Status of R&D Activities on Materials for Fusion Power Reactors

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Abstract. Current R&D activities on materials for fusion power reactors are mainly focused on plasma facing, structural and tritium breeding materials for plasma facing (first wall, divertor) and breeding blanket components. Most of these activities are being performed in Europe, Japan, P.R. China, Russia and the USA. They relate to development of new high temperature, radiation resistant materials, development of coatings that shall act as erosion, corrosion, permeation and/or electrical/MHD barriers, characterization of candidate materials in terms of mechanical and physical properties, assessment of irradiation effects, compatibility experiments, development of reliable joints, and development and/or validation of design rules. Main candidate materials for fusion power reactors are presented, with emphasis on recent achievements and key issues relative to structural materials and perspectives in the field of qualification of fusion materials.

1. Introduction

In fusion reactors, the plasma facing (first wall, divertor) and breeding-blanket components will be exposed to plasma particles and electromagnetic radiation and will suffer from irradiation by an intense flux of 14 MeV neutrons. The high-energy fusion neutrons shall produce atomic displacement cascades and transmutation nuclear reactions within the irradiated materials. From the point of view of materials science, atomic displacement cascades induce the formation of point defects (i.e., vacancies, interstitial atoms, vacancy and interstitial clusters) and segregation of alloying element, while transmutation nuclear reactions produce helium and/or hydrogen gas atoms. Main defect production in steels, at the level of the first wall in a 3-4 GW fusion power reactor, is reported in Table I.

TABLE I: DEFECT PRODUCTION IN STEELS FOR VARIOUS IRRADIATION FACILITIES.

Defect production (in steels)	Fusion neutrons (3-4 GW reactor, first wall conditions)	Fission neutrons (BOR 60 reactor)	High energy protons (590 MeV proton accelerator)	IFMIF (high flux test module)
Damage rate [dpa/year]	20-30	~ 20	~ 10	20-55
Helium [appm/dpa]	10-15	≤ 1	~ 130	10-12
Hydrogen [appm/dpa]	40-50	≤ 10	~ 800	40-50

The final microstructure of the irradiated material results from a balance between environmental conditions, especially radiation damage and temperature, and stress/strain histories. The microstructure evolution in a fusion reactor environment may engender degradation of the physical properties, such as a decrease of the thermal and electrical conductivity, as well as degradation of the mechanical properties, leading to strong hardening and/or embrittlement effects. Gas formation may engender macroscopic swelling of the material, leading to a loss of dimensional stability. These effects are the main factors limiting the choice of candidate materials for fusion power reactors [1,2]. In addition to a good resistance to radiation damage, the materials must show high performance, high thermal stress capacity, compatibility with coolant and other materials, long lifetime, high reliability, adequate resources, easy fabrication with a reasonable cost, and good safety and environmental behavior. The residual radioactivity of a large amount of exposed material is an important concern and will govern the handling methods, dictate the storage periods and the overall waste management and recycling scenarios. The materials R&D strategy that takes into account these limitations has led to the development of the so-called 'low activation' or 'reduced activation' materials.

Current R&D activities on materials for fusion power reactors are focused on plasma facing, functional and structural materials. While in the past substantial R&D work on materials for fusion power reactors has been done already in Europe, Japan, Russia and the USA, P.R. China shares for a short time the development efforts. Current activities relate to the development of new high temperature, radiation resistant materials, development of coatings that shall act as erosion, corrosion, permeation and/or electrical/MHD barriers, characterization of candidate materials in terms of mechanical and physical properties, assessment of irradiation effects, compatibility experiments, development of reliable joints, and development and/or validation of design rules. A number of various designs for the first wall/breeding blanket and divertor components are being investigated in the different countries, which means that various combinations of plasma facing/functional/structural materials are presently envisaged.

This paper is aimed at presenting the main candidate materials for fusion power reactors, i.e., DEMO and beyond, with emphasis on recent achievements and key issues relative to structural materials and perspectives in the field of qualification of fusion materials.

2. Candidate Plasma Facing, Functional and Structural Materials

In a fusion power reactor, the components that will be exposed to the most severe environmental conditions are the plasma facing components (first wall, divertor) and the breeding blanket components. The first wall will consist of a structural material attached to a plasma facing (or armor) material. The breeding blanket will consist of a neutron multiplier, a tritium breeding material, one or several coolants, and a structural material to separate and contain the different materials. Both neutron multiplier and tritium breeding materials are called functional materials. The divertor will consist of a structural material (heat sink) containing a coolant and supporting a plasma facing (or armor) material. Then, three types of materials are of primary concern: the plasma facing materials, which will serve as an armor for the underlying materials, the functional materials, which will have one or several particular functions (e.g., tritium breeding, neutron multiplication, optical transmission, etc.), and the structural materials, which will support the basic structure of the reactor.

