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Scientific and engineering advances from fusion materials R&D

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Abstract

Examples of significant fusion materials R&D accomplishments are summarized for three general categories: (1) Fundamental studies of radiation-induced defects and their effects on material properties, (2) development of improved materials, and (3) development of novel experimental techniques. The fundamental studies R&D (incorporating multiscale modeling and associated experimental studies) is contributing to an improved understanding of the basic mechanisms of radiation damage. Combined studies on the microstructure and properties of irradiated materials are providing important contributions on mechanical deformation and fracture mechanisms. Steady progress is being made on the development of improved, radiation-resistant ferritic–martensitic steels, V alloys and SiC composites. Scoping investigations of refractory alloys (Mo,W alloys) and improved (nanocomposited) oxide dispersion strengthened ferritic steels are providing the technological groundwork for eventual implementation of these high-temperature alloys in future fusion systems. Progress is continuing on the development of innovative miniaturized mechanical property specimens, including fracture toughness and fatigue specimens. Examples of recently developed miniaturized specimen geometries are summarized.

1. Introduction

The severe environment associated with a fusion reactor (radiation, heat flux, chemical compatibility, thermo-mechanical stresses) poses numerous materials science and technology challenges, and will require the development of an array of advanced materials. As noted in previous review articles [1,2], the development of a viable first wall and blanket system for fusion energy may represent the most challenging engineering undertaking of all time. One of the key challenges to establishing fusion as a viable energy source is the development of materials for the in-vessel systems that will provide acceptably high performance and reliability and exhibit favorable safety and environmental features. The worldwide fusion materials research programs are mainly focused on structural materials because their reliability and service lifetime determine, to a large extent, the economics, safety, environmental attractiveness, and hence the overall commercial feasibility of fusion energy [1,3]. Structural materials must operate at elevated temperatures for extended lifetimes under severe conditions characteristic of a fusion power system, including high fluxes of high energy (14 MeV) neutrons, high surface heat fluxes, and high temperature coolants and breeding materials. An underlying R&D mission of international fusion materials programs is to advance the materials science base for the development of innovative materials and fabrication methods that will establish the technological viability of fusion energy and enable improved performance, enhanced safety, and reduced fusion system costs so as to permit fusion to reach its full potential. The current international strategy for fusion materials development has been eloquently summarized by Ehrlich et al. [3].

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Significant progress has been made on the characterization and understanding of the properties and performance limits associated with materials processing, fabrication and joining, physical and mechanical properties, chemical compatibility, and effects of irradiation on properties of the key fusion materials systems. These advances are being accomplished through a combination of theory, experiment and modeling that (1) provides an understanding of the behavior of candidate material systems in the fusion environment and identifies limiting properties and approaches to improve performance, (2) undertakes the development of alloys and ceramics with acceptable properties for service in the fusion environment through the control of composition and microstructure, and (3) provides the materials technology required for production, fabrication, and power system design. Within the fusion materials programs, there has been a concerted effort to effectively use theory and modeling, and to utilize research activities on cross-cutting materials phenomena that broadly impact a wide range of materials systems, i.e. deformation and fracture mechanisms, microstructural stability, corrosion and compatibility phenomena, fabrication and joining science, and other physical and mechanical properties.

International collaborations have been the hallmark of fusion materials research for over two decades [3]. Numerous formal bilateral collaborations that involve joint planning, funding, testing and data analysis, and reporting of materials research and development exist. The international fusion materials research and development effort is coordinated through the International Energy Agency (IEA) implementing agreement on a Programme of Research and Development on Fusion Materials. Japan, the European Union, Switzerland, the Peoples Republic of China, the Russian Federation, and the United States are active participants. Fusion materials research activities coordinated through the IEA include SiC composites, vanadium alloys, advanced ferritic steels, beryllium technology, ceramic insulators, conceptual design of the International Fusion Materials Irradiation Facility (IFMIF), test specimen miniaturization technology, and irradiation damage theory and modeling.

Sustained efforts from international research programs are providing a steady stream of significant accomplishments in fusion materials research and development. This paper surveys some of the recent advances in our understanding (fundamental and applied) of materials behavior, due to fusion-funded research. In the following, examples of significant accomplishments are presented and grouped into three general categories: fundamental studies of radiation-induced defects and their effects on material properties, development of improved materials, and development of novel experimental techniques.

2. Fundamental studies of radiation effects, deformation and fracture

2.1. Fundamental radiation effects processes

Radiation damage is inherently a multiscale phenomenon, with interaction scales ranging from the picosecond and nanometer regimes (e.g., atomic displacement events) to multiple years and macroscopic sizes (e.g., constraint-induced stresses associated with irradiation creep and cavity swelling). At intermediate time and length scales, solute segregation and nucleation and growth of precipitates and extended defects occur.

