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Structural Materials Assessment

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Structural Materials Assessment

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Abstract

The selection of first wall and structural materials is strongly dependent on the proposed design of breeding blanket components and the targets for a fusion reactor development. An envelope of parameters which have to be covered in future R&D activities and which have been adapted in different proposals has been compiled. A short description of interesting material groups like ferritic-martensitic steels, vanadium alloys and ceramic composites, major criteria for their selection and a survey on existing irradiation data is given. This is followed by a comparative assessment of relevant properties and an identification of major issues for each material group.

A more detailed proposal for the future R&D activities is then developed for the ferritic-martensitic steels, the present reference material for the European Breeding Blankets. It describes different phases of development necessary for the qualification of this material for DEMO and gives time schedules which are compatible with parallel component developments. A more selective strategy is proposed for the development of vanadium alloys and the ceramic composite material SiC/SiC. For these alternatives work should be concentrated on identified high-risk issues, before a comprehensive development programme is started.

The necessity of efficient irradiation facilities to study the irradiation behaviour of the materials under simulation – and realistic fusion conditions is discussed. The availability of high flux fission reactors and necessary extensions of irradiation rigs for the next decade is stressed. Finally it is shown that for the qualification of materials under realistic fusion conditions a high-energetic, high-flux neutron source is mandatory. An accelerator-driven d-Li neutron source (IFMIF) can fulfil essential users requirements as test bed for materials and can technically be made available in due time. In combination with ITER and DEMO, where a concept verification and full scale reliability tests of breeding blanket components can be performed appropriate and efficient tools would be available to develop materials and components towards a fusion reactor.

Bewertung des Potentials von Strukturmaterialien für die Kernfusion

Zusammenfassung

Die Auswahl von Strukturwerkstoffen für die sog. Erste Wand und die Brutblankets in Fusionsreaktoren hängt von dem speziellen Design und den Belastungsbedingungen in diesen Komponenten ab. Aus diesem Grunde wird zunächst eine Übersicht über die international genannten Ziele für die Entwicklung von Fusionsreaktoren und die für die Werkstoffauswahl wichtigen Belastungsparameter gegeben. Sie sind für die Planung einer langfristigen Entwicklungsstrategie von Werkstoffen von Bedeutung. Es schließt sich eine Beschreibung der wichtigsten Materialgruppen, ferritisch-martensitischen Stählen, Vanadiumlegierungen und faserverstärkten keramischen Composits vom Typ SiC/SiC an und enthält eine vergleichende Beurteilung wichtiger Eigenschaften, des Verhaltens unter Neutronenbestrahlung und eine Identifizierung möglicher Schwachstellen.

Für die Gruppe der ferritisch-martensitischen Stähle, die als Referenzmaterial für die Entwicklung von Brutblankets in Europa gelten, wird eine detaillierte F&E Strategie entwickelt, um in mehreren Phasen die Qualifizierung für DEMO-relevante Anwendung zu erreichen. Eine andere Strategie wird für die Vanadiumlegierungen und SiC/SiC-Verbundwerkstoffe vorgeschlagen. F+E- Arbeiten sollten sich hier zunächst auf sog. „High-Risk“ Themen konzentrieren, bevor breit angelegte Entwicklungsprogramme gestartet werden.

Das Vorhandensein oder die Beschaffung geeigneter Bestrahlungsquellen zum Testen der Werkstoffe unter realistischen Bedingungen ist für die Durchführung des Entwicklungsprogramms von entscheidender Bedeutung. Während für die nächste Dekade aus Mangel an einer fusionsspezifischen 14-MeV Neutronenquelle Spaltungsreaktoren mit hohem Neutronenfluß und geeigneten Bestrahlungseinrichtungen eingesetzt werden, um Betriebserfahrungen mit diesen Werkstoffen unter Neutronenbelastung zu sammeln, ist der zügige Bau einer Bestrahlungsquelle, die ein fusionsrelevantes, energiereiches 14 MeV-Neutronenspektrum liefert, für die Materialentwicklung und -qualifizierung absolut erforderlich. Eine beschleunigergetriebene d-Li Stripping Neutronenquelle (International Fusion Material Irradiation Facility, IFMIF) erfüllt die Anforderungen der Nutzer und könnte technisch auch in relativ kurzer Zeit verwirklicht werden. Das Testen von großen Komponenten, wie den Brutblankets unter realistischen Bestrahlungs-und Testbedingungen, bleibt jedoch zukünftigen Fusionsanlagen wie dem ITER (International Thermonuclear Reactor) oder einem Demonstrationskraftwerk (DEMO) vorbehalten.

Contents

1. Introduction.

Materials for the first wall (FW), high heat flux- and breeding blanket components belong to the most severely exposed parts of future fusion reactors and pose key problems for the successful implementation of fusion reactors as an efficient source of electric power. This has been stated at many occasions, including very prominent studies elaborated for the International Energy Agency by the Cottrell Blue Ribbon Panel and the Amelinckx Senior Advisory Committee [1, 2]. In accordance with the technical objectives of the International Thermonuclear Experimental Reactor ITER, the realisation of components like FW/breeding blankets and divertors will have priority and needs an extended R&D-programme in the next decades. Hereby the qualification of structural materials for a highly efficient and safe operation is mandatory. Their behaviour impacts the economic competitiveness and determines the environmental attractiveness of fusion.

In this assessment emphasis is given to the development of structural materials to be used as first wall and structural materials in blanket components, taking into account that their selection is strongly dependent on the proposed designs for breeding blankets and hence also on the proposed combinations of structural-, breeding/coolant- and neutron-multiplying materials. Following the introduction it will be tried in Chap. 2 to define realistic targets for the longterm R&D materials programme necessary to build a DEMOnstration reactor (DEMO) or a Commercial Fusion Power Reactor (CFPR) and to extract an envelope of parameters which have to be covered in future R&D activities, independent of specific designs and material combinations. In Chap. 3 a short description of the interesting material groups, major selection criteria and a survey on existing irradiation data and experience stemming from other programmes (mainly Fast Breeder and Materials Test Reactors) is provided. A more detailed R&D proposal is then developed in Chapter 4 for the present EU reference material – a ferritic-martensitic steel – by identifying the major objectives and development phases necessary for the qualification of this material for DEMO. Finally in Chapters 5 and 6 the necessity of efficient tools to simulate the radiation damage in structural materials with existing irradiation facilities and the strategy to develop an appropriate high-fluence and high-energetic neutron test bed for materials research is addressed.

2. Targets for the Materials Research and Development Activities.

The development of structural materials for fusion application follows at present two distinct lines: The first is directed towards the construction of the next step machine ITER. This facility is – with regard to materials issues - characterised by a moderate neutron wall loading and accumulated neutron fluence at the end of its lifetime, a low temperature regime and a strongly pulsed operational mode. It is expected that these moderate demands - even further reduced in recently revised ITER proposals - and compiled in Tab. 1, can be fulfilled by the use of an austenitic stainless steel of type 316 LN-IG which has been successfully applied in conventional fission reactors. The target for the necessary R&D activities is well defined and the international collaboration within the ITER community is arranged through the International Atomic Energy Organisation IAEO. The results are periodically reported at international fusion materials conferences [3,4,5].

Table 1: General performance goals for fusion devices

	ITER	DEMO	REACTOR
Fusion power	0,5 - 1 GW	2 - 4 GW	3-4 GW
Neutron wall loading (first wall)	0,5 - 1 MW/m^2	2 - 3 MW/m^2	2 – 3 MW/m^2
Integrated wall load (first wall) in MWy/m^2 in Displacements per atom*	0,3 – 1 MWy/m^2 3 - 10 dpa	3 - 8 MWy/m^2 30 - 80 dpa	10 - 15 MWy/m^2 100 – 150 dpa
Operational mode	Pulsed (300-1000s) $< 5 \cdot 10^4$ cycles	Quasicontinuous	
Plant lifetime			~ 30 FPy
Net plant efficiency			~ 30 %

* The following relations between neutron wall loading, neutron flux and displacements per atom have been used:
$1\ MW/m^2 \triangleq 3 \cdot 10^{14} n_{tot}/cm^2 \cdot s \triangleq 3 \cdot 10^{-7}$ dpa/s (Fe)
$1\ MWy/m^2 \triangleq 10$ dpa (Fe)
The calculation of dpa according to the Norgett-Robinson-Torrens (NRT) model

2.1 Performance goals for fusion reactors

The longterm development towards a DEMO- or a commercial Fusion Power Reactor aims for materials which can withstand high neutron wall loadings and fluxes under temperature and coolant pressure conditions necessary to drive efficient thermodynamic working cycles. Also the integrated neutron fluences should be high enough to limit the necessary replacement of plasma-near components to a minimum. This is necessary in order to be competitive with commercial conventional and nuclear power plants. Finally the materials should be of "low-activation"-type to maintain one of the most attractive features of fusion.

In general it is assumed that a DEMO will be the major step towards a prototype or a commercial fusion power reactor. This means that all reactor-relevant functions like the breeding of tritium or the successful operation of a divertor have to be demonstrated and successfully tested in DEMO. It is also evident that such components should be built from materials which possess the potential for high-performance so that they could be further qualified for commercial reactors. For the selection of appropriate candidate structural materials not only intrinsic material data like thermophysical or nuclear properties and strength including their possible degradation under extensive neutron irradiation are important, but also their „compatibility“ with other materials like breeding media, neutron multipliers etc.. In this context the „compatibility“ means not only corrosion phenomena but also general interactions between the different materials be they of mechanical, thermal, chemical or irradiation-induced nature. Such additional requirements for the selection and proper combination of materials can be summarised under a term called „material integration issues“ [6]. They are very dependent on the specific design of a component.

In Tab. 1 a range of performance goals presented at different occasions is compiled for DEMO- and commercial power reactors [6,7,8,9,10]. Key parameters for the material development are the expected neutron wall load, which determines the neutron flux and the surface heat load, relevant radiation damage parameters like displacement and transmutation rates for the generation of damaging elements like hydrogen, helium or solid elements and the volume power density. Equally important are the wall loading conditions integrated over the expected lifetime of components. For one of the European DEMO-breeding blanket modules [10,11], the so-called helium-cooled pebble bed (HCPB) blanket, in which a ferritic–martensitic steel with reduced longterm activation, a ceramic breeder material Li_4SiO_4 and the neutron multiplier beryllium have been chosen, detailed calculations of relevant damage parameters have been performed. They are based on detailed MCNP–Monte Carlo neutronic transport calculations and on thermal-mechanical analyses [12]. In Fig. 1 the neutron flux, the yearly displacement rate of atoms and the production of gaseous transmutation products H and He are given for Fe, in dependence of the radial distance to the first wall position. The corresponding max. neutron wall load at the outboard position is 3,5 MW/m² and the maximum displacement rate per annum lies in the range of about 30 dpa. Two observations are of importance: i) the strong decline of neutron flux and dpa along the radius by about nearly two orders of magnitude (30 dpa vs. 0.7 dpa) and ii) the variation of the He/dpa and H/dpa values in Fe (11.1 vs. 2.75 appm He/dpa and 45 vs. 10,3 appm H/dpa). This variation of the damage parameters is due to the reduction of neutron flux and a relative softening of the neutron spectrum. Such figures – though material-dependent - give the range of parameters which have to be covered in irradiation experiments in order to achieve fusion specific conditions in simulation experiments as will be discussed later in Chap. 5. For an envisaged lifetime of 20000hs or

2,28 full power years about 70 dpa, 780 appm He and 3150 appm H will be generated in the First Wall position as a typical target for a DEMO.

