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Silicon carbide composite for light water reactor fuel assembly applications

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ABSTRACT

The feasibility of using $SiC_f–SiC_m$ composites in light water reactor (LWR) fuel designs was evaluated. The evaluation was motivated by the desire to improve fuel performance under normal and accident conditions. The Fukushima accident once again highlighted the need for improved fuel materials that can maintain fuel integrity to higher temperatures for longer periods of time. The review identified many benefits as well as issues in using the material. Issues perceived as presenting the biggest challenges to the concept were identified to be flux gradient induced differential volumetric swelling, fragmentation and thermal shock resistance. The oxidation of silicon and its release into the coolant as silica has been identified as an issue because existing plant systems have limited ability for its removal. Detailed evaluation using available literature data and testing as part of this evaluation effort have eliminated most of the major concerns. The evaluation identified Boiling Water Reactor (BWR) channel, BWR fuel water tube, and Pressurized Water Reactor (PWR) guide tube as feasible applications for SiC composite. A program has been initiated to resolve some of the remaining issues and to generate physical property data to support the design of commercial fuel components.

1. Introduction

The drive to improve fuel cycle cost in the last few decades has led to the increased use of zirconium based alloy materials in fuel designs. The active zone of modern fuel assembly designs are constructed almost entirely of zirconium based alloys. Since its introduction by the US nuclear naval program, significant improvement to the alloys, particularly in cladding corrosion, has been made, making it possible to design fuel assemblies well beyond the US licensing burn-up limit of 62 GWd/MTU. However, despite the gains, operational issues such as assembly/channel bow that are tightly linked with material properties remain unresolved [1–5].

In response to the recent Fukushima Daiichi accident, the high temperature performance of zirconium based alloys is now under scrutiny. Although zirconium based alloys have nominal melting point temperatures higher than 2073 K, the rapid exothermic oxidation at much lower temperatures, 1073–1473 K, is of great concern; from the point of view of hydrogen generation and heat production during the oxidation process [6]. Recent evaluations of several severe accident scenarios suggest zirconium heat of oxidation may equal or exceed that of the decaying heat at around 1473 K, leading to a sharp increase in the fuel temperature and subsequent melting of the fuel [6,7]. An alternative scenario where the cladding is replaced with a more oxidation resistant material does not result in a sharp temperature increase and fuel melting could be delayed by around 10 h as the temperature increases to around 1773 K [6,8]. The extra coping time provided by a more high temperature degradation resistant material gives the impetus to develop alternative fuel materials.

The US industry has initiated development programs to evaluate advanced materials for nuclear fuel applications. This paper explores potential use and benefit of using SiC in LWR fuel applications and focuses on developing material for BWR channel and PWR guide tube applications.

2. Rational for selecting SiC

Searching for suitable alternative materials for nuclear fuel application is a challenging problem given the complex design requirements and hostile operating environments. It is understood the new materials must meet or exceed the performance metric of existing fuel designs while providing additional benefits/relief during severe accidents as well as alleviate some normal operational performance issues. The combined requirements of acceptable neutron capture cross sections and high temperature properties eliminate most of the elements from the periodic table.

Silicon carbide (SiC) has been identified as a potential candidate material. It is not a widely known material but is used in several

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specialized applications, such as, body armor, high temperature heat exchanger, jet engine combustor, and tri-structural-isotropic (TRISO) nuclear fuel. Much of the SiC research for use in the nuclear industry has been focused on the high temperature gas reactor TRISO fuel and fusion reactor structural components [9–12], but interest to use the material for light water reactor application is increasing [13–15]. Data generated thus far suggest this material has excellent performance in LWR environments and is known to be stable to very high fast neutron fluences, after irradiation induced volumetric swelling saturates [16,17,18]. Data also suggests a substantial advantage under high temperature steam conditions, with test results indicating oxidation rates of SiC at a small fraction of zirconium based alloys at up to 1673 K [19–22]. These properties allow revolutionary performance improvements over existing materials.

3. SiC for fuel applications

3.1. Fuel cladding application

The initial concept of using SiC in LWR fuel designs started with the fuel cladding application [11]. This effort was initiated before the Fukushima accident with the intent of developing fuel products that could offer revolutionary performance improvement under normal and transient conditions. The fuel rod hermetic seal requirement necessitated the use of an inner monolithic layer. The monolithic layer was coated with a one dimensional fiber reinforced CVD infiltrated composite layer for strength. Test coupons of this construction performed well in the PWR water chemistry of the Massachusetts Institute of Technology (MIT) research reactor. However, mechanical tests indicate this composite construction may have poor fracture resistance. Additional research is in progress to increase the fracture toughness performance.

As part of a cost-benefit analysis, fuel rod design evaluations were conducted using the FRAPCON fuel performance code. Publically available CVD SiC and CVD SiC composites thermal and mechanical property data were used as input in the analysis. A key finding from this study is the need to build sufficient pellet-cladding gap into the fuel rod design to account for normal operational and transient pellet swelling since SiC is believed to be creep resistant and is not capable of plastic deformation. A persistent and larger than normal pellet-cladding gap, however, can lead to increased thermal resistance and unacceptable fuel pellet centerline temperature. Injecting a liquid metal into the gap is a possible solution but it could introduce additional manufacturing and design complications. An alternative choice explored in the evaluation was the use annular fuel pellet. Preliminary calculations indicate a minimum of 10% fuel removal is needed to bring the fuel interior temperature down to acceptable levels. In principle the solution could work, but in practice the reduced volume is insufficient to pack enough fissile content for many core designs given the 5% enrichment licensing limitation. Many of the existing core designs are already using the practical 4.95% maximum enrichment limit.

