EU DEVELOPMENT OF HIGH HEAT FLUX COMPONENTS

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The development of plasma facing components for next step fusion devices in Europe is strongly focused to ITER. Here a wide spectrum of different design options for the divertor target and the first wall have been investigated with tungsten, CFC, and beryllium armor. Electron beam simulation experiments have been used to determine the performance of high heat flux components under ITER specific thermal loads. Beside thermal fatigue loads with power density levels up to 20 MWm^{-2}, off-normal events are a serious concern for the lifetime of plasma facing components. These phenomena are expected to occur on a time scale of a few milliseconds (plasma disruptions) or several hundred milliseconds (vertical displacement events) and have been identified as a major source for the production of neutron activated metallic or tritium enriched carbon dust which is of serious importance from a safety point of view.

The irradiation induced material degradation is another critical concern for future D-T-burning fusion devices. In ITER the integrated neutron fluence to the first wall and the divertor armour will remain in the order of 1 dpa and 0.7 dpa, respectively. This value is low compared to future commercial fusion reactors; nevertheless, a non-negligible degradation of the materials has been detected, both for mechanical and thermal properties, in particular for the thermal conductivity of carbon based materials. Beside the degradation of individual material properties, the high heat flux performance of actively cooled plasma facing components has been investigated under ITER specific thermal and neutron loads.

I. Introduction

From a heat flux point of view, the divertor in future fusion devices such as ITER represents the most critical plasma facing component. It has to withstand 3000 cycles at a power density level up to about 3 MWm^{-2} for the upper part of the vertical target which will be manufactured from flat W-tiles or W-monoblocks attached to a CuCrZr-heat sink. The lower straight part of the divertor will be subjected to even higher thermal loads (up to 20 MWm^{-2} for about 10 seconds in 10% of plasma discharges during the so-called 'slow transients').

Beside thermal fatigue loads, which mainly influence the integrity of the joint between the plasma facing armor and the heat sink, also transient events which occur during off-normal plasma operation with deposited energy densities up to several tens MJm^{-2} have been investigated experimentally.

Another cause of concern is the degradation of the material properties due to the flux of 14 MeV neutrons. Since qualified test facilities which allow the experimental investigation of neutron effects in this energy range are not available, neutron irradiation experiments have been performed in material test reactors with ITER specific neutron fluences up to 1 dpa. Beside property changes the experimental analyses were mainly focused to a critical evaluation of the high heat flux performance of actively cooled components after neutron irradiation.

The present paper will only focus on the development and experimental evaluation of plasma facing components (PFC) for ITER [1]; the material aspects of other European confinement experiments with high performance PFCs such as Tore Supra or Wendelstein W-7X have been presented elsewhere [2, 3].

II. Plasma facing component development

In principle the design activities for both the divertor and the shield blanket modules follow roughly the same general pattern which is shown schematically in Figs. 1 and 2. The major steps of the R&D activities include the design selection, the qualification of the materials for the plasma facing armour and for the heat sink, the development and improvement of reliable joining techniques. Step-by-step iterations resulted in the production of numerous small scale mockups which were subjected to non-destructive qualification tests and to extensive high heat flux testing, preferably in electron beam test devices. In a further step, selected material samples and small-scale modules were irradiated in material test reactors to ITER specific fluences [4]. In a final step, medium and full-scale components have been manufactured by the European industry. These prototype components were exposed to cyclic ITER relevant thermal loads (divertor) or are now ready for fatigue performance testing (blanket modules).

During the past few years the present design of the ITER divertor has received a well-engineered, technically mature status; this has largely been achieved by the implementation of extensive results from the European R&D programme and in collaboration with other international partners. Taking into account the progress and the experience from European industry, the highly loaded vertical targets of the divertor will be manufactured from a three directional CFC and from tungsten monoblock armour with integrated CuCrZr coolant tubes. These monoblocks are mounted onto a stainless steel structure.

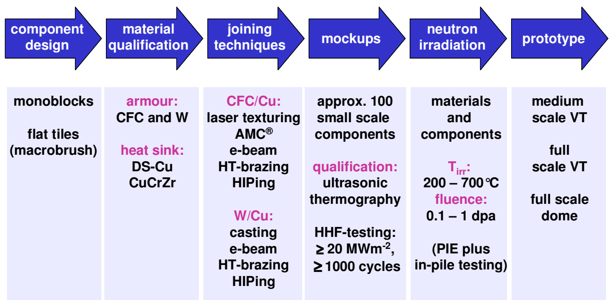


Fig. 1. Schematic presentation of the step-by-step development of the divertor targets

In a joint effort the European industry and the associations have pursued the parallel development and qualification of the so-called monoblock and the flat tile design. In the frame of this study the relevant armour and heat sink materials have been qualified. In addition, a wide spectrum of different joining techniques were applied to the most promising material candidates: firstly the plasma facing material is metallized by a copper casting process (the interface of CFC tiles is textured by a high power laser); secondly, joining of the two partners is established by e-beam welding, high temperature brazing or hot isostatic pressing (HIP). Finally the quality of the bond is being assessed by non-destructive analyses or extensive high heat flux experiments (thermal fatigue testing and simulation of transient events).

