Available online at www.sciencedirect.com
ScienceDirect
Journal of Nuclear Materials 362 (2007) 174–180
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Actively cooled plasma facing components in Tore Supra: From material and design to operation

Ph. Magaud *, P. Monier-Garbet, J.M. Travère, A. Grosman

Association Euratom-CEA, DSM/DRFC/SIPP, CEA-Cadarache, F-13108 St-Paul-Lez-Durance, France

Abstract

In current fusion devices, the components located in front of plasma, the so-called plasma facing components (PFCs), need to sustain severe constraints such as high thermal flux (several MW m^{-2}), erosion and flux of particles. Feedback from these challenging components is essential for the success of the next generation of components, in particular in term of manufacturing or handling intense heat loads. Tore Supra actively cooled high heat flux PFCs are able to sustain up to 10 MW m^{-2} during long plasma pulses. They are at present the only ones in operation in a fusion device. They are described in details from design (including the testing programme used for concept validation) to operation. Lessons learned from the industrialization programme, which could be essential for ITER, will be presented. Finally, the experimental feedback with actively cooled walls, including in situ monitoring to guarantee plasma facing component safety, will be summarized. Another very important feature has been noticed, namely the in vessel progressive uptake of hydrogen, a likely source of concern for ITER.

PACS: 28.52.−s; 28.52.Fa; 52.55.Fa; 81.05.Mh

1. Introduction

At the core of the sun and stars, light nuclei combine – or fuse – to create heavier nuclei. This process releases a significant quantity of energy and is the source of the heat and light that we receive. Harnessing this type of reaction on earth for the purpose of generating energy would open the way to almost unlimited resources. This is the aim of fusion research undertaken by the leading industrial nations. In the sun and the stars, the conditions necessary for fusion: temperature, density and confinement time, are maintained by gravity. This is impossible to implement on earth and at present, magnetic confinements are the best solution to maintain the conditions necessary for fusion and to contain, as if in an intangible box, the hot plasma.

The first components located in front of the intangible box, the so-called plasma facing components (PFCs), need to meet different requirements. The management of this first material interface is critical from a confinement point of view. The main results obtained on confinement were based on the efficient management of this interface. For instance, plasma radiation losses vary significantly with the quantity and type of impurities that it contains

* Corresponding author. Tel.: +33 4 4225 4308; fax: +33 4 4225 4990.
E-mail address: philippe.magaud@cea.fr (Ph. Magaud).

0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2007.01.020

(radiation increases with the atomic number of the relevant impurity). The plasma facing wall must therefore be designed, on the one hand, to minimize this type of pollution and, on the other, to resist to particles and radiation from the plasma leading to erosion, sputtering, etc. It must also extract particles during deconfinement (reaction ashes: helium, deuterium, tritium, impurities) in order to control plasma density and to extract heat (the heat flux can reach several MW m^{-2}). These challenging multi-purpose components are in operation inside the Tore Supra Tokamak device, located in Cadarache. After presenting the development up to industrialization of actively cooled plasma facing components, Tore Supra operation with them is discussed, before a last part is devoted to the ITER challenges in the related fields.

2. Tore Supra plasma facing components

Most of the present fusion devices allow physical studies on time scales ranging from a fraction of a second to a few seconds. Technology is often the major limitation as in the case of non-actively cooled first walls. Tore Supra has been designed and developed to operate with technologies allowing long plasma duration (a few minutes), thanks to the use of superconducting magnets and high heat flux actively cooled plasma facing components. Tore Supra is the only large tokamak including these two features since the startup of the machine in 1988.

The first generation of Tore Supra inner components (Fig. 1) was designed to operate with a plasma discharge of 30 s and sustain an incident heat flux of about 1 MW m^{-2}. These components were made in graphite tiles brazed via a molybdenum/copper interlayer onto stainless steel pipes. This technology showed some limitation leading to debrazing or broken tiles due to the low reliability of the brazing procedure. At that time, non-destructive tests were not able to detect accurately enough some defects such as braze-voids. Moreover, the characteristics of graphite in terms of thermal shock resistance were limited in the case of high energy particle impact during plasma disruptions.

