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First wall and divertor engineering research for power plant in JAERI

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Abstract

JAERI has successfully developed plasma-facing components (PFCs) for ITER application, and the R&D activities are now being shifted to the development of the PFCs for a fusion power plant. This paper presents the R&D activity on the PFCs, such as the first wall and divertor, for a fusion power plant. The PFCs of the power plant will be subjected to heavy neutron irradiation and high heat/particle fluxes from the plasma during the continuous operation. In the present design of the PFCs, the candidate structural material is a reduced activation ferritic-martensitic steel, F82H, selected for low activation and high robustness against neutron irradiation, and the candidate armor material is tungsten, selected for the low sputtering yield and low tritium retention. To achieve PFCs using these materials, there are several R&D issues to be solved: (1) development of a cooling structure with high heat removal capability, (2) development of a suitable form of tungsten, (3) selection of a bonding technique for F82H and tungsten and (4) evaluation of structural integrity. Recent JAERI R&D achievements and status of these issues are presented.

Keywords: Fusion power plant; PFC; F82H; Screw tube; HIP bonding; Thermal fatigue

1. Introduction

In JAERI, design and R&D on the plasma-facing components (PFCs) for a fusion power plant are energetically being performed. In the fusion power plant, the PFCs are subjected to high heat flux, high particle flux and heavy neutron flux comparing to the PFCs in ITER. The PFCs are required to withstand these loadings during the continuous operation of the fusion power plant. Major design parameters for the PFCs of the fusion power plant are summarized in Table 1.

Major difference between the ITER PFCs and the fusion power plant PFCs is in material selection. In ITER, the reference structural materials are stainless steel and copper-based materials, such as austenitic stainless steel (316SS) and CuCrZr. In contrast, due to heavy neutron irradiation, the structural material of the PFCs of the fusion power plant should be low activation materials to minimize rad-waste in case of replacement for the damaged components. In addition, it is essen-

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Table 1
Major design parameters of PFCs of the fusion power plant

	Fusion power plant		ITER	
	First wall	Divertor	First wall	Divertor
Armor material	Tungsten	Tungsten	Beryllium	Carbon fiber composite (CFC) tungsten
Structural material	RAFM (F82H)	RAFM (F82H)	Copper alloy stainless steel (316SS)	Copper alloy stainless steel (316SS)
Surface heat flux (MW/m^2)		∼10	0.5	10–20
Coolant temp. (°C)	380	200	100	100
Coolant press. (MPa)	25	4	3	4

tial for the structural material of the PFCs to have more durability and more reliability from a reactor safety point of view and also to minimize the plant downtime. Hence, the candidate structural material is a reduced activation ferritic-martensitic (RAFM) steel, namely F82H, in the present design of the fusion power plant proposed by JAERI. However, there is also a disadvantage on the usage of F82H as a structural material of the PFCs. Thermal conductivity of F82H is about 1/10 of that of pure copper, which may cause the allowable surface heat flux to the PFCs to be lower than that of copper-based materials case. In addition, the maximum allowable temperature of F82H is limited to less than 550 °C due to creep deformation limits. Those limitations are serious if F82H is used as the structural material of the divertor plate, because the divertor plate is subjected to the highest heat flux among the PFCs in the fusion power plant. Therefore, evaluation of thermal performance of the F82H divertor plate should be carefully done. In this paper, critical heat flux experiments with a high performance cooling tube made of F82H are reported.

The candidate armor material of the fusion power plant is tungsten, selected for its low tritium retention and good refractory characteristic ($T_{melt} = 3400$ °C). Though carbon fiber composite (CFC) material, which is one of the candidate armor materials of the ITER PFCs, has excellent thermal conductivity (>500 W/m/K at room temperature) and has good refractory characteristic, it is difficult to use as an armor material for the PFCs of the fusion power plant because of its relatively high tritium retention and its high sputtering erosion rate. However, there is also an issue with the use of tungsten as an armor material. The mechanical properties of tungsten strongly degrade at elevated temperatures, above its recrystallization temperature of 1500 °C. Hence, the development of the bonding technique for the tungsten materials, which can sustain sufficient thermal performance of the PFCs even if it is irradiated by neutrons, is one of the important issues for the use of tungsten materials. Results are presented of a bonding trial, using a direct bonding method for F82H and tungsten material based on a diffusion bonding technique.

There is another important issue on the use of tungsten armor for the PFCs. The lifetime of the tungsten armored PFCs strongly depends on the sputtering erosion of tungsten. Sputtering erosion experiments have been carried out in JAERI to evaluate the erosion characteristics of tungsten materials, mainly aiming at the ITER application [1–3]. These experiments used a hydrogen ion irradiation facility. The hydrogen ion energy in these experiments was relatively high (∼300 eV) because of the limitation of the ion acceleration system. It is essential for the precise evaluation of the erosion characteristic of the tungsten materials to develop a new acceleration system, which can simulate the cold dense divertor plasma with ion energy less than 100 eV. Development of a low energy ion irradiation facility to simulate the sputtering erosion of the tungsten armor is introduced in this paper.