Owing to the superiority of the monoblock design in terms of the high heat flux removal efficiency, manufacturing of medium and full scale components by the European industry was based on this design option, i.e. tungsten and CFC monoblocks have been utilized for the vertical target using DS-Cu or CuCrZr for the coolant tube. Meanwhile these components have been tested successfully under ITER specific thermal loads.

In contrast to the divertor, the design of the shield blanket modules has undergone substantial modifications during the last few years. To complement the official ITER design, the European home team has proposed a solution with a mechanical attachment to the shield blocks using studs, a straight forward concept which is simple, more flexible in terms of special design requirements, relatively easy to manufacture and less costly. The separation of the first wall from the shield block and the splitting of this component into a number of smaller panels with smaller dimensions (approx. 0.25 m^2) also allows to use brazing processes (by induction or in furnace) as alternative joining techniques instead of a HIP process.

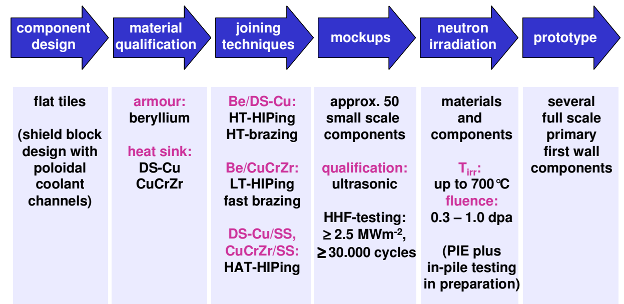


Fig. 2. Step-by-step development of primary first wall components

The reduced panel size also smoothened the way to use precipitation hardened copper (CuCrZr) instead of the more expensive dispersion strengthened copper (DS-Cu, e.g. CuAl25). The application of the less brittle CuCrZr alloy however requires a careful material processing to retain the favorable mechanical properties of this material grade [5].

Finally, a number of qualification tests including neutron irradiation studies and thermal shock tests have been performed on different beryllium grades for the plasma facing armour. Following a careful pre-selection process, actively cooled mockups have been manufactured by brazing or HIPing using both, CuCrZr and CuAl25 as a heat sink layer, and stainless steel 316L(N) as a back plate. Beside these miniaturized test modules also a number of full scale modules have been provided by the European industry.

III. Simulation of ITER relevant heat loads

Plasma facing components in future fusion devices will be subjected to intense particle fluxes by ions, electrons and charge exchange neutrals. In addition high energetic neutrons originating from the D-T-fusion reaction will cause severe volumetric damage to the plasma facing components. The particle bombardment in the eV- and keV-range will generate primarily surface erosion due to a number of plasma surface interaction processes (physical and chemical sputtering, radiation enhanced sublimation etc.) [6]. In addition a substantial amount of heat will be deposited on the PFCs surfaces, i.e. on the first wall and in particular on the divertor targets. Table I specifies these heat fluxes for the present design of ITER. Peak surface heat loads are expected to occur in the separatrix strike zone on the vertical targets with up to 10 MWm^{-2} on the CFC armour. During off-normal operation and a limited number of plasma discharges this heat flux may double for a duration of approx. 10 s. Thermal loads to the first wall region during normal plasma operation are about one order of magnitude below the divertor fluxes; nevertheless, intense transients (although less intensive compared to the divertor region) may still play an important role.

Damage to actively cooled plasma facing components is mainly due to the cyclic character of the plasma pulses which gives rise to thermal fatigue damage; typical cycles numbers for ITER are in the order of several 10^4 pulses; under the prevailing conditions a multiple replacement of the divertor armour is indispensable.

