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ScienceDirect
Journal of Nuclear Materials 367–370 (2007) 659–671
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Current status and critical issues for development of SiC composites for fusion applications

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Abstract

Silicon carbide (SiC)-based ceramic composites have been studied for fusion applications for more than a decade. The potential for these materials have been widely discussed and is now understood to be (1) the ability to operate in temperature regimes much higher than for metallic alloys, (2) an inherent low level of long-lived radioisotopes that reduces the radiological burden of the structure, and (3) perceived tolerance against neutron irradiation up to high temperatures. This paper reviews the recent progress in development, characterization, and irradiation effect studies for SiC composites for fusion energy applications. It also makes the case that SiC composites are progressing from the stage of potential viability and proof-of-principle to one where they are ready for system demonstration, i.e., for flow channel inserts in Pb–Li blankets. Finally, remaining general and specific technical issues for SiC composite development for fusion applications are identified.

1. Introduction

Silicon carbide (SiC) is a unique heat-resistant material that exhibits several attractive properties such as high strength at elevated temperatures in excess of 1500 °C, general chemical inertness, low specific mass, and low coefficient of thermal expansion. Continuous SiC fiber-reinforced SiC-matrix composites (SiC/SiC) are ceramic composites designed to have a pseudo-ductile and predictable fracture mode and tailorable physical and mechanical properties while taking advantage of most of the inherent merits of monolithic SiC. SiC/SiC have been studied and developed for decades because they are promising materials for high temperature applications such as gas turbines and aerospace propulsion systems. For fusion and advanced fission energy applications, additional interest in SiC and SiC/SiC comes from the excellent irradiation performance, which was demonstrated during early

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doi:10.1016/j.jnucmat.2007.03.032

studies on chemically vapor deposited (CVD) SiC for fission fuel coatings [1,2]. Furthermore, safety features arising from the inherent low activation/low decay heat properties [3,4] and low tritium permeability [5] for SiC have been demonstrated.

Recent development of SiC/SiC in the fusion materials community has been intended to address the key feasibility issues when using an essentially new class of materials [6–10]. Benefiting from non-nuclear programs that pursued issues such as manufacturing technology, testing development, and to a limited extent how to design structures from these materials, the fusion community was able to focus on fusion-specific issues. For this reason, the majority of resources in fusion materials programs have addressed the fundamental performance of SiC composites under irradiation along with simulation to define the effects of transmuted helium. Additionally, fusion specific design issues such as the need to develop a leak-tight component, the need for joining the materials with low-activation agents, and issues regarding coolant compatibility have been addressed during the last decade.

In this paper, the recent status for application development and the design requirements for using SiC/SiC in fusion and advanced fission energy systems are first reviewed briefly. The case is made that SiC/SiC development has progressed from the stage of potential viability and proof-of-principle to one where they are ready for system demonstration. In the following sections, recent progress in research and development of SiC/SiC in fusion materials and related programs, namely development of materials, joining, characterization, and irradiation effects studies, is overviewed. Finally, identification of the remaining critical issues, both for general fusion applications and specific to individual target systems, is attempted.

2. Status of application development and design requirements

2.1. Proposed power reactor blanket concepts

Utilization of SiC/SiC for structural components in fusion power reactor blankets was proposed in the early 1990s [11–13]. Reasons for the proposed employment of SiC/SiC were the anticipated attractive energy cost competitiveness and favorable social receptivity associated with the high operating temperature and low activation. The proposed designs were based on self-cooled lead–lithium (SCLL) or helium-cooled ceramic breeder (HCCB) concepts. The latest SCLL conceptual designs, EU Power Plant Conceptual Study (PPCS) Model D [14] and the US ARIES-AT [15], assume the lowest/highest operating temperatures for SiC/SiC structures of ~700/~1000 °C yielding a power conversion efficiency of ~60% for the blanket circuit. For the HCCB concept, the Japanese DREAM assumes the inlet/outlet helium coolant temperatures of ~600/~900 °C with a gross thermal efficiency of ~50% [13]. In Table 1, key properties assumed for SiC/SiC when used in structural or the flow channel insert (FCI) application are summarized and compared with typical values for two presently-available promising materials (2D and 3D CVI and NITE). General reviews of issues regarding blanket designs using SiC/SiC are given elsewhere [16,17]. Many of the assumed basic requirements are already satisfied or advanced to a substantial extent so that the design could rather

Table 1
Key properties for SiC/SiC assumed in blanket designs and typical values for CVI and NITE composites

