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ScienceDirect
Journal of Nuclear Materials 367–370 (2007) 1438–1443
journal of nuclear materials
www.elsevier.com/locate/jnucmat

R&D on tungsten plasma facing components for the JET ITER-like wall project

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Abstract

Currently, the primary ITER materials choice is a full beryllium main wall with carbon fibre composite at the divertor strike points and tungsten on the upper vertical targets and dome. The full tungsten divertor option is a possibility for the subsequent D–T phase. Neither of the ITER material combinations of first wall and divertor materials has ever been tested in a tokamak. To collect operational experience at JET with ITER relevant material combination (Be, C and W) would reduce uncertainties and focus the preparation for ITER operations. Therefore, the ITER-like wall project has been launched to install in JET a tungsten divertor and a beryllium main wall. This paper describes the R&D activities carried out for the project to develop an inertially cooled bulk tungsten divertor tile, to fully characterise tungsten coating technologies for CFC divertor tiles and to develop erosion markers for use as diagnostics on beryllium tiles.

1. Introduction

Currently, the primary ITER materials choice is a full beryllium wall, carbon fibre composite (CFC) at the divertor strike points and tungsten on the upper vertical targets and dome. There is however a concern that tritium retention may still be unacceptably high due to chemical erosion of the CFC near the strike points and transport of hydrocarbons to shadowed areas [1]. The tritium inventory in ITER could reach its maximum permitted level of 350 g after a certain number of D–T pulses, and fuel removal techniques are not fully developed yet. Therefore, an all-tungsten divertor is still included in ITER planning as a back-up option for the D–T phase. However, this divertor option carries its own risks of melt damage and plasma contamination. Neither the ITER reference combination of first wall and divertor materials nor its back up have ever been tested in a tokamak.

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0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2007.04.039

At JET, with its unique scientific and technical capabilities such as the use of tritium and beryllium, it is possible to test a full combination of ITER wall materials; furthermore, the JET size, magnetic fields and plasma currents make it possible to conduct tests at the most ITER-relevant parameters accessible today.

The feasibility of an experiment at JET with tungsten in the divertor and beryllium as wall material has been assessed, and a project has been launched to design, manufacture and test all the necessary components, which could be installed in a dedicated shutdown in 2008.

This paper describes the present status of the ITER-like wall project, the on-going R&D for the tungsten divertor (both tungsten bulk and W-coated tiles) and the development of beryllium marker tiles.

2. The ITER-like wall project

JET has both the technical capabilities of handling tritium and beryllium and the possibility to make experiments with ITER relevant plasmas conditions. Since plasma optimisation is influenced by the wall materials (e.g. erosion, edge radiation, edge plasma control requirements, core radiation, impurity accumulation in advanced scenarios), the JET ITER-like wall (ILW) project can provide operational experience with an ITER relevant combination of materials. Thus, the main objective of the ILW project at JET is to replace CFC in all main limiter and protection tiles (Fig. 1) with bulk beryllium and to minimise in the divertor the amount of carbon surfaces facing the plasma.

The ILW project reference option is a fully tungsten coated CFC divertor, whereas, an alternative option includes CFC tiles for the strike point areas of the divertor (equivalent to the current ITER reference configuration). This choice for the reference configuration is driven by the fact that, if a carbon/tungsten divertor was tested first in JET, and carbon migration was not adequately suppressed by beryllium deposition [2–4], then it might prove difficult to decontaminate the machine for a subsequent all tungsten divertor/beryllium wall phase.

