2. Fusion Engineering Research Project

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

Reactor Design Studies

The conceptual design studies on the helical fusion reactor have been intensively conducted by FERP. In FY2016, the present design was summarized as FFHR-d1. The major radius of FFHR-d1 is 15.6 m, which is four times that of LHD (3.9 m). The heliotron magnetic configuration is similar to that of LHD, having a pair of helical coils with a toroidal pitch number of 10. The toroidal magnetic field has two options at 4.7 T (for FFHR-d1A) and 5.6 T (for FFHR-d1B). The operation point is explored using a design integration code, HELIOSCOPE, incorporating the “Direct Profile Extrapolation” (DPE) method based on the LHD plasma parameters. A self-consistently obtained operation scenario secures the energy multiplication factor $Q \sim 10$. The confinement improvement in the ongoing deuterium plasma experiments in LHD, when confirmed, should lead ultimately to the self-ignition ($Q = \infty$). From FY2017, the design activity has been shifted to a smaller version of FFHR-d1, which is FFHR-c1. The major radius is presently set at 10.92 m (2.8 times LHD).

For both the engineering design of FFHR-d1 and c1, a number of innovative ideas have been proposed from the following three purposes: (1) to overcome the difficulties related with the construction and maintenance of three-dimensionally complicated large structures, (2) to enhance the passive safety, and (3) to improve the plant efficiency. The details are described below associated with the development of each component.

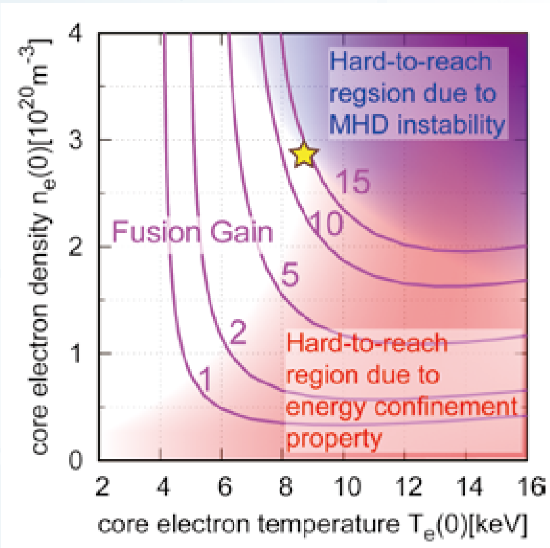

Fig. 1 POPCON plot for the helical fusion reactor FFHR-c1.

Highlight

Innovations for making the helical fusion reactor compact

As is described above, the conceptual design study for the LHD-type helical fusion reactor is now being shifted from the previous design of FFHR-d1 (major radius R = 15.6 m) to a more compact design of FFHR-c1 (R = 10.92 m). There are two innovative factors that have made this shift possible.

(1) One is the employment of the NITA (Newly Installed Twist Adjustment) coils. The NITA coils are the sub-helical coils located outside the main helical coils. The minor radius of the NITA coils is about two times that of the main helical coils. The current is applied in the opposite direction from that of the main helical coils and its amplitude is about 5-10%. By having these NITA coils, it is found that the distance between the helical coils and the plasma (or the ergodic layers outside the last closed magnetic surface) can be increased. A comparison of vacuum magnetic surfaces with and without the NITA coils is shown in Fig. 2. Owing to this innovation, the new design point can be explored as shown in Fig. 3.

(2) The other factor is the employment of the high-temperature superconductors (HTS). For the present FFHR-c1 design, the maximum magnetic field on the helical coils is about 19 T, which is beyond the limit of the low-temperature superconductors (LTS), such as Nb_3Sn. The HTS has been considered also for FFHR-d1, but this was for the purpose of employing the “joint-winding” method. This could be the primary choice also for FFHR-c1.

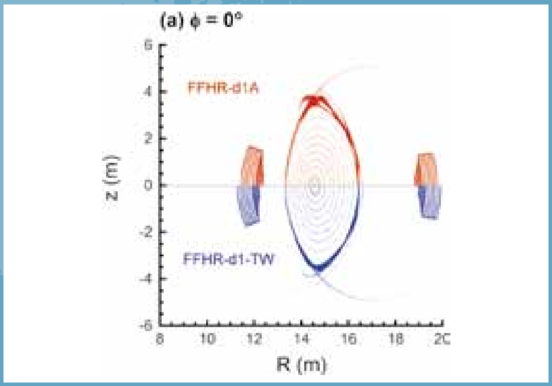

Fig. 2 Comparison of the vacuum magnetic surfaces with and without the NITA coils. The plasma is not very different, but the helical coils are located farther from the plasma with the NITA coils.

