# **Overview of the JET ITER like wall project**

V. Philipps<sup>1</sup>, Ph.Mertens<sup>1</sup>, G. F. Matthews<sup>2</sup>, H.Maier<sup>3</sup>, the ITER like wall project team\* and JET-EFDA contributors to the ITER-like Wall Project

\*P.Edwards<sup>2</sup>, A.Loving<sup>2</sup>, V.Riccardo<sup>2</sup>, H.Greuner<sup>3</sup>, R.Neu<sup>3</sup>, A.Schmidt<sup>1</sup>, M.Rubel<sup>4</sup>, C.Ruset<sup>5</sup>, E.Villedieu<sup>6</sup>

<sup>1</sup>Institut für Energieforschung, IEF4, Plasmaphysik, Forschungszentrum Jülich, Euratom Association, Jülich, Germany

<sup>2</sup> Euratom/UKAEA Fusion Association, Culham Science Centre, Abingdon, UK

<sup>3</sup> Max-Planck-Institut für Plasmaphysik, EURATOM Association, 85748 Garching, Germany

<sup>4</sup> Alfvén Laboratory, Royal Inst. Technology (KTH), Assoc. EURATOM-VR,100 44 Stockholm, Sweden

<sup>5</sup>Nat. Inst. for Laser, Plasma and Radiation Physics, Association Euratom-MEdC, Bucharest, Romania

<sup>6</sup>Association Euratom-CEA, Cadarache, DSM/DRFC, Saint Paul Les Durance, France

## 1. Abstract:

This paper presents an overview of the R&D activities for the ITER-like Wall Project in JET which has been launched in 2005 and will be completed early 2011. A full replacement of the first wall materials in JET will be done to an ITER like wall composition with Be in the main chamber and W in the divertor as foreseen for the second activated phase in ITER. The project is directed to deliver answers to urgent plasma surface interaction questions such as tritium retention or Be-W interaction and to provide in general operational experience in steady and transient conditions with ITER wall materials under relevant geometry and relevant plasma parameters.

#### **2. Introduction:**

The design of ITER which has been decided early 2006 is the collaborative result of fusion research from world-wide fusion activities giving confidence to achieve the scientific and technological objectives within appropriate margins. However critical issues remains which are addressed in ongoing fusion research activities in present devices. Many of these issues are related to plasma-wall interaction processes such as the control of steady state and transient wall

power loads to technically acceptable limits, the control of the in-vessel tritium (T) inventory or the achievement of sufficient lifetime of the plasma facing components (PFCs). ITER is currently designed with a beryllium-clad first wall, tungsten (W) brushes over most of the divertor region and two options for the high power divertor areas, to start with carbon fibre composites (CFC) but to change most probably to a full divertor in the second D-T operation phase. Both material combinations have never been used and tested in a large tokamak experiment and motivated the ITER-like wall (ILW) project at JET [1] in which the present main chamber wall CFC tiles will be exchanged with beryllium tiles and in parallel a fully tungsten-clad divertor will be prepared. Of large importance is in particular the development of plasma scenarios and control schemes compatible will the wall requirements. The project will provide essential information for the expected material behaviour in ITER and a give a technical basis for the development of ITER scenarios.

Preparations for the installation of the ITER-like wall in JET are satisfactory advanced. Series manufacture of beryllium components and the divertor W –tiles has begun and a second long remote handling boom has been commissioned which will allow minimisation of the shutdown time.

