Status of R&D of the Plasma Facing Components for the ITER Divertor

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Abstract: The paper reports the progress made by the ITER Home Teams in the development of robust carbon and tungsten armoured plasma facing components for the ITER divertor. The activities on the development and study of armour materials, joining technologies, non-destructive evaluation techniques, high heat flux testing of manufactured components and neutron irradiation resistance studies are presented. The results of these activities confirm the feasibility of the main divertor components. Examples of the fruitful collaboration between Parties and future R&D needs are also described.

1. Introduction

The ITER divertor design comprises cassettes on to which are mounted replaceable plasma facing components (PFC) such as vertical targets (VT), dome and liners. The design of the divertor and the direction of its development are presented in several recent publications [1, 2]. The majority of the design solutions were based on results of extensive R&D activity performed by the four Home Teams (United States of America (US), the Russian Federation (RF), Japan (JA) and Europe (EU)). During the engineering design activity (EDA) phase (1992-1998) the R&D has provided solutions to some general technical issues such as:

- The selection of the preferred armour materials: Plasma facing materials - carbon fiber composites (CFC) and tungsten and structural materials - CuCrZr alloy and 316 stainless steel;
- Technological feasibility of joining the plasma facing material (armour) and structural materials that can withstand high thermal cycling heat load of 15-20 MW/m^2;
- Possibility to using sub-cooled water for heat removal under the design conditions (20 MW/m^2) with sufficient safety factor (1.4).

During the EDA extension phase (1998-2001) the R&D activity of the Home Teams is mainly focused in the following directions:

- Material grade selection involving characterisation of properties and performance under operating conditions (neutron, heat flux etc.);
- Optimization and characterization of armour to heat sink joining technologies, giving higher reliability during the expected operating and transient conditions;
- Demonstration of industrial manufacturing technologies (full size components with reference materials and all design features);
- Further development of non-destructive testing techniques;
- Design simplification and cost reduction.

The recent results of R&D activities and remaining critical issues are described in this paper.

2. Selection and investigation of armour materials

In the ITER divertor the selection of the armour material is mainly based on the erosion lifetime assessment and the plasma interaction [3]. CFC has been selected for the strike point area because it sublimes rather than melting during disruption events. Tungsten is selected elsewhere due to its low sputtering (plasma impurities) and to limit tritium codeposition effects (safety and operation). Along with the traditional investigation of physical and mechanical properties (thermal conductivity, strength etc.) and plasma-materials interaction, different CFC and W grades have been exposed to neutron irradiation followed by simulated operational and transient thermal cycling.

The JA and EU Home Teams have studied the behaviuor of neutron irradiated CFCs in regimes simulating plasma disruption events, [4, 5]. FIG.1 shows the results of weight loss measurements for different CFC materials in the unirradiated condition and post-irradiation. The analysis of experimental results shows that on average, the neutron irradiation of C-based materials at relatively low temperatures (290-335°C) leads to significant enhancement (2-3 times) of thermal erosion of some CFC grades. The main reason for this is the reduction of the thermal conductivity by neutron irradiation. Some of the candidate grades (e.g. Dunlop Concept 2, SEP N112) showed an unacceptably high level of erosion after neutron irradiation and have to be modified or excluded from further consideration.

