Conceptual Design of Advanced Steady-State Tokamak Reactor

-Compact and Safety Commercial Power Plant (A-SSTR2)-

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Abstract. Based on the last decade JAERI reactor design studies, the advanced commercial reactor concept (A-SSTR2) which meets both economical and environmental requirements has been proposed. The A-SSTR2 is a compact power reactor (R_p=6.2m, a_p=1.5m, I_p=12MA) with a high fusion power (P_f =4GW) and a net thermal efficiency of 51%. The machine configuration is simplified by eliminating a center solenoid (CS) coil system. SiC/SiC composite for blanket structure material, helium gas cooling with pressure of 10MPa and outlet temperature of 900˚C, and TiH_2 for bulk shield material are introduced. For the toroidal field (TF) coil, a high temperature (T_C) superconducting wire made of bismuth with the maximum field of 23Tand the critical current density of 1000A/mm^2 at a temperature of 20K is applied. In spite of the CS-less configuration, a computer simulation gives a satisfactory plasma equilibria, plasma initiation process and current ramp up scenario.

1. INTRODUCTION

The fusion power plant in future should be attractive economically and environmentally compared with other advanced sources. Especially realization of a competitive cost of electricity (COE) is the first priority to be adopted by the utility companies. Even though there are a lot of advantages, such as those in the environmental effects and the abundance of fuel resource, the fusion would never be chosen as major option if its COE is high. The COEn (normalized by the present coal plant without CO_2 control) of the CO_2 controlled fossil fuel plant and the LWR (light water reactor) fission plant, both at ~50 years in future have been predicted as 1.5~1.8 and 1.0~1.5, respectively [1]. A lower COEn less than unity is no doubt desirable. In case of the COEn higher than 1.8, the fusion will not be attractive. For the time being, the COEn of ~1.5 will be a reasonable target for the first generation of fusion reactors.

The fusion COE is mainly governed by a plasma beta value, a toroidal field (TF) strength and a plant thermal efficiency. The COEn of the “ITER[2] like plant” with a normalized beta β_N~3, a maximum TF strength B_{max}~13T and a thermal efficiency η_{th} ~34% is evaluated to be 2.5 [1]. One possible approach for achieving the COEn of ~1.5 is to choose the combination set of higher β_N of 5~5.5 and moderate B_{max} of 13~16T such as CREST [3] or ARIES-RS [4]. Another combination set of moderate β_N<4 and high B_{max}~20T such as DREAM [5] or A-SSTR [6] is also possible. This paper describes the improved version in the later approach, named “A-SSTR2”.

2. OVERALL FEATURE OF A-SSTR2

Table 1 shows the A-SSTR2 operating parameters. The A-SSTR2 is a compact power reactor (R_p=6.2m, a_p=1.5m, I_p=12MA) with a fusion power (P_f=4GW). The machine configuration is simplified by eliminating a center solenoid (CS) coil system. SiC/SiC composite for blanket structure material and TiH_2 for bulk shield material are introduced as shown in Fig. 1. For the toroidal field (TF) coil, a high temperature superconducting wire made of Bi-Ag alloy with the maximum field of 23Tand the critical current density of 1000A/mm^2 at a temperature of 20K is applied. The resultantly yielded space by eliminating CS coil system is used for the TF coil support structure.

A SiC/SiC composite is employed as a structural material of the blanket. A blanket module with a weight of around 300kg is designed and Be and Li_2TiO_3 are used as a neutron multiplier and a tritium breeder. Totally 1950 modules are used in the A-SSTR2. The 1D neutron transport analysis shows a tritium-breeding ratio of 1.29 in this blanket system. The blanket system is replaced as the 30 degree sector unit within 2 months for every 2 years. For obtaining high thermal efficiency of higher than 50%, a helium gas cooling system with inlet/outlet temperatures of 600/900˚C and a gas turbine system are introduced.

By employing TiH_2 pebbles mounted in SiC/SiC holder (68/54cm in thickness for inboard/outboard region) for neutron shield, a low nuclear heating rate (<0.1mW/cc) is realized in the superconducting wire. The A-SSTR2 components layout is shown in Fig. 1.

