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The challenge of developing structural materials for fusion power systems

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Abstract

Nuclear fusion can be one of the most attractive sources of energy from the viewpoint of safety and minimal environmental impact. Central in the goal of designing a safe, environmentally benign, and economically competitive fusion power system is the requirement for high performance, low activation materials. The general performance requirements for such materials have been defined and it is clear that materials developed for other applications (e.g. aerospace, nuclear fission, fossil energy systems) will not fully meet the needs of fusion. Advanced materials, with composition and microstructure tailored to yield properties that will satisfy the specific requirements of fusion must be developed. The international fusion programs have made significant progress towards this goal. Compositional requirements for low activation lead to a focus of development efforts on silicon carbide composites, vanadium alloys, and advanced martensitic steels as candidate structural material systems. Control of impurities will be critically important in actually achieving low activation but this appears possible. Neutron irradiation produces significant changes in the mechanical and physical properties of each of these material systems raising feasibility questions and design limitations. A focus of the research and development effort is to understand these effects, and through the development of specific compositions and microstructures, produce materials with improved and adequate performance. Other areas of research that are synergistic with the development of radiation resistant materials include fabrication, joining technology, chemical compatibility with coolants and tritium breeders and specific questions relating to the unique characteristics of a given material (e.g. coatings to reduce gas permeation in SiC composites) or design concept (e.g. electrical insulator coatings for liquid metal concepts).

1. Introduction

In the next century the world will face the need for new energy sources. As observed by Holdren [1], the era of cheap energy, but energy that has a significant detrimental effect on our environment, is coming to an end. Fusion is one of the most attractive long term energy options. There is an essentially unlimited fuel supply, deuterium from the ocean and tritium from transmutation of lithium using neutrons produced in the D–T fusion reaction. Fusion will not produce CO_2 or SO_2 and thus will not contribute to global warming or acid rain. For fusion to find its way into the energy market place it must compete economically with other energy options, and it must be developed as a safe and environmentally acceptable energy source, particularly from the viewpoint of radioactivity. Achieving acceptable performance for a fusion power system in the areas of economics, safety and environmental acceptability, is critically dependent on performance of the blanket and divertor systems which are the primary heat recovery, plasma purification, and tritium breeding systems. Temperature limits imposed by the properties of materials are a major limitation to achieving high thermal efficiency. Both the blanket and divertor systems will have a finite lifetime and must be maintained and replaced remotely. Reliability and lifetime, which are primarily determined by the performance of materials, will have a significant impact on plant availability, a second major factor in the cost of energy equation. Mechanical properties of structural materials under conditions associated with off-normal events, radioactive isotope

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inventory, and release paths are key considerations in designing for safety. The initial levels of radioactivity of materials on removal from service and the rate of decay of the various radioactive isotopes, dictate acceptable storage and disposal methods and the possibility of recycle of materials, both being major considerations in the environmental acceptability of fusion.

In this paper we will first review the rationale that leads to the options available for structural materials, briefly discuss the design concepts that are acceptable given the fundamental properties of these candidate materials, and discuss material performance requirements and the challenge that these requirements present to the materials science community when developing materials with acceptable levels of performance.

2. Low activation – Rationale for structural material options

There are many factors that must be taken into account in the process of determining which material systems have potential for fusion power system first wall, blanket, and divertor, structural applications. Clearly, it is essential that the material possess the basic mechanical, chemical, and physical properties which provide a useful design window in terms of allowable operating temperatures and heat fluxes, choice of coolants, tritium breeders, etc. An overriding consideration that has evolved throughout the world's fusion research programs is that the structural materials should be "low activation". The idea of building a fusion power reactor using low activation materials was first proposed by Hopkins and Price [2]. The exact meaning or development goals implied in the term "low activation" has evolved with time, and varies from country to country, but in general it is envisioned that through the proper development of materials the safety and environmental attractiveness of fusion power systems can be significantly enhanced beyond that achieved if conventional materials, developed for other applications, would be used. The rationale is quite simple. The fusion reaction does not produce radioactive nuclides and it is not self-sustaining, as is a fission reaction when a critical mass of fissionable material is assembled. Any radioactivity produced in a fusion reactor derives from capture of neutrons in the structural materials, coolant, or tritium breeder. This leads to important potential advantages of fusion relative to fission in the key areas of safety and environmental impact and relative to fossil generation in the area of environmental impact (emissions). Realizing these potential advantages requires "low activation" materials.

With regard to safety considerations the primary concern to the public will be the potential to create and disperse radioactivity in an accident. Volatilization of either the material itself, or its oxide at high temperatures, is the primary way in which materials could be dispersed. Since the fusion reaction is easily and quickly quenched the primary sources of heat to drive such an accident are heat from radioactive decay and heat from chemical reactions (e.g. oxidation or burning of Li or Be). Both the magnitude and time dependence of the generation of heat from radioactive decay can be controlled by proper selection and design of materials. Further, it is possible to minimize the potential for dispersion by selection of materials that are not easily oxidized and/or volatilized; and by design, minimize or eliminate the potential for chemical reactions that produce large amounts of heat.

Since fusion power systems will not produce CO_2 nor SO_2, disposal of radioactive materials will be the primary environmental concern. It is possible to effectively address and minimize this concern. By careful selection of the chemical elements from which we design our structural materials the amount of long half-life radioactivity can be reduced to the point that the materials are innocuous after as little as a 50 or 100 yr decay period. Recycle of materials, which would tremendously reduce the amount of radioactive waste to be disposed of, may be a possibility.

The initial choices for candidate low activation structural materials, SiC and V–Ti–Cr alloys, were developed in the United States in the early 1980s using a combination of activation/decay curves for the elements and the regulations for disposal of low level radioactive waste (10CFR Part 61) [3]. Subsequently the possibility of developing a low activation martensitic steel based on the idea of substituting W for Mo and using V and/or Ta in place of Nb as the carbide forming element(s) was developed [4].

