GA–A27394

FUSION TECHNOLOGY FACILITY – KEY ATTRIBUTES AND INTERFACES TO TECHNOLOGY AND MATERIALS

by
C.P.C. WONG

OCTOBER 2012

GENERAL ATOMICS

DISCLAIMER

GA–A27394

FUSION TECHNOLOGY FACILITY – KEY ATTRIBUTES AND INTERFACES TO TECHNOLOGY AND MATERIALS

by
C.P.C. WONG*

This is a preprint of a paper to be presented at the 24th IAEA Fusion Energy Conference, October 8–12, 2012, in San Diego, California.

*General Atomics, P.O. Box 85608, San Diego, California USA.

Work supported by
the U.S. Department of Energy
under DE-FG02-04ER54698

GENERAL ATOMICS PROJECT 30200
OCTOBER 2012

GENERAL ATOMICS

Fusion Technology Facility – Key Attributes and Interfaces to Technology and Materials

C.P.C. Wong
[1]General Atomics, P.O. Box 85608, San Diego, California 92186-5608, USA

Abstract. On the way to a Demonstration Fusion Power Plant (DEMO), a number of fusion physics and technology issues will need to be resolved including the long burn or steady state DT operation, net tritium breeding ratio of >1 and the application of the Fusion Technology Facility (FTF) as a material and component testing vehicle. This paper focuses on five interface areas between physics and technology that will have significant impacts on the design of FTF. 1. It is clear that tritium supply, inventory, breeding, recovery, safety and leakage are critical for the development of a DT magnetic fusion program, which are impacted by the key parameter of plasma burn-up fraction. 2. Due mainly to parallel heat flux, there is a disconnect between the ITER-specified chamber heat flux of 2-5 MW/m^2 and the ARIES-type power reactor having a specified chamber surface heat flux of ~0.5 MW/m^2. There is another disconnect on the heat flux at the divertor. 3. Transient type-I ELMs, disruptions, and runaway electrons will damage chamber wall and divertor surface materials. Plasma control and transient event avoidance and mitigation are the present approaches. 4. Tungsten (W) is the only surface material being considered for DT machines, but there are problems with the formation of W-fuzz and blisters under high temperature and fluence. The transport of W into the plasma can also limit the performance of the tokamak. 5. There is inadequate data for material damage from 14 MeV neutrons and at high enough fluence for plasma facing components (PFC) and in-vessel components. Plasma facing material will have major impacts on the selection of structural material and corresponding functional materials, such as joining and welding materials. The potential solution approaches to these coupled physics and technology issues will have to be understood and fully assessed before an acceptable solution can be found for FTF.

1. Introduction

Based on the design of ITER and recent U.S. program review reports [1-4], we can identify a number of high priority research areas that address material and material-plasma interface challenges. A number of fusion technology issues will need to be demonstrated including long burn or steady state DT operation, net tritium breeding ratio of >1 and the application of the Fusion Technology Facility (FTF) as a material and component testing vehicle. The FTF in this paper does not stand for the recommendation of a specific machine, but rather a consideration of a generic device with specific attributes that will need to be considered while the fusion community is moving towards the design of DEMO. The focus of this paper will be on the necessary assessment and development of five selected physics and technology interface areas that will have significant impacts on the design of the FTF: 1) impact of tritium burnup fraction to the rest of the tritium system, 2) acceptable chamber wall and divertor peak heat flux, 3) transient events like high-power ELMs and disruption, 4) robust chamber wall surface material, and 5) low activation DEMO-relevant structural material. Close collaboration between the physics and technology communities is needed to understand these interface issues and to resolve them before an acceptable design can be found for FTF.

