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Carbon fiber composites application in ITER plasma facing components

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Abstract

Carbon Fiber Composites (CFCs) are one of the candidate armour materials for the plasma facing components of the International Thermonuclear Experimental Reactor (ITER). For the present reference design, CFC has been selected as armour for the divertor target near the plasma strike point mainly because of unique resistance to high normal and off-normal heat loads. It does not melt under disruptions and might have higher erosion lifetime in comparison with other possible armour materials. Issues related to CFC application in ITER are described in this paper. They include erosion lifetime, tritium codeposition with eroded material and possible methods for the removal of the codeposited layers, neutron irradiation effect, development of joining technologies with heat sink materials, and thermomechanical performance. The status of the development of new advanced CFCs for ITER application is also described. Finally, the remaining R&D needs are critically discussed.

1. Introduction

Carbon Fiber Composites (CFCs) are, together with beryllium and tungsten, candidate armour materials for the Plasma Facing Components (PFCs) of the International Thermonuclear Experimental Reactor (ITER) [1]. The process of selecting a suitable plasma facing material is a trade-off among several conflicting requirements driven by the specific working conditions of each component.

Carbon-based materials are widely used in many fusion devices. However, in ITER the plasma-wall interaction, in terms of particle energy flux and interaction time, will be orders of magnitude higher than in the present machines. Therefore, phenomena such as tritium retention, especially in the codeposited layers, chemical erosion, degradation of properties under neutron irradiation, etc., restrict the application of CFCs to the regions in direct contact with the plasma, such as the divertor vertical target and the dump plate, where only the use of CFCs can provide the needed erosion lifetime and high heat flux durability for the first ITER operational phase.

This paper describes the status of the application of CFCs for the ITER PFCs. The main design requirements and operational conditions are described, as well as the criteria for the selection of the reference CFC grades. Issues related to the use of CFCs such as erosion lifetime, tritium retention and codeposition, neutron irradiation effects, development of joining technologies and the results of the high heat flux testing are reviewed. The baseline properties of advanced CFCs, with or without silicon doping, are compared. Finally, based on

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the analysis of the issues related to the application of CFCs, the R&D needs are discussed.

2. ITER plasma facing components

A detailed description of the operational conditions and the design of the ITER PFCs can be found in [2,3]. The following armour materials have been selected for the different parts of the PFCs: Beryllium has been selected for the primary wall, the port limiter, the upper baffle and, as a back-up, for the divertor dome. The main reason for this choice is the better compatibility with plasma performance. Tungsten has been selected for the lower baffle, the divertor dome, the cassette liner and the upper part of the vertical target. The higher sputtering threshold and, as a result, the highest erosion lifetime under high charge-exchange (C–X) particle fluxes at energies below the threshold for physical sputtering lead to this choice. CFCs have been selected for the lower part of the divertor vertical target and energy dump target. The main reasons are the absence of melting at high power operational conditions (slow transients and disruptions) and therefore the potential for higher erosion lifetime, and the proven thermomechanical performance at high heat fluxes.

The operational conditions in the divertor area (vertical target and dump plate) covered by CFC armour are the following:

Maximum heat flux for partly attached plasma	$\leqslant$ 10 MW/m^2
Maximum heat flux during slow transient (10 s)	20 MW/m^2
Disruption energy density (duration ~0.1–3 ms)	~100 MJ/m^2
Expected max. neutron damage in CFCs (considering three replacement of the divertor during the Basic Performance Phase, BPP)	~0.1 dpa
Inlet coolant temperature	140°C
Inlet coolant pressure	4 MPa

The current design of the ITER divertor is shown in Fig. 1. For CFC armoured components, the monoblock geometry is selected as the reference option [3], the alternative option being the saddle type. This selection is based on the results from the high heat flux testing of different mock-ups (see Section 3.4). This geometry is more reliable under the divertor operating conditions (high heat flux at glancing angle with the PFC surface) [4]. The typical thickness of the sacrificial part of the CFC is ~15–20 mm (maximum 40 mm for high conductivity CFCs), the total surface area is ~100 m^2, and the CFC total weight needed for the manufacture of the first divertor is ~8000 kg.

3. Issues influencing the application of CFCs as armour for ITER PFCs

The specific issues associated with the operation of the ITER PFCs with CFC armour are: erosion lifetime due to chemical sputtering and evaporation during disruptions and slow transients, codeposition of tritium with redeposited carbon mainly in cool areas, durability of the CFC-heat sink material joints at the very high heat fluxes expected during slow plasma transients, and changes of the CFC properties due to neutron irradiation (mainly thermal conductivity).

