FSD Science Coordination Meeting on the JT-60SA divertor

Scientific objectives of JT-60SA and rationale for a metal divertor

18th July 2023

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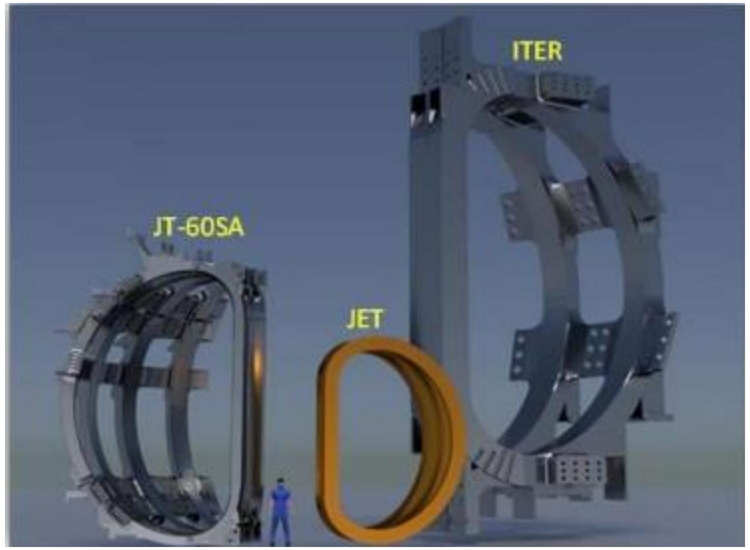
This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

JT-60SA features and its mission

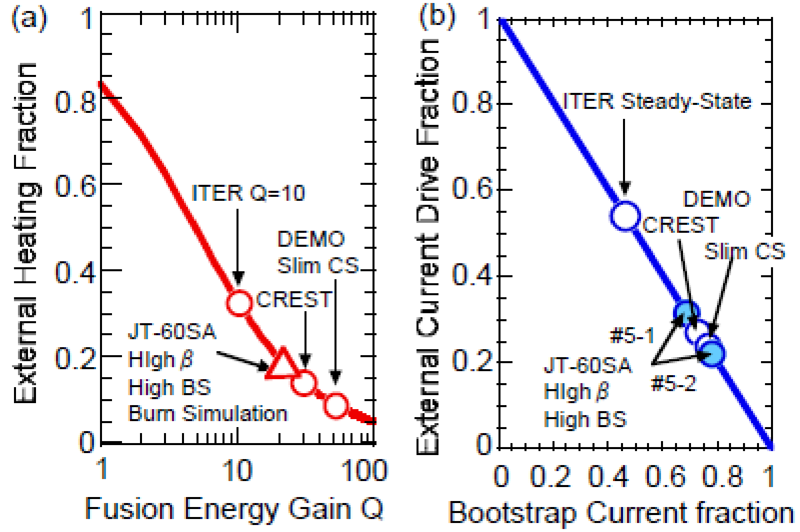
- JT-60SA is the largest tokamak built (before ITER) with low temperature (4K) superconducting coils: the plasma major radius is R_P~3.0 m, and minor radius a_P~1.2 m, the plasma volume ~130 m^3
- JT-60SA is designed for high plasma current: I_P/B_T=5.5 MA/2.3 T, high power and long pulse: 41 MW × 100 s
- JT-60SA has high shape parameters: shaping factor $S=q95I_P/(a_PB_T)$ ~7, aspect ratio A~2.7, elongation k_x ~1.9, triangularity δ_x~0.5 aiming to control the plasma stability at high plasma pressure and to maintain fully non-inductive steady-state for long time

