Overview of Modification of JT-60U for the Satellite Tokamak Program as one of the Broader Approach Projects and National Program

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Abstract. An overview of modification of JT-60U equipped with fully superconducting magnets is presented, which is named as JT-60SA (JT-60 Super Advanced). The tokamak device is characterized with its high power (41MW) long pulse (100s) heating for ITER relevant configuration and low aspect ratio (A~2.6) and high shape factor (S~7) configuration to investigate shape optimization of DEMO. It also enables to investigate possible steady-state operation scenarios for ITER and DEMO at high current (I_p=2.4-3MA) with high beta (β_N>3.5-4.4) high bootstrap current fraction (f_{bs}>0.5).

1. Introduction

In parallel with ITER negotiation, Japanese government (JA) and European commission (EU) are establishing so-called “Broader Approach Program” toward early realization of fusion energy based on tokamak concept. Physics and technical experts from JA and EU (JA-EU satellite tokamak working group) made an assessment in 2005 that JT-60U modification program to a superconducting device [1, 2] can be a strong satellite tokamak program in the BA context if appropriate strengthening for the program is made. Thus this JT-60 modification program becomes combination of JA national program and JA-EU BA program to contribute and supplement ITER toward DEMO with a new device name “JT-60SA (JT-60 Super Advanced)”. Since then, JAEA continued to refine conceptual design of this device to meet its requirement. Bird’ eye view and major parameters of JT-60SA are shown in Fig.1 and Table 1, respectively. This paper describes overview of the JT-60SA program. Technical and physics aspect of this program are described in detail in [3] and [4], respectively.

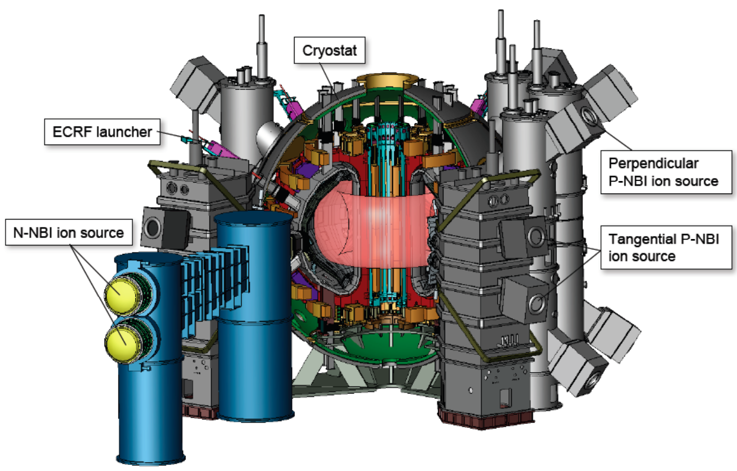

Figure 1 Birds eye view of JT-60SA with surrounding heating and current drive systems

JT-60SA parameters	
Plasma Current	5.5MA/3.5MA
Major Radius	3.01m / 3.16m
Minor radius	1.14m / 1.02m
Elongation κ_{95}	1.83 / 1.7
Triangularity δ_{95}	0.57 / 0.33
Toroidal field B_t	2.72 / 2.59
Safety factor q_{95}	3.77/ 3.0
Flat top	100s (8hours)
H&CD power	41MWx100s
Perp NB	16MW
Co P-NB	4MW
CTR P-NB	4MW
N-NB	10MW
ECRF	7MW
PFC heat flux	$10MW/m^2$
Annual neutron	$4x10^{21}$

Table 1 Major parameters of JT-60SA for low aspect ratio and ITER shaped plasma

2. Scientific Mission of JT-60SA

Main mission of JT-60SA is to support and supplement ITER toward DEMO. Scientific researches of JT-60SA in support of ITER are optimization of operational scenarios for ITER such as ELMy H-mode and hybrid mode, by using ITER-shaped plasma and wide variety of heating and current drive systems for high power density and various current profile controllability, and improved understanding of physics issues of long pulse discharges with comprehensive diagnostics systems, and testing of possible modifications before their implementation on ITER, such as metallic plasma facing components. As for the supplementary role toward DEMO, main missions in support of DEMO are to explore operational regimes of steady-state advanced high beta operation, and control of power and particle complementary to those being addressed in ITER.

