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Plasma Facing Materials for the JET ITER-like Wall

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ABSTRACT
The chosen materials for plasma facing components for the deuterium/tritium phase of ITER are beryllium and tungsten. These materials have already been widely investigated in various devices like ion beam or electron beam tests. However, the operation of this material combination in a large tokamak including plasma wall interaction, material degradation, erosion and material mixing has not been proven yet.

The ITER-like Wall which has been recently installed in JET consists of a combination of bulk tungsten and tungsten coated CFC divertor tiles as well as bulk beryllium and beryllium coated INCONEL in the main chamber. The experiments in JET will provide the first fully representative test of the ITER material choice under relevant conditions.

This paper concentrates on material research and developments for the materials of the JET ITER-like Wall with respect to mechanical and thermal properties. The impact of these materials and components on the JET operating limits with the ITER-like Wall and implications for the ongoing scientific programme will be summarised.

1. INTRODUCTION
Materials for plasma facing components in nuclear fusion devices have to withstand a multitude of extreme loading conditions. Steady state as well as transient heat loads are responsible for material degradation like thermal shock and thermal fatigue cracks, recrystallisation and melting. The impact of neutron induced material degradation on the lifetime of wall components plays also an important role. Furthermore, tritium fuel retention and transmutation of elements is a major concern for the operation of nuclear fusion devices. In addition chemical as well as physical sputtering, which leads to erosion and redeposition processes of mixed layers, has to be taken into account. Finally plasma compatibility of the materials is a main challenge for plasma operation. Especially for high-Z materials the maximum tolerable impurity concentration in the plasma is quite small.

Due to these reasons the number of candidate materials for high thermally loaded plasma facing components is limited. The actual material combination for the deuterium/tritium phase of ITER consists of beryllium for the main chamber wall and tungsten for the divertor. It has to be pointed out that this special material combination of beryllium and tungsten has never been tested in tokamak devices so far. Therefore the ITER-like Wall project has been launched in JET in order to investigate the current choice of plasma facing materials for the deuterium/tritium phase of ITER [1.2.3].

The material configuration for the ITER-like Wall installation in JET is presented in Figure 1. In the high heat loaded divertor area bulk tungsten and tungsten coatings on CFC substrates are applied. Beryllium coatings on INCONEL 625 between the limiters (low heat flux area) und bulk beryllium tiles for the main chamber wall are installed.

This paper gives an overview on material developments of plasma facing components for the ITER-like Wall project. It summarises the material aspects for the beryllium components and major material properties. The information concerning tungsten focuses mainly on embrittlement and melting. In addition to bulk tungsten, also tungsten coatings have been applied partially in the divertor region. The development, testing, manufacturing and performance of these tungsten coatings will be reported as well. Finally, some results of beryllium coating characterisation on INCONEL substrates for low heat flux areas are presented.

2. SCOPE OF THE ITER-LIKE WALL PROJECT

The research topics for the ITER-like Wall project have already been comprehensively explained in literature [4] and will only be shortly summarised in this paragraph. They can be separated in two parts, topics for the beryllium main chamber wall and issues for the tungsten divertor.

Firstly, erosion and migration of the beryllium at the main wall material in ITER relevant scenarios will be examined. Furthermore, the behaviour of molten beryllium and its impact on operation and wall lifetime as well as fuel retention in beryllium tiles and in co-deposited layers will be investigated. In addition the interaction of beryllium with background oxygen, wall conditioning and machine start up will be addressed experimentally [4].

For the tungsten divertor, research is focused on fatigue lifetime and resistance of plasma facing materials to transient power loads (Edge Localised Modes). Additionally the alloying of tungsten with beryllium, which leads to a reduction of the melting point, needs to be taken into account. Moreover the compatibility with ITER relevant plasma scenarios (tungsten sources and accumulation of tungsten in the core of the plasma) and fuel retention are important aspects for the tungsten divertor operation [4].