Perhaps the greatest challenge in using SiC as a cladding material is the lack of a proven solution to seal the tube once fuel pellets are loaded. Conventional welding techniques cannot be applied as the material does not melt, but sublimes/decomposes at 3003 K. Diffusion bonding that forms an intermediate phase has been tried without much success; most of the test coupons detached from each other after only a short exposure in the MIT research reactor. Cause for the failure was not rigorously investigated but differential volumetric swelling rates between the monolithic SiC base block and the intermediate bonding phase is suspected. Other solutions, such as brazing or the use of a metallic "bladder", are also being explored. However, these solutions could not take the full advantage of what SiC has to offer since the brazing or bladder materials do not necessarily have high temperature performance. Recent results from irradiation of a variety of radiation tolerant SiC joints in an inert environment at the High Flux Isotope Reactor at ORNL appear promising [23], but need to be reevaluated in the high-temperature water environments.

The lack of an end-plug sealing solution, the regulatory 5% enrichment limit, and other potential challenges driven by differential-swelling-induced stresses across the structure, significantly curtail development prospect for the material for fuel cladding application.

3.2. BWR structural application

While several BWR fuel design components could be replaced by SiC, the most prominent component is the channel, which makes up approximately 40% of the zirconium in a typical BWR core. Severe channel bow late in life, due to increased cycle length and duty [1,2], has been observed in recent years. Channel bow is primarily caused by fast neutron flux gradients and the less well understood shadow corrosion. Once the material enters the shadow corrosion mode, the rate of corrosion and total hydrogen pickup quickly accelerates. Susceptibility of zirconium based channels to shadow corrosion has been correlated with the time the channel is exposed to control blades early in life. The exact mechanism of shadow corrosion is not well understood but some form of galvanic corrosion in the presence of irradiation is suspected. The difference in the rate of hydrogen accumulation between the shadow and non-shadow faces of the channel quickly leads to differential hydrogen concentration and bowing of the structure. Significant channel bow reduces the inter-channel gaps and thus can lead to control rod insertion difficulties, see Fig. 1. Although most of the BWR channels in use today are fabricated from Zircaloy-2; the industry is working to develop alternative zirconium based alloys to address the issue. The use of Zircaloy-4 in this application has had limited success. Zircaloy-4 is thought to be less sensitive to shadow corrosion effects and has a more predictable hydrogen pickup. Lead channels fabricated from other alloys, such as Ziron and Low Tin ZIRLO™, are also being explored. Although some improvements have been made with the use of alternative materials, a complete solution to the channel bow problem has thus far eluded the industry.

3.3. PWR structural application

The number of structural components a pressurized water reactor fuel design that could be replaced with a SiC composite is

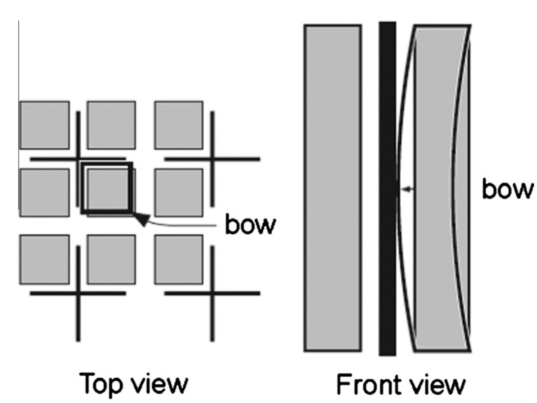

Fig. 1. An illustration of severe channel bow.

limited. The active fuel section of the PWR fuel design is made up of spacer grids connected to each other via guide tubes. The spacer grids have intricate spring forms and are typically stamped from thin metallic strip material approximately 0.5 mm in thickness. A single layer of braided SiC composite has similar thickness but it would be very difficult or uneconomical to fabricate an entire space grids using SiC composite.

The guide and instrumentation tubes, on the other hand, have simple geometries and SiC substitute can be easily fabricated with the current technology. Fuel assemblies fabricated from zirconium alloy based guide tubes are susceptible to assembly bow because of the low creep resistance of the material. Once the fuel assembly buckles, hydraulic lift and spring hold down forces can quickly cause permanent deformation in the assembly. An example of severe fuel assembly bow is shown in Fig. 2. Similar to BWR channel bow, fuel assembly bow can interfere with the insertion of control rods and can lead to premature fuel discharge. Fuel assembly bow has also lead to torn grids and fuel loading difficulties.

4. Benefit of using SiC composites

SiC is known to be irradiation stable in LWR environment, after irradiation induced volumetric swelling saturates at around one displacement per atom (dpa) damage, and therefore could resolve some of the issues discussed in the previous section and at the same time provide margin in a severe accident [14,15]. Both monolithic and SiC fiber reinforced CVD composites have demonstrated irradiation stability after volumetric swelling saturation. The SiC stability and strength at high temperatures offer several advantages over that of the zirconium based alloys. SiC is known to be stable under irradiation, while offering high temperature oxidation resistance. Also hydrogen induced degradation mechanisms common to zirconium based alloys are absent for SiC materials. Many of the proposed new regulations, such as loss of coolant accident (LOCA) [24] and reactivity initiated accident (RIA) [25], are based on zirconium cladding degradation. In both of the proposed regulations, the acceptance criteria are expressed as a function of pre-transient hydrogen concentration and become more restrictive with increasing hydrogen concentration.

Some of the benefits of applying SiC to LWR fuel applications are as follow:

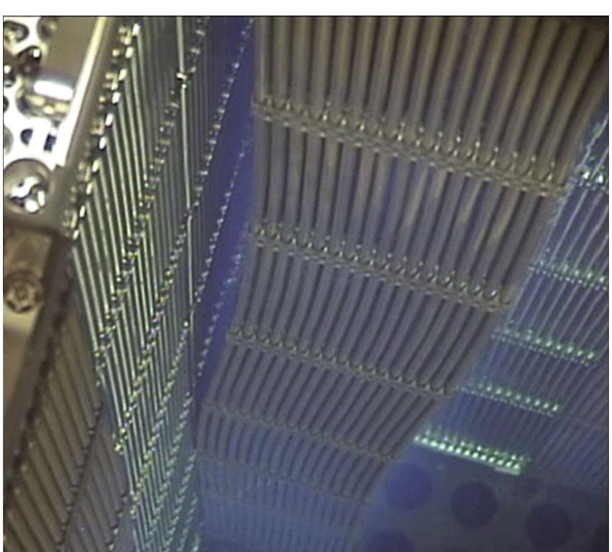

Fig. 2. An example of severe PWR fuel assembly bow.