Several groups have since studied the question of low activation materials [5–8]. Present regulations dealing with handling, storage, transportation, and disposal of radioactive materials vary significantly from country to country. Further, to address the question of induced radioactivity it is necessary to make assumptions relating to the type of coolant and tritium breeder, which effect the neutron energy spectra; and the neutron flux and fluence which the materials receive. The assumptions made in the various investigations differ, but the general conclusions are quite consistent. Piet et al. [5] evaluated all elements of the periodic table (through Bi) with respect to safety, waste disposal (using existing United States regulations), and the potential for recycle. Confirming the initial conclusions, vanadium alloys and ferritic/martensitic steels were identified as possible low activation structural alloys and graphite and silicon carbide as possible low activation structural ceramics. It is clear that control or elimination of impurities will be critical to achieving the low activation goal. Several elements, notably Nb, Mo, and Ag, because they are common alloying elements or impurities in many metallic alloys, must be reduced to very low (ppm–10's ppm) levels. Ehrlich et al. [8] have examined the activation characteristics of vanadium alloys, ferritic steels, and SiC composites containing what they judged to be realistic impurity levels. Forty and Cook [9] compared the specific activity, dose rate, and decay heat for a low activation ferritic steel, vanadium alloy, SiC (composite) and TiAl, an intermetallic alloy under development in the Japanese fusion programs. The specific activity at long times (>100 yr) is a figure of merit (FOM) for waste disposal. With the specific compositions used, the activity at long times was not dramatically different and, with the exception of SiC, the time that the materials transition from intermediate level waste (ILW) to low level waste (LLW) (United Kingdom Regulations) was found to be strongly dependent on impurities. Dose rate as a function of time after shutdown, a FOM for recycle, indicated that all materials could eventually be recycled, with the time for the dose rate to fall below the "recycle" limit being 30–60 yr for the metallic alloys. Decay power is an FOM for safety. In the Forty and Cook study, decay power was integrated for times up to 3 months to obtain a relative measure of a temperature excursion following a LOCA. Decay power does vary from material to material: the ferritic steel having the highest total decay energy, about 6.8×10^6 J/kg, the vanadium alloy and SiC being factors of approximately 9 and 18 below the steel, respectively.

In summary, the desire to maximize the attractiveness of fusion from the viewpoint of safety and environmental impact leads to a focus of material development efforts on SiC composites, vanadium alloys, and advanced ferritic steels. Successful development of these materials will provide significant improvements relative to conventional materials such as Type 316 stainless steel, Modified 9Cr–1Mo martensitic steel, etc.

3. The fusion environment – Requirements on materials performance

When considered in their totality, the requirements for performance of structural materials in a fusion reactor first wall, blanket or divertor, are arguably more demanding or difficult than for any other energy system. The blanket and divertor systems are large systems with combined thermal, hydraulic, and mechanical loading. Recent concept evaluation studies [10,11] indicate the need to move to higher power densities, high thermal efficiencies, and high availability. Higher power densities imply higher heat fluxes. In order to reduce thermal stresses the first wall thickness must be minimized which in turn increases stresses from coolant pressure. To achieve an acceptable tritium breeding ratio, the volume occupied by the structure relative to the tritium breeder must be minimized, a direction that also favors increased stresses. Achieving high thermal efficiencies implies high operating temperatures. Reactor structures will operate in temperature regimes where time dependent deformation (thermal creep) occurs and may be a controlling design property. Although the operating mode of the plasma will be steady state as opposed to pulsed, disruptions and operational interruptions will occur in which case tensile strength, fracture toughness, and fatigue properties of the structural material will determine the response of the structure. In short essentially all modes of deformation and potential fracture will come into play in the design of fusion blanket and divertor structures.

Overlaying the demanding stress, temperature, and operating conditions is radiation damage. The properties of engineering alloys are determined by their microstructure and microchemistry. The continual creation of point defects (vacancies and interstitials), point defect clusters, and transmutations (helium and hydrogen from n,α and n,p reactions) and the diffusion of these defects to sinks causes the microstructure and microchemistry to change or evolve throughout the entire life of the material. Dislocation structures evolve, cavities (voids and helium bubbles) nucleate and grow, and processes such as cascade resolution and radiation induced solute segregation (RISS) alter the phase diagram and precipitation kinetics from that expected at thermodynamic equilibrium. The underlying processes that drive microstructural evolution are dependent on temperature, defect production rates (neutron flux and spectrum) and total defects produced (fluence or dpa). In developing materials we are faced with a situation in which the operating environment dramatically alters the fundamental behavior of the material over the entire lifetime.

4. Fusion power system concepts

Silicon carbide composites, vanadium alloys, and martensitic steels differ markedly in their basic physical, chemical, and mechanical properties. None of the three is compatible with all coolants and tritium breeders that might be considered for all fusion power system concepts. Of the three candidate structural materials, SiC has the highest temperature capability and is best suited for a high temperature helium cooled system employing a Li ceramic breeder. The most attractive option for vanadium alloys is a liquid lithium coolant/breeder design which takes advantage of the chemical compatibility of vanadium and lithium, the excellent heat transfer characteristics of a liquid metal coolant, and offers reasonable high temperature operation and thus thermal efficiency. Martensitic steels have been considered for use in a wide range of concepts including water, PbLi eutectic, and helium cooled systems, with LiPb eutectic or Li ceramic breeders.

5. Development of structural materials – Critical issues

5.1. SiC composites

Silicon carbide composites were initially developed for aerospace and fossil energy applications for which high temperature strength, strength to weight ratio, and corrosion resistance are the most important properties controlling system design and performance. For fusion applications, interest in this material stems not only from its low activation properties but also from its mechanical strength at very high temperature (to 1000°C) which could make possible a high temperature direct cycle helium cooled concept. The SiC composites of interest are synthesized using chemical vapor infiltration (CVI) of performs made of continuous SiC fiber tows. Inherent in the synthesis process is the fact that some void volume fraction (up to 10%) remains in the microstructure and the final structure is permeable to gas. The microstructure consists of SiC fibers, a fiber matrix interphase, and a SiC matrix produced by the CVI process. The mechanical properties of the composite structure are critically dependent on the characteristics of the fiber–matrix interphase [12].

