PROGRESS AND ACHIEVEMENTS OF THE ITER L-4 BLANKET PROJECT

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Abstract

The L-4 Blanket Project embraces the R&D of the ITER Shielding Blanket, and its main objective is the fabrication of prototype components. This paper summarises the main conclusions from the materials R&D and the development of technologies which were required for the prototype specifications and manufacturing. The main results of the ongoing testing activities, and of the component manufacture are outlined. The main objectives of the project have been achieved including improvements of the material properties and of joining technologies, which resulted in good component quality and high performance in qualification tests.

1. INTRODUCTION

The objectives of the L-4 Blanket Project are to demonstrate i) the fabricability of the ITER blanket components and ii) their ability to be properly assembled, iii) to verify the performance of their key engineering features, and iv) to confirm the design choices made. The Project was initiated in June 1995 as a collaborative effort of the ITER Joint Central Team and the Home Teams of the European Union (EUHT), Japan (JAHT), Russian Federation (RFHT) and United States (USHT) [1]. At present the Project has resulted in the selection of the main materials and joining techniques for ITER, improvements in their properties and their characterisation also in irradiated conditions. Neutronic experiments have validated the codes and the libraries used for the shield design. Prototypes or large scale mock-ups of the main components have been completed or are almost manufactured. Assembly tests are being carried out for a preliminary demonstration of component integration.

2. STATUS OF THE SUPPORTING R&D IN THE AREAS OF MATERIALS AND NEUTRONICS

Activities on materials have essentially been focused on stainless steel (SS) and copper alloys for First Wall (FW) and shield application. The early choice of 316 L(N) SS as the main structural material for the ITER blanket has been confirmed, and a particular ITER Grade (IG) has been specified. This SS grade limits the N to a narrow range (0.06 - 0.08%), obtaining high strength and a good weldability also by electron beam (EB), and the B to 10 wppm to reduce the He production under irradiation. Other long term activation impurities (Co, Nb) are also limited.

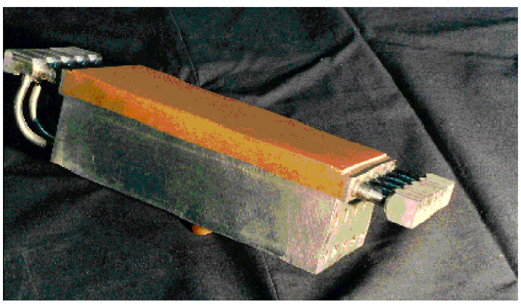

FIG. 1 EUHT primary FW mock-up

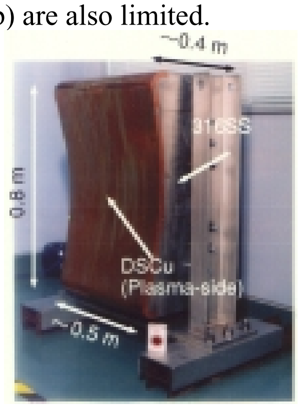

FIG. 2 JAHT medium scale primary wall mock-up

The material characterization was focused on the raw material and on the products obtained by the main manufacturing technologies, Solid and Powder Hot Isostatic Pressing (HIP). Solid HIP SS shows a slightly reduced strength, below the average and closer to the lower scatter band of wrought SS. Grain size increases with each HIP step. The Powder HIP SS strength is closer to the average of wrought SS, and the grain structure is fine. Irradiation tests up to ~2 dpa indicate that the fracture toughness (FT) of the powder HIP material decreases but the reduced values are still relatively high (e.g. ~250 kJ/m^2 at 80 °C at HIP conditions 1125°C, 120 MPa for 10 h), elongation of TIG weldments and their FT decrease (FT ~100-150 kJ/m^2 at 325 °C) whereas the yield strength increases (up to ~700 MPa) compared with the irradiated base metal, and that the EB weld material has properties closer to the base metal (FT ~500 kJ/m^2 325 °C). Reweldability tests of irradiated 316L(N)-IG SS have resulted in the requirement to limit the produced He content to 1 appm.

