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Roadmap to a heliotron reactor

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1. A roadmap to tokamak reactor
2. Original R&Ds for heliotron reactor
3. A roadmap to a heliotron reactor
4. Summary

Contents

(1) Introduce recent roadmaps and R&Ds plans for tokamak reactors.

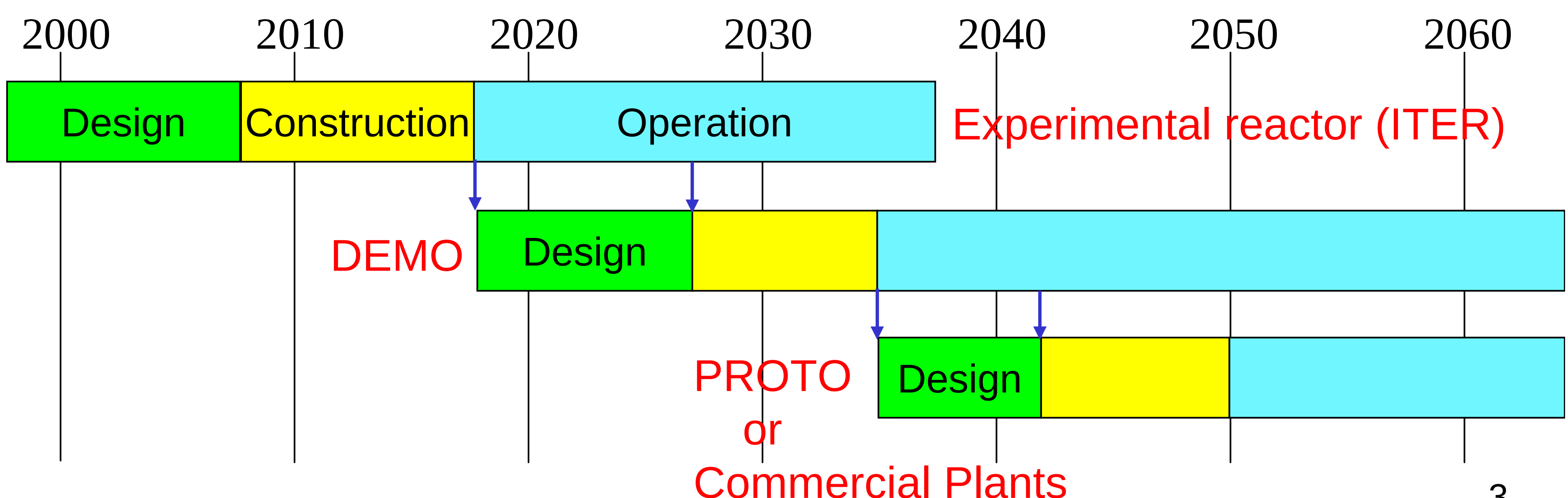
(2) Propose a heliotron DEMO concept and discuss the indispensable R&Ds in addition to those for tokamak reactors.

(3) Propose a roadmap to a heliotron reactor.

(4) Summarize

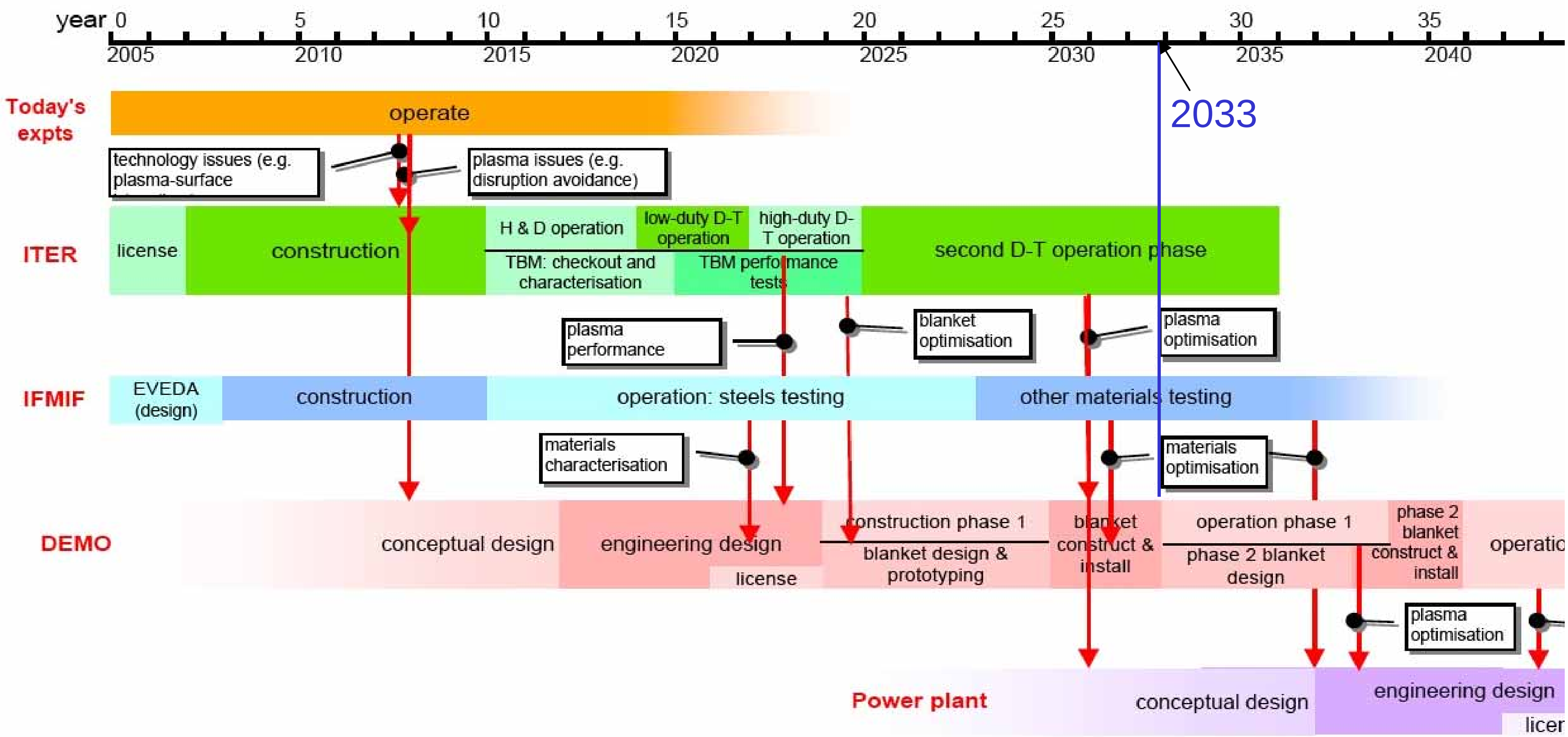
What are Fusion DEMOs?

- ◆Fusion energy should be realized in the second half of this century to contribute to reduction of CO_2 and the energy crisis.
- ◆DEMOs (demonstration plants for electricity production) are expected to generate electric power at a level of several hundreds MWs.
- ◆DEMO plants should prove the capability of reliable operation of fusion power plants (Tritium production is indispensable for the generation).
- ◆The first DEMO reactor should start operation in 2030s to realize commercial use of fusion energy in the second half of this century.



‘Fast track’ Concept in EU

The concept of the "fast track" to fusion power, should ITER continue to demonstrate that the tokamak line of magnetic confinement is the most promising for power generation, has recently been further elaborated in studies at EFDA and at Culham Laboratory. This approach is designed to speed up the development to the extent possible. (http://www.iter.org/a/index_nav_1.htm)