Key properties (Unit)	Structure	FCI	2D CVI	3D CVI	NITE
Thermal conductivity, thru-thickness (W/m K)					
Non-irradiated, 500 °C	$\gtrsim$ 20	<2	~15	25–40	15–40
Non-irradiated, 1000 °C	$\gtrsim$ 20	n/a	~10	20–30	10–30
Irradiated, 500 °C	$\gtrsim$ 20	<2	2–3	5–8	–
Irradiated, 1000 °C	$\gtrsim$ 20	n/a	4–6	12–18	–
Electrical conductivity, thru-thickness (S/m)					
Non-irradiated, 500–1000 °C	$\lesssim$500	$\lesssim$100	0.1–1000	–	–
Irradiated, 500–1000 °C	$\lesssim$500	$\lesssim$100	–	–	–
Tensile properties, in-plane					
Ultimate tensile stress, 500–1000 °C (MPa)	300	n/s	250–350	100–200	300–400
Matrix cracking stress, 500–1000 °C (MPa)	n/s	$\gtrsim$ 100	~150	–	200–250
Modulus, 500–1000 °C (GPa)	200–300	200–300	~250	~200	300–400

n/a: Not applicable; n/s: Not specified; –: Data not available.

be adjusted. Discussion of some of the individual properties will be provided in the later sections.

2.2. Flow channel insert application

The use of SiC/SiC as the insert for a lead–lithium flow channel was proposed initially in the EU advanced lead–lithium blanket concept [18] and the US ARIES-ST blanket design [19]. In this application, an FCI serves as an electrical and thermal insulator in order to mitigate the MHD pressure drop and to allow a significantly higher coolant outlet temperature compared to the temperature limit for the metallic duct structure. The dual-cooled lead–lithium blanket concept (DCLL) has been adopted for the proposed US test blanket module (TBM) and Chinese late-stage TBM to be inserted in the International Thermonuclear Experimental Reactor (ITER) [20]. The design inlet/outlet lead–lithium temperatures for the EU PPCS Model C DCLL blanket are 480/700 °C. This outlet temperature of 700 °C approximately corresponds to the inside temperatures for the FCI, whereas the outside temperature will be $\lesssim$500 °C [21].

2.3. Interaction with fission material programs

Programs considering potential utilization of SiC/SiC for core and/or in-vessel components in advanced fission energy systems have been initiated during the last few years. These programs are associated with the effort for Generation IV (Gen IV) Nuclear Power Plants, which is internationally coordinated by the Generation IV International Forum (GIF). In the US, the use of SiC/SiC is considered as an option for control rod sleeves in a very high temperature reactor (VHTR)/next generation nuclear power (NGNP). This option is based primarily on the promise of SiC/SiC demonstrated by the fusion materials program [22].

The US NGNP composite program assumes the fundamental viability of chemically vapor infiltrated (CVI) SiC/SiC in a fission neutron environment, and focuses on (1) confirmative feasibility issues which include irradiation effects and fabrication of desired shapes and sizes, (2) key technical issues governing the life-time envelope such as time-dependent fracture and irradiation creep, and (3) providing support to test standards and design code development in the frameworks of ASTM International, International Organization for Standardization (ISO), and American Society of Mechanical Engineers (ASME). Similar efforts are on-going or planned in several countries including France and Japan. A common feature in these programs is that addressing the application-specific issues is targeted based on the perceived potential of SiC/SiC as a radiation-resistant high-temperature nuclear material.

SiC and SiC/SiC are also being considered as the primary candidate materials for fuel cladding and core structures in a gas-cooled fast reactor (GFR), which is one of the six Gen IV concepts. The GFR concept features a fast-neutron spectrum, a helium cooled reactor and a closed fuel cycle [23]. General feasibility issues still exist for SiC/SiC used in a GFR application, i.e., the effect of very high fluence neutron irradiation at temperatures ranging to largely beyond 1000 °C, as well as issues which overlap those for fusion power reactor applications.

3. Status of material development and characterization

3.1. CVI composites

For nuclear environments, SiC/SiC made from stoichiometric, high purity, and fully crystalline SiC fibers and matrices are preferred for the harshest conditions [24]. This is because of the demonstrated radiation instability of common matrix second phases such as metallic silicon in the reaction-bonded matrices [25], sintering agent represented by boron in hot-pressed SiC [26], and the amorphous Si–C–O commonly obtained through preceramic polymer routes [27]. Therefore, CVI, which essentially imparts CVD-SiC into a SiC fiber fabric, is an obvious selection of the matrix densification technique for fusion reference grade SiC/SiC [28,29].

Early Generation III (near-stoichiometric and crystalline) SiC fiber, CVI-SiC matrix composites were produced in US–Japan collaboration in 1998 and subjected to a neutron irradiation campaign [30]. Hi-Nicalon™ Type S ('Nicalon-S' hereafter) and Tyranno™-SA fibers were used and various fiber–matrix interfacial phases ('interphases' hereafter) including pyrolytic carbon (PyC) and SiC-based interphases, were examined. Uni-directional (UD) reinforcement architecture was employed for investigation of fundamental aspects of interfacial properties and irradiation effects. Non-irradiated and low dose ($\sim 1\times10^{25}$ n/m^2, $E > 0.1$ MeV, the same shall apply hearafter) irradiation experiments demonstrated the promise of PyC and multilayered (ML) $(PyC/SiC)_n$ interphases [30–32]. The results for higher dose irradiation are discussed later in this review.

