Plasma Commissioning Scenario and Initial Tritium Inventory for Demo-CREST

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Abstract. This paper discusses the plasma commissioning scenario and the relationship between the initial tritium inventory and the commissioning period for a demonstration reactor concept Demo-CREST. The tritium density ratio (T-ratio) control is applied to keep the high density operation preferable for divertor heat-handling during gradual increase of the fusion power in the commissioning phase. It is found that Demo-CREST can start from zero fusion power operation with T-ratio fn_T~0%, in which the divertor heat-handling condition on the SOL density and the radiation power required for divertor heat load less than 10MW/m^2 is similar to that of the ITER steady state operation. An operation route keeping high density by the T-ratio control is also proposed for the commissioning period. This proposed operation route has a consistency with the start-up scenario without the initial tritium loading, and the relationship between the initial tritium inventory and the commissioning period is also evaluated for Demo-CREST

1. Introduction

Demo-CREST is a conceptual design of DEMO proposed by CRIEPI in Japan(Fig.1)[1,2], and Table 1 shows its major parameters. This DEMO concept has two important missions and corresponding two plant operation phases, the demonstration phase and the development one. The first mission for the demonstration phase is to demonstrate net electric power generation in a plant scale as soon as possible. The second one for the development phase is to demonstrate each performance of components required to get the economic competitiveness in the energy market. In the demonstration phase, net electric power up to 500MWe by a thermal efficiency η_{th}~30% is planned with moderate plasma performance similar to that 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 can be applied to a commercial plant such as the CREST design[3]. This two phase strategy is unique, because inevitable blanket replacement in a fusion power plant is considered as a merit in this DEMO concept.

In the first operation phase of Demo-CREST, there should be checklists for plasma performance, plasma control system, blanket and divertor system, tritium fuel system, and so on. To confirm those checklists, the plasma operation in the commissioning phase is supposed to start with a small fusion power and to increase to the rating(full) fusion power, step by step. Hence, operation flexibility from a small fusion power to the rating one is required for

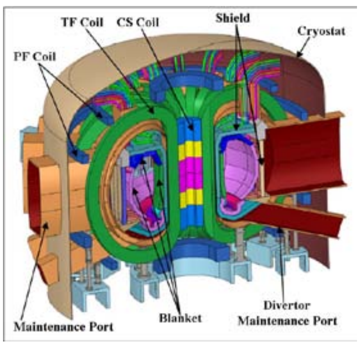

FIG.1. Bird's-eye view of Demo-CREST

TABLE.1: PLASMA PARAMERES FOR DEMO-CREST

Major radius (m)	7.25
Toroidal field (T)	8.0
Plasma current (MA)	13.2 ~ 15.9
Normalized β	1.8 ~ 4.0
HH value	1.0 ~ 1.2
Density ratio to n_{GW}	0.56 ~ 1.3
Ratio of bootstrap current	0.24 ~ 0.73
Current drive power (MW)	107 ~ 190
Fusion power (GW)	1.3 ~ 3.2
Net electric power (GWe)	0.0 ~ 1.1

DEMO. However, such operation flexibility has not been discussed in detail in the conceptual design study for DEMO.

In the previous analysis for the Demo-CREST concept, core plasma property of MHD stability and current drive have been investigated by 2D analysis codes, while the divertor plasma operation to reduce the heat load is pointed out as one of the most important issues in the plasma engineering[1]. Especially, the partial load operation point is more severe on the divertor heat handling, because the partial load operation of Demo-CREST has a larger fusion power($P_f = 1.2$GW) and a lower plasma density($\langle n_e \rangle = 0.62 \times 10^{20}$m^{-3}) than ITER. Hence, when the operation starts even from a small fusion power, it is preferable to keep as high density as the rating operation point for the divertor heat-handling.

