JT-60SA Magnet System Status

S. Davis, W. Abdel Maksoud, P. Barabaschi, A. Cucchiaro, P. Decool, E. Di Pietro, G. Disset, N. Hajnal, K. Kizu, C. Mayri, K. Masaki, J.L. Marechal, H. Murakami, G.M. Polli, P. Rossi, V. Tomarchio, K. Tsuchiya, D. Tsuru, M. Verrecchia, M. Wanner

Abstract **The JT-60SA experimental device will be the world's largest superconducting tokamak when it is assembled in 2019 in Naka, Japan (R=3m, a=1.2m). It is being constructed jointly by institutions in the EU and Japan under the Broader Approach agreement.**

Manufacturing of the six NbTi equilibrium field coils, which have a diameter of up to 12 m, has been completed. So far 13 of the 18 NbTi toroidal field coils, each 7 m high and 4.5 m wide, have also been manufactured and tested at 4 K in a dedicated test facility in France. The first three of four Nb_3Sn central solenoid modules have been completed, as have all of the copper in-vessel error field correction coils.

Installation of the toroidal field magnet, around the previously welded 340° tokamak vacuum vessel and its thermal shield, started at the end of 2016 and is currently underway. The TF magnet will in turn support the EF and CS coils.

Index Terms—**JT-60SA, tokamak, magnet, fusion**

I. Introduction

The JT-60SA tokamak aims to contribute to the timely realization of fusion power by complementing the burning plasma research of ITER [1]-[3]. Its design fulfils its scientific goals while being strongly optimized to reduce its cost. In particular the device aims to develop "advanced" plasma scenarios relevant to achieving fusion conditions in a power plant, i.e. operational modes in which

a) plasma current is driven by means other than induction
b) the ratio between plasma pressure and the confining toroidal field is high
c) plasma conditions and plasma-wall interactions are achieved for durations relevant to steady-state operation.

In a classical tokamak current is induced in the plasma by the transformer effect, with the plasma forming a single secondary turn around the machine axis and a solenoid magnet making many primary turns. While the current in the solenoid is ramped from one extreme to the other current can be driven in the plasma. However this technique limits the plasma duration to one reversal of the solenoid current.

For steady-state operation, desirable for power generation, other non-inductive means to drive plasma current are being developed. These typically exploit other plasma heating technologies such as the injection of high energy particles or resonant heating of the ions or the electrons by electromagnetic waves.

Also to this end when completed JT-60SA will be equipped with 34 MW of neutral beams (the particles must be neutralized once accelerated in order to penetrate the confining magnetic fields) and 7 MW of electron cyclotron resonance heating (gyrotron-generated microwaves broadcast into the plasma). Localized and directional application of these techniques allows not only current drive but also control of the plasma current / temperature / density profile. This can improve stability, avoiding damage from disruptions, and allow improved core conditions relative to the machine size (i.e. steeper gradients).

The generation of one or more "pedestals" in the temperature and pressure profiles is dependent on high auxiliary heating and is essential to achieve fusion-relevant conditions in an affordably-sized device. This allows access to the "H-mode" regime, with improved temperature confinement, and beyond.

Further stability control is provided by stabilizing plates and by fast-acting actively controlled coils, both installed inside the primary vacuum vessel adjacent to the plasma. A pair of copper poloidal coils maintain the vertical position of the plasma while a set of 18 copper saddle coils distributed around the torus react at approximately 3 kHz to limit the development of magnetohydrodynamic instabilities, primarily resistive wall modes.

II. Magnet System Design and Manufacturing

All the toroidal field (TF), equilibrium field (EF) and central solenoid (CS) coils for JT-60SA are superconducting [4]-[7]. This is necessary not only to achieve the fields required in a such a compact device but also to allow longer plasma durations and to avoid resistive power losses, both critical for fusion power generation.

A. Toroidal field coils

S. Davis (email: sam.davis@jt60sa.org), P. Barabaschi, E. Di Pietro, N. Hajnal, V. Tomarchio, M. Verrecchia and M. Wanner are with Fusion for Energy, Broader Approach Programme & Delivery Department, Boltzmannstr 2, 85748 Garching, Germany.

W. Abdel Maksoud, G. Disset and C. Mayri are with CEA-IRFU, 91191 Gif sur Yvette Cedex, France.

P. Decool and J.L. Marechal are with CEA-IRFM, 13108 St Paul lez Durance Cedex, France.

A. Cucchiaro, G.M. Polli and P. Rossi are with ENEA FSN, Via E. Fermi 45, Frascati, Italy.

K. Kizu, K. Masaki, H. Murakami, K. Tsuchiya and D. Tsuru are with QST, 801-1 Mukoyama, Naka-shi, Ibaraki-ken 311-0193, Japan.

