2. Fusion Engineering Research Project

The Fusion Engineering Research Project (FERP) started in FY2010 at NIFS. Along with the conceptual design studies for the helical fusion reactor FFHR, the FERP has been developing technologies of key components, such as the superconducting magnet, blanket, and divertor. The research is also focused on materials used for blankets and divertors, the interaction between the plasma and the first wall including atomic processes, handling of tritium, plasma control, heating, and diagnostics. The FERP is composed of 13 tasks and 44 sub-tasks with domestic and international collaborations. (T. Muroga)

Reactor Design Studies

Conceptual design studies of the helical fusion reactor FFHR-c1, which aims at early demonstration of electric power generation with a reasonable construction cost by reducing the reactor size and increasing the magnetic field strength, has been advanced. In FY2017, it was shown that the steady-state operation with a fusion gain of ~15 can be achieved within the physics condition which has been already confirmed in the LHD experiments. Even though this fusion gain is sufficient for the self-sufficiency of electricity, further increase by an improvement of core plasma performance is still desired to increase the operational margin and to explore a more attractive design with a larger net electric output. Optimization of the winding law of the helical coils is one of the promising methods to improve the plasma performance while keeping several merits of the LHD-type configuration; coil winding with a small variation of the curvature, flexible divertor design by utilizing the rigid and robust divertor field structure, and high maintainability of blankets with large port apertures. In FY2018, the effect of changing the pitch modulation parameter, α, has been examined (Fig. 1). It was found that simultaneous improvement in the MHD stability and energy confinement can be achieved by decreasing α from 0.1 (employed in LHD and former FFHR designs) to 0.0 and selecting an adequate magnetic axis position [1]. It was also found that the change in α within the range of $-0.1 \leq \alpha \leq 0.3$ does not significantly affect the stored magnetic energy and electromagnetic stress on the coil supporting structure [2]. Although this tendency was already known in the design phase of LHD, the calculation reliability has been greatly improved and the design with a sufficient blanket thickness is shown for the first time by reflecting the latest ideas including the installation of the supplemental helical coils (NITA coils), which can enlarge the space between the helical coil and plasma without changing the plasma shape.

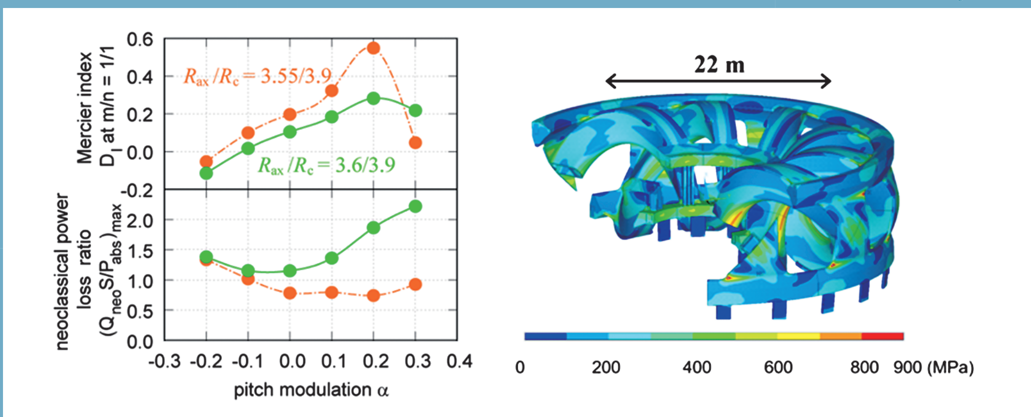


Fig. 1 (left) Dependence of the Mercier index (indicator of MHD instability) and the neoclassical power loss on the pitch modulation parameter α of the helical coil winding with a different magnetic axis position. (right) The stress distribution on the coil supporting structure in the case of $\alpha = -0.1$. The maximum peak stress is within the allowable limit (< 1 GPa).

[1] T. Goto *et al.*, Nucl. Fusion **59**, 076030 (2019).
[2] H. Tamura *et al.*, Fusion Eng. Des. **146**, 586 (2019).

(T. Goto)

Highlight

New ideas on the divertor and blanket

In the latest design of the helical fusion reactor FFHR-c1, a new divertor system using solid tin pebble shower and a new cartridge-type blanket system are adopted. A new tin pebble shower divertor system named REVOLVER-D2 [1] has been proposed. In this system, showers of tin pebble flow are used as the divertor target. The plasma flowing out from the confinement region hits this tin pebble shower before reaching the vacuum vessel wall. Shown in Fig. 1 is the result of an experiment demonstrating that a tin-alloy pebble flow can shut the fire of ~5 MW/m^2 emitted from a gas burner. Related R&D and the design study on the REVOLVER-D2 system are ongoing. A new cartridge-type blanket system named CARDISTRY-B2 [2] has been also proposed (Fig. 2). The main objective is to make construction and maintenance of the complicated helical fusion reactor as easy as possible. Together with the 3D computer-aided design, we are making full use of the recent 3D printing technology to confirm its feasibility (see photos in Fig. 3). Numerical simulations on the structure strength, thermal property, neutron shielding ability and tritium breeding ability of each cartridge, together with discussions on the remote maintenance scenario are now being carried out.

