

## 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) in NIFS is conducting the conceptual design activities of the LHD-type helical fusion reactor FFHR-d1A. In this paper, two design options of “basic” and “challenging” are newly defined. Conservative technologies including what will be demonstrated in the ITER are chosen in the basic option, in which, for example, two helical coils are made of continuously wound cable-in-conduit superconductors of Nb<sub>3</sub>Sn strands, the divertor is composed of the water-cooled tungsten monoblocks, and the blanket is composed of the water-cooled ceramic breeders. On the other hand, new ideas that would possibly be beneficial for making the reactor design more attractive are boldly included in the challenging option, in which, for example, the helical coils are wound by connecting high-temperature REBCO superconductors by mechanical joints, the divertor is composed of the shower of molten tin jets, and the blanket is composed of the molten salt FLiNaBe including Ti powers to increase the hydrogen solubility. Main targets of the challenging option are early construction and easy maintenance of the large and three-dimensionally complicated helical structure, high thermal efficiency, and, in particular, realistic feasibility of the helical reactor.

### 1. Introduction

The conceptual design activities on the helical fusion reactor have been conducted in the National Institute for Fusion Science (NIFS), Japan, since 1994 [1,3]. The first design was named FFHR-1 (Force-Free Helical Reactor 1), which was equipped with three helical coils aiming at a high magnetic field for good plasma confinement with a low magnetic force on helical coils [1]. At that time, the Large Helical Device (LHD) successfully started its operation in 1998 [2]. The magnetic field strength at the helical coil center,  $B_c$ , is  $\sim 3$  T and the helical coil major radius,  $R_c$ , is 3.9 m, respectively, in LHD. Then, 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. The FFHR-2 can be characterized by the high  $B_c$  of 10 T and relatively small  $R_c$  of 10.0 m. Two options with smaller  $B_c$  and larger  $R_c$  of FFHR-2m1 ( $B_c = 6.2$  T,  $R_c = 14.0$  m) and FFHR-2m2 ( $B_c = 4.4$  T,  $R_c = 17.3$  m) were also studied.

These design activities have been succeeded to the Fusion Engineering Research Project (FERP) organized in 2010 [3]. Since then, the FERP has been working on the latest design named FFHR-d1, where “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.*, in the self-ignition state, 2) the plasma parameters should be reasonably extrapolated from the experimental results obtained in LHD without assuming unknown plasma confinement improvement, 3) the arrangement of magnetic coils should be basically similar to that of LHD, and 4) the technologies assumed in the design should be what are already well established, or what are foreseen to be established in the near future. Because of the 3<sup>rd</sup> rule, the MHD equilibrium in FFHR-d1 is similar to that in LHD. This makes the extrapolation of

