Fusion Technology Development for DEMO in the Broader Approach Activities

T. Nishitani 1), H. Tanigawa 1), S. Jitsukawa 1), T. Nozawa 1), K. Hayashi 1), T. Yamanishi 1), K. Tsuchiya 1), T. Hoshino 1)A. Möslang 2), P. Spaetig 3), S. Tosti 4), E.R. Hodgson 5), R. Laesser 6), D. Maisonnier 6), A. Kohyama 7), K. Morishita 7), T. Shikama 8), M. Araki 1)

1) Japan Atomic Energy Agency, 801-1 Mukoyama, Naka, 311-0193, Japan
2) FZK Karlsruhe, P.O Box 3640, 76021 Karlsruhe, Germany
3) CRPP-EPFL, ODGA C110 5232 Villigen PSI, Switzerland
4) ENEA CR Frascati ,Via E.Fermi,45 i-00044 Frascati, Italy
5) CIEMAT Fusion Association, 28040 Madrid, Spain
6) EFDA CSU Garching, Boltzmannstrasse 2, 85748 Garching, Germany
7) Kyoto University, Uji, Kyoto 611-0011, Japan
8) Tohoku University, Aobaku, Sendai 980-8577, Japan

Abstract. As a part of the Broader Approach activities from 2007 to 2016, research and development on blanket related materials and tritium technology have been initiated toward DEMO by the EU and Japan. According to the common interest of EU and Japan toward DEMO, R&D on reduced activation ferritic martensitic (RAFM) steels as a DEMO blanket structural material, silicon carbide composites (SiC_f/SiC), advanced tritium breeders and neutron multiplier for DEMO blankets, and tritium technology will be implemented as a part of the DEMO Design and R&D coordination Center activity of BA. As a preparatory work on the RAFM steels, a 5-ton heat of RAFM steel (F82H) was procured recently, which control of indicated that unexpected impurities was important in the large scale of RAFM steels. Double notch tensile tests to evaluate failure behavior were performed for NITE-SiC_f/SiC composites with different size and different notch depth. It was found that neither significant notch sensitivity nor specimen size effect was observed in proportional limit tensile stress and fracture strength, implying the importance of the stress (or strain) criterion for the failure evaluation of composites. On the neutron multiplier, reactivity of Be-Ti and Be-V alloys with F82H was investigated. The reactivity of the Be-Ti and Be-V alloys was much smaller than that of pure beryllium. Large rods (about 30 mm in diameter) of Be-Ti were fabricated successfully. As a preliminary activity, Li_4SiO_4 pebbles were fabricated by the melt spraying method, and then characterized.

1. Introduction

Several technical R&Ds are newly launched as a part of the Broader Approach (BA) activities [1], which is initiated by the EU and Japan aiming at DEMO. The breeding blanket is one of the most important components in DEMO. A pebble bed type blanket[2,3] is one of the most promising blanket concept both in the EU and Japan, which consists of the first wall and structure material, the tritium breeding pebbles, the neutron multiplying pebbles, and the cooling channels. The DEMO blanket has to withstand high neutron flux and fluence under continuous operation. Therefore the development of blanket structural materials and breeding/multiplying materials which can be used under DEMO operational conditions is essential. According to the common interests of each party towards DEMO, R&D on reduced activation ferritic martensitic (RAFM) steels [4,5] as structural material, SiC_f/SiC composites, advanced tritium breeders and neutron multiplier for DEMO blankets, and Tritium Technology will be implemented through the BA DEMO R&D program.

2. Reduced Activation Ferritic Martensitic Ateels as a Blanket Structural Material

The development of blanket structural materials is crucial on the path to fusion power. Experience is now well established for the development and characterization of reduced activation ferritic martensitic (RAFM) steels, such as F82H [6] in Japan and EUROFER [7] in the EU, which are now being considered as main candidate materials for structural applications in DEMO. For the manufacture and operation of DEMO, the timely availability of sufficient engineering bases, modeling/simulation/prediction methods of materials behavior and design methodology, have become indispensable elements in fusion roadmaps. In particular: The fabrication technology and the materials database must provide highly attractive properties, especially with respect to high thermal efficiency, availability, reliability, irradiation resistance, and reduced activation capability. Appropriate techniques must be defined to incorporate the fracture/rupture properties of the irradiated materials into an engineering procedure that allows one to ensure the integrity of the components for the safe operation of a fusion power reactor. Also methods need to be developed to predict the deformation and fracture behavior of structures under irradiation from materials databases by modeling/simulation of materials behavior.

