Overview of ITER Korea Procurement Activities

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Abstract. ITER Korea was established in 2007 as the Korean Domestic Agency (KODA) entity that is to execute all responsible activities with respect to the ITER Project in Korea. Its main work is designing, manufacturing and delivering on time the Korean ITER contributions to the ITER Organization (IO) complying with procurement specifications and its quality assurance program. Procurement activities are progressing on schedule, and some design and R&D activities are being performed to meet technical specifications for the in-kind procurement packages. This paper will describe the design and technical activities that are currently being conducted in the KODA.

1. Introduction

The main mission of ITER Korea is to design, manufacture and deliver on time the Korean ITER in-kind contributions to the ITER Organization (IO) complying with technical specifications and quality requirements. [1] Since 2007, ITER Korea has performed various procurement activities including engineering design and technical R&D for provision of the necessary technical support and risk containment of the KO procurement packages. As a result, significant progress has been achieved enabling the required schedule and technical specifications to be met. Currently, procurement activities at the KO-DA are soundly progressing in accordance with the project schedule and milestones. In addition to procurement activities, Korea is contributing significantly to outstanding ITER design and R&D through more than 30 Task Agreements between the IO and KO-DA taking advantage of KSTAR experience and nuclear power plant technology, in various areas of the ITER system such as the Hot Cell, EPICS for CODAC, Assembly, Cryogenic system testing at KSTAR, etc. Through the ITER Korea project, the Korean fusion community will eventually have a chance to build industrial infrastructure and human resources for developing commercial fusion power in the future. This paper will describe the design and technical activities that are currently being conducted at the KODA.

2. Status of In-kind Procurement Arrangements

The procurement arrangement (PA) of the ITER toroidal field (TF) magnet conductors, which was the first, was signed between the IO and the ITER Korea in May 2008. The second signing of a PA, that for two sectors of the ITER main vacuum vessel (VV), equatorial ports and lower ports, was in November 2008. In August 2009 and May 2010, the PAs for dedicated assembly tooling and the thermal shield for the ITER tokamak were also signed, respectively. Currently, preparatory work for PAs on the other procurement items such as the blanket shield block, tritium plant storage and delivery system, pulsed power supply AC/DC converters, and selected diagnostics are being carried in Korea.

3. Progress of Procurement Activities

3.1. TF Conductors

The ITER TF magnet system consists of 18 TF coils which provide a magnetic field of 5.3 T at the plasma center with a peak flux density of 11.8 T at the coils. The coils make use of Nb3Sn Cable-In-Conduit Conductor (CICC) with 316LN-IG-HT stainless steel circular conduit, as shown in FIG. 1. The nominal current is 68 kA and the stored energy is 41 GJ. One TF coil uses seven CICC segments, five of which are 760 m long and two which are 415 m. ITER Korea will provide nineteen long CICC segments and eight short CICC segments. Korea has developed an internal-tin process for the manufacturing of Nb3Sn strand. With Cr-plating, the strand diameter is 0.82 mm, and its critical current is greater than 250 A at 4.2 K with 12 T field and -0.25% strain. Hysteresis loss is less than 600 mJ/cc at field variations from -3 T to 3 T at 4.2 K and the residual resistivity ratio (RRR) is greater than 100.

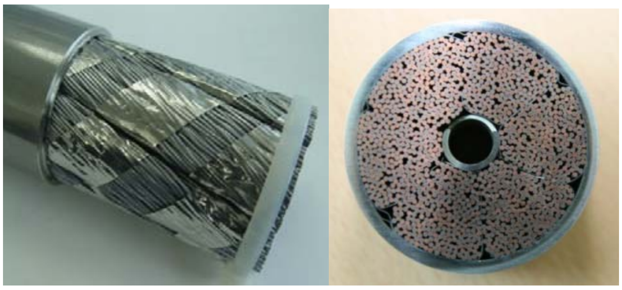

FIG. 1. Dummy CICC Sample.