TABLE I. Wall loading in ITER

	divertor	first wall
normal operation		
peak surface heat flux / MWm^{-2}	10 (CFC) / 5 (W)	0.5 (Be)
duration / s	≤ 450	≤ 450
number of cycles	3.000	30.000
Slow transients		
peak surface heat flux / MWm^{-2}	20 (on CFC)	---
duration / s	10	---
frequency / %	10	---
disruptions		
peak surface heat load / MJm^{-2}	10 - 30	1
duration / ms	0.1 - 3	0.1 - 3
frequency / %	10	10
vertical displacement events		
peak surface heat load / MJm^{-2}	60	---
duration / ms	100 - 300	---
frequency / %	1	---
edge localized modes (ELMs)		
peak surface heat load / MJm^{-2}	1	?
duration / ms	0.1 – 0.5	0.1 – 0.5
frequency / Hz	1	1
neutron fluence		
displacements per atom / dpa	0.7 (W, CFC) 2 (Cu)	1 (Be) 3 (Cu)
(not considering divertor replacements)		

Further material damage is initiated by short, intense transient heat loads which occur during plasma disruptions or vertical displacement events (VDE) on a time scale of a few or a few hundred milliseconds, respectively; the deposited energy density is expected to reach values up to 60 MJm^{-2}. Due to the relatively high frequencies of these events (1 in 10 shots for the disruptions or 1 in 100 pulses for the VDEs) a non-negligible material damage (erosion, cracking and/or melting) will occur. Edge localized modes (type I ELMs) with sub-millisecond pulses which occur at energy densities of about 1 MJm^{-2} and a frequency of ≈ 1 Hz in the plasma edge may trigger additional damage; systematic experiments to investigate these events are on the way in collaboration with the Troitsk Institute for Innovation & Fusion Research, Russia [7].

Compared to a commercial power reactor for electricity generation, the next step device ITER will have manageable neutron fluence with peak values of approx. 1 dpa for the plasma facing armour and up to 3 dpa for the copper alloy in the heat sink.

To investigate these thermal loads experimentally electron beam test facilities (e.g. JUDITH, Juelich, Germany; FE-200, Le Creusot, France) are used routinely. These test devices are rather flexible and allow to simulate most of the above mentioned thermal fatigue and thermal shock loads with ITER relevant parameters. In addition, the JUDITH facility also allows to perform tests on neutron irradiated components; toxic materials such as beryllium can be investigated without major restrictions. The new test facility JUDITH II (fig. 3) at the Research Center Juelich will enable also the simulation of very short events (ELMs) with high repetition rate.

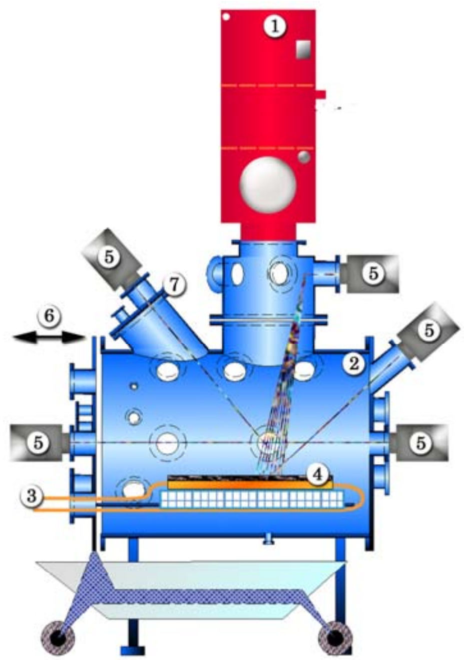
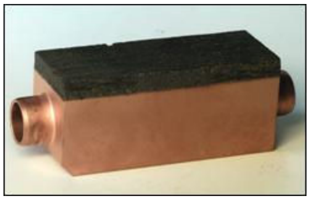
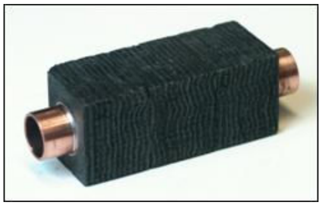


Fig. 3. New 200 kW e-beam test facility JUDITH II
1 – electron beam gun, 2 – vacuum chamber,
3 – high pressure coolant loop, 4 – PFC,
5 – diagnostics, 6 – movable vessel door / stage,
7- alternative position for the electron beam gun

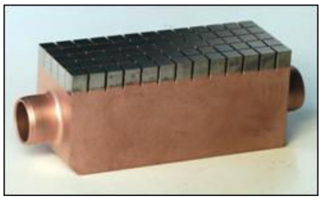
Since qualified neutron sources for the irradiation with energetic (14 MeV) neutrons are not available, the degradation of plasma facing materials and components under ITER specific conditions has been investigated in material test reactors [8] (e.g. HFR Petten, The Netherlands); the post irradiation examination (PIE) of the irradiated specimens includes measurements of the



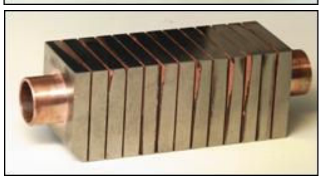
CFC flat tile mock-up
active metal casting of CFC
(silicon doped NS31)
e-beam welding to CuCrZr
heat sink
HHF fatigue testing:
1000 cycles @ 19 MWm^{-2}