It is critically important to obtain a good physical understanding of the atomic displacement process since this represents the source term for all radiation-induced phenomena. Molecular dynamics (MD) simulations of displacement cascades have been performed on a wide range of materials for MD cascade energies up to 100 keV, which corresponds to the average cascade energy for a collision of a 5.1 MeV neutron in Fe [4–7]. This MD energy is in excess of the average primary-knockon atom (PKA) energy for a fusion reactor first wall. Fig. 1 shows an example of the damage efficiency vs. MD energy in Fe, normalized to the Norgett–Robinson–Torrens (NRT) [8] displacement value [7]. The surviving defect fraction decreases steadily for MD cascade energies of 0.1–10 keV due to enhanced point defect recombination within the displacement cascade during the cascade quench [9–12], and then remains nearly constant between 10 and 100 keV due to the onset of subcascade formation above 10 keV.

Several additional important phenomena have been identified in the MD studies. For example, there is now extensive evidence that small clusters of interstitial atoms in both FCC and BCC pure metals migrate one-dimensionally rather than via three-dimensional random walk [12–14]. This can have a profound impact on the

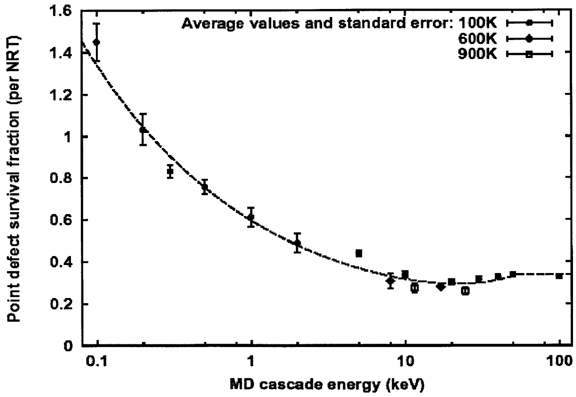

Fig. 1. Surviving defect fraction vs. MD energy in Fe, normalized to the NRT displacement value [7].

theoretical analysis of defect accumulation in irradiated metals [15]. In addition, enhanced defect production can occur at low PKA energies for certain orientations, due to a planar defect creation process [16]. It may be important to include a wide statistical spread of defect production orientations in future stochastic models in order to correctly capture the primary damage state of irradiated bulk materials. Kinetic Monte Carlo [17,18] and rate theory [19] analyses are being used to track the microstructural evolution induced by irradiation, using defect migration and binding energies obtained from MD simulations or experimental studies. By comparing theory to experiments, a cohesive understanding of microstructure evolution under irradiation is emerging, including the intriguing phenomenon of three-dimensional self-assembled microstructures [20,21]. The fundamental contributions of research by fusion scientists in this area have significant impacts on understanding non-equilibrium phase transition phenomena and related material processing technologies.

One of the most important scientific results from fusion materials research that has emerged in the past 15 years is the demonstration of equivalency of displacement damage produced by fission and fusion neutrons. This conclusion has been reached following extensive examinations of the thin foil and bulk microstructures of numerous materials irradiated to low doses (<0.01 dpa) with fission, 14 MeV fusion, and spallation neutrons [22–29], combined with MD cascade simulations [4,6,30] and key property measurements such as radiation hardening [29,31,32]. Fig. 2 summarizes examples of some observations that demonstrate the equivalency of the initial displacement damage produced by fission and fusion neutrons. As indicated in the left-hand transmission electron microscopy (TEM) photos, the predominant defect cluster in both fission and fusion neutron irradiated copper near room temperature is the stacking fault tetrahedron, with similar size and density for a given damage level irrespective of the irradiation source [26,27]. MD simulations such as those depicted for Fe displacement cascades in Fig. 2(c) [33] indicate that the subcascades produced by fusion neutrons are comparable to the cascade structure produced by fission neutrons. Fig. 2(d) shows that the radiation hardening behavior of copper irradiated with fission and fusion neutrons near room temperature is similar when evaluated according to NRT displacements per atom [34]. Of course, a critical unanswered question is the effect of the higher transmutant H and He production in the fusion spectrum on the microstructural evolution at doses above ~0.01 dpa. An intense fusion neutron source such as IFMIF [3,35,36] will be needed to examine these higher fluence fusion neutron conditions.

A longstanding question in the radiation effects literature is whether or not localized (nanoscale) melting occurs within energetic displacement cascades. The concept of a ‘thermal spike’ that could cause in-cascade melting was originally proposed by Seitz and Koehler in

