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THE ROYAL SOCIETY

The development of structural materials for fusion reactors

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Structural materials for first-wall and breeding-blanket components will be exposed to 14 MeV neutrons, plasma particles and electromagnetic radiation. In magnetically confined systems, the operation mode will be quasi-continuous. Typical operation conditions for next-step devices and demonstration plants will be described. The selection of suitable structural materials is based on conventional properties, their resistance to radiation-induced damage phenomena and the additional requirement of low neutron-induced radioactivity. Presently, low-activating ferritic-martensitic steels, vanadium alloys and ceramic composites are investigated as promising candidates. In the paper the present status of knowledge is reviewed and the critical issues for the different alternatives are elaborated. The necessity of an appropriate neutron source for the testing and qualification of the materials under fusion-specific conditions is also stressed.

Keywords: structural materials; radiation-damage parameters; fusion reactor; low-activation materials; intense neutron source; steels, vanadium alloys, ceramic composites

1. Introduction

Materials for the first wall, limiters, divertors and breeding-blanket components are 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 like the Cottrell Blue Ribbon Panel (Cottrell *et al.* 1983) and the Amelinckx Senior Advisory Committee (Amelinckx *et al.* 1986). In parallel to the expected successful demonstration of plasma operation under reactor-typical conditions in the International Thermonuclear Experimental Reactor (ITER), the realization of such components like the first-wall/breeding-blankets and divertors will have priority and needs an extended research and development programme. Here the qualification of structural materials for a highly efficient and safe operation of such components is mandatory. Their behaviour determines both the economic competitiveness and the environmental attractiveness of fusion reactors.

In this paper, the typical operational conditions for structural materials to be used in first-wall/breeding-blanket components and the general targets for the development of magnetically confined fusion reactors and breeding-blanket components are described. Different combinations of structural-, breeding/cooling- and neutron-multiplying materials have been proposed. This is followed by a survey on available structural materials used in energy technology and the existing knowledge about

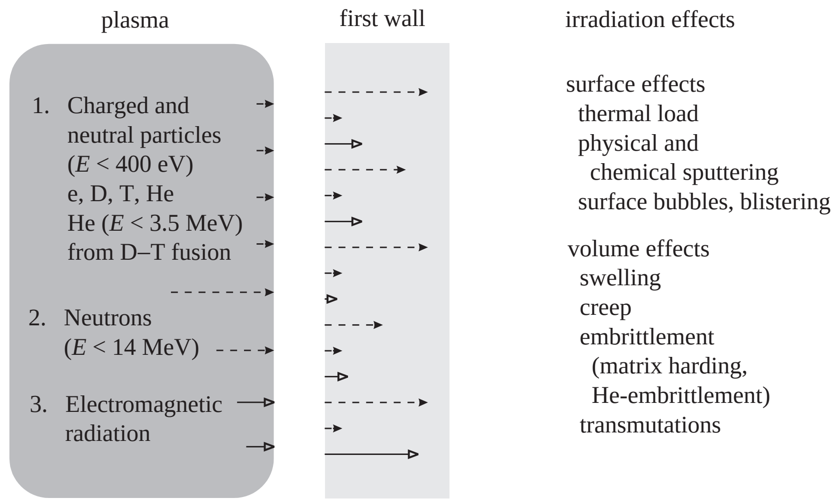

Figure 1. Schematic view of the first-wall loading by plasma particles, electromagnetic radiation and neutrons, and induced radiation-damage effects.

their behaviour in a nuclear environment. A comparison of some important parameters on which a selection of the most appropriate material can be based upon is made. This will include thermophysical and creep-rupture properties, and radiation-damage phenomena. Since structural materials contribute a major part to the radioactivity inventory of fusion reactors, their optimization towards reduced or low activation is an additional and important development target. The paper will conclude with a comparison of the most critical issues for the different material alternatives.

2. Fusion-specific operational conditions and performance goals

Material development is guided by the design and the expected performance goals of components and reactors. Therefore, after an introductory characterization of the fusion-specific interaction of plasma particles and neutrons with materials the general performance goals of demonstration and commercial fusion reactors based on magnetic confinement, and the different breeding-blanket options are shortly summarized.

(a) *Interaction of plasma particles and neutrons with the first wall*

The operational conditions for structural materials are essentially dependent on the interaction with high-energy 14 MeV neutrons, low-energy plasma particles and electromagnetic radiation. In addition, the temperature range and the mode of operation are of equal importance for the lifetime of components. In figure 1, the interaction between impinging particles or electromagnetic radiation with the first wall is shown schematically. Low-energy charged and neutral plasma particles have a limited penetration and lead to well-known phenomena like physical and chemical sputtering or erosion and eventually to near-surface bubble formation and blistering (Post & Behrisch 1986). The resulting implications on material erosion, plasma instability, etc., are not a topic of this paper and are often discussed in the context with the diver-

Table 1. *General performance goals for fusion devices*

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

	ITER	DEMO	REACTOR
fusion power	0.5–1 GW	2–4 GW	3–4 GW
neutron wall loading (first wall)	0.5–1 MW $\mathrm{m^{-2}}$	2–3 MW $\mathrm{m^{-2}}$	2–3 MW $\mathrm{m^{-2}}$
integrated wall load (first wall)	0.3–1 MWy $\mathrm{m^{-2}}$	3–8 MWy $\mathrm{m^{-2}}$	10–15 MWy $\mathrm{m^{-2}}$
	3–10 dpa	30–80 dpa	100–150 dpa
operational mode	pulsed (1000 s)	quasi-continuous	
	$< 5 \times 10^4$ cycles		
plant lifetime			∼30 FPy
net plant efficiency			∼30%

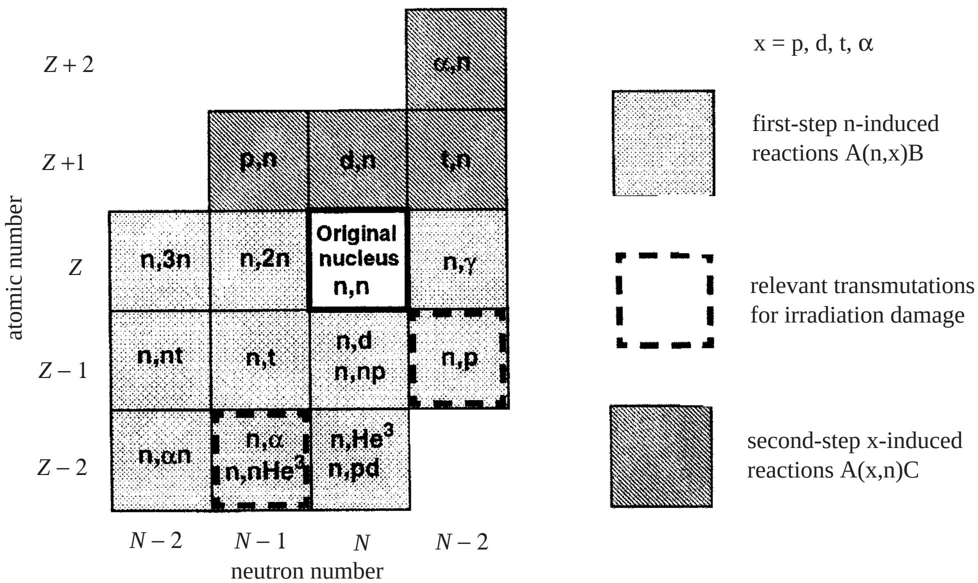

Figure 2. Kinematically possible nuclear reactions of high-energy neutrons with matter ($E_{\mathrm{n}} \leqslant 15$ MeV).

tor development. However, the high thermal surface heat load and induced secondary stresses are very important parameters for the lifetime of first-wall components.

14 MeV neutrons have a long-range penetration and interact either via elastic collisions or inelastic events with the atoms. In both cases the displacement of single atoms from their lattice sites is one of the important primary reactions. The quantitative number of such displacements per lattice atom (dpa) is for a given neutron flux or fluence an agreed measure for the expected radiation damage. It has replaced the usual figures of neutron flux by the term dpa $\mathrm{s^{-1}}$ and the neutron fluence by dpa, which can be calculated by using a standardized Norgett–Torrens–Robinson model (Norgett *et al.* 1975). In table 1 typical defect production rates (dpa $\mathrm{s^{-1}}$)

and lifetime integrated values (dpa) for different fusion devices are given. Their influence on radiation-damage phenomena will be discussed later. In comparison with fission neutrons, high-energy 14 MeV neutrons generally react more often with atoms through inelastic, i.e. nuclear events, since more channels for such reactions are opened by surpassing critical threshold energies. In figure 2 kinematically possible nuclear reactions of high-energy neutrons with atoms are given. They include not only neutron-induced or first-step reactions, but also those induced by charged particles like protons, deuterons, tritons and α-particles, which themselves have been produced through neutron-induced primary events (Cierjacks *et al.* 1990). This is important for the correct calculation of damage parameters including irradiation-induced radioactivity. Transmutation reactions which can lead to the formation of helium and hydrogen and cause specific radiation-damage phenomena, are typically increased by at least one order of magnitude when compared to fission neutrons. The increase of inelastic events also contributes to the overall displacement damage by high-energy recoils and increases the energy of the primary-knocked-on atoms (PKA). A possible consequence is a further shift from a preferential formation of single defects (displacements) and cascades towards subcascades, which could enhance radiation hardening.

