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. Cooperation with the Large Helical Device Project, the Numerical Simulation Reactor Research Project, and the Task Force for Next Research Project are also promoted.

(T. Muroga)

Design Studies for The Helical Fusion Reactor

Regarding the fusion reactor design study, the step-by-step strategy towards the helical fusion power plant FFHR-d1 has been proposed. Currently, three intermediate step devices has been considered before the construction of FFHR-d1: FFHR-c1 (experimental/prototype reactor for the demonstration of steady-state operation of the power plant system, FFHR-b1 (volumetric neutron source for the early utilization of fusion fast neutrons) and FFHR-a1 (non-nuclear system for the examination of improved magnetic configuration and advanced engineering concepts). In this fiscal year, reexamination of the primary design parameters of FFHR-b1 has been conducted. The major radius was enlarged from 3.9 m (just the same as LHD) to 5.46 m (1.4 times larger than LHD) to increase the neutron output and to ensure sufficient thickness of the neutron shielding blanket. The helical coil current density and the maximum field on the helical coil are decreased, resulting in the relaxation of the engineering design requirement. In the last fiscal year, the pitch modulation parameter of the helical coil α was changed from 0.1 to 0.0 to achieve simultaneous improvement in the MHD stability and energy confinement property [1]. This improvement effect has been confirmed through the analysis using the finite-beta MHD equilibrium data obtained by the HINT code (Figs. 1 and 2). According to this analysis, the helical pitch modulation of $\alpha = 0.0$ has been selected for FFHR-b1. In response to these changes in the design parameter, the new design of the volumetric fusion neutron source has been defined as FFHR-b2. The related works on the engineering design concepts have also been advanced. The three-dimensional shape of the blanket modules has been reexamined and the replacement of the modules by a robotic arm has been examined using a 3-D printing model. Regarding the advanced divertor system using a pebble flow and the self-cooled liquid breeder blanket system, adoption of new materials for the improvement of thermal efficiency has been proposed. Regarding the high-temperature superconducting magnet system, improvement of the cooling efficiency using cryogenic coolants other than helium gas is also considered. Resolution of the design issues and enhancement of the design attractiveness are expected by advancing the examination of these new concepts.

[1] T. Goto *et al.*, Nucl. Fusion **59**, 076030 (2019).

(T. Goto)

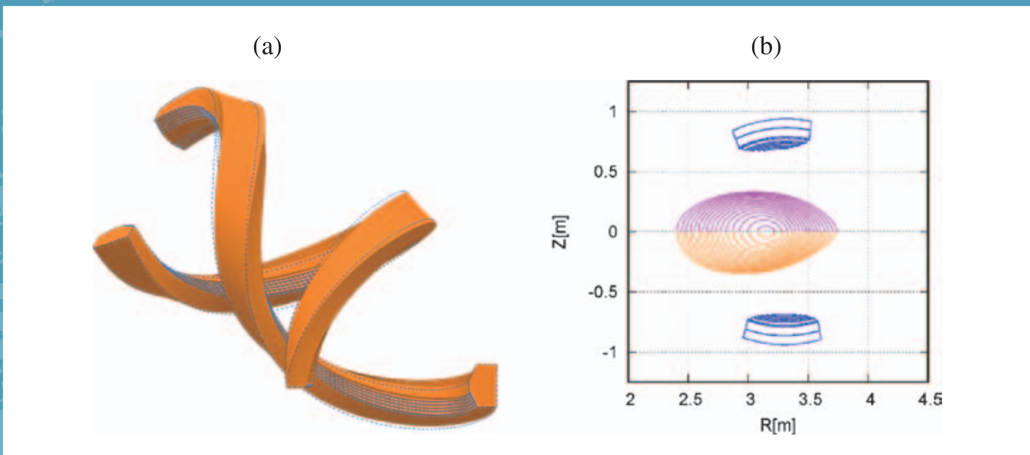

Fig. 1 (a) Variation of the helical winding path with a helical pitch modulation parameter α of 0.1 (blue dotted-curves) and 0.0 (orange block). (b) Cross-sectional shape of the magnetic surfaces (calculated by the HINT code with a peak beta value of 3%) and helical coils at the horizontally-elongated plasma cross-section with a helical pitch modulation parameter α of 0.1 (upper) and 0.0 (lower).