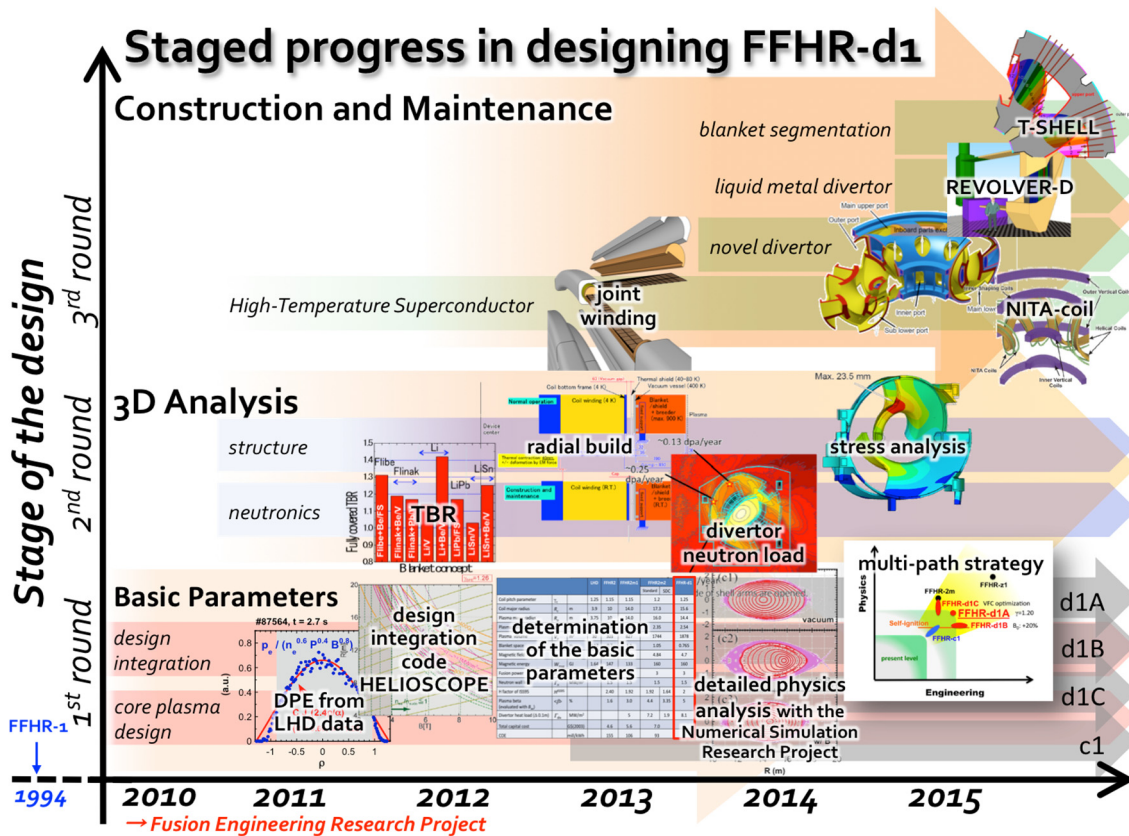


FIG.1. A graphical view of the staged progress in designing the FFHR-d1.

plasma parameters reasonable. The staged progress in designing FFHR-d1 is summarized in FIG.1. In the first round, we have started the design activity from the core plasma design. The plasma parameters are determined by the Direct Profile Extrapolation (DPE) method using the experimental data obtained in LHD [4]. A design integration code named HELIOSOPE has been developed [5]. Using this code, the main parameters were selected, *e.g.*, the device size is four times bigger than that of LHD and the toroidal magnetic field is 4.7 T at the helical coil center. Detailed plasma physics analyses on, for example, the particle and energy transports, the MHD equilibrium and stability, the neoclassical transport, and the alpha particle confinement, *etc.*, have begun in 2014 and it is still continued. The latest study includes the bootstrap current and its effect on the MHD equilibrium, and the power balance between electrons and ions to obtain a self-consistent solution of the density and temperature profiles [6]. In the second 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 ( $B_c = 4.7$  T,  $R_c = 15.6$  m) discussed in this study is the base option, FFHR-d1B ( $B_c = 5.6$  T,  $R_c = 15.6$  m) has a stronger magnetic field to ease the demand for plasma parameters, and for FFHR-d1C, the 3<sup>rd</sup> basic rule, *i.e.*, to use the similar magnetic coil arrangement, is loosened to include flexibility in the magnetic coil design. To be a nuclear test machine that enables a yearlong neutron irradiation test, a compact helical reactor FFHR-c1 ( $B_c = 4.0 - 5.6$  T,  $R_c = 13.0$  m) is also studied. Although the 1<sup>st</sup> basic rule of self-ignition is omitted in FFHR-c1 to reduce the device size, it can be operated in steady state using the self-generated electricity and tritium.