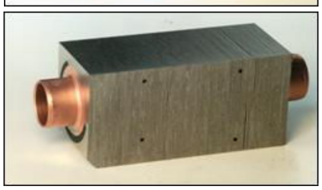
CFC monoblock
drilling of CFC tiles
(SEPcarb NB31)
active metal casting (AMC®)
low temperature HIPing
HHF fatigue testing:
1000 cycles @ 25 MWm^{-2}



W macrobrush mock-up
coating of WLa_2O_3 tiles with
OFHC-Cu, e-beam welding
to CuCrZr heat sink
HHF fatigue testing:
1000 cycles @ 18 MWm^{-2}

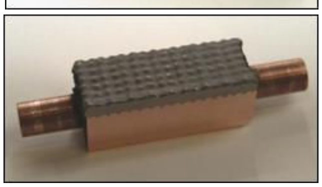


W monoblock mock-up
drilling of W-La_2O_3 monoliths
casting with OFHC-Cu
HIPing
HHF fatigue testing:
1000 cycles @ 20 MWm^{-2}

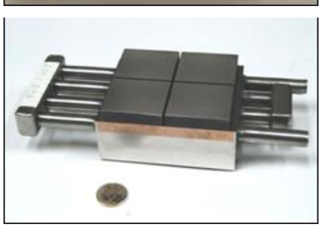


W monoblock mock-up
(lamellae technique)
drilling of W sheets
casting with OFHC-Cu
low temperature HIPing
HHF fatigue testing:
1000 cycles @ 18 MWm^{-2}

PS-W mock-up
vacuum plasma spraying of
tungsten
CuCrZr heat sink
HHF fatigue testing:
1000 cycles @ 5.5 MWm^{-2}



PS-Be mock-up
vacuum plasma spraying
of beryllium (5 mm thick)
manufactured by R. Castro,
Los Alamos Nat. Lab.
HHF fatigue testing:
1000 cycles @ 3.0 MWm^{-2}



Primary first wall mock-up
Be tiles (42 x 47 x 10 mm^3)
CuCrZr heat sink (10 mm)
with 316L coolant tubes
316L backing plate (30 mm)
HHF fatigue testing:
1000 cycles @ 1.5 MWm^{-2}

Fig. 4. Survey of small scale mockups with different plasma facing armour (CFC, tungsten, beryllium) and different design options (flat tile components, monoblock design and plasma sprayed modules)

thermal and mechanical parameters as well as high heat flux tests in the hot cell facility JUDITH. Furthermore in-pile tests; i.e. the simultaneous loading of actively cooled mockups with neutrons and high thermal loads have been performed in the Russian SM-2 fission reactor in Dimitrovgrad.

III.A. Thermal fatigue testing

More than approx. 100 small scale divertor components have been manufactured by the European industry or by the associations. These cover different design options (flat tile, monoblock) and different joining techniques for both, CFC and tungsten armour [9]. In the following a survey of selected plasma facing component with CFC and tungsten armour for the divertor and with beryllium coatings/tiles for first wall applications are summarized; the major characteristics of these modules and the results for medium term thermal fatigue tests are listed in fig. 4.

The results which have been obtained so far can be summarized as follows:

- CFC flat tiles withstood cyclic thermal loads up to 19 MWm^{-2} for 1000 thermal cycles,
- CFC monoblocks have been tested up to 25 MWm^{-2} for 1000 cycles; the CFC armour on the medium scale vertical target sustained 20 MWm^{-2} for 2000 cycles in the FE-200 facility,
- tungsten flat tiles (macrobrush design) were tested at $\leq$ 18 MWm^{-2} (1000 cycles),
- tungsten monoblocks (drilled W-tiles and W-lamellae) withstood up to 20 MWm^{-2} for 1000 cycles,

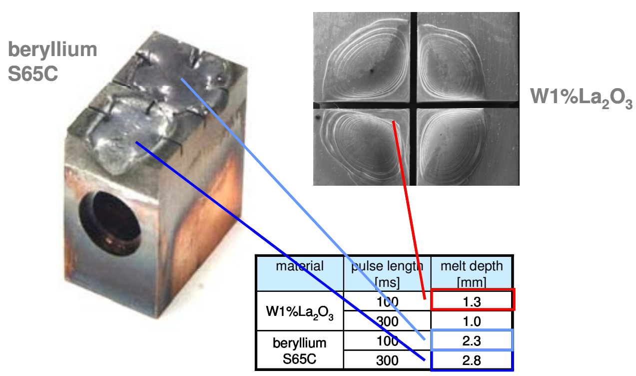
These data show very clearly that technical solutions for the divertor targets are feasible which meet or even exceeded the HHF requirements for ITER.

