Recent Progress in Fusion Technologies under the BA DEMO-R&D in Phase1 in Japan

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Abstract: The R&D on DEMO blanket related materials and tritium technology has been started by both the EU and Japan. In accordance with the common interest of the EU and Japan, the followings subjects have been applied: reduced activation ferritic martensitic (RAFM) steels; silicon carbide composites (SiC_f/SiC): advanced tritium breeders and neutron multiplier; and tritium. In the period from 2007 to 2009 as Phase 1, we have carried out the design and fabrication of the equipment at Rokkasho. Activities for the licensing of the equipment for tritium and beryllium handling have also been carried out. The tritium usage amounts of the equipment per day, per 3 months and per year are 3.7, 185 and 1.35 PBq, respectively. The concentration of beryllium is limited to be less than 0.002 mg/m^3 in a controlled area. For the R&D of SiC_f/SiC, the properties of a NITE-SiC_f/SiC composite were first investigated. Various properties of the plates of the RAFM steel F82H melted and forged were investigated as the initial stage of the R&D on the optimization of fabrication technology of F82H. To study the fabrication method of beryllium intermetallic compounds (beryllides), a series of preliminary tests has been carried out by using Ti-Al and Be-Ti compounds. To study the pebble fabrication methods of the breeding materials, a series of preliminary tests were also carried out with Li_2TiO_3 in this stage.

1. Introduction

The BA Activities were started as collaborative studies between Japan and EU in the field of fusion energy research for a DEMO plant. The BA Activities comprise the following three projects: 1) the Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA); 2) the International Fusion Energy Research Center (IFERC); and 3) the Satellite Tokamak Program[1]. The IFERC functions as the DEMO design and R&D coordination center, with a mission to coordinate scientific and technological activities to DEMO. A series of basic R&D for DEMO has been implemented on key and generic issues of mutual interest, more specifically, on advanced structural materials (reduced-activation ferritic-martensitic steels, RAFM, and SiC composites) ; advanced blanket materials (breeders and multipliers) ; and tritium technologies [1].

These R&D studies are mainly carried out in Rokkasho site in Japan. Hence, we defined the period of 2007 to 2009 as Phase 1, and have carried out the design and fabrication of the equipment at Rokkasho. Activities for the licensing of the equipment for tritium and beryllium handling have also been carried out. In addition, a series of preliminary studies for the above R&D subjects has been started with Japanese universities. This paper describes the recent progress in those R&D activities in Japan, by JAEA and Japanese Universities, under the framework of the BA DEMO R&D program.

2. RI handling equipment at Rokkasho site[1]

The RI handling equipment constructed at Rokkasho site is the first and quite unique facility in Japan, where tritium, beta and gamma RI species, and beryllium(Be) can simultaneously be used as seen in Fig. 1. It has also the first challenge in Japan that researchers in various fields of blanket (for instance, materials and tritium) join R&D program at a facility. A large synergy and combined effect can be expected through the R&D activities at the facility. The tritium usage amounts of the equipment per day, per 3 months and per year are 3.7, 337, and 1350 TBq, respectively. The tritium usage amounts of the

