Engineering Feature in the Design of JT-60SA

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Abstract. The construction of the JT-60U modification (JT-60SA) is now being planned by both parties of Japan and EU as a part of the ITER Broader Approach. It means that a new function of the ITER satellite tokamak is clearly defined as a mission on the JT-60SA program. The magnetic coils and vacuum vessel were designed to ensure flexibility in the plasma shape (aspect ratio A=2.6-3.1). The NbTi superconductor will be adopted to generate the central magnetic field of 2.7 T at R=3.0m. An optimization of neutron and radiation shielding was done on the vacuum vessel and cryostat structures under the neutron yield of about 2×10^{19}/shot produced by the plasma heating systems (NBI and ECRF) of 41 MW, 100 s. The power supplies for the superconducting coils and heating devices will be prepared by the utilization of existing systems. This paper intends to clarify the latest design option of the JT-60SA.

1. Introduction

The JT60-SA originally aims to contribute to DEMO reactor design as well as to ITER with the maximum utilization of existing JT-60 facilities such as plasma heating and current drive systems, power supplies, diagnostics and cooling system [1]. Therefore, the aspect ratio down to 2.6 was chosen to survey the optimum plasma shape for the most cost effective DEMO reactor design [2]. The toroidal field and heating power are deigned at 2.7 T and 41 MW, respectively, to make sure of high β_N H-mode plasma with an ITER-relevant high density. Since the maximum field strength at coil surface is 6.4 T, it is possible to adopt NbTi superconductor for the TF coil. The high power plasma heating systems, such as NBI and ECRF, cause a remarkable increasing of DD neutron yield of about 2×10^{19}/shot. In the design of TF coils, vacuum vessel and cryostat structure, much effort was made to reduce the nuclear heating in the TF coil under a narrow shielding space. The divertor geometry was optimized to produce both high triangularity and ITER shape single-null plasmas with one machine. Semi-closed vertical divertor with flatter dome was adopted to keep higher flexibility of plasma shaping capability. A new AC power system combined with a power grid and existing Motor-Generator set was deigned to satisfy the total energy of 12.9 GJ for the heating systems of 41 MW, 100 s.

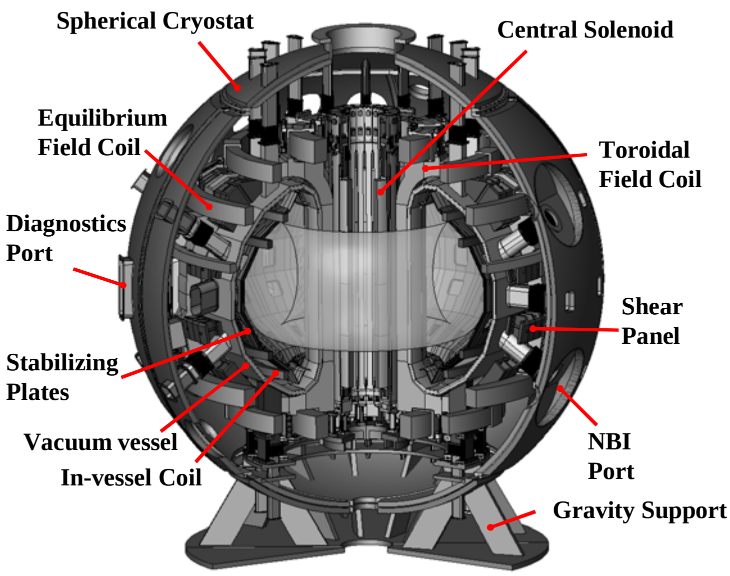

Fig. 1. Cut-set view of JT-60SA.

2. Outline of JT-60SA Device

TABLE I: BASIC PARAMETERS OF JT-60SA.