(1) Irradiation stability

SiC composites fabricated from suitably prepared irradiation stable fibers (nuclear grade) [26,27] have demonstrated stability to very high fluences after an initial volumetric swelling that saturates within one displacement per atom (dpa) damage. The post saturation stability of the material offers revolutionary advantages over zirconium based components,

(a) BWR channels could be immune from flux/shadow corrosion induced bow after saturation
(b) BWR channels could be re-used multiple lifetimes since re-channeling of existing fuel is practiced when necessary
(c) BWR channel and PWR guide tubes would be immune from accelerated irradiation induced growth late in life

(2) Resistance to high temperature steam oxidation

The slow oxidation rate of SiC offers tremendous advantage over zirconium based alloys. At 1473 K zirconium based alloy cladding ductility would be lost after only 5 min of exposure. After approximately 15 min of exposure the fuel would not survive the thermal stresses of a quench and disintegrate upon reflood. SiC is not susceptible to the zirconium alloy based degradation mechanisms. The property of the base material will not be impacted by the high temperature exposure and degradation is through material loss via oxidation that takes place at a significantly slower rate (two to three orders of magnitude slower). Test result indicates extremely low material loss rate up to 1673 K and component integrity could be maintained for hours at this temperature [19–21]. Preliminary unpublished results indicate the material recession rate at 1873 K allows the fuel to maintain geometry for several hours or days depending on the pressure and flow velocity conditions [28].

(3) Creep resistance

SiC is known to have an extremely low irradiation creep rate relative to zirconium based alloys [29–31] while LWR temperatures are well below thermal creep rates for this refractory ceramic. Irradiation creep of zirconium based fuel designs is a key and a primary contributor to fuel geometry changes. In the case of PWR fuel, the buckling of guiding tubes creates the condition of a bending stress which eventually leads to uneven creep and permanent deformation of the fuel assembly. Assembly bow can lead to multiple issues, such as increased control rod drop time, increased core loading time, grid strap damage during loading/off-loading, and power spike from bigger water gap. The low creep rate of SiC is expected to practically eliminate permanent guide tube/skeleton bow even after buckling.

In the case of BWR channels, the higher pressure on the inside of the channel causes outward creep of zirconium based channel box faces (see Fig. 3) and result in channel bulge. Channel bulge reduces inter-channel gap and can reduce safety margin in the form of increased control rod drop time. Channel bulge could be eliminated with the use of SiC channels. With the high creep resistance it is also possible to design channels with a much thinner wall. These two features frees up valuable core volume that can be used to optimize fuel design, such as increased fuel loading.

(4) High temperature strength

One of the major obstacles to increase the regulatory fuel burn-up limit is the phenomenon of fuel pellet fragmentation, relocation and dispersal (FPFRD) of high burn-up fuel. The FPFRD phenomena have been highlighted in recent Halden and NRC sponsored LOCA tests at Studsvik Nuclear AB [32]. Although a significant number

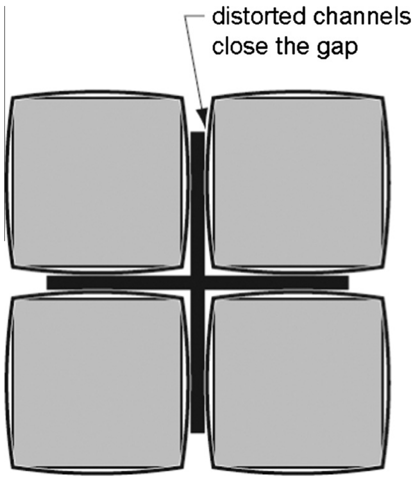

Fig. 3. Gap closure due to channel bulge.

of tests have been conducted, detailed mechanisms are still not well understood. Fuel fragmentation in itself is not an issue, but it could become an issue if significant balloon and burst materialize under high temperature accident conditions. SiC composite based fuel cladding would not be faced with these issues as it maintains high strength to well over 1773 K [33] and when it eventually fails the fiber structure ensures it would not fail catastrophically and fuel would be retained in the fuel rod.

(5) Lower neutron capture cross-section

Silicon carbide has a lower neutron capture cross-section compared to zirconium based alloys. BWR fuel designs using SiC-based channel components can have substantial savings in reduced fuel enrichment requirements. A single assembly analysis indicates the reactivity increase from the lower neutron capture cross-section of SiC is equivalent to approximately 0.1% fuel enrichment. The reactivity increase benefit can be used by decreasing the as-built U_{235} enrichment and can result in approximately $3 million in savings per reload batch.

5. Feasibility evaluation

The design requirements of potential nuclear fuel components that could be replaced with SiC were reviewed against the known properties of SiC in a feasibility evaluation. While previous evaluation suggested it may be possible to design a SiC fuel rod, the lack of a proven sealing solution shelves the concept from further consideration until a solution could be found. However, fuel structural applications are not handicapped by the lack of joining technologies. BWR fuel channel, BWR water tube and PWR guide tube are of simple geometry and in most cases are mechanically attached to the rest of the fuel assembly. Thus, at a high level, the use of SiC in LWR structural applications are promising, however, there are several significant concerns that must be addressed. These concerns are discussed in detail the following subsections.