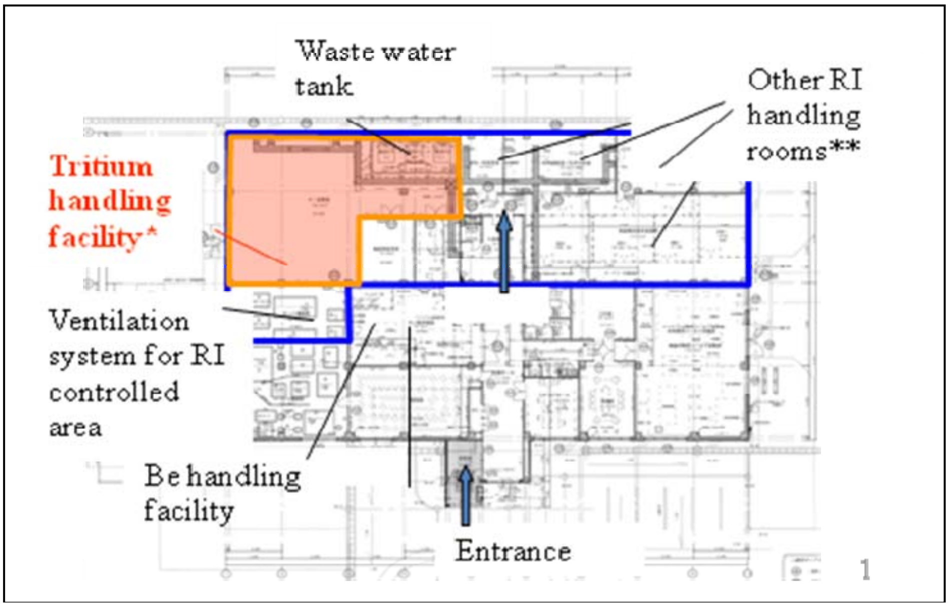

Fig. 1 Conceptual layout of RI handling equipment at Rokkasho site. ***:** In this area, tritium and some RI species (ceramics, steel materials, and nonferrous metals) can be handled. **: In this area, a hot cell and a temperature controlled section are installed. The RI species of ceramics, steel materials, and nonferrous metals are handled.

glove box and hood are 0.37 and 3.7 TBq, respectively. The storage amount is 7.4 TBq for tritium. The other RI species used in the equipment are categorized into four groups: 1) ceramics; 2) steel materials; and 3) nonferrous metals. A typical element used is P^{32} for the ceramics, Cr^{51} for the iron materials, and W^{188} for the nonferrous metals. The amounts of usage are 100, 950, and 44 MBq per day, respectively. The storage amounts for these elements are five times larger values as that of usage.

2.1 Tritium handling facility

Fig. 2 Installation of tritium waste water processing system.

The tritium handling facility is composed of the following systems:1)hoods and a glove box; 2)HVAC(Heating Ventilation Air Conditioning) systems; 3) detritiation systems; and a common service and a waste management systems(See Fig. 2). The radiological control and cold areas are ventilated by two HVAC systems with HEPA filters. Two tritium storage beds are installed into the glove box (nitrogen atmosphere, -40 mm H_20 pressure). The detritiation system is composed of a catalyst bed converting hydrogen to water vapor and a set of molecular sieve beds adsorbing the water vapor. The detritiation factor of the system is design to be 99.9%, and its guarantee value is 99%. A detritiation system (6 m^3/h) is operated in a circulation mode to remove tritium in the atmosphere of the glove box. The other detritiation system (flow rate is 2 m^3/h) is operated in the circulation mode to remove tritium in a vessel. The exhaust gases from the apparatus handling tritium are sent to the vessel. A small amount of gas is also sent to the vessel from the glove box to maintain negative pressure.

2.2 Other RI and blanket materials facility

A hot cell having iron shield with a set of manipulators is installed in a room. Some irradiated samples described above are handled in the cell. An experimental section, where its temperature is strictly controlled is also provided in a room. Some basic material analysis devices are installed in the room. In the building a set of Be handling devices (melting, sintering, and treating devises) are installed in a room (Be handling facility). An independent ventilation system (Frequency of ventilation = 16 times/h, Permissible Be concentration = 0.002 mg/m^3) is installed in the room.

2.3 Evaluation of radiation dose

The calculation procedures of the radiation dose are described as follows. The amount of tritium moved from the hood to the room is 10% of that moved to the hood; and the leak rate of the tritium from an apparatus in the glove box in the hood is 1×10^{-7} Pam^3/s. The amount of tritium transferred from GB, hood, and detritiation systems to room can be evaluated as that of tritium permeating through gloves of the GB (3.25×10^6Bq/h). As the result, the concentration of tritium in the tritium handling room is ~$5x10^{-4}$ Bq/cm^3(~1/1000 of regulation value). In addition, the concentration of tritium at a stack of the building is ~$4x10^{-4}$ Bq/cm^3(~1/10 of regulation value). It was found that the internal exposure by other RI species could be neglected.

