Two Conceptual Designs of Helical Fusion Reactor FFHR-d1A Based on ITER Technologies and Challenging Ideas

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Abstract.

The Fusion Engineering Research Project (FERP) at the National Institute for Fusion Science (NIFS) is conducting conceptual design activities for the LHD-type helical fusion reactor FFHR-d1A. This paper newly defines two design options, "basic" and "challenging." Conservative technologies, including those that will be demonstrated in ITER, are chosen in the basic option in which two helical coils are made of continuously wound cable-in-conduit superconductors of Nb_3Sn strands, the divertor is composed of water-cooled tungsten monoblocks, and the blanket is composed of water-cooled ceramic breeders. In contrast, new ideas that would possibly be beneficial for making the reactor design more attractive are boldly included in the challenging option in which the helical coils are wound by connecting high-temperature REBCO superconductors using mechanical joints, the divertor is composed of a shower of molten tin jets, and the blanket is composed of molten salt FLiNaBe including Ti powers to increase hydrogen solubility. The main targets of the challenging option are early construction and easy maintenance of a large and three-dimensionally complicated helical structure, high thermal efficiency, and, in particular, realistic feasibility of the helical reactor.

Keywords: helical reactor, FFHR, HTS, liquid blanket, molten salt, liquid divertor,

1. INTRODUCTION

Conceptual design activities on the helical fusion reactor have been conducted at the National Institute for Fusion Science (NIFS), Japan, since 1994 [1, 3]. The first design was Force-Free Helical Reactor 1 (FFHR-1), which was equipped with three helical coils aiming at a high magnetic field for good plasma confinement with low magnetic force on helical coils [1]. The Large Helical Device (LHD) successfully started operation in 1998 [2]. The magnetic field strength at the helical coil centre, B_c, is 3T; and the helical coil major radius, R_c, is 3.9m. The next reactor design, FFHR-2 with two helical coils similar to those in LHD, was investigated, reflecting the achievements of LHD in both engineering and plasma physics. FFHR-2 has a high B_c of 10T and a relatively small R_c of 10m. Two options with smaller B_c and larger R_c, FFHR-2m1 (B_c = 6.2T, R_c = 14.0m) and FFHR-2m2 (B_c = 4.4 T, R_c = 17.3m), were also studied. Comparison with other stellarator concepts was also conducted [4].

These design activities transferred to the Fusion Engineering Research Project (FERP), which was organized in 2010 [3]. Since then, FERP has been working on the latest design, known as FFHR-d1 ("d" refers to a fusion "demo" reactor). The following four basic rules have been applied to designing FFHR-d1. (1) It should be operated in steady state without auxiliary heating (*i.e.*, self-ignition state). (2) The plasma parameters should be reasonably extrapolated from the experiment results obtained in LHD without assuming unknown plasma confinement improvement. (3) The arrangement of magnetic coils should be basically similar to that of LHD. (4) The technologies assumed in the design should be those that are already well established or are foreseen to be established in the near future. Because of the third rule, the

