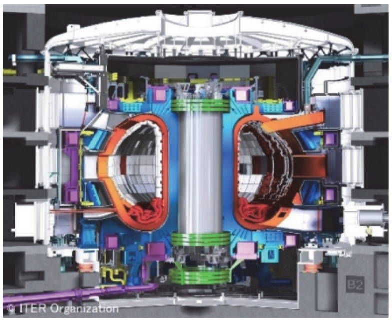
Continuous Challenge and Prospect for Fusion Energy (ITER/ DEMO Reactor)



TAKASHI BABA*1 KATSUSUKE SHIMIZU*2
TATSUSHI MAEDA*1 TOMOHIRO OKAWA*1
MASAHIKO INOUE*2

Fusion energy is expected to be a permanent "dream energy source" because it does not emit CO_2 and fuel for the fusion reaction exists abundantly in seawater. Recently, many fusion startups have been established inside and outside of Japan, and they are developing their technologies while backed by a large amount of funds. Japan is participating in the International Thermonuclear Experimental Reactor (hereinafter referred to as ITER) project, which is being promoted through international cooperation, and is developing in parallel a fusion DEMO reactor in Japan. Mitsubishi Heavy Industries, Ltd. is contributing to the construction of ITER by actual-component manufacturing toroidal field coils and divertors, which are important components for fusion reactions in ITER. In the development of the fusion DEMO reactor, we are promoting design activities based on the knowledge accumulated in Light Water Reactor and ITER projects.

1. Introduction

ITER, currently under construction at Saint-Paul-lez-Durance in southern France, is the world's largest tokamak-type reactor, which confines fusion-reaction-generating plasma in a donut-shaped vacuum vessel. As a result of the revision of the operation plan due to the COVID-19 pandemic and the need for repair of the vacuum vessel and thermal shield [(1)], plasma ignition scheduled originally in 2025 has been delayed, but the original plan to start fusion plasma burning experiments using deuterium around 2035 is still on track and the construction of ITER is still in progress steadily. On the other hand, in Japan, the promotion of the development toward construction of a fusion DEMO reactor to demonstrate power generation is planned, utilizing the knowledge gained from ITER component manufacturing and future ITER operation data.

Mitsubishi Heavy Industries, Ltd. (hereinafter referred to as MHI) was in charge of manufacturing five Toroidal Field (hereinafter referred to as TF) coils for ITER, and in January 2020, became the first in the world to complete such coils and successfully shipped the final one to ITER in August 2023. Besides the TF coils, MHI has been working on manufacturing a divertor, which removes and discharges impurities from plasma and is indispensable for long plasma burning, and completed the manufacturing of its full-scale prototype in March 2024. Currently, MHI is working to optimize the manufacturing process and is proceeding with the sequential manufacturing of the actual component. In addition, MHI is working on the development of the domestic fusion DEMO reactor, such as studying the vacuum vessel structure concept, tritium breeding blanket, and remote maintenance concept for in-vessel components, by utilizing the fusion equipment design and manufacturing technologies that have been cultivated up to now.

2. Challenges to ITER

2.1 Five world's largest superconducting coils shipped to ITER site

TF coils for ITER are large and heavy D-shaped structures with a height of about 16 m, a width of about 9 m, and a weight of about 300 tons (**Figure 1**), and are used at -269°C (4 K absolute temperature) to generate magnetic fields of 11.8 T at the maximum and of 5.3 T at the

*1 Engineering Manager, Advanced nuclear plant engineering Department, Nuclear Energy Systems
*2 Advanced nuclear plant engineering Department, Nuclear Energy Systems

center of the plasma. To maintain stable plasma confinement, the tolerance of magnetic field generated by TF coil is a few gauss for a few-tesla. (i.e., the tolerance is 1/10,000.) Therefore, TF coils are required to have very tight dimensional accuracy in the order of 1 mm for their 10-meter or longer welded structure. To provide products that meet this requirement, MHI conducted elemental tests, verified/demonstrated welding and machining processes using full-scale mockups, and confirmed the deformation behavior in these manufacturing processes through analysis, which are shown in **Figure 2**. This was followed by manufacturing the actual equipment [2] and the first unit was completed in January 2020, which was first in the world. After that, MHI successfully shipped all five coils to the ITER site by August 2023 (**Figure 3**). Note that this completion of a TF coil, which is one of the world's largest superconducting coils, was made possible through collaboration and cooperation with the National Institutes for Quantum Science and Technology (hereinafter referred to as QST) and Mitsubishi Electric Corporation.