5.1. Channel bow due to initial irradiation induced volumetric swelling

A literature review of irradiation data indicates SiC is susceptible to irradiation induced swelling, particularly large at LWR operating temperatures. Isotropic volume change from around 1.5% at 553 K to 0.6% at 1073 K has been reported, but it quickly saturates at around one displacement per atom (dpa) [17]. In the presence of a significant flux gradient, the 1.5% volume change could introduce significant channel bow for a brief period of time in the first cycle of operation. Differential irradiation induced volumetric swelling, shown in Fig. 4, was determined by calculation based on a published volumetric swelling model for two saturation volume changes and an assumed flux gradient of 25% [17,30]. A higher volumetric saturation of 1.9% was included because of large uncertainty associated with the reported test data.

To put the differential volume change into perspective, the allowable channel bow is approximately 5 mm. The active portion of a channel susceptible to bow is on the order 3800 mm. With a channel width of 140 mm, the allowable differential volume change between opposing faces is 0.12% ($8 * 5\ \text{mm} * 140\ \text{mm}/3800^2$) or 0.04% linear dimensional change. The 0.12% allowable volume change is bounded by the 25% flux gradient curve with a saturation volume change of 1.5%. An examination of a typical BWR core design indicates majority of fresh channels experience flux gradient of less than 25%. For the particular core design evaluated, about a dozen fresh channels experience flux gradient between 25% and 29% 3rd row in from the periphery positions. The use of fresh zirconium or previous irradiated SiC channels or minor core design changes could be made to accommodate these positions. It should also be noted that control rod induced channel bow for SiC-based channels is away from the control rod and thus larger channel bow may be acceptable for a short time period if the larger than expected water gap is acceptable.

Volumetric swelling is not expected to be an issue for BWR water tube or PWR guide tube since they are interior to the assembly and much less sensitive to power differentials of adjacent assemblies.

5.2. Fragmentation resistance evaluation

The fact that SiC is a ceramic material immediately conjures up images of brittle failure of common ceramic consumer products. As a fuel component, such concern must be addressed as brittle failures could not only lead to the loss of use of the component but also generate a host of foreign debris issues that could cause fuel failures during operation. For the PWR guide tube application, catastrophic failure of the component could lead to a loss of fuel geometry, rendering the fuel unusable. However, for the proposed application, the component will be used in SiC fiber reinforced composite form. Fragmentation resistance of the composite is expected to improve greatly as prior fabrication and testing of similar material indicates a pseudo ductile failure mode [34]. In this work, partial 3D SiC fiber reinforced composite BWR channel box was

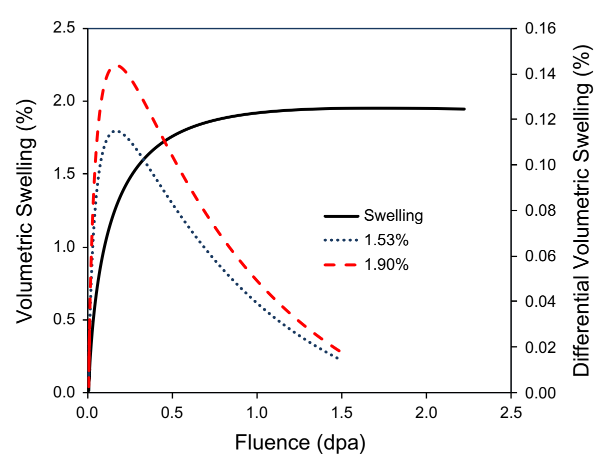

Fig. 4. Flux gradient and differential volumetric swelling.

fabricated and subjected to simulated fuel handling accident impact conditions.

5.2.1. Test article fabrication

SiC fiber reinforced SiC matrix composite (SiC_f–SiC_m) channel boxes, 10.2 cm square cross-section, were produced with irradiation stable, near stoichiometric Hi-Nicalon Type S SiC fiber by chemical vapor infiltration (CVI). A thin pyrolytic carbon (0.10 ± 0.05 μm) was applied to the fiber prior to matrix densification. This composite construction has been previously shown to be irradiation stable. The fiber preform was constructed of four layers of Hi-Nicalon Type S totaling 1.5 mm in thickness. The innermost and outermost layers were constructed of triaxial braids formed continuously around a mandrel with the geometry of the channel box inner mold line. The two layers between the braided layers were formed from high areal weight Hi-Nicalon Type S fabric (8 harness satin construction at 360 g/m^2) with an overlap joint such that the expected axial strength was >30 ksi. After preforming, a pyrolytic carbon fiber coating was applied by the thermal decomposition of hydrocarbons at elevated temperature and reduced pressure. SiC matrix densification was performed by CVI using the thermal reduction of methyltrichlorosilane to beta phase silicon carbide. CVI processing with uniform deposition on all preform surfaces was continued until ∼10% residual porosity remained.

5.2.2. Test conditions

Fragmentation type of failure during normal operation is not expected since the component would only be subjected to minor coolant turbulence forces. Significant impact loads are expected during fuel handling scenarios, particularly in handling accident situations. Three types of impact loading conditions, based on actual fuel handling equipment operational practices, were considered.

1. Normal loading impact: The component must survive normal handling impact loads without damage. This scenario is applicable to normal fuel movement where the SiC channel may come into contact with other fuel assemblies, core baffle or fuel storage structures. An impact velocity of 1.8 m/min is defined for this condition. A single impact location at middle location of the fuel assembly is assumed. The SiC based channel must not sustain any damage as the continued use of the component is expected.
2. Full speed horizontal impact: The second scenario involves a fuel movement accident in which fuel moving horizontally at 18.3 m/min impacts a stationary object. A single impact location at the middle of the fuel assembly is assumed. The fuel would most likely not be used after such an accident and so limited damage may be allowed.
3. Drop impact: The third scenario is the most severe and it simulates the side-impact of an assembly tipping over on its side in water and impacts 10 channel handles. The top span with the largest drop distance of 3.66 m was used to calculate the kinetic energy at the point of impact. This energy was then used to calculate an equivalent impact velocity of a full mass assembly in air. Damage under this scenario is expected but fragmentation would not be acceptable.