3. OPERATION SCENARIO

3.1 Plasma equilibria

The plasma equilibria during the ramp-up phase are found within the reasonable ampere-turns of 120 MAT. The snapshot-like pictures are shown in Fig. 2. Here, the beam driven current profile, the bootstrap current profile and the plasma equilibrium are simultaneously and consistently solved. The energy, power and cross section of the injected beam are controlled for suppressing the heat flux onto the facing blanket first wall less than 1 MW/m^2.

In the CS-less A-SSTR2, the linkage flux of -52.4 Vs is inevitably generated at the end of ramp-up phase. When this flux supply is taken into account, several hours can be shortened for the current ramp-up phase as described in later section.

3.2 Plasma breakdown and initial current ramp-up

In a non-inductive plasma initiation (breakdown) and current ramp-up scheme, the NBI (neutral beam injection) should be replaced by RFW (radio frequency wave) heating and current drive to avoid high shine through and orbital losses in the low density, low current (high q) plasma. Here, an EC (Electron Cyclotron) system is considered for heating and current drive. While even for the CS-less tokamak, a small amount of initial magnetization can be available for the plasma initiation. When the maximum allowable stray field strength is 10 G, the upper limit of the initial magnetization is 0.2 Vs for the A-SSTR2 configuration shown in Fig. 1. The flux swing of 0.4 Vs can be available.

The results of the ASTRA simulation of the plasma breakdown and initial current ramp-up with the external voltage (U=4 V for 0.1 s) is shown in Fig.3 [7]. It takes about 2000 s for the initial current ramp-up phase up to the plasma current of 2 MA.

3.3 Ignition approach

After the plasma breakdown and initial current ramp-up, the plasma current is to be driven by the NBCD from 2 MA to 12 MA. As an example of the ignition approach simulation from the plasma current of 4 MA to 12MA, the related parameters (the electron density n_e, confinement H_H factor, fusion power P_f, NB input power P_{NB} bootstrap current I_{BS}, NB driven current I_{NB}, etc.) are shown in Fig. 4. Here an external magnetic flux is arbitrary supplied for the calculation convenience. Without the external magnetic flux supply, the ignition approach time can be estimated by assuming $(I_{NB}+I_{BS})/I_p$ little more than unity. When $(I_{NB}+I_{BS})/I_p=1.06$ is assumed, the ignition approach time from 2 MA to 12 MA is about 28 hours, where the plasma self inductance of 14.8 μH and the plasma resistance of $3.2x10^{-3}$ μΩ are used. As mentioned before, the linkage flux of -52.4 Vs can shorten the current ramp-up phase from 28 hours to 22 hours.

4. MHD STABILITY

4.1 Plasma vertical position control

The A-SSTRT2 elongated plasma requires the passive shell for vertical positional stability. For the replaceable blanket sector unit, the possible shell structure must be so called “saddle loop” structure located between the shield and blanket structures shown in Fig. 1.

Figure 5 shows that the shell structure of 5cm thick vanadium alloy causes the growth rate of 40 s^{-1}. The solid line is for the both of outer and inner shell structures and the dotted line is for the only outer case. Even though the external SC poloidal coils are used for the feedback control, the control power is less than 100 MVA.

4. 2 Ballooning and kink modes

As to the ballooning mode, two kinds of normalized pressure gradient (α parameter) on each magnetic surface are calculated as a function of shear (s) as shown in Fig. 6. One is the critical pressure gradient and another is the A-SSTR2 operation one. The magnetic surface position is denoted by its shear. The ballooning mode is stable over the whole plasma region.

As to the ideal low n kink-modes, the instability growth rates are calculated for the cases with and without stabilizing ideal shell. Without the shell, both of n=1 and n=2 modes are unstable. With the shell, the n=1 mode requires the shell position r_s closer than $r_s/a_p \leq 1.4$ as the stable condition. And the n=2 mode requires the shell position r_s closer than $r_s/a_p \leq 1.2$.