Due to the design change from the integrated first wall to separate primary first wall (PFW) panels a further production technique, namely HIPing of CuCrZr to stainless steel is made practical by fast cooling. Thus, precipitation hardened copper (CuCrZr) has become the prime candidate for the heat sink in PFM for the EU and additional efforts have been allocated to the development and thermo-mechanical testing of beryllium/CuCrZr-joints. Best performances obtained so far with HHF tests performed in JUDITH on a variety of mockups produced with different joining parameters have shown detachments of the Be tiles at 3 MW/m^2 after cyclic operation for successive 1000, 200, 200 and 200 cycles at 1.5, 2.0, 2.5, 2.75 MWm^{-2}, respectively.

High heat flux testing has also been performed on flat CuCrZr heat sink modules which were coated in a plasma spray process with tungsten and beryllium, respectively (see fig. 4). The PS-tungsten modules have shown a favorable thermal fatigue performance with peak heat loads of 5.5 MWm^{-2} without detectable failure. The Be-coated component didn't show any degradation of the heat removal efficiency up to 3 MWm^{-2}; however, some cracks developed perpendicular to the component's surface (i.e. parallel to the heat flux direction). These findings were predictable since both plasma sprayed mockups have not been castellated so far.

III.B. Transient events

Beside the normal operation scenarios transient loading conditions also have been taken into consideration. Among these events (cf. table I) the so-called vertical displacement events (i.e. the malfunction of the plasma positioning system) may result in severe surface damage due to short term (100 – 300 ms) thermal loads to plasma facing components. Such an event with a deposited energy density of 60 MJm^{-2} (ITER) will mainly affect the surface of components with metallic PFMs (beryllium or tungsten). The melt layer thickness under these events was determined experimentally in electron beam tests and was found to be in the order of a few millimetres (depending on the pulse duration), see fig. 6. Mockups with un-doped CFC armour are more resistant under identical thermal loads since there is no liquid phase; however, some thermal erosion by sublimation has been detected.



material	pulse length [ms]	melt depth [mm]
W1%La$_2$O$_3$	100	1.3
	300	1.0
beryllium S65C	100	2.3
	300	2.8

Fig. 5. Electron beam simulation of vertical displacement events with a deposited energy density of 60 MJm^{-2} [10]

Part of the VDE-tests have been performed on actively cooled components with active cooling of the substrate; there is obviously no damage to the interface between PFM and heat sink. The beryllium mockup in fig. 5 was able to resist 1000 thermal fatigue cycles at an power density of 5 MWm^{-2} following the VDE testing.

More serious material damage is expected during plasma disruptions which occur on a millisecond timescale. For ITER about 10% of the discharges are supposed to be terminated in a plasma disruption. The published data about the expected amount of deposited energy density show some scatter; furthermore, part of the incident plasma energy is absorbed by a dense cloud of ablation vapour which forms above the heat affected surface area. Nevertheless, an absorbed energy density of several MJm^{-2} will be deposited on the PFC surface. Due to the rather short pulse duration ($\Delta t \approx 1$ ms) heat conduction into deeper parts of the PFM does not play any important role and the mayor damage is restricted to a thin surface layer with a thickness of several ten microns.

Under these conditions metallic plasma facing materials such as beryllium or tungsten will melt instantaneously; this mechanism is associated with the formation of bubbles in the melt layer and with the ejection of metallic droplets which finally will contaminate the plasma boundary layer or will be deposited in the form of metallic dust in the gaps behind the PFCs. From a safety point of view this process may generate non negligible amounts of toxic beryllium particles or highly activated tungsten dust which might need periodical removal to avoid safety issues.

The short pulse duration of disruption events will generate steep thermal gradients in the surface of the plasma facing material; this will induce severe thermal stresses which may generate cracks with a depth of several hundred microns and beyond. This effect is of special importance if the temperature of the heat effected material is below DBTT (ductile brittle transient temperature), i.e. at about 400°C for un-irradiated tungsten.