In this study, we propose the control of the ratio of tritium density to the fuel density (T-ratio control) to keep the high density for the partial load operation. The T-ratio control for a commercial plant has already been proposed for plasma commissioning scenario without initial tritium loading in the CREST design[4],[5]. However, the consistency between fusion power control and divertor heat-handling, which means the high density operation for the divertor heat-handling, was not taken into consideration in CREST. Here, T-ratio control is applied to Demo-CREST, and we investigate the core plasma operation to keep high density even for the partial load operation. Furthermore, the divertor transport simulation was carried out to get the outlook for the divertor heat load less than 10MW/m^2. Finally, taking into consideration the plasma commissioning scenario without the initial tritium loading, the initial tritium inventory and the commissioning period is also evaluated for Demo-CREST. In the following section, the core plasma operation is investigated, and the operation route for commissioning is proposed. In the third, the divertor plasma condition along the operation route mentioned in the previous section is analyzed, and the fourth section discusses the relationship between the initial tritium inventory and the commissioning period. The final section is summary and future issues.

2. Core Plasma Operation

2.1. Basic Principle and Calculation Method

The total fusion power P_f is roughly determined from the following equation,

$$P_f \approx n_D n_T \langle \sigma v \rangle_{DT} Q_{DT} V_{plasma} = \left[-\left(\frac{n_T}{n_i} - \frac{1}{2} \right)^2 + \frac{1}{4} \right] n_i{}^2 \langle \sigma v \rangle_{DT} Q_{DT} V_{plasma} \quad (1)$$

where n_D, $n_T$and $n_i (= n_D + n_T)$ are the tritium, deuterium and fuel density, respectively. $\langle \sigma v \rangle_{DT}$, $Q_{DT} (= 17.58\ [\mathrm{MeV}])$, and $V_{plasma}$are the fusion rate, the unit fusion power per reaction, and the plasma volume. When we control the fusion power, the fuel density n_i in eq.(1) is usually a control parameter. On the other hand, the fusion power can be also controlled through T-ratio $fn_T (= n_T / n_i)$ with a constant fuel density. This is the basic idea to keep high density for divertor heat-handling when the fusion power (or heating power) is increased step by step in the initial commissioning phase of DEMO. In the analysis, the profiles for pressure, temperature and plasma current are considered as input parameters, and not only fusion power with the T-ratio control but also other plasma property such as MHD stability are investigated.

In this study, the consistency between MHD stability and current drive property has been investigated including the T-ratio control, when an operation point of Demo-CREST is chosen. We have to consider the relationship among three plasma profiles for pressure, plasma current, and NBI injection. Fig. 2 shows the analysis on how to confirm the core plasma operation point. From the viewpoint of MHD stability, consistency between pressure profile and current one is analyzed by 2D-MHD stability code ERATO[6]. Those results in

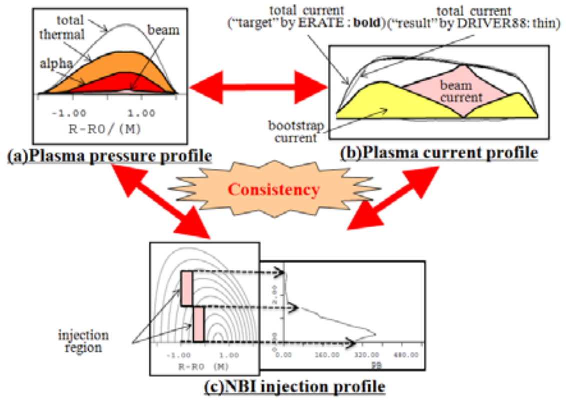

FIG.2 Analysis on consistency among pressure, current and NBI profile by ERATO & DRIVER88