Now the design activity is focused on the construction and maintenance schemes. Although there is no need of current drive and therefore the plasma operation control and steady state sustainment are relatively easy in a helical reactor, we have to solve difficult issues related to construction and maintenance of the three-dimensionally complicated large structures. In some

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

		<b>Basic Option</b>	<b>Challenging Option</b>
<b>SC Magnet</b> $B_c = 4.7$ T $R_c = 15.6$ m	Specification	CICC of Nb <sub>3</sub> Sn (or Nb <sub>3</sub> Al) / 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
<b>Auxiliary Heating</b> 40 – 100 MW (until self-ignition)	ECH (143 GHz fundamental)	w/	w/
	ICRF	w/	w/
	NBI	w/	w/o
<b>Divertor</b>	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
<b>Blanket</b>	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

cases, new and challenging ideas, which are not necessarily well established at this moment, seems to have good possibilities to solve the difficult issues. To include these ideas, we have decided to loosen the restriction of the 4<sup>th</sup> basic rule that allows no unproven technologies to be applied to the reactor design. Then, we newly define two options of “basic” and “challenging” in FFHR-d1A design. Conservative technologies including what will be demonstrated in the ITER are chosen in the basic option. On the other hand, new ideas what 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.

Comparisons between the basic and challenging options are summarized in Table 1. Four items of “Superconducting (SC) Magnet”, “Auxiliary Heating”, “Divertor”, and “Blanket” have different options, respectively. Details of these options are described in Sections 2 and 3. The R&D strategy for realizing the helical fusion reactor is discussed in Section 4. Finally, these are summarized in Section 5.

## 2. The Basic Option

In the basic option, the SC magnet coils adopt cable-in-conduit (CIC) conductors with Nb<sub>3</sub>Sn (or Nb<sub>3</sub>Al) strands cooled by supercritical helium (SHe) at 4.5 K, which is an extension of the ITER technology [7]. The helical coils are wound by react-and-wind method continuously layer

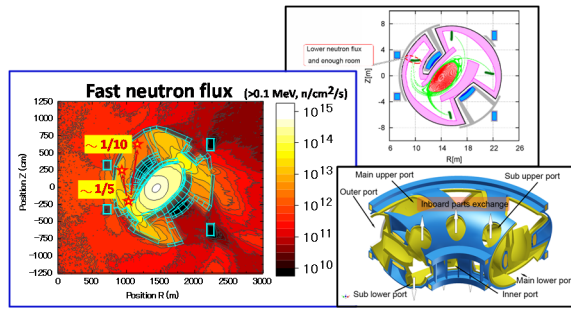
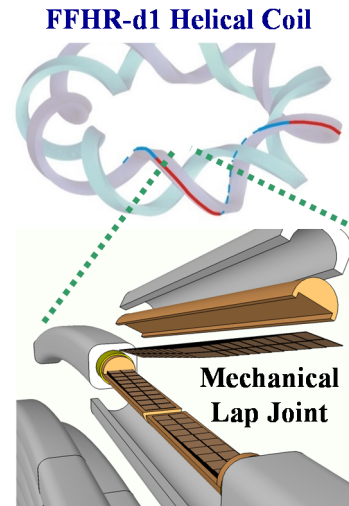


FIG.2. A bird's eye view of the blanket and the coil-support structure for the novel divertor configuration. Inboard parts of the divertor can be exchanged through the vertical maintenance ports at upper inside of the torus.

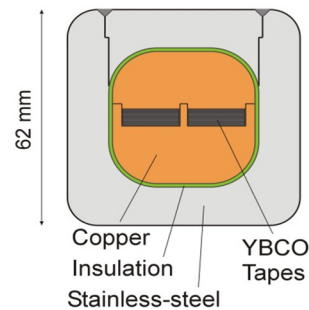
by layer using a large-scale winding machine, which may take  $> 6$  years (considering the fact that it took 1.5 years in LHD). Many other technological difficulties are associated with this option, such as how CIC conductors be precisely bent and twisted to be installed into helical grooves of internal plates and how the Vacuum Pressure Impregnation (VPI) be done by raising the whole coil temperature up to 150 degrees centigrade after winding.