2.1. Plasma Facing Materials

The qualification of plasma facing materials is very demanding. Plasma facing materials will be directly exposed to the fusion plasma, which means that they will submitted to a high heat flux of energetic particles (0.1-20 MW/m^2), high temperatures (775-3475 K), electromagnetic radiation, sputtering erosion, blistering and exfoliation, and high levels of neutron-irradiation (3-30 dpa/year), and to off-normal events like plasma disruptions and Edge Localized Mode (ELM) events. Key issues relate to hydrogen trapping, erosion and high heat loads.

Selection of plasma facing materials is mainly limited by their capability for absorbing heat and minimizing plasma contamination. Main candidate plasma facing materials for fusion power reactors are reported in Table II. Refractory metals and alloys, like tungsten-base alloys, have the advantage of a high melting point (T_m). Tungsten has the highest melting point, T_m = 3695 K, and the lowest vapor pressure, $P_v = 1.3x10^{-7}$ P_a at T_m, of all alloyed and unalloyed metals. In addition, tungsten has a high threshold for physical sputtering energy ($E_{th} \cong$ 200 eV for deuterium), i.e., it is a 'low erosion' material, and exhibits a very good heat load capability characterized by a high thermal conductivity and a good thermal shock capacity [3]. Tungsten-base alloys are considered as the main candidate plasma facing materials for the divertor, while oxide dispersion strengthened (ODS) steels coated with tungsten as well as tungsten-base alloys are candidate plasma facing materials for the first wall.

An alternative to the use of high-melting-point materials would consist in using flowing liquid metal as plasma facing material [4]. Flowing liquids have high heat load capability (up to about 50 MW/m^2) and could allow simultaneous heat and particle removal. Key issues are magneto-hydro-dynamic (MHD) effects on flowing conducting liquids (much modeling is needed), materials compatibility (corrosion), practical methods, and particle pumping capability (mostly helium). Main liquids being considered are lithium, gallium and tin.

2.2. Functional Materials

The qualification of functional materials is also very demanding. Their mechanical resistance under irradiation is presently considered of no primary concern. However, material properties, like the tritium release behavior, the thermal conductivity or the entire structural integrity after prolonged neutron irradiation, are important concerns. As compared to structural materials orders of magnitude more hydrogen and helium isotopes will be generated in functional materials, e.g., in beryllium-type neutron multipliers and lithium ceramic-type tritium breeders. The irradiation resistance of other functional materials, such as ceramic insulators, dielectric and optical windows, optical fibers or complete sensor assemblies, is also an important concern.

The lack of adequate functional materials meeting a very high temperature design window is an important issue for fusion power reactors. Component lifetime will be determined by the resistance of functional materials as well as by the resistance of plasma facing and structural materials. Unfortunately, selection of functional materials is very limited, as it relies mainly upon the properties required by the envisaged function. Main candidate tritium breeding and neutron multiplier materials for fusion power reactors are reported in Table II.

2.3. Structural Materials

The qualification of structural materials is fundamental. They will be submitted to high temperatures and high levels of neutron irradiation, but also to high mechanical and thermo-mechanical stresses. In addition, the thermal efficiency of a reactor, according to the Carnot cycle, is proportional to: (i) the temperature of the coolant at the exit of the reactor, and (ii) the difference between the temperature of the coolant at the exit of the reactor and the temperature of the coolant at the entrance of the reactor. These temperatures are mainly limited by the temperature window for use, i.e., the window of operation of the structural materials, which is mainly limited by their mechanical resistance under irradiation [e.g., 5].

