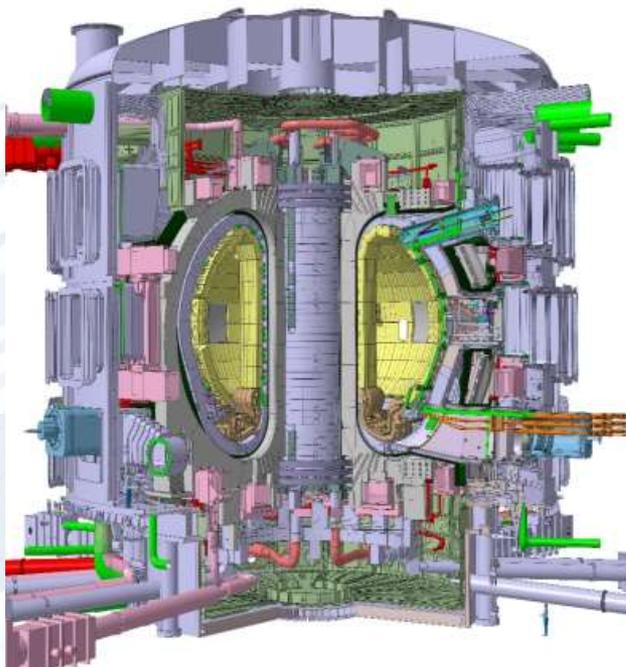


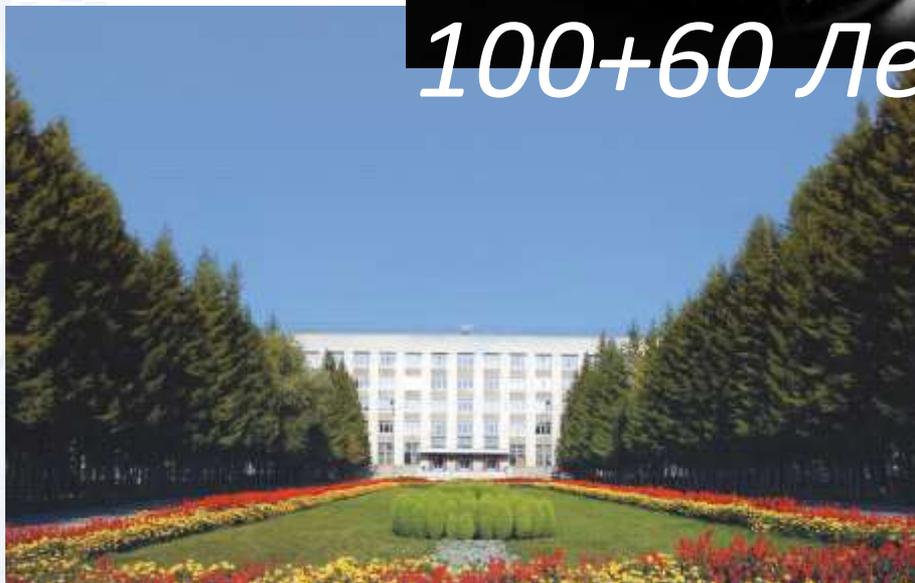


ИТЭР и национальные проекты партнёров – научно-технологическая база термоядерной энергетики



Красильников А.В.
Учреждение ГК Росатом
«Проектный центр ИТЭР»

a.krasilnikov@iterrf.ru



Россия. Токамаки. Основные вехи.

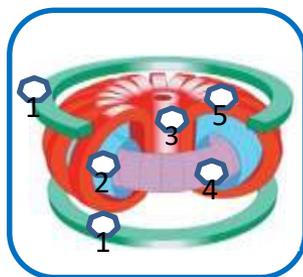
1951 - 1960

1971 - 1990

ТОКАМАК (ТОроидальная КАмера МАгнитная Катушка)



И.Е.Тамм А.Д.Сахаров



1972 году Л.А.Арцимовичем и В.Д.Шафрановым предложена концепция вытянутого сечения с полоидальным дивертором.

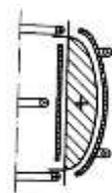


Рис. 1

T-9 Вытянутое поперечное сечение
↓
T-12
TVD Полоидальный дивертор



Л.А.Арцимович В.Д.Шафранов

Термояд в России.
Токамаки.
История

ОТР (РФ, 1988)

1961 – 1995

T-15



INTOR (1980)
↓
ITER



Токамак ТМП - 1953 г.



T-3a

T-7
NbTi

T-10



КТМ



Гибридный термоядерный реактор (1978)

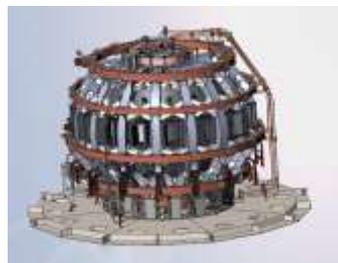


ТСП



Направление Токамаков

T-15 Модернизированный
с теплыми обмотками и
полоидальным дивертором



Глобус-М (0.4 Т, 0.2 МА, 0.15 с)
Глобус-М2 (1 Т, 0.5 МА, 0.4 с)
(полоидальный дивертор,
низкое аспектное отношение)



ИТЭР – основная линия международной научной кооперации

Предложение Е.П.Велихова
сначала М.С.Горбачеву, а
затем Р.Рейгану о создании
международного проекта
термоядерного реактора.



1985

Термояд в России
сегодня

Многопробочная
магнитная ловушка,
ГДЛ



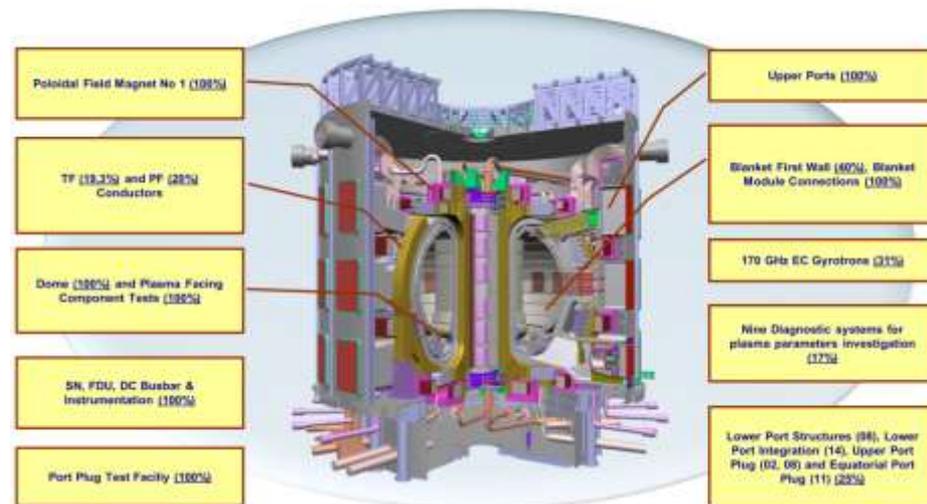
Z-пинчи – Ангара-5



Лазерный УТС



Участие России в ИТЭР

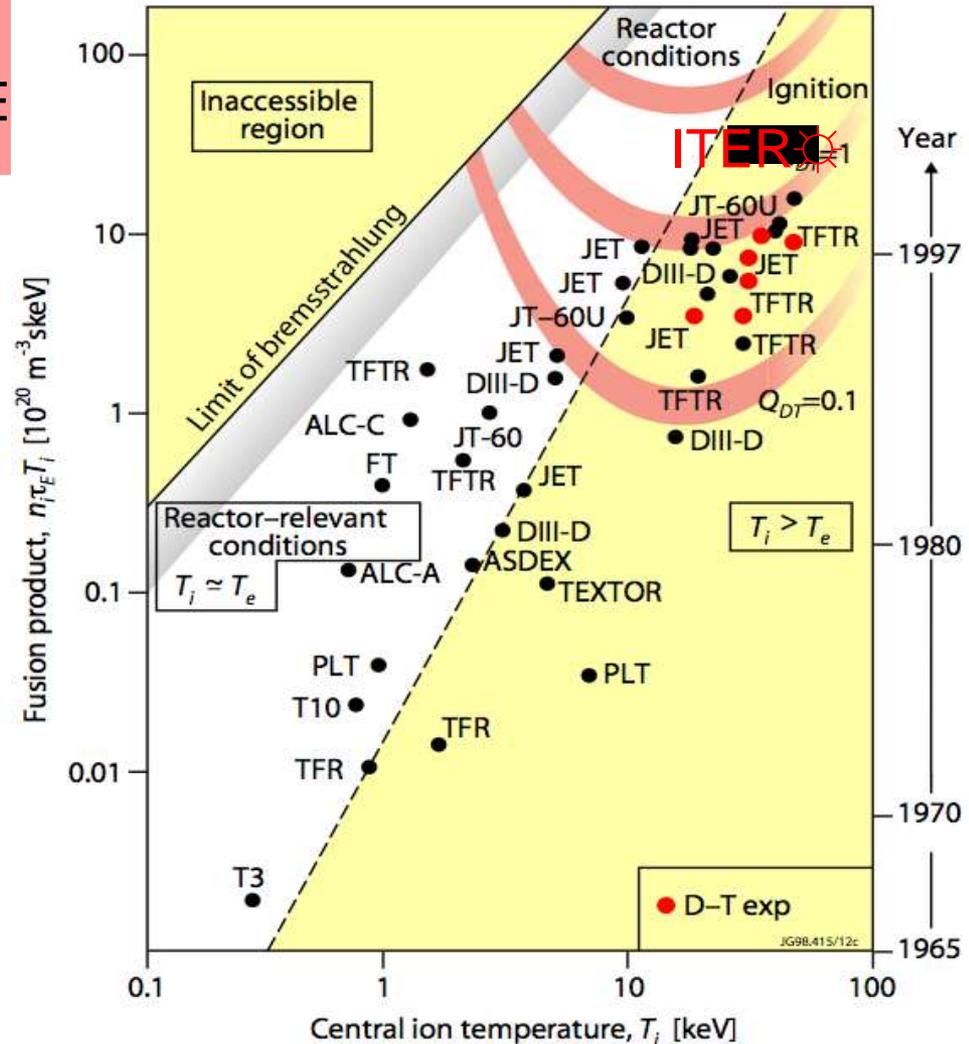


Fusion performance

$$Q = \frac{\text{Fusion Power}}{\text{Input Power}} \sim n_i T_i \tau_E$$

Existing experiments have achieved $nT\tau \sim 1 \times 10^{21} \text{ m}^{-3} \text{ s keV}$ and $Q_{DT} \sim 1$

JET and TFTR have produced DT fusion powers $> 10 \text{ MW}$ for $\sim 1 \text{ s}$
 ITER is designed to a scale which should yield $Q_{DT} \geq 10$ at a fusion power of $400 - 500 \text{ MW}$ for $300 - 500 \text{ s} \rightarrow \text{Baseline scenario}$

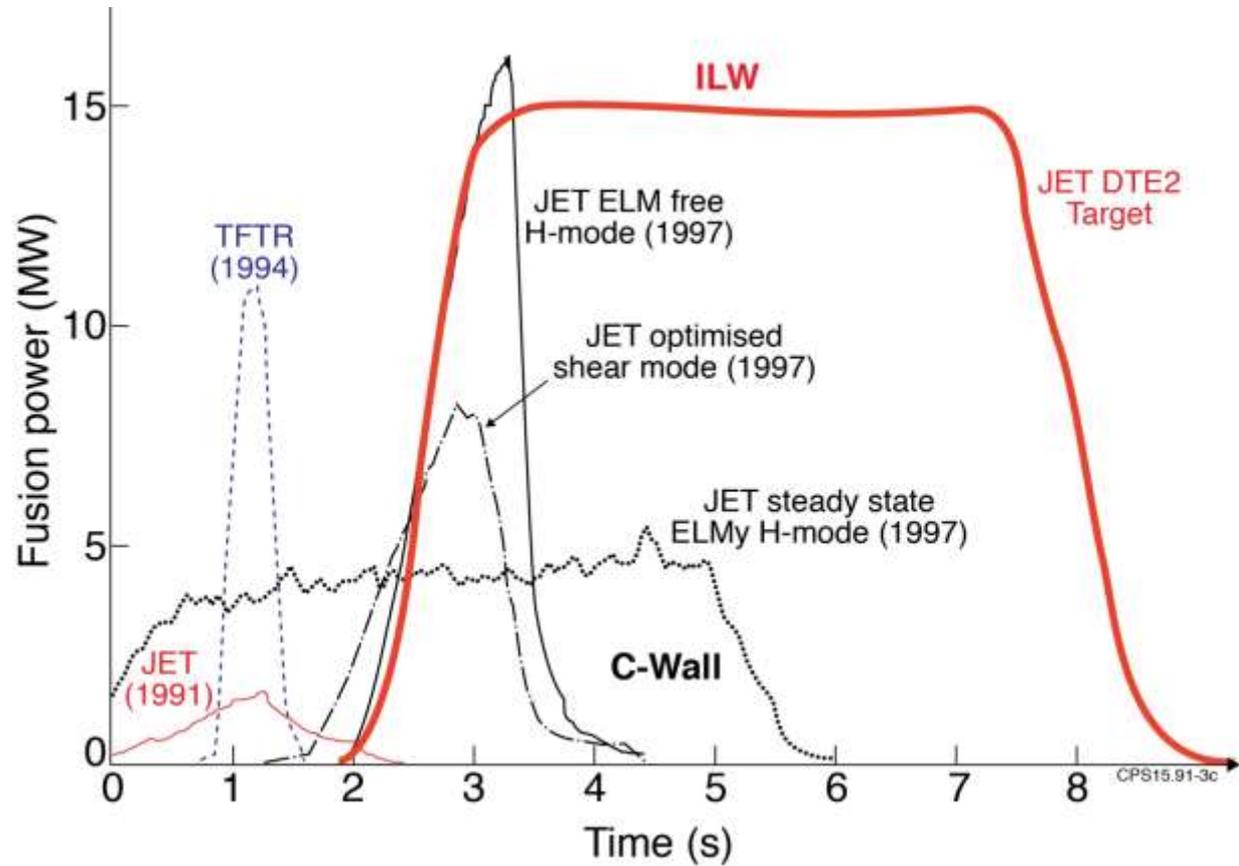
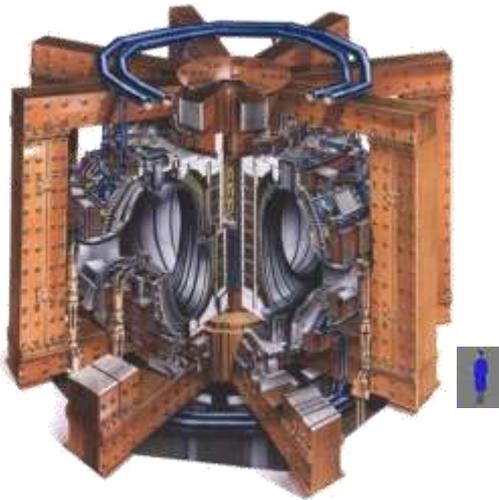


JET

World's largest facility

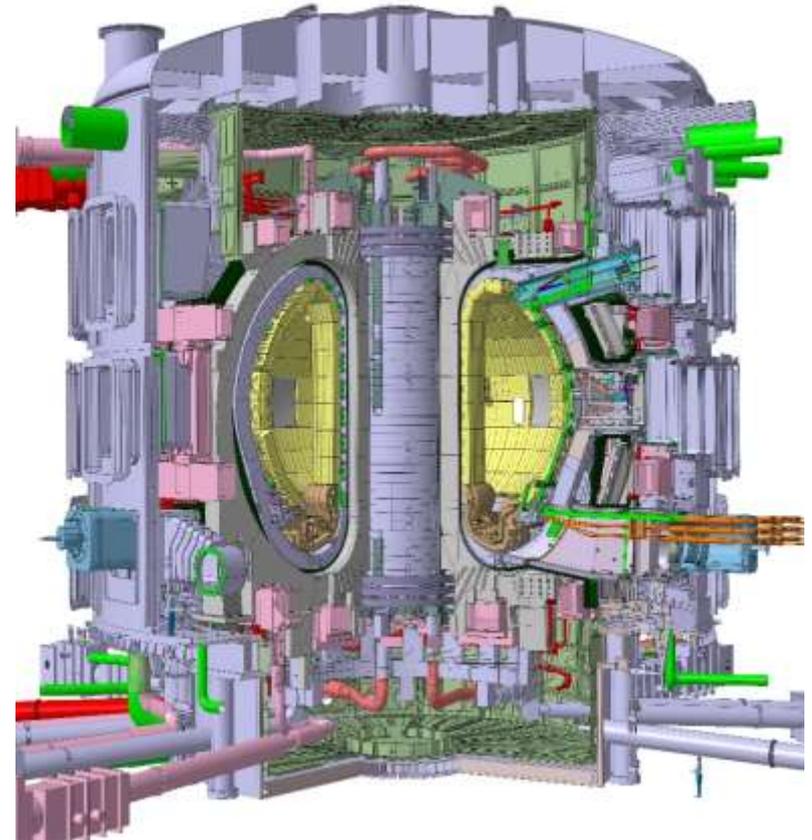
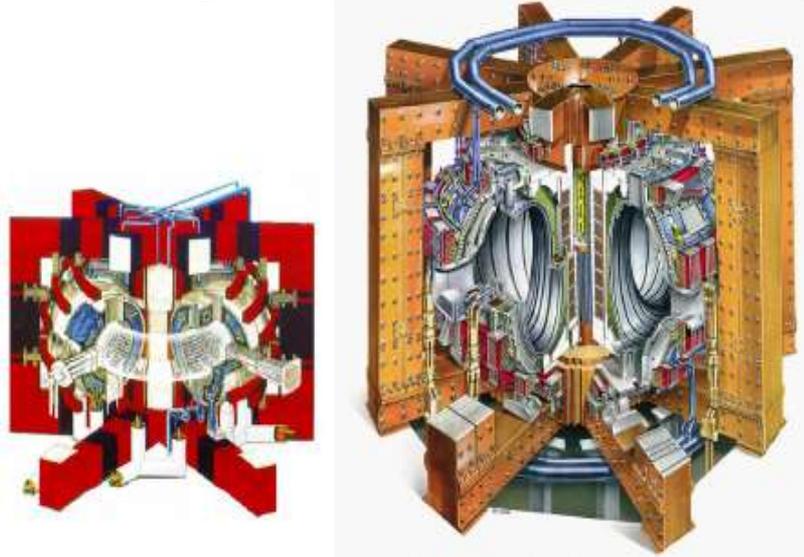
Fusion power record

16MW



ITER – the biggest tokamak ever built

Many tokamaks have been built over the past 60 years. ITER will be twice as big as the biggest tokamak still operating (JET, Culham, UK)



Tore Supra (France)