The divertor system is basically similar to those being developed for ITER, *i.e.*, the water-cooled tungsten monoblock divertor with cooling pipes made of Cu alloy. As in LHD, the entire divertor footprint is covered to form the full-helical divertor. The peak divertor heat load on the full-helical divertor is expected to exceed  $20 \text{ MW/m}^2$ , because of the inhomogeneous divertor heat load profile as observed in LHD. Therefore, we need to develop plasma control methods for divertor heat load reduction, *e.g.*, the divertor detachment, and/or the magnetic field optimization to make the divertor heat load uniform. Maintenance of the full-helical divertor is also one of the difficult issues. Aiming at easy maintenance of divertor plates at the inboard side of the torus, where the divertor heat load is expected to be high, the novel divertor concept has been proposed [8] (FIG.2). In this configuration, the divertor plates in the inboard side are placed behind the helical coils. Ten vertical ports are equipped in the inboard side of the torus for frequent maintenance of the inboard side divertor. The novel divertor mitigates the neutron irradiation on the divertor and makes the use of copper cooling pipes possible.

(a)



(b)



(c)

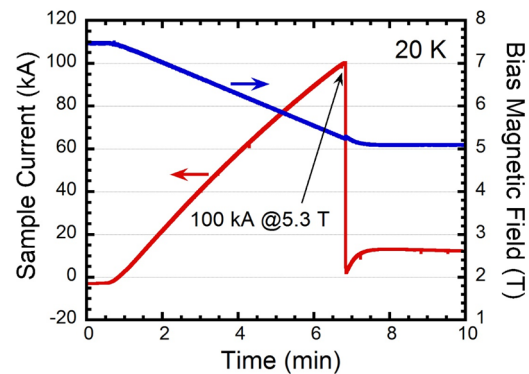


FIG.3. (a) A schematic illustration of the mechanical lap joint used in the HTS helical coil winding. (b) A cross-sectional view of the HTS conductor equipped with an internal insulation. (c) Waveforms in the experiment demonstrating a 100 kA current at a 5.3 T magnetic field and 20 K temperature with a one-turn HTS coil sample.