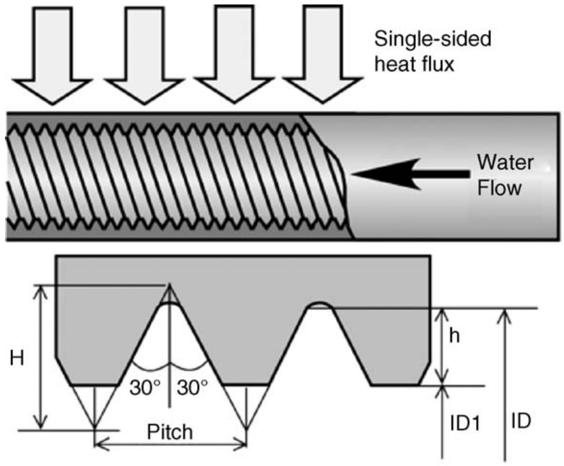
In addition, the lifetime of the PFCs also depends on the durability of the structural material, F82H. The number of cyclic thermal loading on the PFCs of the fusion power plant is expected to be much less than that of the ITER PFCs since the fusion power plant will operate continuously for more than a year without shutdown. However, it is important to investigate the low cycle thermal fatigue behavior of F82H, because F82H is going to be used not only for the PFCs of the fusion power plant but also for the structure of the ITER test blanket modules. Preliminary thermal fatigue experiments and numerical evaluation of fatigue lifetime of an F82H divertor mockup are also reported in this paper.

Basic R&D on tungsten and F82H as PFC materials are being carried out in JAERI. This paper describes the recent R&D achievements and status of the PFCs for fusion power plant.

2. Critical heat flux testing on F82H screw tube [4]

In a design of the fusion power plant proposed by JAERI, the divertor must handle high heat flux of more than 10 MW/m^2. In the design, the divertor is cooled by pressurized water flowing in a cooling structure inside high heat flux components to match a coolant condition of the blanket system. Accommodation of such high heat loads gives rise to several engineering issues affecting the performance and lifetime of the divertor components and influencing the selection of materials and configurations. Extensive research on high heat flux removal using pressurized water has already been carried out in the framework of ITER R&D. In one of these activities, the authors' group has developed a high performance cooling tube with a helical triangular fin on its inner surface [5,6]. Since the fin can be machined by simple mechanical threading, this tube is called a screw tube. In our previous experiments, it was reported that heat removal of the screw tube formed from pure Cu is twice as high as that of a smooth tube.

In the ITER divertor design, the candidate material of the cooling structure is Cu-alloy to effectively remove the high heat flux from the single plasma-facing surface of the components. Most of the ITER activities relating to the critical heat flux (CHF), which causes burnout of a cooling tube, have been done using Cu-alloy tubes or structures. In JAERI's power plant design, however, a candidate structural material for the high heat flux components is reduced activation ferritic-martensitic (RAFM) steel such as F82H instead of Cu-alloy because of the higher neutron load on the components than that of the ITER divertor. As the thermal conductivity of F82H is about ten times smaller than that of Cu-alloy, F82H has scarcely ever been adopted to high heat flux removal as high as 10 MW/m^2. This study is intended to examine heat removal capability of the screw tube formed from F82H under divertor-relevant single sided heating condition.



Name	Pitch	H	ID	ID1	h
M10xP1.5	1.5	1.23	10	8.38	0.81

Fig. 1. Schematic view of screw tube formed from F82H.

For this purpose we have carried out CHF testing on a screw tube formed from F82H by using a hydrogen ion beam heat load. The test sample is the screw tube with the thread of M10 and the pitch of 1.5 mm. Details of the screw tube formed from F82H are shown in Fig. 1. This screw geometry in a pure Cu tube was confirmed to have the highest incident critical heat flux (ICHF) among the other screw geometry of various pitch sizes in previous experiments [6]. The length of the test sample is 566 mm, and the length of screw section is 490 mm. The outer diameter is 12 mm, and the minimum wall thickness is 1 mm. The test sample is exposed to a heat flux on one side by using a hydrogen ion beam irradiation and is cooled by water flowing inside the tube. Inlet temperature and local pressure of cooling water were room temperature and 1 MPa. Flow velocity ranged from 2 to 12 m/s. The incident heat flux at the sample position was measured with a multi-channel calorimeter and it has a Gaussian profile. In the CHF testing the pulse duration of the ion beam irradiation is 10 s to make the test sample reach the thermally steady state condition.