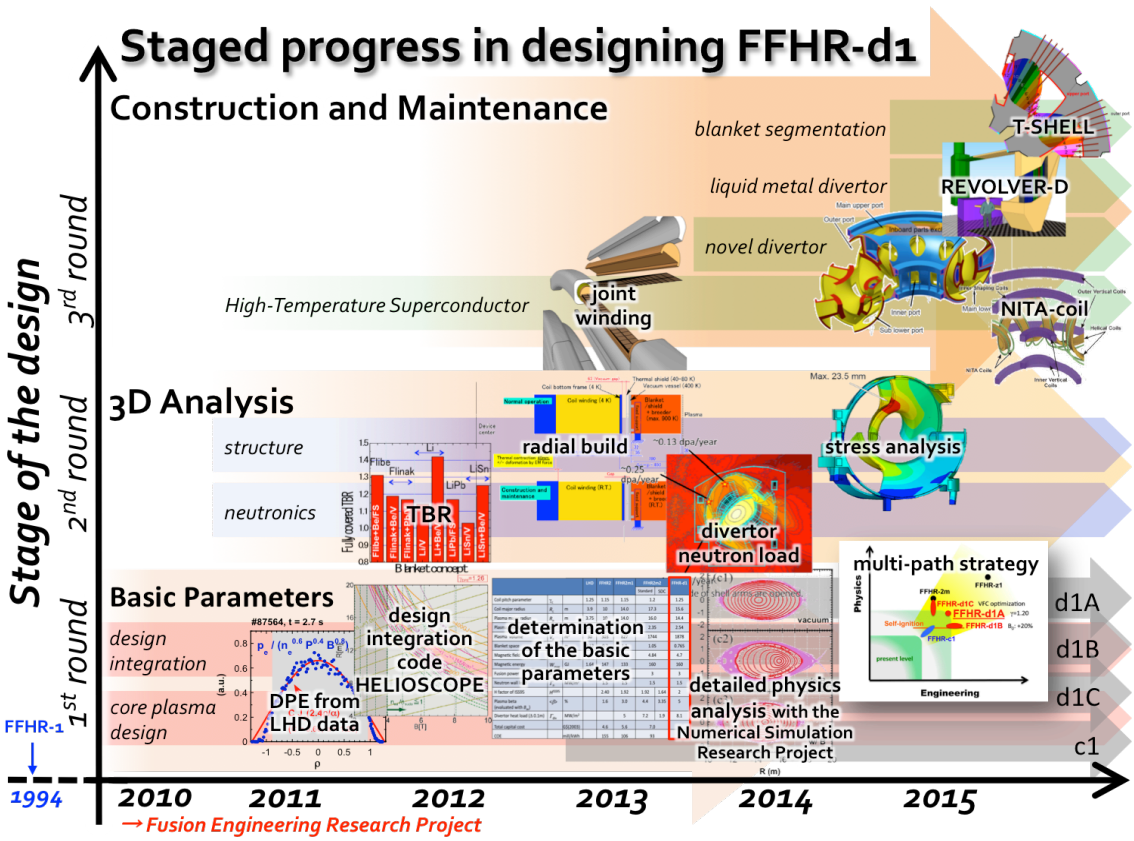

Fig. 1. Graphic view of the staged progress in designing FFHR-d1.

MHD equilibrium in FFHR-d1 is similar to that in LHD. This makes the extrapolation of plasma parameters reasonable. Figure 1 illustrates the staged progress in designing FFHR-d1. In the first stage (named "round" to mean iterative working), we started design activity based on the core plasma design. The plasma parameters are determined by the Direct Profile Extrapolation (DPE) method using the experiment data obtained in LHD [5]. The design integration code HELIOSOPE has been developed [6]. Using this code, the main parameters were selected, (*e.g.*, the device is four times larger than the LHD, and the toroidal magnetic field is 4.7T at the helical coil centre). Detailed plasma physics analyses (e.g., on particle and energy transports, MHD equilibrium and stability, neoclassical transport, and alpha particle confinement) began in 2014 and continue today. The latest study includes the bootstrap current and its effect on MHD equilibrium to obtain a self-consistent solution of density and temperature profiles [7]. In the second stage ("round"), three-dimensional (3D) design of the structures and 3D neutronics analysis were carried out [3]. Since 2015, a multi-path strategy has been taken to include various options in the design. FFHR-d1A (B_c = 4.7T, R_c = 15.6m) discussed in this study is the base option; FFHR-d1B (B_c = 5.6T, R_c = 15.6m) has a stronger magnetic field to ease the demand for plasma parameters; and for FFHR-d1C, the third basic rule (*i.e.*, use a magnetic coil arrangement similar to that of LHD) is loosened to allow flexibility in the magnetic coil design. To develop a nuclear test machine that enables a year-long neutron irradiation test, the compact helical reactor FFHR-c1 (B_c = 4.0 - 5.6T, R_c = 13.0m) is also studied. Although the first basic rule of self-ignition is omitted in FFHR-c1 to reduce the device size, it can be operated in steady state using self-generated electricity and tritium.