In contrast to ITER, all plasma-facing components in the ITER-like Wall are only inertially cooled. Thus the power handling capability of the ITER-like Wall is strongly determined by the heat capacity. The operation of the JET ITER-like Wall started in 2011. That is the reason why this paper contains no results of the current ITER-like Wall operation in JET. The first post-mortem analysis of materials from the ITER-like Wall is foreseen for 2012 [5].

3. BERYLLIUM

Beryllium is a metal with a low atomic number and consists of a hexagonal closed packed crystallographic structure. Beryllium has a high thermal conductivity (190W/(mK) at room temperature) and a good plasma compatibility, which means that it has a high allowable concentration in the fusion plasma. The maximum allowable concentration of beryllium in the plasma is given in literature with 15%. For a light metal, beryllium has an extraordinarily high melting point. Nevertheless, in absolute numbers the melting point of 1287°C is one of the major drawbacks of the material beside erosion and low neutron radiation resistance. Neutron radiation damage in beryllium results in embrittlement by defect formation. Moreover transmutation (tritium and helium formation) leads to gas driven swelling and embrittlement especially for high temperature irradiation. [6] The behaviour in relation to oxygen can be seen from two different sides. On the one hand beryllium is able to remove oxygen from the plasma due to its oxygen affinity; on the other hand beryllium oxide in the volume of the material leads to a decrease in thermal shock resistance.

For the ITER-like Wall project, the former JET S65C VHP scrap beryllium tiles were recycled into new S65J HIP beryllium blocks. Solid beryllium main wall inner and outer guard and protection limiter tiles were designed with a segmented castellated construction mounted on vacuum cast INCONEL 625 carriers in order to reduce eddy currents and thermal stresses on the plasma facing surface. Slots are cut by electro machining (EDM) followed by chemical etching in order to remove surface impurities left from EDM processing. These limiters take most of the main wall power load. As an optimal size for the castellation, an area of 12×12mm^2 with a depth of 16 mm was chosen based on detailed calculations of thermally induced stresses. In Figure 2 a detailed view of the outer poloidal limiter of the tile carrier with fixing bolts and INCONEL cast support is presented. Further details on the engineering aspects of the beryllium components can be found in literature [7, 8, 9, 10]. Details of the application of beryllium as a first wall material in JET in the past are reported in literature as well. JET has extensively used beryllium in a variety of inertially cooled components that have sustained in total, several thousand of plasma discharges. Local melting and micro cracking of small regions was found. Castellation of tiles is mandatory in order to avoid thermal fatigue[11]. The quantification of beryllium erosion is one of the major aspects of beryllium material performance for the ITER-like Wall by post mortem-analysis in order to give the best possible predictions for ITER. Therefore so called marker tiles are applied, which are allocated in special positions on selected tiles of the inner and outer poloidal limiters. The beryllium marker tiles consist of a structure of 8-10μm beryllium on a nickel interlayer of 2-3μm followed by the bulk beryllium [12,13,14]

Nickel was chosen as interlayer material due to the small mismatch in thermal expansion coefficients (13×10^{-6} K^{-1} for nickel and 16×10^{-6} K^{-1} for beryllium at room temperature) and the absence of formation of intermetallic phases with beryllium up to approximately 1000°C. Thermionic vacuum arc technique was selected and optimized in order to deposit the layers [12,13,15].

Before the application in the ITER-like Wall, the marker tile samples were tested in an electron beam test facility at heat fluxes of 1-6 MW/m^2 for 10 s for screening. A damage threshold value of approximately 5MW/m^2 was determined. Furthermore the marker tile samples withstood cyclic loading for 50 cycles up to 3.5MW/m^2 without damage. The formation of intermetallic phases was not observed during the tests and these marker tiles perform well under the operational requirements of the ITER-like Wall of about 1 MW/m^2 for normal operation [13].

