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European DEMO divertor target: Operational requirements and material-design interface

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Recently, an integrated program of conceptual design activities for the European DEMO reactor was launched in the framework of the EUROfusion Consortium, where reliable power handling capability was identified as one of the most critical scientific as well as technological challenges for a DEMO reactor. The divertor is the key in-vessel plasma-facing component being in charge of power exhaust and removal of impurity particles. The DEMO divertor target will have to withstand extreme thermal loads where the local peak heat flux is expected to reach up to 20 MW/m² during slow transient events in DEMO. To assure sufficient heat removal capability of the divertor target against normal and transient operational scenarios under expected cumulative neutron dose of up to 13 dpa is one of the fundamental engineering challenges imposed on target design. To develop the design of the DEMO divertor and related technologies, an R&D work package 'Divertor' has been set up in this consortium. The subproject 'Target Development' is devoted to the development of the conceptual design and the core technologies of the plasma-facing target. Devising and implementing novel structural heat sink materials (e.g. W/Cu composites) to advanced target design concepts is one of the major objectives of this subproject. In this article, the underlying design requirements imposed by the envisaged power exhaust goal and the prominent material-design interface issues are discussed. In addition, the candidate design concepts being currently considered are presented together with the related material issues. Finally, the first results achieved so far are presented.

Keywords: DEMO; structural design; divertor target; plasma-facing component; high-heat-flux; heat sink materials

1. Introduction

Recently, an integrated program of conceptual design activities for a European DEMO reactor was launched in the framework of the EUROfusion Consortium, in which the capability of reliable power handling was identified as one of the most serious technological and scientific challenges for commercially viable fusion power [1]. In this regard, divertor is a key in-vessel component being in charge of power exhaustion and ash (helium) removal. While the design and technology for the ITER divertor has been successfully developed, its applicability for a DEMO is not necessarily assured yet mainly because the neutron irradiation dose expected for the DEMO divertor will be by an order of magnitude higher than that of the ITER divertor. Such strong irradiation is likely to result in the embrittlement of materials affecting the structural performance of the component.

A dedicated R&D work package 'Divertor' (WPDIV) has been set up in the EUROfusion Consortium in order to develop the conceptual design solution for the DEMO divertor system together with the related technology. The objective of the WPDIV is to assure sufficient power exhaust capability and structural reliability of the DEMO divertor against the stationary and transient operational scenarios under cumulative neutron dose of up to 13 dpa. The envisaged lifetime of a DEMO divertor cassette was defined to be two full power years (fpy).

In total, a gross budget of about ten million euros were allocated for the first five years (2014-18), an extension of project period for further two years is foreseen. The final conceptual design report is due by end of 2020.

Fig. 1 illustrates the work breakdown structure of the WPDIV. The WPDIV consists of two subproject areas: 'cassette design & integration' and 'target development'. The former subproject is devoted to the development of a holistic design concept for the divertor cassette system on the basis of neutronic, thermal, hydraulic, structural (static, dynamic), and functional analyses. The latter one is dedicated to developing the conceptual design of the plasma-facing target together with the core technologies such as materials production and mock-up fabrication. To devise novel heat sink materials and to implement them into advanced target design concepts is one of the major emphases of this subproject.

In this article, a brief overview is presented on the work program and the major activities of the subproject 'target'. A report on the companion subproject 'cassette' has been published elsewhere [2]. The aim of this article is to introduce the European divertor target program to the international fusion community. The focus is put on operational requirements, structural design rules, design rationale, engineering approaches, and state-of-the art technologies achieved so far.

