Conceptual Study on a Fusion Power Reactor for Early Demonstration of Electric Generation.

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Demonstration of power generation

Missions of the Demonstration Reactor for Electric Power Generation.

1. **To generate net electric power.**
   (gross electric power > circulation power)

2. **Steady and continuous operation.**
   (tritium self-sufficiency and maintainability)

3. **Licensing as a power plant.**

Principles for Demo-CREST design

- **Minimum extension from the ITER.**
  - ✓ $P_e^{net} > 0$ must be guaranteed with minimum extension from ITER plasma parameters.
  - ✓ Engineering designs are based on minimum extension from ITER engineering design and operation, and ITER test blanket modules. Materials to be developed in parallel with ITER are taken into accounts.
- **Flexibility to increase net electric power.**
  - ✓ Electric power as a scale of commercial use can be generated with advanced plasma parameters attainable during the ITER operation and with advanced technologies and materials.
- **Reasonable plant construction cost.**
  - ✓ Unit cost of electricity is not so important. However, the construction and operation cost must be reasonable in order to achieve public acceptance.

Minimum extension from the ITER

The net electric power output with almost achievable plasma performance and engineering techniques through the ITER program.

- ✓The electric break-even condition (P_e^{net}=0 MW) is achieved with the same plasma performance as the ITER reference operation mode **($\beta_N \leqq 1.9$, $H_H \leqq 1.0$, $f_{nGW} \leqq 0.85$)**.
- ✓**Total NBI input power is restricted to 200 MW** from the consideration of unit NBI input power and maximum port number on the vacuum vessel.
- ✓Plasma current ramp-up is entirely provided with magnetic flux of CS coils.
- ✓**The maximum magnetic field 16 T** is considered according to the outlook Nb_3Al obtained, for example, in JAERI.
- ✓**Thermal efficiency 30%** is considered on the basis of the design of ITER test blanket modules.
- ✓**NBI system efficiency 50%** is considered according to the outlook obtained through the ITER R&D activities.

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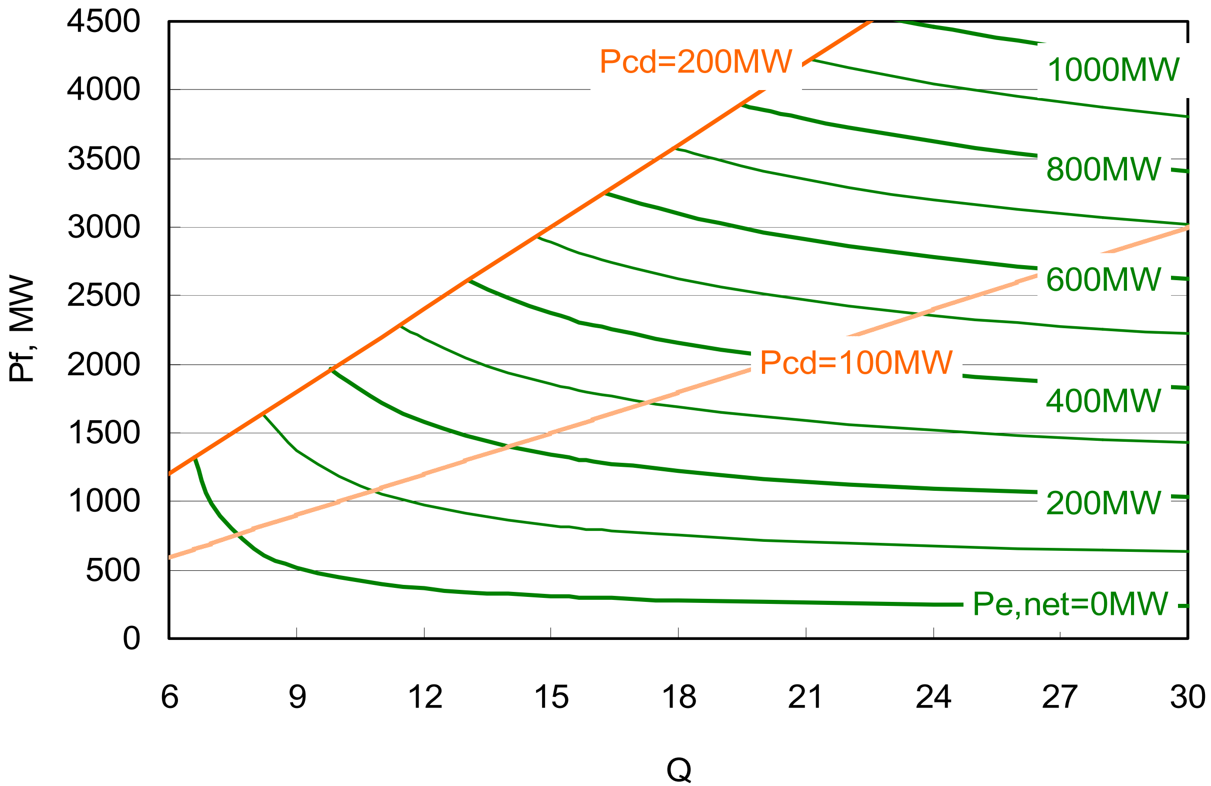
Required Pf and Q for power generation



- ➢Thermal Efficiency: 30 %,
- ➢Plasma heating system efficiency (NBI): 50 %,
- ➢Maximum plasma heating power: 200 MW,
- ➢Recirculation power (excl. plasma heating): 60 MW,
- ➢Neutron coverage of the blanket: 90 %,
- ➢Neutron energy multiplication factor of the blanket: 1.2,

Analysis method of required plasma performance

Database of more than **100,000 operation points*** for a tokamak reactor with **FUSAC****.

Operation points*

Plasma size(major radius, aspect ratio), **major physical parameters,** shape and location of reactor component (TF & CS coil, radial build), weight of each component, construction cost, cost of electricity, fusion power, electric power.

FUSAC**

FUsion power plant **S**ystem **A**nalysis **C**ode.

0D plasma model of ITER Physics Guidelines.

Simple engineering model based on TRESCODE (JAERI).

Generomak Model for economic analysis.