## 3. Overview of scientific aims

The overall scientific objectives of the ILW programme may be generalised such that to gain operational experiences with a beryllium wall which is much more vulnerable against power load excursions leading to melting, the compatibility of a Be wall with a tungsten divertor and the compatibility of a full W divertor with the envisaged ITER operational scenarios, including in particular also advanced operation modes. More specific issues concern the long term T retention behaviour [2] which is limited due to safety reasons and estimated to be reached in the start up wall material configuration (Be-C-W) in several hundred of full performance pulses. In parallel, T removal techniques are not enough elaborated [3] providing the main reason why an all-W divertor has been considered for the deuterium-tritium phase of ITER. The ILW programme is directed also (albeit at a later stage) to operation with increased heating power at high stored energies up to 20 MJ at high Ip and Bt. This requires on one hand the demonstration of reliable ELM and disruption control but also the possibility of (intended or unintended) large ELMs which may exceed the limit for tungsten coatings. This and other reasons motivated the development of a bulk tungsten target for the outer horizontal target row loaded in ITER-like

high triangularity configurations. The W bulk target will also allow other PWI aspects like the tungsten thermal fatigue during repetitive ELM loading, W-melt layer behaviour, W-Be material interaction and material migration into tungsten gaps..

In some more detail the objectives of the JET experiment can be summarised as follows:

- Demonstrate that a Be wall plus an all-W divertor have sufficiently low fuel retention to meet ITER requirements. In addition, prove/test T retention mitigation and detritiation techniques in a Be/W device.
- Analysis of Be erosion, migration and how Be migration to the W divertor interact with W tiles
- Investigate special heating system related effects such as the interaction of fast ions with W surfaces.
- Develop control strategies applicable to ITER to control damage to Be and W plasmafacing components, such as disruption mitigation and edge localized mode (ELMs) power loss control systems. The foreseen power upgrade of JET allows extension of this work to energy densities in transients comparable to those in ITER.
- Develop integrated ITER compatible scenarios for an all-metal machine including impurity seeding strategies to replace the intrinsic carbon radiation to achieve acceptable divertor power loads in the ITER baseline edge scenario.

## 4. Steady state and transient power load boundary conditions.

Taking into account in particular also the input power enhancement in the future JET programme (4) both steady state and in particular transient losses can reach more regularly the material limits, calling for adequate control mechanisms to be an integrated part of the ILW programme. The wetted area in JET for steady state operation is only about 0.7-1m<sup>2</sup> corresponding to a ~ 5 mm power e-folding length at the outer midplane [5] and not much increased during ELMS. With a input power of 40 MW and at low radiation levels (≈0.4), the energy limit of one of the W bulk stacks in the outer divertor would be reached in about 3-4 sec, calling for impurity seeding to operate at higher radiation levels or sweeping of the strike point over larger areas. In previous JET operation [<sup>6,7</sup>] ]type I ELMy H-modes with ELM energy losses  $\Delta W_{ELM} \ge 1$  MJ have already reached which is at the carbon ablation limit . With the heating upgrade, the expected ELM energy at high current (I<sub>p</sub> = 5 MA )and high power (P<sub>inp</sub> = 40 MW) y would be ≈  $\Delta W_{ELM}$  2MJ and

at low plasma density, reaching a pedestal temperature  $T_{ped} = 4 \text{ keV}$  (pedestal collisionality ~ 0.05). This is above the melting for W [8].and much above the ELM loss limit for the W coatings in JET.

The effect of transients (ELMs & disruptions) on the Be first wall is more complicated to estimate since available data are more limited. However with typically 10-20% of the ELM energy reaching the first wall [9], Be melting in large ELMS (2MJ) would also occur assuming an effective wetted area in the order of  $1 \text{ m}^2$ . However quantitative data on the effective area are limited and remain to be determined more quantitatively.

Operation of JET at high stored energy ( $W_{dia} \sim 20MJ$ ) can also reach the material limits both for Be and W. Estimations of the energy loads in a worse case disruption to the divertor in JET [10] result to an energy flux of ~ 3–4 MJ/m<sup>2</sup> in few ms which would approach the tungsten melt limit of about 65 MW/m<sup>2</sup>s<sup>0.5</sup>. Control mechanism to avoid such loads are presently investigated, e.g. by massive gas injection, disruption detection/avoidance and other means. From present knowledge the most critical disruptions are those at high stored energy in advanced scenarios with internal transport barrier, since they can deposit the full stored energy in short time onto the inner wall areas, clad with low melting Be tiles (1278°C). If control mechanism fail or are not sufficient, shallow melting of some areas of the first wall can occur. The dynamics of such a melt layer and its effect on plasma operation remains a key uncertainty in the operation of ITER and will be addressed in JET.