Where possible, beryllium will replace the exposed CFC components of the outer and inner limiters and the upper dump plates (Fig. 1). The use of beryllium instead of CFC required the modification of the design of some of the components in order to satisfy the JET thermo-mechanical requirements. The procurement of the beryllium tiles is compatible with

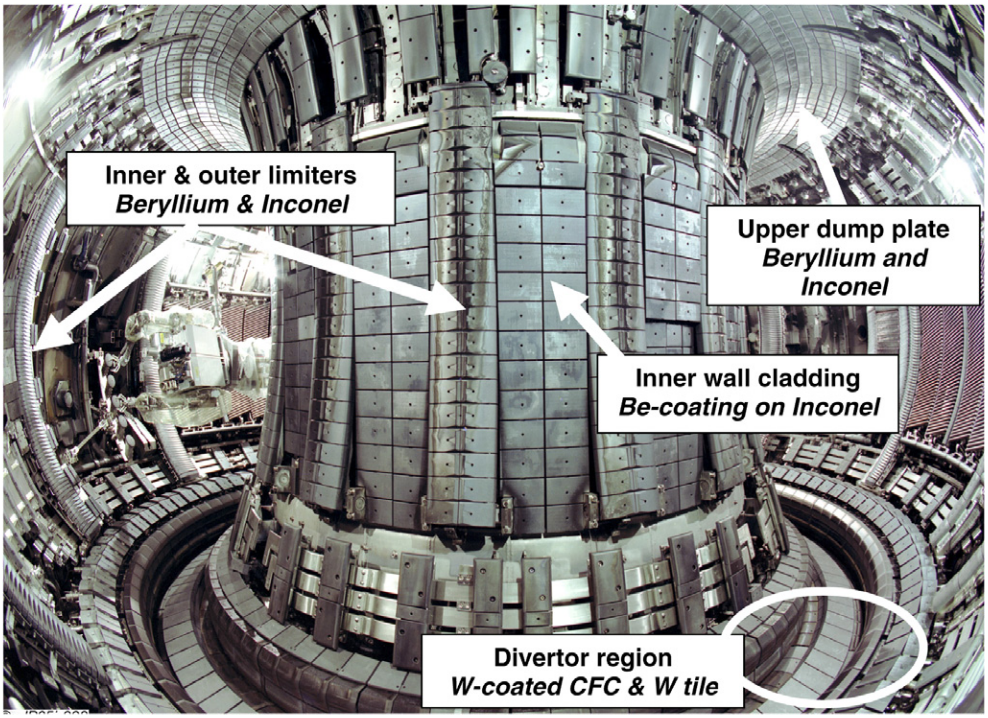

Fig. 1. JET chamber with indicated main plasma facing components.

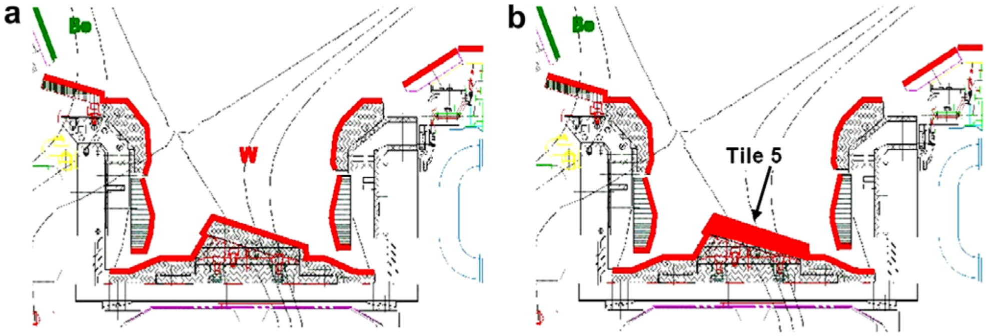

Fig. 2. JET divertor region with the two options: (a) all W-coatings on CFC; (b) W-coating on CFC and bulk tungsten for tile 5.

the general project time schedule and costs due to the availability of 5 tons of beryllium components previously used in JET, which will be recycled.

The tiles of the inner wall cladding are not usually subjected to high heat loads, and only a minimal all-metal (Inconel 625) design is required to protect parts such as the Al–bronze bolts and inserts. To reduce medium Z impurities, it is planned to coat the Inconel tiles with 8 μm beryllium before installation.