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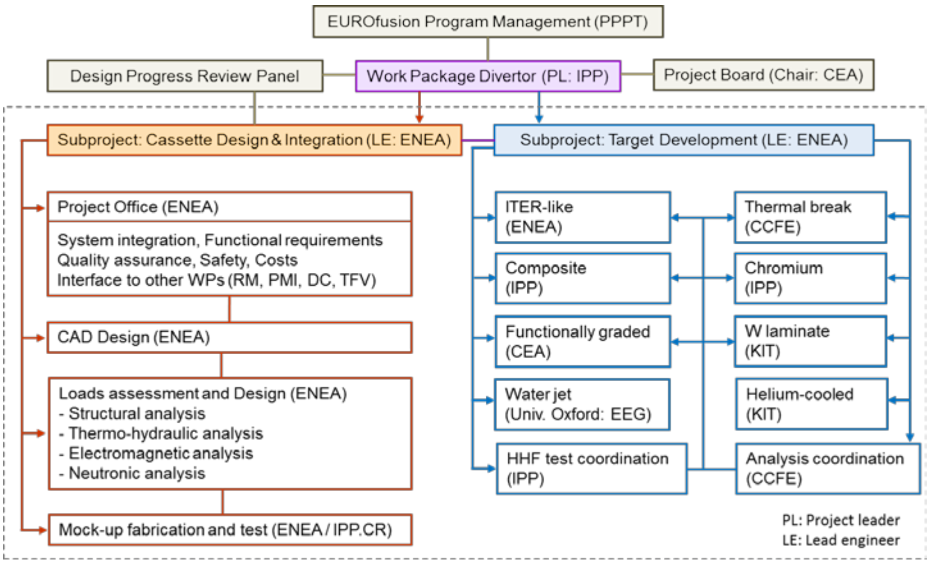

Fig. 1. Schematic project breakdown structure of the EUROfusion work package 'Divertor'.

2. DEMO divertor cassette and target plate

A CAD contour model of the current DEMO divertor cassette design (as of 2015) is illustrated in Fig. 2. The divertor cassette model has a significantly reduced size compared to the ITER divertor. It shall be equipped with a dual coolant circuit system where each piping system runs separately through the plasma-facing components and the cassette body, respectively. Both outboard and inboard baffle parts were removed from the cassette and attached to the breeding blanket. The tritium breeding ratio is predicted to increase from 1.13 to 1.19 owing to the gain of the additional breeding areas.

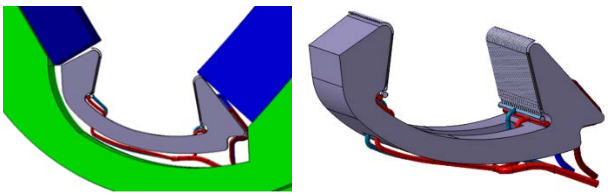

Fig. 2. CAD model of a single DEMO divertor cassette (2nd version as of 2015).

The baseline model for the plasma-facing units of the divertor target plates is the ITER-type monoblock design as illustrated in Fig. 3. It consists of an array of tungsten armor monoblocks connected by a copper alloy cooling tube. The technological ripeness of this design concept has already been demonstrated in numerous high heat flux qualification tests conducted in the ITER divertor project [3]. Variants of the baseline design and exotic design concepts are also considered in the WPDIV.

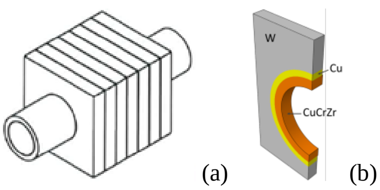

Fig. 3. (a) Schematic illustration of a typical plasma-facing unit of the ITER-type divertor target, (b) half cross section model.

3. Loading environment for target plates

The total power to be deposited on the whole DEMO divertor is estimated to be 259 MW, where 112 MW thereof is expected to be deposited on the plasma-facing components with non-uniform heat flux density profiles. The vertical target plates will have to withstand extreme thermal loads at the narrow band of the strike point (a few mm to a few cm) due to the heavy bombardment of plasma particles. The peak heat flux is predicted to reach locally 10 MW/m² during quasi-stationary operation and even up to 17-21 MW/m² when a slow transient event (i.e. plasma reattachment) takes place. Except the strike point, the average heat flux loads will be kept below 5 MW/m² in the most regions of the vertical targets and the dome (not shown in Fig. 2).

Moreover, the DEMO divertor will be exposed to an intense neutron irradiation. The peak dose is predicted to amount up to 6.5 dpa in the Cu tube and 2 dpa in the W armor per fpy. The continual irradiation of fast neutrons can produce crystal defects and transmutation products e.g. helium gas or brittle intermetallic phases (W/Re) in materials leading to embrittlement and other detrimental effects like reduction of heat conductivity [4]. Irradiation damage can be partially recovered, if the materials are annealed at a sufficiently high temperature. The thermal recovery effect can be utilized to mitigate the irradiation damage by tuning the operation temperature window.

