

Conceptual design of a demonstration reactor for electric power generation

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Abstract: Conceptual study on a demonstration plant for electric power generation, named Demo-CREST, was conducted based on the consideration that a demo-plant 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 and (2) to have a possibility to show an economical competitiveness with advanced plasma performance and high performance blanket systems. The plasma core was optimized to be a minimum size for both net electric power generation with the ITER basic plasma parameters and commercial-scale generation with advanced plasma parameters, which would be attained by the end of ITER operation. The engineering concept, especially the breeding blanket structure and its maintenance scheme, is also optimized to demonstrate the tritium self-sustainability and maintainability of in-vessel components. Within the plasma performance as planned in the present ITER program, the net electric power from 0 MW to 500 MW is possible with the basic blanket system under the engineering conditions of maximum magnetic field 16 T, NBI system efficiency 50 %, and NBI current drive power restricted to 200 MW. Capacities of stabilization of reversed shear plasma and the high thermal efficiency are additional factors for optimization of the advanced blanket. By replacing the blanket system with the advanced one of higher thermal efficiency, the net electric power of about 1000 MW is also possible so that the economic performance toward the commercial plant can be also examined with Demo-CREST.

1. Introduction

Three development stages, demonstration of a fusion reactor operation, demonstration of electric power generation as a power plant and demonstration of economic and safe performance, are usually assumed for realization of a fusion power plant¹. In each stage, there are inevitable missions to be completed. These missions should be structured under the prudent consideration. What kind of missions in each stage should be included remains to be discussed and has to be structured so as to answer the world demand as soon as possible.

On the other hand, early realization of electric power generation with fusion energy has been discussed, recently. This discussion indicates that the second and third stage mentioned above should be achieved with a single demonstration device. The demonstration device for the second and third stage must be constructed just after or during the ITER program. Conceptual study on such a single device corresponding to the second and third stages for early realization of net electric power generation, named "Demo-CREST" was discussed in the present paper. The CREST is a concept of the compact reversed shear tokamak reactor, which is proposed as a commercial reactor by the authors². However, it is not suitable for a demo-plant next to the ITER.

2. Principles for Demo-CREST design

Major missions of the demonstration reactor for electric power generation are to generate plant-scale electric power, to demonstrate steady and continuous operation, and licensing as a fusion power plant. Taking these missions into consideration, principles for Demo-CREST

design are based on the consideration that a demo-plant 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 and (2) to show a possibility of an economical competitiveness with advanced plasma performance and high performance blanket systems. It may be difficult to two capacities with a single device. The Demo-CREST concept tries to realize it by means of replacing breeding blanket from the basic one to the advanced one.

According to these missions and principles, the guidelines to design the Demo-CREST are decided as follows;

- (1) Minimum extension from the ITER: It means that the net electric power must be guaranteed with minimum extension from ITER plasma parameters and engineering designs based on 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: Electric power as a scale of commercial plant can be generated with advanced plasma parameters attainable during the ITER operation and with advanced technologies and materials. Therefore, the Demo-CREST must have a capacity of wide range of power generation.
- (3) Reasonable plant construction cost: Cost of electricity is not so important. However, the construction and operation cost must be reasonable in order to achieve public acceptance of fusion energy development.

3. Plasma Core of the Demo-CREST

The plasma core was optimized to be a minimum size for both net electric power generation with the ITER basic plasma parameters and commercial-scale power generation with advance plasma parameters, which would be attained by the end of ITER operation. 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 FUSAC code^{3,4}. Operation points mean a set of plasma size (major radius, aspect ratio), major physical parameters, shape and location of reactor components, construction cost, cost of electricity, fusion power, and electric power.