[1] T. Ohgo *et al.*, Plasma Fusion Res. **14**, 3405050 (2019).
[2] J. Miyazawa *et al.*, submitted to Plasma Fusion Res.

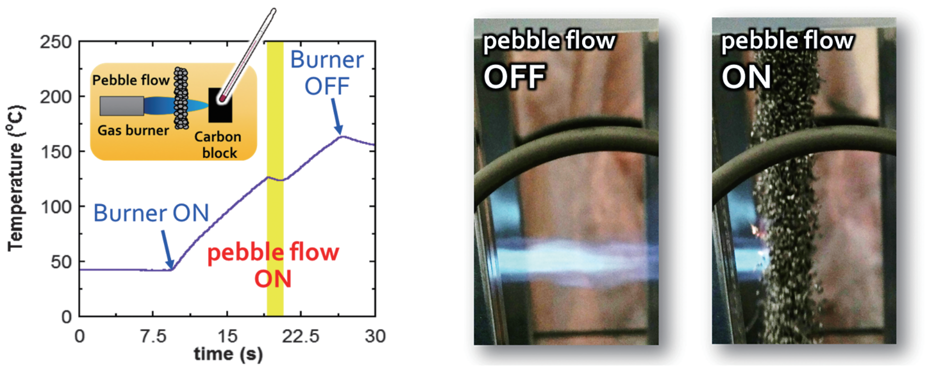


Fig. 1 Experimental results of a gas burner heating experiment on a tin-alloy pebble flow. The temperature of a carbon block, which is set behind the pebble flow and heated by the gas burner, stops increasing when the pebbles are flowing.

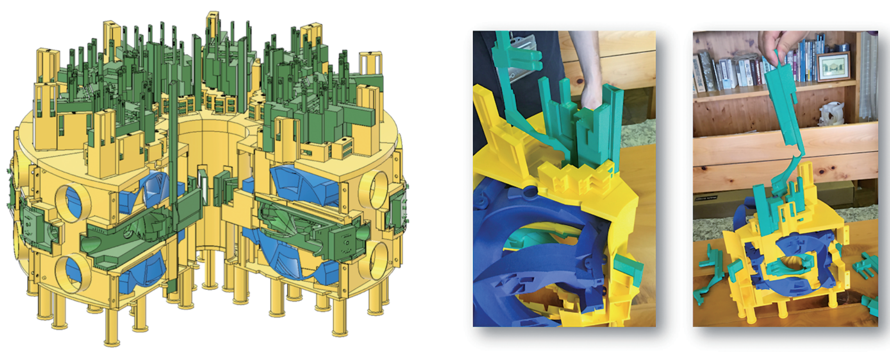


Fig. 2 (left) Bird's eye view of the CARDISTRY-B2 cartridge-type blanket system, and (right) photos of a small model made by 3D printing technology.

(J. Miyazawa)

Research and Development on the Superconducting Magnet

To realize a large-scale superconducting magnet system for the helical fusion reactor, both the Low-Temperature Superconducting (LTS) and High-Temperature Superconducting (HTS) large-current capacity conductors are being developed. For this purpose, the superconductor testing facility, with the maximum magnetic field of 13 T, large bore of 0.7 m diameter, large sample current of 50 kA, and temperature control capability of 4.2–50 K has been operational [1, 2]. Using this facility, an international collaboration experiment was carried out between Massachusetts Institute of Technology (MIT) in the US and NIFS, for the Twisted Stacked-Tape Cable (TSTC) HTS conductor [3, 4]. A one-turn coiled sample, made of a TSTC conductor, was designed and fabricated at MIT, transported to NIFS, and installed into the superconductor testing facility (Fig. 1). The experiment was carried out two times, in September 2018 and in January 2019. Figure 2 shows the cool-down curve of the magnet obtained in the second experiment. The maximum achievable current of the TSTC conductor sample was 19 kA at zero magnetic field. A sample current of 9 kA was measured at the bias magnetic field of 5 T and the inlet temperature of ~5 K.

Apart from this experiment, three kinds of HTS conductors are now being developed to be applied to the next-generation helical experimental device. In addition, as an extension of the ITER technology based on the LTS magnet concept, the development of an internal-matrix-strengthened Nb_3Sn wire is continuing using a ternary Cu-Sn alloy (such as Cu-Sn-Zn) to mitigate the degradation problem of the present Nb_3Sn wires due to mechanical strain [5].