The energy of impact was scaled back 40% for all three scenarios because the nominal dimension of the test article is smaller than actual channel. Only one test article was available and it was impacted three times in the order listed above.

5.2.3. Test apparatus

Impact tests were conducted in air with the SiC test article attached to the middle of a simulated BWR fuel assembly, see Fig. 5. The facsimile fuel assembly has the same mass and external dimension of a typical BWR fuel assembly. During a test, the simulated assembly is suspended on a bridge crane and energy is imparted to the system by pulling back the bottom of the assembly and then releasing it to impact one of two types of fixed impacting surfaces. A plate approximately 15 cm in height was used for conditions 1 and 2 and a production BWR channel handle was used for condition 3.

5.2.4. Test results

No visible damage was detected after the first impact (scenario 1). A small amount of material was dislodged from the surface on the second impact (scenario 2), but no visible structural damage. The test article was punctured on the 3rd impact (scenario 3). Debris generated from impacts 2 and 3 as well as an image of the puncture from impact 3 is shown in Fig. 6.

The test article performed quite well from fragmentation resistance point of view. Most of the debris generated from the 3rd impact came from the channel handle cutting into the side-wall of the test article. There was no crack propagation even with the puncture. Preliminary consensus is that the debris generated would be acceptable but input from a wider industry audience is planned.

5.3. Silicon loss in the form of silica

Material recession, based on weight measurements, was reported for triplex tubing material tested in the MIT research reactor. The MIT research reactor has a neutron flux and spectrum similar to LWR and test samples were placed in a pressurized water loop heated to 553 K. The actual rate of material loss could not be well defined by the MIT research reactor data as the test samples were intended to evaluate material irradiation performance and the fiber ends were exposed to the coolant. Much of the material loss originated from the fiber–matrix interface. The reported material loss in itself does not present a problem as the rate is low and would not compromise component structural integrity. The primary concern is the limited ability of the current plant filtration systems to remove silica. Significant material oxidation and dissolution into the coolant could overwhelm the filtration system and excess silica could be deposited on fuel rods as crud causing fuel performance issues. Recent examinations have shown the presence of zinc silicate in crud pores. The low conductivity of zinc silicate and its preferential deposition in the crud boiling chimneys results in a tenacious crud that significantly degrades the heat transfer and if present in sufficient thickness can lead to crud induced localized corrosion (CILC) failures. Without an accurate material loss rate, it is not know if this will be an issue. Additional filtration system may be needed.

5.4. High temperature oxidation

A concern was raised on potential rapid decomposition of SiC at around 1873 K that could result in more hydrogen generation compared to zirconium based alloys. While all materials would eventually decompose/melt without adequate cooling, it should be noted that compared to the existing zirconium based materials SiC is highly resistant to oxidation at temperatures significantly higher than what zirconium based alloys could survive. The slow rate of oxidation of SiC not only slows down material degradation but more importantly the exothermic heat of reaction and the rate of hydrogen generation is also correspondingly low. Analysis has shown the exothermic heat of reaction of zirconium based

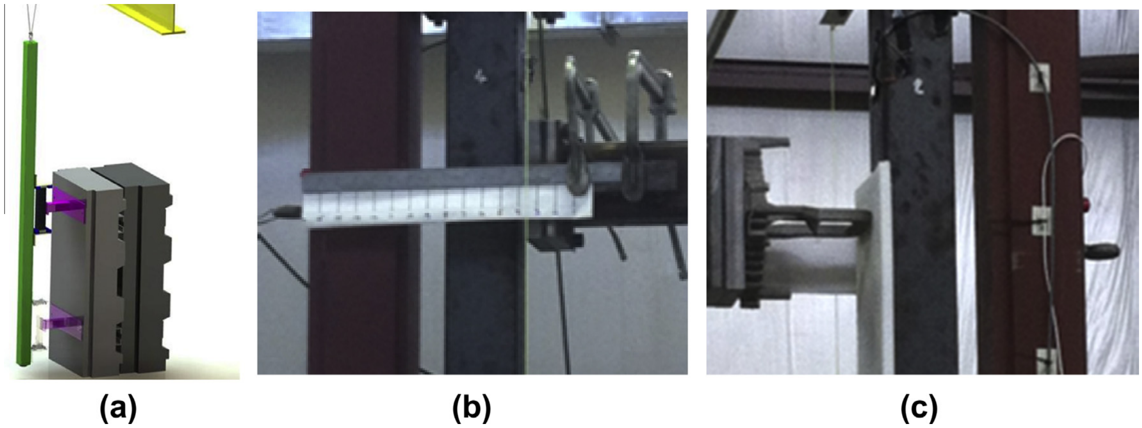

Fig. 5. (a) Model of test apparatus, (b) flat impact surface used for test conditions 1 and 2, and (c) BWR channel handle used as an impact for condition 3.

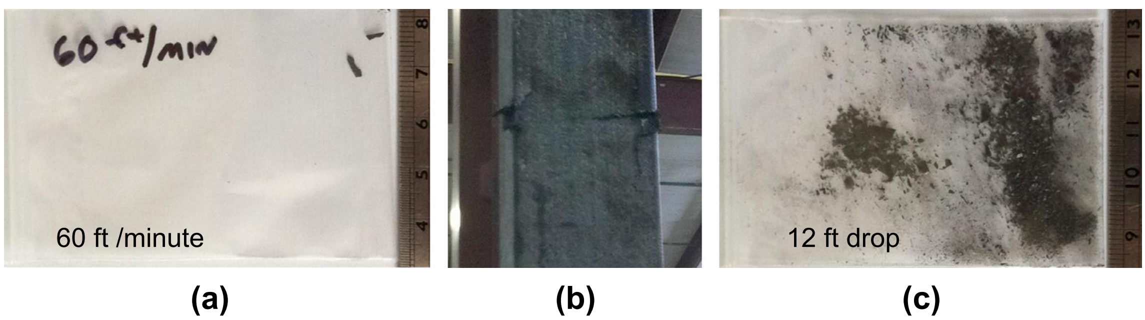

Fig. 6. (a) Debris generated from the 2nd impact, (b) puncture after 3rd impact, and (c) debris generated from the 3rd impact.

materials in the Three Mile Island accident contributed significantly to the core temperature rise. Given the same scenario core melting would have been avoided if SiC has been used as a fuel material.