$V_{\text{plasma}} \sim 25 \text{ m}^3$, $I_p \sim 1 \text{ MA}$

$P_{\text{fusion}} \sim 0 \text{ MW}$

$t_{\text{plasma}} \sim 400 \text{ s}$

JET



$V_{\text{plasma}} \sim 80 \text{ m}^3$, $I_p \sim 3 \text{ MA}$

$P_{\text{fusion}} \sim 16 \text{ MW}$, 1 s

$t_{\text{plasma}} \sim 30 \text{ s}$

ITER



$V_{\text{plasma}} \sim 830 \text{ m}^3$, $I_p = 15 \text{ MA}$, $B_t = 5.3 \text{ T}$

$P_{\text{fusion}} \sim 500 \text{ MW}$, $300 - 500 \text{ s}$, $Q = 10$

$t_{\text{plasma}} \sim 600 - 3000 \text{ s}$, $Q = 5$

First Fusion plasma experiment

Aerial view of platform - September 2013





4.1 Progress in Building construction work



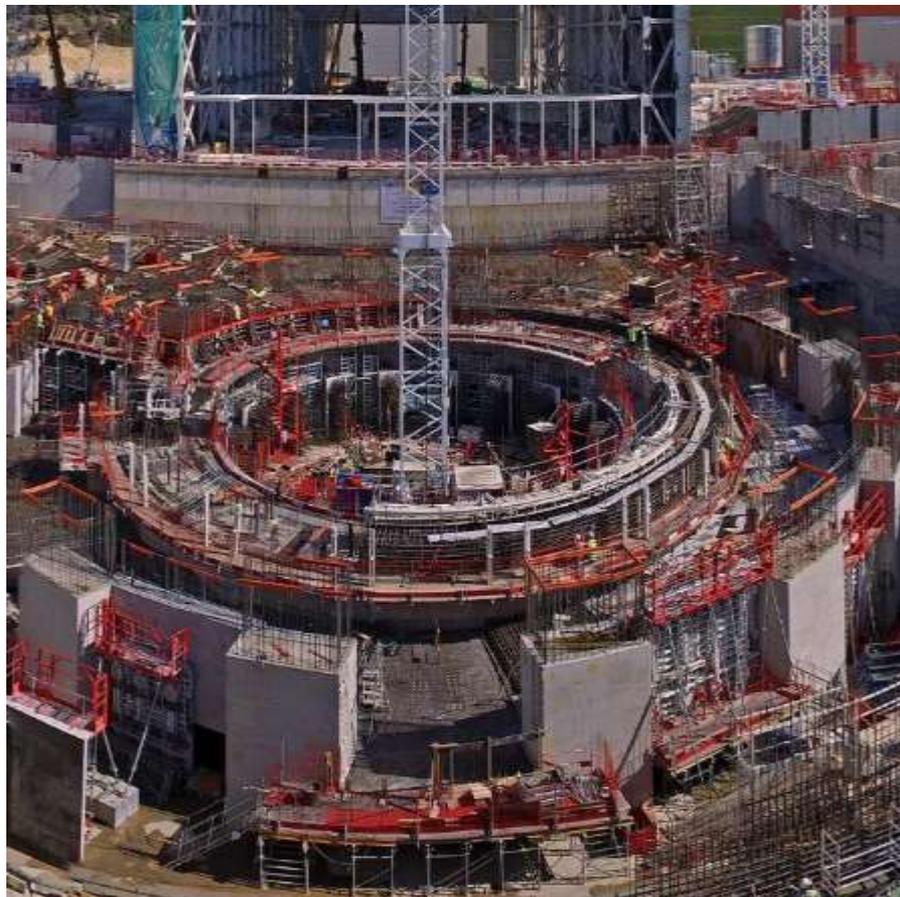
December 2016



April 2017



Tokamak Complex Bioshield raised at full speed



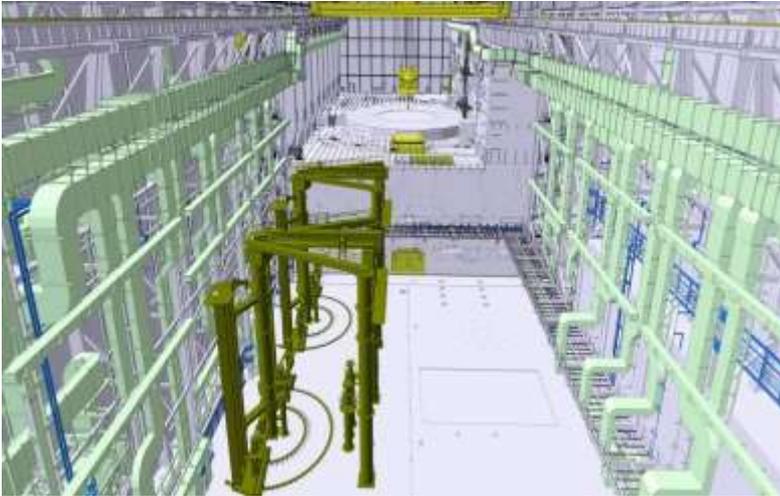
April 2016



April 2017

Начало монтажных работ в зд.11

17 May 2017, Корея

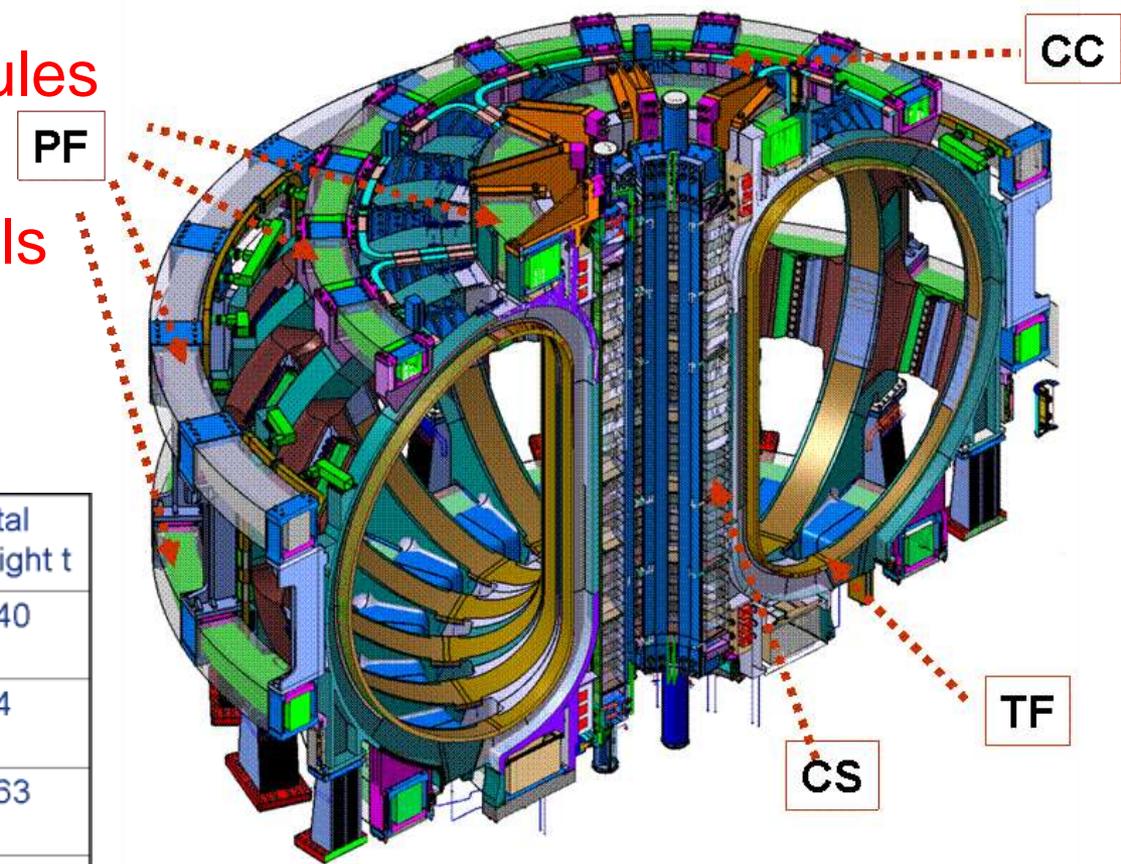


12.02.2018 зд.11 ИТЭР, сборка «сборочного механизма»



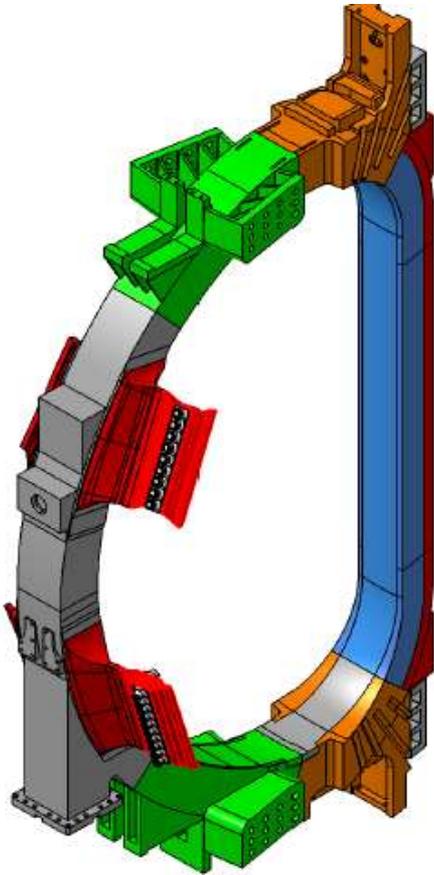
ITER - Magnet Systems

- 48 superconducting coils
- 18 Toroidal Field coils
 - 6 Central Solenoid modules
 - 6 Poloidal Field coils
 - 9 pairs of Correction Coils



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

Toroidal field coils



16 x 9 m, ~360 t
(EU, JP – 18 coils)

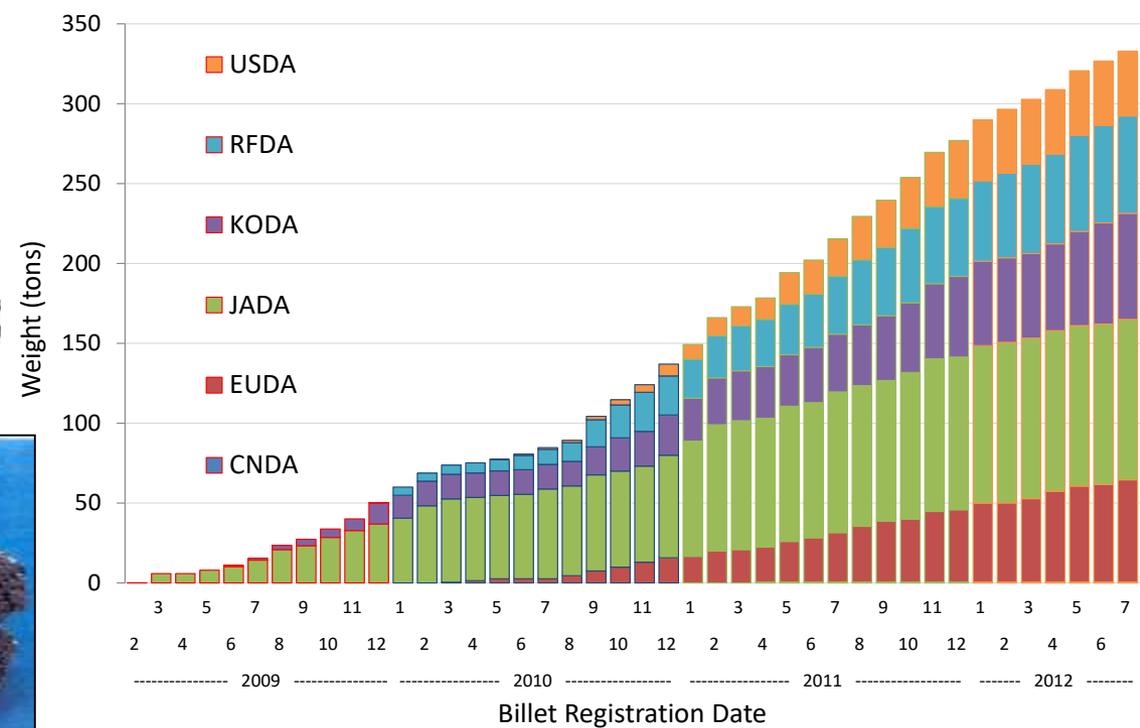
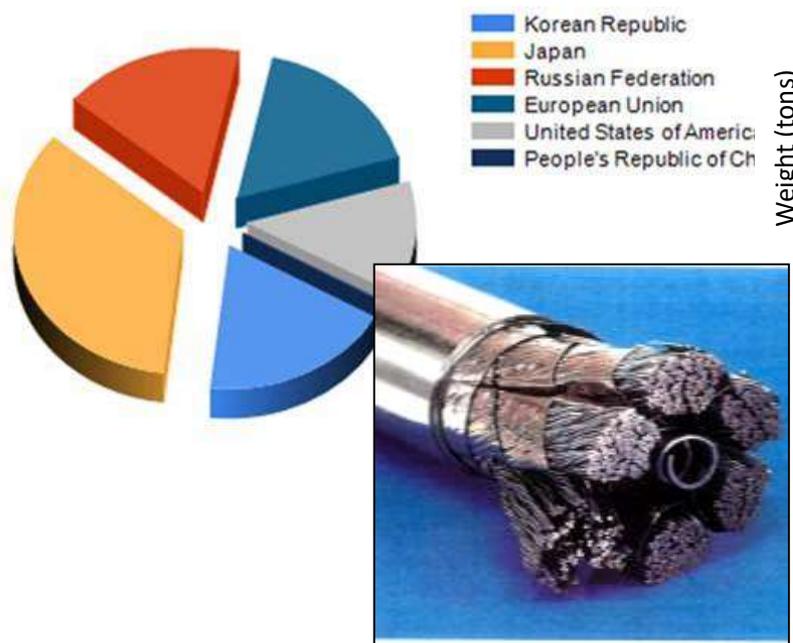


Boeing 747-300
(maximum take-off weight ~377 t)

TF Superconducting Strand Procurement is Largest in History

- Nearly **~450 tons (~90,000 km)** of **Nb₃Sn strands** have been produced (100%)
- The largest Nb₃Sn strand production ever – pre-ITER world production was **~ 15 t/year**.
- **Superconductor industry was created in Russia !!!**

Billet Weight Distribution By DA



TF Strand Production Summary

RF DA. Сверхпроводники для тороидального

и полоидального

П

RUSSIA TF Conductor Production In Russia

Cabling of 760 m Cu Dummy
at VNIIEP, RF (Feb. 09)



Испытания в НИЦ «Курчатовский Институт»



ITER newslines

Russian TF conductor successfully tested in SULTAN
-Alexander Petrov, ITER Russia



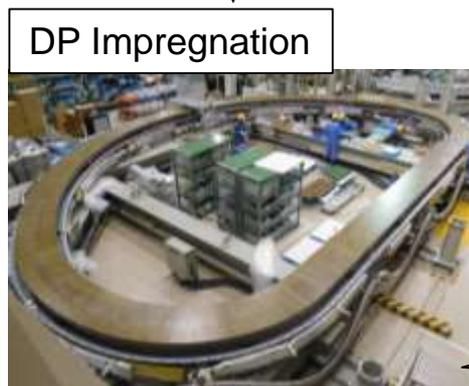
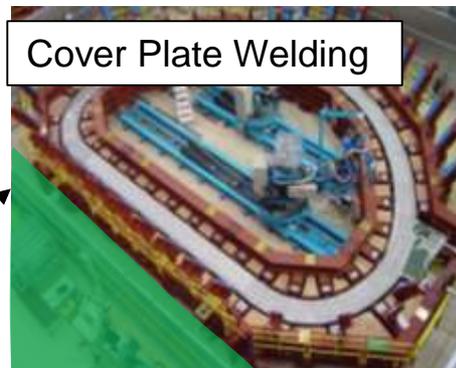
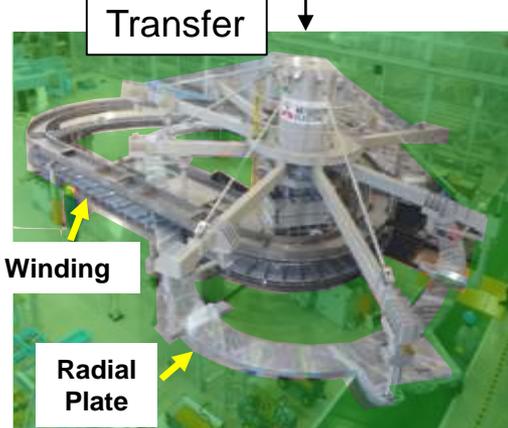
The results obtained with the TFRF4 (Toroidal Field Russian Federation #4) sample show very good agreement with results of the two last samples TFRF2 and TFRF3, which demonstrated the relatively good stability of the conductor during electromagnetic cycling, as well as its good durability during the warm-up/cool-down procedure.



Поставки:

- на SULTAN (Швейцария) для испытаний
- в CRIOTEC Co. (Италия) для джекетирувания

Progress of TF Coil Winding



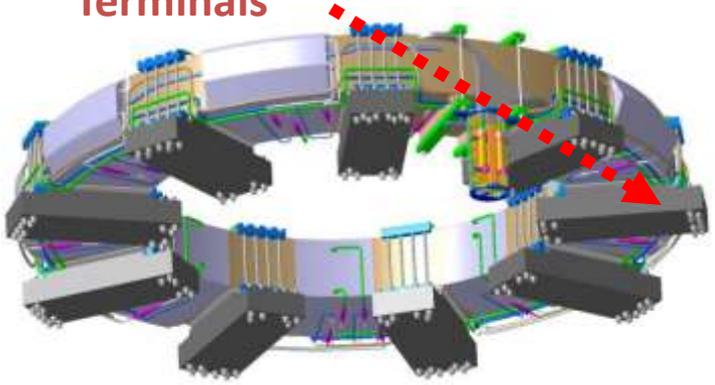
Insulation wrapping of 7 DPs stacked for first JA TF Coil was completed in January 2017, which is timely accomplishment of one of the IC milestones.

Futami Plant of Mitsubishi Heavy Industries Ltd.(MHI)/Mitsubishi Electric Co.

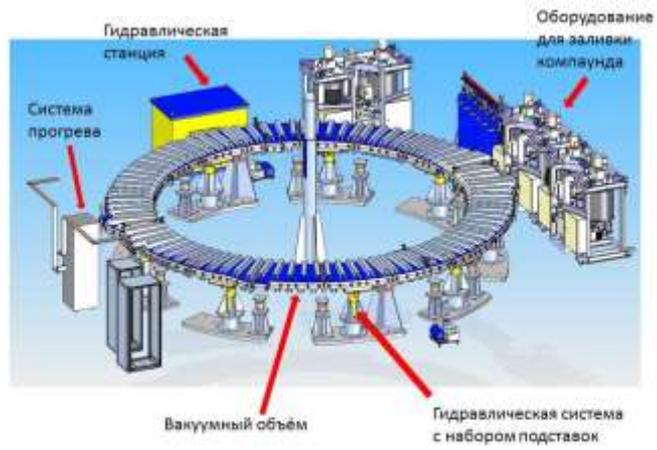
EU & RF

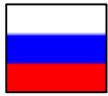
Poloidal Field Coils

Terminals



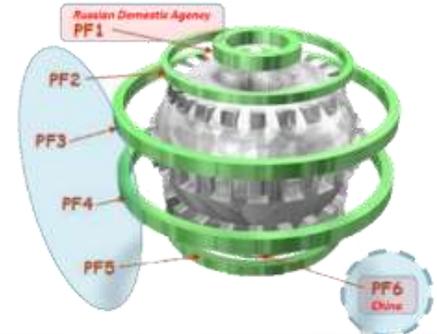
- So large that they most must be manufactured on site
- PF3: 24.5 m dia. & 386 ton
- Building is 250 m long x 45 m wide and is the first building on site!