Since the signing of the TF conductor PA in 2008, the procurement qualification test of a Conductor Performance Qualification Sample (CPQS) has taken place on 5 November 2008 at the SULTAN facility in Villigen, Switzerland, whose results show current sharing temperatures (Tcs) of 6.3~6.6 K both exceeding the ITER requirement of 5.7 K. Based on the results, ITER Korea signed contract awards with industries for superconductor strand and cabling, respectively in 2009. Until now, 18 ton of superconducting strand have been produced, and a 760 m long dummy cable and 100 m superconducting cable were manufactured. Recently, as requested by IO, a CPQS test of manufactured TF conductor was carried out on 7 September 2010. The results (Right & Left) are better than the procurement qualification test (KOTF2) of November 2008, as shown in FIG. 2.

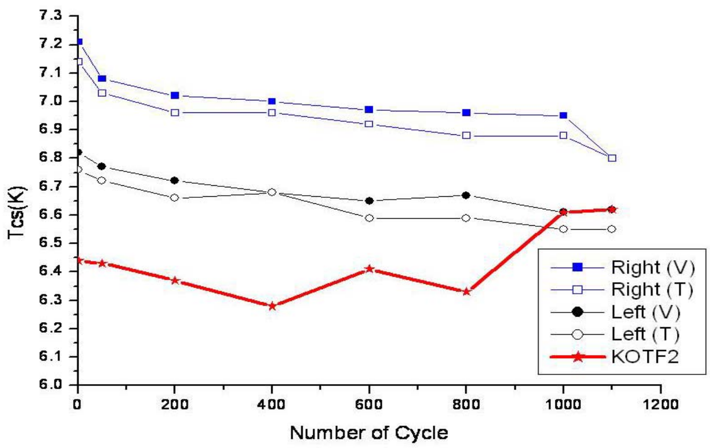

FIG. 2. Conductor Performance Qualification Sample test results at SULTAN.

3.2. Vacuum Vessel Main and Ports

ITER Korea will procure two ITER vacuum vessel (VV) main vessel sectors, along with the equatorial and lower ports including vacuum vessel supports and neutral beam liner. After the

PA signing with IO in November 2008, the KODA proceeded to make a contract with Hyundai Heavy Industries (HHI) in January 2010. With the contract, HHI started to make a partial full scale mock-up to develop their fabrication procedures. [2] The first mock-up is used to develop and stabilize electron beam welding (EBW) techniques. The lower parts of the inboard segment are made for optimization of EBW techniques and development of repair and nondestructive examination (NDE) methods. The second mock-up is a 20 degree upper segment. This is used for verification of forming, development of machining techniques, optimization of welding sequences, and control of distortion. From this R&D, the applicability of the NG-TIG welding method, NDE procedures, and dimension inspection method to production will be verified and the results will be fully incorporated in the final manufacturing process.

One of the major technical difficulties in the fabrication of an ITER main vessel sector is the demanding tight tolerances. The optimum welding sequence and restriction jigs are crucial to meeting the tolerance requirements. To predict the welding deformation and optimize the welding sequence, dedicated analysis is conducted. The applicability of this analysis can be shown in the mock-up of the 20-degree upper segment. The design thickness of the inner and outer shell of the main vessel sector is 60 mm. Rough estimation of plate thickness reduction during the forming process is about 5 - 9 %. To ensure that this reduction does not affect structural integrity, the impact of shell thickness reduction on VV structural analysis is being assessed. Based on this structural assessment, the plate material thickness for the start of fabrication will be determined. In addition, forming analysis is also performed for forming die design and forming method selection. Another difficulty in fabrication is the large amount of single side circular welding between the inner and outer shell to attach the flexible support housing in the box structure. RCC-MR 2007 is the applicable code for the ITER VV, and it requires 100 % volumetric weld inspection. The various welding and NDE coupons are made and tested to validate the procedures. In order to deliver the first sector (#6) to the ITER site on time, the start of real fabrication is scheduled for early 2011.

