Critical Heat Flux Test with the Ferritic Martensitic Steel Mock-ups for the DEMO Blanket First Wall

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1. Introduction

Korea has proposed and designed a DEMO concept considering a Helium Cooled Molten Lithium (HCML) Test Blanket Module (TBM) to be tested in the International Thermonuclear Experimental Reactor (ITER) [1-4]. In these concepts, Ferritic Martensite Steel (FMS) is used as the structural material. The blanket FW of these concepts is an important component which faces the plasma directly and therefore, it is subjected to high heat and neutron loads. The FW is composed of the FMS as a structural material and an armor material such as tungsten and beryllium. Fabrication technology have been being developed expecially for the joining between an armor material and FMS and more the Critical Heat Flux (CHF) should be invetigated for design and safety aspect.. In the present study, three FMS mock-ups without armor material were fabricated with a HIP (Hot Isostatic Pressing), which was developed similarly to the development of the ITER blanket FW in Korea [5]. And they were tested in the high heat flux (HHF) test facility.

2. Test and Results

2.1 Preparation of the Test

Six channels were fabricated with wire cutting for making rectangular channel (20 mm x 10 mm) and one-, two-, and three-channel mockups were fabricated with the HIP methods (1050 ^{o}C, 150 MPa, 2 hours). For recovering the mechanical strength, post HIP heat treatment were performed; normalizing at 950 ^{o}C for 2 hours and tempering at 750 ^{o}C for 2 hours. With the microstructure of the two-channel mockup was observed but there were no pores and cracks. It means that the mockup was successfully fabricated with the proposed HIP conditions. To be installed in the HHF test facility, Neutral Beam Injection (NBI) test stand at KAERI, which was developed for heating up the KSTAR plasma and it can produce 4 MW of a beam for 300 sec and its heat flux can be more than 30 MW/m^2. The manifolds were welded in the mockups to be installed in the test failicty, as shown in Fig. 1 [6].

2.2 CHF evaluation

From the previous studies for the CHF in the one-sided high heat flux loading conditions like the fusion environment, modified CHF at the wall, equation (1) was evaluated, which were developed by A. R. Raffray et. al. for the ITER divertor [7];

$$CHF_w = 0.23 fGH_{fg}\left(1+0.00216\left(\frac{P}{P_c}\right)^{1.8}\mathrm{Re}^{0.5}\,Ja\right)Cf \quad (1)$$

where,

$$f = 8\,\mathrm{Re}^{-0.6}\left(\frac{d_h}{d_o}\right)^{0.32} \quad (2)$$

$$Ja = \frac{\rho_f}{\rho_g}\frac{C_p\,(T_{sat}-T)}{H_{fg}} \quad (3)$$

and where: CHF_w is the CHF at the tube wall (W/m^2); f is the friction factor calcualted from Eq. (2); G is the coolant mass velocity (kg/m^{2}s); T is the local coolant temperature (^{o}C); P is the local coolant pressure (MPa); T_{sat} is the saturation temperature corresponding to P (^{o}C); H_{fg} is the latent heat of vaporisation of water at T_{sat} (J/kg); P_c is the critical pressure, 22.1 MPa; Re is the Reynold number; d_h is the hydraulic diameter; Ja is the Jakob number; d_o is the reference diameter12.7x10^{-3} m. With this equation, the CHF at the wall is about 1.46 MW/m^2 and the test was performed near this value.

A transient CFX analysis was performed at 1.5 MW/m^2 heat flux for each mockups with the same test conditions to compare with the experimental data such as wall and coolant temperature. Figure 2 shows the temperature distribution of the overall mockup and their center plane.

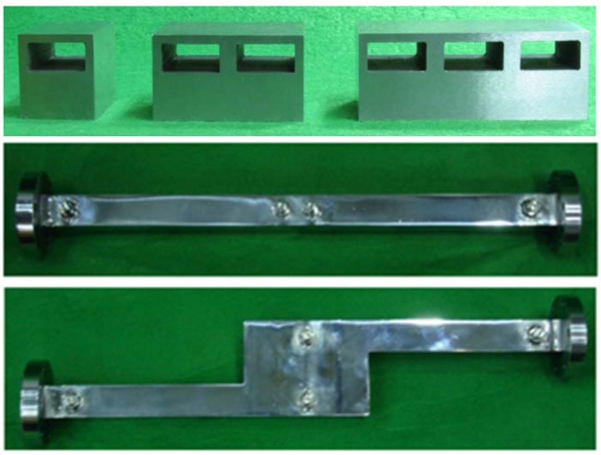

Fig. 1. Fabricated channels and mockups with them.

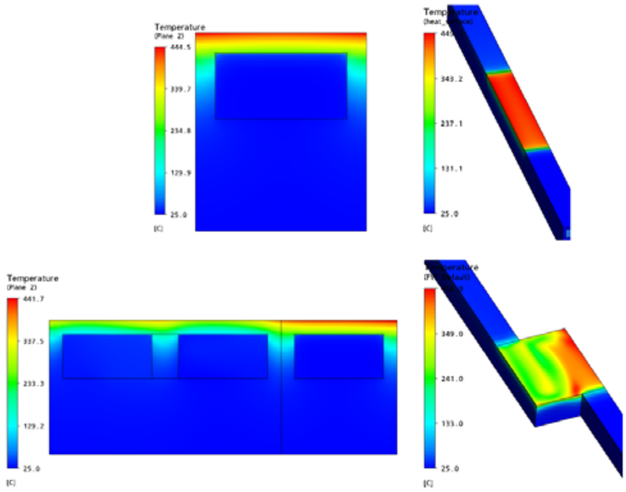

Fig. 2. CFX analysis results for 1.5 MW/m^2 heat flux.