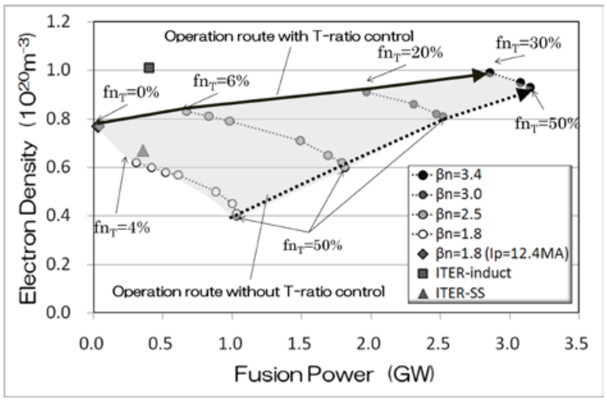

FIG.3 Operational space with T-ratio control for Demo-CREST and ITER operation points

Fig.2 are corresponded to the rating operation point for fusion power $P_f \approx 3.0\text{GW}$ and normalize beta value $\beta_N = 3.4$ of Demo-CREST.

In this analysis, major input parameters are the profiles of pressure and plasma current. When the pressure profile is given, whether a current profile is stable or not from the viewpoint of MHD stability has to be checked. When the current profile is consistent, that current profile is considered as the target profile for NBI current drive in Fig.2(b). Next, we carried out the analysis on current drive property by the current drive analysis code DRIVER88[7]. When the NBI input region and its power are given as shown in Fig.2(c), the resultant driven current profile is analyzed. In this analysis, we have two NBI input regions for on-axis and off-axis region. The bootstrap current profile is also calculated from the density and temperature profiles, and the total current profile is obtained. This total current profile is corresponding to the result in current profile of Fig.2(b). We have to choose the pressure profile, and NBI injection power for off- and on-axis region so that the target and the resultant current profile are overlapped each other(Fig.2(b) purposely shows the result which remains to be optimized to overlap the target and the resultant current profile).

Fusion power is also calculated from the density and temperature profile in DRIVER88. In this calculation, not only D-T fusion but also D-D fusion is considered. Usually, D-D fusion does not contribute to the total fusion power, but in case of small T-ratio like $fn_T \leq 1\%$ it becomes a large effect on the total fusion power and the tritium breeding discussed in Sec.4. Such a small T-ratio is within the target of this study. That is why D-D fusion has been included in the analysis.

2.2. Operational Space and Commissioning Route

The operational points for Demo-CREST are investigated based on the analysis method mentioned in the previous section. Fig. 3 shows the operational space on the fusion power and the averaged electron density for Demo-CREST. The shaded region consists of operational points of the normalized beta values β_N from 1.8 to 3.4 with the total plasma current $I_p = 15.6\text{MW}$. The operation points on the lower and right-hand boundary correspond to the T-ratio $fn_T = 50\%$. In the Demo-CREST concept, the plant commissioning is planned to start from the plasma performance similar to the ITER standard operation of $\beta_N \sim 1.8$. That plasma performance with $fn_T = 50\%$ results in the fusion power $P_f \sim 1.0\,\text{GW}$ and the minimum density of $\langle n_e \rangle = 0.4 \times 10^{20}\text{m}^{-3}$. When β_N is increased, the fusion power and the averaged electron density are also increased along the operation route without T-ratio control in Fig.3. When the normalized beta value $\beta_N \sim 3.4$ is attained, the fusion power becomes about 3.0GW. This operation path was proposed in the previous study[1].

(a)β_N=3.4 (b)β_N=1.8

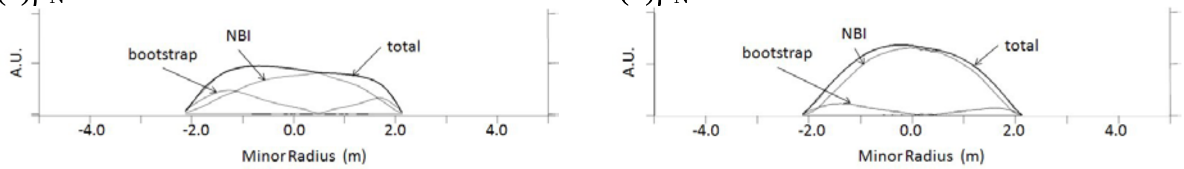

FIG.4 Current profiles for (a) β_N=3.4 case and (b) β_N=1.8 case for T-ratio fn_T=30%