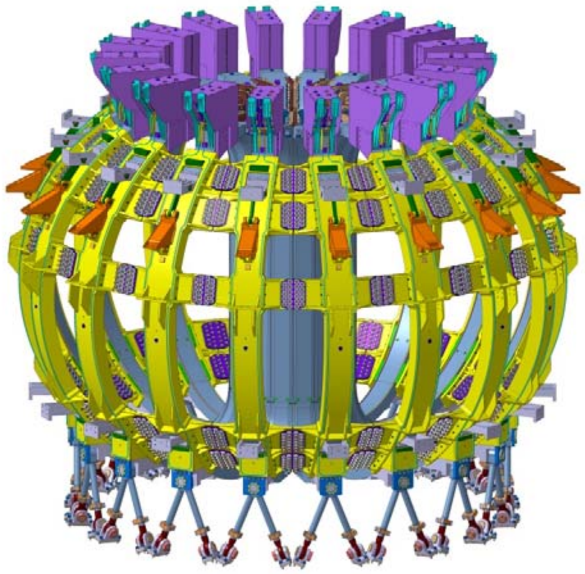

Fig. 2: Toroidal field coil assembly

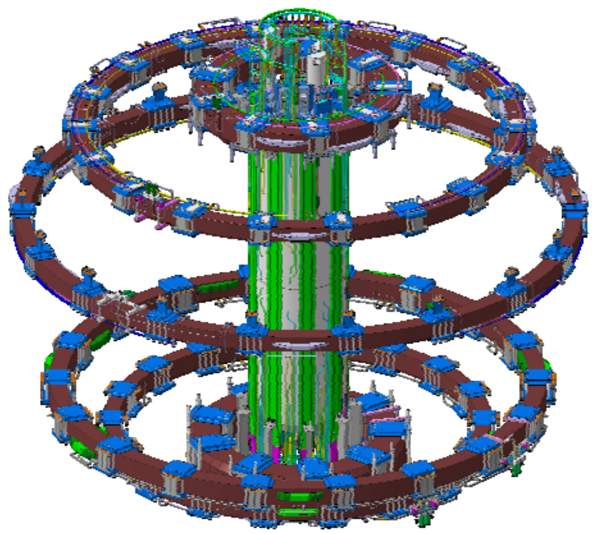

Fig. 1: Equilibrium field coils and central solenoid

The toroidal field coil assembly (Fig. 2) forms the structural backbone of the magnet system [8]. Only net forces acting on the magnets, e.g. the weight of the assembly and loads from earthquakes or plasma disruptions, must be supported externally. This function is achieved by the "gravity supports" at the bottom of each TF coil which pivot on spherical bearings to allow radial movement when the magnets are cooled down to cryogenic temperatures.

There are 18 D-shaped toroidal field coils [9] surrounding the torus-shaped vacuum vessel in which the plasma is generated. At the inboard side their straight legs wedge against each other while at the outboard side their curved legs are free to move radially. They are supported against the toroidal forces (normal to the plane of the each coil) generated by the interaction of their own field and the fields from the plasma and poloidal field coils by the Outer Intercoil Structure (OIS) and the Inner Intercoil Structures (IIS). The OIS allows each coil to be bolted to its neighbors through shear panels while the IIS includes force fit shear pins between neighboring coils at their top and bottom.

Each coil has 72 turns wound from NbTi cable-in-conduit conductor and arranged in 6 double pancakes, forming a compact winding pack with a rectangular cross section. A steady 25.7 kA current produces a nominal 2.25 T field on the tokamak axis (max 5.65 T on the conductor). The toroidal field magnet has a stored energy of 1.06 GJ.

The toroidal field coils are manufactured in Europe [10]-[16] and shipped after testing to Japan. All of the winding packs needed have been wound, including the electrical joints between double pancakes, application of ground insulation and vacuum impregnation. The last coils are now being integrated with their stainless steel casings. After the casing has been welded closed a final epoxy resin impregnation is performed to embed the winding packs inside. Then the mechanical interfaces on the coil casing are machined with respect to the average winding pack centreline and the helium piping, temperature sensors and quench detection cables are fitted.

A series of slight shape changes were observed during the manufacturing of the TF coils. After being released from their tooling after winding and vacuum impregnation the coils were typically slightly 'bean-shaped' rather than D-shaped, with the middle of both the straight leg and the curved leg bowing inwards by a few millimeters. The welding of the casings caused a deformation in the opposite direction, with the coils legs bowing outwards by a few millimeters with respect to nominal to make the coils more round. Hence it was possible to improve the final shape of the winding packs within the casings by wedging them carefully before welding to anticipate this effect.