3.1. Erosion lifetime of the divertor vertical target

An evaluation of the erosion lifetime [5] of the armour in the vertical target shows that it is determined primarily by (1) physical and chemical sputtering during normal operation, (2) evaporation and melting during disruptions and giant ELMs (only for metals), and (3) slow, off-normal transients. The component design lifetime is 3000 full power discharges, including 10% full power disruptions (100 MJ/m^2) and 10% high power transients (20 MW/m^2). In a calculations of the net erosion [6], a strong difference of erosion and redeposition fraction was found between attached, detached, and “radiative” regimes (the latter is characteristic of the ITER semi-detached plasma operation). A redeposition fraction of 90% is assumed for semi-detached operation, i.e. at the lower end of the range for the “radiative” regime [6]. For CFCs, the change in thermal conductivity by neutron irradiation over the life is included in the assessment [5]. The initial thickness of the plasma facing material is based on a heat flux of 5 MW/m^2 with a maximum front surface temperature of 1500°C (assuming 4 cm of Dunlop high heat conductivity CFC). For W alloy an initial thickness of 2 cm was chosen, for Be ~0.7 cm. The End-of-Life (EoL) thickness is 2 mm for all materials. The lifetimes depend strongly on the fraction of the melt layer lost following the transient or disruption heat load (for metals) and on the heat loads, [5]. For this analysis, a weak dependence of the chemical sputtering rate with the particle flux was assumed. The erosion of W alloy and Be were computed for a loss of 10% and 50%, respectively, of the melt layer during transients and disruptions. The results of the lifetime assessment are indicated in Fig. 2 in terms of remaining thickness vs. the number of shots to EoL.

For operation at 5 MW/m^2 (weakly detached or highly radiating partially attached plasma), the Be lifetime is dominated by slow transients, the CFC lifetime is influenced by all three processes and the W alloy lifetime is dominated by disruptions. The Be lifetimes for divertor plate operation is inadequate, the CFC and W lifetimes are acceptable. At this power density, the lifetime in normal operation is (i) 120–230 shots for Be, (ii)

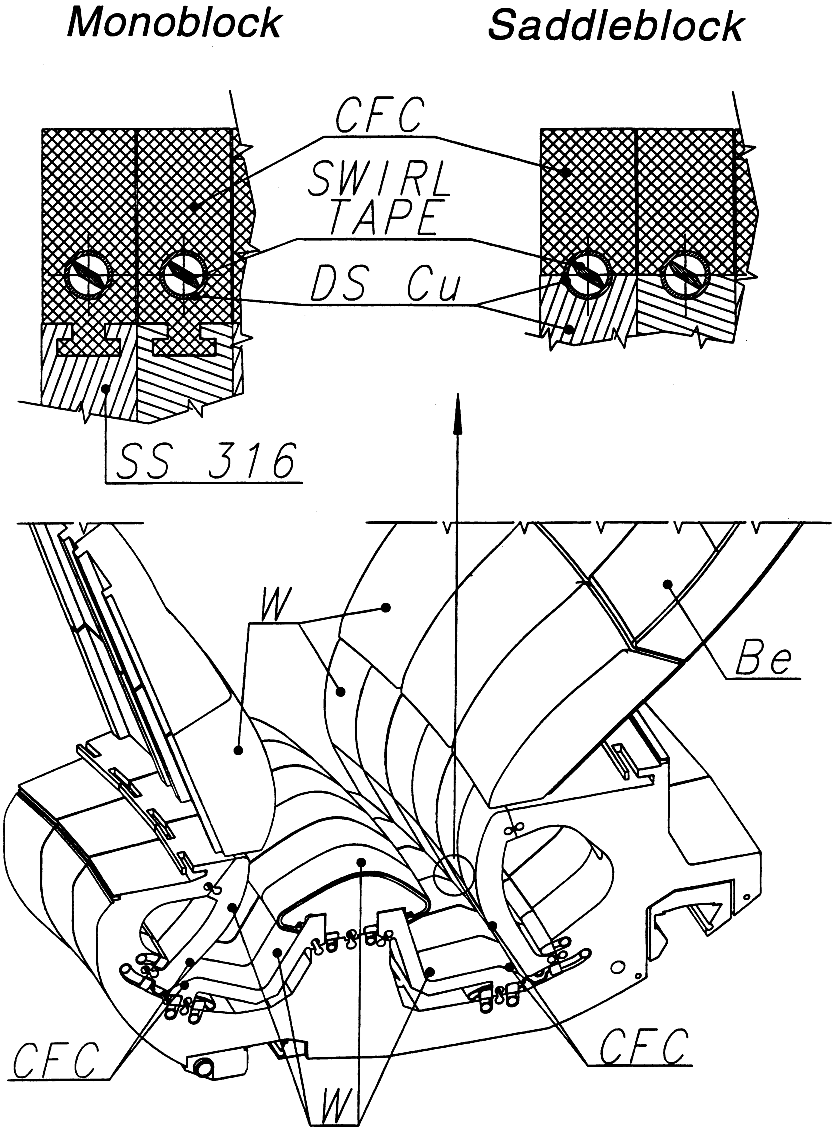

Fig. 1. General view of the current design of the ITER divertor.