Tore Supra inner components were upgraded in 1995 and in 2002. These upgrades were focused on the design itself but they could not be achieved without the development of destructive and

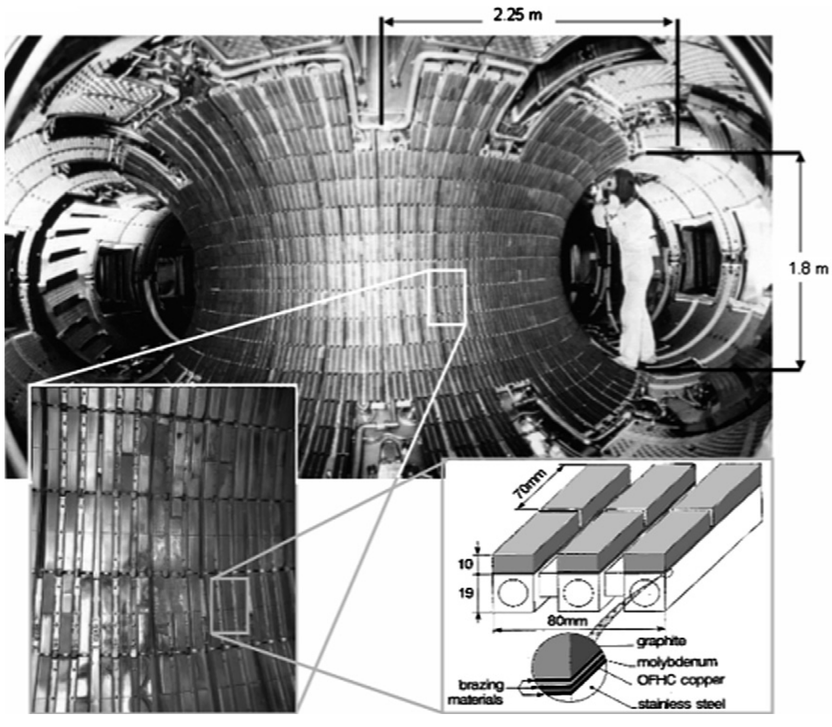

Fig. 1. View of the first generation of Tore Supra inner components. These components were made in graphite tiles brazed via a molybdenum/copper interlayer onto stainless steel pipes and were designed to operate with a plasma discharge of 30 s and to sustain an incident heat flux of about 1 MW m^{-2}.

non-destructive tests. Such components requires qualification tests in high heat flux facilities to assess the development path but also to monitor the industrial process. The CEA, Areva and the EU Fusion program developed in 1991 a collaboration for the design, building and operation of a 200 kW electron beam facility, named FE200, a French acronym for Electron Beam 200 kW (Fig. 2). FE200 is mainly dedicated to thermo-mechanical testing of plasma facing components for fusion applications. The components are installed in a large vacuum chamber (8 m^3), connected to a water loop, heated by the electron beam sweeping and monitored by a number of diagnostics. Since 1992, over 100 000 cycles of thermal fatigue tests have been made on about 100 mock-ups. Thermal hydraulics, critical heat flux (>200 points), disruptions, glancing incidence tests have also been performed thanks to the high level operational flexibility of the facility [1]. The FE200 is currently mainly dedicated to the qualification of plasma facing component prototypes manufactured by the European industry for the ITER Divertor. Concerning non-destructive tests, the CEA developed a specific facility mainly devoted to PFC acceptance tests, the SATIR test bed [2]. The surface temperature behavior of tested components is monitored by infrared thermography during hot water cooling followed by cold water cooling. The surface temperature behavior is compared to a reference component and 3–2 mm defects can be detected. These tests are fully correlated with high heat flux tests in FE200.

The main characteristics of the present Tore Supra Inner Components and Limiters (CIEL, Composants Internes Et Limiteur) are described in [3,4]. These components can extract 25 MW for a plasma discharge of up to 1000 s. The bottom of the Tore Supra inner vessel is a flat toroidal pumped limiter (TPL) covered with 7.5 m^2 of actively cooled tiles in carbon fiber composite (CFC). These components are able to sustain up to 10 MW m^{-2}, 10 times more than the first generation. They are composed of 6 mm CFC flat tiles joined to a thin copper layer, itself welded by electron beam on the structural material, made of CuCrZr. In order to increase joining efficiency, the surface of the tile to be joined is laser treated to obtain micro-holes. This technology, called Active Metal Casting (AMC®) was developed by the Austrian Plansee company. A schematic view of these components is displayed in Fig. 3. About 600 of these components have been manufactured to build the TPL and have been in operation since 2002. They mainly contributed to the world record in steady state operation on December 4, 2003, with 1 GJ injected and extracted from a 6 mn plasma discharge. However, in spite of significant design work, difficulties appeared during the manufacturing phase, mainly due to a relative