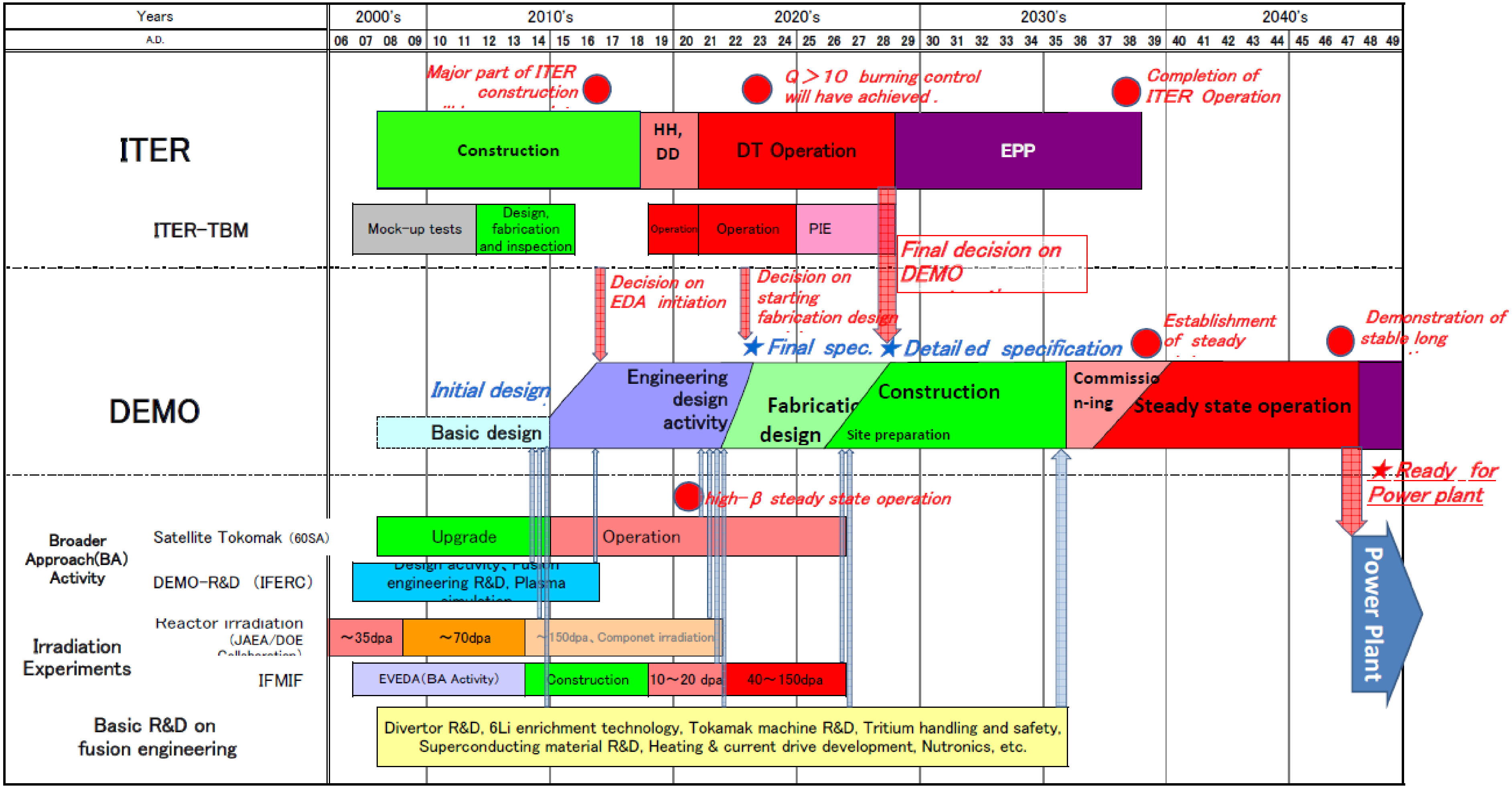


Design Study of Tokamak DEMOs in JA

Reactor design	SlimCS	Demo-CREST	ITER
Fusion power [GW_{th}]	3	1-3	0.5
Energy gain ($Q=P_{DT}/P_{ext.}$)	> 20	> 20	10
Operation mode	Steady state	Steady state	> 400 s
Bootstrap current fraction		0.5-0.73	
TBR	> 1.0	> 1.0	–
Major radius (R_p) [m]	5.5	7.3	6.2
Aspect ratio (A)	2.6	3.4	3.1
Plasma volume (m^3)	~800	~1100	~800
Normalized beta (β_N)	4.3	> 3.4	1.8
Plasma current (Ip) [MA]	16.7	14.7	15
Central field (B_0) [T]	6	7.8	5.3
Max. field (B_{max}) [T]	16.4	16	11.8
External heating, $P_{ext.}$ [MW]	~150	~150	50-73
Divertor heat load [MW/m^2]	10	10	10
Neutron wall load [MW/m^2]	3.2		0.8

(note) TBR: Tritium Breeding Ratio

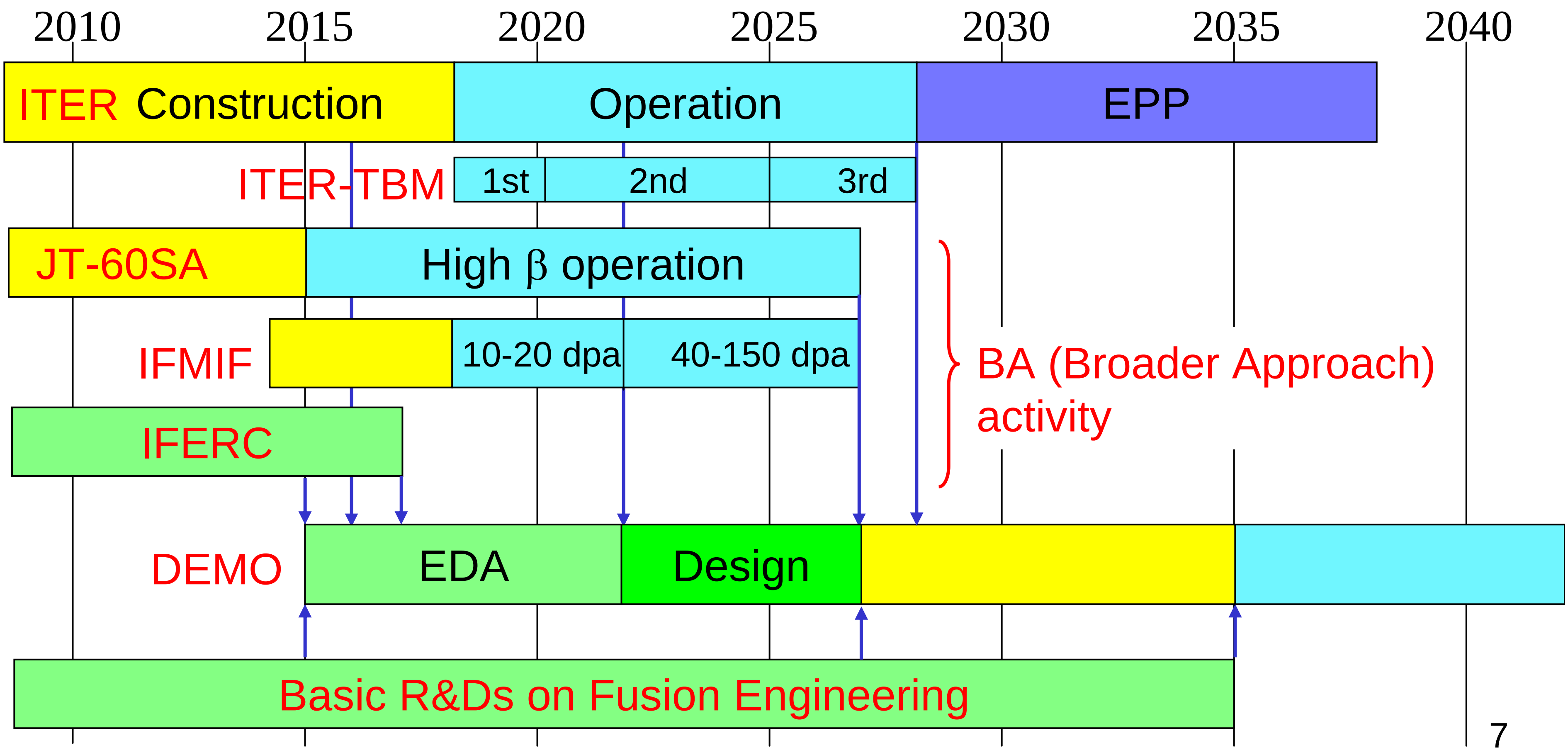
Roadmap to Tokamak DEMO in JA



This roadmap has been studied and presented by **Fusion Energy Forum of Japan** in response of Ministry of Education, Culture, Sports, Science and Technology-Japan, which asked the Forum how to develop Tokamak type DEMO reactor as a case study. The related reports are also available from the following Web site (in Japanese): http://www.naka.jaea.go.jp/fusion-energy-forum/

Roadmap to Tokamak DEMO in JA

- ITER and BA are essential for the first tokamak DEMO.
- In order to utilize the results of D-T operation in ITER, the final design of the first tokamak DEMO will be determined in the middle of 2020s.
- Basic R&Ds in addition to ITER and BA are proposed.



R&Ds for Tokamak DEMOs

(A) ITER : Integration of fusion technology, D-T burning plasma

(B) BA (Broader Approach) by JA and EU

(1) JT-60SA : High β operation for economical reactor, D-D plasma

(2) IFMIF : Material properties in high neutron dose (40-150 dpa)

(3) IFERC : Design activities, Blanket material, Tritium technology, Plasma simulator, Remote operation

(C) Basic R&Ds by JA with international collaboration

(1) Blanket technology : Tritium breeding ratio, neutron damage

(2) Superconducting material R&Ds: Nb_3Al for higher field (16 T)

