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Systematic Technology Evaluation Program for SiC/SiC Composite-based Accident-Tolerant LWR Fuel Cladding and Core Structures: Revision 2015

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August 2015

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE US DEPARTMENT OF ENERGY

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Materials Science and Technology Division

SYSTEMATIC TECHNOLOGY EVALUATION PROGRAM FOR SIC/SIC COMPOSITE-BASED ACCIDENT-TOLERANT LWR FUEL CLADDING AND CORE STRUCTURES: REVISION 2015

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ACRONYMS

2D, 3D	Two- and three-dimensional, respectively
ASTM	ASTM International; formerly American Society for Testing and Materials
ATC	Accident-tolerant core
ATF	Accident-tolerant fuel
BWR	Boiling water reactor
CVD	Chemical vapor deposition
CVI	Chemical vapor infiltration
FCCI	Fuel-clad chemical interaction
FCM	Fully-ceramic microencapsulated (fuel)
FCMI	Fuel-clad mechanical interaction
FOA	Funding opportunity announcement
FP	Fission product
HTGR	High-temperature gas-cooled reactor
LWR	Light water reactor
MIT	Massachusetts Institute of Technology
NDE	Non-destructive examination
NITE	Nano-infiltration and transient eutectic-phase
O/M	Oxygen-to-metal
PIE	Post-irradiation examination
POF	Probability of failure
PWR	Pressurized water reactor
QA	Quality assurance
R&D	Research and development
RIA	Reactivity insertion accident
RZ	Two-dimensional cylindrical coordinate
SA	Severe accident
SiC/SiC	Silicon carbide fiber-reinforced silicon carbide matrix (composite)
TMI	Three Mile Island
TRL	Technological readiness level
TRU	Transuranic or transuranium

EXECTIVE SUMMARY

Fuels and core structures in current light water reactors (LWR's) are vulnerable to catastrophic failure in severe accidents as unfortunately evidenced by the March 2011 Fukushima Dai-ichi Nuclear Power Plant Accident. This vulnerability is attributed primarily to the rapid oxidation kinetics of zirconium alloys in a water vapor environment at very high temperatures. Zr alloys are the primary material in LWR cores except for the fuel itself. Therefore, alternative materials with reduced oxidation kinetics as compared to zirconium alloys are sought to enable enhanced accident-tolerant fuels and cores.

Among the candidate alternative materials for accident-tolerant LWR's, silicon carbide (SiC) – based materials, in particular continuous SiC fiber-reinforced SiC matrix ceramic composites (SiC/SiC composites), are considered a leading option due to its outstanding benefits including exceptional radiation resistance as catalogued in extensive neutron irradiation experiments and data. In addition, they provide outstanding passive safety features in beyond-design basis severe accident scenarios.

However, it is noted that SiC composites are a relatively new class of materials, having found structural applications, primarily in aerospace, only in the last 5-10 years. To date SiC composite in nuclear structures and nuclear fuel clad designs are still in an evolutionary stage yet to define a robust architecture. In order to translate the promise of this family of materials into a reliable fuel cladding a coordinated program of component level design and materials development must be carried out with many key feasibility issues addressed a-priori to inform the process.

The present exercise lays out a plan for the Systematic Technology Evaluation Program for SiC/SiC Composite Accident-Tolerant LWR Fuel Cladding and Core Structures. The primary objective is to brovide a blueprint of a technical program that addresses the critical feasibility issues; assesses design and performance issues related with manufacturing, operating, and off-normal events; and advances the technological readiness levels in essential technology elements.

This document consists of three main elements: a technology review, a critical technology gap analysis, and a draft technical program plan. The technology review and the gap analysis were initially based on discussions during the Workshop on Accident Tolerant Fuels SiC Technology that was held in February 2014 for the U.S. Department of Energy's Fuel Cycles Research and Development Program and then updated with additional information in August of 2015 with input from the community and updated research. Many of the technical gaps identified are related with the three key feasibility issues: hydrothermal corrosion, mechanical failure, and fuel compatibility. Additional performance issues including accident-tolerance features and fission product retention during the normal operation have also been identified.

The program plan was designed to systematically address the key gap issues and is formulated in a work breakdown structure. Simultaneously, the plan is being established to set a technical program for advancing the technological readiness levels of essential technologies in three top-level categories of Design and Failure, Environmental Effects, and Off-normal Behavior. The table below depicts key technical elements of the plan and their relevance with the critical feasibility and performance issues. The present document provides precise descriptions of individual issues to be addressed, the technical approaches proposed, and how individual task elements are anticipated to interact.

Table I – Proposed high level task items and their relevance to critical gap issues for SiC/SiC composite-based accident tolerant fuel technologies for light water reactors.

◉ Primary relevance
○ Secondary relevance

Task	Critical feasibility issues				Critical performance issues
	Coolant compatibility	Cracking-induced failure	Fuel compatibility	Accident-tolerance	Fission product retention
Design & Failure					
Comprehensive analysis tool		◉	◉		◉
Statistical failure assessment		◉			◉
Fission product transport			○		◉
Design and manufacture	◉	◉	○	○	◉
Environmental Effects					
Hydrothermal corrosion	◉			○	○
FCCI and FCMI			◉	○	◉
Irradiation effects	◉	◉	◉	○	◉
Off-normal Behavior					
Steam oxidation		○		◉	○
Thermal shock		○	○	◉	○
Accident analysis		○	○	◉	○

Finally, this STEP document is that it meant to evolve based on community input and through execution and review of the technical program plan progress. The present revision reflects technical progress that was achieved since publication of the initial version in June 2014 and additional input from the research community in SiC accident-tolerant LWR fuels technologies area. Meanwhile, a transition into a phase of technology integration and implementation is anticipated upon positive assessment or successful development of mitigation technologies for critical technical feasibility issues, as schematically illustrated in Figure I.

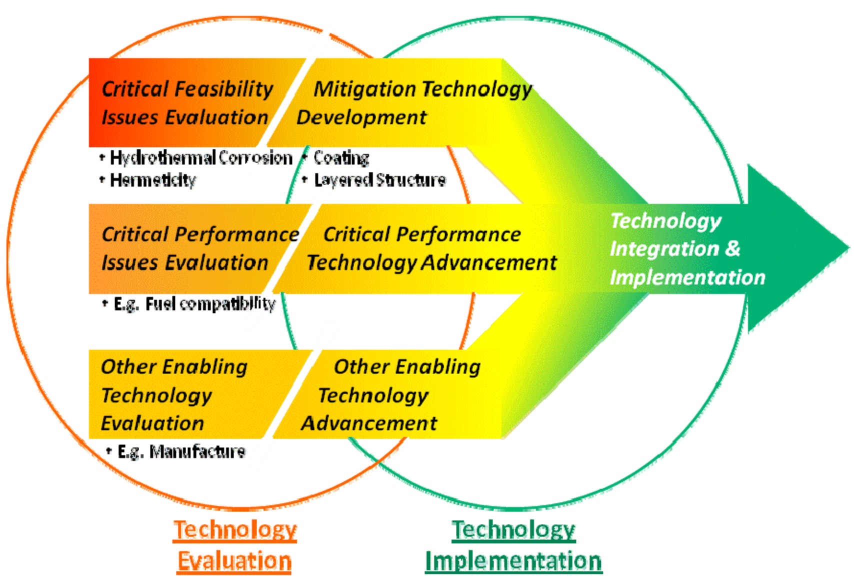

Figure I – A schematic illustration for path-forward to a transition to technology integration and implementation phase for SiC/SiC composite-based accident tolerant fuel technologies for light water reactors.

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1. INTRODUCTION

Fuels and core structures in the current light water reactors (LWR's) are vulnerable to catastrophic consequences in the event of loss of coolant or active cooling, as unfortunately evidenced by the March 2011 Fukushima Dai-ichi Nuclear Power Plant Accident [1-3]. This vulnerability is attributed primarily to the rapid oxidation kinetics of zirconium alloys in a water vapor environment at very high temperatures that results in the production of explosive hydrogen [1, 4]. Current LWR's use Zr alloys nearly exclusively as the materials for fuel cladding and core structures. Silicon carbide (SiC) – based materials, in particular continuous SiC fiber-reinforced SiC matrix ceramic composites (SiC/SiC composites or SiC composites) are among the candidate alternative materials for the LWR fuel clads and core structures to enable so-called accident-tolerant fuels (ATF) and accident-tolerant cores (ATC). SiC and SiC/SiC composites are considered to provide outstanding passive safety features in beyond-design basis severe accident scenarios [3, 5, 6]. The SiC/SiC composites are anticipated to provide additional benefits over the zirconium alloys: smaller neutron cross sections, general chemical inertness, ability to withstand higher fuel burn-ups and higher temperatures, exceptional inherent radiation resistance, lack of progressive irradiation growth, and low induced-activation / low decay heat [7]. Moreover, SiC is considered to be permanently stable in nuclear waste [7].

Although they have a unique combination of these attractive features, SiC/SiC composites are often considered a high risk development option as thermo-structural materials. This is because the use of damage-tolerant composites with brittle matrix material requires a design methodology that significantly deviates from the conventional, often code-qualified design rules that assume certain ductility for structural metallic alloys. However, it is a misconception, which is unfortunately widely spread over nuclear engineering community, that SiC/SiC composite itself is an immature engineering material as compared to metals and metallic alloy options for structural purposes and for advanced fuel cladding. In fact, SiC/SiC composites are finding specialty applications as industrial materials as the application technologies have matured in the decades since the early development [8]. Moreover, SiC and SiC/SiC composites are among the next generation reactor materials that have most extensively been studied for the effects of irradiation for nuclear applications [9].