Incident CHF values of the F82H screw tube are plotted in Fig. 2 as a function of the axial flow velocity,

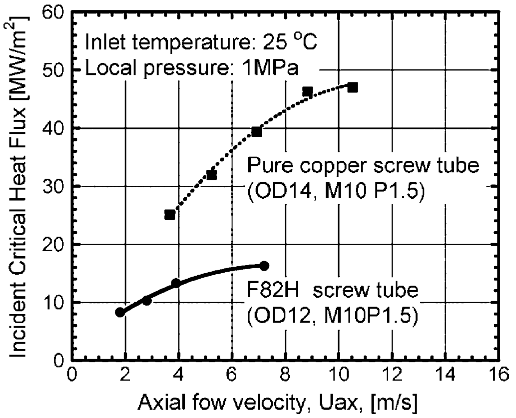


Fig. 2. Incident critical heat flux of the screw tubes formed fromF82H and OFHC-Cu with the thread of M10 and the pitch of 1.5 mm.

U_{ax}. The ICHF of the screw tube formed from pure-copper in the previous experiments [6] is also shown in comparison. The ICHF increased as the axial flow velocity increases in both tube cases. Using the F82H screw tube, ICHF of 13 MW/m^2 is obtained at U_{ax} of 4 m/s. This value is 1.3 times higher than the target value of the JAERI design of the divertor of the fusion power plant, although it is about half value of the OFHC-Cu tube at the same flow velocity. This is a promising result for further studies of the relevant cooling conditions of the fusion power plant at higher pressure and temperature.

3. Development of low energy ion irradiation test facility

As a development activity related to the tungsten armor for the divertor plate, a low energy ion irradiation facility has been developed to simulate sputtering erosion of tungsten armor due to the particle load from the plasma. Sputtering erosion of the tungsten armor is one of the most important issues for the lifetime evaluation of the divertor plate. So far, JAERI has developed the super low energy particle irradiation system (SLEIS) for the sputtering test of plasma-facing materials [1–3]. However, the maximum ion flux of SLEIS is limited to 1–3 × 10^{20} m^2/s at 150–300 eV beam energy. In this study, the ion acceleration system of SLEIS has been modified to obtain a higher ion flux at energy less than 100 eV for the simulation of a divertor plasma. Preliminary beam extraction experiment results using the improved grids are presented.

The modified SLEIS with an improved acceleration system has two specific features. One is thin triple extraction grids with multi-apertures. These grids are made of thin molybdenum plates, which have ion beam extraction area of 38 mm × 38 mm × 0.5 mm^t. There are 725 apertures in total, each with 0.9 mm diameter. The ion beam acceleration system consists of three grids: an acceleration grid, a deceleration grid and a ground grid. The gaps between these grids are 0.5 mm each, which are optimized by a numerical calculation using the BeamOrbit code [7]. The other specific fea-

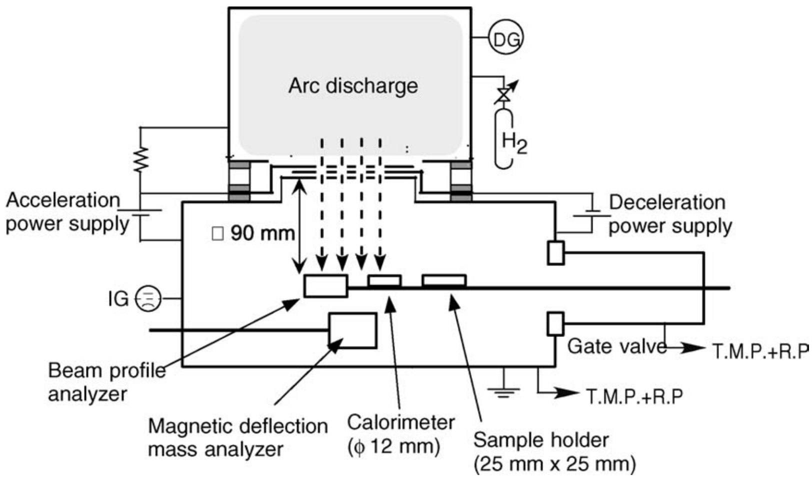


Fig. 3. Schematic drawing of the super low energy ion source (SLEIS).