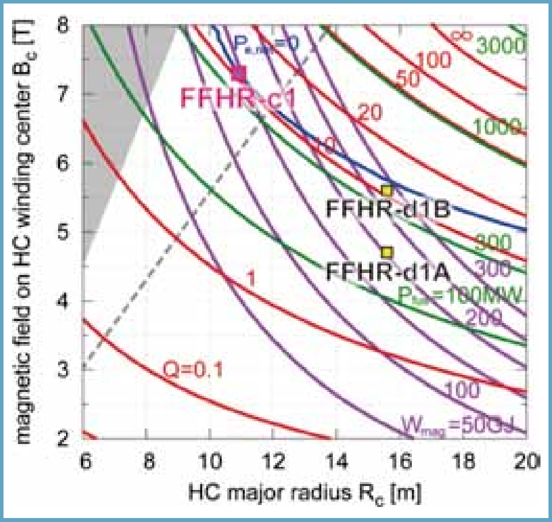

Fig. 3 The design window analysis showing the design points for the FFHR devices. The FFHR-c1 has been selected by considering the core plasma performance and positive net electric power. The shaded region corresponds to where the nuclear heating on the superconducting magnets is too high. The design limit by this nuclear heat moves to the region shown by the broken line if the NITA coils are not used for the blanket space enlargement.

Research and Development on the Superconducting Magnet

The large-scale superconducting magnet system to be applied to the helical fusion reactor has been developed both for the low-temperature superconductor (LTS) and high-temperature superconductor (HTS) options in collaboration with universities and research institutes, domestic and international. For this purpose, the new testing facility equipped with a 13 T magnetic field and a large ϕ0.7 m bore superconducting magnet has started its operation. The facility supplies 4.2-50 K temperature controlled helium and 50 kA sample current. The first cooling test of the 13-T magnet was successfully conducted and the excitation test was carried out up to the magnetic field of ~1.7 T. A full excitation test is planned to be carried out in summer FY2018.

For the LTS conductors, the degradation of the transport current property by mechanical strains on the practical Nb_3Sn wires is a serious problem to be applied to the future fusion magnets operated under higher electromagnetic forces. A development of an internal-matrix-strengthened Nb3Sn multi-filamentary wire is being carried out utilizing the solid solution strengthening mechanism. We successfully developed an Nb3Sn multi-filamentary wire using Zinc (Zn) solid solution ternary Cu-Sn alloy (Cu-Sn-Zn) matrix through the conventional bronze process. The cross-sectional image is shown in Fig. 4. The internally strengthened matrix due to the (Cu, Zn) solid solution is a simpler method than other reinforcement methods, and it has become one of the attractive high strengthening method.

For the HTS conductor, a 100 kA-class STARS (Stacked Tape Assembled in Rigid Structure) conductor has been developed to be applied to the helical fusion reactor FFHR-d1. In the earlier test, a 3-m sample successfully achieved 100 kA at 5.3 T and 20 K. One of the issues associated with this conductor is the non-uniform current distribution among the stacked REBCO tapes. In order to focus on this problem, a down-scaled conductor sample using five tapes was fabricated and tested in liquid nitrogen (Fig. 5). The sample current was supplied from one side of the stacked tapes, and a non-uniform current distribution was formed, which was confirmed by Hall probes measurement. Despite this fact, the transport current reached the expected critical current determined for the whole five tapes. A detailed numerical analysis is being conducted to explain this observation.

Another type of HTS conductor, named TSTC (Twist Stack Tape Cable) has been developed at Massachusetts Institute of Technology (MIT) in US. A 2-m TSTC sample is being prepared to be tested in the 13-T magnet facility in summer 2018.

Fig. 4 Typical cross-sectional image of the internal-matrix-strengthened Nb_3Sn multi-filamentary wire using the solid-solution strengthening mechanism.

