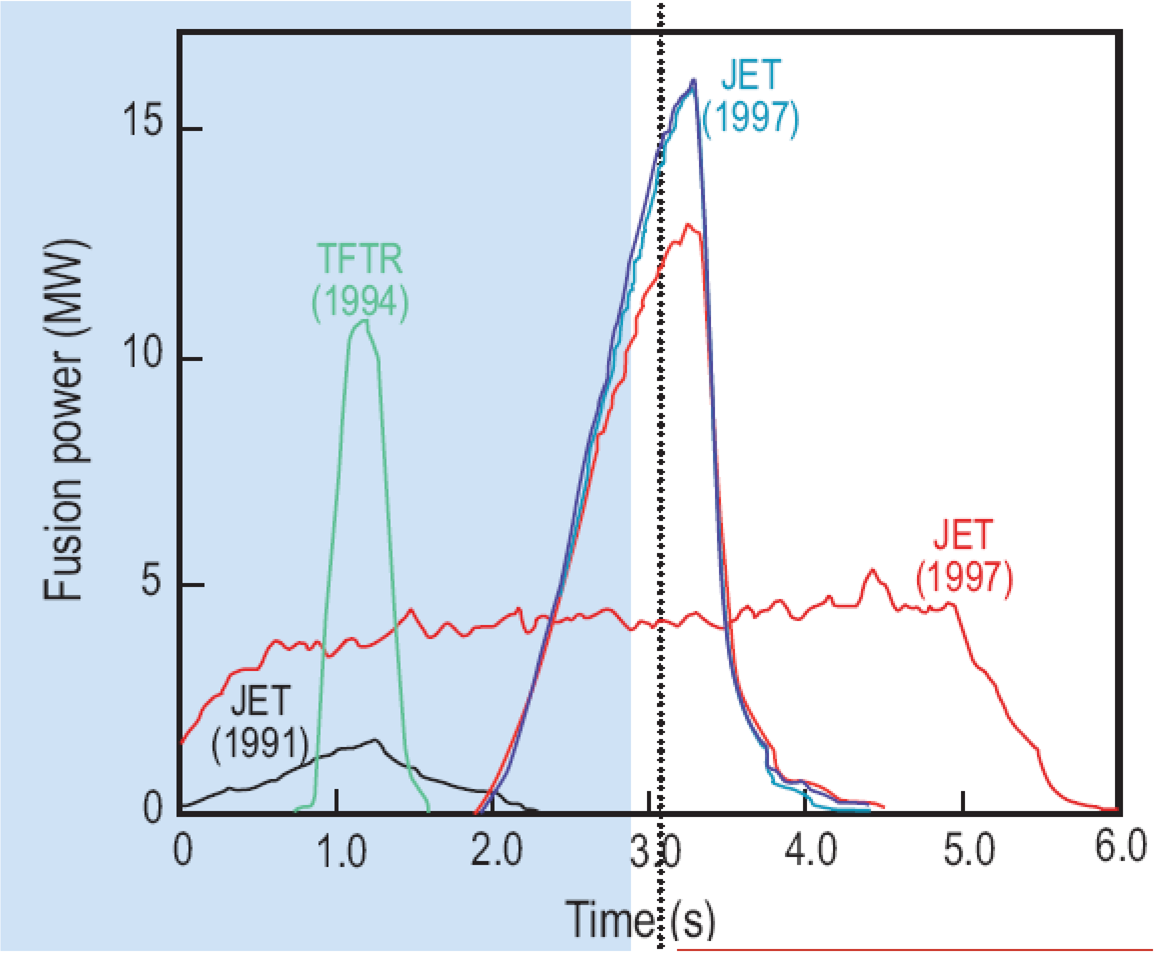
10 years construction, 20 years operation

Cost: ~5 billion Euros for construction, and ~5 billion for operation and decommissioning



History of Tokamaks




Fusion power (MW)
Time (s)
0
5
10
15
0
1.0
2.0
3.0
4.0
5.0
6.0
TFTR (1994)
JET (1997)
JET (1997)
JET (1991)

Lawson: The Fusion Equation

- Confinement Time

$$\tau = \frac{\text{Energy in the Plasma}}{\text{Energy lost per sec}}$$

$$n = 1\text{-}2 \text{ x } 10^{20} \text{ particles} \cdot \text{m}^{-3}$$

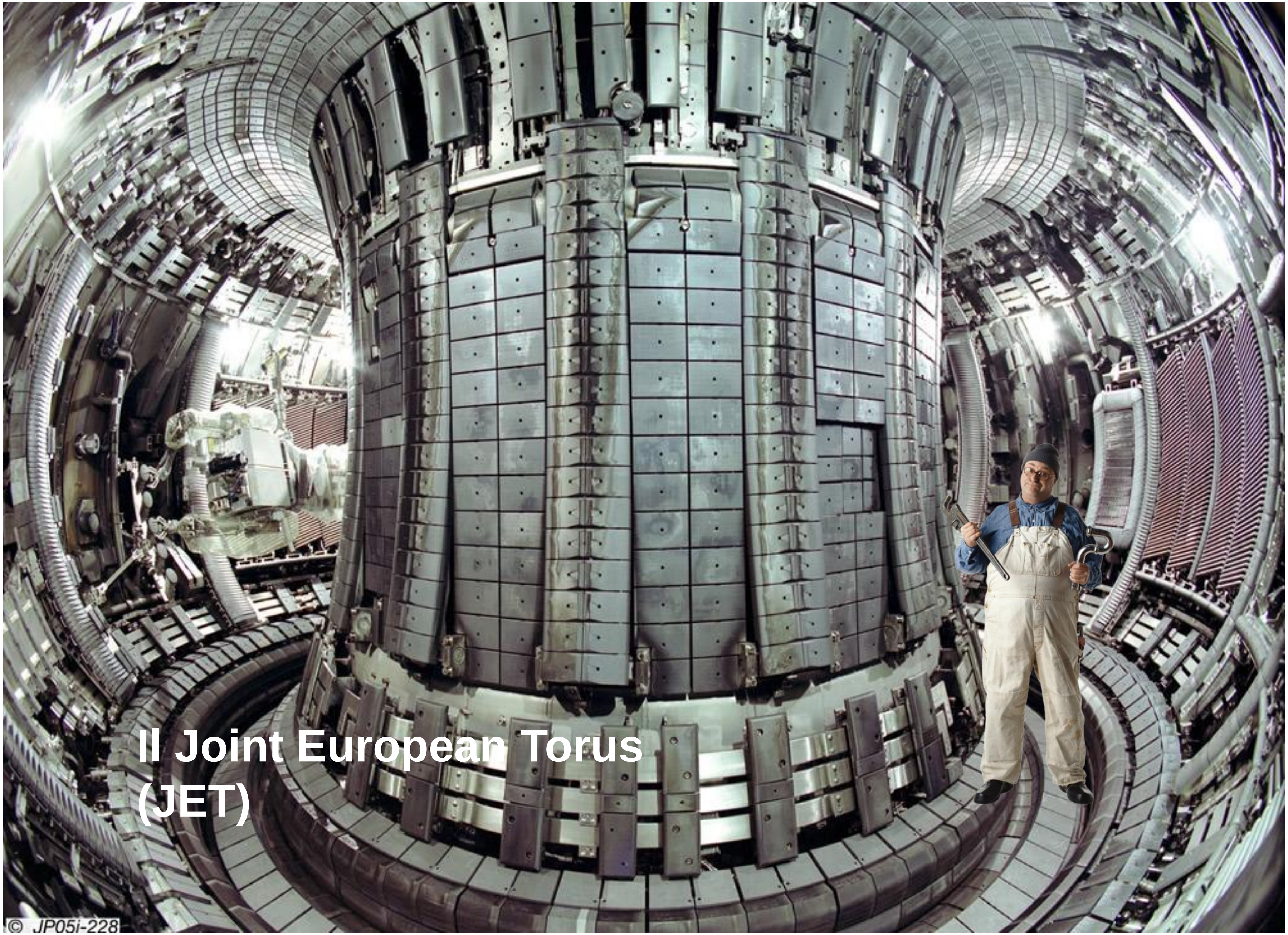
- Density

This image cannot currently be displayed.

$$100\text{-}200 \times 10^{6} \text{ Kelvin}$$

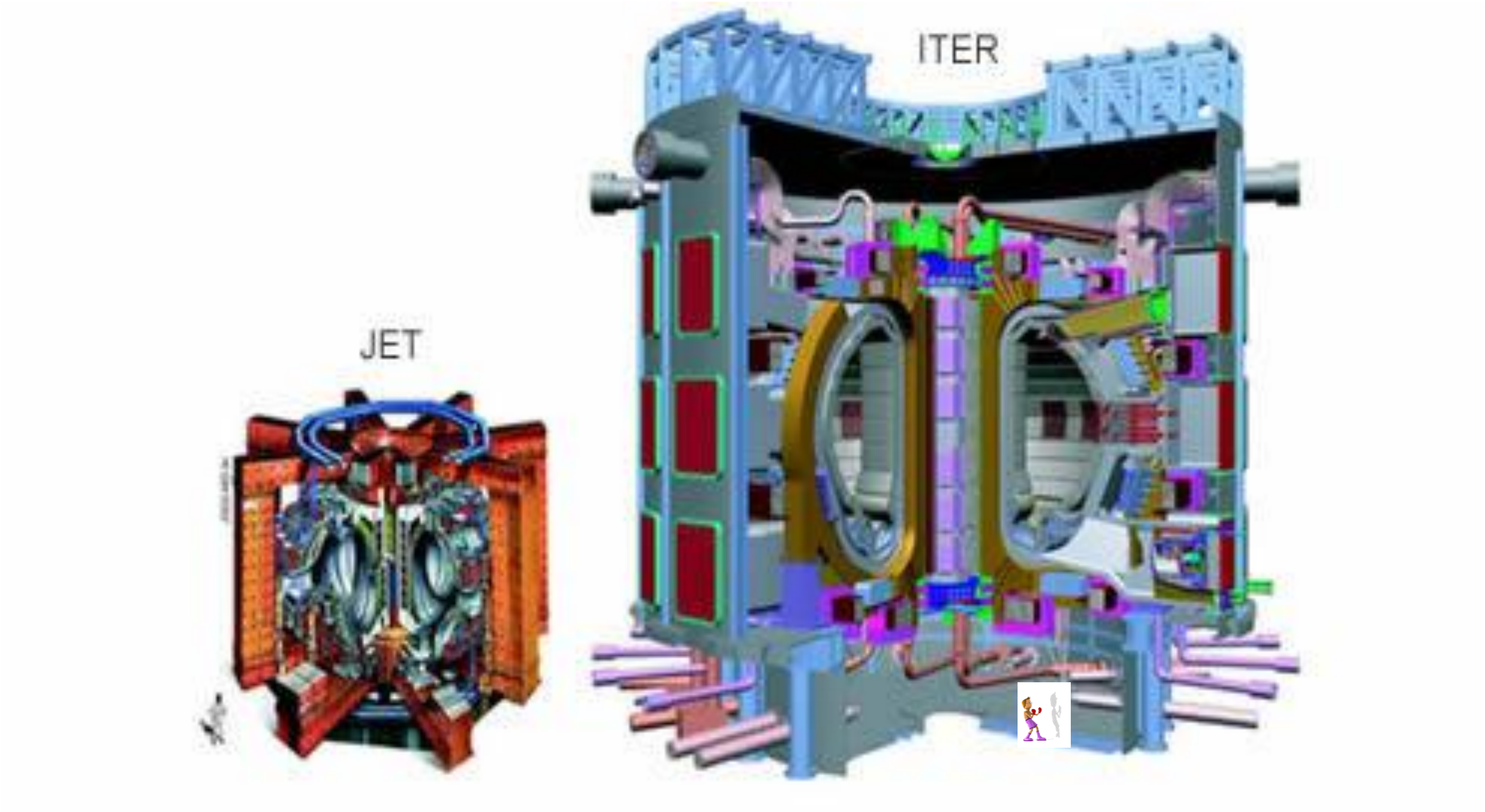
- Temperat

$$\tau \cdot n \cdot T \geq 3 \cdot 10^{28} \frac{Kelvin}{m^{3} \cdot \sec}$$

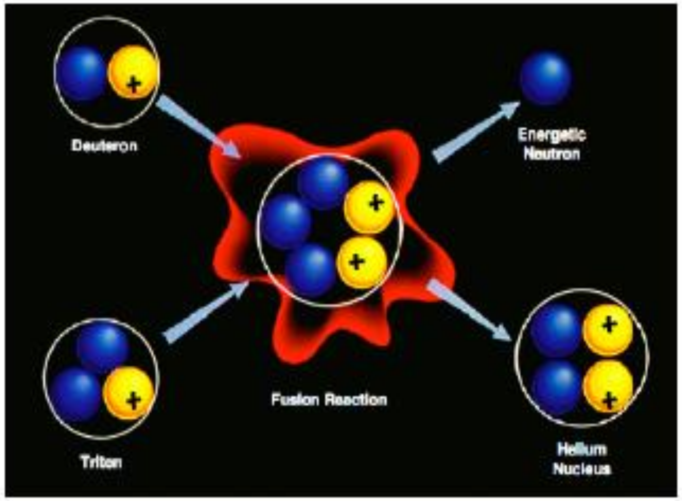


Il Joint European Torus (JET)

JET → ITER



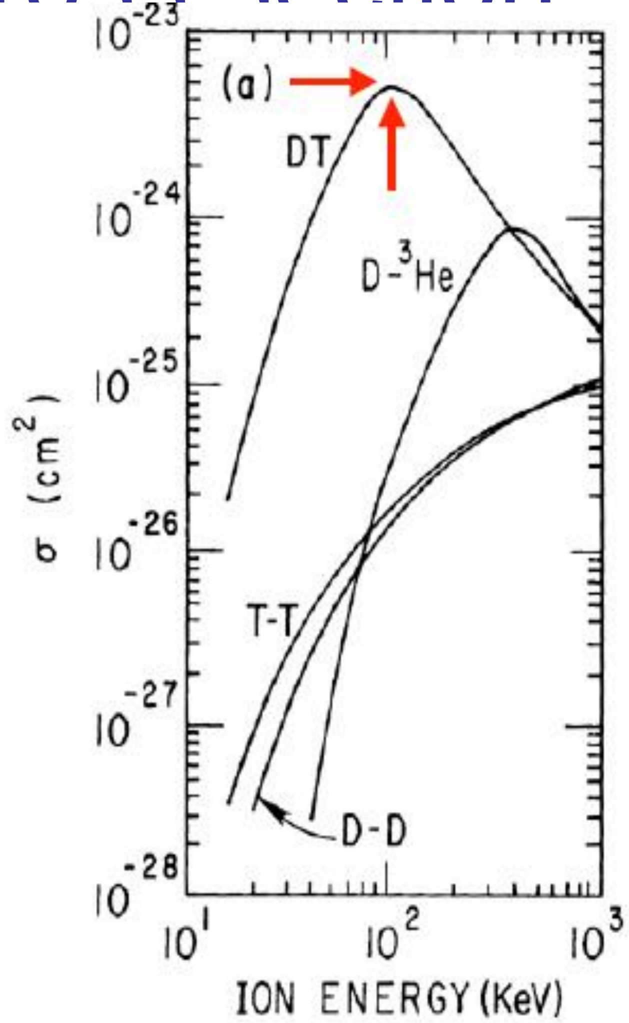
Why D-T: Cross section



The “easiest” fusion reaction uses hydrogen isotopes: deuterium (D) & tritium (T)

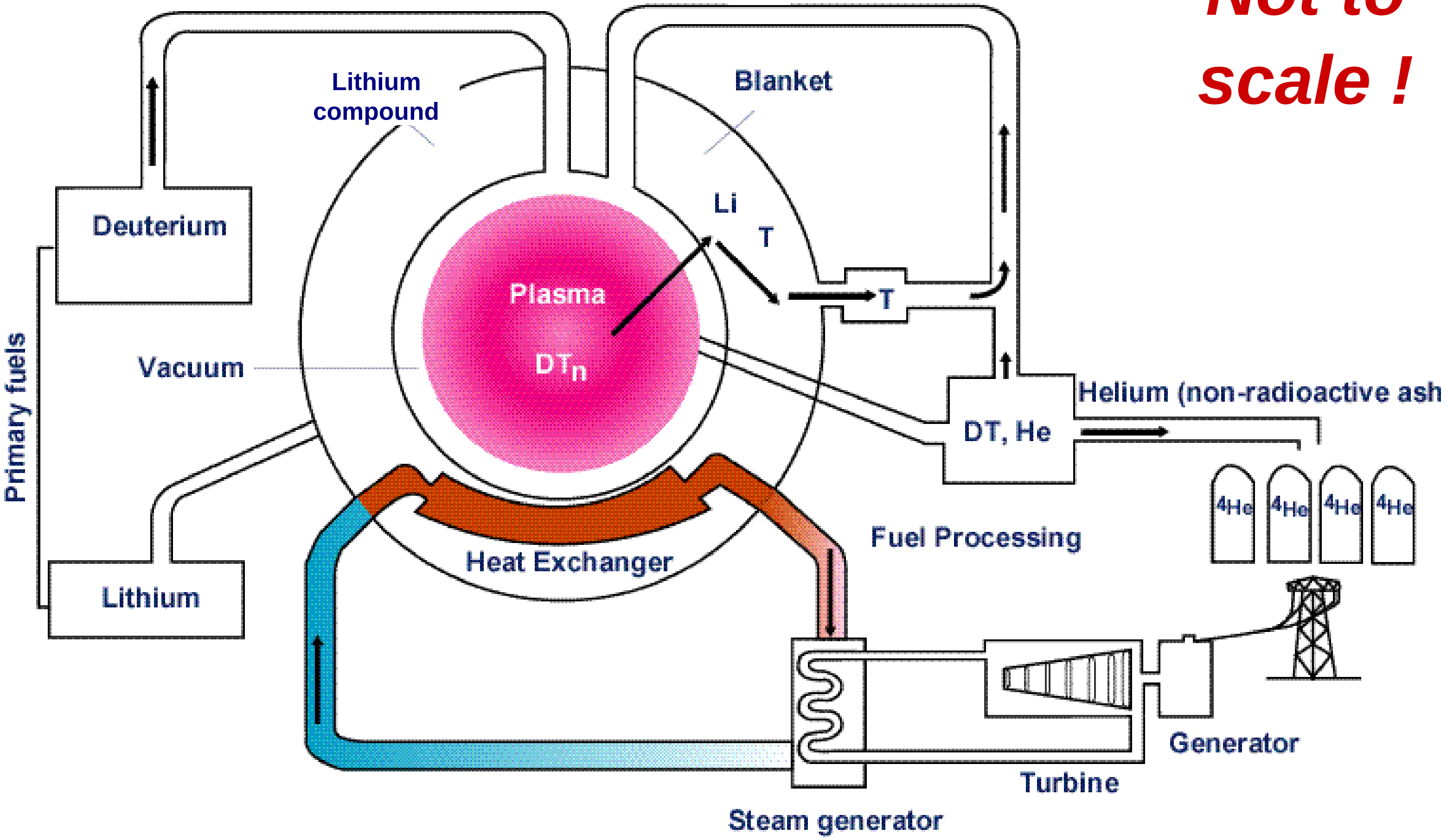
$$ {}_1D^2 + {}_1T^3 \rightarrow \underbrace{\underset{(3.5\ \text{MeV})}{\underset{\uparrow}{{}_2He^4}} + \underset{(14.1\ \text{MeV})}{\underset{\uparrow}{{}_0n^1}}} $$

Energy/Fusion: $\varepsilon_f = 17.6$ MeV



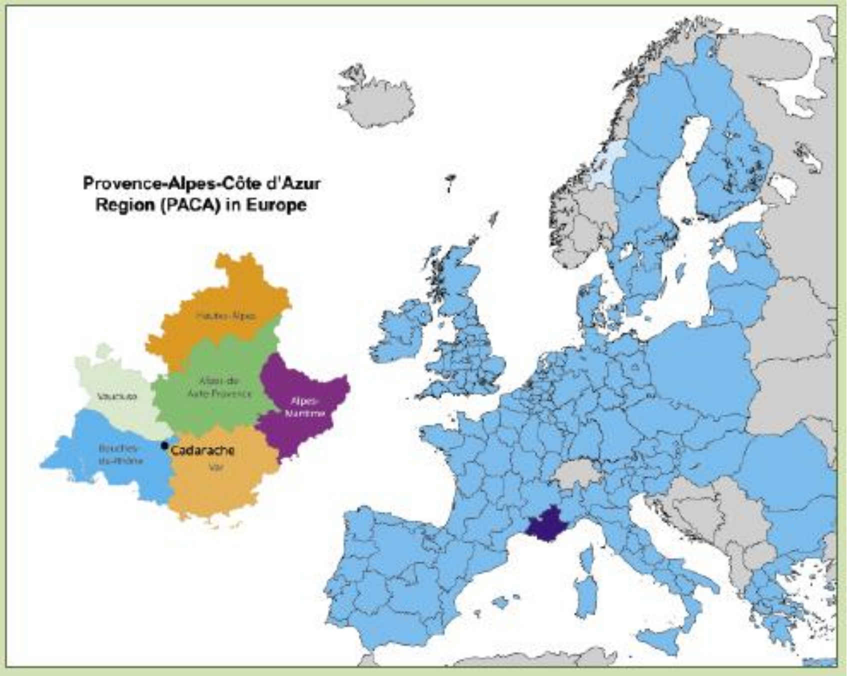
Nuclear cross sections

A Fusion power plant would be like…



Not to scale !

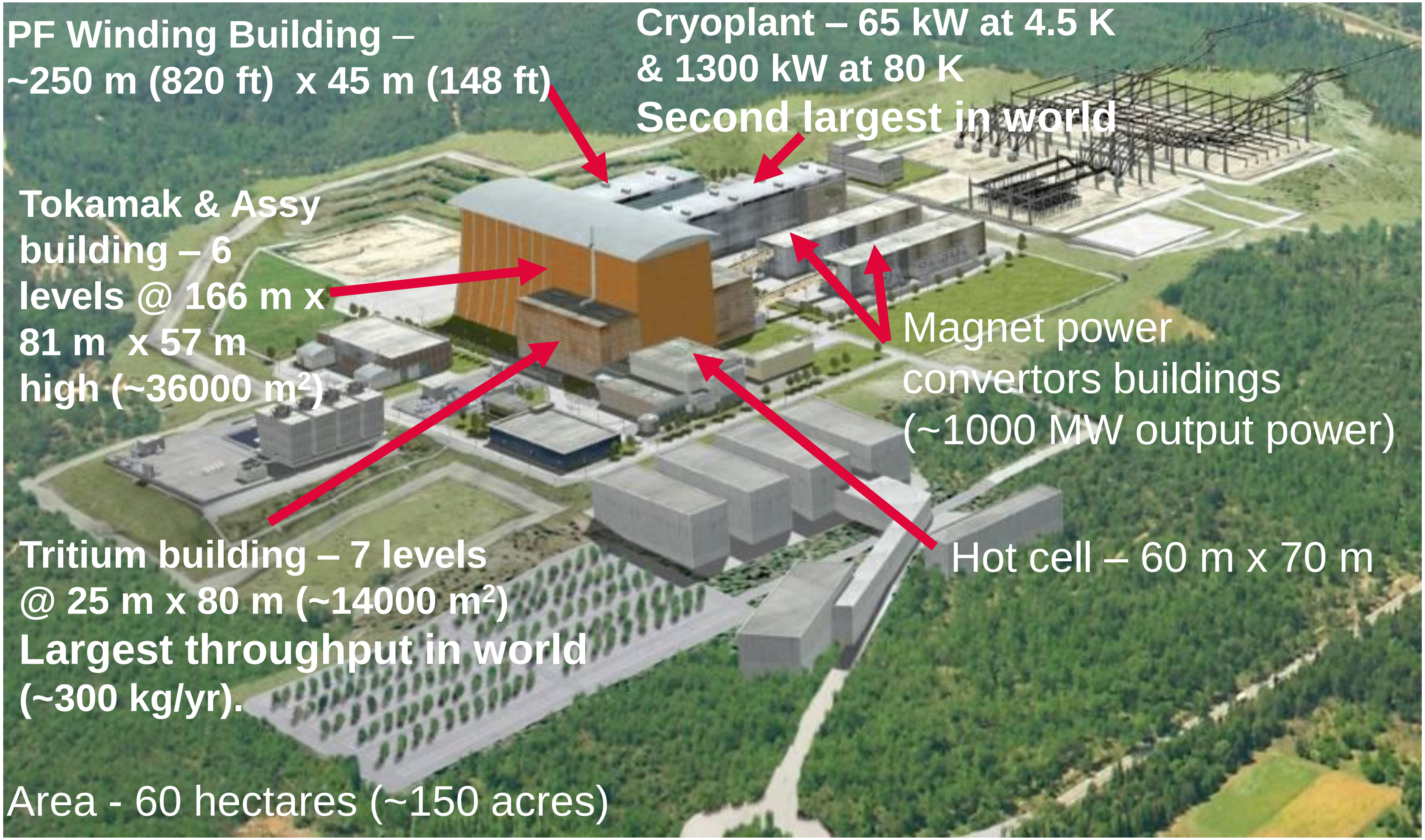
ITER: site



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

…based in Cadarache, Southern France

ITER Buildings and Facilities



PF Winding Building – ~250 m (820 ft) x 45 m (148 ft)

Cryoplant – 65 kW at 4.5 K & 1300 kW at 80 K

Second largest in world

Tokamak & Assy building – 6 levels @ 166 m x 81 m x 57 m high (~36000 m^2)

Magnet power convertors buildings (~1000 MW output power)

Tritium building – 7 levels @ 25 m x 80 m (~14000 m^2)

Largest throughput in world (~300 kg/yr).

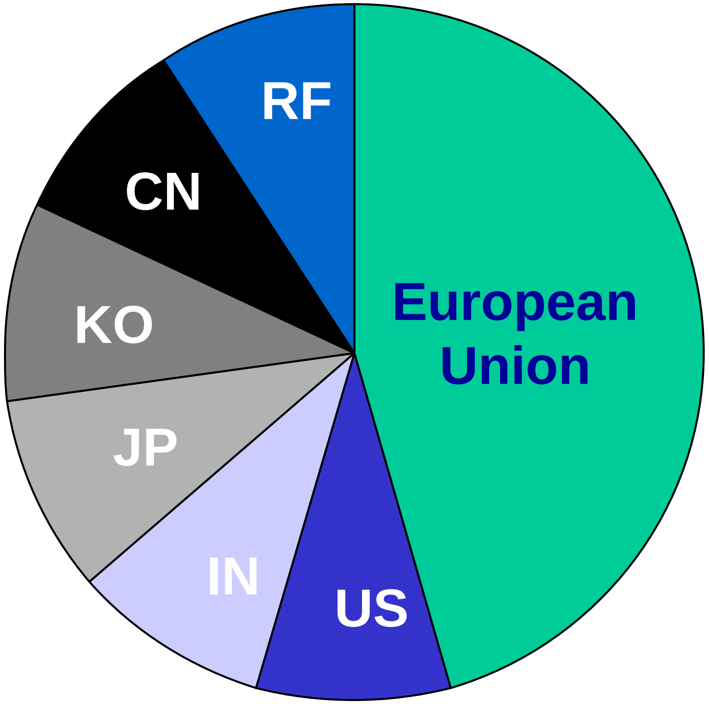
Hot cell – 60 m x 70 m

Area - 60 hectares (~150 acres)

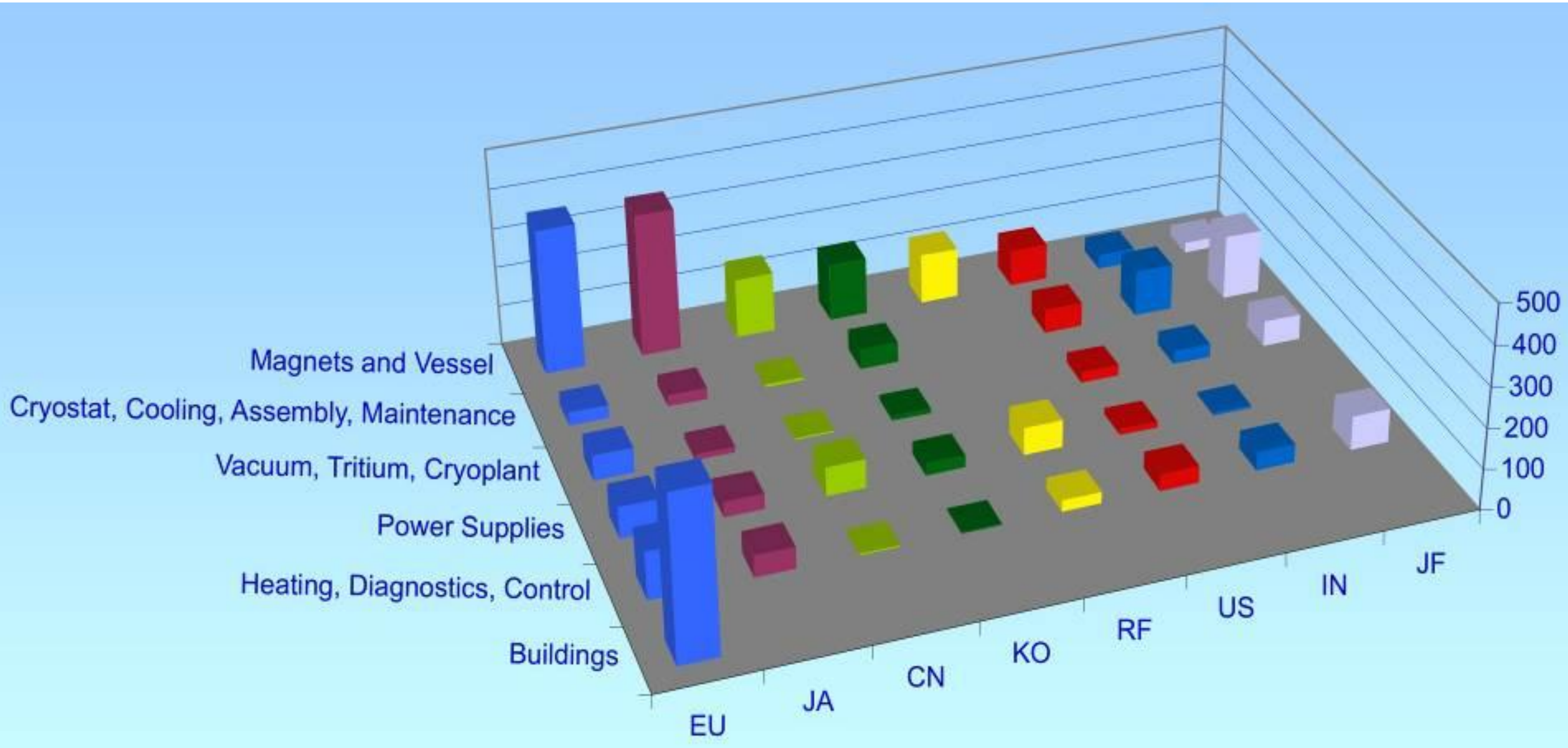
Overall sharing

EU 5/11, other six parties 1/11 each. Overall contingency of 10% of total. Total amount: 3577 kIUA (5365 M€)

Cost: ~5.5 billion Euros for construction, and ~5 billion for operation and decommissioning

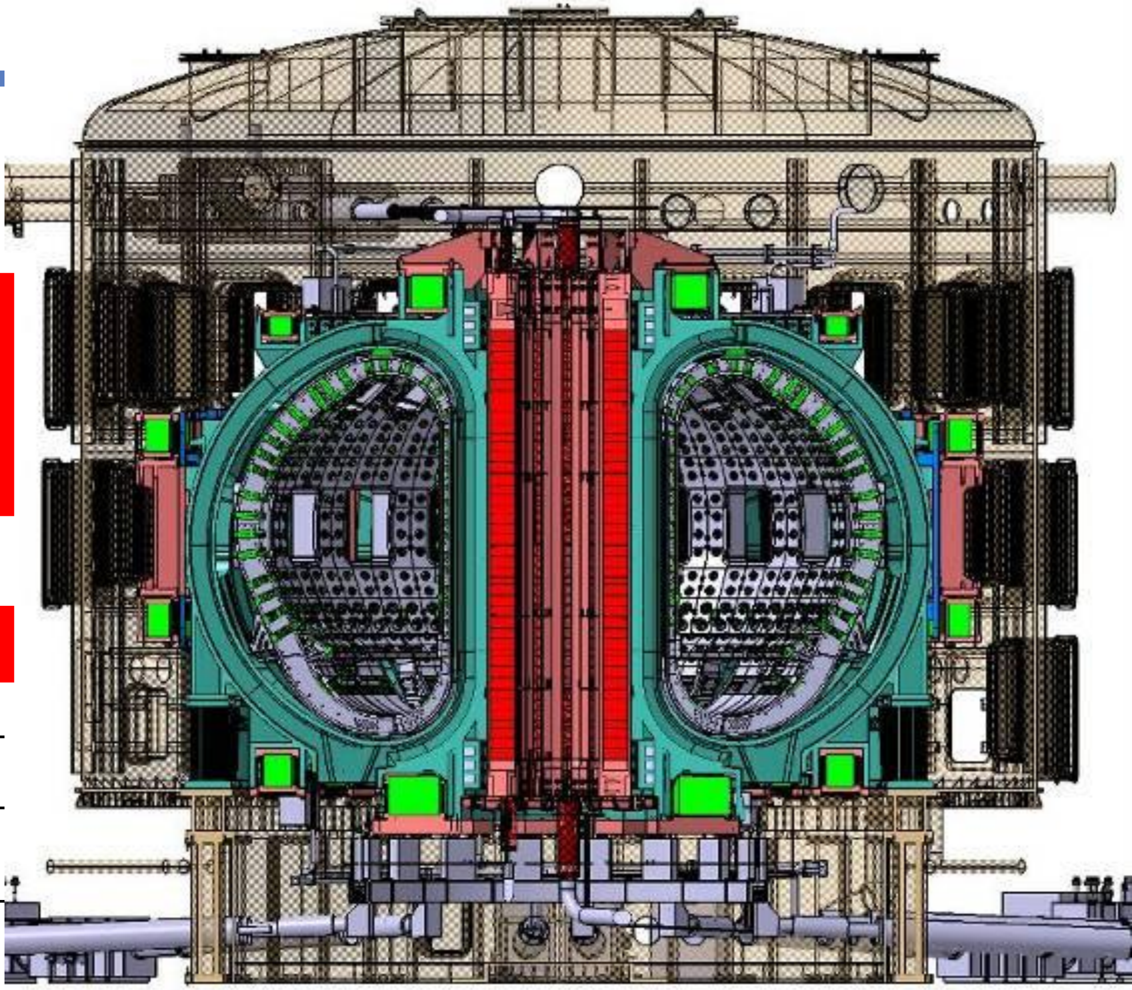


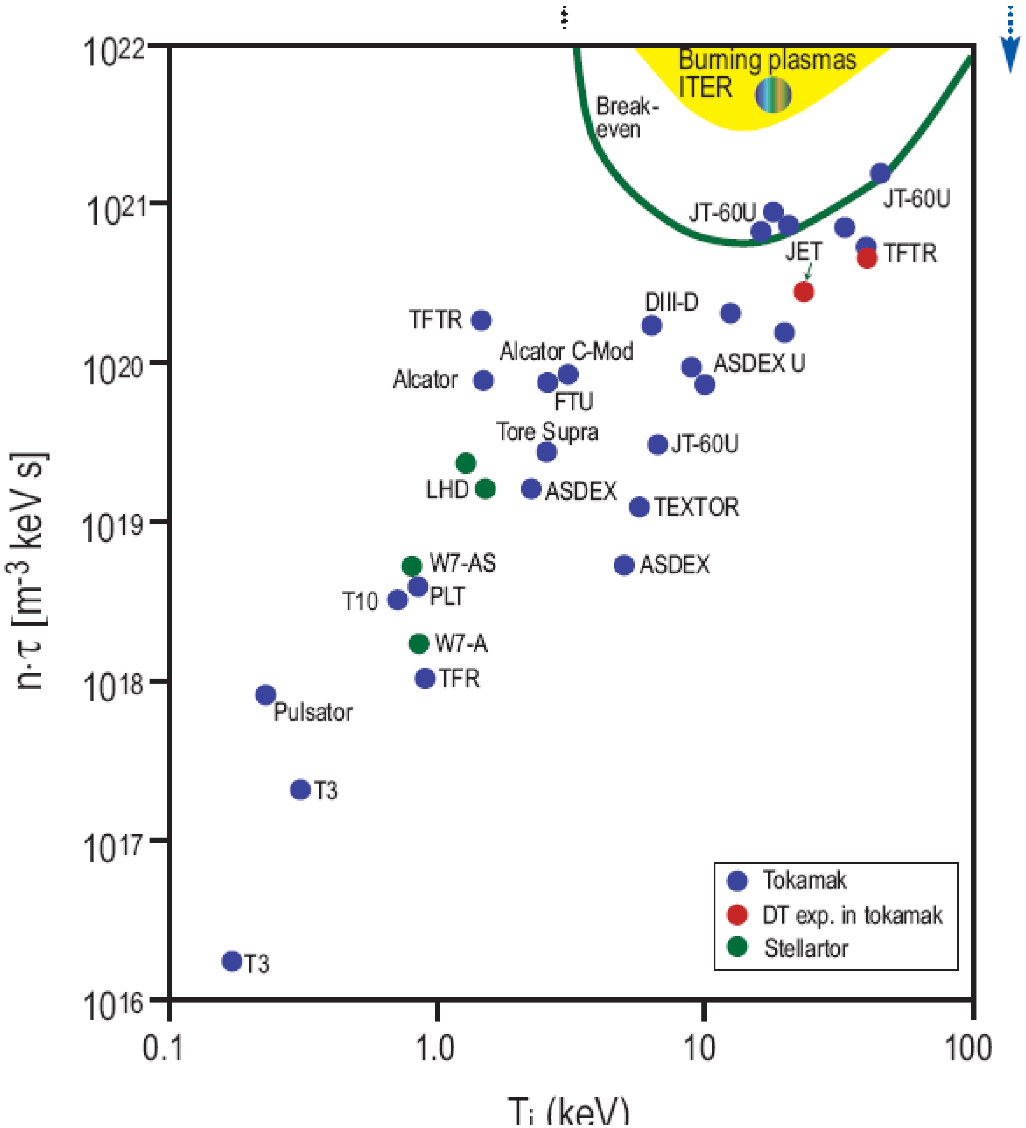
- A unique feature of ITER is that almost all of the machine will be constructed through *in kind* procurement from the Parties