Parameter Region

R (m)	6.0～8.5
A	3.0～4.0
κ	1.7～2.0
δ	0.35～0.45
$T_e(=T_i)$ (keV)	12～20
q_ψ	3.0～6.0
B_{tmax} (T)	(13), **16**, (19)
η_{th} (%)	**30**, (40)
η_{NBI} (%)	(30), **50**, (70)
P_e^{net} (MW)	**0～1,000**

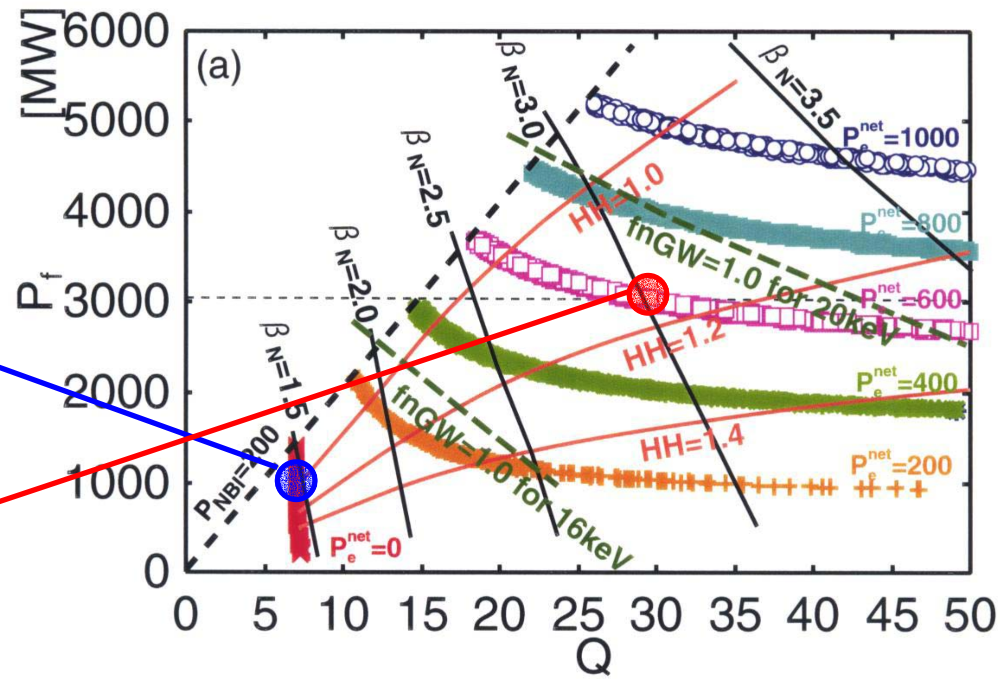
The required region of β_N, f_{nGW}, H_H for P_e^{net}=0～1000MW.

Required plasma performance (1) R = 8.5 m

R = 8.5 m

P_e^{net}=0 MW
(P_f ~ 1,000 MW)
β_N > 1.5, H_H > 1.0

P_e^{net}=600 MW
(P_f ~ 3,000 MW)
β_N > 3.0, H_H > 1.15



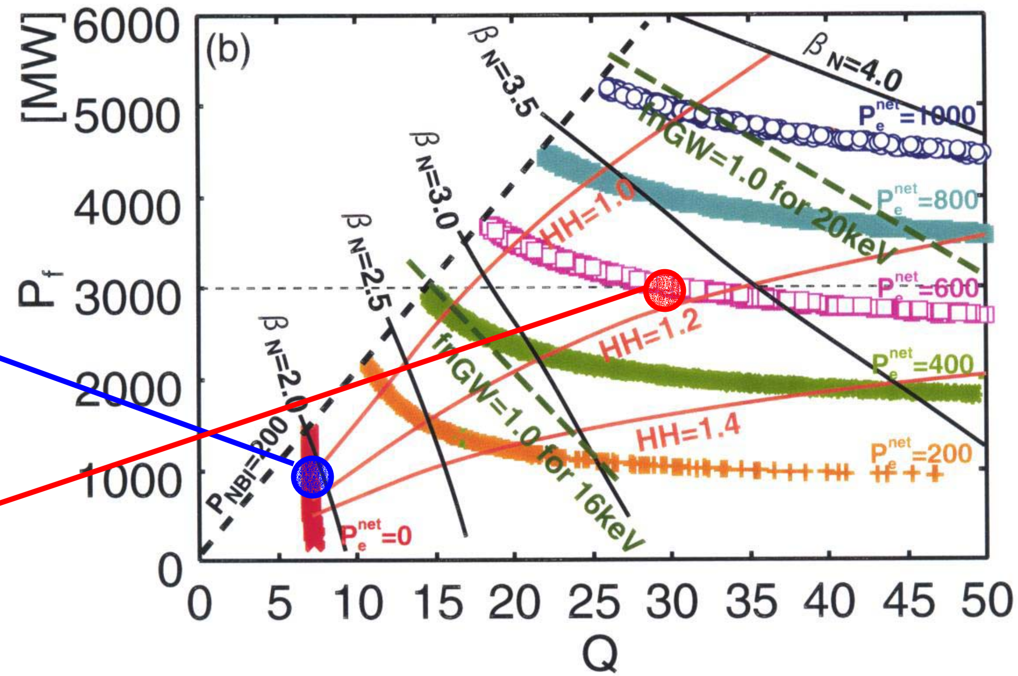
In comparison with the present ITER experimental plan, the outlook of both P_e^{net} =0 MW and P_e^{net}=600 MW will be obtained.

Required plasma performance (2) R = 7.5 m

R = 7.5 m

P_e^{net}=0 MW
(P_f ~ 1,000 MW)
β_N **> 1.9,** H_H **> 1.0**

P_e^{net}=600 MW
(P_f ~ 3,000 MW)
β_N **> 3.4,** H_H **> 1.15**



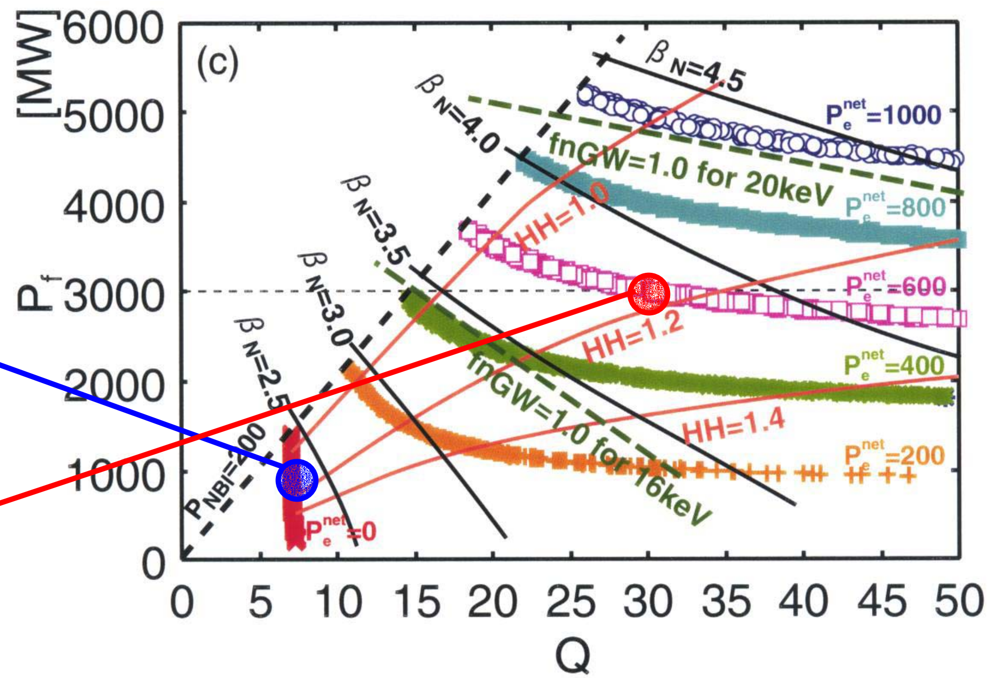
In comparison with the present ITER experimental plan, the outlook of both P_e^{net} =0 MW and P_e^{net}=600 MW will be obtained.

