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Advances of the Design Study of ITER-like divertor target for DEMO

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ITER-like divertor target for DEMO: design study and fabrication test

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As a major in-vessel component of a tokamak-type fusion reactor, the divertor is mainly in charge of removal of particles and partial power exhaust via scrape-off layer. The target plate of the divertor is directly exposed to non-uniform high heat flux on the surface by particle bombardment and radiation. In the case of ITER and a DEMO reactor, the peak surface heat flux is expected to reach up to 10 MW/m² during normal operation and 20 MW/m² during slow transient events like loss of plasma detachment.

This paper reports the results of a preliminary code-based design study and fabrication technology verification test which were conducted for developing an ITER-like divertor target design for the DEMO divertor.

The structural failure evaluation against the ratchetting and fatigue criteria of the ITER SDC-IC showed that the design with reduced dimensions would allow sufficient design margin (reserve factor) for three distinct thermal loading cases. The first trial of mock-up fabrication using a new joining furnace at ENEA was successfully completed. The ultrasonic inspection test made before and after the cyclic HHF tests at GLADIS facility demonstrated high quality of fabrication and robust design concept.

Keywords: DEMO, Divertor, Plasma-facing component, Target plate, Structural design, Cooling condition.

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1. Introduction

As a major in-vessel component of a tokamak-type fusion reactor, the divertor is mainly in charge of removal of particles and partial power exhaust via scrape-off layer. The target plate of the divertor is directly exposed to non-uniform high heat flux on the surface by radiation and particle bombardment. In the case of ITER and a DEMO reactor, the peak surface heat flux is expected to reach up to 10 MW/m² during normal operation and 20 MW/m² during slow transient events like loss of plasma detachment. The temporal fluctuation of temperature in the plasma-facing components (PFCs) lead to a variation of thermal stresses raising the risk of fatigue failure [1]. Neutron irradiation, which is predicted to be significant for the DEMO divertor (e,g. 13 dpa for copper heat sink after 2 fully power year), is an additional challenge for the engineering of divertor components since the damage of materials produced by irradiation may cause severe embrittlement at lower operation temperature [2].

In the frame of the EUROfusion Consortium launched in 2014, a preconceptual R&D program has been carried out in the Work Package 'Divertor' (WPDIV) to develop preconceptual basis for design solution and technologies [2]. In the subproject "Target development", the design concepts and engineering technologies are developed for the PFCs, in particular, the target plate. One of the 8 design concepts being currently considered in this subproject is the ITER-like tungsten monoblock target design developed originally for the water-cooled vertical target of the ITER divertor [3]. This ITER-like target design is adopted as a reference design model with which the other design concepts shall be compared in terms of thermohydraulic and structural performances as well as manufacturing aspects.

In this paper, an overview on the preliminary studies of design optimization, mock-up fabrication and high-heat-flux (HHF) test is presented which was performed in the framework of the subproject 'Target' of WPDIV program. Emphasis is placed on the design optimization in terms of monoblock dimension and geometry.

2 Finite element model and boundary conditions

The design of the ITER-like divertor target component consists of an array of tungsten monoblocks as armor, a CuCrZr alloy tube as heat sink and a thick soft copper interlayer between the cooling tube and the monoblocks. Rolled tungsten and CuCrZr-IG (SAA cw) are used as baseline materials for the armor and tube, respectively. Hot radial press (HRP) technique has been applied as a reference joining method for manufacturing the ITER divertor target [3]. Likewise, HRP process was used also in this study for fabricating mock-ups. Each test mock-up consisted of 15 rectangular monoblocks each with a thickness of 4 mm and a gap of 0.25 mm between two neighboring monoblocks. Considering actual kinematic constraints, equivalent static boundary conditions were applied for the computational simulation (see Fig. 1).

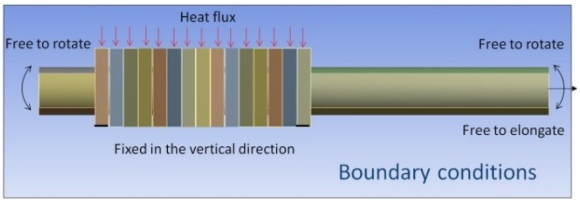

Fig. 1. A schematic view of a reference mock-up of the ITER-like divertor target design concept consisting of 15 tungsten monoblocks and a copper alloy cooling tube. Static boundary conditions are indicated.

Fig. 2 illustrates the model geometry created for finite element analysis (FEA). For the linear elastic FEA-based optimization study the commercial Design Exploration package of ANSYS 16.2 Workbench was used. Only a symmetric quarter part of a single block was modelled. The FEA mesh consisted of 6,400 elements (the element type SOLID186) and 30,000 nodes.