Now, in the third stage ("round"), the design activity focuses on construction and maintenance schemes. There is no need for current drive; therefore, plasma operation control

and steady state sustainment are relatively easy in a helical reactor. However, we must solve difficult issues related to construction and maintenance of three-dimensionally complicated large structures. In some cases, new and challenging ideas seem to offer good possibilities to solve the difficult issues. To include these ideas, we have decided to loosen the restriction of the fourth basic rule that allows no unproven technologies to be applied to the reactor design. Thus, we newly define two options of "basic" and "challenging" in FFHR-d1A design, which has the following design parameters: $n_0 = 1.5\times10^{20}$ m^{-3}, $T_0 = 16.5$ keV, $\tau_E = 1.5$ sec, $P_{fus} = 3.0$ GW, $n\tau T = 37.6\times10^{20}$ m$^{-3}\cdot$ sec$\cdot$ keV, $Q = \infty$ and TBR > 1.05. Conservative technologies including what will be demonstrated in ITER are chosen for the basic option. However, new ideas that would possibly be beneficial for making the reactor design more attractive from the viewpoints of early construction, easy maintenance, and high thermal efficiency are boldly included in the challenging option.

Table 1 compares the basic and challenging options. Superconducting (SC) magnet, auxiliary heating, divertor, and blanket have different options. Sections 2 and 3 describe details of these options. Section 4 discusses the R&D strategy for realizing the helical fusion reactor. Finally, Section 5 presents a summary.

Table 1. Comparison of basic and challenging options for FFHR-d1A.

		Basic Option	**Challenging Option**
SC Magnet $B_c = 4.7$T $R_c = 15.6$m	Specification	CICC of Nb_3Sn (or Nb_3Al)/SHe	STARS of REBCO/GHe
	Fabrication method and period	React and continuous winding with a large reel, > 6 years	Parallel works of jointing, < 3 years
	Key issue	5-in-hand continuous winding with internal plate	Demonstration of the large-scale HTS coil
Auxiliary Heating 40 to 100MW (until self-ignition)	ECH (143GHz fundamental)	w/	w/
	ICRF	w/	w/
	NBI	w/	w/o
Divertor	Material/cooling	W and Cu alloy/water	molten Sn
	Structure	Full-helical	10 positions at inner-X point as the ergodic limiter/divertor
	Key issue	Maintenance, mitigation of heat flux	Plasma irradiation under strong magnetic field
Blanket	Breeder/cooling	Ceramics/water	FLiNaBe with metal powders
	Segmentation	Helical	Toroidal
	Key issue	TBR with a limited blanket space, and maintenance	Demonstration of redox control under neutron irradiation

2. BASIC OPTION

In the basic option, the SC magnet coils adopt cable-in-conduit (CIC) conductors with Nb_3Sn (or Nb_3Al) strands cooled by supercritical helium (SHe) at 4.5K, which is an extension of ITER technology [8]. The helical coils are continuously wound by the react-and-wind method layer by layer using a large-scale winding machine, which may take over 6 years (considering that it took 1.5 years for LHD). Many other technological difficulties are associated with this option, such as how CIC conductors can be precisely bent and twisted to be installed into the helical grooves of internal plates and how the Vacuum Pressure Impregnation (VPI) can be performed by raising the whole coil temperature to 150°C after winding.

The divertor system is basically similar to those being developed for ITER (*i.e.*, a water-cooled tungsten monoblock divertor with cooling pipes made of Cu alloy). As in LHD, the entire divertor footprint is covered to form a full-helical divertor. The peak divertor heat load on this divertor is expected to exceed 20MW/m^2 because of the inhomogeneous divertor heat load profile observed in LHD. Therefore, we must develop plasma control methods for divertor heat load reduction (*e.g.*, divertor detachment and/or magnetic field optimization to make the divertor heat load uniform). Maintenance of the full-helical divertor is also a difficult issue. Therefore, a novel divertor concept has been proposed, aiming at easy maintenance of divertor plates at the inboard side of the torus, where the divertor heat load is expected to be high [9] (Fig. 2). In this configuration, the divertor plates on the inboard side are placed behind the helical coils. Ten vertical ports are provided on the inboard side of the torus for frequent maintenance of the inboard side divertor. The proposed divertor mitigates neutron irradiation on the divertor and enables the use of copper cooling pipes.