Candidate structural materials, as well as candidate plasma facing materials, have a chemical composition that is based on low activation chemical elements (Fe, Cr, V, Ti, W, Si, C) [6]. They are listed in Table II and include mainly reduced activation ferritic/martensitic (RAFM) steels, ODS RAFM and RAF steels, tungsten-base alloys, vanadium-base alloys and fiber reinforced SiC/SiC_f ceramic composites [7,8,9]. Each alternative alloy class exhibits specific problems arising from their intrinsic properties and from their resistance to radiation damage. For the time being, the most promising structural materials seem to be the RAFM steels for which the greatest technology maturity has been achieved, i.e., qualified fabrication routes, welding technology and a general industrial experience are available [10,11,12,13, 14,15,16,17]. Their temperature window of use is presently about 625-825 K, the lower temperature being limited by irradiation-induced embrittlement effects and the upper temperature by a strong drop in mechanical strength. However, a difference of 200 K between the temperature of the coolant at the exit of the reactor and that at the entrance of the reactor should be sufficient to ensure acceptable efficiency of first generation fusion power reactors.

TABLE II: MAIN CANDIDATE MATERIALS FOR PLASMA FACING AND BREEDING BLANKET COMPONENTS.

Function	First wall	Breeding blanket	Divertor
Plasma facing material	W-base alloy, W-coated ODS steel, flowing liquid metal: Li	-	W-base alloy, W-coated SiC/SiC_f, flowing liquid metal: Li, Ga, Sn, Sn-Li
Neutron multiplier material	-	Be, $Be_{12}Ti$, $Be_{12}V$, Pb	-
Tritium breeding material	-	Li, eutectic Pb-Li, Li-base ceramic material (e.g. Li_2O, Li_4SiO_4, Li_4SiO_4, + 2.5 wt.%SiO_2, Li_2TiO_3, Li_2ZrO_3, $LiAlO_2$)	-
Structural material	RAFM steel, ODS steel, V-base alloy, SiC/SiC_f	RAFM steel, ODS steel, V-base alloy, SiC/SiC_f	ODS steel, W-base alloy
Coolant	-	Water, helium, eutectic Pb-Li, Li	Water, helium

3. Status of R&D Activities on Steels and their ODS Variants

3.1. Reduced Activation Ferritic/Martensitic (RAFM) Steels

The development of ferritic/martensitic steels for applications in fusion devices emanates from the limitations of the austenitic stainless steels and the promising high dose experience with ferritic/martensitic steel fuel canning in liquid metal cooled fast reactors. The austenitic steels suffer from helium embrittlement at elevated temperatures and swell to a degree not acceptable for fusion reactor components. The ferritic/martensitic steels exhibit superior performance to austenitic steels in both swelling and helium embrittlement resistance. At a temperature of about 575 K the swelling rate of ferritic/martensitic steels is about 1 vol.% after 100 dpa, while it is about 1 vol.% after 10 dpa for typical austenitic steels. Ferritic/martensitic steels also exhibit a better surface heat capability than austenitic steels (5.4 kW/m at 675 K, i.e., about three times that of austenitic steels) [5], favorable cost, availability and service experience, and their good compatibility with aqueous, gaseous, and liquid metal coolants permits a range of design options. At the same time ferritic/martensitic steels with alloying elements such as chromium, some tungsten, vanadium and tantalum activate little compared to austenitic steels with nickel and molybdenum. In Europe, Japan and Russia it has been demonstrated that it is now feasible to produce RAFM steels on an industrial scale with sufficiently low impurity levels [14,18,19,20]. The promise remains to produce low activation varieties that allow re-cycling within a century [21,22]. Main RAFM steels that are being investigated include the Chinese Low Activation Martensitic (CLAM) steel, the European EUROFER 97 alloy, the Japanese F82H and JLF-1 alloys, and the Russian RUSFER-EK-181 alloy. Their composition lies in the following range: Fe-(7.5-12)Cr-(1.1-2)W-(0.15-0.25)V, in weight percents.

Major issues relate to the following areas:

- Limited strength at high temperatures. RAFM steels exhibit a drop in tensile strength at about 825 K and a strong reduction in creep strength at $T \geq 875$ K. In addition, softening occurs during cyclic loading, which may lead to maximum allowable loads much smaller than the limits predicted by the current design rules.
- Irradiation-induced embrittlement effects at low and intermediate temperatures. Neutron irradiation at temperatures below about 675 K leads to strong hardening and embrittlement effects including loss of ductility, increase in the ductile-to-brittle transition temperature (DBTT) and loss of fracture toughness. The radiation damage in RAFM steels is well documented up to 15 dpa for all kinds of wrought products and welded joints [23,24,25,26,27]. At present results of fast reactor neutron irradiations at 600 K up to 80 dpa are being analyzed [28,29]. As observed at smaller irradiation doses, the rates of hardening and loss of ductility as well as the rate of DBTT increase tend to

decrease with increasing dose, but eventual saturation phenomena still have to be confirmed by more systematic irradiation experiments to high doses.

