

CONFERENCE PRE-PRINT***FIRST JT-60SA PLASMA OPERATION AND PLANS IN VIEW OF ITER AND DEMO***

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Abstract

JT-60SA is the world's largest superconducting tokamak in operation jointly built and exploited by Europe and Japan in the framework of the Broader Approach. JT-60SA aims at addressing some of the technological and physics challenges, such as the long pulse steady-state plasma operation at high beta, which will characterize future tokamaks producing net electrical energy by fusion reactions. The start-up of JT-60SA, which culminated in the first JT-60SA plasma achieved on 23rd October 2023 and Operation-1 (OP-1) until the end of 2023, including the achievement of $>1\text{MA}$ diverted plasmas, paves the way for a new generation of large superconducting tokamaks, such as ITER. Several key scientific topics were investigated during this initial phase. Similarly to ITER, the available parallel electric field ($E_{\parallel}$) is low and yet plasma breakdown was quickly obtained by means of the Trapped Particle Configuration (TPC), used for the first time on a tokamak this size, with the assistance of $\sim 1.5\text{ MW}$ of Electron Cyclotron Resonance Heating (ECRH). The equilibrium reconstruction and plasma control were also demonstrated including a validation of the ITER real-time reconstruction algorithms for the plasma current, centroid position and boundary. A first analysis and classification of the causes for disruptions have been done after the results of OP-1. Vertical Displacement Events (VDE) were responsible for the vast majority of disruptions in highly elongated plasmas, as the stabilization plate was not yet installed in this phase. Therefore, VDE predictors and control algorithms were developed using machine learning techniques with magnetics probe data, showing that these novel techniques are also suitable for the start-up tokamak phases characterized by scarce input data. The generation of runaway electrons, one of the main concerns for ITER, is observed in both disruptive and non-disruptive plasma phases, and it has been correlated to the bursty activity of the tearing mode even in non-disruptive phases. JT-60SA will restart operation in 2026 following a series of upgrades. The experimental programme for future operations is guided in the JT-60SA Experiment Team by significant modelling “predict first” activity, which shows that access to and development of H-mode in conditions of future burning plasmas, i.e. with high electron heating, low torque, fast ion-driven perturbations and high beta will be possible with high Negative Neutral Beam Injection (N-NBI) and ECRH input power. The integration of such elements into a steady-state long pulse operation will be done with the installation of W Plasma Facing Components (PFC) after the initial campaigns.

1. INTRODUCTION

JT-60SA is the largest superconducting tokamak in operation since the 23rd October 2023, when the first plasma was produced. This achievement is a milestone of the JT-60SA project, which started in 2007 based on the agreement between EU and Japan for the joint implementation of the Broader Approach (BA) activities in the field of fusion energy research (BA Agreement) as well as Japanese domestic fusion research and development programme [1]. After 2007, the tokamak assembly started from 2013 and the assembly was completed in 2020. The Integrated Commissioning was conducted during the first plasma operation phase in 2020 and 2023 [2,3].

JT-60SA has three important missions, i.e. (i) addressing potential ITER related issues in advance while having the capability of optimizing ITER operation scenarios at break-even equivalent conditions, (ii) conducting complementary research to ITER, especially studying long pulse steady-state operation and related physics at high β_N and high performance plasmas for providing directions to the DEMO design activity, and (iii) fostering next generation scientists and engineers by building up their experiences through JT-60SA operation. In order to attain these goals, JT-60SA can develop high current, highly shaped plasmas with high heating by systems such as Neutral Beam Injection (NBI) and Electron Cyclotron Resonance Heating (ECRH), which provide flexibility in terms of torque and electron to ion heating. A summary of the main characteristics of JT-60SA can be found in

Table 1 and a description about how the construction and development of the JT-60SA project has been shared between Japan and Europe is shown in Fig. 1.

The scientific exploitation of JT-60SA has been divided into different phases [4] as shown in Table 2. JT-60SA entered into the start-up phase with the Initial research phase I by performing the integrated commissioning during 2020-2021 and 2023. This integrated commissioning had the goal to test the integrity of the machine to fulfil its scientific expectations and to provide a first set of scientific results that could be used to better prepare the following experimental campaigns. It is worth to point out that the start-up of a superconducting device of the JT-60SA size marks by itself an unprecedented achievement that can definitely contribute to the future start-up of other superconducting tokamak devices such as ITER.

TABLE. 1. Typical parameters of JT-60SA.

Magnitude	Value
B_t	2.25 T
I_p	5.5 MA
R/a ($A = 2.5$)	2.96 / 1.18 m
κ/δ	1.93 / 0.5
t (flat-top)	100 s
N-NBI (500 keV)	10 MW
P-NBI (85 keV)	24 MW
ECRH (82, 110, 138 GHz)	7 MW

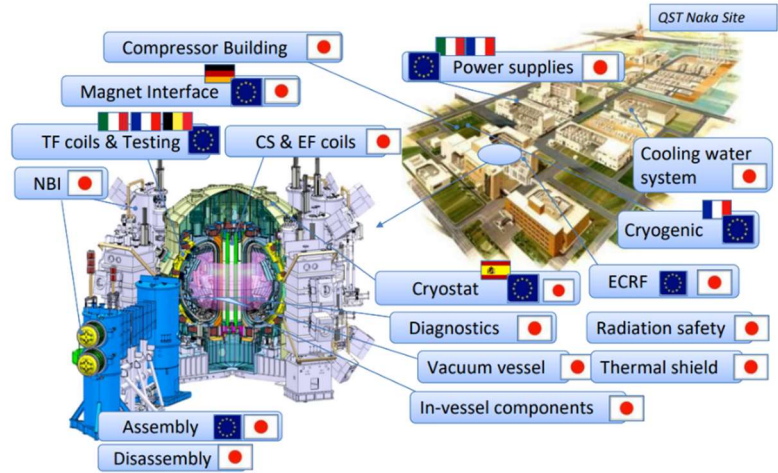


FIG. 1. Origin of the different components used in the construction and upgrade of JT-60SA

Major goals of the JT-60SA integrated commissioning were to perform integrated test of the coils and to develop interlock systems with the aim of ensuring a safe first plasma operation phase. In such a phase, the goals were to achieve a MA-class divertor plasma, demonstrate stable plasma control from current-up to ramp-down including Electron Cyclotron Resonance Heating (ECRH) assisted breakdown in low parallel electric field ($E_{||}$) conditions, qualify wall cleaning and conditioning techniques, to provide a first assessment of disruptions and run-away electron generation and to provide a first evaluation of the plasma performance. All these goals were successfully attained during OP-1. Importantly, while most of the OP-1 was carried out using Helium, Hydrogen was also used to test breakdown and plasma development. Similar results were obtained in H compared to He, which ensures the robustness of the results obtained.