The blanket system is composed of a Neutron Shield Blanket (NSB) and a Tritium-Breeding Blanket (TBB). The TBB in the basic option will be based on the ITER Test Blanket Module (TBM) proposed by Japan (*i.e.*, a water-cooled ceramic breeder blanket). However, detailed design of the TBB has not yet been obtained. Both the NSB and the TBB will be segmented along the helical coils to form blanket modules. How to construct and maintain large and

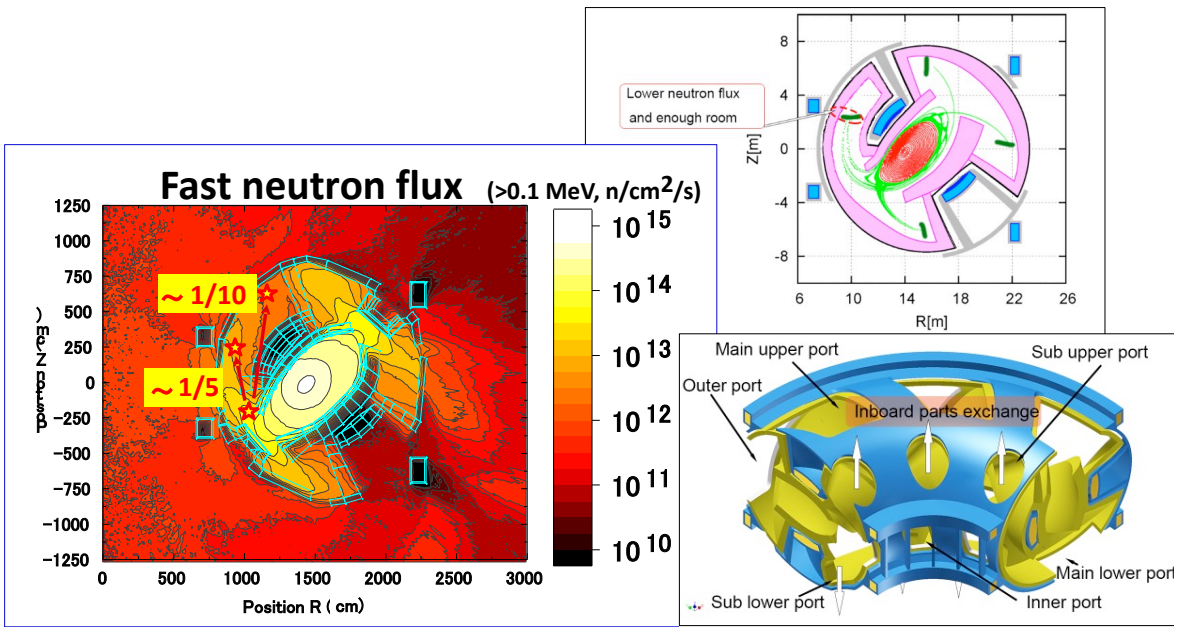

Fig. 2. Bird's-eye view of the blanket and the coil-support structure for the novel divertor configuration. Inboard parts of the divertor can be exchanged through the vertical maintenance ports on the upper inside of the torus.

complicated blanket modules also remains an open issue.

The key technologies needed for the basic option are already well established in LHD or will be established through R&D activities for ITER. However, we must develop construction and maintenance schemes for the helical divertor and blanket with large and complicated 3D structures.