Parameter	Large Plasma (DN)	ITER Similar (SN)
Plasma Current Ip (MA)	5.5	3.5
Toroidal Field Bt (T)	2.72	2.59
Major Radius (m)	3.01	3.16
Minor Radius (m)	1.14	1.02
Elongation, κ_{95}	1. 83	1.7
Triangularity, δ_{95}	0. 57	0.33
Aspect Ratio, A	2.64	3.10
Shape Parameter, S	6.7	4.0
Safety Factor q_{95}	3.77	3.0
Flattop Duration	100 s (8 hours)	
Heating & CD power	41 MW x 100 s	
N-NBI	34 MW	
ECRH	7 MW	
PFC wall load	10 MW/m^2	
Neutron (year)	4 x 10^{21}	

Figure 1 shows the cut-set view of JT-60SA machine. The major components of tokamak will be installed into the inner space of spherical cryostat with a diameter of about 14 m for the thermal shielding of superconducting magnets. The maximum plasma current is 5.5 MA with double null divertor operation and 3.5 MA with ITER like single null divertor operation as summarized in Table I. The most remarkable performance is the plasma heating power of 41 MW, 100s with break-even class plasmas. The nuclear heating of superconducting magnet is not negligibly small so that borated acid water is introduced into the double walled space of vacuum vessel. Further more, boron doped concrete is being planned to be attached onto the cryostat for bio-shielding of torus hall. However, a remote handling device is absolutely necessary for the maintenance and repair of in-vessel devices such as divertor modules and first wall, because the expected dose rate at the vacuum vessel may exceed 1 mSv/hr after 10 years operation and three month cooling. Figure 2 shows the layout plan of JT-60SA torus hall for major components. Four large ports will be devoted to the remote handling devices. In addition, four beam towers of positive ion based NBI(P-NBI), co and counter injection beam tank of P- NBI, and a negative ion based NBI (N-NBI) located at assembling hall will be reused. The wave guides of ECH will be

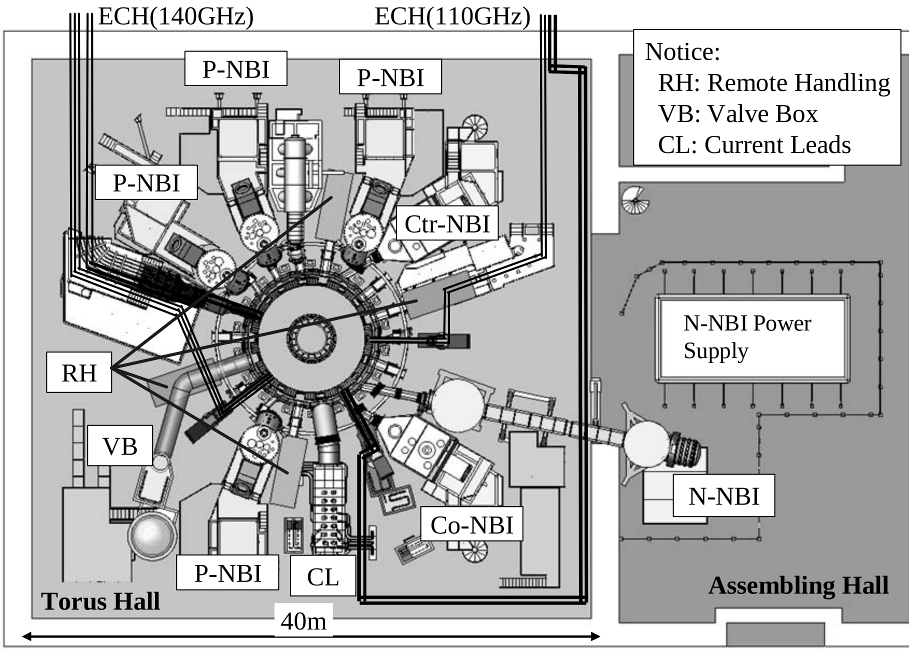

Fig. 2. Layout plan of the JT-60SA torus hall.

introduced from upper neighboring room as illustrated. Also, current lead box, valve boxes and other major equipments are installed nearby the cryostat of JT-60SA. Plasma discharge operation is planned mainly during day time, because the refreshment of divertor cryo-panel is required at every night to cope with high plasma density operation.

