Installation of a Plasmatron at the Belgian Nuclear Research Centre and its Use for Plasma-Wall Interaction Studies

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Abstract. In JET and ITER, the first wall will be covered by beryllium and a full or partial W divertor will be common. In DEMO, only high-Z, low erosion material such as tungsten will be present as a plasma facing material. In present day tokamaks, the very high fluence/low temperature plasma cannot be obtained. Important key issues to be resolved according to plasma wall interaction studies are the tritium retention, dust production, resilience to large steady-state fluences, transient loads, surface erosion, material redeposition and neutron damage. Some linear plasma simulators come close to the very high fluences expected in ITER and DEMO such as PSI -2; PISCES-B; NAGDIS-II and pilot-PSI. In future the larger device MAGNUM-PSI will have even higher fluences and lower temperatures for large scale components. The plasmatron facility VISION I to be installed in Mol, will have the capability to investigate mixed materials (with beryllium/tritium contaminations) and in the long term neutron activated samples. The ETHEL plasmatron VISION I from JRC-Ispra was transferred to SCK•CEN (Mol, Belgium) recently. The equipment is meant to study plasma-wall interaction, in particular the interaction with hydrogen isotopes. The facility is capable to produce relatively cold self-sustained volumetric plasmas with a high plasma flux density at the target of about 10^{20} – 10^{21} ions/m^2.s. The plasmatron has a volume of 18 litres, a target diameter of ~25 cm and modular ion energies in the range of 20-500 eV.

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Keywords: Fusion, Plasma Source, Plasma Wall Interaction
PACS: 52.40-w, 52.40.Hf, 52.50.Dg, 52.72.+v, 94.05.Rx

INTRODUCTION

The current material selection for ITER is beryllium (Be) for the first wall, carbon fibre composites (CFC) in the heavily thermal loaded regions of the divertor, and tungsten (W) or tungsten alloys for the upper vertical target of the divertor, the baffle and the dome. In the near future, JET will undertake 'ITER-like wall' experiments where a combination of these materials is also to be tested **[1]**.

Beryllium (Be) is a low-Z material and it does not lead to significant radiation cooling of the plasma, giving a high performance of the core plasma. It will be used because of its high oxygen gettering capability, good thermal conductivity, very low solubility for hydrogen, and implantation of Be with deuterium (D) or tritium (T) results in saturated layers in the very-near surface with limited inventory. But nuclear reactions will breed T and helium (He) in the material bulk. Therefore, a substantial T inventory in the bulk of Be will be formed after long-term neutron exposure **[2]**. The use of this material is limited mainly by its high erosion yield (physical sputtering due to low surface binding energy (3.31 eV)) and low melting point. The transient heat loads could be much higher than the critical heat load inducing melting of Be. A mechanism which enhances the erosion of Be (and all metals) at high temperatures has been identified. This mechanism is similar to the radiation-enhanced sublimation of carbon and leads to erosion rates which are one order of magnitude larger than physical sputtering (E_{ion}~50eV) for T_{surf} ~1200K **[3]**. These events will shorten the lifetime of the first wall. Enhanced erosion also leads to larger depositions in the divertor area and can give rise to alloy formation in the divertor area (BeW, BeC) **[4]**. These mixed materials will have very different material properties compared to pure materials and exhibits therefore different T retention and erosion damage.

Carbon (C) has good thermomechanical properties, lack of melting under transient power fluxes, low core radiation (low-Z number) as well as effective divertor radiation losses. The high erosion yield and T retention problem (mainly due to co-deposition) gives restrictions to its use and lifetime. Deposition between gaps of plasma facing components (PFCs) and/or sublimation attributed to large transient heat loads such as large ELMs, can increase the retention of T even more. This however is mainly a surface problem which saturates over longer exposure times **[5]**. Again the neutron damage to C results in substantial bulk T retention **[2]**.To minimize the T inventory in the C-T co-deposition layers, several T removal techniques (heating in air or oxygen, laser heating, flash lamps, He-O glow discharges) to minimize in-vessel inventories as well as those used to reduce contamination prior to waste disposal have been investigated but are not completely optimized or validated **[6]**.