In contrast to the internal exposure, the external exposure is mainly due to other RI species. The external dose for radiological workers was evaluated at the distance of 50 cm from the RI species handled. All the RI species used and stored are considered as that they exist simultaneously. The radiation waste stored during the BA program is also considered as that they exist simultaneously. For the hot cell equipment and storage boxes, a radiation shield of iron (7.8 g/cm^3) is assumed to be provided. For the hoods, a radiation shield of lead blocks (10.9 g/cm^3)are assumed to be provided. The maximum value for the external radiation dose of workers was 0.783 mSv/week((working time = 40 hours/week, regulation value = 1.0 mSv/week). The maximum value for the radiation dose at a boundary of the radiation controlled cold area was 1.2 mSv/3 months(working time = 500 hours/3 months, regulation value = 1.3 mSv/3 months). The radiation dose at a site boundary was 43 μSv/3 month((time used = 2184 hours/3 months, regulation value = 250 μSv/3 months).

3. Recent Progress in Phase1 in Japan

3.1 SiCf/SiC Composites

For the R&D of SiC_f/SiC, the properties of a NITE-SiC_f/SiC composite were first studied[2]. The failure behavior of SiC/SiC composites by various failure modes such as tensile, compressive, and shear modes was assessed. The irradiation effects on the two types of SiC, CVD-SiC and NITE-SiC, were preliminarily evaluated. Test results identified an apparent difference of swelling between both materials, and an importance of the presence of sintering additives. The in-situ gamma-ray irradiation experiments were carried out to assess the test equipment including irradiation facilities and materials. The radiation-induced conductivity (RIC) and the increment of the base conductivity without radiation, the radiation-induced electrical degradation (RIED), were successfully detected during irradiation at room-temperature in air. Figure 3 shows the electrical conductivity under gamma-ray irradiation, which indicated that the radiation induced electrical degradation (RIED) was occured, but saturated after the dose of 0.5 MGy. In advance to evaluate the hydrogen transport characteristics as the protonic conductivity of SiC/SiC composites under irradiation in-situ, the hydrogen transport behavior of various types of

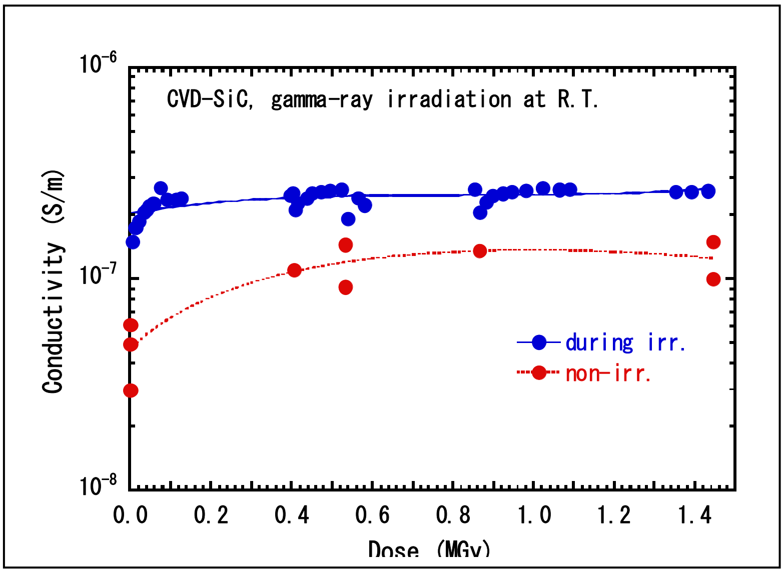

Fig. 3 In-situ irradiation experiments by various irradiation sources showed RIC and indicative RIED.

non-irradiated SiC was evaluated. The preliminary results indicate that the constituent materials of the SiC matrix strongly contributes to the hydrogen transport of SiC/SiC composites.