3. CHALLENGING OPTION

In the challenging option, new technologies of the High-Temperature Superconductor (HTS) [10, 11], the liquid metal ergodic limiter/divertor [12], and the molten salt (FLiNaBe mixed with metal powders) breeder blanket [3, 13] are adopted to solve problems associated with a large winding machine and difficult maintenance

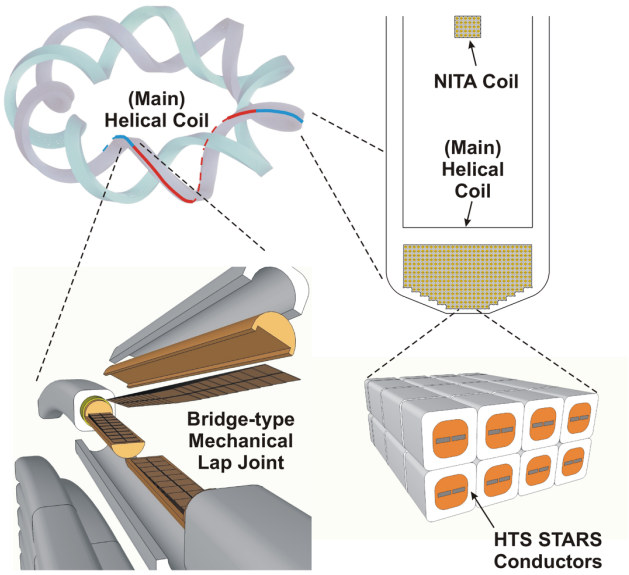

Fig. 3. Schematic illustration of the mechanical lap joint used in HTS helical coil winding. Cross-sectional view of the HTS conductor equipped with internal insulation.

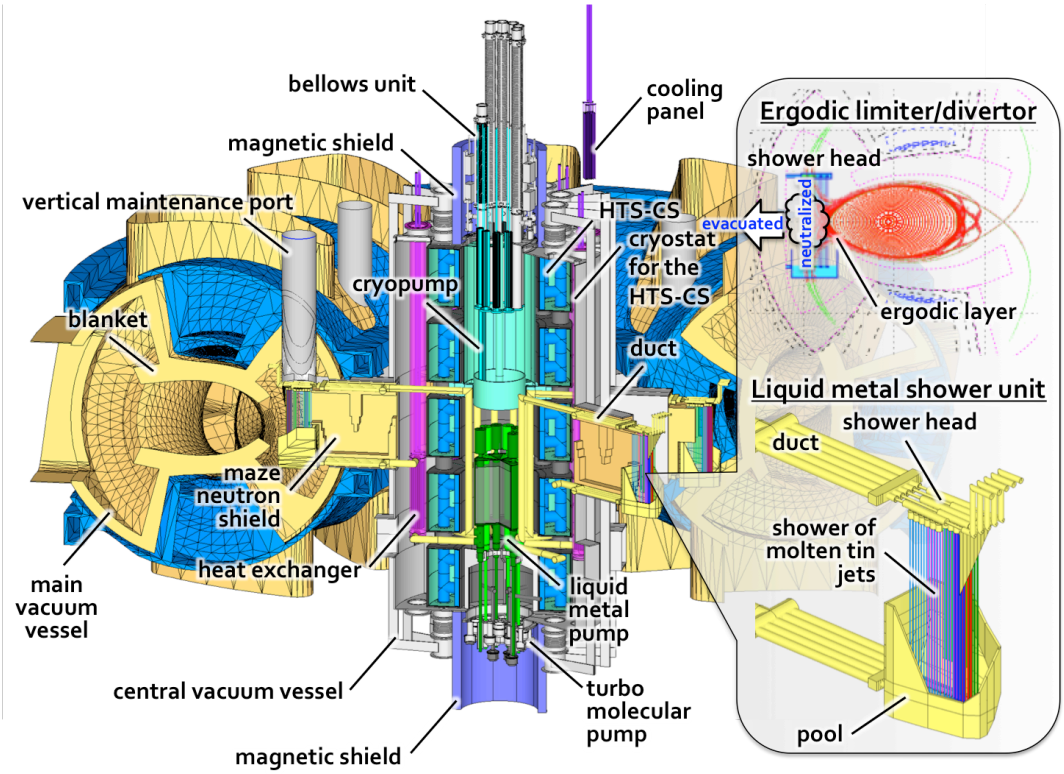

Fig. 4. Bird's-eye view of the FFHR-d1 equipped with the REVOLVER-D. A cross-sectional view of the ergodic limiter/divertor configuration. A close-up view of the liquid metal shower unit is depicted in the balloon.

of divertors and blankets.