(3) Divertor R&Ds : High heat flux, neutron damage, T retention

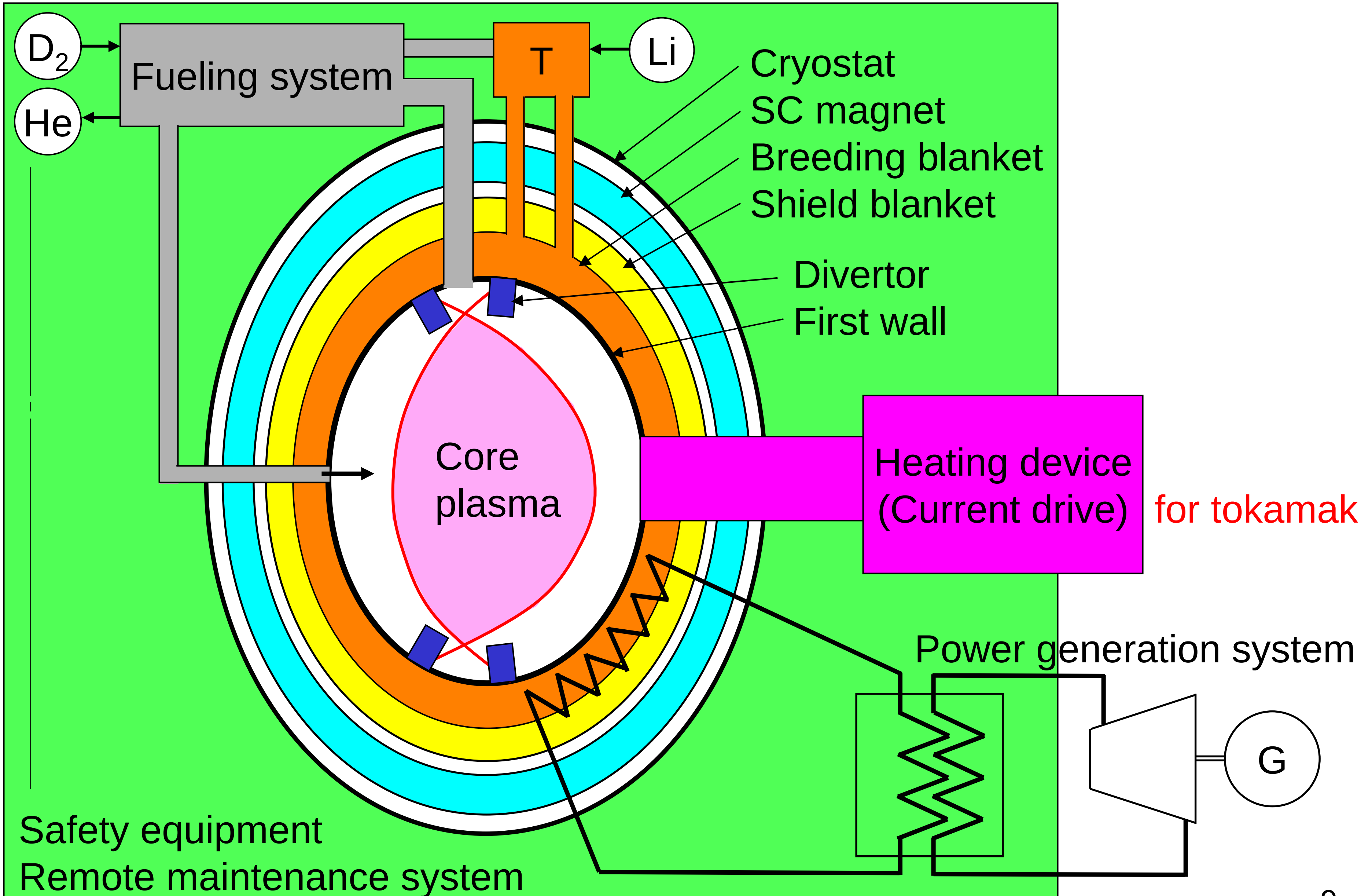
(4) Li^6 enrichment technology

(5) Tritium handling and safety

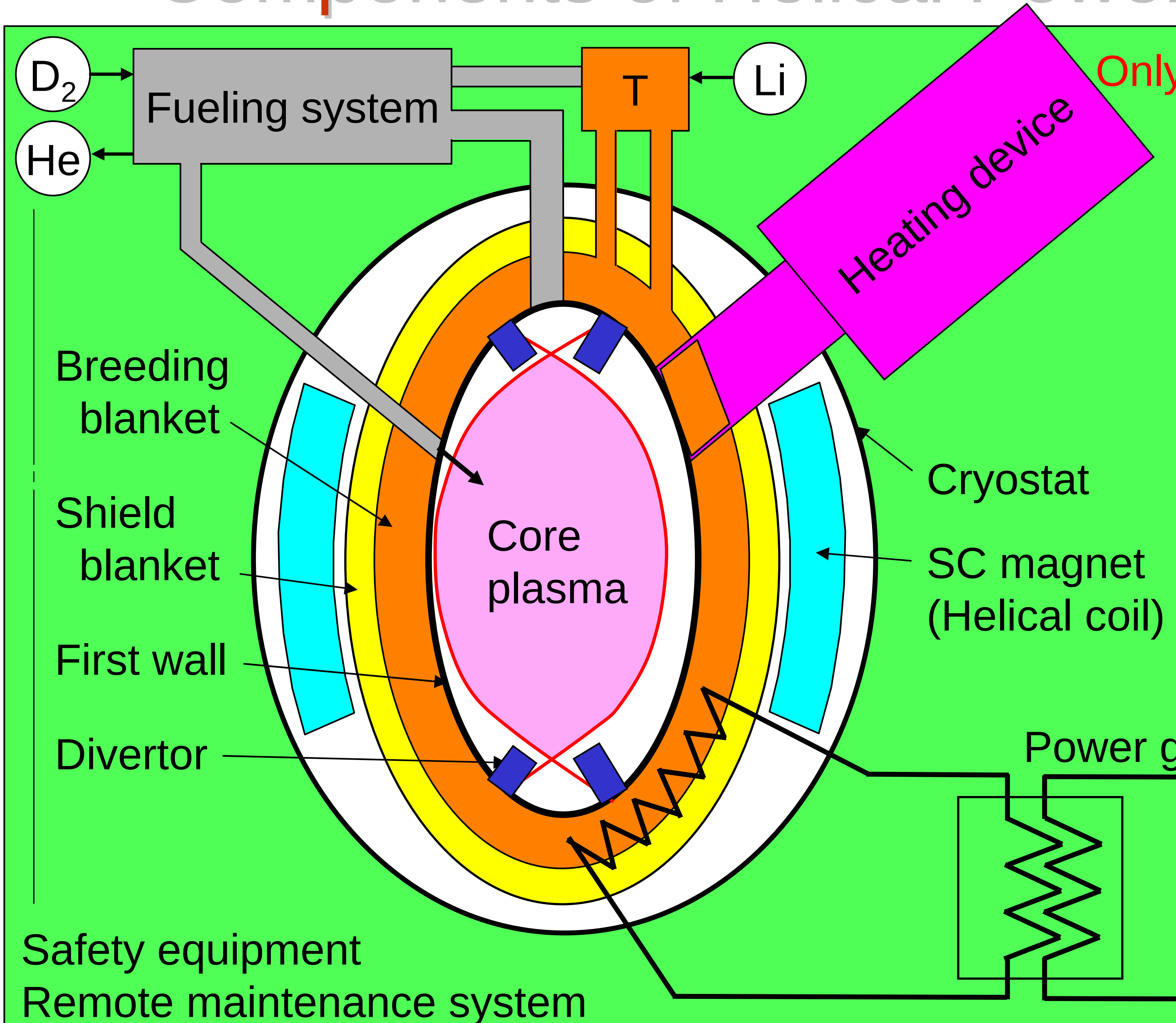
(6) Remote handling and maintenance

(7) Heating and current drive technology : Steady-state operation, high η

Components of Tokamak Power Plant



Components of Helical Power Plant



- Current drive is not necessary.
- The magnet configuration is different from a tokamak.
- The other components are the same as a tokamak.

Features of Heliotron Reactors

○ No current-disruptions (owing to inherently net-current less plasma)

○ No need for current-drive → Self burning operation is possible.

○ Wide space between helical coils for maintenance of blankets

△ Necessarily large major radius to realize the self-ignition condition with sufficient space for blankets. However, neutron wall load is low.

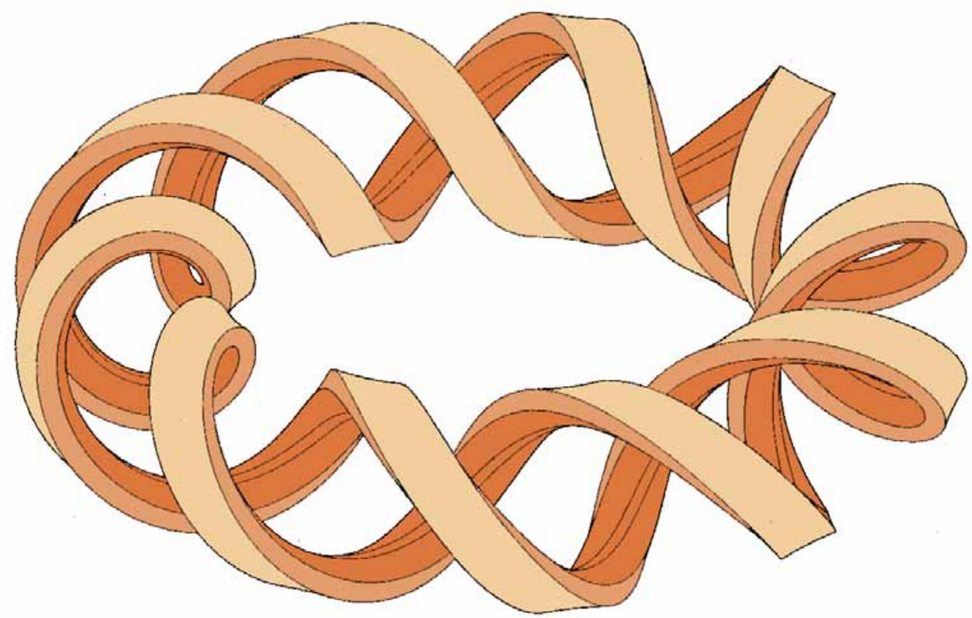
× Difficulty of replacement of HCs

? Plasma confinement at reactor regime

○ High density operation

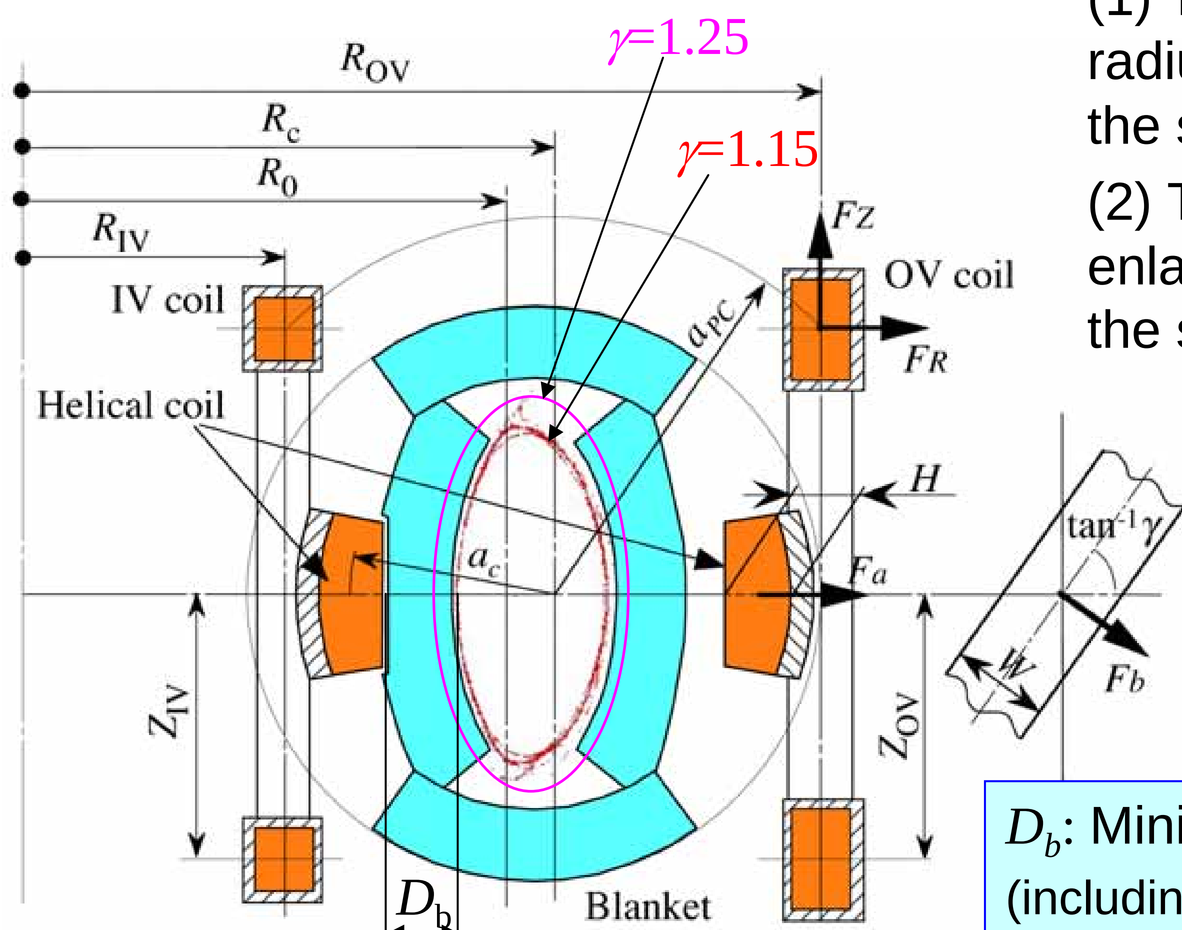
–> Good plasma confinement must be studied with LHD, theory, and simulation research.