A particular important aspect has been the significant predict first modelling activity that has accompanied the preparation of OP-1 in order to guide and optimize the work to be done during the plasma phase development. Such a work has provided valuable information about how to speed-up some of the key research topics in OP-1, such as the plasma breakdown, which was obtained after just two days of operation. Therefore, the JT-60SA start-up paves the way to future large superconducting tokamaks, notably those generating significant fusion power, since they will have to rely on validated modelling to assess the plasma development prior and during their operation, as expected in ITER.

2. RISK MITIGATION STRATEGY AND RESTART OF INTEGRATED COMMISSIONING

As reported in the previous IAEA-FEC, a series of measures were taken, before the integrated commissioning and OP-1 started, to reinforce the risk mitigation to avoid any incident with coils as the one reported in [2]. Such measures combined several safety approaches (i) reinforcement of coil insulation, (ii) avoiding unnecessary voltage application to the coil systems, and (iii) immediate de-energization of the coils when vacuum condition deteriorates. A key activity carried out before plasma operation was the performance of Local Paschen tests (LPT) to verify the right coil insulation where reinforcement was conducted. Furthermore, Global Paschen Test (GPT) were also done to confirm the insulation performance of repaired joints, feeders and other electric circuits inside the cryostat. GPT was carried out at different cryostat pressures. Nevertheless, since reaching a condition of 100% Paschen tight was unlikely, it was decided to proceed to the test phase once the inter-lock system that quick ensures de-energization of the coils when vacuum condition deteriorates was developed.

TABLE. 2. Summary of the initial research phase.

Research phase	Focus of exploitation	Operation Campaign		Divertor	Installed NB power	ECRF	Max. usable aux. power
Initial research phase I	Integrated Commissioning	Op-1	H	Open upper inertially cooled carbon	0	1.5 MW (2 Gyro.)	1.5MW
	Initial stable and reliable operation - H operation for commissioning towards D operation. - Stable operation at high current heated plasma	Op-2		Lower pumped carbon with intershot cooling	PNB 8 units, plus NNB Total 16MW (with H) 23.5 MW (with D)	3 MW (4 gyro)	19MW
	Initial research phase II	ITER and DEMO regime access - Access to ITER-relevant high confinement H-mode at high Ip - High beta access - ITER risk mitigation (ELM, disruption)	Op-3		D		PNB 12 units, plus NNB Total 30 MW
Op-4							

High Voltage tests of the superconducting coils were successfully performed in August 2023 and the energization of the Toroidal Field (TF) and poloidal field (PF) coils was carried out including the validation of quench detector system in case the cryostat vacuum was degraded. Such safety measures enabled the smooth progress of the integrated commissioning phase, the first plasma and the subsequent operation OP-1.

3. MAIN RESULTS FROM INTEGRATED COMMISSIONING AND OP.1

The first plasma was obtained in OP-1 at toroidal current $I_p=130$ kA and Toroidal magnetic field $B_t=2.0$ T while the whole OP-1 reached $I_p=1.2$ MA plasmas in diverted configuration [3]. The development of such initial plasmas has provided important information that is key for future devices such as ITER or DEMO in a number of important topics as describe in the following sections.

3.1. Plasma breakdown

In JT-60SA the available parallel electric field ($E_{||}$) can be as low as 0.15 V/m and therefore the plasma breakdown can be challenging if the usual Field Null Configuration (FNC) approach is used. This feature is common to those tokamaks using central superconducting solenoids such as in ITER and expected to be the case in DEMO. For instance, the inductive electric field in ITER is expected to be 0.3 V/m, which has been a matter for concern about its impact on the ITER start-up phase. While breakdown has been achieved in several tokamaks in FNC conditions at low $E_{||}$, the start-up phase of JT-60SA has provided a unique chance to extend previous studies to a large superconducting tokamak.

With the aim of analyzing this issue before plasma operation, extensive simulations with the TOSCA code were carried out. It was found that the Trapped Particle Configuration (TPC) technique assisted by ECRH could provide a better access to plasma breakdown than the more typical FNC. TPC breakdown was tested and successfully led to the first plasma on the 23rd October 2023 [5]. The first plasma was obtained at $I_p=130$ kA with a limiter configuration. Both fundamental resonance (82 GHz) and second harmonic resonance (110 GHz) ECRH were applied and positively tested during these discharges at a toroidal magnetic field of 2.0 T. The burn-through of the main gas (He) was clearly observed in both resonances. Importantly, the FNC start-up scenario, employing the same EC injection and prefill gas pressure, failed under similar conditions, see Fig.2, which shows that TPC is more efficient for plasma breakdown than FNC in conditions of low $E_{||}$. While most of the discharges during OP-1 were carried out in He, for minimizing wall recycling and speed-up the start-up, breakdown was also tested using H, achieving similar results.

A summary of the operational range for the toroidal electric field and the prefill pressure used during OP-1 in JT-60SA is shown in Fig.2. Clearly, breakdown was obtained in a wide range of prefill pressure at low toroidal

electric field when including either first or second ECRH harmonics or a combination of both. Comparing these results to the typical operation point of JT-60U and the one expected in ITER, JT-60SA has demonstrated breakdown at significant lower toroidal electric field. This achievement supports the expected development of the first plasma in ITER, as ITER is planned to initiate operation with half of its full toroidal magnetic field, utilizing second harmonic EC injection.

Throughout the first operation of JT-60SA, the effective use of ECRH proved critical to plasma start-up at low inductive electric fields, which was achieved in both FNC and TPC configurations. However, the EC-assisted FNC experiments revealed that this configuration was also highly sensitive to device model errors, which caused residual poloidal fields and outward plasma displacement. In contrast, the TPC scenario enabled robust plasma position control owing to the finite vertical magnetic field applied from the breakdown phase.