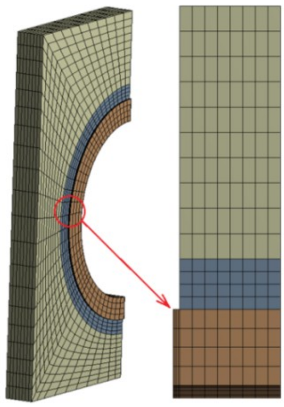

Fig. 2. FEA model (a symmetric quarter part of a single monoblock) and mesh.

The thermohydraulic boundary conditions were defined considering following design requirements [4]:

1. The specified maximum local surface heat flux is 20 MW/m². The thermohydraulic design of cooling was adapted to this transient peak heat flux load.

2. The reference heat flux considered for structural design optimization study is 10 MW/m² that is the quasi-stationary load expected for the normal operation. Structural failure at the transient peak load of 20 MW/m² was also investigated in terms of plasticity fatigue and fracture mechanics, but published elsewhere [5].

3. The local maximum coolant water temperature at the strike point is set to 150°C (or lower) in order to attain a sufficient safety margin to the critical heat flux (CHF) at

the cooling tube wall. The minimum required safety margin to CHF is 1.4.

4. The associated coolant pressure and local minimum velocity is set to 5 MPa and 16 m/s, respectively. Current cooling condition causes only slight pressure drop and thus requires low pumping power.

It could be necessary to reduce the coolant temperature and velocity to prevent corrosion or erosion of the inner tube (e.g. to < 130 °C and < 11 m/s). However, the prominent irradiation embrittlement of the structural heat sink material at lower temperature (< 200 °C) should be also taken into account.

The thermal analysis needed for the stress analysis in the structural design optimization process was carried out for the reference heat flux of 10 MWm^2. The heat transfer coefficient was calculated using the Sieder-Tate and the empirical Tong-CEA correlations [6].

3. Parametric design optimization

The geometrical parameters which were considered as optimization variables were:

i) The inner diameter of the CuCrZr cooling tube,
ii) The thickness of the cooling tube,
iii) The thickness of copper interlayer,
iv) The side minimum thickness of the W block.

The objective function to be minimized was:

i) The maximum temperature in the cooling tube,
ii) The maximum von Mises stress in the cooling tube.

The armor thickness from the surface to the interlayer was fixed at 5 mm. The influence of the varying armor thickness on the thermal and structural performance was studied elsewhere [5]. The optimized dimensions are as follows:

Tube inner diameter: 12 mm
Tube thickness: 1.5 mm
Interlayer thickness: 1 mm
Armor side thickness: 3 mm

4. Thermal and structural design studies

It is important to note that the results of computational stress analysis are essentially affected by the numerical quality and modelling assumptions. In the framework of WPDIV, standard technical guidelines for FEA of stress were produced in order to provide a dedicated simulation strategy for the ITER-like monoblock type joined target component. The elastic version of such guidelines is the 'Monoblock elastic analysis procedure' (MEAP, [9]). In the MEAP, residual stresses to be produced during the fabrication process are taken into account. To assess the residual stress in a more realistic way, the soft copper interlayer is assumed to be elasto-plastic while tungsten armor and copper alloy tube are regarded linear elastic. Selected elastic structural design rules of ITER SDC-IC code are applied to the structural heat sink tube assuming linear elastic behavior [10]. In the following, the fatigue life and the risk of ratchetting are assessed against the corresponding code-based failure criteria. It is noted that the two elastic failure criteria mentioned above can be approximately applied to the monoblock component with a plastic interlayer since these criteria use only stress intensity range instead of absolute stress values, thus are only slightly affected by the presence of residual stress.

4.1. Thermal rules

The MEAP prescribes three thermal rules as limiting guidelines for the structural design of the monoblock-type divertor target:

1. The bulk temperature of the cooling tube should be within the allowed operation temperature range being limited by irradiation embrittlement, irradiation creep and or thermal softening.
2. The tube wall heat flux should be lower than the CHF.
3. The surface temperature of the tungsten armor should be less than 1800°C to avoid overall recrystallization of the major armor bulk.

The local maximum temperature in the cooling tube under the normal HHF load of 10 MW/m^2 was 306 °C which is sufficiently lower than the maximum allowed service temperature for CuCrZr (< 350°C) [10]. The average temperature in the tube was 275 °C which is lower than the temperature limit against irradiation creep for CuCrZr (< 300°C) [10].

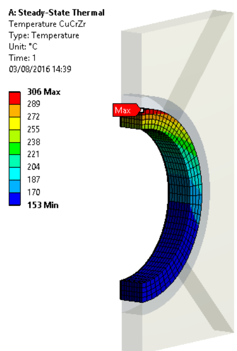

Fig. 3. Temperature distribution in the CuCrZr cooling tube at the surface heat flux load of 10 MW/m^2.