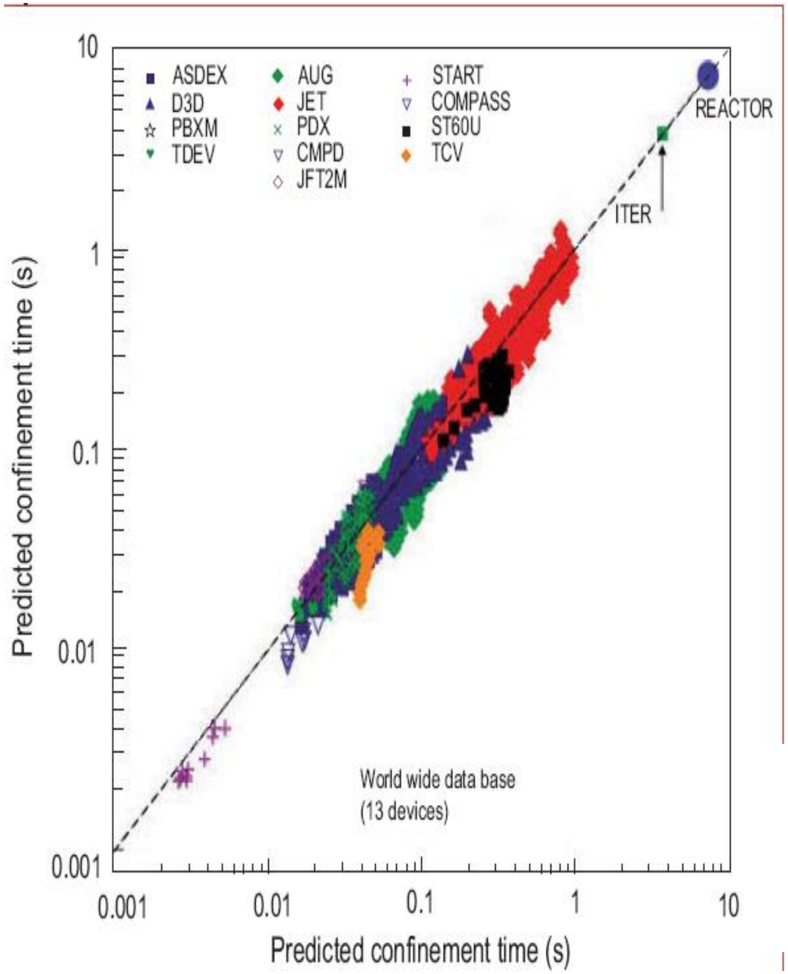
The Core of ITER

Total fusion power	**500 MW**
Additional heating power	**50 MW**
Q - fusion power/ additional heating power	**≥ 10**
Average 14MeV neutron wall loading	**≥ 0.5 MW/m²**
Plasma inductive burn time	**300-500 s ***
Plasma major radius (R)	**6.2 m**
Plasma minor radius (a)	**2.0 m**
Plasma current (I_p)	**15 MA**
Toroidal field at 6.2 m radius (B_T)	**5.3 T**




Burning plasmas
ITER
Break-
even
JT-60U
JT-60U
JET
TFTR
DIII-D
TFTR
Alcator C-Mod
ASDEX U
Alcator
FTU
Tore Supra
JT-60U
LHD
ASDEX
TEXTOR
W7-AS
ASDEX
T10
PLT
W7-A
TFR
Pulsator
T3
T3
Tokamak
DT exp. in tokamak
Stellartor
10^{22}
10^{21}
10^{20}
10^{19}
10^{18}
10^{17}
10^{16}
n·τ [m^{-3} keV s]
0.1
1.0
10
100
T_i (keV)

Scaling of the confinement time



Scaling in termini di parametri fisici

$$\tau_{E,th}^{ELMy} \propto \tau_B \rho_*^{-0.83} \beta^{-0.50} \nu_*^{-0.10}$$
$$\times M^{0.97} q^{-2.52} \varepsilon^{-0.55} \kappa^{2.72}.$$

Scaling in termini di parametri ingegneristici

$$\tau_{E,th}^{ELMy} = 0.0365 I^{0.97} \boxed{B^{0.08}} P^{-0.63} n^{0.41}$$
$$\times M^{0.20} \boxed{R^{1.93}} \varepsilon^{0.23} \kappa^{0.67}$$

Key Design Review Issues - Tokamak

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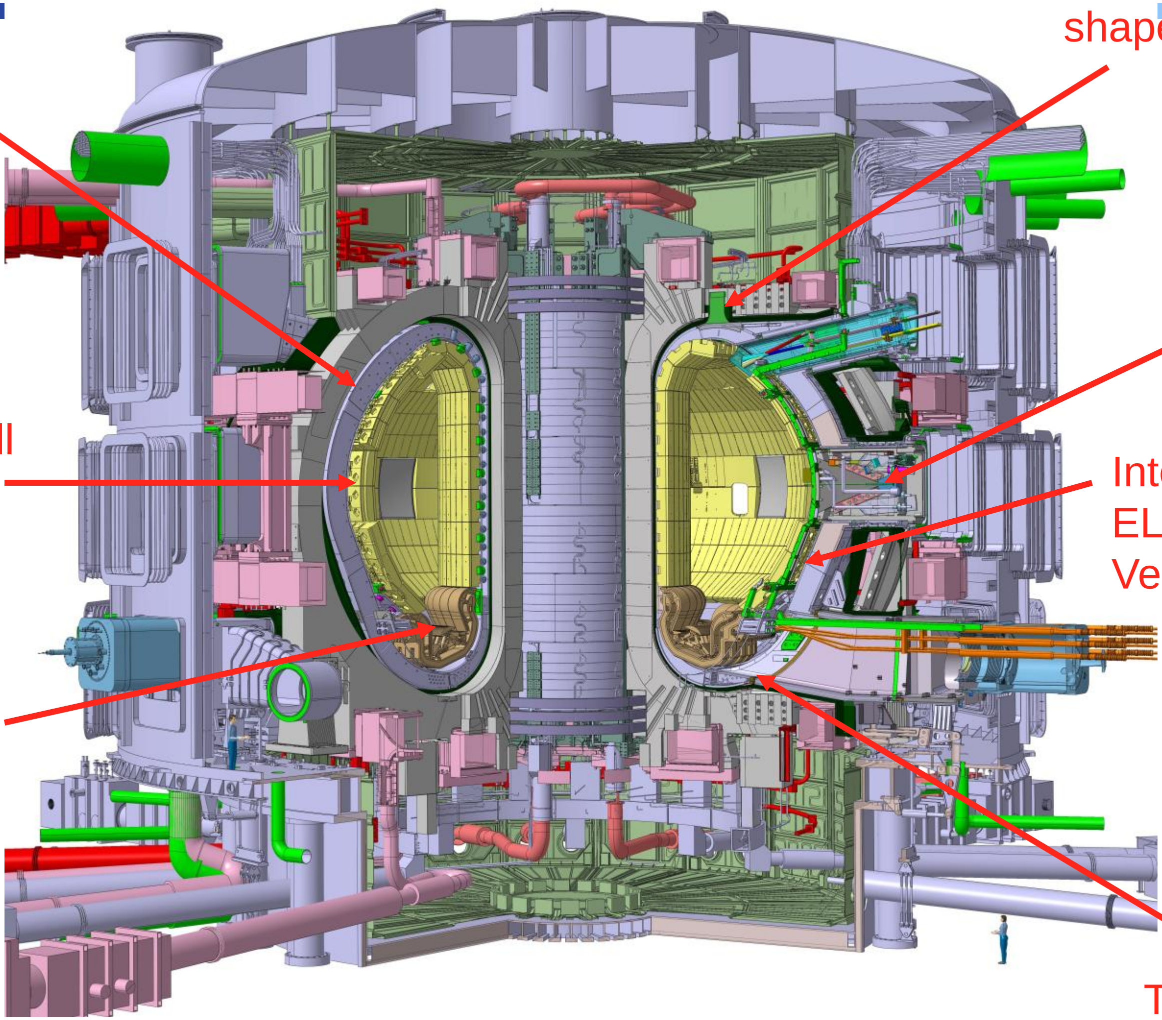


Vacuum vessel em-loads

PF/CS performance: shape and vert. stability

H&CD Systems

Blanket/ First Wall design and RH - Heat Loads

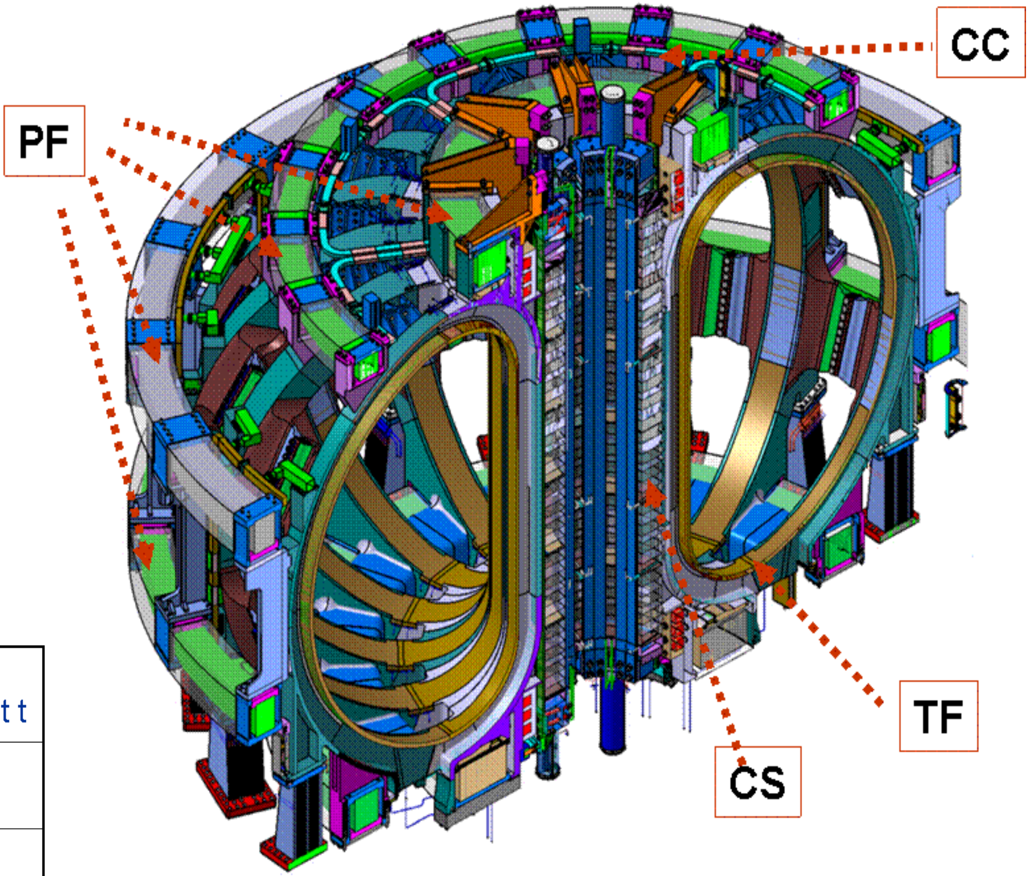
Internal coils: ELM & RWM control, Vertical stability

Divertor material strategy

TF performance: fusion gain

Overview of the Magnet System

- 48 superconducting coils
  - 18 TF coils
  - 6 CS modules
  - 6 PF coils
  - 9 pairs of CC



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

ELM & VS coils – layout of reference option

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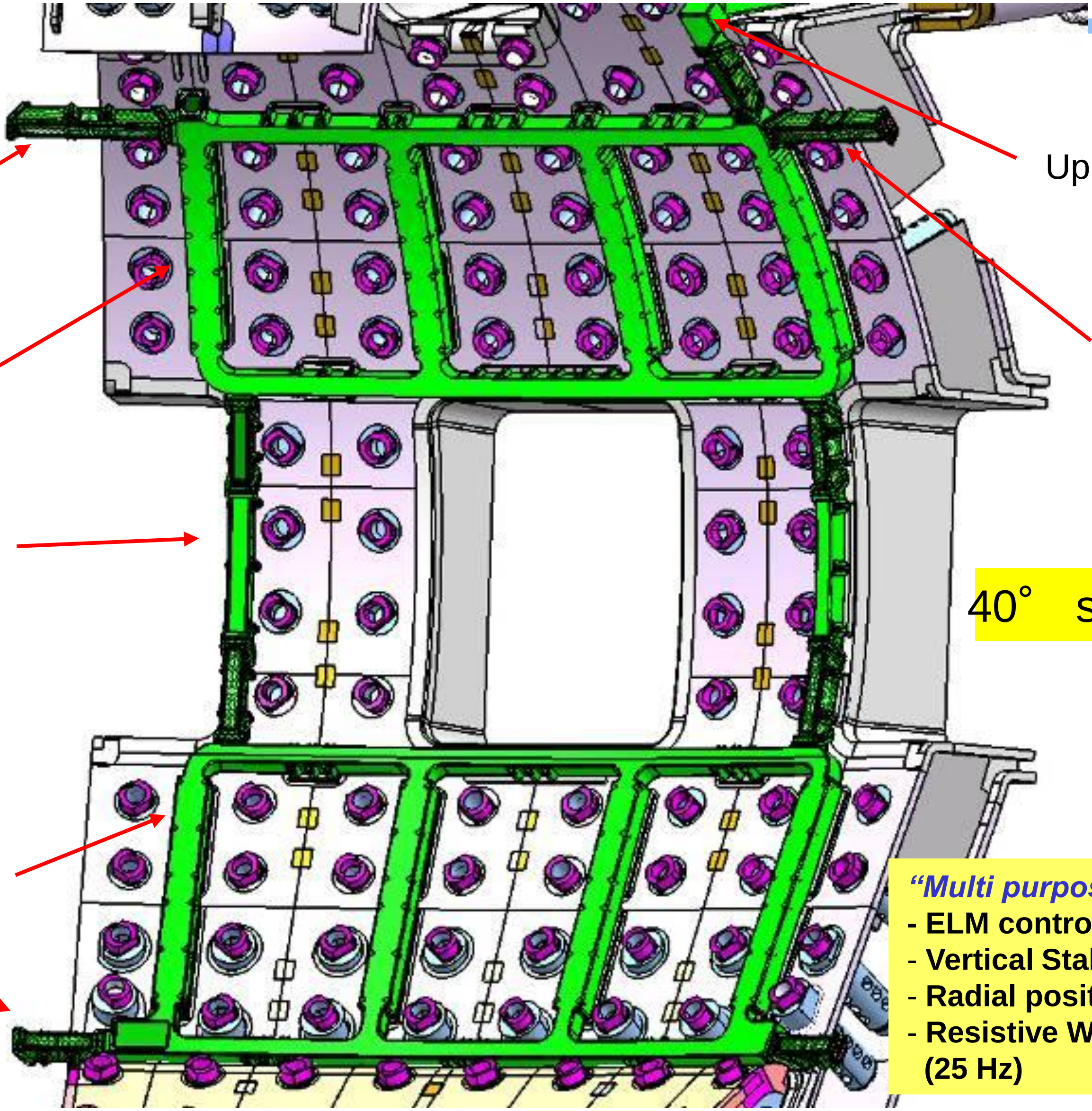
VAC-02 - 3 rows – 9 coils



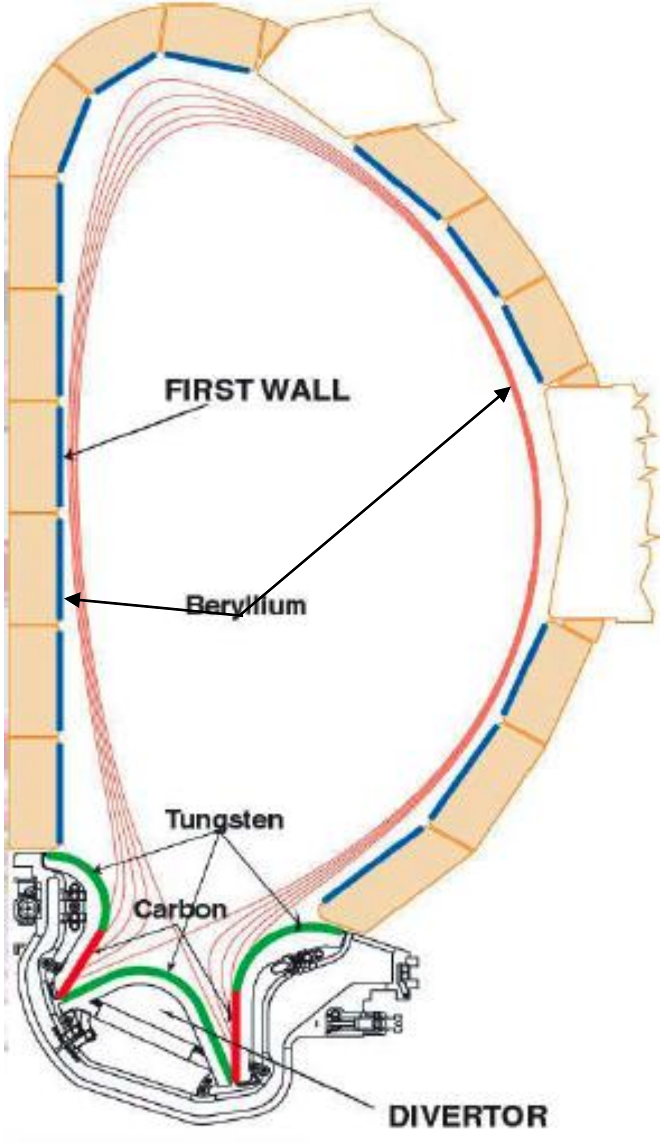
40°   sector Outboard

"Multi purpose" in vessel coils:

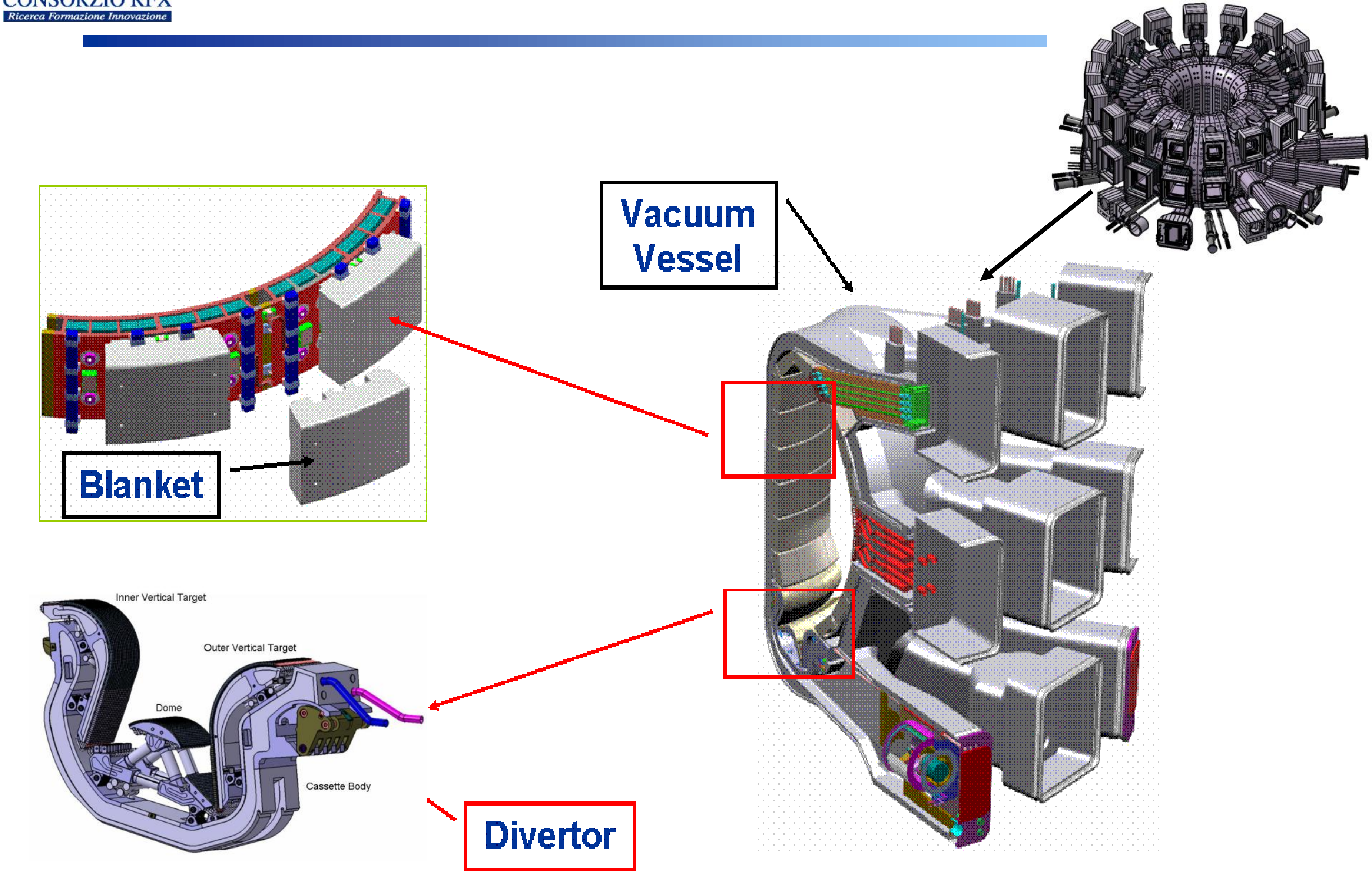
- **ELM control (~2Hz),**
- **Vertical Stability control**
- **Radial position control**
- **Resistive Wall Mode (RWM) control (25 Hz)**

Plasma Facing Material Strategy

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



VV and In-vessel Components

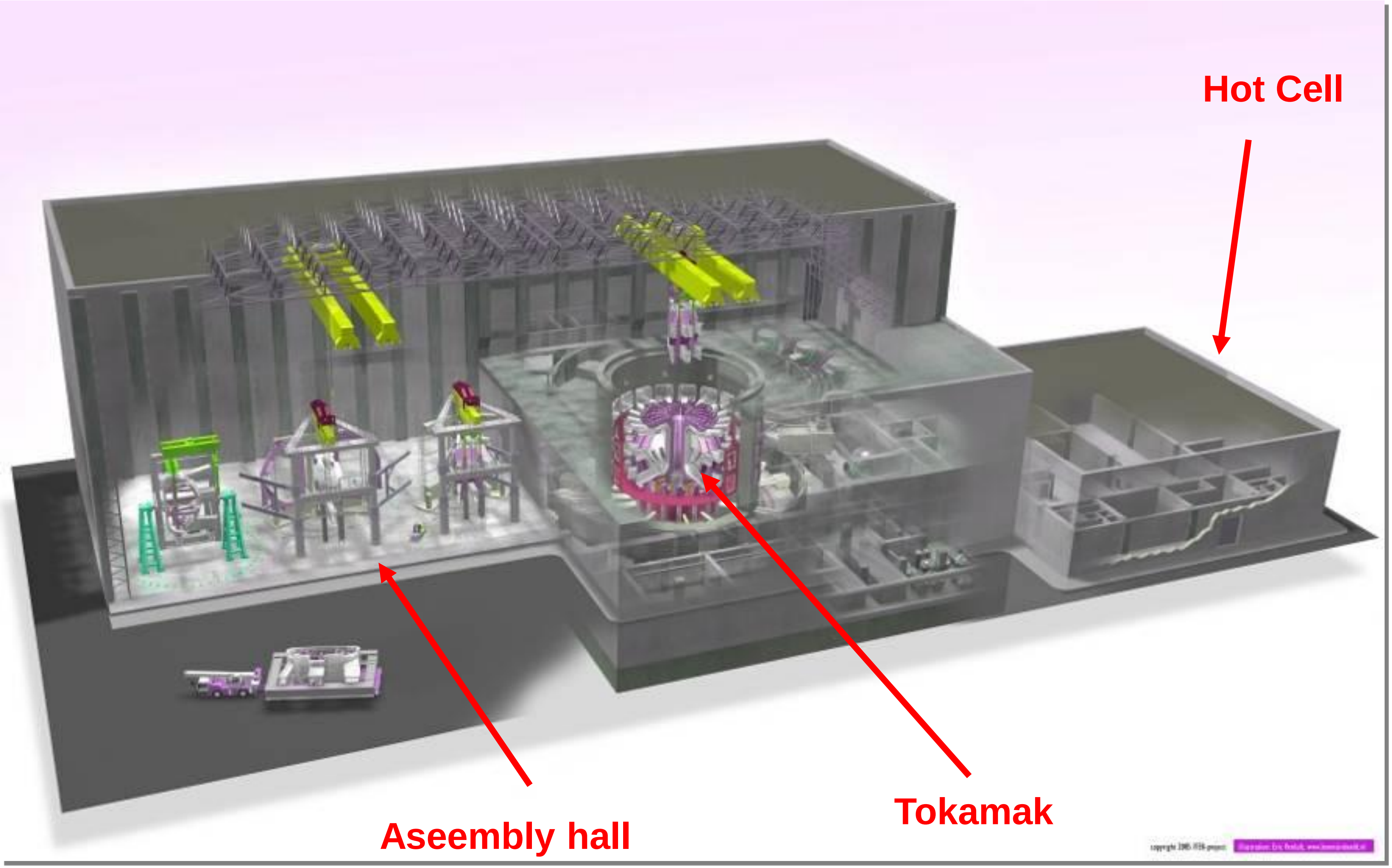


Vacuum Vessel

Blanket

Divertor

Tokamak Building



Hot Cell

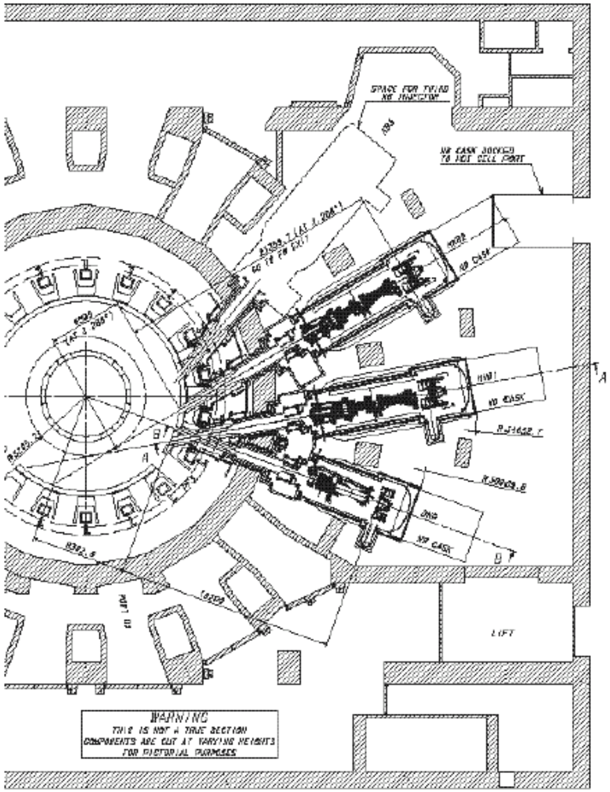
Aseembly hall

Tokamak

ITER : sistema di riscaldamento ausiliario

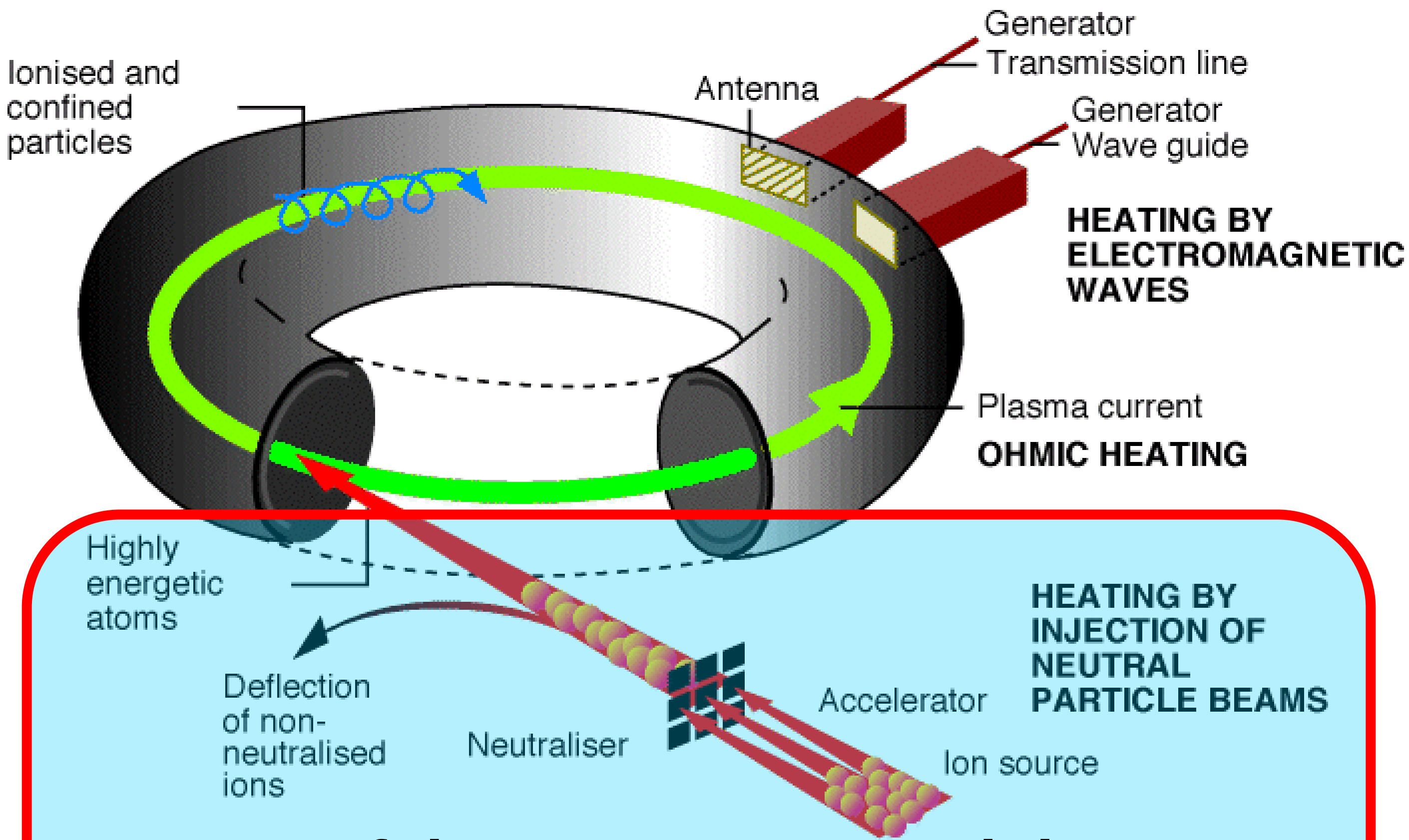
Heating System	Stage 1	Possible Upgrade	Remarks
NBI (1MeV Šive ion)	33	16.5	Vertically steerable (z at Rtan -0.42m to +0.16m)
ECH&CD (170GHz)	20	20	Equatorial and upper port launchers steerable
ICH&CD (40-55MHz)	20		$2\Omega_T$ (50% power to ions Ω_{He3} (70% power to ions, FWCD)
LHH&CD (5GHz)		20	$1.8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

NBI Layout



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

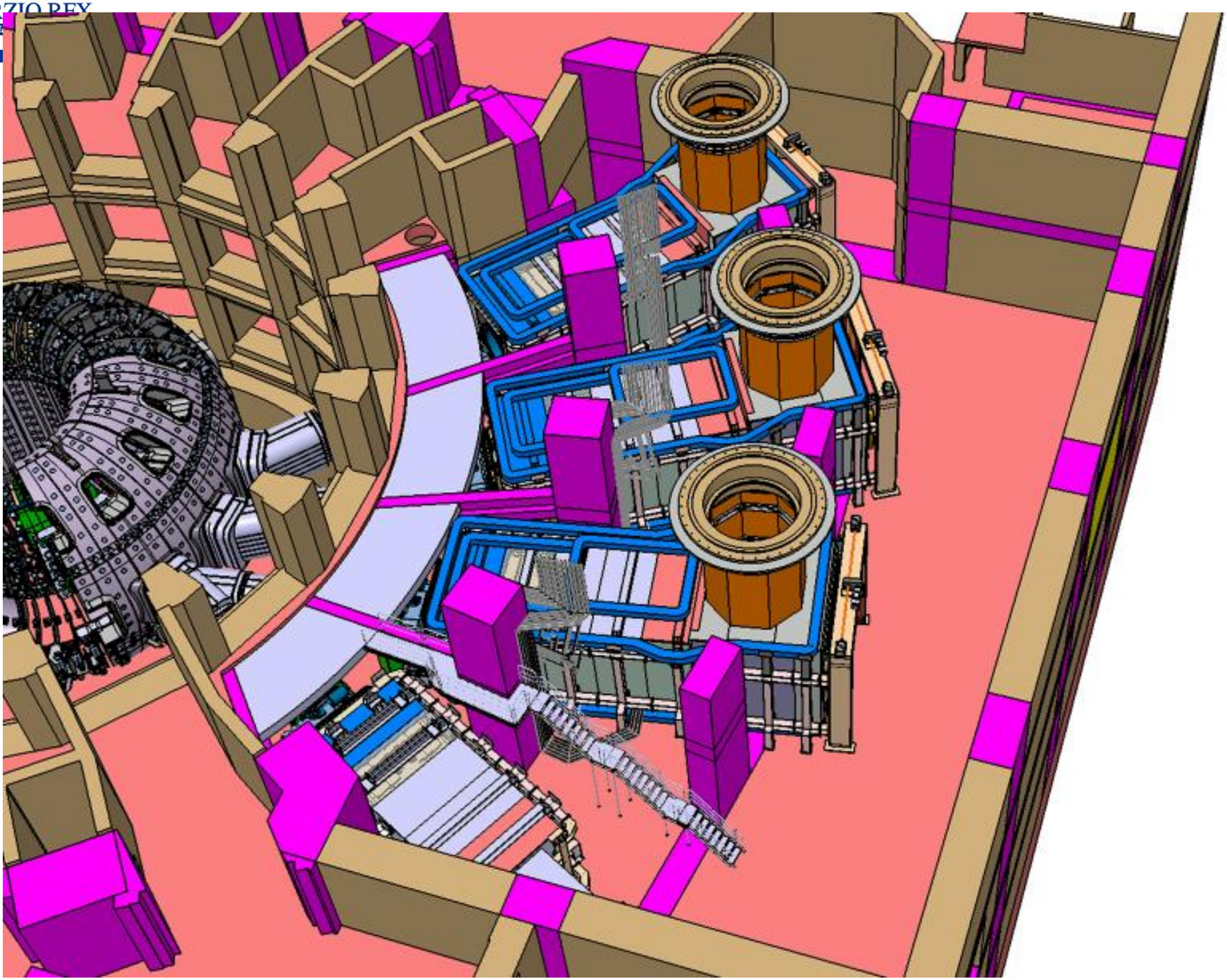
Heating and Current Drive Systems in ITER



R&D of the prototype 1MV injector

R&D of the negative ion source

L1 LEVEL NB CELL



2 (+1) NBI Neutral Beam Injectors based on negative ions are foreseen in ITER

Each beam must provide:

P=16.5MW
I=40A
V=1MV
t_{pulse}=3600s

Effective power deposition in the plasma core implies 1MV acceleration

1MV efficient neutralization by charge exchange implies negative ions

Main functionalities:

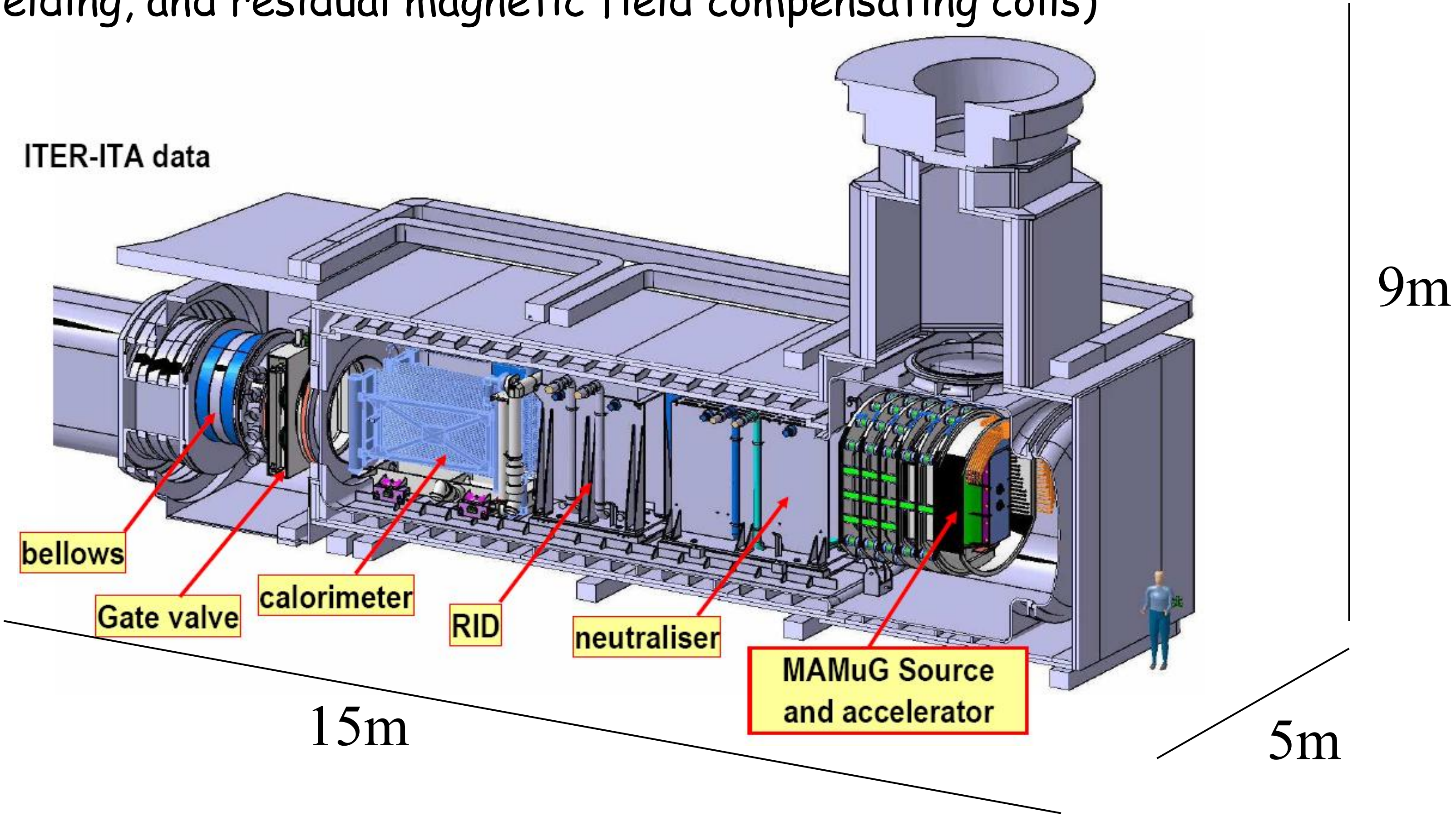
Plasma Heating
Plasma Rotation
Current drive
Plasma parameter profile control
Burn phase control