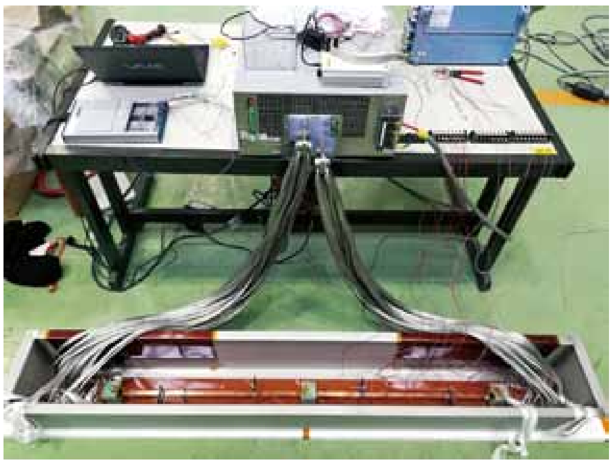

Fig. 5 Experimental setup of a down-sized HTS conductor for the measurement of current distribution among simply-stacked REBCO tapes.

Research and Development on the Blanket

The blanket is a crucial component for a fusion reactor that converts kinetic energy of neutrons to heat and also produces the fuel, tritium, using lithium. This part of the blanket is called the breeder blanket. Behind the breeder blanket, the shielding blanket is situated for the purpose of stopping the residual neutrons so that the superconducting magnet outside the blanket is effectively shielded.

For the helical fusion reactor, FFHR, the liquid-type breeder blanket is considered and designed. For the purpose of developing the blanket, a large-scale forced-convection twin-loop facility of heat and hydrogen, “Oroshhi-2”, was constructed equipped with a superconducting magnet to apply uniform perpendicular magnetic field of 3 T to the flow of either molten salt (Flinak) or liquid metal (LiPb). Using this facility, various researches have been conducted, such as the measurement of the MHD pressure drop in a flow of liquid LiPb through a two-sectioned bending tube in collaboration with Kyoto University. In FY2017, the corrosion characteristics on various materials have been investigated with a molten salt (Flinak) flow at a temperature of higher than 600 centigrade under the magnetic field of 1 T applied in the perpendicular direction to the flow. The experimental setup is depicted in Fig. 6.

For the tritium breeding blanket design, we have chosen the liquid blanket option with molten salt from the viewpoint of passive safety. The present selection of molten salt is FLiNaBe, which has the melting point at 580 K. In order to increase the hydrogen solubility, an innovative idea was proposed to include metal powders, such as titanium. An increase of hydrogen solubility over five orders of magnitude has been confirmed in an experiment, which makes tritium permeation barrier less necessary for the coating on the walls of cooling pipes. Various R&D’s related to this idea are ongoing.

Maintenance is one of the important and difficult issues to realize the helical fusion reactor. For the blanket, a toroidally-segmented system, T-SHELL, was proposed, by dividing the toroidal blanket into every 3 degrees. Another innovative idea is the cartridge-type blanket concept, CARDISTRY-B. The discussion about the maintenance concepts both for the blanket and the divertor is ongoing.

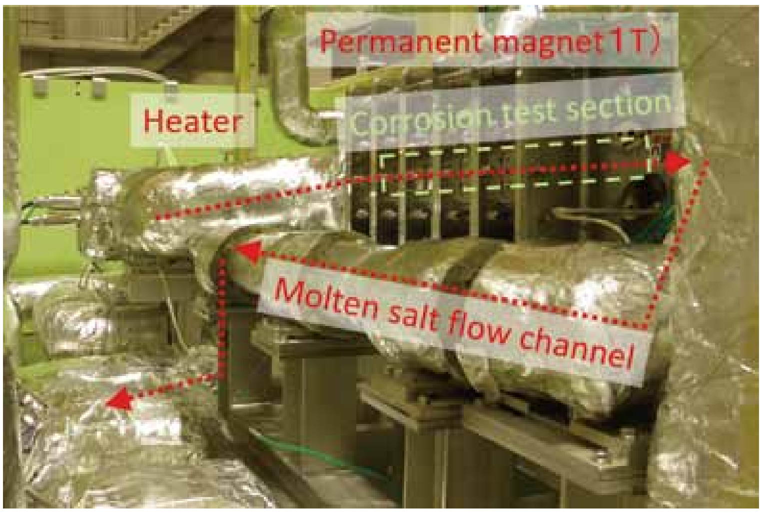

Fig. 6 The corrosion experiment part in the molten salt loop in the Oroshhi-2 twin-loop facility.