National program specifically identify research targets toward DEMO. To clarify the research targets toward DEMO, the definition and scope of DEMO is important. Conventional definition of DEMO mission is demonstration of power production in plant scale. But, recent report of Atomic Energy Commission of Japan stressed that fusion DEMO needs to operate in steady-state and should have certain economical performance for early utilization of fusion energy [5]. Then exploration regime is set out as non-inductively driven high beta (β_N =3.5-5.5) collision-less plasma sustained for more than 100 second. Compatibility of ferritic steel as a first wall material of DEMO will also be addressed since reduced activation ferritic steel is primary candidate of blanket structural material of DEMO.

Wall stabilization is essential element to sustain such high beta plasma in tokamak. Proximity of the wall in DEMO is discussed in relation to disruption tolerance in [6]. It is concluded that we can only expect wall proximity b_{wall}/a_{plasma}~1.3-1.4 since stabilization effect is quite weak for the segmented blanket necessary to withstand disruption force in DEMO. This wall proximity limit sets achievable β_N in DEMO and also effective testing regime in JT-60SA. Behavior of ultra-long tokamak discharge such as change of material property of plasma facing components or operational reliability is one of important concerns in long pulse operation of DEMO as manifested in TRIAM-1M experiments [7]. Day-long (8 hours) steady-state operation will be performed to address this issue and to find out countermeasure towards DEMO as a long-term goal.

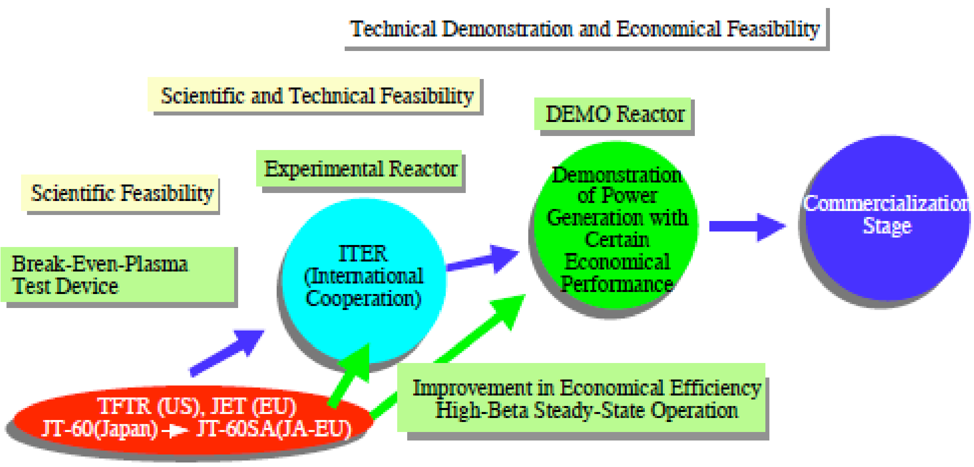

Figure 2 Role of JT-60SA program in fusion development strategy set by Atomic Energy Commission of Japan [5]

3. Tokamak Device

JT-60SA is a tokamak with superconducting magnets capable of confining break-even class high temperature plasma for 100s with heating and current drive power of 41MW. Cross sectional view of JT-60SA is shown in Figure 3.

Toroidal Field coils: The toroidal field at plasma center is modest to allow almost same Greenwald density for ITER-shaped plasma (Bt=2.6T at R=3.15m) and to explore fully non-inductive high β_N operation for DEMO at low aspect ratio (Bt=2.7T at R=3.04m). This choice allows use of NbTi conductor for toroidal field coil with B_{max}=6.5T and RBt=8.2Tm.

Poloidal Field Coils: Poloidal Field (PF) coils consist of 4 Central Solenoid (CS) coils (Nb_3Sn conductor with B_{max}=10T) and 7 Equilibrium Field (EF) coils (NbTi conductor with $B_{max}\leq$6.1T for EF3 and EF4, $B_{max}\leq$5T for other EF coils). These coils produces flux swing of about 40Wb to drive plasma current up to 5.5MA. EF7 coil is necessary to reproduce ITER shaped equilibrium.