However, application technologies for SiC/SiC composites for the LWR fuels and cores are still in early stages of development. While initial concepts were proposed starting in 1990's, serious research and concept development of SiC/SiC as a cladding has only taken place in the last 5 years [6, 10, 11]. Technically proven design concepts are still being formulated and have lacked systematic technology development efforts until recently [5]. For instance, while the potentially serious feasibility issues associated with swelling of SiC in the presence of steep temperature gradients in a nuclear environment was pointed out in 2008 [12], a critical investigation was not been initiated until recently when a modeling study [11, 13, 14] and an experiment [15] predicted and demonstrated cases for the issue, respectively. These models are currently undergoing a validation study to help develop an engineering solution to address this issue. Likewise, an assessment of hydrothermal corrosion in LWR coolant environments had not been investigated until a recent set of experiments in the MIT Reactor, in which a severe mass loss in SiC/SiC composites in the boiling water reactor (BWR) normal coolant chemistry was identified [16] although corrosion seems more limited in pressurized water reactor (PWR) coolant and BWR hydrogen chemistry coolant [17]. While these critical feasibility issues need to be fully addressed in the relatively early stages in the course of the technology development, technically credible contingency plans will also be needed.

Therefore, the main objective of the present exercise for the Systematic Technology Evaluation Program is to develop a draft technical program plan to address critical feasibility issues; to assess design and performance issues related with manufacturing, normal operation, and off-normal events; and to advance the technological readiness levels (TRL's) in essential technology elements for the SiC/SiC composite-based ATF and ATC. Note that this document is considered a living document that will continue to be updated to incorporate anticipated community input and to reflect the evolving technology until an eventual transition to a Technology Implementation Plan commences with the key technical feasibility issues addressed.

The current planning exercise consists of the following elements: a technology overview, a critical technology gap analysis, and a draft technical program plan. The technology overview will be kept brief since an extensive effort in a broader scope was recently performed by Bragg-Sitton et al. with the outcome of study published [5]. However, the general technology issues related with the previously identified critical feasibility issues are reviewed based on the recent findings and discussion. Next, the critical technology gaps were identified as technical elements and briefly analyzed. This part of the exercise was initially based on the discussion at the Workshop on Accident Tolerant Fuels SiC Technology that was held in February 2014 [18] and is updated with additional community input and newly obtained technical information. Finally, a revised technical plan for the Systematic Technology Evaluation Program for the SiC/SiC composite-based LWR fuel cladding and core structures was developed. The program plan was designed to systematically address the gap issues as identified by adopting a work breakdown structure, but also attempts to set a technical program toward steady progress in essential technology areas.

It is noted that the discussion and planning in the present exercise are limited to the concepts that uses SiC/SiC composite or SiC-based composite-monolith layered structure as the sole (or primary) structural element of the fuel cladding. Such concepts include the SiC composite-based fully ceramic claddings and SiC composite-based cladding with an additional layer or additional layers of non-SiC material(s) for certain functions including hermetic sealing (HS) and environmental barrier coating (EBC). Examples of the SiC composite-based cladding concepts are shown in Table 1.

Table 1. Examples of SiC composite-based fully ceramic and non-fully ceramic fuel cladding concepts

Class	Layer Configuration (from inner to outer)	Remarks	Ref.
Fully ceramic	Composite	Monolithic surface layer may be present	
	Composite – Monolith	Monolithic SiC layer as HS/EBC against hydrothermal corrosion	[11, 14]
	Monolith - Composite	Monolithic SiC layer as HS/EBC against FCCI	[10, 19-21]
Metal-assisted Ceramic	Metal – Composite	Metallic layer as HS/EBC against FCCI	[5]
	Composite – Metal	Metallic layer as HS/EBC against hydrothermal corrosion	
	Metal – Composite – Metal	Metallic layer as HS/EBC against hydrothermal corrosion and FCCI	
	Composite – Metal – Composite	Metallic layer as HS	[22]

2. TECHNOLOGY OVERVIEW

2.1 SIC COMPOSITE PROPERTIES

2.1.1 SiC materials overview

The basic properties of SiC materials and the effects of neutron irradiation on them have been extensively studied and summarized in Ref. [23] for high purity chemically vapor-deposited (CVD) SiC and in Ref. [24] for nuclear-grade SiC/SiC composites with chemically vapor-infiltrated (CVI) SiC-matrix and commercial Generation III SiC fibers. The general manufacture and composite design for SiC/SiC composites are well documented in Ref. [25]. A typical high purity CVD SiC is considered to represent the nuclear-grade monolithic SiC and the matrix material for nuclear-grade CVI SiC/SiC composites, because it is a high-purity, stoichiometric, and polycrystalline beta-phase SiC that is practically free from second phases including metallic silicon or carbon pocket. Moreover, the CVI SiC/SiC composites with the commercial Generation III SiC fibers are considered as the reference nuclear-grade SiC composites [24]. The Generation III SiC fibers are defined to be near-stoichiometric and highly crystallized SiC-based fibers [26] represented by Hi-Nicalon Type S (Nippon Carbon, Tokyo, Japan) and Tyranno-SA3 (Ube Industries, Ube, Japan), both of which are commercially available. While the SiC/SiC cladding will continue to benefit from further improvement of the fibers, these current commercial Generation III SiC fibers possess generally acceptable set of properties. In addition to the CVI SiC-matrix composites, SiC/SiC with the matrices processed through certain sintering processes are considered stable in nuclear environments, as demonstrated by the NITE (nano-infiltration and transient eutectic-phase) SiC-matrix composite. Basic properties and the effects of neutron irradiation in NITE composites are found in Ref. [27].

2.1.2 Previous studies

For properties of interest for CVD SiC for the LWR accident-tolerant core applications, Ref. [23] compiles comprehensive information for thermal conductivity, thermal expansion, elastic constants, hardness, fracture toughness, fracture strength, statistical fracture strength, and thermal creep. The effects of neutron irradiation, microstructural development, swelling, thermal conductivity, elastic constants, hardness, fracture toughness, and fracture strength are extensively summarized in Ref. [23]. Irradiation creep of CVD SiC has been studied in a follow-up work which furthered the understanding of creep behavior in relatively low dose regimes [28]. Steam oxidation in conditions relevant to the severe accident scenarios in LWR's has been extensively studied and is being published [29]. Hydrothermal corrosion for certain SiC ceramics and composites have been studied to some extent and published [17]. Limited results from experimental studies on hydrothermal corrosion for certain SiC composites in a radiolytic condition are found in Ref. [21, 30]. The unirradiated properties of CVI SiC/SiC composites, thermal conductivity, thermal expansion, elastic constants, in-plane and trans-thickness strength, statistical aspects of strength, fiber-matrix interface properties, damage tolerance, and rupture envelope are extensively discussed in Ref. [24]. In the same literature, the effects of neutron irradiation are also extensively discussed for swelling, thermal conductivity, thermal expansion, elastic constants, in-plane strength, fiber-matrix interfacial properties, and damage tolerance. Note that the properties compiled in the reference cited above there are primarily for flat geometry materials with two-dimensional fiber architectures, although the discussion is largely applicable to tubular geometry composite components. While not yet as comprehensive as for flat geometry materials, key thermal and mechanical properties have been obtained on tubular components [31] and progress is rapidly being made.

2.2 MANUFACTURE AND INTEGRATION TECHNOLOGY

2.2.1 Manufacture technology

As mentioned above, SiC/SiC composite densification routes that have proved to produce radiation-resistant forms of composite materials are CVI and transient eutectic-phase sintering represented by NITE. CVI SiC/SiC is a mature technology that has already demonstrated scale components with reasonable reproducibility up to large dimensions [25]. The NITE SiC/SiC has more limited experiences, but has demonstrated fabrication of complex-shaped composites such as variable diameter combustor liners, heat exchangers, and screw-ended tubes [27, 32].

Manufacture of thin-walled tubes with a large length-to-diameter ratio remains a challenge for both fabrication routes. For a CVI process, manufacture of such tubes will require a larger production facility that achieves adequate uniformity in temperature distribution and reactant flow condition along the full length of the tubes, which is technically not very difficult but will require a substantial capital investment. Recently, General Atomics became capable of producing ~1 m-long tubes through the CVI process with adequate straightness, wall thickness uniformity, roundness, and surface roughness reproducibly achieved. The Toshiba-Ibiden-Tohoku University team of Japan has developed a CVD facility that is capable of manufacturing ~1 m-long channel boxes, and is now in the process of expanding it to manufacture the full-length (~4 m) components. An alternative CVI process uses a continuous infiltration combined with a rapid densification method, which was pursued in the Ceramic Tubular Products program. NITE SiC/SiC tubes may be produced through a hot isostatic press using a furnace of adequate size. These composite tube fabrication processes will require a substantial R&D effort with the exception of the batch-based isothermal, isobaric CVI. Moreover, reduction of fabrication cost will require substantial R&D.

An alternative to the manufacture of tubes with extreme length-to-diameter ratios such as segmented tube fabrication may also be considered. Possible options may include adhesive joining or mechanical fastening of smaller length tubes to make a full-length fuel rod, and modifying the fuel assembly design to accommodate axially stacked shorter fuel rods that are individually sealed.

2.2.2 End-plug technology

For sealing the ends of the fuel rods, properly designed end plugs and technology for joining them with the cladding tube are required. The joining must provide adequate strength and gas tightness. The joining strength requirement may be relaxed in the case where a mechanical locking mechanism is incorporated. The critical enabling technology here is joining that withstands the irradiation and corrosive environments. Various technologies for joining between SiC-based materials have been studied, developed, and evaluated [33]. In the SiC composite – metal hybrid clad concepts; the required joining may be between the metallic layers instead of SiC composite. A proven radiation-resistant method for joining SiC to SiC is a selective area CVD of SiC. General Atomics has successfully developed a joining method relying primarily on the selected area SiC CVD [34, 35]. Joining of SiC to SiC by a transient eutectic-phase sintered SiC has been demonstrated for manufacture and radiation resistance [33]. However, there may be a significant difference between CVD SiC and sintered SiC in terms of hydrothermal corrosion resistance. Any joining technology that relies on non-SiC substance as the final bonding agent needs to be studied for hydrothermal corrosion and irradiation effects, in particular the likely consequences of transient swelling of SiC.

2.3 FUEL PERFORMANCE

After decades of experience with metallic cladding components in thermal and fast reactors, the transition to use SiC ceramic matrix composites represents a revolutionary paradigm shift. Due to the impact associated with any such a transition, associated challenges will need to be carefully assessed via predictive fuel performance analysis. The fuel performance analysis tools serve to guide the design process to optimize performance for the integral fuel module under normal and off-normal operating conditions. Note that when referring to "fuel", the integral structure consisting of the pellet, the cladding, and other fuel assembly components are all considered here.

