Compact Reversed Shear Tokamak Reactor with a Super-heated Steam Cycle

K. Okano, Y. Asaoka, T. Yoshida, M. Furuya, K. Tomabechi
CRIEPI: Central Research Institute of Electric Power Industry, Komae-shi, Tokyo, Japan
Y. Ogawa, N.Sekimura, R.Hiwatari, T. Yamamoto*
Faculty of Engineering, University of Tokyo, Tokyo, Japan
(*Present address: National Institute for Fusion Science, Toki-shi, Japan)
T.Ishikawa, Y.Fukai, A.Hatayama
Faculty of Science and Technology, Keio University, Yokohama, Japan
N. Inoue, A.Kohyama
Institute of Advanced Energy, Kyoto University, Uji-shi, Kyoto, Japan
K.Shinya, Y.Murakami, I.Senda
Toshiba Research and Development Center, Kawasaki, Japan
S.Yamazaki, S.Mori, J. Adachi, M. Takemoto
Nuclear System Division, Kawasaki Heavy Industries, Ltd., Tokyo, Japan

Abstract

The Compact Reversed Shear Tokamak 'CREST' is a cost competitive reactor concept based on the reversed shear high beta plasma and a super-heated steam cycle which makes a high thermal efficiency (η=41%) possible. The moderate aspect ratio(3.4) and the elongation (2.0) of the CREST are similar to the ITER advanced mode plasma. The current profile control by neutral beam current drive stabilizes the ideal MHD activity up to β_N =5.5. The CREST could generate about 1.2GWe electric power with a competitive cost.

1. INTRODUCTION

The fusion power plant in the future should be made economically and environmentally attractive compared with other advanced energy sources. In particular the achievement of a competitive cost of electricity (COE) is the first priority for electric power industries. The Compact Reversed Shear Tokamak 'CREST' is a cost competitive reactor concept based on the reversed shear (RS) plasma with a moderate aspect ratio A=3.4 which is similar to the ITER advanced mode plasma. Showing such a future perspective based on the ITER is important in the fusion development strategy.

The advanced ODS ferritic steel is adopted for main components, which is compatible with an high thermal efficiency cycle (η=41%) using super-heated steam[1]. The current profile control and the high speed plasma rotation by neutral beam current drive (NBCD) stabilize the ideal MHD activity up to β_N =5.5 with a closed conductive shell, which is installed in the breeding blanket. Our cost assessment study [2] has shown that these high values (η and β_N) are the most effective key parameters for reducing the COE of tokamak reactors.

2. HIGH BETA RS EQUILIBRIUM

The RS equilibria should be sustained by a high bootstrap current (BSC) fraction in order to minimize the current drive power. Since the profile of BSC must be well aligned with the optimized MHD equilibrium, a zero dimensional system code is not useful for the present purpose. A tokamak analysis code used in this study includes the calculations of neutral beam (NB) or RF driven current profiles including BSC[3]. This code has been combined with a two dimensional ideal MHD analysis code EQLAUS/ERATO[4,5] in order to investigate self-consistent MHD equilibria and the stability.

In the first step, we investigated an optimized RS tokamak parameters under the following restrictions: 1) Neutron wall load P_W less than 5 MW/m^2 in average. 2) Neutron shield thickness of 1.4m (incl. the breeding blanket). 3) $2 < q_{min} < 3$ for the MHD stability (note that a large BSC fraction may drop q_{min} into a lower value). The operative window of the RS equilibria is highly restricted because of the above condition (3) on the q profile. It is found that there are two possible passes to RS reactors; high aspect ratio (4<A<6) with a high B_{max}(~16T), or moderate aspect ratio (3<A<4) with a moderate B_{max}(~13T). In the former case, the fusion power restricted to 2.2 GW_{th}

because of the neutron wall load limit (< 5 MW/m^2). It is worth noting that this case is consistent with the STARLITE design [6]. We have chosen the latter case in order to design the CREST with the aspect ratio close to the ITER and to attain 3 GW$_{th}$ output within the wall load limit. The base parameters of CREST are determined as: R_o=5.4m, A=3.4, κ=2.0, δ=0.5.