A decision on the final material configuration in the divertor and whether to adopt only W-coated CFC tiles (Fig. 2(a)) or to include a bulk tungsten tile in the design (Fig. 2(b)) will be taken in the first months of 2006 following further input from the ITER Team and the outcomes of the presently ongoing R&D activities on the characterisation of W-coating technologies for the CFC divertor and inner wall neutral beam shinethrough protection tiles, the design of an inertially cooled W-bulk tile to be used in areas near the strike point, the manufacturing of coated beryllium tiles to be used as markers to study erosion processes, and the evaluation of Be coatings on Inconel.

3. R&D in the ITER-like wall project

3.1. W-coating of CFC

One possibility to equip the JET divertor with an all-tungsten divertor surface is the application of tungsten coatings. The feasibility of this rather inexpensive solution has been proven in the tungsten program of the ASDEX-Upgrade tokamak in Garching (Germany) [5]. To assess the applicability of coatings in JET, new R&D activities have been launched, since, differently from ASDEX-Upgrade, for the JET divertor the coating would have to be performed on a 2-D CFC with a strong anisotropy in thermal conductivity and expansion coefficients. Depending on the position, tungsten coatings with different thicknesses might be necessary. The following reference thicknesses have been chosen: thin coatings (4 and 10 μm), thick coatings (∼200 μm). Thick coatings may be required in areas subjected to particularly high erosion, if a technically and financially viable solution for bulk tungsten technology is not found. Five European Research Institutions (CEA, ENEA, IPP, MEdC, and TEKES) have developed and optimised different coating technologies mostly in co-operation with industry.

Thin coatings with thicknesses of 4 and 10 μm were produced by chemical vapour deposition (CVD) and various methods of physical vapour deposition (PVD). Thicker coatings of 200 μm were produced by CVD and vacuum plasma spray (VPS). In total, 14 different sample types have been manufactured. Characterisation and thermal load testing are ongoing to select the most promising combinations of thickness and production method. The new high heat flux device GLADIS at IPP-Garching [6] is used for the testing. In a first thermal screening, power densities ranging from 4 MW/m^2 for 6–10 s to 22 MW/m^2 for 1–1.5 s were applied producing surface temperatures up to around 2000 °C. Fig. 3 shows a sample with 10 μm PVD tungsten coating and the infrared images of two coated tiles with 10 μm and 200 μm W-coatings at the end of a 10 s pulse at 4 MW/m^2 and a 4 s pulse at 11 MW/m^2, respectively. The 10 μm coating failed by delamination when the temperature at the beam centre was about 1450 °C and the part of the coating without contact to the substrate overheated in the pulse to a temperature beyond the dynamic range of the infrared camera (above 2100 °C). The thermal screening has been completed, and 10 out of 14 samples passed the tests. These samples are now undergoing a characterisation phase in which

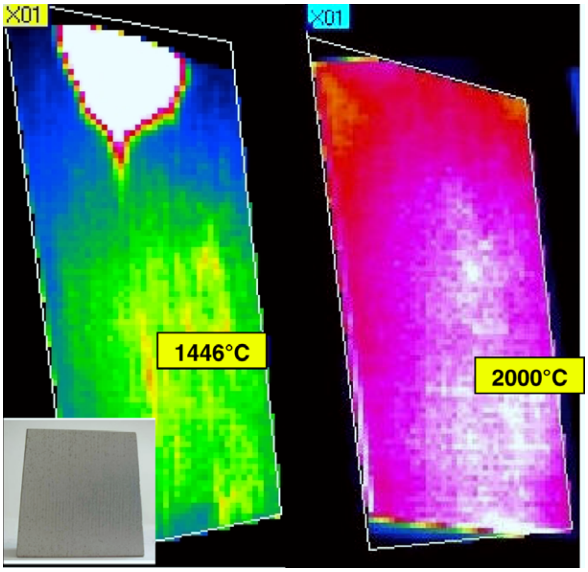

Fig. 3. W-coated tile (lower left corner) and infrared images during high heat flux testing: 10 μm coating, 10 s pulse at 4 MW/m^2 (left) and 200 μm coating, 4 s pulse at 11 MW/m^2 (right). The given temperatures correspond to the centre of the hydrogen beam. The bright white area in the tile on the left is caused by delaminating of the coating, which overheats beyond the dynamic range of the infrared camera.