With T-ratio control($0\% \le \mathrm{fn_T} \le 50\%$), operational density can be increased as T-ratio reduces under the condition of a constant β_N. In case of $\beta_N \sim 1.8$, the operation point with $\langle n_e \rangle = 0.6 \times 10^{20} \mathrm{m}^{-3}$ and $P_f \sim 0.4\mathrm{GW}$ is found under the condition of $\mathrm{fn_T} = 4\%$. This operation point is very close to the ITER steady state(ITER-SS) condition with $\mathrm{fn_T} = 50\%$. Furthermore, when the total plasma current is reduced to $I_p = 12.4\mathrm{MA}$, the operation point with $\langle n_e \rangle = 0.8 \times 10^{20} \mathrm{m}^{-3}$ and $P_f = 0.0\mathrm{GW}$ is also found under the condition of $\mathrm{fn_T} \sim 0\%$. Heating power of this operation point is only the current drive power of about 200MW. This operation point is the starting point on the operation route with T-ratio control from $\mathrm{fn_T} = 0\%$ to 50% in Fig.3.

In case of $\beta_N \sim 3.4$, reduction of T-ratio is not easier than the small β_N case, because the fraction of bootstrap current is large. Fig. 4 shows the current profiles of the operational point for (a) $\beta_N \sim 3.4$ and (b) $\beta_N \sim 1.8$ with T-ratio $\mathrm{fn_T} = 30\%$. The total current profile for $\beta_N \sim 3.4$ is almost flat, but safety factor profile does not have a reversed shear profile. In the periphery region(minor radius r~1.5 m), the bootstrap contribution is large, and the contribution of NBI current drive is small in comparison with the central region. If T-ratio would be reduced less than $\mathrm{fn_T} \le 30\%$, more current drive power was required to keep the current density in the central region, and this would result in the overdriven current in the periphery region. Moreover, the high beta plasma like $\beta_N \sim 3.4$ does not have operation margin enough to change the current profile from the view point of MHD stability, because increase of current density in the periphery region (which would result in a reversed shear q profile) probably requires the nearer conducting wall than that of the present blanket design[2]. This is why lower T-ratio operation with $\beta_N = 3.4$ is difficult to find. On the other hand, the low beta case (e.g. $\beta_N = 1.8$) has a smaller bootstrap fraction than that of $\beta_N = 3.4$, and to find operation point with lower T-ratio is not so difficult.

2.3. Critical Issue on Core Plasma Performance

In the operation route with the T-ratio control, the critical plasma parameter is the confinement improvement factor HH. The high plasma confinement HH=1.57 similar to the ITER steady-state operation is needed for the low T-ratio operation less than $\mathrm{fn_T} \le 10\%$. As T-ratio increases, HH decreases to HH=1.17 at the rating operation. On the other hand, the ratio to the Greenwald density limit $\mathrm{fn_{GW}} \sim 1.0$ is kept through the operation route. This fact suggests that exploring the high confinement performance with the high density is important to keep the operation flexibility for not only the rating operation but also the commissioning and the partial load one. The LH transition threshold power from 50MW to 90MW is required along the operation route. The relatively large current drive power 200MW, which is installed in Demo-CREST[1], is sufficient for the H-mode operation through the operation route with T-ratio control.