The injector

The Injector can be separated in

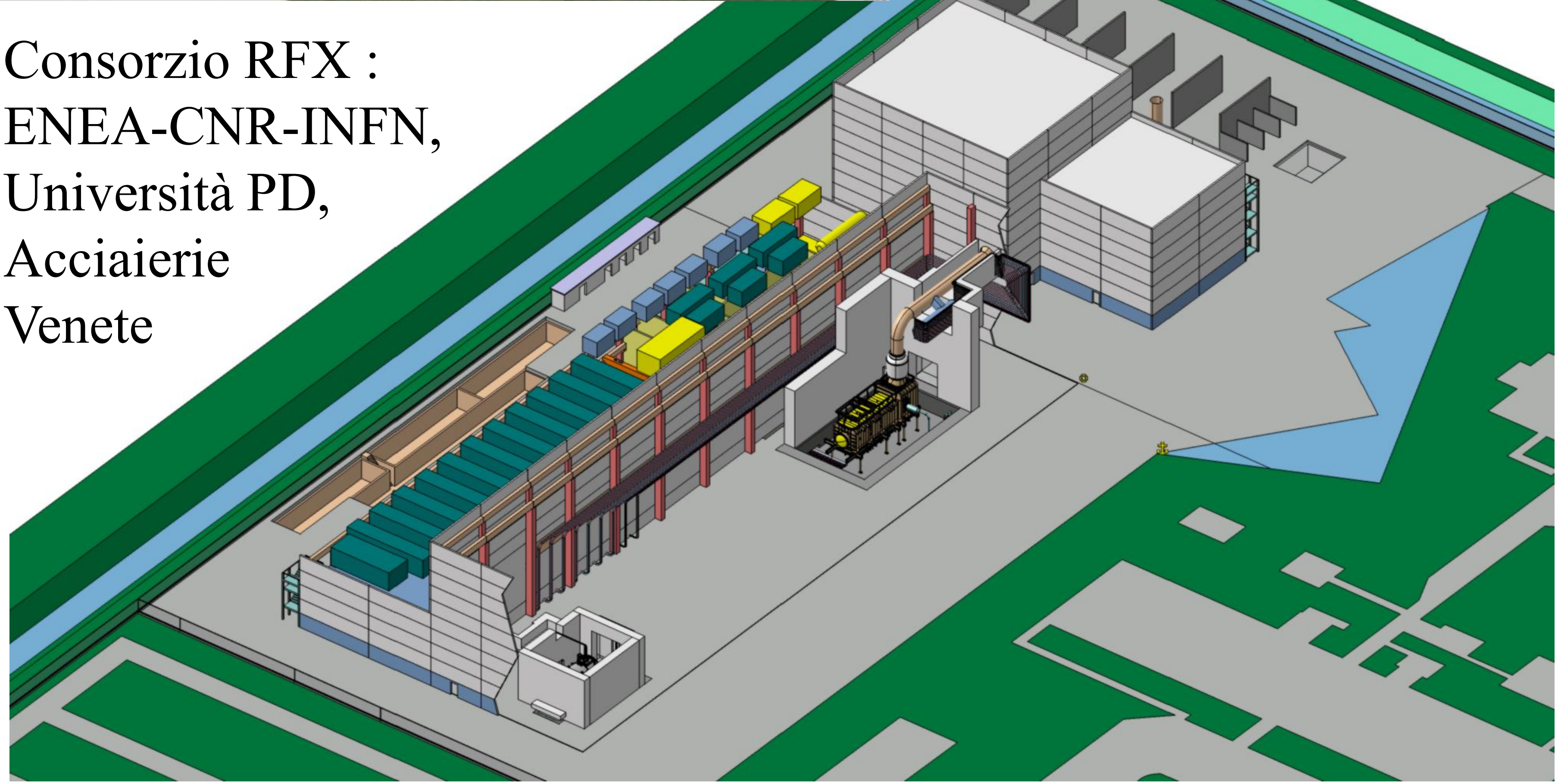
beam components (**Ion Source**, **Accelerator**, **Neutralizer**, **Residual Ion Dump** and Calorimeter)

other components (cryo-pump, vessels, fast shutter, duct, magnetic shielding, and residual magnetic field compensating coils)



Test Facility for the NBI to be realized in Padova at the Consorzio RFX

Consorzio RFX :
ENEA-CNR-INFN,
Università PD,
Acciaierie
Venete



Test Facility

- As most of the issues are strongly coupled, those of them still open can be tackled and solved only by testing a full scale NBI at full performance in D and H.
- A Test Facility where to install and operate a NBI system before operation in ITER has been approved and will be realized in Padova at Consorzio RFX
- Such a Test Facility should also allow adjustments and modifications aimed to optimise the system and will assist ITER during operations

- Status : building call for tender started
  power supply procurement initiated

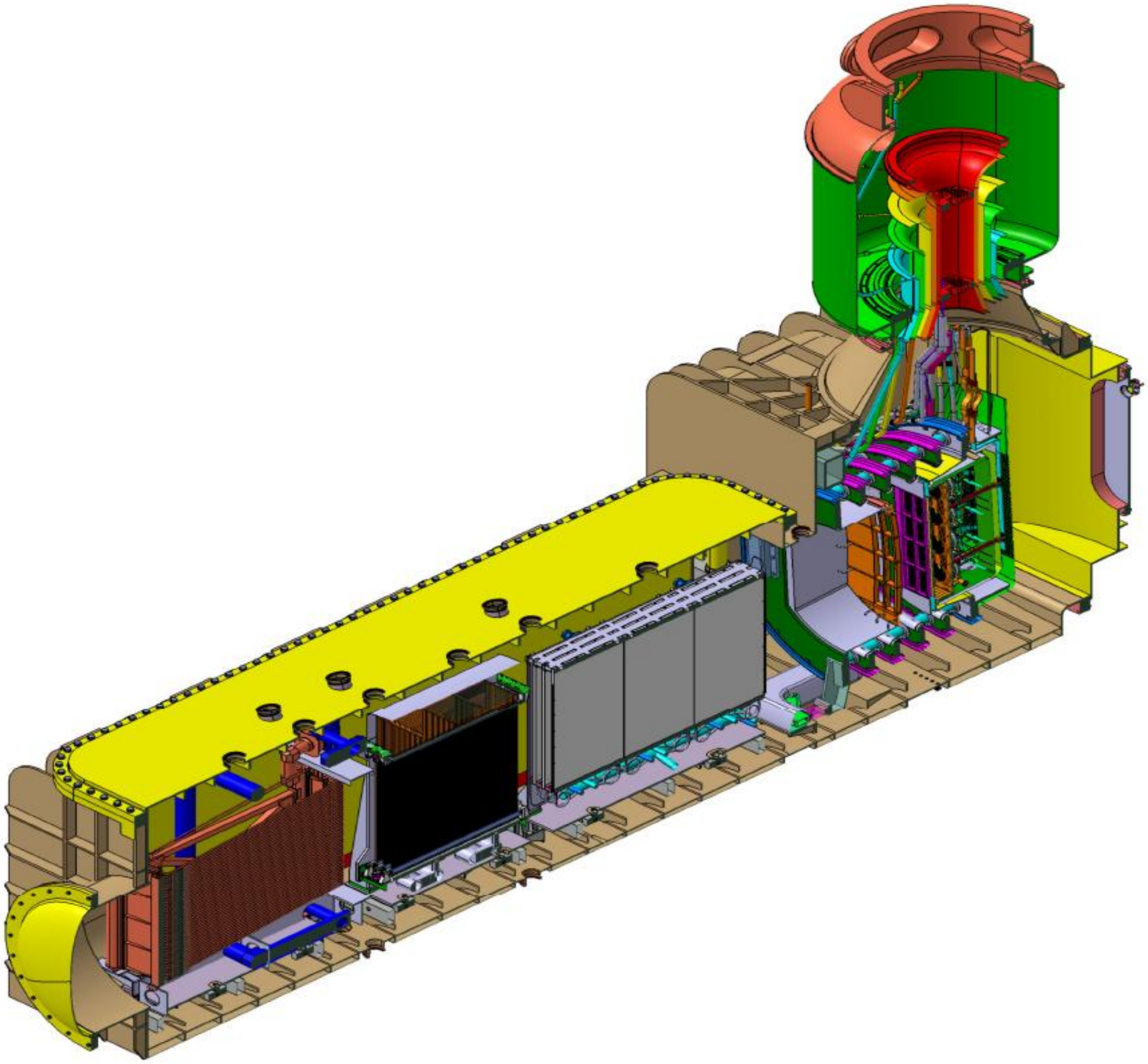
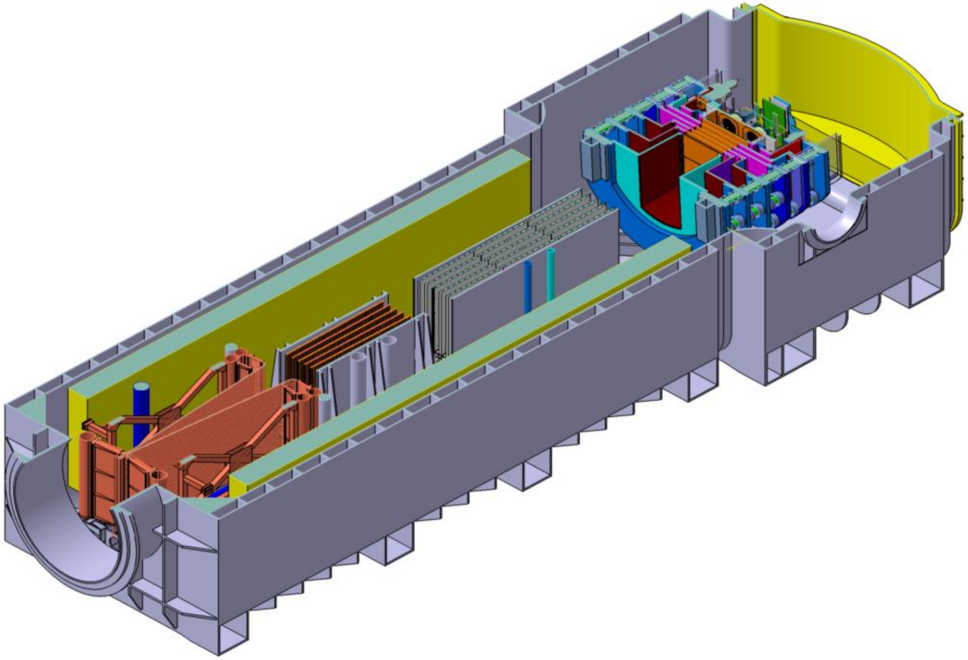
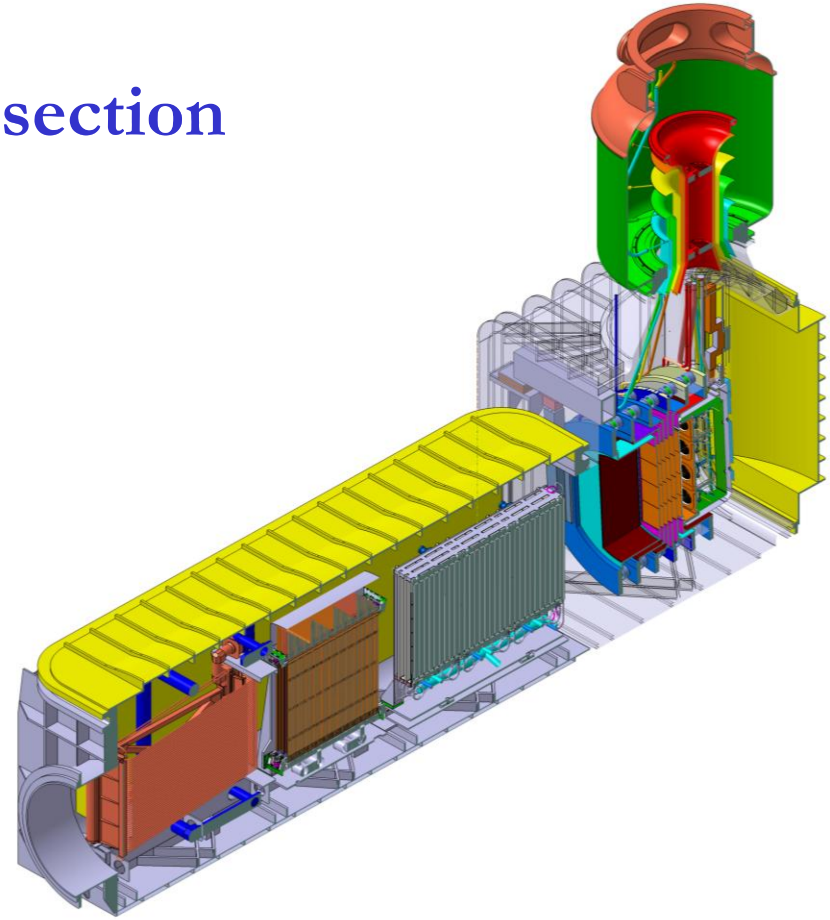


Fig. 9 Progetto del vessel per NBI di ITER

MITICA → The Injector section

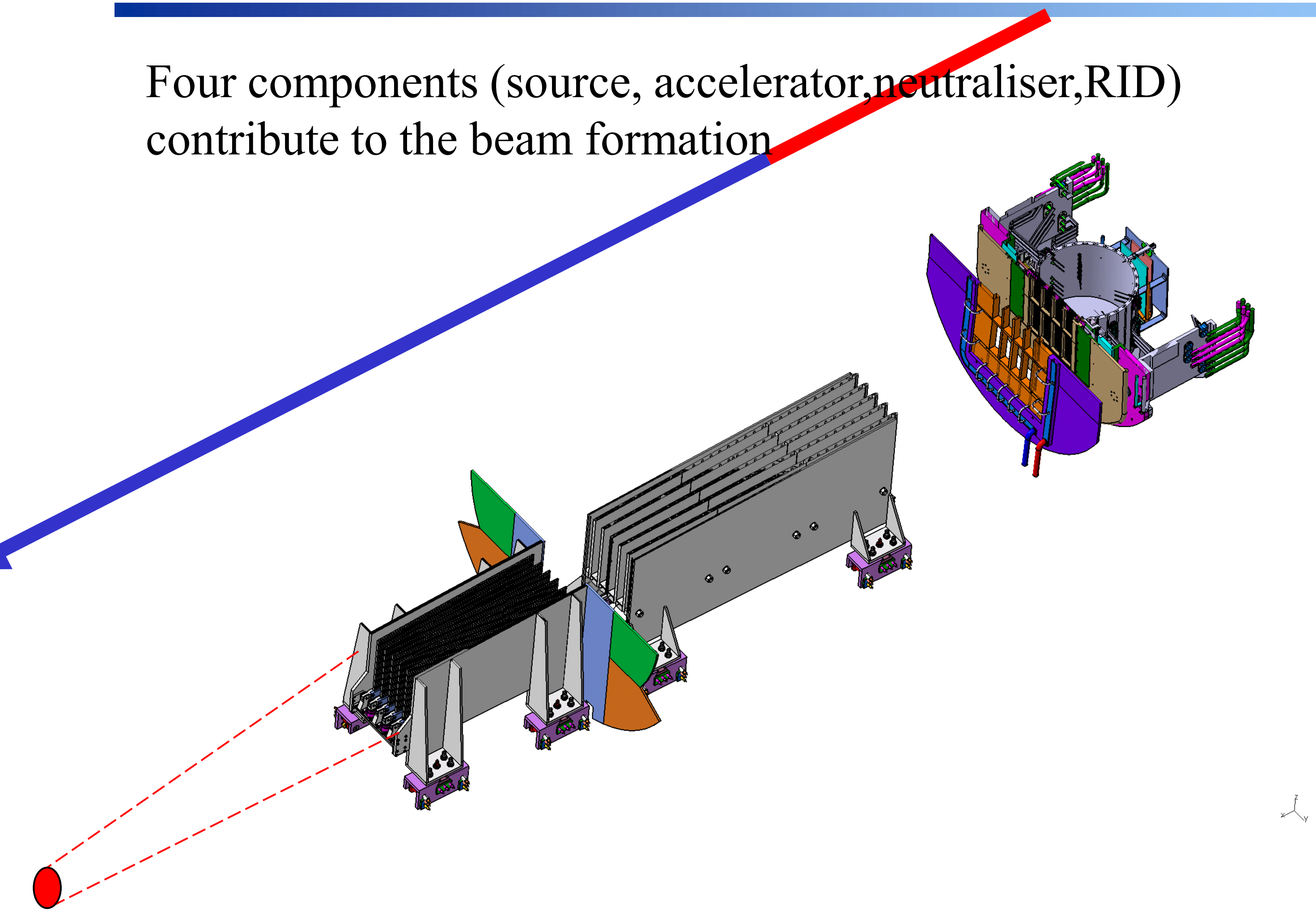




	Unit	H	D
Beam energy	keV	870	1000
Acceleration current	A	49	40
Maximum Beam Source pressure	Pa	0.3	0.3
Beamlet divergence	mrad	≤7	≤7
Beam on time	s	3600	3600
Co-extracted electron fraction (e^-/H^- or e^-/D^-)		<0.5	<1

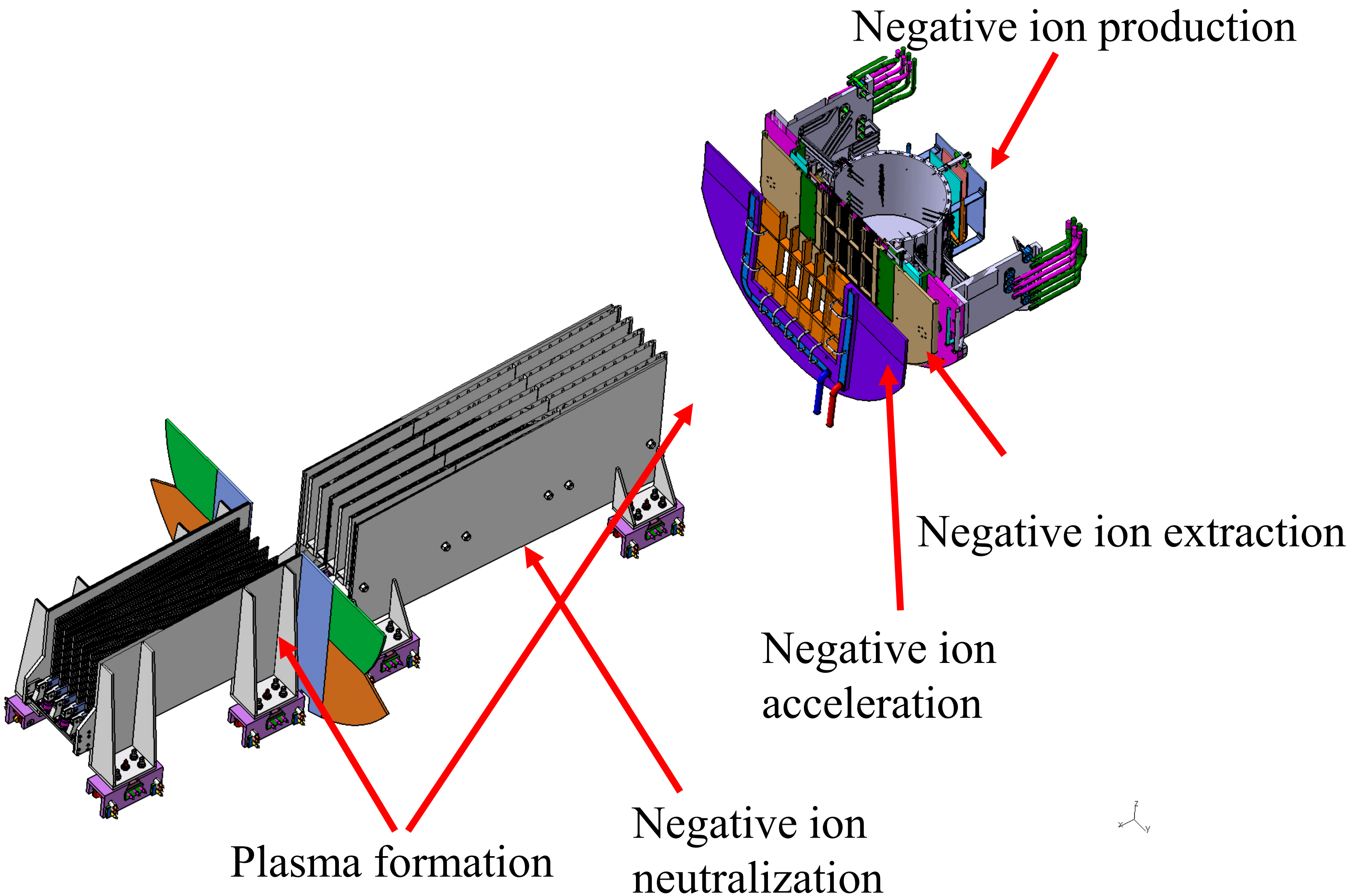
Beam formation

Four components (source, accelerator,neutraliser,RID) contribute to the beam formation



Elliptic beam size 0.6x0.8m

Physics issues



Negative ion production

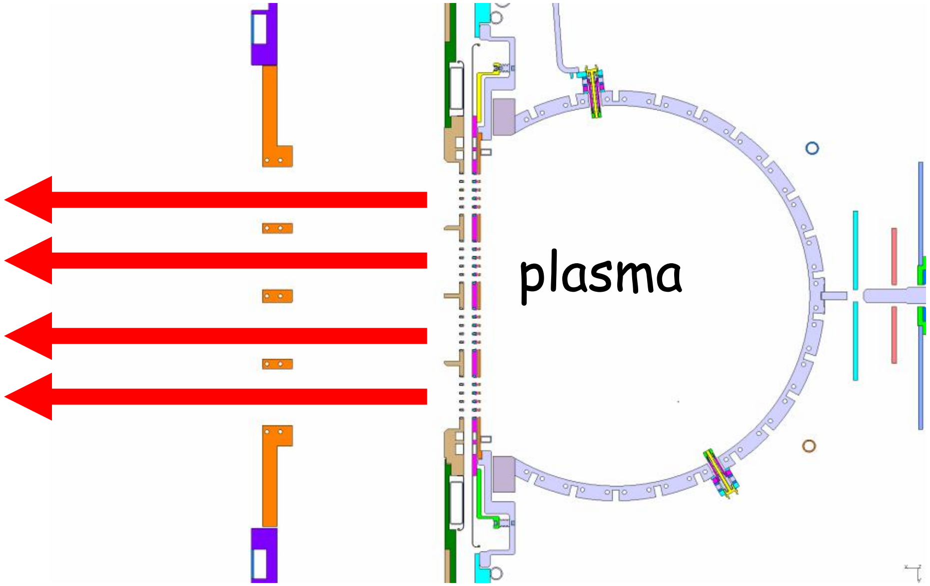
Negative ion extraction

Negative ion acceleration

Plasma formation

Negative ion neutralization

Beam formation: negative ion production



Negative Ion production

- Volume production (attachment of low energy electrons to excited molecules)
- Surface production (attachment of surface electrons to incident atoms)

Cs injection greatly increases negative ion yield as it decreases surface material working function

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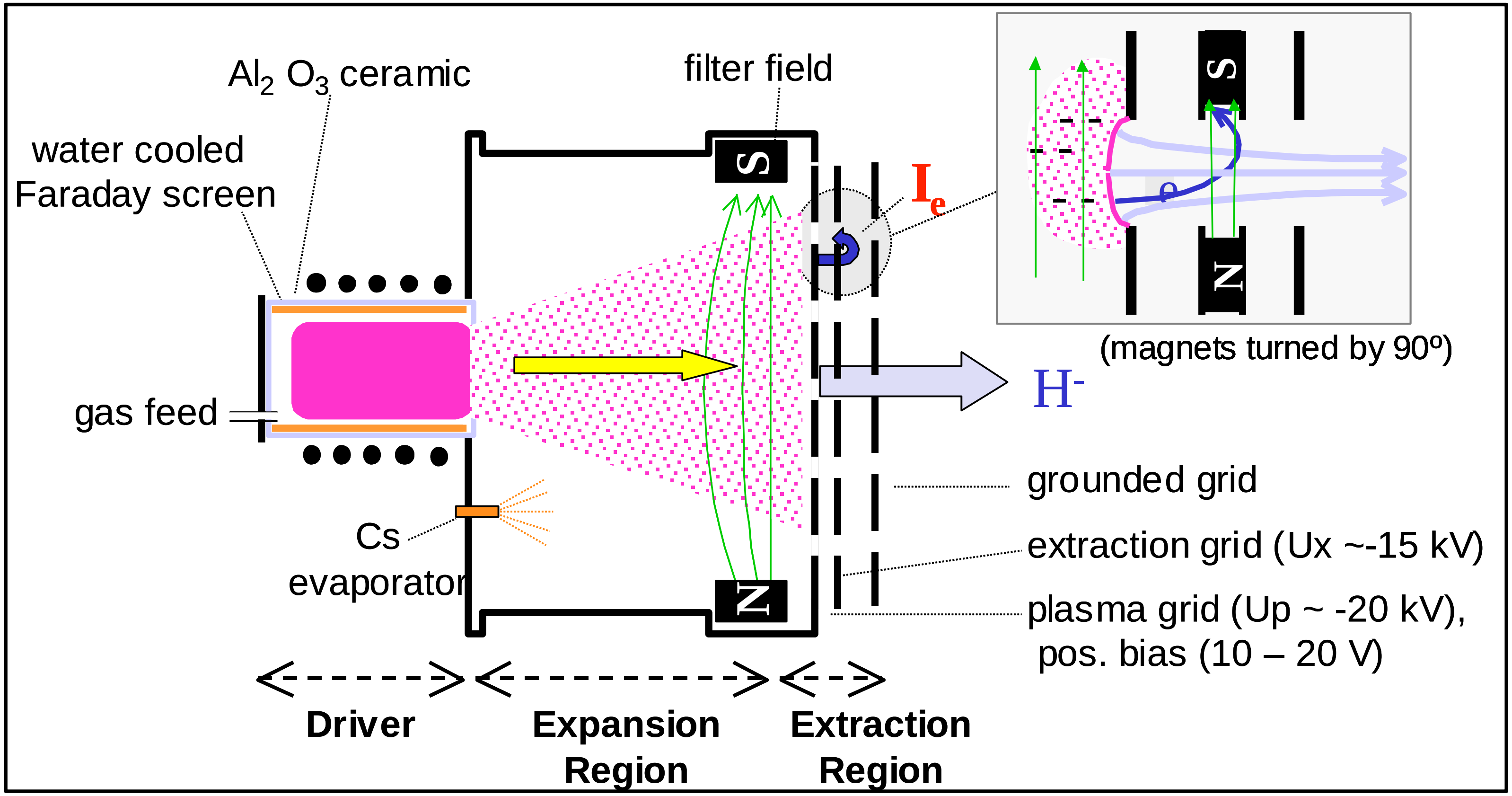
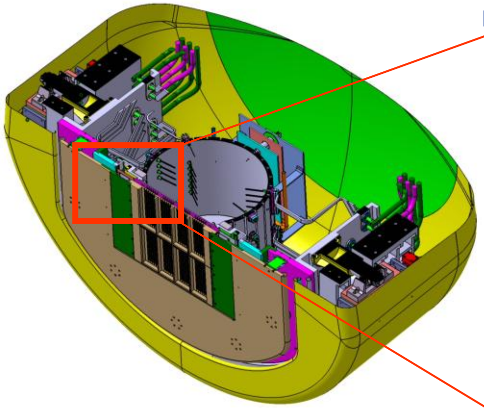
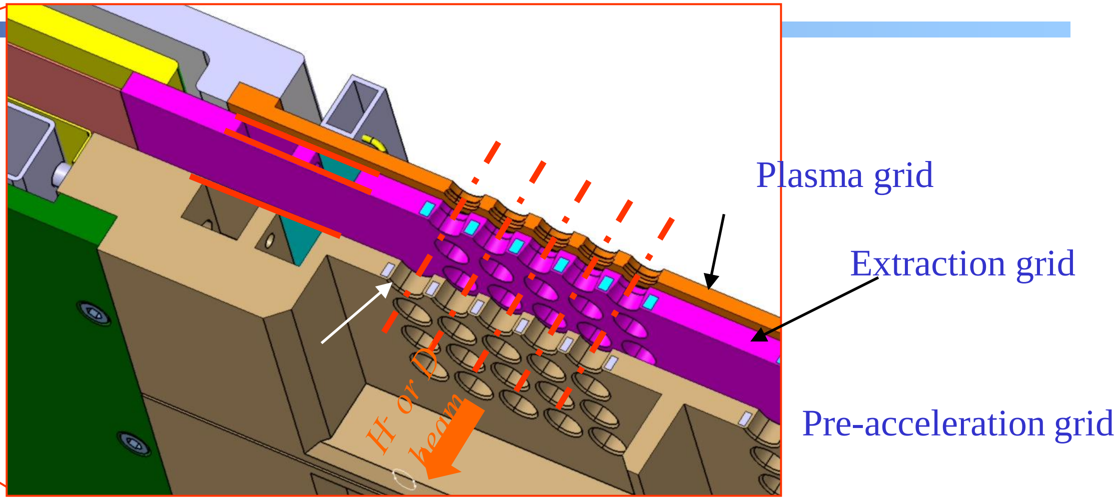
RF Ion Source



Grids design

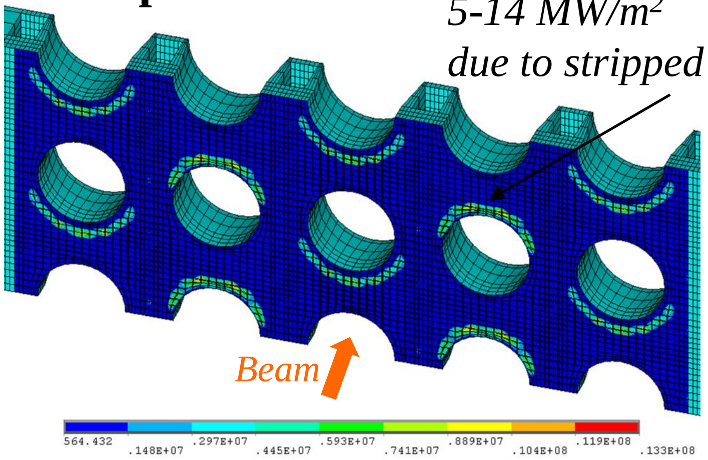


Section view of Beam source in SINGAP configuration



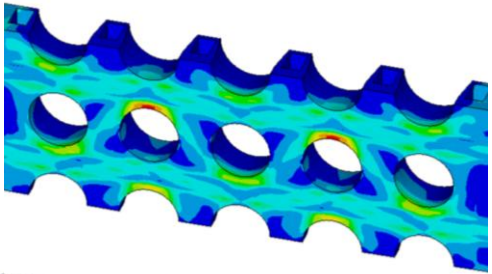
Detail of plasma, extraction and pre-acceleration grids

Extraction and acceleration grids are subjected to large power densities around the apertures

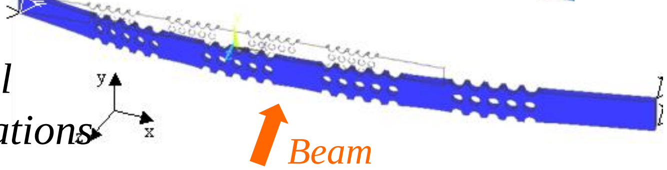


5-14 MW/m^2 due to stripped electrons

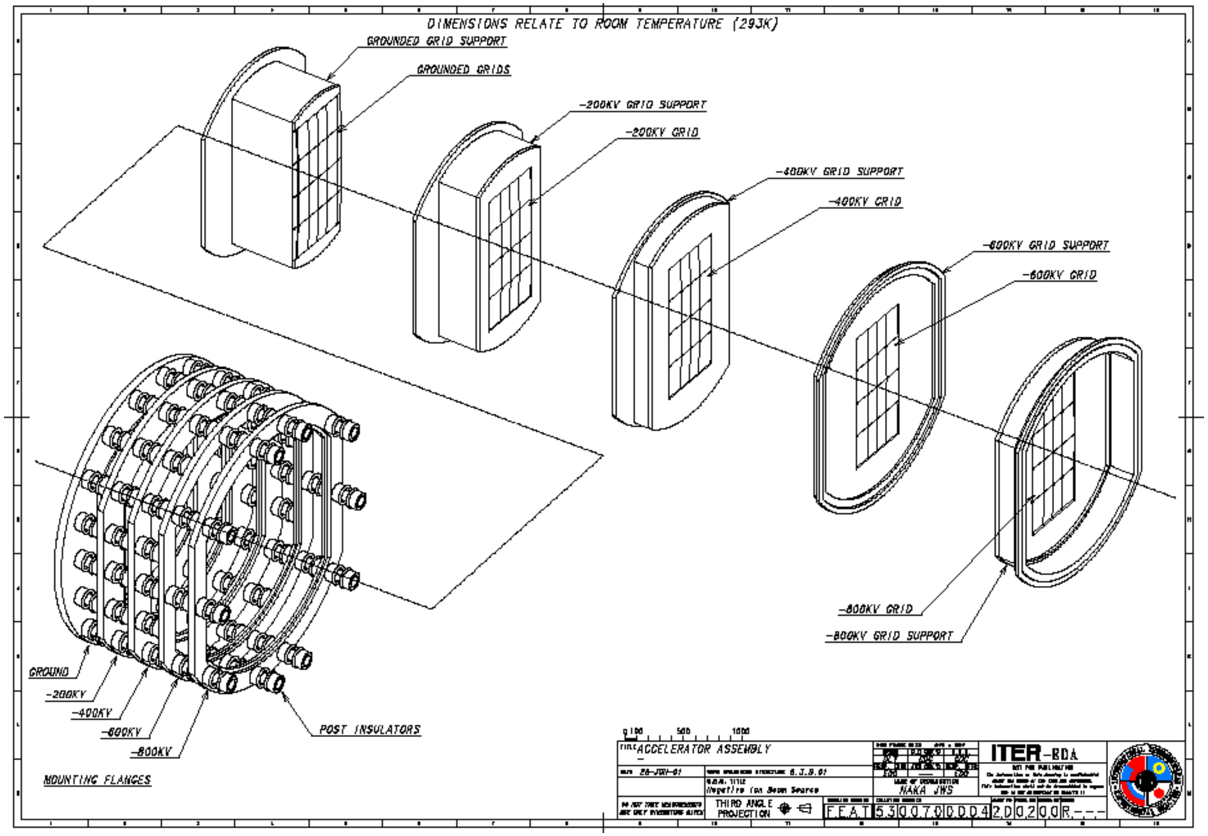
Thermal stresses



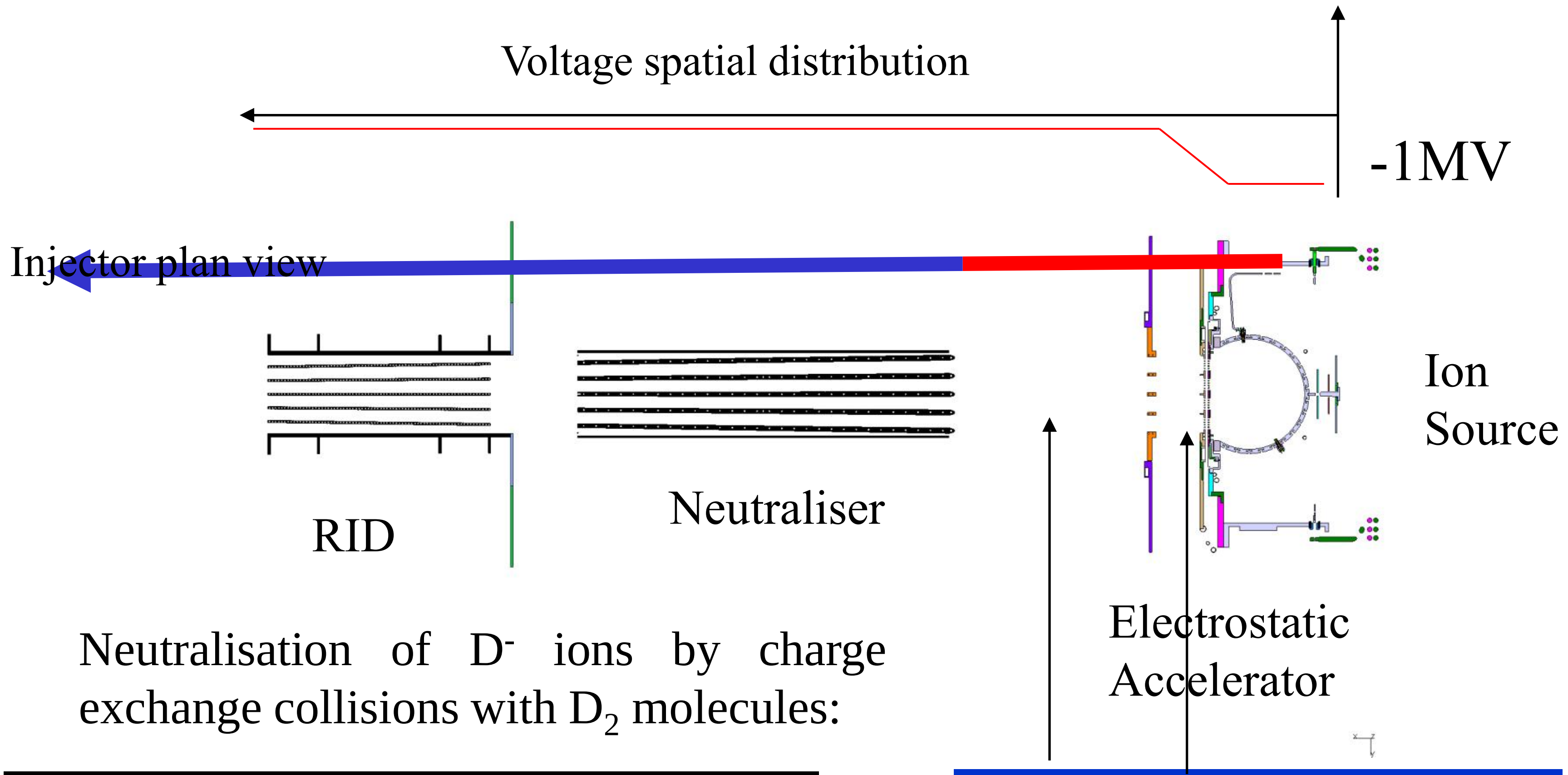
Thermal deformations



NBI in ITER: accelerator



Beam formation (acceleration and neutralisation)