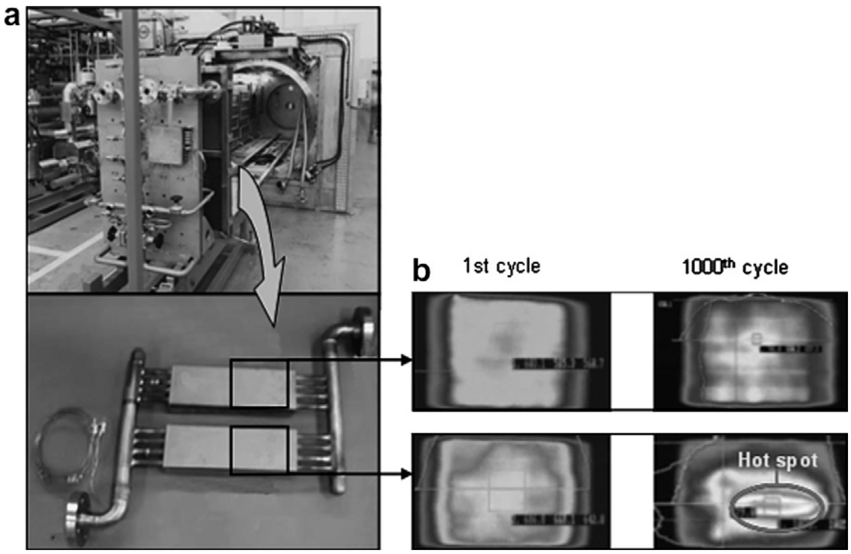

Fig. 2. (a) Vacuum chamber of the FE200 facility, located in AREVA, Le Creusot (France, 71). The FE200 (French acronym for Electron Beam 200 kW) is an high heat flux facility (up to 1 GW m^{-2}) dedicated to the qualification of plasma facing components prototypes. Since 1992, more than 100 000 cycles of thermal fatigue have been operated on about 100 mock-ups. (b) Example of testing: infrared monitoring of thermal fatigue cycling (1000 cycles at 5 MW m^2, 15 s) of a CuCrZr/stainless steel mock-up.

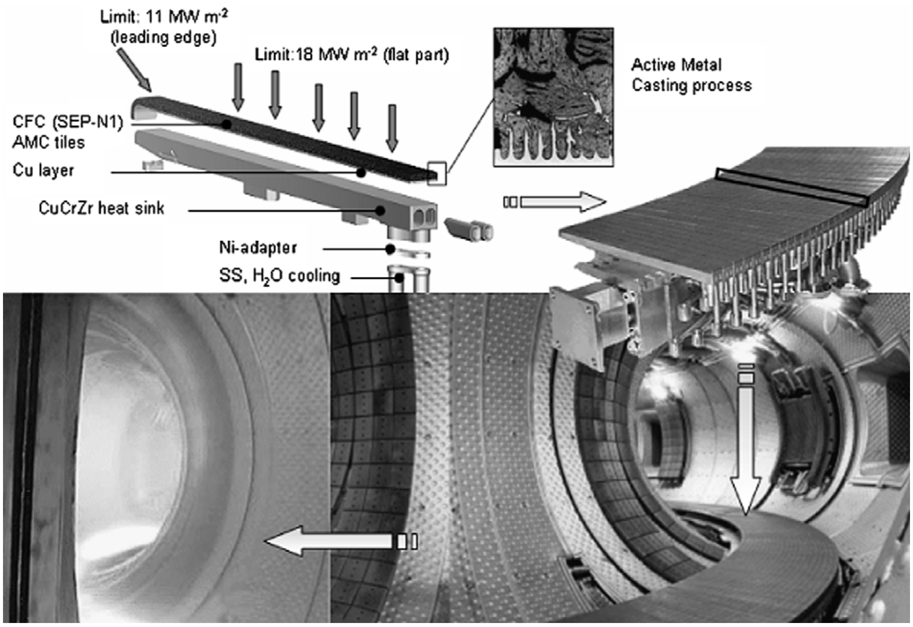

Fig. 3. Schematic representation of the high heat flux TPL element and of the Tore Supra inner vessel without and with plasma. These components are able to sustain up to 10 MW m^{-2}, 10 times more than the first generation.

lack of fundamental knowledge concerning material behavior. Different batches of CFC were provided by the supplier and displayed inhomogeneous characteristics due to the very complex CFC manufacturing route. These led to unexpected defects in the bonding process result and in rejecting different elements. As a consequence, a repair process allowing the replacement of tiles with defects was developed. Another example is the recurrent CuCrZr EB welding difficulties (repeated occurrence of micro-cracks) observed during various Tore Supra component manufacturing. The CEA started in 2005 a research program to understand these phenomena and develop EB homogeneous welding criteria for CuCrZr alloy [5].