Additionally, pulsed fusion operations will generate a cyclic variation of temperature and thermal stress which may cause fatigue damage of materials. The number of heat flux pulses set by the design specification is 5000 for normal operation. A long pulse operation at elevated temperatures may foster irradiation creep and synergetic creep-fatigue interaction [5]. Last but not least, the acute thermal shock by fast plasma instabilities such as ELMs (edge localized modes) or plasma disruption may impose additional thermo-mechanical impacts on the surface of the target plates producing cracks and damage.

4. Design requirements for targets

4.1 Rationale for target design

The highly complex and harsh loading environments addressed above pose a serious engineering challenge in designing and materializing a reliable divertor target for DEMO. The boundary conditions for design are derived from the requirement of maximum local thermal power exhaust at the strike point which has been defined to be 20 MW/m² as a starting assumption. This peak thermal load is the paramount design constraint to which cooling conditions and materials options are subordinated. It is noted that the basic policy of the WPDIV is to assure the power exhaust capability of the targets fully covering the entire operational scenarios of the DEMO divertor (pro tem. including slow transients).

The optimal cooling condition shall be determined as compromise between two competing requirements, i.e.: 1) any coolant bulk boiling and dry-out accident must be avoided at the maximum heat load, and 2) the structural material for the heat sink tube should maintain sufficient ductility (or toughness) and strength to avoid structural failure even at the maximum irradiation dose. It should be noted that the level of coolant bulk temperature has a trade-off effect between two design parameters, namely, the thermal margin to the local critical heat flux (CHF) at the cooling tube and the strain margin to the necking or to the ultimate tensile strain of the heat sink material. In this regard, the thermal and mechanical performances of the irradiated structural material are major drivers for design innovation. Hence, the optimal cooling condition should assure the minimum required heat flux margin to the local CHF at the heat sink and at the same time the permissible operation temperature range for the heat sink and the armor.

4.2 Cooling condition for vertical targets

The design efforts in the WPDIV are mostly focused on the early DEMO (DEMO 1). Here, only water-cooled target concepts are considered as the near-term solution whereas helium-cooled target concept is regarded as a long-term option. The reason for this is the fact that the technology readiness level of the water-cooled target concept is essentially more progressed compared to any of helium-cooled target concepts and mature enough for industrial production. In what follows all discussions are referring to water-cooled target concepts only. It is also reminded that the plasma-facing components (e.g. target) and the cassette body shall be cooled separately by their own cooling circuits, respectively. Thus, there is still an option to cool the cassette body using either helium or water irrespective of target cooling.

In the WPDIV, the minimum required peak heat flux margin to the local CHF at the cooling tube was set as 1.4 for all possible operational scenarios including slow transients. Given that the local heat flux peaking factor of a typical ITER-like target ranges from 1.5 to 1.7 depending on the given cross section geometry, the local maximum heat flux at the tube would reach from 30 to 34 MW/m² when the peak surface heat flux at the target strike point attains up to 20 MW/m². This means that the prescribed heat flux margin requires a cooling condition which is capable of exhausting the local maximum heat flux up to 42 - 48 MW/m² at the tube. In order to meet this requirement an optimal combination of cooling parameters was identified as follows:

- inlet temperature: 150 °C (or lower)
- inlet pressure: 5 MPa
- inlet velocity: 16 m/s
- tube inner diameter: 12 mm (with a swirl tape)

This parameter set was determined using an empirical CHF correlation. According to the vapor pressure curve of water, the margin to the saturation (boiling) point at the local pressure of 4 MPa (assuming a pressure drop of 1 MPa) and the coolant temperature of 150 °C amounts to 1.6, which seems reasonably large.

4.3 High heat flux performance of the baseline model

Prior to defining a design strategy, a preliminary study was carried out to investigate the high heat flux (HHF) performance of the baseline target model by means of finite element analysis [6]. As a reference model for the study, the ITER-type baseline design was investigated. For the sake of brevity, current discussions are focused on the heat sink tube with regard to structural design.

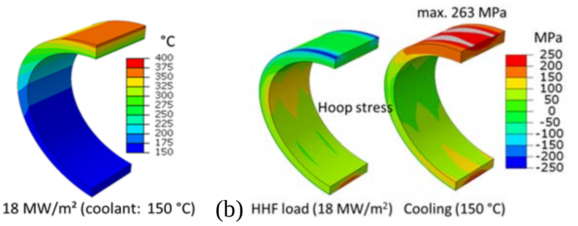

Fig. 4. (a) Temperature and (b) thermal stress fields produced in the cooling tube during steady state HHF loading (heat flux load: 18 MW/m², coolant temperature of 150 °C) [6].