Neutralisation of D^- ions by charge exchange collisions with D_2 molecules:

neutralisation

$$D^- + D_2 \Rightarrow D^0 + D_2 + e$$

re-ionisation (competing reaction)

$$D^0 + D_2 \Rightarrow D^+ + D_2 + e$$

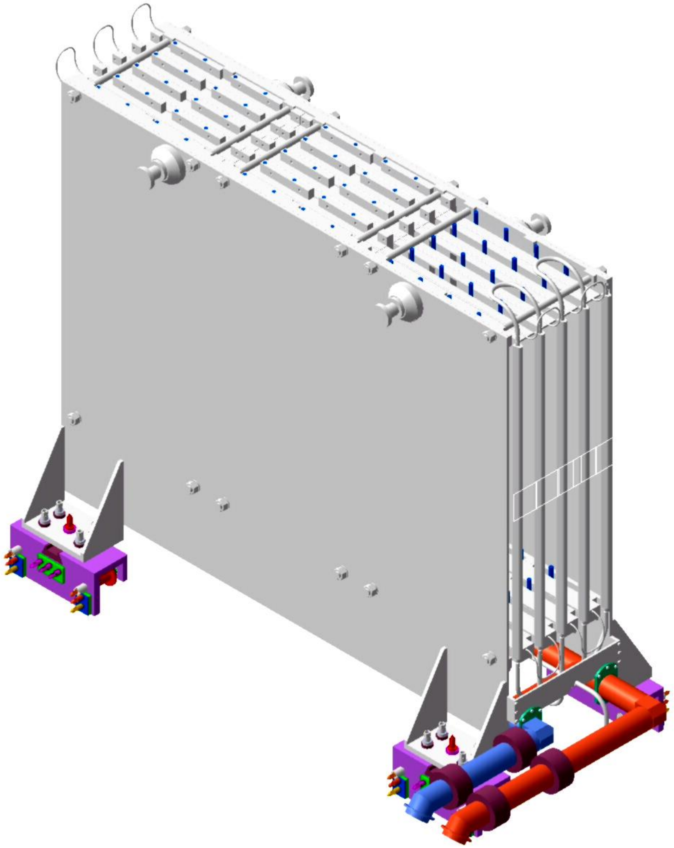
Additional electrons from

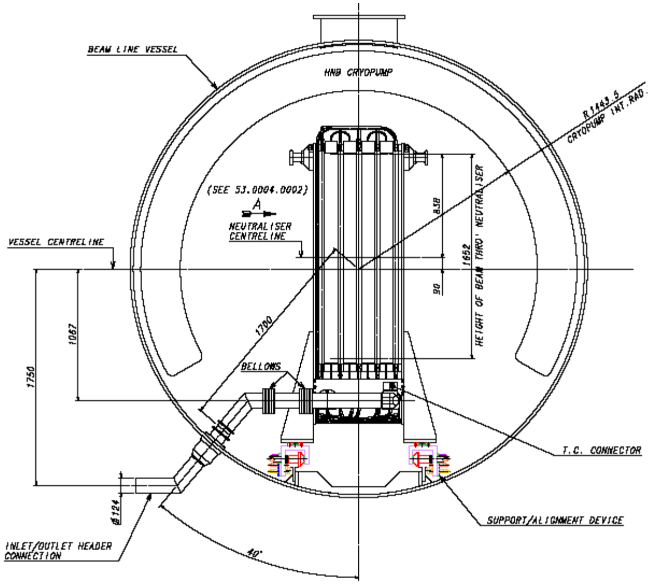
a) Stripping losses ($p<0.3$Pa)

b) Secondary electrons

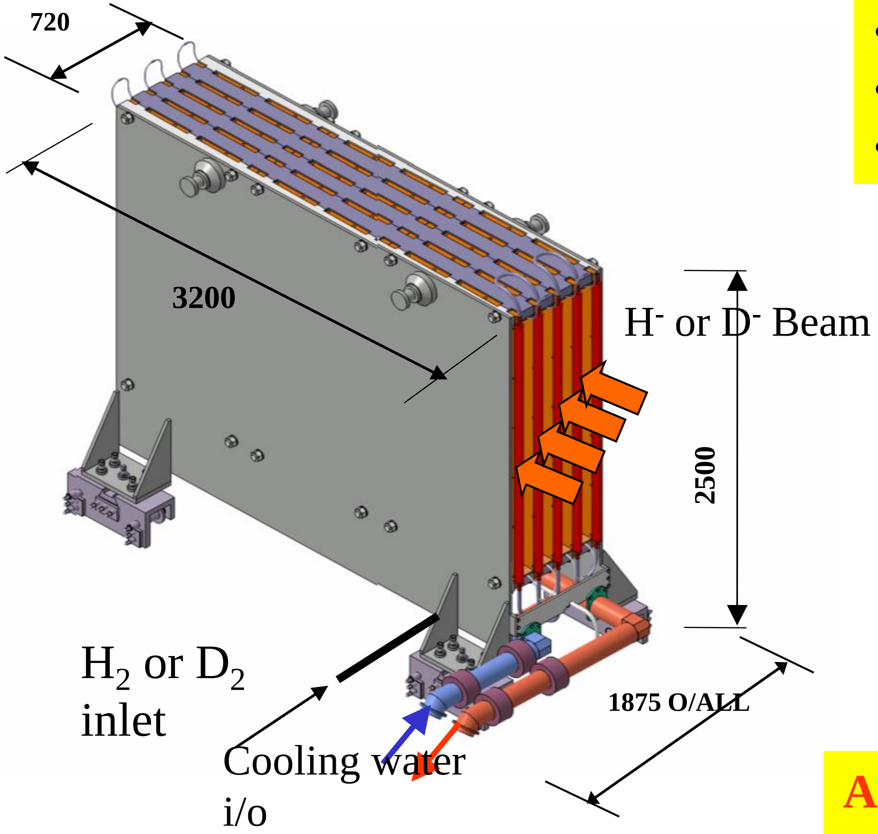
Neutralizer

Maximum neutralization efficency 60%





Neutraliser requirements



Power deposition from ion beam interception:

• on channel walls	4.2 MW (max. 0.5 MW/m^2)
• on leading edges	0.4 MW (max. 2.2 MW/m^2)
• Total power	4.6 MW

Heating cycles during ITER lifetime:

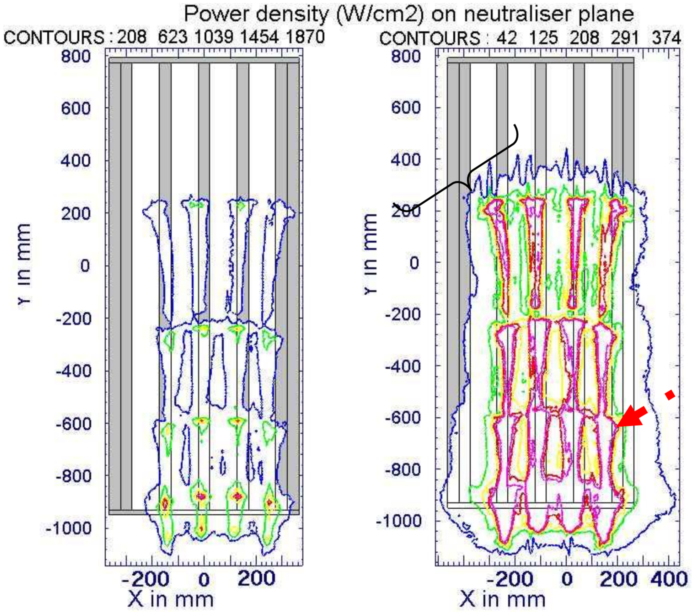
• Beam on/off	5x10^4
• Breakdowns	4.5x10^5

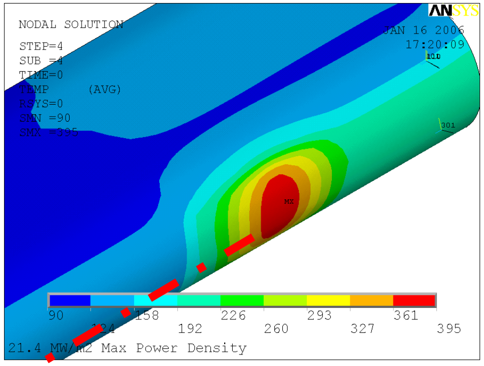
Additional power deposition due to electrons (stripping losses in SINGAP):

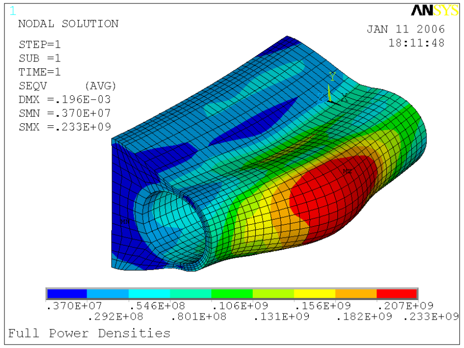
• on leading edges	2.7 MW (max. 26-30 MW/m^2)

Neutraliser : thermo-mechanical issues

Electron power loads on leading edges (front view) in SINGAP configuration







Peak power densities cause large temperature gradients and bending stresses on leading edges .

Work is in progress to :

a) assess optimization of cross section and cooling parameters to satisfy ITER fatigue lifetime requirements.

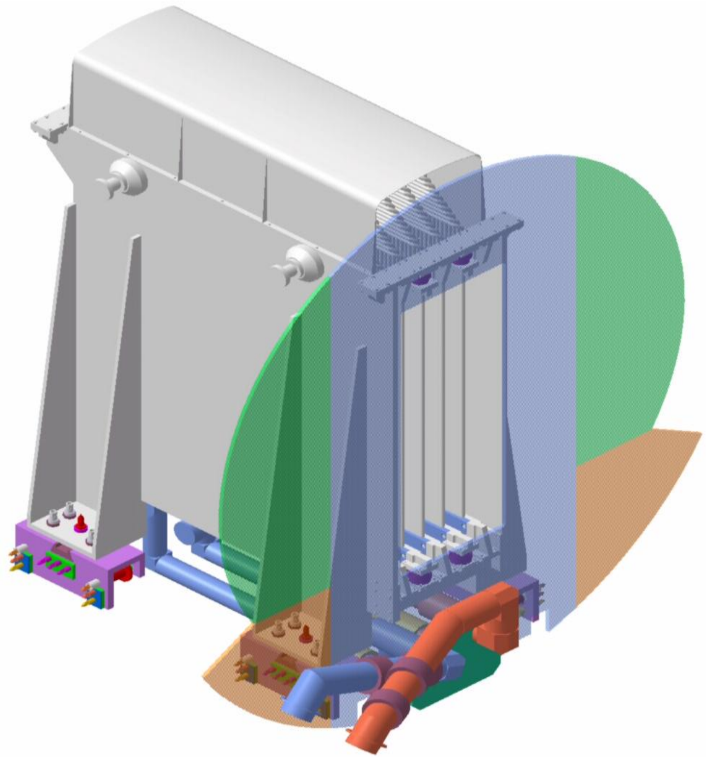
b) design new systems to deflect and dump the electrons

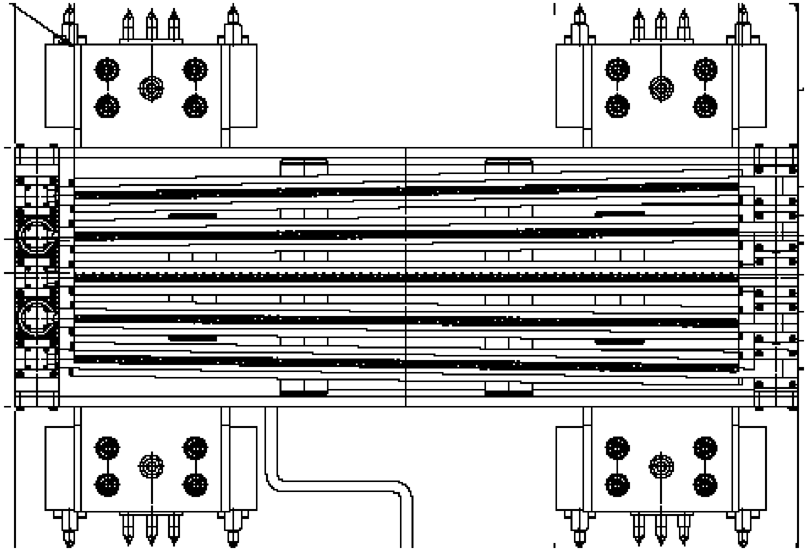
Deviated and focused electrons lead to local peaks of power density up to 25 - 33 MW/m^2

RID

Elettrostatico (10kV)

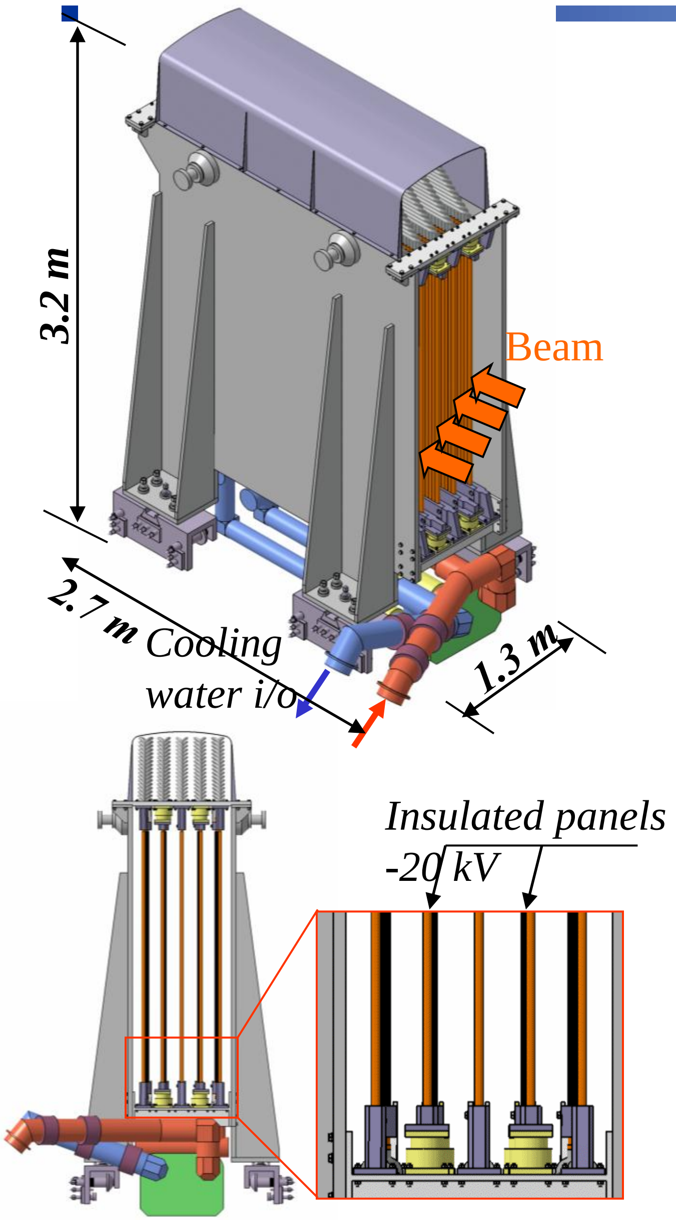
In uscita 60% n, 20% i+, 20% i-





Problemi : carico termico (fino a $20MW/m^2$ per 1 ora)

Residual Ion Dump



Function:

Deflecting by an electrostatic field and dumping on actively cooled panels the ions emerging from the neutraliser.

A bias potential up to - 20 kV is applied and modulated (± 5 kV/50 Hz) to reduce the power densities on the channel walls.

Power deposition from ions and neutrals deposition:

Total power (for 7 mrad beam divergence)	18.5 MW
Peak power density on panels	6 MW/m^2

Required heating cycles during ITER lifetime:

Beam on/off	$5x10^4$
Breakdowns	$4.5x10^5$

Calorimeter

Hydraulic connections

- 2 water coolant headers

Electrical connections:

-No connection

Gas line connections:

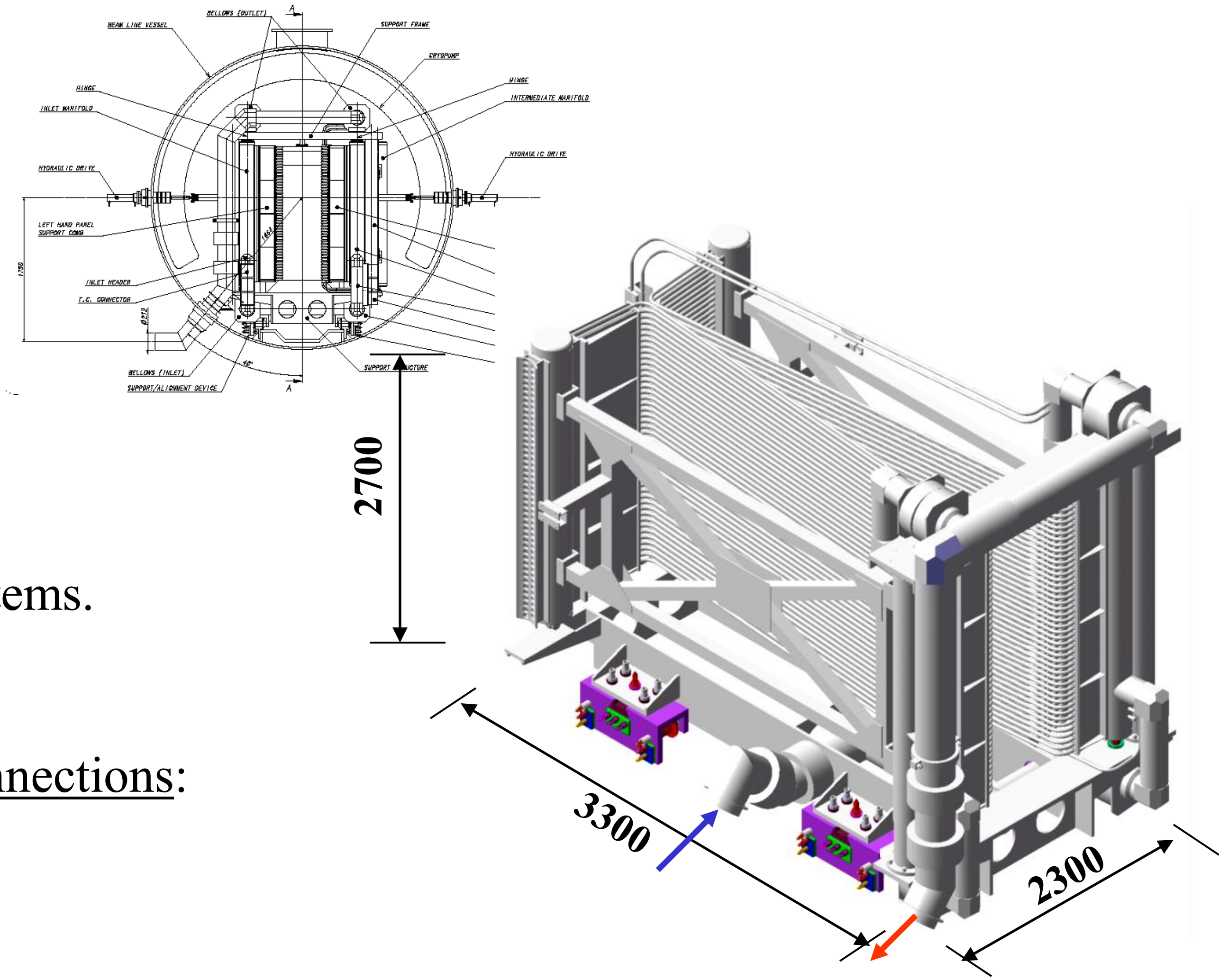
- No connection

Mechanical connections:

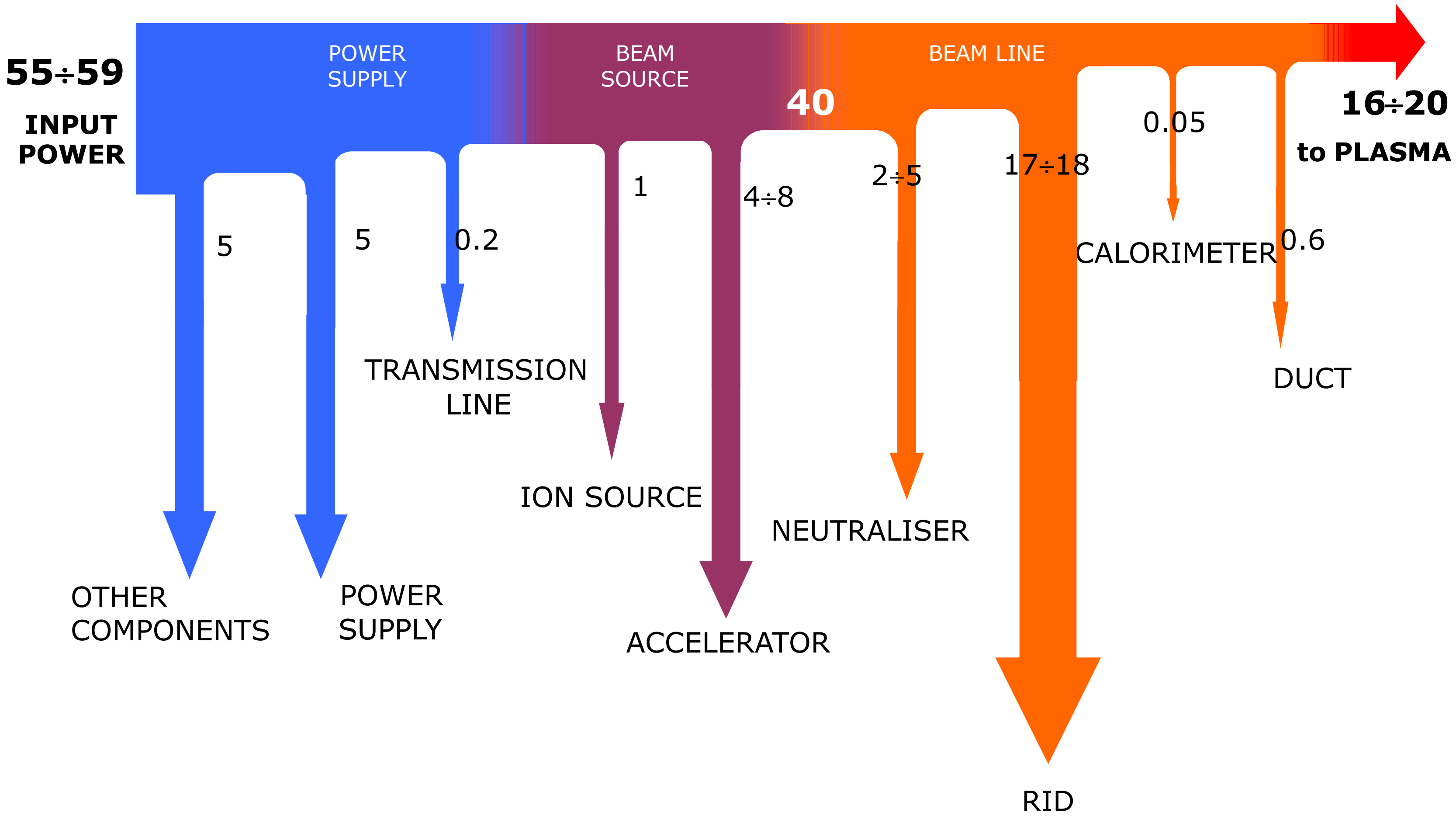
- 4 support/alignment systems.
- 2 hydraulic drives

Diagnostic/monitoring connections:

- 273 thermocouples
- 1 water flow meter



Power balance [MW] with 1MeV D Beam

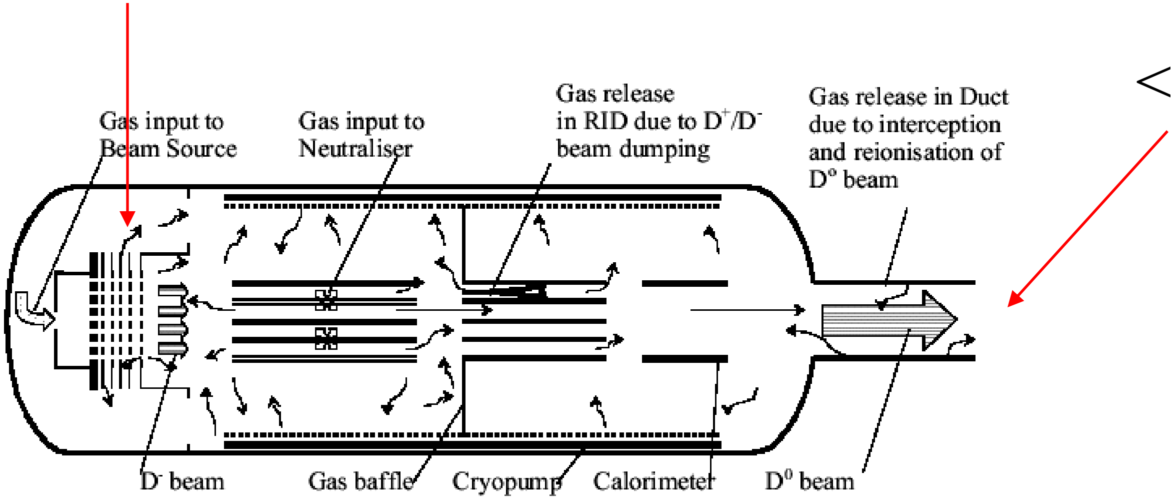


NBI in ITER:Gas flow

Gas input from:
a)source
b)neutraliser
c)surface

0.3Pa

<0.001Pa

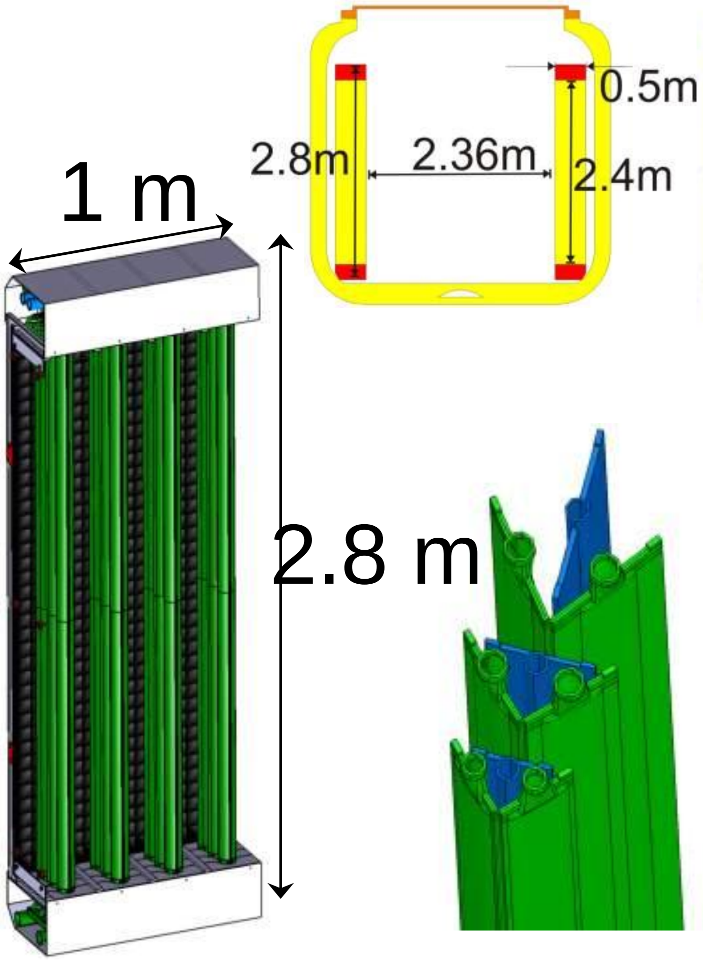


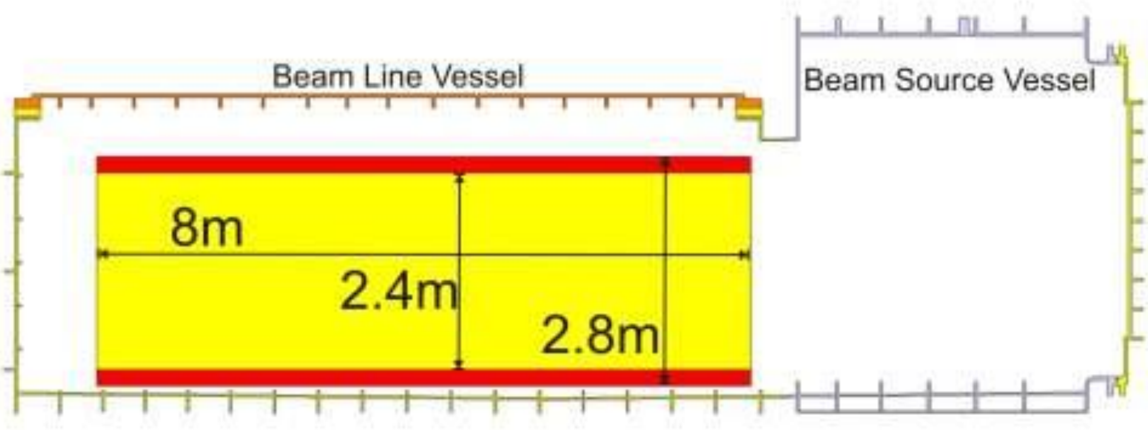
N 53 GR 421 01-06-20 W 0.1

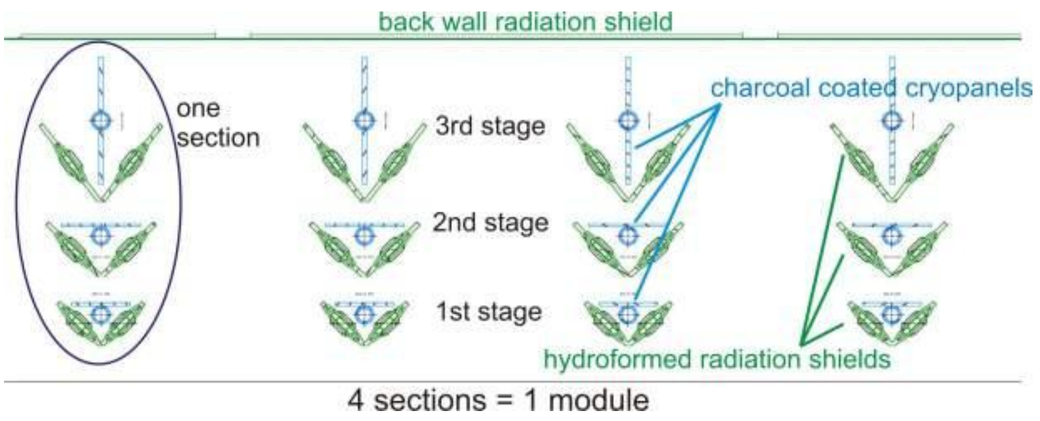
Gas output:
Pumping: 15-30 m^3Pa/s

Cryo-pump: S~$50m^2$

MITICA ➔ Details of the cryopumps







- NBTF Cryogenic needs

  - Cooling for the thermal shields around 80 K
    - -> High Pressure Gas Helium between 65 K and 90 K

  - Cooling for the cryopanels around 4.5 K
    - -> Supercritical Helium between 4.5 k and 6.5 K at 4 bar

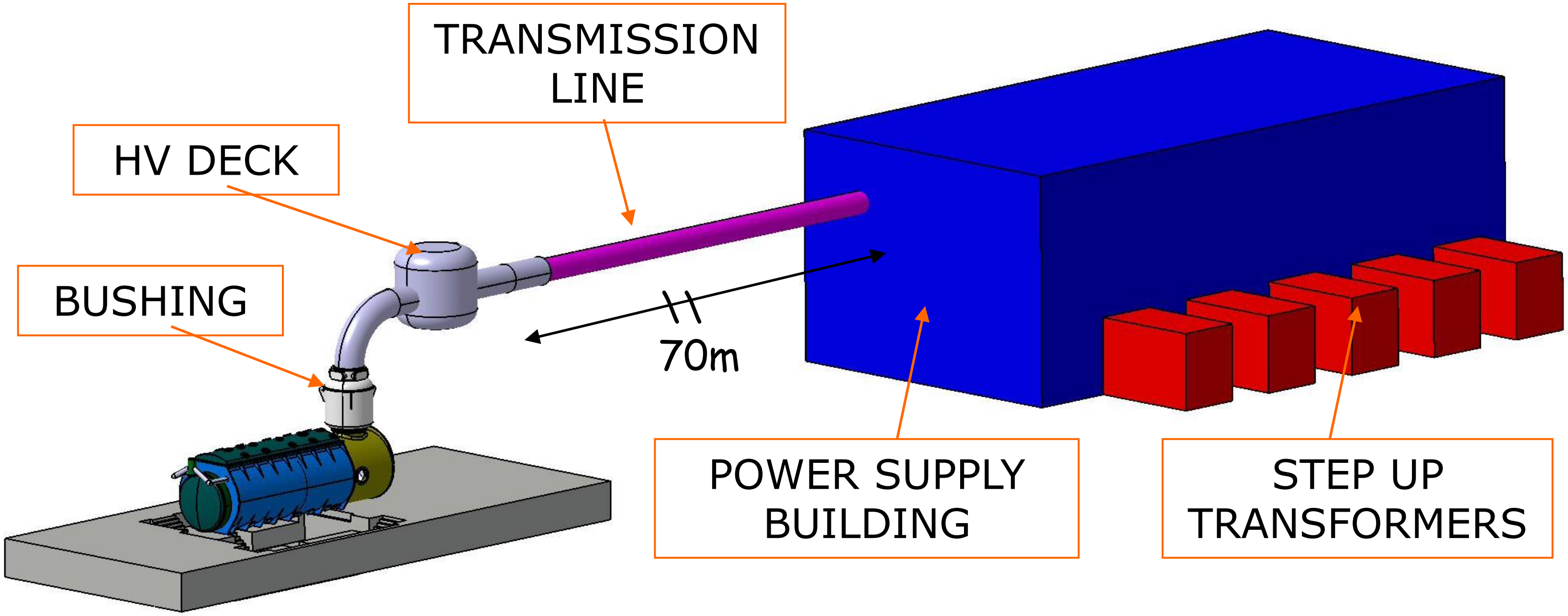
The Power Supply and Voltage Distribution System

The Power Supply (PS) and Voltage Distribution System provides the High Voltage (HV) to the accelerator grids (AGPS) and supplies the ion source (ISPS) and the auxiliary components.

The power is transmitted to the ion source and the acceleration grids via a HV transmission line, SF6 insulated for -1MV dc to ground.

PS for the ion source (IS) and pipes for IS and grids cooling are hosted inside the HVD insulated in SF6

An HV bushing provides the interface between the TL (in SF6) and the ion source and the acceleration grids in vacuum.