Optimum interphase structure and configurations for the advanced CVI composites were then explored. Nicalon-S and Tyranno-SA Grade-3 (SA3) composites with various PyC and ML interphases were fabricated using the forced-flow CVI (F-CVI) technique and evaluated [33–35]. The important finding was the general lack of sensitivity of mechanical properties on PyC interphase thickness for the both SiC fiber composites [34,35]. As seen in Fig. 1, the SA3 composites lacked the steep strength drop at very thin interphase, which is commonly observed for SiC/SiC with older generation fibers. These results show that the interphase can be designed for the optimum radiation and environmental compatibility without compromising the fast fracture strength.

For blanket materials, a high through-thickness thermal conductivity is required to reduce thermal stress. To identify the practical limit and to explore the possibility of further improvement of SiC/SiC thermal conductivity, CVI composites with three-dimensional (3D) fabric architectures with various through-thickness (z) fiber fractions were recently produced and evaluated. A study on orthogonal 3D CVI composites with SA3 reinforcement [36] demonstrated that the presence of z-tows enables higher through-thickness conductivity, because the matrix shells around the tows carry a major fraction of heat. The maximum of the thermal stress figure of merit appeared to occur at relatively small z-fiber fractions [36–38]. Model-based predictive capability for temperature-dependent conductivity of 2D and 3D CVI SiC/SiC has been demonstrated [36,39].

Irradiated thermal conductivity of SiC/SiC will be limited by the irradiation defect accumulation, which is determined primarily by the irradiation temperature [40,41]. Raising thermal conductivity beyond such limits obviously requires incorporation of very high thermal conductivity media [42]. To examine this possibility, 2D and 3D CVI SiC/SiC's supplemented with pitch-based P-120S graphite fibers were produced and evaluated. These 'hybrid fabric' composites demonstrated limited promise, since the CTE (coefficient of thermal expansion) mismatch introduced fine micro-cracks in the SiC matrices surrounding the graphite fibers [43]. The concept requires an improved architectural design, perhaps properly adjusting the hybrid fabric configurations so that the matrix micro-cracking can be suppressed.

In the EU program, manufacturing and extensive characterization of commercial grade 2D and 3D SA3, CVI SiC matrix composites has been pursued [44,45]. The composites were made with a single-layered PyC interphase of ~80 nm-thickness and were optimized mainly for thermal conductivity.

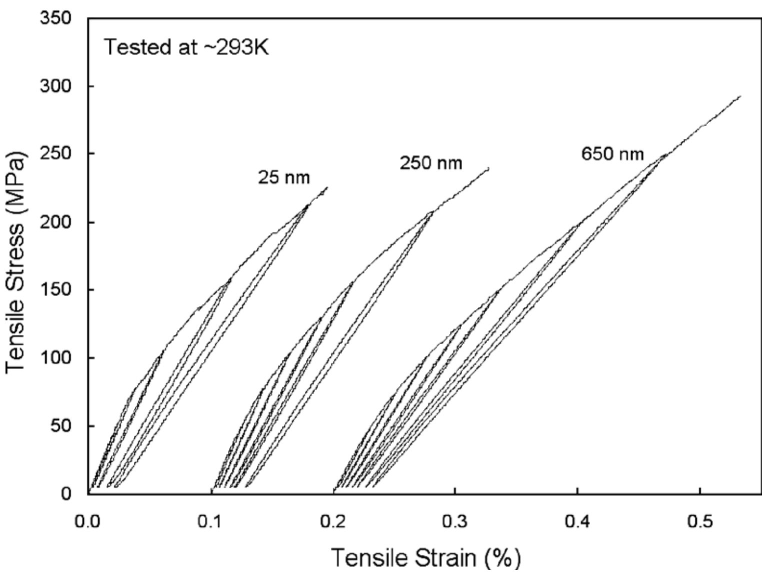

Fig. 1. The influence of carbon interphase thickness (shown in data labels) on tensile properties of Tyranno™-SA3 CVI SiC/SiC. The similarity of the stress–strain curves indicate a general insensitivity of composite strength on interphase thickness.

The 3D composites demonstrated reasonable through-thickness thermal conductivity (reaching ~18 W/m K at 1000 °C) [45]. Low cycle fatigue properties were investigated by 4-point flexural tests and showed satisfactory results at RT and 1000 °C in argon. The creep behavior, investigated by constant stress rupture test in a flexural configuration, exhibited a short lifetime at high stress with run-outs only at approximately half of the room temperature proportional limit stress [44]. The high temperature test results suggest the necessity of further R&D on fiber–matrix interphase and oxidative protection at elevated temperatures.

3.2. NITE composites

The nano-infiltration and transient eutectic-phase (NITE) process was developed by employing a liquid phase sintering (LPS) method utilizing a small amount of oxide additives. The development and fundamental characterization of the early NITE SiC/SiC are summarized elsewhere [46,47]. Based on achievement by the laboratory grade development, pilot commercial grade (PG) production was recently initiated [48]. During the PG productions, several process improvements were achieved.