Fig. 4 illustrates the temperature and stress fields in the cooling tube computed for a high heat flux load of 18 MW/m² at the assumed coolant temperature of 150 °C. The thermal stress (hoop component) is concentrated on the upper part of the tube where conduction heat flux is also concentrated. Since strong tensile stress appears in the upper part during cooling, irradiation embrittlement would be an issue. The steady state temperature values at two critical positions (the uppermost interface and inner wall) are listed in Table 1 for three HHF loads ranging from 10 to 18 MW/m². The maximum temperature of the tube at the uppermost position varies from 263 to 348 °C as the HHF load increases from 10 to 18 MW/m². At the same time, the inner wall temperature varies from 229 to 288 °C. Care should be paid to prevent corrosion there.

Table 1. Maximum temperature of the tube at the interface and at the inner wall under three different heat flux loads.

Heat flux (MW/m²)	10	15	18
Interface at top (°C)	263	316	348
Inner wall at top	229	266	288

5. Material-design interface issues

5.1 Structural design rules

The two baseline materials considered for the plasma-facing armor block and the heat sink tube of the DEMO divertor target are W and CuCrZr alloy, respectively. In addition, advanced Cu matrix composite materials are also considered as alternative heat sink material. CuCrZr alloy has been selected as the baseline heat sink material owing to the excellent thermal conductivity and good mechanical properties. Such an ideal combination is the unique advantage of CuCrZr alloy adequate for high heat flux applications.

It should be noted that the use of CuCrZr alloy for a structural application (NB. heat sink tube is a structural component) is subject to structural design rules which are based on selected structural failure criteria relevant for the anticipated load cases. The most important failure modes relevant for the heat sink cooling tube are:

1. local fracture due to exhausted ductility,
2. plastic flow localization,
3. low cycle fatigue (ratchetting hardly occurs),
4. fast fracture (local , global).

The 'ITER SDC-IC' code (Structural Design Criteria for In-vessel Components) provides structural design rules dedicated to the ITER in-vessel components such as divertor and first wall [7]. The SDC-IC contains both elastic as well as plastic design criteria for the failure modes listed above, respectively. It is thought that these SDC-IC design criteria could be 'selectively' used also for the structural design of the DEMO divertor target as a first approach, provided that the materials data for the expected neutron irradiation dose are available for the code. But, there is a fundamental technical limitation in applying the 'elastic' criteria to a monoblock-type joint component. The problem lies in the fact that the residual stress field produced by thermal strain mismatch during fabrication can hardly be reasonably assessed in a fully elastic analysis. In order to bypass this issue, a dedicated computational analysis procedure has been developed so that the elastic criteria could be more reasonably used as a first approximation. But, this procedure is valid only for the $3S_m$ and fatigue criteria where the intensity of stress ranges is estimated instead of absolute magnitude of stress intensity.

Eventually, proper elasto-plastic failure criteria need to be employed in order to consider the plastic yield effect as well as the cyclic plasticity behavior of the soft Cu interlayer and of the ductile CuCrZr heat sink in a more realistic manner. The four criteria mentioned above can mutually compensate. If at least one of the criteria be violated, then the other criteria could be used for a final judgment. The first and the second criteria are related to the embrittlement and creep effect by irradiation whereas the third and the fourth criteria to the irradiation effect on the fatigue life and the toughness. It should be noted that the fracture toughness of irradiated CuCrZr alloy increases monotonically with decreasing temperature in the range between 20 °C and 350°C [7].

5.2 Design constraints by irradiated CuCrZr alloy

The individual failure criteria are formulated in terms of a sum of relevant stress intensity terms estimated at the most significant line segments and the corresponding design stress limits which indicate material's resistance. In the case of the elastic design rules, the relevant design stress limits for the first three failure criteria are S_d, S_e and $3S_m$, respectively. For brittle fracture, K_{Ic} (or J_{Ic}) is used. The ITER SDC-IC code provides the material data of irradiated CuCrZr alloy for these design stress limits. In Fig. 5 the design stress limits of irradiated CuCrZr alloy are plotted over the temperate range relevant for the operation of heat sink tube [7].