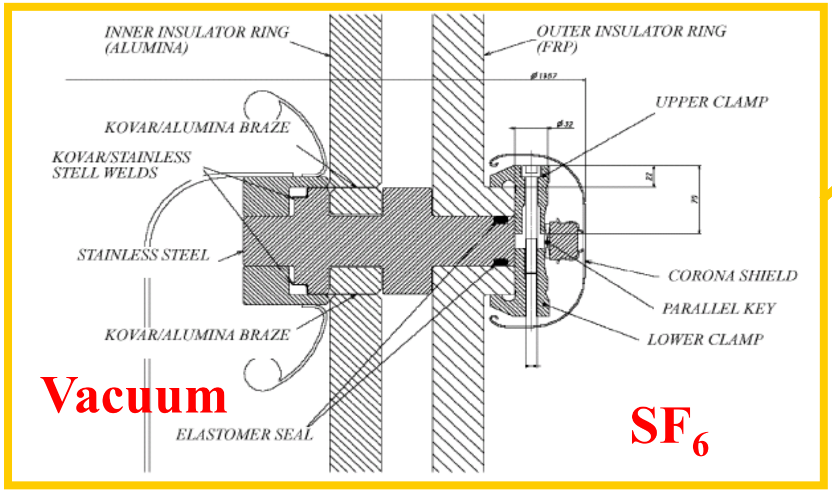


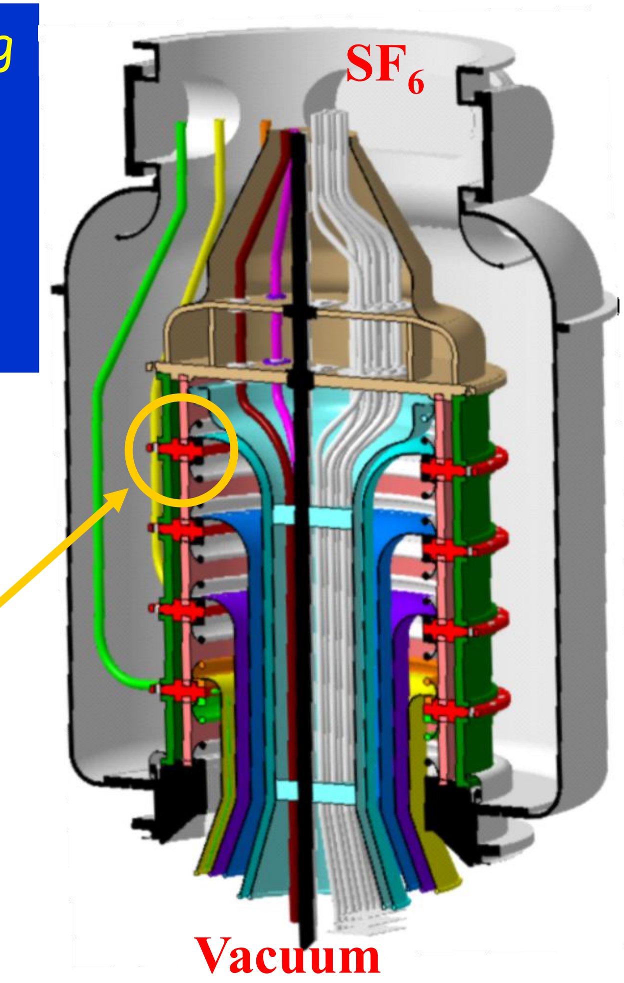
Bushing: reference design

High voltage feedthrough for all electrical and cooling services of the beam source
Interface between the gas insulated (SF_6 at 0.6MPa) transmission line and the ITER primary vacuum
Barrier for tritium containment and vacuum confinement

Large ceramic rings recently manufactured

(T. Inoue et al.,P3-B-362)





Double barrier obtained with two insulating rings (alumina on vacuum side and Fiber Reinforced Polymer (FRP) on gas side) and guard gas in the interspace (N_2)

MAMuG 1MV bushing

Test Facility for NBI :Status R&D

R&D at V~1MV (I ~ 100 mA) at present in two test facilities:
Naka (J) and Cadarache (F)

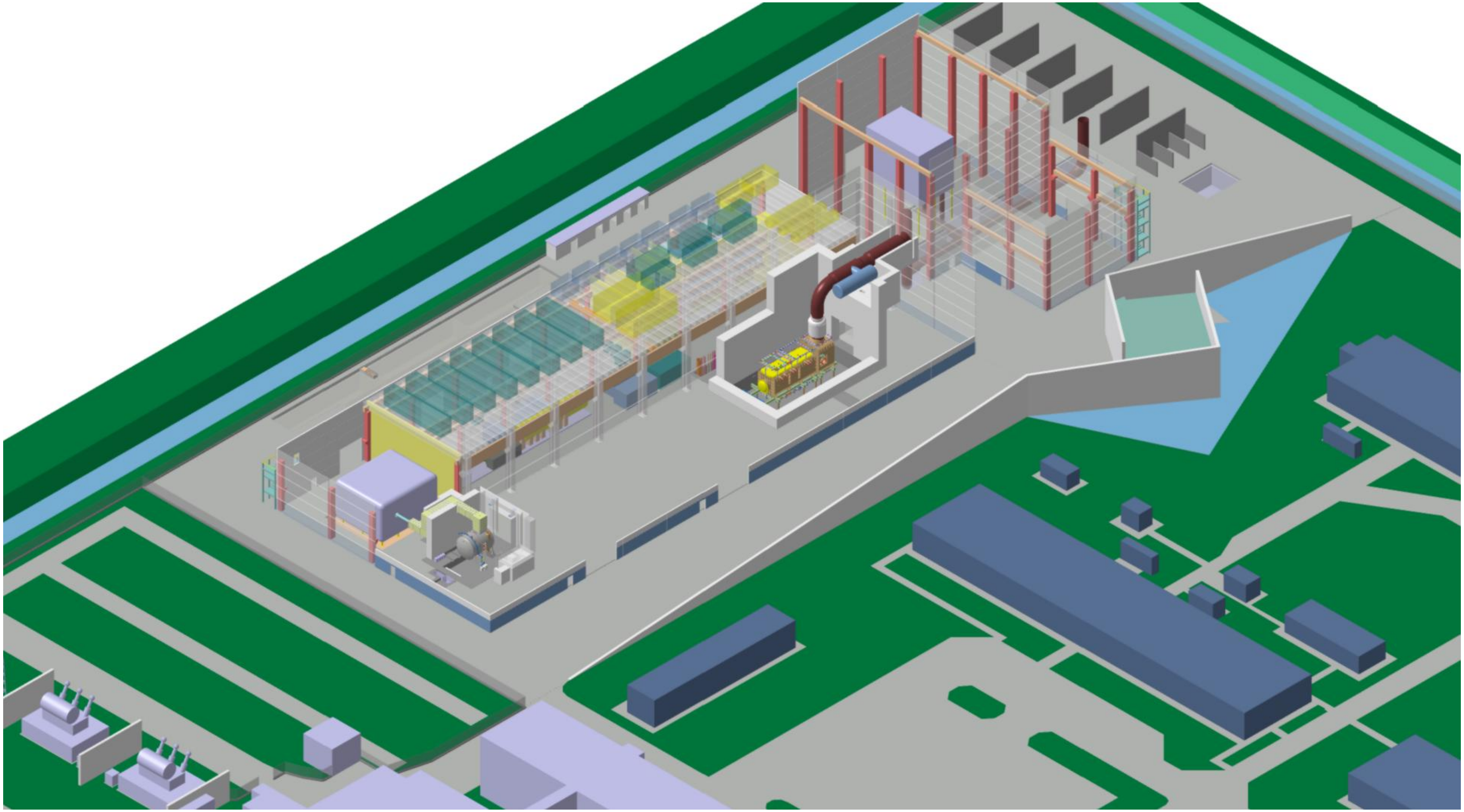
Negative ion NBI best performance in JT-60U :

2.6 MW of 360 keV beams for 10 s in H.
3.4 MW of 418 keV beams for 0.3 s in H
6.2 MW of 381 keV beams for 1.65 s in H.
5.8 MW of 400 keV beams for 0.86 s in D.

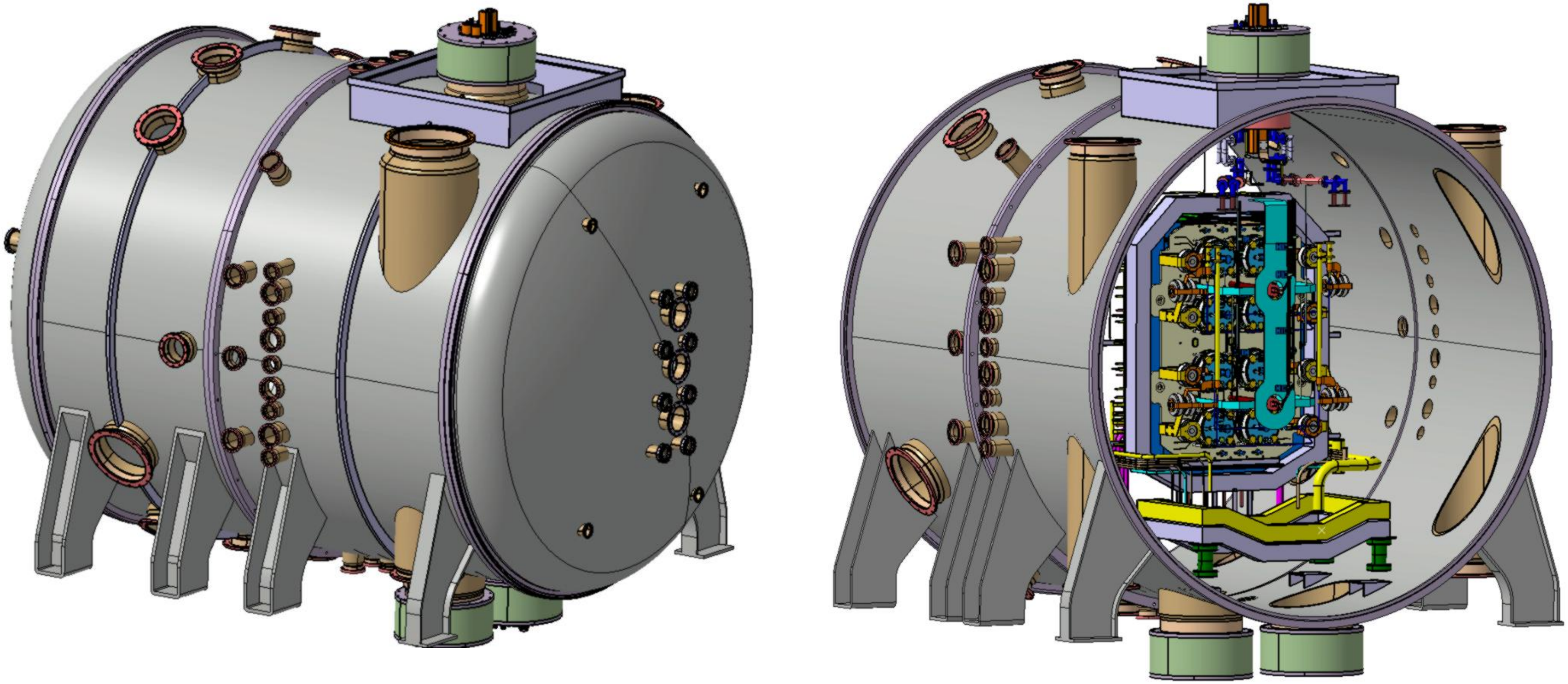
ITER requirements

Power delivered to the plasma per injector	16.5MW
Beam energy	1 MeV
Ion species	D^-
Pulse length	$\leq$ 3,600 s

PRIMA ➔ Buildings and layout of plants



SPIDER ➔ overall view and requirements

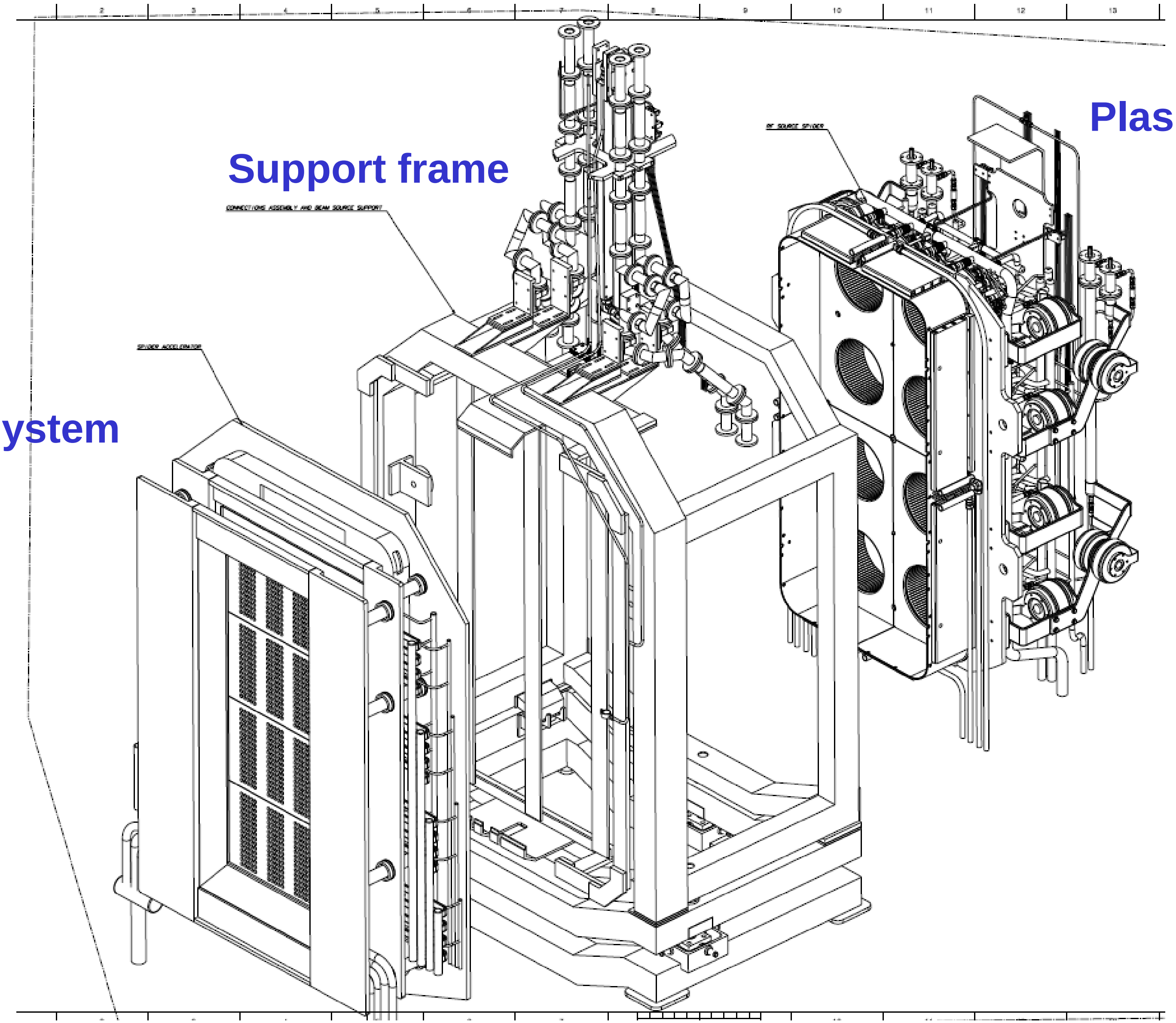


	Ion	Energy (keV)	Stripping loss (%)	**Extr. Ion J (A/m²)**	**Extr. Ion I (A)**	Accel. I (A)
HNB D	D-	1000	29	**290**	**48**	40
HNB H	H-	870	24	**330**	**56**	46

$\frac{e}{D^-} < 1$ for HNB

uniformity

SPIDER ➔ The beam system

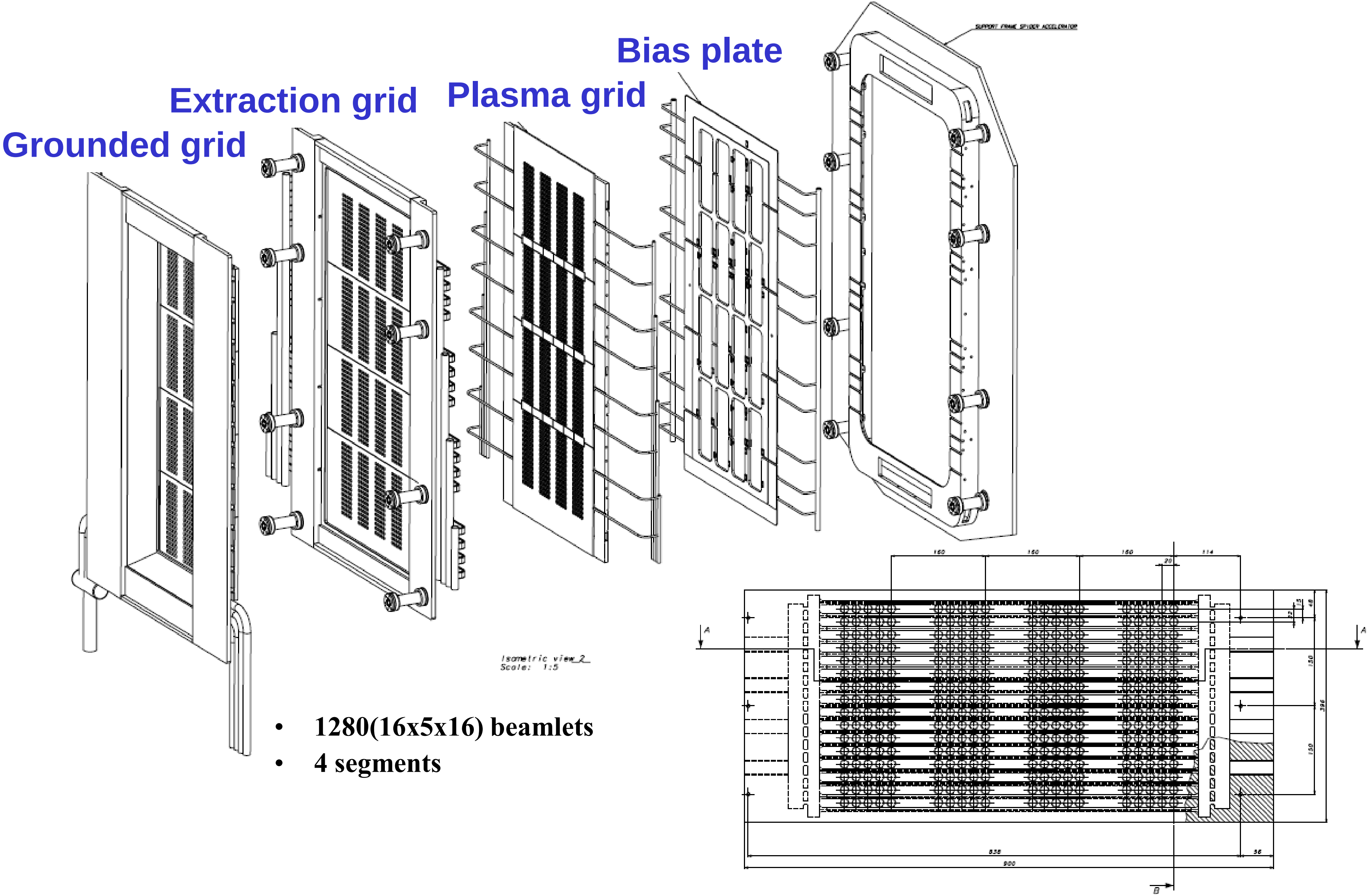


Plasma source

Support frame

Grids system

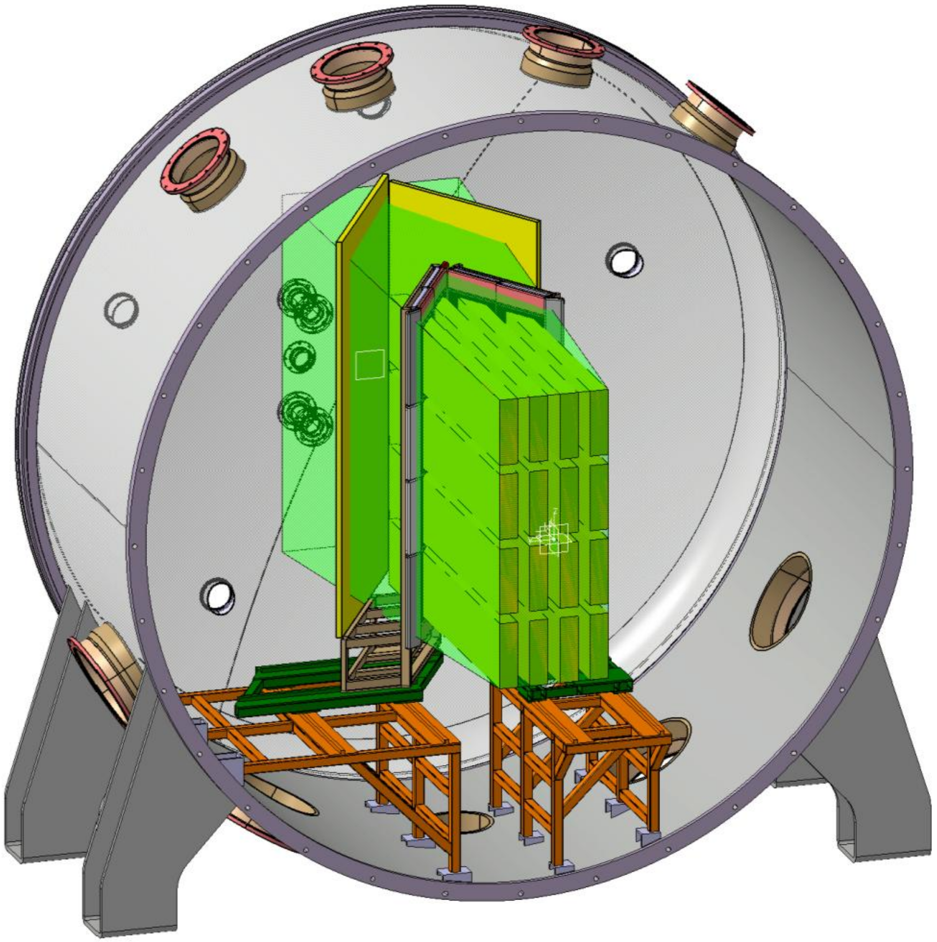
SPIDER ➔ The grids system



- **1280(16x5x16) beamlets**
- **4 segments**

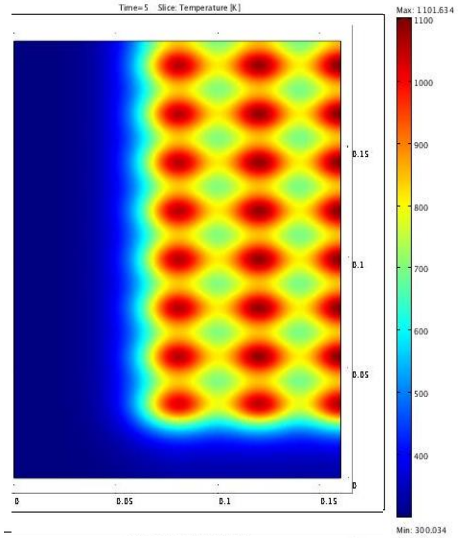
SPIDER ➔ The diagnostic calorimeter

δ = 3mrad; γ = 60°, d=1m, t= 5 s



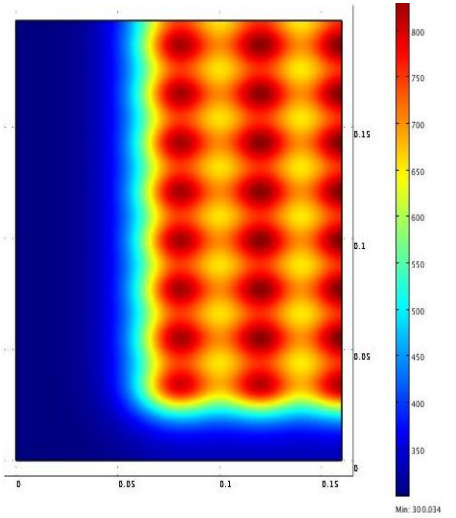
front side

T_{max}>1100 K

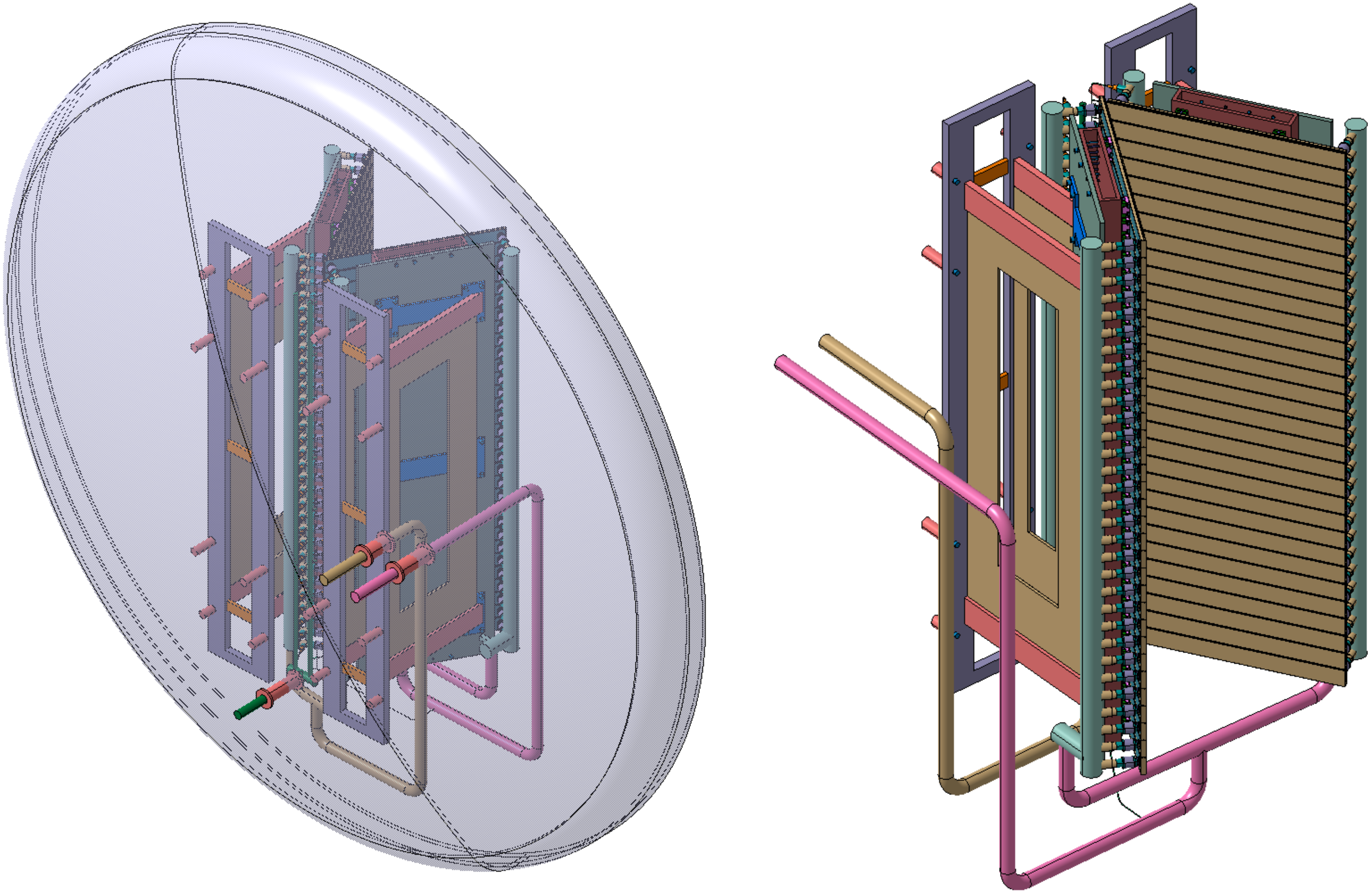


rear side

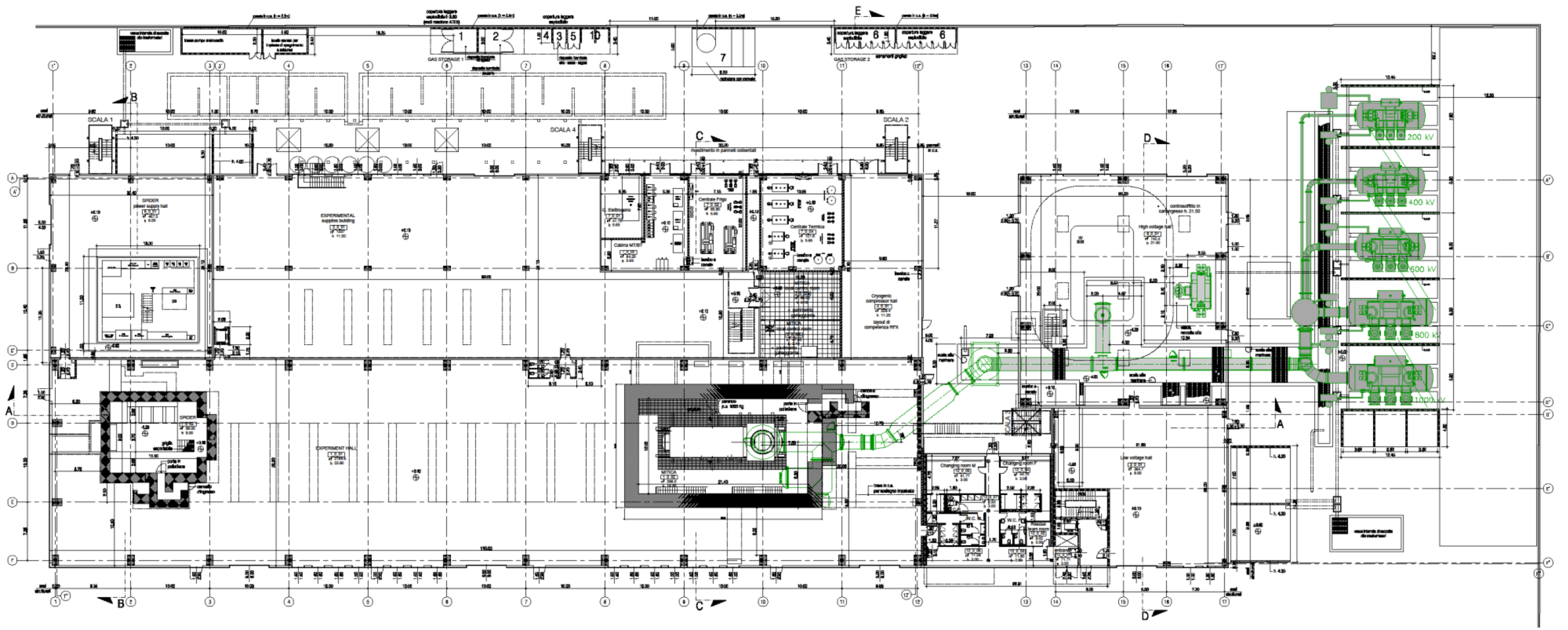
T_{max} ≈ 848 K



SPIDER ➔ beam dump (INDA contribution)



PRIMA ➔ Ground floor



PRIMA - SPIDER - MITICA

PRIMA
Padova **R**esearch on **I**TER **M**egavolt **A**ccelerator

FUSION FOR ENERGY

KIT Karlsruhe Institute of Technology

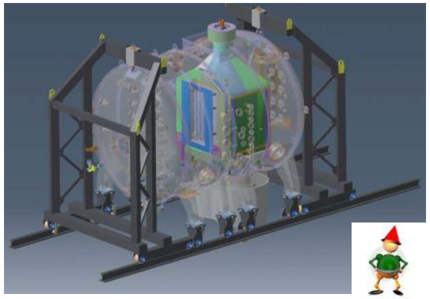
CCFE CULHAM CENTRE FOR FUSION ENERGY

IPP Max-Planck-Institut für Plasmaphysik
EURATOM Association

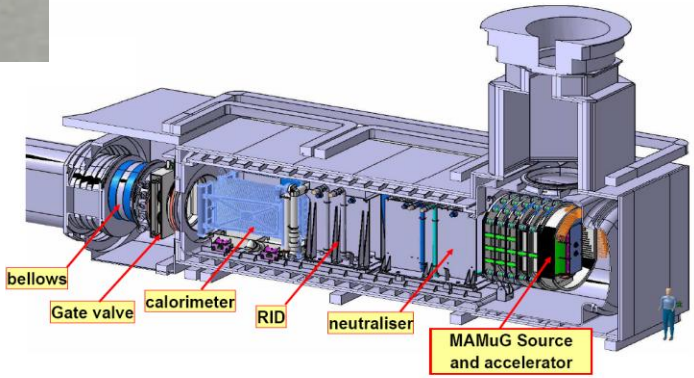
cea

cnrs

UNIVERSITÉ PARIS-SUD 11



SPIDER
Source for **P**roduction of **I**on of
Deuterium **E**xtracted from **R**f plasma



MITICA
Megavolt **IT**ER **I**njector
&
Concept **A**dvancement

Conclusion for the NBI Test Facility

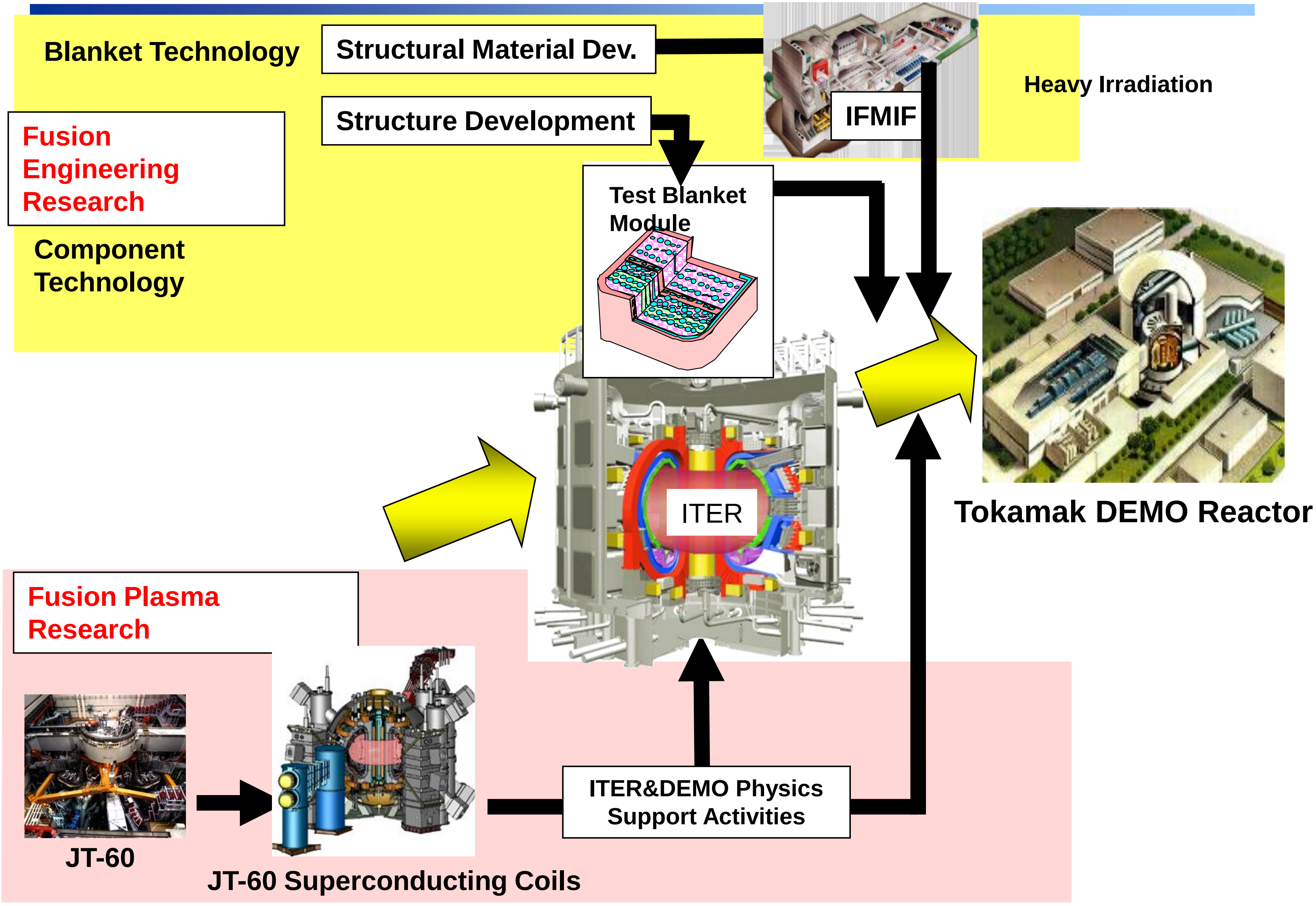
- PRIMA buildings: (1 year to host first SPIDER plants, 1.5 year of overall construction)
  - The design is complete including layout of plants and components
  - The license from the public authorities for the construction of the PRIMA buildings has been obtained
  - The procedure for the Call for Tender has been approved

- PRIMA cooling system design ready at CfT level

- SPIDER (to be ready for the first quarter of 2013)
  - Design at the level of Call for Tender of most of the main components ready
    - ISESP
    - Transmission line and HVD
    - Vacuum vessel
    - Beam source
  - Design at the level of Call for Tender of other components and systems will be ready soon:
    - Vacuum system and gas injection system (also for MITICA)
    - 100 kV PS (INDA)
    - Beam dump (INDA)
    - Short pulse calorimeter
    - Diagnostics not embedded
  - Control system design is ongoing as scheduled and R&D on HW and SW is also in progress

- MITICA (to be ready for the half of 2015)
  - **ISEPS CfT launched by F4E**
  - AGPS design is in progress
  - Design of JADA contributions components is under revision to be ready to sign the PA soon
  - Design of other components and systems are in progress or under review or scheduled to be restarted

Road Map to Fusion DEMO Reactor



Blanket Technology

Structural Material Dev.

