

Conceptual design of a demonstration reactor for electric power generation

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1. Introduction

Three development stages of fusion energy:

- 1) demonstration of a fusion reactor operation --- by ITER
- 2) demonstration of electric power generation as a power plant -- by?
- 3) demonstration of economic and safe performance -- by?

Recently, early realization of electric power generation (~2030) with fusion energy has been discussed **where the 2nd and 3rd stage should be achieved with a single demonstration device.**

The demonstration device for the 2nd and 3rd stage must be constructed from 2020~2025 in our new scenario.

This will be a very difficult target, because, at present (2005), we do not have ITER, IFMIF (nor VNS), steady state tokamak with a size of JT-60 or JET which allows a plasma for $Q \sim 1$, yet.

The allowable plasma and engineering target in the initial operation stage for the single step Demo will be very restricted.

Conceptual study on a device for such single-step-Demo, named “Demo-CREST” was discussed in the present paper.

The CREST, which is proposed as a commercial reactor by the authors (IAEA 98), has been a concept of the compact reversed shear tokamak reactor. However, it is not suitable for a demo-plant next to the ITER.

The design of Demo-CREST should be conservative enough (or **similar to SSTR**) but also should have a capacity to test a possibility toward an advanced tokamak plant like CREST ($\beta_N \sim 5.5$), in the final stage of its operation.

2. Principles for Demo-CREST design

A plant for Demo-CREST should have capacities both

- (1) to demonstrate electric power generation in a plant scale with moderate plasma performance, which will be achieved in the early stage of the ITER operation, and foreseeable technologies and materials
- (2) to show a possibility of an economical competitiveness with advanced plasma performance and higher performance blanket systems.

It will be difficult to assure two capacities in a single device before 2020, possibly. The Demo CREST concept tries to realize it by means of replacing breeding blanket from the basic one to the advanced one.

Our principles are:

(1) Minimum extension from the ITER:

- # minimum extension from ITER plasma
- # minimum extension from ITER engineering design and operation including the ITER test blanket modules
- # Materials to be developed in parallel with the ITER operation are taken into accounts.

(2) Flexibility to increase electric power:

- # Demo-CREST must generate net power of commercial plant scale with advanced plasma parameters and with advanced technologies and materials.

3. Plasma Core of the Demo-CREST

In order to optimize the plasma core of the Demo-CREST, database of more than 100,000 operation points for a tokamak reactor was built with the Tokamak system code FUSAC.

plasma size (major radius, aspect ratio) : major physical parameters
shape and location of reactor components
construction cost : cost of electricity : fusion power : electric power