- The production of high amounts of helium and hydrogen by transmutation nuclear reactions. Possible gas atom effects are still a matter of controversy. It was shown that helium concentrations above about 500 appm have significant detrimental effects on the fast fracture properties of RAFM steels at temperatures below about 675 K [30,31]. A large increase in radiation hardening was measured following dual-beam irradiations with Fe and He ions up to 60 dpa at temperatures between 695 K and 795 K, with respect to single-beam irradiations with Fe ions only, which seems to indicate that helium has detrimental effects also on radiation hardening and not only on fracture properties [32]. The effect of hydrogen isotopes on the mechanical properties of RAFM steels is another primary concern. Results on hydrogen gas trapping and the controlling diffusive transport parameters have been recently reported [33]. The data provide indications about transport mechanisms. Similar measurements need to be performed on irradiated specimens.
- Joining techniques, in particular welding, as RAFM steels are more difficult to weld than austenitic stainless steels. Considerable effort has been put in the demonstration of the feasibility and reliability of the different techniques [14,34]: tungsten inert gas (TIG) welding, electron beam (EB) welding, laser beam welding (LBW), and diffusion welding [35,36]. Important issues are the toughness and radiation resistance following the welding and subsequent heat treatments of the welds. The design complexities, for example many welds in one component mean many post-weld heat treatments to improve joints performance, sometimes compromise the optimum weld preparation conditions [36].
- Limited database at high irradiation doses. The database for RAFM steels is sufficiently large to support Test Blanket Module (TBM) design. Design data for the TBMs include plate, bar tube and weld properties with effects of aging and neutron irradiation to a limited level of about 3-5 dpa [37]. That is enough to predict the TBM behavior, but far too low for power plant conditions aiming at 75 dpa and higher.

3.2. Oxide Dispersion Strengthened (ODS) RAFM Steels

As the upper temperature for use of RAFM steels is presently limited by a drop in mechanical strength at about 825 K, Europe, Japan and the USA are actually researching RAFM steels with high strength at higher operating temperatures, mainly using stable oxide dispersion [38]. The numerous interfaces between matrix and oxide particles are also expected to act as sinks for the irradiation-induced defects. Present R&D activities are aimed at finding a compromise between good tensile and creep strength and sufficient ductility, especially in terms of fracture toughness.

An example of ODS RAFM steel is the European ODS-EUROFER material, which should be used at temperatures up to 925 K in the European long-term dual coolant breeding blanket concept as mm-thick layer plated onto the first wall made of the EUROFER RAFM steel. The ODS-EUROFER is made of EUROFER 97 reinforced with 0.3 wt.% Y_2O_3 (yttria) particles, and it is manufactured by mechanical alloying followed by Hot Isostatic Pressing (HIPping). This material shows an increase in yield strength and ultimate tensile strength by about 50% with respect to the EUROFER 97 alone, a reasonable total elongation, and superior creep strength with respect to the EUROFER 97 [39,40,14,41]. However, ODS RAFM steels do not reach the very low DBTT and high upper shelf energy of RAFM steels. The DBTT of ODS-EUROFER lies presently in the range 195-235 K [14]. ODS RAFM steels exhibit a good resistance to oxidation, much better than that of martensitic steels and similar to that of austenitic stainless steels [26], as well as a good resistance to water corrosion [42].

Major issues relate to the following areas:

- Processing, as manufacturing by powder metallurgy (including mechanical alloying and HIPping) is time-consuming and expensive, and it usually yields to coarse oxide particles with non-uniform size and spatial distribution, as well as to residual porosity. As an alternative to manufacturing of ODS steels by powder metallurgy, strengthening of martensitic steels by more conventional processing techniques (i.e., melting, casting, hot working, cold working, etc.) is being investigated [43,44].

- Full characterization of the pre-irradiation behavior.
- Stability of the Y_2O_3 particles under aging and irradiation.
- Full characterization of the irradiation effects. Neutron irradiation at intermediate temperatures to a few dpa revealed radiation hardening but no significant loss of ductility.
- Development of joining techniques, as the inhomogeneous character of ODS steels makes difficult the manufacturing of sound fusion welds. The feasibility of joining ODS-EUROFER and EUROFER by diffusion welding was recently demonstrated [14]. New technologies, such as friction stir welding, might solve the limitations of fusion welds [45].