5.4.1. Test apparatus

The high temperature steam oxidation tests were performed in the high temperature furnace module of the severe accident test station (SATS) at the Oak Ridge National Laboratory (ORNL). The experimental setup consisted of steam flowing inside an alumina tube heated within two $MoSi_2$ furnaces. As shown in Fig. 7a, the specimen was hung using an alumina rod from the top of the test tube. Steam was produced inside a boiler, initially enters the pre-heated furnace and subsequently flows past the specimen centered in the high temperature furnace enabling exposure at the target

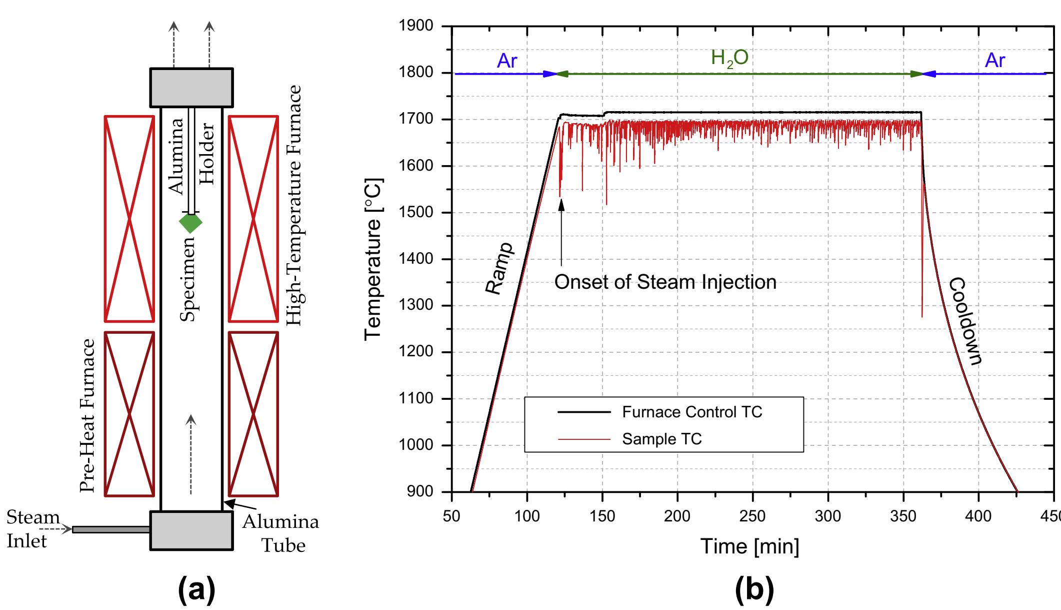

Fig. 7. (a) Schematic representation of furnace used, and (b) typical test sample heat-up profile to 1973 K.

test temperature. During the temperature ramp the specimens were exposed to flowing argon and did not experience any oxidation. Once the target temperature was reached the environment was switched from that of inert gas to pure steam. The steam flow velocity was controlled with the rate of water injection into the boiler using a pneumatic pump. Fig. 7b shows a typical temperature profile during a 4 h steam exposure test at 1973 K.

5.4.2. Test results

A limited number of test samples from the current fabrication were exposed to high temperature steam to verify its performance relative to literature results. The freshly cut test specimens were CVD coated prior to exposure to one atmosphere pressure steam at 1473, 1873 and 1973 K for 4 h. The 1473 K was selected as a reference since it is the limiting cladding temperature per US 10 Code of Federal Regulation (CFR) 50.46 that governs fuel performance and core emergency cooling system requirements in a postulated design basis loss of coolant accident (LOCA). After the 4 h of exposure, no significant weight change was detected at 1473 and 1873 K, and only a small weight change was detected at 1973 K. The results are plotted in Fig. 8. Although the weight reduction at 1973 K appears significant relative to the other two temperatures, it is mostly an artifact of the scale used. The actual material thickness loss is on the order of few micrometers. Important to note is that subsequent characterization of the composite specimens showed that the none of the high temperature exposures (even at 1973 K) compromised the outer CVD SiC layer to expose the fiber–matrix interphase.

5.5. Quench survivability

During the re-flooding phase of a LOCA event, the high temperature fuel assemblies are subjected to rapid cooling from high temperatures. To maintain in a coolable geometry, the channel material must remain intact when water is reintroduced into the core. Under such scenario, the channel may experience tremendous thermal stress as the temperature of the component is rapidly cooled from greater than 1073 K to around 408 K over a short time period. Small composite SiC tubing samples had been exposed to rapid temperature changes, such as quench in air from elevated temperature and direct quench in water from 1473 k without failure. In this work, a section of the test box 7.6 cm in length was rapidly heated to 1423 K in air in an infrared tube furnace. The test sample was introduced into the pre-heated furnace from room temperature and held at 1473 K for approximately 400 s. The heat-up and cool down profile prior to quench, measured by the specimen thermocouple, and a photograph of the test sample inside the furnace are shown in Fig. 9. The test specimen was manually withdrawn from the furnace and dropped into a bucket filled with approximately 12 l of room temperature water. As the test sample was withdrawn from the furnace there was a noticeable change in color, indicating rapid cooling of the test article. By the time it entered water, the color intensity had dim to slightly dark red, suggesting a quench temperature around 973 K. There was very little observable water boiling as the sample entered suggesting a very rapid quench rate.