3. Divertor Plasma Operation

3.1. Operation Condition for SOL Density and Impurity Radiation

The divertor heat load condition was also evaluated by a simple core-SOL-divertor model based on the 0-D core plasma model and the two-point SOL-divertor model[8]. Fig. 5 shows the required radiation power for the divertor heat load less than $10MW/m^2$ along the operation route with and without T-ratio control. The required radiation power along the operation route in Fig.3 and Fig.5 increases with the increase of the SOL density and the fusion power. The initial 1.0GW operation point without the T-ratio control requires about 250MW of radiation power with SOL density $0.13x10^{20}m^{-3}$, under the ITER condition of $n_{sol}=\langle n_e\rangle/3$[9]. Even if $n_{sol}=\langle n_e\rangle/2$ is possible, the SOL density is still smaller than that of ITER. Those divertor operation conditions are more difficult than that of the ITER divertor condition. On the other hand, the initial operation point on the operation route with T-ratio control requires the smaller radiation power 95MW with higher SOL density $0.26x10^{20}m^{-3}$ than the fn_T=50% case in Fig.5. The SOL density is also close to that in the ITER steady state operation. If $n_{sol}=\langle n_e\rangle/2$ is possible, we could start from the radiation power 80MW with higher SOL density $0.39x10^{20}m^{-3}$, which is close to the condition of the ITER inductive operation. Correspondingly, reduction of T-ratio for other β_N operation points expands the operational space to the higher density region in Fig.3. Finally, a control method of divertor plasma start-up assisted by T-ratio control is proposed as the operational route with T-ratio control shown in Fig.3 and Fig.5.

3.2. Heat Load Analysis on the Divertor Plate

To confirm the divertor plasma operation, a two dimensional transport simulation was carried out by SOLPS5.0[10]. Here, we consider the same heat load limit ($\leq$10 MW/m^2) as ITER. In the previous subsection, the operation route is proposed so that the plasma density is kept constant. For example, along the operation route with T-ratio control, SOL density around $n_{sol} = 0.25\sim0.30 \times 10^{10}\ m^{-3}$ for $n_{sol}=\langle n_e\rangle/3$ or $n_{sol} = 0.4\sim0.5 \times 10^{10}\ m^{-3}$ for $n_{sol}=\langle n_e\rangle/2$ is not so changed, while the required radiation power(which is proportional to the total heating power) increased. Hence, if the peak heat load less than $10MW/m^2$ would be possible in the most critical divertor condition at the rating operation point, it could be expected that lower heating power points for partial load operation points were also seemed to be operational. Hence, we carried out the divertor transport simulation for the rating operation point of Demo-CREST as the most severe case.

Fig.6 shows the calculation results by SOLPS5.0, and the V-shape divertor configuration like ITER is applied here[11].The boundary condition on the core plasma interface are as follows: the energy input from core plasma 600MW corresponding to the rating operation point, ion density $0.8 \times 10^{20}\ m^{-3}$, He fraction 5%. The Neon(Ne) puffing of $S_{Ne} = 5.0 \times 10^{21}\ s^{-1}$ is carried out near from the baffle region. Fig.6 shows the heat flux profile on the

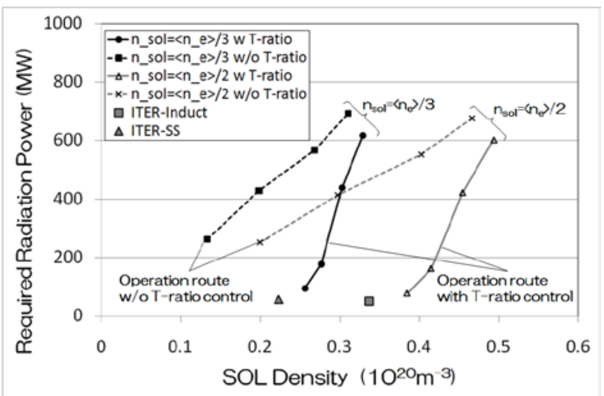

FIG.5 Radiation power required for divertor heat load less than $10MW/m^2$ is delineated along the operation route with or without T-ratio control. Two cases of SOL density condition ($n_{sol} = \langle n_e\rangle/3$ and $n_{sol} = \langle n_e\rangle/2$) are shown.