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

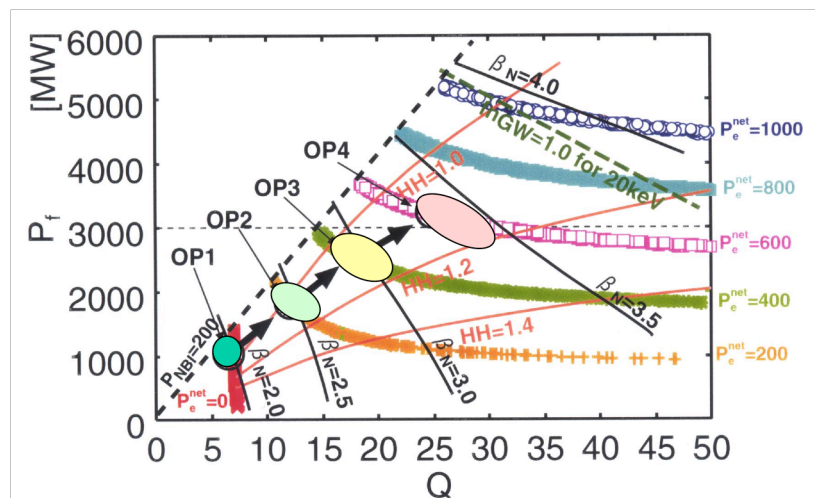


FIG.1. Required plasma performance for electric power generation, in the case of $R=7.25m$.

TABLE I: MAJOR PARAMETERS OF THE Demo-CREST.

		normal shear			reversed shear	
		OP1	OP2	OP3	OP4	
β_N		1.9	2.5	3.0	3.4	4.0
q_0		5.0	5.0	5.0	5.2	6.5
major radius / minor radius (m)				7.25 / 2.13		
a / b				1.85 / 0.35		
f_{nGW}		0.63	0.79	0.87	1.1	1.3
HH		0.96	1.1	1.2	1.2	1.4
P_{NBI} (MW)		188	190	185	191	107
fusion power (MW)		1,260	1,940	2,460	2,840	3,160
basic	gross electric power (MW)	440	650	810	940	N/A
blanket	net electric power (MW)	30	230	390	490	N/A
advanced	gross electric power (MW)	590	870	1,090	1,250	1,350
blanket	net electric power (MW)	230	490	710	850	1,090

the condition of 30% thermal efficiency, 50 % NBI system efficiency, and 16 T maximum magnetic field. FIG.1 shows one of the results of the analysis. 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 $\beta_N > 3.0$ is achieved, The reactor would produce 2.5 GW in fusion power and generate 800 MWe in gross and 400 MWe in net electric power⁴.

As a result, the plasma core of the Demo-CREST is optimized to 7.25 m in major radius and 3.4 in aspect ratio, which is somewhat larger than the ITER. The optimized plasma core will have a capacity of the fusion power of 1,260 MW at low beta ($\beta_N = 1.9$) operation and the fusion power of 2,840 MW at high beta ($\beta_N = 3.4$) operation with plasma current drive power of around 190 MW. At the reversed shear operation ($\beta_N = 4.0$), the fusion power of 3,160 MW is achievable with 107 MW of current drive power. FIG. 2 shows the high β_N reversed shear equilibrium with $\beta_N = 4.0$ at $q_0 = 5.9$ and R_{ext} vs the eigen value ($-\omega^2$) of instability, where n is the toroidal mode number, R_{ext} is the position of shell normalized by the minor radius⁵. However, a conducting shell to stabilize ideal modes is positioned in the breeding blanket in the cases of $\beta_N > 3.5$. Major parameters of each operation points are shown in Table I.

4. Engineering Concept

4.1 Demonstration of tritium self-sufficiency

Demonstration of steady and continuous operation of fusion power plant is indispensable for utilization of the fusion energy. The engineering concept of the Demo-CREST, especially the