Improvements of the dispersion strengthened copper production process have led to the specification of CuAl-25-IG with improved ductility (>20%) and radiation resistance as the reference FW heat sink material. The precipitation hardened CuCrZr-IG is retained as an alternative material. CuAl25-IG keeps a stable structure up to 950 °C and high strength values after irradiation at temperature up to 300 °C, but has a low FT even in unirradiated conditions. CuCrZr alloy loses its properties (e.g. the strength up to 80%) above ~450 °C, but has a FT almost ten times higher than DS-Cu also in the irradiated state. Additional materials selected for ITER are Beryllium S65C or equivalent grade for the armour of the primary wall, limiter and upper baffle mainly because of its good thermal fatigue properties, Tungsten or the W-1%La_2O_3 alloy for the lower baffle armour because of its low erosion rate, necessary in the high neutral density region near the divertor, and Ti-6Al-4V alloy and Inconel 718 for the flexible support and bolting material. Irradiation tests for these materials have been initiated. In conclusion the properties of all the selected materials and weldments are acceptable and they remain satisfactory after irradiation up to 2-5 dpa. Moreover R&D tests show that corrosion/erosion of SS and Cu-alloys is not a critical issue for ITER conditions, if the specified water velocity and water chemistry are used (deareated, demineralised, neutral pH water with hydrogen addition to suppress radiolysis).

Neutronic bulk shield and streaming experiments on small gaps and large penetrations have been conducted using two 14 MeV neutron source facilities to verify the predictive capability of the reference tools for the analysis, i.e. the MCNP-4A code and the FENDL-1 data base, and to assess the design safety margins. Results show that the predictions underestimate the experimental results by a maximum of 30 % at the toroidal field coils. This is covered by the design margins.

3. DEVELOPMENT OF THE BLANKET COMPONENTS

The manufacturing feasibility for the primary wall, baffle and limiter blanket modules has been assessed in two main stages. In the first, several (>20) small and medium scale mock-ups of the first wall part of each module type and of the shielding block have been manufactured and tested (e. g. see Fig. 1 and 2) in order to verify the selected technologies and assess the performance.

TABLE I MAIN BLANKET JOINING TECHNOLOGIES AND L-4 MOCK-UP ACHIEVEMENTS

Be/Cu-alloy **for Baffle / Limiter FW** Requirement 3 / 8 MW/m^2	- RFHT **Fast Amorphous CuInSnNi brazing** of small tiles at ~800°C. Achievements: 4500 cycles at 12 MW/m^2. - USHT **HIP** of small tiles with AlBeMet interlayer at 625°C . Achievements: 1000 cycles at 10 MW/m^2.
Beryllium/Cu-alloy **for Primary FW** Requirement 0.5 MW/m^2	- EUHT **HIP** of large tiles with Ti interlayer at ~800°C. Achievements : 1000 cycles at ~2.5 MW/m^2, failure at ~5 MW/m^2 in screening tests.
Cu-alloy/SS for all modules Req. Strength not worse thar base materials	- EU, JA, RF HTs **HIP** (at 950-1050 °C, 100-120 MPa, 2-4h) **- Friction Welding** (for tubes) Achievements : Satisfactory quality also in irradiated state 7 MW/m^2 for 1500 cycles
SS/SS for all modules Req. Strength not worse thar base materials	- EU, JA, RF HTs **HIP** (at 1050-1090 °C, 100-150 MPa, 2-4h) **- E.B. Welding** **- TIG Welding** Achievements : Satisfactory quality also in irradiated state

FIG. 3 EUHT powder HIP module destructive examin. *FIG. 4 JAHT blanket prototype completed*

The chosen joining technologies are presented in Table I. The results of the thermo-mechanical tests on the small scale mock-ups show the expected performance of the components:

- The primary FW structure (without armour) withstands high thermal loads (up to 7 MW/m^2 for 1500 cycles). In endurance tests at 0.75 MW/m^2 (1.5 x nominal) three FW mock-ups have resisted >20,000 cycles.
- The performance of the armoured mock-ups for the primary wall, baffle and limiter modules coincides so far with the joint performance reported in Table I.

These results show that the FW meets its main thermomechanical requirements reported in Table I. Additional results of thermal fatigue tests will be available during the second half of 1998. In the second stage of the blanket R&D full or near full scale prototypes have been produced, namely:

- an EUHT full scale mock-up of a shield block manufactured with the powder HIP technology
- an EUHT prototype of the double curvature top module (in progress: completion mid 1999)
- a JAHT prototype of an inboard central blanket module
- two EUHT near full scale baffle/limiter FW mock-ups (in progress: completion mid 1999)

Based on intermediate R&D results, the EUHT has chosen to use powder HIP for manufacturing the shield block, and solid HIP for the FW and its attachment to the shield block. Powder HIP results in adequate quality for the shield part. Solid HIP is necessary to achieve the more stringent tolerances of the FW, but results in higher manufacturing costs because of the large machining work. For the powder HIP technology it was decided to manufacture an intermediate full scale mock-up of the shield to demonstrate that adequate material quality and dimensional tolerances could be achieved (Fig. 3). This component was HIPed at ~1090 °C and 120 MPa for 4h and was completed in 1997 with satisfactory quality (tolerances were ~±3 mm and ± 0.2 mm respectively for the radial position and pitch of the front row channels), and manufacturing procedures have been identified by industry to further improve the manufacturing tolerances.

