Overview of Fusion Engineering Research in Japan Focusing on Activities in NIFS and Universities

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Abstract

This paper provides an overview of Japanese fusion engineering research activities focusing on those being carried out by NIFS and Japanese universities (Universities). NIFS is promoting the Fusion Engineering Research Project as one of the three research projects. The majority of the activity in the project is being carried out by collaboration with Universities. Utilizing core facilities installed in NIFS and the unique infrastructures of Universities, collaboration between NIFS and Universities is performed in superconducting magnet, liquid breeder blanket, advanced materials, high heat flux components, and tritium safety. NIFS also carries out international collaboration programs, such as Japan-China, Japan-USA, and IEA-based collaborations, promoting participation of University researchers. Responsibility division with QST, and contributions to ITER-BA and "Action Plan toward DEMO Development" are also reported.

Key Words : superconducting magnet, liquid breeder blanket, advanced materials, high heat flux components, tritium safety

I. Introduction

Japanese fusion energy development program and status were reported in Ref. 1. The program is now in its third phase basic program, where ITER is defined as the core device. In the fourth phase program, construction of DEMO reactor will be the core activity. In addition to the ITER project, complementary activities which are necessary to proceed to DEMO are defined and promoted as BA (Broader Approach) program. These include JT-60SA Tokamak research, and engineering research such as materials, blanket/divertor, safety, and other issues. Japanese program also takes other approaches for fusion reactor development. Although Tokamak is the first candidate for DEMO, Japan is promoting Helical and Inertial Fusion research with high priority (Alternative Options), which are presently carried out as academic research but are recognized to have the potentiality to proceed to the developmental phase toward power reactors. Japan also promotes engineering research which is not directly aiming to contribute to DEMO, e.g., fundamental/academic research, research oriented to long-term advanced technology, and research specific to the Alternative Options.

The major organizations responsible for fusion research in Japan are (1) National Institute for Fusion Science (NIFS), (2) Japanese universities (Universities), (3) National Institutes for Quantum and Radiological Science and Technology (QST), and (4) other institutes and industries. NIFS is an Inter-University Research Institute whose principal role is to promote collaboration with Universities. NIFS and Universities are responsible for the Alternative Options by promoting LHD (Large Helical Device) and FIREX Projects. The research by NIFS and Universities is relatively fundamental or academic in orientation. QST is the National Research and Development Agency, which carries out relatively development oriented research. QST is leading ITER and BA program as their implementing agency. BA is composed of IFERC (International Fusion Energy Research Center), Satellite Tokamak Program (JT-60SA), and IFMIF/EVEDA (International Fusion Materials Irradiation Facility/Engineering Validation and Engineering Design Activity) projects. Engineering research in the three projects are summarized in Ref. 2.

Another key program toward DEMO is ITER-TBM (Test Blanket Module) project, for which the Japanese program proposes and promotes a water cooled ceramic breeder concept. R&D for Japanese ITER-TBM is summarized in Ref. 3. It is commonly recognized in Japanese community that the core facility necessary in the near future is a neutron irradiation facility for testing fusion reactor materials. Based on the outcome of IFMIF/EVEDA project, a reduced size of the neutron irradiation facility, A-FNS, is under design[4]. Materials development strategy, including standardization and licensing procedure, was presented which closely corresponds to the development schedule of the neutron irradiation facility[5].

As for the research strategy toward DEMO, "The Action Plan toward DEMO Development" (Action Plan) was presented in 2016 and amended in 2017 by "The Task Force of Integrated Strategy for DEMO Development" (Task Force), established under the Fusion Science and Technology Committee in the Ministry of Education, Culture, Sports, Science and Technology (MEXT)[6]. The Action Plan is composed of the following 15 categories.

0. Demo Design

1. Superconducting Coils
2. Blanket
3. Divertor
4. Heating and Current Drive Systems
5. Theory and Simulation
6. Core Plasma
7. Fuel System
8. Fusion Materials, Standards and Codes
    - Structural Materials for Blanket
    - Other materials
    - Fusion Neutron Source
9. Safety
10. Availability & Maintenance
11. Diagnostics & Control
12. Cooperation with Society
13. Helical System
14. Laser Fusion

Action Plan defined the following critical decision points :

(1) First check and review around 2020.

(2) Second check and review within a few years from 2025.

(3) Decision of transition to the DEMO construction stage after D-T operation of ITER (2030s).

The distribution of the efforts to be provided will be discussed again based on the results of the first and the second check and review. All the achievements including those of ITER D-T operation will be reviewed for making the decision regarding the transition to the DEMO construction phase. Recently, this policy based on the Action Plan was summarized as “A Roadmap toward Fusion DEMO Reactor”[7].

NIFS and Universities take a different direction for fusion engineering research, e.g., fundamental/academic research, research oriented to long-term advanced technology, and research specific to the Alternative Options, while contributing to ITER and BA in the fusion engineering aspects. As to the DEMO research based on the Action Plan, NIFS and Universities are constructing a collaboration research framework by which sound contributions to achieving the goal defined by the Action Plan should be made. NIFS and Universities have a significant responsibility in education, as well.

The strategy of Japanese fusion research is schematically shown in Fig. 1, including responsibility of the research organizations. An overview of Japanese fusion engineering research focusing on ITER, BA and DEMO research is available[8]. This report mainly overviews the NIFS and University research for fusion engineering.

