Engineering Design of Demo-CREST and Analysis on Critical Development Issues toward Advanced Tokamak Power Plant CREST

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Abstract: A development scenario of the tokamak reactor by three stages (i.e. the experimental reactor such as ITER, a demonstration reactor, and a commercial reactor) is recently being discussed. In order to construct the feasible development strategy, it is necessary to evaluate which component of reactor technologies and to what extent should be developed. From the viewpoint of the future electric supplier, we have proposed the conceptual design of a commercial reactor, CREST and a demonstration reactor, Demo-CREST. On the other hand, the project of the experimental reactor ITER is underway, and its experimental plan and R&D activities are almost completed. Hence, it is most important and reasonable to investigate the demonstration reactor on the track of ITER in order to show a specific development scenario of the tokamak reactor. In this report, we have discussed on engineering aspect in Demo-CREST design, and analyzed the critical development issues toward advanced tokamak CREST. The power flow and power plant system for Demo-CREST are investigated for the improvement of the thermal efficiency in a single devise, and the development issues toward CREST are quantitatively analyzed.

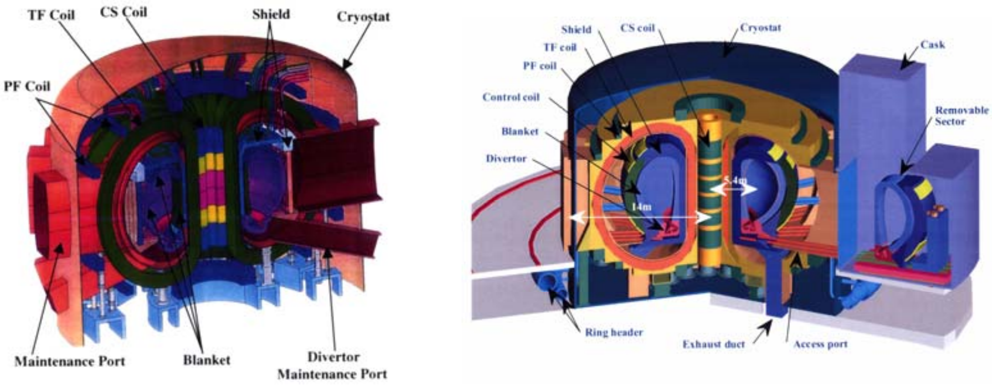
1. Introduction

Now, ITER will be constructed soon, and the realization of fusion energy in the 2030's has been discussed. However, the exploration of an assured path to the commissioning of a fusion power plant in the 2030's has just started. To carry out the effective development toward the next step device after ITER, i.e. the demonstration plant (DEMO), which is also consistent with a commercial plant, the feasibility study based on several development scenarios is required. From the viewpoint of the future electric supplier, we have proposed a commercial plant concept, CREST[1] and a demonstration plant concept, Demo-CREST[2,3]. Indeed, our development scenario of ITER, Demo-CREST and CREST is attractive due to early realization of electric power generation and two steps development from ITER to CREST, but in compensation for the attractiveness, there are critical issues to be resolved after ITER. In this report, we discuss on the power flow and power plant system in Demo-CREST which remained to be solved in the previous report[3], and analyzed the critical development issues toward advanced tokamak CREST in a specific development scenario of the experimental reactor ITER, a demonstration plant Demo-CREST, and a commercial plant CREST.

2. Development Scenario toward CREST

2.1. Concept of demonstration plant : Demo-CREST

The principles for the Demo-CREST design are based on the consideration that a DEMO 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 applicable to CREST. Those requirements are challenging with a single device. The Demo-CREST concept tries to realize it replacing breeding blanket from the basic one to the advanced one. Hence, Demo-CREST has two operation phases, the demonstration phase and the development one[2,3]. The bird's view of Demo-CREST is shown in FIG 1(a).