Required plasma performance (3) R = 6.5 m

R = 6.5 m

P_e^{net}=0 MW
(P_f ~ 1,000 MW)
β_N **> 2.3, H_H > 1.0**

P_e^{net}=600 MW
(P_f ~ 3,000 MW)
β_N **> 3.8, H_H > 1.15**



In comparison with the present ITER experimental plan, the outlook of P_e^{net} =0 MW will be obtained, but it is difficult to achieve P_e^{net}=600MW.

Plasma size and required performance

The plasma size can be decreased with the progress on plasma performance.

	R (m)	8.5	7.5	6.5
P_e^{net}=0 MW	β_N	**1.5**	**1.9**	**2.3**
(P_f~1,000 MW)	H_H	1.0	1.0	1.0
P_e^{net}=600 MW	β_N	**3.0**	**3.4**	**3.8**
(P_f~3,000 MW)	H_H	1.15	1.15	1.15

Thermal efficiency 30%, NBI system efficiency 50%, Max. magnetic field 16T.

Operational region on R-A map

P_e^{net} = 0 MW case

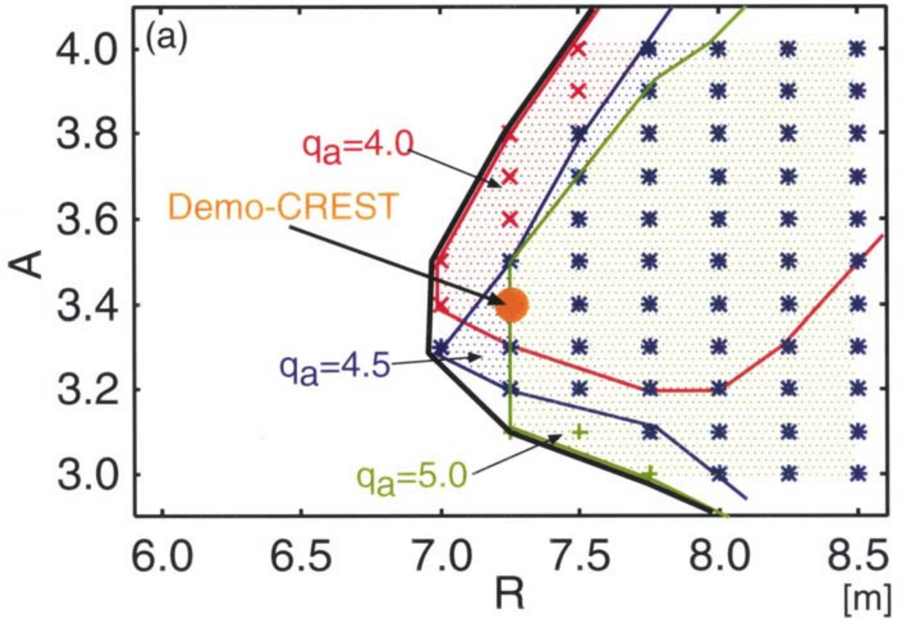
Applied conditions

$\beta_N \leqq 1.9$

$H_H \leqq 1.0$

$f_{nGW} \leqq 0.85$

$\psi_{ramp\text{-}up} \leqq \psi_{CS}$

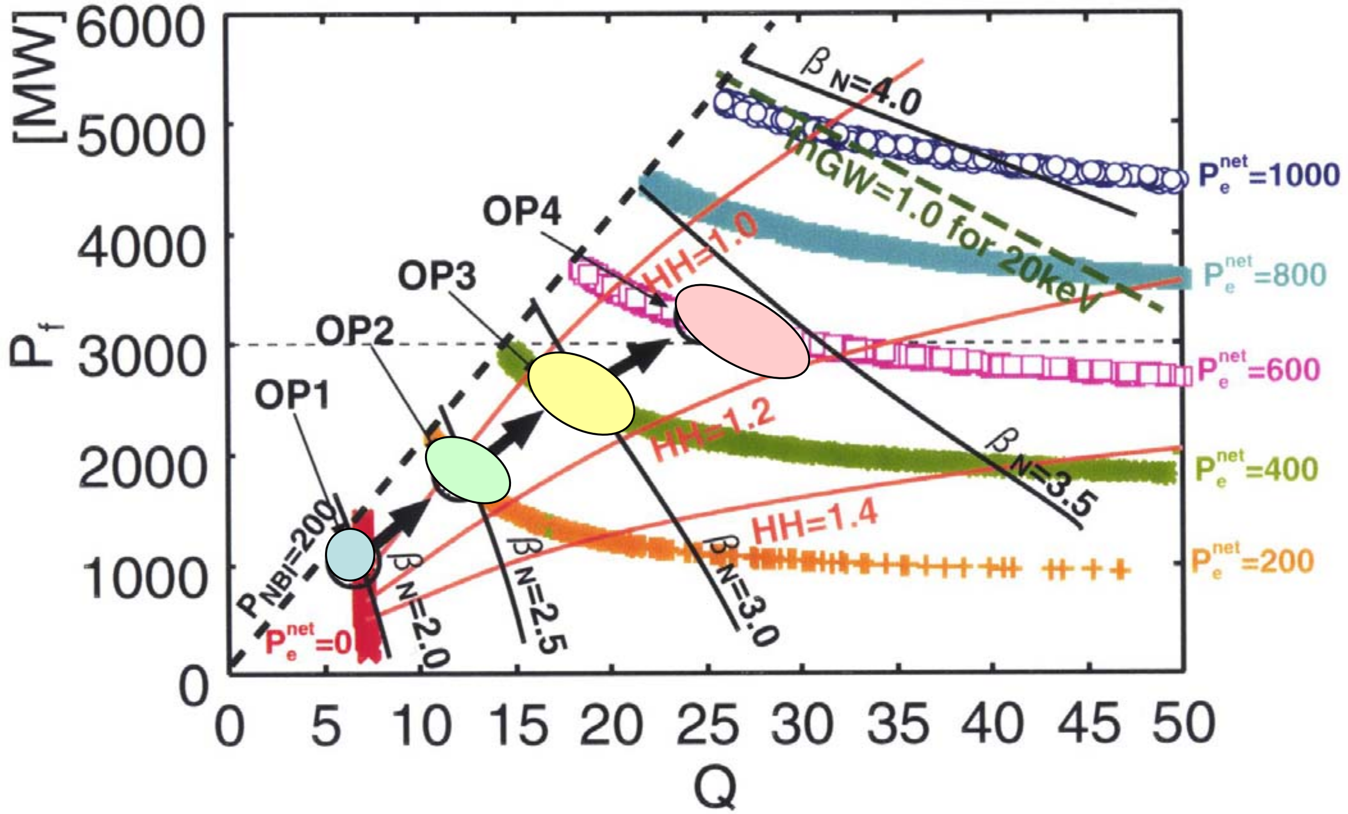


Flexibility for operation plasma current

Demo-CREST plasma configuration: **R = 7.25 m, A = 3.4**

Operation points on Q-Pf map

R= 7.25 m case



Plasma Parameters for OP1-4

	OP1	OP2	OP3	OP4	RS (optional)
β_N	1.9	2.5	3.0	3.4	4.0
R_P (m)/A	7.25/3.4	←	←	←	←
κ/δ	1.85/0.35	←	←	←	←
I_p (MA)	15.9	15.4	15.6	14.7	14.0
f_{bs} (%)	23.7	34.4	40.1	49.7	56.1
q_ψ	5.0	5.0	5.0	5.2	5.2
q_0/q_{min}	1.3/-	1.7/-	2.1/-	2.3/-	3.5/2.9
β_p	1.11	1.55	1.86	2.15	2.42
B_t (T)	8.0	8.0	8.0	7.8	7.2
$\langle T_e \rangle$ (keV)	17.9	18.7	20.7	18.4	18.1
$\langle T_i \rangle$ (keV)	18.8	19.5	21.5	19.2	19.0
Z_{eff}	1.7	1.7	2.1	2.1	2.1
$\langle n_e \rangle$ ($10^{20}/m^3$)	0.625	0.789	0.873	1.05	1.11
f_{nGW}	0.56	0.73	0.80	1.02	1.12
H/HH98	1.5/0.96	1.7/1.1	1.9/1.2	2.0/1.2	2.2/1.3
E_b (MeV)	1.5	1.5	1.5	1.5	1.5
P_{NBI} (MW)	188	190	185	191	160
on axis	164	163	157	166	118
off axis	24	27	28	25	42
P_f (MW)	1260	1940	2460	2840	2850
Q	6.7	10.2	13.3	14.9	17.8

Electric power of Demo-CREST for OP1-4

	OP1	OP2	OP3	OP4