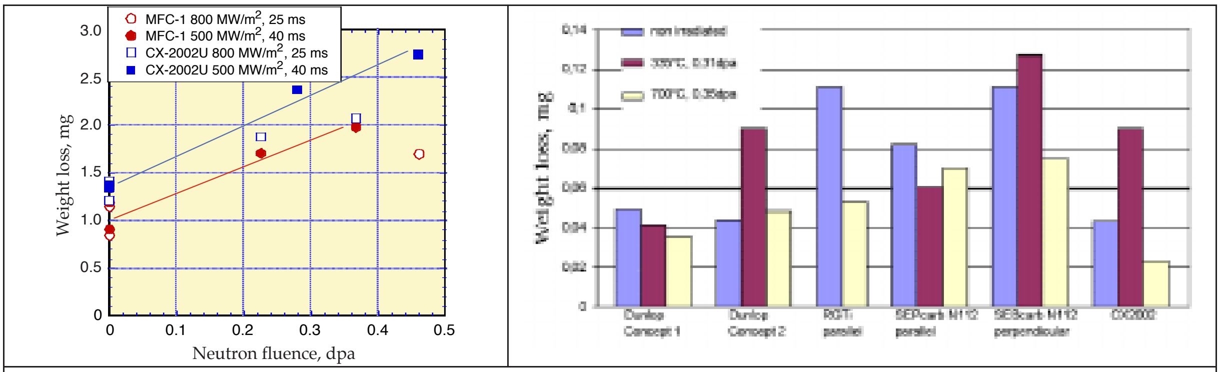

FIG. 1 Thermal erosion of the neutron irradiated CFC, [4,5].

Since W is also a possible candidate for the strike point area (due to the issue of high tritium retention in CFC), systematic studies of different W grades have recently started.

The JA Home Team has performed comparative disruption erosion tests on two candidate W grades using the e-beam facility JEBIS [6]. Weight loss measurements show that unirradiated pure sintered W has significantly lower (at least 2.5 times) erosion in comparison with grade W-1% La_2O_3.

The RF Home Team have performed a comparison of different tungsten grades subjected to plasma disruption simulation events (VIKA and MK-200U plasma guns) followed by one sided thermal-cycling (e-beam facility TSEFEY) thus simulating normal and transient operation regimes [7]. FIG.2 shows surface patterns of different tungsten grades after disruption simulations and subsequent thermal cycling. Generally 20MW/m^2 (without disruption) does not damage the W but, after a disruption simulations (which produces significant surface damage, FIG 2) followed by thermal-cycling there is a distinct difference in the behaviour of the different grades:

- For pure sintered W, pure single crystal W, W-1%Mo, W-1% La_2O_3 and W-2% CeO_2. alloys severe crack formation (up to the full thickness of the tiles) have been observed;
- For single crystal W-0.02%Re alloy and W-5%Re-0.1%ZrC cast alloy no visible cracks have been observed. The possible explanation for this for W-5%Re-0.1%ZrC cast alloy is that the addition of Re improves the ductility and fracture toughness, and in the case of W-0.02Re, the orientation <111> is most favourable for the stress.

The RF and JA Home Team results show that doping of tungsten with oxide particles (W-La_2O_3, W-CeO_2) degrades its performance under disruption like events. The selection of suitable W grades for the strike point area is yet to be made.

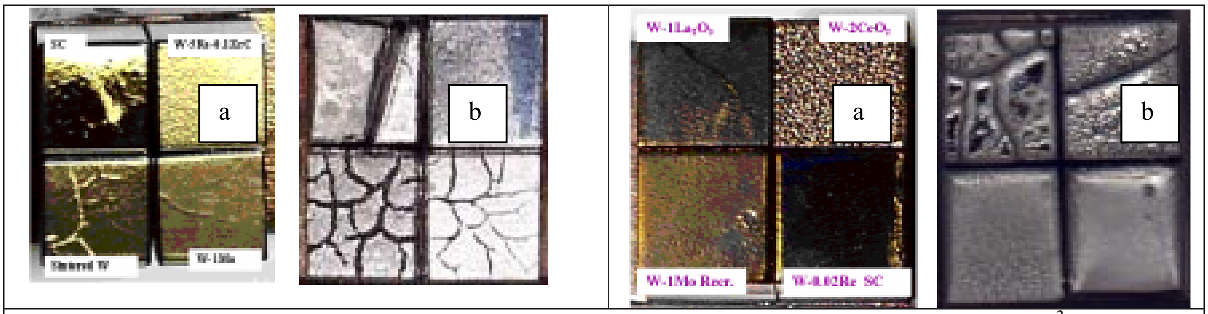

FIG. 2. (a) Surface structure of the several W grades after disruption shorts at 15 MJ/m^2, 0.05 ms and (b).after following thermocycling at 20 MW/m^2.