Research on SiC composites for fusion applications has focused on the effects of irradiation on dimensional stability, strength, and thermal conductivity. The first work explored the response of state-of-the-art composites based on commercially available Nicalon-CG fibers (TM), a carbon interphase, and a CVI SiC matrix. Significant degradation of strength, attributed to dimensional instability of the Nicalon-CG fiber and of the carbon interphase, occurred at relatively low levels of displacement damage [13,14]. The first attempt to improve the performance of the SiC composite structure incorporated a more dimensionally stable fiber and interphase. Low oxygen SiC fibers with higher crystalline perfection than Nicalon-CG (Hi-Nicalon, Nicalon-S, and Dow Corning Sylramic) were found to have improved dimensional stability upon irradiation. Alternate fiber/matrix interphases, porous SiC and interphase of alternating layers of SiC and C, have been developed. The second generation composites utilizing advanced fibers and interphases have higher strength in the unirradiated condition and exhibit smaller reductions in strength than the Nicalon-CG/carbon interphase composite as shown in Fig. 1 [15].

Although efforts to understand and improve the irradiation performance of SiC composite structures will and should continue, there are fundamental questions relating to the performance of SiC itself that must also be addressed. Fig. 2, from the work of Snead, et al. [15] shows the irradiation induced swelling of monolithic SiC, i.e. the matrix of a composite structure, as a function of irradiation temperature. At low temperatures, SiC swells by accumulation of point defects in the lattice. This phenomena saturates at relatively low damage levels and the total amount of swelling decreases with increasing temperature. It was speculated that void driven swelling, which most likely would not saturate but increase monotonically with damage level, would not occur below about 1000°C. The newer results, however, suggest that void swelling may occur at temperatures as low as 800–900°C. It has also been found that with increasing damage level the strength of monolithic SiC continues to decrease [15]. Decreasing strength and volumetric swelling of SiC will act to degrade the strength of a composite structure. All the radiation damage studies conducted to date have investigated only displacement damage. In the fusion spectrum, in addition to displacement damage, approximately 1500 appm He MW a/m^2 will be produced. Using He implantation, Hasegawa et al. [21] found that 150–170 appm He (the amount produced in about 0.1 MW a/m^2) decreased the strength of a SiC composite by about 20%. The temperature dependence of volumetric swelling, the effects of high damage levels on strength, and the effects of high concentrations of helium, up to at least 15,000 appm, upon swelling and strength, are critically important questions with answers that could significantly influence the approach to development of SiC composites for fusion. The relatively low thermal conductivity of SiC composites and the fact that it is further reduced by irradiation damage is a significant concern. With current SiC 2D composites the cross plane thermal conductivity in the unirradiated condition at 800°C is about 10 W/m K and this is reduced to about 5 W/m K upon irradiation. Reactor design studies assume a value of 15 W/m K. Materials are being developed with irradiation stable fibers which have much improved thermal conductivities. These materials are included in present studies and hold promise of meeting the 15 W/m K goal [12].

In addition to the development of a SiC composite that has and will retain adequate mechanical and physical properties in the fusion environment, there are several technological problems that must be solved. Technology for manufacturing components or structures much larger than present state of the art must be developed. The CVI manufacturing process produces a microstructure that has about 10% porosity and is permeable to gases. Methods must be developed and demonstrated that will seal the material and reduce the permeability to acceptably low levels. Methods of joining SiC to itself and to metals must be developed. Joints will be subjected to the same operational conditions as the SiC structure itself. It should be anticipated that a significant amount of field fabrication and joining will

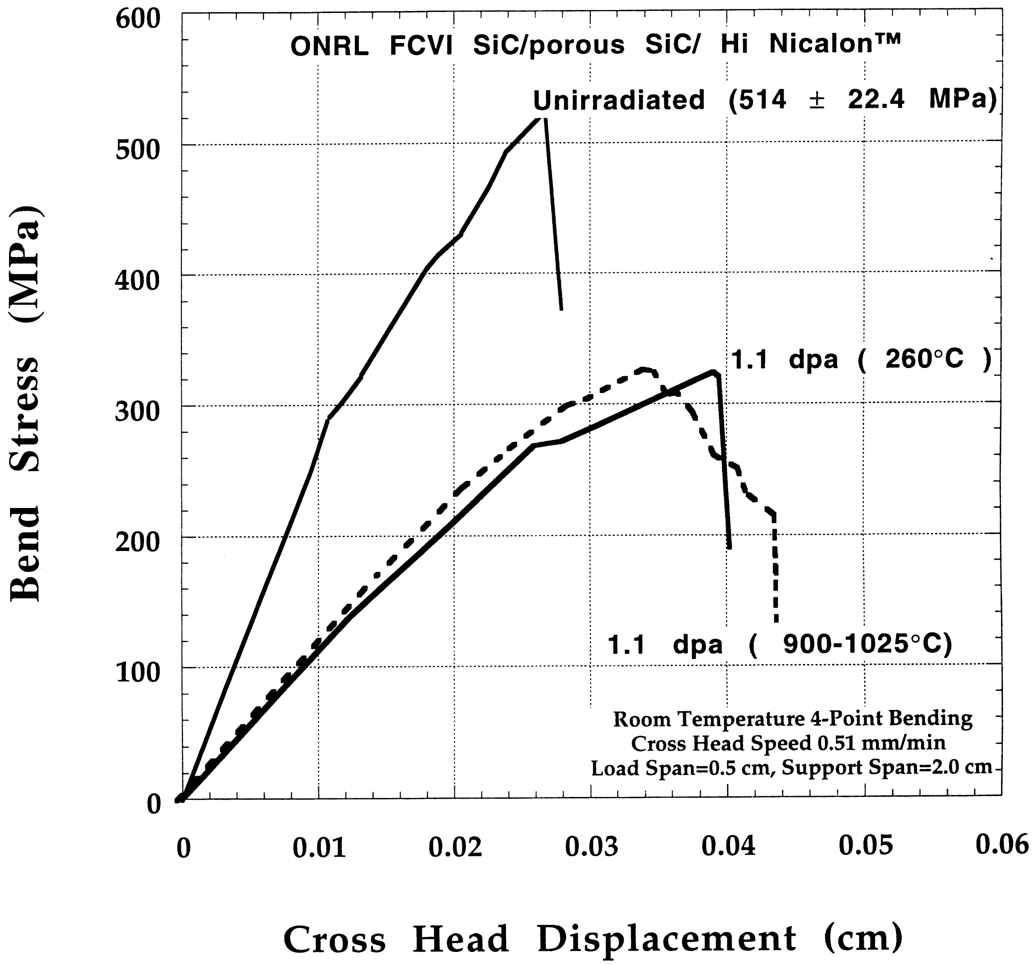

Fig. 1. Load–displacement curves for a SiC/porous SiC interface/Hi Nicalon (TM) composite in the unirradiated and neutron irradiated conditions.