## 5. Technical realisation of the ILW project

The technical objectives for the new wall are driven to preserve and even improve as best we can the performance of the wall for power loading and to withstand in parallel the forces due to EM forces which increases due to higher electrical conductivities of Be and W. The design aims to

- a) Eliminate as much as possible of all carbon sources.
- b) Withstand Eddy and Halo current forces resulting from disruptions of  $(I_p=6MA, B_r=4T)$ .
- c) Limiter energy and power handling equal to the current CFC design.
- d) Remote handling compatible design with a weight limit of 10kg for the main wall area and less than 100kg in the divertor.

e) Neutral beam shine-through energy handling >60MJ to meet the needs of advanced plasma scenarios.

## 5.1. Be main wall

The main constraints for the main wall design have been the needs to preserve the power, energy handling and force limits (due to disruptions) as set for the present CFC wall. These constraints have led to the design of solid Be main wall guard and protection limiter tiles which are inertially cooled, segmented (to minimise eddy forces) and castellated (to avoid thermal stress cracking) with hidden bolts and optimised tile shaping to maximise the power handling [11, 12] (fig 1). Slots are cutted by EDM with chemical etching afterwards to remove surface impurities left from EDM processing. These limiters take most of the main wall power load, in particular also during the limiter start up phase, ELMS and disruptions. The upper dump plates are made from roof like bulk Be limiters. Raw Be material has been delivered from Brush Wellman company, US; and machined in EU companies. Typical tile thickness of 40 mm provides inertial cooling and castellations are used one on hand to minimise thermally induced stresses and Eddy forces on the other hand. Detailed calculations of the thermally induced stress found optimal castellation size and shape of a 12 mm x 12 mm castellation area with 16 mm depth and a keyhole at the end. The EDDY forces on the Be tiles during disruptions are large due to the high electric conductance of Be (0.08  $\mu\Omega$  m at 200°C) and dB/dt of 100T/sec (6MA, 4T) during severe disruptions. Since the requirement for castellations and slicing of the tiles bears the danger of exposed edges on the toroidally facing surfaces, detailed analytical models have done to avoid any edge exposure down to 40µm. The main wall requirements have resulted in complex supports structures for which vacuum cast Inconel was decided to be the appropriate manufacturing technique.

Larger areas of recessed inner wall cladding will be made from 8µm thick Be-coated Inconel tiles [13, 14]since these areas experience much lower particle fluxes and a redesigning of these structures was beyond the scope and resources of the project. Empirical marker erosion data from previous JET campaigns (15) indicate sufficient lifetime of the coating for the aims of the project. In addition, W-coated CFC tiles will be used on special recessed areas of the neutral beam injection (NBI) shinethrough, see fig 2. This decision was driven to keep or even increase the power handling capability of the shine trough area for JET advanced plasma scenarios. Simple calculations indicate a low W sputter source from the beam shinetrough which is

considered not to influence significantly the W plasma contamination. However the analysis of this is also subject of the ILW scientific exploration.

## 5.2 Development of a full W divertor for JET

## 5.2.1. Development of W coatings

The technical solution proposed for the JET full W divertor, as shown on Fig. 3, includes bulk W for the load bearing septum replacement plate (LBSRP) in the outer divertor and W coating on CFC for all the remaining tiles. The decision for this was driven by the fact that, within the technical, financial and temporal constraints, a full W bulk tile solution for the divertor was out of the possibilities. Effort concentrated to provide a bulk W on the most loaded outer divertor tile5 (LBSRP) which is loaded in ITER like high triangularity configurations. To select the most reliable W coating technique, a broad and coordinated R&D programme was launched early 2006, including 14 different types of W coatings, based on chemical vapour deposition (CVD, 4, 10, 200  $\mu$ m), physical vapour deposition (PVD, 4, 10  $\mu$ m), vacuum plasma spraying (VPS, 200  $\mu$ m) and combined ion Magnetron sputter deposition. Finally, large effort was needed under the coordination of IPP Garching (in cooperation with few other partners) to qualify a reliable W coating technique for JET. A detailed description of the physics and technical issues is out of the scope of this contribution, more details can be find in (16, 17, 18, 19). The main scientific and technical issues can be summarised as this.