B. *Equilibrium field coils*

There are 6 equilibrium field coils and 4 independently-controlled solenoid modules (Fig. 2). These coils allow wide ranging and flexible control of the plasma shape. Plasma aspect ratios (major radius / minor radius) down to 2.5 will be possible, allowing efficient exploitation of the available toroidal field. Furthermore both double and single null configurations can be created, i.e. the poloidal field can be configured to

Fig. 3: EF coils 1-3

form a “divertor” or plasma exhaust at both the top and bottom of the torus. This allows the removal of impurities, maintaining core conditions, while managing the high heat flux from the exhaust particles.

The EF coils vary in diameter from 4.4 m up to 12 m. Two different NbTi cable-in-conduit conductors are used with a central spiral coolant channel, one optimized for a peak field of 4.8 T for EF coils 1, 2, 5 & 6 and one optimized for a peak field of 6.2 T for the inboard EF coils 3 and 4. Both carry up to 20 kA. Two pick-up coils are co-wound in each coil for quench protection.

Manufacturing of the EF coils [17]-[20] was completed in 2016. Since their large size (up to 12 m diameter for EF1) prohibits their transportation most were manufactured on site in Naka using modularized tooling. The coils are wound from single or double pancakes, depending on their size and hence how many turns can be made from a single conductor length. The pancakes are press-cured clamped in a mould before they are stacked to form the winding pack. Then the ground insulation is wrapped and cured before clamps are fitted around the winding packs to keep them under compression during operation and provide support points.

Despite their large size excellent control of deviations from circularity was achieved during the winding and stacking of

TABLE 1
EF COIL CIRCULARITY

Coil	Diameter	Deviation from circularity	Requirement
EF1	12.0 m	0.3 mm	≤8 mm
EF2	9.6 m	0.4 mm	≤7 mm
EF3	4.4 m	0.2 mm	≤6 mm
EF4	4.4 m	0.6 mm	≤6 mm
EF5	8.1 m	0.6 mm	≤7 mm
EF6	10.5 m	1.3 mm	≤8 mm

the EF coils, as shown in Table 1. Continuous improvement allowed lower values to be achieved for EF 1-3 (Fig. 3) than for EF 4-6.

C. *Central solenoid*

The central solenoid [20]-[24] is made of 4 modules, each with an outer diameter of 1.65 m and a height of 1.60 m. Each module is made from 6 octa-pancakes and 1 quad-pancake forming 549 turns. The conductor can carry 20 kA, giving a total of almost 11 MA-turns for each module.

The peak nominal field of 8.9 T necessitates a Nb_3Sn c cable-in-conduit conductor and hence the pancakes must be wound before the 650°C heat treatment needed to produce the superconducting strand is performed on the octa / quad pancakes. After heat treatment the turns must be separated carefully, without applying excessive strain, to allow the glass-kapton turn insulation to be wrapped around the conductor. Finally the pancakes are impregnated together to form each of the four modules.

As of August 2017 three of the four solenoid modules have been completed (the first may be seen in Fig. 4) and the insulation of the final module is beginning. Once completed the four modules must be stacked and compressed with tie rods. The finished solenoid will weigh almost 100 tonnes.

Within the available flux swing of 17 Vs JT-60SA aims to drive a plasma current of up to 5.5 MA inductively for 100 s. Non-inductively a current of 2.3 MA is targeted.

Fig. 4: Central solenoid module 1

III. ELECTRICAL TESTING

The performance of the superconducting strand and conductors used in the manufacturing of the coils was demonstrated previously during both qualification tests and routine quality assurance [25]-[30].

A. *Toroidal field coils*

All of the TF coils for JT-60SA are tested under Paschen conditions by their manufacturers, i.e. their ground insulation is tested at 3.8 kV while they are surrounded by gas at a range of low pressures at which a Paschen breakdown could occur. Typically faults observed have been limited to lengths of the helium piping or joints which were insulated by hand. These were corrected and re-tested.

All of the TF coils are also tested under cryogenic conditions at their nominal current and all of them will be quenched in order to demonstrate they have sufficient operational margin with respect to the magnetic field and temperatures present. As of August 2017 13 of the 18 coils had successfully completed these tests at a dedicated facility established at CEA Saclay.

Each quench occurred with a helium inlet temperature of between 7.44 and 7.51K. The quench was initiated in a central pancake, where the field is highest, for only one third of the coils. For half of the coils the quench was initiated in a side pancake.