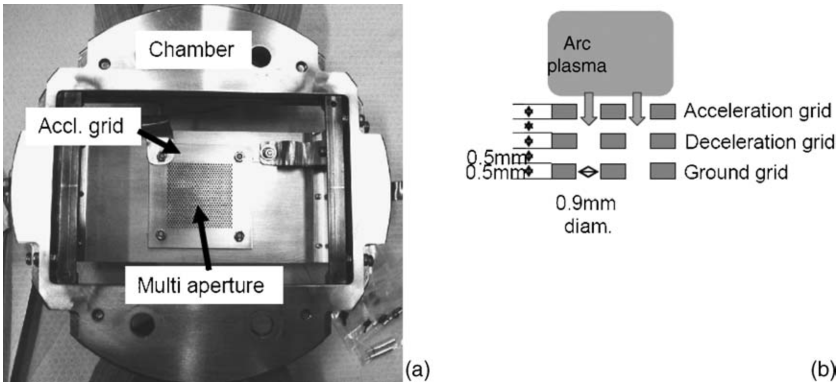


Fig. 4. (a) Photograph of new electrodes. To simplify the structure of electrodes and the installation, the deceleration and the ground grids are installed on the acceleration grid with the insulators. (b) Schematic of the new electrode system. Total number of apertures is 725.

ture is that the ion beam acceleration system utilizes the plasma sheath as the acceleration energy. A schematic diagram and a photo of SLEIS and the new acceleration system are shown in Figs. 3 and 4. To improve accessibility to these grids, the deceleration grid and the ground grid are supported from the acceleration grid by insulators, and the three grids are installed or removed together.

Preliminary ion beam extraction experiments were conducted. The ion beam current was measured at the target sample, and the ion flux was obtained from the target current and from beam composition measurements. The target sample, which has dimension of 25 mm $\times$ 25mm, is placed 90 mm downstream from the ground grid. The beam composition was determined by a mass analyzer with a deflection coil; the beam power profile was also determined by a multichannel beam current analyzer. Fig. 5 is a typical ion beam composition of the new acceleration system. In this experiment, hydrogen ions were generated by arc discharge with an arc power of 2.5 kW at a hydrogen gas pressure of 5.9×10^{-2} Pa. The hydrogen ion beam was extracted at a beam energy of 154 eV. As shown in this figure, the beam species of H_2^+ is dominant in this experiment and the impurities in the beam, such as H_2O, are shown to be less than 3%. Fig. 6 shows the hydrogen ion flux at the target sample as a function of beam energy. The maximum ion flux, more than 5×10^{20} H/m^2/s at the target sample, was obtained at a beam energy of 164 eV. In addition, an ion flux of 1.5×10^{20} H/m^2/s was also obtained at a beam energy of 55 eV. The obtained beam energy of 55 eV can be converted to 28 eV for monoatomic hydrogen since H_2^+ ion is dominant in this experiment, as described above. It is significant that high ion flux can be obtained at such low beam energy which is able to simulate ion irradiation on the divertor plate. Sputtering erosion experiments for plasma-facing materials, such as tungsten, will be started in SLEIS as a next step.

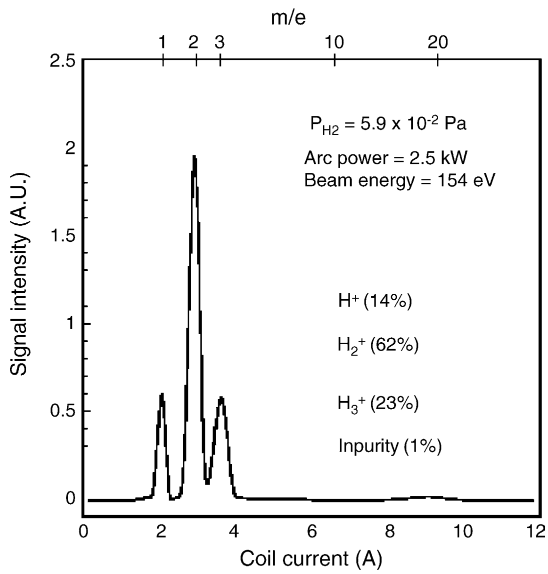


Fig. 5. Typical ion beam composition of the new acceleration system.

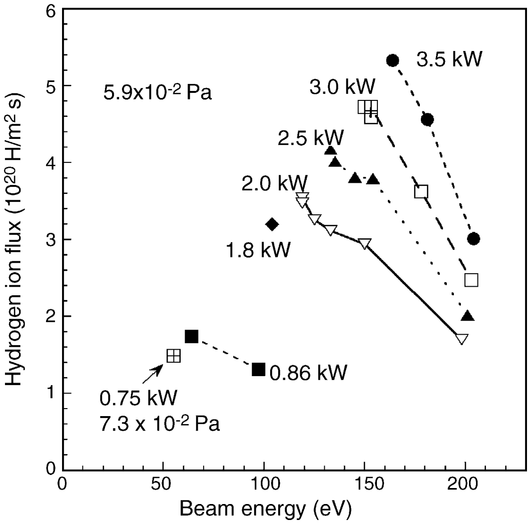


Fig. 6. Beam current at the target sample as a function of beam energy.

