FUSION MATERIALS AND FUSION ENGINEERING R & D IN JAPAN

A. KOHYAMA*, S. KONISHI and A. KIMURA
Institute of Advanced Energy, Kyoto University
1-1 Gokasho, Uji, Kyoto 611-0011 Japan
*Corresponding author. E-mail : kohyama@iae.kyoto-u.ac.jp

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Japanese activities on fusion structural materials R & D have been well organized under the coordination of university programs and JAERI/NIMS programs more than two decades. Where, two categories of structural materials have been studied, those are; reduced activation martensitic/ferritic steels (RAFs) as reference material and vanadium alloys and SiC/SiC composite materials as advanced materials. The R & D histories of these candidate materials and the present status in Japan are reviewed with the emphasis on materials behavior under radiation damage. The importance of IFMIF and technology development for blanket R & D including ITER-TBRG activity is emphasized and the current status of those activities in Japan is also presented.

KEYWORDS : Fusion, Materials, Blanket, Radiation, Japan, R&D, ITER-TBRG

1. INTRODUCTION

Materials R & D in Japan is based on the national fusion R & D strategy, where fusion research is clearly defined as the energy oriented and time-driven to meet ITER and DEMO schedule with the early realization of electricity supply from fusion reactor system [1,2].

After more than two decades of the intensive research activities on fusion engineering at Universities and JAERI, these activities have been unified under the management of the ministry of education, culture, sports, science and technology (MEXT) since 2003 with the emphasis on fusion materials and blanket engineering.

To meet the definition of fusion reactor materials R & D, two categories of structural materials have been studied, those are; reduced activation martensitic/ferritic steels (RAFs) as reference materials and vanadium alloys and SiC/SiC composite materials as advanced materials.

Together with the near term efforts on RAFs as reference materials, long range fundamental studies and development of advanced materials have been extensively pursued especially in the university programs [3,4].

IFMIF is defined as a crucial facility for materials qualification and ITER blanket module tests is also defined as an important milestone for technology integration to satisfy the ITER mission and to meet DEMO schedule. Thus the importance to proceed with IFMIF and ITER in a coordinated way is clearly indicated in the Japanese fusion R & D strategy. Figure 1 is a roadmap for materials and blanket development in Japan.

As is indicated in Fig.1, RAFs development and reference blanket development is the first priority under the JAERI responsibility and the advanced blanket system development coordinated with the advanced materials development is very important but the back-up activities under the Japanese university responsibility [5].

Based on the progresses in structural materials R & D, JUPITER-II, the phase-4 of the Japanese universities and the US DOE collaboration on fusion materials research, has been initiated since April 1, 2001. This is the integration activity of blanket and materials engineering, where self cooled liquid blanket and He-gas cooled solid blanket systems are of concern. The JAERI/ORNL collaboration Phase IV (FY1999-2003) was finished and the Phase V program has just started since April 1, 2004 as five year termed program emphasizing the importance to establish clear understandings and basis for blanket design from "neutron radiation effects up to high dpa".

2. KEY MATERALS R & D

2.1 Reduced Activation Ferritic/Martensitic Steels

Extensive efforts for the development of the reduced activation ferritic/martensitic steels (RAFs) have been accomplished in these decades including large heat productions of F82H and JLF-1. Those materials have been provided as IAE RAFs WG's reference materials.

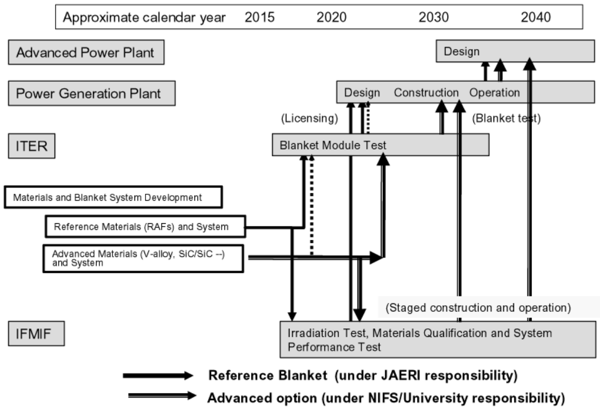

Fig. 1. Roadmap for Materials and Blanket Development in Japan [5]

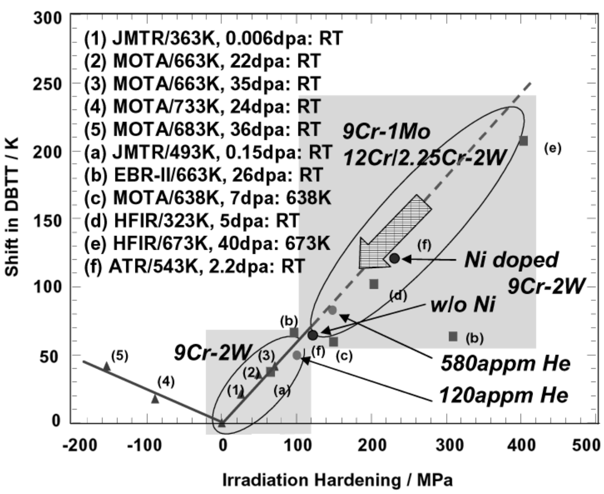

Fig. 2. Optimization of Total Performance for Fusion [6]

Data obtained on various physical, chemical and mechanical properties were compiled in relational database with emphasis on the traceability of the results up to the origin of the materials used. Under the extensive efforts on ITER test blanket module R & D, the importance to produce large heat of the candidate RAFs has been recognized and 9 ton heat of JLF-1 was melted and thick plate fabrication at the production line of Nippon Steel Company Co. Ltd., is on-going as NSC's effort to prepare commercial base production of RAFs for fusion. The production of 20 ton class RAFs for the Japanese fusion material program is under negotiation aiming at the early production in FY 2005.