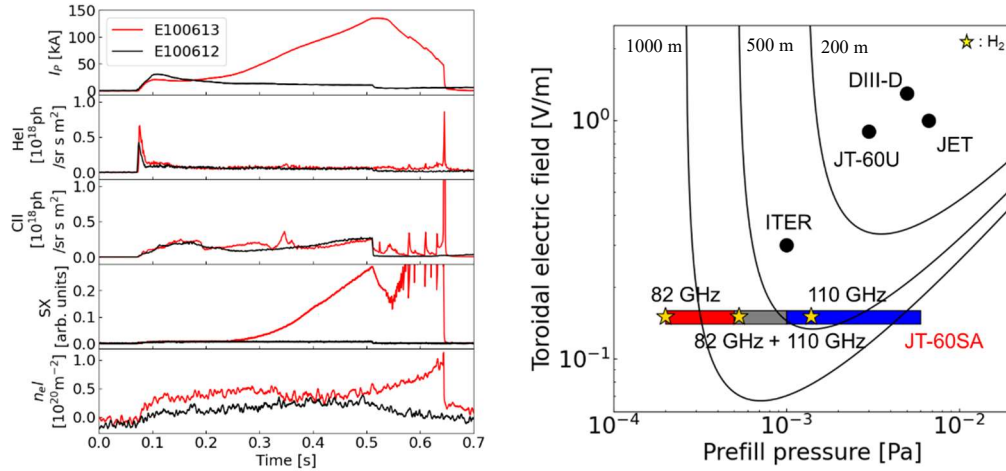


FIG. 2. Comparison of discharge waveforms of initial trials of the TPC scenario (E100613, red) and the FNC scenario (E100612, black) (left). Prefill pressure and toroidal electric field operational parameter space for breakdown in JT-60SA with 82 GHz ECRH, 110 GHz ECRH and simultaneous injection. Operational ranges were examined using He prefill gas whereas successful H₂ discharges were also demonstrated at operational points shown as stars. The minimum electric fields required for Townsend avalanche breakdown for H₂ with connection lengths of 200 m, 500 m and 1000 m, and the typical operational points for JT-60U, DIII-D, JET and ITER, are shown for comparison (right).

3.2. Plasma Shape reconstruction and control

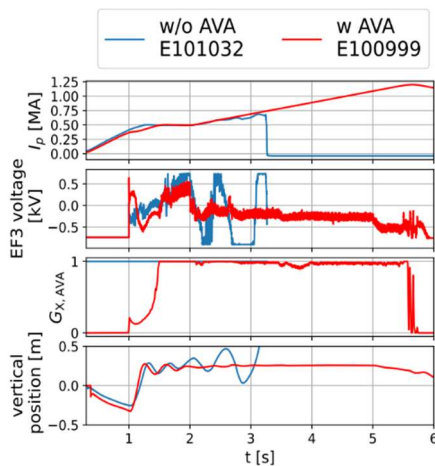


FIG. 3. Comparison between the discharge E100999, using AVA, and E101032 without AVA.

Once the breakdown is achieved, it is essential to ramp-up the plasma to higher current by keeping the plasma position and shape well controlled. This was done in JT-60SA by using several approaches for the plasma equilibrium reconstruction and the plasma shape control.

The position and Last Closed Flux Surface (LCFS) were reconstructed in real time by using a Cauchy Condition Surface (CCS) scheme, in which boundary integral equations are solved inside the LCFS using magnetic measurements [6]. Furthermore, a validation of the ITER real-time reconstruction algorithms for the plasma current, centroid position, boundary and for the poloidal beta has been performed using the data collected during OP-1. The accuracy of the reconstruction is evaluated against plasma equilibria computed with the CREATE-NL nonlinear code. Importantly, the ITER requirements for the estimation of the plasma current are met [7].

Predict-first modelling activity with the free boundary equilibrium non-linear code MECS indicated potential poloidal coils voltage saturation when both the plasma current and shape changed at the same time, notably during the I_p ramp-up phase. This issue is a universal challenge in next-generation large-scale tokamaks, such as ITER and DEMO, which will be characterized by increase in inductance due to coil scaling, voltage saturation, and interference among multiple control objectives (current, position, and shape of plasmas). The adaptive voltage allocation (AVA) scheme [8], which adaptively adjusts the balance between the

position and shape control and the I_p control under the saturated power supply voltage condition, was developed prior to OP-1 using MECS and tested extensively during OP-1. AVA demonstrated a good performance during OP-1 as shown in Fig. 3. Without the AVA scheme, interference between different plasma control schemes lead to voltage saturation in coils resulting in vertical position oscillations. By applying the AVA scheme, these oscillations were suppressed and plasma development up to $I_p=1.2\text{MA}$ was obtained [8]. These results highlight the importance of performing predict first simulations, in particular for plasma control, to ensure smooth and safe start-up phase of superconducting large tokamaks.

3.3. Vertical Displacement Events (VDEs) Control

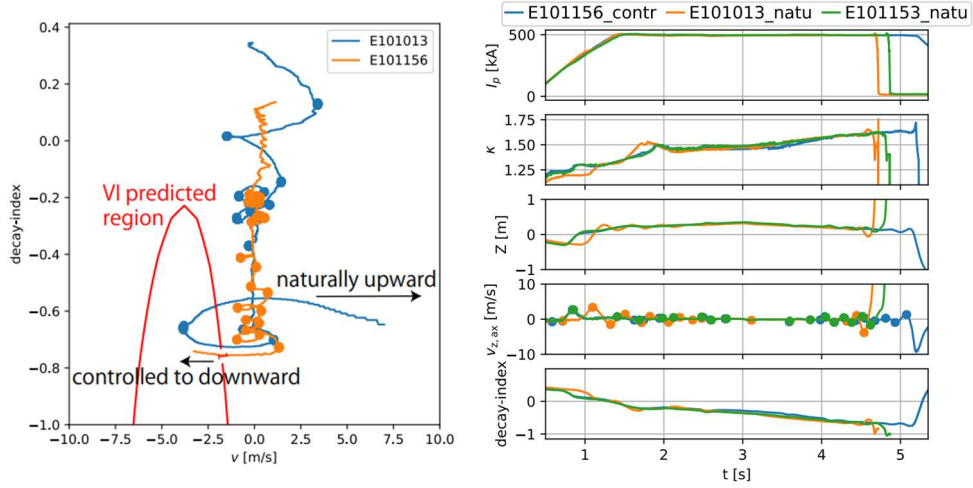


FIG. 4. Machine learning VDE direction control demonstration for the discharge E101156 (left). Comparison between discharges with no VDE control (natural) and with VDE control (controlled) (right).

VDEs are one of the major concerns in the plasma development in large superconducting tokamaks as they can result in plasma disruption and machine damage. Therefore, the development of VDEs control is mandatory to safely conduct the start-up phase. During OP-1 in JT-60SA, VDE predictors and control algorithms were developed using machine learning techniques with magnetics probe data [8]. This is critical and highly relevant development for ITER and DEMO since it is done in a phase with few diagnostics available and scarce input data.