–> SC magnets,especially the helical coils (HCs), must be simplified for high reliability, good maintainability of blankets, and reasonable construction cost.



Layout of Magnets of Heliotron

One set of poloidal coils (PCs) is necessary to adjust the major radius of the plasma, the quadruple field, and the stray field. Two sets of PCs are preferred because of the axis control and the lower stored magnetic energy.



(1) The relative plasma minor radius is the shorter with with the smaller pitch parameter γ.

(2) The space for blanket is enlarged with the smaller γ for the same major radius.

$$\gamma = \frac{m}{l}\frac{a_c}{R_c}$$

D_b: Minimum space for blanket (including the ergodic layer ~0.1 m)

Plasma Parameter of Heliotron Reactors

<Scaling law of energy confinement time>

ISS04: $\tau_E^{ISS04} = 0.134 a_p^{2.28} R_0^{0.64} P^{-0.61} \overline{n}_e^{0.54} B_0^{0.84} \iota_{2/3}^{0.41}$

- P : Plasma heating power per unit volume
- n_e : Average electron density
- B_t : Central toroidal magnetic field
- ι : rotational transform

Under the condition that τ_E, P, n_e, and ι are constant,

$$\tau_E^{ISS04} \propto a_p^{2.28} R_0^{0.64} \left(a_p^2 R_0\right)^{-0.61} B_0^{0.84} \propto a_p^{1.06} R_0^{0.03} B_0^{0.84}$$

In the case of similar shape,

$B_0 \propto R^{-1.30}$ for constant τ_E

Magnetic energy $\propto B^2 R^3 \propto R^{0.4}$

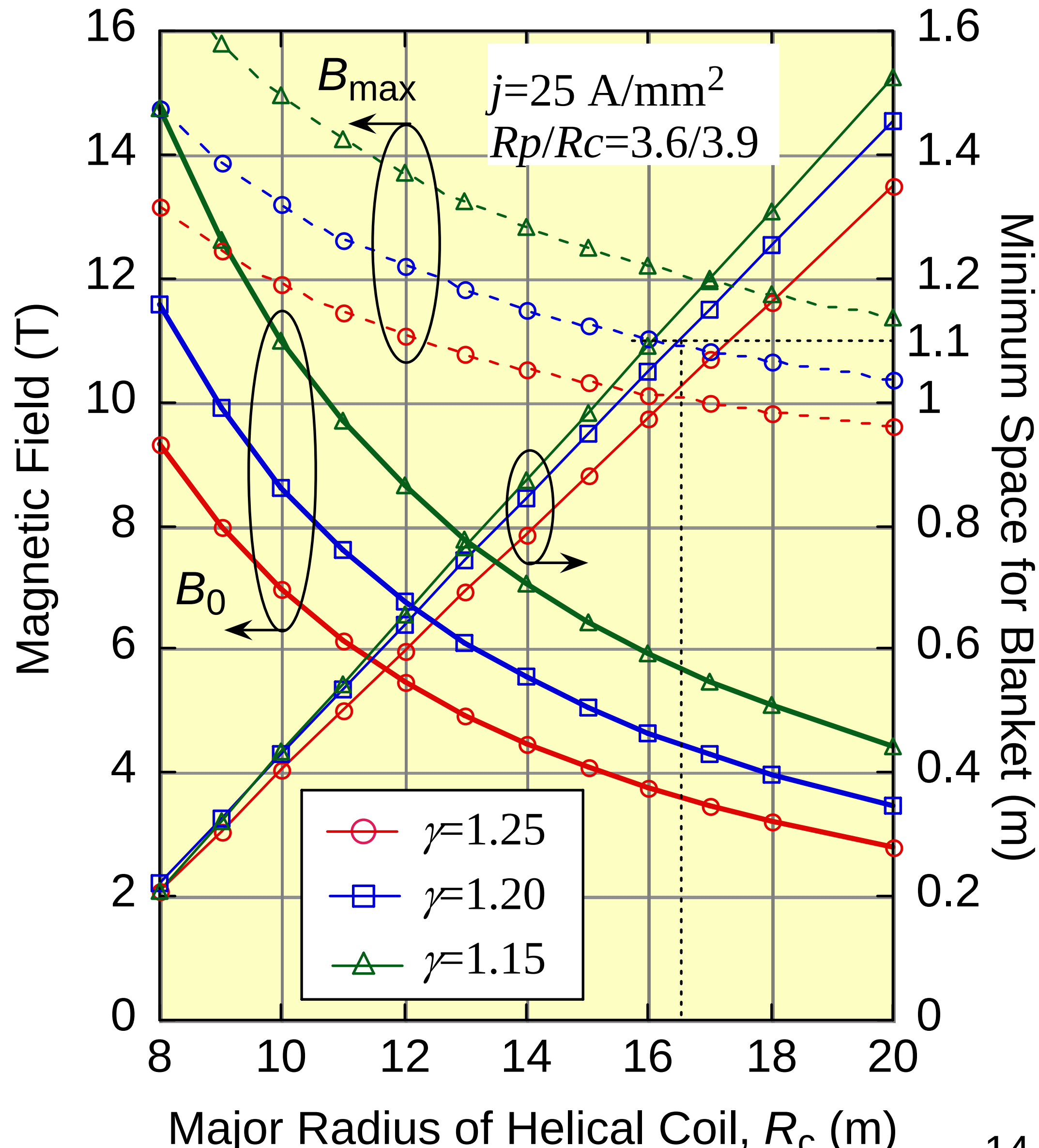
Required Field and Size for Self-ignition

<Plasma parameters>

H_H: 1.2 to LHD (1.12 to ISS04)
$n_e(0)$: 26.6×10^19 m^-3
$T_e(0)$: 15.9 keV
τ_E=const.
He ash ratio: 3%
Oxygen impurities ratio: 0.5%
α heating ratio: 90%
Self-ignition ($P_{ext.}$=0)

(1) B_{max} < 13 T at R_c> ~9 m. Then, the necessary R_0 is determined by the space for the blanket.

(2) The coil major radius of 16-17 m is needed to obtain a blanket space more than 1.1 m.

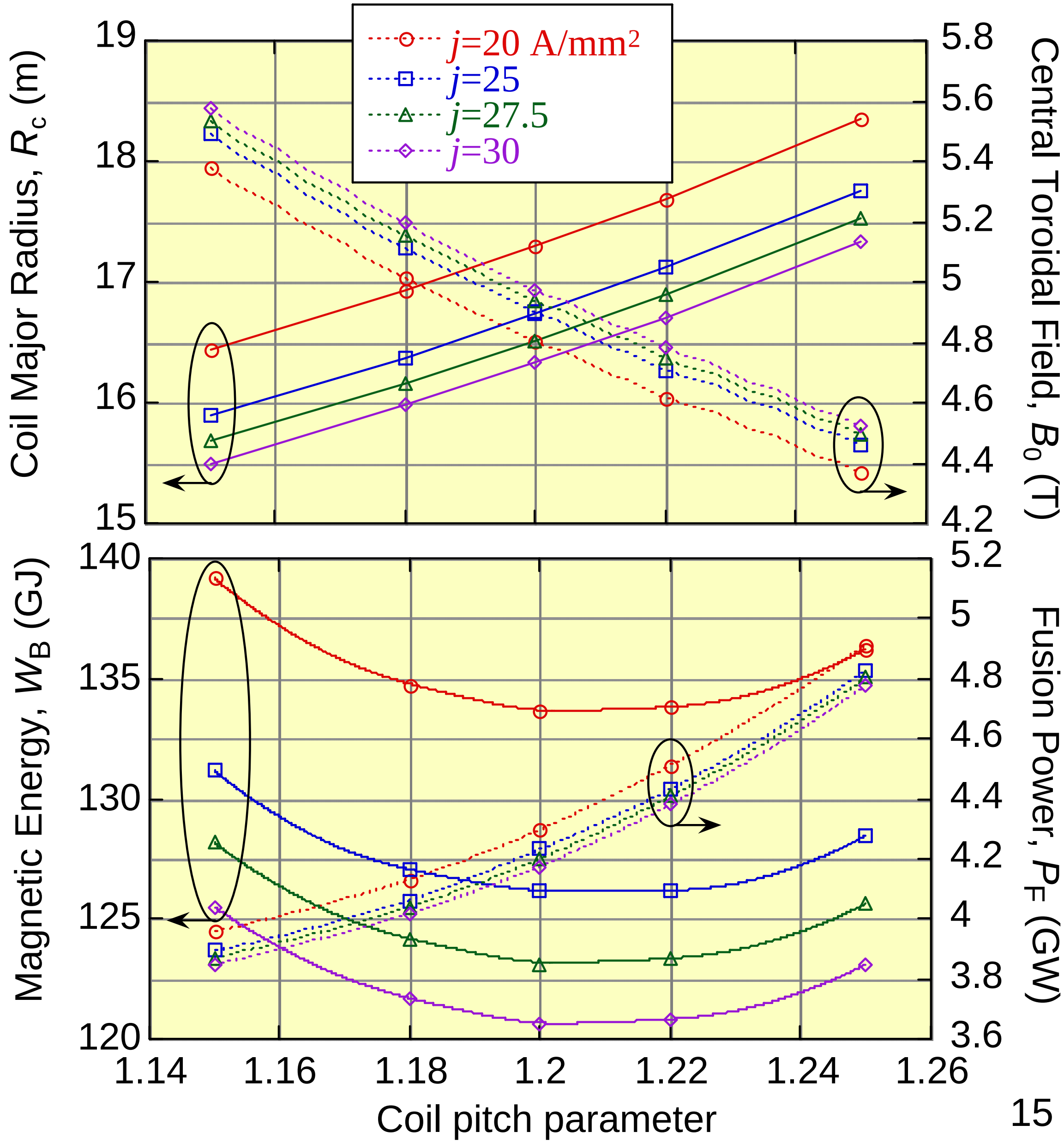


Effect of Pitch Parameter γ

<Design conditions>

(1) Blanket space : 1.1 m
(2) H-factor : 1.2 of LHD
(3) β = 5%
(4) $\langle n_e \rangle = 2.5(PB/R)^{0.5}/a_p$
He ash ratio: 3%
Oxygen impurities ratio: 0.5%
α heating ratio: 90%
Self-ignition ($P_{ext.}$=0)

- $P_F \propto \gamma^{2.2}$ (3.8-4.9 GW_{th})
- W_B = 120-140 GJ
(Change of W_B is small.)
- γ=1.20 is moderate.



