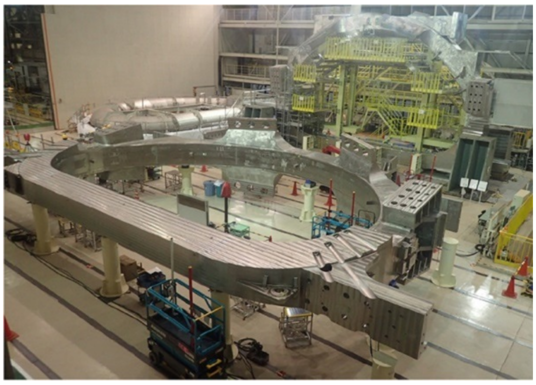
MHI's Continuous Challenge to realize Fusion Energy - Efforts for International Projects of ITER and Broader Approach (BA) -



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Fusion energy does not emit CO_2 and the fuel required for a fusion reaction is abundant in seawater, so fusion energy is expected to be a permanent "dream energy source." Japan is participating in the international project "ITER", which is being promoted through the international cooperation of seven ITER parties (Japan, the European Union (EU), the United States (US), Russia, China, South Korea and India) and also participating in broader approach (BA) activities aimed at the early realization of the fusion demonstration reactor (DEMO reactor), which is being promoted under cooperation of Japan and the EU in parallel with the ITER project. Mitsubishi Heavy Industries, Ltd. (MHI) has completed the world's first toroidal field (TF) coil for ITER and has just started making a prototype to prepare for the manufacture of an actual divertor. For BA activities, MHI is involved in the concept study of plasma confinement vacuum vessels for DEMO reactors, the study of maintenance concepts of equipment inside the vessel and the development of divertors for JT-60SA (SA stands for Super Advanced). This report presents the status of these activities.

1. Introduction

ITER is currently under construction in Saint-Paul-lès-Durance in southern France, and the first plasma operation is scheduled around 2025. The fusion plasma burning experiments using deuterium and tritium will be started around 2035. The missions of ITER will be to attain "long-term fusion plasma burning" and to demonstrate "fusion reactor engineering technology toward the practical application of fusion energy" , and then ITER will acquire the data for the subsequent fusion demonstration reactor (hereinafter referred to as DEMO reactor)to demonstrate electric power generation. The roadmap for the development of the DEMO reactor was presented by the Ministry of Education, Culture, Sports, Science and Technology as the "A Roadmap toward Fusion DEMO Reactor (first report)" (July 24, 2018), in which the role of fusion-related projects including ITER and the action plan for the development of the DEMO reactor was summarized and reported[(1)].

We are responsible for the production of five TF coils for ITER. The first TF coil was completed on January 30, 2020, which was the very first unit in the world, and the first unit was shipped at the end of February, and then delivered to the ITER site at the end of April. The second TF coil was subsequently shipped at the end of May and delivered to the ITER site at the end of July. The following units will be shipped as soon as they are completed. In addition to the TF coil, we are also working on the development of a divertor, which is a device for exhausting impurities and indispensable for plasma confinement, and we have just started trial production of its full-scale mockup. We are currently developing technology for the manufacture of the actual equipment, while optimizing the manufacturing process at the same time.

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For BA activities, based on the fusion device design and manufacturing technology that we have experienced so far, we are performing the concept design of the vacuum vessel structure for the DEMO reactor and that of the remote maintenance scenario and handling tools for in-vessel components such as blanket and divertor, and are participating in R&D activities on the JT-60SA divertor.

2. Activities for ITER

2.1 Completion of the world's first and largest superconducting coil - Activities on TF coil for ITER -

The TF coil for ITER is a large D-shaped heavy structure with a height of about 16 m, a width of about 9 m and a weight of about 300 tons, which is used at -269°C (absolute temperature 4 K) and generates a magnetic field of 11.8 T at maximum and 5.3 T at the center of the plasma. In ITER, 18 TF coils are discretely arranged in a doughnut shape and generate a magnetic field to confine the plasma. As shown in **Figure 1**, the TF coil has a winding pack (WP) and the coil structure that houses the WP consists of the inboard coil structure (AU), the outboard coil structure (BU), the inboard sealing plate (AP) and the outboard sealing plate (BP). The WP consists of 7 layers of double pancakes (DP) and the superconductor is stored in the DP.