Joint winding using the mechanical lap joint technique (Fig. 3) is applied to fabricate helical coils by connecting segmented HTS (REBCO) stacked tape assembled in rigid structure (STARS) conductors. The helical coil winding using this procedure is expected to take less than 3 years [10]. The cooling scheme is simplified by circulating helium gas (GHe) at 20K. VPI can be skipped by having internal electrical insulation in the HTS conductor and by welding neighbouring conductors in the winding package. Newly installed twist adjustment (NITA) coils [14] are supplementary helical coils added to enlarge the blanket space on the inboard side of the torus while keeping the plasma volume unchanged. All these possibilities should be realized by intense development of an HTS conductor beyond the already achieved status of 100kA at 5.3T, and 20K with a short sample.

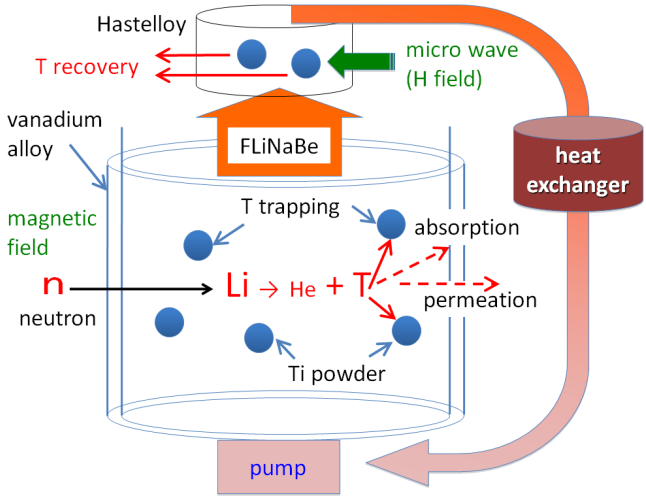

Fig. 5. Principal diagram of Ti powder mixed FLiNaBe.

A new liquid metal limiter/divertor system, REVOLVER-D, has been proposed [12] (Fig. 4). In this system, 10 units forming molten tin (Sn) shower jets stabilized by chains inside each jet are installed on only the inboard side of the torus to intersect the ergodic layer. This works as an ergodic limiter, and the conventional full-helical divertor becomes less necessary. Neutral particles are evacuated through the liquid metal shower. Maintenance can be easily performed using 10 maintenance ports similar to those proposed in the novel divertor concept.

A blanket system using metal powder mixed FLiNaBe (melting point 580K) [3] (Fig. 5) is also a challenging option. Effective increase of hydrogen solubility over five orders of magnitude has already been confirmed [13] with powders of hydrogen storage metal (e.g., Ti). Based on this observation, we assume that vanadium alloys available at over 1000K are