2400–7700 shots for W alloy depending on melt layer loss, and (iii) 5500–8200 shots for CFC (high conductivity Dunlop material) depending on the local power. Even for more frequent slow transients, the analysis [5] indicates that 4000 shots for CFC and 2300–7800 shots for W alloy can still be attained.

In a subsequent assessment, a medium-conductivity CFC (grade SEP NS 31) has been used, since Si doping seems to reduce chemical sputtering (Section 4.2). The present assessment does not take into account this reduction and possible difference in the thermal response during disruptions and transients. This medium-conductivity CFC, optimised to permit occasional operation at 10 MW/m^2, would have an initial thickness of 17 mm (assuming operation for a short time at 1700°C is permissible) and a lifetime of 2400–2900 shots at 5 MW/m^2. Including the effect of some strike point motion and reduced chemical sputtering [7], the expected lifetime exceeds 3000 shots.

From the point of view of erosion lifetime of the divertor plates, for high heat flux components near the strike point, Be is therefore excluded; either CFC or W alloy can be used. The design choice at this location is CFC, since it is a low Z material and does not melt.

3.2. Tritium retention in CFCs

The presence of carbon could adversely affect tritium retention behaviour in ITER. It can result in high rates of tritium codeposition in the C-bearing complex

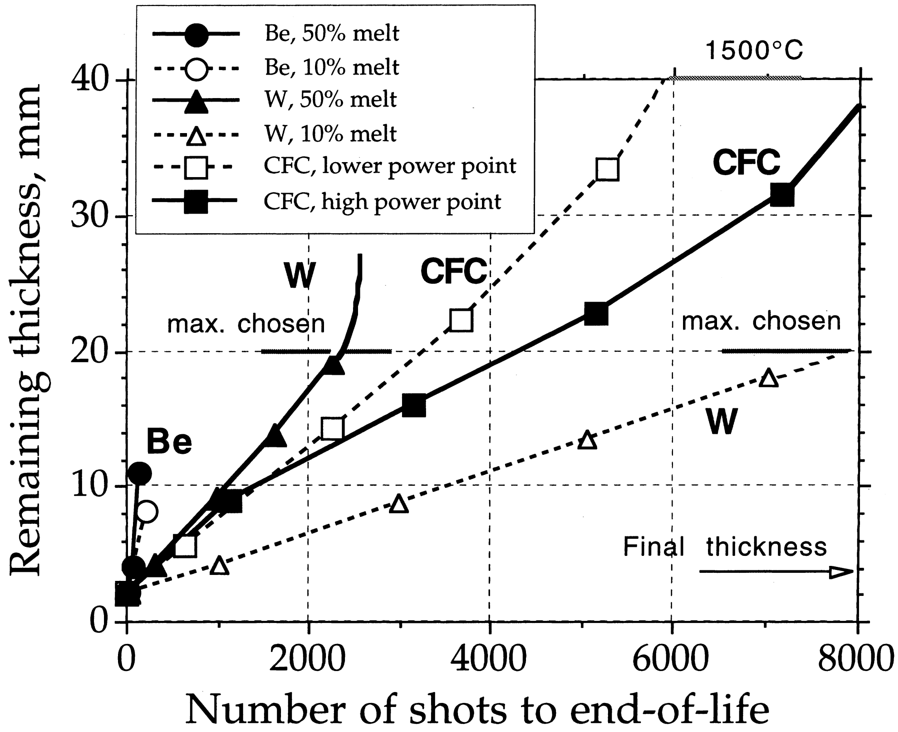

Fig. 2. Remaining thickness of PFC armour measured in number of shots to end-of life for Be, W alloy and Dunlop CFC.

mixed-materials which are expected to form due to erosion and re-deposition [8]. Additional processes that are responsible for the trapping of tritium in C-based materials are adsorption due to internal porosity, saturation of the implant area, and transgranular diffusion with trapping at high temperatures. However, for the present PFC configurations, (i.e., less than 100 m^2 of C with a probably much lower area subject to intense plasma-wall interaction) the contribution of implantation and diffusion in the bulk are deemed to be modest, even in the presence of substantial neutron trapping. Thus the primary concern in ITER remains tritium codeposition.

A considerable level of uncertainty exists in quantifying the overall codeposition rate in ITER and the nature and location of the codeposited films. Simulation of ITER divertor conditions, based mainly on detailed modelling studies of the plasma edge and wall interaction, performed by Brooks et al. [6], predicts a tritium codeposition of 1–20 g/shot. These levels of retention are consistent with those observed in short-pulse present-day tokamaks [8].

Recently, the large tokamaks TFTR and JET, both covered with carbon-based materials, provided precise measurements based on the accounting of tritium in D-T experiments. These studies show that the dominant long-term mechanism for retention in existing tokamaks is codeposition of carbon with hydrogenic fuel species varying from 10% to 40% of the amount of fuel introduced [9]. They also show that the distribution of the deposited material within the machines varies substantially according to the geometry, plasma conditions, etc. Of particular significance for ITER is the observation made recently in the JET Mk II divertor [9] where films and flakes of deuterium-saturated material were found in cooled regions behind the divertor pumping slot. From analysis of these films and flake samples collected, it was concluded that about ~4% of the total fuelling was retained in the films and flakes. This geometry has similarities to the ITER situation, which will have large flow of particles from the divertor slot into the region of the (cold) plenum.