In the near future, more complicated experiments will be performed by the ITER parties looking at the combined effects of disruption events, thermal-cycling and neutron irradiation on various grades of W.

3. Joining technologies, component fabrication and testing

During the ITER EDA, a number of the armour to heat sink joining technologies was developed, which indicate that they could meet the divertor requirements [8]. Several CFC to Cu mock-ups (using active metal casting or brazing technologies) withstood 1000 cycles at 15-24 MW/m^2. For W to Cu joints, different W tile designs (brush, rods or lamella) cast to Cu or directly brazed onto the Cu heat sink, have survived ~ 1000 cycles at 15-20 MW/m^2.

However, there were several reasons to continue efforts to improve the joining technologies:

- at the maximum design heat flux (20 MW/m^2), although the number of cycles satisfy the requirements, they are yet to achieve the goal, 1000 and 3000 cycles respectively, values which take into account safety margins;
- the technologies used for manufacturing present mock-ups are not completely representative with respect to mock-up dimensions, completeness of manufacturing cycle, reference materials, cost reduction issues and so on.

With respect to maximum heat flux it is important to note that in some cases (e.g. the sudden loss of a neighboring flat tile) the operational heat loads on the joint can achieve 30-40 MW/m^2 [9]. This means that selected technologies have to provide the integrity of the armour to heat sink joints at such heat fluxes (at least for a few cycles) in order to prevent a cascade failure scenario occurring. Note: that for a monoblock geometry the sudden loss of a tile has never been observed and is considered unlikely.

For the reference design of the ITER divertor VT, the EU and JA Home Teams have demonstrated the excellent high heat flux performance of the CFC armoured large scale mock-ups. The JA Home Team manufactured a large scale component with CFC to Cu monoblock joints produced by brazing (FIG. 3). Testing showed the feasibility of meeting the requirements with some tiles surviving 1000 cycles at 20 MW/m^2.

The EU Home Team manufactured a medium scale VT prototype that survived 2000 cycles at 20 MW/m^2 on its CFC section and 1000 cycles at 15 MW/m^2 on its W section (FIG. 3).

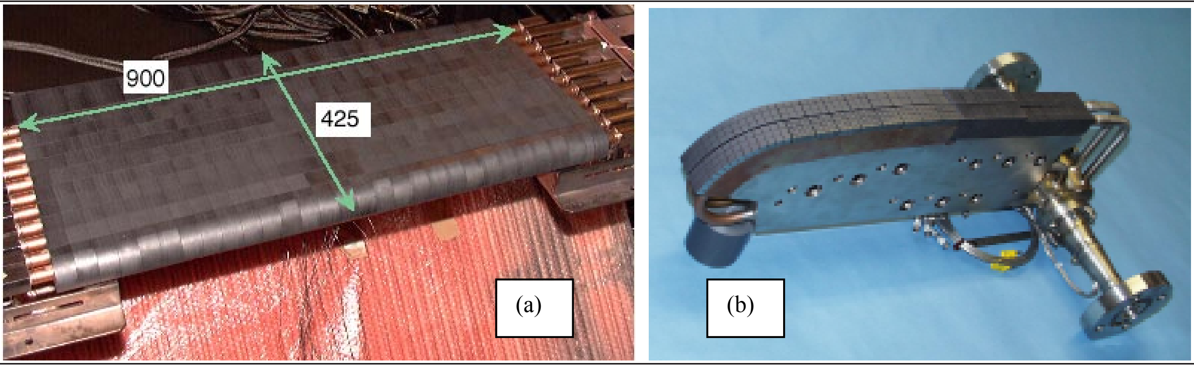

FIG.3. (a) Large scale monoblock type CFC/Cu mock-up, JA Home Team; (b) Medium scale vertical target mock-up with CFC and W armour, EU HT.