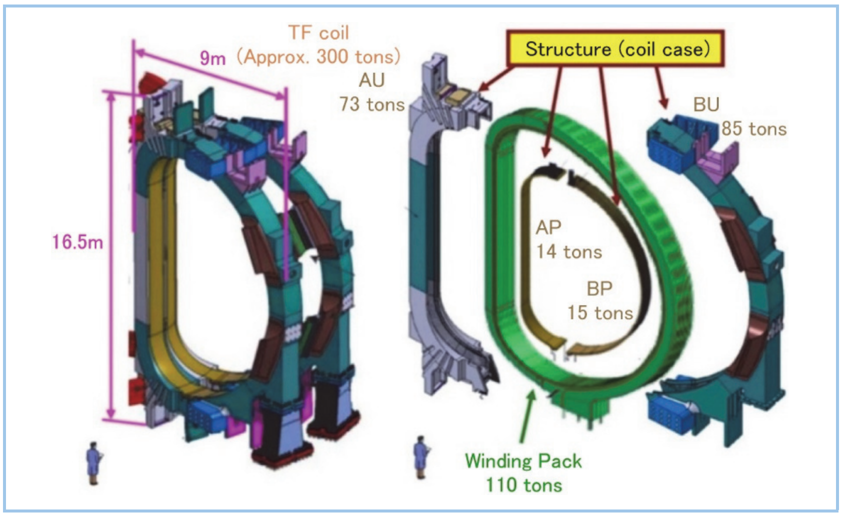


Figure 1　Conceptual diagram of TF coil

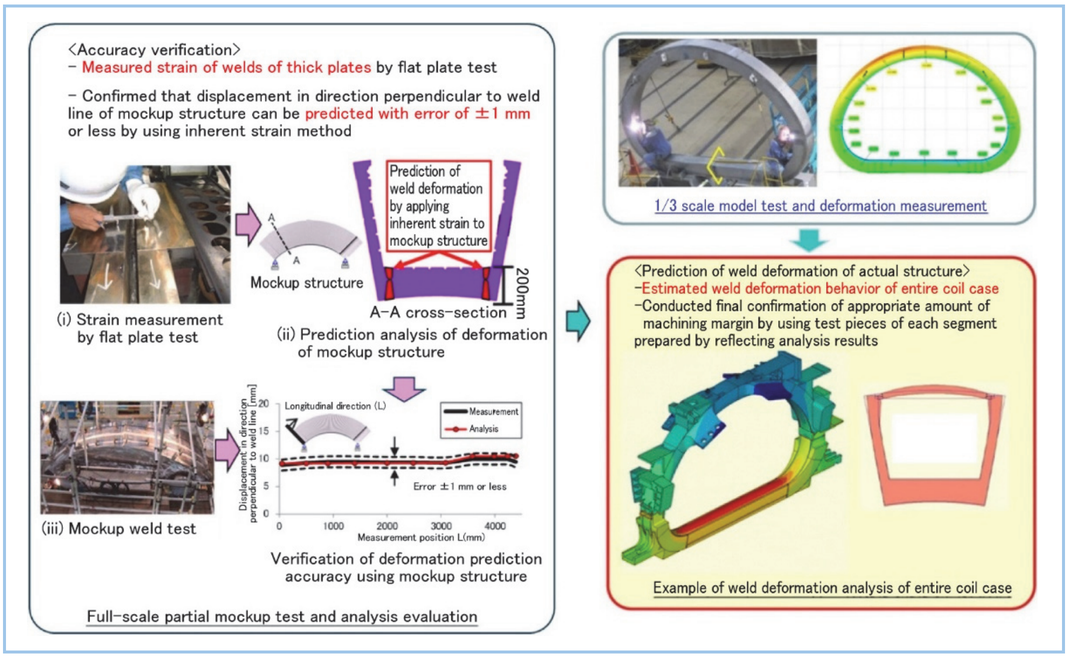


Figure 2　Establishment of integration welding technology (full-scale partial mockup test and deformation analysis)