Spec. of Heliotron DEMO

	H-Power plant	H-Demo	ITER
H-factor of τ_E, H_H	1.2 to LHD	←	
Energy gain, Q	∞ (self ignition)	> 20 (∞ by H_H=1.5 x LHD)	
TBR	> 1.0	> 1.0	
Operation	Steady state	←	
Min. blanket space [m]	> 1.1	> 0.9	
Min. blanket thickness [m]	> 1.0	> 0.8	

The blanket thickness of DEMOs can be reduced, because the neutron fluence in their lifetime should be lower than commercial power plants.

Spec. of Heliotron DEMO

	H-Power plant	H-Demo	ITER
H-factor of τ_E, H_H	1.2 to LHD	←	
Energy gain, Q	∞ (self ignition)	> 20 (∞ by H_H=1.5 x LHD)	10
TBR	> 1.0	> 1.0	
Operation	Steady state	←	
Min. blanket space [m]	> 1.1	> 0.9	
Fusion power [GW_{th}]	4 - 5	~ 2	0.5
Major radius of plasma [m]	15-16	13.2	
Minor radius of plasma [m]	~ 2	1.9	
Central field [T]	~ 5	4.8	
Max. field [T]	11-12	10.6	13.5
Stored magnetic energy [GJ]	120-140	70-80	41 (TF)
Weight of magnets [tons]	~16,000	~10,000	~10,000
Construction costs	< 2 x ITER	< 1.5 x ITER	

Helical Coil with CICC (ITER technology)

<Design criteria of CIC (Cable-in-conduit) conductor [ITER-TF coil]>

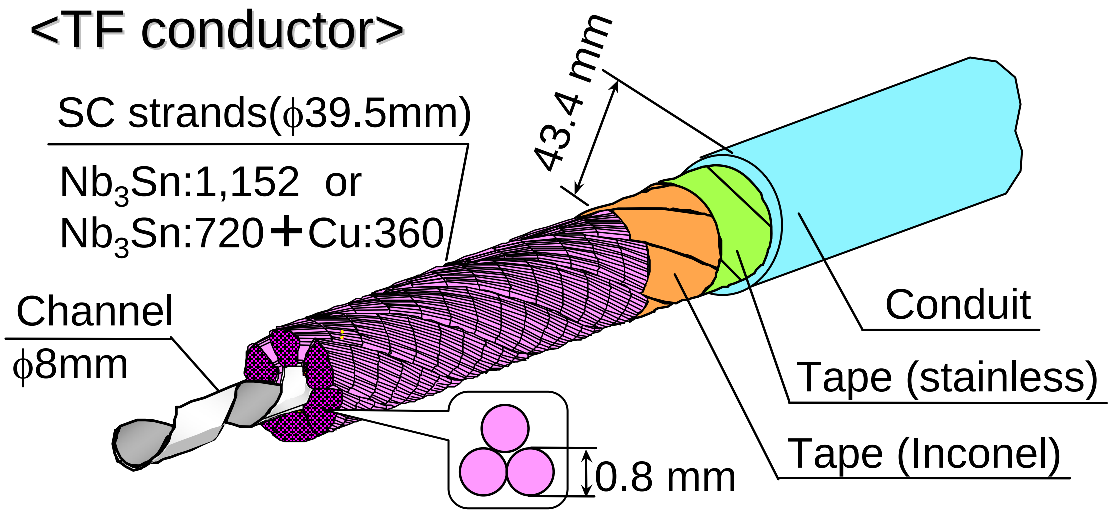
• Max. length of cooling path	< 550 m	[390 m]
• Current	< 100 kA	[68 kA]
• Max. field	< 13 T	[11.8 T]
• Coil current density	< 30 A/mm²	[20.3 A/mm²]

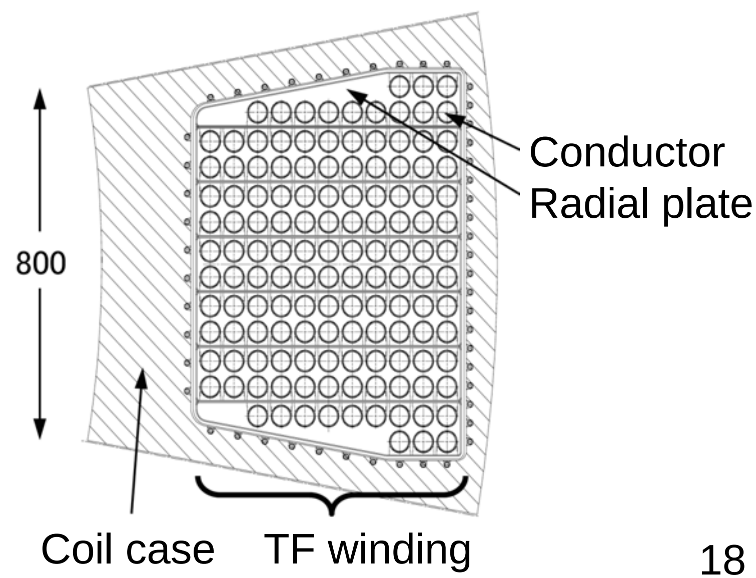
<Concept to adopt CIC conductor for HC (five times longer than ITER-TF)>

(1) To reduce a turn number with a large current conductor

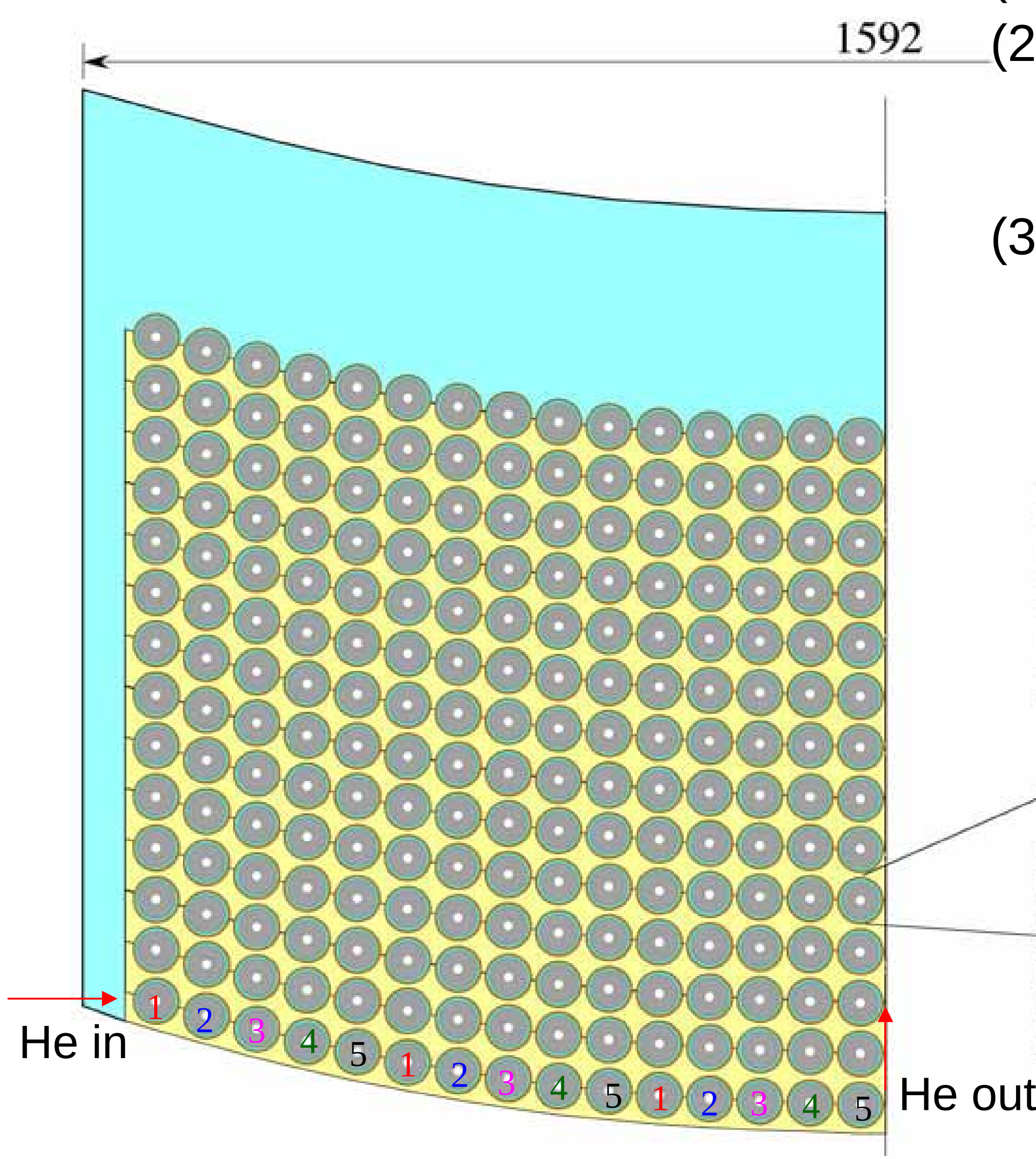
(2) To shorten the cooling length with five parallel winding

<TF conductor>





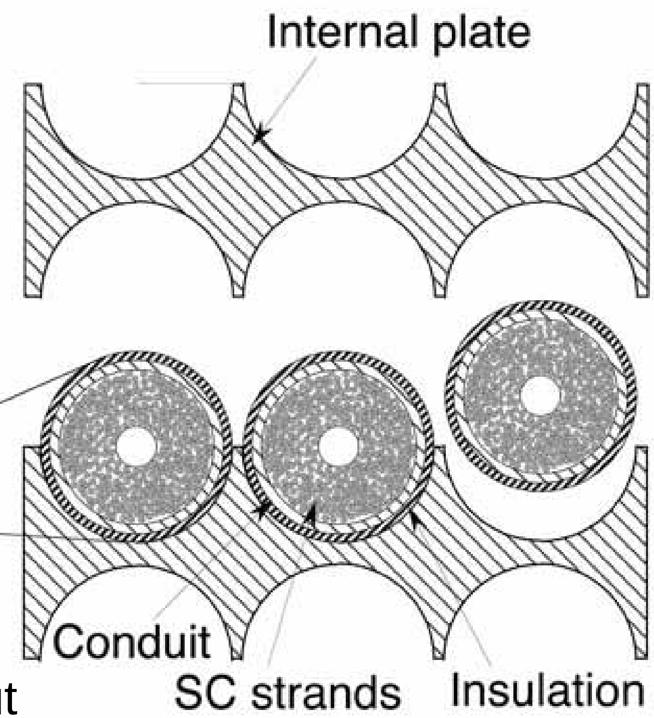
Concept of Helical Coil with CICC



(1) Layer winding by 5 in hand

(2) Winding guides are grooved in the internal plates whose sectors are welded on site.