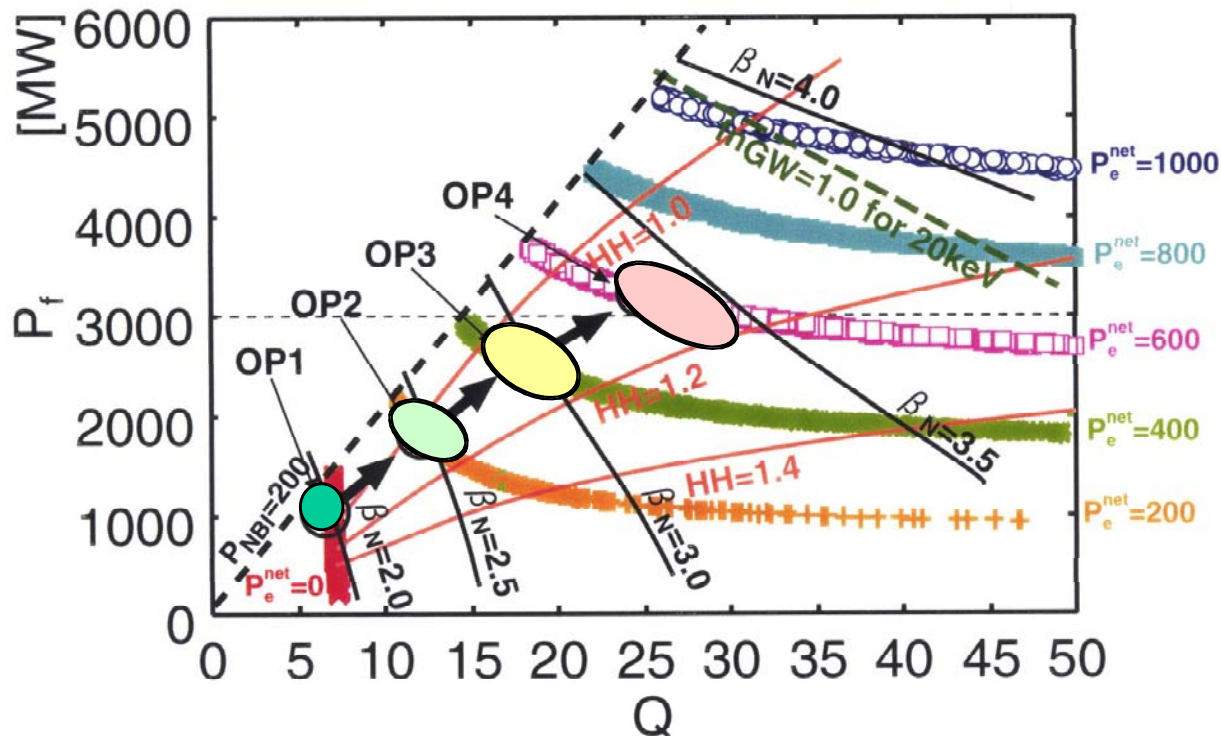


FIG.1.
Required plasma
performance for electric
power generation, in the
case of $R=7.25m$.

By analyzing the database, the required size of the plasma core for electric power generation was determined with an assumed plasma performance, i.e. a set of β_N , f_{nGW} , and HH, under the condition of 30% thermal efficiency, 50 % NBI system efficiency, and 16 T maximum magnetic field.

In the case of 7.25 m in major radius $\beta_N > 1.9$ is required to generate approximately 400 MWe gross electric power, which equal to the circulation power.

If $\beta_N = 3.4$ is achieved, The reactor would produce 2.8 GW in fusion power and generate 940 MWe in gross and about 500 MWe in net electric power.

These features satisfy the minimum requirement **in the initial stage** of Demo-CREST.

TABLE I: MAJOR PARAMETERS OF The Demo-CREST

		<i>Normal shear</i>			<i>Reversed shear</i>
		OP1	OP2	OP3	OP4 RS
	β_N	1.9	2.5	3.0	3.4
	q_ψ	4.0			
	major radius / minor radius (m)	5.0	5.0	5.0	5.2
	κ / δ	6.5			
	f_{nGW}			7.25 / 2.13	
	HH			1.85 / 0.35	
	PNBI (MW)	0.63	0.79	0.87	1.1 1.3
	fusion power (MW)	0.96	1.1	1.2	1.2 1.4
basic blanket	gross elec. power (MW)	188	190	185	191 107
	net elec. power (MW)	1,260	1,940	2,460	2,840 3,160
adv. blanket	gross elec. power(MW)	440	650	810	940 N/A
	net elec. power (MW)	30	230	390	490 N/A
		590	870	1,090	1,250 1,350
		230	490	710	850 1,090

There are equilibria with higher beta ($\beta_N > 5$), because the Demo-CREST plasma ($R/a=3.4$) is a similar figure as the CREST. We will be able to try it with this machine.