In conclusion, high-energy fusion neutrons lead to an enhancement of nuclear reactions, which increases the displacement damage rates and the generation of transmutation products and shifts the spectra of the primary knocked-on atoms to higher energies. These physical parameters play a major role for the development of radiation-damage phenomena and the irradiation-induced radioactivity.

(*b*) *Performance goals and typical damage parameters for fusion devices*

The development of structural materials for fusion application follows the generally adopted strategy that the next steps following the construction and operation of ITER are a demonstration reactor (DEMO) and afterwards a prototype or commercial fusion reactor (CFPR).

The ITER facility is, with regard to materials issues, characterized by a moderate neutron wall loading, a low temperature and a strongly pulsed operational mode. It is expected that these moderate demands, even further reduced in recently revised ITER proposals, and compiled in table 1, can be fulfilled by use of an austenitic stainless steel of type 316 LN-IG. This material has been successfully applied in conventional fission reactors and the research and development activities still necessary are performed within the international ITER community under the auspices of the International Atomic Energy Organisation (IAEO). The results of these investigations are periodically reported at international fusion materials conferences (Kinoshito & Muroga 1998).

The long-term development towards a DEMO or a commercial fusion 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. Finally, the materials should be of 'low-activation'-type to maintain one of the most attractive safety features of fusion.

In table 1 a range of performance goals presented at different occasions is compiled for DEMO and commercial power reactors (Proust *et al.* 1993; Ioshi 1998; Ryabev

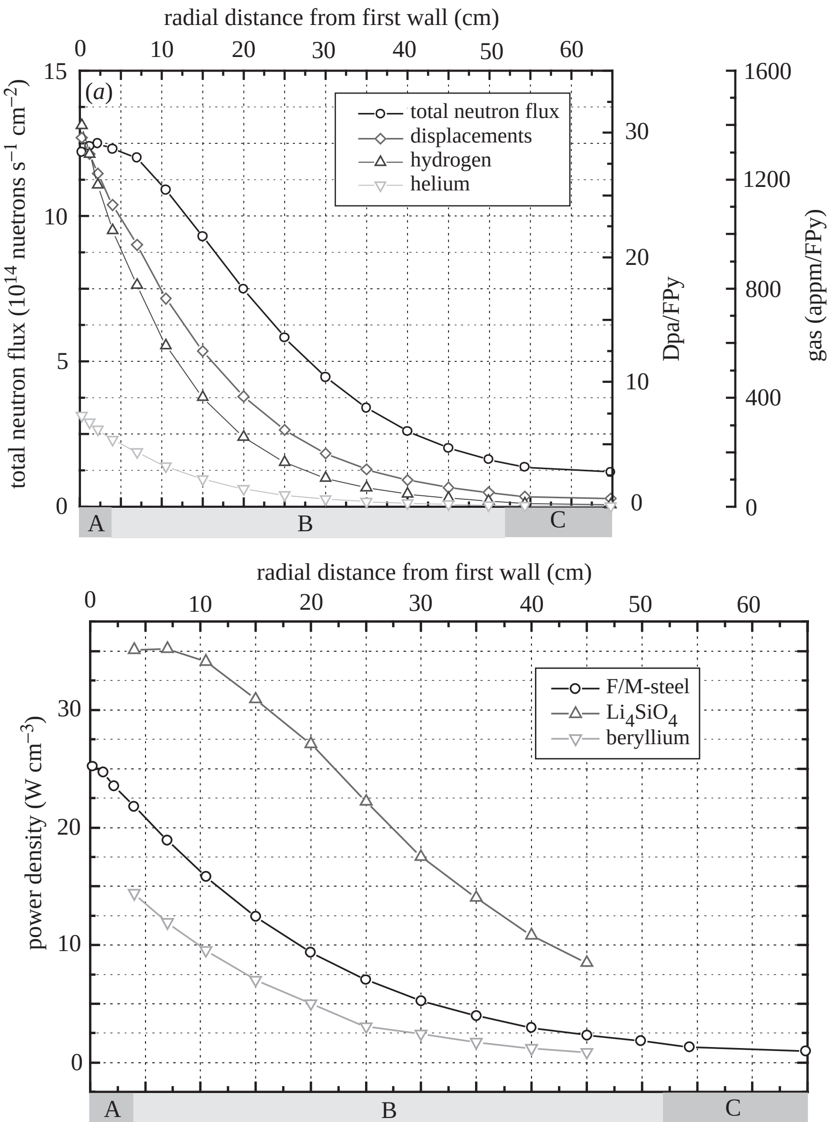

Figure 3. (*a*) Neutron flux and damage parameters for Fe in the helium-cooled pebble-bed outboard breeding blanket in dependence of radial distance from first wall. (*b*) Radial dependence of power density in structural and ceramic breeder material and neutron multiplier in the helium-cooled pebble-bed-outboard breeding blanket. A, first wall; B, breeding material/neutron multiplier/structural material; C, structural material.

& Solonin 1998). Key parameters regarding the radiation exposure are the expected neutron wall load in MW m^{-2}, which determines the neutron flux, the surface and volume power density and relevant radiation-damage parameters.

The data in table 1 refer to the first-wall position and hence present the maximum exposure. In order to give an idea about the radial variation of such damage parameters through a real component like the European helium-cooled pebble-bed outboard breeding blanket (Proust *et al.* 1993), the neutron flux, the yearly displacement rate of atoms and the production of gaseous transmutation products H and He are given in figure 3*a* for a neutron wall loading of 3.5 MW m^{-2}. The data are based on detailed Monte Carlo neutron transport calculations (Fischer & Norajitra 1998). In this blanket module a ferritic-martensitic steel with reduced long-term activation, the ceramic breeder material Li_4SiO_4 and the neutron multiplier beryllium have been chosen. 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 versus 0.7 dpa); and (ii) the decrease of the He/dpa and H/dpa ratios in Fe (11.1 versus 2.75 atomic parts per million (appm) He/dpa and 45 versus 10.3 appm H/dpa). For an envisaged lifetime of 20 000 h about 70 dpa, 780 appm He and 3150 appm H will be generated in this material in the first-wall position. They present a typical development target for a DEMO breeding blanket.

Besides the radiation-damage parameters, the volume power density in W m^{-3}, which is caused by the inelastic interaction of materials with neutrons and plotted in figure 3*b*, is also a limiting parameter. It shows that energy absorption is higher in the breeding material and therefore the allowable maximum temperature for the ceramic breeder material Li_4SiO_4 is reached, before a critical temperature, or strength level, in the structural material is attained. This example shows that in order to determine maximum allowable neutron wall loadings the thermophysical properties of all materials, not only the structural materials, have to be taken into account.

An equally important parameter in table 1 is the mode of reactor operation that is often described to be 'quasi-stationary' or quasi-continuous in magnetically confined fusion reactors. This terminology, for example used for burn pulses in the range of 10^4 s or so, is not 'steady state' regarding the material behaviour, but will still cause a fatigue-dominated operational regime.

In inertially confined fusion reactors, where energy is supplied by the repeated ignition of deuterium–tritium (D–T)-pellets the pulsed operation mode is an intrinsic feature. Typical burn times of nanoseconds with a repetition frequency of several Hz are reported in different reactor studies (Kessler *et al.* 1981; Badger *et al.* 1990). For comparable time-averaged neutron wall loads of several MW m^{-2} the instantaneous values are therefore many orders of magnitude higher than in magnetically confined systems. For instance, whereas in the latter case the typical displacement damage rates are of the order of 10^{-6} dpa s^{-1} at the first-wall position, they increase to 1–10 dpa s^{-1} in the same position for inertial confinement reactors. With increasing radial distance neutron flux intensity decreases and pulse length increases in the breeding-blanket- and reflector/shield positions, but the conditions are always very different from those in a magnetically confined reactor. This raises many questions regarding the effect of high neutron fluxes and pulsed time structure on the development of radiation-damage phenomena like hardening, swelling, irradiation creep, etc. Another important issue is the impulse load on front structural parts and the mechanical response of materials.

In comparison to the case of magnetically confined systems there does not exist any experimental facility which could simulate such high displacement and transmutation rates in an adequate time structure so that first estimates on the material behaviour have to rely on theoretical modelling. A more detailed description of irradiation conditions, materials and concepts for components and reactors for inertial fusion is given in Hogan (1995).

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Table 2. *Major breeding-blanket concepts*

	coolant	breeding material	structural material	neutron multiplier
He/LiCe/FS/Be[a]	He	LiCe	F/M-steel	Be
He/LiCe/SiC–SiC/Be	He	LiCe	ceramic composite SiC–SiC	Be
Li/V	Li	Li	vanadium alloy	Li
H_2O/Pb-Li/FS[b]	H_2O	Pb–Li	F/M-steel	Pb–Li

	operation conditions temperature	pressure
He/LiCe/FS/Be[a]	250–550 °C	5–20 MPa (8 MPa)
He/LiCe/SiC–SiC/Be	450–950 °C	5–20 MPa
Li/V	350–750 °C	∼1 MPa
H_2O/Pb-Li/FS[b]	250–550 °C	12–15 MPa (15.5 MPa)

[a]Helium-cooled pebble-bed blanket (HCPB)/EU; pressure data of both concepts in (brackets).
[b]Water-cooled lithium-lead blanket (WCLL)/EU; pressure data of both concepts in (brackets).
Lithium ceramic (LiCe) breeder materials: Li_2O, Li_4SiO_4, Li_2ZrO_3 or Li_2TiO_3.