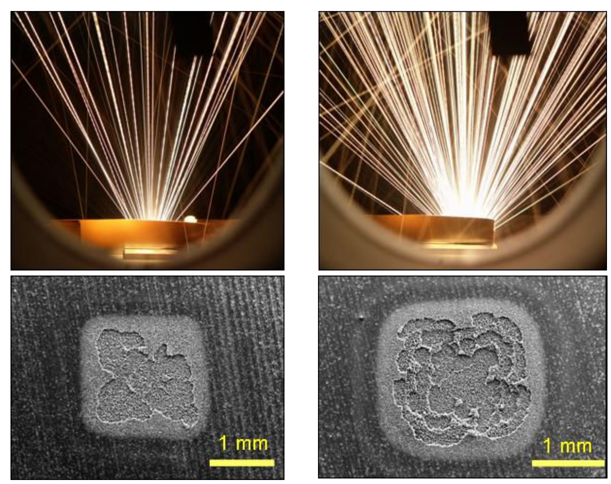


Fig. 6. Brittle destruction of isotropic fine grain graphite at power densities of 3.3. and 4.3 GWm^{-2}, $\Delta t = 2$ ms, top: macroscopic observation with particle trajectories, below: eroded test samples

In contrast to metallic PFMs carbon based materials such as graphites or CFCs do not melt; hence, the formation of dust particles via the above mentioned mechanism does not occur. However, brittle destruction (BD) [11], i.e. generation of thermally induced microcracks in the surface of these materials during intense thermal loads will result in the formation of carbon dust particles, if a critical threshold value of the incident beam power is exceeded. The brittle destruction mechanism is shown schematically in fig. 6 for two power density levels (3.3. and 4.3 GWm^{-2}) below the critical threshold value (< 3 GWm^{-2}) no particle emission has been observed. Up to a 2nd threshold value mainly small and medium sized particles are ejected from the surface of the plasma facing material. In fine grain graphites this process is characterized by the release of the binder phase between the graphitic grains. If the 2nd threshold value is exceeded large dust particles (grains or grain clusters) are emitted from the surface; digital images (cf. fig. 6) taken from this process show clearly the trajectories of the hot particles which may penetrate deeply into the plasma boundary layer. Mayor concern of the carbon dust is the codeposition together with tritium in gaps or remote areas behind the divertor structure. In particular the large particle emission results in a substantial erosion of the graphite surface; this has been clearly demonstrated by weight loss and SEM analyses [10].

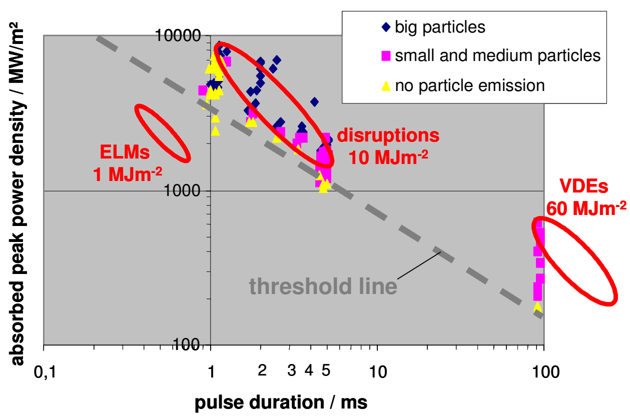


Fig. 7. Threshold values for the onset of brittle destruction on fine grain graphite derived from electron beam simulation experiments [10]. The ovals indicate the relevant range of transient events in ITER.

The threshold values for the onset of brittle destruction have been determined for graphites and CFCs both for disruption and VDE specific pulse durations, i.e. for 1 to 5 and for 100 ms; similar studies for the ELM regime are on the way. According to fig. 7 the thermal loads during plasma disruptions and VDEs in ITER are clearly above the threshold values for brittle destruction, while the ELM regime seems to remain in a safe operation regime. Nevertheless, brittle destruction may also play an important role for ELM specific loads because of the high frequency of these events (1 Hz) and an integrated number of several million incidents during the lifetime of the divertor target in ITER.

Carbon dust particles have been collected and analysed by different methods. The size of these objects covers a rather wide range from a few nanometers to a maximum of about 100 µm, i.e. their dimensions are ranging from nanotubes to graphitic grains or even grain clusters. The particle velocity has been measured by means of a photodiode array and was found to reach maximum values of up to 150 ms^{-1} for large carbon dust particles (graphite R6650, P = 2.4 GWm^{-2}, $\Delta t = 5$ ms) [12]. Simulation tests with carbon fibre composites show a rather similar behaviour compared to fine grain graphites, however, the threshold values are slightly shifted to higher energy densities; this is due to the improved thermal conductivity of this material. The material erosion strongly depends on the architecture of the CFC composite and on the type and orientation of the fibres used.

IV. Material performance after neutron irradiation

To investigate the irradiation induced degradation of mechanical and thermal properties, selected plasma facing materials have been subject to ITER relevant neutron doses in fission type material test reactors, such as the HFR Petten. In addition modifications in the high heat flux performance have been investigated in electron beam tests on irradiated small scale components with CFC, tungsten and beryllium armour.