5. DIVERTOR HEAT LOADS

A large quantity of the divertor related thermal power such as the α-heating power of 800 MW and the current drive power of 60 MW should be managed. Fortunately, since the plasma density and the toroidal field strength are very high, the synchrotron radiation power, the Bremsstrahlung power and the impurity line radiation powers both in the main plasma and the divertor plasma can be in large quantities. Here, the synchrotron radiation power of 100 MW, the Bremsstrahlung power of 100 MW and the impurity line radiation power in the main plasma of 200 MW are appropriated. Therefore the thermal power into the divertor region is estimated as ~460 MW. For the safe operation of the divertor plate, the thermal power load on the plate should be reduced less than 100 MW. More than 360 MW is needed to be radiatively emitted, so called “radiative cooling”.

One of the convincing procedures for the radiative cooling is to seed an inert gas, e.g. Neon or Argon or Krypton. According to a fluid model calculation [8], ~2.5 % Ne seeding into the divertor plasma reduced the thermal power less than 100 MW. On the other hand, the plasma temperature at the vicinity to the divertor plate is also critical issue. It was also found that ~2.5 % Ne seeding lowered the temperature from 200 eV to 20~30 eV. Thus a tungsten plate can be used at the sufficiently low sputtering rate.

6. CONCLUSION

i) A combination set of high toroidal field (11T on plasma axis, 23T for peak) and moderate beta (β_N=4) is chosen for the high fusion power (4 GW) and the compact machine size (R_P=6.2 m). The space for structures supporting the enormous electromagnetic force on the TF coils is made by eliminating the center solenoid (CS) coil.

ii) A computer simulation gives a satisfactory plasma equilibria, plasma initiation process and current ramp up scenario. It takes about 2000 s for the initial ramp-up phase to reach a plasma current of 2 MA. The time to grow from 2 MA to 12 MA is about 22 hours, where the linkage flux of -52.4 Vs by the plasma equilibrium field is taken into account.

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Table 1 A-SSTR2 operating parameters

Total thermal power (GW)	5.0
Gross electric power (GW)	2.7 (54%)
Net electric power (GW)	2.55 (51%)
Tokamak weight (ton)	27880
Mass power dens.(kWe/ton)	91.5
Plasma major radius (m)	6.2
Plasma minor radius (m)	1.5
Plasma elongation (95%)	1.8
Plasma current (MA)	12
Bootstrap current fraction	0.83
Current-drive power (MW)	60
Toroidal field on axis (T)	11
TF coil peak field (T)	23
Coils temperature (K)	20
Fusion power (GW)	4.0
Ave. wall load (MW/m^2)	6.0
Normalized beta	4.0
Structure material	SiC Comp.
Shell material	V-alloy
Shield material	TiH_2
Coolant	He (10Mpa)
Coolant temp. (°C)	600/900°C