3.2 Tritium Technology

The equipment of the tritium handling facility has been made and installed as described above. The safety analysis of the facility has also been carried out as described in the precede section. For the R&D tasks of tritium technology have been carried out by collaborative studies with Japanese Universities. The detailed R&D subjects to be carried out are as follows. For the tritium analysis technology, an imaging plate method has been studied to measure the tritium depth profile in the solid waste. For the basic tritium safety data, the tritium behaviors in the metals used in a fusion reactor (tungsten and stainless steel) and blanket materials have been studied. For the tritium durability test, some basic studies have been started for organic compounds at TPL (Tritium Process Laboratory in JAEA) by gamma and beta ray irradiation tests[3]. Figure 4 shows the effect of the beta ray on the ionic conductivity of the Nafion membrane. The membrane is used in a water detritiation system of ITER. In ITER, it is required that the membrane can be used for two years under the condition that tritiated water concentration is 9.25 x10^{12}Bq/kg (equivalent to 530 kGy irradiation). No serious effect was observed even for the case of 1000 kGy irradiation by beta ray.

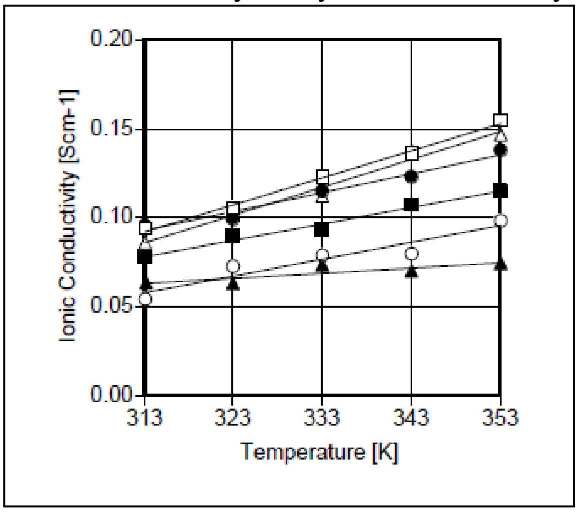

Fig. 4 shows the effect of the beta ray on the ionic conductivity of the Nafion membrane.(□):0kGy, (●): 150kGy, (□):250kGy, (■): 500kGy, (○):1000kGy, (▲):1500kGy

3.3 RAFM steels

A set of various properties of the plates of F82H (F82H-BA07) was started to assess the large-scale fabrication technologies[4]. The detailed evaluation of F82H-BA07 heat was carried out on mechanical properties such as tensile, fatigue, and creep properties. The plates of the F82H-BA07 heat has equivalent performance to the F82H-IEA heat on its high temperature strength. Also it revealed that fatigue life of F82H-BA07 heat show generally longer fatigue life compared to that of F82H-IEA heat. On the joining technology, various welding methods,such as TIG, EB, YAG laser and fiber laser welding, have been accessed on hot/cold cracking sensitivity, residual stress, etc, for the F82H-BA07. There is little or no susceptibility to solidification cracking of F82H with small impact of Ta content, compared with that of stainless steels. On the issue of He and H effect, the environment for DEMO blanket module was studied to simulate with ion accelerators, especially in terms of He/dpa ratio. The establishment of database on the irradiation behavior has been started.

3.4 Advanced Neutron Multiplier

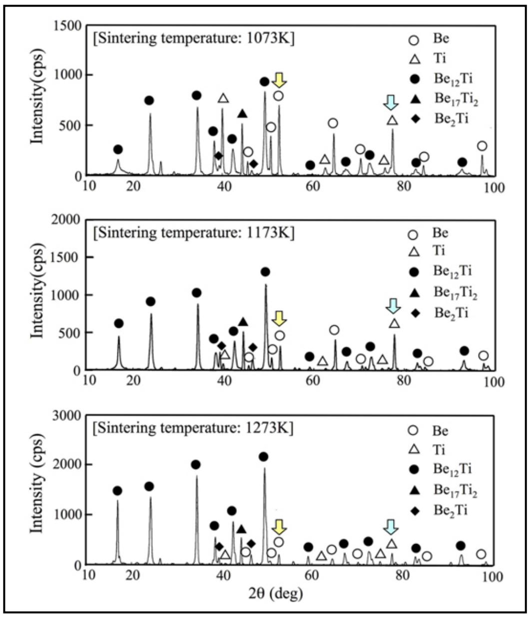

Fig. 5 XRD profiles of beryllide synthesis as a Be-Ti intermetallic by the plasma sintering