II. Fusion Engineering Research Project of NIFS (NIFS-FERP) and Collaboration with Universities

NIFS is operating LHD as the core facility of NIFS and Universities[9]. Through the construction and operation of LHD, collaboration in fusion technology by NIFS and Universities has been highly promoted particularly in superconducting magnet, plasma-facing component technology, and helical reactor design. The Fusion Engineering Research Center was established in 2001 for promoting the collaboration on long-term fusion engineering research such as low activation materials and liquid blankets. Since 2010, the fusion engineering research activity in NIFS was reorganized and extended as Fusion Engineering Research Project (NIFS-FERP), which is one of the three research projects in NIFS together with Large Helical Device Project and Numerical Simulation Reactor Research Project. The majority of the activity in these projects is being carried out by collaboration with Universities. NIFS started D-D experiments of LHD on March 7, 2017[10], which have highly enhanced neutronics and tritium related research.

II-A. Helical fusion reactor design activity

The conceptual design studies on helical fusion reactors have been carried out in NIFS since 1994. The recent design, FFHR-d1, is four times as large as LHD, with the major radius of the helical coils of 15.6 m and the toroidal magnetic field of 4.7 T[11].

Some innovative concepts are being adopted as options to overcome the difficulties related with the construction and maintenance of 3-D complicated large structures, to enhance the passive safety, and to improve the plant efficiency[12]. The innovative options include (1) the High-Temperature Superconductor (HTS), including potential use of segmented coils, as a counter option to the cable-in-conduit conductors with Low-Temperature Superconducting (LTS) Nb_3Sn strands, (2) the liquid blanket option with molten salt (FLiBe, FLiNaBe, FLiNaK) or liquid metals, (3) use of advanced low activation materials such as vanadium alloys or ODS (Oxide Dispersion Strengthened)-RAFM (Reduced Activation Ferritic/Martensitic) steels, (4) advanced solid divertors composed of improved W alloys, Cu alloys, and their interfaces produced by advanced joining technology. As new options, liquid metal limiter/divertor and cartridge-type blankets[13] have been investigated. The options being considered are schematically shown in Fig. 2.

The conceptual design study closely interacts with the engineering research and development activity. For example, the neutronics analysis of radiation environment of in-vessel components supplied the performance of the components to be targeted[14].

II-B. Major fusion engineering facilities in NIFS and their collaborative use

A large bore (700 mm in diameter) and high field (13 T) magnet test facility[15], a 9-T split coil, 75-kA DC power supply, and temperature-variable (4-50 K) refrigerator[16] have been installed for testing various superconducting conductor and coil samples. These facilities have been used for testing the performance of short HTS conductor samples[17], the ITER-TF joint samples[18], and the JT-60SA CS model and module coils[19].

For testing the performance of liquid breeders, a twin loop with 3T superconducting magnet (Oroshhi-2) was installed and is capable of flowing LiPb (~623 K) and FLiNaK (~773 K) with a maximum velocity of 1.5 m/s in a duct of 25 mm in diameter[20]. Sub-loops have been constructed for testing hydrogen transportation in LiPb and compatibility of materials with FLiNaK. Collaborations carried out with Universities include MHD (Magnetohydrodynamics) pressure drop of Li-Pb flow in double-bended pipe[21] and Ultrasonic Doppler Velocimetry (UDV) for Li-Pb[22].

A total of eight thermal creep test machines which are capable of testing small tensile specimens to the maximum temperature of 1173 K in a high vacuum were installed[23]. Creep properties were investigated for RAFM[24], ODS-RAFM[25], and advanced vanadium alloys[26]. Recent major efforts are directed to long-term creep properties for high purity V-4Cr-4Ti alloys, demonstrating that reduction of interstitial impurities from V-4Cr-4Ti enhanced workability and weldability without degrading high temperature creep properties[23].

Hot Isostatic Pressing (HIP) facility, with a maximum temperature of 2273 K, a maximum pressure of 196 MPa, and the dimension of the processing area of 120 mm in diameter and 240 mm in height, was installed in NIFS. Combination of this facility with a planetary ball-milling machine for Mechanical Alloying (MA) and a capsule welding equipment installed in a glovebox made it possible to process without exposing the materials to air, thus enabling precise impurity control of the materials[27]. The major focus of the use of these facilities are fabrication of Dispersion Strengthened (DS) Cu alloys for application of heat sink materials of divertors[28], and joining of Cu alloys with W[29].

High heat load test machine ACT2 (Active Cooling Teststand 2), with 40 keV and ~300 kW electron beam, has been used for characterizing high heat flux test components and materials[30]. Recent experiments include thermal performance of a divertor mock-up with DS-Cu and W using improved brazing technique[31] and characterization of deuterium retention performance of re-solidified W[32].

A Tandem Accelerator of 1 MeV terminal voltage, which is capable of producing H and He ions, has been used for surface analyses such as RBS (Rutherford Backscattering Spectrometry), ERD (Elastic Recoil Detection), and PIXE (Particle Induced X-ray Emission). An example of the results was reported for surface W layer analysis of ion irradiated RAFM by RBS[33].

Some other general materials test facilities are available in NIFS. In particular, FETEM-EDS (Field Emission Transmission Electron Microscope-Energy Dispersive X-ray Spectrometer), FIB

(Focused Ion Beam), GD-OES (Glow Discharge Optical Emission Spectrometer), Imaging Plate system, and TDS (Thermal Desorption Spectrometer)[34] have been installed in the radiation-controlled area of LHD site, allowing characterization of the specimens exposed to D-D plasma of LHD. This activity is promoted mostly by the Large Helical Device Project in NIFS. The major facilities installed in NIFS by FERP and the collaboration with Universities are listed in Table 1. The photographs of the facilities are collected in Fig. 3.