Optimization of manufacturing technology of RAFM steels, such as melting, forging, rolling, milling, to obtain wrought products will be performed. Japan and the EU have experience of small scale heats of RAFM steels. The next step in the development of RAFM steels also focuses on a further reduction of the irradiation-induced activation of large-scale heats (several tons). The long-term activation level is mainly determined by some minor alloying elements and by impurities. It is important to achieve this with industrial scale production technology and not only for lab scale production. As a preliminary activity of this R&D, the vacuum induction melting (VIM) of F82H in 5 tons followed by secondary refinement and forging was performed recently. It was found that Electro Slag Remelting (ESR) was required as for secondary melting process in order to control unexpected impurities. FIGS. 1 and 2 show the pictures of the F82H ingot before ESR and three slabs made as the wrought product.

FIG.1. Picture of the F82H ingot before ESR..

FIG.2. Picture of the F82H slab.

Optimization of joining technology of RAFM steels and dissimilar materials is also an important technology to be established for the DEMO blanket fabrication. DEMO blanket fabrication requires various welding techniques such as tungsten-inert-gas (TIG) welding, electron beam (EB) welding, YAG laser welding, and diffusion welding, and the weldments must be of the highest quality. In Japan, mock-ups of first wall and side walls for the water-cooled pebble bed type TBM were fabricated by Hot Isostatic Pressing (HIP) using F82H to form a built-in cooling channel structure. In order to develop the DEMO specific welding technologies, mechanical restraint factors for the various welding methods, and metallurgy based investigation on hot cracking sensitivity will be performed to obtain basic information on RAFM welding.

Experimental investigation of mechanical properties will be performed using small specimen test techniques, in order to understand the micro-mechanisms of irradiation effects on the mechanical properties of RAFM steels. Not only experimental studies but also simulation will be carried out to characterize the irradiation effects and to quantify the specimen size and geometry effects on the mechanical properties. Fracture toughness, mainly focusing on the applicability of master curve methodology, and tensile tests will be carried out as well as post-test microstructural analyses and finite element modeling of unirradiated and irradiated RAFM steels.

Empirical and theoretical models of the deformation behavior of irradiated RAFM steel structures will be proposed on the basis of detailed analysis of post-irradiation evaluation (PIE) data and on a minimum amount of irradiation experiments. The models are for short-term plasticity/ductile fracture properties (mostly derived from tensile tests), fatigue under irradiation and irradiation creep properties. Irradiated simplified component elements, such as thin plates, plates with holes, and tubes will also be used to evaluate the validity of the models and to prepare basic methodologies for structural design criteria under irradiation in fusion reactors.

3. SiC_f/SiC Composites

Silicon carbide composites have, and are being developed as a candidate material for the high-temperature operating DEMO reactor aiming at high energy conversion efficiency [8,9]. A conservative but new concept supposes the utilization of SiC_f/SiC composites as flow channel inserts for thermal and electrical insulation in the dual-coolant lead-lithium (Pb-17Li) liquid breeder blanket (DCLL) system [10]. Critical issues of the SiC_f/SiC composite development are 1) development of component fabrication technology including mass production, joining and coating technologies, 2) environmental effects for lifetime evaluation, e.g., oxidation mechanism and compatibility with Pb-17Li, 3) irradiation effects including dynamic issue, i.e., irradiation creep, and synergetic effects of helium and hydrogen, and 4) development of design codes and test standards for the development of a material property database. In the BA activity on SiC_f/SiC composites, R&D on mechanical properties of SiC_f/SiC composites and R&D on physical/chemical properties of SiC_f/SiC composites and other functional ceramics are proposed, being of particular importance to develop practical engineering database for the DEMO design.

Of many composite types, a nano-infiltration transient-eutectic-phase sintered (NITE) SiC_f/SiC composites [11] is specifically considered as a main target material in the R&D on SiC_f/SiC composites for the DEMO R&D. The latest "lab-grade" NITE-SiC/SiC composites can be satisfactorily utilized as reference materials in this R&D due to the exceptionally high-competitiveness. As the preliminary R&D, two types of "pilot-grade" NITE- SiC_f/SiC composites: unidirectional and cross-ply plates were preliminarily produced and their fundamental material properties were evaluated in 2007-2008. FIGURES 3 and 4 show the outlook and the micro-structure of the unidirectional NITE-SiC_f/SiC composite prepared as a reference specimen. Most fundamental requirements of these composites were found to be in acceptable level, though the results indicate that the present "pilot-grade" NITE- SiC_f/SiC composites need some minor modifications.