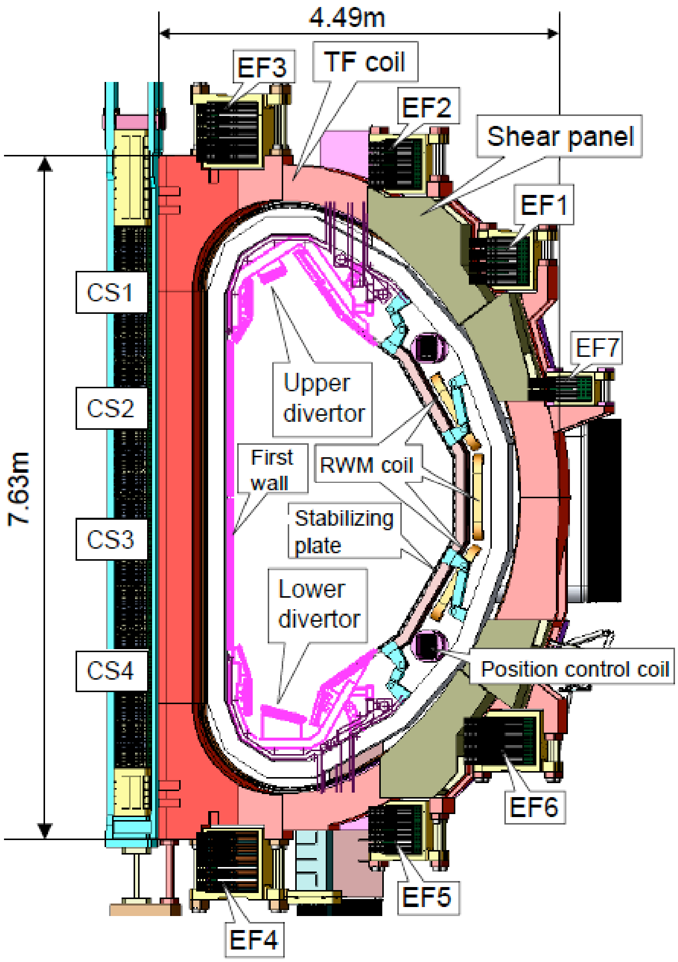

FIG. 3. Cross sectional view of JT-60SA tokamak showing TF, PF magnets, vacuum vessel, top and bottom divertors, stabilizing plates, in-vessel vertical position control coils, and RWM control coils

Error Field Correction Coils: Error field correction coils consist of 6 sets of 3 sector coils (NbTi conductor) located attached outside of TF coils.

Divertor: The device equips with top and botton water-cooled CFC divertors to handle high heating power up to 50MW taking into account of future upgrade of heating system, which are designed to match the ITER shaped plasma with bottom divertor, and the strongly shaped low aspect ratio plasma with upper divertor, respectively. Inner and outer targets of these divertors will be covered with brazed flat type ($q_{heat}\leq$10MW/m^2) and mono-block type ($q_{heat}\leq$15MW/m^2) divertors, respectively. Total water flow rate in the primary cooling loop for divertor is 4800m^3/h.

First Wall: Inner and outer first wall are covered with carbon tiles bolted to Cu alloy heat sink and stainless steel stabilizing plates, respectively.

Stabilizing Plates and In-vessel Coils: The stabilizing plates inside the vacuum vessel stabilize the vertical positional instability with in-vessel vertical position control coils and also stabilize free boundary n=1 and 2 Resistive Wall Modes (RWM) with RWM control coils to sustain DEMO-relevant high beta plasma. The RWM control coil consists of 6 sets of 3 sector coils located at upper, lower and horizontal holes of stabilizing plates. Stabilizing plates will be covered with ferritic plates to simulate ferromagnetic effect of blanket materials.

Vacuum Vessel: The vacuum vessel is made of double-walled 316L SS with low Co content less than 0.05wt% to minimize induced activation. One turn resistance of the vacuum vessel is 13.5μΩ (L/R=0.14s) including contribution of stabilizing plates. The vacuum vessel is filled with boronic acid water whose temperature can be increased to 200°C at 2.5MPa during baking operation.

Cryostat : The cryostat is made of double-walled SS304 filled with boron doped concrete. Activation of Ar is effectively reduced by this boron, which effectively reduces thermal neutron.