(*c*) *Breeding-blanket options and materials*

Since it is assumed that a DEMO will be the major step towards a prototype or a commercial fusion power reactor, all reactor-relevant functions like the breeding of tritium or the successful operation of a divertor have to be demonstrated and successfully tested in such a facility. Components like a breeding blanket should be built from materials that possess the potential for high performance, so that the number of exchanges should be minimized. For the selection of appropriate structural alloys, not only conventional material data like thermophysical and strength properties or their performance under irradiation is important, but also their 'compatibility' with other materials like breeding media, neutron multipliers, etc. In this context, 'compatibility' means not only corrosion phenomena but also general interactions between the different materials be they of mechanical, thermal, chemical or irradiation-induced nature.

Design studies have shown that for integrated first-wall/breeding-blankets only a limited number of combinations of structural materials with breeding and cooling media exist. 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. Solid ceramic breeder materials Li_2O, Li_4SiO_4, Li_2ZrO_3 and Li_2TiO_3 are under discussion, while liquid breeder materials are lithium or lithium–lead. Three major structural materials, ferrritic-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 concepts are compiled in table 2. Here the important parameters are the proposed temperature ranges and system pressures. The upper temperature is limited by high-temperature creep strength or corrosion resistance, whereas the lower temperature is given by the coolant inlet temperature or is limited by irradiation hardening.

Two of the proposed combinations are part of the above-mentioned European Blanket Project (Proust *et al.* 1993). They use a low-activating ferritic-martensitic steel as structural material in combination with either a solid lithium ceramic (LiCe) as breeding, beryllium as neutron multiplier and helium as cooling medium or liquid Pb–Li as breeding medium in a water-cooled lead lithium blanket. From the point of view of mechanical properties, the temperature window is estimated for both blankets to be between 250 and 550 °C, where the creep rupture properties determine the upper temperature and the irradiation-induced ductile-to-brittle transition temperature determines the lower temperature. Further technical targets of these designs are an average neutron wall loading of 2.2 MW m^{-2} at the inboard and 3.5 MW m^{-2} at the outboard blanket, and a lifetime of 20 000 h, which integrates to an accumulated wall loading of 5 and 8 MWy m^{-2}, respectively.

To summarize, there exist four major categories for combined first-wall/breeding-blanket components which take into account the best combinations of structural-, breeding/cooling and other materials. The proposed operational parameters are in most cases based on preliminary design studies and are used here to describe the targets for a focused development of materials.

3. Materials requirements and major selection criteria

There are numerous requirements which have to be fulfilled by structural materials to be used in first-wall- and breeding-blanket components. Their appropriate selection is a very complex process and in the following pages some of the key criteria will be discussed in more detail.

In a first step, for a given component design, the selection has to be based on conventional properties like thermophysical, mechanical and corrosion and compatibility data. For example, depending on the applied heat load and the envisaged operational stress–temperature regimes, the tensile, creep-rupture, creep/fatigue and fracture toughness data are important. Also, corrosion and compatibility including synergistic effects have to be taken into account, to explore the best combinations between cooling medium, breeding, neutron multiplier and structural materials. The selected materials must also have technical maturity, i.e. qualified fabrication and welding technology and a general industrial experience have to be available. Under such conventional conditions many available commercial material groups like the family of austenitic and ferritic steels but also nickel-superalloys could be taken into consideration. The selection of an austenitic stainless steel as structural material for ITER is an example. But also new materials like vanadium alloys and ceramic composites have been proposed in breeding-blanket designs of table 2 for specific temperature and pressure ranges. To illustrate how such conventional properties can influence the materials choice, a comparison of the thermal stress response and the creep-rupture properties—two very important properties that determine the maximum allowable wall loading and the upper limit of operational temperature—will be given in § 3 *a*.

A second important selection criterion is the material behaviour under neutron irradiation since the material has to withstand high integrated wall loadings as mentioned in table 1. Since no experience under fusion-identical irradiation conditions is presently available, the selection process has to be based on existing experience in fission reactor technology and on an assessment how these data can be correctly transferred to fusion conditions. The main source of knowledge stems from struc-

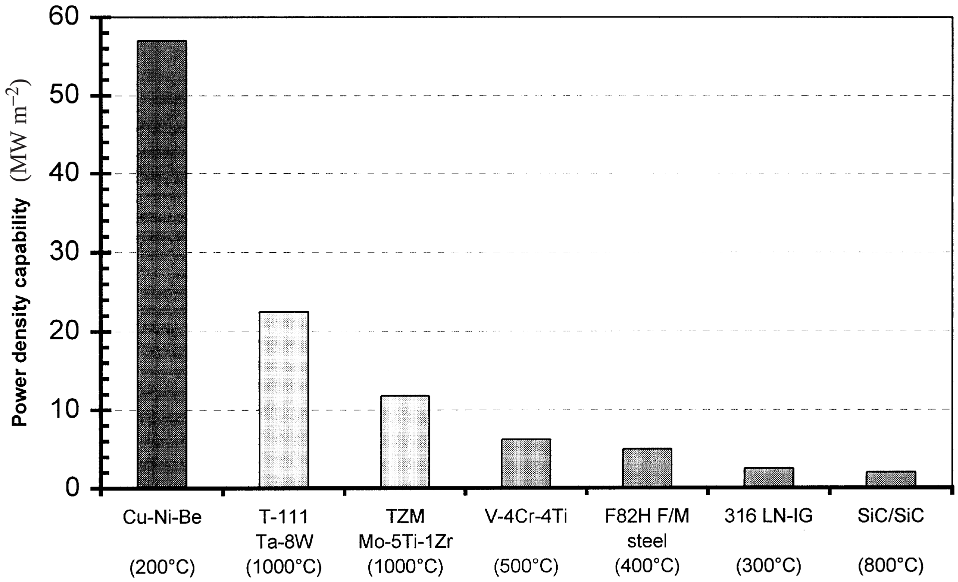

Figure 4. Power density capability of structural and heat-sink materials (Zinkle & Ghoniem 1998). ($\dot{Q} = R_M\lambda(1-\nu)/\alpha E$, where R_M is the tensile strength, λ is the thermal conductivity, α is the thermal expansion coefficient, and E is the elastic modulus.)

tural materials in light water and fast breeder reactors, which will be summarized and discussed in § 3 *b*. This summary allows us at least to reveal the major critical issues to be expected for the different material groups under irradiation.

A third and unique selection criterion for fusion materials technology is the irradiation-induced activation. This is understandable, since the structural materials will provide a major source of radioactivity in a fusion reactor (Raeder *et al.* 1995). Therefore, in § 3 *c* an evaluation of the major alloy groups under investigation with respect to their radiological properties will be elaborated.

All the considerations lead to a concentration of the development to few main lines of materials which at present are followed in the international fusion materials community.

(*a*) *Thermal stress response and creep rupture properties*

The thermal stress response or the power density capability determines the maximum allowable heat load and is therefore an important selection criterion. This parameter includes thermophysical and mechanical properties, and the Poisson ratio ν, and gives the allowable neutron wall loading in MW m^{-2} for a 1 mm thick first wall. In figure 4, such data are compiled for several materials; an austenitic steel 316 LN-IG to be used in ITER, F82H, a reduced activation ferritic-martensitic steel, a vanadium-based alloy of type V–4Cr–4Ti and the ceramic composite material SiC–SiC. The comparison of metallic structural alloys shows that vanadium alloys have the highest potential followed by ferritic-martensitic steels, whereas austenitic steels and Ni-alloys (not shown in this graph) are inferior. Surprisingly, the fibre-reinforced ceramic composites SiC–SiC have the lowest power density capability, though the

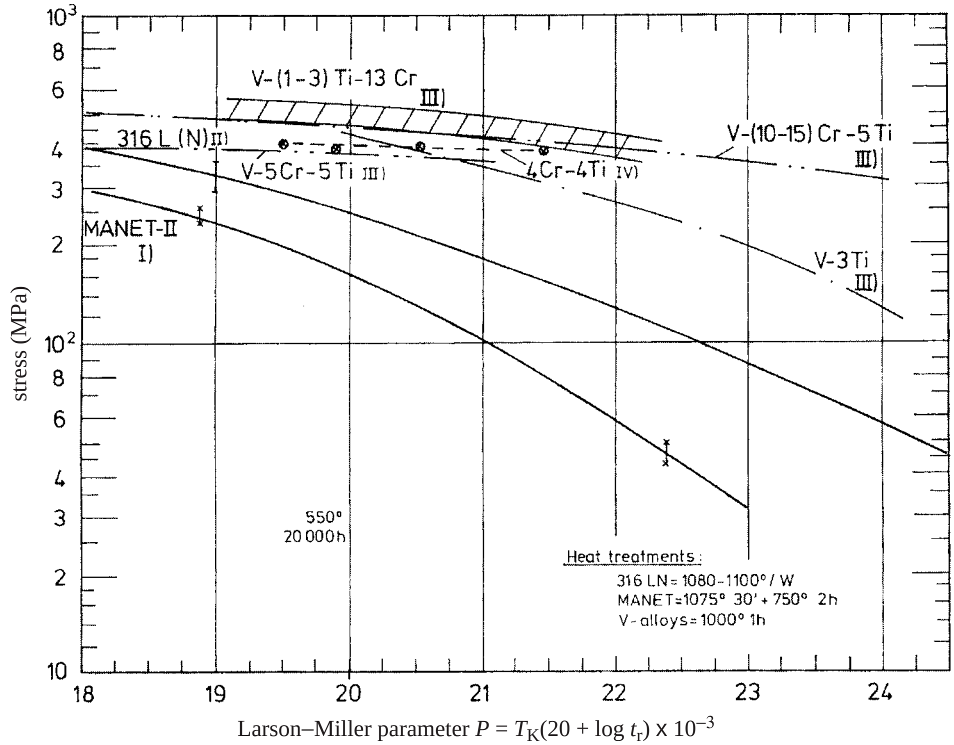

Figure 5. Creep-rupture strength of structural materials. (The Larson–Miller parameter P can be calculated for a given set of temperatures (T_K (K)) and rupture times (t_R (h)) with the indicated formula. It compensates temperature versus time).

monolithic SiC ceramic has good thermal conductivity. The reason for this unexpected behaviour is a strong degradation of this property under irradiation (see § 3 *c*). In more recent design studies refractory alloys on the basis of Ta and Mo have also been proposed as attractive materials in order to achieve very high wall loadings, independent of the question whether or not they also possess advantages in other properties (Zinkle & Ghoniem 1998). A typical heat-sink material Cu–Ni–Be to be used in divertors is also included for comparison.