- The main problem from the beginning on was the anisotropy and mismatch of the thermal expansion coefficient of CFC with respect to tungsten in general, see [17]. Increased difficulties to coat CFC in comparison to fine grain graphite by W has already been reported earlier, probably due to the irregular surface structure of CFC [20]. Moreover it was decided to use the existing type of JET 2D carbon fibre composite tiles as substrate in order to minimise the amount of redesign required, which shows a particular thermal expansion anisotropy with zero expansion in one direction and a large expansion in the other 2 directions. During heating up of the coating in high heat flux test the thermal expansion mismatch of the W-layer to the CFC lead to plastic deformation of the W layer which, during cool down, can be not compensated by the limited elasticity of the W layer leading to cracking of the coating. This results in a typical pattern with cracking perpendicular to the fibres, which is sensitive to delaminate by buckling along the fibres (fig4). This must be accepted as a fundamental property of the CFC/tungsten combination, however this also relax the stress from the thermal expansion mismatch. The delaminated coating

can melt easily due to missing heat contact which finally can lead to partial loss of the coating. The direction to overcome this general difficulty is to move to rather thin coatings, to decrease the W coating density and to improve the W-C adhesion. This ruled out thick and high dense pure W coatings, such as e.g. the CVD layers. Thick VPS coatings with lower density behaved better due to larger intrinsic porosity which enabled to adopt expansion mismatch.

Another principal problem is the possible W-C carbide formation at the W-C interlayer [21, 22] which is temperature activated, governed by the C diffusion in W and the W-C interlayer and leads to a brittle W-C interlayer, which is sensitive against failure by thermal stresses. The method to reduce this risk is to use an interlayer which is less proud for carbide formation and which acts as a barrier for further carbidisation. Both Mo and Re interlayer have been tested in IPP Graching and compared with pure W coatings.

- Upscaling of coating techniques for small scale prototypes to large scale routine tile coatings has been found to be a problem for thick VPS coatings. This problem does not seem unsolvable but could not in the narrow time frame of the project [23]

- In general the reproducibility of the high heat flux behaviour of a special coating from one tile to the other was not satisfactory.

Some more features have been observed which could only partly be understood. One was that the thin (14µm) W-Re multilayers developed a discoulering when stored on air which was indentified to originate from Re-oxide being formed or transported to the top surface [23]. After cleaning by GDC or high temperature (900°C) baking the discoulering reappeared after storing the layer on air for few days (fig 5). Such behaviour is not observed on W-Re multilayer on fine grain graphite and might thus be attributed to the very porous and rough CFC surface structure and has finally ruled out the use of such layers. Carbidisation tests performed at 1350 for 5 hours revealed quite strong W-C carbidisation on the W-Re system, stronger as observed on pure W-C systems. The W-Mo interlayer system showed a better resistance against W-C formation [23].

Finally only 14  $\mu$ m coatings which were deposited by combined magnetron sputtering and ion implantation (CMSII) with a Mo interlayer did not develop a partly delamination in high heat flux (10 MW for 2.5 sec) tests. These coatings technique has been developed in the Romania association and were selected for the large scale W coating in the divertor and main chamber and are presently under production [24 ]