3. Operating with actively cooled walls

Tore Supra has operated with its new components since 2002. After 4 years of operation, no evidence of defects has been observed, thus resulting in the validation of the design and the manufacturing process of such PFCs. On actively cooled machines, major risks associated to overheating of PFCs are component damage or water leaks in the vacuum chamber. In the nuclear working environment of ITER this point will become very important. Two major phenomena must be supervised during plasma operation: (i) defects on actively cooled components like limiter or antennas protections and (ii) transitory events like unexpected hot spots appearance or electric arcing on the LHCD grill.

During plasma discharge, power deposition measurements on the TPL are mainly performed by a set of seven actively cooled endoscopes equally spaced around the torus. The current configuration allows dynamic imaging of four sectors of the TPL, the three ion cyclotron resonance heating (ICRH) antennas and the two lower hybrid current drive (LHCD) launchers. Each endoscope (2.5 m long) is equipped with three viewing lines: two IR cameras able to survey $2 \times 35°$ of the TPL and one radio frequency antenna. The temperature is measured with infrared focal plane array cameras having the ability to deliver maps between room temperature and 1773 K. The infrared data are used for both safety (real time feedback control) and survey (differed time image analysis for overheating study and understanding) (Fig. 4). During the shot, the temperature of the selected areas is monitored and the heating power is controlled (if needed) to prevent temperature from increasing above a selected threshold [6].

In situ examinations are also carried out during maintenance periods (such as lock-in examinations) and could be routinely done in 2007 with an Articulated Inspection Arm (AIA). This remote handling

Fig. 4. Infrared monitoring of the lower hybrid couplers temperature during plasma discharge: (a) visible and (b) infrared imaging.

tool is under development at the CEA within the framework an ITER R&D task. It should operate under vacuum (10^{-6} Pa) and temperature (393 K) and provide visual inspections or diagnostics (Fig. 5) [7]. The development of this tool, its manufacturing and its future operation under real tokamak environment will lead to precious feedback for ITER.

In present Tokamak, carbon is used as main PFC. Carbon was chosen for the present generation of experiments because it is a low Z element, with fully stripped ions and therefore no radiation into the plasma core, and for its good thermal properties. However, it is a common feature on all the devices using carbon, that carbon retains large amounts of deuterium. Deuterium retention into carbon walls is observed with different time scales: (1) the instantaneous rate of deuterium that is injected into the vessel to fuel the plasma discharge is higher than the deuterium exhaust rate, as shown in Fig. 6 [8]. Also, the integrated amount of deuterium recovered at the end of a plasma discharge is generally lower than the integrated injected amount. This corresponds to a 'dynamic' or short term retention and (2) only part of the deuterium retained at the end of a plasma discharge is recovered in between plasma operation by standard conditioning techniques. As a result, the vessel progressively stores deuterium ('long term' retention). The retention rate during a plasma discharge is constant on Tore Supra, up to 6 mn plasma duration, and an identical shot to shot behavior is observed up to 15 mn cumulated plasma operation. This indicates that the actively cooled carbon wall is still not