The most basic fuel configuration utilizing SiC consists of urania pellets clad in an all-ceramic SiC cladding and has been the subject of some analysis in the recent years [11, 36-38]. However, it may be necessary to go beyond the basic design to identify viable fuel systems, so other configurations should undergo evaluation as well. These include alternative pellet fuel concepts such as high-density monolithic fuels (e.g. uranium nitrides, carbides, and silicides) or heterogeneous fuel forms such as the fully ceramic microencapsulated (FCM) fuel [39].

While there are sophisticated fuel performance analysis tools currently in existence for the assessment of conventional Zr-alloy clad urania rods, re-purposing these tools will be challenging for SiC composite clad and alternative fuel pellets. Largely this is because the phenomena that govern fuel behavior are different in these cases (e.g. Zr cladding creeps and anisotropically grows while SiC cladding has limited isotropic creep and swelling), and some materials property data are missing. Even with these hurdles, efforts utilizing the most recent and best available data and understanding, to model the integral performance of these advanced concepts are efficient ways to focus R&D in the right areas and guide the design process.

3. TECHNOLOGY GAP ANALYSIS

This section is intended to achieve the following goals:

i) Establish a technology evaluation structure with the main focus on the critical feasibility issues,
ii) Map the general technology issues into the evaluation structure,
iii) Incorporate other important technology issues in the unified evaluation structure, and
iv) Provide further analysis to individual issues and inter-relations among multiple issues.

The overarching objective of this section is to establish a basis upon which a systematic technology evaluation plan is to be built in Section 4.

3.1 DESIGN AND FAILURE ISSUES

3.1.1 Comprehensive performance analysis

An optimal SiC clad fuel assembly design is only possible when inputs from neutronics, thermal hydraulics, and materials are superimposed and integrated with one another. The focus of this document is on the materials aspects of the design. Meanwhile, it has been shown that if the fuel pellets remains as solid UO_2, minimal differences in reactor physics are expected [40, 41] . Therefore, an essential and useful activity is to examine the behavior of the cladding over the lifetime of the fuel to then gain insights into optimal designs for this component. As was shown by a recent study, the behavior of SiC ceramic cladding differs widely from the experience over the past six decades with Zr-based metallic cladding [11, 13] . Therefore, if experiences with Zr-based cladding are used as a guide for ceramic cladding designs, they will be based on flawed assumptions that will ultimately result in a poor design.

Therefore, what is needed is a full-scale simulation of the ceramic cladding (or metallic-ceramic hybrid structures) based on basic unirradiated and irradiated material property data and phenomena that govern material evolution. The results of this analysis shall:

a) identify sensitive material property data or mechanisms that require further or more accurate investigation,
b) provide a basis for identifying design limits, be it stresses that result in failure when coupled with information in the next subsection or properties that affect the fuel pellets or other parts of the system,
c) provide an efficient and effective tool for examination of various design configurations and reduce the need for costly fabrication and in-pile testing

3.1.2 Statistical failure assessment

Properties related with mechanical failure include the stresses for first major matrix cracking, first penetrating crack formation, and ultimate failure. Statistical aspects for these properties are insufficiently understood for nuclear-grade SiC/SiC composites. The statistical aspects include both the governing statistic laws and the statistical parameters such as Weibull parameters. This is especially important at the extremes of the failure distribution. Figure 1 highlights the need for establishing the statistical failure properties for SiC/SiC composites, showing the very low design allowable stress under a probabilistic design approach assuming the two-parameter Weibull distribution for the matrix cracking stress. The correlation between the statistical mechanical properties and the composite architecture is not adequately understood either.

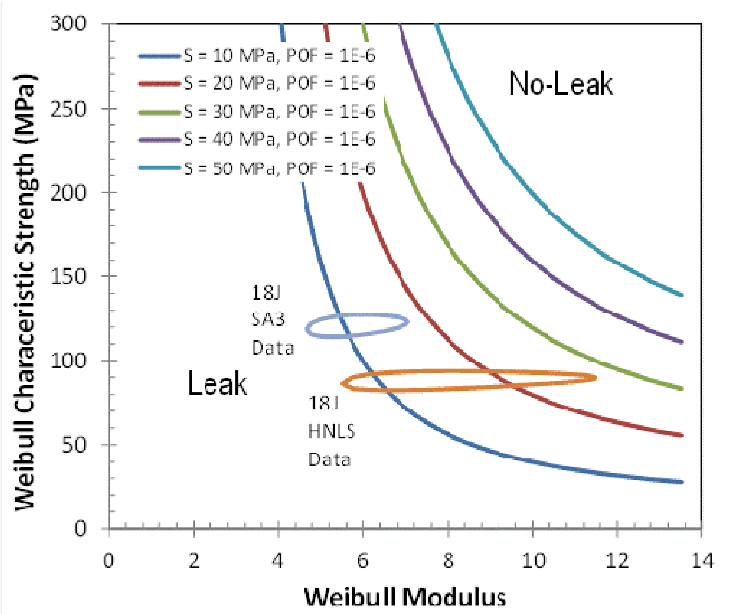

Figure 1.Design space in Weibull modulus and characteristic stress for different uniaxial tensile stress levels at an allowable probability of failure (POF) at 1 parts-per-million. Plotted together are 95% confidence ratio rings for proportional limit stresses for unirradiated CVI SiC matrix composites with Hi-Nicalon Type S (HNLS) and Tyranno-SA3 (SA3) SiC fibers

Only limited information is currently available for statistical mechanical properties of nuclear-grade SiC/SiC composites in two-dimensional orthogonal lay-up architectures under uniaxial loading along one of primary fiber orientations. However, failure behavior of SiC/SiC components in tubular geometries in common fiber architecture for small diameter tubes is anticipated to be significantly different from that for flat two-dimensional architectures. Information and understanding that are necessary to assess the failure probability, either matrix cracking or structural failure, are beginning to be addressed for SiC/SiC fuel rods. Specifically, the following knowledge is critically important for establishing a predictive capability for failure probability of SiC/SiC fuel rods under given loading conditions.

a) Definition of failure
b) Probabilistic law that the failure (as defined) follows in a given loading mode
c) Probabilistic parameters for the failure (as defined) of the test articles (made as prototypical test articles for actual components) in a given loading mode
d) Effects of test article dimensions (typically length) on failure (as defined) in a given loading mode

To achieve a) to d) above, the followings resources are essential.

e) Standard test methods pertinent to the failure/loading modes

f) Experimental data based on standardized test methods (e) with adequate sample quantities

In addition, it is preferred to achieve:

g) Theoretical understanding that supports probabilistic law (b).

3.1.3 Fission product transport

Fission product transport and interactions with SiC, SiC/SiC composites, the composite constituents, and along internal interfaces need to be understood for all gaseous and solid fission products. The FP's of primary interest include noble metals and tritium. The effects of neutron irradiation are important for both transport of the FP and interactions between the FP's and cladding structures.

Experience with SiC as a barrier to fission product transport is limited to its application in high-temperature gas-cooled reactor (HTGR) fuels in the form of a ~30-50 μm coating layer embedded in graphite layers and operating at ~1000 °C. The application under consideration here is for a 0.5-1 mm ceramic composite operating at ~400 °C. Diffusive, bulk, and effusive transport mechanisms all need to be considered. Although reactive transport can be ignored in case of noble gases and other species with low affinity for reaction with SiC, it needs to be taken into account for noble metals and other reactive species. In particular, silver and palladium fission products have been identified as species of concern in TRISO high temperature conditions. Susceptibility, rate, and the onset of fuel rod failure (defined by fission product release) all need to be examined and quantified. Transport of all hydrogen isotopes into and out of the ceramic cladding needs to be understood as well. Moreover, the transport of fission products on internal surfaces, such as microcracks and fiber interfaces, needs to be understood. All transport mechanisms must be ultimately described in a form useful for component modeling.

3.1.4 Design and manufacture

The design and manufacture issues for SiC/SiC fuel rods include engineering of tubular composite structures, developing a method of end-plugging, and manufacturing long (>1 m) tubes. These issues are specific to the pin-type nuclear fission fuels and thus have not previously been explored in other programs researching SiC such as fusion energy sciences or HTGRs. Additionally, due to concerns with cracking probability for fully ceramic clad systems and hydrothermal corrosion rate particularly in BWR normal chemistry coolant conditions, ceramic composite-metal hybrid clad concepts shall be explored. More specifically, issues that must be addressed include:

a) Identification of composite architectures for small diameter tubes that satisfy required mechanical and thermal properties,
b) Develop test methods/standards for evaluating properties of such tubes,
c) Manufacture technology for axially long (~4 m), small-diameter (~1 cm outer diameter), thin-walled (<1 mm) tubes with an adequate quality control,
d) Identification of suitable metallic systems that can be used as the crack seal and environmental barrier coating onto SiC/SiC tube surfaces and development of coating technology, and

e) Engineering technology for end-plugging that withstands mechanical stress from evolving internal pressure, neutron irradiation, and chemical environment combined.

For the end-plug development, the required technologies may further be broken down into the technology elements below.

f) Define failure, performance guidelines, and prototypical design options
g) Develop technology for joining end-plugs with a tube
h) Develop test methods/standards for joints and end-plug mechanical properties
i) Develop test methods/standards for end-plug gas tightness

Note that some aspects of these technology needs, including prototypical designs, joining, and test methods, have been studied in DOE-funded projects by General Atomics [34] and Hyper-Therm High Temperature Composites (currently Rolls-Royce High Temperature Composites). Considerations based on the development and findings in these projects will be incorporated in this evaluation program plan.

Tube manufacture may be considered an economical issue rather than an imminent technical issue. Manufacture of such tubes as described in (b) above through a chemical vapor-infiltration route is possible with the currently existing technology, although costs of infrastructure for a batch infiltration process and manufacture are both expensive. Establishing a continuous infiltration process will reduce the manufacture cost but will require a significant research and development effort, and thus should be considered as a long-range task.

