ITER-FEAT Vacuum Vessel and Blanket Design Features and Implications for the R&D Programme

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Abstract. A tight fitting configuration of the VV to the plasma aids the passive plasma vertical stability, and ferromagnetic material in the VV reduces the TF ripple. The blanket modules are supported directly by the VV. A full-scale VV sector model has provided critical information related to fabrication technology, and the magnitude of welding distortions and achievable tolerances. This R&D validated the fundamental feasibility of the double-wall VV design. The blanket module configuration consists of a shield body to which a separate first wall is mounted. The separate first wall has a facet geometry consisting of multiple flat panels, where 3-D machining will not be required. A configuration with deep slits minimizes the induced eddy currents and loads. The feasibility and the robustness of solid HIP joining was demonstrated in R&D, by manufacturing and testing several small and medium scale mock-ups and finally two prototypes. Remote handling tests and assembly tests of a blanket module have demonstrated the basic feasibility of its installation and removal.

1. Introduction

The ITER vacuum vessel (VV) and blanket design and R&D (L-3 and L-4 Projects) has progressed during the last two years. Many activities have been carried out to assure the feasibility of the vessel and FW (first wall)/blanket design and to reduce the cost for ITER-FEAT.

2. Vacuum Vessel

2.1 Vacuum Vessel Design

The VV discussed here is similar to the earlier 1998 ITER VV in basic features such as structure (double wall), basic shape (torus) and material (SS 316L(N)-IG, ITER Grade) (see *FIG. 1, 2*) [1]. However, the blanket modules will be supported directly by the VV [2], and the blanket cooling manifolds will be mounted on the plasma-side surface of the VV inner wall. The inner and outer shells are both 60 mm plates and the stiffening ribs 40 mm plate (see Table. 1). Poloidal ribs between the inner and outer shells are partially replaced by flexible support housings for the blanket module attachments. The flexible support housings also function as part of the first confinement boundary. They will connect inner and outer shells, and localized ribs between the housings will be used for reinforcement (see *FIG. 3*). The number of continuous poloidal ribs is minimized to reduce the vessel fabrication cost [3]. While the space between the shells will be filled with plates made of SS 304 with 2 % boron (SS 30467), those located under the TF coils are made of ferromagnetic SS 430 to reduce toroidal field ripple. These shielding plates are installed at the factory before shipment to the site (see *FIG. 4*). The VV would be fabricated in the factory as sectors each spanning 40° (see *FIG. 5*). This large sector configuration has advantages with respect to design, fabrication, and assembly.

The vessel will be designed and constructed to the ITER VV code which is based on the ASME code section VIII (Div. 2). The fabrication tolerance of the sector height and width will be –20 mm, which is consistent with the L-3 R&D performance. The sector reference points will be defined so that surface tolerances will be –10 mm. The inner shell is fabricated as the first step as it is the most important for confinement. Butt welds are to be used on the inner shell, and inspections can be performed easily. Most weld joints will have conventional configurations and will be radiographically inspected to assure 100% weld efficiency. These welds will be code/standard qualified. However, the one-sided weld joints between the outer shell and the ribs/housings, and the field joints, cannot be radiographically inspected and so will be inspected by UT (ultrasonic testing), and a "code case" will be justified by testing. The current approach of the weld joint configurations is to minimize required code cases. To reduce the VV fabrication cost, the employment of cast and/or forged structures has been investigated. Precision casting or powder HIPing (hot isostatic pressing) of the blanket support housings would be a cost-effective solution. The maximum weight limit ~100 kg is acceptable for manufacturing the housing by precision casting. The vessel structure in its gravity support region is highly-stressed, requiring reinforcement. Therefore, instead of a shell-welded structure, a forged structure would be a workable solution which would also improve the tolerances.