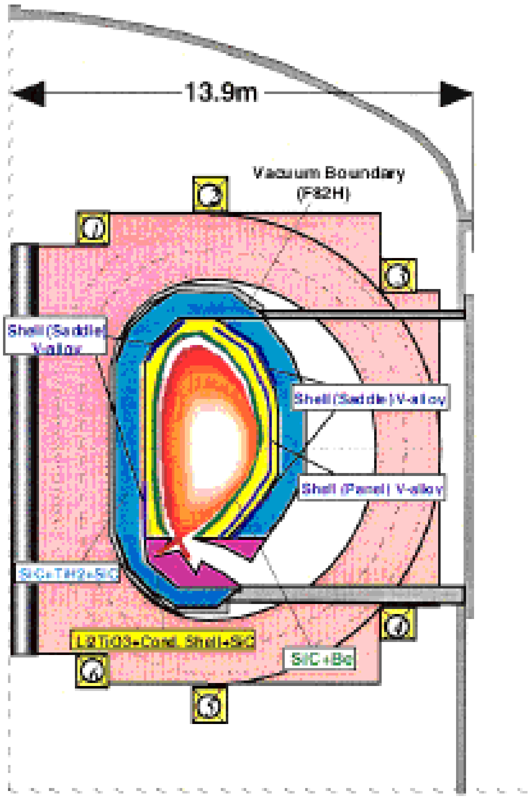

Fig. 1 A-SSTR2 components layout

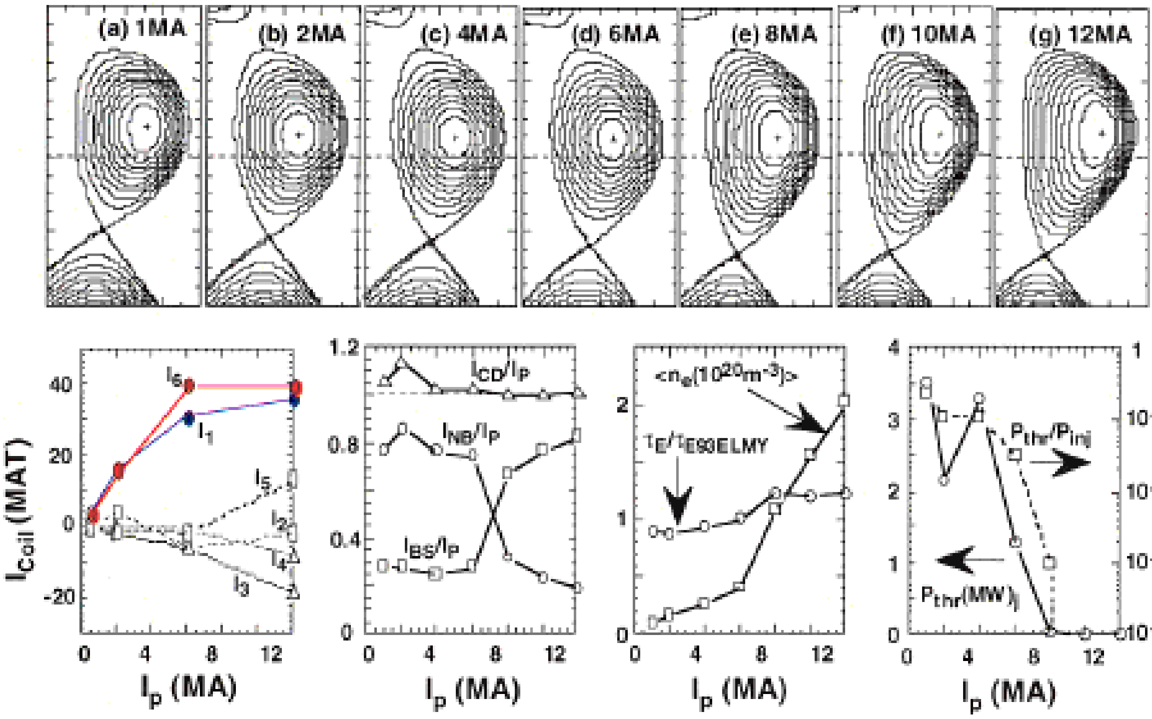

Fig. 2 Snapshot-like equilibrium solutions during the plasma ramp-up phase by the ACCOME code. A suffix number of coil current I_{coil} shown in lower and most left picture corresponds with a PF coil number in Fig. 1. The heat flux onto a facing component by NBI shine-through is less rhan 1 MW/m^2.

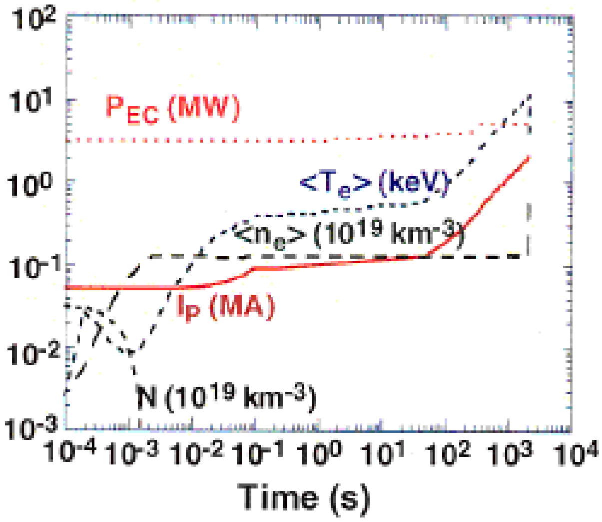

Fig.3 PFC supported ECCD current ramp-up. Loop voltage of 4 V was applied until 0.1 s. ECCD power (P_{EC}) is less than 5 MW. <ne> is the volume averaged plasma density, N is the residual gas density, and <Te> is the density weighted electron temperature.