3.3. Blanket First Wall and Shield Block

The role of the ITER blanket system is to shield and absorb the radiation from the plasma to the vacuum vessel and superconducting magnets. This requires tolerance to extreme heat and radiation by the blanket which lies between the plasma and the vacuum vessel. The blanket first wall is the plasma facing component which is in direct contact with the plasma. The current design of the first wall involves a beryllium outer layer for shielding, a copper alloy layer to efficiently remove the heat generated by the plasma and a stainless steel structure.

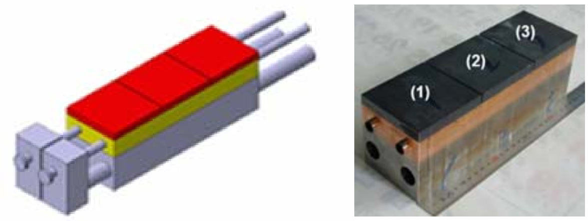

FIG. 3. Blanket FW mock-up design and fabrication for prequalification tests.

The qualification program was agreed by the six procuring domestic agencies in order to guarantee the acceptable quality of ITER blanket first wall modules. In accordance with the manufacturing specifications given by the IO, the first wall mock-up consists of three beryllium armor tiles attached to a Cu alloy (CuCrZr) heat sink plate internally cooled by a SS cooling tube, as shown in FIG. 3. Following the mock-up test plan, the KODA sent two mock-ups to high-heat flux test facilities in the US and EU. Both mock-ups passed their tests and the results were approved by the IO in 2009. [3]

FIG. 4. Blanket shield block mock-ups.

The ITER blanket shield block which consists of a stainless steel 316L(N)-IG structure lying between the blanket first wall and the vacuum vessel insulates the vessel from the heat and neutron radiation generated by the plasma. As part of blanket design tasks, the KODA has participated in blanket design and analysis including electromagnetic, thermo-hydraulic and thermo-mechanical calculations. [4] To verify the manufacturing adequacy of the blanket shield block using conventional methods and to establish the fabrication process, several welding tests such as EBW for thick SS blocks, joining of lid or plug, and attachment of flow drivers, were performed based on the reference design. The single sided EBW technique for 70 and 110 mm thickness blocks were successfully developed by Korean industries, as shown in FIG. 4. EBW parameters to satisfy relevant code requirements and standards were found, and appropriate welding conditions to suppress the growth of voids and porosities in fusion metal were also found. For the weld design of front header lids and plugs, small mock-ups using EBW or narrow gap welding were manufactured. These developments in manufacturing techniques are contributing to the evolution of the blanket design.

3.4. Thermal Shield

The ITER thermal shield (TS) plays the role of reducing the heat load transferred by thermal radiation and conduction from warm components to the magnet structures that operate at 4.5 K. The TS consists of equatorial TS (ETS), upper TS and lower TS. The ETS has three sub-components which are the vacuum vessel TS, equatorial cryostat TS, and support TS. Thermal radiation to the magnet structures is minimized by operating the TS at low temperature and by providing TS surfaces with low emissivity using silver coating. The ITER TS is procured completely by Korea. So, the KODA has been studying the TS design in collaboration with industries since 2007. [5] The conceptual design review (CDR) and preliminary design review (PDR) of the TS were held in July 2009 and in January 2010, respectively. For resolving an issue on the silver coating technology which was one of the most critical chits of the CDR, the KODA was able to suggest a reasonable silver coating method after a series of R&D.

The ITER TS is a very large system and its tolerance requirement is so tight that the precise design and manufacture are indispensable. Therefore, the critical components of the TS should be tested before manufacturing the TS to mitigate possible risks. In 2009, ITER Korea had performed the fabrication and testing of several mock-ups for the ITER TS with support from industry, as shown in FIG. 5.