The casings of the TF coils are not strongly cooled, given that the priority is to cool the winding packs and that there is very limited space for piping, especially on the straight legs. During testing the casings were observed to remain between 5 and 15 K warmer than the winding packs, depending on the location. This is expected to be lower in the tokamak when the coil casings are surrounded by other structures cooled to 4 K rather than the 70 K thermal shield of the test cryostat.

The hydraulic performance of each TF conductor was measured before it was wound and those conductors having below average performance were assigned to the intermediate double pancakes 2 and 5, i.e. away from the highest field and highest temperatures. The pressure drop of each coil is confirmed during the cryogenic tests.

All internal joint resistances were measured, with the maximum resistance measured for an individual joint being 1.57 nΩ. The joints are equipped with temporary voltage taps which are removed before installation in the tokamak.

B. *Equilibrium field coils*

Due to their very large size the production EF coils are not tested under Paschen or cryogenic conditions. The integrity of their ground insulation is demonstrated by spraying the coils with water and covering them with aluminum foil before the 21 kV test voltage is applied.

C. *Central solenoid*

For the central solenoid, a test coil [31] comprising one quad pancake and one double pancake was wound and tested at 4 K. This demonstrated the hydraulic performance of the conductor and the resistance of the pancake joints and the terminal joints used.

Limited cryogenic tests were also performed on the first completed CS module, which indicated that the 0.75 K heating expected from AC losses occurring during a fast discharge should not exceed the 1 K temperature margin. These tests also indicated that the temperature uniformity during cooldown and the cooldown rate were expected to be sufficient.

IV. Assembly status

The assembly of the toroidal field coils is currently underway and as of August 2017 eleven coils have been accurately positioned in the torus hall. EF coils 4-6 were positioned beforehand on the cryostat base in 2013 before the assembly of the plasma vacuum vessel; EF coils 1-3 and the CS will be installed once the TF magnet assembly is complete.

A. *TF coil pre-assembly*

After its cryogenic testing and before its shipment to Japan each TF coil is pre-assembled together with a sector of the Outer Intercoil Structure. For this operation the coil is oriented ‘on edge’, i.e. with its straight leg along the ground and its curved leg up in the air. The U-sectioned OIS is lowered over the coil from above (Fig. 5) and aligned with the symmetry plane of the winding pack. Openings in the coil casing allow tracking points on the winding pack to be measured directly by laser tracker. Steel pads between the OIS and the coil are adjusted to give a 0.5 mm gap. This allows the coil to move radially during cooldown and when energized but to be supported against out-of-plane loads. After the OIS has been fitted each coil is returned to a horizontal orientation, or rather 10° from horizontal given the wedged shape of the OIS, for shipping to Japan.

Fig. 5: TF coil pre-assembly with Outer Intercoil Structure

B. *TF coil positioning*

On arrival on site at QST Naka a number of tests are performed on the coils to ensure they are free from transportation damage, e.g. pressure and leak tests, ground insulation test etc. The coils must be assembled around the vacuum vessel and its thermal shield already welded together. These form an incomplete torus leaving a 20 degree slice open to allow for the insertion of the TF coils. The tokamak assembly is surrounded by a temporary assembly frame capped by a rotary crane that allows each coil, once inserted, to be rotated into its final position.

First each coil is stood upright using a dedicated frame. Then it is lifted to the center of the tokamak using the building crane and a lifting beam to ensure the coil stays level during the lift. The coil is then lowered between the two parallel beams of the rotary crane into the 20° opening in the vacuum vessel. Here it is supported temporarily while it is transferred from the building crane to the rotary crane. While being brought to its target position the coil is shifted slightly outwards radially to increase the clearance between the straight leg and the vacuum vessel thermal shield. The temporary sup-