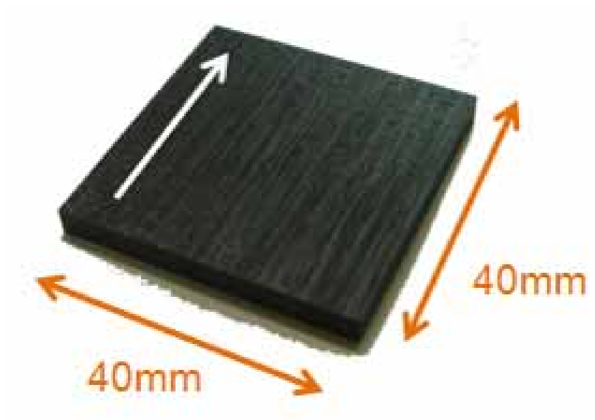

FIG. 3. Picture of the unidirectional NITE-SiC$_f$/SiC composite prepared as a reference specimen.

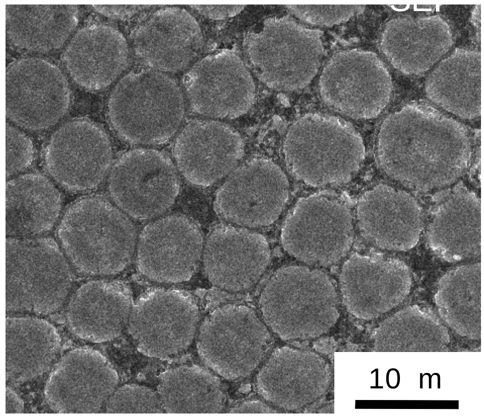

FIG. 4. Micro-structure of the unidirectional NITE-SiC$_f$/SiC composite.

Standardization of testing methods is essential to establish a reliable and practical database for the engineering design of DEMO. However, the test methodologies to evaluate mechanical properties of advanced SiC$_f$/SiC composites are not sufficiently standardized for the practical use of design activities. Establishing the test methodology on high-temperature/lifetime properties such as fatigue and creep must be one of the critical R&D items. Evaluation on the off-axial mechanical properties is another important issue due to the intrinsic anisotropy of the composites. FIGURE. 5 shows the preliminary result of the double notch tensile test method, indicating notch insensitivity and very minor size effect on proportional limit tensile stress (PLS) and fracture strength (UTS) [12]. In parallel, comprehensive modelling of the macroscopic mechanical behavior of advanced SiC$_f$/SiC composites will be developed, also irradiation effects coupled with characterization of microstructural evolution and swelling behavior under irradiation are evaluated for structural modelling.

Transportational properties of SiC$_f$/SiC composite are readily affected by irradiation depending on temperature and dose rate. The electrical conductivity and permeation of He and H isotopes during irradiation, and the possible detrimental effects on the microstructure of the composites due to displacement damage, He and H implantation will be studied as a part of the R&D items. It is worth noting that this study can readily be extended to functional ceramics other than SiC. It is also recognized that the compatibility between SiC materials and a Pb-17Li liquid metal is of particularly high interest and thus the erosion-corrosion behavior of SiC and SiC$_f$/SiC composites in Pb-17Li at high temperatures will be studied. FIGURE 6 shows the experimental apparatus proposed for the test, where temperature up to 1100ºC, relative velocity up to 1 m/s and duration of 3000 h are considered.

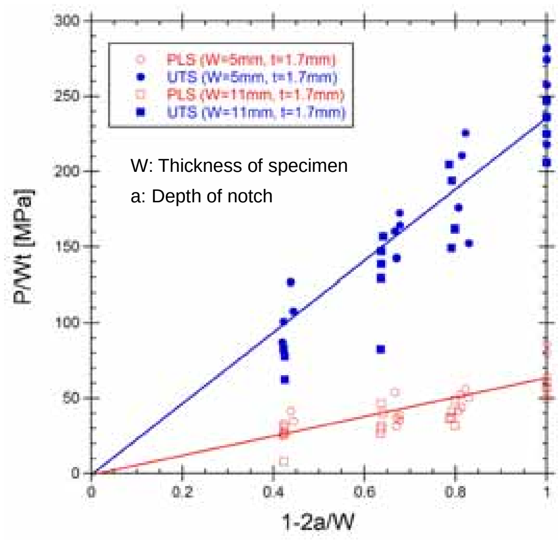

FIG.5. Example of failure evaluation of advanced SiC_f/SiC composites by the double notch tensile test method.