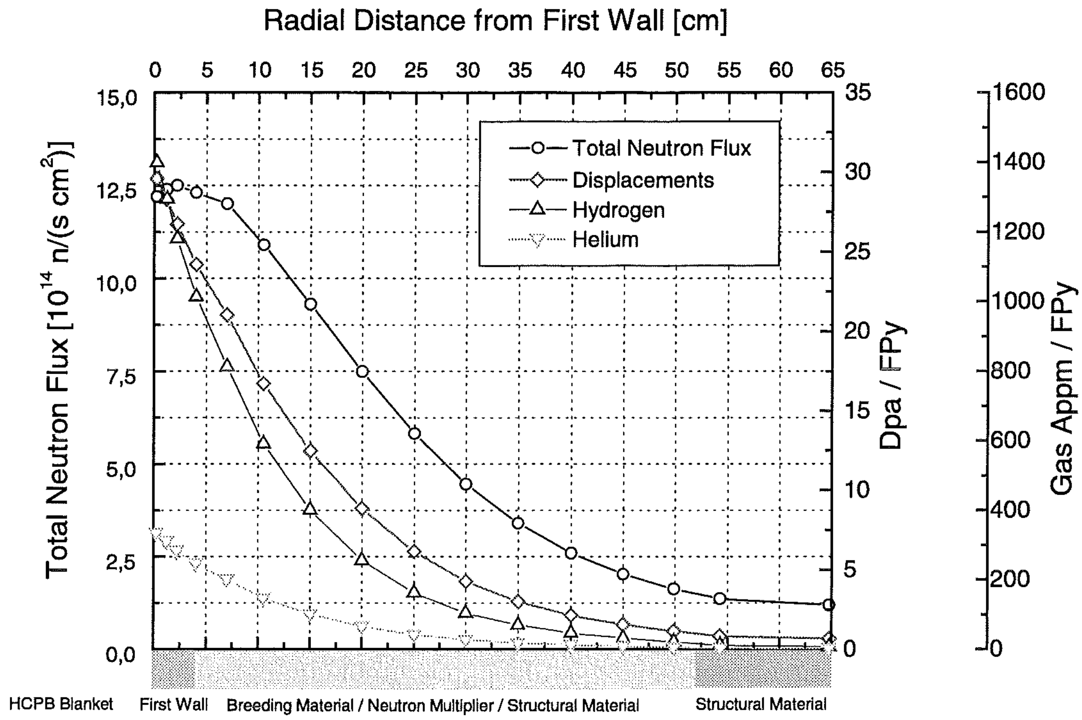

Fig. 1:

Dependence of neutron flux and damage parameters for Fe in the HCPB-outboard breeding blanket as function of radial distance from First wall

Besides the radiation damage parameters the volume power density in W/m³ plotted in Fig. 2, which is caused by the inelastic interaction of materials with neutrons is also important since it could set limits to the thermal or thermally-induced stress response in the materials. Interestingly, for this blanket configuration the allowable temperature limit is reached in the ceramic breeder material Li_4SiO_4 and not in the structural material [12]. This example shows that in order to determine max. allowable neutron wall loadings the thermophysical properties of all materials – not only the structural materials - have to be taken into account.

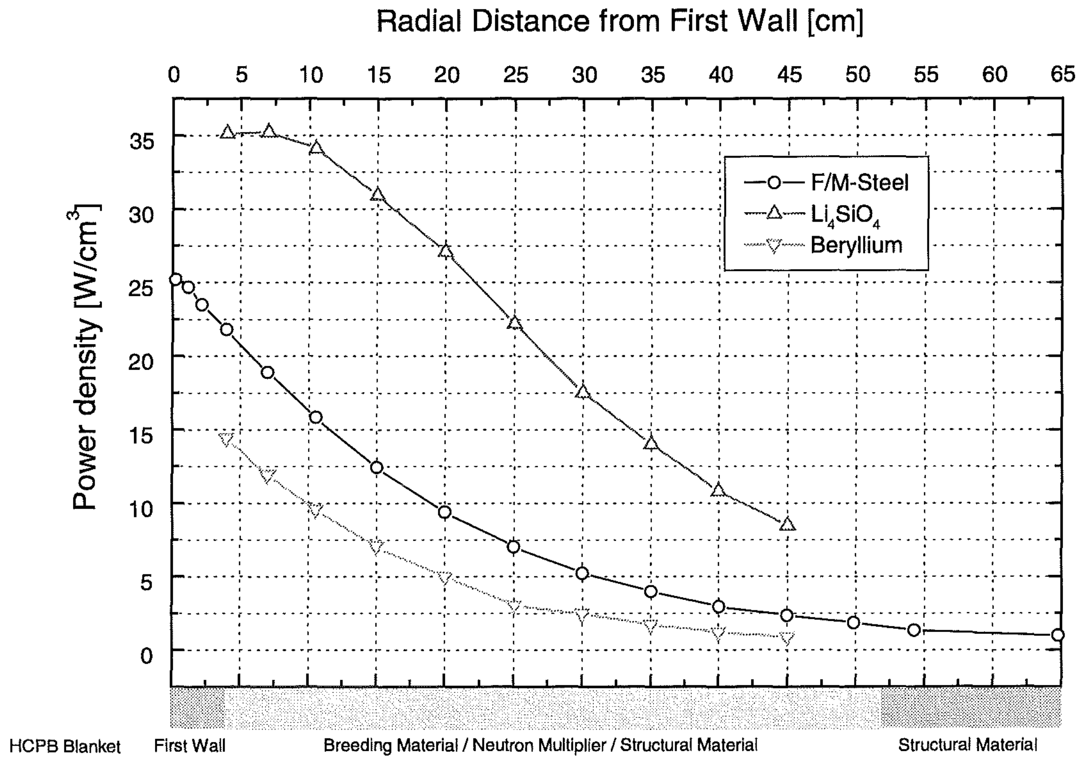

Fig. 2:

Radial dependence of power density in structural and ceramic breeder materials and neutron multiplier Be in the HCPB-outboard breeding blanket

An equally important parameter in Tab. 1 is the mode of reactor operation which is expected to be quasi-continuous or steady state in reactors. In the intermediate development phase (ITER, DEMO) one has to reckon with limited pulse lengths so that the mechanical loading will be fatigue-dominated and secondary stresses induced by thermal cycling or disruption events are relevant. The final aim is of course to arrive at steady state operation which can be achieved by an alternative physics approach like the non-inductive current drive for Tokamaks or the Stellarator concept. For this operation mode creep- and creep-rupture is the dominating material property and the design is limited by primary stresses and temperature. However, even in this case scheduled or unscheduled shutdowns are expected so that appropriate fatigue- and fatigue-creep data have anyway to be available.

2.2 Breeding blanket options

General consensus exists in the fusion community that for integrated FW/breeding blankets only a limited number of combinations of structural materials with breeding/coolant media and special purpose materials exists. They can be classified with regard to the breeding materials into two major categories: a) solid breeders and b) liquid-metal breeders with the options of self-cooled or separately cooled versions. As solid ceramic breeder materials Li_2O, Li_4SiO_4, Li_2ZrO_3 and Li_2TiO_3 are under discussion, whereas liquid breeder materials are lithium or lithium-lead. Three major structural materials, ferritic-martensitic steels, vanadium alloys and SiC/SiC ceramic composites have been considered in different designs. Other options or combinations are derivatives. The major breeding blanket categories are compiled in Tab. 2 and include also the system pressures and the estimated temperature ranges for the structural materials in the designs. These latter data are based on estimates of high temperature strength, corrosion resistance and possible limits at the lower temperature end by coolant inlet temperatures or irradiation hardening, but have not been assessed in detail. Two of the proposed combinations are part of the above mentioned European Blanket Project [10,11].

Table 2: Major breeding blanket concepts

	Coolant	Breeding material	Structural material	Neutron multiplier	OperationConditions	
					Temperature	Pressure
He/LiCe/FS/Be*	He	LiCe	F/M-steel	Be	250-550°C	5-20 MPa (8 MPa)
He/LiCe/SiC/SiC/Be	He	LiCe	Ceramic composite SiC/SiC	Be	450-950°C	5-20 MPa
Li/V	Li	Li	Vanadium alloy	Li	350-750°C	~1 MPa
H_2O/Pb-Li/FS**	H_2O	Pb-Li	F/M-steel	Pb-Li	250-550°C	12-15 MPa (15,5 MPa)

* HCPB - Helium-Cooled Pebble-Bed Blanket / EU; Pressure data in brackets
** WCLL - Water-Cooled Lithium-Lead Blanket / EU; Pressure data in brackets
LiCe= Lithium Ceramic Breeder Materials: Li_2O, Li_4SiO_4, Li_2ZrO_3 or Li_2TiO_3

They use a low-activating ferritic-martensitic steel in combination with either a solid lithium ceramic (LiCe) as breeding material, beryllium as neutron multiplier and helium as coolant (HCPB-blanket), or liquid Pb-Li as breeding medium in the water-cooled lead-lithium WCLL-blanket. From mechanical properties point of view the temperature window is estimated to be between 250 and max. 550°C, where the creep rupture properties limit the upper and the irradiation-induced ductile-brittle-transition-temperature (DBTT) the lower temperature. In the given upper temperature limit additional fatigue effects are not considered. Further technical targets of this development are: an average neutron wall loading of 2,2 MW/m^2 or/and 3,5 MW/m^2 at the outboard blanket, and a lifetime of 20000 hours which integrates to an accumulated wall loading of 5 and 8 MWy/m^2 respectively.

One realistic application for vanadium alloys (e.g. V-4Ti-4Cr alloy) is their combination with liquid lithium as coolant/breeding medium. This option takes advantage of an excellent heat transfer characteristics of lithium. Due to a high creep strength at elevated temperature a potential for a relatively high temperature operation with a high thermal efficiency exists, provided that by appropriate coatings an electrical insulation between the liquid metal and the structural walls can be achieved. The envisaged max. system temperature of about 700 to 750°C relies on the additional assumption that the creep properties are not degraded by irradiation. The restrictions at the lower temperature end will – similar to the case of ferritic-martensitic steels - be due to the irradiation-induced DBTT shift (See Chapter 3).

The fourth concept uses a ceramic composite of type SiC/SiC as structural material in combination with a helium-cooled lithium ceramic compound as breeding material (ARIES-I,IV and DREAM studies). Alternatively, the possibility to use liquid Pb-Li as combined cooling – breeding medium in combination with SiC/SiC has also been investigated recently in the European community (TAURO-concept [13]). Both blanket concepts have nominally the highest temperature capability due to the excellent high temperature strength properties and would lead to an advantageous efficiency. However, as will be discussed later in Chap. 3, a number of principal questions regarding the material behaviour and materials technology has to be investigated before a technical realisation of these concepts can be foreseen.

In summarising, there exist four major categories for combined first wall/breeding blanket components which obviously take into account the best combinations of structural, breeding/coolant and other materials. The proposed operational parameters are based in most cases on preliminary design studies and can be used as more generic targets for the further development of materials. Since the knowledge about and the experience with the proposed materials even in the unirradiated state is very different it is at present very difficult to judge their technical feasibility and potential.

3. Materials, requirements and performance in nuclear environments.

In this chapter a more detailed description of the structural materials mentioned above, their major fields of application, a comparison of some important properties and their performance in nuclear environment is given.

3.1 Materials

Commercial **ferritic-martensitic steels** with chromium contents in the range of 9-12%Cr are used extensively in conventional and nuclear power plants up to 580°C all over the world. They have a balanced spectrum of non-nuclear properties like sufficient high temperature strength, good corrosion resistance and appropriate thermophysical data. As will be shown later also extensive experience in nuclear fission reactor programmes is available [14]. From the beginning of the European Fusion Technology programme in 1983 the Nb-stabilised 9-12% Cr steels, denominated MANET-alloys (**MA**rtensitic Steels for the **N**ext **E**uropean **T**orus) were an essential part of the R&D programme [15]. Parallel R&D programmes were performed in Japan, USA and elsewhere. Extensive collaboration under the IEA Implementing Agreement for the development of Fusion Reactor Materials between Europe, Japan, USA and more recently Russia has pushed this development further, especially in direction of alloys with reduced or low longterm activation, as will be discussed later. Periodic reports give a comprehensive overview on these activities [16,17].