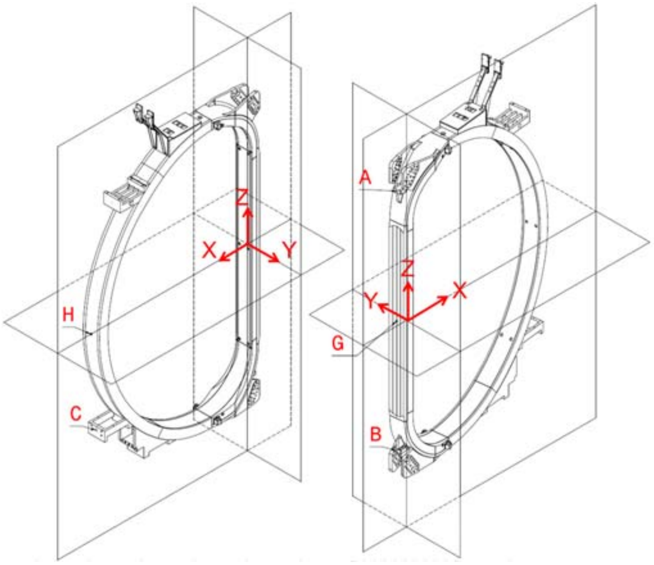

Fig. 6: Main TF coil casing points tracked during assembly

TABLE 2
TF COIL POSITIONS ACHIEVED – VARIATIONS FROM NOMINAL LOCAL CO-ORDINATES (IN MM)

Casing tracking point		B	A	G	H	C
Coil position 3 (11th installed, no. 6)	x	**0.1**	**-0.2**	-1	2.3	1.1
	y	**0.1**	**0.4**	**0**	**0.5**	**0.6**
	z	0.8	0.9	**0.5**	1.4	1.6
Coil position 4 (9th installed, no. 5)	x	**0.1**	**-0.1**	-1.0	2.0	0.7
	y	**-0.4**	**0.0**	**-0.3**	**0.7**	**0.5**
	z	-0.1	0.1	**0.1**	0.6	0.9
Coil position 5 (7th installed, no. 14)	x	0.4	-0.6	-1.3	1.3	0.3
	y	**-0.6**	**-0.5**	**-0.1**	**-0.3**	**-0.3**
	z	0.3	0.2	**0.1**	1.5	1.6
Coil position 6 (4th installed, no. 3)	x	**0.8**	**0.7**	-0.3	2.4	1.2
	y	**-0.2**	**0.8**	**0.3**	**0.1**	**0.0**
	z	0.4	0.7	**0.6**	1.5	1.8
Coil position 7 (3rd installed, no. 1)	x	**0.0**	**0.3**	-0.4	1.8	0.3
	y	**-0.4**	**0.4**	**-0.1**	**-0.4**	**0.3**
	z	0.1	0.5	**0.3**	0.5	0.9
Coil position 8 (1st installed, no. 10)	x	**0.5**	**0.5**	-0.9	2.8	1.1
	y	**0.0**	**0.6**	**0.0**	**-0.3**	**-0.4**
	z	-0.4	0.5	**0.1**	0.9	1.3
Coil position 9 (2nd installed, no. 11)	x	**0.2**	**0.4**	-1.0	2.0	0.6
	y	**0.1**	**0.2**	**0.0**	**0.2**	**-0.2**
	z	0.2	0.2	**0.1**	0.5	0.6
Coil position 10 (5th installed, no. 12)	x	**0.3**	**1.1**	-0.8	2.8	1.1
	y	**0.0**	**0.6**	**-0.1**	**-0.4**	**-0.2**
	z	0.2	-0.4	**0.5**	0.8	1.0
Coil position 11 (6th installed, no. 13)	x	**0.3**	**0.2**	-1.2	2.1	0.8
	y	**0.3**	**0.3**	**0.1**	**1.0**	**0.2**
	z	1.0	0.4	**0.5**	1.3	1.7
Coil position 12 (8th installed, no. 4)	x	**0.3**	**0.6**	-0.2	2.1	0.8
	y	**0.2**	**-0.1**	**0.2**	**-0.6**	**0.8**
	z	0.5	0.7	**0.6**	1.4	1.3
Coil position 13 (10th installed, no. 15)	x	**-0.1**	**0.1**	-1.3	2.8	1.4
	y	**0.6**	**0.4**	**-0.5**	**0.2**	**-0.1**
	z	-0.9	0.5	**0.8**	1.0	1.1

ports of the vacuum vessel must be dismounted and reinstalled one by one as the coil is rotated around it. Laser levels are used to ensure that the vertical positon of the vacuum vessel is maintained.

The positioning of each coil is checked using tracking points machined on the outside of its casing at known positions with respect to the winding pack inside. The primary points tracked (Fig. 6) are A and B at the top and bottom of the straight leg, ensuring its radial position (*x* direction) and verticality (*z*), the middle of the straight leg G, to ensure the coils correct vertical position, and the middle of the curved leg H, ensuring its correct toroidal positioning (*y*).

Three supports are used to position each coil. At the bottom of each straight leg an accurately positioned jig is used to locate the crown plates of the coil, setting both its vertical position and radial position. At the bottom of the curved leg, at the position from which the EF5 coil will eventually be supported, double-acting hydraulic jacks allow the outboard of the coil to be raised or lowered. A little higher up the curved leg near point C screws may be used to adjust the position of the EF6 pedestal in the horizontal plane, i.e. toroidally (*y*) or radially (*x*).

