2. Fusion Engineering Research Project

The Fusion Engineering Research Project (FERP) started in FY2010 at the National Institute for Fusion Science (NIFS). Along with the conceptual design studies for the helical fusion reactor FFHR, the FERP has been developing technologies for key components, such as the superconducting magnet, blanket, and divertor (Fig. 1). The research also focuses 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. There is also a cooperation with the Large Helical Device Project and the Numerical Simulation Reactor Research Project. The FERP is also assisting the ongoing discussion on forming new "Units" for the restructuring of NIFS.

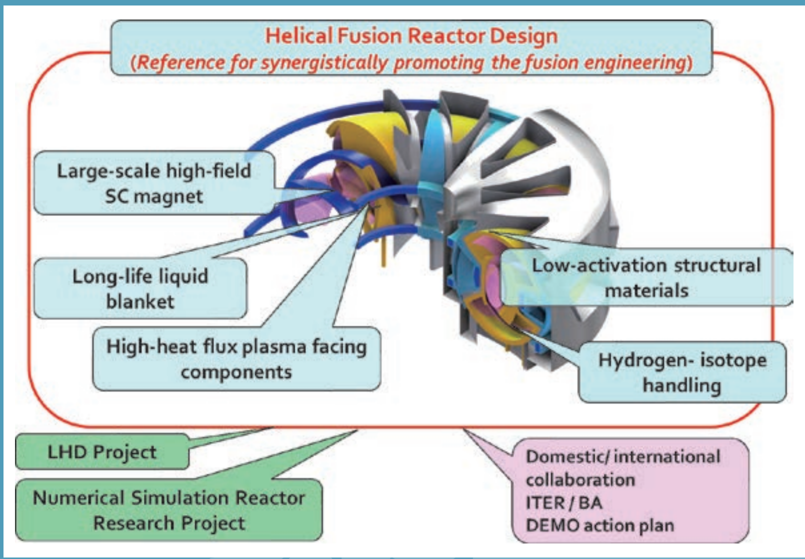


Fig. 1 Research targets of the Fusion Engineering Research Project with collaborations.

(I. Murakami)

Design Studies on Helical Fusion Reactor

In FY2020, a new design called FFHR-b3 was proposed and the step-by-step development strategy for the helical fusion power plant FFHR-d1 was updated. The construction and operation of three intermediate devices are assumed in the updated strategy: FFHR-a1 (non-nuclear system for the examination of improved magnetic configuration and advanced engineering concepts), FFHR-b3 (100 MWe class power plant that can operate for more than five years and twice the size of LHD), FFHR-c1 (500 MWe class power plant that can operate with a self-ignition condition). Because FFHR-b3 requires better plasma performance (MHD stability and energy confinement) with a large coil-to-plasma distance, configuration optimization beyond the variation of the helical coil shape based on the conventional winding law is required, such as shown in Fig. 2 for the operation window. The optimization study using the helical coil optimization code OPTHECS is being conducted in cooperation with the Numerical Simulation Reactor Research Project and a new configuration that has plasma performance equivalent to LHD with a 10% larger coil-to-plasma distance has been identified. The search for further optimized configuration is ongoing.

The design of a cartridge-type blanket for the helical reactor has been updated. The new design CARDISTRY-B3 adopts a slit wall fabricated by alternately stacking solid metal plates and porous ones made of metal or ceramic, as shown in Fig. 3. This slit wall can form a 3D curved surface along the plasma without 3D machining and realize a liquid metal wall including the divertor section by exuding liquid metal through the porous plates. Since the free surface of the liquid metal is exposed to the plasma, the vapor pressure of the working liquid metal should be low enough. Because the liquid metal also works as a coolant and a tritium breeder for the blanket, it should have various functions including a high tritium breeding ratio, low density, low viscosity, a low melting temperature, high heat removal performance, a small amount of radioactive waste generation, low corrosion, a low MHD pressure drop, low toxicity, low chemical activity, and abundant resources. To satisfy these requirements, functional liquid metal (FLM), which is ternary or quadruple alloys including Li, Sn, Pb (or Bi), and Er has been studied.