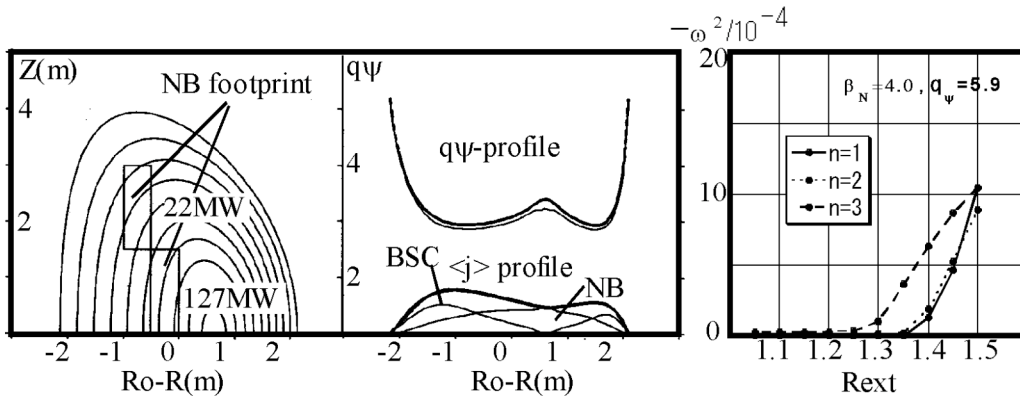


FIG.2. RS equilibrium at $\beta_N = 4.0$ and the stability analysis.

breeding blanket structure and its maintenance scheme, is optimized to demonstrate the steady and continuous operation. Tritium self-sufficiency, net TBR > 1, is one of the most important missions of the power generation demonstration plant. 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 following devices.

Therefore, the blanket design of the demonstration reactor must have a large margin in tritium breeding. For a large margin of tritium breeding, lower neutron wall load, lower blanket temperatures, and high neutron coverage of the blanket are desirable. 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 of the plant. In the case of the Demo-CREST concept, 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

There are two types of breeding blanket for the Demo-CREST. The first one, basic blanket, is for early demonstration of power generation. Relatively low coolant temperature would bring sufficient tritium breeding. The second one, advanced blanket, is for high performance

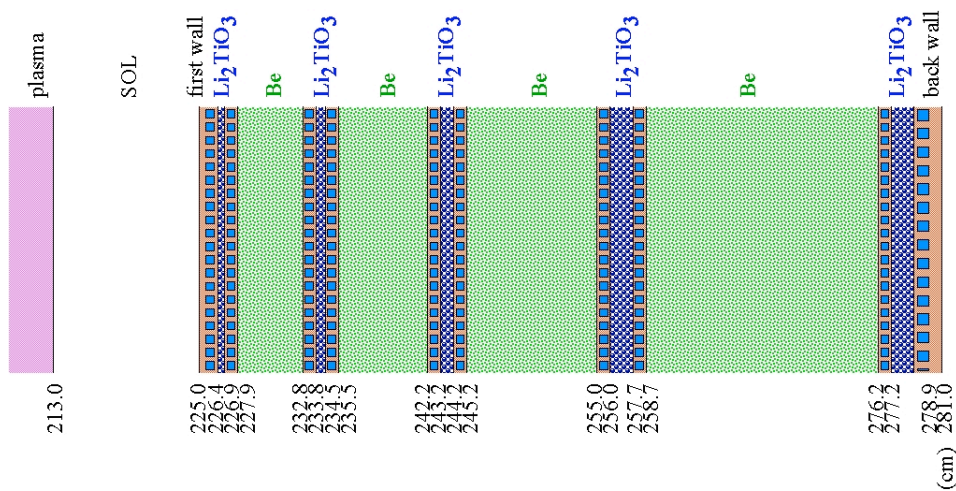


FIG.3. Radial build of the basic blanket of the Demo-CREST.

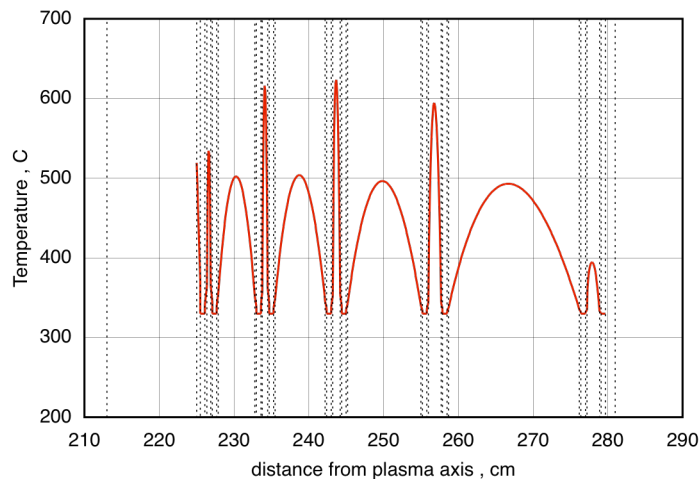


FIG.4. Temperature distribution of the basic blanket of the Demo-CREST.

operation to show a possibility of economical competitiveness. High coolant temperature for high thermal efficiency and installation of conducting shell for reversed shear operation would be required.