3. Superconducting Coil Design

NbTi superconducting conductor is planned to be used for the TF coil and seven equilibrium field (EF) coils including the divertor coil, because their maximum field strength at the conductor surface of TF and EF coils are 6.4 T and 6.1 T, respectively, under the assumption of uniform current distribution. Two kind of NbTi conductor design is expected for the EF coils, because the maximum magnetic field strength is quite different between the outer ring EF coils with large diameter and inner ring coils such as divertor coils. For central solenoid (CS), Nb_3Sn conductor is expected because the maximum field strength is in the range of 9-10 T to ensure the flux swing capability of about 40 Wb which is necessary to sustain the flattop period of 100 s with the rated plasma current. Followings are the details of latest design.

3.1 Toroidal Field Coil

It is not quite realistic to move the existing P-NBI beam tower without large modification or reinforcement of building. It means that the outer leg position of TF coil is limited by the P-NBI beam tower. On the other hand, plasma experiments with break-even class plasmas and more Demo Reactor relevant plasma shape with a lower aspect ratio of around 2.6 are required as mission of JT-60SA. To satisfy above requirement, the plasma equilibrium survey and evaluation of the expected plasma performance are carried out systematically, and then a necessary magnetomotive force of TF coil is defined to 8.2 T·m as a result. Also, the experienced maximum magnetic field strength of 6.4 T is derived considering the contribution from poloidal field under the assumption of uniform current distribution in the winding pack.

There is no commonly recognized design standard and/or methodology for the NbTi conductor design due to a lack of useful experimental data for large scaled coil such as ITER CS model coil. However, it seems to be two different design philosophies at present time. One is ITER criterion which regards critical current as important. Another regards limiting current as much important, which has been used as internal design criterion of JT-60SC/NCT at JAEA[3]. In other words, we still respect the Stekly criteria for NbTi conductor design. Table II shows the results of JT-60SA TF conductor design with our design criterion.

TABLE II: TF CONDUCTOR DESIGN.

Parameter	JAEA Criterion
Operating field Bop	6.4 T
Operating current Iop	26.5 kA
Strand diameter d	0.712 mm
Operating temperature Top	4.6 K
Cu/non-Cu ratio	7.0
Critical Current Density	>2900 A/mm^2
Number of SC strands	720
Cabling pattern	3 x 3 x 4 x 4 x 5
Conduit material	SS316LN
Inner diameter of conduit	22 x 22 mm
Temperature Margin	Top= Tcs-Tmargin Tmargin > 1.0 K
Limiting Current criterion	Ilim (Bop, Top+1K)/Iop>1
Critical Current	Ic / Iop > 1