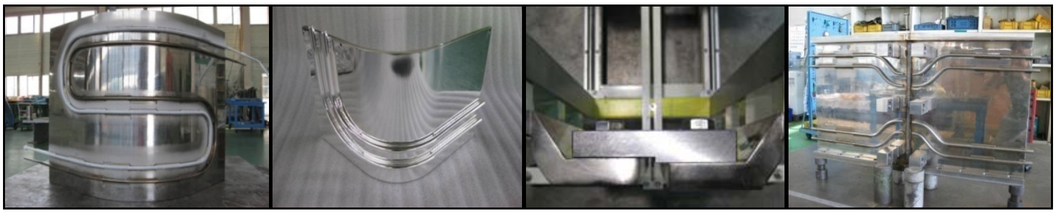

FIG. 5. Mock-up fabrication and tests of the ITER thermal shield.

The TS R&D activities were i) fabrication and tests of panels with cooling tube, ii) fabrication and tests of the silver coating, iii) assembly tests of in-pit joints, and iv) bisecting tests of the VVTS inboard. For the testing of the panels with cooling tubes, welding parameters such as shield gas, welder level, welding rod diameter and weld speed were tested. Staggered (zigzag pattern) welding was performed for eight specimens and their welding quality was carefully examined by cutting the cross section using wire-cutting. Then, thermal cycling (shock) tests were carried out by flowing liquid nitrogen in the cooling tube. No leaks were found after the repeated cool-down and warm-up of the specimens.

The feasibility of low emissivity coating for the TS was studied. Two kinds of mock-ups for the TS were prepared and bath silver electroplating was performed to observe coating defects. Surface emissivity and coating thickness were measured for small specimens. Based on this R&D of several alternative methods of silver coating, the KODA suggested a design change, bisectional method, for cost effective and reliable silver coating. The bisectional method consists of cutting the VVTS in two parts and applying silver electroplating to each sectioned part. The motivation for this approach is to reduce the bath size for the silver coating making it more feasible because the design change is not drastic and its manufacturability is good. The mock-up test was performed for the VVTS in-pit joint proposed by the KODA. The assembly environment around the in-pit joint was prepared and the assembly test was successfully performed using a 3D reverse engineering method.

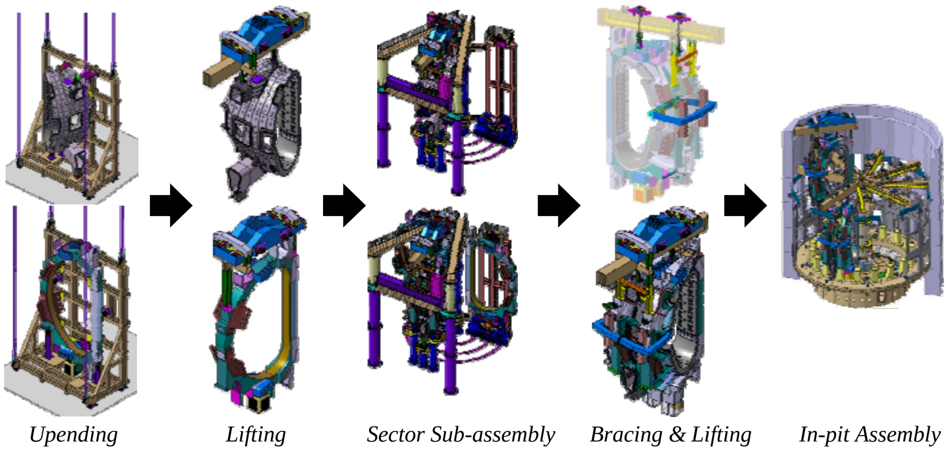

FIG. 6. Overall configuration of the sector sub-assembly and in-pit tools.

3.5. Assembly Tooling

The ITER tokamak assembly tools are purpose-built tools to complete the ITER tokamak machine which includes the cryostat and the components contained therein. [6] Based on the design description document prepared by the IO, ITER Korea has carried out the conceptual design of the assembly tools. The ITER sector assembly tools are classified into 2 groups.