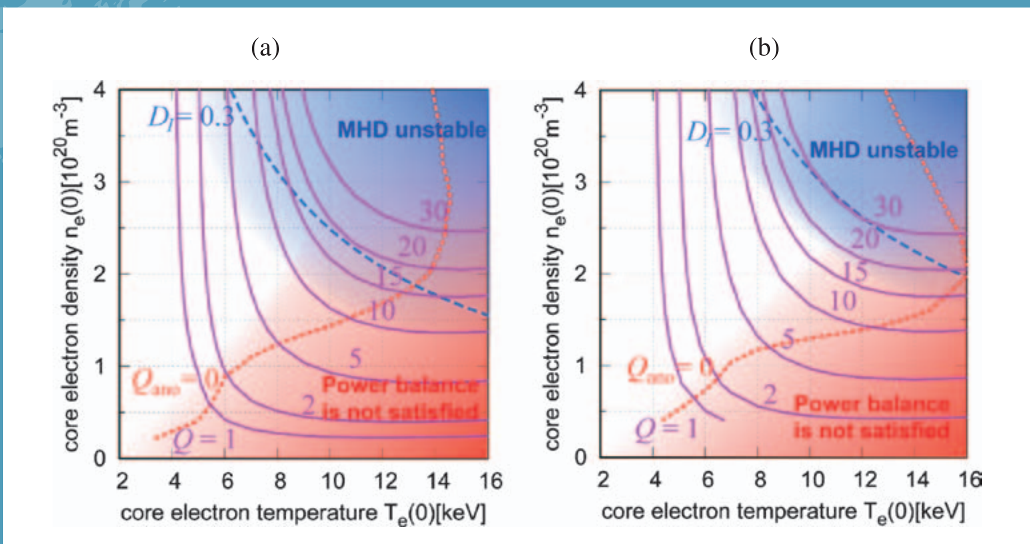

Fig. 2 Comparison of the achievable operation region (the white area without shades) in the diagram of core electron temperature and core electron density for cases of (a) α = 0.1 and (b) 0.0.

Highlight

Enhancement of Link between Design and R&D

Visualization of the link between the critical issues in the helical fusion reactor design and the ongoing R&Ds has been conducted (Fig. 12). All R&Ds has been sorted by research area and “status bars” theme that specify the goal, the member, the execution plan, the current progress and the related budget for each R&D have been made. Regarding the design issue, 22 items have been identified and the “design issue flip boards” that describe the related design parameters assumed in the design study of FFHR-c1 and FFHR-b2 and the necessary knowledge to solve the issues has been made. These status bars and design issue flip boards are linked each other using the reference numbers. The status bars will be continually updated and actively utilized as a research communication tool for the creation of new research field based on the wide-ranged research of the fusion engineering as well as the acceleration of the reactor design activity and the extension of the R&Ds including the cooperative research.

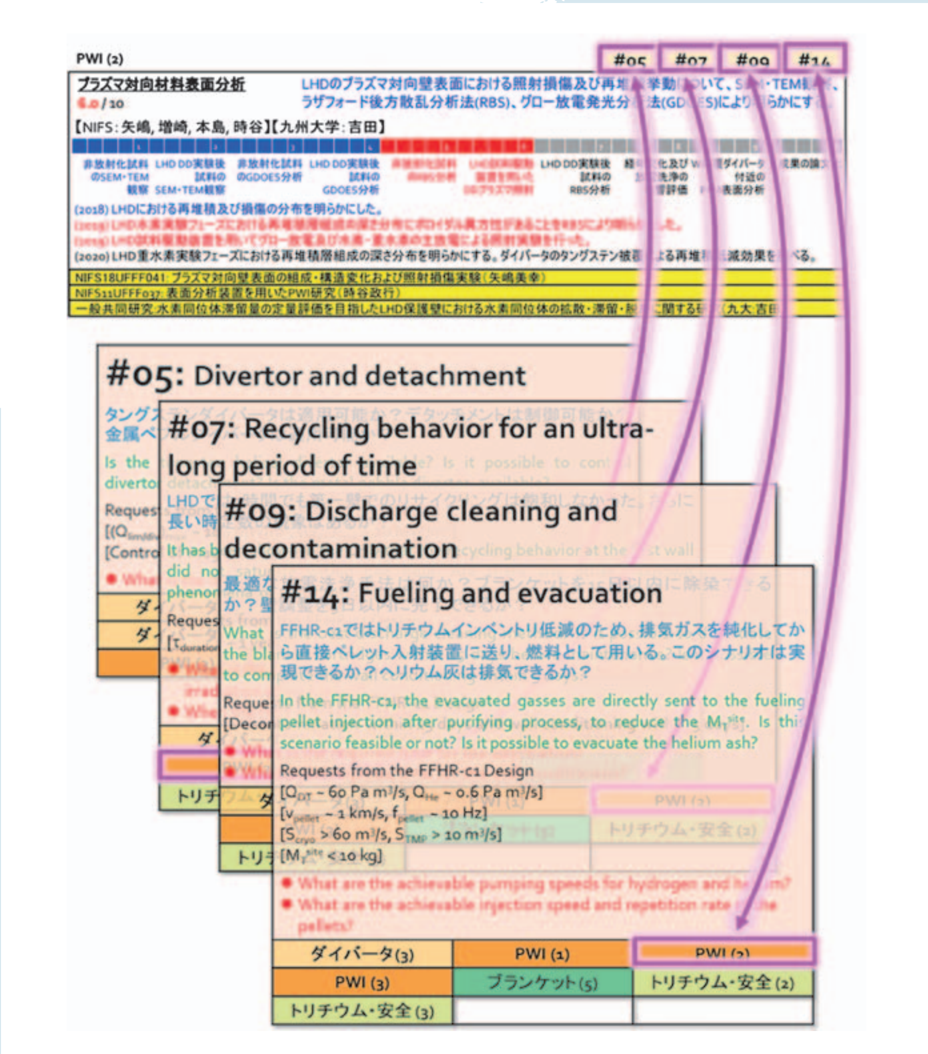

Fig. 12 Example of the visualization of the link between design issues and R&Ds. The design issue flip boards (upper) and the R&D status bars (lower) are linked each other.