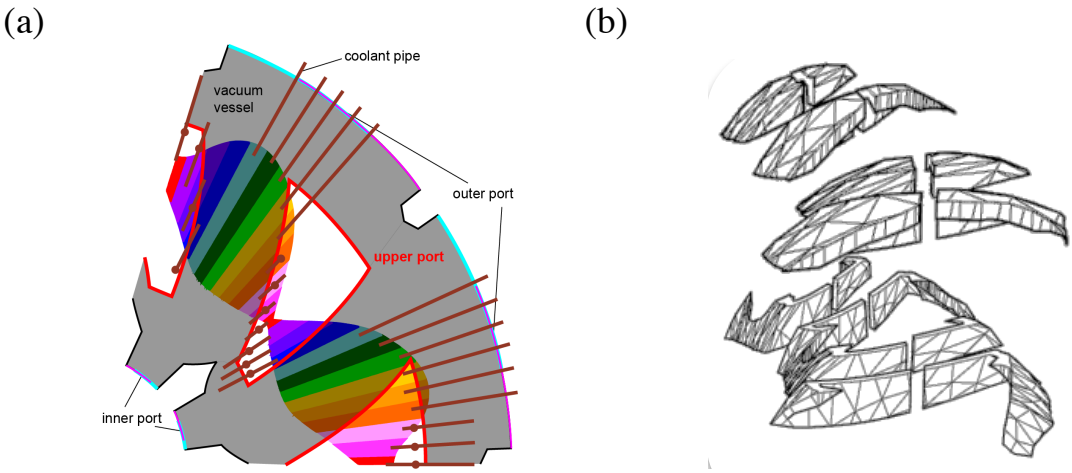

Fig. 6. (a) Top view of the T-SHELL blanket (i.e., toroidally sliced tritium-breeding blanket). (b) Bird's-eye view of the horizontally sliced neutron shield blanket.

applicable, giving higher thermal efficiency exceeding 40% compared with FLiBe/F82H, and making tritium permeation barrier coating less necessary. And there is no MHD effects.

For faster construction with high accuracy, a new type of TBB, the T-SHELL breeder blanket, has been proposed [15] (Fig. 6(a)). This T-SHELL blanket is divided at every 3 degrees of the toroidal angle. With a helically segmented blanket in the basic option, it is necessary to move the blanket units three-dimensionally inside the torus for replacement. With a toroidally segmented blanket such as the T-SHELL, the blanket unit can be replaced using a combination of uniaxial movements and poloidal rotation alone. This increases the feasibility of blanket maintenance. Also for the NSB, toroidal or horizontal segmentation (Fig.6(b)) might make construction easier than helical segmentation. Detailed scenarios of construction and maintenance, including the segmentation method and motion analysis of the blanket units for both TBB and NSB are still under discussion.

4. R&D STRATEGY

Although the new technologies adopted in the challenging option might significantly ease construction difficulties in the basic option, they are not necessarily well established at this moment. The R&D strategy in terms of Technology Readiness Level (TRL) is summarized in Fig. 7. Technologies needed for the basic option are already being developed in heliotrons, stellarators, tokamaks, and linear machines worldwide. These technologies will finally achieve TRL 6 in ITER. However, it is necessary to encourage or start R&D activities to increase the TRL of the new technologies for the challenging option. We have already started R&D (e.g., Fig. 8) with collaboration in wide areas [16]. In the near future, we hope to