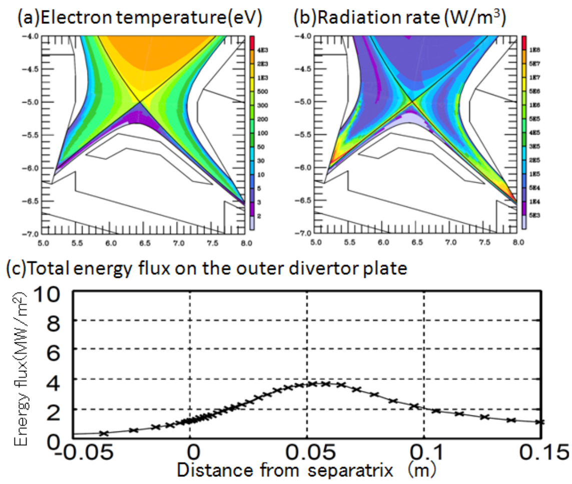

FIG.6 Results of SOLPS, (a) temperature, (b) radiation rate, and (c) energy flux on the outer divertor plate

outer divertor plate and the contour plots of electron temperature and radiation density. In Fig.6(a), the electron temperature near the divertor plate becomes less than 5eV due to the large impurity radiation of Fig6.(b). Consequently, the divertor heat flux less than $10MW/m^2$ is attained, and the outlook for the divertor control for heat flux less than $10MW/m^2$ is found to be promising. However, the optimization for the divertor plasma control still remains to be done. Especially, Ne puff rate has to be optimized, because effective charge number ($Z_{eff}^{SOLPS} \sim 3.0$) in the core plasma area is a little larger than that of the core plasma analysis($Z_{eff}^{core} \sim 2.1$).

4. Tritium Inventory for Commissioning
4.1. Model on Tritium Fuel Cycle

The operation route with T-ration control proposed in this paper has a potential to start-up without initial tritium loading[4],[5]. The DD reactions in a DT-oriented fusion reactor with external power injection by neutral beams produce tritium and neutrons. Tritium produced by the DD reaction together with that produced in the blanket by the 2.45 MeV neutron is re-circulated into the plasma. Then, the DT reaction rate increases gradually, as tritium concentration in plasma builds up towards the level of the rating operation.

The following equation shows the time-differential equation for the tritium particle balance in the core plasma region.

$$\frac{d}{dt} N_T^{pls} = -\frac{N_T^{pls}}{\tau_{p\_eff}^{pls}} - S_T^{DT} + \frac{1}{2} S_T^{DD} + S_T^{fuel} \qquad (2)$$

where N_T^{pls} and $\tau_{p\_eff}^{pls}$ are the amount of tritium and the effective particle confinement time. S_T^{DT} and S_T^{DD} are the loss and source term by the DT and DD fusion reaction. S_T^{fuel} is the source term by fueling, which includes the recovered tritium from the exhaust and the blanket system. Here, eq.(1) shows that half of DD fusion reaction produces the tritium source. In the previous study, the effective confinement time $\tau_{p\_eff}^{pls}$ is assumed to be a constant value (10 sec)[5]. In this study, $\tau_{p\_eff}^{pls}$ is assumed to be proportional to the H-mode energy confinement time $\tau_{p\_eff}^{pls} \propto \tau_E^{IPB(y,2)}/(1-R)$, where R is the particle recycling rate. Accordingly, $\tau_{p\_eff}^{pls}$ is calculated from the core plasma parameters.