Regarding the creep-rupture strength of structural alloys, figure 5 gives in a Larson–Miller plot a comparison of the creep-rupture strength for relevant structural materials with MANET II, a conventional 9–12%Cr ferritic-martensitic steel, 316LN an austenitic alloy and a range of data for different binary or ternary vanadium alloys. As can be deduced from this plot, vanadium alloys have by far the best potential for high-temperature application (Böhm & Schirra 1968), provided that corrosion and compatibility can be managed (Borgstedt & Konys 1998). To give an example, for an envisaged lifetime of 20 000 h at 550 °C, which corresponds to a Larson–Miller parameter P of 20 for the above-mentioned helium-cooled pebble-bed-breeding blanket, the allowable creep rupture stress level is 160 MPa for MANET, 230 MPa for 316LN and more than 400 MPa for the V–Ti–Cr alloy. Correspondingly, a stress level of 160 MPa would allow a temperature level of about 750 °C for vanadium alloys at the same rupture time of 20 000 h.

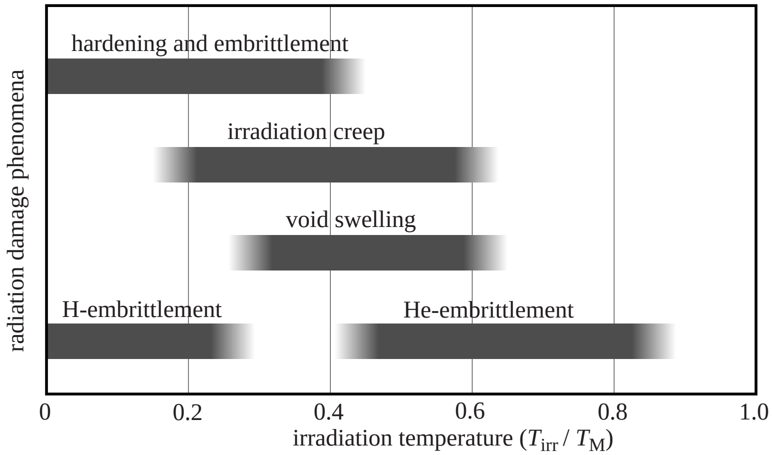

Figure 6. Schematic view of relevant radiation-damage phenomena in metallic structural materials.

(*b*) *Existing experience with nuclear materials*

The interaction of high-energy neutrons with metallic structural materials leads to a number of radiation-damage phenomena (Cierjacks *et al.* 1990) which are schematically plotted in figure 6 as a function of the irradiation temperature $T_{\mathrm{irr}}/T_{\mathrm{M}}$, normalized to the melting temperature T_{M} of the material. There are three major temperature regions in which a degradation of material properties can occur.

(i) At low-temperature radiation, hardening occurs which is connected with a reduction in work hardenability and ductility (embrittlement) and with fracture toughness degradation. This effect already occurs after low fluence irradiations with an obvious tendency for saturation and is specifically important for materials that per se show already in the unirradiated state a so-called ductile-to-brittle transition temperature. This important temperature (DBTT) essentially provides a lower limit to the operating temperature of alloys in the power plant. Above this temperature any dangerously high local stresses in any component can be accommodated by ductile flow, and the plant will remain safe. Below it, there is the possibility of brittle fracture, with catastrophic failure of the component. Alloys are generally chosen so that their operating temperatures will be well above the ductile-to-brittle transition temperature. However, one of the effects of irradiation is that the transition temperature is raised during the irradiation, so that previously safe temperatures can no longer be permitted.

(ii) At an intermediate temperature region where, due to agglomeration of single defects, voids and planar defect clusters form. They cause swelling and enhanced irradiation creep, which leads to dimensional instabilities and for the realistic case of flux/temperature gradients to distortions of components. This is generally a high-fluence phenomenon starting between 10 and more than 100 dpa dependent on the material and will limit the maximum achievable lifetime of a component.

(iii) High-temperature embrittlement caused by formation of helium bubbles at grain boundaries. This latter effect is dependent on the generation rate of He via nuclear n,α-reactions, and can come into effect in some materials already after few appm He. With this very simplified picture in mind, the existing knowledge about material performance under neutron irradiation in fission reactors is discussed. It comprises mainly conventional and optimized austenitic and ferritic steels and Ni-base alloys. In addition on an international level relevant data on vanadium-based alloys and dispersion-strengthened ferritic steels are available. Radiation experience with ceramic composites is very limited.

Austenitic stainless steels are used as cladding and wrapper materials of fuel elements in fast breeder reactors and as core structural materials in light-water reactors. They have been exposed in fast breeder reactors in a large quantity up to fluence levels in the range of 120 dpa and have reached maximum values of 150 dpa in combination with about 20% fuel burn-ups. The performance-limiting radiation phenomenon in these alloys is radiation-induced swelling. Due to material modifications by adding swelling-reducing elements like Ti, Si, and P and by choosing the appropriate material pretreatment, the onset of swelling could be strongly retarded (Bergmann *et al.* 1991). However, since this improvement is obviously only a transient effect, the potential of this material group for high-fluence application is limited. Another important restriction is their sensitivity to helium embrittlement.

High-nickel austenitic steels and Ni-based alloys, mostly strengthened by γ'-precipitates and superior in high-temperature creep strength have also been tested extensively as cladding materials for fuel pins in fast breeder reactors. 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 (Bergmann *et al.* 1991). Ductility exhaustion due to high-temperature helium-embrittlement is the major concern for these materials especially under the aspect that the n,α-cross-sections will strongly increase with neutron energy. A substantial decrease of ductility has recently also been reported at low temperature for the nickel alloy Inconel 718 after 800 MeV proton irradiation and has been ascribed to radiation-induced segregation phenomena in combination with a change in fracture mode (Carsughi *et al.* 1998).

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 fast breeder reactor fuel elements in the temperature range of 360–550 and 600 °C, respectively. 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 (Bergmann *et al.* 1991). At typical fast breeder reactor conditions there were also no indications of a degradation of the 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 ductile-to-brittle transition towards higher temperature occurred (Rieth *et al.* 1995). Though it could be shown that 9%Cr-alloys were less sensitive to this degradation than 12%Cr steels (Klueh & Alexander 1996), this phenomenon is one of the critical issues for the application of ferritic-martensitic steels and will be discussed later in connection with newly developed alloys.

Vanadium alloys (mainly based on V–Ti–Si or V–Cr–Ti compositions) were not applied in fission reactor technology, since attempts to use them as cladding mate-

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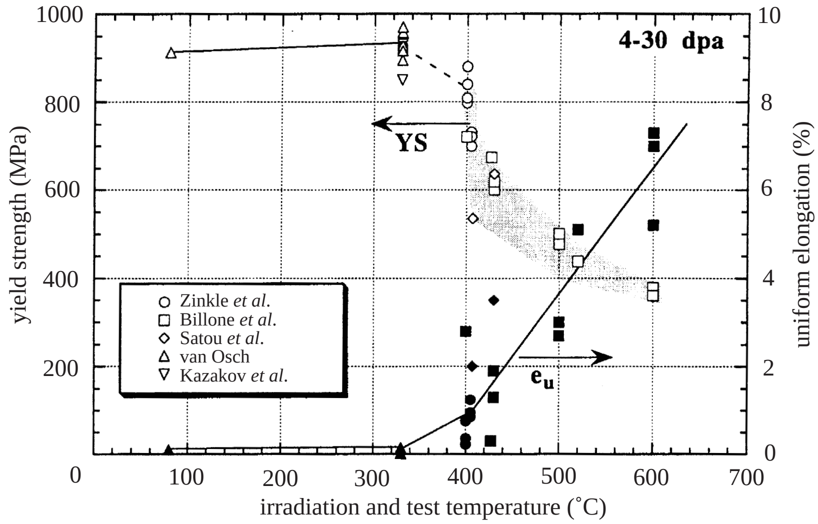

Figure 7. Tensile properties of irradiated V–(4–5%)Cr–(4–5%)Ti alloys (Snead *et al.* 1997).

rial in combination with uranium oxide fuels in fast breeder reactors failed due to thermodynamic incompatibility. However, since very early investigations showed for some alloys a quite good resistance to irradiation-induced He-embrittlement up to 650 °C (Ehrlich & Böhm 1969), and more recently low swelling was reported for selected vanadium alloys under fast reactor irradiations (Matsui *et al.* 1996), the potential for high fluence application seems promising. One major point of concern is, as in the case of ferritic-martensitic steels and other refractory metals, the radiation hardening at low irradiation temperature in combination with a degradation of the impact and fracture toughness properties. Most recent data compiled in figure 7 (from Snead *et al.* 1997) 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 ductile-to-brittle transition in Charpy V tests.