The VDE predictor and a direction control algorithm was developed using only magnetics probe data and a machine learning method called a Support Vector Machine (SVM). The SVM was trained with the discharge E101013, for which the elongation was artificially increased leading to vertical plasma displacement and vertical plasma velocity oscillations and then an upward VDE occurred as shown in fig.4. This behaviour is well predicted by the SVM model as shown in fig.4, in which the unstable vertical displacement region is depicted in red. The discharge E101013 clearly crosses the unstable region resulting in a VDE. The control was active in the equivalent discharge E101156, resulting in a positive plasma direction control by forcing a downward vertical displacement. This development is important since it can redirect the plasma into well protected machine locations. During OP-1, predictions were made using two variables. However, in preparation for OP-2, the system has been expanded to include multidimensional predictions with ten variables and significant improvements in prediction performance have been confirmed [8].

3.4. Scenario development

Once breakdown and plasma control were demonstrated, scenario development was carried out in order to achieve ~ 1 MA diverted plasmas. Such a development was useful to evaluate several parameters, e.g. magnetic flux consumption or plasma performance, before moving to the next phase. The data obtained is important as it allows the validation of codes that are used to assess and predict the next plasma operations.

Example of scenario development at $I_p = 1.0$ MA with $B_T = 2.04$ T, $R = 3.08$ m, $a = 1.28$ m, plasma ellipticity $\kappa = 1.49$ and triangularity $\delta = 0.39$, is given in Fig. 5. The strategy for such development was to use ECRH, $\sim 0.5\text{-}1\text{MW}$, to assist the ramp-up in order to minimize the magnetic flux consumption and to extend the plasma duration as much as possible. At $I_p = 0.5\text{MA}$, the transition from limiter to upper single null diverted plasma was established. The maximum duration of the $I_p = 1$ MA flat-top with divertor configuration was about 3s as

shown in Fig. 5. Importantly, the density control was also demonstrated during this phase as also shown in Fig. 5. The locations of the divertor strike points were confirmed by using the high-resolution visible camera, EDICAM [9] as shown in Fig. 5 and confirmed with divertor Langmuir probes and plasma equilibrium reconstruction. Plasmas at $\sim 1\text{MA}$ were obtained using both H and He gas. The experience obtained during this scenario development allowed to attain several unique achievements in a superconducting tokamak, such as the largest plasma volume, $V=160\text{ m}^3$, and the highest toroidal current, $I_p=1.2\text{MA}$, which demonstrate that JT-60SA is ready to move to the following phase.

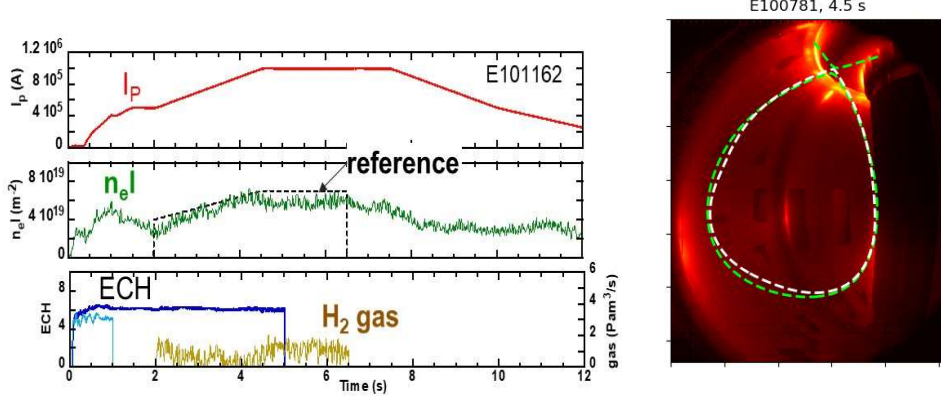


FIG. 5. Main characteristics of the discharge E101162, which reached 1 MA including AVA control and density control (left). Image from the EDICAM camera in the visible spectrum for the discharge E100781. Reconstruction of the plasma equilibrium shows good agreement on the strike point location (right).

3.5. Disruption and Run-away electron (RE) generation

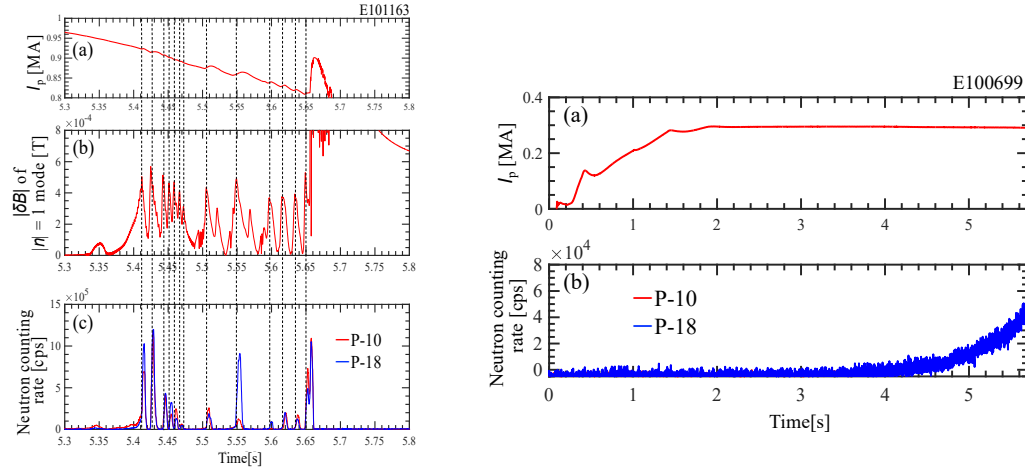


FIG. 6. Time evolution of (a) I_p (b) the magnetic fluctuation amplitude $|\delta B|$ of the toroidal mode number $|n|=1$ and (c) the neutron counting rates measured with NFMs in a typical discharge of the radiative disruption. (left). Time evolution of (a) I_p and (b) the neutron counting rates measured with NFMs at two positions. The continuous RE loss is detected before the plasma disruption (right).

A first analysis and classification of the causes for disruptions have been done after the results of OP-1 [10]. This is important because it is known that disruptions in the initial operation phase are due to the lack of maturity of the operation. Therefore, characterizing disruptions is important for speeding-up the learning curve of operations in future tokamaks such as ITER. In total 82 discharges disrupted out of 151 discharges in which the plasma shape and position were under control in the initial operation phase of JT-60SA. These 82 disruptions have been categorized into three classes according to their characteristics: VDEs, $n=1$ magnetic perturbation and radiative disruptions.