The local peak heat flux assessed at the cooling tube wall (at 20 MW/m^2) was 32 MW/m^2 (CHF: 44.4 MW/m^2). The margin to the CHF was 1.39 (specified minimum margin: 1.4).

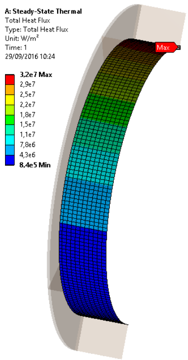

Fig. 4. Wall heat flux of the cooling tube at 10 MW/m².

The maximum armour temperature was estimated to be 880 °C at 10 MW/m², which was much lower than the recrystallization temperature of tungsten.

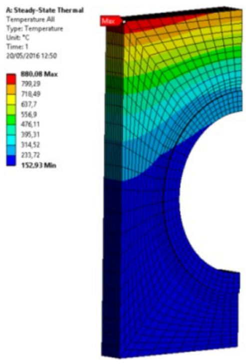

Fig. 5. Temperature distribution of the W armour.

4.2. Ratchetting rule (3 Sm)

The ratchetting criterion defined in the SDC-IC is as follows:

$$Max\left(\overline{Pl+Pb}\right)+\overline{\Delta Q}\leq 3\,Sm$$

where $\overline{Pl}$, $\overline{Pb}$, $\overline{\Delta Q}$ and Sm indicates the local primary membrane equivalent stress, primary bending equivalent stress, range of secondary stress and design stress S_m.

The ratchetting failure risk was assessed against this criterion assuming three distinct component thermal load cases as specified in the MEAP, namely,

1) Shutdown to standby (300 cycles)
2) Shutdown to plasma heat load (300 cycles)
3) Standby to plasma heat load (6,000 cycles)

The results of ratchetting failure assessment based on the $3S_m$ criterion are summarized in Table 1. Linearized stress intensities were computed along four different supporting line segments on cross sections (B1 to B4 and C1 to C4) at different locations. It is seen that the ITER-like target has sufficiently large design margins to the ratchetting loads under normal pulsed plasma operation.

Table 1. Summarized results of ratchetting failure risk assessment based on the 3 S_m design criterion.

	Path	Temp	3Sm	3Sm	P+Q	Reserv Factor	Reserv Factor
		mean	Un-Irr	irr		Un-Irr	irr
load case		°C	MPa	MPa	MPa		
Shut-standby	A0	188	407	N/A	272	1,50	N/A
shut-PlasmaQ	B1	274	371	N/A	313	1,19	N/A
	B2	275	370	N/A	272	1,36	N/A
	B3	154	421	N/A	274	1,54	N/A
	B4	154	421	N/A	324	1,30	N/A
Standby-PlasmaQ	C1	274	371	349	271	1,37	1,37
	C2	275	370	347	203	1,82	1,82
	C3	154	421	421	140	3,01	3,01
	C4	154	421	421	130	3,24	3,24

4.3. Fatigue

The fatigue life of the tube was evaluated according to the SDC-IC prescription. As the assessment against the $3S_m$ criterion has already implied, the sum of the primary membrane plus bending stress range is limited to purely elastic regime, and thus no low cycle fatigue is expected from the primary loads. However, the local peak stress and secondary stress may cause fatigue damage. The total fatigue usage was calculated to be 0.55 (see Table 2) giving a reserve factor of 1.8.

Table 2. Estimated fatigue life and usage of the tube.

		Shutdown to plasma	Shutdown to standby	Standby to plasma heatload	
T	°C	306,0	150,0	306,0	
Δ(P+Q+F)	Mpa	348,0	278,0	312,0	
E	MPa	1,16E+05	1,23E+05	1,16E+05	
e1	%	0,2671	0,2004	0,2395	
neuber K	MPa	92,97	55,70	74,73	
e3	%	0,30	0,22	0,26	
kv	-	1,175	1,077	1,145	
e4	%	0,0467	0,0155	0,0348	
eTot	%	0,34	0,24	0,30	
N	cycles	6,8E+03	3,4E+04	1,2E+04	
requirement	cycles	300,00	300,00	6000,00	Total Usage
Usage V	-	0,04	0,01	0,50	0,55

5. Mock-up fabrication and HHF Tests

For the verification of joining technology, a mock-up of the ITER-like divertor target design for DEMO was fabricated using a dedicated HRP furnace devised for joining divertor target components. For comparison, a mock-up of the original ITER divertor target was also produced as reference by the same process. The block dimensions of the respective mock-up types are:

1. Original ITER divertor target: 28×30×12 mm³
2. ITER-like DEMO divertor target: 23×25×4 mm³

The major reason for the reduced monoblock size for the DEMO divertor case was the fact that the driving force for fatigue crack initiation at the armor surface and deep cracking of the monoblock at 20 MW/m² can be significantly decreased in the smaller geometry Error: Reference source not found.