Because of the tritium supply limitation for operation and due to hazard associated with potential mobilisation and release of trapped codeposited tritium to the surrounding environment during an accident, strict control of the in-vessel tritium inventory is needed in ITER. It is now assumed that the codeposited tritium inside the torus should be limited to 1000 g. This requires an armour material with reduced erosion (e.g. SEP NS31) and also a component design that minimises the tritium uptake. Additionally, efficient techniques need to be developed for tritium removal. Several methods are being considered, for example high temperature (>300°C) baking of the divertor system in an oxygen atmosphere, or low-pressure plasma discharge cleaning with oxygen

(ECDC, ICDC, GDC). For some of the methods, preliminary results are encouraging. However, future experiments need to be conducted to simulate conditions of relevance and applicability to ITER design. The discussion of these methods is included in [10].

3.3. Neutron irradiation influence

Neutron irradiation induces significant changes of the properties of carbon based materials [11–13]. In the current design, CFCs are used relatively far from the burning plasma. The total expected neutron damage is only ~0.1 dpa (taking into account 3 planned exchanges of the divertor during the BPP). This damage level is relatively low and the changes in properties are not expected to be critical (except for thermal conductivity).

Volume change: The dimensional stability of the CFCs is an important factor because of the additional stresses introduced in the material. Generally, CFCs with three-dimensional (3-D) structure, pitch fibers, small cell size and high final graphitization temperature have the highest dimensional stability. For the expected fluence at 300–1200°C the dimensional change (shrinkage) would be less than ~0.1–~0.2% [12,14].

Coefficient of thermal expansion: On the basis of the limited data available for multidirectional CFCs [15] and for nuclear graphites [12] it can be concluded that the change of the coefficient of thermal expansion will be negligible.

Strength and elastic modulus: Irradiation increases the Young modulus and the strength; a change as high as ~30–40% could be expected for fluence of ~1 dpa. Again, for the ITER BPP fluence the change is negligible [13].

Thermal conductivity: Irradiation decreases the thermal conductivity of carbon based materials and this effect depends of irradiation temperature and fluence. The thermal conductivity change has to be known for a correct prediction of the thermal response of CFC armours. The available data on the influence of neutron irradiation on the thermal conductivity have been collected and analysed [13–22]. An empirical equation describing the change of thermal conductivity as a function of the fluence and irradiation temperature has been proposed [23]. Data at low irradiation temperatures, ~150–350°C, are very limited and predictions in this range are very difficult. The observed saturation of the change of thermal conductivity [19,21] with fluence is included in the analysis. Fig. 3 shows the calculated thermal conductivity of the CFC SEP NB31 at different neutron fluences. These values can be used for the prediction of the thermal response of CFC armour.

Slow and fast thermal transients are expected to occur during ITER operation. The CFCs, irradiated at lower temperatures during normal shots, will be heated for a short time to high temperature. In the high temperature areas of the armour (T > 800–1000°C), the radiation defects could be partially annealed (depending

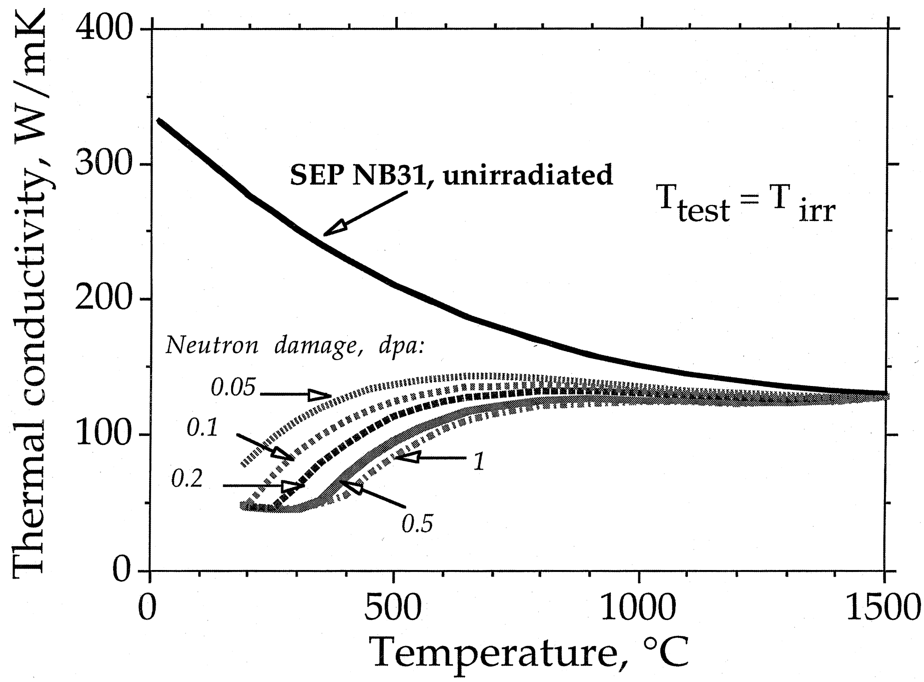

Fig. 3. Calculated thermal conductivity of SEP NB31 at different neutron damage values.