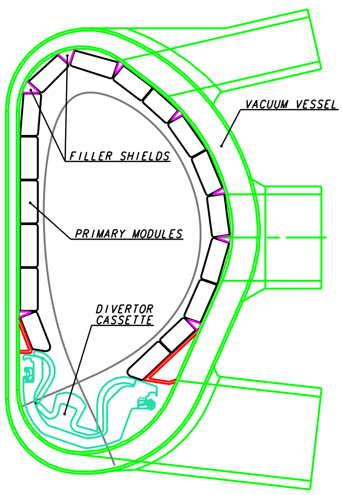


FIG. 1 Vacuum vessel and blanket module poloidal segmentation

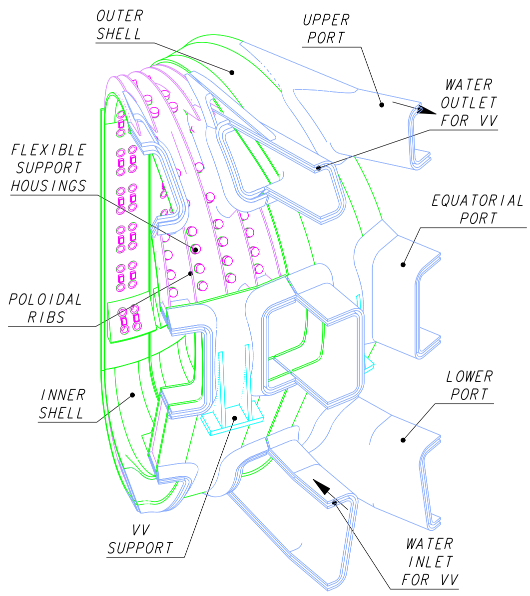


FIG. 2 Vacuum vessel overall structure

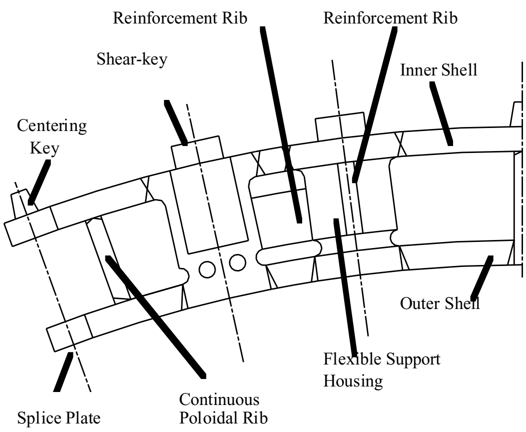


FIG. 3 Double wall VV structure (Inboard)

TAB. 1 VACUUM VESSEL PARAMETERS

Torus height / OD	11.3m /19.4 m
Total double wall thickness	0.34 m - 0.75 m
Number of sectors	9 (40 ° sector)
Shell/rib thickness	60/40 mm
Toroidal/poloidal resistance	8.8/3.8 μΩ
Interior surface area	943 m^2
Total mass (without water)	6500 t
Heat load	10 MW
Coolant inlet temperature	100 °C
Coolant inlet pressure	1.1 MPa
Total coolant mass flow rate	950 kg/s

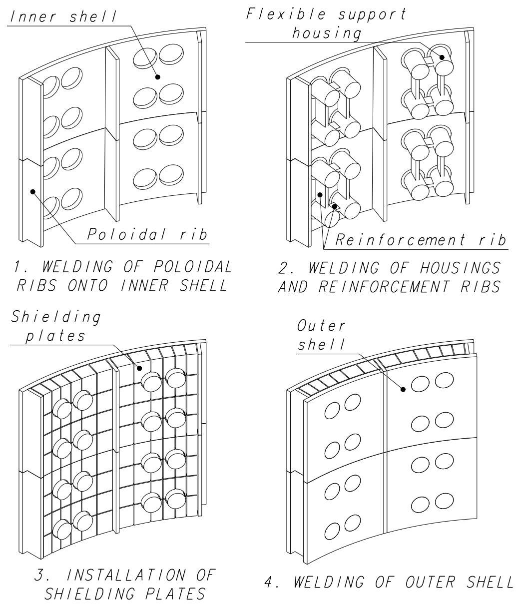


FIG. 4 Fabrication procedure of VV double wall structure (inboard)