III. Fusion Engineering Research by NIFS and Universities

Universities carry out unique fusion engineering research by using their own respective infrastructures. Table 2 lists examples of the facilities in Universities being used for fusion engineering research, and Fig. 4 is the collection of the facilities. This chapter mostly reports the research aiming at advanced options for fusion reactors and taking fundamental/academic approaches carried out by Universities or through collaboration by Universities with NIFS.

III.A. HTS magnet

NIFS has carried out through collaborations with Universities examinations of the feasibility of HTS coils, design of HTS magnets[35], and fabrication and testing of short samples[17]. HTS coils have also been developed for use in small plasma experimental devices[36]. AC loss of HTS coils wound with stacked tape conductors were also investigated[37].

As a challenging option for employing the HTS magnet, the "joint-winding" method has been proposed, in which prefabricated segmented-conductors, having the one-helical-pitch length, are connected in order to facilitate the on-site winding process of the helical coils[35]. As an advanced option, the "remountable" magnet method is also being investigated, and the joint resistance including its bending strain dependence was characterized[38,39]. The continuous winding of the HTS helical coils has been investigated as the key technology as a back-up option, as well[40].

III.B. Liquid breeder blanket

Li-Pb, Li, and molten salts (FLiBe, FLiNaBe, and FLiNaK) are candidates for the liquid breeding materials and have been the major interest for NIFS and Universities. Fundamental studies, i.e., compatibility with structural materials for Li-Pb[41] and Li[42], hydrogen isotope transport for FLiNaBe[43] and Li-Pb[44], and recovery from Li-Pb[45] have been carried out. As for the thermofluid of molten salt, a simulant loop provided valuable information such as the means to enhance the heat transfer in cooling channels[46]. Development of functional materials for liquid breeder blanket application, such as tritium permeation barrier coating[47,48] and advanced shielding materials[49], have also been carried out by Universities.

As a new and advanced option of the liquid breeder blanket system, Ti powder-doped FLiNak is investigated for improved tritium and corrosion control[50].

III.C. Advanced materials

For the development of low activation structural materials, vanadium alloys, SiC/SiC composites, and ODS-RAFM steels, recognized as candidates of advanced low activation structural materials, have been actively studied in NIFS and Universities, in contrast to the development of RAFM steels in which QST is taking the leading role. Recent overview articles for vanadium alloys[51,52], SiC/SiC composites[53], and ODS-RAFM steels[54] are introduced for showing recent activities. NIFS has fabricated vanadium alloys (high purity V-4Cr-4Ti)[55] and ODS-RAFM steels (9Cr and 12 Cr-ODS steels)[56] for collaborative use by Universities. Recent focus in the NIFS collaborations for these materials includes joining of the advanced materials with conventional structural materials for ODS-RAFM steels[57], vanadium alloys[58], and SiC[59]. Irradiation effects are critically important research subjects. Ion accelerators installed in Universities have been used for qualifying the irradiation response of the candidate materials[60,61,62], in addition to elucidating the fundamental radiation damage process in fusion conditions.

W and Cu alloys attract attention as plasma-facing and heat sink materials for application to solid divertors, respectively. Although a large amount of research is being carried out for characterizing W alloys in fusion divertor or fast wall conditions, as will be introduced in section III.D, alloy development effort is rather limited for W alloys. Based on the NIFS collaboration, a series of W alloy materials (with doping with K and alloying with Re) were fabricated, and the characterization was carried out for tensile properties[63] and recrystallization behavior[64].

Cu-base alloys are attractive heat sink materials for divertors. However, for application to fusion reactors, radiation resistance and high temperature strength must be improved. NIFS started collaboration with Universities for developing new Dispersion Strengthened Cu-alloys based on MA-HIP processes, in which various advancements are being made for the production processes[65,66,67].

III.D. Plasma-wall interaction studies

Universities are active research bodies for fundamental plasma-wall interactions using their infrastructure. Among the examples are the effects of re-deposition layer[68], He implantation[69], and displacing irradiation[70] on hydrogen isotope retention in plasma-facing materials. Universities are operating linear plasma simulators, which are quite valuable tools for elucidating the basic process of plasma-wall interactions[71,72].

Characterization of materials exposed to LHD plasma has been carried out by NIFS collaborations, such as microstructure[73], hydrogen retention[74], and W-fuzz structure (nanostructure)[75,76].

A unique effort being made under the Project is that the Compact Divertor Plasma Simulator (CPDS) was installed in the radiation-controlled area of the International Research Center for Nuclear Materials Science (the Oarai Center), Institute for Materials Research, Tohoku University[77]. This has made it possible to study plasma-materials interaction using the materials previously irradiated with neutrons and now being stored in the Oarai Center[78,79].

III.E. Liquid divertors

Liquid divertor is a very advanced option for handling the extremely high heat load. Fundamental studies have been carried out such as thermofluid of the free surface flow in a magnetic field[80] and deuterium transport under plasma exposure[81]. Helical Reactor Design in NIFS adopted a liquid divertor option[82]. Experimental verification of the concept is being carried out[83].

III.F. Tritium

A wide range of tritium studies have been carried out in Universities. University of Toyama has the Hydrogen Isotope Research Center, which functions as the base for collaboration with Universities for tritium research. NIFS supports the Center based on the Bilateral Collaboration Research Program. The researches carried out at University of Toyama by collaborations with Universities include tritium performance in plasma-facing materials, detection and decontamination for tritium control, and fundamentals of biological effects of tritium[84].