3.3. Oxide Dispersion Strengthened (ODS) RAF Steels

The potential of 12-14% chromium steels for increasing operating temperatures up to 1025 K is presently investigated in Europe [46,38,47], Japan [48,49] and the USA [50]. Most of the current R&D activities focus on the characterization of commercial ODS ferritic steels, like the MA957 and PM2000 alloys, in the unirradiated and irradiated states, and on the development of ODS reduced activation ferritic (RAF) steels with a chemical composition in the range Fe-(12-14)Cr-(0.3-0.4)Ti-(1.2-3)W-(0.25-0.3)Y_2O_3 (in weight percents), which are manufactured by mechanical alloying followed either by HIPping or hot extrusion. Such ODS steels show usually excellent high temperature tensile and creep strength, due to the presence of a high density of nano-clusters enriched with Y, Ti and O [51], as well as very favorable oxidation and corrosion resistance [52,53]. Key issues are similar to the ones pointed out for ODS RAFM steels.

4. Status of R&D Activities on Refractory Materials

Refractory materials include group V-B metals (V, Nb, Ta) and group VI-B metals and (Cr, Mo, W). Both groups have bcc structures and are prone to low temperature brittle fracture. In the field of fusion current R&D activities focus on tungsten-base alloys and vanadium-base alloys, due to their reduced activation under neutron irradiation.

4.1. Tungsten-Base Materials

Pure tungsten exhibits high strength at high temperatures and good surface heat capability (11.3 kW/m at 1275 K [5]). It also shows a good resistance to erosion and does not suffer from high activation. Main alloys of interest include WL10 (W-1%La_2O_3), WVM (potassium doped tungsten) and W-(0.3-0.7) wt.% TiC alloys.

Major issues relate to the following areas:
- The inherent low fracture toughness at all temperatures associated with a high DBTT, equal to about 625 K, 1075 K and 1225 K in the case of unirradiated pure tungsten, commercial tungsten and WL10, respectively [54]. Processing of nano-structured materials either by electro-deposition or by severe plastic deformation (SPD) (e.g., high-speed hot extrusion, high-pressure torsion) may help to improve the ductility [55,56]. In pure tungsten such a SPD microstructure is not thermally stable. Alloying can help to overcome this problem.
- High fabrication costs, due to brittleness and high Z of tungsten.
- The low ductility (deformability) and the sensitivity to production history, i.e., to the microstructure (grain size and shape, anisotropy, texture, grain boundary character and impurities), alloying elements, exposure to temperature (re-crystallisation), loading direction, irradiation dose and temperature, for which only a few systematic studies exist.
- Creep rate and strength between 975 K and 1575 K.
- Strong irradiation-induced embrittlement effects at $T < 975$ K. The irradiation-induced increase in DBTT is not really known and could yield a DBTT value well above 875 K.
- The production by transmutation nuclear reactions of considerable amounts of osmium and rhenium under neutron irradiation, which yields the formation of brittle sigma phase.
- Limited knowledge about irradiation effects in general [5]. Series of SPD tungsten-base materials will be irradiated in 2006/2007 in the OSIRIS reactor (France) at 875 K and

1275 K to about 5 dpa (in Fe). This irradiation campaign is referred to as the FURIOSO experiment.
- Joining of tungsten-base alloys to (ev. ODS) RAFM steels, which is a concern in the case of the European helium cooled divertor concepts. The most promising joining technique is high-temperature brazing at 1395-1455 K with nickel-base amorphous foils and brazing pastes as filler material [57,58].

4.2. Vanadium-Base Materials

Candidate vanadium-base alloys for fusion applications include mainly the V-4Cr-4Ti and V-5Cr-5Ti alloys. Vanadium-base alloys are very attractive due to their low activation and good high temperature strength and surface heat capability (6.4 kW/m at 675-875 K [5]). They also exhibit a low DBTT in the unrradiated state (about 80 K for V-5Cr-5Ti), constant strength and good ductility in the range 475-1075 K, better creep resistance than the RAFM steels and austenitic stainless steels (up to 875 K for short durations), good fatigue resistance at 300 K and intermediate temperatures, negligible irradiation-induced embrittlement at T > 675 K, good swelling resistance, and good compatibility with purified helium and liquid metals [e.g., 59,60,61,62]. Their temperature window for use is presently about 675-975 K [63].