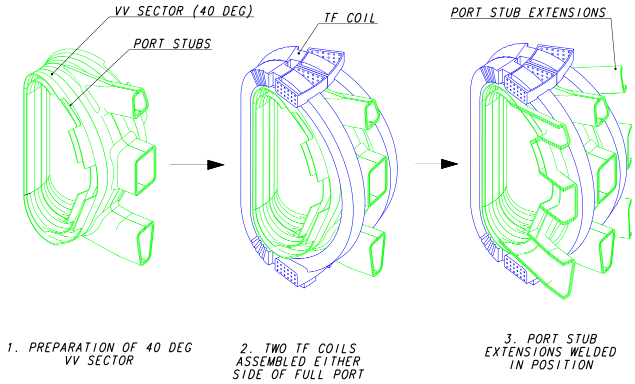


FIG. 5 Assembly of VV sector and TF coils

2.2 Vacuum Vessel R&D

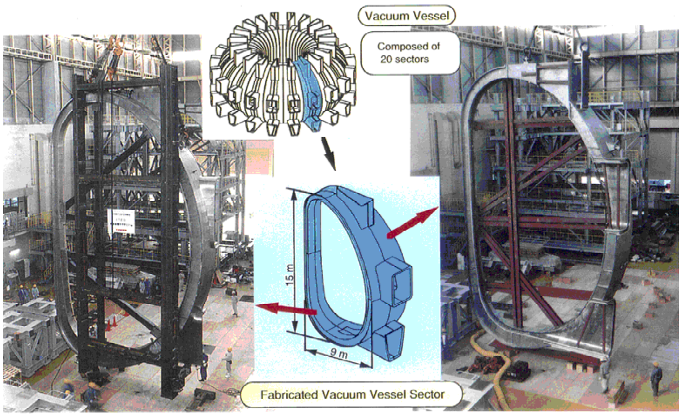
A full-scale VV sector model and a port extension model have been fabricated and tested to provide critical information related to fabrication technology required to produce a high quality component, and to test the magnitude of welding distortions and achievable tolerances [4]. Such information required to complete the design could not be obtained with a small model. The full-scale sector model consists of two 9° half sectors (see *FIG. 6*) [5]. Overall dimensions of the half sectors are 15 m in height and 9 m in width. The geometry of the model is very close to that of the 1998 ITER VV which is larger than the ITER-FEAT VV. The basic structure is a double wall design with the inner and outer shells made of welded plates, 40 to 60 mm in thickness and connected by ribs which space the shells 0.45 - 0.83 m apart. The design of each half sector is different to allow several fabrication methods to be investigated [5]. The full-scale port extension achieved tolerances within 4 mm.

Field joint welding was tested during the assembly of both half sectors [5]. Automatic TIG welding machines mounted on guide rails were used to make the weld joints between two half sectors. Alignment of the outer shell butt weld joint was accomplished by adjusting the sector positions and locally jacking the shells into position before the welding. Alignment of the root gap and root offset of this joint was achieved to < 0.8 mm. The inner shell field joint includes 8 splice plates adjusted to fit the joint for the main shell. A poloidal distribution from the average of weld deformation for the joint was controlled to be uniform to within 1 mm (total values were ~ 4-5 mm) for both butt welding of the outer shell and splice plate welding of the inner shell. Based on the results, the tolerance of the field joint welding is designed to be 5 mm.

The integration test of the port extension has now been performed with the full-scale sector model using remotized welding, cutting and non-destructive testing (NDT) systems (see *FIG. 7*) [4]. The shape of the field joint between the sector model and the port extension is a rectangle 3.4 m high and 2.2 m wide (see *FIG. 8*). A butt weld joint was employed for the outer shell field joint, while splice plates were used for the inner shell field joint. Two sets of remotized welding systems were used for the welding. After the outer shell connection, weld shrinkage in the port axis direction is found to be about 5 mm. Dimensional change observed for the port structure during the welding was up to –1 mm. Cutting of the port extension from the sector is planned to demonstrate the remotized cutting system and to obtain information on deformation before and after cutting.