In addition to the collaboration with Universities via the University of Toyama, research is carried out by NIFS collaboration in relation with the D-D operation of LHD, including device development for monitoring[85], de-tritiation[86], recovery[87], and non-destructive retention measurements[88] for potential application to the tritium control of LHD.

III.G. Inertial fusion oriented engineering studies

In Japan, the research on Inertial Fusion Energy (IFE) is of high priority, and conceptual design of laser inertial fusion reactors and the relevant research have been carried out mainly in Universities. There are a number of technical issues for the reactors. Some are specific to the IFE systems and some are rather common to magnetic confinement fusion systems[89]. The former includes the target technology[90] and the latter includes the chamber technology such as tritium recovery[91], compatibility[92], and liquid metal flow[93].

IV. International Collaborations

Japan-USA Joint Project is one of the key international collaboration activities by NIFS and Universities for fusion engineering research. Summaries of the past projects are given in Table 3. Overviews of past projects are available for RTNS-II[94], FFTF/MOTA[95], JUPITER[96], JUPITER-II[97], and TITAN[98]. Until recently the "PHENIX" Program (FY2013-FY2018), which was focused on materials, tritium, and thermofluid issues for gas-cooled divertors, was carried out[99]. The subsequent project "FRONTIER" Program (FY2019-FY2024) was launched as of April 2019, in which the interface issues for the solid and liquid divertors are the major research targets[100].

Throughout these series of the project, neutron irradiation effects are the core research activity, in which materials delivered from Japan were irradiated in reactors in the United States, and were partly shipped back to Japan, mainly to the Oarai Center for Post-Irradiation Examinations (PIE) using Japanese unique infrastructure[77]. From the TITAN project, some specimens irradiated in HFIR (High Flux Isotope Reactor) in ORNL (Oak Ridge National Laboratory) were shipped to STAR (Safety and Tritium Applied Research) in INL (Idaho National Laboratory) for tritium transfer or plasma exposure experiments. These unique efforts have produced unprecedented results, which were reported in the overview articles[94-99].

As for the Japan-China collaboration, the activity is decreasing slightly after the conclusion of the JSPS-CAS Core University Program (CUP) and JSPS/NSFC/NRF A3 Foresight Program (Japan-China-Korea). Especially in the CUP Program, a wide range of collaboration was carried out in fusion engineering, such as reactor design, superconducting magnet, materials, plasma-wall interaction, and tritium/blanket[101]. Collaboration is being carried out presently based on

MEXT/MOST Agreement Program (JWG) or on NIFS internal budget (named Post-CUP Program)[102]. Recent examples of the collaboration in fusion engineering research include plasma-wall interactions in EAST Tokamak[103].

Japan-Korea collaboration has been mostly based on the A3 Foresight Program, and the major focus is on KSTAR Tokamak. Personnel exchange and workshops have been the major activities in fusion engineering. A collaboration on plasma-wall interactions in KSTAR has recently started[104].

Under the IEA framework, the PWI-Technical Collaboration Program (TCP) has been carried out from 2015 as a successor of IEA TEXTOR Implementing Agreement (IA). NIFS, as the representative institute of Japan, is budgeting PWI-TCP for supporting participation of Japanese university researchers[105]. Plasma-wall interactions have been investigated under this program using the linear plasma devices in the world[32,106,107].

V. Contribution to ITER-BA

NIFS and Universities contribute to ITER and BA for fusion engineering issues. Examples include direct contribution to ITER and BA such as performance tests of the ITER-TF joint samples[18] and the JT-60SA CS model and module coils at NIFS[19]. NIFS and Universities have collaboration contracts with QST contributing to BA. The collaboration in the fusion engineering includes reactor design, blanket, structural and functional materials, and tritium technology in the IFERC Project, and the Li target in the IFMIF-EVEDA Project.

Among the papers published from NIFS and Universities for the BA-IFERC Project collaboration are aging of RAFM[108], small specimen test technology[109], non-destructive evaluation[110], and joining with dissimilar materials[111] for structural materials research, recovery from Li-Pb[112] and permeation in W[113] for tritium research, fracture[114], deuterium permeation[115], and radiation-induced electrical properties[116] for SiC material, hydrogen retention[117] and interaction with steam[118] for Be-alloys, and structural change[119], and vaporization[120] for solid breeding materials. A unique effort being carried out in this project is the characterization of JET ITER-Like Walls (ILW), in which samples extracted from JET in-vessel components were transferred to the International Fusion Energy Research Centre (IFERC), QST, Rokkasho, Japan for subsequent analyses by several scientists in Japan including NIFS and University researchers[121,122].

NIFS and Universities have contributed significantly to the engineering validation of IFMIF for the Li target and the test cell systems. In particular, since the control of flow and the chemistry of Li is the commonly necessary technology to the liquid Li breeding blankets and divertors, infrastructure and experiences in Universities such as Li loop flow test[123], deuterium[124] and nitrogen[125] trapping technology were efficiently used in the collaboration. Also carried out was the analysis of the High Temperature High Flux Test Modules[126].

VI. Examples of the latest and ongoing researches

Finally this paper outlines some of the latest and ongoing researches by NIFS collaboration with Universities and international partners, most of which have not been published in open journals.

VI-1. Tests of HTS conductors

A 2-m HTS conductor named TSTC (Twist Stack Tape Cable)[127], developed at Massachusetts Institute of Technology (MIT), was installed into the large bore (700 mm in diameter) and high field (13 T) magnet test facility[15] in NIFS, and the tests were launched in September 2018. The tests have been carried out without and with application of magnetic field of 5T at temperature of < 30 K.