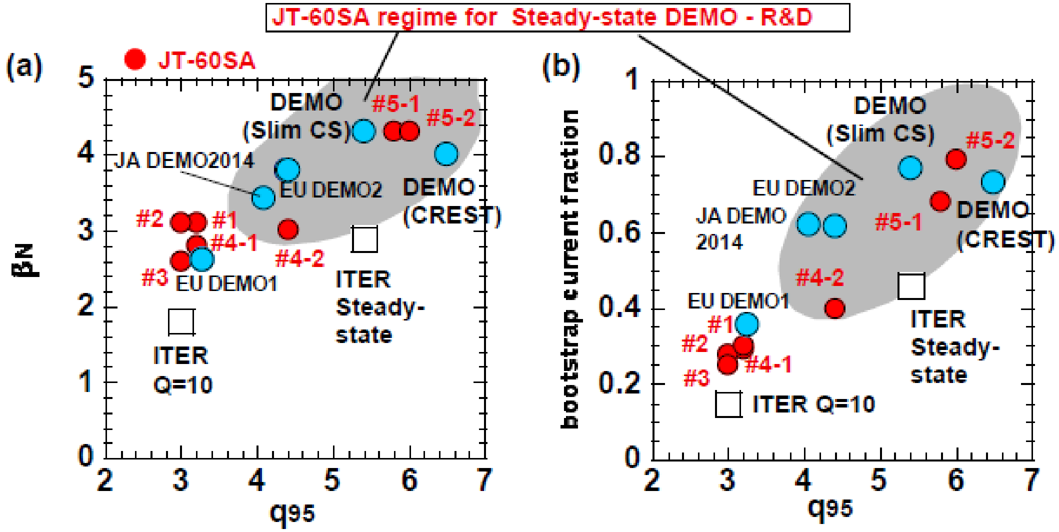


- Mission:
  - confinement and control of breakeven-equivalent class high temperature deuterium plasmas lasting more than the time scales of the key plasma processes such as current diffusion and particle recycling
  - **Aim at fully non-inductive steady-state high β_N operations above the no-wall ideal MHD stability limits, for long time (~3-4τ_R)**
  - **=> address key physics issue for ITER and DEMO**
  - **=> maintain and develop key operational know-how for assembly, commissioning, operation, exploitation of high performance, high complexity superconducting devices for nuclear fusion**

Unique contributions of JT-60SA to the Fusion Roadmap/1

- Steady-state, long pulse, non-inductive, plasma studies at high beta and high Greenwald fraction including ITB formation and sustainment with off-axis current
- Long pulse operation at high thermal energy and triple product
- **λ_q studies at high current, 5.5MA, well above previous JET studies**
- Well confined ~500keV fast ions at 5.5MA allowing fast ions studies, at high density, with relevant diagnostics
- Possibility of studies of well confined alphas in D-^{3}He plasmas
- **Disruption mitigation and suppression, runaway electron** studies at high thermal and magnetic energies, well above previous studies at JET
- Electron heating studies and controllability of electron/ion heating with different source of electron heating (ECRH vs N-NBI)
- **=> Integrated scenario development including all the previous points and sophisticated real time control techniques**