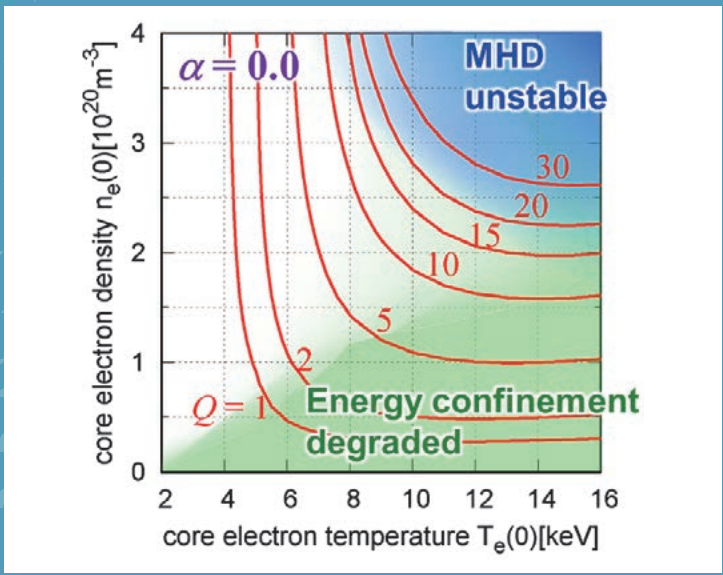


Fig. 2 Operation window for the FFHR-b3 helical fusion reactor on the core electron temperature and density.

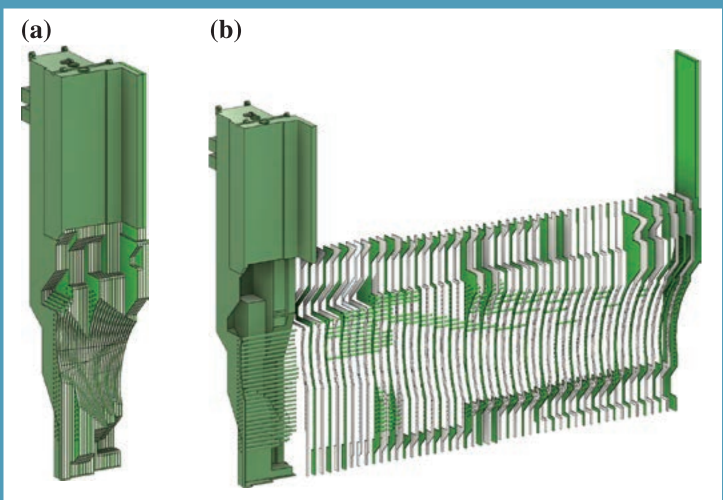


Fig. 3 (a) 3D diagram and (b) its exploded view of a breeding blanket module of CARDISTRY-B3.

(T. Goto and J. Miyazawa)

Research and Development on Blanket

In the FLiNaK/LiPb twin loop system Oroshhi-2, a tritium recovery test from circulating LiPb using a vacuum sieve tray (VST) was continued actively in FY2021. The details of the successful results are described in the following section. In the LiPb loop, a new test section was installed for analysis of impurity concentrations in LiPb circulated in the loop for more than 1000 hours (Fig. 4). LiPb circulated in the loop was introduced into a 1/2-inch diameter tube and rapidly cooled by water to avoid segregation of impurities. After solidification, the tube was taken out of the test section and concentration analysis of impurity elements was performed by inductively coupled plasma mass spectrometry (ICP-MS). Concentrations of Fe, Cr, Ni, etc. were successfully obtained and detailed analyses of the results are being conducted at present. The study will also examine the performance of the cold trap system of Oroshhi-2 which purifies the LiPb circulating in the loop. Regarding the operation of Oroshhi-2, the Department of Engineering and Technical Services has conducted the installation of a remote-control system to enhance the secure and flexible operation of the system.

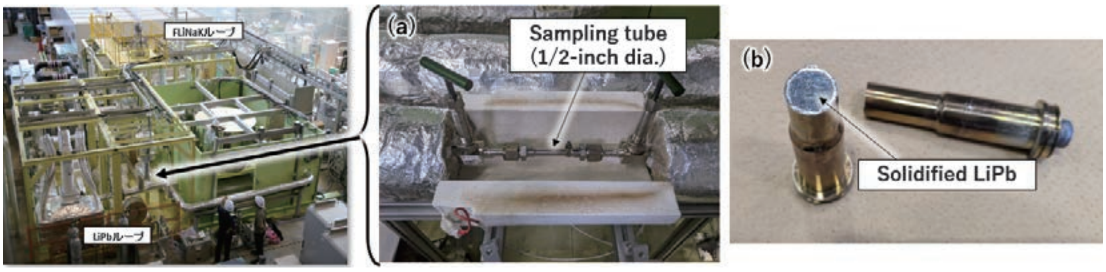


Fig. 4 Photos of (a) new test section for a sampling of LiPb and (b) sampled solidified LiPb.