Vanadium-base alloys have no application in conventional technologies. They gained in interest as cladding materials in gas-cooled Fast Breeder Reactors in the late 60ies because of excellent high-temperature creep strength [18] and appropriate thermophysical and nuclear properties. The results of irradiation experiments with neutrons and after helium implantation were very promising [19,20], but the incompatibility with oxide fuels and with high-temperature helium as cooling medium ended this development in the early 70ies. The interest in this material group was renewed when the development of low activation materials for fusion was discussed [21] and it could be shown that especially alloys with the constituents V-Cr-Ti have by far the best i.e. fastest decay of radioactivity. Since the solubility of vanadium and the alloying elements chromium and titanium is extremely low in liquid alkali metals and has reasonably low values in liquid lead-lithium it was logical to propose these alloys as structural parts in combination with liquid Li and Pb-Li as combined breeding and cooling medium in blankets (see Tab. 2). However, internal corrosion induced by the pick-up of nitrogen and carbon from the liquid metal coolant leads to an embrittlement which hardly can be avoided, even if very low interstitial concentrations in the liquid coolants are guaranteed. The effective max. operational temperature may therefore be restricted by this embrittlement effect [22]. Since vanadium alloys anyway need an electrical insulation in self-cooled breeding blankets to suppress magneto-hydrodynamic effects (MHD), the coating with electrically isolating materials like AlN or CaO which also improve the corrosion behaviour should be an appropriate way to circumvent both problems, provided the coatings are stable under operation. More recent reviews give a comprehensive overview on the state of the art [23-25].

Fibre reinforced SiC/SiC ceramic composites are used in aerospace and fossil energy plants for high temperature applications and have gained interest for the fusion materials community due to low short term activation and decay heat and very favourable high temperature strength properties. In the European Community as well as under the above mentioned IEA implementing agreement a broad R&D programme has been launched and continuous progress has been reported at recent conferences [25,26,27]. Major lack of knowledge exists in the irradiation performance of such innovative materials, but also other technical issues have to be solved before an application can be taken into account.

3.2 Aspects for materials selection

There are many requirements which have to be fulfilled by structural materials in order to satisfy the stringent demands in fusion reactors. Some of the properties and technology requirements on which a materials selection can be based are summarised in Tab. 3. They include appropriate conventional properties like thermophysical, mechanical and corrosion data as well as demands for the availability of necessary materials processing techniques. Of special importance is the knowledge of possible property degradation under neutron irradiation and the demand for minimum activation under neutron irradiation. This is because structural alloys and functional materials like breeding or first wall protecting materials are the major source of radioactivity, nuclear decay heat and radiotoxicity in fusion reactors. Few of important properties are discussed in the following in a comparative way:

Tab. 3: Criteria for materials selection in nuclear technology

Unirradiated properties	Irradiated properties	Technology
• Thermophysical Properties (stress factor)	• Radiation hardening	• Availability (qualified manufacturer)
• Tensile strength	• Embrittlement/Ductility	• Qualified fabrication routes
• Creep strength	• Swelling and irradiation creep	• Weldability/joining
• Fatigue resistance	• Microstructural stability	• Dimensional stability of components fabricated
• Resistance to crack propagation	• Irradiation-assisted stress corrosion cracking	• Non-destructive testing
• Toughness /DBTT	• Reduced activation and radiological properties	• Costs
• Ductility/uniform elongation		
• Corrosion		
• Stress corrosion cracking		

3.2.1 Power density capability (PDC), tensile- and creep rupture strength

The power density capability determines the maximum achievable neutron wall loading and - in combination with the compatibility/corrosion and high temperature strength - the temperature window of operation. It includes the thermal expansion coefficient α, the thermal conductivity λ, the ultimate tensile strength R_M , the Youngs modulus and the Poisson ratio ν and gives the allowable neutron wall loading in MW/m² for a 1mm thick first wall. In Fig. 3 a comparison of several materials which are proposed to be used as structural alloys is given. It includes 316 LN-IG in ITER, F 82H - a reduced activation ferritic-martensitic steel, a vanadium-base alloy of type V-4Cr-4Ti and the ceramic composite material SiC/SiC. More recently in design studies refractory alloys on the basis of Ta and Mo have been thrown in the discussion [28]. For comparison the capability of a typical copper alloy to be used as heat sink material in divertors is also given. The comparison shows that from the classical structural alloys vanadium based materials have the highest potential followed by ferritic-martensitic and austenitic steels. Surprisingly, the fiber reinforced ceramic composites SiC/SiC have the lowest PDC, though the monolithic SiC is well known to have good thermal conductivity properties. The reason for this unexpected behaviour is that a strong degradation of thermal conductivity under irradiation has recently been observed both, for SiC as well as for the ceramic composites [26,27,29].

Regarding the creep rupture strength of structural alloys, Fig. 4 gives in a Larson Miller plot a comparison for relevant structural materials with MANET II as a representative of the conventional 9-12%Cr ferritic-martensitic steels, 316 LN an austenitic alloy and a range of data for different binary or ternary vanadium alloys. As well known, vanadium alloys have by far the best potential for high temperature application [18,30], provided that corrosion and compatibility can be managed [22,31]. To give an example: for an envisaged lifetime of 20000 hours for the above mentioned HCPB-breeding blanket the allowable creep rupture stress level at 550 °C would be 160 MPa for MANET, 230 MPa for 316 LN and more than 400 MPa for a V-Ti-Cr alloy. Correspondingly, a stress level of 160 MPa would allow a temperature level of about 750°C for vanadium alloys for the required creep rupture time of 20000 hours.

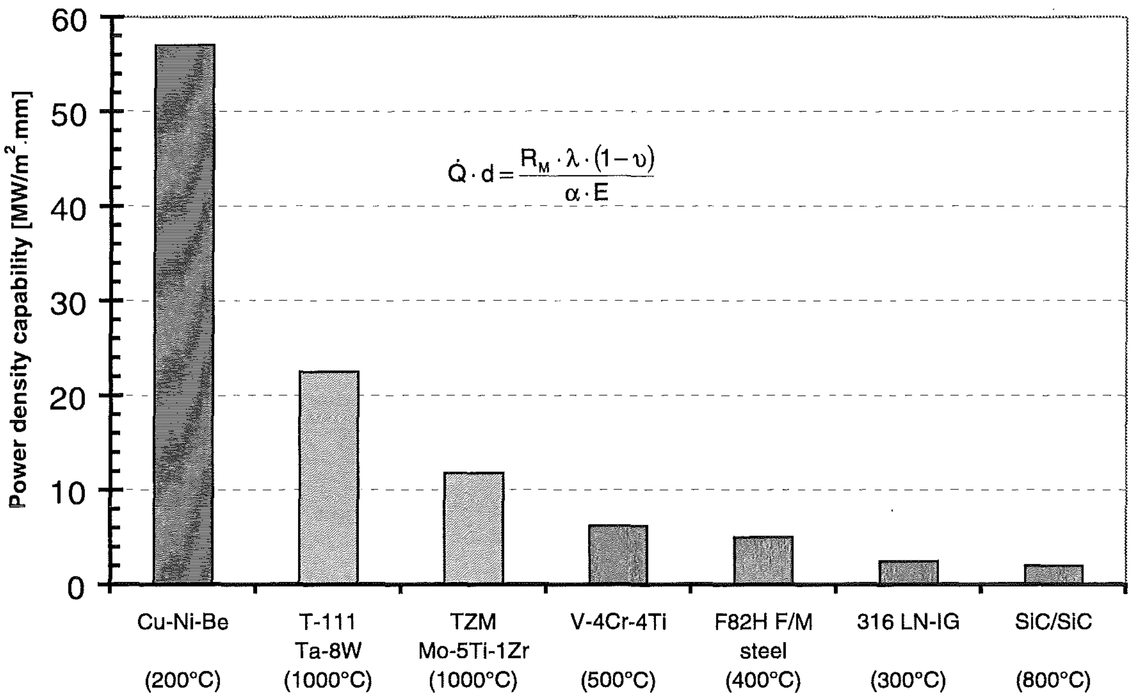

Fig. 3:

Power density capability of structural and heat sink materials [28]

3.2.2 Radiological properties.

In fusion reactors the structural materials will generate a main source of neutron-induced radioactivity which has strong influence on the environmental and safety aspects of fusion power [32]. Therefore the possibility to develop materials with reduced or low radioactivity was early investigated for the major classes of structural materials [33-37]. The number of kinematically allowed transmutation reactions increases strongly with increasing neutron energy for all elements due to the surpassing of so-called Coulomb thresholds. In addition it was shown that charged particles like protons, deuterons, tritons and α-particles, generated by primary transmutation reactions of atoms with neutrons, can themselves produce additional transmutations via so-called sequential reactions [38]. In a number of elements these reactions can surpass the primary neutron-induced activation by orders of magnitude [39].

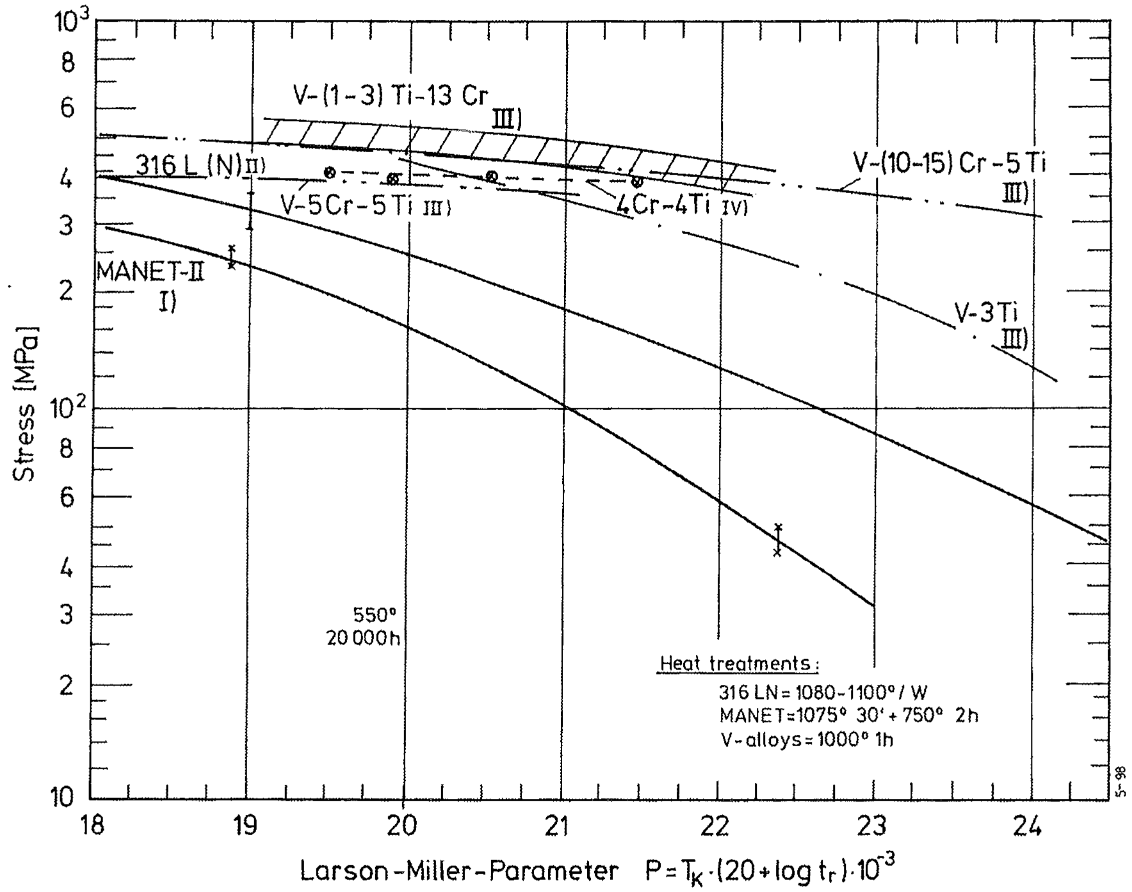

Fig. 4:

Creep rupture strength of structural materials

The development of low activating materials can be viewed under two different aspects: One is coupled to the safety of fusion reactors with the potential to disperse radioactivity in case of an accident. Primary path for such an event is the volatilisation of the material itself or its oxidation or burning (important for materials like carbon, Be, Li etc.). A heat source to drive such events is the so-called decay heat stemming from the radioactive materials after the immediate shut down of a reactor. Materials with a low activation and low decay heat in a time period of days or months after reactor shutdown are therefore preferable, as is the case for the ceramic composite SiC/SiC. Maintenance and repair would also be facilitated by the use of such materials. The second aspect regards the "longterm" activation after 100 years and more, which determines the appropriate ways for waste disposal and material recycling, where some vanadium-base alloys have the best potential. Since none of the above mentioned materials has the lowest activation or decay heat over the complete time scale of decay the decision has to be made where to put the emphasis in the development. In the European long term programme priority has been focussed on a reduced or low irradiation-induced longterm activation.

The first step for the development of low activating alloys with a fast decay behaviour after about 100 years is a thorough analysis of the contribution of all important alloying elements.

For the calculation of data like activity, γ-dose rate, nuclear decay heat or radiotoxicity the FISPACT-Code, especially adapted to high-energetic fusion neutrons, with the input of activation- and decay data from the European Activation System (EASY) is generally used in Europe [40,41]. An important supplement of this computational code was the inclusion of sequential reactions by FZK in the FISPACT Code 4.0 and following versions [39]. At present the FISPACT 97 /EAF 97 is the updated version [42].

For a comparison of the activation behaviour of different material groups the calculated γ-dose rate of OPTIFER, a low-activation ferritic-martensitic steel, the vanadium alloy V-4Cr-4 Ti and the ceramic composite material SiC/SiC composite after an irradiation with a wall loading of 5 MW/m^2 to 12,5 MWy/m^2 (125dpa) is plotted in Fig. 5. In addition a commercial titanium-base alloy has been included for comparison. These calculations are based on the chemical composition of the alloys and contain besides the specified alloying elements also so-called impurity or tramp elements. Compared with previously reported results the data differ in one respect, namely that the predicted advantage of vanadium alloys is orders of magnitude less than calculated for the "pure" alloys. Such divergent results have provoked in the past many discussions about the ranking or superiority of one or the other alloy group. But with the thorough discussion of the influence of impurities in the different materials [43,44,45,46] a more rational view on the realistic possibilities of low-activation materials has been achieved in the materials community. Future activities have to be concentrated on the technical possibilities to keep the level of unwanted impurities as low as possible for each material group . The prospects to reduce unwanted tramp elements to values in the range of appm in technical processes are promising, especially since similar approaches- but with other aims- have been adapted for the production of "super clean" nickel alloys in the past [47].

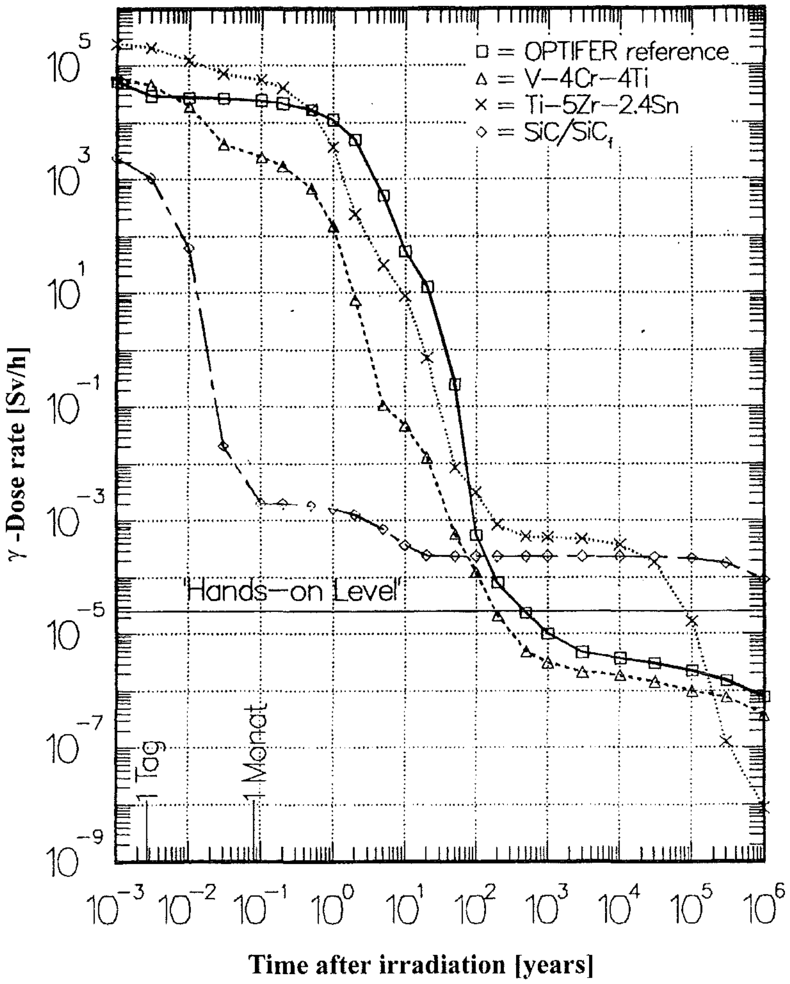

Fig. 5:

Contact-γ dose for structural materials after an integral wall loading of 12,5 MWy/m^2

3.2.3 Existing experience with nuclear materials.

Main knowledge about the material behaviour under high neutron fluences stems in Europe from fuel element development for Fast Reactors and from core structural materials in Light Water Reactors [14,48]. In Tab. 4 a collection of materials tested in Europe in different nuclear components and their operational conditions (temperature range and max. achieved fluence levels) is given. It comprises mainly conventional and optimised austenitic and ferritic-martensitic steels and Ni-base alloys. In USA and Russia similar materials have been tested to about the same fluence levels [49,50]. In addition irradiation tests are known from vanadium-base alloys and dispersion-strengthened ferritic steels. Radiation experience with ceramic composites is very limited up to now [26,27]. The general conclusions to be drawn from these investigations can be summarised for the different materials groups in the following way:

Austenitic stainless steels are used as cladding and wrapper materials of fuel elements in FBRs and as core structural materials in LWRs in a wide temperature range from about 280 to 630°C. They have been exposed in FBRs in a large quantity up to fluence levels in the range of 120 dpa and have reached max. values of 150 dpa in combination with a 20% fuel burn-up. The most limiting radiation phenomenon is radiation-induced swelling and dimensional instability. Such high fluence levels can in austenitic steels only be achieved if appropriate material modifications of existing commercial alloys are made, e.g. by adding swelling-reducing elements like Ti, Si, and P and by choosing the optimal thermo-mechanical material pre-treatment like 20% cold-working. By such investigations the onset of swelling can strongly be shifted to higher fluence levels [14,50,51]. However, since this improvement is only a transient effect, the swelling in combination with swelling-enhanced irradiation creep remains after the onset of swelling the life limiting radiation effect. In solution-annealed austenitic steels this threshold is much lower and as an example for the conventional alloy AISI 316 LN to be used in ITER the expected dose limit is in the range of 30 dpa.

Commercial austenitic stainless steels are not low activation materials. Attempts by substituting Ni through Mn and replacing Mo by other strengthening elements in order to achieve a reduced activation material were not very promising, so that the further development was terminated [35].

Table 4: Experience with structural materials in Nuclear Technology in Europe

Materials group	Field of application		Operational conditions Temperature range, Max. fluence levels	
Austenitic stainless steels of Types 316, Fe-15Cr-15Ni-Ti stab.	FBR:	Cladding and wrapper materials of fuel elements	360-600 (630°C)	Max. 120-150 dpa
High-Ni austenitic steels and Ni-alloys e.g. PE 16, Inconel	FBR:	Cladding materials of fuel pins	360-600 (630°C)	max. 135 dpa
Ferritic-martensitic 9-12% Cr steels e.g. 1.4914, EM-10, FV 448, HT9	FBR:	Cladding and wrapper materials of fuel elements	360-600°C	max. 115-145 dpa
Austenitic stainless steels and Ni-alloys e.g. 18Cr-9Ni-Ti/Nb or 316 L; Inconel etc.	LWR:	Core structural materials	280-320°C	max. 50 dpa

High-nickel austenitic steels and Ni-based alloys, mostly strengthened by γ'-precipitates have also been tested extensively as cladding materials for fuel pins in FBRs. The PE 16 alloy showed in the solution–annealed and aged condition a remarkable swelling resistance, a sufficient ductility and a very good overall performance up to 135 dpa [14]. Ductility exhaustion due to high-temperature helium-embrittlement could eventually limit the lifetime especially if exposed to high-energetic 14 MeV neutrons. A substantial decrease of ductility has recently been reported for Inconel 718 after 800 MeV proton irradiation at low temperatures which is possibly due to radiation-induced segregation phenomena in combination with a change in fracture mode [52]. Since a chemical modification of precipitation-hardened Ni-alloys to achieve low-activation materials is hardly possible, these materials - though superior in high temperature creep rupture strength - are not foreseen as structural parts in fusion reactors.

Ferritic-martensitic steels of the Fe-9-12% CrMoV(Nb) type have mainly been used as wrapper and in few cases as cladding materials in FBR fuel elements in the temperature range of 360-550 and 600°C resp.. Maximum irradiation levels between 115 and 145 dpa have been accumulated. In these alloys a high swelling resistance and nearly no indication of high-temperature helium embrittlement were detected [14,53]. At typical FBR-conditions there were also no indications of a serious degradation of fracture toughness or impact properties up to high neutron fluence. The situation changed when tested below about 400°C, where radiation hardening and embrittlement combined with a remarkable shift of the DBTT towards higher temperature occurred [54]. Though it could be shown that 9% Cr-alloys were less sensitive to this degradation than 12% Cr steels [55], this is one of the critical issues for the application of ferritic-martensitic steels. Another limiting property is their decreasing strength above about 580°C due to recovery. This weakening in strength can even be accelerated by irradiation. Therefore, a further strengthening mechanism by adding oxide dispersions seems a promising way, which should be followed in future R&D activities. As has been mentioned in this class of materials the possibility exists to achieve reduced or low longterm activation by chemical modifications, i. e. the substitution of elements like Mo, Ni and Nb by W, Ta and Ti. This way has been pursued in Europe, USA and Japan since the mid 80ies partly in collaboration under the IEA Implementing agreement /Research and Development of Fusion Reactor Materials [56,57,58,59]. The new series of alloys which in their majority lie in the compositional range of 7-10%CrWVTa have resulted in much better radiological properties (see § 3.2.2) and also attractive fracture toughness- and impact data, including a reduced tendency for irradiation-induced DBTT shift [54,60] Fig. 6 shows as an example the impact properties of a series of "low-activating" (LA) developmental alloys like

OPTIFER and F 82H-mod after low-dose, low temperature irradiation in comparison to conventional alloys of the MANET series. If the encouraging results of low-fluence irradiations can be confirmed by high-fluence tests the new alloys should fulfil the requirements for a DEMO-reference material.