on time), with an increase of thermal conductivity. However, annealing of the irradiation defects will not be possible near the cooling channels where the temperature is low (~300–600°C). The data base for the description of the behaviour of the thermal conductivity at temperatures higher than the irradiation temperature is very limited [16,22,24]. More detailed experimental investigations of the annealing of the radiation defects and recovering of the thermal conductivity are needed, especially at high temperatures and short heating time.

The main difference between the fusion and fission neutron spectra is the He production rate in CFCs. The neutron energies for He production are 7 and 9 MeV [25], which is difficult to obtain in fission reactors. The estimated He generation rate is ~400 appm He/dpa for the ITER neutron spectrum. The influence of He on the properties of CFCs remains to be studied.

3.4. Manufacturing and high heat flux testing of CFC armoured components

In the design of the vertical target and dump plate, CFC tiles are bonded to an actively cooled Cu alloy heat sink. There is a large experience in the development of the joining technologies of carbon based materials to different metals [26]. In the ITER EDA, two additional requirements have been added: (1) avoidance of Ag brazing alloys due to of the transmutation to Cd which is undesirable for high vacuum systems and also has high activation; (2) use of high strength, high conductivity copper alloys for the heat sink.

The R&D program carried out by the ITER Home Teams has been very successful in developing different silver-free joining technologies with three types of armour designs (monoblock, saddle block and flat tile). A short description of the best joining technologies is presented below:

- Active Metal Casting (AMC), originally developed by Plansee AG for Tore Supra limiters [27]. The technology consists of a special treatment of the CFC surface to increase the strength of the joints, followed by casting of pure Cu onto the CFC surface. AMC was applied to monoblock and flat tile configurations. For the monoblock configuration, the AMC Cu layer was brazed to the heat sink using a Ti braze (880°C, 5 min). For flat tiles, the same joint was

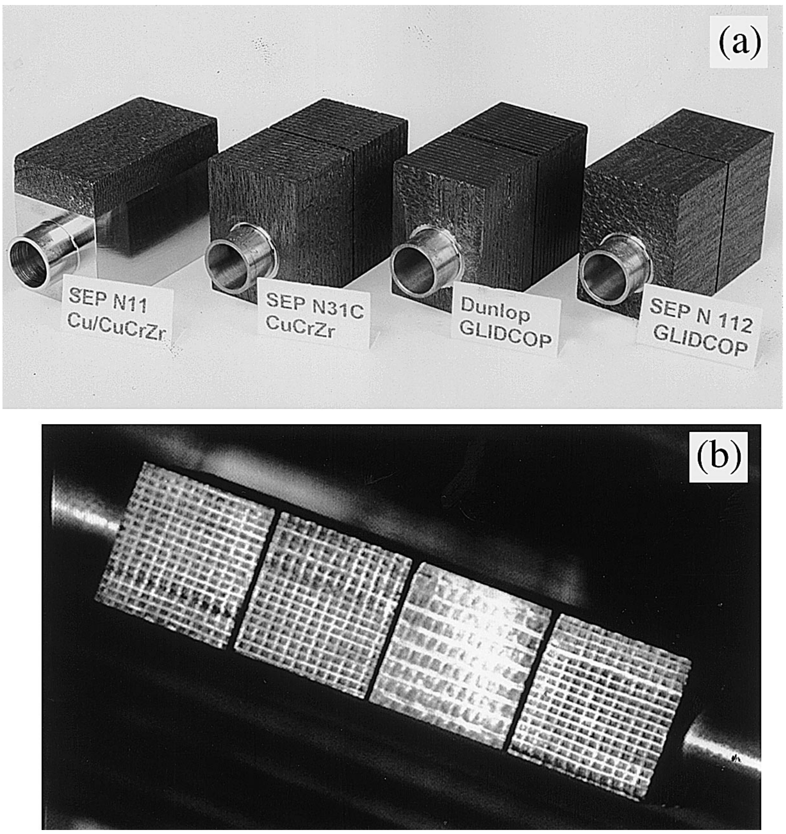

Fig. 4. General view of CFC armoured mock-ups: (a) manufactured by Plansee AG; (b) manufactured by JAERI.

obtained by e-beam welding. AMC technology was applied to different CFC grades (SEP N11, SEP N31, SEP NS31, Dunlop Concept 1) [28].