Near net-shaping techniques for NITE SiC/SiC have also been improved [48]. Small diameter (10 mm ID) tubes were successfully fabricated with the fiber angle of ±15° and ±30° (Fig. 2). The highest density of the tubes was 3.02 g/cm^3, which is close to 3.08 g/cm^3 attained for the PG3 plates. The average diametral ring compression strength was 108 MPa. Also 200 mm diameter GFR fuel compartment cylinders were produced with a wall thickness of 3 mm. These cylinders were successfully engineered to possess 5–40% through-wall porosity.

A limited neutron irradiation data has been acquired for NITE SiC/SiC. The PG1 composite was irradiated in high flux isotope reactor (HFIR) at Oak Ridge National Laboratory (ORNL) to 4.2×10^{25} n/m^2 at 1000 °C. The observed mechanical and dimensional stability suggested decent irradiation tolerance of the matrix material [49]. Also, flexural strength of the NITE matrix material, in the form of a monolithic ceramic, was statistically evaluated after irradiation to 6×10^{24} n/m^2 at 750 °C in Japan Materials Test Reactor (JMTR), and exhibited no major strength degradation [49].

Remaining critical issues in the NITE SiC/SiC development include improved control of matrix quality, stability of the matrix second phases at high temperatures, creep and oxidative resistance associated with the potentially enhanced oxygen transport due to the sintering additives, and radiation stability to higher doses.

Fig. 2. Small diameter tubes of third generation pilot commercial grade (PG3) NITE SiC/SiC produced through the pseudo-isostatic hot-press technique.

3.3. Other matrix densification processes

Other matrix densification techniques historically considered for nuclear applications are polymer impregnation and pyrolysis (PIP) and variations of direct conversion processes represented by melt infiltration (MI). Either technique is not presently attracting much attention for nuclear applications due to the recognized significant challenges [50,51]. However, for PIP, it was suggested that enhanced matrix crystallization and improved stoichiometry could be attained using a very high temperature treatment and novel polymer precursors [50,52]. Primary issue identified for high-crystallinity PIP composites was a loss of matrix integrity due to severe micro-cracking [53]. This problem may possibly be overcome by employing precursors which shrink two-dimensionally into films surrounding fibers, instead of being severely cracked three-dimensionally upon ceramization.

As for the MI, a fine-tuned 'reaction-sintering (RS)' technique was developed to produce monolithic SiC with a minimal amount of unreacted silicon finely distributed as a scattered second phase [54]. The problem for conventional MI-SiC was its high temperature and irradiation instability due to the networked silicon [51]. A Nicalon-S composite with the fine-tuned RS matrix was produced and irradiated in HFIR. The composite, irradiated to 7.7×10^{25} n/m^2 at 800 °C, exhibited flexural strength of 244 ± 18 MPa, which was ~23% reduction from its non-irradiated value of 316 ± 22 MPa. These results imply further reduction of unreacted silicon and microstructural control are required to improve the irradiation stability of RS SiC/SiC.

3.4. Advanced characterization and interphase design

For the advanced characterization of the fiber/matrix interfacial properties, a modified non-linear shear–lag model was developed [55]. Combined with the single fiber push-out experiment, the model was applied to determine the interfacial properties of Nicalon-S CVI composites with PyC and ML interphases. Neutron irradiation effects were investigated in detail. No systematic changes of the interfacial shear properties for the PyC interphase were found, in contrast, a decreased frictional stress was confirmed for the ML interphase. However, sufficient friction after irradiation to 7.7×10^{25} n/m^2 at 800 °C was retained, thus demonstrating relatively good radiation stability for this ML composite. Furthermore, a significant reduction in interfacial friction caused the modification of interfacial crack path. Knowledge obtained through these characterization works will be useful for designing radiation-resistant interphases.

3.5. Slow crack growth

Understanding and predicting time-dependent deformation processes, both thermal and radiation-induced, is an important consideration for structural materials. These time-dependent processes typically occur by slow crack growth in SiC/SiC. The matrix cracks propagate due to the time-dependent elongation of crack-bridging fibers because the fibers carry the highest stresses as they bridge across cracks. Previous studies have indicated that bridged matrix cracks grow via fiber creep processes [56,57]. This is true for composite specimens when creep of the SiC matrix can be ignored. Other time-dependent deformation behaviors result when environmental interactions with the fiber/matrix interface dominate [58]. Introduction of improved SiC fibers has resulted in SiC/SiC with increased slow crack growth resistance. Accordingly, SiC/SiC with Nicalon-S fibers were tested in bending as single-edge notched bars or as compact tension specimens [59].

Several constant stress, 4-point bend tests were performed to induce slow crack growth in unirradiated materials in high purity argon (Fig. 3). The sample tested at 1300 °C failed during the test but the other specimens were removed intact from the test. Optical microscopy indicated that the cracks were growing under a Mode-I crack opening mode and can be considered as classic examples of bridged cracks. Based on the similar response of the composite to single fiber thermal creep, it was concluded that slow crack growth of SiC/SiC made with Nicalon-S fiber is controlled by fiber creep in bridged cracks. Although Nicalon-S fiber is more thermal creep resistant compared to the previous generation SiC fibers, an upper temperature limit for its long-term use in a composite material subjected to moderate stresses is still <1300 °C. Importantly, further tests will be needed to include radiation-induced fiber creep together with any other environmental effects. Efforts are currently underway to model slow crack growth behavior for these other conditions.