4. BERYLLIUM COATINGS

In between the limiters in the low heat flux area 7-9 μm thick beryllium coatings on an INCONEL 625 substrate are applied. Thermal cycling for 50 cycles at 1 MW/m^2 for 10s was applied in high heat flux tests, where no delamination was observed. The microstructure of the coatings before and after these tests is shown in Figure 3.

In Figure 4 the temperature increase during the thermal loading as a function of the energy density is presented. The circle symbols represent the temperature increase of the beryllium coating on an INCONEL substrate with a thickness of 3.3–3.6mm. The square symbols represent thicker INCONEL substrates. It can be concluded, that for the loading conditions similar to those in JET, the heat capacity of the samples (in this case thickness of the INCONEL substrates) is an important factor to determine the maximum surface temperatures. As expected, a lower temperature increase occurred at samples with a higher heat capacity. Using a linear extrapolation to the higher energy density range, the temperature increase of the samples with ~3.5mm thickness would follow the dotted line. The extrapolation indicates that the berylllium coatings would start to melt at loads above 30 MJ/m^2 when the loading starts at the temperature of 200°C (corresponds to the base temperature of the JET ITER-like Wall). Moreover, it has to be mentioned that no delamination of beryllium coatings was found even at the highest tested power density of 2.6 MW/m^2 for 6.2s [1, 16].

It can be concluded that these coatings show a good performance well beyond JET requirements because of the small mismatch in thermal expansion between the coating and the substrate. Moreover no intermetallic phases are formed up to approximately 1000°C [1, 16].

5. Tungsten

Tungsten will be used for the high heat flux areas of nuclear fusion devices. The important advantage of tungsten is its high melting point (3422°C). Moreover tungsten has a high thermal conductivity (173 W/(m K) at room temperature; 100W/(m K) at 1527°C) and a low tritium retention potential. Unfortunately, the cubic body centred crystallographic structure of tungsten leads to a high ductile to brittle transition temperature in comparison with cubic face centred metals due to less crystallographic slip systems. Thus tungsten is sensitive to brittle crack formation due to its limited plasticity in the low temperature range. In this context neutron and hydrogen embrittlement even degrades the deformation capability of the material. Furthermore thermal fatigue has to be taken into account. Results of tungsten material performance and brittle crack formation under transient heat loads in dependence on temperature and absorbed power density can be found in literature [17, 18, 19, 20].

Recrystallisation of tungsten starting at about 1200°C influences the material performance under transients as well. The impact of tungsten melting on the power handling capability of tokamaks and the behaviour of resolidified tungsten material under transients has already been investigated. Resolidified layers are less resistant to thermal shocks and the power handling capability in tungsten melting experiments in the tokamak TEXTOR at Forschungszentrum Jülich was also significantly decreased [21].

A schematic picture of the ITER-like Wall material configuration for the divertor is presented in Figure 5. The limit of deposited energy for the ITER-like Wall tungsten divertor stacks is mainly determined by the engineering limit of 330°C for the clamping system and 600°C for the INCONEL carrier. Engineering aspects of the tungsten divertor material configuration are comprehensively summarised in literature [22, 23]. The lower limit for the tungsten surface temperature is the ductile to brittle transition temperature, which is below 200°C measured in four point bending tests. In the temperature range of 1200°C-2200°C (above the recrystallisation temperature for tungsten of

~1200°C) thermal fatigue of tungsten has to be taken into account [23].

A standard tungsten stack was successfully tested in high heat flux tests in the ion beam test facility Marion at Forschungszentrum Jülich. The exposure of a full scale prototype of the standard tungsten stack in the Marion facility shows that an energy density of up to 60 MJ/m^2 per stack can be handled with the bulk tungsten tile [23].

6. Tungsten coatings

The bulk tungsten divertor tiles could only be accommodated on the horizontal part of the JET divertor due to technical reasons. The majority of the ITER-like Wall divertor and some highly loaded tiles in the main chamber (for example beam shine through areas) consist of tungsten coating on a DMS 780 CFC substrate (Figure 6).