The major accomplishments in these years are, examinations of the effects of neutron irradiation on (1) Ductile

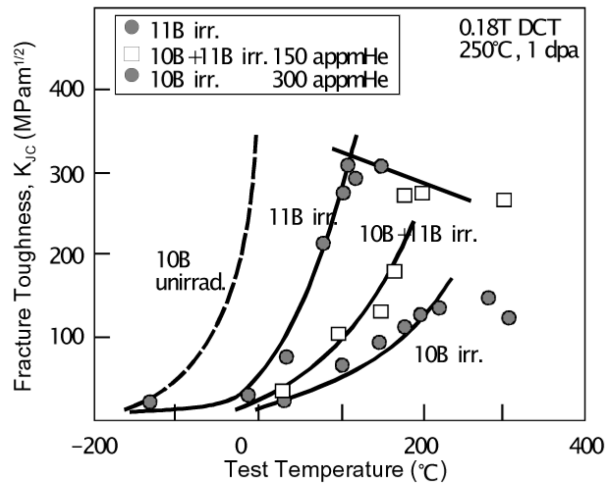

Fig. 3. Temperature Dependence of Post-Irradiation Fracture Toughness of B doped F82H [9]

to brittle transition temperature (DBTT) up to a damage level of 20 dpa to explore lower temperature limit, (2) Enhanced He effect on DBTT shift for Ni/B doped heats (isotopic tailoring method was used for B doping), (3) Fatigue behavior at relatively low temperatures, (4) Susceptibility to environmentally assisted cracking by the slow strain rate tensile tests (SSRT) in a high temperature pressurized water and (5) Flow stress-plastic strain relation obtained by measuring the profile of the specimen during tensile testing, including (6) the improvement of ductility of RAF/M ODS alloys with high temperature strength and other supporting researches. Accomplishments on (1), (2) and (6) are oriented for the materials response to high irradiation damage levels, while (3), (4) and (5) are expected to contribute to ITER TBM development.

Figure 2 indicates the direction to suppress irradiation embrittlement in RAFs. By substituting Mo into W and the optimization of Cr and W to 8-9% and 2%, respectively, together with the optimization of whole process, irradiation hardening and embrittlement have been largely suppressed [6]. The effect of helium shown in this figure is not significant. Mechanistic studies by microstructure evolution analysis and nano-indentation also indicate the less concerns about radiation hardening and embrittlement up to high dpa with helium than those anticipated [7,8]. Heats doped with different ratio of 11B and 10B were used to analyze He effect as isotopically tailoring technique with improved accuracy. The results shown in Fig. 3 are clearly indicating enhancement of embrittlement and hardening with helium but the effect is about one third or smaller than has been indicated previously.

Two concepts of ITER TBM are of concern for us; water cooled/ceramic breeder and He gas cooled/ceramic breeder concepts. The current target is (a) water temperature

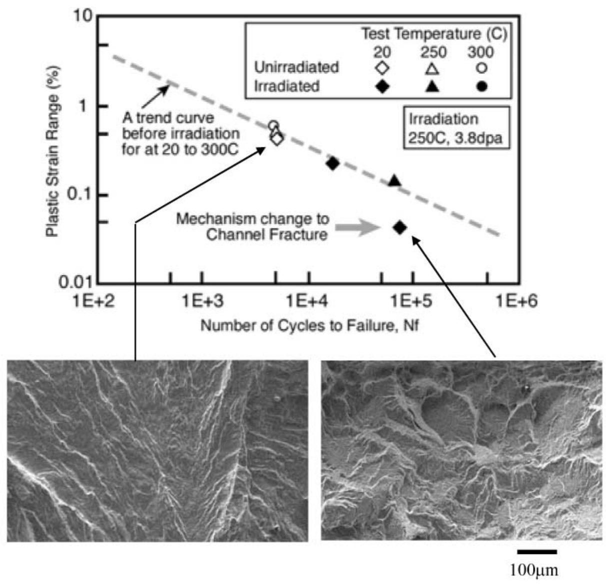

Fig. 4. Fatigue Behavior of F82H After Neutron Irradiation in JMTR [9]

ranging 290 and 520°C and (b) the damage level is about 3 dpa with 30appmHe.

Irradiation effect on fatigue properties has been examined [9,10] with the similar findings. The recent results from JMTR irradiated specimens, shown in Fig.4, indicate that irradiation did not introduce appreciable change in fatigue life, except for the results with a very small cyclic strain range. However, irradiation introduced fatigue mechanism change was observed at a smallest plastic cyclic strain range of about 0.1% and smaller. This will be an issue to be confirmed and be evaluated.