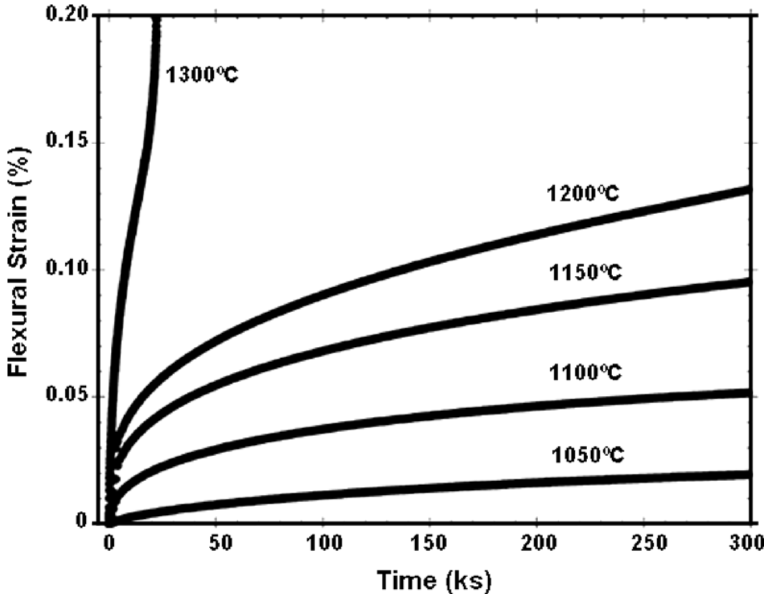

Fig. 3. Slow crack growth data for Hi-Nicalon™ Type-S CVI SiC/SiC as a function of test temperature in argon under a constant applied stress of 260 MPa.

3.6. Joining

A reliable technique of joining must be developed to use SiC/SiC as a primary structure. Joining technologies, including those through the preceramic polymer, direct reaction, and eutectic alloy routes, were overviewed previously [9]. Although recent efforts on joining development for fusion applications are rather limited, steady progress has been achieved. For instance, considering the polymer route, application of polyhydromethylsiloxane (PHMS) was shown to provide a flexible polymer chemistry and high ceramic yield [60]. Initial work demonstrated that PHMS systems with appropriate filler materials for joining SiC, particularly as a field repair technology, were viable. The NITE SiC/SiC matrix processing technique was also applied for SiC joining [61]. The 'NITE Joint' demonstrated tensile strengths in excess of 200 MPa, however, the joining process requires pressurization at high temperatures in a controlled environment. In addition, mechanical fastening of NITE SiC/SiC using screw-threaded tubes and manifolds was successfully demonstrated.

3.7. Pb–Li Compatibility

To compliment prior SiC/Pb–17Li compatibility work [62,63], a series of static capsule tests was performed on high-purity CVD β-SiC specimens by Pint et al. [64]. Studying monolithic SiC avoids SiC/SiC composites processing issues, such as fiber interfaces and porosity. Initial 1000 h exposures at 800 °C and 1100 °C showed no mass change and no increase in the Si content (30 ppma detection limit) of the Pb–Li. Subsequent experiments were conducted at longer times and higher temperatures. After 5000 h at 800 °C, no increase in the Si content was measured. However, after 2000 h at 1100 °C and 1000 h at 1200 °C, increased levels of Si were detected in the Pb–Li (185 and 370 ppma, respectively) [65]. Based on these results, the maximum use temperature of SiC composites in Pb–Li appears to be $\lesssim$1100 °C. To fully determine the compatibility of these materials, future work will need to include testing in a flowing system with a thermal gradient.

3.8. Hermetic behavior

Hermeticity is a major issue for first wall and blanket structure applications that require a pressure boundary and/or gas or fluid containment. A hermetic seal coating likely will be required for such applications, because the matrix micro-cracks, existing more or less in the as-fabricated CVI, PIP or MI SiC/SiC, cause unacceptable gas leak rates for the high pressure helium containment. For the PG3

NITE SiC/SiC, helium gas permeability was recently measured to be on the order of 10^{-10} m^2/s at room temperature [66]. It remained on the same order after unloading from tensile loading in excess of the matrix cracking stress, which indicates that the closed matrix micro-cracks do not significantly deteriorate hermeticity in this order. The helium permeability for the matrix micro-cracked PG3 NITE SiC/SiC was >3 orders smaller than that for typical as-produced MI SiC/SiC.