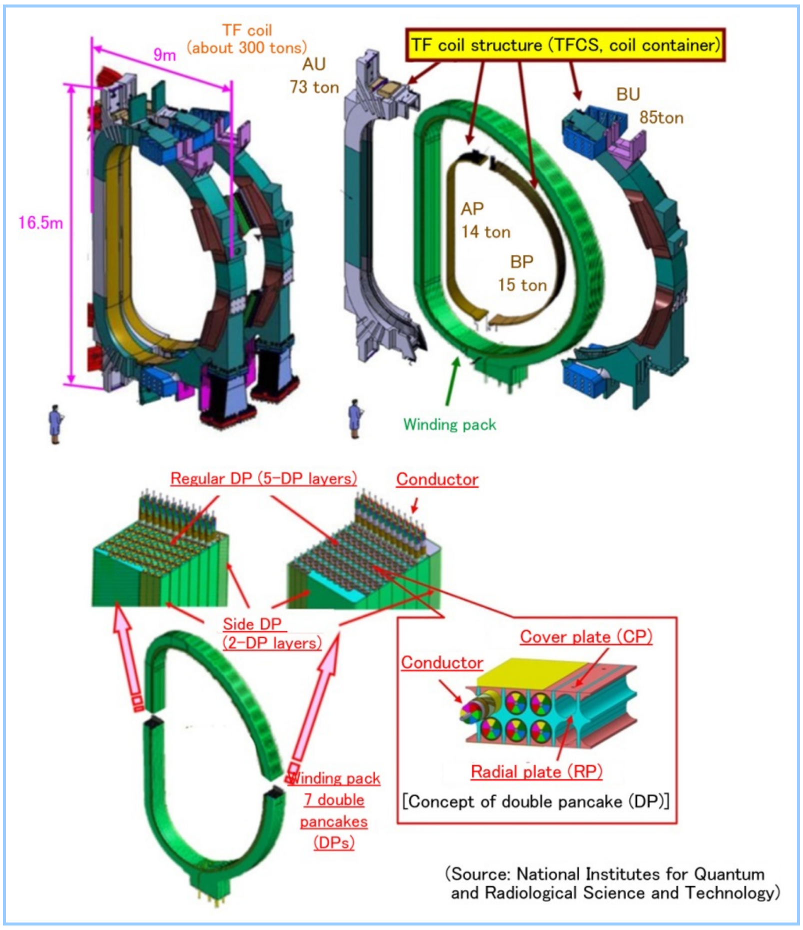

Figure 1 Concept of toroidal field coil

For the purpose of keeping the plasma confinement stable, the tolerance of the magnetic field generated by the TF coil is just several Gauss. Therefore, the TF coil is required to have a dimensional accuracy on the order of 1 mm with respect to the larger-than-10 m welded structure. In order to provide products that meet this requirement, we have adopted electron beam welding, narrow gap TIG welding and laser welding, which have low welding heat input, to suppress welding deformation. In the machining of large structures, we pay attention to change the size of the object as the temperature changes. Therefore, in order to perform machining with high accuracy, we used a large scale and high precision milling machine known as "Super Miller", which is fully equipped with air conditioning that can keep the temperature constant during machining. Moreover, a special tool with 5-axis control was used for its machining because the TF coil is a structure composed of complicated curved surfaces. In advance of manufacturing an actual product, we have performed small-scale tests to determine welding and machining conditions, and then, a full-scale mock-up test was performed to demonstrate and verify processes applicable for the actual product(2). In this way, we manufactured the actual product while steadily solving the technical issues. **Figure 2** shows the typical technologies for manufacturing TF coils.

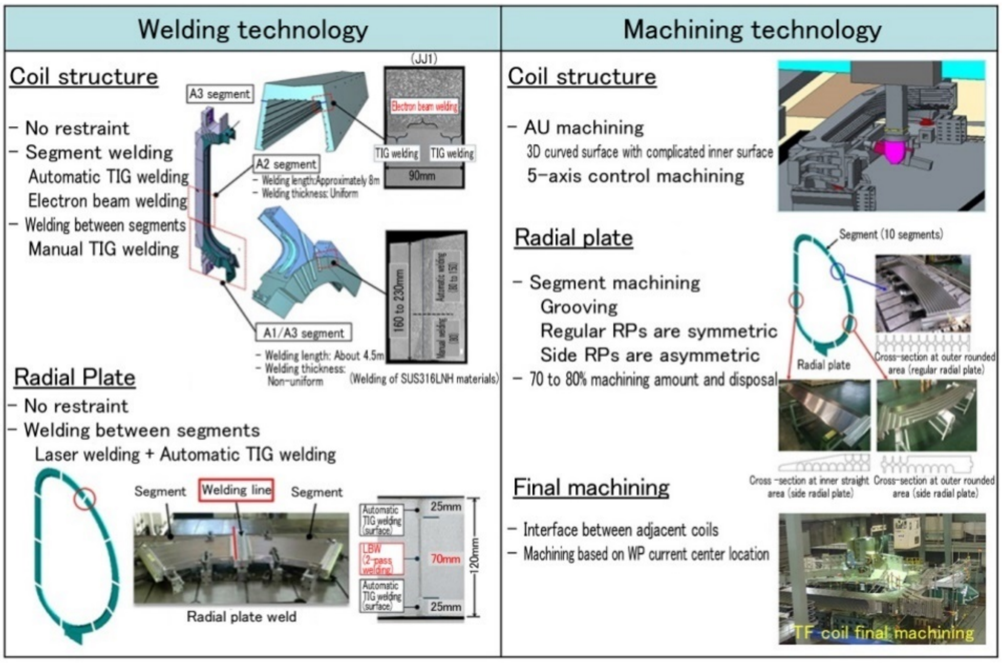

Figure 2 Typical manufacturing technology used for TF coils

Figure 3 shows the process of integrating the coil structure and the WP. The magnetic configuration and distribution which are generated by the TF coil, is dependent on the current center location in the TF coil. Therefore, the current center location is one of important parameters in manufacturing. In the integration process, assembly, machining and so on, are performed based on the current center location marked on the WP surface. The current center location is transferred onto the TF coil structure for the final machining, and the transferred one is used as the basis of on-site installation of the TF coil.