The coil is adjusted such that each tracking point is within 1 mm of its target position as shown in Table 2. Local co-ordinate systems centered on the tokamak vertical axis but aligned with the symmetry plane of each coil allow an easier assessment of the live coil position. The slender coils deform appreciably under their own weight – this may be observed in particular in the radial (x) movement of points G and C at the middle of the straight and the curved legs which bow inwards and outwards respectively.

C. *Straight leg insulation*

To prevent toroidal eddy current circulation, each coil must be insulated from its neighbors and therefore before installation each straight leg is fitted with 2 mm GRP plates on one side. The other side is fitted with stainless steel shims chosen with the aim of ensuring contact where the coils are bolted at their crown plates at the top and bottom of the straight leg and of ensuring a gap of below 0.5 mm along the straight leg.

D. *Inner Intercoil Structures*

The Inner Intercoil Structures (Fig. 7) are a set of bolts and pins fitted between neighboring coils at the top and bottom of

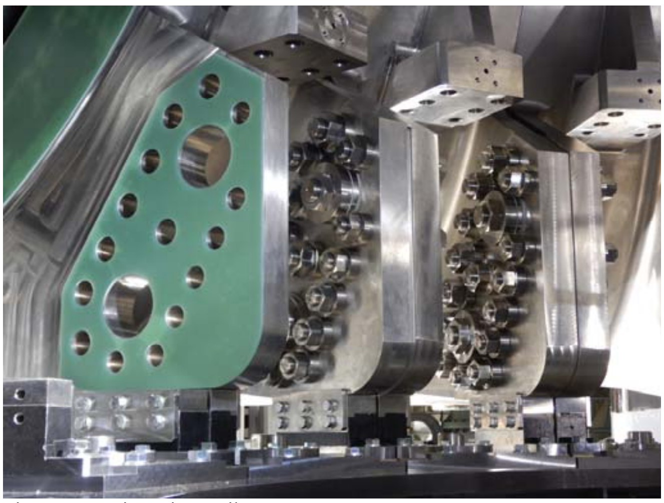

Fig. 7: Lower inner intercoil structures

each straight leg which

a) hold the coils together under the toroidal field , which generally pinches the straight legs of the coils together but also expands them under hoop tension. This acts to separate adjacent crown plates, and
b) withstand the torque arising from the crossing of the TF with the poloidal field from the plasma and EF coils. This causes the crown plates to try and shear against each other.

The M30 and M36 studs and nuts are made in Inconel 718 to withstand the high tensile and bending loads applied during tokamak operation. 20 mm thick titanium washers are used to help reduce the loss of preload that occurs due to the reduced thermal contraction of Inconel compared with the stainless steel crown plates, and to allow larger bolt holes to accommodate potential misalignment between coils. The threads are lubricated with Loctite LB-8012 molybdenum paste and the nuts are tightened using hydraulic wrenches. The extension of the bolts is monitored using an ultrasonic extensometer to ensure the correct pre-load is achieved.

Each pin assembly is made up of a conical Inconel pin and two split bushings. The bushings are made eccentric, i.e. with the axis of their inner conical surface offset from the axis of their outer cylindrical surface. By rotating the bushings in the crown plates, the axes of the conical surfaces may be brought into alignment even if the coils themselves are not perfectly aligned. Allowance is made for the axis of each crown plate pin hole to vary from its nominal position by up to 3 mm in any direction after manufacturing and installation, although the actual misalignments observed are much lower. The pins, which have a plasma-sprayed alumina coating to maintain the electrical separation of neighboring coils, are wedged into place using a hydraulic tensioner.

E. *Gravity supports*

Each of the 18 TF coil gravity supports is pre-positioned on the cryostat base just prior to its respective coil. The main pin between the coil casing and the upper spherical joint of the gravity support is fitted after the position of the coil is finalized. Finally shims are inserted under the two feet of the gravity support and they are bolted to their support pedestals on the cryostat base. However the weight of the magnet will not be transferred to these supports until the complete assembly is finished.

F. *Splice plates*

On each side of each sector of the OIS there are 5 shear panels which must be bolted through pairs of splice plates to those of the neighboring OIS sector. The complete Outer Intercoil Structure is hence able to support the TF coils against toroidal loads. The space between shear panels was maximized to allow large vacuum vessel ports and hence easy access for plasma diagnostics and heating systems.