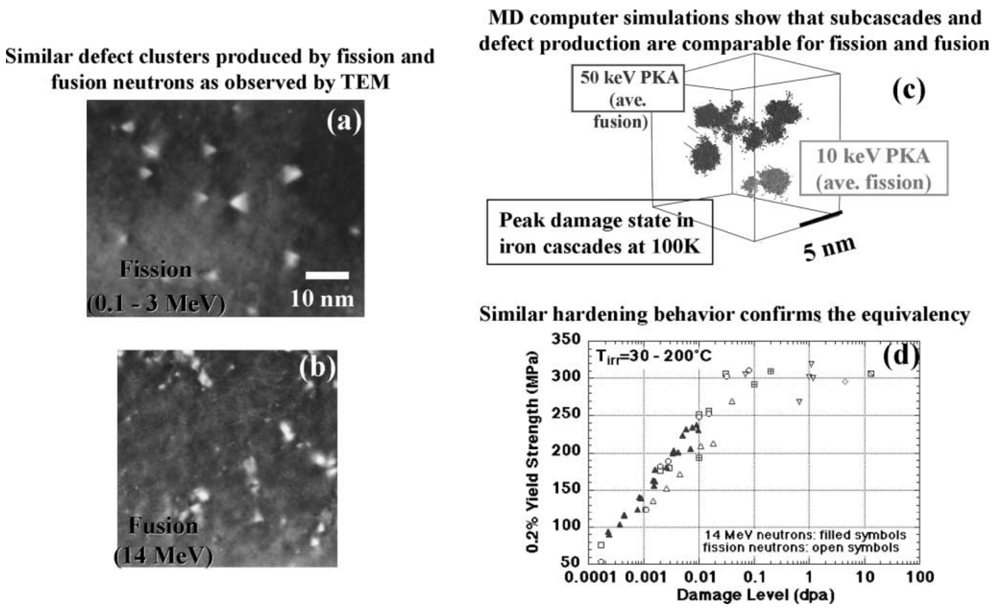

Fig. 2. Comparison of fission and fusion displacement damage characteristics as monitored by: (a) and (b) defect cluster formation in fission and fusion neutron irradiated copper near room temperature [26,27], (c) typical MD cascade morphology at the peak displacement condition for 10 keV (average fission test reactor) and 50 keV (average fusion first wall) simulations in Fe [33], and (d) increase in yield strength in copper irradiated near room temperature with fission and fusion neutrons [34].

the 1950s [37]. Since typical displacement cascade lifetimes are on the order of picoseconds (corresponding to a few lattice vibration periods), questions have continually been raised whether there is sufficient time for melting to occur. A series of recent MD and thermal conduction simulations [5,9,38,39] and experimental studies [40–42] have provided convincing evidence for nanoscale melting within displacement cascades. For example, experimental evidence for localized melting in displacement cascades has been obtained in a recent study of hafnon, zircon and thorite ceramics irradiated with 800 keV Kr ions [42]. Low temperature irradiation produced amorphization in all of these materials due to the rapid quench rate of the displacement cascades (insufficient time for the atoms to rearrange from their disordered state). At elevated irradiation temperatures, the hafnon and zircon decomposed into their component oxides: tetragonal ZrO_2 or HfO_2 and amorphous SiO_2 (Fig. 3). The decomposition occurred in spatially-localized regions with characteristic dimensions comparable to the size of the displacement cascade. As shown in Fig. 3(a), according to the equilibrium phase diagram molten zircon ($ZrSiO_4$) would be expected to separate into ZrO_2 (tetragonal) and SiO_2 during cooling. The critical temperature for the onset of phase separation was dependent on the extent of the two-phase region for the different materials, and could be correlated with estimated cooling times within the two-phase region using typical liquid diffusion coefficients [42].

The importance of grain boundaries (GB) as sinks for the defects produced by the displacement cascades is noted both in experimental observations (i.e. grain boundary segregation) and in rate theory. Recent MD [43] simulations in nanocrystalline Ni where, for the smaller grain sizes, up to 30% of the atoms in the crystal are at or are affected by the GB, illustrate some of the details of the interaction of defects with them. Simulations have been performed in samples with 5 and 12 nm grain size, introducing recoil atoms (PKA's) of 5, 20 and 30 keV at 300 K and have been compared to similar simulations in single crystal Ni. In nanocrystalline Ni, it is found that the GB is a strong sink for interstitials, with direct transport to the GB via replacement collision sequences (RCS). After the first 10 ps, a structure made only of single vacancies and vacancy clusters remains and no significant grain boundary migration is observed. The interstitials are seen to be annihilated at free volume in the boundary. For larger PKA energies, where subcascades are formed, no defects are formed in a case where the subcascade falls on a triple joint. In the case of PKA's of 30 keV, the maximum volume of the cascade is comparable to the grain size of the sample. The structure of the GB remains the same, but a net of stacking faults joins two of the boundaries, an indication that some further grain refinement could take place through this process. These simulations clearly indicate the important role of GB in the resulting damage microstructure: in the presence of a nearby GB, the liquid like core region and subsequent thermal spike region tend to funnel towards the GB, with mass transport occurring via the RCS mechanism, leaving in the grain a structure of mainly single vacancies and clusters.

Fusion material scientists are also obtaining important information on the fundamental properties of atomic and microscopic defects. Analysis of the flux and temperature dependence of amorphization in SiC has demonstrated that long range migration of self interstitial atoms does not occur below ~350 K, as well as providing information on the density change and property changes (elastic modulus, hardness, etc.) associated with the crystal to amorphous transition [44–46]. Analysis of the dislocation loop denuded zones adjacent