§The main challenge for the application of tungsten coatings within the ITER-like Wall project is the fact that the tiles to be coated were made from an anisotropic carbon-fibre reinforced carbon material (CFC). This leads to an anisotropic thermal expansion of the CFC substrate material and consequently to an anisotropic mismatch between the substrate and the tungsten coating. Due to this mismatch between tungsten and CFC a research and development phase was initiated in early 2005 involving various EURATOM associations to produce a total of 14 different types of coatings with respect to the employed deposition processes, coating thicknesses, and interlayer types with the goal of identifying a possible solution to the ITER-like Wall needs [24].

These different types of coatings were then subjected to a programme of qualification tests the main part of which was high heat flux testing in the neutral beam facility GLADIS [24, 25, 26]. After a thermal screening where the coatings were exposed to power densities of up to 23.5MW/m^2 (exceeding 2200°C peak surface temperature), the most promising coating types were subjected to a low cycle fatigue loading programme for 200-300 pulses at 10.5 MW/m^2 corresponding to a peak surface temperature of 1500°C.

The finally selected coating method was the technique of combined magnetron sputtering and ion implantation, which combines conventional magnetron sputtering with the application of high voltage pulses for the purpose of stress relaxation in the coating [27, 28, 29]. 10-20 μm thick tungsten coatings on 3 μm molybdenum interfaces were deposited on a total of about 1700 JET CFC tiles. A picture of the microstructure of these coatings is shown in Figure 7. The coated CFC tiles are presented in Figure 8.

These coatings were subsequently tested in the electron beam facility JUDITH 1 and JUDITH 2 at Forschungszentrum Jülich, Germany [30, 31]. The results of ELM-like heat load tests in JUDITH 1 are presented in Figure 9. Tests were always performed for 100 pulses of 1 ms at different absorbed power densities (79-316 MW/m^2) and temperatures (room temperature up to 400°C). Delamination and cracking start to occur at absorbed power densities of about 158 MW/m^2. Below this value no failure of the coatings was observed. Failure occurrence is mainly dependent on the absorbed power density and less affected by the test temperature of the specimens. It always occurs first on fibres parallel to the surface of the CFC substrate due to the high mismatch in thermal expansion coefficients of the coating and substrate and the bad thermal conductivity of this fibre orientation into the bulk of the material. Beside cracking and delamination even melting is observed at absorbed power densities of about 237MW/m^2. It has to be pointed out that melting of the coating is only a result of the delamination of the coating [32]. Only the very largest JET ELMs approach 158MW/m^2 at the outer strike point (bulk tungsten tile will be used for most experiments).

Finally coating failure due to carbide formation at temperatures of 1350°C for 1-20h was investigated in the neutral beam facility GLADIS. The tests were performed at 16.5MW/m^2 for 1.5s and up to 200 pulses. Clear threshold behaviour for the heating time was found: Coating with a heat treatment above 2-5 hours, corresponds to the carbide formation, are prone for cracking and delamination [33].

SUMMARY AND OUTLOOK

In order to summarize the results, a comparison of the heat load limits between ITER and the ITER-like Wall in JET is presented in Table 1 for the major material choices of beryllium for the first wall and tungsten for the divertor. It has to be pointed out, that a direct comparison is difficult since there are major differences in the technology like the cooling concept for the components and the size of the device. ITER has an approximately ten times higher vacuum vessel volume compared to JET and actively water cooled components.