From a series of design works, the specifications of the devices and ventilation system to be installed in the Be handling facility have been made to meet the Japanese regulations. The preparation work for the licensing of the Be handling facility was also carried out. A survey of fabrication procedure of the beryllide (beryllium intermetallic compounds) was progressed. The powder, pebbles, rods, and blocks of beryllide are made by the sintering, melting, powdering and granulation procedures in the facility. To study the fabrication method of beryllides, a series of preliminary tests has been carried out by using mixed beryllium and titanium powder by the plasma sintering method for the first time(See Fig. 5). The Sintering temperatures were 1073, 1173 and 1273 K. The compound composition was measured by the X-Ray diffraction (XRD). The formation of $Be_{12}Ti$, $Be_{17}Ti_2$ and Be_2Ti intermetallics was observed for each temperature. The peak intensities of elemental Be and Ti decreased with increase in sintering temperature, which shows that sintering of beryllide is enhanced by the increase in temperature. It was thus presumed that beryllides could directly be synthesized by the plasma sintering method from mixed powder particles of Be and Ti. It was also confirmed that the compounds could be granulated by the infrared melting method[5] using Ti-Al compounds.

3.5 Advanced Tritium Breeders

The R&D on the fabrication technologies of the advanced tritium breeders and the characterization of developed materials has been started: 1) production of advanced breeder pebbles; 2) the characterization of the pebbles on physical, chemical, mechanical and other properties; and 3) studies on reprocessing and re-use of tritium breeder[6]. The Li_2TiO_3 pebble fabrication using the rotating granulation method has been studied. Starting powder was a stoichiometric compound Li_2TiO_3, and the particle size of the powder is <5μm, which was confirmed by the XRD and a scanning electron microscope. This raw material powder was set into a rotating drum, and was rotated with binder, and sintered at 1473K. It could be concluded that the pebbles with a sphericity of 1.04 were able to be granulated by the rotating