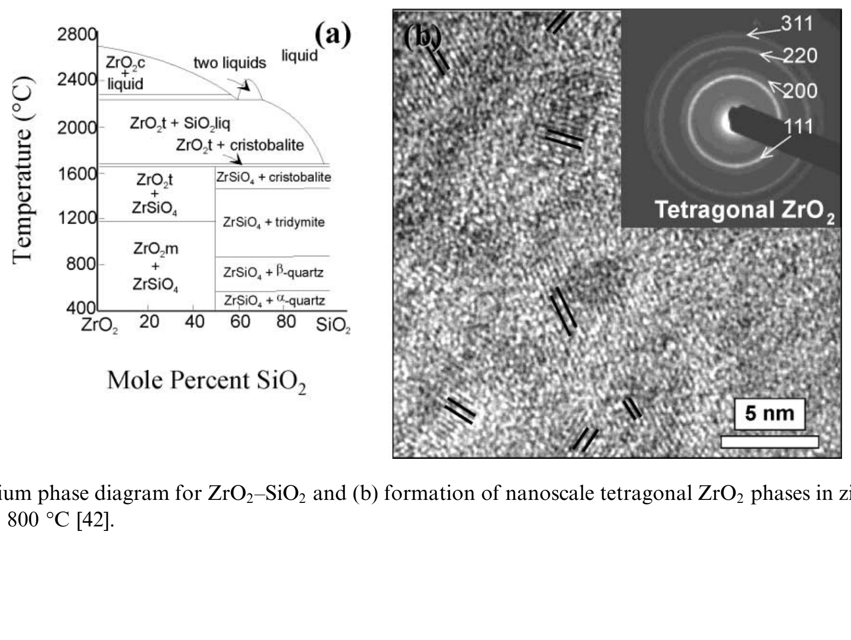

Fig. 3. (a) Equilibrium phase diagram for ZrO_2–SiO_2 and (b) formation of nanoscale tetragonal ZrO_2 phases in zircon irradiated with 800 keV Kr ions at 800 °C [42].

to defect sinks in ion irradiated ceramic insulators [47,48] and loop number density in metals as a function of temperature and dose rate [49,50] has provided important experimental estimates of interstitial migration energies in materials.

Detailed experiments and modeling of the effects of temperature variations during irradiation are in progress [51–55]. Since temperature excursions are expected to be a common occurrence in any commercial reactor due to scheduled startup and shutdown events, it is important to quantify the magnitude of potential enhanced nucleation and growth of defect clusters under varying temperature conditions compared to constant-temperature irradiation conditions.

2.2. Deformation and fracture mechanisms

Recent fundamental research activities on mechanical deformation and fracture mechanisms include: (1) investigation of defect cluster-dislocation interactions (responsible for the observed loss of tensile ductility in metals irradiated at low temperatures), (2) determination of the constitutive equations for deformation of reduced-activation materials over a wide range of temperatures and strain rates, and (3) establishment of micro-mechanics based understanding of fracture processes by development of a master curve for fracture toughness (including investigations of the effects of irradiation on the shape of the master curve).

Impressive advances in understanding the detailed interaction mechanisms between dislocations and defect clusters are being obtained through state-of-the-art computational research, such as MD and 3-D dislocation dynamics (DD). The physical mechanisms responsible for flow localization in metals irradiated at low temperatures (e.g., dislocation channeling, twinning, etc.) are being investigated by a combination of experimental [56,57] and theoretical [58–64] studies. The understanding obtained from these studies may lead to the design of materials that are resistant to these radiation-induced flow localization processes. This research has broad applicability to the general materials science fields of deformation and fracture.

Three-dimensional multiscale simulations of irradiated metals have been used to reveal the mechanisms underlying plastic flow localization in defect-free dislocation channels [60,61,63,64]. The results of experimentally validated atomistic MD and Monte Carlo simulations of the evolution of the defect population in irradiated Cu and Pd have been coupled to three-dimensional DD simulations in order to relate the defect microstructure to the mechanical behavior. The DD simulation box contains a moderate density of Frank-Read dislocation sources, together with an assembly of the experimentally observed defect type and densities. Initial defect decoration of the dislocation lines is included. The details of the dislocation-defect interaction are modeled using MD or analytical calculations. Under such conditions, DD simulations reproduce the experimentally observed yield point and channel formation. The details of the channel formation mechanisms are shown to be as follows [61]: as the Frank-Read sources evolve, a screw segment is pinned by the defects and cross-slips from its primary plane as the stress increases. In the next step, double cross-slip takes place, creating a new source in a parallel slip plane. As this process continues, a set of parallel active slip planes is created. The spreading of such localized deformation band is limited by the dislocation dipoles formed during cross-slip process, which exert a back stress on the segments still trying to cross-slip.

An extensive investigation of mechanical deformation mechanisms in unirradiated and irradiated vanadium alloys is in progress. Experimental studies performed at temperatures from 80 to 1123 K and over eight decades of strain rate have been performed, and the data are being analyzed to develop comprehensive constitutive equations for plastic deformation [65–69]. These results will help establish the operating limits (stress, temperature) for V alloys, and will also provide significant insights into the dominant creep mechanisms in other refractory metals. Similar experimental thermal creep studies on nanocomposited oxide dispersion strengthened (ODS) ferritic steels are also in progress [70–72], and the data are being analyzed to determine the dominant creep mechanisms. It is interesting to note that the thermal creep strength at 800 °C of the Fe–12Cr(W,Y,Ti) nanocomposited ODS ferritic steel is superior to that of V–4Cr–4Ti [72].