Figure 3 Shipment of TF coil

2.2 High heat load receiving divertor manufacturing certification obtained, followed by actual equipment manufacturing

The divertor in a fusion reactor discharges the product particles (helium) generated by fusion reactions, unburned fuel, and other impurities produced by the interaction between plasma particles and the inner wall of the vacuum vessel in the reactor, and is one of the most important component required to maintain stable plasma. The divertor is used in a very harsh environment, facing plasma and exposed to heat load, particle load, and neutron loads. The design heat load conditions in ITER are as high as 10 MW/m^2 at a steady state and 20 MW/m^2 at the peak, requiring a high level of heat resistance and heat removal performance.

Tungsten, a high melting metal, was selected as the plasma facing wall material, and a precipitation hardening copper alloy (CuCrZr) with high thermal conductivity and high strength is used for cooling pipes. For joining dissimilar metals with different thermal expansions, brazing was selected, which has been proven to be highly reliable in such operation through many years of research and development. The brazing uses a special filler metal because the joining interface becomes hot and is subjected to neutron irradiation. To establish the brazing process, it was necessary to select the brazing conditions through elemental tests, verify the process using a medium-scale test piece, and finally manufacture a full-scale long test piece to demonstrate the process.

Figure 4 shows the structural concept of the ITER divertor and **Table 1** shows the main required specifications. The divertor is installed in 3 cassettes in the sector which divides the circumferential direction of the donut of the vacuum vessel into 20° each, and 54 cassettes are installed in total. Manufacturing of the divertors is shared by Japan, Europe, and Russia, and the ITER Organization assembles them. Japan is in charge of manufacturing the outer vertical targets (hereinafter referred to as OVTs). An OVT consists of a plasma-facing unit (hereinafter referred to as PFU), which is a high heat load receiving unit, and a steel support structure (hereinafter referred to as SSS) supporting the PFU. A single OVT cassette has 22 PFUs arranged in parallel, each of which consists of a copper alloy cooling pipe brazed with about 150 blocks made of tungsten (W) with a high melting point. It is necessary to install adjacent PFUs so as to keep the step between them within the required tolerance, and also each gap between blocks must be within the required tolerance (the unevenness of the W surface relative to the design shape must be within 0.5 mm, the step 0.3 mm, and the gap tolerance between blocks about ±0.1 mm). The shape of the W-block is not a simple rectangle and changes slightly in the direction of the cooling pipe axis. Therefore, the 22 PFUs consist of five different shapes, making the OVT a very complex structure.

MHI has been conducting numerous verifications to solve technical issues of the divertor OVT and preparing for manufacturing the actual component (2)(3). Before starting the manufacturing of the actual equipment, the ITER Organization requires prior qualification by using (i) a PFU prototype for high heat flux test and (ii) a full-scale OVT prototype for assembly test as shown in **Figure 5**. MHI has made steady progress in that while utilizing the results of the previous verification, successfully completed the qualification, and is currently moving forward with series production.