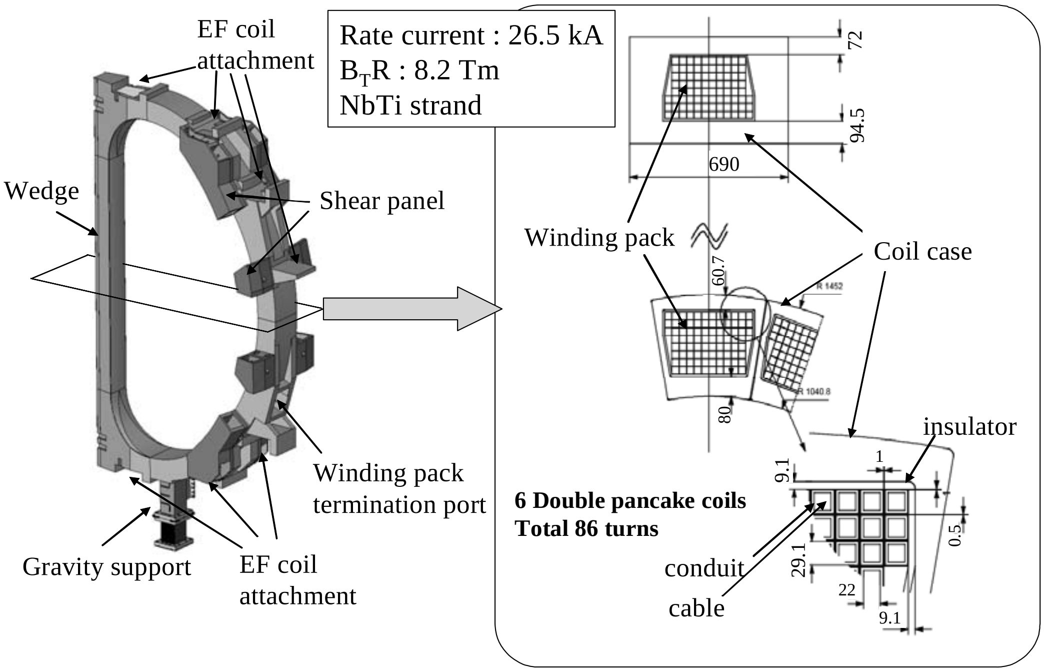

Fig. 3. TF coil structure including coil case, shear panel and winding pack.

Figure 3 shows the TF coil structure and cross-sectional view of winding pack. The seamless conduit with butt welding is assumed for EU industry, while manufacturing with roll forming is foreseen for Japanese industry considering their technical experience. The thickness of electrical insulation layer was designed based on the standard of ITER design. The required current decay time constant of quench protection circuit is 12 s to keep the hot spot temperature rise less than 250 K with strands. Only one dump circuit is enough from the TF coil operating voltage of 10 kV.

3.2 Central Solenoid (CS)

A large flux swing capability of about 38 Wb is necessary to sustain an ITER-relevant high density plasma at Greenwald plasma density of about 0.85, 3.5 MA for 100 s. A new CS conductor[3] was designed to generate 10 T maximum so as to sustain such plasma for 100 s. Figure 4 shows the CS and EF coil configuration. The CS consists of four modules and all modules are bound by 16 tie-plates through buffer zone by hydraulic rams as illustrated in Fig. 5. The middle two CS modules, CS2 and CS3, are planned to be connected in series electrically to minimize the number of current leads. The coil feeder will be placed at the outside side of CS to avoid the higher magnetic field. Each module consists of 8 hexa-pancake coils and one double pancake coil. The CS stack will be supported by links from the bottom side of TF coil case.

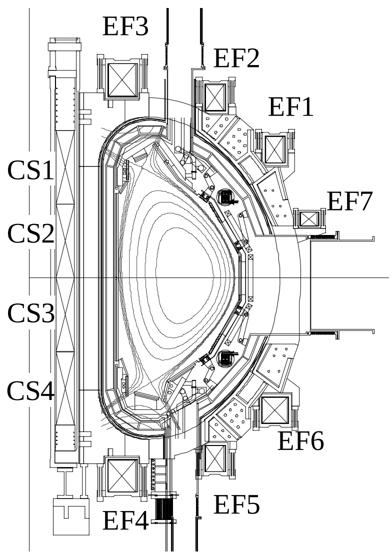

Fig. 4. CS and EF coil configuration.