Due to the concern that irradiation hardening may also increase susceptibility for cracking in high temperature pressurized water, slow strain rate tensile tests with strain rate of about 1 x 10-7/s were conducted for the specimens irradiated to 3 dpa at 250°C in a water environment of PWR condition. So far no obvious change in engineering stress-strain curves (see Fig. 1) and fracture mechanism was detected [9].

To increase the attractiveness of fusion utilizing RAFs, extensive R & D on oxide dispersed strengthening (ODS) steels has been and is on going [11,12]. Figure 5 indicates the progress in ODS development on 9Cr 2W type ODS. The excellence in the newly developed ODS steel compared with PNC-FMS and PNC316. Rupture times at 650 and 700 C are 60 and 100 times longer than those with PNC-FMS, respectively.

2.2 Vanadium Alloys

Recent researches on vanadium alloys have successfully resolved many of the critical issues and enhanced the feasibility of vanadium alloys as fusion blanket structural materials [3,13]. The research emphasis of vanadium alloys for fusion has been made on V-4Cr-4Ti alloy as reference composition. Large and medium heats of V-4Cr-4Ti have been made in the US[14], Japan[3,15] and Russia[16]. Especially high purity V-4Cr-4Ti ingot made by collaboration of NIFS and Japanese Universities (NIFS-HEATs) showed superior manufacturing properties due to the reduced level of oxygen impurities[3,15].

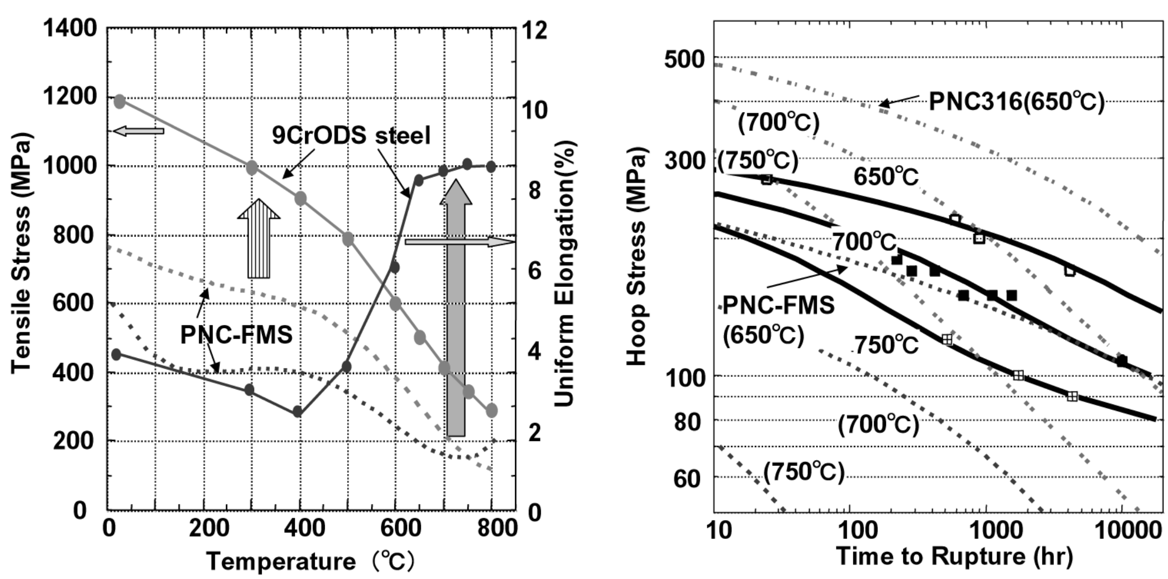

Fig. 5. 9Cr type ODS RAF Steel for Fusion, Comparison with PNC-FMS and PNC316 [12]

Significant progress has been made recently in fabrication and welding technology, applicable to industrial scale manufacturing, for V-4Cr-4Ti alloys by improved control of interstitial impurities. Figure 6 provides recent efforts on NIFS large heat production and efforts to be done. Development of advanced vanadium alloys by minor addition of Y, Al and Si is also in progress for improved radiation and oxidation resistance.

The examination of NIFS-HEAT on process dependence showed that control of Ti-CON precipitates is crucial for mechanical properties of the V-4Cr-4Ti products. Since the band structure of the precipitates should result in anisotropic mechanical properties, rolling to high working degree was necessary for homogenizing the precipitate distribution. Thin pipes, including those for pressurized tube creep specimens, were successfully fabricated controlling the grain size and precipitate distribution by controlling the cumulative working degree between the intermediate heat treatments [17].