Research and Development on the Divertor

The divertor heat flux in fusion reactors is considered to become higher than 10 MW/m^2 in steady state. Important subjects in the engineering research and development for coping with this high heat flux are the material selection, bonding technology between armor tiles and coolant pipes. The design studies on the three-dimensional shape for the helical fusion reactor FFHR is especially important by including the precise neutronics analysis. For FFHR, the water-cooled tungsten monoblocks is considered to be the primary choice. It is expected that a copper-alloy could be applied by placing diverter tiles at the backside of blankets where the incident neutron flux is sufficiently reduced. The peak divertor heat load on the divertor is expected to exceed > 20 MW/m^2 because of the non-uniform divertor heat load profile.

Improvement of copper alloys is being examined in respect to high temperature mechanical properties and radiation resistance. Fabrication process using mechanical alloying and hot isostatic pressing (HIP) is being investigated. Characterization of welding and HIP joints are also carried out for ODS steels and ferritic steels.

An in-situ fabrication process designed to fabricate dispersion strengthened (DS) copper (Cu) alloy with yttria (Y_2O_3) dispersed particles is being proposed, which is an advanced in-situ process combining Mechanical Alloying (MA) and Hot Isostatic Pressing (HIP). For the optimization of the process control, the effects of MA time and Y amounts were investigated. Detailed inspections using XRD, TG-DTA, and SEM confirmed that Cu, Y and CuO powders were mechanically alloyed successfully after MA of 32 hrs. The XRD of these samples also confirmed the formation of Y_2O_3 particles. In addition, the higher Y amounts helps the densification process and enhances the strength of materials. However, it deteriorates the thermal conductivity. Because of the strength-conductivity trade-off, the selection of Y_2O_3 amount should be determined.

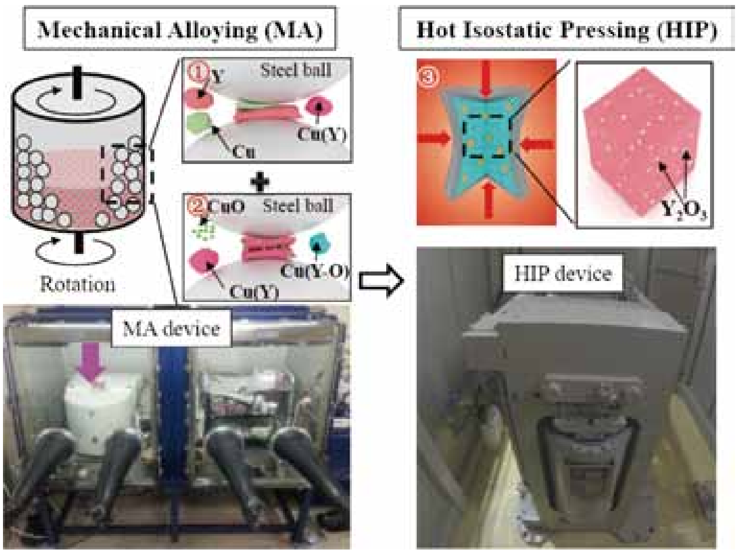

Fig. 7 Fabrication method of Y_2O_3-dispersed copper alloys using an advanced in-situ process combining Mechanical Alloying (MA) and Hot Isostatic Pressing (HIP).

The ACT-2 electron beam facility (maximum power: 300 kW) has been used to apply >10 MW/m^2 of steady-state heat flux to various samples, such as a tungsten block brazed with a copper alloy. This facility is also used to promote collaborations with universities as well as the industry to develop heat removal techniques from water channels. During the last fiscal year, the control system of the electron beam has been upgraded and so that this facility now has a capability of applying also a short pulse (< 1 ms) high power heat flux. The experiment shows that the surface structure on tungsten samples with high power heat flux is significantly varied before and after irradiation by helium ions.

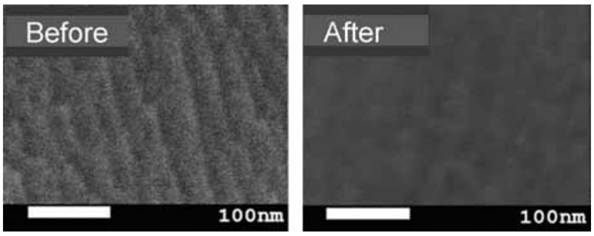

Fig. 8 Surface images of tungsten samples with high heat flux electron beams before (left) and after (right) irradiation by helium ions in the ACT-2 electron-beam facility.