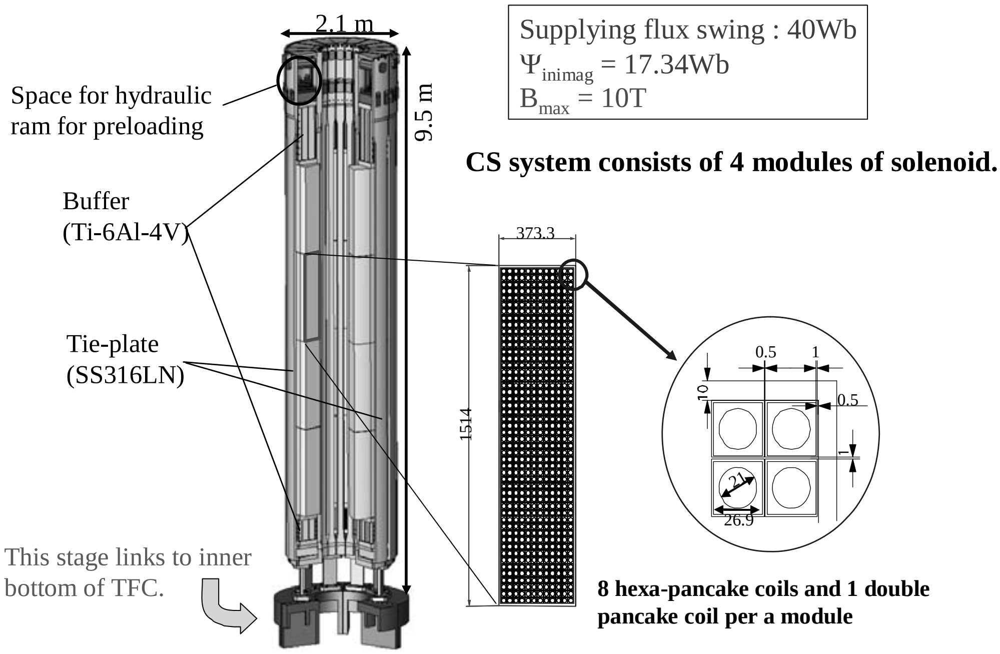

Fig. 5. Outline of CS and the supporting structure.

3.3 Equilibrium Field Coil

The maximum experience field strength at EF coil strongly depends on the plasma equilibrium and that position. Generally, it is usually less than 3 T at EF1, EF6 and EF7 coils, and about 5 T for EF2 and EF5 coils. Then, NbTi conductor can be easily applicable for these outer ring EF coils. However, it may exceed 5T occasionally in divertor coil, EF4, in case of the lower single null divertor operation. As a result, it is concluded that about 6.1 T is the maximum so that NbTi conductor could be applicable. It is quite marginal value because the operating temperature may be a little bit higher than that of TF coil due to the pulse operation. A precise thermal and fluid analysis of conductor should provide a realistic design parameter for the cooling system. From a viewpoint of physical analysis[2], the upper single null divertor operation is also very important as well as lower one. Then upper divertor coil, EF3, should be identical with lower divertor coil.

4. Vacuum Vessel, Divertor and Cryostat

4.1 Vacuum vessel

The acceptable temperature rise of superconductor by nuclear heat generation is 0.3 K at the TF coil, if the inlet coolant temperature of supercritical helium was assumed to be 4.3 K. Considering the flow length of supercritical helium during the plasma heating period of 100 s, the nuclear heating at the TF conductor is required to be less than 0.3 mW/cc as the reference. A filling of the double-wall vacuum vessel with borated acid water was devised to enhance the neutron shielding capability of the vacuum vessel as illustrated in Fig. 6. The estimated nuclear heating values at the inboard leg of TF coil conductor using 1D code of THIDA-2 is 0.27 mW/cc with assumption of 95% condensed ^{10}B at 40 deg. C under the maximum neutron emission rate of 4×10^{17}/s. It is about one third from the pure water case. Since the effect of port streaming can not be neglected, a 3D calculation using MCNP code was conducted. The estimated more precise value is 0.13 mW/cc at the inboard leg of TF coil, and 0.24 mW/cc at the outboard leg. Due to the effect of large diagnostic port, the nuclear heating of outboard is larger than that of inboard side.