4. Development of bonding technology for PFCs

In JAERI, bonding methods for armor and structural materials of ITER PFCs have been developed. Brazing is one of the promising methods for the bonding technology of the ITER PFCs. However, degradation of braze joint due to heavy neutron irradiation will be a serious problem for the PFCs of fusion power plant. Both bond strength and resistance to neutron irradiation should be taken into account for the selection of braze filler materials, if brazing is applied as a bonding technique for the PFCs of a fusion power plant. On the other hand, direct bonding, method, such as diffusion bonding, can avoid such problems. Therefore, direct bonding trial for one of candidate armor materials, tungsten, on F82H substrate was performed. Uniaxial hot compression method using the SPS (spark plasma sintering) facility was utilized in this test. A small tungsten plate is directly bonded on F82H substrate with no compliant interlayer material to mitigate interfacial stress between tungsten and F82H substrate due to mismatch of their coefficient of thermal expansion. Fig. 7 shows a schematic of the bonding test. Compressive force was applied to a tungsten plate and F82H substrate. In parallel, electrical current was applied to those materials to heat them up. This process was done in a vacuum environment to avoid oxidation of the bonded specimen. The compressive force and the bonding temperature were 20 MPa and 960 °C with 30 min of holding time, respectively. The bonding temperature is equivalent to the normalizing temperature of F82H [8]. Hence, this process can combine bonding process with normalizing process of F82H. Then the bonded specimen was cut,

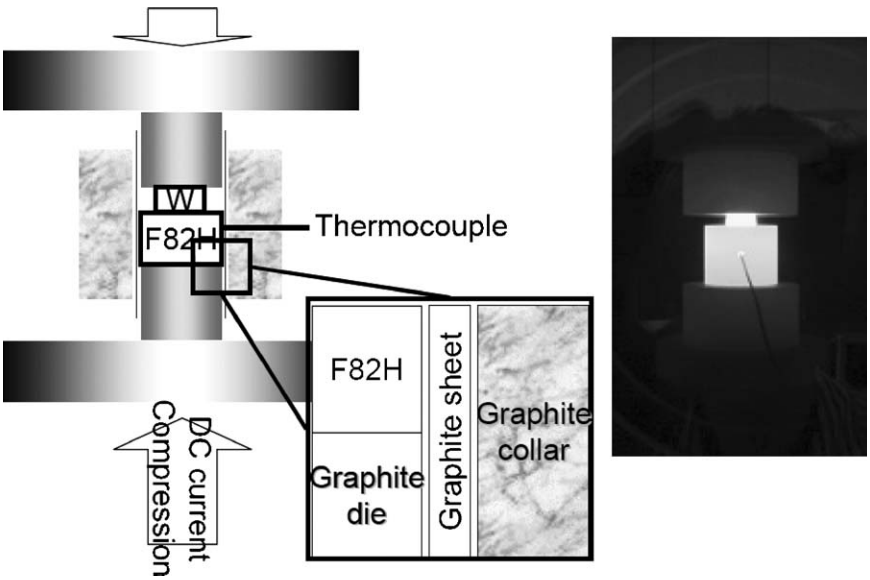
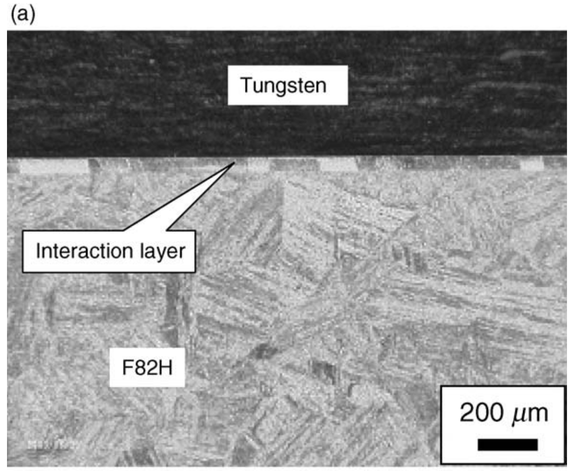


Fig. 7. Schematic of bonding test in SPS facility.



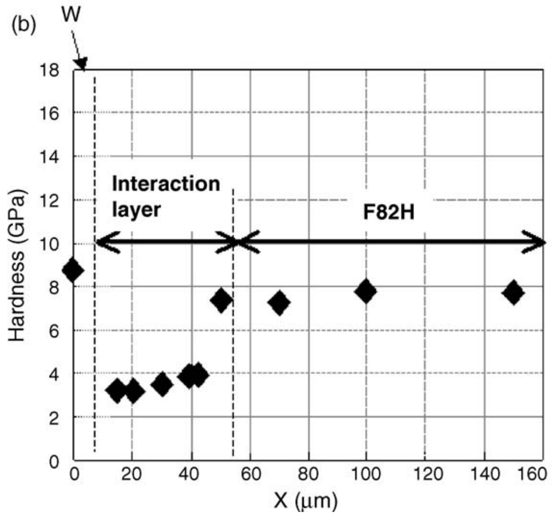


Fig. 8. SEM observation and hardness distribution of the bonded specimen. (a) SEM photo of the bonded area and (b) hardness distribution obtained from a nano-indentation test.