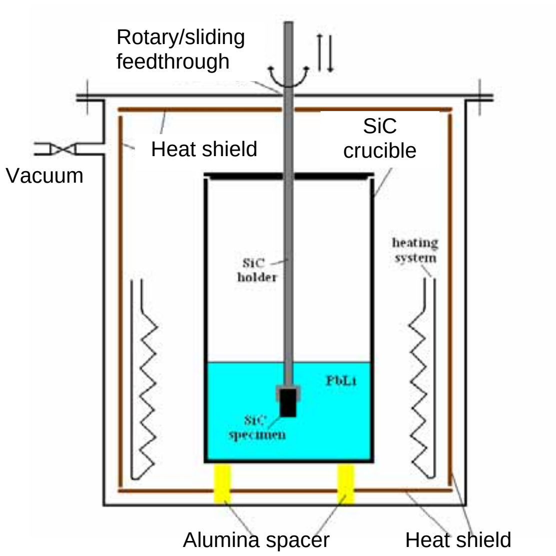

FIG. 6. Experimental apparatus proposed for the compatibility test between SiC materials and a Pb-17Li liquid metal.

4. Advanced Neutron Multiplier for DEMO Blankets

Metal beryllium pebbles are considered as the neutron multiplier in the Japan and EU pebble bed type test blanket modules for ITER. . However, DEMO reactor requires "advanced neutron multipliers" which have low swelling and high stability at high temperature in the blanket design of both EU and Japan. Both parties have been promoting the development of beryllides such as $Be_{12}Ti$ and $Be_{12}V$ as "advanced neutron multipliers". A big challenge is hereby the development of a manufacturing technology that allows the production of pebbles in spite of the extreme brittleness of beryllides. This R&D aims at to establish the fabrication technology of the beryllide mother rod and the pebbles and to evaluate characteristics of those materials.

As the preliminary R&D in Japan, pebbles consisting of Be-Ti alloys were obtained by a small-scale rotating electrode method [13]. Studies on mechanical and chemical properties and irradiation effects were performed for stochiometric $Be_{12}Ti$ (Ti content: 7.7 at%) fabricated by HIP as well as for Be-Ti alloys with the αBe phase fabricated by arc melting [14]. As to compatibility between Be-Ti alloys (Ti content: 0–8.5 at%) and structural material of F82H, the growth rate of the reaction layer on F82H decreased with increasing the Ti content up to 5 at%. Both oxidation and steam interaction were about 1/1000 as small as those for beryllium metal. These results indicate a possibility to reduce a risk of a water or air ingress accident and to achieve a blanket with high efficiency of electric power generation. Recently, compatibility studies between Be-V alloy and structural materials have begun with preliminary tests [15]. FIGURE 7 shows the growth rate of the reaction layer on F82H as a function of the reciprocal temperature. The growth rate of the reaction layer on F82H was about 1/1000 as small as that of beryllium and was approximately the same as that of Be-Ti alloys. Also as the preliminary R&D in EU, large rods (about 30 mm in diameter) of Be-30.8 wt%Ti were fabricated successfully in FZK (see FIG.8). For the produced beryllides, characteristics such as chemical composition, distribution of precipitation and crystalline structure were investigated.

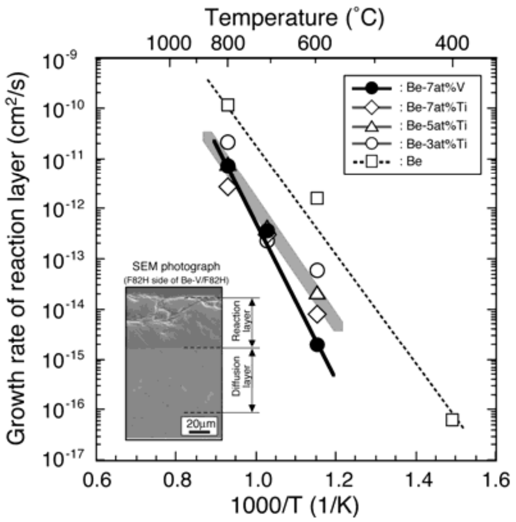

FIG.7. Growth rate of reaction layer of beryllium and beryllium-based intermetallic compounds on F82H as a function of the reciprocal temperature.