With the selected rather thin W coatings the erosion lifetime on the most eroded parts on the outer divertor is an issue. (the inner divertor region in JET has been routinely in a net deposition

regime in the past which is also expected in the future). W markers stripe experiments [25] in previous campaigns have shown a complete erosion of a  $3.5\mu$ m W stripe erosion at the horizontal outer tile 7 during one operation campaign (C15-C17) where the strike point was mostly situated. The observed erosion of the W stripe is due to sputtering by C-impurities while the sputtering by D ions can be neglected. Modelling of the W erosion stripe under C divertor conditions shows a maximal gross erosion during the campaign of about 15 µm which reduces by about a factor of 2-3 due to local redeposition. Under The ILW conditions the net erosion of W will be somewhat reduced due to reduced the sputtering of W by local C release and redeposition and reduced W sputtering by Be. Thus a 10µm thickness might become just acceptable for one operational campaign but at the risk of unwanted C sources at an early stage of operation. This calls to use the outer W solid divertor row as much as possible.

## 5.2. Development of a bulk W divertor row for the outer divertor tile 5

For the most heavily power-loaded tiles, a bulk W tile concept has been developed under the leadership of Forschungszentrum Jülich, Germany. Again, the design was mainly determined by the strong constraint of minimising the electromagnetic forces in disruptions (see conditions above) and optimising mechanical stability. As a consequence of this it turned also out that the whole supporting structure of the tile from the wedge below the PFC tiles down to the adaptor plate which is fixed to the divertor base carrier had to be completely redesigned to scope with the EM forces. The integrated system thus consists of the two tungsten tiles, each splitted into four W stacks, a new wedge which fixes the W-stacks an adaptor plate, which is fastened to the base carrier of JET (fig 6). 6mm thick and 35 mm W lamellas are taken to adopt the power by inertial cooling and to reduce the risk cracking due to the low W ductility. The individual W lamellas are toroidally isolated by Al<sub>2</sub>O<sub>3</sub> coated Mo spacers which provide toroidal electrical isolation to reduce the EDDY currents. For the same purpose 4 poloidal W stacks are chosen and with a corresponding finger like design of the underlying wedge. The main problem was the toroidal fixing of the lamellas which provide the force to hold the lamellas together by friction. No single bolt like design was compatible with the temperature and expansion requirements leading to the use a chain made of high temperature materials, Densamet parts with Nimonic pins which is pulled down by screws and springs below the wedge, providing the thermal expansion tolerances. Each Lamella has dedicated electrical contact points to the support structure to reduce halo forces and avoid arcing.

Detailed thermo mechanical analysis of the temperature distribution and the mechanical stresses have been done, resulting an optimised design for the lamella a shown in fig [26, 27].Critical issues remain like the stability of the end combs of the chain which can reach the maximum temperature of the material (700°C) in high power conditions (70 MW energy load per stack), the temperature of the spring which can reach temperatures above 350C at the upper end of the spring and the thermal fatigue in general of the components under many duty cycles. Some of these issues are still under current investigation with the aim to provide the operational boundary input conditions for JET operation. A more detailed description of the R&D steps involved in this project is out of the scope of this paper but can be found in [28, 29, 30, 31].

A prototype of this concept has been successfully tested in cycling heat flux tests (200) with 7 MW for 10s and failure test with 10 MW for 14s resulting in surface temperatures up to 3000 °C. After HHF tests, microcracks developed at the bulk W surface, as shown in fig 7. This is normally observed under such power load conditions in W. The development of these microcracks is considered to saturate under multiple thermal loading due to stress relieve and, from present view, not considered to pose harm on the W operational behaviour.

As for all wall tiles in JET, each individual W lamella tile is shaped in toroidal direction with additional shaping from one tile to the next and poloidally from stack to stack. The design was driven also by the large variety of JET plasma configurations leading to a range of field line impact angles. Thus a compromise was to be found between avoiding edge exposure at large impact angles and reducing wetted areas of the tile at more glancing plasma impact.