- Brazing with Cu–Mn brazing alloy [29], developed by JAERI. This technology was applied to the saddle block configuration, with a brazing temperature of ~920°C. The heat sink was made of DS Cu with pure Cu compliant layer.

Many mock-ups have been manufactured (examples are shown in Fig. 4) and tested. The most relevant results of high heat flux testing are presented in Fig. 5. AMC with Ti braze for the monoblock geometry, Cu–Mn braze for the saddle type geometry, and AMC with e-beam welding for flat tiles, have demonstrated excellent high heat flux behaviour, exceeding the ITER requirements. The monoblock geometry turns out to be the most robust solution: a mock-up has survived ~24 MW/m^2 absorbed power at 1000 cycles without any damage [28].

In comparison with Be and W armoured components, the high heat flux durability of the CFC/Cu alloy joints is significantly higher [28], and moreover the monoblock geometry (which up to now has been developed only for CFCs) results in a more robust solution when the heat fluxes are oriented at a glancing angle to the surface of the PFCs, flat tiles having a higher risk of failure.

The influence of neutron irradiation on the high heat flux performance of the CFC/Cu alloy mock-ups was studied for the first time by the Jülich Team [30]. The results indicate that neutron irradiation (~0.3 dpa at 350°C) does not lead to significant damage of CFC/Cu joints. The thermal performance of irradiated mock-ups was different compared with unirradiated mock-ups, because of the change of the thermal conductivity of CFCs induced by neutron irradiation.

4. Requirements for the selection of the reference grades and development of advanced CFCs

Among the different existing carbon based materials, carbon fiber composites have been selected as reference due to their high thermal shock and thermal fatigue resistance (low crack propagation) and their high thermal conductivity in comparison with conventional graphites. For the selection of the reference CFC grade, the following criteria have been applied:

- *Thermal conductivity:* minimum allowable initial thermal conductivity ⩾ ~300 W/mK at room temperature and ⩾ ~150 W/mK at 1000°C. The higher the thermal conductivity, the larger the sacrificial thickness and hence the higher the erosion lifetime.

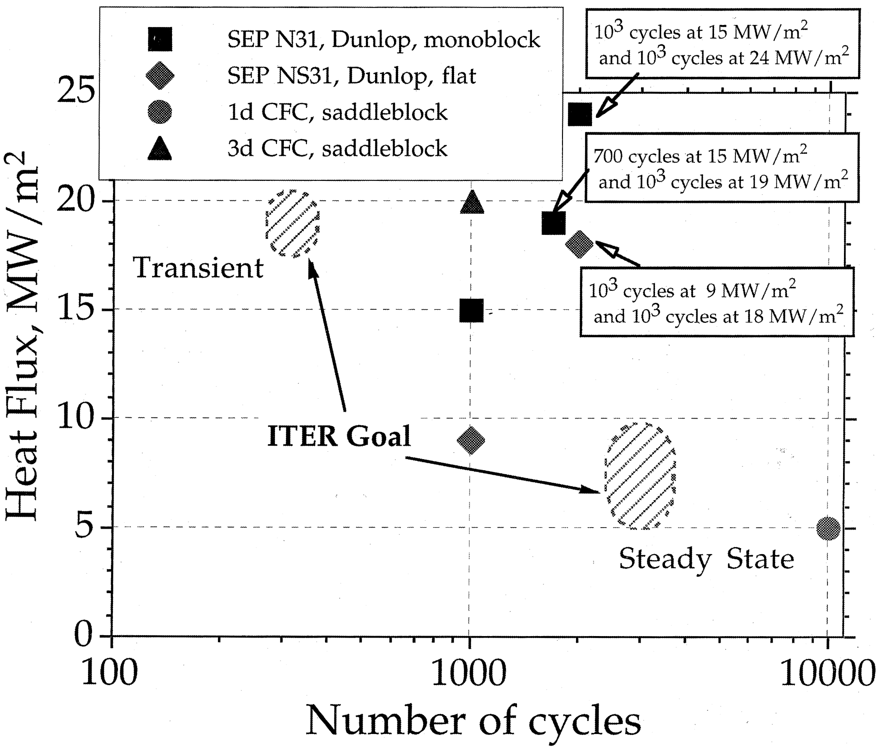

Fig. 5. Results of the high heat flux testing of the CFC armoured mock-ups [28,29].

Thermal erosion is also less for CFCs with higher thermal conductivity in experiments simulating disruption erosion [31];
- *Architecture:* 3-D CFCs are preferable because they have more isotropic properties and higher thermal shock resistance in comparison with 1-D and 2-D materials [32];
- *Density and porosity:* higher density and lower porosity are preferable to minimise gas absorption and outgassing;
- *Chemical erosion:* doped CFCs could possibly reduce chemical erosion, tritium retention and chemical reactivity;
- *Impurity content:* as low as possible to avoid plasma contamination;
- *Oxidation resistance:* higher oxidation resistance is generally desirable for safety reasons;
- *Cost and availability:* reasonable cost and good availability at industrial scale.