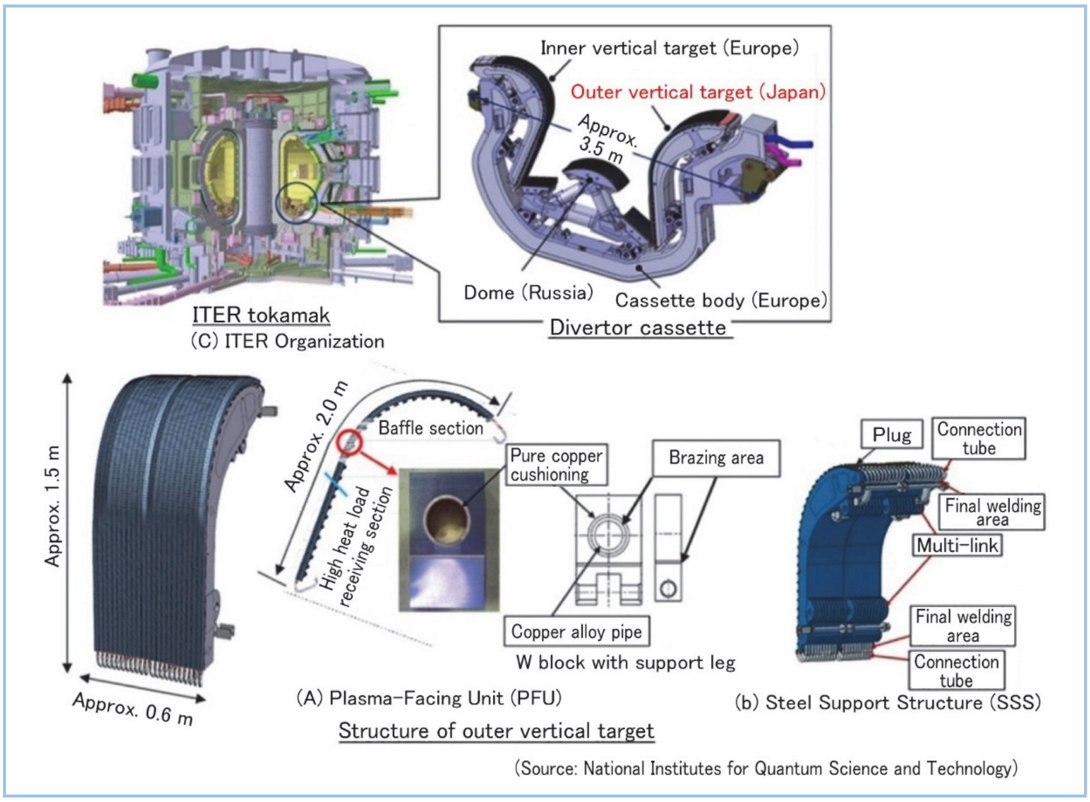


Figure 4 ITER divertor and OVT structure

Table 1 Major required specifications of ITER divertor

Main feature	Item	Specifications and structure
Heat removal	Heat resistance and plasma particle resistance	- Material: Tungsten block - Maximum surface heat load: 10 MW/m^2 (steady state), 20 MW/m^2 (peak) - Profile: 0.5 mm - Step between adjacent PFUs: 0.3 mm or less
	Cooling performance	- Tungsten block penetrated by cooling pipe - Cooling water: 4 MPa, 10 m/s, Inlet temperature 100°C, Outlet temperature 140°C - Brazed (metallurgical joint to ensure heat transfer) - Twisting tape (to promote turbulent heat transfer to improve marginal heat flux)
Strength	Pressure and electromagnetic force support heat elongation absorption	- Cooling pipe: CuCrZr pipe (precipitation hardening copper alloy) - Support leg joint strength: Withstands maximum load (9 kN) - Steel support structure: XM-19 (high-strength steel) - Multi-link, Piping curvature (flexibility)
Maintenance and repair	PFU alone detachability	- Support leg-pin-plug structure
Other	Radioactivation resistance	- Co-regulation

(i) PFU prototype for high heat flux test
(ii) Full-scale OVT prototype for assembly test

Figure 5 Appearance of divertor prototypes for ITER

(1) High heat load test using PFU prototype

The joining quality state of brazed areas affects the heat removal performance and joining strength. Therefore, it is required to conduct a non-destructive inspection using ultrasonic waves (ultrasonic test, hereinafter referred to as UT) on brazed surfaces to confirm that there are no significant defects, and to conduct a high heat load test simulating ITER operating conditions to confirm that the selected brazing conditions are suitable for use in the actual component. MHI has manufactured eight PFU prototypes for the high heat load test and they passed the high heat load test conducted by the ITER Organization in August 2023.

(2) Manufacturing of full-scale OVT prototype

An OVT consists of PFUs and SSS, and their complex shapes must be assembled and completed in accordance with the required dimensions. It was necessary to perform technological studies to verify a single PFU and SSS, followed by a final verification by manufacturing a full-scale prototype in the same process as the actual equipment to show that the required specifications were satisfied. MHI began manufacturing the OVT prototype for the final verification in 2020, and in March 2023, the prototype was confirmed to meet the required specifications and was successfully delivered to QST.