Structure Development

IFMIF

Heavy Irradiation

Fusion Engineering Research

Component Technology

Test Blanket Module

ITER

Tokamak DEMO Reactor

Fusion Plasma Research

JT-60

JT-60 Superconducting Coils

ITER&DEMO Physics Support Activities

Broader Approach

- In February 2007, Euratom and the Japan signed the Broader Approach agreement. This aims to complement the ITER Project and to accelerate the realisation of fusion energy by carrying out R&D and developing some advanced technologies for future demonstration fusion power reactors (DEMO). Within the Broader Approach, three main projects are being implemented

The BA Activities comprise the following three Projects

1) Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA)

2) International Fusion Energy Research Center (IFERC),
   a) A DEMO Design and, R&D coordination Center
   b) A Computational Simulation Center
   c) An ITER Remote Experimentation Center

3) Satellite Tokamak Programme
   Upgrade of JT-60 Tokamak to an advanced superconducting tokamak (JT-60SA)

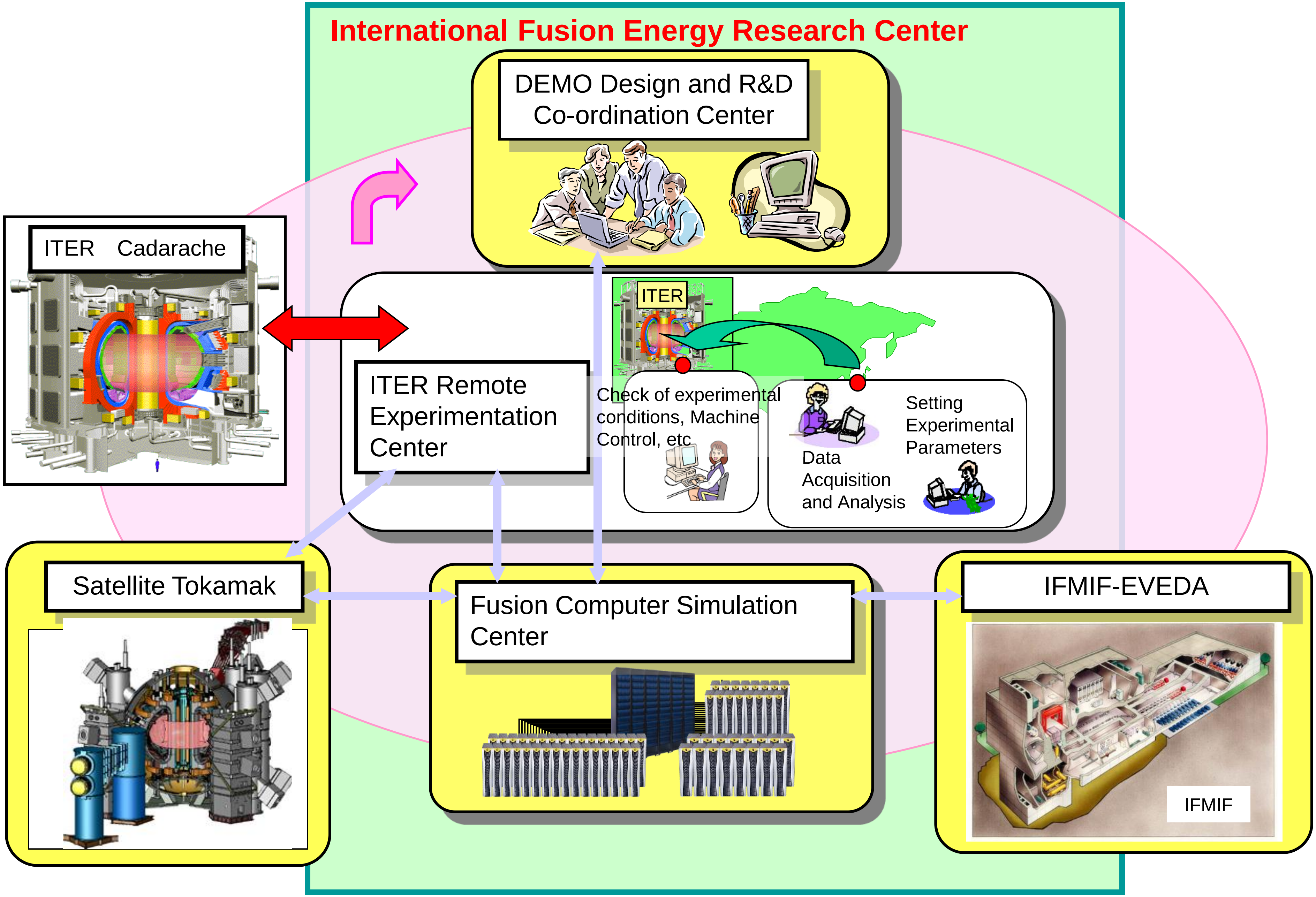
IFMIF

- The first project will complete the detailed and fully integrated engineering design of the **International Fusion Materials Irradiation Facility (IFMIF)**. Fusion as a major energy source will require materials which maintain their essential physical properties and which do not remain highly radioactive for extended periods of time after exposure to the harsh thermal and irradiation conditions inside a fusion reactor. IFMIF will allow testing and qualification of advanced materials in an environment similar to that of a future fusion power plant.

IFERC

- The second project is the **International Fusion Energy Research Centre (IFERC)**. The missions of the centre include the co-ordination of DEMO Design and R&D activities, large scale simulation activities of fusion plasmas by super-computer and remote experimentation activities to facilitate a broad participation of scientists into ITER experiments.

International Fusion Energy Research Center



STP

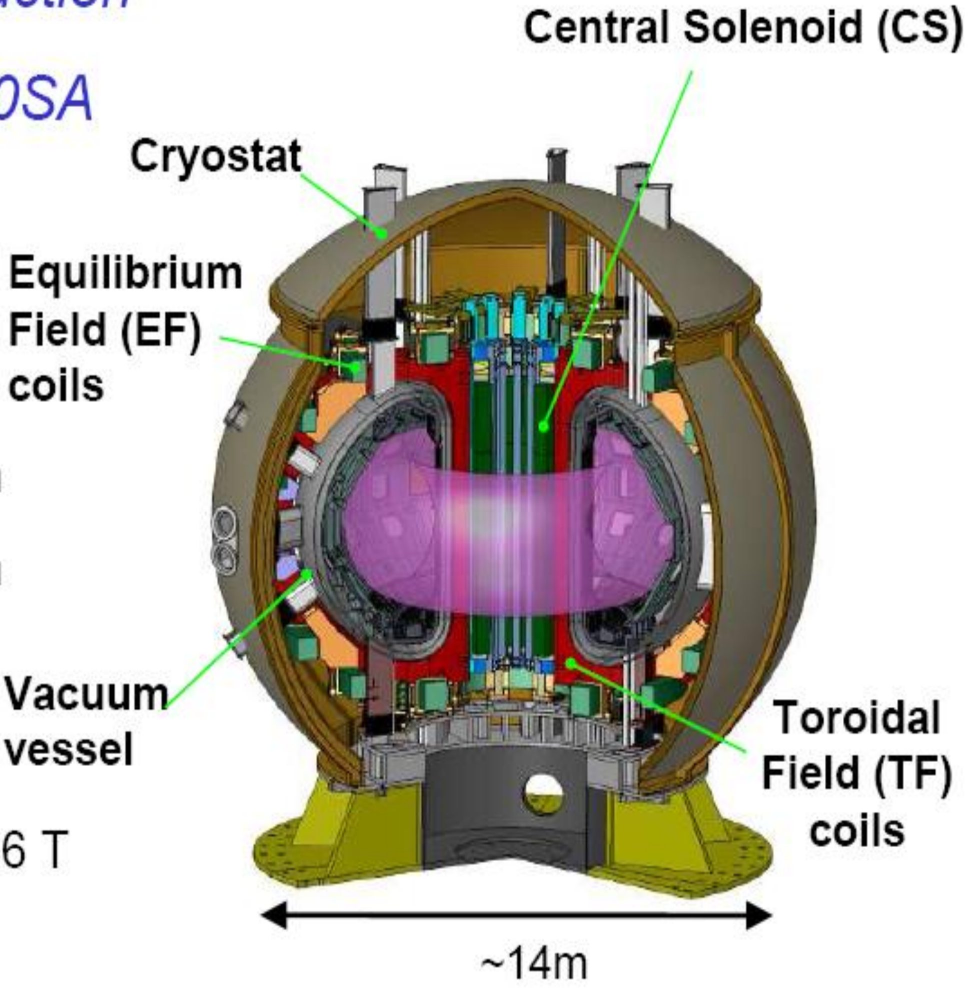
- The third project is the Japan-EU **Satellite Tokamak Programme (STP)**. During ITER construction, major experimental facilities will be required to develop operating scenarios and address key physics issues for an efficient start up of ITER experimentation and for research towards DEMO. The STP in Japan has been identified as a device which could fulfil these objectives. It will therefore be upgraded to an advanced superconducting tokamak and used by Europe and Japan as a “satellite” facility to ITER.

Introduction

JT-60SA

Nominal parameters

- Plasma major radius = 2.95 m
- Plasma minor radius = 1.18 m
- Aspect ratio = 2.5
- Plasma current = 5.5 MA
- Toroïdal magnetic Field = 2.26 T



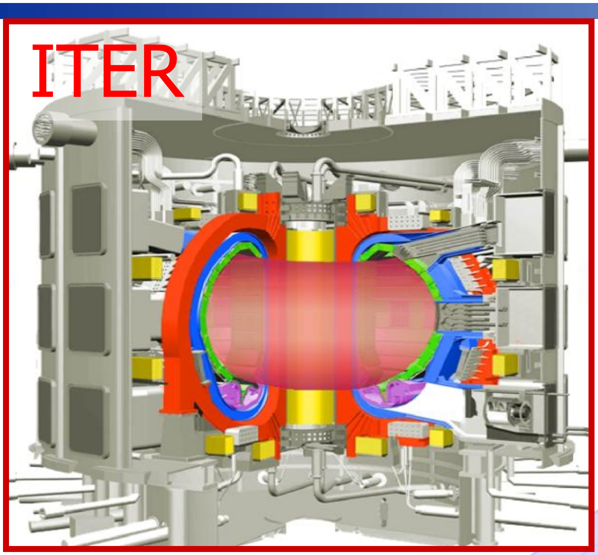
JT-60SA OBJECTIVES

- The mission of the JT-60SA project is to contribute to the early realization of fusion energy by addressing key physics issues for ITER and DEMO.
-
- JT-60SA is a fully superconducting tokamak capable of confining break-even equivalent high-temperature deuterium plasmas.
- It is designed to allow the optimisation of configurations for ITER and DEMO by operating with a wide range of plasma shapes (elongations and triangularities) and aspect ratios (A=R/a down to ~ 2.5) including that of ITER, with the capability to operate in both single and double null divertor configurations.
- The machine will allow the exploration of ITER-relevant high density plasma regimes well above the H-mode power threshold with up to 41 MW of high power electron cyclotron (EC) resonance and positive and negative ion neutral beam (NB) heating.

- JT-60SA is designed to study power and particle handling for 100s at high power with water-cooled divertors compatible with maximum heat fluxes of 15MW/m2.
- The machine will be able to explore full non-inductive steady-state operation with 10MW/500keV tangential NB current drive (CD) and 7MW of ECCD.
- It will be able to explore high beta plasma regimes by using a stabilizing shell covered with ferritic plates, internal resistive wall mode (RWM) stabilizing coils, and a high power H&CD system.
- JT-60SA will be equipped with a remote handing system to allow maintenance of in-vessel components compatibly with the planned annual yield of 1.5x1021neutrons/year.

International Road Map

Advanced Materials are at a critical path



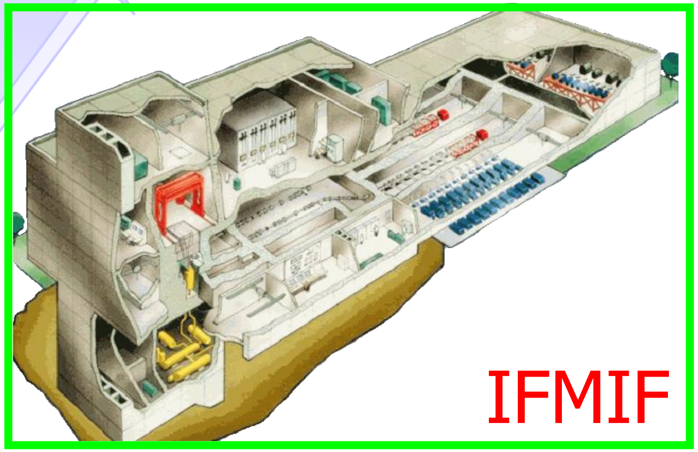
1-3 dpa/lifetime

Plasma Facing Materials

Structural Materials

Functional Materials

< 150 dpa

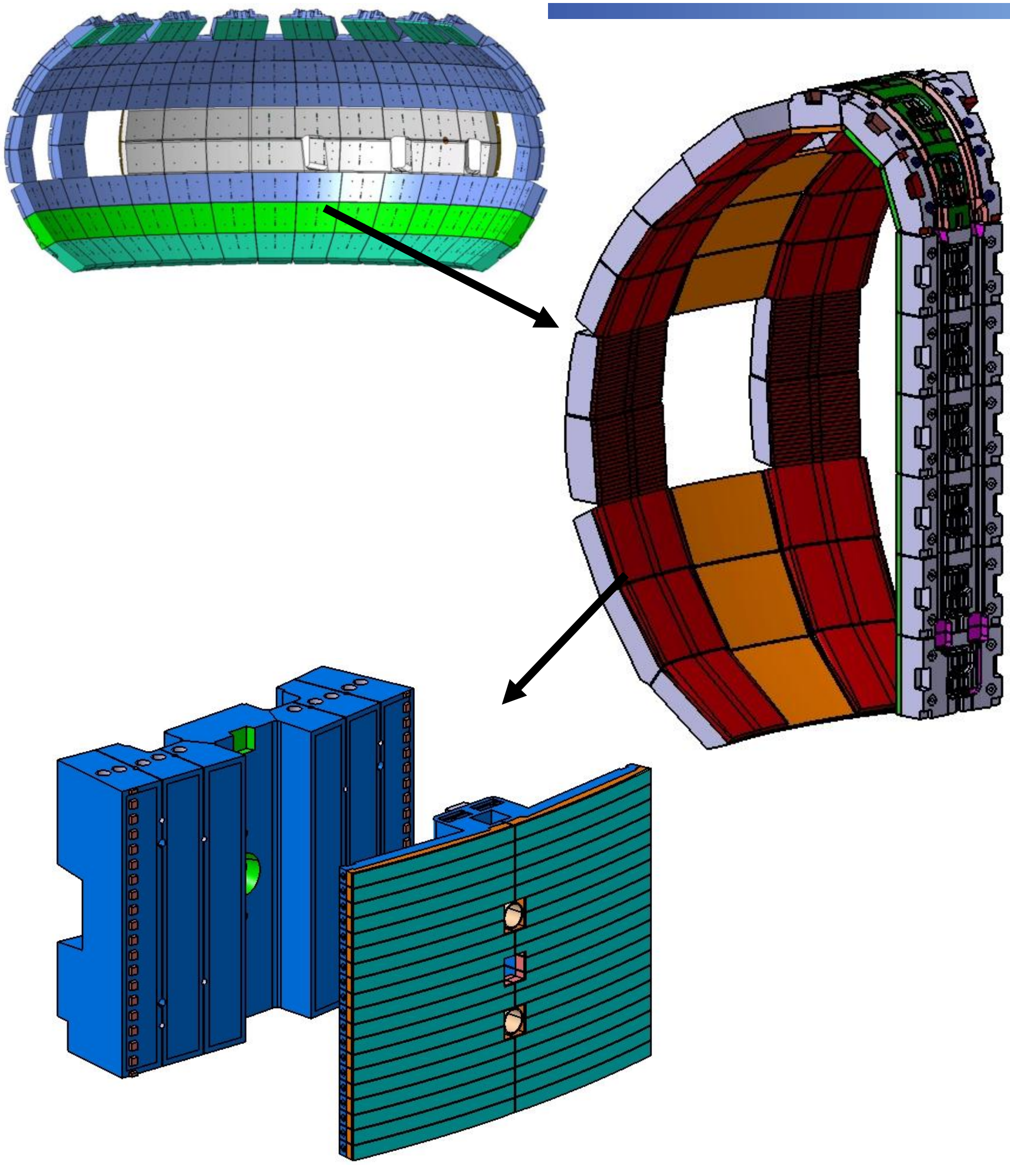


20-40 dpa/year

International Fusion Materials Irradiation Facility

Dpa: displacement per atom: measure the integral of received radiation

Blanket Status & Accomplishments



Status & Accomplishments

- Generic blanket / FW design in progress
- Improved FW shaping in progress (will allow removal of moveable limiters)
- Qualification of mock-ups ongoing with first successful results
- EM analysis benchmarking in progress
- RH design of hydraulic connection in progress
- Thermal & mechanical loads updated and finalized

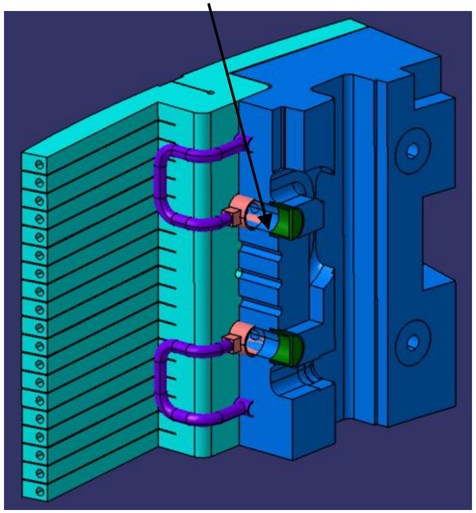
Blanket Activities

Thermal loads have been updated

Parallel Power (MWm^{-2})	Outer Wall	Inner Wall	2nd X-point
ELM's	5		25
Between ELM's	3.5	5.8	8.5
Ramp-up (Limiters)	170		
Ramp-up (Wall)	56	48	
VDE / Loss of Radial Control	800	1640	1300
NBI Shine Through (along beam)	4		

2.2 MWm^{-2}
at 5°

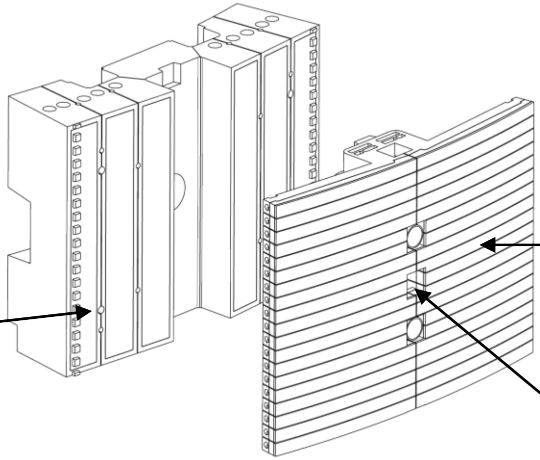
Developing Simplified Hydraulic Connector



Developing in-vessel separable First Wall

No additional cutting/welding in hot cell

No additional access holes for Shield Block mounting



Developing single FW panel to reduce number of operations

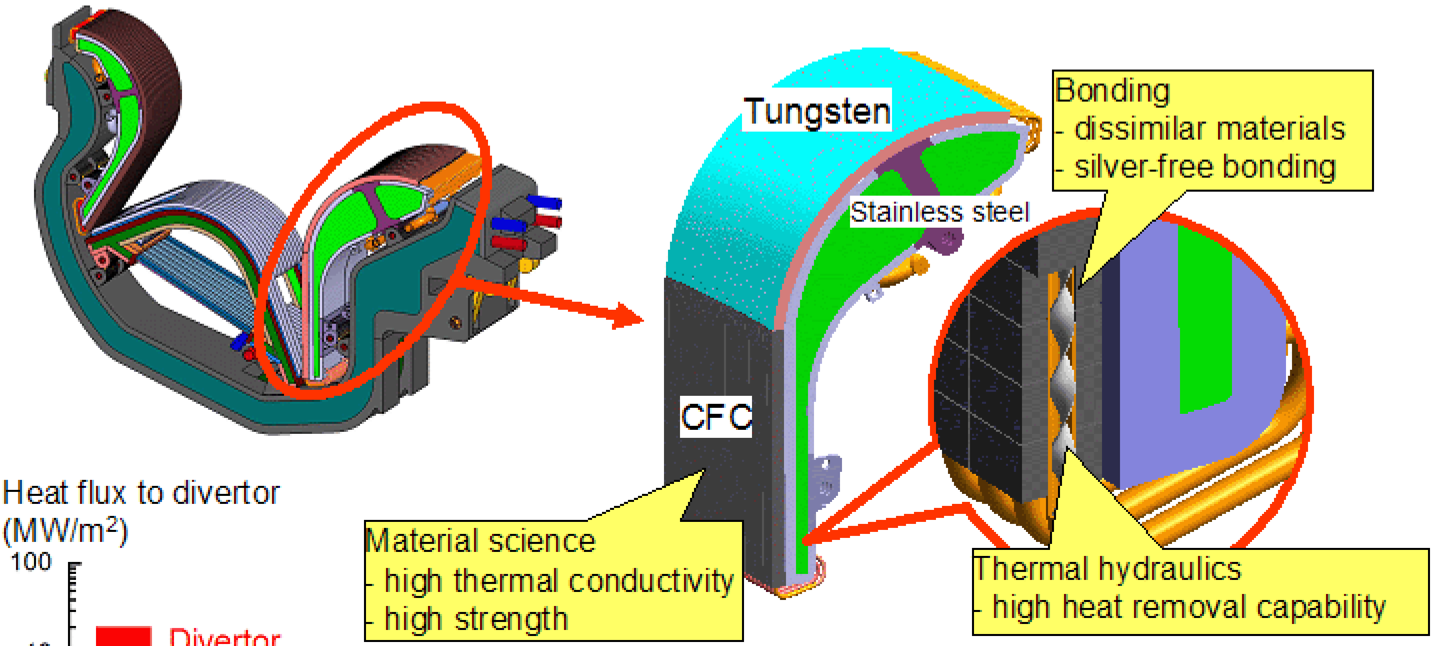
Shaped to shadow leading edges, and shield access holes

Horizontal fingers to reduce Halo and Eddy loads

Fingers to simplify manufacture, and permit other heat flux technologies

Bolted mechanical connection

R&D issues on ITER divertor

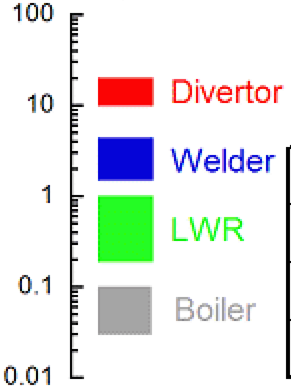


Bonding
- dissimilar materials
- silver-free bonding

Material science
- high thermal conductivity
- high strength

Thermal hydraulics
- high heat removal capability

Heat flux to divertor (MW/m^2)



Candidate materials of ITER divertor

Armor material	Tungsten, Carbon fiber composite (CFC)
Cooling tube/Heat sink	Copper-chromium-zirconium (CuCrZr)
Support structure	Stainless steel (316SS-LN)
Bonding method	Brazing, Diffusion bonding (incl. HIP bonding)

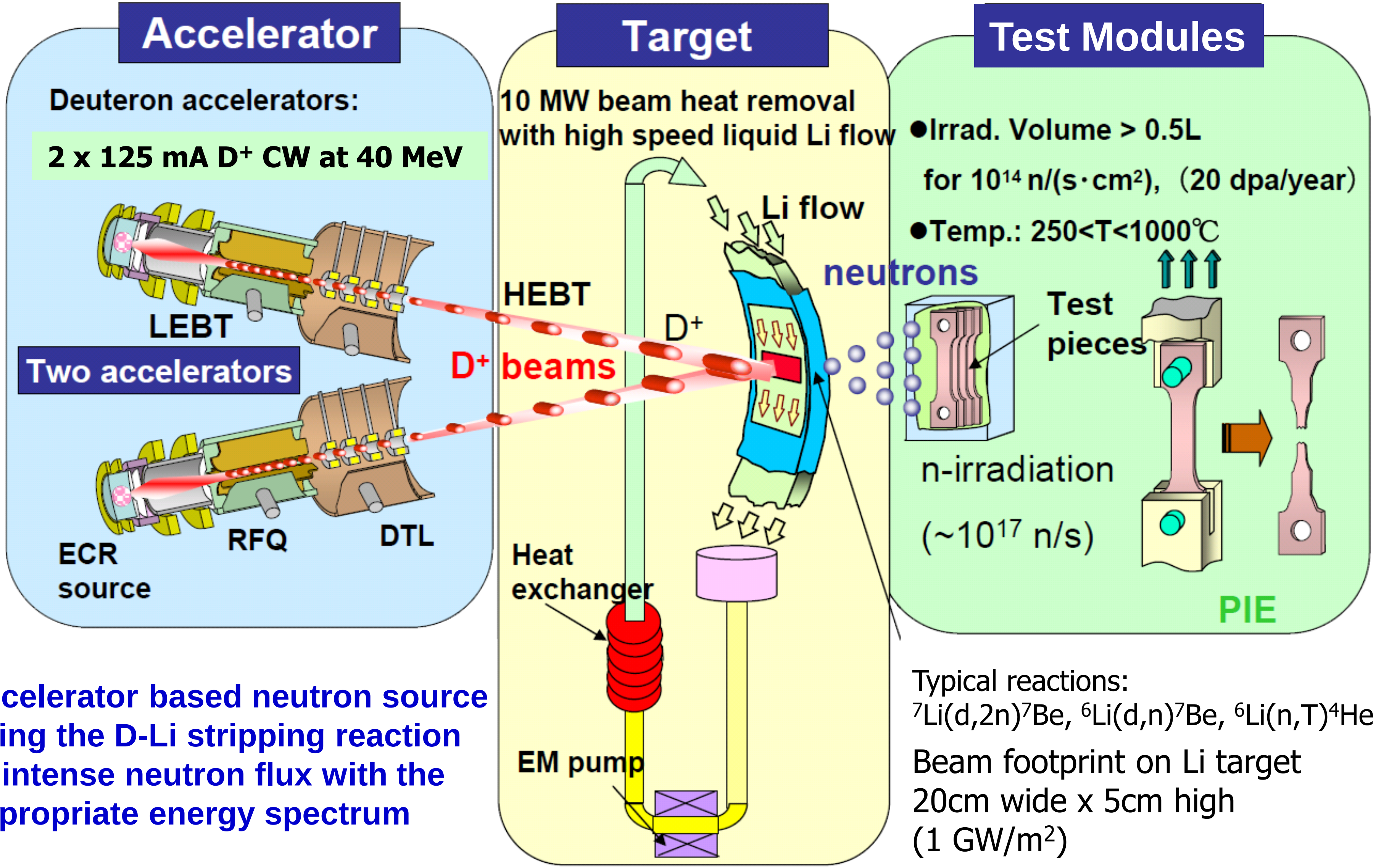
International Fusion Materials Irradiation Facility

- The International Fusion Material Irradiation Facility is an international scientific research program designed to test materials for suitability for use in a fusion reactor. The IFMIF, planned by Japan, the European Union, the United States, and the Russian Federation, and managed by the International Energy Agency, will use an accelerator-based neutron source to produce a large flux of neutrons, in enough volume and over enough time, to test the long-term behavior of materials under conditions like those expected at the inner wall of a fusion reactor.

International Fusion Materials Irradiation Facility

- The IFMIF will consist of two parallel accelerators, each about 50 m long, producing beams of deuterium nuclei. These, on contact with a lithium target, will be converted into high-energy neutrons and used to irradiate materials specimens and test components.
- Preparation for IFMIF construction is going on, however operation testing of materials is not scheduled until roughly 2017. IFMIF is unlikely, therefore, to be useful in the construction of the first-generation ITER reactor, but will provide important construction information for commercial fusion reactors after ITER.

IFMIF Principles



Accelerator

Deuteron accelerators:

2 x 125 mA D^+ CW at 40 MeV

Two accelerators

Target

10 MW beam heat removal with high speed liquid Li flow

Test Modules

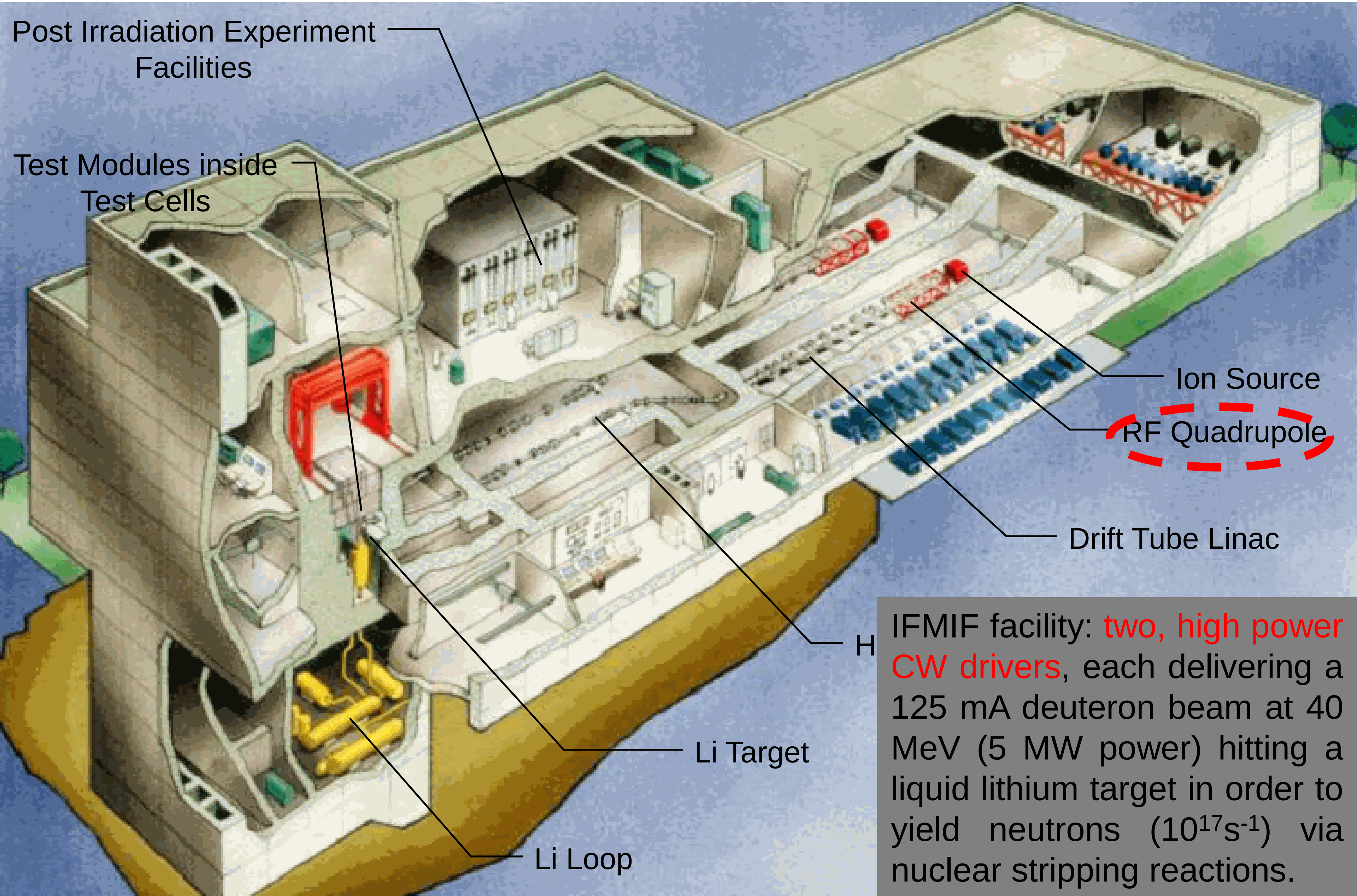
- Irrad. Volume > 0.5L for 10^{14} n/(s·cm^2), (20 dpa/year)
- Temp.: 250<T<1000℃

Accelerator based neutron source using the D-Li stripping reaction ⇒ intense neutron flux with the appropriate energy spectrum

Typical reactions:
$^{7}\text{Li}(d,2n)^{7}\text{Be}$, $^{6}\text{Li}(d,n)^{7}\text{Be}$, $^{6}\text{Li}(n,T)^{4}\text{He}$

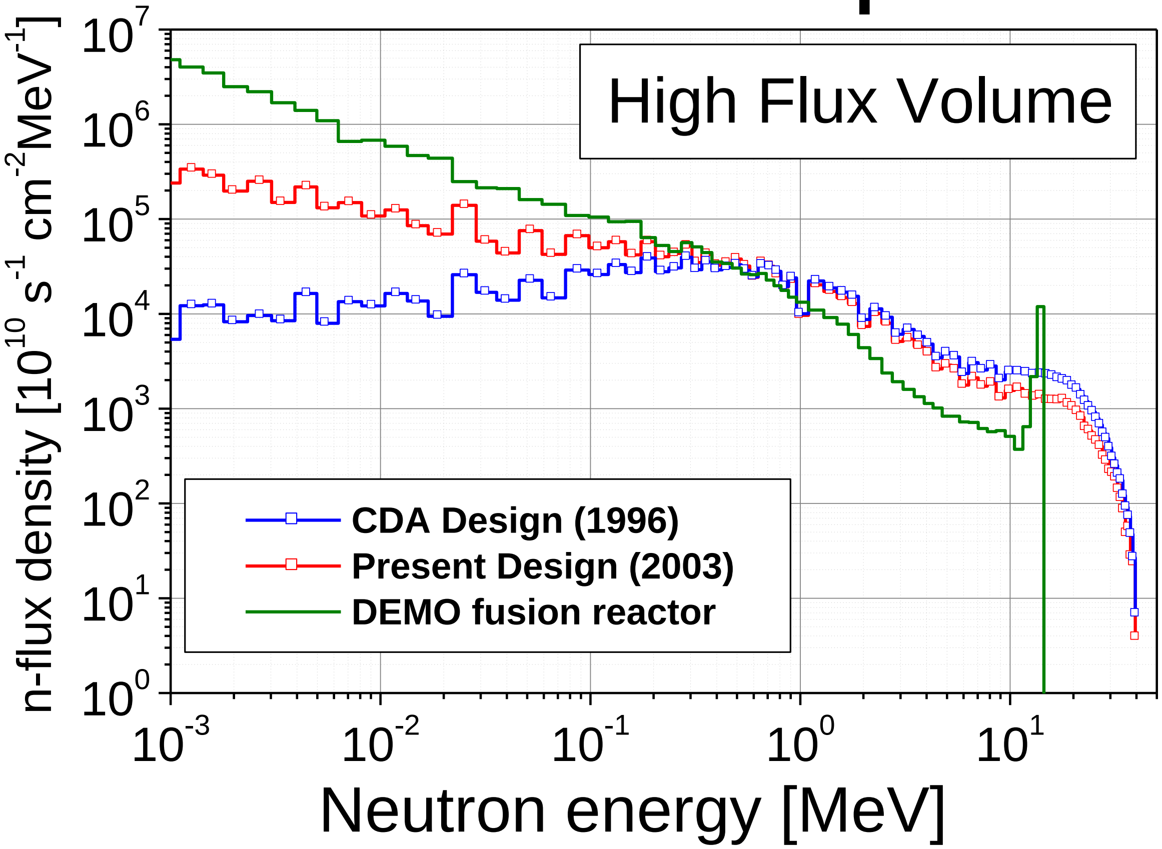
Beam footprint on Li target
20cm wide x 5cm high
(1 GW/m^2)

IFMIF "Artist View"
International Fusion Material Irradiation Facility



IFMIF facility: two, high power CW drivers, each delivering a 125 mA deuteron beam at 40 MeV (5 MW power) hitting a liquid lithium target in order to yield neutrons ($10^{17}s^{-1}$) via nuclear stripping reactions.