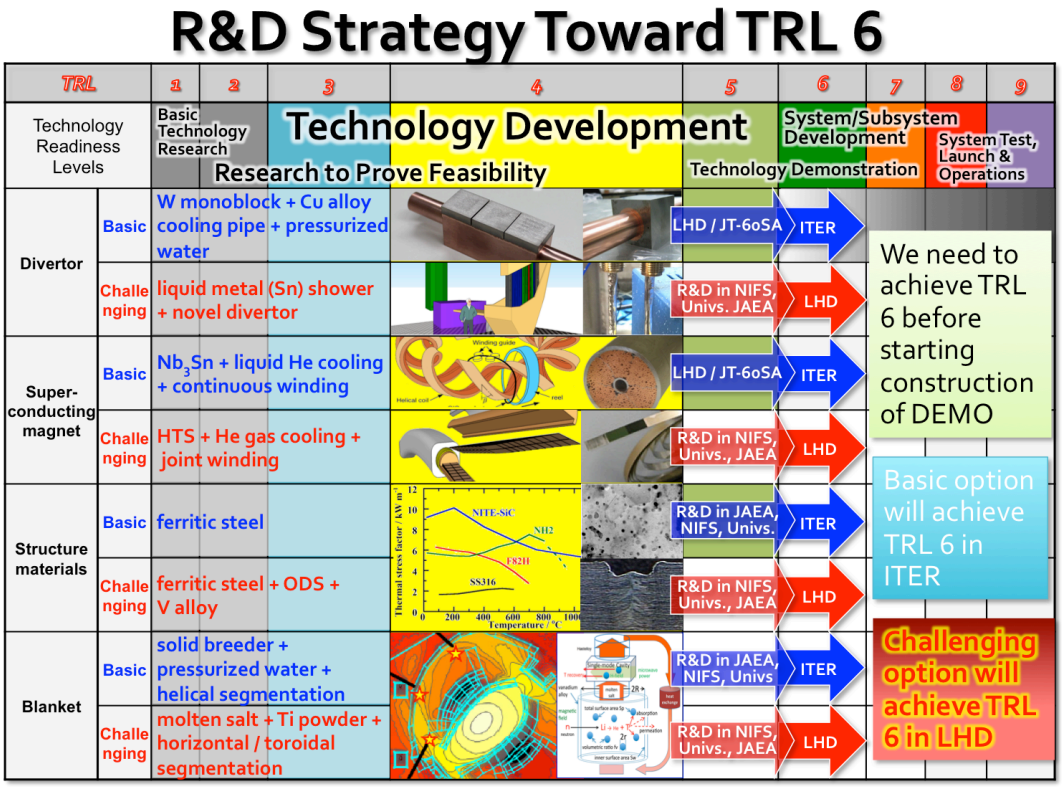

Fig. 7. Summary of the R&D strategy to achieve the Technology Readiness Levels (TRL) for major components in the basic and challenging options of FFHR-d1.

demonstrate these technologies in a reactor-relevant plasma experiment in LHD, to achieve TRL 6 before starting construction of a fusion DEMO reactor.

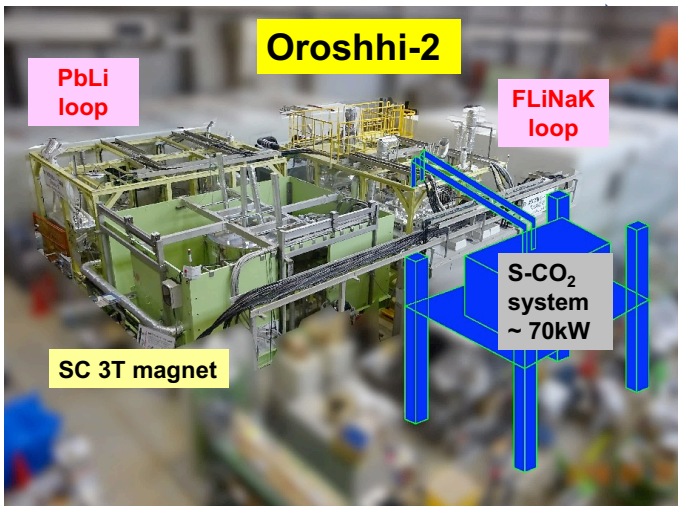

Fig. 8. Photo of the PbLi and FLiNaK twin loop device equipped with a superconducting 3T magnet, Oroshhi-2. A supercritical CO_2 (S-CO_2) turbine system of 70kW is planned to be installed in the future.

5. SUMMARY

Conceptual design activities on the series of helical fusion reactor, FFHR, have been underway since 1994. In the present study, we added two options, basic and challenging, to the latest design of FFHR-d1A. The basic option is based on conventional technologies that are already well established or are being developed worldwide and will finally be established in ITER. The challenging option boldly includes new ideas that would possibly be beneficial for making the reactor design more attractive. In this option, helical coils are composed of helium-gas-cooled HTS magnet coils and fabricated using the joint-winding technique. The divertor is composed of molten tin shower jets inserted into the inboard ergodic layer, and the TBB is self-cooled using molten salt FLiNaBe with Ti powders mixed to increase hydrogen solubility. Both TBB and NSB will be segmented toroidally or horizontally to make construction and maintenance easier. Technologies needed for the basic option will achieve TRL 6 in ITER, while those for the challenging option must be encouraged and finally demonstrated in a reactor-relevant plasma experiment in LHD.

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