Development of a sound understanding of the effects of intrinsic (e.g., matrix strength) and extrinsic (e.g., notch acuity and specimen constraint) factors on the ductile to brittle transition temperature in BCC alloys is of high importance for fusion as well as numerous other engineering applications. The international fusion materials program is making significant contributions to the advancement of knowledge of fracture mechanics in structural alloys. Substantial progress has been made toward establishing a master curve for the fracture toughness of unirradiated V alloys and ferritic steels, which incorporates the key intrinsic and extrinsic parameters [73–76]. The results of this effort will be invaluable to the determination of the reliability of fusion structures, and should also have an impact on the design methodology outside the fusion community.

3. Development of improved materials

As reviewed elsewhere [77–79], the international fusion materials community has successfully developed reduced-activation martensitic steels with unirradiated and irradiated properties comparable or superior to conventional ferritic/martensitic steels. For example, the fracture toughness behavior of F82H reduced-activation martensitic steel is significantly better than that of HT9 (12Cr) commercial ferritic steel following low-temperature neutron irradiation [82]. Fig. 4 summarizes some of the creep rupture data for the HT9 and several reduced-activation martensitic steels specifically developed for fusion. The thermal creep strength of F82H reduced-activation martensitic steel is superior to that of HT9, particularly at high-temperatures and long times. The creep strength of F82H is comparable to that of 9Cr–1MoVNb (high induced radioactivity) steels developed for fast breeder reactors [80]. The creep data for two small laboratory heats of 9Cr–2WVTa reduced-activation martensitic steels shown in Fig. 4 suggest that even more favorable creep properties can be achieved by further optimization of the microstructure. All of the materials exhibit a change in slope in the creep rupture plot for Larson–Miller parameters near 28–29. This change in slope is due to a transition between different creep mechanisms, and further insight requires a detailed analysis using the generalized creep formula [81].

$$\frac{\dot{\varepsilon}kT}{DGb} = A\left(\frac{b}{d}\right)^m \left(\frac{\sigma}{G}\right)^n, \qquad (1)$$

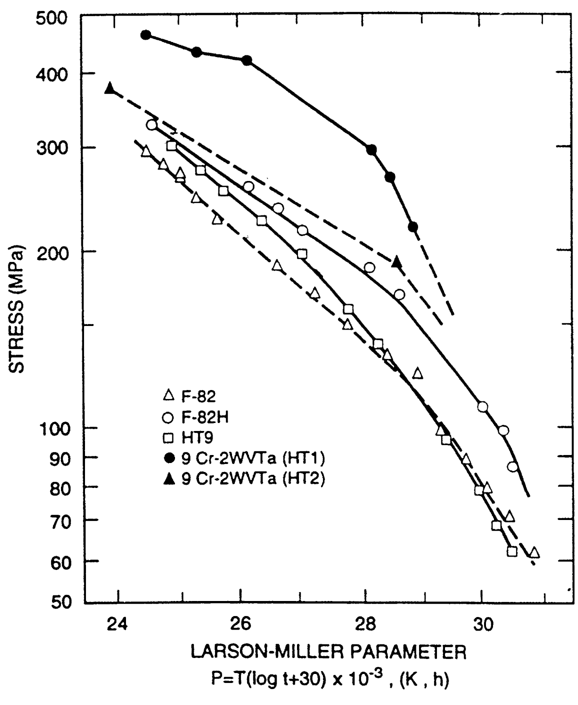

Fig. 4. Larson–Miller plot of the creep rupture data for the 12Cr HT9 commercial ferritic steel and several reduced-activation martensitic steels developed by fusion materials scientists (R.L. Klueh, personal communication).

where $\dot{\varepsilon}$ is the creep rate, k is Boltzmann's constant, T is the temperature, D is the appropriate lattice or interface diffusion coefficient, G is the shear modulus, b is the Burgers vector of the participating dislocations, d is the grain size, and A, m and n are constants that depend on the creep mechanism.

Intensive R&D activities are currently focused on F82H [80], JLF1 [83] and EUROFER [84] large heats of reduced-activation martensitic steel [79]. An alternative TiC dispersion-strengthened ferritic/martensitic steel that offers the potential for further improvements in the thermal creep resistance has been suggested, although further work is needed to produce a reduced-activation version of this steel [85]. One of the major unresolved issues for reduced-activation martensitic steels is whether fusion-relevant He and H levels have an effect on the low temperature fracture or high-temperature deformation behavior.

Improved processing techniques for ODS ferritic/martensitic steels have been developed as the result of intensive study within the Japanese fast breeder reactor program [70,71,86]. A fusion-sponsored fundamental study of the composition of small clusters present in these nanocomposited ODS ferritic steels has determined that the initially pure Y_2O_3 particles can incorporate significant amounts of Ti during ball milling [72,87]. These nanoscale (2 nm radius) clusters have high thermal stability up to 1300 °C, and produce outstanding thermal creep resistance in the steel (six orders of magnitude improvement over existing 8–9Cr ferritic/martensitic steels at ~700 °C) up to temperatures of 800 °C.

With the development of improved fabrication processes, martensitic and ferritic nanocomposited ODS steels are now available with superior high-temperature creep properties compared to previous versions of ODS ferritic steels such as MA956 [72,88,89]. Numerous development issues remain before these new ODS ferritic/martensitic steels can be seriously considered for large-scale fusion structural applications, however, including further improvements in the microstructure homogeneity, production of isotropic mechanical properties, and joining and fabrication issues. Further work is also needed to examine their fracture toughness properties at low and intermediate temperatures.