2. Tritium

It is clear that tritium supply, inventory, breeding, recovery, safety and leakage are critical for the development of a DT device like FTF and DEMO. Due to the lack of external tritium sources, all fusion DT-testing facilities and power plants must breed their own tritium (T) needed for plasma fueling. The net tritium breeding ratio (TBR) during plant operation should be around 1.01 [5]. Even though it seems small, the 1% margin translates into 1-2 kg of excess T generated per year for 2-3 GW fusion power. Such a low net TBR is potentially achievable with advanced physics and technology where the fractional burn-up of T in the plasma exceeds 10%. At the same time any power plant should provide the required start-up inventory for a new power plant to be built every few years. To quantify the start-up T inventory for future plants, the ARIES-CS compact stellarator [6] and ARIES-AT advanced tokamak [7] were used as examples in reference 5. Both designs employed PbLi as the breeder/coolant. For 12% and 36% fractional burn-up of T in ARIES-CS and ARIES-AT plasmas, the T start-up inventories are ~4 kg and ~2 kg, respectively [5]. It is also found that the T startup inventory is very sensitive to the T burnup fraction. For burnup fraction of 2% and 1%, the T-start-up inventory would increase to 12 kg and above 15 kg, respectively [5], which is close to the 27 kg supply of tritium available from CANDU reactors.

An analytical expression was derived for the tritium burnup fraction [8] using parabolic profiles for density and temperature and a global expression for particle balance assuming steady state operation. It was found that f_{burnup} is directly proportional to the fueling efficiency [η_{eff}] and hence an efficient means of fueling, such as high field side (HFS) pellet injection, is important to assure the highest η_{eff}. The biggest uncertainty in the formulation is determining the global recycling coefficient R. R itself can be a function of many variables such as the vessel pumping speed, neutral pressure in the private divertor flux region, impurity seeding, characteristics of the scrape-off layer (SOL) and the choice of plasma facing material (PFM). Recycling will be much lower than in present day tokamaks due to the hotter edge, and this can have important implications in the T fueling design [9]. Recent modeling suggests that the tritium burnup fraction in ITER can be as low as 0.82% and even lower if pellet pacing for ELM control is used [10]. When compared to the above, there are potential disconnects between the desirable f_{burnup} for a power reactor to that is achievable from the analytical results. It can be projected that ITER operation will help to provide experimental benchmarks on the modeling for R for different DT machines. The question becomes the timing of such information for the FTF program.

Therefore, a more complete integrated system study of the effects of plasma fueling, tritium burn-up, wall-recycling, fuel pumping, fuel cycle and plasma operation and choice of PFC materials on the tritium inventory in the selected design option and the complete tritium cycle is needed. This is necessary to generate a complete picture and therefore to identify the necessary R&D to achieve the goal of adequate tritium breeding with low T inventory and corresponding acceptable tritium start up inventory for FTF and DEMO.

3. Heat Flux Distribution

For the heat flux distribution, specifically on the chamber wall and the divertor of a tokamak, there are disconnects between the design parameters for ITER vs. those of conceptual power reactors.

On the chamber wall heat flux, due mainly to impulsive parallel heat flux from high power ELMs, there is a disconnect between the ITER-specified chamber heat flux of 2-5 MW/m^2 [11], and the ARIES-type power reactor having a specified chamber surface heat flux of ~0.5 MW/m^2 [7]. The higher values of heat flux for ITER are acceptable if irradiation damage and the corresponding minimum operating temperature of the structural material does not need to be taken into consideration. To assess the heat removal capability of the conventional chamber first wall design for power reactors, 1-D heat transfer estimates were performed for the helium-cooled chamber wall design with the use of thin layers of W-coating/alloy, reduced activation ferritic martensitic (RAFM) steel, and the oxide dispersion ferritic steel (ODFS). Parameters for the assessment are given in Table 1. The channel and chamber wall layer geometry and material design temperature limitations for the case of using RAFM as the structural material, are shown in Fig. 1. To enhance the heat removal capability of the helium coolant, 1-sided roughened heat removal is assumed for the channel. The high coolant velocity of 100 m/s was also assumed. Results for the maximum material temperatures as a function of surface heat flux for the RAFM and ODFS structural materials are given in Figs. 2 and 3, respectively. For RAFM steel as the channel structural material, due to the minimum temperature limit of >350°C and the maximum temperature limit of <550°C, the maximum heat flux that the design can handle is <1 MW/m^2. Similarly for the use of not-as-well developed ODFS as structural material, due to the minimum temperature limit of >350°C and the higher maximum temperature limit of <700°C, the maximum heat flux that the design can handle is <1.5 MW/m^2, which is still much lower than the ITER design heat flux of 2-5 MW/m^2. This implies that for the FTF, high power ELMs that could lead to high parallel heat flux will have to be

Table 1 Chamber wall heat removal design parameters.