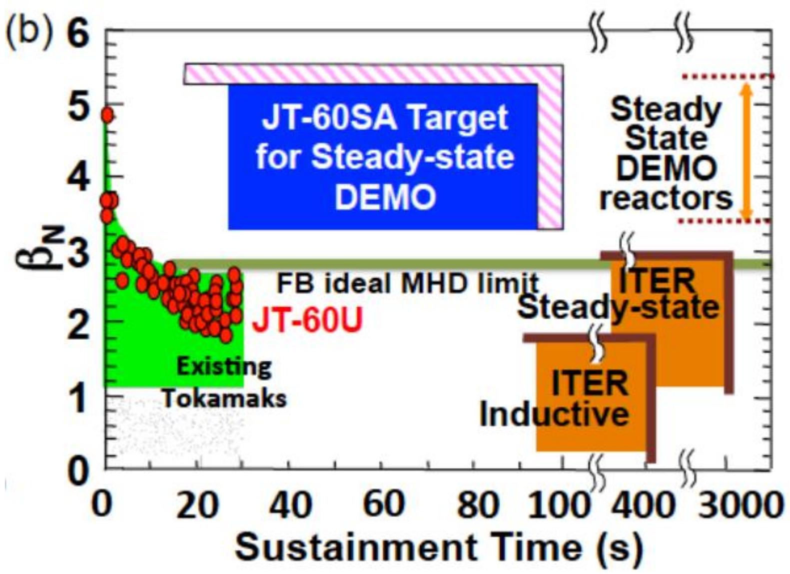




Unique contributions of JT-60SA to the Fusion Roadmap/2

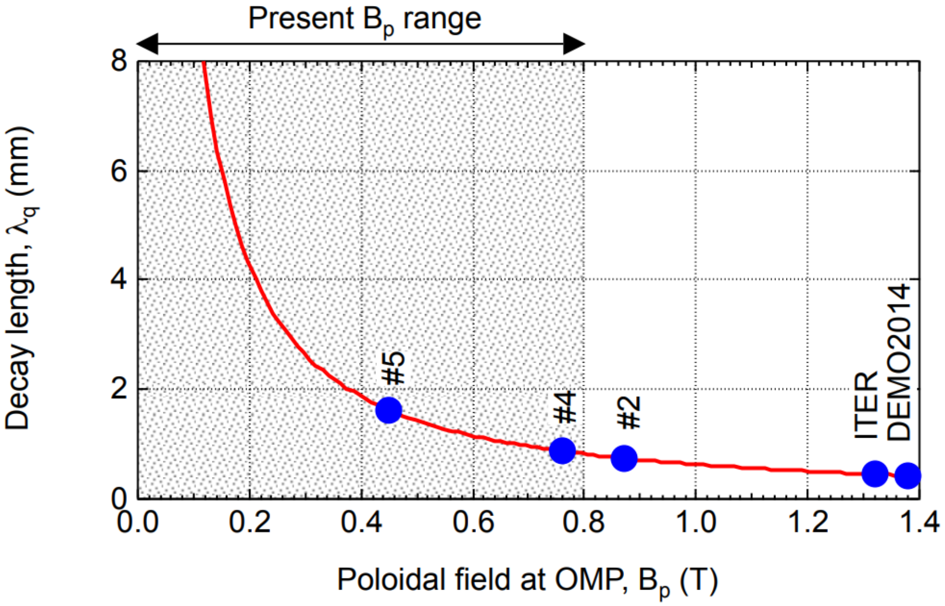
High beta operation (Mission 1)

- Steady-state scenarios development is essential for DEMO
- DEMO relevant long pulse, non-inductive plasmas development are possible in JT-60SA
- Unique operational space at β_N ~4 and Greenwald fraction ~1
- ITB formation and sustainment can be studied with off-axis current
- Long pulse operation at high thermal energy and triple product



Power exhaust (Mission 2)

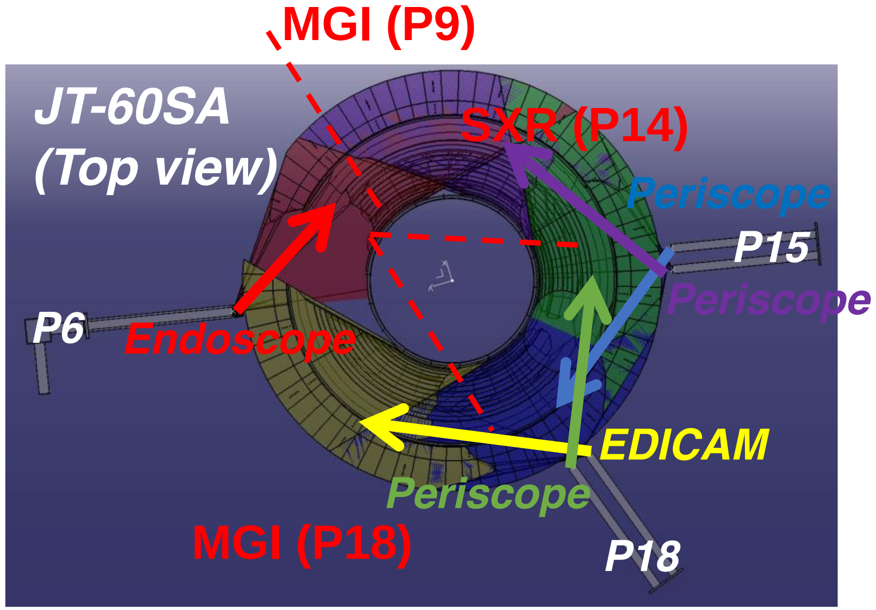
- The power decay length (λ_q) has a dependence mainly on Ip
- Expected to be very small for ITER
- JT-60SA can expand the current domain well above JET results
- Dedicated edge diagnostics can be used for the physics of interpretation of λ_q at high Ip→ Extrapolate to ITER and DEMO