(3) CIC conductors are wound into the groove with insulation.



HC with CICC for FFHR2m1

		H-Demo	H-power plant	ITER-TF
Bmax	T	10.63	11.6	11.8
Magnetic energy	GJ	78.43	136.5	41
Length of a cooling path	m	420.8	493	390
Conductor current	kA	74.8	90.2	68.0
Number of parallel winding		5	5	1
Current density (winding)	A/mm^2	25	25	20.33
SC materal		Nb_3Al	Nb_3Al	Nb_3Sn
Cu ratio of strand		1	1	1
Non-Cu current density	A/mm^2	400.0	400.0	273.4
Ratio of Cu strands (area)		0.452	0.452	0.360
Central tube diameter	mm	12.0	12.0	8.0
Void fraction		0.34	0.34	0.34
Conduit outer diameter	mm	42.2	45.6	43.4
Number of coils		2	2	18
Total length of conductor	km	117.8	138.0	82.2
Total weight of SC strands	ton	353	498	351
Total weight of Cu strands	ton	325	458	206

CIC conductors for a large helical coil can be produced with the same technology for ITER. Improvement of current density is preferred.

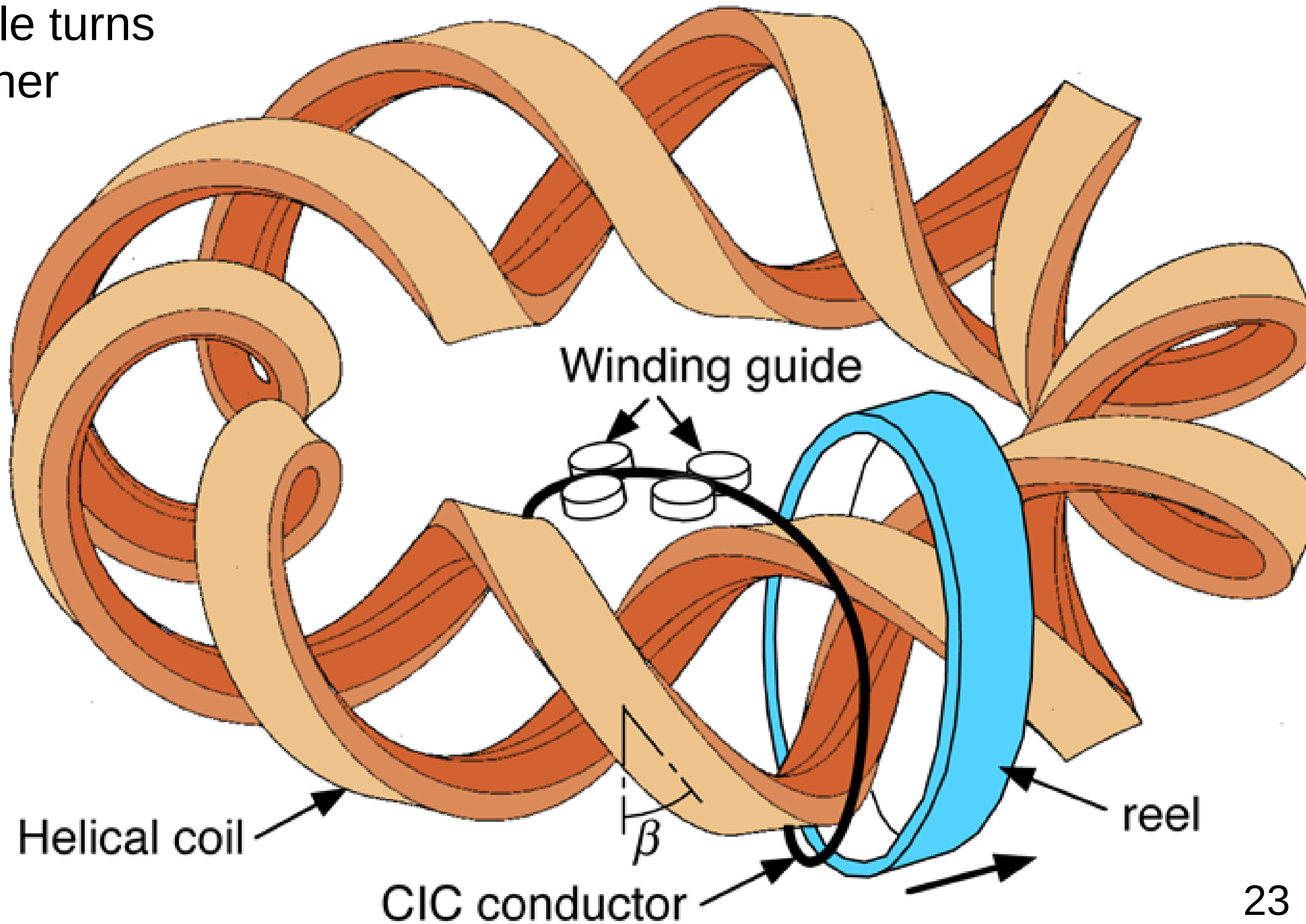
Concept of Winding Helical Coil

(1) Conductors are heated for reaction of Nb_3Al on a bobbin the circumference of which is same as the length of one pitch of the helical coil.
(2) The conductors are transferred to the reel of the winding machine.
(3) The conductors are pulled aside by the winding guide and wound in grooves of the inner plate with being wrapped with glass tapes.
(4) After winding the whole turns in a layer, the next inner plate are assembled.

<Shear strain by winding>

$$r\theta = \frac{r \cdot tan^{-1}\beta}{2\pi a_c / 4} \sim 0.3\% < 0.5\%$$

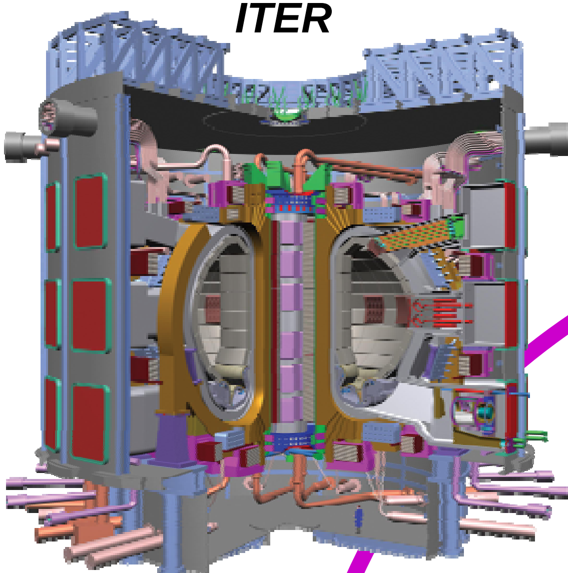
Although the ideal strain is allowable, the helical winding with CIC conductors must be demonstrated.



Roadmap to Heliotron DEMO

Tokamak Experimental Rector

ITER



R&Ds on SC magnets, blankets, and divertor.

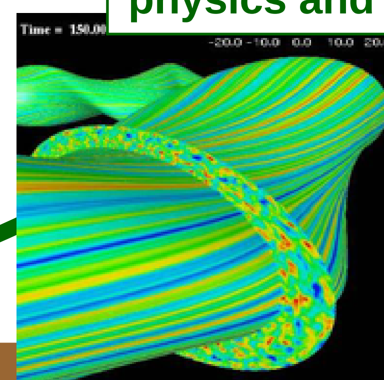
Heliotron Demo

Physics of burning plasmas

Demonstration of steady-state, high-density, high beta by net-current free plasma

