Overview of Large-scale Technology R&D’s for ITER
- Engineering Challenges –

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Variety of technology R&D’s have been performed to confirm the technical feasibility of ITER construction. Special efforts have been made to demonstrate fabricability and performance of ITER critical components such as superconducting magnets, the vacuum vessel, blanket modules, divertor cassettes, and remote handling system. In order to develop these components, seven large R&D projects have been planned and collaboratively carried out by the ITER parties, EU, Japan, Russia and US. These projects include manufacturing the scalable or real-scale components and testing of the performance. The central solenoid (CS) coil project was really a cross-party venture. The size of the test coil is 3.6 m in diameter and 2 m in height, which is almost equivalent to the size of the one module of the ITER CS coil that is composed of 6 modules. The maximum field of 13 T was successfully achieved at a current of 46 kA with a stored energy of 640 MJ. Pulsed operations with a ramp-up rate of 0.6 T/s and ramp-down rate of 1.2 T/s were also successfully carried out. The results demonstrated the fabricability and performance of the ITER CS coil. The full-scale sector model of the vacuum vessel, which has a D-shape of 15m high and 9m wide, was fabricated within an allowable deformation. The blanket modules were fabricated using HIP and powder HIP technologies. The divertor components were developed in EU, JA and RF, achieving the performance of surviving a heat flux of 20 MW/m2 for 2000 cycles. The results proved the performance and fabricability of the divertor cassettes. Remote handling is a key technology to realize high availability operation of ITER. The blanket module of 4 tons should be handled in a narrow space with a high gamma-ray field of around 10^4 Gy/h. The vehicle-type manipulator has been developed for quick replace of the heavy component with a narrow gap between adjacent modules. These projects have successfully demonstrated the feasibility of ITER construction and attainability of its expected performance. The result of these activities has also convinced us to proceed to collaboration for ITER construction.

KEYWORDS: Fusion, ITER, technology R&D, superconducting magnet, blanket, divertor, remote handling

I. Introduction

ITER is a next-step fusion facility, which realize Deuterium-Tritium burning plasmas. One of the objectives of ITER is to achieve extended burn in an inductive operation with an energy gain of greater than 10, for a burn duration between 300 and 500 s. An average neutron wall loading is larger than 0.5 MW/m² and a fluence of greater than 0.3 MWA/m². The key components of ITER are the superconducting magnets, vacuum vessel, blanket, divertor, cryostat, fuel circulation system, plasma heating and current drive system, diagnostic and control system and remote handling system. Required performance of these components are so demanding that extensive technology R&D’s have been planned and performed during the ITER Engineering Design Activities. Figure 1 summarizes 7 large technology R&D’s. Some components such as plasma heating and current drive system are quite unique for fusion facilities, but some components such as blanket and divertor share common issues with advanced fission reactors in terms of high heat flux, neutron irradiation and advanced materials.

In this paper, large-scale technology R&D’s for ITER are overviewed with special emphasis being placed on “what are the issues/objectives?” and “what new technologies have been developed?”

II. Central Solenoid Model Coil

1. Objectives

The objectives of the Central Solenoid Model Coil (CSMC) Program are to develop the fabrication technology of the superconducting coil and to demonstrate the validity of the design by experiments with the CSMC and the CS Insert Coil (CSIC). The size of the test coil is 3.6 m in diameter and 2 m in height, which is almost equivalent to the size of the one module of the ITER CS coil that is composed of 6 modules. The size and the required performance are far beyond the experience and technology being obtained until then. The purpose of the testing of the CSMC and CSIC are:

(1) to demonstrate a reliable cool-down to 4K without any damage to insulators in 480 hours

(2) to confirm DC performance of the CS Model Coil by charging up to 13 T, 46 kA with a stored energy of 640 MJ and of the CS Insert Coil up to 13T, 40 kA.

(3) to achieve pulsed charging performance by a ramp rate of +0.4 T/s to 13T and discharging performance by a ramp rate of −1.2 T/s from 13 T for both of the coils

(4) to measure operational temperature margin for both of the coils

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to perform and demonstrate a cyclic charging of the CS Insert for 10,000 times.

Other important objectives of the CS Model Coil Program are to carry out the fabrication of the superconducting strand, cabling of the strand, Incoloy jacket fabrication, jacketing of the cable, winding, activation heat treatment, insulation and conductor joint fabrication.