(T. Tanaka and Y. Hamaji)

Development of advanced structural materials

Low-activation vanadium (V) alloys are promising candidate structural materials for the first wall/blanket applications in advanced fusion reactor systems. One of the recent studies on the V-Cr-Ti system alloys is to further reduce the radioactive characteristic after use in fusion reactors. Effects of titanium (Ti) and chromium (Cr) concentrations on microstructure and tensile properties of high purity vanadium alloys, which contain less than 300 mass ppm interstitial impurities (e.g., carbon, nitrogen, oxygen), have been investigated. It has been found that Ti can be reduced from 4wt% to 1wt% for scavenging and precipitation. On the other hand, Ti reduction results in the strength degradation of the V-Cr-Ti system alloy for both room temperature and high temperatures. Furthermore, it reveals that tensile strength gradually rises with increasing Cr concentration, indicating that a higher level of Cr can compensate for strength degradation by lowering the Ti concentration. Compared to the yield stress of V-4Cr-4Ti alloy, low-Ti and high-Cr candidates are V-6Cr-3Ti, V-8Cr-3Ti, V-10Cr-1Ti, V-12Cr-0.5Ti, and V-12Cr-1Ti alloys. The Ti and Cr concentrations will be further optimized according to irradiation damage resistance and ductile-to-brittle transition temperature investigations in the future.

(J. Shen)

Research and Development on Divertor

Advanced Multi-Step Brazing (AMSB) for fabricating the plasma-facing component has been developed. AMSB is based on the repetitive application of the Advanced Brazing Technique (ABT), which was originally developed for joining tungsten (W) and oxide dispersion strengthened copper alloy (ODS-Cu; GlidCop®). The new AMSB-type divertor heat removal component with a rectangular-shaped cooling flow path channel and V-shaped staggered rib structure was developed. The component was inserted into the divertor strike point of the Large Helical Device (LHD) and exposed to divertor plasmas for 1,180 shots. The high heat removal capability did not show any degradation over the experiment period. After extracting the component, surface analysis was conducted using an optical microscope, a focused ion beam (FIB), and a scanning electron microscope (SEM). Some micro-scale cracks with a ~50 μm width, and remarkable sputtering erosion and redeposition phenomena due to the strong influx of the divertor plasma were simultaneously confirmed on the W armor, even though the heat removal capability deduced by thermocouples did not change from the initial condition.

Reduced activation ferritic/martensitic (RAFM) steel is one of the candidates for the divertor cooling pipe and heat sink of the dome or baffles in fusion reactors. To obtain a reliable joint between W armor and RAFM steel, an appropriate intermediate layer, such as Cu, is required to be inserted between the W armor and RAFM steel to absorb residual stress during a heat treatment procedure of the joint. As a first step, we tried to fabricate a W/Cu/RAFM steel joint using the ABT between W/Cu and Cu/RAFM steel. Fig. 5(a) shows a photograph of the W/Cu/RAFM steel joint sample. The joint structure was successfully made without any macro-scale cracks, which was confirmed by a scanning electron microscope (SEM). Fig. 5(b) and (c) correspond to the cross-sectional SEM images in the vicinity of the joint interfaces for the W/Cu and Cu/RAFM steel joints, respectively. A very fine microstructure without any small-scale defects, such as delamination or voids, can be confirmed for both interfaces. To apply in the reactor environment, a tolerant material against the neutron dose should be used in the intermediate layer instead of Cu. We will continue to improve the W/Cu/RAFM steel joint structure in the next fiscal year.