Manufacture stages of PF-1 Double Pancakes

IC-63 Milestone: Q1 2021_RF, PF Coil: Manufacturing complete for RF PF1 Coil



The vacuum impregnation (VPI) of 1st Double Pancake .



The process of interlayer isolation



Manufacturing of joint lead



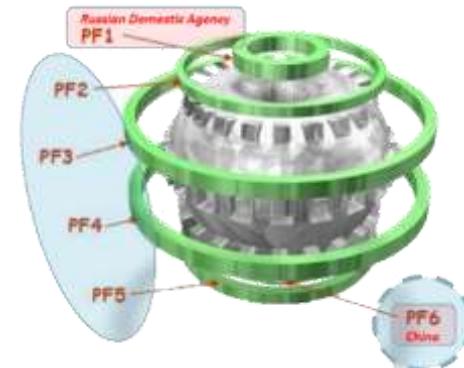
Winding of 2nd Double Pancake ongoing

Technology of large superconducting coil manufacture was created in Russia !!!

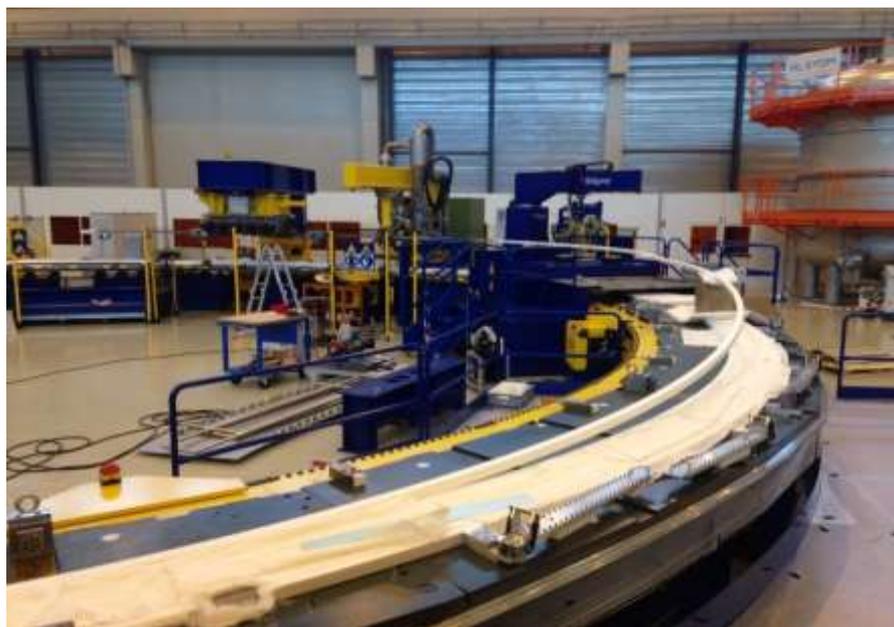


PF2-6 Progress

- PF2-5 pre-dummy was wound using PF4 dummy conductor in September 2016. Dummy DP windings have been completed both in Cadarache for PF5 and in ASIPP for PF6 in February 2017.



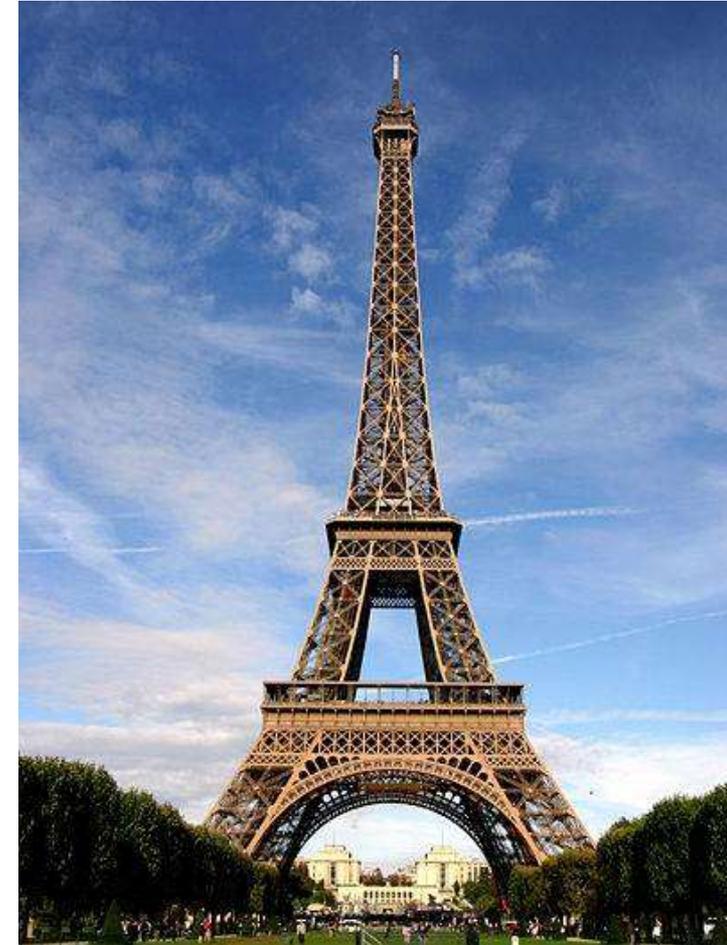
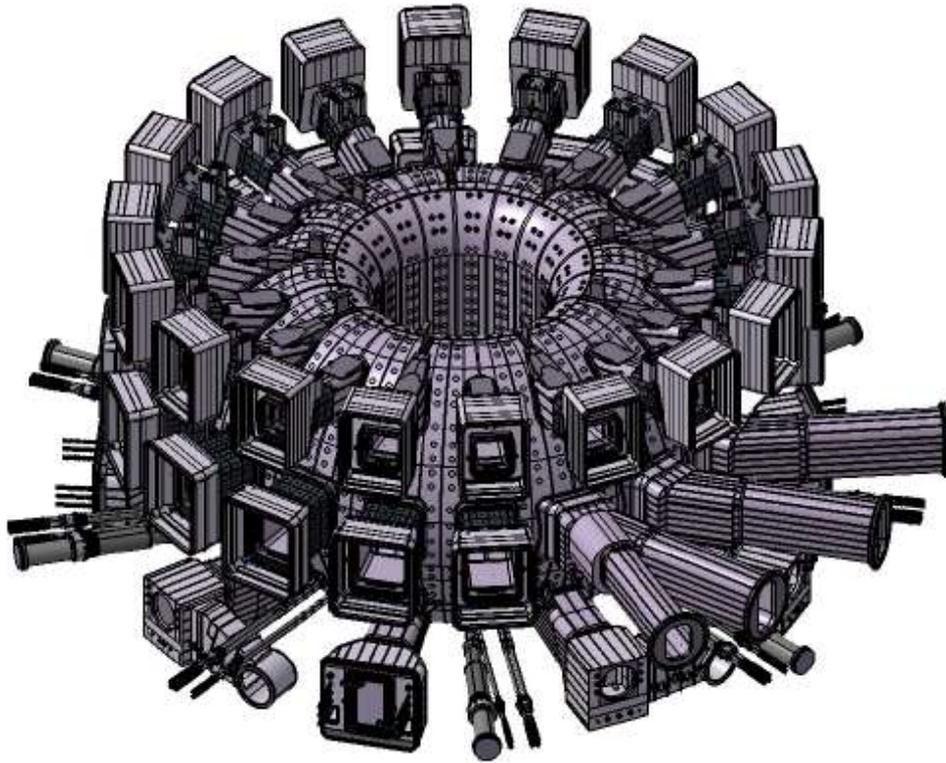
**PF2-5 Pre Dummy DP winding
(September 2016)**



PF5 Dummy DP winding (February 2017)



Vacuum Vessel



7300 t
324 m tall

Weighs about 25% less than the Eiffel Tower (more with in-vessel components included)



Vacuum Vessel: Progress of the First Sector #6 (Upper Segment)



October 2016



April 2017

Progress of VV Ports



KO VV Ports (PBS15)

- ✓ Assembly of inner shells for every Lower Port Stub Extension (LPSE) has been completed and assembly of outer shells is progressing as scheduled.
- ✓ In addition, all Lower Port Extensions (LPE) are being manufactured to schedule.



T-rib to Inner Shell assembly of LPSE



Outer Shell assembly of LPSE



T-rib to Inner Shell of LPE

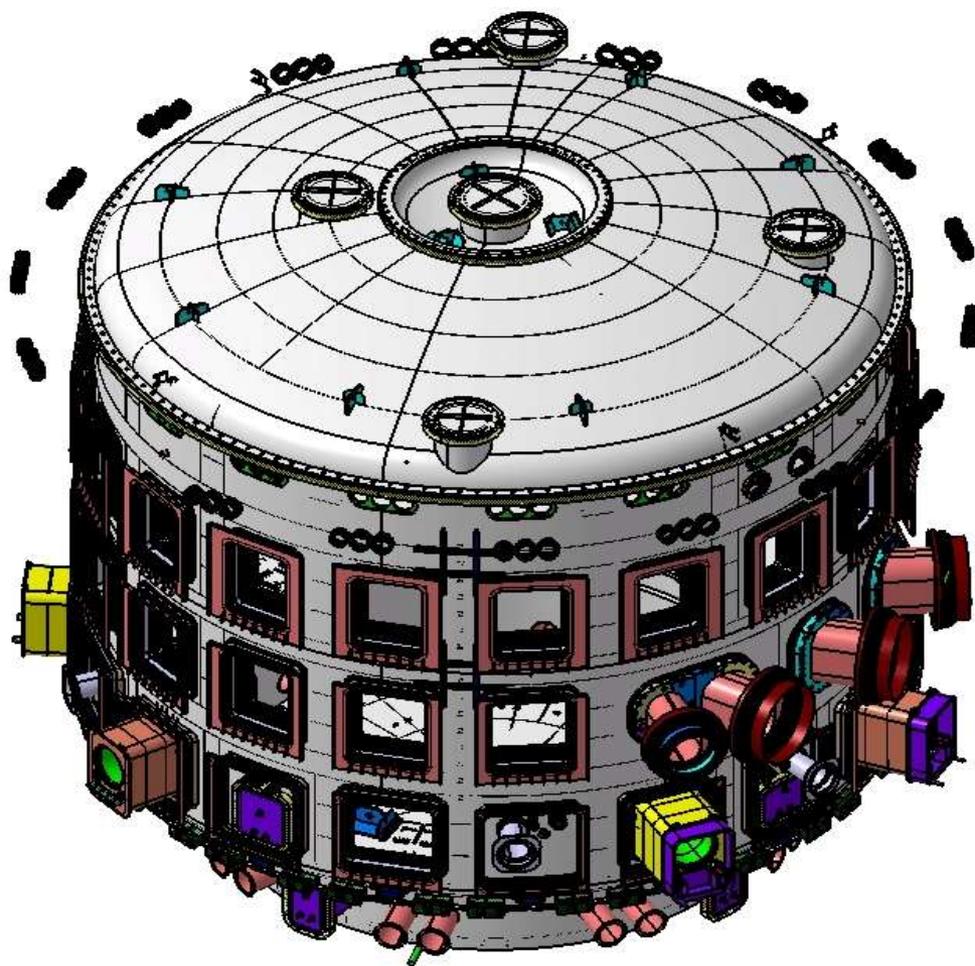


RF Upper VV Ports

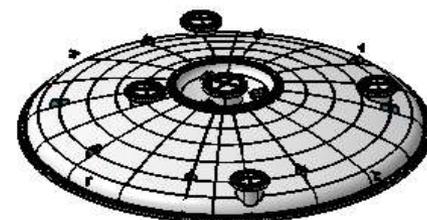
**4 Upper VV ports were manufactured in 2017
First UP12 was supplied to HHI (Korea) in
December 2017 and second one in the first half
of 2018**



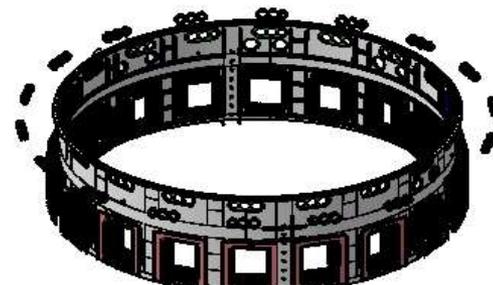
ITER Cryostat



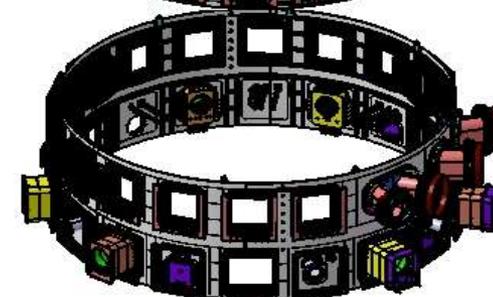
Top lid



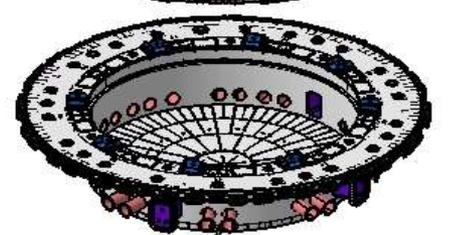
Upper cylinder



Lower cylinder



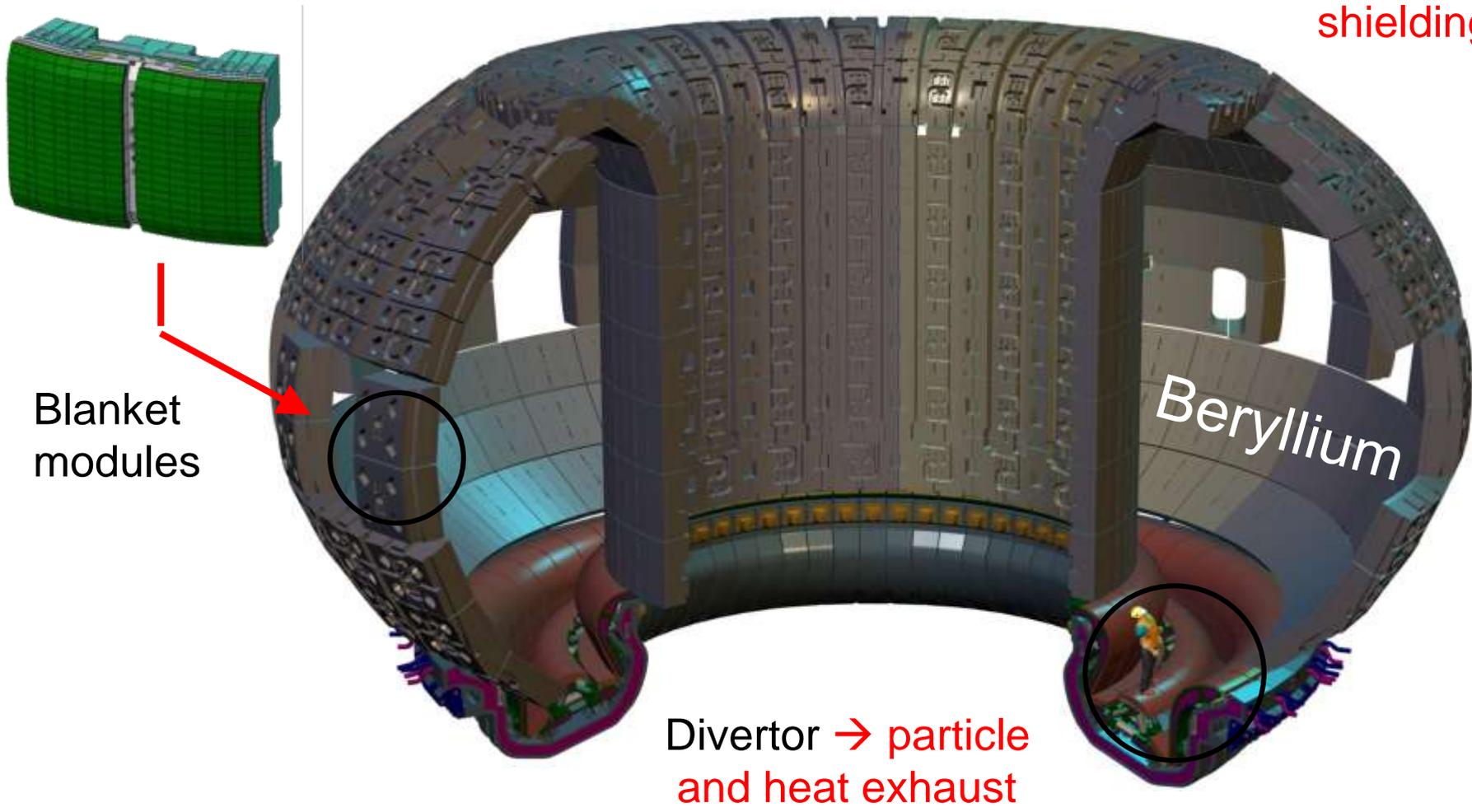
Base section



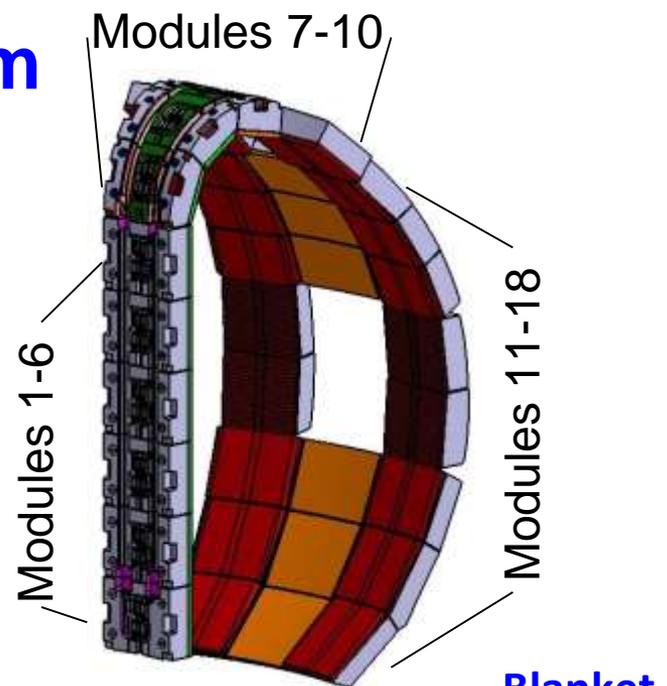
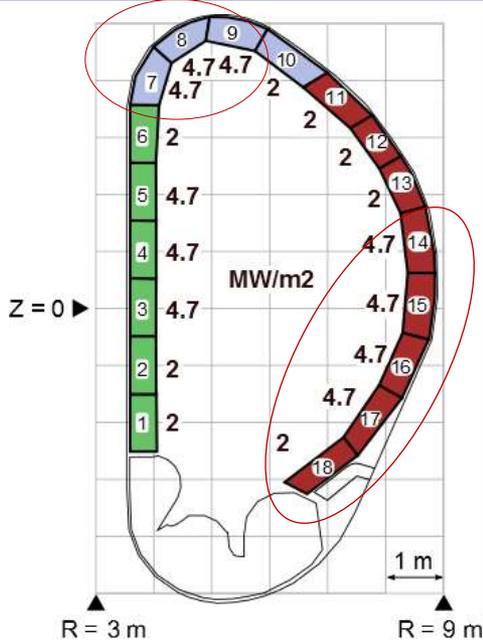
Transfers loads to tokamak floor