Channel geometry:	
W layer thickness, mm	2
Channel length, m	1
Channel width, cm	2
Channel height, cm	2
Channel wall thickness, mm	4
Thermal inputs:	
Neutron wall loading, MW/m^2	3
Kth for RAFM and ODFS, W/m.K	20
Kth for W	50
He Tin, °C	350
He pressure, MPa	8
He velocity, m/s	100
Heat transfer enhancement factor	2

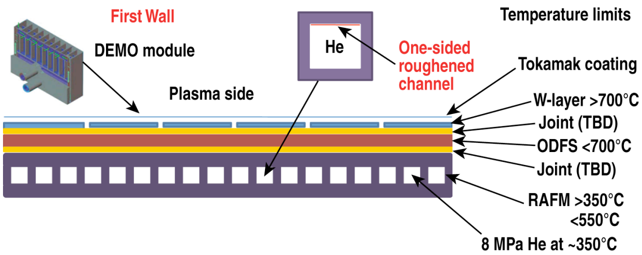

Fig. 1. First wall design model when RAFM steel is used as the structural materials.

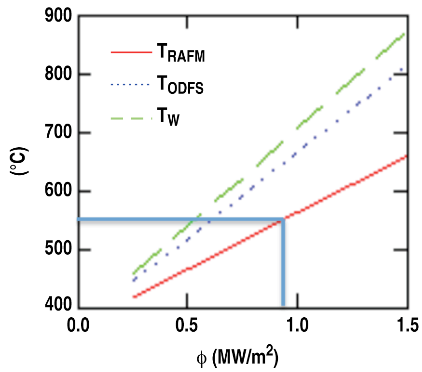

Fig. 2. Chamber wall maximum temperatures as a function of heat flux, with W, ODFS and RAFM steel thickness of 2, 2 and 4 mm, respectively. (The 4 mm is assumed for the wall thickness of the coolant channels.)

avoided for the consideration of the chamber wall design.

For the consideration of the divertor peak heat flux, both water and helium-cooled divertor designs are projected to be able to handle a maximum heat flux of 10 MW/m^2. When extended to the FTF and DEMO with their higher plasma power density, in order to limit the maximum heat flux to ≤10 MW/m^2, both a radiative mantle and radiative divertor should be used, but a stable operation point has not been established. Furthermore, innovative divertor configurations such as the snowflake [12] or super-X [13] divertor concept will most likely be needed. These divertor options will have major impacts on the reactor geometry details.

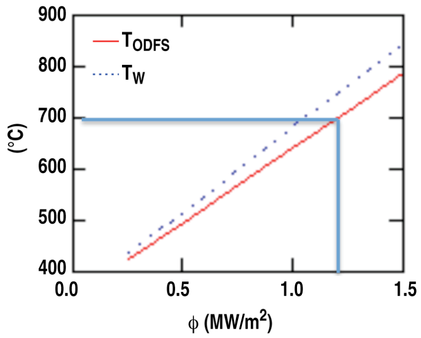

Fig. 3. Chamber wall maximum temperatures as a function of heat flux, with W, ODFS thickness of 2 and 4 mm, respectively (the 4 mm is assumed for the wall thickness of the coolant channels).

For a robust divertor design, based on results from edge localized mode (ELM) and disruption simulation experiments, both high power ELMs and disruptions will have to be avoided in the FTF class of machines; otherwise the surface material will suffer significant damage including the melting of the metallic surface material [14,15].

4. Plasma Transient Events

Transient type-I ELMs, disruptions, and runaway electrons from a tokamak like ITER will damage chamber wall and divertor surface materials as demonstrated in simulation experiments [14,15]**.** Plasma control and transient event avoidance and mitigation are the present approaches [16]. A more comprehensive integrated assessment of FTF operation, required machine availability, and the corresponding maintenance approach will have to be performed as a complete scenario in order to make credible FTF design recommendations. This assessment should include the trade offs on the choice of normal versus superconducting coils, the use of de-mountable superconducting coils and steady state operation vs. daylong pulse operation scenarios. Results of this assessment will have direct impacts on the economic performance of tokamak fusion reactors.