In critical heat flux tests the EU Home Team has demonstrated that CFC to Cu joint can withstand several cycles at 30 MW/m^2 [10].

For the performance of W to Cu joints, significant progress has been made by the RF Home Team. Small scale mock-ups (single crystal 10x10x10mm W tiles with a cast OFHC Cu interlayer and 'fast' brazed (800°C) using CuSnIn filler to CuCrZr heat sink) survived 2 cycles at ~ 43MW/m^2 followed by 1500 cycles at 27MW/m^2 without damage of joints [7] (FIG. 4,a).
The US Home Team demonstrated that small mock-ups (W-1%La_2O_3 rods, Ø3.2 mm - isostatically pressed into OFHC/CuCrZr at 450°C) survived absorbed heat flux of 22 MW/m^2 for 1000 cycles without damage, [11] (FIG. 4,b).

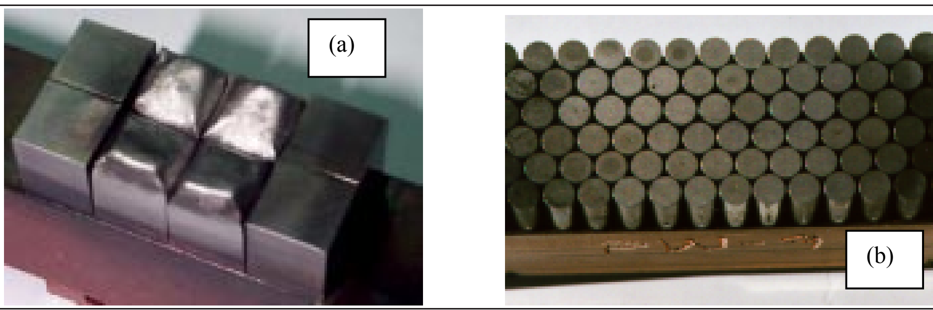

FIG.4. (a) RF HT: W/Cu mock-up after testing; (b) US: W rod (dia 3.2 mm) mock-up.

Optimization of joining technologies has also been directed towards thermal joining cycles compatible with CuCrZr heat sink material properties that are significantly degraded (loss of strength and thermal conductivity) when exposed to temperatures above 600°C. The RF Home Team has demonstrated that a "fast" brazing technology developed for small mock-ups could be applied to the manufacturing of large components (mock-up ~ 1 m long, 20 kg weight) thus maintaining sufficient strength and thermal conductivity of the CuCrZr alloy.

The EU Home Team demonstrated significant progress on the application of new joining technology for the interface "cast Cu to CuCrZr heat sink" based on low temperature (500°C) HIPing. A small CFC-Cu-CuCrZr mock-up (straight monoblock) produced by this HIPing process survived 1000 cycles at 19MW/m^2 and then 680 cycles at 23MW/m^2. The US Home Team demonstrated the use of low temperature HIPing technology (450°C) and HIPing at 850-950°C using fast cooling (~ 2°C/sec) after HIPing for production of divertor components and a large scale dome mock-up while maintaining ~ 80-90% of the fully hardened strength of CuCrZr .

Promising attempts to standardise W and CFC armour geometry and joining technology has been made by the EU Home Team. In this respect it is important to demonstrate the progress of the W monoblock option (compared with excellent results for W-flat tile) and for CFC flat tile option (compared with good results for CFC - monoblock). Mock-ups with W monoblock on CuCrZr tubes made by 500°C HIPing achieved ~ 1000 cycles at 18 MW/m^2 (the maximum capacity of the test facility) and mock-ups with CFC flat tiles survived, before failure, 1000 cycles at 20 MW/m^2 plus 430 cycles at 23 MW/m^2.

For low heat flux loaded W armoured components, the EU and JA Home Teams have developed and tested W plasma spray technologies. Several EU mock-ups with W plasma sprayed armour (thickness ~ 5 mm) have survived ~ 1000 cycles at 5.5 MW/m^2. Cracks in the W armour were observed only at heat flux ~ 7.6 MW/m^2 [10].