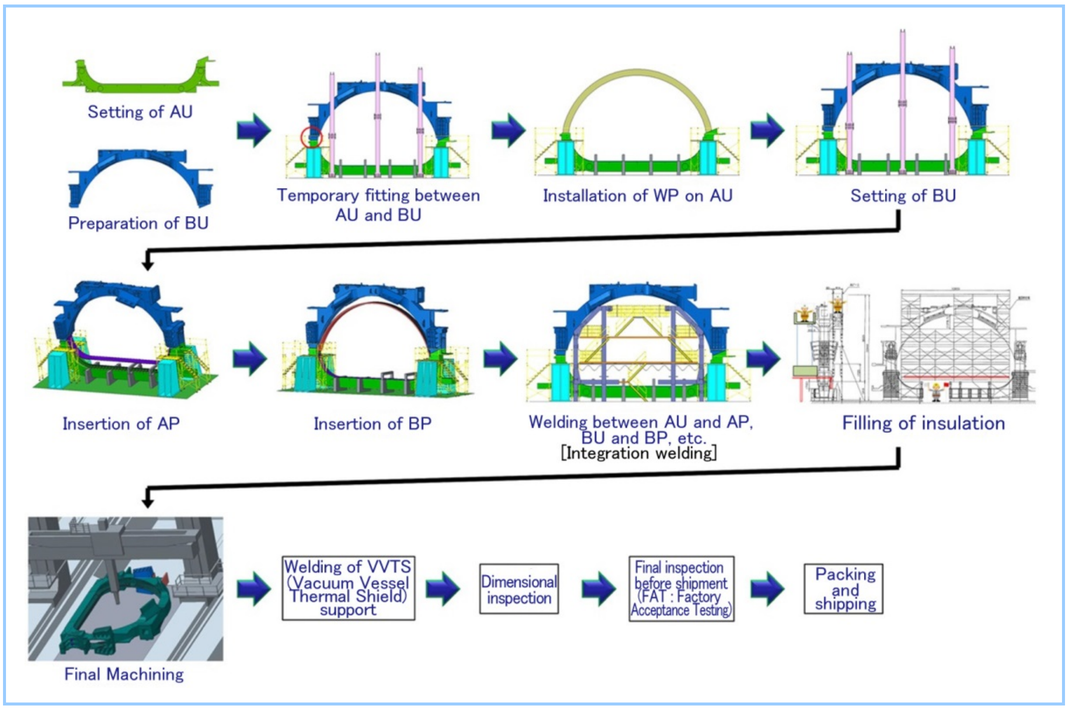

Figure 3 Schematic flow of integration process

After installing the WP into the coil structure AU and covering it with the BU, the center location of the WP is transferred onto coil structure and the AP and BP are welded. In order to secure a gap for pouring insulating materials between the WP and the coil structure, welding is performed as monitoring displacement during welding so that the amount of tilt deformation at the top of the BU is attained within target value (3 mm or less) with consideration of welding sequences. Here, it is also necessary to suppress deformation so as to prevent a shortage of excess thickness for the final machining. Since this integration process allows no backtracking, the weld deformation behavior was assessed by welding deformation analysis and welding sequences were studied in advance with use of mock-ups. As shown in **Figure 4**, the welding strain was first measured by a flat plate test and compared with the analysis results. After we verified the validity of the analysis, the welding deformation analysis was performed to assess the welding deformation with the large full-scale mockup, and the tendency of welding deformation behavior was confirmed by comparison with the measured welding deformation. The trial tests of the integration welding were also performed by using a 1/3 scale model. Based on the data obtained through various welding tests, the welding process applicable for the actual product was established, and the tilt deformation due to welding was suppressed within the target value of 3mm.

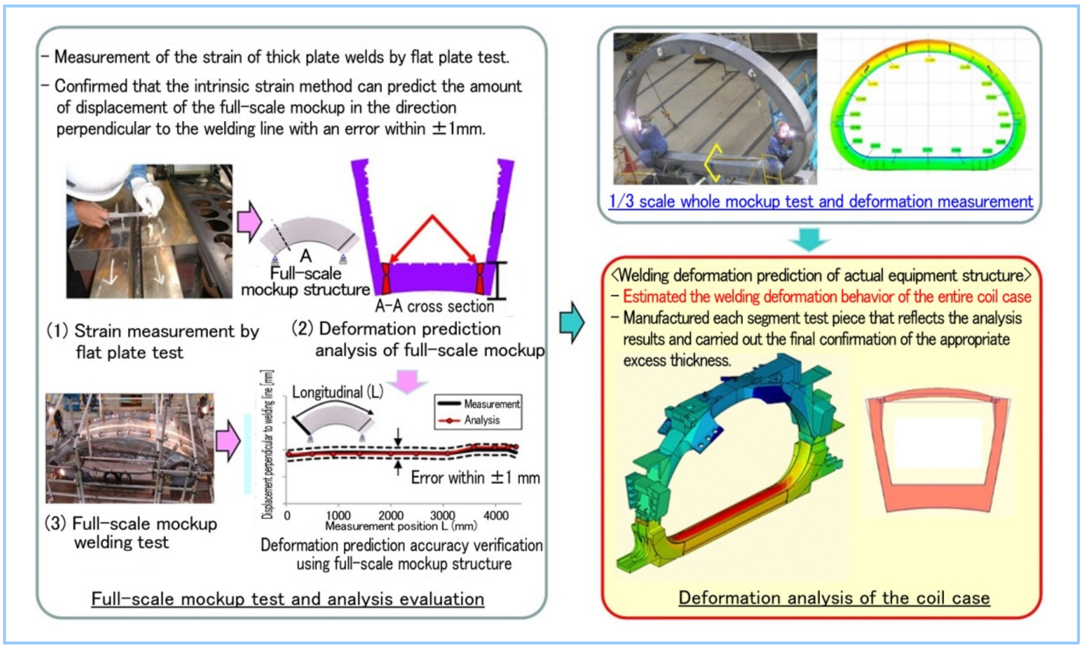

Figure 4 Establishment of integration welding technology (full-scale mockup test and sensitivity analysis)

The technology for controlling the deformation of such a large heavy structure due to welding on the order of millimeters was achieved based on technologies accumulated through experiences of manufacturing and handling of the heavy components in nuclear power plants. We have adopted new technologies such as laser beam welding etc., which is a low heat input to objects, to suppress welding deformation, and have applied the welding techniques without strong constraint jigs by monitoring the deformation during welding.

The very first TF coil in the world was shipped from our shop at the end of February and arrived at the ITER site at the end of April. The second TF coil was also delivered to the ITER site at the end of July. Assembly preparation work is underway at the site (**Figure 5**).

Figure 5 Shipping of the very first TF coil

The accomplishment of completion of the very first TF coil, which is one of the largest superconducting coils in the world, was coming from collaboration and cooperation with the National Institutes for Quantum and Radiological Science and Technology and Mitsubishi Electric Corporation.

2.2 Development of divertor, receiving high heat flux from plasma

A divertor in a fusion reactor is a device that removes particles (helium) generated in a fusion reaction, unburned fuels (deuterium and tritium) and other impurities produced by the interaction between plasma particles and the wall in the reactor, and is one of the important devices related to performance of plasma stable confinement. Divertor elements, such as plasma facing units (PFUs), are exposed to extremely severe thermal environments. In the case of ITER, the PFUs endure a high heat load of 10 MW/m^2 at a steady state and 20 MW/m^2 at a non-steady state, respectively.

Figure 6 shows the concept of the ITER divertor structure and **Table 1** summarizes the typical requirements. Divertors are procured by Japan, the EU and Russia, and Japan is responsible for the Outer Vertical Target (OVT). Three cassettes of divertors are installed in the 20-degree sector in the circumferential direction (toroidal direction) of the vacuum vessel doughnut, so that 54 cassettes are installed in total. Two OVTs are set on one cassette body. The OVT is composed of 11 PFUs (receiving high heat-load from plasma) and a steel support structures (SSS) (supporting the 11 PFUs). The PFU consists of cooling pipe and tungsten (W) block with extremely high melting point. About 150 W blocks are brazed to a copper alloy cooling pipe. It is required to keep the step between adjacent PFUs and the gap between the blocks within the specified tolerances (the step needs to be within 0.3 mm, the overall profile 0.5 mm and the gap between blocks about 0.5 mm to 1 mm). The shape of the W block is not a simple rectangle, but a complicated structure that changes shape in the direction of the cooling pipe axis.