(T. Goto and J. Miyazawa)

Highlight

Topology Optimization of the Coil Supporting Structure

Electromagnetic forces generated by superconducting coils in fusion reactors reaches the order of several tens of MN/m. The supporting structure needs to be strong enough to support this huge force. The conventional design of the supporting structure for the FFHR-c1, the helical fusion reactor aiming to realize steady electrical self-sufficiency, assumes a stainless steel 316LN plate with a basic thickness of 200 mm. The total weight of the supporting structure is estimated to exceed 7,800 tons in this case. In the present study, the "topology optimization", which enables innovative designs that overturn conventional design common sense, was applied to the design of the helical fusion reactor to reduce the heavy amount of the supporting structure. Consequently, unnecessary regions were removed, and total weight was significantly decreased as shown in Fig. 13. From the verification analysis applied for the optimized model, it is confirmed that the stress is in an acceptable level. Furthermore, as a derivative of this research, an effort in structural design of coil supporting structure can be reduced. The topology optimization can be applied to any original shape. For instance, assume that the block covers the entire coil but excludes the space occupied by in-vessel components. The shape obtained by topology optimization is close to the assumed toroidal shape as shown in Fig. 14. By preparing an arbitrary block with the necessary access ports open, it becomes possible to conduct design almost automatically.
This research result was published as H. Tamura *et al.*, J. Phys.: Conf. Ser. 1559, 012108 (2020).

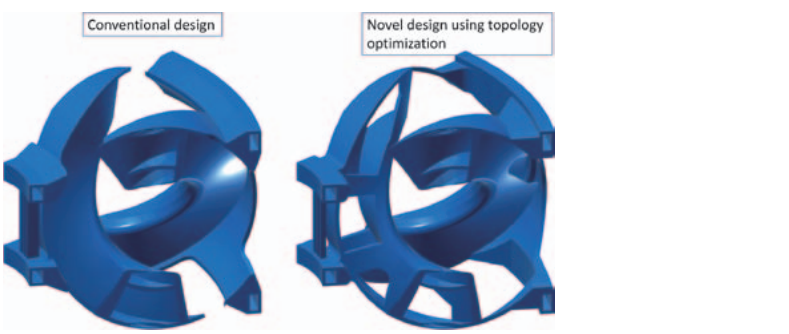

Fig. 13 The shape of the coil support structure obtained by applying the topology optimization (right figure). The figure on the left is based on the conventional design method. Arranging 10 of these structures in the circumferential direction gives the torus-like overall shape. Weight reduction from the conventional design is approximately 2,000 tons.

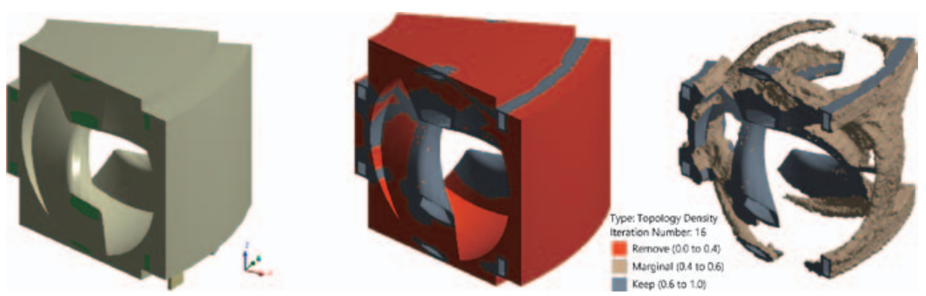

Fig. 14 Example of a design using topology optimization. Left figure is an initial setting, and right one is the optimization result. Red color in the middle figure is the removed region.

(H. Tamura)

Research and Development on the Blanket

The FLiNaK/LiPb twin loop system Oroshhi-2 (Operational Recovery Of Separated Hydrogen and Heat Inquiry-2) is being operated at NIFS as a collaboration platform for integrated experiments on the liquid blanket technologies. In FY2019, two new testing sections have been installed in Oroshhi-2: (i) a test chamber for demonstration of continuous and high efficiency hydrogen isotope recovery from LiPb by the vacuum sieve tray (VST) technique and (ii) a test section for evaluation of heat removal performance of a molten salt flow under an intense magnetic field.

The demonstration of the hydrogen isotope recovery is being conducted by a collaboration with Kyoto University under the LHD-Project Research Collaboration program. In the VST method proposed by the Kyoto University group [1], small droplets of liquid LiPb are made in 0.6 mm diameter nozzles in a vacuum chamber (Fig. 3). Hydrogen isotopes contained in LiPb are quickly released from surfaces of the droplet into vacuum. Circulation of LiPb through the VST chamber has already been confirmed in the test experiment. The world's first demonstration of continuous and high-efficiency hydrogen isotope recovery from LiPb will be performed with the VST chamber in FY2020.

In the design studies of self-cooled molten-salt blanket systems for helical reactors, acquisition of experimental

(a) (b)

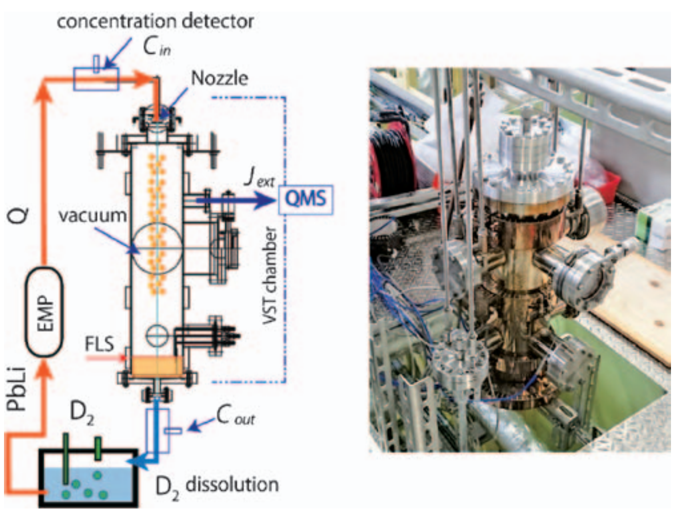

Fig. 3 (a) A schematic drawing of the vacuum sieve tray (VST) technique for continuous and high efficiency hydrogen isotope recovery from LiPb [1], and (b) a photo of the VST chamber installed in Oroshhi-2 in FY2019.