(a) Demo-CREST (b) CREST

FIG. 1 Bird's views of (a) Demo-CREST and (b) CREST

In the demonstration phase, net electric power generation up to 500MWe by a thermal efficiency η_{th}~30% is demonstrated with moderate plasma performance similar to in the early stage of the ITER operation and minimum extension from the ITER technology. In the development phase, the advanced blanket (which has a conducting wall and higher temperature of outlet coolant) enables higher normalized beta β_N>4.0 and thermal efficiency η_{th}>40%, which are applied to the CREST design. The material advancement from Reduced Activated Ferritic steel (RAF), which is applied to the basic blanket system, to oxide dispersion strengthen RAF (ODS-RAF) is assumed during the demonstration phase, and ODS-RAF is applied to the advanced blanket system for Demo-CREST.

2.2. Concept of commercial plant : CREST

The compact reversed shear tokamak (CREST) is a cost competitive reactor concept based on a reversed shear plasma with a moderate aspect ratio which is similar to the ITER advanced mode plasma[1]. The aspect ration and the plasma configuration of CREST should be as close as possible to the ITER advanced mode plasma, because 'testing by ITER' is an important policy toward structuring a fusion energy development strategy and commissioning in the 2050.

The parametric study on COE has shown that a high β_N and a high η_{th} are required in order to achieve a competitive cost[4]. In the CREST design, such high β_N plasma may be realized with a reversed shear (RS) operation of a tokmak. Current profile control and high plasma rotation by neutral beam current drive (NBCD) stabilized the MHD activity up to the normalized beta value (β_N~5.5) with a closed conductive shell, which is installed in the breeding blanket. For both of Demo-CREST and CREST, ferritic steel materials and a water cooling system have been chosen, because a large database and extensive industrial experience exist for this combination, and therefore it seems to be a reliable path to power reactors, at least to the early generation of fusion reactors following the ITER project. In one word, the maximum potential of plasma performance and reactor technology, which has to be demonstrated in Demo-CREST, is applied to CREST for the economic competitiveness. The bird's views of CREST is shown in FIG 1(b).

2.3. Development scenario of ITER, Demo-CREST and CREST

The outline for the development scenario of ITER, Demo-CREST, and CREST is shown in Table I. This development scenario is characterized by an advanced tokamak plasma reactor with a water cooled-RAF blanket system. In the demonstration phase of Demo-CREST, a net electric power P_{net}~500MWe is attainable with β_N~3.5. These plasma performance parameters can be examined as the advanced operation scenario in ITER[5]. In the development phase,

Table I The relationship between net electric power and technology advancement is shown.OP1, OP2, OP3 and OPRS correspond to the operation point of Demo-CREST.

		Development of Reactor Technology			
		ITER R=6.2m, A=3.1-3.4 κ=1.7~1.85 δ~0.4 B_{tmax}=13T	Demo-CREST R=7.3m, A=3.4 κ=1.85~2.0, δ=0.35~0.5 B_{tmax}=16T **Demonstration phase** $\eta_{th}\geq30\%$	**Development phase** $\eta_{th}\geq40\%$	**CREST** R=5.4m A=3.4 κ=2.0 δ=0.5 B_{tmax}=13T η_{th}=41%
Development of Plasma	ITER Reference β_N~1.9	No power generation	**OP1** 0MWe (P_f~1200MWe)		
	ITER Advanced β_N~3.5	No power generation	**OP2** 500MWe (P_f~3000MW)	**OP3** 900MWe (P_f~3000MW)	
	CREST-like Advanced β_N>4.0			**OPRS** 1100MWe (P_f~3000MW)	1200MWe

the advanced blanket system for higher thermal efficiency enable to increase the net electric power, and conducting walls installed in this blanket system break the road to the more advanced plasma performance such as β_N>4.0. During this development phase, the plasma performance and reactor technologies required for a commercial reactor CREST are investigated. In this development scenario, the role of ITER is considered as follows; the completion of the ITER reference plasma operation contributes to get the outlook for the next demonstration reactor, and the advancement of plasma performance for β_N>3.0 clearly shows that fusion energy becomes the promising candidate of alternative energy sources.