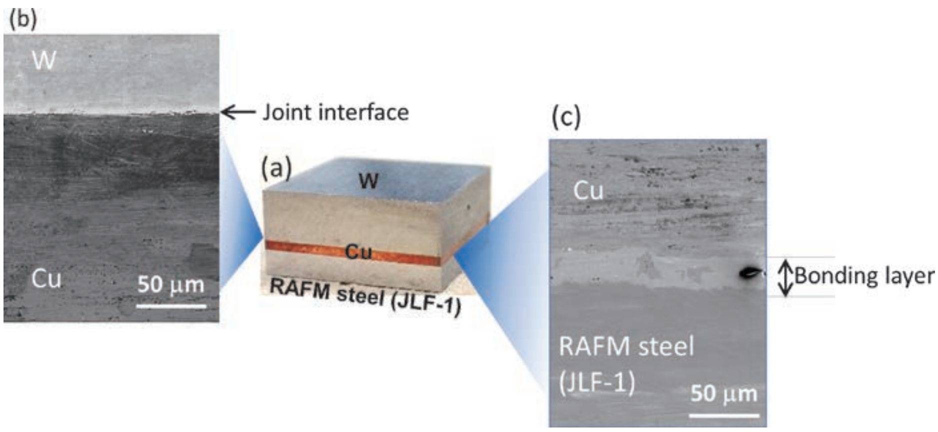


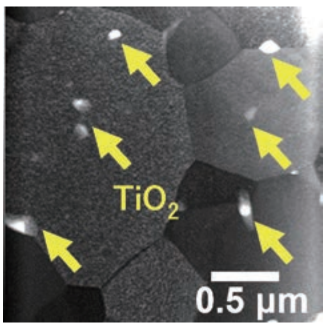
Fig. 5 Photograph of W/Cu/RAFM steel joint sample by ABT (a), and cross-sectional SEM images of the vicinity of the joint interfaces for W/Cu (b) and Cu/RAFM steel joint.

Highlight

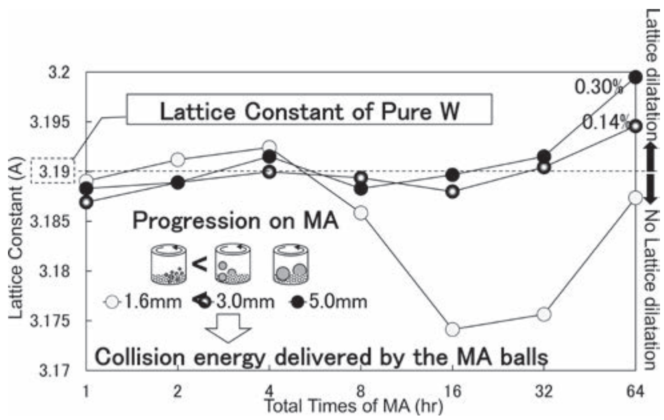
Development of oxide dispersion strengthened tungsten (ODS-W) including titanium oxide

For improvement of plasma facing tungsten on a fusion divertor, we have developed a new oxide dispersion strengthened tungsten (ODS-W) including titanium Oxide as strengthening particles (Fig. 6(a)), fabricated by mechanical alloying (MA)-hot isostatic pressing (HIP), which can inhibit the decrease of a mechanical property even after recrystallization occurs. Our past studies have shown that the condition of the MA process affects the mechanical and thermal properties of the products. For the fabrication of DS-W generally, mechanical alloying is known as an essential process, and important in view of the material design on DS alloys. Even though the MA process needs the understanding of criteria parameters for evaluation of the alloyed state, detailed research of the alloying process on DS-W has not been done. In this study, the influence of the MA ball diameter on the ODS-W products by characterizing the evolution of the milled particles during MA was elucidated. Considering the ball diameter, the differences in the lattice constant and microstructure directly indicate the progress rate of mechanical alloying. The effect of the ball size was interpreted as that of collision energy delivered by the MA balls.

Fig. 6(b) shows lattice constants with MA time. The MA powders with 1.6, 3.0- and 5.0-mm balls exhibited a slight rise and fall of the lattice constant from 0-hr MA to 8-hr MA, then started the re-dilatation from 8-hr MA to 16-hr MA. The slight change of lattice constant indicates that MA less than 16 hr is not sufficient for alloying and forming an inhomogeneous microstructure. The lattice constant of the MA powder with a 1.6-mm ball after 16-hr MA time did not reach that of pure tungsten even though MA of 64-hr had finished, inferring that the forced solid-solution state was not caused between the tungsten matrix and the titanium element. On the other hand, lattice constants of MA powders with 5.0- and 3.0-mm balls exceeded that of pure tungsten from 16 hr to 32 hr, implicating the starting of mechanical alloying between tungsten matrix and titanium element. Finally, both MA powders with 3.0- and 5.0-mm balls exhibited dilatations of 0.14% and 0.29% after 64-hr MA. These results reflect that collision energy caused by the MA ball weight affected the acceleration, suggesting the possible beneficial effect of shortening MA time.