Principal plasma-facing components (PFC)

First wall/blanket → heat exhaust, impurity management, nuclear shielding

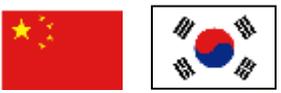


Blanket System



Shield Block
(semi-permanent)

50% 50%

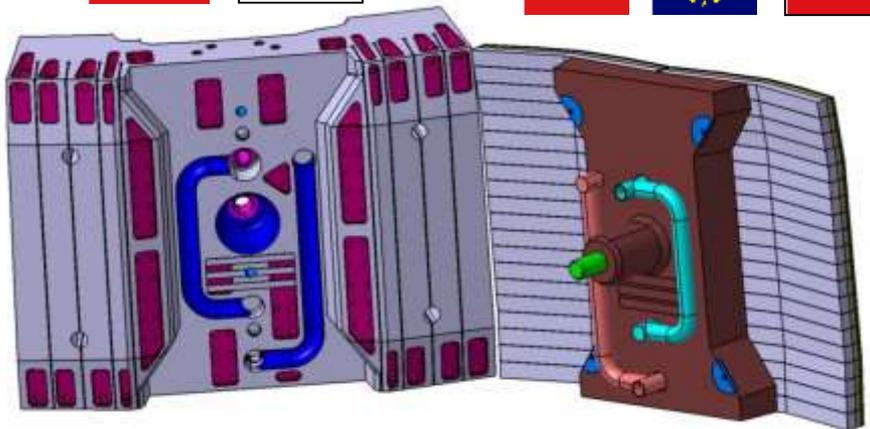
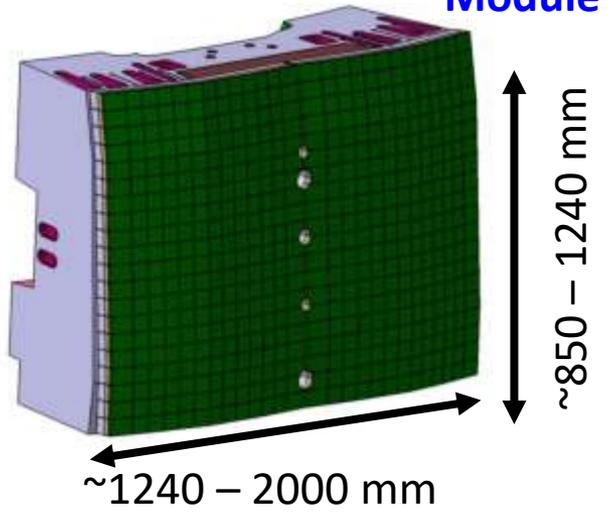


FW Panel (separable)

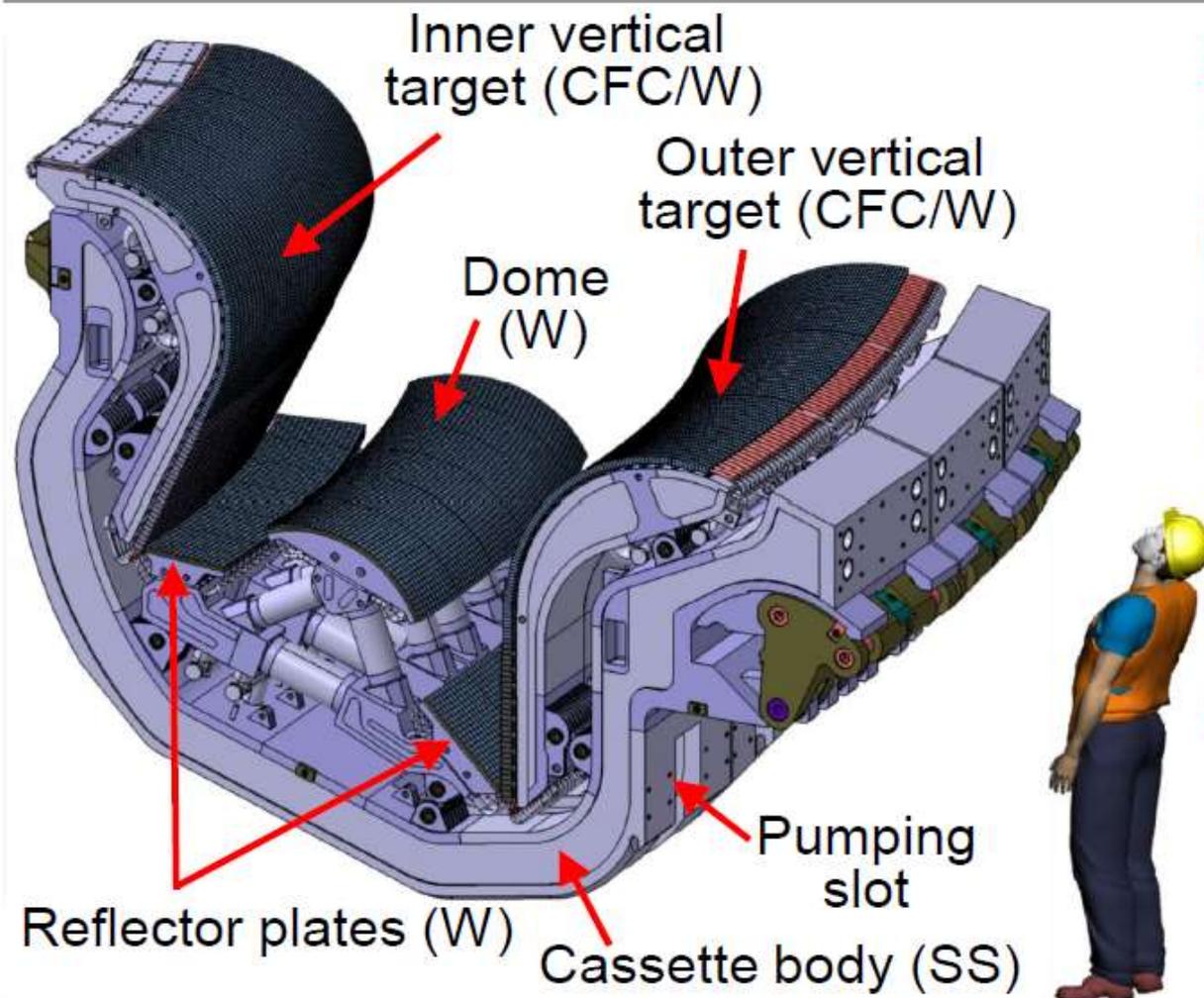
10% 50% 40%



Blanket Module



Divertor assembly



54 divertor assemblies
(~8.7 tonnes each)

4320 actively cooled
heat flux elements

Bakeable to 350°C

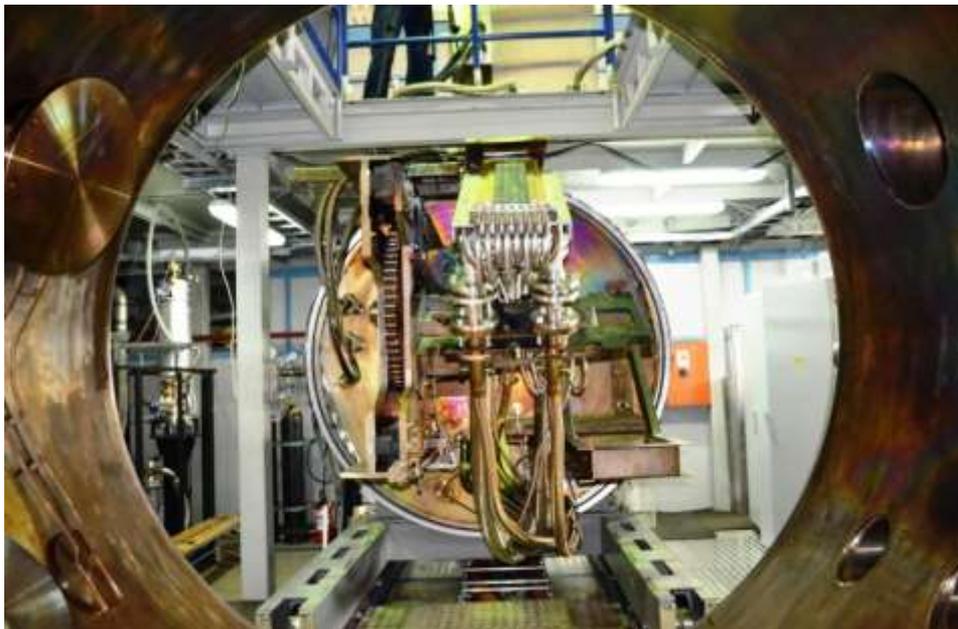
First divertor (non-
active operation):

CFC at strike points
W on the baffles

All-W for nuclear phase

Технологии и производственные линии по созданию элементов
Вольфрамового дивертора созданы в России !!!

**Испытания проводит:
НИИЭФА им.Д.В.Ефремова**



A 800 kW electron injector exposes the components to the same heat loads - up to 20 MW/m² - they will face inside the ITER vacuum vessel.

**IDTF – начало испытаний JA пластин
+ испытания W на КСПУ**



**Трехмерная модель имитатора
европейской испытательной сборки**

**Экспериментальные стенды для испытаний элементов первой стенки и дивертора
созданы в России !!!**



Progress on Electrical Component Fabrication

IC-68 Milestone: Q1 2022_RF, Electrical: Complete deliveries to site for Switching Network, Fast Discharge Units, DC Busbar & Instrumentation



Aluminum busbars PF / CS, SS at the assembly site for Bld.32,33



Copper insert



Test facility for circuit breaker



Section of the SNR CS1 resistor assembly



Busbars—the components that connect the superconducting magnets with their power supply—are capable of carrying 70kA of current.



Protective circuit breaker 33-10/60

Electrical component fabrication and deliveries continue according schedule !!!

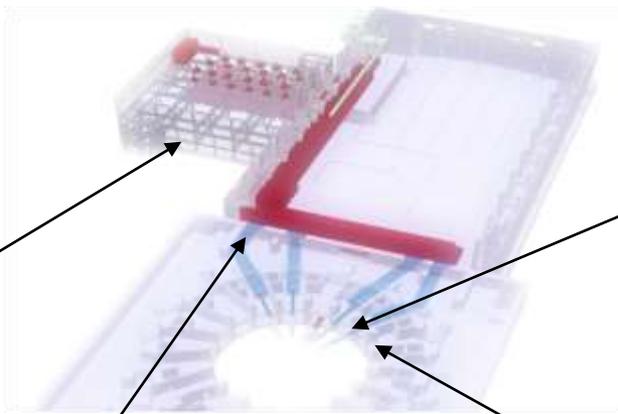
Components for Electron Cyclotron H&CD System are Under Development

Sources: JA and RF Gyrotrons

and their Test Facilities



J
A



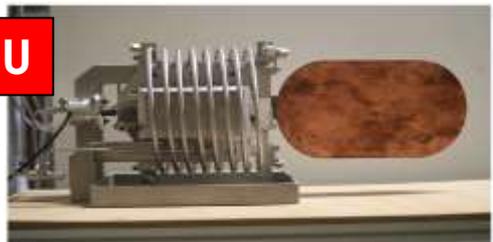
Launchers: JA and EU Tests

J
A



Equatorial Launcher High Power tests

EU



Friction-free Steering Mechanism

TL: Tests at USIPO and JAEA test Facilities

US



J
A



RF



- HV power supplies (55kV, 100A) and (50kV, <1A) (EU, IN)
- Microwave Sources (1MW, 170GHz) (EU, IN, JA, RF)
- Evacuated waveguide components (US, EU)
- Launching Antennas (EU, JA)
- Control systems (JA, EU, RF, IN, US)

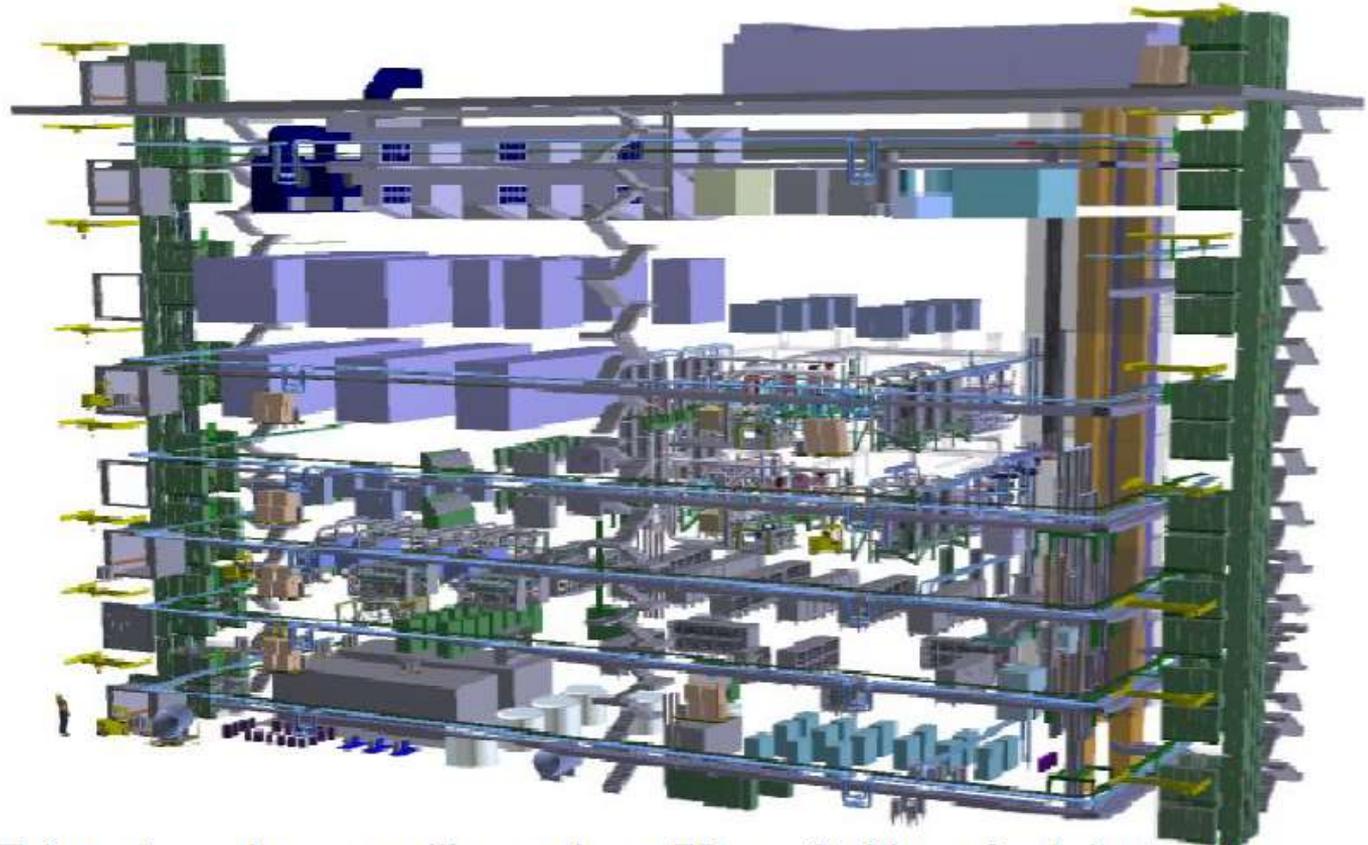
Tritium Plant

7 storeys
(2 below
ground
level)

L = 80 m

W = 25 m

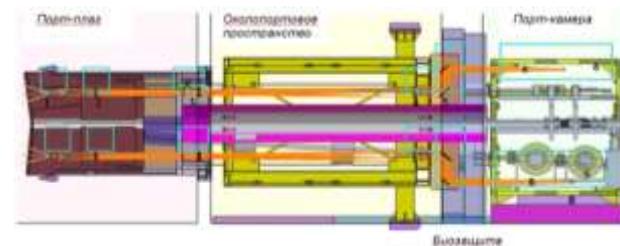
H = 35 m



- A 400 s ITER burning plasma will require ~50 g of tritium fuel, but only ~0.3% will be burned by fusion reactions → T reprocessing mandatory

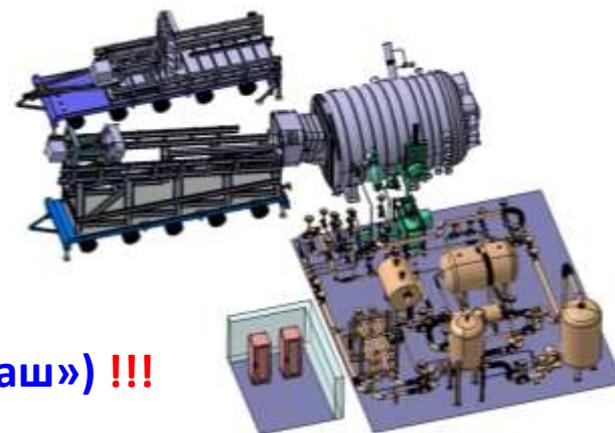
В соответствии Соглашением по ИТЭР Россия имеет право на безвозмездную лицензию для внутренней программы УТС на созданные для ИТЭР тритиевые технологии !!!