3. Power Flow and Power Plant System for Demo-CREST

An improvement method of thermal efficiency by replacing the blanket system with the advanced one is proposed for Demo-CREST. The power flow and power plant system for Demo-CREST, which remains to be solved in the previous report[3], is investigated here. The total thermal power and the available thermal power from the blanket are estimated at 3910MW and 3347MW under the condition of fusion power 3000MW. In the demonstration phase, the power plant system similar to the pressurized water reactor (PWR) is applied, because of the coolant condition of ITER TBM similar to that of PWR. The generated power is 1054MW, and thermal efficiency of 30% for the demonstration phase in Table I is assured.

In the development phase, higher temperature coolant condition of super critical water is considered. Here we propose the direct cycle system shown in FIG2, where the bypass system of coolant from the blanket outlet is required for the reheater system between the high pressure turbine and the low pressure one. While, we should notice that this effect is found to be negligible on the thermal efficiency. The pressure drop in the blanket is evaluated at 0.38MPa from the effect of friction and inlet/outlet. Taken other effect into account, the pressure drop of 1.0 MPa in the blanket system is assumed. The generated power is 1477MW, and thermal efficiency of 40% for the development phase is assured.

In this improvement method, thermal efficiency can be improved step by step. However, the piping system outside the blanket system has to be previously prepared for the super critical water condition at the plant construction. In addition, the turbine system has to be also replaced. Those issues should be paid attention to in this development scenario.

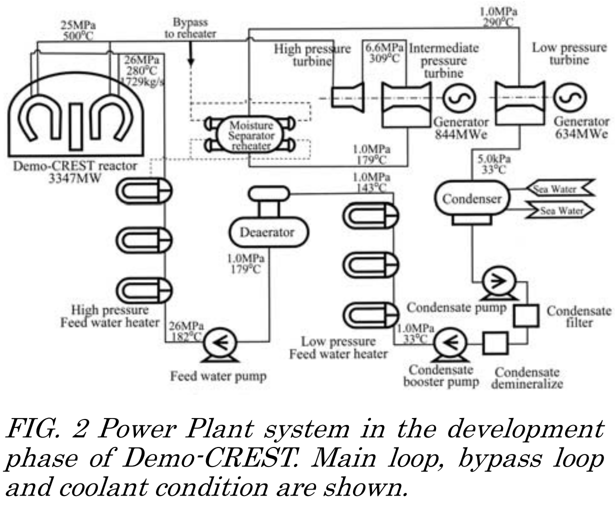


FIG. 2 Power Plant system in the development phase of Demo-CREST. Main loop, bypass loop and coolant condition are shown.

4. Development Issue on Reactor Plasma

4.1. Plasma performance

Plasma performance required in this development scenario is summarized in Table II. In the demonstration phase of Demo-CREST, the plasma performance parameters (β_N, HH, fn_{GW}) completed in ITER are applied to the Demo-CREST operation, step by step. The operation point 1 (OP1) is the starting point of the demonstration phase, and the operation point 4 (OP4) is the last one. The reference operation scenario (Ref.) and a high performance steady stated one (HPSS) proposed in ITER are also shown in Table II. As for β_N and HH, the Demo-CREST parameters are achieved in this HPSS ITER scenario, but fn_{GW} of OP4 for Demo-CREST is a little larger than that of ITER. Hence, the physics of density limit and its attainable region should be examined in the ITER program. In the development phase of Demo-CREST, the β_N value is larger than the ideal wall limit of the present ITER design (β_N~3.8)[12]. Hence, this advanced plasma region should be explored, by other support devices and by itself, and this is why we think Support device is required.

Table II Plasma performance required in this development scenario is shown.