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 723 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/m^2 in average, 4.2 MW/m^2 in peak. Mean heat load is approximately 0.68 MW/m^2 .

A radial build of the basic blanket and its temperature distribution are shown in FIGs. 3 and 4. The local tritium breeding ratio of the basic blanket reaches 1.48 with coolant temperature of 603 K. The coolant pressure is 15 MPa. In the case of normal shear-high beta operation, 930 MWe in gross and 490 MWe in net will be generated with thermal efficiency of 30 %. These

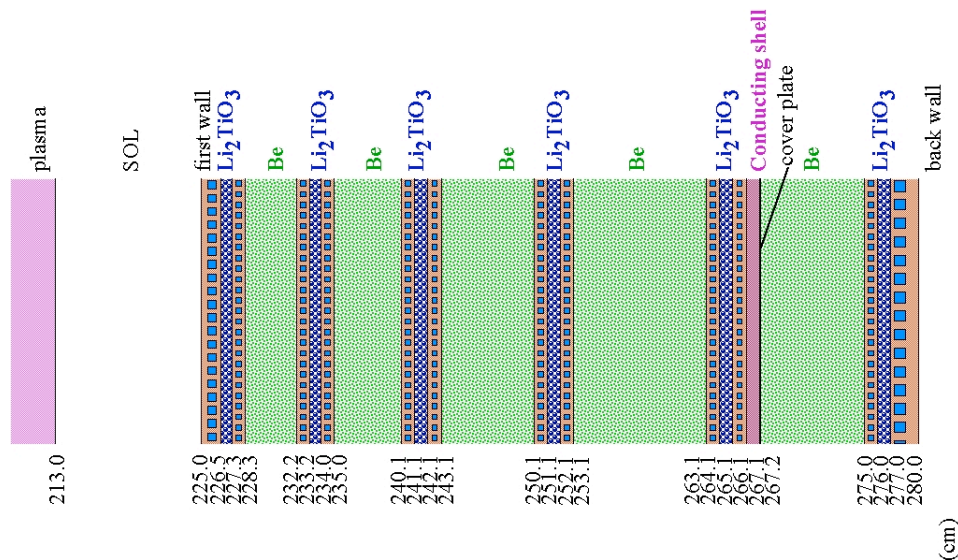


FIG.5. Radial build of the advanced blanket of the Demo-CREST.

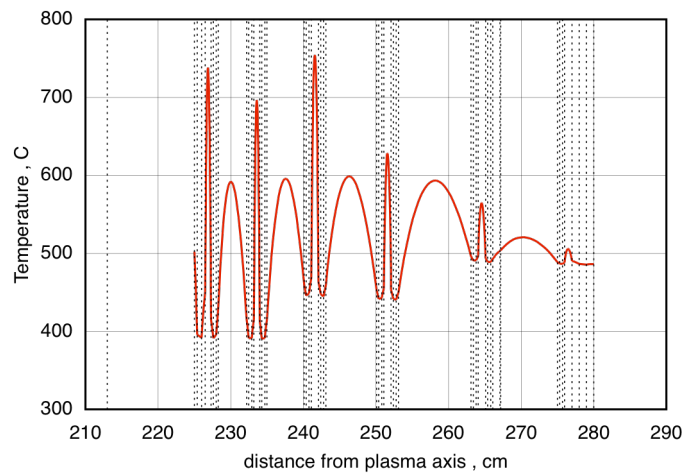


FIG.6. Temperature distribution of the advanced blanket of the Demo-CREST.

TABLE II: COMPARISON BETWEEN THE BASIC BLANKET AND 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

values are sufficient to demonstrate plant-scale power generation, although they are not economically attractive.

4.3 Advanced blanket

In order to show the possibility of economical competitiveness, advanced blanket should be installed to the Demo-CREST. The advanced blanket is intended for high coolant temperature and stabilization of reversed shear plasma. Combination of the ODS-RAF for structural material and the super-critical water for coolant, which are now in progress, are assumed to be attainable for higher thermal efficiency.