4.1. Development of advanced undoped CFCs

Recently investigations of CFCs for high heat flux application have been carried out mainly by the Japanese and the European ITER Home Teams. The activity is primarily aimed at developing advanced CFCs with improved properties for this specific application. The new advanced CFCs are:
- NIC-01, developed for JAERI by the Nisseki Corp., Japan, [33];
- Dunlop Concept 1, 2 and 3, developed for The NET Team by Dunlop, UK [34];
- SEP N31 (A, B, C) developed for The NET Team by SEP, France [35,36].

A short description of these materials and a summary of their properties are presented in Table 1. The temperature dependence of their thermal conductivity is shown in Fig. 6. Due to variations in the manufacturing technology the guaranteed thermal conductivity might be ~10% less. Most of these materials have been successfully joined to Cu alloys and have shown excellent thermomechanical performances.

4.2. Development of the advanced doped CFCs

It is well known that some doped carbon based materials have lower chemical erosion in comparison with pure carbon. The elements which reduce the chemical erosion are Si, B, Ti etc. [37]. To improve the resistance to chemical erosion and to reduce tritium retention of the conventional CFCs, Si doped CFCs (SEP NS31) have been developed for the NET Team by SEP, France [36,38]. SiC doped CFCs have been developed by Tonen, Japan, [39].

Some properties of silicon doped CFC SEP NS 31 are given in Table 1 and Fig. 6. In SEP NS31, part of the Si introduced during liquid impregnation forms SiC (~8–10 wt%) and the resulting material consist of pure Si, SiC, carbon and carbon fibers. It has been demonstrated that chemical erosion of this material is reduced by a factor of 2–2.5 in comparison with pure carbon [7,40].

The other features of Si doped CFCs have to be confirmed by an extensive R&D program. As it was mentioned before the critical issue for application of carbon based materials is tritium retention. It was demonstrated that addition of the pure Si reduced the thermal hydrogen retention [38]. However the amount of retained implanted deuterium in NS 31 did not saturate with ion's fluence [40]. There are no data on the tritium retention in the Si doped codeposited layers. Outgassing rate and oxidation resistance of Si doped CFCs are better in comparison with undoped CFCs [38,41]. No data are available on neutron irradiation effects on the properties of doped CFCs. The final decision on the application of Si doped CFCs will be taken when all issues will be clarified.

5. Required R&D

As discussed in this paper, there are several critical issues related to the application of CFCs for ITER PFCs. The following issues require further work to:
- improve the understanding of the mechanisms of tritium retention at ITER relevant conditions and provide a more precise assessment of the possible total tritium inventory. In this respect the development of efficient techniques for tritium removal from codeposited layers is very important;
- improve the understanding of the mechanisms of erosion of CFCs. This includes chemical erosion at relevant conditions, thermal erosion during slow transients and disruptions;
- study the neutron irradiation influence in the thermal properties, especially:
  thermal conductivity changes at low irradiation temperature (150–300°C);
  behaviour of the thermal conductivity of irradiated materials at high temperatures (slow transient and disruptions) and the possibility to improve the thermal conductivity by annealing;
- continue the development of the joining technologies with the goal of consolidating the positive results to more relevant geometries, improving the joint reliability, determining the fatigue limits (including neutron irradiation) and testing the behaviour of joints in all off-normal conditions.
- continue the development of new advanced doped CFCs.

Table 1
Thermal and mechanical properties of the advanced CFCs [33–36]

Properties	Temperature, °C	Dunlop C1 *z/x/y**	Dunlop C2 *z/x/y**	SEP NB31 *z/x/y**	NIC-01 *z/x/y*	SEP NS11 *z/x/y**	SEP NS31 *z/x/y**
Thermal conductivity, W/mK	RT	430/102/76	377/118/76	323/117/115	537/n/a/n/a	221/212/173	304/100/91
	800	198/53/37	180/61/41	154/58/55	224/n/a /n/a	102/94/77	145/55/48
Specific Heat, J/kg K	RT	693	704	720	717	712	760
	800	1790	1823	1820	1843	1670	1720
Coefficient of thermal expansion, 10^{-6}/K		RT–400°C: −1.32/0.07/3.08	RT–400°C: −1.05/0.07/2.15	RT–800°C: 0.4/1/2.1	RT–400°C: −0.6/0.6/0.87	RT–400°C: 0.21/0.46/1.64	RT–800°C: 0.4/1.7/3.3
		RT–700°C: −0.72/0.61/3.41	RT–700°C: −0.51/0.56/2.75	RT–1000°C: 0.5/1.2/2.7	RT–650°C: −0.2/1.07/2.07	RT–700°C: 0.81/1.08/2.12	RT–1000°C: 0.5/1.2/4.2
Tensile strength, MPa	RT	n/a	n/a	130/30/19	n/a	n/a	160/40/25
Tensile strain, %	RT	n/a	n/a	0.14/0.3/-	n/a	n/a	0.14/0.22/0.22**
Young's Modulus, GPa	RT	n/a	n/a	107/15/12	n/a	n/a	120/55/40
	1000			107/20/12	n/a	n/a	
Poisson's ration	RT	n/a	n/a	0.2/0.1/0.2	n/a	n/a	0.15/0.09//0.15
Compressive strength, MPa	RT	n/a	n/a	102/31/-	n/a	n/a	120/40/-
Density, kg/m^3	RT	1880	1820	1900	1960	2090	2000
Porosity, %	RT	6.63	6.32	8	7	1	5
Material description: type of fibers, etc.		3d, *z*: pitch P120, *x*: PAN, needled, graphitisation 2450°C	3d, *z*: pitch P130, *x*: PAN, needled, graphitisation 2450°C	3d, *z*: pitch P55 (27%), *x*, *y*: PAN (4%), needled, graphitisation 2800°C	3d, Granoc pitch fibers, *z*: 43%, *x*: 18%,*y*: 6%, graphitisation 3000°C	3d, *z*: pitch P25, *x*: PAN, needled; liquid impregnation of 10–12% Si	3d, z: pitch 55, *x*: PAN, needled; liquid impregnation of 10 ± 2% Si