Fibre-reinforced ceramic composites of type SiC–SiC have not yet been used in nuclear reactors and the irradiation experience is based on specific material irradiation tests in reactors and accelerators, mostly at relatively low fluence levels (from 20 to a maximum of 50 dpa). Some major concerns are, in comparison to metallic structural materials, a general lack of knowledge about fundamental damage processes in ceramic materials and the possible influence of high cross-sections for inelastic n,α-processes in Si. In this material 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. One important experimental observation is the strong reduction of thermal conductivity of a neutron-irradiated two-dimensional Nicalon CG–SiC composite in function of the displacement damage in figure 8. With such data in the range of 2 W mK^{-1}, far below the unirradiated and even irradiated values for monolithic SiC (15–20 W mK^{-1}), the very low power density capability shown in figure 4 is explained. Another observed effect in this compound material which consists of a β-SiC matrix, a SiC-fibre material with a different stochiometric composition and an interphase of graphite, is that its dimensional stability under irradiation is depen-

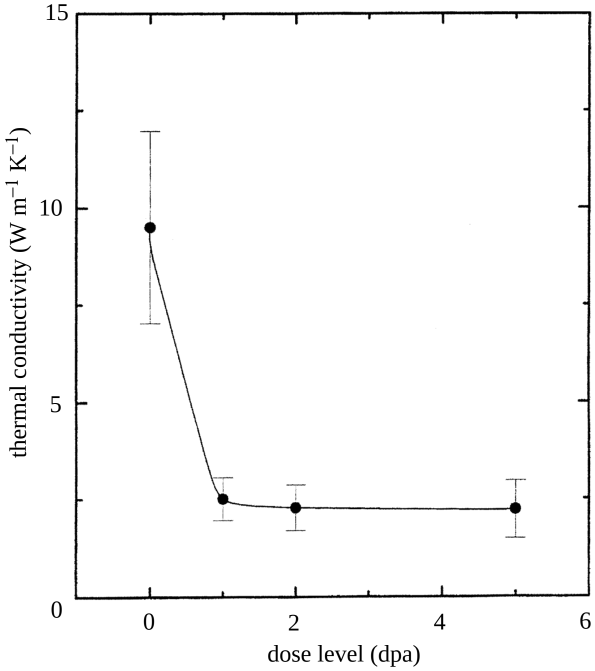

Figure 8. Thermal conductivity of a SiC composite in dependence of displacement damage (Smith *et al.* 1998).

dent on the relative swelling or densification of the different constituents. In the presently available materials, due to such a mismatch between matrix and fibres, a delamination has been observed. This leads to a reduction in strength and fracture toughness of not less than 20% (Jones *et al.* 1997; Zinkle & Snead 1999). It is, however, expected that with advanced fibre materials, an improved swelling resistance and hence reduced delamination effects can be expected.

In summary, the material performance under fission reactor and ion irradiations 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 the difference in important radiation-damage parameters like neutron energy, recoil spectra and transmutation reactions is significant and can have strong effects on the different radiation-damage phenomena.

(*c*) *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 (Raeder *et al.* 1995). As has been explained in § 2, the number of kinematically possible transmutation reactions by neutrons and neutron-produced loaded particles increases strongly when compared with conventional fission reactors. Therefore, the possibilities to develop materials with reduced or low radioactivity were investigated early on an international basis for many classes of structural

Table 3. *Criteria for low activating materials*

	relevant parameters	proposed limits
operation, maintenance, repair	contact dose rate	$< 10^4$ Gy h^{-1} after 24 h or $< 10^7$ Gy total
accidental scenarios[a]	activity	$< 10^{14}$ Bq (off site)
	dose rate	< 100 mSv (off site)
	afterheat/radiotoxicity	
recycling[b]	contact γ-dose rate	< 25 Sv h^{-1} after 100 years
	(hands-on-level)	(recycling)
	afterheat	< 10 mSv h^{-1} (remote recycling)
		< 1 W m^{-3}
waste management deposition[b]	contact γ-dose rate/ afterheat	
	low-level waste	< 2 mSv h^{-1}/< 1 W m^{-3}
	medium-level waste	< 20 mSv h^{-1}/< 10 W m^{-3}
	high-level waste	> 20 mSv h^{-1}/> 10 W m^{-3}

[a]For decay times < 0.1 year.
[b]For decay times of 100 years.

materials (Hopkins & Price 1984; Butterworth 1989; Harries *et al.* 1992; Piet *et al.* 1990; Proceedings of IEA-Workshop 1991).

This development can be viewed under two different aspects. One is coupled to the safety of fusion reactors, where the potential to disperse radioactivity in case of an accident has to be taken into account. The primary path for such an event is the volatilization 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 shutdown of a reactor. Materials with a low activation and low decay heat in a period of days or months after reactor shutdown are therefore preferable under this aspect. Maintenance and repair would also be facilitated by the use of such materials, though remote repair will always be necessary. The second aspect regards the 'long-term' activation after reactor shutdown. The level of long-term activation will determine the appropriate ways for waste disposal and material recycling. In table 3 important criteria for low-activation materials are compiled. Since it will be shown that none of the material groups under discussion has the lowest activation or decay heat over the complete decay time, the decision has to be made where to put the emphasis in the development. In the European long-term programme, priority has been focused on a reduced or low irradiation-induced long-term activation.

The first step for the development of low activating alloys with a fast decay behaviour is a thorough calculation of all radiological properties. For the calculation of such data like activity, γ-dose rate, nuclear decay heat or radiotoxicity, the FISPACT code, especially adapted to high-energy fusion neutrons, with the input of activation, and decay data from the European Activation System (EASY), is generally used in Europe (Forrest *et al.* 1988; Forrest & Kopecky 1991). An important supplement of this computational code was the inclusion of the above-mentioned sequential reactions in the FISPACT Code 4.0 and following versions (Cierjacks & Hino 1990). At present the FISPACT97/EAF97 is the updated version (Forrest & Sublet 1997).

Such calculations of radiological properties revealed that especially important alloying elements like Nb, Mo, Al and Co and a series of impurity elements like rare earth elements, Ag and others have an overwhelming negative influence on irradiation-induced activation, whereas especially vanadium and chromium and, with some restrictions, Ti would be beneficial. These results have led to the exclusion of Ni-alloys, though they are superior in high-temperature creep strength and have reasonably good irradiation behaviour, since a chemical modification is not possible to achieve low long-term activation. Commercial austenitic stainless steels are also not low-activation materials and attempts by substituting Ni through Mn and replacing Mo by other strengthening elements to achieve a reduced activation material were not very promising, so that a further development in this direction was also terminated (Harries *et al.* 1992). For ferritic-martensitic steels the possibility exists to achieve reduced or low long-term 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 with success in Europe, USA and Japan since the mid-1980s, partly in collaboration under the IEA Implementing Agreement/Research and Development of Fusion Reactor Materials (Ehrlich *et al.* 1994; Hishinuma *et al.* 1998; Daum *et al.* 1997). The modifications have led to a new series of alloys, the majority of which lie in the compositional range 7–10%CrWVTa and which have shown promising results.

Therefore, and from other reasons, the developmental work in the last years concentrated mainly on modified ferritic-martensitic steels like OPTIFER, F82H and EUROFER, on vanadium-based V–Ti–Cr-alloys and a ceramic composite of type SiC–SiC which per se have good radiological properties.

For the reduced activation ferritic-martensitic steel OPTIFER, the vanadium alloy V–4Cr–4Ti and the ceramic composite material SiC–SiC the γ-dose rates after an irradiation to 12.5 MWy m^{-2} (125 dpa in Fe) with a dose rate of 5 MW m^{-2} were calculated and plotted in figure 9. 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 in addition to the specified alloying elements also impurity or tramp elements, partly assumed and partly measured in these alloys. Compared with previously reported results, where only the alloying elements had been taken into account, the data differ strongly and reduce the predicted advantage of vanadium alloys regarding the long-term γ-dose rate by orders of magnitude. 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 (Murphy & Butterworth 1992; Ehrlich *et al.* 1996; Forty & Cook 1997; Rocco & Zucchetti 1992), a more rational view on the realistic possibilities of low-activation materials has been achieved in the materials community. Future activities have now to be concentrated on the technical possibilities to keep the level of unwanted impurities as low as possible for each material group. Hereby the potential for an improvement are especially good for V-alloys regarding the long-term activation and also attractive for SiC–SiC for an intermittent timescale between few days up to 10 years. The prospect to reduce unwanted tramp elements to values in the range of appm in technical fabrication processes are also promising, especially since similar approaches, but with other