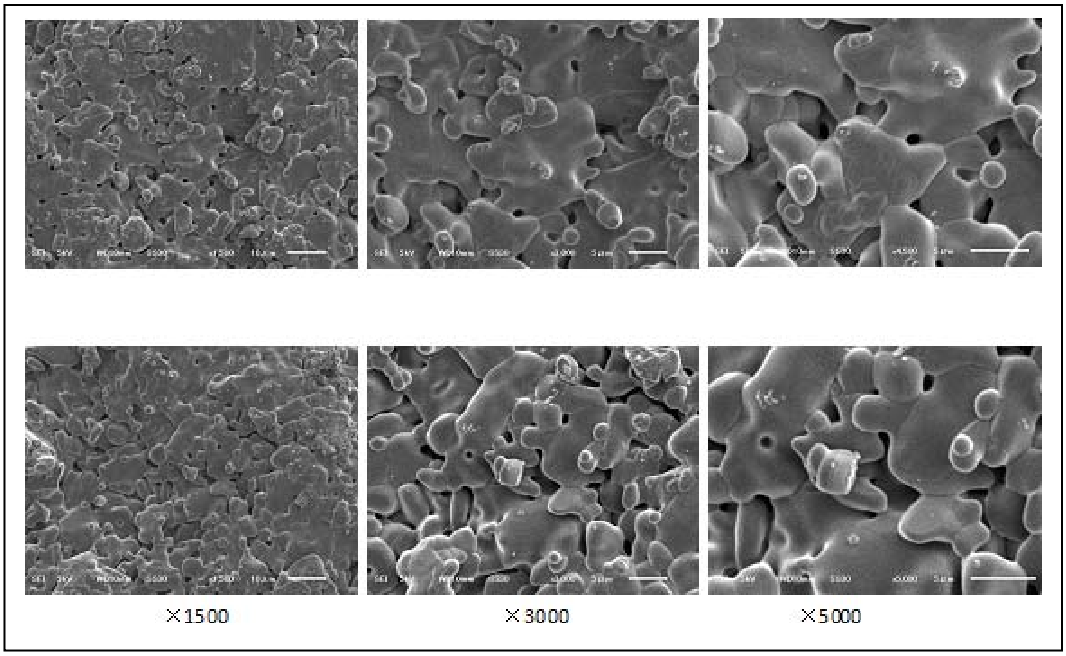

Fig.6 Grain size of sintered pebble.

granulation method. The average pebble diameter and the average grain size after the sintering were 1.14mm and >5μm (see Fig.6), respectively. Concerning the reprocessing and re-use tests, a mixture of aqueous solutions of peroxide hydrogen (H_2O_2) and nitric acid (HNO_3) was selected as a solvent of Li_2TiO_3; and aqueous solutions of HNO_3 was selected as a solvent of Li_4SiO_4 and Li_2O. The results indicate that the dissolution rates of Li from each lithium ceramic powder were larger than 90%.

4. Summary

In accordance with the common interest of the EU and Japan, the followings subjects have been applied: reduced activation ferritic martensitic (RAFM) steels; silicon carbide composites (SiC_f/SiC): advanced tritium breeders and neutron multiplier; and tritium. To carry out these R&D activities, the RI handling equipment has been constructed at Rokkasho in Japan. The tritium usage amounts of the equipment per day, per 3 months and per year are 3.7, 185 and 1.35 PBq, respectively. The tritium handling equipment, glove box, hood; and detritiation systems, have been designed, made, and installed.

For the R&D of SiC_f/SiC, the irradiation effects, as well as the stability at high-temperature and the permeation of hydrogen of SiC_f/SiC, were studied. Various properties of the plates of the RAFM steel F82H melted and forged were studied on the optimization of fabrication technology of F82H. Assessment on the specific welding technologies was also carried. It was presumed that beryllides (beryllium intermetallic compounds) could directly be synthesized by the plasma sintering method from mixed powder particles of Be and Ti. It was also confirmed that the compounds could be granulated by the infrared melting method. To study the pebble fabrication methods of the breeding materials, a series of preliminary tests were also carried out with Li_2TiO_3. It was confirmed that the pebbles of Li_2TiO_3 having the sphericity of 1.04 could be prepared by the rotating granulation method

Refferences

[1] T. Yamanishi et al, “Recent Activities on Tritium Technologies of BA DEMO-R&D Program in JAEA”, the Ninth International Symposium on Fusion Nuclear Technology, October 11-16, 2009, Dallian, China.

[2] T. Nozawa et al., “Japanese Activities of R&D on SiC/SiC Composites in Phase 1 under the Broader Approach Activities toward DEMO”, the 14th Intern. Conf. on Fusion Reactor Materials (ICFRM-14), September 7-12, 2009, Sapporo, Japan.

[3] Y. Iwai, A. Hiroki, M. Tamada, K. Isobe, T. Yamanishi, “Radiation deterioration of ion-exchange Nafion N117CS membranes”, Radiation Physics and Chemistry, 79, 46-51 (2010).

[4] H. Tanigawa et al., “Status and Key Issues of Reduced Activation Ferritic/Martensitic Steels as the Structrural Materials of DEMO Blanket”,the 14th Intern. Conf. on Fusion Reactor Materials (ICFRM-14), September 7-12, 2009, Sapporo, Japan.

[5] M. Nakamichi, et al., “Status of advanced neutron multiplier development for DEMO in the broader approach activities in Japan”, Proceeding of the 9th IEA international workshop on beryllium technology (BeWS-9), 16-20 (2009).

[6] T. Hoshino, K. Tsuchiya, H. Hayashi, K. Nakamura, H. Terunuma and K. Tatenuma, "Preliminary tests for reprocessing technology development of tritium breeders", J. Nucl. Mater., 386-388, 1107-1110 (2009).