GENERAL ATOMICS

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Reference: Operation of DIII-D National Fusion Facility and Related Research
Cooperative Agreement DE-FC02-04ER54698
(GA Project 30200)

Dear Dr. Foster:

Enclosed for your use and information is one copy of the following document to be presented at the *23rd IAEA Fusion Energy Conference* to be submitted in accordance with the referenced Cooperative Agreement:

GA REPORT	TITLE
GA-A26653	**THE DIII-D NATIONAL FUSION PROGRAM** **"Plasma Facing Material Selection: A Critical Issue for Magnetic Fusion Power Development"**

If you have any questions or comments, please do not hesitate to contact me by telephone at 858.455.3057, at FAX 858.455.3545 or by E-mail at Ramona.Gompper@gat.com.

Sincerely,

R Gompper

Ramona Gompper
Sr. Contract Administrator

Enclosure: As Stated

Copy: Per Cooperative Agreement

GA–A26653

PLASMA FACING MATERIAL SELECTION: A CRITICAL ISSUE FOR MAGNETIC FUSION POWER DEVELOPMENT

by
C.P.C. WONG, B. CHEN, D.L. RUDAKOV, A. HASSANEIN,
T.G. ROGNLIEN, R. KURTZ, T.E. EVANS and A.W. LEONARD

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by

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This is a preprint of a paper to be presented at the 23rd IAEA Fusion Energy Conference, October 11–16, 2010 in Daejon, Republic of Korea and to be published in Proceedings.

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This paper proposes a possible development approach that may satisfy the requirements for plasma facing materials for advanced DT devices beyond ITER. Five fundamental requirements will have to be satisfied: heat flux removal, acceptable erosion lifetime, acceptable contamination to the plasma core, radiation resistance and the ability to withstand occasional transient events like disruptions without a major perturbation in reactor operation. To achieve these requirements consideration will have to be given to material selection and design and to the physics of plasma edge control and operation.

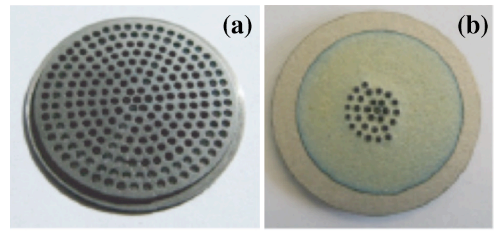
In considering the choice for plasma facing materials, ITER has selected Be as the first wall and C and W as the divertor region surface materials. When extrapolated to an advanced DT device, C and Be tiles will not be suitable due to radiation damage. The remaining material, W, could also suffer neutron radiation damage and damage from helium ion implantation (which causes the generation of tungsten "fuzz" at the divertor) and surface morphology changes at the chamber wall (i.e. the generation of blisters at the chamber wall). Both the tungsten "fuzz" and the tungsten blisters could result in contamination of the plasma core.

Li has been proposed as a possible surface material option. However, surface vaporization causes Li to be transported to the core where it radiates power away from the plasma. To limit this power degradation to 20% of the plasma power, modeling results show that the maximum allowable Li surface temperature is <400°C. Furthermore, in order to maintain an acceptable increase in the ductile-brittle transition temperature (DBTT) of the ferritic steel structural material under neutron radiation, its temperature needs to be >350°C. To work within these temperature limits, heat transfer calculations have been performed to show that the corresponding allowable steady state surface heat flux removal capability for Li will be <0.2 MW/m^2, implying a neutron wall loading of <1 MW/m^2, which is too low for an economical fusion power reactor.

For the development of advanced solid surface materials, it is difficult to foresee any low activation metallic alloy that can outperform W-alloy in high temperature performance, low physical sputtering rates and high thermal conductivity. Ceramic surface materials such as SiC and B_4C could have high temperature capability, but they tend to be brittle, possess low thermal conductivity at high temperature and high neutron fluence, and they have relatively high physical sputtering rates and high generation of helium under high neutron fluence, making them unacceptable for this application.

An alternate approach for plasma facing material development would be the revival of real-time boronization or siliconization, which has been tested successfully in different tokamaks. This requires that the eroded material be replenished during steady state plasma discharges and that the excess B or Si be pumped out of the vessel. With a B or Si layer equivalent thickness of about 10 microns, the surface could withstand an occasional disruption without damaging or melting the underlying metallic substrate, thereby maintaining steady state operation without a major perturbation in reactor operation. This observation is based on modeling results which show that for an ITER-like disruption, the thermal energy will be absorbed by vaporizing B or Si and by radiating power away from the bulk material via the vapor shielding effect.

An innovative approach for maintaining adequate material for vaporization to handle a disruption is the deposition of Si on a W-surface as shown in Fig. 1. The 50% void volume on the W-disc can be filled with Si, such that the surface can withstand occasional ELMs and disruptions while retaining the capability of high heat transmission for power conversion. Results on the development of this Si-infiltrated W surface approach will be reported. An example of the first wall design would be a three-layer design with a coating of W on top of a layer of oxide-dispersion strengthened ferritic steel, which is on top of the ferritic steel substrate structure. This three-layer design could have the capability of removing heat flux in the range of 0.5 to 1 MW/m^2 while satisfying the temperature limits of the respective materials. On top of the W-coating, the low-Z layer could be maintained with the necessary temperature range for the W-coating operation of 700° to 1300°C. These temperatures will also help in releasing tritium from the B or Si. At the same time, the coating could protect against charged particle damage to the metallic substrate and mitigate the formation of W-fuzz. This high temperature operation will also be necessary to maintain a high chamber wall helium coolant temperature to satisfy the requirement of high thermal efficiency for the power reactor. It should be noted that in order to match the coefficient of thermal expansion between the low-Z coating and W substrate, a Si coating may be more suitable than a B-coating. Furthermore, if B is used as the coating material, it would need to be enriched in B-11 to reduce the generation of helium from the (n, ^{10}B) reaction.



Drilled W-disc
2 cm diam disc
1 mm diam, holes

DIII-D boronized
coating on the
middle of W disc
(DIII-D 2/7/2009)

Fig. 1. (a) Drilled 2 cm diameter W-disc with 1 mm diameter holes filled with Si. (b) DIII-D boronized coating of ~0.75 µm thick on the middle of a W-disc with drilled holes

From physics and heat transfer considerations, in order to handle localized transient events like edge localized modes (ELMs), the ITER plan requires that different locations of the plasma chamber wall be designed for surface heat fluxes of 1 MW/m^2 and 5 MW/m^2, in comparison to the average surface heat flux of 0.3 MW/m^2. Fortunately Type-I ELM control techniques, such as ELM suppression, have recently been demonstrated, and plasma-surface interaction modeling is being carried out at DIII-D and JET. However, more will have to be learned about the control of surface material damage from high-energy neutral particles and fast ions and electrons. As noted, the proposed thin low-Z coating on chamber wall surfaces could perform the function of protecting metallic substrates from fast ion implantation. This protection has been demonstrated in laboratory experiments, but it will have to be experimentally demonstrated in tokamak experiments. Furthermore, real time low-Z material coating on the chamber surface for steady state operation needs to be demonstrated. The corresponding impact of low-Z gas consumption and extraction from the plasma chamber and the extraction and purification of tritium fuel under the scenario of real time gas injection need to be understood and accommodated.

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