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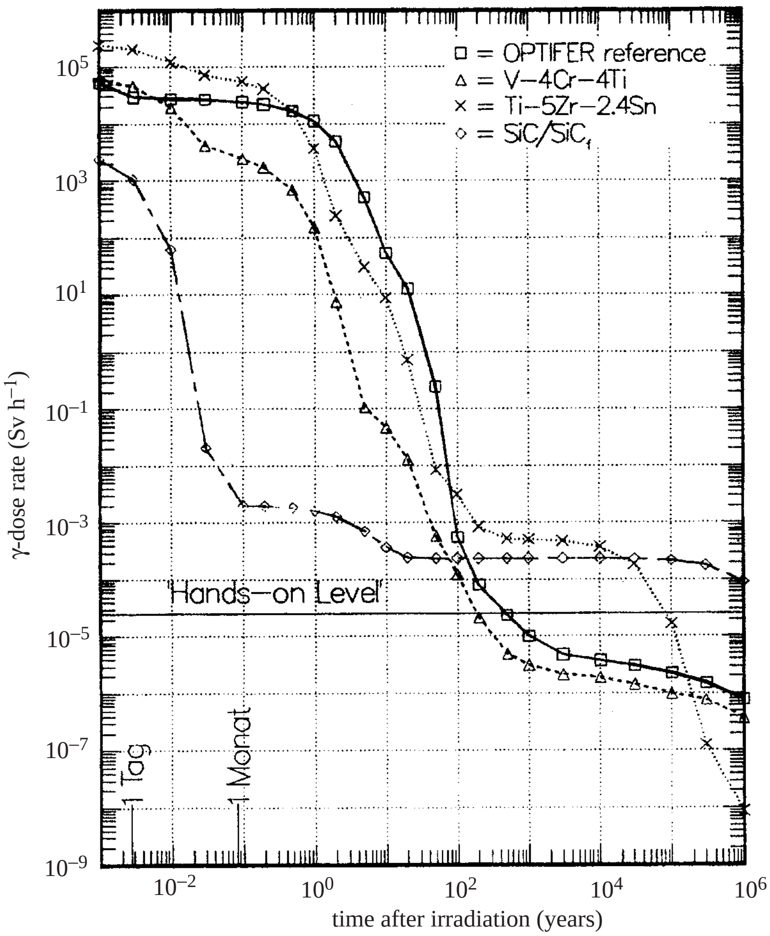

Figure 9. Contact γ-dose rate for different structural materials after an integral wall loading of 12.5 MW m^{-2}.

aims, have been adapted for the production of 'super clean' nickel alloys in the past (Butterworth & Keown 1992).

4. Status of material development and critical issues

Three material groups are at present pursued in national and international programmes.

Ferritic-martensitic steels with reduced activation (RA-F/M-steels), vanadium alloys and ceramic composites of type SiC–SiC. A short status of development, major critical issues and necessary future activities are elaborated for the three alternatives.

The preference for ferritic-martensitic steels is based on reasonable thermophysical and mechanical properties, a good compatibility with major cooling/breeding materials in the foreseen temperature window of application and a low sensitivity to swelling and He-induced high-temperature embrittlement, as mentioned above. Also, a broad industrial experience and application for these materials exists in general energy technology. Since the mid-1980s, research and development work has

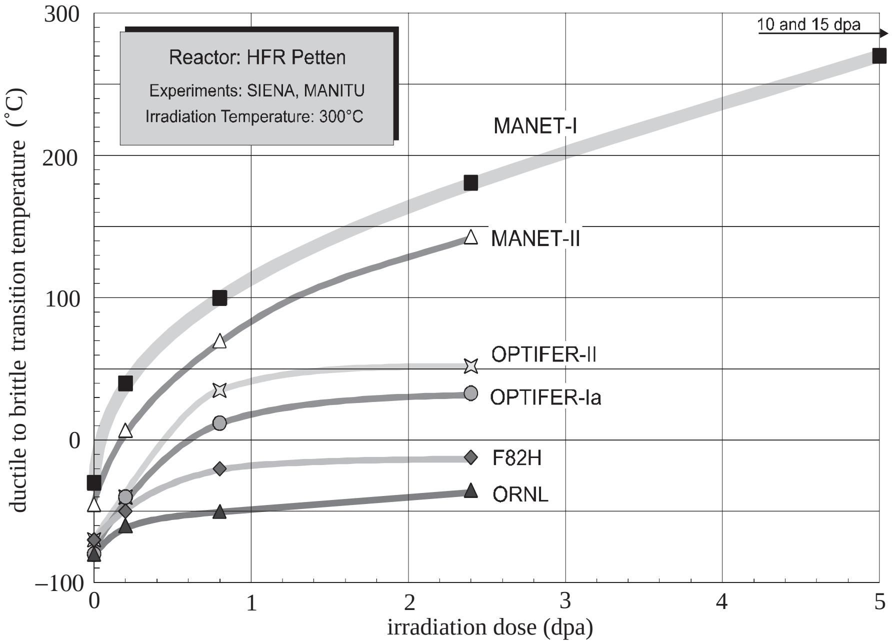

Figure 10. Ductile-to-brittle temperature shift of conventional and RA-F/M steels after low-fluence irradiations.

gradually been redirected from the optimization of conventional alloys to those with reduced activation. The activities were performed in close collaboration between Europe, Japan and the USA under the IEA-Implementing Agreement on Research and Development of Fusion Reactor Materials (Abe *et al.* 1992; Proceedings of IEA-Workshops 1991–1997).

Several experimental alloys like Optifer/Optimax, Batman, etc., have been produced and successfully tested in Europe (Ehrlich *et al.* 1994; Daum *et al.* 1997). With the production of a semi-technical 5 t alloy by NKK/Japan (the so-called F82H alloy), the critical Nb-content could be reduced to a few appm so that the activation could be strongly reduced. This alloy showed also a high degree of homogeneity in structural and mechanical properties. At present in the EU on the basis of the experience gained with these materials a new alloy EUROFER 97 has been ordered, which will be a primary candidate for future investigations.

The metallurgical and mechanical investigations of these alloys in comparison with the conventional alloys like MANET I and II, among others, have revealed better fracture toughness and impact properties combined with slightly reduced strength (Daum *et al.* 1997). First, low-dose irradiations have also indicated that a reduced sensitivity for irradiation hardening and ductile-to-brittle transition temperature shift as shown in figure 10 exists. A critical issue is the unknown influence of increased helium generation on irradiation hardening and fracture toughness properties. Recent parallel experiments in a fission reactor and in an accelerator-driven dual-beam facility, where high He-generation rates can be simulated by direct He-implantation, give a clear indication that helium contributes in addition to the displacement damage to embrittlement and ductile-to-brittle transition temperature shift, as shown in figure 11 (Lindau *et al.* 1999). This result is supported by the independent observation

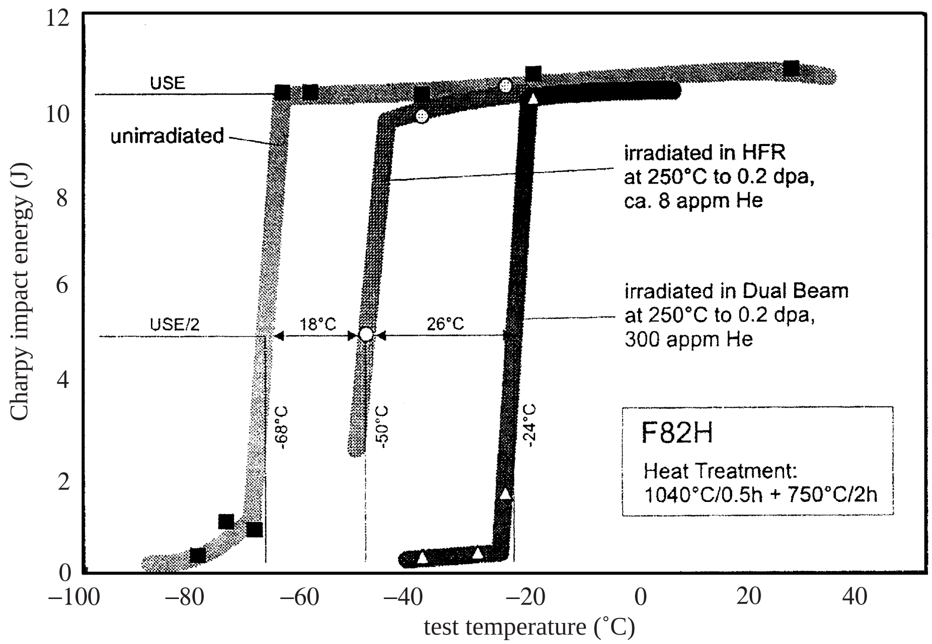

Figure 11. Combined He- and dpa effects on the ductile-to-brittle transition temperature shift in a reduced activation ferritic-martensitic steel.

in figure 10, that highly B-doped conventional alloys like MANET I and II, where due to an intense burn-out of boron high He-concentrations are generated, suffer more strongly from ductile-to-brittle transition-shifts than the low-boron containing developmental alloys. In the light of the expected increase of the He generation rates in fusion devices by at least a factor of 10, in comparison to fission neutrons, this is an important uncertainty, which has to be investigated in an appropriate test device like the International Fusion Materials Irradiation Test Facility (IFMIF) (Martone 1996; Ehrlich & Möslang 1998).

In the field of materials technology, different fabrication processes and fabrication routes like hot isostatic pressing (HIPPING) have to be applied. A general improvement of the mechanical properties for application above 580 °C, possible by the addition of an oxide dispersion strengthener is a further objective in future developmental. This could improve the net plant efficiency of those breeding-blanket concepts which use ferritic-martensitic steels.

Finally, the effect of ferromagnetism on plasma stability in magnetically confined systems has to be investigated.

Vanadium alloys have no general application in technology. The interest for nuclear application was renewed when the development of low-activation materials for fusion was discussed (Loomis *et al.* 1991). It is based on excellent high-temperature strength properties in combination with appropriate thermophysical data, as shown in § 3. The results of irradiation experiments were also promising, as mentioned before (Ehrlich & Böhm 1969; Matsui *et al.* 1996). Most attractive under the aspect of low activity are alloys with the constituents V–Cr–Ti, which possess the best, i.e. fastest, decay of radioactivity for interim and long decay times, as shown in figure 9. This behaviour could be further improved, if the production of superclean alloys with impurity levels of unwanted elements below 1 appm can be achieved.