and metallurgical observation was made with a scanning electron microscope. Fig. 8 shows a cross-section of the bonded interface of the specimen. It is significant that no obvious bond flaw was found around the interface. On the other hand, it was also found that a thin interaction layer, which consisted of ferritic phase, was produced at the bonded interface of the F82H substrate. The thickness of the interaction layer is about 30–40 μm. It is speculated that this layer was transformed from an original martensitic phase of F82H due to decarburization and that it behaved as a "cushioning material" or compliant layer. Since the basic bonding

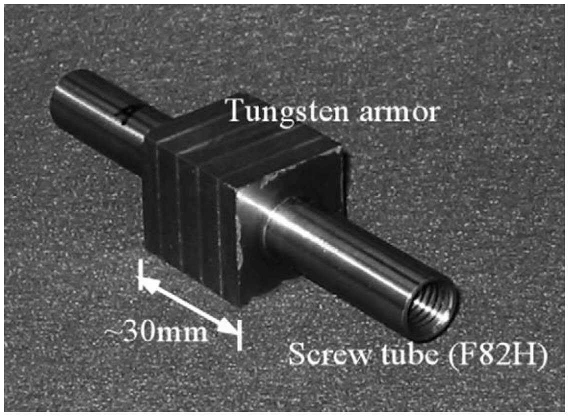


Fig. 9. Monoblock divertor element fabricated by HIP bonding process.

mechanism of this method is similar to that of HIP bonding, it is also significant that this method can not only directly bond tungsten and F82H but also gives suitable HIP bonding condition for the fabrication of PFCs of fusion power plant. Based on this successful result, a divertor element, which has monoblock geometry, was fabricated by HIP bonding method (see Fig. 9). Durability of the bond interface is to be investigated in a high heat flux experiment simulating the first wall and/or the divertor conditions.

5. Lifetime evaluation of divertor mockup made of F82H

It is important to establish a lifetime evaluation method for PFCs made of F82H. In particular, fatigue lifetime evaluation of F82H is a key issue since F82H is to be utilized not only as a structural material of PFCs of fusion power plant but also as that of the ITER test blanket module. Therefore, fatigue lifetime of F82H should promptly be investigated and the results used in the design of those components. Fig. 10 shows a divertor mockup [9]. The mockup was fabricated from a single F82H plate by mechanical machining, and has no armor material. It has no weld/bond part in the test section to eliminate the possibility of an initial weld/bond defect. The slit between adjacent heat sink blocks is to obtain a large stress/strain amplitude in the cooling tube at lower heat flux level. Another distinctive feature of the mockup is that the inner surface of the cooling tube

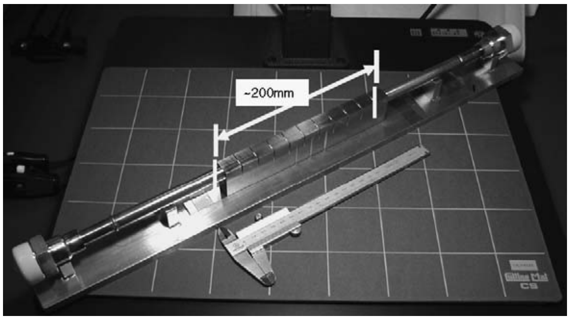


Fig. 10. An F82H divertor mockup thermal fatigue tested in an electron beam test facility.

has screw threads to enhance its heat transfer capability [6]. Based on our previous experiments on the cooling tube, the heat transfer capability of the "screw tube" is found to be comparable with that of a swirl tube. The screw tube, which has simple structure comparing to the swirl tube, is one of promising solutions as a cooling tube for PFCs of a fusion power plant. However, these screw threads may cause fatigue cracking due to stress concentration at the tip of the threads. A thermal fatigue experiment of the divertor mockups made of F82H was performed in an electron beam irradiation facility in JAERI. In the thermal fatigue experiment, surface heat flux was cyclically loaded on the test section of the mockup. The applied heat fluxes were 3 and 5 MW/m^2. The coolant flow velocity and the coolant temperature were 10 m/s and 25 °C, respectively. The temperature change of the heat sink was monitored by thermocouples attached to the side wall of the heat sink: the surface temperature change was monitored by an infrared camera. In parallel, thermal and stress analyses were performed to simulate the fatigue behavior of the mockup by using a finite element code, ABAQUS. Fig. 11 shows the finite element model of the mockup. In the experiment, water leakage due to fatigue cracking of the cooling tube was found at 650 cycles at a heat flux of 5 MW/m^2, whereas the mockup survived more than 10^4 cycles at 3 MW/m^2. After the experiment, the fatigue crack of the cooling tube was observed near the edge of a heat sink block using an optical microscope. Moreover, two fatigue cracks were found in a single