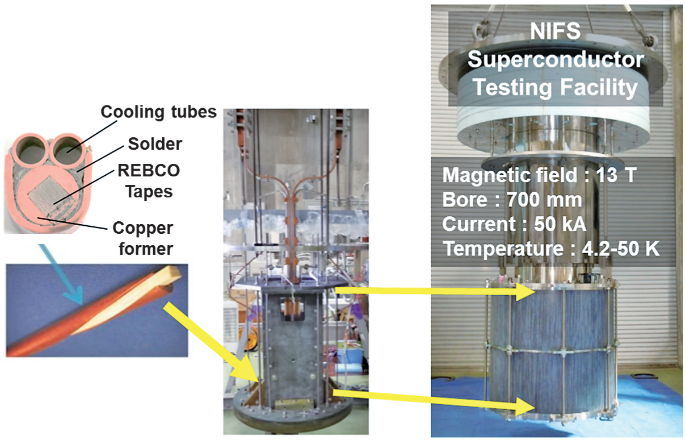


Fig. 1 Experimental setup of a TSTC HTS conductor sample for the international collaboration experiment between MIT and NIFS.

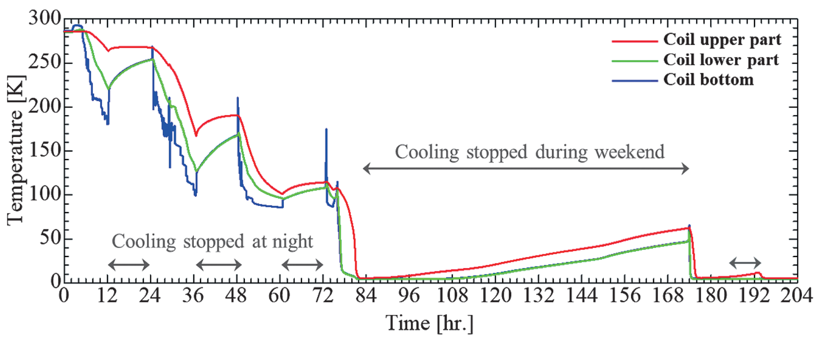


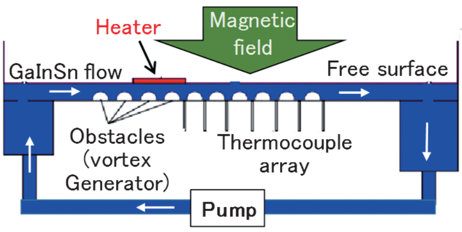
Fig. 2 Cool-down curve of the 13 T magnet from Jan. 8 to Jan. 16, 2019.

[1] S. Imagawa *et al.*, Plasma Fusion Res. **10**, 3405012 (2015).
[2] J. Hamaguchi *et al.*, Plasma Fusion Res. **10**, 3405020 (2015).
[3] M. Takayasu *et al.*, IEEE Trans. Appl. Supercond. **26**, 6400210 (2016).
[4] T. Obana *et al.*, submitted to Cryogenics.
[5] Y. Hishinuma *et al.*, IOP Conf. Series: Materials Science and Engineering **502**, 012175 (2019).

(N. Yanagi)

Research and Development on the Blanket

A FLiNaK/LiPb twin loop system named Oroshhi-2 (Operational Recovery Of Separated Hydrogen and Heat Inquiry-2) is being operated in NIFS as a collaboration platform for integrated experiments on the liquid blanket technologies. In FY2018, thermofluid behaviors of a liquid metal flow were investigated in collaborative experiments with Kyoto University under a strong magnetic field of 3 T provided by the superconducting magnet of the Oroshhi-2 system (Fig. 1). A flow of GaInSn with a free surface was made by an electromagnetic circulation pump in the channel. A magnetic field was applied perpendicularly to the flow. At the bottom of the channel, obstacles are attached as a vortex generator to enhance the heat transfer from the heater set on the free surface to the bottom of the channel. In the experiments, the world's first data of three-dimensional temperature distribution in a liquid metal flow under a strong magnetic field has successfully been acquired using a thermocouple array. The distribution data indicates that the heat transfer from the heater to the bottom becomes nearly parallel to the strong magnetic field, which is perpendicular to the GaInSn flow. The experiments are offering new findings also for the development of liquid divertors.



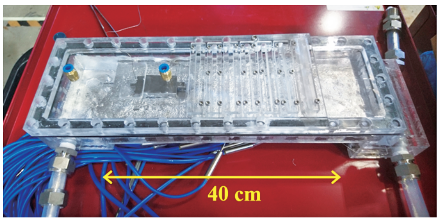


Fig. 1 (left) A schematic drawing and (right) a photo of the experimental device to investigate thermofluid behaviors of a liquid metal flow (GaInSn) under a strong magnetic field of 3 T provided by the superconducting magnet in Oroshhi-2.