4. Irradiation effects studies

4.1. Radiation damage processes and microstructures

Substantial progress has been achieved in understanding the irradiation-induced microstructural evolution in β-SiC at elevated temperatures by neutron and self ion irradiations. The evolutions of various radiation defects (including tiny clusters, dislocation loops, network dislocations, and cavities) were mapped as a function of irradiation temperature and fluence [67]. The formation of black spot defects and small dislocation loops were shown to dominate at relatively low temperatures ($\lesssim$800 °C), whereas these defects grow into Frank faulted loops and finally develop into dislocation networks at a higher temperature (1400 °C). This radiation damage process is similar to that for FCC metals with low stacking fault energy [68]. During self ion irradiation, cavity formation on grain boundaries and stacking faults was observed at >1000 °C and became very significant at >1400 °C [69].

4.2. Strength of composites

The effect of neutron irradiation on strength of SiC/SiC has long been evaluated by flexural tests due to the lack of more relevant test techniques [30,70]. Primarily for studying the neutron irradiation effects, small specimen test techniques (SSTT's) for room and elevated temperature tensile properties [71], in-plane shear strength [72], and trans-thickness tensile strength [73] have been developed in fusion SiC/SiC programs. Reliable tensile properties were obtained using specimens with typical dimensions of $40 \times 4 \times 2$ mm^3; whereas the conventional tensile tests typically require >10 times larger specimen volume. Tensile tests are preferred over flexural tests because they provide more design-relevant property data as well as additional information such as ultimate tensile stress (UTS), strain at load maximum and fracture, interfacial friction, and residual stress [74].

Several Nicalon-S, PyC interphase, UD CVI composites irradiated in HFIR and JMTR were evaluated using the developed tensile test [32,75]. In the HFIR experiment, strength retention of Nicalon-S CVI composites after neutron irradiation to 7.7×10^{25} n/m^2 at 800 °C was confirmed. The flexural tests only indicated a general lack of either the composite proportional limit stress (PLS) or the severe fiber tensile strength reduction, whereas the tensile tests reveal more detailed information such as the fact that the composite UTS slightly increases while other properties such as PLS and modulus do not significantly degrade after irradiation. Moreover, the detailed analysis on the loading–unloading behavior provides insight into the interfacial shear properties and the residual stress. The results for an identical material irradiated in JMTR to at $1–2 \times 10^{25}$ n/m^2 at 800 and 1000 °C indicated similar tensile properties as that for a higher dose and 800 °C [32]. The major differences observed in non-linear behavior are due to the mitigation of residual stress by irradiation creep.

Several 2D Nicalon-S and SA3, PyC interphase, CVI composites were irradiated in JOYO fast breeder reactor (Japan Atomic Energy Agency, Oarai, Japan) up to a relatively high dose of 1.2×10^{26} n/m^2 at 750 °C [76]. In Fig. 4, representative stress–strain curves are shown. The test results indicated no statistically significant changes in tensile properties for composites with either type of fibers, which confirms that SA3 composites also are radiation-stable. Up to this dose range, tensile hysteresis analysis and fractography implied no major degradation in interfacial frictional stress. However, as a major degradation of graphite strength is expected to take place at about this dose level, irradiation to even higher doses will be essential to determine the very high dose effect on the composite strength.

4.3. Swelling and thermal conductivity

Past examinations of SiC swelling for irradiation temperatures >1000 °C are limited and suffered from the technical difficulty of determining irradiation temperature [77–79]. Recent detailed work by Snead et al. [80] examined swelling in high purity CVD and monocrystalline SiC over the 900–1600 °C irradiation temperature range for doses

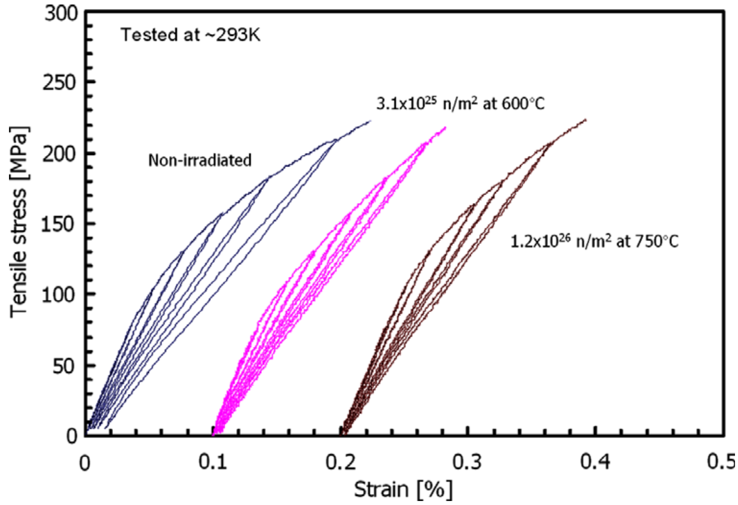

Fig. 4. Tensile stress–strain relationships for 2D Tyranno™-SA3 CVI SiC/SiC non-irradiated and irradiated to 1.5×10^{26} n/m^2 ($E > 0.1$ MeV) at 750 °C.

up to $\sim 1 \times 10^{26}$ n/m^2. Fig. 5 shows the swelling results from this study along with various historical data. Of note is that the swelling at 1100–1300 °C appears not to saturate, increasing from ~0.2% to ~0.5% as the dose increases from 2 to 6×10^{25} n/m^2. Moreover, swelling is seen to increase with increasing irradiation temperature for $T \gtrsim$ 1100 °C. At $\sim 6 \times 10^{25}$ n/m^2 the maximum swelling is ~1.5% at ~1580 °C. The peak temperature for void swelling may be at or above 1580 °C.