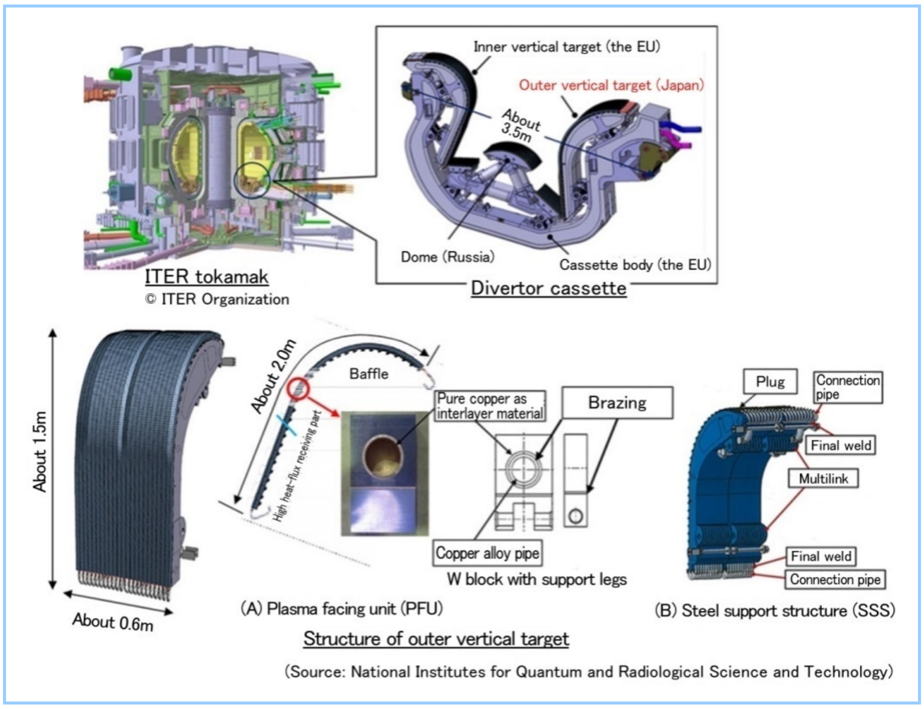

Figure 6 Concept of ITER divertor and OVT (Over Vertical Target) structure

We have been developing divertors for ITER since around 2012, and the PFU consisting of both a carbon fiber-reinforced composite (CFC) monoblock and W monoblock passed the repeated heat load tests required by ITER for the very first time in the world[2]. After that, the PFU design was changed to a full-W PFU from the mixed PFU, and the full-W PFU prototypes were also manufactured. These full-W PFU prototypes cleared the ITER requirements of the heat load tests for the first time in the world as well as the previous CFC-W mixed PFU prototypes (**Figure 7**). These efforts were achieved through collaboration with the National Institutes for Quantum and Radiological Science.

Table 1 Main feature of ITER divertor

Main function	Items	Specifications and structure
Heat removal	Heat and plasma particle resistant	- Material: Tungsten block - Maximum surface heat load: 10 MW/m^2 (normal), 20 MW/m^2 (abnormal) - Profile: 0.5 mm - Step to adjacent channel: within 0.3 mm
	Cooling performance	- Type of cooling pipe penetrating tungsten block - Cooling water: 4 MPa, 10 m/s, inlet temperature of 100°C, outlet temperature of 140°C - Brazing (metallurgical joint to ensure heat transfer) - Twisted tape (promoting turbulent heat transfer and improving critical heat flux)
Strength	Support against pressure and electromagnetic force Heat elongation absorption	- Cooling pipe: CuCrZr pipe (precipitation hardened copper alloy) - Support leg joint welding strength: Can withstand maximum load (9 kN) - Steel support structure: XM-19 (high-strength steel) - Multi-link, piping curvature (flexibility)
Maintenance and repair	Removability of PFU alone	- Support leg-pin-plug structure
Other	Radioactive resistance	- Co control - Nuclear heat removal

Before heat load test　　　　After heat load test

A repeated heat load test was conducted at the Efremov Institute in Russia and the soundness of the joints was confirmed at 10 MW/m^2 x 5000 cycles (ITER requirement: 1000 cycles) and 20 MW/m^2 x 1000 cycles (ITER requirement: 300 cycles) without damage. (satisfying ITER requirements for the first time in the world.)

* The three from the back are made by our company.

(Source: National Institutes for Quantum and Radiological Science and Technology)

Figure 7 High heat flux test of full W-PFU prototype of outer divertor

The main technical issues on the OVT for the actual products are as follows: (1) confirmation of the integrity of the brazing joint, (2) ensuring the integrity of the dissimilar joint between the stainless steel and copper alloy cooling pipes and (3) checking the weldability and machinability of special stainless-steel (XM-19), which is the structural material of the SSS. It is necessary to continue to carry out element tests and full-scale mockup tests etc., and it is important to acquire data related to manufacturability so that the yield reaches 100%.

(1) Confirmation of integrity of brazing joint

Since the joint state of a brazed part affects the heat removal performance and joint strength, it is necessary not only to control the brazing process, but also to perform non-destructive inspection using ultrasonic waves (Ultrasonic Test, UT) to confirm that there are no harmful defects in the joint. The UT is performed from inside the cooling pipe and it is important to select conditions (the probe size and frequency) suitable for the location of the defect in brazed part, the size of the defect to be detected and the material combination of the joint boundary. The conditions of the probe were obtained from the sound wave transmission state using numerical simulation in advance. Prototype of probe was then manufactured, and the preliminary performance test of the prototype was carried out. **Figure 8** shows the results of the numerical simulation and prototype test on the brazed part. Here, Figure 8 (a) is the result of sound field simulation in which the aspect of propagation of ultrasonic wave is numerically evaluated, and Figure 8 (b) is the result of echo simulation in which the intensity of reflection of ultrasonic wave is numerically evaluated. In these simulations, the flat-shaped and spherical-shaped (SR25) probes were assumed, but in both cases, the convergence in the axial direction was slightly inferior to that in the circumferential direction. Resulting from these simulation, we will need to improve a probe with better convergence by optimizing the probe shape and frequency. In the characteristic test using a small-scale test piece as shown in Figure 8(c), signals could be obtained at different boundaries: the boundary between the copper alloy and oxygen-free copper on its outer surface, and the boundary between the oxygen-free copper and the W block. The detectability was verified by the elementary tests. In the actual product, more than 1200 pieces (58 cassettes = 22 pieces x 58 cassettes = 1276 pieces) will be inspected. Therefore, we will optimize the inspection method, including not only the improvement of the detectability (detection sensitivity), but also the reduction of the detection time.