sputtering X-ray photoelectron spectroscopy (XPS), scanning electron microscopy (SEM) and pulling tests will be performed to assess the level of impurities (oxygen and carbon), the thickness (4, 10 and 200 μm), and the uniformity and adhesion of the coatings.

Afterwards, a selected number of coating types will undergo a thermal cycling process at fixed pulse conditions. It has been assessed that, during cyclic loading, the peak surface temperature must be in the range 1300–2000 °C. Therefore, initially the tests are planned with 300 pulses at ~10 MW/m^2 for ~5 s. However, power density and/or power duration could be modified if necessary to reach the required temperatures. This thermal cycling is considered necessary, since the thermal expansion mismatch between thick coatings and the substrate leads to crack formation. The evolution of these cracks and whether they are benign will be assessed.

In parallel to this program, infrared-based non-destructive testing methods (lock-in infrared thermography and flash lamp thermography) are being evaluated for the application for quality assurance during the production of the coatings for the JET divertor. In addition to those two methods, X-ray radiography is being assessed to examine the uniformity of the industrially manufactured coatings.

In early 2006, the distribution of the coatings required in different areas of the JET chamber will be specified. The decision will have to balance the need to have a reasonably high lifetime against erosion (thick coatings) with the necessity of keeping thermo-mechanical stresses in the coatings as low as possible (thin coatings).

3.2. Bulk W tile

Coating performance and lifetime may be a limiting factor in long-term operations. In addition, the availability of a bulk tungsten tile in the divertor region would also make possible the study of the behaviour of melt layers in the divertor created during ELMs or disruptions. Therefore, R&D is being performed for the design of an inertially cooled bulk W tile for use in areas near the strike point (tile 5 in Fig. 2(b)). Various concepts of bulk W tiles are being explored that must satisfy the stringent JET requirements.

One of the critical points in the design of the bulk tungsten tile is to reduce the electromagnetic forces, especially those induced during plasma disruptions. The tiles are supposed to sustain forces due to 18 kA halo current per assembly and eddy currents generated by a magnetic field changing with 100 T/s rate. To reduce forces caused by halo currents, electric paths to the ground have to be defined. To reduce eddy current forces, the possible eddy current loops have to be minimised by segmenting the metallic parts. The solution that is currently considered is the W-lamellae design (Fig. 4). Most W blades have a standard width of 6 mm in the toroidal direction, and thicker blades are used at the lateral edges of each tile ($\leqslant$13 mm) to accommodate heads of fixing bolts in the blade body and to compensate for the toroidal curvature. The supporting structures, ‘wedge’ and ‘adapter plate’ were modified to reduce electromagnetic forces and reinforce the mechanical

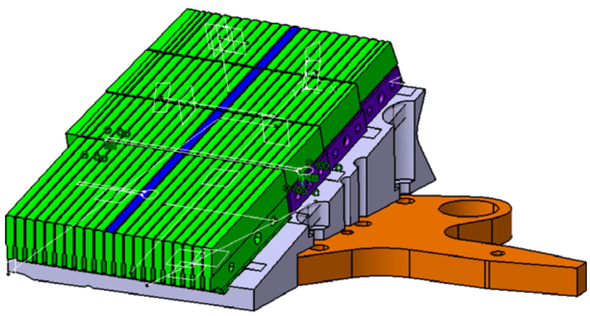

Fig. 4. Bulk W-lamellae design for tile 5.

stiffness. The optimisation of the design of these parts is in progress.