have to be accomplished. Possibly the most difficult problem is the development and demonstration of a design methodology and design codes for use of SiC composites in very large structures in which there will be high temperatures, irradiation, and complex loads and load histories. One must bear in mind that today's design codes for metal structures (e.g. the ASME design codes for boilers, pressure vessels, nuclear systems) have evolved over many decades and are the product of literally thousands of man-years of engineering, construction, and operational experience.

5.2. Vanadium alloys

Vanadium alloys have several inherent properties or characteristics that make them attractive candidates for fusion reactor blanket structural applications. The combination of relatively high thermal conductivity coupled with relatively low thermal expansion and low elastic modulus yields low thermal stresses and thus high heat flux capability. V–Ti–Cr alloys have excellent compatibility with pure lithium making them the material of choice for a liquid lithium coolant/breeder blanket concept. The elements V, Ti, Cr, and Si which make up most vanadium alloys qualify as low activation. In fast reactor irradiations, vanadium alloys exhibit low irradiation induced swelling.

Loomis and coworkers [22–24] have carried out investigations of the effects of Cr and Ti content on the tensile and Charpy impact properties in V–Cr–Ti alloys. Additions of Cr and Ti, separately or together, to V, increase the yield strength, ultimate tensile strength, thermal creep strength, and the DBTT as measured in a Charpy impact test. A trade off in the desire for a low Charpy DBTT (the assumption being that a low DBTT in the unirradiated condition would yield a lower DBTT

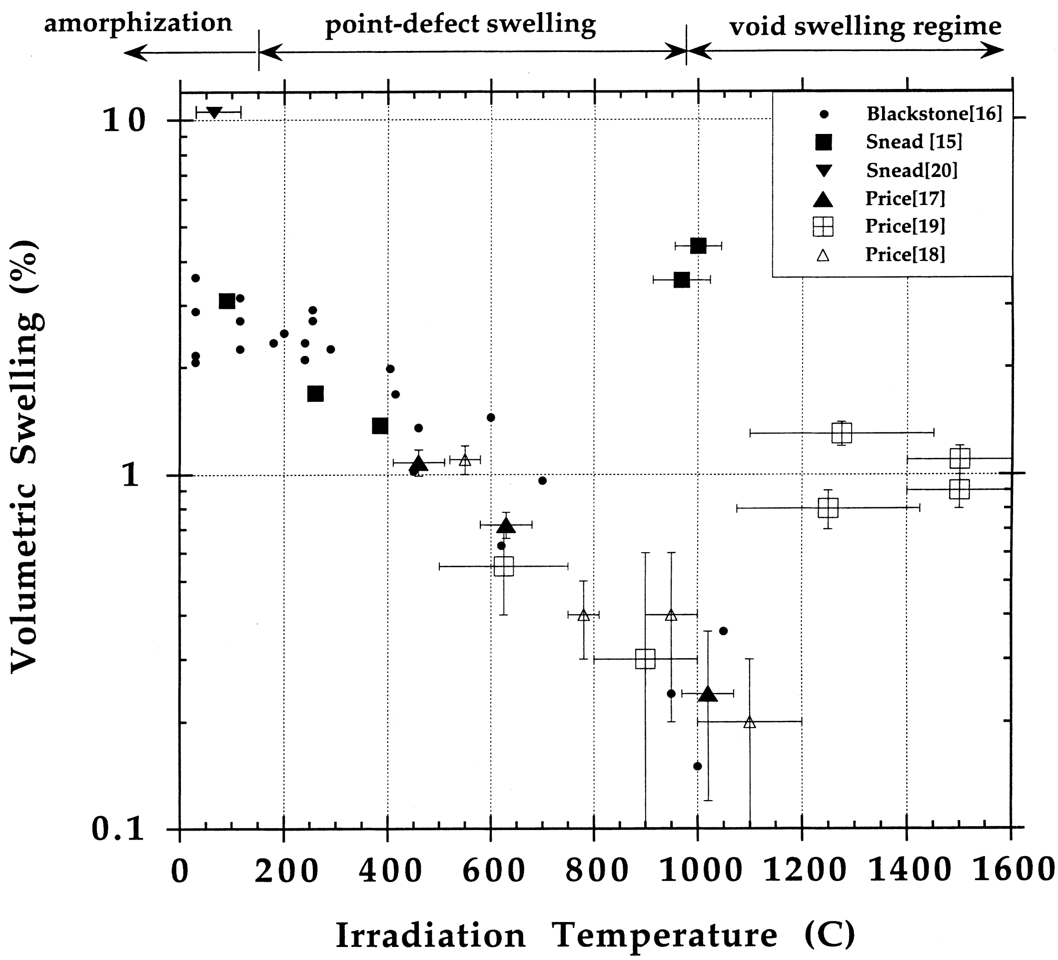

Fig. 2. Volumetric swelling of SiC as a function of irradiation temperature. The regime of void swelling may extend down to temperatures as low as 800°C [15–20].

after irradiation) and high strength led to the selection of an alloy with nominal composition V–4Cr–4Ti as a reference composition [22]. Several small (∼15 kg) and two large (500 and 1200 kg) heats have been melted and fabricated by the US fusion program. Mechanical and physical properties have or are being determined over a wide range of conditions. A significant step in the development of the technology and materials engineering for vanadium alloys is being taken in the DIII-D Radiative Divertor Program in which the V–4Cr–4Ti alloy will be used as the upper section of the new DIII-D double-null divertor [25]. The 1200 kg heat mentioned above was melted and fabricated into rod, plate, and sheet for this project [26]. Properties of this large heat were similar to those of small laboratory melts. Chemical analyses strongly suggest that production of high purity alloys required to meet low activation goals will require dedicated facilities to avoid contamination with unwanted impurities. Resistance, friction and e-beam welding are being developed as joining methods for assembly of the divertor structure [27].