Major radius, R (m)	7.25			
Aspect ratio, A	3.4			
Normalized beta, β_N	1.9	2.5	3.0	3.4
HH factor, H_H	0.96	1.1	1.2	1.2
Ratio to GW limit, f_{nGW}	0.63	0.79	0.87	1.02
NBI power, P_{NBI} (MW)	188	190	185	191
Energy multiplication, Q	6.7	10.2	13.3	14.9
Fusion power, P_f (MW)	1,260	1,940	2,460	2,840
Thermal output, P_{th} (MW)	1,470	2,180	2,720	3,100
η_{th}=30%, η_{NBI}=50%				
P_e, gross (MW)	440	650	810	940
P_e, net (MW)	30	230	490	550
η_{th}=40%, η_{NBI}=60%				
P_e, gross (MW)	590	870	1,090	1,250
P_e, net (MW)	330	600	810	960
Mean wall load P_w (MW/m^2)	1.1	1.8	2.2	2.6

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Demonstration of tritium self-sufficiency

- **Tritium self-sufficiency, net TBR > 1, which is indispensable for steady and continuous operation, is one of the most important missions of the power generation demonstration device.**
- Advantage of fusion energy on the energy security must be shown by demonstration of tritium self-sufficiency.
- Breeding blankets are going to be set in the ITER as test blanket modules. However, it may be difficult to obtain a secure proof of net TBR > 1 for the next device.
- Therefore, the blanket design of the demonstration reactor must have a large margin in tritium breeding.

✓ **Fusion power and size of device (neutron wall loading)**
✓ **Thermal efficiency (temperature of coolant)**
✓ **Blanket structure (neutron coverage of blanket)**
✓ **Structure material development (design window of blanket)**

Tritium breeding blanket

Design of the tritium breeding blanket of the Demo-CREST must be conservative, similar to the plasma parameters.

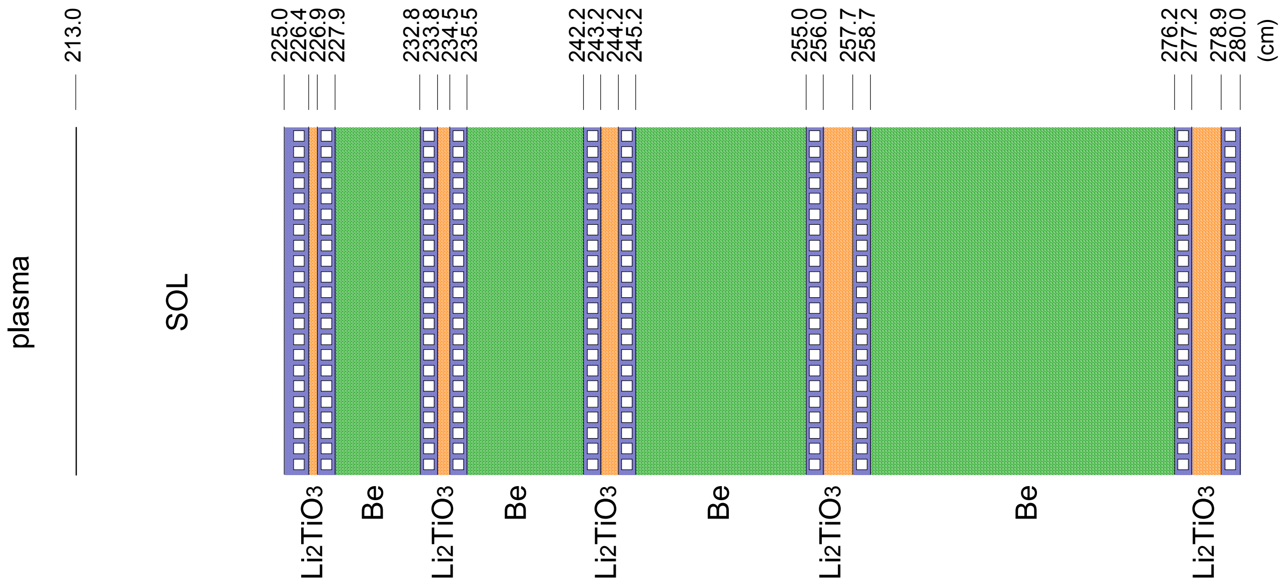
- It must be based on one of the ITER test blanket module designs.
  - ✓ Design target is 3GW in fusion power (OP4).
  - ✓ Neutron wall load in the device of R=7.25 and A=3.4 is approximately 2.7 MW/m^2 in average, 4.2 MW/m^2 in peak. Mean heat load is approximately 0.68 Mw/m^2.