The RS equilibrium for CREST has been optimized by the self-consistent calculation including the ideal MHD analysis. In the early stage of CREST design [7], a beam + RF driven was considered assuming Z_{eff}=1.5. Although this option is attractive for minimizing the total injection power (RF+ NBI), the RF current drive efficiency is very sensitive to Z_{eff}. Since a high Z_{eff} operation with Xe impurity injection is considered for the CREST in order to reduce the heat load of divertor, we have decided to use a new fully beam driven option, shown in TABLE 1 and FIG.1. The Troyon coefficient β_N attains 5.5 without ideal MHD instabilities (ideal kink for n=1,2 and 3, ballooning and Mercier modes have been checked). The fusion power P_f=3GW and the plasma current Ip=12 MA with the bootstrap current (BSC) fraction f_{bs}=83%. The current profile has been optimized by control of the power ratio between two beamlines (one is on mid-plane and another is off mid-plane, 2.5MeV both). The power ratio is shown in the figure. The footprints of beamlines are shown in FIG.1a. In the actual design (FIG.4), the off mid-plane beam is injected in the lower part of the plasma.

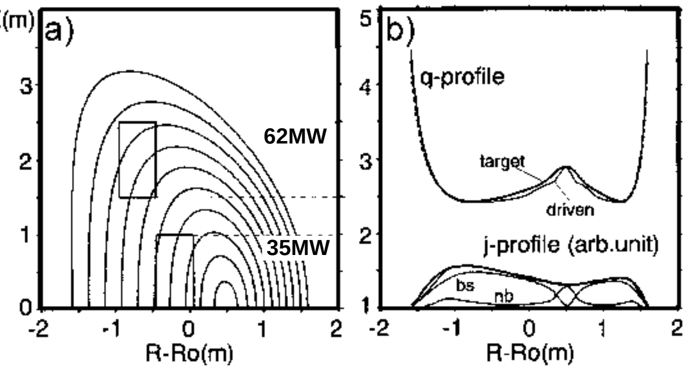

FIG.1. Optimized RS equilibrium for CREST

Because of large momentum input due to the neutral beam injection, the mean toroidal rotation velocity v_{rot} is over 10^5 m/sec (assuming $\tau_{rot}=2\tau_E$). The ideal kink and resistive wall modes will be stabilized by this fast rotation and a conductive shell close to the plasma, which is installed inside of the blanket. On the mid-plane, a_w/a=1.15, where a_w is the effective radius of the conductive shell.

The plasma positional stability is kept by the active and passive copper feedback coils installed inside the TF coils. An example of the feedback system simulation against a small β_p and l_i perturbations ($\Delta\beta_p$=-0.5, Δl_i =-0.05) is shown in FIG.2, where the one-turn resistivity of the passive coils is assumed as 2.7 micro-ohms. The power for the active position control coils is less than 2.5 MW in this example.

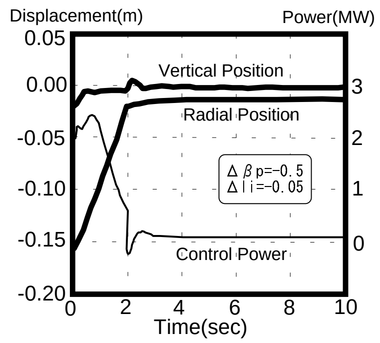

FIG.2. Plasma Position control

3. RADIATIVE COOLING AND PARTIAL LOAD CAPABILITY

The high heat load of divertor is a common issue for all compact tokamak reactors. Making use the maximum advantage due to the insensitibity of NBCD efficiency on Z_{eff}, the CREST can be operated in a wide range of Z_{eff}. A small amount of xenon is added in the main plasma (0.04%, Z_{eff}=2.2) and SOL (0.1%) in order to radiate 85% of the thermal power to the first wall. The radiation powers are 364MW (53%) from the main plasma and 224MW (32%) from SOL, respectively. The peak divertor heat load can be reduced down to 10 MW/m^2, if the separatrix density attains 0.9 × 10 ^{20}m^{-3} (FIG.3).

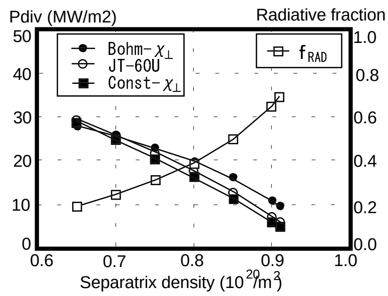

FIG.3. Peak heat load of divertor

By such active Z_{eff} control, the partial load operation from 53% output (Z_{eff}= 1.8) to 100% (Z_{eff}=2.2) will be achievable. The power balance can be maintained by reducing Z_{eff} without changing H value. Such flexibility in operation is attractive in actual commercial use.