Experimental topics to be examined in Oroshhi-2, validities of the experimental methods, schedule adjustment, etc., are regularly discussed in the Oroshhi workshop held at NIFS twice a year. In FY2019, it was decided that the heat removal performance of a molten-salt coolant flow would be examined by a collaborative experiment with Tohoku University through a pebble-bed tube under a 3 T magnetic field. Since the possibility of degradation of the heat removal performance due to suppression of turbulence flow under a strong magnetic field has been pointed out in previous studies, this experiment will be essential for the design of molten-salt-cooled blanket systems. It was also decided that demonstration of continuous and high-efficient hydrogen recovery from circulated LiPb coolant would be performed using the sub-test section of the LiPb loop in FY2019, as the LHD-Project Research Collaboration (see later section).

Other than the research activities using Oroshhi-2, studies on tritium fuel behaviors in high temperature (~500°C) FLiNaK and FLiNaBe tritium breeder/coolants have been performed under a collaboration with Osaka University from FY2017 to FY2018. Low flux neuron irradiation experiments on FLiNaK and FLiNaBe have been performed using an AmBe neutron source at Osaka University, and the world's first tritium release data for both the breeder and coolants has been successfully obtained. Controllability of the chemical forms of released tritium and the tritium release rates by changing the atmospheric condition have also been confirmed [1].

[1] K. Kumagai, Ph.D. thesis, SOKENDAI (2019), in Japanese.

(T. Tanaka)

Highlight

Development of advanced structural materials for fusion reactor blanket

NIFS promotes development of advanced structural materials such as reduced-activation ferritic steels, their oxide-dispersion-strengthened (ODS) alloys, low-activation vanadium alloys, etc., under the collaboration with universities and institutions in the world. ODS steels exhibit excellent high temperature strength and have been considered to extend the maximum operation temperature of blankets to be higher than 600°C, based on hardening by nanometer-size oxide particles dispersed in the alloy matrix. These alloys are made from metal and oxide powders under a high temperature condition, such as 1,150°C, and shaped into plate products. During the hot shaping processes, crystal grains of the alloy matrix are elongated to the preferential crystal orientation. As a result, ODS steels obtain anisotropic mechanical properties, where they lose ductility for tensile deformation if it is vertical to the longitudinal direction of the crystal grains. In order to suppress the anisotropy and improve ductility, recrystallization is required to replace the elongated grains by new ones with less preferential orientation. However, the conventional ODS steels could not complete recrystallization until heating up to 1,400°C, resulting in strength degradation by porosity formation and oxide particle coarsening. Our new fabrication process, two-way cold rolling, effectively introduces crystal defects and their stored energy, and hence induces driving force for recrystallization at even lower temperature, 1,100°C. Figure 1 shows the grain microstructures before and after the two-way rolling, indicating the crystal orientations by colors. The figure proves that the color, the orientation of the crystal, has been completely changed by the new rolling process. At this moment, we have confirmed full recrystallization of the ODS steel after the new rolling process, and we will evaluate high temperature strength and ductility in all directions as a next step. NIFS also leads the development of advanced vanadium alloys which are alternative to the ferritic steels and can raise the operation temperature of blankets up to 700°C. Reduction and optimization of the concentration of the alloying element titanium has been examined to improve their low-activation properties further, therefore, early material recycling is expected after the use in fusion reactors.

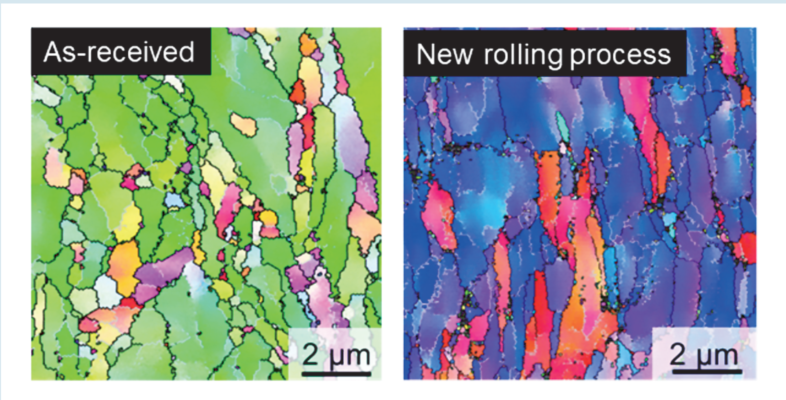


Fig. 1 Images of grain microstructure by electron back scattered diffraction analysis. Black lines indicate the crystal grain boundaries. Red, blue, and green colors are attributed to the crystal orientations of <001>, <111>, and <110> for body centered cubic lattice, respectively.

(T. Nagasaka)

Research and Development on the Divertor

The advanced brazing technique between ODS-Cu (GlidCop®) and tungsten (W) with BNi-6 (Ni-11%P) filler material has been developed for fabricating a divertor heat removal component [1, 2], and optimization of the procedures were greatly progressed in the last fiscal year [3]. By further enhancing the optimized advanced brazing technique, we newly developed a joint of ODS-Cu/ODS-Cu and SUS/ODS-Cu [4]. This newly developed joint has two special features as follows. The first is that these joints have leak tightness against fluids. The second is that the multiple brazing heat treatment can be applicable for fabricating a single divertor component because the prior bonding layer is not affected by the subsequent brazing heat treatment. A small-scale divertor mock-up with a curved cooling flow path channel has been successfully fabricated using the multi-step brazing technique of ODS-Cu/ODS-Cu, SUS/ODS-Cu and W/ODS-Cu, as shown in Fig. 1.