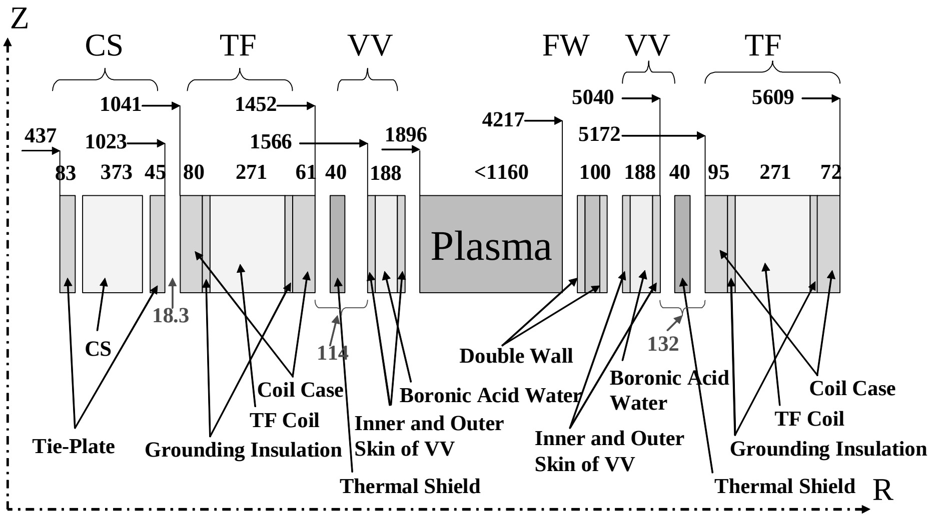

Fig. 6. Radial build of JT-60SA.

Figure 7 shows the basic structure of vacuum vessel including the ports and gravity support. The gravity support will be attached every 40 degree section with spring plates at the bottom of vacuum vessel. The connection plates are introduced to increase the stiffness against the twisting