	ITER		Demo-CREST			CREST
	Ref.	HPSS	OP1	OP4	OPRS	
β_N	1.9	3.6	1.9	3.4	4.0-5.5	5.5
HH	1.0	1.53	0.96	1.2	1.40	1.5
fn_{GW}	0.85	0.86	0.56	1.02	1.31	1.3

4.2. Plasma control

In the demonstration phase of Demo-CREST, applied plasma performance is confirmed in the ITER program. Hence, the plasma control issues, such as avoidance of neoclassical tearing mode (NTM), positional instability, major disruption, and so on, are supposed to be established firmly in the ITER program. FIG. 3 shows the MHD stability analysis for Demo-CREST. In the demonstration phase, plasma performance is improved from OP1 to OP4, assisted by the conducting wall at r_{wall}=1.3a just behind the blanket modules. In this phase, major MHD activity limiting plasma beta value is supposed to be NTM and the resistive wall mode (RWM). NTM probably appears even in the low β_N region corresponding to OP1 and OP2[6]. Hence, an effective control method for NTM has to be firmly established in ITER.

When the plasma performance exceeds the no wall limit (OP3, OP4, OPRS), the suppression of RWM has to be considered. Furthermore, for a reversed shear operation (OPRS) in the development phase, the precise current profile control is required. Roughly speaking, the control issues on a current profile of OPRS are summed up as the radial position control of the minimum safety factor (q_{min}) and its avoidance of a rational q surface with the limited control power. The outer radial location of q_{min} is preferable to OPRS, where the broader negative shear region is effective to the ballooning mode and the kink mode is stabilized by conducting walls. Such controllability of current profile should be examined in ITER. On the other hand, the existence of a conducting wall near the plasma surface induces RWM. Two suppression methods of RWM are considered in ITER[7]; the one is by control coils, the other is by plasma rotation. Theoretically, the plasma rotation speed required to suppress RWM is considered as

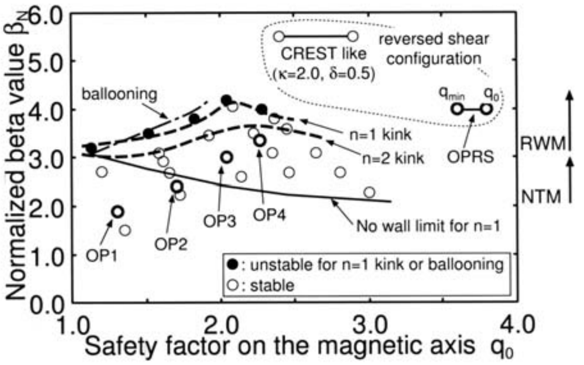


FIG. 3 MHD stability analysis of Demo-CREST on the β_N-q_0 space. The solid line shows the stable limit without conducting wall, and broken lines show the ones with wall at r_{wall}=1.3a. In case of reversed shear configuration, r_{wall}=1.15a is assumed.

several percents of alfven velocity (v_A)[8]. The plasma rotation speed for Demo-CREST is about 1% of v_A at β_N~4.0. According to theoretical predictions, control coils in the vacuum vessel are possibly required. On the other hand, the rotation speed of CREST attains to several percentages of v_A, which is marginal region required to suppress RWM by only plasma rotation. Hence, both methods for RWM should be established in the ITER program, and controllability for RWM only by the plasma rotation should be explored for CREST.

It should be noted that plasma current ramp-up is also a key technology for tokamak power plants. In the Demo-CREST design, 85% of plasma current is induced by CS coils, and other 15% has to be driven by non-inductive method[2]. In the CREST design, non-inductive part of plasma current ramp-up increases up to 50%. The possible operational region for non-inductive current ramp-up should be examined in the ITER program.