A major focus of research and development of vanadium alloys is to understand and develop alloys with acceptable radiation damage resistance. It has been reported that vanadium alloys, with compositions near V–4Cr–4Ti, do not exhibit the classical response to low temperature irradiation and that they are virtually immune to displacement damage [24,28]. This assessment was based upon results of tensile tests that seemed to indicate that although the yield strength was significantly increased by displacement damage the uniform strain remained very high. The most recent experimental results shown in Fig. 3, as well as a reevaluation and correction of previously published results [26] show that this is not correct. For irradiation temperatures of 425°C and below, the yield strength is increased to levels approaching the ultimate tensile strength and the uniform strain is reduced to values below 1%. Over the range

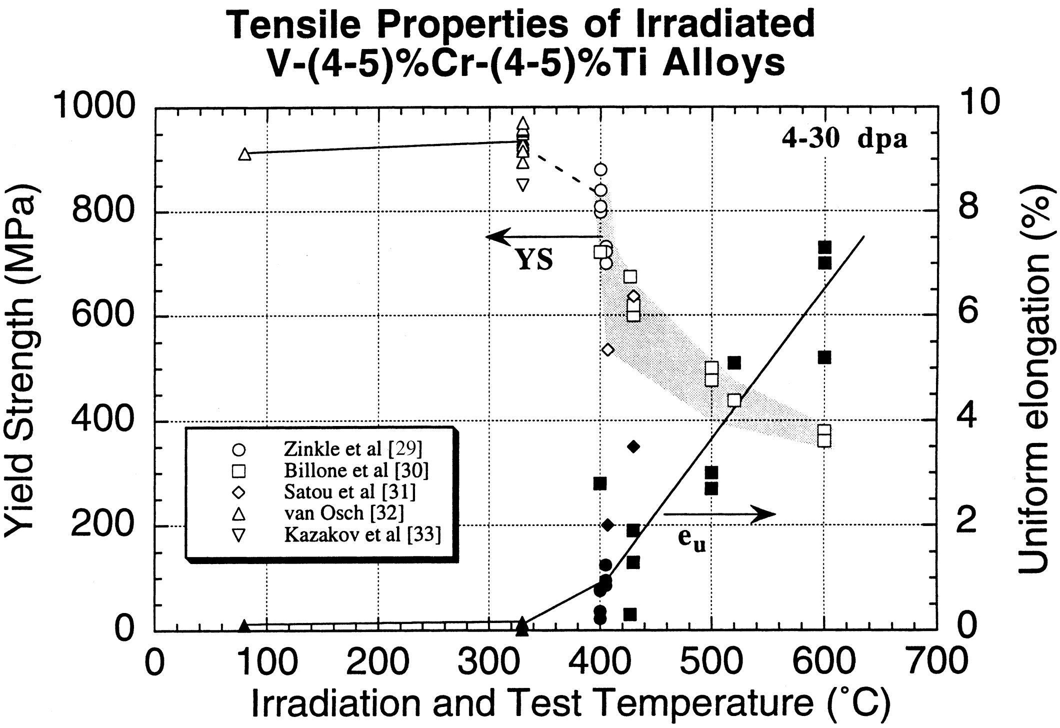

Fig. 3. Effect of neutron irradiation on the yield strength and uniform elongation of V–(4–5)Cr–(4–5)Ti alloys [29–33].

400°C to about 550°C, the magnitude of the irradiation induced increase of yield stress decreases and the yield stress approaches the unirradiated level. For irradiation temperatures up to at least 425°C, irradiation causes a significant upward shift in the DBTT. For example, the 500 kg heat of V–4Cr–4Ti irradiated at 400°C to 4 dpa has a DBTT in 1/3 size blunt notch Charpy samples that is above 290°C. On the basis of these results it would appear that the V–4Cr–4Ti alloy could not be used at irradiation temperature below about 400°C. A critically important temperature regime is 400–500°C. In this temperature range significant hardening persists and some degradation of fracture properties could occur.

At high irradiation temperatures (>~650°C) the primary concern relative to the effects of irradiation on mechanical properties is helium embrittlement. Helium embrittlement is the loss of tensile ductility and ductility and rupture life in creep-rupture tests, resulting from the growth and coalescence of helium bubbles at grain boundaries. The growth of helium bubbles is a stress induced diffusional process and is exacerbated by low strain rates and high temperatures. Injection of helium using high energy α particles, the tritium trick, a modified tritium trick, and a Dynamic Helium Charging Experiment (DHCE), have been used to study the effects of helium on the tensile properties of several vanadium alloys [34–43]. In the DHCE the highest temperature investigated was 600°C which, at tensile strain rates, would be too low to reveal tendencies for helium embrittlement [43]. Helium embrittlement is observed in high temperature (~>700°C) tensile tests and it appears that the magnitude of the effect is dependent upon the strength of the alloy. Of the alloys that have been investigated, including pure vanadium, V–3Ti–1Si, V–20Ti, V–15Cr–5Ti, V–10Ti–10Nb, V–10Ti–20Nb, V–20Ti–20Nb, and Vanstar-7, all exhibit loss of high temperature tensile ductility and indications of grain boundary fracture except pure vanadium and V–3Ti–1Si. These two alloys are much weaker than the more highly alloyed compositions. There have been no investigations of the effects of helium on creep rupture properties of any vanadium alloy and no investigations of the effects of helium on the candidate V–4Cr–4Ti alloy (except in the DHCE). At helium concentrations typical of those produced in a fusion spectrum, helium embrittlement could significantly reduce high temperature creep rupture life and ductility which would reduce allowable stresses and/or impose temperature limits. This is an area that should be addressed in near term fusion materials research for if helium embrittlement occurs in the V–4Cr–4Ti alloy this will have a significant impact on the alloy development strategy.