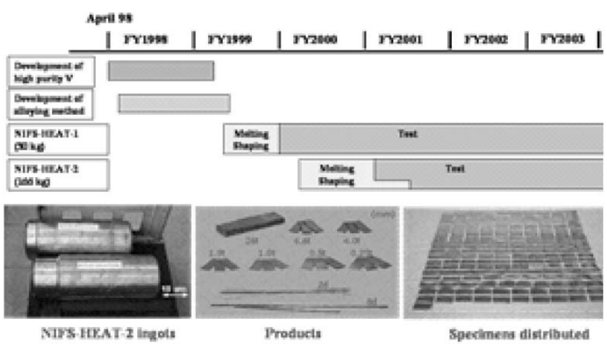

Fig. 6. Production and Characterization of High Purity V-4Cr-4Ti (NIFS-HEATs)

Feasibility of joining of V-4Cr-4Ti was demonstrated by laser welding. Optimization of the Post Weld Heat Treatment (PWHT) was made by controlling the precipitate distribution in weld metals. Limited data on irradiation effects on the weld joint were derived, which showed elimination of radiation–induced degradation of the joint by applying appropriate PWHT conditions [18].

Low pressure plasma-spraying method was applied for coating W on V-4Cr-4Ti. No hardening of substrate V-4Cr-4Ti was observed by the coating [19].

One of the concerns by reducing interstitial impurities in vanadium alloys is a loss of strength at high temperature. Because thermal creep of NIFS-HEATs were shown to be similar to other V-4Cr-4Ti alloys [20], reduction of the impurities to ~100wppm is thought not to induce significant loss of strength. However, significant increase in creep strain rate was reported recently in V-4Cr-4Ti with oxygen level of ~10 wppm[21].

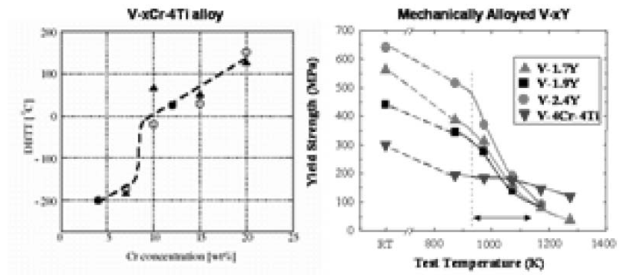

Fig. 7. Improvement of Vanadium Alloys

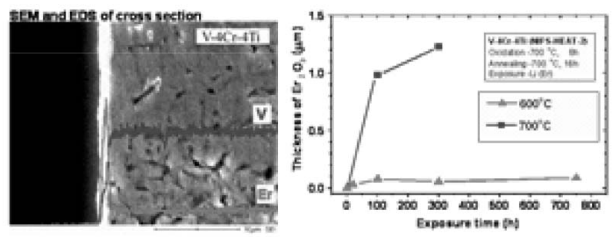

Fig. 8. In-Situ Formation of Er2O3 Layer

The corrosion of vanadium alloys in oxidizing environments is a concern for the performance of the pipe exterior and the impact of air leak. The data should be also useful for the database of vanadium alloys in non-Li coolant system. Addition of Si, Al and Y and increase in Cr level was shown to be effective in suppressing the corrosion in air and water environment, respectively [22,23].

There have been continuous efforts in Japan to improve vanadium alloys by changing composition from V-4Cr-4Ti or to change the fabrication processes. Figure 7 indicate representing results of the efforts. Increase in Cr is known to increase high temperature strength in V-Cr-Ti, bartering with loss of ductility at low temperature. As shown in the figure, the DBTT does not change significantly to the Cr level of 7% in V-xCr-4Ti alloys [24]. Also, mechanically alloyed V-Y alloys were fabricated and their strength improvement from V-4Cr-4Ti below 800C. The irradiation response indicated that the fine grain and oxide dispersion inhibited formation of interstitial loops in the matrix by neutron irradiation, because of the enhanced defect sink [25]. Thus mechanically alloyed vanadium alloys have potential to extend the low temperature limit.

MHD insulator coating is a critical feasibility issue for Li self-cooled blankets with structural vanadium alloys. Significant progress has been made in developing MHD coating recent years, partly enhanced by JUPITER-II program. Promising candidate ceramics of Er_2O_3 and Y_2O_3, which were stable to 1073K in liquid lithium, were identified [26]. Feasibility of the coating with Er_2O_3 and Y_2O_3 on V-

4Cr-4Ti was demonstrated by Arc Source Plasma Deposition [27] and RF sputtering [28].

In addition to the physical deposition, in-situ coating with Er_2O_3 on V-4Cr-4Ti is being developed [29]. As shown in Fig. 8, Er_2O_3 thin insulating layer was formed on V-4Cr-4Ti during its exposure to liquid lithium by reaction of pre-charged oxygen in the vanadium alloy and pre-doped Er in Li. Er_2O_3 layer grows and saturates and was stable at 6oo C to 750 hours.The in-situ coating is a quite attractive technology because it has the potential to coat on complex surfaces after fabrication of components and to heal coating cracks without disassembling the component.

As a result of recent significant progress in developing vanadium alloys, critical issues for future research are now focused into limited numbers. As to the available data, thermal and irradiation creep, helium effects on high temperature mechanical properties and radiation effects on fracture properties are insufficient. Conclusive evaluation of irradiation properties seems to be possible only with the use of 14MeV neutron source, motivating the construction of IFMIF.

Impurity (C, O, N) and precipitate (Ti-CON) control are crucial for the mechanical properties of the fabrication products. Systematic study to optimize the microstructure and mechanical properties are necessary for enhancing the performance of vanadium alloys.