5. Plasma Facing Material (PFM)

Tungsten (W) is the only surface material being considered for DT machines, but there are problems with the formation of W-fuzz and blisters under high temperature and fluence [17]. The transport of W into the plasma can also limit the performance of the tokamak. An effort was made to deposit a significant amount of Si onto the chamber wall, such that the combined Si/W surface materials could become disruption tolerant through the vapor shielding effect of the lower vaporization point of Si to protect the W-surface. However, recent vertical displacement event (VDE) exposures of Si-W samples in DIII-D indicated the formation of the low melting point eutectic tungsten silicide, which forms when the surface temperature reaches 1400°C as shown in Fig 4 [18]. The DiMES module with Si-filled W-buttons is shown.

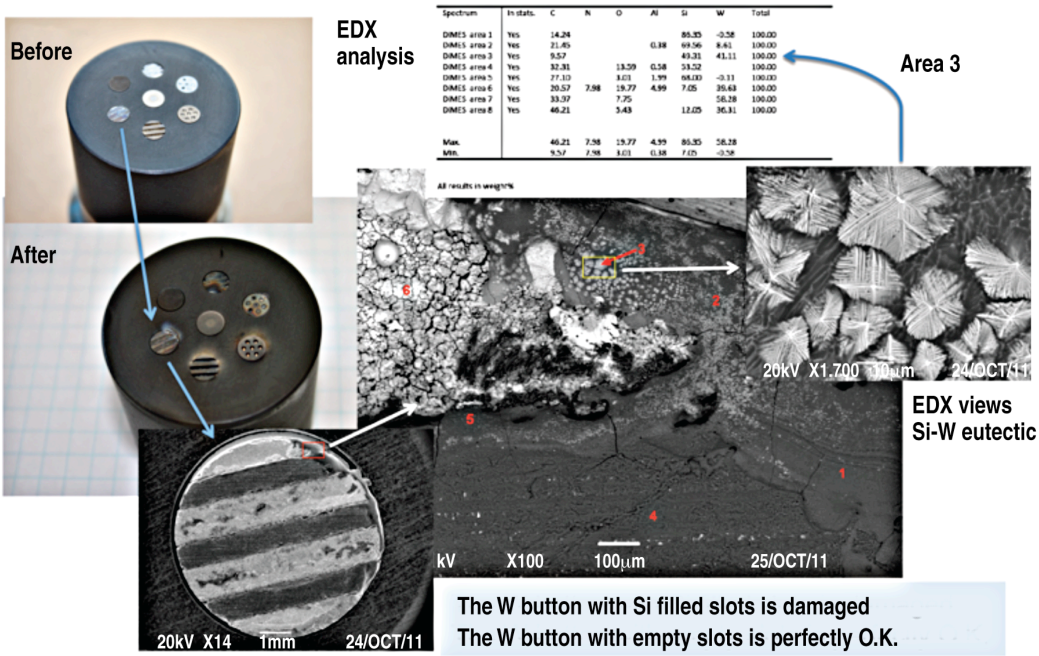

Fig. 4. Exposure of Si-W buttons to several VDEs in DIII-D indicated Si-W reactions and the formation of low melting point Si-W eutectic. This implies that the use of Si and W at >1400°C operation should be avoided.

The module was exposed successfully to six well-controlled VDE exposures. Optical and SEM images of the DiMES module and buttons before (4a) and after (4b) the exposure are shown in Fig. 4. The most informative detail is shown at the right top corner of the Si-filled slotted button in Fig. 4(c). We can see the once melted Si at the edge of the button shown in Fig. 4(d). Upon closer examination, we can see the crystalline structure from the Energy Dispersion X-ray (EDX) examination area 3 view on top of the solidified but once melted Si as shown in the Fig. 4(e) on the right. When compared to the composition of materials at different locations of the melted region as shown with the imbedded EDX analysis table, one can identify the material composition of area 3, which has a composition of 49.31 wt% of Si, 41.11 wt% of W and 9.57% of C. For our sample, the C is from the graphite sample and background impurities in DIII-D. When compared to the Si-W phase diagram [18], the composition is very close to the Si-W mixture of 45.7 wt% of W, which melts at 1850°C. This composition of Si and W is a strong indication that the indicated location had gone up to the temperature of ≥1850°C [19]. With this identification, the explanation for the previous difficulties during the loading of B or Si onto W became clear. The earlier samples were destroyed due to the formation of respective B-W and Si-W low melting point compounds at local temperatures. For B-W, different low melting point eutectic compositions can begin to form at >1970°C, and for Si-W eutectic it can be formed at >1414°C, and the kinetics of such formation is very fast.