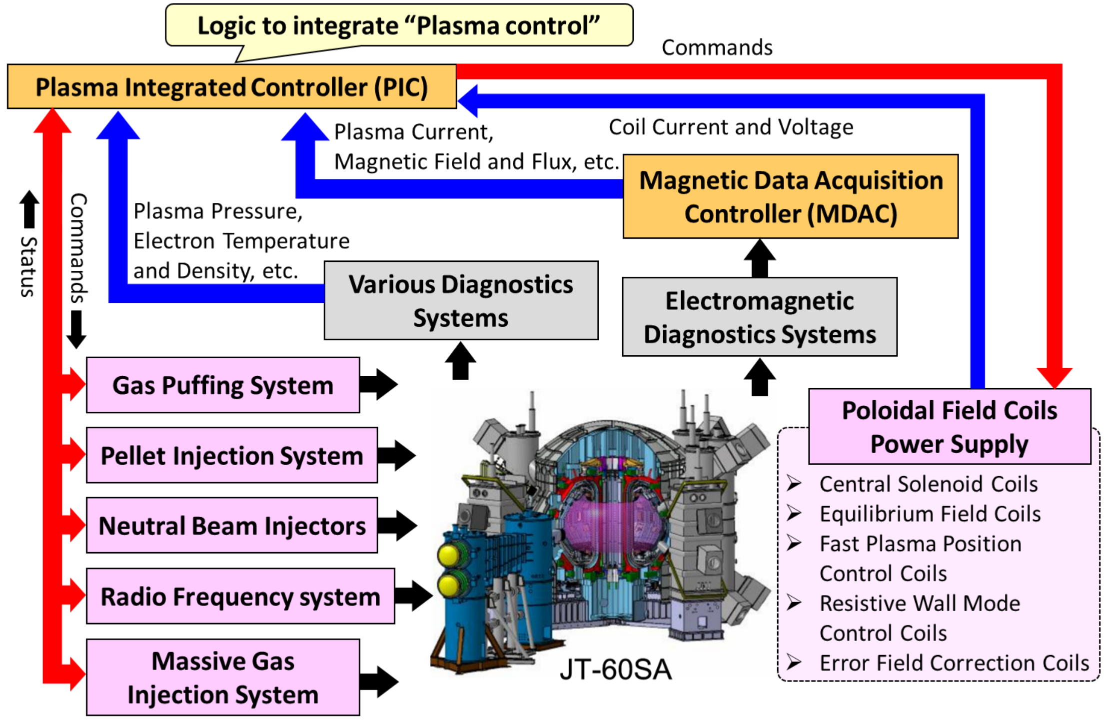


Unique contributions of JT-60SA to the Fusion Roadmap/3

Disruptions and runaway electrons

- JT-60SA can provide evidence of disruption mitigation and suppression at high magnetic and thermal energy, higher than JET
- Essential to extrapolate to ITER and DEMO
- 2 MGI systems provide flexibility for disruption studies (SPI can be considered in future)
- Run away electrons formation and mitigation can be studied at high Ip





Logic to integrate "Plasma control"

Commands

Plasma Integrated Controller (PIC)

Coil Current and Voltage

Plasma Current, Magnetic Field and Flux, etc.

Magnetic Data Acquisition Controller (MDAC)

Plasma Pressure, Electron Temperature and Density, etc.