The "dead-inventory", which is the amount of tritium retained on/in the facing material of the plasma and the process components, cannot be used in the reaction or processing. The dead inventory in the core plasma region is evaluated in the following two models in this paper. In the first model(saturation case), the total number of the tritium dead inventory D_T^{pls} has a certain saturated level as in the previous study[5], and the maximum dead inventory $D_{T,max}^{pls}$ is assumed to be 900g for Demo-CREST. The time-differential equation for the dead inventory is defined as follows.

$$\frac{d}{dt} D_T^{pls} = \beta_{pls} \left(D_{T,max}^{pls} \frac{N_T^{pls}}{N_{T,max}^{pls}} - D_T^{pls} \right) - \lambda D_T^{pls} \qquad (3)$$

In this model, tritium is assumed to become the dead inventory by isotopic exchange, because the dead inventory is initially saturated with deuterium. Amount of the maximum dead inventory $D_{T,max}^{pls}$ and the time constant of isotopic exchange rate β_{pls} are considered as variable parameters according to ref.[5]. These parameters applied here are shown in Table 2. The tritium also decays with the decay constant $\lambda = 1.79 \times 10^{-9} s^{-1}$. The dead inventory for other subsystems are also evaluated based on this principle[5], and the maximum values for other subsystems are also shown in Table2.

TABLE.II: MEAN RSSIDENTIAL TIME AND DEAD INVENTORY IN EACH SUBSYSTEM

	Mean residential time	Max. dead inventory	Exchange reaction rate
Plasma	10sec (at the rating operation)	900g	$1.16\times10^{-6}s^{-1}$(10%/day)
Blanket & tritium recovery	1.0 day	360g	$1.16\times10^{-6}s^{-1}$(10%/day)
Exhaust & fuel clean-up	1.0 hour	125g	$1.16\times10^{-6}s^{-1}$(10%/day)
Isotope separation	1.0 hour	100g	$1.16\times10^{-6}s^{-1}$(10%/day)
Storage & fueling	1.0 hour	75g	$1.16\times10^{-6}s^{-1}$(10%/day)

In the second model(no-saturation case), there assumed to be no saturated level as reported in the evaluation of the tritium inventory in ITER[12]. Recently, the amount of the retained tritium inventory for the several wall materials such as carbon(C), beryllium(Be), and tungsten(W) was evaluated based on the present database. Here, we assume the W wall for Demo-CREST, and the fitting function $f_{dead}^{ITER}(t)$ for the total retained tritium inventory of the W wall for ITER is estimated as $f_{dead}^{ITER}(t) = C_{dead}t^{0.67}$ $(1.51\times10^{21} \le C_{dead} \le 3.75\times10^{21})$ based on the figure 8 of ref.[12]. We assume the retained tritium inventory is proportional to the wall area and the total tritium fluence, and the dead inventory of plasma region for Demo-CREST is estimated as follows.

$$D_T^{pls}(t) = f_{dead}^{ITER}(t) \times \frac{S_{wall}^{Demo-CREST}}{S_{wall}^{ITER}} \times \frac{\int_0^t fn_T(t')dt'}{\int_0^t fn_T^{rating}dt'} \qquad (4)$$

where S_{wall}^{ITER} and $S_{wall}^{Demo-CREST}$ are the surface of the first wall for ITER and Demo-CREST, and $fn_T(t)$ and fn_T^{rating} is the T-ratio at t (sec) derived from the core plasma analysis and the rating T-ratio($fn_T^{rating} = 0.5$). In this case, the dead inventory for other subsystems is temporarily assumed to have a saturation level based on eq.(2), because the amount of dead inventory in the core plasma region is supposed to be dominant among subsystems. Hence, we focus on discussing the effect of the dead inventory in the plasma region on the initial tritium inventory and the commissioning period.