Although recent progress in MHD coating is large, further intensive efforts are necessary. Concept exploration efforts are being made to consider variety of blankets using vanadium alloys, which include potential use of gas and molten salt for the coolants. General common necessary technology would be coating including those for MHD insulation, corrosion protection and tritium diffusion barrier. Close coupling of the materials development and the blanket design is crucial.

2.3 SiC Fiber Reinforced SiC Composite Materials

The concepts of fiber reinforced materials as advanced tailored materials had been developed in the later part of the 20 century and many extensive R & D efforts have been performed in these decades [30,31].

In these decades, R & D efforts on ceramic composites have been very extensive, especially in the fields of aerospace and energy. Among them, C/C and SiC/SiC R & Ds have been very much emphasized in nuclear energy research [32, 33].

As is indicated in Fig. 1, SiC/SiC is under extensive R & D as one of advanced materials for fusion and material R & D is coordinated with technology integration toward blanket fabrication for ITER and DEMO reactors.

The current Japanese efforts have strong concerns on two types of blanket systems, those are helium cooled solid blanket and He/Pb-Li dual cooled liquid blanket. In both systems, SiC/SiC has to be compatible with coolant and tritium breeder and Be for the case of solid blanket at high temperature and has to keep low tritium retention, which may require optional “sealing layer”, such as W coating. The high thermal conductivity requirement is strictly required for the case of solid blanket, but for the case of liquid blanket, low electrical conductivity is required. These two very much different requirements for materials need new methodology of materials R & D.

The unique features of materials R & D methodology for SiC/SiC are just right for fusion materials R & D. The varieties of materials processing options available for SiC/SiC have been investigated and the combined process of Polymer Impregnation and Pyrolysis (PIP) and Melt infiltration (MI), Chemical Vapor Infiltration (CVI) process and Nano-powder Infiltration and Transient Eutectic (NITE) process are current emphasis in the Japanese program [32].

The major efforts of materials R & D are CREST-ACE program (1998-2002) and IVNET program for gas cooled fast reactor (2003-2005) for non-fusion activities [34] and Japan USA collaborative program JUPITER-II (2001-2006) for fusion research [35]. Advanced SiC/SiC fabrication and evaluation is extensively done under JUPITER-II collaboration, mainly with ORNL, Kyoto University and Tohoku

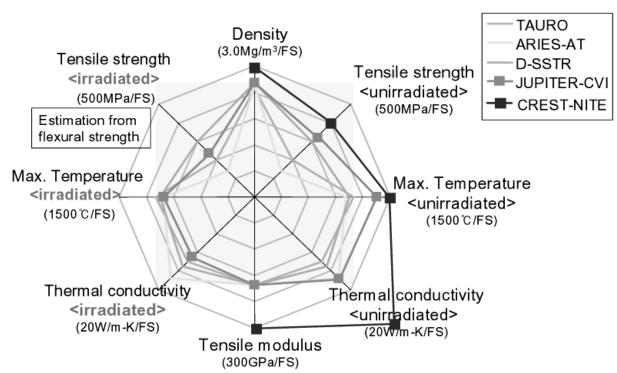

Fig. 9. Current Status of SiC/SiC Composites R & D in Japen

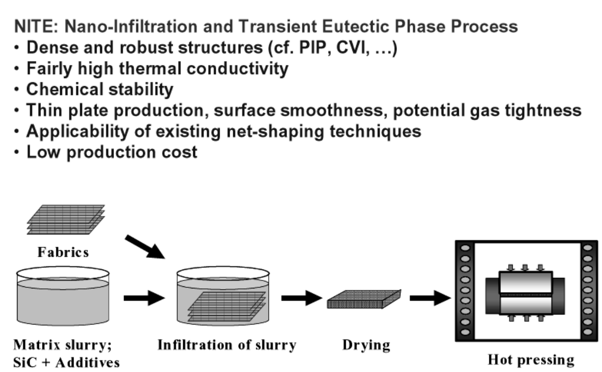

Fig. 10. The Concept of NITE Process

Fig. 11. Shape Flexibility of NITE SiC/SiC

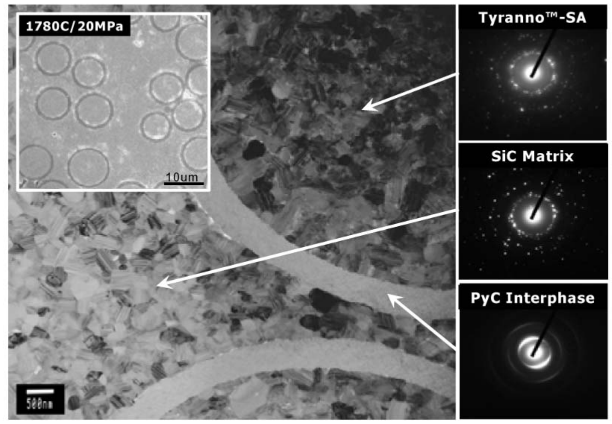

Fig. 12. Cross-Sectional Images of NITE SiC/SiC[4]

University [36, 37]. Figure 9 indicates representing data of SiC/SiC composites developed in Japan with typical design values used for fusion power reactor designs, such as TAURO(EU), ARIES-AT(US) and D-SSTR(Japan). FCVI (Forced-flow Chemical Vapor Infiltration) method has been developed under the collaboration with ORNL and Kyoto University and the properties in Fig. 9 was the data on 1999. CREST-NITE was developed by CREST program (1997-2002) and the value in the figure was the data on 2001. As is shown in the figure, the current status of SiC/SiC is mostly over driven the design values for many power reactor studies.