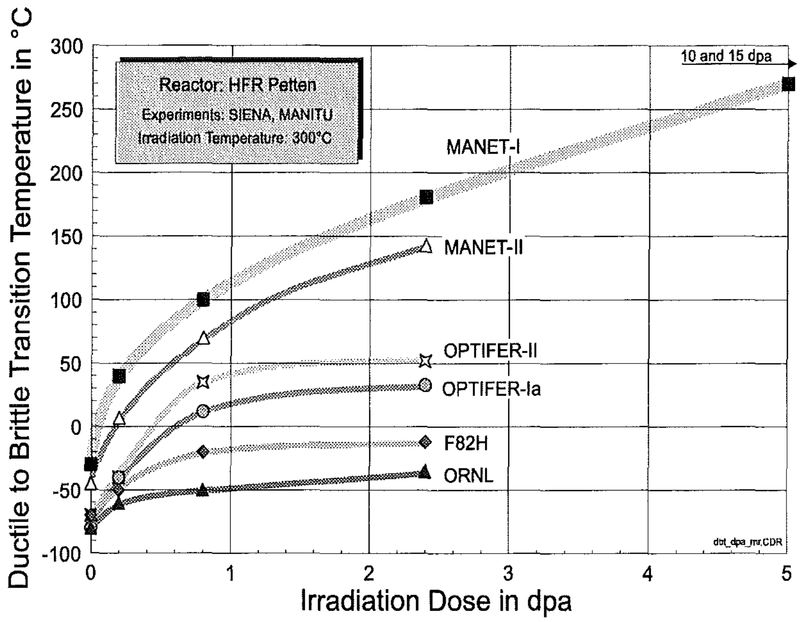

Fig. 6:
DBTT shift of conventional and reduced activation ferrtitic-martensitic steels after low-fluence neutron irradiation

Vanadium alloys (mainly based on V-Ti-Si or V-Cr- Ti compositions) do not have an essential application in conventional or fission reactor technology. The attempt to use them as cladding material in combination with uranium oxide fuels in Fast Breeder Reactors failed due to thermodynamic incompatibility, as mentioned before. However, since very early investigations showed for some alloys a quite good resistance to irradiation-induced He-embrittlement up to 650°C [19,20], and more recently low swelling was reported for selected vanadium alloys under fast reactor irradiations [23], the potential for high fluence application seems promising. Furthermore, the alloys based on the elements V, Ti and Cr have nominally the best conditions for low longterm activation. One major point of concern is - like in the case of ferritic-martensitic steels and refractory metals - the radiation hardening at low irradiation temperature in combination with a degradation of the impact and fracture toughness properties. In contrast to some previously reported results in which radiation hardening was not found [61], most recent data compiled in Fig. 7 [from Ref. 25] show for irradiation temperatures of 425°C and below a remarkable yield strength increase which is combined with a reduction of uniform strain below 1% and a strong shift of DBTT in Charpy-V tests. Similar to the situation in ferritic-martensitic alloys one has, therefore, to reckon with a limitation of the lower end of operational temperature due to radiation hardening. This limit could lie for vanadium in the range of 400 to 500°C.

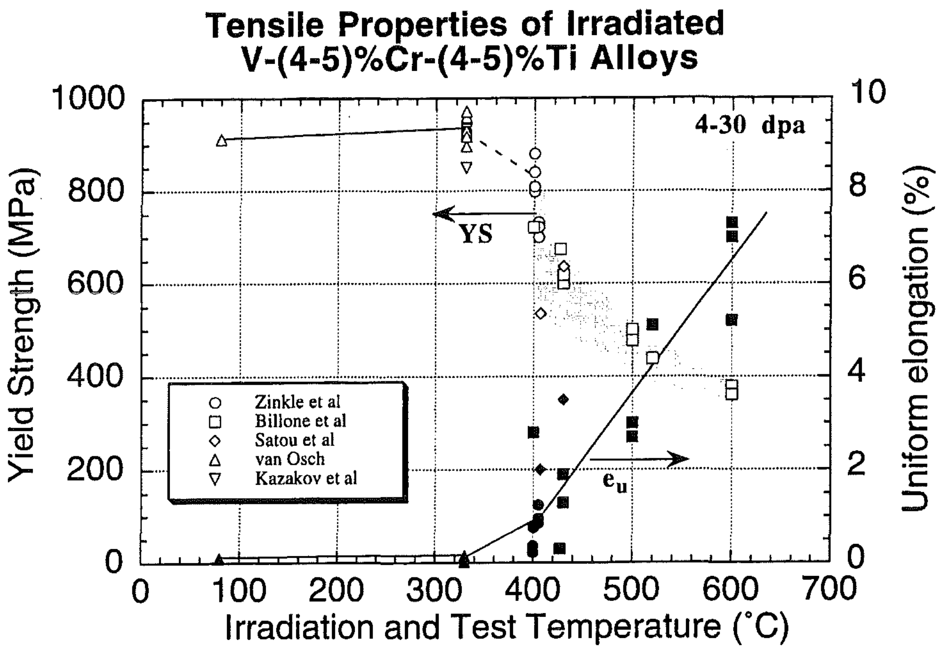

Fig. 7:
Tensile properties of irradiated V-Cr-Ti alloys [25]

Fiber–reinforced ceramic composites of type SiC/SiC have not yet been used extensively in nuclear environments and the experience is limited to specific material irradiations, mostly at relatively low fluence levels (20-max. 50 dpa) in fission reactors and accelerators. In many respects they create questions regarding radiation resistance and stability under 14 MeV neutron irradiation. In comparison to metals and alloys the fundamental displacement damage processes in covalent materials are less understood, including their effects on thermophysical and elastic properties. Also, more fundamental parameters like the displacements per atom or the disorder defect fractions are much more complicated to be calculated than in metallic structural materials. This is also true for partial diffusion processes for the species Si and C, which are often missing [27]. Due to strongly increasing cross sections for inelastic n,α-processes in Si the characteristic He/dpa relation is about one order of magnitude higher than for ferritic–martensitic steels or vanadium alloys and varies for SiC from 150 to 50 appm He/dpa, depending on the location in a fusion blanket. This might have consequences on swelling and embrittlement phenomena at the operational temperature range. Thermal conductivity is in this material caused by phonon scattering which will definitely be increased by radiation-produced defects and hence diminish the conductivity [29]. Fig. 8 from ref. [27] shows such a reduction of a neutron-irradiated 2D Nicalon CG/SiC composite as function of the displacement damage. With such data in the range of 2W/mK, far below the unirradiated and even irradiated values for monolithic SiC (15-20W/mK), the very low power density capability shown in Fig. 3 is explained. Finally since SiC/SiC consists of a β-SiC matrix, a SiC-fiber material with a different stochiometric composition and an interphase of graphite, the dimensional stability of such a composite under irradiation is dependent on the relative swelling or densification of the constituents. In case that a mismatch occurs in the different constituents a delamination of matrix and fibers is the consequence and leads to a reduction in strength and fracture toughness in the order of ≥20% [26,62]. The saturation tendency of the observed strength loss at fluence level of ≥5 dpa has been attributed to the kinetic of the irradiation induced shrinkage process of the fiber in relation to the matrix. Smaller irradiation induced strength changes in combination with improved swelling resistance would be expected in composites where advanced fibers have identical properties to the matrix Therefore this is one of the major development tasks for the future.

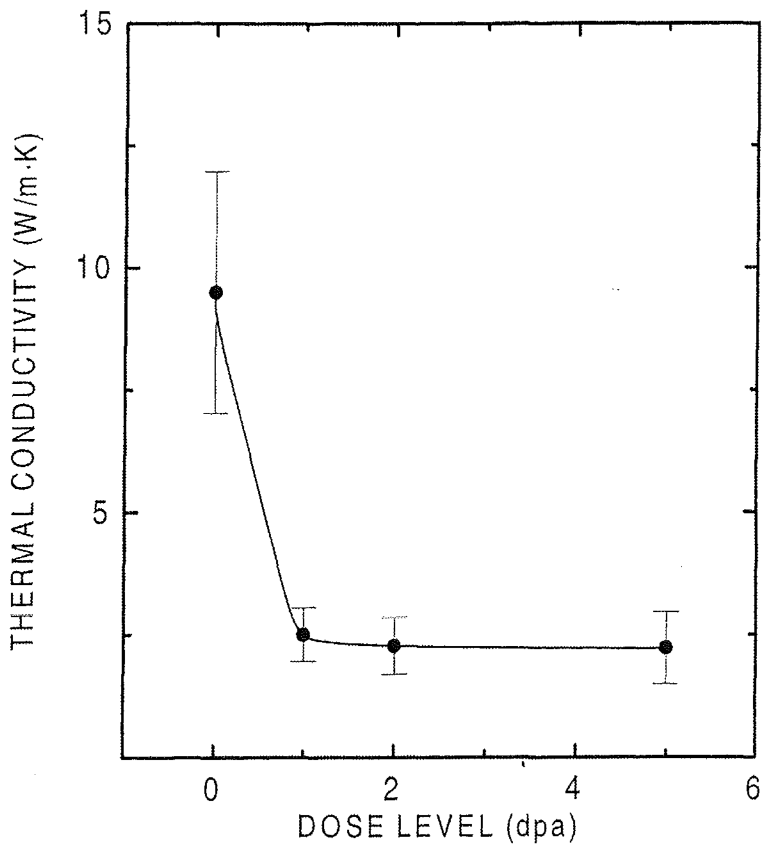

Fig. 8:
Thermal conductivity of a SiC composite as function of displacement damage [27]

In summarising, the material performance under LWR-, FBR- and other simulation irradiations (e.g. accelerators), provides very valuable information, allows a preselection of materials and indicates the major problems to be expected in a typical fusion environment. However, a direct extrapolation of these results to fusion-relevant conditions is at present not possible, since important radiation damage parameters like neutron energy, recoil spectra and transmutation reactions under fission reactor irradiations can significantly differ from irradiation conditions in a fusion reactor environment as will be discussed in more detail in chapter 5. Though we are now able to quantify the differences in irradiation source terms for the different facilities there is no reliable theoretical background or experimental data base to calculate or safely extrapolate to fusion reactor conditions for the manifold radiation damage phenomena [63]. To give an example, the effect of higher helium generation rates or variable He/dpa relations on swelling of a material cannot be forecasted by existing theories. The only realistic strategy is therefore, to provide the fusion materials community with an appropriate tool for fusion-specific irradiations – as will be discussed further in the following two chapters.

4. Objectives and major steps of the European materials programme for DEMO

One of the aims of the European materials programme is the selection and qualification of a reference First Wall- and structural material for the Test Blanket Modules (TBMs) for DEMO. For the construction of these blanket modules conventional optimized 9-12% Cr steels of type MANET had been foreseen and have now been replaced by the newly developed ferritic-martensitic 7-10% CrWVTa steels, optimised for reduced activation and improved in fracture toughness properties. In Fig. 9 a time driven schedule for their further development and qualification is given, showing the main activities and major phases of such a programme. For the alternative materials like V-5Cr-5Ti or the ceramic composite SiC/SiC under discussion a more extended time for their development has to be foreseen:

Phase I: In this phase, terminated approximately with the end of the Fourth European Framework programme end 1998-99, the development and screening of the new alloys with reduced longterm activation by testing of major properties including irradiations to very moderate neutron levels of few dpa has had priority. This programme comprised 8 small melts of European RA-alloys (OPTIFER-,OPTIMAX-,BATMAN- and LA12-alloys) and an industrial batch of a 7% Cr alloy (denominated F82H-mod.), supplied by Japan and investigated in a Round Robin test by laboratories in Japan, Europe and USA under the auspices of the IEA. The results of this exploratory work - though not yet completed – were promising as mentioned above [58-60]. Design-relevant data have already been generated and fed in the data bank for the engineering design of the TBMs - as indicated in Fig. 10.