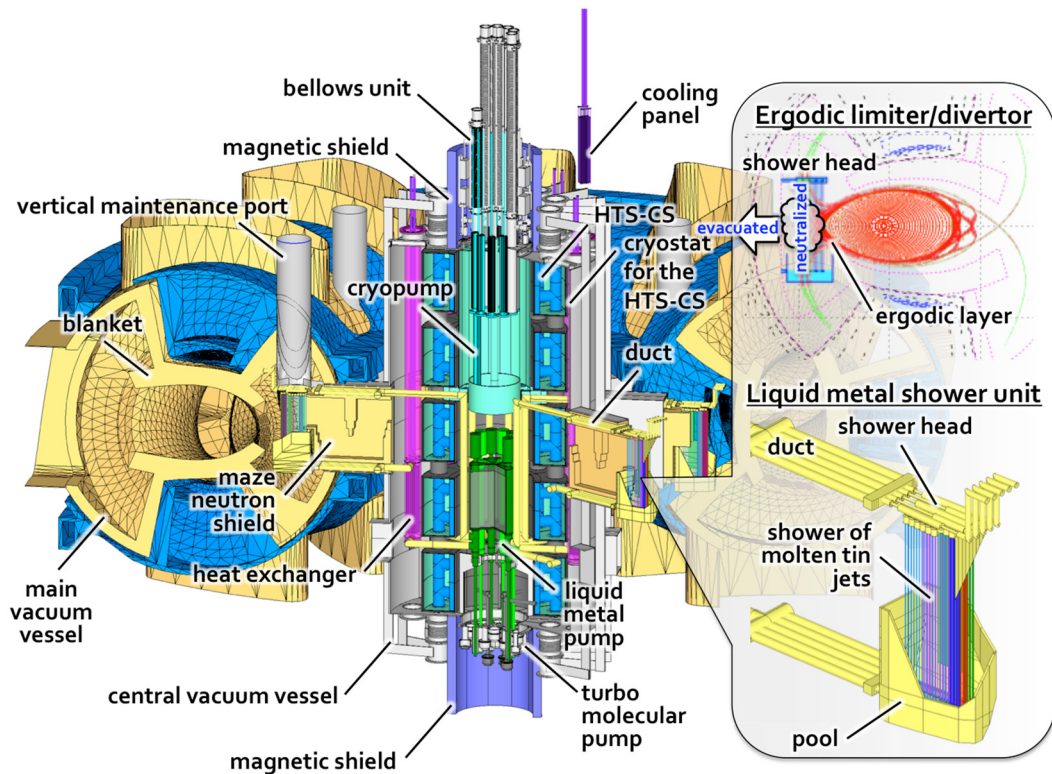


FIG.4. A bird's eye view of the FFHR-d1 equipped with the REVOLVER-D. A cross-sectional view of the ergodic limiter/divertor configuration and a close up view of the liquid metal shower unit are shown in the balloon.

The blanket system is composed of the Neutron Shield Blanket (NSB) and the 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.*, the water-cooled ceramic breeder blanket. Detailed design of the TBB is, however, not obtained yet. Both of the NSB and TBB will be segmented along the helical coils to form the blanket modules. How to construct and maintain the large and complicated blanket modules is also remained as an open issue.

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

### 3. The Challenging Option

In the challenging option, new technologies of the High-Temperature Superconductor (HTS) [9,10], the liquid metal ergodic limiter/divertor [11], and the molten salt (FLiNaBe mixed with metal powders) breeder blanket [3,12] are adopted to solve the problems associated with a large winding machine and difficult maintenance of divertors and blankets.

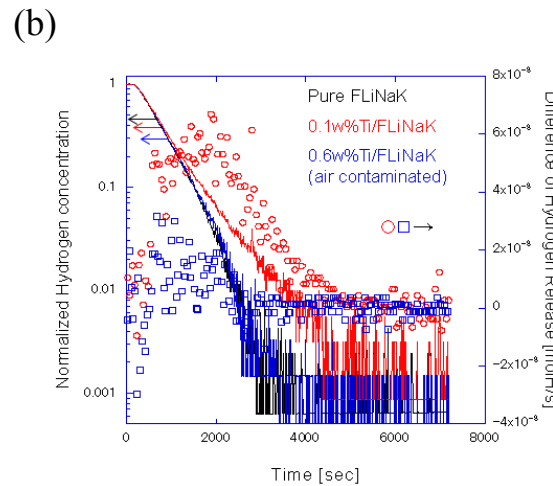
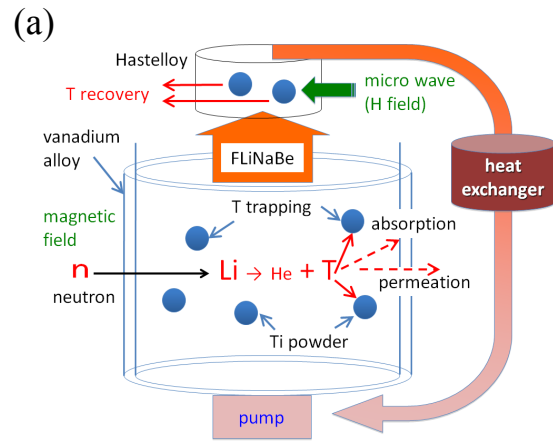
The "joint winding" using the mechanical lap joint technique (FIG.3(a)) is applied to fabricate the helical coils by connecting segmented HTS (REBCO) STARS (Stacked Tape Assembled in Rigid Structure) conductors. The period of helical coil winding by this procedure is expected to take < 3 years [9]. The cooling scheme is simplified by circulating helium gas (GHe) at 20 K. The VPI process can be skipped by having internal electrical insulation in the HTS conductor (FIG.3(b)) and by welding neighbouring conductors in the winding package. The NITA (Newly Installed Twist Adjustment) coils [13], which are supplementary helical coils, are 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 the HTS conductor beyond the already achieved status of 100 kA at 5.3 T, 20 K with a short sample (FIG.3(c)).