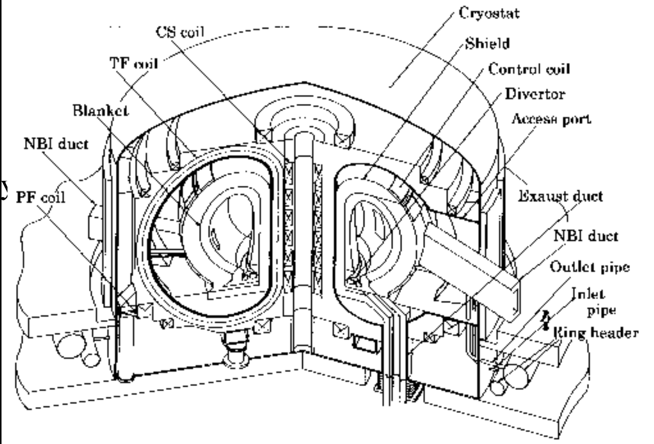

FIG.4. Bird's eye view of CREST

4. CONFIGURATION AND ENGINEERING DESIGN

Figure 4 shows the bird's eye view of the CREST. Full sector removal with horizontal ports for easy maintenance is adopted in order to rise the availability. The cost of the increased size of the TF and PF coils, which is a penalty of the full sector maintenance concept, is expected to be smaller than the cost savings due to the increased availability.

TABLE 1. MAJOR PARAMETERS

major radius, R	5.4 m
aspect ratio, A	3.4
elongation, κ	2.0
triangularity, δ	0.5
safety factor $q_o/q_{min}/q_\psi$	2.9/2.4/4.3
poloidal beta, β_p	2.5
Troyon factor, β_N	5.5
ITER89P multiplier, H	3.2
plasma current, I_p	12 MA
BSC fraction, f_{bs}	83 %
beam energy, E_b	2.5 MeV
beam power, P_b	97 MW
mean elec. temp.,$\langle T_e \rangle$	15.4 keV
mean ion temp., $\langle T_i \rangle$	16.4 keV
mean elec density,$\langle n_e \rangle$	2.1×10^{20} m^{-3}
ratio to GW limit	1.3
mean rot. velocity, v_ϕ	1.2×10^5 m/s
effective charge, Z_{eff}	2.2
He accumulation	15 %
τ_p*(He)/τ_E	7.4
tor. field(on axis), B_t	5.6 Tesla
peak tor. field, B_{max}	12.5 Tesla
neutron wall load, P_w	4.5 MW/m^2
fusion power, P_f	2.970 GW
plant therm.output	3.378 GW
thermal efficiency	41 %
gross elec. power	1.385 GW
net elec. power	1.163 GW

4.1 Materials selection. Reduced activation advanced ferritic steel is selected as a structure material because of good compatibility with water coolant and the large database. The allowable temperature range from 350 K to 900 K is assumed. Mixture of lithium-containing ceramics pebbles and beryllium pebbles is selected for the breeding material. Installation of mixture pebbles of breeder and multiplier is expected to raise the breeding ratio and the thermal conductivity of the breeding zone. Lithium zirconate (Li_2ZrO_3) is selected as the first candidate of the breeder material because of relatively low chemical reactivity and high tritium recovery capability. Zirconium plate is the first candidate of the conducting shell because of its small influence on the tritium breeding.

4.2 Super heated steam cycle The constitution of the breeding zone and the pressure, flow rate and inlet temperature of the coolant were optimized to control the temperatures of structure material, breeder materials, and outlet coolant. The phases of the coolant at first wall and at breeding zones and the pressure loss of the coolant were also controlled. Temperature allowance of breeder materials is from 700 K to 1020K due to tritium recovery and chemical reactivity of the breeder and the multiplier, respectively[8]. Outlet coolant temperature must higher than 750 K to attain the high thermal efficiency.

The pressure of the coolant is 15 MPa, which is similar to that of the pressurized water cooling system of the SSTR[9]. The attainable thermal efficiency is estimated to be 41 % with the outlet temperature of 750K. FIGs. 5 and 6 show the cross section of the blanket and the concept of coolant flow direction. First wall is cooled by pressurized water at relatively low temperature, up to 600 K. The first wall can be cooled in the allowable range at the power loads of 1.2 MW/m^2 in thermal flux