- Materials
  - ✓ Solid breeder(Li_2TiO_3)/Be/RAF/Water structure.

- Restrictions and conditions
  - ✓ Maximum temperature of RAF: 723 K
  - ✓ Maximum temperature of Li_2TiO_3: 1023 K
  - ✓ Maximum temperature of Be: 723 K
  - ✓ Neutron wall load: 2.7 MW/m^2
  - ✓ Heat load on first wall: 0.68 MW/m^2
  - ✓ Blanket coolant temperature: 603 K

Radial build of the breeding blanket

Based on the ITER Test Blanket Module



Local TBR : 1.48

Neutron energy multiplication factor : 1.31

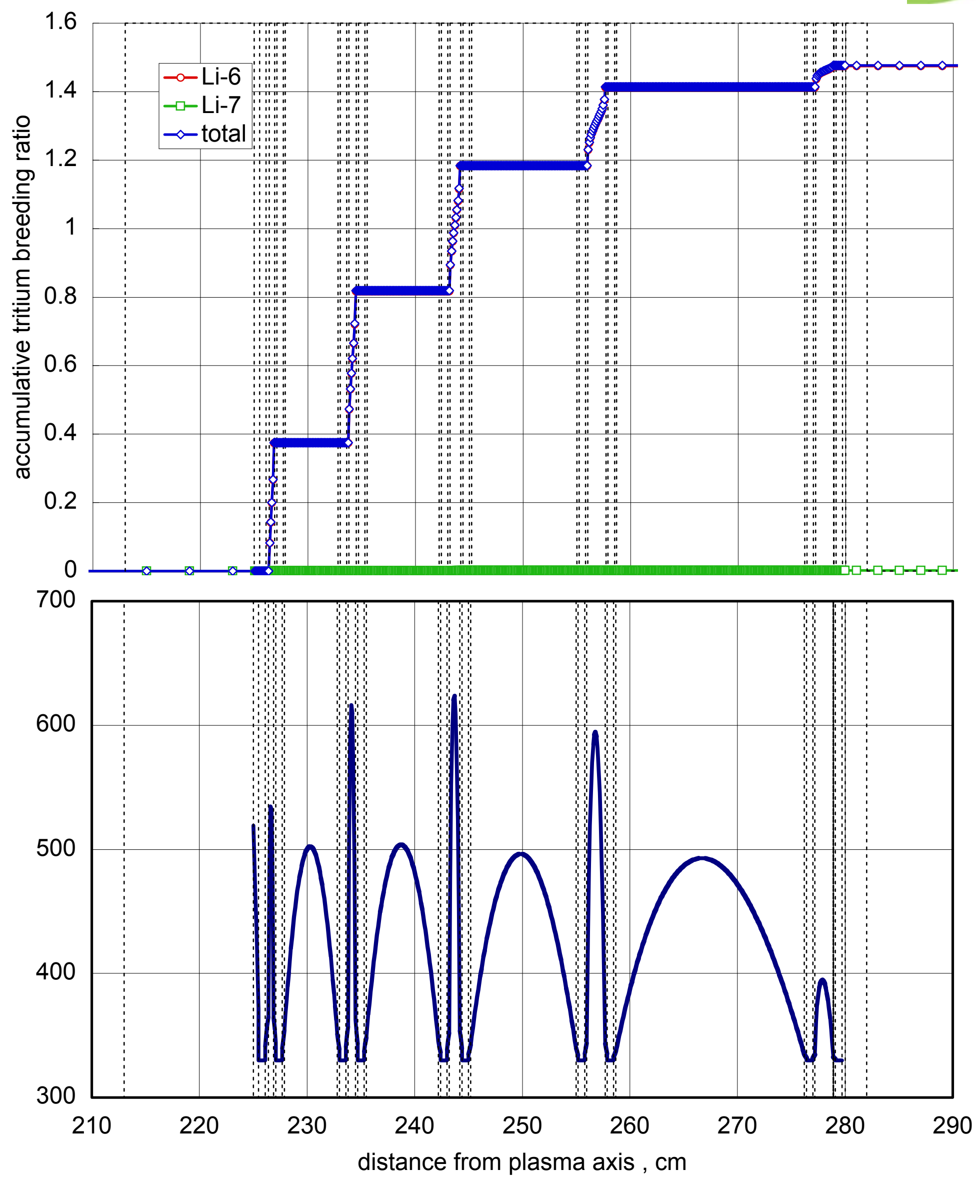
Li_2TiO_3 / Be ratio : 12.5 / 87.5

(Li_2TiO_3: 85 %TD and 80 %PF, Be: 95 %TD and 80 %PF)

Li-6 enrichment: 90 %

Tritium breeding ratio and temperature distribution

Accumulative tritium breeding ratio



Temperature distribution

Limit of beryllium temperature restricts the blankets structure.
Temperature of Li_2TiO_3 is lower than 900K.

Blanket structure (1: small modules)

Small blanket modules and vehicle type maintenance devices (ITER like)

↑Flexibility to replace a small part of blankets, (e.g. accidentally damaged modules, modules exposed to peak wall load)

↓**Long maintenance period to replace all (or a large part of) blankets.**

- ✓ Small modules due to handling weight limit.
- ✓ A large number of modules.
- ✓ Large number of procedures including pipe re-connection and connection.

↓**Reduction of the net TBR.**

- ✓ Gaps between modules, structure materials of blanket side wall and dead space due to curved surface of the blanket and module shape for front access.

In the case of Demo-CREST (larger than the ITER),

↓**Number of modules is larger than that of the ITER.**

↑**On the basis of ITER technology.**

Size of blanket modules & required local TBR

Simple evaluation of required local TBR by neutron coverage ratio of breeding zone in the blanket.