A high heat flux test facility named ACT2 (Active Cooling Test stand 2) has been used for the heat load test of plasma facing components since 2015. ACT2 is capable of performing a reactor relevant heat flux test (> 20 MW/m^2) on a large surface area of >1,000 mm^2. A divertor test component with W-flat tiles brazed on a Cu-alloy heatsink (Fig. 2) was tested in ACT2 at a water-cooled condition (flow speed: 16.8 m/s, pressure: ~0.3 MPa). In this heat load test, a water-cooled copper plate with an aperture called the "beam limiter" was mounted on a test component to maintain the homogeneous heat load and to protect the thermocouples attached to specific points of the test component. The heat load test was successfully carried out without damaging the test component and a good heat removal capability was observed up to 24 MW/m^2. The temperatures measured on the test component with respect to the heat flux are shown in Fig. 2. In this test, the maximum heat flux of 24 MW/m^2 was limited by the capacity of the beam limiter.

New materials of Dispersion Strengthening Copper (DS-Cu) are being developed by means of a combined process of Mechanical Alloying (MA) and Hot Isostatic Pressing (HIP). The DS method can drastically improve the mechanical properties of materials by introducing thermally stable nano-particles (Y_2O_3, for example). Recently, a new process has been developed, in which the dopant materials (CuO) are added to the matrix in the middle of the MA process (Fig. 3). The new process successfully increased the density of the oxide nano-particles. The mechanical properties of the Cu-Y_2O_3 fabricated by the new process will be investigated in future work.

A superior nano-scale fabrication technique of tungsten by using a focused ion beam – electron beam

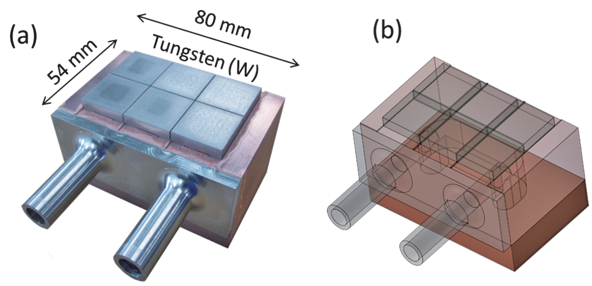


Fig. 1 (a) Photo and (b) a CAD image of a small-scale divertor mock-up with a curved cooling flow path channel fabricated by the multi-step brazing technique.

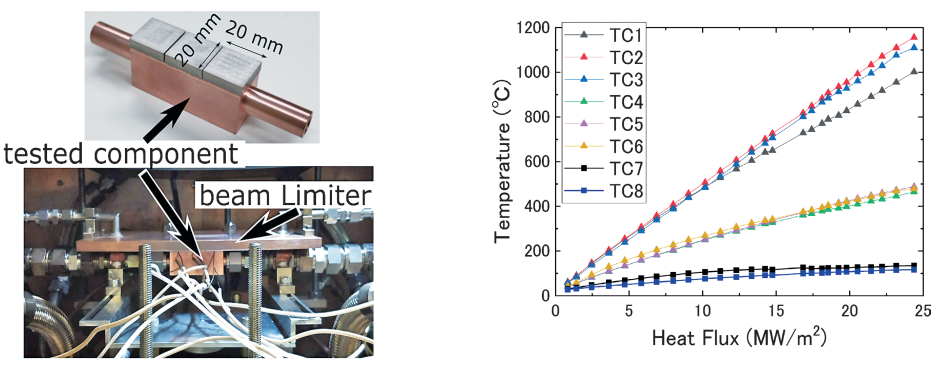


Fig. 2 (left) Photos of a tested component and the experimental setup inside ACT2, and (right) the experimental results of the measured temperature of the tested component with respect to the heat flux.

(FIB-SEM) device has been developed, and is named the "nano-scale sculpture technique" [5]. In order to observe the internal structure of nano-scale materials, the Transmission Electron Microscope (TEM) is commonly used as a powerful tool. In TEM, accelerated electrons are transmitted through target materials. To observe a cross-sectional view of the very close region to the top surface of a tungsten sample by TEM, a small piece of cross-sectional sample is extracted from the surface at first. Then, the extracted small piece is modified to an ultra-thin film sample with a thickness of <~ 100 nm. The FIB-SEM device enables a nano-scale material fabrication by using a focused Ga ion beam. Adjustment of the Ga beam intensity is not an easy procedure, because tungsten is a quite hard material. The very close region to the top surface of the tungsten sample could be lost undesirably, even with a beam intensity slightly stronger than the appropriate value. To solve this problem, we tried to make a special Ga beam operation to maintain the top surface in the ~100 nm level thin film fabrication. Consequently, we succeeded in fabricating an ultra-thin film with a thickness of ~100 nm, or less, while maintaining the top surface of the tungsten sample. Figure 4 shows the details of this "nano-scale sculpture technique." By observing the ultra-thin film using TEM, it becomes possible to clearly identify the atomic level damages formed in the very close region to the top surface of the tungsten sample with high-resolution, as shown in Fig. 5 [6].