It is well established that vanadium alloys are attractive candidate materials for fusion reactor structural components due to their high-temperature strength, high thermal stress factor, and low induced activation characteristics. Research over the past decade has satisfactorily resolved many of the key feasibility issues associated with utilizing V alloys for fusion reactor structures, and an initial property database on a limited number of heats has been assembled [90–95]. Recent work has focused on assessing the thermal creep behavior and low temperature radiation embrittlement phenomena, and the available results suggest that the operating temperature window for V alloys is approximately 400 to 700 °C [96]. The effect of helium embrittlement on the operating temperature window still needs to be investigated. Two ultra-low impurity heats of V–4Cr–4Ti have recently been successfully fabricated [97]. One of the key remaining feasibility issues for V alloys is associated with the development of electrical insulator coatings to reduce the magnetohydrodynamic pressure drop in liquid metal cooled blanket concepts [98].

A concerted multi-institutional effort has resulted in the successful development of techniques to produce high-quality, high-ductility joints in V alloys by either gas tungsten arc (GTA), electron beam, or laser welding [99–102]. Fig. 5 summarizes the improvement in the DBTT as monitored by Charpy impact testing on as-welded V–4Cr–4Ti specimens. The improvement in DBTT was primarily associated with a reduction in oxygen pickup, as determined from chemical analyses of the weld region. This produced a reduction in the Ti(O,C,N) precipitate and submicroscopic interstitial impurity solute hardening contributions in the weld region. Successful welding requires careful control of the surrounding atmosphere, particularly for the case of GTA welds. The use of ultra-high-purity weld wire and alloying additions to control the grain growth during welding were also found to be important. Further work is needed to identify the best technique for field welding of V alloys.

The Group VI (Cr, Mo, W) refractory alloys have attractive thermophysical properties and high-temperature strength, but have well-known processing and embrittlement problems, along with poor oxidation resistance (for Mo, W) at elevated temperatures [96,103,104]. There have been several recent advances on Group VI refractory alloys which offer some encouragement that improved-ductility alloys can eventually be developed. Kurishita and coworkers have fabricated fine-grained Mo and W alloys containing dispersed TiC particles by mechanical alloying techniques [105–107]. The carbon content was limited in the powders in order to minimize the production of the brittle $(W,Ti)_2C$ phase, and a small grain size (∼2 μm) was utilized in order to dilute the harmful effects of oxygen segregation to GB. Three point impact tests on smooth bend bars indicated a significant improvement in the ductility of the new alloys compared to conventional Mo and W alloys both before and after low dose neutron irradiation. Further work on notched or pre-cracked specimens is needed in order to make quantitative assessments of the improved mechanical performance of these new alloys. In a related research effort, controlled additions (50–1600 appm) of Zr, B, and C to molybdenum increased the room temperature tensile ductility of GTA weldments from nearly zero for conventional materials to near 20% for the Zr, B, C-doped alloy [108]. Atom probe analysis showed that the improved ductility was associated with segregation of the Zr, B and C at GB, which caused a corresponding depletion of harmful oxygen at the GB. This offers some optimism that Mo and W alloys might be successfully joined using standard welding techniques (with resulting good mechanical properties) if suitable alloying additions are utilized. Further research is needed to understand the

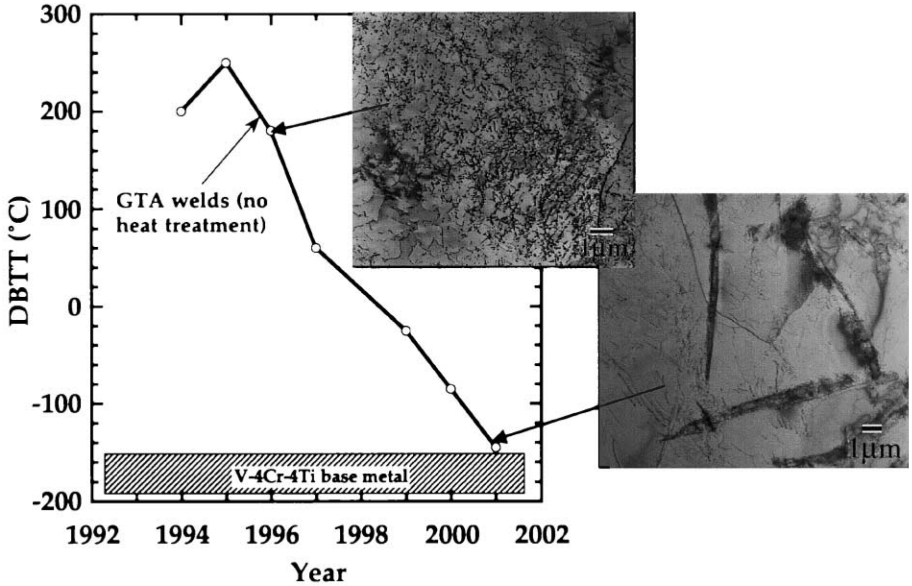

Fig. 5. Summary of improvements in the Charpy impact DBTT of as-welded V–4Cr–4Ti alloys, based on Refs. [99–101]. The inset TEM micrographs show a significant reduction in the Ti(O,C,N) precipitate density as the oxygen pickup was reduced.

mechanisms responsible for the apparent improvement in mechanical properties in the Zr, B, C-doped material.