VI-2. Thermal mixing of MHD liquid metal free-surface film flow

Thermal mixing of liquid metal free-surface film flow has been investigated as a liquid divertor thermofluid study using LMX (Liquid Metal eXperiment) of Princeton Plasma Physics Laboratory (PPPL)[80]. Based on the experience, Kyoto University constructed LMFREX (Liquid Metal FRee-surface EXperiment) and installed to the 3T magnet in Oroshhi-2 loop test facility in NIFS. Thermal mixing of the liquid metal free-surface flow in the vertical magnetic field with the length of ~300 mm is being investigated, including participation of PPPL for verifying an idea of active flow control by conducting electricity.

VI-3. Redevelopment of W-TiC materials

Japanese University research groups have devoted to development of ultra-fine grained W-TiC dispersion strengthened alloys by MA-HIP processes for application to devertor materials[128]. The alloys produced have shown enhanced ductility and radiation resistance. These efforts were, however, suspended for years. With the start of the use of MA and HIP facilities in NIFS, the activity is resumed recently, which includes new effort to systematically investigate the MA conditions and to process the powders after MA with CIP (Cold Isostatic Pressing) prior to the HIP processing.

VI-4. Low activation vanadium alloys recyclable after cooling for ten years

V-4Cr-4Ti has long been a leading candidate for the low activation vanadium alloys. As a further advanced option, potential reduction of Ti level, which is the major source of radioactivity after 25 years' cooling, is investigated. According to the neutronics calculation, showing that recycling and reuse after 10 years' would be possible by reducing Ti level below 1%, the alloy composition is being re-optimized by increasing Cr level for compensating mechanical property degradation by the decrease in Ti level below 1 %. Strict control of impurities which can produce long radioactivity by exposure to neutrons is also necessary.

VI-5. Hydrogen recovery from PbLi droplets in vacuum

Feasibility analysis for tritium recovery from liquid PbLi, in the form of an array of droplets in multiple columns, was carried out[45]. Based on the results, a vacuum sieve tray chamber, composed of a droplet formation nozzle, hydrogen dissolution units, heating units, and evacuation system, was constructed and its performance was tested[129]. The system was then installed into Oroshhi-2 loop test facility in NIFS. Continuous hydrogen recovery tests using the Li-Pb loop in Oroshhi-2 will begin in 2019.

VII. Summary

An Interuniversity Research Institute, such as NIFS, is a unique research organization of Japan for promoting “Big Science”, providing University people, including students, opportunities to use relatively large facilities which are unaffordable for individual Universities. In fusion engineering, collaborative use of the core facilities in NIFS and unique infrastructure in Universities make it possible to perform unique research, while promoting the education of students and young scientists. Such activity also contributes to constructing a network of Universities, thus enhancing various types of collaboration. NIFS also carries out international collaboration programs, such as Japan-China, Japan-USA, and IEA-based collaborations, promoting participation of University researchers.

In the Japanese program, QST is responsible for promoting ITER, BA and DEMO oriented research. NIFS and Universities are, while contributing to those programs, in charge of academic/fundamental research oriented to advanced options of fusion reactors, such as HTS magnet, liquid breeding blankets, advanced low activation materials, advanced plasma-facing components, and the engineering research specific to the Alternative Options of Tokamak, which are recognized to have the potentiality to proceed to the developmental phase toward power reactors. The Japanese fusion engineering research, in this way, maintains variety and robustness.

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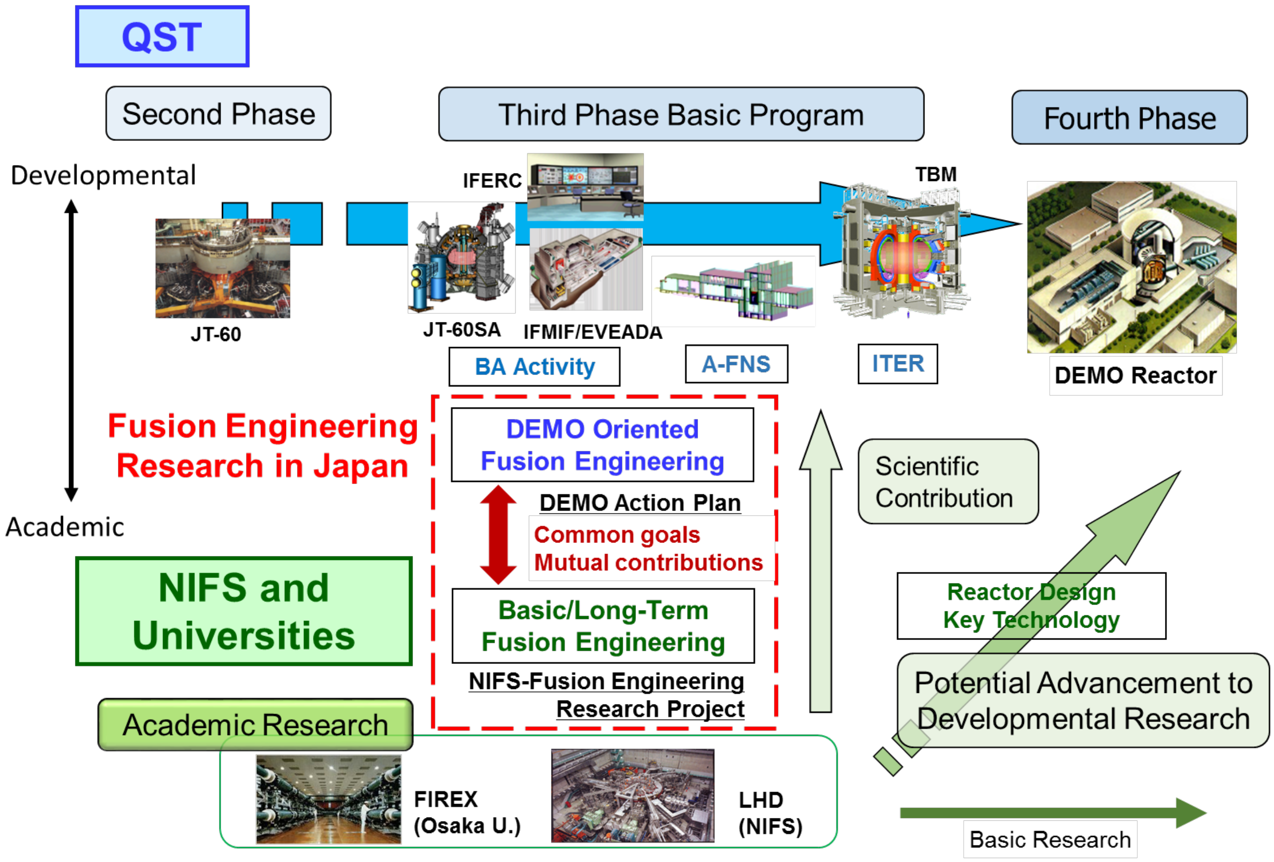
Figure Captions