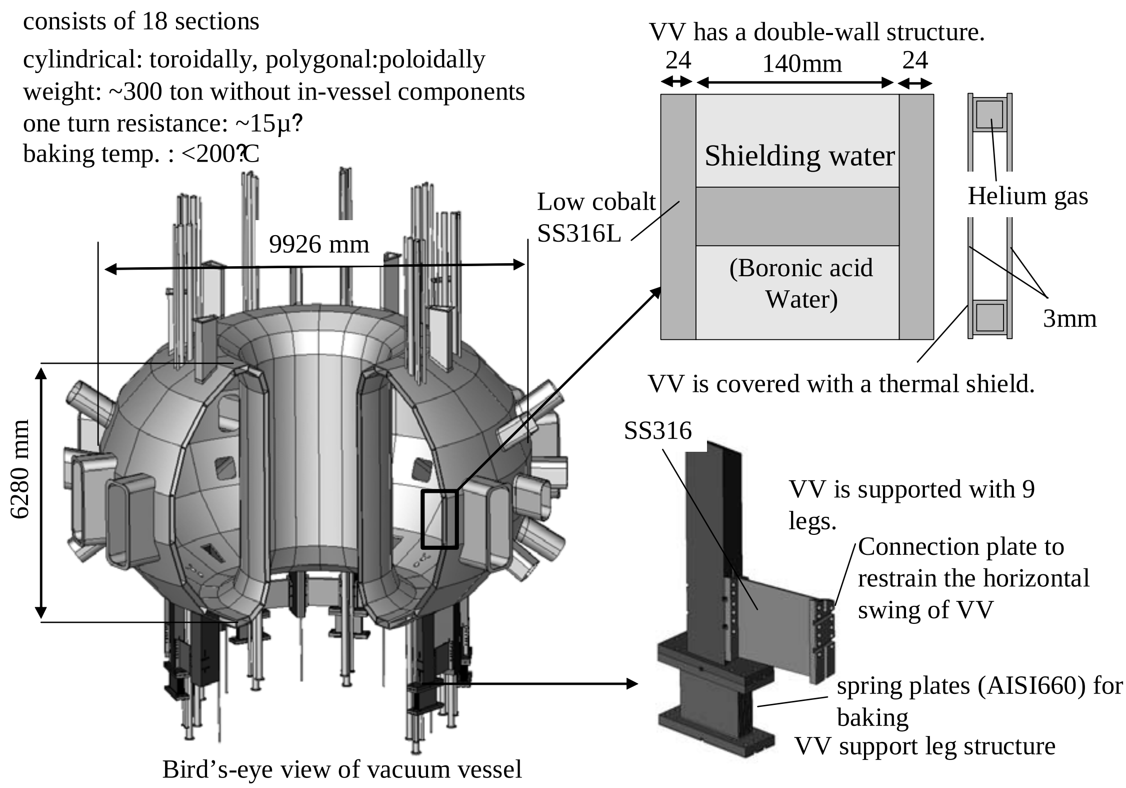

Fig. 7. Structure of vacuum vessel.

moment and seismic forces. The 80 K thermal shield will be placed between the vacuum vessel and the cold structures of superconducting magnets. The vacuum vessel baking with 200 deg. C will be done by heated borated water. It is clear that plasma operation with vacuum vessel baking is preferable for the wall conditioning, but the temperature of in-vessel components such as divertor cassette is not so high. The feasibility study is ongoing considering the stiffness of vacuum vessel and the gravity support, and a performance of Silver coated 80 K thermal shield.

4.2 In-vessel components including divertor cassette module

Figure 8 shows the in-vessel components consists of the divertor cassette, inboard first wall, stabilizing baffle plate and sector coils for RWM control. The upper and lower divertor cassettes are asymmetric to allow us ITER shape plasma and high triangularity plasmas. The mono-block type CFC divertor armor is being planned to withstand heat load of 10-20 MW/m^2 for outboard side divertor armor, while the flat tile type divertor armor is planned for inboard side armor because the expected heat load is about half of that outboard side. They will be attached to a divertor cassette for a remote handling device. The cryo-panels will be installed under the flatter dome and/or the outer baffle plate aiming at a strong pumping capability. Two of in-vessel field coil will be used for a rapid plasma position control. The 18 sector coils for the RWM control will be attached around the hole of stabilizing baffle plate to avoid the magnetic shielding effect. Since the each sector coil has own current lead, the magnetic field mode of m/n = 3/1 or 3/2 can be controlled according to the power supply connection.

5. AC power system and water cooling system

Table III shows the required AC power for the plasma heating and current drive devices. The total plasma heating power of 41 MW-100s requires about 129 MW/155 MVAR and 12.9 GJ at load terminal. Since the maximum deliverable energy from the existing motor-generator set is 5 GJ,