VDEs were responsible for the vast majority of disruptions in highly elongated plasmas [10], as the stabilization plate was not yet installed in this phase. Most of the disruptions with VDE origin were observed at $\kappa > 1.5$ with a clear impact of the gap between the plasma and the limiter as disruptions were likely to occur with lower κ when the gap became large, as reproduced with the MECS code. While MHD events have been observed, sawteeth and $n=1$ tearing modes, a direct link to disruptions has not been established. Disruptions linked to the n

= 1 perturbation mode were caused by the loss of controllability of the plasma shape and position when the mode was growing, rather than the impact of the mode itself. In such conditions, the plasma controller misinterpreted the plasma vertical position as unstable, even though the actual vertical displacement of the plasma was small. Finally, radiative disruptions occurred in conditions of high neutral gas injection and edge cooling and/or ECRH input power removal.

Studying the generation of REs is essential as it is one of the main concerns of future tokamak devices with operational point at high I_p , such as ITER. REs can cause damage of PFCs if they are not well controlled. Therefore, in JT-60SA, there has been a significant focus on analyzing the cause of RE generation in the start-up phase of the device, which is crucial to predict RE behaviour when JT-60SA will reach its full operational capabilities [11]. Due to the lack of gamma diagnostics in OP-1, neutron flux monitors (NFM) with ^{235}U fission chambers are used to indirectly detect the RE loss on PFCs via measurement of the photoneutrons generated by gamma rays impacting atoms in in-vessel components.

Detection of REs is clearly linked to VDEs leading to disruptions as bursty signals of neutron counting rates are observed when the vibrated plasma hit the PFCs. Furthermore, a clear correlation between REs and the onset of $n=1$ activity, caused by tearing modes, is detected. This is shown in Fig. 6, in which during the ramp-down phase of the discharge E101163, a clear correlation exists between the maxima of the fluctuating magnetic field caused by the $n=1$ mode and the neutron emission peaks. Nevertheless, RE emission is also detected in non-disruptive plasma phases as also shown in Fig. 6, in which the discharge E100699 shows a growing RE beam at constant $I_p=0.3\text{MA}$ before disruption occurs.

Finally, an alternative methodology to assess RE generation was carried out by using the visible camera EDICAM, which is sensitive to high energy photons, hence it can be utilized for Hard X-Ray (HXR) dosimetry [12]. In case of high-energy photon - solid-state detector pixel interaction, photoelectrons are generated with significantly higher rate than for visible range photons. This results in a “hot pixel” in the picture, having a significantly higher intensity than the neighbouring pixels. This feature enables the measurement of the high energy photon current quantitatively. This methodology has been applied to RE generation giving similar results to the one using neutron detector for the same discharges.

3.6. Confinement properties

Evaluating the thermal energy confinement properties of the initial JT-60SA plasmas and compare it to the expected confinement from well established scaling laws is essential to check whether the expectations are fulfilled. The energy confinement time obtained for JT-60SA L-mode diverted plasmas has been compared to the ITER89-P scaling with the aim of evaluating whether the expected energy confinement time in ITER is obtained in OP-1 [13]. The plasmas within the database analyzed are either ohmic or heated by ECRH using hydrogen plasmas. The global energy confinement time is evaluated from the energy balance equation ($\tau_E = W_{st}(\dot{P}_{ECH}^{abs} + P_{OH} - dW_{st}/dt)^{-1}$) and from a transient analysis method by which a decay time of W_{st} after the ECRH stops is evaluated.

The energy confinement time is evaluated from 100 ms to 200 ms. The results obtained highly depend on the confinement regime. A clear transition from LOC to SOC confinement modes is observed depending on the electron density, with the LOC regime obtained at low densities. In general, in the first operation phase of JT-60SA, τ_E decreases with increasing total heating power and improves with rising average density $\bar{n}_e$. The global energy confinement time follows the neo-Alcator scaling in the LOC regime, remaining below the ITER89-P scaling. At higher densities, the observed confinement characteristics suggest a transition toward the SOC regime, where τ_E is expected to align more closely with the ITER89-P scaling. These trends indicate that the confinement behaviours in JT-60SA plasmas is consistent with established scaling laws across different density regimes.

3.7. Wall conditioning

Electron Cyclotron Wall Conditioning (ECWC) is considered for the inter-shot cleaning of the first walls in JT-60SA. During OP-1, a systematic study of ECWC with He gas was performed by changing the poloidal magnetic field pattern in TPC configuration and the ECRH conditions. Favourable conditions of the poloidal field configuration and ECRH injections for H removal by ECWC were obtained [14]. In particular, the best results were obtained when He-ECWC plasma was produced on the inboard first wall where the start-up phase of the H tokamak plasma was produced. Nevertheless, the largest H removal rate was low, 1.0% after four tokamak discharges using H as a working gas. Further experiments will be carried out during OP-2.

4. STATUS OF JT-60SA, TIMELINE AND SCIENTIFIC PRIORITIES

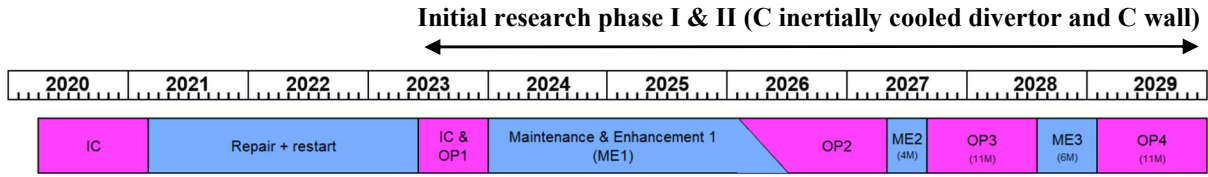


FIG. 7. Overview of the timeline for operation (OP) and Maintenance and Enhancement (ME).

JT-60SA project plan adopts “Phased Approach” with a gradual machine capability improvement, including diagnostics, in order to reach its highest performance and the main scientific goals, as shown in Table 2 for the initial research phase I & II. The NBI and ECRH power will be increased step by step, which will allow to access high performance at high density and I_p . In this initial phase, C Plasmas Facing Components (PFCs) are installed including an intertidally cooled divertor. The initial research phase I & II will serve as a step ladder preparation for the extended research phase, which will be characterized by W PFCs and actively cooled divertor in W, allowing the development of long pulse plasma operation at high β_N in a metallic environment as expected in DEMO.

The current timeline of the initial research phase is shown in Fig.7. By October 2025, JT-60SA is in the Maintenance and Enhancement phase 1 (ME1), while OP-2 will start in 2026. In the upcoming sections the main characteristics of each phase will be described.