An advanced brazing method for bonding tungsten and Oxide Dispersion Strengthened copper (ODS-Cu) is being developed. The small-sized bonding specimens show that the strength of the bonding interface is higher than that of the bulk tungsten. On the basis of this result, a large-scale divertor mock-up was fabricated using 28 tungsten plates bonded to an ODS-Cu block. A reliable bonding technique was established by maintaining a constant gap of 0.5 mm between each tungsten plate.

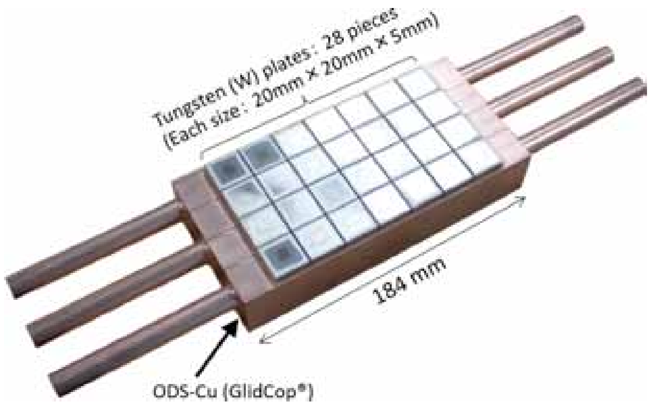

Fig. 9 A large-scale divertor mock-up fabricated using 28 tungsten plates bonded to an ODS-Cu block.

In order to solve the difficult issues on divertors associated especially with the intense heat flux, another possibility is to use liquid metal instead of solid tungsten. A new concept of liquid metal limiter/divertor, REVOLVER-D, has been proposed, having ten units of molten tin shower jets (falls) installed on the inboard side of the torus of FFHR to intersect the ergodic layer. It is proposed that the vertical flow of Tin jets could be stabilized using metal chains embedded in the jets. This system works as an ergodic limiter, and the conventional full-helical divertor becomes less necessary, although they could, or should, be still situated at the backside of the liquid divertor. Neutral particles are expected to be efficiently evacuated through the gaps between liquid metal showers. There are also some other merits of applying liquid divertors, which are that the maintenance scheme could become revolutionary easy compared to the situation for the full-helical three-dimensional divertors and the total volume of waste would be much smaller than the case of all those tungsten plates and copper alloys. The R&D for developing the REVOLVER-D system has begun having a free fall of liquid metal through a channel with an external magnetic field provided by an array of permanent magnets.

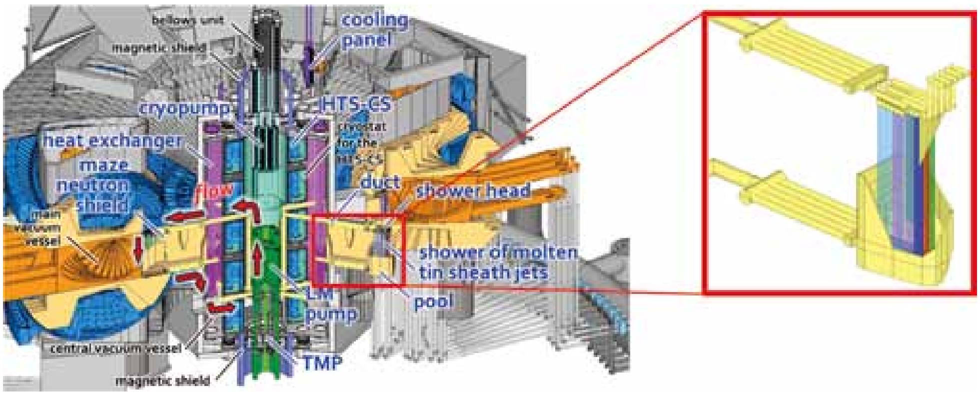

Fig. 10 Schematic illustration of the liquid divertor system REVOLVER-D.

Fig. 11 Experimental setup for investigating the free fall characteristics of liquid metals. Magnetic field is applied to the liquid metal flow using an array of permanent magnets.