Major issues relate to the following areas:
- Strong affinity for solutes such as oxygen, carbon and nitrogen, which leads to matrix embrittlement and reduced compatibility with liquid Li. Concentrations of interstitial impurities can be controlled by small additions of silicon, aluminium and/or yttrium. In particular, yttrium addition is effective in reducing the oxygen level. High purity, large-scale V-4Cr-4Ti-Y ingots have been successfully produced by levitation melting [64].
- High solubility, diffusivity and permeability of tritium, which can lead to embrittlement at low temperatures.
- Severe thermal creep at high temperatures.
- MHD effects in relationship with the use of liquid Li. Candidate coating materials include AlN, CaO, $CaZrO_3$, Y_2O_3 and Er_2O_3 ceramics, as well as multilayer coatings [65].
- Strong irradiation-induced embrittlement effects at T < 675 K.
- Effects of irradiation-induced formation of precipitates and high helium content.
- Welding, which must be done in an inert atmosphere.
- Lack of industrial maturity.

5. Status of R&D Activities on SiC/SiC$_f$ Ceramic Composites

SiC/SiC$_f$ ceramic composites have attractive properties for functional and structural applications. SiC/SiC$_f$ ceramic composites exhibit several advantages: (1) the ability to operate with acceptable mechanical properties at temperatures much higher than steels (up to 1275/1375 K), with little reduction in strength up to 1275 K, which offers potential increase in fusion reactor efficiency, (2) an inherent low level of long-lived radioisotopes, which reduces the radiological burden of the structure, (3) perceived tolerance against neutron irradiation up to very high temperatures, as bulk SiC exhibits moderate swelling in the range 425-1175 K [66]. The chemical composition, density, elastic constants, thermal conductivity and neutron irradiation resistance are different for SiC fibers and bulk SiC. With respect to SiC/SiC$_f$ ceramic composites, the brittle characteristics of bulk SiC, such as low fracture strain and toughness, still impose severe limitations to its practical application. There exist many different SiC/SiC$_f$ materials. They are at the early stages of development and are constantly evolving. Until recently, SiC/SiC$_f$ ceramic composites exhibited significant degradation in mechanical properties upon irradiation due to non-SiC impurities causing easy interfacial debonding. Improved performance was obtained as a result from development of stoichiometric, crystalline SiC fibers and advanced fiber/matrix interfaces such as multilayered interfaces. Chemical vapor infiltration (CVI) CVI of stoichiometric, crystalline fibers is the leading processing method of high purity SiC/SiC$_f$ composites among several currently available ones. Whereas the recent progress was driven by the availability of almost stoichiometric fibers with higher thermal conductivity and higher thermal stability, the next step will be to tailor the properties of the composite to the specific application by choosing the appropriate fiber architecture, fiber to matrix interface and densification processes.

Major issues relate to the following areas [12,67]:

- Low surface heat capability (2 kW/m at 1075 K [5]), which is dependent on processing conditions, type of fibers and fiber architecture.
- Strong irradiation-induced reduction in thermal conductivity.
- High helium and hydrogen production rates under irradiation.
- Large swelling of bulk SiC at temperatures below about 425 K, associated with a crystalline to amorphous phase transition.
- Limited knowledge about irradiation effects in general. SiC/SiC_f materials from Europe, Japan and the USA will be irradiated in 2006/2007 in the OSIRIS reactor to about 5 dpa (in Fe) at 875-925 K and 1275 K (FURIOSO experiment). This irradiation campaign will be performed within the framework of the International Energy Agency (IEA) Agreement on Fusion Materials. This campaign will be followed by two other irradiation experiments, referred to as the MATRIX and FUTURIX experiments, which are aimed at irradiating in the Phenix reactor (France) SiC/SiC_f materials at about 775 K and 1275 K, to about 36 dpa and 72 dpa (in Fe), respectively.
- Leakage of helium gas coolant into the fusion plasma, as most of the available SiC/SiC_f ceramic composites are still porous and anticipated to be vulnerable to widespread microcracking. Appropriate tritium barriers have to be developed.
- Corrosion resistance under flowing liquid Pb-Li at high temperatures (up to 1375 K).
- Fabrication and joining of large sub-components, and structural design methodology.