In the same paper, the authors correlate new data on swelling with the room temperature thermal

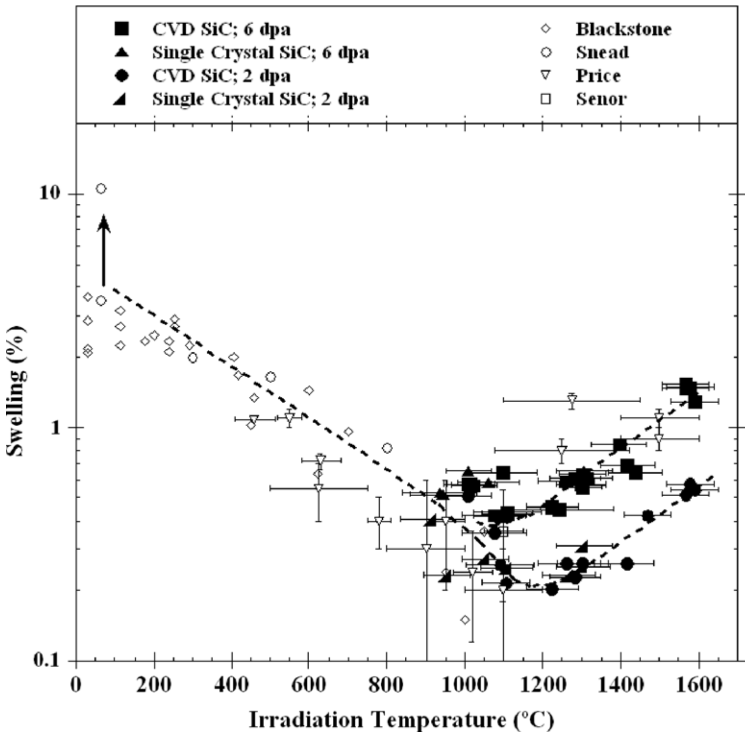

Fig. 5. Volumetric swelling of fission reactor irradiated high-purity SiC to 1600 °C.

conductivity for SiC irradiated at high temperatures as well as for material irradiated in the 200–800 °C range. As the irradiation temperature increases above 1000 °C, the as irradiated conductivity also increases. The non-irradiated room temperature thermal conductivity ~330 W/m K decreased to ~25 W/m K for CVD SiC irradiated at ~800 °C to saturation. For CVD SiC irradiated at ~1595 °C, the thermal conductivity was ~110 W/m K.

The thermal conductivity and swelling data for CVD SiC irradiated from 200 to 1600 °C were examined using the thermal defect resistance approach [40]. A clear, linear correlation is observed between the swelling and the thermal defect resistance in the irradiation temperature range from 200 to 800 °C. This clear correlation between the defect swelling (generally attributed to point defects and small clusters at these low temperatures) and the vacancy defects (which dominate the phonon scattering) suggests that the room temperature thermal conductivity can be estimated from a simple swelling measurement. The linear relationship between the thermal defect resistance and swelling breaks down for irradiation temperatures >1200 °C, which indicates that void swelling becomes dominant and that these defects are less effective as phonon scatters.

The capability of predicting anisotropic thermal conductivity for SiC/SiC composite with any given reinforcement architecture was developed based on the thermal defect resistance model [40] and constitutive models of the composites' transport properties [81,82]. The comprehensive models are able to predict the composites' thermal conductivity at various combinations of irradiation temperature, dose, and test temperature, with appropriately calibrated thermal resistance data for as-grown and irradiation-induced defects [36]. Meanwhile, further understanding of the physics of production, accumulation, and phonon-interaction of irradiation defects in SiC would greatly improve the quality of models.

4.4. Irradiation creep

For estimating the neutron irradiation creep of SiC and SiC/SiC, bend stress relaxation (BSR) technique was applied [83] to an irradiation experiment. In this experiment, thin strip samples were bent at constant strain and irradiated in HFIR and JMTR at 400–1030 °C [84]. Irradiation creep strain at <0.7 dpa exhibited only a weak dependence on irradiation temperature. However, the creep strain dependence on fluence was non-linear due to the early domination of the initial transient creep, and a transition in creep behavior was likely between ~950 and ~1080 °C. Steady-state irradiation creep compliances of CVD SiC at doses >0.7 dpa were estimated to be $2.7(\pm 2.6)\times 10^{-7}$ and $1.5(\pm 0.8)\times 10^{-6}$ $(\mathrm{MPa\,dpa})^{-1}$ at ~600–~950 °C and ~1080 °C, respectively, whereas linear-averaged creep compliances of $1\text{–}2\times 10^{-6}\,(\mathrm{MPa\,dpa})^{-1}$ were obtained for doses of 0.6–0.7 dpa at all temperatures. The larger irradiation creep compliances previously reported are attributed to the domination by transient creep component [85]. This work is ongoing, and further results will be presented as they become available.