One group is the sub-assembly tools including upending tool, lifting tool, sub-assembly tool, VV supports and bracing tools used in the assembly hall and the other group is the in-pit assembly tools that include lifting tools, radial beams, central column and supports, which are shown in FIG. 6. The conceptual designs of the sector sub-assembly tools including upending tool, sector sub-assembly tool and lifting tool have been developed. The design of the sector sub-assembly tools and in-pit assembly tools developed by the KODA satisfied ITER assembly plan and technical requirements. Sufficiently maintaining structural stability to secure assembly tolerances and geometrical requirements requested by IO were verified. Work on the detail design continues for the ITER assembly tools until Oct. 2014.

3.6. Tritium Storage and Delivery System (SDS)

The storage and delivery system of the tritium plant in ITER mainly consists of the fuel storage and delivery system, and tritium loading station. The main function of the fuel storage and delivery system is the delivery of fuel and the safe storage of the tritium-contaminated fuel gases. The main function of the tritium loading station is to handle the tritium transport package and to transfer the tritium gas from the tritium transport package to the fuel storage and delivery system.

Key technologies for tritium SDS procurement are as follows: tritium fuel cycle and plant system, tritium storage bed design and fabrication, tritium accountability, and process design and operation. Currently, ITER Korea is conducting R&D jointly with the Korea Atomic Energy Research Institute (KAERI) and the Korea Electric Power Research Institute (KEPRI). A full scale mock-up of the SDS getter bed was designed and fabricated, and its performance has been tested. In addition, individual subsystems are being tested to ensure performance efficiency, process design and operating scenarios in order to verify the SDS unit process feasibility. As a result of the R&D, an apparatus for simulating a tritium storage container was developed, and a flow-diagram, P&ID, and selection and tabulation of major components have progressed. KAERI is researching storage material characteristics and developing the getter bed, and KEPRI is performing the design of the tritium calorimeter.

3.7. AC/DC Converters

The AC/DC converters for the pulsed power supply consist of a rectifier transformer and 12 pulse thyristor stacks configured in parallel and/or series with inter-phase DC reactors. In preparation for procurement, ITER Korea has been reviewing fundamental concepts and requirements of the AC/DC converters based on FDR2001 in line with the recommendations from the 2007 ITER Design Review, in conjunction with the Chinese Domestic Agency and the IO under the IPT framework. It appears that there is high risk of AC grid overvoltage caused by uncontrolled reactive power changes. A scheme of AC/DC converters with less reactive power generation is being adopted to mitigate this risk.

The KODA has been performing two-staged R&D to establish the technical basis for the ITER AC/DC converter. [7] The first phase R&D demonstrates the main features of the ITER converters such as current sharing of 1.28 and fault suppression capability at the peak current level of 175 kA with a 1/6-scale ITER vertical stability converter. (See FIG. 7) The second phase R&D is to build a 1/2-scale ITER converter that realizes the full integration of all assemblies. A safe structure from severe electromagnetic stress under high fault current levels up to 300 kA will be engineered and verified.

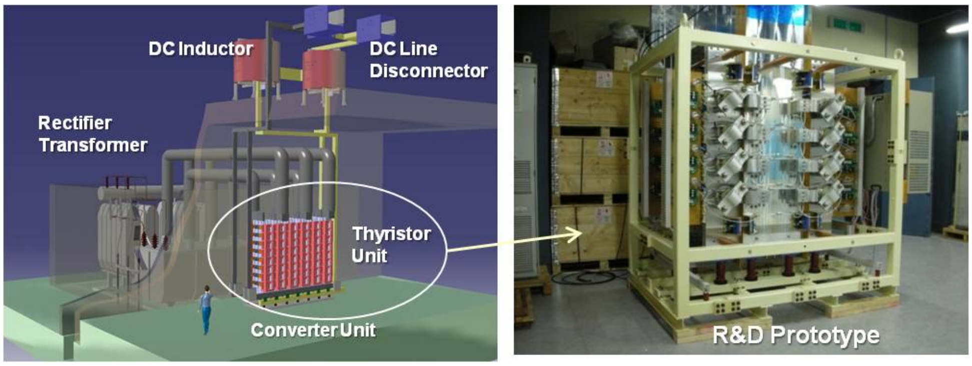

FIG. 7. The Proto-type system of the ITER AC/DC converter.