The two most prominent reasons for pursuing the SiC/SiC-metal hybrid clad concepts are: 1) an external metallic coating is the surest method to mitigate issues of hydrothermal corrosion which may be a significant issue or a potential show-stopper for the fully ceramic clad system, and 2) an internal or external metallic or otherwise ductile layer could prove an effective fission product barrier, thus mitigating the issue of thermo-mechanical cracking and FP release. For 1) a hybrid concept in which a corrosion-resistant metallic layer envelops the entire fuel rod will be evaluated in this exercise. For 2) the metallic outer coating layer must retain sufficient ductility to survive the transient swelling of SiC, retain adequate bonding strength with SiC, and retain strength to withstand microcracking of SiC/SiC structural tubes. It should be noted that fabrication and characterization of these hybrid composite/metal structure are at a significantly less mature level than the underlying SiC/SiC material, although some of the fabrication and test development methodology can be leveraged. In particular exact compositions and fabrication methods for the coatings are still being explored and insufficient information is known about the thermo-mechanical and irradiative stability of the coatings. Initial evaluation of the hybrid concepts will consist of the following technical elements:

j) Identification of candidate coating materials
k) Identification of suitable coating technologies
l) Experimental assessment of coating processing
m) Characterization of coatings and coated material systems for baseline properties and effects of operating environments
n) Assessment of impact of hybridization on tube and joint developments and performance requirements

Standard test methods are obvious needs for design and manufacture of SiC/SiC composite-based cladding, though they are also needed for evaluation of the environmental effects and quality assurance through licensing. Presently only limited test standards are available for tubular geometry ceramic matrix composite components and ceramic joints. Status of the test standards development in ASTM is summarized in Table 2.

Table 2. Development status of test standards that are relevant to SiC/SiC composite-based fuel claddings in ASTM Committee C28 on Advanced Ceramics

Property	ASTM Standard	Status
Axial tensile properties of CMC tubes	C1773	Published
Hoop tensile properties of CMC tubes	WK45794	Subcommittee ballot
Flexural properties of CMC tubes		Under development
Shear strength of ceramic joint by asymmetric four point flexture	C1469	Published
Shear strength of ceramic joint by torsion	WK47657	Subcommittee ballot
Strength of end plug joint		Under development

3.2 ENVIRONMENTAL EFFECTS ISSUES

3.2.1 Hydrothermal corrosion

Recent findings from an in-pile corrosion experiment for SiC/SiC composites in the MIT reactor raised a serious question about hydrothermal reactions of these materials particularly in the BWR normal water chemistry (NWC) and in radiolytic environments [21, 30]. The hydrothermal corrosion of irradiated and non-irradiated SiC and composite constituents and the effect of radiolysis require a complete description. This requirement also applies to every cladding or core structure constituents including the fuel rod end plug and the joining materials that are exposed to coolant.

SiC recession as a result of corrosion in high-temperature water leads to i) cladding thickness loss that could in turn result in increased loading on the structure and exposure of the pyrocarbon-filled fiber-matrix interphase, and ii) deposition of unwelcome corrosion products in the primary water circuit. Therefore, it is essential to quantify the rate of SiC recession along with its dependence on environmental conditions such as pH, oxygen potential, electrochemical potential, solutes in the aqueous systems, radiolysis, etc. While solubility of quartz in pure water has received extensive attention in the geochemistry field, the thermodynamic driving forces for amorphous silica dissolution in water chemistries representative of LWR conditions are not understood. Once such a basis is established and verified against experimental observations, the focus should move toward developing water chemistry control strategies that limit corrosion to the extent that the cladding structure remains intact and the dissolved corrosion products do not present a problem to the overall system. It is also possible that the chemistry control is so challenging that any successful design will require a protective layer on the surface of the cladding that limits its corrosion.

3.2.2 FCCI and FCMI

Fuel cladding chemical and mechanical interactions (FCCI and FCMI) affect the cladding structure and alter its loading state and therefore could be design limiting phenomena.

A bulk of information is available for chemical compatibility of CVD SiC with uranium dioxide fuel in the HTGR fuels studies [42]. Since temperature at fuel-clad interface for LWR fuels is by many hundreds Kelvins lower than the operating temperature of HTGR fuels, FCCI in SiC-uranium dioxide fuel is not anticipated to involve aggressive chemical interactions. However, the fuel pellets and the cladding wall may be in direct contact in the LWR fuels, whereas the CVD SiC layer in a TRISO fuel particle is separated from the uranium dioxide kernel by a porous buffer PyC layer and a dense inner PyC layer. Although SiC has been studied for chemical interactions with other fuel systems including uranium nitride and uranium oxycarbide, the experiences in TRISO systems are not necessarily relevant with the LWR fuels for the same reason. Therefore, FCCI of SiC cladded fuels in LWR still needs to be carefully examined.

With a laboratory test on unirradiated SiC and fresh fuel as a starting point, irradiation effects as well as chemical evolution in the fuel (including increase of oxygen-to-metal (O/M) ratio and Pu production particularly at the pellet rim) all need to be taken into account and addressed. FCCI could lead to stress corrosion cracking by causing corrosion-induced flaws in the material that dramatically reduce the reliability of the ceramic structure under tensile loads. It could also selectively occur in the specific regions of the inhomogeneous composite material and alter the overall properties for the structure.

FCMI that occurs once hard contact between the pellet and the cladding is established can easily lead to cladding failure by inducing significant loads on the structure. Consideration of FP interaction (particularly noble metals) with the cladding is of immediate importance. Internal oxidation is unlikely even though oxygen potential increases significantly with burnup in the pellet since U and TRUs and certain FPs are a stronger getter of O than Si and C. Nonetheless, this point needs to be investigated, especially if susceptibility to silicide formation further increase O/M ratio in the system. The onset of FCMI needs to be determined by meticulous and integrated fuel performance modeling, and once it takes place, its impact should be examined by experimental observations based on pellet expansion inside the ceramic cladding.

3.2.3 Irradiation effects

A reasonable understanding has been developed of the temperature and dose dependence of SiC swelling under neutron irradiation. CVI composite swelling is seemingly similar to that for CVD SiC, though limited differences may exist at lower (LWR relevant) temperatures. However, the detailed swelling build-up behavior; swelling differences among CVD SiC, SiC fibers, and alternative SiC matrix materials; and irradiation creep behavior are not well understood despite their importance for modeling stress evolutions in fuel cladding.

The effects of neutron irradiation on high purity CVD SiC and nuclear grade SiC/SiC composites have been studied extensively as compared to many other emerging nuclear material systems because of technological interest in these materials for such applications as HTGR fuels and core components, fusion reactor blankets, and other advanced high-temperature concepts. However, because most of the previous interest for these applications focusing on high temperature operations (typically ~600°C and up to over 1000°C), published studies on the irradiation effects in the nuclear grade SiC/SiC at low temperatures relevant to LWR applications are rather scarce [43-46]. Reviewing these literature and the recently published article that compiled post-irradiation properties of nuclear grade CVI SiC/SiC composites [24], identifies the following knowledge gaps:

a) **Low temperature, high dose irradiation effects:** Strength and microstructural degradations following irradiation to high fluence levels at ~300°C are apparent for Hi-Nicalon Type-S fiber composites [47]. However, adequate irradiation data relevant to LWR fuel lifetime (>~ 8 dpa) are lacking. Moreover, possible core structures such as BWR channel boxes and core constraints made of SiC/SiC will offer a competitive advantage if it is demonstrated that SiC/SiC core components can withstand substantially longer in their operating environments. It is worth noting that the microstructural examination of highly irradiated composite materials suggests that the radiation instability issue may be specific to the Hi-Nicalon Type S fiber. Therefore, composites made with alternative near-stoichiometric SiC fibers such as Tyranno-SA3 and SCS-Ultra should be examined for high fluence irradiation stability for use in non-cladding core components.

b) **Dimensional evolutions:** Swelling and irradiation creep determine the dimensional evolutions of SiC-based materials in the LWR irradiation environment, at temperatures where thermal creep does not have a noticeable effect. Accurate determination of fluence-dependent and temperature-dependent evolutions of swelling is very important for evaluating SiC/SiC fuel clad and core structures because differential swelling in temperature- and flux-gradients is a major concern for these components [11, 13]. Existing data for swelling of SiC and SiC/SiC composites in LWR-relevant irradiation conditions suffer substantial uncertainties with regards to accuracy of absolute swelling measurement and irradiation temperature determination [23, 24]. Irradiation creep is likely less important than swelling because the anticipated strain caused by creep is significantly smaller than the swelling strain [28]. However, reliable irradiation creep data need to be acquired in the LWR coolant temperature range since the irradiation creep compliance for SiC is reportedly more significant at a lower temperature and could therefore help relax the internal stresses that result from the temperature/flux gradients. Additionally there have been no experimental data reported for such low irradiation temperatures. Irradiation creep should also be determined for relevant SiC/SiC composites, although the reported creep compliances for composites are smaller than those for CVD SiC.

c) **Irradiation effects in joints:** In case a joining material other than fully crystalline and stoichiometric SiC is used to affix the end-plugs in place, the response of the joint to irradiation must be fully understood. As the internal pressure evolves in the fuel rods, the end-plug joints will most likely be the weakest link for possible irradiation creep-induced failure of fuel rods. Currently no irradiation creep data are available for non-SiC joints that are being developed and/or evaluated for end-plugging of SiC/SiC composite fuel clads. In case the SiC/SiC-metal hybrid clad concepts are pursued, end-plug joining based on the coating material may be considered, requiring evaluation of irradiation creep of the particular joining material

3.3 OFF-NORMAL ENVIRONMENTAL EFFECTS ISSUES

3.3.1 Steam oxidation

Until recently, the greatly superior oxidation resistance of SiC materials in pure steam environments compared with Zr-based alloys was an assumption based on data from studies related to aerospace applications; however, notable recent experimental observations confirm the assumption. SiC recession kinetics in atmospheric pressure steam up to 1700 °C have been shown to be at least two orders of magnitude slower and in high pressure steam (up to 2 MPa and 1350 °C) roughly one order of magnitude slower than Zr alloys [4]. Although the enthalpy of oxidation of SiC and Zr alloys is similar [48], the slower kinetics limits the extent of heat generation as a result of oxidation and provides additional safety margins during the course of a severe accident. The areas that require further characterization are oxidation of composite structures with joints and flaws under stress to examine the ability for these structures to preserve their geometry and maintain coolability. Note that the ability to retain fission products inside of the cladding, although desirable, is not a regulatory requirement during design-basis and severe accidents. If metal-ceramic hybrid structures are to be considered to achieve optimal operational performance under normal conditions, the integral high-temperature steam oxidation behavior of these structures will require investigation.