Due to the improvements in reinforcing SiC fibers and availability of fine nano-SiC powders well know liquid phase sintering process was drastically improved to become a new process called the Nano Infiltration and Transient Eutectic Phase (NITE) Process. As is indicated in Fig.10, slurry of SiC nano-powders and additives is infiltrated into SiC fabrics and dried for making pre-preg sheets. After the lay-up of the sheets, hot pressing is applied to make NITE SiC/SiC. For the near net shaping process to make tubes, pipes, turbine blades, blanket for fusion reactors, and this basic process is modified to include pre-preg wire production, filament winding or 3D fabrics weaving and pseudo HIP process. To keep the advantage of NITE process, the followings are essential and to satisfy these requirements in industrial fabrication line production is still under way. Those are, (1) use near stoichiometry SiC fiber with high crystallinity, (2) make protective interface by fiber coating of carbon and SiC, (3) use SiC nano-powders with appropriate surface characteristics [38]. One of the biggest advantages of the NITE process is its flexibility in shape and almost no limitation on size. Figure 11 indicate some examples of composite materials made by the NITE process. About 1 liter volume 2D SiC/SiC composite cubic blocks was successfully produced (as shown at upper left in Fig. 11), where no cracks or cavities were detected by naked eyes. The upper right of Fig.11 is the real size model of 100KW gas turbine combustor liner. Lower left of the figure is 2mm thin plate of 2D SiC/SiC. In these materials, basic properties have been measured. They all presented excellence in high density, high crystallinity, high thermal conductivity and basic mechanical properties. Process improvement and optimization with the emphasis on maintaining sound protection interface are current technical challenges.

The outstanding total performance of NITE SiC/SiC composite material is based on the highly crystalline and highly dense microstructure. Figure 12 shows low magnification SEM image of cross section perpendicular to the fiber axis (top left image) and high magnification TEM image of fiber-interface-matrix. These images present the full dense and small crystalline microstructure of SiC with carbon interface. The selected area diffraction images (right side images) clearly define that fiber and matrix are highly crystalline beta-SiC and interface is pyrolytic carbon. These micro structural features provide excellent thermal stress figure of merit and high hermetic property presented by helium permeability, as shown in Figures 13 and 14, respectively.

Figure 13 represents potential of thermal stress tolerance, M; thermal stress figure of merit, defined by the equation in the figure. Where,

σUTS : Ultimate tensile strength,
K_{th} : Thermal conductivity,
γ : Poisson's ratio,
E : Young's modulus
a_{th} : Thermal expansion coefficient,

Higher M value indicates the excellence in thermal stress tolerance and the well known high temperature metallic material, Inconel 600 is the worst in this figure in all temperature range. The conventional and commercially available CVI-SiC/SiC is slightly better than Inconel 600.

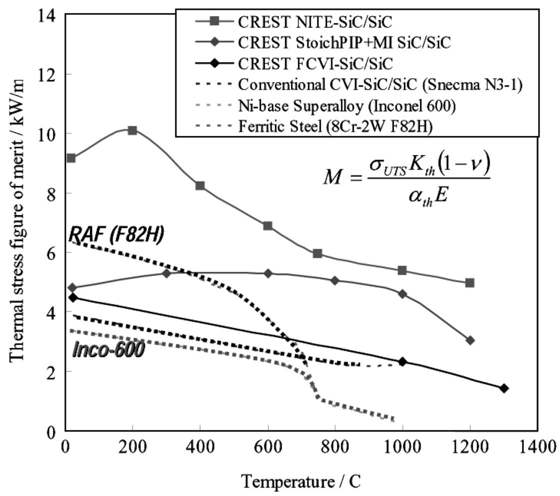

Fig. 13. Thermo-Mechanical Property of SiC/SiC

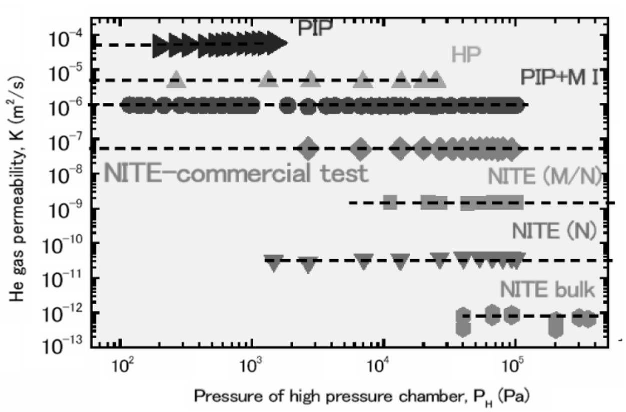

Fig. 14. He Permeability of NITE SiC and SiC/SiC[39]