4.1. Maintenance and Enhancement phase 1 (ME1)

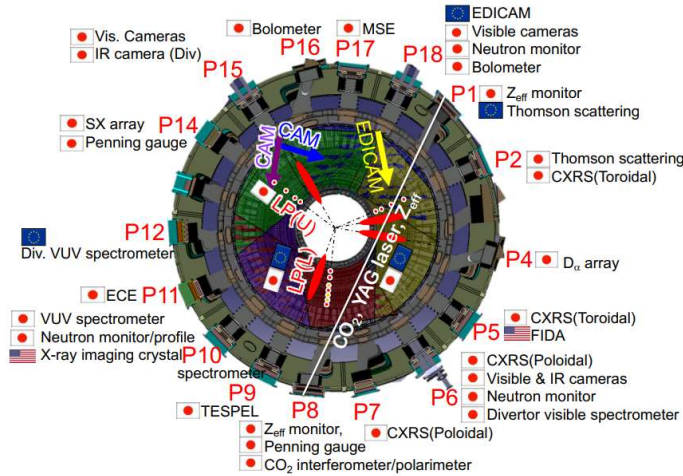


FIG. 8. Overview of the diagnostics available for OP-2.

The massive gas injection (MGI) system is also installed, which will be essential for disruption studies. Furthermore, additional CS reinforcements have been carried out during ME1 in 2025 to ensure that the CS design specifications were met, thus enabling the expected scientific programme to be developed in OP-2.

Most of the diagnostic systems, both from Japan and EU [16] as shown in Fig. 8, will be installed during ME1 for the use in OP-2: Thomson Scattering system, Vacuum Ultra Violet (VUV) divertor spectrometer, Infrared TV camera system, ECE diagnostics system, CXRS diagnostics system, MSE polarimeter system, Neutron profile monitor system, Tracer-Encapsulated Solid Pellet system. Specific agreements with General Atomics (GA) and Princeton Plasma Physics Laboratory (PPPL) were established in order to install the Fast Ion D-alpha (FIDA) diagnostics system, BY GA, and X-ray Imaging Crystal Spectroscopy (XICS) by PPPL.

4.2. OP-2 priorities

During OP-2, due to the installation of many new components and systems in ME1, there will be an integrated commissioning with and without plasma. At the beginning of OP-2, hydrogen operation is developed in order to avoid too much neutron generation and machine activation and hence most of the commissioning will be done in the hydrogen operation. Thereafter, deuterium operation starts as showing in Table 2.

After the successful OP-1, JT-60SA entered in the ME1 phase. JT-60SA is upgraded in order to allow high heating power operation with 23.5MW, including 10MW of Negative NBI at 500 keV, and 3 MW ECRH. Inertially cooled carbon divertor with divertor cryopump is installed to allow heat flux of 10MW/m² for 7.5s and 15MW/m² for 5s. The Stabilizing Plates (SP) with full carbon armours are set up in the torus direction, and behind the SP, in-vessel coils such as Fast Plasma Position Control Coil (FPPCC) [15], Error Field Correction Coils (EFCC) and Resistive Wall Mode Control Coils (RWMCC) are also installed.

The scientific priorities to be developed during OP-2 will be in line with the new machine capabilities after the installation of new components and diagnostics and with the integrated commissioning items to be performed in H. As an example, this is the case of the commissioning of the N-NBI, which will include the increase in energy beam and power. Such a commission will be used for studies of shine-through or fast ion studies by using the N-NBI, which is essential information for ITER. The use of H also allows for dedicated experiments profiting the change of isotope. This is the case of the L-H transition differences in H and D, which can be very well characterized at this stage. For that purpose, the use of ECRH, N-NBI and P-NBI will provide essential information, as it will help to study the impact of electron vs ion heating and the impact of rotation in the L-H transition.

An important activity will be devoted to the tests and development of plasma control schemes, breakdown, error field and plasma formation studies. This is essential for providing stable plasmas for performing reliable experiments. The control schemes will be tested at different I_p , which will be gradually increased with the aim of reaching its maximum capability, i.e. 5.5MA in L-mode. However, as typically happens in the initial phases of any tokamak device, disruptions are expected as the current is increased towards the goal of 5.5MA, therefore disruption studies including suppression of disruptions with MGI will be a priority. In this framework, the generation of runaway electrons is also expected and studies towards the understanding of their mitigation will be also important in ITER. A key element in which JT-60SA can provide key results from this initial phase is the SOL scaling at high I_p , which is a matter of concern for ITER. JT-60SA can provide unique data at >4 MA.

All the previous points are necessary steps to proceed towards the development of the demonstration of ITER-relevant H-mode plasmas which will be a priority in OP-2 as discussed in section 5.1.

4.3. OP-3 and OP-4 priorities

It is expected to achieve higher input power in OP-3 and OP-4 after all the commissioning and conditioning of NBI during OP-2 and the work to be done in ME2 and ME3. This will allow the consolidation of the entry to H-mode plasmas in the ITER-relevant scenario with type-I ELMs but also the development of plasmas at higher beta, hybrid and ITB scenarios, which is a goal towards the demonstration of DEMO relevant scenarios. Access to higher beta naturally means that the scope of the experiments to be performed can be much wider. For example, there will be a special focus on advanced real time control techniques, MHD studies, NTM control, interplay between fast ions and turbulence, type-I ELMs formation, plasma regimes with small or no ELMs, and transport and turbulence studies. Since the range of beta achievable by JT-60SA is high, above the no wall limit, such studies can indeed provide high impact information.

One key element to be considered in this phase is the compatibility of high beta plasmas at high power with heat flux limitations to the inertially cooled divertor. Specific techniques to make compatible high beta studies and divertor limitations will be tested, in particular, impurity seeding studies will be carried out as a preparation for the Integrated Research Phase.

4.4. W-PFC transition

At the end of the Initial Research Phase, it is expected a transition to the Integrated Research Phase with a change from C-PFCs to W-PFCs including an Actively Cooled Divertor (ACD). The installation of an ACD will allow to develop long pulse operation in order to study DEMO relevant plasmas in particular steady-state regimes at high β_N . Furthermore, the installation of W-PFCs also expands the scientific scope of JT-60SA towards W-related ITER relevant scientific topics after its rebaseline. Scientific topics such as the compatibility of steady-state plasma regimes at high β_N in a metallic environment, the source of W at high input power conditions, the possibility of W screening and its control at the plasma edge or the W migration and redeposition will be studied in a metallic environment and in long pulse operation. Such results will expand current studies about long pulse operation in W environment towards high input power and high current.

R&D of the high heat flux (HHF) components of the ACD in W is on-going. They will allow heat flux of 10MW/m² for 100s. In the present plan, the ACD is planned to be installed after OP-4, because much more time is needed to install the ACD comparing with the period of time for Maintenance & Enhancement: ME2 (4 months) and ME3 (6 months).