Size of blanket module	1m×1m		1.5m×1.5m		2m×2m	
Ratio in surface area						
Blanket						
gaps between modules	0.017	0.018	0.011	0.012	0.008	0.009
side wall of modules	0.098	0.104	0.066	0.070	0.050	0.053
coolant headers in modules	0.079	0.084	0.054	0.057	0.041	0.043
ribs between breeding zone	0.014	0.015	0.020	0.021	0.023	0.025
holes for front accessing	0.014	0.015	0.006	0.007	0.004	0.004
breeding zone	0.628	0.665	0.692	0.733	0.724	0.766
sub total	**0.850**	**0.900**	**0.850**	**0.900**	**0.850**	**0.900**
Divertor, duct, others	0.150	0.100	0.150	0.100	0.150	0.100
Required local TBR						
in the case of net TBR=1.00	1.593	1.505	1.445	1.365	1.382	1.305
in the case of net TBR=1.05	1.673	1.580	1.517	1.433	1.451	1.370
in the case of net TBR=1.10	1.753	1.655	1.590	1.501	1.520	1.436

Blanket structure (2: large module)

Large sector blankets and cask type maintenance devices
(CREST, DREAM, ARIES-RS,….)

↑Short maintenance period.

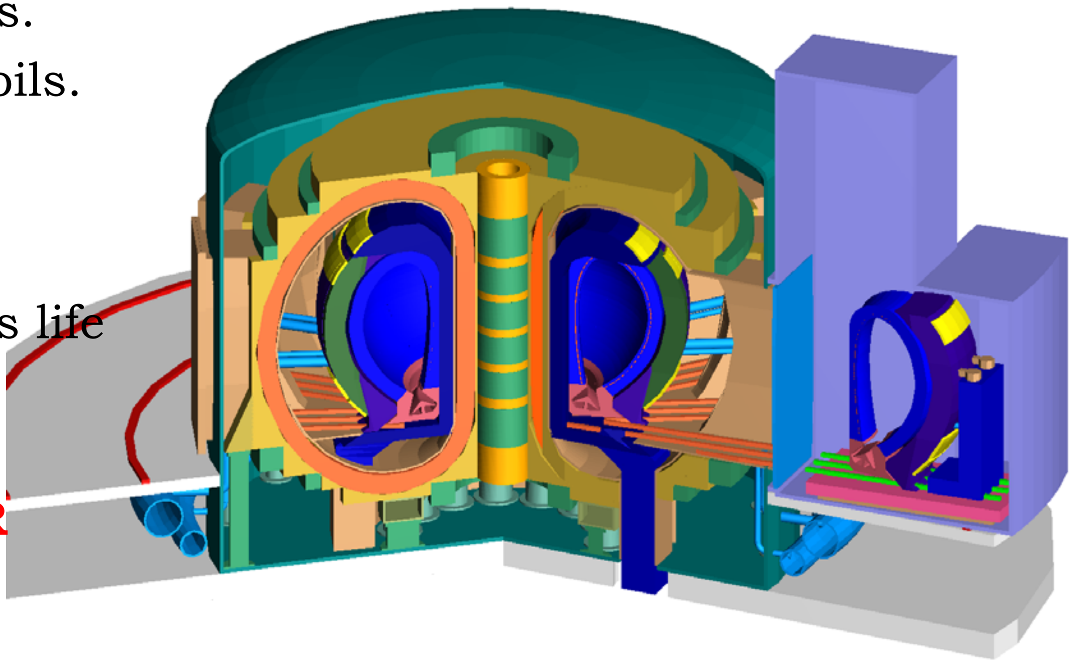
↑High net TBR can be expected.

↓Difficulty of accurate handling of large weight components.

↓Requirement of large TF coils.

↓Replacement of the whole blankets (and divertors) regardless to wall load distribution or components life time.

↓Extension from the ITER Technology.

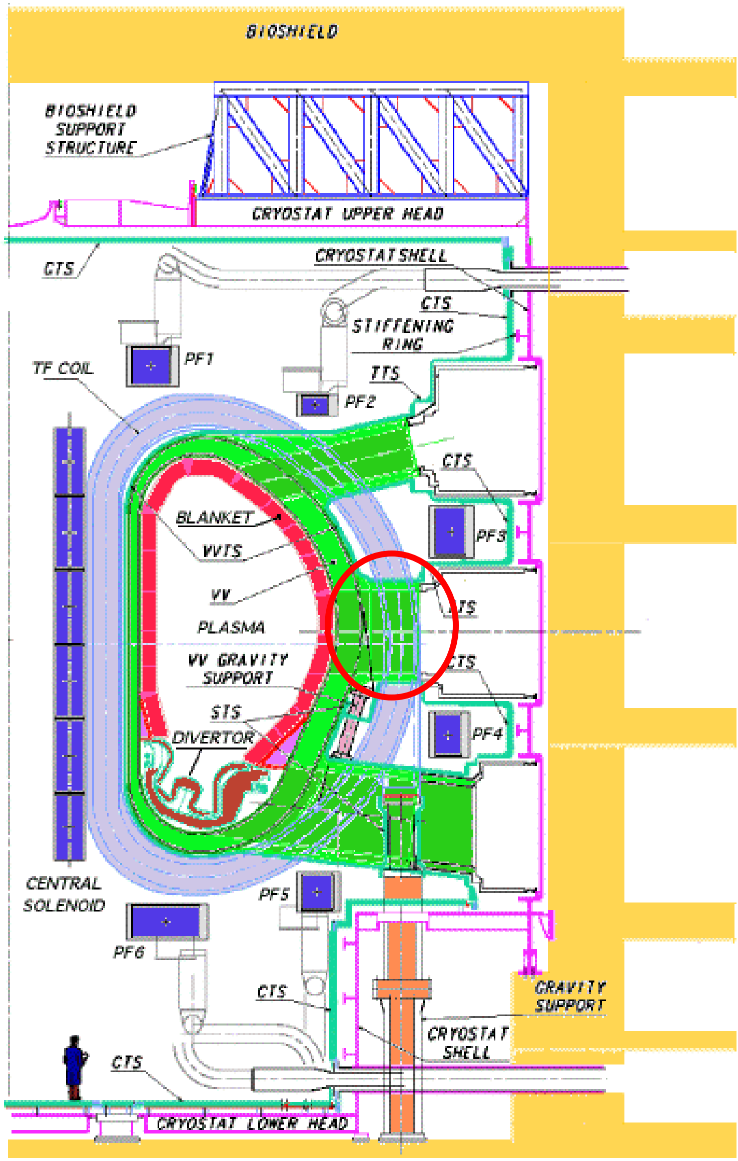


ITER port shield (shield plug)

Minimum extension from the ITER.