Fig. 1 The strategy of Japanese fusion research, including responsibilities of the research organizations. (modified from Fig. 1 in Ref. 1 and http://www.aec.go.jp/jicst/NC/senmon/kakuyugo2/siryo/kettei/houkoku051026_e/betten20.htm highlighting fusion engineering research)

Fig.2 Some options being considered in the design of FFHR-d1[12].

Fig. 3 Core fusion engineering facilities in NIFS.

Fig. 4 Examples of fusion engineering research facilities in Universities.



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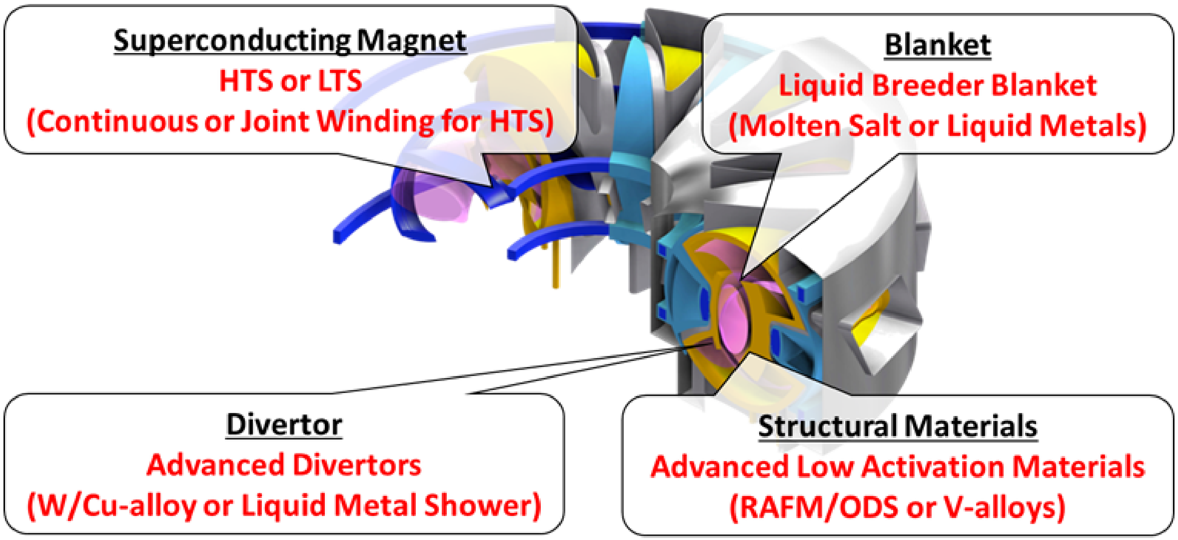