## 6. Summary

JET is underway to completely exchange the present wall materials (CFC carbon) towards a metal dominated ITER-like material mix with mainly solid Be tiles in the main chamber and a full W divertor. This large effort is motivated by a number of outstanding questions associated with the use of metallic PFC and the restricted operational experiences under such conditions. Solid bulk Be tiles will be used for the main wall inner and outer guard limiters and the upper dump plates which will take the major fraction of main wall particle and power load. The inner wall NBI shinetrough areas will use W coated CFC tiles but recessed by 2.5 cm to the Be wall tiles. Be coating (7–9 $\mu$ m) on cast Inconel cladding tiles will be used for the JET inner wall and the upper dump plate carriers. Most of the divertor area will use W coated CFC tiles but a W bulk row for the most loaded outer diverter tile in ITER like configurations. To identify a viable

industrial-scale solution for the W coating of 2D Dunlop CFC tiles, a significant number of different coating types and thicknesses were investigated in high heat flux tests. This has provided a significant gain of experiences on W coating techniques. Finally PVD coatings of  $10\mu$ m micron thickness produced by combined magnetron sputtering and ion implantation (CMSII) in combination with a molybdenum interlayer were selected. The inertially cooled bulk tungsten tile design is based on a tungsten lamellae concept with insulation and dedicated contact points, driven to reduce the EDDY and halo forces in case of disruptions and to increase the mechanical stability under thermal stresses.

The ITER-like Wall Project and the exploration of ASDEX-Upgrade with a full tungsten wall are important steps to tackling the wall materials issues for the medium (ITER) and long term future (DEMO) and demonstrate the strong European commitment in this area. Besides a number of specific topics to investigate, the development of wall compatible plasma scenario and control and machine protection are most important tasks and lessons to be learned for the operation of ITER.

## References

- <sup>1</sup> J. Pamela , G.F. Matthews, V. Philipps et al , J. Nucl. Mat. 363–365 (2007) ,1–11
- <sup>2</sup> Joachim Roth , E. Tsitrone , A. Loarte , J. Nucl. Mat. 390-391, (2009), 1-9
- <sup>3</sup> G Counsell, P Coad, C Grisola, Plas. Phys. Control. Fusion 48, (2006), B189–B199
- <sup>4</sup> J. Pamela , F. Romanelli , M.L. Watkins, Fus. Engin. and Design 82, (2007), 590–602
- <sup>5</sup> W Fundamenski et al., Nucl. Fusion **44**, (2004), 20.
- <sup>6</sup> R.A. Pitts, G. Arnoux, M. Beurskens, J. Nucl. Mat., 390-391, (2009), 755-759
- <sup>7</sup> A. Huber, R.A. Pitts, A. Loarte, J.Nuc. Ma. 390-391, (2009), 830-834
- <sup>8</sup> G Federici et al., Plasma Phys. Control. Fusion 45, (2003), 1523.
- <sup>9</sup> W. Fundamenski, R.A. Pitts , J. Nucl. Mat., 363-365, (2007), 319-324
- <sup>10</sup> V Riccardo et al., Nucl. Fusion 45, (2005), 1427.

- <sup>12</sup> I. Nunes, P. de Vries, P.J. Lomas, Fus. Engin. and Design, 82, 15-24, (2007), 1846-1853
- <sup>13</sup> T. Hirai, H. Maier, M. Rubel, Fus. Engin. and Design, , 15-24, (2007), 1839-1845
- <sup>14</sup> M J Rubel, V Bailescu, J P Coad, J. Phys.: Conf. Ser. **100** Volume 100 (2008), 062028 (4pp)
- <sup>15</sup> M. Mayer, R. Behrisch, K. Plamann, J. Nucl. Mat. 266±269, (1999), 604±610
- <sup>16</sup> G.F. Matthews, P. Coad, H. Greuner, J.Nucl. Ma., 390-391, (2009), 934-937
- <sup>17</sup> H. Maier, R. Neu, H. Greuner, J. Nucl. Mat., 363-365, (2007), 1246-1250
- <sup>18</sup> R Neu, H Maier, E Gauthier, Phys. Scr. **T128**, (2007), 150-156
- <sup>19</sup> H. Maier, T. Hirai, M. Rubel, Nucl. Fusion **47** No 3, (2007), 222-227
- <sup>20</sup> H Maier, Mater. Sci. Forum, (2005), 475–9, 1377–82
- <sup>21</sup> J. Luthin, Ch. Linsmeier, J. Nucl. Mat. 290-293, (2001), 121