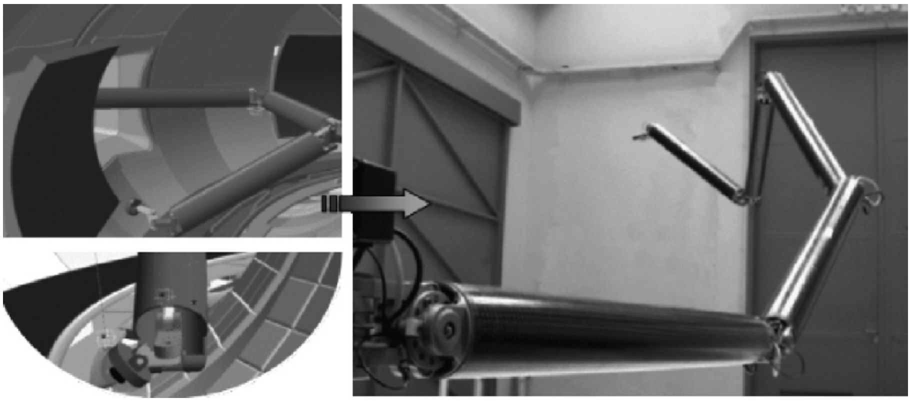

Fig. 5. The Articulated Inspection Arm (AIA), a remote handling tool able to operate under vacuum (10^{-6} Pa) and temperature (393 K). It will provide in situ PFC visual inspections and diagnostics.

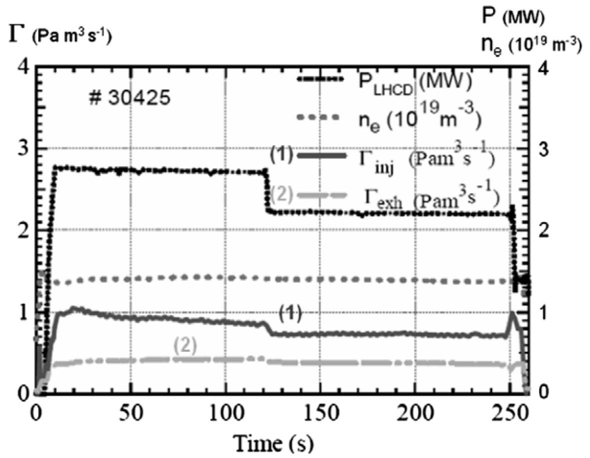

Fig. 6. Time evolution of the injected (1) and exhausted (2) deuterium rate during plasma discharge. Due to wall retention, the integrated amount of deuterium recovered at the end of a plasma discharge is lower than the integrated injected amount.

saturated after this time. In non-actively cooled devices [9], wall saturation is observed, and is correlated to the surface temperature of the main chamber walls. Short term retention is consistent with an adsorption process, showing progressive saturation when porosity is filled [10]. This is a weak physical bond, and deuterium is easily released. On the other hand, long term retention is mainly associated with the two following processes: (1) co-deposition of hydrogen or deuterium with eroded wall material forming hydrogen rich co-deposits in locations which are not wetted by plasma operation and/or usual conditioning methods. Soft films with high T/C ratio (up to 0.8) have been found in JET louvers, hidden from the plasma and kept at moderate temperature (323–373 K). This process is presently generally considered as the dominant long term fuel retention mechanism since the associated vessel inventory increases linearly with the discharge duration and cannot be avoided if carbon erosion and transport take place. In AUG Tokamak, the increased proportion of tungsten in the machine has not modified the deuterium retention showing that the dominant retention processes are not yet significantly modified although 65% of the first wall is not in carbon and (2) an additional long term retention may occur through processes like implantation of deuterium into carbon, followed by diffusion in the bulk. Implantation corresponds to strong chemical bonds, with no release of deuterium except by heating or conditioning. These processes are expected to occur in the main plasma interaction zones. This is supported by recent laboratory experiments of deuterium implantation in CFC samples at high fluence, showing that the retained fraction does not saturate in contrast with pyrolytic graphite [11]. These results may bring severe limitations on the number of plasma discharges that it will be possible to run in ITER before the maximum allowed tritium inventory is reached (350 g). In the present ITER design, the use of carbon is limited to the small region where the highest heat flux is expected, the rest of the vessel being made of beryllium and tungsten. Simultaneously, an intensive research activity on carbon is ongoing, aiming at: (1) better understanding what is the dominant process involved in long term retention, surface trapping in co-deposited layers, or bulk diffusion, and (2) assessing in situ detritiation techniques to remove tritium from the vessel walls.