The quenched sample, photograph shown in Fig. 10, showed no apparent damage. A weight measurement comparison revealed a 39 mg mass change. Upon close examination, two small pieces, approximately 5 mm^2 each in cross-sectional area, detached from one of the inner surfaces. Based on the weight loss, the missing pieces were approximately 60 μm in thickness, which is insufficient to reach the reinforcing fibers. Despite the lack of visual damage, the test article gave a different acoustic response when impacted by a striker, suggesting potential microstructure damage through microcrack development. Additional examination and testing of the quenched test article is planned.

The ability of the test article to remain structurally sound after quench eliminates the quench survivability concern. The fuel assemblies are not expected to be used again after a LOCA event and so some microstructure damage or degradation is acceptable.

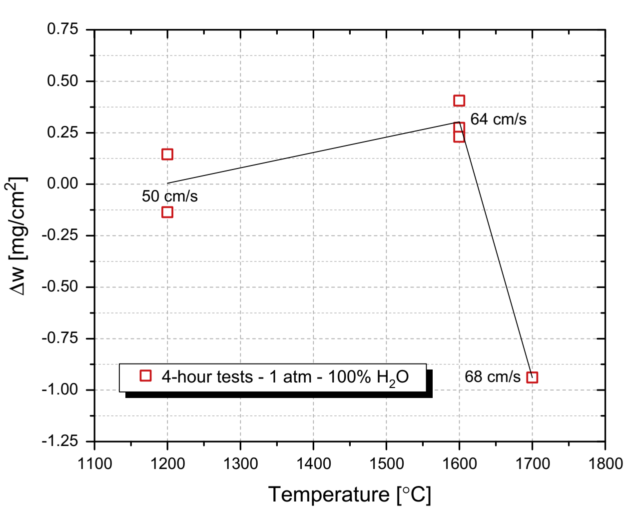

Fig. 8. Sample weight change after 4 h of exposure to pure steam at one atmosphere pressure.

5.6. Neutronic evaluation

The impact of lower SiC neutron capture cross-section was evaluated from reactivity and economic perspectives for channel wall thicknesses of 2, 2.5 and 3 mm at a nominal core void fraction of 40%. As expected all three cases showed positive reactivity change that increases with burn-up relative to a 2.5 mm zirconium based channel. An evaluation of the reactivity increase indicates the fuel enrichment may be decreased by approximately 0.1% for equivalent core reactivity, translating to approximately $3 million in savings per batch of fuel assemblies. However, whether savings could be realized will depend on the manufacturing cost of SiC based channels. Currently bulk of the SiC channel fabrication cost is tied to the expensive irradiation stable Hi-Nicolon SiC fiber. These fibers are made in small batches and key processes are at a laboratory scale. In order for the SiC channel concept to be competitive, an order of magnitude reduction in the fabrication cost of SiC fiber is needed.

A sensitivity evaluation of the effect of SiC channel on the void coefficient of reactivity was also investigated. Relative to a 2.5 mm zirconium channel, the results indicates positive change in the coefficient for 2 and 2.5 mm channel thickness and slightly negative for the 3 mm thick channel.

6. Development program

Although the thermal shock issue was not been entirely resolved in the feasibility evaluation, new data suggests the composite form of the material should perform well under thermal shock and therefore it is not anticipated to be fatal to the viability of the concept. The release of silica into the coolant could cause serious issues if not mitigated but it is possible to filter out the material with plant modifications. An extensive test program is being executed to resolve or determine extent of the issues and to generate the physical property data needed to support the design of commercial SiC-based fuel structural components.

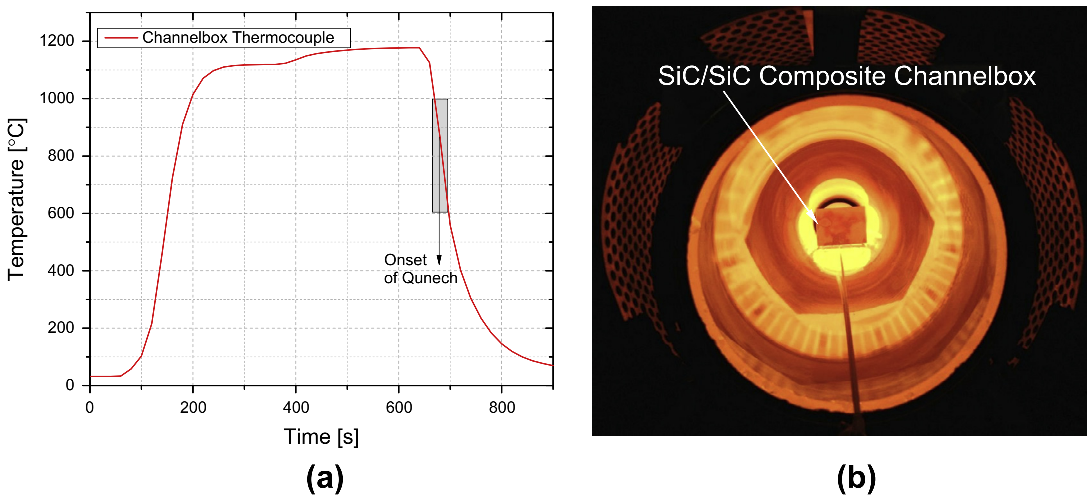

Fig. 9. (a) Channel box heat-up and cool-down profile prior to water quench, and (b) channel heated inside an infrared furnace.