Порт-плаги верхние



Порт-плаги экваториальные

Port Plug Test Facility



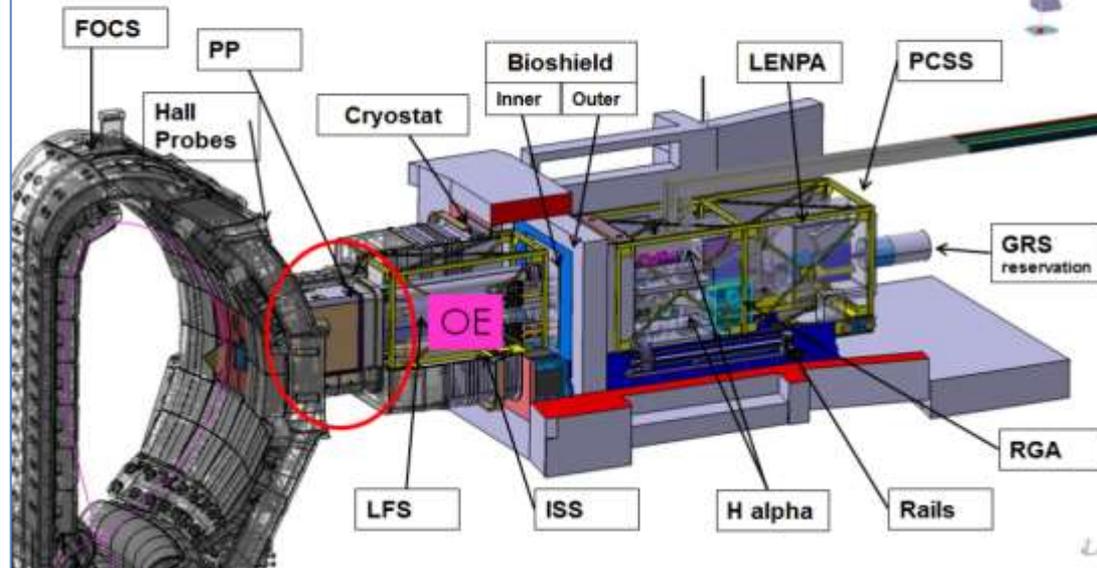
Порт-плаги и Стенды испытаний порт-плагов создаются в России (ИЯФ СО РАН и АО «Криогенмаш») !!!

Equatorial Port Plug 11 integration according to Schedule (ИЯФ СО РАН)

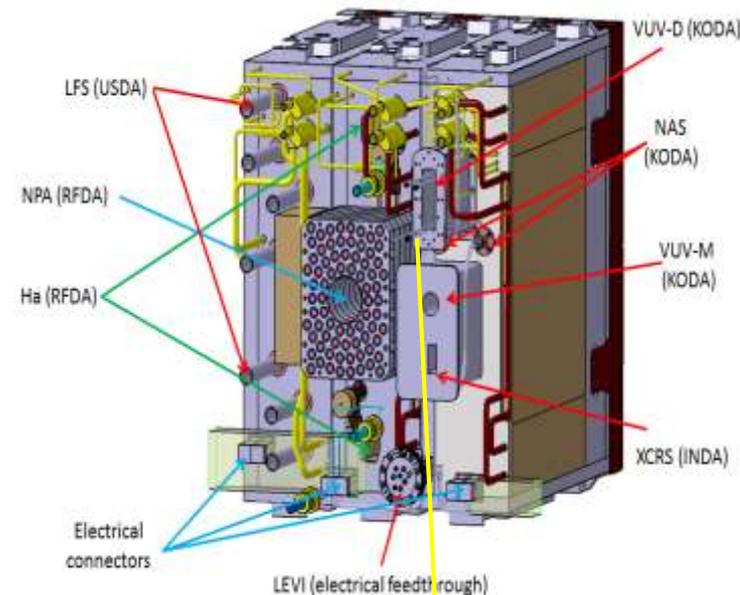


Integrating things together (Super-Project)

- Joint team working closely together to integrate diagnostics and services
- Design for Safety / Function / Reliability / Maintenance

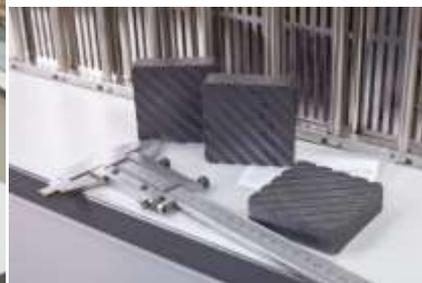
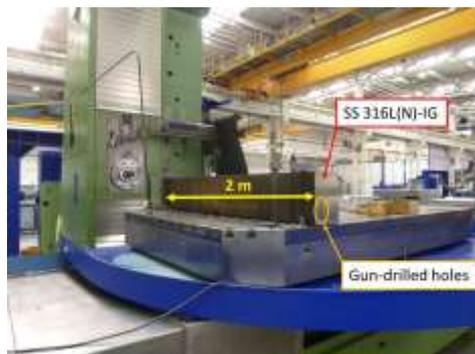


PBS 5.5.P1.RF.06



Design of DFW, DSM & Closure Plate

Integration of diagnostics and neutron shielding optimization is in progress for EPP 11.



The mock-up of in-vessel neutron shielding for DSM

The mock-ups of B₄C bricks for in-vessel neutron shielding

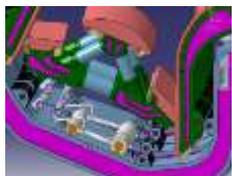


Mock-up of assembly / disassembly of NPA with the possibility of remote control

Successful test of gun-drilling technology for holes of different diameters trough SS piece two meters long.

Нейтронные

Монитор потока нейтронов в диверторе

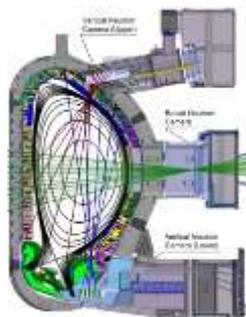


Детекторы на основе U-235 и U-238

Алмазный спектрометр для быстрых атомов и нейтронов



Вертикальная нейтронная камера для томографии



Диверторное томсоновское рассеяние



Полномасштабный прототип внутрикамерной оптики Лазер 2J / 100Hz



Спектральное оборудование:

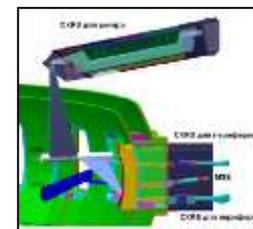
- Полихроматор 5-200 eV
- Полихроматор 0.3-200 eV

• Лазерная система:

- Nd:YAG 1.0645 μm (2 J, 3 ns, 50Hz)
- Nd:YLF laser 1.047 μm (2 J, 10 ns, 5Hz)
- Nd:YAG laser 0.946 μm (0.1 J, 10 ns, 100 Hz)

Оптические

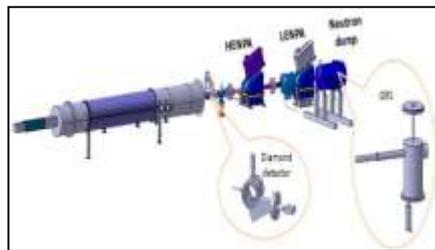
CXRS



Н-альфа спектроскопия

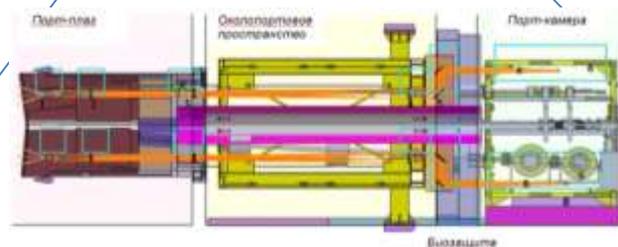
Диагностика плазмы

Анализатор нейтральных частиц



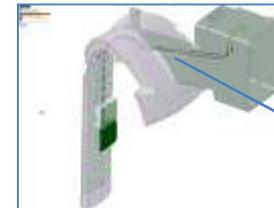
Два анализатора нейтралов: HENPA (0.2 – 2 MeV) и LENPA (10 – 200 keV)

Порт-плаги



Верхний патрубок 02, 08, 14
Нижний патрубок 08
Экваториальный патрубок 11

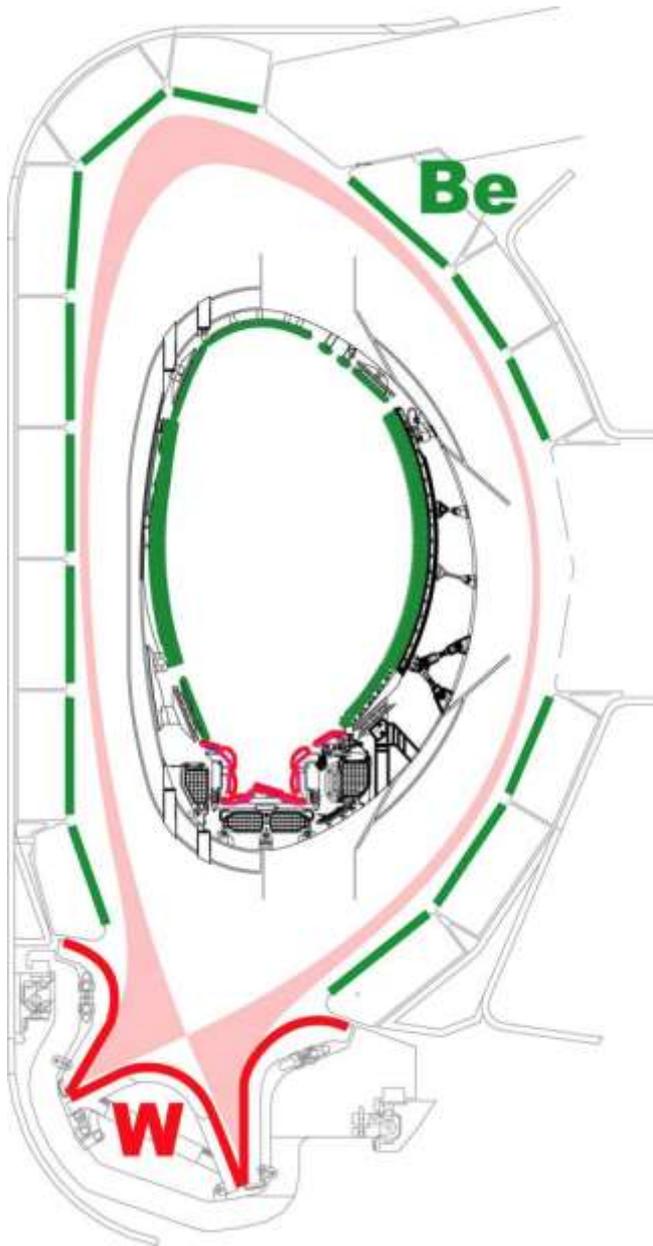
Рефлектометрия



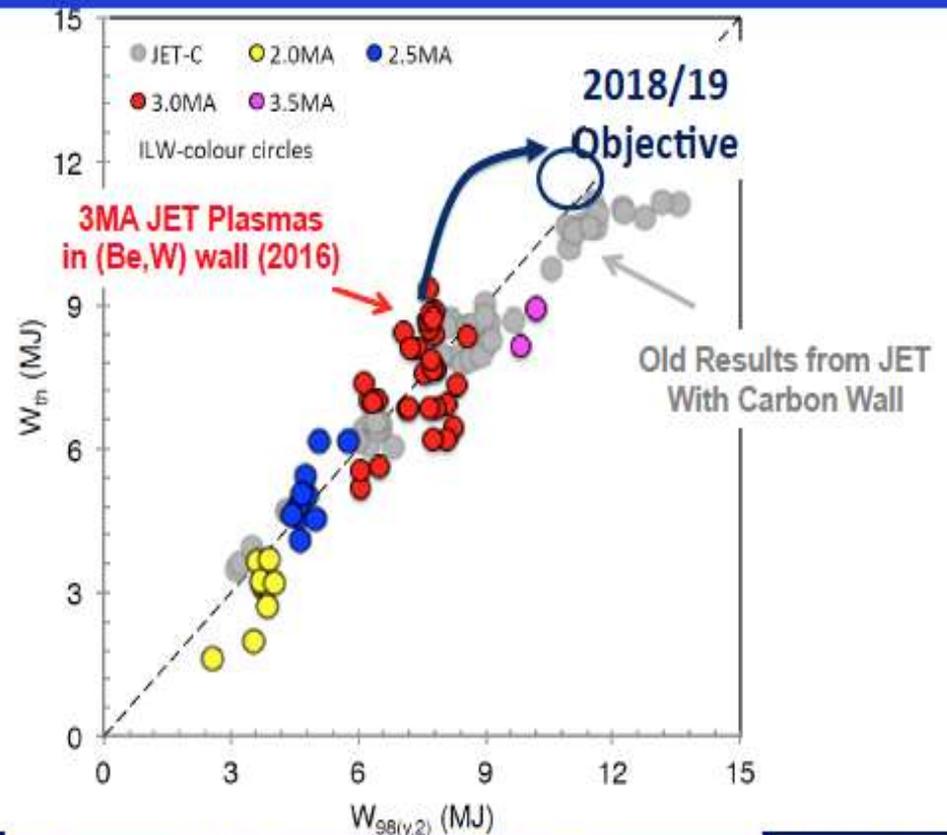
Широкая полоса пропускания от 12 до 140 ГГц с несколькими поворотами и вакуумными окнами

11 диагностических систем ИТЭР создаются в России !!!

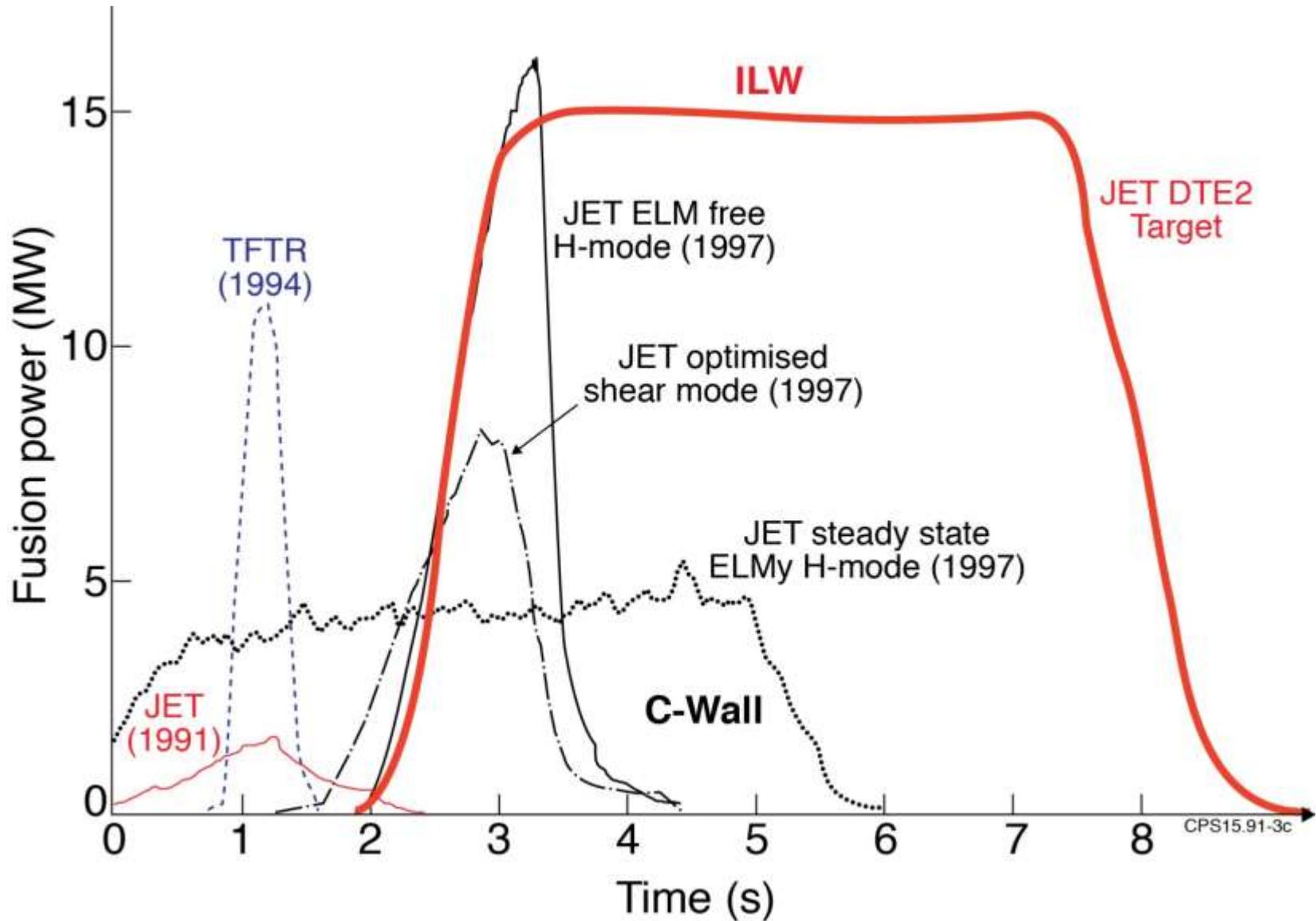
Программы УТС Партнёров. **JET (EC)**



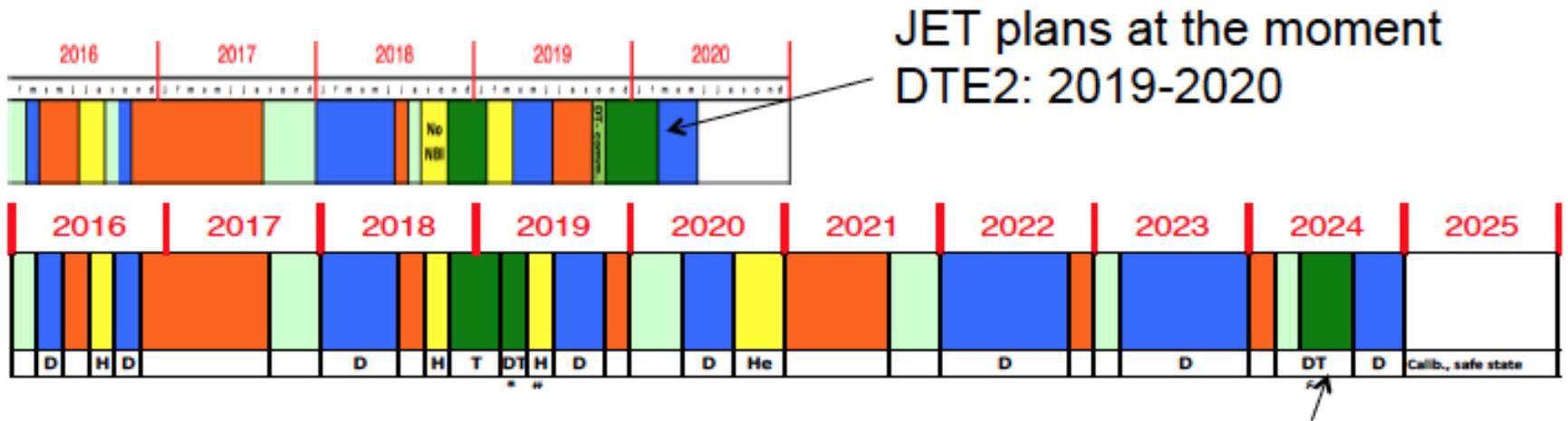
Наилучший результат JET с ITER-подобной стенкой:
30МВт 3МА/2.7Т



JET, TFTR. Fusion power



D-T кампании на JET: DTE3 – 2024?



DTE3:
Possibly 2024?