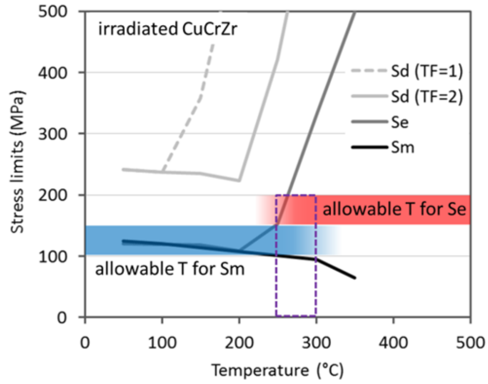

Fig. 5. Design stress limits of irradiated CuCrZr alloy plotted over the relevant temperate range for the heat sink tube [7].

The rapid increase of S_d and S_e at elevated temperature regime (> 200 °C) is attributed to the increasing total and uniform elongations, respectively, due to overall thermal recovery of irradiation defects. This effect is illustrated in Fig 6 where the tensile curves of CuCrZr alloy before and after irradiation are plotted for two test temperatures below and above the threshold of thermal activation [8].

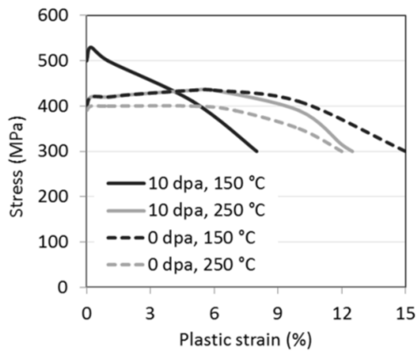

Fig. 6. Tensile curves of CuCrZr alloy tested before and after irradiation at two temperatures (test temp. = irrad. temp.) [8].

Fig. 5 implies that there is a desired optimal operation temperature range for the tube which allows one to avoid softening due to irradiation creep at higher temperatures and irradiation embrittlement at lower temperatures. This ideal temperature window ranges from 250 °C to 300 °C which is obviously too narrow to be practical for the actual operation of target as manifested in Table 1 and Fig. 4. From this situation it can be concluded that there is a need of advanced target design concepts for DEMO.

6. Variants of target design concept

One of the key missions of the WPDIV is to explore and to develop high-performance target design concepts being capable of exhausting the expected high heat flux load for the prescribed cooling condition. In the search for such advanced design concepts, application of novel materials for heat sink or interlayer has been the major approach in this project. Currently, seven kinds of design concepts are being considered for water-cooled target as listed in Table 2. These design concepts are subject to an evaluation test program for final qualification. The target development subproject consists of computational design studies, failure modelling, materials procurement, mock-up fabrication, high-heat-flux load tests and post-mortem analyses. In the following sections, some of the design concepts under development are briefly introduced.

Table 2. Design concepts for DEMO divertor target and dome being considered in EUROfusion WPDIV.

Concepts	Tube	Interlayer	Monoblock
ITER-like	CuCrZr	Cu	W
Thermal break	CuCrZr	Porous Cu	W
Composite 1	W_f/Cu	Cu	W
Composite 2	CuCrZr	W_p/Cu block (W tile)	
Chromium	CuCrZr	Cu	Cr (W tile)
Graded W/Cu	CuCrZr	W/Cu	W
W laminate	W/Cu	Cu	W

6.1 Monoblock with a thermal break interlayer

The underlying idea of this concept is to mitigate the heat flux concentration at the upper part of the tube by means of a thermal barrier interlayer inserted between the W monoblock and the Cu alloy tube. In the presence of the thermal break layer, the heat flux is diverted from the upper part and bypasses around the tube through the armor. The peak heat flux and the maximum temperature at the tube is reduced while the W monoblock is heated more uniformly as illustrated in Fig. 7 which shows the heat flux though the tube plotted along the periphery of the tube for 4 different coverage of the tube by the layer.

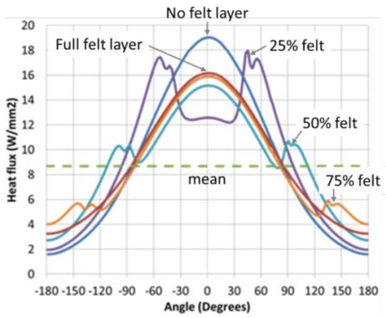

Fig. 7. Heat flux though the tube plotted along the periphery of the tube for 4 different coverage of the tube by the layer.

For the thermal break layer a porous Cu material made of felt or foam (Fig. 8 a) is used. A significant side effect is that the porous thermal break interlayer has such a low stiffness that it interrupts the load transfer at the bond interface reducing the thermal stress concentrated around the tube. This mechanical break gives a beneficial effect.