A new liquid metal limiter/divertor system named the REVOLVER-D has been proposed [11] (FIG.4). In this system, 10 units forming the molten tin shower jets stabilized by chains inside each jets are installed only in the inboard side of the torus to intersect the ergodic layer. This works as the 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 the 10 maintenance ports similar to those proposed in the novel divertor concept.

The blanket system using the metal powder mixed FLiNaBe (the melting point  $\sim 580$  K) [3] (FIG.5(a)) is also the challenging option. Effective increase of the hydrogen solubility over 5 orders of magnitude was confirmed already [12] with powders of hydrogen storage metal such as Ti (FIG.5(b)). Owing to this observation, though without MHD effects, we consider that vanadium alloys available at  $> 1,000$  K become applicable, giving a higher thermal efficiency of  $> 40\%$  than the case of using FLiBe/F82H, and making the tritium permeation barrier coating less necessary.

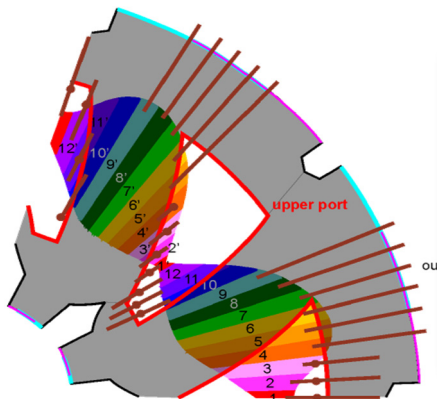
For faster construction with high accuracy, a new type of TBB named the T-SHELL



Hydrogen release behavior from 0.1wt% Ti-FLiNaK  
temp.: 700C, feeding gas  $H_2(20.3 \text{ kPa})/Ar \rightarrow Ar$

FIG.5. (a) A principal diagram of the Ti powder mixed FLiNaBe. (b) The hydrogen release behavior from 0.1 wt% Ti - FLiNaK at 700 degrees centigrade (red circles), showing the amount of hydrogen absorbed in Ti. In the case with the air contamination, the hydrogen absorption decreases even with the higher Ti content of 0.6 wt % (blue squares).

(a)



(b)

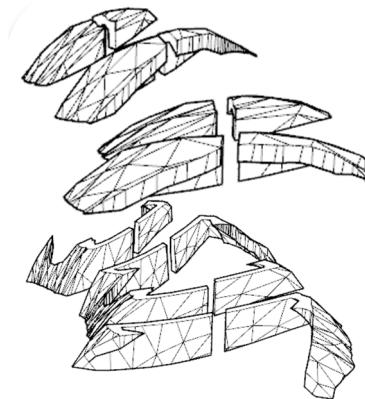


FIG.6. (a) Top view of the T-SHELL blanket, i.e., toroidally sliced tritium-breeding blanket. (b) Bird's eye view of the horizontally sliced neutron shield blanket.

## R&D Strategy Toward TRL 6




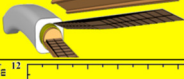
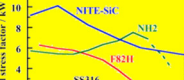
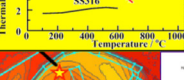
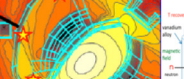

TRL	1	2	3	4	5	6	7	8	9
Technology Readiness Levels	Basic Technology Research		Technology Development			System/Subsystem Development		System Test, Launch & Operations	
	Research to Prove Feasibility			Technology Demonstration					
Divertor	Basic	W monoblock + Cu alloy cooling pipe + pressurized water			LHD / JT-60SA	ITER	We need to achieve TRL 6 before starting construction of DEMO		
	Challenging	liquid metal (Sn) shower + novel divertor			R&D in NIFS, Univs., JAEA	LHD			
Super-conducting magnet	Basic	Nb <sub>3</sub> Sn + liquid He cooling + continuous winding			LHD / JT-60SA	ITER			
	Challenging	HTS + He gas cooling + joint winding			R&D in NIFS, Univs., JAEA	LHD			
Structure materials	Basic	ferritic steel			R&D in JAEA, NIFS, Univs.	ITER	Basic option will achieve TRL 6 in ITER		
	Challenging	ferritic steel + ODS + V alloy			R&D in NIFS, Univs., JAEA	LHD			
Blanket	Basic	solid breeder + pressurized water + helical segmentation			R&D in JAEA, NIFS, Univs.	ITER	Challenging option will achieve TRL 6 in LHD		
	Challenging	molten salt + Ti powder + horizontal / toroidal segmentation			R&D in NIFS, Univs., JAEA	LHD			