(a) (b)

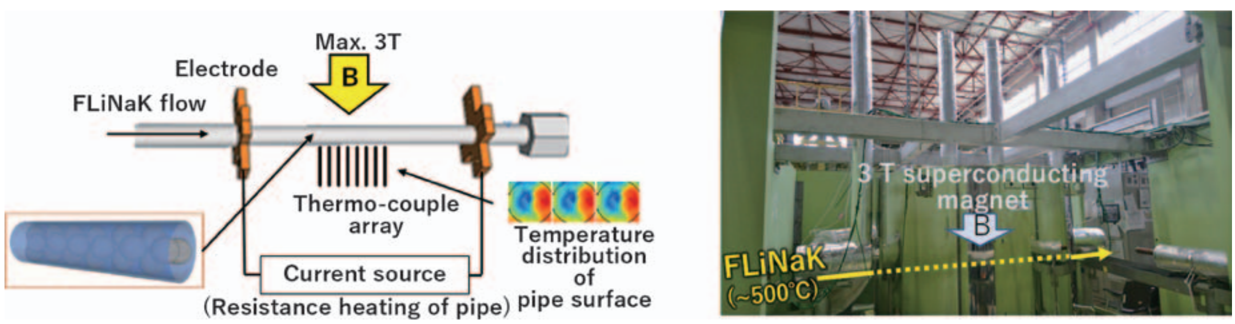

Fig. 4 (a) Schematic drawing of the vacuum sieve tray (VST) technique for continuous and high efficiency hydrogen isotope recovery from LiPb. (b) Photo of the VST chamber installed in Oroshhi-2 in FY2019.

data on heat removal performances of the molten salt coolant flows under intense magnetic field is being required over many years. This is because previous simulation studies indicate that turbulent flow of molten-salts would be suppressed under intense magnetic field and the heat removal performance would be degraded [2]. The test section to evaluate the heat removal performance has been designed and installed at the 3-T superconducting magnet section of the FLiNak loop by a collaboration with Tohoku University in FY2019 (LHD-Project Research Collaboration). Temperature distribution of the surface of the Inconel tube penetrating the magnet body is measured using thermo-couple arrays (Fig. 4). From the changes in the temperature distribution, changes in the heat removal performances due to the applied magnetic field can be evaluated. The FLiNaK circulation and acquisition of data will be started in FY2020 and this will be the world's first evaluation of heat removal performances of molten-salts under intense magnetic field.

[1] F. Okino *et al.*, Fusion Eng. Des. **146**, 898 (2019).
[2] S. Satake *et al.*, Fusion Eng. Des.i **81**, 367 (2006).

(T. Tanaka, Y. Hamaji)

Development of advanced structural materials for fusion reactor blanket

For vanadium alloys, which are leading the world's development of low-activation blanket structural materials, various investigations for dissimilar-metals bonding technologies are ongoing to connect the blankets with the out-vessel components. While the direct melt-welding between the low-activation vanadium alloy NIFS-HEAT-2 and the nickel-based Hastelloy-X alloy is known to be impossible, the casual weld cracking due to intermetallics precipitation has been successfully suppressed by the non-melt explosive welding process under collaboration studies with Kumamoto University, Qilu University of Technology in China, and others. It is indicated that a high-entropy solid solution, which is attractive as a strong and radiation-resistant material, possibly forms in the alloys of weld mixture (Fig. 5). Further investigations will prove whether its solid-solution state can be stable or not during the long-term operation at high temperature.

[1] S. N. Jiang et al., J. Nucl. Mater. 539 (2020) 152322.

(T. Nagasaka)

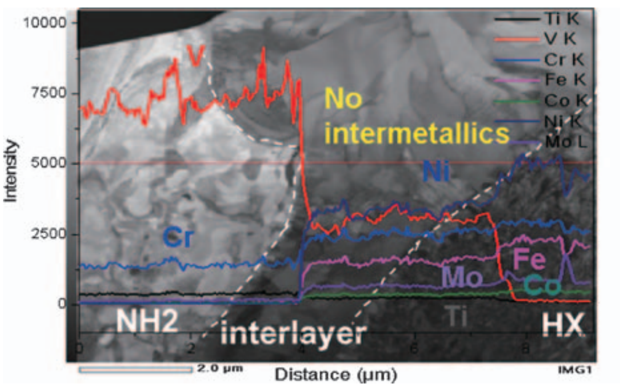

Fig. 5 The transmission-electron microscope image at the interlayer produced by the explosive-weld mixture between the low-activation vanadium alloy NIFS-HEAT-2 (NH2) and the nickel-based Hastelloy-X alloy (HX). Neither intermetallics particles nor homogeneous element distribution are observed in the multi-component interlayer, which is considered to prove formation of high-entropy solid solution [1].