3.3.2 Thermal shock

Thermal shock behavior of SiC composite structures needs to be understood so as to be able to address that the complex contributions from the following: geometrical constraints of the components, irradiation-induced thermal conductivity decrease and its recovery during an off-normal high temperature excursion (high temperature annealing of defects), and combined mechanical loading during the emergency cooling process. It is noted that thermal shock behavior and resistance for SiC/SiC composites have been studied for non-nuclear applications fairly extensively.

To guarantee that the core remains controllable and coolable under design basis and beyond design basis accidents, demonstration of thermal shock resistance for SiC fuel assembly structures is necessary. Preliminary studies have been performed to date on this subject [49, 50], however those studies need to be extended to consider irradiated SiC materials and pressurized cladding structures.

3.3.3 Accident analysis

Once oxidation kinetics, degradation modes, and integral structure response (capturing both physical and chemical degradation modes) under high-temperature, high-pressure, steam environments have been quantified and understood, this information needs to be injected into severe accident (SA) analysis tools to predict core behavior under such postulated events. SA analysis tools are necessary to identify limiting properties and configurations that reduce response time and safety margins. They are also an effective tool to consider various scenarios in an efficient manner and produce remedial actions and procedures to mitigate them.

3.4 CRITICAL FEASIBILITY ISSUES

Among the technology gaps discussed above, hydrothermal corrosion and swelling-induced cracking are identified to be the critical feasibility issues for the SiC/SiC composite LWR fuel cladding and core structures.

Table 3. Critical technical feasibility issues for SiC/SiC composite LWR fuel cladding and core structures emphasized or identified after release of the previous report [5]

Feasibility issue	Description
Hydrothermal corrosion	CVI SiC/SiC composite appeared to undergo unacceptable rate of mass loss in BWR water as evidenced by MIT Reactor experiment. Silicon once dissolved in the primary coolant water is very difficult to retrieve.
Swelling-induced cracking	High temperature side of SiC/SiC fuel cladding is subjected to high tensile stress due to differential swelling of irradiated SiC. Irradiation-induced thermal conductivity decrease in SiC worsen situation.

These two issues need to be either verified to be non-critical or addressed to be resolvable with credible technical pathways identified before the development program transitions from the systematic technology evaluation to technology implementation. In addition to these two issues, issue of fuel temperature rise due to the low thermal conductivity and irradiation-induced swelling of SiC fuel cladding is identified to be an outstanding critical issue, though this is related more closely with the design of fuel itself. These three issues including the fuel temperature are discussed in more details below.

3.4.1.1 Hydrothermal corrosion

Thermodynamics always favors formation of silica (SiO_2) when water is put in contact with SiC under environments characteristic of normal operation in light water reactors. This is also the case for Zr-based alloys where zirconia (ZrO_2) is formed on the surface of the cladding. However, unlike zirconia, silica dissolves in water thereby forming silicic acid, $Si(OH)_4$. Thus, while Zr-based alloys experience mass gain as a result of oxidation, the oxidation-dissolution process experienced by SiC materials results in mass loss with time. In both cases the base material is being consumed and the rate of base material consumption is key, since it determines the remaining thickness of the material in the cladding that continues to bear the applied loads and act as the first barrier to fission product release.

A number of out-of-pile autoclave tests have been performed to date that show mass loss in SiC materials as a function of time [51] . While the slowest mass loss rate was observed for CVD SiC, significantly higher rates were observed for sintered or reaction-bonded SiC variants. More pertinent to the fuel cladding application are the in-pile corrosion tests and autoclave tests on irradiated samples. While such data are rare, a recent in-pile corrosion test on CVI SiC/SiC composites at the MIT reactor under BWR NWC conditions (Figure 2) and an autoclave test in Japan with ion-irradiated CVD SiC samples both indicate accelerated corrosion with irradiation. Previous in-pile tests at MIT reactor with open-ended composite samples where the fibers were exposed, in coolant chemistry close to PWR conditions (low oxygen potential) showed comparable corrosion with autoclave tests. Therefore, one can postulate that the radiolysis, products of which are much more significant under high oxygen potential conditions, has an important effect on SiC corrosion. On the other hand, the accelerated corrosion observed in the autoclave tests of ion-irradiated specimens indicate that radiation defects in SiC can play a role as well.

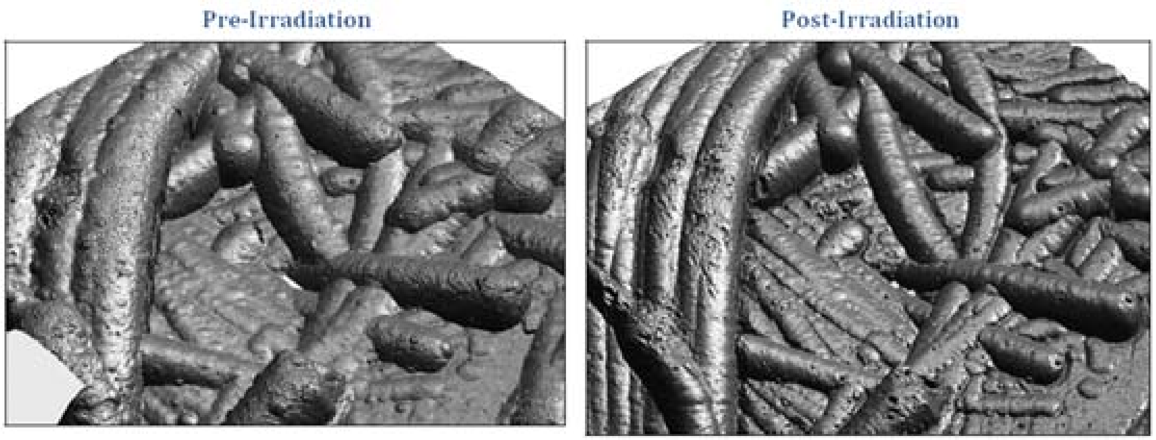

Figure 2. Surfaces of SiC/SiC composites before (left) and after (right) exposure to BWR normal water chemistry in MIT reactor showing uniform recession of CVD SiC overcoat layer. 70 EFPD x ~2.5×10^{14} n/cm^2/s (~1.2×10^{14} n/cm^2/s for E > 0.1 MeV) over ~90 days [30]

The current state of knowledge in the area of SiC corrosion in LWR coolant environments is very shallow, presenting what is perhaps the biggest gap at the moment for this technology. Systematic tests with well-defined samples under well-known conditions aiming to quantify the kinetics and identify the mechanism for corrosion are necessary.

3.4.1.2 Swelling-induced cracking

Fully crystalized SiC ceramics and composites undergo irradiation-induced swelling. The swelling occurs quickly after the start of irradiation and approaches saturation values before 1 dpa is reached [24] (Figure 3). The magnitude of saturation swelling is inversely correlated with irradiation temperature, ~1% at 600°C and >2% at 300°C. The swelling of SiC accompanies a significant decrease in thermal conductivity because both of these irradiation-induced phenomena result from the accumulation of matrix defects. The differential swelling-induced stress or strain has been identified to be a serious threat for SiC-based fuel cladding in which a strong temperature gradient across the tube wall is inevitable due to the radial heat flux. Recent analysis by Ben-Belgacem et al. [13] also predicted very substantial tensile stress near the inner surface of cladding wall. While swelling itself does not result in material microcracking, the resulting thermomechanical stress can result in crack initiation and/or propagation. It is also noted that SiC composites can possess intrinsic microcracks, such that irradiation-induced stress may result in crack propagation.

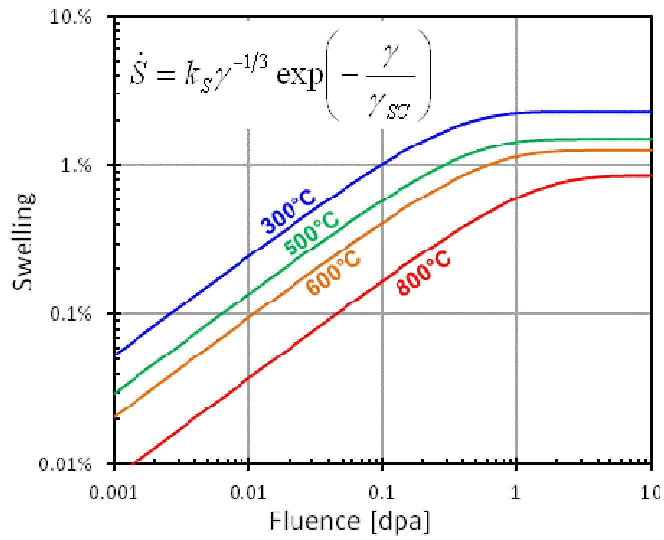

Figure 3. Trends in volumetric swelling for SiC and SiC/SiC composites as a function of fluence and irradiation temperature showing temperature-dependent early saturation for swelling [24]

Recently, Morris et al. performed an examination of a "triplex" SiC-based fuel cladding with standard UO_2 fuel pellets encapsulated following irradiation in the High Flux Isotope Reactor for 403 full power day in VXF-4 position (20 GWd/t, 1.378×10^{21} n/cm^2 thermal) [15]. A large number of cracks were observed that are aligned in the radial direction within the inner monolithic layer. This inner surface is where a strong azimuthal tensile stress (hoop stress) is expected to develop due to differential swelling across the cladding thickness. As expected, due to the limited swelling of the SiC cladding, there was no evidence of fuel-clad mechanical interactions and the fuel pellets slid out of the tube without friction after disassembly. The average design temperature for clad during irradiation was 380°C. The actual irradiation temperature of the clad has not been evaluated.