4.3. Heat and particle control

In the ITER design, peak power load on the targets is limited to q_{div}<10MW/m^2. This condition should be achieved in Demo-CREST and CREST. One of the key parameters is the upstream SOL density n_s. The higher n_s, the lower heat load on the plates[9]. Hence, one of the control issues is increase of n_s without the degradation of core plasma performance. The radiation power required for q_{div}<10MW/m^2 and its fraction to total heating power for ITER, Demo-CREST (from OP1 to OP4) and CREST are shown in FIG. 2. They gradually increase from ITER to CREST. In the Demo-CREST and CREST designs, n_s~2/3<n_e> (which is applied in FIG. 2) enables to keep q_{div}<10MW/m^2 by using impurity seeding in the SOL region[2], while the ITER design is carried out with the conventional case of n_s~1/3<n_e>[10]. Controllability of n_s and impurity seeding level consistent with core plasma performance has to be precisely examined in ITER, and its operational window should be mapped out for the next step devises.

5. Development Issue on Reactor Technology

5.1. Super conducting coil

In the Demo-CREST design, maximum performance of super conducting coil is 16T 10MA/m^2 for TF coils (15MA/m^2 for CS coils), which is higher maximum magnetic field strength (B_{tmax}) with the same coil current density (J_{sc}) as the ITER design[10]. In the CREST design, B_{tmax}~13T, but higher J_{cs} (twice of the ITER design) is required. For these requirements, a super conductor of Nb_3Al has a good potential[11], and a 13T 10MA/m^2 coil of Nb_3Al has been completed in the ITER R&D program[12]. Hence, the development of higher B_{tmax} with the present J_{sc} is a top priority toward the Demo-CREST design.

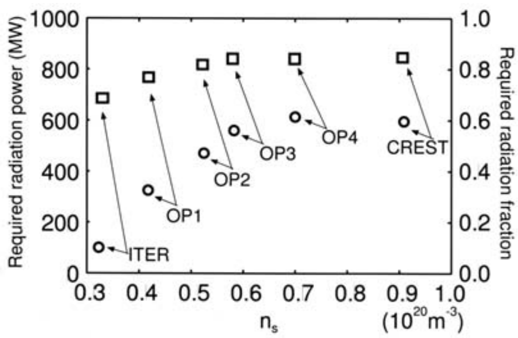
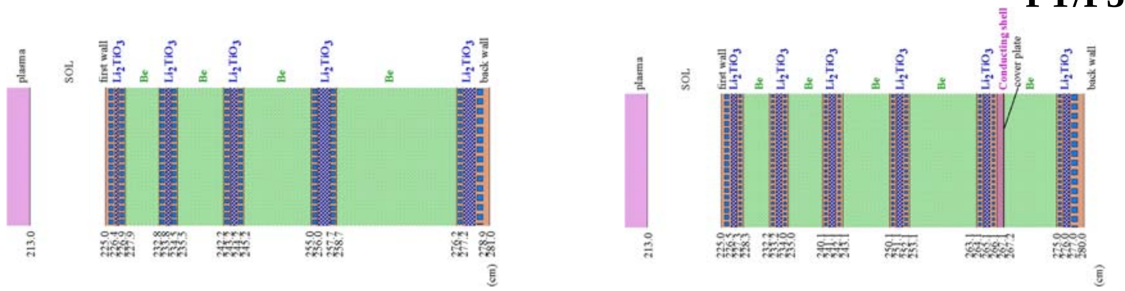


FIG. 4 Total radiation power (circles) and its fraction to total heating power (squares) required to achieve q_{div}<10MW/m$_2$ for ITER, Demo-CREST (OP1 to OP4), and CREST.

5.2. Blanket concept

In the demonstration phase of Demo-CREST, the blanket system should be designed by using the result of ITER test blanket modules (TBM). The critical issues to be demonstrated in this phase are the net electric power generation and the tritium self-sustainability. To ensure these issues, the same outlet coolant condition (15MPa, 603K) as proposed in ITER TBM is applied, and this condition accepts the large breeding zone and the small cooling channel one in the blanket, because



(a) Basic blanket *(b) Advanced blanket*

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

of relatively low temperature. In this blanket concept, the local TBR is estimated at 1.48, which allows the net TBR larger than 1.1. This design enables to attain thermal efficiency more than 30% as mentioned in Sec.3. This value with the ITER reference plasma performance is the starting point to select the core plasma size[2].