A liquid Li cooled/Li breeder concept is the most attractive fusion reactor blanket for use with a vanadium alloy structural material. In a liquid metal cooled concept an electrically insulating coating is required on the surface of coolant channels/ducts in which the flow direction is perpendicular to the magnetic field to reduce large pressure drops associated with magnetohydrodynamic (MHD) forces [44,45]. The coating must be chemically compatible with the vanadium alloy and with the lithium coolant at reactor operating temperatures and for blanket lifetimes. It must be assumed that the coating will be damaged during operation, e.g. by thermal cycling, and thus there must be a mechanism for self-healing during service. The requirement for chemical compatibility with Li limits the practical choices for insulating coatings to two: AlN and CaO. It has been demonstrated that these materials can have adequate insulating capacity in thin layers (a few μm). Practical methods of applying the coatings, adequate mechanical integrity and adherence, compatibility with the Li coolant and the vanadium alloy, and a practical approach to produce a self-healing system are still to be demonstrated [46–51]. This is a critically important area relative to the feasibility of a V alloy–liquid Li coolant/breeder concept.

Interstitial elements such as oxygen, carbon, and nitrogen, are always present in vanadium alloys. These impurities can profoundly affect the mechanical properties, producing a significant increase in strength, loss of ductility, and cleavage or grain boundary fracture dependent on the amounts present and the distribution in the alloy. The technological approach to reduce or minimize these effects is to add a highly reactive element, i.e. Ti in the V–Cr–Ti alloys, that will remove the interstitial elements from solution by forming stable compounds. Interstitial elements can be introduced into the alloy during fabrication, welding and joining, service in gaseous environments, and mass transfer in liquid metal coolants. Depending upon temperature, the precipitation kinetics may be so slow that the interstitials are left in solution in which case the properties will be adversely affected. Experience with other alloy systems shows that during displacive irradiation precipitates can be resolutioned, the composition of precipitates can be altered from that expected under thermal equilibrium, and in some instances phases form that are not expected based on the thermal equilibrium phase diagram form. Although the importance of interstitial impurities on the behavior of vanadium alloys has long been recognized, many aspects of behavior are not understood and continuing research is needed.

6. Low activation martensitic steels

Martensitic steels with chromium contents in the range 9–12% are used extensively in high temperatures (to about 550°C) applications throughout the world. Martensitic steels have the most advanced technology base of the three materials presently being considered for fusion blanket structural applications. High temperature martensitic alloys, e.g. 9Cr–1MoVNb and 12Cr–1MoVW, were investigated for use as wrapper or duct materials in liquid metal reactors (LMR). The database on irradiation induced swelling, irradiation creep, and effects of irradiation on tensile and Charpy impact DBTT developed in the LMR programs throughout the world is extensive. Martensitic steels generally exhibit very low swelling over the entire temperature range. At 200 dpa Gelles [52] found a maximum of 1.76% volume change in the 9Cr–1MoVNb steel and 1.02% in the 12Cr–1MoVW steel. As with all alloys, irradiation at low temperatures produces hardening or an increase in yield strength and a reduction in uniform strain. In these alloys hardening occurs at irradiation temperatures below about 400°C and the effect saturates at 10 dpa or less. Softening, or a reduction in strength, can occur at irradiation temperatures of 500–550°C and above. Common with all BCC alloys the martensitic steels exhibit a ductile-to-brittle transition in fracture behavior as the temperature is reduced. For the 9 and 12 Cr steels the DBTT is usually below room temperature. When irradiated at low temperatures, $<\sim$400°C, i.e. conditions that produce an increase in yield stress, the DBTT is increased. As the irradiation temperature increases the irradiation induced increase in yield strength decreases and the shift in DBTT decreases, becoming negligible at irradiation temperatures above about 400°C. As the yield stress, the shift in DBTT saturates by about 10 dpa.

Initial research to develop low activation martensitic steels explored alloys having Cr contents in the range 2–12%, using W as a replacement for Mo and V and/or Ta as the carbide forming elements, and 2–12% Cr alloys with V as the sole strengthening element [53]. From the initial research results the focus of the international fusion materials programs was narrowed to alloys having Cr in the range 7–9%, 1–2% W and V and Ta as carbide forming elements. The low activation alloys behave very similar to the Mod 9Cr–1Mo alloy from which they were derived. Response of the microstructure to heat treatment and tensile properties are very similar. One aspect of performance that is somewhat different is the DBTT. The reduced activation steels generally have a lower DBTT and the shift induced by irradiation is generally less than that in the standard Modified 9Cr–1Mo steel. Rieth et al. [54] irradiated several reduced activation steels (F82H, OPTIFER I and OPTIFER II, and ORNL 9Cr–2W–VTa) and conventional Cr–Mo steels (MANET I and II) in the Petten HFR reactor at 250–400°C to 0.8 dpa. Fig. 4 from their work, shows that the low activation steels all exhibited lower DBTT than the Cr–Mo steels with the F82H and ORNL 9Cr–2WVTa alloys exhibiting the best performance.

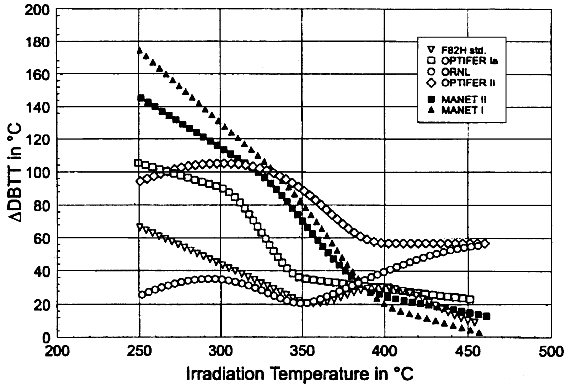

Fig. 4. Irradiation induced shift in the ductile-to-brittle transition temperature (DBTT) as a function of irradiation temperature for four low activation martensitic steels (open symbols and MANET II and MANET I) which are standard activation 10Cr–NiMoVNb martensitic steels.