SiC/SiC composites offer the potential for very high thermal efficiency reactor systems, but are the least developed of the three main reduced-activation fusion structural materials systems [109–112]. The main infiltration methods for SiC composites include chemical vapor infiltration, polymer impregnation and pyrolysis (PIP), and liquid phase sintering. Recent work has shown that composites with low porosity, high thermal conductivity and good tensile properties at relatively low cost can be produced by a combination of PIP and modified liquid phase sintering techniques [113,114]. Research is continuing on investigation of methods to enhance the strength of SiC/SiC composites at high-temperatures by fiber-bridging of propagating cracks [115–117].

As summarized in Fig. 6 [118], a science-based program has led to the development of SiC/SiC composites with improved resistance to radiation-induced mechanical property degradation. First and second generation SiC composites exhibited large degradation in strength after irradiation to damage levels >1 dpa, primarily due to poor radiation stability of the SiC-based fibers. The latest generation of SiC/SiC composites do not exhibit strength degradation up to the maximum doses examined to date (~8 dpa) in the collaborative US–Japan Jupiter program. These improved ceramic composites utilize the latest generation of high-purity, near-stoichometric SiC-based fibers (which have a better elastic modulus match to the SiC matrix and therefore can more effectively partition stresses during loading) and tailored interphase regions between the fibers and matrix such as porous SiC and SiC multilayer interphases. The improved fibers and interphase respond to irradiation in a manner similar to the SiC matrix (all having relatively good structural and mechanical stability in response to irradiation at temperatures 200–1000 °C), and therefore minimize mechanical property degradation in the irradiated composite. The radiation effects research activities on SiC composites are now changing from primarily a screening and evaluation phase (utilizing smooth bend bar specimens) to a new phase where engineering data including tensile properties and fracture toughness data will be obtained. An improved theoretical understanding of the thermal conductivity degradation mechanisms in monolithic and ceramic composite materials as a result of irradiation and mechanical fatigue is also being developed [119–121].

4. Development of novel experimental techniques

Progress is continuing on the development of innovative miniaturized mechanical property specimens, including fracture toughness and fatigue specimens [122,123]. Fig. 7 summarizes the reference miniaturized specimen geometries proposed for the IFMIF intense neutron source [3]. These specimen geometries enable a reduction in volume of 20–150 times compared to conventional specimens. As discussed elsewhere [123], considerable work is in progress to further verify the engineering validity of data obtained using these miniaturized specimens and to investigate the possibility of further size reductions. For example, experimental studies have been recently performed on a range of miniaturized tensile specimen geometries using different

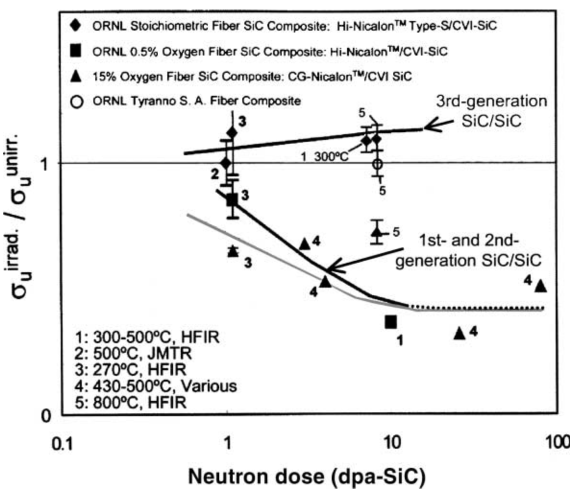

Fig. 6. Normalized flexural strength of neutron irradiated SiC/SiC composites [118].

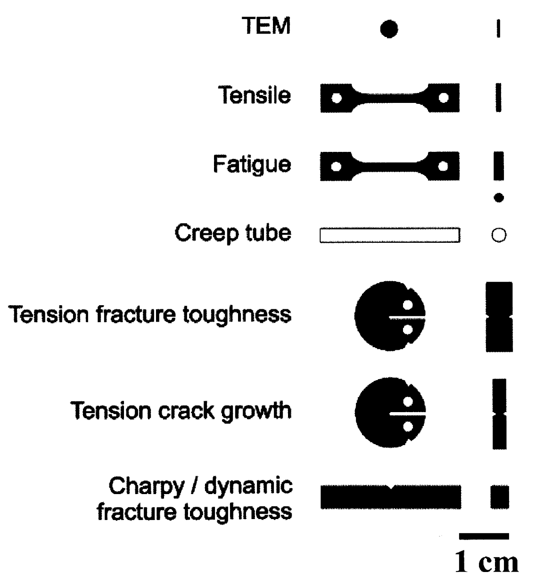

Fig. 7. Summary of reference miniaturized specimen geometries proposed for the IFMIF intense neutron source [3].

materials in order to investigate the detailed size dependence of tensile properties in austenitic and martensitic steels [124]. Finite element analysis of tensile specimen deformation [125] and pileups associated with hardness indentations [129] are also providing important scientific and engineering insight of process such as flow localization and true stress-strain constitutive relations. Considerable experimental and finite element analytical work has also been performed on two miniaturized fatigue specimen geometries that are based on hourglass [126,127] and cylindrical [128] gage regions. Both geometries appear to produce fatigue data in good agreement with larger specimens, and each miniaturized geometry has advantages for some given test conditions.