In the development phase of Demo-CREST, an advanced coolant condition (25MPa, 773K) with supercritical water, which is also proposed in ITER TBM, is applied. Moreover, conducting walls for higher plasma performance are also installed in the advanced blanket system. On the other hand, this advance concept sacrifices TBR because of the larger zone for coolant channels and a conductor wall shown in FIG 5. The local TBR of this advanced blanket system is TBR~1.34. Whether this local TBR is enough or not should be conformed in the previous demonstration phase. When this advanced concept is found to be acceptable, this concept will be applied to the CREST design. When it is found to be impossible to complete this advanced blanket concept, other blanket concept with a superheated water-cooled system is also proposed as a backup option for the CREST design[1].

5.3. Plasma control and heating device

In this development scenario, we focus on NBI current drive as a main tool and electron cyclotron current drive (ECCD) as an additional one. NBI system efficiency (η_{NBI}) should be as high as possible, because of less circulating power. In the Demo-CREST design, η_{NBI}~50% is applied due to development of plasma neutralizer cell. 1.5MeV of the beam energy is also found to be required in the Demo-CREST, in order to get the flexible operation for the plasma profile control[13]. Moreover, 2.5MeV beam energy is required in CREST, because of high density and current drive in the central plasma region. On the other hand, ECCD is required to suppress NTM. Hence, higher frequency of the gyrotron than the ITER design has to be developed because of higher B_t in the design of Demo-CREST.

5.4. Maintenance method

In the Demo-CREST design, the size of blanket modules should be large enough to shorten the maintenance period in comparison with the ITER case. One sector of the blanket, 1/14 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. By using a handling device, the blanket modules can be taken out through each horizontal port as shown in FIG. 6. The flexibility of handling to the toroidal direction is not required in this maintenance scheme. The handling device is considered as a device scaled up form the one for the ITER shielding plug. The maximum weight to be handled in Demo-CREST is 130 ton of the outer shield, while that in ITER is about 40 ton of the shielding plug[10]. This maintenance scheme has 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 for MHD stabilization.

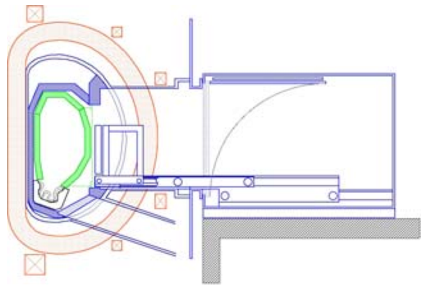


FIG. 6. A vertical view of replaced blanket modules and a handling device of Demo-CREST.

On the other hand, the full sector removal scheme for blanket and divertor systems is applied to the CREST design shown in FIG. 1. The weight of the one sector (1/14 of the torus) is estimated at about 250 ton. This maintenance concept is very effective to the plant availability. In the CREST design, the achievable availability, which is only reduced by the scheduled maintenance of the sector replacement, is estimated at 94%, which enables to achieve more than 80% including an unexpected outage period[14]. However, a system for extraction and attachment of the full sector with precise alignment has to be developed and demonstrated like the ITER maintenance system.

5.5. Development of structure material

The blanket systems proposed in the ITER TBM are applied to the design of Demo-CREST. Hence, RAF is considered to be reasonable, however, it should be noted that the neutron fluence experienced in ITER is not enough for Demo-CREST, and the IFMIF program is indispensable. In the development phase of Demo-CREST, ODS-RAF is probably required as the structure material for higher thermal efficiency. The advanced blanket system made of ODS-RAF also should be demonstrated in the ITER TBM program. In the CREST design, the design condition becomes more severe, because the averaged neutron wall load is 5.0 MW/m^2 (11.3MWa/m^2 for 2.25 FPY), which is larger than 2.7 MW/m^2 (6.1MWa/m^2 for 2.25 FPY) of Demo-CREST. Hence, more advancement of material performance is required during the Demo-CREST operation.