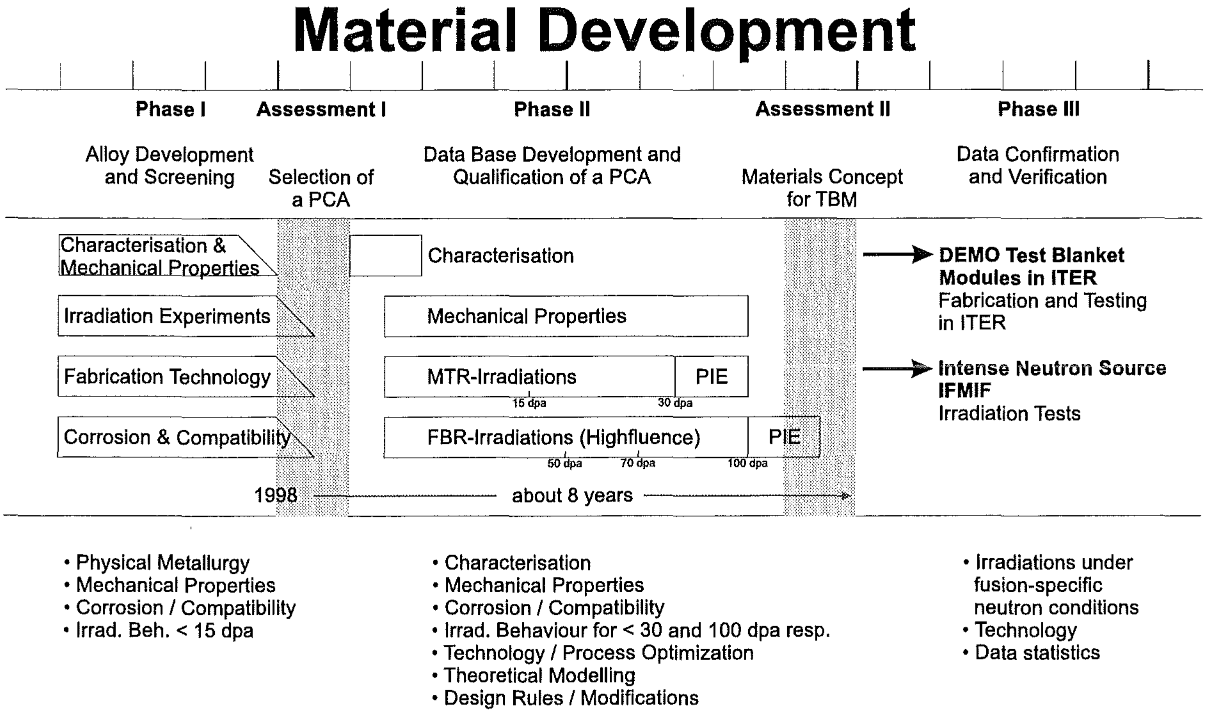

Fig. 9

Activities for the development of RA-ferritic-martensitic steels

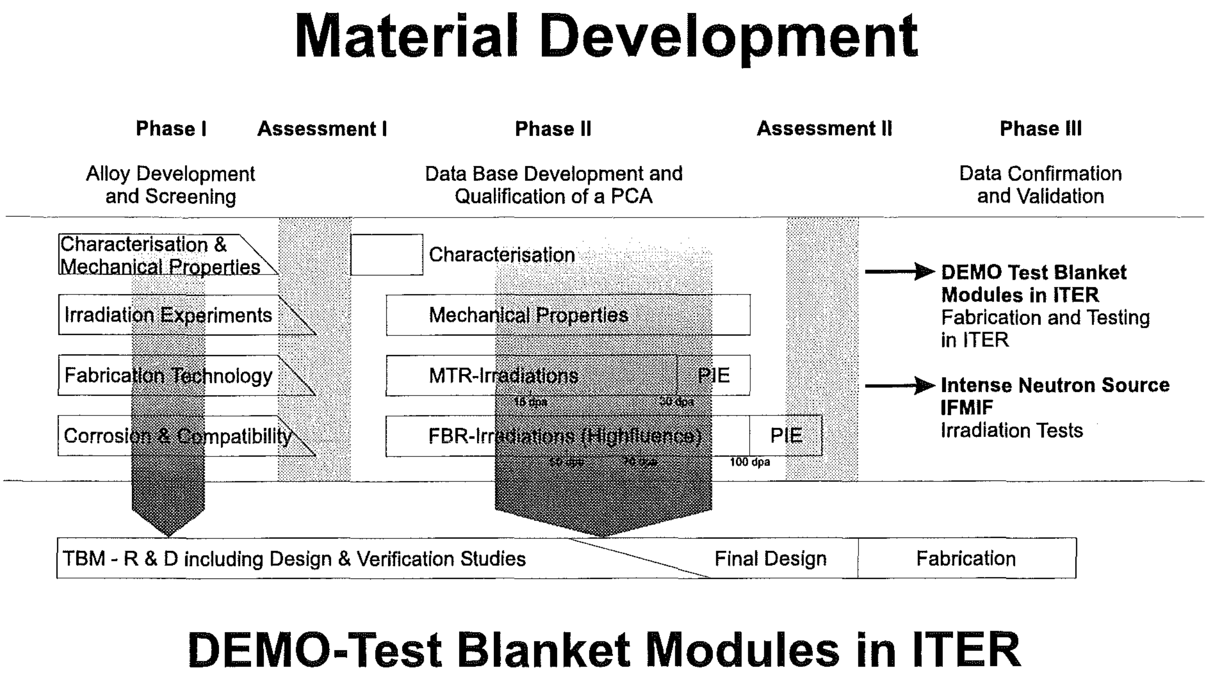

Fig. 10

The interaction of structural materials and DEMO Test Blanket Module development

Assessment I: Based on the overall experience from the preceding work in **Phase I** a primary candidate alloy (PCA) EUROFER 2000 will be specified in the **Assessment I** after an overall evaluation of existing data. This PCA will substitute the preexisting developmental alloys and will be used as reference for further investigations. The follow-on steps in this development are summarized in

Phase II: Main aim in this period is the timely qualification of the PCA for the design and construction of the two DEMO-Breeding Test Blanket Modules (TBMs) in ITER. This includes the determination of conventional properties, the development of appropriate fabrication technologies, the necessary adaptation of existing design rules to fusion conditions and a full scale testing under irradiation for (10) 30 dpa and 70 dpa resp. to cover the requirements for the Basic and Extended Performance Phase in ITER and a DEMO-blanket. The minimum time for these investigations is according to Fig. 9 determined by the long lasting creep- and creep-fatigue experiments and by irradiation experiments including post-irradiation examinations. In addition some open questions like the influence of ferromagnetism on the plasma stability and magnetization forces have to be investigated. Also the possibilities to increase the creep rupture strength of such alloys by adding oxide dispersions in connection with new fabrication methods like hipping have to be examined.

A necessary condition for the punctual performance of the **Phase II**-activities is the availability of fission reactors for testing. In Tab. 5 a survey of Material Test Reactors (MTRs), presently used in the European Longterm Materials programme and hopefully available for the next step of investigations is compiled. Most of the reactors possess instrumented irradiation rigs for temperature-controlled tests in the temperature region relevant for the reference breeding concepts. However, in-situ test facilities like uniaxial creep or fatigue testing devices are not or (no longer) available!

For the class of MTRs an effective 7.5 dpa/year accumulation has been adopted for estimating the necessary time for performing ITER-relevant experiments up to 10-30 dpa in these reactors. To reach a DEMO-relevant damage level of 70 dpa in irradiation times acceptable for the experimentalists only Fast Breeder Reactors are efficient enough. Within the EU only the Phenix reactor will be available for a limited time (until 2004) for uninstrumented tests and a fixed temperature range between 380 and about 600°C. In order to study the very important low-temperature irradiation effects in F/M-steels (and V-alloys) at the relevant temperature between 250 and 400°C, it is, mandatory to develop soon special low temperature/high-fluence rigs in FBR´s which allow to study the critical embrittlement phenomena and DBTT-problems. An international collaboration is necessary in this field.

In general it has to be expected that the number of reactors available for future experiments will be further reduced, which could raise a serious problem for the whole fusion programme. This is a strong argument for an accelerated development and construction of an intense neutron source.

In Fig. 10 an interlink between the different phases of material development and the corresponding activities for the DEMO-Test Blanket Modules is made. The design-relevant property data generated in the materials programme will continuously be fed into the current R&D- design and verification phases of the TBMs and it is expected that the majority of data for the reference alloy - including the time-consuming creep- and irradiation tests - will be available for the final design phase, prior to the fabrication start of the modules. In this phase also the properties of functional materials like neutron multipliers (Be) and ceramic breeder materials and the interaction of materials in components including compatibility, corrosion, mechanical interaction due to irradiation phenomena etc. have to be studied.

Table 5: European Reactors used for Fusion Materials Irradiations

Site	Name	Type*	DPA/FPY(Fe)	Special test rigs and devices	Availability at present
Mol	BR-2	MTR	≤ 11	Yes	+
Petten	HFR	MTR	≤ 9	Yes	+
Saclay	OSIRIS	MTR	≤ 7	Yes	+
Studsvik	R2	LWR	≤ 8	Yes	+
Marcoule	Phénix	FBR	≤ 30	No	Until 2004

* MTR = Material Test Reactor   LWR = Commercial Light Water Reactor   FBR = Fast Breeder Reactor

At the end of **Phase II** which needs a thorough **Assessment II** of data, the reference material for the TBMs should be confirmed, or if unfavourable results appear, a new approach with alternative solutions has to be envisaged with consequences for the time schedule.

Phase III: In this phase the data validation and the verification of the material concept for fusion-specific DEMO-conditions has to be achieved. This means essentially that irradiation data stemming from experiments performed in fission reactors or in accelerators under simplified simulation conditions have to be compared, calibrated and validated with data gained from irradiation tests under fusion-relevant conditions. This work has to be supported by a strong and focussed theoretical modeling of radiation damage phenomena, taking into account physical damage parameters like recoil spectra or the production rates of transmutations like H and He, which deviate significantly between simulation experiments and tests in a real fusion environment.

5. An Intense Neutron Source for Materials Research

A prerequisite for such a strategy is the construction and early operation of a powerful, high-energetic, high-intensity neutron source. Such a need for a powerful test bed for fusion materials studies has been stated in the past at several occasions by high-ranking advisory boards like the Cottrell Blue Ribbon Panel or the Amelincks Senior Advisory Committee [1,2]. A favorable facility which under the auspicies of the IEA has been studied in a Conceptual Design Activity [64,65,66] and has recently been updated in a Conceptual Design Evaluation phase [67] is the **International Fusion Materials Irradiation Facility IFMIF.** IFMIF is an accelerator-based D-Li stripping source, where two parallel operating 125 mA, 40 MeV deuteron beams are focused onto a common liquid lithium target, and produce neutrons via a stripping reaction with a suitable energy spectrum at high intensity Fig. 11. This facility can fulfill essential users requirements, i. e. it adapts for structural materials the physically based damage parameters like dpa, transmutations and PKA-spectra reasonably well to the fusion environment [67,68]. This is demonstrated in the two following Figs. 12 and 13. In Fig. 12 an important parameter which characterizes the displacement events in materials in a more quantitative way than the dpa scale is given, the energy transfer function W(T). It describes the fraction of the damage energy produced by primary knocked–on atoms as a function of their kinetic energy and compares four different irradiation facilities. From this graph one can deduce that the IFMIF neutron source simulates very well the situation in a DEMO reactor. In contrast, in fission reactors like the HFR –Petten, where most of the irradiation experiments in the fusion programme are performed, the majority of PKA events occurs at much lower energy transfers. It is presently under discussion to what extent such differences can affect the formation of cascades and subcascades and hence have an influence on radiation hardening and other radiation damage phenomena.