Further development of advanced methods of cutting, welding and NDT for the VV are underway in order to increase the potential for improved cost and technical performance, such as maintaining the stringent tolerances, tight space constraints, and reweldable surface conditions. The investigated methods include NdYAG laser cutting and welding, reduced pressure electron beam (RP-EB) welding (see *FIG. 9*), and ultrasonic testing systems (see *FIG. 10*) including phased array methods [6].

Since the basic design of the ITER-FEAT VV is similar to the fabricated sector, the fabrication of the full-scale sector model has also confirmed the fundamental feasibility of the ITER-FEAT double wall design. The assembly and field joint welding required for the sectors is common between the 1998 ITER and the ITER-FEAT VV. The technologies developed for the assembly and field joint welding should be applicable to the ITER-FEAT VV. Additional R&D, such as the fabrication of a partial VV sector model including the attachments of the blanket modules, may be required to confirm the improved fabrication technology and associated tolerances.



Sector-B (1/2 sector) Sector-A (1/2 sector)

FIG. 6 Full-scale sector model(JAHT)

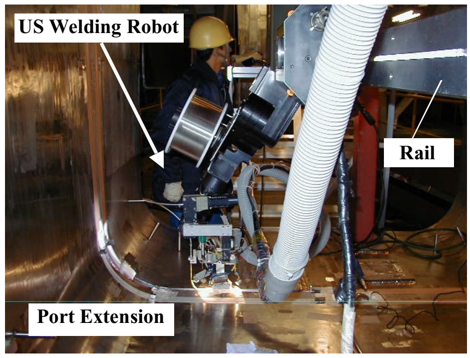


FIG.7 Port extension assembled to VV sector(JAHT)

FIG. 8 Integration of port extension(JAHT, RFHT)

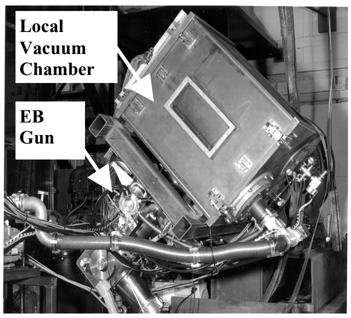


FIG. 9 RP-EB welding machine (EUHT)

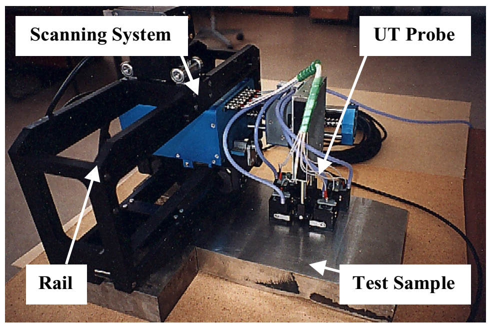


FIG. 10 Ultrasonic testing system (RFHT)

3. FW/Blanket

3.1 FW/Blanket Design

The basic concept of the blanket system of the 1998 ITER has been retained, i.e. a modular configuration with a mechanical attachment system (see *FIG. 11*). However, the ITER-FEAT blanket module design has been improved, (a) to reduce the module cost, (b) to reduce the radioactive waste, (c) to reduce electromagnetic (EM) loads due to disruptions/VDE's (vertical displacement events). The module configuration consists of a shield block to which 4-6 separate FW panels are mounted. The application of multiple flat panels for the FW simplifies the unit design and reduces the associated machining costs. A deeply slitting configuration reduces significantly the induced eddy currents and EM loads in the module. The FW attachment scheme has two options (i) a central shaft and (ii) shear ribs with bolts. In the former option, FW panels are mounted with a central shaft attachment which is welded or bolted to the shield block at its rear side. The welding attachment has a cylindrical shaft ~20 mm thick (see *FIG. 12*) which is welded with a YAG laser. In the latter option (see *FIG. 13*), FW panels are attached to the shield block using a system of bolts and shear ribs [7].