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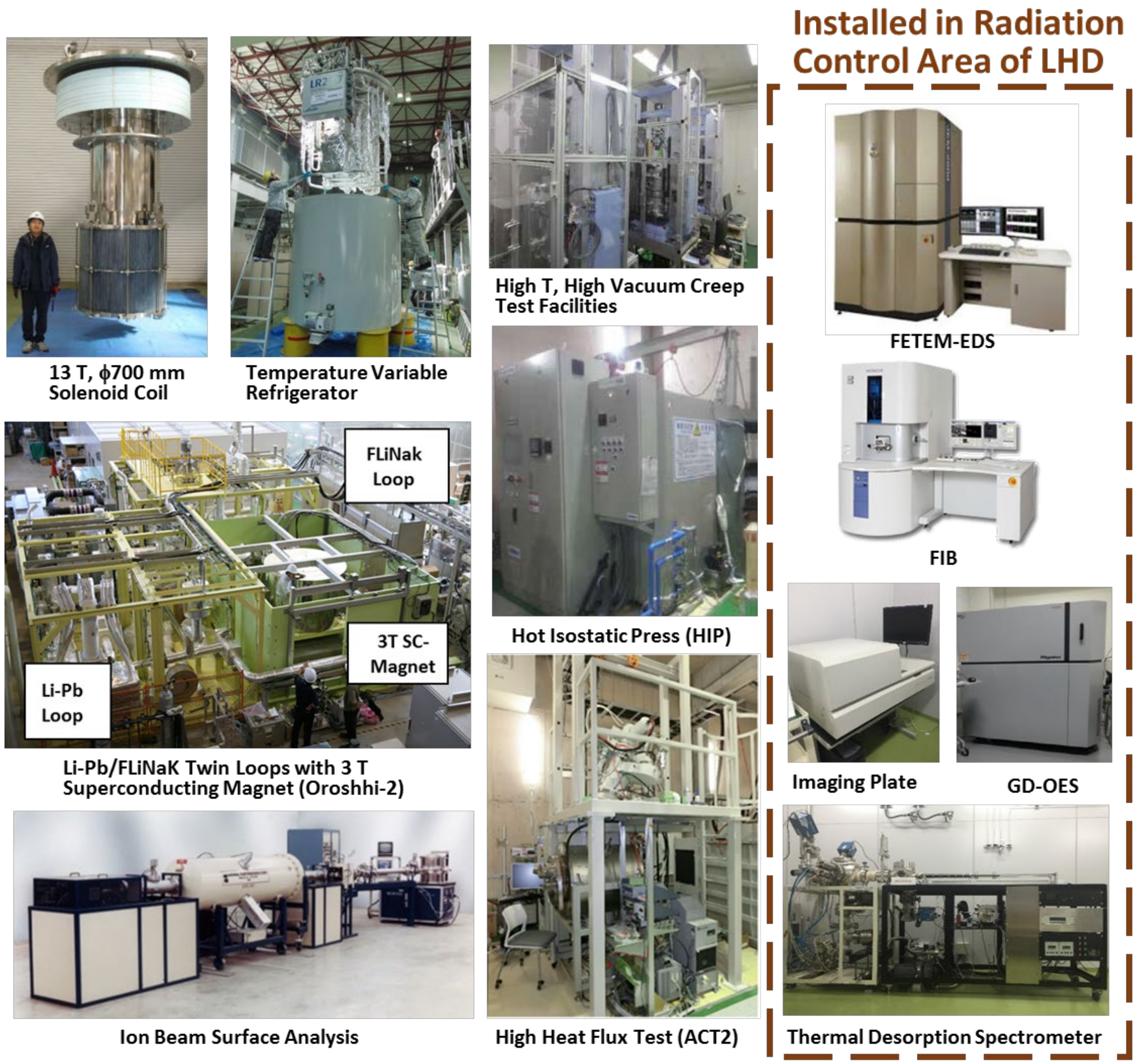

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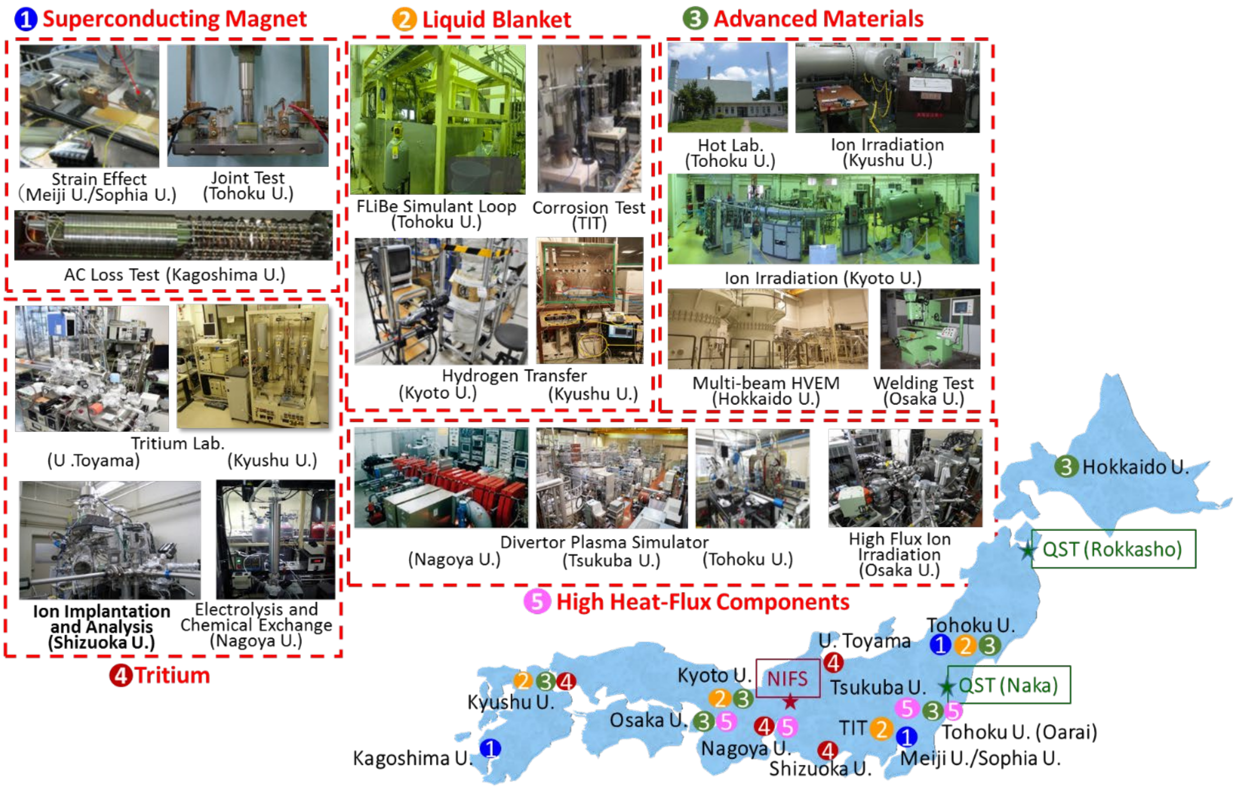

Fig. 4 Examples of fusion engineering research facilities in Universities.