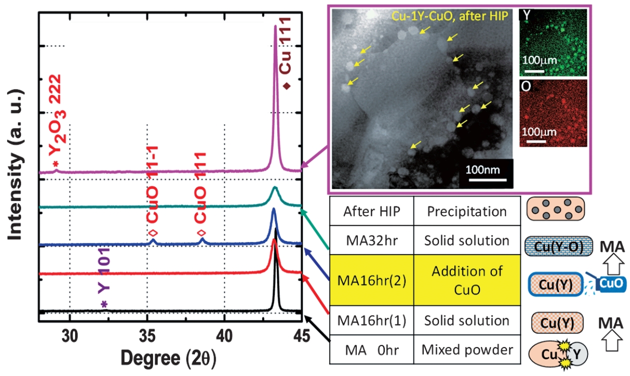


Fig. 3 In-situ fabrication of Cu-Y_2O_3 alloy.

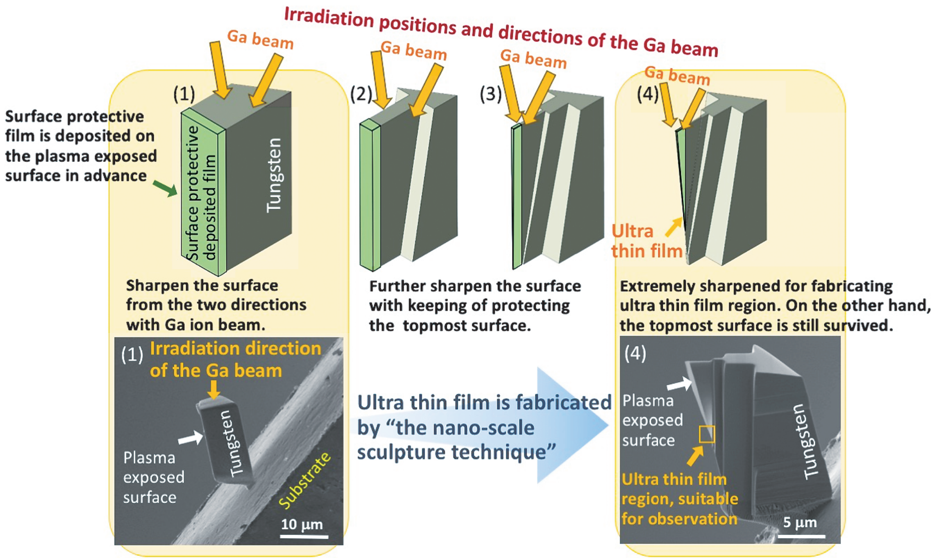


Fig. 4 The "nano-scale sculpture technique" by using a focused ion beam – electron beam (FIB-SEM) device. The upper series shows a schematic view of the fabrication process using a Ga ion beam. The images of (1) and (4) in the bottom series are the corresponding scanning electron microscope (SEM) images to those of (1) and (4) in the upper series.

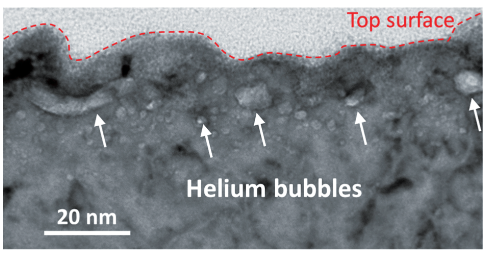


Fig. 5 A cross-sectional TEM image of a tungsten sample exposed to the helium divertor plasma in LHD. The ultra-thin film TEM sample was fabricated by "the nano-scale sculpture technique." This very high-resolution image could not be obtained without the ultra-thin film fabricated by the nano-scale sculpture technique. The bright spherical shape images are helium bubbles. The top surface of the tungsten sample is retained even in an ultra-thin film with a thickness of ~100 nm or less [11].

In-vessel material transport on the plasma-facing walls in LHD has been investigated using the Rutherford backscattering spectrometry (RBS) with 2.8 MeV ^{4}He ions generated by a tandem accelerator (NEC Pelletron accelerator, Fig. 6). Ten Si plates had been placed on the outer side of the first-wall surface for every 36° toroidal sections in LHD, during one experimental campaign. A characteristic RBS spectrum of the mixed-material layers deposited on one of the Si plates is shown in Fig. 6 [7]. The RBS spectrum shows that Cr, Fe, Ni, Ti, Mo, and W are included in the mixed-material layer deposited on the Si plate. Various RBS spectra have been observed for each of the ten Si plates. A detailed data analysis on those spectra is now being carried out. The RBS analysis is a powerful tool for elucidating the material transport (impurity transport) inside the LHD vacuum vessel from the viewpoint of material science.