The FW has Be plasma-facing armour joined to the Cu-alloy heat sink with poloidal cooling channels [8]. The FW panel will be manufactured using solid HIP as the reference method, with powder HIP for serpentine pipes also considered as an alternate method if fabrication is possible within acceptable dimensional accuracy. Based on earlier ITER R&D, HIP using a Ti interlayer is a prime candidate for joining the Be armour to the Cu-alloy heat sink. The blanket R&D program until the end of the EDA includes the fabrication of separate FW panels with different materials and manufacturing methods. This would involve R&D of a CuCrZr heat sink instead of DS-Cu, and alternative Be/Cu-alloy joining techniques such as brazing, plasma spray or diffusion bonding, which could also result in cost saving. The shield block is cooled preferably with radial flow, where efficient cooling will be achieved in the rear part with an irregular configuration due to intensive grooving. Three configurations of the radial flow cooling have been investigated (a) a front header with co-axial flow, (b) front and rear headers, (c) a middle header with co-axial flow [2,7]. *FIG. 14* shows one of the radial flow concepts where the shield block is made by the conventional method of drilling/plugging and machining/welding on flat forged blocks. Powder HIP is also a prime fabrication method which was successfully developed for the shield block in ITER R&D. Fabrication by casting has also been investigated for further cost reduction. Selection will be based on R&D results and cost evaluation.

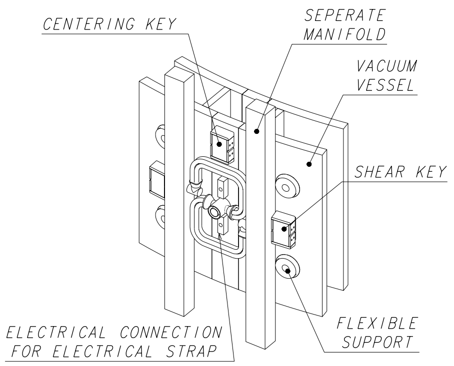


FIG. 11 Blanket module attachment and cooling manifolds

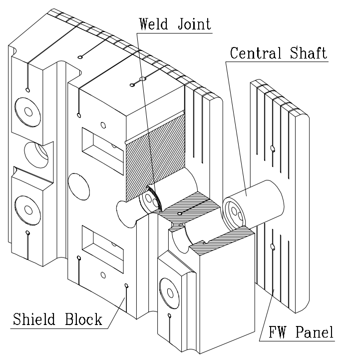


FIG. 12 FW panel attachment (centre shaft attachment)

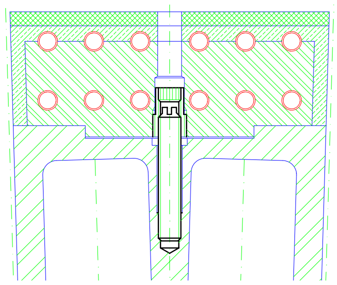


FIG. 13 FW panel attachment (shear ribs and bolts)

TAB. 2 FW/BLANKET PARAMETERS FOR 500 MW FUSION POWER OPERATION

First wall surface area	680 m^2	Heat flux on FW, av./max.	0.2/0.5 MW/ m^2
Number of modules	421	Max. heat flux on limiter	~8 MW/ m^2
Maximum module weight	4.5 t	Total blanket thermal power	690 MW
Total weight of modules	1530 t	Coolant inlet pressure	3.0 MPa
Neutron wall load, av./max.	0.56/0.78 MW/m^2	Coolant temp., inlet/outlet	100 °C/ 148 °C
Total average neutron fluence	0.3 MWa/ m^2	Total coolant mass flow rate	3378 kg/s