Remote Handling: Large annual neutron fluence of 4×10^{21} neutrons/year nearly prohibits the human access inside the vacuum vessel after extensive experimental campaign. So, most of in-vessel components should be compatible with remote handling. The remote handling system of JT-60SA is a vehicle type system adopted in ITER with a possibility to use its rail as a boom.

Pumping and Fuelling : Divertor pumping will be provided by the cryopanels with pumping speed of ~100m^3/s for 100s operation while turbo-molecular pumps becomes main divertor pumping method (pumping speed of ~20m^3/s) for day-long operation. Fuelling will be provided by both gas puffing (several 100Pam^3/s) and pellet injection (up to 50Pam^3/s).

Port Arrangement: Port arrangement of various diagnostics system, H&CD system, RH system, and coil feeders and coolant piping are assigned as shown in Figure 4.

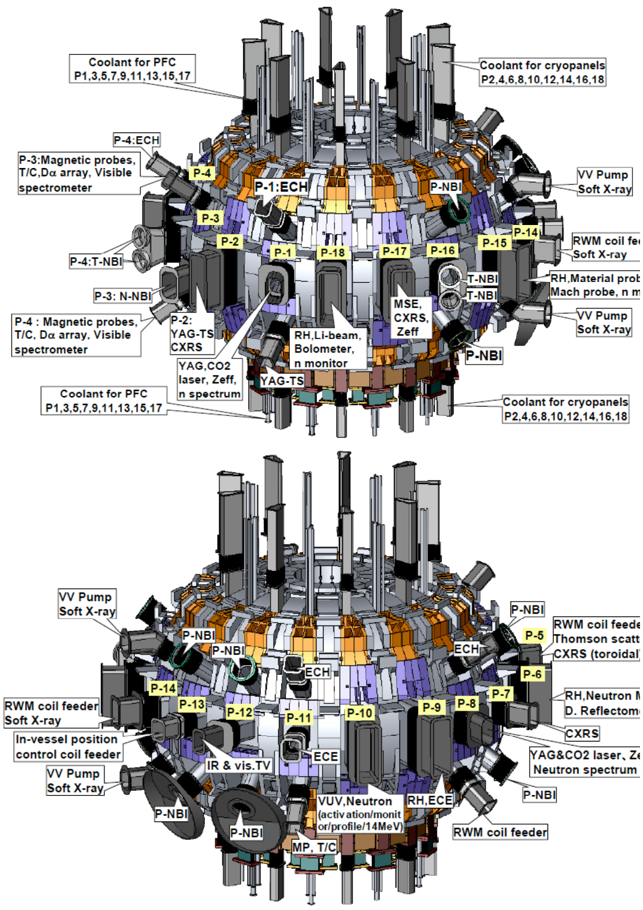

FIG. 4. Port allocations of JT-60SA.

4. Heating and Current Drive system

The heating and current drive (H&CD) system in JT-60SA is upgraded compared with those planned in [2] to allow power density in excess of ITER and to allow independent control of heating, current and rotation profile controls, which are essential to expand advanced tokamak regime. The H&CD system in JT-60SA consists of Positive-ion based NBI (P-NBI) with beam energy of 85keV, Negative-ion-based NBI (N-NBI) with beam energy of 500keV and two ECRF systems (110GHz and 140GHz).

Eight perpendicular P-NBI units provide 16MW of heating power with little current and rotation drives. Two co-tangential (4MW) and two counter-tangential (4MW) P-NBI provide flexibility in off-axis CD and rotation control. A 500keV co-tangential N-NBI (10MW) is unique feature among various satellite tokamaks in support of ITER, which enables efficient CD and produces energetic particles for AE study. The beam-line of N-NBI is lowered by 0.6m from the mid-plane as seen in Figure 5. Preparation of two ECRF systems with different frequencies enables effective NTM stabilization for a range of toroidal field, Bt=1.5-2.7T.