A radial build of the advanced blanket and its temperature distribution are shown in FIGs. 5 and 6. The advance blanket consists of the similar materials of the basic blanket except for the oxide dispersion strengthened reduced activation ferritic steel (ODS-RAF) for structural material and zirconium for the conducting shell. Temperature restriction of neutron multiplier is assumed to be increased to 873 K. Evaluation of tritium breeding ratio and blanket temperature are conducted under the similar condition to the basic blanket.

The local tritium breeding ratio of the advanced blanket reaches 1.34 with coolant temperature of 773 K. The coolant pressure is 25 MPa. Thermal efficiency of more than 40 % would be attained⁶. In the case that 40 % of thermal efficiency and 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. Development of structural materials and advanced tokamak studies in parallel with the ITER program are indispensable.

4.4 Blanket Structure and Maintenance Procedure

Simple and reliable blanket replacement procedure is one of the key technologies to utilize fusion energy. Maintainability of the in-vessel components and a possibility of high availability should be shown by the demo plant. Small blanket modules and vehicle type maintenance devices like ITER have a flexibility to replace a small part of blankets. However, long maintenance period to replace all, or a large part, of the blankets might be required and reduction of the net TBR by gaps and dead space between blanket modules, structure materials of blanket side wall might be occur. Large sector blankets and cask type maintenance devices may have advantages of short maintenance period and high net TBR,

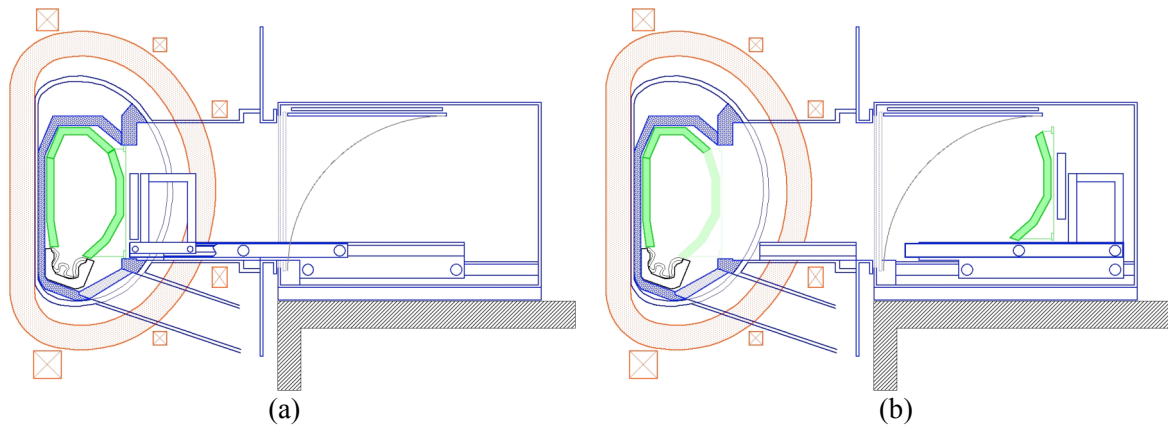


FIG.7. A vertical view of replaced blanket modules and a handling device.

although it has a difficulty of accurate handling of large weight components and a requirement of large TF coils⁷.

A innovative blanket structure and its maintenance procedure are adopted in the Demo-CREST concept. A vertical view of replaced blanket modules and a handling device is shown in FIG. 7. One sector of the blanket, 1/4 of the torus, is divided to three parts. Weights of the outboard, inboard and upper blanket modules are approximately 40, 15 and 20 tons, respectively. The outboard and inboard modules can be taken out through each horizontal port. The upper module is taken down to mid-plane level, and taken out through the port. Although blanket modules are larger than those of the ITER, the flexibility of handling to the toroidal direction is not required. Removable outboard shield, which is removed to open the maintenance port, will be also withdrawn with the similar handling device. 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. 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.

5. Conclusions

We have conducted conceptual study on a fusion power reactor for early demonstration of the electric generation, Demo-CREST. 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 approximately 500 MWe with the basic blanket and approximately 1,000 MWe with advanced plasma parameters and advanced technologies. An innovative concept of blanket maintenance procedure proposed can secure the tritium self-sufficiency and maintainability. 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.

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