2.3 Test Results

The mockups were tested under the following heat fluxes with 20 cycles fore each heat flux; 0.5, 1.0 MW/m^2 heat fluxes for one-channel mockup and 0.5, 1.0, 1.25 MW/m^2 heat fluxes for three-channel mockup. With these lower heat fluxes, the mockups showed no damange or water leakage. And then, both mockups were broken at 1st cycle under 1.5 MW/m^2 heat flux, as shown in Fig. 3. During the test, the temperatures at water inlet and outlet, and two thermocouples for wall temperature were measured. The measured temperatures and predicted ones by ANSYS-CFX were also compared, as shown in Fig. 4. They showed a good agreement.

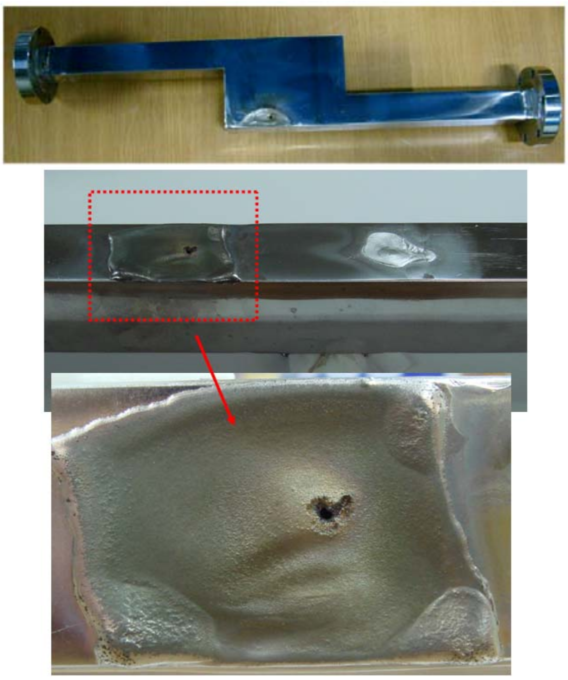

Fig. 3. Photos of the mockups after HHF test.

3. Conclusions

In order to investigate the CHF with the FMS for the DEMO blanket FW, HHF test was performed with the fabricated mockups up to 1.5 MW/m^2 heat flux. CHF at the wall was evaluated from the previous correlation and the value for this test is 1.46 MW/m^2 heat flux. The correlation seems to predict well the CHF at the wall for the fusion application.

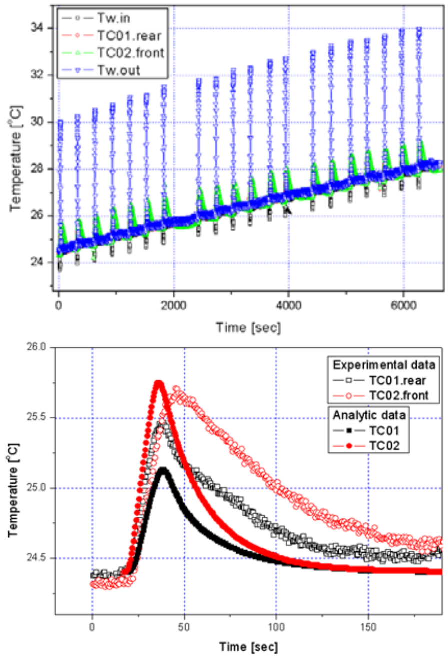

Fig. 4. Measured temperatures during the HHF test with the three-channel mockup (0.5 MW/m^2 heat flux).

REFERENCES

[1] B. G. Hong et. al., " Development of a tokamak reactor system code and its application for concept development of a demo reactor," Fusion Eng. Des. 83 (2008) 1615-1618

[2] D. W. Lee, et. al., "Current status and R&D plan on ITER TBMs of Korea," Journal of Korean Physical Society, Vol.49, Dec. 2006, pp S340-S344.

[3] D. W. Lee, et. al., "Preliminary design of a helium cooled molten lithium test blanket module for the ITER test in Korea," Fusion Eng. Des. 82 (2007) 381-388

[4] D. W. Lee, et. al., "Helium cooled molten lithium TBM for the ITER in Korea," Fusion Sci. and Tech. 52 (2007) 844-848

[5] D. W. Lee, et. al., "Development of fabrication and qualification methods for the first wall of the International Thermonuclear Experimental Reactor," Journal of Korean Physical Society, Vol.51, No.3, Sept. 2007, pp 1210-1215.

[6] D. W. Lee, et. al., "Fabrication and preparation of a high heat flux test with mock-ups for the KO HCML TBM ," Proc. of the KNS Spring Meeting, Jeju, Korea, May 2008

[7] A.R. RAffray et. al., "Critical heat flux analysis and R&D for the design of the ITER divertor," Fusion Eng. Des. 45 (1999) 377-407