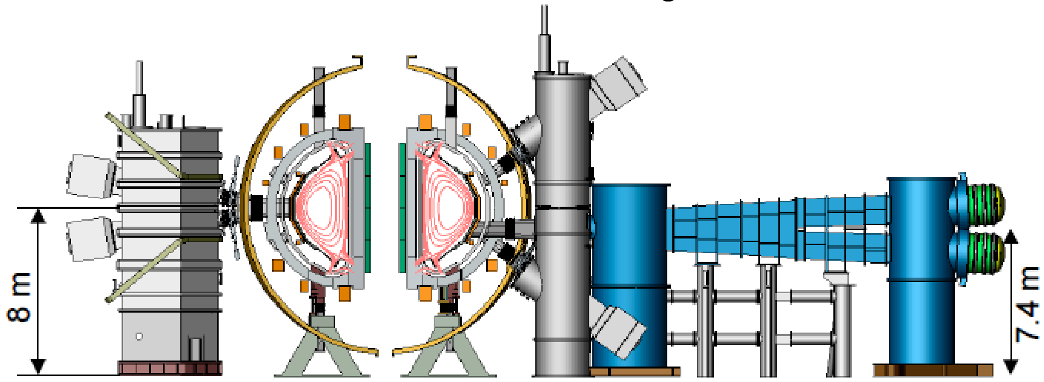

Figure 5 Tokamak and 85keV tangential P-NBI and 500keV N-NBI system.

5. Operating Scenarios of JT-60SA

According to the mission of JT-60SA described in section 2, JT-60SA will explore both ITER relevant regime to support ITER and regimes not covered in ITER to supplement ITER toward DEMO. Figure 6 shows typical plasma shapes in JT-60SA. Plasma shape identical to ITER with a scale factor $R_{ITER}/R_{JT\text{-}60SA}$ of about 2 can be sustained at Ip/Bt=3.5MA/2.6T for 100s allowing contribution to ITER by optimizing operational scenario and further understanding of physics for ITER. It also accommodates low aspect ratio (A~2.6) single null and double null equilibrium to perform configuration optimization of tokamak system.

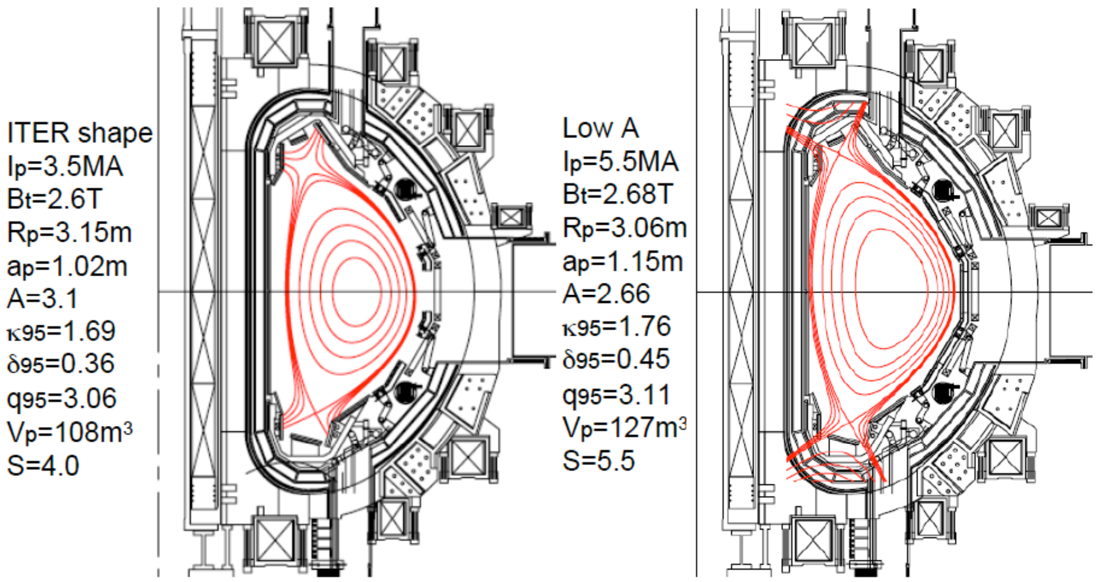

FIG. 6. Left figure shows possible ITER shaped plasma for contribution to ITER with maximum plasma current of 3.5MA. Right figure shows lowaspect ratio(A=2.66) double null plasma with maximum plasma current of 5.5MA.