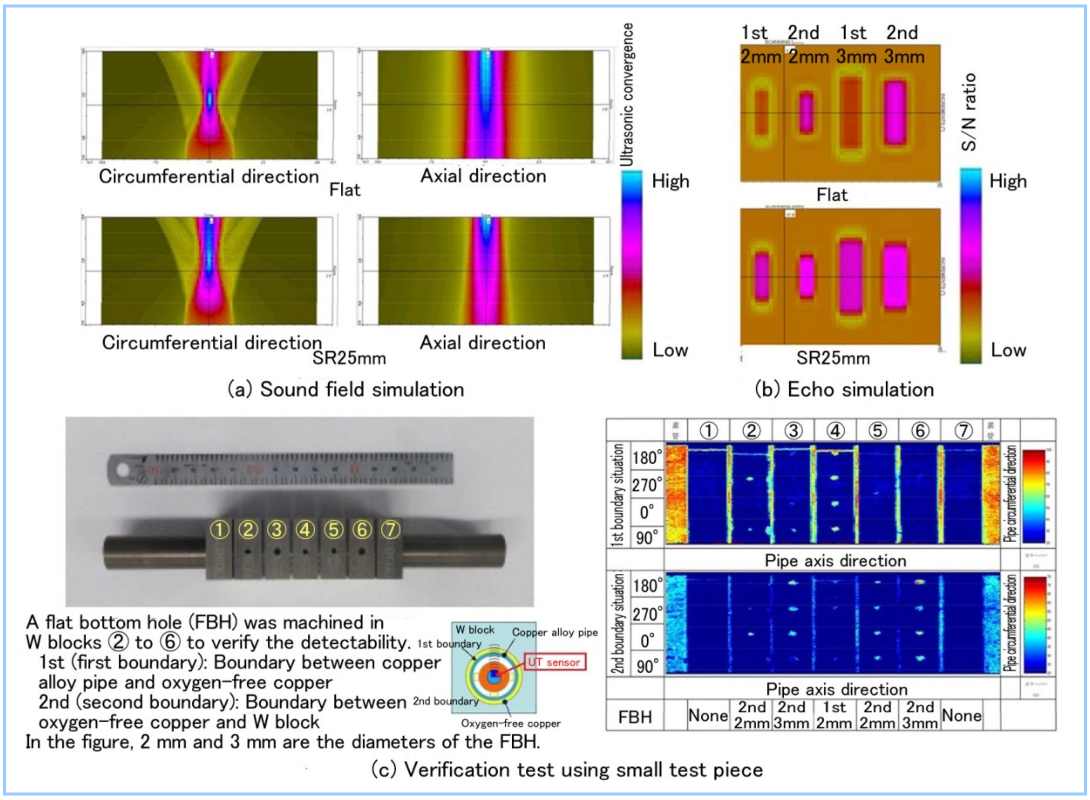

Figure 8 Preliminary tests of performance and sensitivity of UT for brazing joint

(2) Ensuring integrity of dissimilar joint

The weld between the PFU cooling pipe and the header is a dissimilar weld of copper alloy (CuCrZr) and stainless steel, and the weld joint is subjected to a water pressure of several MPa. Therefore, it is important to ensure the integrity of the weld. A beam welding method has been adopted to suppress welding deformation for the dissimilar weld joint as shown in Figure 9. The weld joint is that a transition piece of Ni-based alloy is placed between stainless steel and Cu alloy, so that weld defects will be avoided rather than direct welding of stainless steel to Cu alloy. We conducted element test to establish the beam welding process, and as a result, we obtained the prospect that sound dissimilar joints can be made (**Figure 9**). In the future, we will optimize the welding conditions such as beam penetration depth etc.

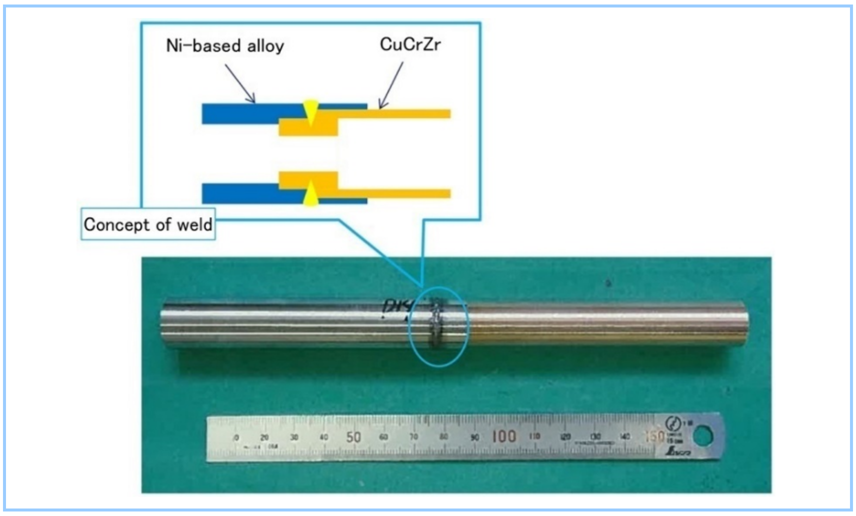

Figure 9 Element test of dissimilar welding (CuCrZr and Ni-based alloy)

(3) Checking weldability and machinability of SSS

XM-19, which is the structural material used for the SSS, contains more Mn and N than ordinary austenitic steels such as SUS316, SUS316Land so on, and it has higher strength than the ordinary ones. Weldability and machinability of XM-19 are harder than those of ordinary stainless-steel, and it will be needed to find appropriate process conditions.