Table 1. Research subjects, key facilities and collaborations in Fusion Engineering Research Project of NIFS (NIFS-FERP).

Research Subjects	**Major Emphases**	**Key Facilities in NIFS**	**Examples of collaborative research**
Superconducting magnet and systems	High temperature superconducting magnet	13 T, ϕ700 mm Solenoid Coil[15], 9T Split Coil, Temperature Variable Refrigerator[16]	Short HTS sample tests[17] ITER-TF Joint Sample tests[18] JT-60SA Central Solenoid tests[19]
Blanket	Liquid breeder blanket	Li-Pb/FLiNaK Twin Loops with 3 T Superconducting Magnet (Oroshhi-2)[20]	MHD pressure drop of Li-Pb[21] Ultrasonic Doppler Velocimetry for Li-Pb[22] Mass transfer control in FLiNaK[50]
Materials	Advanced low activation and divertor materials	High T, High Vacuum Creep Test Facilities[23] Hot Isostatic Press Facility[27]	Creep properties of V-4Cr-4Ti alloys[26] Production of ODS Cu alloys[28] HIP joining of advanced materials[29]
High heat flux components	Solid and liquid divertors	High Heat Flux Test (ACT2)[30] Ion Beam Surface Analysis[33]	High heat flux test of W/Cu-alloy joints[31] Properties of re-solidified W[32] RBS for ion-implanted RAFM[33]
Tritium	Control of low level tritium	Tritium system for LHD[10] Thermal Desorption Spectroscopy[34]	Application of proton conductors[85] Water detritiation[86,87] Non-destructive retention measurements[88]

Table 2. Examples of infrastructure in Universities used for fusion engineering research.

Research Field	Examples of infrastructure in Universities
Superconducting magnet	HTS tape joint test facilities (Tohoku U.)[39] AC loss measurement facilities (Kagoshima U.)[37] Strain effect test facilities (Meiji U./Sophia U.)[40]
Blanket	Molten-salt simulant test loop (Tohoku U.)[46] H/D transfer test facilities (Kyushu U./Kyoto U.)[44,45] Corrosion test facilities (Tokyo Inst. Tech.)[41]
Materials	Hot laboratories (Tohoku U.)[77] Ion irradiation test facilities (Kyoto U./ Kyushu U.)[60,61] Multi-beam High Voltage Electron Microscope (Hokkaido U.)[62] Welding test facilities (Osaka U.)[57]
High heat flux components	Divertor simulator (Nagoya U./Tsukuba U.)[71,72] Divertor simulator in a hot laboratory (Tohoku U.)[78] High flux ion implantation facility (Osaka U.)[33]
Tritium	Tritium laboratory (U. Toyama/Kyushu U.)[84,87] Ion implantation and analysis (Shizuoka U.)[69] Electrolysis and chemical exchange (Nagoya U.)[86]

Table 3. Grand summary of Japan (NIFS/Universities)-USA Joint Projects.

RTNS-II, Rotating Target Neutron Source-II; FFTF, Fast Flux Test Facility; EBR-II, Experimental Breeder Reactor-II; HFIR, High Flux Isotope Reactor; STAR, Safety and Tritium Applied Research; MTOR, Magneto-Thermofluid Omnibus Research Facility; TPE, Tritium Plasma Experiment; PISCES, Plasma Interactive Surface Component Experimental Station; PAL, Plasma-arc Lamp facility; LAMDA, Low Activation Materials. Development and Analysis, LLNL, Lawrence Livermore National Laboratory; ANL, Argonne National Laboratory; ORNL, Oak Ridge National Laboratory; INL, Idaho National Laboratory; BNL, Brookhaven National Laboratory; UCLA, University of California, Los Angeles; UCSD, University of California, San Diego; GIT, Georgia Institute of Technology.

Project Name	Period	Test Facilities	Core Subjects
RTNS-II[94]	1981~1986	RTNS-II (LLNL)	Low dose D-T neutron irradiation Defect production and accumulation Mechanical properties Functional materials
FFTF/MOTA[95]	1987~1994	FFTF (Hanford) EBR-II (ANL-West)	High fluence neutron irradiation Void swelling and mechanical properties Transmutation effects
JUPITER[96]	1995~2000	HFIR (ORNL) ATR (INL) HFBR (BNL)	Transient and variable effects of neutron irradiation Temperature variation effects In-situ conductivity of ceramics Radiation creep
JUPITER-II[97]	2001~2006	HFIR (ORNL) STAR (INL) MTOR (UCLA)	Key issues for advanced blankets Molten-salt blanket Li/V-alloy blanket He-SiC/SiC blanket
TITAN[98]	2007~2012	HFIR (ORNL) TPE/STAR (INL) MTOR (UCLA) PISCES (UCSD)	Mass and heat transfer in first wall and blanket Irradiation-tritium synergism Thermofluid in magnetic field Coating & joining
PHENIX[99]	2013~2018	HFIR (ORNL) PAL (ORNL) TPE (INL) He loop (GIT)	Evaluation of feasibility and safety of He-cooled divertors Heat transfer Tritium Transfer Tungsten materials
FRONTIER[100]	2019~	HFIR (ORNL) LAMDA (ORNL) TPE (INL)	Interface issues for solid and liquid divertors Tritium transfer and radiation effects Compatibility