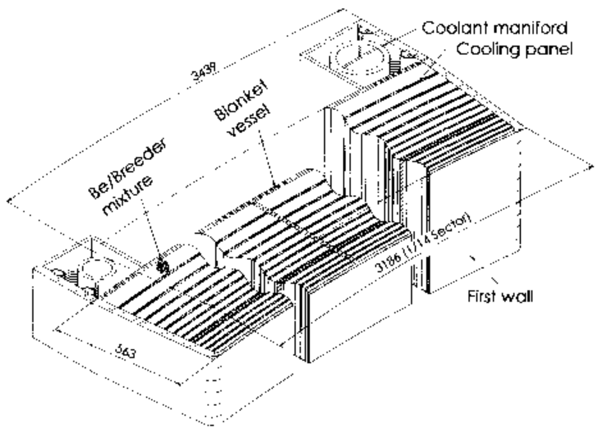

FIG.5. Cross section of the outboard blanket

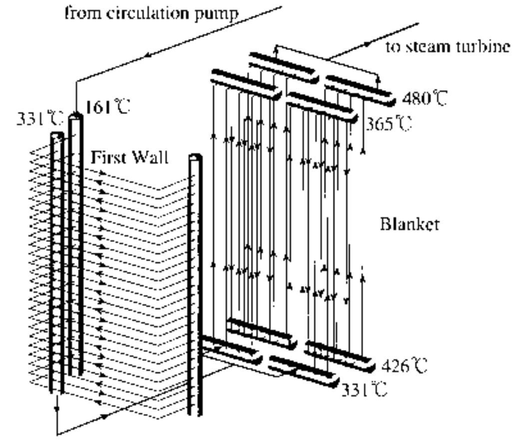

FIG.6. Design concept of coolant flow

and 4.5 MW/m^2 in neutron flux. The pressurized water comes from the first wall is led to the forward breeding zone. The coolant is heated up to 615 K and starts boiling in upward stream channels. The steam led to the backward of the blanket would be super-heated up to 750 K.

Table 2. BLANKET PARAMETERS

	Inboard	Outboard
Thickness (cm)		
FW	1.5	
(FS / water / FS)	(0.4 / 0.8 / 0.3)	
Blanket	54.4	
(breeder / cooling panel / shell)	(40.0 / 13.4 / 1.0)	
Gap	4	18
Shield & VV	56	70
Blanket		
Breeder / neutron multiplier (%)	Li_2ZrO_3 / Be (20 / 80)	
Packing fraction (%)	80	
Li_2ZrO_3 density (%TD)	85	
Be density (%TD)	90	
Shield		
Ferritic steel / water (%)	75 / 25	
FW heat flux (MW/m^2)	1.2	
Average neutron wall load (MW/m^2)	4.5	
Peak neutron wall load (MW/m^2)	5.0	6.5
Damage to magnet (* : at 22.5FPY)		
Peak fast neutron fluence* (n/cm^2)	3.8×10^{18}	4.7×10^{17}
Peak nuclear heating (mW/cm^3)	0.14	0.018
Peak insulator dose* (Gy)	4.7×10^{7}	7.1×10^{6}
Peak dpa to Cu stabilizer* (dpa)	2.4×10^{-3}	2.9×10^{-4}

4.3 Tritium breeding and shielding.

The requirement of net tritium breeding ratio is estimated to be at least 1.10 for the first generation of fusion commercial plants[10] for a favorable introduction pace of fusion. Mixture ratio of Li_2ZrO_3 and Be and Li-6 enrichment were optimized for tritium breeding ratio and its reduction due to burn up. The case of 50 % of Li-6 enrichment and 80 % of Be mixture ratio is selected for the reference case. The initial local TBR and its reduction, which are estimated to be approximately 1.40 and 0.03 for 2.25 FPY operation, i.e. 75% of 3 years, respectively, are acceptable value. The 10mm thickness of zirconium plate at 24 cm from the plasma surface is required to be a stabilization shell. The degradation of TBR due to the Zr shell is estimated to be only 0.05. TABLE 2 summarizes of the shielding performance with the shield of 56 cm thickness for inboard and 70 cm for outboard. The distance between the plasma surface and super conducting coils behind the inboard shield is 1.4 m. Peak neutron wall loading of 5 MW/m^2 (inboard), 6.5MW/m^2 (outboard) and 22.5 FPY operation, i.e. 75% of 30 years, are assumed for calculations of the irradiation damage to the SC magnet.

5. TEST OF CREST LIKE PLASMA BY ITER

The plasma configuration of CREST can be tested in ITER. An example of CREST like RS plasmas for ITER calculated with TOSCA code is shown in FIG.7 : A=3.67, κ=1.89/2.35, δ =0.53/0.56 (top/bottom) and βp=2.5. A monolithic CS coil system was hardly capable of providing equilibrium of CREST like plasmas with high elongation and triangularity. Therefore this calculation adopts a separate CS coil system composed of 6 solenoid coils.

Although many issues still remain to be solved, the CREST study has shown that an attractive reactor concept would be brought about by the success of the ITER project.

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