ITER support scenario

JT-60SA will provide valuable opportunity to develop ITER operating scenario of ELMy H-mode and hybrid scenario under intensive H&CD of 41MW far exceeding H-mode threshold power of ~15MW. With improved heating and CD capability, JT-60SA has distinct characteristics as follows.

(1) Combination of 10MW of N-NBI with beam energy of 500keV and 7MW of ECRF with the frequency of 110GHz and 140GHz provide unique opportunity to simulate ITER

experiments with dominant electron heating and significant energetic particle population.

(2) Capability of 4MW of co-tangential and 4MW of counter-tangential NBI at 85keV is effective to affect Type I ELM characteristics through the modulation of edge plasma rotation in co and counter directions.

(3) 16MW of perpendicular NBI at 85keV provides independent control tool of plasma pressure without current and rotation drives, which also enable to simulate burn control for ITER.

(4) Divertor target of JT-60SA experiences high power and particles fluxes close to those of ITER and provide important information on the performance of flat and mono-block type divertors for ITER.

ITER supplement scenario

JT-60SA also provides regimes not covered in ITER to supplement ITER toward DEMO with wider opportunities in plasma shape, two different divertors in top and bottom, wider variation of H&CD system than those achievable in ITER. Major opportunities are as follows,

(1) Comparison between high tri-angularity upper single null divertor and low tri-angularity lower single null divertor allows investigation of effect of tri-angularity on edge stability and its role on ELM characteristics since upper divertor is designed to match specifically to strongly shaped (high κ and δ) plasma.

(2) JT-60SA can also produce low aspect ratio (A=2.6) strongly shaped ($S=q_{95}Ip/aBt=6$) single/double null plasmas (see Figure 6 right) to explore wider possibility of DEMO configuration such as proposed in recent JAEA DEMO design [8].

(3) Wide variety of current profile control becomes possible with slightly down shifted N-NBI beam line. Upper single null divertor can produce non-inductively driven strongly reversed shear or current-hole plasmas while lower single null divertor provides positive to weakly negative shear plasmas. Figure 7 shows typical example of fully non-inductively driven high beta plasma relevant for advanced DEMO.

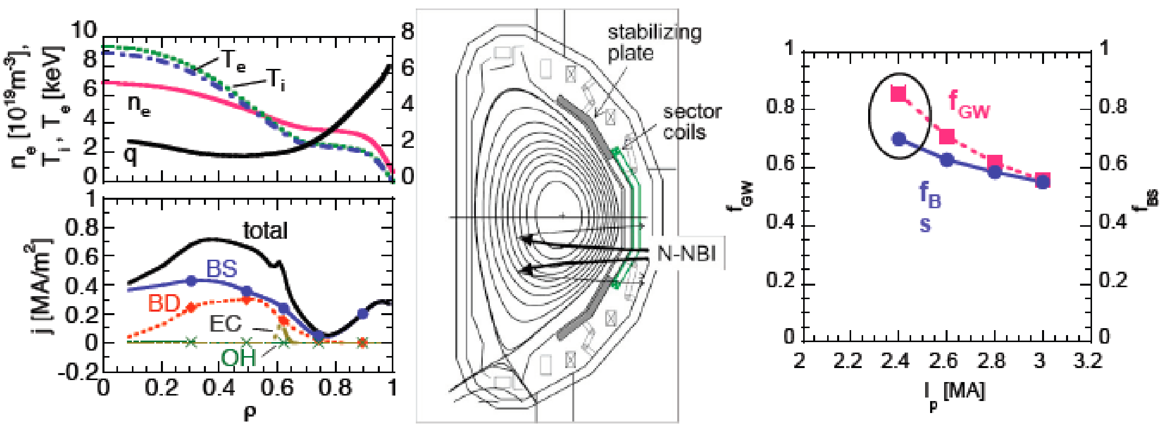

FIG. 7. Current profile and equilibrium of 2.4MA full current drive plasma with A=2.65, β_N=4.4, f_{GW}=0.86, f_{bs}=0.7 which becomes possible with total heating and CD power of 41MW if H_{Hy2}=1.3 is achieved. Right figure shows values of f_{GW} and f_{bs} as a function of non-inductively driven plasma current.