ITER port shield:

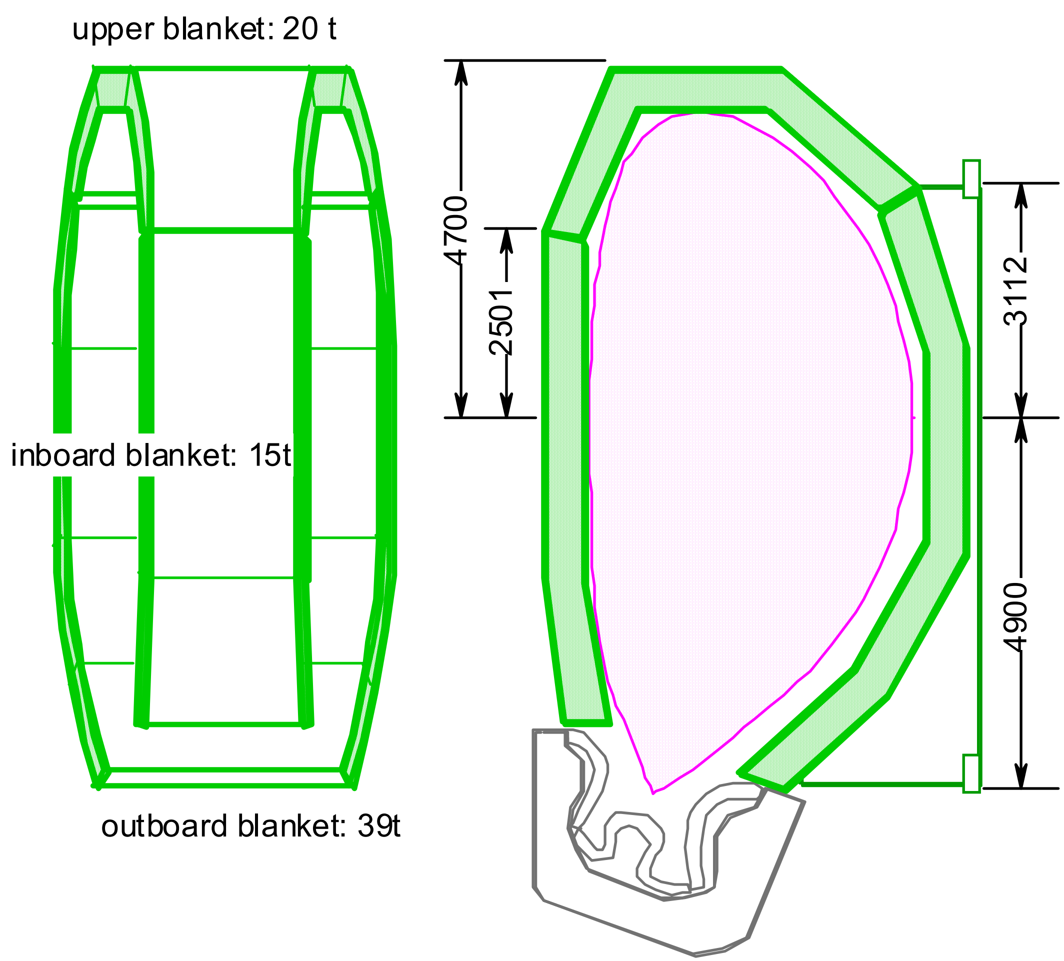
- Removable shield placed at backward of the test blanket modules or other components at horizontal port.
- The weight is approximately 40 tons.
- Removed and transferred by cask type handling device.



Blanket structure of the Demo-CREST

Similar procedures to large sector blanket system.

- Large maintenance port (horizontal port).
- 14 large TF coils.
- Blanket of one sector, 1/14 of torus, is divided to three parts, outboard, inboard and upper.
- The weight of the outboard blanket is approximately 40 tons. Those of the inboard blanket and the upper blanket are around 15 to 20 tons.



Maintenance procedures

Procedures of blanket replacement

Straight withdrawal of the outboard and inboard blankets through each horizontal maintenance port.

Take down the upper blankets to mid-plane level, and withdraw through each horizontal port. Although weight of the upper blanket is larger than that of ITER blanket modules, the flexibility of handling to the toroidal direction.

Outboard shields handling and support of the outboard blanket

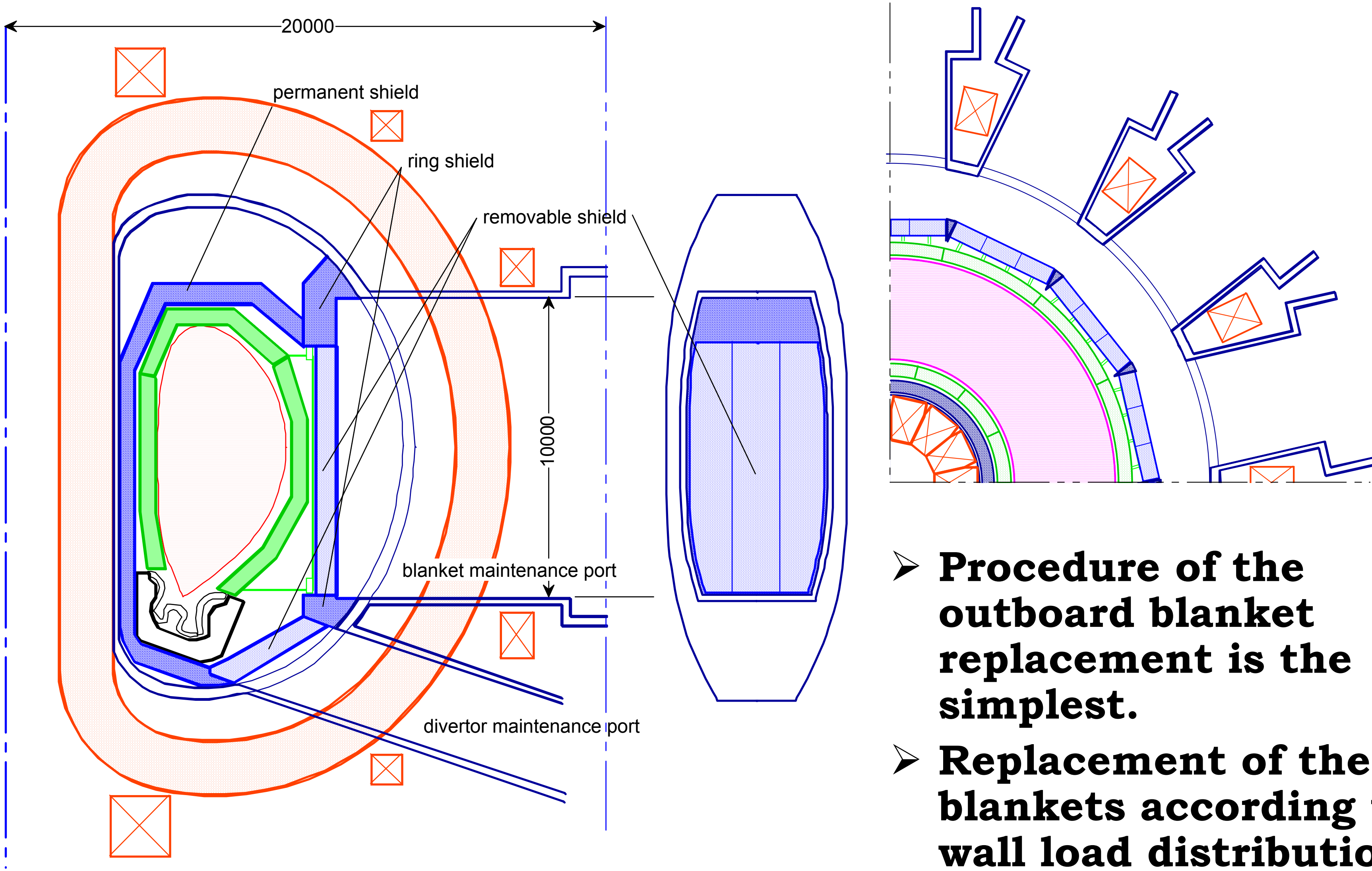
Outboard shield, which must be removed to open the maintenance port, is divide to three parts. Weight of outboard shield to be removed is approximately 126 tons. Therefore weight of each part is 42 tons.