Программы Партнёров **W7-X (EC)**

W7-X is a helical axis stellarator with 5-fold symmetry, $\langle \beta \rangle \approx 5\%$

Reached :

30s, 2MW (ECRH), $3 \cdot 10^{19} \text{m}^{-3}$

$T_e / T_i = 4 / 2 \text{ keV}$

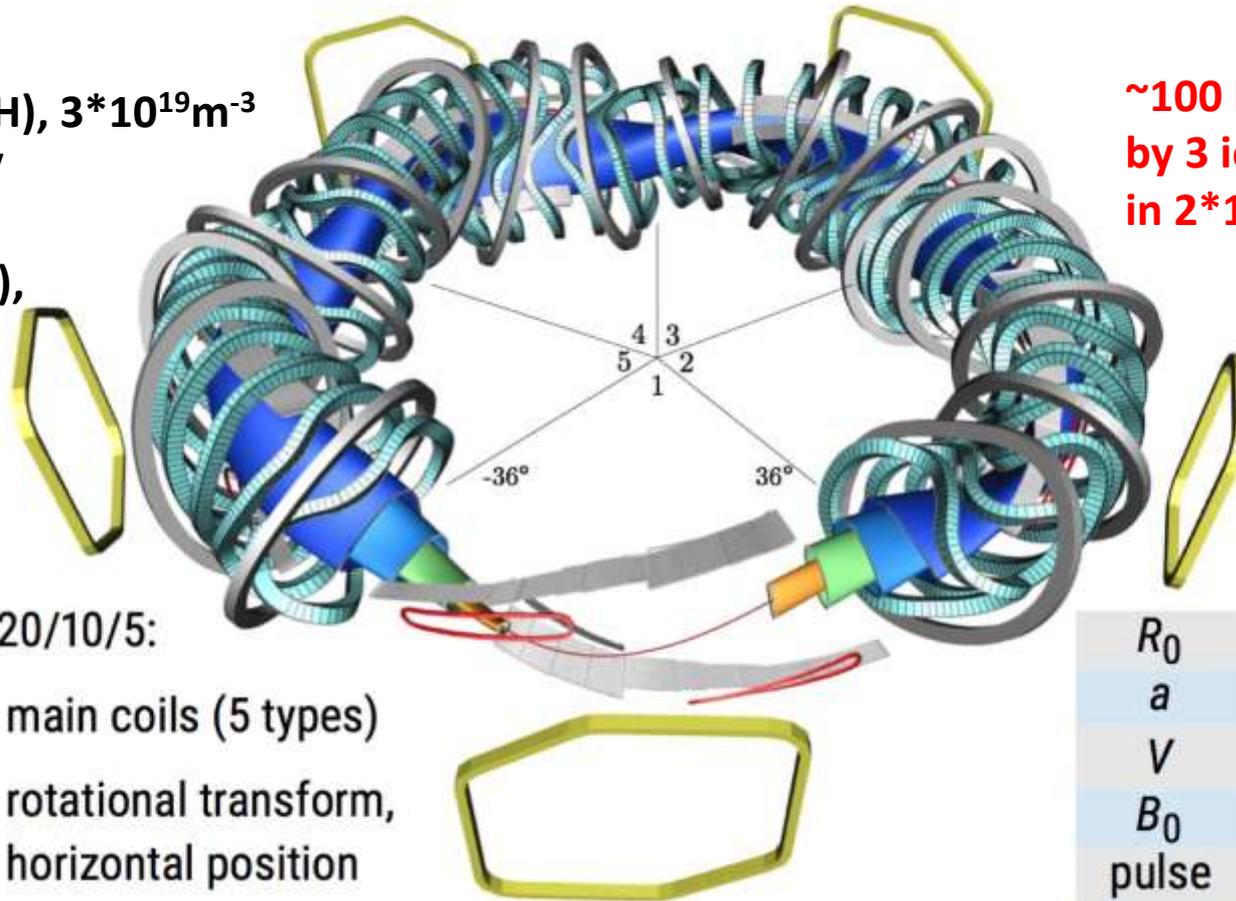
4s, 5MW (ECRH),

$7 \cdot 10^{19} \text{m}^{-3}$

$T_e, T_i = 4 \text{ keV}$

$\tau_e \sim 0.2 \text{ s}$

**$\sim 100 \text{ keV}$ ions
by 3 ions ICRH scheme
in $2 \cdot 10^{20} \text{m}^{-3}$**



Coil systems 50/20/10/5:

non-planar main coils (5 types)

planar rotational transform,
horizontal position

control adjustment of divertor islands

trim compensation of error-fields

R_0	5.5 m
a	0.5 m
V	30 m^3
B_0	$\leq 3 \text{ T}$
pulse	30 min
ECRH	10 MW/30 min
NBI	10 MW/10 s
ICRH	2 MW/10 s

Wendelstein 7-X: Операционные фазы

W7-X experiments are divided in operation phases. The machine capabilities are defined by the in-vessel components and available heating [Pedersen 2015,Wolf 2016] .

OP1.1: 2015-2016



- 5 graphite limiters
- metal first wall
- limited wall conditioning
- ECRH 4 MW, 4 MJ

OP1.2: 2017-2018

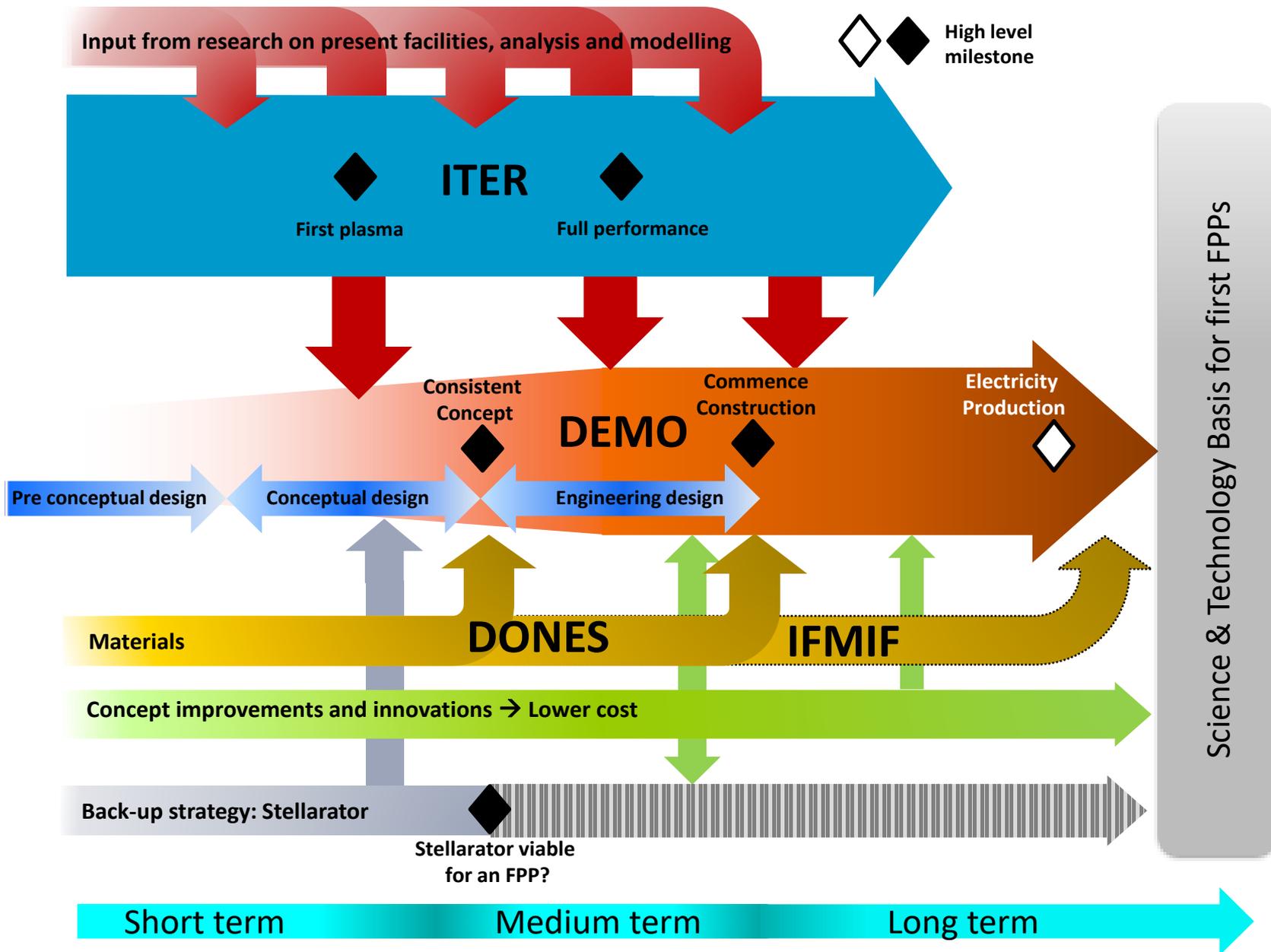


- graphite test divertor
- graphite/metal first wall
- ECRH 8 MW, 80 MJ
- NBI 7 MW

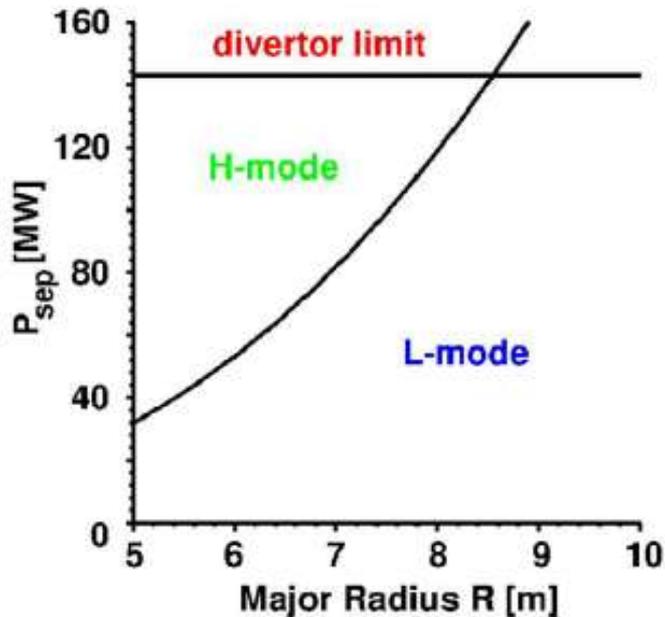
OP2: 2020+



- water-cooled divertor
- water-cooled first wall
- ECRH 10 MW, 18 GJ
- NBI/ICRH 10/2 MW

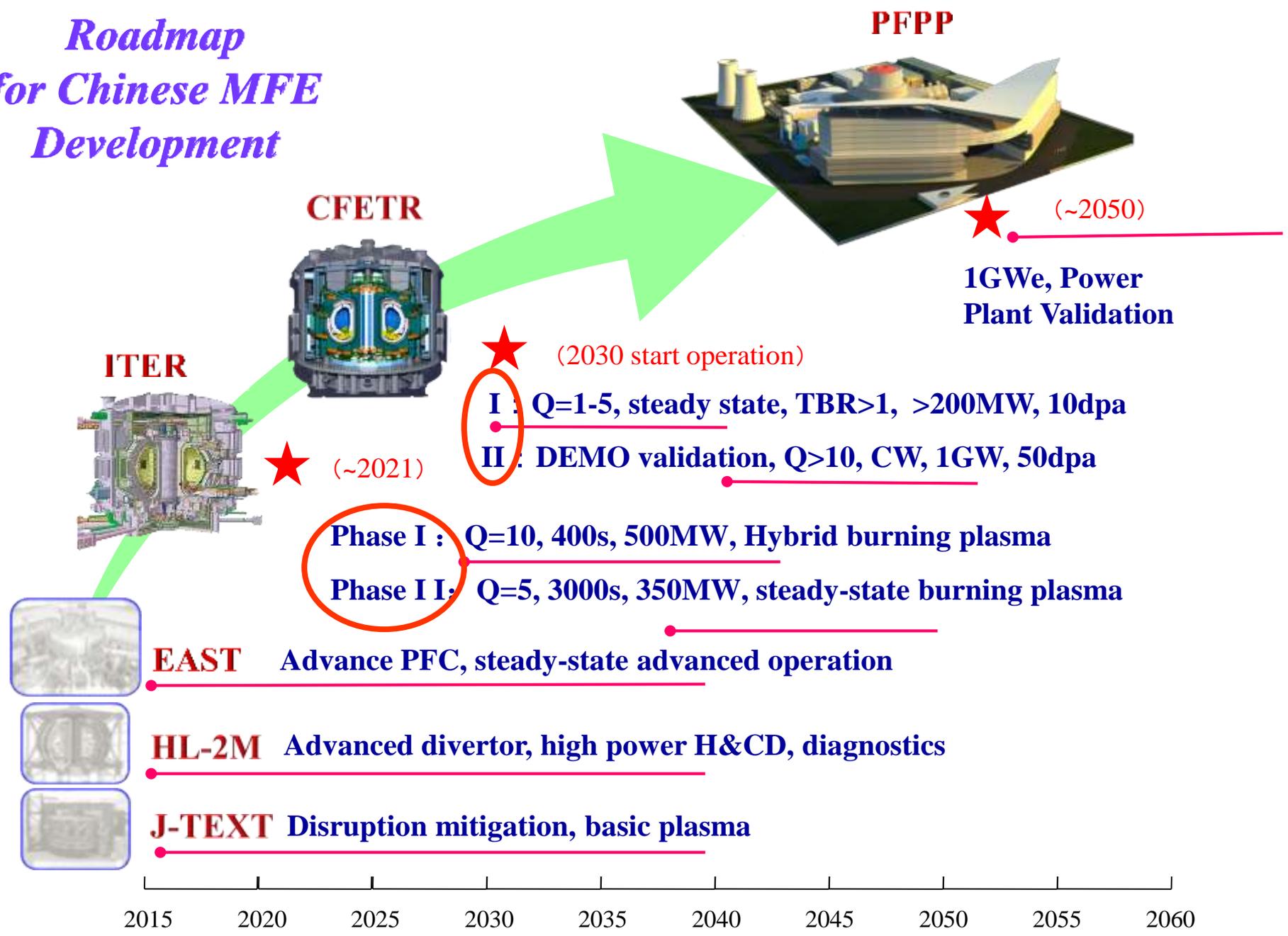


Параметры EU-DEMO 1



Fusion power	2037 MW
Net electric power	500 MW
H&CD power	50 MW
Major radius	9.1 m
Aspect ratio	3.1
Elongation	1.59
Triangularity	0.33
Toroidal magnetic field	5.7 T
Number of TF coils	18
Edge safety factor	3.2
Plasma current	19.6 MA
Average Greenwald density fraction	1.2
Tungsten concentration	5.0×10^{-5}
Energy confinement time	4.24 s

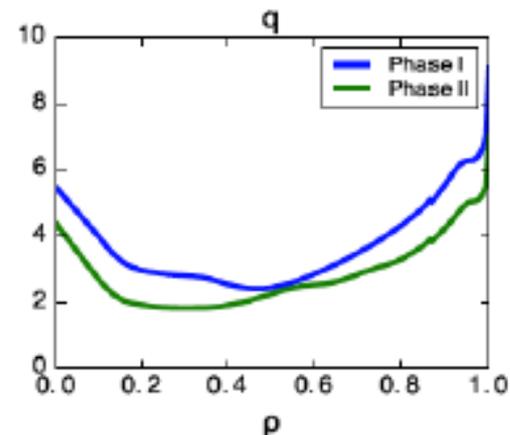
Roadmap for Chinese MFE Development



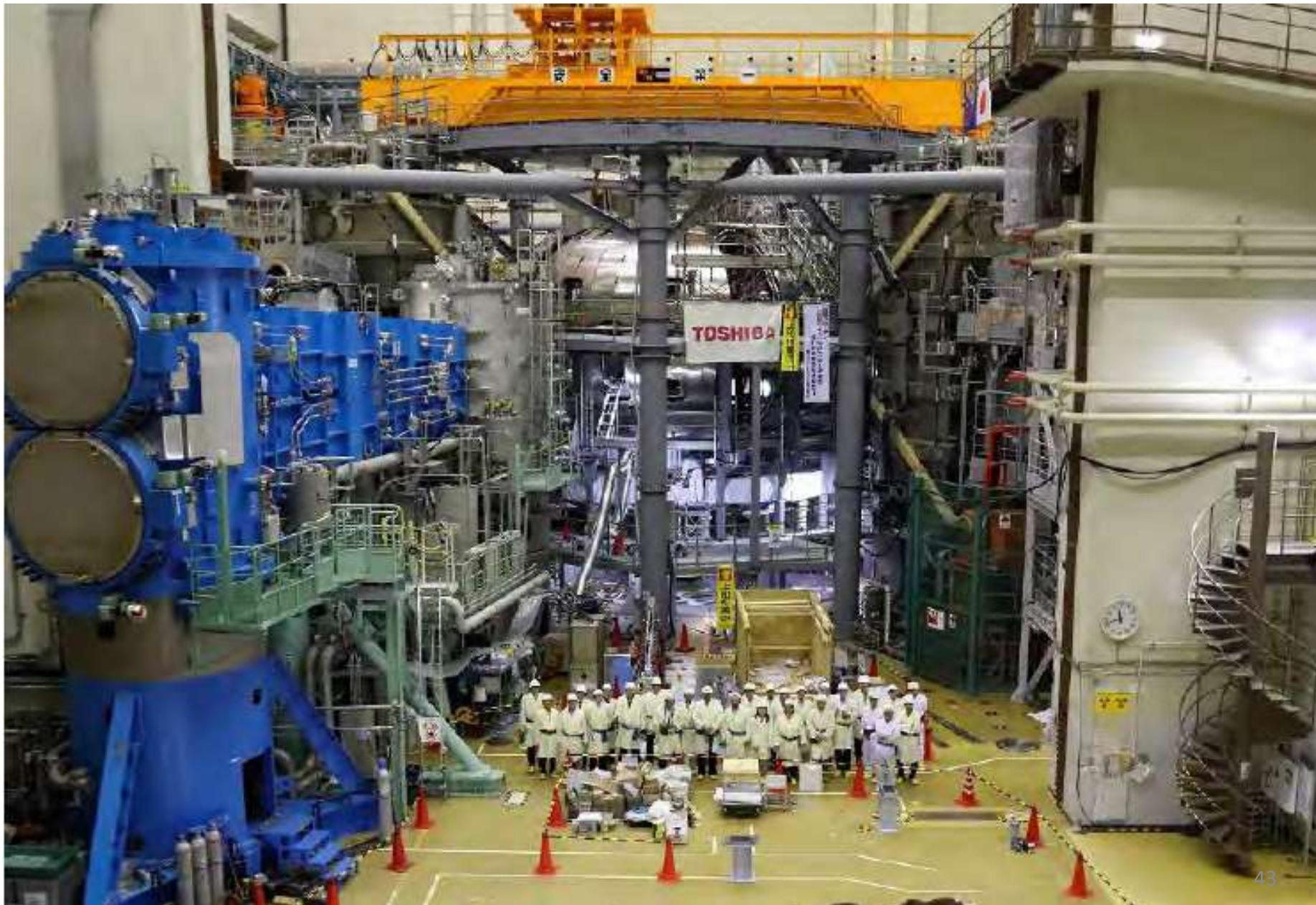
A Steady-State, High Performance, High Radiation CFETR is Demonstrated