For the operation with the ITER-like Wall during the years 2011 and 2012 the material characterization focuses mainly on fuel retention and material migration, material limits and long term samples under transient and steady-state heat loads. A list of material related experiments in this period is provided below [35, 40]:

Recovery wall conditioning

- Initial first wall Be erosion, Be and W material mixing and fuel retention
- C and Be migration in all scenarios
- H-modes prior to long term sample retrieval with tacer injection
- Long term samples analysis

Fuel retention

- Evaluation of fuel retention in all scenarios
- Gas balance analysis with impurity seeding

Tungsten erosion

- Divertor W erosion and ELM induced sputtering
- Long term evolution of W erosion and migration

Beryllium power handling

- Beryllium tile power handling

Divertor power handling

- Bulk tungsten tile power handling

Monitoring pulses

- Beryllium migration monitoring
- ILW status monitoring
- Recovery wall conditioning

One especially interesting material experiment focuses on the behaviour of already recrystallised tungsten lamellas in comparison to non-recrystallised tungsten lamellas under steady state and transient heat loads in the divertor. In a later stage the impact of tungsten melting on the plasma operation will be investigated. The exploration of ITER operating scenarios with the ITER-like Wall and physical topics essential to the efficient exploitation of the ITER-like Wall and ITER will be another focus of the research program [5].

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	Material	Operation limits ITER-like Wall* Not actively cooled	Operation limits ITER* Actively water cooled
First wall	Beryllium	Surface temperature <900°C (due to the beryllium melting temperature) HFF# <22 MW $m^{-2}s^{1/2}$	Power density 1-5 MW/m^2 (steady state)
Divertor	Tungsten	Surface temperature limit <1200-2200°C HFF# 20-35 MW $m^{-2}s^{1/2}$ Transients (ELMs)** ~0.1-0.5 MJ/m^2 (ms range)	Surface temperature limit ~1500°C (depending on design and tungsten armor thickness) Power density ~5-20 MW/m^2 (steady state) Transients (ELMs)** ~0.5-1 MJ/m^2 (sub millisecond range)
	Tungsten coatings on CFC	Surface temperature <1200°C (due to carbidization) HFF# <5 MW $m^{-2}s^{1/2}$ (transients, for 10^2 ELMs)	Not applicable

**without off-normal events*
#HFF = Heat flux factor
***no material parameters, for the material performance of tungsten under transient heat loads see [18, 19].*

Table 1: Comparison of heat loads for the ITER-like Wall and ITER for the relevant plasma-facing materials [34, 35, 36, 37, 38, 39].

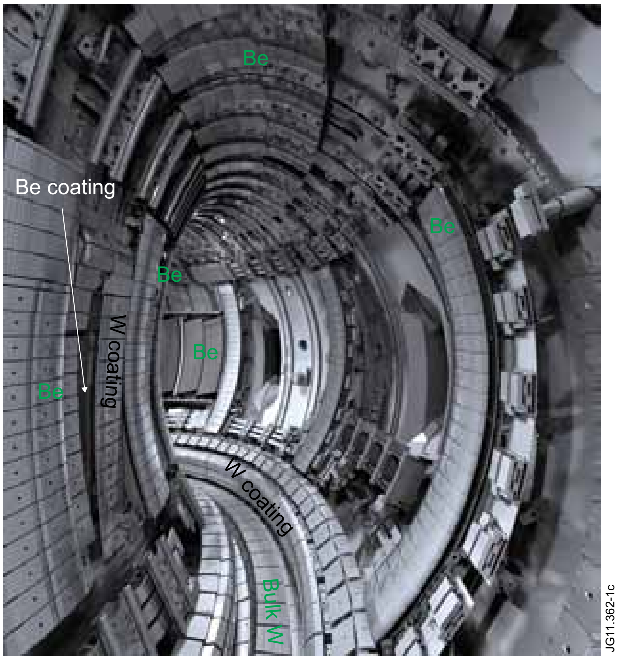

Figure 1: Material configuration for the ITER-like Wall.

Figure 2: Detailed view of an outer poloidal limiter (S65J HIP beryllium) with tile carrier fixing bolts and vacuum cast INCONEL 625 support.