(a)

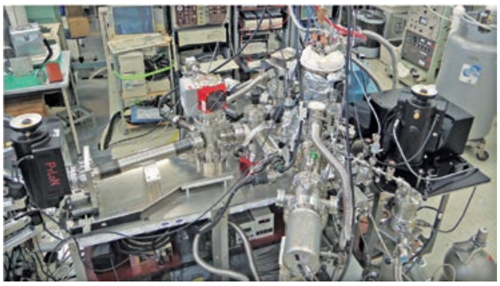


(b)

Fig. 6 (a) Microscopic image of oxide dispersion strengthened tungsten (ODS-W) using titanium oxide, (b) variation of lattice dilatation (LD) with MA time

(H. Noto)

For spectroscopic diagnostics on impurity ions in fusion plasmas, we have been acquiring spectroscopy data of highly charged tungsten ions using an electron beam ion trap (called CoBIT). High-resolution spectral measurements are required to resolve many spectral lines to be identified for performing detailed spectral analysis. Recently, we attempted high-order and high-resolution spectroscopy using higher-order light from a diffraction grating for the extreme ultraviolet wavelength region and succeeded in resolving broad lines around 50 Å of highly charged tungsten ions ($W^{26\sim30+}$). (Fig. 7)



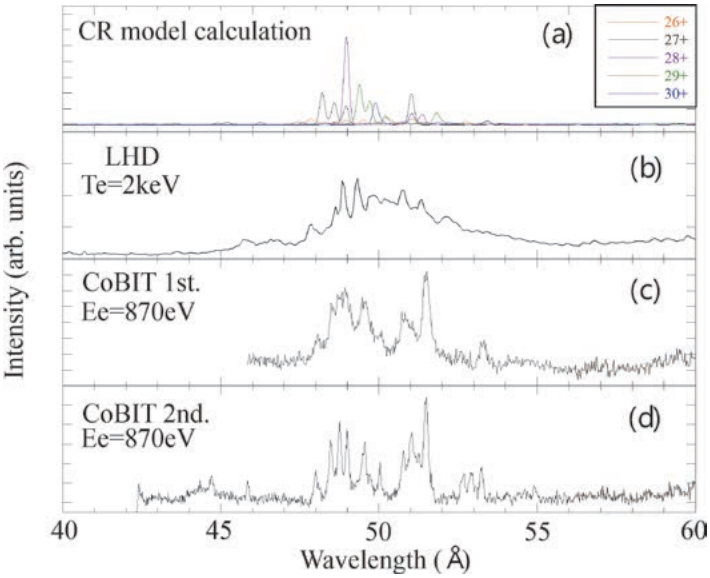


Fig. 7 [Left] Photo of the electron beam ion trap “CoBIT” with installed spectrometers and [right] highly charged tungsten ions spectra: (a) CR-model calculation, (b) typical LHD spectrum, (c) CoBIT spectrum, and (d) second-order spectrum of CoBIT.

(M. Tokitani and H. Sakaue)

Research and Development on Superconducting Magnet

In recent designs of the helical fusion reactor series FFHR, the High-Temperature Superconducting (HTS) magnet is considered and a 100-kA-class HTS conductor has been developed. It is noted that large-current capacity HTS conductors are also being developed in the world to be applied to a variety of designs of fusion reactors. As a prior phase to fusion reactors, applying HTS magnets to the next-generation fusion experimental devices is now being explored. For this purpose, a relatively smaller conductor is required, and presently the target is found at 10–20 kA current in a magnetic field of ~10 Tesla at a temperature of 20 K. Three types of HTS conductors, STARS, FAIR, and WISE (Fig. 8) with different internal structures are being developed with all using REBCO tapes. For these conductor types, 1-3-m long samples have been fabricated and tested in liquid nitrogen (77 K) with no external magnetic field, and the fabrication method has been improved. Then the conductors are tested in a high magnetic field (<8.5 Tesla) and at low temperature (20–50 K) using the large-superconductor testing facility by installing 2-m long conductor samples. A 6-m long, coiled sample was fabricated for the STARS conductor (Fig. 9) and successfully tested at 20 K, 8 T in the large-bore, high-field superconductor testing facility.

