Current Status of Fusion Reactor Structural Materials R&D

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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 ferritic/martensitic 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 brief explanation of Japanese strategy and current status of fusion reactor engineering R&D.

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1. Introduction

Fusion reactor structural 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 (International Thermonuclear Experimental Reactor) and Demonstration Reactor (DEMO) schedule with the early realization of electricity supply from fusion reactor system. After more than two decades of the intensive research activities on fusion engineering at Universities and Japan Atomic Energy Research Institute (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.

The long-term research program of fusion blankets was approved by the Fusion Council of Japan in 1999. According to the established program, JAERI has been nominated as a leading institute of the development of solid breeder blankets, in collaboration with universities, as the primary candidate blanket for the fusion power demonstration plant, while, universities including National Institute for Fusion Science (NIFS) are assigned mainly to develop advanced blankets for potential options of the fusion power demonstration plant and commercial power plants. In the R&D program, ITER test blanket module testing and irradiation testing using International Fusion Material Irradiation Facility (IFMIF) are the most important milestones of the blanket development. The construction of the fusion power demonstration plant will be decided by blanket module performance and integrity data by ITER blanket module testing and material irradiation data by IFMIF. Therefore, Japan is performing the R&D on all types of blankets for primary candidate blanket and advanced options of blankets, aiming at ITER blanket module testing and the fusion power demonstration plant. Based on the R&D progress, Japan is investigating the possibility of testing all types of blankets under ITER-Test Blanket Working Group (TBWG) framework with both of JAERI and universities/NIFS involvements.

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. The reference materials have been developed for early realization of DEMO and first generation power plant, while the advanced materials are for increasing attractiveness of the power plant where cost of electricity (COE) and environmental benignity are of big concerns. 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. 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 this is a back-up activity under the Japanese university responsibility. In all essential issues of reference blanket development, elemental technology development has been almost completed and is now stepping further to the engineering R&D phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to clarify the total structure integrity for final specification decision of test blanket modules. As the advanced blanket options, solid breeder blanket module with SiCf/SiC structure cooled by high temperature helium gas, liquid LiPb breeder with SiCf/SiC cooled by helium, liquid Li self-cooled blanket made by V alloy and molten salt self-cooled blanket made by RAFM are under development by universities and NIFS with cooperation. Key issues have been addressed and critical technologies are being developed.

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 (FY 1999–2003) 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”.

Figure 2 indicates representing fusion facilities in Japan. JAERI Naka Fusion Establishment currently operates JT-60U and JFT-2M Tokomak facilities, where many engineering R&D activities are ongoing including research of ferritic steel effects on plasma behavior. NIFS is operating Large Helical Device (LHD) and Plasma Surface Interaction (PSI) research has been emphasized including collaborative work with TRIAM at Kyushu University. Laser Fusion Research Center at Osaka University is accelerating its efforts to expand engineering activity. Kyoto University operates Heliotron-I and extensive fusion materials/engineering R&D are ongoing based on MUSTER (Multiscale Testing and Evaluation Research) Facility and DuET (Dual-ion irradiation Experiment and Testing) Facility. Other fusion facilities are mainly on plasma physics and fusion science, but are looking for the next step toward fusion engineering.

2. Research Structure in Japan

Fusion engineering and materials research is widely expanding research with the strict requirement of technology integration to realize fusion reactor system for fusion energy power supply within a reasonable time span. This definition requires well organized near term, mid term and long term strategies. “The long-term research program of fusion blankets” was approved by the Fusion Council of Japan in 1999 and the report; “The recommendation for the fusion engineering activity in Japan” was issued by the Nuclear Fusion Special Committee, the Science Academy of Japan in 2002. “White paper on nuclear energy 2003” by Atomic Energy Commission of Japan (AEC) expresses that “ITER project is a joint international project undertaken with the objective of demonstrating the scientific and technological feasibility of using nuclear fusion energy for peaceful purposes” and that “the Council for Science and Technology Policy drew up a set of conclusions in May 2002 stating that in addition to offering full governmental support for the ITER project, the government ought to select the optimal candidate site (Rokkasho-mura) for the project and engage in talks with other governments with an eye to attracting ITER to Japan”. Under the governmental policy, JAERI is defined as the responsible research organization for ITER and near-term fusion research. Naka Fusion Establishment is responsible for fusion research with participation from Tokai establishment for fusion materials, from Takasaki establishment for simulation radiation damage research and from Oarai establishment for fusion functional materials as representing activities. Japanese university activities have been very important for more than three decades and now major fusion research programs are under management and coordination of NIFS and International Research Center for Quantum Materials, Tohoku University at Oarai is taking an important role in fission reactor radiation research and IFMIF R&D. The volunteer base activities to support Japanese fusion research are very important, of which Fusion Research Network, Fusion Forum and Atomic Energy Society of Japan for “Fusion Engineering” and “Nuclear Materials” are the representing activities. Fusion research net work consists of three sub-net works; those are Fusion Science, Fusion Plasma and Fusion Engineering. Figure 3 indicates major participants to Fusion Engineering Net work where (1) materials and fuel, (2) magnet and electromagnetics, (3) system and safety and (4) ICF engineering are current activities. As is indicated in the figure, more than 40 universities with more
Fig. 2 Fusion facilities in Japan.