Tungsten (W) as a high-Z material is the only one that can fulfil the requirements needed for DEMO. The expected large steady-state particle fluxes and fluences to plasma facing materials in DEMO require that the materials have a very low erosion rate under the expected characteristics of these fluxes in DEMO (~few eV at the divertor). W has the highest surface binding energy (8.68 eV) of all elements and thus the lowest erosion rate by light elements (D, T, He). Due to the large sputtering threshold (~209 eV for D on W), erosion by the low temperature hydrogenic plasma (Te < 20 eV), particularly for a detached divertor plasma, will be negligible **[3]**. No substantial chemical erosion by volatile W-oxides has been measured either, which is in line with recent molecular dynamic calculations where chemical erosion of W by O was found to be negligible in the relevant parameter range **[7]**. The high-Z number of W results in a high re-deposition in the divertor area, and the high radiative cooling rate compared to C, especially for electron temperatures in the keV range in the core, limits the concentration to several ppm. The contamination of the core plasma ascribed to W erosion should be limited or an increase of the high-Z impurity control should be made by seeding of noble gases such as Ar and Ne. However, the introduction of these seeded impurities into the plasma edge could result in an increase of the W erosion due to their higher mass and potentially higher impact energy compared to D **[8]**. Recent investigations showed low tritium inventories in pure tungsten materials. Disruptions and ELM heat cycles could lead to roughening and crack formation (that penetrate deep into the bulk along grain boundaries) which causes enhanced melting or evaporation due to irregular roughened surfaces. Studies of T retention in W have shown that most of the retention is due to trapping in the bulk material [9] and is, therefore, deeply influenced by the creation of damage sites by neutron irradiation and by the "effective porosity" of the material, which is affected by the manufacturing technique and by the existence of the cracks caused by thermal cycling of the material under transient loads. Similarly, the influence of He created in the material by transmutation and/or plasma implantation may affect the retention of T in W.

In the **detached regime**, the plasma is strongly coupled to the surface. These divertor conditions are caused by very high dense plasma with a low degree of ionization with plasma temperatures around a few eV. The mean free path for ions, molecules and dust particles eroded from the surfaces is small with respect to the plasma size. Most species will undergo many collisions in the divertor area before eventually disappearing in the vacuum pump, making the plasma physics of this multi component system very complex. These synergistic effects, where two components of the plasma that are individually hardly capable to etch the surface, could together realize significant damage. These effects on H^+ and CH_3^+ have been found for weakly coupled plasma [10], therefore it is expected that the effects will be even stronger for strongly coupled plasmas. [11]

In addition the amount of activated **dust** should be limited for safety reasons. Another issue of dust is the potential reaction with leaking water and the possible production of hydrogen (H) (co-deposits contain large amount of H). The H, or dust itself, could be a risk for explosion with air in case of leakage into the vacuum vessel. It is important to understand the characteristics of dust, to measure its quantity and to remove it. Furthermore, degradation of **in-vessel diagnostic components**, such as mirrors, caused by dust, material deposition and erosion should be predicted and possible methods should be found to avoid or limit them [5].

R&D EXPERIMENTAL CAPABILITIES AT SCK•CEN

The lack or uncertainty of experimental data on the synergistic effects of plasma steady-state flux, material damage by neutrons, tritium retention in pure plasma facing materials (beryllium, carbon, tungsten) and the mixed materials implications are closely related to the unavailability of experimental nuclear devices that are able to adequately reproduce the boundary conditions between the plasma and the first wall.
Studies on hydrogen isotopes interaction with first wall materials have been carried out mainly with low flux, high energy ions, since these features were commonly obtained using standard particle accelerators or ion beam devices. However, it is apparent that results obtained with such devices cannot be easily rescaled to the real operating conditions of a fusion reactor, as both characteristics strongly influence the mechanisms itself (retention, implantation, recycling, …) [12].

Therefore several plasma simulators were build that can reach the very high heat flux and low electron temperatures. Their relevant parameters are given in table 1. PISCES-B, PSI-2 and the plasmatron VISION I cannot compete with the pilot-PSI facility and certainly not with the future Magnum-PSI, but the additional advantage of the PISCES-B and plasmatron facility VISION I is definitely lying in the capability to work with beryllium seeding and beryllium contaminated materials. On the other hand, none of the existing linear plasma generators are capable of operating in a nuclear environment and with tritiated gases. This restricts the research since no neutron degraded materials and/or existing depositions from large tokamaks can be studied since these are contaminated with the radioactive tritium element. Therefore, the plasmatron facility VISION-I would have a unique feature that is still missing today in the existing linear plasma devices that is essential for addressing/solving several key issues for ITER and DEMO.