	Phase I	Phase II
Q_{fus}	3.0	14.9
P_{fus} (MW)	169	811
I_p (MA)	7.6	10
Bootstrap fraction f_{bs} (%)	64	84
β_N	1.9	3.2
H_{98}	1.3	1.3
NB/EC Power (MW)	36/20	35/20
Neutron wall loading (MW/m ²)	0.19	0.92
Divertor heat load P_{div}/R (MW/m)	10.4	25.8
Ion fraction $n_D/n_T/n_{\text{He}}/n_{\text{Ar}}$	0.43/0.43/ 0.05/0.003	0.43/0.43/ 0.05/0.003
Ratio to Greenwald Limit	.83	1.03

- $R=6.7\text{m}$, $B_T=6\text{T}$
- 2 NBs: 100/500 keV
- Additional fueling needed for phase II



Программы УТС Партнёров. JT-60SA (Япония, ЕС)



JT-60SA Research Regime for DEMO

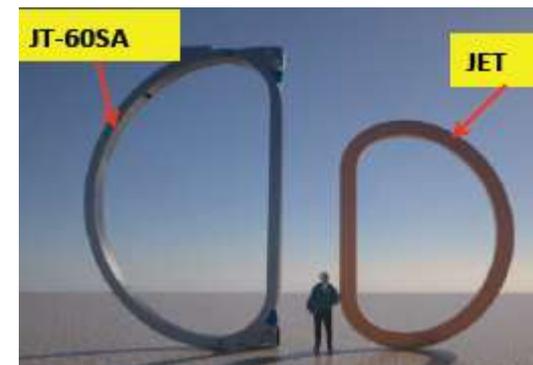
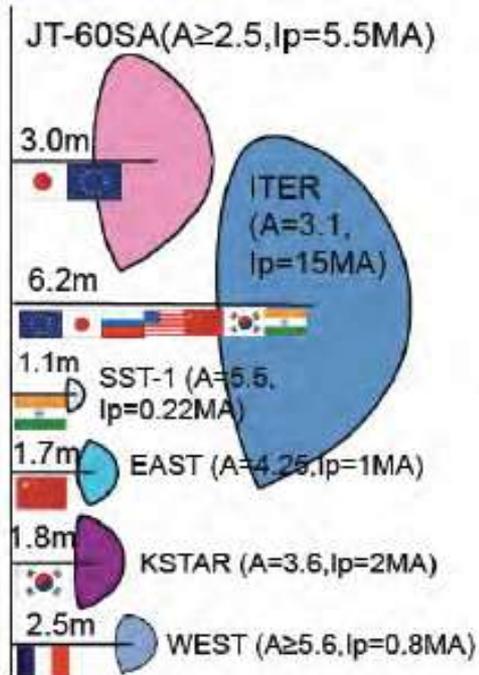
Goal of JT-60SA: 'Simultaneous & steady-state sustainment of the key performances required for DEMO' (= highly self regulating)

JT-60SA should decide the practically acceptable DEMO parameters, and develop & demonstrate a practical set of DEMO plasma controls.

We treat 'the DEMO regime' as a spectrum.

Plasma Current I_p	5.5MA
Toroidal Field B_t	2.25T
Major Radius R_p	2.97m
Minor Radius a_p	1.18m
Elongation κ_x	1.93
Triangularity δ_x	0.5
Safety Factor q_{95}	3
Plasma Volume V_p	133m ³
Flat Top	100 s

Superconducting tokamaks in the World



Программы УТС Партнёров. Гелиотроны (Япония - Япон)

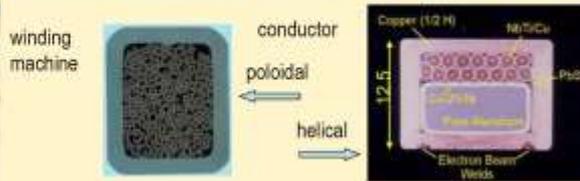
Stellarators: LHD Large Helical Device

LHD Toki Japan
operational 1998 – present
outer diameter = 13.5m
max field on coils = 6.6T

MT-15 Beijing China (1997)
Assembly of the Superconducting Coils & the Cryostat for the Large Helical Device
T.Satow S.Imagawa H.Tamura K.Takahata
N.Yanagi S.Yamada T.Mito A.Nishimura
NIFS Toki Japan
...two superconducting helical coils, 3 pairs of poloidal coils... a 13.5m dia. outside cylinder...



NIFS pictures from T.Mito



Martin Wilson slide36

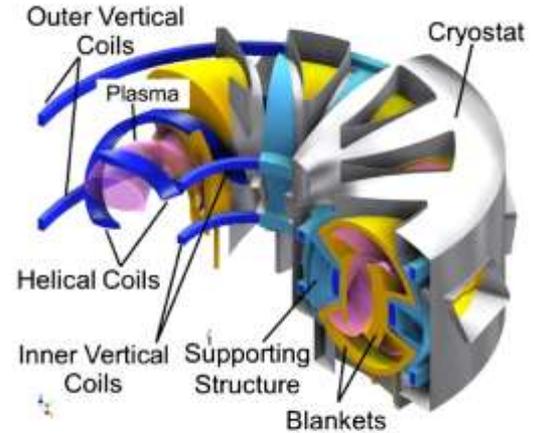


Fig. 1. Schematic illustration of the helical fusion reactor FFHR-d1.

Helical fusion reactor FFHR-d1



LHD-type Helical Fusion Reactor FFHR-d1

Major radius of Helical coil : 15.6 m
Center Magnetic Field of HC : 4.7 T
Maximum Magnetic Field : 12 T
Operating Current : 94 kA
Stored Magnetic Energy : ~160 GJ

Япония работает над реактором-гелиотроном (от LHD с полем ~ 3 Тл к FFHR с полем ~16 Тл на обмотке)

LTS option



Force-cooled Cable-in-conduit conductor

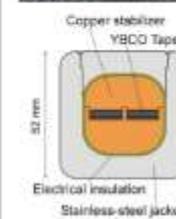


Indirect-cooled LTS conductor

HTS option



Helium gas cooled HTS conductor



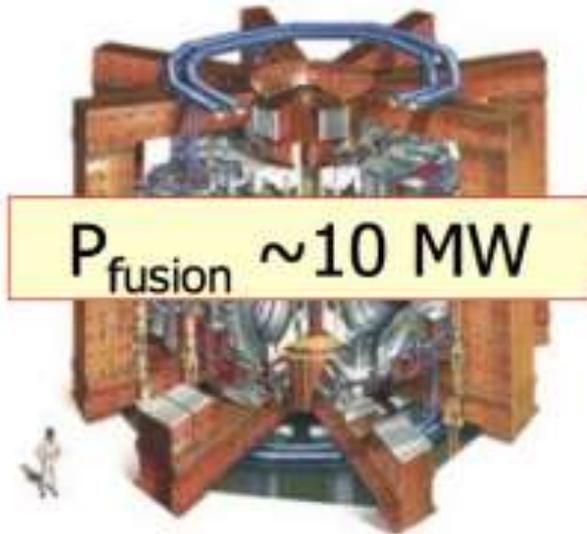
Major Specifications

Operation current 94 kA @ 12 T
Operation temp. 20 K
Conductor size 62 mm × 62 mm
Current density 24.5 A/mm²
Number of tapes 40
Cabling method Simple Stacking

ARC : Compact fusion reactor at high field

JET R ~ 3m

Copper, $B_0 = 3.5 \text{ T}$



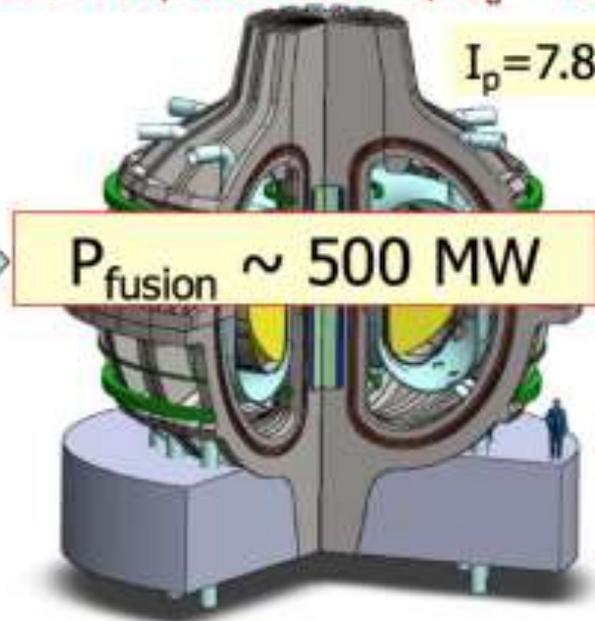
$P_{\text{fusion}} \sim 10 \text{ MW}$

$\times B^4$

ARC R ~ 3.2m

REBCO superconductor, $B_0 = 9.2 \text{ T}$

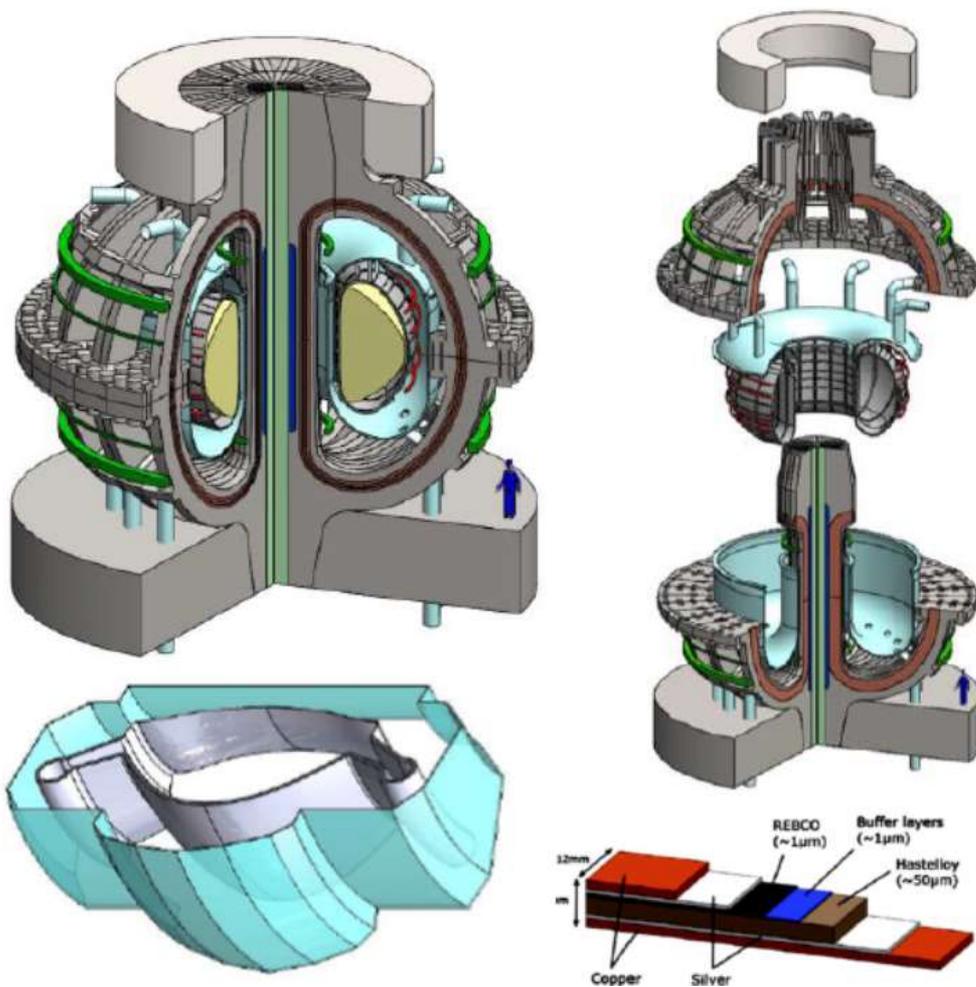
$I_p = 7.8 \text{ MA}$



$P_{\text{fusion}} \sim 500 \text{ MW}$

*B.N. Sorbom, et al., Fus. Eng. Des., 100(2015)378-405

ARC из пакетов ВТСП лент с разъёмными контактами



Design parameter	Symbol	Value
Fusion power	P_f	525 MW
Total thermal power	P_{tot}	708 MW
Plant thermal efficiency	η_{elec}	0.40
Total electric power	P_e	283 MW
Net electric power	P_{net}	190 MW
LHCD coupled power	P_{LH}	25 MW
ICRF coupled power	P_{IC}	13.6 MW
Power multiplication factor	Q_E	3.0
Major radius	R_0	3.3 m
Plasma semi-minor radius	a	1.13 m
Plasma elongation	κ	1.84
Plasma volume	V_p	141 m ³
Toroidal magnetic field	B_0	9.2 T
Peak on-coil magnetic field	B_{max}	23 T
Plasma current	I_p	7.8 MA
Bootstrap fraction	f_{BS}	0.63
Tritium breeding ratio	TBR	1.1
Avg. temperature	$\langle T \rangle$	14 keV
Avg. density	$\langle n \rangle$	$1.3 \times 10^{20} \text{ m}^{-3}$
On-axis temperature	T_0	27 keV
On-axis density	n_0	$1.8 \times 10^{20} \text{ m}^{-3}$
Greenwald fraction	f_{Gr}	0.67
Toroidal beta	β_T	1.9%
Internal inductance	l_i	0.67
Normalized beta	β_N	2.59
Safety factor at $r/a = 0.95$	q_{95}	7.2
Edge safety factor	q_a	4.7
Minimum safety factor	q_{min}	3.5
Fusion power wall loading	P_f/S_b	2.5 MW/m ²
Energy confinement time	τ_E	0.64 s
H89 confinement factor	H_{89}	2.8
H98(y,2) confinement factor	$H_{98,y2}$	1.8
G89 gain factor	G_{89}	0.14

Travelling wave antenna for DIII-D proposed by Victor Vdovin

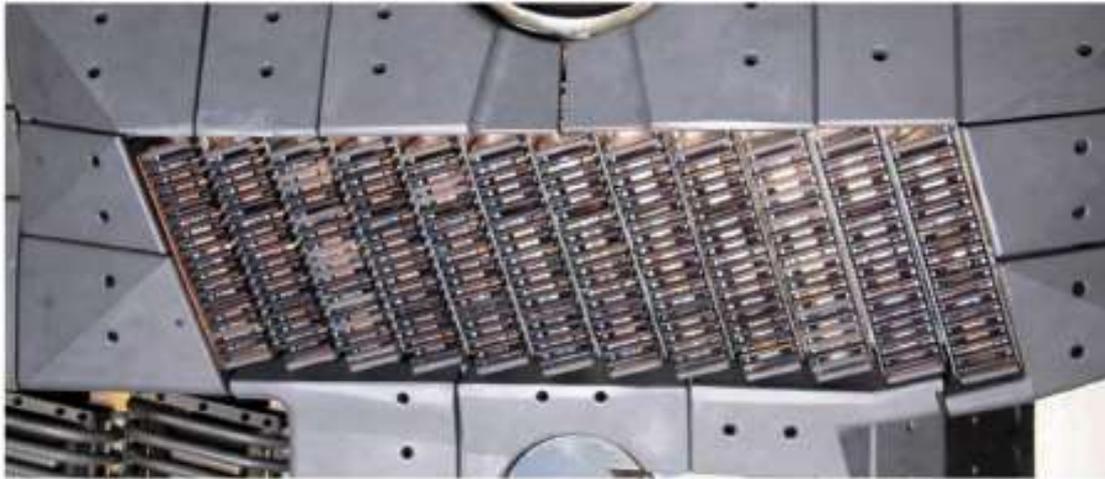
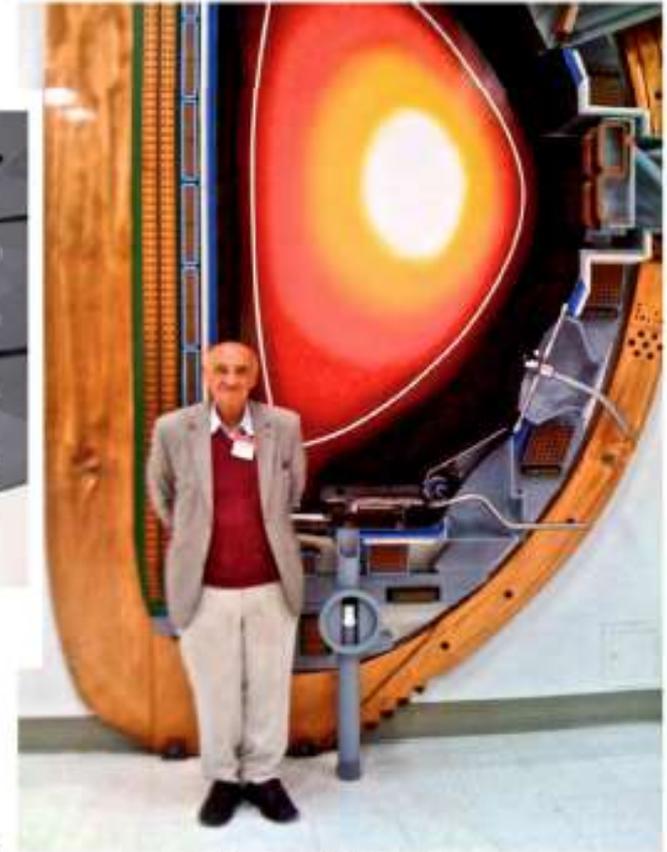


Fig. 1. Photograph of the 12-module low power antenna installed in DIII-D.

J.F. Tooker et al.,
“Development of a high power Helicon system for DIII-D”
Fusion Engineering and Design 123 (2017) 228–231



Victor Vdovin

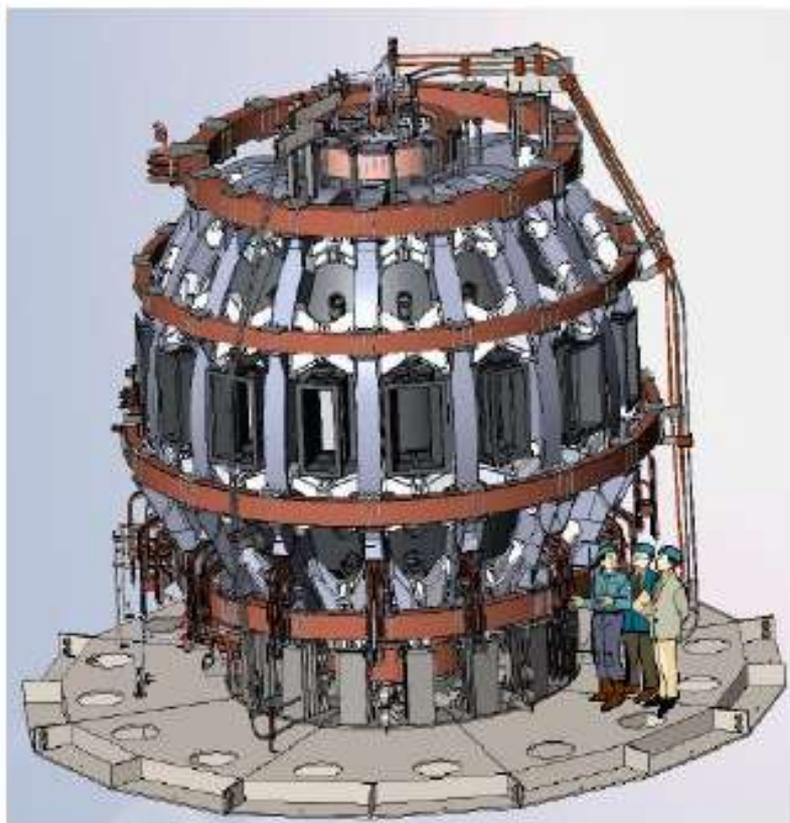
Программа УТС России.