We manufactured the test piece shown in **Figure 10** (a) to perform various tests (tensile test, side bending test, cross-section observation) as preliminary tests for a weldability test. Although in the tensile test a fracture occurred in the weld, the resulting tensile strength was 809 MPa, which satisfied the required value of 690 MPa (at room temperature). No significant cracks were observed in the side bending test. No cracks and no porosity were observed in the cross-sectional observation of the weld. Since these tests are very preliminary ones, we will plan to optimize the welding conditions for the actual equipment in the future.

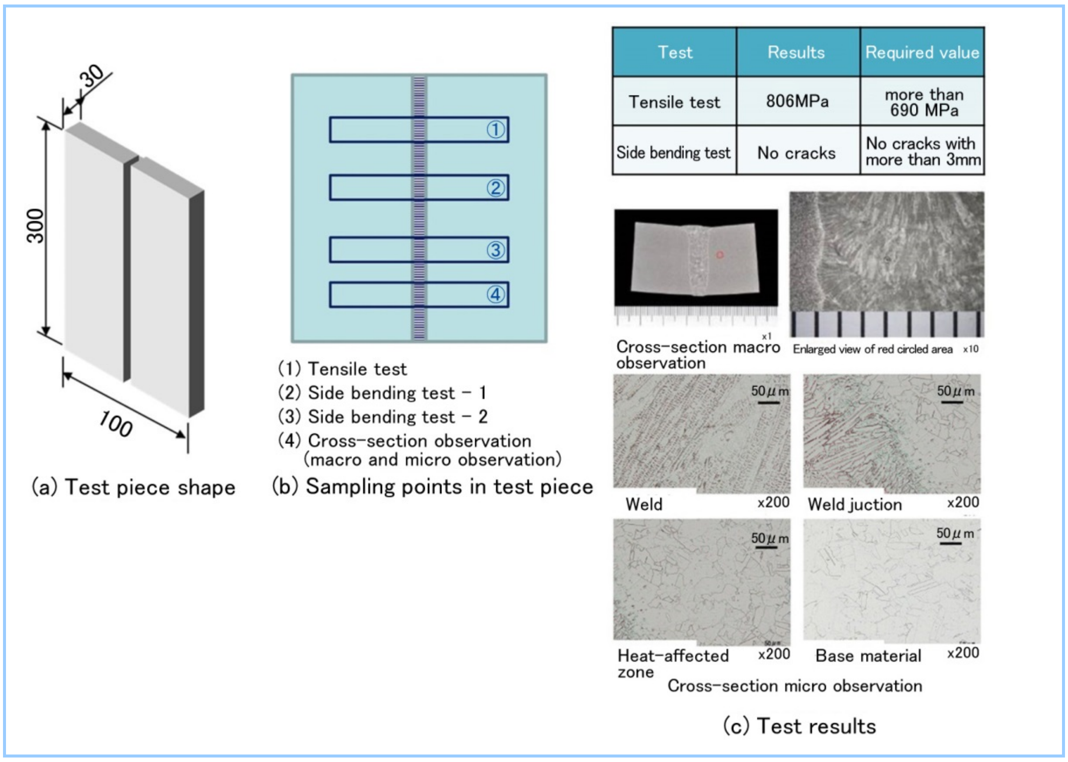

Figure 10 Weldability test of XM-19

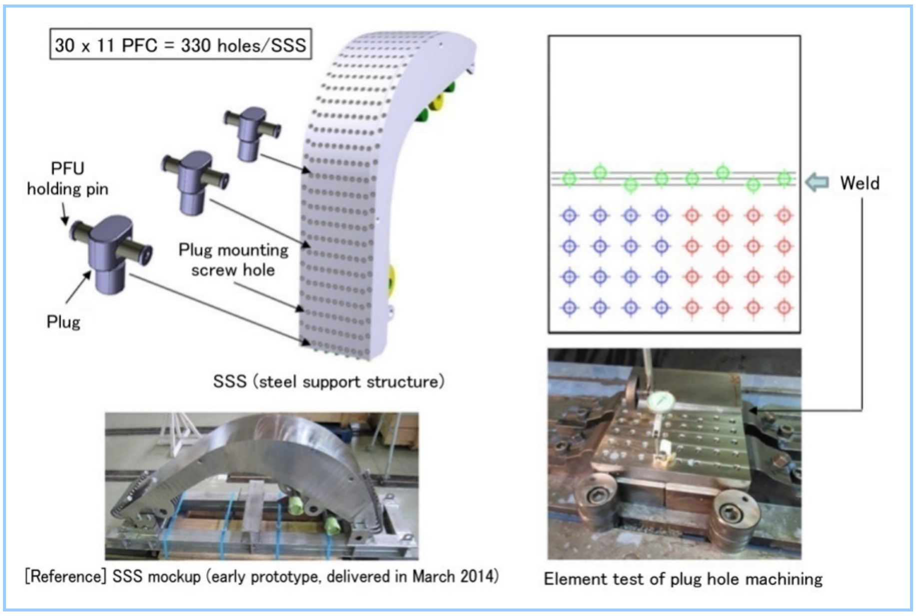

Figure 11 Element test of machining screw holes on SSS

The PFU is supported by the SSS with a pin connection structure and a grooved plug that holds the pin, and screw holes are screwed to the SSS. Therefore, we conducted a preliminary test for screw hole machining (**Figure 11**). It is necessary to machine the plug holes (screw holes) so that they are aligned in the direction of the PFU cooling pipe. However, the machined surface of the SSS has a curved surface and it is important to maintain the posture of the tool during machining. As a result of the preliminary test, we confirmed the correlation between the tool type and the arrangement (alignment), and we obtained the prospect of the screw hole machining on the curved surface.

3. Efforts for BA activities

BA activities are being promoted through bilateral cooperation between Japan and the EU toward the early realization of fusion energy and consist of the following three projects: (1) International Fusion Energy Research Centre project (DEMO Design and Research and Development Coordination Centre, ITER Remote Experimentation Centre, Computational Simulation Centre), (2) Engineering demonstration and engineering design activity project for International Fusion Material Irradiation Facility and (3) Satellite Tokamak Program with JT-60SA as core equipment.

This report describes an overview of our efforts for the DEMO reactor design related to (1) and the JT-60SA related to (3).