(4) Wide variety of feedback control of Resistive Wall Mode (RWM) becomes possible with a comprehensive RWM control coils for n=1 and 2 mode control and co/counter and zero plasma rotation produced by Co/Counter NBI with minimum magnetic braking of toroidal rotation optimized by the error field correction coils. Physics assessment is in

progress and VALEN code study shows n=1 RWM can be stabilized close to β_N=4.3 as shown in Figure 6 [4].

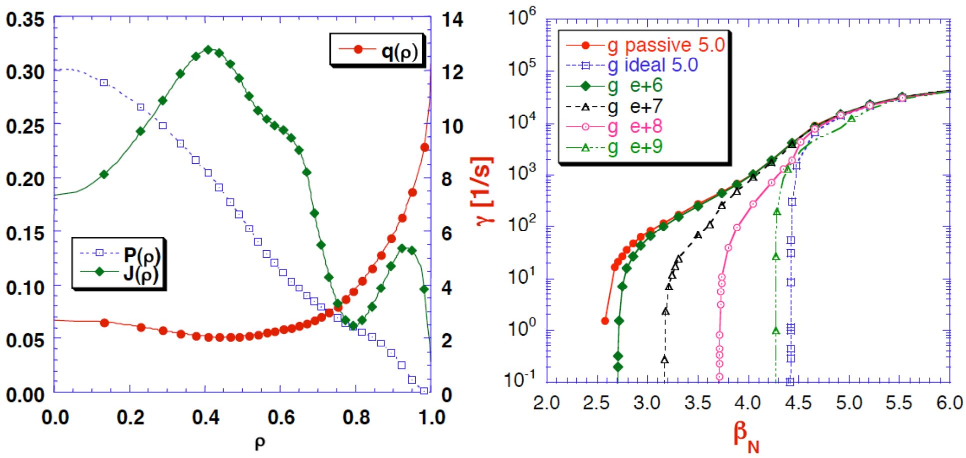

FIG. 6 Pressure, current and safety factor profiles of full non-inductive plasma (I_p=2.4MA, B_t=2.6T, R_p=2.95m, a_p=1.05m, q_{95}=7.5, q_{min}=2.1)(left) and Growth rate of RWM as a function of β_N with various feedback gain g (right)

6. Diagnostics

Comprehensive diagnostics system will be installed in JT-60SA as given in Table 2 to support and supplement ITER. Most of comprehensive set of diagnostics systems listed in Table 2 will be installed from the initial operation of JT-60SA. Basic port arrangements for most of diagnostics systems have been decided and typical examples are shown in Figure 6.

Table 2 Diagnostics system for JT-60SA

Machine Protection and Operation	Fundamental Parameter Measurement	Physics Understanding
^{235}U,^{238}U neutron monitor	50Hz YAG laser Thomson scattering	4 set of 20ch soft X-ray detectors
neutron activation measurement	Tangential and vertical CO2 laser interferometer/polarimeter	6 chord neutronprofile monitor, 14MeV neutron detector, neutron spectrometer
3 visible TVs	ECE(512ch FTS,40ch GP,48ch HR)	Li-beam probe
32ch. Dα emission monitor	CXRS (18ch.toroidal, 18ch. Poloidal)	3ch. O-mode reflectometer
Divertor Langmuir probe	14 ch. Zeff monitor	Reciplocating Mach probe
Upper and lower div. IRTV cameras	VUV spectrometer	Visible spectrometer for divertor
Utility system for Diagnostics	2 sets of 30ch. MSE polarimeter	6 Penning and 10 AUG Gauges
Data acquisition system	32 ch. Core and divertor Bolometers	