Various Diagnostics Systems

Electromagnetic Diagnostics Systems

Status

Commands

Gas Puffing System

Pellet Injection System

Neutral Beam Injectors

Radio Frequency system

Massive Gas Injection System

JT-60SA

Poloidal Field Coils Power Supply

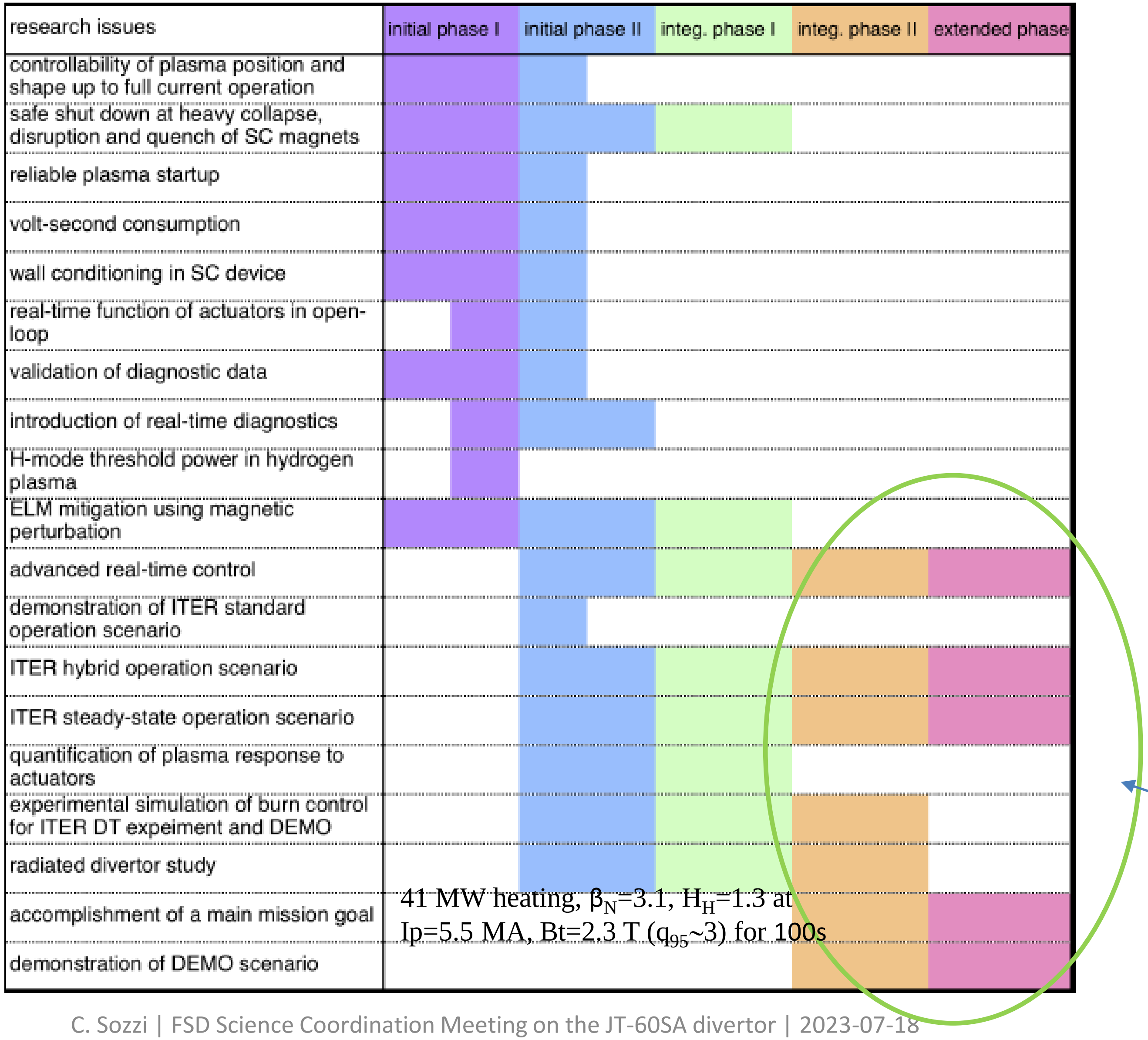
- Central Solenoid Coils
- Equilibrium Field Coils
- Fast Plasma Position Control Coils
- Resistive Wall Mode Control Coils
- Error Field Correction Coils