Highlight

Innovations for the blanket material

The low-activation vanadium alloys of V-4Cr-4Ti, with a chemical composition of 92 (mass)% Vanadium, 4 (mass)% of Cromium and 4 (mass)% of Titanium can be promising candidates for the structural materials of fusion reactor blankets. However, previous alloys exhibited technical problems, such as brittle fracture at weld joint and cracking during tubing process. The embrittlement is induced by ductility loss due to contamination with gaseous impurities, such as C, N and O, in the fabrication processes. National Institute for Fusion Science (NIFS) has been leading the scale-up and purification of vanadium alloys under collaboration with Japanese universities, and developed NIFS-HEAT-2 alloy. The purification has successfully enhanced workability, weldability and also low-activation property. Fig. 12 shows the improvement of weldabilty, indicating much high absorbed energy in impact fracture tests on the weld joint, compared with the previous V-4Cr-4Ti alloy US832665 made by US-DOE (United States of America, Department of Energy) program. No degradation of impact energy was revealed for the weld metal of NIFS-HEAT-2 alloy, while it was under the acceptable level as structural materials for US alloy.

Although many properties were improved by the purification, possible degradation of high-temperature strength due to purification softening was a concern. Therefore, the NIFS fusion energy research project has recently focused on the high-temperature creep property of NIFS-HEAT-2, which limits the operation condition for fusion reactor blanket. Constant load was applied at elevated temperature, and induced deformation and finally rupture of NIFS-HEAT-2 in the creep tests. Fig. 13 plots the loading stress versus creep rupture time at 800°C. It was successfully confirmed that creep strength of NIFS-HEAT-2 in the lower loading condition was comparable to those of US, whereas in the higher loading condition, it was degraded compared with the dashed trend line for US data. Since the loading is expected as 100 MPa or less in the fusion blanket, the purification of NIFS-HEAT-2 does not require any change in design loading condition for vanadium alloy blanket. In conclusion, the purification for NIFS-HEAT-2 improved many properties, such as weldability, workability and low-activation properties, and raised no negative effect on high-temperature creep properties under the blanket loading conditions.

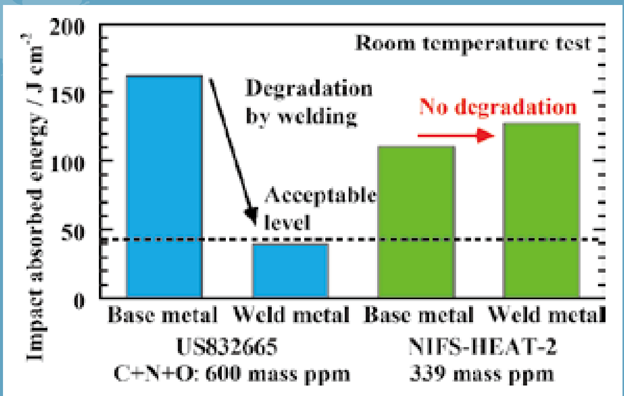

Fig. 12 Impact absorbed energy of the weld metal after welding and the base metal before welding.

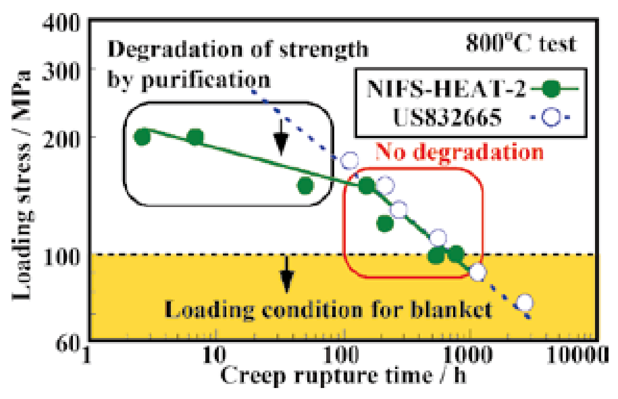

Fig. 13 Creep rupture time under the constant loading stress at 800°C.

LHD-Project Research Collaboration

The LHD Project Research Collaboration program has been contributing to enhancing both the scientific and technological foundations for the research related with LHD as well as the future helical fusion reactors. The characteristics of this collaboration program is that the researches are performed at universities and/or institutions outside NIFS.