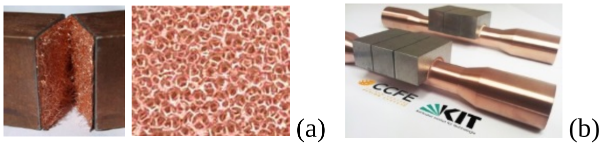

Fig. 8. (a) Porous Cu material used as thermal break (left: felt, right: foam), (b) Target mock-ups with a thermal break layer.

6.2 Monoblock with a wire-reinforced composite tube

The high tube temperature values under slow transient loads (see Table 1) suggest that loss of strength by high temperature and irradiation will be a critical design issue in terms of structural reliability. The composite heat sink concepts utilize the powerful strengthening effect by use of hard tungsten reinforcements in form of either wires or particles. It is well known that W wire-reinforced Cu composites exhibit much higher yield stress and rupture strength than matrix [9, 10]. In Fig. 9, the theoretically predicted tensile stress-strain curves of unidirectional W wire-reinforced Cu composites are plotted for 4 different volume fractions together with the data of W wire, Cu and CuCrZr alloy showing a strong strengthening effect.

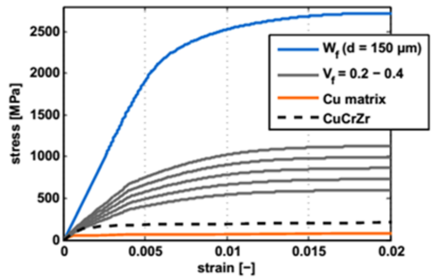

Fig. 9. Predicted tensile curves of W wire-reinforced Cu composites (unidirectional) with 4 different volume fractions. The data of a W wire, Cu and CuCrZr alloy are also compared.

In the design concept 'composite 1', a Cu composite tube reinforced with thin W wires is employed. Long W wires wrap the tube interior in form of a crossed multi-layer architecture. Fig. 10 (a) shows the tube preform consisting of wound W wires before melt casting. This W wire preform was braided using an industrial textile process. Fig. 10 (b) shows the cross section view of a fabricated composite tube. The composite shows no pore or fault indicating high production quality.

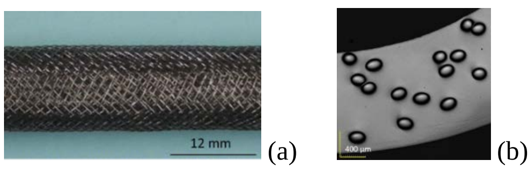

Fig. 10. (a) Tube preform made of wound W wires braided by an industrial process (before casting with Cu melt). (b) Cross section view of a fabricated composite tube.

6.3 Monoblock with a laminate composite tube

Another material used to enhance the high temperature strength of heat sink tube is W/Cu laminate composite. Fig. 11 shows the cross section of a composite tube made of the W/Cu laminate material (a) together with a successfully fabricated test mock-up (b).

The laminated W/Cu composites exhibit an excellent ductility and toughness under impact bending test even at room temperature supporting the idea of this design concept [11]. However, the results of recent irradiation tests show that this material suffers from pronounced embrittlement and loss of strength after irradiation even at 750 °C [11]. This discouraging feature gave a serious impact on concept validation and led to critical scrutiny with regard to the design feasibility for a DEMO reactor.

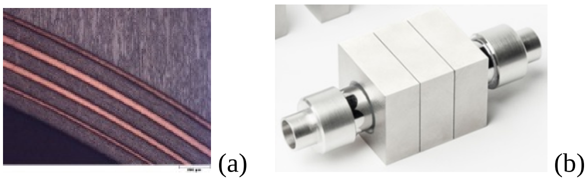

Fig. 11. (a) Cross section of a W/Cu laminate composite tube, (b) Target test mock-up equipped with a W/Cu laminate tube.

6.4 Monoblock with a functionally graded interlayer

In the conventional ITER-type target design the armor and the tube are joined via a soft Cu interlayer. The role of the Cu interlayer is to relax thermal stress by plastic yield enhancing structural integrity. However, there is a concern regarding the deteriorative effects of irradiation e.g. swelling and embrittlement by transmutation [12].