The EUHT Primary Wall Module Prototype is representative of one of the most complex ITER module (#11), it has a double curvature (poloidal and toroidal), and includes Be armour, the eight front access penetrations and the complex back side structure to accommodate the individual elements of the attachment system. All materials used are the reference ones. A multi-step HIP process will be used with optimised parameters according to the different material joints. The shield block is at present being manufactured using powder HIP. Pre-tests are currently underway to guarantee the correct shape and tolerances. The shield block will be available by the end of 1998. In the meantime, optimisation of the manufacturing process for joining the copper and the beryllium armour by solid HIP and for the FW attachment is in progress, with particular attention to the problem of the double curvature.

The JAHT prototype (Fig. 4) has the design features of the ITER module #8 except the Be armour. It has a single (toroidal) curvature, and includes four radial front access penetrations. The shield was made from a SS forged block, and a deep drilling technique was used to produce the coolant channels. After machining, the block was cold bent.

To avoid excessive channel deformations, water ice was used as a filler material. After cooling to ~ -90 °C the shield block bending was executed at room temperature, while the ice filler was kept frozen. The FW parts were prepared by machining the DS-Cu plates with their semicircular grooves, the SS cover plates and the SS tubes which are inserted inside the grooves of the DS-Cu plates. The FW parts were joined together and to the shield block with a single solid HIP process at 1050 °C, 150 MPa for 2h. The Prototype was completed on schedule. Destructive examinations have been performed on a dummy part, cut away from the prototype; maximum deformations on the outer diameter of coolant tube were +1.0 /-0.0mm, and no defects in the joints were detected. Pressure and leak tests were satisfactory.

Two EUHT near full scale baffle/limiter FW mock-ups are being manufactured. One will be equipped with Be armour, one with W and CFC. Selected FW designs are integrated in a common SS shield block in order to be further developed and tested in prototypical mock-ups.

The RFHT is manufacturing two small-scale port limiter mock-ups with all the key features of the limiter, essential for its technological development (Fig. 5). The mock-ups have a Be armoured FW in copper alloy integrated with a SS shield plate. The FW has curved parts which will resemble the limiter edge region. It has cooling channels both inside the FW and the shield plate which are connected in series. The armour will be highly castellated (~6x6 mm) for the best thermal performance. One step solid HIP technology is being used for manufacturing the shield plate, the FW and its attachment to the plates. An amorphous braze alloy (Table I) is used for the Be/Cu joining.

The mechanical attachment of the blanket modules to the supporting structures in ITER comprises bolted flexible cartridges and a centre pin, as well as hydraulic and electrical connections. Furthermore, the individual modules are toroidally connected by big keys. Manufacturing of prototypes of the main components and assembly tests are in progress. The RFHT has manufactured and is going to test prototypes of the flexible cartridge (Fig. 6). The EUHT has started the construction of an assembly test stand. The purpose of this facility is to gain practical experience with the assembly operations and procedures, and in particular with the required tolerances and clearances. The facility includes a full size mock-up of a module, and a part of the supporting structure. The activity includes the manufacture and testing of prototypical bolting, cutting and welding tools.

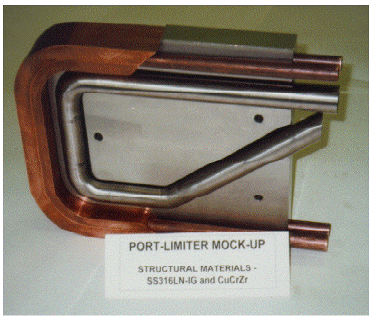

FIG. 5 RFHT limiter mock-up assembled

FIG. 6 RFHT flexible cartridge prototypes

4. CONCLUSIONS

The L-4 Project has generated innovations and a substantial progress in the development of the shielding blanket technology of ITER. The main objectives have been achieved. ITER materials and joining technologies have been selected and qualified. The fabrication of the prototypes of the main components has been or is being completed. Preliminary assembly tests are in progress. The performance tests have already produced satisfactory results, and further tests are underway to substantiate the choices made in the manufacturing processes. Additional activities to be completed are the long term fatigue testing of the FW, the final demonstration of component integration, and the manufacturing feasibility of the blanket supporting structure.

REFERENCES

[1] W. Dänner et al., The ITER ‘L-4” Blanket Project, proc. of the 16th IAEA Fusion Energy Conference, Montreal, Canada, 7-11 October 1996