The uncertainty in actual operating temperature and its gradient across the wall thickness as well as relative swelling rates of composite versus monolithic material in this experiment implies a finite possibility that the issue of differential swelling-induced cracking may be exaggerated (or understated) in this observation. However, the result of the model prediction of stress evolutions in SiC/SiC-based cladding [13] and the current evidence of cracking make a strong case that the differential swelling-induced cracking is a critical feasibility issue for fully ceramic cladding. More detailed investigation into this issue will require comprehensive modeling of fuel and clad behaviors, and a detailed and accurate understanding of swelling, irradiation creep, thermal conductivity change, and statistical failure of the clad material. Moreover, this potential weakness highlights the need to explore clad design options involving engineered and highly controlled monolithic hermetic seal coats and/or seal coating with ductile metallic materials as an alternative approach to address swelling-induced cracking and hydrothermal corrosion simultaneously.

3.4.1.3 Fuel temperature

Irradiation-induced swelling of SiC causes not only stresses arising from differential strain but also results in an upward shift in the fuel temperature profile by two mechanisms: i) the cladding swelling increases the pellet-cladding gap and ii) the low irradiated thermal conductivity of SiC causes a large temperature drop across the cladding (roughly 5 times larger temperature drop than in the case of Zr-based metallic cladding). This increase in the fuel temperature is considered a critical issue since it exacerbates fission gas release from the pellet and could result in limited power rating for the fuel to avoid melting at the pellet centerline which is a design limit by regulation. These concerns were emphasized by Markham et al. in the Workshop. This issue calls for development of alternative fuels that offer higher thermal conductivity and/or improved stability and modified fuel geometry (such as hollow pellets) designed for reduced operating temperature as was investigated in Ref. [40]. Note that the increase in pellet-cladding gap early in life has a positive feedback as well: delaying hard contact between the two to higher burnups compared to Zr-based cladding.

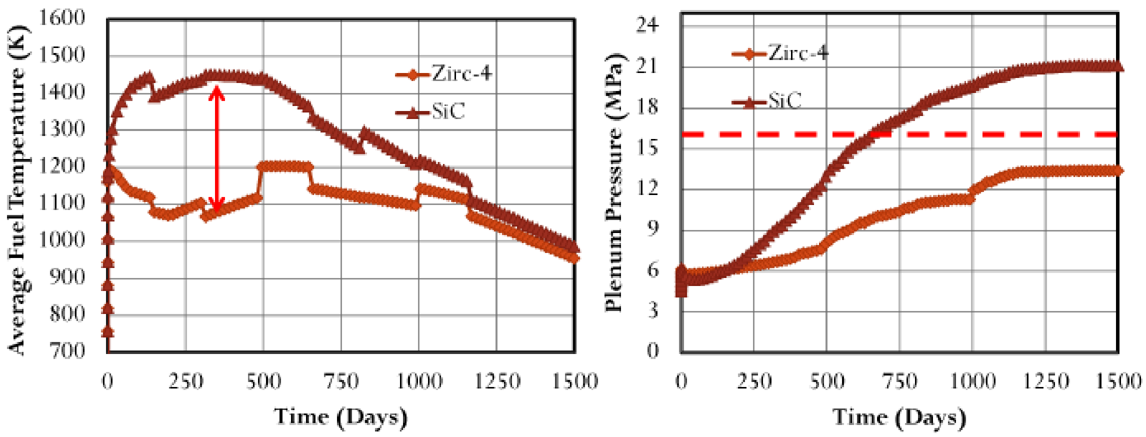

Figure 4. Effects of irradiation-induced thermal conductivity decrease and swelling of SiC-based cladding on fuel temperature and plenum pressure predicted by FRAPCON simulation [52]

4. SYSTEMATIC TECHNOLOGY EVALUATION PROGRAM

In this section, technical work elements that need to be incorporated in the Systematic Technology Evaluation Program for SiC/SiC composite-based LWR accident-tolerant fuels and cores are summarized, based on the critical technology gaps discussed and identified in the previous section, in form of work breakdown structures. The following discussion identifies individual, near-term issues to be addressed, what technical approaches are proposed, and, to a limited extent, timing in cases where the order of technical steps is important. Relative importance of individual breakdown tasks is occasionally mentioned. It should be re-emphasized that the Program is intended to:

i) Address the key feasibility issues as currently identified and establish a foundation to build a more comprehensive technology development program in the near future, and
ii) Contribute to advancing the technological readiness level of this technology and ensure a smooth transition to the anticipated follow-up *technology implementation program*.

The proposed work breakdown structure consists of three top level task categories, namely 1) Design and Failure, 2) Environmental Effects, and 3) Off-normal Behavior. In addition, the plan for additional works that are required to confirm feasibility and develop necessary technology for 4) hermetic coating on SiC-based cladding structures is placed as the fourth top level item. Progress that has been achieved as of August 2015 is noted.

4.1 DESIGN AND FAILURE

The objectives of this top level task are to 1) establish a set of tools and perform comprehensive analysis of the performance of fuel systems with SiC/SiC composite-based claddings, 2) provide ceramic composite science-based analysis for failure probability of SiC/SiC composite clad tubes, 3) provide preliminary analysis to fission product transport and interaction issues for SiC/SiC composite cladding systems, and 4) provide assessment of the state-of-the-art clad tube design and manufacture technologies. These subtasks will collectively provide more decisive information with regard to the probabilistic cracking issue for ceramic claddings and the fuel temperature-related issues, while advancing the general design technologies for the pertinent accident-tolerant fuels. The proposed work breakdown structures in this category are shown in Table 4.

4.2 ENVIRONMENTAL EFFECTS

This top level task category is intended to address the environmental effect issues relevant to normal operation of reactors including: 1) the hydrothermal corrosion, 2) the fuel – clad interactions (including the fission products – clad interactions), and 3) the effects of neutron irradiation. These phenomena are central to understand the behavior of the pertinent cladding systems during operation, limit operational life for them, and predict end-of-life properties. These are important not only as the critical feasibility factors, contributing to address the clad integrity issues and the fuel behavior issues, but also for general safety, economy, and back-end processes. The proposed work breakdown structures in this category are provided in Table 5.

4.3 OFF-NORMAL BEHAVIOR

The Off-normal Behavior top level task category is intended to address the effect issues relevant to accident scenarios based on an analysis of core and fuel evolutions in events of severe accidents. The critical issues that have previously identified include the effects of exposure to very high temperature steam and a quench arising from emergency cooling on mechanical and geometrical integrity of the fuel cladding, assembly, and core structures. However, the accident analysis may reveal additional technical issues to be addressed in future. The proposed work breakdown structures in this category are provided in Table 6.

4.4 HERMETIC COATING

As discussed earlier, there is finite possibility that the fully SiC ceramic fuel clad and/or core structures concepts do not achieve the critical performance requirements including the containment of gaseous fission product and corrosion resistance against high temperature water, particularly of the BWR chemistry. In such a case, solutions will have to be sought toward 1) application of environmental barrier and seal coating with a ductile material (most likely a metal or a metallic alloy) as discussed in Section 3.1.4; 2) modification of SiC-based materials to enable in-situ oxidation/cracking protection through incorporation of a stable oxide former; and 3) modification of the primary water chemistry to achieve acceptable corrosion rate for SiC. Among these directions, application of a coating is currently envisioned as most technically feasible in the near term. Thus, examination of fundamental aspects of coatings was included in the evaluation plan. Since the application of coating will impose a significant impact on various aspects of the technology development, the anticipated influences on the evaluation plan are discussed for individual tasks in

Table 7. More detailed assessment on this issue is planned as a part of the evaluation program, Task 1.4.3.5.

Table 4. Task structure, objectives, and outline in Design and Failure category

No.	Task Name	Description	Status
1	**Design and Failure**		
1.1	**Comprehensive Analysis Tool (MOOSE)**		
1.1.1	Implementation of SiC clad model in MOOSE	1) Implement the current 2D cylindrical (RZ) finite element model in MOOSE, 2) Transition from 2D to 3D analysis, and 3) Implement fuel pellet behavior and pellet-clad interaction (PCI)	Responsible principal researcher identified and engaged for the FY15 work, which is currently in progress.
1.1.2	Steady state analysis	1) Perform steady state RZ analysis as a function of burnup for various design configurations and 2) Perform steady state 3D analysis as a function of burnup for various design configurations (axial and azimuthal flux variation)	Responsible principal researcher identified and engaged for the FY15 work, which is currently in progress.
1.1.3	Transient Analysis	1) Perform pin level power ramp analysis for SiC clad and 2) Perform pin level RIA analysis for SiC clad	Not initiated.
1.2	**Statistical Failure Assessment**		
1.2.1	Definition and Tools		
1.2.1.1	Define failure and performance guideline	The objectives of this task are to provide a definition of fuel cladding failure, the failure modes of interest, and guidelines for mechanical performance of fuel cladding. The cladding failure will be defined by either/both of a loss of gas tightness or/and structural failure. It is also important to determine whether or not commonly used failure criteria such as the proportional limit stress and stress for the first major acoustic emission event may be used as measures for loss of gas tightness. The failure modes of primary interest will be determined based on input from Task 1.1, though they are most likely axial/hoop tension. The performance guidelines will define the allowable failure rates (both gas tightness and structural failure) for full-length articles.	Preliminary definitions and statistical analysis of clad failure are complete. A report on Preliminary Failure Probability Estimation for Fully Ceramic Clad Option was issued. This work is continuing with additional input from Task 1.1.
1.2.1.2	Develop test	This task develops full-consensus test methods for the	Axial tensile test standard for ceramic composite tubes