Research and Development on the Divertor

The advanced brazing technique between oxide-dispersion-strengthened copper (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 between ODS-Cu and ODS-Cu (ODS-Cu/ODS-Cu), and between stainless steel (SS) and ODS-Cu (SS/ODS-Cu) [4,5]. This newly developed joint has two special features. 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. The superior fabrication procedures for divertor heat removal component, "Advanced Multi-Step Brazing (AMSB)", was newly developed by applying these special features. The prototype AMSB component with the rectangle shaped cooling flow path and the "V-shaped staggered rib" structure has been successfully produced as shown in Fig. 6, in which a pre-processed rectangle shaped cooling flow path in the ODS-Cu heat sink is sealed by a SS lid with a leak tight condition.

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 loading test (>20 MW/m^2) on a large surface area of >1,000 mm^2. Several types of the produced AMSB components have been tested in ACT2. For a representative heat loading experiment, a steady state heat loading test was carried out up to the heat flux of ~30 MW/m^2 at a water-cooled condition (flow rate: 15 L/min, pressure: ~0.5 MPa). The heat loading area is depicted in the photograph of Fig. 6 (a). The thermocouples of Channel 4 to 6 were embedded in the positions shown in Fig. 6 (b). Fig. 7 shows the temperature dependence of CH 4 to 6 during a heat loading. The temperature increase during the heat loading of up to ~30 MW/m^2 is acceptable from the viewpoint of structural reliability. The AMSB component shows an extremely high heat removal capability under the reactor relevant condition.

New oxide dispersion strengthened tungsten (DS-W), which applies mechanical alloying (MA)-hot isostatic pressing (HIP) process shown in Fig. 8 (a), is being studied for the development of an advanced plasma facing material (PFM). The initial materials of W and titanium carbide are alloyed using a planetary-type ball mill, consisting of balls of 1.6-mm and 3.0-mm diameter. The mechanically alloyed powders are then pre-sintered. The pre-sintered W alloys are then sintered by HIP at 1,750 degrees Celsius for 1.5 hrs with a pressure of 186 MPa. The HIPed-materials made by this process were evaluated for their mechanical properties at high temperature. In Fig. 8 (b), it is shown that a DS-W after MA using 3.0-mm-diameter balls maintained the hardness and the grain size after annealing temperature of up to 1,900 degrees Celsius, whereas the hardness of a DS-W after MA using 1.6-mm-diameter balls decreased with an increase of the grain size. This result exhibits that the alloying process of DS-W affects the thermal properties after sintering.

The electron beam ion trap (CoBIT) is used for conducting spectroscopy studies of highly-charged ions to provide data necessary for the analysis of fusion plasma emission spectra. In the previous study, we successfullyobserved strong forbidden transitions, with a probability of about 10 billion times smaller than that for the allowed transitions, which is called the "Electric octupole transition (E3)" from highly-charged tungsten ions for the first time in the world [1]. In FY2019, we predicted that the same phenomenon may occur for rhenium multiply-charged ions with an isoelectronic series, and successfully observed it. The spectrum is shown in Fig. 9.

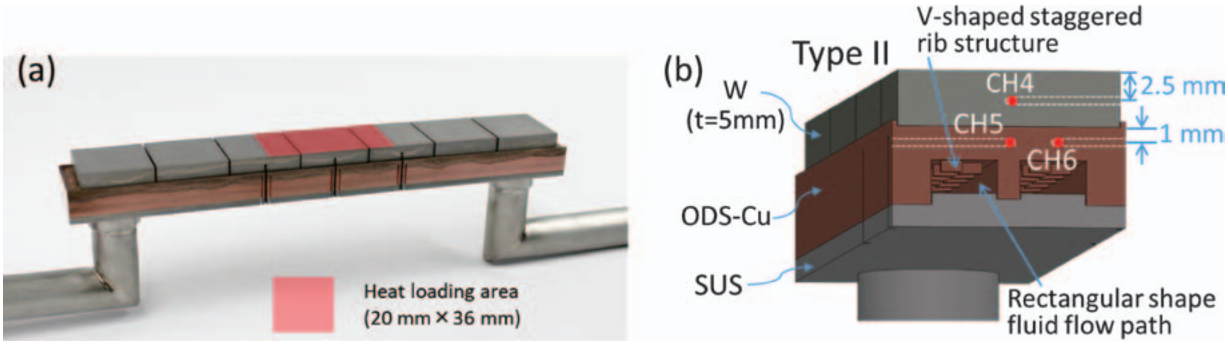

Fig. 6 (a) Photograph of the AMSB divertor heat removal component with the heat loading area for the ACT2 experiment. (b) Schematic cross-sectional view of the AMSB divertor heat removal component at the central region of heat loading area of (a).

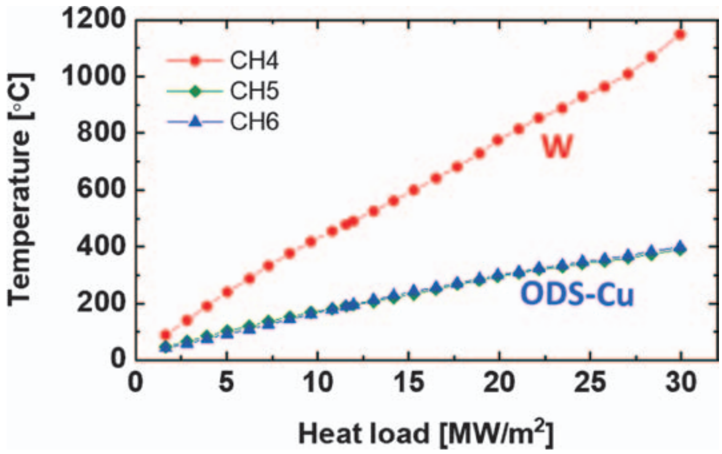

Fig. 7 Temperature dependence of the embedded thermocouples (CH4 to 6) under a steady state heat loading condition in ACT2. The positions of the thermocouples from CH 4 to 6 correspond to the red marker points in Fig. 6 (b).