5. PREPARATION FOR THE SCIENTIFIC EXPLOITATION WITHIN THE JT-60SA EXPERIMENT TEAM

The JT-60SA scientific exploitation is managed under the so-called Experiment Team (ET) [2], which is composed of three Experiment Leaders (ELs), two from Japan and one from Europe, the Topical Group Leaders (TGLs) and the participating researchers. The ET has been focusing in the last two years on the preparation of the

initial JT-60SA campaigns with the aim of expanding the general research activities described in the JT-60SA research plan. A particular focus has been put on the use of predict first modelling activity to assess the research items expected to be developed in the initial research phase I & II, specially for OP-2. In particular, this activity has been carried out for scenario prediction for OP-2, which will start at limited input power of 19MW in H and 26.5MW in D, as it has significantly benefit from the model validation done with OP-1. Nevertheless, other important modelling predict first activities, including projections of W behaviour in JT-60SA, have been also carried out in view of a better definition of the experimental programme. Some of the activities are described in the following paragraphs.

5.1. Scenario preparation and prediction for OP-2

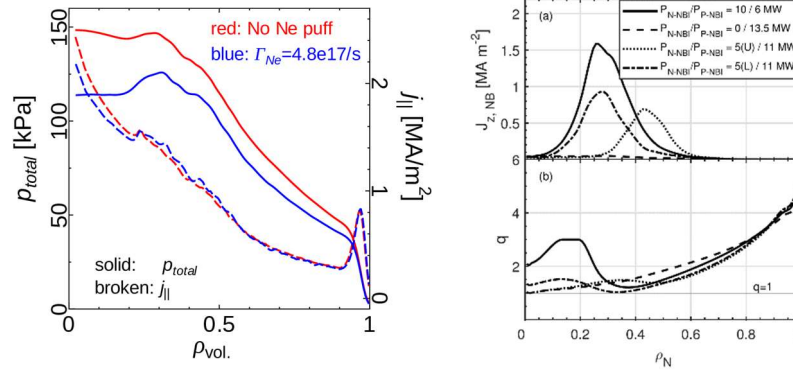


FIG. 9. Differences in plasma pressure with only C injection or by adding Ne injection (left). NBI current drive (a) and q profile (b) obtained at different combinations of N-NBI and P-NBI for the hybrid scenario (right).

An integrated simulation encompassing core-pedestal-SOL/divertor regions was performed to establish H-mode operation scenarios for the JT-60SA initial research phase [17]. The integrated modelling consists of a core transport simulation code GOTRESS+ with the mixed Bohm-gyroBohm model for turbulent transport, the EPED model for the pedestal characteristics, the Saarelma's density pedestal model, and a SOL/divertor simulation code SONIC. The scenarios investigated are the expected baseline scenario in OP-2 with $B_t = 2.28T$, $I_p = 4.6MA$, $\kappa = 1.85$, $\delta = 0.5$, the safety factor at the 95% poloidal flux surface $q_{95} \sim 3.0$, the total auxiliary power $P_{aux} = 19MW$ in D and the hybrid scenario at $I_p=2.7MA$ and $B_t = 1.7T$, with $P_{aux} = 16.5MW$, and $\kappa = 1.8$, $\delta = 0.51$. For the baseline scenario, the L-H power threshold, P_{LH} , predicted by ITPA08 scaling is 8.54MW with Greenwald Fraction, $f_{GW} = 0.5$, meaning that $P_{aux} = 19MW$ is more than twice of P_{LH} . Using this approach, simulations results show that it is possible to obtain the expected performance of baseline scenario, i.e. $\beta_N \sim 2$, $H_{98}(y,2) \sim 1$, in OP-2 conditions. Importantly, when using only D at a rate $\Gamma_{puff} = 1.0 \times 10^{21}/s$, the outer divertor is in an attached condition, resulting the peak heat load of $\sim 10MW/m^2$ with $n_{e,sep} = 2.9e19/m^3$ and $Z_{eff} \sim 1.4$. In order to reduce the heat load to the divertor, Ne is used as an extrinsic impurity, obtaining T_e less than 5eV near the strike point, meaning that a partially detached condition was achieved with the peak heat flux of $6.2MW/m^2$, $n_{e,sep} = 2.5e19/m^3$ and $Z_{eff} \sim 3.4$. As shown in Fig. 9, when using Ne, core pressure reduction is observed due to the increase in radiation power inside the separatrix as $\sim 2.8MW$. Nevertheless, in both with and without Ne puffing, the typical values of baseline scenario are obtained, confirming that the baseline scenario can be studied in OP-2. For the hybrid scenario, simulations show that stationary plasma conditions can be achieved without extrinsic impurity injection by optimizing the deuterium fueling rate while the predicted pedestal structure supports strong impurity screening.

The possibility of developing the hybrid scenario, $q_{min} \sim 1$, $\beta_N \sim 3$, $\beta_p > 1$, has been also studied with the JINTRAC suite of codes after first simulations carried out with the METIS code [18]. Two versions of the hybrid scenario, 3.7 MA/2.28 T and 2.7 MA/1.70 T, with $P_{aux} = 19$ MW, were considered with the aim of maintaining q with a low magnetic shear region, $q_{min} > 1$ and low shine-through losses. The results suggest that a high β_N (~ 3) regime with a high non-inductive current fraction ($\sim 70\%$) could be achieved during the initial research phase at 2.7 MA/1.70 T and at $f_{GW} = 0.4$. Importantly, it was shown that controlling the current drive from NBI, notably the current provided by the N-NBI, is essential to avoid strong q profile reversal as shown in Fig. 9.

Finally, a special effort is done to develop tools able to prepare plasma discharges before the operation in order to minimize risks and prepare ITER operation, in which such tools will be essential. Pulse Design Simulator (PDS) tools such as the one based on METIS+NICE [19] or MECS [20] have been validated with OP-1 experimental data and are being prepared for OP-2. PDS is capable of simulating a discharge from plasma breakdown to the very end of the discharge going through the start of ramp-up plasma, X-point formation, current flat top right to the ramp-down and include heating and transient phases with simplified models for the core response.