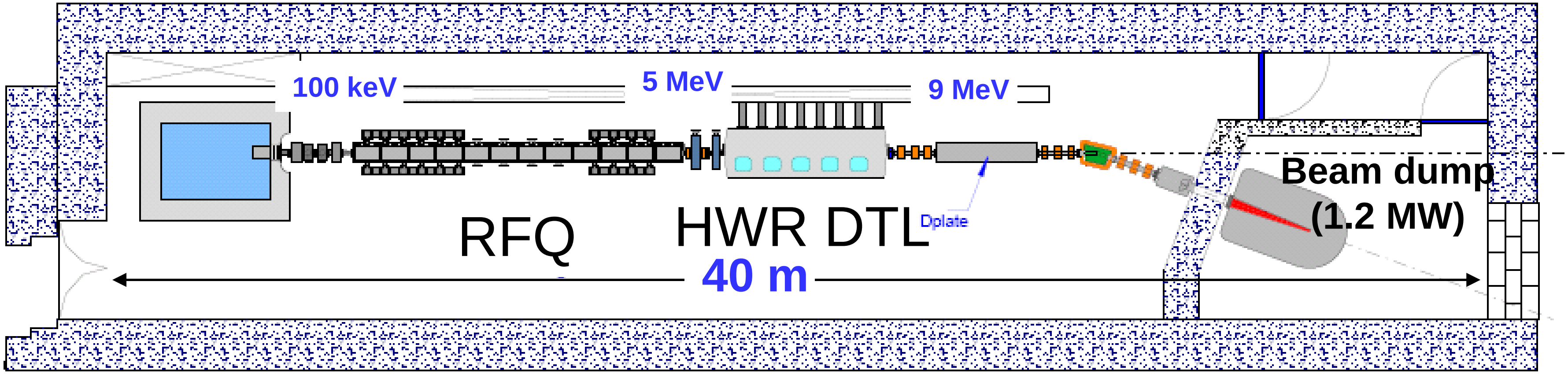
IFMIF neutron spectra



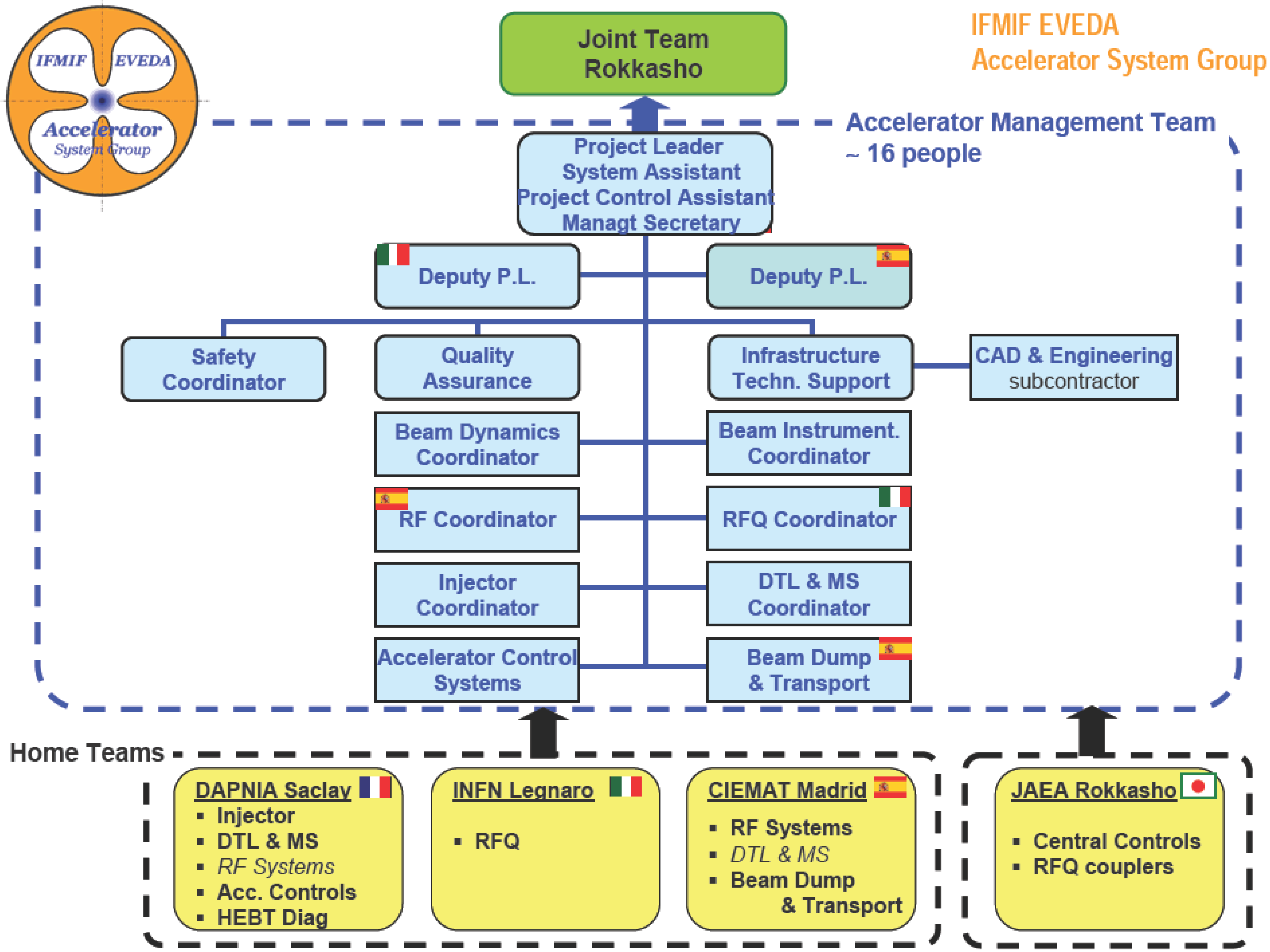
- Neutron moderators and regulators enabled (40 MeV deuterons, about 10^{17}n/s and 10^{10} n/J)
  - To drastically improve the neutron spectrum
  - To increase the irradiated volume

IFMIF EVEDA

- Recently funded within the Broader Approach to Fusion: construction of **a 9 MeV 125 mA cw deuteron accelerator** (to be built in Rokkasho, Japan) based on a high power RFQ followed by a superconducting linac



IFMIF EVEDA
Accelerator System Group

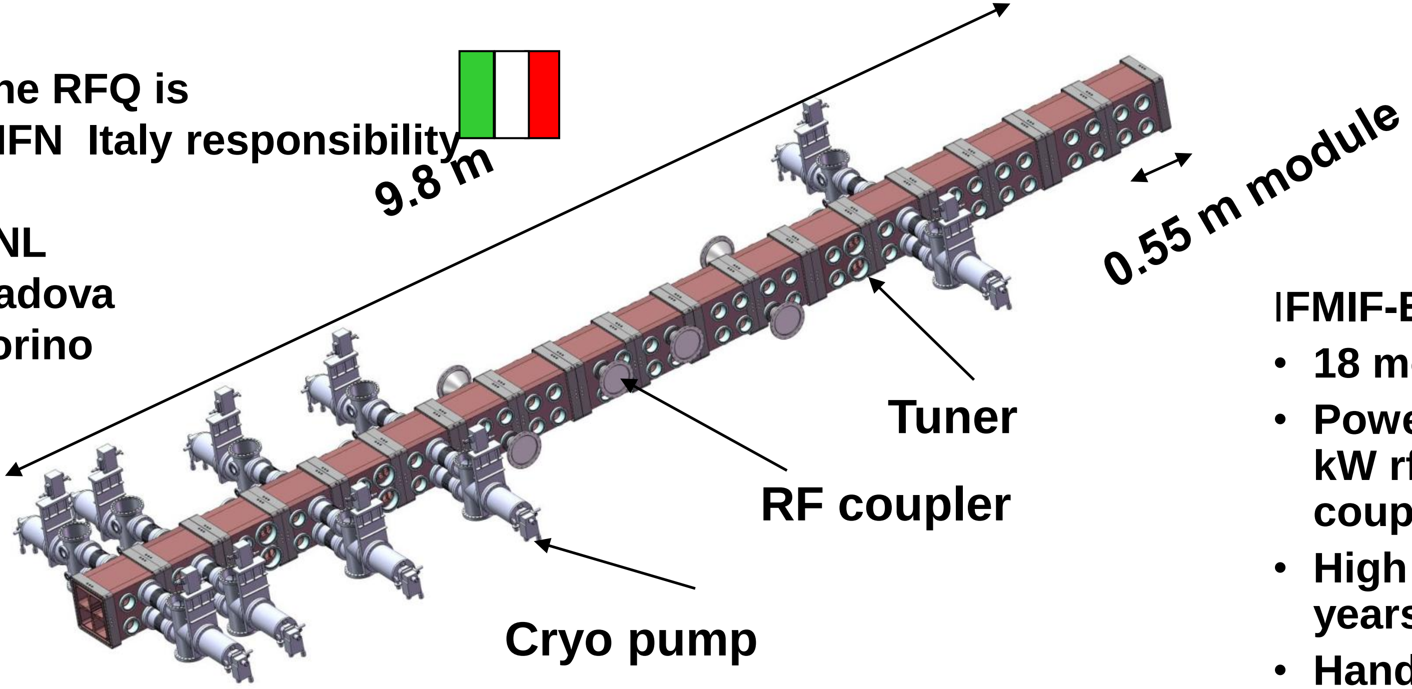


RFQs general parameters

	Name	Lab	ion	energy	vane	beam		RF Cu	Freq.	length		Emax	Power density		operatec
					voltage	current	power	power					ave	max	
				MeV/u	kV	mA	kW	kW	MHz	m	lambda	kilpat	W/cm^2	W/cm^2	
🟦	**IFMIF EVEDA**	**LNL**	**d**	**2.5**	**79-132**	**130**	**650**	**585**	**175**	**9.8**	**5.7**	**1.8**	**3.5**	**30**	**NO**
CW	LEDA	LANL	p	6.7	67-117	100	670	1450	350	8	9.3	1.8	11.4	65	*YES*
	FMIT	LANL	d	2	185	100	193	407	80	4	1.0	1	0.4		*YES*
high p	IPHI	CEA	p	3	87-123	100	300	750	352	6	7.0	1.7	15	120	*NO*
	TRASCO	LNL	p	5	68	30	150	847	352	7.3	8.6	1.8	6.6	90	*NO*

The RFQ is
INFN Italy responsibility

LNL
Padova
Torino



IFMIF-EVEDA RFQ

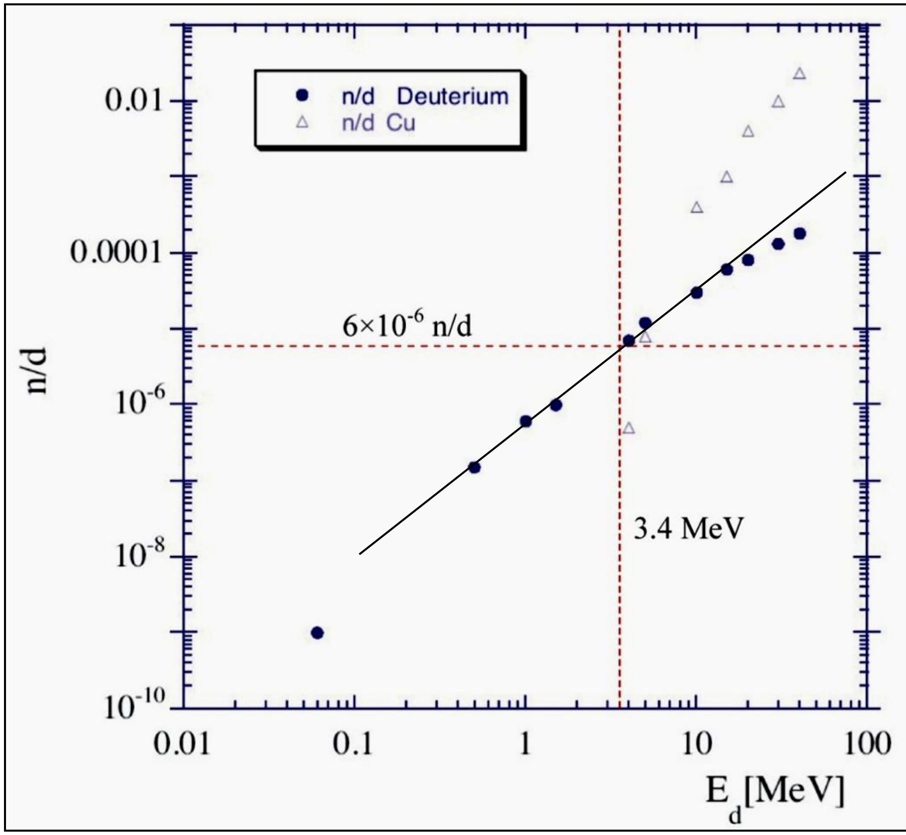
- **18 modules 9.8 m**
- **Powered by eight 220 kW rf chains and 8 couplers**
- **High availability 30 years operation.**
- **Hands on maintenance**
- **First complete installation in Japan**

Beam dynamics essentials

- The beam dynamics design should minimize the activation by beam losses in the RFQ (Hands-on maintenance)
- prepare a high quality beam for a clean transport in the following superconducting linac.
- It is important to consider that the neutron production *n* at low energy (caused mainly by the fusion d-d reaction up to about 5 MeV) scales approximately as:

$$n \propto Nw^2$$

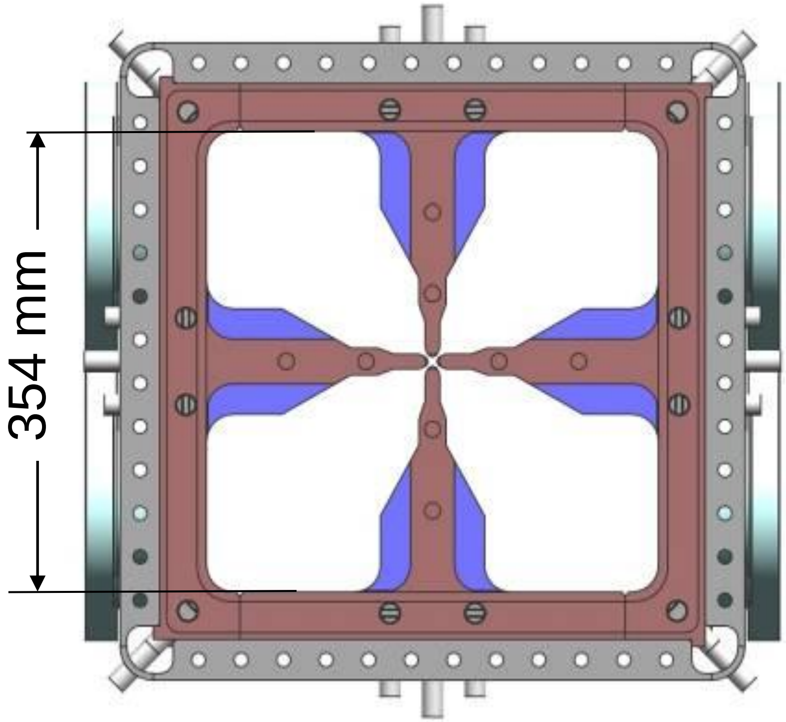
- This means that, one has to concentrate the beam losses at low energy.

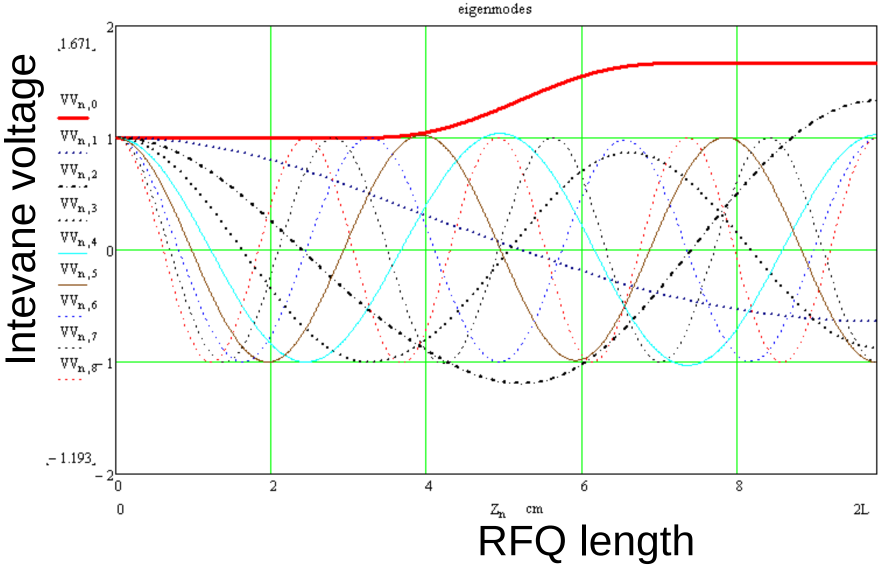


Fit from 0.1 to 5MeV

$$n = 5.15 * 10^{-7} Nw^{2.1}$$

Geometric tolerances





Normal modes in the resonator

Perturbation to the nominal geometry ⇨ Accelerating mode is not pure

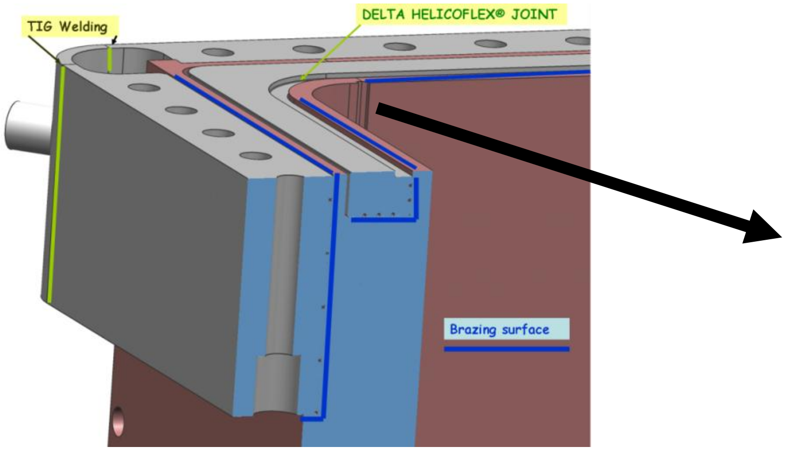
$$\left|E_{210}^{p}\right\rangle=\left|E_{210}\right\rangle+\sum_{n\neq 0}\frac{\left\langle E_{21n}\middle|P\middle|E_{210}\right\rangle}{\omega_{21n}^{2}-\omega_{210}^{2}}\left|E_{21n}\right\rangle+\sum_{n=0}\frac{\left\langle E_{D1n}\middle|P\middle|E_{210}\right\rangle}{\omega_{D1n}^{2}-\omega_{210}^{2}}\left|E_{D1n}\right\rangle+\sum_{n=0}\frac{\left\langle E_{D2n}\middle|P\middle|E_{210}\right\rangle}{\omega_{D2n}^{2}-\omega_{210}^{2}}\left|E_{D2n}\right\rangle$$

Geometrical tolerances: about 5 um **Mechanical construction tolerances: about 20 um**

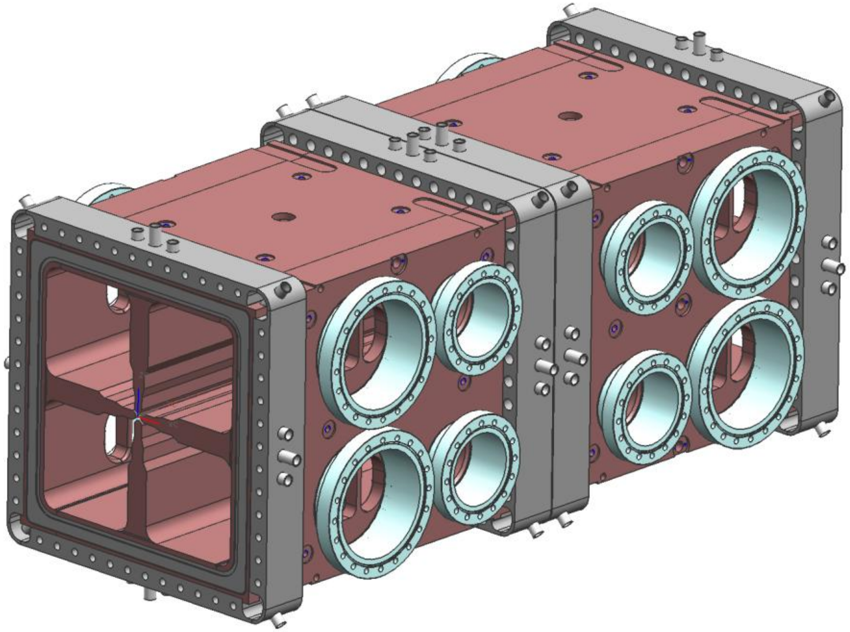
The aluminum real-scale RFQ model

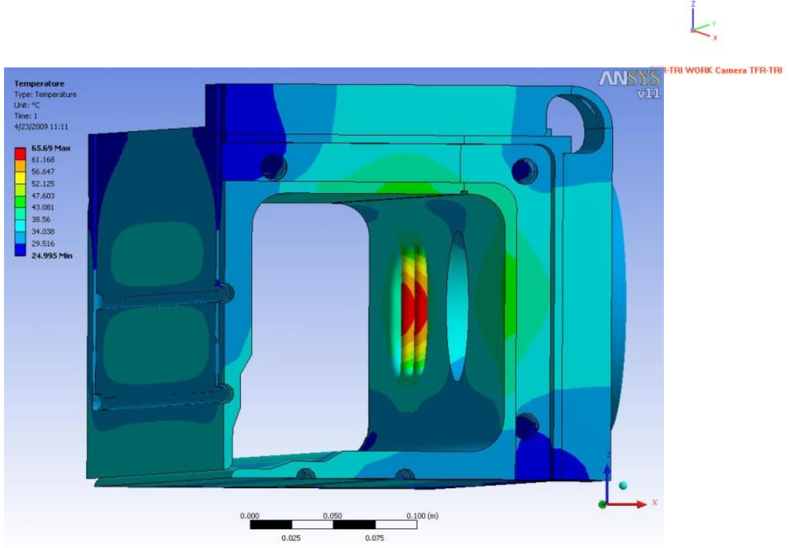
The RFQ model set up at LNL for measurements and equipped with tuners, feeds, dummy plugs and end cells (23/06/09)

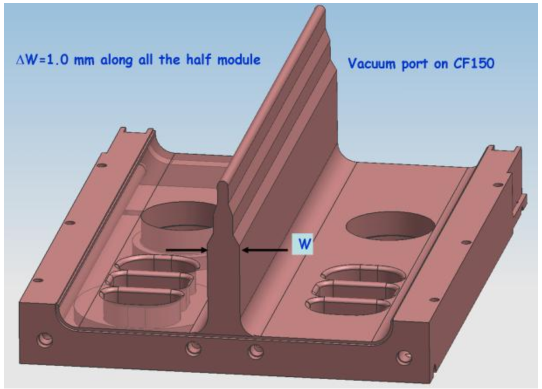
Mechanical design (2)



Head flange

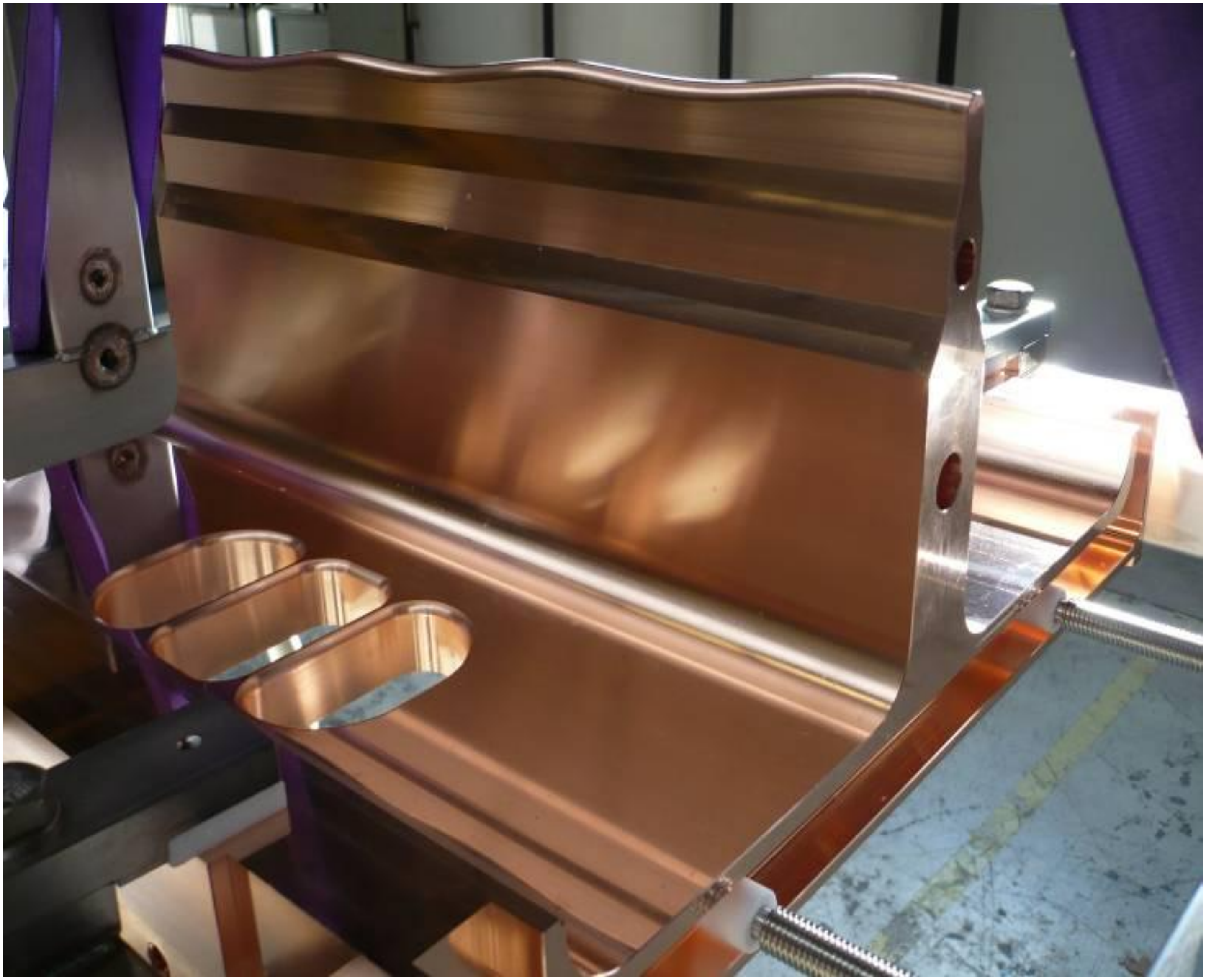






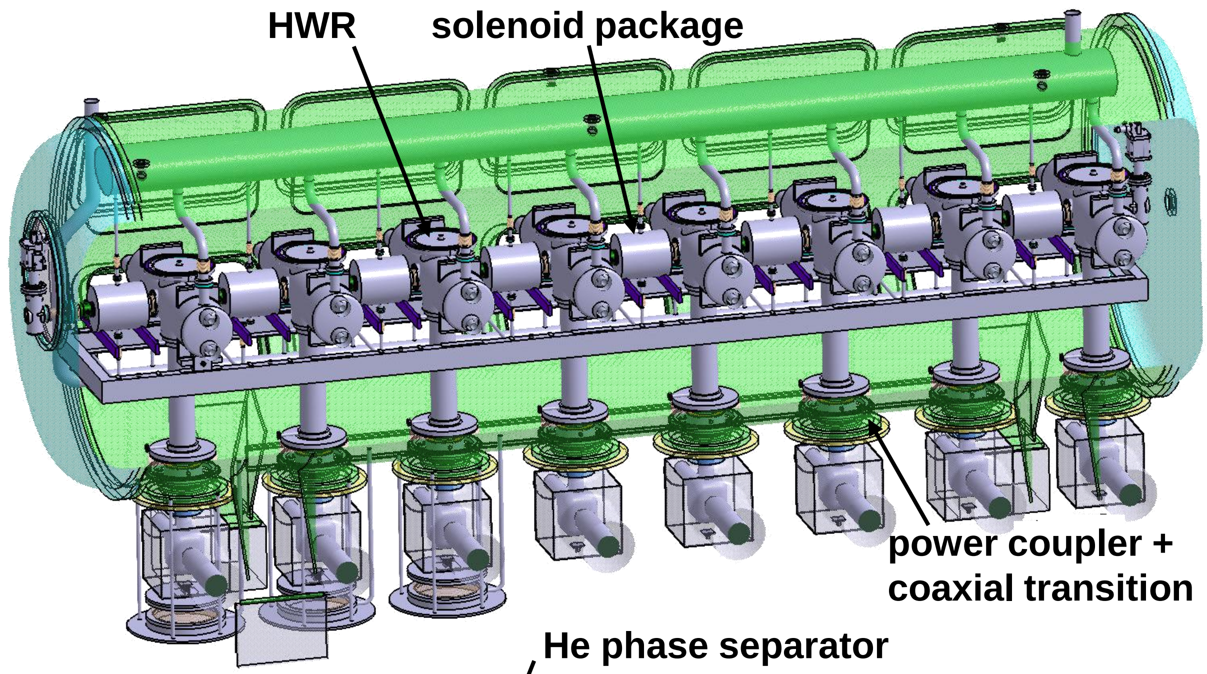
Vacuum grids machined from bulk

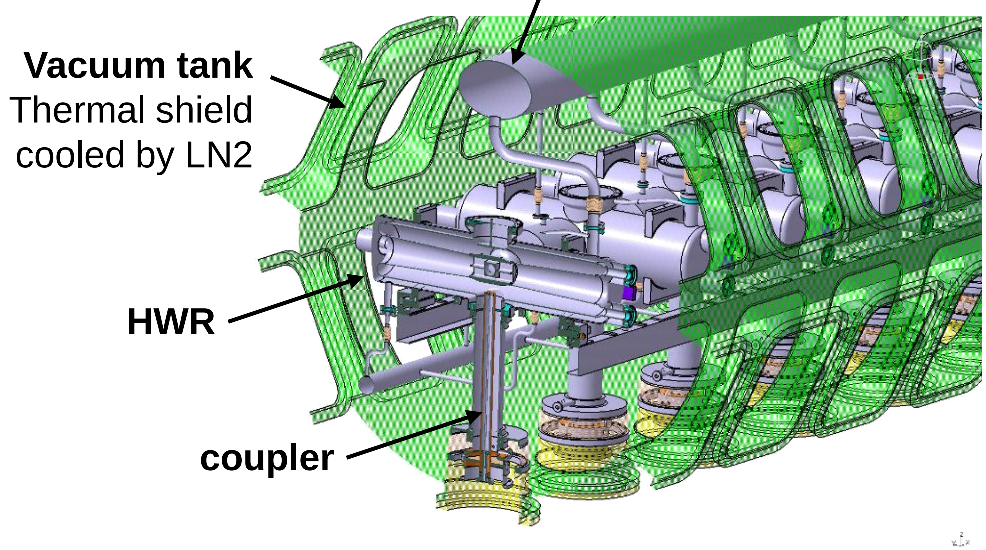
E electrode machined at INFN Torino after 800 deg annealing in LNL vacuum oven



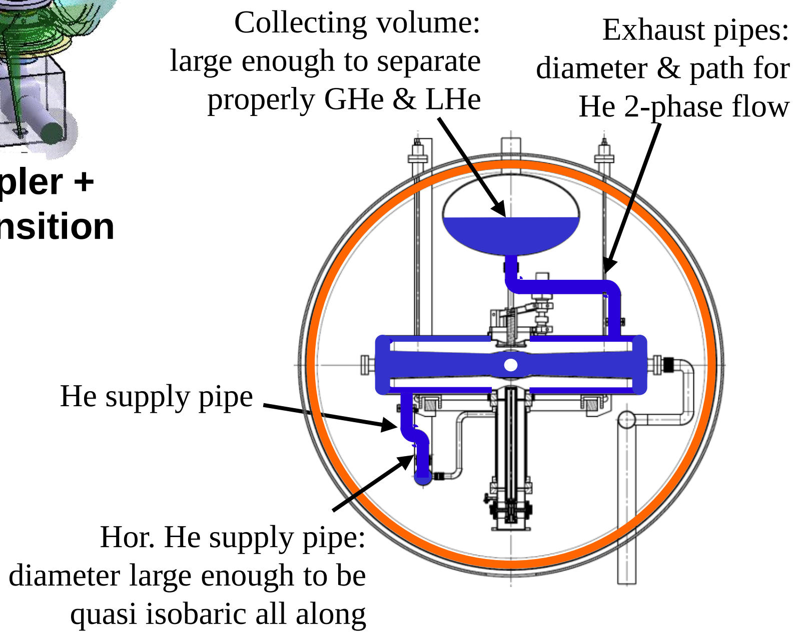
CEA France responsibility

Cryomodule Conceptual design





- ➢ **Conceptual design for:**
  cold mass support, alignment system, cryogenic pipes, vacuum pipes, interfaces, connections with all services


Collecting volume:
large enough to separate
properly GHe & LHe

Exhaust pipes:
diameter & path for
He 2-phase flow

He supply pipe

Hor. He supply pipe:
diameter large enough to be
quasi isobaric all along

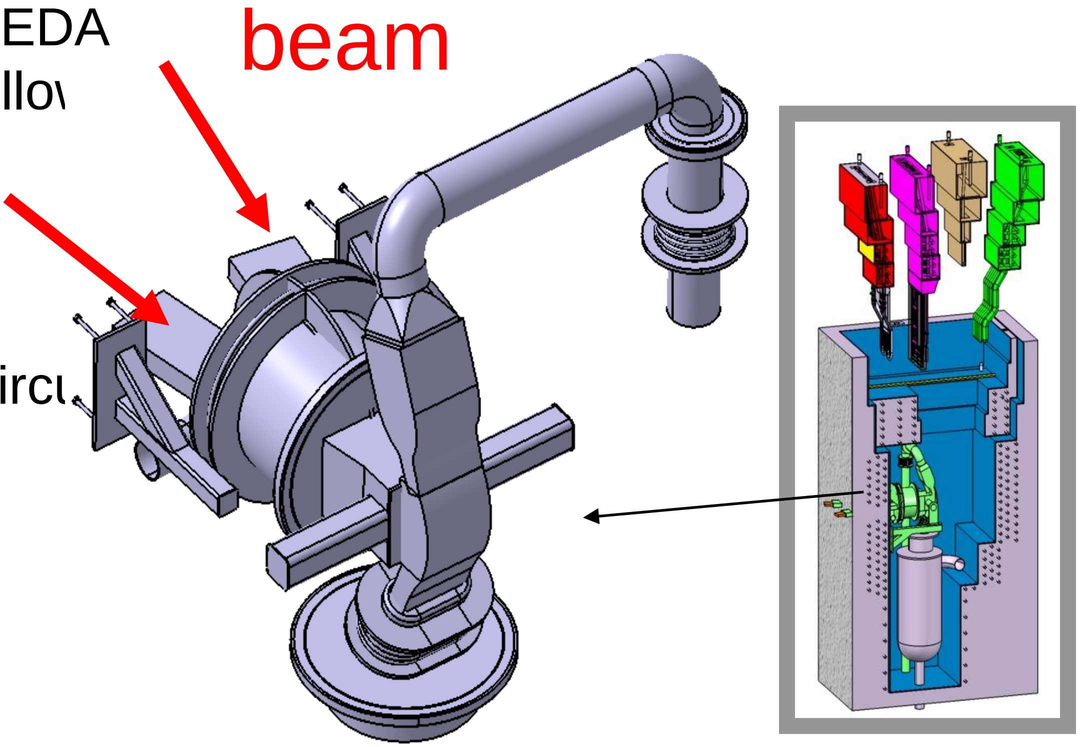
- ➢ **Conceptual design for:**
  He cooling (forced flow mode)

Alban Mosnier, PAC 09

IFMIF EVEDA
(IFMIF Engineering Validation Activities)

- Within the BA (Broader Approach to fusion agreement) the IFMIF EVEDA activities have been launched follo three programs

- Prototype Accelerator
- Prototype of the Lithium target circu
- Experimental facility definition



Li target assembly

Experimental assembly

The Li target concept

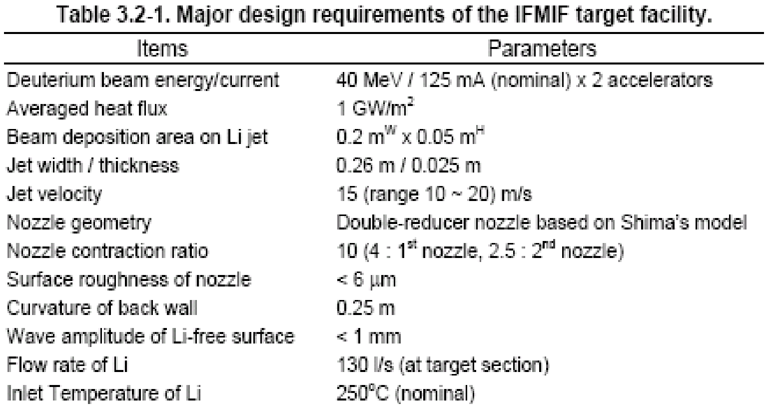
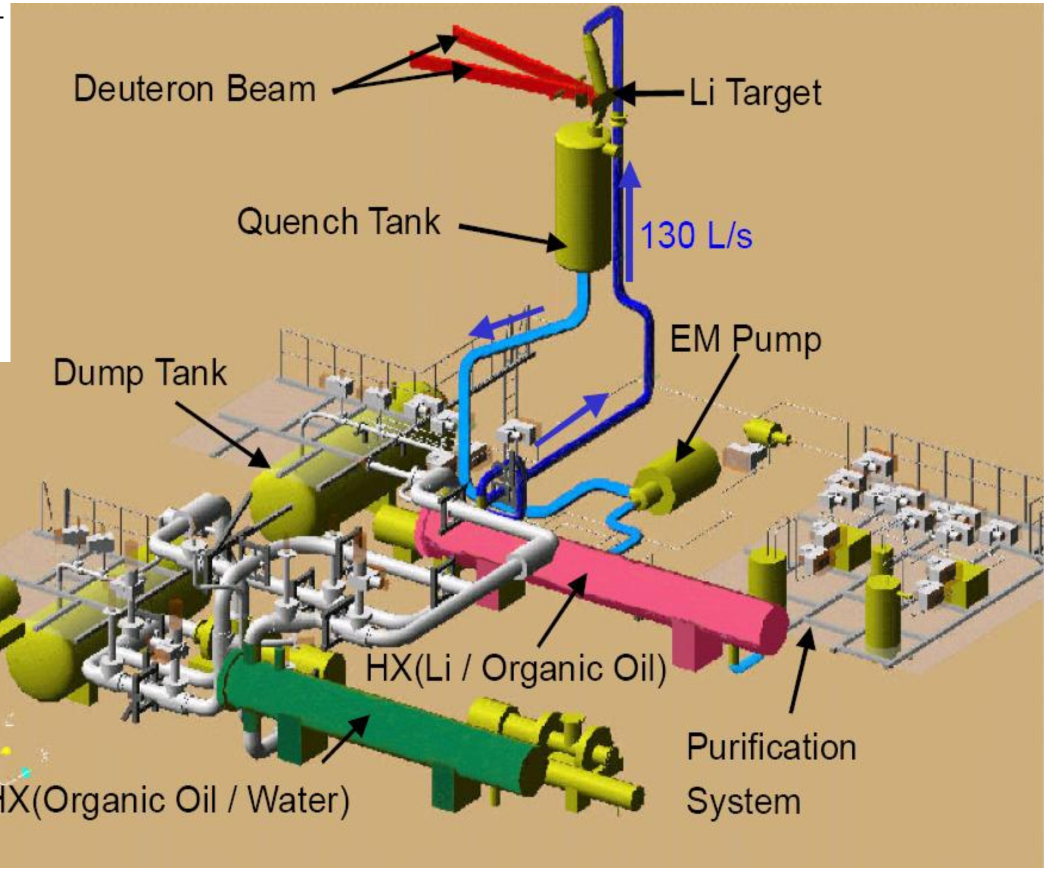


Table 3.2-1. Major design requirements of the IFMIF target facility.