3. Challenges to fusion DEMO reactor

3.1 Fusion DEMO reactor

While working on the construction of ITER, an international cooperation project, MHI is also working on a fusion DEMO reactor that is being developed in Japan.

As a roadmap for the development of fusion DEMO reactors, the Ministry of Education, Culture, Sports, Science and Technology (hereinafter referred to as MEXT) reported “A Roadmap toward Fusion DEMO Reactor (first report)” on July 24, 2018, which summarizes the role of ITER and an action plan for the development of fusion DEMO reactors in Japan (4). The fusion DEMO reactor will play the role of the final runner toward the goal of its practical use, i.e., power generation, utilizing the results of the critical plasma test equipment JT-60 (QST Naka Institute), which achieved an energy multiplication factor $Q > 1$ (energy generation greater than the input energy), and the ITER experiment, which burned deuterium and tritium fuel at $Q = 10$ for over 400 seconds.

MHI has been conducting conceptual studies on the reactor structure and remote equipment of a fusion DEMO reactor, which will be a huge power plant, by utilizing its accumulated technologies on fusion and experience in power plants accumulated through its light water reactors, as well as its knowledge and experience in manufacturing gained through its contribution to JT-60 and ITER.

3.2 Development of blanket modules

The breeding blanket, an in-vessel structure, is a component that produces tritium used in plasma fusion reactions and extract fusion energy. **Figure 6** shows the blanket segment (hereinafter referred to as BS), which consists of breeding blanket modules and their supporting structure, i.e., the back plate. The target value of the tritium breeding ratio (hereinafter referred to as TBR), which is the number of tritium atoms produced per one neutron generated by fusion reactions, is set to be 1.05 or higher for the entire reactor, taking fuel cycle losses and other factors into consideration. To satisfy this target, the target TBR per module is set to 1.19 based on the ratio of the blanket-occupied area in the reactor. As part of its development of the breeding blanket module, aiming for a TBR of 1.19 or higher, MHI established a standard module structure as shown in **Figure 7** through rational structural studies, taking into consideration the manufacturability, structural integrity, and maintainability. This basic structural concept is a tube-plate structure with a welded hemispherical shell on the plasma-facing side, which is a much simpler cooling structure than the conventional structure with cooling pipes inside the module. In the future, MHI will further improve the TBR, the interface with the vacuum vessel will be specified and its integrity evaluated, and the manufacturability will be examined (manufacturing method and manufacturing accuracy) before moving on to the engineering design.