One of the major drawbacks of vanadium alloys is the preferred pick-up of interstitial elements like C, O and H, which immediately leads to embrittlement. From such observations it is conclusive that a combination of vanadium alloys with helium cooling or in contact with solid breeder materials has not been foreseen. Instead, since the solubility of vanadium and the alloying elements chromium and titanium is extremely low in liquid alkali metals it was logical to propose a combination of the V-alloys with liquid Li as breeding and cooling medium in blankets (see table 2). However, even in very pure Li a pick-up of nitrogen and carbon from the liquid metal coolant cannot completely be excluded so that an appropriate coating is more or less mandatory. The coating with electrically isolating materials like AlN or CaO seems an appropriate way to circumvent the corrosion problem and to reduce magnetohydrodynamic effects to a minimum. As in ferritic-martensitic steels a possible degradation of fracture toughness under irradiation through irradiation embrittlement and hardening and the effect of high He-generation rates on swelling and embrittlement are further open points.

Fibre-reinforced SiC–SiC ceramic composites are used in aerospace and fossil energy plants for high-temperature applications (Zinkle & Ghoniem 1998; Jones *et al.* 1997; Bloom 1998). They have gained strong interest for the fusion materials community due to low activation and decay heat data at short and intermediate decay times and because of very favourable high-temperature strength properties. Also acceptable fracture toughness properties and a good compatibility with He as cooling medium have been reported, which favour them as structural material of breeding-blanket components for high-temperature application (see table 2).

In comparison to the above-mentioned metallic structural materials, the irradiation processes are much less understood in ceramic materials (Jones *et al.* 1997). This is true for fundamental processes like the displacements of atoms and the disorder defect fractions of the uneven elements Si and C, but also for changes of thermophysical and elastic properties. A critical issue is the observed strong reduction of thermal conductivity under irradiation, which would limit the wall loading to unacceptably low values (Fenici *et al.* 1998; Rohde 1991). Another concern is the influence of very high n,α-cross-sections in Si (about one order of magnitude higher than in ferritic-martensitic steels or in vanadium alloys) on swelling and embrittlement. A key to a reasonable irradiation behaviour of fibre-reinforced ceramic composites of type SiC–SiC is the development of quasi-stoichiometric fibres with nearly identical properties as the matrix in order to avoid delamination effects between matrix and fibres.

Technical issues to be investigated with high priority are feasibility studies on manufacturing of components including appropriate joining/brazing techniques. The development of acceptable coatings to provide adequate hermiticity and compatibility are further necessary steps. Finally, the development of appropriate design codes for the application in fusion reactors is necessary. Therefore, in comparison to the other alternatives the qualification of ceramic composites of type SiC–SiC for application in fusion reactors is a very long-term task. Finally, all three groups investigated have in common, in agreement with similar statements by other authors (Bloom 1998; Smith *et al.* 1998), the same problem, namely the unexplored behaviour under fusion-specific irradiation conditions.

5. Summary and conclusions

Fusion material development is strongly related to the design and the concepts of components and reactors. For the planning of the next-step and long-term materials programme the DEMO requirements as specified above are a reasonable target, regarding the given integrated neutron wall loading and other parameters.

The choice of structural materials for combined first-wall-breeding-blanket components depends not only on mechanical properties, compatibility with other materials and irradiation performance, but also on their radiological properties.

On the basis of these criteria ferritic-martensitic steels with reduced activation properties, vanadium alloys and ceramic composites of type SiC–SiC are alternatives, which are pursued in all international research and development programmes, partly in close collaboration under an IEA Implementing Agreement. At the present time the ferritic-martensitic steels have achieved the greatest technological maturity.

Some of the identified issues of the three alternatives could reduce the expected availability or efficiency of components through limited performance, but would not in principle endanger their feasibility. A typical example is the determination of a realistic temperature window of application which in comparison to partly optimistic assumptions in proposed designs might be restricted by compatibility or creep properties at high-temperature and radiation embrittlement at the lower temperature end. Other examples are the determination of high neutron fluence limitations due to swelling or dimensional instabilities. The identification of such limitations is the aim of a continuous research and development programme.

Other issues could be of critical importance for the feasibility of proposed concepts. The application of coatings under complex stress and irradiation conditions, the use of non-metallic structures under high neutron fluence exposure and in general the strongly increased formation of transmutation products like He and H belong to this category. These questions have to be investigated with priority.

One major problem which is common to all three alternatives is the inadequate knowledge about their performance under fusion-specific irradiation. The existing irradiation experience in fission reactors or other simulation experiments, though valuable for an early identification of possible problems, cannot substitute an appropriate testing under 14 MeV neutrons. Therefore, the development and construction of a test facility like the International Fusion Material Irradiation Facility is of greatest importance for the development and qualification of all structural materials.

I thank Dr E. Daum, Dr U. Fischer and Dr A. Möslang for providing detailed, partly unpublished results and for helpful discussions. This work has been performed in the framework of the Nuclear Fusion Project of the Forschungszentrum Karlsruhe and has been financially supported by the European Fusion Technology Programme.

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Discussion

R. Bullough (*Reading, UK*). May I first congratulate Professor Ehrlich for his fine and comprehensive discussion of the problems associated with the choice and development of suitable first-wall materials. I am grateful for this opportunity to reinforce briefly some of his concerns that particularly pertain to the microstructural evolution inevitably occurring in the first wall when exposed to the 14 MeV fusion neutron spectrum.

The first wall must withstand the thermal shock associated with plasma disruption in its irradiated state. This depends on whether the thermal conductivity of the material is severely degraded by fast neutron irradiation. Ferritic steels are probably satisfactory since their low swelling behaviour is due to an absence of significant voids. In contrast, the refractory metals such as molybdenum, niobium, etc., are also low swelling, but because they develop very high densities (so-called void lattices) of small voids, such materials suffer a severe reduction in thermal conductivity. The microstructural situation in irradiated V–Cr–Ti alloys is not known, and their low swelling behaviour could be a result of a dearth or a high density of voids, with the latter situation leading to a severe reduction in thermal conductivity. Such a contrasting microstructural response to irradiation by fast neutrons strongly supports the need to have access to a fusion materials testing reactor with the 14 MeV neutron spectrum. Needless to say, the construction of such a reactor would itself require a sensible choice of alloy for its first wall. Clearly, to minimize the inherent risk associated with such a choice, we must have appropriate simulation data (single- and multiple-beam irradiation), and simulation data can only be sensibly extrapolated to predict materials behaviour under neutron irradiation with the aid of physically based rate theory models of the entire microstructural evolutionary process.

In particular, we can briefly identify the following three needs that must be addressed before the final choice of material:

(i) to understand the synergistic influence of helium and hydrogen, as observed in pure vanadium (vanadium, for example, needs dual beam data);

(ii) irradiation creep is almost certainly dominated by the 'gas driven' growth of helium bubbles located on suitably orientated grain boundaries; we therefore need in-pile (beam) studies of such processes;

(iii) internal fatigue crack initiation is possible under periodic thermal stress and periodic irradiation damage; we thus need to know precisely where the helium and hydrogen are located in the evolving microstructure.

K. Ehrlich. I fully agree with the R&D needs that Dr Bullough has elaborated in order to come to a physically based understanding of the microstructural development under 14 MeV irradiation, including He and H-effects and the influence on swelling, creep, crack initiation, etc.

A. Kelly (*Quo-Tec Limited, Amersham, UK*). It is possible nowadays to design materials to withstand specific arduous conditions rather than just relying on the materials with properties listed in the handbooks and which are available via sales catalogues.

I have some experience in the design of materials to withstand heat fluxes of several MW m^{-2}, i.e. of comparable loading to that encountered in the first wall.

In the case of the first wall, one will have to eliminate the damage sustained by neutron bombardment by 14 MeV neutrons. This includes all the point defects generated as well as the transmutation products, e.g. He and H, and for elements heavier than atomic number 9 or so, other more vicious products. If the microstructure is properly designed, and not just the chemical composition and nuclear properties are taken into account, a microstructure can be proposed that will lead to the elimination of the point defects and of the He and H (in the light elements) by ensuring a majority escape to a surface before more are produced. We want to eliminate them, not simply study them. In order to do this, we must produce a lasagna-like structure, within the solid, the dimensions of which can be estimated from the neutron flux, the relevant cross-sections, and knowledge of the diffusion properties of the individual species. It is then possible for the structure to be designed. We know that lasagna-like structures may be produced within a solid. Obviously, the temperature of operation of the wall is an important design parameter.

The proposed microstructure may not be the exact solution, but I am emphasizing the need to design the microstructure (or texture or mesoscale) or other of the names being used. The microstructure must be designed with the same skill as are the nuclear properties and the chemical properties of the first-wall material. This is what is happening in the industries of which I am familiar, where materials are crucial to the performance.

In the case of the first wall using modern computer methods, as are used in other industries, the dimensions of the necessary 'Lasagne-like' structure can be ascertained.