2. CS Model Coil Fabrication

The CSMC is composed of an Inner Module with 10 layers and an Outer Module with 8 layers as shown in Fig. 2. In addition, a single-layer CS Insert Coil is also developed and fabricated for the detailed study on the coil performance. The CS Insert Coil is placed in the bore of the CSMC. The CSMC does not have any sensors in the winding from the viewpoint of an electric insulation. Instead, the CS Insert Coil is well instrumented in order to obtain important information on the CS conductor under the operation condition of 13 T and 40 kA. The temperature sensors, for example, measure the conductor temperatures directly. Ten voltage taps are placed in the 13 T area to observe a possible appearance of normal voltage zones and their propagation. The pressure tap measures the internal pressure at the center of the CS Insert Coil.

The fabrication was carried out by the international collaboration as shown in Fig. 3. The inner module was fabricated in the US, while the outer module was made in Japan. Because the ITER CS coil employs Nb3Sn as superconducting material, the coil must be fabricated by the wind-and-react method, in which the coil is firstly made by winding the conductor and then the coil is heat treated to produce the superconducting Nb3Sn in the conductor.

Formation of Nb3Sn compound requires the reaction heat treatment of 240 hours at a temperature of 650°C. Because the conductor jacket is also subjected to this heating condition, materials for the jacket need to maintain the strength against the treatment condition and to have nearly the same thermal expansion coefficient as that of Nb3Sn.
This is particularly important to minimize the resultant stress generated in the Nb₃Sn associated with the large temperature difference between the heat-treatment temperature of 650 °C and the coil working temperature of around –269 °C.

Incoloy 908 satisfies the above requirement and is selected as the jacket material. However, Incoloy 908 is known to have a potential problem of cracking due to stress accelerated grain boundary oxidation (SAGBO).

SAGBO occurs when the following three conditions are satisfied simultaneously.

- Temperature: from around 550°C to 800°C
- Tensile stress: more than 200 MPa
- Oxygen concentration: more than 0.1 ppm in the atmosphere, or partial pressure larger than 1.33 x 10⁻² Pa in the vacuum environment.

Careful pre-treatments, such as removal of organic compounds and minimization of residual tensile stress, have been applied to overcome SAGBO.

All conductors for the CS Model Coil were successfully heat-treated both in US and in Japan to produce high performance Nb₃Sn superconductors without any SAGBO cracks.

The inner and outer module were assembled and installed in the Test Facility at JAERI.

3. Testing of the CS Model Coil and CS Insert Coil

The test facility is composed of a vacuum system, power supply system, cryogenic system and data acquisition system. The vacuum system consists of a tank of 9.1 m in height and 6.5 m in diameter in which the coils are installed. The cryogenic system has a refrigeration capacity of 5 kW at 4.5 K or liquid helium liquefaction power of 800 liters/hr and a supercritical He cryogenic circular pump of 500 g/s.

The CSMC successfully generates a magnetic field of 13T at a current of 46 kA with a stored energy of 640 MJ. The inlet supercritical He temperature is 4.5 K.

In pulsed operations, a ramp-up rate of 0.4 T/s in charging up to 13T and a discharging rate of -1.2 T/s from 13 T are achieved. These rapid rate charging and discharging performances well satisfy ITER requirements of generating plasma current up to 15 MA and of controlling plasma current and position. These achieved values such as 46 kA with 13 T and 640 MJ and ramp-up and ramp-down times of 30 s and 11 s are factors of 2-20 times higher than the previous records.

Under the background field produced by the CS Model Coil, charging and discharging between (0 kA, 13T) and (40 kA, 13 T) of the CS Insert Coil were repeated for 10,003 times. In this test, the cyclic operation required for the ITER CS coil was simulated. The current sharing temperature was measured at 13T and 40 kA before and after 100, 200, 500, 1000, 2000, 5000 and 10,000 cycles. Decrease of Tcs from 7.65 K to 7.20 K is observed at the end of these tests. Careful investigation of the data concludes that the decrease is attributable to the unexpected temperature rise in the Nb₃Sn filaments associated with artificial quench tests and not to degradation due to cyclic operation under the normal condition.

4. Summary

The CS Model Coil and the CS Insert Coil were successfully designed, developed and fabricated by the international collaboration of the EU, Japan, Russia and the US.

The cool-down was carried out successfully in March 2000 and all the coil modules showed a clear transition to superconducting state at 17.5K. The charging experiments of the CS Model Coil and the CS Insert Coil were carried out and the excellent performances are achieved, fulfilling all the goals of the technology development of the CS Model Coil.

Figure 4 shows the break through of a new technology obtained by this project compared to the former technology frontier of the superconducting magnets that have ever been built. Compared to the record held by the US Demo. Poloidal Coil, the CS Model Coil increases the maximum field by 1.85 times, from 7 T to 13 T, and achieves a stored energy of 640 MJ, which is 21 times as much as that of before.