This further emphasizes the need to maintain the wall temperature below 1400°C if when Si is used as the wall conditioning material, in order to minimize the transport of W to the plasma core. This approach could also mitigate the formation of W-fuzz. The introduction of Si would be via real-time injection during the plasma discharge. This real time low-z conditioning approach can only be successfully developed by close collaboration between experts on plasma operation and SOL physics, experts on material transport, and experts on surface material behavior. The low-Z materials that can be considered are Si, B and C. When operating with real time low-Z material injection to limit the eroded W material transport, consideration also has to be given to external systems designs, especially when tritium extraction, inventory, control, fueling and safety are involved. Techniques for dust and co-deposited layer minimization or removal inside of the vacuum vessel are needed even for FTF.

6. Materials Radiation Damage

The fifth critical interface area between physics and technology is materials radiation damage under 14 MeV fusion spectrums. Recent boron-doped RAFM results to simulate the impacts from transmuted helium indicate the possible increase of the minimum operating temperature of RAFM steel to higher than 350°C [20]. This could significantly narrow the operating temperature window of the RAFM steel at higher neutron fluence, leading to the need for development of new fusion structural material. However, it should be pointed out that B doping need not be the best simulation of the damage effect from He bubbles, because the solubility of boron is low in most metals so it tends to segregate to grain boundary and the results could become misleading. On the other hand, there are other helium implant methods like the addition of nickel to the alloy or the implanter foil concept. The nickel foil implanter approach can provide information on He generation that is volumetric. The fusion materials community has been trying to determine how He changes the microstructural evolution path compared to neutron damage by displacement per atom (dpa) only. Results from these implanted helium atoms indicated damage from high concentrations of helium. This led to the interest in nano-ferritic-alloy (NFA), where the transmuted helium is held around the nano additives without forming the damaging He bubbles. ODS and NFA [21] alloys could be more tolerant to high helium concentration and dpa damage. Similarly, it is clear that fusion-relevant advanced materials are at an early stage of development. This then points to the urgent need for a fusion neutron irradiation testing facility for fusion material development, and the need for the FTF, with true fusion spectrums, to be operated in a staged approach such that advanced fusion materials and corresponding components can be tested in the bootstrap approach for the development of DEMO.

In general, there is inadequate data for material damage from 14 MeV neutrons and at high enough fluence for PFC and in-vessel components. Plasma operation and the understanding of SOL and chamber component design, including the choice of PFM, will have an impact on the detailed design of FTF and DEMO. PFM and PFC themselves will have major impacts on the selection of structural material and corresponding functional materials, such as joining and welding materials. It becomes essential for the physics and material communities to recommend neutron source devices that can provide the correct high dpa, gaseous (helium, hydrogen) generation and metallic transmutations and appropriate helium/dpa ratios in order to project radiation damage caused by DT fusion neutrons. Bootstrap development of the suitable structural material will also have major impacts on the necessary fusion power core change-out approaches for FTF.

7. Conclusion

On the way to a Demonstration Fusion Power Plant (DEMO), a number of fusion physics and technology issues will need to be resolved. This paper focuses on five interface areas that will have significant impacts on the design of FTF. 1. It is clear that tritium supply, inventory, breeding, recovery, safety and leakage are critical for the development of a DT