(a) (b)

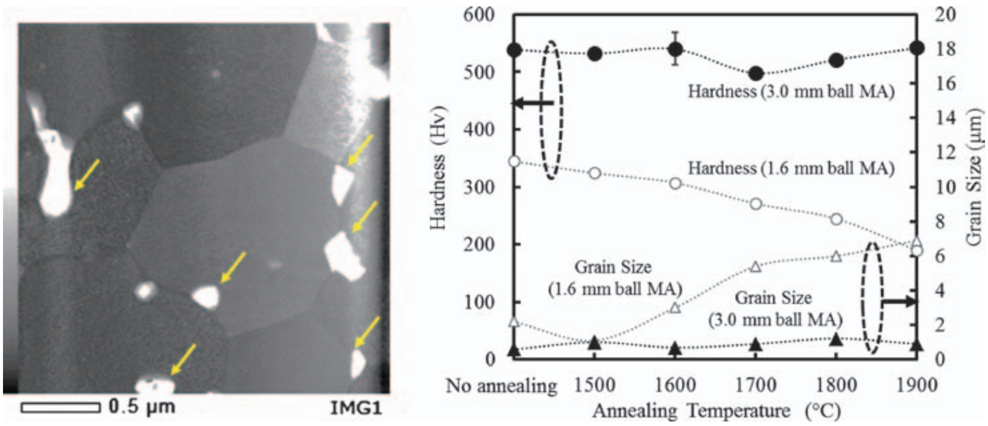

Fig. 8 (a) Transmission Electron Microscope (TEM) image of nano-titanium oxide at grain boundaries after annealing at 1800°C. (b) Relationship between hardness and grain size of DS-W samples made by the MA and HIP process.

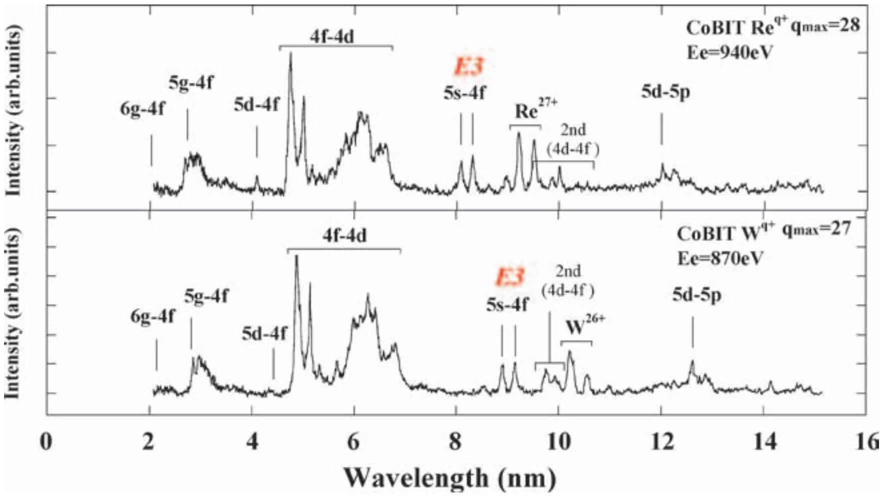

Fig. 9 Extreme ultraviolet spectra of tungsten and rhenium multiply-charged ions observed in CoBIT.

[1] M. Tokitani *et al.*, Plasma Fusion Res. **10**, (2015) 340503.
[2] M. Tokitani, *et al.*, Nucl. Fusion **57**, (2017) 076009.
[3] M. Tokitani *et al.*, Fusion Eng. Des. **146**, (2019) 1733-1736.
[4] M. Tokitani *et al.*, Fusion Eng. Des. **148**, (2019) 111274.
[5] M. Tokitani *et al.*, J. Nucl. Mater. (2020) in press, doi.org/10.1016/j.jnucmat.2020.152264.
[6] H. A. Sakaue *et al.*, Physical Review A100, 052515 (2019)

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

Research and Development on the Superconducting Magnet

For the helical fusion reactor, FFHR, High-Temperature Superconductor (HTS) is considered to apply to the magnet system. Various types of large-current capacity HTS conductors are being designed and developed combining the second generation REBCO HTS tapes. For this purpose, an international collaboration experiment was carried out between Massachusetts Institute of Technology (MIT) in US and NIFS for testing the Twisted Stacked-Tape Cable (TSTC) HTS conductor. 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. 10), which is equipped with a large-bore (700 mm), high magnetic field (13 T), large sample current (50 kA), and temperature control capability (4.2–50 K). The experiment was carried out two times in FY2018. On this occasion, self-field measurements of the TSTC conductor were conducted using Hall sensors, and the detailed analysis was completed in FY2019. Based on the measurement results, the current distribution of the TSTC conductor was analyzed using analytical models. The calculation results indicate that the current distribution of the TSTC is uniform when the operating current is maintained at 10 kA and the temperature is controlled at 34 K. On the other hand, the current distribution is not uniform at the charging and discharging phases with the ramp rate of 50 A/s. Additionally, the current distribution of the TSTC is stable and uniform when the temperature is increased from 29.5 K to 33.5 K at the operating current of 10 kA. To confirm occurrence of a shielding current with a long time constant, self-fields of the TSTC after the discharging were investigated. As a result, a shielding current with a long time constant was not observed in the single turn coil wound with a TSTC conductor.