6. Waste Management Issues

In the field of waste management, the general aim is to avoid geological repository. In what concerns recycling of low activated and contaminated metals, it seems that current existing routes for fission plants could be used, provided the tritium issue is resolved, i.e., industrial detritiation processes become available. In what concerns recycling of highly activated and toxic (e.g., beryllium) materials, several challenges have to be overcome. Recycling techniques have to be developed, which allow separation of the various materials (toxic from untoxic ones, for instance), detritiation processes have to become available for tritium-contaminated water, structural and concrete materials, and recycling plants have to be constructed as the available capacity is too low [68,69]. Hands-on recycling of RAFM steels does not seem to be viable, even after 100 years [70]. However, hands-on recycling should be possible in the case of V-Cr-Ti alloys [71].

7. On Fusion Relevant Irradiation Conditions

As worldwide no intense source of 14 MeV neutrons is presently available, it is necessary to simulate irradiation by 14 MeV neutrons. Up to now, the material fusion research community has made extensive use of fission reactors and particle accelerators to investigate the effects of irradiation on materials. Despite all the achieved successes [9], the limits of employing these types of irradiation have been identified. First, the results and conclusions obtained from a given fission neutron spectrum can be transferred only up to some extent to that of a fusion reactor. Second, there will be always an intrinsic issue of reliability as long as no systematic experimental validation has been performed to prove the validity of the data transfer. Two approaches are being investigated to account for actual fusion relevant irradiation conditions, in terms of temperature, accumulated damage, damage rate, and rates of production of impurities: (1) modeling of radiation damage and radiation effects, (2) construction of IFMIF, an intense source of 14 MeV neutrons.

7.1. Modeling Activities

Main modeling activities are aimed at understanding and predicting radiation effects on Fe, Fe-C, Fe-Cr, Fe-Cr-C, Fe-He, etc., towards RAFM steels, as well as on some other material systems (e.g., SiC/SiC_f), under fusion-reactor-relevant conditions. One of the key elements for achieving such activities is the development of theoretical and numerical predictive tools and validation of these tools against dedicated experiments at the relevant scale. The modeling tools and the validation data are used to correlate experimental data from different irradiation sources, to optimize the IFMIF testing program, and to extrapolate with increased confidence the results to a broad range of irradiation conditions typical of fusion power reactors. The knowledge gained and the tools developed also help in planning R&D activities for other candidate materials such as ODS steels and tungsten-base alloys.

7.2. The International Fusion Materials Irradiation Facility (IFMIF)

There is a fundamental need to build an intense high energy neutron source to serve as a tool (1) for calibrating and validating the data generated using fission reactors and particle accelerators, and (2) for qualifying materials up to about full lifetime of anticipated use in a DEMO-type reactor. The primary mission of IFMIF is to generate a materials database to be used for the design and construction of various components, and for the licensing and the assessment of the safe operation of a DEMO-type reactor.

The irradiation conditions provided by IFMIF will be very close to the ones expected to occur in DEMO-type reactor, at the level of the first wall, in terms of damage rate and rates of production of helium and hydrogen (see Table I). Due to engineering and economic considerations the volume of the high flux test module in IFMIF will be only half a liter in the high flux position (20-50 dpa/fpy). However, by using a set of six different types of extensively qualified miniaturized specimens (tensile, Charpy, fatigue, creep, crack growth and fracture specimens), this volume should be sufficient to achieve within five years for a few materials a 80 dpa database needed for DEMO-predesign and for a variety of materials a 150 dpa database within about 20 years.

On the path to fusion power reactors it is usually considered that the construction of IFMIF is of the same importance as the construction of ITER [72]. According to the European optimized scenario, the Engineering Validation and Engineering Design Activities (EVEDA) phase of IFMIF shall start in 2007 and extend until 2012. Then, construction of IFMIF should take place until 2018, which means that IFMIF should become operational approximately at the same time as ITER.

8. Concluding Remarks

Materials constitute a key issue on the path to fusion power reactors. There is a strong need for materials which (1) are *resistant to irradiation* by a typical fusion neutron spectrum, (2) can *operate at the highest temperatures as possible*, to allow for a good plant thermal efficiency, and (3) are *as low activation as possible*, to make easier public acceptance of fusion as a future energy source.

Material choices will not solve all design problems. The design is complex by nature and further innovations will have to be used to overcome material limitations. Fabrication and processing are major keys to increase the degrees of freedom for design, and the present selection of materials may have to be revised. Close discussions between designers of the various first/wall breeding blanket and divertor concepts and material scientists are required on the way to the construction of attractive fusion power reactors.

9. Acknowledgment

This work, supported by the European Communities, was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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