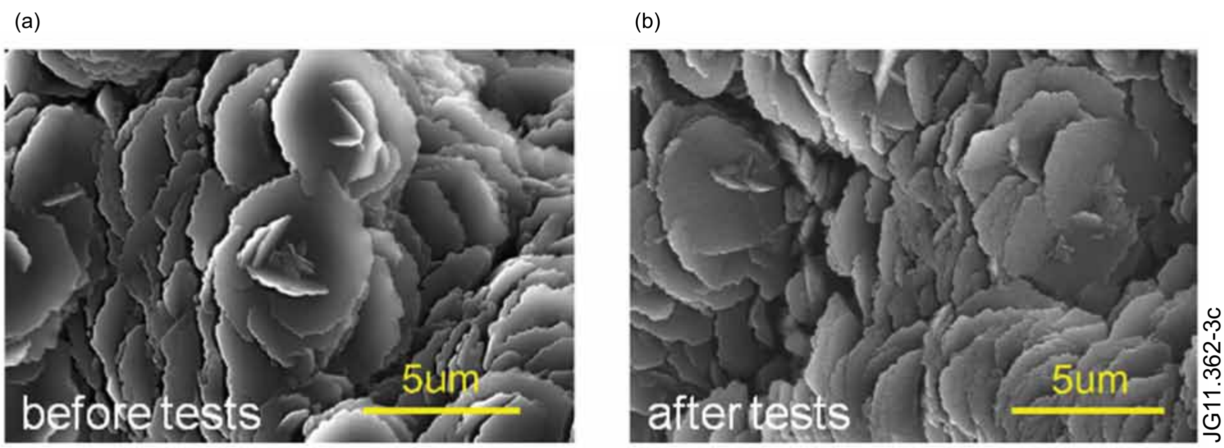

Figure 3: Beryllium coatings on INCONEL substrate (7-9μm thick films) Thermal loads 1 MW/m^2 for 10s, 50 cycles [16].

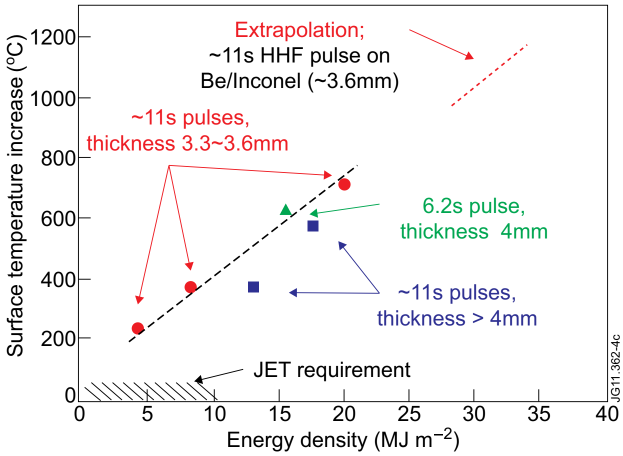

Figure 4: Surface temperature increase as a function of deposited energy (thickness indicates the thickness of the INCONEL substrate) [16].