Multi-layer models covering physics and engineering

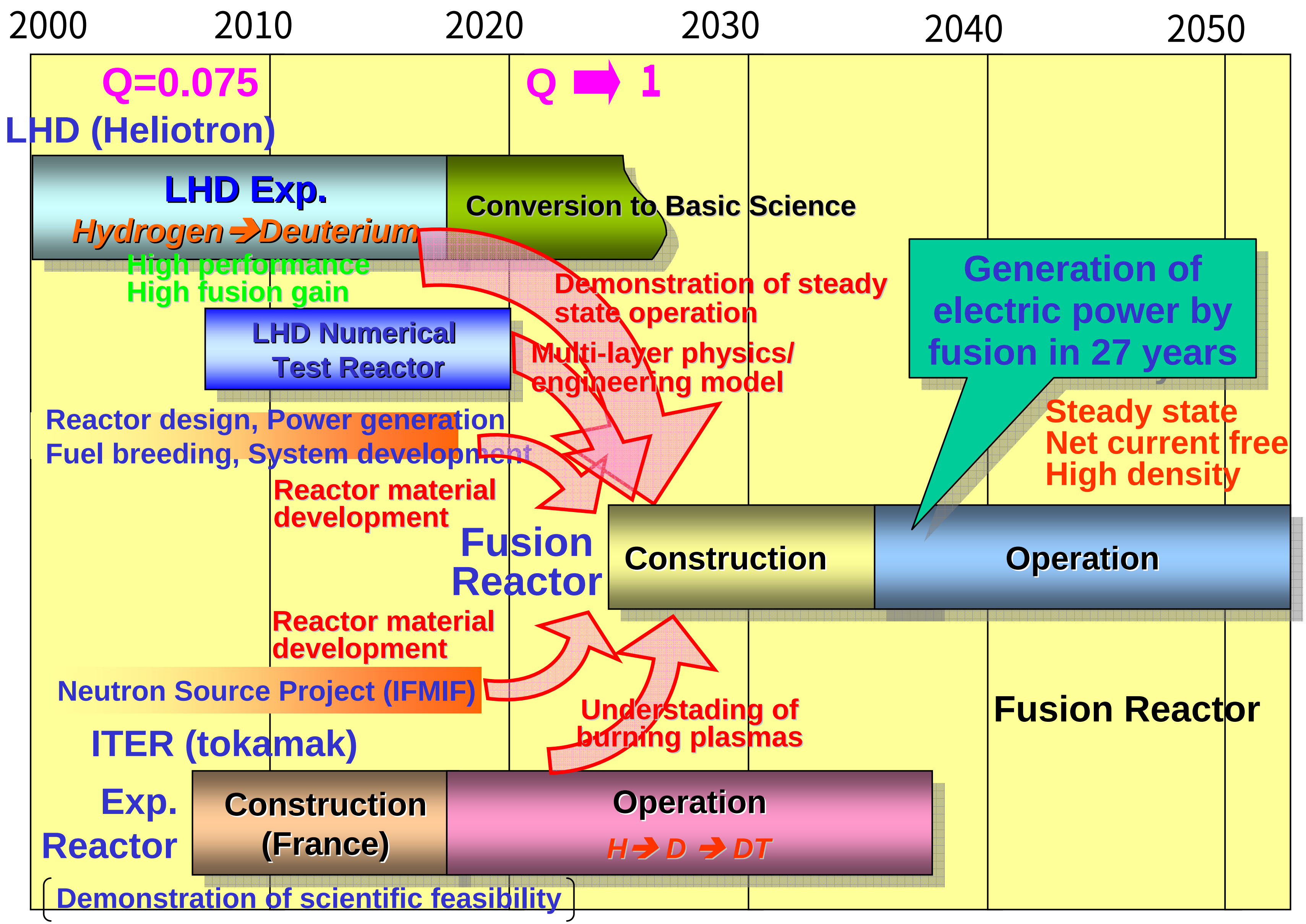
LHD



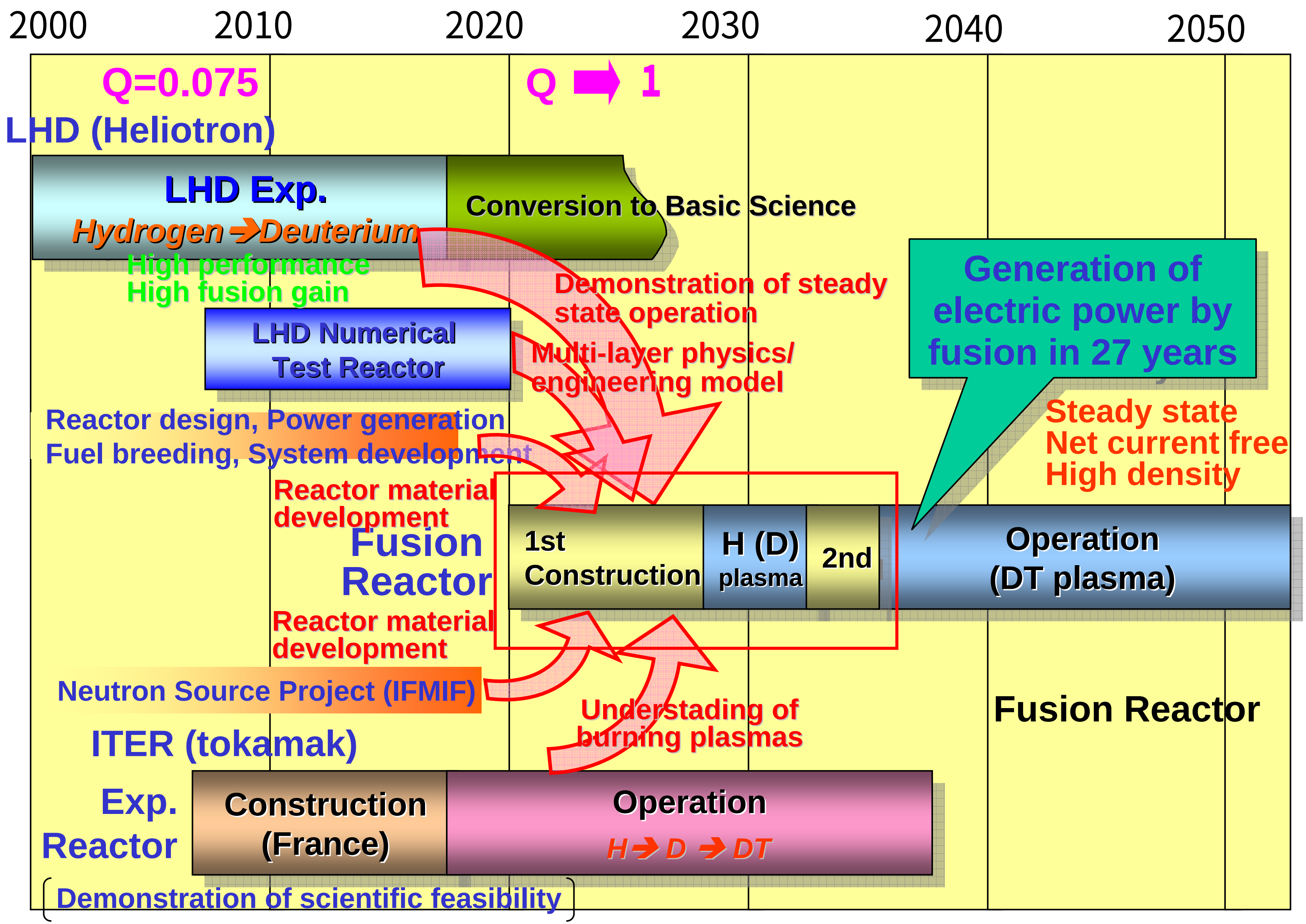
<u>LHD-NT</u>
LHD Numerical Test Reactor

Basic Science

Aiming at Helical Fusion Reactor (Original)

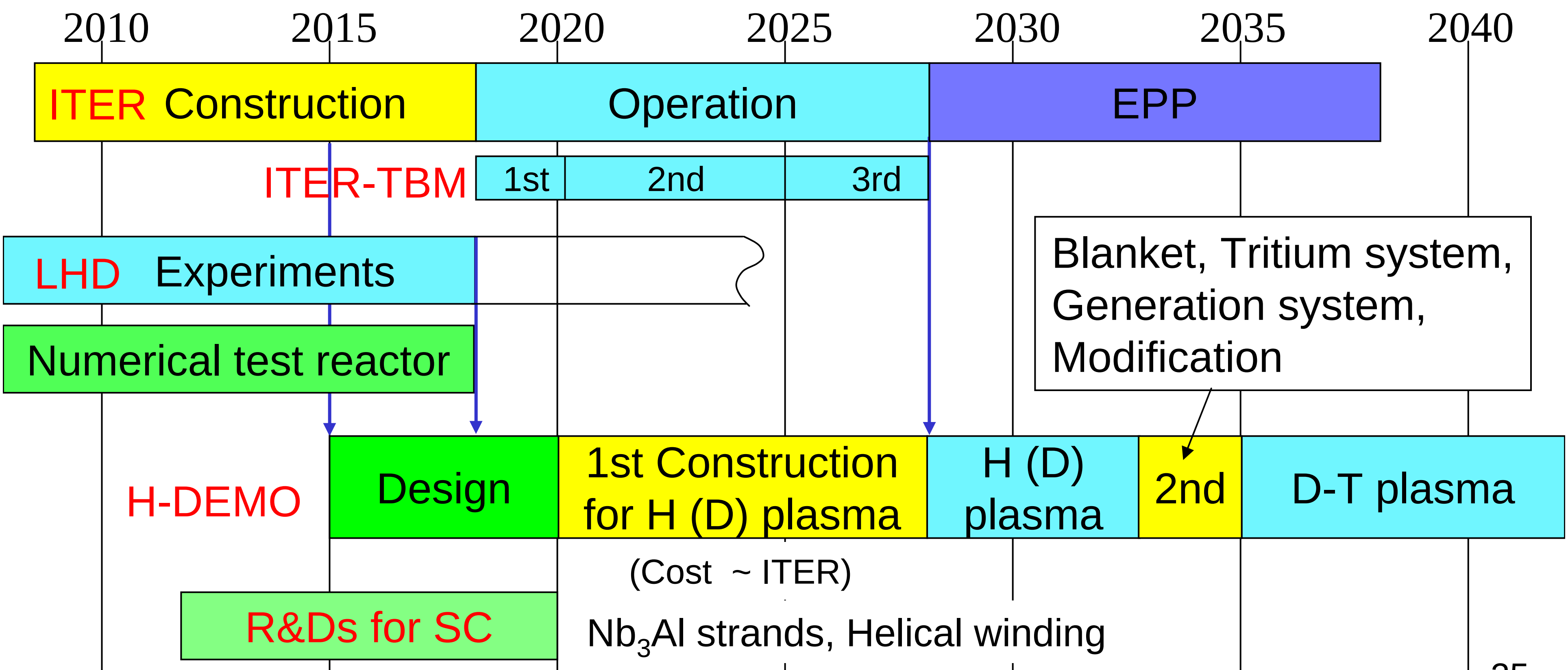


Aiming at Helical Fusion Reactor (Proposal)



Roadmap to Heliotron DEMO (proposal)

- Optimization of magnetic field (design of magnets) should be carried out according to the results of LHD and the numerical reactor.
- R&Ds for the helical winding is indispensable for the 1st construction.
- The performance of plasma confinement is confirmed with H (D) plasma before the construction of the full systems.