TABLE 1. Comparison of various plasma parameters from high fluence, low temperature plasma simulators versus ITER. **[6, 13, 14, 15]**

	PISCES-B	**Magnum-PSI**	**Pilot-PSI**	**NAGDIS-II**	**PSI-2**	**Plasmatron VISION I**	**ITER**
n_e (m^{-3})	10^{17}-10^{19}	10^{19} - 10^{21}	10^{19}-10^{21}	<10^{20}	10^{17}-10^{20}	NO DATA	**10^{19}-10^{22}**
T_e (eV)	~4-40	~0-10	~0-7	~5-10	~1-20	NO DATA	**Div: ~3 MidP:~100**
T_{ion} (eV)	0.1-0.5 T_e	~T_e	~T_e	1-10	0.5-0.6 T_e	20-500	**Div: ~15 MidP: ~500**
σ (m^{-2} s^{-1})	10^{19}-10^{23}	10^{23}-10^{25}	<2.10^{25}	<10^{23}	10^{22}-10^{23}	10^{20}-10^{21}	**10^{24}-10^{25}**
τ (s)	Steady state	Steady state	3-10s	Steady state	Steady state	Steady state	**300-500s - steady state**
Preheat Target	Plasma heating	Plasma heating	Plasma heating	Room temp.	Room temp.	20-600 °C	**Bake temp. < 230 °C**
P_n (Pa)	5 10^{-4}-10^{-2}	<1	~1-10	~0.1-4	0.01	0.05 – 0.5	**1-10**
B (T)	0.015-0.05	<3	0.4 - 1.6	<0.25	0.1	0.2	**5.3**
Target material	C, W, Be, metals, mixed	C, W, metals	C, W, metals	C, W, metals	C, W, metals	C, W, Be, metals, mixed	**C, W, Be, mixed**
Be	Yes	No	No	No	No	Yes	**Yes**
T	No	No	No	No	No	Yes	**Yes**
Nuclear	No	No	No	No	No	Yes	**Yes**

n_e = electron density/ T_e = electron temperature/ T_{ion} = ion temperature/ σ = Ion flux density/ τ = pulse length/ P_n = neutral gas pressure/ B = magnetic field/ Div = detached divertor area / MidP = midplane

The advantages of the linear plasma device VISION-I compared to the existing tokamaks are the easy access to the investigated materials but more important the possible use of tritium, toxic and radioactive materials. Various key questions regarding to the behaviour of plasma facing materials for ITER/DEMO; namely the retention of tritium under neutron irradiation and ITER/DEMO-like plasma particle fluxes; cannot be answered by experiments in the existing tokamaks.

The complete ETHEL **[16]** facility has been shut down ten years ago. During the decommissioning of the ETHEL buildings, SCK•CEN made a contract agreement with JRC ISPRA to transport the device and recover it for fusion applications. Unfortunately, several parts of the facility were appointed for other projects over the long time. Anyway, many of the original drawings and documents could be recovered from the ETHEL archive and the equipment will be reinstalled after a thorough refurbishment. The plasmatron is renamed to VISION I (Versatile Instrument for the Study of ION Interaction – I) and a picture of the present status in the gloveboxes can be seen in figure 1. In addition, the Belgian Nuclear Research Centre has a large spectrum of in-house facilities including a specific tritium and beryllium laboratory. Large hot-cell expertise and infrastructure covering mechanical, physico-chemical analysis systems, irradiated corrosion test systems, specimen preparation workshop as well as microstructure analysis support are present. It is clear that given the difficulties inherent to the transport of radioactive materials and/or tritium containing materials, it is advantageous to have the device and in-house characterization tools at the same location. In one single device, the plasmatron VISION I, makes it possible to trap tritium and explore plasma based removal schemes.

FIGURE 1. Pictures of the present status of the plasmatron facility VISION I in gloveboxes. Left: complete facility, Right: Plasma chamber.

EXPERIMENTAL PARAMETERS OF THE PLASMATRON VISION I

The plasmatron VISION I is an experimental device to investigate issues related to PWI and material properties under high heat and particle flux and low plasma temperatures. Figure 2 shows a schematic view of the plasma chamber. The facility is a vertically mounted cylindrical vessel of 29 cm (outer diameter), 25 cm (inner diameter) and of about 40 cm in height. The cylinder is divided into two parts at a height of about 10 cm from the bottom by an Al_2O_3 insulating ring. The bottom part consists of two major elements; the anode (A) that acts as plasma simulator and the target (T) that constitutes of a replaceable material to be investigated. The top part of the target (T) houses the electrical connections and the conditioning system feedthroughs. The steady state plasma is generated between two heated 1 mm W-cathodes (C) of V shape to enhance electron emission and the anode (A). **[12]**
Ionization takes place when a discharge potential between the cathodes and the anode is switched on where the probability of ionization is augmented by the magnetic field. The design values are 2 kW of heating power and 80 V, 5-100 A/m^2 for the voltage and ion current density ranges, respectively. Samarium-Cobalt (Sm_2Co_{17}) permanent magnets (PM) are placed on the anode external surface and on the sides of the