All these results give us confidence that there are several joining technologies which, with proper armour tile design, can survive the heat loads predicted for the of ITER divertor components.
Another damaging factor, which is being studied, is the effect neutron irradiation. First results are available from the EU Home Team’s neutron irradiation programme (PARIDE) [5]. Small CFC to Cu mock-ups were irradiated in a nuclear reactor at 350 and 700^{0}C up to a fluence of 0.35 dpa. The best results were obtained for the CFC Dunlop monoblock on a DS-Cu heat sink irradiated at 350°C. After irradiation this mock-up was tested in the JUDITH facility and survived 1000 cycles at 15MW/m^2. The maximum achievable heat flux after irradiation (typically between 10 and 15MW/m^2) was limited not by joint failure, but the maximum CFC surface temperature of about 2000°C. This significant increase in surface temperature (FIG. 5a) is caused by degradation of thermal conductivity.

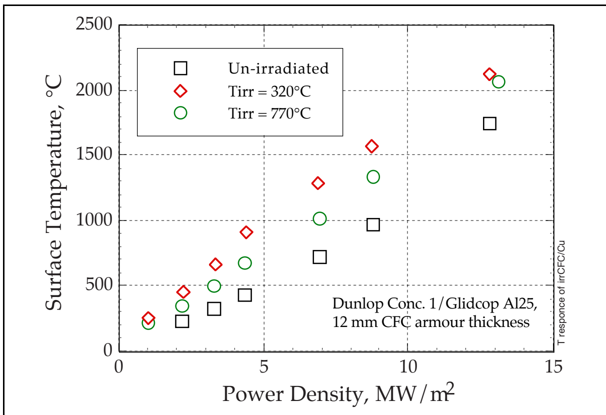

FIG 5a. Surface temperature of the neutron irradiated CFC mock-ups.

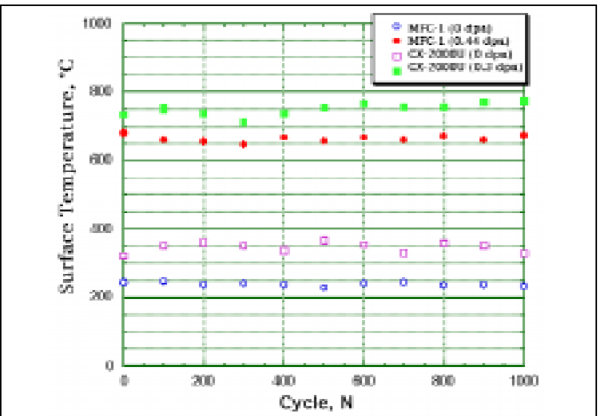

FIG. 5b. Average surface temperature of the CFC through 1000 cycles at 5 MW/m^2.

Similar experiments with irradiated CFC-Cu mock-ups (irradiated up to fluence ~0.3-0.5dpa at irradiation temperature ~300°C) have been performed at the OHBIS – facility by the JA Home Team [12]. No degradation of the measured surface temperature with thermal-cycling, which means no degradation of the quality of CFC/Cu joint, has been observed during the fatigue test, FIG. 5b.

However the possibility to predict the thermal behavior of a mock-up after irradiation exists only for steady-state regimes of irradiation. In reality the neutron irradiation is applied to the structure with significant variations of temperature, thermal stresses and deformations. Therefore the annealing of radiation defects will depend on the synergistic effects of many factors. Additional experiments in this field are still required.

The importance of the R&D activity on the development of non-destructive examination techniques for manufactured components is clear. The activity of all the ITER parties in this area came to a few general conclusions:

- ultrasonic inspection, and x - ray examination are able to detect in many cases the defects in the armour to heat sink joints in the range of ~1-2 mm;
- infrared thermography proved to be a fast and cheap technique to assess the overall thermal performance of a component;
- structural defects in the mm range in many cases have no influence on the working ability of PFCs.