3.1 DEMO reactor design

We have been studying the concept of the reactor structure and the remote equipment for the DEMO reactor since the early stage of BA activities. In the DEMO reactor, the plasma confinement vacuum vessel and the in-vessel structure (blanket and divertor) are required to have a structural concept consistent with the remote maintenance scenario and handling tools. The blanket is an important component that produces tritium (a fusion fuel) and that converts fusion energy to thermal energy. Conceptually, the blanket segments are carried out of the upper port of the vessel and transferred to the storage area when it is replaced for maintenance and so on. As shown in **Figure 12**, the blanket of the DEMO reactor adopts a banana-shaped segment structure and is carried in and out through the upper port. The divertor concept adopts the same cassette concept as ITER and its segmentation was determined, considering a lower port opening area to be able to carry in and out through the lower port and considering interference with the routing of the cooling pipes and so on (Figure 12).

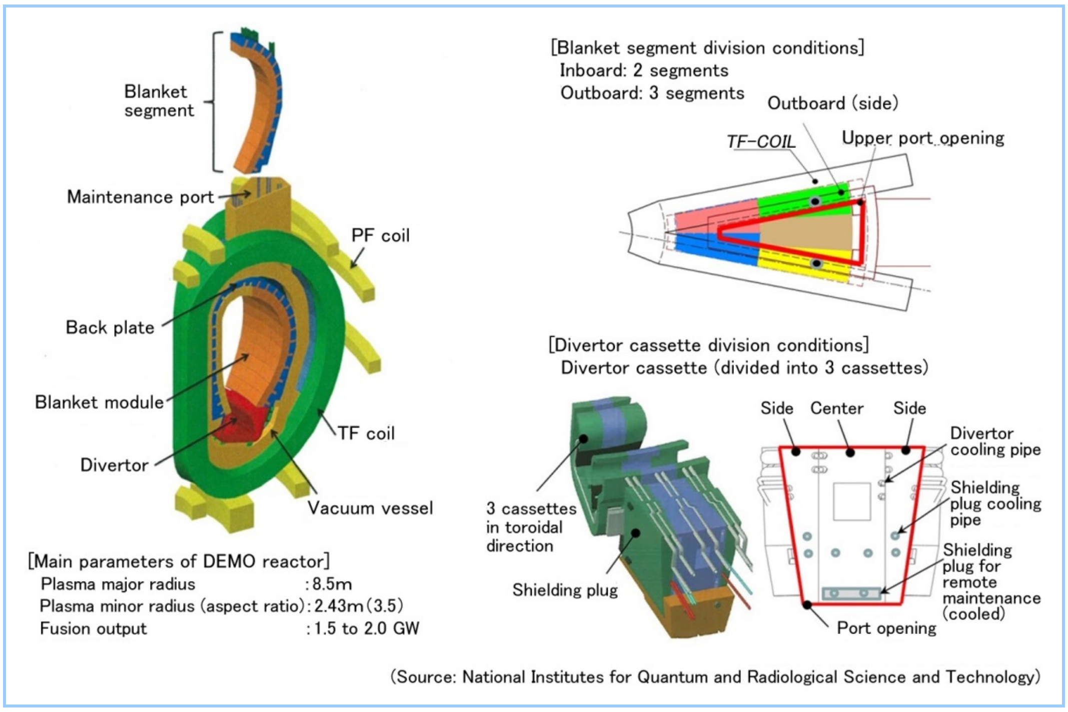

Figure 12 Concept of reactor structure of DEMO reactor

The main loads that determine the concept of the vacuum vessel structure are the electromagnetic force caused by the following: the rapid decay of the plasma current at the loss of plasma control called disruption, and vertical displacement of plasma due to the loss of plasma position control called VDE (Vertical Displacement Event), and high-speed discharge of a TF coil current due to current quench. We have electromagnetic force analysis technology that analyzes the load on the vacuum vessel due to plasma disruptions and so on. As an example of electromagnetic force analysis, we showed a result of an electromagnetic force analysis due to high-speed discharge of the TF coil current, where the discharge time of the TF coil current was assumed to be 30 seconds (**Figure 13**). The maximum electromagnetic force in total is about 20 MN in the radial direction (-X direction) at 2.25 seconds. In the future, it will be needed to evaluate the structural strength in consideration of the rapid decay and movement of the plasma current, and it will be also important to make an evaluation including the dynamic effect on the transient electromagnetic load in order to study the concept of the vacuum vessel structure of the DEMO reactor.