As an extension of the ITER technology based on the LTS magnet concept, the development of the internal

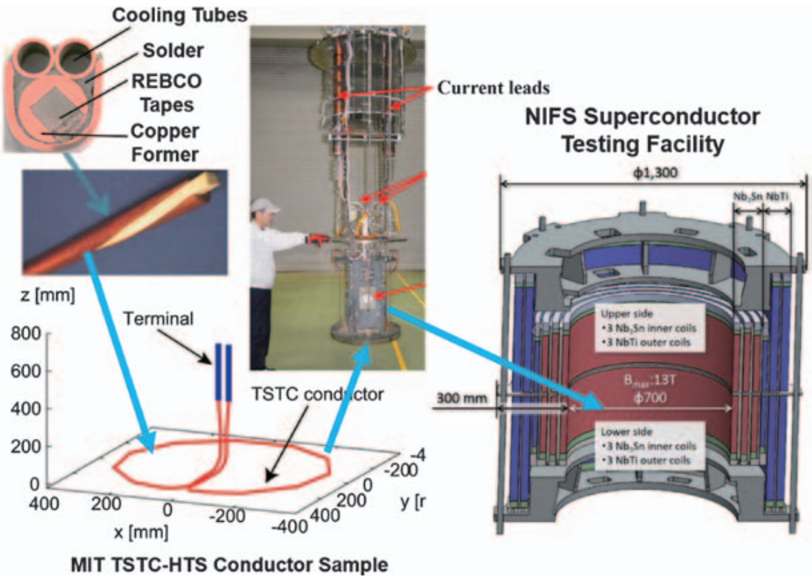

Fig. 10 Experimental setup of the MIT's TSTC-HTS conductor sample in the NIFS superconductor testing facility.

matrix-reinforced Nb_3Sn multifilamentary wires using ternary Cu-Sn alloys have been progressing. We succeeded in fabricating Nb_3Sn multifilamentary wires using Cu-Sn-Zn and Cu-Sn-In ternary alloy matrices, such as shown in Fig. 11 (a). After the synthesis process of Nb_3Sn, the ternary alloy matrices were transformed to a Cu-system solid solution, and they contributed in the improvement of mechanical strength. When a Cu-Sn-In ternary alloy is used as the matrix material, the stress value that gives the maximum critical current is found to exceed that of the CuNb reinforced Nb_3Sn wire, as shown in Fig. 11 (b) [5].

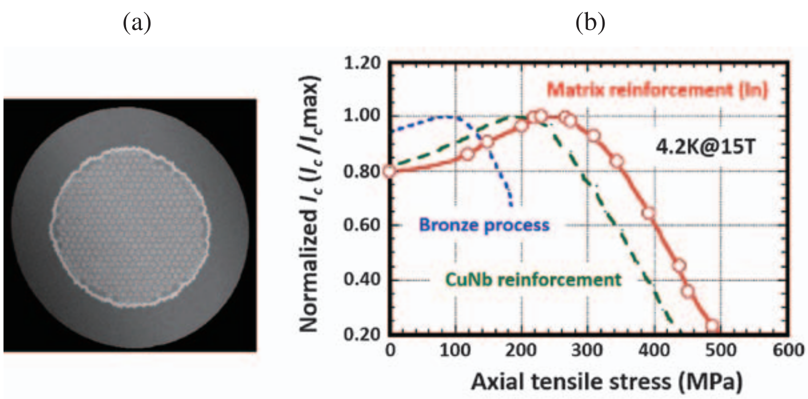

Fig. 11 (a) Typical cross-sectional image of the internally reinforced Nb_3Sn multifilamentary wire. (b) Dependence of the critical current (normalized by each maximum value) on the axial tensile stress for various Nb_3Sn wires.

[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.*, Cryogenics **105**, 103012 (2020).
[5] Y. Hishinuma *et al.*, IEEE Trans. Appl. Supercond. **30**, 6001104 (2020).

(N. Yanagi, T. Obana and Y. Hishinuma)

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 FY2019:

1. Development of irradiation-resistant NDS-Cu alloys for helical reactor divertor
2. Engineering study on lithium isotope enrichment by ion exchange
3. New R&D facility for supercritical CO_2 (sCO_2) gas system as secondary cooling
4. Establishment of high susceptible detection assay for biomolecule response and estimation of the biological effects of low-level tritium radiation by utilizing its assay
5. Development of highly ductile tungsten composite systems
6. Fundamental engineering of tritium recovery process for liquid blanket of helical reactor
7. Development of new rapid-heating and quench processed Nb_3Al large-scaled cables for the helical winding due to the react-and-wind method
8. Field estimation for improvement of environmental tritium behavior model
9. Evaluation of Heat-transfer-enhanced Channel under High Magnetic Field for Liquid Molten Salt Blanket Development
10. Evaluation of multi hydrogen isotope transfer behavior on plasma driven permeation for plasma facing materials
11. Development of effective heat removal method from liquid metal free-surface with local heating under strong magnetic field and its demonstration by Oroshhi-2