Spectroscopic measurement of the light emitted from highly charged tungsten ions has been conducted using an electron beam ion trap (CoBIT), and the data is used to identify the emission lines observed in the LHD plasma experiment (Fig. 7). This time, for the first time in the world, we succeeded in observing a strong forbidden transition called the “Electrical octupole transition (E3)” by using CoBIT. This is a phenomenon that occurs with a probability of about 10 billionth of the normally allowed transition (E1 transition), and we succeeded in elucidating its physics mechanism.

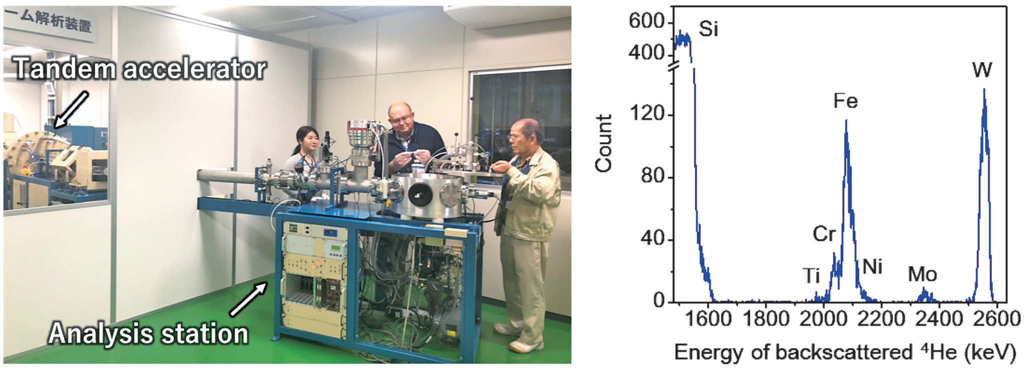


Fig. 6 (left) The ion beam analysis device consisting of an ion beam injector, a tandem accelerator, and an analysis station, and (right) an example of the Rutherford backscattering spectrometry (RBS) spectrum of the mixed-material layers deposited on a Si plate placed inside the LHD vacuum vessel during one experimental campaign.

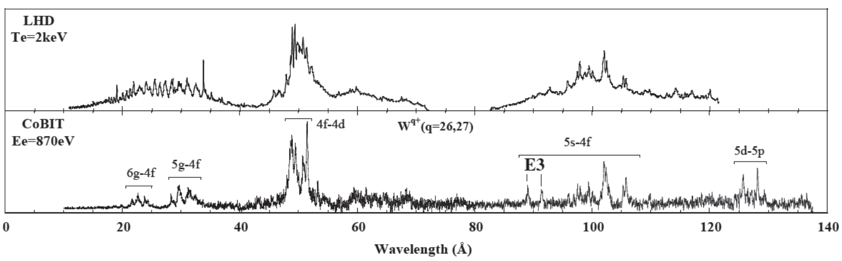


Fig. 7 Comparison of the spectrum observed in LHD with the spectrum observed in CoBIT.

[1] M. Tokitani *et al.*, Plasma Fusion Res. **10**, 340503 (2015).
[2] M. Tokitani *et al.*, Nucl. Fusion **57**, 076009 (2017).
[3] M. Tokitani *et al.*, Fusion Eng. Des. (2019), in press.
[4] M. Tokitani *et al.*, Fusion Eng. Des. (2019), to be published.
[5] D. Nagata *et al.*, Materia, in Japanese. https://doi.org/10.2320/materia.57.604
[6] M. Tokitani *et al.*, Nucl. Mater. Energy **12**, 1358 (2017).
[7] V.Kh. Alimov *et al.*, Fusion Eng. Des. **147**, 111228 (2019).

(M. Tokitani, Y. Hamaji, H. Noto, M. Yajima and H. Sakaue)

LHD-Project Research Collaboration

The LHD Project Research Collaboration program has been contributing to enhancing both the scientific and the technological foundations for the research related with the LHD project as well as the future helical fusion reactors. The characteristics of this collaboration program are that the researches are performed at universities and/or institutions outside NIFS. In the research area of fusion engineering, the following ten subjects were approved and conducted in FY2018:

1. Knowledge and technology transfer from IFMIF-EVEDA accomplishment to systemization of liquid blanket research
2. Development of irradiation-resistant NDS-Cu alloys for helical reactor divertor
3. Development of effective heat removal method from liquid metal free-surface with local heating under strong magnetic field and its demonstration by Oroshhi-2
4. Engineering study on lithium isotope enrichment by ion exchange
5. Tritium behavior in a secondary cooling system in a fusion DEMO reactor
6. Establishment of high susceptible detection assay for biomolecule response and estimation of the biological effects of low-level tritium radiation by utilizing its assay
7. Development of highly-ductile tungsten composite systems
8. Fundamental engineering of tritium recovery process for liquid blanket of helical reactor
9. Development of new rapid-heating and quench processed Nb_3Al large-scaled cables for the helical winding due to the react-and-wind method
10. Field estimation for improvement of environmental tritium behavior model