chamber. The combination of the PMs produces on the one hand the multipolar field of ~0.2T at the anode and negligible at the specimen and secondly increases the probability of ionization of the plasma (helical movement of particles with an increase of the mean free path of electrons). The anode is cooled by water in order to extract the heat produced by the electron load, able of inducing anomalous outgassing signals. Usual working gases generating the plasma could be hydrogen, deuterium and all kinds of noble gases or mixtures of them as well as tritium. The device is an Ultra High Vacuum (UHV) vessel equipped with oil-free pumps especially done for the use of tritium gases. There are two modes of operation, depending on the tritium levels. In the low-level mode, gas is removed by a magnetic bearings turbo-molecular pump and a membrane pump in series. Once the chamber has reached UHV conditions (i.e. ~10^{-6} Pa), the system can be switched to the high-level mode, in which tritium storage and removal are accomplished by two regenerable bulk getter pumps. These pumps are also effective in adsorbing impurities (N_2, O_2, H_2O, CO, …) irreversibly, while hydrogen isotopes are pumped reversibly. [13]

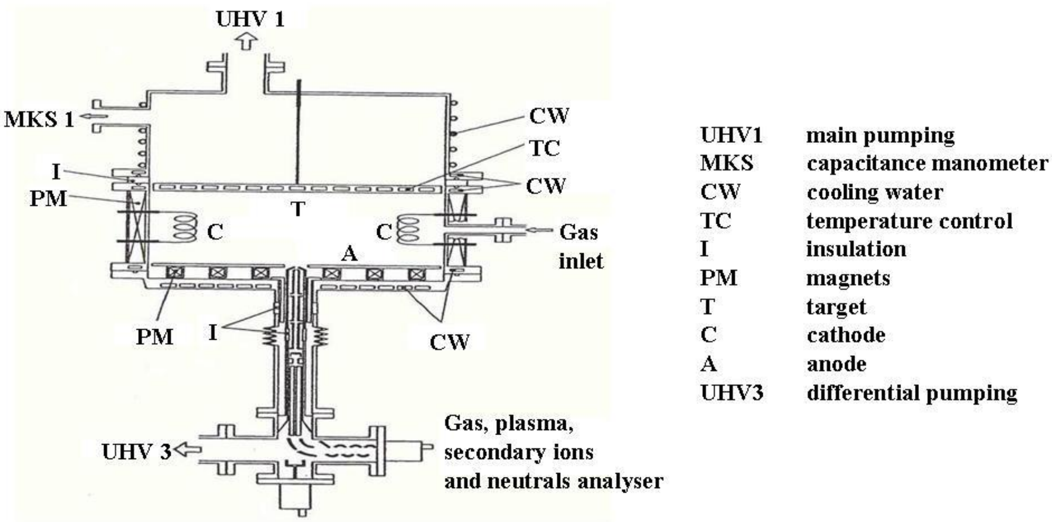

FIGURE 2. Schematic view of ion source chamber of the plasmatron facility VISION I.

CONCLUSION

Plasma facing materials for ITER and DEMO will need to meet many requirements already considered in the R&D programme (fabrication feasibility, resilience to neutron damage & activation, etc.). However, other aspects concerning the behaviour of the materials under exposure to plasma in ITER- and DEMO-like conditions are likely to be as important as those above in determining their performance/use (namely : T-retention, dust production, resilience to large steady-fluences & transient loads, surface erosion and material redeposition) and should be addressed [17]. The key plasma facing material issues are determined by synergistic effects of plasma steady-state flux, transient fluxes and material damage by neutrons. Therefore, it is necessary to have facilities and analysis laboratories that are able to investigate these synergistic effects. The plasmatron VISION I that will be operated at SCK•CEN will have the ability to work on some of the key issues in plasma wall interaction since it can work with tritium, beryllium and in the future on neutron irradiated materials, under high flux densities and low plasma temperatures.

Acknowledgments

The authors greatly appreciate the scholarship awarded to I. Uytdenhouwen and financial support from SCK•CEN (Belgium). They also appreciate the extensive help of the JRC-ISPRA (Italy) people G. Vassallo, R. Garbil and A. Perujo. Thanks to R. Doerner, N. Lopes-Cardozo and N. Ohno for the requested information on the various plasma simulators during personal communications. The many fruitful discussions with P. Coad, G. Counsell, A. Loarte and V. Philipps are greatfully acknowledged.

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