FIG.7. Summary of the R&D strategy to achieve the Technology Readiness Levels (TRL) of 6 for major components in the basic and challenging options of the FFHR-d1.

breeder blanket has been proposed [14] (FIG.6). This T-SHELL blanket is divided at every 3 degrees of the toroidal angle. In the case of a helically segmented blanket in the basic option, it is necessary to move the blanket units three-dimensionally inside the torus for replacement. In the case of a toroidally segmented blanket as the T-SHELL, the blanket unit can be replaced by a combination of uniaxial movements and poloidal rotation alone. This increases the feasibility of the blanket maintenance. Also for the NSB, toroidal segmentation might make its construction easier compared with the case of helical segmentation. Detailed scenarios of construction and maintenance, including the segmentation method and the motion analysis of the blanket units, for both TBB and NSB, are still under discussion.

#### 4. The R&D Strategy

Although the new technologies adopted in the challenging option might significantly ease the construction difficulties in the basic option, these are not necessarily well established at this moment. The R&D strategy in terms of the Technology Readiness Levels (TRL) is summarized in FIG.7. The technologies needed for the basic option are already being developed in heliotrons, stellarators, tokamaks, and linear machines, in the world. These will finally achieve TRL 6 in ITER. On the other hand, it is necessary to encourage, or start the R&D activities to increase the TRL of the new technologies for the challenging

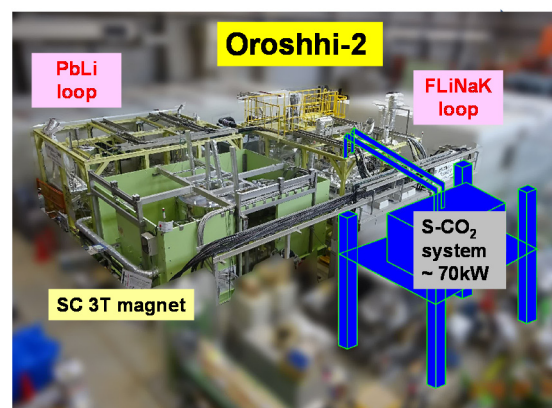


FIG.8. A photo of the PbLi and FLiNaK twin loop device equipped with the superconducting 3 T magnet, Orosshi-2. A supercritical CO<sub>2</sub> (S-CO<sub>2</sub>) turbine system of ~70 kW is planned to be installed in the future.



option. We have already started the R&D as in FIG8 with collaborations in wide areas [15]. In the near future, we would like to demonstrate these technologies in a reactor-relevant plasma experiment in LHD, to achieve TRL 6 before starting construction of a fusion DEMO reactor.

## 5. Summary

The conceptual design activities on the series of helical fusion reactor, FFHR, are underway since 1994. In this study, we added two options of basic and challenging to the latest design of FFHR-d1A. The basic option is based on the conventional technologies that are already well established or being developed in the world and finally will be established in ITER. The challenging option boldly includes the 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 by the joint-winding technique, the divertor is composed of the molten tin shower jets inserted to the inboard ergodic layer, and the TBB is self-cooled by the molten salt FLiNaBe with Ti powders mixed to increase the hydrogen solubility. Both of the TBB and NSB will be segmented toroidally to make construction and maintenance easier. The technologies needed for the basic option will achieve TRL 6 in ITER, while those for the challenging option need to be encouraged and finally be demonstrated in a reactor-relevant plasma experiment in LHD.

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