K. Ehrlich. It is an interesting proposal to produce 'Lasagne-like' structures in order to 'eliminate' defects and transmutation products by proper microstructures or dimensions. Such attempts, based on theoretical rate theory approaches and experimental investigations, have been made in the past. For instance, the introduction of finely dispersed coherent precipitates as internal traps can increase the direct recombination of vacancies and interstitials and hence reduce swelling. The introduction of internal sinks like dislocations, precipitates, etc., to prevent He from collecting at grain boundaries is another example of how helium-induced embrittlement can be reduced. However, to eliminate for example helium from structures with dimensions of centimetres would need extremely high temperatures, which are not compatible with the strength properties of these materials, so that for practical applications only an optimization of the internal microstructures is a way to reduce the negative effects of defects and transmutation products.

J. Sheffield (*Energy Technology Programs, Oak Ridge National Laboratory and the Joint Institute for Energy and the Environment, University of Tennessee, USA*). Do magnetic fields, induced by ferromagnetic materials, disturb the plasma stability, and can such effects be corrected?

K. Ehrlich. In ITER, error fields that are perturbations of the axisymmetry of the magnetic fields should indeed be smaller than 10^{-5} for the so-called $B_{2,1}/B_0$ mode. In ITER studies several sources for perturbations have been studied. Doinikow *et al.* (1996) and Hender (1996) have calculated the effect of a ferritic material inserted as a massive plate with dimensions typical for a first wall in a blanket module. They both calculated error fields less than 5×10^{-6}, which obviously is acceptable. Since the results might depend on the detailed design, additional investigations will have to be made when the concept for the blanket modules is fixed.

Experimental investigations of this problem are planned in the Jaeri Fusion Torus JFT-2F by Suzuki *et al.*, in which the plasma–material compatibility will be studied, including ferromagnetic materials.

Preliminary studies on plasma discharge effects at the HAT-2 Hitachi tokamak, where unintentionally a ferromagnetic material (F82H) was used, showed that the induced magnetic fields were not strong and plasma discharge was disturbed by the material. In addition it was shown that the magnetized material compensated the field ripple of discrete toroidal field coils (Okada *et al*. 1997).

J. Sheffield. Can additional magnetic forces surpass the existing material limits and (or) they be corrected by a proper design?

K. Ehrlich. For the introduction of test blanket modules and shield blankets in ITER breeding blanket modules in DEMO analyses of the induced magnetic loadings under normal operation and for the case of plasma disruptions have been performed. As a general conclusion it was shown that the induced loads are in the accepted limits of material properties and can be handled by proper design (Ruatto *et al*. 1995).

P. Vandenplas (*Laboratory for Plasma Physics, Royal Military Academy, Brussels, Belgium*). Professor Ehrlich mentioned that Va could only be used with Li. Why is this, and what about the practical possibility of licensing of a Li-cooled reactor?

K. Ehrlich. One of the major drawbacks of vanadium alloys is the preferred pick-up of interstitial elements like C, O, N and H, which altogether lead to embrittlement. Therefore, it is conclusive that a combination of vanadium alloys with He-cooling medium and ceramic oxide breeder materials is not foreseen.

Vanadium in combination with liquid Li as a cooling and breeding medium would allow reasonably high operational temperatures (600–650 °C), since the solutibility and hence a mass loss of V in Li is very low. Interstitial pick-ups from the cooling medium (like N) should be controlled to prevent V from embrittlement. In order to reduce magnetohydrodynamic effects, insulator coatings would be necessary. The licensing of a Li-cooled reactor would, however, be a difficult undertaking.

C. Windsor (*UKAEA Fusion, Culham Science Centre, Oxfordshire, UK*). Could Professor Ehrlich estimate the time needed to develop, say, the vanadium alloys to the stage where they could be specified for construction, but with and without an appropriate materials testing source?

K. Ehrlich. Provided that a metallurgical optimized vanadium alloy is already available, which is not the case at the moment, it is estimated that a database development and qualification of such alloys up to reactor relevant fluence levels of 150 dpa needs about eight years, including irradiations in fission reactors and other simulation tools. The data confirmation and concept verification has then to be done in an appropriate fusion materials test facility. Provided this facility has the strength of 2 MW m^{-2}, such tests would need about the same time to come to a reliable result.

E. A. Little (*University of Wales, UK*). It is possible to use even smaller specimens, such as 3 mm diameter discs, and use shear punch tests to determine certain

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mechanical properties, e.g. stress–strain characteristics. The need for these smaller specimens is to use irradiation facilities such as neutron spallation sources, where only small specimen volume irradiations are possible.

K. Ehrlich. The use of 3 mm diameter disc specimens is also planned in IFMIF irradiation experiments and mechanical tests as well as TEM investigations are foreseen on such samples.

E. A. Little. I would like to make some general remarks on the development of ferritic-martensitic steels for fusion applications, and in particular to comment on their void swelling resistance.

Ferritic steels were first investigated intensively in the 1980s to provide candidate materials for fast reactor core components, and essentially the whole class of ferritics proved to exhibit high void swelling resistance under neutron irradiation to high doses in the fast reactor. The 10–12% Cr grades were ultimately selected since they possessed the additional requirements of appropriate mechanical properties and corrosion resistance. In the UK an 11CrMoVNb grade commercially designated FV448 was selected, while in Germany the grade chosen was 1.4914, and in the USA the HT-9 grade was identified.

An important point to make is that the reasons for the high swelling resistance of the ferritic-martensitic grades were never fully explained, although several theories were proposed. These theories, for example, invoked the role of solute elements, such as C or N, or the high dislocation density of the martensite, or the role of populations of dislocation loops with dual Burgers vectors in reducing point-defect supersaturations, and thereby suppressing void nucleation.

The absence of a complete understanding of the mechanism of swelling resistance does, however, leave a window of uncertainty in the behaviour of the ferritic steels under new or untested irradiation conditions. Therefore, we need to be careful in assuming that the ferritics will behave exactly in the manner we expect under fusion reactor irradiation.

As a reminder, I recall a notable example where the swelling resistance of ferritics appears to break down. This is under 1 MeV electron irradiation in the high voltage electron microscope (HVEM), in studies I undertook on the FV448 steel. This steel certainly exhibits high resistance to void swelling in the fast reactor to doses above 100 dpa, but it was found that a high density of voids and several per cent swelling could be induced even at low doses of 30–40 dpa at 420 °C in the HVEM. The reason for the loss of swelling resistance is not obvious. Electron irradiation produces radiation damage in the form of single Frenkel pairs, whereas neutron radiation damage is in the form of cascades. This would not explain the difference in behaviour. However, what we concluded was that there was rapid ingress of surface active gases into the thin foil specimen from the atmosphere of the electron microscope, and that this led to early void nucleation.

When we turn to the case of the fusion reactor, this is also a situation where there are high gas generation conditions, notably He and H transmutation products from (n,α) and (n,p) reactions, respectively. Therefore, by analogy, it is conceivable that ferritics may lose some swelling resistance under fusion irradiation conditions. The swelling resistance of ferritics under 14 MeV neutron irradiation therefore needs to be confirmed.

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K. Ehrlich. Dr Little's comment regarding void-swelling resistance of 9–12% Cr ferritic-martensitic steels under fusion-specific conditions is much appreciated. It shows the difficulty when data from fission-reactor irradiations have to be extrapolated to fusion conditions, and stresses the need to investigate and confirm such results by appropriate experiments.

T. N. Todd (*UKAEA Fusion, Culham Science Centre, Oxfordshire, UK*). How seriously should we view proposals for a component test irradiation facility?

K. Ehrlich. Material properties such as tensile strength and creep under irradiation can all be measured with specimens of only a few centimetres in size. The trend for such properties in the light-water reactor field is to miniaturize the specimen. The exception is fracture toughness testing, which needs larger volumes. However, a high flux test irradiation facility like IFMIF could also give medium fluxes in the range 1–10 dpa per year over a volume of around 8 l, which would be sufficient for fracture toughness testing.

Material tests could well be performed in an accelerator-driven d-Li facility, whose feasibility and suitability has recently been studied under an IEA Conceptual Design Activity for an International Fusion Materials Irradiation Facility (IFMIF) (Martone 1996; Ehrlich & Möslang 1998).

Regarding proposals for a component test irradiation facility, such tests would need much larger test volumes and would certainly need a D–T plasma-based machine. Tests in ITER or a DEMO would be one appropriate alternative for component testing.

I. Cook (*UKAEA Fusion, Culham Science Centre, Oxfordshire, UK*). You can have a volume neutron souce and a reliable DEMO, or no volume source and an unreliable DEMO. The experience of the fast reactor programme in several countries demonstrated how low could be the reliability of components you thought you understood!

J. Sheffield. The argument that is commonly made against a volume neutron source is that the 14 MeV neutrons only affect the immediate first wall. If you are happy that this is not going to fall apart, then the tests on other components, shielded from these neutrons, can be made on a combination of fission reactors and thermohydraulic facilities. Having said that, I personally am in favour of running a small-scale volume source which could give perhaps 20 years of operating experience on D–T systems.

R. J. Hawryluk (*Princeton Plasma Physics Laboratory, USA*). Assuming you have irradiation information from a small sample neutron source, do you need additional information from larger samples?

R. Bullough. There is no escape from large samples in the case of fracture toughness. In the fission pressure vessel studies, one-inch thick steel had to be used to assess the failure probabilities.

R. J. Hawryluk. I would like to rephrase Tom Todd's question as, 'At what level of neutron flux do you need a component testing facility?' 'What total neutron fluence do you need?'

R. Bullough. Something like 100 displacements per atom are needed. A reasonable volume irradiation can be obtained by scanning a high-energy proton beam, so a good proton accelerator could make a possible alternative volume testing facility.

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