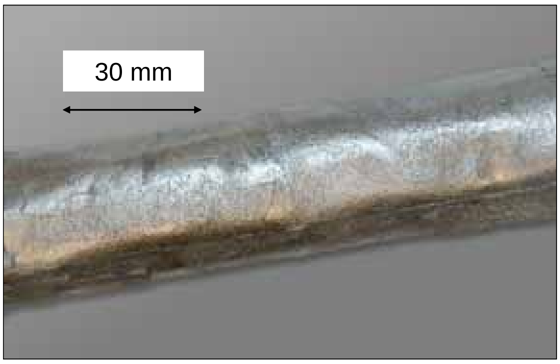

FIG.8 Picture of the large rods of Be-30.8 wt%Ti fabricated in FZK.

5. Advanced Tritium Breeders for DEMO Blankets

DEMO reactors require "^{6}Li-enriched tritium breeders" which have high tritium breeding ratios in the solid breeder blanket designs of both the EU and JA. Both parties have been promoting the development of fabrication technologies of Li_2TiO_3 pebbles by the direct wet process [16] and of Li_4SiO_4 pebbles by melt-spraying [17] as shown in FIG.9.

However, the fabrication techniques of tritium breeder pebbles have not been established for large quantities. In the BA activities, this R&D aims to develop fabrication techniques of advanced Li_2TiO_3 and Li_4SiO_4 pebbles, allowing effective reprocessing of ^{6}Li. The development of the production and ^{6}Li reprocessing techniques includes preliminary fabrication tests of breeder pebbles, reprocessing of lithium, and suitable out-of-pile characterizations.

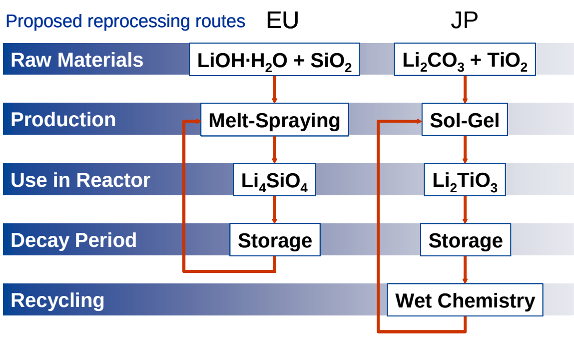

FIG.9. Fabrication flow of Li_2TiO_3 and Li_4SiO_4 pebbles.

As a preliminary R&D, Li_4SiO_4 pebbles were fabricated by the melt spraying method, and then the importance of the development of purification and reprocessing processes was pointed out in order to reduce long-term radioactivity and to control the content of Li isotopes and other components

6. Tritium Technology

The tritium technology is one of the most significant subjects for fusion DEMO plants. A series of basic studies has been carried out to develop an advanced tritium recovery system for an ITER test blanket module. In contrast to ITER, a series of continuous operations is required for the main fuel and the blanket tritium loops for a DEMO plant. The R&D

subjects for monitoring and analysis of tritium in these loops are required from the viewpoint of the control of the loops. In addition, the components of the loops in a DEMO plant are consequently exposed to tritium for a long period. Some new materials will also be used in a DEMO plant. Improvement of the basic tritium behavior database and demonstration of practical operation durability will enhance design, safety, and public acceptance for DEMO. Therefore not only R&D on the monitoring and analysis techniques of tritium but also tritium behavior studies in advanced materials to be used in DEMO will be carried out in the BA activities.

It is required to obtain basic data for the relation between tritium and materials from the viewpoint of the estimation of the amount of tritium permeation. The above basic data should also contribute to enhance safety designs and public acceptance for a DEMO plant. A basic tritium behavior database must be developed containing information such as solubility, diffusivity, permeability, characteristics of trapping, release, replacement, reaction etc. for advanced materials in DEMO. Tritium will be introduced into advanced material samples by various methods, such as thermal absorption, ion (plasma) implantation, etc. Tritium permeation and/or release experiments will be performed using the above samples and the basic characteristics of tritium behavior will be evaluated with the analysis of residual tritium amount in the material. The influence of impurities and traps on the tritium behavior in advanced materials will be also investigated. Also advanced tritium permeation reduction layers (permeation barriers) will be developed. To improve the physical stability of the barrier, oxidation layers for glass materials will be evaluated by the Permeation Reduction Factor (PRF) measurement and alternative barriers with self-repairing function will be investigated. Durability tests for the barrier will be performed by measurements of PRF before and after irradiation or exposure under fusion reactor conditions.