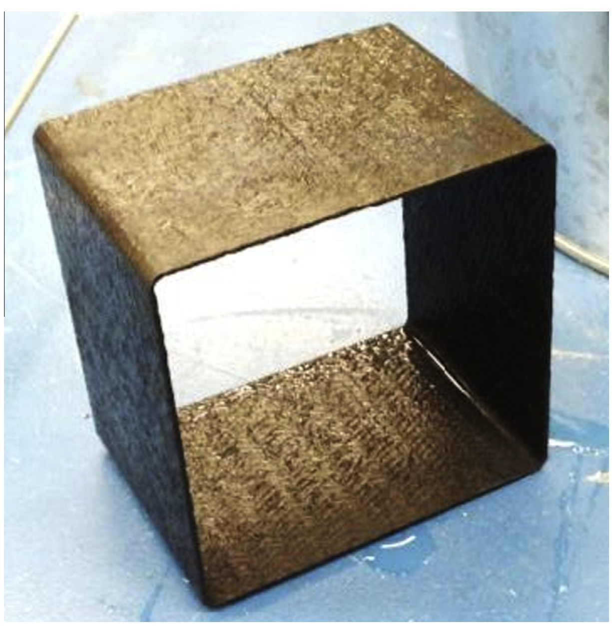

Fig. 10. Test article after quenching in water from 1423 K.

6.1. Non-irradiated out-reactor testing

Key properties to be measured are mechanical strength, density, fatigue, thermal, thermal shock and wear. Most of the tests have standard procedures and therefore details and test parameters are not described. Fatigue testing is being conducted at 573 K and at a cycle frequency of 10 Hz. The test starts with a deflection equivalent to 10% proportional stress limit and increases by 10% every 1000 cycles until failure or a maximum of 25 thousand cycles.

6.2. In-reactor testing

The bulk of the development effort is focused on irradiation effects.

6.2.1. Corrosion

Corrosion testing will be conducted at 553 K using the MIT research reactor. The MIT core offers neutron spectrum similar to commercial reactors. The pressurized water loop will have a BWR like water chemistry and a flow rate of approximately 2 m/s. The flow rate is lower than the typical commercial BWR 10 m/s and may be a source of error if erosion is significant. However, as the flow is parallel to the channel surface, the impact from erosion, if it occurs, would be small. The test samples will be 76 mm × 10 mm × 1.5 mm in dimension. Prior to irradiation, half of the coupons will be pre-conditioned in a standard ASTM G88 633 K water autoclave run for a duration of 3 days to remove any surface silica. Test sample weight will be measured between cycles and coolant silica content may be monitored via Si-31 gamma and/or CIP-AES to evaluate silica release into the coolant.

6.2.2. Irradiation induced volumetric swelling

Volumetric swelling data will be generated at the Oak Ridge National Laboratory (ORNL) High Flux Isotope Reactor (HFIR) and the MIT research reactor. Test samples approximately 25 mm × 4.5 mm × 1.2 mm will be irradiated in the ORNL HFIR Flux Trap positions at conditions outlined in Table 1. Dimensional changes in the MIT corrosion coupons will generate additional volumetric swelling data. The two sources generate complementary data at different damage levels to build a detailed map of the material behavior. It also provides a cross check of the two different testing environments. The HFIR test reactor has a flux density much higher than commercial reactors and thus it raises the question if the data is applicable to LWR conditions. Theoretically if no damage recovery mechanism is involved during irradiation, none expected for SiC, there would be no difference in the test results between the

Table 1
Irradiation induced swelling test plan.

Reactor	Temp (K)	Irradiation Damage (displacement per atom)							
		0.03	0.1	0.25	0.4	0.5	0.75	1	6
HFIR	533	X	X		X			X	X
	553		X		X				
MIT	553			X		X	X	X	

two test environments. Additionally, the test sample temperature in the HFIR reactor is controlled by sample/holder design and therefore could introduce uncertainties. Consistent test results from the two reactors would therefore validate the applicability of using the higher HFIR flux to the study of SiC for the test conditions and serve to further verify the indirect temperature measurement technique used in the HFIR test.

6.2.3. Creep

Test coupons approximately $89\ \text{mm} \times 10\ \text{mm} \times 1.5\ \text{mm}$ in dimension will be utilized in a stress relaxation test in the MIT research reactor. Two-piece sample holders with curvatures necessary to generate sample surface strains of 0.025% and 0.05% will be used. Permanent set in the sample will be used to calculate creep rates as a function of stress.

6.2.4. Small scale integral demonstration

A small scale integral test, using a box sample approximately $35\ \text{mm} \times 35\ \text{mm} \times 610\ \text{mm}$, is planned for the HFIR reactor should the test results prove the concept viable. The test sample will be irradiated in the large 76.2 mm diameter VXF position where a significant flux gradient is expected. The goal of the integral test is to demonstrate/verify at a component level the predicted behavior based on the physical data generated from the irradiation testing program. The differential volumetric swelling induced channel bow and the eventual straightening of the component once irradiation saturation occurs is of primary interest.

Post irradiation mechanical property tests are planned for the corrosion coupons and the integral test sample. Although significant material degradation is not expected with the use of High Nicalon Type – S SiC fiber [13] these tests serve to verify the material performance for the current component construction. The corrosion coupons will be subjected to a mechanical flexural test. Integral samples, in irradiated and non-irradiated conditions, will be subjected to an impact test.

7. Summary

The feasibility of using SiC composite in various LWR fuel component applications was reviewed. SiC offers many benefits over zirconium based alloys and at the same time presents a number of challenges. Initial screening indicates differential irradiation induced swelling due to fast neutron flux gradient, fragmentation and thermal shock resistance may pose the greatest challenge to the use of the material. Detailed evaluation and subsequent testing eliminated most of the concerns. Evaluations conducted thus far suggest SiC composite based BWR Channel/water tube and PWR guide tube concepts are promising. An extensive testing program is in progress to resolve some of the remaining issues and to generate data needed to support the design of commercial fuel assembly components. Should the test data continue to support viability of the concepts, decision for commercial demonstration can be made by 2015.

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