Each splice plate must be customized individually in order to accommodate the variations in shear panel position arising from their manufacturing, pre-assembly with the TF coils and final positioning in the torus. For each joint between adjacent coils the installed position of each shear panel is measured at three bolt hole centers using a laser tracker. These data are used to design and then machine each splice plate accordingly. Both the shear panels and, after machining, the splice plates are sandblasted to ensure consistent friction for the joint. GRP plates are fitted between the shear panels and the splice plates to provide electrical insulation.

Each pair of adjacent shear panels is sandwiched between a pair of splice plates using 22 M42 Inconel 718 studs. These are all tensioned simultaneously using a set of 22 hydraulic tensioners. Simultaneous tensioning minimizes the loss of preload that occurs when load is transferred from the tensioners to the nuts.

As of August 2017 8 sets of splice plates have been customized and delivered just-in-time to Naka. These may be seen fitted to the TF coils in Fig. 8.

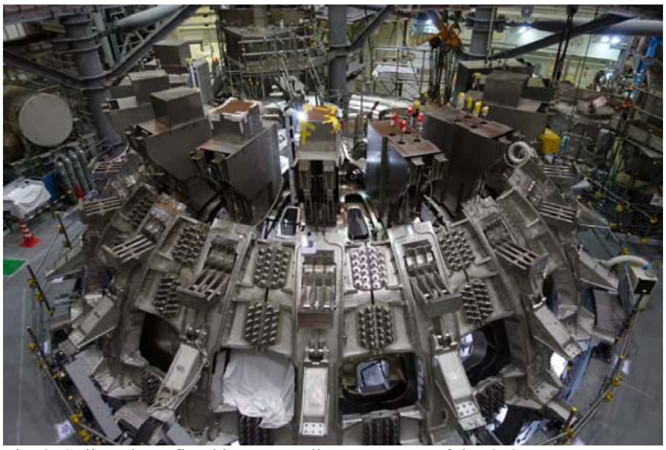

Fig. 8: Splice plates fitted between adjacent sectors of the OIS

V. Error field correction coils

Besides the main superconducting coils described above and the fast plasma position control coils and the resistive wall mode stabilizing coils mentioned in the introduction, JT-60SA will have a set of 18 saddle-shaped error field correction coils [32],[33] installed close to the plasma within the vacuum vessel. These are intended to compensate for any small error in the main field generated by TF and EF coils. Such compensation is beneficial towards the containment of non-axisymmetric plasma modes. The manufacturing of these coils was completed in March 2017.

VI. Conclusion

The manufacturing of the equilibrium field coils and the error field correction coils for JT-60SA is complete, and the manufacturing of the toroidal field coils and the central solenoid is well advanced. The assembly of the TF magnet is underway and will be completed in the first half of 2018.

REFERENCES

[1] H. Shirai et al, "*Recent progress of the JT-60SA project*", Nuclear Fusion, Volume 57, Number 10, June 2017

[2] Y. Kamada et al, "*Progress of the JT-60SA project*", Nuclear Fusion, Volume 53, Number 10 , September 2013

[3] Y. Kamada et al, "*Plasma regimes and research goals of JT-60SA towards ITER and DEMO*", Nuclear Fusion, Volume 51, May 2011

[4] S. Davis et al, "*Status of the JT-60SA magnet system"*, IEEE 26th Symposium on Fusion Engineering (SOFE), 2015

[5] K .Yoshida, "*Construction Status of the Superconducting Magnet System for JT-60SA*", IEEE Transactions on Applied Superconductivity, Volume 26, Issue 4, June 2016,

[6] K. Koide, "*JT-60SA superconducting magnet system*", Nuclear Fusion, Volume 55, Number 8, June 2015

[7] Yoshida, K.et al, "*Conceptual Design of Superconducting Magnet System for JT-60SA*," IEEE Trans. Appl. Super. vol.18, no.2, pp.441,446, June 2008

[8] Tomarchio, V. et al, "*A Global Structural and Electromagnetic Finite Element Model for the Prediction of the Mechanical Behavior of the JT-60SA Superconducting Magnet System*," IEEE Trans. Appl. Super. vol.22, no.3, pp.4703304,4703304, June 2012

[9] Tomarchio, V. et al, "*Design of the JT-60SA Superconducting Toroidal Field Magnet*," IEEE Trans. Appl. Super. vol.20, no.3, pp.572,575, June 2010

[10] P. Decool et al, "*JT-60SA TF Magnets Industrial Production Status at Alstom*", IEEE Transactions on Applied Superconductivity, Volume 26, Issue 4, June 2016