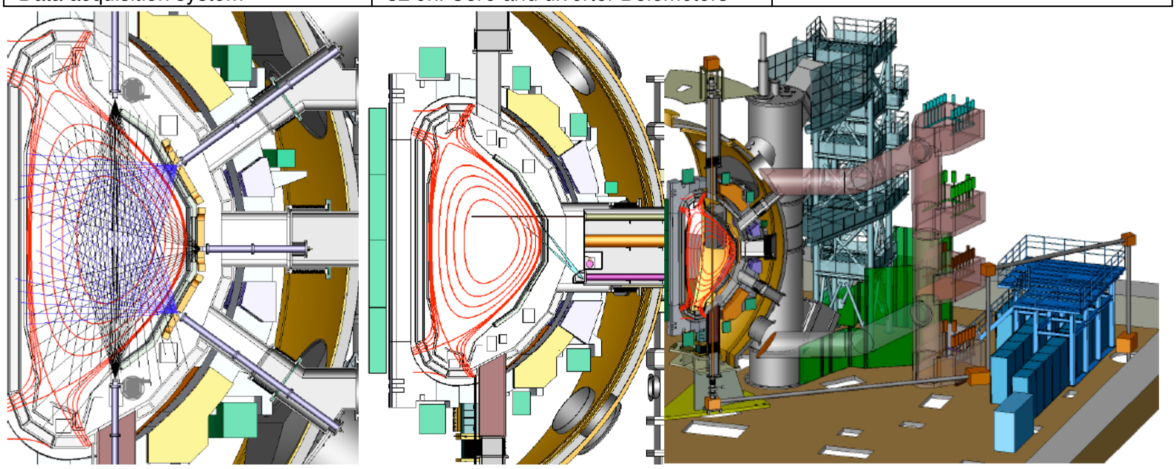

Fig. 6 Examples of diagnostics port arrangement for soft X-ray detector array, Li-beam diagnostics and CO2 interferometer (see also Fig. 4)

7. Safety Consideration

Number of safety measures will be done for improving safety characteristics of JT-60SA. The concrete inside the cryostat is doped with Boron 2Wt% to reduce ^{24}Na gamma dose rate outside of cryostat and ^{41}Ar produced by the thermal neutron days after the yearly operation. Increased air ventilation capacity is planned to keep the air concentration of gaseous radioactivity within legal limit. Intensive DD experiments induce radioactivity in tokamak components. It is important to minimize such induced activity by optimum selection of materials such as low Co concentration SS.

8. Construction Plan

The construction of JT-60SA will take 7 years and 3 years of operation is considered during 10 years of BA period. Basic construction plan is shown in Figure 7. It is agreed that EU contribution is in-kind contribution of Toroidal Field Magnets, Cryostat, Power Supply, Cryogenics System, and ECRF system.

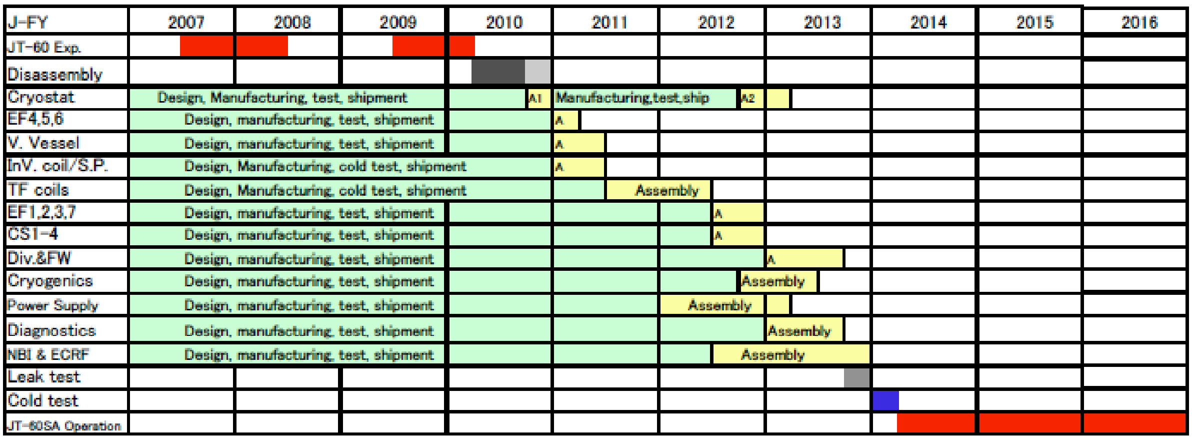

Fig. 7 Basic construction schedule of JT-60SA.

7. Summary and Discussion

Conceptual design of JT-60SA is progressing and will be completed before the start of BA agreement between Japanese government and European commission. The JT-60SA tokamak has a capability to address key physics issues for ITER/DEMO.

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