From the above ten research items, two of them (3 and 7) are briefly described below:

3. New R&D facility for supercritical CO_2 (sCO_2) gas system as secondary cooling in FFHR

In FFHR, supercritical CO_2 (sCO_2) gas system is considered to apply to the secondary cooling system for electricity generation considering the advantages of compact design and higher thermal efficiency under the operational temperature of molten-salt blanket using the FLiNaBe. It is unavoidable that a small amount of tritium is permeated into the secondary cooling system via heat exchangers. Hence, understanding of mass transfer phenomena of gas components including hydrogen isotope in interfaces between sCO_2 and structural materials is necessary. In this study, constructions of sCO_2 experimental setup were done (Fig. 15), and then the analysis of reaction products in high temperature and high pressure CO_2 was done. Using this new facility, sCO_2 condition was successfully operated and CO, H_2, CH_4 were detected as reaction products during hydrogen permeations. These Concentrations of reaction products were observed to increase with a rise of temperature [1].

7. Development of new rapid-heating and quench processed Nb3Al large-scaled cables for the helical winding due to the react-and-wind method

The large helical coils of the helical fusion reactor FFHR-d1 have three-dimensional complex architecture so that the “React and Wind (R&W)” process of the superconductor is preferable to be applied. In this respect, we

(a) (b)

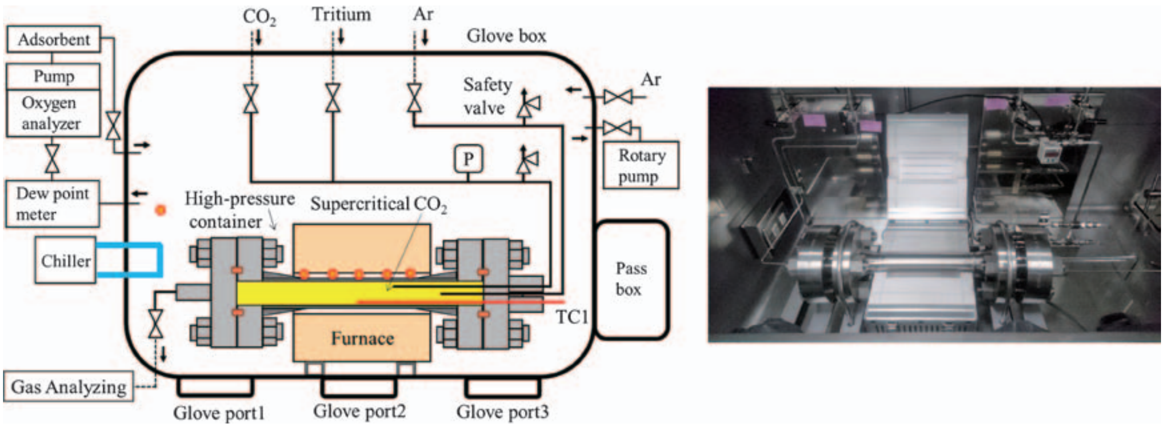

Fig. 15 (a) Design of Supercritical CO_2 gas facility. (b) Photograph of supercritical CO_2 gas facility.

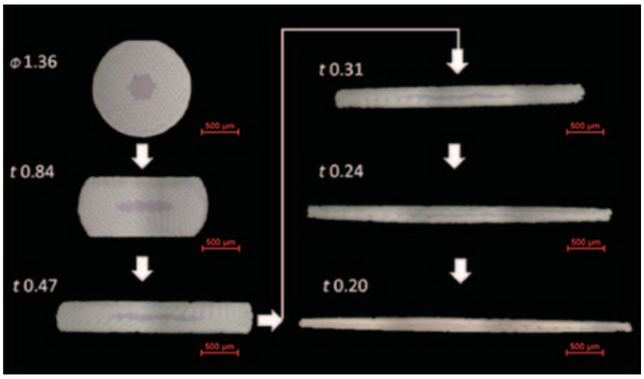

Fig. 16 Fabrication procedure of the tape-shaped RHQT Nb3Al conductor (final thickness $t = 0.20$ mm) starting from a wire-shaped conductor (diameter $\phi = 1.36$ mm).

should note that the A15 superconductors, such as Nb_3Sn, are brittle intermetallic compounds, and the critical current properties of these materials are sensitive to thermal and mechanical strains. However, the Nb_3Al conductor shows an excellent strain tolerance compared to that of Nb_3Sn, and it is suitable for applying the “R&W” process. We investigated a tape-shaped Nb_3Al conductor made by the rapid-heating and quench (RHQT) technique for further improving the mechanical strain tolerance of Nb_3Al shown in Fig. 16 [1]. The bending strain dependence on the critical current property was evaluated. No deterioration of the critical current property was observed even if a bending strain was applied above 0.6%.

[1] K. Katayama, N. Ashikawa, T. Chikada, “Mass Transfer at the interface between stainless steel and supercritical carbon dioxide”, ICFRM (2019) poster presentation.
[2] K. Yamada *et al.*, presented at 10th ACASC / 2nd Asian-ICMC / CSSJ, (2020), 7P-20.

(N. Ashikawa and K. Katayama (Kyushu University),
Y. Hishinuma, K. Takahata and A. Kikuchi (National Institute for Materials Science))