So, at this stage of the investigation it is necessary to clarify the relation between the resolution of different methods, defect geometry and position on the overall performance of components. Mock-ups with artificial calibrated defects, are being subjected to thermal-cycling tests and destructive examination as an essential part of the NDE development plans, which are being actively implemented mainly by the JA and EU Home Teams.

4. Results of R&D activity on the C-T co-deposition in the private zone of the divertor

Co-deposition of tritium with carbon is expected to be the main mechanism for tritium retention in the private divertor zone and in the whole reactor. A semitransparent liner is the main site for such deposition in the private zone and has to be heated to several hundred °C in order to minimise the amount of tritium trapped in deposits. The aim of the current R&D activity is to understand how a hot liner will function chemically as well as to optimise the design to mitigate the co-deposition on the colder surfaces behind the liner [13]. Inductively-coupled RF discharges in methane (simulation of semitransparent structure and pumping duct condition) and magnetron sputtering of graphite in deuterium (simulation of conditions in front of liner) were used to produce a wide spectrum of hydrocarbon radicals as well as atomic hydrogen. Both types of experiments present similar results that indicate that the deposition rate of thermal hydrocarbon radicals drops drastically beginning at about 400K [13]. Therefore, to suppress carbon film formation in the pumping duct, its operating temperature should be above 400K. In order to prevent the hydrocarbon transport to the cryo-pumps it is proposed to use cryogenic catcher behind the cassette. The study of methyl radical (as the most active/mobile hydrocarbon) deposition at liquid nitrogen temperatures is in progress.

The EU Home Team is preparing the experiments on the study of hydrogen isotope retention in deposited films, which will be produced as a result of plasma wall interaction in the steady-state plasma device in MPI Berlin.

5. Future R&D and examples of cooperation

Results of R&D activities already give us confidence that there are several design and technological options which provide the required serviceability of PFCs for a number of years especially during the first stages of ITER operation. Nevertheless, the R&D activity needs to be continued in the future focusing mainly on:

- optimisation of the PFC structure in order to increase the component lifetime and reliability (decrease the number of scheduled and accidental cassette replacements);
- optimisation of the manufacturing of divertor components in order to realise significant cost reduction;
- optimisation of PFC structure in order to improve safety aspects.

The significant progress in the divertor R&D is also a result of the fruitful cooperation between the ITER parties. This cooperation is necessary to provide the best experimental facilities for the testing of components, and for having more confidence in the results (round robin testing of the same mockup in different facilities).

Below are several examples of such joint activity (already realised or that would be useful to realise in the future):

- round robin testing of thermal-hydraulic mock-ups to define the critical heat flux [14];
- round robin testing of the loading parameters of HHF facilities [15];
- use of the JUDITH facility for testing JA and RF neutron irradiated mock-ups and materials [5];
- use of the RF nuclear reactors and plasma guns for testing of EU materials and mock-ups;
- manufacturing, integration and testing of US-JA and EU-RF assembles of full scale dummy PFC prototypes and parts of the cassette body [16,17];
- round robin NDE of mock-ups with artificial defects.

6. Conclusion

Analysis of R&D activities on the ITER divertor PFCs brings the following conclusions:

- during recent years significant progress in the development of CFC and W armoured components with higher resistance to high incident heat fluxes was achieved. For all components there are several options (tile design, joining technologies) which are able to withstand heat fluxes higher than design values;
- large scale mock-ups (~1m) with relevant material combinations and manufacturing processes have been successfully manufactured;
- extensive studies of thermal-cycling and neutron irradiation resistance of components showed that armour lifetime can be longer than presently scheduled if the surface erosion problem is mitigated;
- the ITER Parties demonstrated a high level of cooperation in process development;
- the range of R&D activity and mobility of the ITER parties enabled an adequate and flexible response to the evolving divertor design
- further R&D activity is still required to improve reliability, minimise cost, improve the safety and lifetime of components.

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