z high conductivity direction (Pitch fibers).
*y** needling direction.
** Estimated values.
n/a – data not available.

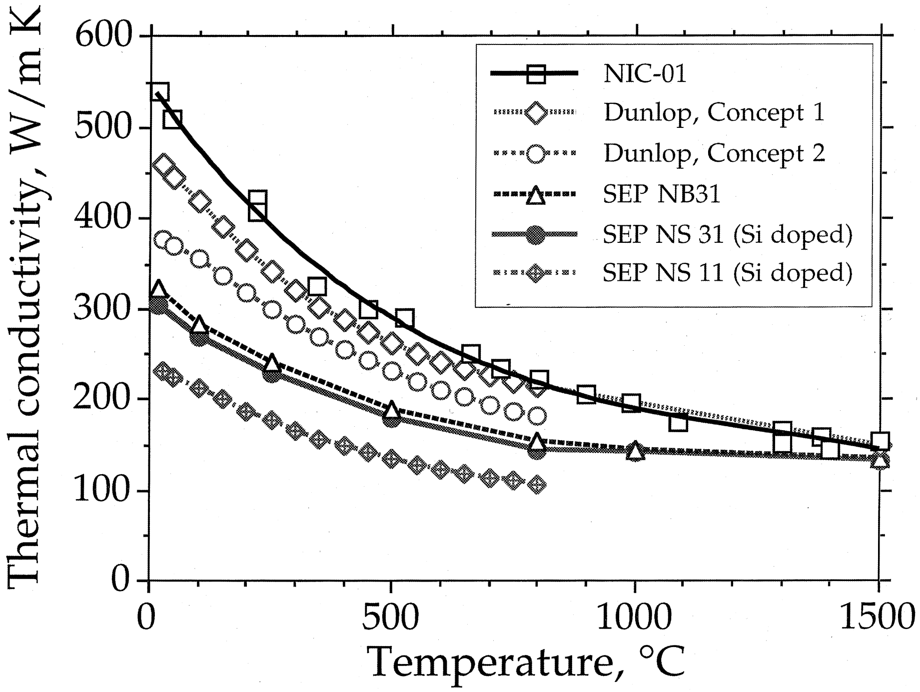

Fig. 6. Thermal conductivity of the recently developed advanced CFCs.

6. Conclusion

Carbon fiber composites have been selected for the divertor vertical target and the dump plate in the area of the plasma strike point. The main reasons for this decision are based on the demonstrated excellent thermomechanical performance of CFCs and the absence of melting, giving higher erosion lifetime compared with Be and W:

- The CFC erosion lifetime, including chemical sputtering, evaporation under slow transient events and disruptions, is ~3000 cycles for a sacrificial thickness of 20 mm;
- The excellent thermomechanical performance of CFC armoured mock-ups has been widely demonstrated. However, more activity is still needed in this area, with the goal of optimising the joining technologies and improving the joint reliability.

A critical issue in the application of CFCs as divertor armour is the tritium codeposition. The design of the ITER divertor components is focussed towards the reduction of the tritium inventory by the use of materials with low erosion and low tritium retention and to develop efficient methods for tritium removal.

For the current design of the ITER divertor the neutron irradiation affects mainly the thermal conductivity of the CFCs. More data is needed to predict the thermal response of irradiated CFCs, especially at low irradiation temperature and at temperatures above the irradiation temperature for typical for slow transient and disruption conditions.

There has been a significant progress in the development of advanced CFCs for fusion applications. An advanced Si-doped CFC has also been developed and proposed for ITER application. This Si-doped CFC has promising properties such as reduced chemical erosion. The studies of other properties related to the ITER conditions are in progress. Assuming positive results, this material would have a high priority for application in ITER.

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