A further physical damage parameter is the formation of transmutation products by inelastic events. In Fig. 13 helium production rate and dpa rate per week are given for different irradiation facilities. One can easily discriminate in this respect typical fission reactors like the HFR, ORR, HFIR and FFTF from the irradiation conditions in DEMO or IFMIF. They produce for comparable dpa-levels one order of magnitude less He per dpa in Fe-based alloys.

High energetic neutron sources like RTNS II-a rotating target D-T neutron source, which no longer exists or the Los Alamos Meson Physics Facility LAMPF can generate the correct He/dpa ratio but have much lower rates than expected for DEMO whereas high and medium flux test regions of IFMIF (HF and MF-positions) agree reasonably well with the conditions for a DEMO first wall position.

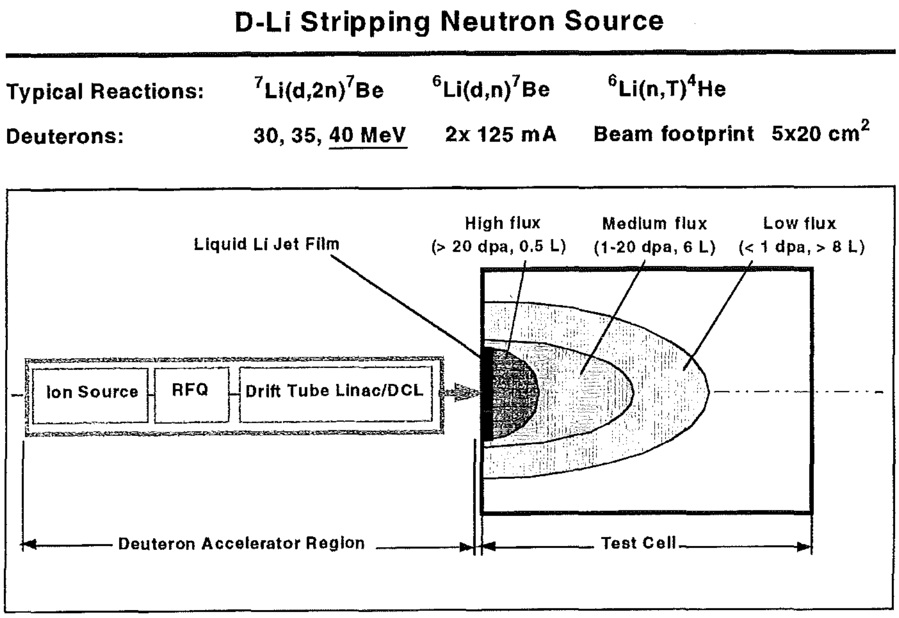

Fig. 11:

The d-Li stripping neutron source IFMIF

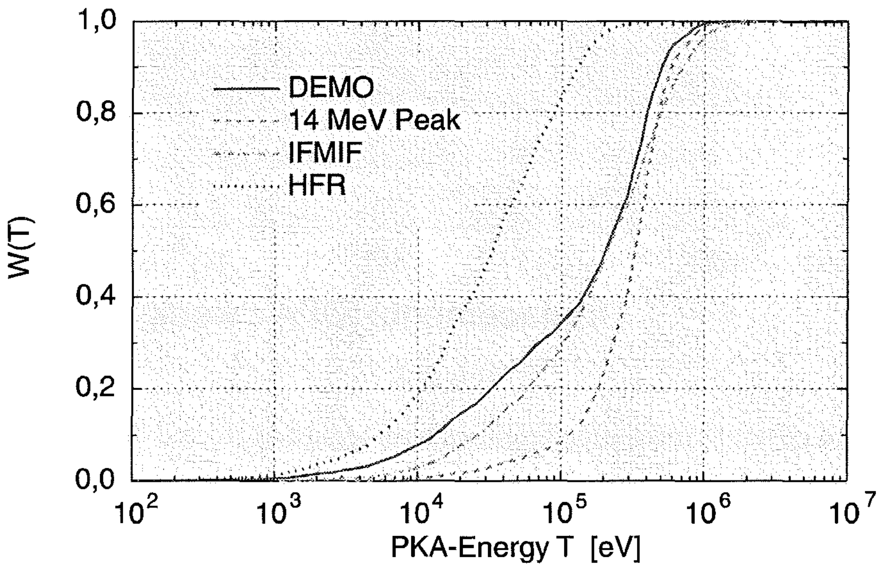

Fig. 12:

Fraction of damage energy produced by PKAs as function of their energy in different irradiation facilities

IFMIF also has - with the given test volumes in the high-, medium and low flux test zones in Fig. 11 - a sufficient capacity to perform necessary types of experiments for structural-, breeding and other materials in the appropriate temperature-, flux- and fluence regimes [66,69]. Such extended matrices – as compiled in Tab. 6 can for the limited irradiation volume in the High Flux Test Module - only be investigated, if miniaturized test specimen are used. For example by using a Small Specimen Test Technique (SSTT) about 1400 specimen including about 320 pieces for fatigue and fracture toughness investigations can be placed in the HFTM for one common irradiation. Recommendations for miniaturized specimen to be used in mechanical tests have already been elaborated [66,70]. The Small Specimen Test Technology is therefore a necessary test technique for IFMIF and has to be qualified to ensure that the results can be accepted for engineering design and licensing procedures.

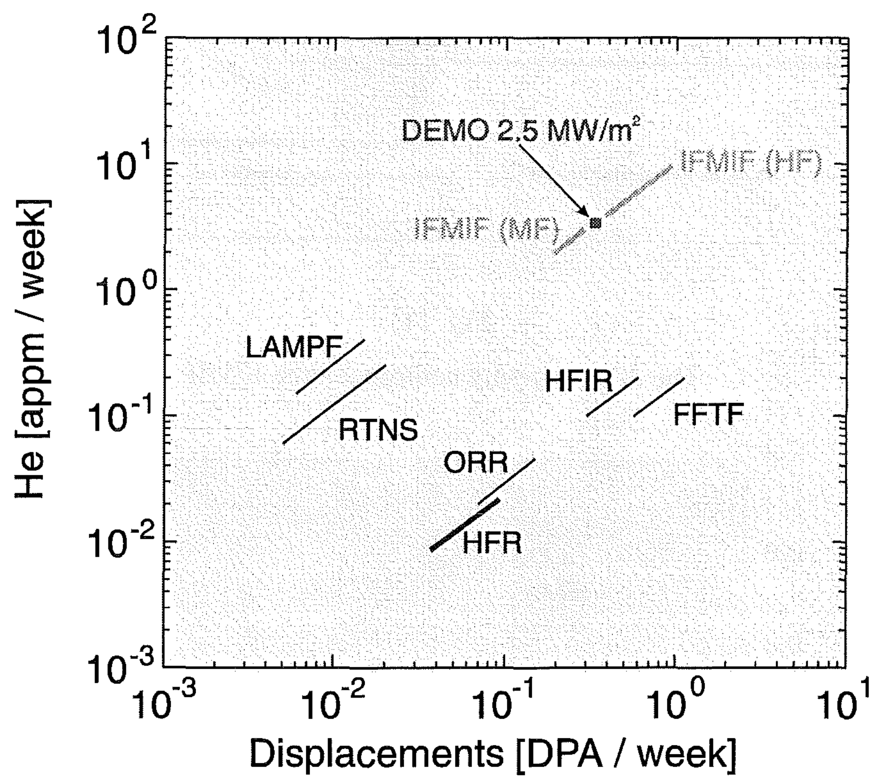

Fig. 13:

Classification of different irradiation facilities with regard to He and displacement production rates in Fe

In comparison to many other proposals this facility is based on proven technology with very moderate extrapolations and could hence be designed and constructed in a foreseeable time.. A technically realistic time schedule for its development is sketched in Fig. 14 - always provided, that positive decisions for its realization will come soon. As indicated there, it is presently proposed to develop in a next step (the KEP-phase) the technology of key elements like the Li-target, the ion source of the accelerators, and specific irradiation and test modules before in a consecutive phase an Engineering Design Activity and the construction of the facility will follow. Final aim is to have this test bed available before, or at least at the same time as ITER.

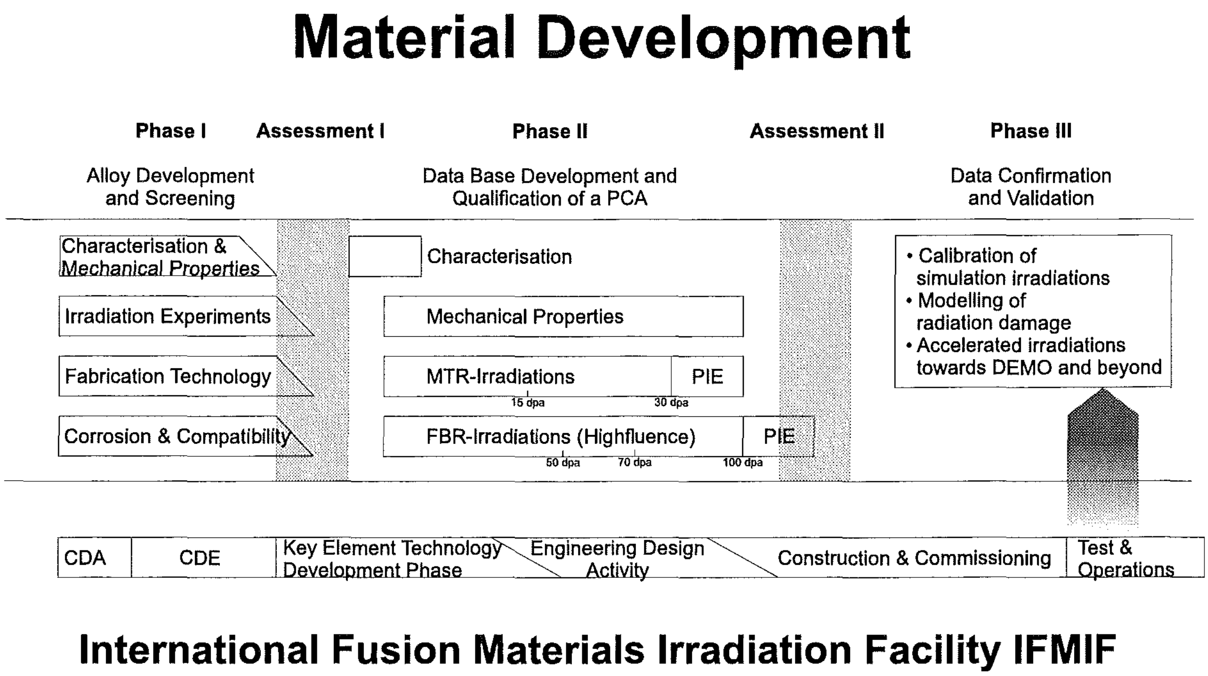

Fig. 14:

The role of an International Fusion Materials Irradiation Facility for the development and qualification of structural materials

From the technical capabilities it is evident that IFMIF is suitable for the screening and qualification of materials and the development of an appropriate materials data base for DEMO-relevant irradiation conditions. Calibration of results from simulation irradiations and - in combination with theoretical modeling - basic studies of radiation damage phenomena are additional fields of application.