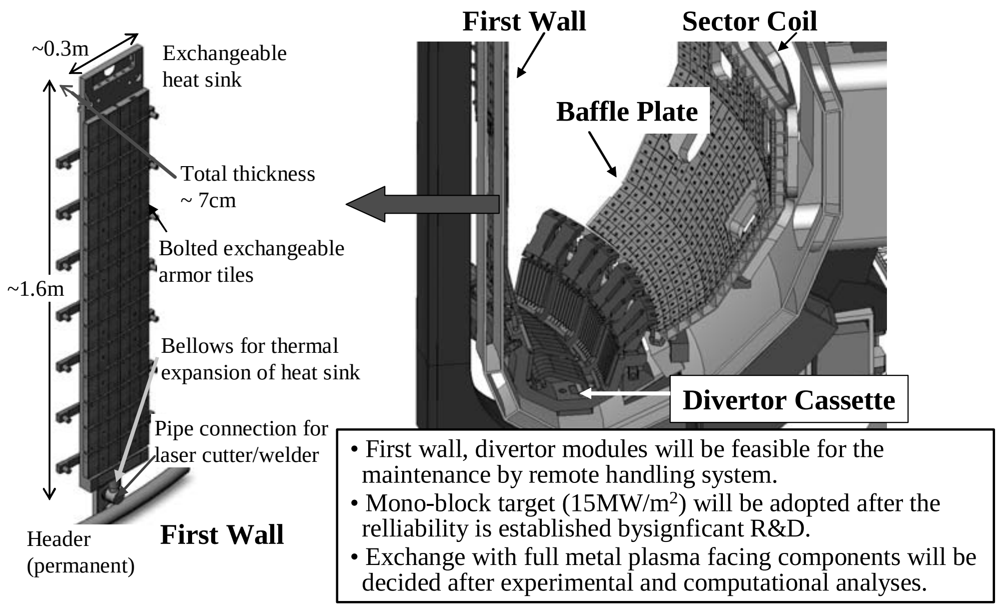

Fig.8. A design of divertor cassette, first wall and baffle plate.

the receiving power from the commercial power grid must be increased. Addition of static VAR compensator, harmonic filter, and dummy loads are also necessary for prevention of distorted power flow out. Figure 9 shows the outline of new AC power system.

TABLE III: SUMMARY OF REQUIRED AC POWER.

	Heating Power (MW)	Efficiency	Power factor	Active Power (MW)	Reactive Power (MVAR)	Energy (GJ)	Comment
P-NBI	24.0	0.40	0.60	60.3	80.1	6.0	2MW-12unit-100s
N-NBI	10.0	0.25	0.60	40.5	54.0	2.0	5MW-2unit-100s
ECH	7.0	0.25	0.80	28.0	21.0	2.8	7 MW
total	41.0	0.32	0.64	128.8	155.1	12.9	41MW-100s

The present primary water cooling system has the continuous heat removal capability of 10 MW for the in-vessel components. Since the maximum plasma heating power is 41 MW, 100 s, somewhat countermeasure is required. Then, temporary water buffer tank of 300 m^3 and powerful pumps are introduced to send cooling water to the divertor modules at the maximum flow rate of 4800 m^3/h for more than 100 s. It is planned to be installed on the underground floor of JT-60 torus building.

6. Conclusion

The JT-60SA design has much progressed and its feasibility has been increased remarkably with help of EU colleagues as a part of ITER Broader Approach. It is definitely possible to start this project immediately after the ratification expected at middle of 2007. The period of manufacturing, construction, assembling and test is expected to 7 years.

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References

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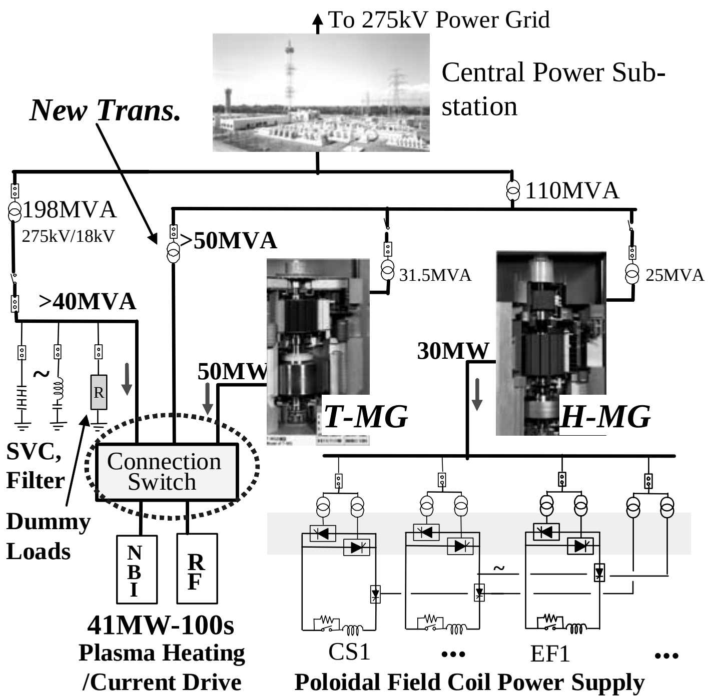

Fig. 9. Outline of AC power sources of JT-60SA.