(Т-15МД, Глобус-М2, Открытые ловушки, ТРТ, Гибрид, ГДМЛ ...)



Т – 15 МД

Основные параметры



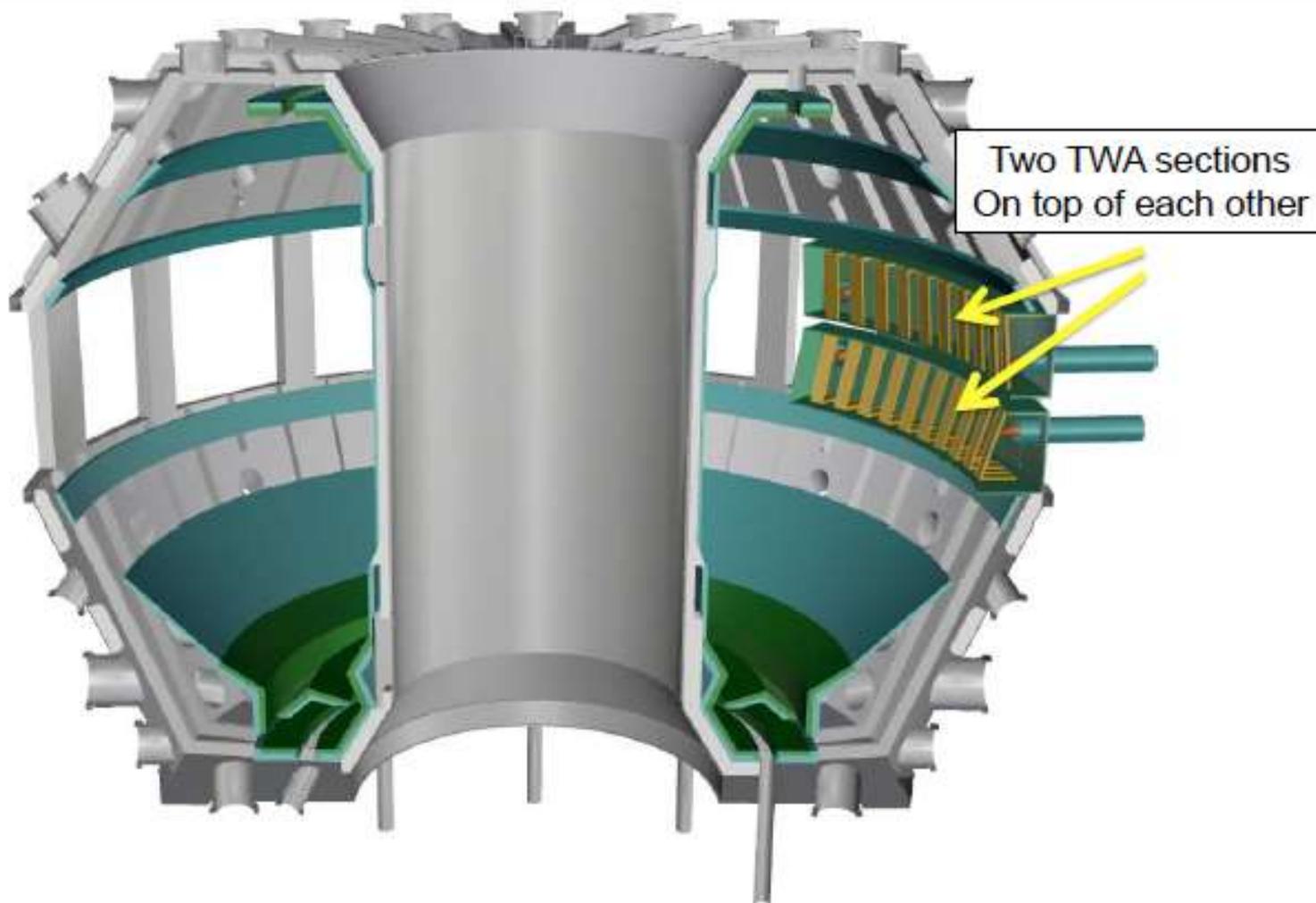
Большой радиус тора R, м	1.48
Аспектное отношение	2.2
Ток плазмы I_p , МА	2.0
Вытянутость, k	1.9
Конфигурация плазмы	SN,DN
Длительность импульса, сек	10
Торондальное магнитное поле, Т	2
Запас магнитного потока в центральном соленоиде, Вб	6
Мощность инжекции нейтралов, МВт	6
Мощность ЭЦР- нагрева, МВт	7
Мощность ИЦР- нагрева, МВт	6
Мощность НГ- нагрева, МВт	4

Литиевая первая стенка, дивертор, вытянутое сечение, квазистационарная генерация неиндуктивного тока геликонами

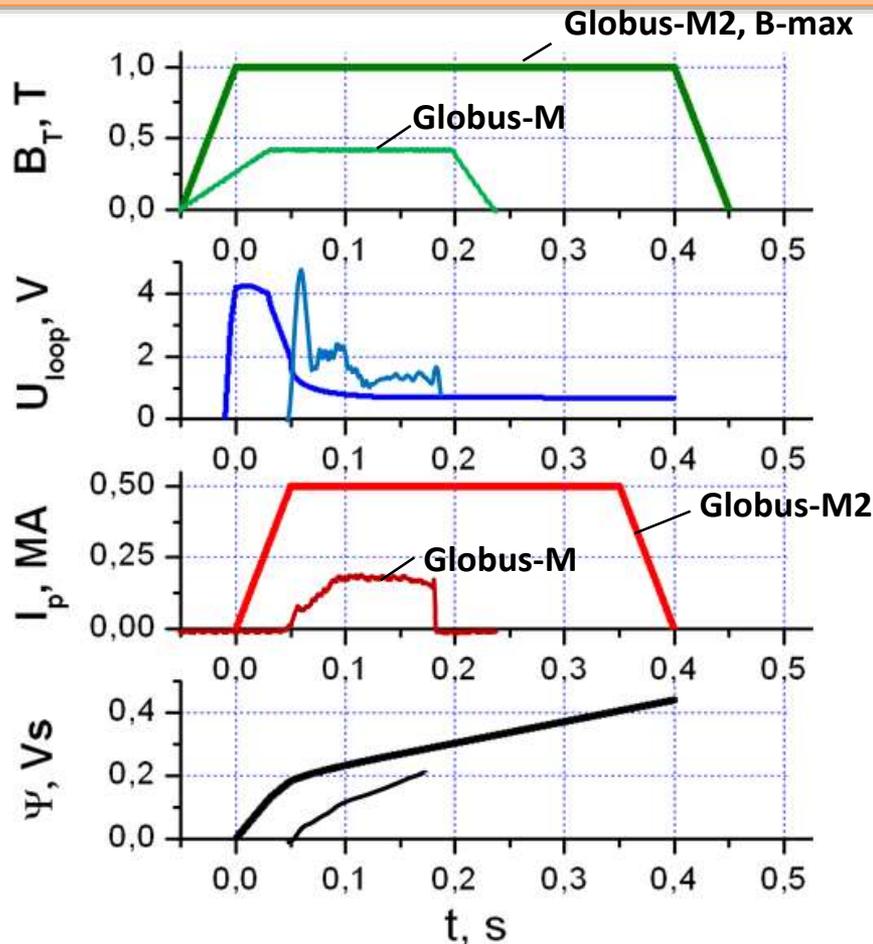
Электромагнитная система и вакуумная камера Изготовление завершено



Proposal for TWA in T15-MD: inside view



Концепция проекта Глобус - М2

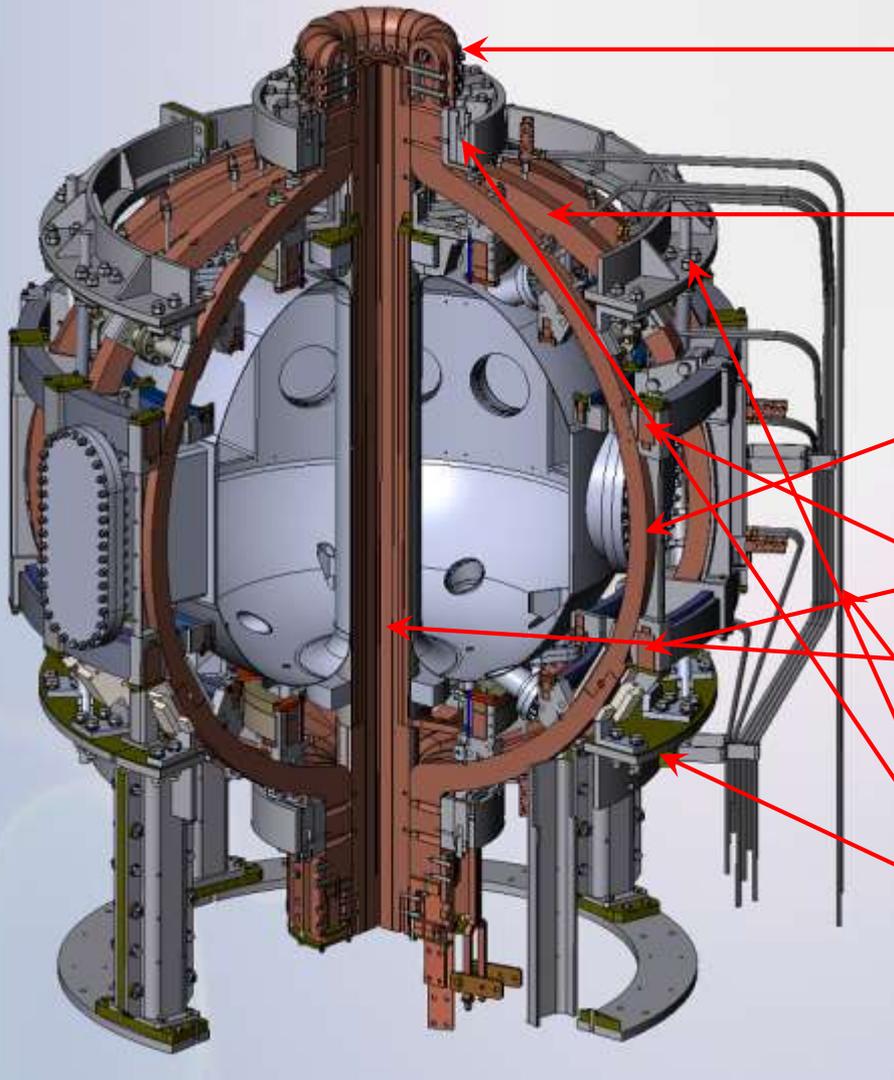


- ✓ увеличение тороидального магнитного поля до 1 Тл
- ✓ увеличение тока плазмы до 0.5 МА
- ✓ увеличение длительности разряда
- ✓ снижение гофрировки тороидального магнитного поля

- Завершена сборка сферического токамака Глобус-М2.
- Физический пуск установки запланирован на весну 2018.
- Полномасштабный плазменный эксперимент начнется до конца 2018.

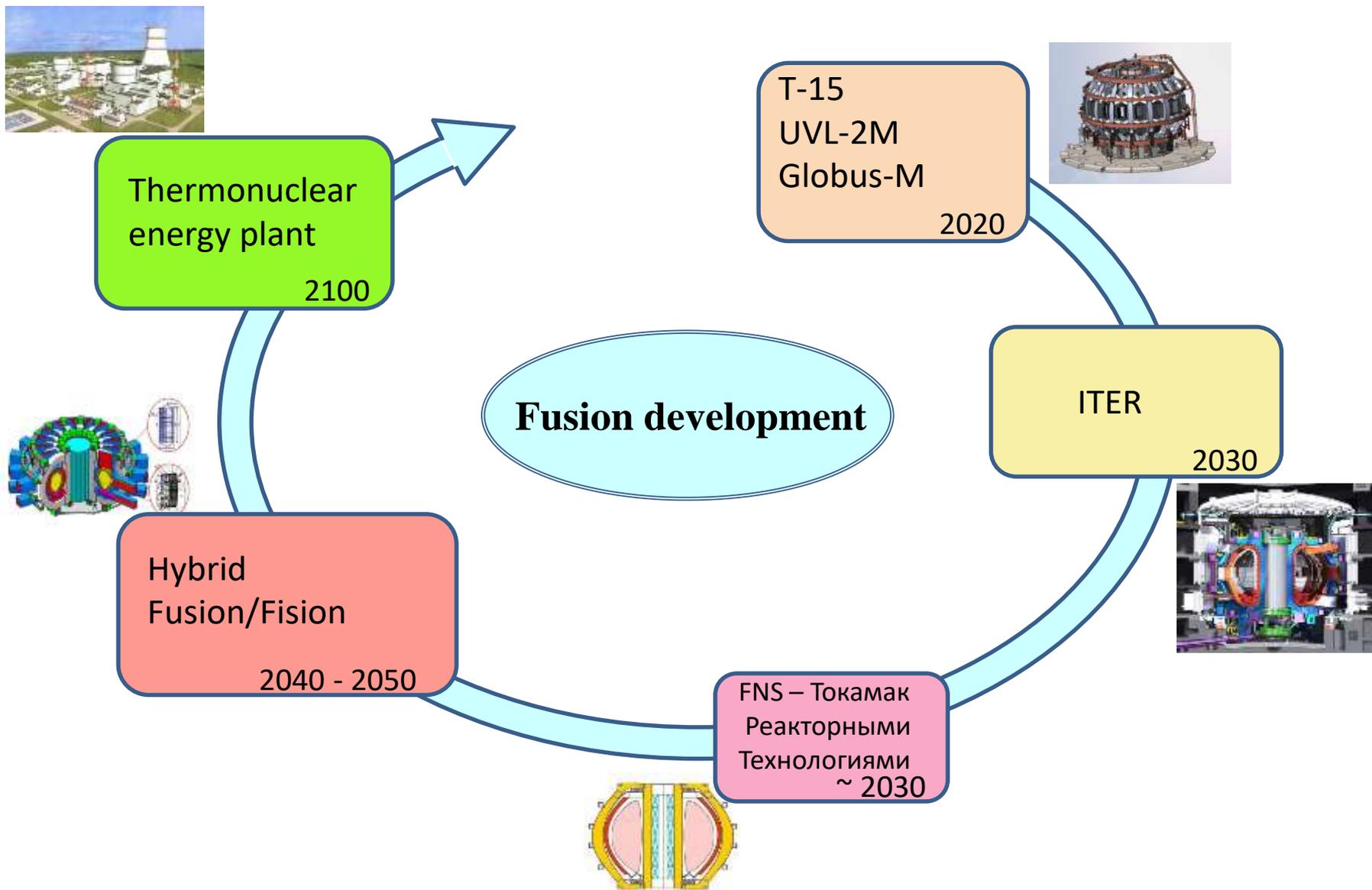
Конструкция ЭМС Глобус-М2

Конструкция Глобус-М2 отличается от Глобус-М рядом существенных деталей



- Гибкие токовые перемычки вынесены из зоны максимальных нагрузок
- Внешние полувитки ОТП изготовлены из высокопрочного медного сплава ($\sigma_{02} > 220 \text{ МПа}$)
- Радиус полувитков увеличен на 40 мм для снижения гофрировки ($< 0.4\%$)
- Конструкция основных ОУ изменена
- Конструкция центрального стержня и индуктора полностью изменена
- Полоидальные ОУ выполнены охлаждаемыми
- Конструкция силовой структуры изменена

[V.B. Minaev, et al., FEC 2014]



Основные компоненты ТРТ, которые в настоящее время могут быть реализованы кооперацией российских предприятий

Компонент проекта	Особенности	Кооперация создателей
Проект (Размеры, магнитное поле, ток)	$R_0 = 3.2 \text{ м}$, $a \sim 1 \text{ м}$ $B_T = 5 - 6 \text{ Тл}$, $I_p > 5 \text{ МА}$	НИЦ «КИ», ТРИНИТИ, НИИЭФА, НИКИЭТ
Электромагнитная система	Сверхпроводящая: Nb3Sn, NbTi, с компонентами ВТСП	АО «ТВЭЛ», АО «ЧМЗ», ВНИИКП, ВНИИНМ, НИЦ «КИ», НИИЭФА
Первая стенка	Be / Li	НИИЭФА, НИКИЭТ / ТРИНИТИ, «Красная звезда»
Дивертор	Be / Li	НИИЭФА, НИКИЭТ / ТРИНИТИ, «Красная звезда»
Дополнительный нагрев плазмы	~40 МВт (Инжекция атомов, ЭЦН (170 ГГц, 1,5 МВт, 1000 с))	ИЯФ СО РАН, ИПФ РАН
Генерация стационарного тока	~ 20-30 МВт (ИН, Геликоны)	ИЯФ СО РАН, НИЦ «КИ», ИОФ РАН, ФТИ РАН
Экспериментальный бридерный blanket	TBR ~ 1	НИЦ «Курчатовский институт», НИИЭФА, НИКИЭТ
Диагностика плазмы	Совместимость с термоядерным реактором	НИЦ «КИ», ФТИ РАН, ТРИНИТИ, ИЯФ СО РАН и др.
Дистанционное обслуживание	Замена первой стенки, blanketа, дивертора	?

International Collaborations

Europe

W7-X  

JET 

MAST-U 

TCV 

IAEA
International Atomic Energy Agency
Atoms for Peace

ITER 

IPP Max-Planck-Institut für Plasmaphysik 

Asia

JT-60SA   

K-STAR  

LHD  

SST-1 

EAST  

IFPP INTERNATIONAL FUSION PROMOTION SOCIETY 

Summary

- **ITER – Technological Platform for Fusion Energy and Creator of World Fusion Industry.**

Россия полноценно в этом участвует (25 систем) и имеет право безвозмездного использования всех технологий, созданных в Проекте ИТЭР, для своей программы УТС.

- **In general ITER is carried out according the Schedule with First Plasma in 2025.**

Россия лидер в части соблюдения графика сооружения ИТЭР

- **DT experiment commissioning is planning for 2035.**

Параллельно России необходимо развернуть внутреннюю программу УТС, которая дополнит технологическую платформу ИТЭР (Li первая стенка, стационарная генерация тока, ВТСП, бланкет, гибрид и др.)

- **Key issues of ITER physics Program are solving *in collaboration with international fusion community.***

России необходимо развивать участие в крупнейших термоядерных экспериментах у партнёров (JET, JT-60SA, LHD, W-7X)

- **Next step – DEMO**

Все партнёры по ИТЭР ведут программы по УТС, разрабатывают варианты DEMO.