As the outboard blanket and the outboard shields must be withdrawn separately,the outboard blanket is supported independently by the ring shield (see next sheet), not supported by the outboard shield.

Replacement of the divertors

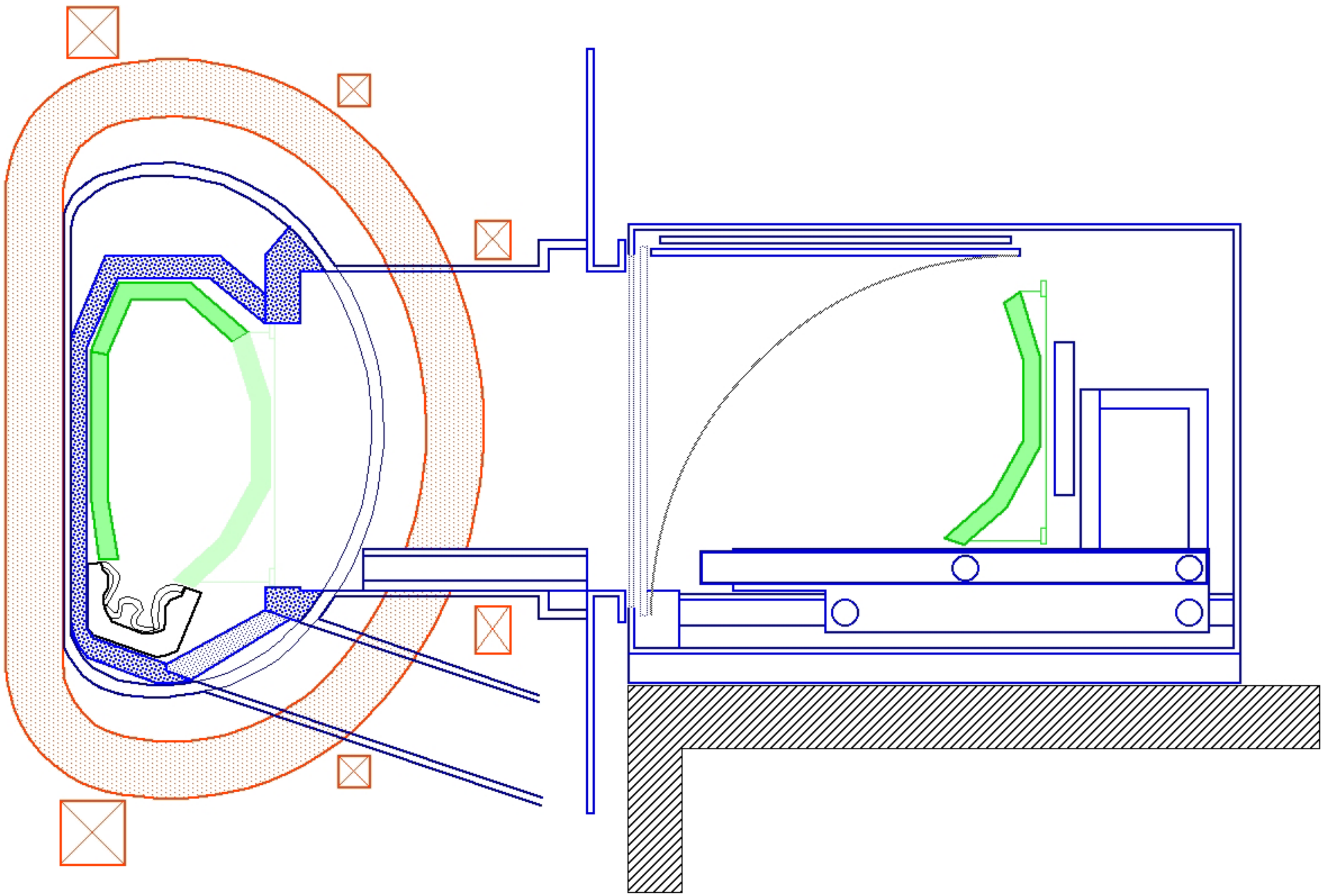
The interval of divertor replacement is considered to be different from that of the blanket. The divertors can be replaced independently with similar procedure to the ITER divertor maintenance.

Elevation view and plane view



- Procedure of the outboard blanket replacement is the simplest.
- Replacement of the blankets according to wall load distribution.

Blanket handling (1)



Summary

We have started conceptual study on a fusion power reactor for early demonstration of the electric generation.

Demonstration of electric power generation (Pe,net > 0) is possible with minimum extension from the ITER plasma parameters, the ITER technologies, and materials to be developed in parallel with the ITER.

And…

- ✓ Up to 3GW fusion power in R=7.25m, A=3.4 device.
- ✓ Possibility to obtain 1,000 MWe with advanced plasma parameters and advanced technologies.
- ✓ Blanket concept based on the ITER test blanket module consists of Li_2TiO_3, Be, pressurized water and reduce activation ferritic steel. The local TBR and neutron energy multiplication factor are 1.48 and 1.31, respectively.
- ✓ The concept of blanket structure and maintenance procedure can secure the tritium self-sufficiency and maintainability, which are indispensable to demonstrate steady and continuous operation. Reduction of the blanket replacement period and the replacement according to the wall load distribution would be possible.

For details

Please refer to our recent papers:

"Plasma Performance Required for a Tokamak Reator to Generate Net Electric Power"

R. Hiwatari, K. Okano, Y. Asaoka Y. Yoshida, K. Tomabechi

J. Plasma Fusion Res., Vol.78, No.10 (2002) 911-913

"Stability of Beam Driven Equilibria with a Closed Stablization Wall"

K. Okano, R. Hiwatari, Y. Asaoka

J. Plasma Fusion Res., Vol 79, No.6 (2003) 546-548