Due to restrictions of the test volume, IFMIF and other accelerator-driven neutron sources will not be able to contribute to the testing and qualification of breeding blanket and other plasma-near components. One possibility to perform such tests on breeding blankets and other plasma-near components would be a high-volume neutron source with sufficient test capacity. In an IEA study the requirements for such a facility have been elaborated [76]. A test area of about 10 m^2, a moderate wall loading of 1-2 MW/m^2 and a target of 6 MWy/m^2 have been defined as necessary conditions in order to achieve concept verifications of different blanket designs in a first (up to 3 MW/m^2) and performance and reliability tests in a second phase (up to 60 dpa or 6 MW/m^3).

Table 6: Materials and irradiation conditions in different flux regions of IFMIF. The dpa rate is given for FPY, that is, for continuous operation

Flux region	Materials and types of experiment	Temperature	Fluence
High flux >20 dpa/yr 0.5 l volume	*First wall and blanket structural materials* Ferritic/martensitic steels Vanadium alloys SiC/SiC composites *Type of experiments* Instrumented irradiation capsules for PIE In a later stage instrumented in-situ tests	 250-550°C 350-700°C 400-1000°C	 ≤ 150 dpa ≤ 150 dpa ≤ 150 dpa
Medium flux 20-1 dpa/yr 6 l volume *Low flux* 1-0.1 dpa/yr 7.5 l volume *Ver low flux* 0.1-0.01 dpa/yr >100 l volume	*Materials* Structural materials Ceramic breeders Ceramic insulator materials RF-windows Diagnostic materials Superconductors, etc. *Types of experiments* In-situ creep-fatigue Tritium diffusion and release Stress corrosion (IASSC) Compatibility, electrical and mechanical integrity	 250-700°C 400-800°C RT-500°C RT-400°C RT-400°C Cryog. temp.	 ≤ 30 dpa ≤ 30 dpa 0.1-10 dpa 0.1-10 dpa 0.001-1 dpa < 0.1 dpa

Several neutron sources based on d-t fusion are under discussion: the mirror-type Gas Dynamic Trap or GDT-facility [72-74] and the tight aspect ratio or Spherical Tokamaks [75,76]. The GDT neutron source will provide a suitable fusion neutron spectrum, an enlarged test volume of about 20 liters for high flux testing, but is effectively a line source with neutron gradients very similar to those in the test cell of IFMIF. Hence it needs also the Small Specimen Test Technology. In the reference version it allows a max. wall loading of 1,8 MW/m^2, too low to perform accelerated testing and in an advanced stage (High Power Neutron Source) about 4-5 MW/m^2 are envisaged. Testing of components will not be possible. The major problem of this proposal is the high potential risk for its verification. Extrapolations of more than an order of magnitude in important plasma parameters like the plasma density and electron temperature are necessary to bridge the gap in knowledge from the present status of plasma physics development to the target parameters of a powerful neutron source.

The Spherical Tokamaks could provide sufficient test volume and would in comparison to the conventional aspect ratio tokamaks consume less tritium. Accelerated testing with higher wall loadings would be difficult and needs again a high effort in developing the plasmaphysical and technical basis for the design of such machines.

In concluding the development of d-t based alternatives needs certainly a more extended timescale and involves much greater technological and physics uncertainties and risks than the IFMIF concept. They can therefore not substitute IFMIF as an appropriate neutron source for materials research.

6. The role of IFMIF and ITER for materials and component testing towards DEMO and a future fusion reactor

As mentioned before the main objective of materials development in **Phase III** is the confirmation of the DEMO-materials concept under fusion-specific irradiation conditions. Also the potential and possible endurance limits of other alternative materials up to DEMO- and even reactor relevant fluence levels have to be explored. This target can only be achieved if an approriate high flux, high-energetic neutron source like IFMIF is made available for materials research **in due time**. This can be deduced from Fig. 15. In this plot a time schedule is shown which is based on the assumption that IFMIF and ITER would start operation at the same time and that a DEMO-reactor would follow about 20 years later. Since a material selection for DEMO has already to be made during the prerunning CDA/EDA phase the actual time window to perform the necessary irradiation is very tight. To give an example, a one-through experiment up to 80 dpa (8 MWy/m^2) with IFMIF operating with a wall loading of 20 dpa/year ($2MW/m^2$) and a machine availability of 80% needs 5 years of irradiation time. If necessary post-irradiation tests are added and a possible reduction of the machine availability in the introductory phase are taken into account, just one go-through experiment is available at the beginning of the CDA(EDA-phase for DEMO.

In parallel in the high-flux region of IFMIF which is limited in volume capacity (see Chapter 5), the potential of the reference and alternative materials in high-fluence experiments can be tested. With an annual displacement rate of 40-50 dpa reactor-relevant fluence levels of 150-200 dpa can be achieved in an acceptable time of 3-4 years. This exploratory work is very important since it can guide the final selection of the most promising materials which afterwards can be fully tested in DEMO and have the potential to be used in prototypic and commercial fusion reactors.

The role of ITER and IFMIF for DEMO

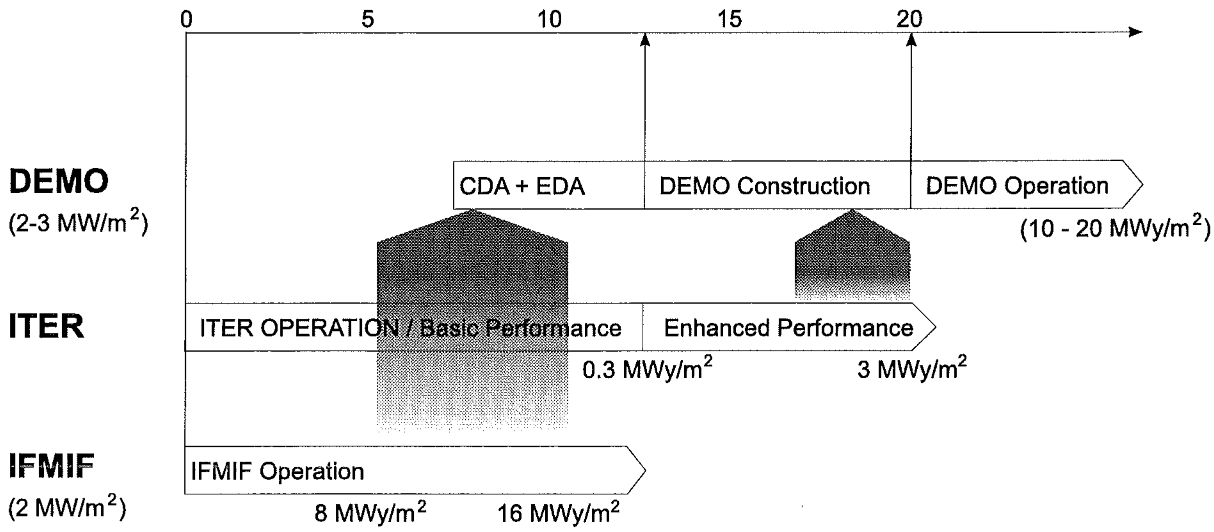

Fig. 15

A possible input of IFMIF and ITER for the development of DEMO

In concluding, the well-timed introduction of IFMIF is a necessary step to execute a straight forward materials development programme for DEMO- and fusion reactor targets.

The role of ITER in the materials and breeding blanket component development is a different one. In Fig. 15 the gain of experience, scaled in MWy/m^2 during the operation based on initial planning – is indicated. It shows that due to a different objective for an ITER in the basic performance phase relevant answers on materials performance under DEMO-conditions cannot be expected. If, however, in an Enhanced Performance Phase a final target of 3 MWy/m^2 can be realized – as initially planned – ITER could provide a substantial test bed for concept verification of alternative breeding blanket concepts including materials.

In summary, with IFMIF the necessary informations on material behaviour up to DEMO-fluence levels can be provided. Informations on the integral component behaviour of breeding blankets, which includes a variety of different materials and complex loading conditions (stress variations, temperature and flux gradients) can be delivered in an Extended Performance Phase in ITER. This gives a reliable basis for the construction of a DEMO reactor, which during its operation will be an ideal test bed for all reactor-relevant components.

7. Summary and conclusions

a) The strategy for the development of structural materials depends very much on design and concepts for first wall and breeding blanket components. One of the important performance targets is the integrated wall loading to be expected. Whereas for the European Test Blanket Modules for DEMO 70 dpa are envisaged, which is well in accordance with a range of 80-100 dpa in other programmes, the goals for prototype or commercial fusion reactors are less defined. A reasonable intermediate target lies in the range of 150 dpa.

b) Structural alloys for combined first wall/breeding blankets are also dependent upon the appropriate choice of breeding-, coolant- and neutron-multiplying materials. An assessment of different combinations leads to four major categories with three groups of structural materials: Ferritic-martensitic steels, vanadium alloys and ceramic composites of type SiC/SiC.

c) Some of the important materials properties like the power density capability - a combination of strength and thermophysical properties-, the creep rupture strength and the radiological properties have been compared and the general experience with these materials in nuclear fission environment has been assessed. This leads to a general judgement of their potential and the identification of major issues which have to be further investigated.

d) For the ferritic-martensitic steels a time-driven and detailed schedule for the R&D and a full qualification up to DEMO-relevant fluences has been developed. It is closely connected to the schedule for the development of the Test Blanket Modules for DEMO. It describes the major activities and development phases and gives milestones for the next decade.

e) A more selective strategy is proposed for the development of alternatives like vanadium alloys and ceramic composites of SiC/SiC-type. In a first phase R&D work should be concentrated on identified high-risk issues, whereas a comprehensive qualification programme should be started after elimination of possible knock-out factors.

   For vanadium alloys the development of coatings to prevent MHD effects and to reduce incompatibility with coolants and breeding materials should have priority. The stability of such coatings under the combined influence of stress and irradiation is of equal importance.

Ceramic SiC/SiC composites – though promising a high potential – need with priority research on fundamental radiation damage (e.g. displacement reactions) and studies on the effects of irradiation on thermal conductivity, dimensional stability and embrittlement. In addition some technical issues like the lack of hermeticity, joining- and welding techniques and the need for the development of design rules for inherently brittle materials are of equal importance.

f) The important role of existing irradiation facilities (mainly fission reactors) for the next step of materials development has been described, and necessary extensions of experimentation rigs in high-flux reactors to achieve DEMO-relevant-fluences have been stressed. The availability of such facilities for experiments in the next decade is mandatory. The limits for the transfer of results from such simulation experiments in fission reactors to fusion-relevant conditions have also been addressed.

g) The importance of a fusion-relevant, high-energetic and high-flux neutron source for the development and qualification of materials for DEMO and reactor-relevant fluence conditions has been shown. Such a facility would also give a possibility to validate data generated in fission reactor and accelerator irradiations. The materials community believes that an accelerator-driven d-Li-neutron source IFMIF can - from its technical capabilities - fulfil the users requirements and can provide in addition a useful test bed for material screening and selection up to reactor-relevant wall loadings in a reasonable time. A prerequisite is, however, that a Small Specimen Test Technology is developed and approved in parallel. IFMIF is presently the only option which can be realized in due time.

h) Though IFMIF can provide comprehensive data on the irradiation performance of structural- and other materials, the engineering testing of complete modules or components is not possible due to the restricted irradiation volume. However, in combination with ITER and DEMO, where a concept verification of different breeding blankets in ITER and afterwards full scale performance and reliability tests in DEMO could be executed, a straight forward strategy to develop materials and components for a fusion reactor is available.

i) The execution of the materials development programme and the installation of appropriate irradiation/test facilities enforces a close international collaboration, which is for the materials and nuclear technology areas at present promoted by the International Energy Agency.

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