Real time control

- JT-60SA scenarios target ELMy H-mode, High Ip, high beta, steady-state, high fast ion content, divertor detachment→ real-time control required
- JT-60SA provides a test-bed for sophisticated techniques based on neural networks and AI
- Essential information for ITER and DEMO

Transition to full metal wall in the **JT-60SA Research Plan** (v.4, 2018)

Table 3-4 Experimental program for operation scenario development



research issues	initial phase I	initial phase II	integ. phase I	integ. phase II	extended phase
controllability of plasma position and shape up to full current operation	🟪	🟦			
safe shut down at heavy collapse, disruption and quench of SC magnets	🟪	🟦	🟩		
reliable plasma startup	🟪	🟦			
volt-second consumption	🟪	🟦			
wall conditioning in SC device	🟪	🟦			
real-time function of actuators in open-loop	🟪	🟦			
validation of diagnostic data	🟪	🟦			
introduction of real-time diagnostics	🟪	🟦			
H-mode threshold power in hydrogen plasma	🟪				
ELM mitigation using magnetic perturbation	🟪	🟦	🟩		
advanced real-time control		🟦	🟩	🟧	🟥
demonstration of ITER standard operation scenario		🟦			
ITER hybrid operation scenario		🟦	🟩	🟧	🟥
ITER steady-state operation scenario		🟦	🟩	🟧	🟥
quantification of plasma response to actuators		🟦	🟩		
experimental simulation of burn control for ITER DT expeiment and DEMO		🟦	🟩	🟧	
radiated divertor study		🟦	🟩	🟧	
accomplishment of a main mission goal				🟧	🟥
demonstration of DEMO scenario				🟧	🟥

41 MW heating, β_N=3.1, H_H=1.3 at Ip=5.5 MA, Bt=2.3 T (q_{95}~3) for 100s

- With JT-60SA's objectives being sustaining high-confinement and high-density plasmas for up to 100 s at high heating power the primary aim of divertor, scrape-off layer (SOL) and plasma wall interaction (PWI) research is the safeguarding of the plasma facing components of JT-60SA.
- Furthermore, JT-60SA shall serve as an experiment to provide a solid scientific basis for power and particle exhaust for supporting the operation of ITER and extrapolating to DEMO, for which integrated scenarios in a metallic wall are a key.
- Therefore, JT-60SA will undergo two distinct operational phases in view of PWI and power exhaust studies. These two phases consist of PFCs being either C or W. The first phase will consist of C as a plasma facing component.

full metal wall, with all PFCs will be covered by either W coated CFC or potentially even solid W (high flux areas)