From the above ten research items, two of them (7 and 8) are briefly described below:

7. Development of highly ductile tungsten composite systems

Tungsten is promising as a plasma facing armor material, because of its high melting point, low plasma erosion rate and low permeation loss of the fuel hydrogens. However, the tungsten armor exhibits disadvantageous ductility loss under the harsh fusion environment. Conventional tungsten reveals the ductility loss at 1,100°C and above due to recrystallization (crystal co arsening), and below 600°C due to their intrinsic brittleness. On the other hand, a potassium (K)-doped advanced tungsten is resistant to the recrystallization and shows no ductility loss up to 1,250°C. In order for compatible low-temperature ductility with the excellent high-temperature ductility, the present project developed layered novel composite systems of the K-doped tungsten and ductile other metals. As shown in Fig. 1, the K-doped tungsten and pure vanadium layer composite successfully maintain ductility at room temperature, even after an exposure to heat loading at 1,250°C.

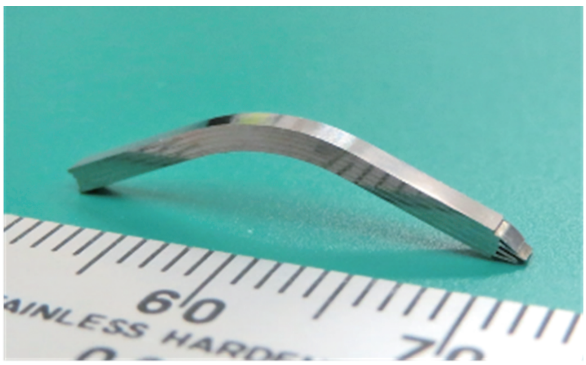


Fig. 1 K-doped tungsten and pure vanadium layers composite after a bend test at room temperature. The visible bending and no fracture prove large ductility.

8. Fundamental engineering of tritium recovery process for liquid blanket of helical reactor

High-efficient hydrogen fuel recovery from high temperature liquid fuel breeder is one of the key technologies required for fusion blanket systems. An innovative idea to recover hydrogen from circulated liquid LiPb breeder has been proposed in Kyoto University and experimental validation is being conducted. In the proposed vacuum sieve tray (VST) system, droplets of liquid LiPb at ~350°C are made with a sieve tray in a vacuum chamber and hydrogen in the droplets is efficiently extracted to vacuum before reaching the bottom (Fig. 2). Stand-alone tests of the function and performance have successfully been accomplished in Kyoto University and the system has been installed to the Oroshhi-2 heat and mass transfer loop at NIFS (Fig. 3). The first demonstration of continuous hydrogen recovery from circulated high temperature LiPb by the VST system will be performed in the summer of 2019.

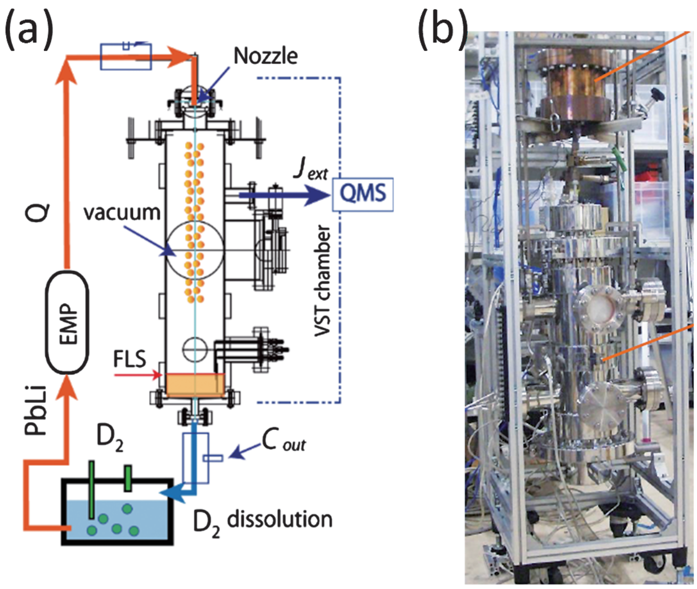


Fig. 2 (a) Schematic drawing of the vacuum sieve tray (VST) chamber for hydrogen recovery. (b) Photo of the VST chamber tested in Kyoto University.

Fig. 3 Expansion of an experimental stage on Oroshhi-2 for the validation experiment of continuous hydrogen recovery from circulated LiPb by the VST technique.

(J. Miyazawa, T. Nagasaka and T. Tanaka)