7. Summary

In the planned Broader Approach (BA) activities, DEMO relevant R&D on reduced activation ferritic martensitic (RAFM) steels as structural material, SiC_f/SiC composites, advanced tritium breeders and neutron multiplier for the blankets, and Tritium Technology will be performed. In the R&D on RAFM steels, the fabrication technology, techniques to incorporate the fracture/rupture properties of the irradiated materials, and methods to predict the deformation and fracture behaviors of structures under irradiation will be investigated. For SiC_f/SiC composites, standard methods to evaluate high-temperature and life-time properties will be developed. Physical and chemical properties such as He and H permeability and absorption under irradiation will be investigated for SiC_f/SiC and other related ceramics. On advanced neutron multiplier, Japan and the EU will develop the fabrication technique of beryllide mother rods such as $Be_{12}Ti$ and $Be_{12}V$, and will try to produce pebbles from the mother rods. On advanced tritium breeder, Japan and the EU plan to establish the production technique for advanced breeder pebbles of Li_2TiO_3 and Li_4SiO_4, respectively. Also, physical, chemical, and mechanical properties will be investigated for produced breeder pebbles. On tritium technology, tritium behavior in advanced materials to be used in DEMO, such as RAFM steels, SiC_f/SiC composites, and advanced breeders/multipliers, will be studied.

Acknowledgment

This report has been prepared within framework of the Agreement between The Government of Japan and the European Atomic Energy Community for The Joint Implementation of The Broader Approach Activities in the Field of Fusion Energy Research.

Reference

[1] MATSUDA S., "The EU/JA broader approach activities", Fusion Eng. Des. 82 (2007) 435-442.
[2] SALAVY, J.-F., et al., "Overview of the last progresses for the European Test Blanket Modules projects", Fusion Eng. Des. 82 (2007) 2105-2112.
[3] NISHITANI, T. et al., "Recent progress in solid breeder blanket development at JAEA", Fusion Sci. Technol. 52 (2007) 971-978.
[4] BALUC, N., et al., "Status of reduced activation ferritic/martensitic steel development", J. Nucl. Mater. 367-370 (2007) 33–41.
[5] TANIGAWA, H., et al., "Status and key issues of reduced activation martensitic steels as the structural materials of ITER Test Blanket Module and beyond", Fusion Energy 2006 (Proc. 21st Int. Conf. Chengdu, 2006), IAEA, Vienna (2007), CD-ROM file FT/1-4Rb and http://www.iaea.org/programmes/ripc/physics/fec2006/html/fec2006.htm.
[6] TAMURA, M., et al., "Development of potential low activation ferritic and austenitic steels", J. Nucl. Mater. 141/143 (1986) 1067-1073.
[7] MOESLANG, A., et al., "Towards reduced activation structural materials data for fusion DEMO reactors", Nucl. Fusion 45 (2005) 649-655.
[8] RAFFRAY, A.R., et al., "Advanced power core system for the ARIES-AT power plant", Fusion Eng. Des. 82 (2007) 217–236.
[9] NISHIO, S., et al., "Improved Tokamak Concept Focusing on Easy Maintenance", Fusion Eng. Des. 41 (1998) 357-364.
[10] NORAJITRA, P., et al., "Conceptual design of the dual-coolant blanket in the frame of the EU power plant conceptual study", Fusion Eng. Des. 69 (2003) 669–673.
[11] HINOKI, T., et al., "Current status of SiC/SiC composites for nuclear applications", Annales de Chimie: Science des Materiaux, 30 (2005) 659–671.
[12] NOZAWA, T., et al, "Determining the Fracture Resistance of Advanced SiC Fiber Reinforced SiC Matrix Composites", to be published in J. Nucl. Mater.
[13] KAWAMURA, H., et al., "Present status of beryllide R&D as neutron multiplier", J. Nucl. Mater. 329-333 (2004) 112-118.
[14] TSUCHIYA, K., et al., "Development of advanced tritium breeders and neutron multipliers for DEMO solid breeder blankets", Nucl. Fusion 47 (2007) 1300–1306.
[15] TSUCHIYA, K., et al., "Compatibility between Be-V alloy and F82H steel", to be published in J. Nucl. Mater.
[16] TSUCHIYA, K., et al., "Effects of gelation and sintering conditions on granulation of Li_2TiO_3 Pebbles from Li-Ti Complex Solution", Fusion Eng. Des. 75-79 (2005) 877-880.
[17] KNITTER, R., RISTHAUS, P., "Fabrication of lithium orthosilicate pebbles by melt-spraying -Reproducibility and yield-", Proc. CBBI-12, Sept. 16-17, 2004, Karlsruhe, Germany, FZKA 7078 (2004) 129-140.