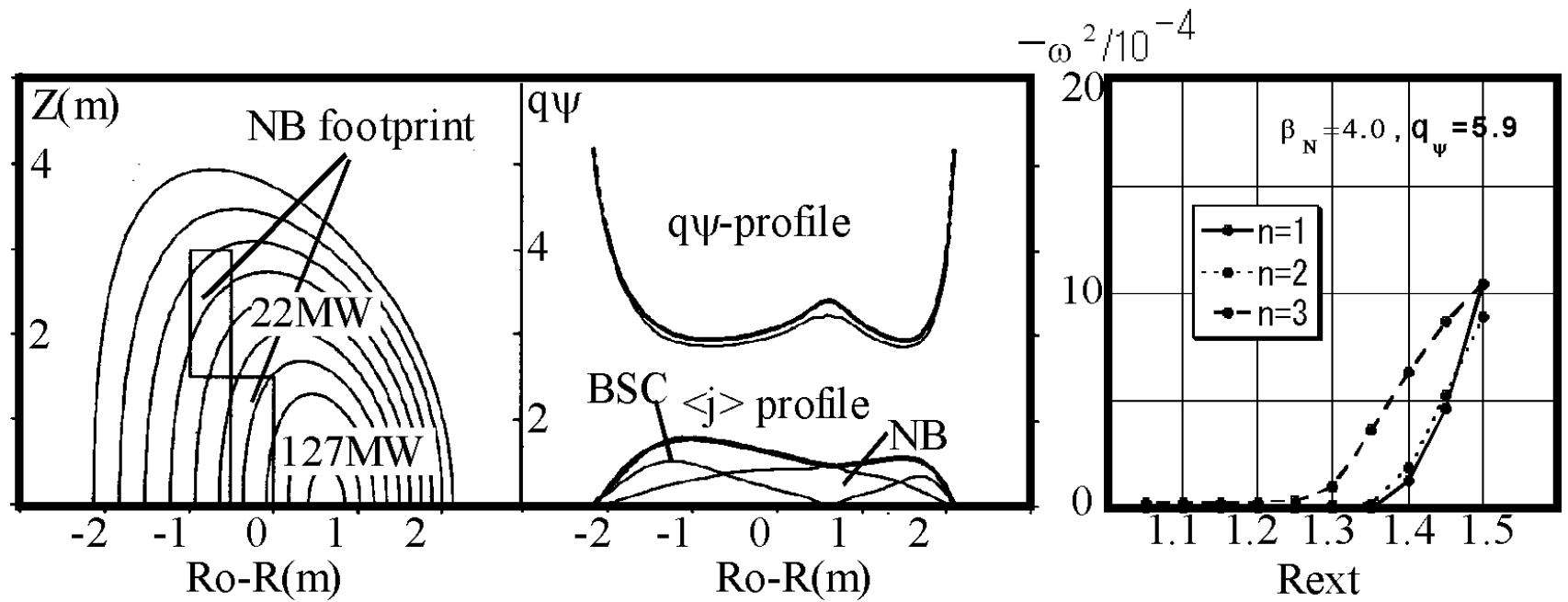


FIG.2. RS equilibrium at $\beta_N = 4.0$ and the stability analysis.

4. Engineering Concept

4.1 Demonstration of tritium self-sufficiency

The breeding blanket structure and its maintenance scheme, are optimized to demonstrate the steady and continuous operation.

"Net TBR > 1" is an important mission for demo-reactors. Therefore the design of the demo-reactor must have a large margin in TBR :

lower wall load, lower temp. → less cooling structure → Higher TBR
higher neutron coverage of the blanket → Higher TBR

However, the low neutron wall load requires small fusion power and large size of the device. Lower blanket temperature would result in low thermal efficiency. These requirements will increase the construction cost.

In the case of the Demo-CREST, we decided to accept "a certain expensiveness" for a secure demonstration of tritium self-sufficiency.