5.1. Fabrication of mock-ups

In the first step, OFHC copper melt was casted into the machined hole of individual tungsten monoblocks and the solidified copper cast was further drilled to make a hole. Subsequently, a CuCrZr alloy tube was bonded to the casted copper layer by the HRP technique in high vacuum at 600 °C and applied pressure of 60 MPa. The furnace could reach the process temperature quickly by

Joule heating. The joining quality of the mock-ups was inspected by means of non-destructive ultrasonic test method before the cyclic HHF tests were performed at test facility GLADIS Error: Reference source not found.

5.2. Original ITER mock-up

One mock-up with the original ITER target geometry was successfully fabricated (Fig. 6 left up) which passed the ultrasonic inspection (Fig. 8 right). The mock-up withstood 500 pulses at 20MW/m^2 without any structural failure (Coolant: 20 °C, 1.15 l/s, 10 bar). A surface crack appeared after 258th cycles (Fig. 6, right). The crack stopped about 3 mm below the surface (no grain growth was observed in that region).

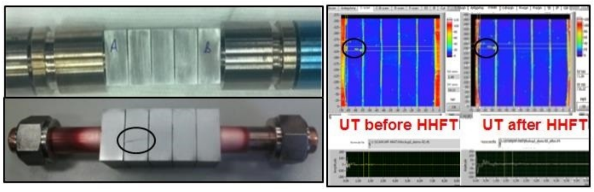

Fig. 6. Original ITER divertor target mock-up (left) and ultrasonic test images (right) before and after HHF test (defects did not increase).

5.3. ITER-like DEMO mock-up

The first ITER-like mock-up for DEMO with reduced size was successfully fabricated (Fig. 7, left up) with a single fault (Fig. 7, right). Up to now, it was loaded with 100 HHF pulses at 10 MW/m^2 and screening test up to 20 MW/m^2 (Fig. 7, right) without any defects.

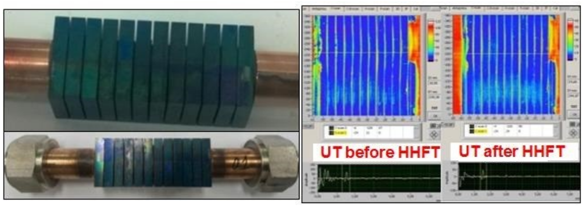

Fig. 7. ITER-like DEMO divertor target mock-up with the reduced size (left) and ultrasonic test images (right) before and after HHF test (defects did not increase).

6. Conclusions

In this paper, the results of a preliminary code-based design study and fabrication technology verification test were presented which were conducted for developing an ITER-like divertor target design for the DEMO divertor. To this end, design optimization was carried out at first to determine an optimal size of a tungsten monoblock. The optimized dimensions were

- Tube inner diameter and thickness: 12 mm, 1.5 mm
- Interlayer thickness: 1 mm
- Block size: 23×25×4 mm³ (side thickness: 3 mm)

The structural failure evaluation against the ratchetting and fatigue criteria of the ITER SDC-IC showed that the design with reduced dimensions would allow sufficient design margin (reserve factor) for three distinct thermal loading cases. Attention should be paid to the limitation imposed on the code-based analysis approach due to the linear elastic assumption. To bypass this shortcoming the soft copper interlayer was assumed to be plastic.

The first trial of mock-up fabrication using a new joining furnace at ENEA was successfully completed. The ultrasonic inspection test made before and after the cyclic HHF tests at GLADIS facility confirmed the high quality of fabrication and robust design concept.

Acknowledgments

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References

[1] J.H. You, Nucl. Mater. Ener. 5 (2015) 7-18.
[2] J.H. You, et al., Fus. Eng. Des. 109-111 (2016) 1598-1603.
[3] E. Visca et al., Fus. Eng. Des. 75–79 (2005) 485-489.
[4] J.H. You et al, Nucl. Mater. Ener. (2016) 10.1016/j.nme.2016.02.005.
[5] M. Li et al, Structural impact of armor monoblock dimensions on the failure behavior of ITER-type divertor target components, Fus. Eng. Des., in press.
[6] L.S. Tong, ASME 75-HT-68 (1975)
[7] R.E. Nygren, J. Plasma Fus. Res. Series, 3 (2000) 225-232.
[8] J. Schlosser et al., Finite element calculations for plasma facing components, in ‘Workshop on high heat flux component cooling’, Grenoble, 1993, 815.
[9] EUROfusion WPDIV Report, ‘Monoblock elastic analysis procedure’ (MEAP), 2015.
[10] Structural Design Criteria for ITER In-Vessel Components (SDC-IC), July 2011.
[11] M. Li, Design options to avoid deep cracking of W armor at 20 MWm^{-2}, submitted to Fus. Eng. Des.
[12] H. Greuner et al., Journal of Nuclear Materials 367–370 (2007) 1444