By the achievements obtained through the CS Model Coil Program, the fabrication technology of the ITER CS coil is developed and its engineering design is validated. The size of the CS model coil is almost same as that of the one module of ITER CS coil, and we are now ready to initiate construction of the ITER CS coil with confidence.

III. Vacuum Vessel

The ITER vacuum vessel is a double-walled structure made of 316LN stainless steel. The cross section is D-shape with a size of 9 m wide and 15 m high. The vacuum vessel is toroidally divided into 20 sectors, which are joined by field welding in the initial assembly. Because of its large scale and high dimensional accuracy, fabrication and testing of a full-scale sector model are performed by international collaboration. The major technical objectives of the program are:

1) to develop and demonstrate the fabrication technologies for double-walled ITER vacuum vessel,
2) to perform the demonstration test of field joint welding
between sectors which is required in the initial assembly.
The program was performed internationally with the joint
effort of the ITER JCT, Japanese, RF and US Home Teams.

The full-scale sector model corresponds to an 18-degree
toroidal sector is composed of two 9-degree sectors, Sector-A
and B as seen in Fig. 5, which are spliced and welded. Two
welding approaches were applied in the fabrication. One is
Tungsten Inert Gas (TIG)/ Electron Beam(EB) welding and
the other is TIG/ Metal Active Gas (MAG) welding. Both
approaches successfully produced the sectors with a
dimensional accuracy to within 3 mm at the field joint edge.
The results have demonstrated the fabrication technology to
be applicable to the ITER vacuum vessel.

The two sectors were shipped to the test site in JAERI for
the final assembly and integration test. The final assembly
includes the field joint welding of the two sectors, which
simulates the initial assembly of ITER vacuum vessel in the
cryostat. The most important test in the field joint was the
demonstration of automatic Narrow-Gap (NG) -TIG welding
with splice plates and non-destructive inspection of the joint
by Penetrant Testing (PT) and Ultrasonic Testing (UT)
method. Figure 6 shows the automatic NG-TIG welding
system used for the field joint welding between outer shells.

No unacceptable weld flaws were detected by PT and UT
inspection. After the completion of assembly, the supporting
structures were removed from the inner bore of two sectors
for the measurement of final dimension and mechanical
behavior. The dimensional changes due to final assembly are
+5.0 mm in horizontal and -1.0 mm in vertical direction.
Based on these results, it has been demonstrated that the
dimensional accuracy of current ITER design, 20 mm in
total width and total height, is attainable by the developed
fabrication and assembly technologies.

The objectives of the Vacuum Vessel Sector Model R&D
have been successfully achieved. Sufficient technical data on
the fabrication and the initial assembly for the ITER vacuum
vessel have been obtained, which verify the feasibility of
current ITER design.

IV. Blanket Modules

The ITER Blanket is composed of about 400
box-shaped modules, each of which has a size of about 1.5m
in width, 1 m in height and 0.5m in thickness. The module
consists of a 0.4 m thick shield block, to which four first
wall panels are attached. The first wall panel receives a
normal surface heat flux of 0.2 MW/m² with a local peak
value of 0.5 MW/m². In off-normal conditions, the panel
has to withstand a heat flux of 3 MW/m² for a period of 1
sec. These modules are attached to the vacuum vessel with
bolts. The main objectives of the blanket module R&D are:
(1) to develop fabrication technologies for the blanket
system,
(2) to demonstrate the manufacturing feasibility by
fabricating prototypical components,
(3) to demonstrate operational performance of the blanket
system.

A partial first wall mock-up (100 mmW x 580 mmH x
85 mmT) was fabricated by bonding a Dispersion Strengthened Cu (DSCu) heat sink to the SS substrate by hot isostatic pressing (HIP) as shown in Fig. 7. The HIP condition is 1050 °C, 150 MPa and 2 hours. The bonded structure was found to withstand a high heat flux of 5 MW/m² which repeatedly applied 1000 times to the heat sink surface.

Increased HIP pressure of 200 MPa improved the impact strength of the bonds by a factor of 2 for the bonds with bonded surface roughness of 12.5 micrometers, compared to that of the bonds with bonded surface roughness of 1.6 micrometers hipped at 150 MPa.

The fabrication technology has been well-established. The performance test on the prototypical components has demonstrated the sufficient performance for use in ITER conditions. The objectives of the program are satisfied.

V. Divertor and Plasma Facing Components

The divertor and plasma facing components of ITER are expected to receive a steady state high heat flux of up to around 20 MW/m² together with high flux of energetic particles and neutrons. The plasma facing components are joint structures that consist of armor materials bonded to heat sink materials. The major objectives of the program are: (1) to demonstrate the feasibility to fabricate PFC’s capable of achieving the high heat flux and lifetime requirements, (2) to develop and demonstrate reliable joining technologies.