5. Other critical issues

5.1. FCI-specific issues

For flow channel insert (FCI), low electrical and thermal conductivities are the primary requirements to lower MHD pressure drop and to maintain modest temperature at the ferritic steel/Pb–Li boundary [86]. However, optimum ranges for electrical and thermal conductivities are not only dependent on various blanket design parameters but also inter-related each other. For example, some preliminary evaluation suggest upper limits for electrical and thermal conductivity of ~100 S/m and <2 W/m K, respectively, for the 5 mm-thick FCI wall in US DCLL TBM [87]. Likewise, irradiation effects on the transport properties must be considered. Other issues for SiC/SiC FCI include: (1) chemical compatibility with lead–lithium in a flow system with strong temperature gradients, (2) hermeticity against lead–lithium, and (3) secondary stresses due to thermal expansion and differential swelling in the presence of strong trans-thickness temperature gradient [20]. Requirement for SiC/SiC for FCI application and typical values for presently available composites are summarized in Table 1. For the FCI application beyond ITER TBM, major technical issues added will be related with high fluence neutron loading and higher operating temperatures.

5.2. He/H effects in irradiated SiC

Relatively extensive production of helium and hydrogen occurs in SiC as a result of nuclear transmutation due to fusion neutrons. Calculation by Sawan and El-Guebaly indicates production ratios of 80–170 appmHe/dpa and 30–70 appmH/dpa at the FW peak radiation regions for various blanket concepts [88]. The helium effect is important because helium interacts with the radiation defects and impurities, and consequently, may alter various irradiation effects on physical or mechanical properties.

Helium/hydrogen effects have been studied mostly by means of dual/triple beam ion irradiation experiments and TEM. Helium effect on swelling was reported [89], and later the enhanced void production in the presence of helium was confirmed [69]. In fact, bubble production in dual/triple-ion irradiated SiC was confirmed at >1000 °C, whereas temperature of >1300 °C was required to produce voids in irradiated, nearly helium-free SiC. Little difference between Si/He dual-beam and Si/He/H triple-beam has been observed [90]. Recently, interesting dislocation–helium interactions were examined by a dual-ion experiment. The presence of helium promotes Frank loop production and dislocations network development, which in turn promotes long-range transport of helium along the dislocation network at 1400 °C [68]. Although much progress has been made, because in SiC the production of helium and hydrogen in the fusion environment is considerable, further investigations of helium/hydrogen effects are needed.

5.3. Solid transmutation, burn-out/burn-in

In SiC-based materials, transmutations other than helium/hydrogen production also are remaining as major issues for fusion (e.g., non-stoichiometric burn-out of SiC and burn-in of impurities including Al, Mg, Li, Be, and P [9,88]). A small amount of impurity burn-in may cause a drastic change in electrical resistivity, hence a major consideration for the FCI application. Accumulation of burn-out and impurity burn-in may cause significant changes in other critical properties; e.g., solid solution Al is known to degrade oxygen corrosion resistance of SiC. General issues related with the solid transmutation were briefly overviewed previously [9]. A most appropriate tool for studying transmutation would be a 14 MeV neutron source with sufficient dose. However, fundamental understanding may also be acquired by ion implantation experiments.

6. Conclusions

The recent status and the critical issues for SiC/SiC research and development for fusion applications, such as for use as a blanket structure in a fusion power reactor and also for relatively near-term application in ITER TBM, were reviewed.

Advanced (Generation III) SiC fiber, CVI SiC matrix composites have been evaluated as the current reference materials. Various interphase configurations and reinforcement architectures have been studied for improved radiation stability, strength, and thermal conductivity. Advanced characterization tests, including fiber/matrix interfacial friction and time-dependent deformation, have been developed. Baseline property characterization and low dose irradiation studies were completed for PG3 NITE SiC/SiC, as a promising alternate material.

Some of the previously identified critical issues for SiC/SiC for fusion applications were addressed. Importantly, after tensile evaluation of CVI composites, no significant degradation of strength properties was observed for composites up to a relatively high dose of 1.2×10^{26} n/m^2. However, potentially significant void swelling in CVD SiC at $>\sim$1100 °C was implied. On the other hand, encouraging irradiation creep resistance at 600–1100 °C was indicated. Models for prediction of the thermal conductivity for irradiated CVI SiC/SiC have been developed. Also, understanding of the fundamental radiation damage processes and microstructural evolution in high purity SiC has been advanced substantially. However, further work will be required for a definitive understanding of irradiation-effects for various forms of SiC and SiC/SiC and the operating physical mechanisms. Nuclear transmutation remains as a critical issue. Finally, issues with respect to specific applications, such as for an FCI, were introduced, and other issues will arise as specific designs are developed.

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