6. Summary

We discussed power flow analysis in Demo-CREST design and the critical development issues for advanced tokamak CREST in the development scenario of ITER, Demo-CREST, and CREST. In Demo-CREST, thermal efficiency can be improved step by step. However, the piping system outside the blanket system has to be previously prepared for the super critical water condition at the plant construction. In addition, the turbine system has to be also replaced.

The critical development issues on this development scenario are also quantitatively analyzed. The main development goals of plasma and reactor technology for Demo-CREST and CREST are summarized in Table III. The essential technologies such as tritium self-sufficiency, steady state operation, material development of reduced activation ferritic steel (RAF) and so on should be firmly established in the ITER and other R&D project. In the demonstration phase of Demo-CREST, there are additionally clear technological gaps from ITER on the divertor performance, B_{tmax}, NBI beam energy, and neutron fluence to first-wall material. The maximum magnetic field B_t=16T is required for early demonstration of electric generation with the ITER reference plasma performance and the smaller major radius than R=8.0m.

In the next step, improvement of β_N and density will be critical issues in the plasma physics in comparison with the present ITER experimental plan. Those improvements over the ITER plasma performance may be explored by itself. The advancement of NBI energy is also required for the precise current profile control. The development of ODS-RAF is supposed to be also required. When the CREST plasma (β_N~5.5) is demonstrated in Demo-CREST, the magnetic field has to be reduce to Bt=10 T to keep the fusion power Pf~3000MW, because of

Table III Main development goals required for Demo-CREST and CREST. The shaded columns show the technological gaps after the ITER program

Category	Issue	Demo-CREST		CREST
		Demo. Phase	Dev. Phase	
Plasma control	steady state operation	need	need	need
	bootstrap current fraction	0.5	0.65~0.73	0.83
	RWM control	n=1	n>1	n>1
	Fraction of non-inductive current ramp-up(%)	20	20	50
Plasma performance	β_N	3.4	4.0~5.5	5.5
	HH factor	1.2	1.4	1.5
	Density ratio to n_{GW}	1.0	1.3	1.3
Divertor perforamance	Required radiation power(MW) / fraction (%)	580/82	580/82	588/85

Category	Issue	Demo-CREST		CREST
		Demo. Phase	Dev. Phase	
Tritium	Net tritium breeding ratio	>1.0	>1.0	>1.08
Blanket	Temperature(K)/Pressure(MPa)	600/15	780/25	←
	Thermal efficiency (%)	>30	>40	←
Coil	B_{tmax}(T)/ coil current density(MA/m^2)	16/15	16/15	13/30
NBI	Beam energy (MeV)	>1.5	>1.5	2.5
Maintenance	Scheme	Large module	Large module	Sector
	Plant availability (%)	>65	>75	>85
Material	Material	RAF	ODS RAF	ODS RAF
	Fluence (MWa/m^2)	6-9	6-9	10-15

heat removal capacity. The technology for CREST is demonstrated here, however, cost of electricity (COE) of Demo-CREST is estimated at about twice of the target COE 12-13 yen/kWh for CREST under the condition of successful operation.

In the last step of CREST, under the condition of the shield thickness of 1.4m and profile controllability for reversed shear plasma, the operation window for β_N=5.5 of P_f=3000MW and 16 T is found to be beyond the engineering restriction of neutron wall load of 5MW/m2. This is the reason why 13T is applied to CREST, and the technology of 16T superconductor (e.g.Nb_3Al) is effectively applied to increase the coil current density for the compact coil size under the condition of the lower magnetic field of B_t=13T. Twice of the coil current density in the ITER design is required for CREST. As for the development of structure material, it is necessary to demonstrate the neutron fluence of structure material up to ~15MWa/m^2, in order to ensure our development scenario ITER/Demo-CREST/CREST.

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