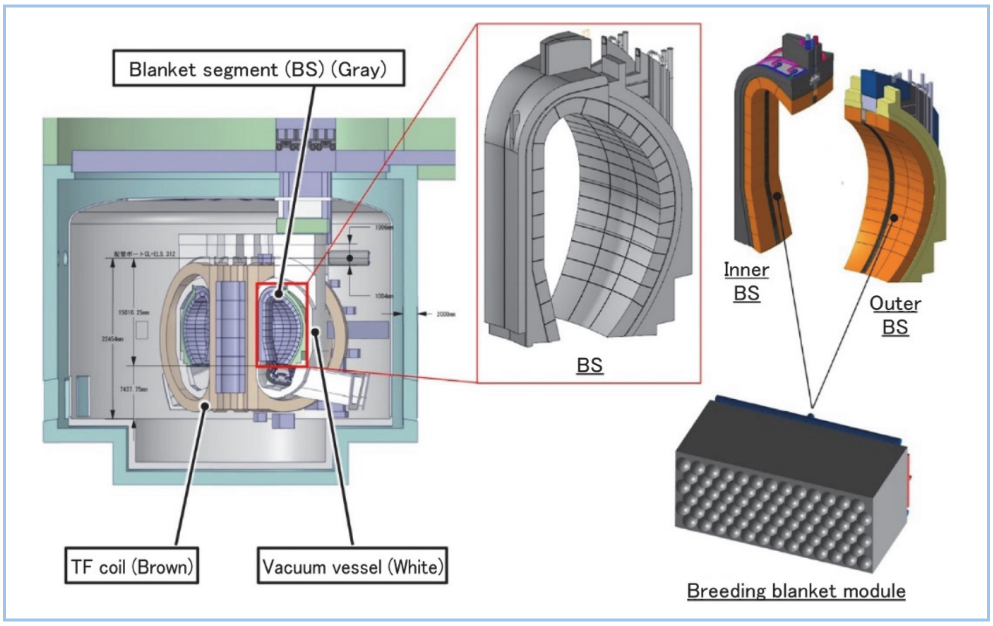


Figure 6　Configuration of blanket segment and breeding blanket modules

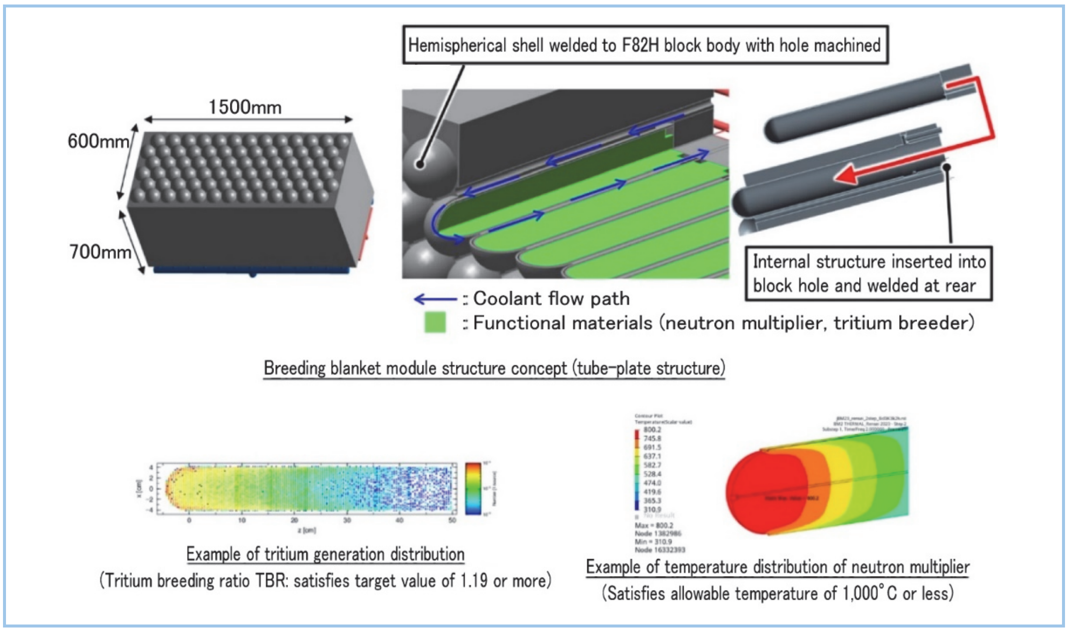


Figure 7　Blanket module for fusion DEMO reactor

3.3　Study of cask system for remote maintenance of blanket

Since the fusion reaction between deuterium and tritium produces high-energy neutrons, components that are heavily irradiated by neutrons, such as blankets, need to be replaced remotely, and it is very important to be able to replace them in the shortest possible time to obtain stable operation rates. Since the BS of the fusion DEMO reactor is a long and heavy structure weighing more than 100 tons, its remote maintenance is very difficult and its replacement takes a long time. Therefore, the maintenance procedure for the BS was planned so as to remove it as a large segment from the upper port of the tokamak using remote maintenance equipment as shown in **Figure 8**, transport it from the tokamak building to the hot cell (repair facility for in-vessel components) using a cask, which is a transport container, and then replace the blanket at the hot cell. The hot cell is required to be divided into various areas such as decontamination/storage area, replacement/repair area and assembly/adjustment/inspection area, and to have sufficient space for transportation.

In the studies so far, based on the maintenance procedure proposal for BS removal, a rotating bridge system has been adopted and its design conditions and equipment configuration have been organized to make a conceptual design. Going forward, we will conduct conceptual design of the cask body, rails, etc., and are aiming to develop concrete plans for the entire cask system for further application.