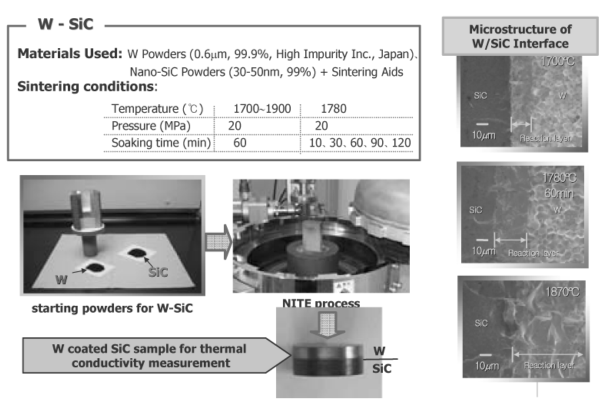

Fig. 15. Tungsten Coating on SiC by NITE Process

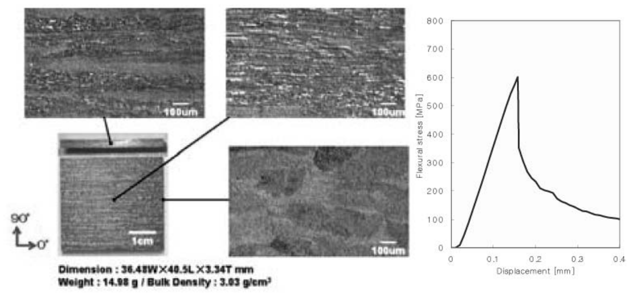

Fig. 16. The Second Large Scale Production Results of NITE SiC/SiC at Ube Industries

F82H is one of the candidate reduce activation ferritic steels for fusion reactor with the chemical composition of 8%Cr and 2%W. F82H steel presents very excellent property, especially below 600°C. The SiC/SiC composite produced through CREST-ACE program [33] by stoichiometry PIP (polymer impregnation and pyrolysis) followed by melt infiltration method, as indicated CREST StoichPIP+MI SiC/SiC, is quite excellent in almost all temperature. But, SiC/SiC made by NITE process, during CREST-ACE program, has very high M value from room temperature to 1300°C. The impact of this high M value is enormous on designing of high temperature component such as gas turbine combustor liner, turbine blade, fuel pin, reactor core components and heat exchangers. For the high temperature gas system application, gas leak tightness or hermetic property is very important, but unfortunately ceramics are well known as very inferior materials from hermeticity view point, especially for ceramic composite materials with high porosity and micro-cracks. The NITE SiC/SiC is becoming the first leak tight ceramic fiber reinforced matrix composite. Figure 14 is the comparison of helium permeability among various SiC/SiC composites and monolithic NITE-SiC. Monolithic SiC by the NITE process keeps its permeability at the level of $10^{-12}m^2/s$, which is near the level of ordinary metallic materials. Laboratory products of NITE SiC/SiC present 10^{-11} to 10^{-9} m^2/s and the first pilot production of the NITE SiC/SiC was not as good as laboratory products, still the level of 10^{-8} m^2/s is outstanding comparing with other SiC/SiC composite materials. This result is encouraging to produce shield fuel pin of SiC/SiC or to produce components of gas-cooled blanket for fusion reactor without applying any shielding layer [40]. As the backs-up option, W coating on the surface of SiC/SiC is also on-going. Figure 15 presents outline of W coating on SiC by NITE process. In this case, starting materials were both fine powders and by lay-up two zones of powders (slurry) followed by hot pressing, two layers of W and SiC were produced with strong bonding. By optimizing the materials used and process condition, SiC, W and their interface reaction layers can be controlled to make acceptable joining, bonding or claddings. Shear strength of SiC/SiC joints has been reported and 31.6 MPa [41] and 15.1 MPa [42] are the typical results, where the shear stress obtained from SiC/SiC joint made by NITE method shows larger than 52MPa even with about 40% unjoined area. This result is encouraging and neutron irradiation and supporting simulation irradiation research are on-going. Although the recent results on radiation effects in advanced SiC/SiC, like NITE–SiC/SiC and FCVI-SiC/SiC, are not provided, the resistance to radiation damage has been greatly improved by these advanced processes [43].

2.4 Refractory Metals and Alloys

Refractory metals such as W and Mo are attractive materials for fusion, but the well known serious embrittlement in several regimes; i.e., low-temperature embrittlement, recrystallization embrittlement and radiation embrittlement has been the serious technical issue. These types of embrittlement are microstructure-sensitive and the efforts to make the most effective microstructure to alleviate such embrittlement consists of fine grains and finely dispersed nano-particles have been reported by materials processing based on powder metallurgical (P/M) methods including mechanical alloying (MA) and hot isostatic pressing (HIP) under a controlled atmosphere with negligible amounts of oxygen and nitrogen. [44,45]. With the improved process W-0.3%TiC-(0.7~1.7)%Mo alloys having fine grains of 0.6~2.0 mm and nano-sized TiC particles around 10~20 nm in diameter, associated with uniform distributions, and relative densities of 99% or more in the as-HIPed state were fabricated. The representing results are shown in Fig. 17. Three-point static bending tests at room temperature showed that the W alloys in the as-forged and as-rolled states exhibited appreciable ductility, whereas those in the as-HIPed state exhibited no ductility before fracture, demonstrating the importance of hot plastic working to improve the ductility of the alloys. It was found that this beneficial effect of plastic working on ductility became prominent with decreasing grain size of the HIPed compacts. This indicates the need of fabricating consolidated bodies with very small grain size less than 0.6 μm. [46].