A target design concept without thick Cu interlayer is being investigated as reference model. In this concept, W monoblocks are joined to the CuCrZr tube via a thin bond coat made of a compositionally graded W/Cu layer. Fig. 11 shows the typical composition profiles of W and Cu in 3 different samples of W/Cu film fabricated using a physical vapor deposition method. The thickness of the bond coat was set at first 17 µm, but variants with larger thickness will be also tested.

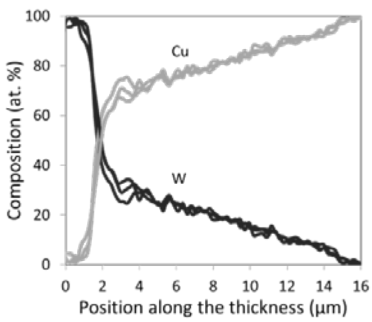

Fig. 11. Composition profiles of W and Cu in three W/Cu bond layers fabricated using a physical vapor deposition method.

Test mock-ups could be successfully manufactured by an industrial scale process (see Fig. 12). Hot radial pressing technique was applied for joining.

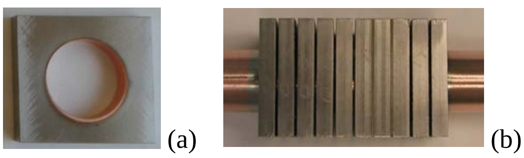

Fig. 12. A test mock-up consisting of W monoblocks and a CuCrZr tube joined via a functionally graded thin W/Cu bond coat (17 µm thick) manufactured by an industrial process.

6.5 Hybrid Cr monoblock with a W flat tile

The plasma-facing components of the DEMO divertor will be exposed to highly non-uniform profiles of heat flux and irradiation dose along the poloidal positions. The dome umbrella and the target plate locating far away from the strike point shall be subjected to lower heat flux but much higher irradiation dose compared to the strike point. In this case, chromium may be an interesting heat sink material offering beneficial properties such as low ductile-to-brittle transition temperature (ca. 300 °C), low activation, acceptable thermal conductivity and medium thermo-elastic constants which reduce thermal mismatch stress (see Fig. 12). Preliminary high heat flux tests are ongoing using a water-cooled monolithic test mock-up.

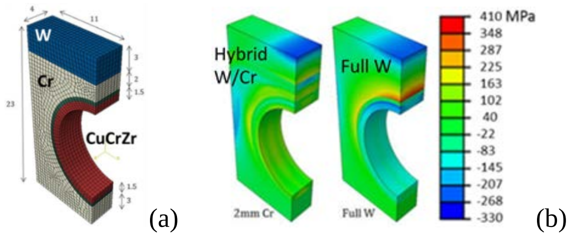

Fig. 12. (a) Hybrid Cr target concept consisting of a W armor tile, Cr heat sink block and a Cu alloy tube, (b) Thermal stress fields (hoop stress) in a hybrid Cr target and full W target.

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References

[1] F. Romanelli et al., Fusion Electricity: A roadmap to the realization of fusion energy, EFDA, 2012.
[2] J.H. You, G. Mazzone, E. Visca, et al., Fusion Eng. Des., DOI: 10.1016/j.fusengdes.2015.11.012
[3] T. Hirai, F. Escourbiac, V. Barabash et al., J. Nucl. Mater. (2015) doi.org/10.1016/j.jnucmat.2014.12.027.
[4] M. Rieth, et al., J. Nucl. Mater. 432 (2015) 482-500.
[5] B.N. Singh, et al., Final Report on In-Reactor Creep-Fatigue Deformation Behaviour of a CuCrZr Alloy, Risø-R-1629(EN) 2008, Risø DTU.
[6] M. Li, J.H. You et al., Fusion Eng. Des., 90 (2015) 88-96.
[7] ITER, ITER structural design criteria for in-vessel components, Appendix A, G 74 MA 8 01-05-28 W0.2.
[8] P. Fenici, et al., J. Nucl. Mater. 212-215 (1994) 399-403.
[9] J.H. You, Nucl. Mater. Energy, DOI: 10.1016/j.nme.2015.10.001.
[10] D.L. McDanels, Tungsten fiber reinforced copper matrix composites, NASA Technical paper 2924, 1989.
[11] L.M. Garrison, Y. Katoh, L.L. Snead et al., J. Nucl. Mater. (ICFRM 2015), submitted.
[12] S.A. Fabritsiev et al. Plasma Dev. Operations 5 (1997) 133-141.