FTF and DEMO, which are impacted by the key parameter of plasma burn-up fraction. The requirement of 10% burnup fraction for low T start up inventory may not be achievable from the physics consideration. 2. Due mainly to parallel heat flux, there is a disconnect between the ITER-specified chamber heat flux of 2-5 MW/m^2 and the ARIES-type power reactor having a specified chamber surface heat flux of ~0.5 MW/m^2. This shows that ELMing operation will have to be avoided. There is another disconnect on the maximum heat flux of 10 MW/m^2 at the divertor between ITER and the higher power density FTF and DEMO. 3. Plasma control and transient event avoidance and mitigation are the present approaches in order to avoid damaging the surface material. 4. Tungsten (W) is the only surface material being considered for DT machines, but there are problems with the formation of W-fuzz and blisters under high temperature and fluence. The transport of W into the plasma can also limit the performance of the tokamak. The technique of real time condition with low-Z material will have to be developed. 5. There is inadequate data for material damage from 14 MeV neutrons and at high enough fluence for plasma facing components (PFC) and in-vessel components**.** 14 MeV neutron testing facilities will be needed to provide the material damage data for fusion material components development. The potential solution approaches to these coupled physics and technology issues will have to be understood and fully assessed before an acceptable solution can be found for FTF.

This work was supported by the US Department of Energy under DE-FG02-04ER54698.

References

[1] GREENWALD, M. (Chair), “Priorities, gaps and opportunities: Towards a long-range strategic plan for magnetic fusion energy,” Report to the Fusion Energy Sciences Advisory Committee, October 2007.

[2] HAZELTINE, R. (Chair), “Research needs for magnetic fusion energy sciences,” Report of the Research needs Workshop (ReNeW),” June 8–12, 2009, DOE/OFES Report, 2009.

[3] KESSEL, C.E. (Chair), “Fusion nuclear science pathways assessment,” Princeton Plasma Physics Report PPPL-4736, February 2012.

[4] ZINKLE, S.J. (Chair), “Opportunities for fusion materials science and technology research now and during the ITER era,” DOE/SC-0149, February 2012

[5] EL-GUEBALY, L., et al., Fusion Engineering and Design **84** (2009) 2072–2083

[6] NAJMABADI, F., et al., Fusion Science and Technology **54**, No. 3 (2008) 655-672

[7] NAJMABADI, F., et al., Fusion Engineering and Design **80** (2006) 3-23

[8] JACKSON, G.L., “An analytic expression for the tritium burn-up fraction in burning plasma devices,” submitted to Fusion Science and Technology, to be published.

[9] POLEVOI, A.R., et al., Nucl. Fusion, **45**, 1451 (2005)

[10] KUKUSHKIN, A.S., et al., J. Nucl. Mater., **415,** S497 (2011)

[11] MITTEAU, R., et al “Lifetime analysis of the ITER first wall under steady state and off normal loads,” PFMC-13 Conference, May 2011, Rosenhelm, Germany.

[12] RYUTOV, R., “Snowflake divertors for power plants,” ARIES Town Meeting on Edge Plasma Physics and Plasma Material Interactions in the Fusion Power Plant Regime, San Diego, CA (2010)

[13] KOTSCHENREUTHER, M., “The super X-divertor for high power density tokamaks,” ARIES Town Meeting on Edge Plasma Physics and Plasma Material Interactions in the Fusion Power Plant Regime, San Diego, CA (2010)

[14] KOVALENKO, D., “Optimization of QSPA-Be plasma gun facility for ITER ELM, disruption, and mitigated disruption simulation experiments. Preliminary results of Be erosion under ELM-like plasma heat loads,” Proc. 13th PFMC Workshop/1st FEMAS Conf., Rosenheim, Germany (2011)

[15] LINKE, J., “Testing of plasma facing materials and components,” Proc. 13th PFMC Workshop/1st FEMAS Conf., Rosenheim, Germany (2011)

[16] HUMPHREYS, D.A., et al., Bull. Am. Phys. Soc. **56**, 189 (2011)

[17] NISHIJIMA, D., et al., J. Nucl. Mater. **329**, 1029 (2004)

[18] WONG, C., et al., "Development of Si-W transient tolerant plasma facing material", 20th International Conference on Plasma Surface Interactions 2012, Aachen, May 21–25, 2012, to be published
[19] "Binary Alloy Phase Diagrams," Vol. 3, 2nd edition, edited by Thaddeus B. Massalski, ASM International, 1990
[20] MOSLANG, A., "Need for a Comparison of Different Neutron Sources with High He/dpa Rates Proc," ICFRM-15 Conf., Charleston, SC (2011)
[21] ODETTE, G.R., et al., "Helium transport, fate and management in nano structural ferritic alloys: In situ helium implanter studies," J. Nucl. Mater. **417**, 1001 (2011)