No.	Task Name	Description	Status
	method/standard	failure modes defined above. ASTM Committee C28 will be chosen as the national standard authority for test methods for ceramic matrix composites. The test method development includes initial publication of an ASTM test standard (through drafting/ballot/revision cycles), determination of precision and bias (through a round-robin program), and a revision to the published standard to add a section on precision and bias. The round-robin testing program will likely be combined with the statistical properties data generation discussed in Task 1.2.2. The work items initially considered will include a) axial tension of tubes, b) hoop tension of tubes, c) acoustic emission, and d) gas permeability.	has been published as an ASTM standard. Hoop tensile test standard for ceramic composite tubes completed the initial drafting and is prepared for the spring ballot at subcommittee level in ASTM C28.07 on Ceramic Matrix Composites. Discussion was initiated toward development of ASTM standards for acoustic emission and gas permeability in ASTM C28 on Advanced Ceramics.
1.2.2	Statistical failure characterization (for each failure mode)		
1.2.2.1	Test article and fixture procurement	This task will develop a specification for procurement of test articles for failure modes of primary importance as determined in Task 1.2.1, place procurement orders, and visually inspect the test articles when they are received. Test fixtures that will be distributed to the round-robin studies will also be procured.	Procurement specification development and identification of a vender are complete. Contract was recently placed.
1.2.2.2	Lead-tests for round-robin studies	Perform lead-tests based on draft ASTM test standards and establish test procedures.	A test specification and procedure document was developed for the round robin testing for axial tensile properties of ceramic composite tubes based on testing of SiC composite tubes that had been previously procured at ORNL. A similar lead-test will be performed after the newly procured tubes are received.
1.2.2.3	Coordination and execution of round-robin studies	This task will identify the participating institutions and responsible personnel for the round-robin studies, develop the test procedure documents and report forms based on the lead-test exercises, coordinate the round-robin procedures, ensure that the tests are performed according to the draft test standards and the procedure	Potential participants have been identified.

No.	Task Name	Description	Status
		documents, and collect results from the participants.	
1.2.2.4	Report on test data acquisition and analysis	This task will reduce and analyze the results from the round-robin studies, provide feed-back on the precision and bias section to the pertinent ASTM standards, and provide a recommendation of the probabilistic theory and parameters to be used for failure probability analysis.	Will start after completion of the round-robin studies.
1.2.3	POF-based feasibility assessment for ceramic tube structures		
1.2.3.1	Report on POF-based feasibility assessment for ceramic tube structures	This task will perform an assessment of technical feasibility for fully-ceramic fuel cladding systems based on the probabilistic failure properties of prototypical test articles, size scaling of failure stresses, and the requirement guidelines for fuel cladding performances.	Will start after completion of the analysis of round-robin results.
1.3	**Fission Product Transport**		
1.3.1	Issues analysis	Analyze and report fission product transport. Linked with gaseous FP transport in 1.2.1.2.4 and solid FP interactions in 2.2.1.	Responsible principal researcher identified and engaged for the FY15 work, which is currently in progress.
1.4	**Design and Manufacture**		
1.4.1	Tubular CMC structures		
1.4.1.1	Composite tube design	This task identifies the reinforcement architectures and composite fabrication methods for small diameter tubes through an interaction with Task 1.1 so that the required mechanical and thermal properties are satisfied with affordable and scalable composite manufacture processes. One or a few reference tube design(s) shall be defined in this task to identify reference tube material(s) in evaluation tasks and reference property values to Task 1.1.	Preliminary discussion was initiated in the R&D community of SiC ceramic composite ATF fuel cladding involving industries and academia. Mechanical and thermal data on several different architectures have been obtained [31].
1.4.1.2	Manufacture issues analysis	This task will perform early analysis of manufacture issues for thin-walled SiC/SiC tubes with an extreme length-to-	Preliminary discussion was initiated in the R&D community of SiC ceramic composite ATF fuel cladding

No.	Task Name	Description	Status
		diameter ratio, based on survey of presently available manufacture technologies for ceramic matrix composites in various application fields.	involving industries.
1.4.1.3	Segmentation options	Combining smaller length tubes into one full-length tube structure is a potential option to build a SiC/SiC-based fuel rod. This task will examine potential methods for manufacturing full length fuel rods from segmented tubes and status and development needs for the required technology. Additionally, issues associated with an alternative fuel assembly design option utilizing axially segmented fuel rods will be examined.	Preliminary discussion was initiated in the R&D community of SiC ceramic composite ATF fuel cladding involving industries.
1.4.1.4	Fuel length rod	Demonstrate the fabrication of a full 12' non segmented rod. This task will examine issues associated with scale up of non-segmented rods such as density and dimensional uniformity and equipment requirements.	Representative lengths up to 3' have been fabricated with design ongoing to move towards a full length rod in the next iteration. [31]
1.4.1.5	Preliminary assessment for NDE and QA requirements	This task will identify requirements and needs for non-destructive examination for SiC/SiC-based fuel clad tubes and fuel rods for the purpose of quality assurance.	Preliminary discussion was initiated in the R&D community of SiC ceramic composite ATF fuel cladding involving industries and academia. Acoustic emission and XCT are being evaluated as QA/QC tools.
1.4.2	End plugs		
1.4.2.1	Failure and performance guidelines	This task will define failure for end-plugs and develop preliminary guidelines for performance of end-plugs. The failure modes will be defined in a consistent manner with those for clad tubes. The performance guidelines will be defined in terms of properties including mechanical, thermophysical, irradiation, and oxidation.	Preliminary discussion was initiated in the R&D community of SiC ceramic composite ATF fuel cladding involving industries involved in FOA projects on end-plug technology development.
1.4.2.2	Joining development	This task develops technology to join SiC/SiC to the end plug material, most likely high purity monolithic SiC. Future program should be built upon recent or ongoing works. Basic joining methods will be developed in geometries of flat face joint and evaluated in simple modes. Development of successful joining methods will be expanded to the end-plug geometries. This task assumes the use of simple end-plug geometries such as butted lap and scarf. Development of more complex end-plug	The flat joint torsion samples were prepared using candidate joining methods for irradiation study. Irradiation results of these studies are positive for many of the joint candidates analyzed. Investigation into development and evaluation of ceramic joints in end-plug relevant geometries, including under irradiation, is ongoing.

No.	Task Name	Description	Status
		technologies such as those involving mechanical locking or fastening mechanisms may be considered, though.	
1.4.2.3	Test methods	Methods for testing end plug functions and performances will be developed. As with Task 1.2, development of ASTM test standards is assumed for all test methods to be developed for the purpose of adequate quality assurance. Effects of neutron irradiation and chemical environment will not be included in this task but will be studied in Task 2. Test methods to be developed will include those for 1) shear strength of flat joints, 2) strength of end-plug joints, and 3) gas tightness of end-plug joints.	Torsional shear strength test standard for flat ceramic joints completed the initial drafting and is prepared for the spring ballot at subcommittee level in ASTM C28 on Advanced Ceramics. End-plug push-out test standard for tubular endplug joined specimens completed initial drafting and is prepared for subcommittee ballot in ASTM C28.

Table 5. Structure, objectives, and outline of technical tasks in Environmental Effects category

No.	Task Name	Description	Status
2	Environmental Effects		
2.1	**Hydrothermal corrosion**		
2.1.1	Develop basic thermodynamic basis for the process	1) Develop and implement the database for all solid, aqueous, and gaseous reactants and products, 2) Quantify the thermodynamic driving force for SiC- and SiO_2-H_2O reactions as a function of environmental conditions, 3) Perform radiolysis simulations and input calculation results to quantify the thermodynamic driving forces for corrosion process under normal operating conditions, and 4) Perform MD and input calculation results to modify the thermodynamic driving forces for GB and other defects	Basic thermodynamics and reaction kinetics data have been acquired and analyzed, and the quantitative evaluation of hydrothermal corrosion for the SiC-H_2O system was performed.
2.1.2	Autoclave testing	1) Test a matrix of SiC samples with varying impurity levels and defect volumes for 12 months and collect data every 30 days, 2) Characterize sample surface after autoclave exposure, 3) Perform mechanical testing on exposed specimens to quantify strength degradation (if any), 4) Perform autoclave testing and examination of neutron irradiated and ion irradiated SiC samples, and 5) Perform autoclave testing on various SiC joints for short term for screening purpose, and then test down-selected joints for long term.	Work in progress with the first few mass evolution data points successfully obtained for unirradiated test specimens.
2.1.3	In-pile corrosion test	1) Test most resistant coupons in PWR/BWR in-pile loops (Note: Need info from autoclave tests) and 2) PIE of in-pile corrosion specimens.	Preliminary discussion has been initiated with external collaborators. A proposal to leverage ATR NSUF resources has been put in place.
2.2	**FCCI and FCMI**		
2.2.1	FCCI	1) Diffusion bond preparation, aging, and examination and 2) Thermodynamic analysis of equilibria (with high-burnup chemical effects)	SiC-Cr diffusion bonding was carried out from 900-1300°C. Microscopic characterization of the bonds was carried out. Isoplethal phase diagram of SiC-Cr was produced using computation thermodynamic tools.
2.2.2	FCMI	1) Plug expansion tests at applicable stress for unirradiated materials (Note: Need input from fuel performance analysis and Tube), 2) Plug expansion tests	Plug expansion tests performed on unirradiated materials. Planning underway for irradiated cladding.