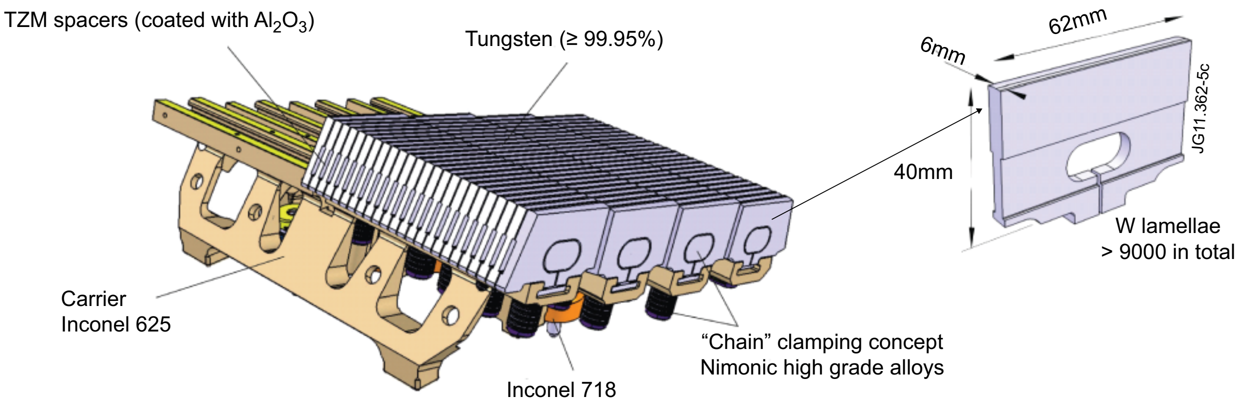

Figure 5: Schematic picture of a bulk tungsten divertor tile (left side); in total over 9000 shaped bulk tungsten lamellas (right side) are installed in the JET-ITER-like Wall.

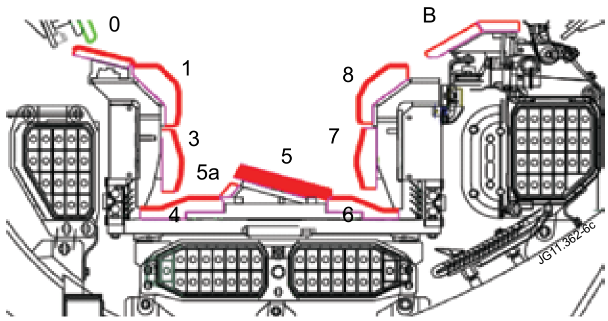

Figure 6: Schematic illustration of JET divertor material configuration in the ITER-like Wall Project.

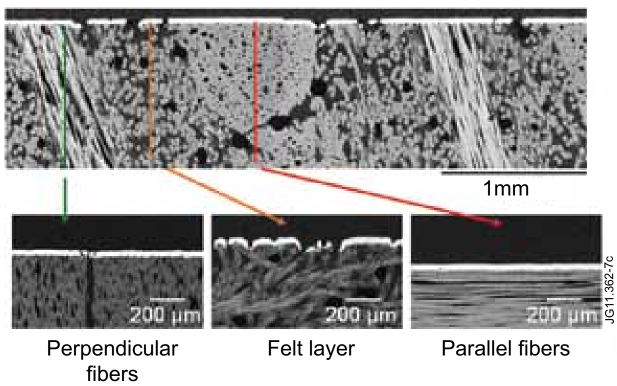

Figure 7: Metallographic images of longitudinal (top) and transversal (bottom row) cross sections of the CFC substrate and the tungsten coating [30].

Figure 9: Failure occurrence of tungsten coatings under ELM-like heat loads in dependence on absorbed power density and test temperature [32].

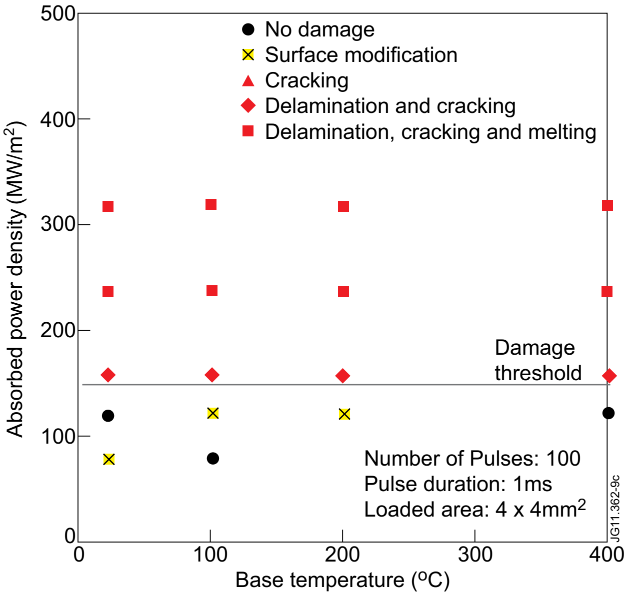

Figure 8: Tungsten coated CFC tiles in the coating device [29].