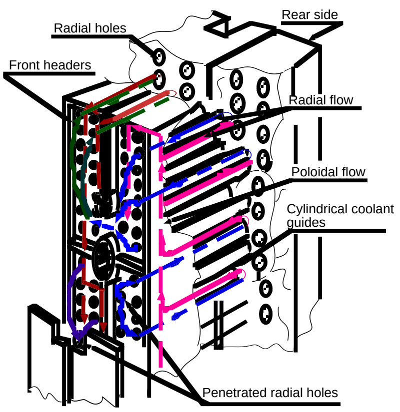


FIG. 14 Radial-flow shield block

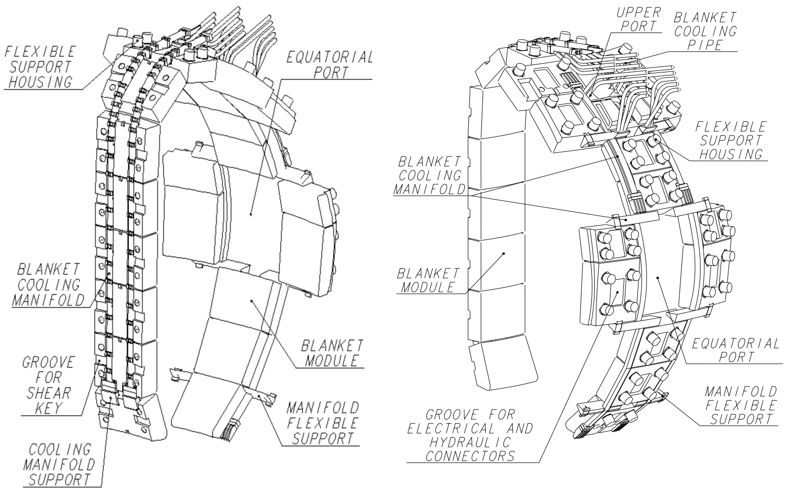


FIG. 15 Blanket modules and cooling manifolds

The supply of the coolant to the blanket modules can be achieved either by channels built inside the vessel wall or by separate manifolds mounted behind the modules. The leak detection procedure for the blanket requires large number of cooling manifolds to reduce the number of modules per loop. Individual cooling loops can be isolated from each other and checked progressively for leaks with tracer elements. The double wall vessel structure is simplified and the ability to use standard welding techniques increased by removing many independent cooling channels. For these reasons, the separate manifold design has been selected for ITER-FEAT [9].

3.2 FW/Blanket R&D

The comprehensive R&D which has been performed gives a sound basis for the present blanket design. There has been substantial progress as a result of the joint effort of the JCT and the Home Teams [10]. All the blanket materials and joining techniques have been characterized and improved, and fulfil all the requirements. Be armour with large tiles (e.g. 50x100 mm) was joined to the Cu-alloy heat sink using a Ti interlayer by HIP (e.g. 800-850°C, 2h, 120 MPa). In thermo-mechanical tests a mock-up has withstood 2.5 MW/m^2 for 1000 cycles without damage, and other mock-ups 13000 cycles at 0.7 MW/m^2 (see *FIG. 16*). The feasibility and the robustness of one-step solid HIP (typical parameters 1050 ^{o}C, 150 MPa, 2h) for joining both Cu-alloy/SS and SS/SS was demonstrated by manufacturing and testing several small and medium scale mock-ups, and finally, a prototype (see *FIG. 17*). The joints resisted thermal loads up to 5-7 MW/m^2 and 0.75 MW/m^2 for 2500 cycles and 30000 cycles, respectively. Large scale mock-ups and prototypes of the blanket module have been manufactured by powder HIPing to reduce costs. In particular, a 4 t 3-D prototype has been successfully manufactured (see *FIG. 18)*. For high heat flux component application, mock-ups of the limiter module with small Be tiles using a CuInSnNi fast amorphous brazing technology (800^{o}C for a few minutes) have resisted up to 12 MW/m^2 for 4500 cycles (see *FIG. 19*). The R&D is now continuing focused on demonstrating the feasibility and performance of the reduced cost separate FW design (see *FIG. 20*). The feasibility of powder HIPing for the fabrication of the FW panels, with particular regard to the tight dimensional tolerances required, still needs to be demonstrated by R&D.

Some prototypes of the titanium flexible supports were also successfully produced, and mechanical tests including fatigue and buckling have been carried out at the operating temperature (see *FIG. 21*). The test results have demonstrated that these components meet their loading requirements.