(a) **(b)** **(c)**

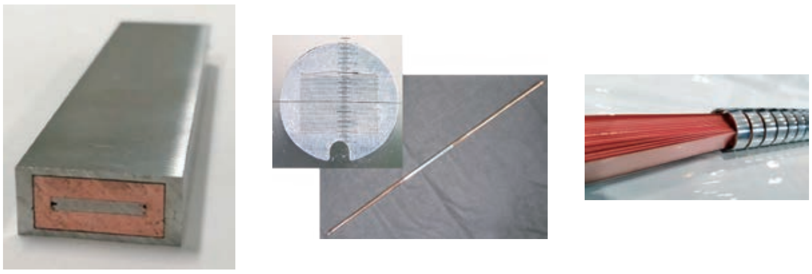


Fig. 8 Schematic drawings of the three types of large-current HTS conductors being developed to apply to the next-generation fusion experimental devices: (a) STARS, (b) FAIR, and (c) WISE conductor.

Fig. 9 A 6-m-long coiled HTS-STARS conductor sample and the research team.

Development of Rapid Heating/Quenching and Transformation (RHQT) processed Nb_3Al multifilamentary tapes has progressed for the new Low-Temperature Superconducting (LTS) magnet option on the DEMO reactor design. We employed a rectangular shape for the improvement of wire flexibility, aiming for the “React and Wind” process. No critical current density (Jc) deterioration was already confirmed, even if a bending strain was applied above 0.6%. Effects of cold flat-rolling on the non-Cu Jc of the RHQT processed Nb_3Al multifilamentary tapes were also investigated. As shown in Fig. 10 (a), the non-Cu Jc property was amplified by increasing the accumulated reduction ratio. And non-Cu Jc values corresponded to above twice higher than that of the conventional round-shaped sample before cold flat-rolling. As shown in Fig. 10 (b), the non-Cu Jc property of the rectangular tape-shaped sample under an external magnetic field above 18 T was higher compared with that of the round-shaped sample. This Jc improvement under a higher magnetic field would be caused by the increase of pinning centers due to the grain boundary density increment associated with the Nb_3Al grain refinement.

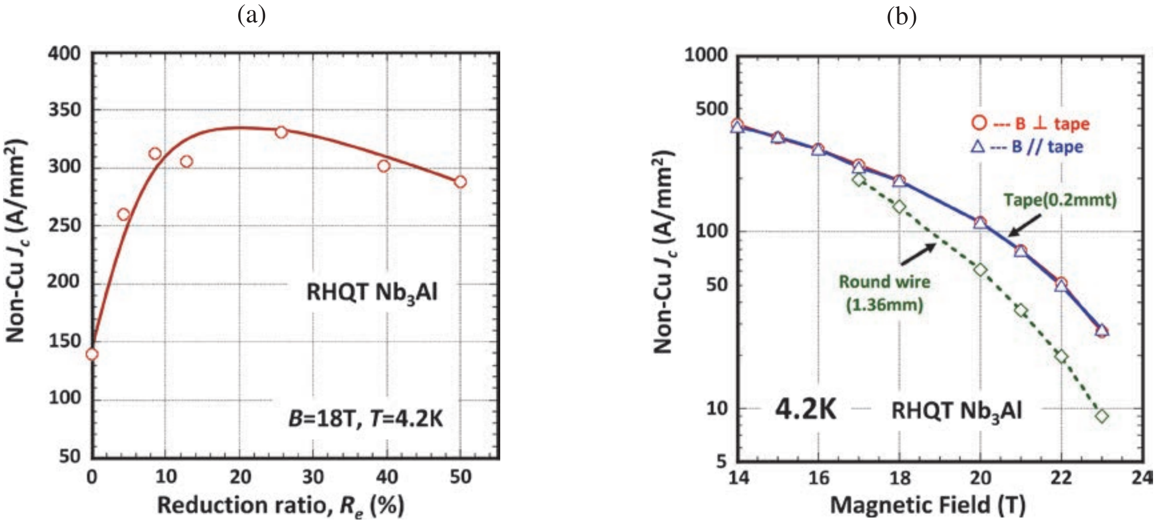


Fig. 10 (a) Non-Cu Jc as a function of the accumulated reduction ratio, (b) Non-Cu Jc of the rectangular tape-shaped sample under an external magnetic field.