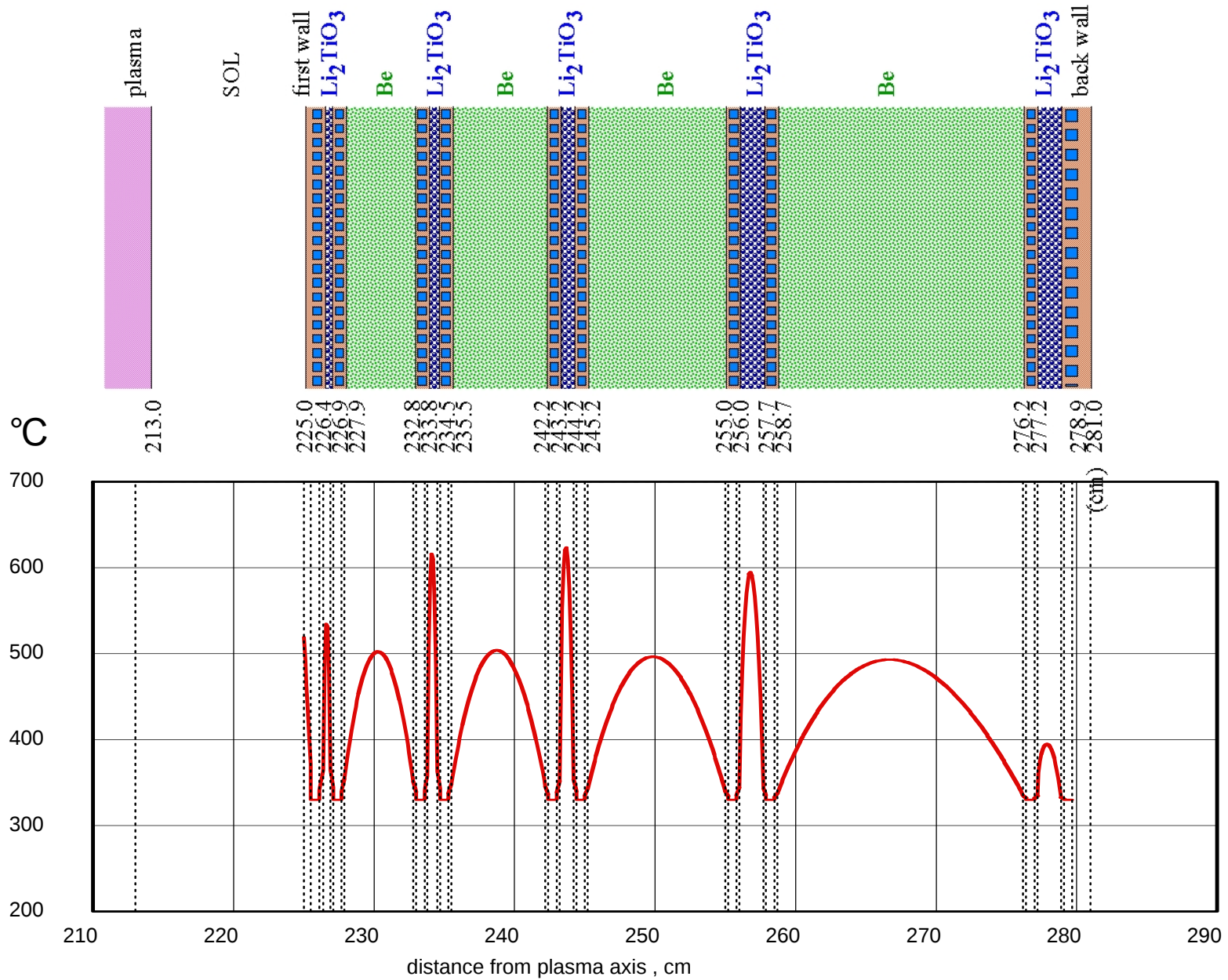
Additionally, the Demo-CREST concept is adopting the innovative blanket structure and its maintenance procedure for high neutron coverage.

4.2 Basic blanket (for the initial stage where $\beta_N < 3.5$)

Basic blanket is for early demonstration of power generation.

RAF + relatively low coolant temperature (603K, similar to SSTR) would bring about a sufficient margin in tritium breeding: (**local TBR = 1.48, or 1.05 w/ 70% coverage**).

The basic blanket consists of reduced activation ferritic steel (RAF) for structural material, 90 % lithium-6 enriched lithium titanate pebbles for breeder, beryllium pebbles for neutron multiplier and pressurized water for coolant. Temperature restrictions for materials are assumed to be similar to those of the ITER test blanket module, i.e. maximum temperature of RAF is 773 K, maximum temperature of breeder material, Li_2TiO_3 , is 1023 K and maximum temperature of neutron multiplier, Be, is 773 K. Evaluation of tritium breeding ratio and blanket temperature are conducted under the condition of 3 GW in fusion power. Neutron wall load is 2.7 MW/m² in average, 4.2 MW/m² in peak. Mean heat load is approximately 0.68 MW/m².



Radial build of the basic blanket and temperature distribution of the basic blanket