[11] P. Decool et al, "*Starting the production of the CEA JT-60SA TF coils at Alstom*," Fusion Engineering and Design, Volumes 98–99, October 2015, Pages 1044-1047

[12] G.M. Polli et al, "*JT-60SA TF coils procured by ENEA: An intermediate assessment*", Fusion Engineering & Design, in press

[13] A. Cucchiaro et al, "*Qualification and preparatory activities for the manufacturing of 9 TF coils of the JT-60SA magnet*", Fusion Engineering and Design, Volume 88, Issues 9–10, October 2013, Pages 1605-1608

[14] P. Rossi et al, "*Manufacturing of the Casing Components for the JT-60SA Toroidal Field Coils*", IEEE Transactions on Applied Superconductivity, Volume 26, Issue 4, June 2016

[15] P. Rossi et al, "*Status of casing manufacturing for JT-60SA toroidal field coils*", Fusion Engineering & Design, in press

[16] P. Decool et al, "*JT-60SA TF magnet joints developments and prequalification*", IEEE Trans. Appl. Super. Vol.22, n°3, 4202504 (2012)

[17] K. Tsuchiya et al, "*Fabrication and installation of equilibrium field coils for the JT-60SA*", Fusion Engineering and Design, Volume 88, Issues 6–8, October 2013, Pages 551-554

[18] K. Tsuchiya et al, "*Manufacture of the Winding Pack and Development of Key Parts for the JT-60SA Poloidal Field Coils*", IEEE Transactions on Applied Superconductivity, Volume 22, Issue 3, June 2012

[19] K. Tsuchiya et al, "*Recent Progress of the Design Activity for the Poloidal Field Coil System in JT-60SA*," IEEE Trans. Appl. Super. vol.20, no.3, pp.525,529, June 2010

[20] K. Tsuchiya et al, "*Progress of Manufacturing CS and EF Coils for the JT-60SA Tokamak*", IEEE Transactions on Applied Superconductivity, Volume 26, Issue 4, June 2016

[21] H. Murakami, "*Development of terminal joint and lead extension for JT-60SA central solenoid*", IEEE Transactions on Applied Superconductivity, Volume 25, Issue 3, June 2015

[22] Tsuchiya, K et al, "*Design of the Precompression Mechanism and Gravity Support for the Central Solenoid Assembly in the JT-60SA Tokamak*," IEEE Transactions on Plasma Science, vol.42, no.4, pp.1042,1046, April 2014

[23] K. Tsuchiya et al, "*Mechanical design of the central solenoid assembly for the JT-60SA Tokamak*," 2013 IEEE 25th Symposium on Fusion Engineering (SOFE), vol., no., pp.1,5, 10-14 June 2013

[24] K. Kizu et al, "*Development of Central Solenoid for JT-60SA*", IEEE Transactions on Applied Superconductivity, Volume 23, Issue 3, June 2013

[25] K. Kizu et al, "*Construction of the Jacketing Facility and First Production Results of Superconductor for JT-60SA*," IEEE Trans. Appl. Super. vol.20, no.3, pp.538,542, June 2010

[26] K. Kizu et al, "*Conductor Design of CS and EF Coils for JT-60SA*," IEEE Trans. Appl. Super. vol.18, no.2, pp.212,215, June 2008

[27] L. Zani et al, "*Completion of TF Strand Production and Progress of TF Conductor Manufacture for JT-60SA Project*", IEEE Transactions on Applied Superconductivity, Volume 24, Issue 3, June 2014

[28] A. della Corte, "*ITER and JT-60SA Conductor Production at ICAS*", IEEE Transactions on Applied Superconductivity, Volume 23, Issue 3, June 2013

[29] L. Muzzi et al, "*The JT-60SA Toroidal Field Conductor Reference Sample: Manufacturing and Test Results,*" IEEE Trans. Appl. Super. vol.20, no.3, pp.442,446, June 2010

[30] H. Murakami et al, "*Current sharing temperature of central solenoid conductor for JT-60SA under repetition excitation*", Proceedings of 24th international cryogenic engineering conference and international cryogenic materials conference 2012

[31] H. Murakami et al, "*Development and Test of JT-60SA Central Solenoid Model Coil*", IEEE Transactions on Applied Superconductivity Volume 24, Issue 3, June 2014

[32] G. Matsunaga, "*In-vessel coils for magnetic error field correction in JT-60SA*," Fusion Engineering and Design, Volumes 98–99, October 2015, Pages 1113-1117

[33] H. Murakami, "*Thermal and Mechanical Design of Error Field Correction Coil for JT-60SA*", IEEE Transactions on Applied Superconductivity, Volume 26, Issue 4, June 2016