ITER Engineering Design Activities - R & D

Vessel, Blanket & Divertor (II)

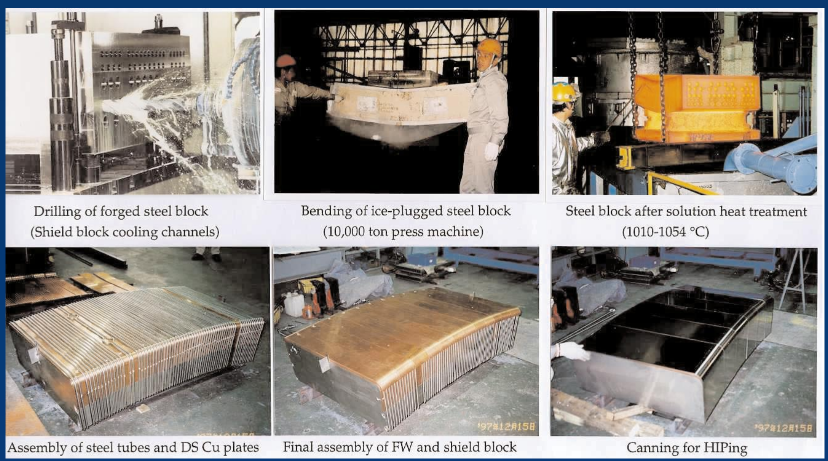
Blanket Module Project (L-4)

Objective

- *Develop and fabricate prototype components for the shielding blanket, in order to assess their manfacturing feasibility.*
- *Assemble them together and develop bolting, welding and cutting tools for the remote removal of the components.*
- *Demonstrate the performance by testing representative parts of the components under relevant conditions.*
- *Obtain confirmation of the design choices by results from accompanying R&D on materials, joining techniques and neutronics using a fast neutron source.*

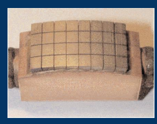
Full scale prototypes include multi-layered first walls made of stainless steel as structural material with copper alloy as heat sink and Be or C as protection material, massive stainless steel shields, and flexible supports.

The feasibility of installation and removal of a blanket module with mechanical attachments has been demonstrated and tested in a prototype assembly. A hydraulic, remotely driven bolting tool has been developed, which achieves high pre-loading using heating rods. High quality remotized hydraulic laser-welded connections have also been made through a 30 mm penetration hole in the front of the module.

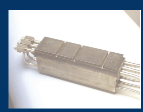


Full-scale prototype manufacture (JA).

Joining Techniques



- *Be/Cu joints of high heat flux components (e.g. limiter): fast amorphously CuInSiNi-brazed small tiles on curved Cu surface (RF), withstood 4500 cycles at 12 MW/m².*



- *Be/Cu joints of lower heat flux components (e.g. first wall): hot isostatic pressing (HIP) of Be tiles with Ti interlayer (EU), withstood 13000 cycles at 0.7 MW/m².*

- *Joining of Cu/SS parts with high precision: solid-solid HIP of the first wall withstood (e.g. JA) 2500 cycles at up to 7 MW/m².*

Flexible supports (RF).

FW prototype (JA).

Shield block prototype - powder HIP (EU).

Port limiter (RF).

Assembly test rig (EU).

Module cut for inspection.

In the frame of this 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.

Divertor Cassette Project (L-5)

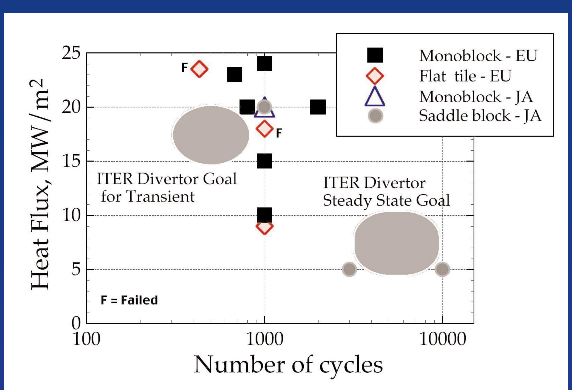
Objective

To develop the technology needed to construct full-scale armoured components capable of providing adequate armour, armour-heat sink joint (CfC-Cu & W-Cu), and heat sink lifetime, and sustaining thermo-hydraulic and electro-mechanical loads, whilst utilizing the most cost effective and reliable manufacturing processes.

Major issues include

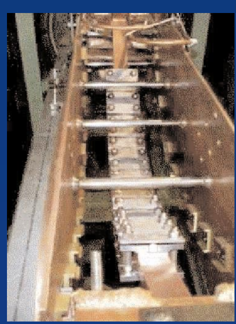
- *the bonding of different plasma facing materials on the same component,*
- *the selection of the heat sink material (CuCrZr now preferred), and the demonstration that it maintains its properties after manufacturing.*

The graph of component test results shows that various tile geometries can meet the ITER requirements. However, the monoblock has proved to be the most reliable with no complete detachment of tiles. Tungsten brush type armour proved to be a solution to having a Cu-W joint able to withstand the large difference in thermal expansion of the two materials under the high heat flux loads.



Results of CfC/Cu high heat flux component testing.

An additional aim of the project was to integrate key plasma facing components together onto a realistic prototype of the cassette body. Following the decision of the US to pull out of ITER, the EU has also constructed an integration prototype. It is not essential to use all the real materials for these prototypes, and dummy components have been made - thermohydraulic equivalents of the real components.



W and Be armour fast brazing to liner CuCrZr heat sink (RF).

Pure Cu-clad DSCu tube armoured vertical target with saddle block CfC and CVD-W armours (JA).

CfC monoblock and W brush armoured vertical target (EU).

Outboard integration mockup prior to installation of liner (EU).

Inboard divertor channel integration mockup undergoing flow tests (US).

Several middle and large scale CfC and W-armoured divertor prototypes have been successfully tested at heat fluxes ~ 20 MW/m² x 1000 cycles, which is consistent with ITER operational needs.