Fig. 3 Fusion engineering network in Japanese university.
than 70 research sectors are joining and Hokkaido U., Tohoku U., U. Tokyo, Nagoya U., Kyoto U., Osaka U., Kyushu U., Toyama U. and NIFS are quite active in R&D and information exchange. Fusion Forum is another important activity, where under the coordination committee (Chaired by Prof. T. Tamano) 4 clusters (Fusion Engineering, Plasma Physics, Fusion and Society, and Fast realization of fusion energy) are currently activated. The main mission of the Fusion Forum is the efficient use of Japanese research resources toward the utilization of fusion energy in the earliest chance including the best use of the ITER project. For the year of 2004, the activity to participate ITER Test Blanket Working Group (TBWG) and the activity to optimize the Japanese R&D program of reduced activation ferritic/martensitic steel as a fusion structural material are on going.

3. Fusion Materials Research

As is clearly defined in the Japanese science and technology policy in 2003, fusion is an important energy option for Japan. Thus fusion energy should be attractive and competitive as the future option. Under the assumption that fusion plasma satisfies self ignition condition and is well controlled, fusion material becomes the most important factor for improving attractiveness and competitiveness of fusion energy through fusion reactor design mainly on power density, energy conversion efficiency, availability, safety and environmental attributes. Figure 4 is an example of fusion reactor design indicating major components of fusion reactor. Each reactor component has its own mission and requirements to materials are different. But, in many areas slight modification of commercially available materials can be used not only to ITER but also for DEMO and Power Reactors. The ITER Design Report provides major materials and material related technology R&D status and “blanket and first wall”, “divertor” and “super conducting magnet” are clearly defined as key technology areas where material and technology innovation are essential.

Although from ITER Design Report, even 316 Stainless Steel looks acceptable to satisfy the ITER operation condition, the following technical issues are important to be sufficiently overcome.

- Materials should keep their performance under high heat load, high neutron load and injection of D.T. He from the core plasma for reasonable time-period.
- Plasma-facing components should be replaced with reasonable time-period to eliminate detrimental effects to core plasma.
- Replacement scenario of blanket or its component should be reasonable for preventing significant loss of economical advantage of the fusion reactor.
- Long lifetime under neutron irradiation is the critical issue for the structural materials.

In this special issue on Fusion Structural Materials R&D, the recent accomplishments and future prospects of materials R&D is precisely reported. However, to give readers the brief outline of major materials R&D the following sections cover reduced activation ferritic/martensitic steels and Oxide Dispersion Strengthened (ODS) steels, SiC/SiC composite materials, Vanadium and its alloys, and refractory metals and alloys. Figure 5 provides current scenario about materials R&D in Japan connecting with the R&D scenario of test blanket development for ITER and beyond including IFMIF as the important materials irradiation facility to confirm materials behavior under fusion neutron exposure.

4. Key Materials R&D

4.1 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. 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., (NSC) 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 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 RAFs and ODS steels 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 test blanket.
module (TBM) development.

Figure 6 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.\(^{10}\) The effect of helium shown in this figure is not significant. Mechanistic studies by microstructure evolution analysis and nano-indentation also indicate the less concern about radiation hardening and embrittlement up to high dpa with helium than those anticipated.\(^{11,12}\) Heats doped same amount of B, but with different ratio of \(^{11}\)B and \(^{10}\)B were used to analyze He effect as isotopically tailoring technique with improved accuracy.

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

The results shown in Fig. 7 are clearly indicating enhancement of embrittlement and hardening with helium but the effect is about one third or smaller than has been indicated previously. The recent results on fatigue behavior from JMTR irradiated specimens are shown in Fig. 8. Irradiation effect on fatigue properties has been examined in Japan\(^{13,14}\) and they concluded 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.