Relating to the use of the martensitic steels as fusion reactor blanket structural materials there are two important areas on which research should focus. The first relates to possible effects of helium upon the fracture properties. Experiments conducted by Klueh and Alexander [55] have been interpreted to indicate that helium, produced by (n,α) transmutations, can act synergistically with hardening caused by displacement damage and precipitation to cause a shift in DBTT that is larger than that attributable to displacement damage and precipitation alone. In the most straightforward experiment, 9Cr–1MoVNb and 12Cr–1MoVW steels were irradiated at approximately 400°C in two different fission reactor spectra: a fast reactor in which very little He was produced, and a mixed spectrum reactor in which 30 and 100 appm He was produced from the 0.1 and 0.5 wt% Ni naturally present in the 9 and 12 Cr steels, respectively. Displacement damage levels were in the range 13–40 dpa and the hardening component of damage had saturated. In both steels the shift in DBTT was significantly larger after the mixed spectrum reactor irradiation even though the yield stresses were very similar. Klueh attributes this difference to the helium generated in the mixed spectrum reactor irradiation. Gelles [56] argues that precipitation of α' and G phase is responsible for the difference in response. Although there are many facets to the discussion, one key factor seems to be the behavior of Modified 9Cr–1Mo which contains only about 0.1% Ni and apparently forms neither G phase nor α' [57]. Papers presented at this conference also suggest that helium can play a role in the fracture process. It is reported that the irradiation induced shift in DBTT in steels containing boron is larger than for steels that do not contain boron, and that the magnitude of the shift scales with boron concentration [58]. The investigators attribute the observed response to helium produced from the 10B(n,α)7Li reaction. It is also reported that helium implanted with a cyclotron causes a shift in DBTT beyond that attributable to displacement alone [59]. It is critically important to establish if, and by what mechanisms, helium effects low temperature fracture behavior, and to obtain estimates of the magnitude of the effect(s) at helium concentrations relevant to fusion blanket lifetimes. The second area of research relates to the upper temperature limit for use, which is usually estimated to be 550°C based on high temperature strength properties. Increasing this temperature to 600°C or 650°C would reduce design problems and make the alloys much more attractive for helium and liquid metal cooled concepts. Development of dispersion strengthened alloys, metallurgical approaches to creating a tempered martensite structure that is stable to higher temperatures, or development of precipitation strengthened ferritic alloys are possible approaches.

7. Conclusions

Realization of fusion as an economically competitive, safe, and environmentally attractive energy source is primarily dependent upon the development of new materials that have unique properties and characteristics for use in the first wall, blanket, and divertor systems. To realize the potential safety and environmental advantages of fusion the materials must have low neutron activation characteristics. This limits the practical choices of material systems from which a structural material can be developed to SiC composites, vanadium–titanium–chromium alloys, and advanced martensitic steels. Low activation goals limit the choice of alloying elements from which the structural material can be synthesized and require that certain elements which are often present as impurities be maintained at very low (ppm) levels. Our level of understanding the science and engineering of the three candidate systems is quite disparate. SiC composites were developed primarily for aerospace applications and have never been used in large systems with mechanical, thermal, and hydraulic loading for long periods of time. Current research is focused on characterizing and understanding the effects of irradiation on the mechanical and physical properties. Composites developed for aerospace applications exhibit degradation of properties associated with dimensional instability of the fiber and the fiber–matrix interphase when subjected to neutron irradiation. The first attempts to tailor composites for radiation damage resistance used fibers and interphases that were more stable and an improved performance of the composite structure was obtained. Development of SiC composites for fusion must be considered to be in the very early stages with many scientific and engineering hurdles to overcome. Vanadium based V–Ti–Cr alloys were investigated and developed in the 1960s and 1970s for use as fuel cladding in the liquid metal reactor programs. There is a reasonably good understanding of critical aspects of the physical, mechanical, and chemical metallurgy of this alloy system. Research has focused on some of the most critical questions relating to the use of these alloys in fusion systems including the effects of irradiation on ductility and fracture at "low" temperatures, the effects of hydrogen and oxygen on mechanical properties, and development of an approach to MHD insulator system. Experimental results obtained in the past two years and a reassessment of previously published tensile data shows significant irradiation hardening and embrittlement for irradiation temperatures below about 425°C. DBTT for V–4Cr–4Ti irradiated at 400°C is above 290°C. The lower temperature limit for use of vanadium alloys in a fusion system is probably near 400°C. It is now important to obtain estimates of the upper temperature limit for use of this alloy system which may be governed by helium embrittlement, the MHD coating system, or possibly thermal creep. The knowledge base for the high temperature martensitic steel system is very extensive. The world's fusion programs have focused on alloys having 7–10% Cr with W substituted for Mo and V and Ta as carbide forming elements as low activation counterparts to the commercial 9Cr–1Mo and 12Cr–1Mo alloys. Properties of the low activation alloys are similar to those of their commercial counterparts with the exception of the fracture behavior, as measured by the DBTT, both before and after irradiation, which is superior. There are indications, however, from previously published research as well as research presented at this conference, that helium in quantities produced in the alloy in a fusion spectrum may have significant detrimental effects on the fracture behavior. This is a critically important question and should be one of focus for future investigations in the martensitic system.