4.3 Advanced blanket (for the adv. stage where $\beta_N \gg 3.5$)

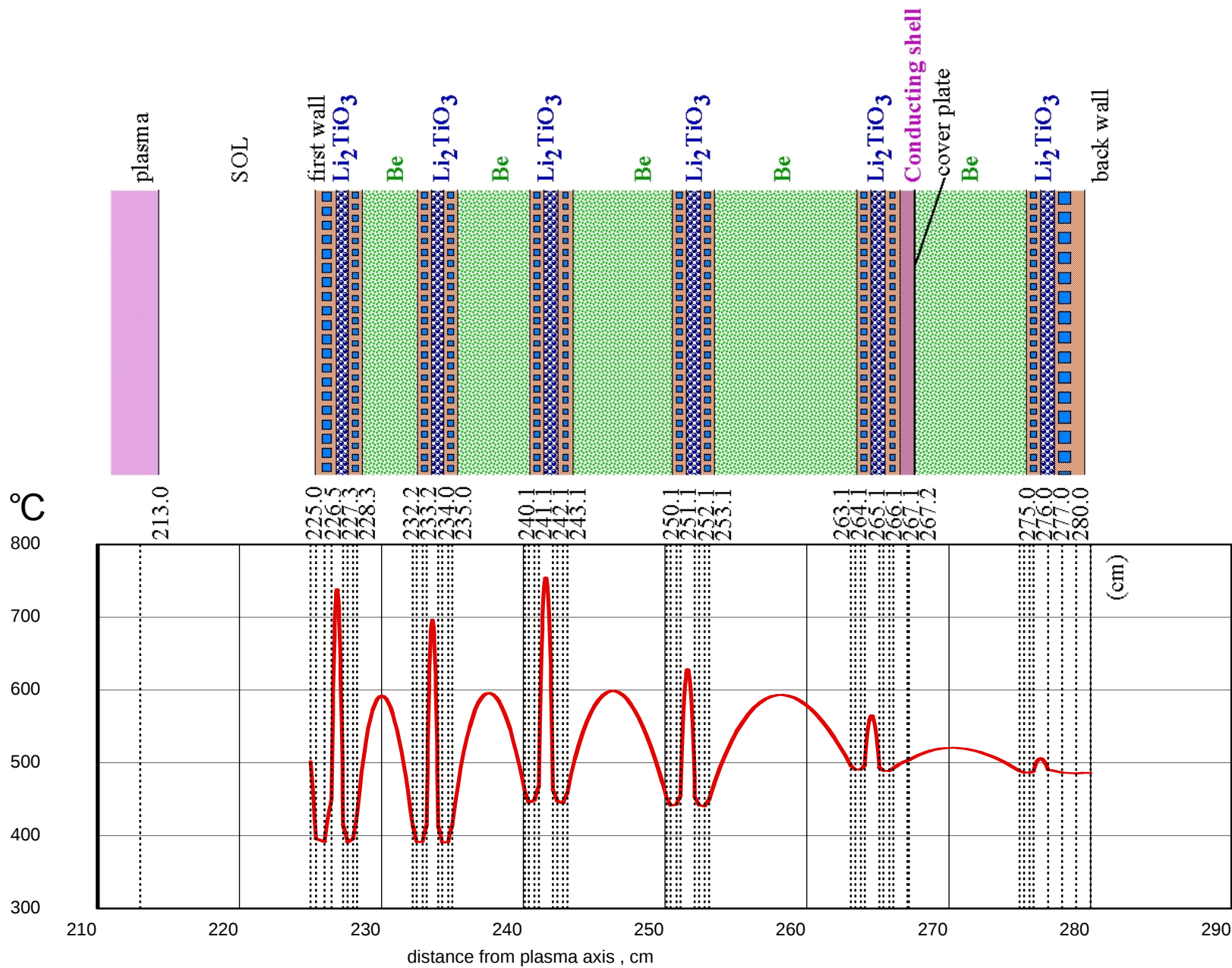
The advanced blanket is intended for high coolant temperature (773K) and stabilization of reversed shear plasma.

ODS-RAF + the super-critical water for higher thermal efficiency. **Local TBR=1.34** (less margin but enough possibly)

Development of structural materials and advanced tokamak studies in parallel with the ITER program are indispensable.

The advance blanket consists of the **ODS-RAF** for structural material and **zirconium** for the conducting shell. Temperature restriction of neutron multiplier is assumed to be increased to 873 K.

The coolant pressure is 25 MPa. Thermal efficiency of more than 40 %. In the case that a reversed shear operation are achieved, the same device would generate 1,090 MWe in net as shown in Table I. Comparison between the basic blanket and the advance blanket is summarized in Table II. A possibility of economical competitiveness of fusion energy may be demonstrated with the advanced blanket and high performance plasma.