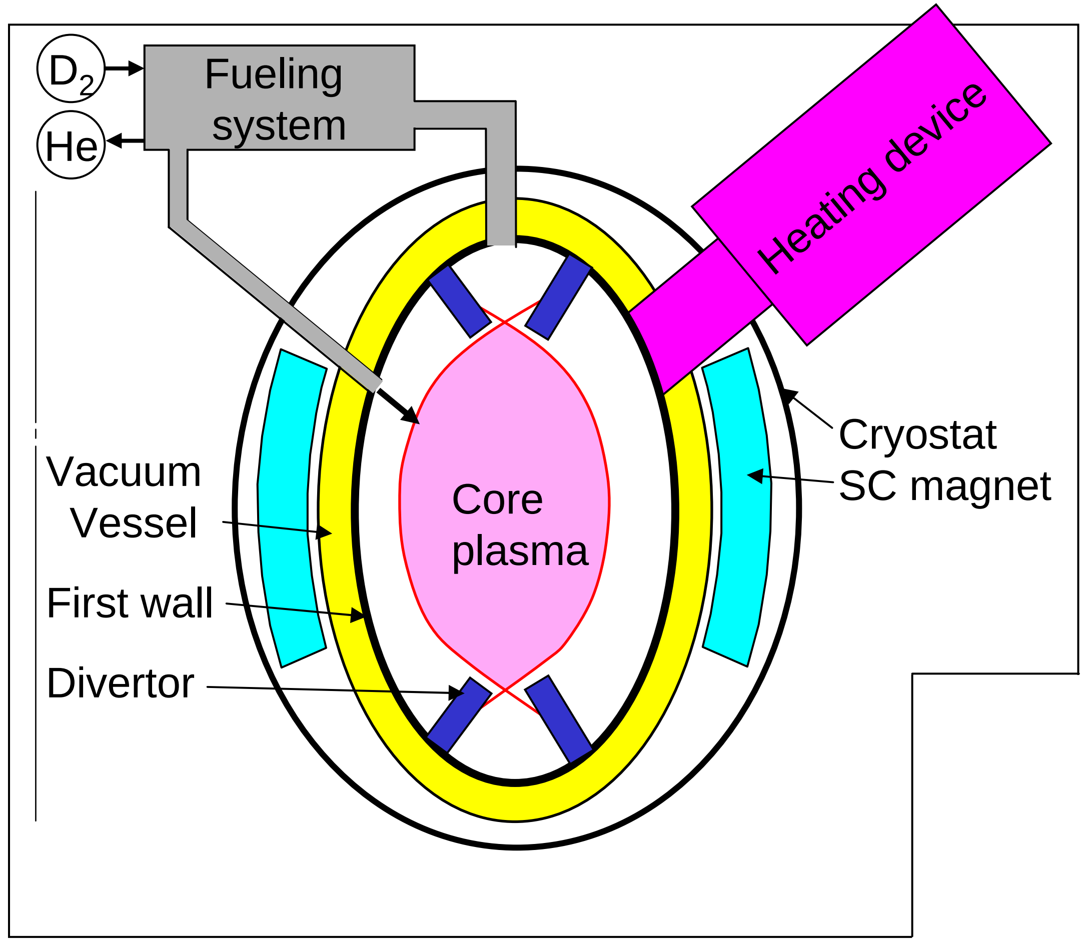


Two Step Construction of H-DEMO

<Objectives at 1st step>

(1) Confirm the plasma confinement at the reactor regime.

(2) Establish the plasma control method and diveror design.



Two Step Construction of H-DEMO

<Objectives at 1st step>

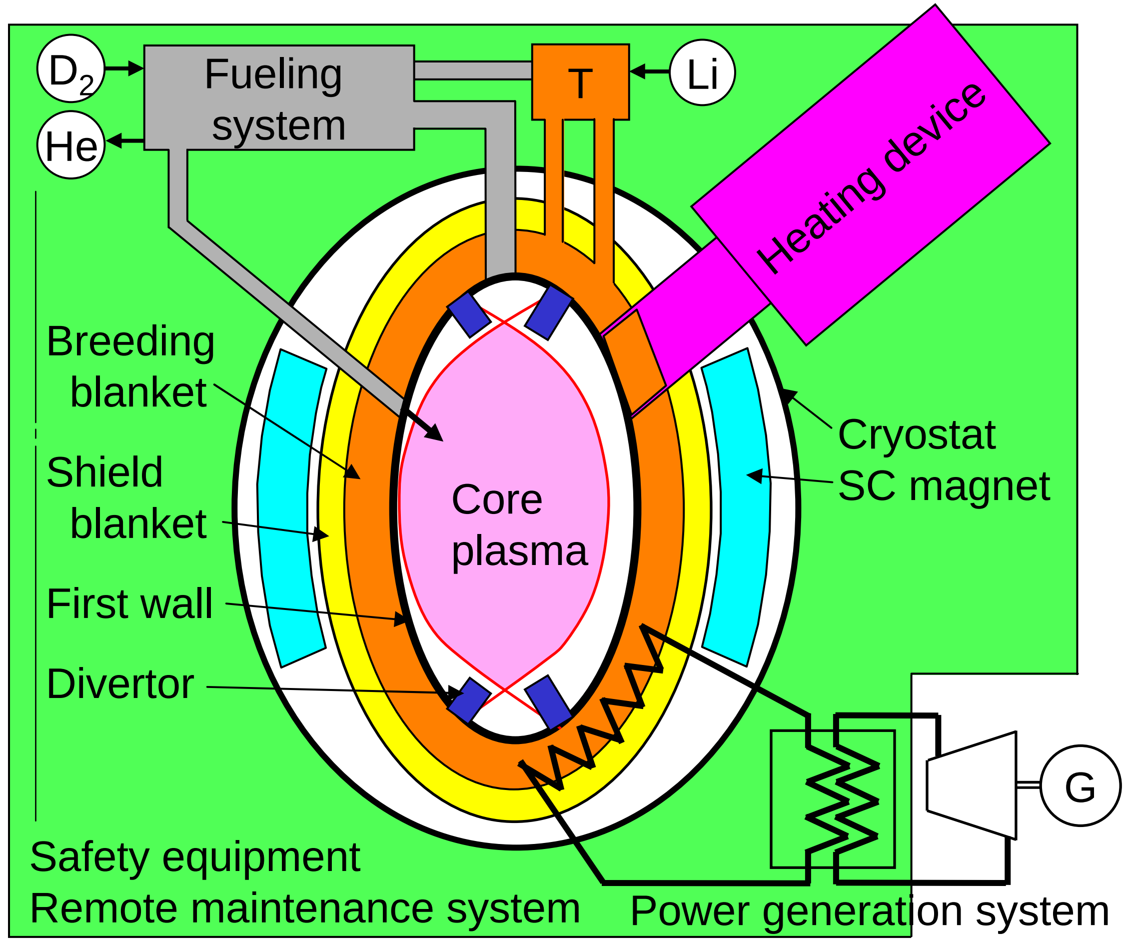
(1) Confirm the plasma confinement at the reactor regime.

(2) Establish the plasma control method and diveror design.

Replace divertor & diagnostics. Install the blanket, T system, and generation system.

<Objectives at 2nd step>

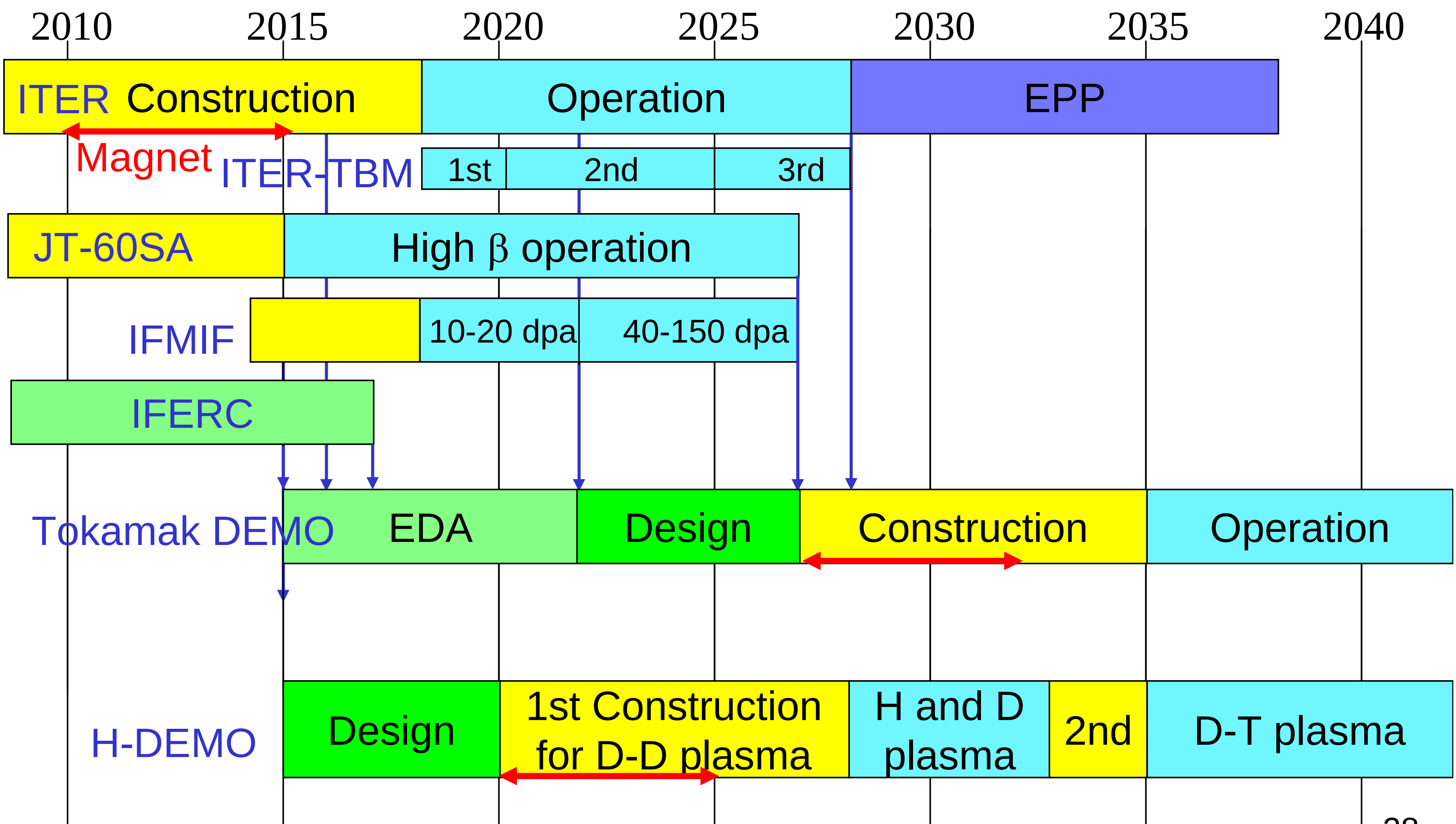
(1) Demonstrate Q>20, TBR>1, steady state operation, and production of electricity.



- ◆Two step construction can reduce the project risk.
- ◆The design of magnets should be optimized from the results of LHD and the numerical reactor.

Continuous Development of Fusion Technology

◆SC magnet technology is continued from ITER to DEMOs.



Summary

(1) The first DEMO reactor should start operation in 2030s to contribute to reduction of CO_2 and the energy crisis.

(2) The necessary size of a LHD-type power plant will be the coil major radius of 16-17 m and the magnetic energy of 120-140 GJ. Although it is three times larger than ITER, Its magnet system can be realized with small extension of the ITER technology.

(3) In order to prove the capability of reliable operation of heliotron power plants, H-DEMO with the magnetic energy of 70-80 GJ is proposed. The amount of its magnets is comparable to ITER.

(4) Two step construction can shorten the start of H-DEMO and reduce the project risk. Also, early construction of H-DEMO contributes to successive development of the fusion engineering.