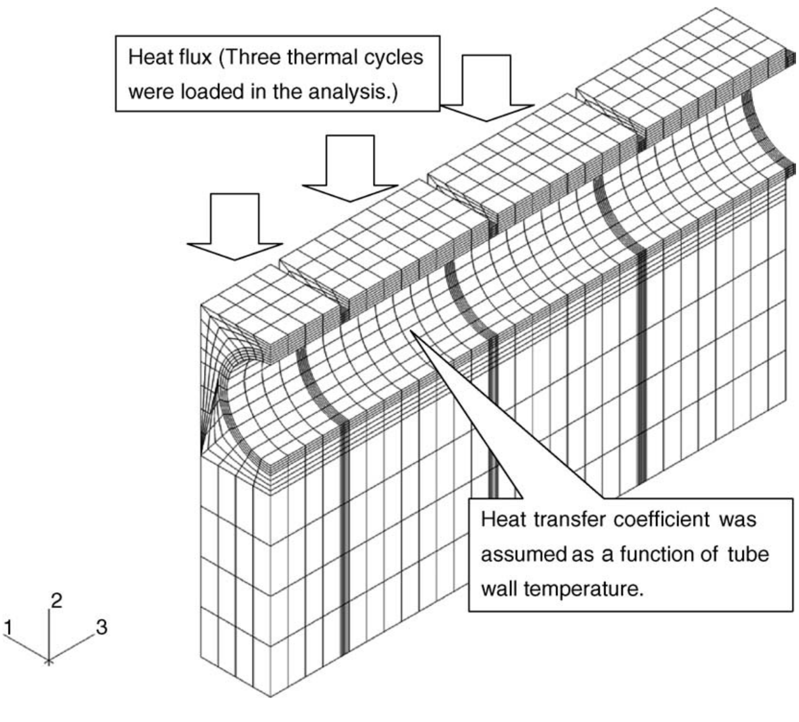


Fig. 11. Finite element model of the divertor mockup.

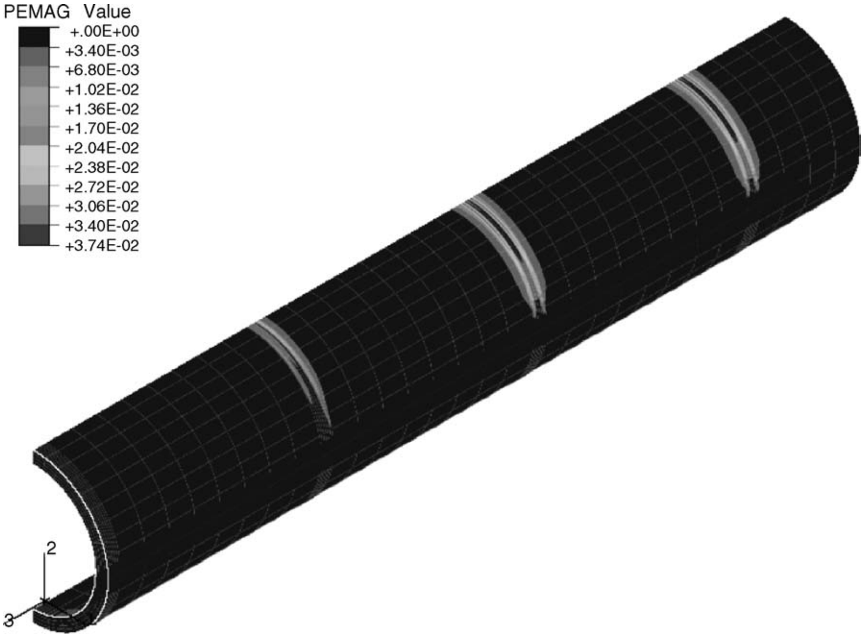


Fig. 12. Plastic strain distribution of the cooling tube of the divertor mockup at a heat flux of 5 MW/m^2 after three thermal cycles. (Plastic strain strongly concentrates at the cooling tube surface between the adjacent armor blocks.)

slit. Fig. 12 shows the strain distribution of the cooling tube of the mockup at 5 MW/m^2. Strong strain concentration appeared at the surface of the cooling tube near the edge of a heat sink block. The fatigue cracking in the experiment is attributed to that strong strain concentration at the cooling tube. On the other hand, the mockup tested at 3 MW/m^2 was also analyzed using the optical microscope. Two fatigue cracks, which do not penetrate the tube wall, were also observed at similar locations. Fig. 13 shows the fractography of the mockup tested at 5 MW/m^2. As shown in this figure, the fatigue crack propagated from the outer surface toward the inner surface of the cooling tube. A striation-like pattern was clearly observed, and it showed the direction of the fatigue crack growth. Though the shape of the divertor mockup does not exactly simulate the realistic divertor plate of a fusion power plant, this result indicates that the stress concentration at the tip of the screw thread