(N. Yanagi and Y. Hishinuma)

LHD-Project Research Collaboration

The LHD Project Research Collaboration program has been contributing to enhancing both the scientific and technological foundations of research related to the LHD project as well as to future helical fusion reactors. A feature of this collaboration program is that all research is performed at universities and/or institutions outside NIFS. For fusion engineering, the following eleven subjects were conducted in FY2021:

1. Development of highly ductile tungsten composite systems
2. Fundamental engineering of tritium recovery process for liquid blanket of helical reactor
3. Development of new rapid-heating and quench processed Nb_3Al large-scaled cables for the helical winding due to the react and winding method
4. Evaluation of heat-transfer-enhanced channel under high magnetic field for liquid molten salt blanket development
5. Evaluation of multi hydrogen isotope transfer behavior on plasma driven permeation for plasma facing materials
6. Studies on liquid hydrogen cooled HTC superconducting magnet
7. The analysis of biological effects elicited by organically bound tritium using life science techniques
8. Technological development of FeCrAl-ODS alloys coexisting with liquid metal cooling system of helical fusion reactor
9. Development of REBCO high Tc coated conductor with conductive micro-path
10. Fabrication technology development toward practical use of functional coating for helical reactor liquid blanket
11. Improvement of environmental tritium transfer model based on the observation near tritium released facilities

From the above eleven collaborative research items, subject 2 is briefly described below:

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

NIFS is collaborating with universities to study liquid metal lithium-lead (LiPb) as one of the candidate tritium breeding and cooling materials for fusion blanket systems. LiPb was already short-listed as a candidate material in the 1980s, whereas it was not treated as a primary design target due to the impractical tritium extraction device size. Estimation of a tritium release rate Q for the device was performed based on a static diffusion coefficient of hydrogen isotopes in LiPb. A breakthrough proof of principle (PoP) was performed in 2010 in a laboratory at Kyoto University. The release rate Q was enhanced by two orders of magnitude when LiPb droplets with dissolved tritium were falling in a vacuum (hereafter named vacuum sieve tray (VST)). This discovery, however, was at a laboratory scale with a single column of droplets, and its technology readiness level (TRL) was not high. To ensure that the VST is a viable blanket technique for a practical fusion reactor, the following requirements must be met. One is the stability of the high tritium release rate Q during a practical operation. Another is proof of the same release rate from multiple falling droplets which is mandatory for supplying a quantity of liquid LiPb for a fusion reactor. To enhance the VST TRL level, a scaled-up setup was integrated into Oroshhi-2 (liquid metal test loop at NIFS) as shown in Fig. 11. A couple of campaigns were undertaken. Results showed the viability of the VST for a practical operation. An epoch-making result was the discovery of a significantly high Q value, which is six orders of magnitude higher than that in the static condition. It is owing to the quick dispersion of tritium in falling LiPb droplets with spherical oscillation. This result affects the tritium inventory analysis of blanket systems, which is one of the major issues of safety and fuel self-sufficiency. The experimental campaign will continue in FY2022.

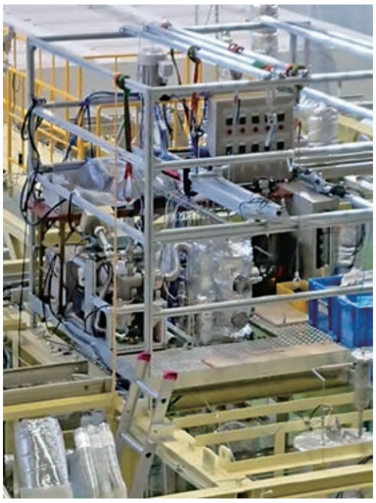


Fig. 11 Overview of the VST experimental setup which was integrated into the Oroshhi-2 facility at NIFS.

(Y. Hamaji, T. Tanaka, J. Yagi (Kyoto Univ.) and F. Okino (Kyoto Univ.))