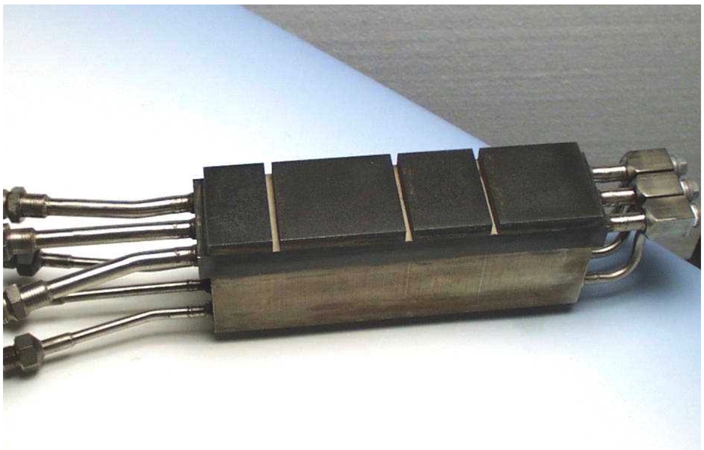


Fig. 16 FW mock-up with HIPed Be armour(EUHT)

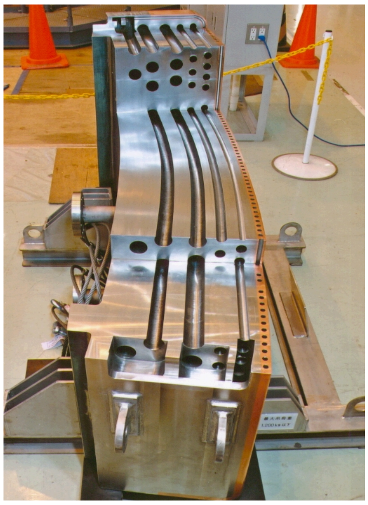


Fig. 17 Blanket module prototype(JAHT)

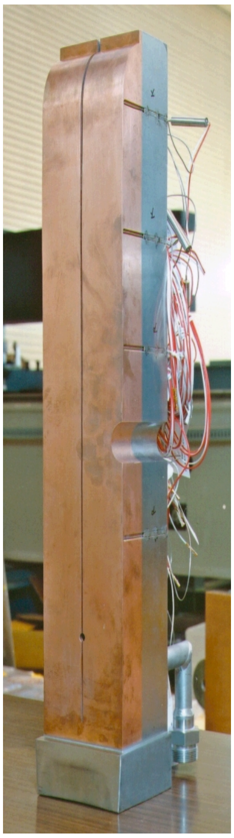


Fig. 20 Separate FW mock-up (JAHT)

Fig. 18 Powder HIPed module prototype (EUHT)

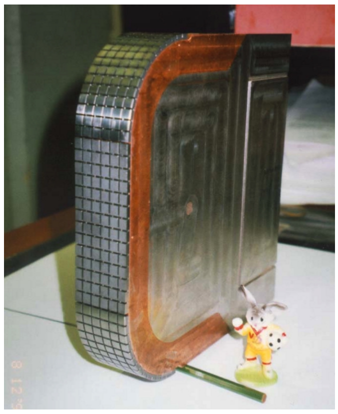


Fig. 19 Limiter mock-up (RFHT)

Fig. 21 Flexible cartridges and test assembly at operating temperature (RFHT)

Remote handling tests have been performed successfully using a remotized in-vessel transporter system for the installation and removal of a 4.5 t blanket module within the specified tolerances. Assembly tests have also demonstrated the basic feasibility of installation and removal of a blanket module with mechanical attachments. A hydraulic, remotely driven bolting tool, which achieves high pre-loading using heating rods, has been developed. Remotized hydraulic connections through a 30 mm penetration hole have also been successfully performed, using a miniaturized YAG laser welding tool.

In the frame of the blanket R&D, innovative technologies have been developed and existing technologies have been improved, giving confidence in the feasibility and robustness of the chosen blanket design.

4. Conclusions

The ITER vacuum vessel and blanket design and R&D have progressed significantly as collaborative efforts by the JCT, the European, the Japanese, and the Russian Home Teams. The design and fabrication methods of the FW/blanket and VV have been assured by the R&D, and optimized methods will be selected based on the results. The design will be finalized and the R&D will be completed during the ITER EDA.

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