All components of the AC/DC converter will be designed, fabricated and tested according to IEC standards. Site tests required according to IEC recommendations and special tests described in technical specifications are to be carried out under the IO approved procedures. Insulation tests, functional tests and protection tests along with other various factory tests are to be included in the qualification control process.

3.8. ITER Diagnostics

ITER Korea will procure one diagnostic upper port plug, a neutron activation system (NAS), and three vacuum ultraviolet (VUV) spectrometer systems monitoring the main, edge and divertor area plasma. [8] Implementation activities including R&D on ITER diagnostics in Korea started shortly after procurement assignment. Currently, ITER Korea is conducting detailed design and engineering work to resolve outstanding technical issues and to finalize technical specifications and is also participating in joint activities for development of technical documents for the procurement arrangement (PA), which include measurement requirements, functional block diagrams, procurement schedule tables, and engineering annexes.

3.9. Tritium Blanket Modules (TBMs)

Korea has proposed a Helium-Cooled Solid Breeder (HCSB) blanket concept with a graphite reflector for the Test Blanket Module (TBM) program which aims to test and validate the concepts relevant to DEMO and/or fusion power plants. It is a unique feature among TBMs that the amount of beryllium is considerably reduced by replacing some of it with graphite satisfying the tritium self-sufficiency condition. However, there are still concerns on the graphite reflector related to nuclear performance as well as material-related issues. The neutron shielding capability of the combination of breeder, multiplier and reflector should also be compatible with the combination of breeder and multiplier. If it is not, radial build should be increased or more power is required to drive the helium refrigeration system for the super-conducting magnets, both of which are not desirable. Nuclear analysis is performed under fusion reactor conditions to address the feasibility of the graphite reflector in the breeding blanket. While the breeding zone thickness is fixed, sensitive tests are performed with varying graphite amount, i.e. increasing graphite and decreasing beryllium or vice versa. Introducing an engineering factor, the optimal TBR/beryllium is justified in terms of reduction of beryllium amount and tritium breeding capability. Currently, R&D activities are on-going in developing joining technology for structural materials, breeder and reflector pebble development, and building up infrastructure for a liquid breeder loop, thermo-hydraulic coolant loop, and purge gas permeation loops.

4. R&D Activities related to ITER Technology

Participation in the ITER project shall lead Korea to find a potential key to approaching the fusion energy utilizing the technical benefits gained by becoming an ITER member country. The investment of 10% in ITER will induce the Korean fusion community to reach 100% in ITER fusion technology. In the end, the ITER Korea project will contribute to building industrial infrastructure and training human resources for developing a DEMO fusion reactor. In line with this perspective, the KODA is committed to building up core technology in the ITER tokamak, even though many components are not part of the 10 Korean procurement items among the total 86 ITER components. To cover the full scope of ITER technology, the KODA is planning on a strategic program that is complementary, focusing on research and development of non-procurement items and the preparation of ITER operation in collaboration with universities and other research institutions. This program together with KSTAR and other fusion R&D will provide an opportunity for the Korean fusion community to reach a worldwide leading position in fusion technology that is necessary for developing a fusion reactor for the future.

5. Conclusions

ITER Korea (KODA) is performing various procurement activities including engineering design and technical R&D in preparation of the necessary technical support and risk containment for the KO procurement packages and has achieved significant progress to be able to meet the required schedule and technical specifications. It is noted that the KODA is keen on building up the core technology in ITER and endeavors to make a contribution to the Korean fusion community for developing a DEMO reactor and commercial fusion power in the future.

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