4.2. Initial Tritium Loading and Commissioning Period for Demo-CREST

Fig.7 shows time evolution of fusion power(P_f) for the initial tritium loading T_{init}=0.0, 0.5, 1.0kg and the dead inventory in the plasma region(D_T) for T_{init}=0.0kg. The lines correspond to the saturation case for eq.(3), and the shaded zones correspond to the no-saturation case for eq.(4). The no-saturation case has the width according to ref.[12]. In Fig.7, the net TBR 1.05 is assumed. In the saturation case, without the initial tritium loading (T_{init}=0.0kg), the commissioning period of 107days is required to achieve the rating operation. As the initial tritium loading increases (0.0 → 0.5 → 1.0kg), that period becomes short (107 → 67 → 43days). Finally, the initial tritium loading of 1.6kg is enough to start with the full power operation. In the no-saturation case, the commissioning period without the initial tritium loading is 75~91days, and shorter than

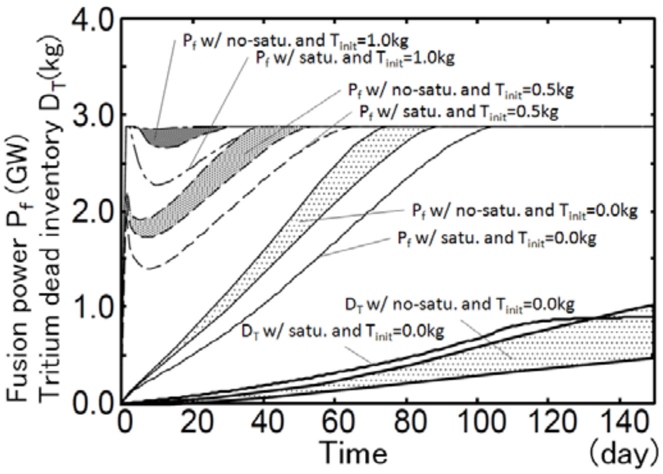

FIG.7 Time evolution of fusion power(P_f) for T_{init}=0.0, 0.5, 1.0kg and dead inventory in the plasma region(D_T) for T_{init}=0.0 kg. The shaded zones corresponding to the no-saturation case and the lines to the saturation case.

that in the saturation case. The key is the increment of the dead inventory D_T. Fig.7 also shows the dead inventory in the plasma region without the initial tritium loading(T_{init}=0.0kg), and the initial increment of D_T for the saturation case is larger than that of the no-saturation case. That is why the commissioning period for the no-saturation case can be shorter than that of the saturation case. The initial tritium loading to start with the rating operation is estimated as 1.3kg in the no-saturation case.

5. Conclusion

In this paper, core plasma operation flexibility by the tritium-ratio control(T-ratio) is proposed in order simultaneously to control the fusion power and to keep a high density for the divetor heat-handling, and the plasma performance required for such operation flexibility is analyzed. The operational space of core plasma in Demo-CREST is expanded to the high density region by reducing T-ratio. Especially, the expansion of operation space for low beta case is significant. When the total plasma current is reduced from 15.4MA to 12.6MA in Demo-CREST, DD-plasma operation ($fn_T \sim 0\%$) is found to be possible. The plasma performance required for such high density operation with T-ratio control is also analyzed. The most important parameter is the confinement improvement factor HH, and high HH similar to that of the ITER steady state operation is required in the Demo-CREST concept. The resultant condition of SOL density and impurity radiation power required for divertor heat load less than 10MW/m^2 is analyzed, and the operation condition as for SOL density and impurity radiation power is found to start from those similar to the ITER steady state condition. The 2D divertor transport simulation supports the promising outlook of the divertor operation with heat flux less than 10MW/m^2. Finally, the initial tritium loading for Demo-CREST is analyzed taken into consideration the possibility of start-up without the initial tritium loading. The initial tritium loading of 1.3~1.6 kg is found to be necessary for Demo-CREST. Furthermore, the plasma commissioning without the initial tritium loading is also the potential method, and the required commissioning period is estimated to be 75~110 days in case of the net TBR 1.05. That possibility strongly depends on the increment of the dead inventory, and understanding the tritium behavior in the plasma region is important to confirm the applicability of this commissioning method to Demo-CREST.

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