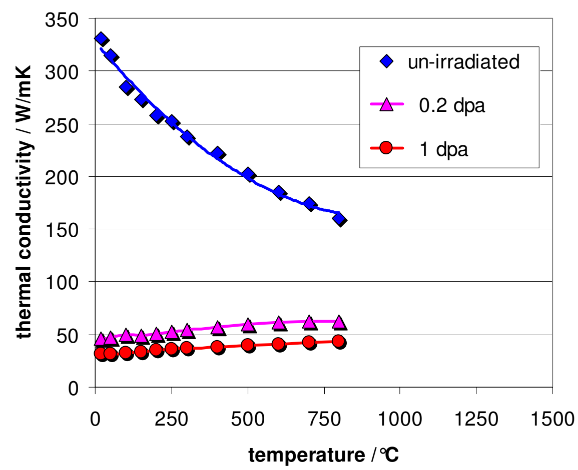


Fig. 8. Thermal conductivity of NB31 before and after neutron irradiation (0.2 and 1.0 dpa, T_{irr} = 200°C, pitch fibre orientation)

The heat removal efficiency of actively cooled components mainly depends on the thermal conductivity λ of the materials. This parameter was determined in laser flash experiments which allows a direct measurement of the thermal diffusivity α in combination with additional recordings of the material density ρ and the specific heat c_p ($\lambda(T) = \alpha(T) \cdot \rho(T) \cdot c_p(T)$).

Carbon based materials show a rather strong decrease in thermal conductivity even after relatively low neutron fluences [4]. The ITER candidate CFC armour material NB31 for example exhibits excellent thermal conductivities before neutron irradiation. Fig. 8. shows laser flash data measured in the high thermal conductivity direction (i.e. parallel to the pitch fibre reinforcement) with RT values exceeding 300 $Wm^{-1}K^{-1}$. Even low neutron fluences have a strong effect on the thermal conductivity with values below 50 $Wm^{-1}K^{-1}$ at room temperature. n-irradiation to 1.0 dpa finally results in a reduction of λ by one order of magnitude. Due to annealing effects the thermal conductivity reduction diminishes at elevated temperatures.

The room temperature thermal conductivity of sintered tungsten is significantly smaller compared to NB31; however, there is only a marginal reduction at elevated temperatures, see fig. 9. For irradiated tungsten the neutron induced degradation of the thermal conductivity λ is also less pronounced; in a temperature range $T \leq 1400°C$ and up to the ITER specific fluence of approx. 0.6 dpa λ remains well above 100 $Wm^{-1}K^{-1}$. For $T \geq 1000°C$ the difference between irradiated and un-irradiated material is negligible.

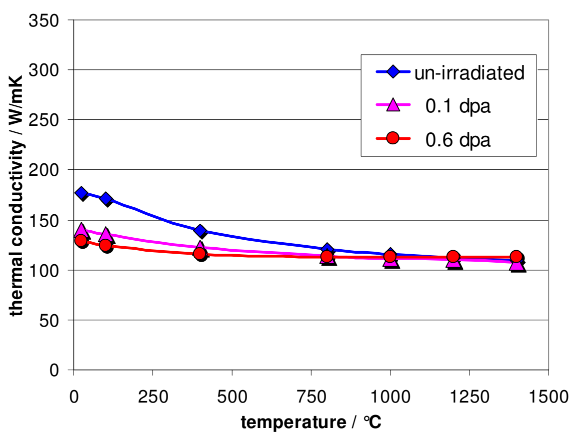


Fig. 9. Thermal conductivity of tungsten before and after neutron irradiation (0.1 and 0.6 dpa, $T_{irr} = 200°C$)

Actively cooled divertor mockups with CFC and tungsten armour (flat tile and monoblock design) have been exposed to the same neutron environment in the HFR reactor. The thermal fatigue behaviour of all modules has been evaluated without and after neutron irradiation. Typical results for a CFC flat tile mockup (NS31 armour) are plotted in fig. 10. To avoid excessive carbon vaporization these experiments were limited to surface temperatures of approx. 2000°C. In compliance with these restrictions the un-irradiated components have been exposed to heat loads beyond 25 MWm^{-2} (screening tests); after neutron irradiation these limits were achieved at approx. 20 and 15 MWm^{-2} for 0.2 and 1.0 dpa, respectively. For low power densities of a few MWm^{-2} the slope of the three curves in fig. 10 shows a very strong dependence on the applied neutron fluence; for higher thermal loads, i.e. when the surface temperature exceeded values of approx. 1000°C part of the neutron induced defects could recover (which can be recognized from the decreasing slope).