To increase the attractiveness of fusion utilizing RAFs,
extensive R&D on oxide dispersed strengthening (ODS) steels has been and is on going.\textsuperscript{15,16)} Figure 9 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.\textsuperscript{16)}

4.2 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.\textsuperscript{3,17)} 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,\textsuperscript{14)} Japan\textsuperscript{3,19)} and Russia.\textsuperscript{20)} 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.\textsuperscript{3,19)}

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 10 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-C-O-N 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 with appropriate grain size and precipitate distribution by controlling the cumulative working degree between the intermediate heat treatments.\textsuperscript{21)}

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, showing elimination of radiation-induced degradation of the joint by applying appropriate PWHT conditions.\textsuperscript{22)}

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 systems. Addition of Si, Al and Y and increase in Cr level were shown to be effective in suppressing the corrosion in air and water environment, respectively.\textsuperscript{23,24)} 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 11 indicates 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.\textsuperscript{25}) Also, mechanically alloyed V-Y alloys were fabricated and their strength improvement from V-4Cr-4Ti below 800°C was confirmed. 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.\textsuperscript{26}) 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\textsubscript{2}O\textsubscript{3} and Y\textsubscript{2}O\textsubscript{3}, which were stable to 1073 K in liquid lithium, were identified.\textsuperscript{27}) Feasibility of the coating with Er\textsubscript{2}O\textsubscript{3} and Y\textsubscript{2}O\textsubscript{3} on V-4Cr-4Ti was demonstrated by Arc Source Plasma Deposition\textsuperscript{28}) and RF sputtering.\textsuperscript{29)} In addition to the physical deposition, in-situ coating with Er\textsubscript{2}O\textsubscript{3} on V-4Cr-4Ti is being developed.\textsuperscript{30}) As shown in Fig. 12, Er\textsubscript{2}O\textsubscript{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\textsubscript{2}O\textsubscript{3} layer grows and saturates and was stable at 600°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 14 MeV 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 studies 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.

4.3 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.31,32) In these decades, R&D efforts on ceramic composites have been very extensive, especially in the fields of aero-space and energy. Among them, C/C and SiC/SiC R&Ds have been very much emphasized in nuclear energy research.33,34)

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 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 of current emphasis in the Japanese program.34)

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 activities35) and Japan USA collaborative program JUPITER-II (2001–2006) for fusion research.36) Advanced SiC/SiC fabrication and evaluation is extensively done under JUPITER-II collaboration, mainly with ORNL, Kyoto University and Tohoku University.37,38)

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 (NITE) 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.39) One of the biggest advantages of the NITE process is its flexibility in shape and almost no limitation on size. Figure 13 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. 13), where no cracks or cavities were detected by naked eyes. The upper right of Fig. 13 is the real size model of 100 KW gas turbine combustor liner. Lower left of the figure is 2 mm 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. These micro structural features provide excellent thermal stress figure of merit and high...
The thermo-mechanical property of SiC/SiC is shown in Fig. 14. The figure represents potential of thermal stress tolerance, $M$, which is defined as the thermal stress figure of merit, $M = \frac{\sigma_{\text{UTS}} K_m (1-\nu)}{\alpha_m E}$, where $\sigma_{\text{UTS}}$ is the ultimate tensile strength, $K_m$ is the thermal conductivity, $\nu$ is Poisson’s ratio, and $E$ is Young’s modulus. Higher $M$ value indicates the excellence in thermal stress tolerance and the well-known high temperature material, Inconel 600, has the worst in this figure in all temperature range. The conventional and commercially available CVI-SiC/SiC is slightly better than Inconel 600. 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 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. Although a concern on this feature is the degradation anticipated under irradiation for the case of SiC/SiC, the excellence remains even under the most conservative estimation. 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 15 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 $10^{-12} \text{m}^2/\text{s}$, which is near the level of ordinary metallic materials. Laboratory products of NITE SiC/SiC present $10^{-11}$ to $10^{-9} \text{m}^2/\text{s}$ and the first pilot production of the NITE SiC/SiC was not as good as laboratory products, still the level of $10^{-8} \text{m}^2/\text{s}$ 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. As the backs-up option, W coating on the surface of SiC/SiC is also on-going. Shear strength of SiC/SiC joints at room temperature has been reported and 31.6 MPa and 15.1 MPa are the typical results, where the shear stress obtained from SiC/SiC joint made by NITE method shows larger than 52 MPa even with about 60% joint efficiency. 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. The recent results from HFIR 12J Rabbit irradiation for 4.2 dpa at 1000°C provide excellent stabilities in microstructure and mechanical property (no degradation in fracture strength). Also dual-ion irradiation experiments suggest no detectable effect of He simultaneous irradiation on swelling between 800 and 1600°C up to 100 dpa under 15, 60 He/dpa condition.

### 4.4 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 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. This can be attributed to the fine grains and the finely dispersed nano-particles. With the
This indicates the need of fabricating consolidated bodies prominent with decreasing grain size of the HIPed compacts. The beneficial effect of plastic working on ductility became improved process W-0.3%TiC-(0.7~1.7)%Mo alloys having fine grains of 0.6~2.0 μm 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. 16. 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.\(^{49}\)

5. Conclusion

Japanese activities on fusion structural materials R&D in these two decades have been providing many progresses in two categories of structural materials, such as reduced activation ferritic/martensitic 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 ITER 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 beyond.

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