It has been twelve years since the LHD Project Research Collaboration started to invite external reviewers from universities and institutions within Japan to form three committees and one advisory council in the selection process of collaboration subjects on fusion engineering, fusion science and plasma physics. Close collaboration among these areas is getting more and more essential for the further progress of fusion research. It is required for NIFS to develop a sound network of fusion research among universities and government institutions, by enhancing information exchange, planning, international collaboration, and education of graduate students. An important criterion for choosing a new collaboration subject is that the proposal is new and innovative, which is useful for the LHD project but is not directly planned at NIFS.

In the research area on the fusion engineering, the following twelve subjects were approved and conducted in FY2017.

1. In-situ LIBS measurements of hydrogen isotope retention and material mixing
2. Development of plasma-spray technique and evaluation of coating properties for LHD tungsten divertor
3. Tritium accumulation and its decontamination of deposition layer
4. Study on development of environmental tritium behavior model incorporating organic bonded tritium
5. Investigation of helical winding application of Nb3Sn cable-in-conduit conductor after heat treatment
6. H, D and T quantitative analyses for plasma facing walls exposed during deuterium experiment
7. Knowledge and technology transfer from IFMIF-EVEDA accomplishment to systemization of liquid blanket research
8. Development of irradiation-resistant NDS-Cu alloys for helical reactor divertor
9. Development of effective heat removal method from liquid metal free-surface with local heating under strong magnetic field and its demonstration by Oroshhi-2
10. Engineering approach to lithium isotope separation using cation exchange resin
11. Behavior of tritium in a secondary cooling loop of a fusion reactor
12. Establishment of molecular biology response and investigation of the biological effects of low level tritium radiation

From the above ten research items, two of them (6 and 10) are briefly described below:

<u>H, D and T quantitative analyses for plasma facing walls exposed during deuterium experiment</u>

A new thermal desorption spectroscopy (TDS) measurement system was designed and installed in LHD to evaluate all the hydrogen isotope desorption behaviors in materials, simultaneously. Using this HI-TDS system, D and T desorption behaviors in implanted or DT gas exposed tungsten samples installed in LHD were examined. It was found that the major hydrogen desorption stages consisted of two temperature regions, i.e., 700 and 900 K. This is consistent with the result obtained in the previous hydrogen plasma campaign, which showed that most of the hydrogen atoms were trapped by the carbon-dominated mixed-material layers. By energetic ion implantation, the major D desorption was found at ~900 K with a narrow peak. For gas exposure, H was preferentially replaced by D and T with a lower trapping energy. In addition, T replacement rate by additional H_2 gas exposure was evaluated. This observation indicates that the hydrogen replacement mechanism might be clearly varied by exposure methods.

Fig. 14 The HI-TDS system installed in the LHD building at NIFS.

Engineering approach to lithium isotope separation using cation exchange resind

The lithium isotope separation by the displacement chromatography using cation exchange resin has been studied. We made cation exchange resins with 50% and 90% cross-linkage, and the SEM photo of the latter one is shown in Fig. 15. The isotope separation coefficient depends on the cross-linkage. The separation coefficients of the synthesized resins are much higher than that of commercially available resins.

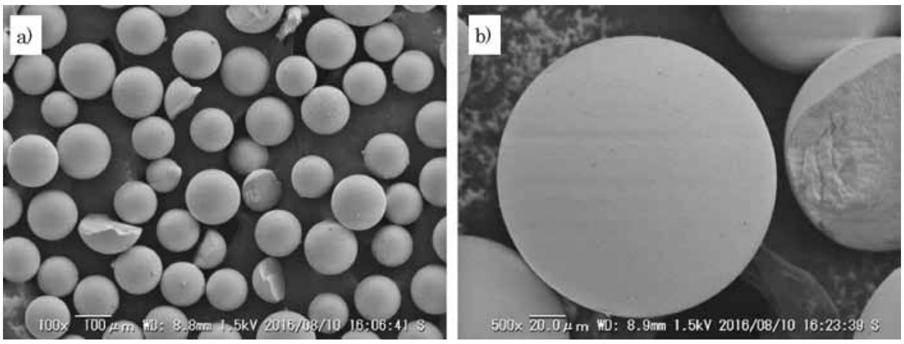

Fig. 15 SEM photo of the cation exchange resin with 90% cross-linkage with two magnifications.

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