Considerable advances have been achieved through a combination of experiment and modeling that may allow fracture toughness information (master curve temperature shift) to be obtained on specimens with cross-sectional areas as small as 3 mm^2 [129]. In particular, a deformation and fracture mini-beam (DFMB) specimen with baseline dimensions of $1.65 \times 1.65 \times 9$ mm has recently been proposed for deformation (smooth and notched bending, hardness) studies and fracture toughness tests up to a reference toughness of 60–80 MPa $m^{1/2}$ [129]. As a further extension of small specimen test techniques, composite DFMB specimens have been proposed where an isotopically-enriched fracture process zone is embedded in the specimen interior [129]. As discussed elsewhere [123,129], opportunities are emerging to develop specimens on which multiple experimental techniques can be applied in order to maximize the amount of information obtained. Multiple measurements on single specimens (e.g., electrical resistivity, hardness, tensile tests, and TEM observation on single tensile specimens) have been performed on many occasions in the past, and recently shear punch and TEM data have been obtained from a single isotopically tailored neutron irradiated TEM disk [130]. However, even more efficient utilization of multiple test techniques on a single specimen geometry can be attained with careful planning.

Tensile test techniques have recently been established for ceramic composites [131]. Due to the requirement of a minimum number of fiber tows in each dimension, the smallest allowable thickness for a ceramic composite tensile is nearly an order of magnitude larger than that for many engineering alloys.

The identification of the crystallography and defect type from observations of electron microscopy of irradiated materials is a difficult but important challenge. It is also important to relate computer simulation results to experiments. The main experimental technique used to study the defect microstructure is their 'post-mortem' (meaning long after their evolution has been completed) observation by TEM. Correlating TEM image simulations with those of MD allows the validation of the atomistic models of the simulated defects, such as dislocation loops and SFT's, by their comparison with experimental observations. A method has been developed [132] which extends the multislice procedure for the simulation of high-resolution images to conventional transmission electron microscopy, both in bright field and weak beam conditions, by appropriate selection of the slicing direction. A virtual specimen is then created from the results of relaxed MD calculations, which is 'observed' at the corresponding electron microscope parameters. The method was initially tested in the characterization of defect geometries in FCC metals. More recently, it has been successfully used first to obtain the image corresponding to the structure of a $\langle 100 \rangle$ loop in Fe resulting from MD calculations. This image was then demonstrated to correspond precisely to $\langle 100 \rangle$ loops observed in the electron microscope [133].

5. Conclusions

Improved understanding of the basic mechanisms of radiation damage is being realized due to multiscale modeling and associated experimental studies. For the metals of interest for fusion applications, it is now firmly established that subcascades produced in 14 MeV fusion neutron collision events are very similar to cascades and subcascades produced in fission neutron events. This allows future attention on fission–fusion correlations to be focused on the role of fusion-relevant H and He generation on the microstructural evolution. The potentially important role played by one-dimensional migration of glissile interstitial clusters is being addressed in a number of modeling and experimental studies.

In the area of mechanical deformation and fracture mechanics, constitutive equations for deformation have been obtained for many of the materials of interest for fusion structural applications and the master curve methodology has been successfully applied to the fracture toughness behavior of reduced-activation martensitic steels and V–4Cr–4Ti alloys. The detailed physical mechanisms responsible for dislocation channeling and accompanying flow localization in metals irradiated at temperatures below $\sim 0.3 T_M$ have not yet been fully identified, although considerable progress has been made in the past few years. Much of the resurgence in interest in dislocation channeling can be attributed to research in support of the ITER project.

Significant incremental improvements have been achieved in the development of radiation-resistant ferritic–martensitic steels, V alloys and SiC composites. It is worth noting that the unirradiated and irradiated properties of reduced-activation martensitic steels are comparable or superior to conventional steels. Vanadium alloys with controllable levels of interstitial impurities have been successfully fabricated, welded, and mechanically tested. State-of-the-art SiC/SiC composites fabricated 5 to 10 years ago showed unacceptably poor radiation stability for damage levels above ~1 dpa (due to poor radiation stability of the SiC-based fibers), whereas current-generation SiC/SiC composites do not exhibit any strength degradation up to the maximum doses studied to date (~10 dpa). Recent international activities on improved (nanocomposited) ODS ferritic/martensitic steels are also yielding promising results, although considerable work is needed to demonstrate the viability of ODS ferritic/martensitic steels for fusion structures. Improvements in processing and weldability of refractory alloys (V, Mo, W alloys), along with other research activities, are providing the technological groundwork for eventual implementation of these high-temperature alloys in fusion systems.

Further optimization of small specimen test techniques appears likely, particularly for fracture mechanics specimens. Utilization of multiple test techniques on a single specimen geometry will likely become increasingly important in future years.

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