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References

[1] J.P. Holdren, Scientific Am. 263 (1990) 157.
[2] G.R. Hopkins, R.J. Price, Nucl. Eng. Des. 2 (1984) 1.
[3] U.S. Nuclear Regulatory Commission, Licensing Requirements for Land Disposal of Radioactive Waste, 10CFR, Part 61, Federal Register.
[4] R.L. Klueh et al., J. Nucl. Mater. 191–194 (1992) 116.
[5] S.J. Piet et al., Fus. Technol. 19 (1991) 146.
[6] G.J. Butterworth, L. Giancarli, J. Nucl. Mater. 155–157 (1988) 575.
[7] T. Noda, J. Nucl. Mater. 233–237 (1996) 1475.
[8] K. Ehrlich et al., in: Effects of Radiation on Materials: 17th International Symposium, 1996, ASTM-STP-1270.
[9] C.B.A. Forty, I. Cook, UKAEA-FUS-365, 1997.
[10] T.C. Hender et al., Fus. Technol. 30 (1996) 1605.
[11] M.S. Tillack et al., Fus. Technol. 30 (1996) 1594.
[12] L.L. Snead et al., J. Nucl. Mater. 233–237 (1996) 26.
[13] L.L. Snead et al., J. Nucl. Mater. 191–194 (1992) 566.
[14] G.W. Hollenberg et al., J. Nucl. Mater. 219 (1995) 70.
[15] L.L. Snead et al., J. Nucl. Mater. 233–237 (1996) 26.
[16] R. Blackstone, E.H. Voice, J. Nucl. Mater. 39 (1971) 319.
[17] R.J. Price, J. Nucl. Mater. 33 (1969) 17.
[18] R.J. Price, J. Nucl. Mater. 46 (1973) 268.
[19] R.J. Price, J. Nucl. Mater. 48 (1973) 47.
[20] L.L. Snead et al., submitted to Nucl. Instr. and Meth. B.
[21] A. Hasegawa et al., in: Proceedings of IEA International Workshop on SiC/SiC Ceramic Composites for Fusion Structural Applications, in press.
[22] B.A. Loomis, D.L. Smith, J. Nucl. Mater. 191–194 (1992) 84.
[23] B.A. Loomis, D.L. Smith, J. Nucl. Mater. 179–181 (1991) 783.
[24] B.A. Loomis et al., J. Nucl. Mater. 212–215 (1994) 799.
[25] J.P. Smith et al., J. Nucl. Mater. 233–237 (1996) 421.
[26] W.R. Johnson, J.P. Smith, these Proceedings.
[27] J.P. Smith et al., presented at the 8th Int. Conf. on Fusion Reactor Materials, Sendai, Japan, 1997.
[28] H.M. Chung et al., J. Nucl. Mater. 233–237 (1996) 466.
[29] S.J. Zinkle et al., these Proceedings.
[30] M.C. Billone et al., these Proceedings.
[31] M. Satou et al., J. Nucl. Mater. 233–237 (1996) 447.
[32] E.V. van Osch, in: Proceedings of the Second IEA Workshop on Vanadium Alloy Development for Fusion, ECN-R-96-012, Netherlands Energy Research Foundation ECN, 1996, p. 417.
[33] V.A. Kazakov et al., J. Nucl. Mater. submitted.
[34] K. Ehrlich, H. Bohm, in: Proceedings of IAEA Conference on Radiation Damage in Reactor Materials, Vienna, Austria, 1969, p. 349.
[35] M.P. Tanaka et al., J. Nucl. Mater. 103&104 (1981) 895.
[36] K. Ehrlich, D. Kaletta, in: Proceedings of Conference on Radiation Effects and Tritium Technology for Fusion Reactors, CONF-70989, Gatlinburg, TN, 1975, p. II-289.
[37] R.F. Mattas et al., in: Proceedings of the Second Topical Meeting on the Technology of Controlled Nuclear Fusion, CONF-760935, 1976, p. I-199.
[38] A.T. Santhanam et al., in: Proceedings of the International Conference on Defects and Defect Clusters in BCC Metals and their Alloys, Gaithersburg, MD, 1973, p. 302.
[39] R. Carlander, W.F. Murphy, Presentation to AEC Working Group Meeting, AEC Germantown, MD, 1967.
[40] D.N. Braski, in: Proceedings of 13th International Symposium on Influence of Radiation on Material Properties, Part II, ASTM STP 956, 1987, p. 271.
[41] D.N. Braski, in: Proceedings of Symposium on Reduced Activation Materials for Fusion Reactors, ASTM STP 1047, 1990, p. 161.
[42] M.L. Grossbeck, J.A. Horak, in: Proceedings of 13th International Symposium on Influence of Radiation on Material Properties, Part II, ASTM STP 956, 1987, p. 291.

[43] H.M. Chung et al., J. Nucl. Mater. 233–237 (1996) 466.
[44] D.K. Sze et al., Fus. Technol. 21 (1992) 2099.
[45] S. Malang et al., Fus. Eng. Des. 27 (1995) 570.
[46] K. Natesan, Development of Aluminium Nitride Insulator Coatings for Fusion Reactor Applications, Argonne National Laboratory Report ANL/FPP/TM-278, 1995.
[47] J.H. Park, T.F. Kassner, J. Nucl. Mater. 233–237 (1996) 476.
[48] J.H. Park et al., in: Proceedings of Third International Symposium on Fusion Nuclear Technology (ISFNT) 1994.
[49] J.H. Park, G. Dragel, Fusion Materials Semiannual Progress Report for the Period Ending 31 March 1993, DOE/ER-0313/14, 1993, p. 405.
[50] J.H. Park, W.D. Cho, Fabrication of Intermetallic Coatings for Electrical Insulation and Corrosion Resistance on High Temperature Alloys, 1996, p. 151.
[51] J.H. Park, W.D. Cho, Fusion Materials Semniannual Progress Report for the Period Ending 30 June 1996, DOE/ER-0313/21, 1996, p. 52.
[52] D.S. Gelles, J. Nucl. Mater. 233–237 (1996) 293.
[53] R.L. Klueh et al., J. Nucl. Mater. 141–143 (1986) 1081.
[54] M. Rieth et al., J. Nucl. Mater. 233–237 (1996) 351.
[55] R.L. Klueh, D.J. Alexander, J. Nucl. Mater. 218 (1995) 151.
[56] D.S. Gelles, J. Nucl. Mater. 230 (1996) 187.
[57] R.L. Klueh, D.J. Alexander, J. Nucl. Mater. 230 (1996) 191.
[58] M. Rieth et al., these Proceedings.
[59] R. Lindau et al., presented at the 8th Int. Conf. on Fusion Reactor Materials, Sendai, Japan, 1997.