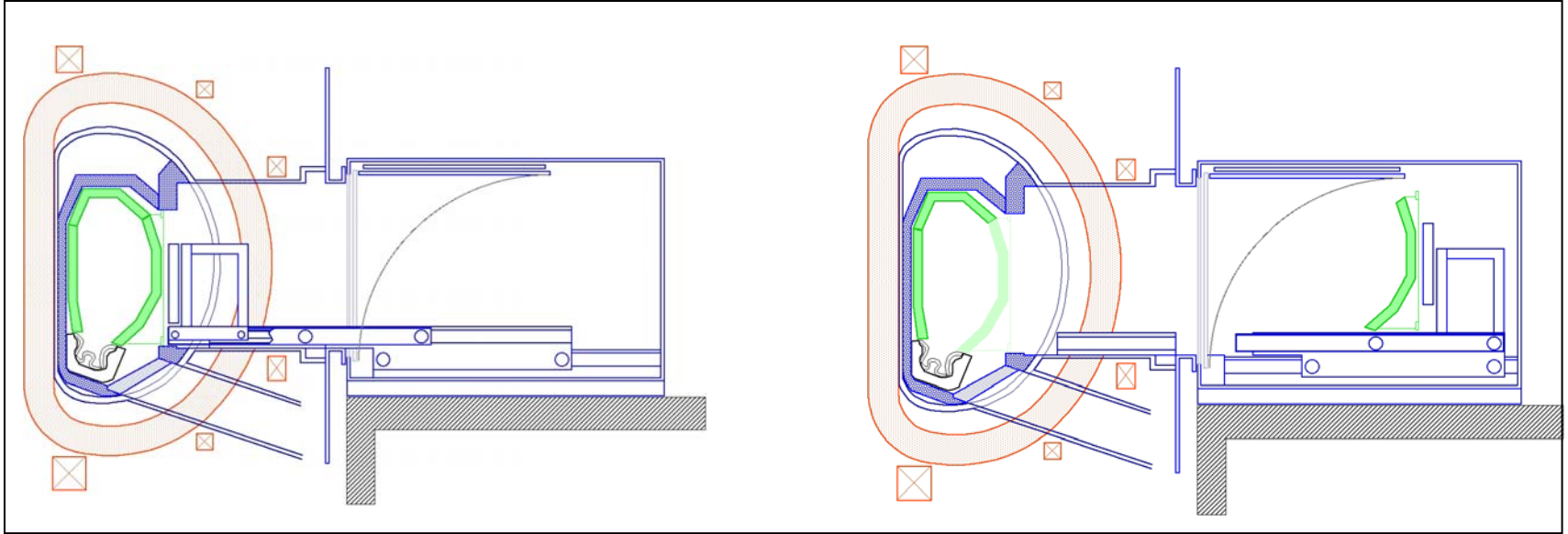
Radial build and temperature distribution of the advanced blanket

	basic blanket	advanced blanket
structure material	RAF	ODS- RAF
breeder material	Li_2TiO_3	Li_2TiO_3
neutron multiplier	Be (<773K)	Be (<873K)
coolant	water (603K, 15MPa)	water (773K, 25MPa)
local TBR	1.48	1.34
thermal efficiency	> 30 %	> 40 %
Note		conducting shell for kink stabilization The steam turbine will be replaced, too.

TABLE II: COMPARISON BETWEEN THE BASIC BLANKET AND THE ADVANCED BLANKET

4.4 Blanket Structure and Maintenance Procedure

An innovative blanket structure and its maintenance procedure are adopted in the Demo-CREST concept.



One sector of the blanket is divided to 3 parts. Weights of the outboard, inboard and upper blanket modules are approximately 40, 15 and 20 tons, respectively. Weight of outboard shield to be removed is approximately 126 tons.

As the outboard blanket and the outboard shields are withdrawn separately, the outboard blanket is supported independently, not supported by the outboard shield.

The divertors are replaced independently through the divertor maintenance port as the interval of divertor replacement is considered to be different from that of the blankets.

The blanket partition and the replacement procedure proposed here would be acceptable from a viewpoint of the plant availability and would be able to show the development path to the **one-sector replacement**, which used in the CREST design.

Additionally, they have advantages of an ability of blanket replacement according to wall load distribution, less reduction of tritium breeding by structural materials, and a capacity to install conducting shell in the outboard blanket for stabilization of reversed shear plasma.

Some additional control coils may be required for control of RWM. In the present design, those are not installed.

5. Conclusions

- 1) We have conducted conceptual study on a fusion power reactor for early demonstration of the electric generation, Demo-CREST.
- 2) Demonstration of electric power generation is possible with minimum extension from the ITER plasma parameters, the ITER technologies, and materials to be developed in parallel with the ITER. Demo-CREST produces up to 3 GW fusion power in R=7.25m, A=3.4 device.
Net electric power of up to 500 MWe with the basic blanket is attainable.
- 3) Net electric power 1,000 MWe is attainable with advanced plasmas ($3.5 < \beta_N < 5.5$) and advanced technologies. But the advanced plasma and materials must be developed in parallel with the ITER.
- 5) An innovative concept of blanket maintenance procedure proposed can secure the tritium self-sufficiency and maintainability.
- 6) The Demo-CREST has a capacity to demonstrate both the net electric power generation in the early stage and a possibility of economical competitiveness in the advanced stage.