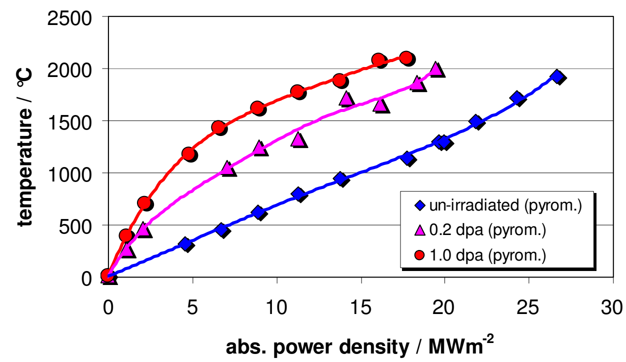


Fig. 10. Surface temperature at steady state vs. power density for flat tile CFC mockups before and after neutron irradiation

Beside screening tests with small cycle numbers, thermal fatigue experiments have been performed with n = 1000 cycles [8] in agreement with the experiments on un-irradiated components in chapter III.A. The results which have been obtained so far can be summarized as follows:

- CFC flat tiles have been exposed to cyclic thermal loads up to 15 MWm^{-2} (at 0.2 dpa and 1.0 dpa) and for 1000 thermal cycles without any failure,
- CFC monoblocks have been tested up to 12 MWm^{-2} for 1000 cycles; screening tests performed at 14 MWm^{-2} have been terminated caused by vaporization losses due to high surface temperatures,
- tungsten monoblock modules didn't show any failure up to 18 Wm^{-2} (0.1 and 0.6 dpa).
- tungsten flat tiles (macrobrush) withstood 1000 cycles at 10MWm^{-2} (0.1 and 0.6 dpa;).

The fatigue tests on the macrobrush modules were characterized by a non-negligible increase of the surface temperature. The configuration of the individual tungsten blocks (5 x 5 x 6 mm^3) appeared to be disarranged after 1000 electron beam pulses. According to metallographic analyses this may be due to the accumulation of point defects in a thin layer of soft copper which has been used to align and fix the individual tungsten cubes.

Neutron irradiation experiments with primary first wall mockups (low temperature irradiation at 0.6 dpa) are in preparation.

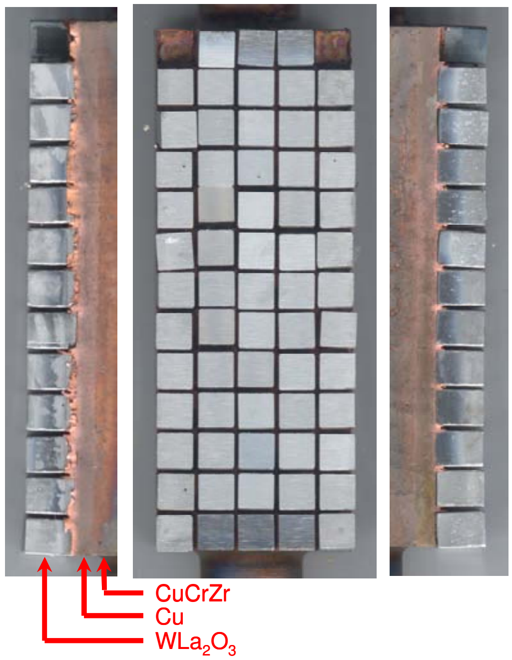


Fig. 11. Top and side view of a neutron irradiated tungsten macrobrush module after thermal fatigue testing (0.1 dpa at T_{irr} = 200°C; 1000 cycles at 10 MWm^{-2})

V. CONCLUSIONS

The present R&D activities on the development and testing of high heat flux components for ITER have demonstrated that plasma facing components for the divertor and for the first wall can be reliably manufactured. HIPing or electron beam welding in combination with copper casting are the preferred joining techniques; brazing is retained as a back-up solution. Components produced with these technologies show an excellent heat removal efficiency up to 20 MWm^{-2} for 1000 cycles and beyond.

When exposed to transient thermal loads during off-normal plasma operation (plasma disruptions or vertical displacement events) CFCs and tungsten alloys show an excellent thermal shock resistance (if $T > DBTT$). However, melt layer instabilities or brittle destruction trigger a non-negligible material erosion, which is associated with the formation of toxic, radioactive or tritium contaminated dust particles.

Neutron irradiation of plasma facing materials or components to ITER relevant fluences (≤ 1 dpa) has significant influence on the physical properties. Particularly carbon based materials show a drastic reduction in thermal conductivity which again deteriorates the high heat flux performance of actively cooled components. The neutron induced degradation of PFCs with tungsten armour is less distinct. In general components with monoblock structure are more robust compared to flat tile solutions.

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