4. ITER and future fusion reactors: new constraints

Tore Supra PFCs are capable of withstanding continuous thermal fluxes of several MW m^{-2}, which is the ITER requirement. The ITER plasma facing component design was derived from this concept. The armour tile is made either of a low Z material like CFC or beryllium, or a refractory material like tungsten for the most exposed sections of the PFC and is joined to the water cooled structure with technologies similar to those used for Tore Supra. These ITER technologies were successfully tested in specific installations under thermal fluxes reaching 20 MW m^{-2} over thousands of cycles but remain to be qualified at the industrial level. Moreover, ITER will also add a very challenging and new constraint: its nuclear environment due to the presence of both tritium and 14 MeV neutron irradiation. For PFCs, this means a concept able to take into account neutron damage (monoblock design instead of flat tile) which leads for CFC to a significant decrease of thermal conductivity properties [12]. Safety criteria, notably those involving authorized tritium inventories are a severe constraint for CFC PFCs. This concern will be manageable in ITER by applying time-consuming in situ detritiation methods but will probably be prohibitive for CFC in a fusion power plant environment, where economic criteria must be taken into account. The choice of PFC material for a fusion reactor could be made by elimination: C was not retained because of the tritium retention aspect and Be because of erosion problems.

As regards the reactor, plasma facing components and the divertor specifically remain an open issue. Tungsten is presently the only choice for armour material. The main difficulty consists in integrating complicated phenomena that control the plasma edge and technological constraints due to materials. ITER will provide some crucial answers in this area.

5. Conclusions

An important R&D programme led to achieve reliable high heat flux component in Tore Supra. They are able to sustain heat fluxes comparable with ITER heat fluxes. Even if a future fusion device adds other constraints such as neutron irradiation and thus different designs, lessons from industrial feedback appear important. In particular, detailed acceptability procedures (criteria, test facilities) and fallback issue (repair processes) will be of utmost importance during the manufacturing process. The development of repair processes appears highly valuable and must to be implemented as soon as possible.

As regards the operation of long plasma discharges, safety processes, in situ inspections and/or diagnostics will be key points for ITER. These aspects are already being addressed by Tore Supra.

If the heat exhaust seems well controlled in Tore Supra, particle control is an open issue. Tore Supra long discharge results show that half of the injected flux remains buried inside the vessel components. Tritium retention in CFC will probably be managed in ITER by applying time-consuming in situ detritiation methods but will probably be prohibitive in a fusion power plant environment where economic criteria must be taken into account.

References

[1] I. Bobin-Vastra, F. Escourbiac, M. Merola, P. Lorenzetto, Fus. Eng. Des. 75–79 (2005) 357.
[2] A. Durocher, N. Vignal, F. Escourbiac, J.L. Farjon, J. Schlosser, F. Cismondi, Fus. Eng. Des. 75–79 (2005) 401.
[3] P. Garin, Fus. Eng. Des. 49&50 (2000) 89.
[4] A. Grosman, Tore Supra Team, Fus. Eng. Des. 74 (2005) 49.
[5] A. Durocher, D. Ayrault, Ch. Chagnot, M. Lipa, W. Saikaly, in: Proceedings of the 12th International Conference on Fusion Reactor Materials, Santa Barbara, J. Nucl. Mater., submitted for publication.
[6] D. Guilhem, J.L. Bondil, B. Bertrand, C. Desgranges, M. Lipa, P. Messina, M. Missirlian, C. Portafaix, R. Reichle, H. Roche, A. Saille, Fus. Eng. Des. 74 (2005) 879.
[7] Y. Perrot, J.J. Cordier, J.P. Friconneau, L. Gargiulo, E. Martin, J.D. Palmer, A. Tesini, Fus. Eng. Des. 75–79 (2005) 537.
[8] Th. Loarer, E. Tsitrone, C. Brosset, J. Bucalossi, J. Gunn, E. Joffrin, A. Loarte, P. Lomas, P. Monier Garbet, J. Ongena, B. Pégourié, P. Thomas, in: Proceedings of the 30th EPS Conference on Controlled Fusion and Plasma Physics, St Petersburg, 2003.
[9] Y. Sakamoto, JT-60 Team, in: Proceedings of the 32nd EPS Conference on Controlled Fusion and Plasma Physics, Tarragona, 2005, Plasma Phys. Contr. Fus. 47 (2005) B337.
[10] M. Warrier, R. Schneider, E. Salonen, K. Nordlund, Contrib. Plasma Phys. 44 (2004) 307.
[11] J. Roth, V.Kh. Alimov, A.V. Golubeva, R.P. Doerner, J. Hanna, E. Tsitrone, Ch. Brosset, V. Rohde, A. Herrmann, M. Mayer, J. Nucl. Mater., doi:10.1016/j.jnucmat.2007.01.164.
[12] M. Roedig, W. Kuehnlein, J. Linke, D. Pitzer, M. Merola, E. Rigal, B. Schedler, E. Visca, J. Nucl. Mater. 329–333 (2004) 766.