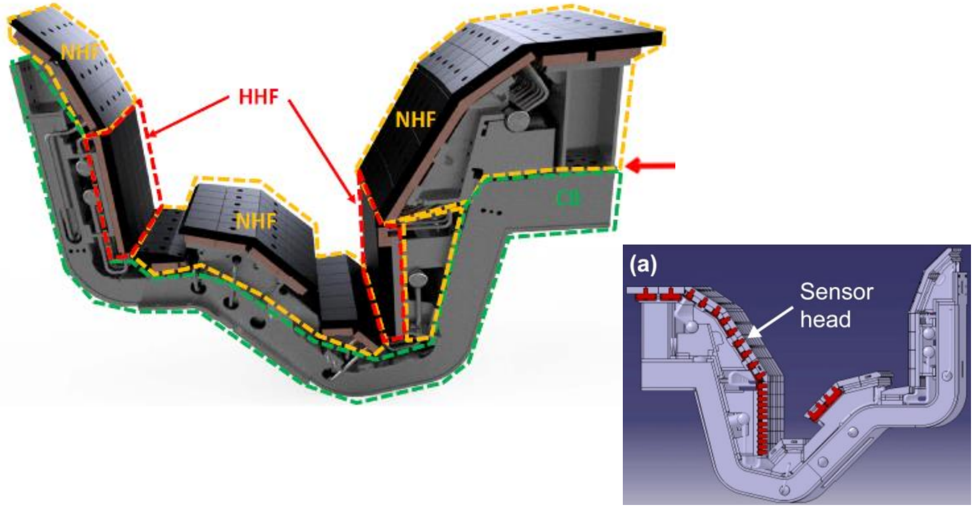
European interests in JT-60SA

EU Strategic priorities in the JT-60SA research program as in https://idm.euro-fusion.org/?uid=2NPW2R&version=v1.1

- 1. Development and investigation of high performance scenarios compatible with future W-PFCs.
  - Compatibility of small/no ELM, radiative divertor and high beta
  - Pedestal characteristics at high density with pellet fueling
  - Detachment physics and code validation
- 2. Avoidance and mitigation of disruptions and runaways
- 3. Fast ion physics
- 4. Development and validation of high level real-time control strategies

The configuration of the tokamak will evolve in line with its exploitation and the available resources. In particular

- the available heating power is increased progressively
- at least 3 different lower divertors are expected to be installed successively: inertially cooled carbon, actively cooled carbon and actively cooled tungsten (peak power density to the actively cooled divertor of up to 10 MW/m^2 for 100 seconds and up to 15 MW/m^2 for 5 sec for an integrated number of up to 13000 pulses)
- as high-power and/or long-pulse operation with deuterium plasmas approaches, in-vessel components should be equipped to be compatible with remote maintenance



Contribution to the divertor development in FP9

- WPDIV
  - development of the JT-60SA divertor components
- WPPWIE
  - support in the optimization of the divertor design for plasma-surface interaction
- WPSA
  - support to the JT-60SA Experiment Team for the scenarios development/adaptation, including the modelling activity
  - development of the diagnostics, in cooperation with F4E
- WPTE link with experimental activity in EU

Long term planning

Research phase	Focus of exploitation	Operation Campaign	Expected operation schedule	[1]	Annual neutron limit	RH	Divertor	Installed NB power	ECRF	Max. usable aux. power[2]
-	Integrated pre-plasma Commissioning	Op-1	2020-2021 (6M) 2023 (6M) First plasma 2023	H	- (N_2 in VV interspace[3])	R&D	Open upper inertially cooled carbon[4]	0	1.5 MW (2 Gyro.)	1.5MW
Initial research phase I	**Initial stable and reliable operation** • H operation for commissioning towards D operation. • Stable operation at high current heated plasma									
		Op-2	2025-2026 (9M)				Inertially cooled lower pumped carbon[5] (limits high power heating duration)	PNB 8 units, plus NNB Total 16MW (with H) 23.5 MW (with D)	3 MW (4 gyro)	19MW
				D						26.5MW
Initial research phase II	**ITER and DEMO regime access** (high power and high Ip with short pulses) • Access to the ITER standard scenario • High beta access • ITER risk mitigation (ELM, disruption)	Op-3	2026-2027 (9M)		3.2e19 (N_2 in VV interspace)					
		Op-4	2028 (8M)					PNB 12 units, plus NNB Total 30 MW		33 MW
Integrated research phase I	**High beta long pulse** **Burning plasma relevant** • ITER standard and hybrid stationary (~2-3tR) • High beta steady-state (~2-3tR), DEMO contribution	TBD	TBD		4e20 (water in VV interspace)		Actively cooled lower pumped carbon[6]		7MW (9 gyro.)	37MW
Integrated research phase II	**High beta and metal wall compatibility** • Radiative divertor with impurity seeding • Impurity pumpout from core	TBD	TBD		1e21 (water in VV interspace)	Use	Actively cooled lower pumped tungsten			
Extended research phase		TBD	TBD		1.5e21 (boronated water in VV interspace)		Actively cooled tungsten advanced structure (Upper div. TBC)	34MW[7]		41MW