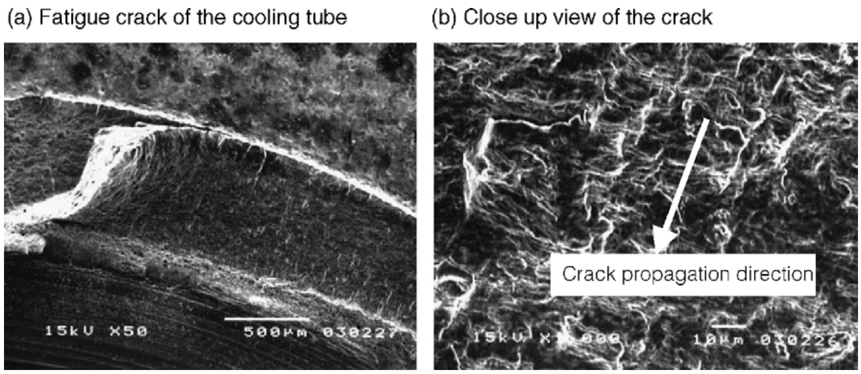


Fig. 13. Fractography of the cooling tube of the divertor mockup tested at 5 MW/m^2.

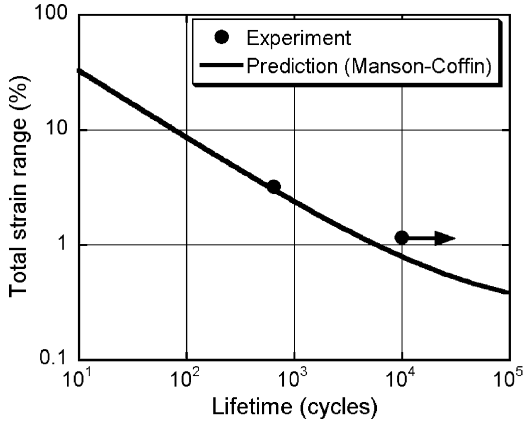


Fig. 14. Fatigue lifetime evaluation of the divertor mockup based on the finite element stress analysis. (Total strain range of the cooling tube was obtained from the numerical analysis.)

has little effect on the fatigue crack initiation from the cooling tube. Strain amplitude at the outer surface of the cooling tube mostly affects the fatigue crack initiation. Fig. 14 shows the lifetime evaluation of the divertor mockup. Manson–Coffin's law can reasonably predict the fatigue lifetime of the mockup. However, the fatigue lifetime obtained from the experiment gives qualitatively longer lifetime than the numerically predicted value. Though this finite element stress analysis can calculate the stress/strain amplitude of a "sound" mockup, it cannot take the stress mitigation due to crack opening into account. The stress reduction due to crack opening causes the fatigue lifetime from the experiment to be longer than the numerically predicted value. Further experiments to accumlate the fatigue lifetime data is necessary to obtain more reliable evaluation.

6. Conclusion

JAERI has energetically been conducting R&D on the plasma-facing components for the fusion power plant. Recent achievements and status of the R&D activities are summarized as follows:

(1) A critical heat flux experiment on the screw tube made of F82H was carried out. The incident CHF value of 13 MW/m^2 was obtained at a flow velocity of 4m/s. This ICHF value is more than 30% higher than the design heat flux of the divertor of the fusion power plant. In addition, the flow velocity of 4 m/s is less than design values for the divertor of the fusion power plant.

(2) The maximum ion flux of more than 5×10^{20} H/m^2/s was obtained at a beam energy of 164 eV in SLEIS. In addition, an ion flux of 1.5×10^{20} H/m^2/s was also obtained at a beam energy of 28 eV(H$^+$). It is revealed that SLEIS can perform sputtering erosion test of armor materials, which simulates the ion irradiation to the divertor plate of the fusion power plant.

(3) A direct bonding test of tungsten and F82H was successfully conducted using the SPS facility. It was found that a thin interaction layer, which consisted of ferritic phase, may serve as a compliant layer and lead to successful bonding between tungsten and F82H. Based on the bonding condition obtained from this test, a divertor element, which has monoblock geometry, was fabricated by HIP bonding method.

(4) A thermal fatigue experiment of divertor mockups made of F82H was carried out. The fatigue lifetime of the mockup can be reasonably predicted using Manson–Coffin's law. Further experiments to accumulate fatigue lifetime data is necessary to obtain more reliable fatigue lifetime evaluation.

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