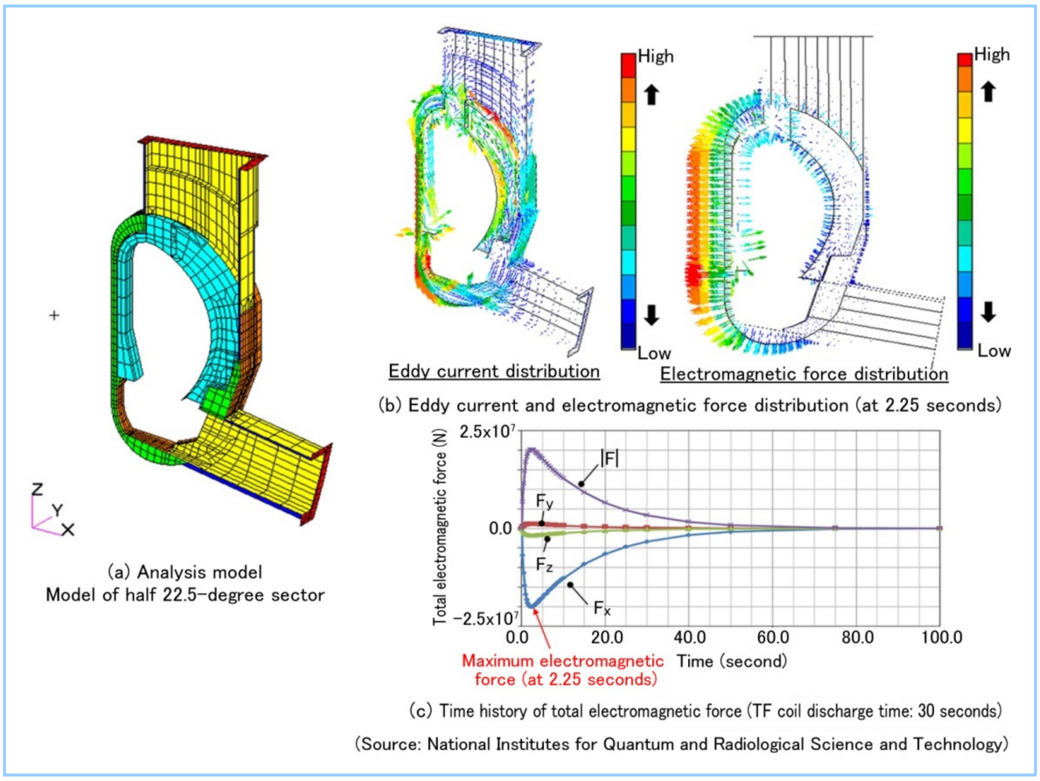

Figure 13 Electromagnetic force analysis on vacuum vessel

3.2 Divertor for JT-60SA

The JT-60SA is an experimental device which is targeted at being used to complement ITER plasma experiments and develop plasma for the DEMO reactor and is under construction in Japan at this time. We are participating in the development of brazing type divertors, to which our basic technology cultivated in divertors for ITER can be applied, and have manufactured a small prototype for a preliminary heat load test. CFC, of which thermal conductivity is higher than that of W, is adopted as the plasma facing material to the JT-60SA divertor, unlike the ITER divertor adopts W. In contrast to metal, CFC has a porous structure, so the brazing material penetrates easily. Therefore, it is a critical issue that the brazing conditions are established in consideration of compatibility of the brazing material with CFC. **Figure 14** shows the structural concept and element test pieces of the JT-60SA divertor. The CFC monoblock used for the JT-60SA has a length of approximately 30 mm in the cooling pipe axial direction and its size is larger than the block used for ITER. Therefore, there is a concern about thermal stress in the vicinity of the joint boundary. In the future, we will optimize the joining conditions including the block size and shape, as well as understand the UT characteristics at the brazing joint and develop the signal processing to improve detectability.

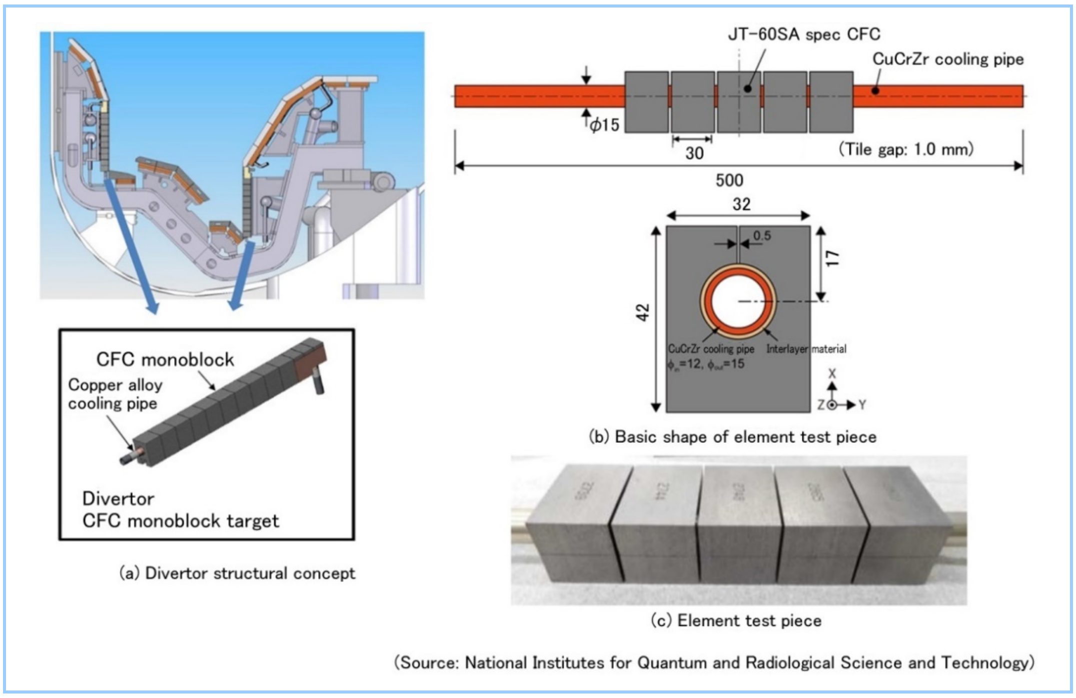

Figure 14 Structural concept and shape of element test piece of JT-60SA divertor

4. Conclusion

In order to complete the ITER’s TF coil first unit, which is one of the world’s largest super-large superconducting coils, we utilized existing technologies, our ingenuity and the latest technologies, and, as a result, we provided the product that is fully satisfied with the ITER requirements. The TF coils are followed by the OVT, and we are diligently working to eliminate technical issues on the divertor, which is the main component for ITER, through research and development tasks and in-house research toward the establishment of actual product. The divertor PFU consists of many parts and is manufactured with high precision while ensuring quality of parts and confirming dimension at step by step in the processes. Since the special steel, XM-19, is used for the steel support structure, the weldability and machinability of XM-19 will be investigated and the processes of welding and machining will be established until start of series production.

In BA activities, designing of the DEMO reactor is currently in the phase of the transition from conceptual study to the conceptual design. In the future, we will carry out the design with manufacturing in mind. Since the DEMO reactor is expected to have a scale about twice that of ITER, it will be needed to design it rationally with cost-consciousness In addition, since the DEMO project is a long-term project, it is also necessary to pass on the basis of design and manufacturing technologies on the nuclear fusion equipment to young engineers from senior engineers, and we will also progress our design and manufacturing technological capabilities for the future. As computer designs and AI-based manufacturing technology innovations become widespread, we will establish new methods that can integrate individual technologies and experience so that rational manufacturing can be performed.

We will continue to conjoin old technologies with new ones and create new methods, and we will also utilize the experience and technology obtained from ITER’s TF coil production, as well as that accumulated in future divertor-related activities, for the DEMO reactor design. We will proceed with development toward the practical application of fusion reactors.

References

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(2) Katsusuke Shimizu et al., MHI's Efforts towards ITER - Challenges to Build Large-size High - performance Superconducting Coils -, Mitsubishi Heavy Industries Technical Review Vol. 52 No. 1 (March 2015) p.15～23