Another critical point to be considered is the heat capacity of the W blade, which impacts on the temperatures during plasma pulses. The bulk W tile will be exposed to high heat flux without direct active cooling (~7 MW/m^2 for 10 s). Consequently, during a shot, the surface temperature will increase continuously until the end of plasma discharge. With the given design constraint on the total height of the bulk W tile (blade's height), it has been estimated that the W surface temperature will reach more than 2300 °C during ELM loads. W grade selection is ongoing with respect to re-crystallization resistance and mechanical strength at elevated temperature. W brazing on CFC and W/TZM joining by electron beam welding are also being considered as fall-back options.

In addition to the compatibility with JET thermal and mechanical requirements, the feasibility of the investigated designs at an industrial scale is also an essential parameter to be taken into account in the final assessment.

3.3. Beryllium marker tiles

The main chamber wall is a net-erosion dominated area [4]. The assessment of the beryllium erosion rate is of great interest for ITER, and it is also a necessary and important objective of the overall ILW project. To measure the eroded thickness, the so-called marker tiles will be used. These are standard-size beryllium tiles coated with a sandwich stripe of high-Z metal (2–3 μm) and beryllium (7–8 μm) films on the plasma-facing surface. The high-Z metal interlayer is applied between the Be tile and the surface film to enable the measurement of the eroded Be thickness of up to 8 μm. Several metals have been considered for the interlayer (Ni, Re, W), taking into account the thermo-mechanical compatibility (thermal conductivity and linear expansion coefficient). The final selection of the metal will be based on the results of the R&D programme that comprises the manufacturing and characterisation of the coatings: morphology and behaviour under moderate heat flux testing.

To ensure the assessment of erosion deeper than 8 μm, a series of precise notches 10 and 20 μm deep will be machined on the tile edges. Pre-characterised marker tiles will be placed at several locations in the arrays of outer poloidal and inner wall guard limiters (Fig. 1). After a few-months campaign, the tiles will be retrieved from the torus. Ex-situ analyses using a number of material characterisation methods will provide the information on the net erosion. The proof of principle has been tested in short- and long-term experiments [4,7,8] and marker tiles have been in regular use at JET during all campaigns since 1999.

4. Summary and final remarks

The ILW project has been launched at JET with the aim at testing the reference material combination chosen for ITER. In this frame, challenging R&D has been launched on technologies for W-coating of CFC tiles, manufacturing of beryllium marker tiles and design of an inertially cooled bulk W divertor tile module. The first results on W-coating and on the design of the bulk W tile for the divertor are promising. Based on the outcomes of these activities, a decision will be made in early 2006, on the thickness to adopt for the coatings in different part of the divertor and whether it is feasible to install full tungsten tiles.

In this way, JET, with its unique scientific and technical capabilities such as the use of tritium and beryllium, could bring operational experience in steady and transient conditions with the ITER first wall and divertor materials (Be, C and W). This could have significant benefits for the efficiency of ITER exploitation, by reduction of the uncertainty regarding tritium retention in ITER, the strategies needed to limit it, the evaluation of the physics viability of the all-tungsten back-up option and development of plasma control and diagnostic strategies for protecting the beryllium wall. The on-going R&D, besides being a successful example of collaboration in Europe between several different national research institutions, is a most comprehensive test of different technologies for W-coating of CFC. This has potential relevance for the use of W-coating on other tokamaks such as ASDEX-Upgrade, the other European tokamak with ITER shaped plasmas and full metal tungsten plasma facing components.

Acknowledgments

This work has been performed within the European Fusion Development Agreement and the Authors want to acknowledge gratefully the essential contributions provided by E. Gauthier, R. Mitteau, X. Courtois, C. Grisolia from CEA-France, G. Maddaluno, B. Riccardi from ENEA-Italy, C. Ruset and C. Lungu from MEdC-Romania, J. Likonen from TEKES-Finland, G. Counsell from UKAEA-UK and I. Uytdenhouwen from SCK-CEN-Belgium.

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