5.2. Fast ion, alpha particle and turbulence studies

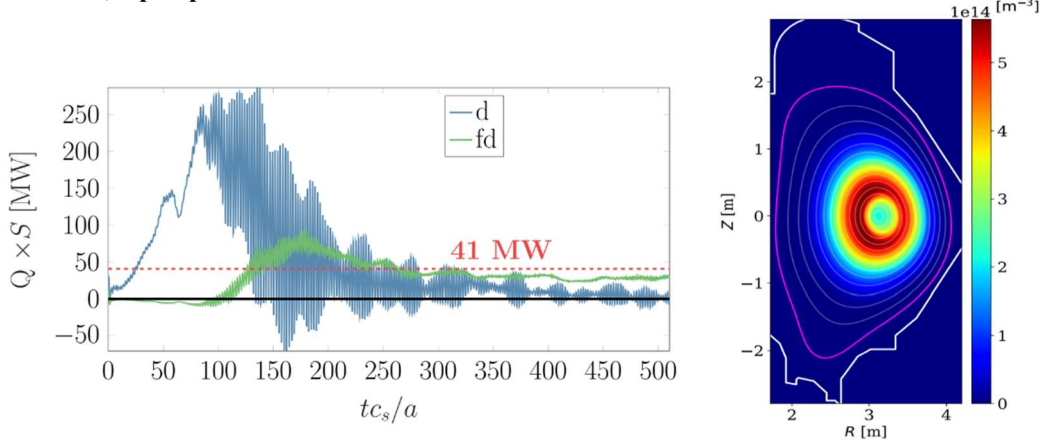


FIG. 10. D and fast D heat fluxes obtained from GENE simulations (left). Slowed-down alpha particle poloidal distribution as obtained from the ASCOT code (right).

JT-60SA has strong heating system flexibility. Notably its N-NBI will provide 10MW of heating power with energy beam up to 500 keV. This feature will allow detailed fast ion studies during the initial research phase, such as the destabilization of fast ion related instabilities, the analysis of the fast ion generation and losses or the interplay between fast ions and thermal turbulence [21]. Turbulence in the presence of fast ions has been studied in JT-60SA by performing GENE simulations of the scenario 3 from [22] at 41MW of auxiliary power. A clear interplay between turbulence and fast ion modes was found in electromagnetic simulations, which led to thermal fluxes oscillate with the high frequency of the fast ion driven modes. While thermal turbulent fluxes are generally low and the energy is mainly transported by the fast ion species, as shown in Fig. 10, they are quite sensitive to the specific thermal plasma background and the thermal fluxes can transit to the non-zonal transition (NZT), where zonal-flow-based saturation is disabled due to enhanced magnetic stochasticity at high $\beta_e \sim 2.7\%$ and/or profile gradients.

As of 2025, Deuterium-Tritium plasmas are not planned in JT-60SA. However, the fusion reaction $D_{\text{beam}} + {}^3\text{He} \rightarrow {}^4\text{He} (3.6\text{MeV}) + p (14.7\text{ MeV})$, which has its maximum cross-section close to 500keV for fast D, has been envisaged to study alpha particle production [23]. By using the Monte-Carlo code ASCOT, it is found that at 10% ${}^3\text{He}$ concentration and with 10MW of N-NBI power, $\sim 8 \times 10^{16} \text{s}^{-1}$ birth rate is obtained, four times higher than the JET equivalent results at 20-25% concentrations although at reduced NBI power (7MW). The slowed down alpha distribution is slightly off-axis, as shown in Fig. 10, due to the off-axis N-NBI deposition. The impact of MHD on alpha particle losses or the non-linear interplay between alpha particles and MHD modes in conditions of high q_{min} or reversed shear q profile driven by N-NBI can be investigated. This is an important information for steady-state plasma scenarios in D-T in ITER and DEMO, which might be characterized by low or even negative magnetic shear.

5.3. Plasma edge and divertor studies towards W operation

Long pulse operation at high beta, auxiliary power and plasma current will be the main characteristic of JT-60SA. Therefore, integrated power and particle exhaust scenarios represent a key element in operating the machine in safe conditions. This is particularly important when transiting to W-PFCs, in which potential W pollution of the plasma core by sputtering might limit the plasma performance. Several modelling studies have been carried out with the codes SOLPS-ITER and SOLEDGE in order to evaluate the controllability of the heat and particle flux to divertor in conditions of high auxiliary power. Simulations with SOLEDGE show that at an intermediate edge power of 15 MW, there is a need for impurity injection during the first period of exploitation of the machine as 10 MW/m^2 is obtained in attached cases, and achieving detachment is challenging, requiring upstream electron densities at least above $4 \times 10^{19} \text{m}^{-3}$. By using Ne seeding, it is shown that at 15 MW of power injected into the edge plasma, the inner target is easily detached and presents low heat loads. Therefore, JT-60SA will probably need to be operated in a deep detached regime in its first phase of exploitation for discharges longer than a few seconds. These results are similar to those obtained using SOLPS-ITER which in addition shows that the V-shaped outer divertor corner, designed to favour the detachment at low midplane plasma density, is particularly effective in trapping neutral particles in the corner, thus enhancing the momentum loss in the strike point region. The influence of these results on plasma operation when W divertor will be installed is being modelled in parallel to studies about potential optimized W divertor geometry [24].

6. CONCLUSIONS

After the construction phase, JT-60SA has entered into the initial research phase I by performing an integrated commissioning in 2023, which included significant amount of plasma operation and scientific development during OP-1. Several key results relevant for ITER and DEMO have been obtained in this initial phase. Breakdown was obtained in low $E_{||}$ conditions as expected in ITER by using the novel TPC technique, which was extensively prepared before operation. Plasma position and shape control has been obtained by using the AVA scheme, which allows simultaneous control of plasma shape and I_p in conditions of saturated power supplies. Such achievements allowed JT-60SA to enter into the MA-class tokamaks with a maximum I_p current of 1.2MA in diverted plasmas, hence becoming the largest tokamak in operation.

JT-60SA is currently in a phase of maintenance and enhancement in which important upgrades, such as the lower inertially cooled divertor in C, the NBI heating, in vessel coils and a large number of diagnostics are being installed to resume operation in 2026. Three campaigns will be developed using C-PFCs, in which essential scientific topics, such as L-H transition, scenario development at high current and density and performance, high β_N scenarios, plasma wall interaction and ELM control or disruption mitigation and prediction, will be explored. After these campaigns, a transition to W-PFCs, with an actively cooled divertor, is expected. The compatibility of long pulse steady-state plasma operation in high beta conditions, as expected in DEMO, will be a priority.

In the speedy development of plasmas during OP-1, the role of predict first modelling activities has been essential. The same type of activity has been followed to assess priority scientific items for OP-2. The results show that the development of plasma scenarios in H-mode at high performance and I_p is possible in D, notably in conditions of high electron heating and low rotation, as expected in ITER.

Finally, the start-up of JT-60SA marks a landmark as it shows the importance of new techniques, such as predict first modelling or machine learning in conditions with scarce input data, to guide the operation and scientific development of a new large superconducting tokamak. The lessons learned are essential for the new generation of researchers that will certainly be fostered in the scientific exploitation of a large superconducting machine within the JT-60SA project before ITER and DEMO operation.

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