<sup>&</sup>lt;sup>11</sup> V. Thompson, Y. Krivchenkov, V. Riccardo, Fus. Engin. and Design, 82, 15-24, (2007), 1706-1712

<sup>22</sup> D. Hildebrandt, P. Wienhold, W. Schneider, J. Nucl. Mat. 290-293, (2001), 89

<sup>23</sup> H. Maier, R. Neu, H. Greuner, 12th PFCM workshop, Jülich , May 2009, accepted for publication in Physica scripta

<sup>24</sup> C Ruset, E Grigore, H Maier Phys. Scr. **T128** ,(2007), 171-174

<sup>25</sup> M. Mayer, J. Likonen, J.P. Coad, J. Nucl. Mat., 363-365, (2007), 101-106

<sup>26</sup> P.H. Mertens et al 12th PFCM workshop, Jülich , May 2009, accepted for publication in Physica scripta

<sup>27</sup> S. Grigoriev, Ph. Mertens, O. Neubauer, Fus. Engin. and Design, 84, 2-6, (2009), 853-858

<sup>28</sup> Ph. Mertens, H. Altmann, T. Hirai, Fus. Engin. and Design, 84, 7-11, (2009), 1289-1293

<sup>29</sup> Ph. Mertens, H. Altmann, T. Hirai, J. Nuc. Mat., 390-391, (2009), 967-970

<sup>30</sup> Ph. Mertens, T. Hirai, J. Linke, Fus. Engin. and Design, 82, 15-24, (2007), 1833-1838

<sup>31</sup> T. Hirai, H. Maier, M. Rubel, Ph. Mertens, Fus. Engin. and Design, 82, 15-24, (2007), 1839-1845

## Figure captions:

Fig 1: Outer poloidal limiter assembly (right) and detailed view of one of the tile carrier with fixing bolts and Inconal cast support (left).

Fig2: Map of the inner wall material configuration. The tungsten areas were needed to increase the neutral beam shine-through energy handling and maintain the shine-through power handling. The beryllium tiles are 2.5cm forward of the W coated CFC tiles in the shinethrough area

Fig 3: Material configuration for the tungsten divertor with the lower single null W divertor (left) and tile configuration (right). Tiles 0,1,3,4,6,7,8 will be 2D CFC with 14  $\mu$ m (CSMII, see text) W-coatings. Tile 5, the high heat flux area for ITER-like configurations will be made from bulk tungsten

Fig4: Typical appearance of cracking of W coatings after high heat load (vertical direction) with also buckling of the coating along the CFC fibres (horizontal direction).

Fig 5: Appearance of VPS W-Re multilayers as received (1), after baking in vacuum at 960C (3h) and 3 days after storage in air

Fig 6: Design of the outer W bulk tile, consisting of the W tile with 4 stacks (1) and individuals lamellas (5), the supporting wedge with individuals fingers (2) and the base carrier (3). The lamellas insulated by coated Mo spacers

(7) and are hold together by a chain made of Densamet and Nimonic (6) which is pulled down by screws with springs (8).

Fig 7: Development of microcracks in W lamellas after high heat exposure (7.5 MW, 10s) in e-beam facility (JUDITH







Fig 3: Material configuration for the tungsten divertor with the lower single null W ivertor (left) and tile configuration (right). Tiles 0,1,3,4,6,7,8 will be 2D CFC with 14  $\mu$ m (CSMII, see text) W-coatings. Tile 5, the high heat flux area for ITER-like configurations will be made from bulk tungsten





Fig 5: Appearance of VPS W-Re multilayers as received (1), after baking in vacuum at 960C (3h) and 3 days after storage in air