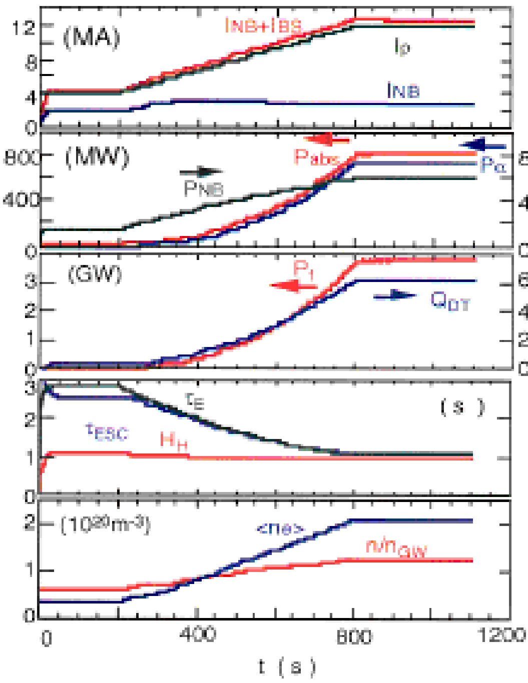

Fig.4 Ignition approach simulation by TOPICS. Here an external flux supply is considered for the calculation convenience. The true ramp up time from 2 MA to 12 MA is about 22 hours.

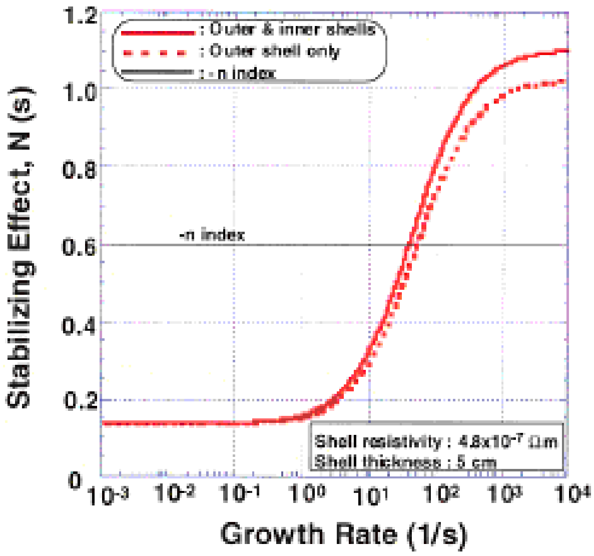

Fig. 5 Growth rate dependence on the stabilizing effect. The shell is 5 cm V-alloy and located behind blanket. The plasma elongation is 1.8. Destabilizing force denoted by n-index (decay index of plasma equilibrium field) and the passive stabilizing force denoted by N(s) are well balance each other at the growth of 40 s-1.

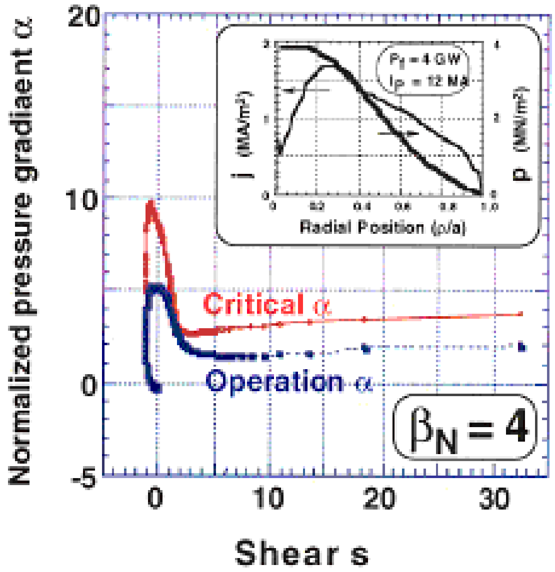

Fig. 6 Critical and operational α (normalized pressure gradient) for ballooning mode are shown in A-SSTR2. The shear s denotes the position of magnetic surface. The ballooning mode is stable in all plasma region. The pressure and current profiles, denoted by **p** and **j**, respectively are also shown.