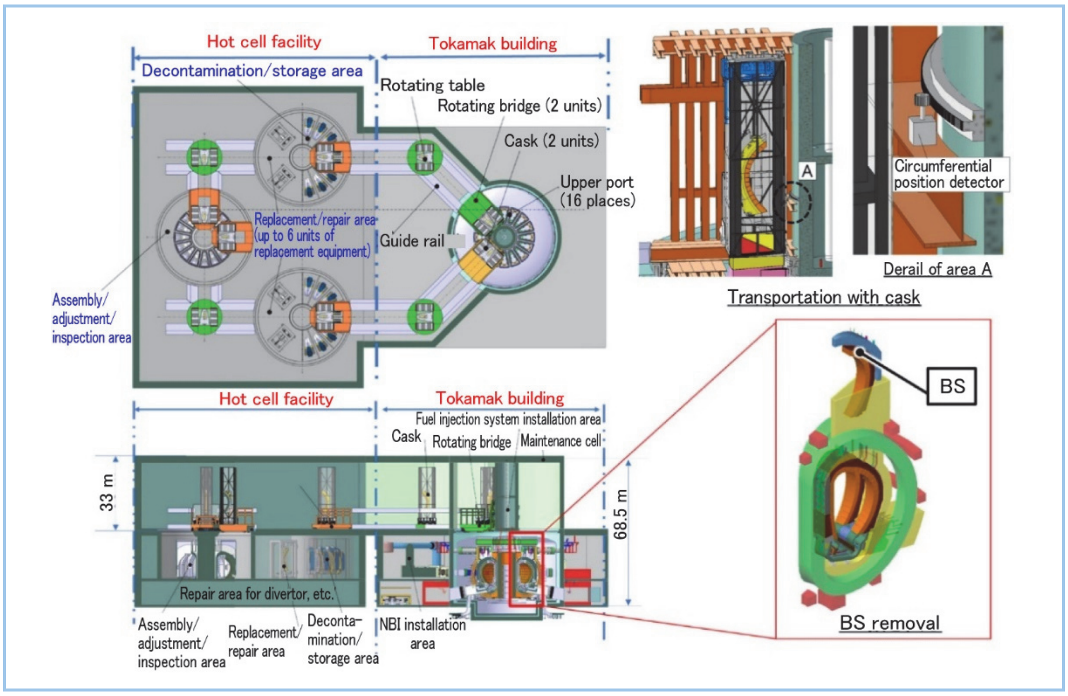


Figure 8　Cask system for remote maintenance of blanket for fusion DEMO reactor

4. Conclusion

MHI was in charge of manufacturing 5 TF coils, which are some of the world's largest superconducting coils, for ITER, and all of them were shipped in August 2023. As for the divertor, which is a key component of ITER as well as the TF coils, MHI has completed the acquisition of qualification for PFU joining and the manufacturing of a full-scale prototype, after eliminating technological risks through R&D and in-house research. Based on these findings, MHI is now moving forward with series production of the actual component.

As for the fusion DEMO reactor, MHI is currently in the conceptual design phase, and will conduct design with manufacturing in mind and move on to the next stage, engineering design. This is a long-term project, and therefore, it is also necessary to hand down, maintain, and improve the design and manufacturing technologies that serve as the foundation. As technological innovations in design using CAD/CAE/CAM and manufacturing using AI become more widespread, MHI will establish methods to integrate individual technologies and experience so that rational manufacturing can be carried out.

MHI will continue to apply the experience and technology gained through the manufacturing of TF coils and divertors for ITER, as well as its knowledge of light water reactors, to the design of the fusion DEMO reactor to proceed with development toward the practical use of fusion reactors.

References

(1) MEXT Science and Technology Committee on Fusion Energy, “Progress of ITER Project” July 10, 2024 (in Japanese)

(2) K. 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 (2015)

(3) K. Shimizu et al., MHI’s Continuous Challenge to realize Fusion Energy - Efforts for International Projects of ITER and Broader Approach (BA) -, Mitsubishi Heavy Industries Technical Review, Vol.57 No.4 (2020)

(4) MEXT Science and Technology Committee on Fusion Energy Report, “A Roadmap toward Fusion DEMO Reactor (first report), July 24, 2018”