In the divertor, C, Carbon-Fiber-Composites (CFC) and W are used as the plasma facing armor material and copper alloys are for the heat sink material. Joining of these materials has been intensively studied.

Joining of W to Cu alloy has been successfully developed using soft-Cu as a compliant material. Other methods such as casting Cu on W and plasma sprayed W on Cu have also provided successful results.

CFC/Cu joints have been fabricated with a compliant layer by brazing with silver-free alloys such as CuSiAlTi.

The progress of these joining technologies enables to fabricate large-scale divertor mock-ups as seen in Fig. 8. These mock-ups have been subjected to high heat flux testing and demonstrated to withstand a heat load of 5 MW/m², 30 s for 3000 cycles, and 20 MW/m², 10 s for 1000 cycles. After the demonstration of large-scale mock-ups being able to withstand the ITER-relevant heat loads, major efforts have been concentrated to develop technologies for increasing reliability and saving costs. From these points of view, an annular swirl tube has been developed.

The annular swirl tube is fabricated and tested to study the critical heat flux in an ion beam test facility in JAERI. The outer tube is made of CuCrZr with the outer diameter of 21 mm, and the inner diameter of 18 mm. The inner tube is made of stainless steel with the inner diameter of 9 mm. Swirl fins are machined on the outer surface of the inner tube. Water flows through the inner tube, returns at the end plug, and flows back in the annular section along the swirl fins. One of the key questions of this tube is “Is the critical heat flux as high as that of the circular swirl tube?” Because the inner tube is just inserted in the outer tube, small clearance exists between the outer tube and the fins, which may degrade the critical heat flux due to the leak flow through the clearance.

Results of the CHF test are shown in Fig. 9. In the tests, a normal swirl tube having the same dimension as the outer tube, is also tested for comparison. The heat flux is defined
by the incident heat flux on the outer surface. The coolant pressure is kept to be 1 MPa during the tests. It is confirmed that the incident critical heat flux of the annular swirl tube is almost the same as that of the swirl tube. The annular swirl tube has successfully demonstrated that it is applicable to the ITER divertor.

In summary, the progress has demonstrated the feasibility of fabricating the divertor that can withstand the ITER heat load conditions.

VI. Remote Handling of Blanket Modules
Tokamak components installed in the vacuum vessel are called in-vessel components, which include the blanket and divertor. Because the in-vessel components are activated by the intense neutrons produced in the fusion plasma, their replacement and maintenance must be made remotely. This requires the development of the remote handling system.

The system has to handle the blanket modules with a dead weight of up to about 4 t within a ±0.25 mm positional accuracy in the constrained in-vessel space and under intense gamma radiation of ~ 1 kGy/h. This is really a challenging job and demonstration of the system needs to be performed.

A concept of rail-mounted vehicle manipulator system has been developed to meet these requirements and tested in the blanket test platform as seen in Fig. 10.

The critical issue of the vehicle manipulator system for blanket maintenance is the feasibility of the deployment and storage of the articulated rail composed of simple structures without any driving mechanism in the joints. The new driving mechanism, procedure and control scheme for the rail deployment and storage were proposed based on the requirement of the simplified system to increase the reliability. The proposal was then tested using a full-scale vehicle manipulator system in order to demonstrate the feasibility of the rail deployment and storage. As a result, the respective rail links of the articulated rail have been successfully deployed and stored within 6 hours by means of the synchronized control of the driving mechanisms. In addition, a position control with torque-limit control was effective in order to suppress the overload of the driving mechanisms and to continue the reliable operation. It is therefore concluded that the feasibility of the rail deployment and storage of the articulated rail composed of simple structures without any driving mechanism in the joints has been demonstrated by the proof of principle tests using the full-scale vehicle manipulator system.

VII Conclusion
ITER Engineering Design Activities have been carried out to establish the engineering design of ITER and to perform technology R&D’s for demonstration and confirmation of the technical feasibility of ITER construction.

This paper overviews the typical technology R&D’s for ITER. The technical challenges can be summarized as follows.

- large size : superconducting magnet, vacuum vessel
- high heat flux : divertor
- remote handling of heavy components with high positional accuracy : remote handling system
- various joint and bonded materials : blanket, divertor
- neutron irradiation : blanket, divertor, remote handling system

Through the technology R&D’s, the fabrication technologies have been developed and the engineering design has completed. ITER is expected to start construction in the near future, and further efforts for better, economical and simple solutions to cope with these challenging issues should be pursued.

Reference