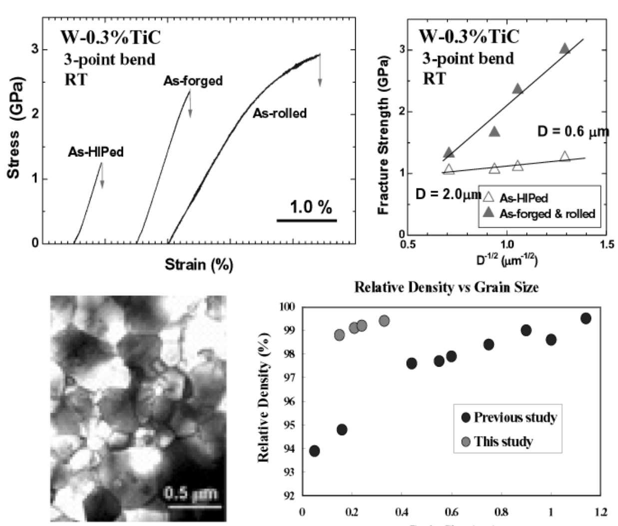

Fig. 17. Development of Ultra-Fine Grained, Nano-Particle Dispersed W-TiC Alloys with Improved Ductility[46]

2.5 Integration of Breeding Blankets and Materials for ITER TBM

As written in the Introduction and the Fig.1., Japanese strategy for the development of materials aims at its target to integrate them into the concepts of the blanket. All the possible concepts consider the ITER test blanket module (TBMs) as the most important milestone, and the R&D will be focused on either the first or the later generations of the TBMs. In Japan, solid breeder cooled with water with RAFMs as structural material is the primary candidate concept for the blanket module, and pursued mainly by JAERI with universities participation[47]. This type of blanket is developed for the first generation tokamak fusion plant in Japan[48]. Alternative advanced concept of the blankets are mainly studied by universities. Particular interests on the liquid blankets are focused on FLiBe, liquid Li, and LiPb, studied by NIFS and universities. A helical reactor design, FFHR [49] is based on the Flibe self-cooled concept with the data in universities. [50,51]. R&D for Li/V blanket is lead by the development of V-4Cr-4Ti as structural materials[3].

Particular interest is attracted by the LiPb dual coolant concept that is studied as an option in TBM for ITER, while early works on LiPb were conducted on basic characteristics such as solubility and diffusivity of tritium [52]. This strategy involves development of module with stagnant LiPb breeder and He coolant, and SiC insert for electrical and heat insulation to flow LiPb at higher temperature and flow rate to utilize heat to better utilize the fusion heat. The ultimate goal of this concept is LiPb/SiC blanket[53] to be operated at above 900 degree C for high efficiency and hydrogen production [54]. By the dual coolant concept, LiPb blanket can be tested at lower temperature in ITER with conservative technology, and is expected to gradually improve its performance of advanced blanket.

Above three liquid blanket concepts have been studied by Universities and NIFS, mainly as fundamental researches so far. However considering the situation of ITER, and necessity for the next phase beyond it, these advanced concepts are required to plan the next phase of research of the engineering development in ITER/TBM with international collaborative efforts for development.

One of the marked recent activities is the Korean involvement in the TBM study in ITER[55]. Korean effort is currently focused on Helium Cooled Solid Breeder and Helium Cooled Molten Lithium concepts. Since they are in early stage of development now, if the research activities come to the adequate phase, they will have to be pursued by international collaboration frameworks including Japanese involvement in near future. Korean nuclear program includes strong tritium related activities related to the heavy water reactor and capability of tritium production with them [55,56]. Such a tritium technology is of extreme importance in integration of blanket development and the contribution is highly expected.

3. CONCLUSION

Japanese activities on fusion structural materials R & D in these two decades have been providing many progress in two categories of structural materials, such as reduced activation martensitic/ferritic steels (RAFs) as reference material and vanadium alloys and SiC/SiC composite materials as advanced materials. Those results are encouraging to emphasize the efforts to integrate materials knowledge and fusion reactor engineering and technology via ITE to DEMO and Power Reactors. Although the recent results on radiation effects were not provided here, the on-going Japan-USA collaborations on blanket engineering and materials, JUPITER-II Program and on neutron radiation effects, JAERI/ORNL Phase IV and V are continuously producing important data, which will give the stronger basis to proceed ITER and the beyond.

As described in the previous section, material studies are entering the new phase of engineering development. One of the focuses of the efforts is irradiation with fusion relevant neutron source such as IFMIF, and another important activity is material integration into the blanket modules for ITER and beyond. Particularly in the development of blanket for fusion power plant, tritium processing technology required for the development of both recovery of bred tritium and extraction of tritium from heat transfer media are major subject to be studied. Korean involvement is highly expected because of its experience and capability related to heavy water reactor technology.

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