No.	Task Name	Description	Status
		on irradiated cladding (if unavailable, mechanical tests on irradiated specimen), and 3) Coupling FCMI results with fuel performance analysis tool to predict failure threshold.	
2.3	**Irradiation Effects**		
2.3.1	Swelling and irradiated properties		
2.3.1.1	Low temperature swelling	Existing data for swelling of SiC and SiC/SiC composites in LWR-relevant irradiation conditions involve uncertainties with regard to accuracy of swelling measurement and irradiation temperature determination. This task will 1) Perform experimental determination of swelling of high purity CVD SiC and nuclear-grade SiC/SiC composites as a function of neutron fluence at LWR-relevant low temperatures. Put emphasis on accurate quantitative measurement of dimensional changes and irradiation temperature as compared to previously conducted experiments, and 2) Establish high accuracy fluence-dependence swelling development for high purity CVD SiC through an instrumented in-situ reactor experiment.	Accurate determination of swelling of high purity CVD SiC and a nuclear grade SiC composite is complete for specimens irradiated up to 0.3 dpa in HFIR in a collaboration project with Electric Power Research Institute. The planned 1 dpa irradiation experiment failed due to unexpected malfunction of the hydraulic rabbit facility of HFIR. Irradiation temperature determination is underway for individual specimens.
2.3.1.2	Irradiated properties	Recent experimental study shows significant and progressive degradation of Hi-Nicalon Type S, CVI-SiC matrix composites' mechanical properties at high fluence levels at an LWR-relevant low temperature. This task will clarify fluence-dependent evolutions of baseline mechanical and thermal properties at an LWR-relevant temperature for Hi-Nicalon Type S and alternative SiC fiber (Tyranno-SA3 and possibly SCS-Ultra SiC fibers) composites. Irradiation effects data are needed for a dose range 8 to at least 30 dpa at ~280°C.	A new set of rabbit vehicles started irradiation in HFIR after completion of capsule design, construction, and safety approval. The materials irradiated include CVI SiC-matrix composites with Hi-Nicalon Type S, Tyranno-SA3, and SCS-Ultra SiC fibers. The first two capsules completed irradiation in August 2015.
2.3.1.3	Post-irradiation permeability	Impermeability of irradiated material needs to be evaluated. Through-clad temperature and stress gradients evolve during irradiation under LWR conditions.	Planning underway.
2.3.2	Irradiation creep		

No.	Task Name	Description	Status
2.3.2.1	In-pile creep of SiC	Small but finite irradiation creep has been reported for relevant SiC materials. This task will determine the irradiation creep behavior of high purity CVD SiC at an LWR-relevant temperature to a high accuracy through an instrumented reactor experiment.	Experiment in Halden reactor facility has been designed, implemented, and the irradiation experiment nitiated in March 2015.
2.3.2.2	In-pile creep of joints	In SiC/SiC-based fuel clads, the end-plug joints will likely be the weakest link for possible irradiation creep-induced failure. This task will experimentally evaluate irradiation creep behavior of candidate joints or joining materials in a shear deformation mode after down-selection of most promising joints in Tasks 1.4.2 and 2.3.3.	Not yet started.
2.3.3	Irradiation effects on joints		
2.3.3.1	Irradiation effects in flat joint test articles	This task will evaluate the effects of neutron irradiation on shear strength and microstructures of joints down-selected in Task 1.4 using torsional shear test specimens with flat face joint layers. Establishment of an applicable test method in Task 1.4.2.3 is a pre-requisite. Structural and phase stability of the bonding materials and interfaces and consequences of likely differential swelling behavior between SiC and bonding material are of primary interest.	Irradiation program in HFIR is in progress with the flat joint torsion samples prepared using candidate joining methods, irradiation vehicles designed, constructed and safety-approved, and all vehicles completed ~8 dpa irradiation in reactor in February 2015. Post-irradiation examination (PIE) will be initiated after necessary cooling, capsule transfer and disassembly, and specimen shipment to the PIE facility.
2.3.3.2	Irradiation effects in end-plug geometry	In three-dimensional joint structures, the effect of differential dimensional evolutions between SiC and bonding material will be more severe than in flat face joint geometries. Moreover, the end-plug geometry is more appropriate both for measurement of gas permeability and for other properties (primarily strength) evaluation using in a relevant geometry. This task will evaluate gas permeability and mechanical properties of prototypical end-plug section test articles following neutron irradiation in LWR-relevant conditions. Test articles shall be made using the reference and/or candidate SiC/SiC tubes, as defined in Task 1.4, and candidate joining technologies that completed the	This effort will be initiated after successful development of test articles and test method.

No.	Task Name	Description	Status
		baseline irradiation study in Task 2.3.3.1, and be tested by test methods developed in Task 1.4.2.3.	

Table 6. Structure, objectives, and outline of technical tasks in Off-Normal Behavior category

No.	Task Name	Description	Status
3	Off-normal Environmental Effects		
3.1	**Steam oxidation**	1) Steam exposure of monolithic and model composite materials to high temperatures, 2) Steam exposure of SiC Joints to high temperatures, and 3) Steam exposure of candidate fully ceramic composites and hybrid SiC concepts to high temperatures.	Shown to be more than two orders of magnitude improvement over Zircaloy [29]. Planning is underway for the hybrid concepts.
3.2	**Thermal shock**	1) Thermal hydraulic analysis of quench conditions, 2) Quench test setup and execution for various cladding configurations (Note: Need input on tube design), and 3) Quench stress analysis for pressurized and de-pressurized cladding.	Initial evaluation suggested adequate shock tolerance for composites [50, 53, 54]. Planning underway.
3.3	**Accident analysis**	Using TMI-2 or Fukushima-3 as the baseline accident case, evaluate the performance of 1) Fully ceramic SiC/SiC cladding (all other materials remain standard), 2) SiC/SiC – Zircaloy hybrid cladding, and 3) SiC/SiC structures with standard Zircaloy cladding (assess the relative impact of SiC as a cladding vs. structural material with regard to total hydrogen production, maximum cladding temperature, oxidation heating, time to core melt, etc.). As the result, working MELCOR 1.8.6 code structure that allows fully decoupled structural and cladding components in the PWR case will be developed.	Initial assessment was performed [55, 56]. Planning underway for additional works.

Table 7. Structure, objectives, and outline of technical tasks in Seal Coating category

No.	Task Name	Description	Status
4	Seal Coating		
4.1	**Assessment**		
4.1.1	Material systems	This task identifies appropriate coating materials or material systems that provide adequate hydrothermal corrosion protection in normal LWR operating environments. The coating material also needs to survive transient swelling of SiC and retain sufficient gas tightness when penetrating cracks develop through a SiC/SiC clad structure. The impacts of coating will be addressed in comparison with fully-ceramic options.	Candidate coating materials adequate for providing hydrothermal corrosion resistance in normal LWR operating environments were identified.
4.1.2	Failure criteria and probability	Impacts of presence of a hermetic layer or layers on the criteria and probability of clad failure will be addressed.	
4.1.3	Tube manufacture and NDE/QA	Impacts of need for coating on manufacture of SiC/SiC tubes and requirement of NDE and quality assurance will be addressed.	
4.1.4	End plug	Impacts of presence of a hermetic layer or layers on the end plugging technology will be addressed.	
4.2	**Processing technologies**	Promising coating methods for candidate coating materials or material systems selected in Task 4.1 will be identified and developed. Trial-fabricated coated samples will be examined for baseline evaluation including physical integrity, homogeneity, microstructures, and bonding strength.	Three technically viable methods for metallic coating onto SiC surface were identified.
4.2.1	Plating	Develop plating technology for metallic coating on surface of SiC/SiC cladding.	A method for electroless seed coating was developed. A method for surface treatment that enables electroplating onto SiC surface was identified.
4.2.2	Plasma spray	Develop vacuum plasma spray technology for metallic coating on surface of SiC/SiC cladding.	Methods for vacuum plasma spray coating with a variation of metals are being developed.
4.2.3	Transient heating	Develop transient heating technology for metallic coating on surface of SiC/SiC cladding.	A preliminary study to evaluate feasibility of using plasma arc lamp processing technique to apply a metallic coating on SiC is initiated.
4.3	**Environmental effects**	Effects of neutron irradiation and chemical environment will be evaluated per methods used in Task 2.	

No.	Task Name	Description	Status
4.3.1	Hydrothermal corrosion	Corrosion of coating material will become a primary hydrothermal corrosion issue if coating is on the outside. However, corrosion of structural SiC/SiC will remain important.	
4.3.2	FCCI and FCMI	Chemical interactions of coating material and fission products that diffuse through SiC/SiC layer needs to be addressed. Also if the coating is on the inside it could greatly affect (to enhance or reduce vulnerability to) FCMI processes.	
4.3.3	Irradiation effects	General irradiated properties of coating material and coated material systems need to be experimentally addressed. As the first step, effects of neutron irradiation on integrity of the sealing function and the substrate-coating bonding strength in an LWR-relevant irradiation condition need to be determined. More focused objectives and technical approach of this task will be defined after the candidate coating concepts and the specific technical issues are defined in Task 1.4.3. Should end-plug joining technology is affected by coating, irradiation effects need to be addressed, particularly for irradiation creep.	This effort will be initiated after successful development of candidate coated materials.
4.4	**Off-normal Environmental Effects**	Behaviors of coating materials and coated material systems at off-normal high temperatures and presence of steam environment needs to be clarified.	

5. CONCLUSIONS

The initial technical plan for the Systematic Technology Evaluation Program for SiC/SiC Composite Accident-Tolerant LWR Fuel Cladding and Core Structures was laid out, in support of the Fuel Cycles Research and Development Program for the Office of Nuclear Energy, the U.S. Department of Energy. The primary objective of the planning was to develop a blueprint of a technical program that addresses the critical feasibility issues; assesses design and performance issues related with manufacturing, operating, and off-normal events; and advances the technological readiness levels in essential technology elements for SiC/SiC composite-based LWR fuel cladding and core structure concepts.

The presented planning exercise consists of three main tasks: a technology review, a critical technology gap analysis, and a technical program planning. Various technical gap issues were identified during the technology review process and the analysis revealed that many of them are related with the three key feasibility issues for SiC/SiC composite fuel cladding: hydrothermal corrosion, cracking failure, and fuel-clad system compatibility. Additional very important gap issues have also been identified and analyzed. The technical program plan was designed to systematically address these key technology gaps and feasibility issues, and then lay out the tasks in a work breakdown structure, which consists of three top level categories: Design and Failure, Environmental Effects, and Off-normal Behavior. Simultaneously, the plan attempts to set an early technical program toward advancing the technological readiness levels of essential technologies in these three areas.

It is emphasized that this technical program plan document is intended to receive continued updates beyond the current version. The future updates will incorporate input from broader community knowledge and reflect the evolving technology and findings. The current technology review and the gap analysis are largely based on the previous technical report [5] and discussions at the recent program workshop (Appendix). Input that contributes toward a technically sound evolution of this program plan is sought from the expert communities of not only nuclear engineering but also materials science, composite technology, and other application engineering areas where the ceramic composites are considered as important enabling materials. Moreover, it is believed that the additional workshops and focused technical meetings will be useful for deepening the discussion and further refining the program plan.

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