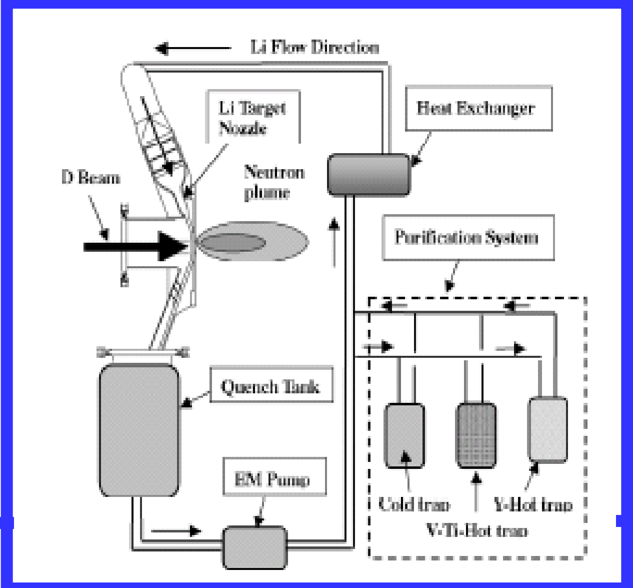
Items	Parameters
Deuterium beam energy/current	40 MeV / 125 mA (nominal) x 2 accelerators
Averaged heat flux	1 GW/m^2
Beam deposition area on Li jet	0.2 m^W x 0.05 m^H
Jet width / thickness	0.26 m / 0.025 m
Jet velocity	15 (range 10 ~ 20) m/s
Nozzle geometry	Double-reducer nozzle based on Shima's model
Nozzle contraction ratio	10 (4 : 1st nozzle, 2.5 : 2nd nozzle)
Surface roughness of nozzle	< 6 µm
Curvature of back wall	0.25 m
Wave amplitude of Li-free surface	< 1 mm
Flow rate of Li	130 l/s (at target section)
Inlet Temperature of Li	250°C (nominal)



10 MW

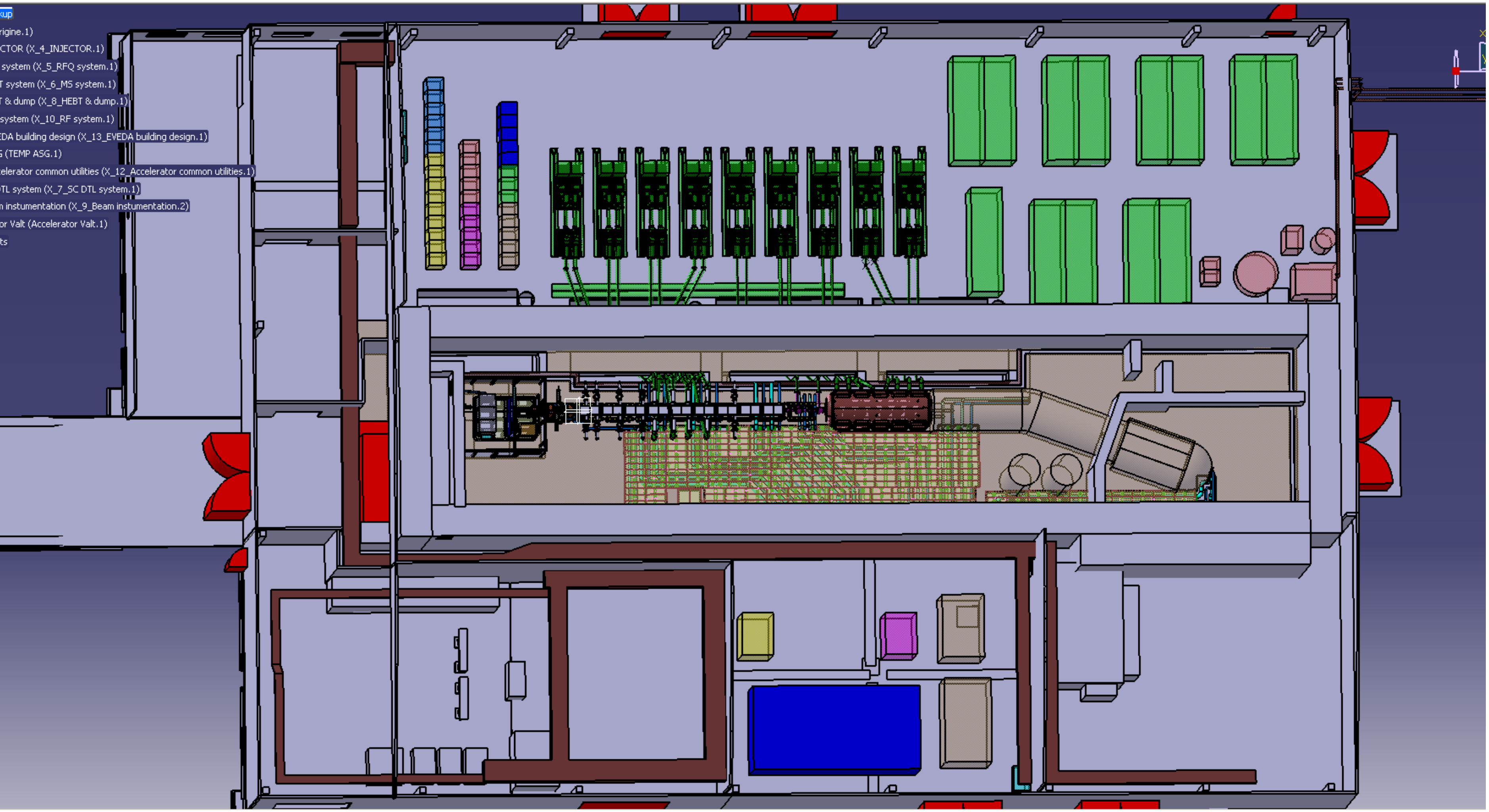
Li solidification 200 deg C

Li boiling about 1000 deg C



IFMIF/EVEDA building (Rokkasho)

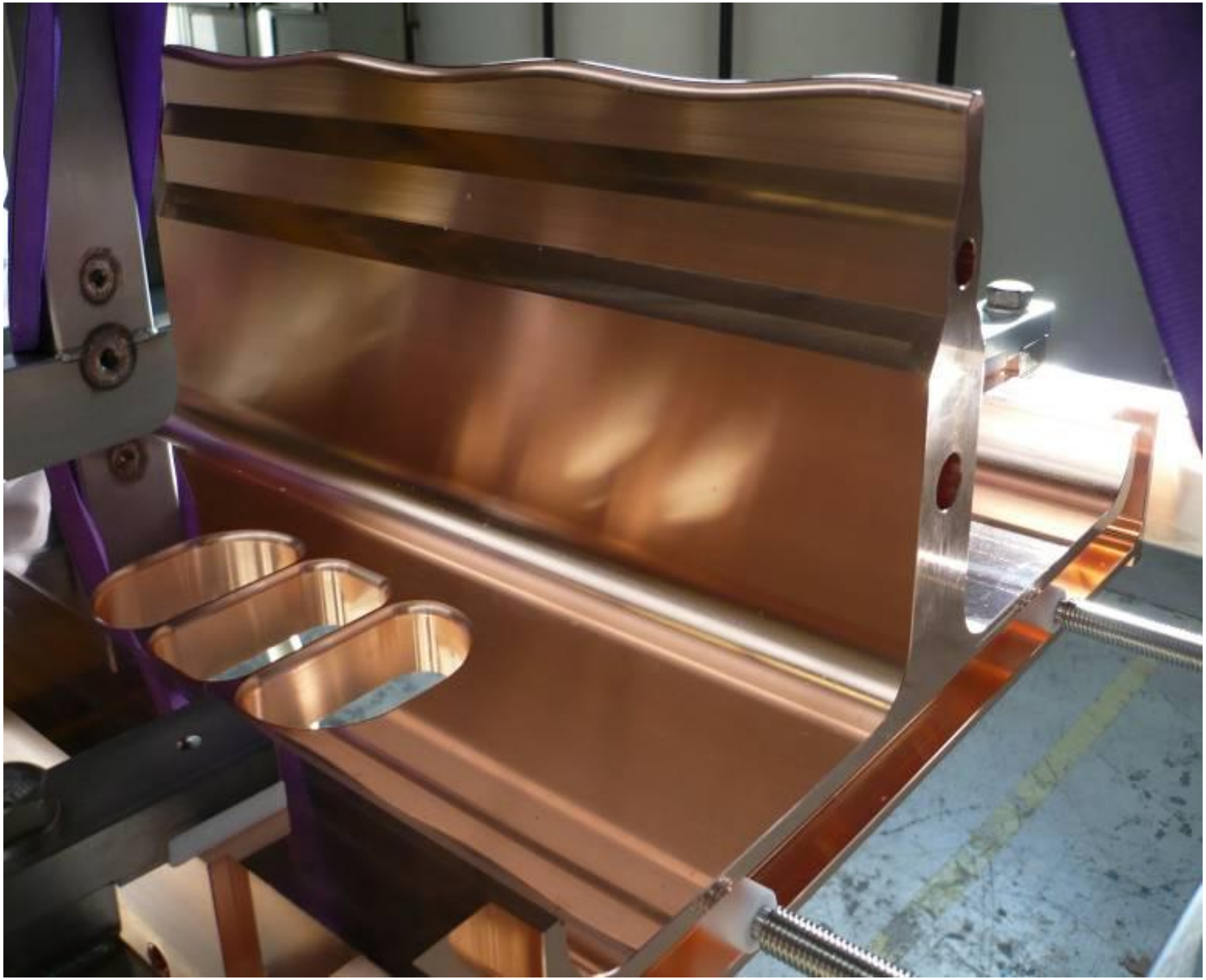
RFQ in the building



The aluminum real-scale RFQ model

The RFQ model set up at LNL for measurements and equipped with tuners, feeds, dummy plugs and end cells (23/06/09)

E electrode machined at INFN Torino after 800 deg annealing in LNL vacuum oven



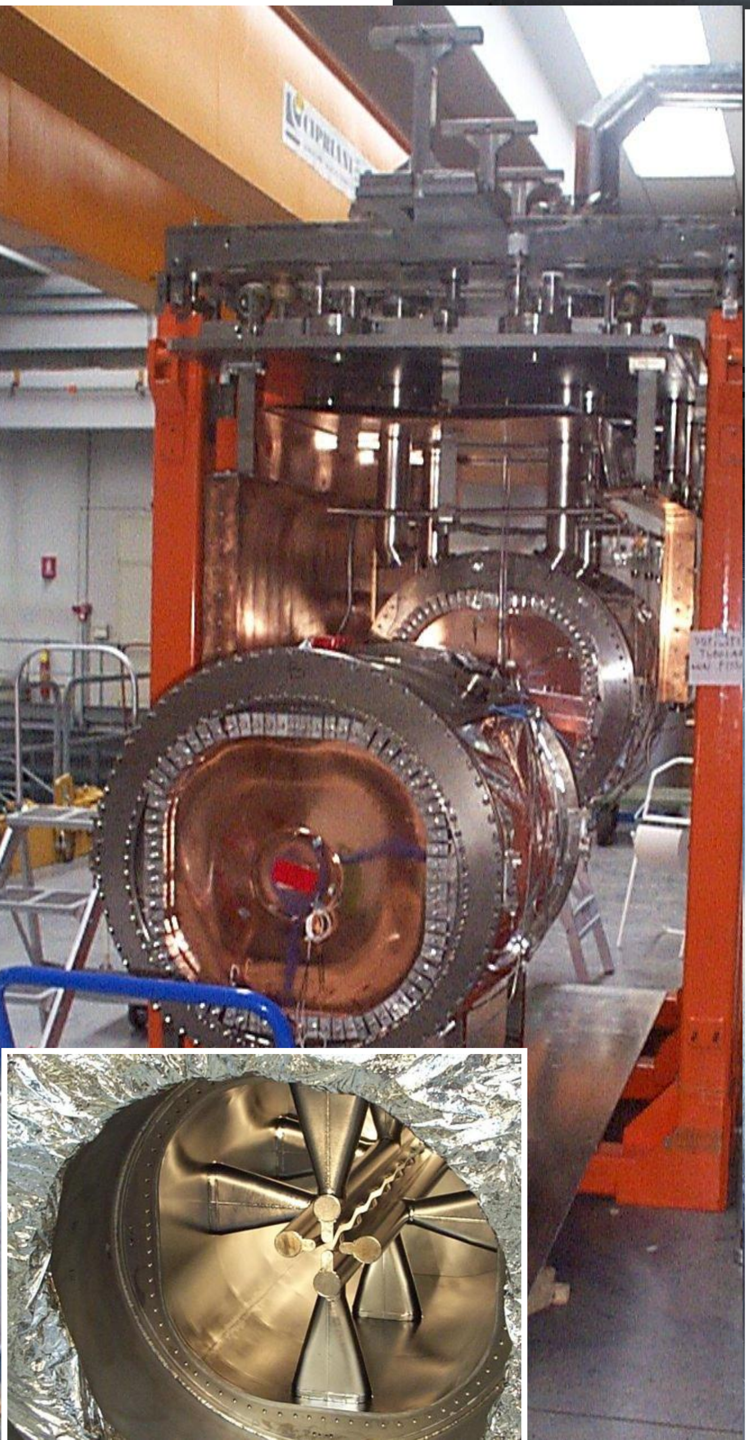
High power RFQ

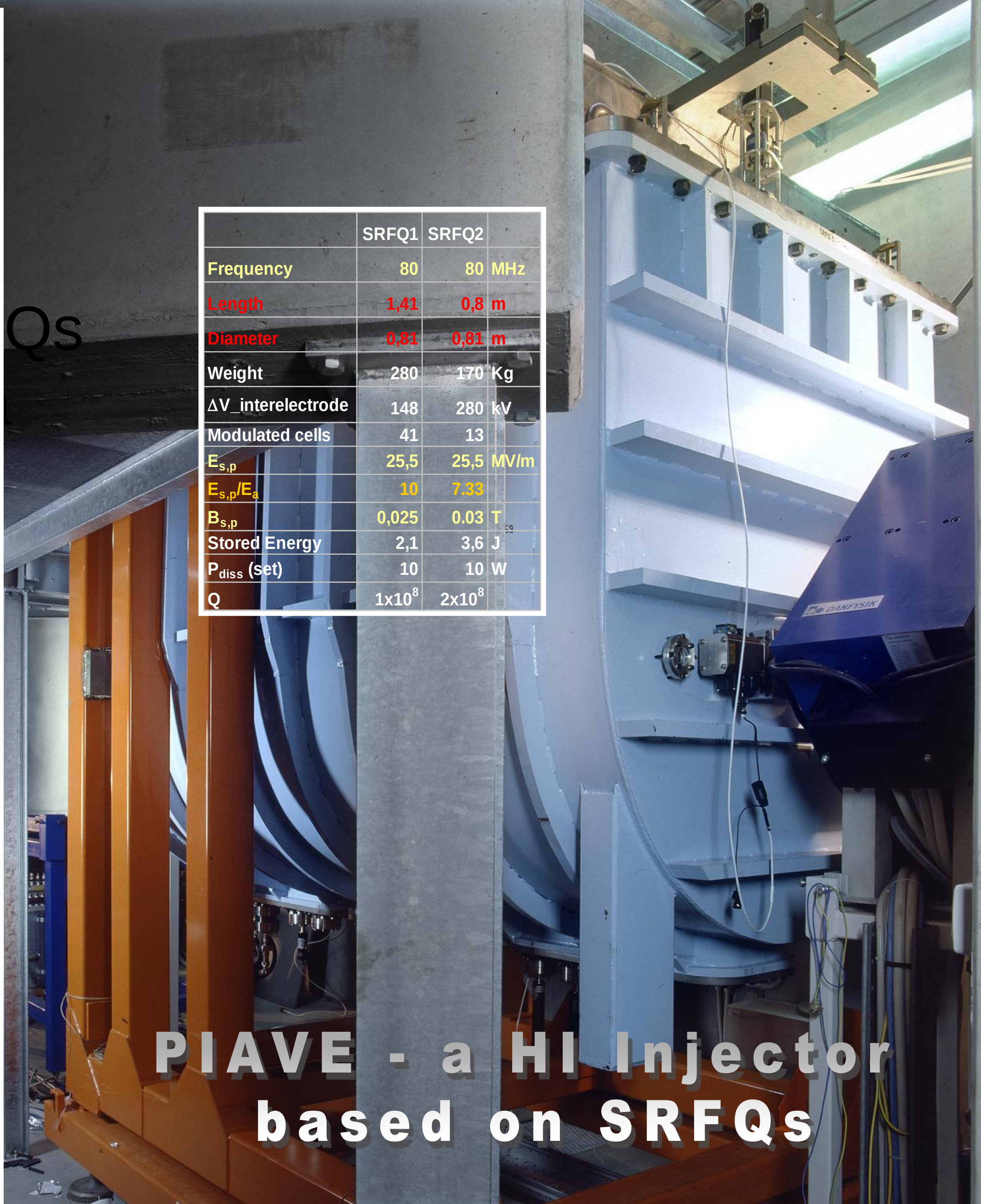
- High power linear accelerators constitutes a very lively field, with many applications to:
  - Energy
  - Fundamental physics
  - Material science
- New technologies have been developed in the last 10 years and are now mature
  - Superconducting structures
  - High power RFQs

Applications of high power beams

- Main applications:
  - Nuclear fusion: **Fusion reactors Material Irradiation** tests under large neutron fluxes
  - Nuclear fission: **nuclear waste transmutation**, i.e. processing of the nuclear reactor fuels to eliminate (ideally) long lived radiactive components, material science Fundamental Physics
- Contribution of **Particle Accelerators** to the development of clean nuclear power
- Contribution from **High Energy Physics and Nuclear Physics** community

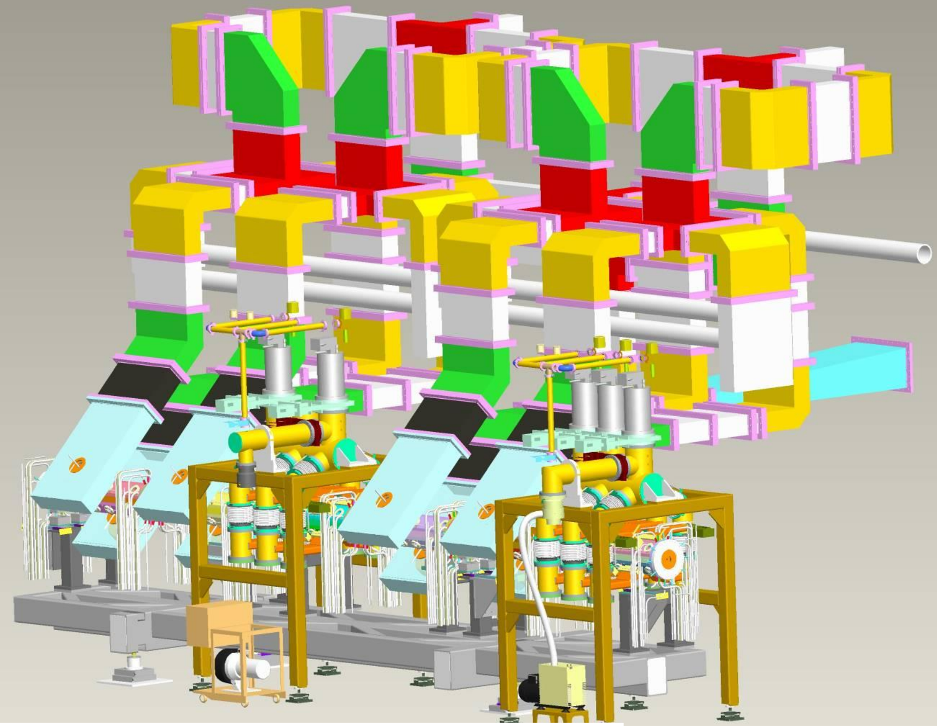
PIAVE - a HI Injector based on SRFQs

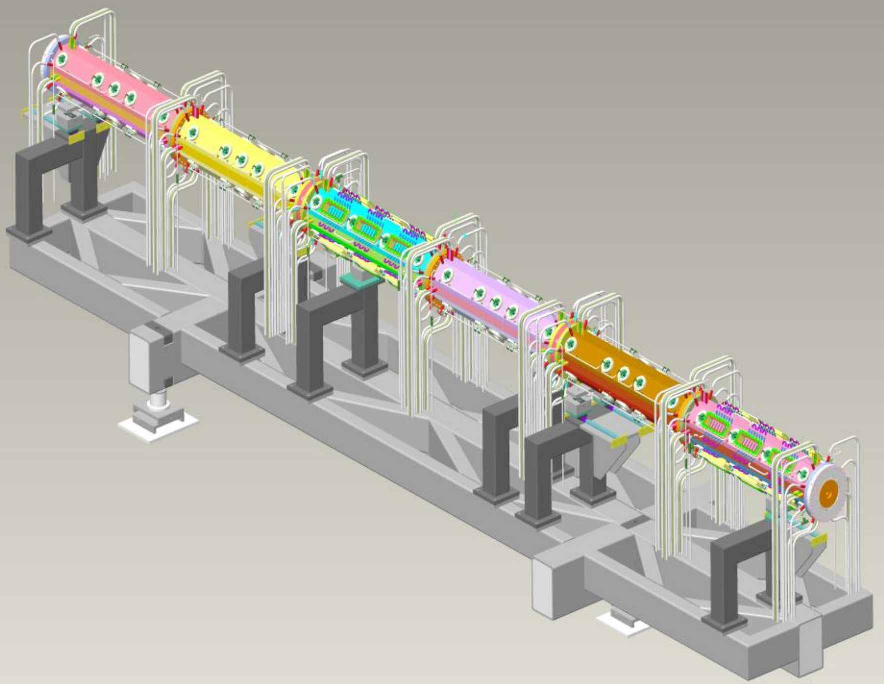




Qs

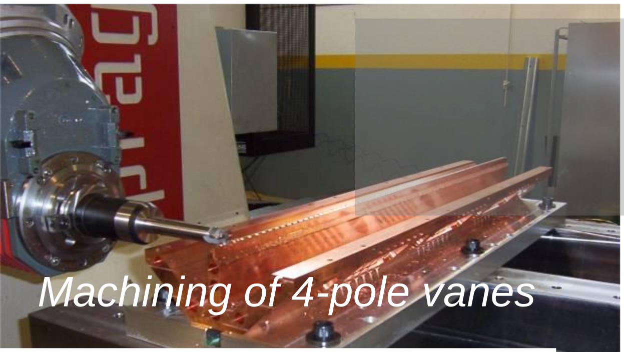
	SRFQ1	SRFQ2	
Frequency	80	80	MHz
Length	1,41	0,8	m
Diameter	0,81	0,81	m
Weight	280	170	Kg
ΔV_interelectrode	148	280	kV
Modulated cells	41	13	
$E_{s,p}$	25,5	25,5	MV/m
$E_{s,p}/E_a$	10	7.33	
$B_{s,p}$	0,025	0.03	T
Stored Energy	2,1	3,6	J
P_{diss} (set)	10	10	W
Q	$1x10^8$	$2x10^8$	



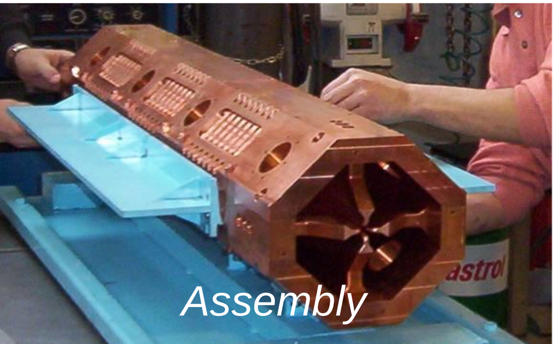


The TRASCO-SPES RFQ

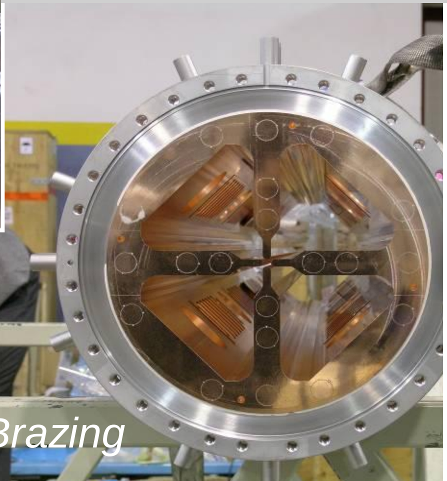
(TRAsmutazione di SCOrie)



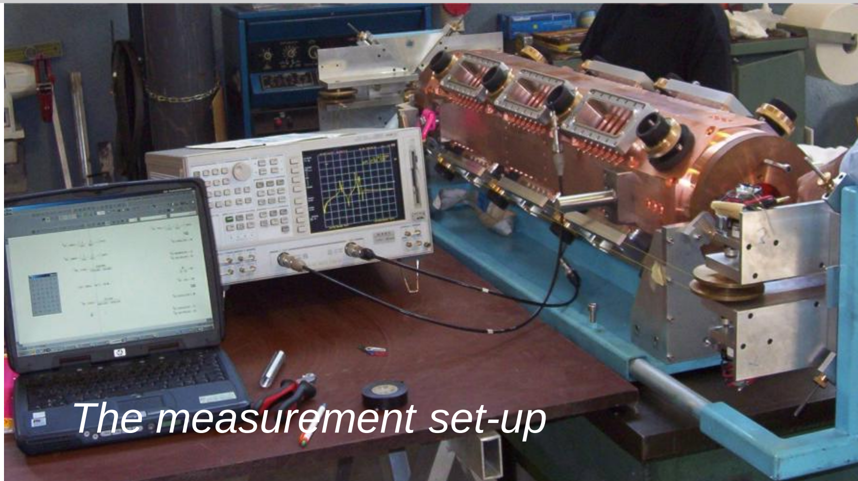
Machining of 4-pole vanes



Assembly

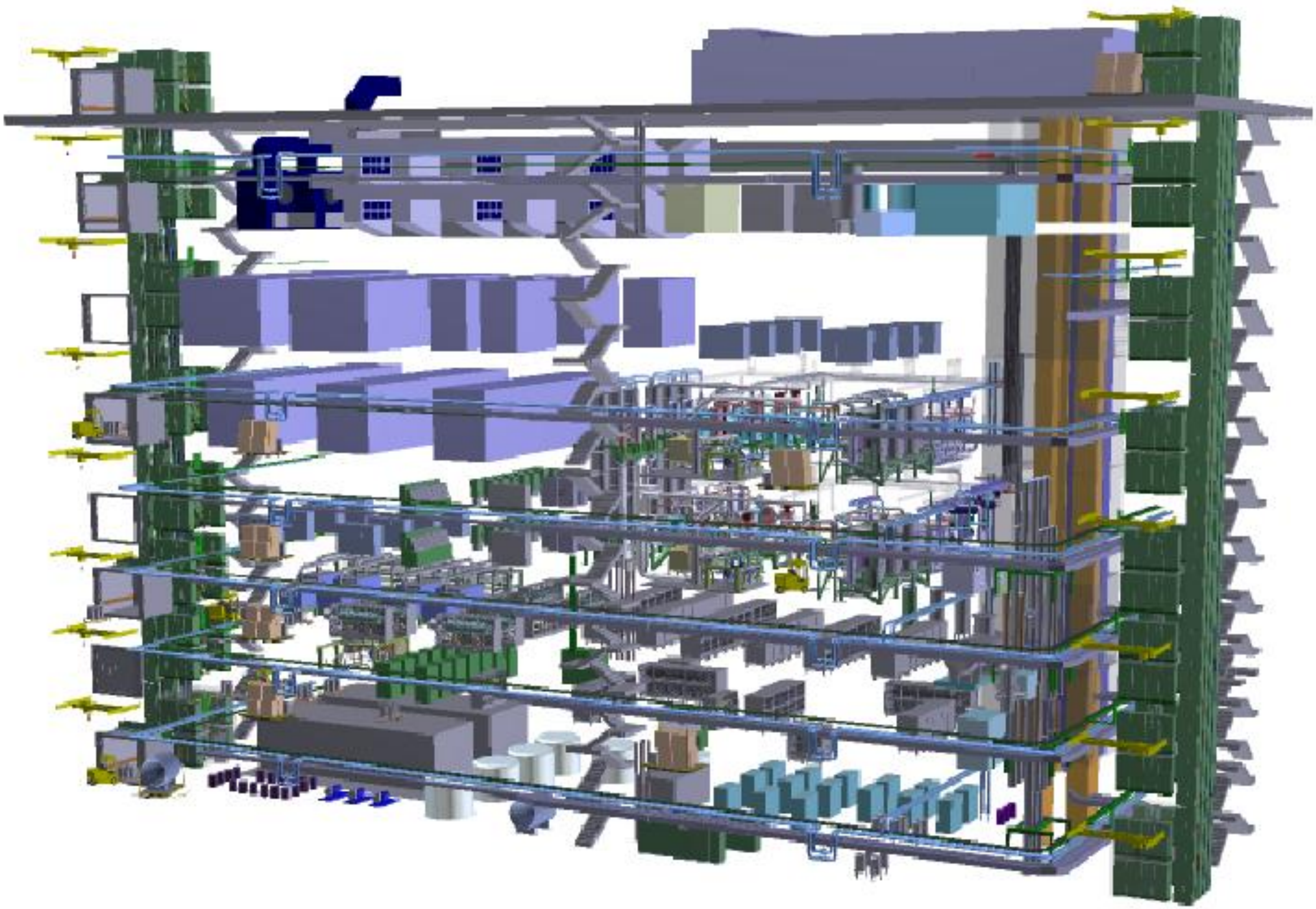


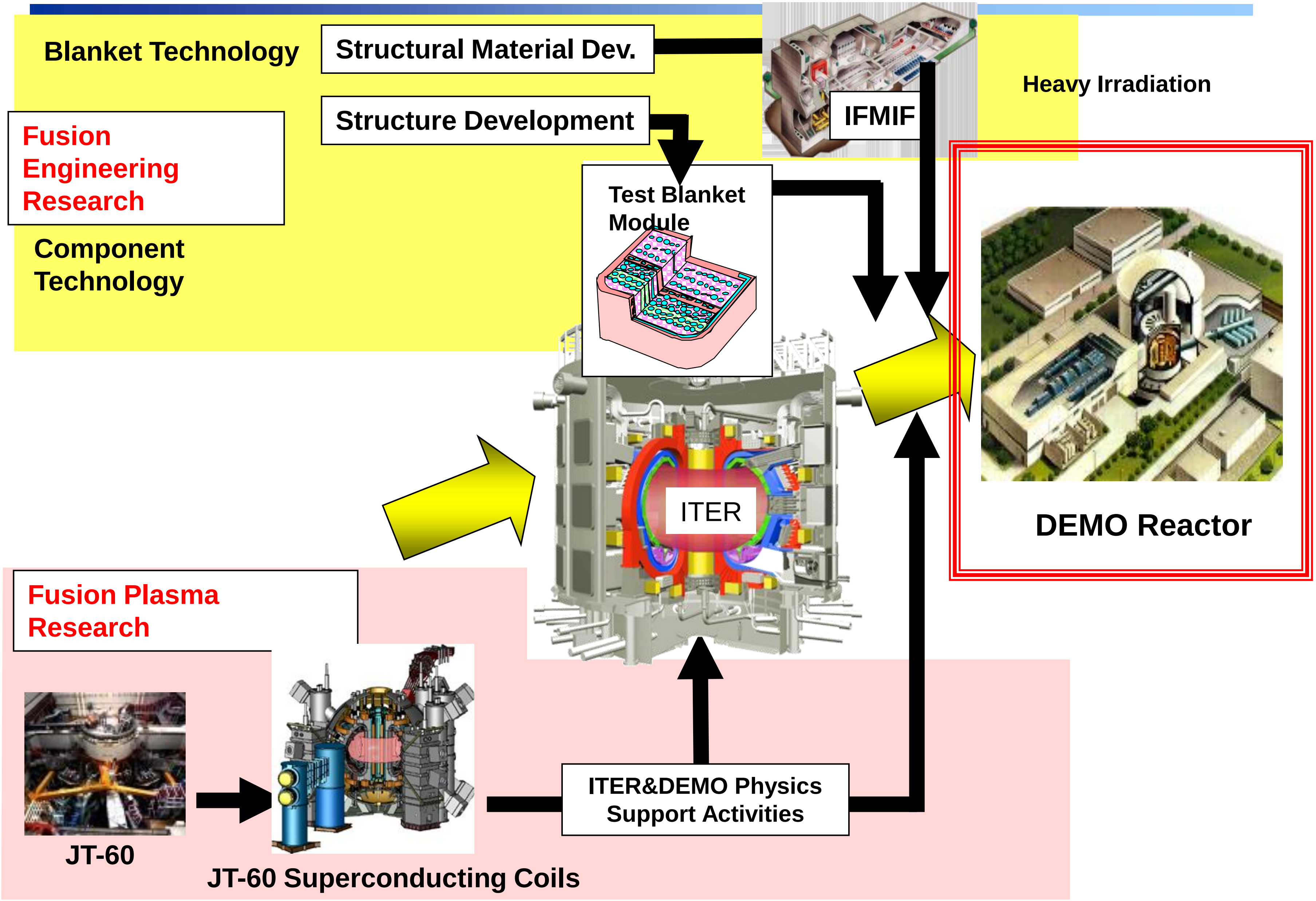
Brazing



The measurement set-up

Tritium Plant Building Systems Layout





Blanket Technology

Structural Material Dev.

Structure Development

IFMIF

Heavy Irradiation

Fusion Engineering Research

Component Technology

Test Blanket Module

ITER

DEMO Reactor

Fusion Plasma Research

JT-60

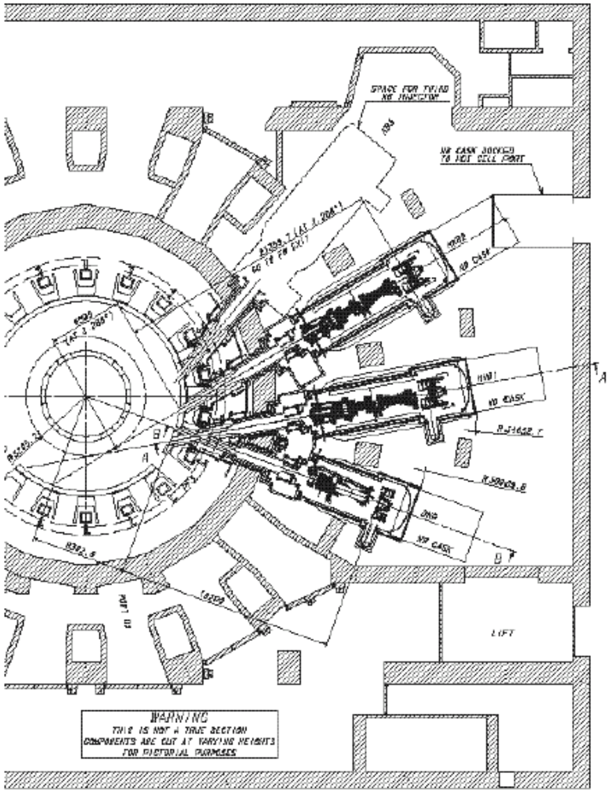
JT-60 Superconducting Coils

ITER&DEMO Physics Support Activities

ITER : auxiliary heating systems

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

NBI Layout



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

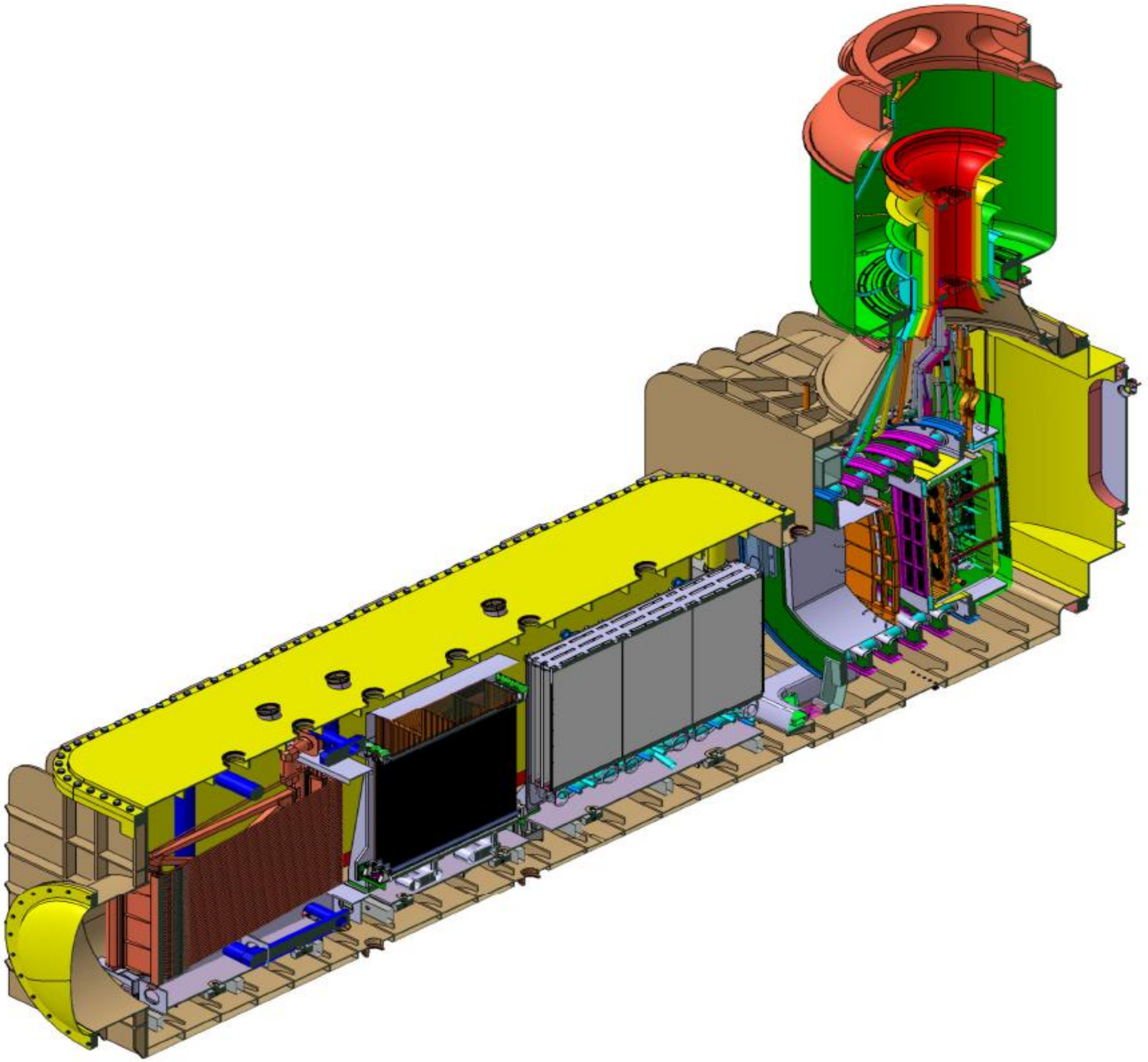


Fig. 9 Progetto del vessel per NBI di ITER